THE AMSTER CONCEPT (ACTINIDES MOLTEN SALT TRANSMUTER)

J VERGNES P. BARBRAULT,D LECARPENTIER, Ph TETART
EDF Pole Industrie Division Recherche et Développement
Département Physique des Réacteurs
1 Avenue du Général de Gaule
92140 CLAMART   France
e mail: jean.vergnes@edfgdf;fr

H MOUNEY
EDF Pole Industrie Division Ingéniérie et Services
CAP AMPERE
93282 SAINT DENIS CEDEX France
e mail: henri.mouney@edfgdf;fr

ABSTRACT

In the frame of the French law related to the management of long lived radioactive wastes, a reactor concept aiming both to generate electricity and to minimize existing transuranics (TRU) and long lived fission product is proposed.

This concept named AMSTER is a Molten Salt Reactor, the molten fuel being a mixing of Transuranics and enriched Uranium.

The salt processing is integrated and allows both to supply U-235 and to separate fission products while minimizing transuranics losses.

Two options have been analyzed :
- a TRU self-burning option, releasing only a few grams/Twhe of TRU in the wastes for geological storage. It needs an equilibrium inventory of 1.6 metric tons/Twhe in the reactor and a consumption of 10 metric tons/Twhe of natural uranium (compared to 21 for PWR) ;
- a TRU incineration option which would permit, for example to burn the TRU present in PWR spent fuels at a rate of 28 kg/TWhe, for an inventory at equilibrium of 4.7 metric tons/Twhe of TRU. The consumption would be then of 2.2 metric ton/TWhe of PWR spent fuel and 3.7 metric ton/TWhe of natural Uranium.

1. INTRODUCTION

This study falls within the framework of French law concerning the primary area of research into the management of nuclear waste. This area covers separation and transmutation. The AMSTER (Actinides Molten Salt Transmuter) concept is part of the research program into a fission reactor minimising the production of waste, in other words a reactor for which the mass of transuranium elements (TRU) and long-lived fission products (LLFP) destined for geological storage will be as small as possible.

The residual TRUs and LLFP will comprise:

    i. the total TRU and LLFP losses during reprocessing and production of recycled fuels,

    ii. the TRU inventory in the cycle at the end of the scenario. The TRU inventory in the cycle is the quantity of transuranium elements needed to run the reactors recycling them.

    i. To reduce the total losses (i) multi-recycling of all the TRU and LLFP is required. Moreover, one must at the same time:

- process as little TRU and LLFP as possible, by using high burn-up rate fuels

- obtain a TRU and LLFP loss rate which is as low as possible.

ii To reduce the inventory in the cycle (ii) the mass of TRU and LLFP both inside and outside the active core must be reduced.

Six of the main TRU incinerating reactors examined so far were compared in a plutonium limitation scenario using either a single type of TRU self- burning reactor or two types of reactors: PWRs producing TRU and TRU incinerating systems.. This shows considerable dispersion of the TRU masses handled per TWh and of the inventories in the cycle per GWe. This study showed that:

- the lowest in-reactor inventories are obtained with the highly thermalised reactors
- the inert support enables the maximum theoretical incineration rate per TWh to be obtained, but considerably deteriorates the isotopic quality of the TRU. In association with highly thermalised reactors, it thus appreciably increases the TRU inventory in the reactor. It also significantly increases the loss of reactivity, which requires very frequent reprocessing, thereby increasing losses. For dynamic reasons (small fraction of delayed neutrons), it requires the use of a sub-critical accelerator driven system
- following this analysis, the AMSTER concept was proposed with the aim of cumulating the various advantages:

  1. the advantages of an IFR type integrated concept: no transport of highly active fissile materials outside the plant, simpler reprocessing

  2. highly thermalised reactors leading to much smaller fissile inventories in the active core, very high Pu isotope and LLFP capture fission cross-sections;

  3. continuously reprocessed molten salt reactors, reducing stray capture by the FP in the active core

  4. critical reactors, in which the power distribution is flat enough to minimise the active core volume;

  5. an enriched uranium support:

- allowing injection of reactivity into the core without increasing the TRU inventory,

- avoiding overly damaging the TRU isotopic quality by adding Pu 239 through capture on U238.

6. dry pyrochemical processing of the molten salt which, by eliminating insolubles, should save one or two decades in the reprocessing loss rates, as compared with solid fuels subjected to aqueous reprocessing.

The aim of this document is to describe the proposed reactor, to present the initial reactor calculation results, to evaluate its potential and list both its advantages and its drawbacks.

2. DESCRIPTION OF AMSTER

AMSTER is a continuously loaded, graphite moderated, molten salt critical reactor, using a Uranium 238 support, slightly enriched with U235 if necessary.

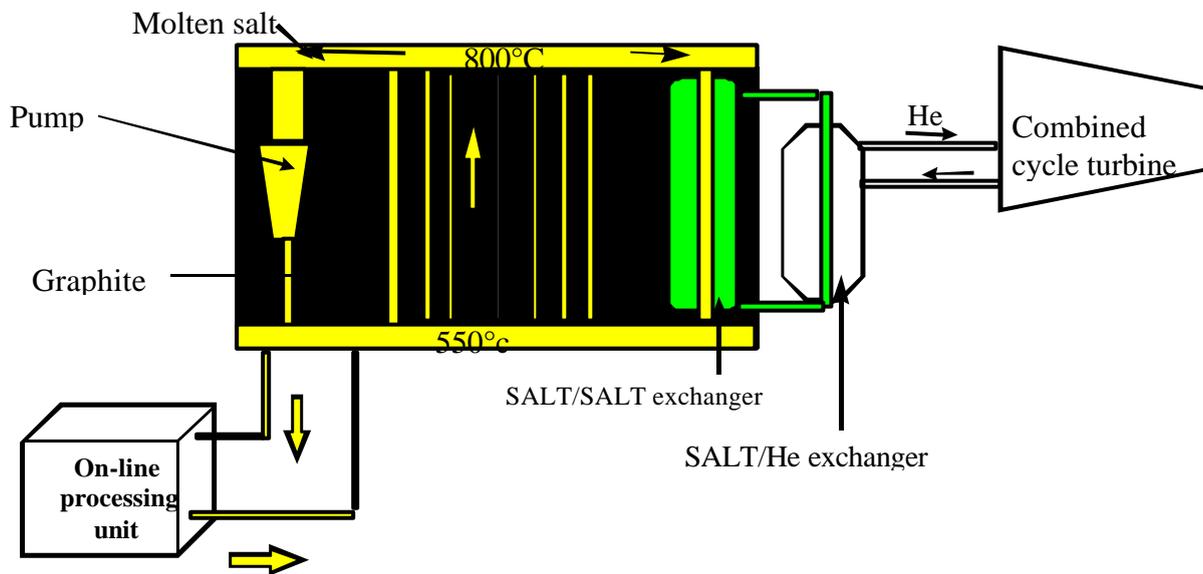Figure 1 gives the working principle of this type of reactor.

Figure 1: Schematic diagram of the Molten Salt Reactor

The reactor comprises three parts:
1. a thermal energy production part, consisting of the core and the primary circuit;
2. an on-line salt processing unit allowing separation of FP and introduction of heavy nuclei
3. a part converting the thermal energy produced, here consisting of a salt/He exchanger, turbine and generator

2.1 MODERATING ARRAY

The core of the molten salt reactor consists of an array of graphite hexagons identical to those of the Saint-Laurent A1 UNGG reactor, with a hole drilled for salt circulation. The diameter of the salt hole is 8 cm for a hexagon 13 cm on a side (see figure

Graphite of density
2.25 g/cm3

SALT
8 cm

13 cm
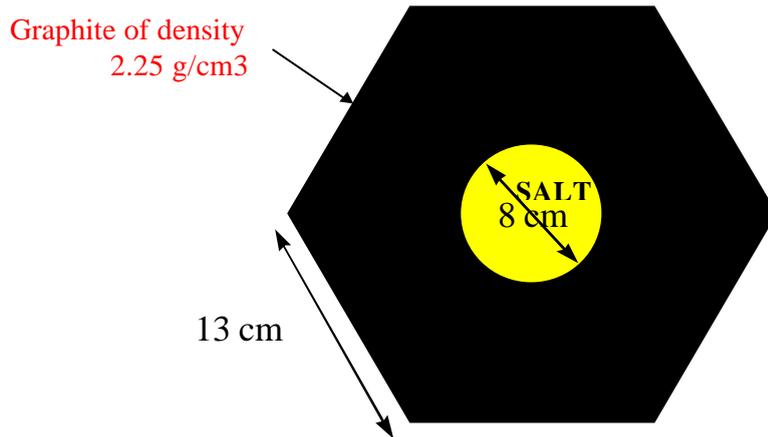
2)

Figure 2 Basic cell of the graphite array

The salt used is of the same type of that of the MSBR detailed project [3]. The molar composition is

61 Li F-21 BeF$_2$-18 NL F$_4$

2.2 OPERATING PRINCPLE

When the molten salt fuel penetrates the graphite array, it becomes critical and heats up. It enters at a temperature of about 500°- 600° C and leaves at a temperature of 700 to 800°C.

- The core entry temperature is set by the salt melting temperature, which itself depends on its composition (450 to 600°C).

- The exit temperature is set by the resistance of the materials other than the graphite (hastelloy).

Once heated, the salt is entrained by pumps and passes through salt/salt exchangers which extract the thermal energy thus produced. On exiting the core, an on-line processing unit collects a small fraction of the fuel to extract the fission products from it. This process is accompanied by the injection of new heavy nuclei, U235 or U238, or TRU, to replace the fissioned heavy nuclei.

The secondary salt heats either steam, or helium, which supplies either a combined cycle turbine and a generator, or other high temperature applications

2.3 FUEL MANAGEMENT OPTIONS

A large number of fuel management options are possible with this type of reactor. They can be characterised by:
1. the elements used in the initial composition of the salt
2. the operations performed at the moment the salt is processed:
3. the elements multi-recycled in the reactor:
4. the elements released as waste

In this document, two management options are considered, defined in the following sub-paragraphs (2.3.1 and 2.3.2)

2.3.1 TRU SELF- BURNING OPTIONS

For this option :
- The heavy nuclei in the initial composition of the salt are only U235 and U238.
- The heavy nuclei added during salt processing are natural uranium (0.711% of U 235).

- In the processing unit, the substances left in the salt and subsequently multi-recycled are isotopes of uranium, TRU and eventually certain LLFP.
- The substances released as waste are FPs, a very small quantity of TRU (reprocessing losses), depleted uranium containing U236.

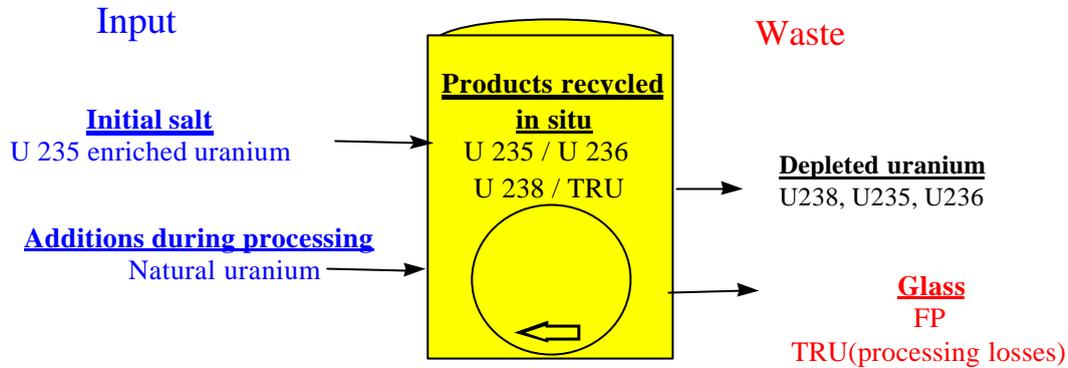This management option is shown on the diagram in figure 3

Figure 3: TRU self-burning option

2.3.2 TRU INCINERATING OPTION

- In this management solution, the elements initially introduced into the molten salt fuel are U235, U238 and TRU.
- The multi-recycled substances are isotopes of U and the TRU.
- During of on-line processing, the heavy nuclei U235 and U238, and TRU coming from spent PWR fuel and additional U235 coming from natural uranium (0.71 % of U235) are added.
- The substances released as waste are FPs, a very small quantity of TRU (reprocessing losses) and depleted uranium containing U236.

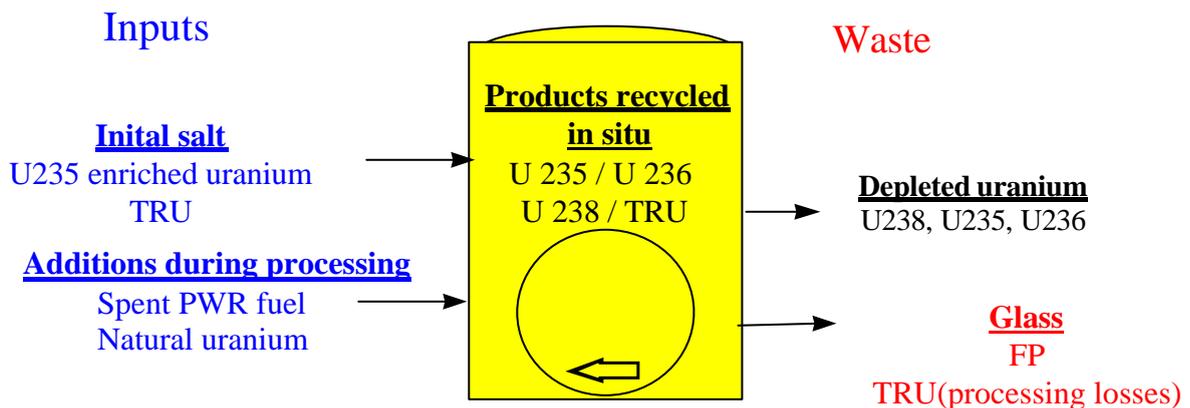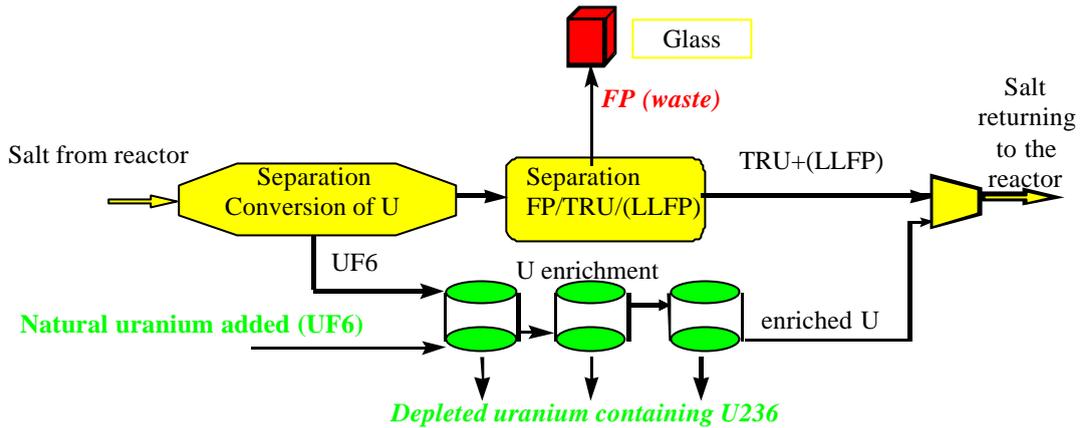This management solution is shown on the diagram in figure 4.

Figure 4: TRU incinerating management

2.4 PROCESSING UNIT

The operation of the processing unit is shown on the diagram in the following figure

In this processing unit, a small part of uranium is extracted from the salt and converted into UF_6. This enables U235 to be added by enriching this UF_6 with additional natural Uranium, also in the form of UF_6.

Enrichment is carried out using centrifuges. Not many would be needed owing to the relatively small quantity of UTS to be added (about 50 t of UTS per year, per GWe, or 1200 units)

During this enrichment process 35% of the U236 are extracted from the spent salt.

After extracting uranium from the salt derived from the reactor, the fission products are separated from the TRU and LLFP that must be incinerated Additional TRU from other reactors spent fules, such as PWR can also be loaded.

The short-lived fission products could be vitrified and stored like today's glasses. The transuranium elements, the enriched uranium and selected LLFP are reinjected into the reactor.

3. NUMERICAL SIMULATION PRINCIPLE

Numerical simulation of this type of reactor requires iterative work, the principle of which is described below (figure 6).
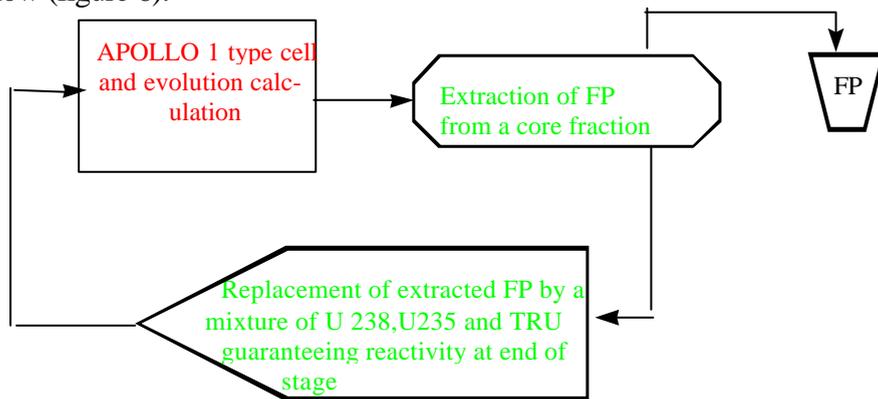

Figure 5: Schematic diagram of the AMSTER on-line processing unit

Figure 6: Schematic diagram of the evolution calculation

Cell calculations in an infinite, evolving medium are conducted with the APOLLO 1 code. This code requires only little computing time and has already been used to simulate and manage the natural uranium gas graphite (UNGG) reactors. Calculation accuracy has been checked by a reference calculation made with the Monte Carlo TRIPOLI code (discrepancy of 400 pcm for a k_∞ of 1.2).

At the end of this calculation stage, FP are extracted from a fraction of the core. The FP are

replaced by a mixture of TRU, U235 and U238 in order to guarantee reactivity at the end of the stage.

To limit the calculation times, the time between two reprocessing operations must be long enough. Initial equilibrium was simulated with an interval of 10 days and then, in the light of the initial results (little change in the k_∞) this interval was increased to 100 days.

4. AMSTER INCINERATION-TRANSMUTATION CHARACTERISTICS

As with all enriched uranium reactors recycling their TRU, the fuel evolves after a certain time towards a composition at equilibrium.

In particular, the mass of TRU in the cycle then reaches an equilibrium value which can no longer be reduced without reducing the energy produced. If the quantity of TRU in the reactor is higher than the equilibrium value, the reactor - which tends towards equilibrium - will consume the surplus. It then becomes a TRU incinerator.

AMSTER equilibrium depends:

1. **on the frequency of processing operations and more exactly on the fraction of the core processed daily**.

2. on the quantities of U236, TRU and LLFP which are extracted or injected.

The smaller the core fraction reprocessed daily, the more fission products will remain in the core. If a core fraction of $\dfrac{1}{n}$ is reprocessed daily, the quantity of FP in the liquid fuel at equilibrium is identical to that of a solid fuel which has achieved a burn-up of n efpd (see appendix 2) This is how is defined the **equivalent burn-up of salt**. It should be noted that the significance of the word burn-up no longer represents the fraction of initial heavy nuclei which have fissioned, but the proportion of FP present in the core.

The calculations designed to determine the incineration-transmutation characteristics of AMSTER, were made as follows:

The following assumptions were made:

- a calculation pitch of 100 days (to minimise calculation times),
- a k_∞ at the end of cell evolution, of 1.05 (taking account of neutron leaks),
- an electrical power of 1 GWe (2250 MWth) produced by AMSTER,
- a salt volume of 48 m^3 (30 in active core and 18 in auxiliaries identical to that of the MSBR project).

1. A reference case corresponding to TRU self- burning option in which one third of the core is processed every 100 days (burn-up of 300 efpd, or one calendar year). This first simulation showed that all or part of the U236 formed by capture of U235 had to be purged (see § 4-1). Partial purging of the U236 (30 % of the initial U236) was adopted for the rest of the study.

2. We then varied the equivalent burn-up to values of 500, 1000, and 3000 efpd.

3. The Tc99 was recycled for burn-ups of 300 and 3000 efpd.

4. A TRU incinerating configuration was simulated, by fixing a mass of TRU present in the reactor. In this study this mass varies between 1.56 t (mass at equilibrium of the reference case) and 10 t, for an equivalent burn-up of 300 efpd.

These results as a whole already cover a large part of the possible uses of AMSTER: production of energy from spent PWR fuels or natural uranium, incineration of TRU and LLFP contained in the spent fuels.

4.1 THE NEED TO PURGE U236

The equilibrium of an AMSTER fed only with uranium was first of all determined. Initial results show that if we add highly-enriched uranium, equilibrium would not be reached because of the build-up of U236. The U236 was produced by capture from U235. It poisons the core, which means that the quantity of U235 present in the reactor has to be increased. This leads to a regular increase in the quantity of U236.
The U236 therefore has to be purged either completely or partially. Several methods are possible:

- Total purge at each processing operation, provided that a highly-selective enrichment processes, such as laser type process, is available on site.

- Total purge of a fraction of the core at given intervals. In this case, all the uranium in the reprocessed salt would be replaced by the enriched uranium purged beforehand. Partial purge performed at each recycling when enriching the uranium contained in the salt. To do this, the uranium from the salt is mixed with an adequate quantity of natural uranium. This principle, , was adopted for the study and enables up to 40% of the U236 to be eliminated

The simulation shows that partial purging of 30% of the U236 (figure 7) can lead to stabilisation of the masses of U235, U236 and TRU in the reactor.
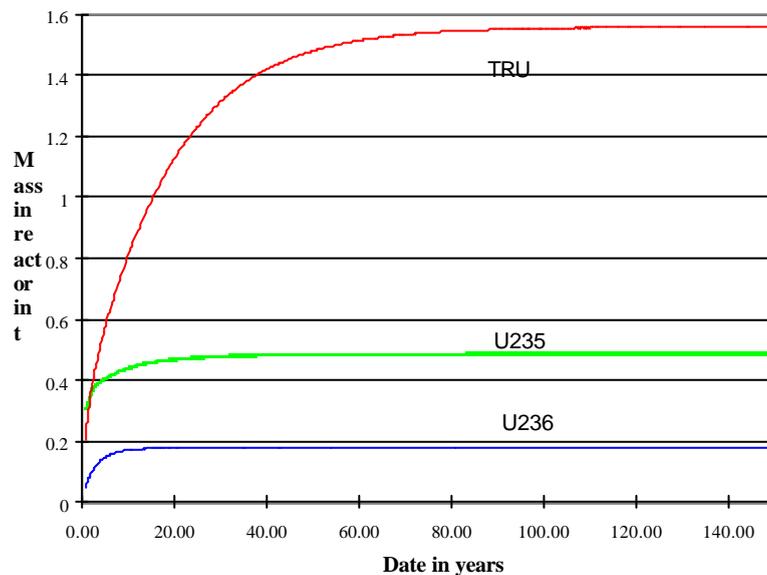

Figure 7: Evolution of masses in the core with partial purging of U236

4.2 EVOLUTION OF THE ISOTOPIC COMPOSITION OF THE TRU

The evolution in the composition of the transuranium elements with partial purging of the U236 is shown in figure 8. The highly thermalised spectrum of the reactor leads to a build-up of Pu242 and Cm244, but in a lesser proportion when compared with a reactor on an inert support of the same type.

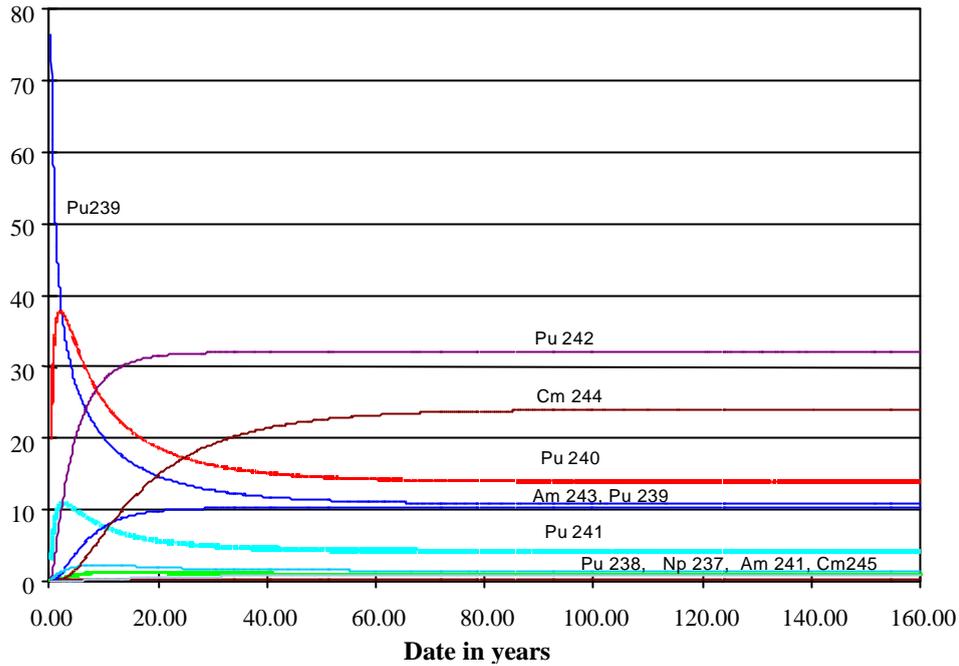**Isotopic composition in %**

Figure 8: Evolution of the isotopic composition of the TRU in self-stabilising configuration for an equivalent burn-up of 300 efpd and partial purging of the U236

4.3   URANIUM CONSUMPTION

Two other important parameters are the mass of U235 and U238 consumed per TWhe during operation. The masses of U238 and U235 consumed per TWhe are shown in figure 9, from start-up to equilibrium.
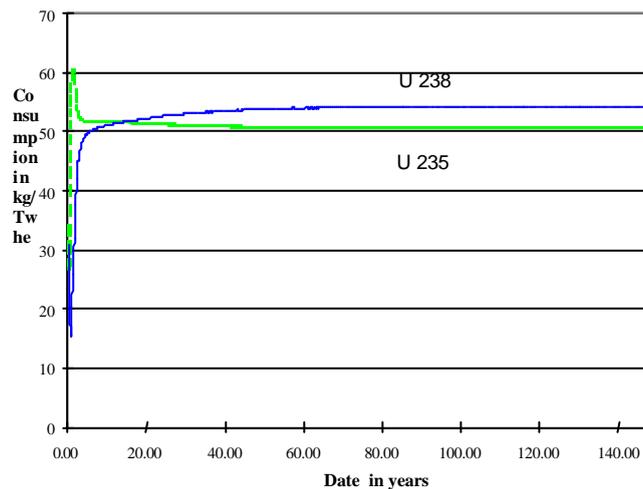

Figure 9: Masses of U238 and U235 consumed per TWhe

It would appear that this reactor is a good converter, as about 50 % of the energy is produced by the Pu239 coming from U238. The U235 only represents 50 % of fission, or 50 kg/TWhe. This result shows that in these operating conditions, one can have excellent use of natural uranium in such

a reactor, while maintaining a very low-level of transuranics element inventory.

4.4 SENSITIVITY AT EQUIVALENT BURN-UP

The increase in equivalent burn-up increases core poisoning by the FP and therefore requires increased U235 enrichment of the salt. The larger quantity of U235 in the core leads to a corresponding rise in the quantities of U236, Np237 and Pu238 along with higher final consumption of U235 per TWhe as shown by the graphs in figures 10 and 11.
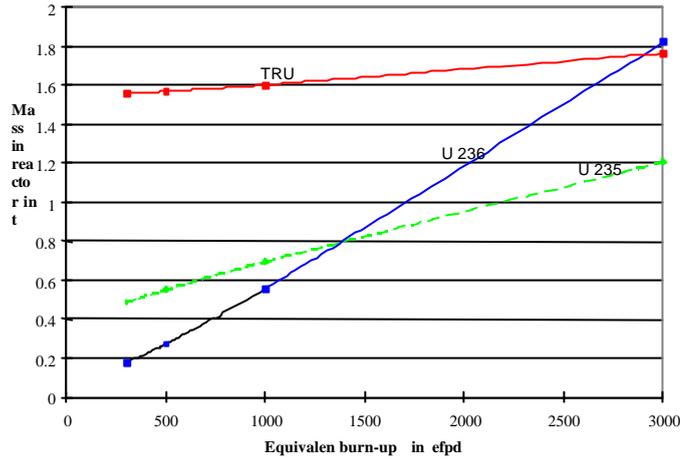

Figure 10: Evolution of masses in the reactor at equilibrium, versus equivalent burn-up,
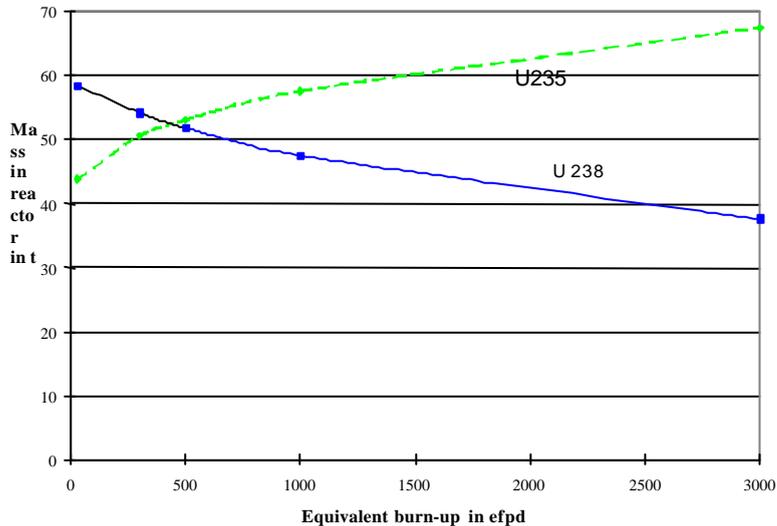in the case of partial purging of U236

Figure 11: Evolution of masses consumed at equilibrium versus burn-up
in the case of partial purging of U236

4.5 TRU LOSSES PER TWH

These are equal to the product of the mass of TRU processed per TWh by the unit «loss» rate for processing of this mass.

The mass processed depends on the equivalent burn-up. Table 1 shows the changing mass of TRU processed according to the equivalent burn-up.

Burn-up in efpd	300	500	1000	3000
Mass of TRU in the reactor (kg)	1560	1570	1600	1760
Mass of TRU processed per TWh (kg)	216	130	67	24

Table 1 – Mass of TRU processed per TWh according to the equivalent burn-up in kg

The loss rate depends on the pyrochemical processes utilised. Molten salt has the advantage of having no insolubles, which account for most solid fuel losses in aqueous reprocessing (10^{-3}). The important parameter is the quantity of TRU remaining with the FP rather than the TRU/FP separation factor. A few percent of FP remaining in the fuel can be tolerated, but the quantity of TRU remaining the FP must be brought as low as possible. The separation effort must therefore be concentrated on a smaller mass of chemical substances (about 100 kg/TWhe).
TRU losses of about 10^{-5}, 10^{-6} are in theory possible. This theoretical value has yet to be proven in an industrial process. In table 2, we have analysed the sensitivity of losses in grams to the equivalent burn-up and the unit loss rate.

«Loss» rate	Equivalent burn - up in efpd			
	300	500	1000	3000
10^{-3}	216	130	67	24.5
10^{-4}	21.6	13	6.7	2.45
10^{-5}	2.16	1.3	0.67	0.245

Tableau 2: Reprocessing «Losses» per TWh
according to equivalent burn-up and unit loss rate in grams

The extreme loss values are 216g/TWhe and 0.25 g per TWhe.
It should be noted that to limit TRU losses, one is inclined to boost the equivalent burn-up, possibly even beyond 3000 efpd. This should not lead to any neutron problems.

4.6   OPTIMISATION OF EQUIVALENT BURN-UP

As the studies currently stand, it is hard to optimise the equivalent burn-up, in other words, the fraction of the core reprocessed daily. For this, the following would have to be taken into account:

1. the stability of the salt with large amounts of FP, which could be prejudicial to high equivalent burn-ups

2. the cost of enriched uranium 235 which would affect the natural uranium and UTS investment per GWe, but also the consumption of natural uranium and UTS per TWhe, favourable to low equivalent burn-ups;

3. the cost of the processing installation on the basis of its mass flow and its TRU release rate, favourable to high equivalent burn-ups

Of these parameters, those concerning (i) the consumption of Unat and UTS, (ii) the mass flow and TRU releases from the processing installation are compared in table 3, for the two extreme equivalent burn-ups examined.

	Equivalent burn-up	
	300 efpd	3000 efpd
Mass of natural uranium consumed per TWhe at equilibrium (t)	9.78	13.11
Mass of UTS consumed per TWhe at equilibrium(t)	7.1	16.76
Mass of TRU processed per TWhe (kg)	220	25
Reprocessing "Losses" per TWhe for a «loss» rate of 10^{-5} (g)	2.20	0.25

Table 3 Comparison of burn-up optimisation parameters

A complete technical/economic evaluation of the TRU and uranium processing installation according to the salt throughput and loss rates, therefore has to be carried out.

4.7   TC TRANSMUTATION

Tc_{99} is one of the LLFP to be taken into consideration for possible transmutation in the reactor.

An AMSTER with equivalent burn-ups of 300 efpd and 3000 efpd for a self-stabilising configuration in which the Tc_{99} is not extracted and thus remains in the salt, was simulated.

At equilibrium, for about 360 and 560 kg/GWe of Tc_{99} are in the reactor respectively. As this Tc_{99} poisons the core, U235 must be added, 75 kg and 120kg respectively,. Incineration of the Tc99 would therefore imply an additional investment of 16 and 10 % U235, with a corresponding increase in U235 consumption of 6% and 3%, depending on the equivalent burn-ups considered (300 and 3000 efpd).

The transmutation of Tc99 would therefore be feasible in this type of reactor, but relatively costly.

4.8   INCINERATING CONFIGURATION

To make this reactor an incinerator, the TRU inventory in the core would have to be higher than the self-stabilising configuration inventory.

A parametric study was conducted for an equivalent burn-up of 300 efpd, according to the quantity of TRU in the reactor, varying the TRU inventory from 1.56 t (mass at equilibrium in the self-stabilising version) to 10 t of TRU.

4.8.1 CONSUMPTION OF TRU AND U235 AT EQUILIBRIUM

The evolution of the consumption of TRU, U235 and U238 at equilibrium for the incinerating solution, is presented in figure 12.
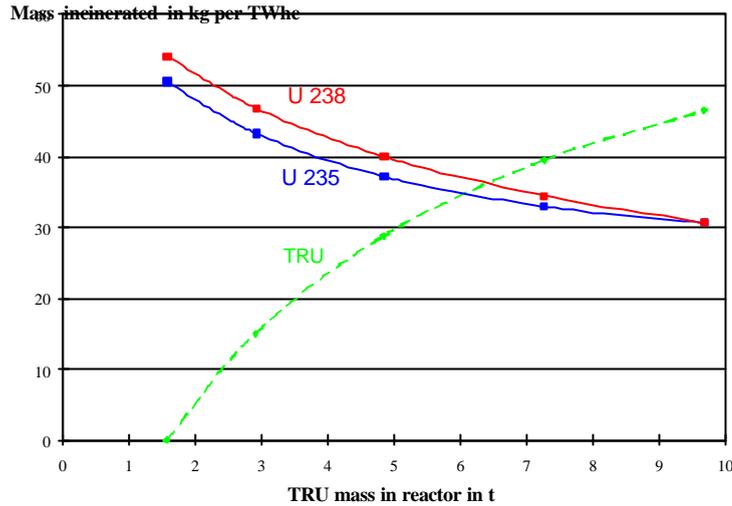
Figure 12: Consumption of TRU and U235 at equilibrium per TWhe
according to the mass of TRU in the reactor

When the mass of TRU in the reactor exceeds 4 t, we can see a trend towards a tailing off (i) in the increase of incineration (ii) and a drop in consumption of U 235.

Elsewhere, increasing the mass of TRU requires a corresponding increase in the mass of U235 added to the reactor, of about 0.07 kg of U235 per kg of TRU, as shown in figure 13.
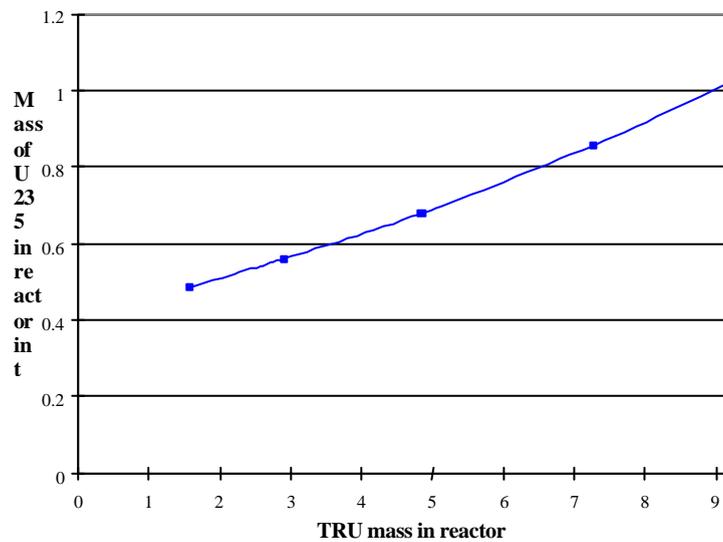

Figure 13: Evolution of the mass of U235 added to the reactor according to the mass of TRU
for the incinerating configuration with an equivalent burn-up of 300 efpd

To incinerate one kg of TRU we will generate a certain quantity of «losses». The losses are proportional to the mass of TRU processed. If we plot the curve of TRU mass processed per kg of TRU incinerated (figure 14), we obtain an optimum for an inventory of between 4 and 5 t. We must then process 23 kg of TRU in order to burn 1.
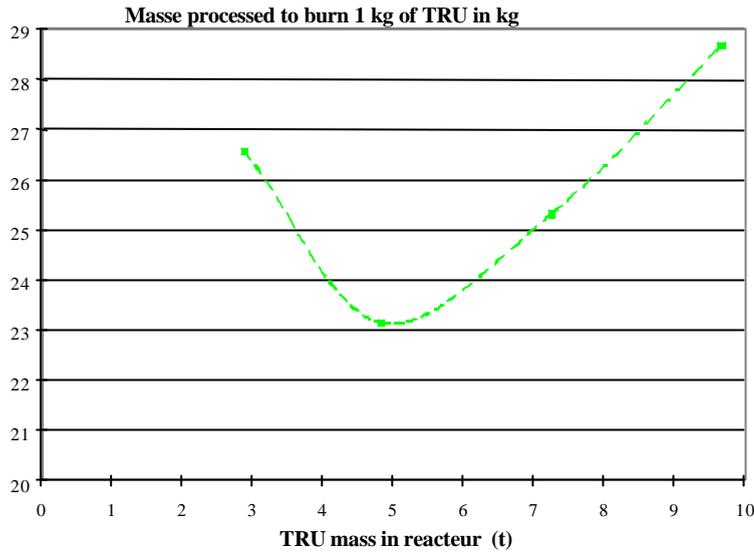
Figure 14 Mass of TRU processed per kg of TRU incinerated

4.8.2 ENERGY PRODUCED FROM SPENT PWR FUEL

The use of the fissile materials (TRU and U235) present in the spent PWR fuels will lead to natural uranium savings, with corresponding energy production in AMSTER.
The uranium savings provided by incineration of spent PWR fuels depends on their burn-up. They would allow the production of between 56 and 91 % of the energy that the spent fuel had already produced.

5. ILLUSTRATION OF SOME AMSTER MANAGEMENT SOLUTIONS

The results presented and commented on in the previous section are summarised below, for an equivalent burn-up of 300 efpd and a unit «loss» rate of 10^{-5}, in the two characteristic AMSTER management solutions studied here:

1. TRU self-stabilising management at equilibrium.

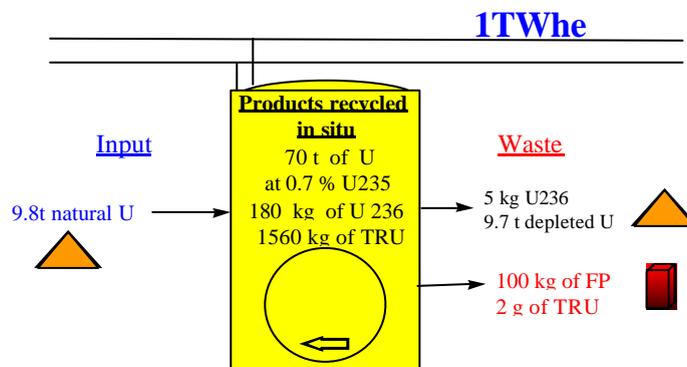2. optimised TRU incinerating management.

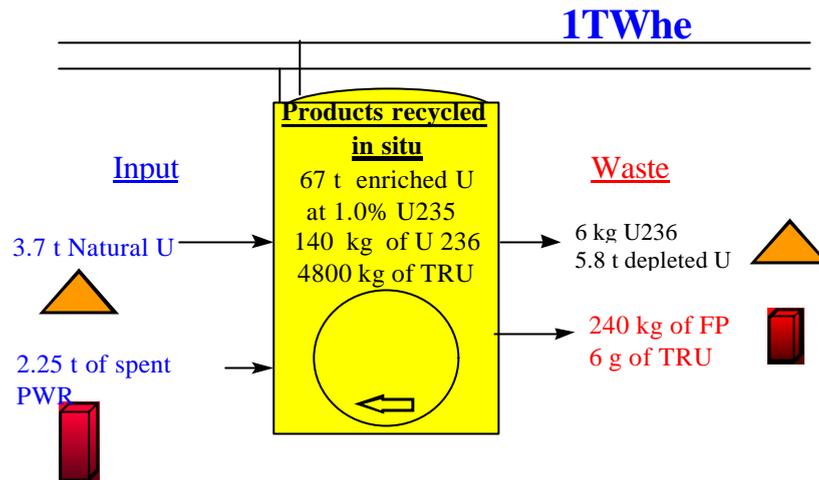Figure 15: Mass balance of TRU self-stabilising management solution

Figure 16: Mass balance of TRU incinerating management solution

6. ADVANTAGES AND DRAWBACKS OF AMSTER

The benefits and difficulties inherent in the AMSTER concept are listed below

ADVANTAGES

Waste production

Very small TRU and LLFP inventory

Very small TRU and LLFP losses (a few grams per TWhe) in theory, but the separation factor of 10^{-5} has yet to be confirmed.

In relation to the open cycle, reduction of the alpha activity masses and waste radio-toxicity of 4 decades (see appendix 4)

Transport of highly radioactive materials

Integrated cycle.

No transport of very active materials.

Proliferation

Non-proliferating fuel, as containing only 10% of Pu 239 mixed with 24 % of Cm 244.

Use of resources

Good use of natural uranium: 10 t/TWhe for an equivalent burn-up of 300 efpd as against 21 t/TWhe for a standard open cycle PWR. This advantage is partly due to the thermodynamic efficiency of AMSTER. The consumption of natural uranium rises with the equivalent burn-up (13t/TWhe for 3000 efpd)

Spent PWR fuels can produce 56 to 91 % of the energy they have already provided.

Security

Combines the advantages of molten salt reactors (negative temperature coefficients, high thermal inertia) with those of a reactor with very low enrichment (very low reactivity of the fuel outside the core)

Flexibility

Salt accepts many types of heavy nuclei: U235, TRU, U233, U238 or Th 232. It can easily and quickly switch from one type of management to another.

DRAWBACKS

New technology

The molten salt reactor series, based on a cycle different from that used in AMSTER, the thorium cycle, was studied at Oak Ridge in the 60s, with a prototype, the MSRE, which worked remarkably well (80 % load factor) but which was abandoned for mainly budget-related reasons in the early 70s.

Hot salt corrosion

Hot salt is extremely corrosive. Oakridge claims to have solved the problem with Hastelloy.

Dry reprocessing (pyrochemistry)

Promising technique, but still at the laboratory stage.

Cm 244

As with all reactors multi-recycling TRU, the fuel at equilibrium contains Cm 244 (25%) and consequently poses radiation protection problems. This will be offset by the fact that the quantity of Cm244 present in the reprocessing shop will be all the smaller as the equivalent burn-up of salt is high and the fact that the Cm will not leave the reactor building.

7. FIRST CONCLUSIONS

The AMSTER concept would allow energy production with a very small quantity of TRU waste to be disposed of (about a few g per TWhe) with no transportation of highly radiotoxic products and half the natural uranium consumption of a PWR. It would also be able to reuse PWR fuels, producing from 60 to 90 % of the energy they produced initially, but once again producing very little TRU waste for disposal (a few g per TWhe).

Consequently, AMSTER could begin to eliminate the TRU produced by PWRs. This could be done with the addition of very little natural uranium (about 3.5 t per TWhe). Once the stocks of spent fuel had been processed, one would then have the choice between:

1. reducing the TRU inventory in the cycle by reducing the installed power
2. continuing with the production of nuclear power, generating very little TRU waste for disposal, just a few grams per TWhe as compared to 30,000 g of TRU per KWhe produced by an open-cycle PWR

Although new, the technology adopted - the molten salt reactor - was already experimented at Oak Ridge in the 60s. The prototype built at the time worked remarkably well. EDF and the CEA examined molten salt reactors up until 1983.

This preliminary study should be followed by a minimum R&D and engineering program in order to evaluate both pyrochemical reprocessing (loss rates to be confirmed), technological feasibility and the safety of the AMSTER concept.

It will then be necessary to optimise the reactor characteristics: unit power, size of the graphite array, equivalent burn-up of salt, quantity of TRU in the reactor, and so on. Its economics will also have to be evaluated.

In closing, it should be stated that these studies are a part of the current Research and Development program required by the Law of December 1991 and can in no case be construed as constituting EDF's short or medium term industrial strategy.

8. REFERENCES

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[2] ORNL 45-41 Conceptual design study of a single fluid molten salt breeder reactor

 [3] R.N. Hill, D.C. Wade, J.R. Liaw, and E.K.Fujita "Physics Studies of Weapons Plutonium Disposition in the Integral Fast Reactor Closed Fuel Cycle", *Nuclear Science and Engineering,* 121 pp17-31(1995)

9. APPENDIX 1 PROCESSING THE URANIUM FROM SALT

Note:

$EU5_{sel\ usé}$	U 235 enrichment of the used salt uranium
$EU5_{sel\ neuf}$	U 235 enrichment of the new salt uranium
$EU6_{sel\ usé}$	U 236 enrichment of the spent salt uranium
$EU6_{sel\ neuf}$	U 236 enrichment of the new salt uranium
$EU5_{Unaturel}$	U 235 enrichment of the natural uranium to be mixed in (0.711%)
$EU5_{Uappauvri}$	U 235 enrichment of the depleted uranium (0.711%)
$EU6_{mélange}$	U 236 enrichment of the mixture
$EU5_{mélange}$	U 235 enrichment of the mixture
mU5	the mass of uranium 235 to be added to the spent salt
MUnat	the mass of natural uranium to be added to the uranium extracted from the salt
$MU_{sel\ usé}$	the mass of uranium contained in the spent salt
$MU_{selneuf}$	the mass of uranium contained in the new salt
$MU_{mélange}$	the mass of uranium contained in the mixture

Principle

After being extracted from the salt, the uranium from the spent salt is mixed with an adjusted quantity MUnat of natural uranium.

The mixture thus obtained is enriched by ultra-centrifuges (modular and with easy U232 decontamination)

The figure shows the equations governing this process (we initially assumed that the U236 co-enrichment factor in relation to U235 is three-quarters)

At equilibrium $MU_{sel\ usé}$ is equal to $MU_{selneuf\ f}$ minus the U238 and U235 burned. To simplify the calculation, we assumed that $MU_{sel\ usé}$ is equal to $MU_{selneuf}$

The mass mU5 to be added is determined by the neutron calculation. The $EU5_{sel\ neuf}$ enrichment is calculated so that the mass of U235 in the new salt is equal to the mass of U235 in the spent salt plus mU5.
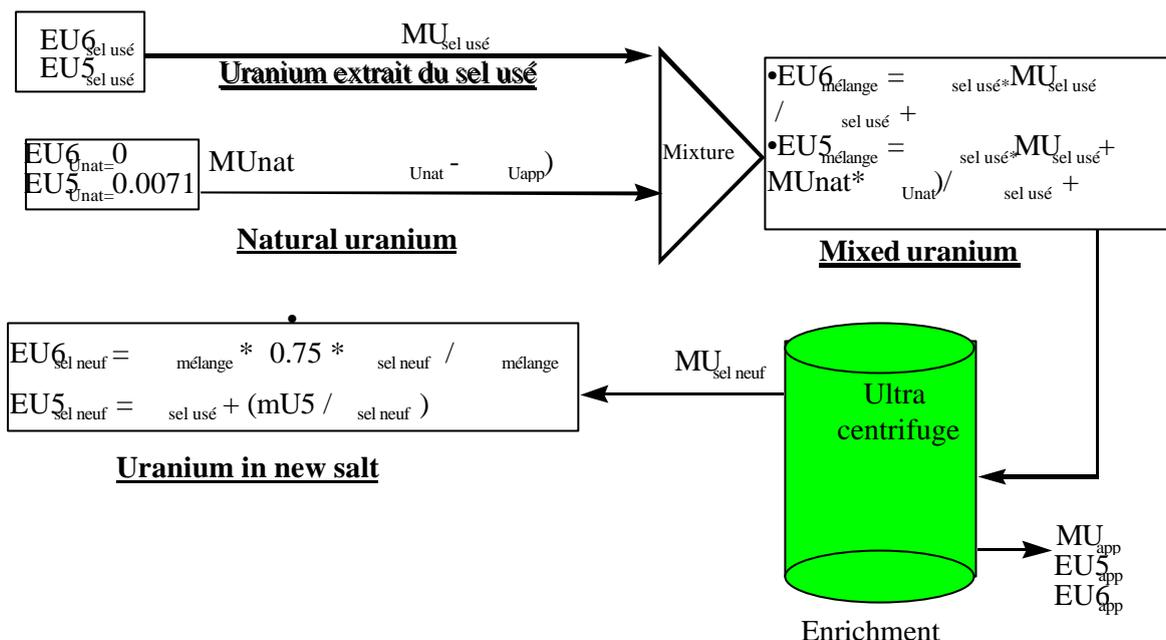
Figure 17: Equations governing the support uranium enrichment process

The following table gives numerical results for self-stabilising management at two equivalent burn-ups of 300 and 3000 efpd

	Equivalent burn-up	
	300 efpd	3000 efpd
$MU_{sel\ usé}$ (t)	10.00	1.00
$EU5_{sel\ usé}$ (t)	0.70	1.80
$EU5_{melange}$ (%)	0.71	0.79
$EU5_{sel\ neuf}$ (%)	1.20	8.50
$EU6_{sel\ usé}$ (%)	0.28	2.74
$EU6_{mélange}$ (%)	0.14	0.19
$EU6_{sel\ neuf}$ (%)	0.18	1.57
%U6 extract (%)	36	43
MUnat (t)	9.78	13.11
MUTS (t)	7.10	15.95

Table 4 numerical results for self-stabilising management at two equivalent burn-ups of 300 and 3000 efpd

10.   APPENDIX 2: EQUIVALENT BURN-OF A MOLTEN SALT REACTOR

The salt condition is characterised by the quantity of FP present in it. This depends on the average burn-up of the salt and the quantity of FP extracted daily, and thus the α fraction of the core volume reprocessed each day.

Thus Irr(n) average core burn-up on date n (in efpd). On date n+1 the average burn-up will be Irr(n+1) such that

$$Irr(n+1) = Irr(n) * (1 - a) + 1$$

At equilibrium $\qquad Irr(n+1)=Irr(n)=Irr_{équ}$

Hence

$$\text{Irr}_{equ} = \frac{1}{a}$$

The quantity of FP in the salt at equilibrium is that of an average burn-up of $\frac{1}{a}$ efpd

We will say that at equilibrium the **equivalent burn-up** of the salt is $\frac{1}{a}$ efpd.

12.   APPENDIX 4   TRU RADIOACTIVITY AND RADIOTOXICITY

The isotopic composition of the TRU contained in the salt differs considerably from that of the TRU present in a spent PWR fuel (Table 6)

	Spent PWR (BU of 60GWd/t)	AMSTER (equivalent burn-up of 300 efpd)
Np237	0.06	0.01
Pu238	0.03	0.01
Pu239	0.44	0.11
Pu240	0.21	0.15
Pu241	0.14	0.04
Pu242	0.08	0.32
Am241	0.01	0.00
Am243	0.02	0.10
Cm244	0.01	0.24
Cm245	0.00	0.01

Table 5 Isotopic composition of the TRU present in a spent PWR fuel and in the self-stabilising AMSTER salt

We then calculated the alpha activity and radiotoxicity of these two types of TRU (figure 18 and 19)
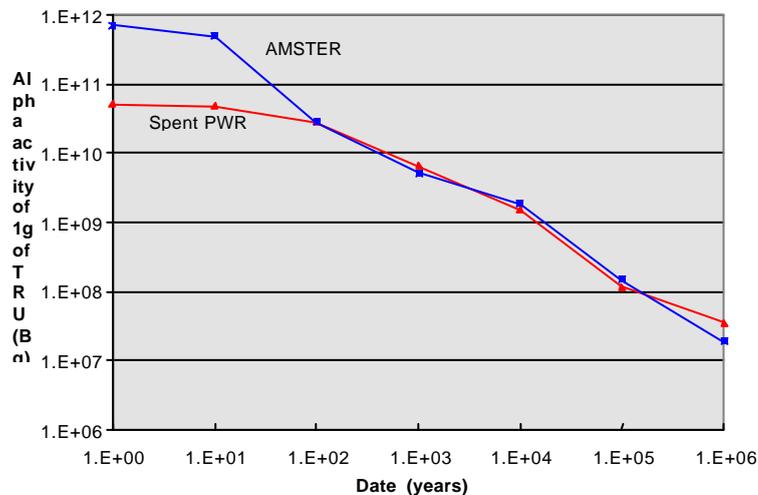

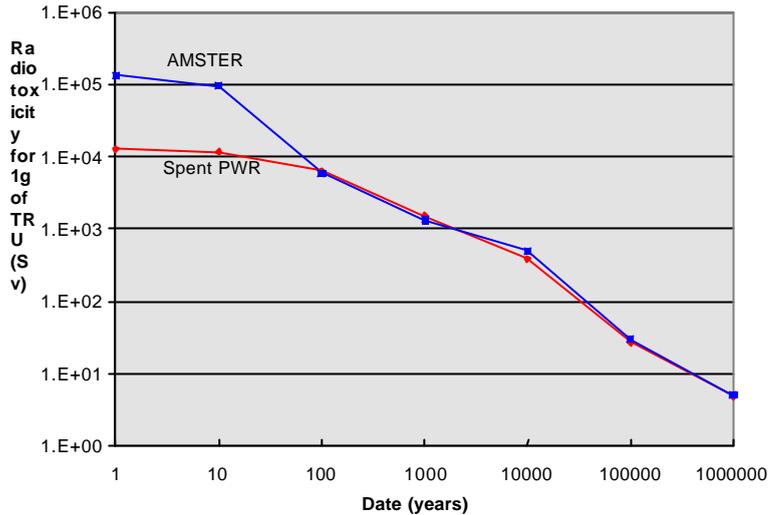Figure 18 Alpha activity of the TRU from a PWR and present in AMSTER

Figure 19 Radiotoxicity of the TRU from a PWR and present in AMSTER

These graphs show that once the Cm244a has decayed (after 100 years), the alpha activity and the radiotoxicity of one gram of TRU contained in the AMSTER salt and in a spent PWR fuel, are very close.

The reduction in TRU mass brought about by AMSTER thus corresponds to an equivalent reduction in alpha activity and radiotoxicity after 100 years. The highest activity between 0 and 100 years is not a problem, owing to the preponderance of FP radiotoxicity during this period.

12.   APPENDIX 5 COMPARISON BETWEEN IFR AND AMSTER

The IFR concept [3] and the AMSTER concept are based upon an integrated back end fuel cycle. IFR is a fast reactor and AMSTER a thermal reactor. In the table 6, main characteristics of the two reactors are compared.

	IFR power production	IFR moderate destruction	AMSTER power production	AMSTER moderate destruction
By GWe in t In core TRU inventory In cycle TRU inventory	7650 10300	9130 13300	1000 1600	3000 4800
By TWhe in kg TRU burned by Twhe TRU processed by TWHe	0 170	44 175	0 220	28 670
Decay half life time		22		17

Tableau 6 Comparison between IFR and AMSTER

The TRU inventories are higher in the fast reactor (factor 6 for the energy producer and 3 for the moderate burner)

The TRU burned mass is better in the fast reactor

The processed TRU mass on which depend the loses is higher in the AMSTER. This mass could be reduced by increasing the equivalent burn up in the AMSTER. But this increase implies an increase in the burning of natural uranium.