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# PHENIX: Interpretation of ECRIX-H transmutation experiment with TRIPOLI4D®

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**Abstract** - *The ECRIX experiments aimed at demonstrating the transmutation capacity of sodium fast reactors regarding minor actinides (Americium, Neptunium, and Curium). These artificial atoms, produced in reactors, are mainly responsible for radiotoxicity and heat production of nuclear waste and thus concur to the complexity of storage facilities. Transmutation aims at reducing the amount of these products by irradiating them with neutron beams in dedicated facilities. The ECRIX-H experiment, performed from 2003 to 2006 in the French sodium cooled fast reactor Phénix, was focused on Americium transmutation in a softened neutron spectrum. The experimental pin was filled with a mixed material composed of americium oxide and magnesia. It was placed inside a calcium hydride container whose goal was to soften the neutron spectrum in order to enhance transmutation. Local neutron slowing-down due to hydrogen scattering reactions is a complex phenomenon which represents a challenge in terms of computer simulation. Previous interpretation campaigns highlighted limitations of standard fast reactor calculation schemes regarding both flux and depletion calculations with significant deviations (up to 50%). This paper will discuss of a new interpretation based on up-to-date calculation schemes using the stochastic transport calculation code TRIPOLI4D® shipping a new depletion calculation module.*

## I. INTRODUCTION

Phénix is a medium power sodium-cooled fast reactor (SFR) built in the early seventies in France and stopped in 2010 after nearly 35 years of operation [Ref. 4]. During its operating period a high number of irradiation experiments were conducted in order to gather knowledge on various fields such as material behavior under irradiation, minor actinides transmutation, production of radioactive isotopes for medical applications. Among these experiments, The ECRIX experiment, performed from 2003 to 2006, aimed at demonstrating americium transmutation capacity of fast reactors. In order to enhance the transmutation process inside experimental pins, the flux spectrum was softened using moderator materials such as enriched boron carbide ($^{11}B_4C$, ECRIX-B) or calcium hydride ($CaH_2$, ECRIX-H). This paper is focused on the interpretation of ECRIX-H experiments with the burnup simulation code TRIPOLI4-D based on a stochastic flux solver and a deterministic nuclide depletion solver [Ref. 1, 2].

## II. TRANSMUTATION OF AMERICIUM

Minor actinides (Americium, Neptunium and Curium) are artificial atoms produced in nuclear reactors as a result of unwanted radioactive capture reactions in fuel materials. They are mainly responsible for radiotoxicity and heat production of nuclear waste. Transmutation phenomena can be defined as a process in which an atom is transformed in another as a result of a nuclear interaction (radioactive decay, interaction with neutron …). Natural decay process of minor actinides is very slow (see half life in Table I). We can not count on it to significantly reduce amounts of these isotopes in nuclear waste.

Table I : Half-life of main minor actinides

| Isotopes | Half-life (years) |
|---|---|
| $^{241}Am$ | 432 |
| $^{243}Am$ | 7 370 |
| $^{244}Cm$ | 18 |
| $^{245}Cm$ | 8 500 |
| $^{237}Np$ | 2.1E6 |

Active transmutation using a high neutron flux is therefore a necessity. Incoming neutrons can induce fission or capture reaction on actinides. Fission reactions are able to transform actinides in less radiotoxic elements; on the contrary capture reactions create upper actinides. To enhance transmutation, fission reactions should therefore be favored. Table II shows ratios between fission and capture cross sections for $^{241}Am$ in different neutron spectra and flux levels. Reactions rates depends both of neutron flux and cross section levels. Compared to a thermal reactor, the flux level is high in a fast reactor (factor ~ 10) and the ratio between fission and capture is the best but cross section levels are very low (factor ~ 0.01). The main characteristic of ECRIX experiments are the presence of moderator materials around transmutation targets in order to enhance cross section levels. The best characteristics are obtained for ECRIX: high flux, high cross sections.

Table II : Comparisons of $Am^{241}$ cross sections (1 Group) and flux amplitude for PWR, Phenix and ECRIX

| | Thermal Spectrum (PWR) | Fast Spectrum (Phenix) | ECRIX Spectrum |
|---|---|---|---|
| Fission (F) | 3,9 barns | 0,1 barns | 4,1 barns |
| Capture (C) | 634,9 barns | 10,3 barns | 595,2 barns |
| Ratio F/C | **0,006** | **0,008** | **0,007** |
| Flux (n/cm²/s) | ~1E14 | ~ 4E15 | ~ 1.5E15 |

## III. PHENIX REACTOR DESCRIPTION

Phenix is a sodium cooled fast reactor built in France by a joint venture between CEA and EDF in 1974 [Ref. 4]. It is a pool-type (integral) reactor, with an initial power of 563 MWth (250MWe). Since 1993, it was operated at a lower power rating (350 MWth). The figure 1 gives a quick overview of the reactor design. The core consists in a lattice of hexagonal sub-assemblies containing 217 fuel pins filled with a mix of uranium and plutonium oxide (MOX). Each fuel S/A is composed of three axial zones: the upper and lower fertile axial blanket, and the fissile zone. The driver core is divided into a low plutonium enriched inner zone and a high enriched outer zone, surrounded first by an annular fertile zone and then by steel reflectors and shielding rods (see figure 2). Reactor control is ensured by the mean of 6 control rods containing boron carbide rods enriched in $^{10}B$. The core is located in the main vessel which holds around 800 tons of primary sodium.

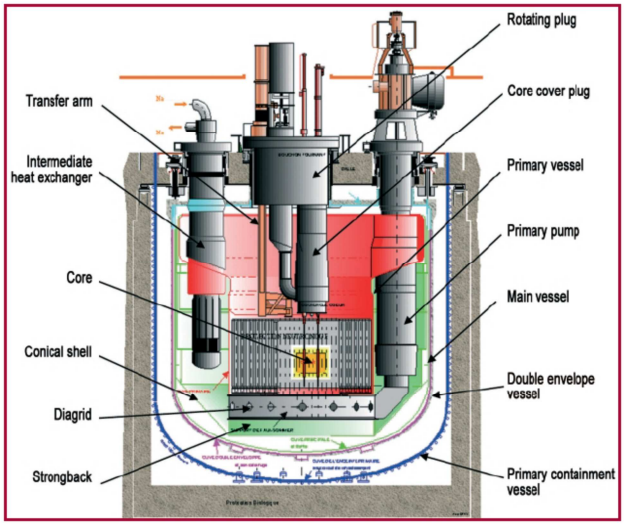


Figure 1 : Schematic of the reactor

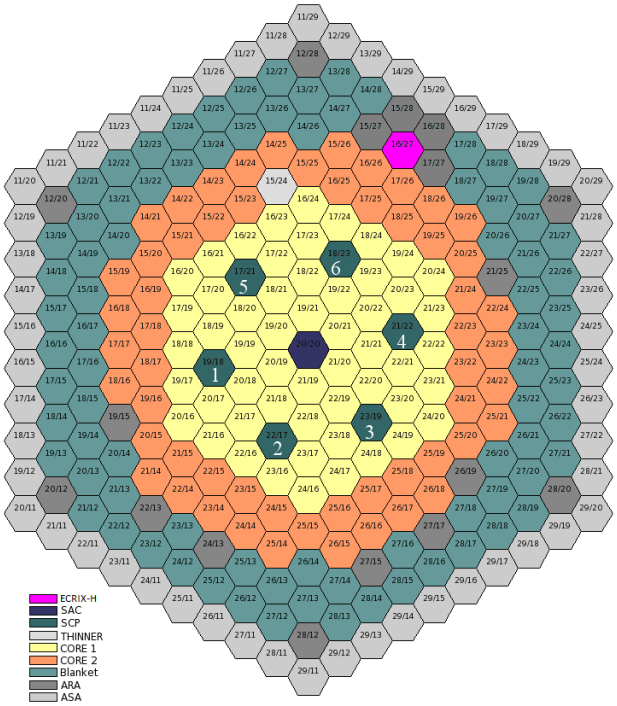


Figure 2 : Phenix Core description

## IV. DESCRIPTION OF THE ECRIX-H EXPERIMENT

The ECRIX-H experiment was performed from 2003 to 2006. The experimental assembly was located in the seventh core ring (see figure 2). It was surrounded by two fissile assemblies belonging to the external core and 4 reflector assemblies whose goal was to focus neutron flux in experimental zone. This assembly was made of two distinct parts (see figure 3): a fix part containing the calcium hydride tank responsible for the neutron slowing-down and a moving capsule containing the test pin. Experimental pellets were made of a mixed material containing americium oxide ($AmO_2$) and magnesia (MgO). The Table III presents the main characteristics of the ECRIX-H materials.

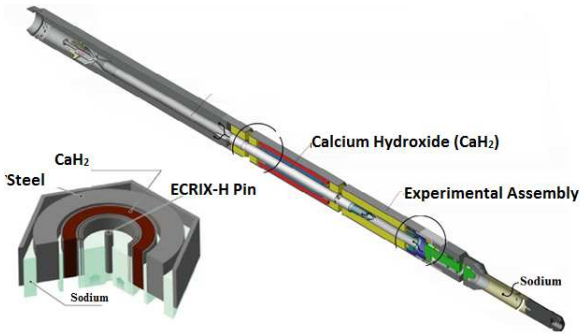


Figure 3 : ECRIX-H description: Experimental Assembly + Test pin

Table III : Main characteristics of ECRIX materials

| | |
|---|---|
| $CaH_2$ : ratio H/Ca | ~1.9 |
| $CaH_2$ : density | 1.6 g/cm$^3$ |
| ECRIX Pin : $AmO_2$ mass fraction | 16,6% |
| ECRIX Pin : ratio 243Am/Am | ~5 % |

The figure 4 presents the simplified irradiation history of ECRIX-H assembly which had seen successive irradiation (551 days) and decay periods (397 days).

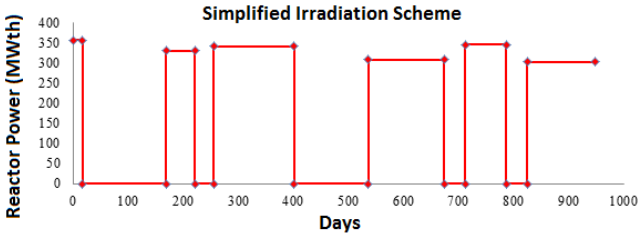


Figure 4 : Simplified irradiation scheme of ECRIX-H test pins

In order to characterize final irradiation results, we can introduce two physical quantities, known as:

- The global transmutation rate which measures the amount of americium lost by fission and capture reactions (Eq. 1):

$$TR = 1 - \frac{Mass_{Am}^{final}}{Mass_{Am}^{initial}} \quad (1)$$

- The global fission rate which measures the amount of americium and its descendants (Pu, Cm, U, Np) lost by fission reactions (Eq. 2):

$$TF = 1 - \frac{\sum_{i\in[Am,Cm,Np,Pu,U]} Mass_i^{final}}{Mass_{Am}^{initial}} \quad (2)$$

This latter ratio takes in account not only americium lost by direct fission but also losses of plutonium and curium isotopes created by unwanted capture reactions on americium

## V. PHENIX MODELING

### 1. Description of TRIPOLI4-D

TRIPOLI-4D [Ref. 1, 2] is a burnup code developed at CEA. It relies on the TRIPOLI-4 as neutron transport code and MENDEL as nuclide depletion solver. TRIPOLI-4 is a radiation transport code based on a stochastic approach to solve the Boltzmann equation [Ref. 3]. It simulates the propagation and the interactions with matter of both neutron and photon in general three-dimensional geometries. Pointwise interaction cross sections along with probability tables for the unresolved range are directly used in the simulation without the multigroup approximation. Raw cross-sections can possibly be corrected of temperature effects such as the Doppler resonance broadening effect [Ref. 5]. It is considered as a reference neutronics code compared to deterministic code. MENDEL is a nuclide depletion solver used for burnup calculations [Ref. 2]. It solves the Bateman equation using multigroup flux computed by an external flux solver (TRIPOLI-4 here) and nuclear data such as filiation chain and nuclear decay data. TRIPOLI4-D acts as a manager responsible for the successive communications between flux calculation steps using fixed isotopic concentrations and depletion calculation steps using fixed fluxes (see figure 5).

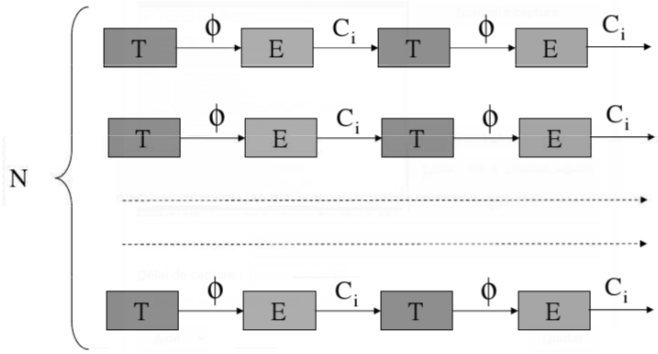


Figure 5 : Parallelization scheme used in T4D calculations

In order to compute statistical uncertainties on evolving parameters, the code uses the method of independent replicas coupled with the Central Limit Theorem. The figure 5 presents this method: the transport/depletion loop is done many times on different simulators with different seeds in order to obtain distribution of results. If the number of independent simulation is sufficient the Central Limit Theorem ensures that these distributions can be well represented by Gaussian distribution. Average values along with standard deviations can then be easily estimated.

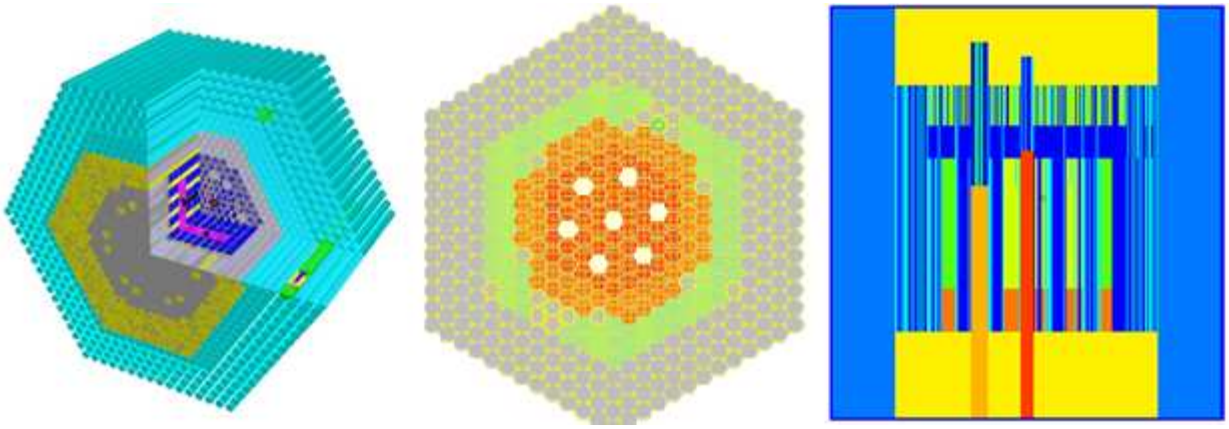

Figure 6 : ROOT model of the core

Note that systematic uncertainties induced by the following causes are not addressed by the former technique:

- Fluxes and reaction rates are considered constant during a time step.
- Isotope densities are considered uniform in each depletion region.
- Non linearity of Bateman equation with respect to the flux.

The first two causes are not specific to Monte-Carlo burn up simulations. We can then use standard solutions implemented in evolution schemes with deterministic transport tools. Fluxes stationary during a time step can be addressed by using a fine temporal scheme either by increasing the number of temporal points or by using predictor/corrector models. Following schemes are implemented in TRIPOLI4-D:

- EULER scheme (order 1 – linear – Eq. 3) :

$$\begin{aligned} N_{t+1} &= MENDEL(N_t, \phi_t, \delta t) \\ \phi_t &= TRIPOLI(N_t) \end{aligned} \quad (3)$$

- MEAN scheme (order 2 – quadratic – Eq. 4) :

$$\begin{aligned} N^1_{t+1} &= MENDEL(N_t, \Phi_t, \delta t) \\ \phi_t &= TRIPOLI(N_t) \\ N^2_{t+1} &= MENDEL(N_t, \phi_{t+1}, \delta t) \\ \phi_{t+1} &= TRIPOLI(N^1_{t+1}) \\ N_{t+1} &= 0.5 * (N^1_{t+1} + N^2_{t+1}) \end{aligned} \quad (4)$$

The problem of uniform isotopic densities can be addressed by using a fine spatial mesh of evolving media in order to take in account skin effects.

The problem of non linearity is more specific to Monte-Carlo burn up simulations. This systematic error is linked to the number of neutrons involved in each simulation. To reduce it, we have to increase the number of simulated neutrons.

Cures for all these uncertainties have a negative impact on calculation times and memory consumptions. An efficient equilibrium state should then be found.

## 2. Description of INCA

In order to simplify the workflow of neutron fast reactor conception and studies, the CEA is currently developing a design software named INCA, whose goal is to offer user-friendly tools to technical design teams. INCA, as shown on figure 7, is on top of neutronic calculation software such as TRIPOLI4-D or APOLLO3® [Ref. 8]. Tasks such as geometrical design, calculation scheme definition or advanced post-processing were rethought using user based data model. Numeric engines are automatically driven by a service based mechanism.

INCA has the ability to convert its own geometry model into ROOT geometry model [Ref. 7]. ROOT is an object-oriented library developed in C++ by CERN and integrated in TRIPOLI-4. It was originally developed for particle physics data analysis and provides functionalities to manage complex tri-dimensional geometry and handle materials associations.

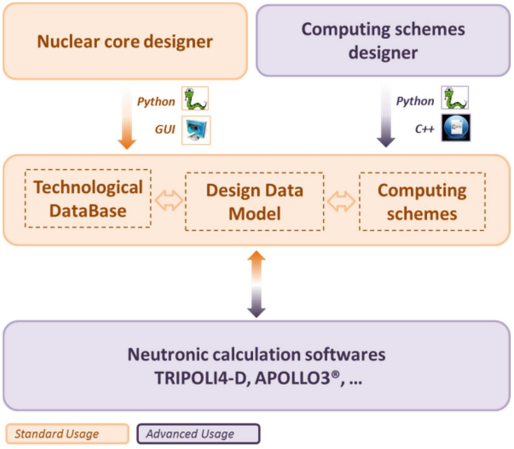


Figure 7 : Description of INCA

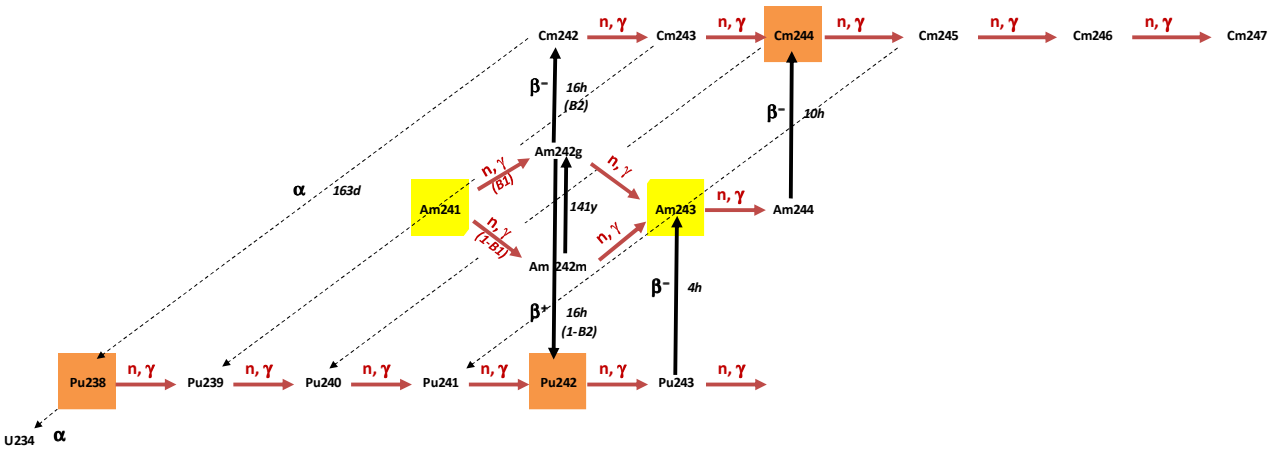


Figure 8 : Depletion chain for Americium

## 3. Core geometry and compositions

The tridimensional reactor core was entirely modeled within INCA. Assembly models were close to the reality. Pellets, clad, wrapper tube were described for all kind of sub-assemblies (fuel, blanket, control rods, shielding and ECRIX-H – see figure 9). Reactor model was done at 20°C (dimensions, material compositions, cross sections) except for sodium whose density was fixed according to sodium average temperature during normal operation state (~450°C). These simplifications do not have a significant impact on americium depletion results (<2%).

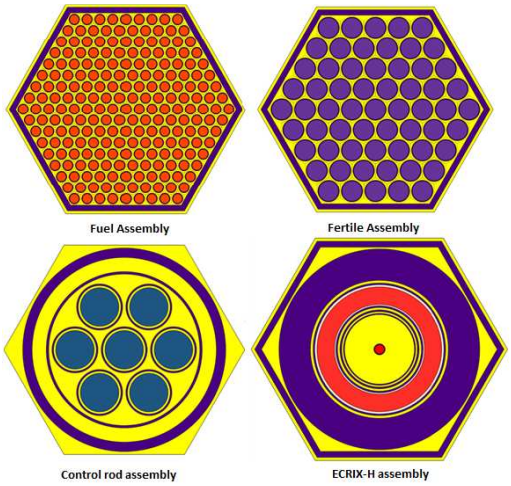


Figure 9 : Geometry models of main assemblies

Phenix core management tools included in the ERANOS-2.2 code [Ref. 6] were used to produce averaged compositions of inner, outer and fertile fuel assemblies in order to dispose of a representative core in terms of burn up. Note that these tools simulate real fission products using a reduced number of pseudo fission products with modified cross sections to ensure equivalent absorption rates. As such pseudo fission products are not available in TRIPOLI4, they were replaced by Molybdenum atoms while ensuring an equivalent absorption rates.

Finally as TRIPOLI4-D depletion calculations are memory and time consuming, we have limited burnup calculations to americium pellets only. These pellets were divided in 5 axial zones and 6 radial zones in order to take in account the axial and radial flux shape. The core was not depleted and was used as a neutron source whose level was fixed by the irradiation scheme.

## 4. Nuclear Data libraries and depletion chain

European nuclear data libraries JEFF-3.1.1 was used for the study. These libraries include not only pointwise interaction cross sections and probability tables for all isotopes but also energy released data and decay data (half-life, yields).
TRIPOLI4-D needs the definition of a depletion chain to perform burn up calculations. The americium depletion chain used for this study is presented in figure 8. B1 and B2 correspond to the branching ratios of respectively the capture reaction on $Am^{241} \rightarrow Am^{242}{}_g$ and the decay of $Am^{242}{}_g \rightarrow Cm^{242}$. B1 depends of the incident neutron energy and thus of neutron spectrum. It was calculated using a ponderation algorithm based on a fine energy mesh (281 groups) and on the computed capture rate of $Am^{241}$ in ECRIX assembly (Eq. 5). B2 is equal to 84%.

$$\overline{B_1} = \frac{\sum_{g\in[1,281]} B_1{}^g(E) * \tau^{Am241}_{capture\,g}(E)}{\sum_{g\in[1,281]} \tau^{Am241}_{capture\,g}(E)} \qquad (5)$$

Table IV : Branching ratio ($Am^{241} \rightarrow Am^{242}{}_g$)

| B1 $Am^{241} \rightarrow Am^{242}{}_g$ | Fast spectrum (Core centre) | ECRIX spectrum |
|---|---|---|
| JEFF-3.1.1 | 83,7% | 87% |

## VI. RESULTS

### 1. ERANOS-2.2 results

The ERANOS code, developed by CEA in the framework of a European collaboration, is a deterministic neutronics calculation system adapted to the physics of fast reactors [Ref. 6]. It benefits of an important validation database composed of various experiments realized on MASURCA, Phenix and SUPERPHENIX fast reactors. It includes various core solvers (diffusion, transport solvers relying on SN or PN methods), cell solvers (ECCO, HETAIRE), core management and post-processing tools. This tool can be plugged with various nuclear libraries such as JEF-2.2, ERALIB-1 and JEFF311. This tool was intensively used to perform former analysis of ECRIX experiments.

However this latter was not able to describe a realistic three dimensional version of the ECRIX-H assembly due to solver limitations. Such description is nevertheless mandatory to take in account strong local moderation in the hydride container. The following procedure based on a three stage scheme coupled with JEFF3.1 nuclear library was then used to correctly estimate flux in the target pins:

- Cell calculations were done with the cell solver ECCO in order to produce homogenized and self-shielded cross sections in a 33 group energy mesh. Self shielding and slowing down on heavy isotopes were dealt using a fine energetic mesh (1968 groups). This stage was done at the initial irradiation time.

- Core calculations were done with the transport solver VARIANT on a three dimensional geometry in order to compute the flux intensity in the ECRIX-H assembly. All assemblies including ECRIX-H were described as homogeneous media using cross-sections coming from ECCO. This stage was done after each depletion step.

- A fine flux calculation with the transport solver BISTRO on a realistic two dimensional geometry of ECRIX-H. This flux was then normalized using the former 3D flux and used for depletion calculations. This stage was done after each depletion step.

The used irradiation history is presented in figure 4. ERANOS2-2 results are presented in the Table V. Flux normalizations were done by imposing a total reactor power according to the irradiation history. Used branching ratios are presented in Table IV. All calculations were done on a cold model of the reactor (dimension + material compositions at 20°C) for the sake of comparisons with TRIPOLI4-D.

Table V : Comparison between experimental and calculated fission and transmutation rates with ERANOS-2.2/JEFF3.1

| | Fission rate | Transmutation rate |
|---|---|---|
| *C/E* | ***1,51 ± 0,29*** | ***0,99 ± 0,01*** |

### 2. TRIPOLI4D results

#### A. Flux normalization

Flux normalisation was performed using the nominal reactor power (350 MW). TRIPOLI4 is able to compute the total energy deposited in the reactor for a given initial neutron source. Deposited energy values by reaction come from nuclear data libraries. The following assumption was made: the energy is locally deposed in order to avoid computation of the propagation of emitted gamma.

Real flux in the reactor can then be estimated with the following formula (Eq. 6) where $C_{MeV\rightarrow J}$ is the conversion coefficient from MeV to Joules.

$$\Phi_{reactor} = \phi_{calculated} * \frac{P_{reactor}}{E_{calculated} * C_{MeV \rightarrow J}} \quad (6)$$

The normalisation coefficient was computed using simulations involving 800 millions of neutron histories in order to have a good statistical convergence on multiplication factor ($\sigma_{keff}$ ~ 3 pcm). Table presents $k_{eff}$ and flux calculated values with their uncertainties. Mean neutron energy in ECRIX target is reduced by a factor two regarding the situation in the core center which illustrates the calcium hydride efficiency in term of neutron slowing-down. Note that flux intensity in ECRIX target is reduced by a factor 3.

Table VI : ECRIX-H Calculated parameters at initial step

| Parameters (JEFF-3.1.1) | | Value | Sigma (%) |
|---|---|---|---|
| $K_{eff}$ | | 1,04673 | 0,003% |
| ECRIX | **Flux (n/cm²/s)** | 1,44E+15 | 0,76% |
| | **$E_{moy}$ (MeV)** | 0,25 | 0,73% |
| Core Center | **Flux (n/cm²/s)** | 4,32E+15 | 0,29% |
| | **$E_{moy}$ (MeV)** | 0,5 | 0,27% |

#### B. Depletion results

Each depletion calculation involved 32 independent burn up simulations. Each burn up steps involved the processing of 64 millions of neutron histories to compute the flux in ECRIX pellets. Final uncertainties were estimated by TRIPOLI4D using the central limit theorem. EULER and MEAN temporal schemes were used. As their results were very close, we give only MEAN results in the following sections.

Table VII and figure 10 presents the comparisons between calculated and measured fission and transmutation rates.

Previous interpretation campaigns highlighted limitations of standard fast reactor calculation schemes, based on determinist methods, regarding both flux and depletion calculations with significant deviations (up to 50%). These deviations are now reduced but still noteworthy with a 30% relative overestimation of the fission rate.

Table VII : Comparison between experimental and calculated global fission and transmutation rates

| | Fission rate | Transmutation rate |
|---|---|---|
| *ERANOS2.2 - C/E* (2σ) | ***1,51 ± 0,29*** | ***0,99 ± 0,01*** |
| *TRIPOLI4D - C/E* (2σ) | ***1,34 ± 0,26*** | ***0,99 ± 0,01*** |

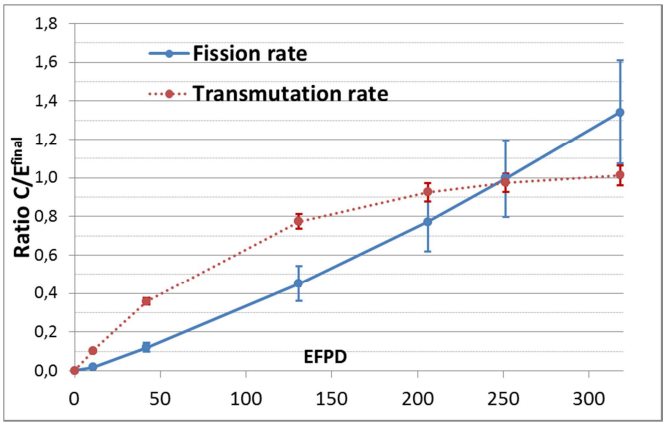


Figure 10 : Evolution of C/E ratio of global rates during irradiation

Table VIII presents the comparisons between calculated and measured masses of minor actinides still present after cooling time. Calculated heavy nuclides mass is systematically lower than measured one which is coherent with an overestimation of the fission rate by calculations. TRIPOLI4 results are slightly better than ERANOS ones in particular uranium and plutonium elements. However except for curium element whose measured uncertainty is very low, discrepancies on element masses are of the same order of magnitude as measured uncertainties (Pu, Am).

Finally even with an up-to-date calculation scheme taking into account a realistic three dimensional description of the core and based on a Monte-Carlo reference simulation tool for flux estimation, discrepancies between calculated results and measured ones are still noteworthy. Following section will discussed of potential explanations of such discrepancies.

Table VIII : Final mass - Ratio C/E

| **Atom (Experimental. uncertainty 2σ)** | **ERANOS22 JEFF-3.1 (2σ)** | **TRIPOLI4D JEFF-3.1.1 (2σ)** |
|---|---|---|
| **Uranium (30%)** | 0,59 ± 0,18 | 0,72 ± 0,22 |
| **$PU^{238}$ (8,5%)** | 0,72 ± 0,06 | 0,78 ± 0,07 |
| **$PU^{239}$ (8,5%)** | 1,38 ± 0,12 | 1,22 ± 0,11 |
| **$PU^{240}$ (8,5%)** | 0,86 ± 0,07 | 1,68 ± 0,15 |
| **$PU^{241}$ (8,5%)** | 1,22 ± 0,10 | 0,89 ± 0,08 |
| **$PU^{242}$ (8,5%)** | 0,63 ± 0,05 | 1,29 ± 0,11 |
| **$AM^{241}$ (18.6%)** | 1,27 ± 0,24 | 0,98 ± 0,19 |
| **$AM^{242}{}_{m}$ (18.6%)** | 0,82 ± 0,15 | 0,66 ± 0,13 |
| **$AM^{243}$ (18.6%)** | 1,01 ± 0,19 | 0,80 ± 0,15 |
| **$CM^{242}$ (6.5%)** | 1,25 ± 0,08 | 0,56 ± 0,04 |
| **$CM^{243}$ (6.5%)** | 0,53 ± 0,03 | 0,49 ± 0,04 |
| **$CM^{244}$ (6.5%)** | 0,86 ± 0,06 | 0,82 ± 0,06 |
| **$CM^{245}$ (6.5%)** | 0,80 ± 0,05 | 0,40 ± 0,04 |
| **$CM^{246}$ (6.5%)** | 1,62 ± 0,11 | 0,81 ± 0,07 |
| **$CM^{247}$ (6.5%)** | 2,14 ± 0,14 | 0,96 ± 0,11 |
| **U** | 0,59 ± 0,18 | 0,72 ± 0,22 |
| **Pu** | 0,79 ± 0,07 | 0,92 ± 0,08 |
| **Am** | 1,10 ± 0,20 | 0,86 ± 0,16 |
| **Cm** | 0,84 ± 0,05 | 0,72 ± 0,05 |
| **NL** | 0,82 ± 0,06 | 0,88 ± 0,06 |

### *C. Sensitivities to input data*

Table IX summarizes the main sensitivities of final results to model parameters such as:

- The quantity of hydrogen atoms in the moderator which have a huge influence of flux spectrum in targets.
- The flux intensity in the target.
- The branching ratio of the reaction $Am^{241} \rightarrow Am^{242}{}_{g}$.

The code ERANOS-2.2 was used in order to compute these results along with the JEFF-3.1 nuclear library. These results were obtained by modifying each input parameter by 1% which induces a modification of the final rate of s % where s is the sensitivity (See equation 7).

$$S(Rate_i) = \frac{\delta Rate_i / Rate_i}{\delta P / P} \qquad (7)$$

Table IX : Sensitivities of rate results to the main parameters of ECRIX-H experiment

| | $S_{[CaHx\ density]}$ | $S_{Flux}$ | $S_{B(Am241 \rightarrow Am242g)}$ |
|---|---|---|---|
| **$U^{234}$** | -0.75 | -0.63 | 0.97 |
| **$U^{235}$** | 0.36 | 0.45 | 1.05 |
| **$U^{236}$** | 1.70 | 1.63 | 1.05 |
| **$PU^{238}$** | -0.88 | -0.70 | 0.96 |
| **$PU^{239}$** | -0.41 | -0.29 | 1.00 |
| **$PU^{240}$** | 0.84 | 0.35 | 0.43 |
| **$PU^{241}$** | 1.07 | 0.73 | 0.92 |
| **$PU^{242}$** | -0.51 | -0.73 | 1.17 |
| **$AM^{241}$** | -5.41 | -3.79 | -0.06 |
| **$AM^{242}{}_{m}$** | -6.97 | -4.41 | -5.89 |
| **$AM^{243}$** | --0.49 | -0.59 | -0.18 |
| **$CM^{242}$** | -1.71 | -1.32 | 0.77 |
| **$CM^{243}$** | -1.49 | -0.74 | 0.95 |
| **$CM^{244}$** | 0.63 | 0.46 | -0.43 |
| **$CM^{245}$** | 0.01 | 0.72 | -0.41 |
| **$CM^{246}$** | 2.03 | 2.04 | -0.40 |
| **$CM^{247}$** | 2.34 | 2.86 | -0.34 |
| **NL** | -0.68 | -0.55 | 0.69 |
| **TF** | **1.28** | **1.05** | **-1.31** |
| **TT** | **0.15** | **0.11** | **0.01** |

The global fission rate is very sensitive to the quantity of hydrogen in the moderator, to the flux level in pellets and to the branching ratio Am241/Am242g. Final actinide masses are sensitives to these parameters but also to the cross sections level given in the nuclear library.

The use of a realistic three dimensional description of the core and of a Monte-Carlo reference simulation tool represents today the best way to estimate the flux in ECRIX pellets. However the method of independent replicas used in TRIPOLI4-D is so memory and time consuming that it limits the potential number of simulated neutrons by depletion step. This parameter has an impact on flux estimation in pellets and then on the final fission rates (see Table X). The scheme involving Euler scheme with 2 millions of simulated neutrons is converged with respect to the reference Mean scheme.

Table X : Impact of the number of simulated neutrons by depletion step

| Temporal scheme | Neutrons by step | Fission rate | Transmutation Rate |
|---|---|---|---|
| **Mean (Ref)** | 2 000 000 | 1,00 | 1,00 |
| **Euler** | 4 000 000 | 1,03 | 1,00 |
| **Euler** | 2 000 000 | 1,00 | 1,00 |
| **Euler** | 500 000 | 0,87 | 0,97 |

The hydrogen quantity in the moderator is a technological parameter. To improve knowledge on this parameter new measurements have to be performed.

Branching ratios and cross sections are nuclear parameters. To improve knowledge on these parameters new studies have to be performed which are not in the scope of this manuscript.

## VII. CONCLUSION

The ECRIX experiments aimed at demonstrating transmutation capacity of sodium fast reactors regarding minor actinides (Americium, Neptunium, and Curium). They were performed from 2003 to 2006 in the French sodium cooled fast reactor Phénix, and were focused on Americium transmutation in a softened neutron spectrum in order to boost reaction rates. Local neutron slowing-down due to hydrogen scattering reactions is a complex phenomenon which represents a challenge in terms of computer simulation. Previous interpretation campaigns highlighted limitations of standard fast reactor calculation schemes regarding both flux and depletion calculations with significant relative deviations (up to 50%).

New interpretations based on up-to-date calculation schemes, using the stochastic transport and burnup code TRIPOLI4D®, show a slight improvement on global fission rate estimations. Discrepancies between calculated results and measured ones are still noteworthy with a relative error of nearly 30% on the global fission rate in ECRIX pellets.

These errors can be linked to integral parameters such as the quantity of hydrogen in the moderator, the flux level in pellets and nuclear data like the branching ratio of the reaction Am241/Am242g. The use of a realistic three dimensional description of the core and of a Monte-Carlo reference simulation tool represents today the best way to estimate the flux in ECRIX pellets provided that the number of simulated neutrons by depletion step is sufficient. Finally to improve knowledge on the hydrogen quantity in the moderator and the nuclear data involved in this experiment, new studies or experiments have to be conducted.

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