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A REALISTIC PLUTONIUM ELIMINATION SCHEME WITH FAST ENERGY AMPLIFIERS AND THORIUM-PLUTONIUM FUEL

C. Rubbia, S. Buono¹⁾, E Gonzalez, Y. Kadi and J.A. Rubio

Abstract

In a previous report [1] we have presented the conceptual design of a sub-critical device designed for *energy amplification* (production). The present note further explores the possibilities of the Energy Amplifier (EA) in the field of the *incineration* of unwanted actinide "waste" from Nuclear Power Reactors (PWR) and from the disassembly of Military Weapons.

The key idea which is developed is the one of using a Thorium-Plutonium mixture which is much more effective in eliminating Plutonium at acceptable concentrations ($\leq 20\%$) than the conventional mixture of Uranium-Plutonium. The device operates as an effective Plutonium to ^{233}U converter. The latter can be later mixed with ordinary or depleted Uranium and it constitutes an excellent fuel for the PWRs. The EA sub-critical mode is preferred over the conventional Fast Breeder Reactor, because of the much smaller risks associated to the narrow criticality window (approximately $\pm 0.15\%$ in Δk) of a Fast Reactor, the negative void coefficient related to the presence of Plutonium and the inevitable presence of large amounts of highly toxic Plutonium.

It is shown that a cluster of EAs operated in conjunction with existing PWRs is a very effective and realistic solution to the ultimately complete elimination of the accumulated Plutonium and Minor Actinide stockpiles and it greatly alleviates the problem of definitive geologic disposal. Preliminary economical considerations show that Plutonium incineration when compared to direct geological disposal is not only environmentally more acceptable but also an economically profitable alternative.

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¹⁾Sincrotrone Trieste, Trieste, Italy

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1.- General considerations

The linearly growing accumulation of Plutonium and higher Actinides (Am, Cm, Cf, etc.) due to civil Nuclear Power and the stockpile of bomb-grade Plutonium from Military Arsenals are a cause for serious concern [2]. The projected situation in the year 2010 for the present civil nuclear power capacity is given in Table 1.1, taken from Schapira [3]. The surplus from destroyed nuclear warheads is somewhat uncertain. We will base our considerations on Ref. [3] and assume the amount of military grade Plutonium which needs to be eliminated is about 100 tons out of a total of about 300 tons, of which about 180 tons still in warheads.

Many schemes have been proposed in order to dissipate such a major liability to mankind. Underground geologic disposal is the only realistic solution so far retained for implementation and it begins to be deployed¹ in some countries. But it is not without concern and is a subject of strong public opposition, mainly in view of the very long period of perfect retention required, generally in excess of millions of years. Other schemes, like for instance sending the offending materials in space have been proposed but they do not seem practical in view of the large amounts of material involved and the risks of a crash of the vectors².

Studies on alternative methods based on nuclear transformation of the reprocessed fuel have gained a considerable momentum [3]. The majority of these

Table 1.1.- Projected amount of Radioactive waste from ≈ 400 GWe Nuclear Reactors (world-wide) in year 2010. [3]

Total spent fuel	300,000	tons
Plutonium isotopes	3000	tons
Neptunium isotopes(²³⁷ Np)	140	tons
Americium and higher Actinides	120	tons
Long lived Fission fragments		
⁹⁹ Tc	250	tons
¹³⁵ Cs	90	tons
¹²⁹ I	60	tons

¹ In some countries, deep underground disposal is forbidden by law. In others, only "domestically produced" waste can be disposed. For more details see [2].

² The reliability of a normal launch now is estimated to be about 95%. The manned mission reliability is about 98%.

Figure 1.1 General scheme of coupled EA-PWR. The mass flows refer to one year of operation. The power of the PWR is adapted to the figures for a single EA complex, as described in Ref. [1].

schemes are based on the idea of "incineration" in which neutrons or other particles induce nuclear reactions which transform the bulk of the unwanted materials into more acceptable nuclear species. In the case of Actinides the leading processes are fissions of fissionable elements and transformation of poorly fissionable nuclides into fissionable ones by neutron capture(s). In the case of long-lived non-fissionable radio-nuclides, like for instance Fission Fragments (^{99}Tc , ^{129}I , etc.) in general, methods proposed transform long-lived into short-lived or stable species with a neutron capture (e.g. $^{99}\text{Tc}[2.1 \times 10^5 \text{ y}] + n \rightarrow \gamma + ^{100}\text{Tc}[15.8 \text{ sec}] \rightarrow \beta^- + ^{100}\text{Ru}[\text{stable}]$). These captures give particularly spectacular effects with Fission Fragments since nuclei which are the outcome of fission are already "neutron rich" and an extra neutron has a strong de-stabilising action.

Many schemes based on critical Reactors and on spallation target driven sub-critical devices have been discussed over the last several years [4]. The present proposal is directly derived from the Energy Amplifier (EA) of which we have recently released a conceptual design [1] in which a first description of these ideas has been already given. It is based on a coupled operation of an (existing) PWR and

of a specially operated EA. We propose a fuel load for the EA specifically designed to achieve an effective incineration and made of a mixture (MOX) of the unwanted Plutonium plus *all* other associated higher Actinides³, *imbedded in a Thorium matrix*. The presence of the Thorium, instead of the usual Uranium in the mixture eliminates the ^{239}Pu producing chain $^{238}\text{U} + n \rightarrow ^{239}\text{U} \rightarrow ^{239}\text{Np} \rightarrow ^{239}\text{Pu}$, replacing it with the ^{233}U producing $^{232}\text{Th} + n \rightarrow ^{233}\text{Th} \rightarrow ^{233}\text{Pa} \rightarrow ^{233}\text{U}$. While the former chain is producing additional ^{239}Pu , thus reducing the efficiency of incineration, the latter produces ^{233}U which can be reused and fully burnt in a Light Water Reactor once mixed with ^{238}U ⁴.

In short (Figure 1.1) the EA acts as a $\text{Pu} \rightarrow ^{233}\text{U}$ transformer and the PWR burns the ^{233}U , producing a smaller amount of Pu which is re-injected in the EA. The system PWR-EA operates indefinitely in a closed fuel cycle with an intermediate Fuel Reprocessing and Fuel Manufacturing phase. The cycle is fed externally by additional Pu, either from civil or military applications and smaller amounts of fresh Uranium and Thorium to compensate the amounts which are burnt and it discharges the Fission Fragments (FFs), which have a much shorter activity lifetime than the Actinides. Some specific long lived FFs (^{99}Tc , ^{129}I , etc.) can also be incinerated re-injecting them in the EA⁵. The net results [1] are (1) no more Actinides to be disposed in a Geologic repository and (2) the (ingestive) radio-toxicity of the FFs is after some 600 years in a "Secular Repository" about 10^4 times smaller than in the case of an open PWR cycle for the same produced energy. These discharged radio-elements have a negligible amount of α -activity and have a residual radio-toxicity which matches the best expectations of Fusion [1].

In order to illustrate the advantages of the Thorium-Plutonium mixture over the conventional methods based on Uranium-Plutonium we show in Figure 1.2 the Pu elimination efficiency as a function of the relative concentration of Pu in the Uranium-Plutonium fuel of a CAPRA scheme [5]. At a relative concentration of the order of 15%, the newly produced and the incinerated Pu exactly balance with no net improvement of the Pu stockpile. While appreciably below such a critical concentration the Reactor is a "Breeder", for higher concentrations it becomes an

³ We underline that in our method all Actinides, with the exception of the Uranium, are incinerated together. In the majority of the studies in the literature most often specific elements (e.g. Am, Np, Pu etc.) are separately targeted.

⁴ As pointed out later on, this is one of the many possibilities, which are application dependent. For instance the produced ^{233}U could be used instead as (initial) fuel for another, new EA.

⁵ Evidently by mass conservation the amount of fresh Th, U and Pu to be injected in the cycle is equals to the mass of the produced FFs, plus a significant mass due to the mc^2 term in the mass to energy transformation.

Figure 1.2: Plutonium elimination efficiency as a function of the relative concentration of Pu in the Uranium-Plutonium fuel for a fast Breeder. (Source: CEA, unpublished)

“Incinerator”. The incineration efficiency becomes dominant only at Pu concentrations which are very high: evidently the best performance is achieved with a pure Pu Reactor.

Such very high concentration of Pu fuel has considerable problems since (1) its bare critical mass is very low (2) its reactivity is choked by the emergence of Fission Fragments after a small relative change of the Pu concentration⁶ and (3) the Fast Breeder is harder to operate than with its standard fuel because of the increased negative void reactivity coefficient⁷ (due to the high Plutonium concentration) and the reduced Doppler effect. Dissolving the unwanted Pu instead in a fertile Thorium matrix greatly alleviates such problems and a major relative fraction of the inserted Pu even at acceptable concentrations can be burnt without interventions or manipulations of the fuel, since the bred ^{233}U is an effective substitute to Pu in order to maintain a viable and constant criticality [1].

⁶ In a pure Plutonium fuel, fissioning about 10% of the mass produces fission fragments concentration sufficient to reduce the multiplication coefficient k by about 7%. Hence one is led to very frequent refills and manipulations. At the limit one is led to the “chemistry on-line” proposed by the Los Alamos group [6].

⁷ Expressed as $\Delta k/(\Delta \rho/\rho)$, i.e. a decrease in coolant density arising from the formation of voids in the core of a Fast Reactor produces an increase in reactivity.

Many variations besides the scheme of Figure 1.1 are of course possible. More generally, the ^{233}U is easily recovered chemically⁸ at the end of the Thorium-Plutonium cycle in the EA and it constitutes a valuable product, since for instance

- (1) It can be mixed with depleted or ordinary Uranium⁹ to produce additional enriched fuel for PWRs without isotopic separation (Figure 1.1) or
- (2) It can be re-injected in the EA in order to increase significantly the neutron inventory, as required in order to incinerate the long lived Fission Fragments (FFs), like for instance ^{99}Tc , ^{129}I , etc., produced both by the PWR and the EA. It is well known that the main problem with incinerating FFs is the availability of the appropriate fraction of the produced neutrons which must be sacrificed to this task.
- (3) It can be used as "seeds" to develop a line of EAs purely operated on Thorium, as described in Ref. [1].

We believe that the fast EA coupled with the Thorium-Plutonium mixture is the winning formula and this for very specific reasons:

- (1) It is well known that *fast neutrons* are more effective than thermal neutrons in incinerating Actinides from the discharge of a thermal PWR since the fission process is more abundant at high neutron energies and generalised to the majority of the elements involved in the incineration

Figure 1.3: *Effect of a fast (15 ms) 2.5 \$ reactivity insertion in the device of Ref. [1], operated as a Fast Breeder or as a sub-critical device, with two possible multiplication coefficient values, namely $k = 0.98$ and $k = 0.96$. The magnitude of the reactivity insertion corresponds roughly to the accidental extraction of all the control bars.*

⁸ ^{233}U is the dominant Uranium isotope, the rest being made of small amounts of ^{232}U , ^{234}U and ^{235}U . The effects of the activity of the ^{232}U descendants (^{208}Tl) are discussed further on.

⁹ Mixing with ^{238}U of course "denatures" the mixture and inhibits military diversions of the newly produced fuel.

chain. Therefore a fast neutron stage in the incineration cycle appears to us as almost a necessity.

- (2) A *sub-critical operation*, namely an Energy Amplifier, offers considerable safety and operational advantages over a critical Fast Reactor. We show in Figure 1.3 taken from Ref. [1] the effect of an accidental fast reactivity insertion for a critical and a sub-critical operation. A significant fast accidental reactivity insertion ($\Delta k/k \approx 7 \times 10^{-3}$), of the order of magnitude of the full extraction of the control bars will cause a major, catastrophic power excursion, but in the case of the EA the effect is acceptable with mild consequences. The curves of Figure 1.3 have been calculated for the geometry of the EA of Ref. [1]. Rief and Takahashi [7] have reached the same conclusion in the case of a Sodium cooled Fast Breeder. The indicated reactivity insertion will produce the instant vaporisation of $\approx 85\%$ of the Sodium coolant in the critical case and no appreciable consequence for the sub-critical device.
- (3) As already pointed out, *Thorium* matrix has the advantage of producing far less Plutonium than the Uranium. In the present scheme we have introduced Thorium only in the fast neutron part of the (closed) fuel cycle, since we would like to make a minimum number of changes in the thermal PWR operation. However the introduction of Thorium also in the thermal neutron segment is beneficial since it reduces the production of Plutonium in the burning of the ^{233}U . This point is further discussed in paragraph 5.

It is evident that the added safety of operation offered by the EA is of particular value in this application in which one has necessarily to deal with large amounts of highly hazardous Plutonium. As an added safety it may be advisable to operate the Thorium-Plutonium mixture at a lower value of k than $k = 0.98$ which was chosen for the pure EA in Ref. [1]. For instance a

Figure 1.4 Fraction of the total produced electric power recirculated in the Accelerator, as a function of the value of the multiplication coefficient k . A reasonable range of choices for the operating range of k is also indicated.

value¹⁰ of $k = 0.95 \div 0.96$ will ensure that during the very long, unperturbed incineration cycle ($120 \div 200 \text{ GW} \times \text{day/t}$) the EA behaves as safely as a “beam dump”. The main drawback of a lower k is an increase of the power to be recirculated through the accelerator, which however remains within acceptable limits, as shown in Figure 1.4.

Plutonium or other Actinides complete elimination through fissions produces a very large amount of energy, namely $940 \text{ MW} \times \text{day}$ for 1 kilogram. In practice the energy yield is higher since for instance fissions occur in the Thorium matrix (few per cent) and in the ^{233}U bred in the Thorium. As shown later on this implies some 30% additional energy production, and therefore the energy produced by the incineration of 1 kg of unwanted Actinides is about $1200 \text{ MW} \times \text{day}$. Typically a nominal 1.0 GW-electric PWR produces about $900 \text{ GW} \times \text{day}$ of thermal energy yearly and a total “Dirty Plutonium” waste (N, Pu, Am, Cm, Cf etc.) of 271 kg. Hence “incineration” of such a waste will inevitably lead to the production of some $271 \times 1.200 = 325.2 \text{ GW} \times \text{day}$, namely $325.2/900 = 36.13 \%$ of the power produced by the initial PWR, close to the “theoretical” limit in which only the Plutonium is burnt, namely $271 \times 0.940/900 = 28.30 \%$ of the PWR power. The energy accumulated in the Actinide “waste” and which can be recovered with our method is therefore very large, since it amounts to about $1/3$ of all the nuclear power produced so far. In particular an EA-incinerator will eliminate the Actinide waste at the rate produced by a PWR with only about $1/3$ of the installed power.

Since only a fraction of the neutron interactions produce fissions in each reaction, in order to reach complete elimination, the fuel must undergo a long chain of nuclear transformations each with a specific, variable degree of fission probability. These reactions are sustained by the neutrons coming from the fission reactions. It is a fortunate circumstance that in our scheme the number of such neutrons is sufficient to complete the job throughout the evolution, essentially at a constant criticality [1]. As shown in Ref. [1] after a sufficiently long time an equilibrium situation is reached between the constant inflow of material to be incinerated and the reaction chain with the fission elimination channel.

A number of simplifying assumptions permits to calculate analytically the essential features of the evolution of the concentration vector $c_{(A,Z)}(\phi)$, where ϕ is the (average) neutron flux. We ignore the discontinuity of the refills and assume a constant inflow of the father element and neglect the $(n,2n)$ and other channels which may introduce “loops” in the (A,Z) evolution plane. We assume that in the presence

¹⁰ We recall that in the case of a Geologic Repository (for very long periods) the value $k \leq 0.95$ has been deemed acceptable.

of the neutron flux ϕ , for all elements there is only one transformation channel (either with neutron capture averaged cross section $\sigma_{capt}^{(i)}$ or radioactive decay with decay rate $\lambda^{(i)}$, whichever is dominant) and a dissipative, fission channel with spectrum averaged cross section $\sigma_{fiss}^{(i)}$. For very high values of A spontaneous fission and other forms of nuclear instability will contribute to such dissipative terms. The rate of transformation in a neutron flux ϕ is $\phi\sigma$ and the total rate $\mu^{(i)} = \phi(\sigma_{capt}^{(i)} + \sigma_{fiss}^{(i)})$ or $\mu^{(i)} = \phi\sigma_{fiss}^{(i)} + \lambda^{(i)}$ if the transformation is decay dominated. The survival, chaining coefficient, which represents the probability of continuation to the next step of the evolution chain is defined as $\alpha^{(i)} = \sigma_{capt}^{(i)} / (\sigma_{capt}^{(i)} + \sigma_{fiss}^{(i)})$ or $\alpha^{(i)} = \lambda^{(i)} / (\lambda^{(i)} + \sigma_{fiss}^{(i)} \times \phi)$ respectively. The procedure is schematically shown below:

Chain	$P \rightarrow$	$N_1 \rightarrow$	$N_2 \rightarrow$	$N_3 \rightarrow$	$N_i \rightarrow$
	$\downarrow$	$\downarrow$	$\downarrow$	$\downarrow$	$\downarrow$
Initial amount	$N_1(0)$	0	0	0	0
Removal rate	$\phi\sigma_{fiss}^{(1)}$	$\phi\sigma_{fiss}^{(2)}$	$\phi\sigma_{fiss}^{(3)}$	$\phi\sigma_{fiss}^{(i)}$	
Transfer rate	$\phi\sigma_{capt}^{(1)}, [\lambda^{(1)}]$	$\phi\sigma_{capt}^{(2)}, [\lambda^{(2)}]$	$\phi\sigma_{capt}^{(3)}, [\lambda^{(3)}]$	$\phi\sigma_{capt}^{(i)}, [\lambda^{(i)}]$	
Survival coeff. $\alpha^{(i)}$	$\frac{\sigma_{capt}^{(1)}}{\sigma_{capt}^{(1)} + \sigma_{fiss}^{(1)}}$	$\frac{\sigma_{capt}^{(2)}}{\sigma_{capt}^{(2)} + \sigma_{fiss}^{(2)}}$	$\frac{\sigma_{capt}^{(3)}}{\sigma_{capt}^{(3)} + \sigma_{fiss}^{(3)}}$	$\frac{\sigma_{capt}^{(i)}}{\sigma_{capt}^{(i)} + \sigma_{fiss}^{(i)}}$	
Total rate $\mu^{(i)}$	$\phi\sigma_{fiss}^{(1)} + \lambda^{(1)}$	$\phi\sigma_{fiss}^{(2)} + \lambda^{(2)}$	$\phi\sigma_{fiss}^{(3)} + \lambda^{(3)}$	$\phi\sigma_{fiss}^{(i)} + \lambda^{(i)}$	

Assuming first no refill ($P = 0$) and an initial number of nuclei $N_1(0)$, the time evolution is given according to the Bateman equation ($i > 1$):

$$N^{(i)}(t) = N^{(0)}(t) \left(\prod_{j=1}^{j=i-1} \alpha^{(j)} \right) \times \left[\left(\prod_{j=1}^{j=i-1} \mu^{(j)} \right) \times \sum_{j=1}^{j=i} \frac{\exp(-\mu^{(j)}t)}{\prod_{\substack{k=1 \\ k \neq j}}^{k=i} (\mu^{(k)} - \mu^{(j)})} \right]$$

If alternatively, there is refill at the constant rate P per unit time and no initial nuclear sample, i.e. $N_1(0) = 0$, the formula becomes ($i > 1$)

$$N^{(i)}(t) = P \left(\prod_{j=1}^{j=i-1} \alpha^{(j)} \right) \times \left[\left(\prod_{j=1}^{j=i-1} \mu^{(j)} \right) \times \sum_{j=1}^{j=i} \frac{1 - \exp(-\mu^{(j)}t)}{\mu^{(j)} \prod_{\substack{k=1 \\ k \neq j}}^{k=i} (\mu^{(k)} - \mu^{(j)})} \right]$$

In practice, both refilling and initial nuclei are present and the actual number of nuclei will be simply the sum of the two above terms. Note that for a power density

$\rho \approx 100 \text{ W/g}$, $\phi \approx 5 \times 10^{15} \text{ cm}^{-2} \text{ s}^{-1}$ and that the sum of cross sections is of the order of magnitude of $\approx 2 \times 10^{-24} \text{ cm}^2$, leading to an evolution time constant $1/\mu^{(i)}$ of the order of ≈ 3 years.

The asymptotic distribution is reached at the limit $t \rightarrow \infty$. At this stage the process is dominated by the refill term P and one can easily calculate the equilibrium amounts:

$$N^{(i)}(t \rightarrow \infty) = P \frac{\left(\prod_{j=1}^{i-1} \alpha^{(j)} \right)}{\mu^{(i)}} = N^{(1)}(t \rightarrow \infty) \frac{\mu^{(1)}}{\mu^{(i)}} \left(\prod_{j=1}^{i-1} \alpha^{(j)} \right)$$

The time required by $N^{(i)}$ to grow to $N^{(i)}(t \rightarrow \infty)(1 - 1/e)$ is approximately given by $\sum 1/\mu^{(j)}$ where the sum is extended up to i . Since the order of magnitude of the time constant is typically 3 years, equilibrium is reached after $\approx 3(i-1)$ years where we have used $i-1$ to take into account that the step through the ^{233}Pa is fast. The fast decrease of $N^{(i)}(t \rightarrow \infty)$ with the rank in the chain is due to the product of the $\alpha \ll 1$ terms. To a fast decreasing degree of concentrations, the whole table of elements is eventually involved. As already pointed out, in practice the chain is not open-ended since spontaneous fissions and other instabilities ensure very small α -values toward the end. Fast neutrons are highly preferable over thermal neutrons, since the fission yield is much larger and opened to more elements.

Specific scenarios can be visualised. Evidently the EA incineration facility will be added to existing PWRs only many years after the beginning of operation. Therefore both the already accumulated Plutonium stockpile and the continuous production will have to be eliminated. Marchetti [8] has shown that the implantation of Nuclear Power in the western world follows an "epidemic equation" of the type $dN = aN(\bar{N} - N)dt$, namely the number of new adopters dN during time dt is proportional to the number of actual adopters N times the number of potential adopters $(\bar{N} - N)$, where $\bar{N}$ is the final number of adopters. Integrating the equation and replacing $\bar{N}(F = N / \bar{N})$ we get the solution $\log(F/1 - F) = at + b$, where the time constant $\Delta T = 4.39/a$ is the time to go from $F \approx 0.1$ to $F \approx 0.9$ and the central date is $T_0 = -b/a$. For the Western World $T_0 = 1981$, $\Delta T = 19$ years and a saturation value of 315 GWe. Parameters of the Marchetti's fit are listed in Table 1.2. By now all entries have reached in practice the asymptotic value, i.e. the market "niche" is saturated. The useful lifetime of the PWR is assumed to be 40 years. The nuclear energy supply from existing Reactors will then come to an end as shown in Figure 1.5. For instance in order to maintain the total nuclear power at its constant value, new power sources are required with $T_0^{EA} = 1981 + 40 = 2021$. The amount of accumulated Plutonium waste is shown in Figure 1.6.

Figure 1.5: Available power from existing nuclear reactors in the Western World and future evolution under the assumption of a 40 years lifetime for the power plants. The required new power sources in order to maintain the total power constant is also shown.

Table 1.2.- Parameters of the penetration of Nuclear Power in World nations according to Marchetti [8]. Values refer to the "Epidemic Equation".

Country	$\bar{N}$ (GWe)	ΔT (years)	T_0 (50%)
US(1) ¹¹	56	8	1974
US(2)	52	6	1985
USSR	70	20	1986
Sweden	12	14	1980
France	70	12	1983
Japan	20	15	1980
FRG	27	18	1983
Canada	16	20	1982
Western World	315	19	1981

¹¹ US exhibits two cycles.

Figure 1.6 Accumulated "Dirty Plutonium" waste in the Western World as a function of time for three possible scenarios, namely (1) All power plants are rebuilt in order to maintain the total power at the present level (2) No more nuclear plants are built and new power sources are of conventional, non nuclear type and (3) Nuclear power is renewed with EA-PWRs scheme with a fraction f of the total power produced by EAs.

We can distinguish three limiting scenarios:

- (1) If the replacement plants are ordinary PWRs and no incineration technique is applied, the waste accumulation will continue linearly with the spectacular amount of the order of 7000 tons¹² of Pu and minor Actinides by circa 2050.
- (2) If the Nuclear option will come to a halt, namely no new Reactor is built in the future, the residual Pu waste will be still of the order of 3800 tons.
- (3) Introducing the EAs has spectacular effects. We assume that a fraction f of the total nuclear power is produced by EAs initially fed with Plutonium¹³ and that the electric energy supply from nuclear sources is kept constant, the Western World's stockpile of "Dirty Plutonium" will be reset to zero by circa 2045 ($T_o^{EA} + 24$) for $f = 0.85$ and 2060 ($T_o^{EA} + 39$) for $f = 0.65$ (Figure 1.6).

Figure 1.7 : Residual stockpile of "Dirty Plutonium" and its descendants as a function of cycle in the EA. After a first phase of incineration in which the Pu is injected at each cycle, the EAs are operated as Amplifiers, with fuel recycling but no external Pu injection. (a) the injected Pu is "Dirty Plutonium" from PWR discharge; (b) the Pu is of military origin (bomb grade).

We expect that the lifetime of an EA plant be longer than the one of a PWR, provided the necessary maintenance and replacement of components are carried out¹⁴. Assuming an EA useful lifetime of ≈ 80 years, the park of EAs will come to the end of the useful life by circa $T_o^{EA} + 80 = 2101$. Hence, after the existing Plutonium has been eliminated one may wish to continue with several EA cycles as a pure Thorium

¹² For the open cycle and no reprocessing, the total amount of spent fuel "waste" is about 100 times larger, namely 0.7 million tons.

¹³ The EAs are operated with Pu-Th mixture in such a way that all the Pu (pre-existing and produced by the new 1-f fraction of PWRs) is eliminated. Extra power is produced with EAs operated in the energy Amplification mode, namely with Th and recycled ^{233}U . The required, initial ^{233}U stockpile is produced by $\text{Pu} \rightarrow ^{233}\text{U}$ transformation (EA).

¹⁴ In our view a well designed EA, after some part replacements could have a lifetime well in excess of 50 years.

burner as described in Ref. [1], namely with no Pu injection. This latter mode will be vastly cleaner than the PWRs [1] and in addition it will help to incinerate more completely the most neutron resilient radio-toxic residues accumulated in the EA fuel due to Pu burning initial cycles. As shown in Figure 1.7, the amount of residual “waste” drops by about one order of magnitude during the incineration phase and to about few units of 10^{-3} after the second phase. Since the burning of the unwanted Actinides is progressively growing parasitic with the reduction of the volume of the surviving waste, additional incineration may be pursued beyond the indicated 10^{-3} level of waste, if still considered excessive.

Assume finally a minimal programme in which one wants to eliminate *only* the accumulated waste, with no concern about maintaining the total nuclear power constant. If the useful lifetimes of the EA and of the PWR are respectively Γ_o^{EA} and Γ_o^{PWR} , the EA must be able to incinerate Pu at a rate $\eta = \Gamma_o^{PWR} / \Gamma_o^{EA}$ the PWR production rate. At the end of the operation one may wish to run n EA cycles as a pure Thorium burner. If $N = \Gamma_o^{EA} / \Gamma_{cycle}^{EA}$ represents the total number of cycles each of duration Γ_{cycle}^{EA} in the lifetime of the EA, the Pu incineration rate, again in units of the PWR production rate is $\eta = (\Gamma_o^{PWR} / \Gamma_o^{EA}) [N / (N - n)]$. Note that for a cycle time of 3 (5) years and a useful minimal plant lifetime of ≈ 90 years¹⁵, $N \approx 30$ (18) namely only a limited number of refills and major manipulations are necessary over the lifetime of the EA. Assuming that the lifetime of the PWR to be 1/2 of the one of the EA, we could set $n = N/2 \approx 15$ (9) and therefore $\eta = 1$, namely EA and PWR burn and produce waste at the same rate with a power production in the EA which is about $f = 0.33$ of the one of the PWRs¹⁶ and a second, subsequent phase of pure EA burning is carried out in order to complete the elimination of the unwanted waste.

The combined EA-PWR cycle is very effective also in eliminating Military Surplus Plutonium. The elimination of the totality of the estimated surplus of 100 tons of ^{239}Pu will produce a total of $400 \text{ GW} \times \text{year}$ in the EAs. *Two clusters of 3 EA units each for a thermal power of 4500 MW and an electric power of 1870 MW could eliminate it in about 45 years.* As mentioned above, an additional period of similar duration with the EAs operated as “amplifiers”, in which the ultimate burn-up is performed “parasitically” will ensure the final reduction of the residue to the level of few 10^{-3} of the initial mass ($\leq 200 \text{ kg}$) (Figure 1.7). The 86.18 tons of ^{233}U produced in

¹⁵ We have slightly stretched the useful lifetime of the PWRs and of the EAs, since we assume that if the technology will have to be terminated, one would try to get the maximum benefit from the investment.

¹⁶ We have taken into account that the thermal efficiency of an EA is higher than the one of a PWR because of the higher operating temperature [1]. If one assumes equal efficiencies, then $f = 0.40$.

the transformation can be used to produce as much $86.18/0.02375 = 3629$ tons of useful fuel for PWRs (2.375% enrichment). The produced fresh fuel is sufficient to run a standard 1000 MW (electric) PWR for about 100 years.

When compared to the relatively established MOX-Fuel technology applied to Thermal PWRs [9], the present scheme has the advantage that while the former method allows essentially a single pass through the Reactor and then it becomes “waste”, with a reduction of about a factor 2 in the unwanted Actinides, the present scheme allows many passages without appreciable loss in reactivity, eventually completed by a final phase as energy generating EA (without fresh Plutonium injections) and therefore (as explained in detail later on) the surviving Plutonium and higher Actinides are several orders of magnitude smaller than the total injected stockpile.

Finally Plutonium incineration when compared to deep geological storage is not only environmentally more acceptable but also economically profitable, since the cost of the energy produced by the EAs is competitive in price to the one of PWRs [10]. This point will be discussed more completely in a forthcoming paper [14].

2.- The Energy Amplifier

A conceptual design of a fast neutron operated EA is described in detail in Ref. [1] to which we refer for all details. In short the EA is made of two main coupled elements, (1) a specially designed proton accelerator of high energetic efficiency and high reliability, capable of delivering a beam typically of $10 \div 20$ mA at a kinetic energy of 1 GeV (Figure 2.1) and (2) a Lead cooled beam dump in which the beam power is greatly increased by the secondary interactions of thermonuclear cascade in a sub-critical medium and extracted by conventional heat exchangers (Figure 2.2).

In order to enhance the “passive” nature of the “beam dump” we have chosen for that design a pure convective cooling¹⁷. In spite of the large power involved, we have reached a sensible set of parameters, mainly because of the unique properties of the molten Lead as coolant. A general layout is shown in Figure 2.3 and the list of main parameters is given in Table 2.1 both taken from Ref. [1].

¹⁷ Evidently pumps, especially electromagnetic pumps, could be used instead, if further analysis proves them more convenient.

Table 2.1 : Main parameters of the Energy Amplifier (Thorium operated)

Gross Thermal Power/unit	1500	MW
Primary Electric Power	625	MW
Type of plant	Pool	
Coolant	Molten Lead	
Sub-criticality factor k, (nominal)	0.98	
Doppler Reactivity Coefficient, ($\Delta k / \Delta T$)	$- 1.37 \times 10^{-5}$	
Void coefficient (coolant) $\Delta k / (\Delta \rho / \rho)$	+ 0.127	
Nominal energetic Gain	120	
Accelerator re-circulated Power	30	MW
Fraction Electric Power recirculated in Accel.	0.0465	
Control Bars	none	
Scram systems(3)	CB ₄ rods	
Seismic Platform	yes	
<i>Main Vessel</i>		
Gross height	30	m
Diameter	6 m	m
Material	HT-9	
Walls thickness	70	mm
Weight (excluding cover plug)	2000	ton
Double Liner	yes	
<i>Proton Beam and Spallation Target</i>		
Accelerator type	Cyclotron	
Number of beams	1	
Accelerator overall efficiency ¹⁸	43%	
Kinetic energy	1.0	GeV
Nominal current	12.5	mA
Nominal beam Power	12.5	MW
Maximum current	20	mA
Spallation Target material	Molten Lead	
Beam radius at spallation target	7.5	cm
Beam window	Tungsten, 3.0 (1.5)	mm
Max. power density in window	113	W/cm ²
Max. Temp. increase in window	137	°C
Window expected lifetime	≥ 1	year

¹⁸ Beam power/Mains Load.

Fuel Core

Initial fuel mixture	ThO ₂ + 0.1 ²³³ UO ₂		
Initial fuel mass	28.41	ton	
Cladding material	low act. HT-9		
Specific power	52.8	W/g	
Power density	523.	W/cm ³	
Average Fuel Temperature	908	°C	
Maximum Clad Temperature	707	°C	
Dwelling time (eq. @ full power)	5.0	years	
Average Burn-up	100.0	GWd/t	

Breeder Core

Initial fuel mixture	ThO ₂		
Initial fuel mass	5.6	ton	
Cladding material	low act. HT-9		
U ²³³ stockpile at discharge	242.7	kg	
Power density at end cycle	3.0	W/g	

Primary cooling system

Approximate weight of the coolant	10,000	ton	
Pumping method	Nat. Convection		
Height convection column	25	m	
Convection generated primary pressure	0.637	bar	
Heat exchangers	4 × 375	MW	
Decay heat removal	RVACS		
Inlet temperature, Core	400	°C	
Outlet temperature, Core	600	°C	
Coolant Flow in Core	53.6	ton/s	
Coolant speed in Core, average	1.5	m/s	

Decay Heat Passive Cooling (RVACS)

Riser channel gap width	18	cm	
Downcomer channel gap width	57	cm	
Trigger Temperature	500	600	700 °C
EA Coolant max Temperature rise	110	83.5	64.5 °C
Time to max. Temperature rise	17.5	11.2	9.5 hours
Outlet air Temperature (@ max. temp.)	273	302	334.3 °C
Outlet air Speed (@ max. temp.)	13.4	14.2	15.2 m/s
Air flow Rate (@ max. temp.)	52.8	56.1	60 m ³ /s
Extracted Heat (@ max. temp.)	8.57	9.65	10.84 MW

Figure 2.1: General scheme of the particle Accelerator. Note the modest scale of the components. For more details we refer to Ref. [1].

Figure 2.2: General layout of the Energy Amplifier/Beam Dump [1].

Figure 2.3: Principle diagram of the Energy Amplifier complex [1].

Figure 2.4: Ingestive radiotoxicity in the "waste" of the Secular Repository as a function of time. The corresponding values for a standard open cycle PWR is also shown.

For simplicity we have used in this paper the same parameters of Ref. [1] namely a nominal unit power of $1500 \text{ MW}_{\text{th}}$. The system is modular and several units can be clustered if needed. The fuel is mixed oxide (MOX) clad with low activity steel¹⁹, essentially identical to the one used in Fast Breeders. The operating temperature is of the order of $650 \text{ }^{\circ}\text{C}$.

The fuel (see Figure 2.3) is kept sealed inside the EA for an extended burn-up, typically in excess of $100 \text{ GW} \times \text{day/t}$, and then “regenerated” removing the fission products (FFs), and topping it with additional fresh primary fuel which in the simplest mode is natural Thorium. The power produced is controlled by the proton beam current and there are no control bars. The device is operated at all times well below criticality and it is intrinsically safe against melt-down or other thermal run off due to malfunctioning of the accelerator or other unplanned accidents.

All produced Actinides are re-injected in the EA in an indefinite, closed cycle. In a more advanced version, FFs are also reprocessed in order to extract some of the very long lived radio-nuclides (^{99}Tc , ^{129}I , etc.) which are also “incinerated” in the EA, thus further reducing the activity of the FFs after $500+1000$ years. Even without this precaution, the (ingestive) radio-toxicity after a first-cool down period of about 600 years is reduced to about 5×10^{-5} of the one of the waste from an ordinary PWR for the same produced energy (Figure 2.4).

3.- Plutonium incinerating mode of the EA

The EA can operate with a variety of fuels. In the present application we start with a mixture of “Dirty Plutonium”, which is the standard discharge from a PWR, topped with natural Thorium. The assumed composition of the “Dirty Plutonium” is given in Table 3.1. However the performance of the EA is largely independent of the exact mixture used [1]. The neutron inventories (Table 3.2) are only slightly different from the ones with pure Thorium operation [1]. The neutron spectra are essentially the same as for standard EA operation (Figure 3.1) and change insignificantly during the evolution of the fuel composition, from the initial load to the asymptotic mixture. The corresponding effective cross sections which are used as input to the evolution calculations [1] are listed in Table 3.4.

¹⁹ The exact choice of the cladding depends on the effects of Lead corrosion, which needs further investigations (see Ref. [1]).

Table 3.1. Assumed composition of "Dirty Plutonium". From Ref. [11]

Radio-Nuclide	Relative Concentration	Radio-Nuclide	Relative Concentration
²³⁷ Np	0.075066	²⁴¹ Am	0.48572E-02
²³⁹ Np	0.75434E-08	²⁴² Am	0.43789E-04
²³⁶ Pu	0.92361E-06	²⁴³ Am	0.91257E-02
²³⁸ Pu	0.02204	²⁴² Cm	0.48940E-03
²³⁹ Pu	0.52988	²⁴³ Cm	0.72123E-05
²⁴⁰ Pu	0.21747	²⁴⁴ Cm	0.33522E-02
²⁴¹ Pu	0.10193	²⁴⁵ Cm	0.20386E-03
²⁴² Pu	0.03550	²⁴⁶ Cm	0.22925E-04

The relevant safety parameters are listed in Table 3.3. The most notable difference is that the void coefficient $\Delta k/(\Delta \rho/\rho)$ notoriously positive for standard EA operation is now negative, and it amounts to $\Delta k/(\Delta \rho/\rho) = -0.035$ at the beginning of the fill, when no ²³³U is present²⁰. Such a void coefficient²¹ is far less of a concern than in a conventional Sodium cooled Fast Breeder since (1) the boiling point of Lead is very high (1743 °C at normal pressure) and (2) the sub-criticality margin of the EA is vastly larger than the maximum change due to extreme void effect. However it cannot be excluded that an accidental loss of the molten Lead might occur²².

In such a hypothetical event, part or the totality of the core and the surrounding region will be made void of Lead coolant and moderator. We have simulated the emptying of the beam dump in a variety of conditions and looked at the changes in the reactivity coefficient k . (Figure 3.2). We find that during the emptying process the resulting void effect is always negative. We also remark that, as expected, at the end of the cycle (EOC), because of the beneficial presence of the ²³³U the reduction is more pronounced than at the beginning of the cycle (BOC).

²⁰ Evidently during operation while the initial Plutonium load is slowly incinerated, the newly bred ²³³U moves the operating point slowly towards the standard EA operating conditions, therefore improving the void coefficient.

²¹ Here the void coefficient is defined as the variation of k due to a relative change in coolant density ρ . A positive value implies that k decreases with decreasing density.

²² We underline the extreme unlikelihood of such a happening. A small crack in the vessel (doubly lined) will produce a leak which will stop as soon as the molten Lead flow comes into contact with cold environment, since the solid Lead will act as a plug. Only a massive destruction of the container and of the concrete walls of the pit will permit the loss of a major fraction of the coolant. Note the large excess of molten Lead above the core, which has to be lost before the core becomes exposed. The whole installation is seismically protected. [1].

Table 3.2 : Neutron capture inventory of the F-EA for the Thorium-"Dirty Plutonium" cycle.
(first fill ; $k_0 = 0.9896$; 16.5% MOX).

Zone-Wise			Fraction
Core			0.8970
Blanket			0.0425
Plenum			0.0257
Diffuser			0.0279
Beam Tube + Window			0.0005
Main Vessel			0.0064
Leakage			0.0018
Material-Wise			Fraction
Fuel (Th + U)			0.8577
Breeder (Th)			0.0395
Lead of which	percentage	Abs. Fraction	
<i>Diffuser</i>	(46.81 %)	0.0279	
<i>Plenum</i>	(12.08 %)	0.0072	
<i>Core</i>	(39.09 %)	0.0233	
<i>Blanket</i>	(02.01 %)	0.0012	
Lead Total			0.0596
Structures of which	percentage	Abs. Fraction	
<i>Cladding</i>	(84.06 %)	0.0364	
<i>Window</i>	(01.15 %)	0.0005	
<i>Main Vessel</i>	(14.78 %)	0.0064	
Structures Total			0.0433
Leakage			0.0018

Table 3.3: - Safety-related physics parameters of the F-EA.

	Th-U	Th-Dirty Pu Asymptotic		Th-Weapon Pu
	Asymptotic			Asymptotic
	EOC	BOC	EOC	BOC
Doppler Coefficient, $\Delta k/\Delta T$	- 2.27E-5	- 1.12E-5	- 1.38E-5	- 1.66E-5
Change in % $\Delta k/k$, emptying				
Top Core	- 0.1029	- 0.0552	- 0.0714	- 0.0680
Half Core	- 4.0513	- 2.3235	- 3.1485	- 2.9560
Full Core	- 6.4844	- 0.3468	- 2.9342	- 2.3909
All Regions	- 11.923	- 4.1138	- 7.4858	- 6.7542

Figure 3.1: Neutron spectra in Fuel Core for different fuels in the EA. Spectra are normalized to equal integrated flux.

Figure 3.2: Relative variation of the multiplication coefficient k as a function of the fraction of the vessel voiding in a variety of conditions. (EOC), (BOC) are respectively the end and the beginning of the cycle. There is a small difference between the fillings with PWRs Plutonium waste (Dirty Pu) and the one with surplus Weapon Plutonium (Wea Pu). The case of the standard Thorium-Uranium operated EA (Th-U) is also shown as a reference.

Table 3.4: - Averaged cross sections (barn) of Actinides relevant to the fast EA.

Element	Capture	Fission	Elastic	(n->2n)	(n->n')	Total
²³⁰ Th	0.167803	0.025697	13.218194	0.001447	1.123411	14.536536
²³² Th	0.308828	0.008570	10.087153	0.001402	0.819348	11.073583
²³¹ Pa	2.957017	0.225269	8.817699	0.000978	1.213704	13.214634
²³³ Pa	0.976008	0.055470	7.792018	0.000407	1.833241	10.657163
²³² U	0.652538	2.068427	8.993785	0.000750	0.487607	12.203108
²³³ U	0.257299	2.684459	7.637661	0.000746	0.419488	10.999662
²³⁴ U	0.559472	0.310451	9.705321	0.000150	0.791574	11.366984
²³⁵ U	0.507261	1.813811	8.607527	0.000965	0.688819	11.618405
²³⁶ U	0.410061	0.088795	10.565082	0.000727	0.970194	12.034825
²³⁷ U	0.417513	0.618110	8.794017	0.001956	0.584155	10.415781
²³⁸ U	0.308613	0.036678	10.165911	0.001064	0.913203	11.425482
²³⁷ Np	1.470774	0.296474	8.992845	0.000231	0.699920	11.459522
²³⁸ Np	0.400021	2.091224	8.931683	0.000723	0.285945	11.709574
²³⁹ Np	1.827500	0.430915	8.857689	0.000319	0.925176	12.041578
²³⁶ Pu	0.356359	1.424113	9.138505	0.000296	0.346369	11.265685
²³⁸ Pu	0.513813	1.073784	9.522254	0.000077	0.484591	11.593491
²³⁹ Pu	0.461561	1.751134	8.810845	0.000290	0.636930	11.640947
²⁴⁰ Pu	0.468515	0.353373	9.840660	0.000225	0.646495	11.248711
²⁴¹ Pu	0.521293	2.426248	8.400412	0.001785	0.432803	11.780483
²⁴² Pu	0.425562	0.239415	10.411948	0.000573	0.740759	11.809869
²⁴³ Pu	0.356255	0.779540	8.941603	0.004175	0.738341	10.819971
²⁴⁴ Pu	0.185471	0.200118	10.316672	0.001769	0.904391	11.608458
²⁴¹ Am	1.753717	0.246073	9.221643	0.000016	0.628412	11.849841
²⁴² Am	0.568606	3.072716	8.130913	0.000364	0.335152	12.107742
^{241m} Am	1.401599	0.188868	9.556294	0.000071	1.038566	12.185221
²⁴¹ Cm	0.190791	3.025927	8.138980	0.000032	0.261699	11.617401
²⁴² Cm	0.291108	0.142550	9.883826	0.000017	0.805147	11.122590
²⁴³ Cm	0.352351	2.896037	8.587559	0.001300	0.268673	12.105916
²⁴⁴ Cm	0.819713	0.392233	10.052383	0.000347	0.581884	11.845643
²⁴⁵ Cm	0.301665	2.315751	8.479863	0.001682	0.934156	12.033182
²⁴⁶ Cm	0.214099	0.237992	10.388494	0.000473	0.848259	11.689314
²⁴⁷ Cm	0.309619	1.923196	8.803992	0.002529	0.439674	11.479051
²⁴⁸ Cm	0.238365	0.274690	10.860856	0.000578	0.884598	12.259142
²⁴⁹ Bk	1.247552	0.150108	9.798133	0.000135	1.282295	12.478257
²⁴⁹ Cf	0.615083	2.538316	8.764503	0.000441	0.489086	12.407440
²⁵⁰ Cf	0.381355	1.031658	8.567427	0.000943	0.544598	10.526023
²⁵¹ Cf	0.303871	2.307346	8.484874	0.002975	0.503585	11.602709
²⁵² Cf	0.309327	1.337872	10.545552	0.000414	0.581779	12.774944

There is a small difference between the fillings with PWRs Plutonium waste (Dirty Pu) and the one with surplus Weapon Plutonium (Wea Pu). We conclude that even in extreme conditions the behaviour of the EA is safely sub-critical.

The maximum duration of the burn-up is determined by many practical considerations like the radiation damage effects on the fuel and the emergence of FFs captures. The duration of each fuel cycle T_{cycle} is related to the length of the burn-up B_o by the average power density ρ in the fuel, $T_{cycle} = B_o / \rho$. The mass of the fuel M_{fuel} is determined by the total power P to be produced, $M_{fuel} = P / \rho$. In a first approximation the incineration rate $\dot{M}_{inci} = dM_{inci} / dt$ is only dependent on P , namely independent of ρ and M_{fuel} for a given B_o . A large variety of choices in these parameters is therefore possible. In the energy production version of Ref. [1], a relatively low fuel power density has been chosen $\rho \approx 55$ watt/gr (oxide), in order to ensure a maintenance free condition for a long period, $T_{cycle} = 5$ years and $M_{fuel} = 25$ ton. In our specific application shorter cycles may be advisable. In this case one might adopt the value $\rho \approx 100$ watt/gr (oxide) which is ordinarily used in the Fast Breeders (Superphenix, MONJOU, EFR, ALMR etc.) and therefore $T_{cycle} = 3$ years and $M_{fuel} = 15$ ton). As already pointed out, these choices do not affect the incineration rates, as long as the nominal power is set.

The evolution programme used to make predictions of this work takes into account the neutron losses elsewhere than in the fuel according to Table 3.2 and the emergence of the Fission Captures, treated as a neutron sink. A fraction of the neutron inventory can be preassigned for incineration of long lived FFs. At this stage this feature is also treated as a neutron sink. For more details we refer to Ref. [1], where a more sophisticated method based on full Montecarlo simulation has been also used. It has been shown that the presently used evolution method is in good agreement with the full Montecarlo method and that its precision is adequate for the present analysis. In the programme

- i) We specify first the initial value of the multiplication coefficient k_o . The initial concentration of "Dirty Plutonium" topped up with ^{232}Th is set accordingly.
- ii) The burning is then followed for a specified period of time at a constant power density or constant neutron flux or at constant accelerator current, as preferred.
- iii) At the end of the cycle the fuel is "discharged", cooled for a specified time period (with only decays active) and a new refill is prepared with (1) the full Actinide residue from the preceding run minus the Uranium isotopes which are separately recovered, (2) a fresh amount of "Dirty Plutonium" chosen in order to start with the initially chosen multiplication coefficient k_o ; (3) fresh ^{232}Th to reach correct total initial fuel mass.

We repeat then for an arbitrary number of cycles points ii) and iii) in succession, thus following the incineration procedure. Results of the calculations are

now discussed. They show that the short and long term evolutions are extremely well behaved.

- i) The multiplication coefficient k remains remarkably constant over a long burn-up, because of the compensation between the rising amount of ^{233}U , the loss of mass and fissionable mass of the higher Actinides and the growing capture rate of the FFs. This is visualized in Figure 3.3. The duration of the cycle is therefore determined primarily by the emergence of radiation damage mainly in the fuel cladding and by the fact that the Pu to ^{233}U conversion becomes less efficient since the fraction of ^{233}U which is burnt in the EA grows with time. We have chosen as a guideline $120 \text{ GW} \times \text{day/t}$, but larger values may be attainable.

Figure 3.3: Variation of the multiplication coefficient k with and without FFs captures. The stabilizing effect of the captures is evident. The choice of the initial value of k is arbitrary.

- ii) The continuous recycling with periodic addition of fresh “Dirty Plutonium” can be repeated indefinitely without loss of criticality coefficient k (Figure 3.4) and with a steady amount of fresh additions (Figure 3.5). The first load is evidently larger than the others, since significant reactivity is maintained in the Actinide residue which is transferred from the previous cycles.
- iii) Asymptotic mixture of the fuel is reached after a few cycles and maintained indefinitely. The content of Actinides (Figure 3.6) after an initial growth,

Figure 3.4: Variation of the multiplication coefficient k as a function of burn-up for different, successive fillings and a specified, arbitrary initial value.

Figure 3.5: Fractional composition at the beginning of the cycle, as a function of the cycle number. The amount of (1) new "Dirty Plutonium", (2) the residue from the previous cycle, excluding Thorium, (3) the fresh Thorium addition are shown.

Figure 3.6: Fractional weight in the fuel of Actinides as a function of the cycle number. Asymptotic concentrations are evident. Note that notwithstanding the changes in composition, the multiplication coefficient remains remarkably constant (Figure 3.4).

stabilizes in the residue because of equilibrium between incineration due to fissions and production due to captures in the father element(s).

- iv) The operation of the EA is normally controlled by the intensity of the beam current from the Accelerator. The beam current variation for the constant power of 1500 MW_{th} is shown in Figure 3.7. With the exception of the first fill which requires a somewhat higher current and could eventually be run at somewhat reduced power, an accelerator current with a nominal value of some 12 mA and a maximum value of about 18 mA is required. This is well within the design parameters of the accelerator of Ref. [1].
- v) The power produced is primarily due to fissions. In Figures 3.8a and 3.8b we give the fractional power contribution for each of the elements in the mixture as a function of the burn-up. For definiteness we show the case of the asymptotic mixture, but the behaviour is not very different for the early fillings. At the beginning of the cycle, burning is concentrated on the Plutonium (fresh and remnants) and on the higher Actinides. During the cycle, while the reactivity of these elements decreases, freshly produced ²³³U takes over. At the end of the

Figure 3.7: Accelerator current in mA for protons of 1 GeV kinetic energy in order to maintain the delivered power to the constant value of 1.5 GW. Note that the first fill requires a larger current: it may be preferable to operate at somewhat lower delivered power.

cycle, some 45% of the produced power is due to ^{233}U . The integrated fractional power produced by the burning of the Thorium/Uranium component amounts to 0.30 of all fissions²³ after 120 GW $\times$ day/t, corresponding to about 70% of the theoretical limit²⁴ of pure “Dirty Plutonium” burning (Figure 3.9).

The scheme allows continuous supply and incineration of “Dirty Plutonium” with an essentially constant residue which is continuously recycled. The accumulated amount of incinerated material and of the fresh ^{233}U as a function of the number of cycles performed is shown in Figure 3.10. We remark the almost linear elimination rate of the “Dirty Plutonium”, amounting to 9.20% of the initial metal fuel mass for each cycle (120 GW $\times$ day/t) and the almost linear production rate of ^{233}U amounting to 5.801 % of the fuel mass. Hence the Pu $\rightarrow$ U conversion efficiency is 63%. The isotopic composition of the produced Uranium is primarily ^{233}U with noticeable contaminations of ^{232}U (4.63×10^{-3}), ^{234}U (0.0859), ^{235}U (6.83×10^{-3}) and ^{236}U (1.00×10^{-3}). Once the produced Uranium is diluted with ordinary Uranium to constitute new fuel for the PWRs, the relative concentration of the γ -emitting ^{232}U drops to about 100 ppm.

²³ The fraction of the power produced by Thorium fissions alone is $\approx 2\%$.

²⁴ We recall that the complete fission of 1 ton of “Dirty Plutonium” will produce 940 GWatt $\times$ day.

Figure 3.8a,b : Fractional power contributions (fissions) as a function of the burn-up for an asymptotic fuel composition. (a) large concentrations; (b) small concentrations.

At a constant energy production rate of $1500 \text{ MW}_{\text{th}}$ the EA will accumulate a yearly energy production of $547.5 \text{ GW} \times \text{day}$, corresponding to the elimination of $547.5/120 \times 0.0920 = 0.420$ tons of “Dirty Plutonium” independently of the actual amount of fuel mass. The correspondingly produced Uranium is $547.5/120 \times 0.05801 = 0.265$ tons which can be used to manufacture about 13 tons of PWR fuel with 1.87 % of ^{233}U . We remark that for a nominal 1000 MW_e PWR²⁵ [10], corresponding to an integrated thermal energy of $900 \text{ GW} \times \text{day}$, the burnt fuel Uranium metal mass is 27.271 ton/year and the total “Dirty Plutonium” (namely Pu, Np, Am and Cm) accumulated amounts to 0.271 Ton/year [11].

The main conclusions of these calculations are that a single ($1500 \text{ MW}_{\text{th}}$) EA module operated as an incinerator:

- i) It can match the total Pu, Np, Am and Cm production rates of $0.420/0.271 = 1.55$ nominal (1000 MW_e) PWRs and in addition

²⁵ Basis for the calculations are a thermal efficiency of 0.325, a dwell time of 1100 days, a burn-up of $33 \text{ GWatt} \times \text{day/t}$, and a capacity factor 0.80. The fuel contains initially 3.3% of ^{235}U . These exemplificative figures come from Ref. [11].

- ii) It produces through the $\text{Pu} \rightarrow \text{U}$ conversion enough fresh fuel for $13/27.271 = 0.48$ PWRs.
- iii) Incineration of the “Dirty Plutonium” of PWRs will finally produce an incremental energy in the EAs equal to $547.5/(900 \times 1.55) = 0.392$ of the one already produced in the PWRs. We believe that this extra energy is produced to very competitive costs.
- iv) Ample allowance (10% of the full neutron stockpile) has been set aside for FFs incineration. Although the method is not completely perfected [12] we expect that it should be sufficient to eliminate the main long lived FFs produced namely ^{99}Tc , ^{129}I and although with previous mass separation ^{135}Cs .

For more details on possible combined strategies we refer to paragraph 5.

Figure 3.9: Fractional power generated by the “Dirty Plutonium” and its descendants, averaged over the fuel cycle, for different final burn-ups. For small burn-ups almost all the power comes from Pu burning, the small difference ($\approx 2.5\%$) being due to fissions in Thorium. At larger final burn-ups, burning in the freshly produced Uranium contributes significantly.

Figure 3.10: Cumulative amount of incinerated material and of the fresh ^{233}U as a function of the number of cycles.

4- Fuel utilization in an ordinary PWR

The stockpile of the produced ^{233}U accumulated in the discharge of the EA constitutes a valuable asset. At any rate it represents a substantial toxicity and it must be burnt. We shall consider here the possibility of using it as a fuel in an ordinary PWR. In short the PWR is used to incinerate the ^{233}U "waste" from the EA, while the EA will incinerate the "Dirty Plutonium" from the PWR and eventually some of the long lived FFs produced by both units.

Since the purpose of the exercise is the one of having no Actinide in the repository, following Figure 1.1, the idea we pursue is the one of recycling the spent fuel from the PWR in the same way as for the EA, namely dividing it in three streams:

- (1) all Actinides other than Uranium and eventually some of the most offending long lived radio-nuclides are recycled in the EA-fuel,
- (2) the FFs are disposed in the secular repository and
- (3) the Uranium is re-used for new PWR-fuel fabrication.

The PWR is evidently an existing machine. It is important that the minimum changes from ordinary operation are introduced. Recycling the bulk of the Uranium at each new fuel fabrication is reasonable since the ^{233}U enriched fuel has already many of the features of the customary MOX, including the presence of a substantial γ -activity, primarily due to the already mentioned presence of ^{232}U . The programme ORIGEN [13] has been used to compare the performance of this new and of the ordinary PWR fuels. As reference for ordinary fuel we have taken 3.3% ^{235}U -enriched Uranium in the form of Uranium Oxide. The nominal burn-up has been set to $33 \text{ GW} \times \text{day/t}$ and the power is kept constant at all times. The dwell time of the fuel in the Reactor is three years, with three separate refill phases, performed once a year²⁶. The utilization factor is 0.80. We shall consider the first fill and the evolution of the fuel composition due to recycling in the PWR.

- (1) For the first fill we have adjusted the concentration of the Uranium produced by the EA (^{233}U -rich mix) in a natural Uranium fuel in order to obtain the same average reactivity $\langle k_{\infty} \rangle = 1.165$ as the reference fuel over the specified burn-up period.

²⁶ At each refill the oldest third of the fuel is extracted and substituted with new fuel.

- (2) For the subsequent fills, the bulk of Uranium comes from the previous discharge. The amount of ^{233}U -rich mix is kept constant and the fuel mass is preserved by injecting an appropriate amount of fresh Uranium²⁷. As already pointed out, after several refills, element concentrations tend to equilibrium values.

It is a very fortunate circumstance that the behaviour of this "asymptotic mixture" in a PWR is very similar to the one of the initial fill. It is this feature that makes our scheme possible and attractive when compared with other procedures which suffer from a rapid "decay" of the properties of the recycled fuel. The relevant yearly quantities, normalized to 1 GW nominal thermal power are given in Table 4.1 for successive fuel cycles. The multiplication factor is remarkably independent of the fuel cycle²⁸, as shown in Figure 4.1.

Figure 4.1: Dependence of the multiplication coefficient k_{∞} half way through the fuel cycle. The asymptotic level is also shown.

The differences of the reactivity k_{∞} as a function of the burn-up within the cycle is shown in Figure 4.2, for the first and the 11-th combined cycle (representative of an asymptotic mixture)²⁹ and compared with the ^{235}U reference case. The difference in behaviour between the two fuels and the standard reference case are extremely small and they become even smaller with the asymptotic mixture. The slightly higher value of k_{∞} of ^{233}U and its slightly faster fall-off are further compensated when one considers fuel shuffling exercises which are needed to uniformize the reactivity of the

²⁷ Since the composition of the bulk of the recycled Uranium is ^{238}U , the exact amount of fresh Uranium is not very critical and it is mainly added for mass conservation. In the calculations we have used Natural Uranium but depleted Uranium could be used instead.

²⁸ It could be eventually further trimmed changing slightly the ^{233}U -Mix, although it does not seems required, since the control bars could easily correct this small difference.

²⁹ Note that each fuel cycle will last at least 3 years of dwell time in the PWR and 1 year for reprocessing. Hence the 10 cycles will span over 40 years which corresponds to the lifetime of a PWR.

fuel during burn-up. Variations of Uranium concentrations during burn-up are displayed in Figure 4.3. Note that at the 11-th cycle and beyond concentrations of ^{234}U , ^{235}U and ^{236}U remain essentially constant, as the resultant of burning and formation by neutron capture.

Table 4.1: Inventory of PWR operation at initial load and after $33 \text{ GW} \times \text{day/t}$.

	Cycle Number									
	0	1	2	3	4	5	6	7	8	10
<i>General parameters</i>										
k_{∞} @ 1/2 burn	1.166	1.156	1.153	1.154	1.156	1.158	1.159	1.16	1.161	1.161
Flux $\times (10^{14})$	3.36	3.22	3.19	3.14	3.10	3.07	3.05	3.03	3.02	3.01
<i>Inventory at initial load, Kg/ GW(t)</i>										
Natural U		195.7	196.5	197.8	197.4	197.4	197.1	196.1	196.4	197.2
^{233}U -Mix	189.8	189.8	189.8	189.8	189.8	189.8	189.8	189.8	189.8	189.8
<i>Inventory at discharge, Kg/ GW(t)</i>										
Actinides $\neq \text{U}$	84.75	85.33	85.20	84.91	84.70	84.74	84.81	84.97	85.11	85.43
He,Pb,Th	0.023	0.032	0.037	0.039	0.040	0.041	0.042	0.042	0.042	0.043
Np	1.583	2.191	2.496	2.775	3.064	3.351	3.621	3.874	4.098	4.473
Pu	81.29	81.32	80.89	80.32	80.00	79.73	79.56	79.46	79.39	79.31
Am	1.591	1.578	1.539	1.494	1.456	1.428	1.408	1.394	1.384	1.372
Cm	0.262	0.258	0.245	0.231	0.219	0.211	0.204	0.200	0.197	0.194
FF -Total	301.2	300.9	300.9	301.2	301.2	301.2	301.2	301.2	301.2	301.2
^{99}Tc	6.32	6.23	6.23	6.26	6.26	6.26	6.28	6.28	6.28	6.28
^{129}I	2.23	2.33	2.35	2.34	2.34	2.33	2.33	2.33	2.33	2.33
^{135}Cs	2.59	2.58	2.60	2.63	2.65	2.65	2.67	2.67	2.67	2.67

It is well known that a substantial amount of the produced energy is due to fissions in Plutonium, rather than Uranium. We display in Figure 4.4 the relative contributions of these two elements as a function of the burn-up. It appears that for the first fill the contribution due to Plutonium is slightly larger than the reference case. For the subsequent cycles, since the concentration mix of the fuel is getting richer with higher Uranium isotopes, the effects of Plutonium are slightly smaller. All these differences are small and do not represent problems. The Plutonium

isotopic composition during burn-up is extremely similar for all cases, since it is driven by the ^{238}U bulk material. Neptunium production is about 50% the one of the standard fuel.

Figure 4.2: Reactivity k_{∞} as a function of the burn-up within the cycle for the first, the 11-th cycle (representative of an asymptotic mixture) and for the ^{235}U reference case.

Figure 4.3: Uranium isotopic concentrations a function of burn-up for the first, the 11-th cycle (representative of an asymptotic mixture) and for the ^{235}U reference case.

Looking in more detail at the evolution of the fuel composition as a function of the fuel cycle, we give in Figure 4.5 the fuel composition at the load of the n -th cycle, as a function of n . The ^{235}U concentration drops from the initial concentration of natural Uranium and settles to a level of $\approx 0.32\%$, very close to the one of ^{236}U (0.34%). The other noticeable isotopes, besides the bulk of ^{238}U are: ^{234}U (0.09%) and the highly toxic and γ -emitting ^{232}U (2.0×10^{-4}).

Figure 4.4: Relative contributions to produced energy of Uranium and Plutonium isotopes as a function of the burn-up for the first, the 11-th cycle (representative of an asymptotic mixture) and for the ^{235}U reference case.

Finally since the aim of the exercise is the one of eliminating the Plutonium “waste”, we briefly mention the possibility of replacing also in the PWR the Uranium fuel matrix with a Thorium matrix. In this way the Plutonium production will be minimized at production and the role of the EA limited to the fresh fuel production for the PWR through $\text{Pu} \rightarrow \text{U}$ conversion of some, already existing Plutonium. The initial PWR fuel for the first fill is made of Thorium Oxide mixed with ^{233}U -rich Oxide mix from the EA in a percentage of 3.651% , chosen (as previously) such as to equalize the mean k_{∞} to the one of the reference mixture. The dependence of the fuel reactivity is very close to the one of the reference case (Figure 4.6). At the end of the cycle, Fission Fragments are separated from Actinides with the help of reprocessing and destined to the Secular Repository. The fuel for the next cycle is made with the total Actinide residue from the previous cycle (with eventually some of the most toxic long lived FFs), topped with fresh Thorium in order to maintain the active mass of fuel constant. However only a smaller fraction of ^{233}U -rich Oxide mix must be added, since some of the ^{233}U is bred in the PWR during the burn-up. The

concentration of ^{233}U depends on the balance between the amount burnt and the one bred through the process

. As well known for thermal neutrons and at sufficiently small neutron flux the relative ^{233}U concentration tends to a breeding equilibrium of $C_{\infty} = 1.35\%$ [1]. In the present configuration the externally added fraction is important ($C_0 = 3.29\%$ of ^{233}U) and the concentration $C(t)$ tends as a function of time roughly exponentially to the breeding equilibrium, namely $C(t) = (C_0 - C_{\infty})\exp(-\lambda t) + C_{\infty}$, as shown in Figure 4.7. The exponential constant (fitting of ORIGEN prediction) is $1/\lambda = 38.82 (\text{GW} \times \text{day/t})^{-1}$. The net fraction of the ^{233}U which has been burnt and which has to be replaced by a fresh

supply from the EA is therefore $\Delta C(T) = (C_0 - C_{\infty})(1 - \exp(-\lambda T))$, where T is the design fuel burn-up. With our parameters $\Delta C(T) = 1.11\%$, namely only 33.8% of the initial load. The ^{233}U -rich mix from the EA in the PWR has therefore an energy yield with a Thorium matrix which is 2.96 times better than with a Uranium matrix. The trans-uranic Actinides produced are very small³⁰, and they can be safely neglected³¹.

However, operation with a Thorium based fuel of an ordinary PWR is significantly more complicated, since the long lifetime of the ^{233}Pa produces, as well known, a reactivity variation as a function of shut-down and of power variations which has a magnitude and features which resemble the well known Xe-effect. As already pointed out this effect is absent in Fast Neutron devices and in particular in the EA. The phenomenology is well described in the literature [11]. The maximum change in reactivity switching off the Reactor (which grows with the 27 days ^{233}Pa half-life), is approximately equal to the ratio of ^{233}Pa and ^{233}U concentrations, $(\Delta k/k)_{T=\infty} \approx N(^{233}\text{Pa})/N(^{233}\text{U})$ and it is shown in Figure 4.8 as a function of the burn-up.

Figure 4.5: Initial fuel Uranium isotopic concentrations as a function of the cycle number.

³⁰ At the end of the cycle: Np (75.87 gr/GWatt(t)) and Pu (21.29 gr/GWatt(t)). Other Actinides (Am, Cm, etc.) are truly negligible.

³¹ They could eventually be separated and incinerated in the EA with negligible consequence to the inventory stockpile.

Figure 4.6: Reactivity k_{∞} as a function of the burn-up within the cycle for the first cycle with Thorium Oxide Matrix and for the ^{235}U reference case. The ^{233}U -rich mix from the EA is mixed at the percentage of 3.651 %.

In the most unfavourable conditions it may amount to a $\Delta k/k = +0.08$ change with a time constant of $27 \times 1.406 = 38$ days. Finally at the neutron flux conditions³² of the present example, there is a significant probability that the ^{233}Pa may be captured before it decays $^{233}\text{Pa} + n \rightarrow ^{234}\text{Pa} \xrightarrow{\tau_{1/2}=6.7h} ^{234}\text{U}$, thus missing the highly fissionable ^{233}U step and causing a reduction in $\Delta k/k$. For a more quantitative analysis we refer to Ref. [1].

Finally the presence of ^{233}U in a Thorium bulk material could open the way to military diversions of the highly fissile ^{233}U by simpler chemical separation of the fuel. This question has been amply discussed in Ref. [1], to which we refer for more details. In short, the presence of a strong γ -emitter in the ^{232}U decay chain will make the realization of such a bomb very hard, at any rate harder than just extracting the "Dirty Plutonium" from the reactor fuel during standard operation. Hence the use of the EA-derived fuel should not increase the diversive risks with respect to the same PWR already operating with ordinary fuel. Should it not be considered sufficient, it is possible to "denaturate" the Uranium in the fuel [11], by adding a significant amount of ^{238}U in such a way that the resultant critical mass of the (chemically) extracted Uranium is considerably increased. In this way the production of Plutonium waste,

³² The neutron flux is $\phi = 2.7 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ at the beginning of burn-up and grows approximately linearly to $\phi = 3.8 \times 10^{14} \text{ cm}^{-2} \text{ s}^{-1}$ at the end of the cycle

Figure 4.7: ^{233}U relative concentration in the fuel as a function of the burn-up within the cycle for the first cycle with Thorium Oxide Matrix. The fit formula is given in the text. **Figure 4.8:** $^{233}\text{Pa}/^{233}\text{U}$ ratio of atomic concentrations in the fuel as a function of the burn-up within the cycle for the first cycle with Thorium Oxide Matrix.

although larger than without “denaturation” will be much less than the one with Uranium based fuel.

5- Mass flow in the combined EA-PWR incineration cycle

The general mass flow is shown in Figure 1.1. Values shown are for asymptotic conditions. We have chosen a cluster of 3 EA units for a total thermal power of 4500 MW and an electric power of 1870 MWe. The fuel burn-up is set to 120 GW $\times$ day/t.

The EA units are fed with fresh metal Thorium at the rate of 1.251 tons/y and “Dirty Plutonium” at the rate of 1.260 tons/y. They produce 794.5 kg of fresh metal Uranium³³ and 1.66 Tons of Fission Fragments (the slight mass defect is due to the energy produced). The fuel fabrication is based on known MOX-technologies and the chemical reprocessing is only required to separate out of the spent fuel the three main components, namely (1) Uranium (794.5 kg/y) (2) Actinides other than Uranium (8.771 ton/y of Th and 2.404 ton/y of Pu and other Actinides) and (3) the fission fragments (1.66 tons/y), fuel claddings plus miscellanea.

The PWR size has been chosen in order to match the rate of ²³³U-rich Mix from the EAs³⁴. The corresponding total thermal power is 4187 MW(t) for a utilization factor of 0.80 and a burn-up of 33 GW × day/t. The total amount of fuel of each early batch is 37.213 tons. The dwell time of each fuel batch in the PWR is three years.

As already mentioned, at each cycle fresh Uranium (approximately 825 kg) and new ²³³U-mix from the EA must be added in the fuel fabrication. We have kept constant the amount of ²³³U-mix and slightly adjusted the amount of added fresh Uranium in order to maintain the fuel mass constant. At the end of the cycle, by reprocessing, all other Actinides other than Uranium and the FFs are collected. The spent fuel (with a slight loss due to the mass to energy conversion) is reprocessed in the same way as the one of the EA and partitioned³⁵ in (1) Uranium (35.59 tons/y) (2) Actinides other than Uranium, called “Dirty Plutonium” (358 kg/y of Pu and other Actinides) and (3) the fission fragments (1.261 tons/y), fuel claddings plus miscellanea. The Actinides are actually subdivided as follows: Np (18.7 kg/y), Pu (332.3 kg/y), Am (5.74 kg/y) and Cm (0.81 kg/y).

The “Dirty Plutonium” locally generated is incinerated together with the surplus of external origin which has to be incinerated. Hence the “surplus” incineration capacity of the complex drops to 0.902 ton/year. The PWR nominal power associated to such a surplus Pu yield is of the order of $4187/358 \times 902 = 10.55$ GW(t), namely about the “waste” of 3 additional 3.51 GW(t) power stations in addition to the one already fed with ²³³U. In total a single 4.5 GW(t) EA can match the “waste” rate related to a total of $10.55 + 4.187 = 14.737$ GW(t) produced in PWRs.

³³ The actual isotopic composition is as follows: 716.35 kg/y of ²³³U; 3.68 kg/y of ²³²U; 68.25 kg/y of ²³⁴U; 5.43 kg/y of ²³⁵U; 0.79 kg/y of ²³⁶U. See Ref. [1] for more details.

³⁴ We do not mean that we must have 1 PWR which has exactly this power. This is the equivalent power which has to be generated with the manufactured fuel in order to burn all the produced fuel. In practice it can be achieved in a number of ways and with different mixtures of new and standard fuel.

³⁵ Actual numbers refer to the 11-th cycle. For more details we refer to paragraph 4.

Figure 6.1: Variation of the multiplication coefficient k as a function of burn-up for different, successive fillings and a specified, arbitrary initial value.

These are idealized conditions and are for illustration purposes. For more realistic schemes we refer to paragraph 1.

Other very interesting schemes are possible if, as already mentioned, the PWR is operated with a Thorium based Fuel. We recall that the ^{233}U -rich mix from the EA in the PWR has therefore an energy yield with a Thorium matrix which is 2.96 times better than with a Uranium matrix. Hence the same 14.737 GW(t) of power could be produced in PWRs using the ^{233}U -mix from the only one single EA module of roughly 1.50 GW(t)³⁶. The "Dirty Plutonium" incineration rate required to keep the system going will obviously be only $1260/3 = 420 \text{ kg/y!}$

6- Elimination of Military Plutonium

The possibility of extending our scheme to the elimination of military Plutonium is briefly considered. The Plutonium is now essentially pure ^{239}Pu , which implies a much higher fission cross-section. During incineration higher Pu isotopes and minor Actinides are formed and progressively incinerated. But since they are not supplied with the initial incinerable material, their concentration is significantly lower.

³⁶ Following Ref. [1] we have defined a EA power station a cluster of three modules.

The multiplication coefficient with periodic Pu fillings with initial concentration set to produce a specified initial value of the multiplication coefficient k_0 is very constant over an extended burn-up and many cycles, reaching an asymptotic concentration of acceptable properties (Figure 6.1). In this respect the two fillings of “Dirty Pu” and pure ^{239}Pu have rather similar behaviour. Required concentrations of the mix as a function of the cycle number is shown in Figure 6.2 . The accumulated amount of incinerated material and of the fresh ^{233}U as a function of the number of cycles performed is shown in Figure 6.3 .

Figure 6.2: Fractional composition at the beginning of the cycle, as a function of the cycle number.

Figure 6.3: Cumulative amount of incinerated material and of the fresh ^{233}U as a function of the number of cycles.

We remark the almost linear elimination rate, amounting to 7.56% (9.20%) of the initial metal fuel mass for each cycle (110 GW $\times$ day/t) and the almost linear production rate of ^{233}U amounting to 6.52 % (it was 5.801 % in the “Dirty Pu” case) of the fuel mass. Hence the $\text{Pu} \rightarrow \text{U}$ conversion efficiency is quite large, 86.18 % (it was 63%). The energetic efficiency in burning ^{239}Pu is of the order of 70% for a cycle lasting 110 GW $\times$ day/t. The contribution of various Actinides to the (fission) power is shown in Figures 6.4a and 6.4b. It is not very different than the corresponding figures for “Dirty Plutonium”.

At a constant energy production rate of $1500 \text{ MW}_{\text{th}}$ a single EA unit will accumulate a yearly energy production of $547.5 \text{ GW} \times \text{day}$, corresponding to the elimination of $547.5/110 \times 0.0756 = 0.376 \text{ ton/y}$ of ^{239}Pu independently of the actual amount of fuel mass and cycle duration. The correspondingly produced Uranium is $547.5/110 \times 0.0652 = 0.324 \text{ tons}$ which can be used to manufacture about 12.5 tons of PWR fuel with 2.3 % of ^{233}U . This means that the elimination of the totality of the 100 tons of ^{239}Pu will produce a total of $400 \text{ GW} \times \text{year}$ in the EAs. Two clusters of 3 EA units each for a total thermal power of 4500 MW and an electric power of 1870 MWe could *eliminate in about 45 years the 100 tons of unwanted ^{239}Pu* . An additional period of similar duration with the EAs operated as “amplifiers”, in which the ultimate burn-up is performed “parasitically” will ensure the final reduction of the residue to the level of few 10^{-3} of the initial mass ($\leq 200 \text{ kg}$) (Figure 1.7). The 86.18 tons of ^{233}U produced in the transformation can be used to produce as much $86.18/0.02375 = 3629 \text{ tons}$ of useful fuel for PWRs. The produced fresh fuel is sufficient to run a standard 1000 MW (electric) PWR for about 100 years.

Figure 6.4a,b Fractional power contributions (fissions) as a function of the burn-up for an asymptotic fuel composition. (a) large concentrations; (b) small concentrations.

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