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# ON THE BENEFIT OF FAST-NEUTRON REACTOR FUEL DEPLETION FOR TRANSPORTATION

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**ABSTRACT**

*In recent years, some efforts have been devoted in France to research projects for handling Phenix Fast-neutron Reactor (FR): wet storage, reprocessing process and transportation. This paper assesses the benefit of taking account of the depletion of FR fuel elements for transportation.*
*After a brief presentation of the calculation tools and models for depletion and criticality calculations, the depletion and criticality options are discussed. Then, interesting results show that the use of a low burnup level may allow an optimised loading of FR fissile fuel assemblies in the TN® 17/2 transport cask. Furthermore, the results for FR fissile assemblies also encompass criticality calculations for FR fertile assemblies. Indeed, due to TN®17/2 transport cask design, substantial margins are available for FR fertile assemblies, even for strongly conservative hypotheses on fuel inventory. Consequently, considering their irradiation is not of interest for this case, but could be for other transportation configurations. Finally, the paper focuses on depletion codes validation, a key element for a straightforward and effective implementation of this approach.*

**KEY WORDS**

*Criticality-Safety, Used Nuclear Fuel, Transportation*

## 1. INTRODUCTION

In recent years, some efforts have been devoted in France to research projects for handling Phenix Sodium Fast-neutron Reactor (SFR): wet storage, reprocessing process and transportation. This paper assesses the benefit of taking account of the depletion of FR fuel elements in transportation. So far, criticality-safety assessments have used a conservative approach assuming a fresh fissile fuel and penalizing plutonium content and isotopic composition for both fissile and fertile assemblies. The effect of the irradiation – which can lead to an important gain in reactivity – is consequently not considered.
Nevertheless, taking credit for the irradiation of the FR used fuel elements requires, among many other aspects, the definition of a set of penalizing hypotheses that ensure the conservatism of both the isotopic composition and the criticality calculations. In particular, the choice of the isotopic modelling approach can have an impact on the criticality-safety analysis of FR used fuel elements transportation.
The first part of the paper presents the calculation tools and models as well as the depletion calculation conditions for fissile and fertile used fuel assemblies. Those parameters are then used in the criticality analysis of an Orano TN's representative transport cask for FR used fuel transportation.
The second part of the paper evaluates the cask reactivity gain in comparison to the fresh fissile fuel assumption.
Finally, the last part of the paper concludes on the potential applicability of the study in terms of criticality-safety evaluation needs. It focuses specifically on depletion code validation, a key element for a straightforward and effective implementation of this approach.

## 2. DEPLETION COMPUTATIONAL MODELS AND ASSUMPTIONS

### a. Computer codes and nuclear data

The PEPIN2 depletion solver of the DARWIN2.3 [1] package calculates the isotopic concentrations at the end of the irradiation or after a given cooling time.

Figure 1 describes the sequence between the irradiation and the depletion calculations.

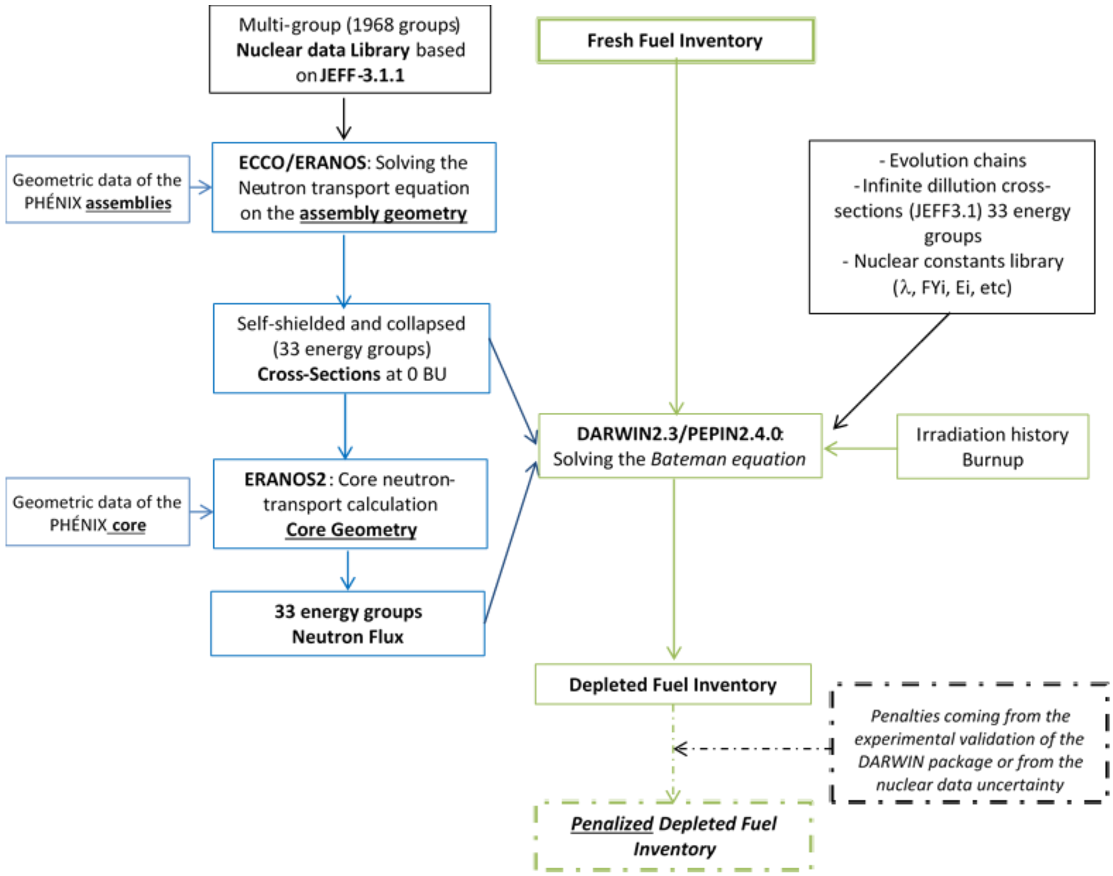


**Figure 1. Overview of the Fast-neutron Reactor depletion and criticality calculation chain**

DARWIN2.3 is the French reference calculation package for fuel cycle applications, such as fuel inventories and decay heat. It performs the nuclide depletion calculation (PEPIN2 solver), involving nuclear data libraries on the one hand and neutronics data on the other hand (Figure 1). All the decay data and fission yield values come from the JEFF-3.1.1 evaluation [2], whereas the self-shielded cross-sections and neutron spectra are provided by a deterministic neutron transport code, namely ERANOS2 [3], resolving the Boltzmann equation on the whole reactor core for Fast-Neutron Reactor studies. When needed, complementary cross-sections are directly taken from JEFF-3.1.1 evaluation. It is to notice that DARWIN2.3 is experimentally validated for FR fuels [4][5].

b. The Phenix core depletion configuration

As previously mentioned, this paper focuses on the French Sodium Fast Reactor (SFR) Phenix elements. The depletion calculations were performed on the whole Phenix core.
As illustrated in Figure 2, two rings of **radial blankets** (fertile assemblies) surround the fissile internal cores (namely core 1 and core 2) inside the Phenix core.
Moreover, the **axial fertile elements**, named **Lower** and **Upper Axial Blankets** (respectively LAB and UAB), are part of the fissile fuel pins. It is to mention that, in order to be transported, stored and reprocessed, the Upper Axial Blanket is removed from the other part of the pin.

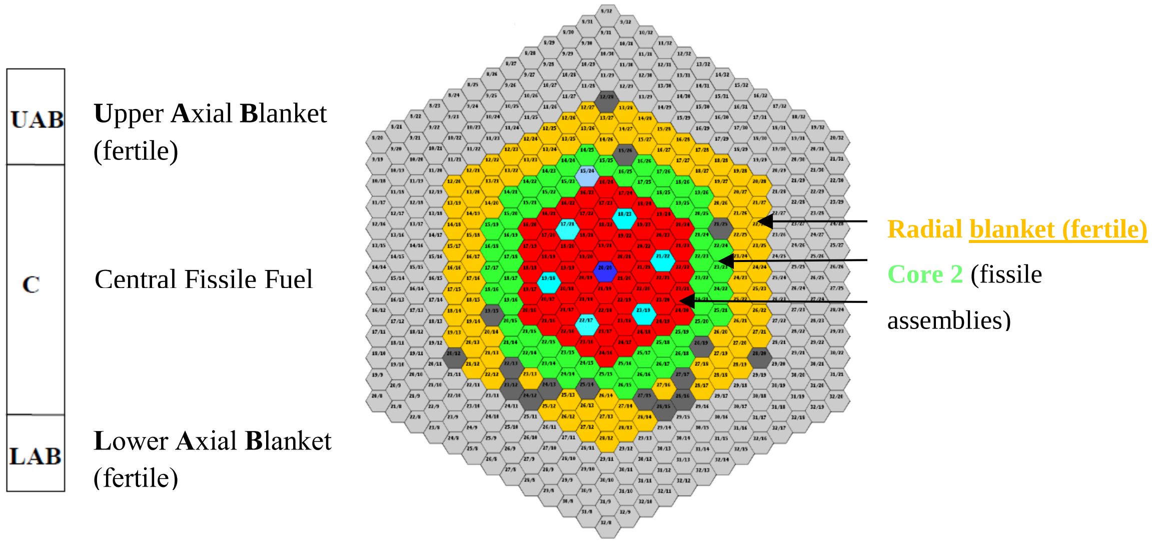


**Figure 2. An illustration of a Phenix fissile pin and of an ERANOS2 model of the SFR Phenix core**

c. Fissile Fuels

**Fresh Fuel Inventory**

The state-of-the art approach for transportation criticality-safety is to assume that irradiated fissile assemblies are fresh fissile fuels with given plutonium content and composition. In this study, an isotopic composition and a plutonium weight ratio (**Table I**) covering the ones of the Phenix fissile fuels are used.
As a reminder, the fissile fuels are made up of Mixed Oxide (MOx) fuel.

**Table I. Phenix fissile fuel isotopic composition and plutonium content used in the depletion calculation**

| Isotopic composition [wt. %] | | | | | | Pu content : Pu/(U+Pu) [wt. %] |
|---|---|---|---|---|---|---|
| $^{235}U$ | $^{238}U$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ | |
| 0.7 | 99.3 | 74.4 | 15 | 9.7 | 0.9 | 30 |

**Irradiated Fuel Inventory**

Penalizing irradiation conditions have not been defined for this study. Standard operational conditions values were used for the irradiation parameters such as the fuel, coolant and structure temperatures, the control road

insertion and the environment. Moreover, the irradiation calculations were done on the central part of the assembly in the center of the fissile core. Various burnup values were calculated from 0 to 125 GWd/t.
Furthermore, the nuclides of interest for burnup calculation have been selected on the basis of the 27 BUC nuclides (12 actinides and 15 fissions products) chosen for LWR-MOx fuels [6].
The results of the depletion calculations are presented in Figure 3 for the plutonium composition and content and in Figure 4 for the minor actinides and fission products.

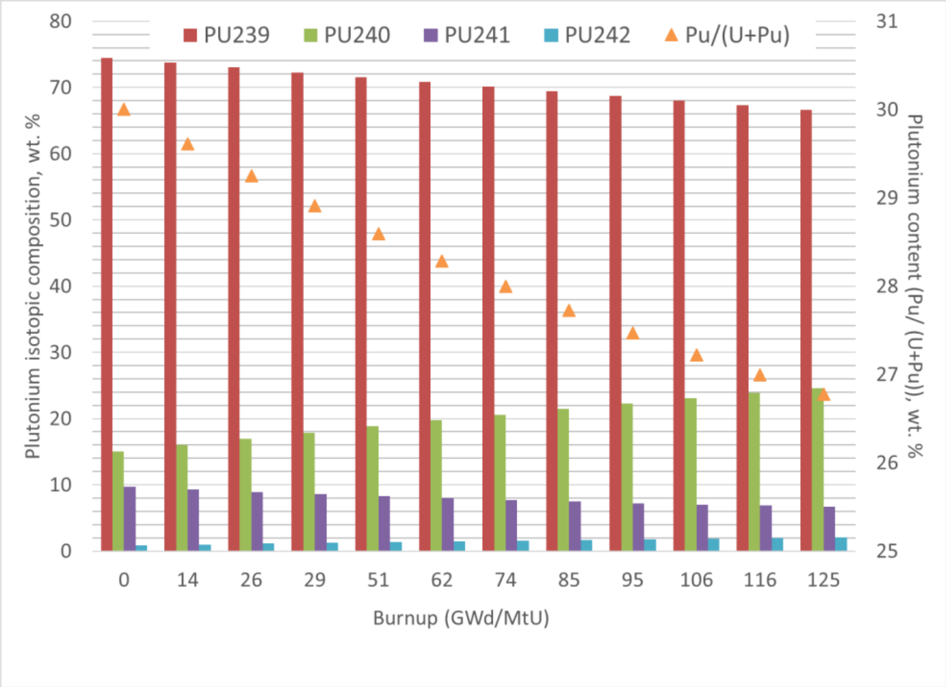


**Figure 3. Evolution of the plutonium content of the fissile fuel as a function of the burnup**

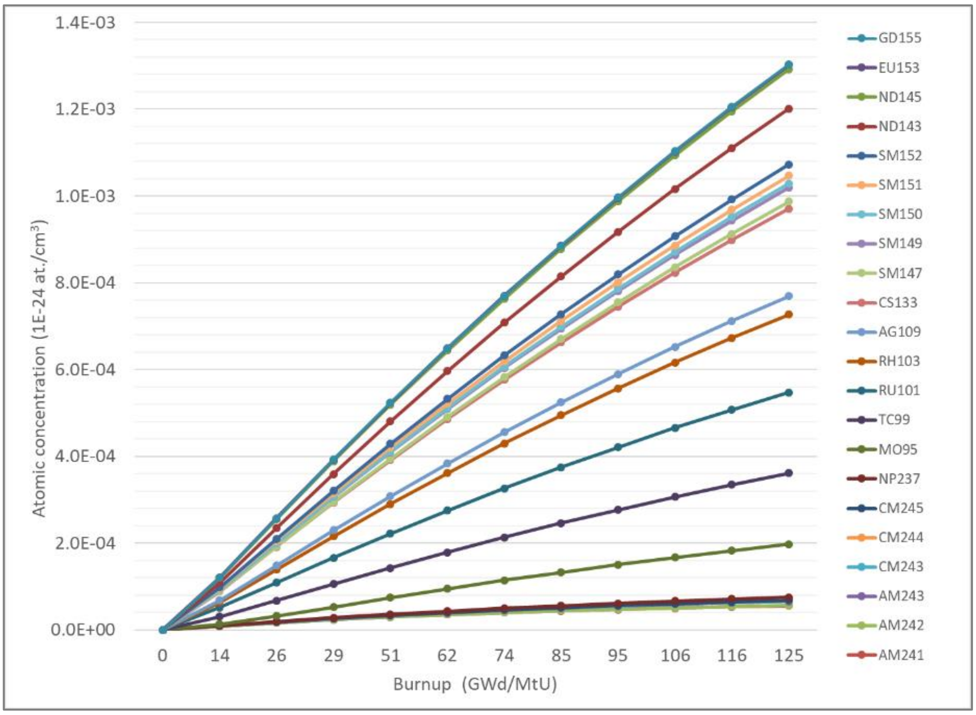


**Figure 4. Minor Actinides and Fission Product content of the fissile fuel as a function of the burnup**

d. Fertile Elements

**Fresh inventory**

**The fertile elements** (lower axial and radial blankets) are made up of depleted uranium. Their reactivity increases with increasing irradiation. This is essentially due to the uranium-to-plutonium conversion under the fast neutron spectrum conditions of the Phenix SFR.
In criticality-safety studies, the blankets are usually regarded to as fresh fissile fuel with a penalizing initial composition (see subsection 3.). Considering them as irradiated fertile fuels offers substantial criticality margins but requires the use of a conservative approach. In particular, the evaluation of the maximal reactivity of the fertile element needs to be provided. The definition of such an approach has been presented in a previous study [7].
Thus, a set of conservative depletion options were defined to account for the fertile element irradiation. As regards the assumption on the fresh fuel composition to be used, sensitivity studies (100% $^{238}U$ vs. depleted uranium) have shown that it was not of importance beside the other hypotheses like the irradiation conditions. Thus, **depleted uranium with 0.2% of $^{235}U$** was used for this work in the depletion calculations.

**Irradiated Fertile Element Inventory**

Regarding the irradiation and depletion condition of the fertile elements, the results from a previous study focusing on the definition of a conservative approach to account for irradiated fertile element in criticality-saftey studies [7] have been used. To obtain a conservative irradiated blanket inventory -from the criticality point of view- the irradiation parameters leading to maximizing the $^{239}Pu$ production have been chosen. $^{239}Pu$ coming from the radiative capture of $^{238}U$, the highest the neutron-flux level is, the most efficient the $^{238}U$ capture and so the $^{239}Pu$ production is. Thus, the configuration maximizing the neutron-flux level received by the fertile assemblies was defined. This penalizing configuration (see [7]) has been used to calculate the irradiated inventory of the fertile element presented in Table II.

**Table II. Isotopic composition and plutonium content of irradiated Phenix fertile elements**

| Isotopic composition [wt. %] | | | | | | | | Pu content : Pu/(U+Pu) [wt. %] |
|---|---|---|---|---|---|---|---|---|
| $^{235}U$ | $^{236}U$ | $^{238}U$ | $^{238}Pu$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ | |
| 0.08 | 0.03 | 99.89 | 0.10 | 86.22 | 12.67 | 0.95 | 0.06 | 7.8 |

## 3. CRITICALITY COMPUTATIONAL MODELS AND ASSUMPTIONS

a. Computer code and nuclear data

Criticality calculations are performed using the “standard route” APOLLO2 MORET 4 of the French criticality safety package, CRISTAL V1 [8].
The “standard route” makes use of the CEA93 172-energy group nuclear data library (derived from the JEF2.2 evaluation) and employs the APOLLO2 deterministic code [9] for flux, self-shielded and homogenized cross-sections calculations. The MORET 4 multigroup Monte Carlo code [10] provides the neutron multiplication factor associated for the studied configuration.

b. Transport cask model

Criticality calculations are carried out on the TN® 17/2 Orano TN transport cask which consists of a thick forged steel shell (~300 mm). The internal arrangement of the cask for Phenix used fuel transportation is constituted of two baskets. Each basket has 12 compartments which can be loaded with two types of Phenix irradiated fuel rods: $(U\text{-}Pu)O_2$ fissile fuel rods or $UO_2$ fertile fuel rods (radial an upper axial blankets, see subsection 2.).

The calculation model is an individual cask loaded in isolation according the IAEA regulation [11] for the transport of radioactive material and under Accidental Conditions of Transport (ACT). The forged shell of the cask is directly surrounded by a 200-mm water layer and it is assumed that the integrity of the Phenix fuel rods was not guaranteed under ACT (see Figure 5). Hence, fuel rods are considered completely damaged in each compartment and, in a conservative manner, only the fissile material (fresh $(U\text{-}Pu)O_2$ or composition $UO_2$ fertile after irradiation) is modelled.

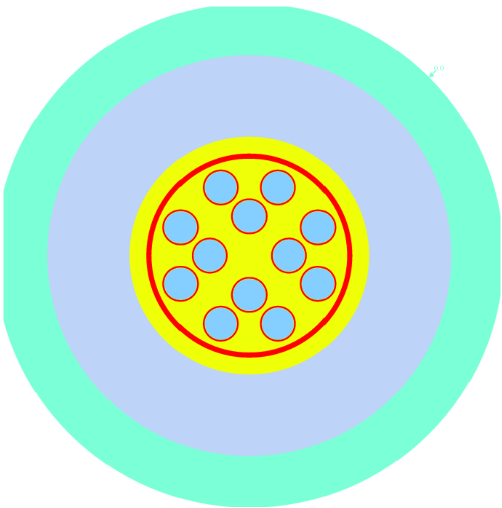

**Figure 5. Radial view of the TN® 17/2 criticality model**

Criticality calculations for the TN® 17/2 TN transport cask loaded with Phenix fuel rods are carried out using the conservative isotopic compositions give in Table III. It is important to notice that the isotopic composition used for the fertile Phenix fuel rods corresponds to a hypothetical and unrealistic isotopic composition after irradiation. It is used, in conservative manner, for criticality-safety analyses of irradiated fertile Phenix fuel rods.

**Table III. Phenix fertile and fissile fuel rods isotopic composition and plutonium content used in the criticality calculation**

| | Isotopic composition [wt. %] | | | | | | Pu content : Pu/(U+Pu) [wt. %] |
|---|---|---|---|---|---|---|---|
| | $^{235}U$ | $^{238}U$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ | |
| Fissile fuel rods | 0.7 | 99.3 | 74.4 | 15 | 9.7 | 0.9 | 30 |
| Fertile fuel rods | 0.7 | 99.3 | - | - | 100 | - | 8 |

Furthermore, according the IAEA requirements [11] it is demonstrated under ACT that a limited quantity of water (less than 1 litre) can enter the cavity of the cask. Therefore, the neutron spectrum of the industrial application is not thermalized.

## 4. TRANSPORT CASK REACTIVITY GAIN

### a. Fissile Fuel

The burnup reactivity credit ($\Delta k_{eff}$) is calculated as follow:

$$\Delta k_{eff} = k_{eff}(BU) - k_{eff}(BU = 0)$$

Figure 6 presents the reactivity credit as a function of the fissile fuel burnup and regarding the selected nuclides. Three sets of nuclides were studied: the major actinides, the minor actinides, and the 15 Burnup Credit Fission Products.

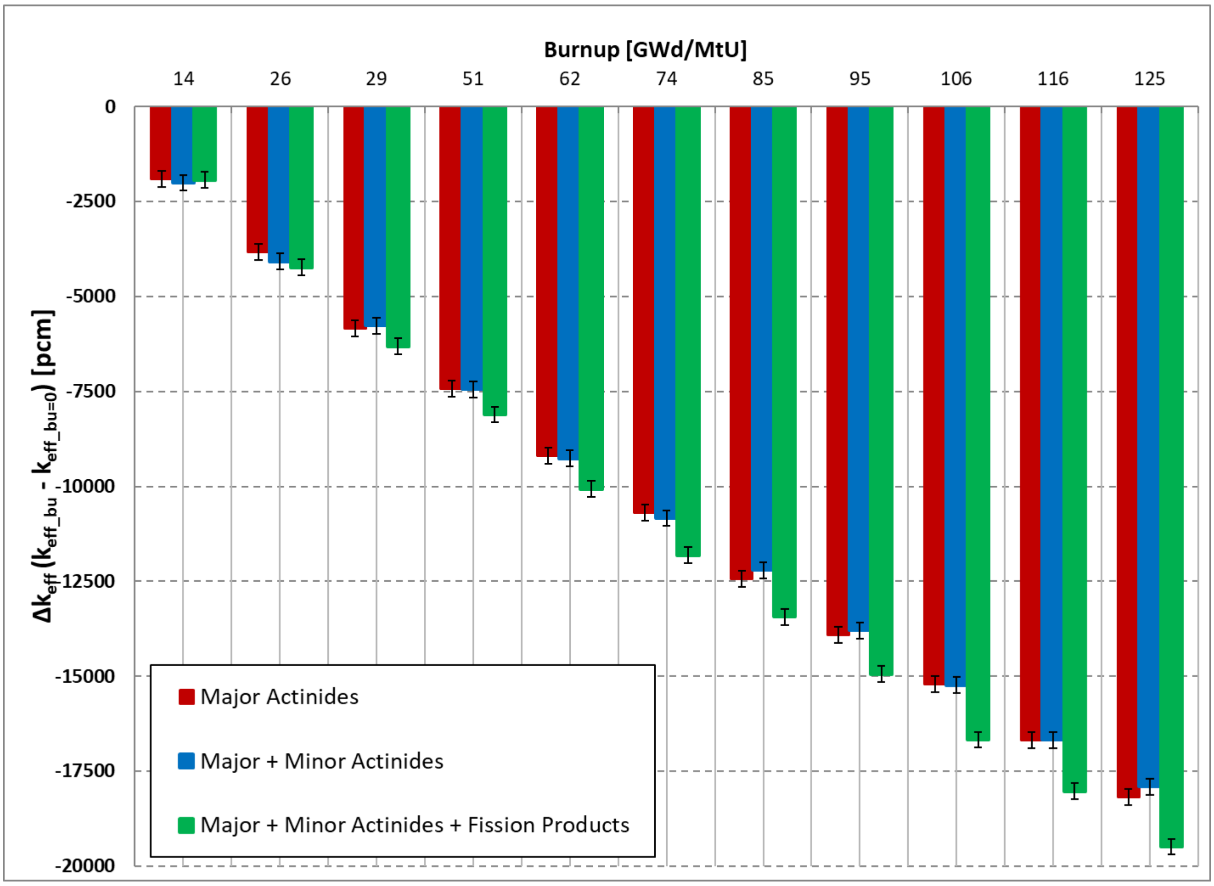


**Figure 6. Transport cask reactivity gain as a function of the fissile fuel burnup**

Those results are interesting and show that the use of a low burnup (14GWd/t) is enough to meet the $k_{eff}$ accidental safety-criterion and to allow the loading of a larger number of assemblies. Thus, substantial margins could be expected. It is to be noticed that taking credit from minor actinides and fission products is not of interest for this cask configuration, the neutron spectrum being not thermalized.

The most significant parameter regarding the burnup credit is the decrease of the plutonium content and the variation of the plutonium isotopic composition (Figure 3).

### b. Fertile element

The $k_{eff}$ of the fertile configuration using a 100% $^{241}$Pu content is about 0.75. The TN® 17/2 cask loading with fertile fuel rods is covered - in term of reactivity- by the fissile fuel rods loading. Therefore, margins could not be really expected on this specific case.

However, the use of irradiated fertile elements on this cask configuration show that substantial negative reactivity gain (around +40%) could be expected. This gain may be considerably interesting to greatly reduce costs if applied to the design of new baskets.

## 5. APPLICABILITY: DISCUSSION AND REQUIEREMENTS

One of the most significant requirement to expand “burnup” credit is the validation of depletion codes.

Experiments for Phenix fissile and fertile assemblies, respectively TRAPU and DOUBLON, were performed between 1977 and 1981 in the Phenix reactor.

As for the **fissile fuel** transport cask configuration, it was shown that the reactivity credit mostly comes from the change in plutonium content and composition. In this regard, the TRAPU experiment provides useful information. In this experiment, fuel pins were irradiated in a well characterized neutron spectrum near the centre of the core. Three pins were analysed with plutonium isotopic contents and ratios (Table IV) similar to the ones used in this study.

**Table IV. Isotopic content of the TRAPU experiment pins (% wt.) [4]**

| | $^{238}$Pu | $^{239}$Pu | $^{240}$Pu | $^{241}$Pu | $^{242}$Pu | (Pu/(U+Pu)) |
|---|---|---|---|---|---|---|
| TRAPU1 | 0.12 | 73.26 | 21.92 | 3.99 | 0.71 | 19.6 |
| TRAPU2 | 0.77 | 71.37 | 18.54 | 7.42 | 1.9 | 19.25 |
| TRAPU3 | 0.22 | 33.97 | 49.4 | 10.03 | 6.38 | 28.04 |

The interpretation of these experiments for the experimental validation of the DARWIN2.3 package [4] has shown that the final quantities of $^{239}$Pu and $^{240}$Pu are well predicted with calculation-over-experiment ratios about 1.004 ± 0.011 and 0.991 ± 0.012. However, a slight underestimation of $^{241}$Pu has been noticed, around 0.969 ± 0.007, which is related to an underestimation of the integral capture of $^{240}$Pu. Moreover, these results are consistent with the interpretation of separate sample irradiation experiments, as PROFIL, related to the nuclear data validation [8] [13].

The DOUBLON experiment was dedicated to the validation of **radial fertile blankets** calculations and so may be of interest for potential applications of irradiated fertile assemblies in transportation. The experiment performs a detailed study of two fertile assemblies of the first and second row of the Phenix core and is part of the DARWIN2.3 depletion calculation package experimental validation database. The main results of the DOUBLON interpretation are summarized in Table V.

**Table V. Average Calculation-over-Experiment ratios on the final nuclides amount of the DOUBLON experiment [4]**

| | **C/E ± σ** |
|---|---|
| **$^{238}$Pu / $^{239}$Pu** | 1.835 ± 1.669 |
| **$^{239}$Pu / $^{238}$U** | 0.984 ± 0.069 |
| **$^{240}$Pu / $^{239}$Pu** | 0.992 ± 0.165 |
| **$^{241}$Pu / $^{239}$Pu** | 0.881 ± 0.281 |
| **$^{242}$Pu / $^{239}$Pu** | 0.395 ± 0.298 |

The average calculation-over-experiment ratios on the final amounts of $^{234}$U, $^{235}$U, $^{236}$U and $^{239}$Pu are excellent. This is all the more important as they are the main isotopes in the final inventory. Concerning the $^{240}$Pu – which is produced in much smaller quantities – the average calculation-over-experiment ratio is also very good, but with higher uncertainty. On the contrary, the calculation-over-experiment ratio of the $^{238}$Pu, $^{241}$Pu and $^{242}$Pu are high, with a very important pin-to-pin dispersion.

Nevertheless, the criticality calculations have shown that the benefit to account for irradiated fertile elements are substantial. So, should important penalties result from the validation of the depletion calculations, the interest to account of the irradiation will remain of interest for transport applications.

## 6. CONCLUSION

This paper assesses the benefit of taking account of the depletion of FR fuel elements in transportation. So far, criticality-safety assessments have used a conservative approach assuming a fresh fissile fuel and penalizing plutonium content and isotopic composition for both fissile and fertile assemblies. The effect of the irradiation – which can lead to an important gain in reactivity – has not been consequently considered yet.

Interesting results show that the use of a low burnup level may allow an optimised loading of FR fissile fuel assemblies in the TN® 17/2 Orano TN transport cask. Furthermore, these results also encompass criticality calculations for FR fertile assemblies. Indeed, due to TN® 17/2 transport cask design, substantial margins are available for FR fertile assemblies, even for strongly conservative hypotheses on fuel inventory. Consequently, considering their irradiation is not of interest for this case, but could be for other transportation configurations.

Finally, the depletion code validation plays a crucial role for a straightforward and effective implementation of this approach. A brief analysis of the depletion code experimental validation for FR fissile and fertile assemblies show that useful information form relevant experiments is available, which provides accurate calculation-over-experiment ratios as well as uncertainties.

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