

THE AMSTER CONCEPT

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Abstract

AMSTER is a concept for a graphite-moderated molten salt reactor, in which the salt treatment installation has been redesigned in order to reduce waste production. Using this concept, one can define a large number of configurations according to the products loaded and recycled. This document presents a configuration which self-consumes transuranium elements and generates fissile material with a mixed thorium and uranium support. This gives a reactor, which is highly economical in uranium and thorium consumption, leaving only a few grams of transuranium elements per billion KWhe in the ultimate wastes to be disposed of.

1. Introduction

On 30 December 1991 the French parliament adopted a law concerning research into the management of radioactive waste [1]. This law stipulated that three types of work should be conducted simultaneously:

- A search for solutions allowing the separation and transmutation of long-life radioactive elements present in this waste.
- A study into the possibility of reversible or irreversible storage in deep geological formations, in particular by creating underground laboratories.
- A study into long-term packaging and storage conditions for this waste on the surface.

The studies presented in this article aim to provide a foundation for work in the first area of research defined by this law.

Analysis of the problem of reducing long-life radioactive products leads us to propose a new concept: AMSTER (Actinides Molten Salt TransmutER). We will present this concept and the results of a preliminary study into the reactor physics associated with this concept.

2. Description of the AMSTER concept

AMSTER is a continuously reloaded, graphite-moderated molten salt critical reactor, using a Uranium 238 or Thorium 232 support, slightly enriched with ^{235}U if necessary.

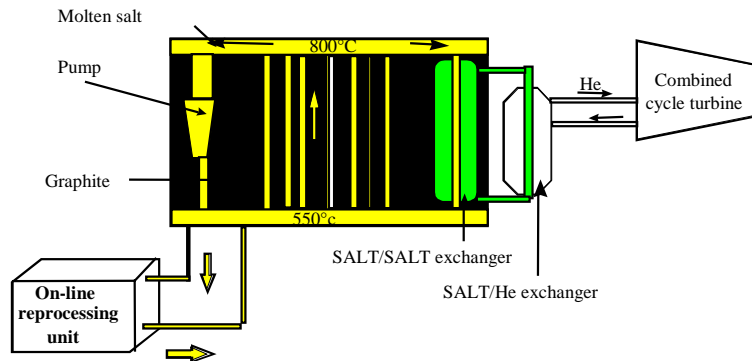
2.1 General presentation

Critical molten salt reactors were extensively studied in the 60s and 70s. Research was carried out in the Oak Ridge National Laboratory, where an 8 Mwth prototype was operated, the Molten Salt Reactor Experiment (MSRE). This experiment was followed by a 1 Gwe project, the Molten Salt Breeder Reactor (MSBR) [2], on which certain aspects of AMSTER are based. It should also be noted that the MSBR project was extensively examined in France (at the CEA and at EdF) in the 1970s.

The aim at the time was breeder reactors. Today, this type of reactor is again of interest to the specialists, owing to its incinerating capacity.

Figure 1 gives the basic layout of this type of reactor.

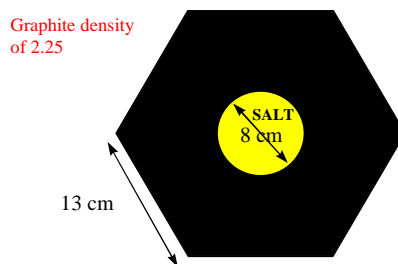
Figure 1. Layout diagram of the molten salt reactor



2.2 Reactor core

The core of a molten salt reactor consists of an array of graphite hexagons identical to those of the Saint-Laurent B1 UNGG reactor. Each hexagon contains a hole through which the salt circulates. We used the results of a study conducted by the EdF reactor physics department in 1976, which defines a salt hole diameter of 8 cm for a hexagon 13 cm on a side. The diameter of this hole was optimised to favour reactor conversion (transmuting as much ^{238}U as possible into ^{239}Pu).

Figure 2. AMSTER reactor cell



We used salt of the same type as that of the MSBR project [2]. The composition adopted, $61\text{LiF} - 21\text{BeF}_2 - 18\text{NLF}_4$, enables a moderate quantity of uranium and transuranium nuclei to be introduced into the core (NL here stand for heavy nuclei).

2.3 Operating principle

When the salt enters the array, it becomes critical and heats up. It enters at a temperature of about 550°C - 600°C and leaves at a temperature of 800°C .

The core inlet temperature is determined by the salt melting temperature, which itself depends on the composition of the salt (500 to 600°C).

The outlet temperature is determined by the strength of the materials other than graphite (hastelloy).

Once heated, the salt is entrained by pumps and passes through salt/salt exchangers which enable the thermal energy produced to be recovered. **On leaving the core, an on-line reprocessing unit which we have entirely redesigned, takes a small fraction of the fuel for reprocessing, in other words it extracts the fission products from it. This reprocessing is accompanied by injection into the salt of new nuclei, ^{235}U , ^{232}Th , ^{238}U or transuranium elements, to replace the heavy nuclei already fissioned.**

The secondary salt heats up either steam or helium, which feeds either a combined cycle turbine plus alternator, or thermal applications such as the production of hydrogen if the salt temperature is high enough.

2.4 Configurations examined

A large number of configurations can be envisaged with this type of reactor, depending on:

- The products placed in the reactor: isotopes of uranium, transuranium elements, isotopes of Thorium, long-life fission products (LLFP).
- Substances multi-recycled in the reactor.
- The substances leaving the reactor:
 - Vitrified losses from reprocessing of transuranium elements, uranium, thorium, LLFP and all the short-life fission products (SLFP) considered being waste.
 - Depleted uranium when the ^{235}U currently stored is used for enrichment.

The various configurations are characterised by:

- The type of the support: uranium, thorium, mixture of uranium and thorium. The thorium support has the advantage of producing significantly fewer transuranium elements than the uranium support and of having a thermal spectrum regenerating more fissile material than the uranium support.
- The presence of fertile blankets allowing regeneration of the fissile nuclei.
- The possible input of transuranium elements coming from other reactors, determining the incinerating capacity of the reactor. If no transuranium elements are input from the outside, the reactor consumes its own transuranium elements. The reactor loaded with outside transuranic become an incinerator.

We thus examined 7 configurations. Table 1 describes the various configurations examined. There are 4 configurations without fertile blanket and 3 configurations with fertile blanket.

Table 1. Definition of configurations examined

Without fertile blanket

Support	Uranium	Thorium	Uranium + thorium
Self consuming	<u>X</u>	<u>X</u>	
Incinerating	<u>X</u>	<u>X</u>	

With fertile blanket

Support	Uranium	Thorium	Uranium + thorium
Self consuming	<u>X</u>	<u>X</u>	<u>X</u>
Incinerating			

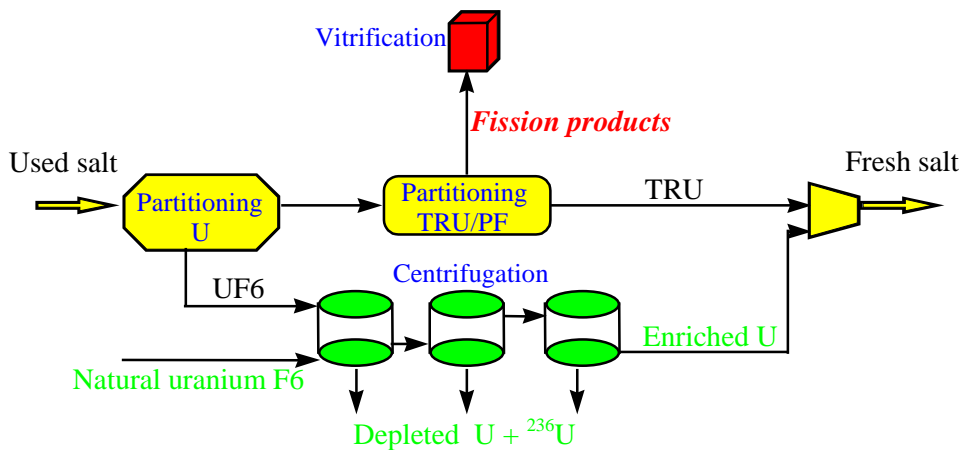
Below, we will present the transuranium elements consumption configuration, with fertile blanket and a mixed uranium-thorium support.

Processing unit: for each configuration examined, more or less complex salt reprocessing is required.

2.5 General recycling principle

Below, we present the recycling principle for the uranium support reactor. For the thorium support, the principle is the same, with the thorium separated after the transuranium elements and before the FP.

Figure 3. Layout diagram of the AMSTER salt processing unit



More specifically, the transuranium elements are confined in the core – reprocessing unit assembly by separate extraction of the transuranium elements and rare earths by a Bismuth counter-flow.

The salt processing unit includes the cycle front-end (salt enrichment) and back-end (FP extraction).

- Cycle front-end

Uranium is first of all extracted from the salt (Figure 3). It is in UF_6 form, which is mixed with an adjusted mass of natural uranium (also in UF_6 form). This mixture is enriched with ^{235}U if necessary, to the required new salt ^{235}U enrichment value, for example using an ultra-centrifuge.

The residual depleted uranium is evacuated, taking with it a large proportion of the ^{236}U in the spent salt (about 35 %) [3]. This prevents the core being poisoned with this isotope.

This solution would require a small number of centrifuges, owing to the small quantity of ^{235}U to be added.

- Cycle back-end

The transuranium elements are separated in a salt – liquid metal exchanger. Given the good separation factor in this operation (about 10), and by using 6 consecutive stages, the salt would only contain a residue of about 10^{-5} times the initial mass of transuranium elements. Then the thorium and the fission products, except for the LLFP to be incinerated, are extracted from the salt with no need for a high separation capacity. The residual transuranium elements in the salt are extracted with the FP, which can be vitrified and stored in the same way as fission products today.

3. Numerical simulation principle

Numerical simulation of this type of reactor requires an iterative working method, the principle of which is described below.

We begin with an APOLLO 1 type calculation of a cell evolving in an infinite medium. We chose APOLLO 1, as this code requires little calculation time and has already been used to simulate and manage Gas Graphite Natural Uranium Reactors (UNGG). We checked the accuracy of the calculations against a reference calculation using the Monte Carlo TRIPOLI 4 code (discrepancy of 400 pcm for a k_{∞} of 1.2).

At the end of this calculation step, we extract the FP from a fraction of the core. They are replaced by a mixture of TRU, ^{235}U and ^{238}U so as to guarantee the reactivity at the end of the step.

To limit the calculation time, the time between two reprocessing operations must be sufficiently long. We achieved initial equilibrium with a pitch of 10 days. Then, in the light of the first results (slight evolution of k_{∞}) we raised this pitch to 100 days.

We adopted a calculation pitch of 100 days and a cell k_{∞} at the end of evolution of 1.05 (to take account of leaks), an electrical power produced by AMSTER of 1 GWe or 2 250 MWth, a salt volume of 48 m³ (30 in the active core and 18 in the auxiliaries) for a reactor without blanket.

We defined a reference case in which we extract one third of the core every 100 days (burn-up of 300 efpd). This initial simulation showed that it was necessary to purge all or part of the ^{236}U formed by ^{235}U capture.

We therefore adopted a partial purge (30%) of the ^{236}U for the rest of the study.

The transuranium elements, enriched uranium, thorium and possibly other transuranium elements, would be re-injected into the salt and then into the reactor.

4. Feasibility of an AMSTER self-generating uranium concept

4.1 The concept

In the case of the thorium support, the consumption of fissile uranium is low enough for it to be produced in the form of ^{233}U , in an extra core zone. This fertile zone located on the core periphery, would be under-moderated by increasing the diameter of the salt hole.

In this concept, the size of the fertile zone would be adapted to make the reactor only just a ^{233}U self-generator (the production of the fertile zone would exactly compensate the consumption of the fissile zone).

Salt processing would simply be by removal of the FP and replacement by the same mass of ^{232}Th or (and) ^{238}U support.

4.2 Calculation method

Only an equilibrium situation using a complete core calculation would be able to validate this concept completely. However, its feasibility can be evaluated by cells calculations.

In the fertile zone, the radius of the cylinder in which the salt circulates is set at 8 cm. This value will then be optimised when the core calculation verification is made.

We also supposed that the power density in the fertile zone was half that in the fissile zone. An exact determination of this ratio is only possible with a core calculation.

We calculate the volume of fissile and fertile salt to obtain the power level sought.

In these conditions, the mean power density of the fertile salt is equal to one-eighth the mean power density of the fissile salt. So that heating of the fertile salt is the same as that of the fissile salt, we slow down the salt flowrate in the blanket by a factor of 8 and thus multiply the time the salt spends in the blanket by 8. The production of heavy nuclei in the blanket is thus equivalent to the production from a volume of salt during the time the salt passes through the fissile core.

Given these hypotheses, the reactor was thus balanced for a given ratio of fertile and fissile salt volumes. At each time step Δt :

- The fuel was placed in each zone for Δt , and the 2 zones were mixed pro rata the core volumes.
- Part of the fission products is removed and is replaced by the same mass of a mixture of ^{233}U and ^{232}Th .
- The enrichment of this mixture is calculated to keep the fissile core k_{∞} at 1.05.

The volume of salt in the fertile area is that for which the mass of ^{233}U to be added at equilibrium is zero.

Equilibrium was achieved for several ratios of salt volume in fertile zone to salt volume in fissile zone. For reasons of precision, we took a time pitch of 10 efpd

4.3 The thorium support

In the fertile zone the ^{233}U balance is positive: More ^{233}U comes out of the blanket than goes in. If we increase the blanket volume, we increase the quantity of ^{233}U produced and at any given moment, the ^{233}U contribution of the blankets will compensate the ^{233}U consumption by the fissile core. The core will generate its own uranium. To determine the volume of the blankets needed, we varied the ratio between the fertile salt volume and the fissile salt volume between 1.5 and 2.5.

For each of these ratios, we give (Table 2) the consumption of ^{233}U , the masses in the reactor at equilibrium, the increase in core size (volume and radius) necessary, and the proportion of power given off in the fissile zone.

These calculations show the feasibility of the concept. This configuration is particularly interesting because:

- It does away with the uranium enrichment phase.
- It requires no extraction of ^{236}U .
- It offers a very small transuranium elements inventory.
- It reduces consumption of heavy nuclei to 100 kg of ^{232}Th per TWhe, thus reducing mining waste accordingly.
- It considerably reduces the masses of depleted and reprocessed uranium.

4.4 The thorium-uranium support

It is also possible to add uranium (i.e. depleted U in stock) to the thorium support. We thus increase the quantity of transuranium elements at equilibrium, while remaining within a reasonable range. But we then burn ^{238}U and make ^{232}Th savings. We significantly reduce the fissile nuclei enrichment of the uranium in the core, thus making the reactor non-proliferating.

As a counterpart, we will have to increase the production of ^{233}U and thus the volume of the blankets.

We thus varied the proportion of ^{238}U added to the load from 0 to 100% for 2 volumes of fertile salts: 2 times the fissile volume and 2.5 times the fissile volume.

Figure 4 shows the mass of ^{233}U consumed (or produced) versus the percentage of uranium 238 in the fertile material added;

The more uranium is placed in the support, the more ^{233}U must be added and the more the volume of the blanket must be increased. Between 0 and 50 % ^{238}U the ^{233}U input is small enough to be

conceivable. Thus by increasing the relative volume of fertile salt to 2.5, self-generation is achieved for 50% ^{238}U .

Table 2. Characteristics of the two-zone AMSTER, depending on the ratio of salt volume in fertile core to the salt volume in fissile core with a 100% thorium support

Fertile salt volume to fissile salt volume ratio	1.5	2	2.5
Core inventories (kg/GWe)			
^{232}Th	138 800	153 580	167 360
Uranium	3 354	3 650	3 900
Transuranium elements	53	54	54
^{233}U consumption (kg/Twhe)	2.11	-0.12	+1.91
Core radius (m)	4.95	5.04	5.11
Total volume of salt in the reactor (m^3)	82	91	99
Power produced in the fissile area (MWe)	1 895	1 800	1 714

The presence of ^{238}U increases the mass of transuranium elements at equilibrium. This rises almost linearly with the percentage of ^{238}U and is equal to 1 500 kg for a percentage of 50%.

The presence of ^{238}U also modifies the isotopic composition of the transuranium elements. Figure 5 shows the different masses in the reactor for 0%, 50% and 100 % ^{238}U in the fertile material inputs.

Figure 4. Mass of ^{233}U consumed (or produced) versus the percentage of ^{238}U in the fertile material input

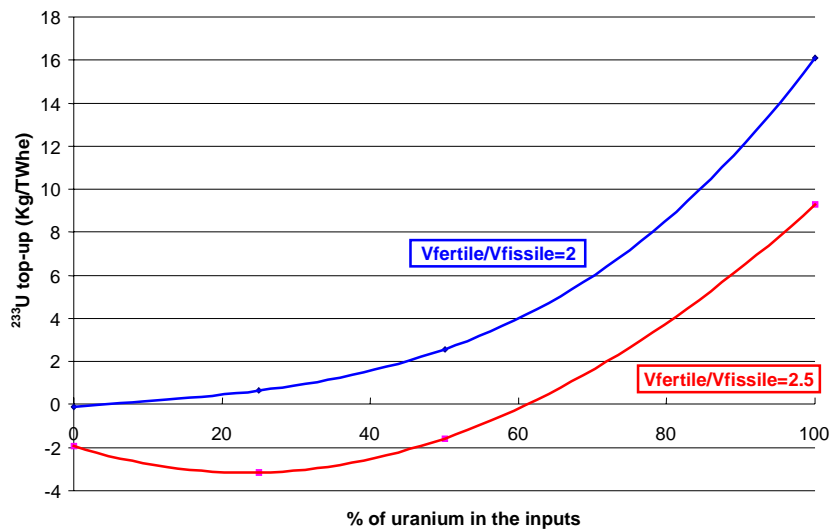
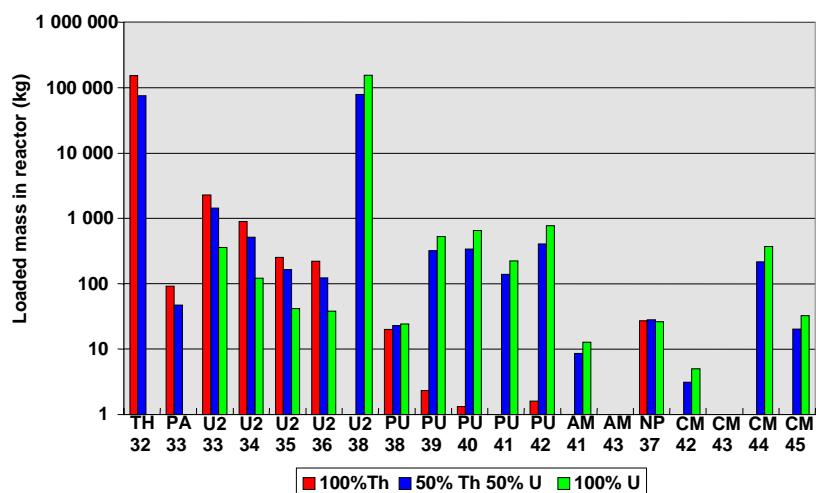


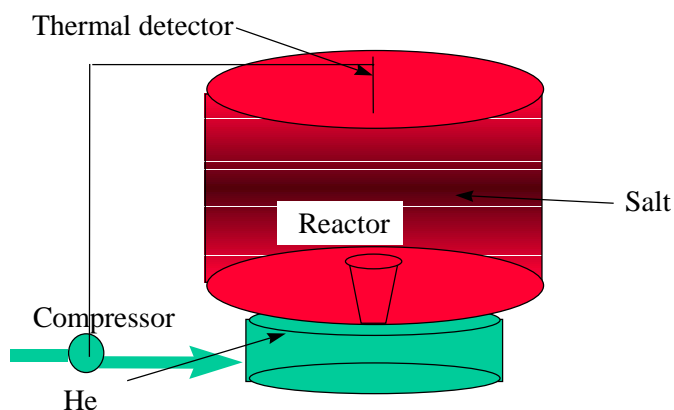
Figure 5. Core mass balance for 0.50% and 100% ^{238}U in the fertile material input for $V_{\text{fertile}}/V_{\text{fissile}} = 2$



5. A major safety asset: core drainage

The salt is at its maximum reactivity in the graphite. A safe fallback position can thus be obtained by draining the core. As the fuel is liquid, it can be extracted from the core at any moment. For this, we adopted a concept proposed by EdF and the CEA, which consists in placing a drain tank under the core, which is permanently connected to it. Salt is confined within the core by a helium back-pressure (Figure 6). One therefore need simply interrupt the electric power supply to the He compressor for gravity drainage of the core. This feature, allied with the considerable thermal inertia of the reactor and the difficulty of rapidly inserting reactivity, should make the reactor particularly safe.

Figure 6. Core drainage principle



6. R&D needed to validate the AMSTER concept

To make the transition from concept to technology, much research and experimentation is required.

Among the subjects covered will be:

- Salt chemistry, structural corrosion by salt.
- Processing chemistry, containment of transuranium elements in the salt.
- Reactor dynamics.
- Safety.

It should be recalled that major R&D work was already carried out in the 60s/70s and a prototype functioned remarkably well (MSRE). A detailed preliminary project for a breeder reactor (MSBR) was conducted. The AMSTER concept is similar to these two reactors and the experience acquired would be directly applicable to it.

7. Fertile and fissile material utilisation

The AMSTER concept should allow the production of energy with very reduced quantities of transuranium element waste (a few g per TWhe), and with no transportation of highly radiotoxic substances.

AMSTER with a mixed thorium-uranium support with a peripheral fertile zone, breeding its own uranium, should further improve this performance. This reactor should in fact offer incinerating performance identical to that of the thorium AMSTER supplied with ^{235}U , while eliminating the need for uranium enrichment and making the reactor non-proliferating. Furthermore, this self-breeding reactor would consume only 50 kg of thorium and 50 kg of ^{238}U per TWhe, which in the light of estimated resources (1 to 4 million tonnes of thorium and 1 to 3 million tonnes of uranium), would allow the production of 20 to 70 million TWhe (the world's annual electricity consumption from all sources together is about 15 000 TWhe); The depleted uranium stored in France (200 000 t), together with 200 000 t of thorium, could produce 4 million TWhe (annual production is 400 TWhe).

Figure 7 compares the mass balance entering and leaving the reactor for a standard open-cycle PWR and a self-generating AMSTER with a support of 50% U and 50% Th.

The inputs are natural or depleted uranium and thorium.

The outputs comprise:

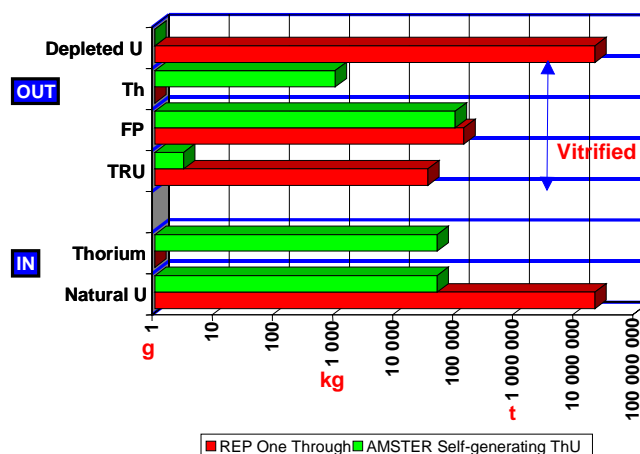
- Transuranium element losses, U and Th vitrified with the FP.
- Depleted uranium produced at enrichment of the support.

This theoretical study shows that:

- AMSTER would be far “cleaner” than a PWR (4 decades reduction of transuranium element waste).

- As we saw earlier, it offers “virtually inexhaustible” resources.
- It should be safer, owing to the fact that the fuel can be rapidly extracted from the core if necessary.

Figure 7. Mass balance entering and leaving the reactor for a standard open-cycle PWR and for a self-generating AMSTER with a support of 50% U and 50% thorium



8. Initial conclusions

Although new, molten salt reactor technology was already experimented at Oak Ridge in the 1960s. The prototype built at the time operated remarkably well. Furthermore, EdF and the CEA studied molten salt reactors until 1983. This preliminary study should be followed by a minimal R&D and engineering program in order to evaluate both the pyrochemical reprocessing process (loss rates to be confirmed) and the technological feasibility and safety of AMSTER.

The reactor specifications would then have to be optimised: unit power, size of graphite array, salt burn-up, quantity of transuranium elements in the reactor, etc., and its economics evaluated.

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