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URANIUM-THORIUM FUEL CYCLE IN A VERY HIGH TEMPERATURE HYBRID SYSTEM.

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ABSTRACT

Thorium is a potentially valuable energy source since it is about three to four times as abundant as Uranium. It is also a widely distributed natural resource readily accessible in many countries. Therefore, Thorium fuels can complement Uranium fuels and ensure long term sustainability of nuclear power.
The main advantages of the use of a hybrid system formed by a Pebble Bed critical nuclear reactor and two Pebble Bed Accelerator Driven Systems (ADSs) using a Uranium-Thorium ($U + Th$) fuel cycle are shown in this paper.
Once-through and two step $U + Th$ fuel cycle was evaluated. With this goal, a preliminary conceptual design of a hybrid system formed by a Graphite Moderated Gas-Cooled Very High Temperature Reactor and two ADSs is proposed. The main parameters related to the neutronic behavior of the system in a deep burn scheme are optimized. The parameters that describe the nuclear fuel breeding and Minor Actinide stockpile are compared with those of a simple Uranium fuel cycle.

Key Words: Uranium-Thorium fuel cycle, deep-burn, transmutation, hybrid-system

1. INTRODUCTION

Deep burn transmutation in Very High Temperature Reactors (VHTRs) is based on using the neutron thermal spectrum to reach a high fuel burn-up. Simplified comparisons made between capture and fission cross sections of fast and thermal neutrons concluded that transmutation using thermal reactors is not factible. Nevertheless, more detailed studies carried out later proved that thermal neutrons are capable to transmute Minor Actinides (MA) in appropriate conditions [8].
In the deep burn concept, transmutation of long lived wastes composed by transuranic elements (TRU) from the nuclear reactors and almost the whole destruction of the useful materials for

the nuclear weapons fabrication (Plutonium isotopes), is obtained with only once-through reprocessing cycle [6]. One of the key benefits of the system proposed in this paper is the utilization of the fuel confined into TRISO particles. Graphite coated fuel particles present some attractive advantages for deep burn transmutation as they have high resistance to irradiation damages, mechanical stress, and very high melting points, enabling to reach a high fuel burn-up [7].

The different TRISO particle's layers constitute excellent fission product and radionuclide barriers in the geological repository. Ceramic fuel particles also keep impenetrable to humidity longer than current metallic containers destined to preserve the spent fuel of conventional nuclear reactors. These features make TRISO particles a robust and attractive waste container [11].

VHTRs belong to next nuclear plants generation, the Generation IV [4]. VHTRs are expected to have attractive features like low electricity generation costs and short construction periods. The development of this technology is based on the experience from High Temperature Gas Reactors (HTGR) like Dragon Peach Bottom, from England, and German Work Group Experimental Reactor (AVR) and Thorium High Temperature Reactor (THTR), from Germany. The above mentioned are experimental reactors built on 60's to prove their viability for electricity and heat cogeneration. They enabled to reach high coolant temperatures at the core's outlet. Current projects, i.e. HTTR-2000, from Japan and HTR-10, from China, which became operatives on the years 2000 and 2003 respectively, enable to reach coolant outlet temperatures higher than 950℃[9].

Taking advantage of high coolant outlet temperatures of the VHTR core, it is possible to produce Hydrogen from water and heat utilizing the Iodine-Sulfur thermochemical process, or from water, heat and natural gas applying steam reforming. Electricity production with a high thermodynamic efficiency is also possible in a direct cycle with a gas turbine. The heat extracted from the reactor is also useful in refineries, petrochemical and metallurgic industries. Because of its constructive characteristics, TRISO fuel can reach high burnups as mentioned above. It could work in a Very High Temperature Reactor and later in a Very High Temperature Accelerator Driven System. This strategy would enable to reach the high burn-up allowed for this kind of fuel, with no structure damages due to irradiation or high temperatures.

An ADS is composed by a heavy particle accelerator, where high energy protons are generated. Protons react with metallic spallation target material to produce neutrons. The spallation target is located at the center of a subcritical core which contains the nuclear fuel. The nuclear fuel may be composed by traditional nuclear fuel, transuranic elements, long lived fission products or Uranium and Thorium mixture as in the present study. Fissile elements are disposed in a way that nuclear chain reaction cannot be sustained without a neutron source (in this case, neutrons from spallation reactions). Neutrons and charged particles are generated from spallation reactions when the accelerated protons hit the target. These neutrons are tightly linked to the kinetic behavior of the system, like delayed neutrons in critical reactors. Therefore, the low fraction of delayed neutrons of spent fuels, loses importance in the chain reaction control, which is determined by the neutron source and the accelerator.

Neutrons produced from spallation reactions in the target and multiplied by fission neutrons in the nuclear fuel, can transmute transuranic elements and long lived fission products into stable elements. As a result of fission reactions that take place in the transmutation processes, large quantities of heat are produced and can be transformed into electricity.

In the present work, the feasibility of a once-through and two steps $U + Th$ fuel cycle is evaluated, and a preliminar conceptual design of a hybrid system composed by a Graphite-Gas Very High Temperature Pebble-Bed critical reactor, and two ADSs with similar characteristics, is accomplished.

2. URANIUM + THORIUM FUEL CYCLE

In a $U + Th$ fuel cycle, three fissile isotopes mainly sustain the criticality of the reactor: U^{235}, which represents a certain percent of the fresh Uranium, U^{233}, which is produced by transmutation of fertile Th^{232}, and Pu^{239}, which is produced by transmutation of fertile U^{238}. In order to compensate U^{235} depletion with U^{233} and Pu^{239} breeding, the quantity of fertile nuclides must be much larger than that the quantity of U^{235} because of the small capture cross section of the fertile nuclides in the thermal neutron energy range, compared to the U^{235}'s. At the same time, the amount of U^{235} must be large enough to set the criticality of the reactor [9]. In nature, Thorium exists as a radioactive metallic element. Thorium fuels and fuel cycles have the following benefits [5]. Thorium is 3 to 4 times more abundant than Uranium, widely distributed in nature as an easily exploitable resource in many countries. Thorium fuel cycle is an attractive way to produce long term nuclear energy with low radiotoxicity waste. In addition, the Thorium fuel cycle could be done through the incineration of weapons grade plutonium (WPu) or civilian plutonium.

The Th^{232} absorption cross-section for thermal neutrons (7.4 barns) is nearly three times the U^{238}'s (2.7 barns). Hence, the conversion of Th^{232} into U^{233} is more possible than the conversion of U^{238} into Pu^{239}. Thus, Thorium is a better fertile material than U^{238} in thermal reactors, but Thorium is less useful to depleted Uranium as a fertile material in fast reactors since the resonance integral of Th^{232} is one-third the U^{238}'s. For the fissile U^{233} nuclide, the number of fast neutrons liberated per thermal neutron absorbed (represented as η) is greater than that for U^{235} and Pu^{239}. Thus, in contrast to U^{238}-Pu^{239} cycle, in which breeding can be obtained only with fast neutron spectrum, in the Th^{232}-U^{233} fuel cycle it can be obtained with fast, epithermal or thermal spectra. Compared to U^{238}-Pu^{239} fuel cycle, Th^{232}-U^{233} cycle produces much less quantities of plutonium and Minor Actinides (MA: Np, Am and Cm). Therefore, the radiotoxicity associated to spent fuel is minimized. Consequently, the production of radioactive waste is less in a Thorium fuel cycle than in the $U - Pu$ cycle used in traditional Light Water Reactors (LWRs). Thorium fuel cycle also have some challenges, whose solutions are under investigation [5].

3. ONE-THROUGH AND TWO STEPS OPEN CYCLE IN A VERY HIGH TEMPERATURE HYBRID SYSTEM

In the present work, the performing of a once-through and two steps fuel cycle in a very high temperature hybrid system for the deep burn of a fuel mixture composed by 60% of Uranium and 40% of Thorium (Figure 1), is outlined. TRISO fuel is burned in a Very High Temperature Critical Reactor during a certain period of time under a power level. When the spent fuel cannot maintain the criticality of the system, pebbles are moved to two Accelerator Driven Systems (ADSs) to continue the deep burn scheme. Pebbles will remain in the ADSs a further period of time producing heat, which can be used to hydrogen and/or electricity production.

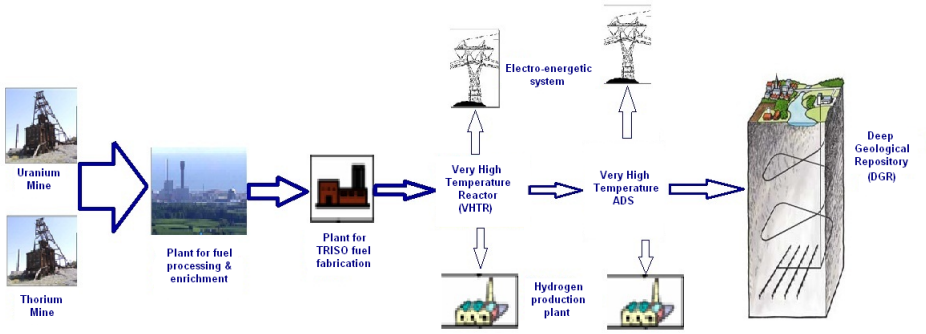


Figure 1: One-through and two steps $U + Th$ fuel cycle in Advanced Very High Temperature Nuclear Systems.

Table I: Critical reactor core's parameters.

General data	Core's height	Core's diameter	Average power density	Core's volume	Fuel volume
Value	2.9394 m	4.175 m	5 MW/m^3	40.24 m^3	6547 m^3

3.1 The Critical Reactor

The proposed reactor is a pebble bed gas-cooled and graphite moderated Very High Temperature Reactor (VHTR), designed for the deep burn of a fuel composed by Thorium and enriched Uranium. This reactor presents a high energy conversion efficiency as it uses a cycle of gas, which has a higher thermodynamic efficiency than the characteristic LWR cycles (saturated steam cycles), and allows high core's outlet coolant temperatures.

In the preliminary conceptual design of the once-through and two steps fuel cycle, the following dimensions and parameters of the reactor's core are proposed (Table I). The fuel in the reactor's core is composed by TRISO coated particles confined in 263 465 pebbles approximately. The fuel is burned up at a thermal power of 200 MWt until the system reaches the subcritical state. The core is surrounded by a radial graphite reflector and an axial one on the top and the bottom.

3.2 Accelerator Driven System

As a second step of the deep burn strategy with $U + Th$ fuel mixture, the burn-up of the fuel extracted from the reactor in two ADSs of the type described in [3], is proposed. The suggested accelerator is a cyclotron with six separated sectors and four accelerating cavities enabling to reach energies of 1GeV with a proton beam intensity of 10 mA [3]. The spallation target is based on eutectic lead-bismuth in melted state with an innovative geometrical design for the

separation window between the beam tube and target. This target was studied in detail in [2]. Analogue to the critical reactor's core, the ADS core is a cylinder containing the fuel elements. The central part is occupied by the spallation target that acts as the neutron source. The core is filled with the graphite fuel pebbles that were previously burned up in the reactor. Although the main advantage of the pebble bed is the possibility of a continuous refuelling, this option was not taken into account in the proposed model since the spent fuel parameters are the most important issue for the cycle features analysis.

Helium gas was chosen as coolant even though it is more expensive than other gases like CO_2 or Nitrogen, because it is transparent to neutrons and hence, neutron spectrum is only determined by the fuel design as there is no neutron moderation. On the other hand, there is no reactivity coefficient due to coolant temperature since a Helium density variation would produce a negligible increment on macroscopic absorption and remotion cross sections. Helium is also chemically inert and does not provoke graphite oxidation.
The fuel used in this cycle step is composed by the pebbles extracted from the reactor. These pebbles have the same design features with the difference that they were already exposed to radiation inside the reactor. Therefore, the pebbles contain fuel and fission products.

4. CALCULATION METHODS

The MCNPX 2.6e version code as main tool of neutronic probabilistic computer modelling was used for the calculation of particle transport and fuel burning. This software was created in 1994 by Los Alamos National Laboratory. It is a program based on Monte Carlo probabilistic method for computer modelling of time dependent transportation of many types of particles, for different geometries and a wide range of energies. Monte Carlo method models probabilistic events which are include in a process and determines statistically their probability distribution. The statistical nature of the process is based on generating random numbers to form histories. These random numbers represent each logical event that neutrons may experience and the study of the neutron trajectory from its birth until its capture or escape (neutron history).

MCNPX 2.6e version incorporates new possibilities respect to previous versions. Some of them were used in the present study, such as CINDER90 code for fuel burning, which works with a 63 energy groups structure. Cross sections for these 63 groups are condensed using a generic spectrum. The incorporation of the LAHET code allows to describe the neutron cascade produced by the proton beam and its transport at high energies. Hence, the simulation of protons and neutrons transport in the spallation target and the entrance of neutrons to the subcritical core as fast neutrons, can be performed in one run. The libraries of the LAHET BERTIN model, which have the physical constants needed for the required parameters calculation are included in the BERTIN file. Another MCNPX improvement is the LA150N library, which contains 42 data tables of high energy neutrons and LA150H library, which contains protons data tables for 41 isotopes.

5. CONCEPTUAL DESIGN OF A VERY HIGH TEMPERATURE HYBRID SYSTEM FOR A DEEP BURN ONCE-THROUGH U+TH CYCLE

In the first cycle's stage, the fuel is burned up during 121 days in the VHTR under a 200 MWt power. When the critical state of the system cannot be maintained, the fuel pebbles are moved

to two ADSs where will keep working during three periods of 66 days under 100, 80 and 60 MWt respectively. The core's dimensions design was performed assuming a working power of 200 MWt and an average power density of 5 MW/m^3, since this value is under the permitted limitable value of 7 MW/m^3 corresponding to this type of systems. A packing fraction of 0.74 was chosen for the reactor's core, as this is the highest possible packing fraction in an infinite ordered honeycomb pebble bed configuration [10]. The fuel burn-up in the reactor was modelled in three time intervals (25, 50 and 46 days).

In order to obtain the optimal reflector thickness, the keff dependence on different reflector thickness values was studied. Results shown in Figure 2 indicate that the maximun reflector savings is obtained with a radial reflector thickness of 180 cm. Hence, radial and axial graphite reflector thicknesses of 180 cm and 100 cm respectively, were considered for the cylindrical core. The control rods and the gas coolant supplier ducts are proposed to be located in the radial reflector. The results obtained in the studies of the reactor reflector design were extrapolated to the ADSs cores.

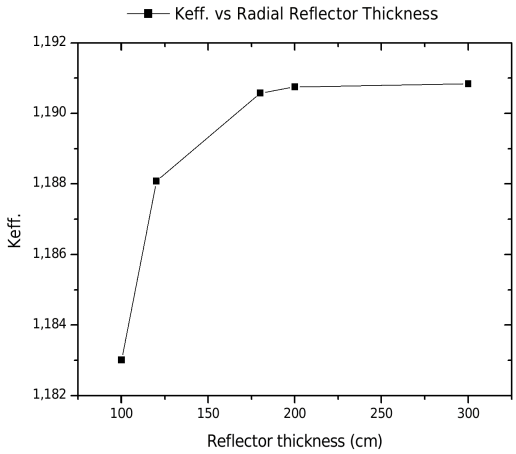


Figure 2: Keff dependence on the radial reflector thickness.

5.1 Study for the Fuel Physical Parameters Optimization

A deep burn fuel cycle based on a mixture of 40% Thorium and 60% Uranium, was implemented. The pebble fuel mass was optimized to obtain the largest Keff value for different U^{235} enrichments in a way that the critical conditions of the system would be ensured in a burning cycle as large as the standard LWRs (18-20% of the excess of reactivity). Although an important advantage of the proposed device is the allowance of on-line refuelling, in the present study a burning cycle with fresh fuel is modelled. Therefore, an initial excess of reactivity is necessary in order to study the burning effect.

An homogeneous composition was considered for the inner part of each pebble (except for the coating), using the adequate mass fraction of carbon, silicon and fuel. It was considered that fuel particles were diluted in a carbon matrix. For simulation proposes, the core was divided into ten horizontal levels considered in the calculation as homogeneous zones.

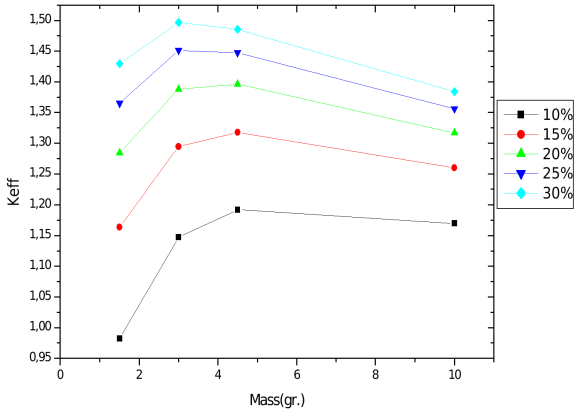


Figure 3: Keff dependence on the pebble fuel mass for different U^{235} enrichments.

The Keff values for the different proposed states were calculated with the MCNPX code. In Figure 3, the calculations results for different pebble fuel masses (1.5 g, 3 g, 4.5 g, 10 g) and U^{235} enrichments (10%, 15%, 20%, 25%, 30%), are shown. For all U^{235} enrichment values, a maximum Keff value was obtained between 2 and 5 g of pebble fuel mass. A Keff value of 1.19050 corresponding to 4.5 g of pebble fuel mass with a 10% of U^{235} enrichment was obtained, which was high enough to guarantee an adequate cycle's duration. The last demonstrates that a relatively low U^{235} enrichment in a $U + Th$ mixture was enough to obtain a satisfactory cycle's duration in the critical system. In previous studies about $U + Th$ fuel cycles [9], a 20% U^{235} enrichment for a prismatic reactor was proposed.

After optimizing the U^{235} enrichment value and the mass in the fuel mixture, radial and axial power density profiles, and radial and axial neutronic flux distributions for the beginning of the cycle (BOC) and the end of the cycle (EOC) in the VHTR core, were obtained (not shown for space constraints).

Flux and power density profiles in the proposed VHTR present a similar behavior to standard nuclear reactors. The peak factors are 1.25 and 1.16 for radial and axial distributions respectively, both at BOC. This shows a uniformity in the power density profiles of the reactor core.

After 121 days of fuel burning in the reactor core, a redistribution was made with the fuel pebbles in two ADSs cores with the same characteristics (Table II). The ADS parameters were chosen based on a conceptual design reported on [1], and called from this point the "reference ADS". ADS geometry was designed in a way that exactly the half amount of the reactor pebbles were placed on each ADS. The distribution of the downloaded pebbles from the reactor to the ADS was done considering the different burn-up of the ten horizontal layers in which the ADS was divided in the simulations. The pebbles of the five most burned zones were put into the first ADS, and the rest pebbles into the second one.

For the fuel burn-up modelling in the ADS, only one burning period, divided into 3 stages of 66 days, working each one under different power values (100, 80, 60 MWt) was considered. The

Table II: Main parameters of the proposed and reference ADS.

Parameter	Core's outer radius (cm)	Total volume (m^3)	Fuel volume (m^3)	Fuel mass (Kg)	Number of pebbles	Keff	Thermal power (MW)
Ref. ADS	125.75	14.38	10.24	124.5	94092	0.94	100
ADS 1	148.42	20.12	8.62	510.1	131732	0.88	Variable
ADS 2	148.42	20.12	8.62	510.1	131732	0.85	Variable

same packing fraction of the reactor (0.74) was chosen.

Radial and axial power density profiles, and radial and axial neutronic flux distributions for the beginning and the end of the cycle (BOC and EOC), were obtained for the two ADSs. It was observed a similar flux and radial and axial power density behavior, to that obtained in the reference ADS.

5.2 Neutronic Behavior of the Critical Reactor and ADSs

The reactor behavior using as fuel 4.5 g per pebble of a mixture composed by 60 % in weight of Uranium with a 10% of U^{235} enrichment, and by 40% in weight of Thorium, was modelled. The last guarantees an excess of reactivity enough to maintain the criticality of the reactor during more than 100 days working under 200 MWt thermal power. Keff variation as function of time is showed in Figure 4. It depletes until the criticality of the reactor cannot be maintained.

The reactor burnup was calculated using equation 1:

$$Burnup(GWd/ton) = \frac{Energy(GW) * Time(Days)}{FuelMass(ton)} \tag{1}$$

The fuel reaches a burn-up of 23 GWd/ton in a 121 days cycle. One of the main goals of the proposed once-through $U + Th$ cycle analysis is the calculation of the contribution of the fissile initial isotope U^{235} and those that originates from conversion (Pu^{239}, Pu^{241} and U^{233}), to the energy production. An analysis about each fissile isotope contribution to the total produced energy was done. Results are shown in Table III. It can be observed that the contribution of the Plutonium fissile isotopes is greater than the contribution of U^{233}.

The fuel burn-up behaviour in the ADSs is shown in Table IV. The first value corresponds to the burn-up value of the fuel downloaded from the reactor.

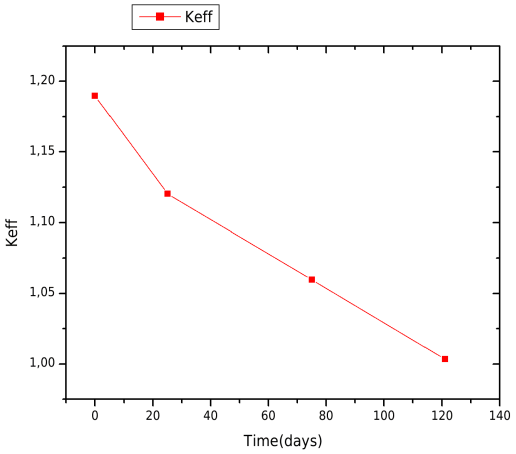


Figure 4: Keff of the reactor as a function of time.

Table III: Contribution of each fissile isotope to the total energy produced in the reactor.

Isotope	U^{233}	U^{235}	Pu^{239}	Pu^{241}	Total
Energy (MWd)	491.07	20365.81	3702.63	284.32	24843.83
%	1.98	81.98	14.90	1.14	100

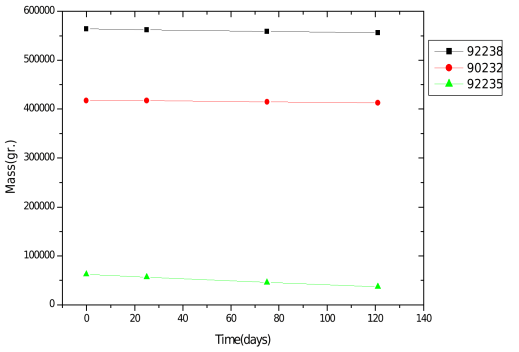
5.3 Variation of the Isotopic Composition

The mass variation of the fuel isotopes, Plutonium isotopes, Uranium isotopes and other isotopes in the reactor as function of time, under a nominal power of 200 MWt, are shown in Figures 5(a), 5(b) and 5(c) respectively.

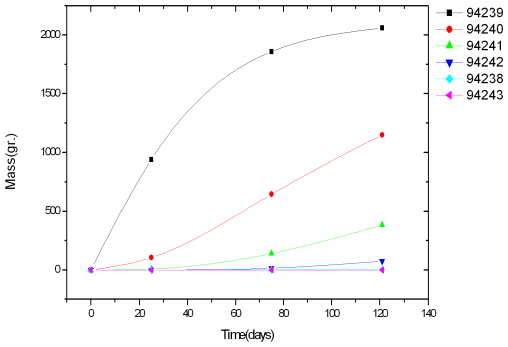
The sum of the fuel isotopes initial masses is about a tonne (1043.6 Kg). This permits to refer the final mass quantities of a given isotope as final mass per ton of initial fuel, approximately.

Table IV: Fuel burnup (GWd/ton) in the ADSs.

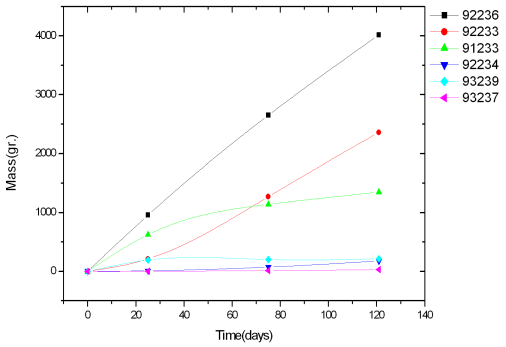
Time (days)	0	33	66	99	132	165	198
ADS 1	23.2	29.7	36.2	41.4	46.6	50.5	54.4
ADS 2	23.2	29.7	36.2	41.4	46.6	50.5	54.4



(a) Mass depletion of the fuel isotopes (U^{238}, Th^{232}, U^{235}).



(b) Plutonium isotopes variation in the reactor (Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}, Pu^{243}).



(c) Variation of the composition of Uranium, Neptunium and Protactinium main isotopes (U^{233}, U^{234}, U^{236}, Np^{237}, Np^{239}, Pa^{233}) in the reactor.

Figure 5: Mass variation in the reactor

Fertile isotopes U^{238} and Th^{232} present not much relative variation with burnup, as it can be conclude from Figure 5(a). Fissile isotope U^{235} presents much larger relative variation since it has a high fission cross section in thermal energy spectrum. In figure 5(b) it is shown that Pu^{239} mass tends to reach a constant value.

The mass variation of the rest of Uranium, Neptunium and Protactinium isotopes as function of time, are illustrated in Figure 5(c). The U^{236} and fissile U^{233} grow up linearly. The high absorption rate of thermal neutrons in U^{235} provokes the large U^{236} growing up. About 16% of spent U^{235} mass becomes U^{236}. In the reactor fuel cycle, the transmutation of approximetely 1% of Thorium (4 Kg) takes place, becoming U^{233} almost completely (final mass: 2.36 Kg) and Pa^{233} (final mass: 1.35 Kg). On the other hand, 1.4% of initial U^{238} mass (8 Kg) is transmuted into Pu^{239}, Pu^{241} and Np^{239}. Pu^{239} produces fission with a significative contribution to energy production (15%). However, U^{233} generates less than 2%.

The Np^{237}, which appears due to neutron absorption by U^{236} (very small σ_a), reaches a small

Table V: Comparison of MA accumulation (Mass in Kg per ton of initial fuel)

Isotope	Am^{241}	Am^{243}	Cm^{244}
$U+Th$ cycle (Sum of both ADSs quantities)	0.536	0.062	0.011
LWR (After a cooling time of 15 years)	0.77	0.14	0.031

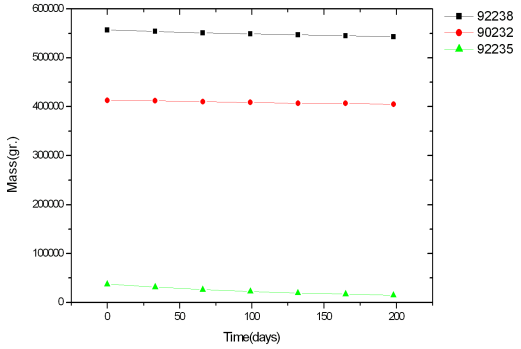
mass. Americium and Curium masses are not shown because insignificant values of these elements are obtained. Fuel Breeding Coefficient (BC) values were calculated for the reactor cycle. The following expression which represents a fissile isotopes mass balance was used:

$$BC = \frac{mass_{initial}U^{235} - mass_{final}U^{235} + massPu^{239} + massPu^{241} + massU^{233}}{mass_{initial}U^{235}} \quad (2)$$

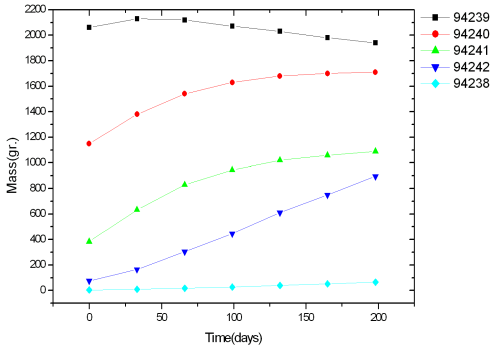
In this equation, Pu^{239}, U^{239} and Np^{239} masses are summed because U^{239} and Np^{239} have a very small $T_{1/2}$ and constitutes intermediate isotopes between U^{238} and Pu^{239}. The same procedure was done with Pa^{233}, which is also an intermediate isotope. The reactor's BC is 0.51 for $U+Th$ cycle.

In figures 6(a), 6(b), 6(c) and 6(d), the mass variation in time of Pu, U and MA fuel isotopes in the ADSs that worked under power values of 100, 80 and 60 MWt, in time intervals of 66 days, are shown. A relative variation of 2.3% and 1.9% in U^{238} and Th^{232} masses respectively, was obtained in the ADSs. In the whole cycle, a variation of 3.7% of U^{238} and 2.9% of Th^{232} relative to the masses initially loaded in the reactor, is produced. The total U^{235} mass depletion is 76.4%. The final Pu^{239} mass in the ADSs decreases respect to the loaded mass, as shown in Figure 6(b), corroborating that in a deep burn scheme, the final quantities of this isotope are minimized (final mass: 1.9 Kg per ton). Nevertheless, U^{233} mass in the ADSs maintains its increasing tendency along the whole cycle. The final value is 5.71 Kg per ton of initial fuel as shown in Figure 6(c). If the final Pa^{233} value (1.29 Kg) is added to the final U^{233} value, a final U^{233} quantity of 7 Kg per ton of initial fuel is obtained.

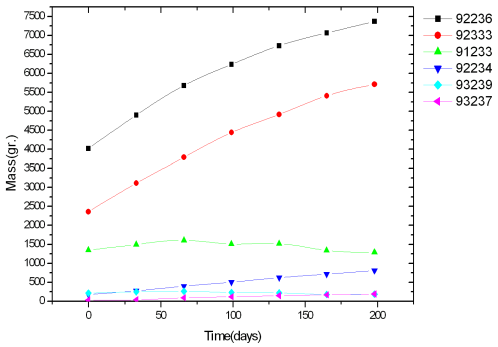
The increase of MA considered null in the reactor, such as Am^{241}, Am^{243} and Cm^{244}, which are responsible for the long live wastes radiotoxicity, is illustrated in Figure 6(d). The MA that increases the most is Am^{243}. A comparison between the transuranic accumulated masses per ton of initial fuel values in the proposed cycle, and the values published in [4] for conventional LWR under a 40 GWd/ton burn-up and a cooling time of 15 years, is illustrated in Table V. The Am^{241} quantity formed from Pu^{241} decay at the end of the whole $U+Th$ cycle was added to the Am^{241} quantity accumulated after a cooling time of 15 years. For the proposed $U+Th$ cycle in the VHT hybrid system, much smaller values of Am and Cm masses are obtained, despite they were calculated under a much higher burn-up (54 GWd/ton).



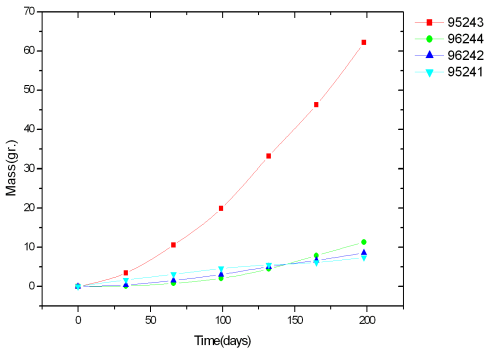
(a) Initial fuel isotopes variation in the ADSs (U^{238}, Th^{232}, U^{235}).



(b) Plutonium isotopes mass variation in the ADSs.



(c) Main isotopes mass variation in the ADSs.



(d) Minor Actinides isotopes mass variation (Am^{243}, Cm^{244}, Cm^{242}, Am^{241}).

Figure 6: Mass variation in the ADSs

6. COMPARISON BETWEEN U+TH CYCLE AND CONVENTIONAL URANIUM CYCLE

A comparison between the $U + Th$ fuel cycle with a 40% of Thorium and a 60% of Uranium with a 10% of U^{235} enrichment, and a similar fuel cycle using only Uranium with an enrichment equivalent to the total previous mixture enrichment (6%), was carried out. Both cycles were subjected to the same power levels in the reactor and ADSs, in the same time intervals.

Keff behavior in both cycles is shown in Figure 7. In the Uranium cycle, Keff value keeps slightly higher than in $U+Th$ cycle since U^{238}'s microscopic fission cross section at high energies is larger than Th^{232}'s and for this reason, the U^{238} contribution to reactivity is higher than the Th^{232}'s. The U^{238} and Th^{232} fission to absorption rates at these energies are 0.3% and 0.09% respectively in the considered multiplicative system. The mass variation of fissile and fertile isotopes in Kg between BOC and EOC is given in Table VI. The accumulated U^{233} mass in the $Reactor + ADS$ hybrid system for $U + Th$ cycle, is three times the Pu^{239} mass. In Table VII, the masses in Kg

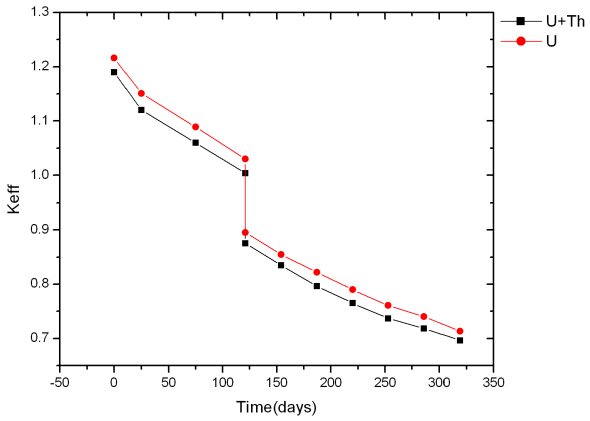


Figure 7: Keff behavior in both cycles.

Table VI: Mass variation in Kg of fissile and fertile isotopes.

	Isotope	U^{238}	Th^{232}	U^{235}	Pu^{239}	Pu^{241}	U^{233}
$U + Th$ cycle	BOC	563.5	417.4	62.61	0	0	0
	EOC	543.3	405.3	14.75	12.12	1.09	7.00
Uranium cycle	BOC	982	0	62.61	0	0	0
	EOC	954.2	0	17.42	3.43	1.65	0

that become energy by fission of fissile isotopes in both cycles, are given. A greater contribution of fissionated U^{235} mass in $U + Th$ cycle (66%) is observed. The contribution of new generated fissile isotopes is similar in both cycles. The fuel's Breeding Coefficient (BC) values for each whole cycle (Reactor+ADS) were also calculated, using Eq. 2 for $U + Th$ cycle and Eq. 3 for Uranium cycle:

$$BC = \frac{mass_{initial}U^{235} - mass_{final}U^{235} + massPu^{239} + massPu^{241}}{mass_{initial}U^{235}} \tag{3}$$

The BC for $U + Th$ cycle with 10% of U^{235} enrichment was calculated for the whole cycle ($Reactor + ADS$), and a value of 0.93 was obtained. For the Uranium cycle with 6% of U^{235} mass enrichment, a value of 0.8 was obtained. The last confirms the advantage of using the

Table VII: The masses in Kg that become energy by fission of fissile isotopes in both cycles.

Isotope	U^{233}	U^{235}	Pu^{239}	Pu^{241}
$U+Th$	4.29	40.49	16.36	0.19
$Uranium$	0	38.20	20.50	0.57

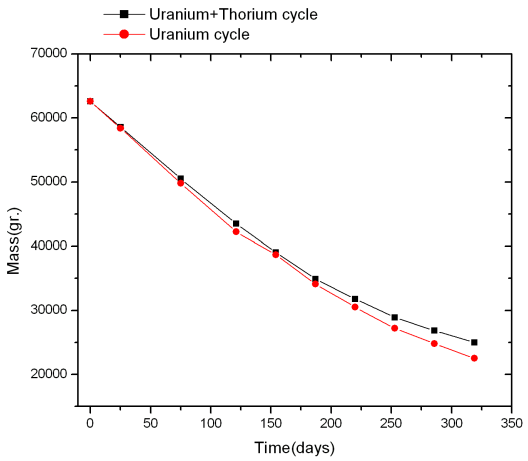


Figure 8: Fisile isotopes mass variation as funtion of time for both cycles.

$U+Th$ cycle in a deep burn scheme for fuel breeding. For $U+Th$ and Uranium cycles only in the reactor, values of 0.51 and 0.46 respectively were obtained.

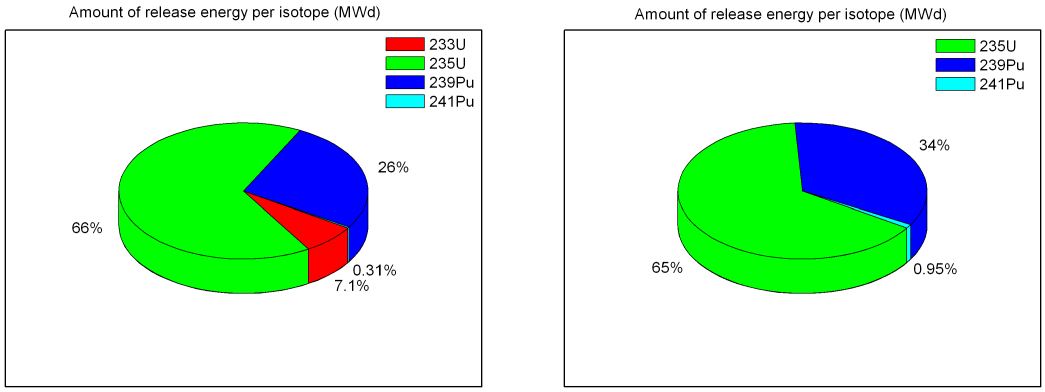
The mass variation of fissile isotopes as a function of time for both cycles is illustrated in Figure 8. The fissile fuel mass obtained at the end of $U+Th$ cycle is larger than that obtained at the end of the Uranium cycle and therefore, a larger BC is obtained for the whole cycle.

The contribution of each fissile isotope to the total energy produced in the $Reactor+ADS$ system for both cycles, is shown in Figures 9(a) and 9(b). The contribution of Pu^{239} is approximately three times the U^{233} contribution.

7. CONCLUSIONS

The proposed hybrid system composed by a critical reactor and two Very High Temperature ADSs, a deep fuel burn-up (54 Gwd/ton) with a Breeding Coefficient close to one (0.93) in a $U+Th$ once-through and two-steps fuel cycle, in which less MA quantities than in a conventional cycle are accumulated, is achieved.

In the present work, the fuel's physical parameters of the proposed $U+Th$ mixture, composed by 40% of Thorium and 60% of enriched Uranium were optimized. It was demonstrated that a



(a) (b)

Figure 9: Energetic contribution of each fissile isotope in $U + Th$ cycle (on the left) and U cycle (on the right).

relatively low U^{235} enrichment (10%) and a mass per pebble of 4.5 g in the reactor, permits to obtain an excess of reactivity high enough to reach an adequate cycle's duration in the critical system.

In the analyzed system, power density distributions uniform enough to ensure adequate heat removal from the cores, are obtained for both the reactor and the ADSs.

The comparison between the proposed $U + Th$ cycle and a Uranium cycle with 6% of U^{235} enrichment, confirms that a high fuel breeding in both $U + Th$ cycle (BC=0.93) and Uranium cycle (BC=0.8), can be obtained. The final amount of U^{233} obtained is the main contributor to the breeding coefficient in the whole $U + Th$ cycle, however, a very high Pu^{239} contribution to energy production is achieved (26% by Pu^{239} vs. 7% by U^{233}). It demostrates that the proposed $U + Th$ cycle is better as a closed fuel cycle than as a once-through fuel cycle.

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