# FUEL CYCLE STUDIES ON MINOR ACTINIDE BURNING IN GAS COOLED FAST REACTORS

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**Abstract.** The gas cooled fast reactor (GFR) is a Generation IV reactor type considered as an alternative fast neutron reactor design aimed to improve the sustainability of nuclear energy by improving the uranium utilization efficiency. Besides fuel breeding the especially fast neutron spectrum of the GFR also provides excellent opportunities for minor actinide (MA) burning. Development of the design of a 2400 MWth full-scale and a 70 MWth demonstrator GFR (ALLEGRO) is going on in the Euratom sponsored GoFastR project. The paper presents fuel cycle studies performed at the Budapest University of Technology and Economics (BME) for the assessment of the transmutational capabilities of the 2400 MWth GFR design in the framework of the GoFastR project. Homogeneous recycling of minor actinides into the GFRs was considered and their MA consumption was investigated. The results confirm the expectation that GRFs can also be applied for burning of MAs produced in other reactors.

## 1. Introduction

Fast reactors have the unique ability to be sustainable by, not only being able to generate their own fuel, but through being able to burn minor actinides to reduce the quantity and radiotoxicity of nuclear wastes. The latter ability enables fast reactors to not only burn the minor actinides produced by themselves but, in addition, the minor actinides arising from legacy wastes and thermal reactors in the nuclear park. The Generation IV International Forum [1] has identified six systems which merit development to achieve the goals of sustainability, proliferation resistance, economics and improved safety. Of these six systems, three are fast reactors, the gas-cooled fast reactor (GFR) being one of these. The sustainability goal has been developed further within Europe through the establishment of the Sustainable Nuclear Energy Technology Platform (SNE-TP). As well as setting out a vision for the development of sustainable nuclear energy within Europe, the SNE-TP has devised a Strategic Research Agenda (SRA) [2] that identifies the priorities for research through which this vision can be realized. In this context, sodium-cooled fast reactors have been identified as the near-term technology that would allow rapid deployment of fast reactors. The SRA also identifies that gas-cooled and the lead-cooled fast reactors could be deployed in the longer-term. Both of these technologies will be capable of operating at higher temperatures than the use of a liquid sodium coolant will allow. Therefore higher efficiency electricity generation and a wider range of non-electrical applications becomes possible, such as the generation of high quality process heat and efficient mass production of hydrogen. In addition, the harder neutron spectrum improves the transmutation capabilities allowing minor actinides to be destroyed more effectively.

The Euratom sponsored GoFastR project [3][4] concentrates on the gas-cooled fast reactor (GFR). The design goals for GFR are ambitious, aiming for a core outlet temperature of around 850°C, a compact core with a power density of about 100MW/m$^3$, a low enough plutonium inventory to allow wide deployment, a self-sustaining core in terms of plutonium consumption, and a proliferation resistant core by not using specific plutonium breeding elements.

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The initial core design of the 2400 MW thermal power GFR was provided by the CEA at the beginning of the project [5]. The core is composed of carbide fuel element cladded with ceramic composite SiC/SiC fibre material. The ceramic cladding is needed in order to reach high operation temperature and withstand as high as 1200 °C cladding temperature during transients.

At the Institute of Nuclear Techniques of the Budapest University of Technology and Economics (BME) fuel cycle studies were performed to investigate the fuel breeding and minor actinide burning capabilities of the GFR2400. Fuel cycle studies concerning transmutation options pose two main challenges:

- The evaluation of the different transmutation options can be performed based on the detailed composition of the final waste, which requires the tracking of a wide range of isotopes in the fuel cycle and the determination of the accurate composition of the spent fuel.
- Minor actinide recycling options in transmutation fuel cycles also results in a wide range of possible isotopic compositions of the core, influencing the neutron spectrum and therefore the burnup process.

Most scenario codes contain only cross-section sets at a few burnup steps, which is not flexible enough for such analysis[6]. On the other hand detailed burnup calculations are too time consuming to be inserted in the simulation of the complete fuel cycle. Development of a quick and flexible burnup model was started at the BME in order to cope with these problems.

## 2. Calculational method

### *2.1. Mathematical principles of the burnup model*

The evolution of the isotopic composition in the core during the burnup can be described by the well-known Bateman differential equation system, which represents a balance equation for the number density of each isotope:

$$\frac{dN_i}{dt} = \sum_{j \neq i} \left(\sigma_{j\to i}\Phi + f_{j\to i}\lambda_j\right)N_j - \left(\sigma_i\Phi + \lambda_i\right)N_i\,, \qquad (1)$$

where $N_i$ is the number density if isotope $i$ in the core, $\sigma_{j\to i}$ is the microscopic cross-section of the reaction leading from isotope $j$ to isotope $i$, $\sigma_i$ is the total cross-section of the reactions consuming isotope $i$, $f_{j\to i}$ is the branching ratio of the decay form of isotope $j$ leading to isotope $i$, $\lambda_j$ is the decay constant of isotope $j$. $\Phi$ is the average one group neutron flux in the core, which can be determined from the power:

$$\Phi = \frac{P}{\sum_{i=1}^{n} E_{f,i}\sigma_{f,i}N_i + E_\gamma \sum_{i=1}^{n} \sigma_{\gamma,i}N_i}, \qquad (2)$$

where $P$ is the thermal power of the reactor, $E_{f,i}$ is the energy released in a fission of fissile isotope $i$ (~200 MeV), $\sigma_{f,i}$ is the fission cross-section of isotope $i$, $E_\gamma$ is the average energy released in the core in an (n,γ) reaction (~5 MeV) while $\sigma_{\gamma,i}$ is the (n,γ) cross-section of isotope $i$. Obviously the above one-group cross-sections have to be generated with a proper weighting in order to reproduce the average reaction rates in the core. Combining the number densities into a single isotope vector **N,** the differential equation system (1) can be written in a matrix form:

$$\frac{d\overline{\mathbf{N}}}{dt} = \overline{\overline{\mathbf{A}}}_k\,\overline{\mathbf{N}}\,, \qquad (3)$$

where the matrix $\mathbf{A}_k$ is composed of coefficients which are determined by the cross-sections and decay constants. Index $k$ distinguishes between the coefficient matrices at different time steps of the

burnup, since the change in the isotopic composition results in the change of the spectrum and therefore also the one-group collapsed cross-sections. This is why the time consuming core calculations need to be repeated during the burnup calculations or burnup dependent cross-section sets have to be used. In order to develop a quick and flexible burnup calculation method, we chose to describe the cross-sections as functions of the isotopic composition based on fitting to the results of numerous core calculations.

The 16 most important actinide isotopes were considered for the description of the isotopic composition ($^{234\text{-}236,238}$U, $^{237,239}$Np, $^{238\text{-}242}$Pu, $^{241,242m,243}$Am, $^{244,245}$Cm) as well as the total quantity of fission products. Due to the large number of variables, fitting of the cross-sections is feasible only with the Moore-Penrose pseudo inverse [7]. A limitation of this method is that it is only applicable for fitting of functions (*f(x)*) which are linear in their $a_j$ coefficients:

$$f(\bar{x}) = \sum_j a_j f_j(\bar{x}) \tag{4}$$

where $f_j(x)$ can be any function of $x$. Polynomials meet this condition and since preliminary investigations had shown that accuracy of the fit does not improve with the insertion of third order terms, the following polynomial was chosen:

$$\sigma(\bar{\mathbf{N}}) = \mathrm{a}_0 + \sum_{\mathrm{j}=1}^{\mathrm{n}} \mathrm{a}_\mathrm{j} \mathrm{N}_\mathrm{j} + \sum_{\mathrm{j}=1}^{\mathrm{n}} \sum_{\mathrm{k}=\mathrm{j}}^{\mathrm{n}} \mathrm{a}_{\mathrm{j,k}} \mathrm{N}_\mathrm{j} \mathrm{N}_\mathrm{k} \tag{5}$$

(5) can be written in a vector form:

$$\sigma(\bar{\mathbf{N}}) = \bar{\mathbf{N}}' \bar{\mathrm{a}}\,, \tag{6}$$

where the n+n$^2$ element of the extended isotope vector $\bar{\mathbf{N}}'$ contains besides the isotope vector $\bar{\mathbf{N}}$ the cross-products in the last term of (5). If the rows of a matrix $\bar{\bar{\mathbf{N}}}'$ contains the above extended isotope vectors $\bar{\mathbf{N}}'$ of the different compositions and vector $\bar{\boldsymbol{\sigma}}_\mathrm{C}$ contains the corresponding calculated cross-sections, then the vector **a** of the fitting parameters $a_j$ and $a_{j,k}$, can be determined by a least-square-fit based on the theorem of the Moore-Penrose pseudo inverse:

$$\bar{\mathbf{a}} = \bar{\bar{\mathbf{N}}}'^{+} \bar{\boldsymbol{\sigma}}_\mathrm{C}\,, \tag{7}$$

where the pseudo inverse $\bar{\bar{\mathbf{N}}}'^{+}$ can be determined by definition as:

$$\bar{\bar{\mathbf{N}}}'^{+} = \left( \bar{\bar{\mathbf{N}}}' \bar{\bar{\mathbf{N}}}'^{*} \right)^{-1} \bar{\bar{\mathbf{N}}}'^{*} \tag{8}$$

By substituting (5) in (1) the differential equation system (3) takes the following form:

$$\frac{d\bar{\mathbf{N}}}{dt} = \bar{\bar{\mathbf{A}}}\left(\bar{\mathbf{N}}\right)\bar{\mathbf{N}}\,, \tag{9}$$

where matrix $\bar{\bar{\mathbf{A}}}\left(\bar{\mathbf{N}}\right)$ is a function of isotope vector $\bar{\mathbf{N}}$, since it contains the coefficients determined by the cross-sections depending on the isotopic composition. This is not a linear differential equation system anymore, but its numerical integration is still fast enough to be integrated into a fuel cycle simulation.

## *2.2. Core calculations*

In order to perform the least-square-fit in (7) the one group cross-sections have to be calculated for numerous different compositions with detailed core calculations. Due to the large number of fitted parameters practically a few thousand calculations have to be performed, therefore a simplified core model had to be set up, which could provide the homogenized one-group cross-sections for the core.

In the case of the GFR2400 the codes of the SCALE code system [7] were used for this purpose. First the elementary cell of the GFR-2400 was modeled which consists of the ceramic fuel pin with the surrounding He coolant. One dimensional cylindrical model was built, where the height of the core was taken into account with buckling correction. The T-XSEC sequence produced the resonance self-shielded 238 groups cross-section set based on ENDF/B-VII data. The XSDRN-PM one dimensional discrete ordinates code was applied to generate the cell weighted homogenized cross-sections for the elementary cell. The hexagonal fuel assembly was also modeled in a one dimensional cylindrical geometry containing the homogenized fuel region, the ceramic assembly wall and the gap between the assemblies. Other core components (axial and radial reflector, rod followers and gas plenum) were homogenized in infinite homogenous medium approximation. The 3D model of the core was built from the homogenized fuel assemblies and other core components as it can be seen in Fig. 1. Full core calculations were performed by the KENO-VI multigroup Monte Carlo criticality code in 238 groups. The correctness of the homogenization techniques applied was checked against detailed Monte Carlo calculations.

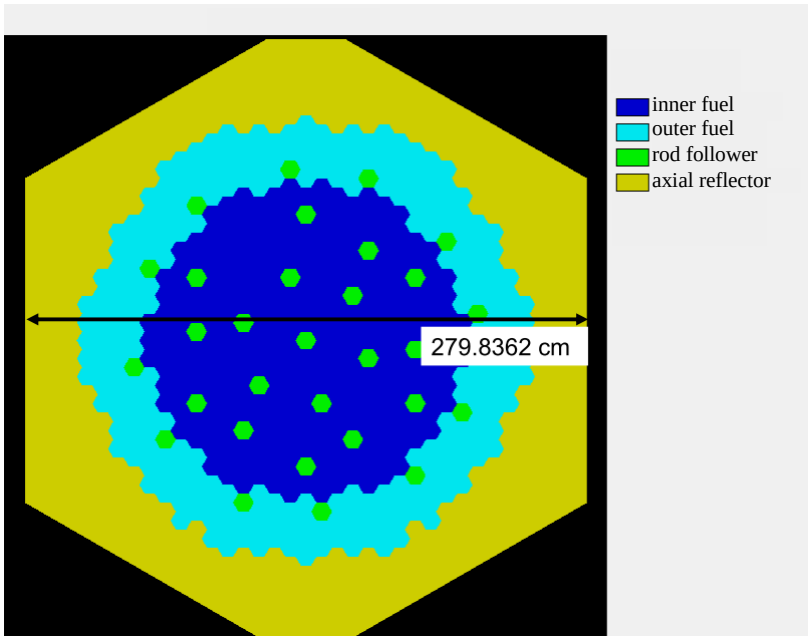



*FIG.1. KENO-VI model of the GFR2400 core*

More than 2000 core calculations were performed with the above simplified core model assuming different isotopic compositions for the fuel. The actinide composition of the fuel was randomly sampled for the different calculations taking into account the following constraints:

- Pu number density in the fuel changed between ~10-25 % of the total actinide number density. The ratio of the Pu content of the inner and outer core was kept at 0.8 which is the case also for the initial loading.
- MA number density in the fuel changed between 0-10 % of the total actinide number density..
- The rest of the actinide content of the fuel was U.
- Fission products were considered with a general fission yield vector. Their quantity is directly related to the burnup level of the fuel which was varied between 0 and approximately 10%.

— Isotopic composition of the heavy metal elements was also randomly distributed, but limits were set to every isotope considering the isotopic composition of the initial charged fuel and the equilibrium composition estimated by preliminary calculations[8].

The actinide reaction rates, the average flux in the inner and outer core and the $k_{eff}$ were recorded from the calculations. The reaction rates are defined as:

$$R_i^j = \sigma_i^j \Phi^j N_i^j V^j , \tag{10}$$

where index *j* refers to the inner or outer core region. Since the simplified burnup model is a point model where the outer and the inner regions are not handled separately and cross-sections, number densities and flux are recorded only for the complete core, a weighted average cross-section needs to be calculated. The following weighting scheme was chosen:

$$\overline{\sigma_i} = \frac{\sum_j R_i^j}{\sum_j \Phi^j N_i^j V^j} \tag{11}$$

which implies the use of a weighted average number density and average flux:

$$\overline{N_i} = \frac{\sum_j \Phi^j N_i^j V^j}{\sum_j \Phi^j V^j} \tag{12a}$$

$$\overline{\Phi} = \frac{\sum_j \Phi^j V^j}{\sum_j V^j} \tag{12b}$$

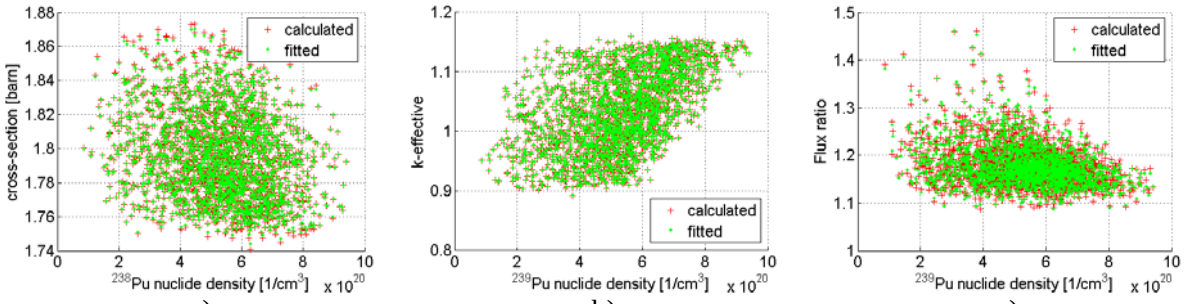
in order to preserve the total reaction rate:

$$\sum_j R_i^j = \overline{\sigma_i}\ \overline{\Phi}\overline{N_i} V_{core} \tag{13}$$

Fitting according to the above mentioned Moore-Penrose method was done for the weighted cross-sections in (11). Examples for the results of the fitting process can be seen in Fig. 2. Results show the fitted functions can reproduce the calculated cross-sections with a few percent of maximum deviation.

### *2.3. Burnup model*

The fitted cross-sections were applied in a burnup model developed in MATLAB for the numerical solution of (9). Since in this burnup model the flux and the number densities are calculated only for the complete core and not for the inner and outer region, the weighted average number density $N_i$ cannot be calculated according to (12a). Instead the fact is utilized that the ratio of the number density in the inner core and in the outer core is kept constant for all isotopes. The ratio of the flux in the inner and outer core changes with the composition, but this dependence can be determined by applying the same fitting process as in the case of the cross-sections (see Fig. 2.). If one defines number density ratio *C* and the flux ratio *F*:




a) b) c)

*FIG. 2. Results of the fitting compared to the calculated points as a function of $^{239}Pu$ nuclei density: a) $^{239}Pu$ fission cross-section, b) $k_{eff,}$ c) flux ratio of the inner and outer core region F.*

$$C_i = \frac{N_i^{in}}{N_i^{in} + N_i^{out}} \tag{14}$$

$$F = \frac{\Phi^{in}}{\Phi^{out}} \tag{15}$$

then (13a) can be written as:

$$\overline{N}_i = \left(N_i^{in} + N_i^{out}\right)\frac{CFV^{in} + CV^{in} - V^{in}}{FV_{in} + V_{out}} \tag{16}$$

The full core calculations also determined the $k_{eff}$ for the different compositions. This gives the opportunity to fit the $k_{eff}$ as a function of the composition as well (see Fig. 2.), which has the great advantage that in the fuel cycle model the fissile loading for the required excess reactivity can be determined.

The above described burnup model was used to build the simplified fuel cycle model shown in Fig 3. The model was developed in MATLAB and contained the GFR burnup model and followed the material flows between reactors and storages. The fuel cycle contains two type of reactors: the GFR and conventional light-water reactors (LWR). LWRs operate in once-through cycle: they are fed by 3.6% enriched U and the fuel is discharged after 33 MWday/kgU burnup. The spent fuel was considered with the composition in Table 1. No recycling into LWRs was considered but the spent fuel was moved to partitioning and the Pu and MA fractions were recycled into the GFR. GFR was fed with depleted U (as fertile material) produced during the enrichment of U for LWR fuel, and Pu and MA from the reprocessed LWR fuel. The reactor operates in three-batch cycle: one third of the core is discharged and replaced with fresh fuel in every cycle (481 EFPD) and each fuel element spends 3 cycles in the core. The Pu content of the charged fuel was calculated with iteration in order to set the excess reactivity of the beginning of cycle (BOC). The multiplication factor at the end of the cycle $k_{eff}^{EOC}$ was expected to be 1.005 therefore multiplication factor for the beginning of the next cycle $k_{eff}^{BOC'}$ was set based on the multiplication factor at the beginning of the actual cycle the following way:

$$k_{eff}^{BOC'} = k_{eff}^{BOC} - k_{eff}^{EOC} + 1.005 \tag{17}$$

MA were considered to be loaded homogenously into the core and different options were investigated concerning the MA content of the charged fuel. Fuel discharged from GFR was partitioned after 5 years of cooling and sent to the corresponding storage (U, Pu or MA). The U, Pu and MA need of the GFR was taken from the storage and depleted U or Pu and MA from LWR spent fuel was used only when the amount in the storage was not enough. The developed tool is capable to follow the above

fuel cycles for a long term (hundreds of cycles) in a reasonable CPU time (few minutes) which is important to reach the equilibrium in the system.

Table 1. Considered LWR spent fuel composition after 33 MWday/kgU and 5 years cooling time

| MA isotope | Ratio | Pu isotope | Ratio |
|---|---|---|---|
| Np-237 | 0.495 | Pu-238 | 0.027 |
| Am-241 | 0.316 | Pu-239 | 0.564 |
| Am-242m | 0.00103 | Pu-240 | 0.240 |
| Am-243 | 0.148 | Pu-241 | 0.099 |
| Cm-242 | $4.13 \cdot 10^{-9}$ | Pu-242 | 0.0697 |
| Cm-243 | 0.000436 | | |
| Cm-244 | 0.0369 | | |
| Cm-245 | 0.00262 | Component | Ratio |
| Cm-246 | 0.000477 | Total Pu | 0.00905 |
| Cm-247 | 0.0000101 | Total MA | 0.00067 |

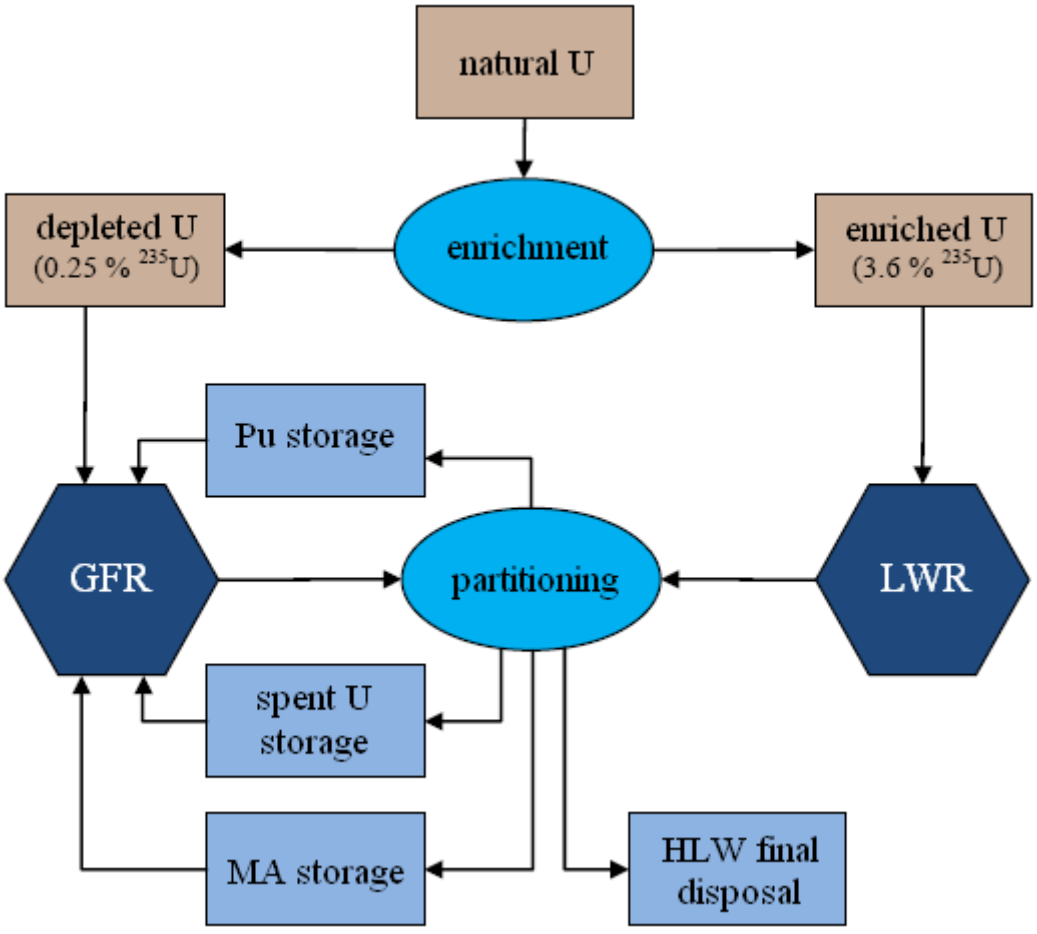



*FIG. 3. The considered fuel cycle model. Pu and MA from LWR is fed to GFR only when the required amount is not present in the corresponding storage.*

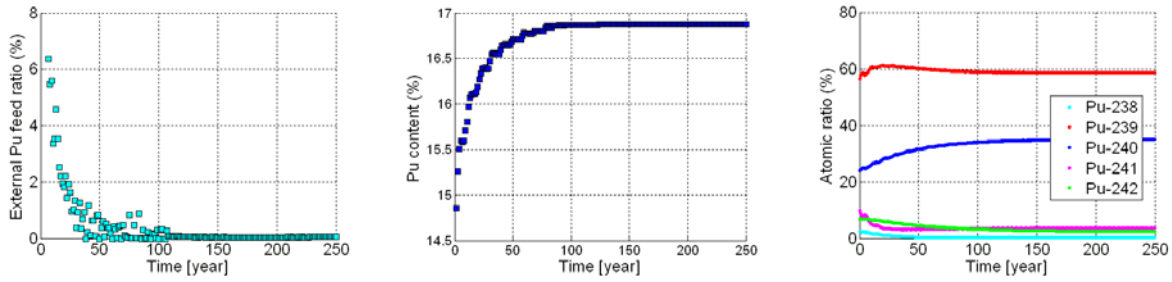
In order to compare the investigated cases the fuel utilization efficiency was defined in the following way. The total thermal power generated in the system is the thermal power of the GFRs plus the thermal power of the LWRs producing the required Pu amount for the GFR load. The fuel utilization efficiency is the ratio of the total thermal power to the theoretical energy content of the natural U

(1000 MWday/kg) consumed to produce enriched U fuel for the LWR. The fuel utilization efficiency of an LWR with the above parameters operating in once-though cycle is 0.44 %.

## 3. Results

All simulations were continued till the equilibrium was reached in the system. Although the time needed to reach the equilibrium was too long to be realistic in a fuel cycle simulation (several hundreds of years) but the investigation of the equilibrium state of the system is important for its evaluation.

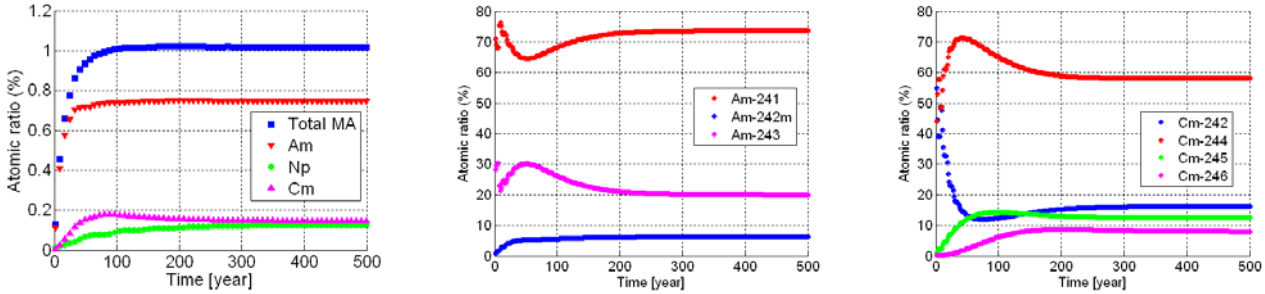
The first investigated option assumed only Pu recycling into the GFR in order to provide a reference case for further investigations. The simulation shows, that Pu feed from LWRs quickly decreases and in the equilibrium no external Pu feed is needed, in fact a small amount of Pu is even produced, which results in a breeding gain of 0.022. This shows that the GFR is a true self-breeder. The initial 14.77% Pu content of the core increases to 16.84% due to the pile up of $^{240}$Pu and the decrease of the fissile $^{241}$Pu which is typical to fast reactors (see Fig 4.).




a) b) c)

*FIG. 4. Results for the case without MA recycling: a) external feed of Pu, b) Pu content of the core, c) isotopic composition of Pu. Increase of the Pu concentration is due to the accumulation of $^{240}$Pu*

The next case was the recycling of Pu and MA into GFR but without adding MA from LWRs. An important question in this case is whether an equilibrium can be reached and at what MA concentration. The results show that the equilibrium is reached slightly above 1% of MA concentration in the core. Isotopic compositions also reach an equilibrium which means that all MA is consumed by fission and no Cm accumulation occurs due to the MA recycling (see Fig 5.). This proves that the GFR can be applied as a MA burner. The MA content of the core even slightly improves the breeding capabilities of the reactor. This is explained by the fertile nature of some MA isotopes, while the equilibrium Pu content is also lower in the core due to the fissile MA isotopes fed into the core. The amount of $^{238}$Pu is increased in the equilibrium Pu composition, which is produced from $^{237}$Np.

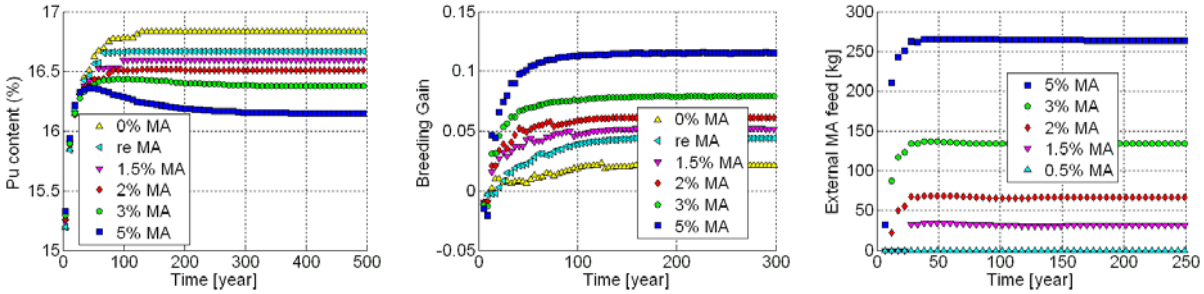



a) b) c)

*FIG. 5. Results for the case with MA self-recycling: a) MA composition, b) Am composition, c) Cm composition. All isotopes reaches equilibrium which means that GFR is a net MA burner.*

The MA equilibrium concentration of 1 % suggests that higher MA concentration in the core also allows external feed of MA and the GFR turns into a net MA burner. In order to verify this statement

simulations were performed with the assumption of a fixed ratio of MA in the charged fuel varying from 0.5 % to 5 %. As it was expected with 0.5 % MA ratio the need for external MA feed diminishes as the equilibrium is approached while with 1.5 % or more MA a significant external MA load stabilizes (see Fig. 5.). The external feed needed for the 5 % MA load almost completely consumes the MA produced by more than 10 LWRs of the same thermal power (see Table 2.). This means that a symbiotic nuclear energy system can be set up where the LWRs produce the Pu required for the start-up of the GFR by the utilization of enriched U, while the GFRs burn the MA produced by the LWRs. In this way the MA output of the system reduces to the losses during the partitioning process. It is also worth to note that the increased MA feed also improves the fuel utilization. Although the higher MA content provides better performance from the fuel cycle point of view, one also has to look into the safety parameters of the core, which may limit the MA content at a lower level.




a) b) c)

*FIG. 6. Comparison of results for fuel cycle simulations with 0.5, 1.5, 2, 3 and 5 % MA feed: a) Pu content, b) Breeding gain, c) External MA feed.*

Table 2. Fuel utilization and MA burning parameters of different scenarios in equilibrium

| Case | Equilibrium Pu content (%) | Equilibrium Pu production [kg] | Equilibrium breeding gain | External MA feed [kg] | LWR MA power ratio (%) |
|---|---|---|---|---|---|
| Pu recycling | 16.84 | 29.04 | 0.022 | - | - |
| MA self-recycling | 16.67 | 63.38 | 0.044 | - | - |
| 0.5 % MA feed | 16.82 | 45.06 | 0.033 | 0.00 | 0.00 |
| 1.5 % MA feed | 16.59 | 76.72 | 0.052 | 31.04 | 56.97 |
| 2 % MA feed | 16.51 | 91.98 | 0.061 | 66.09 | 73.82 |
| 3 % MA feed | 16.38 | 119.94 | 0.079 | 134.03 | 85.12 |
| 5 % MA feed | 16.15 | 171.69 | 0.115 | 262.57 | 91.81 |

## 4. Conclusions

A quick and flexible burnup model was successfully developed based on the fitting of cross-sections and other parameters as a function of the isotopic composition. The model was integrated into a nuclear fuel cycle scheme containing conventional LWRs and GFRs. The fuel utilization and MA burning capabilities of such a system was investigated.

It was found that the GFR achieves self-breeding and therefore only the initial Pu load needs to be fed from LWR spent fuel. In case of recycling of MA into the GFR, the MA content in the GFR core reaches equilibrium at about 1 %. If more MA is fed into the GFR, it turns into a net MA burner and it

is able to consume MA from an LWR park. The higher MA content of the core even results in higher breeding gain.

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# Why R&D for Generation IV reactors should be subsidised ? A strictly economic point of view based real option theory

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**Abstract.** Generation IV fast reactors make better use of natural uranium than current reactors. Since there is a significant risk that the uranium market could come under pressure before the end of the 21st century, fast reactors are in a position to play a vital 'sustainable' role by making the resource usable on longer time scales. However, given their likely higher investment costs compared to previous generations, their competitiveness is not guaranteed. This paper aims at assessing the R&D budget available to develop this technology, considering that it would be deployed to counterbalance an important price rise of uranium. The deployment decision thus depends on its relative competitiveness, which is determined by its overcost and uranium price. A model based on the real options theory shows that the budget willingly allocated to R&D is positive even in cases of overcost and uranium price forecasts unfavourable to fast reactors.

## 1. Introduction

Today, Light Water Reactors (LWR) occupy a predominant place in the nuclear fleet worldwide, generating 17% of electricity in the world, 35% in Europe and almost 80% in France (see [1]). Nevertheless, their main weak point resides in their use of uranium, since only 0.5% to 1% of the natural uranium is actually used to generate energy by fission. Given the identified world resources, such a performance guarantees about 80 years' operation for the reactors currently in service[1]. The technological progress in mineral exploration, together with unconventional resources, could expand the available uranium, but on the other hand a potential growth of the world's nuclear fleet could have an important impact on the demand for uranium. Despite the accident of Fukushima, the possibility exists that nuclear energy should still expand on a long-term scale. There is thus a significant risk that the uranium market could come under pressure before the end of the 21st century.

Generation IV Fast Reactors would allow to avoid such pressure thanks to the benefits of self-breeding or even of breeding. Several thousand years of fission energy can be guaranteed by using a greater fraction of natural uranium. In the case of France, given the age of the fleet and state of technology, the need for a potential industrial development of Fast Reactors appears in 2040. Having this option would allow to handle any upcoming pressure on the uranium market. However, the competitiveness of this innovative technology is uncertain, due to the additional investment costs involved. The relevance of such an option is therefore to be confirmed in the future. Today, only the sodium-cooled fast reactor (SFR) technology seems capable of meeting this requirement by 2040 owing to its high level of maturity.

The question we seek to answer in this study is whether it is worth pursuing R&D on SFRs beyond 2012. In order to shed light on this issue, we developed a simplified model based on the real options theory to try and compare the consequences of the two possible outcomes. As literature review shows,

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[1] See *Uranium 2009*, AI EA and AEN

real options theory has already been applied to such fields as energy, nuclear energy and R&D investments ([2],[3],[4],[5],[6],[7],[8]).

## 2. Model

We use a model based on the real options theory to estimate the R&D economic value for Generation IV nuclear reactors. Uncertainty focuses on the overcost of SFRs compared with LWRs and on the future price of natural uranium with the deployment of nuclear energy worldwide.

### *2.1. Decision to conduct research in 2012*

The decision to be made in 2012 by the French authorities is assumed to be binary: "halt R&D on Generation IV reactors" or "finance R&D in this field".
The aim of the study is to compare the two possible choices in 2012, through the total discounted costs at this date associated with nuclear electricity generation (frontend cycle, electricity production, backend cycle) for both choices over the 2012 - 2150 period, in order to minimize it.

### *2.2. Opportunity of SFRs industrial development in 2040*

The choice to start a new technology implies that the necessary preliminary stages have been successfully completed : research and prototype testing, which would make the start of SFRs industrial development possible around 2040.
Given these conditions, and in the case where the R&D option is chosen in 2012, the decision-maker will have to make another decision in 2040: "give the go-ahead to start building fast reactors" or "veto its industrial-scale construction". In the second case, France would then continue to operate LWRs (since it is assumed there are only two technologies in competition, LWRs and SFRs).
For the study, we consider to be within a French context with no technology exchange outside its borders, which means that, if no R&D is conducted in 2012, there will be no Generation IV reactors in 2040. Nevertheless the model could allow us to consider a larger area including other nuclear fleets in other countries in the case of an association for conjoint research : the electricity production would be higher, which would affect the results proportionnally. The window of opportunity is fixed (dates 2012 and 2040) as in Henry's value option models [9]. This model includes two periods (model with simple real options) : the first period ranges from 2012 to 2040 while the second ranges from 2040 to 2150.

### *2.3. Conducting research : a flexible decision*

According to Henry [9], three conditions are needed to use real options theory : having incresing information "*We will know better about tomorrow than we know now about after tomorrow*", being "*in an uncertain universe*" and being faced with "*choices of variable flexibility*".
As previously mentioned, the uncertainty on the price of uranium and the overcost associated with fast reactors are the determining factors to their competitiveness. It is assumed that the information on the competitiveness will be revealed in 2040, making it possible to choose (or not) the fast reactor technology with full knowledge of the facts. This is why the decision to conduct or cancel R&D in 2012 is considered flexible.
The research question here is to know whether the cost of flexibility is justified. This cost corresponds to the R&D subsidies for SFRs.

Before calculating the costs associated with alternative decisions, the relative competitiveness of the LWR and SFR technologies has to be analyzed.

### *2.4. LWR and SFR relative competitiveness*

In this paragraph we seek to establish a simple relationship between LWR and SFR future costs in order to assess their relative competitiveness and determine the zone of equivalence where their costs are equal.

The following assumptions were used to define this zone of equivalence (Figure 1):

(1) The annual electricity production Q is stable over the entire period of study. The availability of LWRs and SFRs is supposed to be the same and have no influence on annual production.

(2) For uranium at €100/ kg, fuel cost represents 5% of the total generation cost for LWRs. We suppose that, even if the price of uranium grows, the part of fuel in the total LWR cost is fixed to 5% .

The total cost of the LWR fleet (frontend cycle, backend cycle and electricity production) needed to generate the annual production, with the uranium price at €100/kg is written "Cost $LWR_{100}$". If the price of uranium increases by a factor p, then:

$$\text{Cost } LWR_p = \text{Cost } LWR_{100} \times (1+0.05p). \qquad \textit{Eq. (1)}$$

(3) The generation cost for SFRs does not depend on the uranium price, nor does it depend on the price of plutonium which is assumed to be free of charge in France (since the plutonium is already generated by the reprocessing of LWR waste - nonetheless in other contexts, it would be relevant to take a higher cost into account.) SFR overcost is mainly due the higher investment cost compared with a LWR. We nonetheless take into account the overcost as total generation overcost (investment, production, frontend, backend), that could also inclue plutonium costs in cases where it cannot be considered free.

Given that s represents the overcost of an SFR in relation to an LWR where uranium is worth €100/kg, then:

$$\text{Cost SFR} = \text{Cost } LWR_{100} \times (1+s). \qquad \textit{Eq. (2)}$$

Generation costs are the same for LWRs and SFRs when:

$$\text{Cost } LWR_{100} \times (1+s) = \text{Cost } LWR_{100} \times (1+0.05\,p). \qquad \textit{Eq. (3)}$$

That is to say when:

$$\mathbf{s = 0.05\,p} \qquad \textit{Eq. (4)}$$

The zone of equivalence is linear: a straight line that cuts the (p x s) graph in two aeras : SFR competitive area and LWR competitive area.

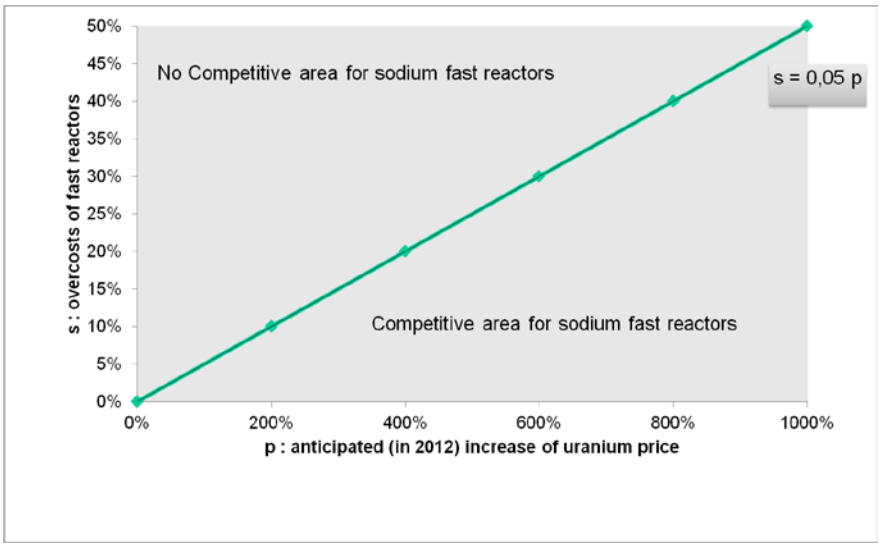


*FIG. 1. SFR and LWR competitive areas from 2040 and line of equivalence for the two technologies from an economic viewpoint*

## *2.5. Uncertainty*

As previously mentioned, there is uncertainty both on the price of uranium from 2040 and on the overcost of SFRs.

### *2.5.1. Price of uranium*

The uranium price is estimated at €100/ kg for the first period. It is then assumed to rise in 2040 by p% and to remain stable throughout the second period. The rise, p, is expressed as a percentage of the price prior to 2040 and is assumed to follow a Gaussian distribution with a mean $p_m$ and a standard deviation $\sigma_p$.

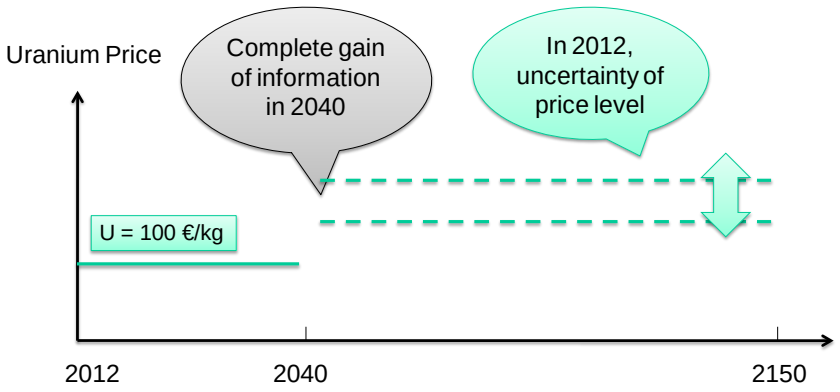


*FIG. 2. Uranium price rise in 2040*

### *2.5.2. SFR overcost*

Over the second period, it is assumed that the SFR overcost, compared with a LWR in the first period, follows a Gaussian distribution with a mean $s_m$ and a standard deviation $\sigma_s$.

### *2.5.3. Implication of introducing uncertainty in the model*

As a consequence of introducing uncertainty in the form of Gaussian distributions for the uranium price and SFR overcost, the separation between SFR and LWR competitive areas is not binary anymore. The line of equivalence still represents the zone where SFR and LWR are equally competitive; but there is a non-zero probability that SFR could be competitive in the LWR competitive area, which means that SFR integration could occur in the nuclear fleet, and vice versa.

## *2.6. Decision tree*

The decision tree (see Figure 3) explains the choices public authorities have to face in 2012 and in 2040. In 2012 they will have to choose between continuing research on future reactors or halting this research taking into account the impact of their choice on future costs. Continuing R&D will open a new window of opportunity in 2040 which involves choosing to build (or not) the innovative technology. The costs are calculated using a decision tree according to a *backward induction* method where the costs are minimised at every step (node) of the decision process, considering first the [2040, 2150] period and then the [2012, 2040] one, on every branch of the tree (see [10] and [11]).

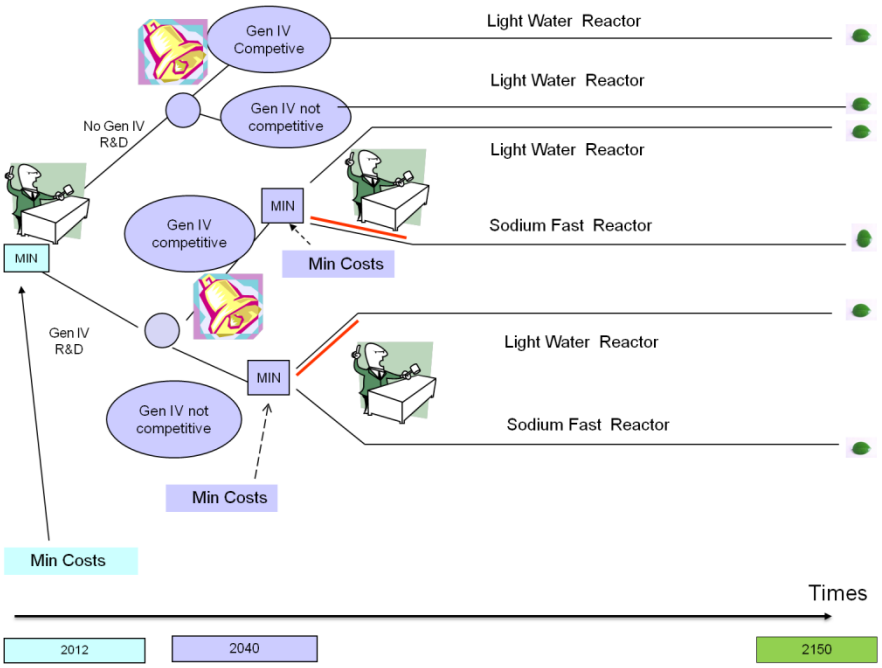


*FIG. 3. Decision tree*

### *2.7. Case 1 : Discounted cost of the decision to halt R&D*

If the decision made is to give up R&D on SFRs in 2012, France will dispose of the LWR technology only. The total discounted cost Z for this case consists in the discounted generation cost for annual production Q with LWR costs taking into account uncertainty on uranium price through the density probability function.

### *2.8. Case 2 : Discounted cost of the decision to conduct R&D*

In the case where R&D is conducted, electricity will be generated by:

- A LWR-only fleet from 2012 to 2040,
- A LWR-only fleet after 2040 if SFRs are not competitive, a LWR and SFR fleet if SFRs prove competitive.

The discounted cost of R&D over the first period is taken into account as A.
B represents the production cost during this period (only for the LWR technology).
For the second period, the discounted generation cost C is calculated based on the fact the electricity will be generated by LWRs if the SFR are not competitive and generated by LWRs and SFRs if they are competitive: since you have got the information about the project's context, you cannot lose money. The density probability function models uncertainty and proportional factors models progressive integration of SFRs in the fleet

Finally, the cost of the decision to conduct R&D in 2012 amounts to the sum of the three expressions, A, B and C:

$$\overline{COST}(SFR\ R\&D) = A + B + C \qquad \textit{Eq. (5)}$$

### *2.9. Comparing the option value with the R&D amount*

The two discounted costs need to be compared and the R&D amount needs to be defined for which both decisions "conduct R&D" or "halt R&D" are considered to be equivalent.

The difference between the cost to halt R&D and the cost to conduct R&D (which is positive, due to the flexibility associated with the decision to conduct R&D : you can make the economic optimal choice, so you do not lose money) represents the maximum R&D budget that can be allocated to Generation IV fast reactors, i.e.:

$$Z - (B+C) \qquad \text{Eq. (6)}$$

Strictly speaking, the value of the electricity produced by the prototype should be integrated into the R&D costs. We have not taken this aspect into account in order to simplify the model, since it would mean making assumption about electricity market price in the future. Nevertheless this simplification penalises slightly the decision to conduct R&D.

## 3. Applications and simulations

This section describes the numerical applications and simulations performed using the model.

Firstly, the assumptions defining all the parameters of the model are detailed, i.e. : i) nuclear electricity production Q which is assumed to be stable, ii) annual cost of the LWR fleet (Cost LWR $fleet_{100}$), iii) discount rate for the first and second period, iv) proportion of SFRs in the fleet and its progress over time, v) means and standard deviations of probability density functions, vi) overcost of SFRs, and vii) uranium price rise.

The numerical applications provide an assessment of the costs for each decision, as well as an estimate of the limit not to be exceeded for the R&D budget allocated to Generation IV reactors. The simulations are used to calculate these same costs by varying the parameters of the model (mean of the overcost and of the uranium price rise, uncertainty, discount rate, etc.) so as to visualise different decision-making contexts.

### *3.1. Assumptions of the model parameters*

#### *3.1.1. Nuclear electricity production and discounting*

Our study was based on the total annual costs for an entire fleet producing a quantity Q = 430 TWh of electricity. The total annual cost of the LWR fleet is: Cost LWR $fleet_{100}$ = €20 G.
The discount rate applied is the public rate: $a_1$ = 4% before 2040 and $a_2$ = 2% after 2040.

#### *3.1.2. SFR integration*

The progressive integration of SFRs into the fleet from 2040 is taken into account on the basis of past LWR constructions, their life spans and the available plutonium resources (for SFRs).

#### *3.1.3. Reference assumptions for the probability density functions*

The uranium price rise p in 2040 is given as a percentage of the price of the first period and is assumed to follow a Gaussian distribution with a mean $p_m$ = 240% and a standard deviation $\sigma_p$ of 100%. This rise represents a punctual rise in 2040 and the new price is then stable over the [2040, 2150] period (as well as the characteristics of the Gaussian distribution that describes it : mean and standard deviation). SFR overcost s is also stable over the period [2040, 2150] and follows a Gaussian distribution with a mean $s_m$ = 12% and standard deviation $\sigma_s$ equival to 1/30, i.e. 3.33%.

This combination of mean values for the distributions s and p was chosen as follows:

- The mean of the s distribution is based on an expert analysis in which the SFR overcost is estimated in relation to the LWRs in service in the first period. Assuming that uranium costs €100/ kg and in light of this overcost, the assessment of the overall overcost (investment, operation, cycle) amounts to 12%.

- Once $s_m$ has been calculated, $p_m$ (mean of the p distribution) is chosen so that the ($p_m$, $s_m$) combination is located on the line of equivalence for both technologies $s_m = 0.05\ p_m$, which leads to a $p_m$ of 240%.

The standard deviations were chosen to include an appreciable level of uncertainty while limiting scatter around the mean.

### *3.2. Numerical applications*

The numerical applications were performed with the Maxima software.

$$\overline{COST}(LWR) = Z = 668{,}4\ \text{G€}$$

$$\overline{COÛT}(SFR\ R\&D\ decision) = B + C = 664{,}9\ \text{G€}$$

Considering the model's simplifying assumptions, with a mean uranium price rise predicated at 240% and an mean overcost of 12% for SFRs compared with LWRs (with moderate uncertainty on these two random variables), the public authorities will be able to spend up to €3.5 G for research on future reactors.

Varying the model's parameters in a set of simulations allows us to observe the variation in the amount that the public authorities are willing to spend on R&D.

### *3.3. Simulations*

#### *3.3.1. Different combinations (mean uranium price rise $p_m$ and mean SFR overcost $s_m$)*

As mentioned in 2.5.3, uncertainty introduces non-zero probability of having competitive SFRs in the LWR competitive area and vice versa.

Simulations were performed with ($p_m$, $s_m$) combinations that differed from the reference combination but with the same standard deviations ($\sigma_p$,$\sigma_s$). These simulations allow us to observe the maximum amount (A) that would be allocated to R&D according to different positions on the graph ($p_m$ x $s_m$):

- on the LWR-SFR line of equivalence,

- in the LWR competitive area,

- in the SFR competitive area.

Figure 4 shows the results of these simulations: the maximum amount (A) (in €G) is indicated for each combination.

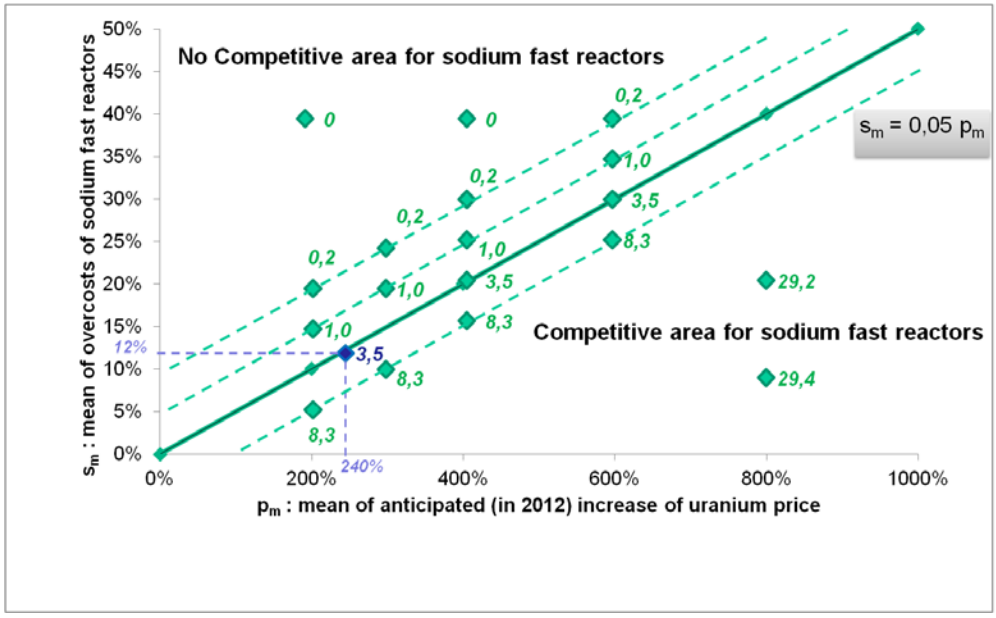


*FIG. 4. Simulation results: (A) given in €G*

The results show that the amount (A) allocated to R&D becomes non-zero on the line of equivalence. As expected, this amount nevertheless grows increasingly smaller when moving away from the line of equivalence in the SFR non-competitive area and increasingly higher when going in the competitive area of SFR.
One striking results is that practically the same amount (A) allocated to R&D is found for the ($p_m$, $s_m$) combinations located on the line of equivalence. On each line parallel to this equivalence line the amount remains the same for all combinations belonging to this line. At the same level of uncertainty in absolute, the amount allocated to R&D is determined by the relationship between $p_m$ and $s_m$.

## 4. Discussion and conclusion

### *4.1. Results*

The option value model revealed the following results:

Faced with uncertainty on the future price of uranium and the SFR overcost, the option value associated with the decision to conduct research is non-zero, even in the area where SFRs are in principle (and in probability) not competitive. The option value is generated by : uncertainty, increasing information, flexibility in the choice (having different technologies available makes optimal choice possible and thus guarantuees not to lose money).

This is also equal to the maximum budget that the authorities are willing to invest in R&D. It is estimated at €3.5 G based on the reference assumptions for the model which assesses the mean overcost of SFRs at 12% compared with LWRs, and taking into account the case where the probability of SFR reactor being competitive is equal to the probability of LWR reactor being competitive (50%) (which corresponds to a mean uranium price increase of 240%).

With all other assumptions being equal, if the mean overcost of SFRs is increased by a 5% increment i.e. 17% instead of 12% (meaning they are not competitive), the maximum budget allocated to R&D is reduced to €1 G. If the mean overcost of SFRs is lowered by a 5% increment (meaning they are considered competitive in relation to LWRs), this maximum budget for R&D amounts to €8.3 G.
In the same way, all else being equal, if the mean uranium price increase is a 100% increment higher (SFRs are competitive), the maximum budget for R&D amounts to €8.3 G. If the mean uranium price increase is a 100% increment lower (SFRs are not competitive), this maximum budget for R&D amounts to €1 G.

Another result is that if we increase the quantity of electricity production Q, the budget allowed for R&D will increase proportionnally. The electricity production Q has a direct impact on the cost of the nuclear fleet since Cost $LWR_{100}$ represents a total production cost and is determined so as to follow the same variations as Q.

No matter how informative, it nevertheless remains that these first results have been produced by a simplified economic model that will need to be further developed in order to continue our research. The main limits of the model are described below.

### *4.2. Modelling limits and conclusions*

It is assumed that R&D will necessarily lead to the development of the SFR technology and that there will be no problem with public acceptance of this technology.
The first assumption can be loosened by weighing the amount dedicated to R&D by a probability function reflecting the success of R&D. The second assumption being particularly debatable in the wake of the Fukushima disaster, additional uncertainty can be introduced into the model by including a random variable on the public acceptance of the technology. But considering their advantages in terms of waste toxicity, will SFRs have a better chance of being accepted? The cost of safety will rise significantly. This will also have an impact on both LWRs and SFRs, which is why it has no impact on our results.
Moreover, the valuation of the electricity produced by the prototype should be integrated into the R&D costs.
It is also assumed that the part of uranium in the LWR total cost will not change (5%).
Resctricting our study to France is, of course, only an approximation of the reality since technology exchanges between countries should be taken into account. The case of a free rider who profits from the effects of R&D without contributing to its funding should be taken into consideration. However, it is very unlikely that France behave as a free rider in light of its behaviour in the past. Otherwise, France could receive royalties from the sale of its innovation overseas, which has not been integrated into the model.
Lastly, this study does not take into account the feedback effect of Generation IV deployment on the uranium market and thus its price. Massive SFR integration in the fleet could nonetheless lessen the pressure on uranium market.

This model, though simple, indicates that there is an economic interest in pursuing R&D when the decision maker is confronted to uncertainty and that more information is expected to come in the future. In order to assess more precisely the mean $p_m$ of uranium price rise p in 2040, it would be interesting to connect our model with an additionary model on uranium price that would allow us to give a more precise value to the option of pursuing R&D (instead of having a range of values as in Figure 4).

This work was extended and sophisticated in order to take into account SFR integration in the fleet in uranium price evaluation as mentioned in the last limit of the model, which is the object of a paper submimtted in an economic journal. These extended works show that subsidising research and development for Generation IV reactors would be beneficial for the competitiveness of both Generation IV and Generation III reactors, thus for the whole nuclear sector.

## ACKNOWLEDGEMENTS

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# 3 Investments Scenarios for Fast Reactors in Europe
## *2012, November 30 - Preliminary Version*

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**Abstract.** The article aims at widening the scope of MIT report « The future of nuclear power after Fukushima » (2012) to the European electric supply by studying the particular conditions of industrial development of Fast Reactors. We specially focus on France, Germany, United Kingdom, Spain and Italy, from now up to 2040, for the technology is supposed to be available by then. These conditions can be either favorable or not to FRs, according to 3 main dynamically quantified drivers: "technical change", i.e. relative evolutions of efficiency and costs of available technologies (gas, coal, wind…); "policy", i.e. incentive framework given by European energy policies (nuclear, climate...); "economic", i.e. structure of electricity markets (level of centralization...). A total of 24 scenarios are developed using an imaginative approach, i.e. assuming different possibilities for the future change according the latter 3 main drivers. 3 of them proving to be favorable to the FRs are then discussed in view of the quantitative drivers mentioned above and of an additional driver about public acceptance.

## 1. Introduction

In a context of post-Fukushima nuclear development as well as climate protection, this article addresses the issue of investment in future nuclear technologies in Europe, and in particular Fast Reactors. MIT publication "The future of nuclear power after Fukushima" [1] reports the expected growth of nuclear power in the world fleet (1% per year through 2035 in OECD countries and 6% per year in non-OECD countries through 2035) will not be significantly reduced, expect for Germany, Japan and Switzerland, which would mean increased consumption of natural uranium. This nuclear growth and carbon emissions reduction measures such as in the European Climate Action and Renewable Energy Package (see [2]), could make Generation IV Fast Reactors a viable choice for further electricity generation capacities. This technology using fast neutrons instead of thermal neutrons effectively allows better use of natural uranium and could insure several thousands of years of nuclear generation whereas identified resources in uranium only allow about one century of generation with Generation II and Generation III thermal neutrons Reactors [3].
The evolution of the generation mix and potential integration of Generation IV Fast Reactors in the mix from 2040 depends of course on many factors such as climate and energy policies, but in the end is determined by the actual decisions of power generation companies, who will invest in new capacities in order to replace their ageing capacities and satisfy a growing demand. This is why we have chosen to focus on investors, i.e. power generation companies, and analyze their behavior regarding investments in generation capacities.

There are thus two research questions we seek to answer in this article: What are the drivers for investors' decisions on the European electricity market, regarding investments in power generation

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capacities? How do they affect the evolution of the European generation mix, and the development of future nuclear fast reactors in the mix?

We focus on France, Germany, United Kingdom, Spain and Italy, for they represent 65% of EU27 power generation [4]. The time horizon is fixed to roughly 2040, for Fast Reactor technology is assumed to be ready for industrial development by then, and most reference scenarios' time limit are situated between 2030 and 2050 ([2], [5]).

To identify the drivers to investors' decisions, we proceed by analyzing:

- Historical aspects of the European generation mix constitution and of the European market liberalization (Section 2);
- Investors' profiles through a few key characteristics such as the shareholding structure, market capitalization, annual revenue, generation mix (Section 3);
- Technologies investments conditions such as costs and incentives (Section 4).
- In the end, 3 key drivers are identified and scenarios for future generation mix are built upon a couple of low/high hypothesis for each driver (Section 5).

## 2. Generation Mix Constitution and Market Liberalization in Europe

It is necessary as a first step of our analysis to look back on historical aspects, and mainly two of them: the constitution of the European generation mix from the fifties to now, in order to understand past investment choices, and the European market liberalization that started in the nineties, in order to understand which kind of context today's investors are confronted to.

This historical analysis shows that European countries have massively privileged local resource, such as coal in Germany and Spain, gas in the UK (when resources were found in the North Sea), and hydropower in France and Italy, the latter also using its resources in gas and oil. When local resources became unsufficient, they chose to develop a locally well-mastered technology such as nuclear in France and to a lesser extent in the four other countries, or to import resources such as gas, hich was mostly the case for Spain. This tendency was reinforced after the two oil shocks in the seventies, leading European power companies to insure security of supply at high costs [4][6]. The driver to these decisions was the **state policy**, whose purpose was to ensure energy independency.

After the counter shocks of the eighties, a market reform occurred in Europe in the nineties, in order to create a unique European competitive market from all the national markets in place, often integrated monopolistic markets. The reform was unequally applied in the different countries : very much in the UK, which was a pioneer of liberalization, and in Italy, where the government limits the market shares of electricity market actors ; very little in France, where the the natural monopoly model – see Baumol 1997 [7] for theoritical aspects- was considered a success within the rule of the Ramsey-Boiteux pricing, as well as in Germany and Spain, where the authorities tend to protect their electricity companies from the effects of liberalization. This disparity of attitudes leads to various market structures and concentrations that will constitute very different environments for investors [4]. **Market structure** is thus another driver for investors' decisions.

## 3. Investors' Profiles

The second step of our analysis consists in defining who the investors are and how their characteristics will influence their own investment decisions.

Investors' profiles can be analyzed through a few key characteristics that are:

- The shareholding structure, which will give an indication on the kind of value the shareholders expect from the company, and thus on the investment strategy of the company : private shareholders such as institutional shareholders or public float will only expect economic value in high priced shares or dividends ; whereas state shareholders such as a state, ministry, local collectivity, will expect economic value at a broader scale i.e. supplying consumers and national industry at acceptable costs, social value in supplying all citizens, geopolitical value in energy independence, ethical and environmental values by insuring the company respects government policies regarding these matters (environment protection, labour law…) ; the weight of the different types of shareholders will influence the company's investment strategy ;
- The market capitalization and annual revenue, that indicate the size of the company from a financial point of view and the size of the investments the company can support,
- The total annual production, that indicate the size of the company from an industrial point of view;
- The generation mix, that shows the expertise fields of the company;
- The market shares on the markets where the company is active, which show the international scope of the company.

As a result from this analysis, most of the power generation companies today are former historical operators who used to be in a dominant market position [4], [8]. Their shareholders are state actors such as the government, a ministry, or local communities, institutional investors such as banks and insurance companies, and private shareholders (public float), the weight of each type of shareholders depending on the national position towards market reform and the particular history of the company [8]. This means that the state policy is a direct driver for investment decisions in numerous cases in Europe.

Their annual revenue and market capitalization represent several dozen billion € and annual production around a hundred TWh [8]. Their dominant technologies are mostly coal and gas, (and nuclear for EDF). Most of them have crossed the border of their initial market and started being active on neighboring markets: for instance EDF is present in UK and Italy, EOn in UK, Italy and Spain. We can also observe concentrating movements between these companies: as few examples among many others, Italian operator ENEL owns Spanish one Endesa, French operator EDF owns British Energy and Italian company Edison, and Spanish operator Iberdrola owns Scottish Power.

Yet another type of profile seems to be emerging with the market reform, the one of small power companies. Such companies are generally young, dating from the nineties or years 2000 such as wind operator Theolia [8]. Their shareholding structure have no state actors component; their revenue is around a few million € and annual production less than 1 TWh. They mostly specialize in one technology since their size does not allow them to diversify, mostly recent technologies such as renewable or CCGT, and can be local or international operators, representing minor market shares in any case.

As we said above, national positions regarding the market reform differ from one country to another, which affects power generation companies' evolution. France, Germany and Spain tend to protect their historical operators on their inside markets and promote their international development thanks to the reform, as though UK and Italy are really promoting competition on their own market, with Italy limiting market shares for the different actors on the Italian market for instance. The evolution of investors' profiles towards multinational concentrated companies or towards small power operators will depend on global market structure evolution, in link with the market reform policies lead in EU countries.

## 4. Investment Conditions in Electricity Generation Technologies

After analyzing the history of the market and the investors themselves, the third step will look into the technologies and the investment conditions for each technology. All major power generation technologies are considered: coal, gas, nuclear, hydro, wind, solar. For each technology, investment conditions are examined: building and generation costs, and load factors, that will directly impact the expected profits, but also all the parameters that will make the technology more or less easy to acquire for the investor, which are building period, average size of the plant for this technology, technology complexity, variety of financing methods, positive and negative incentives coming from policies (such as carbon costs, subsidies for renewable, insurances from the government, or radical decisions such as nuclear phase-out decisions in Germany and Italy).

The review of these investment conditions shows strong differences from a technology to another. We will thus summarize our approach by presenting an overview of the generic risks investors are confronted to when investing in an electricity generation capacity, and mention the most striking ones for each technology.

For any investment in electricity capacity, there is a list of generic risks investors have to face [9]. There are globally three types of risk :

- The risks before operation, on conception and building phases;
- The risks during operation : operating default, fuel supply, sale on electricity market (this risk is all the more important since the liberlization), plant's end of life management;
- The risks concerning any phase : political risk, regulatory risk, financial risk on loans, catastrophy or cases of force majeure

Each risk will be more or less strong according the technology's characteristics, as it is summarized in table 1 [9][10] [11].

Table 1. Major investment risks for electricity generation technologies

| Technology | Major risks for investor |
|---|---|
| Nuclear | political risk due to nuclear opposition<br>conception (complex technology)<br>building cost (2/3 of generation costs),<br>risk on sale<br>perceived risk on end-of-life costs |
| Coal | political risk due to carbon emissions<br>building cost<br>risk on fuel supply |
| Gas | political risk due to carbon emissions<br>risk on fuel supply |
| Hydro | political risk due to environmental opposition |
| Recent renewable :<br>wind, solar | low profitability due to low load factors<br>(offset by subsidies for now) |

In order to understand investment choices, it is relevant to confront investors' profiles and technologies' investment conditions: for instance, capitalistic investments such as coal or nuclear plants are *a priori* achievable only for companies with sufficient revenue and capitalization to support the building costs, and low capital cost technologies such as small renewable facilities are at all investors' reach. But the thorough investigation of investment conditions show that original financing methods such as conjoint investment from a power generation companies consortium or financing

from long-term electricity purchasers can broaden the scope of companies able to make capitalistic investments [12][13][14].

The evolution of these investment conditions depend on both **policies** and **technical progress**, policies impacting investments through incentives, and technical progress being the key to cost reduction. However, among the technologies being studied, coal, gas, hydro and nuclear are considered to be time-tested and expect less progress than wind and solar[1].

## 5. Building of Scenarios

*A.Assumptions*

The analysis has thus allowed us to identify three drivers for investors' decisions:

- Policy, i.e. incentive framework given by European energy policies;
- Economic driver, i.e. structure of electricity markets, with level of centralization, concentration and competition...
- Technical change, i.e. relative evolutions of efficiency and costs of available technologies for gas, coal, nuclear, hydro, solar, wind.

In order to build investments scenarios based on these drivers, it is necessary to define more precisely the key aspects of these drivers we have chosen to focus on and to extract from our previous analysis assumptions regarding their evolution in the time horizon of our study.

The policy driver actually contains three dimensions:

- Climate policy, which will determine the incentives regarding carbon emissions, and promote low carbon energies, which are at the heart of our study. For our scenarios, we identify a strong climate policy scenario and a moderate climate policy scenario that can be quantified by their carbon price ranges, carbon pricing being the key tool of climate policy. Moderate climate policy would consist in pursuing carbon emission trading through the EU Emission Trading System, with carbon pricing increasing from a dozen \$/ton (today's price) to \$45/t$CO_2$ in 2040. Strong climate policy would increase carbon price up to 120 \$/t $CO_2$ in 2040 [5];
- Nuclear policy, for the use of this energy can be controversial according to the national context, the positions in the 5 studied countries being very different. For our scenarios, we identify the three positions currently observable and assume that they do not change within the considered period[2]: France has adopted a strongly pro-nuclear position, UK a moderate pro-nuclear position, Germany, Italy and Spain an anti-nuclear position;
- Electricity market reform policy, which will have a direct influence on investors' environment and investors' profiles themselves. For our scenario elaboration, this driver is included in the second one: "economic driver".

[1] It is true though that nuclear technology still experiences innovation, but even new generations of nuclear reactors (Generation III, Generation IV) are based on experienced concepts : Pressurized Water Reactors for Generation III, which is one of the most current concepts in operation today, and Sodium-Cooled Fast Reactors for Generation IV, the technology of which was experienced in France in the eighties with demonstrators Phenix and Superphenix, and is today in operation in Russia on a few reactors (BN-600, BN-800).

[2] This assumption may be considered a limit of the scenarios elaboration, in particular, France could change its position from "strongly pro-nucear" to "moderate "pro-nuclear"; nevertheless, such political positions commit long term industrial behaviors and for this reason assuming certain inertia of the pro or anti-nuclear position is relevant.

The market structure driver contains several aspects: the level of concentration and competition of the market that can be characterized by the number of actors present on the market and the Herfindahl–Hirschman Index (HHI[3]) and related to that, the market policy lead by the country, which will have an influence on both the market concentration and market coordination, which is essential to investors' decision. As a first approach, we will assimilate the market coordination aspect as part of the energy policy aspect[4] and for the "market structure" driver; we will only consider it under the concentration angle, and have a high concentration and low concentration market assumptions using the HHI: as in the European Commission Guidelines about competition, we consider a market in which HHI is lower than 1000 as competitive and low concentrated whereas a market in which HHI is in excess of 2000 is highly concentrated.

The technical change driver corresponds mostly, as we said earlier, to the expected technical change for recent renewable technologies, i.e. wind and solar. For this driver, we make a high technical change assumption and a low technical change assumption. The technical change would impact investment costs, generation costs, capacity factors of each technology; WEO 2011 scenarios allow us to make estimates about expected cost reduction [4]. Since the impact on these different costs is quite homogenous according to the expected progress for one technology, overnight investment cost reduction is a relevant indicator: Table I gives orders of magnitude of investment cost reduction for the two assumptions, which shows that progress is mostly expected for solar technologies (PV and CSP).

Table 2. Investment cost reduction between 2010 and 2040

| Technology | Low technical change | High technical change |
|---|---|---|
| Onshore wind | 10% | 20% |
| Offshore wind | 25% | 50% |
| Solar PV (utility and rooftop) | 50% | 75% |
| Concentrated solar power | 40% | 90% |

Own calculs based on [5], [15]

As a results of the number of hypothesis (2 "climate policy" hypothesis, 2 "market structure hypothesis, 2 "technical change" hypothesis, and 3 "nuclear policy" hypothesis), a total of 2*2*2*3 = 24 different scenarios are possible. Since we assume that nuclear policy is given data and remains unchanged, we have for each country 8 possible scenarios presented in next Table:

Table 3. Scenarios and corresponding assumptions in favour of Fast Reactors

| | | | |
|---|---|---|---|
| Scenario 1 | strong climate policy | low technical change | concentrated |
| Scenario 2 | strong climate policy | low technical change | not concentrated |
| Scenario 3 | strong climate policy | high technical change | concentrated |
| Scenario 4 | strong climate policy | high technical change | not concentrated |
| Scenario 5 | low climate policy | low technical change | concentrated |
| Scenario 6 | low climate policy | low technical change | not concentrated |
| Scenario 7 | low climate policy | high technical change | concentrated |
| Scenario 8 | low climate policy | high technical change | not concentrated |

Scenarios are classified from the most favorable to Fast Reactors investment to the least favorable, as we will explain in detail in section B.

[3] HHI definition, with $s_i$ the market share of firm i in the market, and N the number of firms:

$$H = \sum_{i=1}^{N} s_i^2$$

The more HHI is low, the more the market is competitive, and the more HHI is high, the more the market is concentrated.

[4] For instance, a strong climate policy will imply market coordination towards low carbon energies.

*B. Scenarios Discussion*

Scenarios identified as favorable to Generation IV nuclear are the ones with strong climate policy hypothesis i.e. scenarios 1 to 4.

As our technical change hypothesis mostly corresponds to progress in recent renewable technologies, and technological maturity of Fast Reactors in 2040 is assumed to be achieved, the ones most favorable to Fast Reactors are the ones that combine strong climate policy and low technical change, i.e. scenarios 1 and 2. Since the hypothesis of a non-concentrated market implies multiple actors with small market shares, we could think that scenario 1 should be the most favorable one to capitalistic investments such as nuclear ones. However, this assumption has limits, for scenario 2 does not exclude the possibility for capitalistic investments. The presence of multiple actors with small shares on a market does not necessarily mean they have little investments capacity: given the multinational profile of some investors, they may have important investment capacities despite their low market share. Moreover, market coordination and conjoint investments could also make capitalistic investments possible on a non-concentrated market. We thus consider scenarios 1 and 2 as both favorable to Fast Reactors. Lastly, considering the nuclear policies in the different countries we focus on, scenarios 1 and 2 are favorable to future nuclear reactors in France and UK only. In Germany, Italy, and Spain, they tend to be favorable to renewable.

Scenarios 3 and 4 would then be the ones favorable to both time-tested and recent low carbon technologies like nuclear, hydropower, wind and solar. The concentration factor makes scenario 3 in favour of nuclear in France and UK as well as scenario 1 and 2, whereas scenario 4 would be more favourable to small generation faticilities such as recent renewables. In Germany, Italy, and Spain, they tend to be favorable to renewable again.

Scenarios 1 to 4 are thus favorable to investment in both renewable and nuclear in France and UK, scenarios 1 and 2 tending to have more nuclear investment and scenario 3 and 4 more renewable investment. They are favorable to investment in renewable in Germany, Spain and Italy that are anti-nuclear countries. In all countries, fossil fuel based technology will lose market shares according to these scenarios. This means, especially for scenario 4, that back-up generation due to renewable intermittency will be ensured by non-intermittent hydraulic power and by nuclear power. It is necessary to point out that such a situation means a lower load factor for nuclear power and thus an important loss of competitiveness on generation costs. As a consequence such massive low carbon investments situations would possible only if climate policies were strong enough to maintain nuclear investment attractive compared to fossil fuels and especially gas, or if technical change could bring solutions to intermittency such as mastering long term storage or interconnection between numerous sources. For investment in both nuclear and renewable, scenarios 1 and 2 are thus more favorable.

Scenarios 5 to 8 are the ones with low climate policy and are thus more favorable to coal and gas investments than the previous ones.

Scenarios 5 and 6 are still favorable to nuclear due to the low technical change factor, scenario 5 being slightly more favorable due to market concentration factor. In these two scenarios and especially in scenario 6, gas investment will be promoted, for it is a low-capital, flexible technology technically suited to be a back-up capacity to renewable and economically suited to low load factors.

Scenarios 7 and 8 are the least favorable ones to low carbon technologies, 7 being more favorable to gas than coal due to market concentration factor and 8 to both gas and coal.

More generally, among low carbon technologies, these scenarios tend to reduce nuclear investment in favor of gas and coal.

## 6. Conclusion

This study identifies the key drivers of investors' choices and builds scenarios of European generation mix evolution based on these drivers' evolution in the future. On a total of 24 scenarios, 3 have proven favorable to a "fast" development of Fast Reactors (which would mean in the next decades). A description was given of these and other scenarios as well. This scoring may appear as not highly favorable in the present context of economic crisis. However, in the case of a rise of a stringent limitation of carbon emissions, they may appear as a necessity.

In addition, there is also an indirect driver "public acceptance of the technology" which is for now included in the nuclear policy driver. However, public rejection could appear for renewable as well because of land use and landscape transformation.

Thus, this driver may play a double role: on first hand, it can limit the development of new energy technologies (renewable and nuclear). On the other hand it can push new technologies with improved mastering of risks, such as Generation IV Fast Reactors.

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# Downscaled estimate of uranium resources produced from phosphates and impact on the penetration of sodium fast reactors

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**Abstract.** Future energy demand scenarios elaborated by international organisations are always more ambitious in terms of the installed nuclear power required, particularly when it comes to meeting requirements to curb climate change. The availability of natural uranium resources is a major constraint in terms of meeting this demand.

The first part of this paper outlines the current assessment of uranium resources contained in phosphate rocks, which are far from being as abundant as some studies may have first predicted.

Furthermore, the production of uranium as a by-product of phosphate is determined by the world production of phosphoric acid. The exorbitant cost of producing uranium alone (without recovering the phosphoric acid) would not overcome this limit on the production capacity.

The second part of the document analyses the impact of such a reduction in uranium resources on how the world's nuclear fleet evolves.

Different assumptions on the global uranium resources are taken into consideration. Sodium fast reactors (SFR) will be built in line with the availability of plutonium, including fast breeders when necessary.

The capacity of the nuclear industry to meet the different power demand forecasts around 2150 is therefore examined.

## 1. Introduction

Future world energy demand scenarios are increasingly ambitious in terms of the installed nuclear power required to curb greenhouse gas emissions.

Current light water reactors use thermal neutrons and burn uranium, whereas some future Generation IV reactors using fast neutrons will be capable of recycling their own plutonium and already-extracted uranium (self-sufficient or breeder fast reactors). It is therefore important to have an accurate estimate of the available uranium resources in order to plan for the world's future nuclear reactor fleet.

This paper aims at demonstrating the importance of making sure the uranium and plutonium resources can meet these nuclear power requirements.

After having reviewed current knowledge on conventional uranium resources, the first part of this paper focuses on unconventional resources recovered from phosphate rocks. Today, the most accurately identified resources will only allow a nuclear fleet to operate for a limited period of time; additional resources – unknown or unconventional – will be required to ensure the operation of a nuclear power fleet for several decades.

If we can provide an accurate estimate of the potential uranium resources, we can then check that they correspond to the future energy demand scenarios. The second part of this paper sets out to do this, by first examining light water reactors which burn uranium, and then examining a mixed fleet with both light water reactors and fast reactors which use plutonium.

## 2. Primary supply of uranium

Since the mid-sixties, in cooperation of the Member Countries and States, the OECD's Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA), have regularly updated their report which summarises the current status of uranium exploration, resources and production, together with the nuclear power plant requirements. The latest version of this report, generally called the "*Red Book*", is titled "*Uranium 2011: Resources, Production and Demand*" [1].

Resource assessments in this Red Book are divided into distinct categories that stand for different degrees of certainty concerning the indicated amounts. The resources are subdivided into ranges on a production cost basis, i.e. the cost of recovered uranium at the ore processing plant.

So-called '**conventional resources'** are those that allow uranium to be recovered as a primary product, a co-product or a major by-product (e.g. in copper or gold mines). '**Unconventional resources'** are very low-grade resources and those from which uranium is only recoverable as a minor by-product.

Conventional resources consist of identified resources and undiscovered resources.

- **Identified resources** consist of 'Reasonably Assured Resources' (RAR) and 'Inferred Resources' recoverable at costs lower or equal to USD 260/kg U: 6.3 MtU [2]
- **Undiscovered resources** consist of 'Prognosticated Resources' and 'Speculative Resources': 10.4 MtU [2].

Though there are multiple unconventional uranium resources, we only take into account those having already been exploited on an industrial level, i.e. phosphate rock deposits.

Phosphate deposits can be divided into two categories: igneous phosphate rocks (13%) and sedimentary phosphate rocks (87%) [3]. The presence of uranium in phosphate rocks can be explained on an atomic scale, by the substitution of a calcium atom by a uranium atom in the crystal lattice of the phosphates (apatite) [4][5][6].

The USGS in 2011 [7] estimates that the world phosphate reserves are equivalent to 65 Gt. These reserves are essentially found in Morocco (77%), China (6%), Algeria (3%) and Syria (3%). However, the production of phosphate (176 Mt in 2010) differs: China ranks as the first producer (37%) followed by Morocco (15%) and the US (15%). The US phosphate production has been declining since the beginning of the century, whereas China has considerably increased its highly cost-effective production. The phosphate production in Morocco has not really changed over the past twenty years.

The Red Book provides several different estimates of the uranium reserves in phosphate rocks to underline the uncertainties of all these reserves:

- 22 MtU based on the De Voto & Steven report [8],
- 9 MtU in the AIEA report 2001 [9],
- 7.3-7.6 MtU reported in 1965-1993 Red Books.

Assuming the global phosphate rock reserves to be 65 Gt [7] and based on the assumption of an average 100 ppm uranium concentration in phosphate rocks [10][2][4][3][11], it can be said that

6.5 MtU is contained in phosphate rocks. This is very close to the last Red Book figures at 7.3-7.6 MtU, but different from the 9 MtU given in the 2001 IAEA assessment.

Nonetheless, this figure does not take into account inevitable losses due to the imperfection of many processes. Uranium can only be recovered from phosphate rocks by using phosphoric acid, which is a by-product of the wet phosphoric acid process and the first step to produce fertilizers. When phosphate rocks are dissolved with sulphuric acid, it generates both phosphoric acid and phosphogypsum. The majority of uranium passes into the phosphoric acid (93%) while only a minor proportion remains in the phosphogypsum [12]. In addition, only 72% of phosphate rocks are used to produce phosphoric acid [13][14]. Finally, the rate of uranium recovery from phosphoric acid can reach 90% with the DEHPA-TOPO process [15][16]. **Considering all these losses, 3.9 MtU is expected to be recovered from the 6.5 MtU contained in the phosphate reserves.** In fact, this figure could fall to 3.4 MtU when excluding igneous phosphates rocks (13% of the global phosphate rock reserves) which are known to contain a lower uranium concentration and could be unprofitable.

The production of a by-product depends on the production of the main product, which is why uranium recovery from phosphates is limited by the phosphate production.

Assuming an annual phosphate production of 176 Mt with a concentration of 100 ppm of uranium in phosphate deposits, and assuming that 72% of the phosphate production is devoted to phosphoric acid with a global rate of recovery of 84%, then the maximum uranium production from phosphates will be 10.6 ktU per year. The global recovery rate includes 7% of uranium lost in phosphogypsum and 10% misplaced during the recovery of uranium from the phosphoric acid with the DEHPA-TOPO process.

To get around this limited production and meet the world demand at the same time, uranium should be recovered from phosphate rocks as a primary product. In this case, uranium should bear all the costs: extraction, phosphoric acid production and uranium recovery.

The expected cost of production – according to the different hypotheses - should range between 1,300 and 6,300 $/kgU, which is largely prohibitive and therefore hardly feasible [17].

## 3. DEPLOYING NUCLEAR POWER WITH LIMITED URANIUM RESOURCES

The available quantities of uranium are limited and will thus hinder the development of the world's nuclear power fleet. If we can provide an accurate estimate of the potential uranium resources, we can then check that they correspond to the future energy demand scenarios.

This paper takes into account four uranium resource limits defined as given in the Red Book and according to our own estimate of unconventional uranium resources recovered from phosphate rocks:

- 6 Mt, which represents the quantity of identified conventional uranium resources,
- 20 Mt, which comprises 16 Mt of both identified and undiscovered conventional uranium resources, together with 4 Mt from phosphate rocks,
- 38 Mt, which comprises 16 Mt of both identified and undiscovered conventional uranium resources, together with 22 Mt from phosphate rocks,
- 90 Mt, which takes into account the hope that mining exploration will find substantial new resources; this figure is based on a very optimistic view rather than an evaluation.

### *3.1. Energy scenarios*

In our past foresight studies on the development of nuclear power [18][19], we chose to work with the IIASA (1998) scenarios which, at the time, were the only scenarios providing an energy mix for the 21st century. More recent scenarios (IIASA 2007 [20], WNA2010 [21], AIEA 2010 [22]) predict much higher installed nuclear power estimates, yet even meeting these first much less ambitious scenarios proves to be problematic as will be seen further on in this paper.

This is why we have chosen to conserve the IIASA scenarios from 1998 [23] for this study.

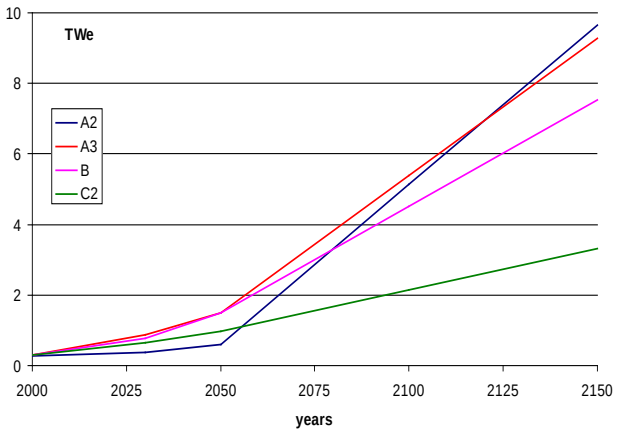


Figure 1: IIASA Scenarios - Requested nuclear power

The **A2 Scenario** is a strong global growth scenario of around 2.7% per year, with the preferred short-term use of oil and gas resources. Nuclear energy represents 4% of world energy demand in 2050 and 21% in 2100.

The **A3 Scenario** is also a strong global growth scenario with a more gradual introduction of nuclear energy than in scenario A2; nuclear energy represents around 11% of world energy demand in 2050 and 22% in 2100.

The **B Scenario** is a "business as usual" world growth scenario during the 21st century (around 2% per year).

The **C2 Scenario** corresponds to a strong intention to protect the environment against global warming. Nuclear energy represents around 12% of world demand for primary energy in 2050; this is close to twice as much as it represents today.

The IIASA scenarios consider a strong increase in the world demand in primary energy. Even if the share of nuclear power is less than 20% of the total, it supposes a quite significant increase in the installed nuclear power capacity.

These scenarios were established prior to the Fukushima accident but are still deemed relevant since the effects should be short or medium term, especially since the countries with the highest nuclear power demands such as China, India and Russia have not modified their nuclear energy policy.

### *3.2.Reactor technologies*

This study takes into consideration three different reactor technologies:

- Pressurised water reactors (PWR). These Generation II reactors are representative of the reactors currently operating in the world today.
- European Pressurised (or Evolutionary Power) Reactors (EPR). These Generation III reactors are representative of what is currently being built or scheduled for construction throughout the world.

- Fast reactors (FR). These Generation IV reactors use plutonium as fissile material.

It is considered that there is no recycling of fissile material in the PWRs and EPRs.

The quantity of uranium consumed during the lifetime of the reactor is called 'engaged uranium'. An EPR is only built if there is some foresight on the availability of uranium resources. When the consumed and engaged uranium exceeds one of the above-mentioned limits, it will be impossible to build a new reactor requiring uranium, i.e. an EPR in our case. The only reactors that can be built once this limit has been reached are fast reactors which operate with plutonium. Considering that plutonium has to be produced and is not available in limitless quantities, it will become possible one day that we cannot build enough reactors and no longer match supply to demand.

### *3.3.Exclusive deployment of EPRs*

Our first calculations are based on the assumption that only EPRs can be built and that fast reactors will never become available. This assumption underlines the importance of Generation IV reactors for the future of nuclear power.

Figure 2 illustrates the demand and production of nuclear power (in TWhe) for the A3 and C2 scenarios.

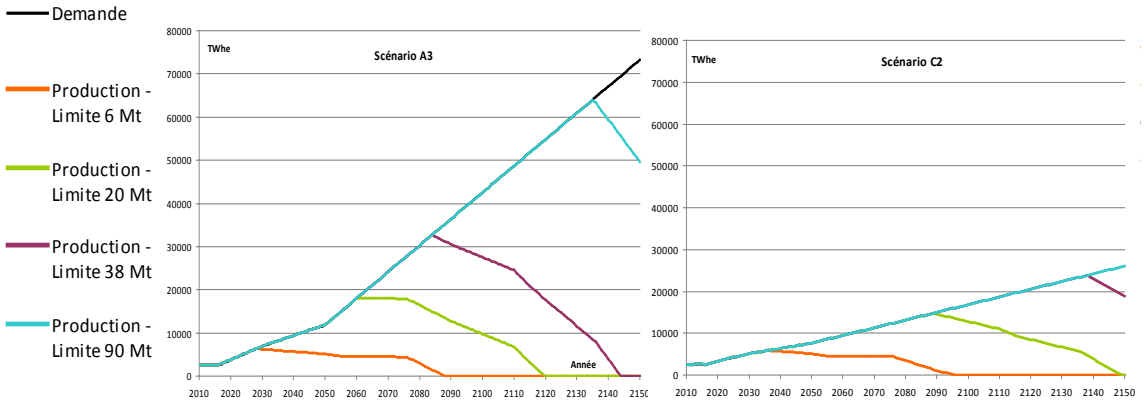


Figure 2: Light water reactors only – Demand and power production (in TWhe) according to the natural uranium limit

When no other reactor can be built due to the lack of uranium, the installed power starts to drop and the reactors cannot be replaced at the end of their lifetime.

The A3 scenario shows that EPRs do not meet the entire nuclear power demand even with a uranium limit of 90 Mt. The problem becomes quickly apparent (in less than 20 years) if the limit is only 6 Mt.

The C2 scenario represents the slowest growth of nuclear power with less pressure on the uranium resources. Nonetheless, only the uranium limit of 90 Mt will meet the nuclear power demand up to 2150.

These results stress the need to develop fast reactors in order to ensure the sustainable production of nuclear power.

### *3.4.Possible deployment of FRs from 2040*

From 2040, our calculations give top priority to building fast reactors (self-sufficient reactor or breeder reactor with a gain of 0.3 if necessary). However, EPRs will be built if there is an insufficient stock of available plutonium and a sufficient supply of uranium. If both uranium and plutonium are lacking, no reactors will be built and it will be impossible to meet the demand.

Figure 3 shows the possible nuclear power production for the A3 and C2 scenarios based on the availability of plutonium and the uranium limit taken into consideration.

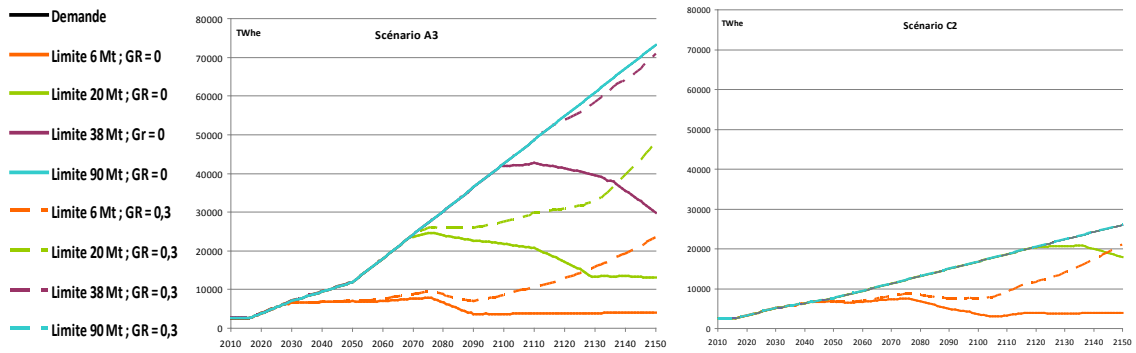


Figure 3: Light water reactors and fast reactors – Demand and power production according to the natural uranium limit (TWhe)

For the C2 scenario with limited nuclear power growth, only the very restrictive limit of 6 Mt curbs the deployment of nuclear energy. Fast reactors with a breeding gain are necessary when the uranium limit is 20 Mt.

Considering the availability of plutonium, it is not possible to build as many EPRs and FRs as intended in the other scenarios.

## 4. Conclusion

Knowledge of the uranium resources is a prerequisite to studying nuclear power deployment scenarios. The largely quoted estimate of 22 Mt of uranium recovered for phosphate rocks remains very uncertain and will probably be seriously downscaled. Based on our current knowledge of phosphate resources, 4 Mt of recoverable uranium seems to be a feasible reference value.

Considering light water reactors exclusively, 38 Mt of uranium is required for the least ambitious C2 scenario up to 2100 when it will no longer be possible to build reactors. For the most ambitious A3 scenario, 38 Mt of uranium would make it possible to build reactors up to 2080, whereas 6 Mt would only stretch as far as 2030. Without any other technology besides light water reactors, the nuclear power production is not sustainable for more than 100 years.

The downscaled estimate of the uranium resources available in phosphate rocks means that problems with deploying an EPR fleet will arise 20 to 50 years earlier than expected depending on the scenario.

Fast reactors are therefore essential. Their deployment is nonetheless restricted by the availability of plutonium and they do not meet the forecast energy demand in all the scenarios.

The deployment of nuclear power on a global scale requires more uranium than that provided by the reserves currently known to exist. Mining exploration is therefore essential in the hope of discovering new uranium resources.

It is also essential to examine how we can optimise the use of uranium. This raises the question of deploying fast reactors. To overcome the problem of plutonium availability, their first loads can use enriched uranium. So the deployment of a fast reactor fleet could be fast-tracked but at the expense of PWRs whose uranium resources would become even scarcer. Consequently, the growth and composition of the nuclear fleet would be modified (paper to be published).

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# Economic relevance of starting an SFR with enriched uranium

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**Abstract.** Though it is currently estimated that the identified uranium reserves could meet our demands for another hundred years or so at our present rate of consumption, an increased number of future nuclear power plants in numerous countries will reduce the potential duration of our stocks.

Since the natural uranium market deals with *engaged uranium (Future uranium stocks required to operate already-installed reactors for their remaining lifespan) rather than consumed uranium*, this market is expected to feel the pressure well before the 100-year mark.

Sodium fast reactors (SFR) use uranium much more efficiently than the current industrial reactors, which makes them a feasible solution for the sustainable development of nuclear energy. A sufficient quantity of plutonium is nonetheless needed to start up an SFR, with this plutonium already being produced in pressurised water reactors (PWR) for instance and representing about 1% of the mass of spent fuel.

Past studies have revealed that there would be a lack of available plutonium stocks produced by PWRs if all the new future reactors built in the world were to be SFRs only. Owing to the lack of available plutonium, the only remaining solution is to start up SFRs with enriched uranium.

The purpose of this study is to assess the option of starting up SFRs with enriched uranium, which we have dubbed SFR-U5. Our research focuses on the competitiveness of this solution by comparing it with an EPR in terms of the cost of natural uranium ($U_{nat}$), the possible excess cost of SFRs and the discount rate.

For a reactor lifespan of 60 years, the SFR-U5 consumes three times less uranium than the EPR and represents a 60% reduction in terms of separative work units (SWU), though its requirements are concentrated over the first 7 years of operation. The SFR-U5 will therefore consume two times more natural uranium and 2.5 times more SWUs than the EPR during the first 7 years of operation.

This beginning-of-life consumption rate has a negative economic impact, which is all the more so since the discount rate is high.

Though starting up an SFR with enriched uranium is therefore a technically feasible solution, it nonetheless appears to be a very costly one. Such efforts should therefore at least be optimised by sodium fast breeder reactors.

## 1. Introduction

According to almost all forward-looking studies[1][2], the world energy consumption should increase in the future decades, mostly because of the growing world population with an ever-increasing standard of living. This is occurring at a time when greenhouse gas emissions have to be sharply curbed and when the security of energy supplies has to be optimised.

Nuclear energy is a currently available technology capable of rising to both challenges. In this context, nuclear power should be a key strategic component in meeting this increasing energy demand.
Nevertheless, the known reserves of natural uranium will only meet the demand over the next hundred years at the current rate of consumption [3]. This period will be even shorter if the world light water

reactor (LWR) fleet increases in size. Sodium fast reactors (SFR) use natural uranium-much more efficiently, which makes them a feasible solution for the sustainable development of nuclear energy.
On a global level, however, scenario studies show that the transition from the current LWR fleet to an all-SFR fleet is expected to be long (post-2150), mainly due to the lack of available plutonium [4].

We therefore examine the possibility of starting such reactors with enriched uranium as recommended in ref. [5], thereby circumventing restrictions due to the quantity of available plutonium. The issue for electric utilities is therefore to know whether it is more advantageous to immediately start building an SFR using enriched uranium (hereafter referred to SFR-U5, which gradually becomes an SFR-Pu since it is an iso-generator) or to first build a LWR that, at the end of its lifespan, will have produced enough plutonium to be replaced by an SFR started with Pu. This issue will be examined in terms of the competitiveness of this solution by comparing an SFR with an EPR (European Pressurised or Evolutionary Power Reactor) in relation to the price of natural uranium ($U_{nat}$), the possible additional cost of SFRs, and the discount rate.

## 2. Methodology

Assuming the same lifespan of 60 years for PWRs (pressurised water reactor as a representative for LWR) and SFRs, the electric utility will be faced with the same situation at the end of this period (operation of a PWR or of an SFR started with enriched uranium): it will have a sufficient quantity of plutonium to replace this reactor with an SFR started with plutonium. This quantity of plutonium corresponds to two cores: the first core and an equivalent quantity for the next few reloads (the reactor will then produce its own Pu).

This first 60-year period will therefore be decisive, i.e. spent providing electricity with a PWR or an SFR and accumulating the necessary quantity of Pu to start up an SFR-Pu at the end of this period.

We therefore consider that there must be enough enriched uranium for the first core and the following 5 reloads, after which there will be no need to provide additional enriched uranium. The following reloads will use the reprocessed fuel from this SFR. This reactor will contain increasingly less $^{235}U$ and increasingly more Pu, yet the fissile material recovered will be equivalent and could be reloaded without having to top it up since SFRs are iso-generators.
We will be comparing these two types of reactors according to the quantities of uranium used and in economic terms.
The costs of reprocessing spent UOX from EPRs or of ensuring its disposal via vitrification have been disregarded. These costs will only have an impact at the end of the reactor's lifespan and are therefore considered to be second order.
We will also determine the economic value of Pu by equivalence between an SFR started with uranium and an SFR started with Pu. In this way, our study differs from that cited in reference [6], which compares the costs of an SFR started with uranium or plutonium, with the economic value of Pu for the first core and the following reloads being defined by the cost of reprocessing the spent fuel from PWRs.

### *2.1. Assumptions applied to the reactors and fuel reloads*

Concerning EPR fuel, the first core and its annual reloads are taken into consideration, i.e. the following costs:
Natural uranium + conversion + enrichment + fabrication

Concerning SFR-U5 fuel, it is assumed that the core is reloaded by $1/5^{th}$ each time with the fuel remaining a little longer than 5 years in the core. It is assumed that the length of the fuel cycle out of the reactor is 5 years (cooling period after unloading up to the fabrication of a new sub-assembly that can be loaded into the reactor). The first core and the following 5 reloads must therefore contain

enriched uranium, while the subsequent reloads use the plutonium produced by the SFR thereafter. We therefore have the following costs:
At the beginning: Natural uranium + conversion + enrichment + fabrication
For each reload every year: reprocessing + fabrication.

Table 1. Characteristics of reactor technologies and their fuel

| Characteristics of the technology | **Unit** | **EPR-type PWR**[1] | **SFR-U5** |
|---|---|---|---|
| **Lifespan** | years | 60 | 60 |
| **$U_{235}$ enrichment** | % | 4.9 | 14.4 |
| **Mass of $U_{235}$ in core** | Tonnes of $U_{235}$/GWe | 3.9 | 8 |
| **Reloading** | Tonnes of $U_{235}$/GWe/year | 0.78 | 1.4[2] |

The tails assay ($T_{\text{depleted uranium}}$) is optimised in relation to the natural uranium price, its conversion and its enrichment. The more expensive the natural uranium, the more it will be depleted to extract the maximum amount of uranium-235. This determines the quantity of natural uranium and the separative work units (SWU) needed to produce enriched uranium.
The quantities of uranium required will therefore depend on the assumptions that we will apply as regards the variations in the price of natural uranium.
Table 2 indicates the optimal tails assay and the cost of reloading an EPR for a conversion equivalent to €10/ kg and an enrichment equivalent to €110/ SWU.

Table 2. Optimised tails assay and cost of reloading an EPR as a function of the natural uranium price for 1 GWe

| Natural uranium price (€/kg) | Tails assay (%) | Natural uranium per reload (tonnes) | SWU per reload | Cost of reload (€/MWh) |
|---|---|---|---|---|
| 80 | 0.25 | 160 | 123 | 4.42 |
| 100 | 0.23 | 154 | 129 | 4.81 |
| 200 | 0.16 | 138 | 152 | 6.65 |
| 300 | 0.13 | 131 | 168 | 8.34 |
| 400 | 0.11 | 127 | 181 | 9.97 |
| 500 | 0.09 | 124 | 192 | 11.56 |

[1] For comparison with an SFR, we chose the characteristics of the EPR (assumptions may differ in relation to reference [5]). The figures are given in relation to an equilibrium cycle
[2] For the first 5 reloads of an SFR. The U5 enrichment is given for the first core : it constanly decreases as the SFR becomes an SFR-Pu

It is considered that the purchase of uranium and its conversion into UOX fuel take place 2 years prior to its loading, with enrichment and fabrication taking place 1 year beforehand.
It is considered that reprocessing and the fabrication of MOX fuel take place 1 year prior to loading.

Two discount rates are studied:
1) 8% for 30 years and then 3%,
2) 4% for 30 years and then 2%.

## *2.2. Costs covered*

### *2.2.1. Investment and operating costs*

The construction cost of an EPR is estimated at €2,000/kWe. Prime contracting represents 11% of this cost, while pre-operation represents 7%. A construction period of 5 years is taken into consideration.
The dismantling cost corresponds to 15% of the full investment costs.
The operating costs are broken down into two parts: a fixed amount of €50.90/kWe and a variable amount which depends on production, equivalent to €0.00061/kWh.
The investment and operating costs of an SFR are considered in relation to those of an EPR. With an extra cost of x% on investment, the extra cost for operation is fixed at (x/2) %. A parametric study on the additional investment cost is detailed further on in this document.

### *2.2.2. Fuel cycle unit costs*

Table 3 indicates the unit costs for the different stages of the fuel cycle as a function of the discount rate.
As the UOX fuel from the first core and the following five reloads of the SFR is enriched with $^{235}U$ by 14.4%, we have estimated that its fabrication cost is closer to that of UOX fuel for an EPR than that of MOX for an SFR.

Table 3. Fuel cycle unit costs

| **Short-term/ long-term discount rate** | **8% then 3%** | **4% then 2%** |
|---|---|---|
| Conversion | €10/ kg | |
| Enrichment | €110/ SWU | |
| UOX fabrication in €k/t HM | 300 | 270 |
| UOX fabrication for an SFR in €k/t HM | 500 | 450 |
| MOX fabrication for an SFR in €k/t HM | 1,500 | 1,100 |
| Reprocessing of SFR fuel in €k/t HM | 1,500 | 1,100 |

## 3. Results and sensitivity study

### *3.1. Material flows during the different stages of the fuel cycle*

Table 4 specifies the material flows for the different stages of the fuel cycle under consideration, as well as the enrichment requirements for the reactor lifespan when the price of natural uranium is €100/kg for the reactor's entire service life (flows vary depending on the price of natural uranium through the optimisation of the tails assay). Year 0 corresponds to the year the reactor is commissioned.
Over the reactor's 60-year lifespan, it can be seen that the SFR-U5 uses three times less uranium than the EPR and requires 60% fewer SWU. Yet if we compare the fuel requirements over the first 7 years of operation, the SFR-U5 uses 2 times more natural uranium and 2.5 times more SWU than the EPR.
The following section analyses the economic benefits.

Table 4. Annual flow of materials (tonnes) and enrichment requirements (million SWU) for 1 GWe

| | EPR | | | SFR | | | |
|---|---|---|---|---|---|---|---|
| Year | Flow of natural uranium | Number MSWU | Flow of uranium enriched at 4.9% | Flow of natural uranium | Number MSWU | Flow of uranium enriched at 14.4 % | Reprocessing flow |
| -2 | 769 | 0.65 | | 1,628 | 1.67 | | |
| -1 | 154 | 0.13 | 80 | 293 | 0.30 | 56 | |
| 0 | 154 | 0.13 | 16 | 293 | 0.30 | 10 | |
| 1 | 154 | 0.13 | 16 | 293 | 0.30 | 10 | |
| 2 | 154 | 0.13 | 16 | 293 | 0.30 | 10 | |
| 3 | 154 | 0.13 | 16 | 293 | 0.30 | 10 | |
| 4 | 154 | 0.13 | 16 | 161 | 0.17 | 10 | |
| 5 | 154 | 0.13 | 16 | | | 6 | 10 |
| 6 to 57 | 154 | 0.13 | 16 | | | | 10 |
| 58 | | | 16 | | | | 10 |
| 59 | | | | | | | |
| Total | 9,844 | 8.27 | 1019 | 3,256 | 3.34 | 111 | 540 |

### *3.2. Competitiveness between the EPR and the SFR-U5: Impact of the natural uranium price*

To analyse the impact of the natural uranium price on the cost of a kWh, we have taken into account a linear variation of this price over time, starting around €100/kg when investing in a new reactor.

We have determined the limit of competitiveness of SFRs, i.e. variations in the price of natural uranium associated with a possible additional cost of SFRs, which makes it possible to level out the cost of kWh for both technologies.

Table 5 lists the results for discount rates of 8% and then 3%, and for 4% and then 2%. It gives the evolution (linear variation) under 60 years that should take the uranium price to make the MWh produced by an EPR equal to that produced by an SFR. We will refer to this as the "equivalence price". It also indicates the cost of MWh at the equivalence.
It can be seen that the results are rather different depending on the discount rate.

In the case of an 8% short-term discount rate, the impact over the first few years is significant. It emerges that the start-up of an SFR using uranium enriched at 14.4% is costly. The SFR kWh will only be competitive if uranium becomes extremely expensive at the end of the reactors' lifespan (exceeding €780/kg for no additional investment cost, and exceeding €1,700/kg if the SFR represents an additional investment cost of 10%).
When the discount rate is 4% and then 2%, the situation is slightly more favourable but nonetheless complicated: for an additional investment cost of 30%, the price of uranium will need to increase from €100/kg to €1,050/kg in 60 years for the SFR-U5 to be considered competitive.

Table 5. Cost of MWh and equivalence price of natural uranium as a function of the additional investment cost of SFRs and the discount rate

| | Construction cost of SFRs compared with EPRs | Equal | +10% | +20% | +30% |
|---|---|---|---|---|---|
| Discount rate: 8% then 3% | Cost of MWh (€) | 44 | 47 | 51 | 55 |
| | Equivalence price of natural uranium (€/kg) | 100 to 780 | 100 to 1,700 | 100 to 2,500 | 100 to 3,500 |
| Discount rate: 4% then 2% | Cost of MWh (€) | 27 | 29 | 31 | 32 |
| | Equivalence price of natural uranium (€/kg) | 100 to 150 | 100 to 430 | 100 to 750 | 100 to 1,050 |

The SFR-U5 could thus only be developed in the case of great pressure on the uranium market. In these conditions, its smaller use of natural uranium could justify its construction. Yet it is to be feared that numerous electric utilities will take the same line at the same time, which would result in enormous yet relatively limited SWU needs over time. The industrial feasibility of such a scenario is therefore far from being reached.

### *3.3. Estimating the economic value of Pu by equivalence between an SFR-U5 and an SFR-Pu*

In previous studies we have recommended a Pu economic value determined in terms of a holder of spent fuel faced with the choice of either direct disposal or reprocessing it to recover the uranium and plutonium. We have found it to be worth about €50/g.

By comparing an SFR started up with enriched uranium against an SFR started in the conventional manner with plutonium, we can determine a new equivalent economic value of Pu.

Table 6 indicates that the flows of uranium and plutonium required during the first few years to start the reactor and to meet the first reloads before being able to recover the Pu from the first spent reloads.

Table 6. Economic value of Pu for a natural uranium price of €100/kg and a discount rate of 8%

| For U at €100/kg | SFR started up with enriched uranium | | | | | SFR started up with Pu | | |
|---|---|---|---|---|---|---|---|---|
| Year | Natural U (tonnes) | Discounted cost of uranium (M€) | Discounted conversion cost (M€) | Discounted enrichment cost (M€) | Discounted fabrication cost of UOX (M€) | Pu flow (tonnes) | Discounted Pu flow (tonnes) | Discounted fabrication cost of MOX (M€) |
| -2 | 1,627.95 | 189.9 | 19.0 | | | | | |
| -1 | 293.32 | 31.7 | 3.2 | 198.7 | 30.0 | 8.75 | 9.45 | 82.62 |
| 0 | 293.32 | 29.3 | 2.9 | 33.1 | 5.0 | 1.57 | 1.57 | 13.77 |
| 1 | 293.32 | 27.2 | 2.7 | 30.7 | 4.6 | 1.57 | 1.46 | 12.75 |
| 2 | 293.32 | 25.1 | 2.5 | 28.4 | 4.3 | 1.57 | 1.35 | 11.81 |
| 3 | 293.32 | 23.3 | 2.3 | 26.3 | 4.0 | 1.57 | 1.25 | 10.93 |
| 4 | 161.33 | 11.9 | 1.2 | 24.4 | 3.7 | 1.57 | 1.16 | 10.12 |
| 5 | | | | 12.4 | 1.9 | 0.87 | 0.60 | 5.21 |
| total | 3, 256 | 338.3 | 33.8 | 354.0 | 53.5 | 17.49 | 16.83 | 147.20 |
| Discounted cost of UOX fuel (M€) | | 780 | | | | | | |
| Pu economic value = [discounted cost of UOX fuel – discounted cost of MOX fabrication]/ discounted Pu flow = 38 €/g | | | | | | | | |

The economic value of Pu is €38/kg for a natural uranium price of €100/kg. The same calculation for a natural uranium price of €200/kg gives a Pu economic value of €56/kg. It can again be seen that the price of natural uranium represents about 50% of the reloading cost; if the natural uranium price is doubled, the reloading cost is increased by about 50%, as is the cost of Pu accordingly.
We checked that the discount rate has very little impact since the calculation is only performed over 7 years (modification of the first decimal).

## 4. Conclusion and future prospects

As the known reserves of natural uranium will only meet the demand over the next hundred years at its current rate of consumption, SFRs that make better use of natural uranium are therefore expected to be built to ensure the sustainable development of nuclear energy.
Nonetheless, we have shown in previous studies that the current availability of plutonium would restrict their expansion.

This study therefore analyses the possibility of starting SFRs with enriched uranium in terms of competitiveness by comparing an EPR with an SFR as a function of 1) the price of natural uranium, 2) the relative cost of SFRs and EPRs, and 3) the discount rate.

For a given reactor lifespan, the SFR-U5 consumes three times less uranium than the EPR and represents a 60% reduction in terms of separative work units (SWU), though these requirements are concentrated over the first years of operation.
The SFR-U5 will therefore use two times more natural uranium and 2.5 times more SWUs than the EPR during the first 7 years.

This beginning-of-life consumption rate has a negative economic impact, which is all the more so since the discount rate is high.

With a discount rate of 8% and then 3% (short term and long term), it appears that the kWh for the SFR-U5 is only competitive if there is a severe shortage of uranium: if uranium becomes extremely expensive at the end of the reactors' lifespan (exceeding €780/kg for no additional (reactor?) investment cost, and exceeding €1,700/kg if the SFR represents an additional investment cost of 10%). Discount rates of 4% and then 2% are slightly more favourable for SFR-U5. Nonetheless, the natural uranium price would have to increase from €100/kg to €1,050/kg over 60 years to justify using an SFR-U5 since SFRs are 30% more expensive in terms of investment costs and 15% more expensive in terms of operation.

By comparing an SFR started with enriched uranium against an SFR started in the conventional manner with plutonium, we can determine a new equivalent economic value of Pu. It is about €40/kg for a uranium price of €100/kg and €55/kg for a uranium price of €200/kg.

Though starting an SFR with enriched uranium is a technically feasible solution, it nonetheless appears to be a very costly one. This would also require a high level of availability in natural uranium and a considerable enrichment capacity over short periods of time, which raises industrial issues in terms of acquiring the necessary capacity. Such efforts should therefore at least take advantage of the breeding possibility. Scenario studies will make it possible to assess the relevance of SFR-U5 reactors in the interest of developing the nuclear reactor fleet.

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# Non-Electric Applications of Fast Reactors

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## Abstract

*One advantage of the technology of fast reactors (FR) compared to light water reactors (LWR) is the high coolant fluid temperature. In addition to allowing a better overall thermodynamic electrical efficiency, that feature opens the possibility to investigate uses going beyond present nuclear technologies. In particular, a variety of non-electric industrial applications could be achieved, lying in the 250°C-500°C temperature range. In this paper, we shall focus on one of those, namely petroleum refining. It is shown that the use of a fast reactor enables delivering all main energy utilities required by the refinery plant (electricity, hot steam, hydrogen and water). The result turns out to be highly beneficial, enhancing the fuel output production by 8 to 10% for the same quantity of oil input. The corresponding value asset is tremendous and would by itself justify taking the financial risk for the plant owner. In addition, almost all $CO_2$ emissions associated to gas burning on site would completely vanish.*

## 1. Introduction

Today, the main use of nuclear energy in industry is realized by drawing out electrical power through the electric network. But that fails to grasp about one half of the overall energy needs utilized in the form of heat. The corresponding required heat temperatures are almost equally split in three regions: low temperatures, medium temperatures and high temperatures (Figure 1).

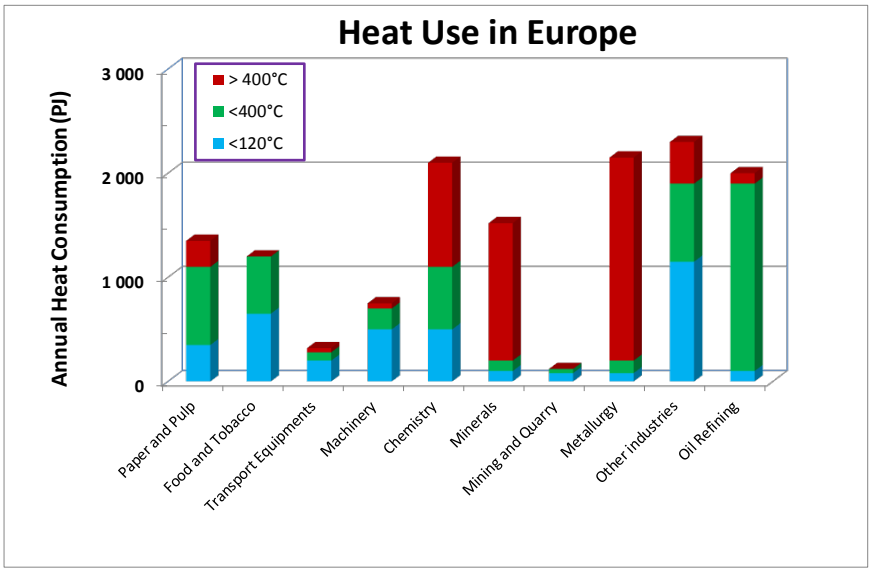


*Figure 1 – Annual heat consumption in the European industries [from **1**]*

Energy-intensive industrials usually own dedicated power plants able to fulfill the required energy supply on site. These combined heat and power (CHP) plants located close by the industrial process units are typically modern gas fired involving Combined Cycle Gas Turbines (CCGT) to

enhance efficiency (> 50%). In the past, the replacement of these CHP plants by nuclear reactors have been studied in many countries and for many different applications [**2**]. However, until now, no industrial company has ever constructed a commercial nuclear power plant (NPP) delivering both electricity and heat to feed in an industrial unit. The reasons for this cautiousness are known: the large initial investment, the nuclear safety and its corresponding penalties and the risk liability including the ultimate waste management. Notwithstanding, the benefits of nuclear energy are obvious. Nuclear delivers large amounts of cheap energy, guaranteed over a very long period of time and independent of other uncontrollable energy prices fluctuations. Moreover, nuclear is an effective carbon-free energy showing indirect greenhouse gas (GHG) emissions even lower than renewables [**3**]. In light of these advantages, it is therefore wise to reconsider which industrial applications might take best profit of the use of nuclear cogeneration.

## 2. Energy in Industry

Many industrial processes rely on energy for matter transformation, especially in chemistry (fertilizers, plastics, rubbers, fibers, chemical reactions,…), iron and steel industry (furnaces), food industry (sugar, milk, starch, fruits, vegetables,…), paper and pulp, cement production, metallurgy, glasses and ceramics. All these industries need electric power to activate machines like conveyor belts, pumps, centrifuges, extractors, filters or control-commands but they also need heat power to achieve their processes. Heat is usually brought in as hot water or steam at temperatures starting as low as 60°C but could sometimes reach up to 1700°C (Figure 2).

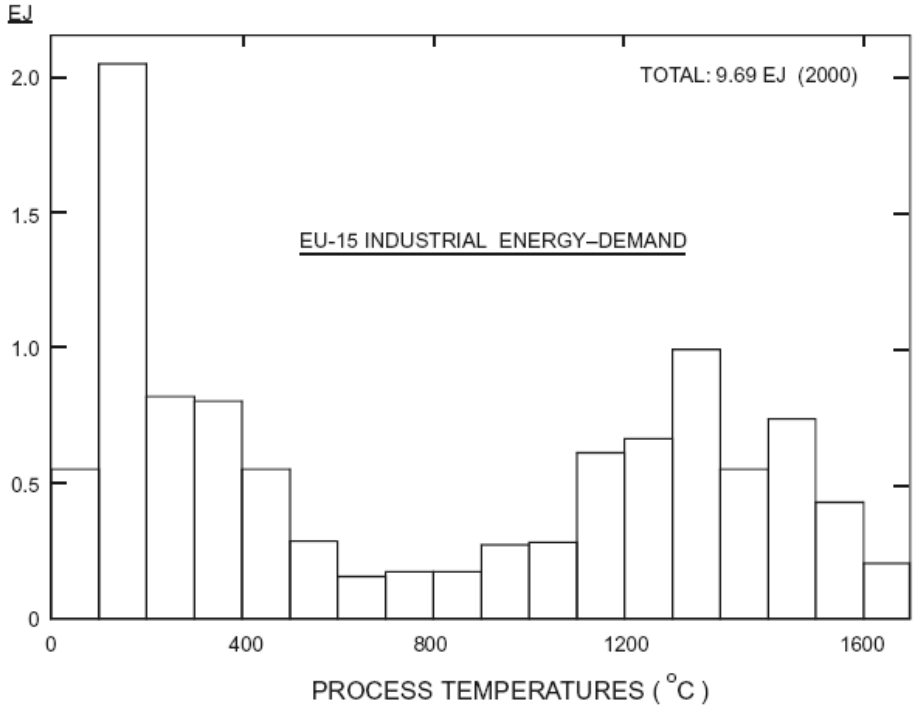


*Figure 2 – Amount of annual heat energy needed in the European industry as a function of the required process temperature [from **4**,**5**]*

Today, due to the high oil prices, most of the industrial heat is produced on site using gas in CCGT plants (Figure 3). Only a small fraction of the total energy (5.6%) is produced externally and delivered through heat networks. As energy in the European industry is mostly based on fossil fuels (74% of the 271 Mtoe in total), this results in large amounts of $CO_2$ emissions. Moreover, one should add to this figure 88.5 Mtoe of consumption for the energy industry own uses encompassing oil refineries and electric power plants, 85% of which (75.4 Mtoe) are fossil fuels based.

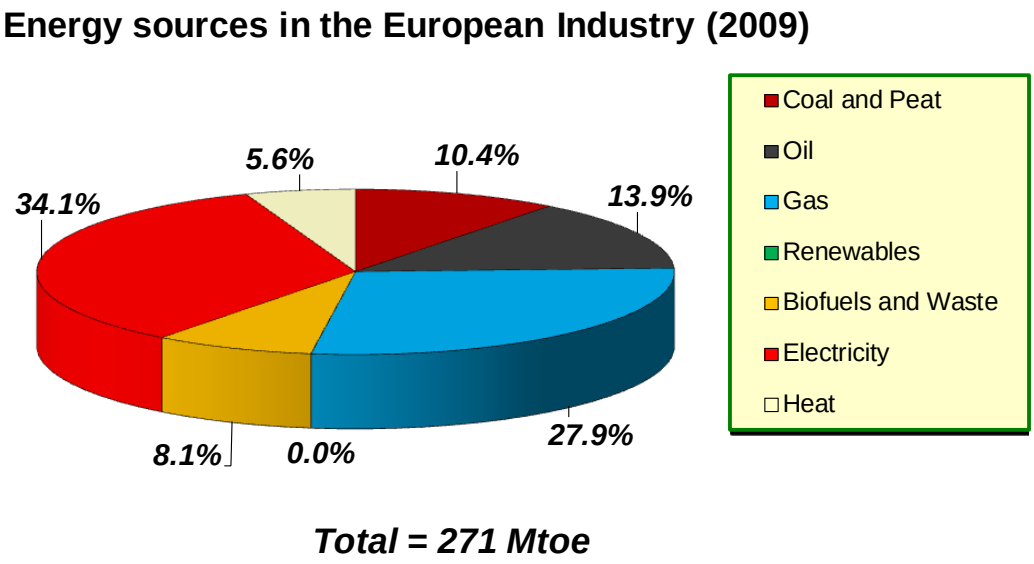


*Figure 3 – Energy consumption in the European industry by energy source [from **6**]*

## 3. What can offer nuclear energy?

One may classify the heat requirements in industry in three main categories: low temperatures below 200°C, medium temperatures in between 200°C to 500°C and high temperatures above 500°C. The three categories are about equal size in terms of energy needs. Clearly, steel fabrication and metallurgy lie in the high temperature class whereas food and paper industries will belong to the low temperature range. However, the medium category is of relevance for oil refineries and covers large areas of paper, food and chemical processes. Interestingly, the three categories correspond to three different reactor types. Light Water Reactors (LWR) are suitable for the low temperature region, fast Reactors (FR) for the medium temperature zone and (Very) High Temperature Reactors (HTR/VHTR) may cover at least a part of the high temperatures applications (Figure 4).

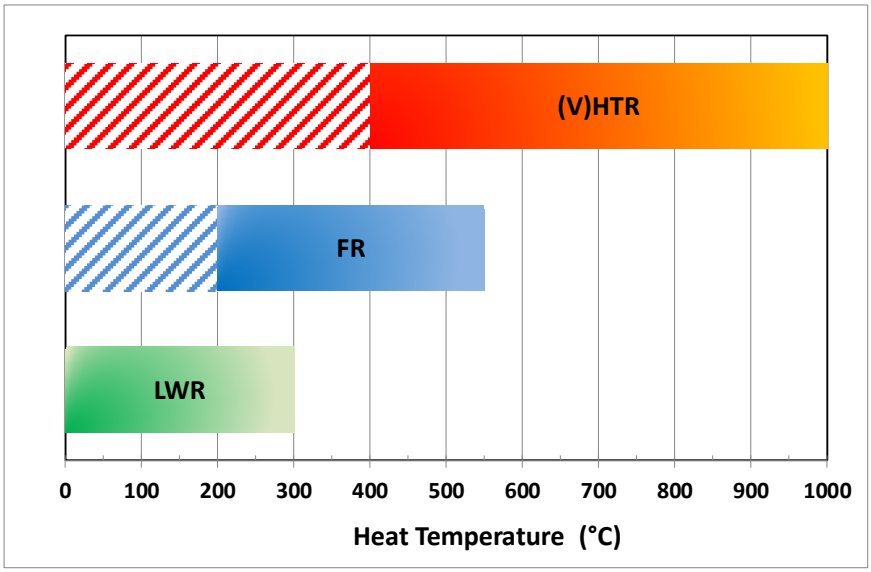


*Figure 4 – Three types of nuclear reactors fulfill the three temperatures zones covering most of the heat uses in industry.*

An important contribution of nuclear energy for industry would be to supply both electricity and heat for at least part if not all of the energy needs in replacement of fossil fuels fired CHP plants. Its primary advantage is economical, reducing the share cost of energy in the final product. Moreover, nuclear offers a continuous energy supply providing a guaranteed and stable cost over a very long lifetime. Finally, a NPP will ultimately wipe out all GHG emissions due to fuel combustion for the production site. Whenever carbon emissions are considered as an issue for energy intensive industrials, then switching to nuclear energy would make sense.

## 4. Coupling a NPP to an industrial site

Coupling a NPP to an industrial site is a key point to achieve successful operation. Safety related issues should be carefully analyzed depending on the application requirements. Being interconnected, the operational configurations in between the NPP and the industrial processes should be clearly and safely defined. In any case, the NPP should be located at a minimal safety distance away from the industrial operational units.

A NPP may produce either heat or electrical power or a mixture of both in a cogeneration mode. If the final industrial use requires additional intermediate products like steam, hydrogen, syngas, distillated water or any other chemical compound then intermediate factories will be required on site to provide those products. Each corresponding unit will be fed by heat and electricity coming from the NPP (Figure 5).

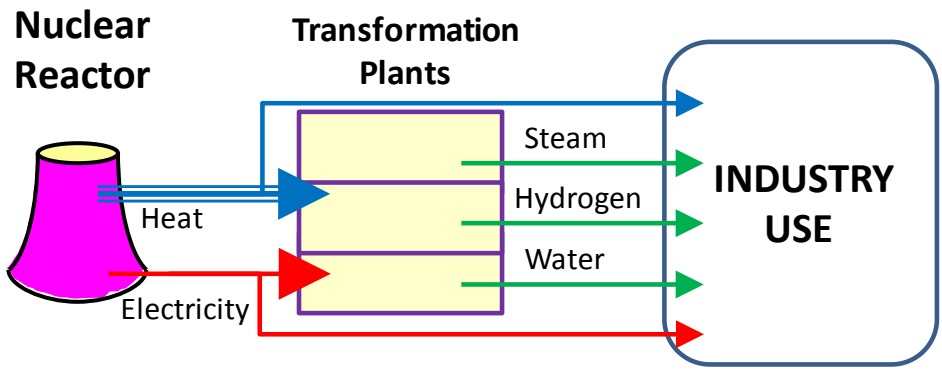


*Figure 5 – General scheme for coupling a nuclear reactor to an industrial site.*

## 5. Application to an oil refinery

One of the most promising applications of FR for industry is the case of oil refinery. The amount of energy involved is 48 Mtoe in Europe and 5.2 Mtoe in France. Actually, it is one of the largest consumption of medium temperature heat in industry. We shall take the Gonfreville-L'Orcher refinery (Figure 7) located in Normandy, France as an example to assess the benefits of implementing a NPP in replacement of a CHP gas plant.

*Figure 6 – Aerial view of Gonfreville refinery in Normandy, France. The CHP gas plant can be seen on the left side at distance from the distillation towers.*

The Gonfreville refinery has an annual capacity of treatment of 15 Mt of crude oil. The refined products supply 15% of the French market through pipelines delivering the Paris area. Crude oil heated at 385°C in distillation units is cracked in hydrocarbon products. The required heat demand in some main process units is estimated in Table 1.

| Main unit | Heat demand (kWh/t) | Temperature (°C) |
|---|---|---|
| Crude Distillation Unit (CDU) | 200 | 385 |
| Vacuum Distillation Unit (VDU) | 90 | 390-450 |
| Hydrotreater | 170 | 280-430 |
| FCC (Fluid Catalytic Cracker) | 20 | 20-540 |
| Hydrocracker | 30 | 290-400 |
| Gasoline reformer | 100 | 430-540 |
| Coking | 40 | 900 |
| Waste Water cleaning | 1 | 20-60 |

*Table 1 – Estimated values of heat demand and the corresponding steam temperatures in some units of the refinery.*

| | | |
|---|---|---|
| **Electricity** | **117** | kWhe/t |
| **Heat** | **551** | kWh/t |
| **Hydrogen** | **5.9** | kg/t |
| **Water** | **1.0** | m3/t |

*Table 2 – The refinery energy needs per ton of crude oil input.*

The overall energy needs of the refinery are summarized in Table 2.

## 6. A matching Fast Reactor

A Sodium Fast Reactor (SFR) is perfectly convenient to deliver the steam required by the refinery at the right temperature of 430°C. A preliminary conceptual design shows that the overall core thermal power should be in the order of 1500 MW. This happens to be quite precisely the size of the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) project [**7**,**8**]. This new FR concept is under design in France with a plan to start construction around 2017. The hot part of the sodium cooled intermediate circuit being at temperatures above 500°C, the required steam production for the refinery could be easily obtained. One may even consider replacing the steam methane reforming plant of the factory by a more recent technology for hydrogen production, using for example medium temperature water electrolysis around 500°C. A desalination/deionization plant may also provide the needed industrial water input for the refinery. To summarize, settling a SFR reactor quite similar in size to the ASTRID project close by Gonfreville would be sufficient to deliver all the required commodities for the oil refinery plant, primarily electricity and heat, but eventually in addition hydrogen and water (Table 2).

| | | |
|---|---|---|
| **Electric power** | **250** | MWhe |
| **Steam at 400°C** | **450** | t/h |
| **Hydrogen** | **13** | t/h |
| **Water** | **2300** | m3/h |

*Table 3 – A 1500 MW thermal SFR will be able to simultaneously deliver all these commodities outputs to feed the oil refinery.*

## 7. Economic benefits

The main advantage obtained from replacing a gas CHP plant by a nuclear reactor is savings in energetic fuels. Today, up to 10% of the energetic contents of the output fuels are utilized to produce the own energy consumption of an oil refinery. Therefore, out of the 15 Mtons of oil processed every year, only 13.5 Mt are finally sold to the market as refined products. Going nuclear would either enhance the value of the final output products and/or require less crude oil inputs to generate the same yield of products. At an oil barrel price of 100$/bbl, the estimated annual economical profit could turn out to be in the billion Euros ballpark. That would certainly make the energy switch from gas to nuclear attractive enough to overweigh the heavy initial investment in such a new fast reactor. In addition, the refinery owner would almost completely wipe out the 4 Mt of GHG emissions his factory is releasing every year. Even though the carbon price is nowadays quite low and the corresponding economic incentives not appealing, that would offer the oil company a social credit showing a voluntary involvement in fighting climate change.

## 8. Conclusion

Today, most of nuclear reactors worldwide are only used for electricity production while many industrial energy needs are still in the form of heat. Standard LWR could provide heat at temperatures below 250°C. However, medium heat temperatures are only accessible using either fast reactors or high temperature reactors. New and safer SFR designs are almost readily available and may cover all industrial uses below 500°C. One example has been reported showing the benefits upon replacing a standard gas plant by a nuclear FR at an oil refinery site in France.

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# SELECTION OF INITIAL FUEL COMPOSITION FOR THE ESFR CORE BASED ON THE KNOWLEDGE OF ITS EQUILIBRIUM CLOSED CYCLE PARAMETERS

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**Abstract.** Advanced sodium cooled fast reactor should be capable to recycle its own fuel and thus to work in a closed fuel cycle. The aim of this paper was to propose initial fuel composition, which will provide smooth and fast transition to the equilibrium closed cycle. For the transition simulation, the ERANOS based EQL3D procedure was used, assuming that all the actinides are recycled, the cooling and reprocessing time is the same as the residence time, and that the removed fission products are replaced by depleted uranium. The optimized fuel was based on LWR spent fuel, or actually on a mixture of its U, Pu, Np, and Am vectors. Isotopic enrichment of these vectors was excluded. The proposed initial fuel provided relatively smooth reactivity evolution during the transition. Nevertheless, in the first several batches it was influenced by the fission products build-up, which has also impact on the safety related parameters. The short term reactivity and safety related parameters evolution were not the objective of the optimization. However, the proposed method can be, in the future, extended also for these purposes.

## 1. Introduction

The fast reactors proposed by the Generation-IV International Forum [1] should be capable to utilize $^{238}$U as a resource for energy production. This feature is enabled by the suitable fission and capture cross-sections of the actinides in a fast spectrum. In comparison with the Light Water Reactors (LWR), in fast reactors $^{239}$Pu can be produced through the neutron capture on $^{238}$U without generating too many higher Pu isotopes and Minor Actinides (MA). Accordingly, these reactor can be operated in a closed cycle with Pu and potentially also MA multi-recycled. The repetitive recycling with constant reprocessing and refuelling scheme may in long-term lead to an equilibrium state. It is represented by an iso-breeding state, where only the depleted uranium is used as a feed replacing the fission products and reprocessing losses. In general, a balance between $^{239}$Pu production and destruction roughly represents such an iso-breeding or break-even state. The main $^{239}$Pu production path is the above-mentioned neutron capture on $^{238}$U and the main destruction paths are its fission and neutron capture. In the fast iso-breeding reactor, this balance is typically established for $^{239}$Pu/$^{238}$U mass ratios around 0.125, which is more or less independent from the coolant type [2]. Since the initial fuel composition usually relies on the Pu vector from spent LWR fuel, it differs from the equilibrium (EQL) fuel vector (see right Table 1.). Accordingly, the transition to equilibrium may introduce strong variations of the safety and performance parameters. The aim of this paper is to analyse these variations and to propose an initial fuel vector, which provides smooth and fast reactivity transition to the equilibrium closed cycle. The applied tools are the ERANOS [3] based EQL3D procedure [4] and a dedicated MATLAB script. This study represents a summary of a master thesis [5] completed at the Paul Scherrer Institut by V. Brankov in the frame of EPFL ETHZ Joint MSc. in Nuclear Engineering in July 2012.

## 2. ESFR core description and computational tools

### *2.1. ESFR core description*

The analysed sodium cooled system is based on recent reactor designs prepared in the frame of European project CP ESFR [6]. The core design [7] is based on pin-type fuel with U-Pu oxide pellets

in ODS steel cladding. A steel wrapper surrounds each assembly. The initial $^{239}Pu/^{238}U$ ratio is proposed around 0.9 what is below the equilibrium value 0.125, thus the core acts as a slight breeder or iso-breeder, in dependence on the selected recycling strategy. The nominal power is 3600 $MW_{th}$. The core design includes description of the safety and control rods. However, in the calculations reported here, these were placed in the upper parking position. The foreseen fuel residence time in effective full power days (EFPD) and the number and length of batches are 2050 = 5×410. For reshuffling and refuelling 30 days are envisaged. The total cycle length, time between two beginnings of cycle, is thus 5×(410 + 30) = 2200 for the ESFR specific five-batch cycle. This time is assumed for the long-term cooling and reprocessing of the unloaded fuel. The effective full power years (EFPY) unit or the actual cycle number are used to present the results.

### *2.2. EQL3D procedure*

An equilibrium cycle procedure for fast reactors EQL3D, based on ERANOS code, has been applied in this paper to obtain the closed cycle equilibrium state and to simulate the evolution towards it. Generally, the EQL3D procedure can be used [2, 8, 9] to yield the description of two basic situations: i) the equilibrium of an open fuel cycle (result of periodic operation with fresh fuel without any recycling) and ii) the equilibrium of a closed fuel cycle (the asymptotic state resulting from the operation with a fixed fuel management scheme involving recycling of the reactor's own fuel). In both cases, the explicit cycle-by-cycle reactor operation under specified periodic fuel management is simulated until the equilibrium state is reached. The EQL3D version applied in this study is based on the ERANOS 2.2 code, used in conjunction with the JEFF3.1 data library.

The fundamental assumption of the EQL3D methodology is to keep the actinides mass in the manufactured fuel constant. In this study, 100% of all actinides (including uranium) are recycled; the reprocessing losses are set to zero. Another assumption is the imposed constant core power and periodic fuel management scheme. In practical terms, this means that the fuel assembly path through each multi-batch cycle is the same. Imposing these conditions, the cycle-by-cycle simulation of the reactor represents the convergence path, which is the objective of the optimization. The corresponding equilibrium cycle is an inherent core characteristic, which does not depend on the initial fuel composition. It is a consequence of the regular periodic operation with fixed fuel management, where the reloading pattern, cooling time, reprocessing losses, feed composition, and the mass of actinides in manufactured fuel are constant.

## 3. Methodology

### *3.1. Fuel Composition Constraints*

It was assumed that the LWR spent fuel can be reprocessed and that its U, Pu, Am, and Np vectors can be partitioned. Utilization of these components, without isotopic enrichment, represents a constraint for the initial ESFR fuel composition. To estimate these vectors, the spent fuel production by Swiss nuclear power plant Betznau (KKB) during its entire past and planned future operation was analysed. The fuel produced by both reactor units of this power plant has been stored for additional 30 years before reprocessing. The obtained composition corresponds to relatively old spent fuel vector; majority of $^{241}Pu$ is already decayed to $^{241}Am$ (see Table 1.). The $^{239}Pu$ share in the vectors corresponds to the historical evolution of the KKB burn-up level from 35 up to 60 GWd/t. The comparison with design specific ESFR fuel compositions show, that the ESFR vector includes less $^{241}Am$ and more $^{240}Pu$. Accordingly, the ESFR composition corresponds to LWR fuel with higher average burn-up and shorter cooling time before reprocessing or Am removal before manufacturing. The maximum available absolute and relative masses of these vectors in the spent fuel were reflected in the proposed initial fuel mixture. Nevertheless, the absolute amount of Pu in the KKB spent fuel is not sufficient even for the first fuel loading of the ESFR core. It corresponds roughly only to ¾ of the necessary amount.

Table 1. Isotopic vector composition of the spent LWR KKB fuel (left), share of each vector in the fuel (middle), and comparison of the original CP ESFR Pu vector with the corresponding spent KKB LWR vector and the equilibrium closed cycle vector (right).

| KKB vector compositions in % | | | |
|---|---|---|---|
| $^{234}$U | 0.02 | $^{238}$Pu | 2.31 |
| $^{235}$U | 0.81 | $^{239}$Pu | 59.56 |
| $^{236}$U | 0.49 | $^{240}$Pu | 27.22 |
| $^{238}$U | 98.69 | $^{241}$Pu | 3.08 |
| $^{241}$Am | 85.81 | $^{242}$Pu | 7.84 |
| $^{243}$Am | 14.19 | $^{237}$Np | 100.00 |

| Vectors share % | |
|---|---|
| U | 98.85 |
| Pu | 0.94 |
| Np | 0.06 |
| Am | 0.15 |

| | ESFR | KKB | EQL |
|---|---|---|---|
| $^{238}$Pu | 3.57 | 2.04 | 2.09 |
| $^{239}$Pu | 47.39 | 52.44 | 53.94 |
| $^{240}$Pu | 29.66 | 23.97 | 34.13 |
| $^{241}$Pu | 8.23 | 2.71 | 4.22 |
| $^{242}$Pu | 10.37 | 6.90 | 3.46 |
| $^{241}$Am | 0.78 | 11.94 | 2.16 |

## *3.2. Equilibrium cycle properties and half-time for rebuilding*

The equilibrium cycle properties were analysed in the sense of the mean transmutation paths for the isotopes evolution. All production and destruction reactions for each isotope were enumerated. Furthermore, for each isotope also the reactivity importance and half-times for rebuilding were analysed. These half-times represent a time necessary for given isotope to rebuild half of its original concentration after it was fully removed from the equilibrium core, whereas the concentration of all other isotopes stayed constant. The results are summarized in Fig. 1. As it can be seen almost all isotopes depends mainly on the previous element of the chain. Nevertheless, a strong path from $^{241}$Am to $^{238}$Pu through the neutron capture, $\beta^-$ decay and final $\alpha$ decay of $^{242}$Cm can be identified. Similarly as $^{241}$Am, also $^{237}$Np is predecessor of $^{238}$Pu. Accordingly, the function of Am and Np vectors seems to be similar. Their negative reactivity worth and the impact to safety parameters may, however, differ. Compared to $^{237}$Np, the $^{241}$Am presence also accelerate the Cm isotopes build-up.

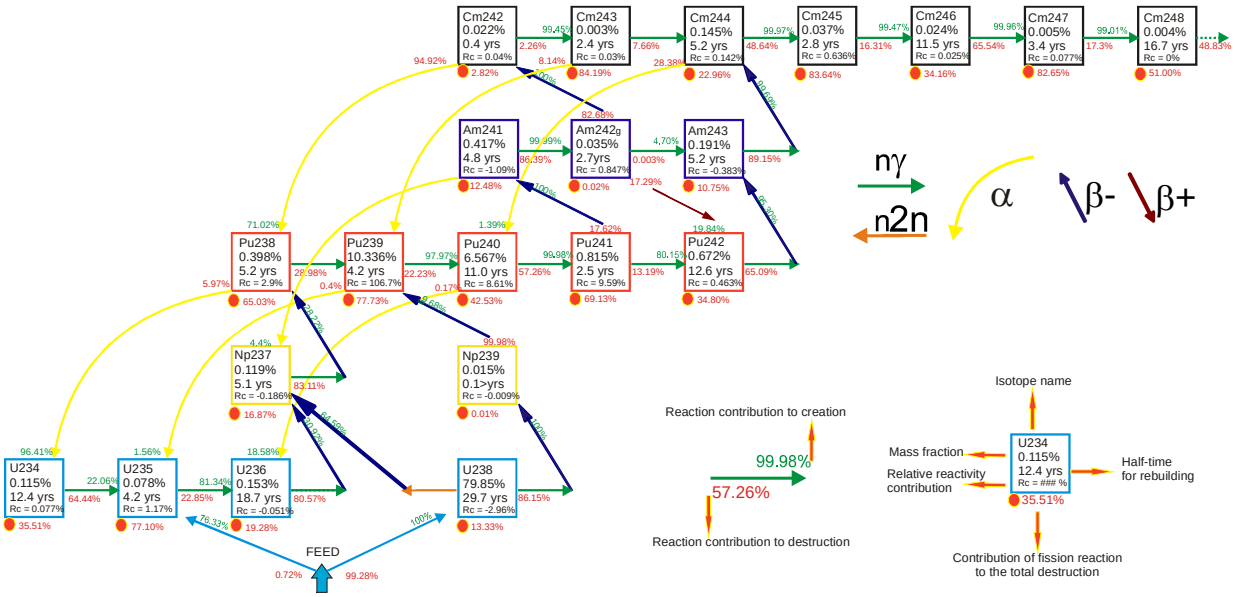


*FIG. 1. Isotope evolution chain for the ESFR core operating at equilibrium.*

Since, the equilibrium vector is unique (for given core geometry, feed, and reprocessing schemes), the figure is independent of all initial fuel vectors. The goal of this study is to propose initial fuel vector that achieves faster convergence to equilibrium and the understanding of these relationships is necessary. The half-times can be used as a characteristic constant for the elements. For instance $^{242}$Pu has particularly large half-time, which is explained by the fact that conversion path from $^{241}$Pu represent only 13%. The key variables for the Cm series are $^{241}$Am and $^{243}$Am. Similarly, the Am vector depends on $^{241}$Pu and $^{242}$Pu, but also on the initial Am load.

The $^{239}Pu/^{238}U$ ratio tends to converge to 0.125. Accordingly, the initial ratio should be close to this value. On the other hand, higher Pu content in the initial core may accelerate higher isotope build-up. In the same time, it can increase the initial reactivity. Advantage of the Np and Am vectors is, that they can reduce the initial reactivity and accelerate $^{238}Pu$, and in case of Am also Cm, build-up.

### *3.3. Criteria for optimization*

Based on a fast scoping study accomplished by MATLAB script, relying on the cross-section provided by ERANOS code, the above described dependencies and characteristics of each vector were analyzed. The script enabled time efficient simulation of reactivity and fuel composition evolution towards the equilibrium for many different initial fuel compositions. In all these calculation, only the one-batch approximation of the cycle was assumed.

The main disadvantage of the MATLAB script is that the cross-sections provided by ERANOS are either for fresh or equilibrium fuel. Accordingly, the results are accurate only in the given area. Nevertheless, the gained understanding helped to propose two simple criteria, which provided the base for the optimization method.

#### *3.3.1. Criterion 1*

The first criterion requires symmetric reactivity swing for the initial cycle around the mean equilibrium reactivity. The mean value is obtained by averaging the equilibrium beginning of cycle (BQC) and end of cycle (EQC) reactivities. Mathematically it can be easily expressed as follows:

$$\frac{\rho_1 + \rho_2}{2} = \frac{\rho_{BQC} + \rho_{EQC}}{2},$$

where $\rho_1$ and $\rho_2$ represent reactivity at the beginning (BOC1) and end (EOC1) of the first cycle. Fulfilling this relation should be favourable for both single and multi-batch schemes. The criterion is graphically illustrated in Fig. 2.

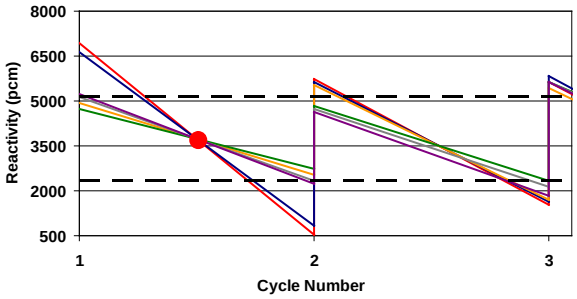


*FIG. 2. Illustrative reactivity evolutions fulfilling the criteria 1.*
*The dashed lines represent the BQC and EQC values.*

#### *3.3.2. Criterion 2*

The second criterion requires that the reactivity at the beginning of third cycle (BOC3) is equal to the BQC value. The requirement may assure relatively small deviation of the evolving reactivity from the equilibrium level; because the major changes in the fuel composition happens already during the first two cycles. The criterion is graphically illustrated in left Fig. 3. The Mathematical expression is trivial:

$$\rho_3 = \rho_{BQC}$$

Combination of criteria 1 and 2 leads to a reactivity evolution, which is illustrated in the right Fig. 3.

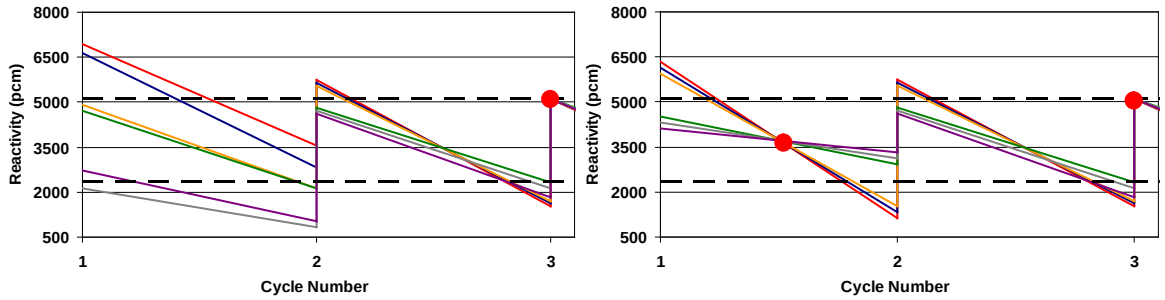


*FIG. 3. Illustrative reactivity evolutions fulfilling the criteria 2 (left) and both criteria 1 and 2 (right). The dashed lines represent the BQC and EQC values.*

### *3.3.3. Criterion 3*

In case that there is more than one solution fulfilling both criteria, then the one with minimal reactivity swing during the first cycle should be chosen. Nevertheless, the observed differences between the found solutions are so small that the importance of this criterion is limited. It may be replaced in the future with additional requirements, e.g., on the safety related parameters.

## *3.4. Approach for fulfiling the criteria*

To fulfill the two criteria defined above, the impact of each Pu, Am, and Np vector variation to the reactivity evolution was enumerated. Three necessary reactivity points were selected for this purpose: beginning and end of the first cycle BOC1 and EOC1 and beginning of the third cycle BOC3. The impact of each particular vector variation to these three points was enumerated in a form of reactivity coefficients (reactivity/kg). Illustration of the three selected reactivity points can be seen in Fig. 4.

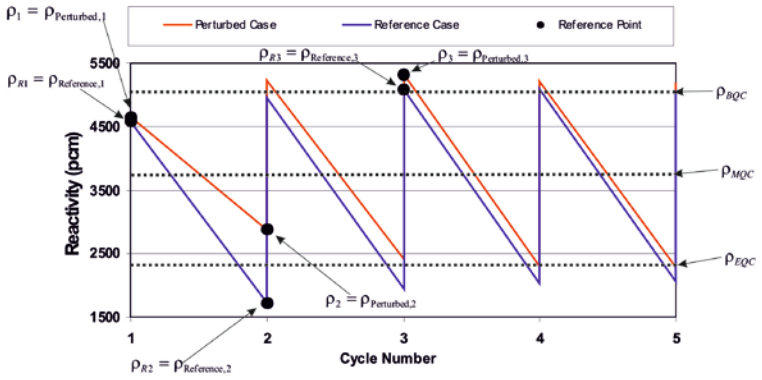


*FIG. 4. Illustration of the three selected reactivity points for the reference and perturbed case.*

The reactivity coefficients are calculated using nominal and perturbed simulations as follows:

$$C_1^i = \frac{\Delta\rho_1^i}{\Delta m^i} = \frac{\rho_1^i - \rho_{R1}^i}{m_{perturbed}^i - m_{reference}^i}, \quad C_2^i = \frac{\Delta\rho_2^i}{\Delta m^i} = \frac{\left(\rho_1^i - \rho_2^i\right) - \left(\rho_{R1}^i - \rho_{R2}^i\right)}{m_{perturbed}^i - m_{reference}^i}, \quad C_3^i = \frac{\Delta\rho_3^i}{\Delta m^i} = \frac{\left(\rho_1^i - \rho_3^i\right) - \left(\rho_{R1}^i - \rho_{R3}^i\right)}{m_{perturbed}^i - m_{reference}^i},$$

where index *i* stands for *Pu*, *Am*, or *Np*. The first coefficient represents the change of BOC1 reactivity per kg of the selected vector. The second coefficient represent the reactivity swing change Δ(BOC1-EOC1) per kg of the selected vector. And finally the third coefficient represent the change of reactivity difference Δ(BOC1-BOC3) per kg of the selected vector. These coefficients were evaluated by both MATLAB script for the scoping study and by ERANOS code for more exact application. As it can be seen from Fig. 5 these coefficients also depends on the absolute mass of selected vector in the fuel.

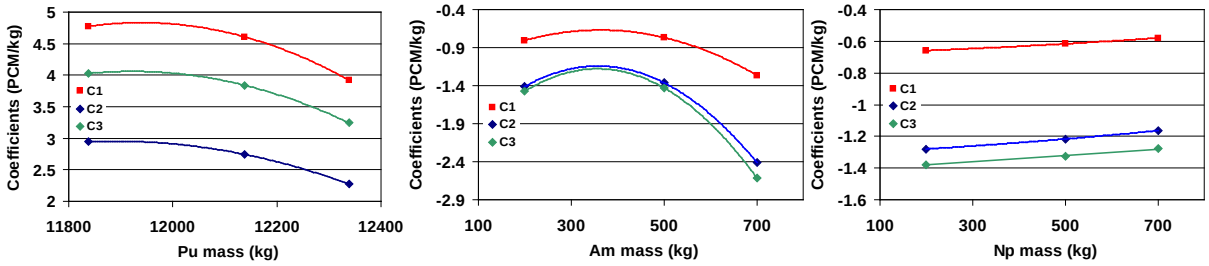


*FIG. 5. Enumeration of the three coefficients by the ERANOS code for the KKB Pu (left), Am (middle), and Np (right) vectors.*

The dependency of the coefficients on the absolute mass, especially for Am vector, is strongly nonlinear. Nevertheless, in the method applied here only the average values presented in Table 2. are used for the optimization. If a higher accuracy is required, the coefficient may be expressed as a polynomial function of the mass and the respective reactivity can be expressed as a mass integral from these functions. The resulting set of equation will then change form linear to integral form.

Table 2. Coefficients for the Pu, Am and Np vectors based on the KKB and ESFR compositions.

| KKB | $C_1$ (PCM/kg) | $C_2$ (PCM/kg) | $C_3$ (PCM/kg) |
|---|---|---|---|
| Pu | 4.459 | 2.666 | 3.724 |
| Am | -0.920 | -1.677 | -1.782 |
| NP | -0.618 | -1.219 | -1.326 |

| ESFR | $C_1$ (PCM/kg) | $C_2$ (PCM/kg) | $C_3$ (PCM/kg) |
|---|---|---|---|
| Pu | 4.397 | 2.618 | 3.764 |
| Am | -1.052 | -1.858 | -1.754 |
| NP | -0.900 | -1.617 | -1.641 |

The average coefficients presented above are used to express the reactivity for the three points of interest (BOC1, EOC1, and BOC3) in the following way:

$$\rho_1 = \rho_{R1} + C_1^{Pu}\Delta m^{Pu} + C_1^{Am}\Delta m^{Am} + C_1^{Np}\Delta m^{Np}$$
$$\rho_2 = \rho_1 - (\rho_{R1} - \rho_{R2}) - C_2^{Pu}\Delta m^{Pu} - C_2^{Am}\Delta m^{Am} - C_2^{Np}\Delta m^{Np}$$
$$\rho_3 = \rho_1 - (\rho_{R1} - \rho_{R3}) - C_3^{Pu}\Delta m^{Pu} - C_3^{Am}\Delta m^{Am} - C_3^{Np}\Delta m^{Np}$$

These expressions relate the reactivity for selected points with the mass of Pu, Am and Np vectors. The variables $\Delta m^{Pu}$, $\Delta m^{Am}$, and $\Delta m^{Np}$ represent the mass difference from the reference case for the three respective vectors. By substituting these expressions into the two criteria a final set of two equations for three variables is obtained. After simplifications it writes as follows:

$$\Delta m^{Pu} = \frac{\rho_{BQC} + \rho_{EQC} - (\rho_{R1} + \rho_{R2}) - (2C_1^{Am} - C_2^{Am})\Delta m^{Am} - (2C_1^{Np} - C_2^{Np})\Delta m^{Np}}{2C_1^{Pu} - C_2^{Pu}}$$
$$\Delta m^{Pu} = \frac{\rho_{BQC} - \rho_{R3} - (C_1^{Am} - C_3^{Am})\Delta m^{Am} - (C_1^{Np} - C_3^{Np})\Delta m^{Np}}{C_1^{Pu} - C_3^{Pu}}$$

## 4. Results

### *4.1. Optimized fuel composition*

Since there are only two equations for three variables to be solved, there exists infinite number of solutions. Therefore, the Np mass was selected as a parameter which was varied from 0 to 700 kg. From these solutions, three cases were selected and tested with the EQL3D procedure. These cases are presented in Table 3 for both KKB and ESFR available vector sets.

Table 3. Coefficients for the Pu, Am and Np vectors based on the KKB and ESFR compositions.

| KKB | Pu (kg) | Am (kg) | Np (kg) |
|---|---|---|---|
| Solution 1 | 11463 | 1157 | 50 |
| Solution 2 | 11459 | 955 | 300 |
| Solution 3 | 11454 | 754 | 550 |

| ESFR | Pu (kg) | Am (kg) | Np (kg) |
|---|---|---|---|
| Solution 1 | 11842 | 1500 | 50 |
| Solution 2 | 11839 | 1240 | 300 |
| Solution 3 | 11836 | 979 | 550 |

The corresponding reactivity evolutions calculated by the ERANOS based EQL3D procedure for these three solutions and two different vector sets are shown in Fig. 6. It can be seen both of the criteria are relatively well fulfilled. Furthermore, the results for each three cases are similar; therefore, the solution 3, with highest Np share, was selected for further refined multi-batch simulation.

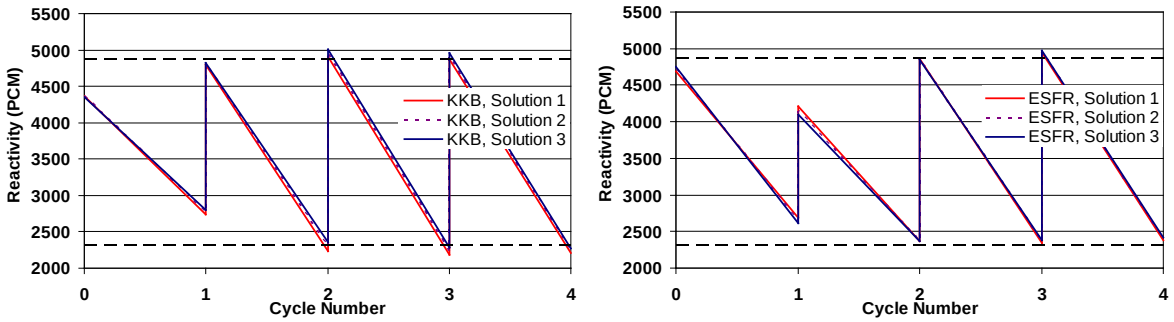


*FIG. 6. Reactivity evolution for three selected solutions and for the KKB vectors (left) and ESFR vectors (right) obtained by the ERANOS based EQL3D procedure.*

The evolution of nominal and optimized cases, based on the solution 3 for KKB and ESFR vectors, using the one-batch approximation and multi-batch simulation is shown in Fig. 7. Both optimized cases provide smooth and flat reactivity evolution towards the equilibrium. Even the initial fission product build-up in the multi-batch case, which was not the objective of the optimization, introduces only limited perturbation. Nevertheless, the optimized multi-batch results present slight overshooting of the reactivity. This may be avoided, if the method would be directly based on multi-batch simulation and not only on the one-batch approximation.

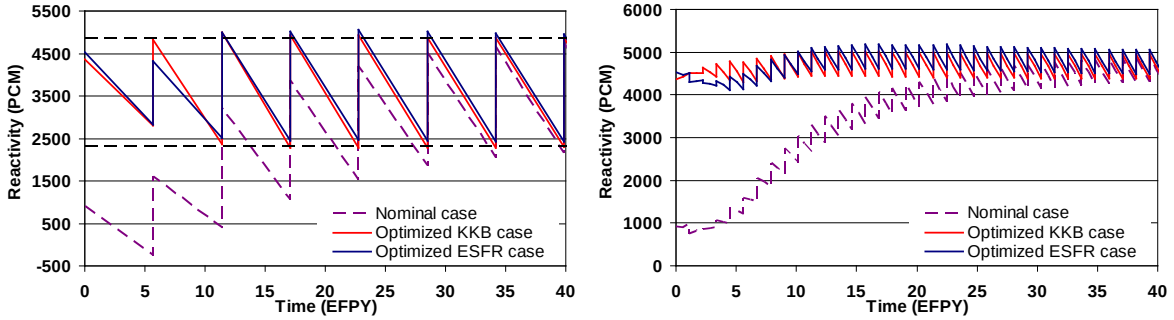


*FIG. 7. Reactivity evolution for the nominal and optimized cases, based on the solution 3 for KKB and ESFR vectors, using one-batch approximation (left) and multi-batch simulation (right).*

### *4.2. Safety related parameters*

The safety related parameters have not been included in the optimization criteria. Nevertheless, their evolution for nominal and optimized cases was also simulated. As seen in left Fig. 8 the Doppler constant, $Dc = \Delta\rho / \Delta\ln(T)$, evolution is relatively smooth. On the other hand the void reactivity, for active core voiding, presents stronger initial increase (right Fig. 8), which is, however, smaller than for the nominal case. For both discussed safety related parameters small overshooting, that peaks around cycle 7 can be seen. This may represent a safety related issue and will probably require that the future optimizations include also the coefficients based on safety parameters.

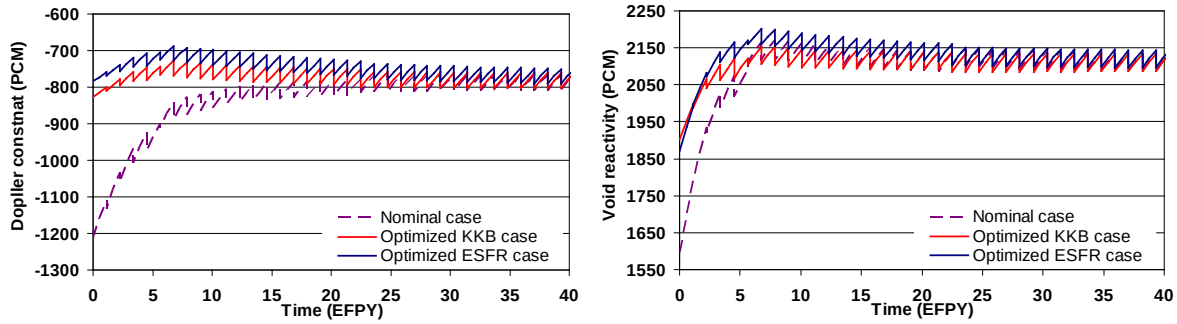


*FIG. 8. Doppler constant (left) and void reactivity (right) evolution for the nominal and optimized cases, based on the solution 3 for KKB and ESFR vectors*

### *4.3. Reactivity level reduction*

The reactivity evolution for optimized cases presented in Fig. 7. is smooth. However, the absolute reactivity level is relatively high. Since the equilibrium state does not depend on the initial fuel composition, the methods for its reduction are constricted. The equilibrium cycle and so the reactivity depends only on the core design, reloading pattern, cooling time, reprocessing losses, feed composition, and the mass of actinides. The easiest way for the reactivity reduction is through the fuel density. The above-derived equation for BOC1 reactivity can be used to estimate the density reduction factor *w*. The equation should be modified to the following form:

$$\rho_1 = \rho_{R1} + C_1^{Pu}\left(\left(m_{reference}^{Pu} + \Delta m^{Pu}\right)w - m_{reference}^{Pu}\right) + C_1^{Am}\left(\left(m_{reference}^{Am} + \Delta m^{Am}\right)w - m_{reference}^{Am}\right) + C_1^{Np}\left(\left(m_{reference}^{Am} + \Delta m^{Am}\right)w - m_{reference}^{Am}\right)$$

The resulting *w*=0.82 represent density reduction by 12%. The respective results are compared with the original evolutions in Fig. 9. As it can be seen, the reactivity level was reduced; however, the evolution curve is still smooth as before. Nevertheless, this method may have some drawbacks for the safety related parameters. Other solution for the reactivity lowering may be the core height or assembly number reduction, which may even improve the safety parameters. On the other hand, it may also introduce difficulties for thermal-hydraulic design of the core. Finally yet importantly, the reactivity may be reduced by simulating artificial or realistic reprocessing losses, which are assumed zero in this study or by many other solutions.

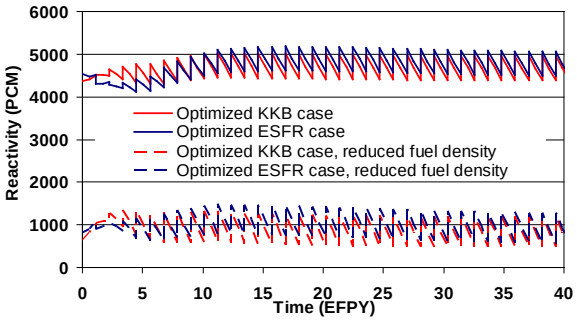


*FIG. 9. Reactivity evolution for the nominal and reduced fuel density based on the solution 3 for KKB and ESFR vectors and using multi-batch simulation.*

## 5. Summary and conclusion

In this paper, the initial fuel composition for the sodium cooled ESFR reactor was proposed. This composition assured smooth and fast reactivity evolution towards the equilibrium closed cycle. The transition was simulated by ERANOS based EQL3D procedure and based on assumptions that all the actinides are recycled, the cooling and reprocessing time is the same as the residence time, and the

removed fission products are replaced by depleted uranium. The composition was optimized using two simple criteria and utilizing LWR spent fuel U, Pu, Np, and Am vectors, without isotopic enrichment. The resulting reactivity evolution is smooth and converges swiftly to the closed cycle equilibrium. Nevertheless, in the first several batches it is influenced by the fission products build-up. Furthermore, its absolute level is relatively high. To demonstrate one of the possibilities for final reactivity adjustment the fuel density reduction was simulated.

The proposed optimization method, and so the resulting initial fuel composition, can be improved in the future. Especially, the more precise multi-batch simulations can be used to derive the reactivity coefficient. Furthermore, the safety related parameters may became one of the goals of the optimization method.

## ACKNOWLEDGEMENTS

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# Scenarios for Fast Reactors Deployment with Plutonium Recycling

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**Abstract.** Plutonium-management is crucial towards sustainability of nuclear energy and demands a progressive recycling policy starting in LWRs, by use of MOX-fuel, up to the multi-recycling of Pu in Fast Reactors, such as Sodium Fast Reactors (SFRs). This paper focuses on the transition where SFRs perform a multi-recycling of Pu. In order to sketch a few of the possible futures towards sustainability, illustrative scenarios for SFR deployment in progressive replacement of the French PWR fleet with different chronologies (from 2040 or 2080) were analyzed in the context of the French 2006 Law. The sensitivity of scenarios to the SFR core design is evaluated by considering a homogeneous core (SFR V2B) or a new heterogeneous core with a significant gain on sodium void effect (CFV). Scenarios of symbiotic fleet, in which a 60 GWe PWR fleet is maintained while SFRs are progressively introduced from 2040 depending on the plutonium produced by the PWR fleet, are also envisaged. The first evaluation focuses on the maximal achievable installed SFR power using plutonium from PWR spent fuel. Then, different key drivers to modulate SFR deployment are presented (PWR and SFR spent fuel cooling time, reprocessing capacities). These studies allow to identify the minimal installed SFR power in function of the Pu yearly produced by PWR fleet.

## 1. INTRODUCTION

For the future, the long term sustainable nuclear systems would include fast reactors which allow full use of uranium with no enrichment needs, efficient recycling of plutonium and potentialities for improving waste management. SFRs offer the flexibility to adjust the Pu-balance in nuclear energy systems composed of LWRs and SFRs. A variety of scenarios for the introduction of SFRs are considered in many countries, ranging from a transition to a full SFR-park in the future to more progressive scenarios where SFRs are deployed allowing to ensure a symbiotic recycling of Pu in LWRs and SFR. In France, these SFRs will be initially fuelled with plutonium coming from spent MOX fuels, the breeding gain being adjusted according to energy needs, Pu-balance and isotopics needs.

In this context, this paper describes the results of some scenarios considering the deployment of SFRs replacing gradually the French PWR fleet. The results are assessed with the simulation COSI code (some results have already been detailed in [1]), the objective being to investigate the Pu-balance for such scenarios taking into account the latest characteristics of SFRs, as developed in France within the GEN IV reactor program. Different chronologies are considered to evaluate the capability of SFRs, in function of Pu availability, to renew a 60 GWe PWR fleet and the sustainability of this deployment in function of plutonium availability. A first scenario considers the deployment of SFRs in two steps with 20 GWe from 2040 to 2050 and 40 GWe in addition from 2080 to 2100. The impact of a SFR deployment delayed to 2080 with 60 GWe deployed from 2080 to 2110 is assessed in a second

scenario. The sensitivity of scenarios to the SFR core design has been evaluated by considering a homogeneous core (SFR V2B) and a new heterogeneous core with a significant gain on the sodium void effect (CFV).

In addition, scenarios for a symbiotic fleet, in which a 60 GWe PWR fleet is maintained while SFRs are progressively introduced from 2040 depending on the plutonium produced by the PWR fleet, can be envisaged. The first evaluation focuses on the maximal achievable installed SFR power using plutonium from PWR spent fuel. Both cases of PWR fleet with and without MOX fuel are evaluated. Then, different key drivers to modulate the SFR deployment are presented.

## 2. SCENARIOS ASSUMPTIONS

### *2.1. Calculation scheme*

Scenario studies are carried out with the simulation software COSI [2] (Simulation Code for nuclear strategy studies). COSI simulates the evolution of a nuclear reactor fleet and of the associated fuel cycle facilities over a defined period (an order of magnitude of centuries), to analyze from a physical point of view the consequences of the choices made over the fleet, fuel types and fuel cycle process. The evolution calculation is performed by coupling COSI with CESAR [3] (Simplified Evolution Code Applied to Reprocessing). Both COSI and CESAR are developed by the SPRC (Reactor Physics and Fuel Cycle Service) at the CEA Cadarache center. CESAR is the reference code used at the AREVA NC La Hague reprocessing plant to calculate quickly the isotopic composition of irradiated fuel from the initial isotopic composition, the fuel burnup and the irradiation time. CESAR uses neutron data libraries (cross sections sets) supplied by the CEA reference calculation codes for neutron physics: APOLLO2 [4] for thermal spectrum systems and ERANOS2 [5] for fast spectrum systems.

### *2.2. Reactors and fuel cycle assumptions*

In the scenarios considered for the study, the current French fleet, which contains 58 PWR reactors loaded with UOX fuel (one third are loaded with 30% of MOX fuel), phases out progressively from 2020 to 2050. The studied scenarios compare different strategies of evolution of the French PWR fleet, by a mixed fleet of EPR™ reactors and SFRs (the Generation IV reactors are supposed industrially mature from 2040). The lifetime of future reactors is fixed to 60 years.

Natural uranium is enriched to 4.5% from 2015 on for UOX fabrication. The depleted uranium enrichment coming from the enrichment plant is 0.25% until 2019, then 0.2%, and is used for PWR MOX fabrication. Aside recycled uranium re-enriched to feed two 900 MWe PWR units of the PWR fleet, recycled uranium is mostly used with plutonium to produce SFR fuel, to take advantage of the U235 enrichment (close to 0.8%), higher than in depleted uranium, which allows some plutonium saving. The fabrication lead time is fixed to 2 years (including fuel transport until loading in core). The minimum cooling time before reprocessing aimed for all fuel types is 5 years and the effective cooling time is adapted to meet fissile materials need. Facilities capacities (fabrication plant, reprocessing plant) remain steady over several decades and storage capacities for spent fuels or materials separated in reprocessing plant are limited to realistic values determined consistently with current values.

The SFR V2B [6] core concept has been developed by the CEA and the French partners in 2008. A new heterogeneous core named “CFV” [7] (the preliminary version V0 is used) with a significant gain on sodium void effect is also considered (see Figure 1), to evaluate the sensitivity of scenarios studies to SFR core. The Table 1 presents the main neutronic characteristics of EPR and SFR cores. Basically, both SFR V2B and CFV cores have a breeding gain close to zero, but up to two rows of fertile blankets with a depleted $UO_2$ matrix can be added to increase the breeding gain.

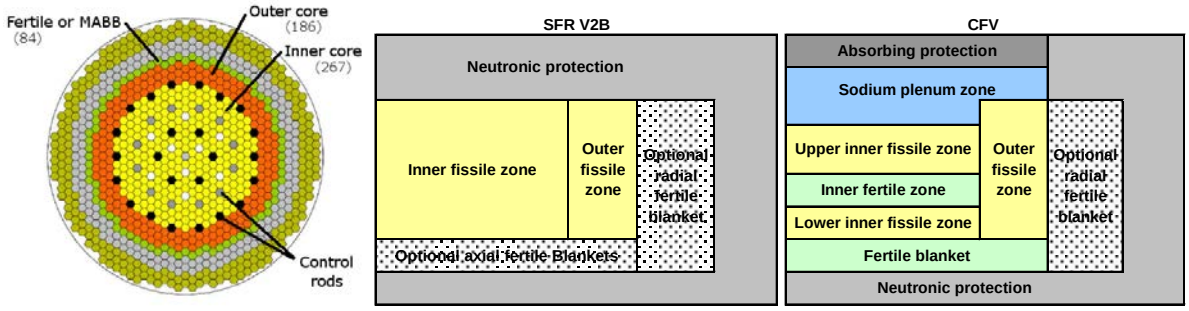


*FIG. 1. Radial geometry of SFR core design, axial geometry of SFR V2B and CFV core design.*

Table 1. Main neutronic characteristics for EPR and SFR cores

| | **EPR™ reactors** | **SFR V2B** | **CFV V0** |
|---|---|---|---|
| Thermal / Net electrical power (MW) | 4500 / 1550 | 3600 / 1450 | 3600 / 1450 |
| Net yield (%) / Load factor (%)[a] | 34.4% / 81.8% | 40.3% / 81.8% | 40.3% / 81.8% |
| Heavy metal mass in core (tons) | 120.0 | Core: 74<br>Axial fert. blankets: 22<br>Radial fert. blank. (2 rows): 41 | Core: 51<br>Axial fert. blankets: 37<br>Radial fert. blank. (2 rows): 44 |
| Core management × irradiation time (EFPD) | 4 × 367 | 5 × 410 | 5 × 400 |
| Fuel type | 17×17 UOX<br>17×17 MOX | MOX (U-Pu) | MOX (U-Pu) |
| Fuel assemblies average burnup (GWd/tons) | 55 | Core: 98<br>Fertile blankets: 10 | Core: 119<br>Axial fertile blankets: 23<br>Radial fert. blank. (2 rows): 16 |
| Equivalent $Pu^{239}$ weight content (wt %) | - | 11.0% | 16.7% |
| Average initial Pu weight content (wt %) | - | ~16.0% | ~24.5% |
| Breeding gain | - | Core: 0.01 to 0.06[b]<br>Axial fertile blankets: +0.11<br>Radial fert. blank. (2 rows): +0.10<br>TOTAL: 0.22 to 0.27 | Core: -0.25 to -0.20[b]<br>Axial fertile blankets: +0.25<br>Radial fert. blank. (2 rows): +0.14<br>TOTAL: 0.15 to 0.20 |

[a] The load factor is defined as the energy production (TWhe/y) divided by the installed power (GWe).

[b] The breeding gain depends on the Pu isotopic composition and increases with Pu recycling.

## 3. RESULTS ON SFR DEPLOYMENT IN REPLACEMENT OF THE FRENCH PWR FLEET

The Figure 2 presents two scenarios of SFR V2B deployment in replacement of the French PWR fleet. In the two cases, the nuclear energy production remains constant to its current level (430 TWhe per year), produced by 63 GWe of current PWR then 60 GWe of EPR™ and SFR, which is made possible by an improvement of the reactors load factor. In the first scenario, SFRs are progressively deployed from 2040 to 2100, with 20 GWe from 2040 to 2050 and an additional 40 GWe from 2080 to 2100. In the second one, SFR deployment is delayed in 2080 and concentrated on 30 years. The number of EPR™ reactors is adjusted to maintain the nuclear energy production. The purpose of these scenarios is to evaluate the capability of SFR to replace PWR and the associated impacts on fuel cycle and the sustainability in terms of fissile material resources. The impact of a change of SFR core is also presented in the paragraph 3.2, by considering the CFV core instead of the SFR V2B core.

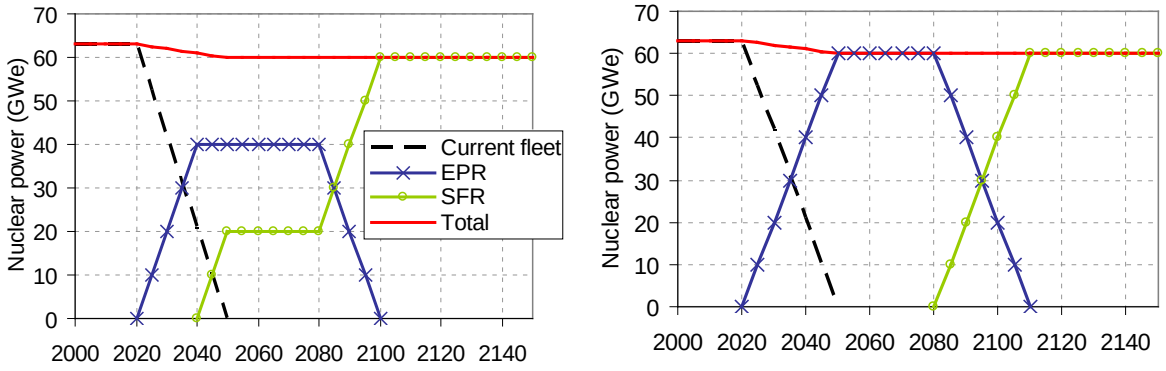


*FIG. 2. Possible evolutions of the French nuclear fleet with SFR deployment from 2040 or 2080*

## *3.1. Impacts of SFR deployment*

### *3.1.1. Fuel cycle facilities capacities*

The fresh fuel fabrication needs for each scenario are shown on Figure 3 and follow the evolution of the nuclear power repartition. The current PWR fleet requires on average 1,000 tons/year of UOX fuels and 100 tons/year of MOX fuels (Pu recycling with MOX in PWR is stopped before SFR deployment), whereas the SFR fleet at equilibrium requires 450 tons/year. The Pu flow in fabrication plants increases in the two scenarios from 10 to more than 70 tons/year.

The cumulated natural uranium consumption to feed the PWR fleet since its commissioning is function of the number of EPR™ reactors:

- 728,000 tons if 40 GWe of EPR™ are deployed;
- 870,000 tons if 60 GWe of EPR™ are deployed.

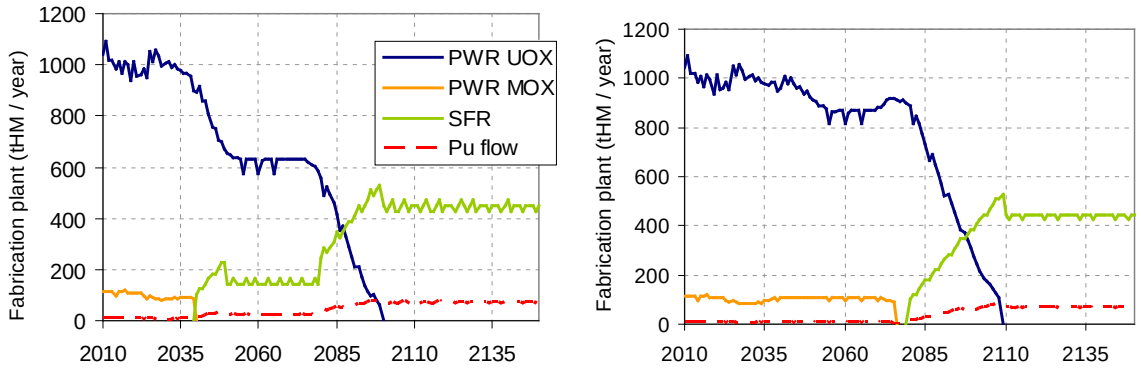


*FIG. 3. Fresh fuel fabrication needs with SFR deployment from 2040 (left) or 2080 (right)*

In order to timely feed the SFRs with separated Pu from PWR fuels, deployment scenarios for the reprocessing plants were considered as in Figure 4 indicating the amount of fuels being reprocessed at various times for the two illustrative scenarios. These fuels reprocessing requirements have been optimized to maintain a steady global capacity over several decades. PWR fuels (UOX and MOX) are reprocessed in priority before SFR fuels. Given that the scenarios imposed a SFR-deployment, and thus consequently an annual Pu-flow from reprocessing plants allowing for SFR-fuel fabrication in time, the reprocessing capacity is obviously directly related to the introduction rate of SFRs which is steeper in the second case with SFRs deployment from 2080 on. As such, the reprocessing capacity for today's PWR UOX and MOX fuelled park suffices in the first SFR deployment case though additional capacity is required temporarily to match the higher SFR-deployment rate in the second scenario.

During the transient period, the capacity of reprocessing plant reaches 1,100 tons / year if the SFR deployment is spread over 60 years from 2040 whereas it reaches 1,600 tons if the SFR deployment in concentrated during 30 years from 2080. Due to the increased production of PWR MOX fuels until 2075 in this last case, the maximal fraction of reprocessed PWR MOX fuels is doubled with SFR deployment from 2080 by comparison with SFR deployment from 2040. The needs are equivalent once the SFR fleet is totally deployed, with 450 tons reprocessed per year, in balance with the fresh fuel fabrication. It is also worth mentioning that the ratio Pu / (U+Pu) increases during the scenarios from 1% to 18%.

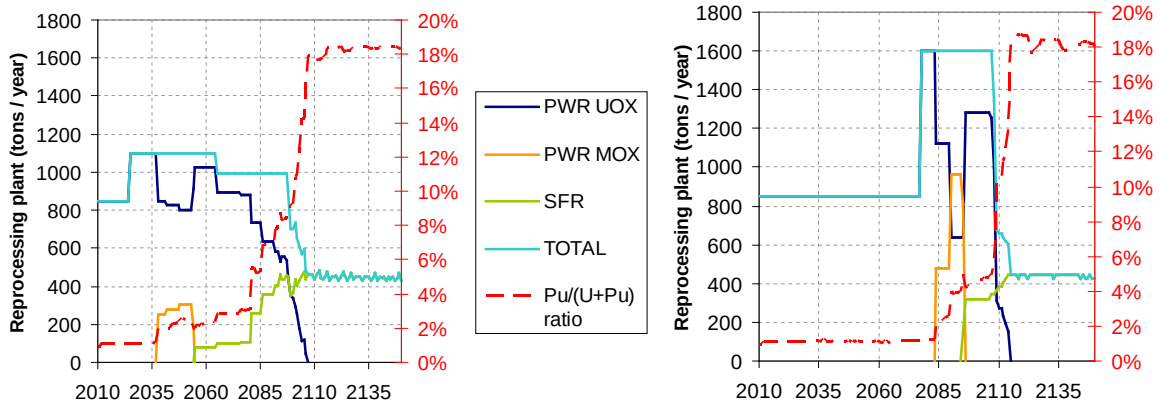


*FIG. 4. Spent fuel reprocessing needs with SFR deployment from 2040 (left) or 2080 (right)*

Accordingly to the reprocessing capacity, the spent fuel storage is displayed on Figure 5. PWR UOX and MOX fuels are accumulated during the operation of PWR fleet. Then, the storage decreases due to fuels reprocessing to produce the Pu required to SFR deployment. The spent fuel storage peaks at 17,700 tons (including 4,000 of PWR MOX fuels) if SFR are deployed in 2040. If the SFR deployment is delayed in 2080, the spent fuel storage peaks at 26,700 tons (including 8,500 tons of PWR MOX fuels). In the two cases, the SFR fuel storage stabilizes at less than 2,000 tons once the SFR fleet is at equilibrium.

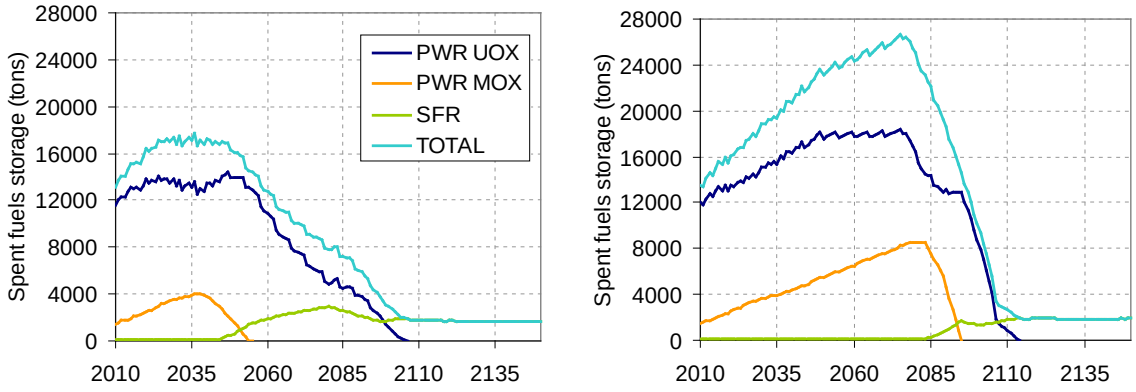


*FIG. 5. Spent fuels storage with SFR deployment from 2040 (left) or 2080 (right)*

### *3.1.2. Pu recycling in SFR V2B*

The Figure 6 presents the evolution of the initial Pu content and the Pu quality of fresh SFR V2B fuel for the scenarios of introduction of SFR in 2040 (full line) or in 2080 (dotted line). The Pu quality is defined as the ratio of odd isotopes in Pu and represents the variability of the isotopic composition of Pu produced by the reprocessing plant. The Pu content is adjusted to compensate the variability of Pu isotopic composition using an equivalence model [8].

The lowest Pu quality comes from PWR MOX fuels reprocessing: 51.7% in case of SFR introduction from 2040 and 46.4% if SFR introduction is delayed in 2080. These values lead to the highest Pu content (17.0 wt% and 18.9 wt%) required in the SFRs. Nevertheless, the impact on reactivity coefficients (sodium void effect, Doppler coefficient) is limited to less than 10%. At the equilibrium, the Pu content is deliberately increased to 16.4% to adjust the breeding gain to zero and to stabilize Pu inventory.

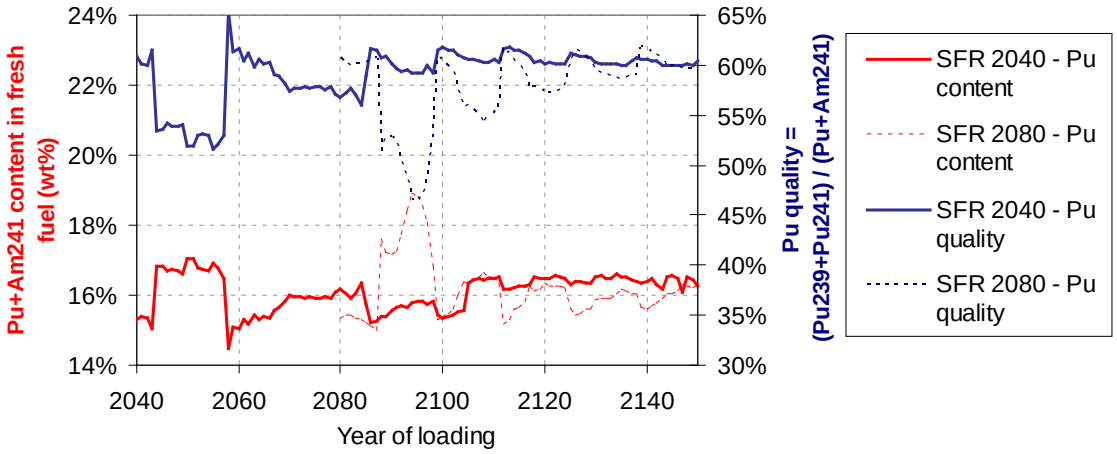


*FIG. 6. Composition of fresh SFR fuel*

### *3.1.3. Sustainability of SFR deployment*

To evaluate the sustainability of SFR deployment in terms of fissile materials availability, the Pu distribution in the fuel cycle must be studied, as displayed on Figure 7 which shows the Pu mass in facilities, in storage (spent fuels and separated materials at reprocessing plant), in fuels irradiated in reactors and in ultimate waste. The final distribution of Pu inventory is similar in both scenarios and is consistent with the transit time of materials at each step: 50% in reactors (almost 7 years of irradiation), 15% in facilities (mainly 2 years of fabrication) and 35% in storage (5 years of cooling).

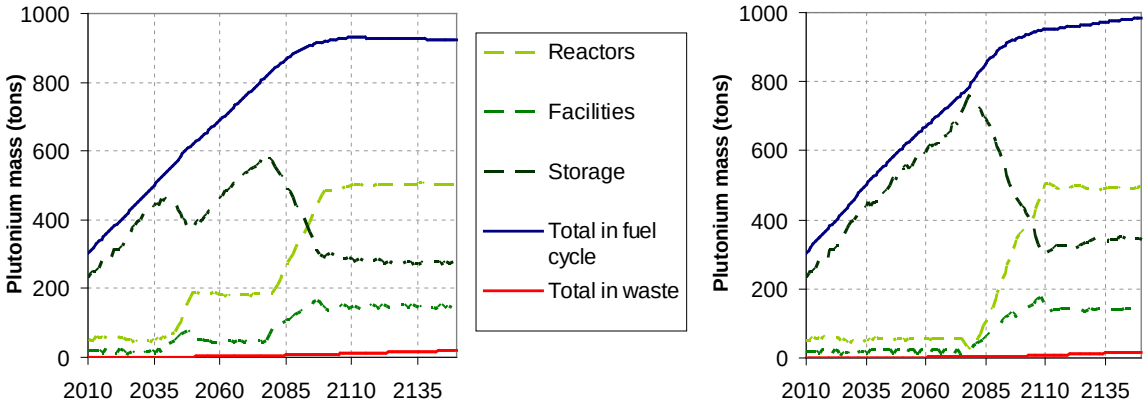


*FIG. 7. Pu inventory with SFR deployment from 2040 (left) or 2080 (right)*

From the distribution of Pu inventory, the Pu margin representing the available Pu to start SFR is defined as the sum of separated Pu and Pu in fuels enough cooled to be reprocessed. A negative value of Pu margin represents a Pu shortage that has to be compensated by an input of external Pu. The evolution of Pu margin and the drivers allowing to adjust the margin to meet SFR deployment requirements are represented on Figure 8. With a minimum cooling time of 5 years for spent fuels, a Pu deficit appears at the end of SFR deployment. The delay to 2080 of SFR introduction reduces this

deficit without removing it completely. Nevertheless, it is possible to maintain a positive Pu margin through SFR deployment:

- by reducing the minimum cooling time (Tc) of SFR spent fuels to less than 5 years from 2080 (3.4 years is SFRs are deployed since 2040 and 4.1 years if SFRs are deployed since 2080);
- by adding axial fertile blankets to increase SFR breeding gain.

To a lesser extent, reducing the use of MOX fuels in PWR before the SFR deployment can be another way to save Pu for SFRs and to limit the constraint on Pu availability.

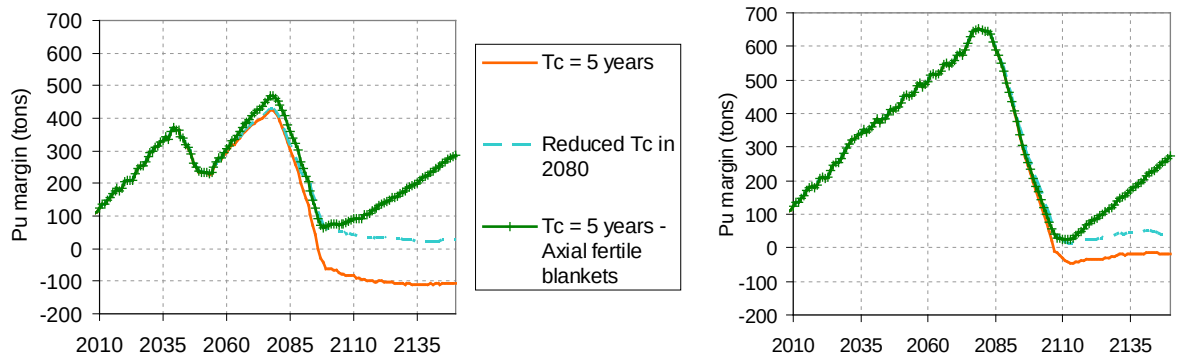


*FIG. 8. Pu margin with SFR deployment from 2040 (left) or 2080 (right)*

### *3.2. Sensitivity to SFR core design*

To evaluate the sensitivity of scenarios results to SFR core, the Table 2 compare the Pu need of SFR V2B and CFV V0 core designs. Two examples of Pu isotopic composition are used. The first one named "Transition" is representative of the transition period of SFR deployment (Pu coming from the reprocessing of PWR fuels) and is independent of the SFR core design. The second one named "Equilibrium" is representative of the equilibrium composition of Pu recycling in SFR (without any adjustment). The Pu quality at equilibrium is higher in CFV than in SFR. The Pu contents in fresh fuel for each Pu composition are not directly comparable, due to the difference of heavy metal in core (74 tons in SFR and 51 tons in CFV). But if the Pu need is expressed in tons per GWe, it is obvious that the Pu need is of the same order of magnitude in the two cases, although with a difference of 5%.

Table 2. Comparison of the performances of SFR V2B and CFV V0

| | | **SFR V2B** | | **CFV V0** | | **Difference (CFV – SFR)** | |
|---|---|---|---|---|---|---|---|
| | | Transition | Equilibrium | Transition | Equilibrium | Transition | Equilibrium |
| Pu isotopic composition | Pu238 | 3.6% | 0.5% | 3.6% | 0.4% | 0% | +4% on Pu239 |
| | Pu239 | 47.8% | 58.9% | 47.8% | 61.6% | | |
| | Pu240 | 29.9% | 33.9% | 29.9% | 32.6% | | |
| | Pu241 | 8.3% | 3.3% | 8.3% | 2.8% | | |
| | Pu242 | 10.4% | 3.4% | 10.4% | 2.6% | | |
| | Pu quality | 56.1% | 62.2% | 56.1% | 64.4% | 0% | +4% |
| Initial Pu content | | 16.3% | 15.5% | 24.9% | 23.3% | - | - |
| Initial Pu mass | | 8.3 t/GWe | 7.9 t/GWe | 8.8 t/GWe | 8.2 t/GWe | +5% | +4% |

The constraints on Pu availability are slightly increased in a scenario of CFV deployment, due to the difference of Pu mass in core, but the feasibility is not questioned by the change of core design. As in the case of SFR V2B deployment, the Pu margin can be adjusted to meet fissile material requirement by reducing the minimum cooling time of SFR fuels from 2080 (3 years instead of 5 years) or by adding radial fertile blankets (axial blanket being already part of the break-even core).

The impact of the change of SFR core design on facilities remains limited. At equilibrium, the annual need for fuel fabrication for CFV is 540 tons per year (315 tons of fissile fuels and 225 tons of fertile),

to compare to 450 tons per year for SFR. Even if the global capacity is modified, fabrication processes are not affected. The maximal size of spent fuels storage depends mainly on PWR fuels and is consequently not affected by the change of SFR core. The distribution of Pu inventory in cycle and in waste is also similar whatever the SFR core.

## 4. RESULTS ON SFR DEPLOYMENT IN COMPLEMENT OF THE FRENCH PWR FLEET

In order to assess the impact of SFR introduction in the French nuclear reactor park, illustrative scenarios with various deployment schedules for SFR in view of renewal of the French PWR fleet as well as in complement to the French PWR fleet are studied. In this context, SFRs (SFR V2B core design) are assumed to be progressively introduced from 2040 on to provide a complementary service to PWR fleet (for example, to take advantage of Pu in PWR MOX spent fuels). Consequently, the aim of these scenarios is to determine the number and the pace of SFRs deployed in function of the Pu resource produced by the PWR fleet.

### *4.1. Maximal achievable SFR deployment*

In a first time, the maximal achievable SFR V2B deployment in function of the Pu produced by the PWR fleet is evaluated, as displayed on Figure 9. A 60 GWe EPR™ fleet is deployed from 2020 to 2050 to renew the current French PWR fleet. The case of continuing MOX in EPR™ (30% of MOX fuels and 70% of UOX fuels in core, for one third of EPR™ fleet, which could be equivalent to 10% of MOX over whole of the fleet) and the case of stopping MOX in EPR™ are both taken into account. In the first case, SFRs are deployed using Pu of MOX spent fuels whereas in the second case, SFRs are deployed using Pu from MOX and UOX spent fuels.

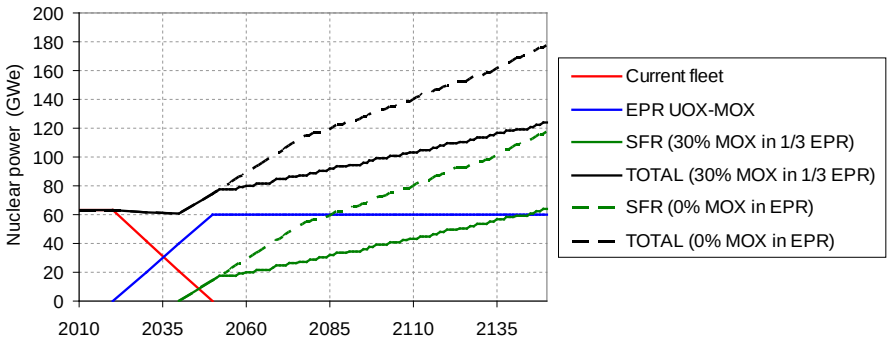


*FIG. 9. Maximal achievable SFR deployment with and without MOX in EPR™ reactors fleet*

Two main phases can be observed:

- a first stage of SFR deployment starting in 2040 using Pu of accumulated PWR spent fuels (the pace of SFR deployment is limited to 1.45 GWe per year). This stage leads to the deployment of 17 GWe of SFR in 2052 using accumulated PWR MOX fuels and of 55 GWe of SFR in 2078 using both MOX and UOX fuels. This is made possible by the reprocessing of all fuels accumulated and available, at the maximum rate of 500 tons/year of MOX fuels or 3,700 tons/year of MOX and UOX fuels.
- at the end of the first stage, a phase of regular SFR deployment starts, whose pace depends on the amount of Pu yearly produced by PWR fleet. With 30% of MOX in EPR™, the mean pace of SFR deployment is 0.5 GWe per year (one unit of 1.45 GWe started every 3 years) whereas it reaches 0.9 GWe per year (one unit started every 1.7 years) assuming no MOX being used in EPR™.

It is clear that SFR deployment at even very significant rates is possible from a Pu-balance perspective in France and that the progressive introduction of SFRs is possible after the Pu-recycling in PWRs.

## *4.2. Key drivers to modulate SFR deployment*

### *4.2.1. Sensitivity to the spent fuel cooling time*

Different key drivers can be identified to modulate SFR V2B deployment in this scenario. The Figure 10 presents the sensitivity of SFR deployment to the cooling time before reprocessing (Tc) of PWR fuels (left figure) and SFR fuels (right figure). Results are given in the case of 30% of MOX fuels in one third of EPR™, but similar results are obtained without MOX in EPR™.

The cooling time of PWR fuels impacts mainly the available amount of Pu to deploy SFRs during the first stage, using Pu in accumulated PWR spent fuels. Consequently, the first stage of SFR deployment is reduced if PWR fuels cooling time is increased. If the PWR fuels cooling time is superior to 40 years, the SFR deployment can be delayed after 2040. Besides, the cooling time of SFR fuels affects the delay for a SFR to be able to use its own Pu and by consequence the Pu inventory required to start a new SFR. Finally, the pace of SFR deployment during the regular stage can be reduced by a factor two if the SFR fuels cooling time is increased from 5 to 15 years.

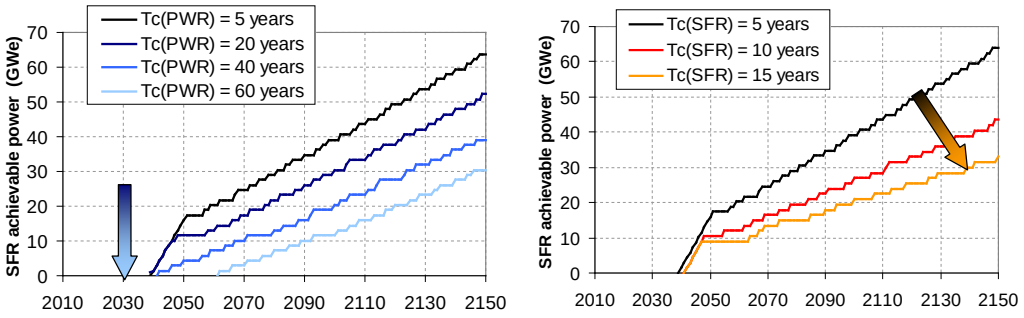


*FIG. 10. Sensitivity of SFR deployment to the spent fuel (PWR and SFR) cooling time*

### *4.2.2. Sensitivity to the capacities of fuel cycle facilities*

The capacity of fuel cycle facilities is another parameter impacting the amount of available Pu for SFR deployment. The results obtained without limiting the reprocessing plant capacity, corresponding to the maximal achievable SFR deployment detailed at paragraph 4.1, are presented in the black curve on Figure 11. In that case, the reprocessing of 500 tons per year during 10 years of accumulated MOX and REPU spent fuels, followed by the reprocessing of MOX and SFR spent fuel yearly outputs, allow the deployment of 39 GWe in 2100. If the reprocessing plant capacity is limited to a realistic value (for example, 130 t/y of MOX spent fuels, in accordance with the French MELOX plant capacity, and 20 t/y of REPU spent fuels), the first stage of SFR deployment is suppressed in favor of the phase of regular deployment, leading to the deployment of 25 GWe of SFR in 2100.

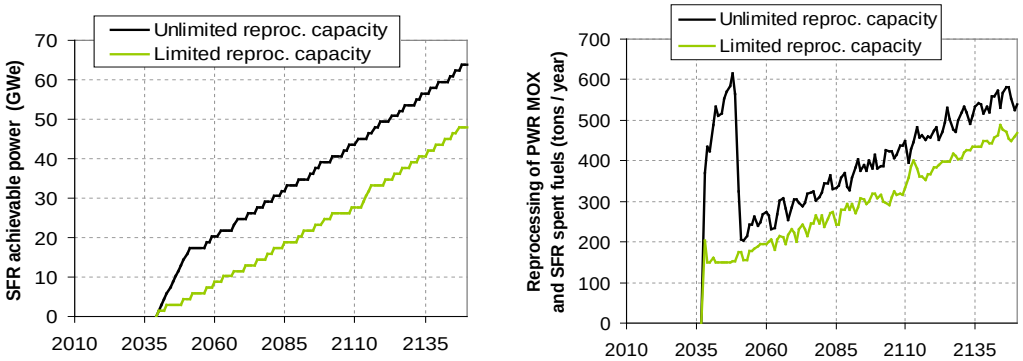


*FIG. 11. Sensitivity of SFR deployment to the capacities of fuel cycle facilities*

Accordingly to these assumptions, the evolution of PWR MOX and SFR spent fuels storage is displayed on Figure 12. In the scenario of maximal SFR deployment, accumulated PWR MOX spent fuels are reprocessed from 2040 to 2050 and the storage stabilizes at 660 tons, corresponding to 5 years of cooling of annual production. With limited reprocessing capacity, the PWR MOX spent fuels storage stabilizes to its level of 2040 (4,000 tons). Assuming that the maximal UOX spent fuels storage reaches 16,000 tons (Pu from UOX spent fuels being used for PWR MOX fabrication), spent fuels storage needs are consistent with current French storage capacities. Furthermore, the SFR spent fuels storage increases according to the evolution of the number of SFRs deployed in each scenario.

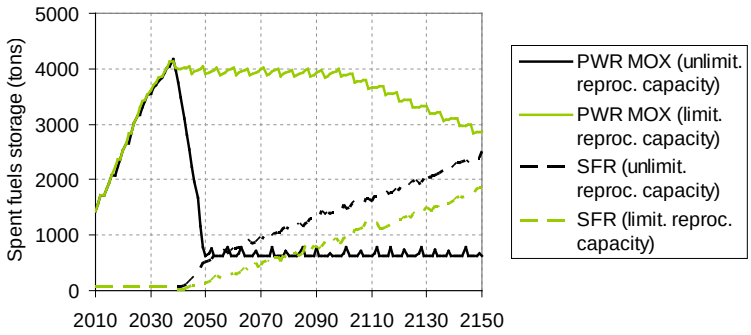


*FIG. 12. Sensitivity of spent fuels storage to the capacities of fuel cycle facilities*

The progressive deployment of 25 GWe of SFR from 2040 to 2100 (at the pace of 0.4 GWe per year) corresponds to the minimal SFR deployment allowing to benefit from the Pu annually produced by the PWR French fleet while stabilizing PWR spent fuels storage to current values and requiring realistic reprocessing capacities. This scenario follows a logic of preparing for the future, in anticipation of the need of Fast Reactors facing the risk of natural uranium shortage.

## 5. CONCLUSION

Several illustrative scenarios with contrasting assumptions have been studied to sketch the capability of SFR deployment:

- to renew the French PWR fleet, with a constant energy production, for different time scales of SFR introduction (progressively in two stages from 2040 or over 30 years from 2080) and for different SFR core design (homogeneous core SFR V2B or more innovative heterogeneous core CFV with a gain on sodium void effect).
- from 2040, in complement of a 60 GWe EPR™ fleet in the continuity of the current fleet, to take advantage of Pu in PWR spent fuels through multi-recycling instead of immobilizing Pu in spent fuel.

The SFR deployment in replacement of French PWR fleet is made feasible in terms of Pu availability, whatever the date of SFR introduction, by reducing the minimum cooling time of SFR spent fuels to less than 5 years from 2080 or by adding fertile blankets to SFR cores. The impacts of SFR deployment on fuel cycle facilities are limited, even if spent fuels storage and reprocessing capacities are significantly higher if SFR deployment is delayed in 2080. The isotopic composition of fuels loaded in SFR is compatible with the range of operation of these reactors and with handling and transport constraints. The consideration of CFV V0 (preliminary core design) does not challenge these findings, the fissile material need being similar for CFV and SFR V2B.

Scenarios of SFR deployment in complement of a 60 GWe EPR™ fleet highlight a first stage of SFR deployment starting in 2040 using Pu of accumulated PWR MOX and eventually UOX spent fuels, followed by a phase of regular SFR deployment, the pace of which is 0.5 GWe per year with MOX in EPR™ and 0.9 GWe per year if MOX might be stopped in EPR™. Different key drivers have been identified to modulate SFR deployment in this scenario. An increase of PWR spent fuels cooling time limits the amount of available Pu for SFRs and thereby reduces the first stage of SFR deployment. An

increase of SFR spent fuels cooling time increases the fissile material inventory required to start a new break-even core and consequently reduces the pace of SFR deployment. If the reprocessing plant capacity is a continuation of today's available capacity, the first stage of SFR deployment is suppressed in favor of the phase of regular deployment. This leads to the progressive deployment of 25 GWe of SFR from 2040 to 2100, which is the minimal SFR deployment allowing to benefit from the Pu annually produced by the PWR French fleet while stabilizing PWR spent fuels storage to current values.

Future studies will deal with scenarios based on symbiotic PWRs using MOX fuel and SFRs park ; this option allows namely to take into account the recycling in PWRs of the plutonium coming from SFR spent fuels and advanced industrial optimizations.

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# DEVELOPMENT OF NUCLEAR FUEL CYCLE IN VIETNAM

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**Abstract.** Fossil fuel energy as well as hydropower sources in Vietnam and in the world are being exhausted. And nuclear power offers energy more effectively, safely and economically, while simultaneously combating greenhouse gases. In Vietnam, nuclear power would contribute to ensuring the energy security of the nation and provide power for development, industrialization and modernization. So Vietnam has been considering nuclear power as an option of energy supply since the 1980s. In 2002, the Prime Minister established the Governmental Steering Committee for the Development of Nuclear Power in Vietnam, and in 2006 this Committee developed "the Long-term Strategy for Peaceful Utilization of Atomic Energy in Vietnam up to 2020", which aims to introduce the first nuclear power plants (NPPs) by 2020. Vietnam has begun to implement this "Long-term Strategy" by establishing a comprehensive Master Plan, enacting the Atomic Energy Law, and approval by the National Assembly on 25 November 2009 for the construction of the first NNPs. The agreed timescales are that the first reactor will start construction in 2014 and operation in 2020, and that four reactors (4000 MW) will be in operation by 2025 under a basic demand scenario.

Institute for Technology of Radioactive and Rare Elements (ITRRE), one of institutions of Vietnam Atomic Energy Institute (VINATOM), is responsible for development of nuclear fuel cycle (NFC) in order to guarantee nuclear fuel supply security for the NPPs. On the base of The Decision - No. 906/QD-TTg – "Approving orientations for planning nuclear power development in Vietnam through 2030", three main articles in the front end of the NFC, namely study on Vietnam's uranium mining-milling (to product yellow cake), study on nuclear fuel (to fabricate nuclear fuel) and study on radioactive waste treatment (to manage radioactive waste) are developed in Vietnam.

## I. INTRODUCTION

### 1.1. Nuclear fuel cycles

There are six major steps in NFC [2].

**(1) Uranium mining and milling**

Uranium is the starting fuel for all fuel cycles. Uranium mining and milling is similar to the mining and milling of copper, zinc, and other metals. Uranium is often found with copper, phosphates, and other minerals and thus a co-product of other mining operations. About 200 tons of natural uranium is mined to fuel a 1000-MW(e) light-water reactor (LWR) for one year.

**(2) Uranium conversion and enrichment**

The uranium is chemically purified. Uranium contains two major isotopes: uranium-235 and uranium-238. Uranium-235 is the initial fissile fuel for nuclear reactors. Natural uranium contains 0.7% uranium-235. In the uranium enrichment process, natural uranium is separated into an enriched uranium product containing 3 to 5% uranium-235 and ≥95% uranium-238 that becomes LWR fuel and depleted uranium that contains ~0.3% uranium-235 and ~99.7% uranium-238. There will be 10 to 20 times as much depleted uranium as product.

**(3) Fuel fabrication**

The enriched uranium is converted into uranium dioxide and fabricated into nuclear fuel. An LWR requires ~20 tons of fuel per year.

**(4) Light-water reactor**

About 90% power reactors in the world are LWRs. The initial fuel is uranium-235 that is fissioned to produce heat. The fuel also contains uranium-238, a fertile non-fuel material. In the nuclear reactor some of it is converted to plutonium-239 – a fissile fuel that is also fissioned to produce heat. The heat is converted into electricity. With a fresh fuel assembly, all the energy is from fissioning of uranium-235. When the fuel is discharged from the reactor as spent nuclear fuel (SNF), about half the energy being generated is from the fissioning of plutonium-239 that was created in the reactor.

**(5) Storage of SNF**

A typical LWR fuel assembly remains in the reactor for three to four years. Upon discharge of the SNF, it contains ~0.8% uranium-235, ~1% plutonium, ~5% fission products, and uranium-238. The SNF is stored for several decades to reduce radioactivity and radioactive decay heat before disposal.

**(6) Waste disposal**

After interim storage, the SNF is disposed of as a waste in a repository.

*Nuclear fuel cycles are different from fossil fuel cycles because nuclear reactors burn only a fraction of the fuel before the fuel is discharged as SNF. Full burn up of the fuel before discharge is not possible [2].*

- The reactor produces heat by fissioning uranium-235 or plutonium-239. The resultant fission product "ash" in high concentrations will shut down the reactor.

- The materials of fuel element construction have a limited endurance in the reactor and limit fuel burn up. Because reactors cannot fully utilize the fissile and fertile materials in a fuel assembly, there are many possible fuel cycles.

- LWR partly closed fuel cycle (Top two lines of Fig. 1):

The fissile material in LWR SNF can be recycled back into LWRs. The LWR SNF is reprocessed, the plutonium and uranium recovered, and the plutonium and some uranium are fabricated into fresh fuel, and the resultant transuranic fuel is sent to the LWR. Because of the low fissile content of the LWR SNF, recycle of the plutonium reduces uranium fuel demand by only 15% and recycle of the uranium reduces uranium fuel demand by only 10%. The high-level waste (HLW) from reprocessing is stored for several decades to reduce radioactivity and radioactive decay heat before disposal. LWR SNF recycle changes the plutonium isotopes such that the SNF can only be recycled one or two times. The recycle SNF must either wait to go to a repository or could fuel fast reactors. Several countries recycle LWR SNF.

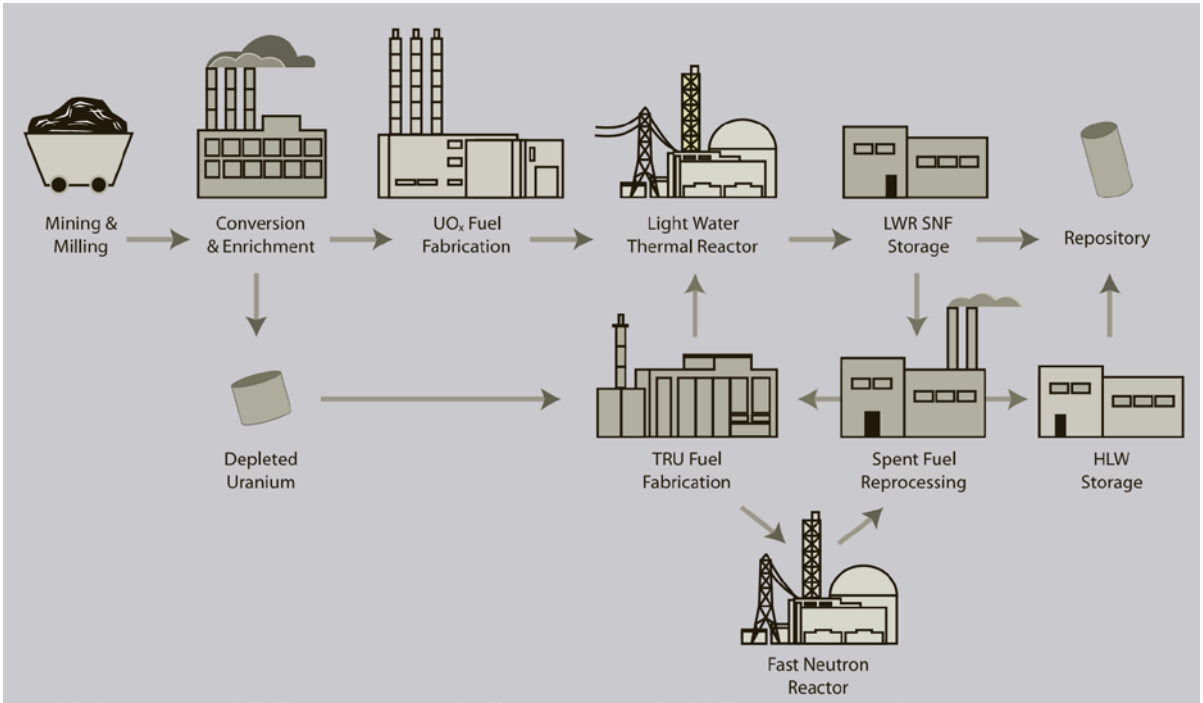


***Figure 1.*** *Alternative Fuel Cycle [2].*

- Fast reactor fuel cycle:

Fast neutron-spectrum reactors can convert fertile uranium-238 to fissile plutonium-239 faster than they consume that fuel and thus convert all uranium-238 into fissile fuel over time. This enables full utilization of the depleted uranium from LWR uranium enrichment facilities, the uranium in LWR SNF, and the plutonium in LWR SNF. Such reactors can recover 50 times as much energy per kilogram of mined uranium as an LWR; however, fast-reactor startup requires a large fissile inventory. The traditional strategy is to reprocess LWR SNF and use the recovered plutonium to fabricate fast reactor fuel. The plutonium in LWR SNF from 30 years of operations is required to start one fast reactor with a high conversion ratio. After fast reactor startup and operation, fast-reactor SNF is reprocessed to recover plutonium and uranium. Plutonium and uranium from fast reactor SNF, and makeup depleted uranium are used to fabricate new fast reactor fuel assemblies. Each fast-reactor SNF assembly has sufficient plutonium for a new fast reactor fuel assembly. Fast reactors are under development in several countries but are today uneconomic and have not been deployed.

### 1.2. Vietnam's policy on nuclear fuel supply [1, 3]

Though Vietnam has an estimated 210 thousand tons of uranium ore deposits, it does not currently have the operational or technological capability to extract or process it into nuclear fuel. As such, Vietnam will need to either import the nuclear fuel rods from abroad or include a provision in the tender offer that requires the eventual winning bidder to supply the nuclear fuel necessary to run the reactors.

The Decision - No. 906/QD-TTg – "APPROVING ORIENTATIONS FOR PLANNING NUCLEAR POWER DEVELOPMENT IN VIETNAM THROUGH 2030" [1] indicated the route map for assurance of nuclear fuel supply security as follows:

***By 2015:*** To study mechanisms, policies and solutions for assuring security of nuclear fuel supply for the nuclear power plant. To investigate and explore natural uranium and develop mechanisms and policies on exploitation and commercial use of natural uranium.

***By 2020:*** To adopt policies on assuring nuclear fuel supply security, including formation of a fuel security assurance fund. To work out a roadmap for localizing production of fuels from imported enriched uranium. To develop mechanisms and policies on exploitation and use of natural uranium. To be able to absorb nuclear fuel manufacturing technologies and have a system of laboratories on modern uranium technologies.

***By 2030:*** To master the technology of manufacturing nuclear fuels, build investment projects on facilities to manufacture domestic nuclear fuels from imported enriched uranium. To start commercial exploitation of natural uranium in the country.

To ensure the operation of the first NPPs, nuclear fuel for the plant must be supplied by the nuclear reactor supplier. Nuclear fuel supplier and Vietnam, a nuclear fuel recipient, would negotiate bilaterally the terms and conditions of a fuel supply contract including its duration, the quantities of fuel involved, the pricing mechanism and the relevant fuel performance guarantees. However, the annual fuel replacement must be performed for all the lifetime of the reactor and the spent fuel must be stored and disposed. Therefore, these problems must be carefully considered in the long-term plans of the nuclear power program. Thus, the supply contract would have two components, one related to the supply of fresh fuel assemblies and the other dealing with spent fuel management. Under the fuel supply contract, spent fuel would have to be returned to the Supplier State after a minimum cooling time (e.g. 2 years or less if technically possible and economically justified).

Vietnam needs to build an Integrated Nuclear Infrastructure, including infrastructure of nuclear fuel supply guarantees and study on the purposes, problems to be solved, assurance mechanism of initiatives/proposals on nuclear fuel supply assurance from Global Nuclear Energy Partnership (GNEP) – now the International Framework for Nuclear Energy Cooperation (IFNEC), Russian LEU Fuel Reserve in Angarsk, international organizations (IAEA, NTI, WNA), etc. Purpose of these initiatives/proposals is non-proliferation of nuclear weapon, to persuade states without facilities of NFC (both states possessing NPPs and states having plan to build NPPs) not to develop sensitive

nuclear technologies: uranium enrichment and spent fuel reprocessing and to set up framework for nuclear fuel supply assurance in future. The policy and program on security of nuclear fuel supply for nuclear power program of Vietnam, localization of several stages of NFC need to be considered in order to improve the nuclear fuel supply policy.

Vietnam needs to study and employ machanisms of nuclear fuel service and market in present in order to buy LEU and nuclear fuel fabrication service. And in long – term, Vietnam will set up a nuclear fuel fabrication facility from import LEU. The State should study to formulate policies on fuel supply security for nuclear power program and use services and the current fuel market according to the present mechanism, study on the issues of international law and implementation of the inter-governmental agreement, etc. to buy natural or low enriched uranium and to receive services of fuel production as well as to construct fuel facilities and to transfer technology of nuclear fuel fabrication.

Vietnam needs to improve the investigation, exploration, and research process, economic evaluation of natural uranium to formulate policies on appropriate uses of a part of raw materials from uranium ore, study on the ability to build fuel fabrication factory from imported enriched uranium to provide fuel directly to the reactors. This is un-inhibited and un-hindered by International Community with respect to the non-proliferation of nuclear weapons. Vietnam in the near future owns the nuclear fuel fabrication to be of great advantages to meeting the fuel design, increasing the storage capacity of preventive fuel and taking the initiative in nuclear fuel under all circumstances.

Vietnam needs to participate in initiatives and international proposals to ensure fuel supply in case of disruption of the fuel supply for political reasons. The initiatives and proposals are constantly being implemented and amended, but the biggest international trends are to develop international fuel centers, fuel banks and to build the multi-lateral and multi-layers mechanism to ensure the supply of fuel. These trends are currently supported by most of all countries, IAEA as well as policy researchers of fuel supply and non-proliferation of nuclear weapons. To bring these centers to reality, it is depended on the efforts not only of the countries that use nuclear fuel but also of the nuclear powers, and global trend whether nuclear power promotion or not. In international exchanges, Vietnam has the right to discuss and require guaranteed supply of fuel with the mechanism of the new entrants to build their own nuclear power and Vietnam's own. Vietnam also needs to participate in the international research center for fuels not only by commitments, but also with shares, administration, etc. The other proposals should also be considered such as fuel leasing service, reactor leasing service and spent fuel collection service, etc.

The current views of Vietnam are to support these initiatives in order to ensure fuel supply security and energy security. Vietnam needs to focus on the trend towards the international centers for nuclear fuel supply with the mechanisms to ensure multi-lateral and multi-layers.

In company with the problem, Vietnam still has significant work to do in preparation for the introduction of a nuclear power plant including the creation of a standard system of building and operating NNPs, developing human resource on nuclear power that meet international standards, and drawing up effective plans to deal with nuclear power plant breakdowns. These tasks, while achievable, will require continued collaboration with the IAEA and international partner countries.

## II. DEVELOPMENT OF NUCLEAR FUEL CYCLE IN VIETNAM

There is adequate time before any choices for deployment need to be made to move away from the current open fuel cycle. Uranium resources are relatively abundant with respect to the uranium requirements for credible growth rates of the nuclear power system. Evolution from the open cycle will in any case be gradual. The preferred long-term path forward is not certain today. For the long term, the incentives for development of alternative fuel cycles are: extension of fissile resources; possible mitigation of waste management challenges; and possible minimization of proliferation concerns.

However, in the last decade there have been major changes in our understanding of uranium resources, implications of different fuel cycle assumptions such as the conversion ratio for advanced reactors, and new technologies. Multiple factors will influence the ultimate choice of a NFC, including (1) the pace and scale of nuclear power deployment and (2) evolving technical, economic, and safety

performance of fuel reprocessing methods, reactor types (both LWR and fast spectrum reactors), and disposal pathways for waste streams, and (3) the relative importance society places on different goals.

Vietnam uses the once-through open fuel cycle to fuel LWRs and, in the near future, to localize partially NFC. This fuel cycle is the simplest and the most economic one today (Top line of Fig. 1).

### 2.1. Vietnam's uranium resources

Uranium exploration in selected areas of Vietnam began in 1955. Since 1978, a systematic regional exploration program has been underway throughout the entire country. About 330,000 $km^2$, equivalent to almost 100% of the country, have been surveyed at the 1:200,000 scale using surface radiometric method combined with geological observations. About 103,000 $km^2$ (31% of the country) have been explored at 1:50,000 scale. Nearly 80,000 $km^2$, or 24% of the country, has been covered by an airborne radiometric/magnetic survey at the 1:25,000 and 1:50,000 scales. Selected occurrences and anomalies have been investigated in more detail by 75,800 m of drilling and by underground exploration working.

Uranium exploration is conducted by the Geological Division for Radioactive and Rare Elements and the Geophysical Division of the Department of Geology and Minerals of the Ministry of Industry and Trade. From 1997 through 2002, exploration was concentrated on evaluation of the uranium potential of the Nong Son basin, Quang Nam province. Exploration activities were concentrated on three projects: (1) evaluation of the An Diem deposit hosted in sandstone; (2) exploration of the Pa Rong area and (3) exploration of the Dong Nam Ben Giang area in the southeast Ben Giang-Nong Son basin.

Vietnam reports RAR (Reasonably Assured Resources) reserve of 113 T of $U_3O_8$, EAR-I (Estimated Additional Resource) of 16,563 T of $U_3O_8$, EAR-II (Estimated Additional Resource) 15,153 T of $U_3O_8$ and SR (Speculative Resource) of 186,338 T of $U_3O_8$. Total of the uranium reserves in Vietnam is 218,167 T of $U_3O_8$.

***Table 1.*** *Uranium reserves in sandstone (Nong Son basin)*

| Area | Reserve (ton of $U_3O_8$) | |
|---|---|---|
| | EAR-I + EAR-II | EAR-I + EAR-II+SR |
| Khe Hoa-Khe Cao | 10,040 | 22,000 |
| Pa Lua | 5,420 | 12,000 |
| Pa Rong | 1,766 | 4,695 |
| An Diem | 2,266 | 2,266 |
| South-East Ben Giang | 1,500 | 10,500 |
| **Total** | **20,992** | **51,461** |

### 2.2. Study on uranium ore treatment

Technology flowsheet for uranium ore processing in Vietnam is indicated in figure 2. After the uranium ore is mined, uranium is chemically extracted from the ore in order to produce a partially refined product with a uranium content of at least 65%, yellow cake. Uranium milling is based primarily on hydrometallurgical operations such as leaching, solvent extraction or ion exchange, and precipitation.

The basic steps of acid leaching- uranium ores in the flowsheet are [4]:

(1) Crushing and grinding;

(2) Leaching;

(3) Solid-liquid separation and washing;

(4) Solvent extraction or ion-exchange;

(5) Yellow-cake precipitation and drying.

The run-of-mine ore is crushed and then ground to the consistency of fine sand. Most uranium mills use wet grinding, and the resulting slurry is fed to a leaching circuit where sulphuric acid is added. Leaching times vary from a few to more than 24 hours. With the ores, an oxidant such as manganese

dioxide or sodium chlorate has to be added to achieve satisfactory uranium extraction. The oxidant is needed because most ores contain uranium in the reduced, or quadrivalent, form. The reduced uranium is only slightly soluble in the acidic leach solutions; the oxidant provides the driving force to convert the uranium to the hexavalent state which is readily soluble. Leach recoveries normally range from 85% to 95%, and the resulting leach solutions are relatively dilute but complex acidic sulphate solutions containing a wide variety of ions. The uranium concentration is normally 1 to 2 g/liter; concentrations of the other ions can vary greatly, depending on the composition of the specific ore being treated. After leaching, the solids and liquids are separated, and the solids are washed to recover the adhering leach solution. In most mills, the washing operations are conducted in countercurrent thickener circuits. Both the thickener techniques and the flocculants developed for use in uranium mills are now widely used in other hydrometallurgical industries. Flocculants are chemical agents that can gather suspended particles into aggregations which settle much faster than the individual particles.

The uranium is separated from the leach solutions by solvent-extraction or ion-exchange. The uranium is then stripped from the organic complex or ion-exchange resin by contacting it with an back-extraction or eluent solution. The yellow-cake is precipitated from the strip or eluent solution, and the resulting solid is dried and packaged.

### 2.3. Study on fuel preparation

Generally, there are three main stages in the fabrication of the nuclear fuel structures used in LWRs and PHWRs [5]:

(1) Producing pure uranium dioxide ($UO_2$) from incoming $UF_6$ or $UO_2$;

(2) Producing high-density, accurately shaped ceramic $UO_2$ pellets;

(3) Producing the rigid metal framework for the fuel assembly – mainly from zirconium alloy; and loading the fuel pellets into the fuel rods, sealing them and assembling the rods into the final fuel assembly structure.

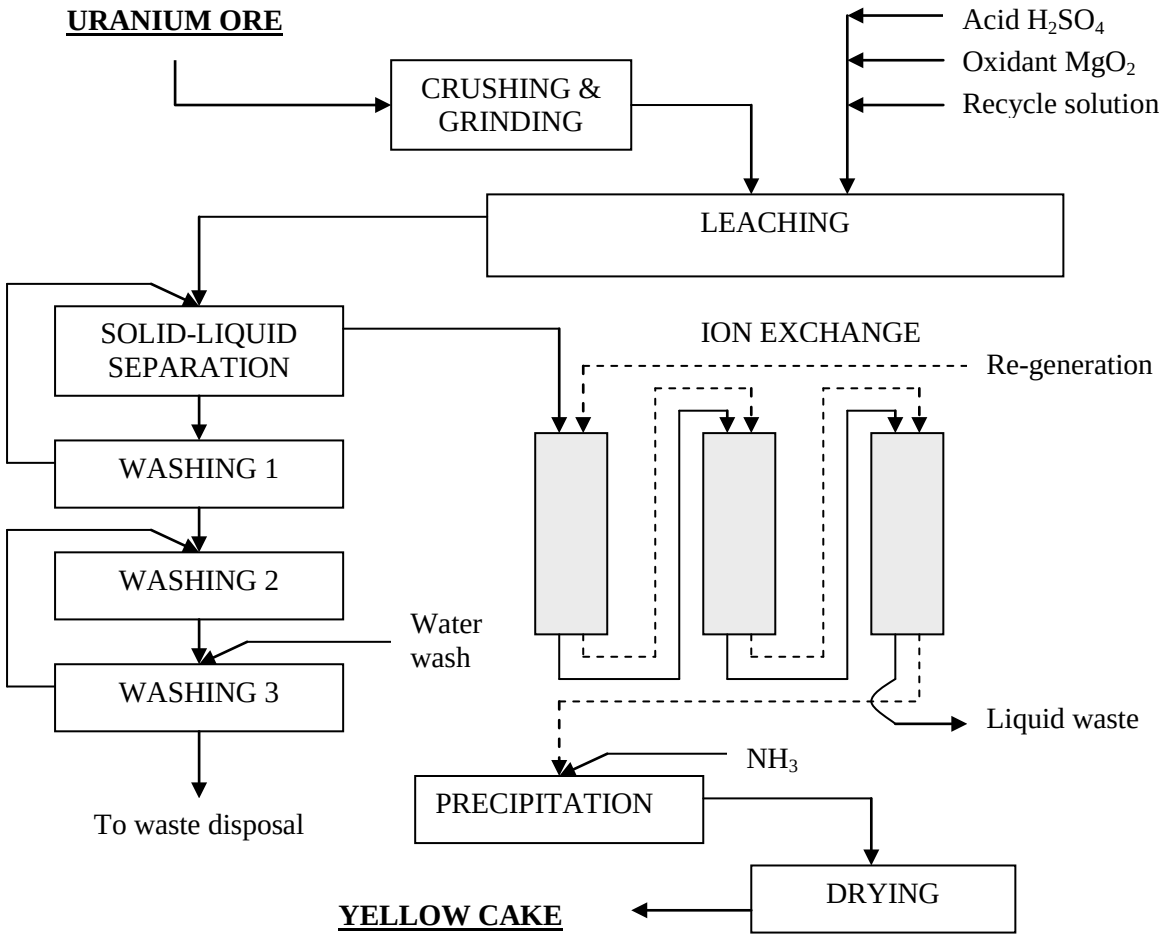
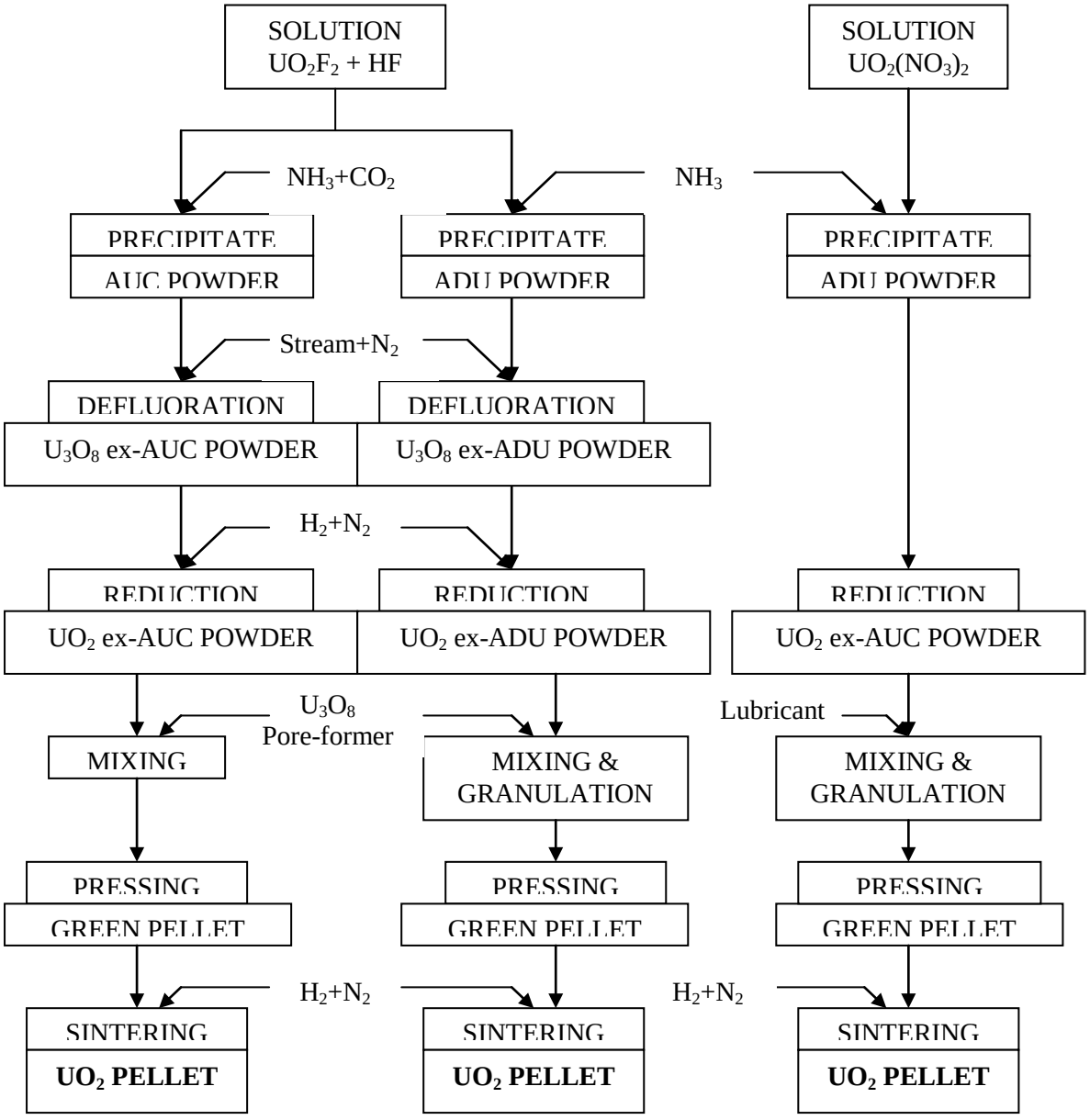


***Figure 2.*** *Uranium ore-processing flowsheet.*

Conversion of $UF_6$ to $UO_2$ can be done using "dry" or "wet" processes. In the dry method, $UF_6$ is heated to a vapor and introduced into a two stage reaction vessel (e.g. rotary kiln) where it is first mixed with steam to produce solid uranyl fluoride ($UO_2F_2$) – this powder moves through the vessel to be reacted with $H_2$ (diluted in steam) which removes the fluoride and chemically reduces the uranium to a pure microcrystalline $UO_2$ product. Wet methods involve the injection of $UF_6$ into water to form a $UO_2F_2$ particulate slurry. Either ammonia ($NH_3$) or ammonium carbonate ($(NH_3)_2CO_3$) is added to this mixture and the $UO_2F_2$ reacts to produce; ammonium diuranate (ADU - $(NH_3)_2U_2O_7$) in the first case, or ammonium uranyl carbonate (AUC – $UO_2CO_3.(NH_3)_2CO_3$) in the latter case. In both cases the slurry is filtered, dried and heated in a reducing atmosphere to pure $UO_2$. The morphology of $UO_2$ powders deriving from the ADU and AUC routes are different, and this has a bearing on final pellet microstructure. Wet methods are slightly more complex and give rise to more wastes, however, the greater flexibility in terms of $UO_2$ powder properties is an advantage [5].

In Vietnam, the preparation of $UO_2$ powder from incoming uranyl nitrate and uranyl fluoride of nuclear grade via ADU and AUC route (figure 3).




**Preparation of simulative $UO_2$ pellet for LWR (AUC and ADU methods)**

**Preparation of simulative $UO_2$ pellet for PHWR**

***Figure 3.*** *The simulative $UO_2$ pellet preparation process in Vietnam.*

Uranium used for the study on fuel preparation in Vietnam is natural uranium. And the uranyl nitrate and uranyl fluoride solutions are preparated from Vietnam's yellow cake. The $UO_2$ powder may need further processing or conditioning before it can be formed into pellets:

- Homogenization: powders may need to be blended to ensure uniformity in terms of particle size distribution and specific surface area.

- Additives: $U_3O_8$ may be added to ensure satisfactory microstructure and density for the pellets. Other fuel ingredients, such as lubricants and pore-formers may also need to be added.

Conditioned $UO_2$ powder is fed into dies and pressed biaxially into cylindrical pellet form using a load of several hundred MPa – this is done in pressing machines operating at high speed. These "green" pellets are then sintered by heating in a furnace at about 1750°C under a precisely controlled reducing atmosphere (usually mixture of Ar or $N_2$ with $H_2$) in order to consolidate them. This also has the effect of decreasing their volume.

For most reactors pellets are just under one centimeter in diameter and a little more than one centimeter long. A single pellet in a typical reactor yields about the same amount of energy as one tonne of steaming coal.


**Technical (76%) $U_3O_8$**

**Simulative $UO_2$ powders and ceramics**

***Figure 4.*** *Products of yellow cake and simulative $UO_2$ ceramic pellet.*

### 2.4. Study on radwaste management

Types of radioactive wastes in Vietnam consist of exempt waste and very low level waste (VLLW) that contains radioactive materials at a level which is not considered harmful to people or the surrounding environment, low-level waste (LLW) that is generated from hospitals and industry, as well as the uranium ore processing and $UO_2$ pellet preparing.

Traditional uranium mining generates fine sandy tailings, which contain virtually all the naturally occurring radioactive elements naturally found in uranium ore. These are collected in engineered tailings dams and finally covered with a layer of clay and rock to inhibit the leakage of radon gas and ensure long-term stability. In the short term, the tailings material is often covered with water. After a few months, the tailings material contains about 75% of the radioactivity of the original ore. Strictly speaking these are not classified as radioactive wastes.

For low level radwaste handling, because of the different characteristics of solids, liquids, and gases, each must be processed differently. The waste must also be processed in such a manner as to minimize the risk of exposure to the public. These processes might include [6]:

(1) Filtering,

(2) Routing through demineralizers,

(3) Boiling off the water (evaporation) and leaving the solid impurities (which are then processed as solid radioactive waste), and/or

(4) Storing the liquid for a time period to allow the radioactive material to decay.

After processing, the water will be sampled. If samples show the water meets the required standards, the water can be placed in the storage tanks for use in the plant or be released to the environment. If the samples show the water does not meet the standards, it will be reprocessed.

Some materials, such as the evaporator bottoms (solids that remain after the water is evaporated off), will be mixed with some material to form a solid (such as concrete). This is also sometimes done with spent demineralizer resins. After mixing with a hardener, the material is processed as solid radioactive waste.

Figure 5 indicated the LLW handling system in Vietnam. There is not high level radwaste in Vietnam.

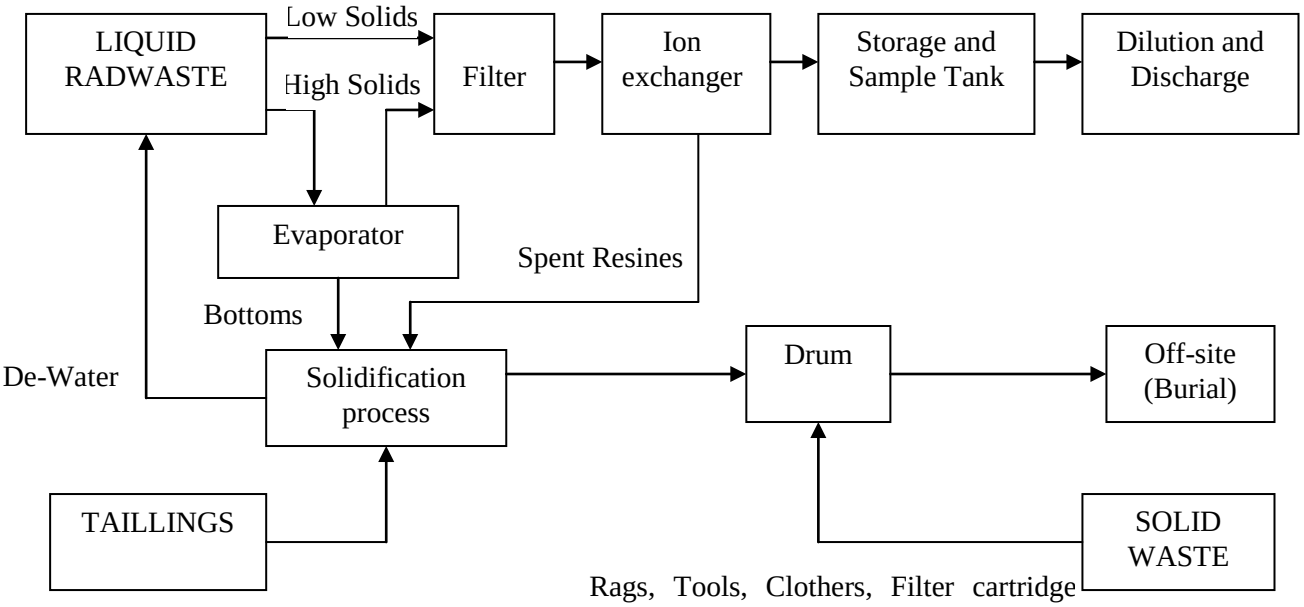



***Figure 5.*** *Radwaste handling system.*

***Figure 6.*** *An interim storage for radioactive waste in Vietnam.*

## III. CONCLUSION

The application and development of nuclear energy in Viet Nam is for peaceful purposes, and for the development of the national economy and society. Therefore, Vietnam wants to make sure that the nuclear application and development will be absolutely harmless to the people and the environment. Vietnam always considers nuclear safety and security a top priority in using nuclear technology for peaceful purposes. Vietnam abides by and implements all treaties and major international legal documents in nuclear disarmament, non-proliferation, nuclear safety and security, including the Nuclear Non-Proliferation Treaty (NPT), the Comprehensive Test Ban Treaty (CTBT) and UN Security Council Resolution 1540.

Vietnam will localizate several stages of NFC (treatment of Vietnam's uranium ore and nuclear fuel fabrication from imported LEU) and need to improve the nuclear fuel supply policy. During from 2025 to 2030, Vietnam will set up a nuclear fuel fabrication facility from import LEU.

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# Fast Reactor Systems and Innovative Fuels for Minor Actinides Homogeneous Recycling

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*Presented by R. Calabrese*

**Abstract. T**his work is focused on the performance of critical fast reactor systems aimed at the transmutation of minor actinides (Np, Am, Cm) homogeneously dispersed in the MOX driver fuel. In particular, the paper deals with two scenarios in the 2050 time horizon, at first evaluating an extension of once-through fuel cycle strategy, hence introducing fast reactors in a closed fuel cycle strategy beyond 2030. The synergistic use of the DESAE and NFCSS scenario codes permitted to evaluate key indicators for natural resources usage, waste management, proliferation issues, and fuel cycle infrastructures needs. The paper aims at discussing the sustainability of a high development of nuclear energy to promote a transition to a low-carbon energy future. Finally, the results of scenarios analysis are discussed in the light of the ongoing studies moving ahead in the development of innovative fuels for minor actinides transmutation (e.g., PELGRIMM EU projects), where ENEA is actively involved on the track of related past activities.

## 1. Introduction

The concept of sustainable development addresses the capability to meet present needs without compromising the possibility to answer to the needs of future generations. Energy is fundamental to improve living standards and to support societal development. If on one hand fossil energy sources have been capable to account for about 80% of the global primary energy supplies, on the other hand this choice leads to phenomena of increasing concern like urban air pollution, regional acidification and human-induced climate change [1]. A fundamental driver for a future development of nuclear energy is the capability to tackle greenhouse gases (GHG) emissions at a competitive cost [2]. Management of nuclear waste and their proliferation risks are recognized drawbacks of nuclear energy deployment. Safety is a crucial factor as a localized accident may cause globally changes in the national energy policies with the decision to phase-out or abandon any plans to enter nuclear business as confirmed in the recent accident of Fukushima-Daiichi. Main reasons that have been promoting the recent nuclear renaissance are still valid as suggested by the limited reduction in the projections of the nuclear energy published by the international organizations [3].

A sustainable development of nuclear energy is therefore strongly dependent on the introduction of innovative reactor systems and advanced fuel cycle strategies. In this regard, the Generation IV Forum identified six plant technologies capable to tackle various issues of concern such as the use of natural resources, the management of spent fuel inventories and their burden on the final repository as well as proliferation and economics issues [4][5].

The paper discusses the transition from an open fuel cycle strategy to a closed fuel cycle strategy where innovative reactors are capable to safely manage MA-bearing oxide fuels as for the ABR (Advanced Burner Reactor), and ELSY (European Lead-cooled SYstem) fast reactors investigated in this study [6][7]. In this scenario, next-generation FRs, deployed from 2030, reach in 2050 a value of installed capacity of 5% of adopted projections. The curve of nuclear energy development meet the requirements of a scenario where nuclear energy pursues, together with others energy sources, the objective to limit the concentration of $CO_2$ in the atmosphere at 450 ppm [2][8]. The DESAE code (Dynamic Energy System-Atomic Energy) and the NFCSS code (Nuclear Fuel Cycle Simulation System) permitted to model and analyse these scenarios [9][10][11][12].

 

Last section of the document addresses the needs in R&D for the qualification of MAs-bearing oxide fuels as a crucial factor for the deployment of proposed transition to a close fuel cycle.

## 2. Status and projections of development of the nuclear energy sector

Following the Fukushima-Daiichi accident, some countries announced the decision to phase-out from nuclear business in the next two decades (Belgium, Germany, and Switzerland) or to abandon any plan to reintroduce nuclear energy in the near term (Italy); nevertheless, several countries confirmed their nuclear option revising the projected rate of development [2]. Recent data reports that 436 nuclear power plants are in operation with a total net installed capacity of about 370 GWe, 5 nuclear power reactors are in long-term shutdown and 63 under construction [13]. Pressurized light water reactors (272 units) and boiling water reactors (84 units) account for about 88.5% of the total capacity. The share of pressurized heavy water plants is 6.2% with 47 units. Gas-cooled, light water-cooled graphite-moderated and fast breeder power plant technologies cover the residual fraction of nuclear fleet. In 2004 the total inventory of spent nuclear fuel was close to $2.68 \cdot e^{+05}$ $t_{HM}$ reaching in 2011 a value of $3.41 \cdot e^{+05}$ $t_{HM}$ with an increase by about $0.11 \cdot e^{+05}$ $t_{HM}$ per year [14][15]. Reprocessing plants, nowadays with a nominal capacity of 4100 $t_{HM}$ per year, treated about 30% of the spent fuel stockpiles [16].

The curves of nuclear development adopted in the analysis account for the effects of the Fukushima-Daiichi event with a reduction in the range 5-13% of pre-accident estimates; see Table 1 [8][17]. Other international organizations confirmed these indications [2]. In this paper, the curve of total nuclear capacity is consistent with a high rate of deployment in a scenario where nuclear energy plays a fundamental role in limiting GHG emissions.

Table 1. Projections of nuclear energy development, GWe

| | **2020** | | **2030** | | **2050** | |
|---|---|---|---|---|---|---|
| | **low** | **high** | **low** | **high** | **low** | **high** |
| **Nuclear capacity** | 429 | 525 | 501 | 746 | 560 | 1228 |

## 3. General assumptions and rate of deployment of innovative fast reactors

Main assumptions in presented results are as follows:

- scenarios extend from 1960 up to 2050;
- SNF is at first delivered to the interim storage nearby power plants for cooling hence either to the final repository or to the reprocessing plants;
- plutonium and MAs recovered from the reprocessing of spent UOX is recycled either in ABR or ELSY (needed only during the dynamic phase of deployment);
- reasonable assured and inferred identified uranium resources 5.404 million tonnes (< USD 130/kgU) [18];
- undiscovered uranium resources 10.4 million tonnes [18];
- total reprocessing capacity 4100 $t_{HM}$/yr [16];
- tails assay of natural uranium enrichment 0.3%;
- load factor of operating plants 0.8 (0.9 for innovative FRs).

Presented analysis deals with following strategies for the management of spent fuel: open fuel cycle (OFC) and closed fuel cycle (CFC) by assuming high projections of nuclear energy development. In particular, in the OFC, spent fuel is piled-up in a long-term storage after cooling at the interim storage nearby NPPs. In this strategy, the deployment of heavy water reactors is constant accounting for 6.2% of the total installed capacity and, from 2010, third-generation light water reactors are connected to the

grid in the place of second-generation; see Fig.1. In the CFC, next-generation fast reactors, either ABR or ELSY, reach from their introduction a fraction of 5% in two decades; see Fig. 2. In this hypothesis, the foreseen development of fast reactors should occur mainly in the second half of the century.

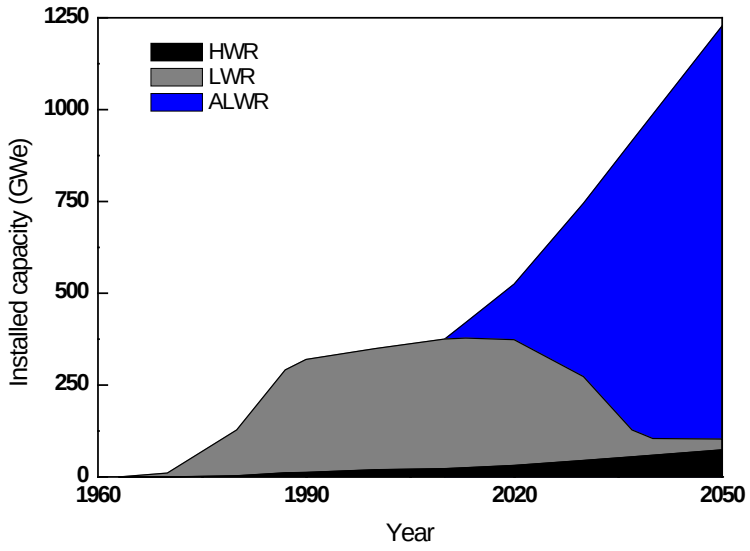


*FIG. 1. Open fuel cycle.*

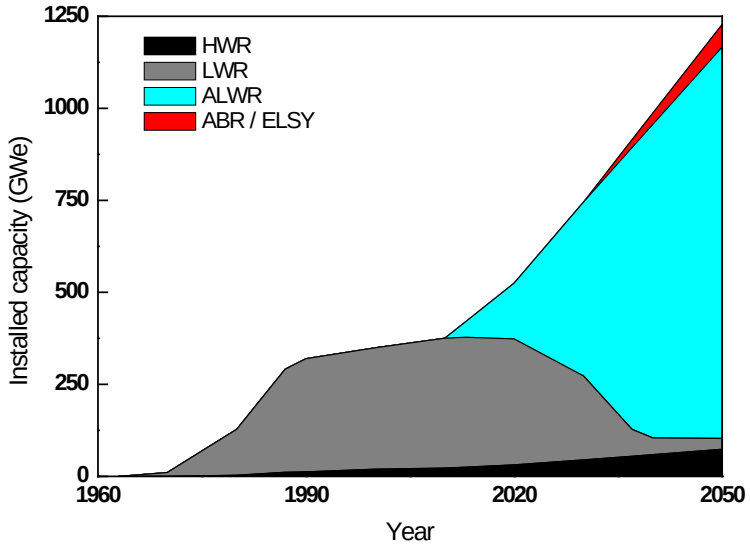


*FIG. 2. Closed fuel cycle.*

Presented study depicts a homogeneous world where nuclear technology knowledge, natural and infrastructural resources are shared at a global level as a preliminary step of a deeper investigation on the sustainability of presented fuel cycle strategies.

## 4. DESAE and NFCSS codes

The DESAE 2.2 code allows predicting financial and material resources needed for a sustainable nuclear energy policy at country, regional and global level [9][10]. For calculations, the user defines the deployment scenarios where reactors, fuel cycle facilities, and energy demand projections are introduced in the proper input sections. The code offers to the user various options for the modelling of both open and closed fuel cycles taking into account uranium and thorium recycling. DESAE does not perform burn-up or core management calculations for that reason fresh, equilibrium and spent core isotopic composition are required. The code deals with 18 nuclides[1] and one additional variable accounting for the fission products as a single group. The layout of investigated nuclear energy systems may be composed up to seven different nuclear reactor models and four recycling plants. In this regard, the code does not take into account the reprocessing losses.

NFCSS (formerly known as VISTA) is a scenario code developed by the IAEA to estimate, for investigated strategies, the requirements for long-term nuclear material and fuel cycle services as well as the related material flows [11][12]. Calculations, performed in a country or worldwide nuclear power fleet, provide estimates of natural uranium, conversion, enrichment, and fuel fabrication needs. Furthermore, the code, through the analysis of quantity and quality (isotopic composition) of unloaded fuels, let the user apply a recycling strategy. Users can introduce in the analysis different reactor types in addition to seven built-in models to study commercial plants. NFCSS calculates the overall nuclear materials flow as well as the discharge and the accumulation of each nuclide through a two-step calculation. The module named CAIN (Calculation of Actinide Inventory) is the first step and performs the analysis, according to one-group cross sections, of the fuel depletion under irradiation and the change of isotopic composition during cooling and reprocessing. In the second phase, the code defines material flows and needs for services in the front-end and back-end according to the energy system designed by the user. The code deals with 14 nuclides.[2] The modelling of proposed scenarios relies on the synergistic use of DESAE 2.2 and NFCSS codes.

## 5. Nuclear power plant models

Different values of uranium enrichment, to cope with the burn-up of discharged fuel, are distinctive parameters of light water reactor (LWR) and advanced light water reactor (ALWR); see Table 2. HWR is the abbreviation for pressurized heavy water plant.

The ABR, developed on the so-called Super-PRISM design, is a sodium-cooled fast reactor designed to reach a transuranics conversion factor of ~0.3 and a one-year cycle length (3 batches) [6]. The oxide core is loaded with transuranics recovered from the reprocessing of the LWR spent nuclear fuel. The ELSY reactor was investigated within the so-called adiabatic core concept where the core produces energy burning $^{238}$U and releasing to the environment only fission products of in-reactor fission reactions and the losses at the reprocessing stage [7][19]. Equilibrium compositions are mainly dependent on neutronics and quite similar for lead- and sodium- cooled fast reactor [20]. The ELSY design is therefore capable to transmute its own actinides where the ABR could act as burner for the MAs legacy. The value of burn-up of the ABR system is typical of Generation IV fast reactors.

The NFCSS code provided the isotopic composition of fuel after irradiation and after cooling. This information together with others, some of which presented in Table 2, form the reactor models introduced in the DESAE code and applied in calculations. The schema of the reaction/decay chains of NFCSS proved to calculate with a reasonable accuracy the total amount of MAs with deviations of about -10% for both ABR and ELSY [6][19]. In this regard, the total amount of curium proved to deviate by about -50% while for americium the deviations were below +6%.

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1 $^{230}$Th, $^{232}$Th, $^{232}$U, $^{233}$U, $^{234}$U, $^{235}$U, $^{236}$U, $^{238}$U, $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu, $^{237}$Np, $^{241}$Am, $^{244}$Cm, $^{129}$I, $^{99}$Tc.
$^{2235}$U, $^{236}$U, $^{238}$U, $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Pu, $^{237}$Np, $^{241}$Am, $^{242m}$Am, $^{243}$Am, $^{242}$Cm, $^{244}$Cm.

The SNF isotopic composition calculated by means of DESAE is consistent with the results provided by NFCSS, in the long-term, the estimates of $^{237}$Np show a deviation; see Figs. 3 and 4. These figures highlight the different range of isotopic composition in ALWR and ABR.

Table 2. Parameters of reactor models

| **Reactor** | **HWR** | **LWR** | **ALWR** | **ABR** | **ELSY** |
|---|---|---|---|---|---|
| **Reactor capacity**, GWe | 0.837 | 1 | 1.516 | 0.8 | 0.6 |
| **Heavy nuclei loading**, $t_{HM}$ | 119.0 | 78.7 | 133.0 | 16.2 | 35.1 |
| $^{235}$**U,** wt.% of uranium vector | 0.711 | 3.7 | 4.9 | 0.2 | 0.1 |
| **Fissile plutonium**, wt.% | - | - | - | 19.4 | 10.4 |
| **Burn-up**, GWd/$t_{HM}$ | 7.7 | 45.0 | 60.0 | 135.0 | 78.0 |
| **SNF cooling time**, yr | 6 | 5 | 5 | 5 | 4 |
| **Plant lifetime**, yr | 57 | 50 | 50 | 60 | 60 |

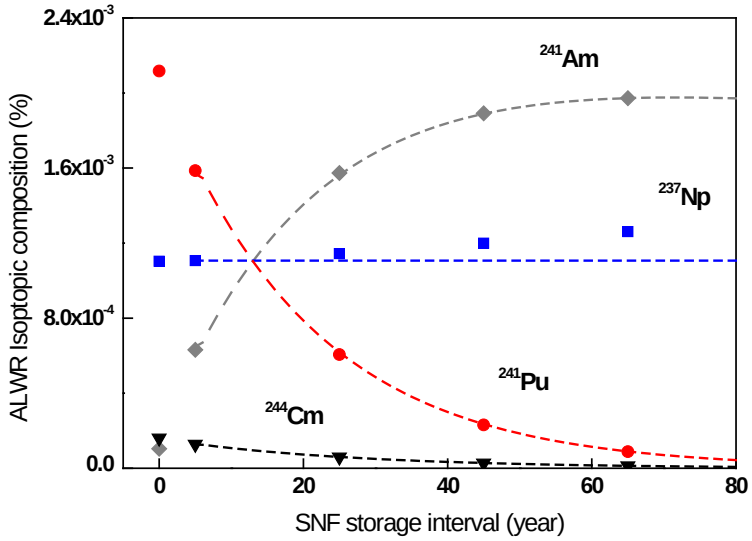


FIG. 3. ALWR: Comparison DESAE (dashed lines) vs. NFCSS (scatter).

## 6. Results

The consumption of natural uranium is well above the total indentified resources in the domain of undiscovered resources; see Table 3. In this regard, the introduction of innovative fast reactors has, in this analysis, a limited impact. To meet sustainability criteria on natural resources, unless considering significant investments on geological inspections, requires higher rate of introduction of studied systems or the deployment of breeder fast reactors. On the contrary, the closure of fuel cycle markedly reduces the stockpiles of SNF and fissile plutonium. These values depend on the operating reprocessing capacity in each scenario, a parameter that was defined, assuming the current reprocessing capacity, pursuing the objective to have no separated fissile plutonium at 2050, see Fig. 5. In this regard, ELSY has a better performance in comparison with ABR, meeting this constraint by using lower values of reprocessing services thanks to the adiabatic concept. In this analysis, the reprocessing of ABR SNF is not performed. The value of the separated minor actinides is 223.5 and 191.8 $t_{HM}$ for ELSY and ABR respectively, confirming the capability of this latter in transmuting the MAs legacy.

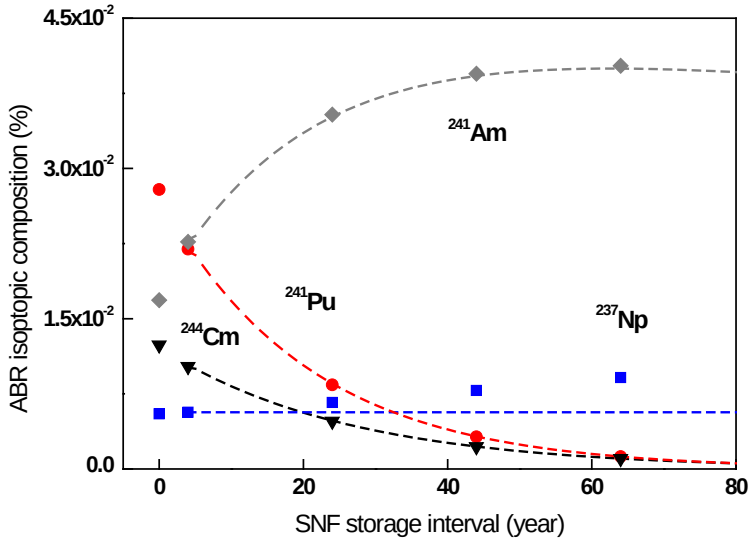


*FIG. 4. ABR: Comparison DESAE (dashed lines) vs. NFCSS (scatter).*

Table 3. DESAE integral results at 2050, $t_{HM}$

| | **$U_{nat}$** | **SNF** | **Fissile plutonium** | **MAs** |
|---|---|---|---|---|
| **OFC** | 8059723 | 792837 | 4159 | 1347 |
| **CFC-ABR** | 7915167 | 619089 | 3409 | 1007 |
| **CFC-ELSY** | 7915172 | 674549 | 3460 | 1060 |

The need for fast reactor fuel fabrication is, at the end of the introduction phase, 374 $t_{HM}$ per year for ABR and 647 $t_{HM}$ per year for ELSY, values consistent with the different level of burn-up and of the same order of the current fabrication capacity for MOX fuel that is 440 $t_{HM}$ per year.

## 7. Fuels for MAs recycling in the homogeneous strategy

The experience gained on sodium-cooled fast reactors proved that MOX fuel have been performing well in a domain of burn-up up to 170 MWd/$kg_{HM}$ with linear heat rate of the order of 450 W/cm. The performance of cladding is a technological constraint in fast reactors: swelling, irradiation creep and corrosion may seriously limit the lifetime of fuel pins at high burn-up. For ELSY, the corrosion behaviour in lead suggested to limit the temperature of cladding operating in normal conditions. Fuel rods of Generation IV systems are designed to operate in domains of 100-200 MWd/$kg_{HM}$ for fuel burn-up and up to 200 dpa for cladding neutron damage to meet economical and transmutation requirements.

The knowledge of relevant properties of MOX fuel in this domain has not fully assessed yet and the impact of parameters such as burn-up and stoichiometry is still an open issue [21]. The description of the performance of MOX at high burn-up is considered a cross-cutting issue relevant for the development of next-generation fast reactor [5]. The addition of MAs has several effects on key properties such as fuel thermal conductivity and melting temperature, where the re-distribution of actinides under irradiation makes even more complex the description of fuel behaviour. MA-bearing MOX proved an enhanced production of helium that could lead to either high swelling or high release.

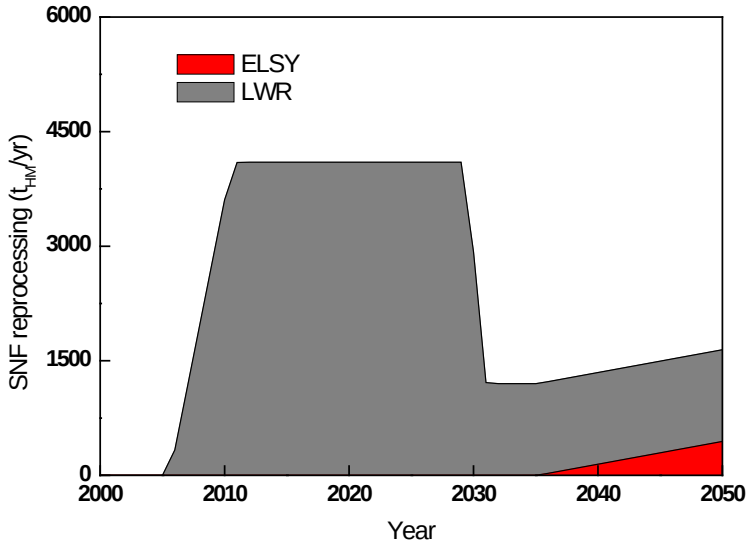


*FIG. 5. Reprocessing of SNF in the CFC-ELSY scenario (in the CFC-ABR, not shown, the reprocessing is constant at a value of 4100 $t_{HM}$ /yr since 2030).*

Significant efforts in R&D are therefore required for the assessment of the performance of MA-bearing MOX fuel. Experimental findings indicated that the addition americium and neptunium to MOX proved to have a limited impact on the values of un-irradiated thermal conductivity up to 1770 K; melting temperature was affected at an extent of 4 K per wt.% [22][23][24]. The Radial re-distribution of actinides in MA-bearing MOX, noted in the SUPERFACT experiment, was confirmed already at the early stages of irradiation [25][26]. The SUPERFACT experimental programme suggested that the production of helium under irradiation is nearly four times higher than the reference MOX [26].

Fuel fabrication is also demanding due to the high decay heat and radiation field associated with the use of minor actinides. The EU PELGRIMM project (PELlets versus GRanulates: Irradiation, Manufacturing & Modelling) is studying the performance of sphere-pac fuel in comparison with standard pellets. Sphere-pac fuel should be well performing under irradiation accommodating the internal stress due to the large amount of gas release, both fission products and helium, and to fuel swelling in the inter-particles free volumes. The deployment of such an advanced fuel cycle strategy goes far beyond the state of the art and significant R&D programmes are needed on recycling, fabrication, and fuel qualification to achieve this ambitious objective [27].

## 8. Conclusions

The capability of nuclear energy source to limit GHG emissions at a competitive cost is a potential driver for its development in the near- and medium-term. Several issues such as the shortage of natural uranium resources, the management of steadily increasing inventories of spent nuclear fuel as well as competitiveness and proliferation are of concern for the sustainability of nuclear energy. Nuclear technology should be, for its societal acceptability, affordable, safe and featured by low proliferation risks. In this regard, the design of innovative fast reactors should markedly improve the performance in each of addressed areas.

This paper confirms the expected performance of investigated ABR and ELSY systems noting, in presented scenario, that the rate of their introduction is dependent on the availability of plutonium and the shortage of natural uranium shortage is still an open issue. The reduction of GHG emissions by

means of a steep expansion of nuclear energy needs therefore careful investigations, together with the improvement of reactors performance, of several parameters of planned fuel cycle strategy such as the rate of introduction and breeding capability of innovative systems as well as the capacity of reprocessing plants. In this regard, a multi-criteria approach is of crucial importance to optimize the planning of advanced nuclear energy systems.

In this frame, the development of MA-bearing oxide fuel is of great importance to meet the requirements expected by next-generation nuclear reactors and investigations are ongoing. Significant efforts in R&D are fundamental for a detailed fuel qualification where different options for fabrication are under considerations.

As final remark, the synergistic use of DESAE and NFCSS proved to be capable in modelling a transition scenario to an innovative fuel cycle strategy providing consistent results regarding the performance of reactor models applied. The estimate of the codes accuracy in the evaluations of MAs would be an important in the discussion of results and objective of future analyses.

## ABBREVIATIONS

| | |
|---|---|
| ABR | Advanced Burner Reactor |
| ALWR | advanced light water reactor |
| CAIN | Calculation of Actinide Inventory |
| CFC | closed fuel cycle |
| DESAE | Dynamic Energy System – Atomic Energy |
| ELSY | European Lead-cooled SYstem |
| FR | Fast Reactors |
| GHG | greenhouse gases |
| HM | heavy metal |
| HWR | heavy water reactor |
| IAEA | International Atomic Energy Agency |
| INPRO | International Project on Innovative Nuclear Reactors and Fuel Cycles |
| LWR | light water reactor |
| MA | minor actinides |
| MOX | mixed oxide fuel |
| NFCSS | Nuclear Fuel Cycle Simulation System |
| NPP | Nuclear Power Plant |
| OFC | open fuel cycle |
| PELGRIMM | PELlets versus GRanulates: Irradiation, Manufacturing & Modelling |
| PRIS | Power Reactor Information System |
| SNF | spent nuclear fuel |

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# A sociotechnical analysis of the French FBR programme:

# evaluation as a cornerstone

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**Abstract.** Based on a three-year study of the French FBR program, this paper aims at showing the dynamics of evaluation of FBR demonstrators, with methodological inputs from the "Science and Technology Studies" branch of sociology. Such a reactor has to demonstrate the feasibility - including safety, technical and economic viability - of a promising technology regarded as a potentially inexhaustible energy source for the future. The research shows that until the mid-seventies, the need for an FBR fleet was regarded as urgent, entailing a focus on proving technical feasibility with demonstration reactors. But after the mid-seventies, the evaluation of FBR projects gave more importance to other topics: not only did they have to prove the technical feasibility of the programme, but also its safety and economic viability. The analysis of the Superphénix case is used to illustrate the difficulty of reconciling the three elements of the evaluation in a changing context.

## 1. Introduction

Fast Breeder Reactors (FBR) technology was developed immediately after the Second World War, with prototypes of ever-increasing sizes in different countries [1]. Yet there is nothing linear about this history: depending on the time and the country, development underwent phases of acceleration, of slowdown, of stoppage and of renewal.

In France, in order to learn from the past, experts in this technology implement "operational feedback" and record what they learned from the technical choices that were made, from global "loop" or "pool" concepts, to fine steel grades. Nevertheless, the story of the development of sodium-cooled fast breeder reactor technology is not just one of technical objects, there are also human and even socio-technical elements. In this sense, since 2009 EDF R&D have been carrying out research from the perspective of the sociology of science and technology.

This paper focuses on one of the many lessons learned from this research, that of the crucial role of evaluation in explaining the phases of project slowdown or acceleration. In this context, evaluation is understood in a very broad sense, as being one's appraisal of the technology, making it possible to qualify it from a technical, economic or safety standpoint. The result of these evaluations is that every apparently technical decision, such as to build a prototype, to make improvements to safety or to recalibrate the project, is in fact the realization of the discourses which qualify the project.

First of all, we will set out the conditions and line of attack of the research; we will then present the three phases of the history of Sodium-cooled Fast Breeder Reactor technology in France, relating to three different ways of assessing the technology. The final section will discuss several key issues relating to the problems involved in assessing prototypes and demonstrators.

## 2. The sociology of science and technology applied to the Superphénix project: a demanding and comprehensive method

This socio-technical analysis of the development of FBR technology in France is the result of three years of research carried out by EDF R&D, as part of a project devoted to fourth-generation reactors. The sources for this work were extensive reading of existing literature, the consultation of archives and approximately thirty interviews of project actors. Some of the interviewees were involved in the first steps of fast-breeder technology, before reorienting their careers towards other areas of nuclear power; others were involved from start to finish, devoting their entire careers to the development of the technology; finally, some came from other professions and were only briefly involved. We met scientists and engineers who took part in the conception, construction or operation of Superphénix, as well as members of the board of directors, experts from the nuclear safety authority, and some opponents of the technology.

The history of fast breeder technology in France, and of Superphénix in particular – a reactor which was stopped earlier than planned – is a controversial one. Researchers who examine this case are confronted with a profusion of written documents of a highly diverse nature: within the literature we studied, we found 65 publications on fast breeder technology, not to mention the extensive press archives. The initial difficulty is one of developing a methodological and interpretive framework which will make it possible to organize these sources in a coherent manner. The method we chose is rooted in the sociology of science and technology and in particular in the work by Bruno Latour, "Aramis or the Love of Technology"[2]. This book traces the history of a public transport project called "Aramis" which was intended to serve the south of Paris with the combined advantages of rail transport and individual cars, but which never reached the commercial stage. Above and beyond a case study, this work offers lessons on factors for success or failure for such innovative projects, along with a methodological stance from which to talk about the past from the point of view of the researcher's situation in the present. The narrator talks to his student as follows:

"*Always assume that people are right, even if you have to stretch the point a bit.* […] *otherwise, you play the sly one at the expense of history. You play the wise old owl.* […] *life is a state of uncertainty and risk, of fragile adaptation to a past and present environment that future cannot judge.*" ([2], p.35-37)

This obligation to show goodwill is one of the features of the method used during this analysis of fast breeder technology. Another requirement is that of applying a rigorous critical approach: such human and social science research involves an iteration between sources, theoretical frameworks and constantly revized intermediate hypotheses, until one succeeds in producing an interpretation which, in a coherent fashion, brings together and integrates all of these elements in a "reciprocal double-fitting" (Baldamus quoted by [3]). This critique-based approach is common to the sociology of science and technology [2] or to history [4]. In particular, we refused to reread the history of a technology on the basis of its developments which were known to the researcher but unpredictable for the actors in the on-going project, as illustrated in this quote from Rip [5]:

*"the direction of technological development was determined by the actual paths and the expectations of what could be next steps [...]. Our retrospective idea of steps in the direction of the situation as we know is irrelevant"*.

These methodological requirements allowed us to interact with actors who have been involved in the development of SFR technology for many years, whether this was within the framework of investigative interviews, exchanges, or regular presentations as part of the "fourth generation nuclear" research project at EDF R&D, the other work packages of which relate to technical developments and feedback.

Using numerous existing accounts of the history of sodium fast breeder technology in France, some of which separate the technical from the political, the originality of our approach is that it offers a socio-technical analysis. During our research, we felt the issue of FBR project evaluation to be crucial: it

traverses the entire period under consideration, with different modalities, and this will be the subject of this paper.

## 3. A history in three periods

This research allowed us to offer a "socio-technical" chronology in three periods which interlink visions of the future and evaluations of prototypes and the FBR technology :

- The beginnings: demonstration of the feasibility of fast-breeder technology (1954-1975);
- From the programme to construction of the "industrial prototype": the evaluation broadens out to a triptych – technology / economics / safety (1975-1986);
- The challenge of operation: justifying safety and revamping objectives (1986-1997).

This chronology will allow us to show that each stage of the development of the fast breeder technology is the result of an evaluation of its necessity and potential. More particularly, the decision of whether or not to move on to the next stage of the programme was always the result of an assessment of the merits of the programme, based on criteria which evolved over the many decades under consideration.

### *3.1. The beginnings: demonstration of the feasibility of fast breeder technology (1954-1975)*

In France, FBR projects were launched throughout the 1950s and 1960s, a period when the vision for the future which made them necessary was shared by the decision-making bodies:

- forecast of an increasing demand for energy leading to the development of nuclear power;
- concern regarding the depletion of energy resources and the increasing cost of uranium, giving a significant advantage to FBR technology which is capable of regenerating its own fuel;
- desire for national energy independence;
- vision for the nation's future which would come about through national technologies;
- technical and economic potential of these reactors deemed to be highly promising.

We might view the development of experimental and then demonstration reactors as a consequence of these visions for the future of energy. In France, FBR project developers were driven by the conviction that this technology would receive overall support if they could provide proof of its technical feasibility. To this end, the first experimental reactor in France, called "RAPSODIE", was developed and built in Cadarache by the Commissariat à l'Energie Atomique (CEA) with a contribution from Euratom; it diverged in 1967 [6].

At the end of the 1960s, debates in France on the type of nuclear power to choose for the industrial fleet were an opportunity to compare different visions for the future: on the one hand, the rhetoric of national technological excellence supported the graphite-gas industry developed in France; on the other hand, the rhetoric of the economics of energy supply was in favour of light-water nuclear plants (LWR), available from American constructors at very attractive prices. Chosen in 1969, LWRs were seen as a short-term economic solution to meet energy requirements. Everyone then agreed to plan for a future nuclear fleet based on FBRs, a technology which combines the challenge of national technical excellence with that of energy-saving [7]. In the long term, FBR technology, the cornerstone of the nuclear system, should allow the nuclear industry to supply abundant low-cost energy to the entire world. Over the medium term, the challenge for competing countries was therefore to be the first to develop plants which were industrial (powerful and reliable) and commercial (capable of equipping the national fleet and of being exported).

In France, the next stage was to create a 250 MWe prototype, the characteristics of which were inferred from an industrial 1000 MWe pre-project [6]. The Phénix project diverged in 1973, and was hailed as a technical success, brought to fruition within the deadlines due to innovative project

organization ("integrated project") which brought the project owner and engineering and construction companies together within a single project team.

During this period, the elements of project evaluation were as follows:

- the vision of the future of energy made FBR technology necessary over the short-medium term;
- the challenge was to prove its feasibility, and projects were essentially analyzed from a technical standpoint with a safety condition, control of which was passed to an *ad hoc* department within the CEA [8]. Under such a regime of research and demonstration, the function of every experimental installation or prototype was to answer the same implicit question: "does it work?"

At the beginning of the 1970s, the evaluation was positive: the satisfactory commissioning of Phénix was proof of the viability of FBR technology. The decision was therefore taken to launch the phase for the Superphénix industrial prototype, which would mark FBR's move from the experimental era into the industrial era. Five times more powerful than Phénix, Superphénix was launched as an "industrial and European prototype".

### *3.2. From the programme to the construction of an "industrial prototype": evaluation expands into a technology / economics / safety triptych (1975-1986)*

During the decade which constituted the second period of our chronology of the French FBR programme, the development and evaluations of FBR projects evolved in parallel in various areas which we will set out in the following order (the order is theme-based and not chronological):

(1) The Superphénix "industrial prototype" was built on the Creys-Malville site, located in the south-east of France, between Lyon and Geneva;
(2) Safety assessment and economic evaluation were of increasing importance in institutional and academic circles, with an international dimension;
(3) The economic evaluation found its material translation within EDF's plant design division, in an effort to make the reactors of the future fleet less costly;
(4) The question of the future need for fast breeder technology and the shift to the industrial fleet was also subject to evaluation;
(5) The characteristics of the Superphénix industrial prototype were subject to "expert" militant criticism which echoed academic criticism from French research laboratories or foreign institutions.

#### *3.2.1. The "industrial prototype" worksite*

In a manner which was more visible, the 1975-1986 decade was that of the Superphénix "industrial prototype" worksite. But the mission embedded in the characteristics of Superphénix was more vast than the demonstrator missions of the previous period, Rapsodie or Phénix, which were designed to prove technical feasibility within the CEA research agency. Deconstructing the mission assigned to a prototype is a useful way of analysing related debates and controversies: Superphénix now had to validate the full-size industrial operation of a technology deemed to be ready for commercialization. This mission was embedded just as much in Superphénix's technical characteristics (a size of 1200 MWe, industrial choices), as in its organisational characteristics: the project owner was a joint venture company made up of French, Italian and German electricity supplyers (to mention just one of the elements of a complex organization targeting the serial production and commercialization of FBRs in the near future [9]).

On site, the project managers had to overcome numerous difficulties: the creation of a "world first" meant that they were constantly facing technical challenges. The project engineers speak volubly about this difficult job of work where they used all their technical and innovation skills to resolve unprecedented problems. With the help of the sociology of science and technology, we can consider

the worksite phase as a time when the project was weighed down by all of the technological detours that had to be invented in order for it to see the light of day [10]. The finished prototype was thus not exactly the same as on the pre-project plans: it was more complicated, and the provisional budgets and scheduling had to be extended. Case studies such as Aramis show that this is the cost of completing an innovative project in the form of a technical installation. One key question is therefore to find out whether the way in which people promoters talk about this technical installation is coherent with its new material form [2][11].

### *3.2.2. Safety assessment and economic evaluation simultaneously of increased importance*

In parallel to this huge worksite, though in a less visible manner, economic and safety evaluations were of increasing importance during the decade under consideration. In France, the Service Central de la Sûreté des Installations Nucléaires (SCSIN) was created in 1973 as part of the French Ministry of Industry: Superphénix, which prefigured a future industrial fleet was very carefully examined by this department, which was no longer part of the CEA. Modifications had to be made to the prototype to take account of earthquakes and the evacuation of residual power; these modifications were necessary for the project to take its place in the reality of a safety regulatory regime at a given point in time.

Just like the detours necessary to resolve the practical difficulties at the worksite, the modifications changed and expanded the original pre-project plans.

Hecht [7] shows that the examination of the energy production industry from the standpoint of economic evaluation appeared in France at the end of the 1960s. The economic assessment of fast breeder programmes took on new dimensions at the beginning of the 1980s, when academic economists examined the dossier and pointed out – often in an accusatory manner – the successive revisions of cost estimates for fast breeder projects, depending on the actors and criteria taken into consideration [12]. They challenged the hypotheses made by research agencies, such as the ratio used to predict the cost of a production reactor compared to a prototype reactor, or the law predicting a drop in the specific investment costs when the size of the reactor increases. Their evaluations made the extrapolation from prototype to industrial fleet in line with their own criteria, and they invalidated the utility of an FBR fleet for electricity production, on the basis of cost overruns and the technical problems of the prototypes built.

In a less accusatory tone, expert reports demanded by institutions in the United Kingdom and the United States also examined the potential and the costs of fast breeder technology. The official British “Flowers” report [13] questioned both the safety and economics of the industry, on the basis of the experience gained with the prototypes. Whilst taking account of the numerous uncertainties, it concluded that fast breeder reactors constituted a form of insurance against a possible depletion of energy resources in the future. Combined with this insurance-based rationale, the report’s assessment led it to conclude that it was preferable to delay commitment to the 1000 MW commercial demonstration plant that was envisaged at that time in the United Kingdom.

Compared to the previous period, the evaluation extended to the three dimensions of a triptych that had to be kept together: a prototype must provide proof that the technology is feasible, whilst at the same time ensuring its safety and economic viability. The transcription of technological innovation and the guarantee of safety to the prototypes built showed that the first two aspects increased the cost of the project.

### *3.2.3. Reducing the cost of the reactors of the future fleet*

As a result, EDF’s engineering teams in Lyon worked on defining the characteristics of the production reactors which would succeed Superphénix, looking for ways to simplify the prototype so as to meet competitiveness requirements. The economic evaluations which compared FBR technology with other means of electricity production were of increasing importance within a new context [14].

### *3.2.4. Assessing the long-term need for fast breeder technology*

During this decade, parallel to the Superphénix project, to the increasing importance of evaluations, and to thinking on how to design competitive industrial reactors, we can see a fourth type of event which affects fast breeder technology:

The slowing demand for energy, due to the economic slump which had followed the oil crisis, began to eat away at the urgency of developing a fast breeder fleet. Then in 1979, the Three Mile Island nuclear accident put a sudden stop to orders for nuclear reactors in the United States, and thus a very major slowdown in growth forecasts for nuclear power throughout the world. During the years that followed, the perspective of uranium depletion over the long term disappeared, and the urgency for a fast breeder programme diminished even further. As the decade advanced, plans for FBR industrial fleets were steadily postponed, with different representations depending on the actors and countries.

The way of envisaging the future is a determining factor for evaluation and for resulting decisions. During the previous period, developments of fast breeder technology were decided on the basis of an argument of necessity: it represented a source of inexhaustible energy which justified costly developments in order to prepare for the future; the prospective of growth in energy demand which had given rise to Superphénix seemed to have stabilized. But during the following decade, this argument of necessity had to coexist with economic evaluations which transformed fast breeder technology into something relative, contingent, for which it was necessary to evaluate the service provided in terms of the cost and possible alternatives.

Such an evaluation took place in an official context in the United Kingdom, such as the Royal Commission on Environmental Pollution in 1976 [13] or the House of Commons Committee of Public Accounts in 1984 [15]. In France, EDF's general management put on hold the decision to commit to industrial production, so as to have a full year of feedback on the operation of Superphénix: the principle of an industrial fleet was then conditionally validated, and postponed to a future date.

### *3.2.5. "Expert" militant criticism echoes academic criticism*

To finish the overview of this decade, we need to describe an "expert" militant criticism that was less visible than the radical activism at the origins of the demonstrations which marked people's memories. This "expert" criticism came from associations, university researchers, physicists and economists, who echoed the critical stances of Anglo-Saxon countries and insisted on an assessment of FBR safety and technology. Their criticism related to the modalities of the Superphénix project, in particular to its size: they felt that it was premature to build an industrial-size reactor. In France, the issue was debated in public (but not institutional) arenas, and the government confirmed the importance of energy independance, which justified Superphénix [16][17].

In conclusion, the decade from 1975 to 1986 for the development of FBR technology was not limited to the Superphénix worksite alone. Fast breeder technology was assessed from the standpoints of technique, economics and safety, three interdependent aspects of the same triptych. It was difficult to hold them together and to satisfy increasing performance requirements when the time frame for the industrial fleet was postponed to a later date: little by little, support for FBR technology became conditional.

The discourses assessing the success of the technology or, on the contrary, the "failure of fast breeder reactors" [14] were based on feedback from the first reactors: the shifting or the extrapolation from the reactor to the industrial fleet is a point that must be stressed. We can see the evaluation of fast breeder technology become an academic (especially in economic science) and institutional activity (institutions in the field of nuclear energy assess the safety, opportunity and time frame of an industrial fleet, on economic bases). These different fields were linked: the need that one might have for the technology rendered the imposed competitiveness criteria more or less strict – which was reflected in the calculations that included different trajectories for forecasts of the cost of uranium.

During this decade, what was asked of demonstrators was no longer to simply "prove that it works": the debate related to the capacity of the prototypes to provide proof that the technology could satisfy the evaluations with regard to the three aspects of technique, safety and economics, the criteria becoming more strict.

### *3.3. The operation hurdle: justifying safety and revamping objectives (1986-1997)*

During the third period of our FBR chronology in France, evaluation activity, now official, took place in a public context and was the subject of discussions in more wide-reaching arenas. To simplify matters, we will split this period in two, even though some of the developments were concomitant: first of all, the early years of Superphénix's operation were marked by assessments of its safety; then, discussions on the reconversion of Superphénix into a research facility gave a new turn to plant evaluation activities, the criteria for which were changing once again.

#### *3.3.1. The early years of Superphénix's operation were marked by evaluations of its safety*

It was in 1986 that Superphénix began operation as an industrial installation at the Creys-Malville site. It was operated as part of EDF's nuclear fleet, alongside plants using the more proven LWR technology. As for other innovative projects, early operations had their fair share of technical difficulties. In March 1987, there was a sodium leak in the fuel storage tank; it was replaced by a "fuel transfer unit" which only performed some of the original functions. The modified Superphénix was thus no longer truly representative of the way in which the industrial fleet to follow would be operated. In 1990, a pollution of primary sodium led to a lengthy phase of public questions about the safety and purpose of Superphénix: would it not be better to convert it into a research facility? During four years of investigation and debate, the plant was stopped, and major works were required for safety reasons. It finally started to operate again in 1994, but another leak, this time argon, meant a further six months of stoppage.

At the time that Superphénix was shut down by order of the government in 1997, written arguments designed to defend the plant put forward the notion of technical success: operating time was compared not to total time, but to time without any administrative blockages; the situation of the definitive stoppage after a year (1996) of satisfactory operation was harrowing; updates were completed. The technical success was in fact a legacy from the first period of our chronology, where the discourses on the opportunity offered by fast breeder technology converged and the challenge lay in the technical demonstration. One may therefore quantify the days of operation, kWh or other objectifiable variables to show the extent to which the technology met future energy needs. Yet the challenge of operating Superphénix was not just to demonstrate technical feasibility; in the post-1986 world, marked by the aftershock of the oil crisis, Chernobyl and Almeria sodium fire, discourses on opportunity were less unequivocal, and the challenge for the installation became that of "proving that it is safe". This challenge was a subject of negotiation, less tangible and objectifiable than the production of kWh.

Following the sodium leak in the fuel storage tank in 1987 and then the entry of air which oxidized the primary sodium in 1990, the safety of Superphénix was closely examined by experts from the Institut de Protection et de Sûreté Nucléaire (IPSN), and was the object of reports by the Autorité de Sûreté. What was new in the 1990s was the public nature of the evaluations and the debates: the safety of Superphénix was also debated in the French National Assembly, and assessed by the parliamentary office for the assessment of scientific and technological choices (OPECST), which allowed concerned groups and academic experts to voice their views. There was the issue of the fuel storage tank, the risks of hydrogen or of sodium fires, relating not to a research installation but to a reactor which was industrial by its size, by its operation as part of EDF's fleet and by the FBR fleet that it prefigured.

The trial of fast breeder technology through Superphénix provided partial answers to the question of its feasibility. Within this framework, the meaning of each event was interpreted in conflicting ways, depending on the point of view: "teething problems" or "significant design issue relating to safety" … The discourse of the project managers linked the negative aspects to the installation itself, which did not compromize the potential of the forthcoming fast breeder technology – and thus the utility of this

first step. But for the opponents, the very same events were proof of the non-viability of the technology and of the urgency of closing the plant. In France, in the 1990s, the debate took place in open arenas where various different arguments were exchanged and considered (in the United Kingdom a debate had taken place in the 1970s, concerning PFR and DFR prototype plants, with similar arguments: cf. [15]).

Furthermore, safety improvements led to significant costs which had a disastrous effect on the plant's financial results and hence on the technico-economic evaluation of fast breeder technology. In the 1970s, it was felt that the technology was ready to enter a commercial phase; in the 1990s, it was struggling to live up to its promise when confronted with the combined demands concerning technique, economics and safety, in a context where the need for an industrial fleet was diminishing. This was one reason why, at the start of the 1990s, public authorities began to consider converting the commercial demonstrator into a research facility.

### *3.3.2. Converting Superphénix into a research facility*

In May 1992, while Superphénix was stopped, the very opportunity for its operation was discussed during a public debate on "the possibility of restarting Superphénix and the future of FBRs". This debate, held under the auspices of the OPECST, took up the safety issue and examined the question of converting the plant into a research facility. Indeed, once the perspective of developing an FBR fleet had been pushed back to the long term, Superphénix's finality might have been to be part of a research programme and of experiments on the "incineration" of nuclear waste, within the framework of a law passed on this issue in 1991.

This project for conversion into a research facility meant that pluralist scientific commissions were entrusted with giving expert opinions, in 1992 and then again in 1996; they gave positive but unenthusiastic opinions on the utility of such a project.

In the 1990s, evaluation of Superphénix took place officially and openly within the public arena: no less than 18 official reports were produced between 1990 and 1997. The themes were a continuation of those of the previous decade, but the official nature of the assessments and the institutional framework were a radical novelty. These official reports concerned the safety, viability and finality of Superphénix, all of which were interlinked. In 1996, the Cour des Comptes produced an economic evaluation of Superphénix, which was considered to be a budgetary item; having determined the real cost overruns compared to what had been forecast, it assessed future income and expenditure in accordance with several options of availability and of the date for cessation of operation [18]. It was no longer a case of using these costs to extrapolate them to an industrial fleet: the aim was simply to assess the cost to the community of running such a research installation, and of asking, in budgetary terms, the question of whether or not to continue its operation.

The years of Superphénix's operation were studded with numerous events in the public arena which would take too long to enumerate (cf. [9]); we will content ourselves with questioning the link between the criteria for assessing Superphénix and its conversion into a research facility.

From the 1980s to the beginning of the 1990s, the criteria for "technical / economic / safety" evaluation constituted a triptych which was difficult to hold together, due to the innovative nature of the project which led to cost overruns. Furthermore, the more the project for an industrial fleet faded away, the stricter the technology's objectives of economic competitiveness became – in a climate of controversy where the modalities of extrapolation between "industrial prototype" and production reactor were being debated. It was no longer possible to maintain all of the aspects of the triptych in the objectives for competitiveness.

We put forward the following hypothesis: the choice of conversion into a research facility constitutes a form of response to the question of evaluation. With the reactor's change in finality, the economic evaluation criteria changed: Superphénix no longer prefigured a future industrial fleet, but in itself Superphénix constituted a research installation, which was set objectives of technical demonstration of

the feasibility of certain experiments. It was an attempt to return to the criteria of the first period, i.e. to demonstrate the feasibility of industrial electricity production and the feasibility of certain experiments, as shown by the discourse of certain project managers when the plant was closed. We might compare this outcome with that of Aramis: when the project for an industrial prototype is unable to satisfy the criteria of its evaluation, qualifying the installation as a research project confers upon it a more suitable framework. It was then assessed by the Cour des Comptes, in an accounting manner, as a public expenditure item, and no longer as an industrial installation whose purpose was to meet criteria of profitability and competitiveness in a near future.

## 4. Conclusions: challenging the evaluation criteria which evolve over time

Although evaluation appeared to be a key point, we would like to discuss certain aspects of this result: what is assessed? Using what modalities? The answers to these questions varied, depending on the period we were examining.

The evaluation of a prototype is linked to the question asked of it, and it is always useful to ask what a demonstrator must demonstrate. In the early stages of research, it was technical feasibility, within the framework of financing innovation; but it very rapidly became a case of holding several dimensions together: technique, safety, economy.

This research shows that the "severity" of an evaluation of a prototype evolves, depending on the vision for the future and on the industry's needs; if the necessity is shared, then the technical problems and cost overruns are acceptable. If the industrial fleet is only hypothetically required, and in the long term, the criteria for assessing the prototypes will be more severe.

In addition, in the evaluations of fast breeder technology we found considerable ambiguity, or more accurately, a permanent shift from the assessment of a prototype to that of the technology, which we would like to stress here (even if this theme deserves dedicated development). The process of innovation is such that an enthusiastic vision of a technology leads to the development of a prototype installation (cf. [10]). The prototype is qualified as promising or as being weighed down with cost overruns and technical issues, and these evaluations "shift" towards technology as a whole. We found three cases in point in these evaluations of FBR technology:

- The prototype is assessed as promising, so the technology will be promising;
- Even if the prototype is weighed down with cost overruns and technical problems, the technology is deemed to be promising (in which case the project promoters refer to "teething troubles");
- As the prototype is weighed down with cost overruns and technical problems, the technology is doomed to failure (extrapolation made by academic economists and by opponents).

The shifting of prototype evaluations from prototype to technology is a form of extrapolation - the criteria for which are debatable. Indeed, certain aspects of this extrapolation are very uncertain, such as the ratio used to forecast the cost of a production reactor compared to a prototype reactor, or the law predicting a drop in specific investment cost when the size of the reactor increases.

Finally, in this paper we have tried to show that the design choices for a plant such as Superphénix incorporate certain questions that the plant must answer and certain elements that it must prove. But during the long period under consideration, the questions asked of the plant evolved, moving away from the formulation contained in the original technical design. The issue of technical demonstration with the guarantee of safety was expanded to include commercial, European, economic and then research aspects, which had to satisfy evaluations which went well beyond the argument of "proving satisfactory technical operation".

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# Optimising environmental steps for the ASTRID project

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**Abstract.** Often felt as a stress in the 90s, projects can no longer ignore the ISO environmental directives governing design. As the 4th generation reactor project, the ASTRID team has decided to go further in its approach by taking full responsibility to optimise the environmental steps within a sustainable development strategy which systematically looks to minimise the environmental impact of the project per kilowatt-hour of electricity generated. This also illustrates the importance of public acceptance, by considering the environmental requirements as a positive way of ensuring a responsible performance label.
The paper describes a series of practical measures taken by the CEA as the project owner (identification of environmental requirements in the structuring documents, ecological inventory of the prospective site's biotope, impact assessment, etc.) and by their engineering partners in their pre-conceptual studies. For example, natural resources will be optimised and minor actinides will be incinerated (radioactive waste) in the core; design options will be compared on the basis of environmental criteria for the nuclear island; and in terms of non-nuclear installations, a High Environmental Quality approach will be used to design the building housing the turbine.

This paper describes the expected status of the nuclear island at the end of the pre-conceptual design phase; some reservations do remain in this paper since certain pre-conceptual design decisions will only be confirmed at the end of 2012.

## 1. Proactive approach by the project owner

The project to select a site for ASTRID – the GEN IV reactor – needs to begin early, to be well managed and to establish good communications with all stakeholders, including the regulators. It is widely recognised that the choice of a site for such a project involving a power plant is likely to be politically contentious. Even though France relies on nuclear power to generate a large proportion of its electricity, there still could be significant opposition to the construction of ASTRID. However, owing to the very nature of the concept, ASTRID is gradually becoming synonymous with sustainable development as it pursues the industry's objective of designing more environmentally friendly plants that reduce waste, recycle materials and use uranium-238, while improving safety features. The environment is therefore an important issue that should not be underestimated. Integrating environmental issues is a way of ensuring that the right site is chosen on the basis of safety, environmental, technical, economic and social factors, increasing the probability of completing the project on schedule. If not properly planned and executed, it is likely to result in major delays or even failure to complete the project. It is therefore important to set up a comprehensive management system and an efficient organisation to support the schedule and implementation of the siting activities, while ensuring that the required quality of the activities is achieved. It is also important to share good practices on these issues.

### *1.1. Organisation*

The project team was created in mid-2010 and the environmental aspects were carefully examined even during this early period devoted to structural choices. The CEA short-listed a site for the project, though its choice has not been finalised and other sites will be examined later before final selection. A 'site correspondent' was appointed to manage the first site studies focusing on the archaeological features, the biotope of the fauna and flora, and the chemical, hydrological and geological surveys. Feedback from various operational reactors was also taken into account.

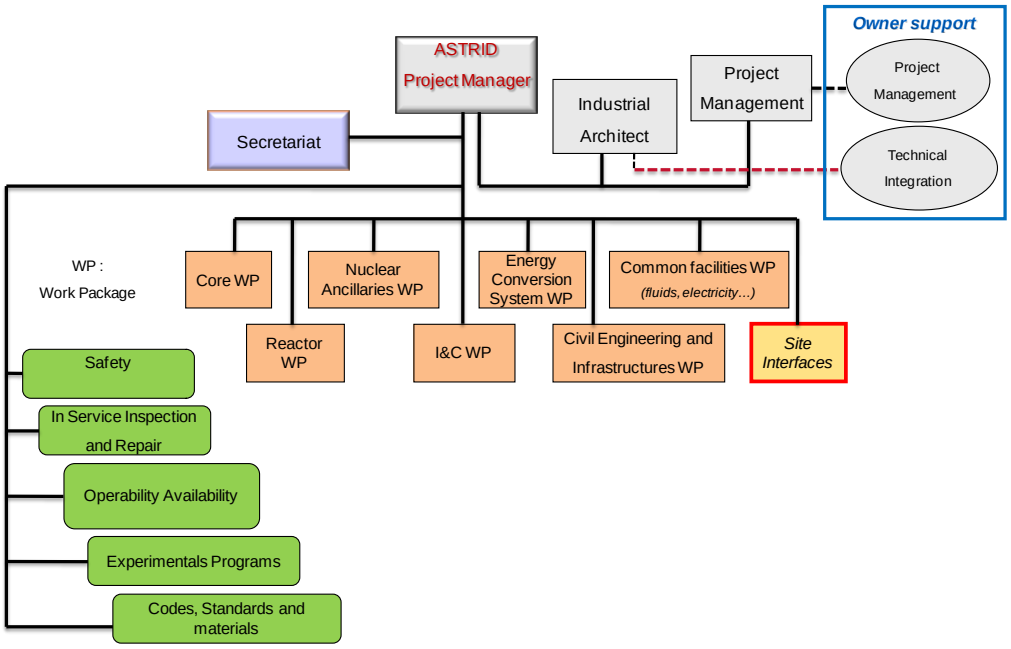


FIG.1. Astrid project organisation

### *1.2. Engineering input data*

The major engineering documents were drafted during this early period. These structuring documents specify the way to conduct the design studies, covering all the environmental aspects in great depth.

- The functional specifications define the objectives for sustainable development: using uranium-238; demonstrating transmutation; limiting waste and the activation of structures; recycling secondary sodium; regenerating cold traps; limiting the environmental impact of tritium discharges,
- In the project management specifications document, a dedicated requirement stipulates: "*Within the framework of their studies, the engineering firm will list all the technical measures taken in favour of the environment in a specific document. The result of these studies will be resumed in the technical specification of the concerned sub-systems*".
- The design package also asks for two documents to be produced by the end of 2014: 1) the principle, quantification and means of minimising waste and effluents; 2) the waste and effluent zoning plan.
- Finally, the performance management plan also identifies the above-mentioned environmental requirements.

### *1.3. Site studies*

The French regulatory obligations applicable in the field of assessing the impact of building a nuclear facility are defined by Decree No. 2007-1557 of 2 November 2007 and its ensuing texts. The impact assessment must include:

- Analysis of the initial condition of the area and the environmental features likely to be affected by the project;
- Analysis of positive and negative impacts on the environment, whether direct or indirect, temporary or permanent, and short, medium or long term. The different phases of facility construction and operation are to be taken into consideration.

The approach recommended by the French Ministry of Ecology, Sustainable Development and Energy, is to "*avoid, reduce, offset the environmental impacts*". The design should first seek to avoid any impact on the environment, including through the level of fundamental design choices. This phase is a prerequisite to minimising the environmental impact of the project, to reducing the impact and, where necessary, to compensating for any residual impact after avoidance and reduction. This approach implies taking the environment into account as early as possible during the project design, which also leads to optimising the investments required in this field.

Several examples of how this approach was implemented on a practical level are given for the ASTRID project.

<u>Impact on fauna and flora</u>

The main ecological impacts likely to be generated by a project include:

- Destruction of the flora or fauna and their natural habitats, some of which are protected,
- Disturbance around the area due to effluent discharges, transport, etc.

To apply the "avoid, reduce and offset" approach, a detailed ecological inventory first has to be done, which requires a sufficient period of time.
This involves compiling a bibliographical study, with the field survey being an essential step. All the habitats have to be explored systematically to cover the different ecological types and all types of vegetation, paying particular attention to the ecologically richest habitats (afforestation, wasteland, etc.). It includes the fluvial environments, with the river itself, and eventually the side channels and the oxbows.

Following state-of-the art practices, this requires the use of:

- Photographies and records,
- Temporary capture instruments (net),
- Temporary equipment (ultrasonic sensors, traps, photos, plaques on the ground) to identify animals that usually remain invisible,
- The sampling of plants on the ground and in the water.

In this way, a list of species and habitats is compiled taking care to identify the most remarkable features on maps and photographs. The role of micro-regional sectors can be highlighted (travel routes of mammals, amphibians, wintering waterfowl, etc.).
It is also necessary to identify the protection zones in the vicinity, either national ones such as ZNIEFF (natural areas of ecological interest, flora and fauna) or European ones such as Natura 2000 (network of special areas of conservation), to take into account their specific regulatory obligations.

The impact assessment results from the comparison between the project characteristics and this ecological inventory. The site selection for ASTRID is the first way to avoid or reduce the impact of the project: a site that already has an industrial or agricultural activity will be preferred, although there can be functional complementarities between natural and artificial environments (artificial light attracting insects for instance).
Compensatory ecological measures are possible to offset the project impacts. They can be offered in a larger ecological entity in the vicinity by improving its ecological capacity:

- Aging of afforestation areas with old trees that provide reservoirs of biodiversity. Cavities provide shelter for nesting birds and bats, rotten-wood-boring insects in trees are prey for other species, etc. So islets of aging trees, free of any human interference, could be put into place;
- Insofar as this does not affect the safety of the area (including fire prevention), maintaining 'islands' of shrubs will provide shelter and feeding areas for the target species.

Nonetheless, these compensatory measures have to be anticipated so the right areas can be found and so an environmentally sound management can be organised.

Other impacts

In case of major constructive projects, the Prefect of the Region often requires an archaeological evaluation. To avoid interferences with eventual archaeological remains, an early diagnostic by the Regional Archaeological Department will help to implement the buildings where the probability is the lowest. It will reduce the risk of delays, or even of late changes in the design of buildings or of the site.

Transport

Feedback from the construction of other reactors indicates that the transport of materials (sand, cement, aggregates, steel, etc.) could reach about 50 trucks per day, and the transport of large components could require several hundred exceptional convoys over the whole duration of the project.

If ASTRID is sited near a river, river transport would limit the increase in road traffic and its nuisances for the local populations, while reducing carbon emissions (2 to 4 times less than road transport). A river convoy of 5,000 tonnes is equivalent to 250 trucks by road or 125 cars by train. Exceptional convoys by road significantly disrupt traffic, with an increased risk of accidents, especially when crossing through small villages.
In areas where the road network is undersized, the use of river transport would reduce the needs for enlarging existing roads or creating new roads, thus increasing the artificial surface area taking up natural or agricultural lands.
River transport services could be provided not only at the beginning of the construction phase, but possibly also for the operation and dismantling of ASTRID, with thousands of tonnes of waste to transfer to specific storage sites.
However, the implementation of river services requires building one or several docks on the river bank, taking into account the flood risks, and the impacts on water withdrawal and releases, on the groundwater, and on the fauna and flora. Regulatory obligations should be anticipated to be able to use the river transport at the beginning of the construction phase.

FIG.2. River transport

Decommissioning and dismantling

Decommissioning and dismantling operations have to be considered well in advance in order to reduce the impact of the ASTRID site at the end of its life span. To leave enough leeway for the future dismantling policy, the project is considering the option of immediate dismantling, i.e. long-term decay is not required before dismantling activated and highly contaminated equipment. ASTRID aims to minimise the quantity and radiotoxicity of waste produced by the plant during its dismantling. For instance, the reduction in the use of Stellite is being studied to avoid generating $^{60}Co$ by activation in the reactor.
The overall design of ASTRID takes into account feedback from the dismantling of the Phenix and Superphenix reactors: the teams in charge of these projects have been participating in the design reviews. For instance, the need for a hot cell to dismantle experimental devices or fuel sub-assemblies before their transport has contributed to the decision to integrate a hot cell into the reactor support facilities.
Dismantling reviews of ASTRID will be organised at the major milestones of the project, which will strengthen the dismantling plan that has to be submitted with its licensing application (requesting authorisation to build the facility).

## 2. Engineering methodology

### *2.1. Core design*

In terms of the core design, two fast neutron reactor principles are favourable to the environment:

- On the one hand, the use of uranium-238: fast neutron reactors are able to transform all of the uranium-238 into plutonium-239. Thanks to this, it allows to use no more only 0.7% (as third generation reactors used to do) but the whole natural uranium to electrical purposes. The world availability in primary fissile resources can therefore be multiplied by almost 100. The stock of depleted uranium in France means we have the potential to meet our electrical needs for hundreds of years.
- On the other hand, the core is designed to be able to transmute minor actinides. Two modes are still being studied (the homogeneous or heterogeneous incorporation of minor actinides in the pellets); transmutation provides a way to close the cycle while reducing the volume and radiotoxicity of ultimate waste.

### *2.2. AREVA methodology developed for the nuclear island design*

#### *2.2.1 Purpose of the methodology*

The aim of the methodology developed by AREVA for the ASTRID project is to:

- Help designers to develop systems with a minimum environmental impact,
- Build an environmental management system that makes it possible – among others – to justify the choices of the different systems composing the nuclear island from an environmental point of view,
- Respond to great expectations in this field, particularly in terms of the current French regulations.

To reach these objectives, two tasks need to be managed:

- Definition of environmental specifications,
- Quantification of environmental performance levels for the different design options in order to offer an environmental criterion of choice as cost criteria for example.

#### *2.2.2 Defining the environmental specifications*

The most important French ministerial orders taken into account to define environmental specifications is the French order dated 7 February 2012 defining the general rules relative to licensed nuclear facilities [1]. Among others, this order specifies that the facility operator must ensure that:

- Its facility is designed, manufactured/built, operated, maintained, decommissioned and dismantled with the lowest level of risk and environmental impact deemed economically acceptable,
- The best available techniques are applied whenever possible,
- All measures are applied to offset any negative impacts that cannot be avoided or sufficiently reduced.

Environmental specifications are also defined under the AREVA best practices. A Green Project (or environmental friendly project) guide summarises the key environmental points to take into account in a new project.

#### *2.2.3 Quantifying the environmental performance of the different design options*

To compare the environmental performance of the different design options, the impact of these options on the environment is assessed.
That assessment comprises three tasks:

- Defining the environmental performance indicators,
- Prioritising options in relation to their environmental impact. Thus, efforts focus on the most important contributors,
- Performing a life cycle assessment of the most important contributors to evaluate the environmental performance of the different options.

##### *2.2.3.1 Defining the environmental performance indicators*

A preliminary list of the environmental performance indicators has been defined on the basis of the directives developed in the ISO 14031 standard and the best available techniques defined in the French Order of 26 April 2011 [2].
These indicators are summarised in the following table:

| Impact category | Environmental Performance Indicator | Unit | Comments |
|---|---|---|---|
| **Consumption and use of natural materials and resources** | Depletion of raw materials | kg Sb eq | |
| | Energy consumption | MJ eq | |
| | Water consumption | $m^3$ | |
| | Gas consumption | $m^3$, litres | Ar, $N_2$, $CO_2$, $H_2$, $NH_3$, … |
| | Others consumables | kg, $m^3$, … | |
| **Emissions/Releases** | Greenhouse gas emissions | kg eq $CO_2$ | releases into the atmosphere |
| | Air acidification | kg eq $SO_2$ | releases into the atmosphere |
| | Photochemical pollution | kg eq $C_2H_4$ | releases into the atmosphere |
| | Eutrophication | kg eq $PO_4$ | releases into the water |
| | Aquatic eco-toxicity | kg eq 1,4-DB | releases into the water |
| | Human toxicity | kg eq 1,4-DB | releases into the water |
| **Waste** | Solid/liquid waste | kg | |
| | Toxic/eco-toxic waste | kg | |
| | Radioactive waste | kg and Bq | |
| | Recyclability/ reusability | kg | |
| **Pollution** | Luminous pollution | Lux | |
| | Radiation | Bq or Sv | |
| | Thermal pollution | Watt | |
| | Olfactory pollution | $OUE/m^3$ | European olfactory unit / $m^3$ |
| | Noise pollution | dBA | |

TAB.1. Environmental performance indicators

Depending on the indicator, an equivalent unit could have been used. For example, the depletion of raw materials could also have been expressed in antimony (frequently used unit).

#### *2.2.3.2 Prioritising the most important contributors*

Each system composing the nuclear island is quickly evaluated to identify its potential environmental impact for each life step. Three mains life cycles are taken into consideration:

- Manufacturing,
- Operation,
- Dismantling.

The different environmental performance indicators are estimated for each life step. This evaluation is formalised in a rating sheet.
This rating sheet specifies the consumption rates, emissions, waste and pollution (if known). If not, a cut-off criterion, specific to each indicator, makes it possible to determine if the consumption, emission, waste production or pollution is significant or not.
At this stage of the ASTRID pre-conceptual design, no indicator could be precisely quantified especially since manufacturing and dismantling phases have not been completely defined.

Finally, the systems are prioritised in relation to the number of indicators retained. A second condition has to be checked to justify a life cycle assessment: the existence of different solutions that can be compared. If there are no different solutions to study, a short life cycle assessment is performed to compare the solution with the best available techniques so as to reduce the environmental impact and thus evaluate the

difference between them as stipulated in the Order dated 26 April 2011 on the implementation of the best available techniques as provided for in Article R.512-8 of the French environment Act [2]. Such a study is also useful for identifying the origin of the impacts and to orient improvements.

The preliminary results of this exercise show, for example, the relevance of analysing the life cycle of the sodium purification systems as a first priority.

### *2.2.3.3 Assessing the life cycle of the different options*

Once the systems of the nuclear island with an important environmental impact have been prioritised, a life cycle assessment can be performed.
The assessment is limited to assessing the different environmental performance indicators of the different solutions and alternatives in order to provide the designers with a criterion of choices.
The principle of this assessment is illustrated in the diagram below:

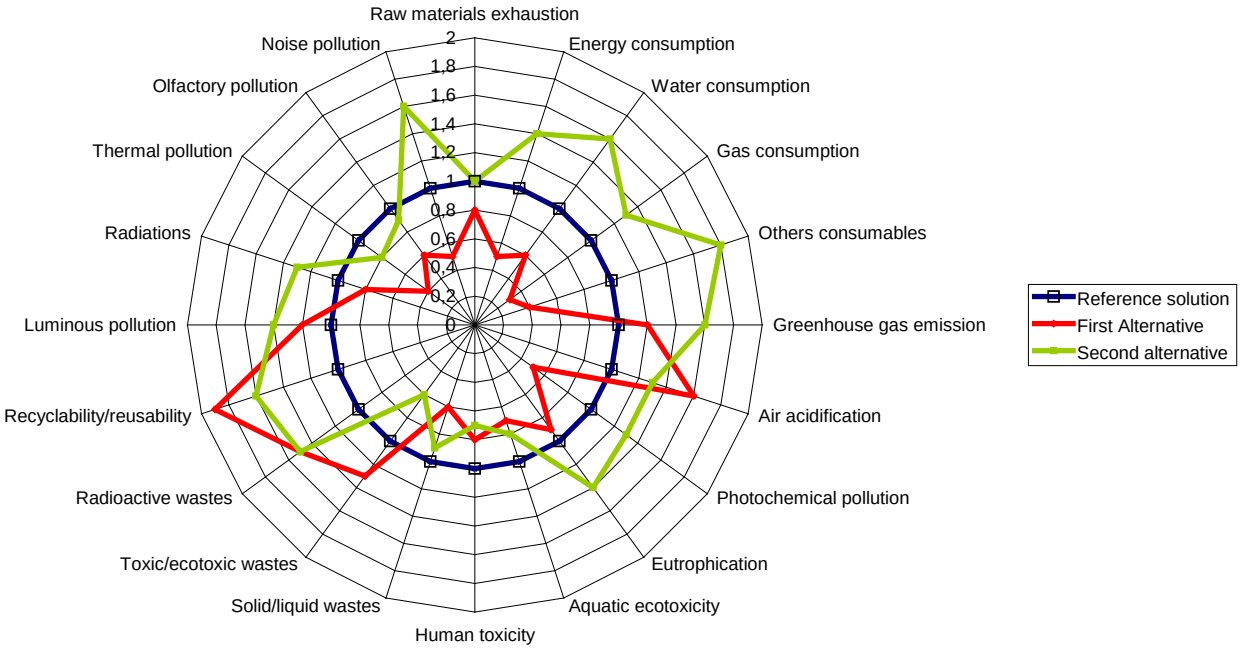


FIG.3: Principle of an options environmental assessment

This assessment also makes it possible to identify the weak points of the systems in order to optimise them and evaluate the environmental consequences.
In this example, recommendations could aim to improve the performance of the first alternative in terms of air acidification, recyclability/reusability, and radioactive and toxic/eco-toxic waste so it is preferable to the reference solution. These aspects will be analysed in-depth within the scope of a subsequent optimisation approach.

## *2.3. Balance of plant design*

The balance-of-plant design is based on two main features: the cooling reactor system and the High Environmental Quality approach for the turbine building design.

Cooling

The French Act of 7 February 2012 states that "*the open circuit cooling by fresh water from the environment is prohibited unless explicitly stated in the licensing decree authorising the construction of the facility. To qualify for this exemption, the operator must justify the acceptability of this practice, particularly with regard to the impact of thermal discharges on the environment.*"
Rivers will be greatly impacted by climate change. Snow will fall in thinner layers on the mountains, melt earlier and give way to rain. Rivers will be subject to severe flooding in late winter and long periods of drought until autumn. During the low-flow period, the warmer water could also affect the fauna and flora in the rivers.

Therefore, one option considered for ASTRID is cooling with air coolers, which would only require 1 to 2 $m^3/s$ to compensate for evaporation, compared with 30 $m^3/s$ required for open-circuit cooling. With this option, extra attention must be paid to the treatment of purge water released into the river due to the use of scale inhibitors to avoid deposits on pipes and the use of biocides to prevent the development of bacteria, including Legionella.

However, feedback from other sites shows that the use of cooling towers more than 100 m high with a plume of steam could be a problem in regions trying to develop tourism and agriculture due to their visual impact and the shadow impact of the plume. The possibility of using new-generation cooling towers is being studied for ASTRID since they can be better integrated into the landscape.
The Hamon Company, for instance, builds fan-assisted cooling towers with a low profile and invisible plume. On the site of Moorburg, in Germany, a single plume-abated cooling tower with 130 m base and a height of 60 m can cool two coal-fired 800 MWe power plants, instead of a cooling tower with a height of 120 m and visible plume based on a classic design.

FIG.4. Plume-abated fan-assisted cooling tower for the new Vattenfall power plants in Moorburg

Heat recovery

It might also be worth recovering part of the heat produced by ASTRID for heating purposes. This has already been done in the Phenix reactor. From 1974 to 1990 and from 2004 to 2009, the Marcoule Atomic Centre was mainly heated by steam at 30 bar produced by Phenix, representing only 5% of its steam output. New technologies like pre-insulated pipes have significantly reduced heat losses as low as 2% for every 100 km, making it possible to supply heat at long distances. This would also make it easier to build a redundant network for the maintenance periods of ASTRID.

## 3. Conclusion

This paper describes a series of environmentally conscious measures implemented by the project owner and the engineering companies. The most promising options are highlighted and must be strengthened during the conceptual design.
Our project is firmly rooted in a sustainable development approach that seeks to minimise the environmental impact per kilowatt-hour of electricity generated as early as the pre-conceptual design phase. The preliminary results are encouraging, if only in terms of the estimated consumption of nuclear material expressed per kilowatt-hour produced. Nonetheless, the subjects discussed in this paper will be developed in more detail during the successive optimisation phases of the project.

## NOMENCLATURE

ASTRID Advanced Sodium Technological Reactor for Industrial Demonstration

## REFERENCES

[1] French ministerial order of 7 February 2012 setting the generals rules relatives to licensed nuclear facilities
[2] French ministerial order of 26 April 2011 on the implementation of the best available techniques provided for in Article R.512-8 of the French Environment Act

# EDF research scenarios for closing the Plutonium cycle

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**Abstract.**

EDF long-term objective is to fully close the plutonium cycle, in order to achieve the nuclear industry sustainability. In order to compare the main strategies conceivable, we have simulated three types of prospective scenario families:
- Sodium cooled fast reactors (SFR) are progressively deployed as from 2040 for replacing ageing PWRs, so that the French fleet is entirely composed of SFRs at the end of the century (scenario S_1);
- SFRs deployment, started in 2040, remains partial, and the French fleet is thus composed of PWRs, loaded either with UOX or MOX fuels, and SFRs (scenario S_2);
- SFR deployment being delayed, advanced PWRs with a high conversion ratio allowing to multi-recycle the plutonium are deployed as from 2040 (scenario S_3).

These scenarios present various ways of using plutonium from spent MOX fuel which is not possible in the current fuel cycle. They allow at various degree limiting the need of natural uranium. Scenario S_1 is obviously the most efficient but relies on a very large deployment of SFR which may not be possible before the end of the century. Scenario S_2 is a good way to optimize the fleet with less advanced reactors: with only one fourth of SFR installed, we can reduce by half the need of natural uranium. Scenario S_3 shows that the use of SFR is not the only option to use MOX spent fuel and that High Conversion PWR can be considered to close the fuel cycle. This solution might be a more economic way to close the fuel cycle, but its performances in terms of uranium savings, fuel cycle capacities and high level waste production are low compared to scenarios S_1 and S_2. Therefore R&D to improve HCPWR designs in order to increase the conversion ratio and to decrease the plutonium inventory in reactor should be continued to make this solution more attractive.

 

# 1. INTRODUCTION

EDF long-term objective is to fully close the plutonium cycle, in order to achieve the nuclear industry sustainability. In order to compare the main strategies conceivable, we have simulated three types of prospective scenario families:

- Sodium cooled fast reactors (SFR) are progressively deployed as from 2040 for replacing ageing PWRs, so that the French fleet is entirely composed of SFRs at the end of the century;
- SFRs deployment, started in 2040, remains partial, and the French fleet is thus composed of PWRs, loaded either with UOX or MOX fuels, and SFRs;
- SFR deployment being delayed, advanced PWRs with a high conversion ratio allowing to multi-recycle the plutonium are deployed as from 2040.

For all these scenarios we will present and compare the main fluxes and inventories (uranium consumption, High Level Waste production, Pu inventories, etc), and the fuel cycle facility capacities that are requested.

Each scenario has been computed with the EDF R&D fuel cycle simulation code TIRELIRE-STRATEGIE [1] and optimized to meet various fuel cycle constraints such as using the reprocessing facility with long period of constant capacity, keeping the temporary stored mass of separated plutonium and MA under imposed limits and recycling older assemblies first.

# 2. GENERAL SCENARIOS ASSUMPTIONS

## *2.1. Reactors assumptions*

### *2.1.1. Pressurized Water Reactor ( current PWR and Generation III EPR)*

In our simulations, current PWRs and generation III EPRs differ only by the net yield (32.4% for PWRs versus 34% for EPRs), their lifetime (42 years versus 60 years) and the final burn-up of the spent UOX fuel (47 GWd/t versus 55 GWd/t). Spent MOX fuel burn-up is constant at 46 GWd/t. EPR main characteristics are listed in Table 1. The average Pu content in MOX fuel is given for a fleet with mono-recycling of plutonium. This value depends on plutonium quality and will therefore evolve as the fleet is turned into a symbiotic fleet. For safety considerations, the Pu content in PWR-MOX fuel must be kept under 12 % in order to keep a negative void coefficient. This is the reason why we cannot consider multi-recycling in the current freench fleet: the quality of the plutonium in PWR-MOX spent fuel is to low to be used again in PWR-MOX. No limitation has been considered on the fraction of PWRs fed with MOX fuel.

### *2.1.2. Sodium Cooled Fast Reactor (SFR)*

Sodium Fast Reactor CFV concept used for the study has been developed by the CEA [2]. This design is optimised to reduce the void effect in order to enhance the safety. It uses internal axial breeder zone, upper sodium plenum, upper absorbing zone, small core height and different heights depending on the radius. In order to reach a conversion ratio of 1.0 a lower axial blanket is included in the design. This design has been modified to create a 3600 MWth core, its main characteristics are listed in Table 1. In order to increase the breeding gain of SFR-CFV, two rows of radial breeding blankets of depleted uranium can be added to the design. It permits to reach a breeding gain of nearly 0,18.

### *2.1.3. High Conversion Pressurized Water Reactor (HCPWR)*

In order to be able to multi-recycle plutonium in light water reactors, we have to use an innovative design with an increased conversion ratio. Several designs of PWRs using heterogeneous arrangement in the assembly and in the core are being evaluated by CEA with the external size of the core consistent with the standard EPR pressure vessel. These designs try to combine the double advantage of a low moderation ratio and of the presence of blanket elements in the core. Among these concepts, one of them, the HCPWR-HEXA [3], has a moderation ratio lower than 1 (~ 0.83) which is obtained

assuming a lattice using a triangular arrangement. The core consists in 163 fuel assemblies (in each fuel rod, there is the presence of 3 blanket inserted axially between 4 fissle sections) and 43 fertile fuel assemblies. The fuel assembly consists in 439 fuel rods, 30 control rods in a hexagonal arrangement. We use the Fissile Inventory Ratio $FIR = \frac{Mass(^{235}U + ^{239}Pu + ^{241}Pu)_{EOC}}{Mass(^{235}U + ^{239}Pu + ^{241}Pu)_{BOC}}$ as a characteristic indicator of the convertion. For this design of HCPWR, the FIR is 0.86. The main characteristics of this core are given in Table 1. The plutonium content of this reactor is really high compared to other plutonium fueled reactors.

Table 1. Reactor characteristics

| Characteristics | EPR | | HCPWR | SFR |
|---|---|---|---|---|
| Thermal power | 4500 MWth | | 4250 MWth | 3600 MWth |
| Net electrical power | 1550 MWe | | 1450 MWe | 1450 MWe |
| Net yield | 34.4 % | | 34. % | 40.3 % |
| Load factor | 81.81 % | | 81.81 % | 81.81 % |
| | UOX | MOX | | |
| Core management | 4 * 366.6 | 3 * 366.6 | 4 * 369 EFPD | 5 * 400 EFPD |
| $^{235}U$ enrichment | 4.5 % | | | |
| Average Pu content (% of fissile assembly) | | ~ 9 % | ~ 21 % | ~ 24 % |
| Average Pu content (tons/Gwe) | | 6.8 | 14.5 | 8.7 |
| FIR | | 0.7 | 0.86 | >1. |
| Fuel average burn-up | 55 GWd/t | 46 GWd/t | 45.2 GWd/t | 121 GWd/t |

## *2.2. Fuel cycle assumptions*

In the current fuel cycle, MOX for PWR is produced with plutonium issued from reprocessing of PWR-UOX fuel and with depleted uranium. In the future cycles, plutonium can come from different sources but the content of plutonium in fresh PWR-MOX fuel must remain under 12 % for safety reason. MOX for SFR or HCPWR can be produced from recycling of plutonium from every type of fuel.

For every fuel types we have considered a two-year minimum for the fabrication time and we imposed no limitation to the fabrication plant capacity.

The reprocessing plant is supposed to be an evolution of the current reprocessing plant of AREVA NC at La Hague. The assemblies have to be cooled enough to be washed and transported before being reprocessed. We have considered a five-year minimum of cooling time for the spent fuel as an industrial limit. We consider losses of 0.1 % of U and Pu during the reprocessing. The High Level Waste (HLW): Fission Products (FP) and minor actinides are gathered in glass canisters (CSD-V). We have assumed that the future reprocessing plant will be able to deal with high plutonium content spent fuel, even if we know that the higher the plutonium content is, the more difficult are the dissolving and the separation.

## *2.3. Calculation tools*

The scenario simulations have been computed with EDF R&D fuel cycle code TIRELIRE-STRATEGIE [1]. This is a code simulating a pool of nuclear electricity generating plants with its associated fuel cycle facilities. It is used to assess various medium and long-term options for nuclear fleet evolution and fuel management. Main TIRELIRE-STRATEGIE results are fuel inventory and isotopic composition in all reactors and fuel cycle plants. The physics of the code is based on

evolution models for calculating the composition of irradiated fuel and equivalence models for calculating the fissile content of fresh fuel that is required to reach the targeted burn-up. Those models are validated over reference codes.

- PWRs fuel equivalence is computed with the code ECRIN and its isotopic evolution is computed with the code STRAPONTIN developed by EDF R&D [4].
- FBRs fuel equivalence is computed from $^{239}$Pu equivalent isotopic weight and its evolution is computed with a perturbation matrix; the isotopic weights and matrix coefficients are computed by a set of calculations with fast spectrum reactor neutronic code ERANOS [5].
- HCPWRs fuel equivalence is computed from $^{239}$Pu equivalent isotopic weight and its evolution is computed with a perturbation matrix; the isotopic weights and matrix coefficients are computed by a set of calculations with thermal spectrum neutronic codes APOLLO2-CRONOS2.

## 3. PLUTONIUM FUEL CLOSURE

### *3.1. Current Nuclear Fleet*

Before the transition to an advance fleet, the French fleet is supposed to be composed of PWRs fed with UOX and MOX fuel. UOX-PWRs accounts for 54.75 GWe over the 60 GWe of installed power of the fleet, the remaining 5.25 GWe coming from MOX-PWRs. The installed power of 60 GWe as well as the yearly electricity production of 430 TWe are kept constant during the whole simulation. The renewal of Generation II PWR fleet is carried between 2020 and 2050 at a pace of 2 GWe per year. The first 40 GWe are replaced by Generation III reactors, of EPR type. From 2040, each scenario uses its own deployment path towards plutonium fuel cycle closure.

As a comparison, we have computed the scenario “business as usual” S_BAU where the current fleet is replaced by a 60 GWe EPR fleet between 2020 and 2050 and remains the same after this year.

### *3.2. Full Generation IV deployment: scenario S_1*

The final objective of the fuel cycle closure is to minimize the use of natural uranium. In accordance with this aim, scenario S_1 simulates a deployment of a 60 GWe SFR fleet before 2100 as displayed in FIG. 1. This kind of scenario has already been studied by EDF R&D with a previous SFR design [6][7]. The deployment of 60 GWe of SFR-CFV requires the use of all fertile blankets during the transition phase (2040-2100).

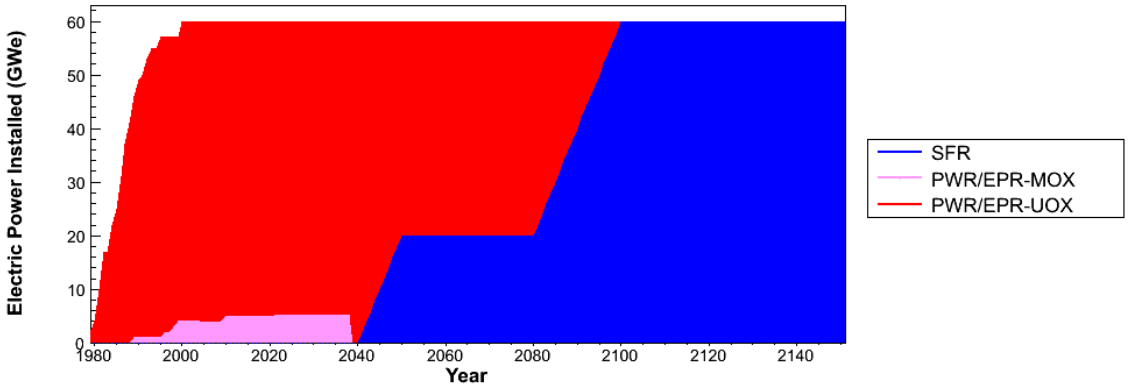


*FIG. 1. Power installed per type of reactor in scenario S_1*

### *3.3. Partial Gen-IV deployment: scenario S_2*

Scenario S_1 is the most direct evolution toward complete fuel closure but deploying an entire fleet of SFR may be technologically too challenging. A first step toward fuel closure is to be able to multi-recycle plutonium. EDF R&D as shown that it is possible with a symbiotic fleet composed of PWR-UOX, PWR-MOX and SFR optimized to reach an equilibrium between plutonium production and consumption [8]. This scenario minimizes the number of SFRs while maximizing the energy produced with PWRs-MOX which use the plutonium resource as fuel and are already widely used in the current french fleet. The plutonium cycle considered is displayed in FIG. 2: the plutonium produced in UOX-PWRs is used to feed MOX-PWRs (as it is already done in France) and SFRs are used to recycle plutonium from spent PWR-MOX fuel and to improve its quality so that it can finally be used together with spent UOX fuel to produce fresh PWR-MOX fuel. We used both row of radial fertile blankets in SFR through all the scenario in order to maximize the plutonium used in PWR-MOX. With SFR-CFV design, the equilibrium is 29.5 GWe of PWR-UOX, 14.5 GWe of PWR-MOX and 16 GWe of SFR. The transition to this fleet follows this pattern: from 2040 ageing Generation II PWRs are replaced by SFR until SFR installed power reaches its targeted value, and then by Generation III PWRs as displayed in FIG. 3. PWR-MOX is gradually growing from 5.25 to 14.5 GWe in 2065. The plutonium content of PWR-MOX fuel required to reach 46 GWd/t remains under 11 %.

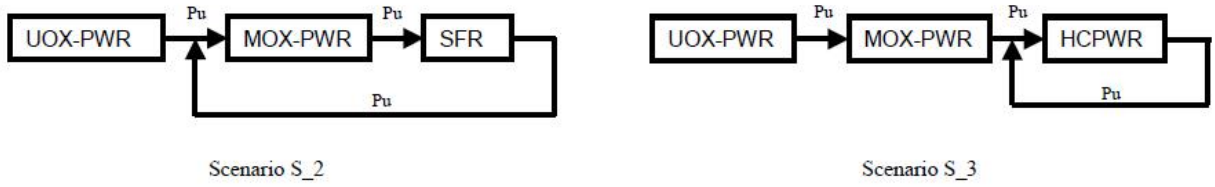


*FIG. 2. Plutonium fuel cycle for scenarios S_2 and S_3*

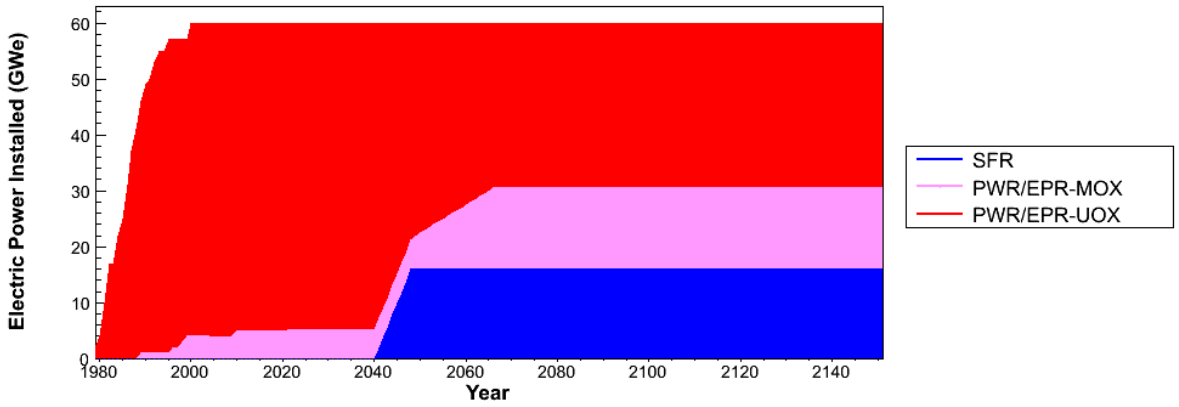


*FIG. 3. Power installed per type of reactor in scenario S_2*

### *3.4. Partial HCPWR deployment: scenario S_3*

The previous scenarios assume that technological and economical requirements for Generation IV industrial deployment will be meet in 2040, which may not be the case. Therefore we propose a scenario with the same objective that S_2 but with HCPWR instead of fast reactors. As current PWRs are not able to multi-recycle plutonium, the use of HCPWR is a roundabout solution to plutonium cycle closure. This solution is less efficient that the previous one as HCPWR has a FIR significantly

lower than 1 and furthermore it does not improve the quality of plutonium, thus the plutonium can only be recycled in HCPWRs as shown in the fuel cycle displayed in FIG. 2. The equilibrium can be reached in one deployment phase beginning in 2040. The fleet is composed at equilibrium of 43.2 GWe of PWR-UOX, 4.8 GWe of PWR-MOX and 12 GWe of HCPWR. The power installed is displyed in FIG. 4.

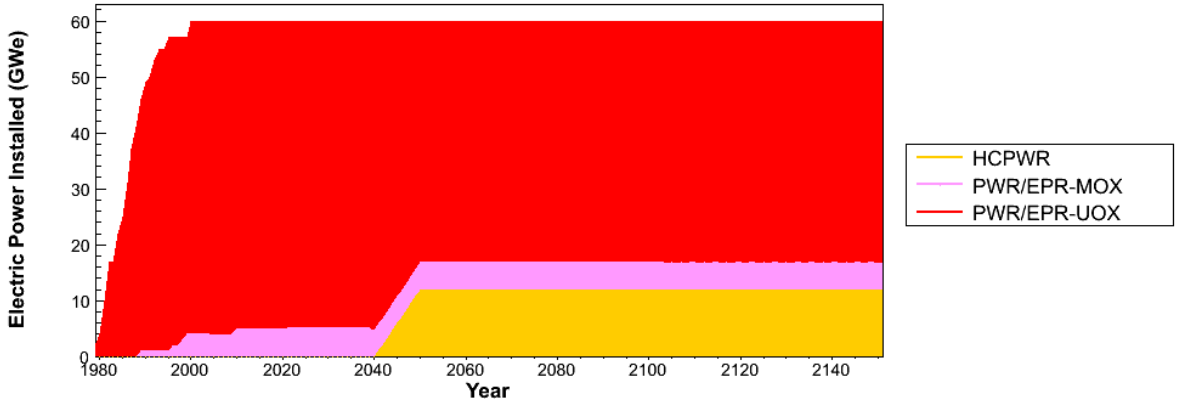


*FIG. 4. Power installed per type of reactor in scenario S_3*

## 4. RESULTS

### *4.1. Uranium consumption*

The multi-recycling of MOX fuel permits to limit natural uranium consumption. The table 2 gives the total uranium consumption in 2100 and 2150 and the yearly consumption at equilibrium for all the scenarios. We can note that the reduction of uranium consumption is still very limited at the end of the century, whatever the case considered. At equilibrium, scenario S_1 does not need natural uranium to create energy whereas the other scenarios still relie on this resource. Nevertheless, the mass of natural uranium preserved per GWe of advanced reactor installed compared to scenario S_BAU is a good indicator of the efficiency of the cycle. For scenario S_1 this ratio is 120 tons/year/GWe, for scenario S_2 it is 210 tons/year/GWe and for scenario S_3 it is 125 tons/year/GWe. This indicator shows that scenario S_2 could be considered as making a better use of its advanced reactors. Scenario S_3 has the same ratio as scenario S_1 but with HCPWR instead of SFR that are technologically more advanced.

Table 2.

| | Scenario S_1 | Scenario S_2 | Scenario S_3 | Scenario S_BAU |
|---|---|---|---|---|
| 2100 ($10^3$ tons) | 741 | 731 | 822 | 915 |
| 2150 ($10^3$ tons) | 741 | 927 | 1,109 | 1,279 |
| consumption at equilibrium ($10^3$ tons/year) | 0 | 3.91 | 5.74 | 7.26 |

### *4.2. Plutonium inventory*

The inventory of plutonium in cycle is displayed in FIG. 5. In S_1 and S_2, the inventories of plutonium are at equilibrium respectively in 2100 and 2070. While S_3 fleet composition does not evolve after 2065, the plutonium inventory continue to grow after 2150 (the equilibrium is near 800 tons). The explanation of this growth is that plutonium quality is decreasing during the multi-recycling in HCPWR as would show the growth of plutonium content in fresh HCPWR fuel in order to reach the same irradiation from 21 % in 2040 to 25 % in 2150. We can note that the plutonium inventory at

equilibrium in S_1 is really high: over 1100 tons which is the plutonium inventory for S_BAU in 2130. Scenarios S_2 and S_3 have also high plutonium inventories of the same magnitude (compared to nowadays inventory) but as we have shown in the previous section, S_2 has better cycle performance in term of uranium consumption.

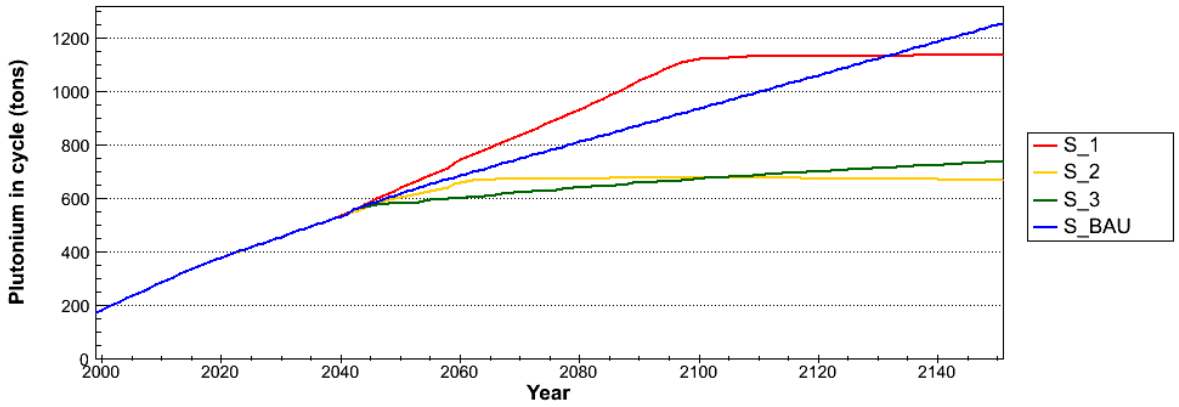


*FIG. 5. Plutonium inventory in cycle*

### *4.3. Spent fuel processing*

The spent fuel reprocessing capacity is driven by the need of plutonium to fuel reactors. We have computed these scenarios without any limit of reprocessing capacities. The FIG. 6 shows that S_3 has really high needs of reprocessing capacities to deploy the HCPWR because these reactors have a very high plutonium inventory. This means that limited reprocessing capacities would slow down the deployment of HCPWR. At equilibrium, the reprocessing capacity needed by scenarios S_2 and S_3 are roughly the same and is nearly the capacity in mass of the current plant of AREVA NC La Hague. For scenario S_1, the reprocessing capacity needed is half in mass but as we can see in FIG. 7., the plutonium content of spent fuel at equilibrium is 2.5 higher than for other scenarios of plutonium multi-recycling which will be a challenge for the reprocessing plant.

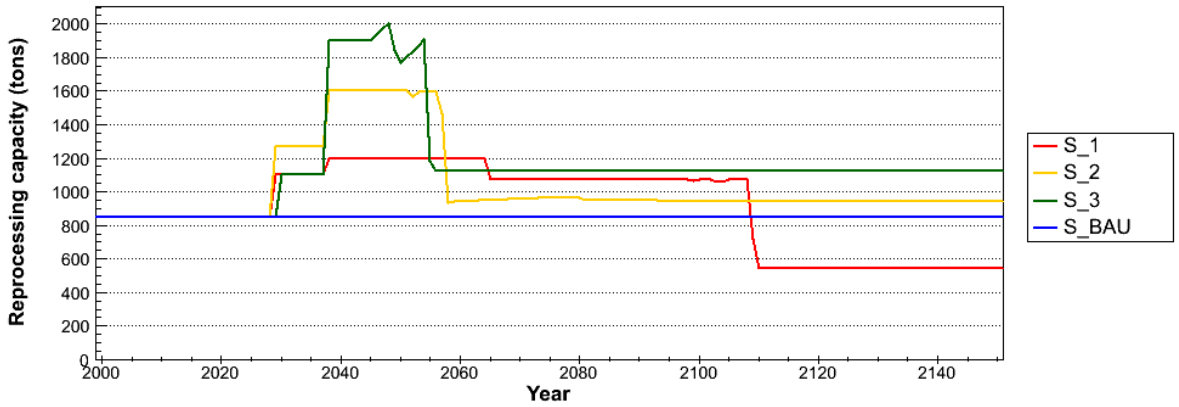


*FIG. 6. Reprocessing capacity*

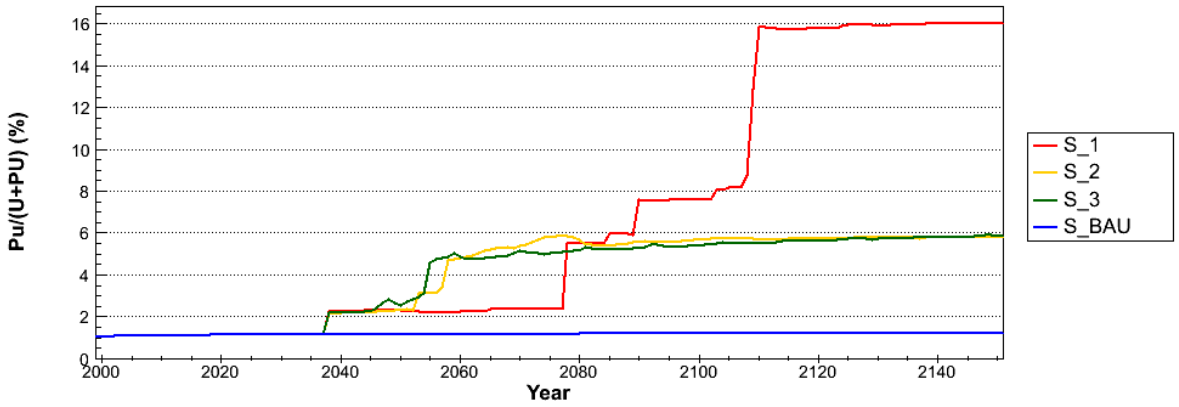


*FIG. 7. Ratio Pu/(U+Pu) in the reprocessing plant*

### *4.4. Spent fuel storage*

After irradiation in core, spent fuel assemblies are stored in cooling pools in order to let short lived isotopes to decay, so the decay heat and the neutron and gamma sources decrease. We have considered that the minimum time spent in storage before reprocessing is an industrial limit of 5 years. Spent fuel storage are displayed in FIG. 8. Current fuel cycle cannot use the plutonium from spent MOX fuel. Therefore, the sub-assemblies of PWR-MOX stay in cooling pool in S_BAU which increases the capacity needed. All scenarios using plutonium multi-recycling permits to keep the spent fuel storage under 19,000 tons which is consistant with the current french capacity for spent fuel. At equilibrium, the storage needed is really low compared to the current state: around 5000 tons for S_2 and S_3 and 2700 tons for S_1. Nevertheless, scenarios S_1 and S_2 have to use sodium cooling pools before spent fuel washing which is technologically more advanced than current water pools.

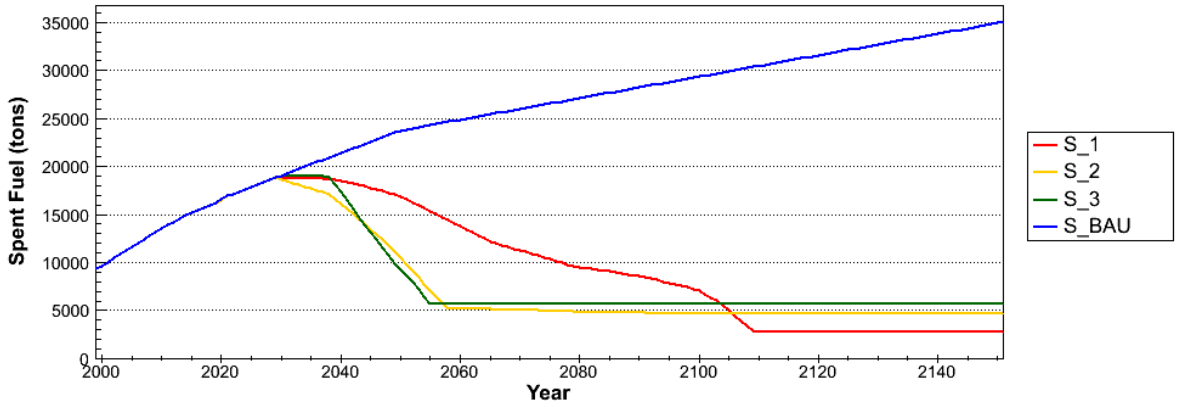


*FIG. 8. Spent fuel storage*

### *4.5. Waste fluxes and minor actinides inventories*

As all scenarios assume the same production of electricity, the only difference in the mass of fission products melted in glass canisters is the net yield of the reactors. As the net yield is higher in SFR than in light water reactors (40.3 % versus ~34. %), scenarios S_1 produces less FPs than S_2 that produces less FPs than S_3 and S_BAU. The number of glass canisters is roughly proportional to the FPs sent to

waste. In 2150, a total of 8150 tons of FP has been sent to waste through the reprocessing unit for S_1, 8450 tons for S_2 and 8740 for S_3.

Minor actinides (MA) are considered in our scenarios as final waste. At equilibrium, the mass of MA in cycle for scenarios S_1, S_2 and S_3 are low, respectively 14 tons, 22 tons and 27 tons. The total inventory (in cycle and in waste) of MA is displayed in FIG. 9. In 2150, scenario S_3 has a total inventory of MA of 570 tons which is more than a half more than scenario S_1 (370 tons in 2150). Scenarios S_2 and S_BAU produce the same amount of MA (500 tons in 2150).

We have displayed in Table 3 the minor actinides produced per TWhe for each reactor type and cycle in 2150. Thermal spectrum and even more epithermal spectrum favor neutron capture over fission of heavy nuclides which explains why PWR-MOX and HCPWR produce so more MA. Scenario S_2 and S_3 produces more than two times more MA than the scenario S_1. We can note that scenario S_2 produces more MA than the linear combination of reactor production displayed in the Table 3. The reason is that the quality of plutonium in fresh PWR fuel is lower than the one of plutonium from PWR-UOX spent fuel and the quality of plutonium in SFR fresh fuel is also decreased by the irradiation in PWR-MOX.

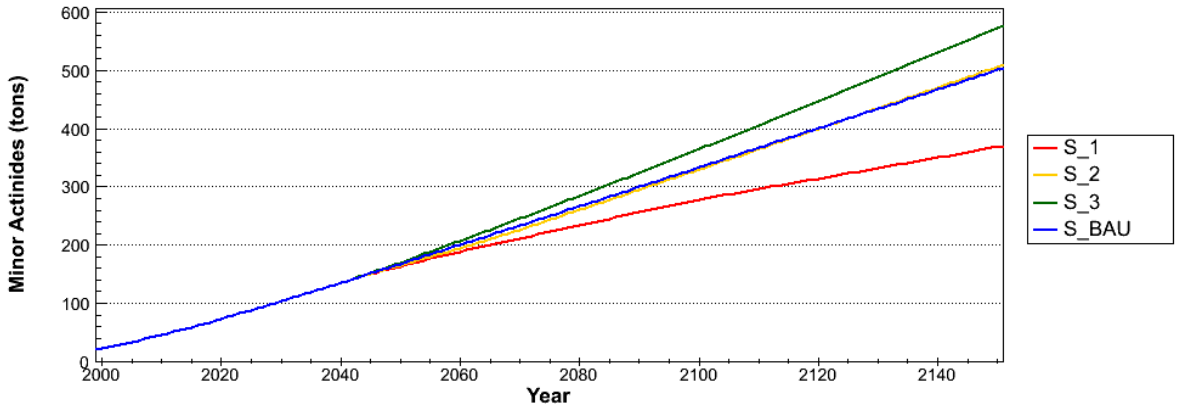


*FIG. 9. Minor actinides in cycle and in waste*

Table 3. Minor actinides production for each reactor type or cycle at equilibrium (kg/TWhe)

| | PWR-UOX | PWR-MOX[(1)] | HCPWR | SFR-CFV[(2)] | S_1 | S_2 | S_3 |
|---|---|---|---|---|---|---|---|
| Neptunium | 1.70 | 0.39 | 0.45 | 0.38 | 0.38 | 1.05 | 1.35 |
| Americium | 1.54 | 14.83 | 26.40 | 3.90 | 3.90 | 6.95 | 7.58 |
| Curium | 0.22 | 2.81 | 2.05 | 0.20 | 0.20 | 0.98 | 0.79 |
| MA | 3.46 | 18.04 | 28.90 | 4.48 | 4.48 | 8.99 | 9.71 |

[(1)]PWR-MOX fueled with plutonium from PWR-UOX spent fuel only.
[(2)]SFR-CFV fueled with plutonium from SFR-CFV spent fuel only.

## 5. CONCLUSION

The scenarios we have studied correspond to various way of using plutonium from spent MOX fuel which is not possible in the current fuel cycle. They allow at various degree limiting the need of natural uranium. Scenario S_1 is obviously the most efficient but relies on a very large deployment of SFR which may not be possible before the end of the century. Scenario S_2 is a good way to optimize the fleet with less advanced reactors: with only one fourth of SFR installed, we can reduce by half the need of natural uranium. Scenario S_3 shows that the use of SFR is not the only option to use MOX spent fuel and that HCPWR can be considered to close the fuel cycle. This solution might be a more economic way to close the fuel cycle, but the performances of its cycle are low compared to scenarios S_1 and S_2: the reduction of uranium consumtion is limited, the plutonium inventory in cycle is still really high, the high level waste (fission products and minor actinides) production is higher and the capacity of the reprocessing plant has to be very high during the deployment of the HCPWR. Therefore R&D to improve HCPWR designs in order to increase the conversion ratio and to lower the plutonium inventory in reactor should be continued to make this solution more attractive.

**AKNOWLEDGEMENT**

We would like to thank Frédéric Damian (CEA) and his team for the calculation of the HCPWR fuel evolution.

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# Perspectives of the SVBR-1000 during the Transition Period to the large-scaled energetic based on FBR

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*Presented by Y.S. Fedorov*

**Abstract.** The effective life of the SVBR-100 reactor working on the standard uranium fuel lasts about 6 years. The full charge of the core of SVBR-100 is about 9 tons of heavy metals, and in this case the reactor can function for a long time without new fuel supplies and SNF removal. The density of the standard UO2 fuel is 10.5 g/cm$^2$, and the breeding ratio is ~0.84. In case of the usage of more dense fuel the characteristic values improve. For example, using the cermet fuel UO2+10%Umetal with the density 11.4 g/cm$^2$ (this fuel was recently patented in Russia) or the nitride fuel with the density 13.2 g/cm2 the effective life can increase up to 8 and 11 years, and the breeding ratio can reach 0.87 and 0.9 respectively. The reprocessing of the SNF SVBR-100 can be realized or together with SNF VVER-1000 or together with MOX SNF BN-1200 or BREST (using uranium-plutonium fuel). Therefore, SVBR-100 can function both in NFC with thermal reactor and in Closed NFC with the fast reactor.

## 1. Introduction

Further development of atomic energy is connected with fast neutron reactors, though the launch of a significant park of such reactors will require resolution of a number of financial problems, as well as problems of technical organization of Closed Nuclear Cycle (CNC). Due to these and some other reasons the launch of large scale fast neutron energy production will be shifted to the future. Meanwhile first industrial fast neutron reactors of low capacity can be launched within several years. Among them is lead-bismuth cooled fast reactor SVBR-100 which can operate both in open nuclear cycle and closed nuclear cycle with thermal and fast neutron reactors.

SVBR-100 is a lead-bismuth cooled fast reactor [1]. The reactor thermal power is 280 MW equivalent to 100 MW of electric power. The height of the core is 900 mm, the diameter is 1643 mm, oxide fuel has 16% of enrichment of uranium 235. The effective campaign of SVBR-100 reactor operating on regular uranium fuel lasts for about 6 years. Full loading of the core is about 9 t HM., the campaign lasts for approximately 50000 effective hours, which means that the reactor can operate for a long time without new fuel and without extraction of spent nuclear fuel. The density of regular $UO_2$ fuel is 10.5 g/cm$^3$, when breeding ratio makes up ~0.84.

## 2. SVBR-100 reactor in Open Nuclear Cycle

One of the main features of SVBR-100 reactor is that using enriched uranium oxide fuel its effective campaign lasts for 6 years, which significantly surpasses the same indicator for other types of both thermal and fast neutrons reactors. With the use of fuel of higher density the effective campaign will rise significantly. For example, with the use of ceramic-metal fuel

$UO_2$+10% $U_{metal}$ with density 11.4 g/cm$^3$ [2] or mononitride fuel [3] with density 13.2 g/cm$^3$ the effective campaign will increase up to 8 and 11 years and breeding ratio – up to 0.87 and 0.9 accordingly.

The original technique on synthesis of special cermet fuel was developed. This fuel is based on uranium dioxide matrix containing nano-size inclusions of metallic uranium. Nano-particles of metallic uranium are formed during sintering of fuel pellets and they are located in pores of uranium dioxide grains [2]. Further investigation may provide development of high-dense fuel that would be very prospective for the use in SVBR.

Thus within 50 years of operation one SVBR reactor will accumulate about 75t of spent fuel from enriched uranium fuel and 60t of spent fuel from nitride fuel. After decommissioning all nuclear fuel can be transported for long term storage or reprocessing.

## 3. SVBR-100 reactor in Closed Nuclear Cycle with thermal-neutron reactors.

Composition of spent fuel from SVBR-100 reactor [1] and thermal neutron VVER-1000 reactor [4] is represented in Table 1. Table 1 shows that spent fuel from SVBR-100 contains more uranium fission isotopes than spent fuel from VVER-1000 reactor which makes it reasonable to reprocess it.

Table 1. Contents of uranium and plutonium isotopes in spent uranium fuel from SVBR-100 (burnup – 10%, cooling time 7 years) and VVER-1000 (burnup 50 GW·d/t, cooling time 5 years) kg/t HM.

| Nuclide | SVBR-100 | VVER-1000 |
|---|---|---|
| U-235 | 103.1 | 8.85 |
| U-236 | 13.3 | 5.79 |
| U-238 | 840.5 | 973 |
| Pu-238 | 0.095 | 0.28 |
| Pu-239 | 40.7 | 6.65 |
| Pu-240 | 2.28 | 3.16 |
| Pu-241 | 0.06 | 1.39 |
| Pu-242 | - | 0.87 |
| ∑ Pu | 43.14 | 12.36 |
| U-235+Pu-239+Pu-241 | 143.9 | 16.89 |

It can be reasonable to reprocess spent fuel from SVBR-100 together with spent fuel from VVER-1000 as it is practiced at RT-1 (PA “Mayak”) where spent fuel from BN-600 is reprocessed together with spent fuel from VVER-440 [5]. Since SF from SVBR-100 contains less even uranium and plutonium isotopes and more fission nuclides than SF from VVER-1000, the extracted uranium and plutonium can be used to produce fuel for VVER.

According to calculations, combined reprocessing of 1t of SVBR-100 SF together with 3t of VVER-1000 SF will give 4t of uranium-plutonium regenerate (restore waste) which can be used to produce fuel for VVER-1000. Total amount of fission nuclides will be about 4.9% of which plutonium isotopes will make up 1.6%, the rest – U-235. Such fuel composition corresponds to REMIX fuel [4] for VVER-1000 reactors which were obtained without using natural enriched uranium.

## 4. SVBR-100 reactor in closed nuclear cycle with fast-neutron reactors

With the beginning of construction of commercial fast neutron reactors such as BN-1200 and BREST-1200 SVBR may shift to MOX fuel or nitride fuel, which will have to be produced in significant amounts without any limits. Then it will be possible to reprocess SF received from operation of SVBR-100 in combination with either MOX SF or nitride SF at the reprocessing power that meets the needs of fast neutron reactors.

## 5. Conclusion

SVBR-100 is a fast neutron reactor with can operate with almost any kind of fuel (oxide, ceramic-metal or nitride fuel). The launch of such reactors requires a special infrastructure which is necessary for large commercial reactors. SVBR-100 fuel cycle easily fits the existing infrastructure of thermal reactors. Thus SVBR-100 is an ideal reactor in the period of transition from thermal-neutron reactors to fast neutron reactors.

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# Fast reactors and nuclear nonproliferation

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**Abstract.** Problems are discussed with regard to nuclear fuel cycle resistance in fast reactors to nuclear proliferation risk due to the potential for use in military programs of the knowledge, technologies and materials gained from peaceful nuclear power applications. Advantages are addressed for fast reactors in the creation of a more reliable mode of nonproliferation in the closed nuclear fuel cycle in comparison with the existing fully open and partially closed fuel cycles of thermal reactors. Advantages and shortcomings are also discussed from the point of view of nonproliferation from the start with fast reactors using plutonium of thermal reactor spent fuel and enriched uranium fuel to the gradual transition using their own plutonium as fuel.

## 1. Introduction.

For more than 50 years, the world of nuclear power has passed a long way in its development and wide geographical expansion of use. After all this time, the main nuclear technologies having changed with improvements in economics and safety remain the heritage of military activity and so demand close attention to providing a regime for assuring nonproliferation.

Development of the system of nuclear power and its accompanying infrastructure leads to the generation of a large quantity of fissile and fissionable materials and creates motives or preconditions in which the peaceful use of material in the nuclear fuel cycle (NFC) can be switched to the creation of nuclear weapon (NW) or is stolen for the purpose of creating nuclear explosive devices (NED). At this, switching of materials of NFC can be carried out by certain state entities or the theft of materials by terrorist or other subnational criminal elements and groups.

The existence of large quantities of fissile and fissionable materials in the NFC of nuclear power induces some short-sighted politicians to express a very close connection between the application of peaceful nuclear power and the acquisition of nuclear weapons, and thus the inevitability of the diversion of materials from its use for peaceful nuclear power to outside the NFC for other applications. This has led to the often heard call for the termination of the use of peaceful nuclear power because under cover of nuclear power it would be easier for threshold states to implement a clandestine nuclear weapons program.

Indeed, nonproliferation is an important factor which needs to be considered in choosing ways for nuclear power development. However, it should be kept in mind that nuclear power is a long and expensive way to nuclear weapons. All established nuclear weapons states developed their nuclear weapons when they did not have nuclear power.

 

If a state makes a political decision to acquire nuclear weapons, it would be able to choose more straightforward and less expensive options. The development of a uranium nuclear explosive device requires resources and separation work about 2-3 orders of magnitude lower than the start-up of one nuclear reactor. Similarly, a reactor about 2-3 orders of magnitude less powerful than a typical power reactor is capable of producing plutonium in an amount sufficient for a plutonium explosive device during a year. Both these ways have been demonstrated (Pakistan and India).

Nuclear nonproliferation is a multilateral problem. Its peculiarities should be considered with respect to the status of a state: (1) established nuclear weapons states; (2) de facto nuclear weapons states; (3) states possessing key nuclear technologies; and (4) states developing or planning to develop nuclear power. Different approaches might be needed to prevent horizontal proliferation and counter nuclear terrorism from non-governmental organizations. Nuclear reactors and nuclear fuel cycle technologies also differ in the risk of proliferation.

In a number of publications claimed that proliferation risks for fast reactors are much higher than that for thermal reactors [1]. Also, there is a diffused opinion that the closed nuclear fuel cycle inherently bears the risk of proliferation.

The decision of President Carter Administration based on the danger of nuclear proliferation when using plutonium in fast reactors resulted in essential harm to the development of this technology in the U.S. and no less than in some other countries. Besides, such efforts, generally thanks to the U.S, in the world created a negative opinion on fast reactors and their NFC as being the most dangerous from the standpoint of nuclear proliferation.

At the same time there have been carried out a number of studies which demonstrate that technological proliferation barriers can be implemented for fast nuclear power and closed fuel cycle as well [2]. Of course, technological barriers should be combined by institutional measures including advanced control and accounting system, and improved physical protection system.

It is also important to reduce or eliminate stimuli for states in acquiring nuclear weapons. The IAEA Director General Mohamed ElBaradei told a news conference at the Japan National Press Club in Tokyo, Japan 30 November 2006, that countries try to develop nuclear weapons because they sometimes feel insecure, they sometimes feel threatened and they sometimes feel that nuclear weapons will bring some power and prestige. And he added that we need to work on all these motivations.

We need have a comprehensive set of requirements for countries developing nuclear power, which would be free of political preferences. It is necessary to develop both incentives for strict adherence to these requirements as well as penalties for their violation.

## 2. Nuclear materials.

A certain risk of proliferation through technologies and materials of nuclear power can not be excluded completely. In a nuclear fuel cycle there is a large quantity of nuclear materials, including fissile and fissionable ones – many hundreds and thousands of tons. For manufacturing of one nuclear bomb there are enough tens of kilograms and even less.

Fissile materials of a nuclear fuel cycle at some additional processing can potentially use for production of the nuclear weapon or nuclear explosive devices. From this point of view interest is represented, of course, by the materials, capable to support chain fission reactions. In other words, the material should have not rather big critical mass. This is a key property of a material that is necessary, but not a sufficient condition, for a material choice for purpose specified above.

So far as the main fissile materials for production of the nuclear weapons high-enriched uranium and plutonium produced especially for this purpose – weapon plutonium have been used.

While high-enriched uranium and plutonium can be used for a NED, between them there is a basic distinction from the point of view of decrease in risk of proliferation. It is that the high-enriched uranium can be "mechanically" diluted with the low-enriched or natural uranium with the resulting loss of its ability to sustain an uncontrolled chain fission reaction sufficient to effect an explosion. To restore this ability it is necessary to use uranium enrichment technology and appropriate equipment to which access is very limited. Separation of plutonium from other elements, with which it can be mixed, is much simpler, since only chemical processing is required.

On the other hand, if to compare nuclear physical properties of uranium and plutonium, it appears that plutonium unlike uranium possesses certain properties of self-protection which can complicate creation of NED, such as neutron background from spontaneous fission, decay heat from alpha and beta decay along with spontaneous fission, and the radioactivity belonging to such properties. Certainly, manifestation of these properties depends on the amount of certain isotopes in plutonium. The enriched uranium also possesses similar properties, but they are orders of magnitude below those of plutonium, and therefore can not render appreciable influence on the creation of a NED using high-enriched uranium.

In Table 1 nuclear physical property of uranium and plutonium isotopes for comparison are listed [3].

Table 1.

Nuclear physical property of uranium and plutonium isotopes (rounded off numbers)

| № | Isotope | Half-life, year | Neutron background, n/sec·kg | Decay heat, W/kg | Bare critical mass, кg | Activity, Ci/кg |
|---|---|---|---|---|---|---|
| 1. | U-232 | 68.9 | $4.0\cdot10^{-3}$ | $7.2\cdot10^{2}$ | ~5 | $2.2\cdot10^{4}$ |
| 2. | U-233 | $1.6\cdot10^{5}$ | 1.2 | 0.3 | 16 | 9.6 |
| 3. | U-234 | $2.5\cdot10^{5}$ | 7,2 | 0.2 | ~40 | 6.2 |
| 4. | U-235 | $7.0\cdot10^{8}$ | 0.4 | $6.0\cdot10^{-5}$ | 50 | $2.2\cdot10^{-3}$ |
| 5. | U-236 | $2.3\cdot10^{7}$ | 4.1 | $1.8\cdot10^{-3}$ | ~170 | $6.5\cdot10^{-2}$ |
| 6. | U-238 | $4.5\cdot10^{9}$ | 10.8 | $7.2\cdot10^{2}$ | - | $3.4\cdot10^{-4}$ |
| 7. | Pu-238 | 87.7 | $2.7\cdot10^{6}$ | $5.7\cdot10^{2}$ | 10 | $1.7\cdot10^{4}$ |
| 8. | Pu-239 | $2.4\cdot10^{4}$ | $2.0\cdot10^{1}$ | 2.0 | 10 | $6.2\cdot10^{1}$ |
| 9. | Pu-240 | $6.6\cdot10^{3}$ | $1.0\cdot10^{6}$ | 7.1 | 36 | $2.3\cdot10^{2}$ |
| 10. | Pu-241 | 14.4 | 2.4 | $1.3\cdot10^{2}$ | 13 | $1.0\cdot10^{5}$ |
| 11. | Pu-242 | $3.7\cdot10^{5}$ | $1.8\cdot10^{6}$ | 0.1 | 92 | 4.0 |

High-enriched uranium is most attractive for nuclear weapons development and hence it is most hazardous from nonproliferation point of view. Its low neutron emissions allow a simple design of gun type. This design does not require sophisticated adjustment and testing, and such a nuclear device was used without testing, for example, in the bomb dropped on Hiroshima. The development of a nuclear

weapon based on enriched uranium would apparently be the most accessible for non-governmental criminal and terrorist groups [4].

That is the reason why many scientists believe that uranium enrichment technology poses severe threat to nonproliferation and its abandoning would become the most valuable contribution to nonproliferation regime.

Plutonium devices require a much more sophisticated design and more sophisticated technologies with a need to exploit implosion schemes which have to be carefully tested in appropriate experiments. The higher neutron emission, radioactivity and generated heat, it is the more difficult to build an adequate nuclear device because requirements for its design get more complicated and it becomes necessary to use remote manufacture and assembly technologies, high heat-resistance chemical explosives and heat removal equipment.

Nevertheless, plutonium of any kind can with more or less difficulties be used in nuclear weapons. In 1977 the U.S. Department of Energy reported a successful test of a reactor-grade-plutonium-based device held in 1962.

## 3. Start-up and operation of plutonium-fuelled fast reactors.

At a dawn of nuclear power E. Fermi put forward idea that the first fast reactors will be started up on plutonium which will be produced in thermal reactors with peaceful uses of nuclear power (civil plutonium).

Provided the first fueling with mixture of plutonium (for example, recovered from the spent nuclear fuel of thermal reactors) and natural or depleted uranium, plutonium can then be produced in amounts to this or that extent higher than that initially loaded. If civil plutonium is used for start up of fast reactors the same quality of plutonium with large content of the highest even-numbered isotopes (non-weapons grade) will then be produced during operation in the reactor core. In this context fast reactors do not require fuel with enriched uranium due to their peculiarities related to plutonium breeding.

In Table 2 isotopic compositions of civil plutonium which produces in the irradiated fuel of thermal reactors of various types are presented, here for comparison the isotopic composition of weapon plutonium is presented too [5][6].

Table 2.

Isotopic compositions of civil plutonium in irradiated fuel of thermal reactors of various types

| Reactor type | Fuel burn-up, GW·d/t | Plutonium isotopic composition, % | | | | |
|---|---|---|---|---|---|---|
| | | Pu-238 | Pu-239 | Pu-240 | Pu-241 | Pu-242 |
| PWR | 33 | 1.6 | 56.5 | 23.8 | 12.8 | 5.4 |
| PWR | 60 | 3.8 | 51.8 | 23.0 | 14.2 | 7.2 |
| PWR | 100 | 7.8 | 47.0 | 21.7 | 14.7 | 8.8 |
| MAGNOX | 5 | ~ 0 | 68.5 | 25.0 | 5.3 | 1.2 |
| CANDU | 7.5 | ~ 0 | 66.5 | 26.5 | 5.5 | 1.5 |

As it can be shown from the presented data, there is a considerable amount of the highest even-numbered isotopes of plutonium-240 and-242 in civil plutonium which give a neutron background

from spontaneous fission by far exceeding that for weapon-grade plutonium. In addition, rather high content of plutonium-238 leads to a considerable decay heat in civil plutonium, and decay of plutonium-241 leads to the high level of radiation.

Fuel on the basis of civilian plutonium irradiated in fast reactor will contain the plutonium which isotopic composition essentially in principal does not differ from isotope composition of initial plutonium in fresh fuel.

Moreover, according to the «Agreement between the Government of the United States of America and the Government of the Russian Federation concerning the management and disposition of plutonium designated as no longer required for defense purposes and related cooperation" from September 2000 and the Protocol from April, 2010 with changes introduced into this Agreement the Russian side will utilize 34 tons of the plutonium removed from defense programs in MOX fuel using fast reactors BN-600 and BN-800. Under operating modes in these reactors, MOX fuel with plutonium of weapon quality will be irradiated to satisfy "spent fuel standard". "Spent fuel standard" definition includes such degradation of initial isotopic composition of plutonium of weapon quality at which the relation of concentration of plutonium-240 to plutonium-239 will be more than 0.1.

Quite another matter is with fast reactors which have internal or external breeding zones – blankets. It is well-known that the blankets of fast reactors produce plutonium with an isotopic composition close to weapon-grade.

The refusal of blankets in fast reactors, on the one hand, eliminates production of plutonium that would be close to weapons-grade plutonium in composition, on the other hand, leads to a decrease in the reactor breeding ratio, and as a result leads to the loss of additional plutonium which could be used to speed the expansion of fast nuclear power. Nevertheless, fast reactors without blankets are capable of fuel self-sufficiency and can become a source of plutonium for starting new fast reactors at a rather limited pace of development, though.

Along with institutional measures, the following solutions of this problem are possible:

- joint management and reprocessing of spent fuel subassemblies from the core and irradiated blanket subassemblies;
- excluding separation of pure plutonium when reprocessing spent fuel and blanket assemblies, for example, the mix consisting of 50 % uranium and 50 %plutonium in separated material streams;
- elimination of breeding blankets in fast reactors which are designed for export to the countries which do not have nuclear weapon;
- organization of international centers for rendering services in the nuclear fuel cycle;

Various options of denaturation of plutonium proposed in recent years do not consider problems related to the management with such plutonium in the nuclear fuel cycle of nuclear power and do not take into account economic consequences of such denaturation.

## 4. Start-up of fast reactors using enriched uranium fuel.

Recently, at least, in Russia, the option is being investigated of the start up of fast reactors on enriched uranium with the subsequent gradual transition to a mix of uranium-plutonium fuel with the use of its own bred plutonium. Such an option allows fast reactors to be independent of availability of plutonium from the reprocessing of thermal reactor fuel.

In this case, during reactor start up and especially in first refueling plutonium with small amounts of the highest isotopes will be produced in the core. Later on with transition to a mix of uranium-plutonium fuel the composition of discharged plutonium will tend to the equilibrium non-weapons grade. Certainly, if fast reactor has a blanket, it will also produce high-grade plutonium.

At start up of fast reactors on uranium fuel with the subsequent transition to mixed uranium-plutonium fuel, the sensitive technologies specified above will be used: uranium enrichment and reprocessing of spent fuel. And, two hazardous materials – enriched uranium and high-grade plutonium – will be present in the fuel cycle.

The isotopic compositions of plutonium which is produced in enriched uranium fuel of the core in BREST type lead cooled fast reactor, estimated on the basis of preliminary calculations, are given in Table 3 [7]. The same figures of isotope concentrations to some extent has plutonium produced in enriched uranium fuel of the core in BN type sodium cooled fast reactor.

Table 3.

Plutonium composition in enriched uranium fuel of BREST-type fast reactor, %

| Plutonium isotope | In uranium fuel after the first micro-campaign | In uranium fuel after full campaign |
|---|---|---|
| Pu 238 | 0,02 | 0,11 |
| Pu 239 | 98,73 | 95,47 |
| Pu 240 | 1,24 | 4,28 |
| Pu 241 | 0,01 | 0,14 |
| Pu 242 | 0 | 0,003 |

## 5. Uranium enrichment and nuclear nonproliferation.

The development of the option to start-up fast reactors on enriched uranium is drawing more attention to the problem of nuclear proliferation by way of uranium enrichment capability. In this connection appropriate studies have been carried out which allows the formulation of the following main conclusions: the state-proliferator, which obtains enrichment technology in some fashion, will aspire to carry out the plans to produce nuclear weapons at the expense of reaching as high of uranium enrichment as possible due to the following reasons:

- needs comparably small amount of fissile material to manufacture "faster" device, that
- leads, as a rule, to a bigger energy yield during explosion;
- needs a smaller quantity of an initial material (natural uranium);
- needs the less number of SWU;
- needs less quantity of electric energy;
- needs less time.

Function of attractiveness of the enriched uranium will have the following appearance:

$$A(x) = \mathrm{mod}\left[1.8 \times (2x-1) \times \ln\left[x/(1-x)\right] - 0.9\right] \qquad (1)$$

Substituting figures of uranium enrichment x, the corresponding attractiveness can be calculated and graphed. Figure 1 presents a graph of the function A(x) versus enrichment x.

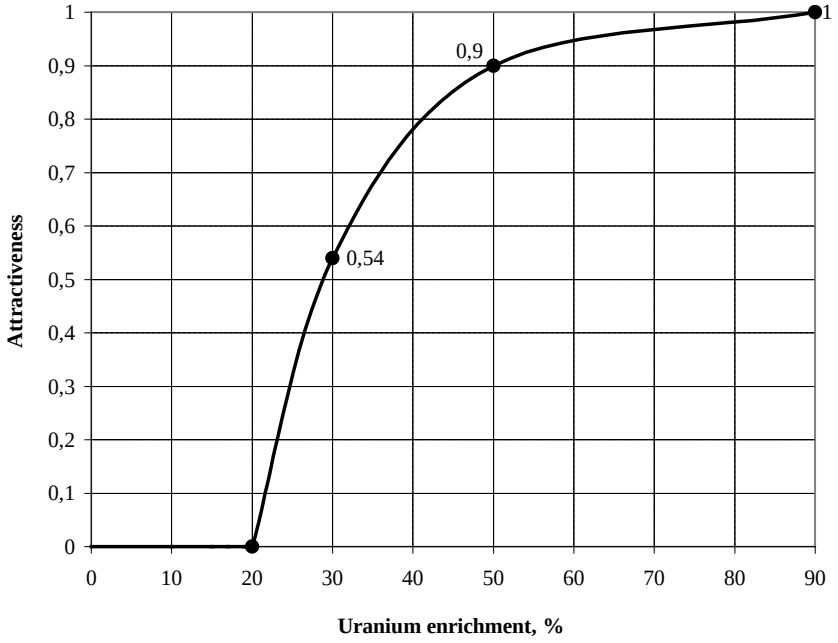


FIG. 1. Attractiveness of enriched uranium versus enrichment.

From the graph it follows that attractiveness of uranium at 50 % enrichment is quite high and equal to 0.9.

In Table 4, some comparative estimates are presented by quantity of the feed material and by quantity of necessary SWU for production of 25 kg of 90 % enriched uranium (HEU). The enriched uranium fuel for thermal reactors (TR) and for fast reactors (FR) with typical figures of enrichment level are used as feed materials. For the estimates, 0.2 % of uranium-235 is assumed for the enrichment tails.

Table 4.

Amount of feed material and number of SWU needed for the production 25 kilograms of HEU

| | Input material to produce HEU | | | |
|---|---|---|---|---|
| | Natural uranium | Uranium fuel of TR | Uranium fuel of FR | |
| | 0.711% of U-235 | 4% of U-235 | 15% of U-235 | 20% of U-235 |
| Mass of feed material | 4.4 t | 590 кg | 150 кg | 110 kg |
| Number of SWU | $5.7 \cdot 10^3$ | $1.8 \cdot 10^3$ | 750 | 500 |

From the presented results it is noted that the use of fuel of nuclear reactors as feed material for enriching to HEU considerably reduces both the required quantity of a feed material, and number of

SWU resulting in an essential reduction of time to produce needed amount of HEU. When using fuel of nuclear power reactors as feed materials, in essence, there will be enough feed material from about two fuel subassemblies to produce HEU for one significant quantity.

The potential proliferator, will not significantly save on expenses in HEU production, it can allow the larger content of uranium-235 in enrichment tails that leads to reduction of SWU number and consequently reduces time, but on the other hand it leads to increase of required feed material mass.

Thus, it is possible to assume that supply of enriched uranium nuclear fuel to other countries there is a potential risk that this fuel can be used as feed material for HEU production.

## 6. Possible scenarios of fast nuclear power development.

A phase of accelerated development of nuclear power with fast reactors is needed in order to occupy in foreseeable future a significant place in the Russian (and global) energy mix, followed by a stationary state (or, possibly, development at a rather moderate pace). Such a scenario for Russia was considered in [8].

Start with reactor-grade plutonium (from spent nuclear fuel of thermal reactors) cannot provide the pace required for the phase of accelerated development.

To increase the amount of bred plutonium it is necessary the use of a blanket with natural or depleted uranium, the isotopic composition of plutonium produced in the blanket will be close to that of weapons-grade plutonium.

Plutonium close to weapons-grade by isotopic composition will also be appeared at least in the first reloads if fast reactor starts-up with the use of uranium fuel.

So, both options of accelerated development of fast nuclear power, either start-up with using reactor-grade plutonium and blanket or start-up with uranium fuel without blanket, would result in generation of weapons-useable plutonium at least in the first reloads. In addition, the second scenario would require a tremendous effort to develop uranium enrichment capabilities.

Spent nuclear fuel reprocessing with no plutonium and uranium separation was proposed as an additional technological barrier as far as this separation is not absolutely necessary for the operation of fast reactors.

Now in Russia and in other countries, spent nuclear fuel is reprocessed with chemical technologies immanently capable of separating plutonium and other actinides because their chemical properties strongly differ.

The proposed scenarios of chemical technology with no plutonium and uranium separation pose strict requirements for regimes and chemicals to be used. Their violations (hardly controllable) will allow separation of pure plutonium.

The use of physical (plasma) technologies would become a radical solution to the problem but at present they have reached the level of maturity at laboratory experiments only.

Possibly, a compromise scenario would be optimal for the first, intensive phase of fast power development. For this scenario it is supposed that the start-up of fast reactors will be done with a mixture of plutonium and uranium recovered from spent nuclear of thermal reactors, with adding enriched uranium in amounts to satisfy criticality conditions. This scenario would help reach a rather high pace of development with moderate demand for enrichment technology and without need for plutonium and uranium separation. In this case the isotopic composition of produced plutonium would be far from that of weapons-grade plutonium.

Later on, as fast nuclear power expands and thermal reactors will be replaced by fast ones, requirements for the pace of development might decrease and, accordingly, the demand for enriched uranium will reduce which allow abandoning enrichment technology some day in the future. In this way nuclear power would be able in the future to get rid of the "black spots" of proliferation – enriched uranium and high-grade plutonium.

Special attention should be given to the export of fast reactors with an on-site fuel cycle. The return of irradiated fuel for technological storage and reprocessing to the exporter state would be an alternative. On the one hand, transportation of nuclear materials could be vulnerable to theft, and this is an argument in favor of on-site fuel cycle. On the other hand, the accounting and control of nuclear materials in bulk form is difficult. This and the sprawl of spent fuel reprocessing technologies are arguments against on-site fuel cycles to be exported together with fast reactors.

## 7. Conclusions.

1. At the start-up and operation of fast reactors with the use of plutonium fuel, plutonium with large amount of the highest isotopes, which is a little suitable for the use in the nuclear weapon, is formed in the core, however without a blanket is problematic to provide rather intensive development of power on fast neutrons.

2. Use of blanket can help attain a higher pace of development but generate high-grade plutonium, most suitable for nuclear weapons.

3. At start-up of fast reactors with the use of uranium fuel, plutonium close to weapons-grade by isotopic composition is formed in the core.

4. A compromise scenario where the reactors start-up with a mixture of plutonium from spent fuel of thermal reactors and enriched uranium can be considered; it allows an acceptable rate of development and does not generate high-grade plutonium.

5. In long-term perspective transition to fast nuclear power allows abandoning enrichment technology which is most hazardous from the nonproliferation point on view.

6. The use of spent fuel reprocessing technology with no uranium and plutonium separation would create an additional technological barrier to proliferation.

7. The issue of exporting fast reactors with the on-site fuel cycle needs further consideration and comprehensive analysis.

8. It is desirable to intensify the development of physical (plasma) technologies for spent nuclear fuel reprocessing without the separation of plutonium and other actinides.

9. Time has come to remove from fast reactors and their nuclear fuel cycle the label unfairly identifying them as the most dangerous installations of nuclear power from the standpoint of being a proliferation problem.

**ACKNOWLEDGEMENTS**

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# Analysis of scenarios of the inclusion of fast reactors in the nuclear power of Russia in the context of sustainable development with the use of the INPRO methodology

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**Abstract.** The analysis of several transition scenarios to a new technology platform of nuclear power in Russia is made with use of the INPRO methodology for assessing nuclear energy systems (NESs) taking into account requirements for sustainable development worked out under the IAEA auspice. The technological platform under consideration consists of thermal reactors of the VVER type, fast reactors of the BN type and installations of a closed nuclear fuel cycle. Indicators used for assessing sustainability of the options of the future NES of Russia were adopted from the set of indicators elaborated by the participants of the international project GAINS implemented in the framework of the INPRO project. The study was focused on the analysis of evolutionary scenarios of nuclear power deployment simulated with the use of recently developed computer code CYCLE, designed for mathematical modeling scenarios of a NES with the complex structure of the nuclear fuel cycle. Results of calculations of the nuclide flows and characteristics of irradiated nuclear fuel are discussed in the paper in terms of approaching to sustainable nuclear energy system.

## 1. Introduction.

Currently, the world nuclear power is based on the technology of light-and heavy-water reactors and a once-through nuclear fuel cycle (OTFC). It's generally acknowledged that this technology can not be a basis for a future large-scale nuclear power. The consumption of natural uranium resources in the OTFC is one of the reasons which make this cycle unable to meet requirements of sustainable development since the energy potential of uranium-235 being used in thermal reactors is too small to compare with the energy potential of fossil fuels. Within technological capabilities available today only NES with fast reactors and closed nuclear fuel cycle (CNFC) has the potential for fuel supply of nuclear power comparable in scale to the energy potential of fossil fuels. Besides the global challenges in the area of nuclear resources there are many other challenges to be met by the future nuclear power in order to make a significant contribution to sustainable economic and social development of mankind. At international level, the requirements of sustainable development for NES were developed under the auspices of the IAEA in the INPRO methodology [1]. This paper presents simulation results of a few scenarios of a transition from the current technological platform of nuclear power in Russia to a new one and provides analysis of the scenarios with the key indicators established and used within the framework of the INPRO collaborative project GAINS [2].

## 2. Development of transition scenarios for two-comonent nuclear energy system of Russia.

Middle-term projections of nuclear power deployment in Russia are presented in the Energy Strategy of Russia up to 2030. The long-term strategy of nuclear power development in Russia (to 2050 and on) does not contain any specific projections, but defines the fundamental principles of the nuclear power deployment in Russia and certain technological directions for the future development. As an important guideline for developing scenarios up to 2050 and on, the long-term strategy states that in order to meet the challenges of sustainable development and increase the share of nuclear power in electricity generation up to 30% the fuel cycle of a national nuclear energy system has to include fast reactors and to be closed with respect to fissile materials. Thus, for a long time the nuclear power of Russia is supposed to be a two-component system, consisting of thermal and fast reactors with a gradual transition from a OTFC to a closed NFC.

The principle features of fuel flows in a two-component NES (100 GWe) consisting of FR and LWR are shown in Figure 1.

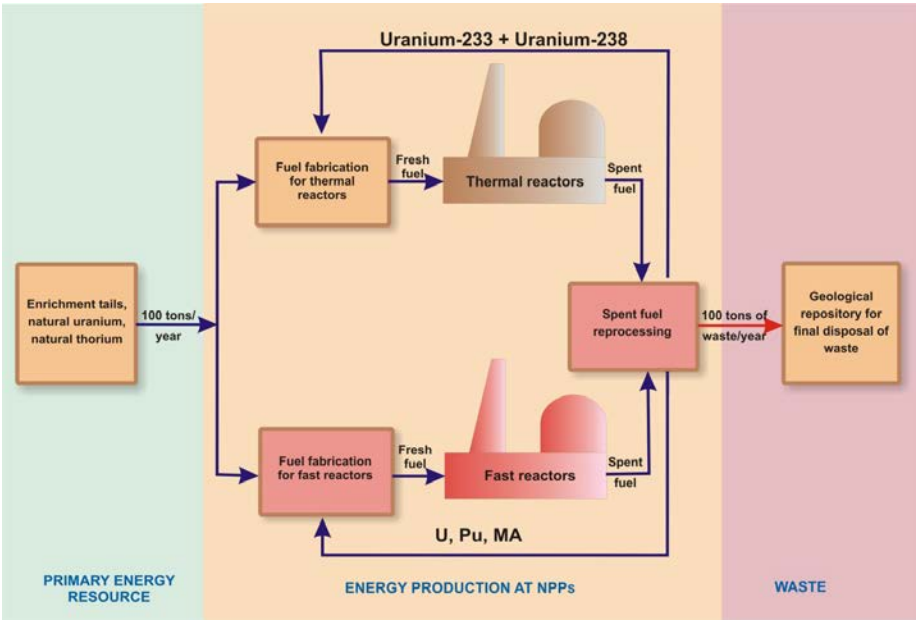



FIG. 1. Example of a two-component NES of fast and thermal reactors.

Simulation was performed for three scenarios of two-component NES based on a closed NFC with VVER-1000 reactors of PWR type and BN fast reactors of SFR type. The scenarios differ from each other by the time of fast reactors introduction into the nuclear energy system. In the first (base) scenario the serial deployment of fast reactors BN-1200 is assumed to start in 2020 after six years of operation of the first Russian MOX fueled fast reactor BN-800 (Figure 2). Comparing to the base scenario, deployment of the BN-1200 series is supposed to delay of 20 years in the second scenario and of 40 years in the third scenario. In each scenario the rate of fast reactors commissioning is gradually increasing. Intensive introduction of the commercial fast reactors with the rate up to ~ 2.4 GWe/year begins after about 20 years of the first of a kind BN-1200 unit. Spent fuel of the VVER reactors is reprocessed and a part of extracted plutonium is used in the fuel of BN reactors. A given growth of electricity generation is provided by both VVER and BN reactors. One of the ultimate goals for the two-component NES is to reach balance between plutonium production and consumption.

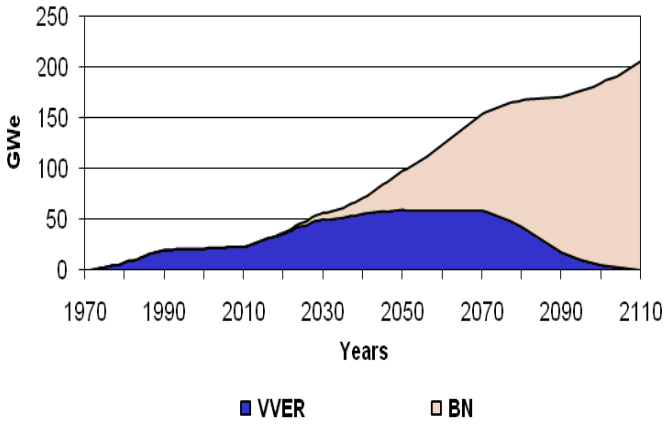



FIG.2. Refference scenario for the two-component NES of Russia

## 3. Results of scenarios simulation.

Modeling of the scenarios was performed with a computer code CYCLE [3], recently developed in IPPE. The code is dedicated for mathematical modeling of thermal and fast reactors operating in the once-through and closed nuclear fuel cycle. A distinctive feature of the complex is the detailed modeling of the nuclide composition of the fuel at all stages of the nuclear fuel cycle. This makes it possible not only to calculate the correct characteristics of the material balance, but also to assess the thermal, and ecological characteristics of the fuel in the NFC. The CYCLE code designed to simulate the loading and unloading of fuel in the system of nuclear energy, including reactors and necessary infrastructure - storage pools, storage of irradiated fuel assemblies and repository for the final disposal of waste. The given function describing the input reactors in time and some technical characteristics of reactors are input data for the code.

Several key indicators of the GAINS project [2] were selected to analyze results of the scenarios. The set of indicators used in study includes: the total consumption of natural uranium, the amount of accumulation of spent nuclear fuel, capacity requirements for separation work, capacities for fuel fabrication and reprocessing of spent nuclear fuel, the amount of accumulated spent nuclear fuel at the storages, amount of plutonium and minor actinides (MA) in the storages and repositaries.

Figure 3 shows the total consumption of uranium in the three scenarios under consideration.

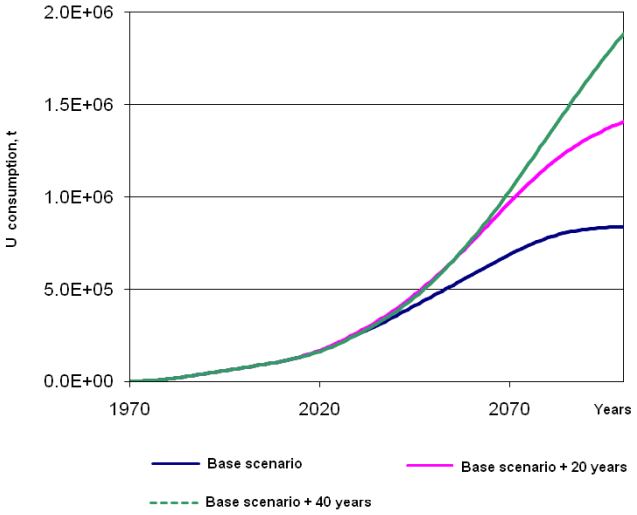



FIG. 3. Total consumption of natural uranium in the base case and in scenarios with delayed commissioning of fast reactors.

As can be seen in Fig. 3 delay in fast reactors input of 20 years results in the growth of the total consumption of natural uranium up to 70%. An even more significant increase in the consumption of natural uranium is observed when input of fast reactors is delayed for 40 years. In the last case, park of reactors will consist of thermal reactors up to 2060. The total requirements for natural uranium would reach ~ 1.9 million tons that is twice more than the amount of uranium consumption in the base case and exceeds the estimated reliable Russian reserves of natural uranium (~1.5 million tons).

Significant effect can also be seen in the required separate work capacities (Fig. 4).

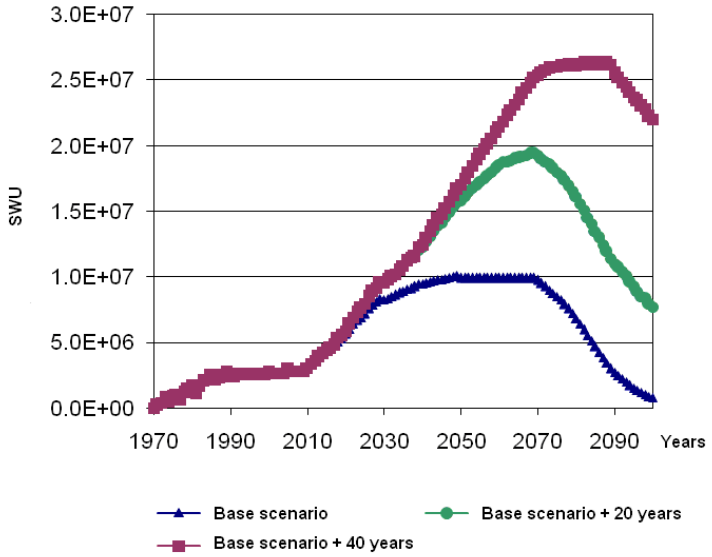
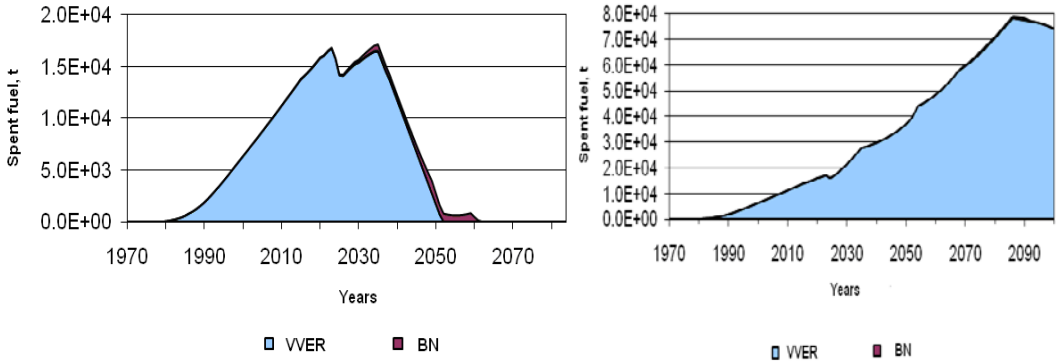



FIG. 4. Need in separation work

The graph in Fig. 4 demonstrates that in case when input of fast reactors would be delayed 40 years the capacities of enterprises for enrichment of the VVER fuels would have been increased by 2070 to about 10 times as compared with currently available Russian enrichment facilities.

One of the important indicators to judge how the NES under consideration solves the problems in the areas of "Waste Management", "Non-proliferation" and "Environment" is the amount of storage of spent nuclear fuel, plutonium and MA. Data on the accumulation of spent nuclear fuel are shown in Fig. 5. A "pit" in Fig. 5 (a) is associated with the removal of the VVER spent fuel from the storage to reprocessing with an aim to extract plutonium for MOX fuel of fast reactors.




a. b.

FIG. 5. Amount of accumulated spent fuel of thermal and fast reactors in the base scenario (a) and in case of the delay of the BN reactors input for 40 years.

As follows from Fig. 5 (a), spent fuel of thermal reactors in the baseline scenario is not going to further accumulation after 2050. The calculations show that in case of 20 years delay for the input of fast reactors the peak accumulation of spent fuel of the VVER reactors would reach ~ 30 thousand of tons by 2076 and would increase approximately 2-fold comparing to the base case. In the third scenario (40 years delay for the input of fast reactors) the peak accumulation of spent fuel of the VVER reactors would reach ~ 78 thousand of tons by 2080 as shown in Fig. (b).

The same trends are observed for the plutonium accumulation (Fig. 6).

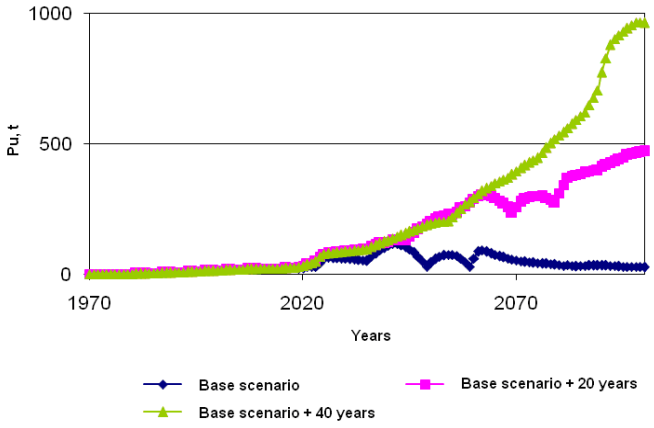



FIG. 6. The amount of accumulated plutonium in spent fuel of VVER-1000 and BN-1200.

Likewise in the case of spent nuclear fuel accumulation, the shortest time for minimizing of the amount of the separated plutonium storage can be achieved in the base case. Presently, the amount of separated plutonium in the storage is about 50 tons. This is mainly plutonium extracted from the spent fuel of VVER-440 and BN-600. The spent fuel of VVER-1000 is not reprocessed today. In the base scenario it is possible to keep the balance between the separated plutonium from the VVER-1000 and plutonium used in BN-1200 at the minimal operational level (Fig.6). Delay of 20 years for the fast reactor input would result in the accumulation of ~500 tons of plutonium in spent fuel by the end of the century. Delay of 40 years for the fast reactor input would result in the accumulation of ~1000 tons of plutonium in spent fuel by the end of the century.

The scenarios developed in the presented study did not include options with dedicated reactors for transmutation of MA. Variants for MA transmutation were restricted with the options of MA burn up in the core of fast reactors under assumption of homogeneous inclusion in the fuel of 1% of MA with the composition of neptunium/americium/curium as 0.2/0.7/0.1. Preliminary results show that burning of neptunium and americium in fast reactors could solve the problem of these minor actinides accumulation and, consequently, reduce the radiotoxicity of HLW destined for final disposal. As for the curium, although utilization of this element in BN is also technically possible, many experts are inclined to think that there is no much sense in its transmutation. On the one hand, handling of curium is rather difficult because of the high radioactive alpha- and gamma- radiation of Cm-244. On the one hand, after separation, curium could be stored for 100 years in special protected containers until practically full decay to plutonium-240 suitable for utilization in fast reactors. It is planned to continue comprehensive consideration of scenarios for MA management keeping the fast reactor option as a promising direction in the MA utilization.

## 4. Economic studies.

The code MESSAGE [3] for the energy system simulation was used for economic assessments. This code allows determine the optimal strategy within available resources under the given boundary conditions and the limitations of technologies.

The nuclear fuel cycle cost analysis shows that there is considerable uncertainty in the economic data for the technologies that have not yet passed the stage of demonstration and commercialization. Therefore, it is important to perform a sensitivity analysis of economic parameters that will determine the upper limits of the prices at different stages of NFC under which the fast reactor technology remains competitive. It means within MESSAGE model that the inclusion of fast reactors into the optimal plan for a two-component nuclear energy system has to be demonstrated.

The base scenario of a two-component NES described above was chosen for the sensitivity analysis. Specific capital costs for construction of a new VVER-1200 reactors and BN-1200 reactors were taken equal. Figure 7 shows the optimal plan for the case. The share of BN reactors in the optimal plan by 2050 is 39% and goes on after 2050.

The study on the impact of capital costs to the competitiveness of the BN-1200 in a two-component NES showed that the increase in capital cost of BN-1200 by 10% can reduce the share of BN reactors in the optimal plan by 2050 from 39% to 13%. Yet, by the end of this century fast reactors continue to play a leading role. Growth of the capital cost of BN by 50% comparing to VVER results in a radical change in the ratio of thermal and fast reactors in favor of thermal reactors (Fig. 7, b). Thus, for an economically viable operation in two-component NES of thermal and fast reactors excess of capital costs for the construction of the BN reactors compared with the advanced thermal reactor VVER should be no more than 10%.

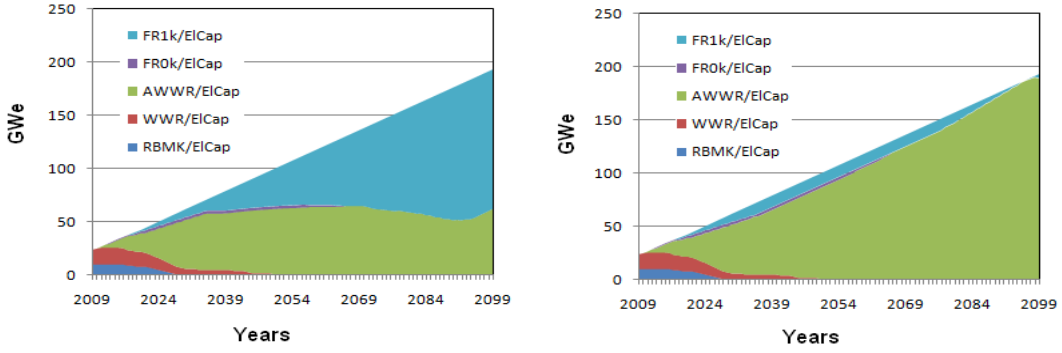



FIG. 7. An optimal plan of a NES under equal capital cost for the construction of VVER (a) and BN reactors and under growth of capital cost of the BN by 50% over VVER (b).

Another important economic parameters affecting the competitiveness of fast reactors, is the cost of the fuel cycle. The fundamental properties of fast reactors confirmed by the evaluations made in France [4] allow expect reducing of the fuel cost share in the total cost of electricity generation of fast reactors to be lower than the share of fuel cost share of thermal reactors. However, preliminary calculations performed with the expected prices for the main conversion stages of a closed nuclear fuel cycle show that the share of fuel cost share of electricity fast reactors is comparable with the share of fuel cost of thermal reactors, or even higher.

This situation is associated with a relatively low cost of uranium fuel, determined by the offer of the current global market for natural uranium, low price of uranium enrichment at the national market of Russia, a relatively low cost of dry storage of spent nuclear fuel of thermal reactors and the estimated costs of the final disposal of spent nuclear fuel. Perhaps, the economic attractiveness of the OTFC option of NFC will remain in many countries for a long time. Currently, the task of a comprehensive economic study of all stages of the nuclear fuel cycle in the Russian conditions, especially of reprocessing and fuel fabrication stages becomes very relevant. It is expected that the efficient use of uranium resources, reducing the amount of fissile materials storage, minimizing the environmental risks of nuclear waste disposal and other improvements consistent with the principles of sustainable development of NES will be adequately taken into account in the evaluation methodology and give an additional stimulus for the development of nuclear fuel cycle technologies.

## 5. Conclusions.

The need to make a transition to a closed nuclear fuel cycle with fast reactors and the optimal time for such a transition are among the most important and disputable issues of nuclear power development. The strategy of nuclear power development in Russia defines transition to the closure of the nuclear fuel cycle, but the choice of specific technological options is not completed. Currently, a few R&D programs are under implementation, which has to be resulted in a final decision on the balance between evolutionary development and radical innovation. In this paper an evolutionary scenario was discussed. It is based on inclusion of reactor technologies and technologies of the closed nuclear fuel cycle used or demonstrated in Russia: PWR type reactors (VVER), sodium fast reactors (BN), pellet UOX and MOX fuels, aqueous technologies of spent fuel reprocessing.

It is shown that the evolutionary two-component system of advanced VVER and BN reactors can provide a substantial contribution to enhancing sustainability features of a national nuclear energy system in terms of nuclear fuel supply and waste management:

- under timely deployment of fast reactors the installed capacities of nuclear power of Russia could reach by 2050 about 100 GWe with use of the proven national resources of natural uranium;
- the problem of spent nuclear fuel accumulation can be basically solved by 2050 since plutonium from the spent fuel to be used in MOX fuel of BN reactors;
- a system based on a closed fuel cycle with MOX fueled fast reactors of BN type would save natural uranium and would allow to expand export of thermal reactors with an attractive package of fuel supply (supply of fresh fuel and take back spent fuel).

Along with a set of indicators of sustainable development that demonstrated significant advantages of the two-component system of thermal and fast reactors operating in a closed fuel cycle, the application of the INPRO methodology confirmed the presence of "weak links" of the system to date. For the area embraced by the study, a number of economic and infrastructure challenges for fast reactors were indicated. These challenges are addressed in the Federal target program on transition to a closed fuel cycle with fast reactors which is currently under implementation in Russia.

From the point of view of the authors of this paper the commercialization of fast reactors by 2030 and their widespread use in the coming years will require to:

- put into operation by 2014 and ensure reliable operation of the reactor BN-800 fueled by pilot and industrial facilities for reprocessing and MOX fuel fabrication;
- start up the construction of the first of a kind BN-1200 by 2016-2018, put the reactor in operation by 2021-2023 and demonstrate its reliable operation;
- deploy a small series of BN-1200 with a mixed uranium-plutonium fuel, specify the system requirements for the NPP with BN and associated NFC with focus on economic characteristics of the whole system;
- continue implementation of R&D programs on improving the aqueous reprocessing methods that can be used in Russia or in the international nuclear fuel cycle centers under the auspice of the IAEA.

As it was mentioned above the scope of the paper is restricted with the evolutionary reactor and fuel cycle technologies. The “Federal target program of transition to a closed nuclear fuel cycle” provides R&D for a wide spectrum of promising nuclear energy technologies of fast reactors and includes development of fast reactors with lead and lead-bismuth coolant, dense fuel and fuel with minor actinides, new construction materials, innovative technologies of reprocessing such as pyrochemical methods, etc. Results of this program will provide a reliable basis for decision on the future of the nuclear power of Russia.

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# About some scenarios for neptunium management in nuclear power

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**Abstract.** Scenarios of neptunium utilization in conventional sodium cooled fast reactors cores with its homogeneous addition to MOX fuel, as well as radiation characteristics of fresh regenerated MOX-fuel containing neptunium are discussed. The results of calculation studies on $^{237}$Np utilization in BN-K type fast sodium reactor are given. The analysis of radiation characteristics of fresh regenerated MOX-fuel are presented, that allowed to make the preliminary recommendation on the practical implementation of $^{237}$Np recycling in fuel of BN-K type reactor from the viewpoint of radiation aspects. It is shown that non-separated mixture of plutonium and neptunium obtained as a result of reprocessing of SNF from VVER-1000 reactor could be used as fuel for BN-K type reactor.

## 1. Introduction.

Russia has a considerable experience in the area of reprocessing of spent nuclear fuel (SNF) of thermal reactors. Up to now SNF reprocessing is accompanied by extraction of neptunium in a kind of neptunium dioxide, which is currently stored in temporary storages. A little part of extracted neptunium is used to obtain valuable isotope $^{238}$Pu.

According to the estimations ~45 tons of civil plutonium have been extracted by now during reprocessing of SNF of thermal reactors (VVER-type) at reprocessing plant RT-1 in Russia. Assuming that SNF of VVER reactor contains approximately 0.06-0.07% of neptunium (cooling time for SNF is 3 years), one can estimate that about ~ 3 tons of neptunium have been accumulated today at the RT-1. Currently, the annual discharge of spent fuel from VVER-440 and VVER-1000 reactors is estimated as ~87 t and ~220 t, correspondingly. The estimations show that about 9 tons of Np additionally will be accumulated over the next ten years, if to take into account the projected growth of nuclear power in Russia in the short-term period and planned development of reprocessing of SNF from VVER-1000 reactors.

At the same time neptunium is a highly toxic material, it is also included in the main list of nuclear materials subject to control and accounting for non-proliferation. In this connection, it requires special equipped storages that will require significant additional expenses and won't solve the problem of storing neptunium. Currently, there are two real and affordable ways to reduce storages of neptunium: vitrification of neptunium for its final disposal and use of Np in fuel of fast reactors operating in closed nuclear fuel cycle.

There is an opinion that due to low contribution of neptunium into radiotoxicity and residual power of SNF the best way for neptunium management is to vitrify it together with fission products for further disposal. However, it should be noted that neptunium has a very low sorption and high migration capacity, higher than other radionuclides, that creates a very complex technical problem with its sealing. According to data given in [1], migration of neptunium in natural rock formations is 7-fold higher than migration of plutonium and about 25-fold higher than migration of americium and curium.

 

Besides, one can expect high leaching of Np from conditioned waste due to its very low solubility in glass matrix [2]. This risk does not allow guaranteeing the long-term safety of the repository during a significant period ($^{237}$Np has a half-life of 2144000 years). In accordance with the existing Nuclear Regulations non-usable products of SNF reprocessing as dangerous radioactive materials should be isolated and stored for the period of time, during which they causes the danger. For $^{237}$Np this period is more than millions years.

At the same time neptunium can be transmuted into valuable energy products by its irradiation in reactor: $^{238}$Pu is used as energy source in thermoelectric generators for pacemakers, space satellites, navigation beacons; and $^{239}$Pu is the most important fissile material.

Utilization of neptunium can be implemented in fast reactors more effectively, it can be considered as a secondary function of BN-800 and BN-K sodium fast reactors operation. However, it should be noted that $^{237}$Np can also be added in the fuel for thermal reactors for improving nonproliferation resistance. Addition of $^{237}$Np into fresh UOX or MOX fuel for thermal reactors (LWR) can reduce significantly the risk of nuclear proliferation that will prevent theft or sabotage with the aim of weapons use, because the amount of $^{238}$Pu, which is α-active, increases in discharged fuels. Studies have shown [3] that from the viewpoint of reducing proliferation risk it is enough to add about 1% and 0.5% of $^{237}$Np into UOX-fuel and MOX-fuel, correspondingly (restriction is connected with the requirement of negative value of sodium void effect).

Currently a Pilot Demonstration Centre (PDC) for SNF reprocessing technologies is under construction at the FSUE "Mountain Chemical Combine". The PDC is designed for mastering a new SNF reprocessing technology for generation III reprocessing plant, which is to ensure no discharges of liquid waste into the environment and to reduce the overall cost of SNF reprocessing. A qualitative analysis of the problem of neptunium management will allow chemical engineers to optimize reprocessing of SNF in better way, and thereby to eliminate accumulation of neptunium at storage.

## 2. Utilization of neptunium in BN-K type reactor.

The use of conventional fast sodium reactor cores for neptunium utilization is of great interest for adding neptunium homogeneously into MOX-fuel (at a level of few percents of h.a.). Calculation studies of this option were carried out for BN-K type fast reactor operating in closed nuclear fuel cycle using the package of codes RZA/ PC [4] in two-dimensional (RZ) geometry and CONSYST code, which provides 26-group cross section data [5]. The core of BN-K type fast reactor is surrounded by lower axial blanket and two rows of radial blanket. Sodium plenum is above the core that provides an acceptable value of sodium void reactivity effect.

The calculations simulated the operation of BN-K type reactor in closed nuclear fuel cycle, in which neptunium is added into fresh MOX-fuel fabricated on the basis of “own” plutonium and neptunium extracted from SNF discharged at previous cycle of BN-K operation and some amount of "external" neptunium from storage. Several options of neptunium content in annual fuel loading are considered. Neptunium is located homogeneously in all fuel rods, replacing the appropriate amount of depleted uranium. To ensure the reactivity at the end of cycle initial plutonium content in fuel is corrected. Cooling time for SNF before reprocessing is equal to 3 years.

As result of calculations the equilibrium plutonium composition was established and its required content in fuel loading. Table 1 gives the isotopic composition of plutonium discharged in equilibrium state for different neptunium content in fuel loading. Table 2 also presents the composition of plutonium discharged without plutonium and neptunium recycling for comparison. The isotopic composition of plutonium discharged, except $^{238}$Pu depends weakly on amount of neptunium in fuel loading.

Figure 1 depicts a change of $^{238}$Pu content in discharged fuel depending on amount of external Np added into fuel loading.

Table 1.

Pu and $^{237}$Np recycling

| Annual loading of "external" $^{237}$Np, kg | Pu content in fuel, % | Average $^{238}$Pu content in discharged fuel, % | $^{237}$Np content in fuel loading, % | $^{237}$Np content in Pu and Np mixture, % |
|---|---|---|---|---|
| without Np recycling | 16.9 | 0.3 | | |
| 6.1 | 16.8 | 0.7 | 0.07 | 0.4 |
| 15 | 16.8 | 1.3 | 0.4 | 2.5 |
| 30 | 16.8 | 2.2 | 0.8 | 4.6 |
| 60 | 16.8 | 4.0 | 1.6 | 8.5 |
| 90 | 16.6 | 5.7 | 2.7 | 13.9 |
| 120 | 16.5 | 7.4 | 3.1 | 18.8 |

Table 2.

Isotopic plutonium composition, %.

| Pu isotopes | Pu discharged, NFC option without Pu and Np recycling | Pu discharged depending on annual adding of "external" Np, NFC option with Pu and Np recycling, kg | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | 0 | 6.1 | 15 | 30 | 60 | 90 | 120 |
| $^{238}$Pu | 1.5 | 0.3 | 0.5 | 1.3 | 2.2 | 4.0 | 5.7 | 7.4 |
| $^{239}$Pu | 64.2 | 67.5 | 67.5 | 67.3 | 67.0 | 66.5 | 66.1 | 65.4 |
| $^{240}$Pu | 26.2 | 26.7 | 26.7 | 26.3 | 25.7 | 24.7 | 23.8 | 23.0 |
| $^{241}$Pu | 3.6 | 3.7 | 3.6 | 3.5 | 3.5 | 3.4 | 3.1 | 3.0 |
| $^{242}$Pu | 5.7 | 1.8 | 1.7 | 1.6 | 1.6 | 1.5 | 1.3 | 1.2 |

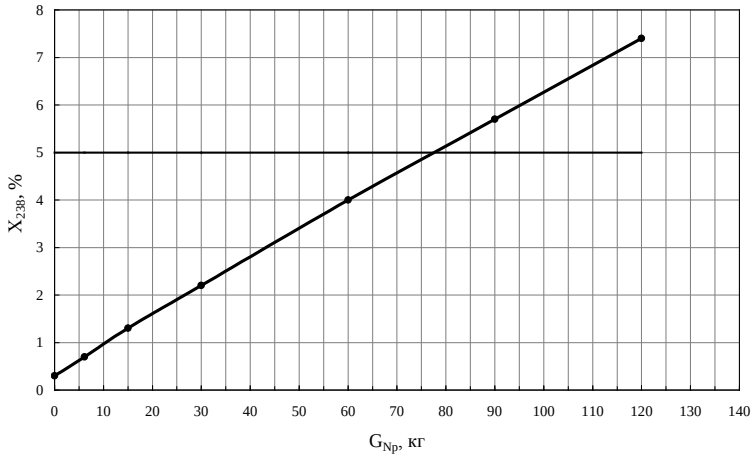


FIG. 1. $^{238}$Pu content in discharged fuel depending on amount of external Np added into fuel loading.

It is shown that the maximum permissible adding of neptunium is about 80 kg in case of recycling their own plutonium. It should be noted that these results were obtained for the case of co-reprocessing of spent nuclear fuel from the core and radial blanket.

The presented results allow to evaluate the possibility of using neptunium together with plutonium as fuel for fast reactors. Neptunium fraction in a mixture of neptunium and plutonium for SNF from VVER-1000 is 5.4% or 0.9% related to all heavy nuclei. According to Table 1 fuel with such content of neptunium is acceptable for fast reactor. This corresponds to annual consumption of neptunium in amount of about 50 kg, when the content of $^{238}$Pu in discharged plutonium is about 3%.

Thus, the technology of reprocessing of VVER SNF is acceptable, in which neptunium and plutonium are not separated from each other and are used for manufacturing MOX-fuel for fast reactors.

## 3. Analysis of radiation characteristics of fresh regenerated MOX-fuel

When organizing multiple recycling of neptunium, it is important to understand how the radiation situation deteriorates at production of fresh regenerated MOX-fuel. The studies have been carried out, in which content of neptunium in fresh regenerated MOX-fuel is varied. To assess radiation characteristics the following options of nuclear fuel cycle (NFC) were considered:

1. The steady state composition of fuel at multiple recycling of MOX-fuel in BN-K type reactor based on reactor grade plutonium without involving Np;

2. Introduction of some amount of $^{237}$Np into regenerated MOX-fuel based on reactor grade Pu from SNF of thermal reactors in such way that $^{237}$Np content in initial MOX-fuel was at a level of 1% by weight (~ 0.95 kg/SA).

3. The steady state composition of regenerated MOX-fuel at multiple recycling in the BN K reactor based on reactor grade plutonium without extraction of $^{237}$Np, "external" $^{237}$Np is added into MOX-fuel (~0.7 kg/SA) that its total content in initial MOX fuel was 1.57 % by weight .

4. The steady state composition of regenerated MOX-fuel at multiple recycling in the BN K reactor based on reactor grade plutonium without extraction of $^{237}$Np, "external" $^{237}$Np is added into MOX-fuel (~1.4 kg/SA) that its total content in initial MOX fuel was 3.1 % by weight .

Table 4 presents radiation characteristics of fresh MOX-fuel depending on temporary storage time after fabrication of MOX-fuel and NFC option.

Table 4.

Dependence of radiation characteristics of fresh MOX-fuel on temporary storage time after manufacture and NFC option.

| Parameter | Temporary storage time after MOX-fuel manufacture, years | NFC option | | | |
|---|---|---|---|---|---|
| | | 1 | 2 | 3 | 4 |
| Heat release, κW/sec SA | 1 month | 0.091 | 0.347 | 0.417 | 0.714 |
| | 1 year | 0.093 | 0.349 | 0.416 | 0.711 |
| | 2 years | 0.095 | 0.351 | 0.415 | 0.707 |
| Gamma source, κW/sec SA | 1 month | 2.279E+13 | 4.367E+13 | 6.283E+13 | 9.985E+13 |
| | 1 year | 2.254E+13 | 4.905E+13 | 6.493E+13 | 1.014E+14 |
| | 2 years | 2.802E+13 | 5.412E+13 | 6.688E+13 | 1.029E+14 |
| Neutron sources, | 1 month | 7.101E+06 | 1.630E+07 | 1.611E+07 | 2.436E+07 |
| n/sec SA | 1 year | 7.148E+06 | 1.634E+07 | 1.608E+07 | 2.425E+07 |
| | 2 years | 7.193E+06 | 1.638E+07 | 1.604E+07 | 2.414E+07 |

Heat generation in fresh MOX-fuel subassemblies increases with the growth of $^{238}$Pu content in plutonium loaded. This parameter effects essentially on temperature of manufactured MOX fuel SA. For example, preliminary estimates indicate that for heat generation at the level of 0.5 kW/SA temperature of central fuel rod cladding in air can be ~ 250°C and temperature of SA wrapper tube in air – ~130°C.

Intensities of neutron and gamma sources in fresh MOX fuel indicated in Table 4 depend on NFC option and temporary storage time after manufacture of fresh fuel. The intensity of neutron source is associated primarily with the content of $^{238}$Pu in plutonium loaded;

Equivalent dose rate at the surface of MOX-fuel SA wrapper tube is an important radiation parameter characterizing the radiation situation when handling with fresh regenerated MOX-fuel SA. The values for this parameter for different NFC options are shown in Table 5.

Table 7.

Dependence of Equivalent dose rate at surface of MOX-fuel SA wrapper tube on temporary storage time after MOX-fuel manufacture and NFC option, mSv/h

| Cooling time | Radiation type | NFC option | | | |
|---|---|---|---|---|---|
| | | 1 | 2 | 3 | 4 |
| 1 month | neutron | $2.02 \cdot 10^3$ | $4.5 \cdot 10^3$ | $4.34 \cdot 10^3$ | $6.49 \cdot 10^3$ |
| | photon | $4.69 \cdot 10^2$ | $1.7 \cdot 10^3$ | $2.06 \cdot 10^3$ | $3.69 \cdot 10^3$ |
| | total | $2.49 \cdot 10^3$ | $6.2 \cdot 10^3$ | $6.4 \cdot 10^3$ | $1.02 \cdot 10^4$ |
| 1 year | neutron | $2.03 \cdot 10^3$ | $4.51 \cdot 10^3$ | $4.34 \cdot 10^3$ | $6.46 \cdot 10^3$ |
| | photon | $6.92 \cdot 10^2$ | $8.43 \cdot 10^3$ | $1.41 \cdot 10^4$ | $2.75 \cdot 10^4$ |
| | total | $2.72 \cdot 10^3$ | $1.29 \cdot 10^4$ | $1.85 \cdot 10^4$ | $3.4 \cdot 10^4$ |
| 2 years | neutron | $2.04 \cdot 10^3$ | $4.52 \cdot 10^3$ | $4.33 \cdot 10^3$ | $6.43 \cdot 10^3$ |
| | photon | $8.44 \cdot 10^2$ | $1.32 \cdot 10^4$ | $2.29 \cdot 10^4$ | $4.48 \cdot 10^4$ |
| | total | $2.89 \cdot 10^3$ | $1.78 \cdot 10^4$ | $2.72 \cdot 10^4$ | $5.12 \cdot 10^4$ |

The analysis of data presented shows that introduction of neptunium into initial MOX-fuel with once-through and multiple recycling leads to growth in equivalent dose rate of MOX-fuel SA due to:

1) gamma radiation of $^{233}$Pa ($T_{1/2}$ = 27 days), which is formed due to decay of $^{237}$Np;

2) higher content of isotopes $^{232}$U and $^{236}$Pu, whose decay leads to formation of $^{208}$Tl ($E\gamma$=2.614 MeV) in fresh MOX-fuel which is a hard gamma emitter, so that there is a growth in equivalent dose rate with increase in temporary storage time after MOX-fuel manufacture;

3) involvement of $^{237}$Np into NFC cycle leads to increase in $^{238}$Pu content in regenerated fuel, which in turn leads to an increase in neutron component of the equivalent dose rate of fresh MOX-fuel subassemblies.

Thus, the increase of Np content leads to increase in the equivalent dose rate as much as 3 to 10 times or more, depending on temporary storage time after MOX-fuel subassemblies manufacture. Therefore, it is expedient to limit the weight content of $^{237}$Np in fresh MOX-fuel for BN-K type reactor at the level of 1-1.5% from the standpoint of radiation aspects of management with regenerated MOX-fuel.

In case of non separation of neptunium from plutonium during reprocessing of SNF from VVER reactor, neptunium content in fresh MOX-fuel is less than 1%, which is within the recommended range of values.

## 4. Conclusions.

Calculation studies of homogeneous transmutation of neptunium in BN-K type reactor demonstrated the possibility of using non-separated mixture of plutonium and neptunium obtained during reprocessing of SNF from VVER-1000 reactors as fuel for fast reactors.

In this case, the fraction of neptunium in mixture of neptunium and plutonium for SNF from VVER-1000 is 5.4%, and the content of $^{238}$Pu in plutonium in SNF of fast reactor will be about 3%. Neptunium fraction in fresh regenerated MOX-fuel would be less than 1%, which is within the recommended range of values of $^{237}$Np content in fresh MOX-fuel from the viewpoint of radiation aspects of management with regenerated MOX-fuel.

**ACKNOWLEDGEMENTS**

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# Feasibility Study on AFR-100 Fuel Conversion from Uranium-based Fuel to Thorium-based Fuel

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**Abstract.** The feasibility study of converting a fast reactor from uranium-based fuel to thorium-based fuel was studied using the 100 MWe Advanced Fast Reactor (AFR-100). Several fuel conversion scenarios were envisioned in this study.

The first scenario is a progressive fuel conversion without fissile support. It consists of progressively replacing the burnt uranium-based fuel with pure thorium-based fuel without fissile material addition. This was found to be impractical because the low excess reactivity of the uranium-fuelled AFR-100 core, resulting in an extremely short cycle length even when only a few assemblies are replaced.

A second scenario consists in operating the reference LEU fuelled AFR-100 core for 24 years and then replacing one fuel batch out of four every 7.04 years with thorium-based fuel mixed with transuranics. The transuranics weight fraction required during the transition period is identical to that required at equilibrium and is equal to 18.6%. The original uranium-based fuel is discharged with an average burnup of 120 GWd/t and the Th-TRU fuel with an average burnup of 101 GWd/t. The thermal-hydraulic and passive safety performances of this core are similar to those of the reference AFR-100 design. However, Th-TRU fuel fabrication and performance needs to be demonstrated and TRU separation from the LWR used nuclear fuel is necessary.

The third scenario proposed consists of replacing the whole AFR-100 core with fuel assemblies made of several thorium and 20% enriched LEU layers. The mode of operation is similar to that of the reference AFR-100 core with the exception of the cycle length which needs to be reduced from 30 to 18 years. The average LEU and thorium discharge burnups are 79 GWd/t and 23 GWd/t, respectively. The major benefit of this approach is the improved inherent safety of the reactor due to the reduced coolant void worth.

## 1. Introduction

In this study the feasibility of fuel conversion in a fast reactor is assessed and several possible scenarios are proposed. The analyses are performed using the Advanced Fast Reactor (AFR-100) design [1], a fast reactor core concept recently developed by ANL. The AFR-100 is a small 100 $MW_e$ reactor developed under the US-DOE program relying on innovative fast reactor technologies and advanced structural and cladding materials. It was designed to be inherently safe and offers sufficient margins with respect to the fuel melting temperature and the fuel-cladding eutectic temperature when using U-10Zr binary metal fuel.

Thorium-based metal fuel was preferred to other thorium fuel forms because of its higher heavy metal density and it does not need to be alloyed with zirconium to reduce its radiation swelling. The various scenarios explored cover the use of pure thorium fuel as well as the use of thorium mixed with transuranics (TRU).I

The design objectives and requirements, the computation methods used as well as a description of the core concept are provided in Section 2. The three major scenarios considered are introduced in Section 3 and the neutronics performances of those scenarios are discussed in the same section. The orificing

strategies used and the steady-state thermal-hydraulic performance are provided in Section 4. The kinetics and reactivity coefficients, including the inherent safety characteristics, are provided in Section 5, and the Conclusions in Section 6.

## 2. Design Bases, Computation Methods and Core Concepts

### *2.1. Comparison of Uranium and Thorium Based Fuels*

When used in a thermal reactor thorium is typically a better fertile material than uranium, but in fast reactors use of thorium fuel in place of uranium fuel may impair the reactor performance. Since natural thorium does not contain any fissile isotope, it is necessary to support thorium-based fuel with fissile material such as low enriched uranium (LEU), plutonium or transuranics (TRU), until enough $^{233}$U is bred. Due to the lower η value of $^{233}$U, compared to $^{239}$Pu, and to the significantly lower heavy metal density of thorium, the breeding ratio achievable with thorium-based fuel in a fast reactor is typically smaller than that achievable with uranium-based fuel. The lower heavy metal density may also require loading a larger volume of fuel or shortening the cycle length to keep the core critical. The lower delayed neutron fraction of $^{233}$U compared to uranium isotope makes for a higher reactivity insertion but due to the cross-section resonances occurring in a lower energy range than for $^{239}$Pu and $^{235}$U, and smaller η value change with spectral change. Overall better reactivity feedback performance is potentially achievable.

### *2.2. Design Objectives and Requirements*

The primary objective of this study is to assess the feasibility of transitioning a sodium-cooled fast reactor from uranium-based fuel to thorium-based fuel within the same fast reactor core geometry to extend the possible uses of an installed fast reactor facility. The fuel conversion scenarios must meet the original design constraints initially adopted in the SFR core concept with the uranium-based fuel and retain the favorable inherent safety features without changing the mechanical and physical design parameters.

The feasibility of the fuel conversion from uranium-based to thorium-based fuel was studied using the 100 $MW_e$ Advanced Sodium-cooled Fast Reactor (AFR-100). This core concept was developed using U-Zr metal fuel with an assumed peak fast flux fluence limit of $6\times10^{23}$ neutrons/cm$^2$, based on the expectation that advanced cladding material will be more resistant to irradiation damage.

For consistency to the fuel form adopted in the reference AFR-100 core concept, the thorium-based fuel was also considered as a metal fuel. Thorium metal fuel does not need to be alloyed with Zr because its face centered cubic crystal structure is less prone to irradiation swelling than un-alloyed uranium [2][3]. When fissile material needs to be mixed with thorium, it is theoretically possible to use Th-Pu, Th-TRU, Th-LEU or Th-$^{233}$U. In this study, TRU and LEU were preferred because of the proliferation concern of separating plutonium from minor actinides and the current lack of significant $^{233}$U reserves. The density of $^{232}$Th is taken to be 11.7 g/cm$^3$ and that of Th-TRU is taken to be 13.4 g/cm$^3$, which corresponds to approximately 20wt% of TRU. The density of U-Zr is taken to be 15.7 g/cm$^3$.

The capacity factor is assumed to be of 90%, the coolant inlet and bulk outlet temperatures are taken to be 395°C and 550°C, respectively, and the fuel-cladding eutectic and maximum fuel temperatures are assumed to be 700°C and 1200°C, respectively. Due to the high coolant temperature, a high thermal efficiency is expected and is assumed to be 40%. The fuel smear density and burnup axial expansion of the thorium-based fuels are taken to be respectively 75% and 8%, the same as for U-10Zr fuel.

### *2.3. Computation Methods*

The ANL suite of fast reactor analysis code system was used for the neutronics design. Fuel cycle analysis was performed with the DIF3D/REBUS-3 code system[4][5][6][7]. Region-dependent multi-group neutron cross-section sets generated for a typical metallic fuelled sodium-cooled fast reactor under the USDOE Integral Fast Reactor (IFR) program were utilized in this study.

The reactor effective delayed neutron fraction and prompt neutron lifetime are determined using the VARI3D perturbation code [8] and the reactivity feedback coefficients are determined by the direct differences in the eigenvalues for the nominal and perturbed conditions using the DIF3D code. Perturbed cross-section sets using 33-energy groups based on ENDF/B.VII database are used to calculate the reactivity feedback coefficients.

The sub-channel analysis code SE2-ANL [9] was employed for whole core temperature calculations. SE2-ANL is a modified version of the SUPERENERGY-2 [10] thermal-hydraulic code interfaced with ANL heating calculations. The SUPERENERGY-2 code is a multi-assembly, steady-state sub-channel analysis code designed specifically to perform efficient calculations of the detailed core-wide coolant temperature profiles in sodium cooled fast reactor core geometries.

### *2.4. AFR-100 Core Concept*

The AFR-100 is a 250 $MW_{th}$ sodium cooled fast reactor using metal fuel operating on a once-through fuel cycle with no fuel shuffling or reloading for 30 years. FIG. 1. shows its radial core layout which is divided into three different enrichment zones: 30 assemblies for the inner core, 48 assemblies for the middle core and 72 assemblies for the outer core. The fuel assemblies contain 91 fuel pins arranged in a triangular pitch array. The fuel pin diameter and cladding thickness are 1.49 cm and 0.5 mm, respectively. The active core height is 110 cm and the overall assembly height is 302.3 cm including the lower reflector, shield and fuel handling structures. The dimensions of all the assemblies are the same as those used in the reference AFR-100 design [1]

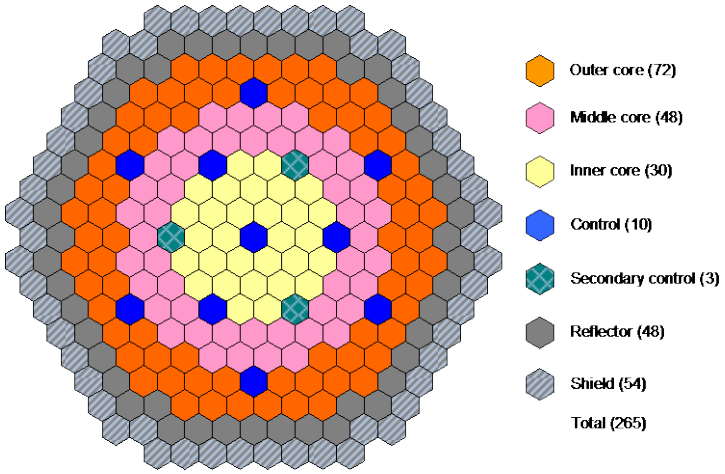


*FIG.1. Radial Core Layout of AFR-100 Core*

## 3. Fuel Conversion Scenarios

### *3.1. Progressive Fuel Conversion Scenario without Fissile Support*

The first scenario envisioned is ***progressive fuel conversion without fissile support***. The core is initially operated for a given number of years and a fraction of the original uranium-based fuel is replaced with thorium-based fuel. At the end of each cycle a fraction of the fuel is replaced with thorium-based fuel. The fresh thorium-based fuel is made of pure thorium. This is a breed and burn mode of operation [11]. The timing of the fuel conversion and number of assemblies replaced play major roles in maintaining criticality.

In order to minimize the number of transition cycles needed to achieve fuel conversion, it is desirable to replace as many assemblies as possible but this is restrained by the excess reactivity of the reference AFR-100 core when the fuel conversion starts. It is noted that the core may become sub-critical by replacing too many uranium-based fuel assemblies with thorium-based fuel assemblies because the thorium-based fuel assemblies do not contain any fissile material. Sensitivity studies were performed to determine the most favorable starting time for the fuel conversion, most favorable region of the core where the replacement takes place and maximum number of assemblies which can be replaced at a time. The fuel conversion starting time ranges from 2 to 22 years, for the replacement occurring in the inner, middle or outer core, and for a number of replaced fuel assemblies ranging from 9 to 30 out of 150.

It was found desirable to start the fuel conversion when the excess reactivity of the AFR-100 is the highest after 18 years of operation and replacing assemblies located in the outer core region to minimize the reactivity drop. The results obtained for this configuration for various number of fuel assemblies being replaced is shown in FIG. 2. It is found that for the core to remain critical at most 9 uranium-based fuel assemblies can be replaced. However, the corresponding cycle length is only 4 years, resulting in a discharge burnup significantly smaller than achieved with the reference AFR-100 core. Overall the progressive fuel conversion of the AFR-100 core is impractical without fissile support.

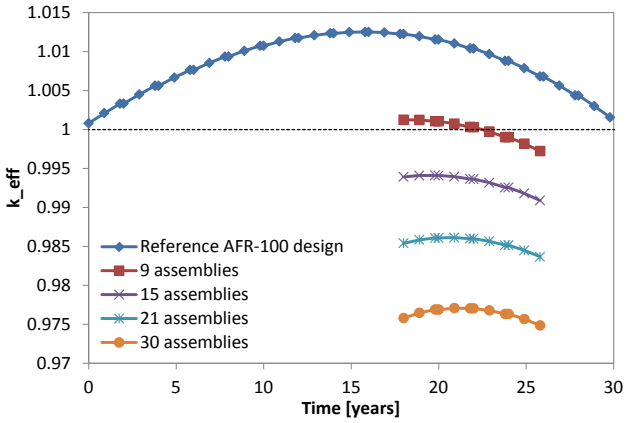


*FIG. 2. $k_{eff}$ Evolution for Various Number of Assemblies Replaced in the Outer Core Region*

### *3.2. Progressive Fuel Conversion Scenario with Fissile Support*

The second scenario envisioned is a ***progressive fuel conversion with fissile support***. This is similar to the previous scenario except that the fresh thorium-based fuel contains fissile material. In this study, the TRU recovered from the Pressurized Water Reactor (PWR) used nuclear fuel with a discharge burnup of 50 GWd/t is used as fissile material. The same enrichment zoning strategy as in the reference AFR-100 design [1] is used in order to obtain an acceptable radial power distribution and burnup reactivity swing. After all the original fuel assemblies have been replaced with Th-TRU fuel,

the same fuel replacement strategy is continued and the equilibrium mode of operation is reached after a few cycles. The core layout is shown in FIG. 3 for the 4-batches strategy.

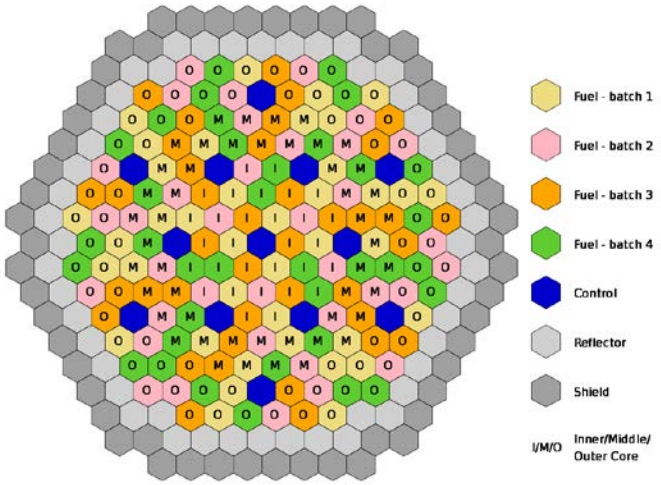


*FIG. 3. Th-TRU Fuelled AFR-100 Core Layout for a 4-batches Strategy*

The AFR-100 core is first operated for 24 years with LEU assemblies before the first batch is replaced with Th-TRU (18.6wt%) fuel. The time at which the first fuel replacement occurs was determined by a sensitivity study. It corresponds to the longest cycle length allowing maintaining the core critical during the transition period. Following the first replacement, another fuel batch is replaced every 7.04 years and after five cycles the equilibrium operation is reached. The main core performance parameters are provided in Table 1.

Table 1. Core Performance Parameters for the Th-TRU fuelled AFR-100 Core

| Characteristic | Transition | Equilibrium |
|---|---|---|
| Thermal power, MWt | 250 | 250 |
| Refueling interval, yr | 24 then 7.04 | 7.04 |
| Number of batches | 4 | 4 |
| Fuel form | LEU & Th-TRU | Th-TRU |
| TRU fraction, wt% | 18.6 | 18.6 |
| Initial HM loading, t | 23.9 | 22.9 |
| Burnup reactivity swing, %Δk | 1.25 | 0.77 |
| Average power density, W/cm$^3$ | 58.3 | 58.3 |
| Peak power density, W/cm$^3$ | 114.8 | 110.3 |
| Radial power peaking factor | 1.45 | 1.32 |
| Average discharge burnup, GWd/t | 120.4 | 100.9 |
| Peak discharge burnup, GWd/t | 222.2 | 162.3 |
| Peak fast fluence, $10^{23}$n/cm$^2$ | 9.00 | 5.52 |

The burnup values and peak fast fluence provided for the transition pertain to the LEU fuel only. It is observed that the average discharge burnup of the LEU fuel is larger than that of the reference AFR-100 design because of the longer residence time and higher enrichment. Subsequently, the peak fast

fluence of the LEU fuel reaches is significantly larger than the value currently demonstrated for HT-9 (~$4.0*10^{23}$) and the value assumed for the reference AFR-100 design ($6.0*10^{23}$). This can be easily solved by starting the fuel replacement after 15-20 years instead of 24 years. The burnup reactivity swing and the radial power peaking factor during the transition are identical to those of the reference AFR-100 design. At equilibrium, the smaller discharge burnup and different power distribution result in a peak fast fluence smaller than that of the LEU fuel during the transition period. It is also smaller than that of the reference AFR-100 core for which the fuel is discharged at a similar burnup

At equilibrium, approximately 84.9 kg of heavy metal are consumed per year. At discharge, the fissile plutonium content of a fuel has been reduced by approximately 59%, corresponding to a yearly net consumption of 49.8 kg for the whole core. About 69.6 kg of $^{232}$Th are yearly converted into $^{233}$U of which 37.4 kg are discharged. On average, 662 kg of $^{232}$Th and 151 kg of TRU are needed yearly to feed the Th-TRU fuelled AFR-100 core.

### *3.3. Whole Core Fuel Conversion Scenario with Fissile Support*

The last scenario envisioned is the ***whole core fuel conversion with fissile support***. Compared to the previous two scenarios, all the fuel assemblies are replaced at once with thorium-based fuel assemblies using 20% enriched LEU as the supporting fissile material. This one-batch fuel management scheme is similar to that used for the reference AFR-100 core concept: no shuffling occurs and every fuel assembly is replaced with a fresh one when the reactor becomes subcritical. In order to improve the $^{233}$U breeding ratio, a heterogeneous core configuration was considered rather than a homogeneous one. Due to cooling considerations, the separation of the LEU and pure thorium fuel was considered axially: i.e., the fuel pins are made of several layers of LEU and thorium. The fuel assemblies, loaded in the inner, middle and outer core region, as shown in FIG. 1, are made of a different number of $^{232}$Th and 20% enriched LEU layers and layer thicknesses. The inner, middle and outer core assemblies are made of 9, 7 and 7 layers, respectively, as shown in FIG. 4. The total mass of $^{235}$U required, 3442 kg, is almost equal to that required in the reference AFR-100 core. The layer thicknesses have been chosen so as to obtain an acceptable radial power distribution, small burnup reactivity swing and to maintain the core critical for as long as possible without refueling or shuffling. At BOL the thorium represents about 25% of the total heavy metal mass. The main core performance parameters are provided in Table 2.

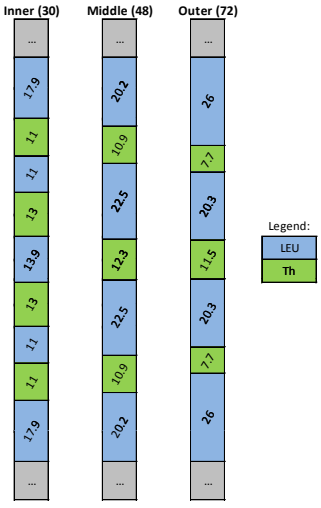


*FIG. 4. Inner, Middle and Outer Fuel Assemblies Layers (in cm)*

Table 2. Core Performance Parameters of the LEU/Th Fuelled AFR-100 Core

| Characteristic | Value |
|---|---|

| | |
|---|---|
| Thermal power, MWt | 250 |
| Refueling interval, yr | 18 |
| Number of batches | 1 |
| Fuel form | LEU/Th layers |
| LEU enrichment, % | 20 |
| Burnup reactivity swing, %Δk | 1.16 |
| Specific power density, MW/t | 10.96 |
| Average power density, W/cm$^3$ | 58.3 |
| Peak power density, W/cm$^3$ | 123.1 |
| Radial power peaking factor | 1.35 |
| Average discharge burnup in LEU/Th, GWd/t | 79.4/23.1 |
| Peak discharge burnup in LEU/Th, GWd/t | 126.5/40.7 |
| Peak fast fluence, $10^{23}$ n/cm$^2$ | 3.47 |

It is found that with no shuffling or refueling, the core remains critical for 18 years. The average discharge burnup of the LEU fuel is 79.4 GWd/t and that of the thorium fuel is 23.1 GWd/t. Compared to the reference AFR-100 core where no thorium is used, the LEU fuel discharge burnup is approximately 21% smaller and the amount of energy produced per kg of $^{235}$U is ~40% smaller. The burnup reactivity swing and the radial power peaking factor are slightly smaller than those observed for the Th-TRU fuelled AFR-100 core and for the reference AFR-100 design.

After 18 years of operation, 1.5 tons of heavy metal have been consumed. About 1659 kg of uranium and 403 kg of thorium have been consumed by fission or converted by neutron absorption into TRU and $^{233}$U, which are in turn partially consumed in situ. The discharged fuel contains approximately 262 kg of $^{233}$U, 2123 kg of $^{235}$U and 513 kg of fissile plutonium. The required natural uranium and separation work unit to fill the initial core are 827 tons and 660 tons, respectively, with an assumed tail enrichment of 0.3%.

## 4. Orifice Design and Steady-State Thermal-Hydraulic Performance

Reactor orifice zoning and flow allocation were determined through steady-state thermal-hydraulic analysis using the sub-channel analysis code SE2-ANL. The assembly orifice zoning and the flow rate allocation to the assemblies in each orifice zone were iteratively determined until all thermal-hydraulic design criteria were met. The orificing and thermal hydraulic analyses are performed for the two scenarios discussed in Section 3.2 and 3.3. For the progressive fuel conversion scenario using Th-TRU fuel, the analysis is performed for the cycle where the maximum power peaking occurs which is the first transition cycle

The fuel assemblies are grouped into nine orifice zones for the progressive conversion scenario and in eight zones for the whole core replacement scenario. In both cases the non-fuel assemblies are grouped into four orifice zones. FIG. 5a and 5b show the coolant flow rate and velocity in each assembly for the two scenarios. The coolant flow rates and velocities are small because of the low power density, compared to conventional fast reactors.

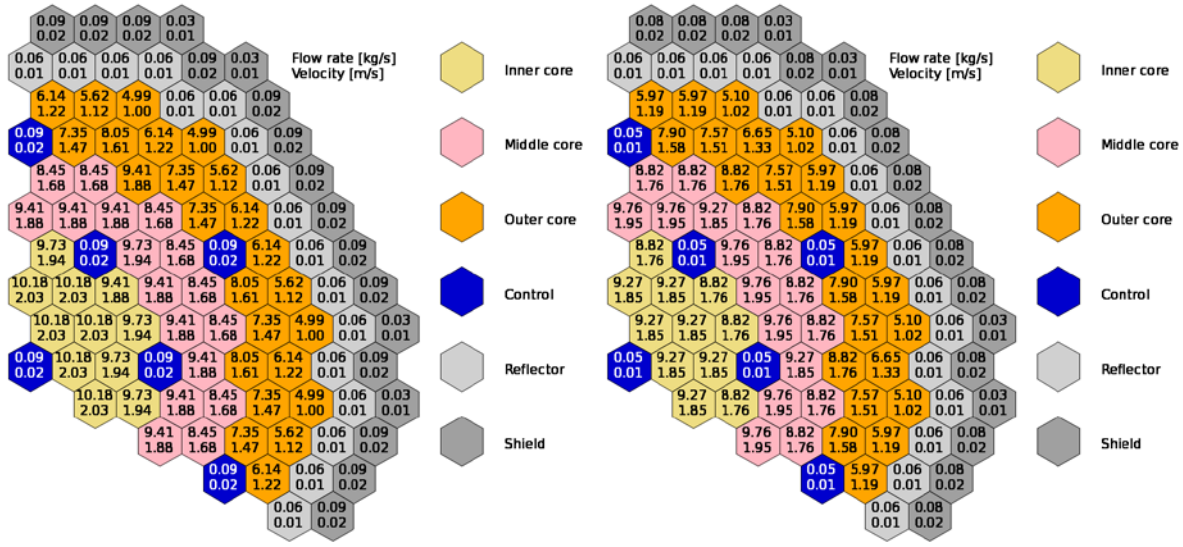


*FIG. 5a and 5c. Coolant Flow Rate and Velocity for the Progressive Fuel Replacement (left) and Whole Core Fuel Replacement Scenarios (right)*

FIG. 6a and 6b show the peak cladding inner wall temperatures with 2σ hot channel factors that were employed in the CRBR core design [12] for the two scenarios. The maximum peak 2σ cladding inner wall temperature occurs at BOC in a LEU fuel assemblies located in the outer core region for the progressive replacement scenario. For the whole core replacement scenario it occurs at BOC in several of the fuel assemblies loaded in the outer core region. The maximum peak temperatures for those two scenarios are 681°C for the progressive replacement scenario and 669°C for the whole core replacement scenario. Those temperatures are lower than the target fuel-cladding eutectic temperature for advanced metal fuel (700°C). Due to the little data available regarding thorium-based metal fuel, especially with respect to the fission product migration which limits the fuel-cladding eutectic temperature, the same eutectic temperature as for metallic uranium fuel is used throughout the core.

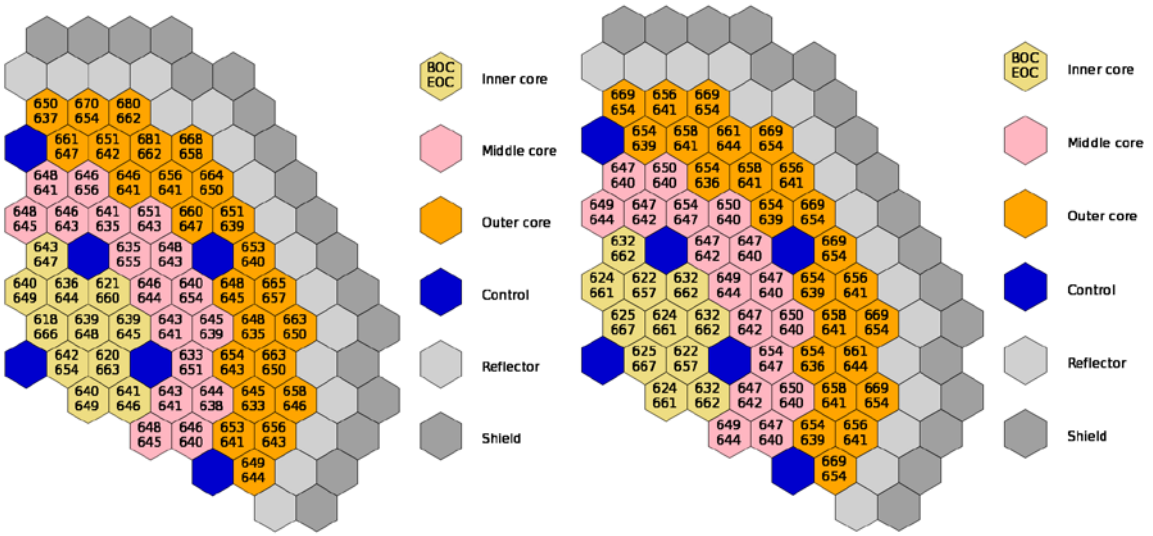


*FIG. 6a and 6b. Peak 2σ Cladding Inner Wall Temperatures (°C) for the Progressive Fuel Replacement (left) and Whole Core Fuel Replacement Scenarios (right)*

The overall peak fuel temperature is 766ºC for the progressive replacement scenario and 699°C for the whole core replacement scenario. No data is available regarding the melting temperature of Th-TRU, so the same melting temperature as for U-Zr, 1200°C, is assumed. This assumption appears to be conservative since the pure thorium melting temperature, 1755°C, is higher than that of pure uranium, 1132°C, and that the thorium based fuels have typically a larger melting temperature than the equivalent uranium based fuel [13]. The margin to fuel melting remains larger than 400°C for both scenarios which is much higher than that of a typical fast reactor due to the derated power density.

## 5. Kinetic Parameters and Reactivity Feedback Coefficients

The reactor kinetics parameters and reactivity coefficients were evaluated using the VARI3D and DIF3D codes for the scenarios discussed in Sections 3.2 and 3.3. In order to measure the capacity of the AFR-100 cores studied to attain inherent safety responses to Anticipated Transients without Scram (ATWS), the integral reactivity parameters of the quasi-static reactivity balance analysis [14] were calculated. For acceptable asymptotic core outlet temperatures resulting from possible unprotected accident scenarios such as Loss of Heat Sink (ULOHS), Transient Over-Power (UTOP), Loss of Flow (ULOF), chilled inlet, and pump over-speed, the integral reactivity parameters A, B, and C should satisfy the following three sufficient conditions:

$$A/B < 1 \text{ and } 1 < C \cdot \Delta T_c / B < 2 \text{ and } \Delta\rho_{TOP}/|B| < 1$$

The transient overpower initiator ($\Delta\rho_{TOP}$) was calculated using the number of control assemblies of the primary control system (10) and a first rod out interaction factor of 1.15.

### *5.1. Th-TRU Fuelled Core based on Progressive Fuel Conversion Scenario*

For the progressive replacement scenario, the calculated sodium void worth, Doppler coefficient and axial and radial expansion coefficients are presented in Table 3 at the BOC and EOC of the two first transition cycles as well as at the BOC and EOC of the equilibrium cycle. The reactivity coefficients for the initial 24 years cycle are the same as those of the reference AFR-100 design [1].

Table 3. Kinetics Parameters and Reactivity Feedback Coefficients for the Th-TRU Fuelled Core

| | Transition cycle 1 | | Transition cycle 2 | | Equilibrium | |
|---|---|---|---|---|---|---|
| | BOC | EOC | BOC | EOC | BOC | EOC |
| Effective delayed neutron fraction ($\beta_{eff}$) | 0.0045 | 0.0042 | 0.0038 | 0.0037 | 0.0029 | 0.0029 |
| Radial expansion coefficient, ¢/°C | -0.16 | -0.18 | -0.20 | -0.21 | -0.28 | -0.27 |
| Axial expansion coefficient, ¢/°C | -0.05 | -0.06 | -0.07 | -0.07 | -0.09 | -0.09 |
| Sodium void worth, $ | 2.57 | 2.97 | 3.48 | 3.61 | 4.71 | 4.24 |
| Sodium density coefficient, ¢/°C | 0.08 | 0.10 | 0.11 | 0.12 | 0.15 | 0.14 |
| Doppler coefficient, ¢/°C | -0.07 | -0.07 | -0.07 | -0.07 | -0.07 | -0.06 |

At the beginning of the transition period, the effective delayed neutron fraction corresponds to that of U-Pu metal fuel and as the LEU fuel is being replaced with Th-TRU fuel, the effective delayed neutron fraction decreases. Within the transition cycles the delayed neutron fraction is decreasing as $^{235}U$ is consumed and $^{239}Pu$ bred. At equilibrium no change is observed within a cycle because the effective delayed neutron fraction of $^{233}U$ and $^{239}Pu$ are similar.

During the transition period, it is observed that the sodium void worth is increasing between cycles as a result of replacing LEU fuel assemblies with Th-TRU fuel assemblies as well as within a cycle. This

latter effect is due to the increasing concentration of $^{239}$Pu in the core. At equilibrium $^{239}$Pu is consumed and $^{233}$U is bred, slightly decreasing the sodium void worth within a cycle. The sodium void worth is always positive and reaches its maximum at equilibrium because of the small effective delayed neutron fraction.

The Doppler coefficient always remains approximately constant because the reactivity increase due to $^{239}$Pu is offset by the decreasing effective delayed neutron fraction. The axial and radial expansion coefficients follow an evolution inverse to that of the effective delayed neutron fraction. They are getting more negative during the transition period and remain approximately constant at equilibrium.

The results of the quasi-static reactivity balance analysis are provided in Table 4 at BOC and EOC for the first two transition cycles and at BOC and EOC for the equilibrium cycle. The values reported in this table indicate that the Th-TRU fuelled AFR-100 core has overall favorable inherent safety features.

Table 4. Integral Reactivity Parameters for the Th-TRU Fuelled Core

| | Transition cycle 1 | | Transition cycle 2 | | Equilibrium | |
|---|---|---|---|---|---|---|
| | BOC | EOC | BOC | EOC | BOC | EOC |
| $A/B < 1$ | 0.21 | 0.20 | 0.19 | 0.18 | 0.14 | 0.13 |
| $1 < C\Delta T_c/B < 2$ | 1.10 | 1.08 | 1.05 | 1.04 | 1.01 | 1.03 |
| $\Delta\rho_{TOP}/\lvert B\rvert < 1$ | 0.77 | 0.46 | 0.59 | 0.09 | 0.78 | 0.07 |

## *5.2. LEU/Th Fuelled Core based on Whole Core Fuel Conversion Scenario*

The reactor reactivity feedback coefficients calculated for the whole core replacement scenario are presented in Table 5 at the BOL, middle-of-life (MOL) and EOL.

Table 5. Kinetics Parameters and Reactivity Feedback Coefficients for the LEU/Th Fuelled Core

| | BOL | MOL | EOL |
|---|---|---|---|
| Effective delayed neutron fraction ($\beta_{eff}$) | 0.0071 | 0.0063 | 0.0056 |
| Radial expansion coefficient, ¢/°C | -0.11 | -0.12 | -0.14 |
| Axial expansion coefficient, ¢/°C | -0.03 | -0.04 | -0.04 |
| Sodium void worth, $ | -0.14 | 0.18 | 0.55 |
| Sodium density coefficient, ¢/°C | ~-0.01 | 0.01 | 0.02 |
| Doppler coefficient, ¢/°C | -0.03 | -0.03 | -0.03 |

All the reactivity coefficients vary almost linearly during the cycle. With the exception of the sodium void worth reactivity coefficient, the reactivity coefficients variation is mostly due to the change in the effective delayed neutron fraction. At BOL the delayed neutron fraction corresponds to that of LEU fuel. As $^{239}$Pu and $^{233}$U are bred, the delayed neutron fraction decreases. At BOL the sodium void worth is slightly negative because of the large reactor neutron leakage probability and use of thorium fuel. As the $^{239}$Pu concentration increases and the radial power distribution shifts toward the core center, decreasing the neutron leakage probability, the sodium void worth increases. It remains significantly smaller than the sodium void worth observed in Section 5.1 and for the reference AFR-100 core.

The results of the quasi-static reactivity balance analysis are provided in Table 6. These results indicate that the inherent safety features of the whole core replacement scenario are favorable. All the sufficient conditions are met with comfortable margins, in particular due to the small sodium void worth.

Table 6. Integral Reactivity Parameters for the LEU/Th Fuelled AFR-100 Core

| | BOL | MOL | EOL |
|---|---|---|---|
| $A/B < 1$ | 0.12 | 0.11 | 0.11 |
| $1 < C\Delta T_c/B < 2$ | 1.24 | 1.20 | 1.17 |
| $\Delta\rho_{TOP}/\lvert B\rvert < 1$ | 0.43 | 0.57 | 0.00 |

## 6. Conclusion

The feasibility of fuel conversion from uranium to thorium in a fast reactor was assessed along with the impact on the reactor performance. The results were obtained for a small size fast reactor core, the AFR-100, but the overall conclusions are expected to be also valid for a more conventional fast reactor design. In addition to the neutronics performance, the thermal hydraulics and inherent safety performance of the scenarios proposed have been assessed.

Three main scenarios were envisioned in order to load thorium-based metal fuel in the AFR-100 core. The first scenario is a progressive fuel conversion without fissile support. It consists in progressively replacing the uranium-based fuel with pure thorium-based fuel without fissile material addition. This was found to be impractical since only 9 out of 150 fuel assemblies can be replaced and the resulting core remains critical for only four years, resulting in a very low fuel discharge burnup.

The second scenario consists of operating the reference AFR-100 core for 24 years and then replacing a batch every 7.04 years with thorium-based fuel mixed with TRU. The TRU mass fraction required is 18.6%. The original uranium-based fuel is discharged with an average burnup of 120 GWd/t and the thorium-based fuel with an average burnup of 101 GWd/t. The thermal hydraulic and preliminary inherent safety performances of this core were found satisfactory and comparable to those of the reference AFR-100 core.

The last scenario proposed consists in replacing all the original uranium-based fuel assemblies at once with assemblies made of several layers of LEU and thorium. This is a once-through fuel cycle, similar to the reference AFR-100 core. By using the same mass of $^{235}U$ as that used in the reference AFR-100 core, the thorium corresponds to 25% of the total heavy metal mass and the core remains critical for 18 years without refueling or shuffling. The average LEU and thorium discharge burnups are 79 GWd/t and 23 GWd/t, respectively. The thermal-hydraulic performance of this core was found satisfactory and the preliminary inherent safety performance of this core was found to exceed that of the reference AFR-100 core because of the significantly smaller coolant void worth.

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# Fast reactors as a solution for future small-scale nuclear energy

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**Abstract.** The world safe and sustainable development in the 21st century depends on the ability to build a new global energy infrastructure that could provide efficient access to energy resources and reduce greenhouse gas emissions. In a growing number of countries there is an increasing interest in decentralized, small-scale energy systems. Small nuclear power plants, based on fast reactor technology, could become a new platform of the decentralized energy system.

## 1. A new platform of the decentralized energy system

In the 21st century, safe and sustainable development of the world's economy largely depends on the ability to build a better energy infrastructure that would provide efficient access to energy resources globally while reducing greenhouse gas emissions.

Since 1992 more than 311 GW of generating capacity was built in the USA, all in the form of 100-300 MW natural gas power plant units with no investments in large energy-generating facilities. In Russia small-scale energy units cover about 70% of installed capacity.

Small nuclear power plants can become a better platform for decentralized energy systems providing better levels of accessibility, safety and environmental friendliness. Small-scale reactors can complement large nuclear and non-nuclear plants and help to expand to previously untapped markets. Small nuclear complexes can also provide an extended range of products: electricity, heat / steam and desalinated water to remote residential areas and industrial installations.

Currently there are dozens of concepts for innovative SMRs which are under development in the IAEA Member States. An increasing number of countries show strong interest in novel small & medium -scale nuclear energy systems (SMR) [1]. For example:

| | | |
|---|---|---|
| Korea | SMART | In July 2012, the Korean Nuclear Safety and Security Commission issued theStandard Design Approval for the 100 MW(e) SMART |
| USA | NuScale<br>mPower<br>W SMR | US-DOE is funding 452M$ in 5 years for two (2) out of the four US competing iPWR SMRs |
| Argentina | CAREM-25 | Site excavation for CAREM-25 was started in September 2011 |
| Japan | 4S | Toshiba had promoted the 4S for a design certification with the US NRC |
| Russia | KLT-40s | 2 modules marine propulsion-based barge-mounted KLT-40s are under construction, 90%. |
| | SVBR-100 | The lead-bismuth cooled SVBR-100 is planned to be launched by 2018 |

## 2. Is SMR something new?

According to the IAEA classification Small-sized nuclear reactors are the reactors with less than 300 MW(e) of installed capacity. , Medium-sized reactors have capacity from 300 to 700 MW(e). Based on this definition at the beginning of 2011 the world-wide number of operating SMRs was about 125 (based on 1970-80s technologies) with about 57 GW(e) generation capacity. The number of countries with SMR was 28.

Despite the vast accumulated operating experience and more recent design innovations none of the innovative SMRs is currently commercially available [2]. Only about two dozen SMR concepts reached the advanced design stage with much lesser number reaching the licensing stage.

Accordingly we should attempt to explore the reasons behind the slow progress, and define the key motivational drivers for deployment of innovative SMRs.

## 3. Key drivers for SMR deployment

Major benefits of the atomic energy are widely known: energy supply stability, zero-carbon emission, low operation costs. These are accompanied by equally well known challenges: higher capital costs, non-proliferation resistance, spent nuclear fuel (SNF) and radioactive wastes management.

Additionally, small nuclear complexes can have challenges related to safety, load following capabilities, operational and maintenance simplicity, and economical efficiency. The industry professionals state among the major challenges to the SMR commercialization issues of safety, proven technology and public acceptance [2].

To compete against alternative solutions SMRs must exhibit some evident technical and economical advantages. These advantages should be actively promoted to the stakeholders including the general public, policy makers, energy consumers. More importantly, these advantages must be capable for demonstration and verification via successful operation of pilot units.

In our opinion, among the major drivers that can accelerate SMR deployment are the reactor`s inherent safety, which is based on natural physical and chemical principles, and LUEC competitiveness within its market niche. Non-proliferation resistance, SNF and radioactive waste management etc. should also be resolved using the best practices (existing or developing) of the nuclear industry.

## 4. Inherent safety

Growing safety requirements for the nuclear power plants (NPP) demand application of a growing number of active and passive safety systems and defense-in-depth barriers, which mitigate the probability of severe accidents and their consequences. This probability, failures of the equipment, safety systems, protection barriers, and personnel’s errors are considered as random events. Probabilistic safety analysis (PSA) reports severe accidents probability being very low (~ 10-6 per reactor-year and less) but do not exclude their realization.

The NPP hazard is determined by two factors [3]:

- radiation potential accumulated, i.e., total radiotoxicity contained in the reactor,
- probability of radioactivity release into the environment.

The first factor does not depend strongly on the reactor type, and is determined mainly by the amount of fission products depending primarily with thermal power of the reactor and duration of its operation, i.e., by energy production.

The second factor is determined by the reactor type and depends on reactivity margin, feedbacks, design, and potential energy accumulated in the reactor (nuclear energy, internal thermal energy,

coolant compression energy, chemical energy), which can cause exhaust of radiotoxicity into the environment.

The nuclear fission energy which can be released under conditions of reactivity accidents must be minimized in principle, at the reactor design stage by limiting the reactivity margin, use of feedbacks, and by various engineering solutions to exclude.

Otherwise, potential (non-nuclear) energy stored in the reactor cannot be changed by engineering solutions. The crucial coolant characteristic is a value of potential energy stored in a volume unit of coolant (specific energy). This parameter defines a safety level of the NPP (Table 1).

Table 1. The values of specific potential energy for different coolants

| Coolant type | Water | Sodium | Lead, Lead-Bismuth eutectic |
|---|---|---|---|
| Parameter | P = 16 MPa, t = 300ºC | t = 500ºC | t = 500ºC |
| Maximum potential energy, GJ/m3, including: | ~ 21,9 | ~ 10 | ~ 1,09 |
| Thermal energy | ~ 0,90 | ~ 0,6 | ~ 1,09 |
| Potential compression energy | ~ 0,15 | None | None |
| Potential chemical energy of interaction | ~ 11,4 (with zirconium) | ~ 5,1(with water)<br>~ 9,3 (with air) | None |
| Potential chemical energy of interaction of hydrogen | ~ 9,6 | ~ 4,3 | None |

Non-nuclear energy “stored” in the reactor and associated with reactor coolant is a crucial factor determining the nuclear power plant safety/hazard performances as it is inherent safety feature. The data in the Table 1 illustrates the inherent safety advantage of the fast reactors (especially with lead or lead-bismuth coolant) over more traditional water-cooled thermal-neutron reactor.

## 5. LUEC competiveness

It was discussed many times [4], however not proved that SMRs in its mature state (i.e. serial production) will possess such practical and “selling” points as modularization, factory readiness, learning curve, match of supply to demand, shorter construction period, affordability (lesser investment requirements) etc.

All these values can to some extent smooth the difference between capital cost (per kW) for large scale (1 GW) and small reactors (different estimations show 5-20% difference).

However the scale factor influence cannot be overcome totally. Accordingly SMR is not a competitor of large scale power source of any type of generation, and its competiveness should be measured and

adjusted in LUEC1 terms within its market niche, conservatively taking into account cost saving factors.

The potential market for SMR is different than those for large scale reactors: decentralized grids, remote areas, co-generation & fresh water production, industrial heat etc.

Table 2. SMR estimated LUEC and range of competiveness for selected countries

| Country | LUEC*, $/MW·h | Competiveness range |
|---|---|---|
| China | 60-70 | CHP coal plants 100-300 MW(e); co-generation, desalination, renewable energy - wind |
| India | 80-100 | Coal plants, Solar PV |
| Brazil | 60-70 | Coal plants 100-600 MW(e), Gas CCGT, Solar PV |
| Russia | 60 | Coal plants 100-400 MW(e), co-generation/municipal heat, desalination, renewable energy - wind |
| Turkey | 80-100 | Coal plants, Gas CCGT, co-generation, desalination, renewable energy - wind |
| Indonesia | 100-120 | Coal plants, Gas, desalination, renewable energy – wind, Solar PV |
| South Africa | 100-120 | Coal plants, Gas, renewable energy – wind, Solar PV |

[*] *Discounting rate is taken as 10%*

The potential SMR market volume for 2020-30 years for the selected states can be evaluated as 25-30 GW(e).

## 6. SVBR-100 inherent safety reactor

SMR deployment technical and economic challenges have been addressed in multi-modular nuclear power complexes which are based on SVBR-100 reactor – an integral type 100 MW(e) lead-bismuth fast reactor with inherent safety and high proliferation resistance features.

SVBR-100 is developed by leading Russian nuclear research and design institutes [5]. The project is managed by JSC “AKME-engineering” - a 50/50 joint venture of the State Russian Atomic Corporation “Rosatom” and private partner En+ Group, which is a part of the Basic Element Company. It’s planned to launch the first pilot unit by 2018 in the city of Dimitrovgrad (1000 km east to Moscow). SVBR can potentially become world’s first innovative commercial reactor.

SVBR design is based on more than 80 reactor-years operational experience of Pb-Bi cooled reactors for propulsion applications [6]. Table 3 presents SVBR-100 power plant specification.

Table 3. SVBR-100 Power Plant Specifications

| Parameter | Value |
|---|---|
| Reactor thermal output | 280 MW(th) |

[1] Levelized unit of electricity cost

| | |
|---|---|
| Electricity | 101 MW(e) |
| Process steam* | 580 tonnes/hour, saturated steam, p=6.7MPa, T~282.9°C |
| Municipal heat* | max. 81 MW(th) |
| Desalinated water* | max. 200 000 tonnes/day |
| Design load factor | 90% |
| Fuel campaign duration | 7-8 years (for UO2 fuel with 16.3 wt% U-235/U enrichment ) |
| Load following capability | 0.5-2% per minute in 70-100% power range |
| Reactor module weight | 270tonnes |
| Reactor module dimensions | 4.5 / 7.86 meters (diameter/height) |

* *if appropriate equipment is installed*

Due to the natural properties and design of SVBR-100 the reactor has sufficient non-proliferation protection which along with site & transportation safeguard and organizational measures minimizes the risk of un-authorized access to the fissile materials during the NPP lifetime.

## 7. Conclusions

Small nuclear power plants can provide a better platform for decentralized energy supply providing better levels of accessibility, safety and environmental friendliness. The optimal solution for SMR deployment isfast reactors with inherent safety that can meet growing safetyrequirements.

Small modular reactors with lead-bismuth coolant (SVBR-100) which are being developed in Russia in the framework of public-private partnership can be considered a prospective solution for small and decentralized energy future.

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# EC-FP7 ARCAS: technical and economical comparison of Fast Reactors and Accelerator Driven Systems for transmutation of Minor Actinides

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**Abstract.** The ARCAS project aims to compare, on a technological and economical basis, Accelerator Driven Systems and Fast Reactors as Minor Actinide burners. It is split in five work packages: the reference scenario definition, the fast reactor system definition, the accelerator driven system definition, the fuel reprocessing and fabrication facilities definition and the economical comparison. This paper summarizes the status of the project and its five work packages.

## 1. WP 1: reference scenario definition

The reference scenario considered in the frame of the ARCAS project refers to PATEROS project [1] where a regional scenario, at a European level, was analyzed in detail. Scenario 1 was taken into consideration, in which spent UOX and MOX fuel discharged from LWR is reprocessed (mono-recycled) in order to separate TRU from fission products (which, together with reprocessing losses, are sent to a geological repository). Reprocessed Pu and MA are recycled in the regional transmuter facility, which in this case is the ADS-EFIT (*Accelerator Driven System – European Facility for Industrial Transmutation*) [2] and blended with TRU separated from spent fuel of subcritical transmuter fuel cycle (as soon as available) in subsequent cycle passes (Figure 1). The final goals of the scenario are:

***a)*** to fully reprocess spent fuel legacy of some European countries (*Group A*), which are supposed to be in a stagnant or phasing-out scenario: *Belgium, Czech Republic, Germany, Spain, Sweden* and *Switzerland*, in order to eliminate all the TRU stocks, before the end of the present century;

***b)*** to store Pu (for a possible future use for the deployment of fast reactors, which were not simulated in this case) and to stabilize the MA inventory in European countries (*Group B*) pursuing nuclear energy generation: *France* was considered in this case.

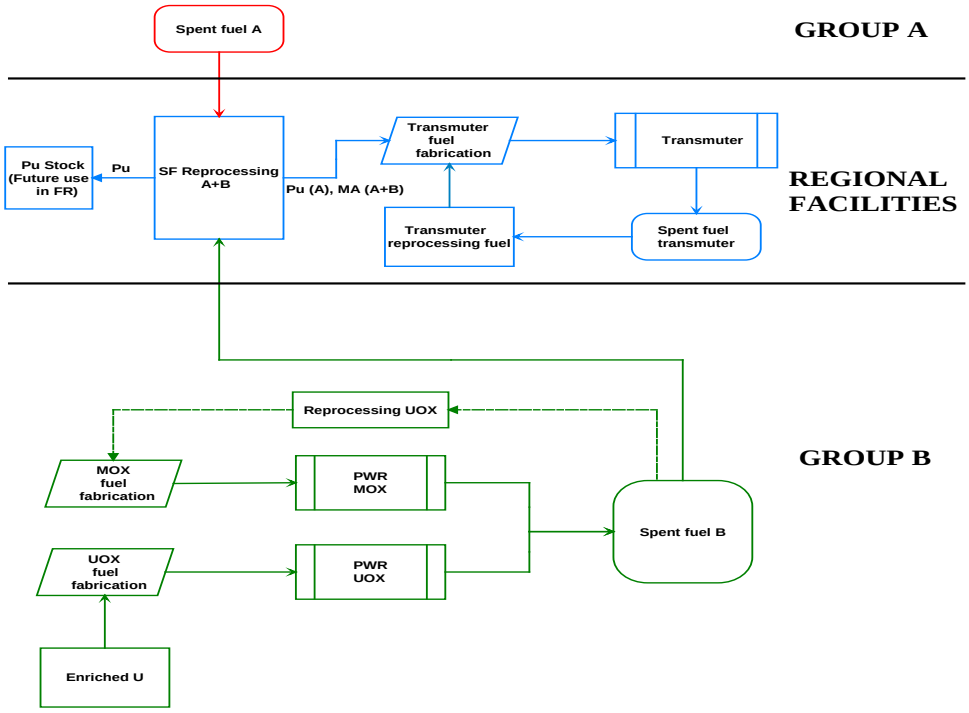


*FIG. 1. PATEROS simplified flow scheme*

According to scenario assumptions ADS-EFIT will be deployed in a regional centre starting from 2045 (this hypothesis should appear rather unrealistic) up to 2090 – then a constant energy production level – i.e. number of transmuters - is assumed, regional fuel cycle facilities such as reprocessing and fuel fabrication plants for innovative fast reactor fuel, and a spent fuel (SF) interim storage are considered. In particular a reprocessing capacity of 850 tonnes/year was assumed for Group A reprocessing plant, while 1700 tonnes/year were necessary in order to stabilize the inventory of Group B.

Calculations were performed with COSI6 – ver. 6.0.1, a code developed by CEA (Cadarache)[3]. In particular simulations addressed the MA streams (and their isotopic composition) evaluation from Group A (i.e. coming from a spent fuel storage after some decay time) and Group B (i.e. coming from a continuous feed from a PWR fleet with a 50,000 MWd/t burnup, fuelled by 90% UOX and 10% MOX, a 5 years cooling time and a total yearly energy production of 430 $TWh_e$). The outcome of the simulations is shown in Table 1: MA composition and a range of minimum and maximum annual values are indicated. It should be noted that the hypothesis that all European nations except France will phase-out during the present century may appear too optimistic from the MA waste stream amount point of view. If we consider that the nuclear power installed in France is today 63,130 $MW_e$, while the total for Europe is 169,932 $MW_e$ [4], a factor of ca. 2.7 should be considered (as it appears that a phase-out of nuclear energy in the near future in OECD countries is unlikely, due also to environmental concerns about global warming, as stated by IPCC or IIASA scenarios [5][6]).

Table 1. Proposed reference MA composition

| Nuclide | Content (%) |
|---|---|
| $Am^{241}$ | 39.55 |
| $Am^{242m}$ | 0.22 |

| | |
|---|---|
| **$Am^{243}$** | **22.34** |
| **$Np^{237}$** | **32.91** |
| **$Cm^{243}$** | **0.059** |
| **$Cm^{244}$** | **3.97** |
| **$Cm^{245}$** | **0.95** |

Min. MA annual stream[1]: 2.3 tonnes/year
Max. MA annual stream[2]: 6.5 tonnes/year

It is important to stress however that the composition indicated above is just indicative: in order to adopt an efficient transmutation strategy in fact it is mandatory that fast systems (both critical and subcritical) run a closed cycle, by reprocessing their own fuel and recycling it in their fuel fabrication plants as first choice. This strategy will present some relevant consequences, which should in principle affect fuel cycle costs heavily:

Recycling of transmuters fuel, and then blending it with fresh fissile material in order to balance the fissioned mass, will obviously modify fuel isotopic vector at every new reactor load: this fact will require probably to modify fuel shares (e.g. inert matrix/heavy metal, amount of uranium, etc.), which, in their turn should affect safety coefficients, performance, burning capacity, etc.;

Recycling fuel in a closed cycle fashion will cause probably an accumulation of heavy elements, thus increasing fuel gamma and neutronic emission, such as also decay heat power: as new technologies, making use probably of remote handling and improved shielding issues, should be required this parameters should be taken into consideration accurately in costs evaluation.

Finally it should be taken into consideration in transmuters evaluation that if nations with a phasing-out policy are considered, as in case of PATEROS scenario, plutonium management should be an issue, especially if adopted transmuters are not specifically designed for this goal. Simulations show that if maximum MA stream cited above is adopted (i.e. present European nuclear fleet) a plutonium annual stream of ca. 24 tonnes/year results, which means an accumulation of more than 2100 tonnes of fissile material by the end of the present century. It should be pointed out also thatsmall nations that decide to continue to produce nuclear energy without planning to use produced plutonium in fast reactors (such as France) will have to adopt a proper strategy for its final disposal (regional transmuter design should take this issue properly into account).

## 2. WP2: Definition of the Fast Reactor System

In the frame of the EU CP-ESFR project [7], a basic SFR concept was proposed as a 'Working Horse' (WH) design, which was further optimized in an effort to improve the original reactivity coefficients. A short description of the optimized reactor concept is provided in this section, which will be used for analytical estimations in ARCAS. The cross section of the core is depicted in Figure 2.

The 3600 MWth core is composed of 225 sub-assemblies (S/A) in the inner core and 228 S/A in the outer core; 453 S/A in the whole reactor.

The core S/A are MOX type, where the composition is as follows (in weight percentage of the total Heavy Metal):

[1] PATEROS scenario
[2] PATEROS extended to all European countries with present energy production

- Inner core, 8 S/A active rows: 85.12% Depleted Uranium, 14.76% Pu, and 0.13% Am (as a result of Pu decay during fabricated fuel storage).
- Outer core, 4 S/A active rows: 82.72% Depleted Uranium, 17.15% Pu, and 0.12% Am.

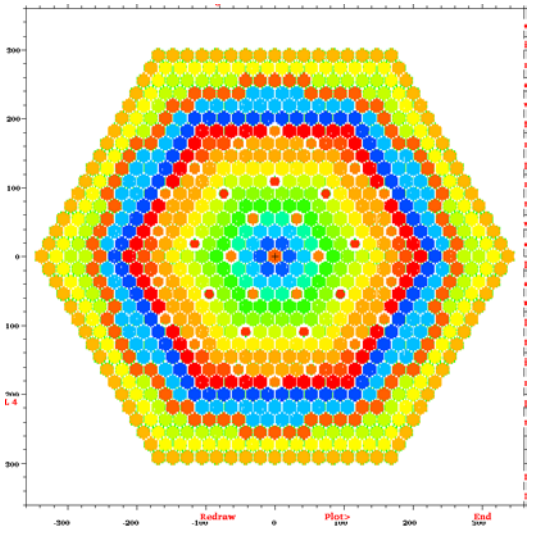


*FIG. 2. Cross section view of CP-ESFR core*

The core optimization is called CONF2 case and the features are:

- A lower axial blanket made of depleted Uranium dioxide.
- An upper Sodium plenum to enhance neutron leakage in the region in case of plenum voiding.
- An upper neutron absorber layer, above the Sodium plenum.

Before the optimization process, the basic 'Working Horse' design consisted only in the active length, 1 m high, with no lower axial blanket, sodium plenum and upper absorbing layer. The WH design was intended to be a break-even core.

It is found that MA homogeneous loadings in the reference reactor lead to moderate transmutation values, up to 6.9 kg/TWhth for 4%w loading, and noticeably deteriorates reactivity coefficients (Doppler constant and core void worth). However, the deterioration depends very much on the exact core configuration. Hence, dedicated core design strategies for lowering the MA impact have an important effect for the Doppler constant, whose deterioration may decrease from 40 to 15%. The reduced void worth deterioration is found similar before and after application the optimization guidelines (some 25% in both cases). On the other hand, the extended void worth significantly decreases compared to the reduced void worth, which means that optimization guidelines are promising and should be further pursued, even targeting negative core void worth. In the meanwhile, a combination of lower MA loading, 2.5%, and optimization guidelines seem to be a promising concept, as deterioration will be lower.

Heterogeneous blanket configurations lead in general to low deterioration of safety parameters or even to little improvement when core optimization guidelines are considered. However, concerning transmutation values, virtually no net transmutation is found or just a small net value after optimization guidelines. An interesting case between homogeneous and heterogeneous has been also presented with MA loading in the outer core (together with Pu) leading to medium transmutation values (3.6 kg/TWhth) and no deterioration of reactivity coefficients.

In any case, the impact of MA loading on reactivity coefficients expand over a range of results, which illustrates the necessity for rigorous safety analysis in order to advance the issue of the core feasibility

from the point of view of licensing. This is indeed an open field for research, as no fully dynamic safety analysis is yet available. Concerning first European scenario analysis, it is obtained that medium transmutation values (as of 2.8 kg/TWhth) in all reactors could lead to elimination of the neptunium and americium stock, out of the reactor site, in the frame of a century, and that the result is compatible with Pu breeding in all reactors. The curium mass stock, however, is not eliminated but just somewhat decreased. Such an objective would imply the fabrication of a very large number of MA bearing fuel assemblies at low contents, 2.5%w. Also, the Pu amount involved in fuel fabrication would be very large.

## 3. WP3: Definication of the Accelerator Driven System

The aim of Work Package was to select and characterize the reference Accelerator Driven System to be used in the ARCAS project. As, in Europe, there is only one design for an industrial transmutation facility available, the choice of the reference system was rather easy: the EFIT, European Facility for Industrial Transmutation, as designed in the 6$^{th}$ European Framework programme IP-EUROTRANS [8].

The accelerator foreseen in the EFIT design is an 800 MeV proton accelerator delivering 20 mA of current. This beam impinges on a windowless spallation target, where the induced spallation reactions produced the required source neutrons. The 19 central positions of the hexagonal core lattice house the spallation target which is surrounded by fuel assemblies. The number of fuel assemblies is such that the core, by design, will not become critical (even in accidental conditions). The reactor core is cooled by pure lead (as opposed to lead-bismuth eutectic as foreseen in the experimental facility XT-ADS). This allows a high inlet and outlet temperature (400°C and 480°C respectively) and as a consequence a rather high thermodynamic efficiency of 40% .

For the fuel one opts for uranium-free fuel since this avoids extra build-up of plutonium (by capture in U-238). Because there is a relationship between the energy produced and the material destroyed by fission (one fission produces about 200 MeV of energy), the final balance is always a loss of 42 kg/TWh$_{th}$ [9]. The design goal of EFIT was hence to have a loss of minor actinides as close to 42 kg/TWh$_{th}$ as possible and a loss of plutonium close to 0 kg/TWh$_{th}$ [10][11]. The second goal for the design of the core was to have a reactivity swing as close to zero as possible, reducing the power fluctuations during the cycle without the need to compensate for this using the proton accelerator.

Two types of advanced fuels have been analyzed in the EUROTRANS project: the CERCER option and the CERMET option. The former uses a MgO matrix, the latter a Mo matrix. For the CERMET, two sub-options have been analyzed: a matrix with natural Mo and a matrix enriched in the ligther isotopes of Mo, avoiding excessive neutron capture.

The ADS EFIT core used for this study is based on the one defined in the deliverables D3.2 and D3.3 from the AFTRA (**A**dvanced **F**uels for **TRA**nsmutation systems) domain (DM3) within the EURATOM FP6 integrated project (IP) EUROTRANS [12][13]. For this comparative study, we used EFIT-400 (AFTRA) core with one zone configuration, a thermal power of 400 MW and with the two selected fuels for EFIT core: CERCER with MgO matrix and CERMET with Mo matrix enriched in $^{92}$Mo. As shown in Figure 3, the core contains 6 rings of fuel assemblies (FAs), surrounded by 2 rings of reflector ones and a cylindrical core barrel with 30 mm as thickness. The spallation target and the surrounding region (containing mainly lead) occupy the space created by the withdrawal of 19 fuel assemblies from the central region proposed in the EFIT reference design [14].

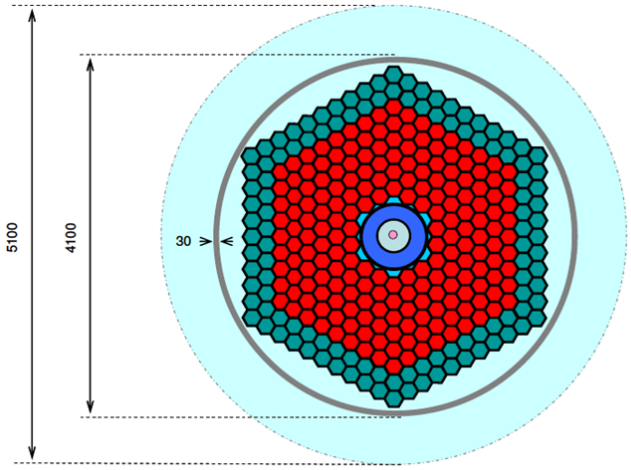


*FIG. 3. Single-zone reference core model of the EFIT-400: radial layout*

The results of MA transmutation rate are calculated using the ALEPH code (SCK•CEN home-made code) [15]. The ALEPH code is designed to combine a Monte Carlo codes (MCNP or MCNPX) for spectral calculations with a modified version of ORIGEN-2.2 code [16] for evolution calculation. The nuclear data used are based on the JEFF-3.1 library [17].

As shown in Table 2 and Table 3, in terms of total MA transmutation rate, for both ARCAS and EFIT vectors, the transmutation performances are the same: reaching values of 39kg/TWh and 36kg/TWh for EFIT-400 CERCER fuel and EFIT-400 CERMET fuel respectively.

Table 2. MA Transmutation rates (kg/TWh) for EFIT-400 with CERCER fuel

| | **MA ARCAS vector** | **MA EFIT vector** |
|---|---|---|
| **Np** | -17.379 | -1.331 |
| **Am** | -29.589 | -44.734 |
| **Cm** | 8.206 | 7.349 |
| **Total MA** | -38.76 | -38.71 |

Table 3. MA Transmutation rates (kg/TWh) for EFIT-400 with CERMET fuel

| | **MA ARCAS vector** | **MA EFIT vector** |
|---|---|---|
| **Np** | -15.583 | -1.215 |
| **Am** | -27.096 | -43.692 |
| **Cm** | 7.010 | 6.771 |
| **Total MA** | -35.67 | -35.5 |

## 4. WP4: Definition of the fuel reprocessing and fuel fabrication facilities

The objective of ARCAS WP4 is to define a fuel fabrication plant and a reprocessing plant for fast reactor (FR) and ADS fuels in order to compare costs. Clearly, these plants may be different for each neutron system, but that difference will only depend on the fuel types and their irradiation conditions.

The output from WP4 comprises baseline information which identifies process differences upon which an economic assessment of heterogeneous fuel fabrication plant and spent nuclear fuel reprocessing plant for FR and ADS can be made in WP5. Inert matrices of Mo or MgO (Yttria Stabilised Zirconia) and pure MgO have been selected as model fuels for ADS and FR systems respectively. The case for reprocessing of ADS fuel using pyrochemical technology and the fabrication of ADS fuel using Sol Gel is made and appropriate high level case studies completed. Similarly, the case for reprocessing FR fuel using aqueous technology and the use of powder metallurgy as the preferred fuel fabrication route for $UO_2$ blanket and U/Pu oxide core fuel is made. Sol Gel is the preferred route for minor actinide (MA) fuel fabrication.

Reprocessing options are expected to fall into two “camps”. These are:

- Materials well suited to existing fuel fabrication processes and compatible with nitric acid/organic phase, PUREX or GANEX type, separation processes and;
- Those that are not where non-aqueous process routes (i.e. pyrochemical) are most likely to be deployed.

The following assumptions are made for the fast reactor fuel

- Heterogeneous actinide and MA fuel pins are U/Pu and inert matrix (MgO)/MA;
- Based on an oxide system;
- High Pu content in FR core;
- MgO is soluble and easily diverted within an aqueous reprocessing option, therefore only aqueous reprocessing options are required for this scenario;
- Oxidation and dissolution of high Pu content fuel and MA is possible;
- Aqueous reprocessing solvents are sufficiently stable to very high burn-up fuel, however, in extreme cases, the effect of radioactive content in aqueous reprocessing can reduce the effect of solvent extraction dramatically. This is especially true in the first stages of these processes where the organic solvent is in contact with fission product activity in the aqueous solution;
- An organic phase clean-up and recovery step will be included in any reprocessing scheme to maintain process efficiency, and;
- Aqueous processes will be assessed in conjunction with an appropriate scenario (e.g. sufficient cool-down time).

and for the Accelerator Driven System fuel

- Fuels are based on an actinide oxide dispersed in an, inert matrix;
- Heterogeneous actinide and MA fuel pins – Mo or $MgO/PuO_2$ and Mo or $MgO/MAO_2$;
- Provided Mo is recovered and recycled at an early stage of reprocessing, then aqueous processes should be considered viable, if not;
- Pyro processing due to the potential issues of CaesiumPhosphoMolybdate (CPM) and insoluble product formation in aqueous processes is proposed;
- Pyro-processing is used for spent fuel when fabrication techniques have required a $ZrO_2$ or Yttria Stabilised Zirconia (YSZ) in the MgO matrix;

- Where aqueous reprocessing is selected, the assumptions shown under FRs above also apply here ;
- All fuels will be subject to very high burn-up.

Carbide and nitride systems have been discounted from this work due to their very low technological readiness levels in comparison to oxide and metal fuels.

Both aqueous and non-aqueous process steps have been investigated and various conclusions drawn. Two baseline processes for fuel fabrication have been considered Powder Metallurgy and Sol Gel. MOX fuel production is, of course, based on powder metallurgy and is well established at industrial scale however, the production of separate MA oxide fuels is not.

Preparation and production of heterogeneous oxide fuels for the double strata advanced fuel cycle using either FR or ADS is extremely challenging due to high alpha, high decay heat, high neutron emission and high gamma activity. Any fuel processing facility will therefore necessarily need to have very high integrity containment to prevent the spread of highly mobile alpha activity, include heavy shielding for the penetrating radiation and almost certainly require the deployment of remote engineering technology for some plant operations / plant maintenance purposes. No preferred technology for fuel fabrication was identified due to the low Technology Readiness Levels, however, with dedicated production lines for each fuel type, the technology of choice can be selected when suitable technological maturity is obtained. Costs were expected to be closely related to the number of unit operations rather than technology selection.

It should be noted, the ARCAS study has been bounded to include Cm heterogeneous targets, and therefore fuel fabrication plants are required to include heavy neutron shielding. Should the decision be taken to sentence Cm to a dedicated decay store, then shielding requirements become less demanding for fuel fabrication. This scenario however, is outside the scope of ARCAS.

For reprocessing plants, an analysis of the different unit operations for both aqueous and pyrochemical options was completed. A basic gap analysis highlighted the technical immaturity of both technology options and, as expected, they were found to have very low TRLs of 2-3.

All fuel fabrication processes, with the exception of U and MOX fuel are technically immature and assigned low Technology Readiness Levels of 2-3. Costs were expected to be related to the number of unit operations, shielding requirements and remote technology deployment rather than the technology selection, per se.

## 5. WP5: Economical comparison

The last work package of the ARCAS project is to gather all information from the other work packages in order to able to present a comparison between the two options of fast reactors or accelerator driven systems. An economic analysis and a business case description are being prepared for the EFR and EFIT nuclear plant designs with transmutation capabilities.

Two methods are used to determine the cost structures for the two options. One is to calculate the cost per kilowatthour electric and per metric ton actinide waste destroyed for each design separately. The GIF tool G4Econs is being used for this. The other method is to calculate these costs for certain defined scenarios of reactor parks. Three scenarios are being considered: only Fast Reactors with heterogeneous targets, double strata with Fast Reactor burners, and double strata with ADS transmuters. (The double-strata nuclear fuel cycle consists of the commercial reactor fuel cycle (the 1st stratum cycle) and nuclear transmutation fuel cycle (the 2nd stratum cycle) based on FR or ADS that transmute the minor actinides generated in the 1st stratum.)

With the number of units needed per GWe of LWR installed and the investment cost of a transmutation unit, the investment cost per GWe is determined. For selected nuclear evolution

scenarios, the total investment cost needed for transmutation can be determined. Also, the total generating costs are compared, giving an answer to the question on how much the MA transmutation would add to the cost of kWh. These costs would include both the investment, operational and fuel cycle costs. The fuel cycle costs consist of all the parts of the closed cycle, including reprocessing and fuel/target fabrication.

Both FR and ADS have transmutation capabilities. As expected from their fuel loadings and spectra, the project work packages 2 and 3 have demonstrated that ADS have very superior capability for transmutation compared to FR. Also the required transportation of nuclear spent fuel and dedicated burner fuel can be limited because of the high concentration of minor actinides in ADS fuel. The challenging question is whether these advantages could compensate for the extra difficulties and then costs of building these facilities. Table 7 shows the cost advantages and disadvantages for the three reactor systems considered. Given the extra complexity of its design (need for a reliable powerful accelerator), ADS most probably have a higher LCOE (€/MWhe) than FR, who in turn have a higher LCOE than LWR. If transmutation is not needed, if Pu is not managed separately, then utilities using LWR have no incentive to pay the extra costs of MA transmutation and LWR are by far the best and probably only choice. If however utilities would be obliged legally to manage their heavy nuclides and in particular the remaining MA after Pu removal, then a market could emerge for transmutation. Calculating the electricity costs for a nuclear park consisting of LWR and transmutation facilities, the higher costs of electricity produced by ADS may then be balanced by its limited share in the energy mix and the bigger share of lower cost kWh produced by LWR.

Table 4. Comparison of LWR, FR, ADS costs advantages and disadvantages

| | LWR/BWR | Fast Reactor | ADS |
|---|---|---|---|
| **Construction cost** | + | - | -- |
| **Operation & maintenance cost** | + | - | -- |
| **Fuel costs** | | +- | - |
| **MA transmutation capacity** | -- | + | ++ |

## 6. Summary

The ARCAS project tries to address a crucial issue in the partitioning and transmutation debate: which options are technologically feasible and at what price. As a CSA project, it does not aim to perform Research & Development in the field, but rather to gather the available information and combine it in a global study. At the moment, the inventory and feed stock of minor actinides has been established, the reference Fast Reactor System and Accelerator Driven System have been defined. The fuel reprocessing and fuel fabrication facilities are being assessed and their choice finalized. The main work yet to perform is the combination of all in an economic comparison in the final work package.

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# Options for UK plutonium in SFR fuel cycles
## *Overview of current work*

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**Abstract.** The current study examines potential long-term options for stockpiled plutonium use in a UK fast reactor (FR) programme and the situations where FRs could become more advantageous than current options proposed by the NDA. This paper consists of a review of initial considerations and outlines the approach to be taken for further work. The UK has more than 90 tons of separated plutonium stockpiled due to its 1960s commitment to reprocessing. Originally intended for use in FRs, which never became economically viable, it is now labelled as a "zero value asset". Present NDA options aim to reduce the stockpile through irradiation of MOX in thermal reactors; direct disposal; or disposal following a period of storage. Recently, the UK's consideration of GE-Hitachi's PRISM reactor for plutonium burning suggests that different uses of FRs could, in the future, become advantageous with changing political and economic circumstances. These include: reducing dependence on resources; minimising long-term radio-toxicity of waste; proliferation concerns, or a combination of these factors, which will be reviewed.

## 1. Introduction

The basis of this study is to consider likely fast reactor (FR) fuel cycles in the UK and assess their relative benefits in terms of five high order metrics: (1) Impact on UK stored material and requirements for imported fissile material; (2) Technology readiness level; (3) Repository requirements; (4) Proliferation resistance; (5) Economic competitiveness. This paper sets out the initial considerations for the study and a review of relevant materials, which will form the main body of work and its analysis. Previous studies in this area have not considered the UK with its unique position – owning a large stockpile of civil plutonium.

FR fuel cycles are of little use at present due to the low cost of thermal reactor systems and abundance of uranium. Given time, this may no longer be the case. Future needs for a more sustainable base-load electricity generator could make FRs an obvious choice, and subsequently make plutonium a valuable commodity. Equally, given the evolution of technology and changes in public opinion, the opposite could happen and the need for nuclear power may vanish, making plutonium a liability.

### *1.1. The UK*

At present, the UK is continuing down the nuclear route, with 16 GWe of new nuclear power expected by 2025. The Department of Energy and Climate Change's (DECC) reasons for supporting new nuclear power is based on it being "the UK's most significant source of low carbon energy" and that prices do not fluctuate greatly with raw material prices[1]. However, past this round of new build it is uncertain what may happen and, at present, stock piled plutonium is classed as a "zero value asset"[2]. If the UK decides to move away from nuclear, disposing of this plutonium would be advantageous, eliminating any storage costs. However, if nuclear electricity generation continues in the UK and worldwide in the long-term, this stock of fissile material could become valuable as a substitute for dwindling uranium resources. With there being no "cost nothing"[3] option and uncertainty in the future of nuclear power, three high level options for plutonium have been outlined by the Nuclear Decommissioning Authority (NDA):[2][3][4]

- Long-term storage before disposal;
- Reuse of plutonium in thermal reactors as MOX before disposal or
- Immobilisation and disposal.

Recently, alternatives have been suggested and are being considered by the NDA. One option, which has made significant progress, is the reuse of plutonium in GE-Hitachi's PRISM Sodium Fast Reactor (SFR) prior to disposal[5]. This plan irradiates all plutonium in metallic fuel to the spent fuel standard before reloading it into the reactor to achieve the maximum burnup and thus generating as much electricity as possible. This increases the rate of return on initial investment. This option's consideration shows that there are situations where FRs could be considered in the UK, whether they be based on public opinion; international pressure or economically favourable solutions provided by the private sector.

All NDA options focus on the UK's large civil stockpile and methods to reduce it. Despite stockpile reduction being part of the study, a lot of options outlined focus on drivers other than this, considering long-term aspects such as sustainability. All options consider the use of SFRs rather than thermal reactors. Whilst thermal reactor fuel cycles have been extensively assessed in the UK and are suitable for comparison, they do not form part of this work.

## 2. Technical barriers

### *2.1. Sodium fast reactors*

Only SFRs are considered as they are the most commercially developed, have more experience (both in the UK and internationally) and a reasonable timescale. Two reactor types will be compared throughout the study. One will be a more commercially developed, large scale MOX reactor like that of the CDFR and EFR project, which the UK has experience with in terms of the fuel cycle and reactor technology. The other will be a smaller, modular reactor with metallic fuel like that of GE-Hitachi's PRISM reactor: the associated fuel cycle and reactor technology have significantly less operational experience. The fuel cycles for each reactor will be considered. For the MOX reactor, the more common approach of aqueous reprocessing and pelletised fuel at centralised facilities will be used. For the smaller metallic fuelled reactor type, on-site fuel cycle facilities will be considered using pyro-reprocessing techniques. The purpose of this comparison is to see if developing a less experienced fuel cycle that has intrinsic advantages over current technology is worth developing over a more "off the shelf" approach with lower development costs.

#### *2.1.1. Reactor experience and barriers*

Worldwide there is significant FR experience and the UK has extensive experience as part of the DFR (metallic fuel); PFR (MOX fuel) and input on the CDFR and EFR projects. One of the main barriers is technical expertise and a large skills gap exists if the UK was to manufacture and operate facilities. These would be significant and limiting factors in the time scale of deployment.

Despite such operational experience, the construction and running of a number of reactors has highlighted several reliability issues with SFRs. Some demonstration reactors, such as PFR; MONJU and Superphenix suffered prolonged shutdowns as a result of a number of problems: refuelling issues; sodium leaks and reactivity spikes, to name a few. As such, advances in materials; components and safety analysis would be highly useful before large scale deployment.

Table 1. Cumulative load factor of demonstration SFR reactors [6].

| **Reactor** | **PFR** | **BN-600** | **Phenix** | **Superphenix** |
|---|---|---|---|---|
| **Cumulative load** | 20.57% | 71.51% | 33.72% | 6.6% |

Examining the cumulative load factor of demonstration reactors, only BN-600 has a reasonable cumulative load factor. The reason for BN-600's high cumulative load factor is its ability to operate despite sodium fires. By engineering around issues such as this, with materials advances and improved component design, it should be possible to achieve consistent reliabilities for future SFRs. Considerations such as oxide dispersion strengthened steels; pumps; welding and leak detection, amongst others, are very important. A key area identified for UK research comes under the umbrella of power conversion systems. Advances in steam generators or alternative power conversion systems can reduce the probability and severity of sodium-coolant interactions, which cause reactor outages. Alternative power conversion systems such as helium or supercritical CO2 Brayton cycles could be employed and the UK is particularly well placed to use printed circuit heat exchangers, for example[7].

### *2.1.2. Fuel cycle barriers*

Metallic fuel experience is limited to EBR-II development as part of IFR. UK has experience with U-Mo fuel but this is unlike modern zirconium-TRU fuel design. Therefore, fabrication experience and reprocessing experience is very low.

Worldwide there has been considerable experience of fabricating thermal MOX. As of 2000, there had been: Belgium-BN/Desse 467 tHM; Germany-Siemens 158 tHM; France-CFCa 248 tHM; France-Melox 455 tHM, India-BARC 3 tHM, Japan-PFFF 120 tHM, UK-MDF 14 tHM, UK-SMP 5.5 tHM (2007). There has also been a lot of fast reactor MOX: Germany-Siemens 5.9 tHM; France-CFCa 110 tHM; Japan-PFFF 4 tHM; Japan-PFPF 10 tHM; UK-MDF/Sellafield 13 tHM; Russia-Paket 1.4 tHM and Russia-ERC 4.3 tHM[4][8][9][10]. Despite considerable experience there have still been issues with production. In the UK, the SMP never reached its name plate capacity and was shut down without fulfilling its contractual obligations. The large scale and complexity of such plants makes the failure of such large scale centralised facilities very expensive and impossible for private industry to finance alone, due to risk.

Despite considerable reprocessing experience there has only been a small quantity of MOX fuel being reprocessed. As of 2006, less than 171 tHM of MOX has been reprocessed in large scale purex plants: France-UP2/3 150 tHM; France-APM 2.8 tHM; and Japan-TRP 18 tHM. Fast reactors even less: France-UP2/3 100 tHM (diluted with thermal MOX); France-APM 10.7 tHM; France-AT1 1 tHM; Russia-RT1 450 tHM; Russia-RIAR 7 tHM and UK-UKAEA-RP 14 tHM[9][11][12]. As a result the assumption that these operations can be scaled up, may not be true and the building of plants which under perform or go over budget has the potential to make them economically unfavourable at first and have a knock on effect in terms of public acceptance.

The UK has experience of aqueously reprocessing metallic fuel (not zirconium based fuel). The B205 Magnox reprocessing plant operated successfully with a capacity of 1500 t/year of Magnox fuel and later suffered outages and required refurbishment before finishing its planned run. Reprocessing oxide fuel has been done in the Thorp plant with a capacity of 1200t/year. Unfortunately it has never been reliable, with changing throughput due to outages, major accidents and required modifications[13].

### *2.2. Partitioning and transmutation (P&T)*

P&T scenarios have been widely considered due to their impact on long-lived radiotoxicity of waste and how this reduces the burden on a geological repository. Thermal loading in a repository is significantly reduced so more material can be stored[14]. Due to inefficiencies, transmutation schemes cannot remove the need for TRU disposal, instead it reduces repository requirements[15]. In a study considering Yucca mountain, the capacity could be increased by a factor of 4.4-5.7 with all minor actinides removed (depending on separation efficiency)[16]. However, it is not possible to remove the need for a repository. As a result it is worth considering long-lived isotopes, which have higher thermal loadings, or alternatively, the more easily transmuted and fissioned TRUs.

## 3. Assessment criteria

Below are the criteria which were used to assess fuel cycles:

(1) Sustainability – Considers the impact on UK material over time, depending on the system doubling time of different fuel cycles. It also includes the requirements for imported fissile material;
(2) Technology readiness level – There is significantly more experience with some facilities, which have been demonstrated on a commercial scale. Some techniques, such as advanced reprocessing methods or metallic fuels, have only been demonstrated on a small or lab scale, which greatly affects deployment timescale and cost;
(3) Repository requirements – The size of a repository and the length of time that waste must be kept secure and out of the environment is significant. This is assessed in terms of the time it takes for waste toxicity to reduce to the levels of natural uranium and the heat generated by waste over time. It has repercussions in terms of cost and burden to future generations and the public acceptance of nuclear power;
(4) Proliferation resistance – Systems which are more diversion resistant with higher intrinsic barriers to production/diversion of material are preferable (through minimising onsite materials; time to produce significant quantities; time to detect diversion and detection methodologies). In the UK this is, to a certain extent, less of a concern being a weapons state. However, the development of a fuel cycle which is deployable everywhere is a key factor;
(5) Economic competitiveness – Certain fuel cycles will be more economically favourable than others. Whilst it is understood that a lot of the above criteria cannot all be met in one system, there is the potential for a more ideal system, based on the UK requirements and public/political opinion. However, this may not prove cost effective and the most technologically ready systems might be the only economically effective route.

## 4. UK Fuel-Cycle-Options (FCO) for consideration

The UK's stance on nuclear power and plutonium disposition has been very changeable. With this in mind, and from a review of literature, common criteria for fuel cycle options (FCOs) became apparent. As such, six logical options and their justification have been outlined for further investigation:

- (FCO-1) Sustainability – Meet the requirements of limited uranium resources and the potential for large growth in terms of nuclear capacity;
- (FCO-2) Sustainability with high proliferation resistance – As above but with high intrinsic barriers to proliferation. Large growth in sustainable nuclear power could include insecure, non-weapons states. As a result, any fuel cycle deployed in the UK should be deployable in other states without fear of separated plutonium being easily diverted;
- (FCO-3) Proliferation resistance and minimising fissile stock – FRs to minimise the proliferation risk of any separated, weapons usable material;

- (FCO-4) Waste minimising – Reduce the burden on geological repositories and minimise the lifetime of all SNF so the burden of nuclear waste on future generations is minimised;
- (FCO-5) Sustainability with minimum waste – A sustainable FR system that keeps the lifetime of waste produced to a realistic minimum;
- (FCO-6) Feasibility – Whilst multiple FR systems have inherent advantages, there are issues with under developed technology. Therefore, if FRs become advantageous over thermal reactors in the near term, the most deployable route will be favourable.

Multi-attribute decision analysis methodologies were used to determine fuel cycle scenarios that best meet the above options. This was done by applying weightings, based on personal consideration of literature and previous studies, in terms of: proliferation; sustainability; waste and feasibility. These weightings were applied to aspects such as: the initial fuel stream; fuel constituents (whether higher actinides are included), and how fuel cycles are setup (to be self sustaining; breeding or net consumers of fissile material). Different permutations of these were considered and the scorings gave the following results for each fuel cycle option (shown in figure 1):

- FCO-1 – Breeding with only plutonium and uranium fuel and both LWR and FBR fuel is reprocessed;
- FCO-2 – Stockpiled plutonium has been through LWRs as MOX once (reducing the stockpiled mass and storing it in an active matrices). FRs use this as an initial driver fuel and actinides are homogeneously included (and potentially some lanthanide for self shielding purposes) in the fuel and set up for breeding;
- FCO-3 – No reprocessing, only use stockpiled plutonium as fuel to achieve very high burnup before disposal;
- FCO-4 – Multiple recycle of all TRUs from LWR and FBR fuel, with all MAs in target assemblies;
- FCO-5 – Same set up as FCO-1, but with americium homogeneously mixed in with the fuel in relatively small quantities to reduce one of the major components of long-lived waste;
- FCO-6 – The same as FCO-1 but with the potential (given a long deployment timescale) that some plutonium has been recycled in LWR MOX.

There is also the option to mix some of the sustainability and waste minimising scenarios in an equilibrium system to meet certain requirements. However, despite these being outlined as sensible routes to study, time constraints may require limiting these to a few of the more relevant fuel cycles for comparison. Similarly, some scenarios can be considered more relevant due to public opinion. There is also the potential for further analysis to remove some FCOs from the study for not being realistic. However, they are included here for completeness. The multi-attribute analysis was based on weightings given by the user and, despite material in previous studies supporting this assessment, it is important to realise that others may weight aspects differently.

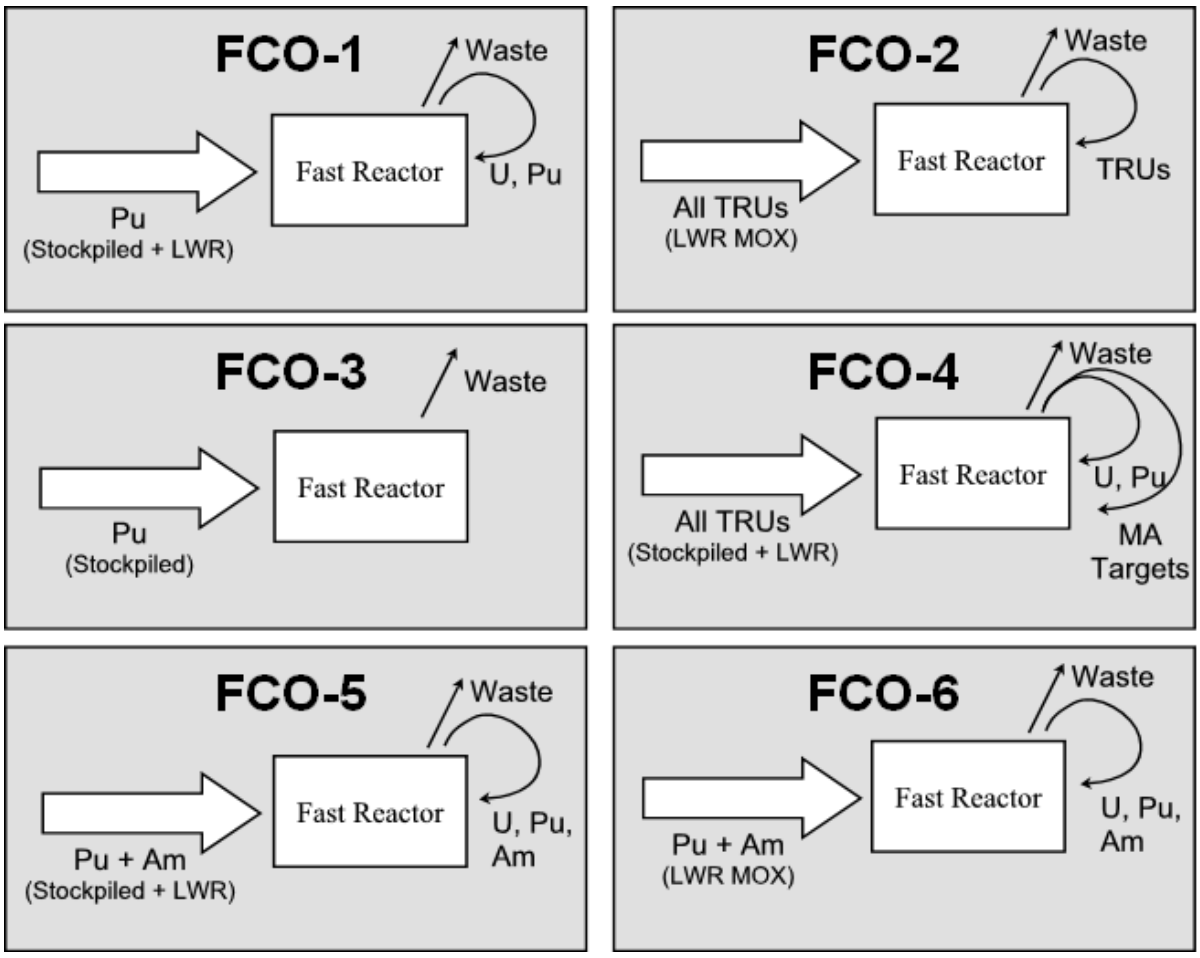


*FIG. 1. Fuel cycle options showing basic material flows.*

### *4.1. Deployment*

Studies have shown that, in a high nuclear build scenario, world nuclear capacity would level off before increasing again if FRs are deployed too late. This is due to the reduction in uranium resources and the time required for breeding to build up a stockpile of fuel[17][18]. As such, consideration must be paid to whether build rates and stockpiled resources are appropriate. The scale of deployment will look at current considerations for new build as a minimum (16 GWe) and enough to begin decarbonisation of the UK as an upper limit (75 GWe).

### *4.2. Material stockpile*

Reprocessing of spent fuel in the UK was originally intended for the weapons program. Over time, with the expected expansion of nuclear power worldwide and limited uranium resources, reprocessing was continued to provide fuel for a FR program[13]. However, the rate of nuclear expansion tailed off and the known, economically recoverable uranium resources gradually increased. This postponed the need for expensive FR programs but left the UK with the world's largest civil plutonium stockpile. Uranium fuels are still far cheaper to use than plutonium fuels and it is unlikely that the price of uranium will rise to a level where plutonium fuels would be competitive for decades[19]. However, the UK is left with 90.3 tons of civil plutonium (and 32 tons in spent fuel, giving a total of 122.3 tons as of 2011, and over 60 000 tons of depleted uranium and enrichment tails), which has no disposal route and is a proliferation concern[3].

A large FR such as the EFR, fuelled with TRUs from thermal reactors will use roughly 5 tons of plutonium per GWe of generating capacity. This means stockpiled material could roughly fuel 24 GWe of FRs as an upper limit. However, UK electricity demand peaks at over 50 GWe and, in the

future, this would be higher if the UK moves away from fossil fuels for transport and heat applications. As such, if FRs are used as a sustainable base-load electricity source, there is a requirement for much more plutonium. Studies looking at variations of burnup, breeding ratio and doubling time show that, depending on the design, compound system doubling time can take between 16.5-43.5 years (this may be reduced with pyro-processing and shorter cooling times)[14]. To meet the current peak electricity demand with FRs, twice as much plutonium will be needed to seed reactors. This has a significant impact on the time to deploy a large FR program due to limited fissile stocks.

It is also unlikely that radial breeder blankets will be used due to proliferation concerns and as a result will make doubling times much longer. Therefore the scale of FRs deployment could be limited by the amount of fissile material stockpiled. This will be a more significant limiting effect on other countries, which do not have similar stockpiles. An alternative could be seeding reactors with enriched uranium, however, if FR deployment comes about due to expensive and dwindling uranium resources, this may not be feasible.

If one assumes an optimistic doubling time of 20 years (reference of heterogeneous reactor with breeding gain of 0.39) and a build rate of 1 GWe per year it would still take over 40 years to double the current plutonium inventory. A more realistic scenario, without radial blankets and breeding ratios slightly greater than one, could lead to a doubling time of up to 100 years, even with reprocessing of all new build LWR fuel. As a result, the UK would be dependent on LWRs and uranium in the interim. Therefore, deployment of FRs to meet future demand would have to begin decades before economically recoverable uranium resources begin to run low.

## 5. Methodology

Established codes will be used to obtain fuel cycle and reactor information, which will then contribute to the fuel cycle options analysis. The neutronics code WIMS10 with the ECCO cell lattice code will be used for FRs, whilst fuel cycle scenarios will use National Nuclear Laboratories fuel cycle analysis code ORION.

The approach to analysing these scenarios is as follows: an initial fuel cycle model will be run to its equilibrium point, which will then aid the selection of appropriate scenarios to investigate further, performing more detailed criticality designs in WIMS with more exact geometry. This will be used to check realistic doubling times, with and without radial breeders, and to obtain time dependent depletion calculations for further use in ORION. These depletion calculations will be used to more accurately model the fuel cycle, determining waste buildup and fuel utilisation for each FCO. Finally, FCO results will be analysed in terms of feasibility, repository requirements, sustainability, proliferation concern, and estimated cost. This body of work is set to begin immediately, and support with fuel cycle and neutronics modelling has already been arranged with AMEC and the National Nuclear Laboratory.

## 6. Summary

It is important to understand that this project is in its infancy. As a brief description of the approach and main considerations of the study, it should give an overview of the issues being examined and the criteria by which they will be assessed in further work. The potential areas that could be covered by this study are very large, so restricting the number of options to be looked at in detail will be important. Although all fuel cycle options highlighted would be interesting for comparison, they may not all be taken to the same level of detail.

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