International Journal of Engineering, Mathematical and Physical Sciences
ISSN: 2517-9934
Vol:6, No:3, 2012

The MUST ADS Concept

J-B. Clavel, N. Thiollière and B. Mouginot

***Abstract*—**The presented work is motivated by a French law regarding nuclear waste management. A new conceptual Accelerator Driven System (ADS) designed for the Minor Actinides (MA) transmutation has been assessed by numerical simulation. The MUltiple Spallation Target (MUST) ADS combines high thermal power (up to 1.4 GWth) and high specific power. A 30 mA and 1 GeV proton beam is divided into three secondary beams transmitted on three liquid lead-bismuth spallation targets. Neutron and thermal-hydraulic simulations have been performed with the code MURE, based on the Monte-Carlo transport code MCNPX. A methodology has been developed to define characteristic of the MUST ADS concept according to a specific transmutation scenario. The reference scenario is based on a MA flux (neptunium, americium and curium) providing from European Fast Reactor (EPR) and a plutonium multi-reprocessing strategy is accounted for. The MUST ADS reference concept is a sodium cooled fast reactor. The MA fuel at equilibrium is mixed with MgO inert matrix to limit the core reactivity and improve the fuel thermal conductivity. The fuel is irradiated over five years. Five years of cooling and two years for the fuel fabrication are taken into account. The MUST ADS reference concept burns about 50% of the initial MA inventory during a complete cycle. In term of mass, up to 570 kg/year are transmuted in one concept. The methodology to design the MUST ADS and to calculate fuel composition at equilibrium is precisely described in the paper. A detailed fuel evolution analysis is performed and the reference scenario is compared to a scenario where only americium transmutation is performed.

***Keywords*—**Accelerator Driven System, double strata scenario, minor actinides, MUST, transmutation.

I. Context and Motivation

THE nuclear waste management R&D priority axes in France are defined in the 1991 and 2006 law. The Accelerator Driven System (ADS) potential for Minor Actinides (MA) transmutation is being investigated in this framework.

The present paper attempts to describe ADS influence on MA inventory in double strata scenarios. The nuclear plants stratum dedicated to electricity production generates MA which feeds ADS fleet stratum devoted to transmutation.

J-B. Clavel is with Subatech, Subatomic physics and associated technology laboratory, 4 rue Alfred Kastler La Chantrerie BP 20722 44307 Nantes cedex 3, France (phone: +33 2 51 85 84 63; fax: +33 2 51 85 84 79; e-mail: Jean-Baptiste.Clavel@subatech.in2p3.fr).

N. Thiollière is with Subatech, Subatomic physics and associated technology laboratory, 4 rue Alfred Kastler La Chantrerie BP 20722 44307 Nantes cedex 3, France (phone: +33 2 51 85 86 42; fax: +33 2 51 85 84 79; e-mail: thiollie@subatech.in2p3.fr).

B. Mouginot is with Subatech, Subatomic physics and associated technology laboratory, 4 rue Alfred Kastler La Chantrerie BP 20722 44307 Nantes cedex 3, France (phone: +33 2 51 85 85 64; fax: +33 2 51 85 84 79; e-mail: Baptiste.Mouginot@subatech.in2p3.fr).

TABLE I
Variables and Units

Symbol	Quantity	Units
IV	isotopic vector	
MA	minor actinide	
FP	fission product	
K_{eff}	multiplication factor	
ρ	reactivity	
φ_{Max}	maximum neutron flux inside the reactor core	$1.cm^{-2}.s^{-1}$
φ_m	mean neutron flux inside the reactor core	$1.cm^{-2}.s^{-1}$
P_{MaxPin}	maximum pin thermal power	1 W
P_m	Total thermal power	1 W
$<E_f>$	average fission energy	$1\ J \rightarrow 6.2415096471204 .10^{12}$ MeV
σ_f	fission cross section	$1\ b \rightarrow 10^{-24}\ cm^{-2}$
N_N	nuclei number	
Γ_{ff}	flux form factor	
N_{pin}	pin number per assembly	
N_{Ass}	total fuel assembly number	

To define the electricity generator stratum, two cases have been taken into account. On one hand, world nuclear power has been considered as constant, which implies that uranium reserves can provide the electricity needs. With this scenario, breeder reactors are not required and European Pressurized Reactor (EPR) can be used as the first stratum. On the other hand, a significant increase in global electricity production from nuclear energy is considered, which implies the use of fast breeder reactors leading to uranium saving.

A major part of the long-term radiotoxicity of nuclear waste is due to the plutonium resulting from the used nuclear fuel. This implies a plutonium closed cycle strategy in the AM management. As a consequence, a plutonium multi-reprocessing into MOX fuel is taken into account for both cases. Figure 1 illustrates scenarios assessed in the present work.

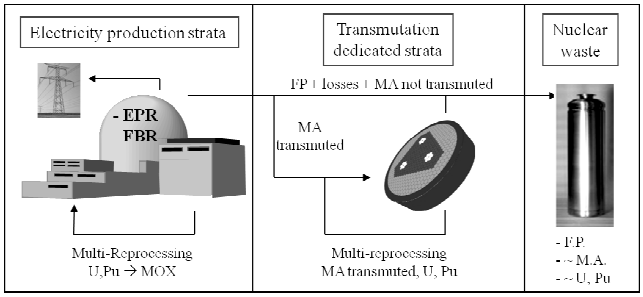

Fig. 1 Schematic drawing illustrating double strata scenario principle

The EFIT (European Facility for Industrial Transmutation) [1], [2] ADS concept is usually the reference for MA transmutation studies. The induced MA inventory in nuclear cycle (reactors, treatment and fuel fabrication factories) is large compared to transmutation in fast reactor scenarios [3]. Moreover, many ADS are needed to burn MA flux produced by the electricity strata. The MUST (MUltiple Spallation Target) ADS concept has been developed at Subatech Laboratory with the aim of avoiding those major drawbacks.

MUST ADS consists of three spallation targets disposed inside a sub-critical core (K_{eff} ~ 0.97). To produce the neutron flux required to maintain a constant number of fission inside the core, a high energy (1 GeV) and high intensity (~30 mA) proton beam is divided into three secondary beams transmitted the spallation targets. A three neutron sources concept provides a regular and high specific power inside the subcritical core.

The MUST ADS is characterized by a higher thermal power between 1.2 GWth and 1.5 GWth. In addition, the irradiation time during which minor actinides are burned (cycle time) is extended up to five years.

The following paper describes the methodology used to design the MUST ADS concept, including neutron and thermal-hydraulic calculations.

II. Simulation Tools

A. MURE

The code MURE (MCNP Utility for Reactor Evolution) [4] performs reactor time-evolution and is distributed by OECD NEA. MURE provides a C++ interface to the Monte-Carlo transport code MCNP [5] providing an easy way to build MCNP input file. The nuclides evolution calculation during irradiation or cooling phase is done by solving the Bateman equations.

The BATH class (Basic Approach of Thermal Hydraulic) [6], [7] included in the code MURE calculates temperature through coupled neutron/thermal-hydraulics simulation. This class is the faster way to manage thermal-hydraulics in MURE, while more precise methods are implemented. This class provides coolant and fuel temperature as a function of the axial position inside a pin element lattice. The method is based on the heat and mass transfer equations. Inputs parameters are the coolant type, the coolant inlet temperature, the coolant mass velocity, and the fuel pin thermal power.

B. ACDC

The MURE code provides precise results at the cost of a high cpu time required by the Monte Carlo simulations. Thus, a simple evolution code named ACDC (ACtinides Depletion Code) has been developed to perform preliminary evolution calculations.

ACDC manages 21 connected nuclides, builds and solves Bateman equation with the following isotopes properties: proportions, cross sections, average number of neutrons emitted per fission and half-lives. An input file containing the scenario strategy and the corresponding microscopic mean cross sections $<\sigma(E)>$ is provided to ACDC. In (1), $\sigma(E)$ and $\phi(E)$ are the microscopic cross sections and the neutron flux according to energy.

$$<\sigma(E)>=\frac{\int \sigma(E).\phi(E)dE}{\int \phi(E)dE} \quad (1)$$

The program performs the nuclei evolution considering a constant neutron flux. Several cycles are computed and the approximated fuel composition at equilibrium state was obtained. A benchmark with the code MURE has been performed from a typical five years reactor cycle leading to an error margin under 10%.

III. Core Design and Homogeneous Fuel Characterization

A methodology has been developed to set the assembly's geometry, the distance between spallation targets and the core size. The homogeneous fuel, i.e. the isotopic vector at equilibrium, used in the ADS at the beginning of cycle is also estimated.

A. Scenario parameters

The MUST concept contains spallation targets composed by a 65 cm lead-bismuth cylinder to have the maximum neutron production at the bottom half of the core.

The coolant must induce a fast spectrum to maximize the fission reaction [8]. Lead and sodium are investigated as it is most mature technologies.

The homogeneous fuel is mixed with inert matrix to adjust the reactivity and improve the thermal conductivity. The fuel type and the inert matrix composition determine pins operating temperature.

The reactors that compose the stratum dedicated to electricity production have been discussed in part I. This is important to stress that isotopic composition of MA that feed ADS strongly depends on the considered first stratum.

The first transmutation strategy consists in burning neptunium, americium and curium all together. A second strategy accounts for the neptunium and the americium transmutation. Finally, americium transmutation is also considered.

Table II lists the set of the investigated options. Nevertheless, in this work, we focus on presenting the methodology and results related to the reference case only.

TABLE II
Scenario Parameters and ADS Characteristics

Composition of first stratum	EPR[a], FBR, …
Fuels of first stratum	MOX multi-recycled[a] …
Transmutation strategies	Am | Np,Am | Np, Am,Cm[a]
ADS coolant's	Na[a], Pb, gaz,…
Fuel type	Oxide[a], metallic, …
Inert matrix fuel	MgO[a], Mo, …

[a] Reference case

B. Sodium Cooled Assembly Design

Fuel assembly and pin dimension are based on the SFR (Sodium Fast Reactor) design [9]. The control assembly used to maintain the sub-critical reactivity is composed by 37 B4C rods [10]. The fuel and control assembly characteristics are summarized in the Table III and presented in figure 2.

C. Sub-critical Core Design

The reactor size and the distance between spallation targets are optimized to obtain a regular flux in the sub-critical core while maximizing the thermal power.

The first composition at equilibrium (IV1) dedicated to fill the sub-critical core is determined using ACDC calculation. The averaged neutron flux is 10^{15} n.cm^2.s^{-1} and cross sections are extracted from ENDFB7 [11] at 500keV.

A one spallation target sub-critical reactor is simulated with MCNPX. The core radius is chosen large enough (about 4 meter) to be considered as a R-infinite core. The inert matrix quantity is adjusted to get K_{eff} ~ 0.97. Finally, the one target neutron flux cartography is calculated.

We assume the neutron flux, in a three spallation target subcritical core to be the sum of the three independent target fluxes. This hypothesis allows for rapid testing of numerous configurations. φ_{Max_1T} defines the maximum neutron flux in ADS with one spallation target, φ_a the core averaged neutron flux and φ_l the lower local neutron flux. Following the defined nomenclature, three constraints are imposed and tested for a different ADS configurations:

1. $\varphi_a \geq 0.6\ \varphi_{Max\_1T}$
2. $\varphi_l \geq 0.3\ \varphi_{Max\_1T}$
3. Assemblies number should be maximum

Figure 2 shows the final ADS MUST concept, which needs 12 assembly control to maintain a sub-critical level. The distance between spallation targets and center is 119.396 cm.

TABLE III
CORE AND ASSEMBLY CHARACTERISTICS FOR 3 TARGETS AND SODIUM COOLED MUST ADS

Fissile length (mm)	1000
Assembly grid pitch (mm)	206.8
Structure thickness (mm)	0.8
Fuel pellet and hole diameter (mm)	9.5 and 2
Fuel pin void thickness (mm)	0.15
Fuel pin cover thickness (mm)	0.7
Number of pin per assembly	271
Fuel assembly number per MUST ADS	213
B_4C and cover control rod diameter (mm)	24.4 and 16.4
Number of control rod per assembly	37
Coolant mass velocity (kg.m^{-1}.s^{-1})	2094
Coolant inlet temperature (K)	673

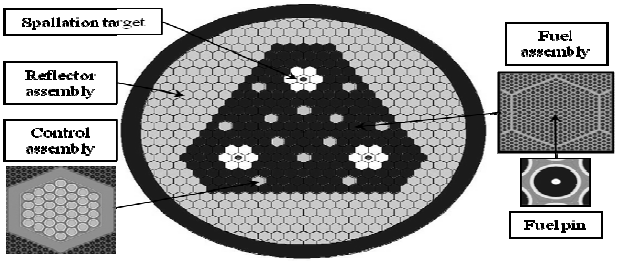

Fig. 2 Schema of three spallation targets and sodium MUST ADS

D. First Fuel Composition Update

The first fuel composition update has three phases. Microscopic cross sections are recalculated from an assembly simulation with MCNPX. Then, an ACDC calculation including new cross sections and IV1 is performed, providing a refined isotopic vector at equilibrium (IV2). A core simulation with MCNPX permits to adjust the inert matrix amount to reach a K_{eff}~0.97 in the ADS filled with the new fuel composition.

E. Thermal-Hydraulic Study

The MURE thermal-hydraulic calculations are computed with a fuel pin thermal power ranging from 26 kWth to 37 kWth and the set of parameters listed on table III.

For each thermal power, the temperature at the pin center according to axial position is extracted. The temperature profile (figure 3) is then fitted following:

$$T(z) = a + bz + A\exp(-\frac{(z - z_m)^2}{2.\sigma^2}) \quad (2)$$

With a and b the zero and first order polynomial coefficients respectively, A, z_m and σ the Gaussian fitting parameters. Finally, the hot point temperature (T_{hp}) corresponding to the function maximum is determined. A typical hot-point distribution profile according to the pin thermal power is presented figure 4.

The dissociation temperature for a CERCER fuel is about 1653 K [2]. A five per cent lower temperature (1570 K) is chosen to provide an adequate level a safety. The maximum pin thermal power is then calculated using linear interpolation, as illustrated in figure 4.

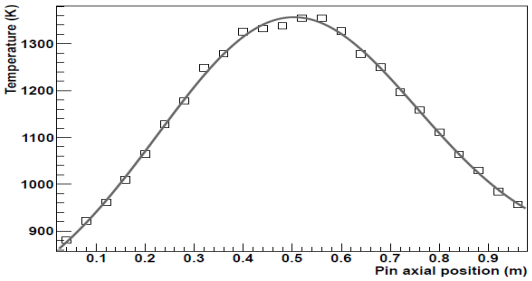

Fig. 3 Fuel hot point temperature according to pin axial position.Grey line represents the distribution fit

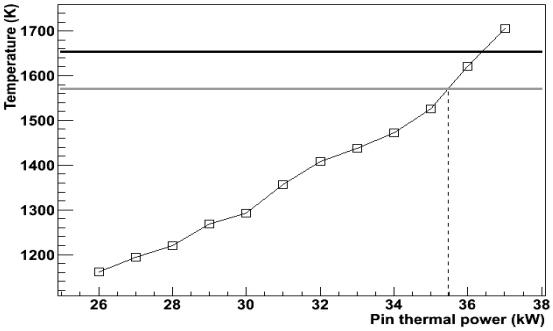

Fig. 4 Hot point temperature according to pin thermal power.Black and grey lines represents 1653K and 1570K. Dashed vertical line is the pin power selected for the reference case.

F. Second Fuel Composition Update

Further ACDC calculations were performed to account for the mean neutron flux inside the ADS resulting from the thermal-hydraulic results. The MUST ADS flux spatial distribution is calculated with MCNPX from a 3 dimensional mesh (figure 5). The form factor (Γ_{ff}) defined as the ratio between the maximum flux (φ_{Max}) and the averaged flux (φ_m) is then evaluated. The neutron flux and thermal power of the selected MUST ADS concept can be obtained from (3), (4) and (5).

$$\varphi_{Max} = \frac{P_{MaxPin}}{[< E_f > . \sum_i N_{Ni} . \sigma_{fi}]} \quad (3)$$

$$\varphi_m = \frac{\varphi_{Max}}{\Gamma_{ff}} \quad (4)$$

$$P_m = \frac{P_{MaxPin} . N_{Pin} . N_{Ass}}{\Gamma_{ff}} \quad (5)$$

$<E_f>$ corresponds to the fission energy (200 MeV), N_{Ni} is the number of fissile atoms inside the pin and σ_{fi} is the microscopic fission cross section. N_{pin} and N_{ass} stand for the number of pins per assembly and the number of assemblies inside the MUST ADS respectively. The induced isotopic vector calculated with ACDC is found to be closed to equilibrium.

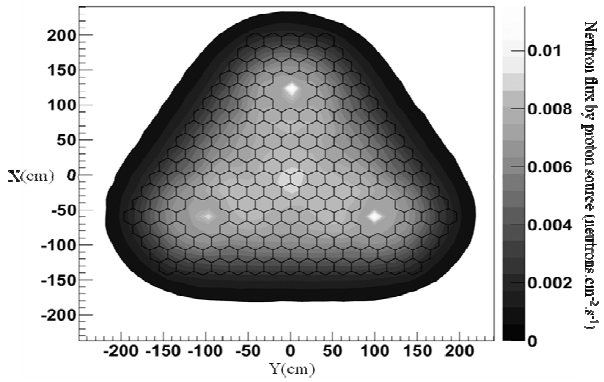

Fig. 5 Integrated fluxes over the entire height depending on the position

MCNPX calculation is then achieved to adjust the inert matrix amount to get a K_{eff} around 0.97 with the control rods lowered of 1/5. From each K_{eff} adjustment results an increase of the inert matrix amount. Since inert matrix improves the thermal conductivity, this increase implies a reduction of the functioning temperature in comparison to the one calculated during the thermal-hydraulic step.

IV. Cycle Evolution

A. Control of Reactivity

Twelve control assemblies are displayed inside the sub-critical core to maintain the K_{eff} around 0.97. Evolution without control assemblies has shown that the reactivity increases by few per-cents during the cycle. The initial control assemblies position is thus located at a fifth of the core height.With the aim of evaluating the new control assemblies position at each time step in the evolution, a new control assemblies management method has been implemented in MURE. It consists of calculating the reactivity at three control assembly positions: without control assemblies, 1/4 inserted and fully inserted. From the reactivity induced data, the following equation parameters are calculated.

$$\rho = 1 - \frac{1}{K_{eff}} \quad (6)$$

$$\alpha = \frac{\rho(H) - \rho(0)}{H} \quad (7)$$

$$\rho(x) = \rho(0) + \alpha . x - [\rho(0) + \frac{\alpha . H}{4} - \rho(\frac{H}{4})] . \sin(\frac{2 . \pi . x}{H}) \quad (8)$$

The set of equations represent the reactivity according to the control assemblies position inside the core. The position which corresponds to k_{eff}=0.97 is calculated and updated to carry on the calculation. The method precision depends on the CPU time dedicated to the three calculations and the time step evolutions.

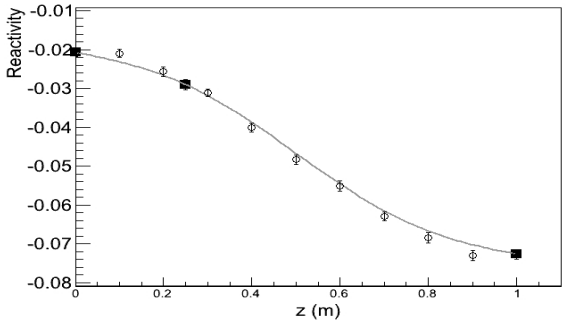

Fig. 6 Reactivity according to the control assembly position.The black square are used to calculate the three parameters of (8). The grey line represents the corresponding function. Circle represents MCNP calculation for other positions.

Figure 7 shows the k_{eff} evolution as a function of the irradiation time. For a five years irradiation phase, K_{eff} is contained between 0.965 and 0.975. The seven years cycle time evolution has been superimposed showing that the cycle time can be extended with a constant K_{eff} (to be confirmed by a dpa estimation).

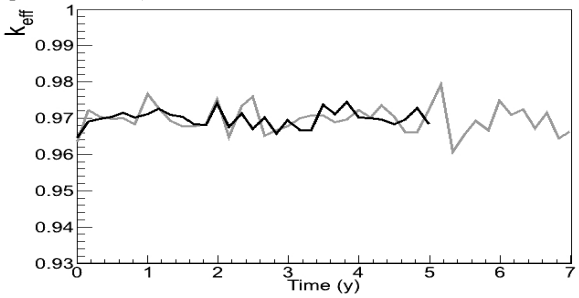

Fig. 7 The black line is the K_{eff} evolution for MUST ADS in the reference case scenario, and the grey line corresponds to 1.45 GW_{th} MUST ADS with seven years irradiation time

B. Fuel Cycle Evolution

Isotopic inventory has been precisely calculated with the code MURE.

Figure 8 represents main neptunium and americium isotopes contributors. The irradiation phase occurs until five years and a cooling phase is taken into account from 5 to 12 years. The depletion phase confirms the efficiency of neptunium and americium transmutation.

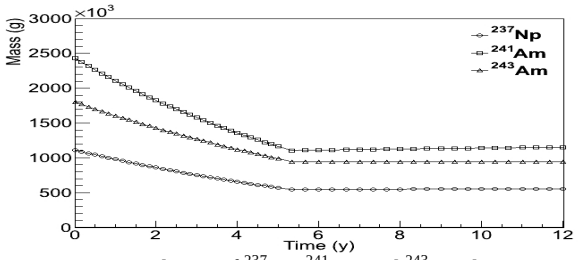

Fig. 8 Masse evolution of ^{237}Np, ^{241}Am and ^{243}Am during one cycle.

Figure 9 shows most important curium isotopes evolution on one irradiation cycle following by 7 cooling years. It shows that the ^{244}Cm amount decreases during the cycle. Other curium isotopes are not transmuted. This can be explained by the fact that curium is transmuted and simultaneously generated by lower atomic masses isotopes.

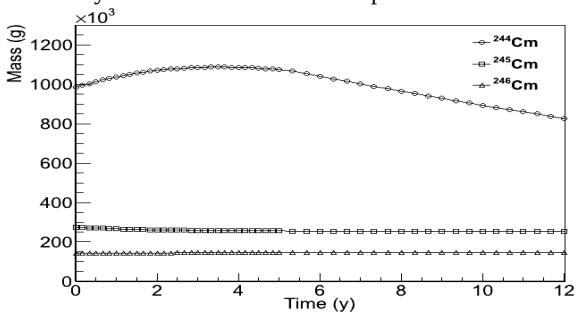

Fig. 9 Masse evolution of main curium isotopes during one cycle

Figure 10 represents uranium and plutonium isotopes with significant amount. Plutonium and uranium isotopic vector is clearly at equilibrium.

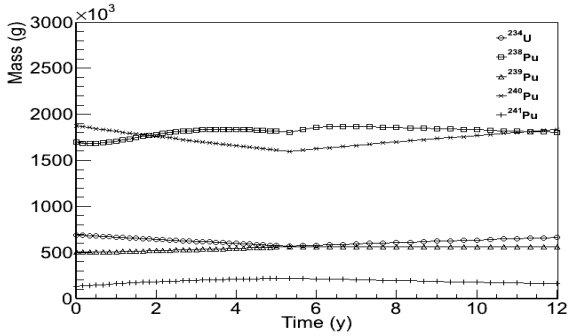

Fig. 10 Masse evolution of main uranium and plutonium isotopes during one cycle

C. Equilibrium scenario analysis

The table IV provides masses balance for most important isotopes. Only nuclides with initial mass higher than 1% of the total masse have been taken into account.The last columns provides isotopes variation rate, which are for the three most important transmuted nuclides (^{237}Np, ^{241}Am and ^{243}Am), close to 50%. The two major plutonium isotopes (^{238}Pu and ^{240}Pu) are closed to equilibrium since there is few per-cent variation rates.Nevertheless, additional calculations will be done in the future to improve the equilibrium state for nuclides, such as ^{241}Pu.The total transmuted mass can be obtained from the table IV. Closed to 570 kg of MA per year can be burned in the MUST ADS for this equilibrium scenario. As EPR, including a plutonium multi-reprocessing strategy produces around 110 kg/year [12] of MA, a MUST ADS can approximately manage the flux providing by five.

The table V provides equivalent data as table IV for a scenario including the americium transmutation without neptunium and curium management.The two main americium contributors transmutation rates are closed to 40%. Most important plutonium and uranium isotopes are closed to equilibrium state. Nevertheless, more than 700 kg of ^{244}Cm are generated, which is not negligible. This shows that scenarios excluding curium management leads to an additional accumulation.

V. Conclusion and Prospects

The first research objective in the scope of French law related to nuclear waste management concerns MA separation and transmutation. A new ADS concept, characterized by a high thermal and specific power has been developed. Three spallation targets disposed inside a sub-critical core are irradiated by a ~30 mA and 1 GeV proton beam. The precise methodology to design the core and to calculate the fuel isotopic vector at equilibrium has been described. Coupled neutron and thermal-hydraulic analysis has showed that the MUST ADS can operate at 1.4 GW_{th}, without reaching critical fuel temperature. Transmutation capabilities have been presented for the reference case. The presented scenario considers that EPR provides electricity including a plutonium multi-reprocessing strategy.

TABLE IV
Isotopic Evolution

Isotope	Initial masse (kg)	Mass variation (kg)	Variation rate %
234U	691,151	-26,60	-3,85
237Np	1111,73	-555,98	-50,01
238Pu	1698,52	104,34	6,14
239Pu	501,362	61,88	12,34
240Pu	1874,57	-40,94	-2,18
241Pu	131,513	29,18	22,18
241Am	2430,27	-1277,65	-52,57
243Am	1800,27	-855,12	-47,50
244Cm	983,739	-156,65	-15,92
245Cm	275,533	-20,89	-7,58
246Cm	142,413	4,62	3,24

Masses balance for the reference case scenario in one cycle including cooling. Isotopes which have more than 1% of the total mass are represented.

TABLE V
ISOTOPIC EVOLUTION

Isotope	Initial mass (kg)	Variation mass (kg)	Variation rate %
234U	1270,46	-22,64	-1,78
235U	198,61	10,68	5,38
236U	175,43	3,05	1,74
238Pu	2402,21	14,76	6,15
239Pu	638,00	50,59	7,93
240Pu	1245,73	105,23	8,45
241Am	5278,19	-2193,66	-41,56
242Am*	204,54	-17,46	-8,54
243Am	4235,30	-1464,32	-34,57
244Cm	0,00	735,16	

Masses balance for a scenario related to the transmutation of americium only. Isotopes which have more than 1% of the total mass are represented.

Np, Am and Cm flux from EPR are transmuted into a sodium cooled MUST ADS with a five years irradiation time. The ADS homogeneous fuel is mixed with MgO to limit the core reactivity and improve the fuel thermal conductivity. One MUST ADS transmutes MA up to 570 kg/year for the reference case.

Further methodology development aim to account for safety criteria, such as void and temperature coefficient calculation and dpa estimation closed to spallation targets.

Finally, the proposed method will be used to study extended MA transmutation scenarios. A code is currently developed to manage complex scenario including reactor transition. The potential of the MA transmutation with MUST ADS will be assessed for the French long-term nuclear scenario.

REFERENCES

[1] A. Barbensi et al. EFIT: The European Facility For Industrial Transmutation of minor actinide. In: Proc. 8th Int. Topical Mtg. on Nuclear Applications and Utilization of Accelerators, ACCAPP'07. Pocatello, Idaho, USA, 2007, pp. 885–892.
[2] C. Artioli, et al. Optimization of the minor actinides transmutation in ADS: the European Facility for Industrial Transmutation-EFIT-Pb Concept. In: Proc. Int. Conf. ACCAPP'07, Pocatello, USA, July 30–August 2.
[3] V. Romanello, et al. Comparative study of fast critical burner reactors and subcritical accelerator driven systems and the impact on transuranics inventory in a regional fuel cycle. Nuclear Engineering and Design 241 (2011) 433–443
[4] O. Méplan et al., 2009. MURE, MCNP Utility for Reactor Evolution: couples Monte Carlo transport with fuel burnup calculations. Computer Program Services of OECD Nuclear Energy Agency. <http://www.oecd-nea.org/tools/abstract/detail/nea-1845>.
[5] J.F. Briesmeister, 2000. MCNP – A General Monte Carlo N-Particle Transport Code – Version 4C. Manual LA-13709-M, Los Alamos National Laboratory.
[6] N. Capellan et al., 2009. 3D coupling of Monte Carlo neutronics and thermal-hydraulics calculations as a simulation tool for innovative reactor concepts. In: Proc. Int. Conf. GLOBAL 2009, Paper 9274.
[7] N. Capellan, 2009. Couplage 3D neutronique thermohydraulique. Développement d'outils pour les études de sûreté des réacteurs innovants. Thesis, France, November 2009.
[8] J. Guidez et al. Phenix: the irradiation program for transmutation experiment. In Eighth Information Exchange Meeting "Actinide and Fission Product Partitioning & Transmutation", USA,November 2004.
[9] G. Rimpault, L. Buiron, P. Sciora and F. Varaine. Towards GEN IV SFR design: Promising ideas for large advanced SFR Core Designs. In proceedings *International Conference on the Physics of Reactors PHYSOR'2008*, Interlaken, Switzerland, September 2008.
[10] L. Buiron, D. Blachet and N. Stauff. Sodium Fast Reactor Task Force. In meeting NEA WPRS EGRPANS on May 2011.
[11] M.B. Chadwick et al. ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data. Nuclear Data Sheets, vol 112, 2011.
[12] S. David. Private communication. Institut de Physique nucléaire d'Orsay, 2012