COSAC: A computational code for investigations on nuclear fuel cycles

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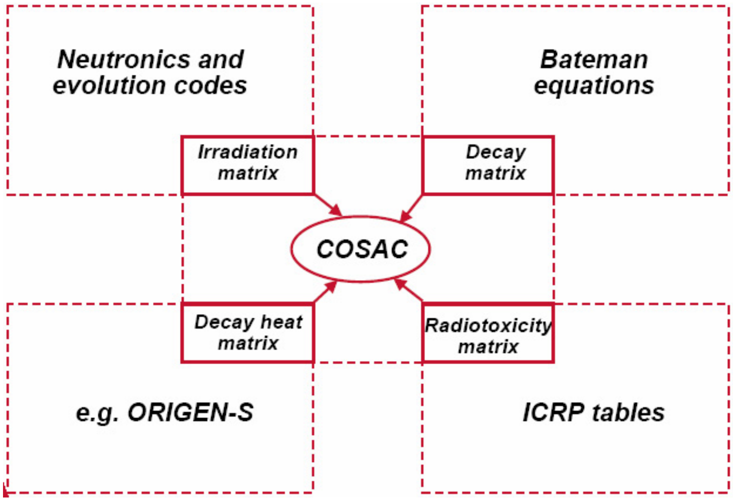
Abstract

In support to reactor design studies, AREVA NP has developed the COSAC code for the purpose of evaluating scenarios based on reactor technologies and related fuel cycles. This allows AREVA NP to provide customers with some elements to select a reactor technology, a fuel cycle and the appropriate deployment schedule according to some specific needs bound to the countries. COSAC code has been developed for quick computation of the results. So, some simplified assumptions have been made in the code, especially in neutronics and other reactor physic issues. Such issues are addressed outside the code and are then integrated to the code via matrices. Matrices are the expression of neutronics evolution from fresh to discharged fuel. They result from calculations in neutronics and reactor physics codes, and are introduced into COSAC after an appropriate data treatment. The underlying assumption behind this simplified matrix-based method is that fuel linearly depletes versus its initial isotopic composition, as long as the fuel discharge occurs at the same burn-up and the same reactivity level. To do so, an equivalency formula is used in order to automatically adjust the mass content in the fuel to its initial isotopic composition. Such an approach, based on a linear relationship between a few known inputs (typically, less than ten) and their corresponding outputs derived from some computer depletion simulations, let the so-built matrices to be then broadly applied to a wide range of other inputs. The advantage of this approach is to offer an efficient but yet reliable means to evaluate various spent fuel contents over multiple reactor fleet cycles, and to quickly compute many different options relative to a scenario family, a reactor technology or a given fuel cycle. The accuracy of this matrix-based method has been successfully tested for several types of reactors and fuel cycles, separately or combined together: PWR UOX fuel, PWR MOX fuel, and some preliminary studies have been carried out with COSAC for advanced fuel cycles such as minor actinides combined with (U, Pu, O_2) burning in sodium fast reactor according to the current trend of scenario studies: This paper highlights the high variety scenarios studies that can be achieved with COSAC. Its simplified linear approach of fuel depletion makes simple or complex fuel cycle evaluations possible in a quite short time, and allows users to bring out the global trends from many possible scenario options.

Introduction

COSAC is a scenario-based computer model for the simulation of the fuel cycle. It has been developed by AREVA for the ten past years in the specific aim to estimate main fuel cycle parameters such as: natural uranium need, enrichment and fuel fabrication requirements, spent fuel arising in intermediate or ultimate storage, and reprocessing capacity requirements.

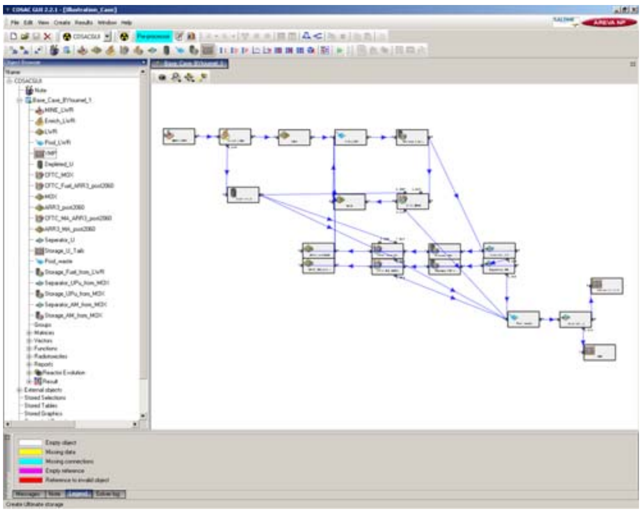
No physics is directly supported by COSAC itself. On the contrary COSAC uses results from some external physics codes to take account of some physics phenomena in fuel cycle calculations, such as depletion and radioactive decay. Results from external physics calculations are then introduced in COSAC in the form of matrices.



Nuclear material flows

Overall material flow for a nuclear fuel cycle can be sketched by tracking the nuclear materials entering and exiting each process in a nuclear fuel cycle. COSAC and its graphical user interface (GUI) allow concrete and clear connections between the processes. Once the user has created all the processes in a scenario and defined the features of each, he can join the processes each other to define the circulating paths of nuclear material between processes.

An example of COSAC modelling appears in the snapshot below, where each process is explicitly modelled by a "box", and circulating paths for nuclear material are materialised by "blue lines" as connections between processes.



Calculating isotopic inventories, radioactive decay, decay power and radiotoxicity

Each calculation step in COSAC is processed with specific matrices and vectors:

- *Irradiation matrices* (noted here [*I*]) for depletion calculations of nuclear material in reactors:

$$V' = [I].V$$

- *Decay matrices* (noted here [*D*]) for radioactive decay calculations of nuclear material in storage installations:

$$V' = [D]\frac{T}{T'}.V$$

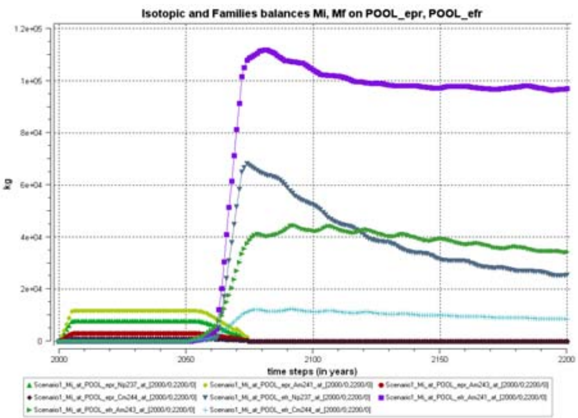
- *Decay power vector* (noted here [Π]) for power decay calculations emitted by nuclear material in storage installations:

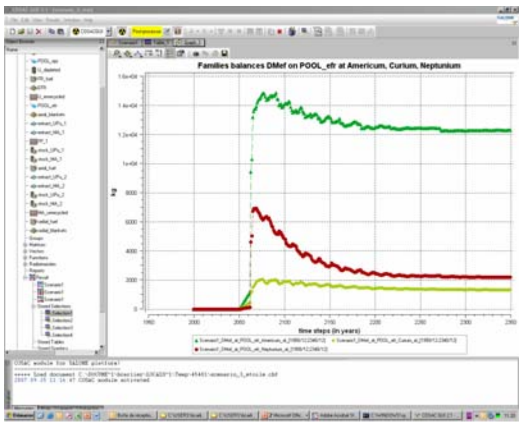
$$PR^{tot} = \sum_i PR_i = \langle \Pi | V \rangle = \sum_i \pi_i V_i$$

- *Radiotoxicity vector* (noted here [*D*]) for gamma (*γ*) and neutron (*n*) radioactivity calculations emitted by nuclear material in storage installations:

$$D^{(total)} = D^{(\gamma)} + D^{(n)} = \sum_{r=\gamma,n}\sum_{g}\left[R^{(r)}_{radio}\, lg\, g * \sum_i \left[A^{(r)}_{activity}\, lg\, i * V_i \right]\right]$$

The result is an extraordinary computational speed and a nearly unlimited flexibility to computing new scenarios: each computational run lasts between a few seconds to a few minutes, and there is practically no limit in adding new fuel cycles and new nuclear reactor types, and in estimating new scenarios as shown in the below graphs:





Flexible transition system from LWR cycle to FBR cycle*

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Abstract

In order to establish the future ideal FBR fuel cycle system from current LWR fuel cycle, the transition scenario from LWR to FBR is investigated in detail and the Flexible Fuel Cycle Initiative (FFCI) is proposed as a suitable transition system. The FFCI removes ~90% uranium from LWR spent fuel in LWR reprocessing and residual material named recycle material (RM), which contains ~50% U, ~15% Pu and ~35% other nuclides, is treated in FBR reprocessing to recover Pu and U. If the FBR deployment rate becomes lower, the RM (~1/10 volume of original spent fuel) will be stored until the higher rate again. Storage form of the RM will be either liquid or solid considering the easiness for Pu/U recovery afterwards and RM storage.

The FFCI has some merits compared with ordinary system that consists of full reprocessing facilities for both LWR and FBR spent fuels during the transition period. The economy is better for FFCI due to the smaller LWR reprocessing facility (no Pu/U recovery and fabrication). The FFCI can supply high Pu concentration RM, which has high proliferation resistance and flexibly respond to FBR introduction rate changes. Volume minimisation of spent LWR fuel is possible for FFCI by its conversion to RM.

Several features of FFCI were quantitatively analysed such as Pu mass balance, reprocessing capacities, spent LWR fuel amounts, RM amounts, and proliferation resistance to compare the effectiveness of the FFCI system with other systems. The calculated Pu balance revealed that the FFCI could supply enough but no excess Pu to FBR. These analyses demonstrated the applicability of FFCI system to the transition period from LWR cycle to FBR cycle.

* The present study includes the result of "Research and Development of Flexible Fuel Cycle for the Smooth Introduction of FBR" entrusted to Hitachi-GE Nuclear Energy, Ltd. by the Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT).

Introduction

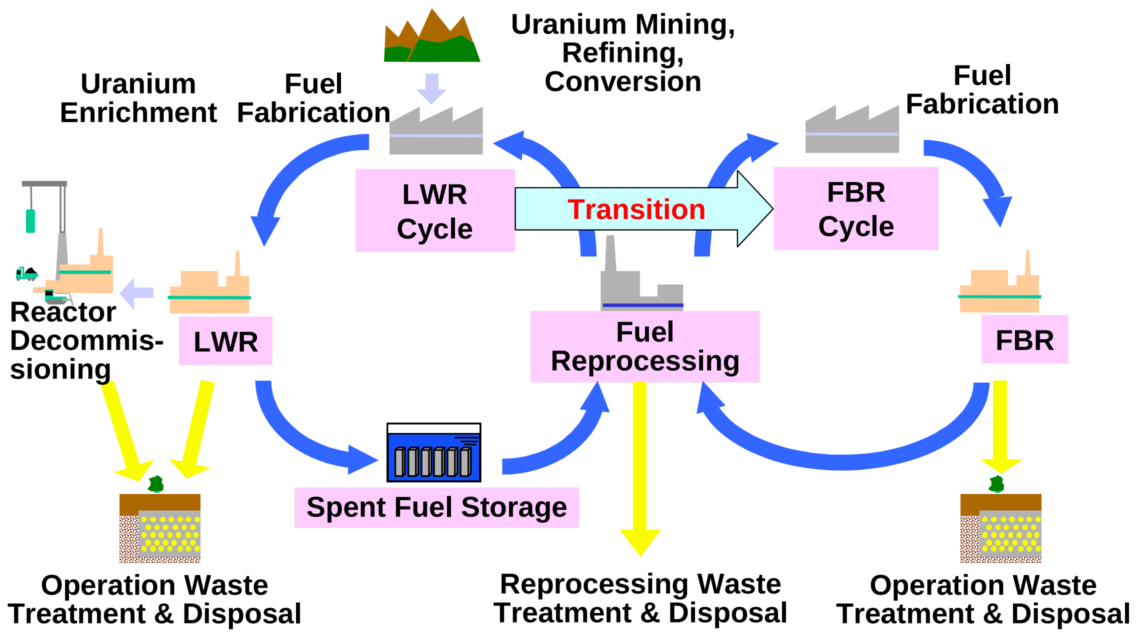
Japan aims at the maximum utilisation of energy sources and minimum emission of green house gases by the introduction of fast breeder reactor (FBR) cycle. The Framework for Nuclear Energy Policy states that commercial FBR deployment will start around 2050 under the suitable conditions for the replacement of light water reactor (LWR) with FBR. Conditions to be considered are FBR economy, U price, innovative technologies, international trends, proliferation resistance, and so on, which depend on many uncertain future factors including FBR deployment start-time and rates. Thus flexible management of the transition period fuel cycle from LWR to FBR is quite important to establish the equilibrium FBR cycle.

The transition scenario from LWR to FBR is investigated in detail and the Flexible Fuel Cycle Initiative (FFCI) is proposed as a suitable transition system. In this paper the characteristics of FFCI were evaluated considering the factors such as Pu mass balance, reprocessing capacities, spent LWR fuel amounts and proliferation resistance to compare the effectiveness of FFCI with other systems.

Transition fuel cycle and its issue

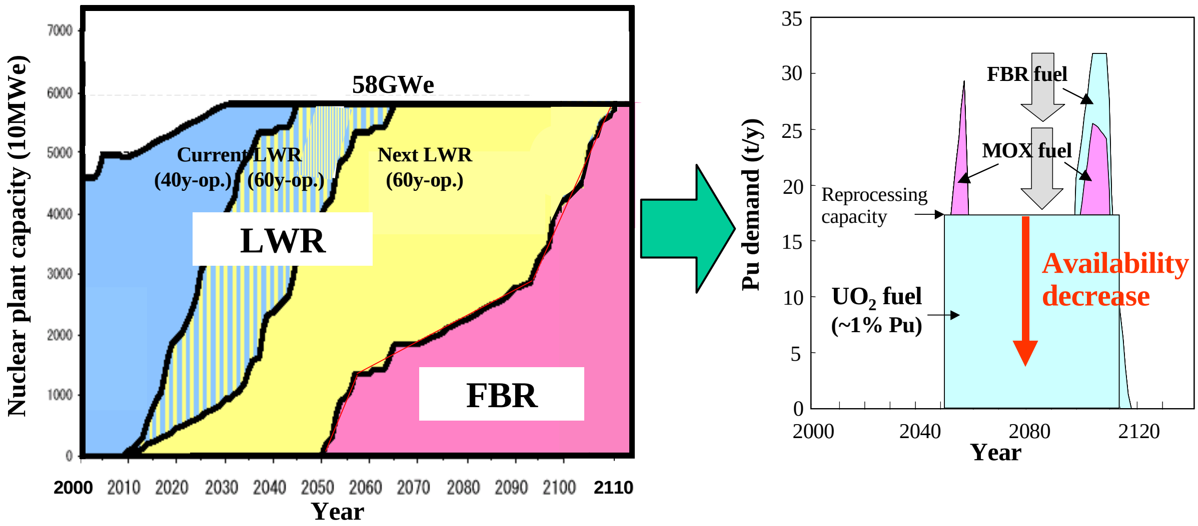
Figure 1 shows the outline of the current LWR fuel cycle and the future FBR fuel cycle. We have 54 light water reactors in Japan. Uranium mining, refining and enrichment are carried out in foreign countries, and uranium enrichment (partially) and uranium oxide fuel fabrication are carried out in Japan. After generating electricity in LWR spent fuel is stored inside the nuclear power plant site and will be stored in the interim storage facility outside the site. Spent LWR fuel is partially reprocessed in Tokai Reprocessing Plant (TRP) and will be commercially reprocessed in Rokkasho Reprocessing Plant (RRP). It is important to consider how to treat recovered plutonium of about 8 t/y in RRP and Japan uses the plutonium in LWR as mixed oxide (MOX) fuel with uranium for maximum resource utilisation and nuclear non-proliferation. The capacity of RRP is 800 t/y and the spent fuel is generated about 1 000 t/y, then we need an interim storage facility for the excess amount of spent fuel and another reprocessing plant to reprocess the stored spent fuel.

Figure 1: LWR and FBR fuel cycles



Recovered Pu obtained by the LWR spent fuel reprocessing is necessary for the deployment of FBR, *i.e.* for the transition from LWR to FBR. The left figure of Figure 2 shows the image for the transition from LWR to FBR described in the Framework for Nuclear Energy Policy. The deployment of FBR will

Figure 2: Typical transition scenario and Pu demand for FBR deployment



start around 2050 under the suitable conditions for its construction and operation. Total nuclear plant capacity is assumed to be 58 GWe. The figure indicates the replacement of LWR after 60 years operation with FBR and shows the non-linear deployment curve (non-constant deployment rate). The FBR deployment rate is fast (~2 GWe/y) at the beginning and the end of the transition period from around 2050 to 2110, and slow (~0.5 GWe/y) at the middle.

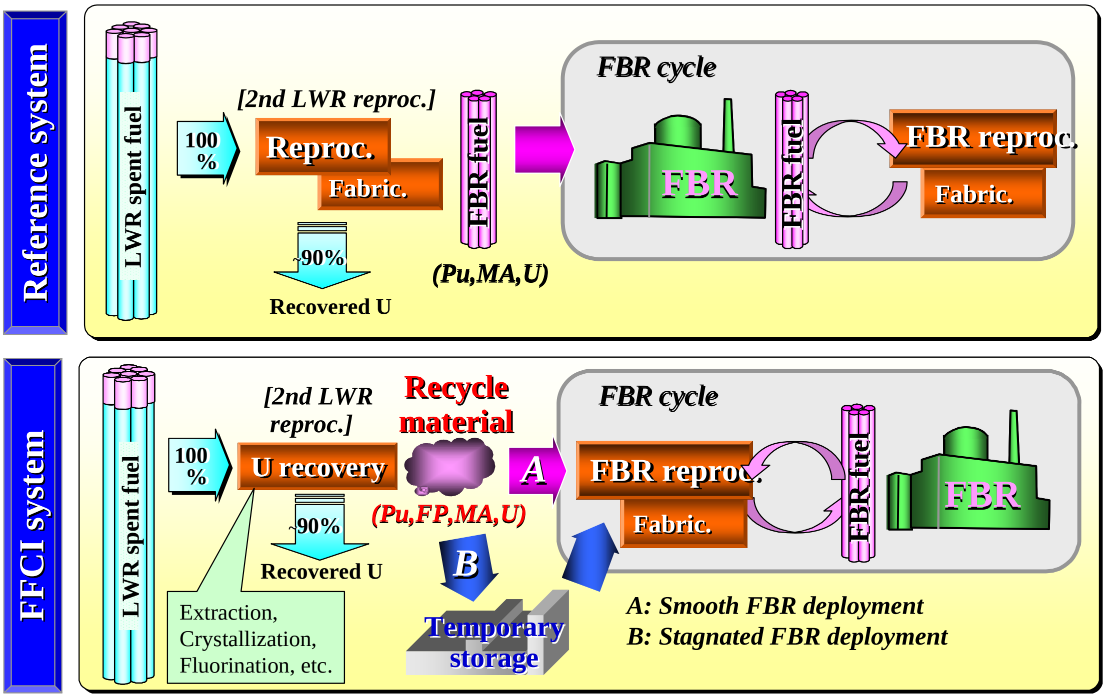
The right figure of Figure 2 is calculated Pu amount required for FBR deployment. The FBR deployment needs initial MOX fuel (Pu) from outside and the required Pu amount can be calculated by the FBR deployment rate. The Pu demand will be high at the beginning and the end of the transition period even utilising (reprocessing) the LWR MOX spent fuel (higher Pu content) and FBR spent fuel (bred Pu), and low at the middle. As it is better to minimise the excess Pu amount in Japan, the availability of the LWR reprocessing plant of which capacity should be high for the begging FBR deployment might be lowered, which makes the fuel cycle economy worse. Thus the flexible fuel cycle system is desirable to respond uncertainties predicted in the transition era from LWR to FBR.

Proposed fuel cycle system

Figure 3 shows the proposed fuel cycle system. Upper system is the currently proposed fuel cycle system (reference system), where the LWR and the FBR reprocessing plants independently have head-end, U recovery, Pu/U recovery, and FBR fresh fuel fabrication systems. The second LWR reprocessing plant separates U from LWR spent fuel, and remove radioactive waste from the remainder (extract Pu/U), and fabricate fuel in an FBR fuel fabrication facility, then supply the fuel to FBRs. The role of second LWR reprocessing is to supply Pu needed for FBR deployment, and its capacity and availability are highly dependent on the FBR deployment status.

The lower system of Figure 3 is the new fuel cycle system FFCI (Flexible Fuel Cycle Initiative) that is aimed to enhance the flexibility to accommodate various uncertainties in transition cycle from LWR to FBR. In FFCI, LWR reprocessing only carries out about 90% U removal from LWR spent fuel, then the composition of remaining spent fuel (recycle material) is about 40% U, 15% Pu and 45% other nuclides. Uranium can be recovered by the methods such as solvent extraction, crystallisation and fluoride volatility. Recovered U will be purified and utilised in LWR after re-enrichment. Recycle material (RM) is transferred to FBR fuel reprocessing to recover Pu/U followed by FBR fresh fuel fabrication. Conventional solvent extraction and pellet fabrication methods can be applied for this purpose. If the FBR deployment rate becomes lower (stagnation), the RM (~1/10 volume of original spent fuel) will be stored until the higher rate again. The FFCI has some merits compared with ordinary system that consists of full reprocessing facilities for both LWR and FBR spent fuels during the transition period, that is smaller LWR reprocessing facility, spent LWR fuel reduction, storage and supply of high proliferation resistant and high Pu density RM that can flexibly respond to FBR deployment rate changes.

Figure 3: Fuel cycle systems for the transition from LWR to FBR



For the stagnation period for FBR deployment (lower FBR deployment rate period in Figure 2), reference system will store product Pu without considering excess Pu storage (non-proliferation) or store LWR spent fuel with considering non-proliferation. On the other hand, FFCI system will store recycle material which has high proliferation resistance due to the coexistence of fission products (FP), minor actinides (MA) and U.

Mass balance analysis

The fuel cycle material balance was calculated for reference and FFCI systems based on the transition scenario shown in the left figure of Figure 2. Calculated values are the amounts of LWR spent fuel reprocessing, FBR spent fuel reprocessing, Pu storage and LWR spent fuel interim storage. The storage forms of Pu (Pu containing material) during low FBR deployment rate are reprocessing product or LWR spent fuel for reference system and recycle material for FFCI system.

The calculation results are shown in Figure 4 for the reference system with reprocessed Pu storage (no Pu stock limit) and with LWR spent fuel storage (Pu product stock limit of 30t), and FFCI system with recycle material storage (Pu product stock limit of 30t). Storage limit of 30t for reprocessed Pu product is the same value for Rokkasho Reprocessing Plant (RRP). The FFCI system has the following characteristics compared with reference system: *i)* The LWR reprocessing function (U recovery only) is minimised compared with reference systems and its capacity is reduced compared with the Pu limit case. *ii)* The FBR reprocessing construction should be earlier and a little bit larger than the reference systems. *iii)* The Pu product (pure Pu) storage can be avoided. *iv)* The storage amount of LWR spent fuel is decreased compared with reference system of the Pu limit case.

Conclusions

The newly proposed FFCI system has evaluated in this work, which revealed the flexibility and applicability to the transition fuel cycle from LWR to FBR. Although the FFCI must construct the FBR reprocessing plant earlier and a little bit larger than the reference systems, the FFCI can minimise the LWR reprocessing function and capacity, avoid the Pu product (pure Pu) storage and proliferation resistance decrease, and steadily reduce the storage amount of LWR spent fuel.

Figure 4: Calculation results for reference system with product Pu storage (left), reference system with LWR spent fuel storage (middle) and FFCI system with recycle material storage (right)