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Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes

NUCLEAR ENERGY AGENCY

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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Foreword

Under the auspices of the NEA Nuclear Science Committee (NSC), the Working Party on Scientific Issues of the Fuel Cycle (WPFC) has been established to co-ordinate scientific activities regarding various existing and advanced nuclear fuel cycles, including advanced reactor systems, associated chemistry and flowsheets, development and performance of fuel and materials, accelerators and spallation targets. The WPFC has different expert groups to cover a wide range of scientific fields in the nuclear fuel cycle.

The Expert Group on Fuel Cycle Transition Scenarios Studies was created in 2003 to study R&D needs and relevant technology for an efficient transition from current to future advanced reactor fuel cycles. The objectives of the expert group are to (1) assemble and organise institutional, technical, and economics information critical to the understanding of the issues involved in transitioning from current fuel cycles to long-term sustainable fuel cycles or a phase-out of the nuclear enterprise and (2) provide a framework for assessing specific national needs related to that transition.

After reviewing national, regional or worldwide transition scenarios, the expert group performed a benchmark study to compare the existing codes in terms of capabilities, modelling and results. The benchmark was conducted in two phases: (1) depletion calculations for PWR UOX, PWR MOX and fast reactor calculations and (2) transition calculation using various scenario codes (COSI, FAMILY21, VISION, EVOLCODE and DESAE) using three different transition scenarios (once-through, limited plutonium recycling in LWRs and plutonium and minor actinides recycling in fast reactors). The comparison mainly focused on the mass flow and the composition of heavy elements depending on time, i.e. natural uranium needs, enrichment needs, fresh fuel fabrication needs, fuel irradiation, inventory of spent fuel and nuclear materials, reprocessing needs, etc.

Acknowledgements

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Executive Summary

Several scenario codes have been developed to study the future of nuclear energy in different countries and institutes. These codes allow simulating scenarios for nuclear energy at national, regional or worldwide level. They simulate the whole fleet of nuclear reactors with associated fuel cycle plants and storage for nuclear materials. The codes all model the dynamic transition from an initial state to a final state for nuclear energy. The results are mainly the mass flow and the composition of heavy elements depending on time: natural uranium needs, enrichment needs, fresh fuel fabrication needs, fuel irradiation, inventory of spent fuel and nuclear materials and reprocessing needs. Some codes also allow calculation of nuclear waste production, economic aspects and non-proliferation criteria.

The Expert Group on Fuel Cycle Transition Scenario (FCTS) is working under the guidance of the Working Party on Scientific Issues of the Fuel Cycle (WPFC) of the NEA. National, regional or worldwide transition scenarios are studied in this expert group with different existing tools devoted to scenario studies. After reviewing existing national scenarios, one of the first missions of this expert group is to compare the existing scenario codes in terms of capabilities, modelling and results. Thus, it was decided that a benchmark should be performed between the existing codes.

The benchmark on scenario codes is divided into two parts:

- A depletion part: the objective was to compare simple depletion calculations in the case of well-known standard reactors. The codes involved in this part were ALEPH developed at SCK-CEN-Belgium, CESAR4 developed at CEA-France, FAMILY21 developed at JAEA-Japan and a code package of KIT, which couples neutronic with burn-up modules using DANTSYS and TRAIN. The results indicate a good agreement for some uranium and plutonium isotopes and some significant differences for other heavy nuclides.
- A transition scenario part in which 5 scenarios codes were involved: COSI6 developed at CEA-France, DESAE2.2 developed at ROSATOM-Russia, EVOLCODE2.0 developed at CIEMAT-Spain, FAMILY21 developed at JAEA-Japan and VISION2.2 developed at INL-USA. The purpose was to compare the methodologies used by the codes and a set of important results. These codes have different capabilities. For instance, some codes allow economic assessments, waste package calculations and non-proliferation criteria assessment. Thus, to make the comparison possible, only the common capabilities were compared and the benchmark focused on heavy nuclides calculations: U, Pu, Am, Np and Cm.

Three different scenarios were selected of three different levels of complexity. The first scenario is the open cycle simulating a LWR fleet with direct disposal of spent fuel. The second scenario is more complex and simulates the single recycling of plutonium in the LWR. The third scenario is the most complex and simulates the transition between a LWR fleet and a Generation IV fast reactor fleet recycling Pu and minor actinides. These scenarios are not considered as realistic options for the future, they are only case studies allowing comparison of the methodologies and the physical models used in the scenario codes.

The results and the analysis of the calculations lead to the following conclusions:

- The general trends observed for each code are the same for the three scenarios calculated in the benchmark.
- All the scenario codes give very close results for scenario 1. However, this scenario is very simple and cannot serve as a reference for the comparison of the codes.
- When the level of complexity of the scenario increases, some differences appear. The comparison of the results in the second scenario demonstrates the importance of initial assumptions and the common interpretation of the hypotheses and magnitudes in the comparison.
- A tuning of the assumptions is always necessary. This necessity is due to the difference of interpretation for initial conditions and to some missing assumptions which may appear. Thus, several iterations can be necessary to converge.
- Once the tuning has been made, some differences remain and come from:
  - The capacity of modelling of the codes. Particularly, the loading of the fuel batches is annually averaged for all codes except COSI6 which treats discrete batches (Figure 26), and can lead to a more complex comparison of the results.
  - The transition period in scenario 3: Figures 57 and 58, Tables 22 and 24 indicate that the results are very close for the equilibrium period but some differences appear during the deployment of fast reactors. A reduction of these differences would have necessitated several more iterations.
  - The difference in physical models: how the decay of heavy nuclides is taken into account in the interim storages (Figure 43 to Figure 46), differences in depletion calculations (Chapter 4.1 Depletion), differences in equivalence calculations for Pu and minor actinide fractions in the fresh fuel (Figure 42).
  - The simulation of the heterogeneous cores of fast reactors which is not possible in VISION version 2, the version used for the benchmark.
  - The flexibility offered by the codes to simulate or not the first cores and the renewing of the reactors (case of DESAE2.2, Figure 50).
  - Some remaining unexplained behaviours (case of DESAE2.2).

Some useful code capability additions would be the simulation of the decay of nuclear materials during the storage of separated materials and wastes (case of VISION), the simulation of TRU losses from reprocessing (case of DESAE2.2), the simulation of annual averaged batches of fuel (case of COSI6) and the use of equivalence models for the calculation of Pu or TRU content in the fresh fuel of LWR MOX and fast reactors in the case of EVOLCODE, FAMILY21, VISION2.2 and DESAE2.2).

Also, some differences will remain whatever the level of the analysis and the number of iterations because of the different methods of calculations.

This benchmark was limited to comparison in heavy elements material flows. A comparison for isotopes would probably have led to other differences and would have necessitated a more detailed investigation of the physical models used by the codes.

A similar benchmark co-ordinated by MIT (MIT-NFC-TR-105, April 2009), involving COSI6, VISION2.2, CAFCA and DANESS led to the same type of conclusions.

1. Introduction

1.1. Background

Past studies of the implementation of partitioning and transmutation performed in the Nuclear Science Committee of the OECD/NEA have mostly concentrated on equilibrium mode scenarios, where the global infrastructure is fixed and mass flows of materials are constant. These studies have resulted in a deep understanding of the possibilities of partitioning and transmutation (P&T) to address nuclear waste issues and have indicated the infrastructure requirements for several key technical approaches. While these studies have proven extremely valuable, several countries have also recognised the complex dynamic nature of the infrastructure problem: severe new issues arise when attempting transition from current open or partially closed cycles to a final equilibrium or burn-down mode. While the issues are country-specific when addressed in detail, it is believed that a series of generic issues exists related only to the current situation and to the desired end point. Some examples are listed here:

- time lag to reach equilibrium, which can take decades to centuries;
- a wide range of transmutation performance for the various technologies involved;
- accumulation of stockpiles of materials during either a transition phase or a growth period;
- very significant and possibly prohibitive investments required to reach equilibrium;
- complex interactions with final waste disposal paths.

The Expert Group on Fuel Cycle Transition Scenario (FCTS) is working under the guidance of the Working Party on Scientific Issues of the Fuel Cycle (WPFC). National, regional or worldwide transition scenarios are studied in this expert group with different existing tools devoted to scenario studies. After reviewing existing national scenarios, one of the first missions of this expert group was to compare the existing codes in terms of capabilities, modelling and results.

A similar exercise involving COSI6, VISION2.2, CAFCA and DANESS was co-ordinated by MIT (MIT-NFC-TR-105, April 2009).

1.2. Objective of the study

The objective of this study was to perform a benchmark between the different available scenario codes devoted to nuclear energy transition scenarios. The first part of the benchmark involved depletion calculations for PWR UOX, PWR MOX and fast reactor calculations.

The objective of the second part was to compare the various scenario codes (COSI, FAMILY21, VISION, EVOLCODE, DESAE), applied to 3 different transition scenarios: once-through, limited plutonium recycling in LWRs, plutonium and minor actinides recycling in FRs. The codes enable analysis of the dynamic transition between an initial and a final

state for nuclear energy. The results compared are mainly the mass flow and the composition of heavy elements depending on time: natural uranium needs, enrichment needs, fresh fuel fabrication needs, fuel irradiation, inventory of spent fuel and nuclear materials, reprocessing needs, etc.

2. Computer codes selected for the benchmark

The following chapters briefly describe the five codes selected for the benchmark exercise.

2.1. FAMILY21

2.1.1. Introduction and history

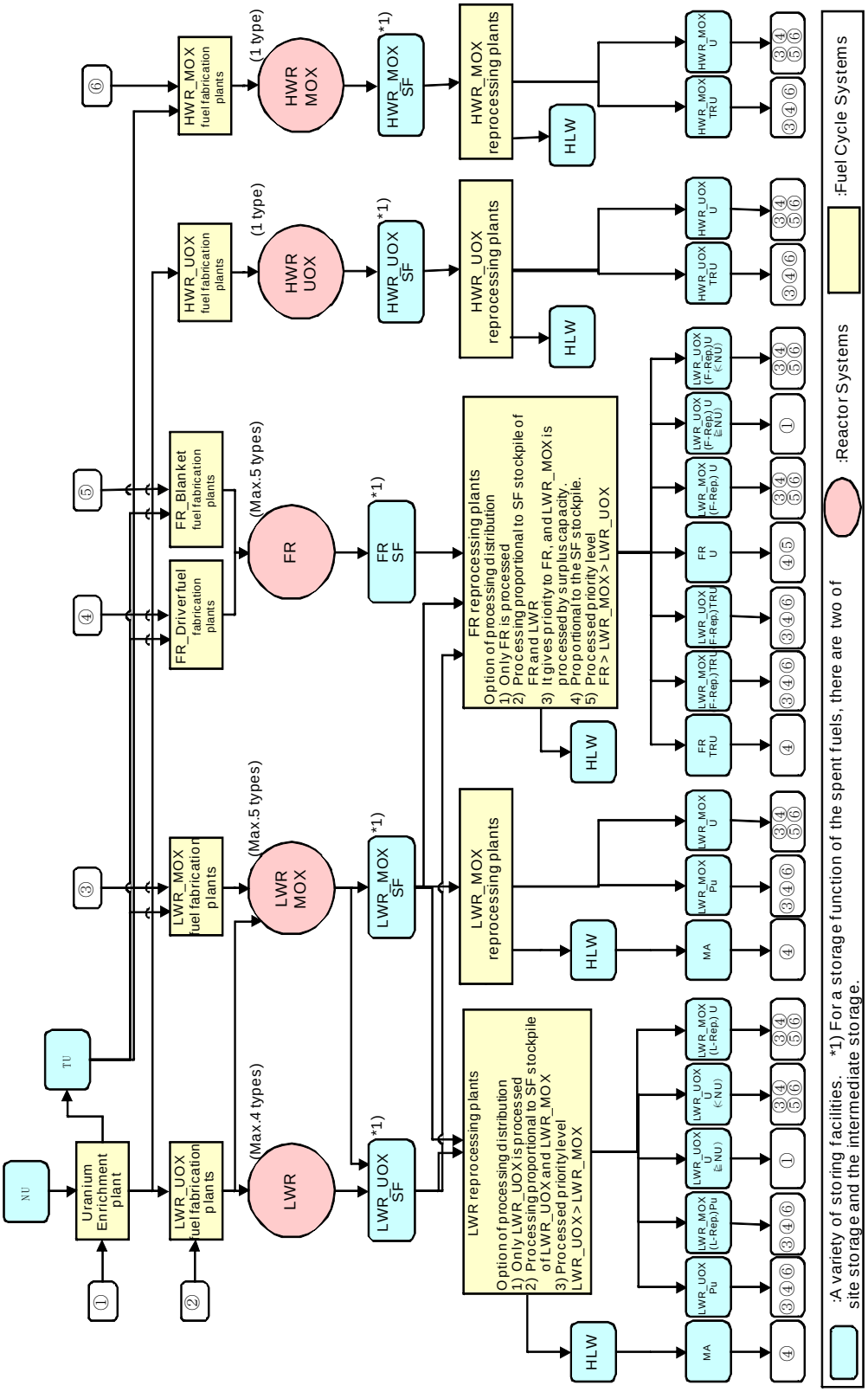
In the study on transition scenario from thermal reactor to fast reactor, the simulation code was necessary in order to have a clear grasp of the reactor system and its nuclear fuel cycle system adaptable to future uncertain nuclear needs. The quantitative evaluation on the time-dependent change of balance and composition for nuclear fuel material in the LWR cycle and the FR cycle was particularly important.

FAMILY code was developed for the purpose of a policy and technical risk control for future nuclear plans. The first edition programmed by FORTRAN language was executed with a mainframe computer until about ten years ago. Then, FAMILY Excel version (FAMILY-EX), which is stand-alone system type, was developed with the advent of high performance personal computer and convenient spreadsheet software. This FAMILY-EX has been used for a simple evaluation since 1998. FAMILY21 used for the benchmark study is the latest version of the FAMILY code series. This code has two features. One is usability like FAMILY-EX and another is consideration of the change of the isotopic composition of the nuclear fuel. A plot-type of FAMILY21 was developed in 2000 and FAMILY21 has been used for the activities of the Japanese New Nuclear Policy-planning Council and the JAEA's FaCT Project (Fast Reactor Cycle Technology Development Project) after 2003.

2.1.2. FAMILY21 capabilities

FAMILY21 is composed of input tool, solver, graphic tool and post-processor and can simulate mass balance of the whole or a part of the system shown in Figure 1. In addition, this code can calculate simultaneously up to 15 types of reactors (LWR x 9, HWR x 1, and FR x 5) in combination with all the different coolants and fuels. The end users can choose and add reactors data in their own way. Sodium-cooled reactor, gas-cooled reactor, lead-bismuth-cooled reactors and water-cooled reactors are treated in the fast reactor model. Furthermore, one of three types of reprocessing plants can be chosen for the reprocessing of LWR-MOX spent fuels, namely dedicated reprocessing plant for LWR-MOX spent fuels, LWR reprocessing plant (mixed reprocessing of LWR-UOX spent fuels and LWR-MOX spent fuels) and FR reprocessing plant.

Figure 1: Material flow scheme of FAMILY21 code



Input tool

The input data necessary for a calculation by FAMILY21 are classified into 'cycle option data' and 'system characteristic data'. The cycle option data are existence or non-existence of reprocessing, a reprocessing method for LWR-MOX spent fuels, minor actinide (MA) recycling conditions, control conditions for automatic calculations for installation capacity of fast reactors and reprocessing plants, etc. Most cycle option data can be set by the input tool based on a graphical user interface (GUI) function. The system characteristic data are nuclear power generation capacity, reactor data, transfer coefficient of various nuclides in each process of fuel cycle facilities, etc. Maintenance and management of the system characteristic data can be easily performed with Microsoft Excel spreadsheets. For example, tail uranium concentration of the uranium enrichment plant is given as variable annual data using EXCEL spreadsheets.

Main input data are as follows:

- reactor core characteristic data (fuel mass balance); set standard weight and material composition of loading and discharge fuels for initial core, equilibrium core and final core of each type of reactors;
- ex-core time period; set for each type of reactors;
- uranium tail enrichment; select fixed value during calculation or variable values;
- transfer coefficient or loss factor of various nuclides; set for enrichment plant, conversion plant, fuel fabrication plant and reprocessing plant;
- reprocessing plant capacity; select manual or automatic operation;
- priority order of utilisation of uranium recovered from reprocessing; set for each type of reactor; it is also possible to apply the uranium recovered from LWR reprocessing plant to matrix of FR fuels;
- MA recycling conditions; select full recycle, delayed recycle, or partial recycle; set MA cooling time after separation at reprocessing plant and upper limit of MA content of new FR fuels;
- priority order of utilisation of plutonium; it is also possible to estimate the coexisting scenario of LWR-MOX cycle and FR cycle, in which the plutonium recovered from FR spent fuels is preferentially applied to LWR MOX fuels and then plutonium recovered from LWR-MOX is refreshed in FR cycle;
- reprocessing methods of LWR-MOX spent fuels; select dedicated reprocessing plant for LWR-MOX spent fuels, LWR reprocessing plant and FR reprocessing plant; direct disposal option is also available for LWR-MOX fuels.

Solver

The solver plays a crucial role in the simulation functions of FAMILY21. The computation function of the solver is modularised to the frontend facilities, nuclear reactors (a depletion calculation) and the backend facilities. In consideration of the performance of the personal computer and response time of the parametric analysis, the number of isotopes treated in calculation of the solver is limited to total 20 (except the number of isotopes in the depletion calculation is total 38). They are uranium (^{233}U-^{236}U, ^{238}U), plutonium (^{238}Pu-^{242}Pu), neptunium (^{237}Np only), americium (^{241}Am-^{243}Am) and curium (^{242}Cm-^{246}Cm). The decay calculation of the actinide nuclides (total 20 isotopes) and the core depletion calculation (total 38 isotopes) are processed by each matrix function made from the results of depletion calculation of the ORIGEN2 code. Therefore, FAMILY21 can calculate time variation of the isotopic composition in multi-recycling with high precision without using other analysis codes together.

Furthermore, the solver has automatic calculation functions to regulate the installation capacity of fast reactors and reprocessing plants. Computation time of a few days needed for optimisation works by manual operation was reduced to about 20 minutes for national analysis by this automatic function and the repeatability of the calculation was improved markedly.

The software specifications and the hardware requirements of the solver are as follows:

- language: Microsoft Visual Basic (Japanese edition);
- executable lines: 16 500 steps (about 200 subroutines);
- computation time; about 20 mins for national analysis, about 1-2 hours for global analysis (CPU: Intel Core2 Duo E8600 3.33GHz, RAM: 3.25GB);
- simulation period: 200 years.

Graphical tool

The calculation results of the solver are output as a CSV format file by each computation module. The graphical tool programmed by Microsoft Visual Basic (Japanese edition) generates total 100 graphs with their data tables based on an output file of the solver for the purpose of various scenario evaluations. Those graphs and data tables can be freely edited by using Microsoft Excel spreadsheet.

Post-processor

The post-processors are individual calculation functions corresponding to each evaluation purpose, for instance, calorific value of FR new fuel assemblies, amount of the HLW disposal, etc. The following typical calculation items are treated in the post-processor of FAMILY21.

- calorific value of FR new fuels for MA recycling;
- composition and quantity of fission products (FP) in the reprocessing plant;
- quantity and property of high-level radioactive vitrified waste at vitrified waste production step;
- composition;
- decay heat;
- radioactivity;
- geological repository area.

Function expansions by the post-processor are the effective means that do not damage a calculation function of the main body of FAMILY21. We are now planning to add a new post-processor for the low-level radioactive waste soon.

2.1.3. Brief description of the methods used in FAMILY21

The outline of the frontend module and the depletion matrix, a typical function of FAMILY21, are shown in this section.

Frontend module

The frontend module includes the models of uranium conversion facilities, enrichment facilities, fuel fabrication facilities and mainly calculates the natural uranium demand, TRU (plutonium and MA) demand and fuel fabrication amount. These are

calculated based on standard fuel composition of initial core and equilibrium core, lead-lag time and loss factor given by an input tool. When uranium recovered from reprocessing plant is enriched, a higher uranium concentration than the usual concentration requirement is given by frontend module in consideration of the reactivity compensation by thermal neutron absorption effect of ^{236}U. However, if the enrichment separate work for recovered uranium is higher than that for natural uranium, the recovered uranium does not enrich.

In addition, as the composition of TRU products always changes somewhat, in FAMILY21, fissile plutonium concentration of LWR-MOX fuels and the FR fuels are calculated by the following simultaneous equations (1), (2), and (3) with an equivalent fissile coefficient.

$$\varepsilon_U \sum_i \alpha_i \eta_i + \varepsilon_{Pu} \sum_j \beta_j \eta_j + \varepsilon_{MA} \sum_k \gamma_k \eta_k = E_0 \quad (1)$$

$$\varepsilon_U \sum_{i=MA} \alpha_i + \varepsilon_{Pu} \sum_{j=MA} \beta\alpha_j + \varepsilon_{MA} \sum_{k=MA} \gamma_k = X \quad (2)$$

$$\varepsilon_U + \varepsilon_{Pu} + \varepsilon_{MA} = 1 \quad (3)$$

where:

E_0 : equivalent fissile content in core fuel (wt%)

X : minor actinide concentration in core fuel (wt%)

εU : uranium ratio in fuel(wt%)

εPu : plutonium ratio in fuel(wt%)

εMA : minor actinide ratio in fuel(wt%)

α : uranium isotope composition(wt%)

β : plutonium isotope composition(wt%)

γ : minor actinide isotope composition(wt%)

η : equivalent fissile coefficients (shown in Table 1)

i,j,k : isotope number

i=MA,j=MA,k=MA: minor actinide in each material

Here, E_0 (equivalent fissile content in core fuel calculated by standard composition) is obtained with the following equation (4) in advance.

$$E_0 = E_U \sum_i A_i \eta_i + E_{Pu} \sum_j B_j \eta_j + (1 - E_U - E_{Pu}) \sum_k \Gamma_k \eta_k \quad (4)$$

E_U and E_{Pu} are weight ratio of the uranium and plutonium in a standard composition, respectively. In addition, A_i and B_i are uranium and plutonium isotope compositions (ratio by weight), and Γ_i is isotope composition of other nuclides. The equivalent fissile coefficients used in a benchmark study are shown in Table 1.

Table 1: Equivalent fissile coefficients

	PWR-UOX (FAMILY-21)	PWR-MOX (FAMILY-21)	FR (FAMILY-21)	FR (specification)
U233	5.19E-01	1.12E+00	1.31E+00	
U234	-1.50E-01	-5.61E-01	-5.17E-03	2.55E-02
U235	3.37E-01	4.78E-01	7.51E-01	7.75E-01
U236	-6.16E-02	-2.80E-01	-9.40E-02	-6.19E-02
U238	-5.31E-03	-2.14E-02	-7.02E-02	0.00E+00
Np237	-2.33E-01	-6.44E-01	-3.42E-01	-2.70E-01
Pu238	-1.70E-01	-1.66E-01	5.25E-01	5.78E-01
Pu239	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Pu240	-1.69E+00	-8.18E-01	6.52E-02	1.22E-01
Pu241	1.21E+00	1.59E+00	1.54E+00	1.47E+00
Pu242	-2.11E-01	-4.33E-01	1.74E-02	8.26E-02
Am241	-8.33E-01	-1.23E+00	-3.71E-01	-3.37E-01
Am242M	8.03E+00	8.15E+00	2.23E+00	2.18E+00
Am242	3.14E+00	3.97E+00	2.34E+00	
Am243	-3.63E-01	-1.23E+00	-3.52E-01	-3.24E-01
Cm242	-7.06E-04	1.13E-02	5.19E-01	3.11E-01
Cm243	1.30E+00	3.26E+00	2.40E+00	2.50E+00
Cm244	-1.17E-01	-4.03E-01	1.53E-01	2.09E-01
Cm245	2.19E+00	3.26E+00	2.46E+00	2.43E+00
Cm246	-1.15E-02	-4.58E-02	1.50E-01	2.29E-01

Depletion matrix

ORIGEN2 code used for core combustion analysis calculates production of a large amount of FP nuclides. On the other hand, mass flow calculation in the fuel cycle by FAMILY21 is performed for a limited number of actinide nuclides (total 20 isotopes of U, Np, Pu, Am, Cm). Since 'ORIGEN2/FAMILY21 Coupled Model' is thought to be inefficient because of the large demand of calculation resources, a depletion matrix (total 38 isotopes of ^{232}U- ^{252}Cf) for combustion calculations in FAMILY21 was developed. Actinide nuclides considered in depletion matrix and decay matrix are shown in Tables 2 and 3. The decay matrix is used for the material balance calculations of the nuclear fuel cycle.

The image of the depletion matrix in Excel spreadsheet is shown in Figure 2. The transmutation factors between the nuclides in the depletion matrix are stored based on results of combustion calculation by ORIGEN2 code. The making of a flow diagram of the depletion matrix is shown in Figure 3. In the combustion calculation by ORIGEN2 code, the transmutation factors between the nuclides in the depletion matrix are calculated step by step, using one group cross-section, neutron flux and irradiation periods corresponding to each type of reactor. These nuclear transmutation factors are stacked and completed as a matrix of 38×38 in length and width. The depletion matrix is prepared for each type of reactor. For LWR core, it consists of 2 regions of UO_2 fuel and MOX fuel and for FR core, depletion matrixes are provided for the core fuel, axial blanket fuel and radial blanket fuel.

Table 2: A nuclide for calculation by using depletion matrix (only use for combustion calculation)

Nuclides	Isotopes
U	^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{239}U
Np	^{236}Np, ^{237}Np, ^{238}Np, ^{2396}Np
Pu	^{236}Pu, ^{237}Pu, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{243}Pu, ^{244}Pu
Am	^{241}Am, ^{242m}Am, ^{242}Am, ^{243}Am
Cm	^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm
Bk	^{249}Bk, ^{250}Bk, ^{251}Bk
Cf	^{249}Cf, ^{250}Cf, ^{251}Cf, ^{252}Cf

Table 3: A nuclide for calculation by using decay matrix (fuel cycle whole)

Nuclides	Isotopes
U	^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U
Np	^{237}Np
Pu	^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu
Am	^{241}Am, ^{242m}Am, ^{242}Am, ^{243}Am
Cm	^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm

Figure 2: Sample of depletion matrix

Reactor LWR6 Library

Reactor Type APWR(60GWd/t) Core =WR47.03&LUX= 2.52E+14 N/CM**2-SEC

Fuel Type UO_2 Inside blankets 0

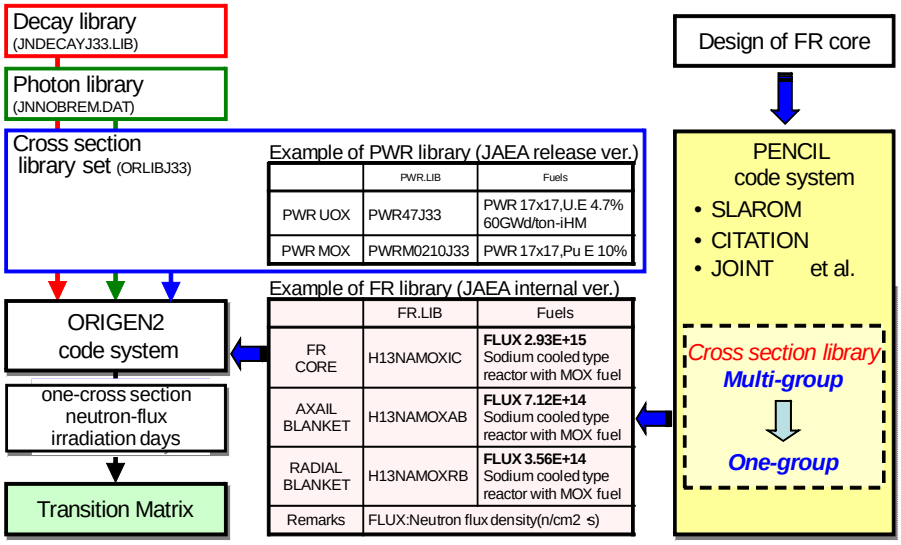
Reactor 37.7 (MWt) :ulation pc 530.5 d(atch ght of reactor core : 77.1 t-HM Radial blankets 0 FLUX= N/CM**2-SEC

Transition matrix of actinoid Axial blankets 0 FLUX= N/CM**2-SEC

Core	U232	U233	U234	U235	U236	U238	U239	Np236	Np237	Np238	Np239	Pu236	Pu237	Pu238	Pu239	Pu240	...	Bk249	Bk250	Bk251	Cf249	Cf250	Cf251	Cf252
U232	7.74E-01	1.02E-05	1.50E-07	5.52E-10	1.67E-08	8.12E-12	2.35E-13	4.28E-07	4.88E-07	7.29E-09	2.33E-13	2.22E-01	2.80E-07	7.41E-09	2.35E-13	4.08E-12		1.18E-20	2.59E-22	2.78E-24	1.04E-19	2.60E-22	2.78E-24	2.14E-23
U233	5.50E-02	5.25E-01	1.25E-05	1.88E-08	1.39E-08	6.59E-12	1.69E-12	4.70E-08	2.84E-07	7.52E-08	1.68E-12	9.20E-03	1.16E-05	7.64E-08	1.69E-12	2.35E-11		6.94E-20	2.40E-21	1.52E-23	6.19E-19	2.41E-21	1.52E-23	1.19E-22
U234	2.17E-03	4.79E-02	7.92E-01	3.17E-05	4.77E-05	2.49E-08	3.32E-07	1.40E-04	1.43E-03	8.52E-03	3.30E-07	4.04E-04	2.22E-03	8.61E-03	3.33E-07	5.71E-06		2.18E-14	8.03E-16	6.07E-18	1.80E-13	8.09E-16	6.07E-18	4.46E-17
U235	1.50E-04	4.99E-03	1.60E-01	6.05E-01	2.54E-05	1.02E-07	1.61E-05	1.08E-05	1.07E-04	8.78E-04	1.60E-05	2.30E-05	1.90E-04	9.00E-04	1.61E-05	2.45E-07		7.31E-16	1.03E-16	1.62E-19	6.37E-15	1.03E-16	1.62E-19	1.26E-18
U236	3.80E-06	1.73E-04	8.00E-03	7.12E-02	9.03E-01	3.20E-07	1.88E-05	6.47E-06	6.61E-06	3.19E-05	1.87E-05	6.88E-07	8.47E-06	3.28E-05	1.88E-05	7.78E-05		8.61E-15	1.42E-14	4.74E-18	2.76E-14	1.43E-14	4.74E-18	2.62E-17
U238	5.83E-10	3.38E-08	2.49E-06	2.79E-05	9.41E-04	9.89E-01	2.44E-08	2.70E-09	2.55E-09	6.84E-09	1.46E-08	1.16E-10	2.02E-09	7.29E-09	1.48E-08	1.44E-07		1.44E-12	3.19E-12	1.32E-15	2.97E-13	3.20E-12	1.32E-15	7.18E-15
U239	3.00E-16	1.74E-14	1.26E-12	1.44E-11	4.72E-10	4.95E-07	1.26E-14	1.39E-15	1.28E-15	3.52E-15	7.53E-15	5.96E-17	1.04E-15	3.75E-15	7.59E-15	7.43E-14		7.41E-19	1.64E-18	6.81E-22	1.53E-19	1.64E-18	6.80E-22	3.68E-21
Np236	2.07E-14	1.37E-12	1.22E-10	1.58E-09	6.63E-08	3.29E-11	9.41E-13	8.49E-02	9.35E-07	5.74E-11	9.34E-13	5.99E-08	6.04E-07	5.55E-12	9.42E-13	1.17E-11		5.33E-20	3.12E-22	2.19E-23	3.92E-19	3.13E-22	2.19E-23	1.46E-22
Np237	4.96E-08	2.87E-06	2.08E-04	2.32E-03	7.59E-02	3.84E-05	1.69E-06	1.46E-01	6.96E-01	6.73E-06	1.67E-06	6.72E-02	4.63E-01	6.73E-06	1.68E-06	1.70E-05		1.04E-13	7.73E-16	5.86E-17	6.95E-13	7.75E-16	5.86E-17	3.59E-16
Np238	1.07E-10	6.15E-09	4.47E-07	4.98E-06	1.71E-04	8.65E-08	3.62E-09	3.15E-04	1.51E-03	1.45E-08	3.60E-09	1.44E-04	9.95E-04	1.45E-08	3.62E-09	3.66E-08		2.24E-16	1.69E-18	1.26E-19	1.50E-15	1.69E-18	1.26E-19	7.72E-19
Np239	3.05E-13	1.76E-11	1.28E-09	1.48E-08	4.88E-07	7.14E-05	7.46E-10	7.72E-07	3.72E-06	8.65E-11	7.39E-10	3.54E-07	2.44E-06	9.92E-11	7.45E-10	8.80E-09		1.22E-13	2.63E-13	1.26E-16	2.54E-14	2.64E-13	1.25E-16	6.97E-16
Pu236	3.58E-14	2.47E-12	2.32E-10	3.18E-09	1.49E-07	7.35E-11	3.54E-12	2.50E-06	2.69E-06	3.76E-08	3.51E-12	5.19E-01	1.69E-06	3.80E-08	3.54E-12	5.39E-11		2.16E-19	4.93E-21	7.07E-23	1.71E-18	4.94E-21	7.07E-23	4.97E-22
Pu237	2.67E-15	1.95E-13	2.02E-11	2.89E-10	1.56E-08	1.30E-11	3.87E-10	7.19E-08	3.19E-07	8.70E-07	3.84E-10	8.33E-03	2.26E-06	8.77E-07	3.83E-10	2.00E-09		9.93E-18	3.66E-19	3.49E-21	7.51E-17	3.67E-19	3.49E-21	2.42E-20
Pu238	2.82E-09	1.95E-07	1.89E-05	2.58E-04	1.32E-02	7.33E-06	1.42E-04	3.28E-02	2.51E-01	7.17E-01	1.41E-04	3.26E-02	2.78E-01	7.22E-01	1.42E-04	2.03E-03		1.12E-11	4.06E-13	4.50E-15	8.17E-11	4.07E-13	4.50E-15	3.00E-14
Pu239	1.15E-10	9.00E-09	9.57E-07	1.62E-05	9.47E-04	4.76E-03	2.28E-01	2.91E-03	2.46E-02	1.24E-01	2.26E-01	3.09E-03	3.76E-02	1.24E-01	2.25E-01	1.32E-04		4.10E-12	8.59E-12	2.89E-15	5.27E-12	8.60E-12	2.89E-15	1.70E-14
Pu240	7.40E-12	6.57E-10	7.75E-08	1.62E-06	1.08E-04	9.22E-04	1.58E-01	3.96E-04	3.59E-03	3.01E-02	1.57E-01	4.49E-04	7.74E-03	2.99E-02	1.57E-01	3.00E-01		2.81E-10	4.97E-10	2.05E-13	6.08E-10	4.99E-10	2.05E-13	1.13E-12
Pu241	7.46E-13	7.35E-11	1.43E-08	2.35E-07	4.82E-05	6.99E-04	7.20E-02	7.30E-05	1.58E-03	8.72E-03	7.15E-02	8.72E-05	1.94E-03	9.17E-03	7.16E-02	2.89E-01		7.11E-09	1.15E-10	1.22E-11	3.32E-08	1.16E-10	1.22E-11	5.81E-11
Pu242	2.76E-14	3.02E-12	7.23E-10	1.26E-08	3.47E-06	8.51E-05	1.13E-02	4.94E-06	1.43E-04	9.56E-04	1.12E-02	6.27E-06	1.77E-04	1.09E-03	1.13E-02	8.98E-02		1.63E-06	2.96E-06	2.66E-09	3.63E-07	2.97E-06	2.66E-09	1.27E-08
Pu243	4.61E-18	5.06E-16	1.41E-13	2.11E-12	7.21E-10	1.77E-08	1.89E-06	8.27E-10	2.95E-08	1.60E-07	1.88E-06	1.05E-09	2.97E-08	1.83E-07	1.89E-06	1.50E-05		2.73E-10	4.96E-10	4.52E-13	6.07E-11	4.96E-10	4.51E-13	2.22E-12
Pu244	8.06E-20	9.87E-18	3.71E-15	5.39E-14	2.38E-11	9.60E-10	1.29E-07	2.65E-11	1.32E-09	8.45E-09	1.28E-07	3.62E-11	1.27E-09	1.09E-08	1.29E-07	1.75E-06		2.43E-08	5.50E-08	1.02E-07	4.77E-09	5.51E-08	1.02E-07	4.07E-07
Am241	4.67E-15	5.10E-13	1.27E-10	2.08E-09	5.20E-07	1.20E-05	1.72E-03	8.08E-07	2.31E-05	1.51E-04	1.71E-03	1.02E-06	2.83E-05	1.72E-04	1.71E-03	1.27E-02		1.60E-10	1.38E-12	1.49E-13	9.11E-10	1.39E-12	1.49E-13	8.13E-13
Am242M	2.95E-17	3.42E-15	9.66E-13	1.61E-11	5.14E-09	1.45E-07	1.97E-05	6.91E-09	2.33E-07	1.57E-06	1.96E-05	8.98E-09	2.74E-07	1.85E-06	1.97E-05	1.70E-04		1.80E-12	1.24E-14	1.33E-15	1.08E-11	1.25E-14	1.33E-15	7.63E-15
Am242	8.14E-18	8.88E-16	2.19E-13	3.64E-12	1.04E-09	2.39E-08	2.99E-06	1.41E-09	4.14E-08	2.64E-07	2.97E-06	1.78E-09	4.93E-08	3.00E-07	2.98E-06	2.21E-05		2.84E-13	1.14E-14	2.65E-16	1.59E-12	1.14E-14	2.64E-16	1.44E-15
Am243	6.88E-16	8.28E-14	2.93E-11	4.33E-10	1.78E-07	6.45E-06	8.53E-04	2.05E-07	9.33E-06	5.87E-05	8.47E-04	2.75E-07	9.30E-06	7.34E-05	8.53E-04	1.01E-02		1.42E-07	3.06E-07	1.46E-10	2.94E-08	3.06E-07	1.46E-10	8.09E-10
Cm242	2.74E-16	3.25E-14	9.80E-12	1.63E-10	5.55E-08	1.80E-06	2.64E-04	7.48E-08	2.74E-06	1.94E-05	2.62E-04	9.86E-08	3.19E-06	2.35E-05	2.64E-04	2.71E-03		2.37E-11	8.83E-13	1.48E-14	1.50E-10	8.84E-13	1.48E-14	8.84E-14
Cm243	1.04E-18	1.35E-16	4.86E-14	8.29E-13	3.26E-10	1.48E-08	2.63E-06	4.45E-10	1.98E-08	1.63E-07	2.61E-06	6.24E-10	2.34E-08	2.09E-07	2.63E-06	3.75E-05		6.82E-12	1.97E-13	1.11E-14	3.27E-11	1.97E-13	1.10E-14	5.26E-14
Cm244	3.05E-17	4.00E-15	1.88E-12	2.58E-11	1.40E-08	7.08E-07	9.91E-05	1.44E-08	8.86E-07	5.90E-06	9.84E-05	2.05E-08	7.99E-07	8.01E-06	9.94E-05	1.62E-03		2.60E-08	4.82E-08	2.51E-11	3.54E-08	4.82E-08	2.50E-11	1.36E-10
Cm245	4.29E-19	6.05E-17	3.51E-14	4.64E-13	3.03E-10	1.98E-08	2.96E-06	2.95E-10	2.23E-08	1.61E-07	2.94E-06	4.48E-10	1.91E-08	2.32E-07	2.97E-06	6.04E-05		1.41E-04	4.66E-07	5.27E-07	5.25E-04	4.66E-07	5.26E-07	2.15E-06
Cm246	5.67E-21	8.59E-19	5.98E-16	7.87E-15	5.84E-12	4.98E-10	8.41E-08	5.72E-12	5.06E-10	4.19E-09	8.35E-08	9.19E-12	4.33E-10	6.38E-09	8.46E-08	2.17E-06		1.49E-02	2.10E-02	5.34E-05	3.72E-03	2.10E-02	5.34E-05	2.19E-04
Cm247	1.35E-23	2.18E-21	1.78E-18	2.34E-17	1.94E-14	2.03E-12	3.88E-10	1.91E-14	1.93E-12	1.80E-11	3.85E-10	3.27E-14	1.65E-12	2.86E-11	3.91E-10	1.21E-08		3.80E-04	6.39E-04	1.65E-04	8.58E-05	6.39E-04	1.65E-04	7.85E-06
Cm248	1.65E-25	2.81E-23	2.64E-20	3.51E-19	3.21E-16	4.10E-14	8.75E-12	3.18E-16	3.62E-14	3.78E-13	8.67E-12	5.82E-16	3.11E-14	6.23E-13	8.82E-12	3.24E-10		3.13E-02	5.80E-02	8.48E-02	6.58E-03	5.81E-02	8.47E-02	2.78E-01
Bk249	5.04E-28	9.80E-26	1.02E-22	1.37E-21	1.34E-18	1.95E-16	4.47E-14	1.38E-18	1.64E-16	1.85E-15	4.43E-14	2.56E-18	1.41E-16	3.11E-15	4.51E-14	1.84E-12		9.91E-03	1.66E-03	2.23E-03	1.99E-04	1.66E-03	2.23E-03	6.74E-03
Bk250	7.22E-31	1.40E-28	1.46E-25	1.99E-24	1.91E-21	2.80E-19	6.40E-17	1.91E-21	2.35E-19	2.64E-18	6.35E-17	3.66E-21	2.03E-19	4.45E-18	6.46E-17	2.63E-15		1.42E-05	2.44E-06	3.28E-06	2.94E-07	2.45E-06	3.28E-06	9.88E-06
Bk251	1.82E-35	3.53E-33	3.67E-30	4.92E-29	4.81E-26	7.04E-24	1.61E-21	4.81E-26	5.92E-24	6.65E-23	1.60E-21	9.21E-26	5.10E-24	1.12E-22	1.63E-21	6.62E-20		3.58E-10	6.16E-11	8.26E-11	7.39E-12	6.16E-11	8.25E-11	2.49E-10
Cf249	#######	5.54E-27	6.48E-24	8.80E-23	9.26E-20	1.58E-17	3.92E-15	9.34E-20	1.25E-17	1.54E-16	3.89E-15	1.88E-19	1.09E-17	2.64E-16	3.96E-15	1.82E-13		5.00E-02	4.31E-04	7.00E-04	1.18E-01	4.31E-04	7.00E-04	2.42E-03
Cf250	3.18E-29	1.76E-26	2.03E-23	2.76E-22	2.89E-19	4.86E-17	1.20E-14	2.91E-19	3.88E-17	4.71E-16	1.19E-14	5.88E-19	3.35E-17	8.09E-16	1.21E-14	5.50E-13		6.85E-02	2.85E-02	1.90E-03	3.14E-02	2.84E-02	1.90E-03	6.26E-03
Cf251	2.15E-35	2.77E-27	3.53E-24	4.85E-23	5.38E-20	1.02E-17	2.69E-15	5.46E-20	7.81E-18	1.01E-16	2.67E-15	1.15E-19	6.78E-18	1.77E-16	2.72E-15	1.36E-13		8.13E-02	4.93E-02	6.79E-03	2.84E-02	4.93E-02	6.78E-03	3.57E-03
Cf252	#######	1.70E-28	4.00E-25	5.60E-24	6.68E-21	1.48E-18	4.25E-16	6.85E-21	1.07E-18	1.51E-17	4.21E-16	1.54E-20	9.31E-19	2.68E-17	4.29E-16	2.41E-14	...	1.86E-01	2.54E-01	2.53E-01	4.45E-02	2.54E-01	2.53E-01	6.26E-01

PWRUOX					
UOX	Calculation results				
Loading	1Cycle	2Cycle	3Cycle	4Cycle	
0.000E+00	1.108E-08	3.538E-08	9.338E-08	1.909E-07	
0.000E+00	7.227E-07	9.473E-07	9.561E-07	8.849E-07	
5.052E-02	4.017E-02	3.197E-02	2.560E-02	2.077E-02	
4.950E+00	3.003E+00	1.823E+00	1.108E+00	6.746E-01	
0.000E+00	3.529E-01	5.328E-01	6.113E-01	6.311E-01	
9.500E+01	9.395E+01	9.292E+01	9.190E+01	9.089E+01	
0.000E+00	4.712E-05	4.660E-05	4.609E-05	4.558E-05	
0.000E+00	1.095E-08	4.633E-08	9.001E-08	1.293E-07	
0.000E+00	1.514E-02	4.791E-02	8.161E-02	1.093E-01	
0.000E+00	3.289E-05	1.063E-04	1.806E-04	2.413E-04	
0.000E+00	6.783E-03	6.709E-03	6.638E-03	6.562E-03	
0.000E+00	2.274E-08	1.217E-07	2.846E-07	4.700E-07	
0.000E+00	2.667E-09	1.558E-08	4.100E-08	7.454E-08	
0.000E+00	1.974E-08	1.241E-02	3.251E-02	5.883E-02	
0.000E+00	4.523E-01	5.516E-01	5.715E-01	5.747E-01	
0.000E+00	8.760E-02	1.852E-01	2.296E-01	2.462E-01	
0.000E+00	6.641E-02	1.396E-01	1.917E-01	2.179E-01	
0.000E+00	8.085E-03	3.760E-02	8.055E-02	1.241E-01	
0.000E+00	1.682E-06	6.844E-06	1.487E-05	2.338E-05	
0.000E+00	9.120E-08	1.228E-06	5.278E-06	1.390E-05	
0.000E+00	1.140E-03	4.825E-03	9.071E-03	1.226E-02	
0.000E+00	1.378E-05	6.505E-05	1.283E-04	1.778E-04	
0.000E+00	2.271E-06	8.681E-06	1.607E-05	2.163E-05	
0.000E+00	6.127E-04	5.555E-03	1.749E-02	3.508E-02	
0.000E+00	1.710E-04	1.097E-03	2.453E-03	3.652E-03	
0.000E+00	1.406E-06	1.922E-05	6.007E-05	1.109E-04	
0.000E+00	6.726E-05	1.382E-03	6.735E-03	1.854E-02	
0.000E+00	1.881E-06	7.527E-05	4.895E-04	1.618E-03	
0.000E+00	4.683E-08	4.472E-06	4.633E-05	2.171E-04	
0.000E+00	1.928E-10	4.002E-08	6.290E-07	3.919E-06	
0.000E+00	3.895E-12	1.790E-09	4.501E-08	3.890E-07	
0.000E+00	1.862E-14	1.374E-11	4.266E-10	4.134E-09	
0.000E+00	2.660E-17	1.968E-14	6.115E-13	5.925E-12	
0.000E+00	6.688E-22	4.959E-19	1.542E-17	1.493E-16	
0.000E+00	1.501E-15	2.084E-12	9.007E-11	1.059E-09	
0.000E+00	4.617E-15	5.918E-12	2.398E-10	2.693E-09	
0.000E+00	9.690E-16	2.028E-12	1.030E-10	1.304E-09	
0.000E+00	1.406E-16	6.249E-13	5.033E-11	8.672E-10	
1.000E+02	9.799E+01	9.630E+01	9.488E+01	9.363E+01	

Figure 3: Example of the making flow diagram on depletion matrix



2.2. COSI6

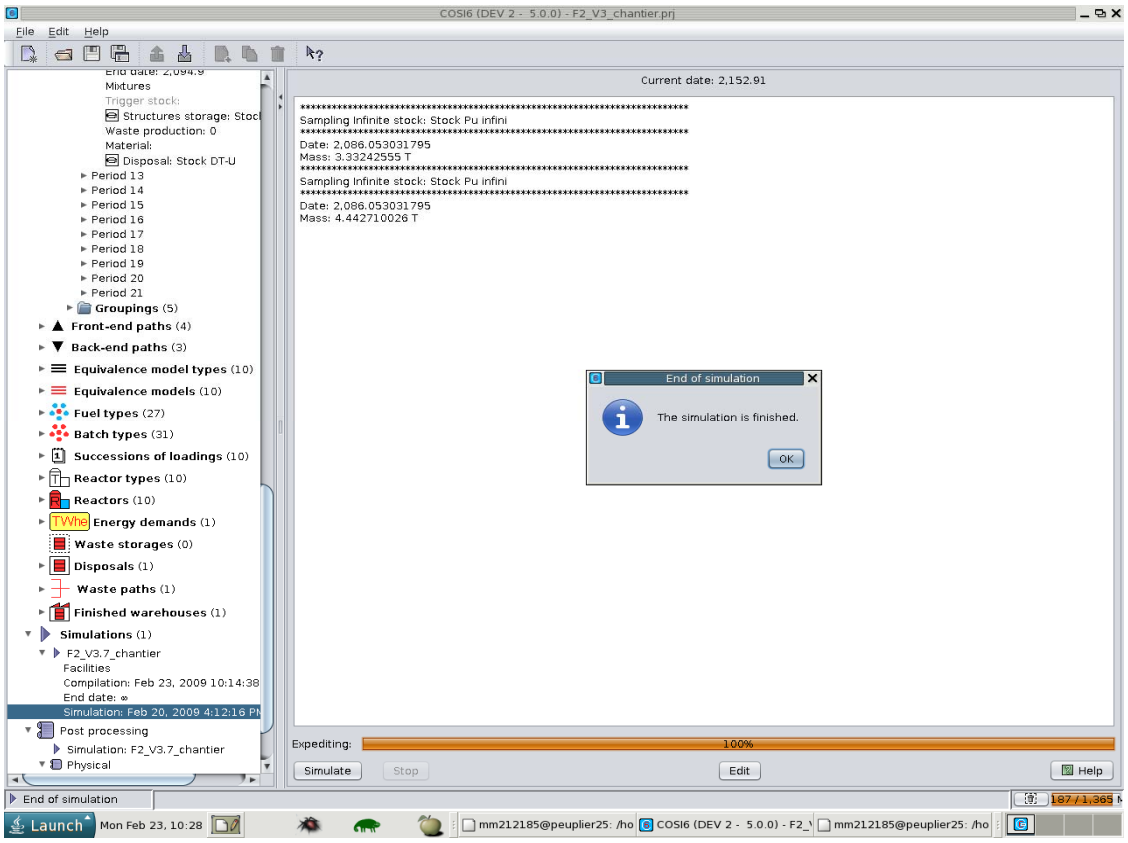
2.2.1. *Introduction and history*

The insertion of a new concept must be evaluated in the global electronuclear system with an analysis of the impact on the fuel cycle (enrichment, fuel fabrication, reactor, processing, interim storage, waste storage, and cost of cycle). This concept can be a new type of fuel, a new strategy for the management of plutonium or minor actinides, or a new type of reactor. Therefore, the Nuclear Energy Direction at CEA (the French Atomic Energy Commission) has developed the software named COSI. It simulates a pool of nuclear power plants with their associated fuel cycle facilities.

This code has been designed to study various short- medium-and long-term scenarios for the introduction of various types of nuclear reactors and for the use of associated nuclear materials, with due consideration to isotopic composition.

The COSI6 user interface is shown in Figure 4.

Figure 4: COSI6 Graphical user interface



2.2.2. *COSI6 capabilities*

The simulation that can be performed with the COSI code is shown schematically in Figure 4, which includes:

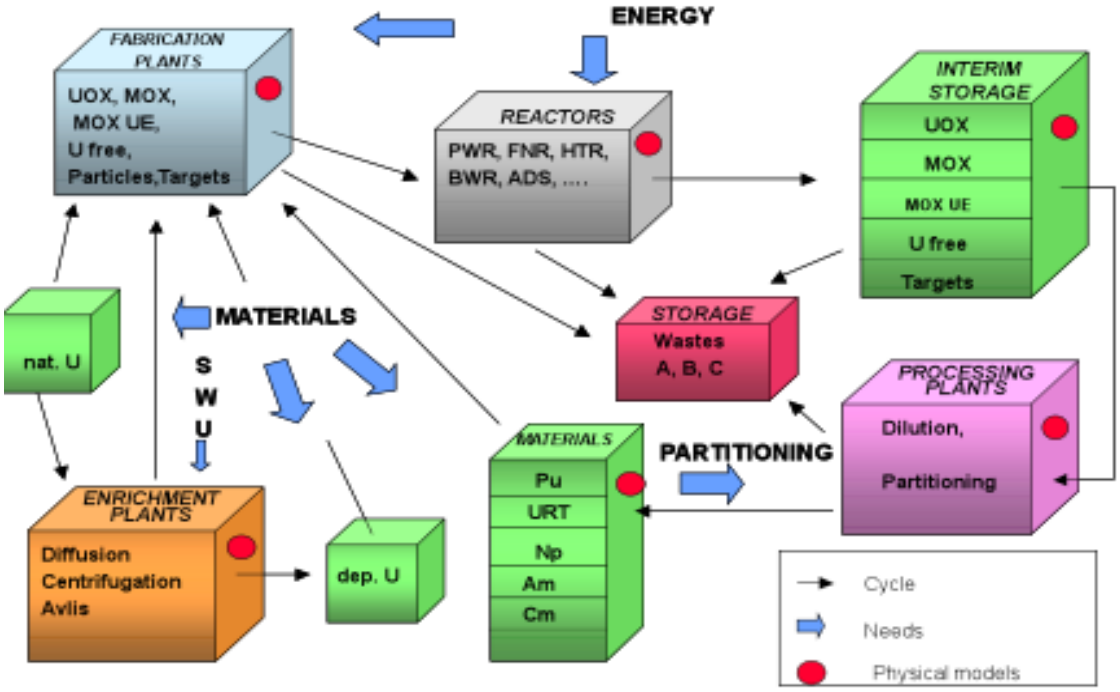
- the facilities of the fuel cycle (mines, enrichment, fabrication facilities, reactors, processing facilities, stockpiles, waste storage, geological disposal);
- the input data for the simulation (energy demand, fuel and nuclear materials requirements), described by the thick arrows;
- the transfers of nuclear materials, described by the thin arrows;
- the steps for which the change in the isotopic composition of the fuel is taken into account (irradiation, cooling time, aging time) using physical modelling, described by the full circles.

A simulation scenario is mainly described by the date of commissioning or decommissioning of the reactors and is based on the reactors' needs of fertile and fissile materials in order to produce the required electricity. The results of the back-end fuel cycle parameters (irradiated stockpiles, reprocessing mass flow, plutonium, uranium from reprocessing, wastes, etc.) are calculated. The required front-end fuel cycle parameters (fuel elements, nuclear materials, etc.) complete the cycle description.

The COSI code permits exploration of different electronuclear scenarios involving:

- a pool of reactors: light-water reactors (LWR), sodium-cooled fast reactors (SFR), high-temperature reactors (HTR), gas-cooled fast reactors (GFR), accelerator-driven systems (ADS);
- the entire fuel cycle facilities;
- the different types of fuels.

Figure 5: COSI6 modelling scheme



The detailed analysis of nuclear materials in COSI takes into account the isotopes of each element:

- U, Pu, and minor actinides isotopes if COSI is coupled with the evolution code CESAR4;
- U, Pu, MA and 200 fission products if COSI is coupled with the evolution code CESAR5.

The following constraints in the operation of the fuel cycle facilities can be selected:

- processing plant capacity in heavy metal and in plutonium;
- minimum cooling period prior to spent fuel processing.

The user can choose alternative possibilities for the processing of spent fuel:

- “first-in-first-out” or “last-in-first-out”;
- various types of dilution;
- partitioning of minor actinides.

COSI gives a detailed computation of the materials balances, including the computation of the plutonium content or ^{235}U enrichment entering in the fuel fabrication based on:

- the composition of the various batches of plutonium used;
- the origin of the uranium (natural, depleted or reprocessed uranium);
- the core management;
- the burn-up.

COSI can also assess the high-level and intermediate level waste: waste package inventory, isotopic compositions, activity, radiotoxicity and decay heat.

The evaluation of the economics of reactors and fuel cycle facilities can be made in order to obtain a cost for the kWh. The economic model of COSI takes into account:

- the investment, operations and decommissioning costs for each reactor and facilities, and their associated planning;
- the cost of nuclear materials : natural uranium, plutonium, etc;
- the actualisation rate.

2.2.3. Brief description of the methods used in COSI6

There is no algorithm that controls the deployment of fast reactors in COSI. The user chooses the reactor type for each reactor and the associated installed capacity. In the benchmark cases the reactors started are either FR or LWR, depending on the specification:

If there is insufficient TRU to fuel the selected FR installed capacity, the FRs are not fed in the simulation and the user has to change the installed capacity and launch a new simulation.

Concerning reprocessing capacity, there are three kinds of reprocessing plants:

- "fictitious" plants are able to reprocess as much fuel as required by the fissile material need (determined by the fabrication plants);
- "limited fictitious" plants are able to reprocess as much fuel as required by the fissile material need (determined by the fabrication plants), but takes also into account the maximum mass flow constraint given by the user;
- "real" plants reprocess the quantity of fuel specified by the user.

All plants are also limited by the spent fuel availability. Two physics models are used in COSI.

The CESAR model

For each fuel batch, the exact isotopic composition is taken into account by COSI. In COSI, the CESAR code is used for the in-pile calculations. It solves the differential equation system that describes the fuel evolution in pile.

The number of equations is identical to the number of the isotopes N (A, Z). For actinide calculation the equation is:

$$\begin{aligned}\left(\frac{dN(t)}{dt}\right)_{\left[\begin{smallmatrix}A\\Z\end{smallmatrix}\right]} &= \Phi(t).\left[\sigma_c(t).N(t)\right]_{\left[\begin{smallmatrix}A-1\\Z\end{smallmatrix}\right]} + \Phi(t).\left[\sigma_{n,2n}(t).N(t)\right]_{\left[\begin{smallmatrix}A+1\\Z\end{smallmatrix}\right]} \\ &+ \left[\lambda_{\beta^+}.N(t)\right]_{\left[\begin{smallmatrix}A\\Z+1\end{smallmatrix}\right]} + \left[\lambda_{\beta^-}.N(t)\right]_{\left[\begin{smallmatrix}A\\Z-1\end{smallmatrix}\right]} + \left[\lambda_{\alpha}.N(t)\right]_{\left[\begin{smallmatrix}A+4\\Z+2\end{smallmatrix}\right]} + \left[\lambda_{TI}.N(t)\right]_{\left[\begin{smallmatrix}A^m\\Z\end{smallmatrix}\right]} \\ &- \Phi(t).\left[\left(\sigma_c(t)+\sigma_f(t)+\sigma_{n,2n}(t)\right).N(t)\right]_{\left[\begin{smallmatrix}A\\Z\end{smallmatrix}\right]} - \left[\left(\lambda_{half-life}\right).N(t)\right]_{\left[\begin{smallmatrix}A\\Z\end{smallmatrix}\right]}\end{aligned}$$

For fission products and for some activation products, we can add a global fission yield, which is the sum of the fission rates of the fissionable actinides multiplied by the yields of the fission products for the fissionable actinides.

$$Y_{G\left[\begin{smallmatrix}A\\Z\end{smallmatrix}\right]} = \sum_{\substack{\text{fissionable}\\ \text{actinides}\\ j}} \gamma_{j\left[\begin{smallmatrix}A\\Z\end{smallmatrix}\right]} \tau_j$$

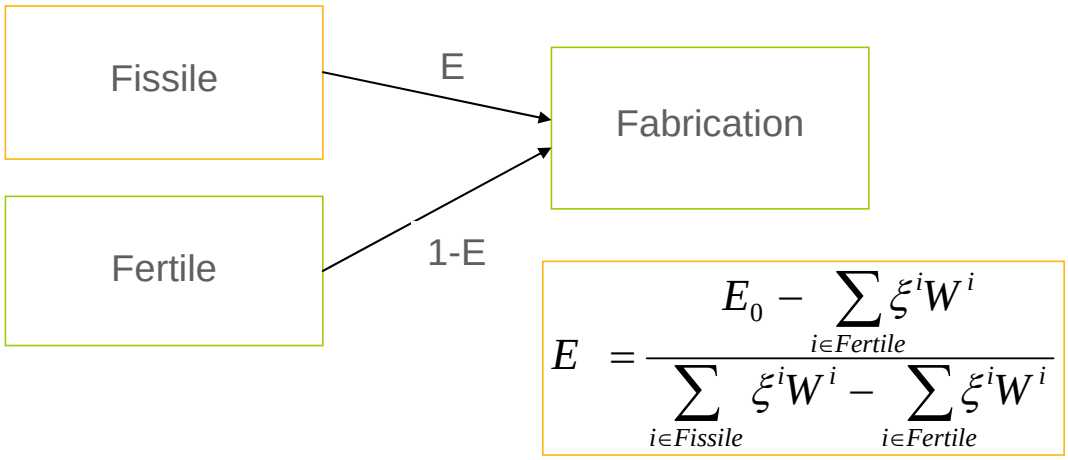
The Runge-Kutta method

Cross-section sets come from the reference calculation codes for neutronics: APOLLO for the thermal spectrum systems and ERANOS for the fast spectrum systems. The activation products data come from the European Activation File. These cross-section sets are given as a function of burn-up and initial ^{235}U enrichment, or initial Pu content in MOX fuel.

CESAR is validated against the French post-irradiation experiments (PIE) database.

The equivalence model for fuel based on uranium and transuranics, the initial TRU fraction is calculated by taking into account the TRU composition and using an adapted formulation expressed in equivalent ^{239}Pu. The formula takes into account the reactivity effect of each isotope of U, Pu and minor actinides.

Figure 6: COSI6 Equivalence model for fast reactors



The parameters of the model are:

- ^{239}Pu equivalent rate (E_0);
- reactivity weights Wi for fissile and fertile materials.

The isotopic compositions ξ^i for fissile and fertile materials are taken from the simulation.

The result of the equivalence model is the fissile content in the mixture: E.

There are other formulations for thermal neutron spectrum reactors.

2.3. EVOLCODE

2.3.1. Introduction and history

The potential benefits of the partitioning and transmutation (P&T) strategy are based on the development of new concepts in nuclear systems, such as fast reactors (FR) or sub-critical systems (ADS) coupled to an external source of neutrons. From the beginning of the studies of these new systems, the computer simulation has been the most widely used tool due to the lack of experimental facilities or demonstrators giving precise experimental values of the neutronic characteristics of the system. At the same time, large developments in the available simulation systems were crucial to handle as many requirements as possible of the problems studied in the field. Indeed, in this kind of problems, a special simulation code combining neutronics and isotopic evolution calculation was needed due to the specific necessities of the studies, in particular, the high fuel burn-up and corresponding large variations in isotopic compositions and the multiple recycling of materials in the cycle.

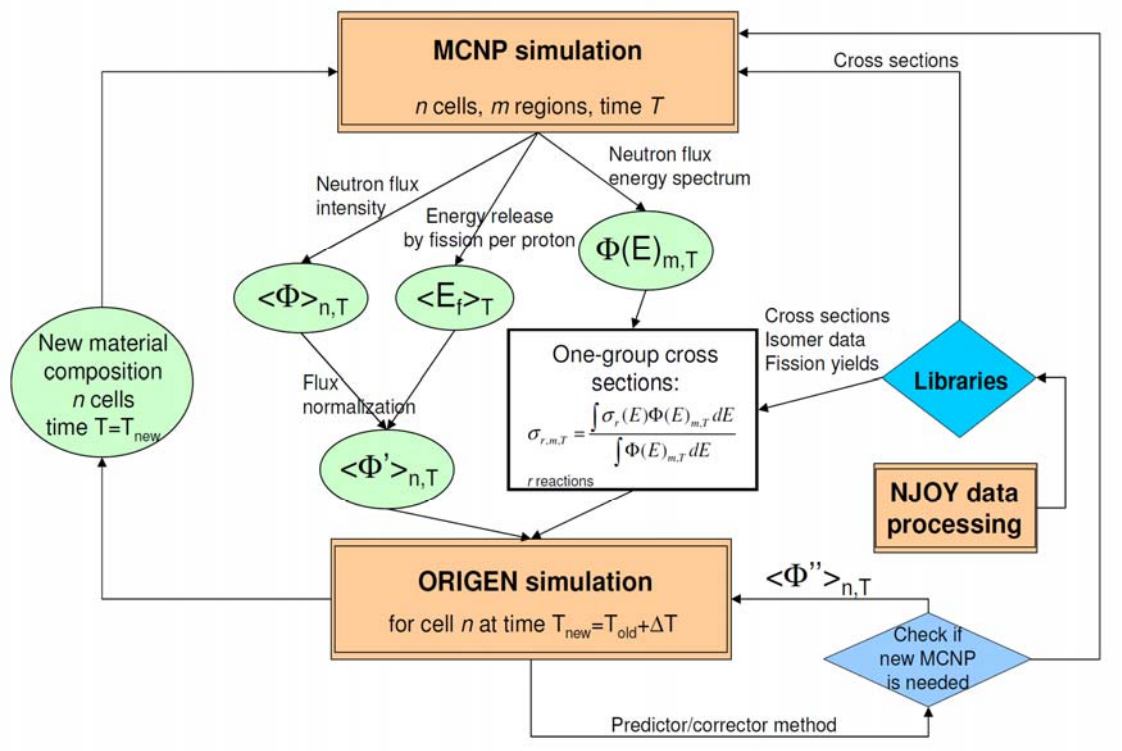
In 1997 a new group was setup at CIEMAT to study advanced fuel cycles and the role of ADS in those cycles. In order to address these aspects, CIEMAT developed the first version of a coupled transport-depletion code, the EVOLCODE system, published in 1999 [1]. Since then, CIEMAT has been continually validating, updating and upgrading the code from the experience gained in participations in international benchmarks [2] and projects organised by international organisations such as NEA [3], IAEA [4], UE (RED-IMPACT) [5], EUROTRANS [6], PDS-XADS [7] and other applications SAD [8].

In the present version of the code, EVOLCODE2.0 [9], the neutron transport calculations are implemented by the general Monte Carlo N-Particle Transport Code MCNPX [10]. The Isotope Generation and Depletion Code ORIGEN [11] currently implements the isotopic burn-up evolution of the geometry zones, requested by the user, although the modern depletion code ACAB [12] has been implemented in the EVOLCODE2 system to provide results without uncertainties and to enlarge the number of nuclear reactions taken into account by the depletion calculations. The complete substitution of ORIGEN is planned for the near future. EVOLCODE2 automatically links all the information produced by these codes in order to perform a detailed simulation, properly exchanging the information required by them.

2.3.2. EVOLCODE2 capabilities

The scheme of the EVOLCODE2 iterative procedure, the so-called “main” cycle, is shown in Figure 7. The cycle begins with the calculation by MCNPX of the neutronic parameters (power, fluence, spectra) of the initial configuration, which determine the evolution of the reactor characteristics. After processing this information, EVOLCODE2 calculates the reaction rates and the corresponding collapsed cross-sections for the evolving cells (requested by the user). Considering a certain interval of irradiation time (the step duration) and using the assumption of constant neutronic parameters during irradiation, ORIGEN calculates the burn-up evolution of the isotopic composition of each evolving cell. The burn-up evolution of the whole reactor is then estimated as the union of the burn-up evolution of the different evolving cells. With the new reactor composition available after a partial irradiation, a new step of EVOLCODE2 is performed, obtaining the new neutronic parameters of the system and afterwards the evolution of the materials isotopic composition for successive partial irradiation steps. In this way, the full irradiation is simulated after several cycles and the final solution is estimated as a piecewise-defined function for both neutron flux and material isotopic compositions.

Figure 7: EVOLCODE2 cycle data flow scheme



The main capabilities that make EVOLCODE2 a reference transport-depletion code are described below.

- Reaction rates (and cross-sections) are calculated by EVOLCODE2 outside MCNPX convoluting the cross-section data taken from the libraries by the neutron flux energy spectrum. Although MCNPX could calculate online these values in the most accurate way (as long as it works with adequate libraries) since it computes the exact value of the neutron energy for each nuclear reaction, the memory/CPU time requirements would be very demanding. This is because of the high number of operations needed to evaluate the cross-sections of the available nuclear reactions required for hundreds of isotopes, for every evolving cell of the nuclear system, and with a sufficient number of collisions per cell. This can force the user to define a smaller number of cells (and hence geometrically larger cells) in the MCNPX input, losing precision in the isotopic burn-up evolution. The energy spectrum of the neutron flux required by EVOLCODE2 can be obtained from MCNPX dividing the energy range in iso-lethargy energy bins in such a way that all the energies, for which the cross-section information is available, are covered. As an example, in the case of a simulation with 100 cells and 300 isotopes, the calculation of the cross-sections is around 30 times faster in EVOLCODE2 (using 80 000 iso-lethargy energy bins) for the same statistical accuracy.
- The difference between the values of the one-group cross-sections obtained from both methods is small (well below 1%) if the MCNPX statistics are good enough for a sufficiently detailed binning. The reason for these low deviations is that the second method uses the same linear interpolation that MCNPX in the neutron flux energy spectrum: the number of energy bins can be as large as the number of pointwise cross-sections values in the database (selected to describe the cross-sections better than 1% accuracy with a linear interpolation). For typical cases, the number of cross-section entries is about 77 000 for ^{238}U and about 50 000 for ^{239}Pu

and the iso-lethargy energy binning can be set to 80 000 bins for the range between 10^{-9} eV and 20 MeV.

- EVOLCODE2 includes a predictor/corrector method to optimise the cycle length, guaranteeing that the variation of the neutron flux is limited to values compatible with the specified precision in the materials evolution. The method reduces the cycle length whenever the variation of the thermal power during irradiation is larger than the limit fixed by the user. The use of the predictor/corrector method avoids the need for setting, a priori, unnecessary short cycle lengths controlling the CPU use but maintaining the required precision.
- The number of isotopes that EVOLCODE2 can handle is only limited by the availability of database information for each particular isotope. Every possible reaction and decay chain (if available in ORIGEN/ACAB) is treated. For an optimum use of this information, the isomer production reactions are also treated since they may have very different half-lives and reaction cross-sections compared with the ground state isotope, leading to different transmutation chains and possibly having a considerable impact on neutronics or waste management issues. EVOLCODE2 specifically manages the isomer production data for reactions (n,涤γ)* and (n,2n)*. The same nuclear reactions are allowed for ground and isomer states of an isotope.
- The cross-sections libraries created by EVOLCODE2 for the different ORIGEN executions include the information about the fission product yields of fissionable isotopes. ORIGEN considers eight isotopes as explicitly fissionable species (^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{241}Pu, ^{245}Cm and ^{252}Cf) with the rest of the fissionable species adjusted to a nearby actinide with explicit fission yields, called "nearest connected actinide". In the ENDF format database, a maximum of three different nuclide-yield lists are given as a function of the determined energy of the incident neutron. These energies are 0.0253 eV for thermal spectrum, 500 keV for fast neutron spectrum and 14 MeV for fusion spectrum fission yields. EVOLCODE2 obtains the precise values of the fission product yields as a new nuclide-yield list interpolating in energy, according to the library rules included in the fission yield libraries, for the cell averaged neutron energy at fission.
- MCNPX requires a complete description of the physics involved in the neutronic interactions (the neutron transport data) in order to give accurate results, although this description is nevertheless only available for a certain number of isotopes and elements. For this set of isotopes, EVOLCODE2 is pushed to use the same libraries with the aim of ensuring consistence. For the rest of the isotopes, EVOLCODE2 reads and computes the activation databases, which contain the cross-sections information of activation reactions. These databases are not always usable for transport by MCNPX but include the necessary information for EVOLCODE2 to make the proper convolutions of isotopes without transport data. Therefore, the ORIGEN depletion calculation uses a longer isotope list than the MCNPX transport. According to this, the ORIGEN isotope information must dominate over the MCNPX material description in order not to lose precision. After the ORIGEN calculations, the new MCNPX input file is created updating the material compositions but excluding those isotopes without neutron transport data. The excluded MCNPX isotopes will be included in the ORIGEN isotopic composition description of the next cycle, so the precision loss is minimised.
- The propagation of cross-section uncertainties has proved to be an essential feature in order to discover the precision of the results due to the basic data libraries. The depletion code ACAB has been successfully implemented into the EVOLCODE2 system to provide the propagation of cross-section uncertainties from the basic database information in the inventory of actinides and fission products. The impact of cross-section uncertainties on relevant fuel cycle parameters can

also be calculated using this tool. This tool also has the advantage of taking into account some different nuclear reactions that may be relevant for the isotopic evolution in new nuclear systems operating in different neutron spectra.

Figure 8: General fuel cycle representation of buffers and mass flows in TR_EVOL

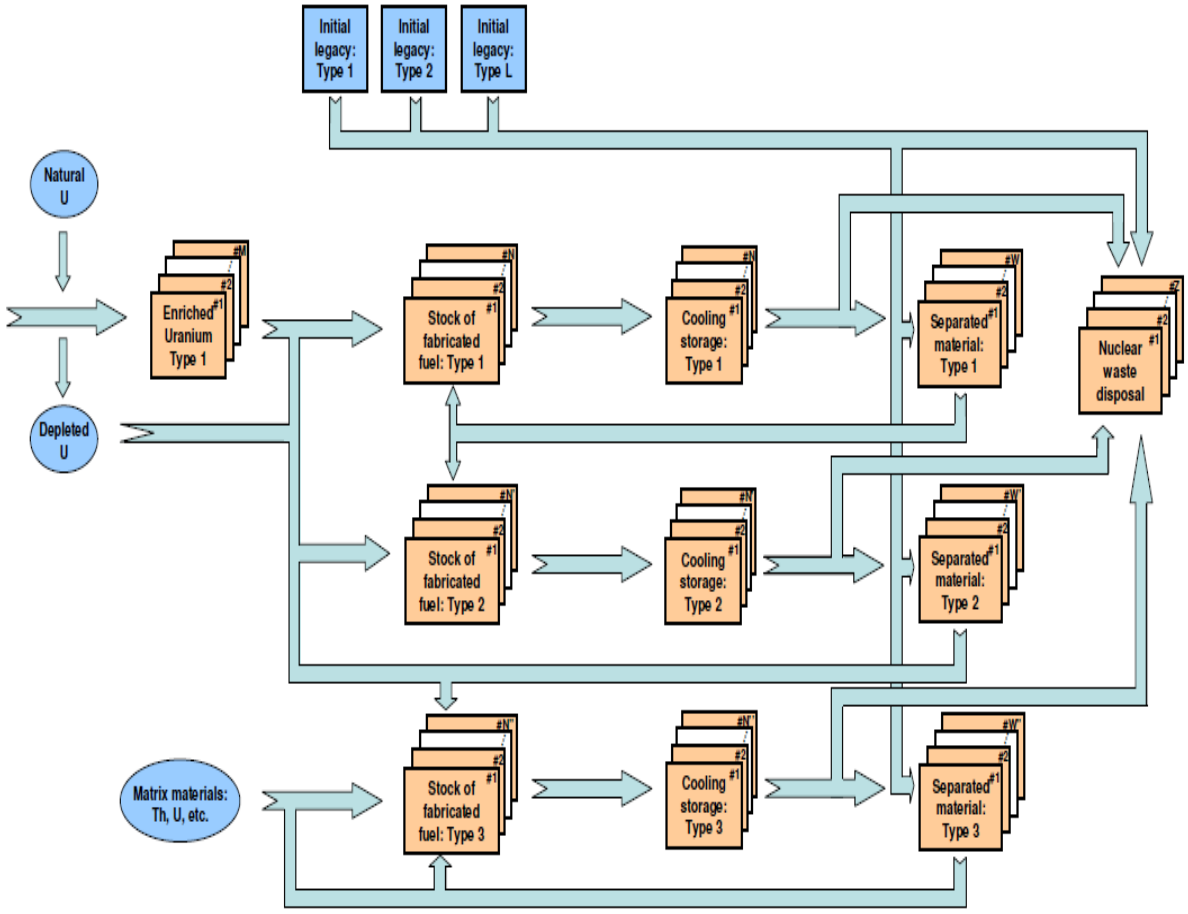
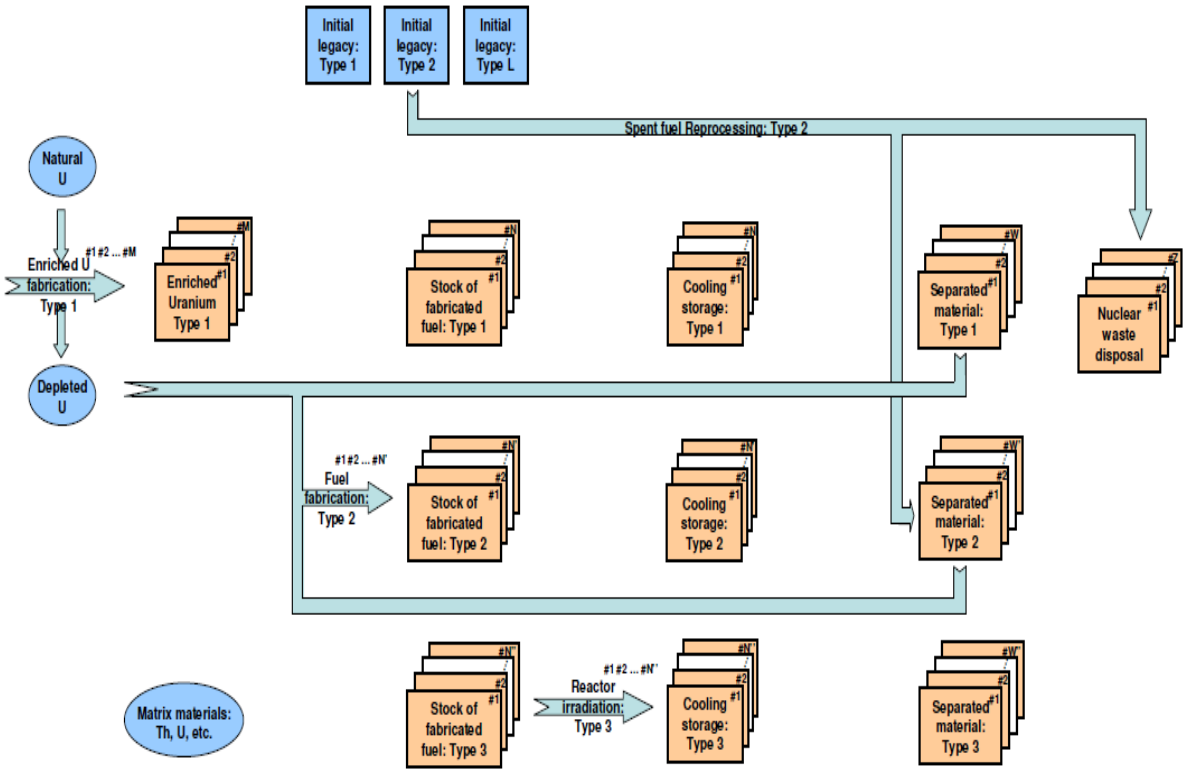


Figure 9: Representation of operational rules in TR_EVOL



2.3.3. *Upgrading of capabilities for transition fuel cycle scenarios*

The EVOLCODE2 nuclear system has been upgraded with the attachment of a new module to make transition fuel cycle calculations. The transition evolution module (TR_EVOL) is intended to estimate the mass balance of every stream (fresh fuel for nuclear reactors, reprocessed material, etc.) defined in any user-defined fuel cycle component as a function of time, without any limit for the number of streams or facilities.

TR_EVOL data treatment is made using buffers representing material in storage. Each buffer contains the isotopic vector and the total amount of material present in that storage. Figure 8 shows a general fuel cycle including three different types of reactors. Storage facilities taken into account in this fuel cycle (others could be included when necessary for other cycles), represented as brown boxes in the figure, are:

- enriched uranium;
- fresh fuel for nuclear reactors;
- spent fuel in cooling storage;
- separated material from reprocessing and
- nuclear waste.

Each fuel cycle storage facility can be represented by one or several different buffers. For instance, the Type 1 nuclear park might consist in a series of N PWR with different ^{235}U enriched fuels. Hence, data concerning fresh fuels with different enrichments would be stored in N different buffers.

Connections between buffers represent mass flows and are represented by light blue arrows. In the figure, some possible connections have been removed for simplicity, such as connections from uranium enrichment, fuel fabrication or fuel irradiation to the nuclear waste buffer. External material is shown in the figure as dark blue boxes and includes the initial legacy of spent fuel, natural or depleted uranium (not managed in the fuel cycle) and other materials such as the fertile or inert matrix for target fabrication. Connections can flow from one buffer to another, but can also join other connections or divide towards different buffers.

The parameters of the cycle facilities and the time-dependent interconnections are described in TR_EVOL using a series of basic operational instructions or rules. Each rule specifies a particular action that is applicable to a particular buffer (decay of stored material) or to a particular interconnection (fuel irradiation, fuel fabrication, reprocessing, etc.). The period of time for which that particular action is active is also specified (for instance, advanced reprocessing may be only applicable from a certain year on). Figure 9 shows some examples of operational rules. Typical rules shown in the figure are:

- The fabrication rule: material for the fabrication of nuclear fuel is taken in this fuel cycle from different buffers, such as separated material type 1 and separated material type 2 (considering one or more different elements from each material type) and external material. A different rule is needed for each different fabricated fuel, represented by each different buffer N' in the figure.
- The irradiation rule (for type 3 reactors in the example) connects two general buffers, 'fabricated fuel' and 'cooling storage'. Each general buffer can be divided into different buffers in case reactors use fuels with different enrichment or technology (UOx vs. MOX or breeder vs. converter fast reactors).

Some other details have to be included for completing the information of the rule, for instance, the amount of material to be reprocessed (and previously extracted from the cooling storage buffer) in the case of a reprocessing rule. All information concerning buffers and rules is provided in a TR_EVOL input file. Other general information of the fuel cycle scenario has to be provided, including the burn-up of the reactors and the energy generated by each reactor park during the operation period.

The main capabilities and methods used by TR_EVOL module include:

- The mass and the isotopic composition balances of each stream defined in the fuel cycle are calculated as a function of time.
- Buffers and rules can be active for the whole duration of the fuel cycle or only for a limited duration. The user specifies the time validity of each buffer or rule in the input files.
- Any kind of nuclear facility can be simulated in this module, since its operation can be described as a set of buffers plus a series of rules.
- The irradiation and decay of the nuclear materials is simulated through ORIGEN executions. Hence, the Bateman equations are used for depletion calculations, taking into account the reaction and decay chains available in ORIGEN. The isomer production data for reactions (n,γ)* and (n,2n)* are also managed by this module, allowing the same nuclear reactions for ground and isomer states of an isotope. Fission product yields can also be taken into account in the irradiation simulations.
- The number of isotopes that TR_EVOL can handle is only limited by the existence of database information for each particular isotope. The isotopic information is always preserved for informational purposes, while results can be presented at the elemental level.

- Any kind of nuclear reactor (with any fuel) can be considered, although TR_EVOL implements different approximations for the calculation of the mass balances depending on the available information of the reactor. On one hand, streams coming from an advanced nuclear reactor (such as ADS or FR) require the fully detailed study of the system using EVOLCODE2 in order to provide an adequate set of cross-sections (for ORIGEN) representative of the advanced nuclear reactor park. The simulation is implemented using the isotopic composition of a representative situation in time. Neutron fluence and spectrum are averaged over the whole reactor for better representativeness. For more standard nuclear reactors, the ORIGEN reference reactor libraries can be also used, avoiding the necessity of a whole 3D reactor design (this is usually the case of the LWR).
- In addition, a nuclear reactor can also be represented by more than one set of cross-sections for both temporal and geometrical reasons. For instance, large changes in the isotopic composition with time may lead to better representativeness if more than one cross-section set is used and blankets in fast reactors may have a different neutron spectrum than the driver and hence need a different set of cross-sections.
- The fuel fabrication process can use the concept of equivalent ^{239}Pu if required. This concept is used in the simulation of fuel fabrication where different isotopes of uranium and plutonium (and minor actinides) come from different buffers. It gives a reactivity worth to each isotope present in the mix. Then the ratio of material coming from each buffer is calculated for the total reactivity of the fuel to match the reactivity of equivalent ^{239}Pu, which is fixed beforehand. If this concept is not necessary, a fixed ratio between uranium, plutonium and minor actinides can be used instead.
- Reprocessing is simulated as a set of coefficients of element recovery. The rule is applied to a certain amount of material extracted from a buffer. Recovered material is sent to one or more separated material buffers and the reprocessing losses can be sent to a nuclear waste buffer. Each type of technology can be then simulated by this procedure, with the same or different recovery coefficients.
- Spent fuel in a cooling storage buffer can be saved in two different ways. First, it can be homogenised but the information about the introduction date in the buffer is lost. Second, material can be stored with the date information for the simulation of a reprocessing strategy of first-in-first-out.
- Nuclear waste can appear as a by-product of different rules such as reprocessing or fuel fabrication. In addition to actinides, these buffers can contain fission products (and hence intermediate level waste) since the information of more than 3 000 isotopes is taken into account in the simulations. Some post-processing of data is needed for the creation of waste packages.
- Nuclear waste can decay or not, depending on the user's choice. Nuclear waste decay would, however, constrain the option of post-processing the results for the creation of waste packages.
- TR_EVOL allows the introduction (at any time of the fuel cycle) of external material such as an initial legacy of spent fuel from past generations or special material for fuel fabrication (possibly material for target matrices or natural uranium).
- Results are provided per year. Hence, large fluctuations of operational parameters on individual cycle facilities are averaged over the year assuming that the nuclear park is large enough.
- For results presentation, buffers can be managed in elements or groups (such as minor actinides).

- Execution time for a complex fuel cycle is on the order of half an hour (plus the required EVOLCODE2 simulations to evaluate the one-group cross-sections for the different reactors).

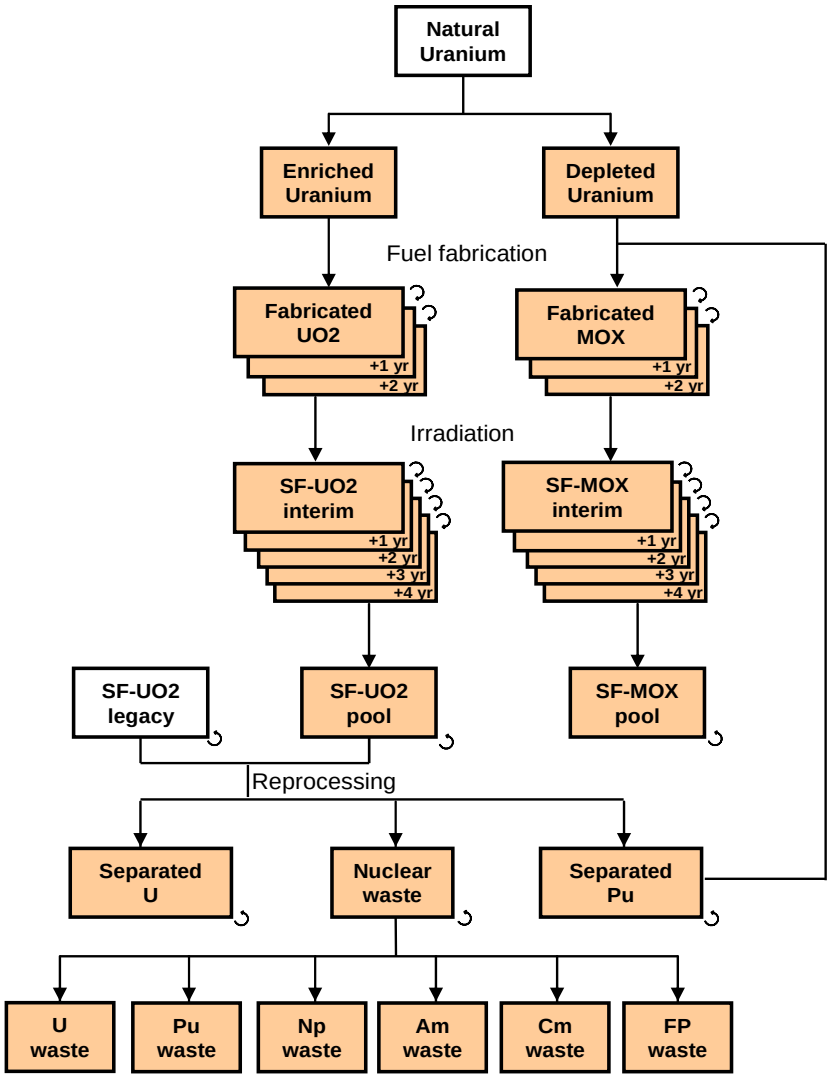
EVOLCODE2 (TR_EVOL) has been used to calculate the evolution of the transition fuel cycle scenarios of the present benchmark, particularly scenario 2 (Monorecycling of Pu in PWRs) and scenario 3 (Monorecycling of Pu in PWRs and deployment of Generation IV fast reactors). The implementation of scenario 2 included 37 buffers and 38 rules (19 for buffer decay, five with reprocessing purposes, two for irradiation, six for fuel fabrication, one for initial legacy of spent fuel, and five for informational purposes). While the detailed description of scenario 2 is included in the forthcoming chapters, the implementation of this scenario, displaying the different buffers is shown here (Figure 10). The ORIGEN reference reactor libraries have been used for the PWR with enriched uranium and with MOX fuel. The TR_EVOL module execution took 31 minutes.

Scenario 3 calculations included 62 buffers and 67 rules (32 for buffer decay, 14 for reprocessing of spent fuel, five for irradiation, ten for fuel fabrication, one for initial legacy appearance and five for informational purposes). For the simulation of irradiation in PWRs with enriched uranium and with MOX fuel, the ORIGEN reference reactor libraries have been used. For the simulation of the FR, two different one-group cross-section libraries have been created by means of a detailed EVOLCODE2 simulation: one for the FR core (averaged over the whole driver) and the other for the FR axial and radial blankets. The TR_EVOL module execution took 33 minutes.

For the simulations of these fuel cycles, some simplifications have been used:

- The minimum time unit for irradiation and decay is one year. Shorter simulations are not allowed in the current version of the module.
- The first and the last cores are not simulated in this benchmark. However, it is possible to manage these cores in TR_EVOL if required.
- The fact that operational parameters on individual cycle facilities are averaged over the year has some impact on the energy generation of a nuclear reactor park. Since it is supposed that the nuclear park is large enough, at every time there will be reactors in operation and in the refuelling stage. This is implemented considering that the total cycle length is equal to the sum of the equivalent full cycle length plus the refuelling period (cycle length/load factor). The generated energy per year for a particular reactor is equal to the total energy generated divided by the total cycle length (in years).
- In addition, TR_EVOL manages the amount of fuel to be fabricated, irradiated and stored for cooling, etc. by year instead of by core. In this sense, strategies of refuelling are also averaged.
- Quantities provided by TR_EVOL are based on actinides (and fission products). For instance, UO_2 fuel fabrication is managed only as uranium and not as a mixture with oxygen.
- Instantaneous rules, such as the extraction of material from a buffer for reprocessing, are considered to be done in TR_EVOL on the 1st of January to be consistent with other codes in the benchmark, although the result is shown by year. If the extraction is considered to be done, for instance, on the 31st of December of the previous year, the decrease in the amount of the buffer material would occur in the previous year and a phase shift of one year would appear in results versus other codes or calculations.

Figure 10: Scheme of the implementation of Scenario 2 with TR_EVOL, including 31 active buffers plus six management buffers



Finally, in all the calculations, we have applied the fuel cycle hypotheses of the benchmark in a very rigorous manner. Indeed, this was important as we have observed that the results are very sensitive to the strict implementation of the hypotheses.

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2.4. VISION

2.4.1. Introduction and history

The verifiable fuel cycle simulation (VISION) model is the primary dynamic simulation model used for systems studies for the US Fuel Cycle Technologies (FCT) Programme and its predecessor, the Advanced Fuel Cycle Initiative (AFCI). VISION is used to perform dynamic scoping trade studies of alternative fuel cycles to obtain qualitative and quantitative comparisons of resource requirements, reactor types and mix, sequencing and timing, waste streams and geologic repository requirements, with a capability to provide cost estimates of levelized cost of electricity and cash flow/funding requirements. The model provides a number of parameters for the comparison of fuel cycle options, including repository capacity and performance, separation capacity, interim storage, energy recovery, proliferation resistance and safety. Specific waste parameters include waste mass, wasteform mass, wasteform volume, long-term radiotoxicity and long-term heat commitment to a repository.

In 2003 the AFCI Systems Analysis campaign reviewed current systems codes and adopted the existing DYMOND code developed by Argonne National Laboratory as the starting point for the systems code. The existing elemental mass flow model was significantly extended by a multi-laboratory group led by the Idaho National Laboratory, including tracking mass by isotope, addition of waste and economic modules, and inclusion of numerous new algorithms and indicators. The VISION name was adopted

when the code exceeded the memory limits of the underlying software (Stella) and the model was reorganised and ported to PowerSim, which provided better support for the multiple arrays used to track isotopic data and other fuel cycle attributes. The current model is used by national laboratories and universities to conduct fuel cycle studies.

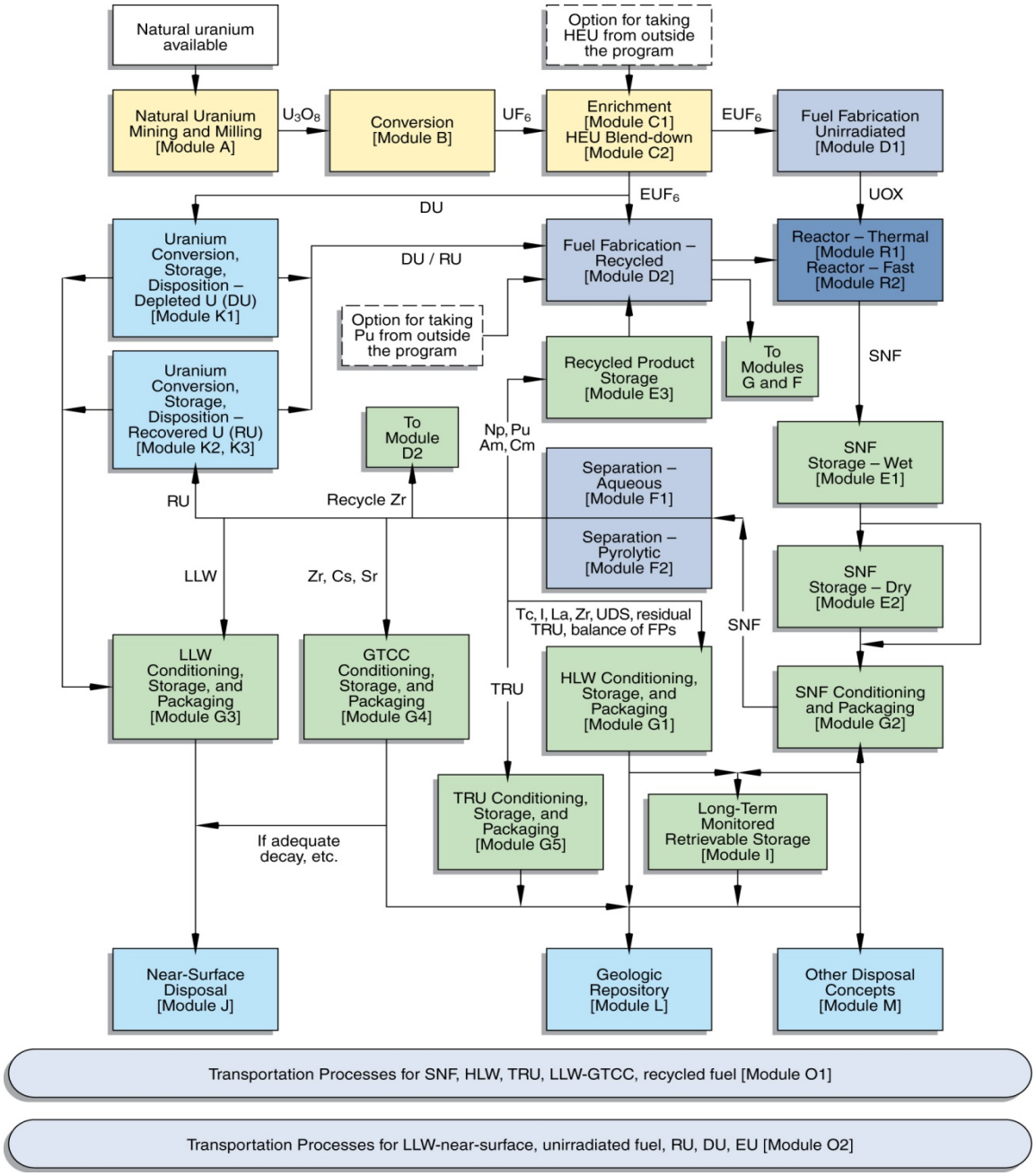
2.4.2. VISION capabilities

The full VISION implementation includes a series of Excel input and output files interacting with the core VISION model running in PowerSim. Input files include an extensive fuels library, model initialisation information for the US legacy system and a primary user file that includes a large number of standard fuel cycle cases (which the user may modify) along with 5 user-definable cases. The core model is designed as a full system dynamic model including feedback and logic that allows the model to forecast and make decisions to evolve all aspects of the nuclear infrastructure based on minimal high-level direction such as an overall growth rate. The user may override this logic as desired via the input file, specifying everything from individual reactor builds to split fractions of materials in separations. However, the underlying logic will still verify user direction and override that direction when physical constraints are exceeded (such as requesting more reactors that can be fuelled). Output Excel files provide an extensive range of parameters along with automatic graphing of both individual cases and comparisons between up to 5 cases. The users may develop their own output file containing specific graphs, etc. via linkage to the provided files. VISION is based on a modular view of a generic nuclear infrastructure, including reactors and front- and back-end facilities as depicted in Figure 11. The emphasis is on the back-end of the fuel cycle, as that is the primary focus of FCT R&D efforts.

The code logic is built around a series of backbone structures that model core features of the system. These include a fuel cycle mass flow structure, a facility life-cycle structure for reactors, storage, fabrication, separations, and disposal facilities, and the reactor park. These structures provide quarterly and annual capacity and mass flow information that form the basis for the numerous parameters that are calculated for the user and for internal control of the model. VISION does not explicitly model individual front-end facilities other than fuel fabrication, as mining, conversion and enrichment are well-established commercial ventures that operate based on market conditions and modelling of total fleet capacities and throughputs are sufficient for calculating indicator values. Back-end facilities are explicitly modelled and the user can specify any level of control desired from only indicating when a technology is first available for use to specifying the specific capacity and construction timing of each facility.

VISION uses a fuel cycle library reference to calculate transmutation performance for each reactor/fuel combination. The model adjusts the pre-calculated fuel “recipes” to reflect the actual isotopic and fissile content of feed materials. The model logic supports both one-tier and two-tier recycle systems, with different fuel, reactor, and separations technologies for each stage (e.g. thermal reactors with first-run material, thermal reactors with recycled material, fast reactors). The user can specify how many times fuel is recycled at each stage (from zero to unlimited). Fuel recycle compositions are modelled reflecting evolution of isotopic content through up to five cycles, at which point the fuel is considered to be at an equilibrium composition. Start-up and shut-down fuel loads for individual reactors are not modelled explicitly at this time. Discharged material is decayed to reflect isotopic changes while in storage prior to disposal or reuse. Separated materials are not decayed, because they are assumed to be reused shortly after reprocessing. Waste materials can be decayed via post-processing analysis, but are not decayed within VISION because the important decay times for waste streams are typically much longer than the length of simulations.

Figure 11: Fuel cycle modules used in VISION



VISION tracks 81 separate groups of isotopes. All isotopes potentially important as fuel or target materials are tracked individually, while lighter isotopes coming from decay or generated directly as fission products may be grouped based on behaviour during reprocessing. Detailed assessment of reprocessing and waste management options require identification of key isotopes and residual masses for Group 1A/2A elements (Rb, Cs, Sr, Ba), inert gases (Kr, Xe), halogens (Br, I), lanthanides, transition metals, transuranic (TRU), uranium, actinide decay products. Tracked isotopes include almost all actinides and fission products with half lives of more than 0.5 years. Shorter-lived isotopes and

very low yield isotopes are ignored, with their mass assigned to their longer-lived progeny, so that all mass is tracked without the need for any groups of "actinide other" or "fission product other" categories. For example, ^{232}Pa at reactor discharge is assigned to ^{232}U. Similarly, heat and decay energy is included when short-lived isotopes decay, e.g. ^{90}Y decay energy is included when ^{90}Sr decays.

2.4.3. Brief description of the methods used in VISION

For the benchmark simulation, VISION input included a specified nuclear electricity demand, an initial legacy reactor fleet with a defined retirement profile and an initial store of used fuel. An introduction date for fast reactor technology was provided and new fast reactors were given preference over new thermal reactors. The benchmark also included a detailed specification of separations and fabrication capacities for both LWR and FR used fuel. Based on the requested nuclear generation, VISION forecast reactor retirements and ordered replacement reactors such that licensing, construction and initial core fabrication were completed in a timely manner. Fuel for the start-up core was fabricated during reactor construction. Since this timeframe overlapped with the time period when refuelling requests were ending for the retiring reactor, this limits fabrication spiking effects.

The "wet storage" (called "cooling storage" in the benchmark) capacities were used to model inventories of discharged fuel undergoing the minimum cooling period, while "dry storage" (called "cooled storage" in the benchmark) was used for fuel that exceeded the cooling period but had not been recycled. Other VISION fuel storage features such as interim storage were not utilised for the benchmark. All materials in storage were decayed to reflect isotopic evolution.

Separations and fuel fabrication facilities were ordered, licensed and constructed based on the requested separations and fuel fabrication profiles. The multi-year start-up ramp rate capability of VISION for modelling first and nth separations facilities was not used, with 100% capacity assumed at start-up to better match the capabilities of the other models.

During the period when UOX separation was operating but fast reactors were not yet available, separated TRU was stored and decayed. Once fast reactor technology was available, VISION was instructed to preferentially order a new fast reactor to replace a retiring reactor as long as sufficient fuel feedstock was available.

VISION provides the user with a number of controls that can be used to tune a model. One of these is a parameter to indicate how conservative the model should be with respect to feedstock availability, which is especially important when modelling burner fast reactors which require an ongoing supply of fresh TRU throughout their life. Since the fast reactors in the benchmark were operating at a conversion ratio of 1.0, the model was directed to only ensure sufficient TRU feedstock for the initial core and 2 years of refuelling, after which time the reactor would be capable of sustaining itself with only a source of recycled U or depleted U. This setting also resulted in the closest match to the other models for the ratio of LWRs to fast reactors during the period after mid-century, when the initial round of reactor retirements/replacements has been completed.

Another feature of VISION, the TRU bank, was not utilised in the benchmark activity so as to better match the other models. The TRU bank allows the user to indicate how much TRU should be retained in storage as insurance against temporary facility interruptions such as the recent ~18 month shut-down of THORP due to a broken pipe. If the TRU bank had been utilised, VISION would have built fewer fast reactors than the other models due to the TRU held in reserve.

2.5. DESAE

2.5.1. *Introduction*

DESAE2.2. (Dynamics of Energy System of Atomic Energy, version 2.2. [1], [2]) is a 'Systems' code developed by the Russian Kurchatov Institute. It calculates the nuclear fuel cycle requirements, material balances and economic parameters for a given combination of nuclear reactors and recycling plants during a specified time period. It allows the following options:

- the study of (independent) regions simultaneously;
- the commissioning of new reactors and the decommissioning of old reactors during the time period of a scenario, allowing the study of transition scenarios;
- the modification of reactor characteristics, both to start the scenario and in a time-dependent way during the scenario;
- the creation of new reactor types;
- the study of both open and closed fuel cycles;
- the saving of scenarios for later use or modification.

2.5.2. *DESAE Reactor models*

A DESAE reactor model contains the following main parameters. All parameter lists can be accessed by a GUI change mechanism (for one at a time changes) or by an EXCEL interface (for multiple changes).

DESAE holds spent fuel in two locations: in the 'NPP' (nuclear power plant) during the post-irradiation cooling time (parameters 13-15), in 'storage' otherwise. Fuel in storage is not described by isotope unless recycled at the 'NRP' (nuclear recycling plant).

Parameters 19 and 20 expand to the lists in Tables 5 and 6. The content of ^{235}U in DU (depleted uranium tailings from the enrichment plant) is the parameter defining the efficiency of the enrichment process for fresh fuel.

Table 4: DESAE Reactor parameters

№	Characteristic	Units
1	Reactor capacity (thermal)	GW
2	Capacity factor	none
3	Efficiency (electricity)	none
4	Efficiency (high temperature)	none
5	Efficiency (low temperature)	none
6	Efficiency (water production)	none
7	Fuel loading (core)	tonnes
8	Fuel loading (axial blanket)	tonnes
9	Fuel loading (radial blanket)	tonnes
10	Core residence time	days
11	Axial blanket residence time	days
12	Radial blanket residence time	days
13	Cooling time of spent fuel (core)	year
14	Cooling time of spent fuel (axial blanket)	year
15	Cooling time of spent fuel (radial blanket)	year
16	Construction duration	year
17	Reactor lifetime	year
18	Decommissioning duration	year
19	Resources and materials	various
20	Economic	various
21	Core isotopic composition	wt%
22	Axial blanket isotopic composition	wt%
23	Radial blanket isotopic composition	wt%

Table 5: Expansion of resources and materials parameter

№	Characteristic	Units
1	Content of ^{235}U in DU	wt%
2	Staff	man/GW(el)
3	Iron consumption	kg(Fe)/kW(t)
4	Copper consumption	kg(Cu)/kW(t)
5	Aluminum consumption	kg(Al)/kW(t)
6	Zirconium consumption	kg(Zr)/kg(fuel)
7	Water consumption	kg(H_2O)/kWh(el)
8	Heavy water consumption	kg(D_2O)/kW
9	Graphite consumption	kg(C)/kg(fuel)
10	Heavy metal consumption	kg(metal)/kW
11	Electricity consumption	kWh(el)/kWh(t)
12	User material	kg/GWh(el)

Table 6: Expansion of economic parameter

№	Characteristic	Units
1	Natural uranium price	$/kg
2	Separative work price	$/kg
3	Natural thorium price	$/kg
4	Reprocessed Pu fission price	$/kg
5	Reprocessed ^{233}U price	$/kg
6	Fuel fabrication price	$/kg
7	Spent fuel stored at NPP	$/kg/year
8	Spent fuel in long-time storage	$/kg/year
9	Back end dumping of spent fuel	$/kg
10	Construction cost of 1 kW plant	$/kWt
11	Decommissioning cost of 1 kW plant	$/kWt
12	Monthly salary paid in plant	$/(man·month)
13	O&M	$/kWh

The core, axial and radial blankets are considered as separate regions in the core with their own fuel characterisations (input and output), burn-ups and residence times. Parameters 21-23 expand to the list shown in Table 7 (all expressed in wt.%), for which the values must be calculated by an external code.

Table 7: Expansion of core, axial blanket or radial blanket parameters

№	Characteristic	№	Characteristic	№	Characteristic
1	Content of Th-230 in the first loading	20	Content of Th-230 in the equilibrium loading	39	Content of Th-230 in spent fuel
2	Content of Th-232 in the first loading	21	Content of Th-232 in the equilibrium loading	40	Content of Th-232 in spent fuel
3	Content of U-232 in the first loading	22	Content of U-232 in the equilibrium loading	41	Content of U-232 in spent fuel
4	Content of U-233 in the first loading	23	Content of U-233 in the equilibrium loading	42	Content of U-233 in spent fuel
5	Content of U-234 in the first loading	24	Content of U-234 in the equilibrium loading	43	Content of U-234 in spent fuel
6	Content of U-235 in the first loading	25	Content of U-235 in the equilibrium loading	44	Content of U-235 in spent fuel
7	Content of U-236 in the first loading	26	Content of U-236 in the equilibrium loading	45	Content of U-236 in spent fuel
8	Content of U-238 in the first loading	27	Content of U-238 in the equilibrium loading	46	Content of U-238 in spent fuel
9	Content of Np-237- in the first loading	28	Content of Np-237- in the equilibrium loading	47	Content of Np-237- in spent fuel
10	Content of Pu-238 in the first loading	29	Content of Pu-238 in the equilibrium loading	48	Content of Pu-238 in spent fuel
11	Content of Pu-239 in the first loading	30	Content of Pu-239 in the equilibrium loading	49	Content of Pu-239 in spent fuel
12	Content of Pu-240 in the first loading	31	Content of Pu-240 in the equilibrium loading	50	Content of Pu-240 in spent fuel
13	Content of Pu-241 in the first loading	32	Content of Pu-241 in the equilibrium loading	51	Content of Pu-241 in spent fuel
14	Content of Pu-242 in the first loading	33	Content of Pu-242 in the equilibrium loading	52	Content of Pu-242 in spent fuel
15	Content of Am-241 in the first loading	34	Content of Am-241 in the equilibrium loading	53	Content of Am-241 in spent fuel
16	Content of Cm-244 in the first loading	35	Content of Cm-244 in the equilibrium loading	54	Content of Cm-244 in spent fuel
17	Content of I-129 in the first loading	36	Content of I-129 in the equilibrium loading	55	Content of I-129 in spent fuel
18	Content of Tc-99 in the first loading	37	Content of Tc-99 in the equilibrium loading	56	Content of Tc-99 in spent fuel
19	Content of xp1- in the first loading	38	Content of xp1- in the equilibrium loading	57	Content of xp1 in spent fuel

'xp-1' is a free parameter which can be identified with an isotope of interest to the user.

DESAE reactor models have no parameters for the composition of the core load just before decommissioning. The decommissioned core load, therefore, is assumed to be the same as an equilibrium core load.

DESAE applies the following decay parameters (Table 8), taken from the DESAE Users Manual [1] to all spent fuel in NPP, storage and to the isotopes at the NRP.

Table 8: Spent fuel decay parameters

Element	Isotope	Decay type	$T_{1/2}$
Tc	99	β	2.1×10^5 years
I	129	β	1.57×10^7 years
Th	230	α, SF	7.54×10^4 years
U	232	α	68.9 years
Pu	238	α, SF	87.7 years
	241	β	14.4 years
Am	241	α	432.2 years
Cm	244	α	18.1 years

The density of decaying isotope *k* in spent fuel for time step *j*+1 (ρ_k^{j+1} ρ_k^{j-1}) is calculated from the density of isotope *k* in year *j* (ρ_k^j), and that of its parent ρ_{k-1}^j as follows:

$$\rho_k^{j+1} = \rho_{k-1}^j \cdot \left(1 - e^{-\lambda_{k-1} dt}\right) + \rho_k^j \cdot e^{-\lambda_k dt}$$

DESAE comes with many pre-defined reactor types, including:

- Na-cooled FRs with breeding ratios of 1.2, 1.4 and 1.6, and a Pb-cooled FR with a BR of 1.05;
- VVER 440 and VVER 1 000;
- an RBMK 1 000;
- LWRs with 4%, 5% and MOX fuel;
- two GCRs;
- a PHWR with natural uranium fuel;
- two HTRs.

The usual method for creating a new reactor type would be to find a similar existing reactor type as a starting model.

2.5.3. DESAE scenario models

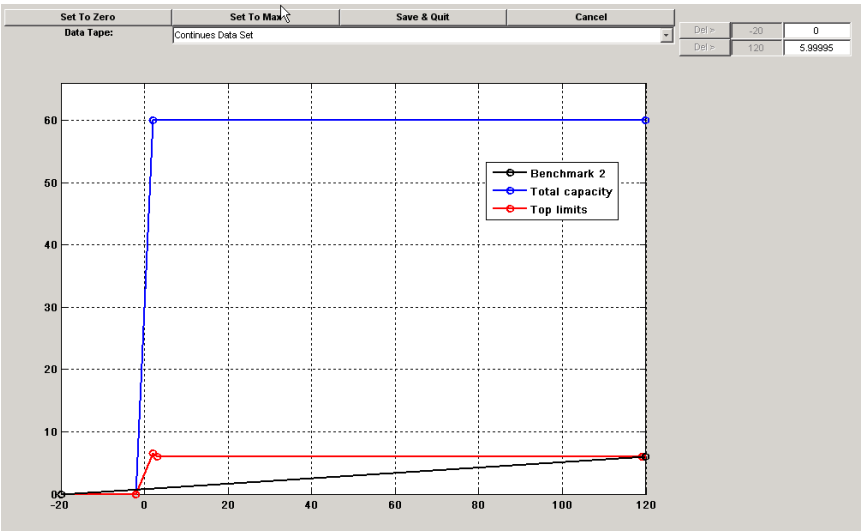
A DESAE scenario is defined as:

- a global power vs. time requirement;
- individual power vs. time requirements for each of up to seven defined reactor types;
- the presence or absence of recycling (closed vs. open fuel cycles);
- recycling plant capacities as a function of time for up to four recycling plants.

Figure 12 shows how power requirements are input for a scenario where a global power requirement of 60 GW is to be supplied by two reactor types ('Benchmark 1', and 'Benchmark 2'). Input is done via a graphical user interface (GUI) in which the variation of

total capacity (either thermal[1] or electrical) is defined by drawing a line on an input graph ('screen'). The total capacity target having been input on a first screen, and 90% of that capacity having been supplied on a second screen by Benchmark 1, the GUI input for Benchmark 2 (third screen, shown in Figure 9) shows the starting value (black line) and a red line representing the power requirement necessary so that Benchmark 1 and Benchmark 2 add to the total requirements. The user would then normally adjust the black line to meet the red.

Figure 12: An example input screen for entering scenario power requirements



Nuclear power and material in storage, is assumed to be zero at the start of any scenario, so a scenario requiring pre-existing reactors cannot be simulated. DESAE builds and rebuilds the fleets according to Table 4 individual reactor specifications. In particular, reactors are built over the time period for construction (parameter 16) and decommissioned (parameter 18) at the end of their defined lifetime (parameter 17). If the power requirements are for longer than the reactor lifetimes, DESAE decommissions the reactors and builds a second identical fleet, with associated mass flows equal to the full core loads.

Recycling plants are attached to reactors as shown in the example in Figure 13, with the first two reactor types feeding recycling plant 1, the third reactor type feeding recycling plant 2 etc. All recycled transuranic elements are put into storage and are available as input fuel to any of the reactors if the run time option 'closed fuel cycle' is selected. Unavailable isotopes required for fresh fuel are 'borrowed' from storage and result in negative values there (corresponding to the requirement to import these isotopes from another region).

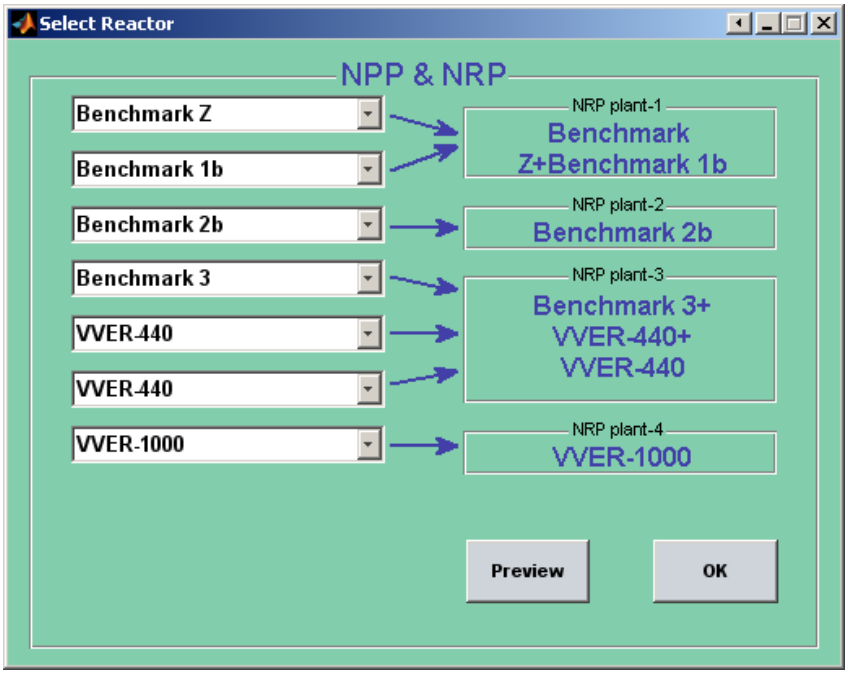
Recycling of uranium and thorium for new fuel is selected via two other independent run time options. In the case of uranium recycling, the wt.% of ^{235}U in U in the spent fuel must be at least twice the '^{235}U in DU parameter' (Table 5, parameter 1), for recycling to occur. Recycled uranium from storage is used as fuel for the fleet in a particular year only

1 The capability of DESAE to normalise to thermal power demand allows the construction of scenarios where the reactor fleet is used for other purposes, such as desalination or the production of high-temperature steam for industrial use.

if there is sufficient quantity there to supply the entire requirement for fresh fuel; otherwise newly mined natural uranium is presumed to be the source of supply.

Fuel at individual recycling plants is assumed to be processed on a first in, first-out basis. There is no capability for applying rules (such as 'fuel from this reactor fleet has priority over fuel from that reactor fleet'). There is no parameter corresponding to recycling losses.

Figure 13: DESAE reprocessing plant attachment to reactors



A DESAE world model consists of a number of non-interacting regions, each with its own reactor types and reprocessing capacity. Reprocessed fuel is not traded across regional boundaries. Some interaction between regions can be obtained by making the total power requirement the global requirement.

2.5.4. *DESAE output*

In DESAE, the reactor mix can be relatively easily tuned to optimise a number of critical parameters:

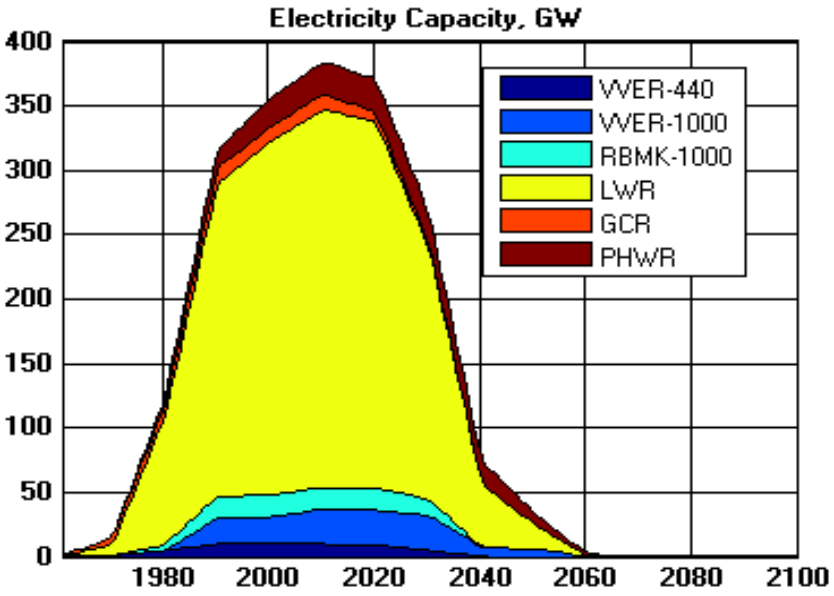
- natural uranium consumption;
- Pu and ^{233}U build-up in storage;
- total scenario power;
- recycled fuel mass flow;
- economic parameters.

Reactors can be replaced at this point, or have their characteristics changed, or have their contribution to the scenario altered in a time-dependent way until the user is satisfied with the scenario. A detailed scenario report is then produced. This contains output graphs (or Excel tables) of the following scenario characteristics, broken down by reactor type and by recycling plant:

- energy/power vs. time:
  - thermal & electrical capacity;
  - power available for non-electrical uses (water, high/low temperature steam).
- fuel use:
  - uranium and thorium utilisation;
  - depleted uranium production.
- recycling throughput:
  - spent fuel in NPP storage;
  - spent fuel in NRP storage;
  - amount recycled per year and total recycled;
  - ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{237}Np, ^{242}Pu, ^{241}Am, ^{244}Cm in storage;
  - ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{230}Th, ^{232}Th in storage;
  - radiotoxicity of fuel in storage.
- fission products (only ^{129}I and ^{99}Tc):
  - composition of spent fuel;
  - decay heat.
- economics:
  - use of non fuel materials (water, steel, electricity) to run the fleets;
  - cash flow per year (operational costs, reactor commissioning and decommissioning costs).
- composition of spent fuel for each reactor type (% fHE):
- fresh and spent fuel movement:
  - fresh fuel inloading/spent fuel outloading per year;
  - recycled Pu inloaded/created per year;
  - ^{233}U created in spent fuel per year.

An example of the form of the output is given in Figure 14, taken from the DESAE Users Manual [1].

Figure 14: A scenario electricity capacity plot



2.5.5. *Methods*

Overall, DESAE keeps power output, spent fuel storage, and fuel recycling separate for up to 7 reactor fleets and 4 recycling plants, but has common enrichment, fuel manufacturing, and disposal, as shown in Figure 15. From the perspective of the flow of isotopes, a DESAE model looks like Figure 16.

For calculations, DESAE utilises a year-by-year finite difference method where integrations are broken into summations with an index running over 1/10th of a year intervals. In general, the inputs to the scenarios are interpreted as differentials (for example, the rate of increase of capacity) and these are integrated using a limited summation:

$$F(J) = \sum_{j=1}^{J} f_j dt \, .$$

In the case of instantaneous changes (for example, sudden capacity introduction), DESAE changes the input to make the differential finite by forcing the change to occur in the space of one year, then breaks down the year into 10 equal intervals.

Figure 15: A DESAE scenario

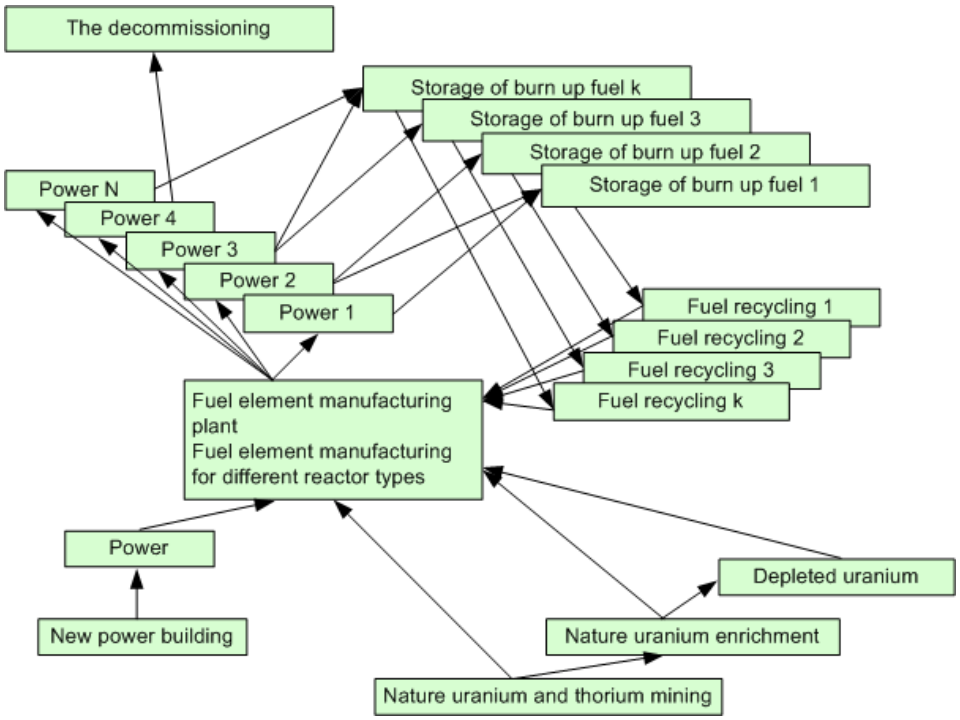
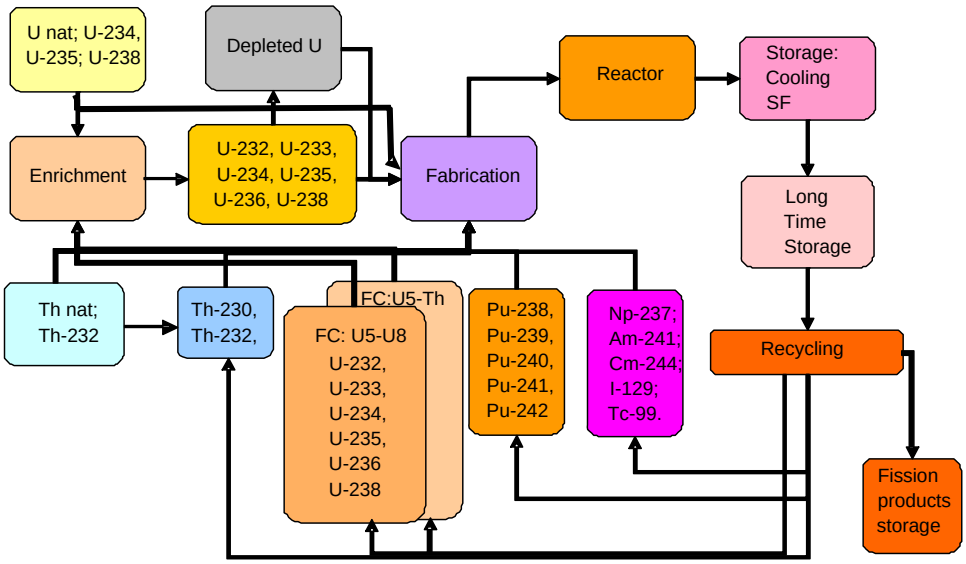


Figure 16: Flow of isotopes in a DESAE scenario



Some treatment of output fuel is essentially heuristic. The energy emitted by fission products is calculated by the formula:

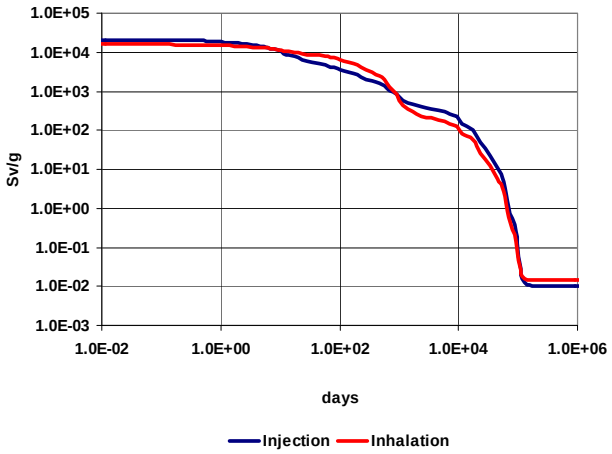
$$E_{fp}(t) = \int_0^t E_{fp\,0}\, f(t-\tau) G_{fp}(\tau) d\tau$$

Where the specific fission product heat generation function:

$$f(t) = 4(t+10)^{-12} - 3.48(t+2\cdot 10^{7})^{-12} - 2.45\cdot 10^{-4} e^{-\frac{t}{2040}} - 1.59\cdot 10^{-6} e^{-\frac{t}{290000}}$$

is defined. Similarly, since precise fission product isotopics of spent fuel are not defined as input to DESAE, radiotoxicity and inhalation hazard estimates in DESAE output are based on the time variation of the fission product activity in 50 GWd/tonne VVER spent fuel multiplied by the biological dose coefficient for each fission product (Figure 17).

Figure 17: VVER 50 GWd/tonne inhalation and injection weighting function



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[1] E.A. Andrianova, V.D. Davidenko, V.F. Tsibulskiy, Dynamic of Energy System of Atomic Energy (DESAE2.2) code User Manual.

[2] V. Tsibulskiy, S.Subbotin, M. Khoroshev, F.Depisch (2006), DESAE (Dynamic Energy System- Atomic Energy) Integrated Computer Model for Performing Global Analysis in INPRO Assessment Studies, International Conference on Nuclear Engineering "ICONE 14" 17-20 July 2006, Miami, Florida, USA.

2.6 Comparison of code capabilities

Table 9: Comparison of code capabilities

	COSI6	DESAE2.2	EVOLCODE2.0	FAMILY21	VISION
Language/Software	Java		Fortran	Microsoft Visual Basic(Japanese Edition)	System dynamics/power sim
Facilities: Discrete/continuous	Discrete	Continuous	Discrete	Discrete	Continuous
Fuel (batches)/continuous	Discrete	Discrete	Discrete	Discrete	Continuous(except for ordering of first cores)
User interface	Graphical user interface	Graphical user interface	Text interface	Graphical user interface and Microsoft Excel	MS-Excel spreadsheet and/or Graphic User Interface
Simultaneous advanced technologies scenarios	Any combination of LWR, HTR, FR(SFR and GFR), ADS+different types of fuels	Yes	Any reactor with any fuel	Any combination of LWR, HWR, FR(SFR, GFR, LFR and ADS+different types of fuels	One-tier, two-tier scenarios(+choice of the number of recycling)
Isotopics tracking	Y(Isotopes of U/Pu/MA/200FP)	U,Pu, minor actinides	Yes (~3300 isotopes)	Yes(Isotopes of U/Pu/MA/880 FP)	Yes(Follows up to 81 isotopes)
Choice of fuel	User	User	User	User	User
Calculation of transmutation performance in cores	Embedded CESAR code (Fortran) with one-group cross-sections libraries based on deterministic APOLLO 2 and ERANOS systems Direct coupling with ERANOS possible	No coupling with transmutation code	Creation of one-group cross-sections with EVOLCODE2. Possibility of choosing reference libraries.	Stored depletion matrix based on results of depletion calculation by the ORIGEN2 code	Precalculated Fuel recipes with interpolation(as a function of the number of cycles)
Start-up and shut-down fuel loads	Yes	Start-up only	Yes	Yes	No

Table 9: Comparison of code capabilities (continued)

	COSI6	DESAE2.2	EVOLCODE2.0	FAMILY21	VISION
Front-end fuel cycle facilities	All facilities represented	Enrichment	Enrichment	Enrichment fabrication	Only fuel fabrication and enrichment facilities represented
Reprocessing plants	Represented	Represented	Represented	Yes	Represented
Reprocessing capacity deployed	Automatic/manual	Manual	Yes	Automatic/manual	Automatic/manual
Spent fuel to be reprocessed	User choice: "first-in first-out" or "last-in-first-out"	First-in first only	User choice: "first-in first-out", "last-in first-out" or homogeneous	User choice: "first-in first-out" or "last-in-first-out"	User choice: oldest/youngest (with min cooling time) and from onsite/ MRS/retrievable repository
Fissile material availability forecast	No	No	No	No	Y (user defined degree of conservatism in ordering new NPP)
User parameter for deployment of reactors	No	No	Yes	Yes	User defined scenario/growth driven/level of conservativeness
Waste-radioactivity	Yes	No	Yes	Yes(at vitrified waste production step)	Yes
Waste-decay heat	Yes	Yes	Yes	Yes(same as above)	Yes
Waste-radiotoxicity	Yes	Yes	Yes	No	Yes
Waste conditioning modelling	Yes	No	No	Yes (Heat rate and weight of oxides are considered at vitrified waste production.)	Yes
Repository requirement assessment	Yes	No	No	No	Yes(several types of disposal)
LLW modelling	Yes	No	No	No	Yes(A/B/C/GTCC)
Economics assessment module	Yes	Yes	No	No	Yes

Table 9: Comparison of code capabilities (continued)

	COSI6	DESAE2.2	EVOLCODE2.0	FAMILY21	VISION
Economics optimisation	No	Yes	No	No	Uncertainty analysis
U price model	No	Yes	No	No	Uncertainty range
Transportation costs	Yes	Yes	No	No	No
Proliferation metrics	Heating rate from Pu(W/kg) weight fraction of even isotopes inventory(significant quantities defined by IAEA) concentration(SQ/t)	No	No	Heating rate form Pu(W/kg) at Pu storage and MOX fuel fabrication steps weight fraction of U,Pu MA isotopes amount of Pu storage after reprocessing	Yes-multiple

3. Benchmark specification [1]

3.1. Depletion part

Some scenario codes use a depletion module to calculate the evolution of isotopic composition in the different fuels (UOX, MOX, etc); it is a simplified calculation to have an average composition at each step in the fuel and back-end cycle. The first stage of the benchmark is to compare the results obtained by these depletion modules.

The benchmark was performed on the depletion module of the scenario codes for 3 types of fuel:

- UOX fuel for PWRs loaded with 100% of UOX;
- MOX fuel for PWRs loaded with 100% of MOX;
- MOX fuel for Na-FRs loaded with 100% of MOX; minor actinides are introduced in this fuel.

3.1.1. PWR UOX fuel composition (wt%)

The UOX fuel has an initial enrichment of 4.95 wt% ^{235}U. The composition to be used in the benchmark is presented in Table 10.

Table 10: Initial composition for UOX fuel

Nuclide	wt%
^{234}U	0.0445
^{235}U	4.95
^{238}U	95.0055

3.1.2. Irradiation history for PWR UOX fuel

The calculation is made in one step for a burn-up of 60 GWd/t (1 760 EFPD) and a cooling time of 5 years.

3.1.3. PWR MOX fuel composition (wt%)

The MOX fuel has an initial content of 9.026 wt% (Pu+^{241}Am). The composition to be used in the benchmark is presented in Table 11.

Table 11: Initial composition for MOX fuel

Nuclide	wt%
^{235}U	0.2056
^{238}U	90.7684
^{238}Pu	0.2816
^{239}Pu	4.6565
^{240}Pu	2.1951
^{241}Pu	1.0606
^{242}Pu	0.7257
^{241}Am	0.1065

3.1.4. Irradiation history for PWR MOX fuel

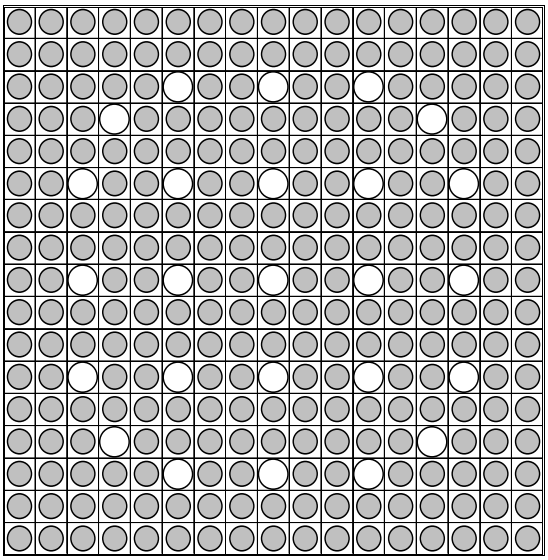
The calculation is made in one step for a burn-up of 60 GWd/t (1 760 EFPD) and a cooling time of 5 years.

3.1.5. Other hypotheses for UOX and MOX in PWR

The geometric data are corresponding to a standard fuel assembly type FRAGEMA 900 MWe (17×17), as follows:

- 264 fuel rods
- number of thimble guide : 24
- 1 instrumentation tube
- no extra water hole
- length of the network : 1.264916 cm
- radius of the pellet : 0.41266 cm
- intern radius of the clad : 0.41266 cm
- extern radius of the clad : 0.474364 cm
- density of the clad : 6.49012 g/cm^2
- thickness of the water between 2 sub-assemblies : 0.10768 cm
- intern radius of the thimble guide : 0.572945 cm
- extern radius of the thimble guide : 0.6132012 cm

Figure 18: Scheme of the fuel assembly



- material composition / densities
  - density of UO_2 = 10 07 g/cm^3
  - density of U-PuO_2 = 10 02 g/cm^3 (at nominal temperature)
- average boron concentration
  - 456 ppm for UOX
  - 600 ppm for MOX (suggested to be constant during irradiation)
- boundary conditions
  - calculations are made in an infinite network, the coefficient B2 is adapted to have k_{eff} = 1
- temperatures
  - UOX: 600°C for the fuel, 306°C for the moderator (choose an approximate clad temperature)
  - MOX: 650°C for the fuel, 305°C for the moderator (choose an approximate clad temperature)

Irradiation could be divided into small steps. However, the time for unloading and loading of the fuel between two cycles was ignored.

The cladding material is Zircalloy 4, same as the thimble guide. Mass composition for fabrication is:

- Sn: 1.2 – 1.7 %
- Fe: 0.18 – 0.24 %
- Cr: 0.07 – 0.13 %
- O: 0.10 – 0.14 %
- Zr: ~98% (may vary upon sum of other composition)

The thimble guides and instrumentation tube are filled with water. For the instrumentation tube, the material is Zircalloy 4 (same as thimble guide).

3.1.6. MOX FR fuel composition (wt%)

The MOX Na-FR fuel has an initial content of 22.21 wt% Pu. The composition to be used in the benchmark is presented in Table 12.

Table 12: Initial composition for MOX Na-FR fuel

Nuclide	wt%
^{234}U	0.000538
^{235}U	0.188200
^{238}U	75.091897
^{238}Pu	0.875900
^{239}Pu	12.670000
^{240}Pu	6.889000
^{241}Pu	0.702600
^{242}Pu	1.074000
^{241}Am	0.858200
^{242m}Am	0.048340
^{243}Am	0.511400
^{237}Np	0.500000
^{242}Cm	0.002424
^{243}Cm	0.006541
^{244}Cm	0.469900
^{245}Cm	0.083910
^{246}Cm	0.027150

The main characteristics of the fuel are described in Chapter 3.

The composition of the fuel assembly is the following:

Table 13: Composition for MOX Na-FR fuel

	Composition (volume)
Fuel	37.51
Na	32.94
Structure	23.59

3.1.7. Irradiation history for MOX Na-FR fuel

The calculation is made in one step for a burn-up of 136 GWd/t (1 700 EFPD) and a cooling time of 5 years. Calculations are made in an infinite lattice at criticality (k_{eff} = 1).

3.2. Transition scenarios

Three scenarios are included in the second part of the benchmark:

- open cycle;
- monorecycling of the plutonium in the PWRs;
- monorecycling of the plutonium in the PWRs and then deployment of the Generation IV fast reactors recycling plutonium and minor actinides.

The common hypotheses are:

- duration of the scenario: 120 years
- constant installed power: 60 GWe
- constant electrical annual production: 430 TWhe (load factor: 0.8176)
- variation rate for every type of reactor: ± 2 GWe/year.

Table 14: Data compilation for the benchmark study

Fuels / blankets				
	Unit	PWR UOX	PWR MOX	FR
Fissile burn-up	GWd/tHM	60	60	136
Axial blankets burn-up	GWd/tHM	-	-	15
Radial blankets burn-up	GWd/tHM	-	-	25
Minimum cooling time	y	5	5	2
Fabrication time	y	2	2	2
Fresh fuel ^{235}U enrichment	%	4.95	0.25	0.25
Moderation ratio		2	2	-
Equivalent Pu content	%	-	-	14.5
Cores				
	Unit	PWR UOX	PWR MOX	FR
Electrical nominal power	GW	1.5	1.5	1.45
Efficiency	%	34	34	40
Load factor	-	0.8176	0.8176	0.8176
Heavy metal masses				
Fissile	t	128.9	128.9	41.4
Axial blanket	t	-	-	18.0
Radial blanket	t	-	-	13,5
Breeding gain		-	-	≈1
Cycle length	EFPD	410	410	340
Core fraction (fuel)		¼	1/4	1/5
Core fraction (radial blankets)		-	-	1/8
Reprocessing plants				
	Unit	PWR UOX	PWR MOX	FR
Priorities		First in –first out	First in –first out	First in –first out. First fuel then blankets
Losses (U and Pu)	%	0.1	0.1	0.1

Table 15: Data compilation for the benchmark study-Initial spent fuels

Initial spent fuels				
	Unit	PWR UOX	PWR MOX	FR
Initial mass	t	1 0000	0	0
Isotopic composition			-	-
^{232}U	%	2.78E-07	-	-
^{233}U	%	3.08E-07	-	-
^{234}U	%	1.75E-02	-	-
^{235}U	%	7.56E-01	-	-
^{236}U	%	6.87E-01	-	-
^{238}U	%	9.09E+01	-	-
^{236}Pu	%	6.53E-08	-	-
^{238}Pu	%	5.11E-02	-	-
^{239}Pu	%	6.37E-01	-	-
^{240}Pu	%	3.11E-01	-	-
^{241}Pu	%	1.53E-01	-	-
^{242}Pu	%	1.12E-01	-	-
^{241}Am	%	5.05E-02	-	-
^{242m}Am	%	1.57E-04	-	-
^{243}Am	%	2.94E-02	-	-
^{237}Np	%	9.16E-02	-	-
^{239}Np	%	2.52E-08	-	-
^{242}Cm	%	1.89E-06	-	-
^{243}Cm	%	1.89E-04	-	-
^{244}Cm	%	1.21E-02	-	-
^{245}Cm	%	1.05E-03	-	-
^{246}Cm	%	1.46E-04	-	-
^{247}Cm	%	2.87E-06	-	-
^{248}Cm	%	4.90E-07	-	-
Other isotopes	%	6.17E+00	-	-
Total	%	100	-	-

3.2.1. Scenario 1 - Open cycle

Scenario 1 simulates an open cycle nuclear fleet. Figures 19 and 20 show both flow chart and installed capacity of scenario 1.

Figure 19: Scenario 1 – Flow chart

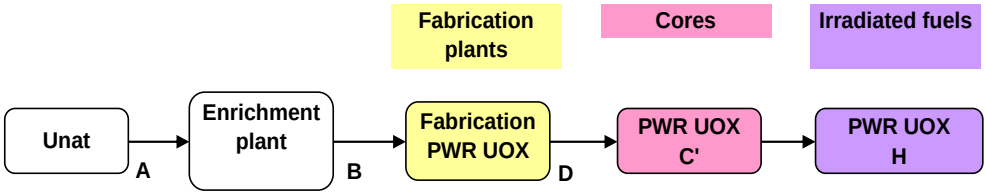
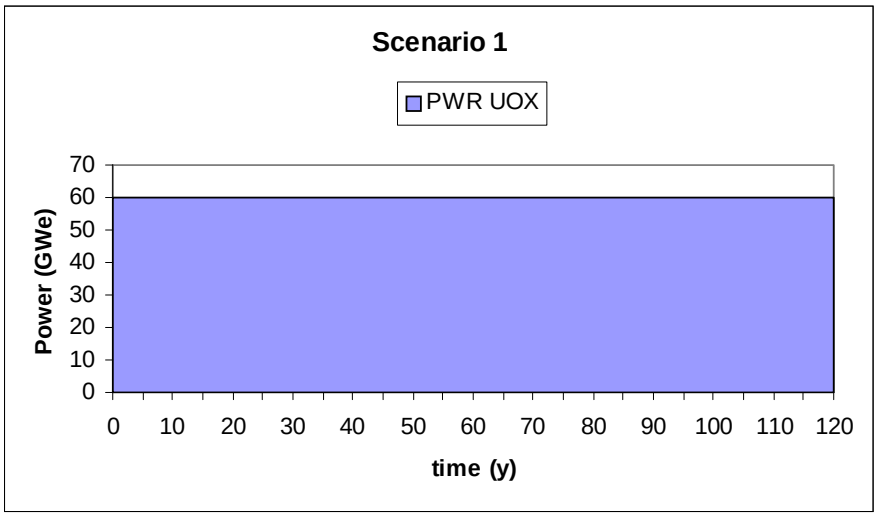


Figure 20: Scenario 1 – Installed capacity



3.2.2. Scenario 2 - Monorecycling of plutonium in PWRs

Figure 21 shows the flow chart of scenario 2. The installed capacity, which is a linear function, is shown in Table 16 and Figure 22.

Figure 21: Scenario 2 – Flow chart

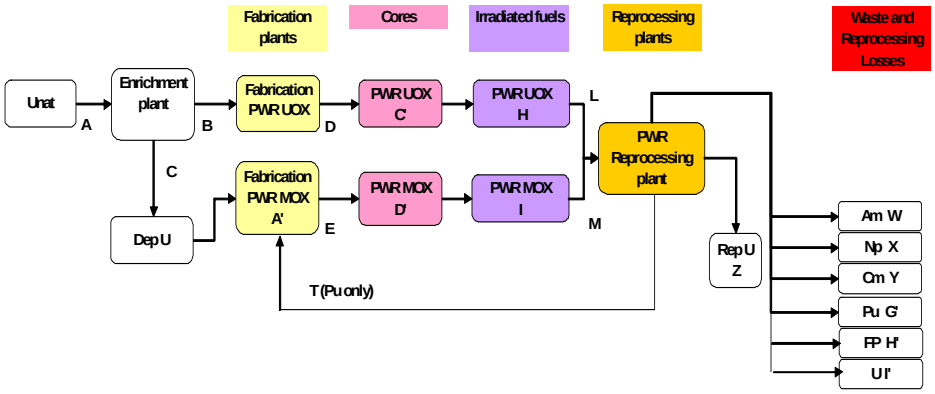


Figure 22: Scenario 2 – Installed capacity

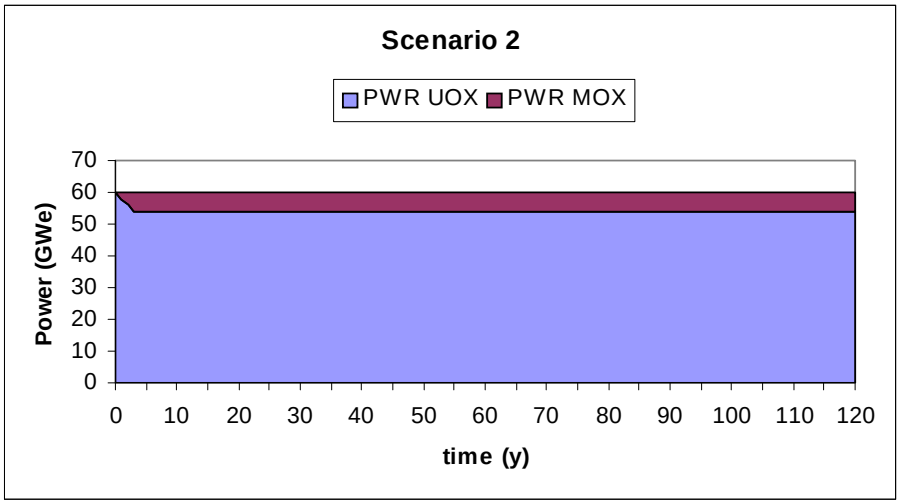


Table 16: Scenario 2 – Installed capacity

Time (y)	PWR UOX (GWe)	PWR MOX (GWe)
0	60	0
3	54	6
70	54	6
73	54	6
80	54	6
110	54	6
120	54	6

3.2.3. Scenario 3 - Monorecycling of plutonium in PWRs and deployment of Generation IV fast reactors

Figure 23 shows the flow chart of scenario 3. The installed capacity, which is a linear function, is shown in Table 17 and Figure 24.

Figure 23: Scenario 3 – Flow chart

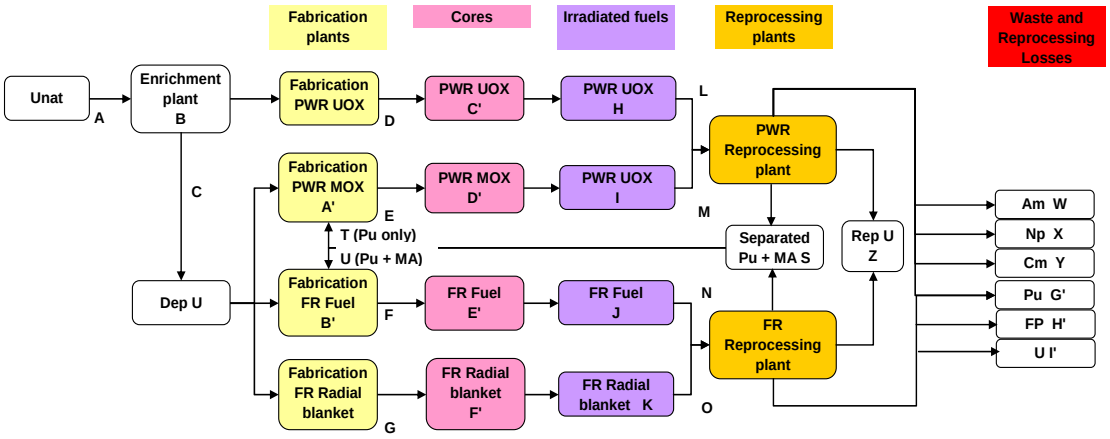


Figure 24: Scenario 3 – Installed capacity

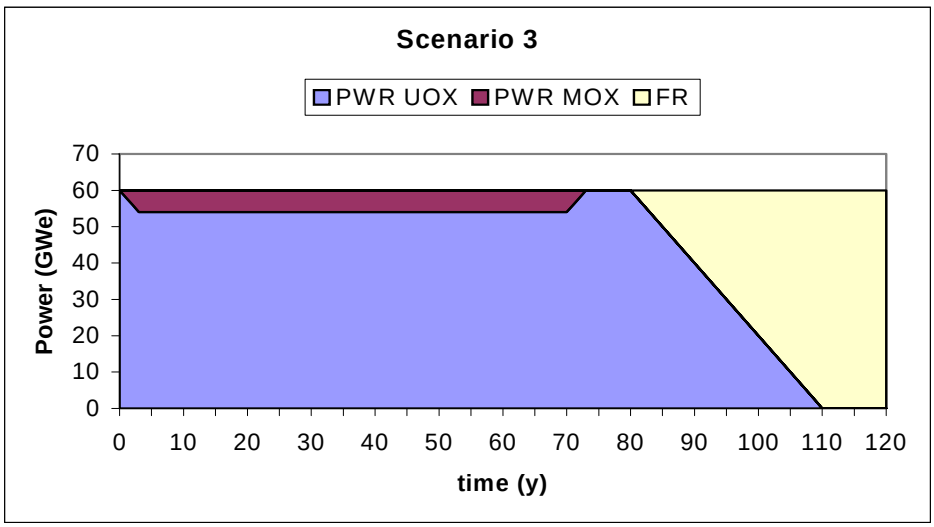


Table 17: Scenario 3 – Installed capacity

Time (y)	PWR UOX (GWe)	PWR MOX (GWe)	FR (GWe)
0	60	0	0
3	54	6	0
70	54	6	0
73	60	0	0
80	60	0	0
110	0	0	60
120	0	0	60

3.2.4. Other specifications

Reactors

- scenario 1
  - first load of PWR UOX : year -2
  - last load of PWR UOX : year 120.2
- scenario 2
  - first load of PWR UOX : year -2
  - last load of PWR UOX : year 120.2
  - first load of PWR MOX : year 0
  - last load of PWR MOX : year 120.8
- scenario 3
  - first load of PWR UOX : year -2
  - last load of PWR UOX : year 109.2
  - first load of PWR MOX : year 0
  - last load of PWR MOX : year 72.8
  - first load of fast reactor : year 80
  - last load of fast reactor : year 119.8

First core

The first cores of the reactors are not simulated. The first batch of fuel has the same mass and composition as equilibrium batches.

Enrichment plant

The enrichment of the tails is 0.25% ^{235}U.

Fabrication plant

- scenarios 1, 2 and 3

  - fabrication of PWR UOX fuel starts at year –4 to feed the reactor PWR UOX at year -2
- scenarios 2 and 3
  - fabrication of PWR MOX fuel starts at year –2 to feed the reactor PWR MOX at year 0
- scenario 3 for fast reactor fuel fabrication
  - fabrication of fast reactor fuel and blankets start at year 78 to feed the fast reactor at year 80.

The fabrication of the driver fuel is made with a mix of depleted uranium (tails from enrichment), Pu and MA, with an equivalent ^{239}Pu fraction is 14.5%. The reactivity coefficients are detailed in Table 18.

Table 18: Reactivity coefficients

Isotope	Coefficient	Isotope	Coefficient
^{234}U	0.0255	^{242m}Am	2.1763
^{235}U	0.7749	^{243}Am	-0.3236
^{236}U	-0.06192	^{237}Np	-0.2695
^{238}U	0	^{239}Np	-0.3078
^{238}Pu	0.5779	^{242}Cm	0.3109
^{239}Pu	1	^{243}Cm	2.5015
^{240}Pu	0.1223	^{244}Cm	0.2086
^{241}Pu	1.4717	^{245}Cm	2.4319
^{242}Pu	0.08263	^{246}Cm	0.2294
^{241}Am	-0.3374	^{247}Cm	1.5522

If there is insufficient TRU coming from the fast reactor reprocessing plant, Pu from the PWR reprocessing plant is used.

The fabrication of axial and radial blankets is made with depleted uranium coming from the enrichment plant (0.25% ^{235}U).

Reprocessing plants

No reprocessing plant is used in scenario 1. For scenarios 2 and 3, initial reprocessing is applied to the initial pool in the first years of the calculation, until irradiated UO_2 is created by the PWR UOX reactor and cooled. Time for reprocessing is assumed to be 0. Table 19 gives the reprocessing assumptions.

Table 19: Reprocessing plant assumptions

	PWR reprocessing plant: scenario 2	PWR reprocessing plant: scenario 3	Fast reactor reprocessing plant: scenario 3
First year of reprocessing	-2	-2	85
Last year of reprocessing	120		
Type of fuel reprocessed	100% PWR UOX	From -2 to 70 : 100% PWR UOX From 71 to 120 : 25% PWR MOX -75% PWR UOX 100%PWR UOX if PWR MOX not available	100% of fuel assemblies (fissile part + radial blankets) 100% of radial blankets if fuel assemblies are not available
Priorities	Oldest batch are reprocessed first		
Annual capacity of initial heavy metal	850 tonnes	850 tonnes	600 tonnes
Separation efficiency	99.9% of annual flux for U and Pu, 0% for MA	From -2 to 74 : 99.9% of annual flux for U and Pu, 0% for MA From 75 to 120 : 99.9% for U, Pu and MA	99.9% of annual flux for U, Pu and MA

Spent fuel

- scenarios 1, 2 and 3
  - the initial mass of spent fuel (10 000 tonnes) is accounted at year –7 , with a minimum cooling time of 5 years (thus available for reprocessing at year -2)

3.2.5. Results expected to be reported

Expected results to be reported for the benchmark include the following annual values:

- natural uranium consumption;
- SWU needs;
- fuel fabrication flows;
- interim storage (spent fuel, depleted uranium, plutonium, etc.);
- processed spent fuel;
- Pu and MA mass flows;
- plutonium and minor actinides losses from reprocessing.

Reference

[1] OECD/NEA (2007), Specification For The Benchmark Devoted To Scenario Codes, NEA/NSC/DOC(2007)13/REV3.

4. Benchmark results

4.1. Depletion results

The results of the depletion part of the benchmark are presented in detail in Annex 1 and summarised in the following figures. The values indicate the difference of SCK, JNC and KIT compared to CEA results. This way of presentation is used to illustrate the differences between codes. It does not mean that CEA results can be considered as a reference compared to the other codes.

Figure 25: Depletion part – UOX results

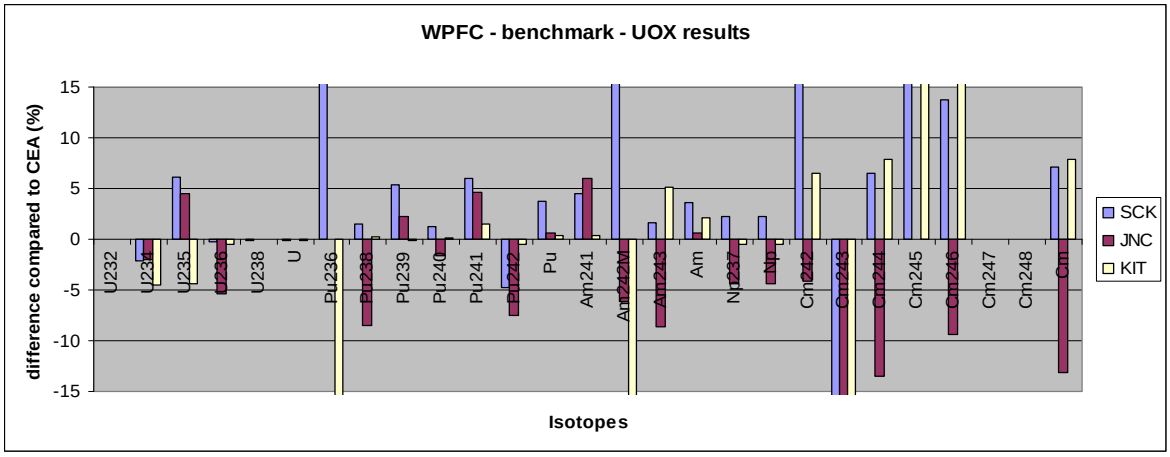


Figure 26: Depletion part – MOX results

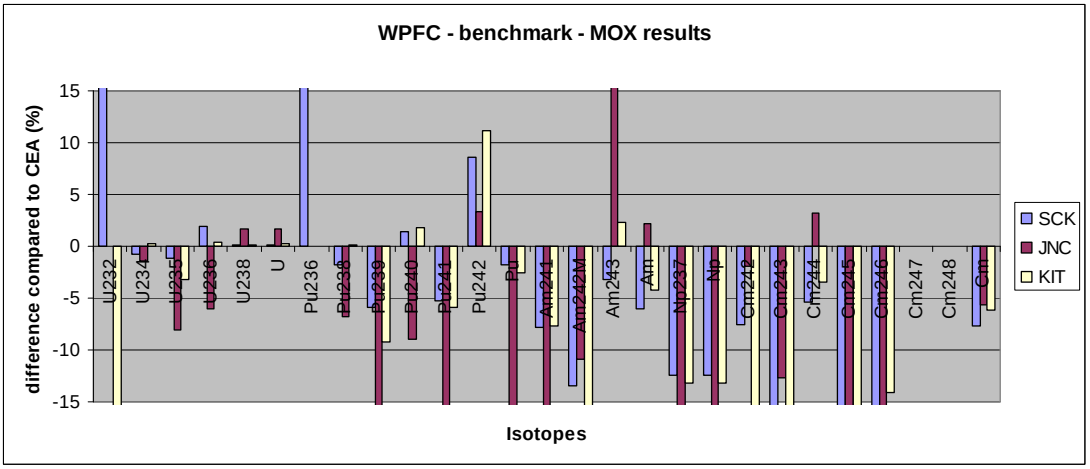
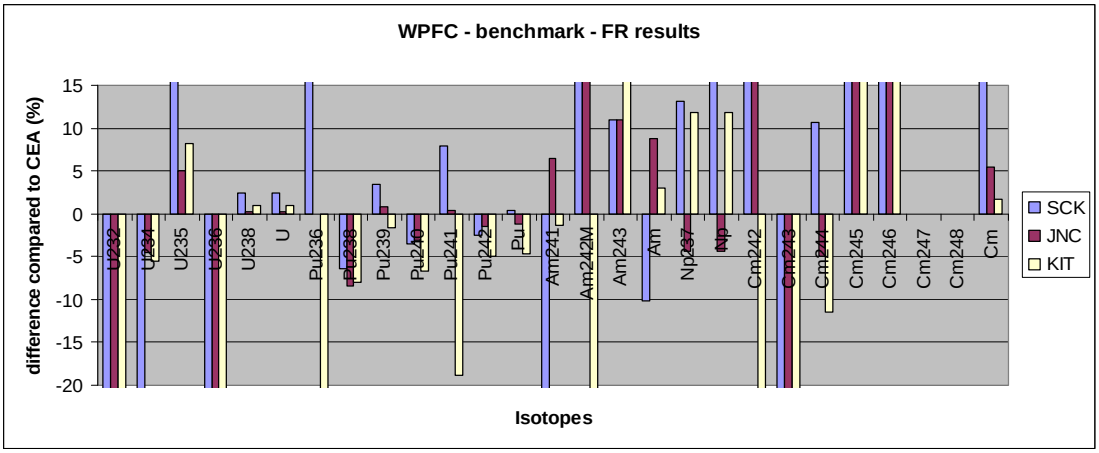


Figure 27: Depletion part – FR results



The results indicate that:

- For PWR UOX calculations there was good agreement (difference around or less than 5%) for uranium and plutonium isotopes (except ^{236}Pu), ^{241}Am and ^{237}Np. The differences are more significant for the other isotopes, particularly Cm isotopes.
- For PWR MOX calculations between SCK, KIT and CEA results, there was good agreement (difference around or less than 5%) for ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{241}Pu. The differences are more significant for the other isotopes, particularly Cm Isotopes. JNC calculations indicate very different trends.
- For FR calculations there was good agreement (difference around or less than 5%) for ^{238}U, ^{239}Pu, ^{240}Pu and ^{242}Pu. The differences are more important for the other isotopes, particularly Am and Cm isotopes.

4.2. Transition scenario results

4.2.1. *Scenario codes used in the benchmark study*

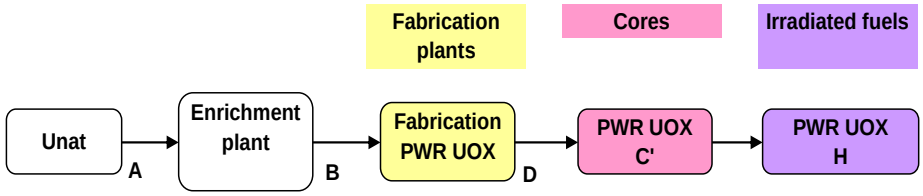
The scenario codes used in the benchmark are presented in Table 20:

Table 20: Scenario codes

Code	Version Date	Full name	Contributing Organisation	Scenarios
COSI6	4.0.0 (Dec 2008)	Commelini-Sicart	CEA	1, 2 and 3
VISION	-	Verifiable Fuel Cycle Simulation	INL	1, 2 and 3
FAMILY21	-	-	JAEA	1, 2 and 3
EVOLCODE	2.0.0 (Nov 2008)	Evolution Code	CIEMAT	2, 3
DESAE2.2	2.2	Dynamics of Energy System of Atomic Energy	AECL	1, 2 and 3

4.2.1. *Scenario 1*

Figure 28: Scenario 1 – Flow chart



Scenario 1 was analysed by four codes: COSI6, FAMILY21, DESAE2.2 and VISION2.2.

Since scenario 1 is very simple, the comparison will be focused on the front end of the fuel cycle: mining, enrichment, fuel fabrication and on spent fuel inventory. The results are presented in Figures 29-33 and are discussed below.

Figure 29: Scenario 1 – Natural uranium needs

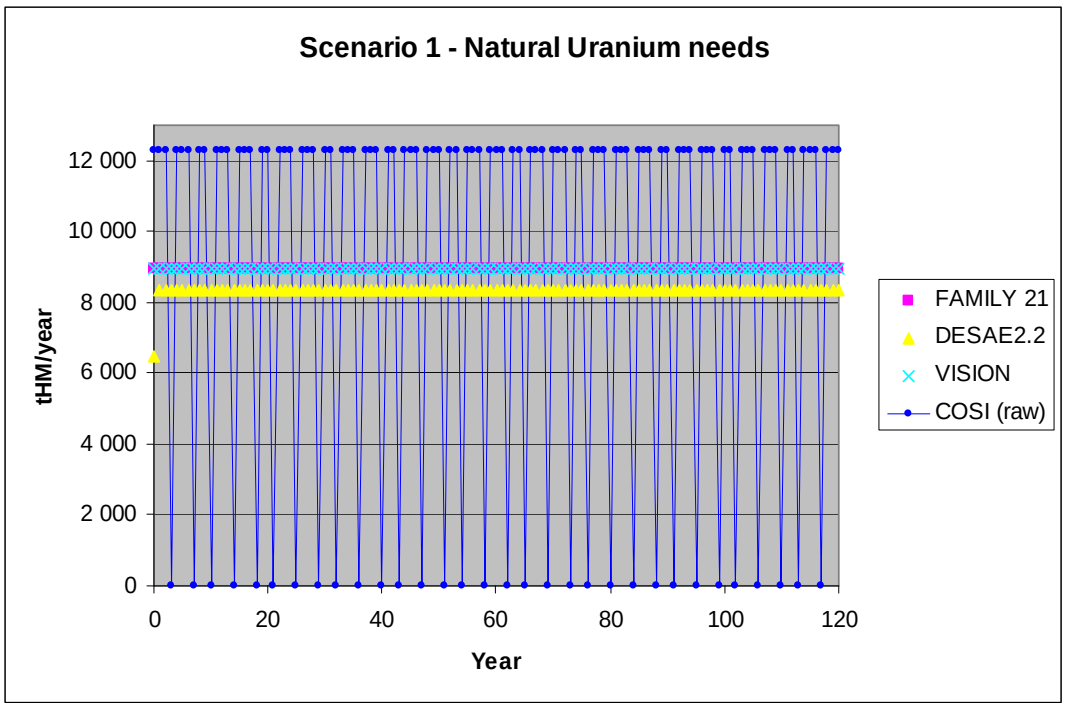


Figure 30: Scenario 1 – SWU needs

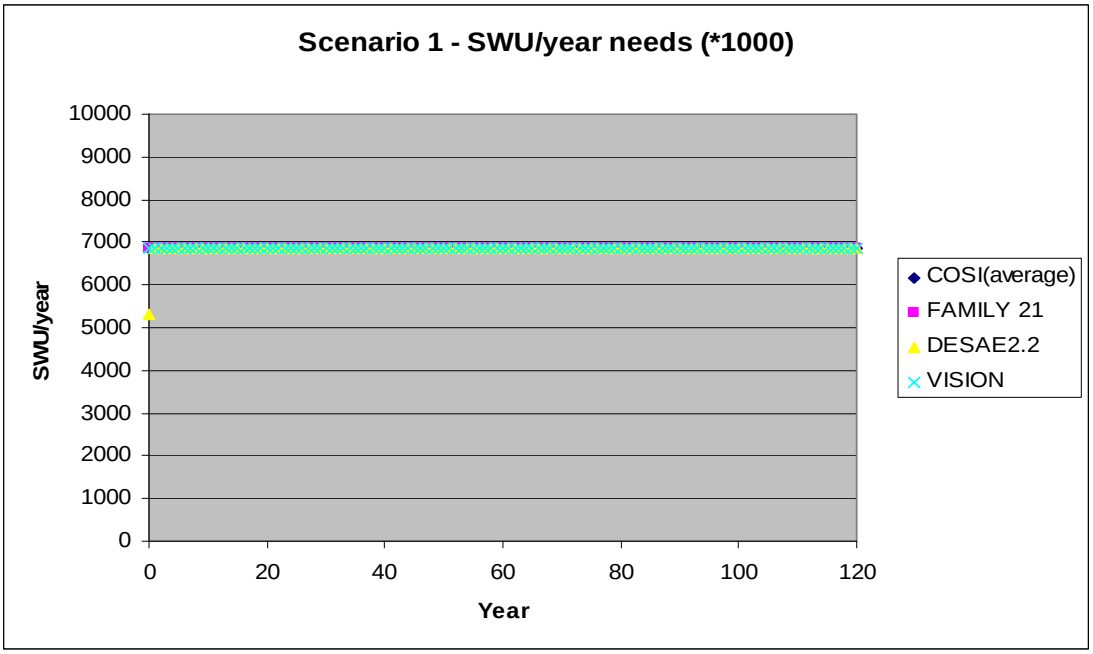


Figure 31: Scenario 1 – Enriched uranium needs

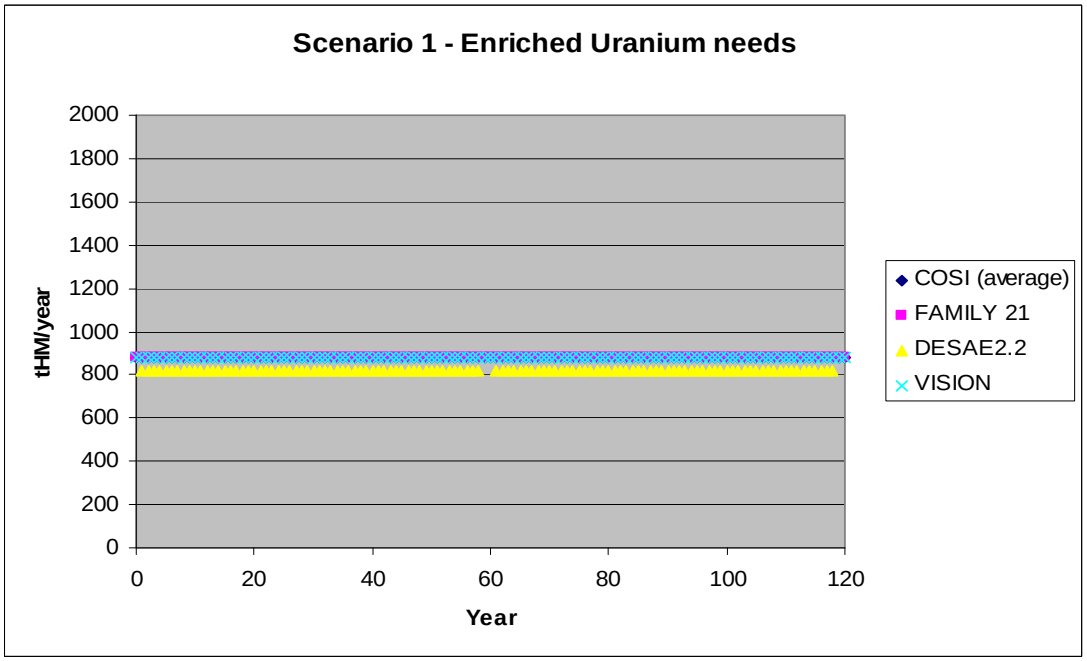


Figure 32: Scenario 1 – UOX fabrication needs

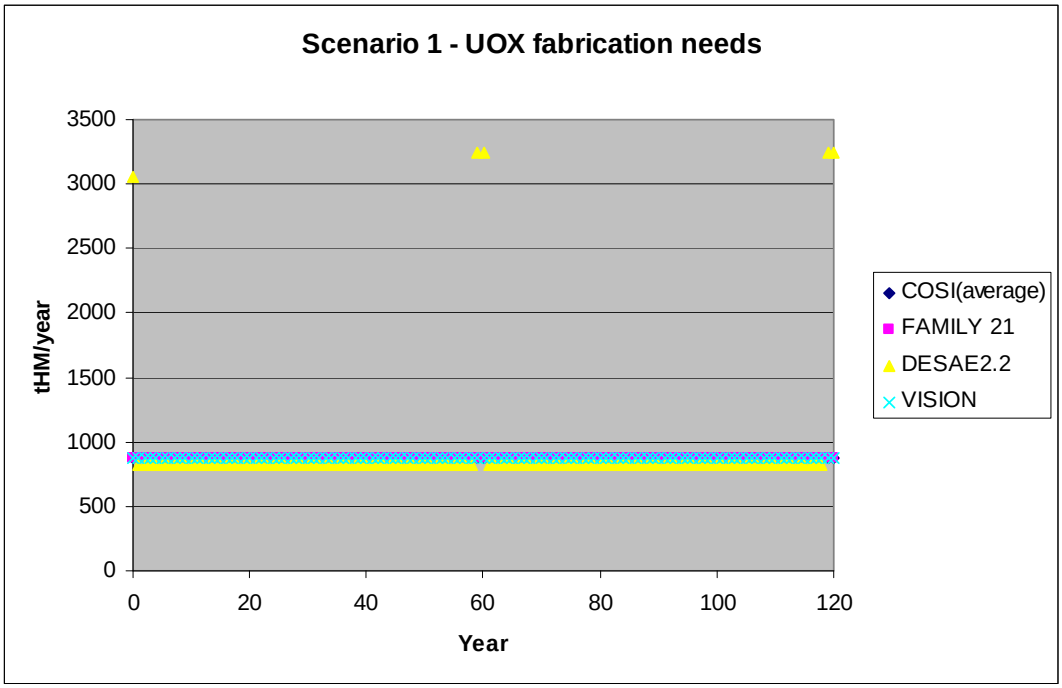
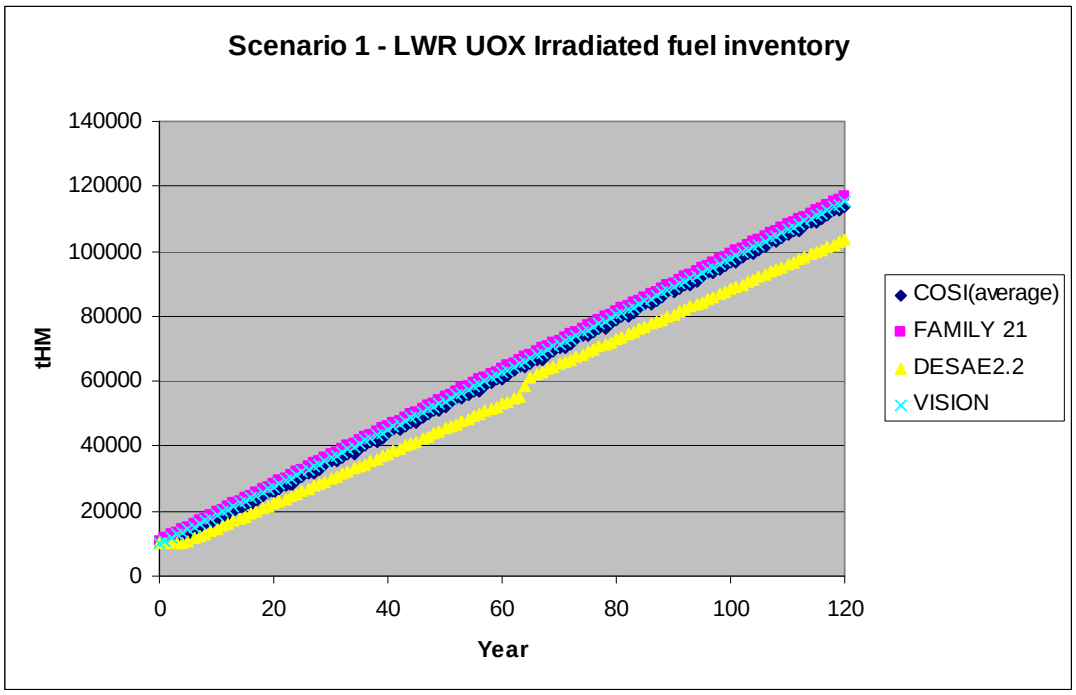


Figure 33: Scenario 1 – LWR UOX irradiated fuel inventory



The oscillations that are observed in COSI results for natural uranium needs come from a different representation of the fuel loading. COSI individually tracks the loading of batches in each reactor, which occurs every 16.4 months because all the reactors (1.5 GWe each) are simulated with one single fictive reactor of 60 GWe. On the other hand, the other codes provide average values for the annual loading of fresh fuel ("annual flow"), which is derived from the total number of reactors in operation (average annual loading per reactor × number of reactors).

In the following, COSI average results will be presented as much as possible to avoid having a more complex comparison with other codes.

From most of the codes, the results given for scenario 1 are very close. For instance, for LWR UOX fabrication, the average values given by the codes are:

- COSI: 877.7 tonnes / year
- FAMILY21: 878.4 tonnes / year
- VISION: 877.7 tonnes / year
- DESAE2.2: 820.7 tonnes / year

Only DESAE2.2 is a little bit different from the other codes: -6.5% compared to COSI, because of a difference in core loading assumptions. COSI and VISION have identical values.

Another small difference appears in the LWR UOX irradiated fuel inventory, due to an accumulation effect. Thus, at the end of the scenario, the results given by the code are:

- COSI: 113 700 tonnes
- FAMILY21: 116 905 tonnes
- VISION: 115 326 tonnes
- DESAE2.2: 103 627 tonnes

In the case of DESAE2.2, a little step can be noticed at year 64. At this year, about 6 000 tHM of spent fuel arrive in the storage instead of 820 tHM, due to the unloading of the initial cores of the reactor renewing the nuclear park. In the other calculations, the initial cores are not taken into account. However, the LWR UOX irradiated fuel inventory is lower at year 120, due to the lower annual core loading.

In the FAMILY21 calculation, the total LWR UOX irradiated fuel inventory at year 120 is higher because of annual unloading, which is a little bit higher than COSI and VISION.

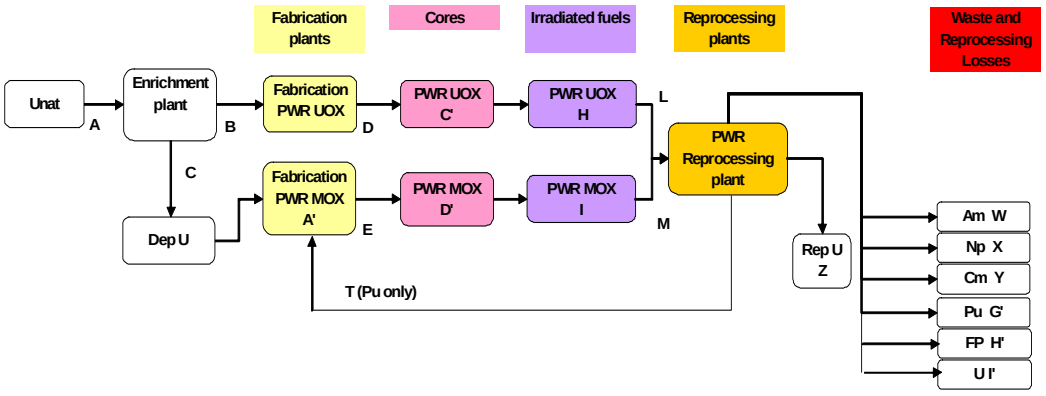
The year of the first unloading of spent fuel has also an impact on the accumulated LWR UOX irradiated fuel. The values given by the codes are:

- COSI: year 3
- FAMILY21: year 2
- VISION: year 1
- DESAE2.2: year 1

In summary, there was good agreement for all of the codes for scenario 1, with all showing the same trends and only minor differences in values. These minor differences are mostly explained by different implementations of the initial condition assumptions.

4.2.2. *Scenario 2*

Figure 34: Scenario 2 – Flow chart



Five codes used for scenario 2 analysis are COSI6, FAMILY21, DESAE2.2, EVOLCODE and VISION. The following figures present the results on the front end of the fuel cycle.

Figure 35: Scenario 2 – Natural uranium needs

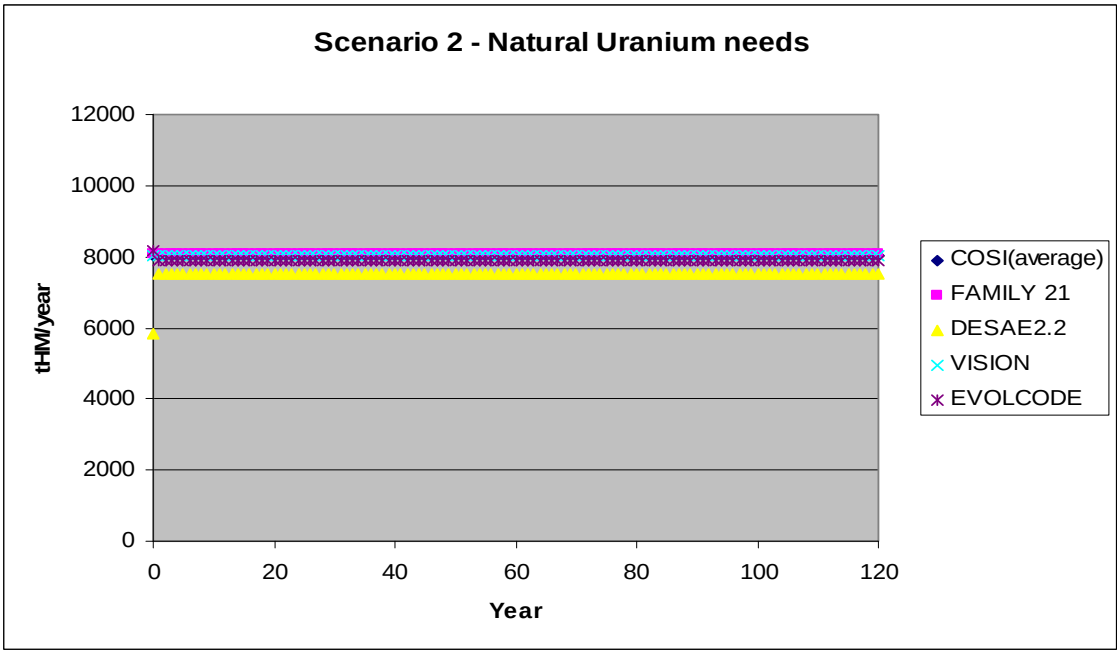


Figure 36: Scenario 2 – SWU needs

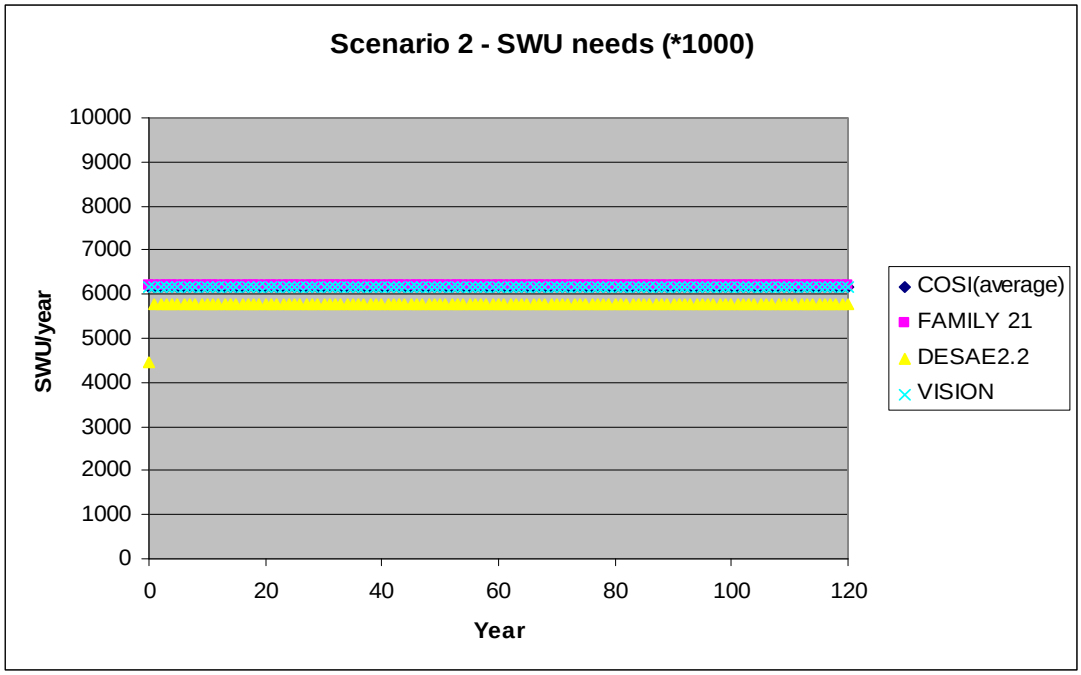


Figure 37: Scenario 2 – UOX fabrication needs

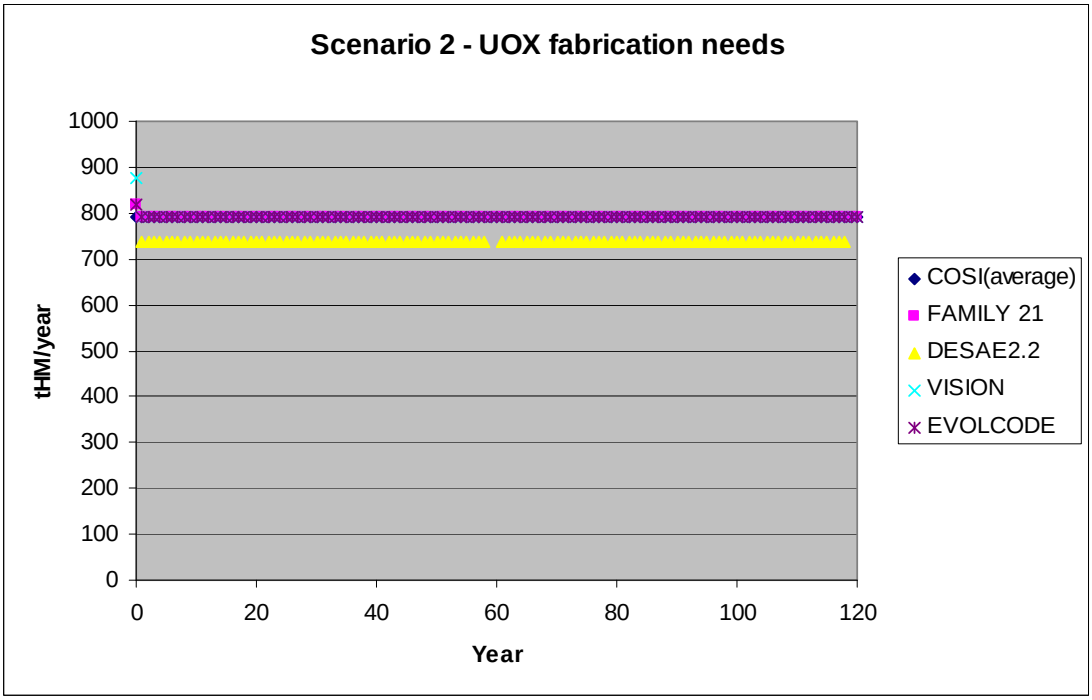
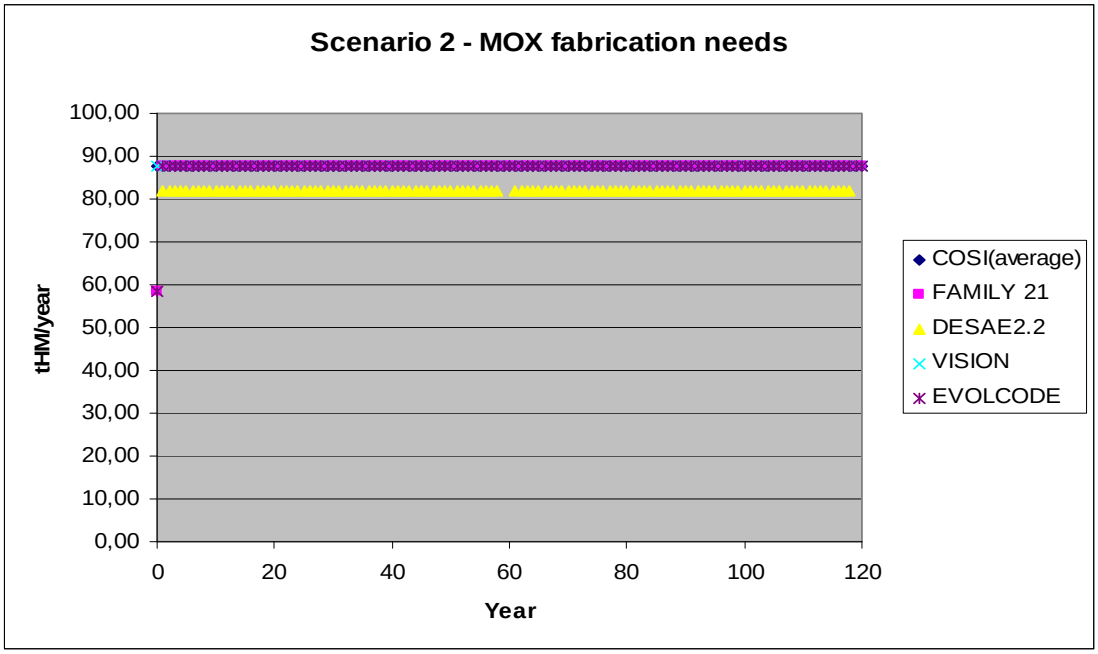


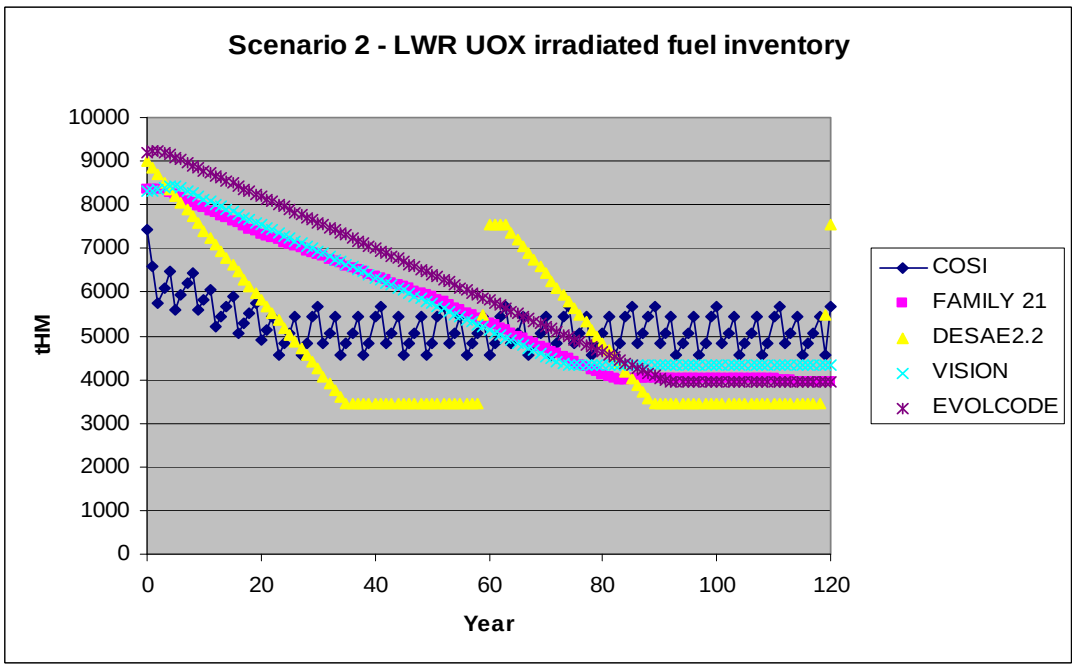
Figure 38: Scenario 2 – MOX fabrication needs



Among all the contributing codes, COSI, FAMILY21, EVOLCODE and VISION calculate the same values for natural uranium needs, SWU needs, UOX and MOX fuel needs. For DESAE2.2, natural uranium needs, SWU needs, UOX and MOX fuel needs are lower. The reason could be a difference in core loading assumptions.

The results on the back end of the fuel cycle are presented below:

Figure 39: Scenario 2 – LWR UOX irradiated fuel inventory

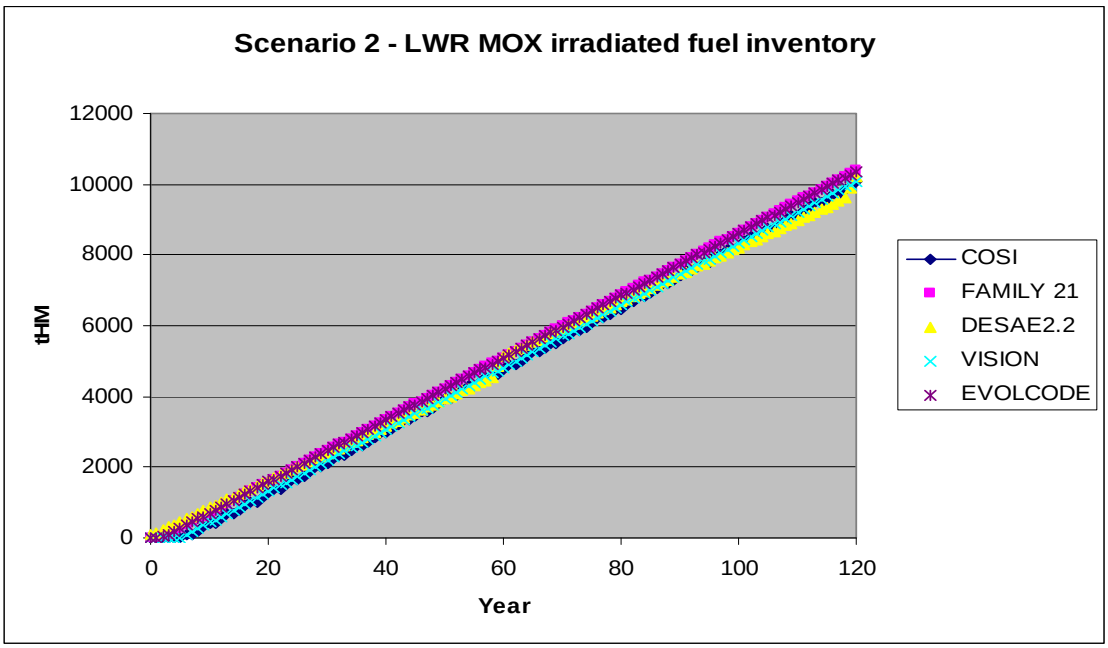


For UOX spent fuel inventory, the results are similar for FAMILY, VISION, and EVOLCODE, with COSI and DESAE showing different patterns. Until year 25 approximately, the decreasing slope of all results (excepting DESAE2.2) agrees very well, according to the difference between the reprocessing requirement assumption (850 tHM of spent fuel per year) and the amount of spent fuel unloaded from LWR-UOX (790 tonnes of spent fuel initial heavy metal per year). The different offsets of the curves come from differences in the first year of spent fuel unloading. For instance, between EVOLCODE and VISION, this difference is one year and leads to an offset of approximately 790 tHM. For COSI, the first year of unloading is year 3, with a difference of two years with VISION and 3 with EVOLCODE2. In scenario 3, COSI calculation has been adapted (the "COSI adjusted" results) to have the same initial condition as the VISION calculation. In that case, COSI, FAMILY, EVOLCODE and VISION all have similar values and trends (see Section 4.2.3 Scenario 3).

Beyond year 25, COSI reaches equilibrium in the spent fuel inventory due to a lack of spent fuel availability (see below, Figure 41). For FAMILY, EVOLCODE and VISION, this stabilisation takes place at different years due to the differences mentioned above in the first unloading. The level of stabilisation is around 4 000 tHM, corresponding to the mass of spent fuel in cooling before reprocessing. For COSI, the averaged level is around 5 000 tHM and the minimum value is around 4 500 tHM. This difference is due to the fact that COSI takes into account the actual cycle length of the reactors: 16.4 months. Thus, the mass of spent fuel unloaded is equal to 1 085 tHM. Moreover, the reprocessing is annual and occurs in COSI in the middle of the year. For these reasons, the spent fuel in cooling can be higher than 4 000 tHM.

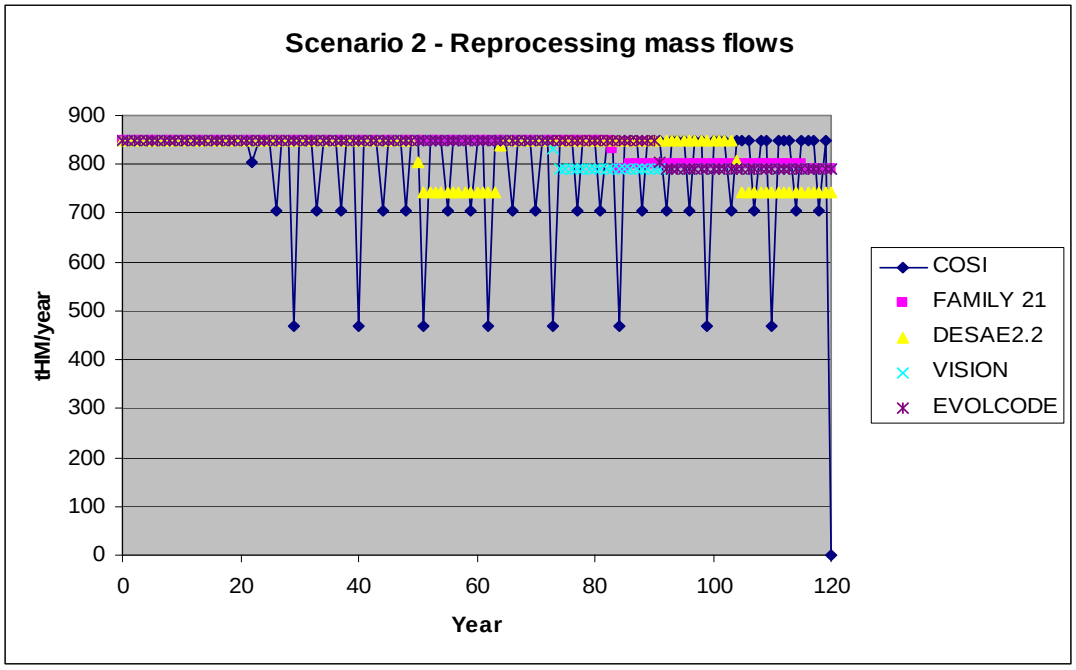
For DESAE2.2, the step at year 64 comes from the unloading of the first cores of the new reactors that replaced the original fleet when they reached their end of life. The renewal occurs in the same year for all the reactors of the park, as was the case in scenario 1 (Section 4.2.1). However, the minimum spent fuel inventory is about 3 500 tonnes and should be at least equal to 3 693 tonnes, which is 5 (years of cooling) multiplied by 738.6 tonnes (annual unloading in DESAE).

Figure 40: Scenario 2 – LWR MOX irradiated fuel inventory



The small differences in MOX spent fuel inventory are primarily a consequence of differences in MOX annual fabrication since MOX fuel is not reprocessed in this scenario, and differences in the year of first MOX unloading: year 3 for EVOLCODE, year 5 for COSI, 6 for VISION, 2 for FAMILY21, 0 for DESAE.

Figure 41: Scenario 2 – Reprocessing mass flows



Reprocessing mass flows are a consequence of the reprocessing assumptions: 850 tHM of UOX spent fuel per year, and of the fuel availability for reprocessing. Each code reaches, sooner or later, a point where the available spent fuel for reprocessing is insufficient to satisfy the specified reprocessing requirements (850 tHM), after that point the amount of annual reprocessing is equal to the annual mass of UOX spent fuel unloaded from the reactors five years before (790 tHM).

The oscillations observed with COSI are a result of the fuel availability in spent fuel storage. When the quantity of fuel available (having the minimum cooling time) is lower than 850 tHM, COSI reprocesses the fuel as much as possible. However, the averaged mass of spent fuel from reactor to storage is about 790 tHM per year, lower than the reprocessing demand: 850 tHM. For that reason, once the initial legacy of spent fuel is consumed around year 20, the annual reprocessing oscillates between 480 tHM and 850 tHM, depending on reactor unloading and fuel availability.

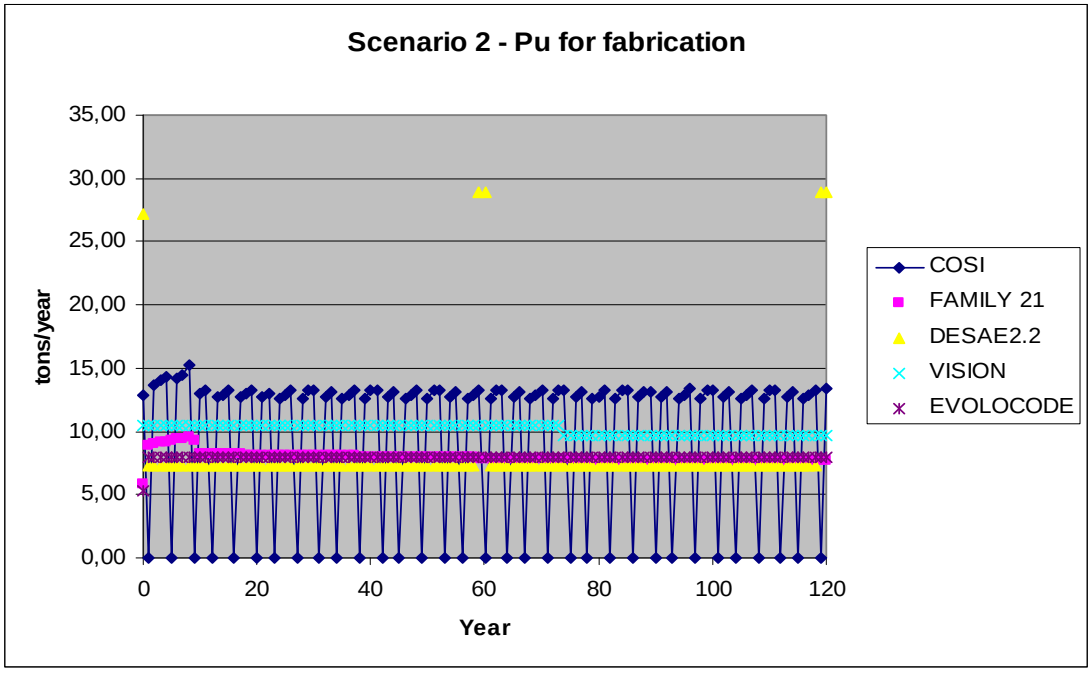
With VISION, the initial condition of the scenario leads to a higher value for the spent fuel inventory at year 0 (8 300 tHM against 7 350 tHM with COSI) and the first batch of spent fuel arrives earlier (year 1 against year 3). For these reasons, the amount of spent fuel available for reprocessing is higher than with COSI. After year 73, the annual reprocessing is in accordance with the annual mass of UOX spent fuel unloaded from the reactors: 790 tHM per year.

In scenario 3, COSI calculation has been adapted to have the same initial condition as the VISION calculation. In this case, COSI and VISION are consistent (see Section 4.2.3 scenario 3).

For DESAE2.2, the annual reprocessing is not consistent with the availability of spent fuel in interim storage: the spent fuel inventory is stabilised around year 35 whereas the

first lack of spent fuel for reprocessing occurs in year 50. After year 89 and until year 103, the annual reprocessing is 850 tonnes whereas the UOX spent fuel available seems lower. During this period of time, the UOX spent fuel inventory available for reprocessing should be equal to the annual UOX spent fuel discharge (738.6 tonnes for the case of DESAE2.2) since the UOX spent fuel inventory is at the minimum value.

Figure 42: Scenario 2 – Pu for fabrication



The Pu for fabrication of MOX is derived from MOX fabrication needs and the Pu fraction in the fresh MOX. In the assumptions, no equivalent ^{239}Pu amount or initial fixed content of Pu has been proposed for PWR MOX fuel in the transition scenario part of the benchmark.

Different codes provide a different value of the Pu for fabrication. Again, it is due to a different interpretation of the Pu fraction.

For COSI, the Pu fraction is calculated with an equivalence model established by neutronic calculations. The model takes into account the isotopic composition of plutonium for fabrication, the final burn-up of the fuel and the core fraction. The less fissile isotopes in plutonium, the higher the initial fraction is. The annual average value given by COSI is around 9.4 tonnes. As in scenario 1, the oscillations are due to the individual tracking of each fuel batch and because all the reactors (1.5 GWe each) are simulated with one single fictive reactor having a 16.4 month cycle length.

For VISION, the Pu fraction is set at 12% and the Pu mass for fabrication is constant at 10.45 tonnes from year 0 to year 73.

For EVOLCODE, the Pu mass for fabrication is a fixed value corresponding to the assumption made for the depletion part of the benchmark: 9.026% Pu in MOX fuel.

The amount of TRU in reprocessing losses is derived from irradiation calculations (amount of TRU in the spent fuel), cooling time of spent fuel (because of ^{241}Pu, ^{241}Am, ^{242}Cm and ^{244}Cm decay) and from the amount of spent fuel reprocessed each year.

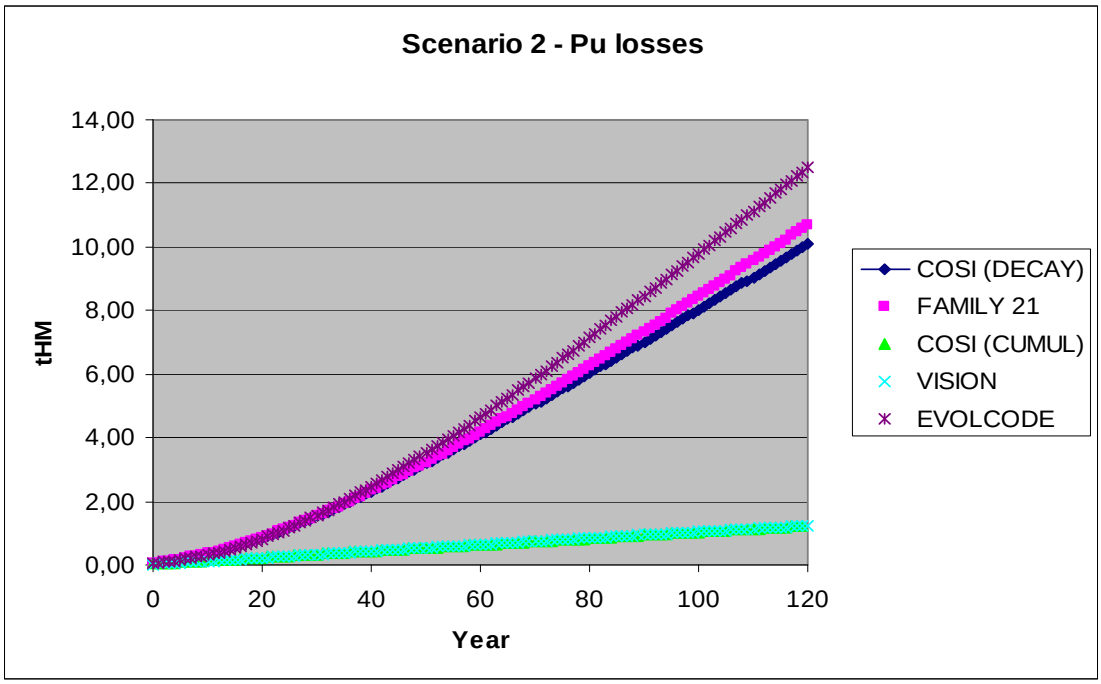
The amount of plutonium present in the waste is shown in Figure 43, except for DESAE, which cannot simulate reprocessing losses.

In COSI and EVOLCODE calculations, the Pu losses and minor actinides losses are mixed together in the same "box". Thus, for the Pu losses from reprocessing, two calculations are possible:

- the amount of Pu taking into account the decay of other isotopes (^{242}Cm and ^{244}Cm) mixed with the plutonium in the HLW : COSI (DECAY) and EVOLCODE;
- the amount of Pu without decay : COSI (CUMUL).

Two groups of results can be seen in the following figure. The first one includes the simulations without decay of the waste: COSI (CUMUL) and VISION. The second group includes COSI (DECAY), EVOLCODE, FAMILY and takes into account decay. For this latter group, the main contribution comes from the decay of ^{244}Cm, leading to ^{240}Pu. This decay (on the order of eight tonnes at the end of the scenario) matches the Cm evolution shown below (Figure 46). The reasons for the larger amount of Pu in the EVOLCODE simulation are a larger creation of Cm during irradiation and a 7% higher amount of annual spent fuel reprocessed.

Figure 43: Scenario 2 – Pu losses

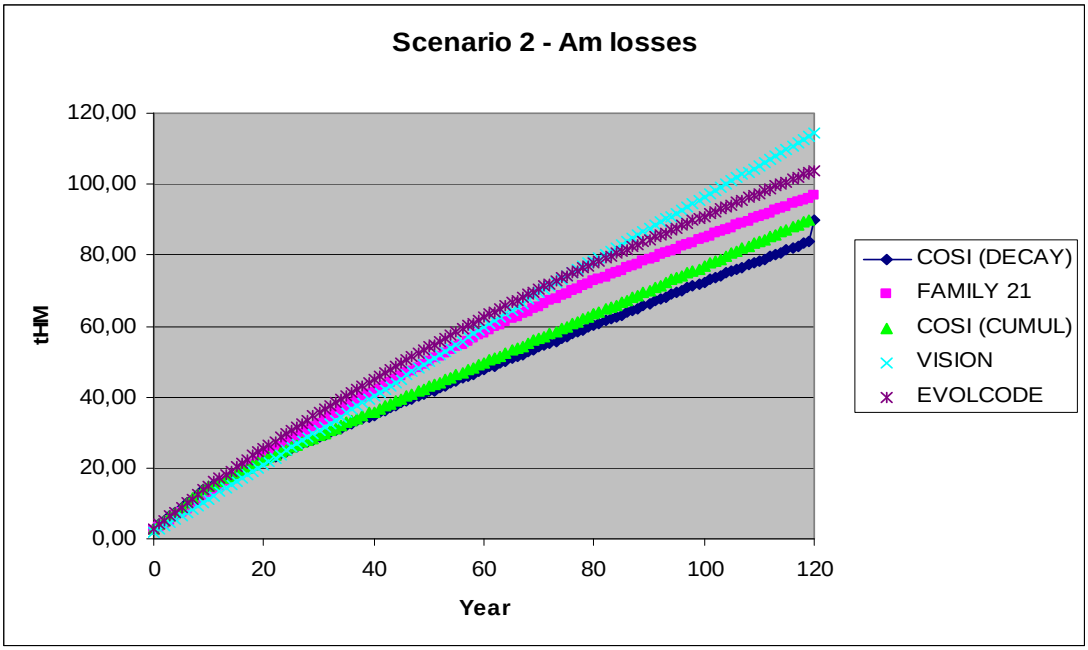


Thus, in order to explain the differences between codes for the amounts of Pu and minor actinides in the waste, it is necessary to consider 4 factors:

- A different reprocessing amount of heavy metal: in DESAE, EVOLCODE and FAMILY simulations, 850 tHM are reprocessed per year, whereas for COSI and VISION the averaged value of the reprocessing is 790 tHM per year (7% smaller amount).
- In COSI and EVOLCODE, the reprocessing strategy of "oldest batches are reprocessed first" is applied. However, due to the lack of spent fuel availability detected in COSI, the effective decay before reprocessing of spent fuel is different.

- Each code has applied its own neutron spectra and cross-sections, leading to slightly different isotopic compositions in the spent fuel.
- COSI, EVOLCODE and FAMILY codes account for the decay of nuclear waste after reprocessing, whereas DESAE and VISION do not consider this decay.

Figure 44: Scenario 2 – Am losses



For americium losses, two groups can also be observed. The first one is formed by COSI (DECAY) and COSI (CUMUL), with very similar results. The difference between both curves comes from the decay of ^{241}Am ($T_{1/2}$ = 433 years). The second group of results includes FAMILY21, EVOLCODE and VISION. For these codes, the effective cooling time before reprocessing is larger than for the first group, leading to a higher accumulation of ^{241}Am by decay of ^{241}Pu before reprocessing. After reprocessing, the decay of ^{241}Am is taken into account by EVOLCODE, FAMILY, and COSI (DECAY), reducing the Am accumulation rate for these codes versus VISION, which decays spent fuel prior to reprocessing but does not decay HLW.

For neptunium losses, the results for COSI (DECAY), COSI (CUMUL) and VISION are very close. On the other hand, the results given by FAMILY21 and EVOLCODE are different from COSI and VISION, due to the larger amount of ^{241}Am decay to ^{237}Np as described in the previous paragraph. The impact of decay of ^{237}Np is negligible due to the very long period of ^{237}Np ($T_{1/2} = 2.14 \times 10^6$ years).

Figure 45: Scenario 2 – Np losses

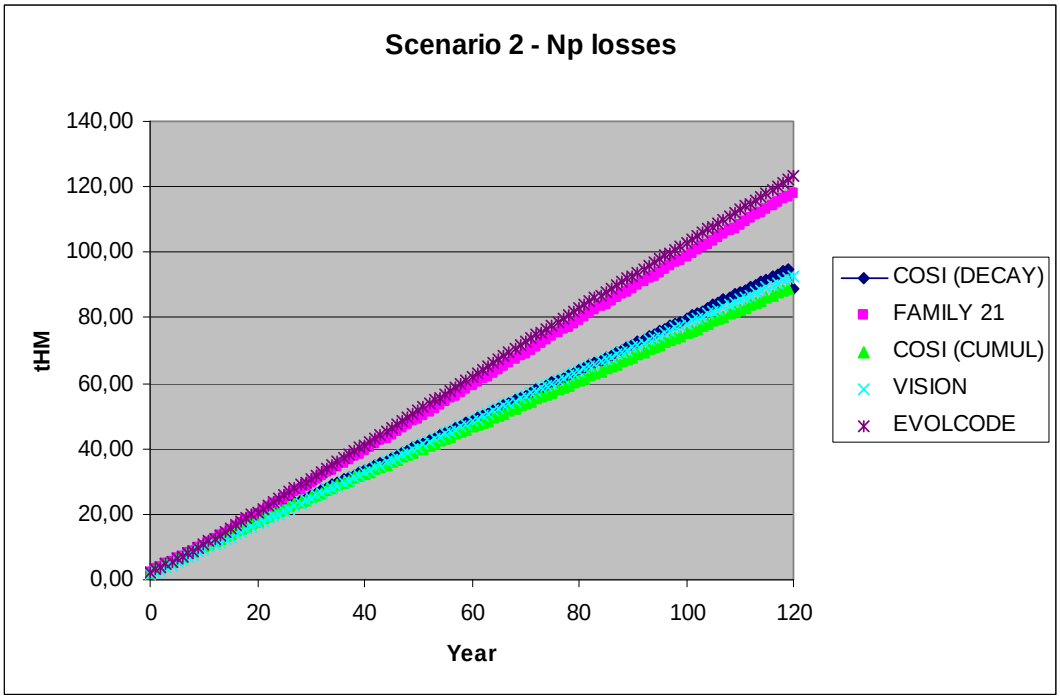
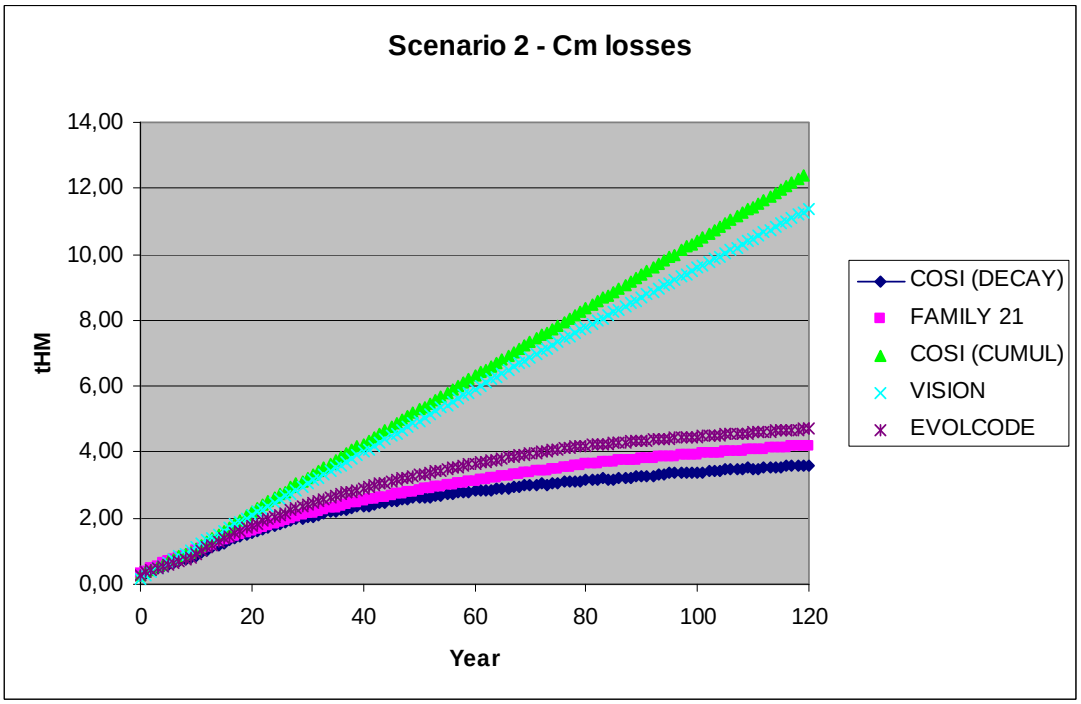


Figure 46: Scenario 2 – Cm losses



For the Cm losses, as for plutonium, the three codes with isotopic evolution of the wastes COSI (DECAY), FAMILY21, and EVOLCODE, approximately agree between themselves, whereas COSI (CUMUL) is closer to VISION. As mentioned above, the difference between both groups of codes is approximately equal to eight tonnes of Cm at the end of the scenario and represents the decay of ^{244}Cm into ^{240}Pu. The different creation of Cm at irradiation (and with minor importance the 7% lower spent fuel amount reprocessed each year in the FAMILY21 and COSI simulations) leads to the remaining differences between EVOLCODE, FAMILY21 and COSI (DECAY).

Figure 47: Scenario 2 – Reprocessed U

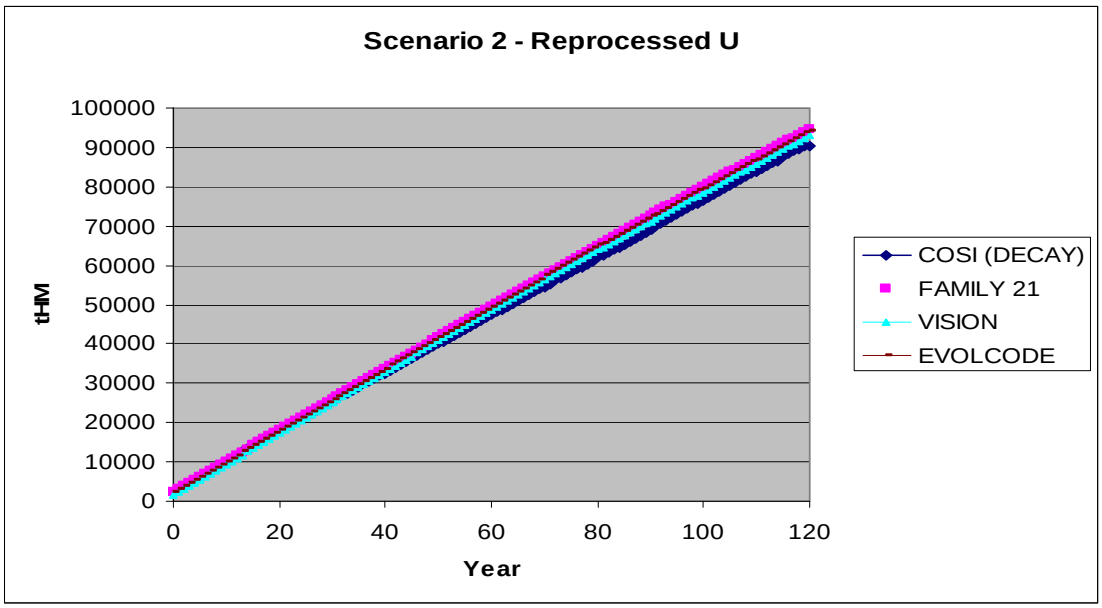
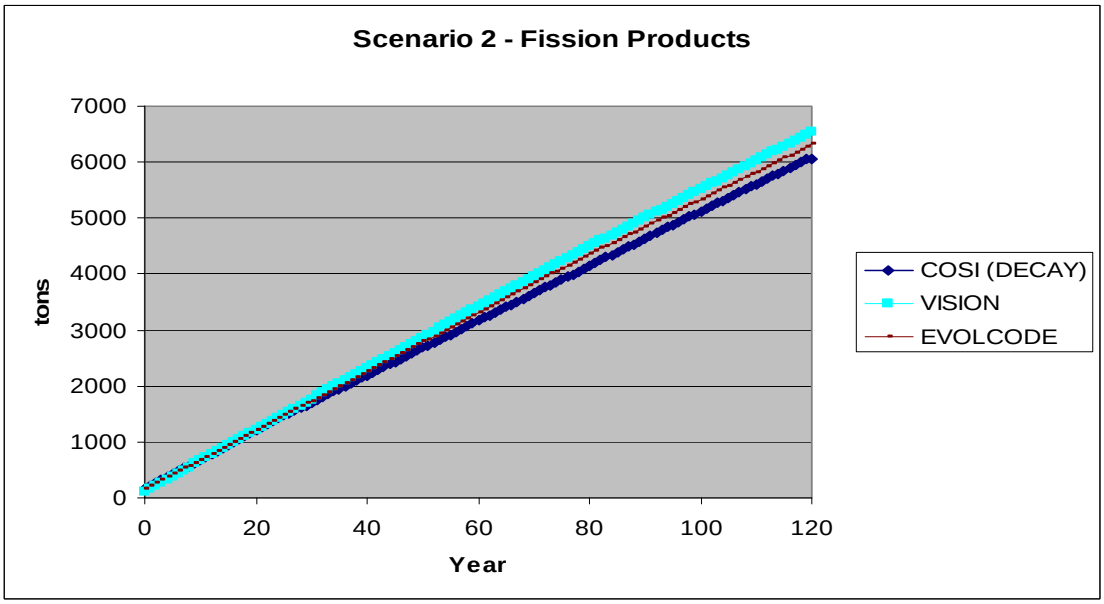


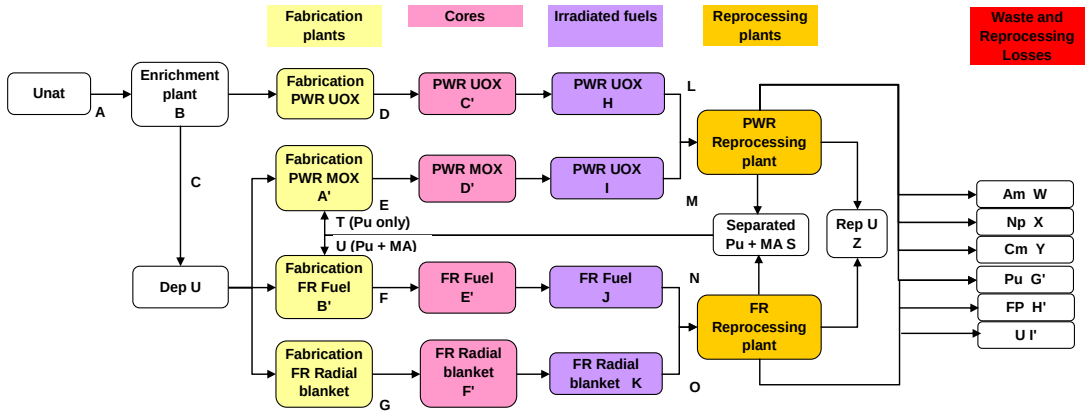
Figure 48: Scenario 2 – Fission products



The results on reprocessed uranium are similar for all four codes (maximum difference = 5%). The results on fission products are also similar (maximum difference = 10%).

4.2.3. *Scenario 3*

Figure 49: Scenario 3 – Flow chart



Five codes were used for scenario 3: COSI6, FAMILY21, DESAE2.2, EVOLCODE and VISION2.2. Figures 50 and 51 present the results on the LWR UOX and LWR MOX.

Figure 50: Scenario 3 – LWR UOX fabrication

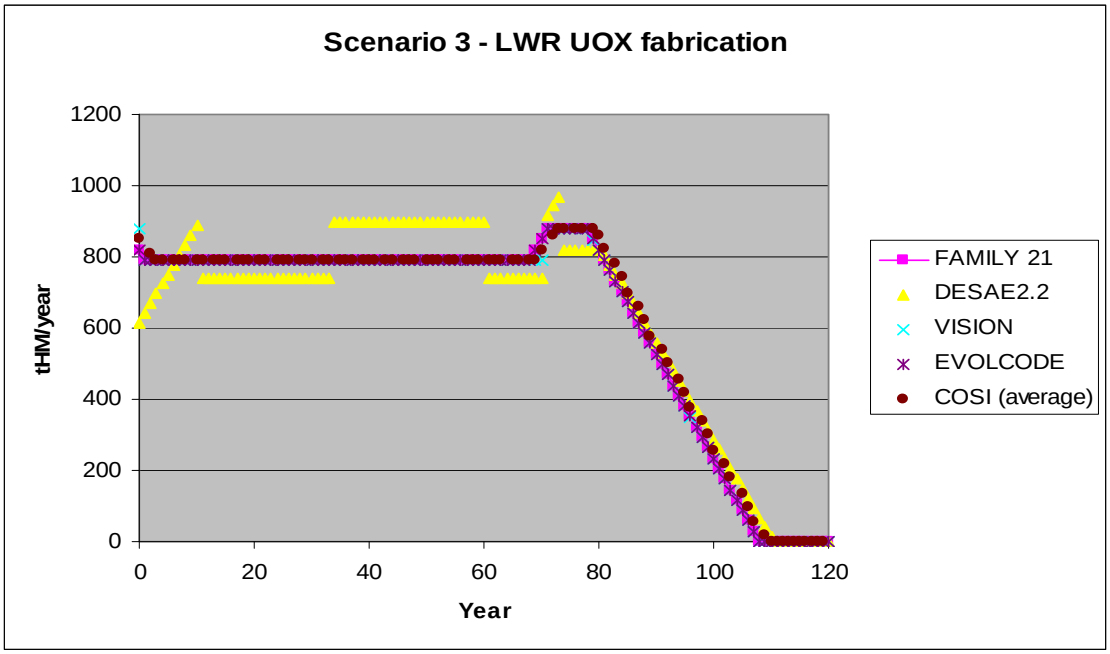
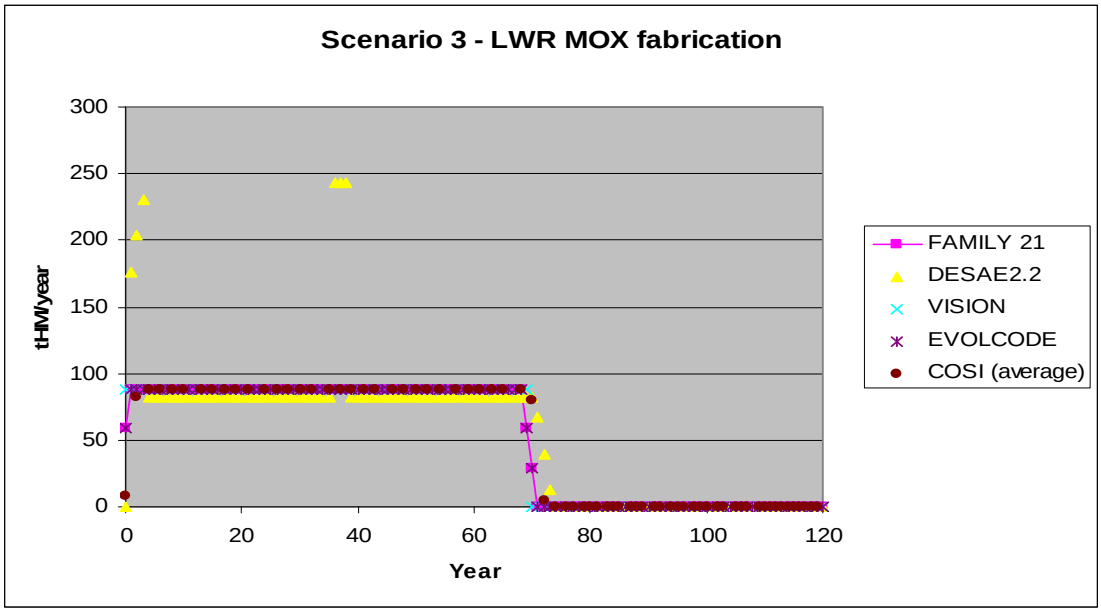


Figure 51: Scenario 3 – LWR MOX fabrication



Among all the contributing codes, COSI, FAMILY21, EVOLCODE and VISION calculate the same values for UOX and MOX fuel needs. For DESAE2.2, the patterns of LWR UOX needs are different, even though the total integrated fabrication across the full simulation is similar. An important difference appears from year 34 to year 60 but the reason for this difference was not identified. The reason could be the impact of the first core, which is taken into account in DESAE calculations, but in this case, the lifetime of the reactors would not be 60 years. For MOX, the three points around year 40 are not explained.

Figure 52: Scenario 3 – LWR UOX irradiated fuel inventory

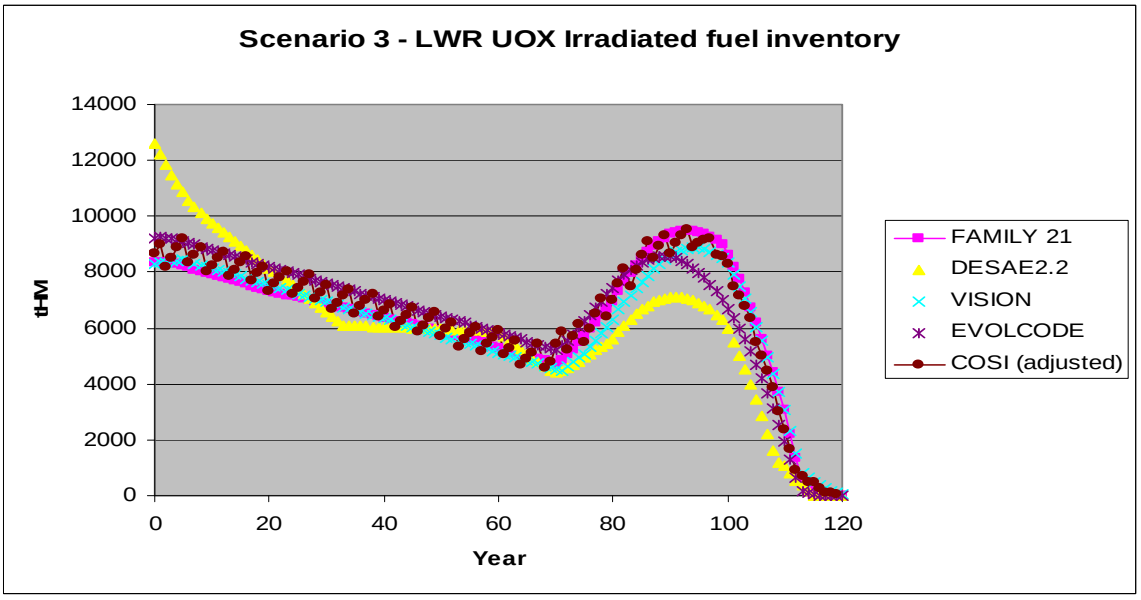


Figure 52 exhibits results for COSI using a different assumption for initial legacy spent fuel (COSI adjusted) compared to scenario 2. With this assumption, the results between COSI, VISION, FAMILY21 and EVOLCODE are very consistent, except for the small oscillations given by COSI, already explained in Section 4.2.1, scenario 1. The results given by DESAE are different and remain unexplained.

Figure 53: Scenario 3 – LWR MOX irradiated fuel inventory

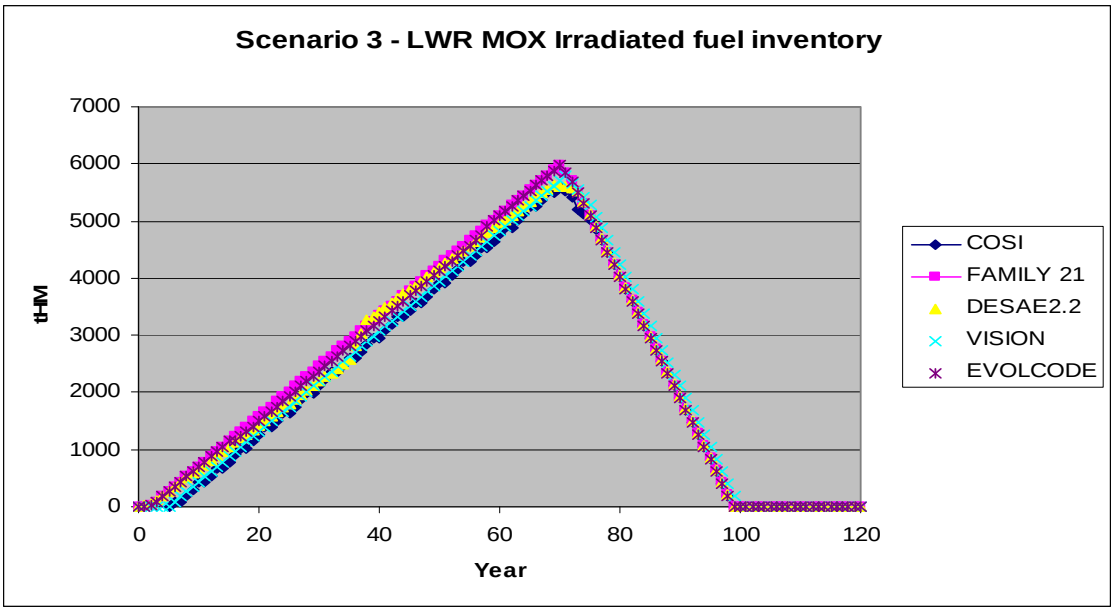
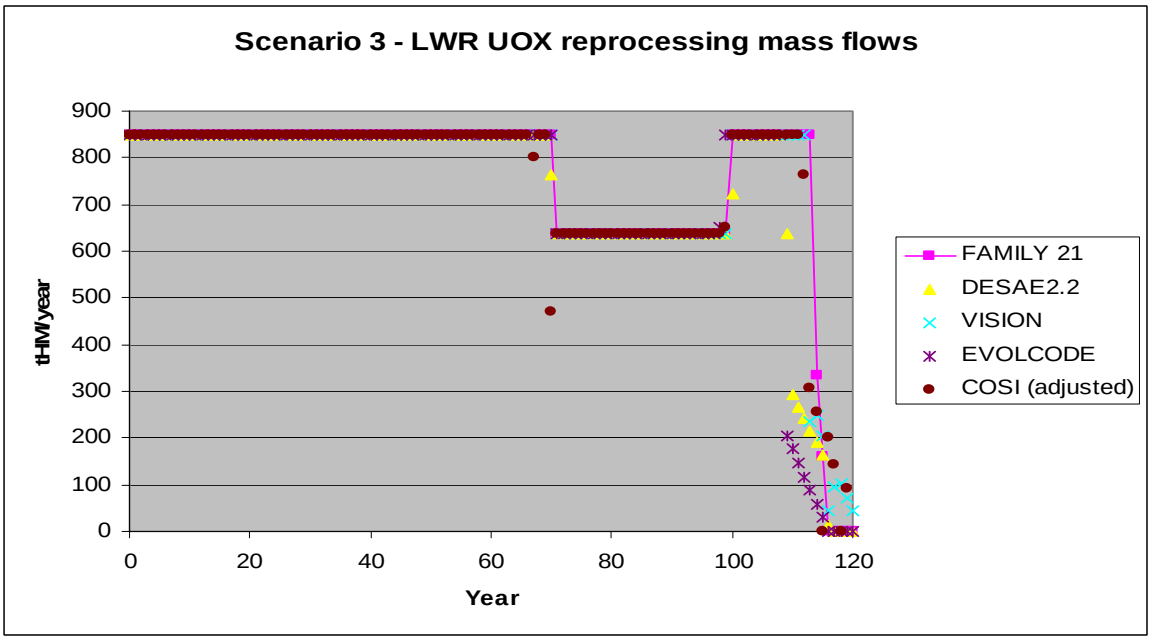


Figure 54: Scenario 3 – LWR UOX reprocessing mass flows



For the LWR MOX irradiated fuel inventory, the results are very consistent and again, the small differences in MOX spent fuel inventory are a consequence of differences between codes in the time of the first MOX unloading.

Figure 55: Scenario 3 – LWR MOX reprocessing mass flows

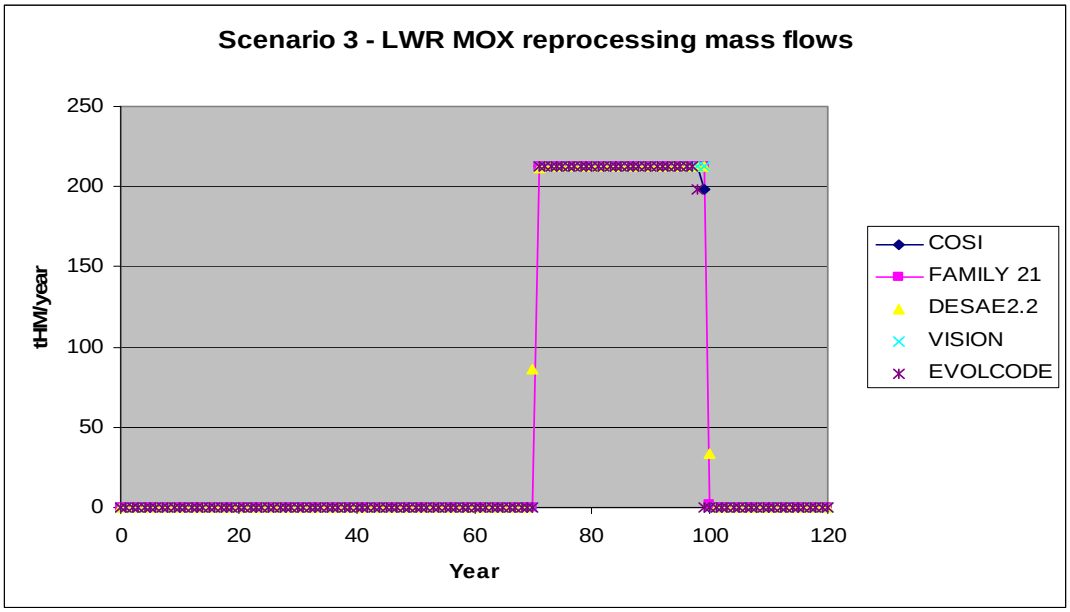
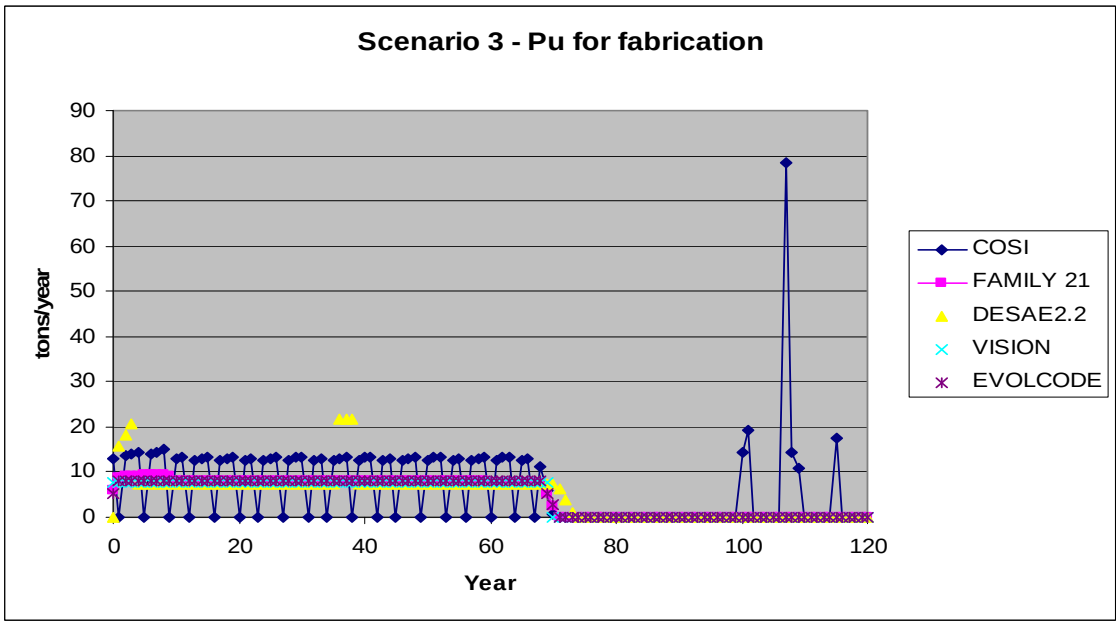


Figure 56: Scenario 3 – Pu for fabrication



For the LWR UOX and MOX reprocessing mass flow, when COSI uses the same assumptions as VISION (COSI adjusted), the results given by COSI, VISION, EVOLCODE, DESAE2.2 and FAMILY21 are consistent until year 110.

For the plutonium used for fabrication FAMILY21, DESAE2.2, VISION and EVOLCODE are in good agreement: between 7.8 and 8.4 tonnes per year. The annual average value given by COSI6 is around 9.5 tonnes per year. As for scenario 1, the oscillations are due to the individual tracking of each fuel batch and because all the reactors (1.5 GWe each) are simulated with one single fictive reactor having a variable power. After year 100, COSI uses additional Pu for FR fuel fabrication to compensate for a lack of TRU.

Table 21 gives the average plutonium fraction in PWR MOX fresh fuel. The average is calculated between year 4 and year 64.

Table 21: Scenario 3 – Plutonium fraction in PWR MOX fresh fuel: Annual average values

	COSI	FAMILY21	VISION	EVOLCODE	DESAE2.2
Annual Pu mass	9.45 tonnes	8.39 tonnes	7.83 tonnes	7.93 tonnes	7.32 tonnes
Average heavy metal mass	88.9 tonnes	87.84 tonnes	87.77 tonnes	87.83 tonnes	82.07 tonnes
Pu fraction	10.62 %	9.56 %	8.92 %	9.03 %	8.92%

As in scenario 2, no equivalent ^{239}Pu amount or initial fixed content of Pu has been proposed for PWR MOX fuel in the transition scenario part of the benchmark. The Pu fraction proposed by COSI is higher than other codes. In COSI, this fraction is calculated with an equivalence model taking into account Pu isotopic composition, final burn-up of the fuel and core fraction. The model has been calculated with the deterministic neutronic code APOLLO 1. For EVOLCODE, a fixed amount of 9.026% of Pu in the MOX fuel has been considered, independently of the isotopic composition (assumption for the depletion part of the benchmark for the fast reactor).

The second set of figures present the results on the fast reactors:

Figure 57: Scenario 3 – FR MOX + axial blanket fabrication

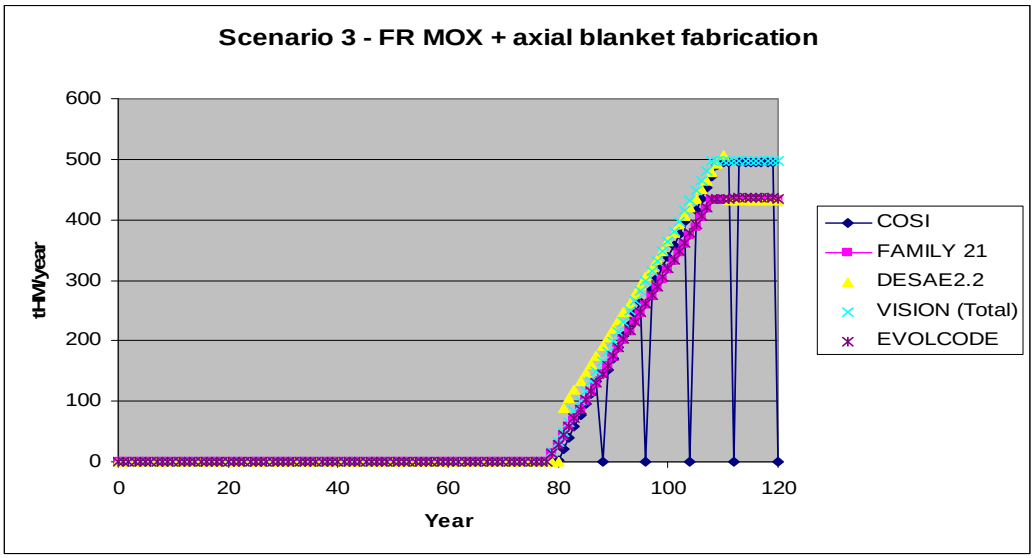
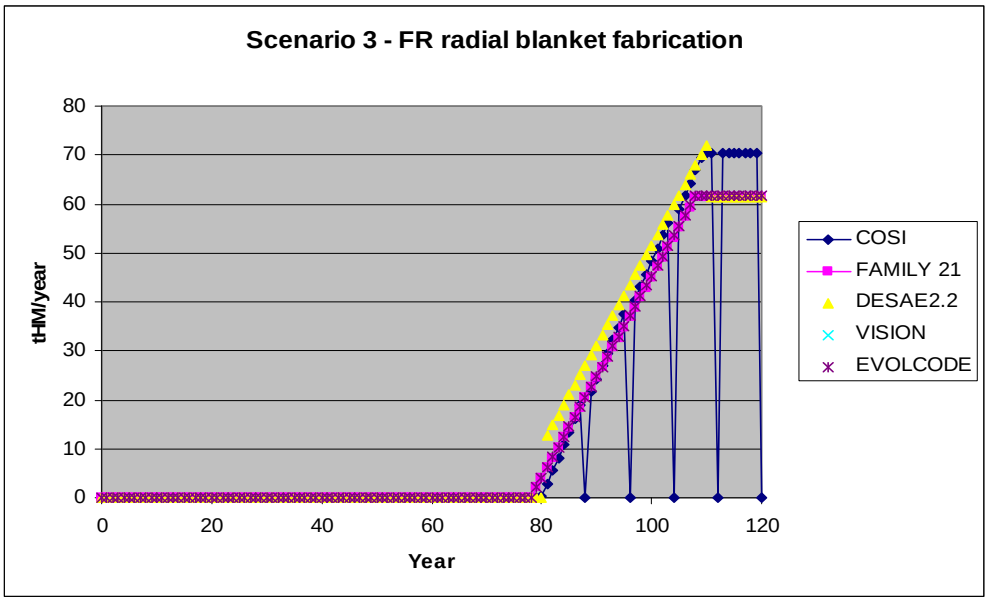


Figure 58: Scenario 3 – radial blanket fabrication



For fast reactors fuel and blankets fabrication, the apparent differences between codes can be explained. All the codes exhibit the same trend. However, COSI gives a value equal to 0 for the year 88, 96, 106, 112 and 120. In COSI, the whole fast reactor fleet is simulated with one single fictive reactor with a variable power. Due to the cycle length for fuel and blankets, which is different from one year, there is no reloading every 8 years in the fast reactor. It is also important to note that the version of VISION used in the benchmark could not separate fuel, axial blankets and radial blankets in the simulation and the results indicate the total of all the contributions. (A more recent version of VISION now has this capability.)

When using the annual average of the values after deployment of the reactors, there is only a difference of 1% between all the codes for FR MOX + axial blanket, and again 1% for the radial blanket. Table 22 summarises the average annual values during the equilibrium period after 2100.

Table 22: Scenario 3 – Fast reactor annual average fabrication results

	COSI	FAMILY21	VISION	EVOLCODE	DESAE2.2
Annual average MOX + axial blanket fabrication after 2100	435.0 tonnes	431.8 tonnes	-	435.4 tonnes	431.8 tonnes
Annual average radial blanket fabrication after 2100	61.8 tonnes	61.3 tonnes	-	61.8 tonnes	61.3 tonnes
Total	496.8 tonnes	493.1 tonnes	497.4 tonnes	497.2 tonnes	493.1 tonnes

Figure 59: Scenario 3 – FR MOX + axial blankets reprocessing mass flows

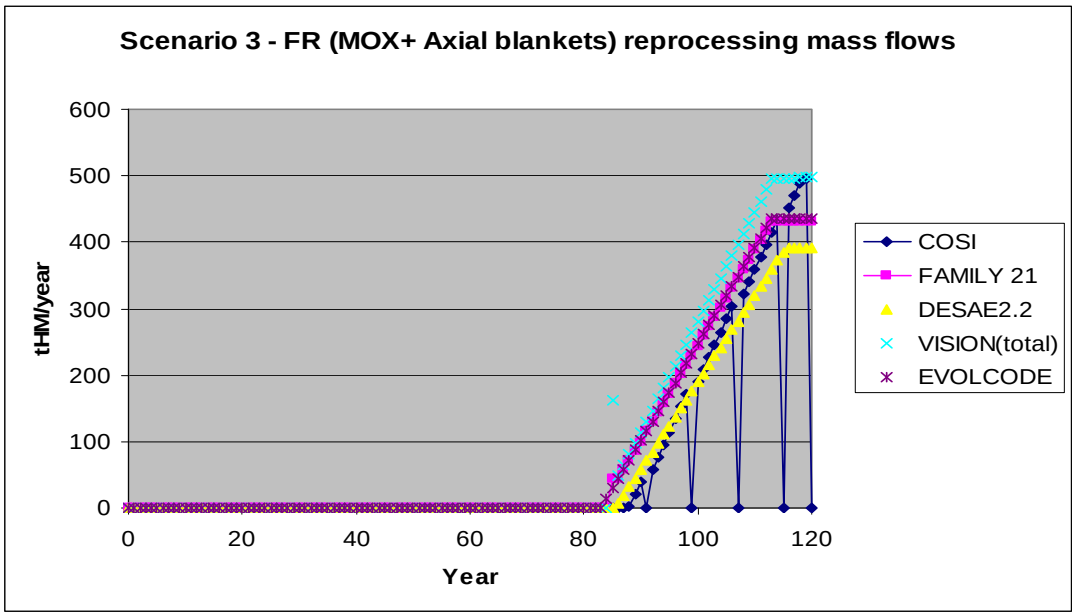
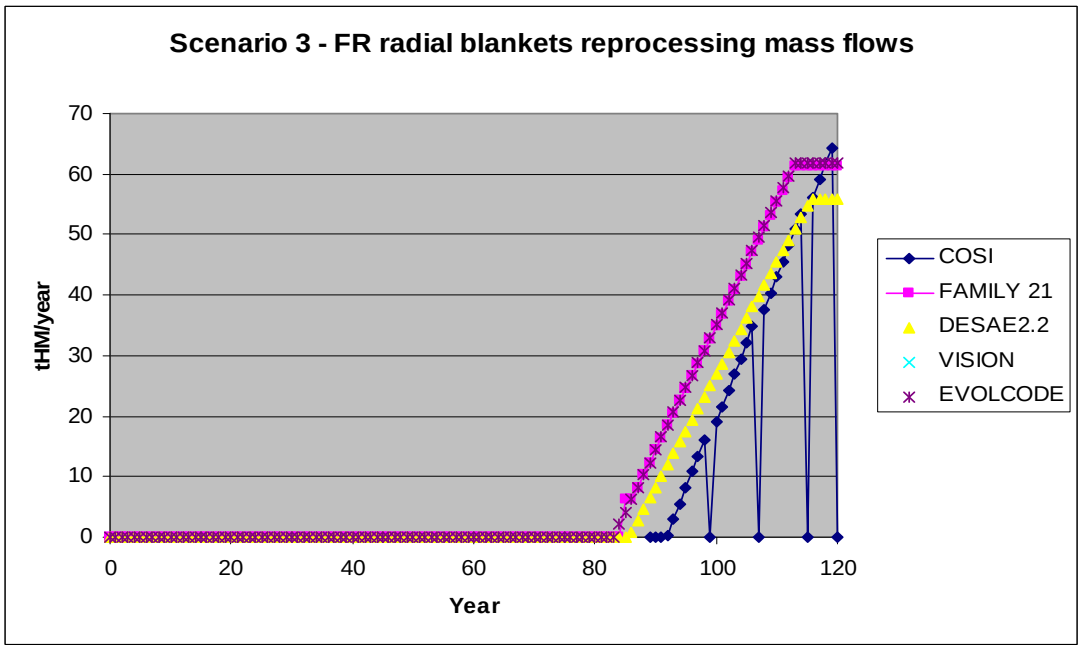


Figure 60: Scenario 3 – FR radial blankets reprocessing mass flows



For fast reactors fuel and blankets reprocessing, the apparent differences between codes are important and cannot be neglected. However, all the codes exhibit the same trend. COSI6 gives a value equal to 0 every 8 years, as was the case for fabrication and for the same reasons. As for fabrication, Table 23 summarises the average annual values for the equilibrium period:

Table 23: Scenario 3 – Fast reactor spent fuel - annual average reprocessing results

	COSI (year 119)	FAMILY21	VISION	EVOLCODE	DESAE2.2
Annual average MOX + axial blanket reprocessing after 2100	435.0 tonnes	431.5 tonnes	-	435.4 tonnes	392.6 tonnes
Annual average radial blanket reprocessing after 2100	56.5 tonnes	61.3 tonnes	-	61.8 tonnes	55.8 tonnes
Total	491.5 tonnes	492.8 tonnes	497.3 tonnes	497.2 tonnes	448.3 tonnes

COSI, FAMILY21, VISION and EVOLCODE give close values for FR spent fuel reprocessing, in good agreement with the average annual values of the FR fuel fabrication for the equilibrium period (Table 22). DESAE2.2 gives lower values.

Figure 61: Scenario 3 – FR MOX + axial blanket irradiated fuel inventory

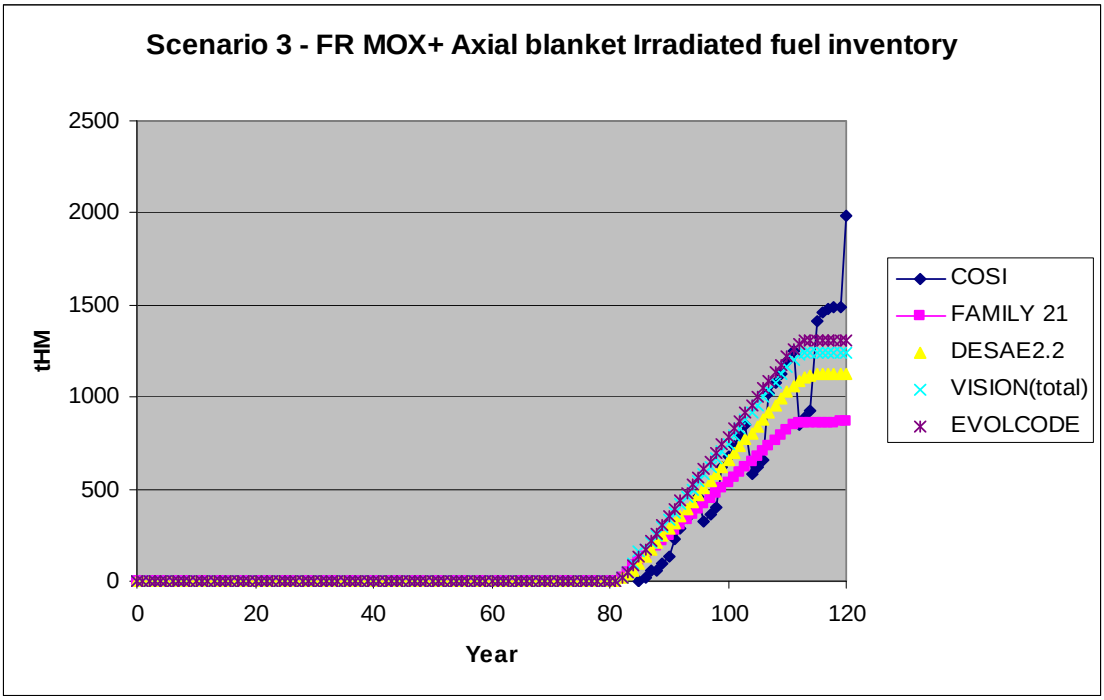
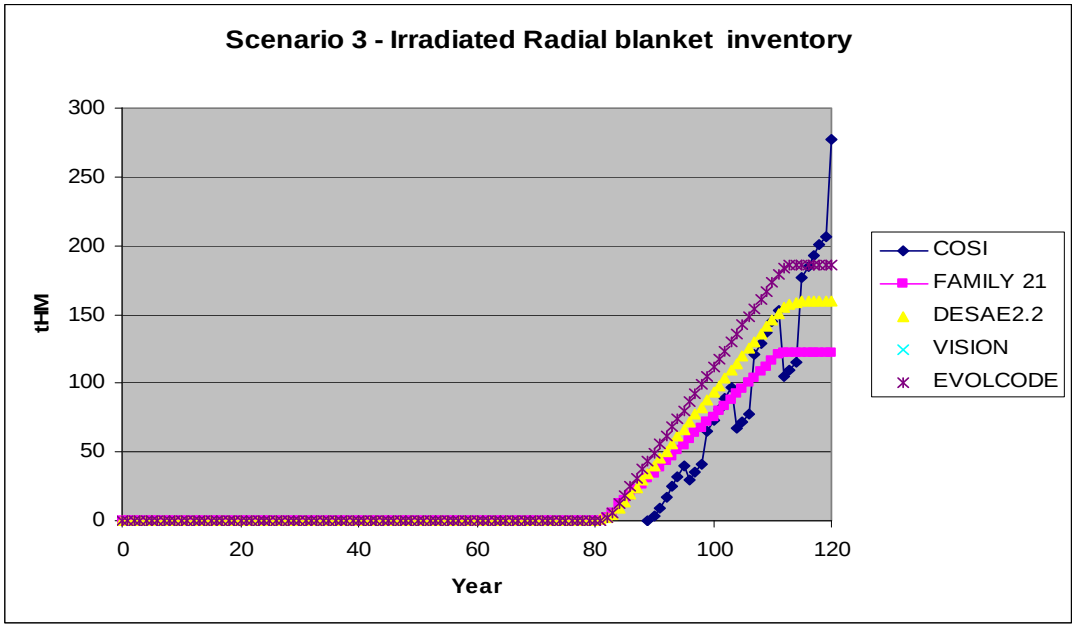


Figure 62: Scenario 3 – Irradiated radial blanket inventory



For the fast reactors irradiated fuel inventory, the discrete values given by COSI exhibit the same trend as FAMILY21, VISION, EVOLCODE2 and DESAE2.1. However, the values for the equilibrium period are very different. The irradiated fuel inventory is the difference between the fuel discharged from the reactors and the fuel reprocessed. The mass of fuel discharged from the reactors is the same as the mass of fuel leaving the fabrication plant with a delay due to the irradiation in the reactor. For the equilibrium period, these values are the same and the irradiated fuel inventory remains constant. Thus, the differences observed in the irradiated fuel inventory at year 120 are due to the transition period. Finally, the difference between cumulated fabrication and cumulated reprocessing represents the total "irradiated fuel + fuel in the reactors". Here again, the important differences are due to the transition period. Once again, we have to keep in mind that VISION2.2 could not separate fuel, axial blankets and radial blankets in the simulation and the results indicate the total of all the contributions.

Table 24: Scenario 3 – Fast reactors fuel + axial blankets – cumulated fabrication and reprocessing

	COSI	FAMILY21	VISION	EVOLCODE	DESAE2.2
(1) Cumulated fabrication at year 120	11 095 tonnes	11 874 tonnes	13 678 tonnes	11 974 tonnes	13 274 tonnes
(2) Cumulated reprocessing at year 120	7 130 tonnes	9 700 tonnes	11 240 tonnes	9 797 tonnes	7 831 tonnes
(1) – (2)	3 965 tonnes	2 174 tonnes	2 437 tonnes	2 177 tonnes	5 442 tonnes

Figure 63: Scenario 3 – TRU for fabrication

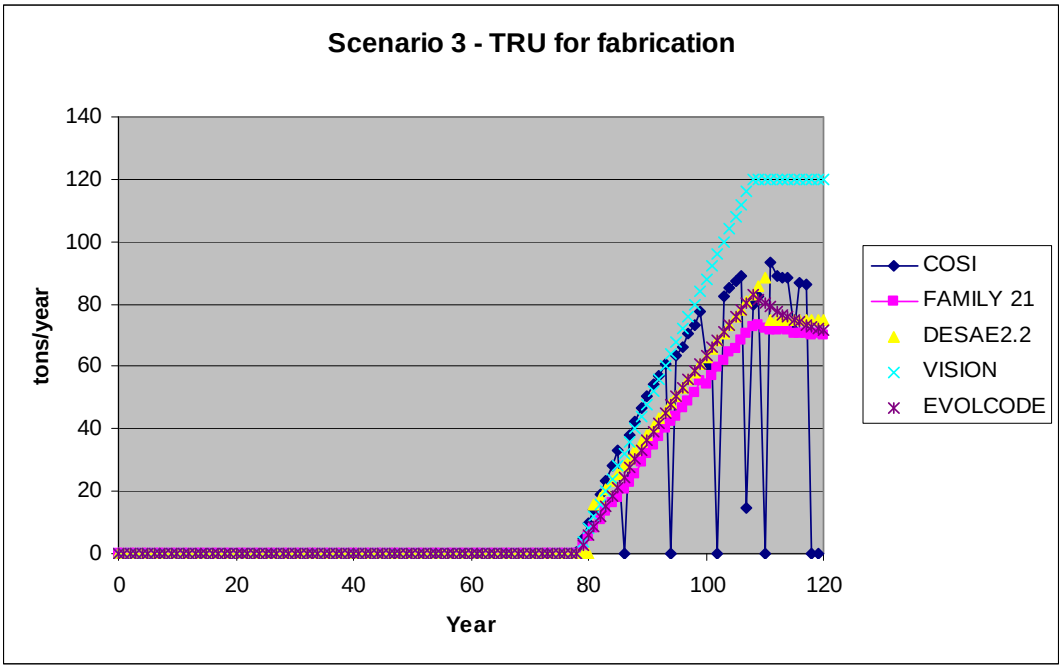
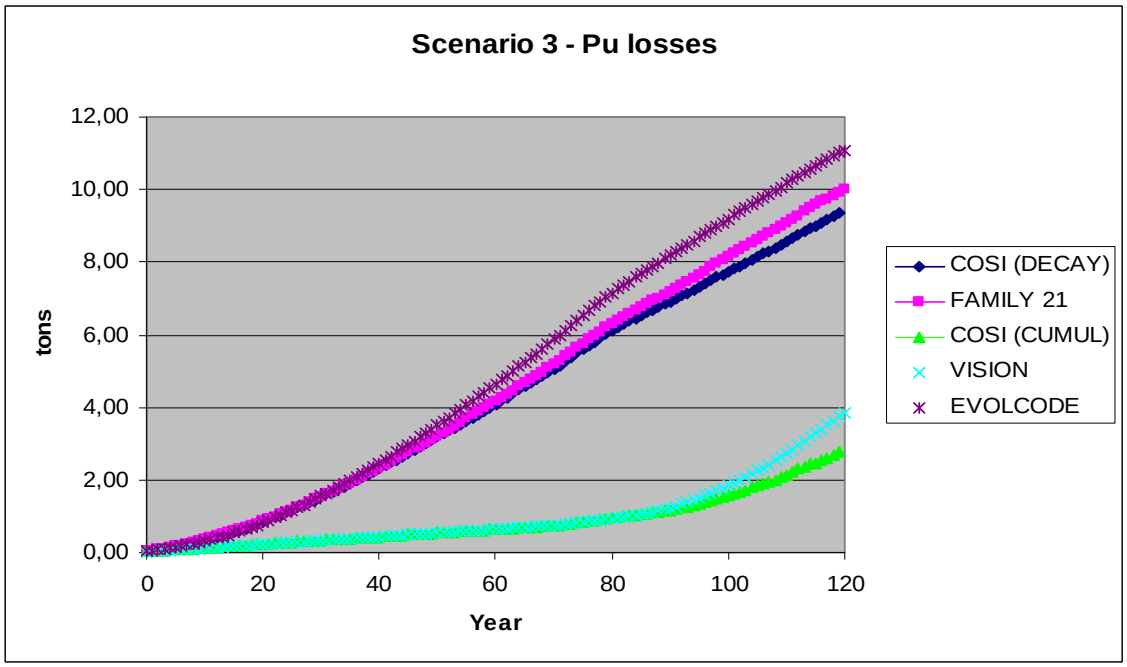


Figure 64: Scenario 3 – Pu losses



For the TRU for fabrication, the trends are the same but the values are very different, particularly between VISION and the other codes. In COSI and EVOLCODE, the TRU mass is calculated with an equivalence model using:

- the equivalence reactivity coefficients described in Table 18;
- the equivalent ^{239}Pu fraction given in Table 14.

However, the results between COSI and EVOLCODE are different, probably because of a difference in TRU isotopic composition used for fabrication. These values have been calculated by the neutronic package ERANOS dedicated to fast reactor neutronic calculations. The following set of figures presents actinides losses and reprocessed uranium, except for DESAE, which cannot simulate reprocessing losses.

Figure 65: Scenario 3 – Am losses

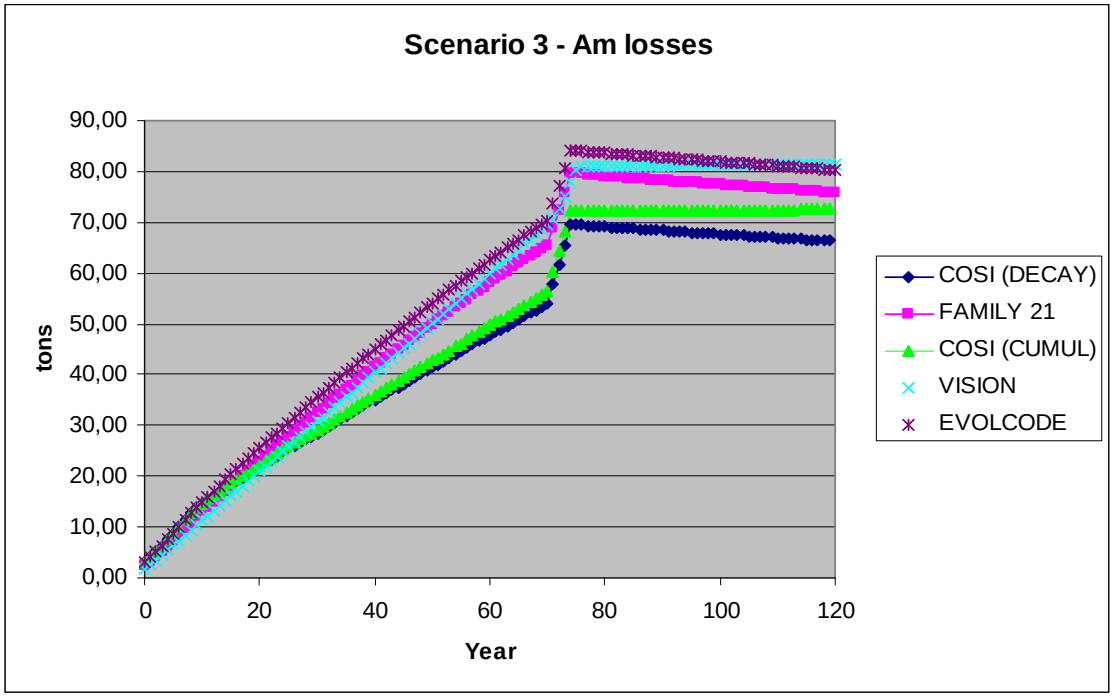


Figure 66: Scenario 3 – Np losses

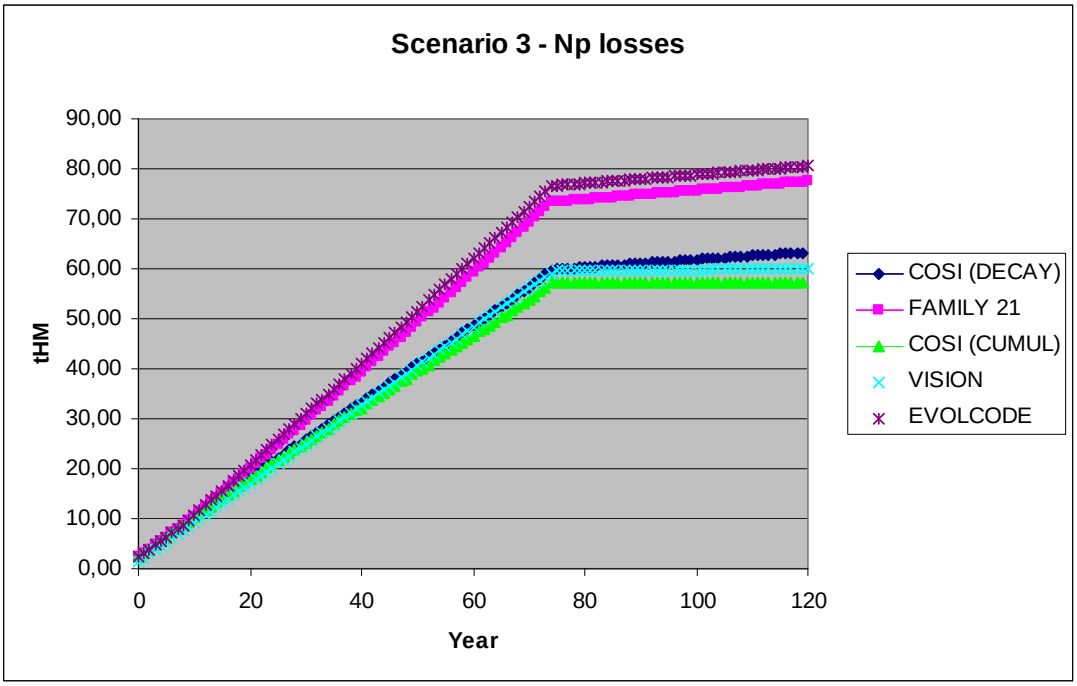


Figure 67: Scenario 3 – Cm losses

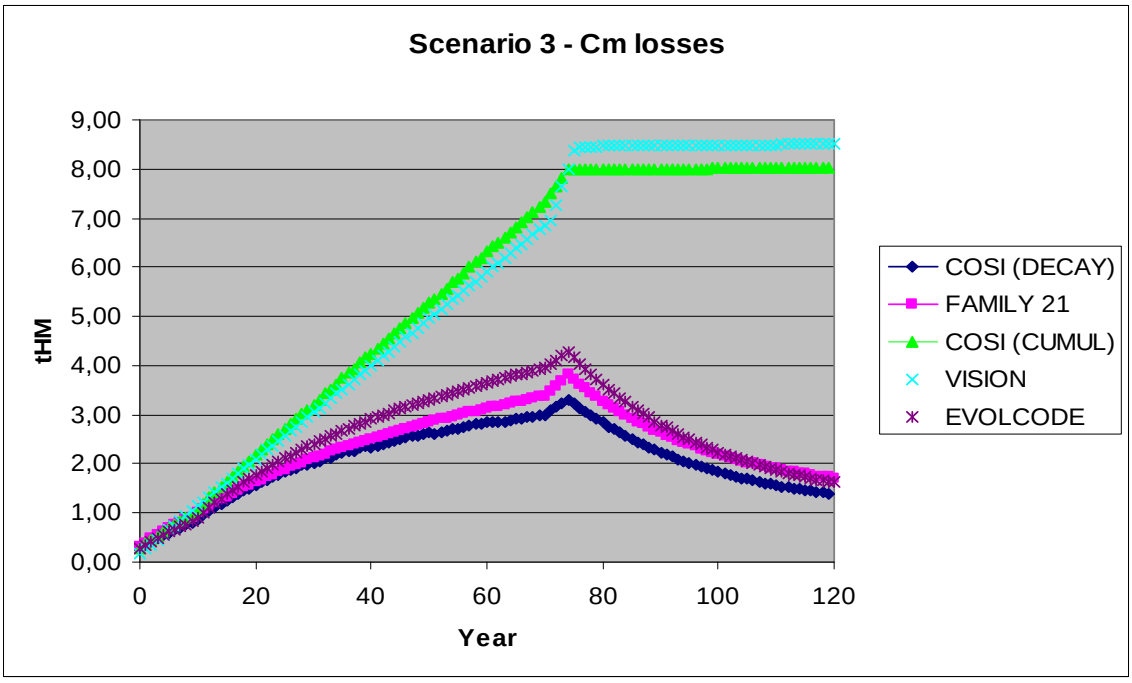
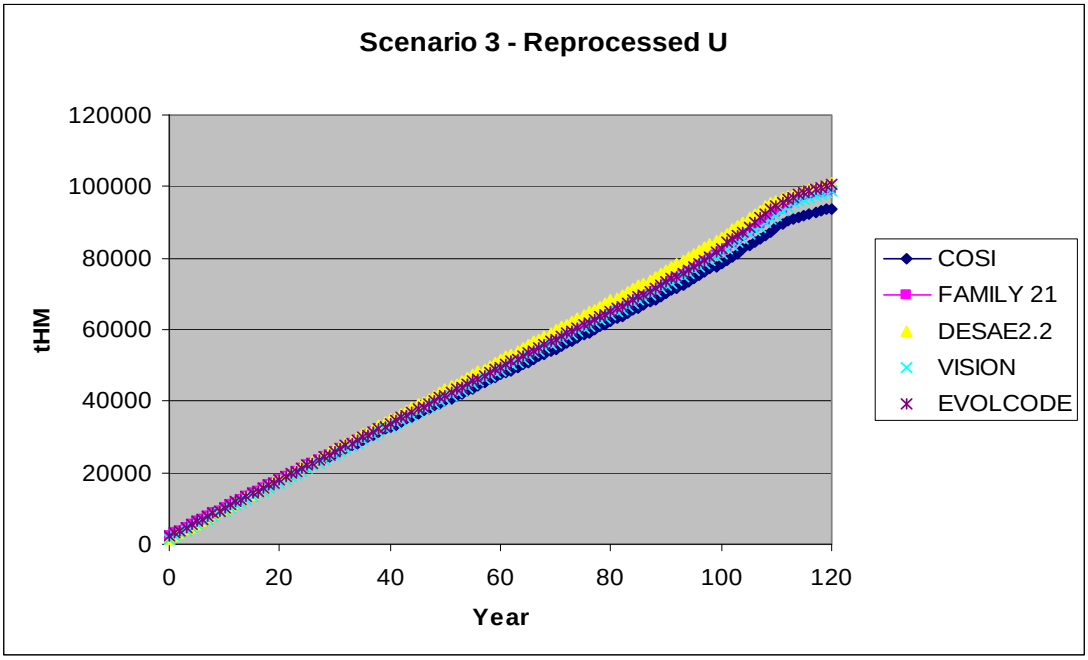


Figure 68: Scenario 3 – Reprocessed U



As for plutonium losses, minor actinides losses and reprocessed U, the trends observed are exactly the same as scenario 2 and the analysis leads to the same conclusions (cf 4.2.2 scenario 2). The only difference is due to the addition of advanced reprocessing beginning in year 75, causing the change in slopes for the minor actinide curves.

5. Conclusion

The benchmarking results and the analysis of the calculations lead to the following conclusions:

1) The general trends observed for each code are the same for the 3 scenarios calculated in the benchmark.

2) All the scenario codes give very close results for scenario 1. However, this scenario is very simple and cannot serve as a reference for the comparison of the codes.

3) When the level of complexity of the scenario increases, some differences appear. The comparison of the results on the second scenario demonstrates the importance of initial assumptions and the common interpretation of the hypotheses and results in the comparisons.

4) A tuning of the assumptions is always necessary. This necessity is due to the difference of interpretation for initial conditions and to some missing assumptions which may appear. Thus, several iterations can be necessary to converge results.

5) After the tuning iterations were completed, some differences remained, due to:

- The capacity of modelling of the codes. Particularly, the loading of the fuel batches is annually averaged for all codes except COSI6, which treats discrete batches (Figure 29), and can lead to a more complex comparison of the results.
- The transition period in scenario 3: Figures 57 and 58, Tables 22 and 24 indicate that the results are very close for the equilibrium period but some differences appear during the deployment of fast reactors. A reduction of these differences would have necessitated several more iterations.
- The difference in physical models: how the decay of heavy nuclides is taken into account in the interim storages (Figure 43 to Figure 46), differences in depletion calculations (Section 4.1. Depletion), and differences in equivalence calculations for Pu and minor actinides fraction in the fresh fuel (Figure 42).
- The simulation of the heterogeneous cores of fast reactors, which was not possible in VISION2.2.
- The flexibility offered by the codes to simulate or not the first cores and the replacement of retiring reactors (case of DESAE2.2, Figure 50).
- Some remaining unexplained behaviours (case of DESAE2.2).

Some useful capability additions would be the simulation of the decay of nuclear materials in the separated materials and wastes (case of VISION), the simulation of TRU losses from reprocessing (case of DESAE2.2), the simulation of annual averaged batches of fuel (case of COSI6), and the use of equivalence models for the calculation of Pu or TRU content in the fresh fuel of LWR MOX and fast reactors (case of EVOLCODE, FAMILY21, VISION and DESAE2.2).

Also, some differences will remain whatever the level of the analysis and the number of iterations, because of the different calculation methods used.

This benchmark was limited to comparison in heavy elements material flows. A comparison for isotopes would probably have led to other differences and would have necessitated a more detailed investigation on the physical models used by the codes.

Finally, we note that a similar benchmarking effort organised by MIT (MIT-NFC-TR-105, April 2009), involving COSI6, VISION2.2, CAFCA and DANESS, led to the same type of conclusions.

Appendix A: Results of the depletion part

Table A.1: Depletion part – results of PWR UOX calculations (g/tihm)

Nuclide	Initial	SCK		JNC	CEA	KIT	SCK / CEA	JNC / CEA	KIT / CEA
	g/t iHM	g/t iHM	g/t iHM						
		EOL	5 y cooling	5 y cool.	5 y cool.	5 y cool.			
U232	0.00E+00	1.90E-03	4.69E-03		3.70E-03				
U234	4.45E+02	1.93E+02	2.13E+02	2.13E+02	2.17E+02	2.07E+02	-2.07	-2.03	-4.47
U235	4.95E+04	8.06E+03	8.06E+03	7.93E+03	7.59E+03	7.26E+03	6.16	4.54	-4.36
U236	0.00E+00	6.86E+03	6.86E+03	6.51E+03	6.88E+03	6.84E+03	-0.24	-5.37	-0.52
U238	9.50E+05	9.08E+05	9.08E+05	9.09E+05	9.09E+05	9.09E+05	-0.10	0.02	-0.01
U	1.00E+06	9.23E+05	9.23E+05	9.24E+05	9.24E+05	9.23E+05	-0.08	-0.02	-0.09
Pu236	0.00E+00	4.26E-03	1.27E-03		9.65E-04	0.00E+00	31.60		-100.00
Pu238	0.00E+00	4.86E+02	4.99E+02	4.50E+02	4.92E+02	4.94E+02	1.50	-8.51	0.30
Pu239	0.00E+00	6.62E+03	6.71E+03	6.51E+03	6.37E+03	6.36E+03	5.35	2.20	-0.14
Pu240	0.00E+00	3.12E+03	3.15E+03	3.06E+03	3.11E+03	3.11E+03	1.25	-1.59	0.10
Pu241	0.00E+00	2.08E+03	1.63E+03	1.61E+03	1.54E+03	1.56E+03	5.99	4.66	1.49
Pu242	0.00E+00	1.08E+03	1.08E+03	1.05E+03	1.13E+03	1.12E+03	-4.78	-7.52	-0.53
Pu	0.00E+00	1.34E+04	1.31E+04	1.27E+04	1.26E+04	1.27E+04	3.71	0.62	0.40
Am241	0.00E+00	7.53E+01	5.17E+02	5.24E+02	4.95E+02	4.97E+02	4.46	5.96	0.36
Am242M	0.00E+00	2.71E+00	2.65E+00	1.27E+00	1.35E+00	0.00E+00	96.52	-6.11	-100.00
Am243	0.00E+00	2.99E+02	2.99E+02	2.69E+02	2.94E+02	3.09E+02	1.66	-8.58	5.10
Am	0.00E+00	3.77E+02	8.19E+02	7.95E+02	7.90E+02	8.07E+02	3.62	0.57	2.11
Np237	0.00E+00	9.20E+02	9.36E+02	8.76E+02	9.16E+02	9.12E+02	2.23	-4.34	-0.46
Np	0.00E+00	1.01E+03	9.36E+02	8.76E+02	9.16E+02	9.12E+02	2.23	-4.34	-0.46
Cm242	0.00E+00	3.07E+01	1.96E-02	1.62E-02	1.69E-02	1.80E-02	16.11	-4.14	6.51
Cm243	0.00E+00	9.86E-01	8.73E-01	8.11E-01	1.73E+00	1.00E+00	-49.52	-53.13	-42.08
Cm244	0.00E+00	1.56E+02	1.29E+02	1.05E+02	1.21E+02	1.31E+02	6.54	-13.47	7.85
Cm245	0.00E+00	1.31E+01	1.31E+01	1.05E+01	1.05E+01	1.25E+01	24.83	-0.04	18.67
Cm246	0.00E+00	1.66E+00	1.66E+00	1.32E+00	1.46E+00	1.75E+00	13.71	-9.38	19.73
Cm247	0.00E+00	3.22E-02	3.22E-02			3.00E-02			
Cm248	0.00E+00	2.85E-03	2.85E-03			3.00E-03			
Cm	0.00E+00	2.03E+02	1.45E+02	1.17E+02	1.35E+02	1.46E+02	7.12	-13.07	7.93

Table A.2: Depletion part – results of PWR MOX calculations (g/tihm)

Nuclide		SCK		JNC	CEA	KIT	SCK / CEA	JNC / CEA	KIT / CEA
	g/t iHM	g/t iHM	g/t iHM						
	BOL	EOL	5 y cool.	5 y cool.	5 y cool.	5 y cool.			
U232	0.00E+00	7.78E-04	2.00E-03		1.61E-03	1.00E-03	24.26		-37.89
U234	0.00E+00	7.34E+01	1.94E+02	1.92E+02	1.95E+02	1.96E+02	-0.72	-1.47	0.31
U235	2.06E+03	8.95E+02	8.98E+02	8.35E+02	9.08E+02	8.79E+02	-1.10	-8.06	-3.15
U236	0.00E+00	2.65E+02	2.74E+02	2.53E+02	2.69E+02	2.70E+02	1.91	-5.98	0.33
U238	9.08E+05	8.69E+05	8.69E+05	8.83E+05	8.68E+05	8.69E+05	0.13	1.67	0.16
U	9.10E+05	8.70E+05	8.70E+05	8.84E+05	8.69E+05	8.71E+05	0.17	1.70	0.21
Pu236	0.00E+00	1.87E-03	5.56E-04		4.64E-04	0.00E+00	19.75		
Pu238	2.82E+03	2.89E+03	3.04E+03	2.88E+03	3.09E+03	3.10E+03	-1.75	-6.77	0.16
Pu239	4.66E+04	2.24E+04	2.25E+04	1.26E+04	2.39E+04	2.17E+04	-5.96	-47.19	-9.25
Pu240	2.20E+04	1.79E+04	1.82E+04	1.63E+04	1.79E+04	1.82E+04	1.44	-9.03	1.73
Pu241	1.06E+04	1.10E+04	8.66E+03	7.55E+03	9.14E+03	8.60E+03	-5.29	-17.40	-5.91
Pu242	7.26E+03	8.60E+03	8.60E+03	8.18E+03	7.92E+03	8.80E+03	8.56	3.30	11.10
Pu	8.92E+04	6.28E+04	6.09E+04	4.75E+04	6.20E+04	6.04E+04	-1.73	-23.36	-2.58
Am241	1.07E+03	1.19E+03	3.52E+03	3.14E+03	3.82E+03	3.52E+03	-7.78	-17.67	-7.75
Am242M	0.00E+00	3.46E+01	3.38E+01	3.48E+01	3.91E+01	2.19E+01	-13.45	-10.94	-43.96
Am243	0.00E+00	2.30E+03	2.30E+03	3.20E+03	2.38E+03	2.43E+03	-3.18	34.43	2.27
Am	1.07E+03	3.53E+03	5.86E+03	6.38E+03	6.24E+03	5.98E+03	-6.08	2.23	-4.17
Np237	0.00E+00	2.13E+02	2.34E+02	1.59E+02	2.67E+02	2.32E+02	-12.41	-40.45	-13.15
Np	0.00E+00	2.84E+02	2.34E+02	1.59E+02	2.67E+02	2.32E+02	-12.41	-40.45	-13.15
Cm242	0.00E+00	2.68E+02	1.97E-01	2.09E-01	2.13E-01	1.78E-01	-7.53	-1.97	-16.43
Cm243	0.00E+00	1.45E+01	1.28E+01	1.40E+01	1.60E+01	1.33E+01	-19.94	-12.67	-16.69
Cm244	0.00E+00	1.64E+03	1.35E+03	1.48E+03	1.43E+03	1.38E+03	-5.45	3.19	-3.50
Cm245	0.00E+00	2.30E+02	2.30E+02	1.45E+02	2.78E+02	2.30E+02	-17.39	-47.72	-17.45
Cm246	0.00E+00	1.94E+01	1.94E+01	1.70E+01	2.31E+01	1.98E+01	-16.22	-26.46	-14.16
Cm247	0.00E+00	5.25E-01	5.25E-01			4.86E-01			
Cm248	0.00E+00	4.09E-02	4.09E-02			3.40E-02			
Cm	0.00E+00	2.17E+03	1.61E+03	1.65E+03	1.75E+03	1.64E+03	-7.73	-5.59	-6.11

Table A.3: Depletion part – results of FR MOX calculations (g/tihm)

Nuclide	JNC	SCK	CEA	KIT	SCK / CEA	JNC / CEA	KIT / CEA
U232		4.037E-02	5.735E-02	6.000E-03	-29.6		-89.5
U234	5.408E+02	2.244E+02	5.663E+02	5.348E+02	-60.4	-4.5	-5.6
U235	6.769E+02	7.491E+02	6.441E+02	6.973E+02	16.3	5.1	8.3
U236	3.108E+02	2.622E+02	6.441E+02	3.017E+02	-59.3	-51.7	-53.2
U238	6.461E+05	6.601E+05	6.442E+05	6.505E+05	2.5	0.3	1.0
U	6.476E+05	6.613E+05	6.461E+05	6.521E+05	2.4	0.2	0.9
Pu236	0.000E+00	8.330E-02	1.462E-02	2.000E-03	470.0		-86.3
Pu238	7.374E+03	7.543E+03	8.056E+03	7.409E+03	-6.4	-8.5	-8.0
Pu239	1.014E+05	1.041E+05	1.006E+05	9.893E+04	3.4	0.8	-1.7
Pu240	6.807E+04	6.779E+04	7.028E+04	6.560E+04	-3.5	-3.1	-6.7
Pu241	8.083E+03	8.678E+03	8.045E+03	6.528E+03	7.9	0.5	-18.9
Pu242	9.930E+03	9.827E+03	1.008E+04	9.573E+03	-2.5	-1.4	-5.0
Pu	1.949E+05	1.979E+05	1.971E+05	1.880E+05	0.4	-1.1	-4.6
Am241	6.897E+03	4.878E+03	6.475E+03	6.387E+03	-24.7	6.5	-1.4
Am242M	5.325E+02	5.704E+02	4.304E+02	2.722E+02	32.5	23.7	-36.8
Am243	3.969E+03	3.967E+03	3.575E+03	4.140E+03	11.0	11.0	15.8
Am	1.140E+04	9.417E+03	1.048E+04	1.080E+04	-10.1	8.8	3.0
Np237	2.372E+03	2.805E+03	2.481E+03	2.773E+03	13.1	-4.4	11.8
Np	2.372E+03	2.989E+03	2.481E+03	2.773E+03	20.5	-4.4	11.8
Cm242	1.542E+00	3.805E+02	1.269E+00	8.830E-01	29886.8	21.5	-30.4
Cm243	5.184E+01	5.312E+01	7.005E+01	4.684E+01	-24.2	-26.0	-33.1
Cm244	3.836E+03	4.468E+03	4.039E+03	3.575E+03	10.6	-5.0	-11.5
Cm245	1.036E+03	1.087E+03	8.002E+02	1.043E+03	35.8	29.4	30.4
Cm246	3.374E+02	3.756E+02	8.380E+01	3.701E+02	348.2	302.6	341.7
Cm247	0.000E+00	4.007E+01	0.000E+00	4.116E+01			
Cm248	0.000E+00	3.824E+00	0.000E+00	2.728E+00			
Cm	5.263E+03	6.408E+03	4.994E+03	5.080E+03	28.3	5.4	1.7

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Appendix A: Results of the depletion part

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ROMANELLO, Vincenzo	FZK
SCHWENK-FERRERO, Aleksandra	FZK
JAPAN	
ONO, Kiyoshi	JAEA
KOREA (REPUBLIC OF)	
KIM, Young-In	KAERI
SPAIN	
GONZÁLEZ, Enrique Miguel	CIEMAT
ALVAREZ-VELARDE, Francisco	CIEMAT
UNITED KINGDOM	
GREGG, Robert W. H.	NNL
ZIMMERMAN, Colin H	NNL

UNITED STATES OF AMERICA

DIXON, Brent	INL
FINCK, Phillip J.	INL
IRELAND, John R.	INL
MCCARTHY, Kathryn A. (Chair)	INL
PASAMEHMETOGLU, Kemal O.	INL

INTERNATIONAL ORGANISATIONS

DYCK, Gary	IAEA
INOZEMTSEV, Victor	IAEA
CHOI, Yong-Joon (Secretary)	NEA