



The Effects of the Uncertainty of Input Parameters on Nuclear Fuel Cycle Scenario Studies

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Foreword

Under the auspices of the Nuclear Energy Agency (NEA) Nuclear Science Committee (NSC), the Working Party on the 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 fuels and materials, accelerators and spallation targets. Various expert groups were established to cover the above-mentioned topics.

The Expert Group on Advanced Fuel Cycle Scenarios (EGAFCS) was created in 2010, replacing the Expert Group on Fuel Cycle Transition Scenarios Studies, to study R&D needs associated with the transition from current or future advanced nuclear fuel cycles. The objectives of the expert group are 1) to assemble, organise and understand the scientific issues of advanced fuel cycles; 2) to provide a framework for assessing specific national needs related to the implementation of advanced fuel cycles.

After conducting a benchmark study to compare existing codes in terms of capabilities, modelling and results, the expert group performed a benchmark study to identify and communicate the impact of uncertainties in fuel cycle analyses using systems codes. The purpose of this study was to assess and quantify the importance of input parameters in scenario analysis. The benchmark was conducted in different phases: 1) base case scenarios based on the previous benchmark scenario; 2) system codes normalisation; 3) parameter identification; 4) parameter studies; 5) uncertainties representation. Results from code calculations from seven organisations were compared. Seventeen input parameters were selected and their impact of uncertainties was assessed on 22 scenarios outputs or indicators.

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List of abbreviations and acronyms

BOL:	Beginning of life
BR:	Breeding ratio
BU:	Burn-up
CEA:	Commissariat à l'énergie atomique et aux énergies renouvelables
CIEMAT:	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
CNL:	Canadian National Laboratory
CR:	Conversion ratio
CSD:	Control and shutdown device
DSD:	Diverse shutdown device
EFIT:	European facility for industrial transmutation
EFR:	European fast reactor
EFPD:	Effective full-power day
EGAFCS:	Expert Group on Advanced Fuel Cycle Scenarios
EK:	Hungarian Academy of Sciences, Centre for Energy Research
ENEA:	Italian National Agency for New Technologies, Energy and Sustainable Economic Development
EOL:	End of life
ESFR:	European sodium fast reactor
FP:	Fission product
FR:	Fast reactor
INL:	Idaho National Laboratory
KIT:	Karlsruhe Institute of Technology
LAB:	Lower axial blanket
MA:	Minor actinides
MOX:	Mixed oxide
NPP:	Nuclear power plant
NU:	Natural uranium
P&T:	Partitioning and transmutation
PWR:	Pressurised water reactor

RU:	Reprocessed uranium
SA:	Sub-assembly
SFR:	Sodium fast reactor
SWU:	Separative work unit
TRU:	Transuranic
UAB:	Upper axial blanket
UGP:	Upper gas plenum
UOX:	Uranium oxide
WH:	Working horse definition of the European SFR core

1. Introduction

1.1. Purpose of the study

Fuel cycle scenario analysis is a common method for identifying and communicating potential nuclear energy futures, especially when assessing the impact of new technologies. System codes are used to assess the behaviour of different nuclear energy systems and to examine scenarios involving transition of infrastructure (reactors, fuel cycle facilities, etc.) to accommodate growth and adopt new fuel cycle management approaches (e.g. recycling).

By their nature, scenario analyses involve assumptions about the future, including future demands for energy, evolution of energy infrastructure, and performance of existing and future technologies. In addition, the systems codes used in scenario analyses are imperfect representations of the real world. Uncertainties inherent in scenario specifications and system codes impact the accuracy of analyses. Since these analyses may contribute to decisions on policy, technology selection, and research, development and demonstration (RD&D) budgets, it is important to identify and communicate the impact of these uncertainties.

There are several sources of uncertainty in fuel cycle analyses using system codes, including scenario assumptions, facility operational values, reactor core physics calculations, etc., and their impacts vary based on the fuel cycle and the performance metrics of interest.

The purpose of this study is to systematically identify these sources of uncertainty and use sensitivity studies to assess their impacts on system level results.

The results of this study should help nuclear energy system analysts to improve model accuracy and better communicate the impact of unavoidable uncertainties in their assessments. In particular, it will indicate which uncertain parameters may have large impacts necessitating sensitivity assessments while also indicating where additional modelling detail is not warranted because the system is relatively insensitive to associated parameter values.

1.2. Organisation of the study

This study was conducted in several phases.

1.2.1. Base scenario definition

First, a base scenario was identified and carefully defined. The Expert Group on Advanced Fuel Scenarios (EGAFCS) has learned from past efforts that a clear and concise scenario definition is necessary to reduce the potential for multiple interpretations of the specification.

The base scenario needed to be simple but non-trivial and be able to demonstrate the impact of variations in key parameters. The expert group had previously conducted a systems code benchmarking activity [1] that used a scenario with many of the needed features, including the use of more than one reactor type, more than one fuel type, an initial steady

state configuration with constant energy demand, and transition of the reactor and fuel cycle infrastructure to a new system.

However, the benchmark scenario included some features that were either undesirable or unnecessary for the uncertainty study. First, the benchmark scenario was tightly constrained such that a small variation could result in a large change in the overall system, which was undesirable because it would tend to amplify the effects of uncertainty beyond that seen in most analyses. In essence, the benchmark scenario was like driving along the edge of a cliff, where a slight change of direction would result in a “broken scenario” with the system no longer meeting performance objectives. Second, the scenario included an additional third fuel that was not needed to demonstrate the impacts of uncertainty of parameter values. For these reasons, the benchmark scenario was simplified by removing the third fuel and relaxing the scenario timing to make it harder (but not impossible) to have a broken scenario.

Another reason for simplifying the base scenario was to ensure that the scenario was within the modelling capabilities of all of the system codes that were to be used in the study. Since the system codes have been designed by different organisations, each has unique capabilities in areas of interest to their sponsoring organisation. Members of the expert group have had considerable experience in previous benchmarking studies in identifying the common core of functionality each code includes and ensuring the base case only requires that common core.

The scenario definition included a standard set of output parameters that each code needed to produce to document the results of the analyses. The Expert Group drew on our collective experience in scenario analyses to identify the common set of parameters to be used. In addition to the common set, each analyst was instructed to include additional outputs as needed to explain any unusual observed behaviour.

The resulting base scenario is 200 years in duration, begins with a fleet of thermal reactors (PWRs) in equilibrium using low-enriched uranium oxide fuel, transitions to a fleet of sodium fast reactors (SFRs) near mid-scenario that use a uranium/plutonium mixed oxide fuel, reaching equilibrium for the new fleet before the end of the scenario and maintaining constant electricity production throughout. Standard outputs include reactor numbers and types, mass flows of fuel and waste materials, and inventories of materials in storage and disposal. The detailed scenario specification for the base case is documented in Chapter 2.

1.2.2. System code normalisation

The study involved the use of multiple system codes, with some codes used by more than one organisation. The next phase of the study involved all of the participants modelling the base scenario and adjusting their models to produce similar behaviour. This accomplished several objectives.

First, the analyses of the base case by multiple organisations uncovered any remaining ambiguities in the scenario specification. Each organisation independently interpreted the specification and any differences in interpretation were revealed by differences in the scenario results. The expert group discussed these differences and clarified the scenario documentation as needed to ensure consistent interpretation.

Second, the remaining differences in results revealed the impact of differences in code architectures. The system codes used in this report include most of the major codes used by member countries for scenario analyses. Each code was developed independently and there are variations between codes in the level of detail modelled for different functions in the fuel cycle. These variations can result in small differences in results for the same scenario, which is one form of uncertainty. Where these differences occurred, their drivers were identified and discussed in the text.

Finally, the base case results for each code provided a basis for identification of the effects of uncertainty of key parameters. Since each code provides slightly different results, each needed to establish and document a basis of comparison for the parameter studies to come.

Chapter 3 provides the output of the base case for all codes and participants, including an assessment of any significant variations in results.

1.2.3. Parameter identification

The most important component of this study was the identification of the key parameters to be evaluated. Again, the expert group drew from our collective expertise to develop the set of parameters that in our experience most often are sources of uncertainty during the construction of scenario specifications. An important consideration was the independence of these parameters, such that variation in one did not immediately require variation in a second, coupled parameter. Coupled parameters were considered as secondary inputs for the scenario specification where their values were determined by the primary inputs.

The second step in this process was to identify an uncertainty range for each key primary parameter to be used as the basis of a sensitivity analysis. Again, experience was used as a basis, with each parameter varied over ranges typically used in past analyses. The group considered that this approach would provide a more accurate assessment of the impacts of uncertainty than varying each parameter by the same amount (e.g. +/- 10%) because we knew from experience that some parameters can be defined more accurately while others are more uncertain.

The list of primary parameters included in the sensitivity analyses and the ranges used for the parameters are documented in Section 2.2.

1.2.4. Parameter studies

The primary activity in this study was conducting sensitivity analyses on the key parameters to identify the effects and quantify the impacts of uncertainty. To the extent possible, each parameter was varied independently without change to any other parts of the specification, including any secondary parameters. In some cases, this was not possible due to direct interdependencies of secondary parameters. For example, the enrichment of fresh thermal reactor fuel limits the burn-up potential of the fuel; higher enrichment is necessary for higher burn-up. Changes to any secondary dependent parameters were identified and included in the specification of the sensitivity scenario (see Appendix A).

While every participant performed the base scenario analysis, the parameter studies were divided up between participants. In most cases, this was based on the availability of personnel to perform the analyses, while in some cases it was based on the specific

capabilities of the system codes. Some codes have capabilities beyond the common core that enabled easy assessment of certain parameters, and these capabilities were taken into consideration.

During the sensitivity analyses, some scenarios “broke” and had insufficient fuel material for the fast reactor fleet. This was noted in the analysis and then an effort was made to “fix” the scenario, via one of two methods at the discretion of the analyst. One option was to modify additional parameters to address an imbalance in the scenario. For example, if the scenario also resulted in a significant inventory of unprocessed used fuel then the reprocessing capacity could be increased to supply more fuel material. The other option was to add an external source of fuel material and note the amount of extra material needed to complete the scenario. This was the only option in some cases when there were no other system imbalances (e.g. a growth case). Broken scenarios and the approaches to heal them are discussed in more detail in Section 4.4.

Chapters 5, 6, and 7 provide detailed discussion and results of the sensitivity analyses performed in the study.

Chapter 5 focuses on the effects and impacts of uncertainty in the general scenario specification parameters, including:

- total energy demand;
- cooling time of used nuclear fuel prior the reprocessing for recycle, by fuel type;
- fabrication time of fuels, by reactor type;
- the year of initial introduction of fast reactors in the scenario;
- the rate of introduction of fast reactors in the scenario.

Chapter 6 focuses on uncertainty related to reactor performance parameters, including:

- thermal reactor fuel burn-up;
- fast reactor fuel burn-up;
- fast reactor fuel minor actinide content;
- fast reactor breeding ratio;
- reactor lifetime.

Chapter 7 focuses on the impacts of uncertainty related to the operations and performance of fuel cycle facilities, including:

- enrichment tails assay;
- the year of initial reprocessing for both thermal reactor used fuel and fast reactor used fuel;
- the annual reprocessing capacity for thermal reactor uranium oxide fuel and fast reactor mixed oxide fuel;
- losses to the waste stream during reprocessing for uranium, plutonium and minor actinides;

- reprocessing priorities (newest fuel versus oldest fuel).

1.2.5. Uncertainty representation

While the members of the expert group all have experience with sensitivity analyses, for most this was the first time so many parameters were being compared at the same time. A method for summarising the impact of parameter uncertainty was needed that would work for all parameters included in the study and would communicate the relative magnitude of the impacts. The group had considerable discussion on how to do this, given the large number of output parameters used and the range of impacts expected. An important consideration was how to treat impacts that were transient, where there was a notable effect during transition but the system ultimately achieved the same final state by the end of the scenario. Several methods of summarisation were considered, as discussed in Chapter 4. The summarised results are presented in Chapter 8, along with a discussion of major conclusions of the study.

1.3. About the expert group

The Expert Group on Advanced Fuel Cycle Scenarios is organised under the Working Party on Scientific Issues of the Fuel Cycle within the NEA Nuclear Science programme. The mandate of the group is to perform tasks and study needs associated with the transition from current to future advanced nuclear fuel cycles.

The objectives of the expert group are:

- to assemble, organise and understand the scientific issues of advanced fuel cycles;
- to provide a framework for assessing specific national needs related to implementation of advanced fuel cycles.

The expert group was created in 2010 with a current mandate extending to mid-2019. Membership is open to all NEA member countries. The current membership of the group is listed in Appendix D.

References

- [1] NEA (2012), “Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes”, NEA/NSC/WPFC/DOC(2012)16, OECD, Paris.

2. Specifications of the base case scenario

2.1. Scenario description

2.1.1. Depletion part

Some scenario codes use a depletion module to calculate the evolution of the isotopic composition of the different fuels (UOX, MOX, etc.) whereas others use an average composition at each step in the fuel and in back-end cycle.

The scenario considers two types of fuel:

- UOX fuels for PWRs loaded with 100% of UOX;
- MOX fuels for SFRs loaded with 100% of MOX (minor actinides being sent to the waste). Minor actinides fuel content is considered in a second case.

2.1.1.1. PWR UOX fuel

UOX fuels have an initial enrichment of 4.95 wt% of ^{235}U . The composition used in the benchmark is detailed in Table 2.1-1.

Table 2.1-1: Initial composition of UOX fuels

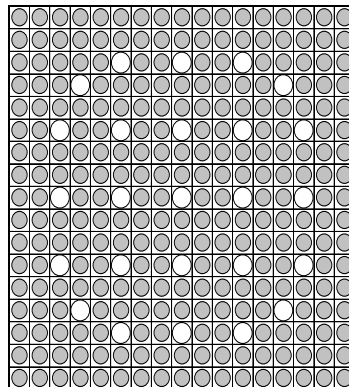
Nuclide	wt%
^{234}U	0.0445
^{235}U	4.95
^{238}U	95.0055

The geometric data corresponds to the one of a standard fuel assembly type FRAGEMA 900 MWe (17x17), as follows (Figure 2.1-1):

- 264 fuel rods;
- 24 thimble guides;
- 1 instrumentation tube;
- 0 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/cm²;
- 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;
- material composition / densities:
 - density of UO₂ = 10.07 g/cm³;
- average boron concentration:
 - 456 ppm for UOX;
- boundary conditions:
 - calculations are made in an infinite network, the B2 coefficient is adapted to have $k_{\text{eff}} = 1$;
- temperatures:
 - UOX: 600°C for the fuel, 306°C for the moderator (choose an approximate clad temperature);

Figure 2.1-1: Scheme of the fuel assembly



Irradiation could be divided into small steps but the fuel loading and unloading times between those steps is ignored.

The cladding, the thimble guides and the instrumentation tubes material is Zircalloy-4 whose composition is (in wt%):

- 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 the sum of other composition)

The thimble guides and instrumentation tubes are filled with water.

The depletion calculation is made in one step for a burn-up of 60 GWd/t (1640 EFPD) and a cooling time of 5 years.

2.1.1.2. MOX FR fuel

MOX FR fuels are composed of a fissile part (69.7 wt%) and an axial blanket (30.3 wt%). The fissile zone has an initial equivalent ^{239}Pu content of 13.8 wt%, corresponding to an initial Pu content going from 20 wt% to 21.5 wt% in function of the Pu isotopic composition.

The reactivity coefficients are detailed in the Table 2.1-2. The composition of the fuel assembly is detailed in Table 2.1-3.

Table 2.1-2: Reactivity coefficients

Isotope	Coefficient	Isotope	Coefficient
^{234}U	0.0255	$^{242\text{m}}\text{Am}$	2.1763
^{235}U	0.7749	^{243}Am	-0.3236
^{236}U	-0.0619	^{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

The depletion calculation is performed in one step for a burn-up of the fissile zone of 136 GWd/t (1700 EFPD). Calculations are carried out in an infinite lattice at criticality ($k_{\text{eff}} = 1$).

Examples of compositions (homogeneous composition of fissile and fertile zones) used for fresh and spent MOX FR fuels are presented in Table 2.1-4. The final composition of the fuels corresponds to the composition of unloaded fuel coming out of the reactor.

Table 2.1-3: Composition for MOX FR Fuel

	Composition (vol%)
Fuel	37.51
Na	32.94
Structure	23.59

Table 2.1-4: Initial and final compositions for MOX SFR Fuel

	Beginning of SFR deployment (2080)		End of SFR deployment (2110)		SFR fleet equilibrium (2200)	
	Initial	Final	Initial	Final	Initial	Final
²³⁴ U	0.1647%	0.1159%	0.0046%	0.0089%	0.0019%	0.0041%
²³⁵ U	0.2220%	0.1092%	0.2149%	0.0847%	0.2155%	0.0842%
²³⁶ U	0.0207%	0.0485%	0.0010%	0.0300%	0.0010%	0.0301%
²³⁸ U	84.1396%	74.0736%	85.5555%	75.0666%	85.7885%	75.2811%
²³⁸ Pu	0.3845%	0.2965%	0.2214%	0.1383%	0.0842%	0.0822%
²³⁹ Pu	8.6885%	8.3409%	8.1381%	8.2179%	8.2399%	8.2568%
²⁴⁰ Pu	4.3327%	4.5778%	4.4101%	4.5680%	4.8054%	4.8325%
²⁴¹ Pu	0.0586%	0.5548%	0.4354%	0.6256%	0.4180%	0.6601%
²⁴² Pu	1.5360%	1.1139%	0.9727%	0.7510%	0.4033%	0.3831%
²⁴¹ Am	0.4314%	0.2144%	0.0463%	0.1113%	0.0422%	0.1138%
^{242m} Am	0.0000%	0.0169%	0.0000%	0.0059%	0.0000%	0.0059%
²⁴³ Am	0.0000%	0.2502%	0.0000%	0.1654%	0.0000%	0.0769%
²³⁷ Np	0.0212%	0.0478%	0.0001%	0.0361%	0.0001%	0.0362%
²⁴² Cm	0.0000%	0.0168%	0.0000%	0.0081%	0.0000%	0.0083%
²⁴³ Cm	0.0000%	0.0022%	0.0000%	0.0007%	0.0000%	0.0007%
²⁴⁴ Cm	0.0000%	0.0975%	0.0000%	0.0653%	0.0000%	0.0291%
²⁴⁵ Cm	0.0000%	0.0073%	0.0000%	0.0050%	0.0000%	0.0022%
²⁴⁶ Cm	0.0000%	0.0003%	0.0000%	0.0002%	0.0000%	0.0001%
PF	0.0000%	10.1156%	0.0000%	10.1109%	0.0000%	10.1126%
TOTAL	100.0000%	100.0000%	100.0000%	100.0000%	100.0000%	100.0000%

2.1.2. Transition scenario assumptions

Three scenarios were included in the past benchmark [1]:

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

In the present study, a simplified version of the third scenario (where only UOX fuel is considered to be loaded in PWR) has been selected to be used as a base for uncertainties studies. The main assumptions of this scenario are:

- duration of the scenario: 200 years;
- constant installed power: 62.4 GWe¹;
- constant electrical annual production: 430 TWhe (load factor: 0.786);
- variation rate for every type of reactor: ± 2 GWe/year.

More data are reported in Table 2.1-5.

1. (*) This power corresponds to a fleet of 39 PWR UOX as described in Table 2.1-5. However, it does not strictly correspond to a whole number of SFR. The power has been kept constant to simplify the scenario, and the mass loaded in SFR has been slightly adjusted to fit 62.4GWe.

Table 2.1-5: Data compilation for the benchmark study

		PWR UOX	SFR
Fuels/blankets			
Fissile Burn-up	GWd/tHM	60	136
Axial blankets burn-up	GWd/tHM	-	15 (*)
Minimum cooling time	y	5	2
Fabrication time	y	2	2
Fresh fuel ²³⁵ U enrichment	%	4.95	0.25
Moderation ratio		2	-
Equivalent ²³⁹ Pu content	%	-	13.8
Cores			
Electrical nominal power	GWe	1.6	1.45
Efficiency	%	34	40
Load factor	-	0.786	0.786
Heavy metal masses			
Fissile	t	128.9	41.4
Axial blanket	t	-	18.0
Breeding gain		-	≈1
Cycle length	EFPD	410	340
Core fraction (fuel)		1/4	1/5
Reprocessing plants			
First year of reprocessing	y	35	85
Priorities	-	First in - first out	First in - first out First fuel then blankets
Annual capacity	tHM	850	600 (max.)
Losses (U and Pu)	%	0.1	0.1
Enrichment plant			
Enrichment tail	%	0.25	-

(*) Some codes (e.g. VISION) are not capable of modelling heterogeneous cores explicitly, and so require a core averaged burn-up.²

The scenario simulates an open cycle nuclear fleet followed by the continuous recycling of Pu in fast reactors. The flow chart of the scenario is depicted in Figure 2.1-2. The installed capacity, which is a linear function for each period, is shown in Figure 2.1-3 and Table 2.1-6.

2. This is nominal FR, for more information see Appendix A.

It should be noted that the transition time between a PWR and a FR fleet has been chosen arbitrarily with no reference to some specific member states scenario.

Figure 2.1-2: Flow chart

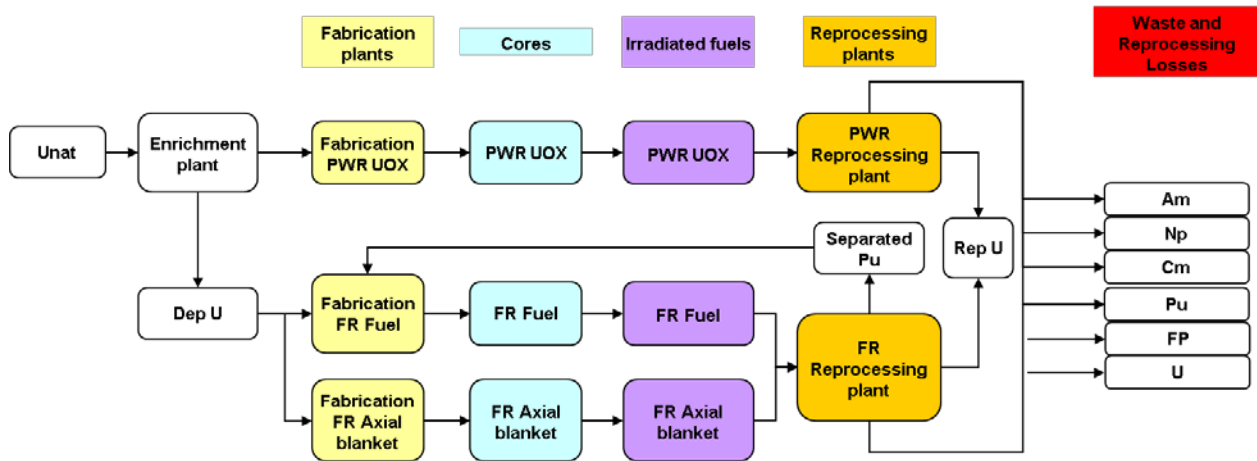


Figure 2.1-3: Installed capacity

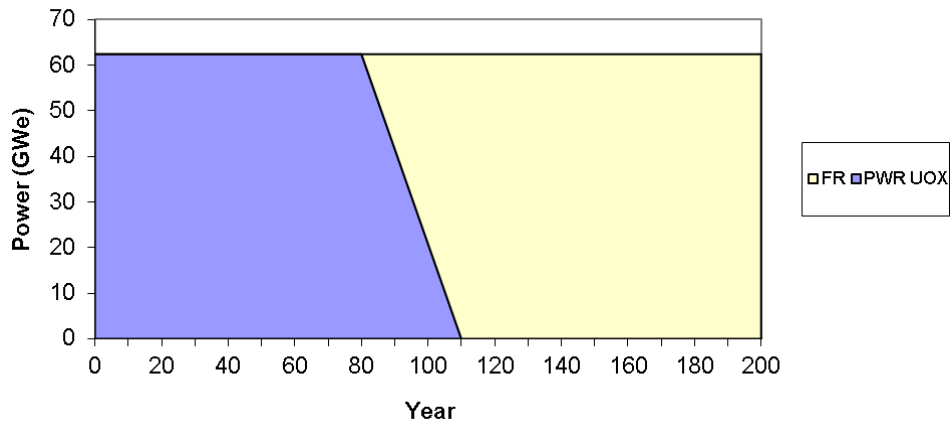


Table 2.1-6: Installed capacity

Time (y)	PWR UOX (GWe)	FR (GWe)
0	62.4	0
80	62.4	0
90	41.6	20.8
100	20.8	41.6
110	0	62.4
200	0	62.4

Reactors

- 1st load of PWR UOX: determined to have an equilibrium PWR fleet at year 0;
- last load of PWR UOX: year 109;
- 1st load of fast reactor: year 80;
- last load of fast reactor: year 200.

The reactor lifetime is supposed to be infinite in the reference scenario. First and final cores are as follows:

- PWR: the first cores of these reactors are not simulated. The final cores are simulated. The characteristics of the batch loaded in final cores are those of equilibrium batches.
- FR: the first cores of these reactors are simulated. The characteristics of batch loaded in first cores are those of equilibrium batches. The final cores are not simulated.

Enrichment plant

The enrichment tail of ²³⁵U is 0.25%.

Fabrication plant

Fuel fabrication starts 2 years before the first load in each reactor.

The fabrication of the FR fuel is made with a mix of depleted uranium (tails from enrichment) and Pu.

The fabrication of axial blankets is made with depleted uranium coming from the enrichment plant.

Reprocessing plants

Time for reprocessing is assumed to be 0 (the fabrication time and the spent fuel cooling time will be considered as parameters for uncertainty studies).

Spent fuel

The minimum spent fuel cooling time before reprocessing is of 5 years for PWR fuels and 2 years for FR fuels.

2.1.3. Expected results

The expected results which have to be reported in the benchmark include the following annual values:

- natural uranium consumption;
- Separative Work Unit (SWU) and enriched uranium needs;
- PWR UOX and FR fuel fabrication flows;
- plutonium mass flows for fabrication;
- spent fuel (PWR UOX and FR) interim storage;
- processed spent fuel (PWR UOX and FR);
- plutonium inventory in fuel cycle (fabrication and reprocessing plants, nuclear power plant, spent fuel storage);
- materials in interim storage (depleted uranium, reprocessed uranium, separated TRU, etc.);
- plutonium and minor actinides (Am, Np, Cm) inventory into the waste (including losses from reprocessing).

Other results can be added according to each code capability.

2.2. Uncertainties studies description

Table 2.2-1 lists the field of variation considered for each parameter of interest for uncertainties scenario studies (the standard values of the parameters are represented in blue) along with some expected results and the repartition of studies between participants.

Table 2.2-1: Field of variation for parameters considering in uncertainties studies

		PWR UOX	SFR	Expected results	Institute
Reactor characteristics					
Fissile BU	GWd/tHM	40, 50, 60	100, 136	U consumption, enrichment, fab., Pu for fab, reprocessing	CEA, ENEA
Fresh fuel ²³⁵ U enrichment <i>Part 6.1</i>	%	4.95 (adjusted with BU)	-		
Equivalent Pu content <i>Part 6.2</i>	%	-	13.8 (adjusted with BU)		
Cycle length <i>Part 6.2</i>	EFPD	410 (adjusted with BU)	340 (adjusted with BU)		
Breeding gain ³ <i>Part 6.2</i>		-	0.75, 0.9, 1, 1.1, 1.25 Can be achieved by adding / removing fertile blankets or by changing FR fleet composition	Pu inventory	ENEA, CIEMAT, KIT
Reactor lifetime <i>Part 6.3</i>	y	Infinite, 60, 40 Commission /Decommission when needed	Infinite, 60, 40	All results	CNL, AREVA, JAEA
		PWR UOX	FR	Expected results	Institute
MA recycling					
Initial MA weight content <i>Part 6.2</i>	%	-	0%, 1%, 2% Homogeneous and Heterogeneous loadings (up to 5%) or MA loaded in dedicated burners	MA storage and inventory	CEA, CIEMAT, ENEA, KIT
Recuperation rate (MA) <i>Part 6.2</i>	%	-	0% (without P&T) 99% 99.9% (with P&T)	Waste	CEA, CIEMAT, ENEA

3. For more information on the different FRs used, see Appendix A.

General Scenario Assumptions					
Total nuclear energy demand <i>Part 5.1</i>	TWh/y	430	430 (steady) Increasing / Decreasing (fixed by participants)	All results	All participants
Minimum cooling time <i>Part 5.2</i>	y	2, 5, 8	2, 5, 8	SF storage and reprocessing	CNL, JAEA, CEA, AREVA
Fabrication time <i>Part 5.3</i>	y	1, 2, 3	1, 2, 3	Fuels fabrication	CNL, JAEA, CEA, AREVA
Introduction date of FR <i>Part 5.4</i>	y	-	Year 70, 80, 90, 130	All results	CNL, JAEA, CEA, AREVA, KIT
Rate of introduction <i>Part 5.5</i>	y	-	Over 20, 30, 40 years	All results	CNL, JAEA, CEA, AREVA

		PWR UOX	FR	Expected results	Institute
Facilities					
First year of reprocessing <i>Part 7.1</i>	y	25, 35, 45	85, 95, 105	Storage and reprocessing	CNL, JAEA, CEA, AREVA
Annual reprocessing capacity <i>Part 7.2</i>	tHM	700, 850, 1000	400, 600, 800	Storage and reprocessing	CNL, JAEA, CEA
Losses (U and Pu) <i>Part 7.3</i>	%	0.05, 0.1, 0.2		Waste	CNL, CIEMAT, JAEA
Reprocessing priority <i>Part 7.4</i>	-	FIFO to FILO		FIFO to FILO	CEA, CIEMAT, KIT
Enrichment tail <i>Part 7.5</i>	%	0.15, 0.25, 0.35		Enrichment, U consumption	CEA, CIEMAT, JAEA, AREVA

Minor actinides (MA) recycling is considered in the sensitivity studies (the MA content in fresh SFR fuel and the MA recuperation rate at reprocessing plant are consequently adjusted).

Sensitivity studies on FR breeding gain can be achieved by adding radial fertile blankets or by removing axial fertile blankets. Alternative FR designs with different breeding gain have also been compared.

To perform studies of sensitivity to UOX fuels characteristics, the burn-up, the initial ^{235}U enrichment and the cycle length must be linked. Suggested values for these parameters and the corresponding initial and final compositions are given in Table 2.2-2.

Table 2.2-2: Initial and final compositions for UOX fuels in sensitivity studies

BU (GWd/t)		40	50	60
Fuel management (EFPD)		3 x 320	4 x 325	4 x 410
Initial composition	^{234}U	0.0345%	0.0423%	0.0496%
	^{235}U	3.5000%	4.2500%	4.9500%
	^{238}U	96.4655%	95.7077%	95.0004%
Final composition after irradiation	^{234}U	0.0186%	0.0206%	0.0220%
	^{235}U	0.7426%	0.7681%	0.7604%
	^{236}U	0.4525%	0.5721%	0.6878%
	^{238}U	93.5141%	92.1844%	90.9054%
	^{238}Pu	0.0240%	0.0358%	0.0493%
	^{239}Pu	0.5692%	0.6074%	0.6366%
	^{240}Pu	0.2565%	0.2846%	0.3088%
	^{241}Pu	0.1500%	0.1741%	0.1947%
	^{242}Pu	0.0711%	0.0915%	0.1123%
	^{241}Am	0.0083%	0.0089%	0.0092%
	$^{242\text{m}}\text{Am}$	0.0001%	0.0002%	0.0002%
	^{243}Am	0.0148%	0.0217%	0.0293%
	^{237}Np	0.0546%	0.0726%	0.0902%
	^{242}Cm	0.0021%	0.0027%	0.0033%
	^{243}Cm	0.0001%	0.0002%	0.0002%
	^{244}Cm	0.0052%	0.0092%	0.0146%
	^{245}Cm	0.0003%	0.0006%	0.0010%
	^{246}Cm	0.0000%	0.0001%	0.0001%
	FP	4.1157%	5.1453%	6.1744%
	TOTAL	100.0000%	100.0000%	100.0000%

References

- [1] NEA (2012), "Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes", NEA/NSC/WPFC/DOC(2012)16, OECD, Paris.
- [2] P. Sciora et al. (2009), "A break even oxide fuel core for an innovative SFR: CEA neutronic studies", *Proceedings of GLOBAL 2009*, Paris, France.
- [3] L. Buiron et al. (2011), "Transmutation abilities of a 3600 MWth SFR core", *Proceedings of GLOBAL 2011*, Makuhari, Japan.

3. Outputs from the base case scenario

The outputs from the base case scenario obtained by all participants are presented and compared in this chapter. The different codes used to obtain those results are the following:

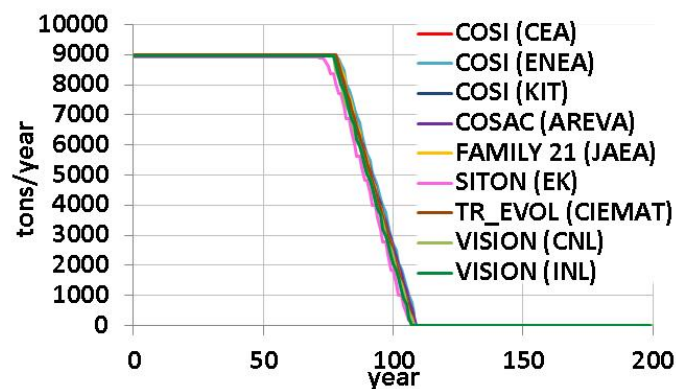
- **COSI**, used by CEA, ENEA and KIT [1];
- **COSAC**, used by AREVA [2];
- **FAMILY 21**, used by JAEA [3];
- **SITON**, used by EK [4];
- **TR_EVOL**, used by CIEMAT [5];
- **VISION**, used by CNL and INL [6].

3.1. Font-end cycle

3.1.1. Natural uranium consumption

The natural uranium consumption (Figure 3.1-1) follows the PWR's shape of the energy demand. Before the PWR fleet phase out, the natural uranium consumption is about 9000 tonnes/year. Very good agreement for all the codes was observed on this result.

Figure 3.1-1: Base case scenario – Natural U consumption



3.1.2. Enrichment needs

The uranium enrichment need also (t/year in Figure 3.1-2 and Separative Work Unit per year in Figure 3.1-3) follows the PWR's shape of the energy demand. The average values before the PWR fleet phase out are respectively 900 tonnes/year and 7.10^6 SWU/year. Results are in good agreement for all the codes.

Figure 3.1-2: Base case scenario – Enriched U needs

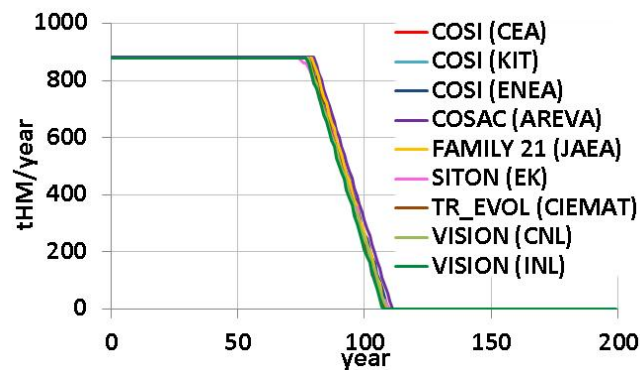
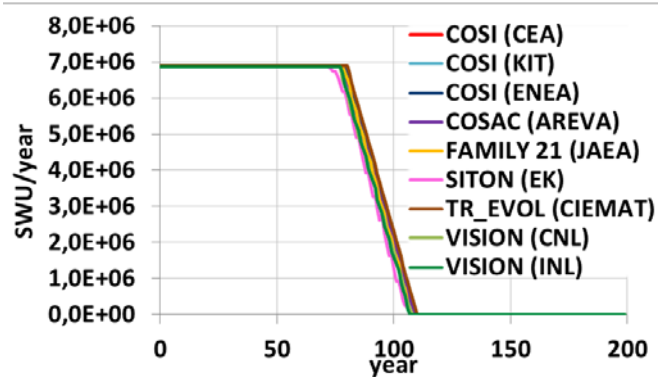


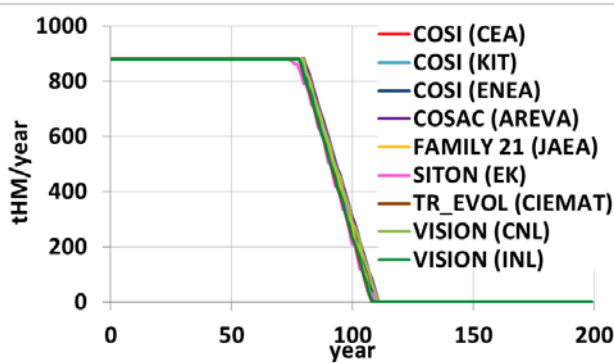
Figure 3.1-3: Base case scenario – SWU needs



3.1.3. Fabrication needs

The UOX fuel fabrication need remains constant until year 80 (at about 900 tonnes/year for the 60 GWe PWR fleet), and decreases linearly to reach zero in year 110 (Figure 3.1-4). There is a very good agreement on this result.

Figure 3.1-4: Base case scenario – PWR UOX fabrication needs



The progressive introduction of FR from year 80 to year 110 leads to an increase in the need in FR fuels fabrication (first cores are taken into account).

Once the 60 GWe FR fleet is deployed the need in FR fuel fabrication become stable at 300 tonnes/year of fissile fuel and 120 t/year of fertile fuel (including axial blankets). The results are shown in Figure 3.1-5 and Figure 3.1-6.

Figure 3.1-5: Base case scenario – FR fissile fuels fabrication needs

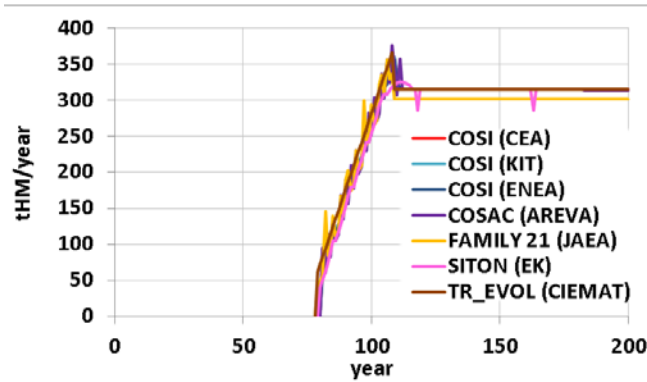
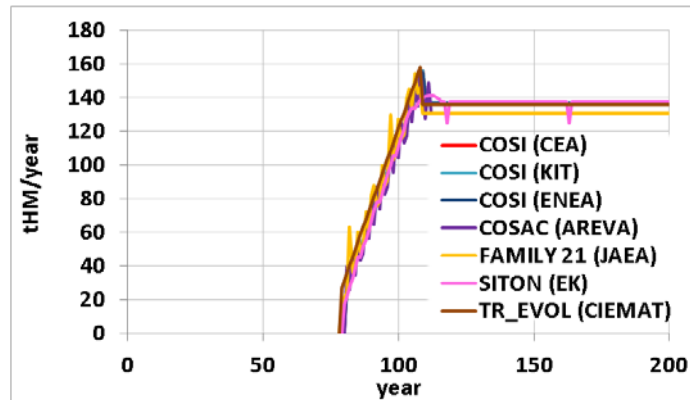


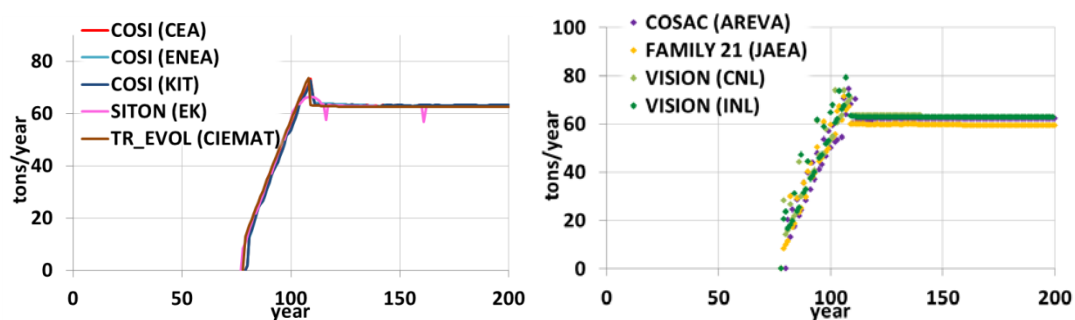
Figure 3.1-6: Base case scenario – FR fertile fuels fabrication needs



There is a good agreement on this result. The slight discrepancies are due to inconsistencies in the initial data (estimation of the need in fuel loading from the energy production –in COSI6 for example – or from the installed nuclear power – such as in FAMILY).

The annual Pu flow follows the fissile FR fuel fabrication (Figure 3.1-7). Its average value after FR deployment varies between 60 and 65 tonnes/year. This value is consistent with the average Pu content (22.2%wt) given in the scenario specifications. There is a good agreement on this result.

Figure 3.1-7: Base case scenario – Pu flow for fabrication



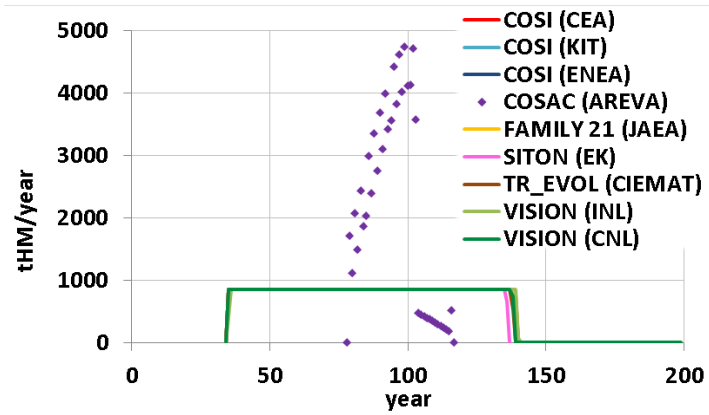
3.2. Back-end cycle

3.2.1. Spent fuel reprocessing need

UOX spent fuels reprocessing is constant and evaluated to produce enough plutonium for FR deployment. It is fixed at 850 tHM/year from year 35 to year ~140, when all UOX spent fuels have been reprocessed.

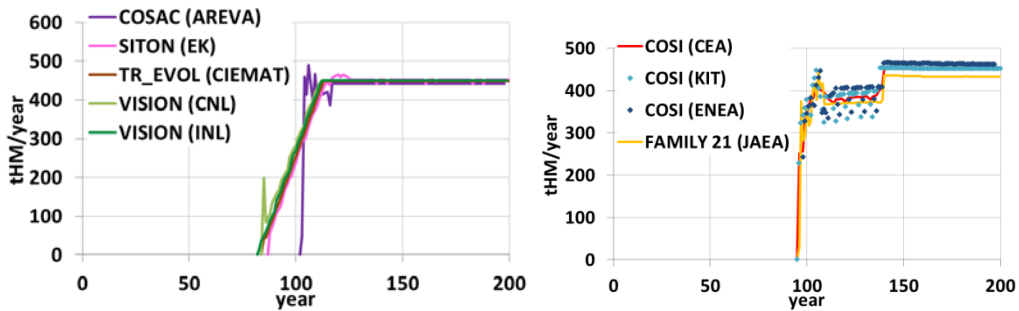
With COSAC, all available spent fuels are reprocessed in function of the need in Pu without any anticipation (Appendix A). This explains the different behaviour obtained in Figure 3.2-1. For the other codes, there is a good agreement on this result.

Figure 3.2-1: Base case scenario – PWR UOX spent fuels reprocessed



FR spent fuels reprocessing is following FR spent fuels production and limited to 600 tHM/y. Once the equilibrium is reached, it stabilises at about 450 tHM/y.

Figure 3.2-2: Base case scenario – FR spent fuels reprocessing



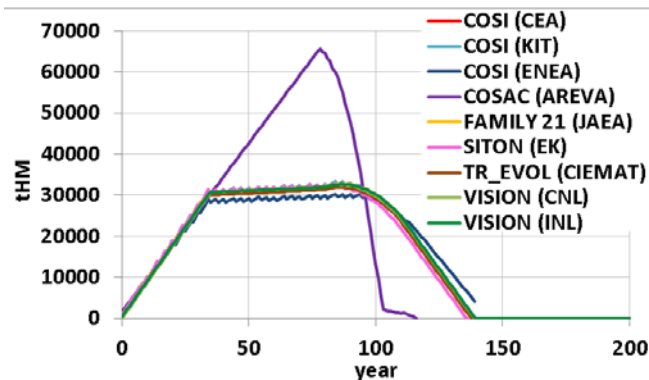
Some differences are observed between the different codes during the FR deployment phase as shown in Figure 3.2-2 and are mainly due to differences on the spent fuels isotopic composition and Pu content and to the FR breeding gain (Appendix A). For TR_EVOL, VISION and SITON, a difference in the reprocessing strategy is also observed, the FR spent fuels reprocessing is imposed, no matter if the separated Pu is needed or not for fuel fabrication (Appendix A).

3.2.2. Spent fuels storage

The PWR UOX spent fuel storage depends on the UOX spent fuels reprocessing. Its maximum is around 33 000 tonnes.

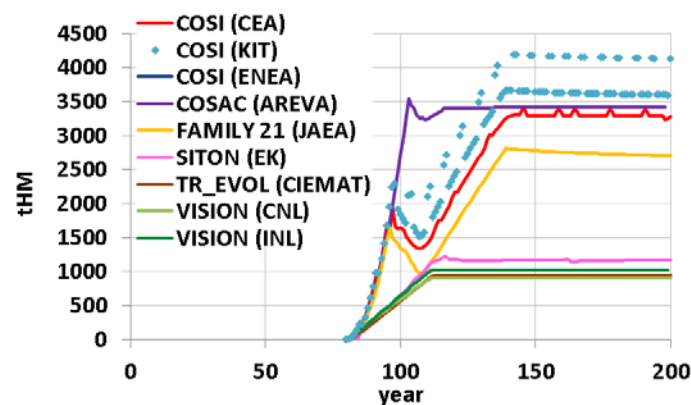
Good agreement was found between codes apart from COSAC. With COSAC, the reprocessing follows the need in Pu for FR fuels fabrication (Appendix A). PWR UOX spent fuels storage increases up to 66 000 tonnes, as indicated in Figure 3.2-3.

Figure 3.2-3: Base case scenario – PWR UOX spent fuels storage



FR spent fuels storage depends on the FR fuels reprocessing and FR spent fuel quality. Once all FR are deployed, the results reported in Figure 3.2-4 vary from 940 tonnes up to 4200 tonnes if no smoothing procedure is applied (KIT case, Appendix A).

Figure 3.2-4: Base case scenario – FR spent fuels storage

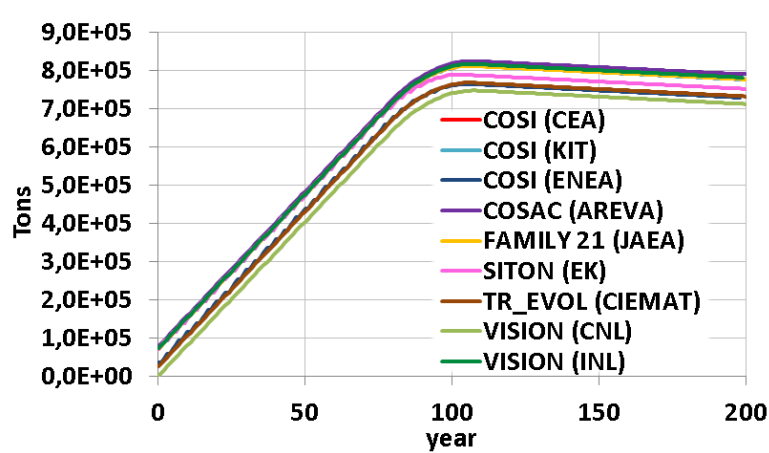


For TR_EVOL, VISION and SITON, FR spent fuel reprocessing is imposed independently of the need for spent fuel fabrication (see appendix A). With those codes, FR spent fuels storage remains constant at around 940 tHM/y. The discrepancies observed between the other codes are due to differences at reprocessing.

3.2.3. Materials interim storage

Depleted uranium storage is stabilised when the enrichment plant is stopped (slight consumption for fertile blankets fabrication). There is a relatively good agreement on this result (Figure 3.2-5). The relatively small difference between codes comes from the different interpretation of the beginning of the scenario, so that at year 0, a different amount of depleted uranium is stored (mainly the one corresponding to one or two years of fuel enrichment).

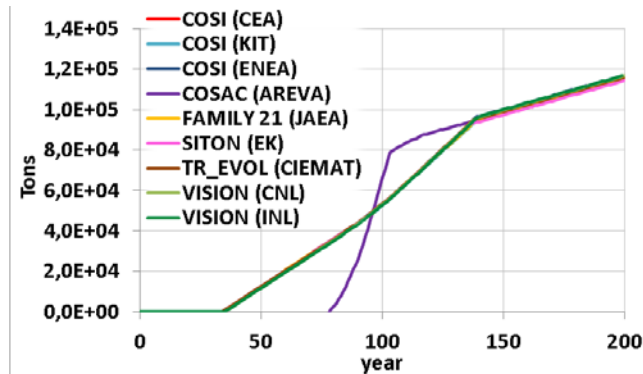
Figure 3.2-5: Base case scenario – depleted U storage



The reprocessed uranium storage increases during the scenario.

Due to differences on the PWR UOX spent fuels reprocessing strategy (Appendix A), the reprocessed uranium storage evaluated with COSAC is different, until year 140, from the one evaluated with the other codes. As indicated in Figure 3.2-6, from year 140, COSAC results are in agreement with the results obtained by the other codes. There is a good agreement on this result for the other codes.

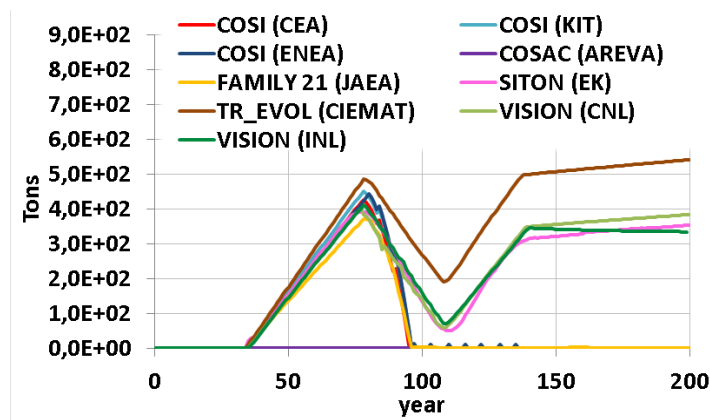
Figure 3.2-6: Base case scenario – reprocessed U interim storage



Separated plutonium waiting to be used in the FR fabrication plant is stored in a plutonium interim storage (Figure 3.2-6).

Due to the anticipated UOX spent fuels reprocessing, the Pu storage increases before the first FR fuel fabrication and reaches a first maximum ranging from 350 tonnes to 500 tonnes. It then drops to zero, the FR spent fuels reprocessing being adapted to the Pu need for FR fuels fabrication.

Figure 3.2-7: Base case scenario – Pu interim storage

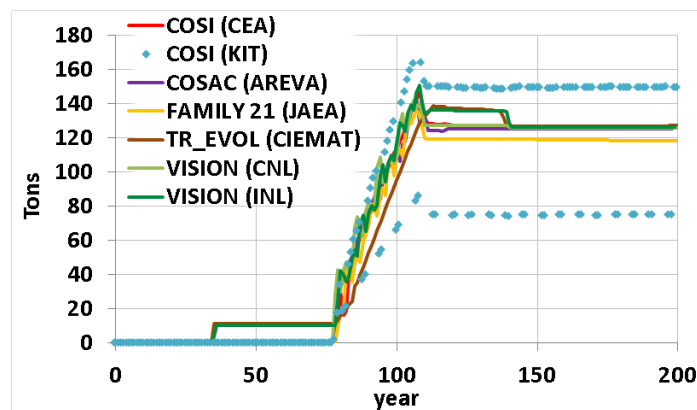


With COSAC, the Pu interim storage is always empty (Appendix A). With TR_EVOL, VISION and SITON, the FR spent fuels reprocessing does not strictly follow the need in Pu for FR fuels fabrication (Appendix A) resulting in a large inventory of Pu in the Pu interim storage after year 100 (Figure 3.2-7). The differences in the breeding gain or in PWR discharge composition (Appendix A) can also explain some discrepancies.

3.3. Inventories

3.3.1. Pu inventories in fuel cycle

Figure 3.3-1: Base case scenario – Pu inventory in plants

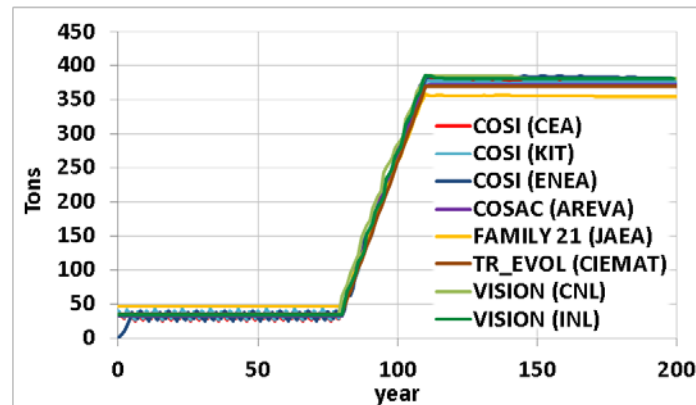


The reprocessing time being regarded as null, the Pu inventory in plants (Figure 3.3-1) represents the Pu in the fabrication plant. Thus, it follows the need in FR fuels fabrication; it increases from year 78 and it is stabilised around 125 tonnes in year 110.

The results obtained by the CEA with COSI are smoothed, which is not the case of the results obtained by the KIT (Appendix A).

The Pu inventory in NPP (Figure 3.3-2) is stabilised around 35 tonnes while PWR UOX are in operation. It increases between year 80 and year 110, during the FR fleet deployment, and is stabilised around 375 tonnes when the FR fleet reaches equilibrium.

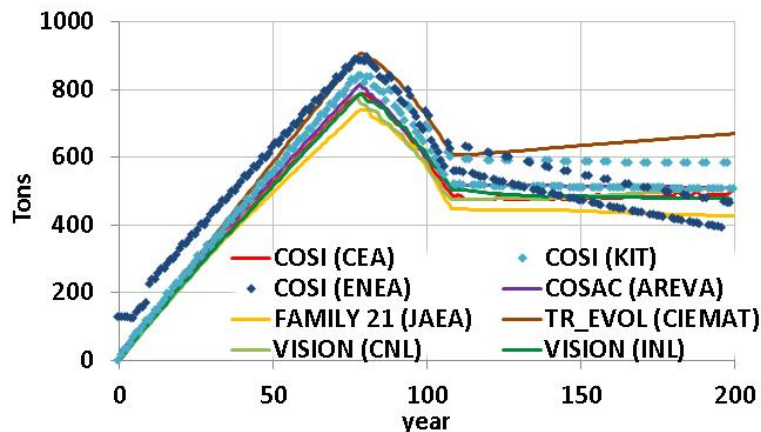
Figure 3.3-2: Base case scenario – Pu inventory in NPP



The Pu inventory in storage increases before the deployment of FR fleet since UOX spent fuel and separated Pu are accumulated in anticipation for the deployment. Its maximum is reached in year 80 and ranges between 740 and 900 tonnes, depending on the code.

It decreases during the FR fleet deployment and is stabilised with most of the codes between 480 and 550 tonnes.

Figure 3.3-3: Base case scenario – Pu inventory in storage

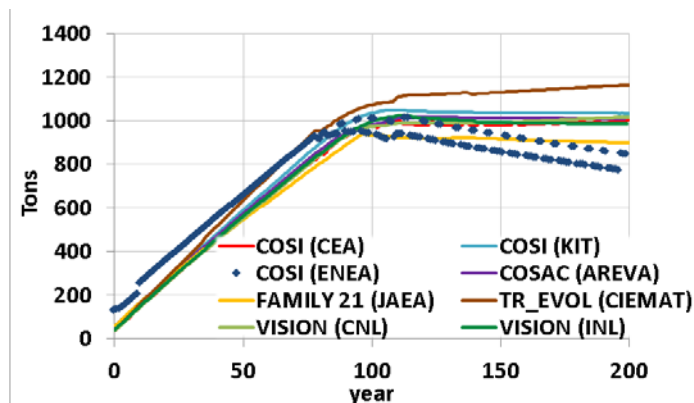


It should be noted that the FR core concept used with TR_EVOL, COSI (CEA, ENEA), and FAMILY is not strictly break-even, which leads to an increasing or decreasing Pu inventory in storage (Appendix A). The results obtained by the KIT with COSI are not smooth (Appendix A).

The total Pu inventory in cycle (Figure 3.3-4) is the sum of the plutonium in plants (fabrication and reprocessing), in reactors and in spent fuels storage and of the separated

plutonium. The differences indicated for Figure 3.3-1, Figure 3.3-2 and Figure 3.3-3 are also observed in Figure 3.3-4.

Figure 3.3-4: Base case scenario – Total Pu inventory



The total Pu inventory calculated with COSI (CEA, KIT) and VISION stabilises around 1000 tonnes at the end of FR deployment (break-even SFR core). With TR_EVOL, the total Pu inventory slowly increases to 1200 tonnes in year 200 (FR breeding gain > 0). With FAMILY and COSI (ENEA), the total Pu inventory slowly decreases to 900 t in year 200 (FR breeding gain < 0).

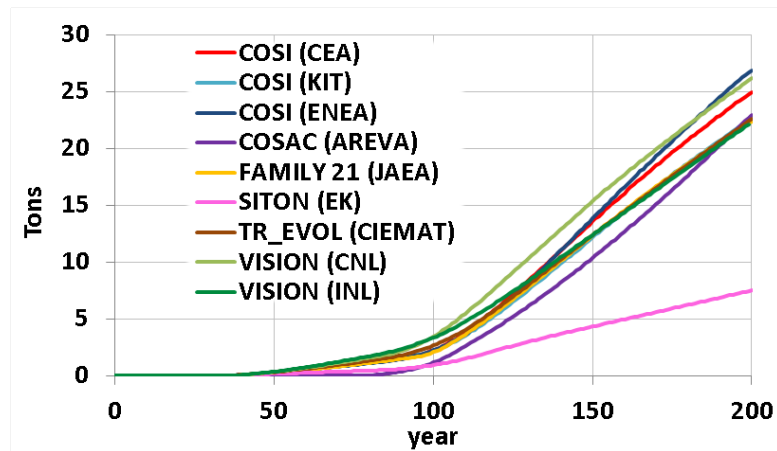
3.3.2. Inventories in waste

Pu and MA inventories in waste respectively depend on Pu and MA losses in the reprocessing plant: 0.1% Pu, 100% MA. There is no MA transmutation in the reference scenario.

Pu inventory in waste is represented in Figure 3.3-5. It includes, for all the codes, except for SITON, the Pu losses at reprocessing and the Pu production by decay. It increases until around 25 tonnes in year 200, showing that all codes can correctly model materials cooling.

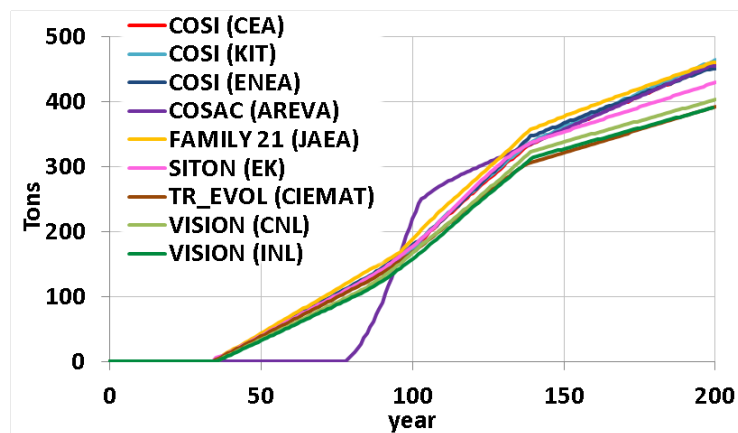
Inventories cannot be evaluated with SITON so that the Pu inventory in waste represents the cumulated Pu losses at reprocessing only. The Pu produced by MA decay is counted in the "MA inventory in waste" (which represents the cumulated MA losses at reprocessing).

Figure 3.3-5: Base case scenario – Pu inventory in waste



The MA inventory in waste (Figure 3.3-6) increases and reaches 400 - 450 tonnes in year 200. The discrepancies observed results calculated with COSAC, up to year 140, are due to a difference in the UOX reprocessing strategy (Appendix A). There is a good agreement on these results.

Figure 3.3-6: Base case scenario – MA inventory in waste



References

- [1] C. Coquelet-Pascal, M. Tiphine, G. Krivtchik, D. Freynet, C. Cany, R. Eschbach, C. Chabert (2015), "COSI6: a tool for nuclear transition scenarios studies and application to SFR deployment scenarios with minor actinide transmutation", *Nuclear Technology*, vol. 192(2), 2015, pp. 91-110.
- [2] B. Carlier (2013), "COSAC: a computational code dedicated to fuel cycle back-end simulations in AREVA", AREVA document # PTDN-2013-037, 2013.
- [3] NEA (2012), "Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes", NEA/NSC/WPF/DOC(2012)16, OECD, Paris, pp. 13-21.

- [4] A. Brolly et al. (2015), "Development and application of SITON, a new fuel cycle simulation code", *Proceedings of the 13th Information Exchange on Actinide and Fission (IEMPT)*, OECD/NEA, Seoul, Republic of Korea, 23- 26 September 2014.
- [5] F. Álvarez-Velarde et al. (2010), "TR_EVOL, upgrading of EVOLCODE2 for transition scenario studies", *Proceedings of the First Workshop on Technology and Components of the ADS (TCADS-1)*, OECD/NEA, Karlsruhe, Germany, 4-6 June 2007.
- [6] J.J. JACOBSON et al. (2010), Verifiable Fuel Cycle Simulation Model (VISION): A Tool for Analysing Nuclear Fuel Cycle Futures, *Nuclear Technology*, 172 (2), pp. 157–178.

4. Presenting sensitivity analysis results

The analysis presented in this report aims at determining the sensitivity of 20 fuel cycle metrics to changes in 15 parameters. With each fuel cycle metric being evaluated for each year of the scenario, it would be useful to quantify the impact on metric value for the entire scenario due to a change of a single parameter value. Therefore, a sensitivity value will be defined that captures the change in value of a given metric for the entire scenario relative to the reference case.

This sensitivity value for each fuel cycle metric and parameter change could then be displayed in a format that would concisely show the results of the sensitivity analysis. This chapter presents two different ways of calculating and displaying the results, the Tornado Diagram described in Section 4.2, and the Sensitivity Table described in Section 4.3. Section 4.4 discusses another important issue of how results were obtained when the analyses failed to complete the full 200-year scenario.

4.1. Output indicators

To quantify the impact of the variation of an input parameter on the calculation, it is necessary to define the output indicators that can measure this impact.

First, it should be noted that an output indicator stems from an output parameter but is not the output parameter itself. Indeed, any output parameter has as many values as there are computational steps in the calculation. As a result, no output parameter can efficiently be used to quantify the impact of an input parameter variation on the whole calculation.

An output indicator must thus be built on the basis of an output parameter but, on the contrary to the latter, it will be representative of the evolution of the output parameter over the whole calculation.

From a practical point of view, three types of output indicators were retained in this study:

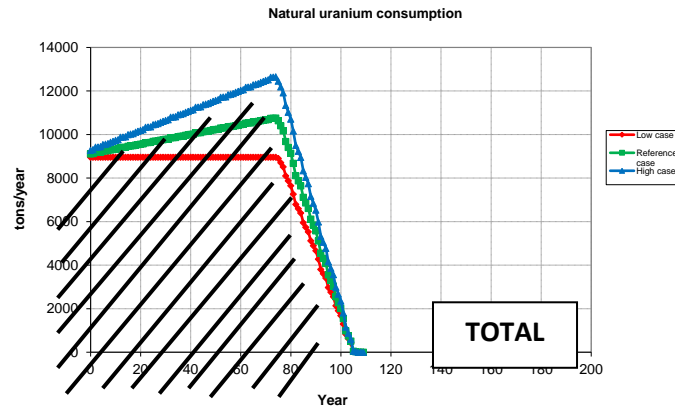
- the cumulated value of an output parameter over the scenario calculation;
- the maximum value reached by an output parameter over the whole scenario calculation;
- the final value reached by an output parameter at the end of the scenario calculation.

Depending on the nature of each output parameter, the related output indicator was chosen among these three above types. Typically:

- all the output parameters related to an annual flow –such as “natural uranium consumption”, “enriched uranium needs”, “fuel fabrication” and “spent fuel

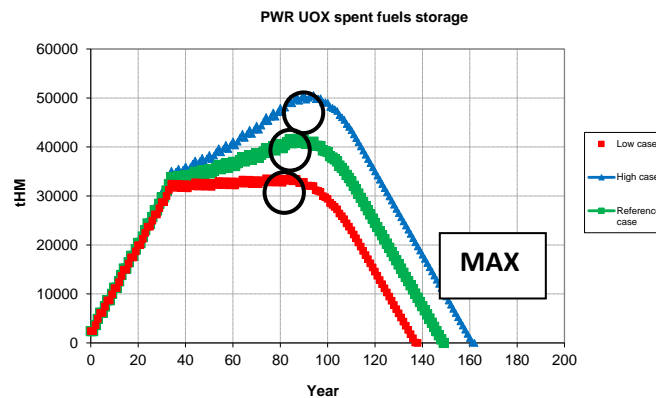
reprocessing”– were associated to output indicators defined as the **cumulated value** of the concerned output parameter, as depicted in the below graph:

Figure 4.1-1: Impact of energy demand on natural U consumption



- all the output parameters related to an inventory in an interim storage –such as “spent fuel storage”, “reprocessed uranium storage” and “depleted uranium storage”– were associated to output indicators defined as the **maximum value** (that may happen at different points in time) of the concerned output parameter, as depicted in the below graph:

Figure 4.1-2: Impact of energy demand on UOX spent fuel storage



- all the output parameters related to a global inventory inside the cycle or the waste –such as “plutonium inventory in the cycle”, “plutonium inventory in the waste”, “minor actinide inventory in the cycle” and “minor actinide inventory in the waste”– will be associated to output indicators defined as the **final value** of the concerned output parameter, as depicted in the below graph:

Figure 4.1-3: Impact of energy demand on Pu inventory in the cycle

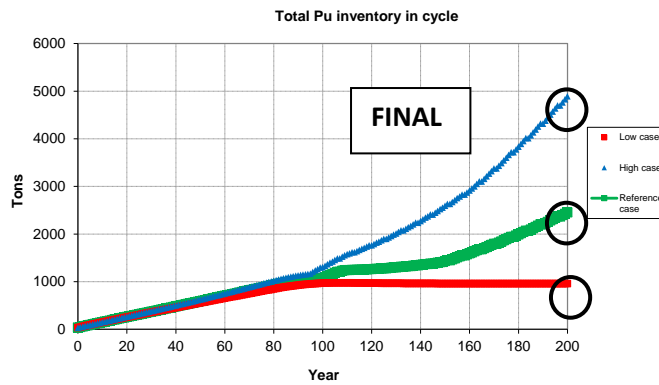


Table 4.1-1 details the method that is used to define each output parameter:

Table 4.1-1: Method used to define each output parameter

Output Parameter	Summary method
Annual natural uranium consumption (tons/year)	Total
Separative Work Units needs (SWU/year)	Total
Enriched uranium needs (tonne Heavy Metals (tHM/year))	Total
Annual PWR UOX fabrication needs (tHM/year)	Total
Annual FR fissile fuel fabrication needs (tHM/year)	Total
Annual FR fertile fuel fabrication needs (tHM/year)	Total
Annual Pu flow (input) for fabrication (tons/year)	Total
PWR UOX spent fuels inventory (tHM)	Maximum
FR spent fuels inventory (tHM)	Maximum
Annual flow of PWR UOX spent fuels reprocessed (tHM/year)	Total
Annual flow of FR spent fuels reprocessed (tHM/year)	Total
Depleted uranium interim storage (tons)	Maximum
Reprocessed uranium interim storage(tons)	Maximum
Separated Pu interim storage (tons)	Final value
Pu inventory in fab. and reprocessing plants (tons)	Final value
Pu inventory in nuclear power plants (tons)	Final value
Pu inventory in storage (separated Pu, spent fuels) (tons)	Final value
Total Pu inventory in cycle (tons)	Final value
Pu inventory in waste(tons)	Final value
MA inventory in waste (tons)	Final value

Sections 4.2 “Tornado Diagram” and 4.3 “Sensitivity Table”, describe two methods of quantifying and displaying the impact of the variation of an input parameter on the calculation. As the same output indicators were used by the both methods, the impact of an input parameter variation on the calculation can be compared and discussed on the same basis.

4.2. Tornado Diagram

It is informative to determine which parameters have the largest impact on the metrics and which metrics are most affected by a given parameter. The results of this type of analysis can be presented using tornado diagrams. Determining the relative impact that changing a parameter has on a given metric requires the definition of a sensitivity value, which can then be used to create a tornado diagram.

The sensitivity analysis proceeds as follows. First, the fuel cycle metrics are evaluated with all 15 parameters at their reference values. Then for each parameter, the fuel cycle metrics are evaluated for the following sensitivity cases: the given parameter is set to its low value; the parameter is set to its high value. All other parameters are set to their reference values. These two cases are referred to as the low and high sensitivity cases, respectively.

It should be noted that some of the sensitivity cases in which only a single parameter is different from the reference scenario may result in an insufficient amount of separated plutonium to fuel the FRs. In these cases it is assumed here that a sufficient supply of contingent plutonium is available to make up the short-fall (see Section 4.4). Contingent plutonium is assumed to have the same vector as the plutonium in SFR fuel at the beginning of the transition to FRs in the reference scenario, the composition of which is shown in Table 4.2-1.

Table 4.2-1: The composition (%wt.) of contingent Pu

^{238}Pu	2.6%
^{239}Pu	57.9%
^{240}Pu	28.9%
^{241}Pu	0.4%
^{242}Pu	10.2%

4.2.1. Sensitivity value definition

The sensitivity value is used to quantify the % change in an output indicator per unit % change in input parameter value, thus giving an estimate of the relative impact of each parameter on each output indicator.

The sensitivity value is calculated using Equation (1)

$$q = \frac{p_{ref}(R_{ref} - R_s)}{R_{ref}(p_{ref} - p_s)} \quad (1)$$

where

- p_{ref} is the reference value of the input parameter;
- p_s is the value of the input parameter in the sensitivity case;
- R_{ref} is the value of the output indicator when the input parameter is equal to p_{ref} ;
- R_s is the value of the output indicator when the input parameter is equal to p_s .

There are two values of q for each input parameter: one for the case in which the given parameter is at its low value, and one for the case in which the parameter is at its high value. A positive value of q indicates that increasing/decreasing the value of a given input parameter results in an increase/decrease in the value of the given output indicator. A negative value of q indicates that increasing/decreasing the value of a given input parameter results in a decrease/increase in value of the given output indicator.

The method used to calculate the value of R depends on the fuel cycle output being evaluated. For example the value of R for NU consumption is the cumulative NU consumption over the entire scenario; the value of R for spent fuel in storage is the maximum spent fuel in storage; and the value of R for plutonium in waste is the plutonium in waste at the end of the scenario. Table 4.1-1 shows the method used to calculate R for each fuel cycle output.

4.2.2. Example results with Tornado Diagrams

The results presented in this section are from VISION (CNL), COSI (CEA), and TR_EVOL (CIEMAT). Figure 4.2-1 shows an example tornado diagram in which the input parameters that have the largest impact on a given output parameter are shown. In these diagrams the vertical axis represents the input parameters that have a q value for the given output parameter of more than 1×10^{-4} , which are sorted in descending order of the difference between the high and low values of q from top to bottom. The horizontal axis represents the value of q for the given input and output parameters. Horizontal bars are plotted to show the value of q for the low and high value of the given input parameter.

A tornado diagram also shows how a given output parameter responds to changing an input parameter from its reference to its low value, and from its reference to its high value. This means that the following inferences can be made based on the relative values of q for the low and high input parameter value.

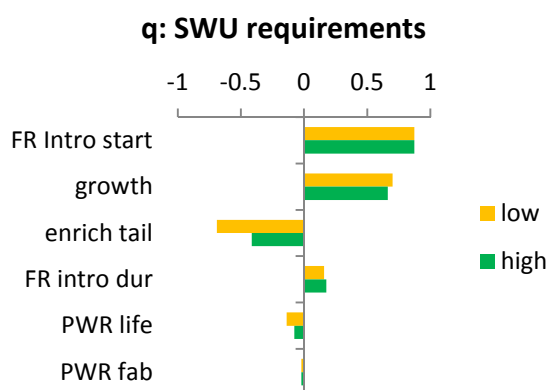
Similar q values (sign and magnitude): the relationship between the input parameter and output parameter is approximately linear over the given range of the input parameter.

q values with the same sign but different magnitudes: the change in output parameter value per unit change in input parameter value differs when going from the reference to the low input parameter value versus going from the reference to the high input parameter value.

q values with different signs: the given range of the input parameter contains at least one critical value across which the relationship between the input parameter and the output parameter changes from negative to positive.

Note that the sensitivity cases that require changing more than one input parameter (eg., PWR fuel and MA fuel loading) are excluded from this analysis.

Figure 4.2-1: The sensitivity of SWU requirements, to each parameter



The figure shows the largest impacts on SWU requirements to be the date of introduction of FRs, the energy demand growth, and the residual assay of fissile material in the enrichment tails. The later (higher) the FR introduction, the greater the total SWUs needed as the PWRs run longer and need more fuel. An earlier FR introduction (lower) results in a lower total amount of SWUs. Since the sign of the change is the same, the bar shows a positive value.

The energy demand growth increases the size of the PWR fleet, resulting in increased fuel consumption and, therefore, NU consumption and SWU requirements. Since the growth in energy demand is exponential, resulting in a relatively larger increase in energy demand after the transition to FRs, the bar for the low growth case is slightly larger than for the high growth case.

For the enrichment tails, a higher assay results in less SWUs needed, so the bar is negative (opposite signs). Similarly, a lower tails assay results in higher SWU requirements. Since more work is required to extract additional fissile material from lower assay material, the bar for the low enrichment tail case is larger than for the high enrichment tail case. A full set of sensitivity analysis tornado diagrams are provided in Chapter 8.

4.3. Sensitivity table

A sensitivity indicator (hereafter noted “S”) aims at quantifying the variation (in %) of an output indicator (hereafter noted “R”) to the variation of 1% of an input parameter (hereafter noted “p”), provided the linearity of the output indicator “R” as a function of the input parameter “p”.

For instance, delaying the introduction date of the FRs has a significant impact on the PWR uranium needs because of the PWR prolongation as a result of the FR deployment

For instance, delaying the introduction date of the FRs has a significant impact on the PWR uranium needs because of the PWR prolongation as a result of the FR deployment delay. Based on the results from the various codes involved in the benchmark, this impact is quantified at 0.9. This means a delay of 1% of the FR introduction date will increase by 0.9% the cumulated uranium needs of the scenario. In this example, “R” is the cumulated amount of uranium over the whole scenario, “p” is the introduction date of the FRs, and the resulting sensitivity indicator “S” is equal to 0.9.

“S” is expressed as $\frac{p_{ref}}{R_{ref}} \cdot \frac{\partial R}{\partial p}$ where p_{ref} is the nominal value of the input parameter “p”, ∂p is the variation of the input parameter, R_{ref} is the value of the output indicator “R” when $p = p_{ref}$, and ∂R is the variation of the output indicator “R” when the input parameter is shifted from p_{ref} to $p_{ref} + \partial p$.

The factor $\frac{\partial R}{\partial p}$ is estimated by least squares linear regression of the output indicator “R” as a function of the input parameter “p”.

Nota Bene:

- It can be noticed that, when p_{ref} is equal to the mean value of p_{min} and p_{max} where p_{min} and p_{max} are respectively the low value and the high value of the input parameter « p », then the sensitivity indicator “S” here defined is mathematically equal to the mean value of the low and high “q” values used in the Tornado diagrams and calculated in Section 4.2.1.
- This definition of “S” does not allow the consideration of the case of simultaneous variations of several parameters as, for instance, the variation of fuel characteristics that implies burn-up, enrichment and cycle length are changed at the same time.

The definition of the output indicator “R” depends on the examined output parameter, as it was previously explained in Chapter 4.1. To summarise, here are the three main output indicator types:

- for output parameters considering flows entering or exiting an installation (such as natural uranium from the mine), the output indicator “R” is defined as the cumulated value (or total value) of the output parameter over the whole scenario;
- for output parameters considering materials stored in an installation (such as spent fuel in an interim storage), the output indicator “R” is defined as the maximal value taken by the output parameter during the scenario;
- for output parameters considering some inventory in the cycle or in the waste (such as plutonium inventory in the cycle or in the waste), the output indicator “R” is defined as the final value reached by the output parameter at the end of the scenario.

The assumption of the linearity of the output indicator “R” as a function of the input parameter “p” is the keystone for considering the sensitivity indicator “S” as a valid parameter. For this reason, the linearity is checked by calculating the coefficient of determination (noted “r²”) as follows:

$$r^2 = \frac{\left[\sum_i (p_i - \langle p \rangle) \cdot (R_i - \langle R \rangle) \right]^2}{\sum_i (p_i - \langle p \rangle)^2 \cdot \sum_i (R_i - \langle R \rangle)^2}$$

When the coefficient of determination “r²” is close to 1 (i.e. greater than 0.9), then the sensitivity indicator “S” can be retained as a valid parameter. Otherwise (i.e. if the coefficient of determination “r²” is lower than 0.9) the sensitivity indicator “S” is suppressed.

It must be noticed that each sensitivity indicator “S” is calculated from a small amount of points (maximum of three points for each sensitivity indicator, i.e. one for the reference case, one for the upper sensitivity case, and one for the lower sensitivity case), so that the results should be considered as tendencies rather than absolute results. Compared with the Tornado Diagram that displays two values of sensitivity (qmin and qmax), the sensitivity indicator S is more global as it is based on a least squares linear regression upon the three available points (reference, upper sensitive case and lower sensitive case). Furthermore, when a same sensitivity indicator “S” can be calculated several times because several sets of points from different codes are available, then the most representative sensitivity indicator “S” is selected and presented in this report.

Table 4.3-1 is an example summary of the sensitivity indicators “S” applied to the input parameter “Reactor lifetime” and showing the various output parameters (one cell for each output parameter). As these results should be considered as tendencies, colours only are indicated in Table 4.3-1:

Table 4.3-1: Example summary of the sensitivity indicators “S” applied to the input parameter “Reactor lifetime”

S (r ² ≥ 0,9)	PWR cycle			FR cycle		Reprocessing	Inventory									
	Front-end		Back-end	Front-end	Back-end		Pu				MA					
	Uranium	Enrichment	Fabrication	Storage	Pu	Fabrication	Storage	PWR	FR	Plants	Reactors	Storages	Cycle	Waste	Cycle	Waste
Reactor characteristics																
Reactor Lifetime						?										

When a sensitivity coefficient is red, this means it is positive: an increase (resp. a decrease) of the reactor lifetime induces an increase (resp. a decrease) of the output parameter. The darker the red sensitivity coefficient is, the more positive it is. On the contrary, when a sensitivity coefficient is blue, this means it is negative: an increase (resp. a decrease) of the

reactor lifetime induces a decrease (resp. an increase) of the output parameter. The darker the blue sensitivity coefficient is, the more negative it is.

When the response between input and output parameters is non-linear, or when the sensitivity can hardly be quantified into a single value, then the sensitivity coefficient is replaced by a question mark “?”: typically, this can occur when the coefficient of determination “ r^2 ” is lower than 0.9.

It can be noted that all the material flows are impacted by a change in reactor lifetime, but each impact is quite small: the colours are mostly pale. The most significant impact is on the plutonium inventory in storage, which appears with a blue colour a bit darker than the other ones. This is due to an increased amount of spent fuel in the pools and interim storages as decommissioning is more frequent when the reactor lifetime is shorter. In general, the coefficients of input parameter on the “reactor lifetime” are negative (blue), which expresses the fact that, when the reactor lifetime is shorter and the commissioning/ decommissioning more frequent, then more fresh fuels are manufactured, more spent fuels are reprocessed, and more waste is stored. Blanks present in the table indicate that the output parameters are not impacted by a change in the reactor lifetime, so that the related sensitivity indicators are not available. A complete table of the sensitivity indicators for each input parameter is provided in Chapter 8.

4.4. Broken scenarios

As previously indicated, in the activity performed, we have run into some cases (depending on the changes applied for the input parameters considered) for which it was impossible to conclude the fuel cycle simulations for the whole period considered of 200 years.

These cases have been referred to as “broken scenarios” cases. Several types of broken scenarios have been found. In the majority of the cases, the scenario fails due to a temporary lack of plutonium for fabricating the required FR fuel (e.g. in cases in which the FR introduction date has been put forward by 10 years). This temporary lack generally occurs during the transition phase. For those scenarios, small variations of other input parameters may solve the problem (e.g. by increasing the annual reprocessing capacity, by advancing the PWR reprocessing plant start-up date or by slightly changing the FR introduction rate).

For other cases, such as the case of a strong increase in energy demand (e.g. +1.5% cased considered by CIEMAT as indicated in Section 5.1), the lack of Pu becomes permanent. Under this condition, the only option available for allowing the scenario to run for 200 years is the adoption of an external Pu feed. This option, however, may be also used for cases with a temporary lack of Pu.

In general, only two main options were chosen by participants to the uncertainties investigations benchmark study in order to solve the problem of “broken scenarios”:

- Option 1: adoption of an external Pu stock that it is called to feed the scenario once the Pu available from reprocessing is not enough (e.g. if minimum cooling time before reprocessing is not achieved). This implies the use of a stock with fixed Pu vector (namely a stock in which the ageing of Pu is not considered). The overall fuel cycle

system may, then, be considered as “open”. This solution has been applied to scenarios with temporary or permanent lack of Pu.

- Option 2: increase of the PWR annual reprocessing capacity. This option allows overcoming the temporary lack of Pu in the scenario maintaining the scenarios as isolated. However, this option is not applicable to all cases as discussed above.

Several other options may be considered for solving the problem of broken scenarios and all of them are valuable options to be considered during an optimisation process. Some alternatives adopted by the participants are listed in Table 4.4-1.

In addition, in this study, the impact of choosing one of the two main adopted options (external Pu feed and PWR reprocessing capacity) has been analysed with respect to some specific cases.

Two test cases, with different complexity, have been selected among the cases considered in the uncertainties study to show the impact of selecting different strategies for responding to a broken scenario.

- CASE A: FR fleet composed by ESFR systems. Only Pu is recycled – MA are sent to the waste.
- CASE B: FR fleet composed by a mixed fleet of ESFR and ASTRID-like burners (1/3). Pu and MA are recycled (Pu in both systems and MA only on burners).

Table 4.4-1: Options considered for avoiding broken scenarios

Option	Description	Used by:
Option 1 (main option)	Adoption of an external infinite Pu stock	CNL, CEA
Option 2 (main option)	Increasing of the PWR annual reprocessing capacity.	KIT, CIEMAT, JAEA
Option 3	Changing a start-up of the PWR reprocessing	JAEA
Option 4	Changing the FR introduction rate	JAEA
Option 5	Reducing FR fuel fabrication time	JAEA
Option 6	Reducing the PWR SNF minimum cooling time	JAEA

As expected, the impact on selecting option 1 or 2 is mainly noticeable for quantities related to FR and to the transition period. Therefore, in the following part, figures of merit related to the PWR front end have been not included.

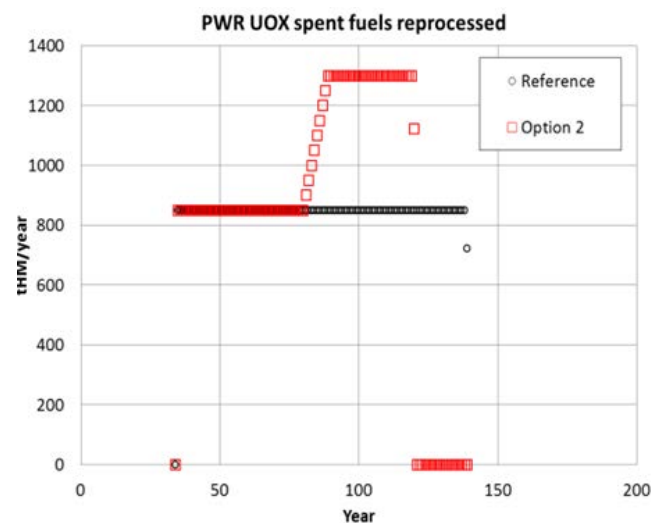
In particular, for option 1 (external Pu feed), a stock of Pu characterised by the fixed vector was used by CEA/KIT as indicated in Table 4.4-2. The other partners have used a

similar vector for their simulation⁴. For option 2, the PWR SNF annual reprocessing capacity, shown in Figure 4.4-1 (annual capacity increased from 850 t/y to 1300 t/y), has been used. Other options may be considered as well.

Table 4.4-2: Option 1: Pu vector adopted for the external feed

Pu isotopes	CEA/KIT	CNL
	%wt.	
²³⁸ Pu	2.7	2.6
²³⁹ Pu	56	57.9
²⁴⁰ Pu	25.9	28.9
²⁴¹ Pu	7.4	0.4
²⁴² Pu	7.3	10.2
²⁴¹ Am	0.7	-

Figure 4.4-1: Option 2: PWR SNF annual reprocessing capacity



4.4.1. CASE A: FR fleet composed by ESFR systems

The first analysed case is the case in which the FR fleet is mainly composed of ESFR systems. This case has been compared with the reference case in Section 6.2.3. In Figure 4.4-2 the FR fissile fuel fabrication needs have been compared. As expected no effect is underlined in using option 1 or 2.

However, the adoption of an external Pu feed (option 1) blended with the Pu coming from the cycle has an impact on the Pu loaded in the FR (see Figure 4.4-3-close to the end of

4. CNL derived its Pu vector from the composition of SFR fuel at the beginning of the transition to FRs in the reference scenario.

FR introduction where the two curves are very similar but not identical). The adoption of different Pu streams with different qualities may slightly affect the Pu content (assuming for instance the same equivalent model) of the batches loaded in core.

The adoption of option 2 (larger PWR reprocessing capacity) has an impact on the reprocessed U interim storage. As indicated in Figure 4.4-4, the impact is limited to the transition period. In the long term, the same behaviour is obtained by adopting the two options.

The adoption of option 1 or 2 has a larger effect on the Pu inventory in the cycle, as indicated in Figure 4.4-5. The first introduction of external Pu feed is clearly visible in Figure 4.4-5 with the jump around year 100.

Concerning the Pu in waste (Figure 4.4-6), the effect is very limited. The difference is due to the amount of reprocessing steps performed (different in the two cases). Similar conclusion can be done for MA in waste (Figure 4.4-7) where the difference is very small and mainly important during the transition period.

Figure 4.4-2: CASE A: FR fissile fuel fabrication needs

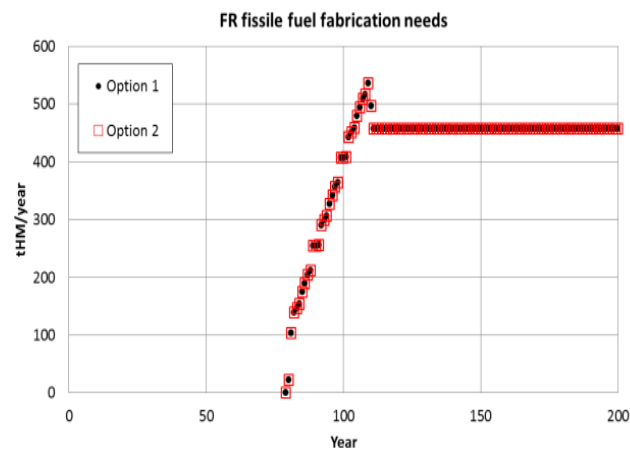


Figure 4.4-3: CASE A: Pu flow for fabrication

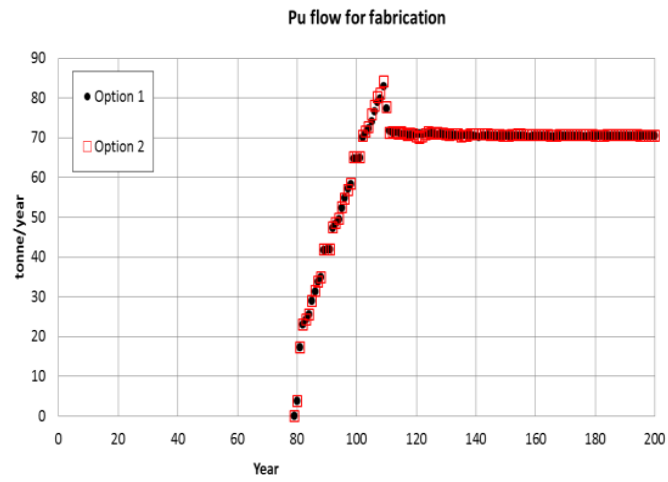


Figure 4.4-4: CASE A: Reprocessed U interim storage

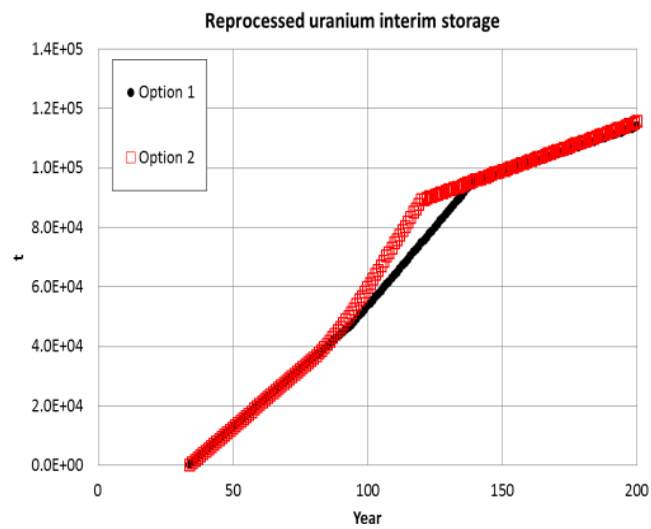


Figure 4.4-5: CASE A: Total Pu inventory in the cycle

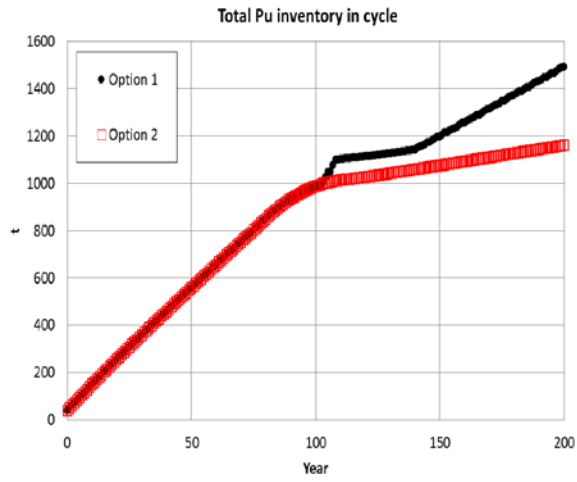


Figure 4.4-6: CASE A: Total Pu inventory in waste

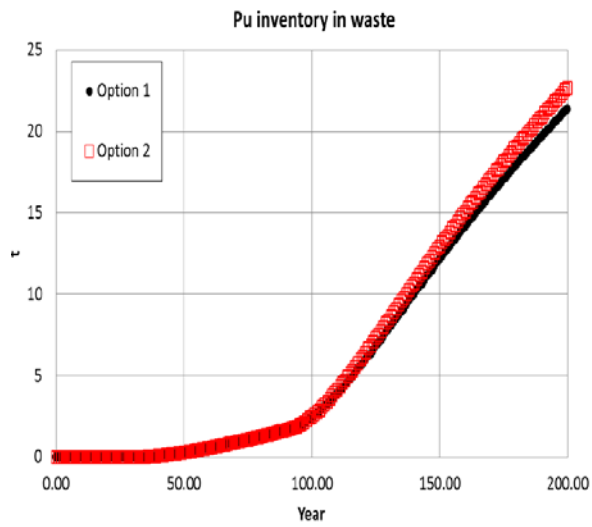
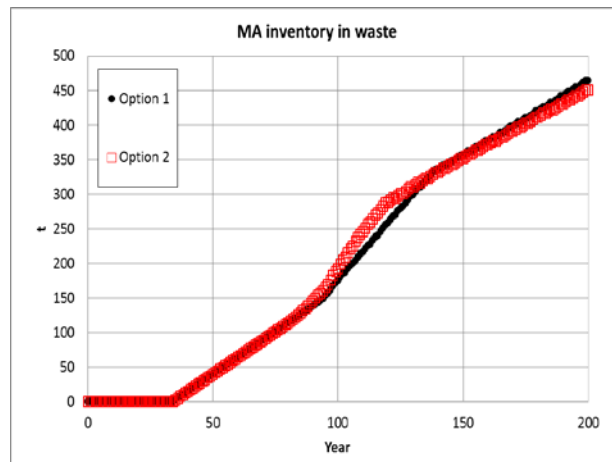


Figure 4.4-7: CASE A: Total MA inventory in waste

4.4.2. CASE B: FR fleet composed by a mixed fleet of ESFR and ASTRID-like burners

The second analysed case is a case in which the FR fleet is composed of a mixed fleet of ESFR systems (2/3) and ASTRID-like burners (1/3). This case has been compared with the reference case in Section 6.2.2 as a possible option for MA burning. In Figure 4.4-8, the Pu flow for fabrication is shown (every batch is represented). Looking at an average value, the difference in using option 1 or 2 is quite pronounced. This is mainly due to the impact of the Pu quality used for fabricating the ESFR and burner cores.

The Pu inventory in the cycle is shown in Figure 4.4-9. As already indicated for case A (Figure 4.4-5) the first introduction of external Pu feed is clearly visible by a jump around year 100.

The introduction of external Pu also leads to an increase of the Pu in storage, as indicated in Figure 4.4-10. Also in this case, the difference appears after year 100.

Finally, Figure 4.4-11 exhibits the total MA inventory in the cycle where a difference between the two options can be seen starting from year 100. This difference can be attributed to the contribution coming from the additional ^{241}Pu (Pu in the cycle introduced by external feed) which decays into ^{241}Am .

Figure 4.4-8: CASE B: Pu flow for fabrication

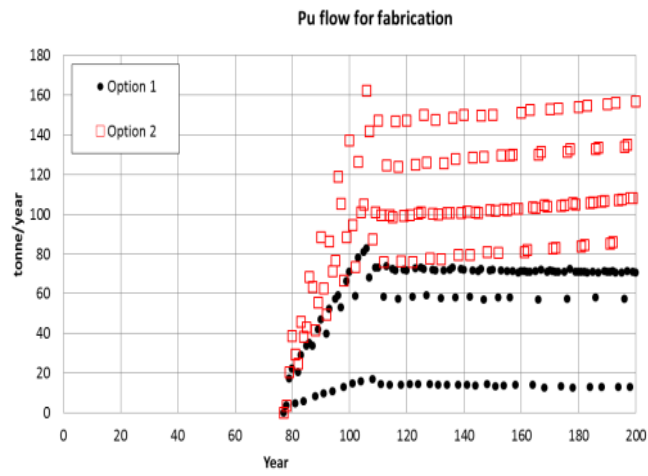


Figure 4.4-9: CASE B: Total Pu inventory in the cycle

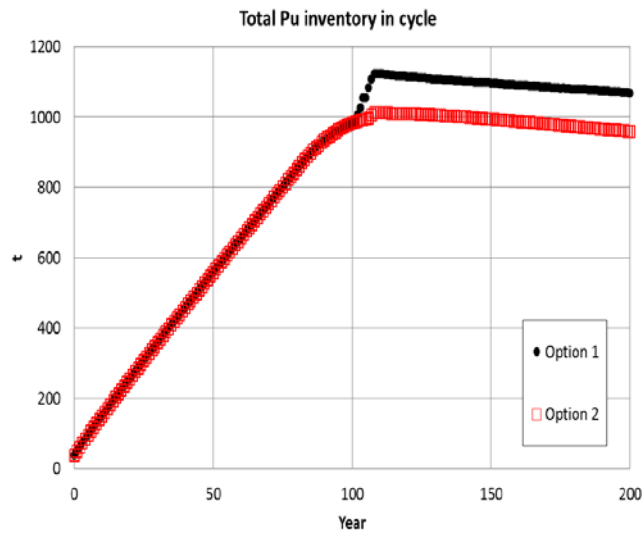


Figure 4.4-10: CASE B: Total Pu inventory in storage

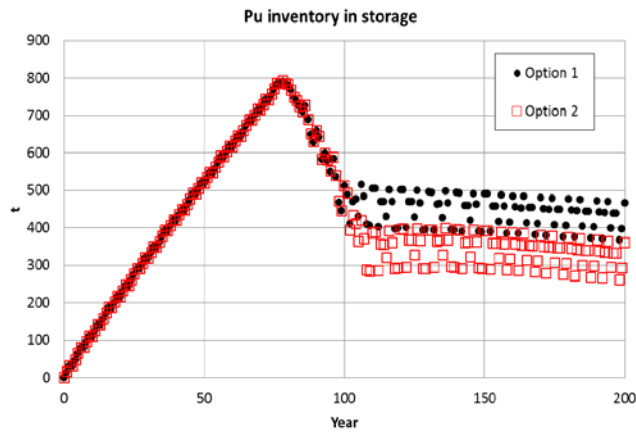
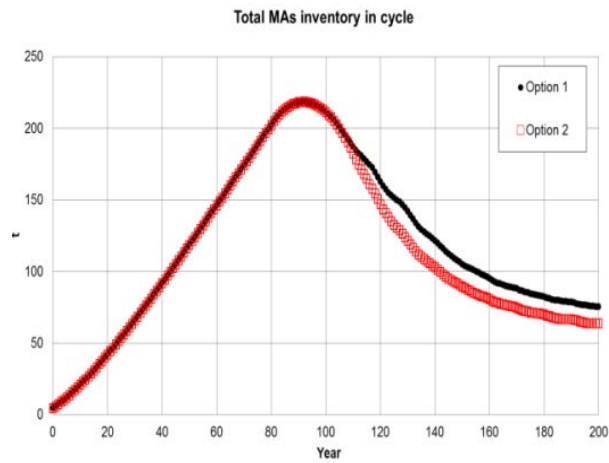


Figure 4.4-11: CASE B: Total MA inventory in the cycle



5. Effects of the uncertainty of the general scenario assumptions

5.1. Total nuclear energy demand

Satisfying an energy demand scenario implies the investigation of a wide spectrum of the nuclear fuel cycles to highlight potential weaknesses in terms of performance of the nuclear fuel cycle components, as well as weaknesses/difficulties of the scenario codes to appropriately describe the adopted scenario.

Such a wide spectrum investigation, as a parametric study, proves to be useful for collecting information on the numerical-behavioural of the parameters in order to create a “set of data” for use as a base for further development, dealing with uncertainties evaluation.

The typical approaches generally adopted for the energy demand- with respect to the temporal dependence - are: fixed amount variation, linear variation and/or exponential variation. In the case of a transition scenario, the time-dependence options could be split into some other sub-options.

In the following section, the results of evaluations of the impact of the energy demand variation will be presented and discussed. The main objective is to highlight the potential peculiarities of each type of energy variation on a given parameter of the investigated scenario.

The constraint of the sustainability of an energy demand scenario involving FRs deployment is assured by the availability of plutonium (Pu) in the fuel cycle, even though external sources. With the particular character of this type of studies, such a constraint has sometimes been intentionally not satisfied. Such a choice allows evaluating the amount of Pu lack, information which is necessary for the sustainability of the scenario itself. When available, the information will be explicitly mentioned. Results of this section have been provided by using the followings codes:

- COSI (CEA, KIT, ENEA);
- TR_EVOL (CIEMAT and ;
- VISION (CNL).

Each organisation has analysed some specific, among the possible scenarios. A common unit for the displayed scenario quantities, tHM/year, was used.

5.1.1. Increase/decrease by a fixed amount

Evaluations have been performed by ENEA, assuming a constant variation of $\pm 5\%$ of the energy with respect to the reference scenario. In the case of a 5% increase, for scenario sustainability, 48.9 tonnes of Pu are required (indeed, a lack of 48.9 of Pu tonnes has been

observed at year 108 to 110), the LWR shut-down being completed at year 110. The missing amount of Pu can be recovered by anticipating the start of the LWR SF reprocessing by 5 years or by an increase of the LWR reprocessing plant capabilities. No Pu lack has been observed in case of 5% decrease of the energy demand.

A first result concerns the common behaviour, in equilibrium regime, of the scenario parameters: they vary of $\pm 5\%$ respectively, with respect to the reference case. Figures 5.1-1, 5.1-2 and 5.1-3 show this behaviour, for scenario parameters dealing with the uranium needs of the LWRs. SWU/year has been evaluated through the indicated formulation, [1,2] and the enrichment and tail specification/requirements.

Figure 5.1-1: Natural U needs (tHM/year)

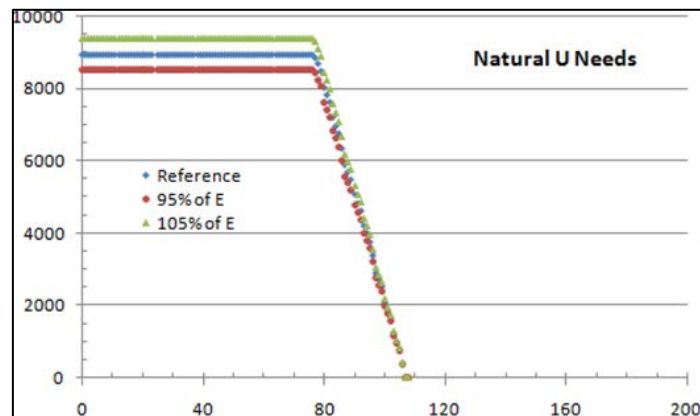


Figure 5.1-2: Enriched U needs (tHM/year)

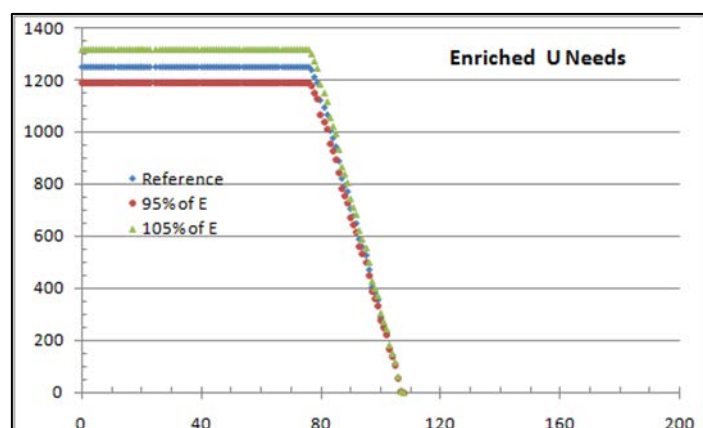
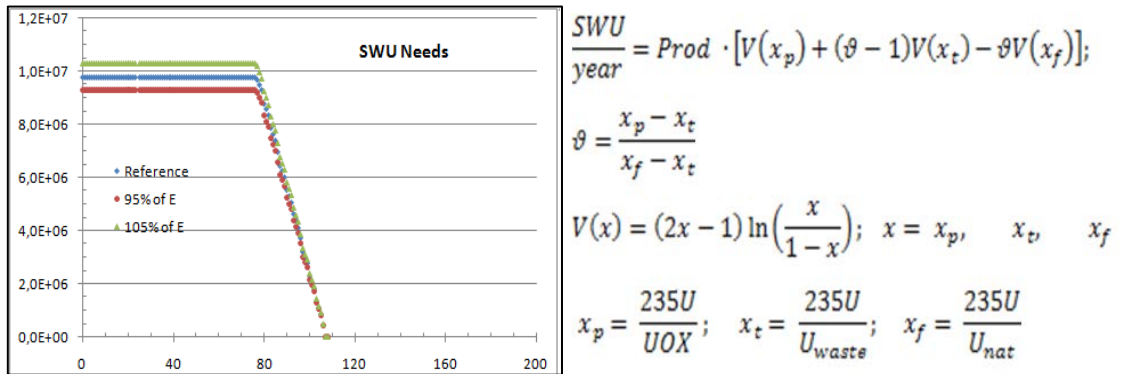


Figure 5.1-3: Separative work unit needs (SWU/year)



A similar behaviour has been observed for the materials in interim storage: depleted uranium and reprocessed uranium (Figures 5.1-4 and 5.1-5). At the end of the scenario, the amount of reprocessed uranium reaches values between about 106 and 116 thousand tonnes, while depleted uranium reaches values between about 725 and 800 thousand tonnes at the end of the PWRs shut-down (year 110).

Figure 5.1-4: Depleted U in interim storage (tHM)

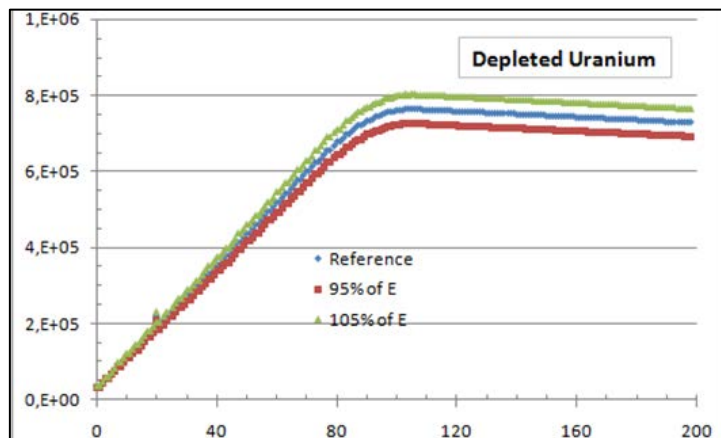
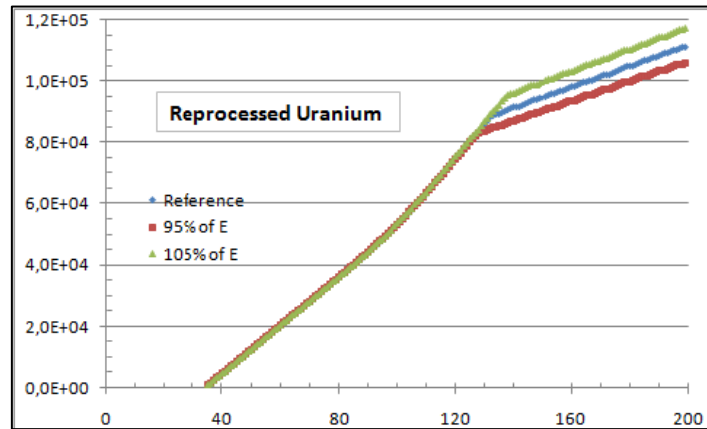


Figure 5.1-5: Reprocessed U in interim storage (tHM)

The behaviour of scenario parameters is quite similar, as shown in Figures 5.1-6 and 5.1-7 for the Pu flow for fabrication and FRs reprocessed spent fuel respectively, or FRs fuel fabrication needs (see Figures 5.1-8 and 5.1-9).

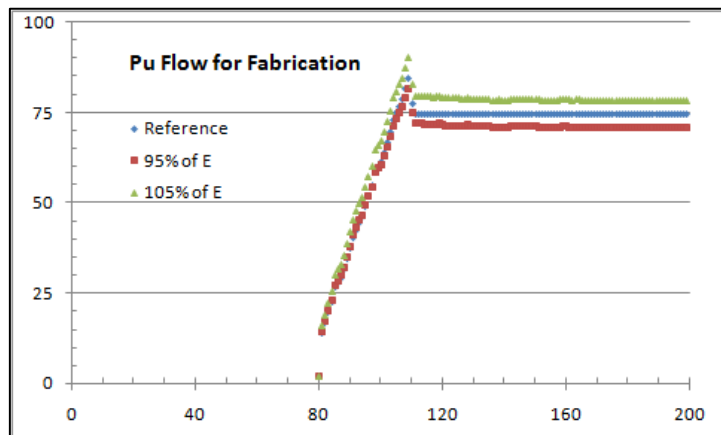
Figure 5.1-6: Pu flow for fabrication (tHM/year)

Figure 5.1-7: FRs reprocessed spent fuel (tHM/year)

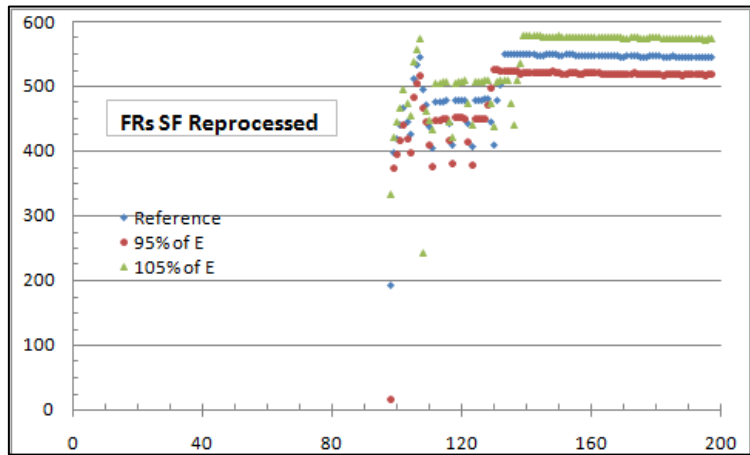


Figure 5.1-8: FRs fissile needs (tHM/year)

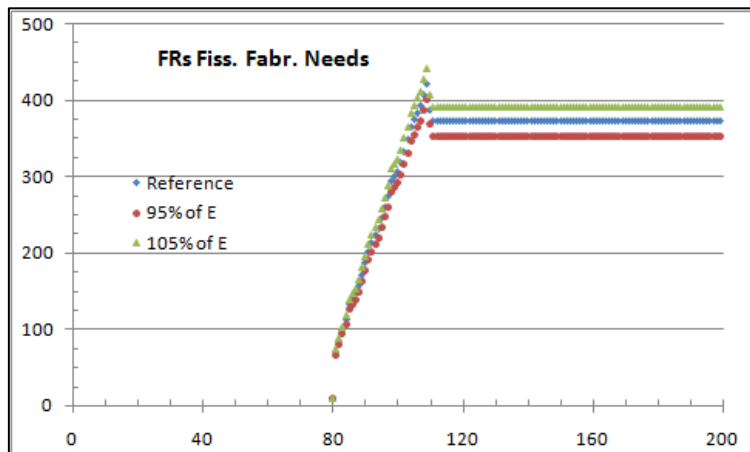


Figure 5.1-9: FRs fertile needs (tHM/year)

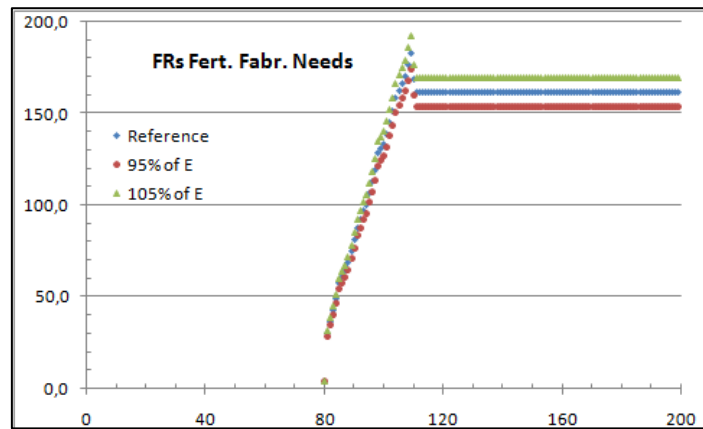
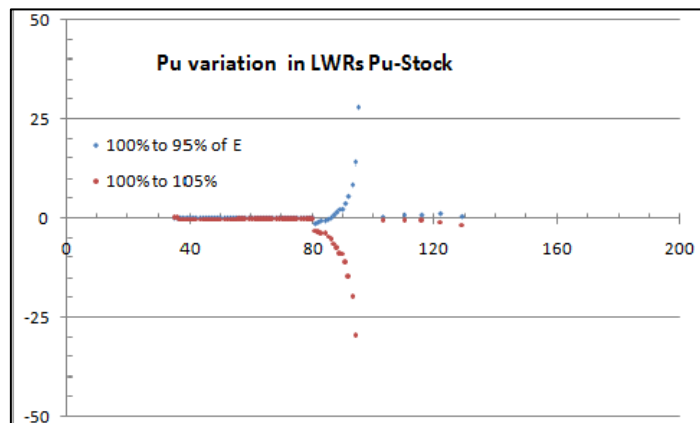


Figure 5.1-10: Percentage variation of separated Pu in LWR Pu stock



It is worth mentioning that the energy variation does not significantly affect the LWRs Pu in interim storage until the start of FRs fuel fabrication, as displayed in Figure 5.1-10, which shows the behaviour of the percentage variations with respect to the reference case. Concerning the Pu inventory in the fuel cycle, the inventory in plants and reactors follows the same behaviour. Figures 5.1-11 and 5.1-12 show the percentage variations with respect to the reference case; both confirm the above mentioned behaviour.

A significantly different situation concerns the Pu inventory in the whole fuel cycle, which includes Pu inventory in Plants, NPPs and Storages. The last one increases until the start of the reprocessing of the LWRs spent fuel, and decreases with FRs deployment, as displayed in Figures 5.1-13 and 5.1-14, respectively for the Pu inventory in storage and in the fuel cycle.

Figure 5.1-11: Percentage variation of Pu in NPPs

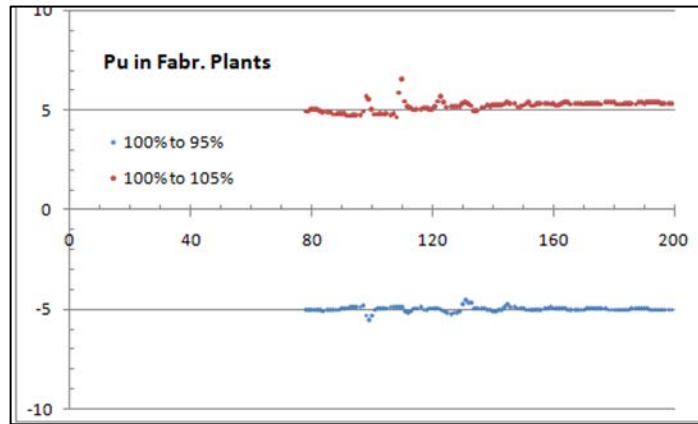


Figure 5.1-12: percentage variation of Pu in NPPs

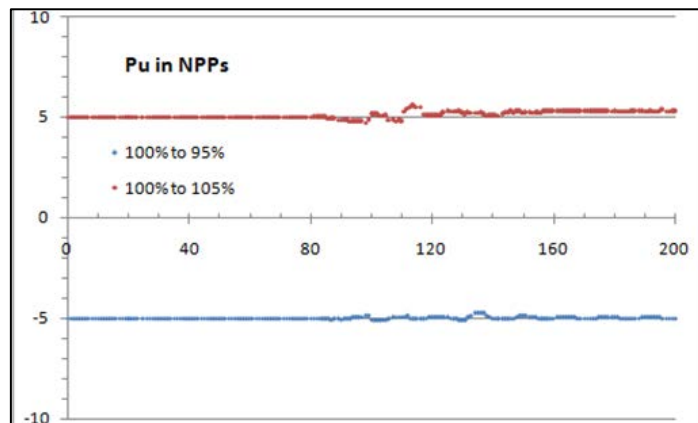
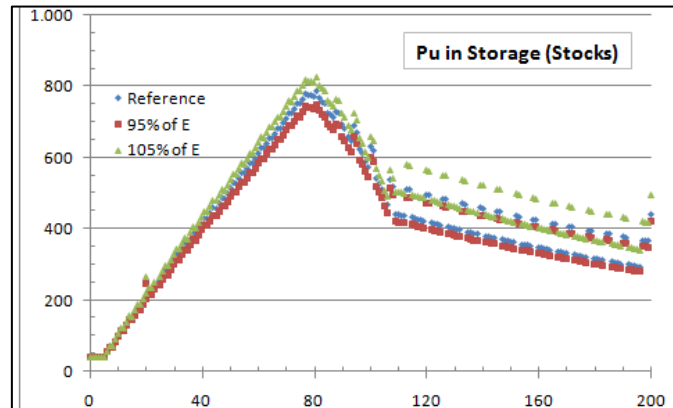
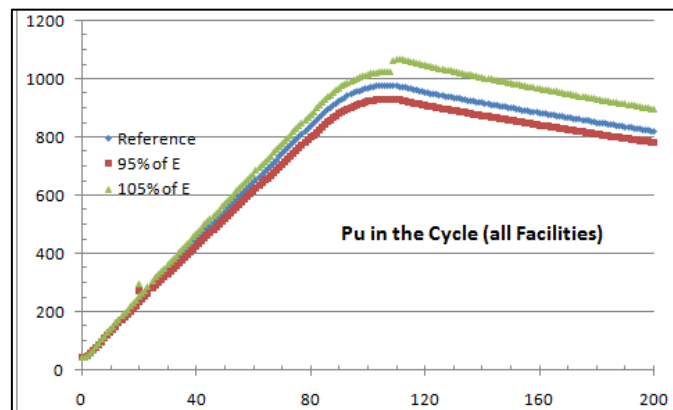


Figure 5.1-14 shows the behaviour of the Pu inventory in the fuel cycle: the discontinuity, around year 108 (Figure 5.1-13), in the case of 5% increase of the energy demand, shows the time and size of the external Pu need, for the scenario sustainability. This discontinuity corresponds to the gradient variation of the Pu inventory in the interim storage.

Figure 5.1-13: Pu inventory in storage (tHM)**Figure 5.1-14: Pu inventory in fuel cycle (tHM)**

The consequences of this discontinuity, due to a lack of Pu, are evident in Figures 5.1-15 and 5.1-16, showing the percentage variations with respect to the reference case.

Figure 5.1-15: Percentage variation of Pu in storage

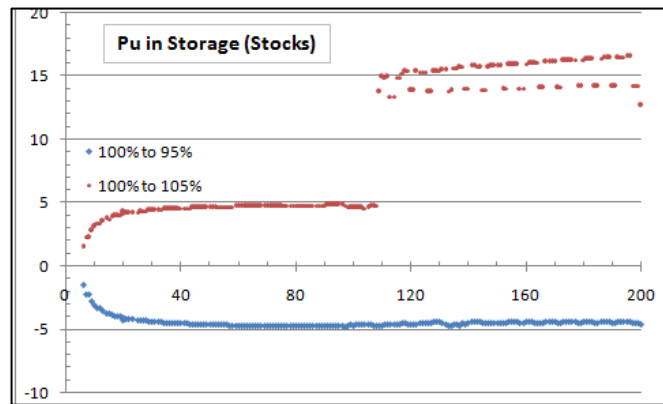
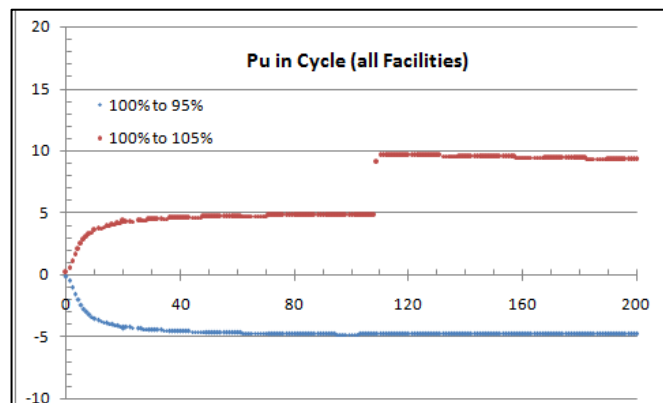


Figure 5.1-16: Percentage variation of Pu in fuel cycle



In the case of lack of Pu, a supplementary uncertainty should be taken into consideration, even if, a priori, it could be not evaluated. Indeed about 50 tonnes of Pu lack, in a relatively short period of the scenario (about 2 years), introduces and propagates, until the end of the scenario, a doubling of the considered energy's percentage variation for the Pu inventory in cycle, while the same energy's percentage variation is trebled in case of Pu inventory in storage.

When the Pu lack is not observed, as in the scenario at 95% of energy demand, the Pu inventory variation follows the general rule, previously mentioned, of the energy variation, i.e. it varies a - 5% with respect to the reference case.

Figure 5.1-17: Pu inventory in waste (tHM)

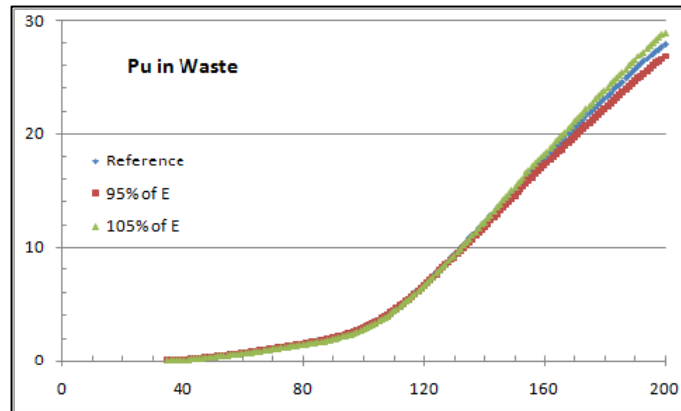
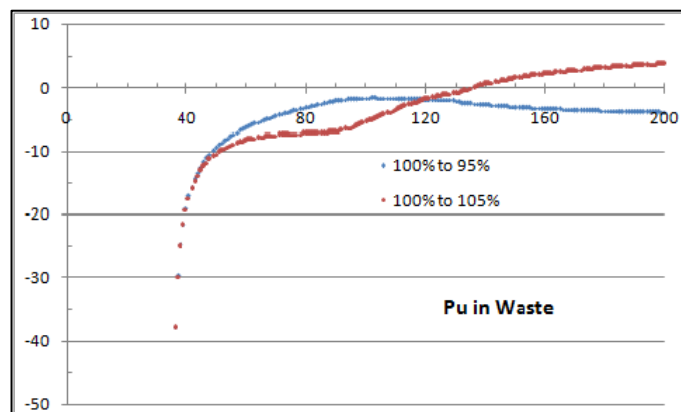


Figure 5.1-18: Percentage variation of Pu in waste

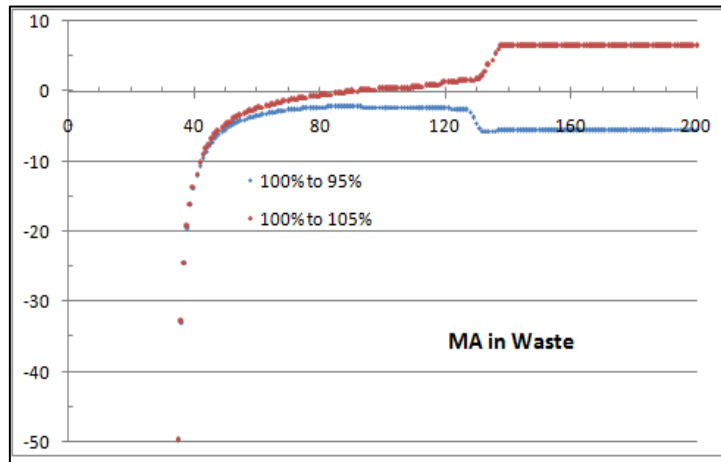
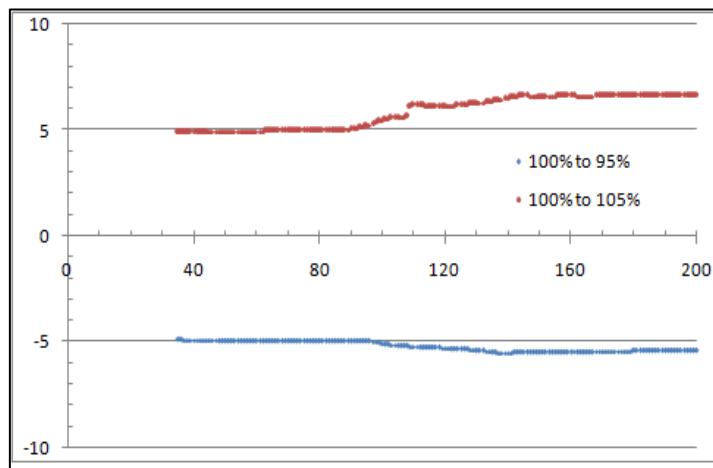


Reprocessing plant losses of 0.1% produce, at the end of the scenario, about 28 tonnes of Pu for the reference case, which are kept in a “Pu Waste” stock.

The time distribution of the Pu inventory in waste is displayed in Figure 5.1-17, while the percentage variations of the two other scenarios, with respect to the reference one, are displayed in Figure 5.1-18. These variations do not exceed the $\pm 5\%$ of the energy demand variation, even if (in principle) it is not an immediate evaluation especially in the transition phase.

The MAs inventory is the direct consequence of 100% “losses of the reprocessing plants” because there are no MAs recycling in the scenario. The MAs inventory reaches about 420 tonnes for the reference case at the end of the scenario. The MAs inventory in the fuel cycle reaches about 432 tonnes and about 12 tonnes⁵ in all the other components of the fuel cycle.

5. These numbers referred to a figure not shown in this report; nonetheless Figure 5.1-43 (refers to a linear increase in energy demand) includes and shows the MA inventory for the reference case scenario.

Figure 5.1-19: Percentage variation of MA in waste**Figure 5.1-20: Percentage variation of MA in cycle**

The shape of the time distribution is very similar to the one of reprocessed uranium, Figure 5.1-5, while the behaviour of the percentage variations of the two other scenarios, with respect to the reference one, are displayed in Figures 5.1-19 and 5.1-20. Even in the case of the presence of MAs in waste, the variations with respect to the reference case do not follow the constant variation of the energy demand. On the contrary, in the case of MAs inventory in the cycle, these variations are close to the energy demand variation, but with different numerical responses. The case of +5% variation of the energy demand is impacted by the lack of Pu, i.e. of the external Pu need, as also seen in Figure 5.1-20, which would require supplementary reprocessing of the spent fuel.

5.1.1.1. Recovering the Pu Lack

In recovering the Pu lack, two options have been adopted: (1) start reprocessing spent fuel 5 years earlier; or (2) increase the reprocessing plants annual capacity to 900 tHM/year instead of the nominal 850 tHM/year.

Figures 5.1-21 and 5.1-22 show the Pu inventory in the fuel cycle for these options, as well the percentage variations with respect to the case where external Pu is needed.

**Figure 5.1-21: Pu inventory in cycle
(all cases)**

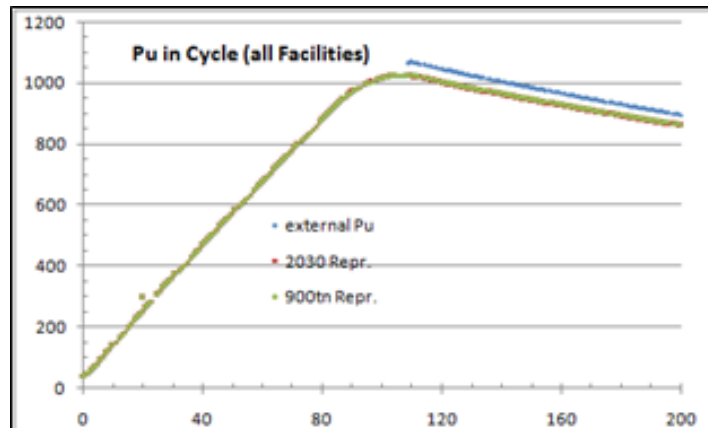
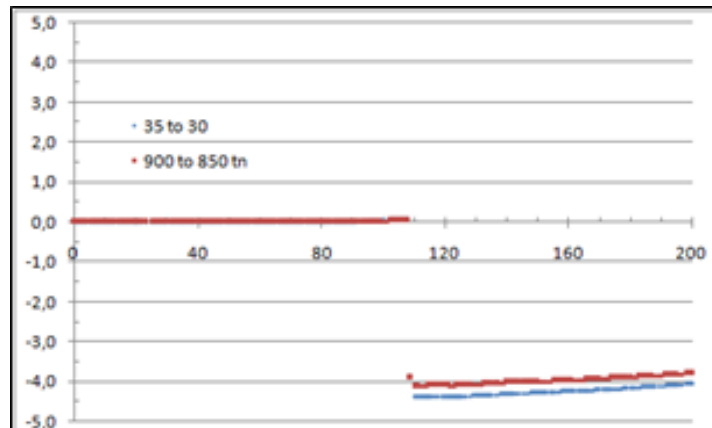


Figure 5.1-22: Percentage variation of Pu in cycle



Both options provide very similar results. They seem to be equivalent solutions to solve the issue with lack of Pu, which are valid for variations of the energy demand of some units.

5.1.2. Linear increase in energy demand

Three different scenarios have been analysed, corresponding respectively to increases of +12.5%, +50% and +100% of the reference energy demand. For all three scenarios, a lack of

Pu has been observed, respectively of 62, 1450 and 3825 tonnes. Even when the same scenario code was used, studies were performed in slightly different conditions. These discrepancies sometimes lead to differences on the results even of the reference cases, for example on nominal energy values. Therefore, any comparison, when available, will refer to the own reference case.

The obtained results show many similarities with respect to the ones for the constant increase of the energy demand. This was also observed for the common behaviour in equilibrium regime, i.e. to vary with respect to the reference case following the law of linear increase in energy demand. For instance, for the scenario parameters dealing with the uranium needs of the LWRs, Figure 5.1-23 shows natural uranium needs, with peak values respectively of 9393 (in year 76), 10752 and 12646 tHM/year (both in year 74). Enrichment and SWU needs show the same behaviour, following the same shape (see Figures 5.1-2 and 5.1-3), reaching the peak values respectively of 1312 (in year 76), 1507 and 1772 tHM/year (both in year 74) for the enrichment needs and $1.028E7$ (in year 76), $1.178E7$ and $1.385E7$ SWU/year (both in year 74) for the SWU needs.

It should be noted that the percentage variations, with respect to own reference cases, of the parameters of all the uranium needs are governed by the same behaviour and values, as shown in Figure 5.1-24, including PWR UOX fabrication needs. These percentage variations are shown in Figure 5.1-23 with peak values of 1316 (in year 78), 1507 and 1772 HMt/year, respectively.

Figure 5.1-23: Natural U needs (tHM/year)

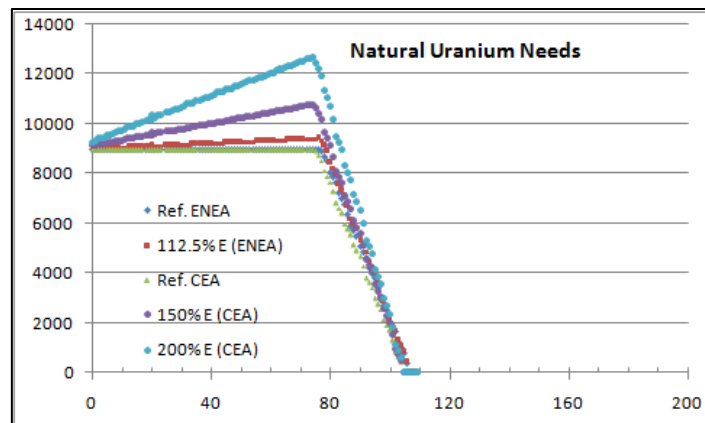
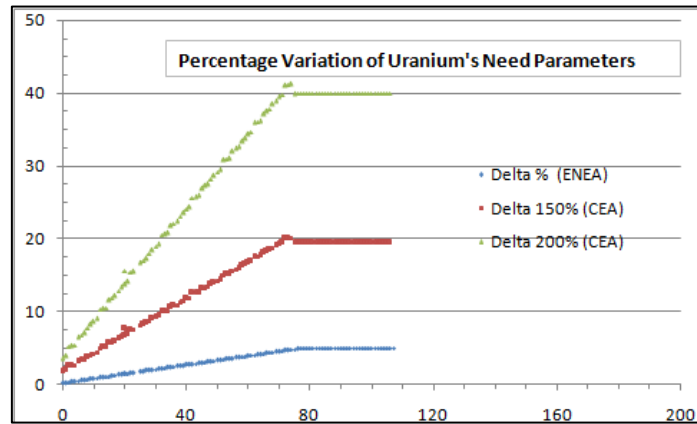


Figure 5.1-24: Percentage variation of U needs



Results for the FRs fuel fabrication needs are displayed in Figures 5.1-25 and 5.1-26, while Figures 5.1-27 and 5.1-28 show the Pu inventory in reprocessing and fabrication plants and, the Pu flow for fabrication respectively.

Figure 5.1-25: FRs fissile fabrication needs (tHM/year)

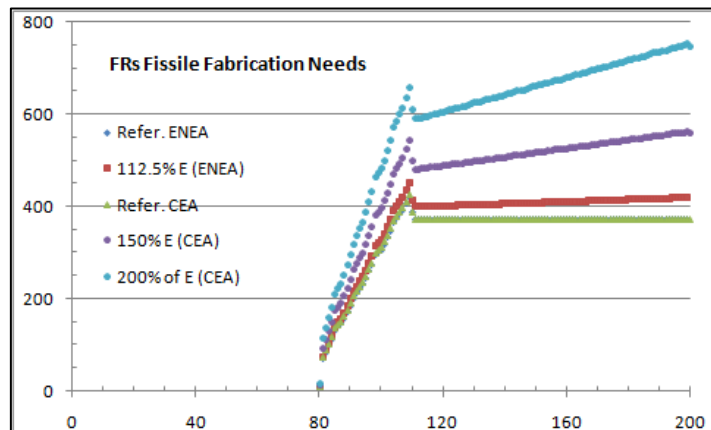


Figure 5.1-26: FRs fertile fabrication needs (tHM/year)

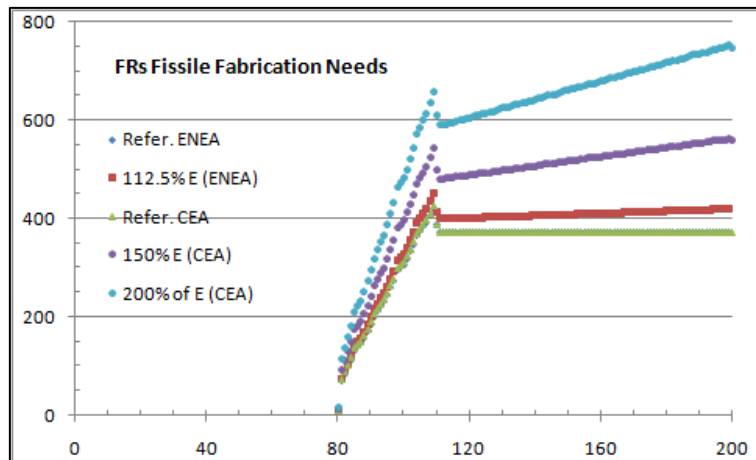


Figure 5.1-27: Pu inventory in NPPs (tHM)

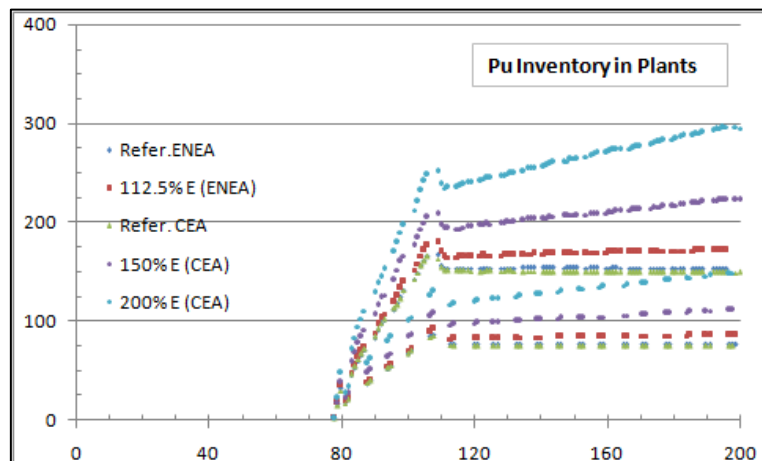
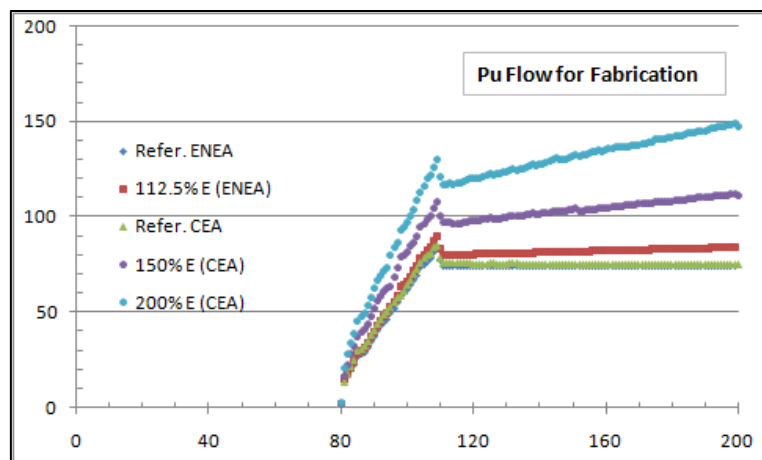


Figure 5.1-28: Pu flow for fabrication (tHM/year)



The percentage variations, with respect to the own reference cases, showed in Figure 5.1-29, are very close to the corresponding percentage variations of Pu inventory in the fabrication and reprocessing plants, as well as of the Pu flow for fabrication (see Figure 5.1-30). In equilibrium regime, the gradients of the variation follow the laws of linear energy increase, which does not happen during the transition phase.

Figure 5.1-29: Pu inventory in NPPs (tHM)

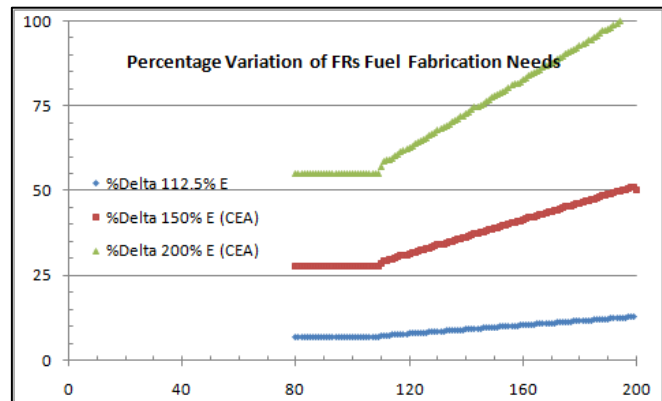
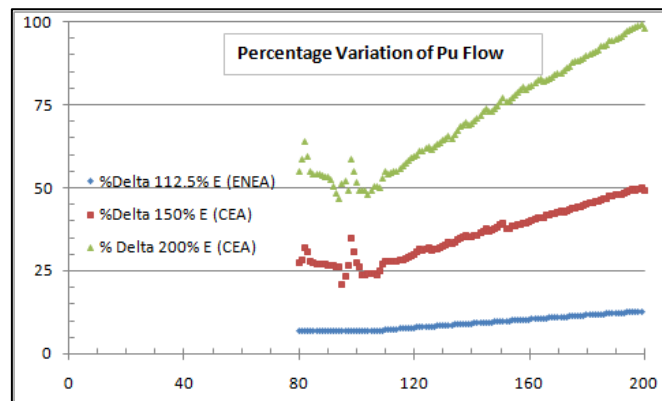


Figure 5.1-30: Pu flow for fabrication (tHM/year)



Other interesting results show the amount of spent fuel in storage, and the spent fuel reprocessing plant capacities. Figures 5.1-31 and 5.1-32 show the PWR UOX and FR MOX spent fuel in the storage.

Figure 5.1-31: UOX spent fuel in storage (tHM)

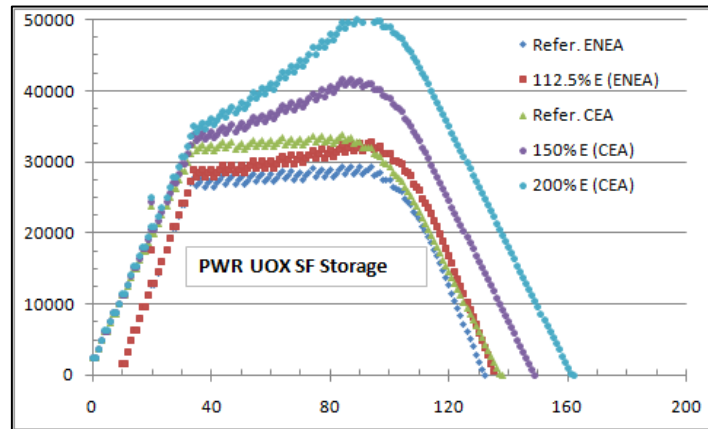
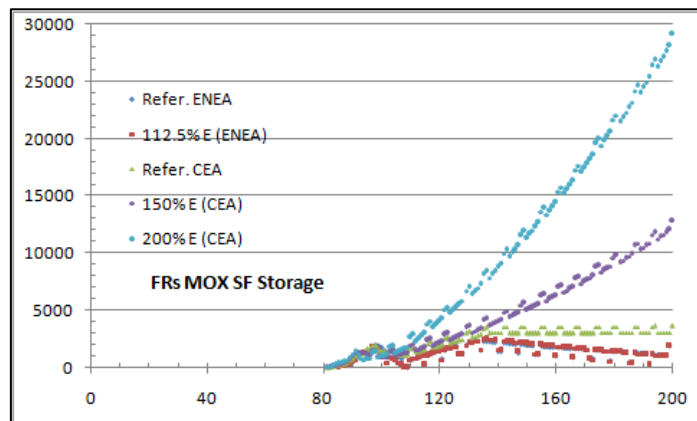


Figure 5.1-32: MOX spent fuel in storage (tHM)



Both these parameters of the scenario are characterised by some differences between the own reference cases related to the different options adopted such as, for example, the breeding gain. Regarding the spent fuel reprocessing plant capacities, the UOX fuel reprocessing plant has been fixed at 850 tHM/year. FRs spent fuel reprocessing rate performances are shown in Figure 5.1-33. They are not fixed but on-demand (see Appendix A.3). It should be noted that the percentage variation, with respect to the own reference case, shows a significant spread of variability in Figure 5.1-34.

Figure 5.1-33: FRs SF reprocessing capacity (tHM/year)

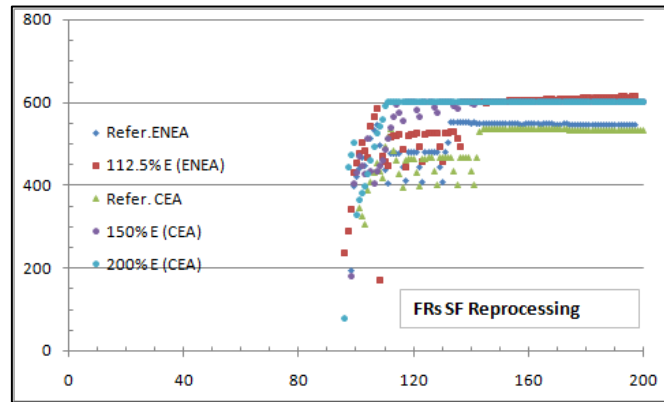
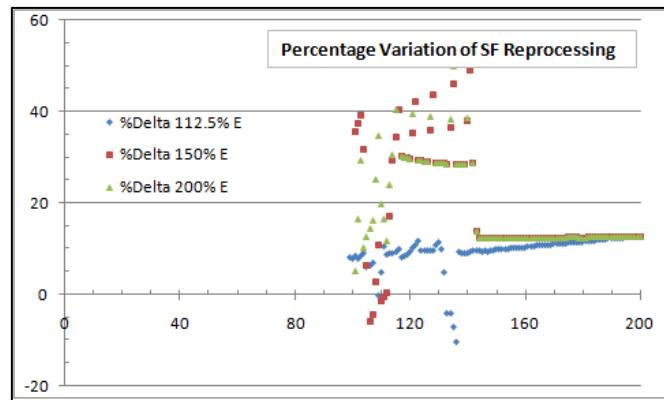
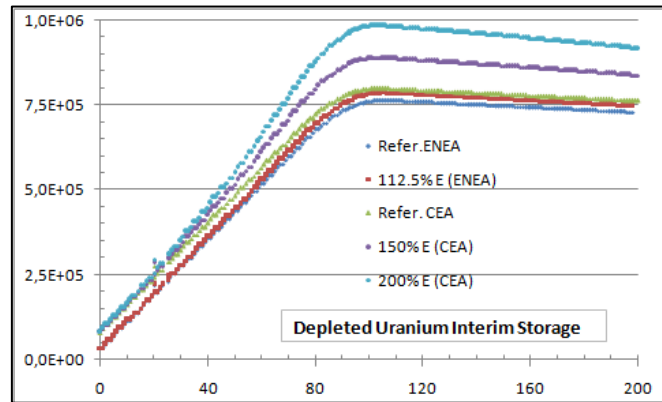
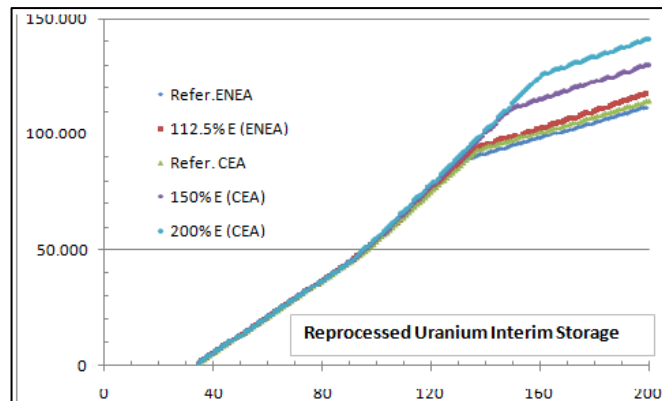


Figure 5.1-34: Percentage variation FRs reprocessing



Concerning the materials in interim storage, the depleted uranium and reprocessed uranium behaviours are similar to the one described in Section 5.1.1. Figures 5.1-35 and 5.1-36 show the results.

Figure 5.1-35: Depleted U in interim storage (tHM)**Figure 5.1-36: Reprocessed U in interim storage (tHM)**

For both parameters, despite the considered energy variations, the percentage variations prove to be about four (4) times lower than the energy ones.

From the point of view of the uncertainties, the situation is significantly different concerning TRU inventories in the fuel cycle. It has been shown that for the Pu inventory in the plants (reprocessing and fabrication) the gradients of the variations follow the laws of linear energy increase, reaching respectively the energy demand variations, i.e. 12.5%, 50% and 100%.

Similarly, some equivalent conditions rule the Pu inventory in the NPPs. Both the Pu inventory and its corresponding percentage variation follow the imposed linear energy increase, except for the transition regime. The results are displayed in Figures 5.1-37 and 5.1-38.

Figure 5.1-37: Pu inventory in NPPs (tHM)

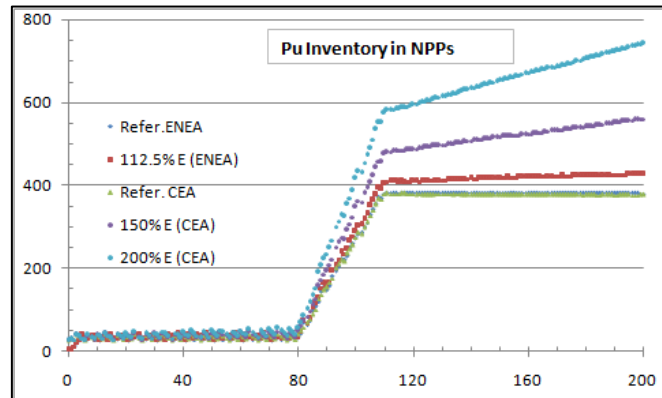
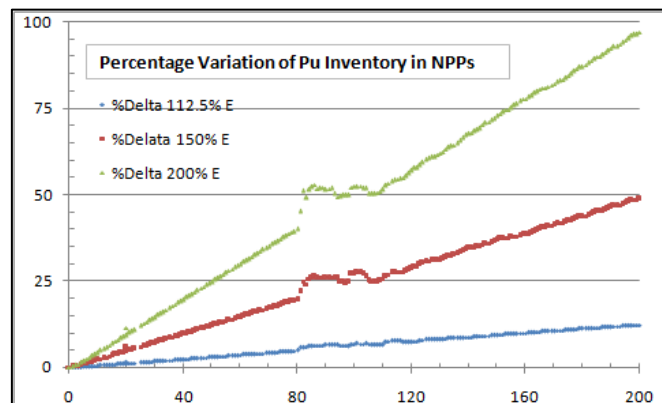


Figure 5.1-38: Pu inventory in NPPs percentage variation



Concerning the Pu inventory in storage and in all facilities (whole fuel cycle), an increase of four to five times was observed (see Figures 5.1-39 and 5.1-40), which implies a percentage variation four to five times higher than the increase of the energy demand (Figures 5.1-41 and 5.1-42), in particular in the case of 100% increase of the energy demand.

Figure 5.1-39: Pu inventory in storage (tHM)

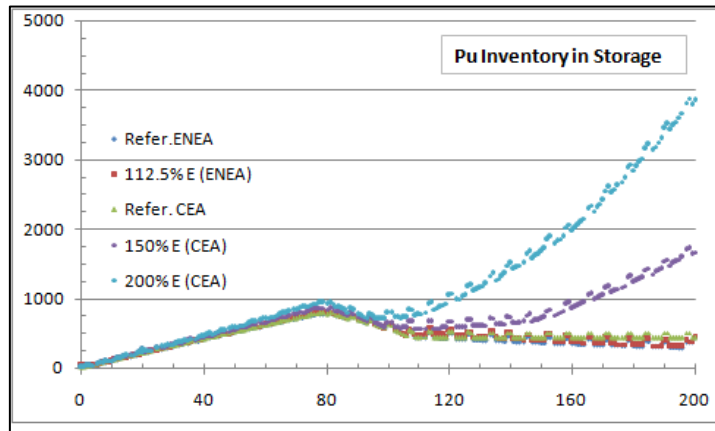


Figure 5.1-40: Pu inventory in cycle

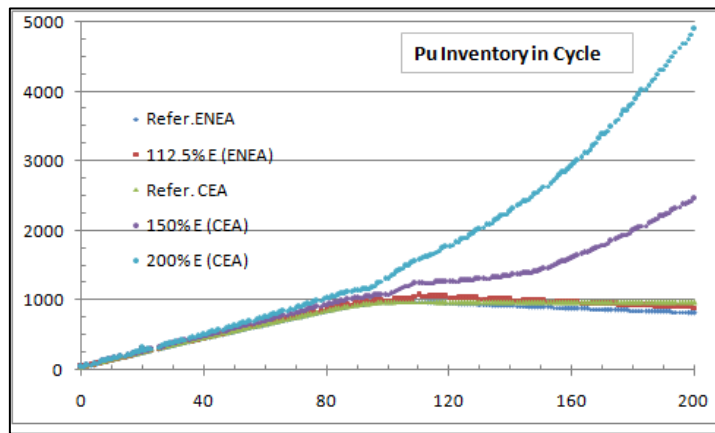


Figure 5.1-41: Pu inventory in storage percentage variation

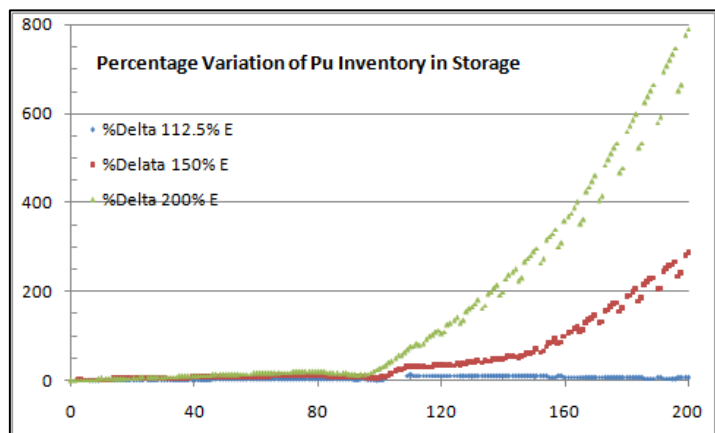
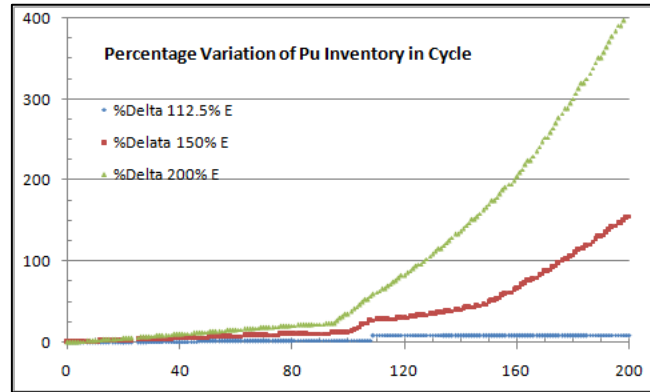


Figure 5.1-42: Pu inventory in cycle percentage variation



Finally, for the MA inventory, the situation is close to the one of Pu. For the MA inventory in waste stock, the increases with respect to the own reference cases reach approximately half of the energy increase for each case as shown in Figures 5.1-43 and 5.1-44.

Figure 5.1-43: MA inventory in waste (tHM)

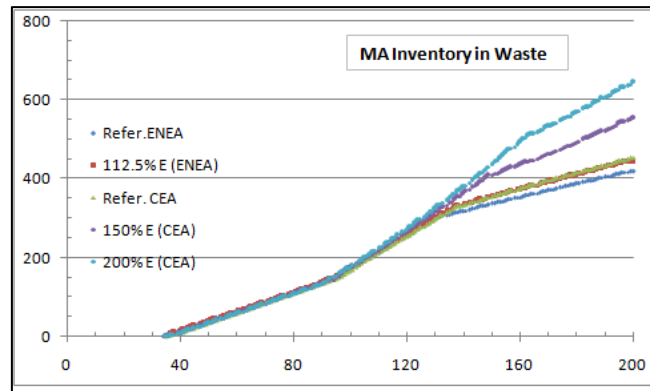
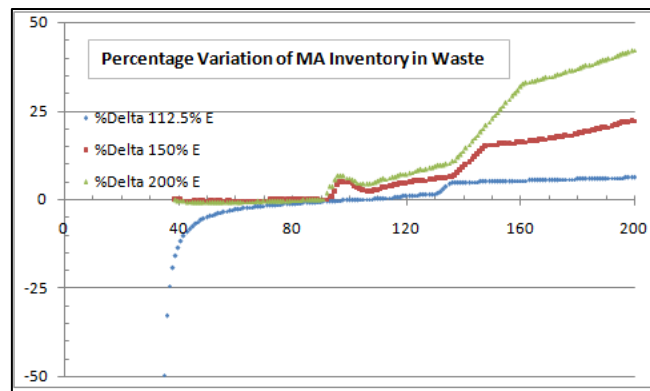


Figure 5.1-44: MA inventory in waste percentage variation



The above results refer to 100% of the MA losses from reprocessing of the spent fuel and depend on the non-linearity of both the MA build-up and the spent fuel decays.

The same situation has been observed in the MA inventory in all the other components of the fuel cycle. Figures 5.1-45 and 5.1-46 show the MA inventory in the cycle (inventory in waste is excluded). In this case, the impact of the energy demand increase, implying more fuel in the fuel cycle components (under non-linear effects of the transmutation rates), is more evident. Relative maximum peaks, respectively of 66.3% (at year 132), 168.1% and 485.3% (at year 138) were observed (see Figure 5.1-46) while, at the end of the scenario, the percentage variations reach the values of 9.1%, 348.2% and 1104.7%, respectively.

Figure 5.1-45: MA inventory in cycle (tHM)

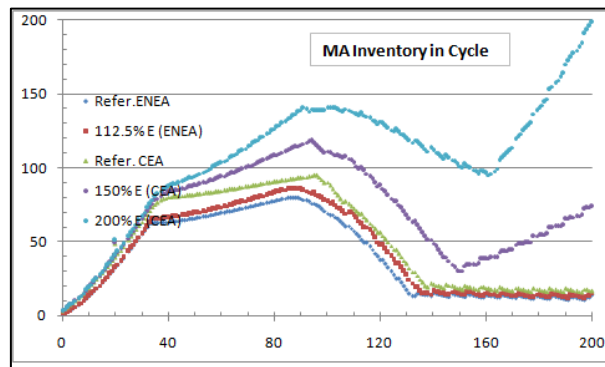
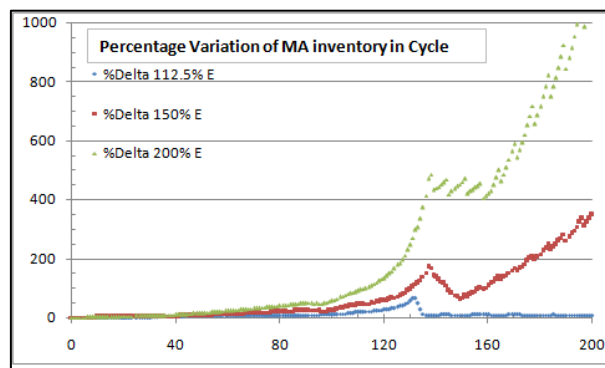


Figure 5.1-46: MA inventory in cycle (percentage variation)



5.1.3. Linear decrease in energy demand

In this section the behaviour of the scenario parameters has been investigated by varying the nominal value of the energy demand, following a linear variation law:

$$E(t)=E_0 \pm \Lambda t; \quad \text{with: } E(0) = E_0; \quad E(80) = (1 \pm 5\%) E_0; \quad \Lambda \equiv \text{Constant}$$

the reached value of the energy being kept constant in the period from year 80 until the end of the scenario in order to compare the results with the ones from the study at constant energy variation, of $\pm 5\%$, presented and discussed in Section 5.1.1.

Similar to the energy demand constant variation cases, the LWRs fuel needs, i.e. consumption, enrichment, separative work units and PWR-UOX are ruled by the same law, of the linear energy variation. The percentage variations, for all the parameters, are described by the same figure with identical numerical results (see Figures 5.1-47 and 5.1-48), and symmetric behaviour with respect to the reference scenario. The profiles follow the imposed energy demand variation law, of fixed amount variation after year 80.

Figure 5.1-47: Natural U needs (tHM/year)

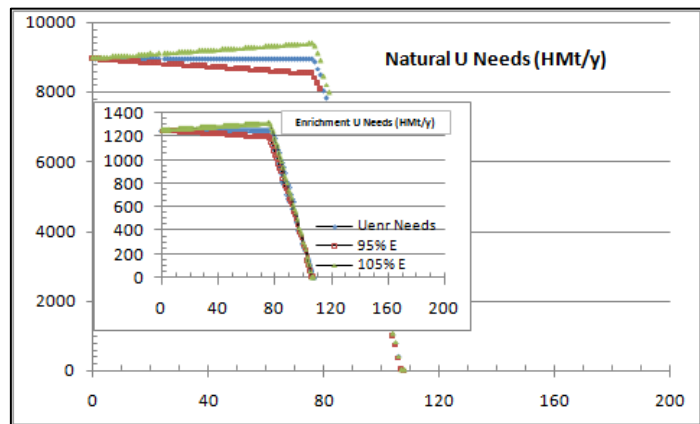
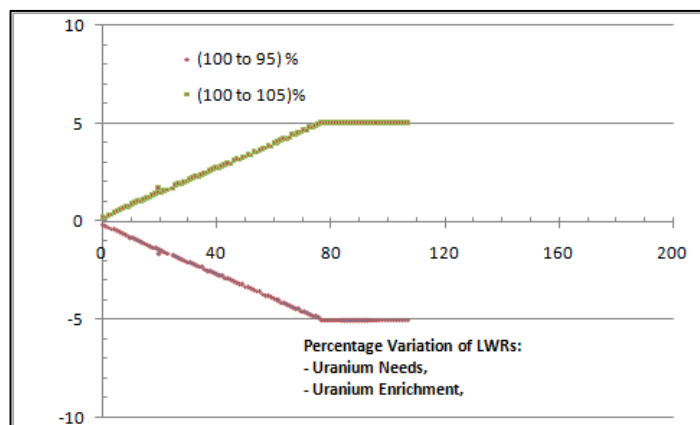


Figure 5.1-48: Percentage variation of U needs



The same kind of considerations can be made with respect to scenario parameters such as depleted and reprocessed uranium. In Figures 5.1-49 and 5.1-50, where percentage variations are shown, it was observed that variations follow the law of the imposed variation on the energy demand and do not exceed the limits of the imposed variations.

Figure 5.1-49: Natural U needs

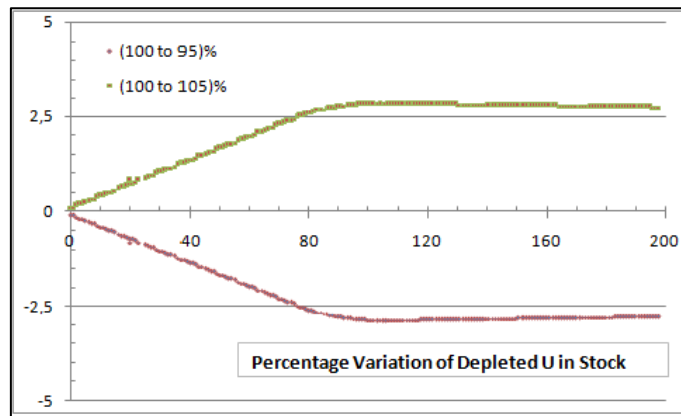
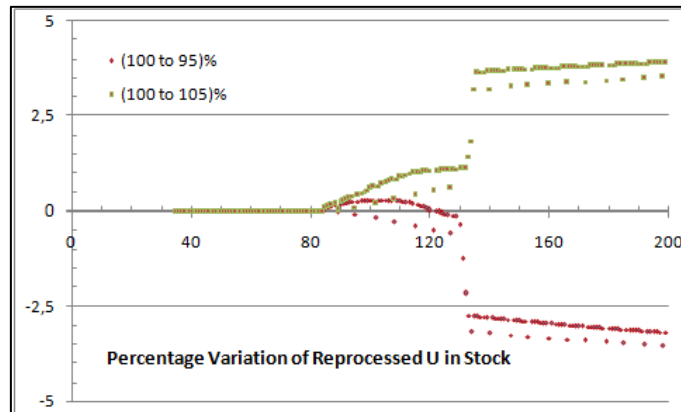


Figure 5.1-50: Percentage variation of U needs



A fully equivalent situation has been observed for the scenario parameters concerning the FRs. Indeed, profile of FRs fuel needs, i.e. fissile and fertile needs, Pu flow for fabrication, and FRs reprocessed spent fuel needs, follow the same behaviour as indicated in Figures 5.1-26, 5.1-27 and 5.1-28. For example, Figures 5.1-51 and 5.1-52 show, respectively, the FRs' variations of reprocessed spent fuel and percentage variations of the same parameter. The last one shows the symmetric behaviour and the asymptotic stabilisation of the percentage

variations; both results were expected due to the law of variation adopted for the energy demand.

Figure 5.1-51: FRs SF reprocessing (tHM/year)

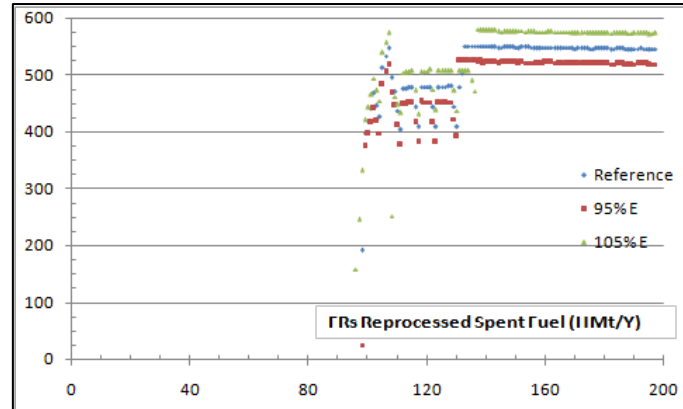
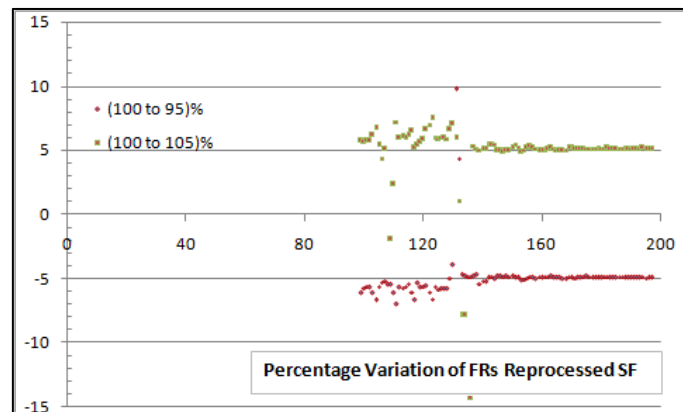


Figure 5.1-52: Percentage variation of FRs SF reprocessing



Regarding the Pu inventory in cycle, the obtained results exhibit some differences, with respect to the increase/decrease (by fixed amount) scenarios, following the adopted law of the energy demand variation. Instead of the behaviour displayed in Figure 5.1-12, referring to the percentage variation of the Pu inventories in NPPs, the current situation is showed in Figures 5.1-53 and 5.1-54, while Figures 5.1-55 and 5.1-56 show the situation regarding the Pu inventories in storage.

Figure 5.1-53: Pu inventory in NPPs (tHM)

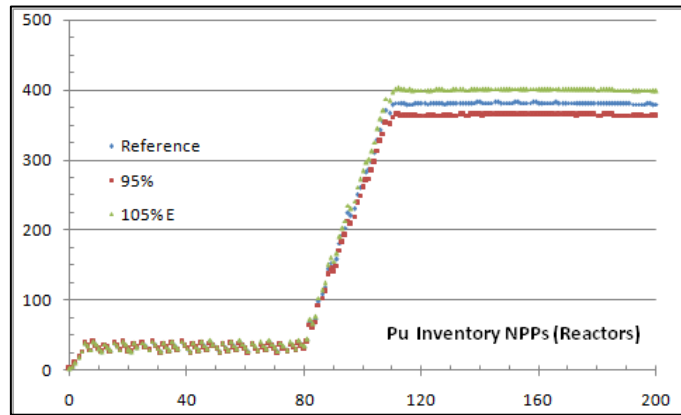


Figure 5.1-54: %- variation Pu inventory in NPPs

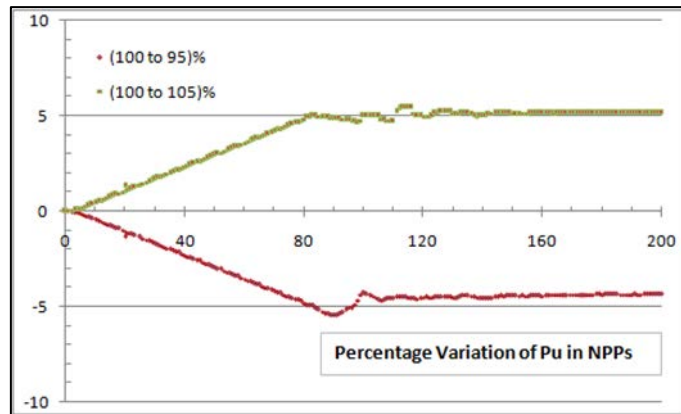


Figure 5.1-55: Pu inventory in storage (tHM)

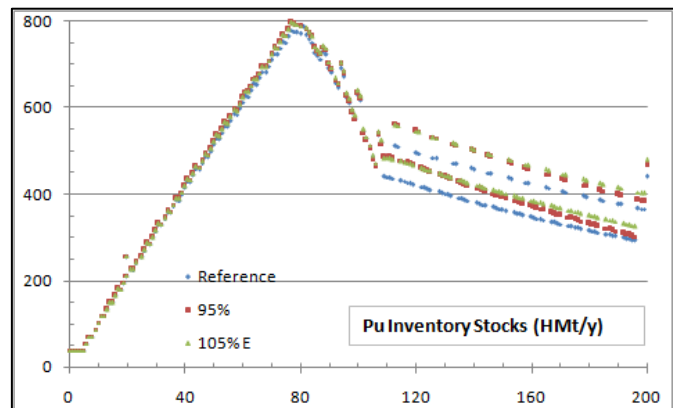
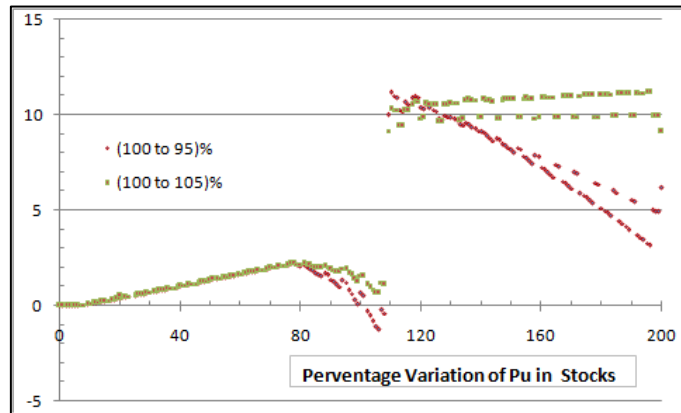


Figure 5.1-56: Percentage variation Pu inventory in storage

The discontinuity observed in the percentage variations (Figure 5.1-56) corresponds to the gradient change in the Pu inventory in stocks between year 106 and year 107. This behaviour, which does not differ from the one in Figure 5.1-15, shows the impact of the assumed law on the energy demand variation.

At the same period, a Pu lack of 39 tonnes has been observed. In Figures 5.1-57 and 5.1-58 the behaviour of Pu inventory in cycle, and relative variations, are displayed.

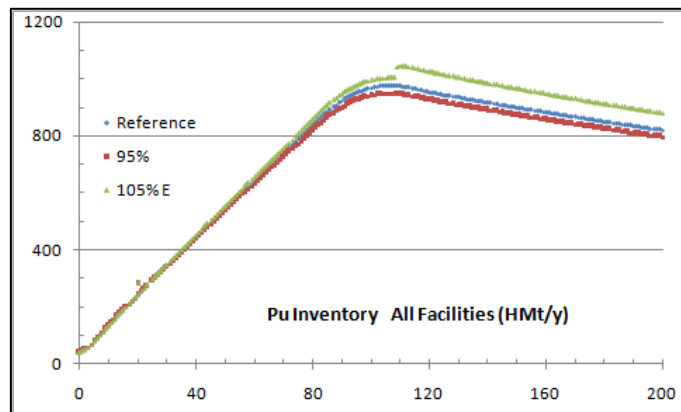
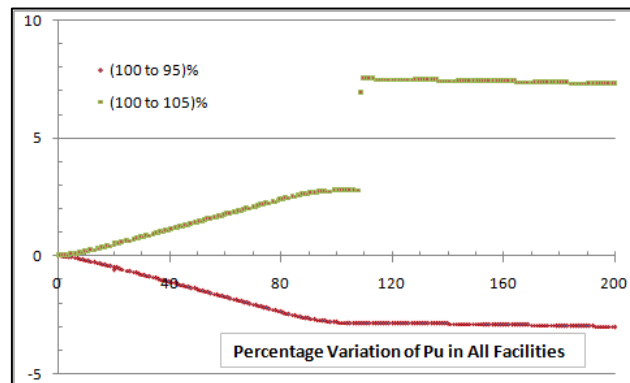
Figure 5.1-57: Pu inventory in cycle (tHM)

Figure 5.1-58: Percentage variation Pu inventory in cycle

The percentage variations of Pu, with respect to the ones in Figures 5.1-14 and 5.1-16 corresponding to the scenarios with a fixed increase in energy demand are very similar and they follow the assumed law of energy demand variation, exceeding the maximum range of energy variation since the Pu lack.

5.1.4. Exponential increase/decrease in energy demand

Different scenarios have been analysed, from a reduction of 63.3% to an increase of 171% of the original energy demand, with an extension up to +1864.3% by modifying the reprocessing plants capacity. In detail, based on an exponential variation law:

$$E(t_N) = E_0(1 \pm \Lambda)^N; \text{ with: } E(t_0) = E_0 = E_{200}; \Lambda \equiv \text{Constant, } [N] = \text{years.}$$

The following values for the parameter Λ have been considered, from the different organisations:

- CEA: $\Lambda \equiv +0.347\%/\text{y}$, or $E(t_{200}) = 2.0 E_0$, energy increase of 100%;
- CIEMAT: $\Lambda \equiv +0.2\%/\text{y}$, or $E(t_{200}) = 1.491 E_0$, energy increase of 49.1%;
 - $\Lambda \equiv +0.5\%/\text{y}$, or $E(t_{200}) = 2.712 E_0$, energy increase of 171.2%;
 - $\Lambda \equiv +0.5\%/\text{y}$, or $E(t_{200}) = 2.712 E_0$, energy increase of 171.2%,
at 2 times reprocessing plants capacity;
 - $\Lambda \equiv +1.5\%/\text{y}$, or $E(t_{200}) = 19.64 E_0$, energy increase of 1864%,
at 2 times reprocessing plants capacity;
- CIEMAT: $\Lambda \equiv +1.5\%/\text{y}$, or $E(t_{200}) = 19.64 E_0$, energy increase of 1864%,
at 10 times reprocessing plants capacity;
- CNL: $\Lambda \equiv +0.02\%/\text{y}$, or $E(t_{200}) = 1.041 E_0$, energy increase of 4.081%;
 - $\Lambda \equiv +0.04\%$, or $E(t_{200}) = 1.083 E_0$, energy increase of 8.327%;
- KIT: $\Lambda \equiv -0.2\%/\text{y}$, or $E(t_{200}) = 0.670 E_0$, energy decrease of 33.0%;
 - $\Lambda \equiv -0.5\%/\text{y}$, or $E(t_{200}) = 0.367 E_0$, energy decrease of 63.3%.

Due to the multiplicity of the investigated scenarios, a direct comparison of the results was carried out, to avoid introducing misunderstandings on the interpretation of the results. Such a choice relies also on some indications coming from the obtained results regarding uranium needs (Figure 5.1-59) and the Pu inventory in cycle (Figure 5.1-60) for the reference cases. Figures 5.1-61 and 5.1-62 show the percentage variations compare to the reference scenario provided by CEA.

Figure 5.1-59: Natural U needs – Reference cases (tHM/y)

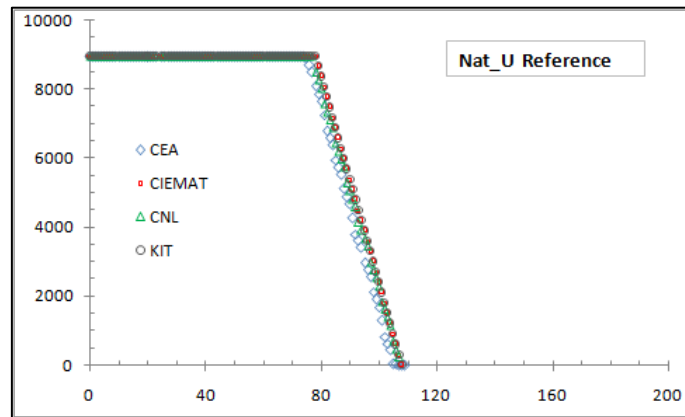


Figure.5.1-60: Total Pu in cycle – Reference cases (tHM)

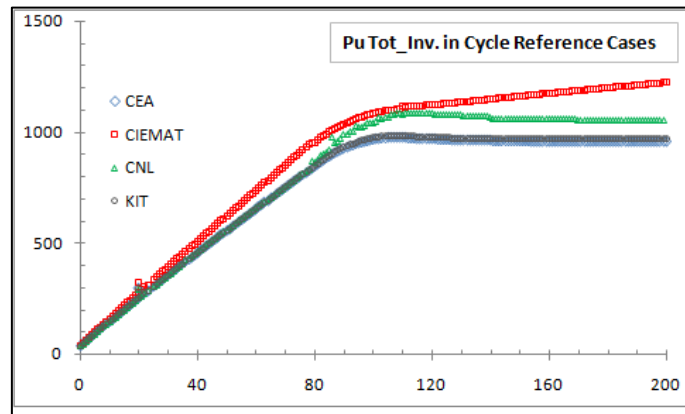


Figure 5.1-61: Percentage variation of natural U needs

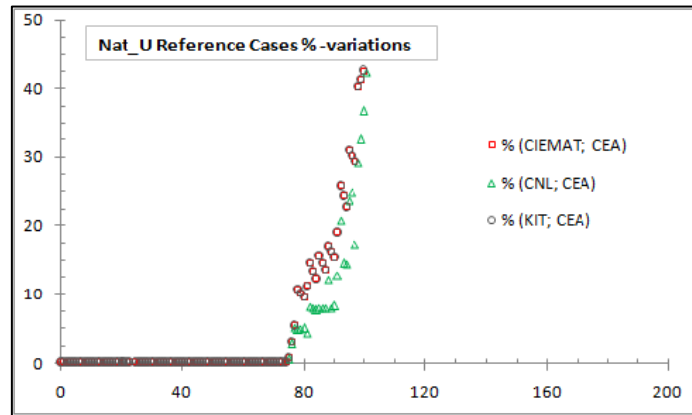
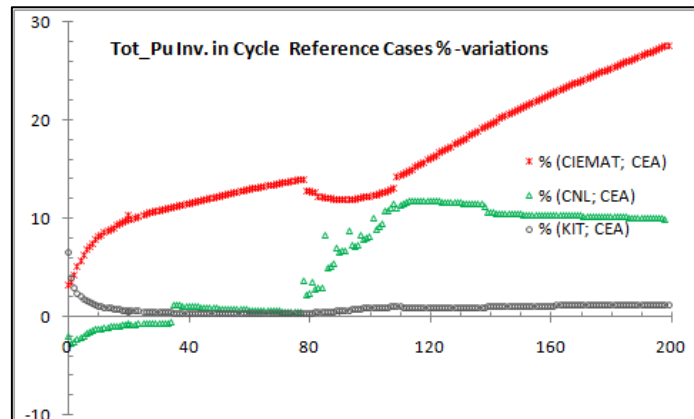


Figure 5.1-62: Percentage variation of total Pu in cycle



The uranium needs follow the shape of the PWRs policy within the energy demand scenario. Significant percentage variations (higher than +45%) in the transition phase have been observed, with almost coincident results between CIEMAT and KIT ones.

For the Pu inventory in cycle, which follows the shape of the PWRs and FRs policies within the energy demand scenario, significant percentage variations (up to +30%) for almost the whole scenario period have been found. Similar results between the CEA and the KIT were observed, while discrepancies were observed between CIEMAT and CNL results, accentuated after beginning of the FRs deployment.

Concerning the PWRs scenario parameters, a representative answer may be given by Figures 5.1-63 and 5.1-64 which show, respectively, the enriched uranium and the UOX fabrications needs. Both scenario parameters behave following the shape of the adopted energy variation.

Figure 5.1-63: Enriched U needs (tHM/y)

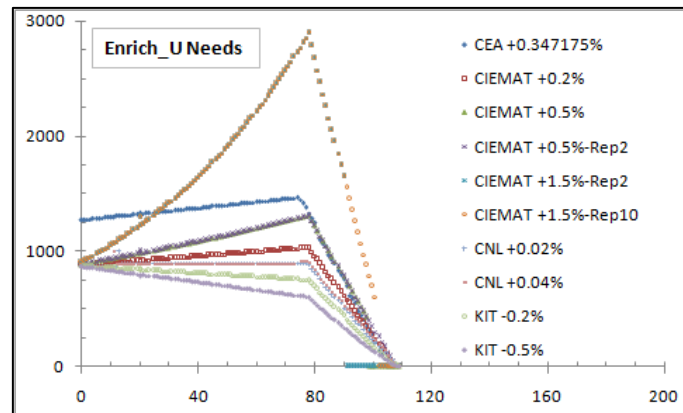
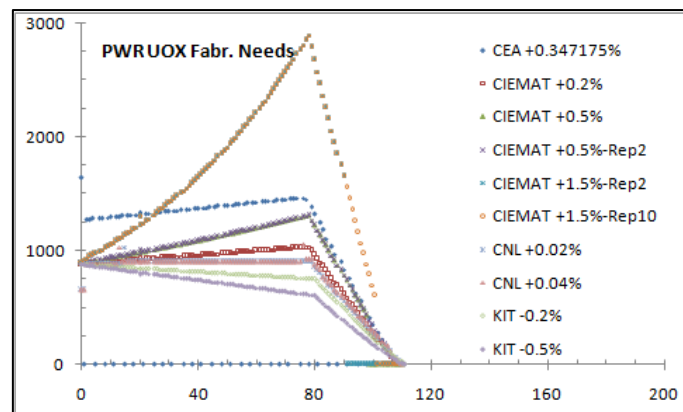


Figure 5.1-64: PWRs fuel fabrication needs (tHM/y)

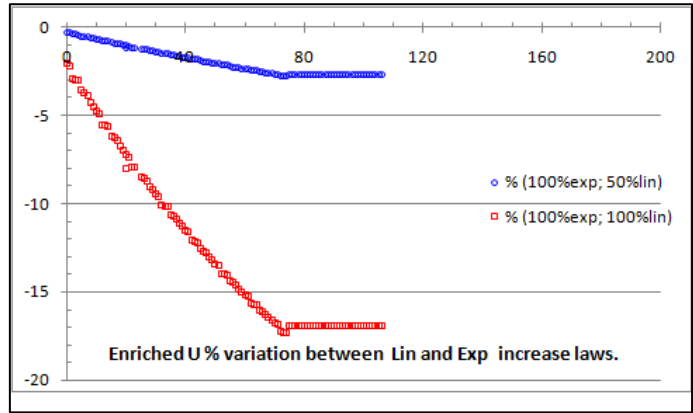


Natural uranium consumption and SWU follow the same profile, reaching respectively the peak values (scenario: CIEMAT, $\Lambda = 1.5\%/y$) of 29510 tonnes and $2.2634E7$ SWU at year 80.

It should be noted that when the reprocessing capacity is not constraining, results are independent of the reprocessing capacity.

It should be highlighted that, for the scenario parameters related to the needs of LWRs, comparison of the CEA results show that the exponential increase of the energy of 100% requires performances similar to those of a linear increase of the energy of 50%, with percentage variations less than 2.5%. On the contrary, these performances are quite far from those of a linear increase of the energy of 100%, with percentage variation up to about 17%. Figure 5.1-65 shows the percentage variation of the enriched uranium needs for the exponential increase option with respect to the ones of the linear increase. Integrating in the time the energy variation laws, the whole energy production could be obtained by exponential increase and is 3.83% lower than the whole energy production by 100% of linear increase, and 15.42% higher than the whole energy production by 50% of linear increase.

Figure 5.1-65: Percentage variation of enriched U needs



Since the lower uranium needs, in the case of energy exponential, increase with respect to the linear one at the same boundary conditions, the exponential energy increase should be advantaged with respect to the linear one.

Figure 5.1-66 shows the results obtained for the FRs fuel fabrication needs, which naturally follow the deployment of the FRs themselves. The maximum value of 1504 tHM/year was observed in year 101, for the CIEMAT scenario ($\Lambda = 1.5\%/y$ - Rep10).

Similar results were obtained for the fertile fabrication needs, which follow the same profile of the fissile fuel, on a different scale. For the CIEMAT scenario, as before, a peak value of 907 tHM/year at year 101 has been observed, while 179.7 tHM/year is the corresponding value for the CEA scenario at $\Lambda = 0.3472\%/y$.

Figure 5.1-66: Enriched U needs (tHM/y)

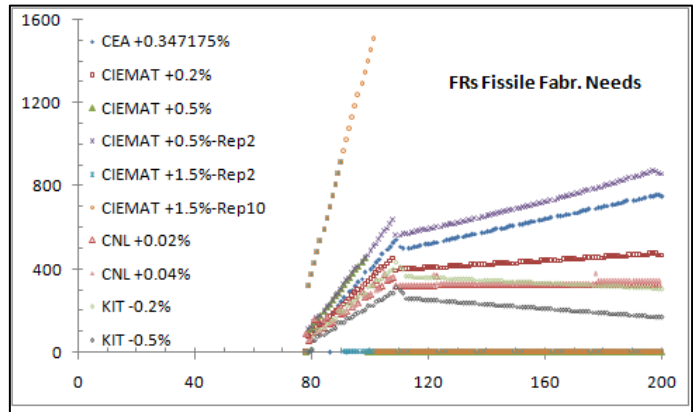
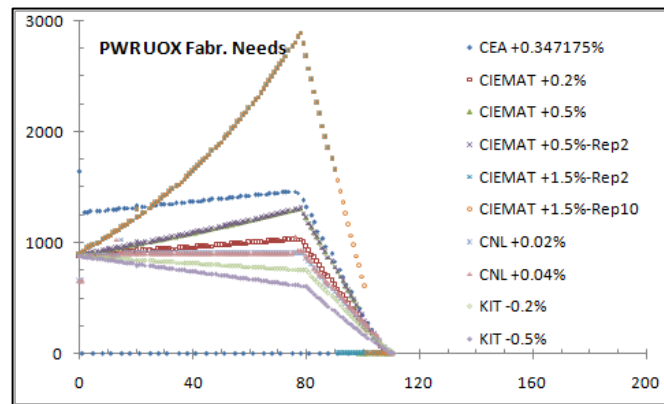


Figure 5.1-67: LWR fuel fabrication needs (tHM/y)



Even in the case of FRs, both fissile and fertile fuels show the same behaviour and trend, with respect to the independence from the reprocessing capacity (in the CIEMAT scenarios). All the results suggest that the time behaviour of those parameters is mainly conditioned by the energy increase.

Equivalent time behaviour has also been observed for Pu for fuel fabrication needs. A peak value at 287 tHM/year was observed for the scenario run by CIEMAT ($\Lambda = 1.5\%/y$ - Rep10) in year 101, compared to the 85 tHM/year of the corresponding value of the CEA scenario with $\Lambda = 0.3472\%/y$ (see Figure 5.1-67).

A different situation occurs in the cases of spent fuel inventory and spent fuel reprocessing. In the case of LWRs spent fuel inventory, except for the scenarios with no nominal reprocessing capacity, all the other scenarios behave according to their own energy demand variation. Nonetheless, the results of the CIEMAT scenario of $\Lambda = 0.2\%/y$ (for an energy increase of 49.1%) almost coincide with the CEA scenario of $\Lambda = 0.3472\%/y$ (for an energy increase of 100 %) (see Figure 5.1-68).

For the FRs spent fuel inventory, apart from the CEA results which follow an exponential law, reaching the value of 23000 tHM at the end of the scenario, all results do not exceed 4000 tHM (maximum value of 3850 tHM for the KIT results) following the FRs deployment shape (see Figure 5.1-69).

Figure 5.1-68: PWR spent fuel in storage (tHM)

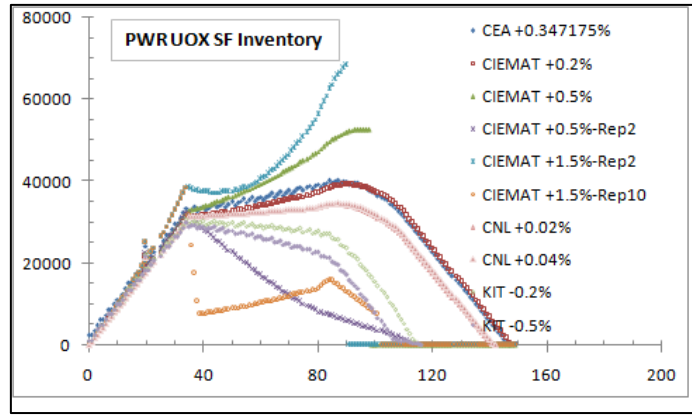
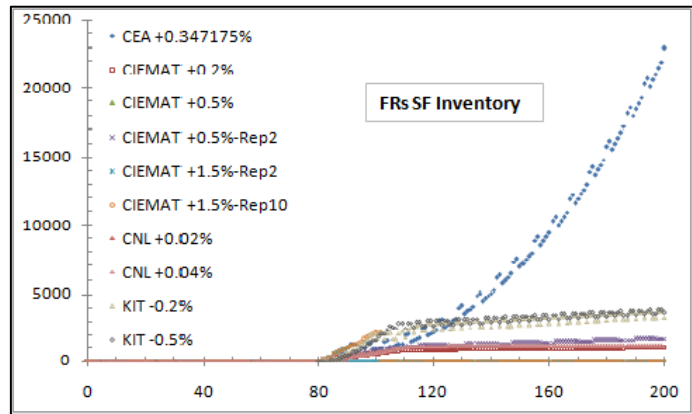


Figure 5.1-69: FR spent fuel in storage (tHM)



It should be noted that KIT calculation results, dealing with energy decrease scenarios, were found to be higher than all the remaining scenarios as a consequence of a “phase out” scenario.

Regarding LWRs spent fuel reprocessing results, the scenarios of CEA, CIEMAT at nominal reprocessing capacity, CNL and partially KIT, were developed keeping the nominal value of 850 tHM/year. The KIT scenarios at the beginning of the FRs deployment, in year 80, needed of an increase of the reprocessing capacity, reaching the value of 1300 tHM/year in some ten years. Once the FRs deployment is completed, a reduction of the reprocessing capacity occurs, starting from the scenario at $\Lambda = -0.5\%/y$ (energy decrease of 63.3%) (see Figure 5.1-70).

Figure 5.1-70: PWR SF reprocessed (tHM/y)

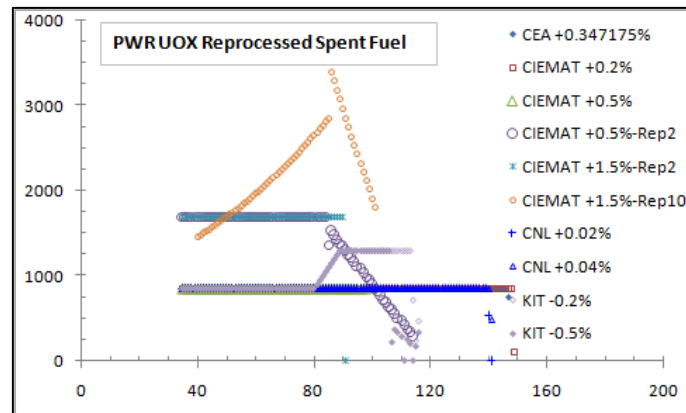
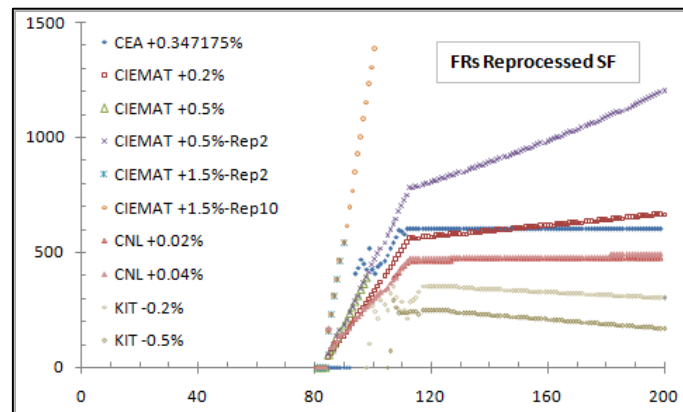
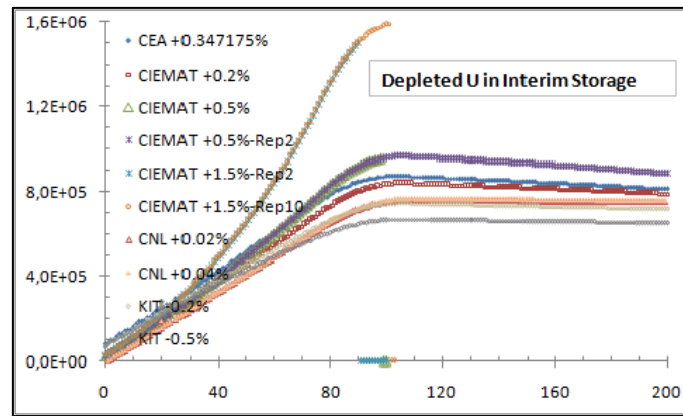
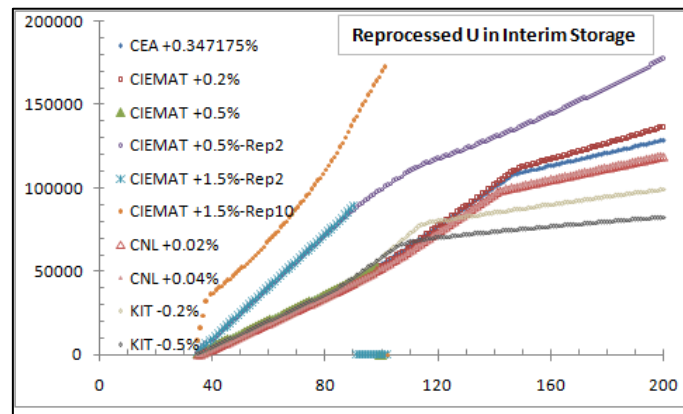


Figure 5.1-71: FR SF reprocessed (tHM/y)



In CIEMAT scenarios a rapid decrease of the reprocessing capacities occurs starting in year 85 at double and/or decoupled reprocessing capacity. The FRs reprocessed spent fuel follows the profile of the own energy demand variation, as shown in Figure.5.1-71. While CEA and KIT results show a relative minimum at the end of the transition phase, CIEMAT and CNL results display a monotonous increase. Moreover the CIEMAT scenario at $\Lambda = 0.2\%/y$, corresponding to an energy increase of 49.1%, starting at year 150 requires reprocessing more fuel than the CEA scenario at $\Lambda = 0.3472\%/y$, corresponding to an energy increase of 100%. Analogous behaviour may be observed for the interim storage depleted Uranium, as well as for the reprocessed one. At the end of the scenario and for the scenarios at nominal reprocessing capacity, depleted uranium values between about 650 to 870 thousands of tHM have been found, while for the reprocessed uranium the values between about 82 to 136 thousands of tHM have been detected. In the case of not nominal reprocessing capacities, variation respectively of 66% and 30% has been found as depicted in Figures 5.1-72 and 5.1-73.

Figure 5.1-72: Depleted U in interim storage (tHM)**Figure 5.1-73: Reprocessed U in interim storage (tHM)**

Concerning the separated Pu inventory in interim storage (starting in year 35), for the scenarios at nominal reprocessing capacity, the results around year 80 are spread out between 390 to 470 tHM/year with a monotonous increase. After year 80, a decreasing behaviour is observed with two distinct trends: CEA and KIT in the first one, showing a progressive reduction up to exhaustion close to year 100; CIEMAT and CNL in the second one, reaching a relative minimum around the year 110 with a consequent increase and reaching of a stabilisation, between year 140-150, at the values respectively of about 480 and 320 tHM/year. For the scenarios at no-nominal reprocessing capacity, two of them (at $\Lambda = +1.5\%/y$; double and decoupled reprocessing capacity) they follow the first mentioned trend, while the last one (at $\Lambda = +0.5\%/y$, double reprocessing capacity) follows the second mentioned trend. Such results also depend on the boundary conditions assumed in the development of the scenarios.

Once again, concerning the CIEMAT scenarios, the behaviour highlighted in the uranium needs is repeated. Once the energy increase has been fixed, the same results are obtained regardless of the reprocessing capacity.

As previously highlighted for the CEA scenarios, even in the case of spent fuel, the behaviour of the LWRs spent fuel in storage, originated by exponential energy demand variation of 100%, follows the behaviour originated by a linear energy demand variation of 50%, while such a behaviour is not observed for the FRs spent fuel inventory. Such behaviour cannot be observed in the reprocessed spent fuel of the PWRs since a fixed reprocessing capacity was imposed. However, it is still valid for depleted and reprocessed uranium inventories, as well as for the separated Pu inventory in interim storage. In this last case, all the scenarios follow the same profile and the same values.

For scenarios involving FRs deployment, the Pu inventory in the fuel cycle components is a very important fact. The general trend of the Pu inventory in the fuel cycle components follows both FRs deployment strategy and energy demand variation. Figures 5.1-74 and 5.1-76 show the behaviour of Pu inventories respectively in Plants (fabrication and reprocessing) and Reactors for all the considered scenarios, confirming the mentioned general trend.

Figure 5.1-74: Pu inventory in NPPs (tHM)

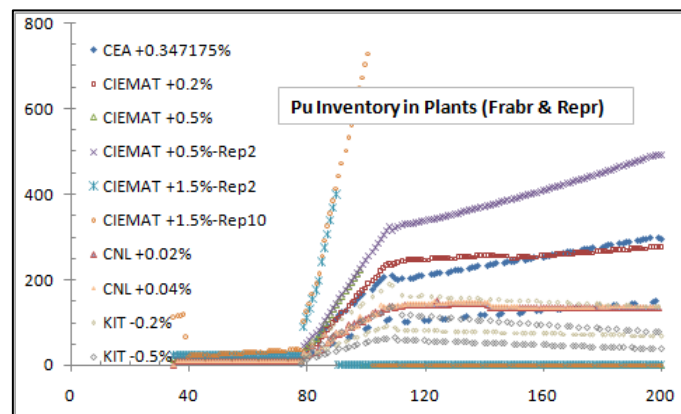
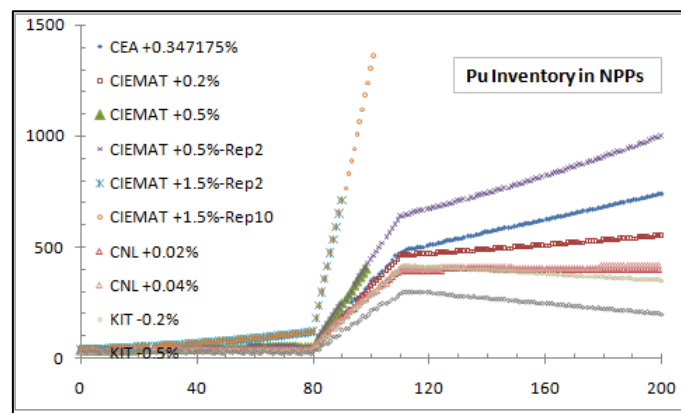


Figure 5.1-75: Pu inventory in reactors (tHM)



Regarding CNL scenarios, despite the slight increase of the energy demand (energy increase of 4.1% and of 8.3%), the behaviour of the Pu inventory in the reactors deviates slightly from the energy demand profile, remaining almost constant, respectively at about 400 and 410 tHM/y, during the whole equilibrium regime of the scenario.

Also for these scenario parameters, CIEMAT scenarios follow the general trend of dependence of the results from the energy demand variation, regardless of the reprocessing plants capacity.

Of course, the behaviour and performance of the FRs spent fuel inventory of the CEA scenario, displayed in Figure.5.1-69, dominates the Pu inventory in storage, more than the scenarios of not-nominal reprocessing capacities. Figure 5.1-76 collects the Pu inventory in storage for the entire analysed scenarios.

Figure 5.1-76: Pu inventory in storage (Stocks) (tHM)

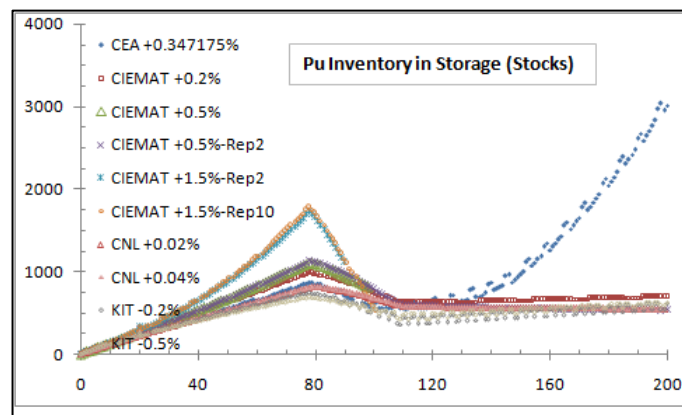
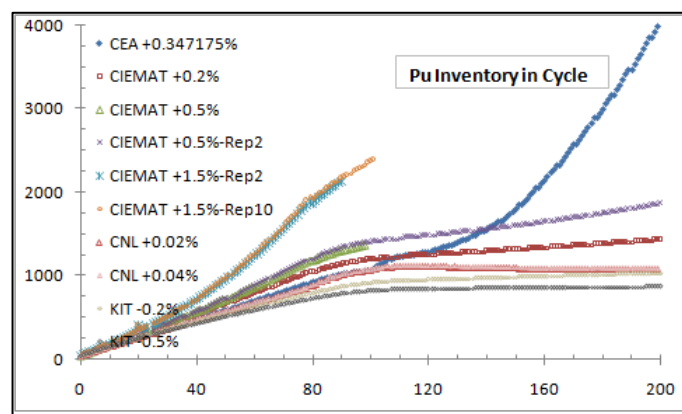


Figure 5.1-77: Pu inventory in the cycle (tHM)



Again, the scenarios of nominal reprocessing capacity, behave following the FRs deployment policy and the own energy demand variation, except for that of the CEA which behaves following an exponential law, reaching the value of about 3000 tHM at the end of the scenario.

The total Pu inventory (i.e. the Pu in all the fuel cycle components) will be dominated by the Pu inventory in storage in the CEA scenario (see Figure 5.1-77) while the behaviour of the Pu inventory in cycle in all the other scenarios follows the reactors deployment and the own energy demand variation.

The Pu inventory, in the fuel cycle components, in the scenarios with no-nominal reprocessing capacity, behaves following the usual trend already highlighted.

This rule-behaviour seems to not be respected in the case of Pu waste inventory. The 0.1% of losses in the spent fuel reprocessing produces some tonnes of Pu waste as shown in Figure 5.1-78. The Pu waste production, for the scenarios of no-nominal reprocessing capacity, seems to depend on the energy demand variation instead on the reprocessing capacity.

Figure 5.1-78: Pu inventory in storage (Stocks) (tHM)

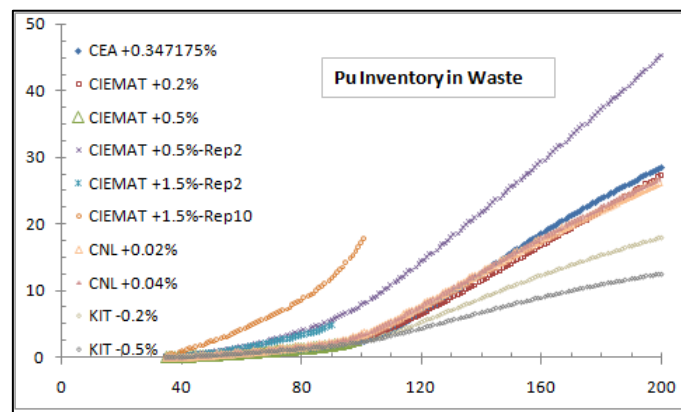
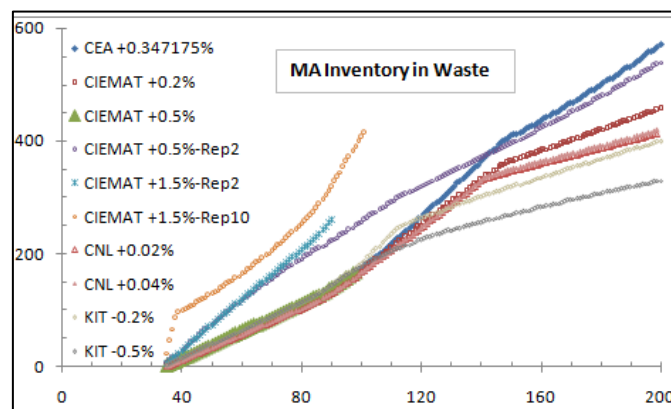


Figure 5.1-79: Pu inventory in the cycle (tHM)



Moreover, the scenarios of nominal reprocessing capacity of CEA, of CIEMAT and of CNL (all these at increasing energy demand variation) follow the same profile and almost the same value. However, in the KIT scenarios, depending on their energy demand variation, a reduction of about 30% or 55%, with respect to all the other scenarios, may be observed. These results are a consequence of a partial “phase out” scenario. Due to no MAs recycling, the 100% of losses together with the spent fuel decays, produce a significant amount of MAs in waste (see Figure 5.1-79). The indicated results follow both the energy demand variations and the reprocessing capacities.

Finally, concerning the MAs inventory in waste, the results of the CEA's scenario studies show that the ones originated by an exponential law of the energy increase, of 100% of the original energy demand, exceed almost completely the ones originated by a linear law of the energy increase, of 50% of the original energy demand. Such behaviour is not observed in the Pu inventory in all the components of fuel cycle.

5.2. Cooling time

In this section the effect of changing the minimum cooling time of spent fuel before reprocessing is examined. The results presented are based on COSI (CEA) calculations.

5.2.1. Spent PWR UOX fuel

Cooling time of the spent PWR UOX fuel was changed from the reference case's 5 years to 2 and 8 years. According to the scenario, the first irradiated fuel batches from the equilibrium PWR UOX fleet are discharged in the first year of the scenario while their reprocessing starts in year 35. The amount of the annually discharged irradiated PWR UOX fuel is around 880 tonnes while the annual reprocessing capacity is slightly less: 850 tonnes. This means that spent fuel is reprocessed after more than 30 years of cooling time in the reference case. Because the annual reprocessing capacity is lower than the spent fuel discharge, accumulation of spent fuel can also be observed, which can be regarded from the reprocessing's point of view as a reserve.

Changing the cooling time of spent PWR UOX fuel does not change the above described situation, i.e. it has no impact on any metrics of the fuel cycle. This can be explained with starting date of reprocessing and its capacity. Increasing the reprocessing capacity or changing the reprocessing strategy (to reprocess spent fuel as soon as possible or to change the order of reprocessing from FIFO to LIFO) would yield different results.

5.2.2. Spent FR fuel

Cooling time of the spent FR fuel was changed from the reference case's 2 years to 5 and 8 years. The spent FR fuel's cooling time is one of the parameters which influences the recycling time of the FR's spent fuel, i.e. the difference between the discharge of the irradiated fuel and the introduction of separated material (in this case plutonium) from the spent fuel into the FR's core. The FR fuel fabrication time is the other parameter in this benchmark which influences the recycling time.

In this sensitivity study the FR spent fuel reprocessing plant works on-demand, i.e. as plutonium need arises in the FR fuel fabrication plant the reprocessing plant starts to operate: it takes spent fuel from the spent fuel interim storage and reprocesses it to fulfil this

need. The amount of spent FR fuel available for reprocessing is influenced by the FR spent fuel's cooling time: if cooling time is increased then this amount is decreased. It is worthy to note that similar results are obtained if as much of the FR spent fuel is reprocessed as the reprocessing capacity allows. (Results obtained with VISION, CNL).

In the reference case, when the spent FR fuel's cooling time is 2 years, there is no limitation on the reprocessing, i.e. the reprocessing plant can follow the FR's fuel fabrication plutonium needs and can reprocess enough spent FR fuel to provide the required amount of plutonium. However, when the cooling time is increased, there will be plutonium shortage in the fuel cycle. If the cooling time is 5 years, 100 tonnes of external Pu is used between years 102 and 109, while for 8 years 288 tonnes of external Pu is used between years 99 and 115. On the other hand this means that the reprocessing capacity is adapted, i.e. decreased, to the amount of available FR spent fuel and the amount of FR spent fuel in the interim storage is increased compared to the reference case as shown in Figures 5.2-1 and 5.2-2.

Figure 5.2-1: Amount of reprocessed spent FR fuel as function of its cooling time

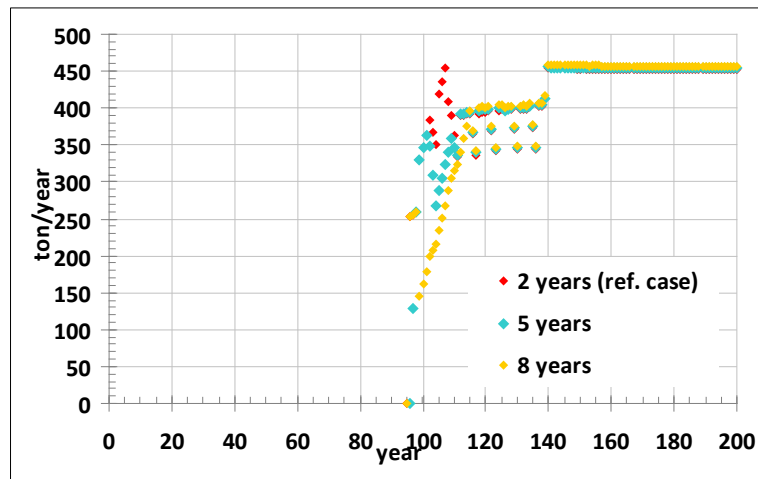
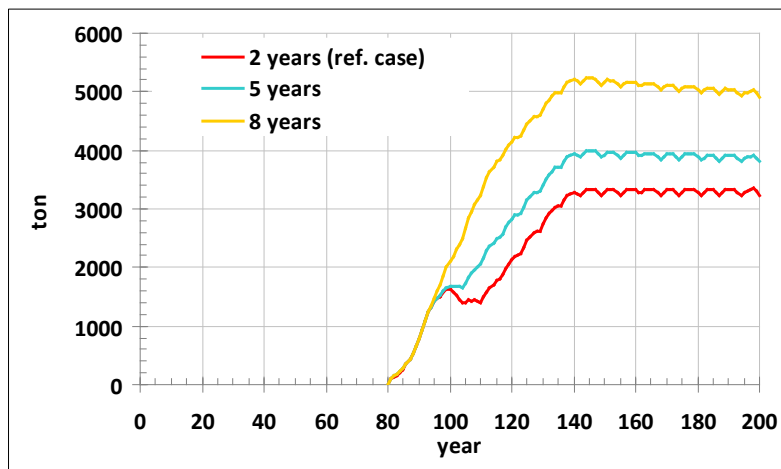


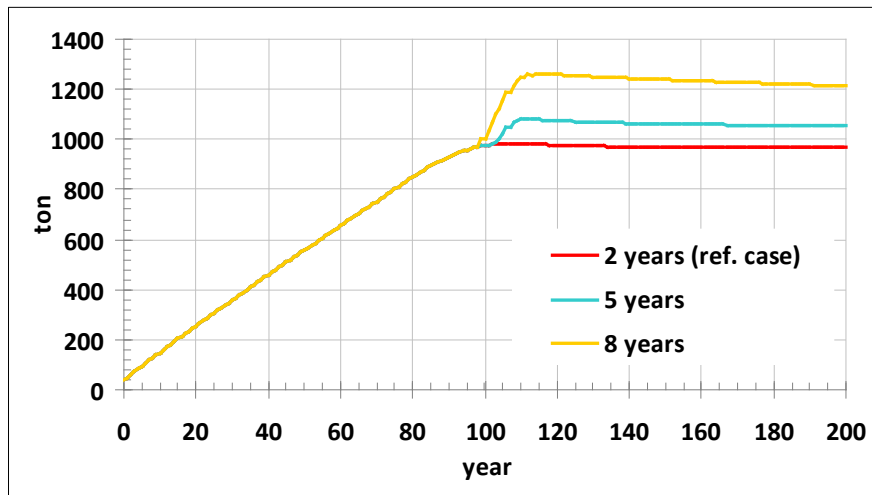
Figure 5.2-2: Amount of spent FR fuel in the interim storage as function of its cooling time



The impact of spent FR fuel's cooling time on other parameters, like plutonium or minor actinides inventory in the waste, is limited.

The plutonium deficit in the fuel cycle can be seen in Figure 5.2-3 after year 100, close to the end of the introduction period of the FR fleet. In these years an external plutonium source is used to cover the FR fuel fabrication's plutonium need.

Figure 5.2-3: Total Pu inventory in cycle as function of spent FR fuel's cooling time



Lack of plutonium after year 100 can be observed for all cases except the reference one.

The lack of plutonium might be counterbalanced with tuning the spent PWR UOX fuel's reprocessing capacity. This helps only if the spent FR fuel's cooling time is 5 years, which means that in this case the fuel cycle has inherently enough plutonium to cover the lack. If the spent FR fuel's cooling time is 8 years, the Pu lack can be mitigated, but cannot be covered by the Pu contained in the fuel cycle: an external Pu source is needed anyhow to let the cycle work.

If the spent FR fuel's cooling time is 5 years then increasing the spent PWR UOX fuel's reprocessing capacity up to 1000 tonnes/year or keeping the capacity on the reference value (850 tonnes/year) but starting the reprocessing earlier, in year 24, supplies enough Pu for the FR fuel fabrication. For the former all the spent UOX fuels will be reprocessed by the year 124, while for the latter 545 tonnes of separated Pu will be accumulated by the year 78.

5.3. Fabrication time

In this chapter the effect of changing the fuel fabrication time of fresh fuel is given. The results presented are based on COSI (CEA) calculations.

5.3.1. PWR UOX fuel

Fabrication time of PWR UOX fuel was changed from the reference case's 2 years to 1 and 3 years. This change has effect only on the front-end of the PWR fleet. Increasing the fabrication time will result in earlier enrichment and natural uranium need. This effect cannot be detected at the equilibrium PWR fleet, but only in the phase-out period where shifting of

the natural uranium consumption and SWU needs curves with ± 1 year can be observed as shown in Figures 5.3-1 and 5.3-2.

Figure 5.3-1: Natural U consumption as a function of PWR UOX fuel's fabrication time

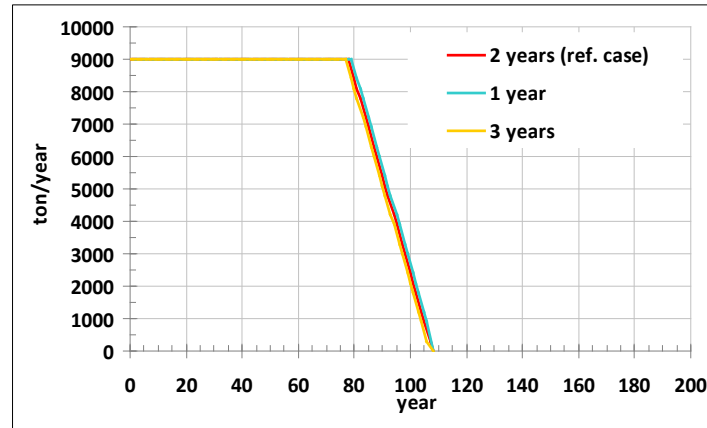
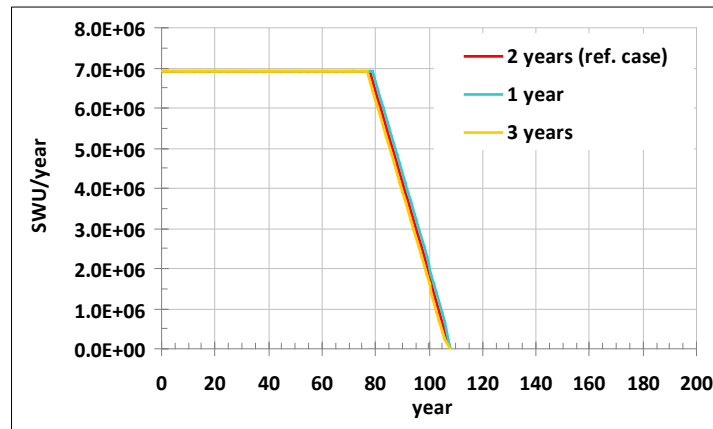


Figure 5.3-2: SWU needs as a function of PWR UOX fuel's fabrication time



It also should be noted that changing the PWR UOX fuel's fabrication time has a slight effect on the depleted uranium inventory. As a result of the change of the fabrication time, the enrichment plant starts to fill the depleted uranium stock in a different time. This stock has one input stream and one output stream. The latter goes to the fabrication plant of the FR fuel. Because the output stream is unchanged, compared to the reference case, the shape of the depleted uranium inventory curve during the operation of the PWR fleet is slightly different. This small difference vanishes after the phase-out of the PWR fleet.

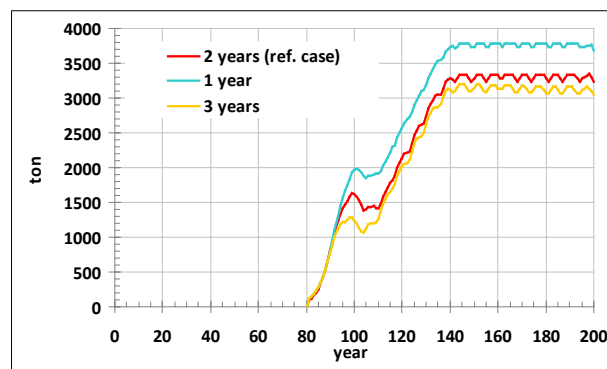
5.3.2. FR fuel

Fabrication time of FR fuel was changed from the reference case's 2 years to 1 and 3 years. This parameter is, among others, the one which influences the recycling time of the FR's spent fuel as mentioned earlier in Section 5.2.2. Changing this parameter affects several results including:

- starting date of reprocessing (if reprocessing works on-demand);
- amount of FR spent fuel in interim storage (if reprocessing works on-demand);
- amount of separated Pu in interim storage (if reprocessing works with fixed capacity);
- Pu flow into the fabrication plant;
- Pu inventory in the fabrication plant.

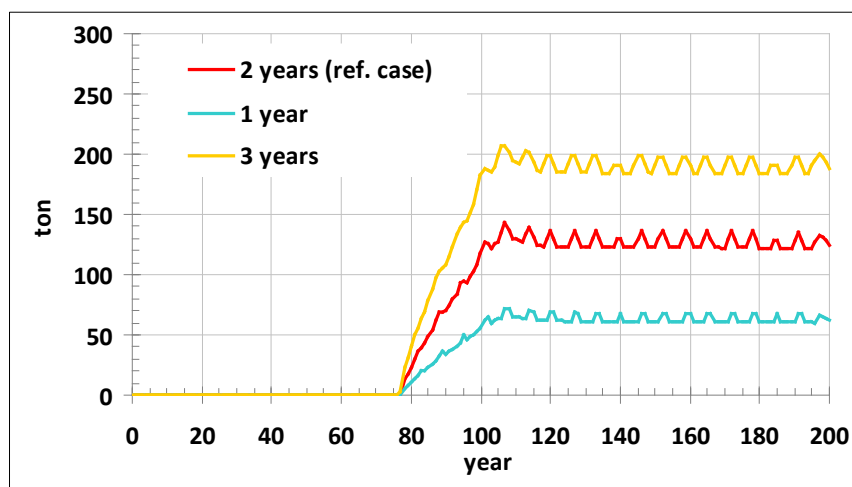
Increasing the FR fuel's fabrication time causes earlier starting of the on-demand working reprocessing plant whose effect has an impact on the spent FR fuel interim storage, as shown in Figure 5.3-3, increasing fabrication time decreases the amount of FR spent fuel in the interim storage. If fabrication time is 1 year, then the average amount of FR spent fuel after year 140 is 3761 tonnes. As fabrication time is increased to 2 years this value decreases to 3302 tonnes (87.8% of the value belonging to 1 year fabrication time). If fabrication time is 3 years the corresponding value is 3133 tonnes (83.3%).

Figure 5.3-3: Amount of spent FR fuel in the interim storage as a function of the fresh fuel's fabrication time



If the reprocessing of the spent FR fuel works with a fixed capacity, i.e. it doesn't work on-demand, then the amount of separated plutonium in the interim storage decreases as the fabrication time increases. Concerning the inventory of plutonium in the fabrication plant, it increases with the fabrication time and is a linear function of it, as shown in Figure 5.3-4. If fabrication time is 1 year the average amount of Pu in plants after year 120 is 63 tonnes. As fabrication time is doubled or tripled this amount is also doubled and tripled to 126 tonnes and 190 tonnes, respectively.

Figure 5.3-4: Pu inventory in the FR fuel fabrication plant as a function of the fabrication time



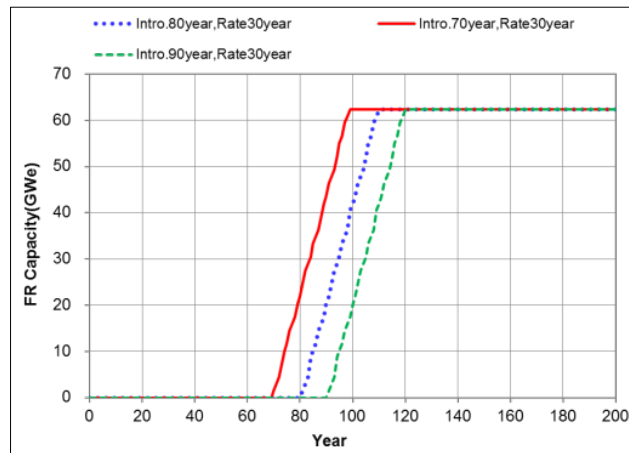
If the FR fabrication time is 3 years there will be a shortage of plutonium between years 105 and 108. This lack of 48 tonnes was covered by an external Pu. This shortage can also be counterbalanced by starting the spent PWR UOX fuel's reprocessing earlier (in year 29 instead of year 35 as in the reference case). In this case 479 tonnes of separated plutonium will be accumulated until year 79.

5.4. Introduction date of the fast reactor

This section describes the effects of changing the start year of FR introduction. A calculation with FAMILY-21 code revealed that when the starting time of FR introduction was changed from year 80 (standard case) to year 70, the shortage of 46.7 tonnes of Pu (fissionable Pu: 33.7 tonnes) in total occurred during the period from year 93 to 98 under the FR introduction conditions shown in Figure 5.4-1. In addition, it also showed that this Pu shortage was avoided by the change of the FR introduction rate from the standard 30 years to 40 years.

In this section, we will report on the effects on the PWR fuel cycle and the FR cycle by the change in the FR introduction start year from year 80 to years 70 and 90 with the standard FR introduction rate of 30 years.

Figure 5.4-1: FR introduction conditions for FAMILY-21 code



5.4.1. PWR cycle

5.4.1.1. Natural uranium consumption and SWU needs

Figures 5.4-2 and 5.4-3 show the calculation results of natural uranium consumption and enriched uranium separation work units respectively. The time differences shown in the two process amounts in the front-end are equal to that of the FR introduction start conditions, minus 10 years for the 70 year case and plus 10 years for the 90 year case from the standard case.

Figure 5.4-2: Natural U consumption

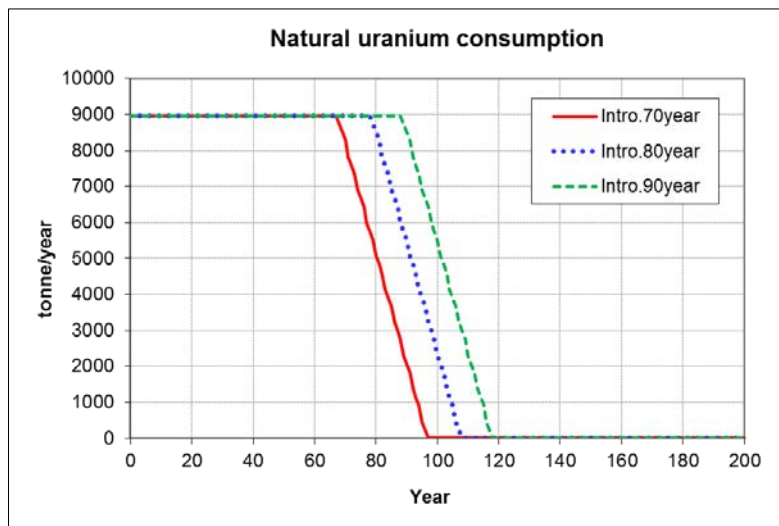
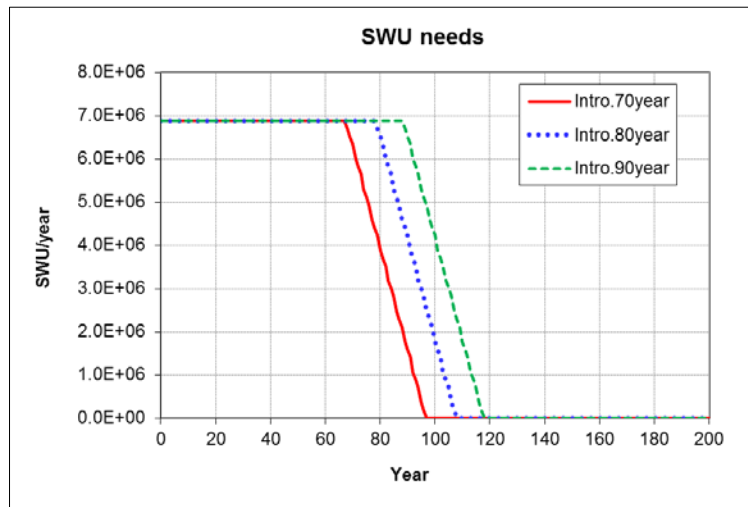


Figure 5.4-3: Enriched U separation work units



5.4.1.2. Enriched uranium demands and PWR-UOX fuel fabrication

Figures 5.4-4 and 5.4-5 show enriched uranium demands and PWR UOX fabrication demands respectively. Amounts of enriched uranium demands and PWR UOX fabrication amounts of both cases, the 70 and 90 years, gradually decrease with the same slope as those of the standard case.

Figure 5.4-4: Enriched U demands

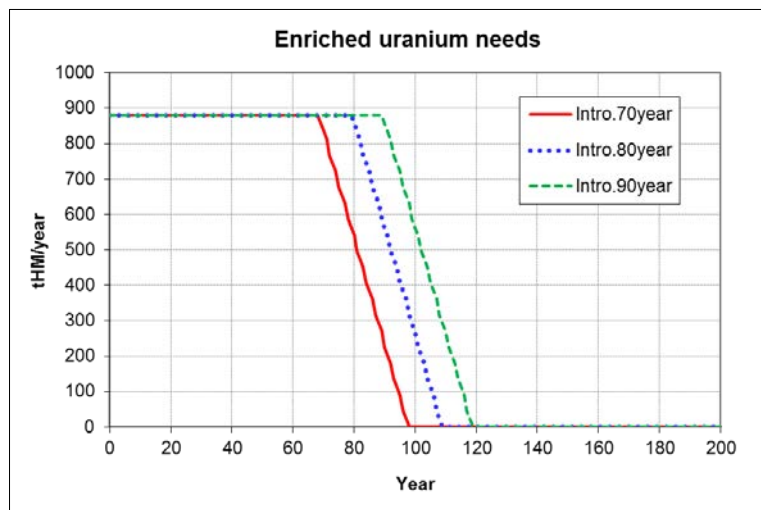
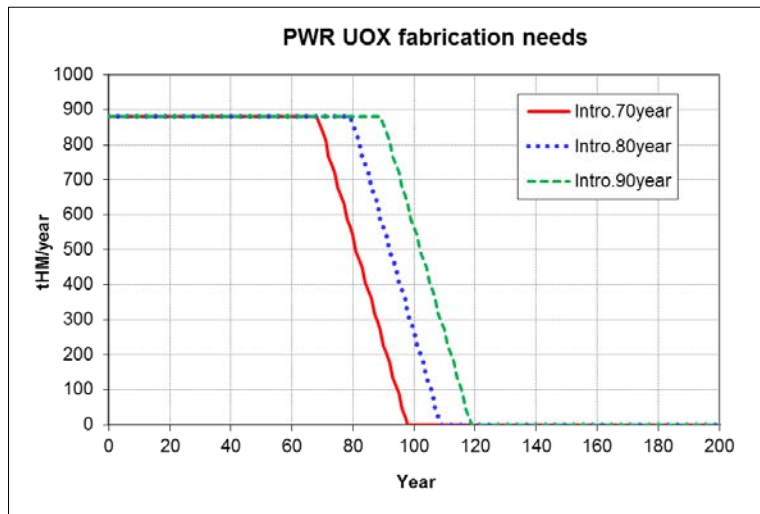


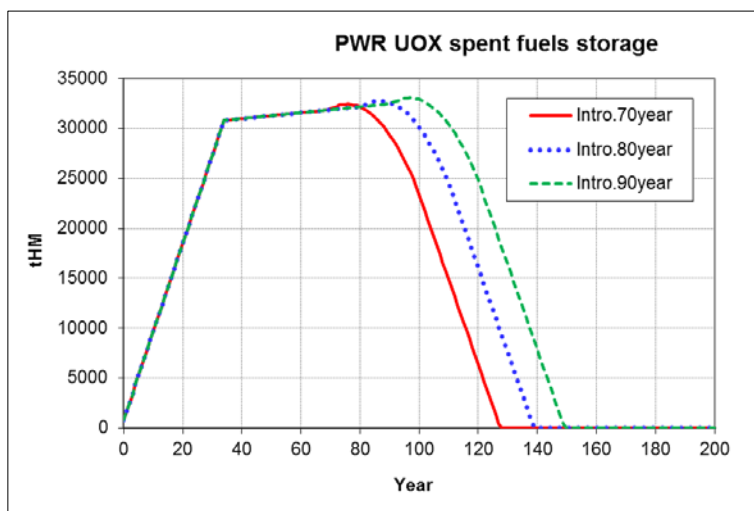
Figure 5.4-5: PWR-UOX fuel fabrication



5.4.1.3. Spent PWR-UOX fuel storage

Figure 5.4-6 shows spent PWR UOX fuel storage amounts. Reduction curves of both the 70 and 90 year cases are similar to that of the standard case. The maximum storage amounts of spent PWR UOX fuel with the 70 year case is minus 330 tonnes and the 90 year case plus 300 tonnes from that of the standard case. It indicates that the storage amount of spent PWR UOX fuel becomes higher as the timing of the FR introduction becomes later under the same condition for PWR UOX reprocessing.

Figure 5.4-6: Spent PWR-UOX fuel storage



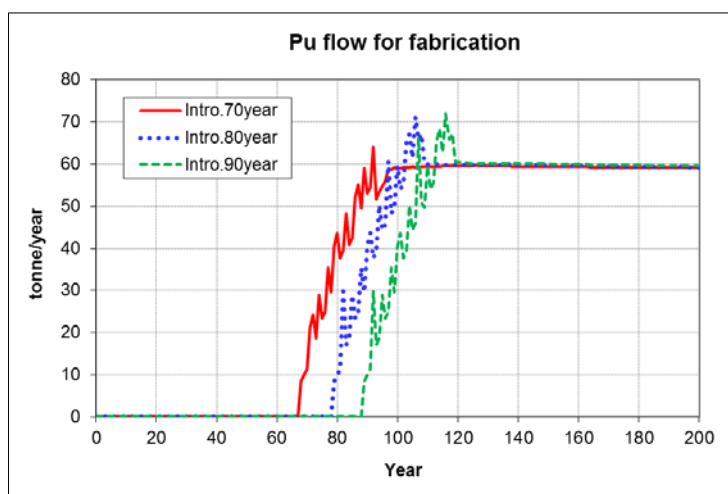
5.4.2. FR cycle

5.4.2.1. Plutonium supply to FR fuel fabrication

Figure 5.4-7 shows the amounts of plutonium (Pu) supply needed for FR fuel fabrication. In the 70 year case, the peak of Pu supply to FR fuel fabrication is lower than those of the two other cases. In addition, the trend of changes until the Pu supply reaches a substantially constant amount (about 60 tonnes/year) in the 70 year case is clearly different from those of the other two cases. These differences are caused by a Pu shortage. More specifically, the relative reduction of recovered Pu from PWR UOX fuel is due to the 10-year-earlier FR introduction.

FAMILY 21 automatically adjusts the amounts of Pu procurement from outside, FR introduction rate (the FR introduction amount balanced with Pu supply) and reprocessing amount as a Pu shortage prevention function, and in this analysis for uncertainty of parameters the Pu shortage amount was compensated with Pu from outside.

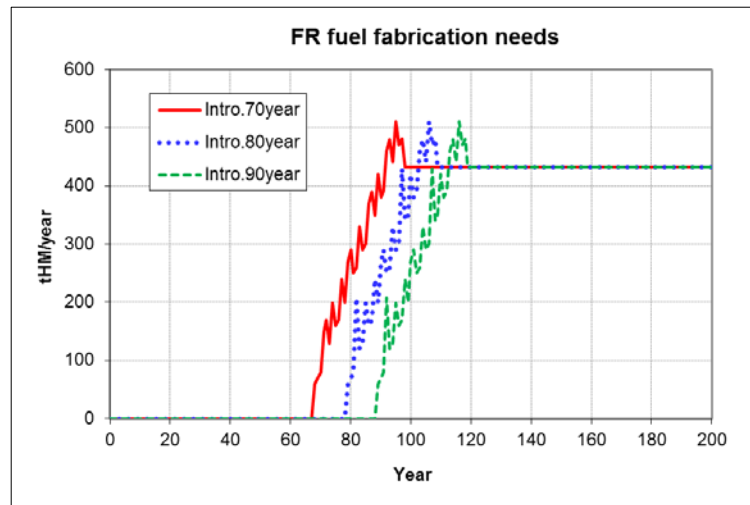
Figure 5.4-7: Pu supply to FR fuel fabrication



5.4.2.2. FR fuel fabrication

Figure 5.4-8 shows FR fuel fabrication needs. The variation patterns of FR fuel fabrication amount (sum of fissile fuel and fertile fuel) in the three cases are almost the same.

Figure 5.4-8: FR fuel fabrication

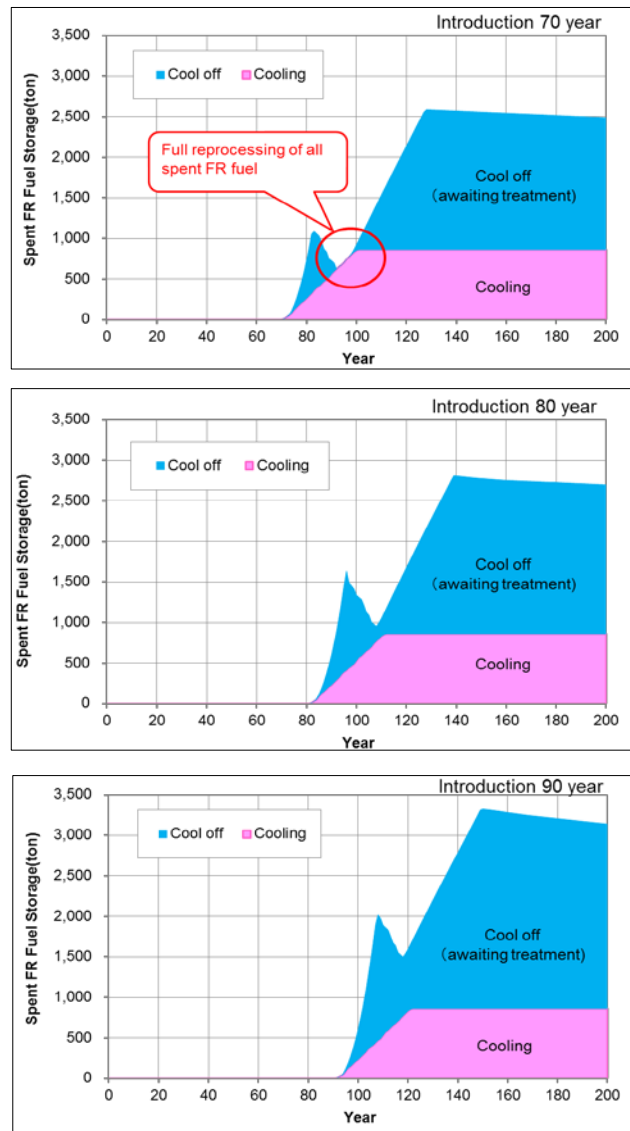


5.4.2.3. Spent FR fuel storage

In the calculation performed with FAMILY 21, the difference between the Pu supply amounts from intermediate Pu storage and the Pu needs for FR fuel fabrication was complemented by the recovered Pu from FR reprocessing facilities. Thus, the annual reprocessing amounts of spent FR fuel was curbed to the least required to meet the Pu needs for FR fuel fabrication.

Details of spent FR fuel storage are shown in Figure 5.4-9. In the 70 year case of FR introduction, all the spent FR fuels waiting for reprocessing were reprocessed during the period from year 82 to 97. As the timing of FR introduction became early for 10 years, Pu from intermediate Pu storage was started to be consumed earlier and the inventory decreased to almost zero in and after year 82.

Figure 5.4-9: Details of spent FR fuel storage



5.4.3. Spent fuel reprocessing

As shown in Figure 5.4-10, the annual reprocessing amounts of PWR UOX fuel for the three cases are the same with 850 tonnes/year. The completion time of spent PWR UOX fuel reprocessing of the standard case (year 80) was year 140, while that of the 70 year case was year 129 and that of the 90 year case year 151, which are shifted plus and minus 11 years from that of the standard case. Thus, it is indicated that the amount of recovered Pu equivalent to about 9 000 tonnes of spent PWR UOX fuel decreases in the 70 year case compared to that of the standard case.

Figure 5.4-10: Annual throughput of reprocessed spent PWR-UOX fuel

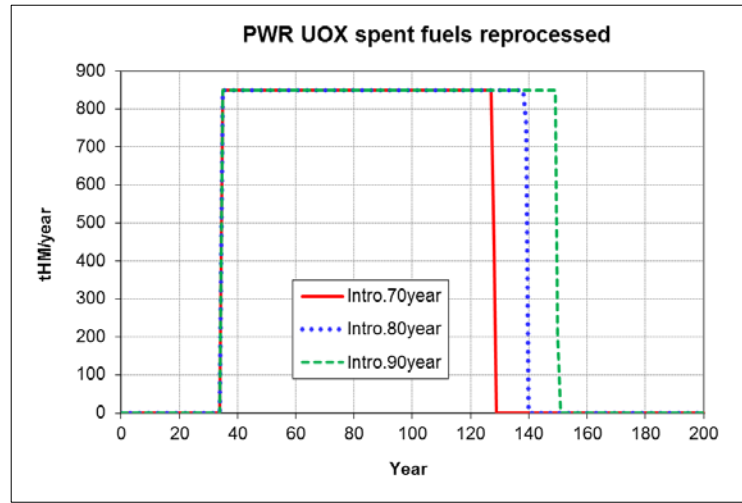
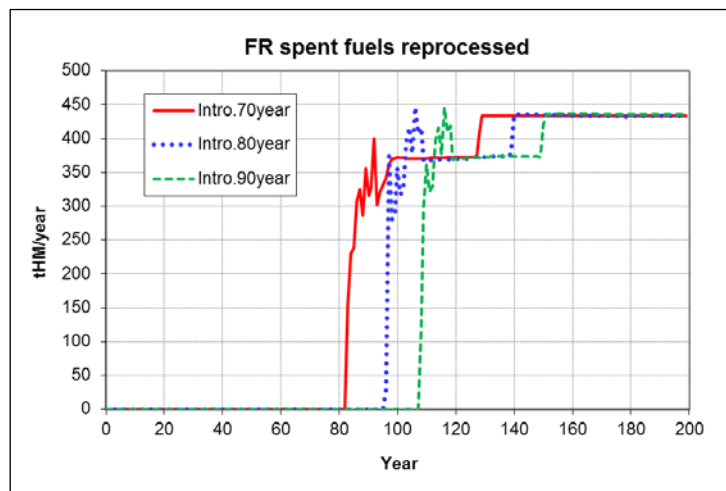


Figure 5.4-11 shows the annual throughput of spent FR fuel. The term from the FR introduction to the start of reprocessing became longer as the FR introduction timing is delayed. The reprocessing start time of the 70 year case was year 83 (13 years later), that of the standard case year 96 (16 years later) and that of the 90 year case year 108 (18 years later). As the FR introduction is delayed, the amount of recovered Pu from spent PWR UOX fuel increased, putting off the start of FR reprocessing plants. Furthermore, in the 70 year case the peak of reprocessing amount after about 10 years was lower than those of the other two cases. This is due to the shortage of spent FR fuel waiting for reprocessing as mentioned earlier.

Figure 5.4-11: Annual throughput of spent FR fuel



5.4.4. Inventories of the main radionuclide

5.4.4.1. Plutonium inventory

This section explains Pu inventory as follows:

- fabrication and reprocessing plant;
- nuclear power plant;
- plutonium intermediate storage;
- nuclear fuel cycle;
- waste.

Figure 5.4-12 shows Pu inventories at FR fuel fabrication facilities and reprocessing facilities. The calculation with FAMILY-21 assumed that the Pu shortage amount is compensated by imports from abroad (Pu fissile ratio: 0.72). In the 70 year case where 46.7 tonnes of Pu shortage occurred in the period from year 93 to 98, the Pu inventory during the time was slightly lower than those of the other two cases because the shortage was made up with high fissile Pu from outside.

Figure 5.4-12: Pu inventory in FR fuel fabrication plant and reprocessing plant

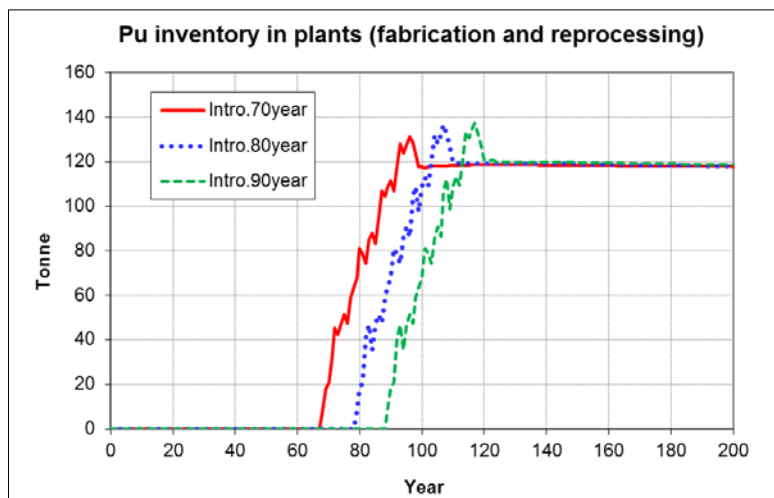
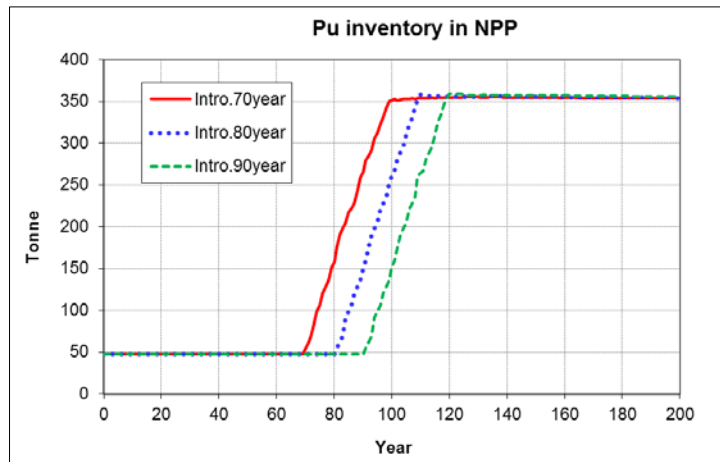


Figure 5.4-13 shows Pu inventories in NPPs. The increase was due to the increase of the FR capacity introduced instead of PWRs. Although the start of increase in the Pu inventories of three cases differ according to its FR introduction year, the slope of three lines are the same because all cases were calculated with the standard FR introduction rate (30 years).

Figure 5.4-13: Pu inventory in NPPs



As shown in Figure 5.4-14, Pu inventories of intermediate storage increase up to two years before the FR introduction of each case and change to decrease due to Pu consumption for FR fuel fabrication. The later the FR introduction, the larger is the capacity of Pu interim storage required.

Figure 5.4-14: Pu inventory of intermediate storage

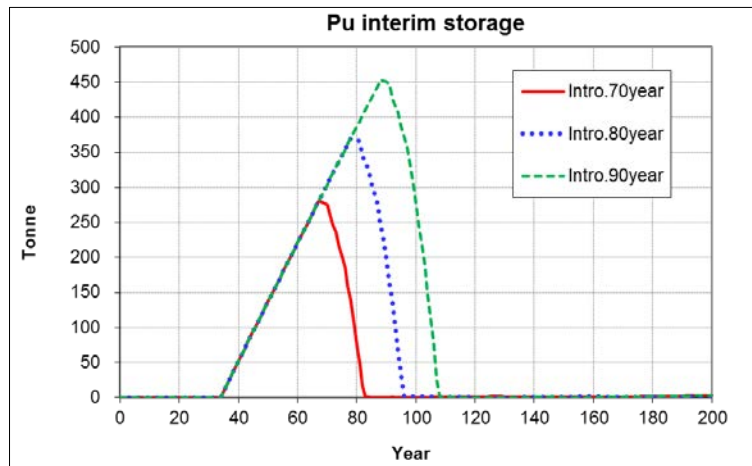
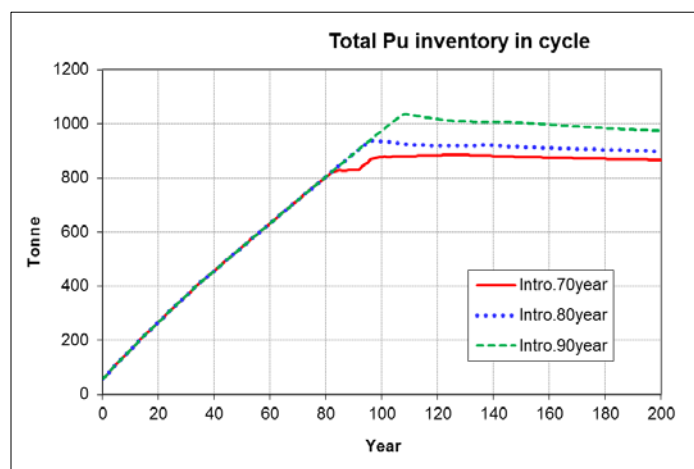
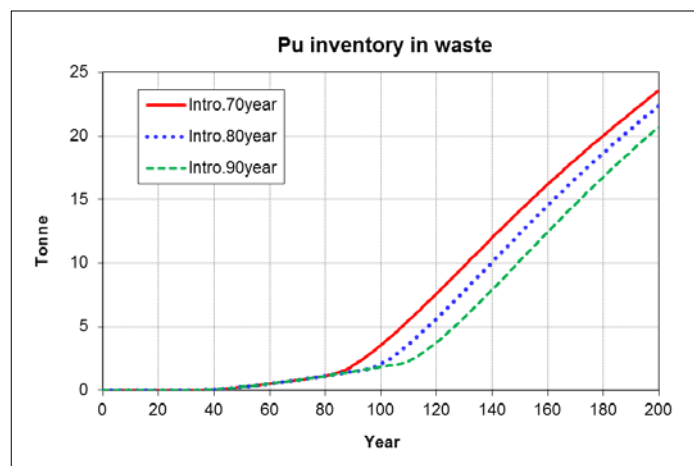


Figure 5.4-15 shows total Pu inventories in the nuclear fuel cycle. The difference between the three cases was due to the difference of the spent FR fuel storage amount waiting for reprocessing. In the 90 year case where the FR introduction is the latest, Pu supply from spent PWR UOX fuel increases, leading to the increase in the spent FR fuel storage for reprocessing.

Figure 5.4-15: Total Pu inventories in nuclear fuel cycle

As shown in Figure 5.4-16, Pu inventories in waste increase rapidly by reprocessing of spent FR fuel with high concentration Pu. Therefore, the increase of Pu inventory in waste by the start of reprocessing of spent FR fuel appeared first in the 70 year case.

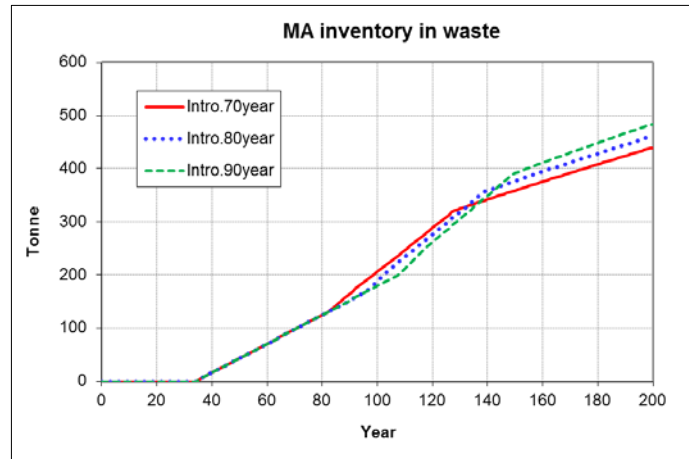
Figure 5.4-16: Pu inventories in waste

5.4.4.2. Minor actinide inventory

Figure 5.4-17 shows minor actinide (MA) inventories transferred to waste. Major MAs include ^{241}Am , ^{243}Am and ^{237}Np , which account for about 98% of total MAs. The shift of FR introduction year has impacts on the MA inventories, which indicate the increase from the start in the range between year 83 and 103 due to the start of spent FR fuel reprocessing and the decrease from the start in the range between year 129 and 150 due to the completion of spent PWR UOX fuel reprocessing. By comparison of the three cases it was found that the longer the term of spent PWR UOX fuel reprocessing (the annual reprocessing amount of

spent PWR UOX fuel was 850 tonnes/year for all cases), the more obvious the increase of the MA inventory.

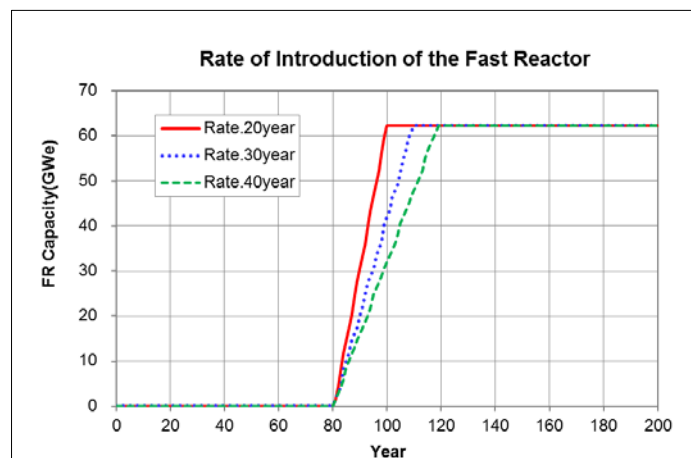
Figure 5.4-17: MA inventories in waste



5.5. Rate of introduction of the fast reactor

This section explains effects of the change of FR introduction rate. Figure 5.5-1 shows FR introduction rates calculated with the rate of 20 years and 40 years besides the standard rate of 30 years. In the 20-year case there was Pu shortage (total 38.1 tonnes from year 96 to 100) but it was found that this shortage can be avoided by changing the lead time of FR fuel fabrication facilities to one year from the standard two years. We will report major impacts on the PWR fuel cycle and the FR fuel cycle by the changes of FR introduction rate to 20 and 40 years from the standard 30 years.

Figure 5.5-1: Transitions of FR capabilities with each FR introduction rate



5.5.1. PWR cycle

5.5.1.1. Natural uranium consumption and SWU needs

Figures 5.5-2 and 5.5-3 show calculation results of natural uranium consumption and enriched uranium separation work units respectively. Decrease rates of each case in these two process amounts in the front-end are inversely proportional to its FR introduction rate.

Figure 5.5-2: Natural U consumption

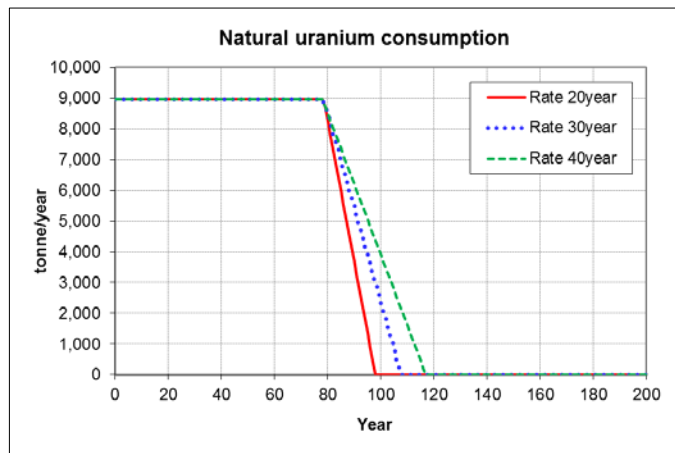
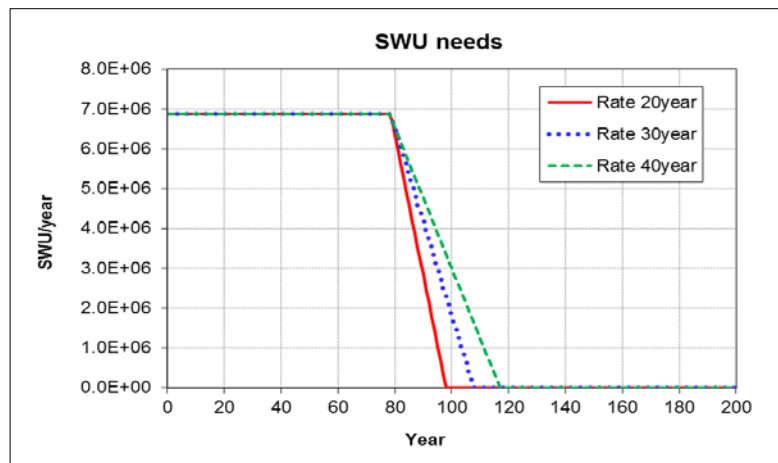


Figure 5.5-3: Enriched U separation work units



5.5.1.2. Enriched uranium demands and PWR-UOX fuel fabrication

As shown in Figures 5.5-4 and 5.5-5, calculation results also show that decrease rates of each case in the enriched uranium demands and PWR UOX fuel fabrication amount are inversely proportional to its FR introduction rate.

Figure 5.5-4: Enriched U demands

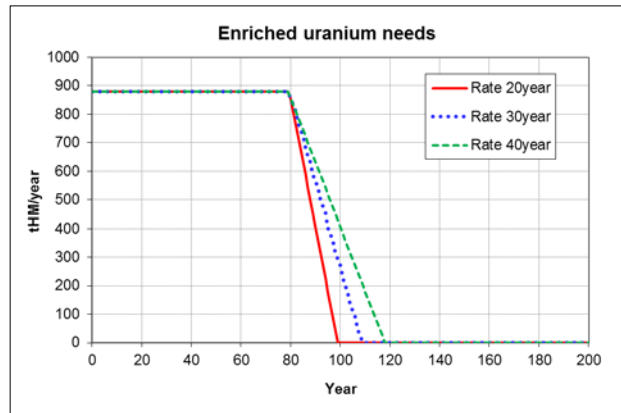
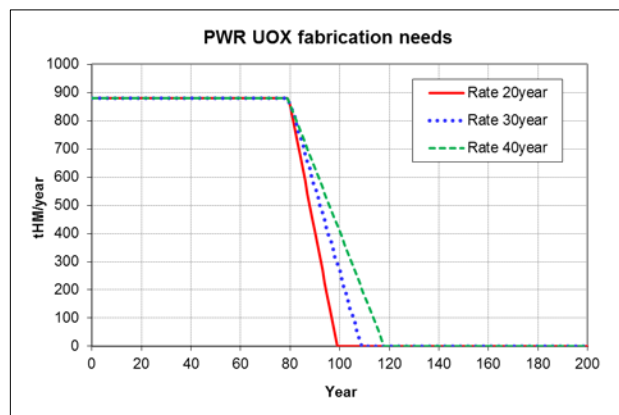


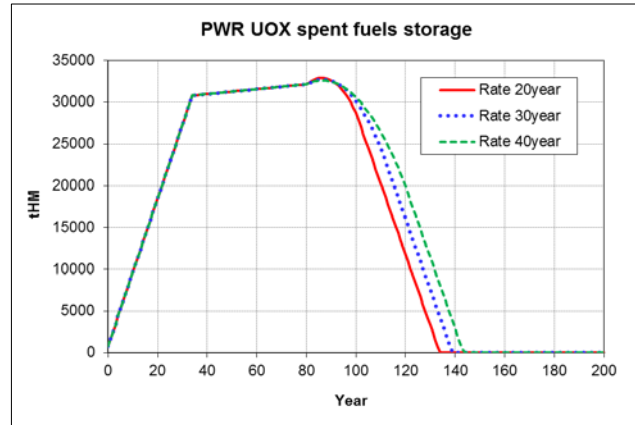
Figure 5.5-5: PWR-UOX fuel fabrication



5.5.1.3. Spent PWR-UOX fuel storage

Figure 5.5-6 shows the calculation results of spent PWR UOX fuel storage amount. The term from the start of FR introduction to the completion of spent PWR UOX fuel reprocessing of the standard 30 year case was 56 years, that of the 20 year case 53 years and that of the 40 year case 63 years. Thus, the shift of the FR introduction rate from the standard 30 year case is plus and minus 10 years, but its impact on spent PWR UOX fuel storage amount was around plus and minus 5 years from the standard case.

Figure 5.5-6: Spent PWR-UOX fuel storage

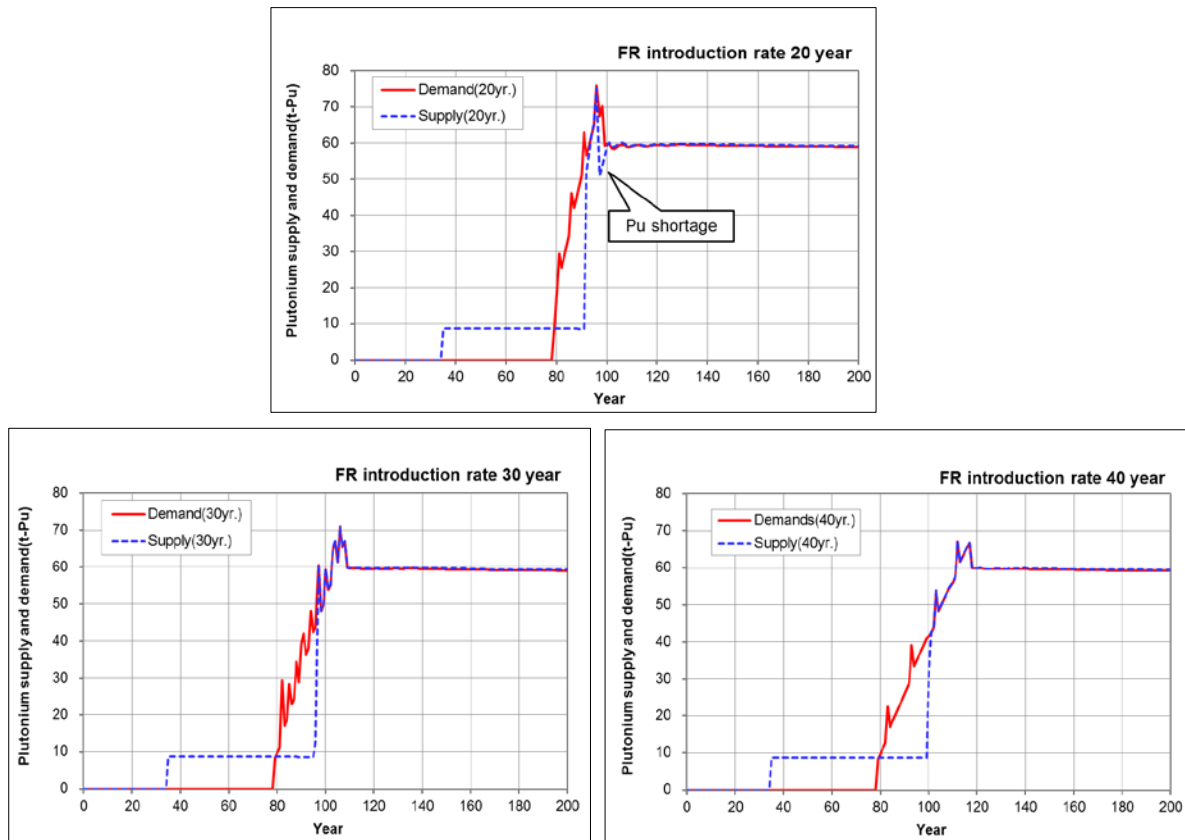


5.5.2. FR cycle

5.5.2.1. Plutonium supply and demand of FR fuel fabrication

Figure 5.5-7 shows Pu balance of each case of FR introduction rate. Pu supply means the annual recovered Pu amount from PWR UOX reprocessing and FR reprocessing and Pu demand means the annual Pu consumption for FR fuel fabrication. In the 20 year case, there was Pu shortage of 38.1 tonnes during the period from year 96 to 100, which is just before the end of transition from the PWR to FR. In the calculation with FAMILY 21, however, it is assumed that the Pu shortage was compensated by imports from abroad, so the Pu demand for FR fuel fabrication was secured.

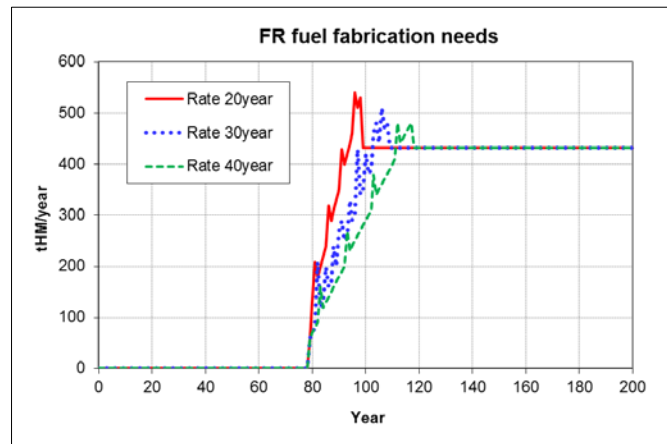
Figure 5.5-7: Effects on Pu balance with each FR introduction rate



5.5.2.2. FR fuel fabrication

Figure 5.5-8 shows FR fuel fabrication amounts for each FR introduction rate case. Initial loading fuel and replacement fuel for FRs are fabricated at almost the same time in FR fuel fabrication plants in the period of transition between PWR and FR. In the case of shortening the FR introduction rate, it is required to increase the capacity of FR fuel fabrication plants and even improve Pu supply capability. The annual average amount of fuel fabrication was 328 tonnes/year for the 20 year case, decreased to 293 tonnes/year for the standard 30 year case and further decreased to 276 tonnes/year for the 40 year case.

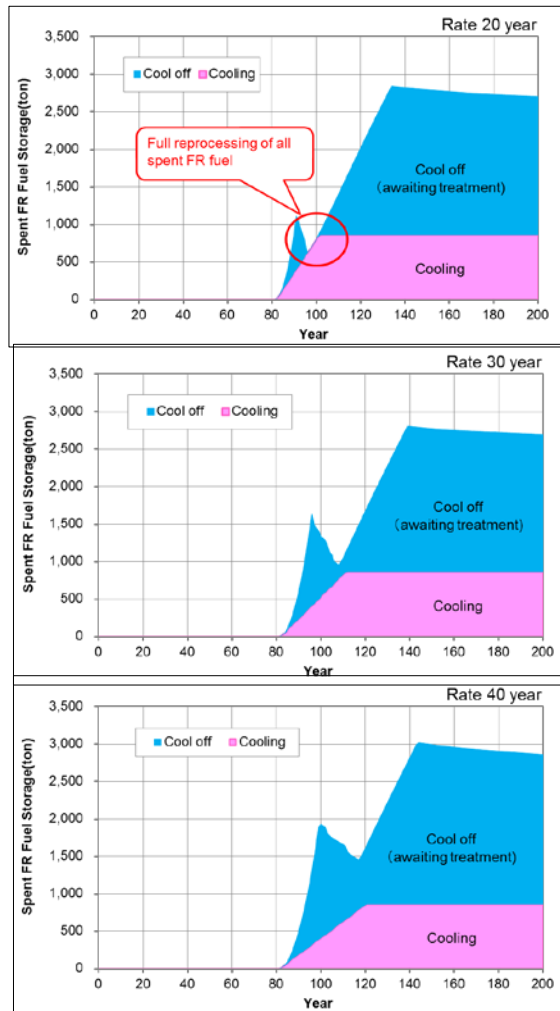
Figure 5.5-8: FR fuel fabrication



5.5.2.3. Spent FR fuel storage

Figure 5.5-9 shows details of spent FR fuel storage. In the case of 20 years of FR introduction rate, all spent FR fuels waiting for reprocessing were reprocessed in the period between years 96 to 100, wherein there was Pu shortage. This is because the annual amount of FR fuel fabrication increased due to the reduced FR introduction rate, resulting in reprocessing all spent FR fuels to meet the increased Pu demand affected by the increase of the annual FR fuel fabrication amount. Since Pu shortage occurred even with reprocessing of all spent FR fuels, it was compensated with Pu from abroad in the 20 year case.

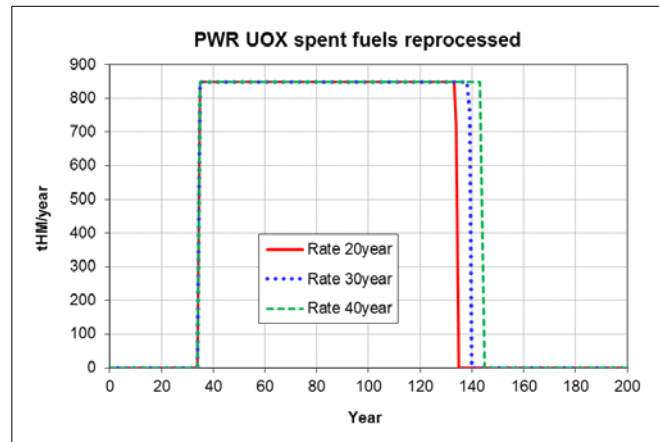
Figure 5.5-9: Details of spent FR fuel storage



5.5.3. Spent fuel reprocessing

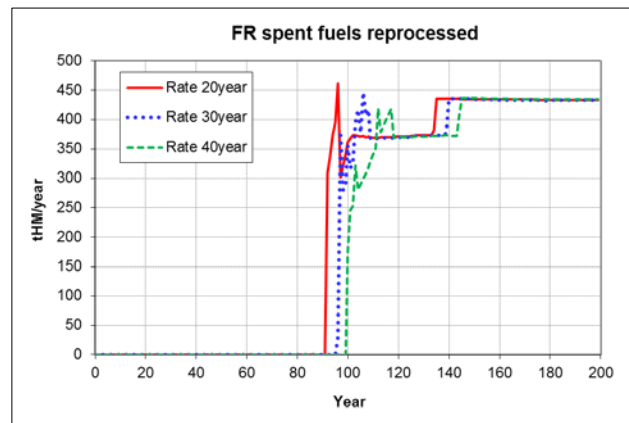
Figure 5.5-10 shows annual throughput of reprocessed spent PWR UOX fuel. The shift of the FR introduction rate from the standard 30 year case is plus and minus 10 years but its impact on the timing of the completion of spent PWR UOX fuel reprocessing is limited to plus and minus 5 years from the standard case.

Figure 5.5-10: Annual throughput of reprocessed spent PWR-UOX fuel



As shown in Figure 5.5-11, the slopes of the annual throughput of spent FR fuel reprocessing in the period of transition between PWR to FR are about the same with those of Pu demand for FR fuel fabrication plants. Meanwhile, the increase in steps during the period from year 134 to 144 is due to the increase of Pu from FR reprocessing to compensate the decrease of recovered Pu caused by the completion of spent PWR UOX fuel reprocessing.

Figure 5.5-11: Annual throughput of spent FR fuel



5.5.4. Inventories of the main radionuclide

5.5.4.1. Plutonium inventory

This section explains Pu inventory in the following.

- fabrication and reprocessing plant;
- nuclear power plant;

- plutonium intermediate storage;
- nuclear fuel cycle;
- waste.

Figure 5.5-12 shows Pu inventories at FR fuel fabrication facilities and reprocessing facilities. The Pu shortage of 38.1 tonnes occurred in the 20 year case was compensated by imports from abroad. The smaller the FR introduction rate, the greater the peak of Pu inventory just before the end of transition from PWR to FR.

Figure 5.5-12: Pu inventory in FR fuel fabrication plant and reprocessing plant

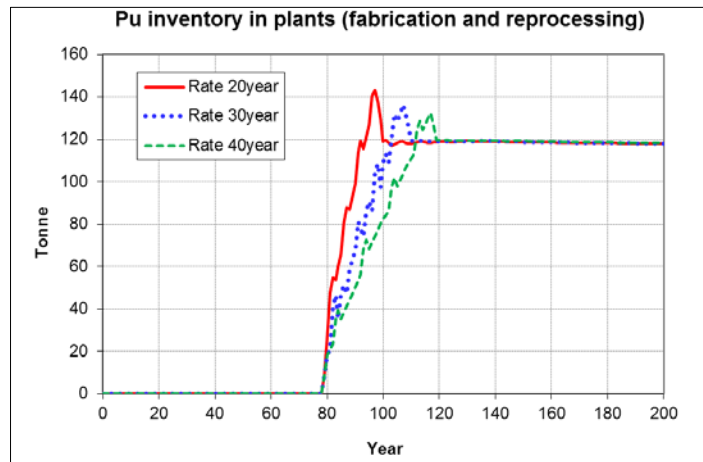


Figure 5.5-13 shows Pu inventories in NPPs calculated with each FR introduction rate. The rates of increase in Pu inventory with each FR introduction rate are in proportion to the increase curves of FR introduction amount described at the beginning.

Figure 5.5-13: Pu inventory in NPPs

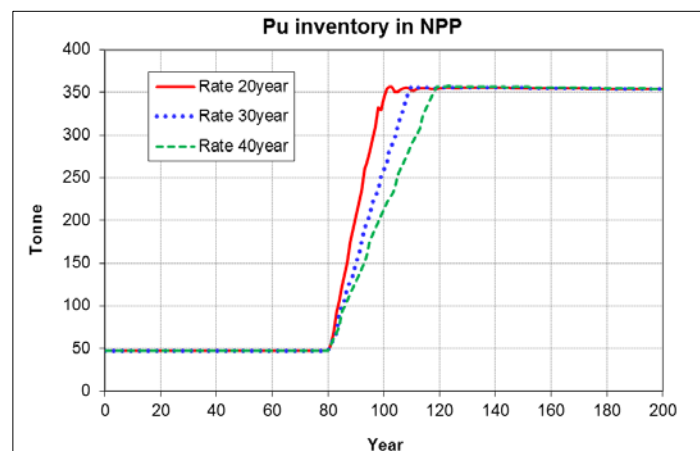


Figure 5.5-14 shows Pu inventories in interim storage calculated with each FR introduction rate. The Peak of the Pu inventory in each case is almost the same. Meanwhile, the speed of decrease in Pu interim storage amount tends to be faster as the FR introduction rate is shortened because FR fuel fabrication amount and Pu needs per year increase as the FR introduction rate is shortened.

Figure 5.5-14: Pu inventory of intermediate storage

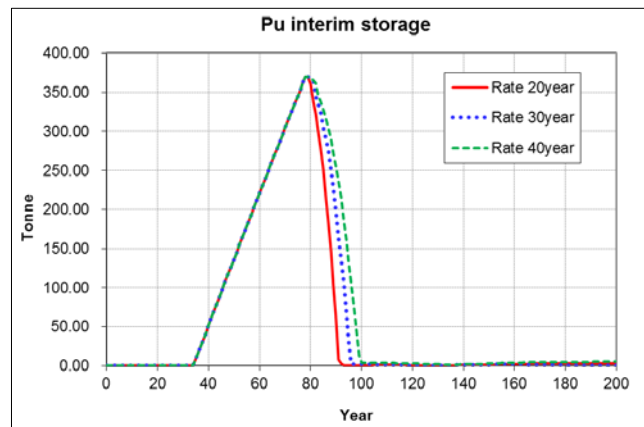


Figure 5.5-15 shows Pu inventories in the nuclear fuel cycle. The differences among each FR introduction case result mainly from differences in spent FR fuel storage amount waiting for reprocessing. After the resolution of Pu shortage the Pu inventories in the nuclear fuel cycle of the 20 year case and the standard 30 year case are almost the same.

Figure 5.5-15: Total Pu inventories in nuclear fuel cycle

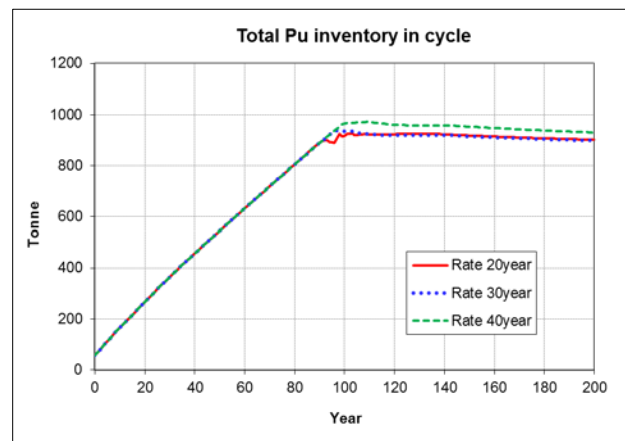
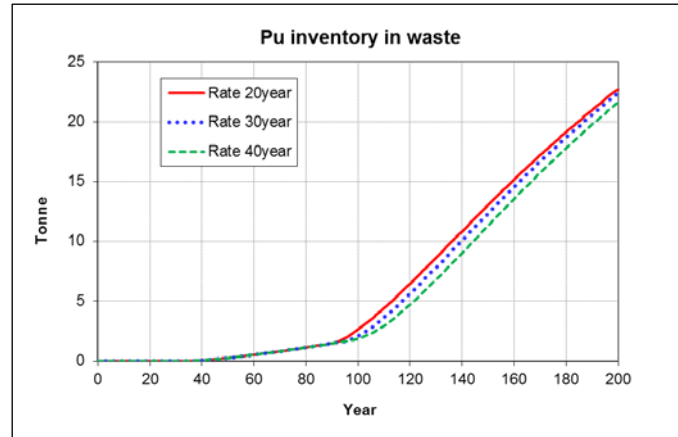


Figure 5.5-16 shows Pu inventories in waste. It rapidly increases by reprocessing spent FR fuel with high-concentration Pu. In the calculation it was found that the reprocessing of spent FR fuel is advanced to start as the FR introduction rate is shortened. Thus, differences in the

start timing of increase of Pu inventory in waste are dependent on the differences of the start timing of spent FR fuel reprocessing.

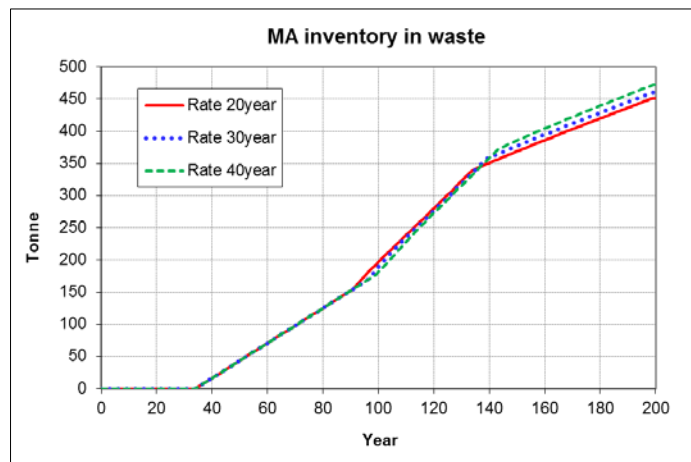
Figure 5.5-16: Pu inventories in waste



5.5.4.2. Minor actinide inventory

Figure 5.5-17 shows MA inventories in waste calculated with each FR introduction rate. The amount of MA transferred into waste change mainly with reprocessing conditions. MA inventories in waste increased in the period from around year 35 to 90 by PWR UOX and FR reprocessing. Its increase in the period from around year 90 to 135 is attributed to MAs recovered from reprocessed PWR UOX and FR fuels and this period is when the amounts of recovered MAs are largest. Meanwhile, the reduction of increase rate after year 135 is due to the decrease of recovered MAs resulting from the completion of PWR UOX reprocessing. The difference in MA inventory around year 135 with each FR introduction rate comes from the difference in the total reprocessing amount of spent PWR UOX fuel.

Figure 5.5-17: Minor actinide inventories in waste



References

- [1] K. Cohen (1951), *"The Theory of Isotope Separation as Applied to the Large-scale Production of U235"*, New York: McGraw-Hill.
- [2] R. Scott Kemp (2009), "Gas Centrifuge Theory and Development: A review of U.S. Program", *Science and Global Security*, 17; 1-19, 2009, Princeton University, Princeton, NJ.

6. Effects of the uncertainty of reactor parameters

6.1. Variation of PWR Burn-up

The results on the study of sensitivity to the PWR characteristics are described in this chapter. The considered PWR fuels characteristics are the following:

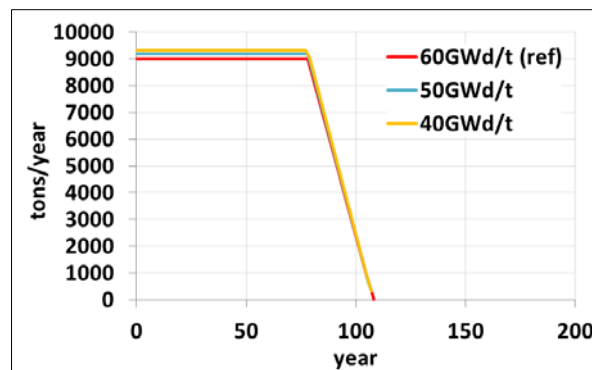
- 3.50% ^{235}U - 40 GWd/t - 3x320 EFPD;
- 4.25% ^{235}U - 50 GWd/t - 4x325 EFPD;
- 4.95% ^{235}U - 60 GWd/t - 4x410 EFPD.

6.1.1. Front-end cycle

6.1.1.1. Natural uranium consumption

The decrease in the PWR fuel BU leads to a slight increase in the natural uranium consumption (Figure 6.1-1) due to an increase of the loading frequency.

Figure 6.1-1: Sensitivity to PWR – Natural U consumption

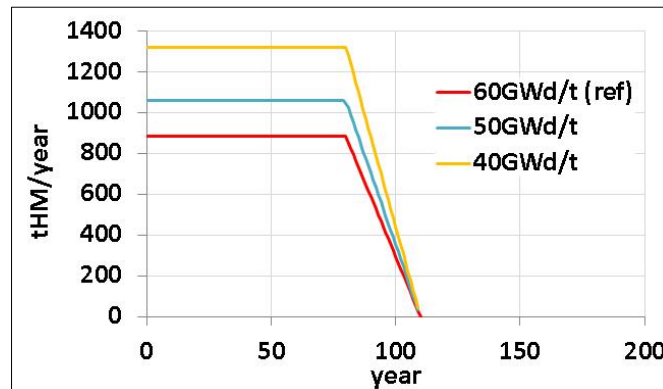


Between year 0 and year 80, the yearly uranium consumption is 9000 tonnes/year for the reference case, 9175 tonnes/year if the BU is reduced to 50 GWd/t (+2%) and 9320 tonnes/year if the BU is reduced to 40 GWd/t (+3.5%).

6.1.1.2 UOX fabrication needs

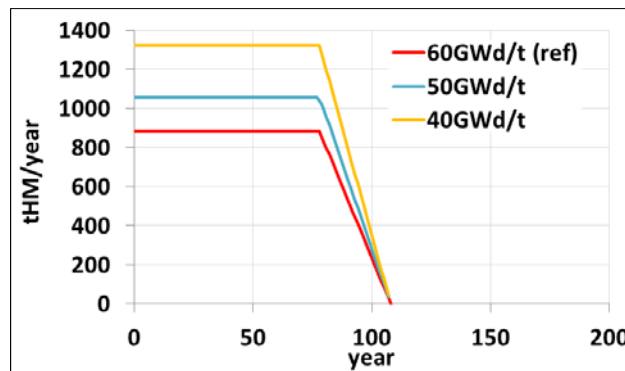
Due to an increase in the loading frequency, a decrease in the BU leads to higher needs in PWR fuels fabrication. A decrease of the BU to 50 GWd/t leads to a 20% increase in the UOX fabrication needs and decrease to 40 GWd/t leads to a 50% increase in the UOX fabrication needs as shown in Figure 6.1-2.

Figure 6.1-2: Sensitivity to PWR – PWR UOX fabrication needs



6.1.1.3. Enrichment needs

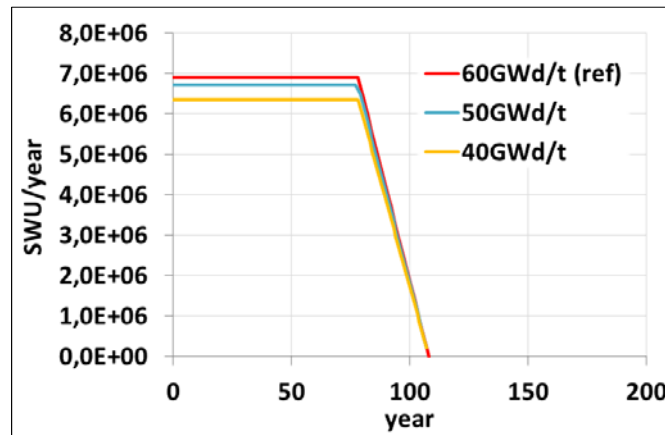
Figure 6.1-3: Sensitivity to PWR – Enriched U needs in tHM/y



The enriched uranium needs (see Figure 6.1-3) in tHM/y follow the UOX fabrication needs: they are increased by 20% when the BU is reduced to 50GWd/t and by 50% when the BU is reduced to 40 GWd/t as indicated in Figure 6.1-3.

Due to the decrease in the ^{235}U enrichment linked with the decrease in BU, the enrichment need in SWU/y is less impacted and is positively related to the change in BU. It decreases by 3% when the BU is reduced to 50 GWd/t and by 8% when the BU is reduced to 40 GWd/t (see Figure 6.1-4).

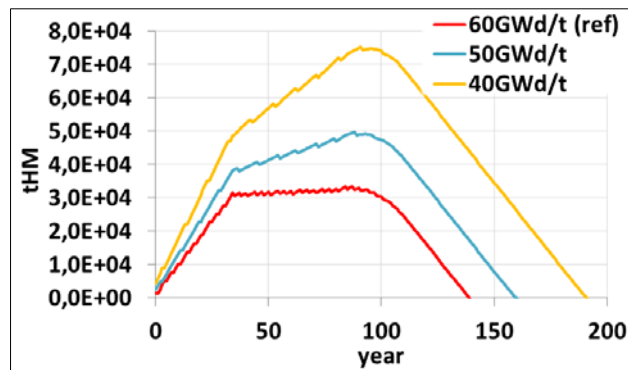
Figure 6.1-4: Sensitivity to PWR – Enriched U needs in SWU/y



6.1.2. Back-end cycle

6.1.2.1. Spent fuels storage

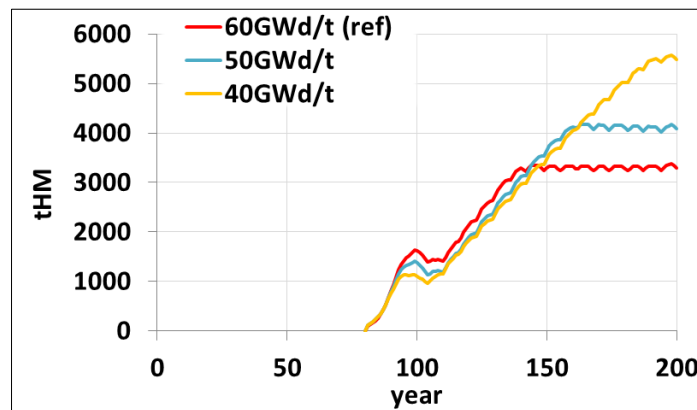
Figure 6.1-5: Sensitivity to PWR – PWR UOX spent fuels storage



When the UOX BU is reduced, the higher UOX fuels fabrication (due to an increase in the loading frequency) leads to an increase in the UOX spent fuels storage. When the BU decreases, the UOX spent fuels storage is no longer stabilised and reaches a maximum of 49 630 tHM (+ 49%) for a 50GWd/t BU and of 75 150 tHM (+ 125%) for a 40 GWd/t BU (see Figure 6.1-5).

The UOX spent fuel reprocessing being fixed at 850 tHM/y, the UOX spent fuels are consumed more or less quickly. Their storage drops to zero in year 139 in the reference case, 160 in the 50 GWd/t case and 191 in the 40 GWd/t case.

Figure 6.1-6: Sensitivity to PWR – FR spent fuels storage

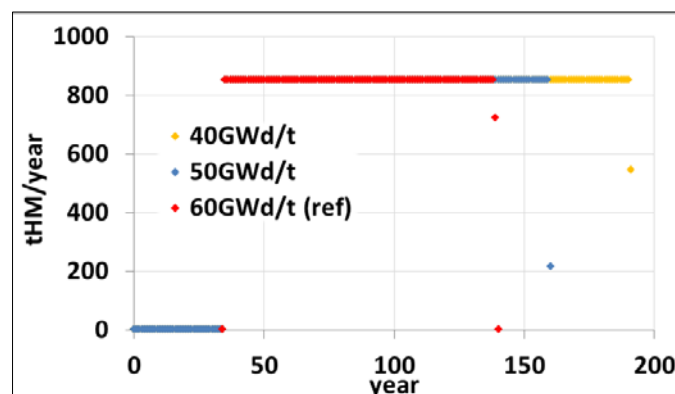


The reprocessing of FR spent fuel is adapted to meet the need in plutonium for fresh fuels fabrication. When the UOX burn-up is reduced, the reprocessing of UOX spent fuels over a longer period leads to reprocess less FR spent fuels at the beginning of their reprocessing (see Figure 6.1-8). That explains the increase in the stored FR spent fuels which stabilises at 4125 tHM (+ 25%) if the BU is 50 GWd/t and at 5415 tHM (+ 64%) if the BU is 40 GWd/t as shown in Figure 6.1-6.

6.1.2.2 Spent fuel reprocessing need

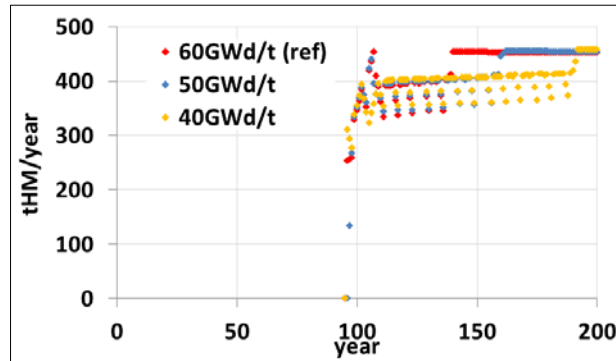
The UOX spent fuel reprocessing is fixed at 850 tHM/year. Depending on the UOX spent fuels availability, it stops in year 160 (+ 21y) if the burn-up is reduced to 50 GWd/t and in year 191 (+ 52 y) if the burn-up is reduced to 40GWd/t (see Figure 6.1-7).

Figure 6.1-7: Sensitivity to PWR – PWR UOX spent fuels reprocessed



The reprocessing of FR spent fuel is adapted to meet the need in plutonium for fresh fuels fabrication. Whatever the UOX burn-up, it stabilises at 455 tHM/year when the reprocessing of UOX spent fuels stops (see Figure 6.1-8).

Figure 6.1-8: Sensitivity to PWR – FR spent fuels reprocessed

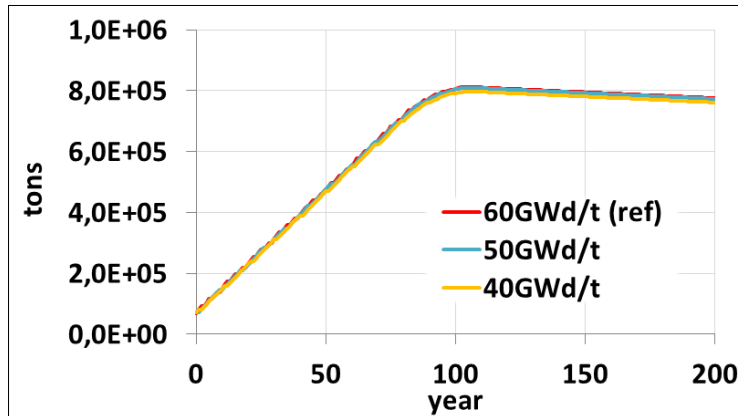


If the UOX burn-up is reduced to 50 GWd/t, in years 107 and 108 there is not enough FR spent fuels available for reprocessing and 16 tonnes of external plutonium are necessary to run the simulation. If the UOX burn-up is reduced to 40 GWd/t, 62 tonnes of external plutonium are required between 104 and 109. To avoid using external plutonium, it would have been possible to increase the UOX spent fuels reprocessing from year 80 up to 950 tHM for the 50 GWd/t case and 1150 tHMGWd/t for the 40 GWd/t case.

6.1.2.3. Materials interim storage

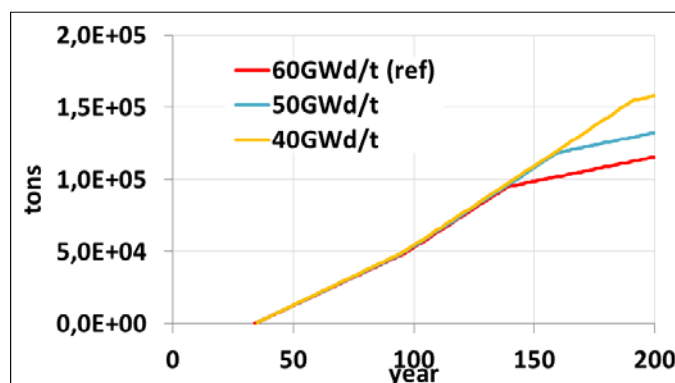
The depleted uranium storage (see Figure 6.1-9) is slightly impacted by the change in the enrichment need. It is reduced by 0.4% if the BU is reduced to 50 GWd/t and by 1.9% if the BU is reduced to 40 GWd/t.

Figure 6.1-9: Sensitivity to PWR – Depleted U storage



The reprocessed uranium storage (see Figure 6.1-10) is impacted by the longer UOX spent fuels reprocessing. In year 200, it is increased by 14% if the BU is reduced to 50 GWd/t and by 37% if the BU is reduced to 40 GWd/t.

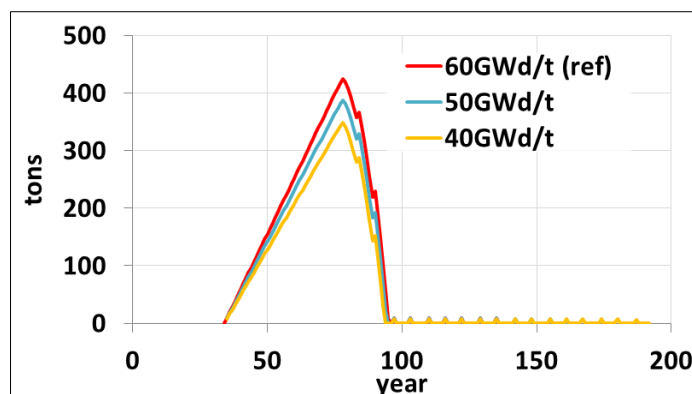
Figure 6.1-10: Sensitivity to PWR – Reprocessed U interim storage



The Pu quantity and quality in UOX spent fuels are affected by the modification of UOX characteristics. That impacts the separated Pu storage (see Figure 6.1-11).

The maximum Pu interim storage is reduced by 9% if the BU is 50 GWd/t and by 18% if the BU is 40 GWd/t.

Figure 6.1-11: Sensitivity to PWR – Pu interim storage



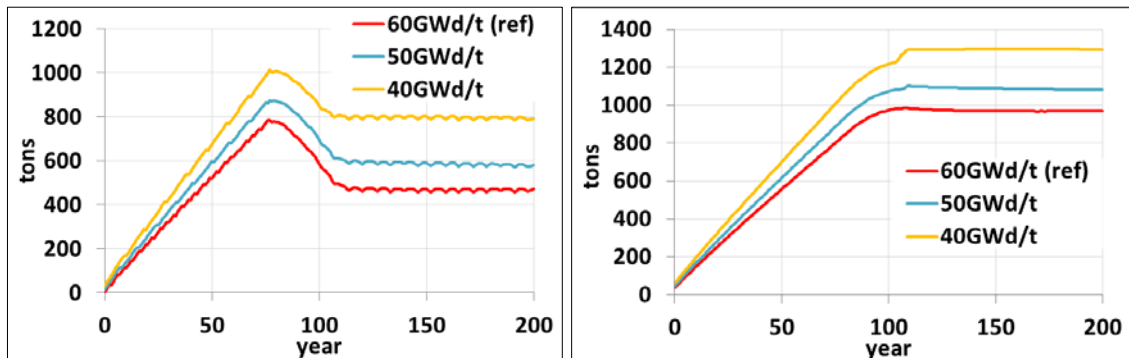
6.1.3. Inventories

6.1.3.1. Pu inventory in fuel cycle

Due to an increase in the Pu inventory in spent fuels storage, a decrease in the UOX BU leads to an increase in the Pu inventory in cycle.

In year 110, the Pu inventory is stabilised at 970 tonnes in the reference case, at 1090 tonnes (+12%) in the 50 GWd/y case and at 1295 tonnes (+33%) in the 40 GWd/t case.

Figure 6.1-12: Sensitivity to PWR – Pu inventory in spent fuels storage (left) and in cycle (right)

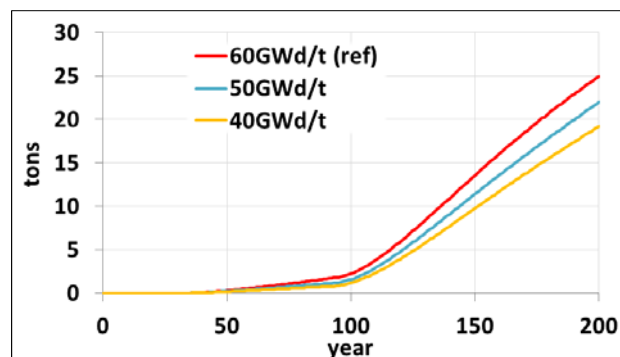


6.1.3.2. Pu inventory in waste

A decrease in the burn-up leads to a decrease in the Cm production. Thus, the Pu production by Cm decay in the waste is reduced.

In year 200, the Pu inventory in waste reaches 25 tonnes in the reference scenario, 22 tonnes (-12%) in the 50 GWd/t scenario and 19 tonnes (-23%) in the 40 GWd/t scenario (see Figure 6.1-13).

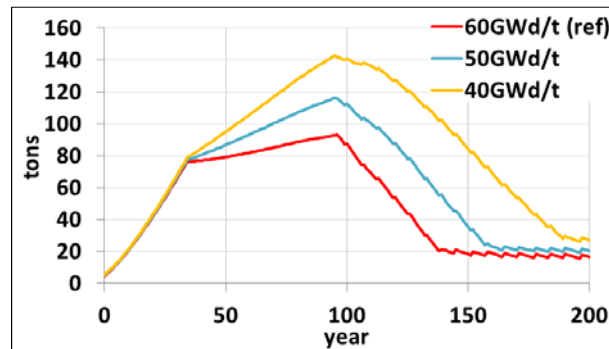
Figure 6.1-13: Sensitivity to PWR – Pu inventory in waste



6.1.3.3. MA inventories

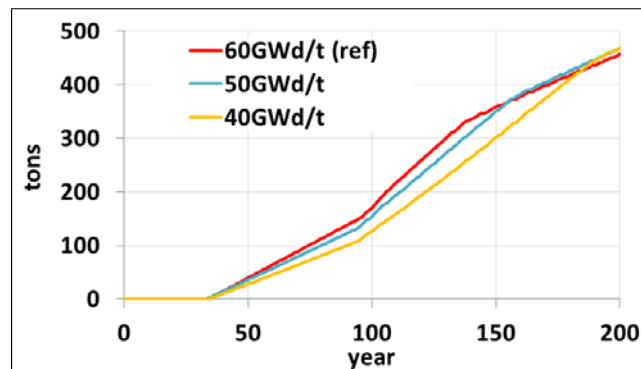
Up to year 191, the MA inventory in cycle (see Figure 6.1-14) is impacted by the differences on the UOX spent fuels storage. It reaches a maximum of 116 tonnes (+ 25%) in the 50 GWd/t case and of 143 tonnes (+ 53%) in the 40 GWd/t case. From year 192, because of the differences on the FR spent fuels storage, it stabilises at 21 tonnes (+ 22%) in the 50 GWd/t case and at 28 tonnes (+ 62%) in the 40 GWd/t case.

Figure 6.1-14: Sensitivity to PWR – MA inventory in cycle



The MA inventory in waste (Figure 6.1-15) is impacted by a change in the UOX burn-up, mainly because of the differences on the FR spent fuels reprocessing. From year 192, the MA inventory in waste becomes almost the same for the three studied cases.

Figure 6.1-15: Sensitivity to PWR – MA inventory in waste



6.2. Fast reactor parameters

In this section, the parameters related to fast reactors (FR) design are investigated. Main attention has been focused on burn-up (BU) variation, initial minor actinides (MA) content and breeding ratio (BR).

As shown in the following, the variation of the mentioned parameters has a large effect mainly on quantities related to the fuel cycle back-end and to the transition period from PWR to SFRs fleet. Results reported in this section have been provided by using the COSI code (CEA, KIT, ENEA) and the TR_EVOL code (CIEMAT). Discrepancies between COSI and TR-EVOL codes have been already discussed in Chapter 3 (reference scenario).

6.2.1. Variation of burn-up

The reference SFR model considered in the study, the European Fast Reactor (EFR) concept, has a BU of 136 GWd/tHM (for the fissile core) obtained by considering 5 cycles of 340 effective full power days (EFPD) each. The description of the reference system is available in Chapter 2.

In this section only the results obtained by ENEA for a 5% reduction of the BU at discharge have been reported. The COSI6 code has been used for this purpose. Two cases have been considered by ENEA: 1) reducing the BU (input to COSI) from 136 to 129.2 GWd/tHM (cases indicated in Figure 6.2-1 and Figure 6.2-2 as “95%BU (ENEA)”); and 2) by reducing the EFPD/cycle from 340 to 323 EFPD (cases indicated in Figures 6.2-1 and Figure 6.2-2 as “323_efpd (ENEA)”).

Changing the SFR model, e.g. from EFR to the more recent European Sodium Fast Reactor (ESFR) design [1], implies also some changes on the discharge BU (from 136 GWd/tHM to 100 GWd/tHM, for EFR and ESFR systems respectively). The results obtained by the adoption of different SFR concepts are not included in this section but they are considered as part of the section related to BR variation (Section 6.2.3).

The two options for BU reduction considered by ENEA have been compared with the reference case (EFR). The Pu inventory in the cycle is reported in Figure 6.2-1. The results obtained by the two options considered are in good agreement. The 5% reduction of BU leads to, at the end of the scenario (y = 200), ca. 20% less Pu in the cycle.

In Figure 6.2-2, the Pu inventory in waste is shown. Also in this case, the results obtained by the two options considered are in good agreement. At the end of the scenario (at year 200), ca. 12% more Pu is in waste due to more reprocessing steps needed to complete the scenario.

In Figure 6.2-3, the MA inventory in waste is shown. Also in this case, the results obtained by the two options considered are in good agreement. At the end of the scenario (year 200), ca. 7-8% more MA is in waste due to more reprocessing steps needed to complete the scenario.

Figure 6.2-1: Pu inventory in cycle (BU variation)

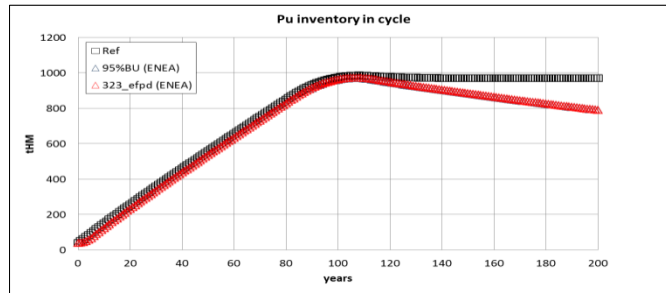


Figure 6.2-2: Pu inventory in waste (BU variation)

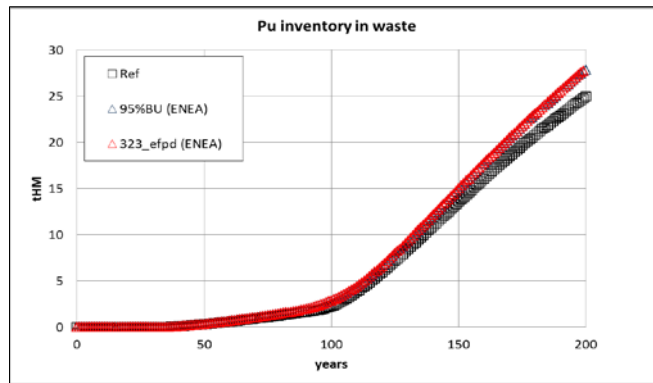
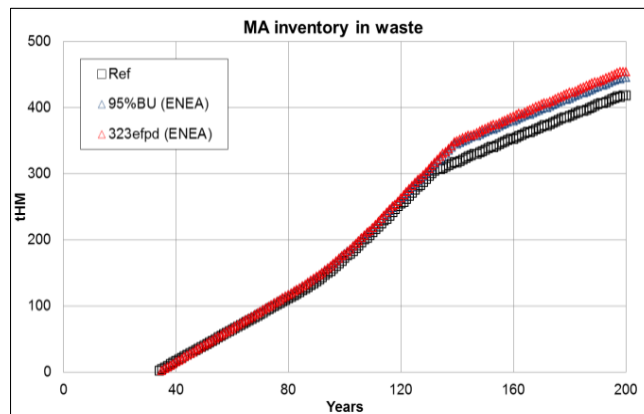


Figure 6.2-3: MA inventory in waste (BU variation)



6.2.2. Initial MA content

Concerning the initial MA content, a large set of options has been considered by CEA, CIEMAT and KIT. A summary of the significant results are reported in this section.

Different kinds of SFR concepts have been considered for the introduction of MA into the fuel. More details concerning the reactor models are reported in Appendix A. The reactors considered for analysing the impact of initial MA content are as follows:

- (1) The reference SFR model (EFR concept) has been considered by CEA and CIEMAT. Both organisations considered homogeneous MA (Am, Np, and Cm) loadings. An initial content of 1% and 2% Am has been considered for the study.
- (2) The European Sodium Fast Reactor – Working Horse (ESFR WH) concept has been considered by KIT. This system is characterised by a core with a Breeding Ratio (BR) ca. 1 obtained without the adoption of fertile blankets [1]. Homogeneous americium (Am) loadings have been considered for this case, namely neptunium and curium are not loaded in core. An initial content of 1%, 2% and 4% Am has been considered for the study.
- (3) The ESFR CONF-2 concept has been considered by KIT. This core represents an optimised core (from the point of view of safety) with respect to ESFR WH core obtained during the FP7 – CP-ESFR project [2]. The system is characterised by a core BR ca. 1 to which lower fertile blankets (LAB) have been added (the system BR is closed to ca. 1.2). For this option, homogeneous and heterogeneous (in LAB) americium (Am) loadings have been considered. As homogenous case, only 4% Am (loaded in core in LAB) has been considered, while for the heterogeneous case the following options have been used: 5%, 10% and 20% Am in the LAB (the core is loaded only with standard MOX fuel).
- (4) A mixed fleet of ESFR WH (no loaded by MA) and ASTRID-like MA burners has been considered by KIT. The ASTRID-like MA burners are critical low CR (ca. 0.7) fast reactors developed at KIT in the framework of partitioning and transmutation studies [3,4]. Under this option, different shares (1/3, 1/5 and 1/9) of the SFR fleet have been substituted by the MA burners. This is a quite extreme case where large variations of the reference scenario have been considered. However, as shown in the following, the results obtained may be considered as complementary to the ones provided with EFR or ESFR systems and therefore included in the study.

Concerning the impact of changing SFR design (from EFR to ESFR WH or ESFR CONF2) more considerations are included in Section 6.2.3 where the SFR systems loaded with standard MOX fuel are compared. Concerning point 4), the change on the fleet composition has also an impact on the “effective BR” of the fleet (reducing the “effective BR” below unity). However, this aspect has not been considered again in Section 6.2.3 related to BR impact.

In order to avoid mixing the effects related to the type of SFR systems with the effects related to MA content, it has been decided to show the results grouped per type of SFR. For each group, a reference case has been established.

For point 1) the comparison is done with respect to EFR reference design (namely the reference scenario, chapter 3). For point 2) the comparison is done with respect to ESFR WH concept (fuel without MA) and for point 3) with respect to ESFR CONF-2 concept. Concerning point 4, the reference is a fleet composed by 100% ESFR WH, therefore same reference as at point 2.

In general, the options considered have a limited impact on the quantities related to the PWR front-end (e.g. natural U demand, etc.). Therefore, here the focus is only on quantities related to the fuel cycle back-end and to the transition period from PWR to SFRs fleet.

In Figures 6.2-4 and 6.2-5, the Pu inventory in plants (fabrication and reprocessing plants) is shown for the EFR and ESFR WH cases, respectively. The introduction of MA in fuel does not affect the results when compared with the reference cases. Same trends are obtained for all the considered SFR cases (including the ESFR CONF2 cases). Adding some changes on the fleet leads to some differences (not smoothed results) as indicated in Figure 6.2-6. However, the average values remain around 150 tonnes for the mixed fleets as for the ESFR WH case.

In Figures 6.2-7 and 6.2-8, the Pu inventory in reactor is shown. The adoption of different amount of MA into the SFR fuel does not change significantly the behaviour as indicated for the EFR case (see Figure 6.2-7) and for the ESFR WH, CONF2 and mixed fleet cases (see Figure 6.2-8). Some variations in Figure 6.2-8 can be due to the fact that large amount of MA in fuel may slightly change the BR of the systems and the discharge Pu quality.

Figure 6.2-4: Pu inventory in plants (EFR case – initial MA content)

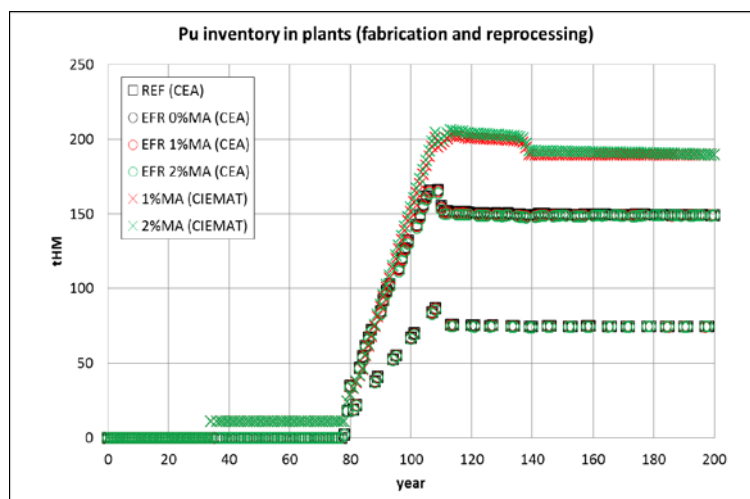


Figure 6.2-5: Pu inventory in plants (ESFR case – initial MA content)

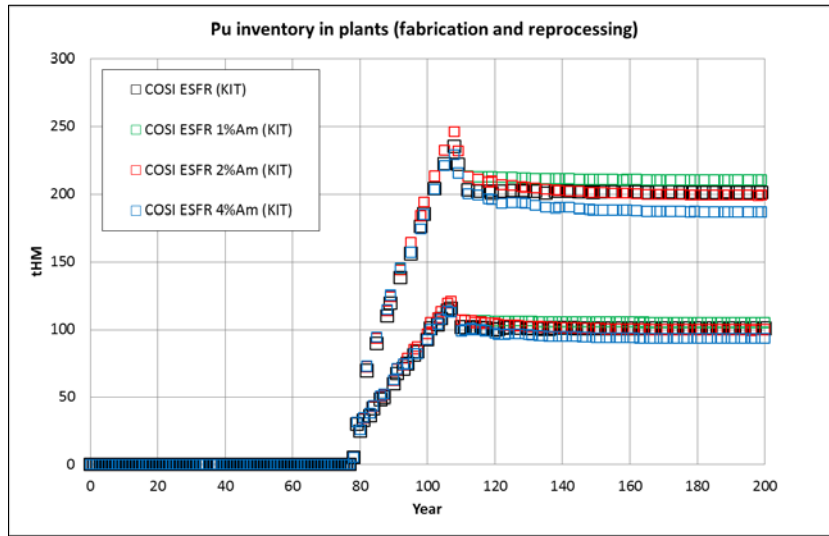


Figure 6.2-6: Pu inventory in plants (ESFR+burner case – initial MA content)

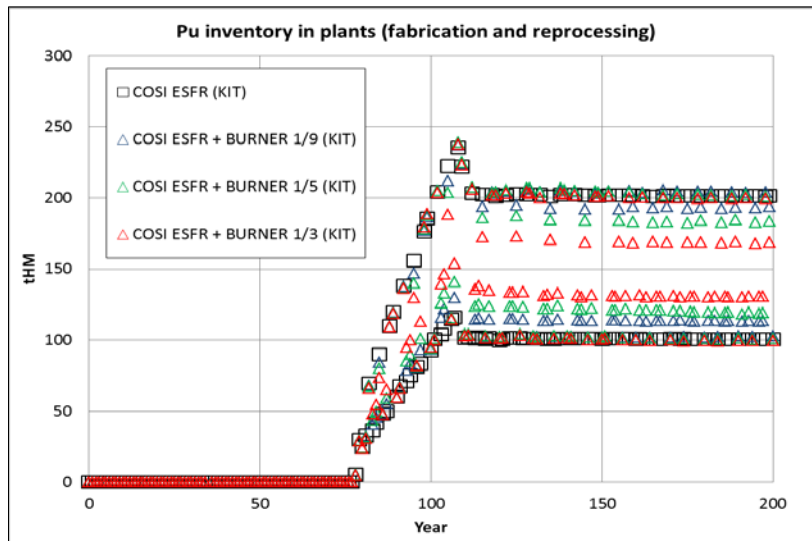


Figure 6.2-7: Pu inventory in NPPs (EFR case – initial MA content)

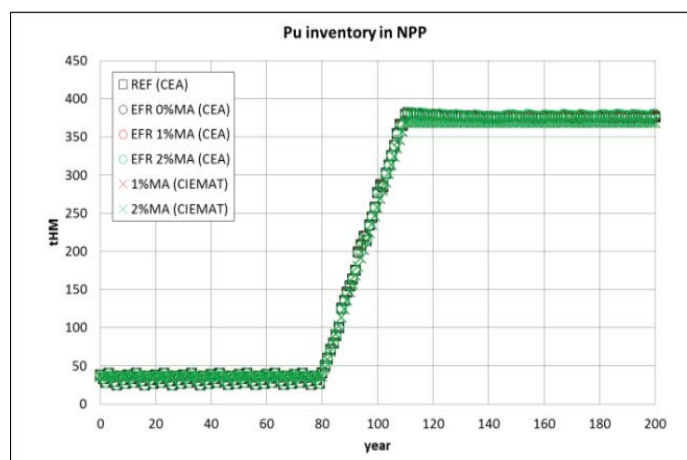
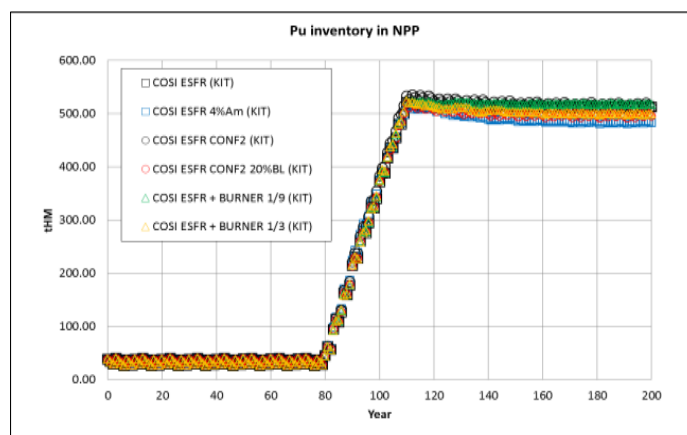


Figure 6.2-8: Pu inventory in NPPs (ESFR WH, ESFR CONF2 and mixed fleet cases – initial MA content)



In Figures 6.2-9 to 6.2-12, the Pu inventory in cycle (all facilities excluded the waste storages) is shown. The introduction of 1%-2% of MA in core leads to an increase on the Pu in the cycle (slightly affecting the BR of the fast reactors). At the end of the scenario (in year 200), an increase of up to 16% may be obtained. Similar results (see Figure 6.2-10) are obtained for the case based on ESFR WH. The differences are due to the FR model and to the fact that in ESFR only Am (and not Am, Cm and Np) is loaded.

If MAs are heterogeneously loaded in the LAB (see Figure 6.2-11) of the ESFR CONF2 case, the total Pu inventory in the cycle remains unchanged. At the contrary the adoption of the mixed fleet leads to a reduction in the total inventory (ca. 17% less at end of scenario). This is because the burner systems considered have a BR(Pu) lower than 1 (the systems are mainly burning MAs but also slightly Pu [3]). This characteristic is shown in Figure 6.2-12.

In Figures 6.2-13 to 6.2-15, the MA inventory in waste has been shown for the ESFR WH, ESFR CONF2 and mixed fleet cases. When only Am is loaded in the systems (cases with ESFR WH and CONF2) the MA inventory in waste stabilises to ca. 130 tonnes (in year 200) that results in a reduction of 72% when compared to the reference model (either ESFR WH or ESFR CONF2). The main contributors to this inventory are Cm and Np isotopes.

The adoption of the mixed fleet where all MA are loaded in the burners systems implies a large reduction of MA inventory in waste (indeed all MA remain in the cycle) to values of the order of 1-1.2 tonnes in the year 200. The inventory of MA in waste is similar to the ones obtained by CIEMAT while loading EFR with 1-2% of MAs.

Figure 6.2-9: Pu inventory in cycle (EFR case – initial MA content)

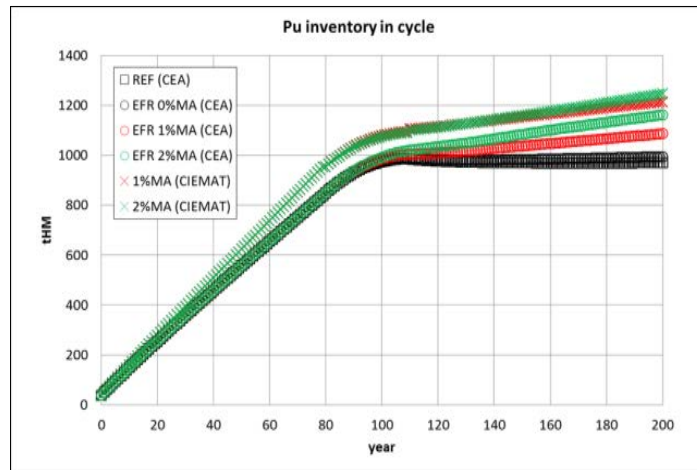


Figure 6.2-10: Pu inventory in cycle (ESFR WH case – initial MA content)

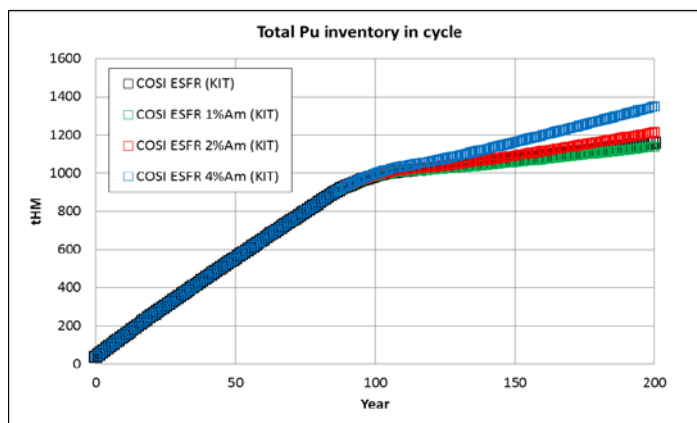


Figure 6.2-11: Pu inventory in cycle (ESFR CONF2 case – initial MA content)

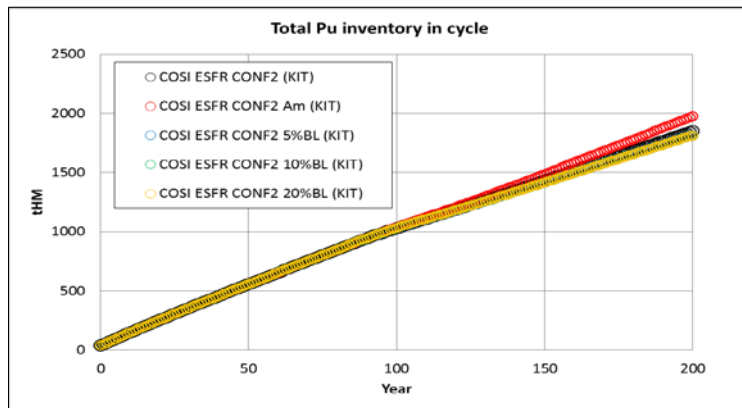


Figure 6.2-12: Pu inventory in cycle (ESFR+burner case – initial MA content)

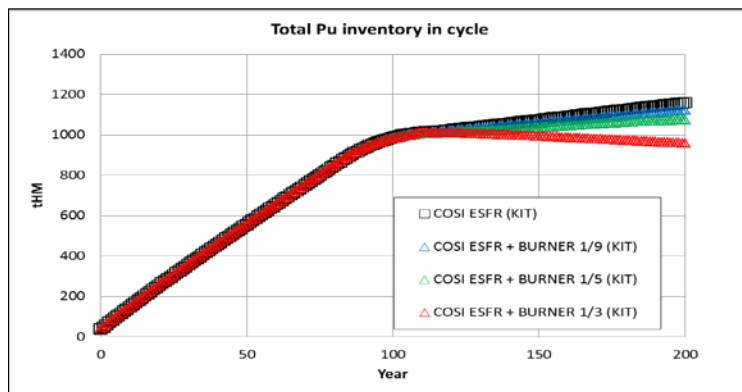
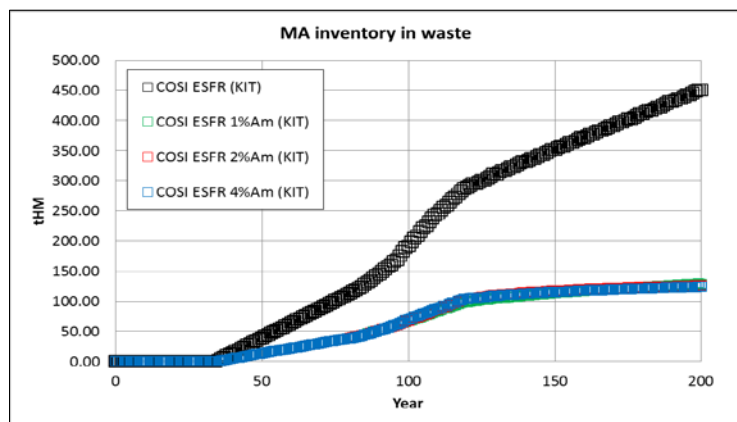
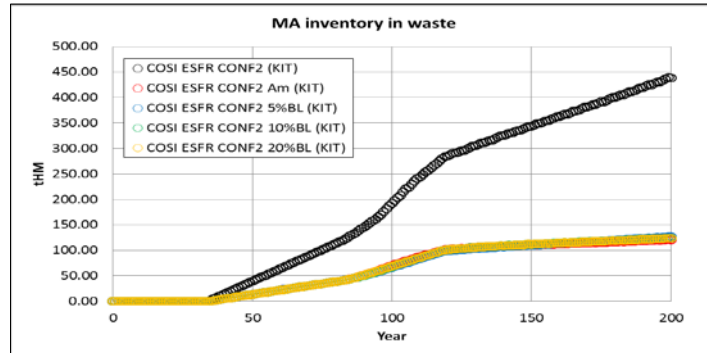


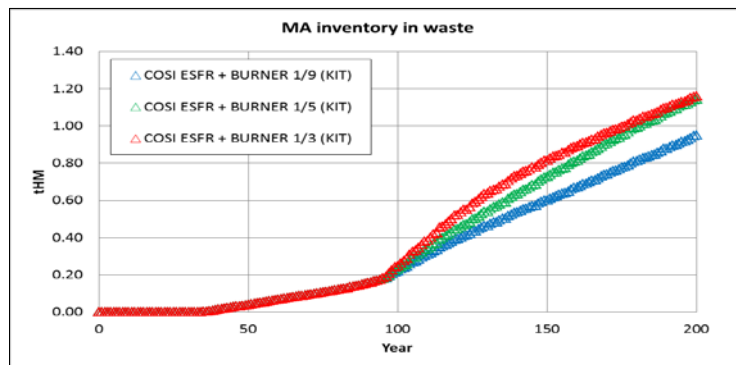
Figure 6.2-13: MA inventory in waste (ESFR WH case – initial MA content): Separation efficiency 99.9%



**Figure 6.2-14: MA inventory in waste (ESFR CONF2 case – initial MA content):
Separation efficiency 99.9%**



**Figure 6.2-15: MA inventory in waste (ESFR+burner case – initial MA content):
Separation efficiency 99.9%**



The MA inventory in the cycle (all facilities excluded waste storages) has been analysed as well. The results are shown in Figures 6.2-16 to 6.2-18. Loading MA into the fuel leads to a larger inventory in the cycle compared to the reference case. Data provided by CEA and CIEMAT concerning EFR cases are in good agreement (Figure 6.2-16). The results show that with 1% of initial MA content, the inventory in the cycle may be stabilised to 300 tonnes in the year 200 but it can be reduced to 130 tonnes if a larger amount is charged in core (2% case). In Figure 6.2-17 the ESFR WH case is shown. The results are consistent with the ones in Figure 6.2-16 showing that larger amount of MA (4% of Am) may further reduce the inventory in the cycle (to ca. 70 tonnes in the year 200).

If different configurations are considered the amount of MA in the cycle in the year 200 shows a large spread as indicated in Figure 6.2-18.

Figure 6.2-16: MA inventory in cycle (EFR case – initial MA content)

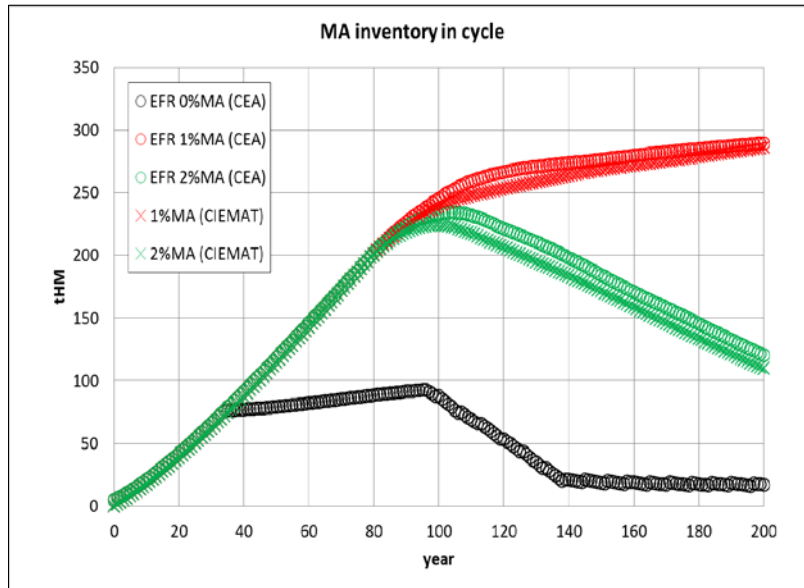


Figure 6.2-17: MA inventory in cycle (ESFR WH case – initial MA content)

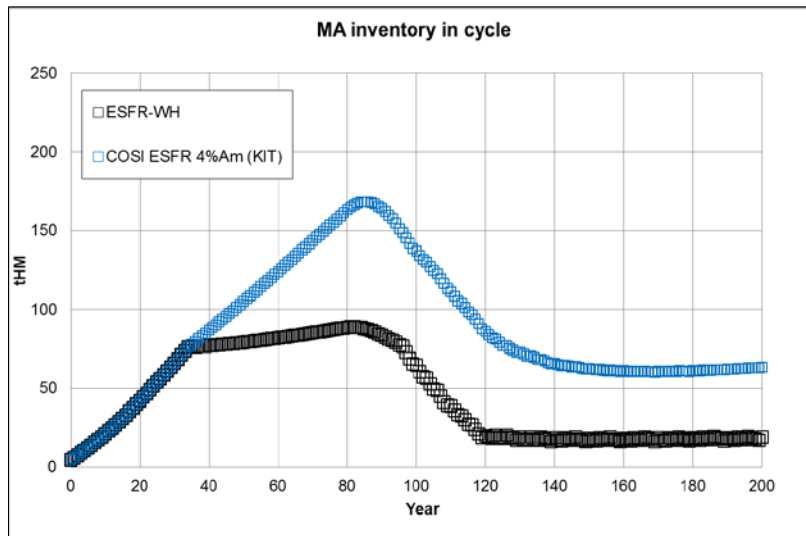
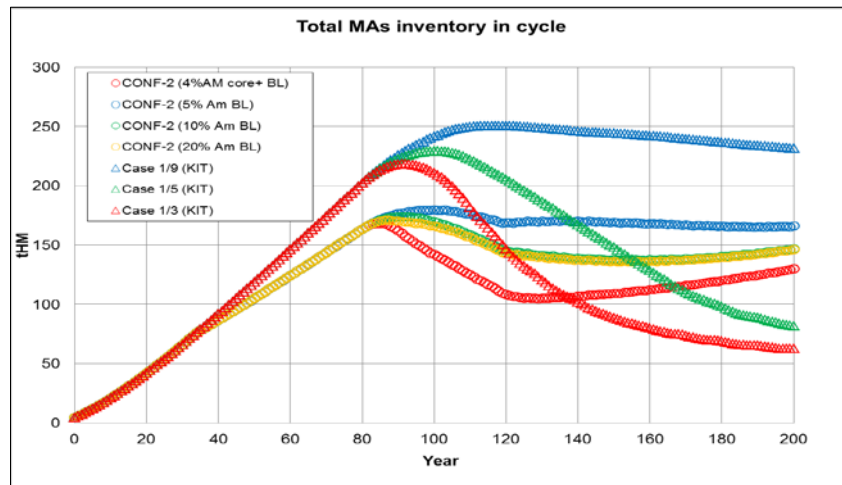


Figure 6.2-18: MA inventory in cycle (ESFR CONF2 and ESFR+burner cases – initial MA content)



6.2.3. Breeding ratio

As already indicated in Section 6.2.1, different SFR designs have been considered in the study. More details are indicated in Appendix A.

- (1) The reference SFR model (EFR concept) as described in chapter 2 and it is characterised by a breeding ratio (BR) close to 1 (label in figures as “*BR=1 REF (CEA)*”).
- (2) The ESFR WH concept (core with BR ~ 1.02 , without blankets) has been considered by KIT (label in figures as “*BR=1.02 ESFR WH (KIT)*”). A variation of this configuration is considered by CIEMAT (label in figures as “*BR=1.034 WH(*) (CIEMAT)*”). The CIEMAT configuration is characterised by the same geometry as the original SFR design (Chapter 2) but with different cross-section libraries associated to the core zone. The cross-section libraries refer to ESFR WH core. In the CIEMAT configuration axial blankets are maintained.
- (3) The SFR-V2B concept (core with BR equal to 1, without blankets) has been considered by CEA (label in figures as “*BR=1 SFR-V2B (CEA)*”). This core is very similar to the ESFR WH systems.
- (4) The ESFR CONF-2 concept (core with BR ~ 1 plus lower fertile blankets for a total BR closed to 1.2) has been considered by KIT (“*BR=1.2 ESFR CONF2 (KIT)*”). A variation of this configuration is considered by CIEMAT. The CIEMAT configuration is characterised by the same geometry as the original SFR design (chapter 2) but with different cross-section libraries associated to the core zone. The cross-section libraries refer to ESFR CONF2 core. In the CIEMAT configuration axial blankets are maintained. These configurations are characterised by a slightly increase of the BR compared to the WH core, label in the figures as “*BR=1.047 CONF2 (*) (CIEMAT)*”.
- (5) A case with reduced BR derived from the EFR system by removing the blankets have been considered by CIEMAT. In particular, the case called “*BR=0.91 REF (*) NO BL (CIEMAT)*” corresponds to a core with BR equal to 0.91. The adoption of smaller BR implies less quantity of Pu in the cycle and the possibility of not having enough material for completing the scenario, as indicated in the following figures where around year 165 the scenario fails.

The burners considered in Section 6.2.2 as part of a mixed fleet have not been considered for the BR case, as already indicated.

Also for the BR cases, larger impacts are on quantities related to the fuel cycle back-end and to the transition period from PWR to SFRs fleet.

The SFR fuel fabrication needs (fissile and fertile) are shown in Figures 6.2-19 and 6.2-20, respectively. The ESFR WH and SFRv2B design shows the same trends (fissile needs stabilise to around 450 tonnes/year and there is no need of fertile fuel). The fissile needs for CONF2 are in agreement with the ESFR WH, as expected because the fissile core zone of the two systems is fully comparable. However, results provided by CIEMAT (including WH and CONF2 cross-section libraries cases) are in agreement with the reference scenario based on EFR. For the case with $BR < 1$ the scenario stops around year 165 (Pu no longer available in the cycle as indicated in Figure 6.2-21).

The Pu flow for fabrication is shown in Figure 6.2-22. EFR based systems stabilise to ca. 60 tonnes/year while new models (ESFR WH and SFRv2B) stabilise to ca. 70 tonnes/year. The large quantity of Pu needed for loading the core has an impact on the spent fuel storage. For new systems, indeed, the PWR UOX spent fuels storage is emptied earlier compared to the reference case (Figure 6.2-23). This behaviour is also due to an increased PWR UOX spent fuels reprocessing capacity, which is needed to use the Pu in the cycle during the SFR start-up as shown in Figure 6.2-24.

Figure 6.2-25 shows the SFR spent fuels storage. Large differences may be underlined. In particular, the adoption of ESFR CONF2 leads to a larger amount of fuel in storage at the end of the scenario. This behaviour is mainly due to the better quality Pu produced by CONF2 with respect to ESFR WH and to the presence of the axial blankets.

For systems considered, the SFR spent fuels reprocessed stabilised to the same value (ca. 450 tonnes/year) as indicated in Figure 6.2-26. During the transition period from PWR to SFR fleet larger variations are visible in Figure 6.2-26.

Figure 6.2-19: FR fissile fuel fabrication needs (BR variation)

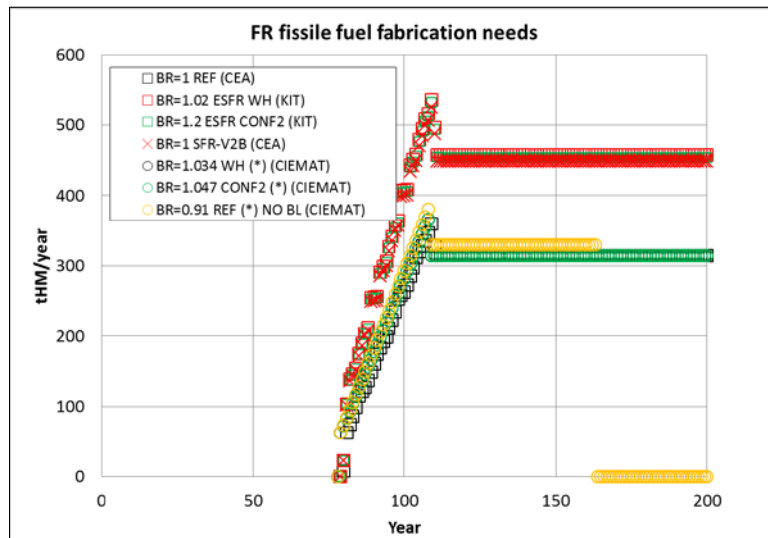


Figure 6.2-20: FR fertile fuel fabrication needs (BR variation)

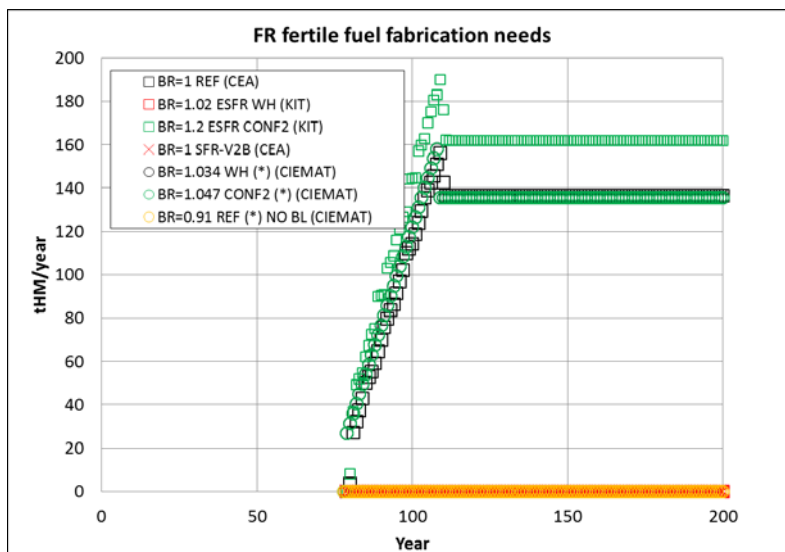


Figure 6.2-21: Pu in interim storage (BR variation)

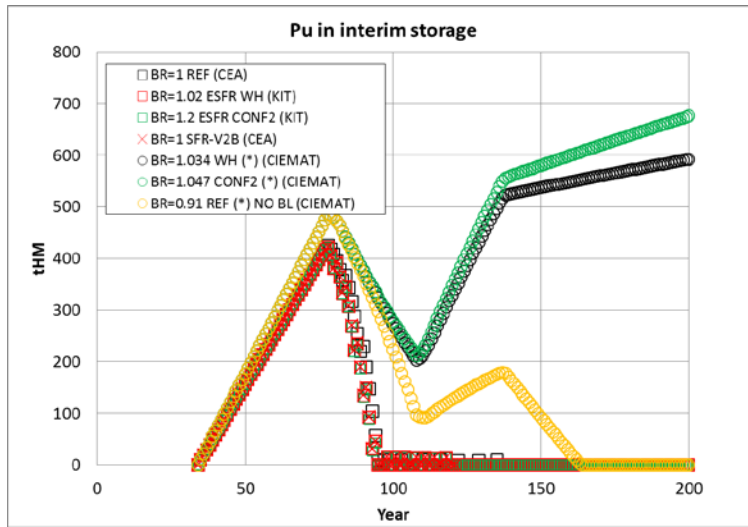


Figure 6.2-22: Pu flow for fabrication (BR variation)

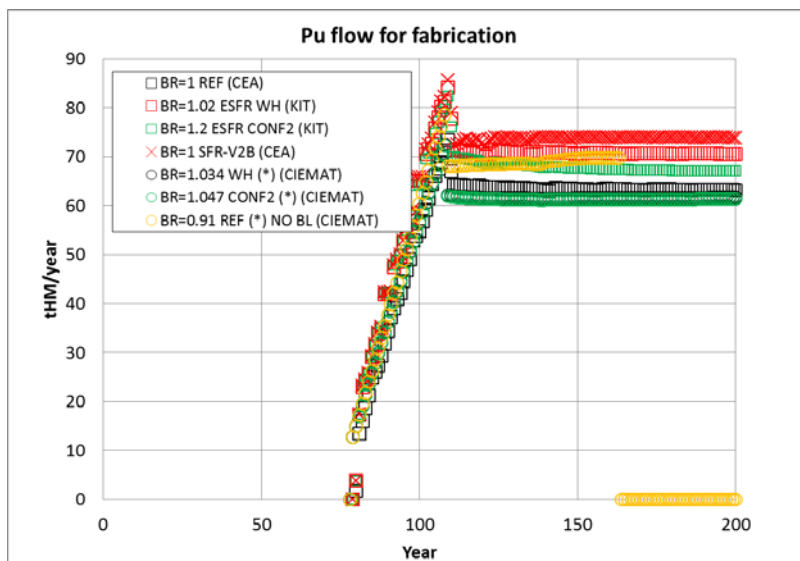


Figure 6.2-23: PWR UOX spent fuels storage (BR variation)

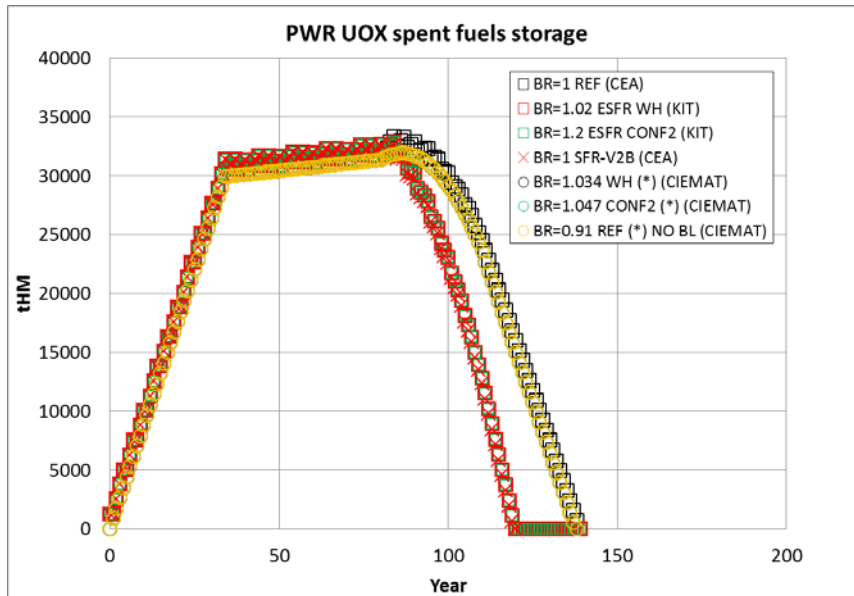


Figure 6.2-24: PWR UOX spent fuels reprocessed (BR variation)

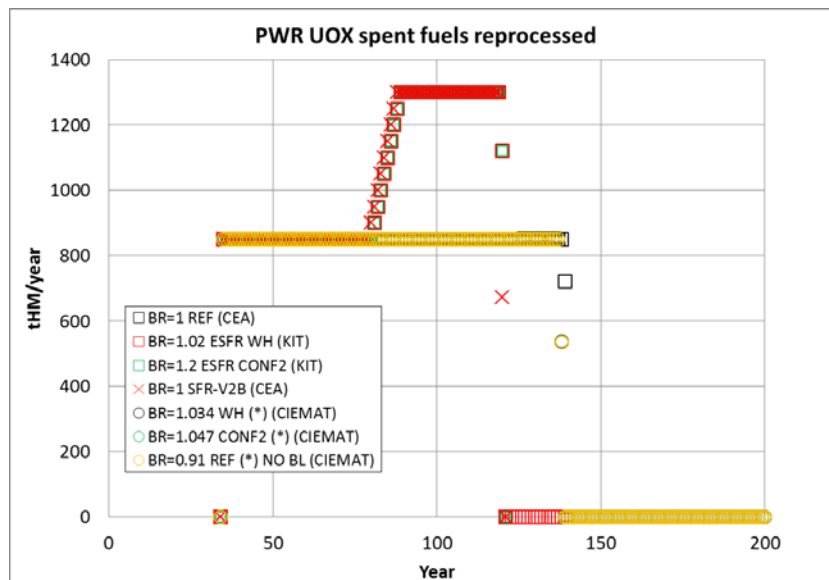


Figure 6.2-25: FR spent fuels storage (BR variation)

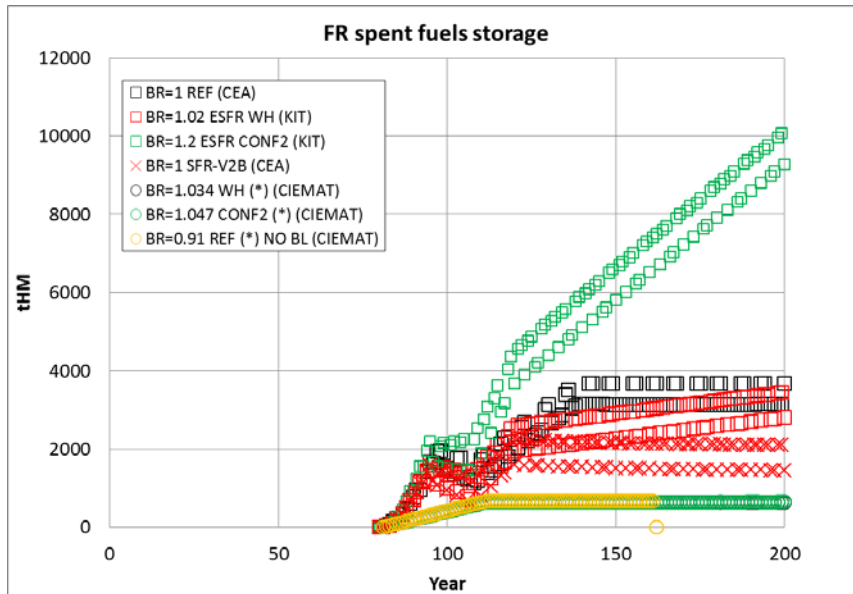
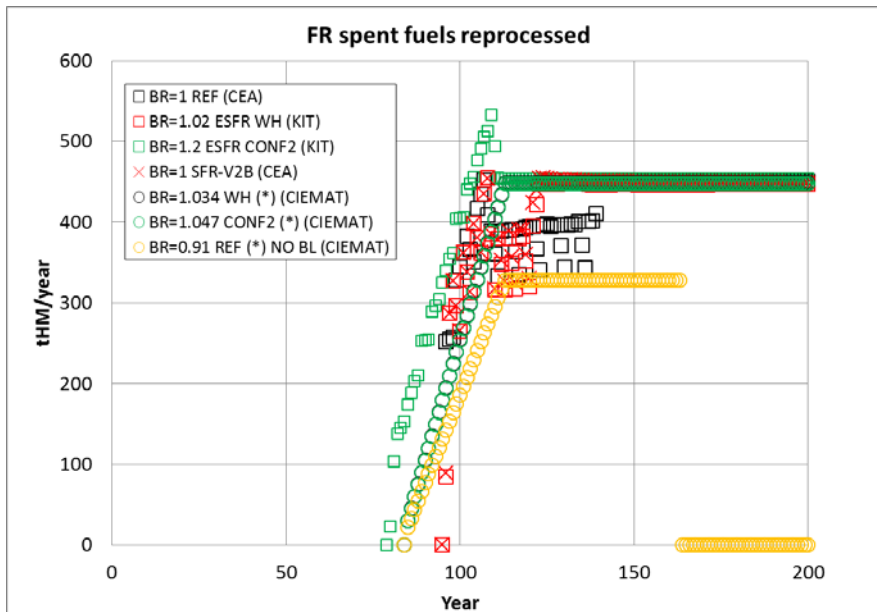


Figure 6.2-26: FR spent fuels reprocessed (BR variation)



Concerning the reprocessed U interim storage (see Figure 6.2-27) all systems follow the same behaviour except the ESFR CONF2 case (KIT). This difference comes from the fertile blankets.

The Pu inventory in NPPs (see Figure 6.2-28) looks similar to the SFR fissile fuel fabrication needs (see Figure 6.2-19). Old designs stabilise to ca. 380 tonnes while new designs (ESFR, SFRv2B) stabilise to larger values (ca. 500 -520 tonnes).

The Pu inventory in storage (see Figure 6.2-29) increases by increasing the BR as indicated by the CIEMAT cases (WH versus CONF2) and by the KIT cases (WH versus CONF2). In cases considered by KIT, where the BR variation is quite high, the Pu inventory in storage at the end of the scenario (year 200) is almost double when compared with the value obtained for ESFR WH. The ESFR WH models considered either by KIT and CIEMAT are slightly breeder (BR equal to ca. 1.02 for KIT and ca. 1.034 for CIEMAT). The two systems follow the same trends (slope) as indicated in Figure 6.2-29 while the SFRv2B has a BR=1 as also visible in Figure 6.2-29. The Pu in the cycle (Figure 6.2-30) reflects somewhat the results discussed for the Pu inventory in storage. For the case with BR < 1 it is visible that around year 165 there is not enough Pu for loading the systems and the scenario is ended.

From increasing the BR, the Pu inventory in waste is reduced (see Figure 6.2-31) from 25 tonnes at year 200 in the reference scenario case to ca. 20 tonnes for CONF2. This behaviour is due to the better quality Pu produced by CONF2 with respect to the reference scenario.

Concerning MA inventory in waste, the different SFR designs do not change the figure of merit as indicated in Figure 6.2-32. The difference between COSI and TR_EVOL codes was already underlined for the reference scenario (see Chapter 3).

Figure 6.2-27: Reprocessed U interim storage (BR variation)

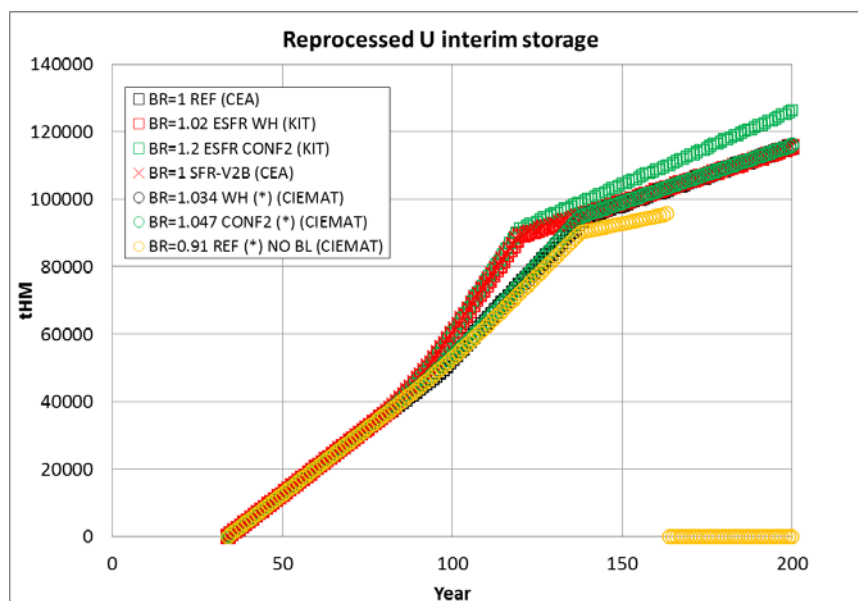


Figure 6.2-28: Pu inventory in NPPs (BR variation)

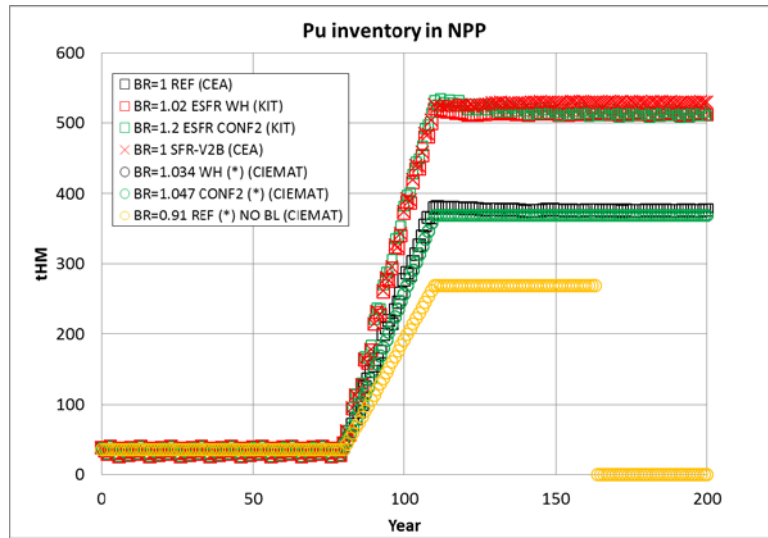


Figure 6.2-29: Pu inventory in storage (BR variation)

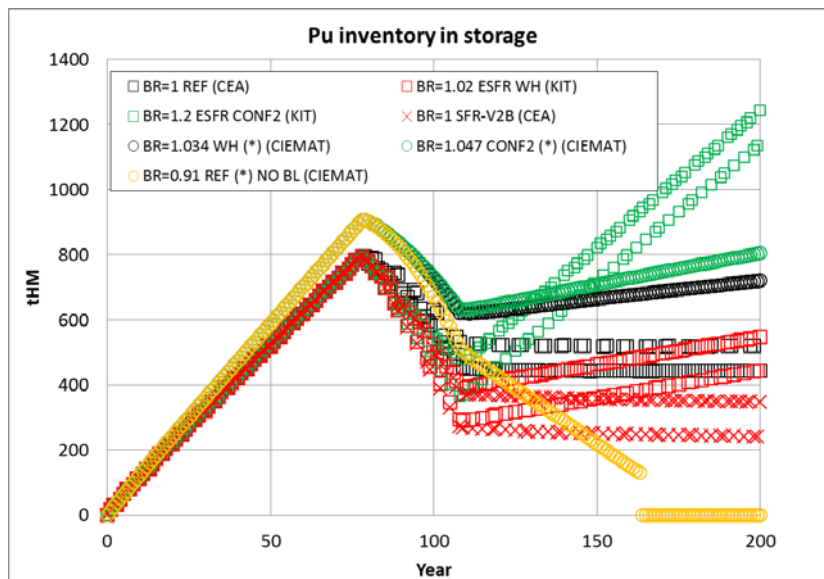


Figure 6.2-30: Pu inventory in cycle (BR variation)

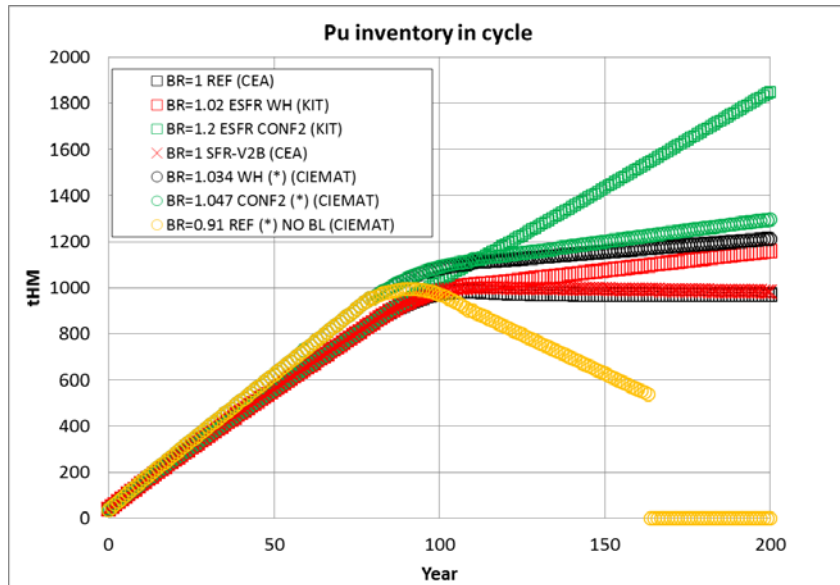


Figure 6.2-31: Pu inventory in waste (BR variation)

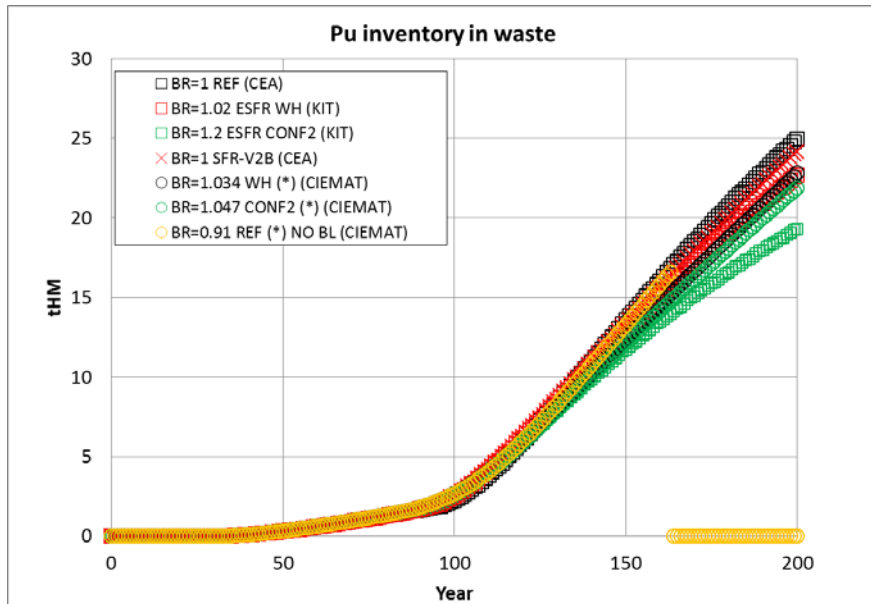
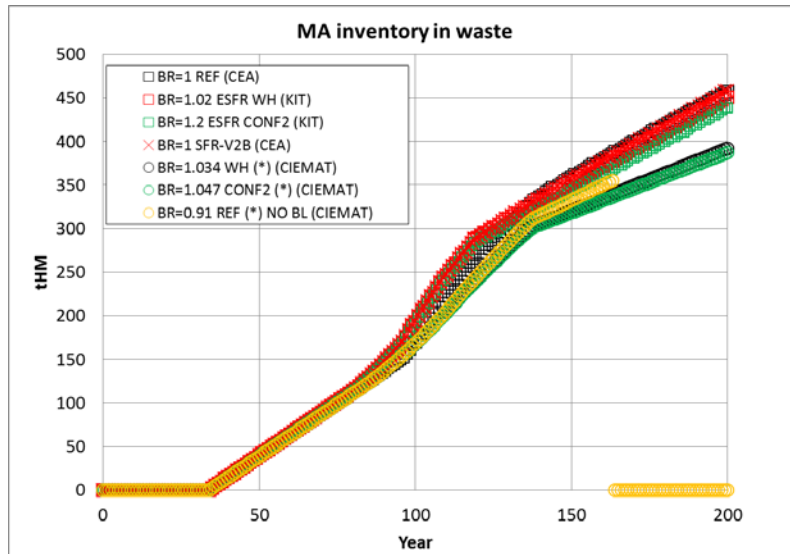


Figure 6.2-32: MA inventory in waste (BR variation)



6.3. Impact of the Reactor Lifetime

This chapter describes the impact of changing the reactor lifetime (e.g. 40 years or 60 years instead of infinite lifetime) on the results of the code. The code here chosen is COSAC as long as it is representative of the other codes. When it is not, the COSAC results are compared to the results from another code, COSI, in order to complete the description of the impact of changing the reactor lifetime. The detailed explanation of the different strategies used in COSAC and in the other codes like COSI can found in Appendix B.

6.3.1. Impact on the natural uranium consumption

Figure 6.3-1: Impact on the natural U consumption

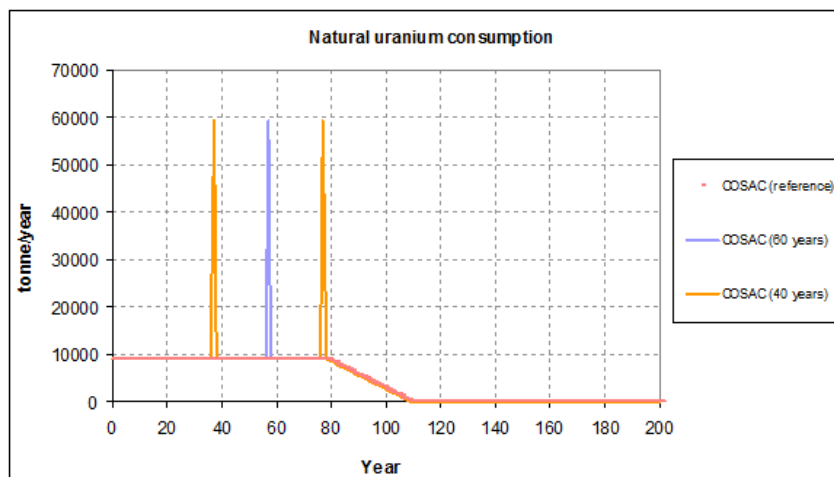
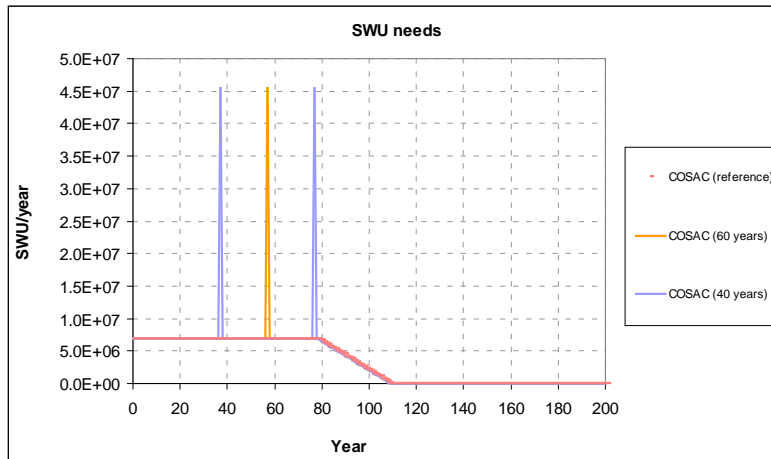


Figure 6.3-1 exhibits a peak of the natural uranium consumption every 40 years (resp. 60 years).

Since all PWRs are commissioned together in the first year of the scenario, the impact of limiting the reactor lifetime at 40 years (resp. 60 years) is the repetition of the initial investment of natural uranium mass every 40 years (resp. 60 years) to be loaded in the first cores of all the PWRs at the same time.

6.3.2. Impact on the enrichment needs

Figure 6.3-2: Impact on the enrichment needs

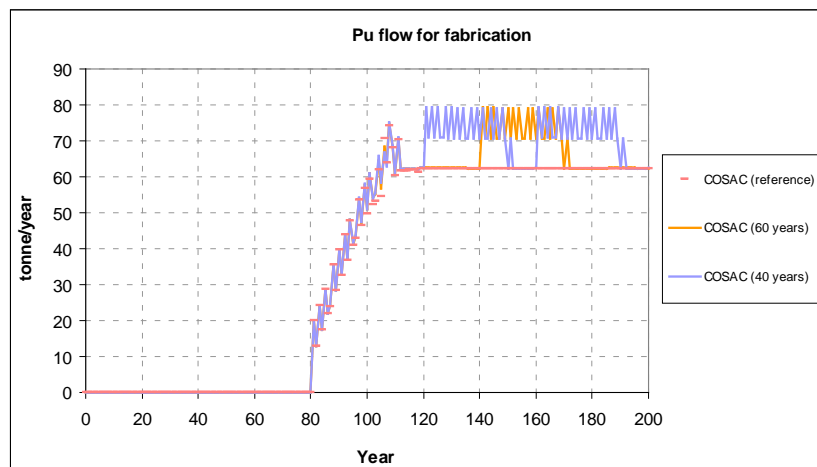


A peak for enrichment needs (see Figure 6.3-2) is observed every 40 years (resp. 60 years) as previously seen for natural uranium consumption.

The enrichment of the natural uranium is achieved two years after the uranium extraction. Therefore enrichment needs follow the natural uranium consumption with a time shift of two years.

6.3.3. Impact on the plutonium flow for fabrication

Figure 6.3-3: Impact on the Pu flow for fabrication

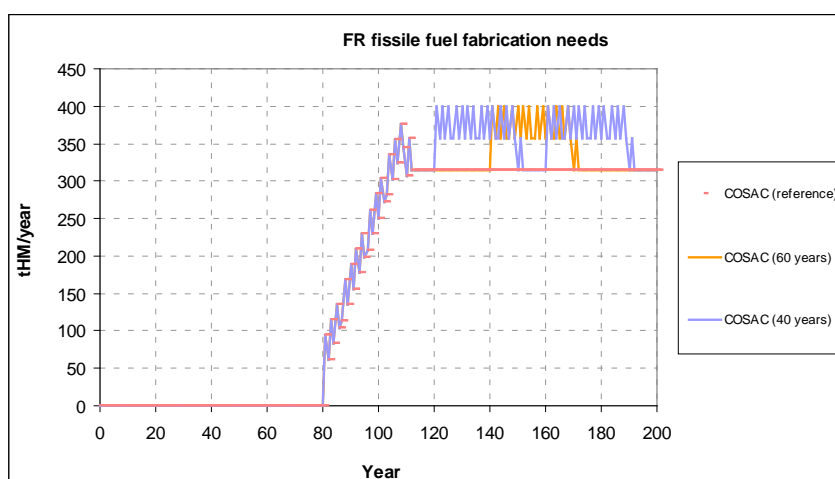


The plutonium flow for fabrication (see Figure 6.3-3) increases every 40 years (resp. 60 year) compared to the reference scenario. In both cases (40 years and 60 years), this increase in plutonium flow for fabrication lasts 30 years, that is, the construction time for the new FR fleet. The increased values alternatively reach a maximum and a minimum value. Compared to the reference value, the maximum value is twice higher than the minimum value.

Since FRs are first commissioned between years 80 and 110 (over 30 years), their replacement every 40 years (resp. 60 years) lasts 30 years. This replacement needs an increased amount of plutonium to be loaded in the first cores every 40 years (resp. 60 years). As the commissioning of the new FRs occurs every 8.5 months (43 FRs to be commissioned in 30 years = one FR commissioned every 8.5 months approximately), two FRs are sometimes commissioned during the same year, whereas one FR is commissioned the other years. This explains why the increased value of plutonium flow may reach either a maximum or a minimum value at a given year. The maximum value represents the amount of plutonium to be loaded in two first cores at the same year, and the minimum value represents the amount of plutonium to be loaded in one first core at another year.

6.3.4. Impact on the fuel fabrication flows

Figure 6.3-4: Impact on the fuel fabrication flows



The same observation can be done for the PWR fuel (resp. FR fuel) fabrication flow as for the natural uranium consumption (resp. the plutonium flow for fabrication).

The fuel fabrication flow for PWRs follows the natural uranium consumption after a lap of time of two years. The same lap of time of two years is applied to the fuel fabrication for FRs after plutonium retrieval from the reprocessing, so that the fuel fabrication flow for FRs follows the plutonium flow for fabrication.

6.3.5. Impact on the spent fuel reprocessing

Figure 6.3-5: Impact on PWR UOX spent fuel reprocessing

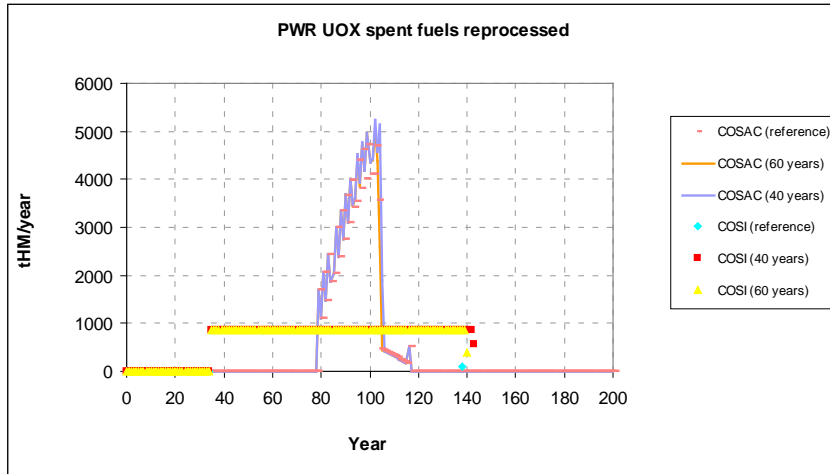
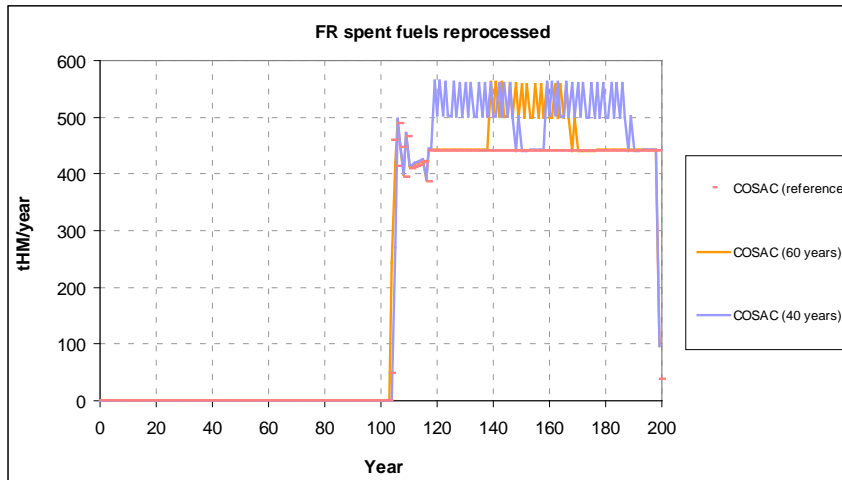


Figure 6.3-6: Impact on FR spent fuels reprocessing



There are two series of curves displayed for reprocessed PWR UOX spent fuels (Figure 6.3-5), one for the COSI calculations that are representative of steady reprocessing capacity (here, 850 tonnes of heavy metal per year), and another for COSAC calculations that are representative of reprocessing capacity following the Pu need at fabrication without anticipation. More details between the two different strategies about reprocessing that are used in COSAC and the other codes like COSI can be found in Appendix B. For both series of curves, the same observation can be made: the curve for reprocessed PWR UOX spent fuels in the case of a reactor lifetime of 40 years (resp. 60 years) is apparently the same as for the reference scenario. For the reprocessed FR spent fuels (Figure 6.3-6), the curve is also apparently the same in the case of a reactor lifetime of 40 years (resp. 60 years) as for the reference scenario, except for the very beginning of the curves (years 103, 104 and 105)

where the various cases differ. On the contrary, the dismantling of FRs after 40 years (resp. 60 years) of operation has a large impact on the FR spent fuel reprocessing: we can observe the same variations between a max value and a min value of the spent fuel reprocessing as for the plutonium flow for fabrication.

Despite a higher amount of PWR UOX spent fuels produced by the dismantling of PWRs after 40 years (resp. 60 years) of operation, the impact on the amount of plutonium available for being loaded in FRs after reprocessing is small. In the case of a reactor lifetime of 40 years (resp. 60 years), the extra amount of PWR UOX spent fuel produced by the dismantling of PWRs is of about 9,200 (resp. 4,600) tonnes of heavy metal, but in the case of COSI this extra amount only allows to carry on the PWR UOX spent fuel reprocessing for five years, i.e. from year 138 to year 143 (resp. for two years, i.e. from year 138 to year 140), and in the case of COSAC calculations it only allows to postpone the beginning of the FR spent fuel reprocessing by one year, i.e. from year 103 to year 104 (resp. two years, i.e. from year 103 to year 105).

This small impact on the date of the beginning of FR spent fuel reprocessing is especially due to the low content of plutonium (around 1%) in the PWR UOX spent fuel. On the contrary, when the FRs are renewed, there is no more PWR UOX spent fuel to be reprocessed, and the reprocessing of FR spent fuel is fully driven by the newly commissioned FRs that need to be first loaded. Therefore the FR spent fuel reprocessing is following the same curve as the plutonium flow for fabrication.

6.3.6. Impact on the spent fuel storage

Figure 6.3-7: Impact on PWR UOX spent fuels storage

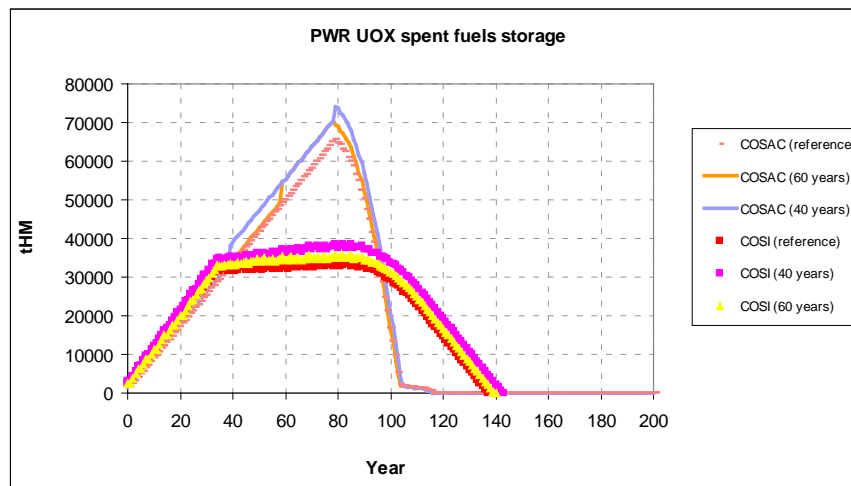
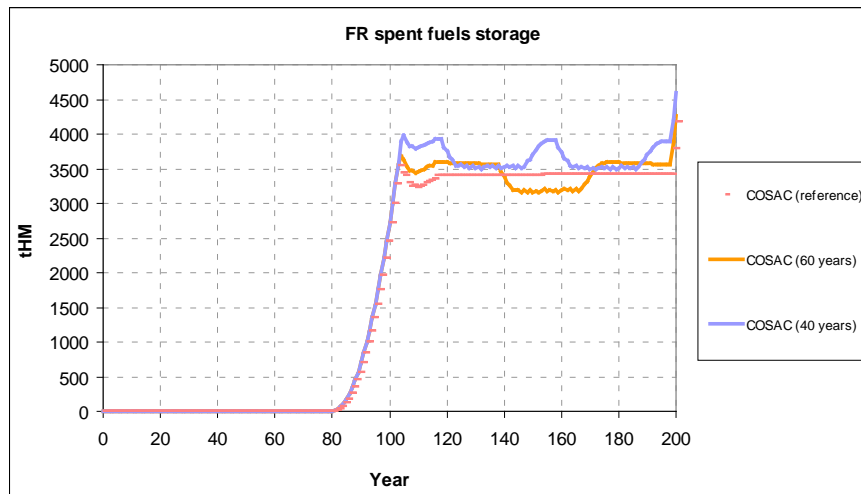


Figure 6.3-8: Impact on FR spent fuels storage

Here again, there are two series of curves displayed for PWR UOX spent fuel storage (Figure 6.3-7), one for COSI calculations that are representative of steady reprocessing capacity (here, 850 tonnes of heavy metal per year), and another for COSAC calculations that are representative of reprocessing capacity following the Pu need at fabrication without anticipation. Please refer to Appendix B to find more details about the impacts onto the spent fuel storages induced by the two different strategies about reprocessing that are used in COSAC and the other codes like COSI. For the codes such as COSI with steady reprocessing capacity, the PWR UOX spent fuel storage is cancelled all the later since the reactor lifetime is shorter. For the codes such as COSAC that do not consider spent fuel reprocessing before the need to use plutonium occurs, the value of the PWR spent fuel storage is increased every 40 years (resp. 60 years) that is each time the PWR fleet is renewed. For the FR spent fuel storage, the value can either be increased or decreased compared with the reference scenario, depending on the reactor lifetime (40 years or 60 years) and on the period of the scenario where the comparison is done.

When the PWR fleet is renewed at year 40 (resp. at year 60), all the old PWRs are decommissioned and their cores are fully discharged. The result is an increase of the PWR UOX spent fuel storage. For codes such as COSI, the extra amount of PWR spent fuel in the storage makes the time when the storage is emptied postponed for five years (resp. two years), because the reprocessing capacity is the same whatever the amount to be reprocessed. For codes such as COSAC, the extra amount of PWR spent fuel in the storage remains unused until the first FRs are commissioned, that is in year 80, because COSAC doesn't consider PWR spent fuel reprocessing before plutonium is needed by FRs. Once the FRs are deployed, the extra amount of PWR UOX spent fuel in the storage is quickly reprocessed, and the time when the storage is emptied is practically the same whatever the reactor lifetime is. In the particular case of a reactor lifetime of 40 years, the second renewal of the PWR fleet coincides (at year 80) with the beginning of the replacement of PWRs with FRs. In this case, the extra amount of PWR spent fuel storage produced by the second renewal of the PWR fleet is very soon cancelled because PWR spent fuel is reprocessed to provide plutonium to the newly commissioned FRs. That is the reason why the shifted curve

of the PWR spent fuel storage in the case of a reactor lifetime of 40 years does not differ a lot after the year 80 from the case of a reactor lifetime of 60 years.

For the FR spent fuel storage (Figure 6.3-8), there is a competition between the extra amount of spent fuel that is produced each time the FR fleet is decommissioned (i.e. every 40 years or 60 years), and the need in plutonium that is required to be loaded in the first cores of the newly commissioned FRs. In the case of a reactor lifetime of 40 years, the extra amount of PWR spent fuel produced by the two renewals of the PWR fleet at year 40 and year 80 is enough to make the need in FR spent fuel reprocessing always lower than the extra amount produced by the FR fleet renewal. In this case, the curve of the FR spent fuel storage is then always higher than the one for the reference scenario. On the contrary, in the case of a reactor lifetime of 60 years, the need in FR spent fuel reprocessing to provide plutonium to the newly commissioned FRs is sometimes higher sometimes lower than the extra amount produced by the FR fleet renewal. In this case, the curve of the FR spent fuel storage is sometimes higher sometimes lower than the one for the reference scenario.

6.3.6. Summary of all the impacts: Sensitivity coefficients

Table 6.3-1 below shows the sensitivity coefficients “S” as defined in Chapter 4.3 and applied to the input parameter “Reactor lifetime”.

Table 6.3-1: Sensitivity coefficient “S” applied to the input parameter “Reactor lifetime”

S ($r^2 \geq 0,9$)	PWR cycle				FR cycle				Reprocessing		Inventory						
	Front-end			Back-end	Front-end		Back-end	PWR	FR	Pu						MA	
	Uranium	Enrichment	Fabrication	Storage	Pu	Fabrication	Storage			Plants	Reactors	Storages	Cycle	Waste	Cycle	Waste	
Reactor characteristics																	
Reactor Lifetime							?										

All the material flows are impacted by a change in reactor lifetime, but each impact is quite small: the colours are mostly pale. The most significant impact is on the plutonium inventory in the storages, which appears with a blue colour a bit darker than the other ones. This is due to an increased amount of spent fuel in the pools and interim storages as decommissioning are more frequent when the reactor lifetime is shorter.

Generally speaking, all the coefficients are negative (blue), which expresses the fact that, when the reactor lifetime is shorter and the commissioning/decommissioning are more frequent, then more fresh fuels are manufactured, more spent fuels are reprocessed, and more waste is stored. The question mark “?” present in the table indicates a coefficient of determination “ r^2 ” lower than 0.9. Blanks present in the table indicate the output parameters are not impacted by a change in the reactor lifetime, so that the related sensitivity indicators are not available.

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- [1] G.L. Fiorini (2009), "The Collaborative Project on European Sodium Fast Reactor (CP ESFR)", *Proceedings of the International Conference on Euratom Research and Training in Reactor Systems (FISA2009)*, Prague, Czech Republic, 22–24 June 2009.
- [2] B. Vezzoni, F. Gabrielli, et al. (2012), "Safety-Related Optimisation and Analyses of an Innovative Fast Reactor Concept", *Sustainability* 2012, 4, 1274-1291; doi:10.3390/su4061274.
- [3] F. Gabrielli et al. (2013), "ASTRID-like Fast Reactor Cores for Burning Plutonium and Minor Actinides", *Proceedings of the International Conference, INES-4*, Tokyo (Japan), 6-8 November 2013.
- [4] B. Vezzoni, F. Gabrielli, et al. (2014), "Innovative TRU burners and fuel cycles options for phase-out and regional scenarios", *Proceedings of the 13th Information Exchange on Actinide and Fission (IEMPT)*, OECD/NEA, Seoul, Republic of Korea, 23-26 September 2014.

7. Effects of Uncertainties on Fuel Cycle Facilities

7.1. First year of reprocessing plant operation

The results presented in this section were calculated with the VISION code (CNL) unless stated otherwise.

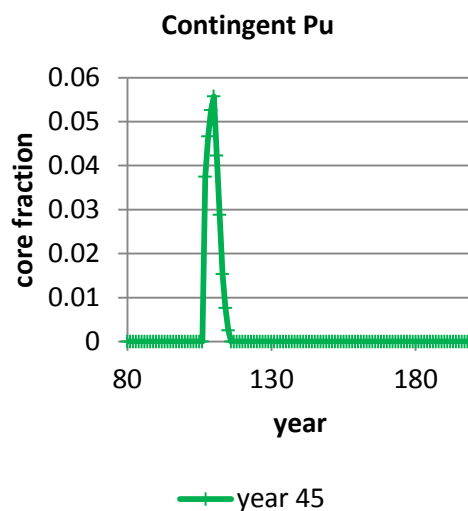
7.1.1. PWR UOX reprocessing plant

Low	Reference	High
year 25	year 35	year 45

In the reference scenario the PWR UOX reprocessing plant begins operation at year 35 to allow enough plutonium to be separated in order to fuel the transition from a fleet of PWRs to a fleet of FRs.

Delaying the first year of reprocessing PWR UOX spent fuel to year 45 results in an insufficient amount of separated plutonium for completing the transition to FRs. In this case, up to 6% of the plutonium in FR fuel must come from contingent plutonium reserves between years 107 and 115, as shown in Figure 7.1-1. This results in an increase of the inventory of plutonium in the cycle.

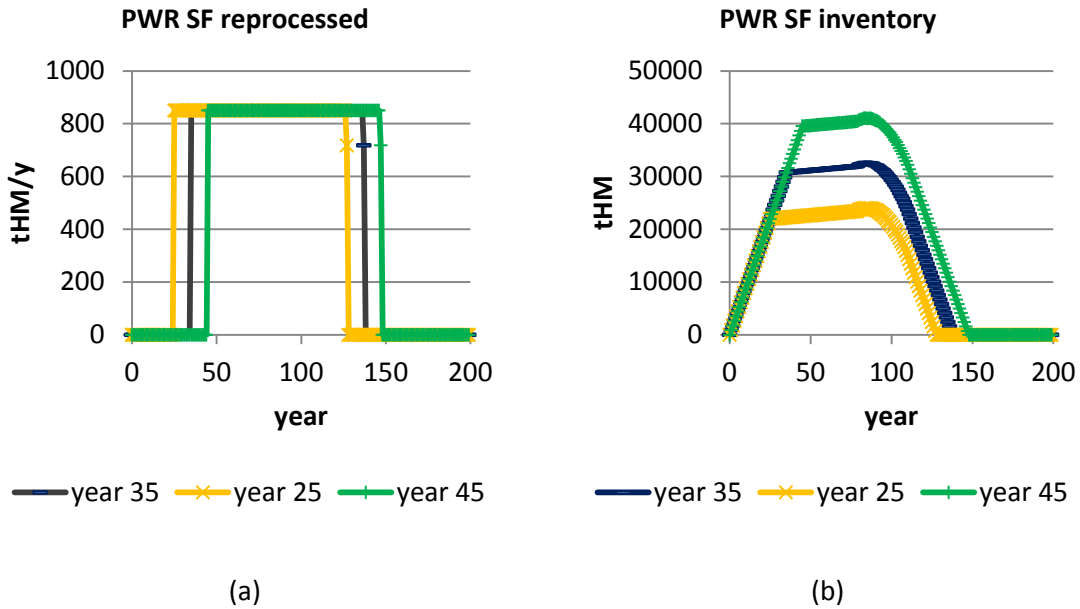
Figure 7.1-1: The use of contingent Pu



Although the same amount of spent PWR UOX fuel is reprocessed regardless of the year that reprocessing begins, Figure 7.1-2a shows that the PWR UOX spent fuel reprocessed shifts backward and forward in time for the low and high reprocessing start years, respectively.

When the first year of reprocessing is changed to year 25 the amount of separated material in storage is higher and the amount of PWR spent fuel in storage is lower when compared to the reference scenario until all of the PWR spent fuel has been reprocessed. Conversely, delaying the first year of reprocessing to year 45 decreases the amount of separated material in storage and increases the PWR spent fuel in storage. These relationships are also shown in Figure 7.1-2b and Figure 7.1-4. Note that starting the reprocessing plant in year 45 results in more separated plutonium in storage at the end of the scenario due to the contingent plutonium that is needed to complete the transition to FRs.

Figure 7.1-2: PWR UOX spent fuel (a) reprocessed and (b) in storage



If as much of the FR MOX spent fuel is reprocessed as the capacity of the reprocessing plant allows, as is the case with VISION (CNL), then changing the first year of reprocessing PWR UOX spent fuel has no effect on the reprocessing of FR spent fuel. If the FR spent fuel is reprocessed on demand then the amount of FR spent fuel that is reprocessed in a given year depends on the amount of separated plutonium in interim storage, which depends on the first year of reprocessing of PWR spent fuel. The effects of changing the first year of reprocessing PWR spent fuel on the reprocessing of FR spent fuel and on the FR spent fuel in storage are shown in Figure 7.1-3. Note that the results shown in this figure are from COSI (CEA). Advancing the first year of reprocessing PWR spent fuel to the year 25 results in a larger inventory of separated plutonium at the time that the reprocessing of FR spent fuel would begin if the reprocessing of PWR spent fuel began in the year 35 (i.e., the reference

case). This causes a temporary decrease in the amount of FR spent fuel that is reprocessed and a corresponding increase in the amount of FR spent fuel in storage until all of the PWR spent fuel has been reprocessed. Once all of the PWR spent fuel has been reprocessed then the reprocessing of FR spent fuel becomes much higher than in the year 35 case between the years 130 and 139, after which the amount of FR spent fuel that is reprocessed in the year 35 case becomes nearly equal to that of the year 25 case. Conversely, delaying the first year of reprocessing PWR spent fuel to the year 45 results in a temporary increase in the amount of FR spent fuel that is reprocessed, followed by a decrease in reprocessed FR spent fuel in the year 45 case versus the year 35 case, prior to these cases becoming nearly equal for the remainder of the scenario. There is more FR spent fuel in storage by the end of the scenario in the year 45 case than the year 35 case because of the contingent plutonium that is required in the year 45 case.

Figure 7.1-3: FR MOX spent fuel (a) reprocessed and (b) in storage, assuming FR spent fuel is reprocessed on demand (Courtesy of CEA)

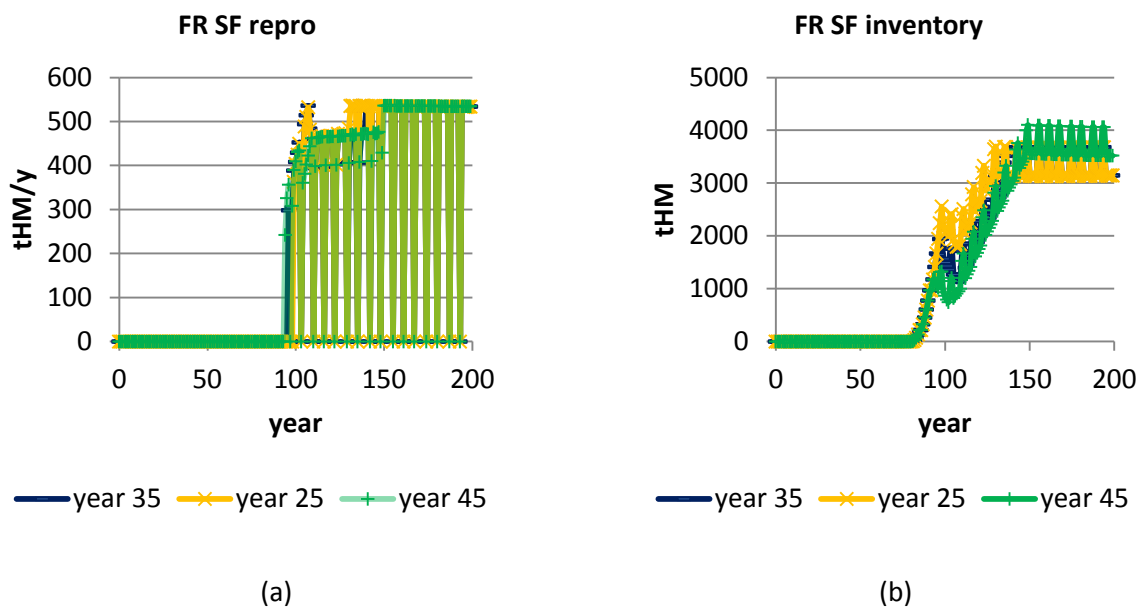
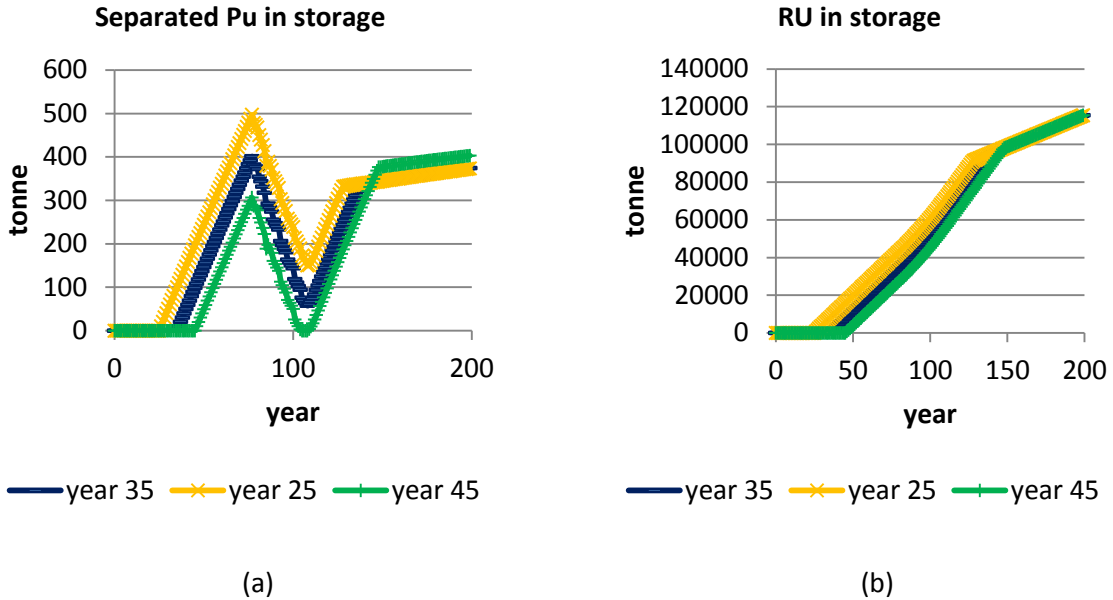
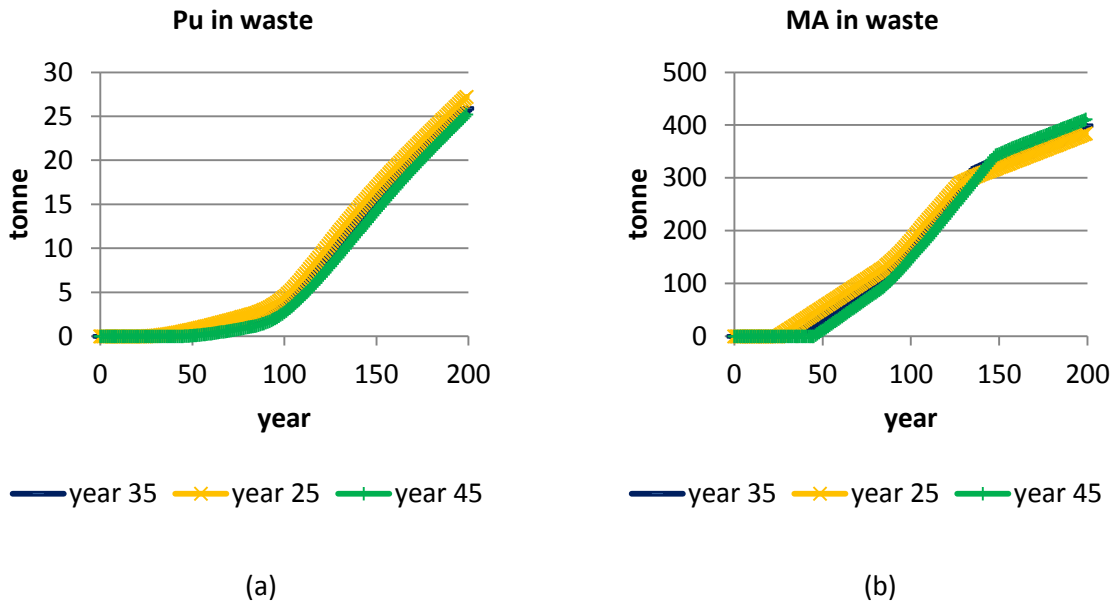


Figure 7.1-4: Inventory of (a) separated Pu and (b) reprocessed U



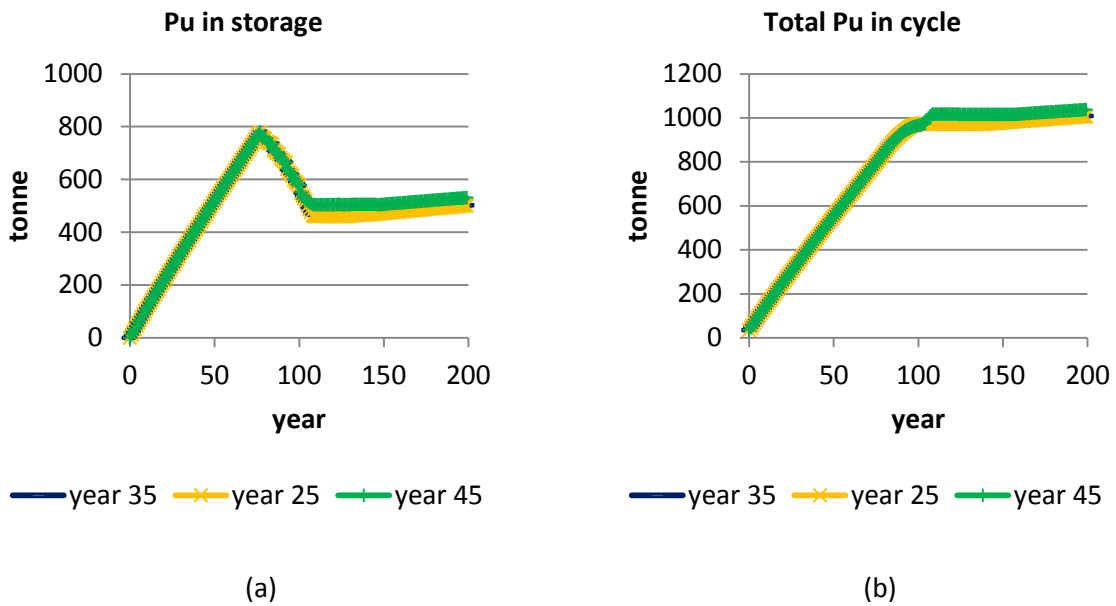
The effects of the first year of reprocessing PWR spent fuel on the inventory of plutonium and MA in the waste are shown in Figure 7.1-5. The first reprocessing year affects the amount of plutonium and MA in the waste in two ways. First, changing the first reprocessing year affects the amount of spent fuel that has been reprocessed prior to each year afterward until all of the PWR spent fuel has been reprocessed. In the case of advancing the first year of reprocessing, the inventory of plutonium and MA waste is higher than in the reference case between the years 25 and 140. In the case of delaying the first year of reprocessing, the inventory of plutonium and MA waste is lower than in the reference case between the years 35 and 144. After the PWR spent fuel has all been reprocessed, the effect of the first year of reprocessing on the amount of plutonium and MA in the waste is mainly due to the decay of ^{241}Pu to ^{241}Am . Advancing the first reprocessing year reduces the amount of ^{241}Pu that decays prior to reprocessing, and hence increases the amount of plutonium and decreases the amount of MA in waste. Conversely, delaying the first reprocessing year decreases the plutonium and increases the MA in waste.

Figure 7.1-5: The (a) Pu and (b) MA inventory in reprocessing waste



The inventory of plutonium in storage and in the entire fuel cycle is mainly affected by the amount of contingent plutonium that is added to the fuel cycle. Figure 7.1-6 shows that in the case of delaying the first year of reprocessing, adding contingent plutonium between the years 107 and 115 results in higher plutonium inventories for the remainder of the scenario.

Figure 7.1-6: The Pu inventory in (a) storage and (b) in the fuel cycle



The plutonium flow to fabrication and the plutonium inventory in fabrication and nuclear power plants (Figure 7.1-7) are affected by the source of plutonium, where FR fuel, comprised of plutonium from PWR spent fuel, has a larger proportion of plutonium than FR fuel comprised of plutonium from FR spent fuel. Since plutonium from reprocessed PWR spent fuel is used with a higher priority than plutonium from FR spent fuel, the proportion of FR fuel from PWR plutonium depends on its availability. Advancing the first year of reprocessing PWR spent fuel allows the plutonium from PWR spent fuel to be fabricated into FR fuel sooner, resulting in a short-term increase in plutonium flow to fabrication during the transition, and a short-term decrease in the plutonium flow to fabrication after the transition is completed and the separated PWR plutonium is completely consumed. Delaying the start of reprocessing to the year 45 has the opposite effect, although contingent plutonium is required to complete the transition to FRs. Since the contingent plutonium is assumed to come from PWR spent fuel, an increase in plutonium flow to fabrication occurs when contingent plutonium is required. These effects can also be seen in the inventory of plutonium in the fabrication and nuclear power plants, which are shown in Figures 7.1-7b and 7.1-8.

It should be noted that the time to reprocess spent fuel is assumed to be zero, therefore there is no plutonium inventory in the reprocessing and fabrication plants prior to the year 78 in Figure 7.1-7b. If the time to reprocess spent fuel is not zero then there would be an inventory of plutonium in the reprocessing plant during its years of operation.

Figure 7.1-7: (a) Flow of Pu to fabrication and (b) Inventory of Pu in the fabrication and reprocessing plants

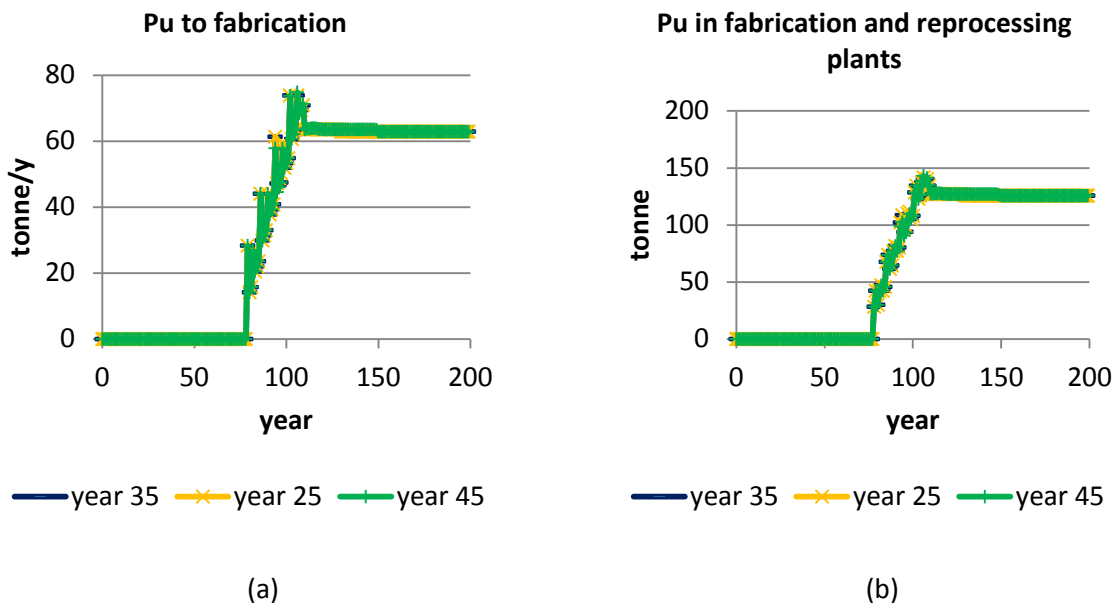
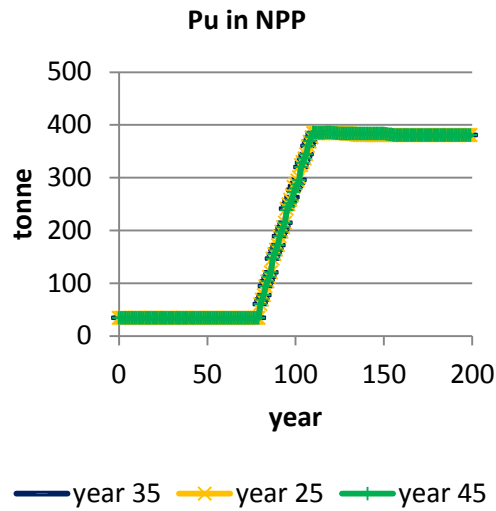


Figure 7.1-8: Pu inventory in the NPPs

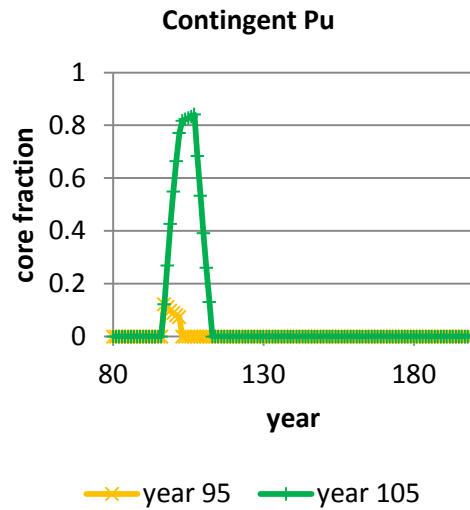
7.1.2. FR MOX reprocessing plant

Reference	Low	High
year 85	year 95	year 105

In the reference scenario, the FR MOX reprocessing plant begins operation at year 85 to help supply plutonium for the transition to FRs, and to enable the continued operation of FRs once the plutonium from PWR spent fuel has been depleted.

Delaying the first year of reprocessing FR MOX spent fuel to the year 95 results in an insufficient amount of separated plutonium for completing the transition to FRs. If the FR MOX reprocessing plant begins operation in the year 95 then up to 12% of the plutonium in FR fuel must come from contingent plutonium reserves, which is needed between the years 97 and 102 as shown in Figure 7.1-9. If the FR MOX reprocessing plant begins operation in the year 105 then up to 85% of the plutonium in FR fuel must come from contingent plutonium reserves, which is needed between the years 97 and 112.

Figure 7.1-9: The use of contingent Pu



Although the same amount of spent FR MOX fuel is reprocessed regardless of the year that reprocessing begins, Figure 7.1-10a shows that delaying the first year of reprocessing FR MOX spent fuel results in a number of years of the FR MOX reprocessing plant operating at full capacity before the reprocessing requirements drop to their reference values.

As was the case for delaying the first year of PWR UOX reprocessing, delaying the reprocessing of FR MOX spent fuel results in: a higher inventory of FR spent fuel in storage relative to the reference case (Figure 7.1-10b); and a lower inventory of separated plutonium until the transition to FRs is completed, after which it is higher than the reference case due to the addition of contingent plutonium to the fuel cycle (Figure 7.1-11). Note that delaying the first year of reprocessing by 20 years results in a gradual decrease in separated plutonium inventory after the year 139 until the end of the scenario. This decrease is caused by the additional decay of ^{242}Cm to ^{238}Pu and ^{244}Cm to ^{240}Pu prior to reprocessing, resulting in a higher proportion of ^{238}Pu and ^{240}Pu in the separated plutonium inventory, which in turn causes an increase in the amount of separated plutonium that is withdrawn from storage to fabricate FR fuel. If the reprocessing of FR spent fuel is on-demand then the impact of changing the first year of reprocessing FR spent fuel depends on the first year that FR spent fuel is required to be reprocessed. Figure 7.1-10a shows that in the reference case FR spent fuel is not required to be reprocessed until the year 96, therefore changing the first year of operation of the FR spent fuel reprocessing plant to the year 95 would have no effect on the results.

Figure 7.1-10: FR MOX spent fuel (a) reprocessed and (b) in storage

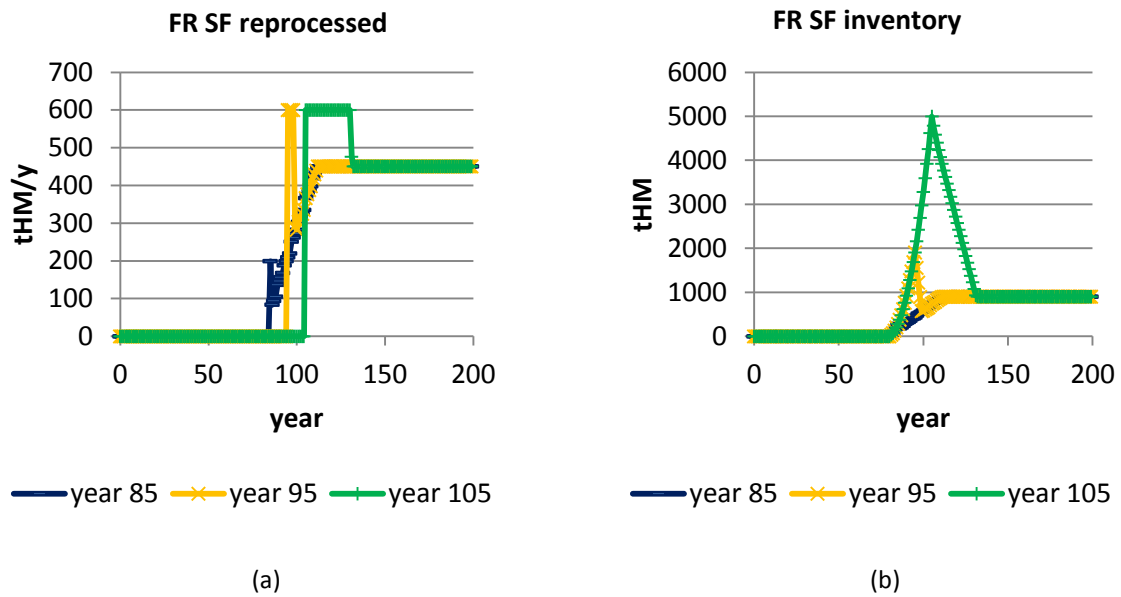


Figure 7.1-11: Inventory of separated Pu

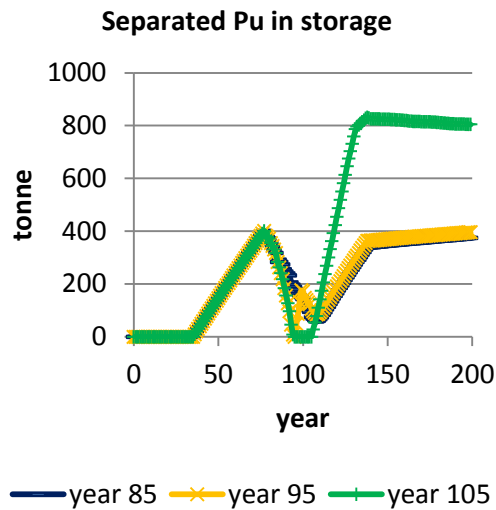
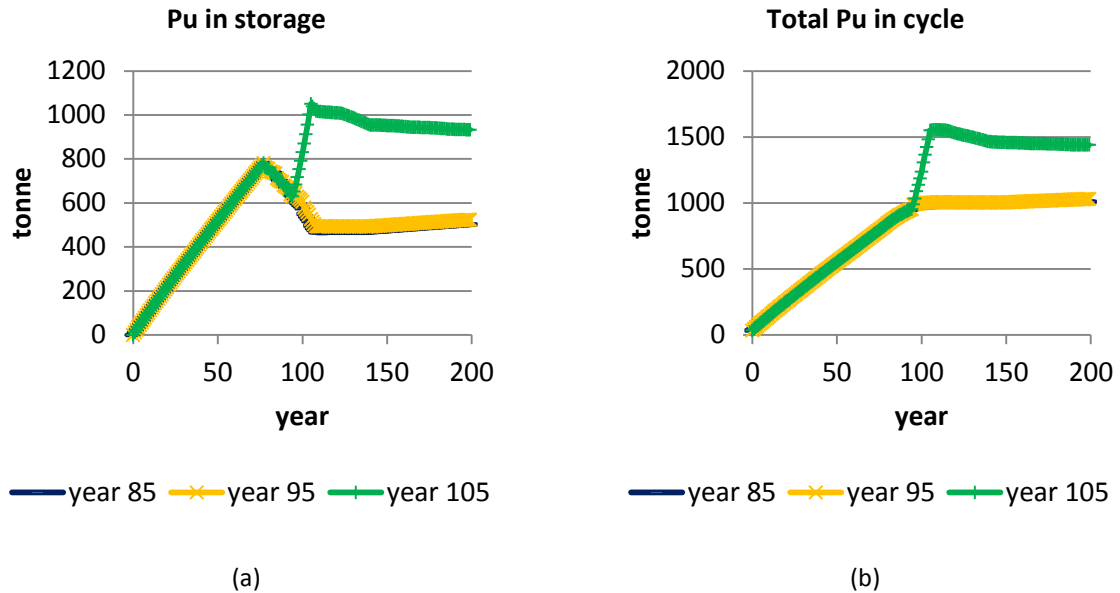


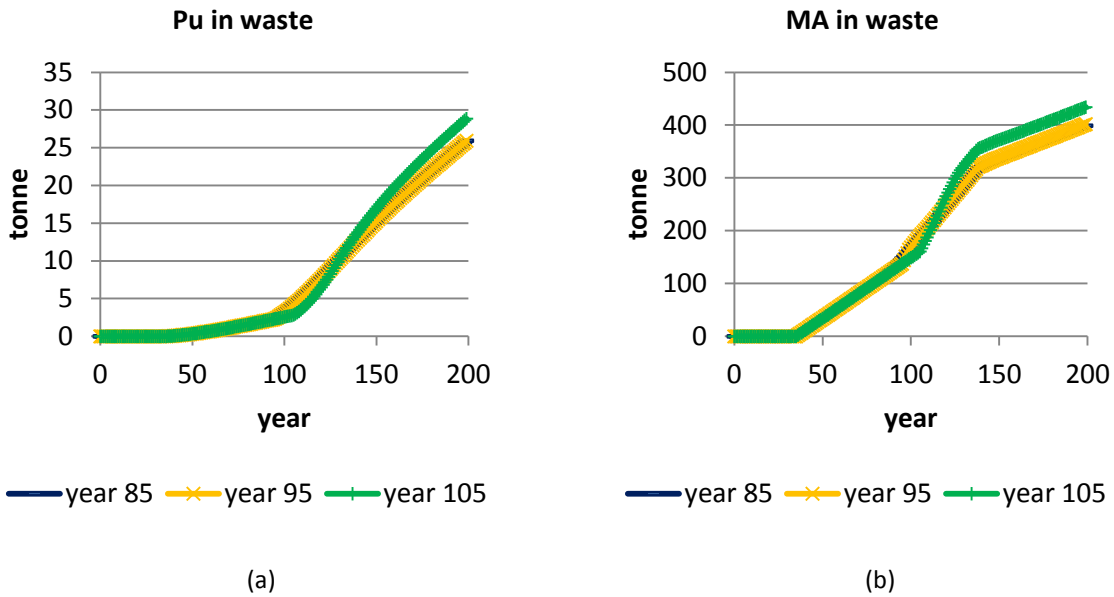
Figure 7.1-12 shows that in the case of delaying the first year of reprocessing, adding contingent plutonium results in higher plutonium inventories for the remainder of the scenario. When the reprocessing of FR spent fuel begins in the year 95, contingent fuel is added between the years 97 and 102. When the reprocessing FR spent fuel begins in the year 105, contingent fuel is added between the years 97 and 112.

Figure 7.1-12: Total Pu in (a) storage and (b) in the fuel cycle



The effects of the first year of reprocessing FR spent fuel on the inventory of plutonium and MA in waste are shown in Figure 7.1-13. In the case of starting the reprocessing in year 95, the inventory of plutonium and MA in waste is slightly lower than in the reference case between the years 85 and 98. After the year 98 only a little difference is observed. In the case of starting reprocessing in year 105, the inventory of plutonium and MA waste is lower than in the reference case between the years 85 and 113. After the year 113 the effect of the first year of reprocessing on the amount of plutonium in the waste is mainly due to the addition of contingent plutonium to the cycle, where the inventory of plutonium in the waste exceeds that of the reference case by the end of the scenario. After the year 113 the effect of the first year of reprocessing on the amount of MA in the waste is mainly due to the reprocessing of contingent plutonium and the decay of ^{241}Pu to ^{241}Am , where the inventory of MA in the waste becomes higher than the reference case to the end of the scenario.

Figure 7.1-13: The (a) Pu and (b) MA inventory in reprocessing waste



Changing the first year of reprocessing FR spent fuel affects the flow of plutonium to fabrication, the inventory of plutonium in the fabrication plants, and the plutonium in nuclear power plants as shown in Figures 7.1-14 and 7.1-15. It should be noted that the plutonium in fabrication and the nuclear power plants are affected by the source of plutonium, where FR fuel that is comprised of contingent plutonium has a larger proportion of plutonium than FR fuel that is comprised of plutonium from FR spent fuel. Delaying the first year of reprocessing FR spent fuel results in the use of contingent plutonium and, therefore, more plutonium in fabrication and more plutonium in the nuclear power plants when contingent plutonium is being used.

Figure 7.1-14: The (a) flow of Pu to fabrication and (b) the inventory of Pu in the fabrication and reprocessing plants

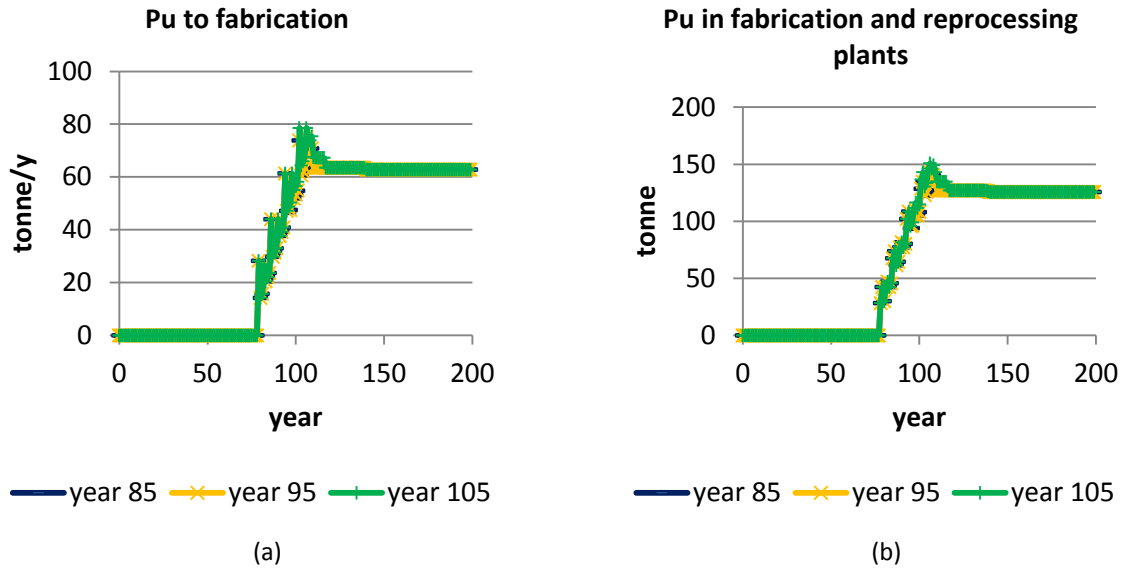
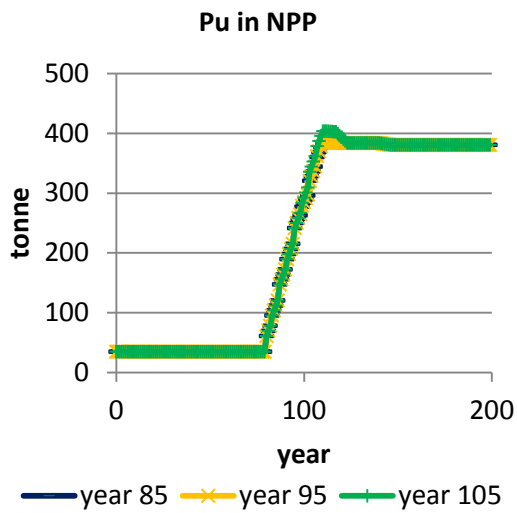


Figure 7.1-15: The Pu inventory in NPPs



7.2. Reprocessing capacity

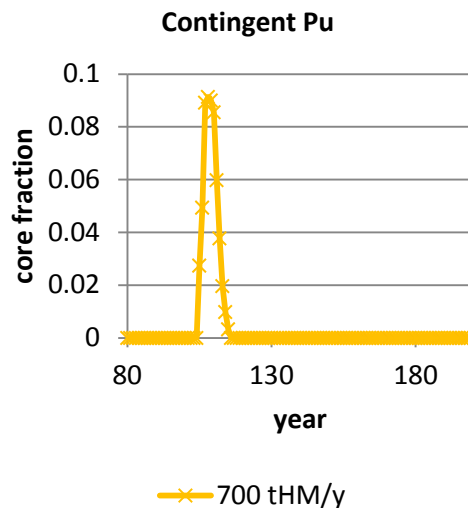
7.2.1. PWR UOX reprocessing plant

Low	Reference	High
700 tHM/y	850 tHM/y	1000 tHM/y

In the reference scenario, the capacity of the PWR UOX spent fuel reprocessing plant is 850 tHM/y to allow enough plutonium to be separated in order to fuel the transition from a fleet of PWRs to a fleet FRs.

A reprocessing capacity of 700 tHM/y is not enough to complete the transition from PWRs to FRs without the use of contingent plutonium. In this case, up to 9% of the plutonium in FR fuel must come from contingent plutonium reserves, which is needed between the years 104 and 116, as shown in Figure 7.2-1.

Figure 7.2-1: The use of contingent Pu

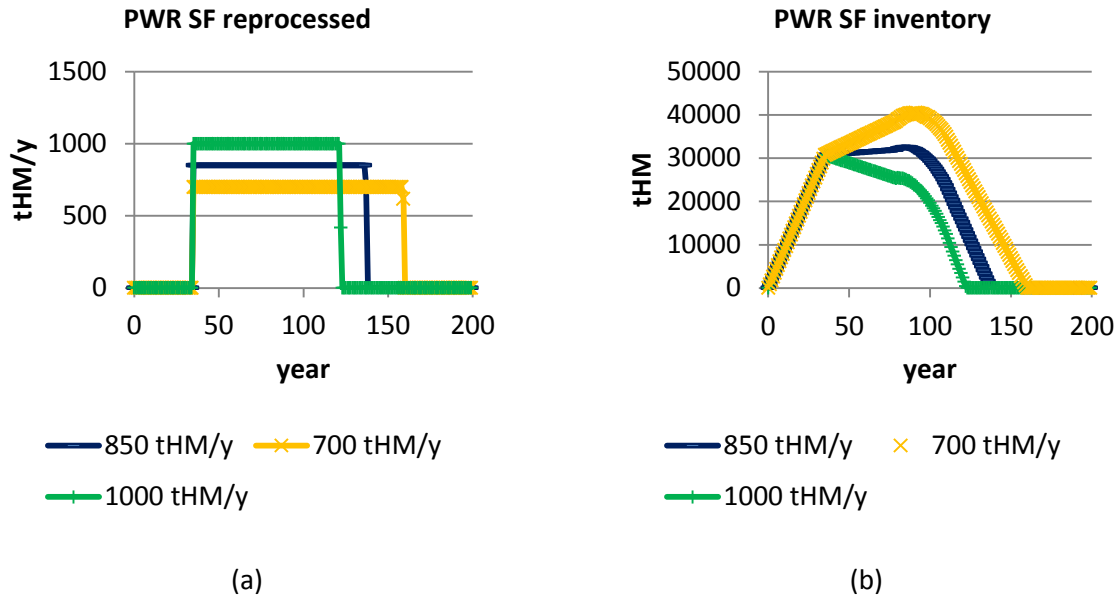


Although the same amount of spent PWR UOX fuel is reprocessed regardless of the capacity of the reprocessing plant, Figure 7.2-2a shows that the maximum PWR UOX spent fuel reprocessed annually and the time taken to reprocess all of the PWR UOX spent fuel depend on the reprocessing capacity. Specifically, a lower reprocessing capacity reduces the maximum PWR UOX spent fuel reprocessed annually and increases the time required to reprocess all of the PWR UOX spent fuel. The opposite occurs for higher reprocessing capacity.

When the reprocessing capacity is lowered to 700 tHM/y the amount of separated material in storage is lower and the amount of PWR spent fuel in storage is higher when compared to the reference scenario until all of the PWR spent fuel has been reprocessed. Conversely, increasing the reprocessing capacity to 1000 tHM/y increases the amount of separated material in storage and decreases the PWR spent fuel in storage. These

relationships can be seen in Figures 7.2-2b and 7.2-4. Lowering the reprocessing capacity to 700 tHM/y results in more separated plutonium in storage at the end of the scenario due to the contingent plutonium that is needed to complete the transition to FRs.

Figure 7.2-2: PWR UOX spent fuel (a) reprocessed and (b) in storage



If as much of the FR MOX spent fuel is reprocessed as the capacity of the reprocessing plant allows, as seen with VISION (CNL), changing the capacity of the PWR spent fuel reprocessing plant has no effect on the reprocessing of FR spent fuel. If the FR spent fuel is reprocessed on demand, then the amount of FR spent fuel that is reprocessed in a given year, depends on the amount of separated plutonium in interim storage, which depends on the capacity of the PWR spent fuel reprocessing plant. The effects of changing the capacity of the PWR spent fuel reprocessing plant on the reprocessing of FR spent fuel and on the FR spent fuel in storage are shown in Figure 7.2-3. The results shown in this figure are from COSI (CEA). Increasing the capacity of the PWR spent fuel reprocessing plant to 1000 tHM/y results in a larger inventory of separated plutonium at the time that the reprocessing of FR spent fuel would begin if the capacity of the PWR spent fuel reprocessing plant was 850 tHM/y (i.e. the reference case). This causes a temporary decrease in the amount of FR spent fuel that is reprocessed and a corresponding increase in the amount of FR spent fuel in storage until all of the PWR spent fuel has been reprocessed. Once all of the PWR spent fuel has been reprocessed then the reprocessing of FR spent fuel becomes much higher than in the 850 tHM/y case between the years 125 and 139, after which the amount of FR spent fuel that is reprocessed in the 850 tHM/y case becomes nearly equal to that of the 1000 tHM/y case. Conversely, decreasing the capacity of the PWR spent fuel reprocessing plant to 700 tHM/y results in a temporary increase in the amount of FR spent fuel that is reprocessed, followed by a larger decrease in reprocessed FR spent fuel in the 700 tHM/y case versus the 850 tHM/y case, prior to these cases becoming nearly equal for the remainder of the scenario. The larger

amount of FR spent fuel in storage by the end of the scenario in the 700 tHM/y case than the 850 tHM/y case is due to the requirement of contingent plutonium in the 700 tHM/y case.

Figure 7.2-3: FR MOX spent fuel (a) reprocessed and (b) in storage, assuming the reprocessing of FR spent fuel is on demand (courtesy of CEA)

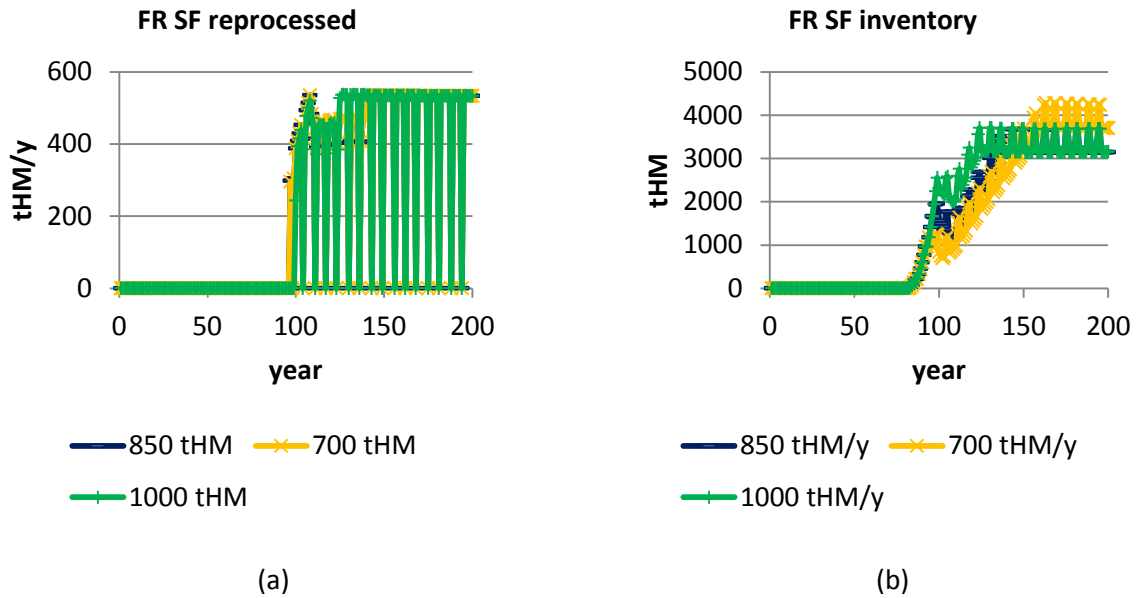


Figure 7.2-4: Inventory of (a) separated Pu and (b) reprocessed U

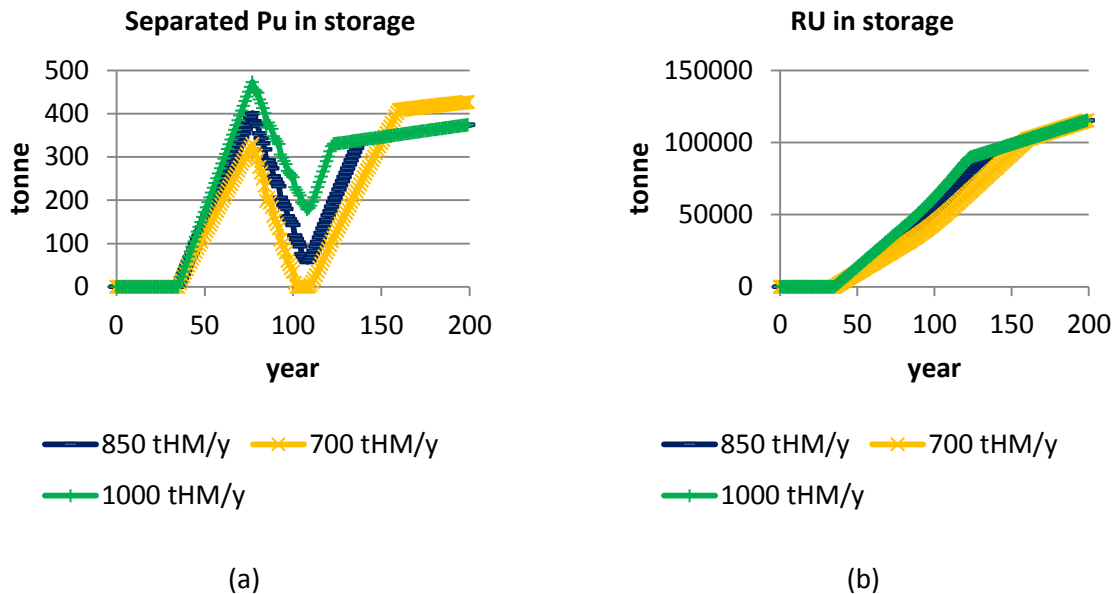
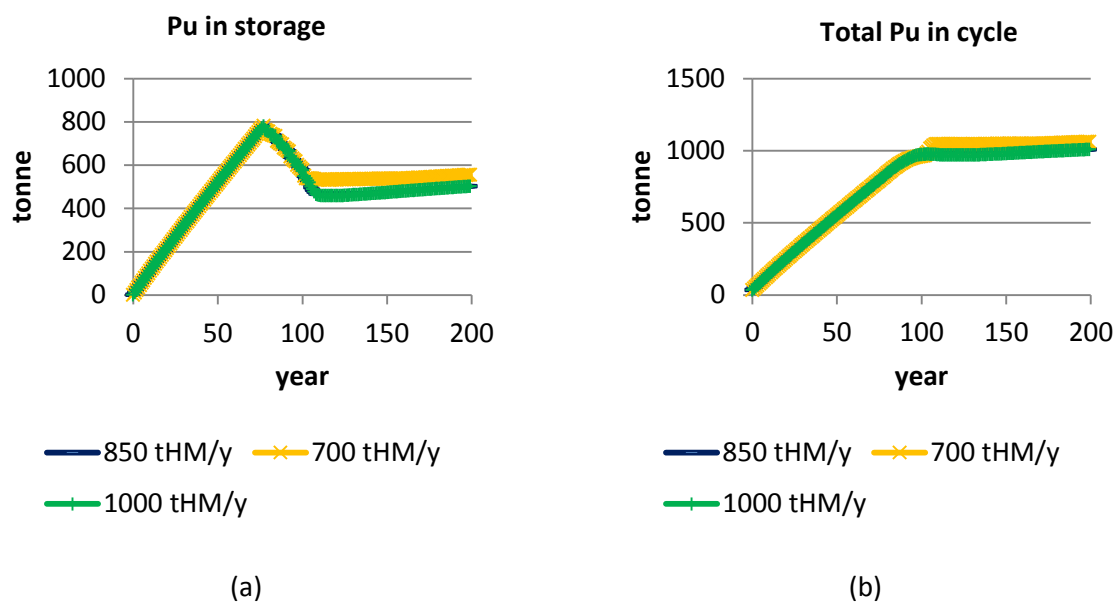


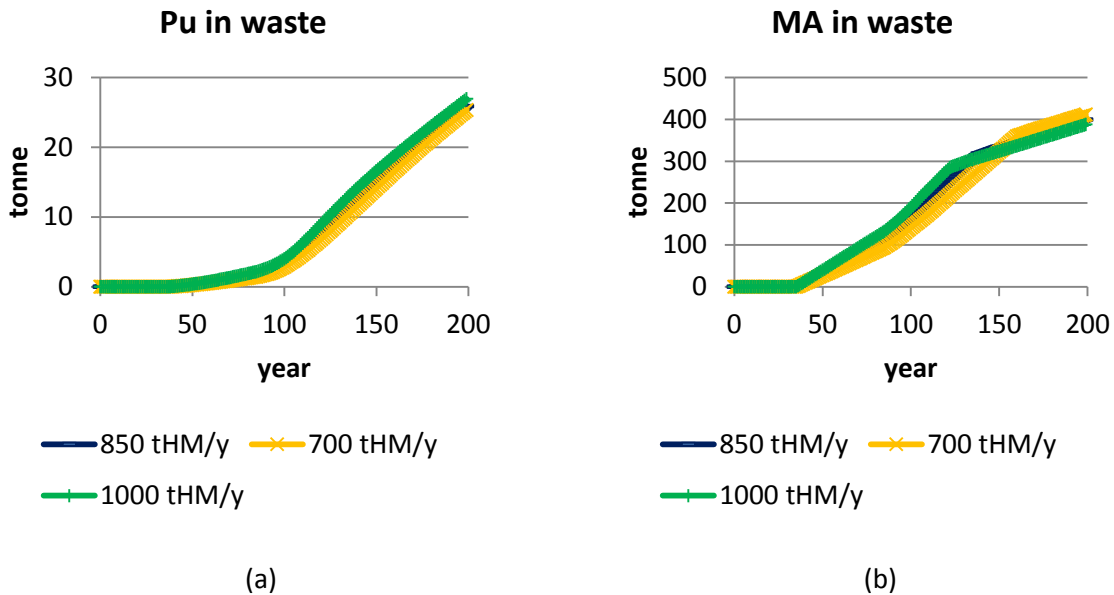
Figure 7.2-5 shows that in the case of decreasing the reprocessing capacity, adding contingent plutonium between the years 104 and 116 results in higher plutonium inventories for the remainder of the scenario.

Figure 7.2-5: The Pu inventory in (a) storage and (b) in the fuel cycle



The effects of the PWR spent fuel reprocessing capacity on the inventory of plutonium and MA waste are shown in Figure 7.2-6. Changing the reprocessing capacity affects the amount of spent fuel that has been reprocessed prior to each year afterward, until all of the PWR spent fuel has been reprocessed. In the case of increasing the reprocessing capacity, the inventory of plutonium and MA in waste is higher than in the reference case between the years 35 and 136. In the case of decreasing the reprocessing capacity, the inventory of plutonium and MA waste is lower than in the reference case between the years 35 and 156. After the PWR spent fuel has all been reprocessed, the effect of the first year of reprocessing on the amount of plutonium and MA in the waste is mainly due to the decay of ^{241}Pu to ^{241}Am . Increasing the reprocessing capacity reduces the amount of ^{241}Pu that decays prior to reprocessing, and hence increases the amount of plutonium and decreases the amount of MA in waste. Conversely, decreasing the reprocessing capacity decreases the plutonium and increases the MA in waste.

Figure 7.2-6: The (a) Pu and (b) MA inventory in reprocessing waste



The plutonium flow to fabrication and the plutonium inventory in fabrication and nuclear power plants are affected by the source of plutonium, where FR fuel that is comprised of plutonium from PWR spent fuel has a larger proportion of plutonium than FR fuel that is comprised of plutonium from FR spent fuel. Since plutonium from reprocessed PWR spent fuel is used with a higher priority than plutonium from FR spent fuel, the proportion of FR fuel from PWR plutonium depends on its availability. Increasing the capacity of the PWR spent fuel reprocessing plant increases the amount of plutonium from PWR spent fuel available to be fabricated into FR fuel resulting in a short-term increase in plutonium flow to fabrication during the transition, and a short-term decrease in the plutonium flow to fabrication after the transition is completed and the separated PWR plutonium is completely consumed. Decreasing the capacity of the reprocessing plant has the opposite effect, although contingent plutonium is required to complete the transition to FRs. Since the contingent plutonium is assumed to come from PWR spent fuel, this results in an increase in plutonium flow to fabrication when contingent plutonium is required. These effects can also be seen in the inventory of plutonium in the fabrication and nuclear power plants, which are shown in Figures 7.2-7 and 7.2-8

It should be noted that the time to reprocess spent fuel is assumed to be zero, therefore there is no plutonium inventory in the reprocessing and fabrication plants prior to the year 78 (Figure 7.2-7). If the time to reprocess spent fuel is not zero, there would be an inventory of plutonium in the reprocessing plant during its years of operation.

Figure 7.2-7: The (a) flow of Pu to fabrication and (b) the inventory of Pu in the fabrication and reprocessing plants

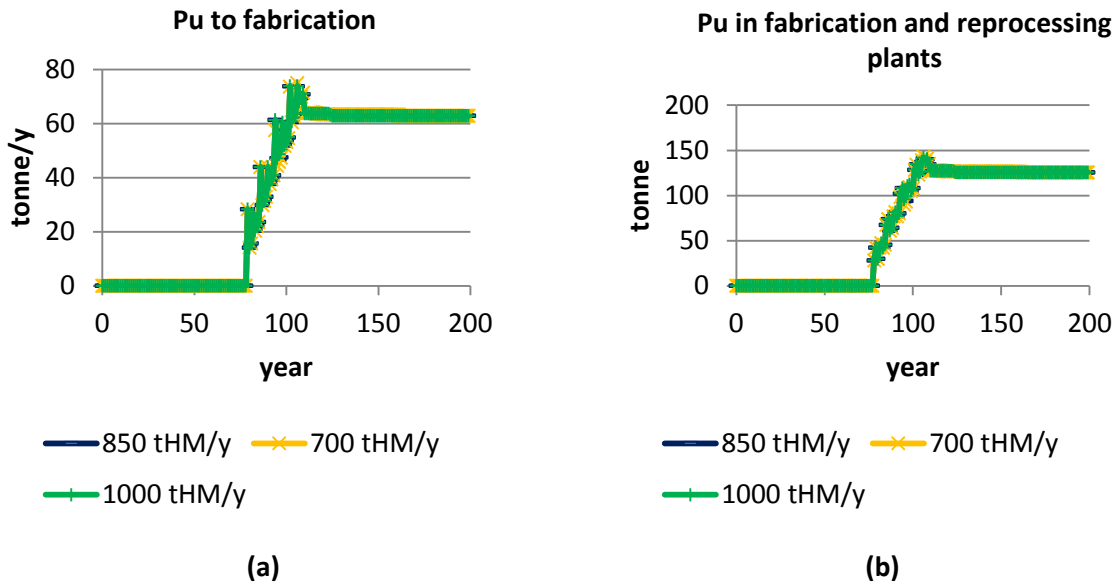
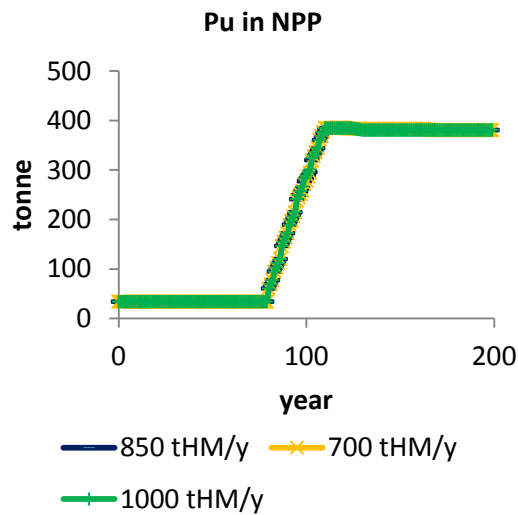


Figure 7.2-8: The Pu flow to fabrication



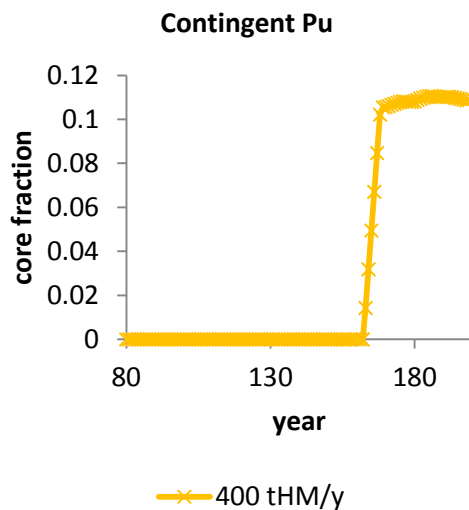
7.2.2. FR MOX reprocessing plant

Low	Reference	High
400 tHM/y	600 tHM/y	800 tHM/y

In the reference scenario the capacity of the FR MOX reprocessing plant is 600 tonnes/yr to help supply plutonium for the transition to FRs, and to enable the continued operation of FRs once the plutonium from PWR spent fuel has been depleted.

Figure 7.2-9 shows that a 400 tonnes/year reprocessing capacity of the FR MOX reprocessing plant is sufficient to complete the transition to FRs, but is insufficient to maintain enough plutonium to fuel the entire fleet of FRs beyond the year 162. After the year 162, up to 11% of the plutonium in FR fuel must come from contingent plutonium reserves.

Figure 7.2-9: The use of contingent Pu



The lower reprocessing capacity results in the build-up of FR spent fuel in storage starting in year 111 and growing steadily until the end of the scenario, as shown in Figure 7.2-10b.

According to the reference scenario approximately 450 tHM/y of FR spent fuel needs to be reprocessed in order to maintain a constant amount of FR spent fuel in storage, as is shown in Figure 7.2-10a. This means that increasing the reprocessing capacity to 800 tHM/yr does not result in more spent fuel being reprocessed each year.

Figure 7.2-10: FR MOX spent fuel (a) reprocessed and (b) in storage

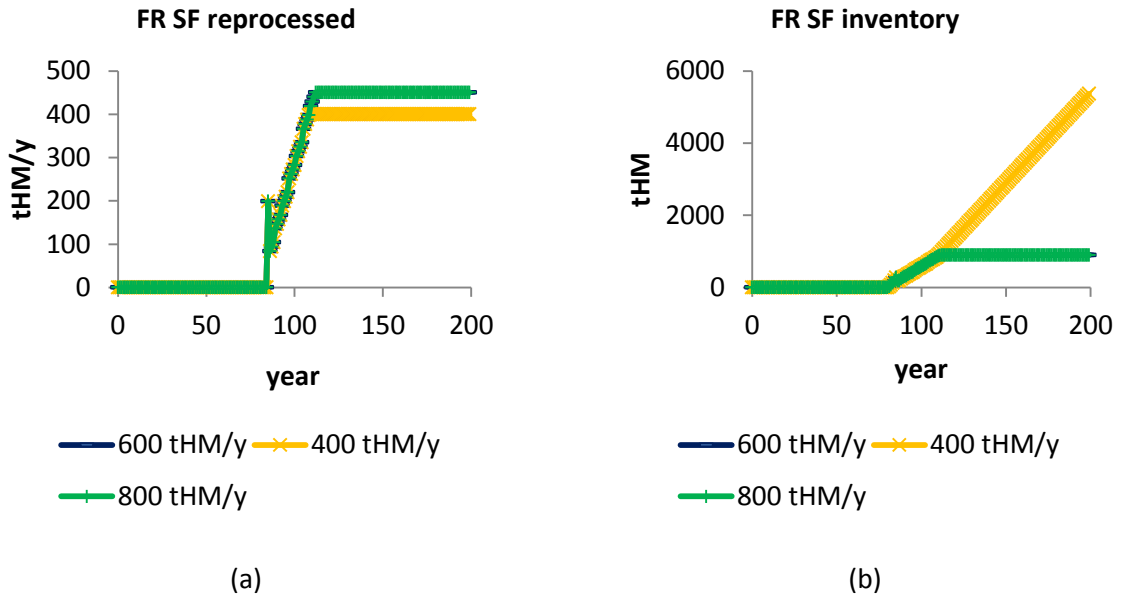
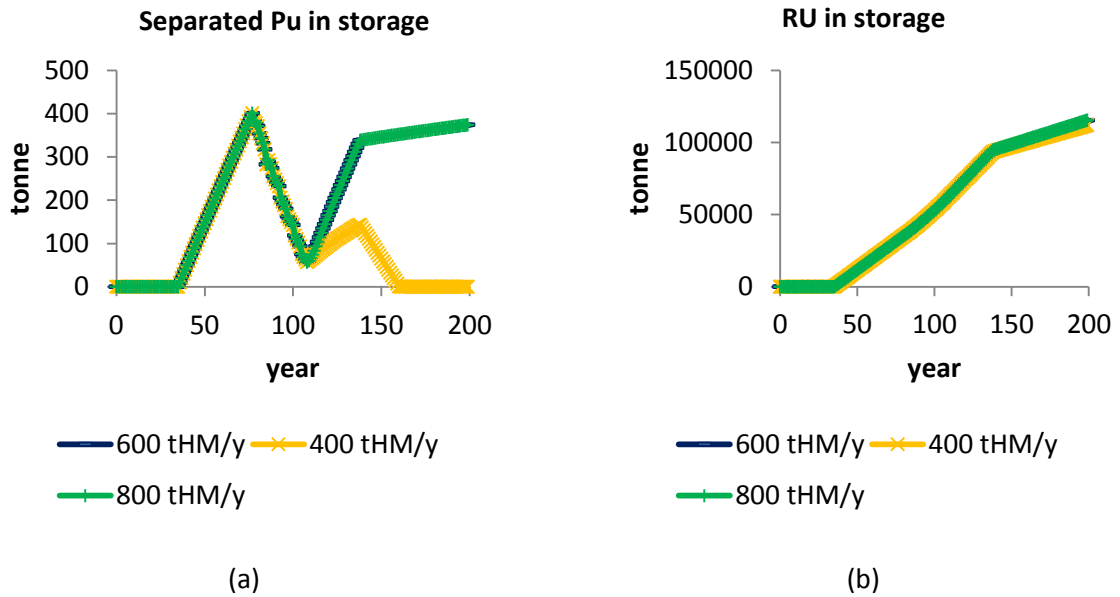


Figure 7.2-11a shows that a 400 tonnes/year reprocessing plant produces enough plutonium to fuel the transition to FRs while also building up a stock pile of plutonium. But once there is no more plutonium from PWR UOX spent fuel, then the stock of plutonium from FR spent fuel is depleted in 22 years, after which contingent plutonium is then required. The lower reprocessing capacity also results in less reprocessed uranium in storage by the end of the scenario, as is shown in Figure 7.2-11b. These figures also show that a higher reprocessing capacity has negligible effect on the inventory of separated materials.

Figure 7.2-11: Inventory of (a) separated Pu and (b) reprocessed U



The inventories of plutonium in storage and in the fuel cycle are only affected by the use of contingent plutonium, which is required when the capacity of the FR reprocessing plant is 400 tHM/y. This can be seen in Figure 7.2-12

Figure 7.2-12: The Pu inventory in (a) storage and (b) in the fuel cycle

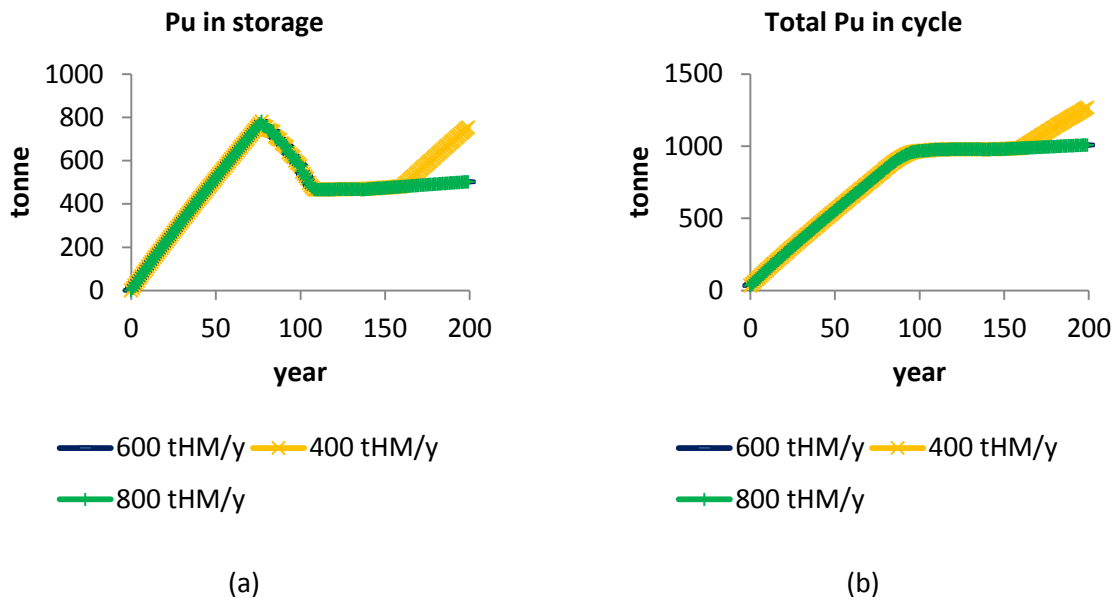
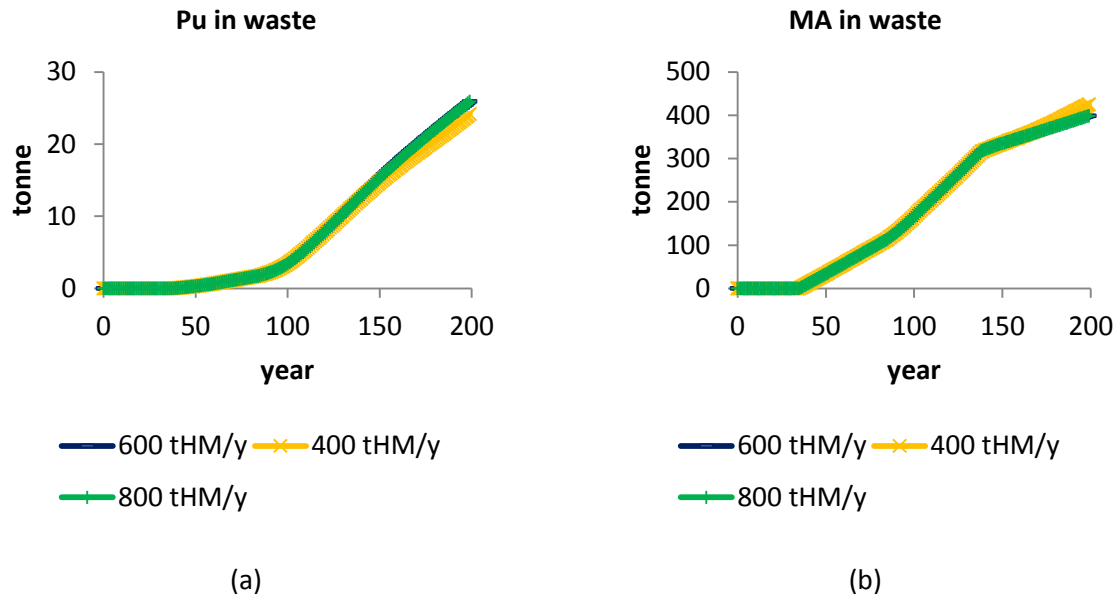


Figure 7.2-13 shows that increasing the capacity of the FR MOX spent fuel reprocessing plant has very little effect on the plutonium and MA inventory in the reprocessing waste, while decreasing the capacity to 400 tHM/y decreases the inventory of plutonium and increases the inventory of MA in the waste. This is mostly due to the longer time that spent fuel spends in storage, building up ^{241}Am from the decay of ^{241}Pu , prior to being reprocessed.

Figure 7.2-13: The (a) Pu and (b) MA inventory in reprocessing waste



Figures 7.2-14 and 7.2-15 show that increasing the capacity of the FR MOX spent fuel reprocessing plant has very little effect on the plutonium in fabrication and the nuclear power plants, while decreasing the capacity to 400 tHM/y results in higher plutonium after all of the plutonium from PWR spent fuel has been sent to fabrication.

Figure 7.2-14: The (a) flow of Pu to fabrication and (b) the inventory of Pu in the fabrication and reprocessing plants

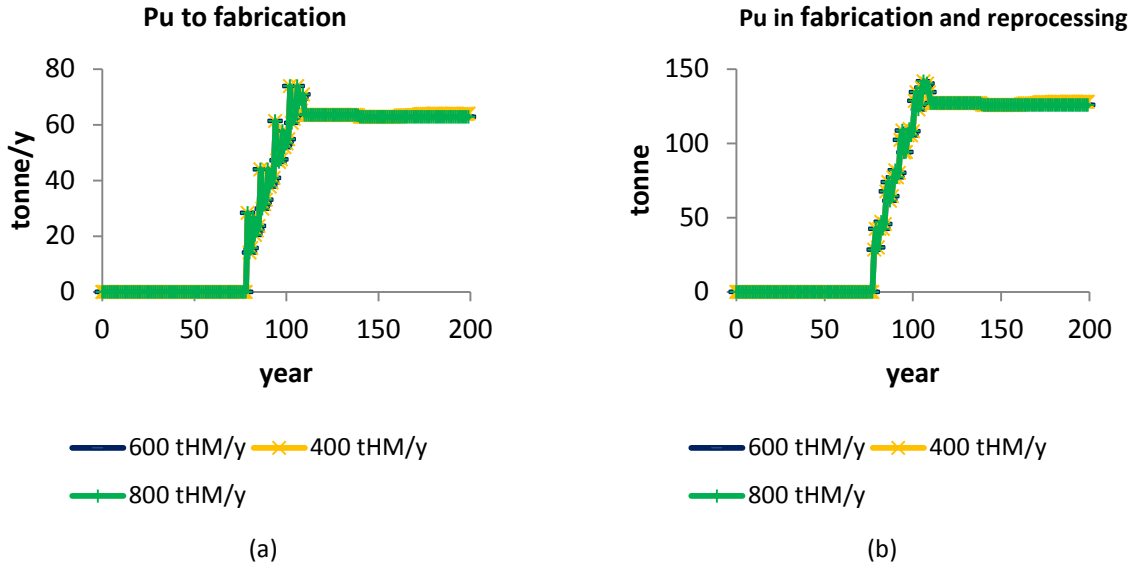
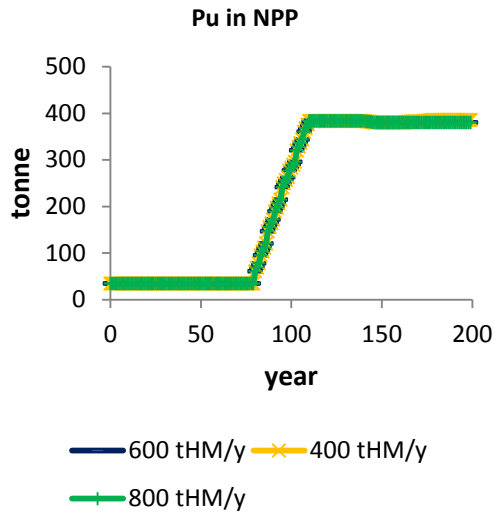


Figure 7.2-15: The Pu inventory in NPPs

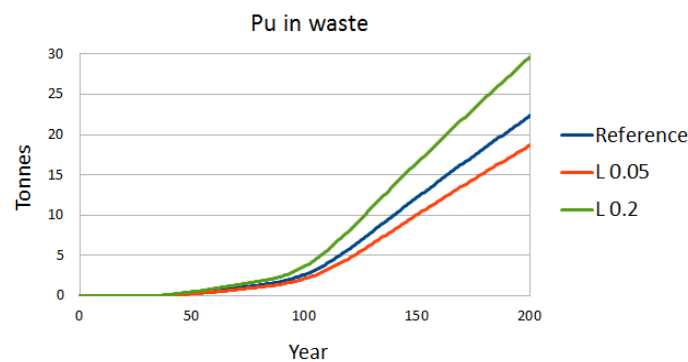


7.3. Reprocessing losses

7.3.1. U and Pu

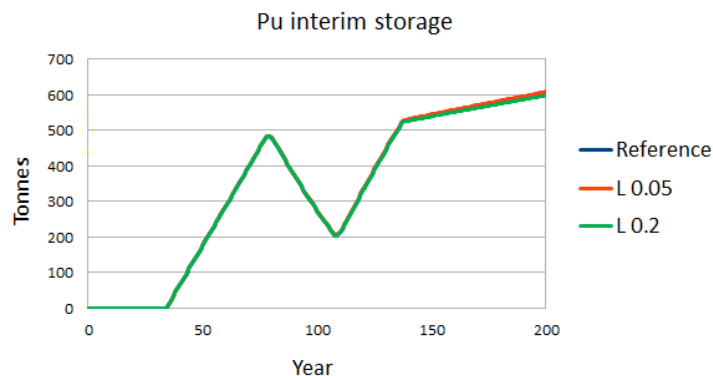
The reference scenario assumes that a PUREX (Plutonium and Uranium Refining by Extraction) process is used in both PWR and FR reprocessing plants for the separation of all the isotopes of U and Pu from the minor actinides and fission products. The U and Pu separation efficiency of large reprocessing plants such as La Hague (France) or Sellafield (UK) is considered to be 99.9%, which is the value taken in the reference scenario. The variation of this parameter to 99.95% and 99.8% allows evaluating the impact it has in the mass of both elements at different stages of the fuel cycle.

Figure 7.3-1: Sensitivity of Pu in waste to reprocessing losses

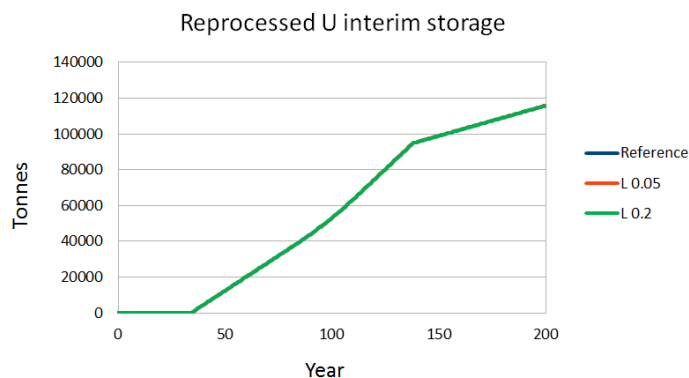


The main impact of this parameter can be appreciated in the amount of Pu present in the final waste. The difference between scenarios can be seen from the first year reprocessing starts (year 35), but it remains relatively small until the FR fleet is deployed and MOX begins to be reprocessed. Once this FR fleet is fully set up, the higher concentration of Pu in the spent fuel (63.77 tonnes/year for the FR vs 11.04 tonnes/year for the PWR) leads to an increasingly bigger difference, which at the end of the scenario is of -3.66 and +7.36 t respectively for the case at 99.95% (indicated in figure as L 0.05) and the case at 99.8% (L 0.2), as shown in Figure 7.3-1.

The amount of Pu in the intermediate separated pool is likewise affected by the reprocessing losses. This difference remains very small while LWR spent fuel elements in storage are being reprocessed. After the deployment of the FR fleet, given the higher amount of Pu in the spent fuel, the different Pu recovery causes significant differences in this separated pool of Pu. This effect is shown in Figure 7.3-2.

Figure 7.3-2: Evolution of Pu in interim storage

The amount of reprocessed U (see Figure 7.3-3) is likewise affected by this parameter, although the quantity of the total mass in this case makes the difference almost unappreciable.

Figure 7.3-3: Sensitivity of reprocessed U of reprocessing losses

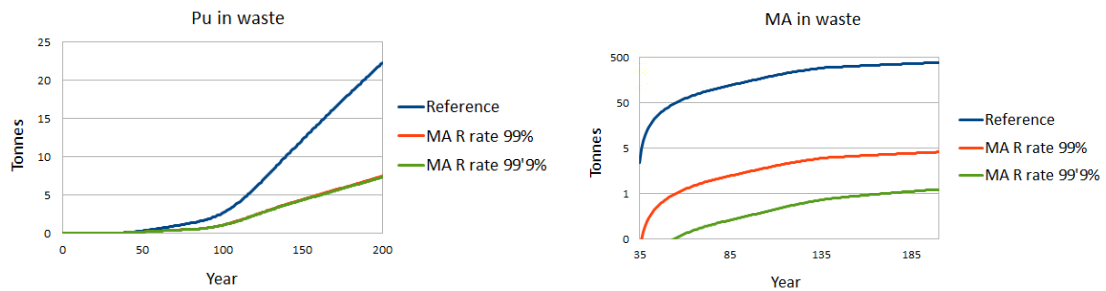
7.3.2. Minor actinides

The reference scenario considers only the reprocessing and reuse of Pu as part of the MOX to be consumed by the FR. The effect of a similar separation of the most relevant MA (americium, neptunium and curium) with different efficiencies has also been explored and is shown in Figure 7.3-4. It is supposed that these MA are stored and kept away for a later use outside of the cycle, which is why they are not sent to the final disposal. On the one hand, the extracted MA are stored apart and therefore their decay into isotopes of Pu does not contribute to the amount of this element in the final waste or inventory stream (mainly ^{242}Cm into ^{238}Pu). The recovery of the MA leads to approximately 15 tonnes less of Pu in the final

waste disposal, which makes for a 33 wt% less in both parametric cases at the end of the cycle.

On the other hand, the amount of MA present in the final repository is reduced in more than two orders of magnitude, in accordance with the selected reprocessing efficiency.

Figure 7.3-4: Sensitivity to MA recuperation rate variation

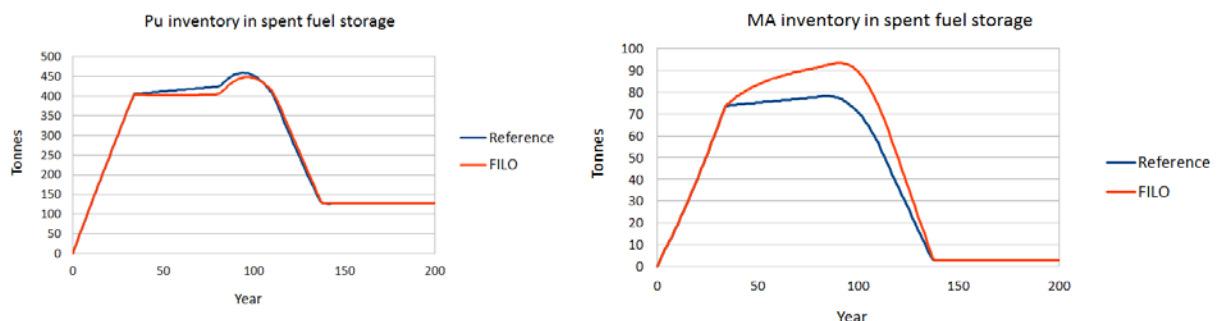


7.4. Reprocessing priority

7.4.1. FIFO to FILO

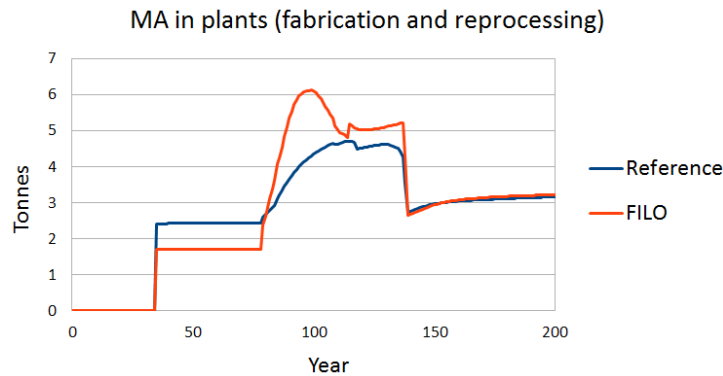
The reference scenario considers the oldest fuel elements of the PWR fleet to be the first ones being reprocessed for obtaining the Pu needed for the fabrication of MOX fuel. This “First In First Out” strategy is changed to a “First In Last Out” strategy in this sensitivity study. This change causes differences mainly at isotopic level. In particular, the first reprocessed fuel elements have different Pu composition (mainly more ^{241}Pu , since its rapid decay into ^{241}Am is avoided, and also less ^{238}Pu , given that the decay of ^{242}Cm in this isotope is also avoided). Figure 7.4-1 shows this behaviour in the amount of Pu and MA in storage. Since in the FILO strategy (red curve) the elements richer in Pu are reprocessed first, the amount of Pu is slightly smaller than in the reference strategy (in the period between years 35 and 140 when PWR spent fuel are reprocessed). In the case of the MA, the FILO strategy shows a larger amount of MA during this period, since the spent fuel contains a larger amount of ^{241}Am .

Figure 7.4-1: Pu and MA storages' sensitivity to reprocessing strategy



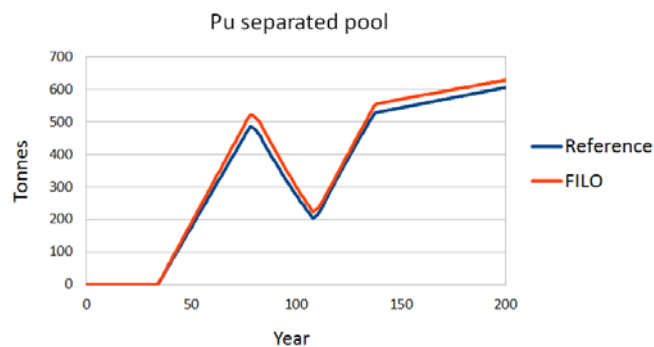
The amount of MA present in reprocessing plants is accordingly reduced in the FILO strategy until the PWR fleet starts being decommissioned, as shown in Figure 7.4-2, since Pu-rich spent fuel is reprocessed first. After this period, the MA in reprocessing plus fabrication plants is larger in the FILO strategy, due to a larger amount of ^{241}Am in the Pu stream for MOX fabrication (decayed from ^{241}Pu after reprocessing) and also because the Pu-rich spent fuel has been exhausted and MA-rich elements are reprocessed. Once the PWR spent fuel storage has been completely emptied, this effect can no longer be appreciated as the FR elements are reprocessed as soon as they leave the cooling pools.

Figure 7.4-2: MA in plants' sensitivity to reprocessing strategy



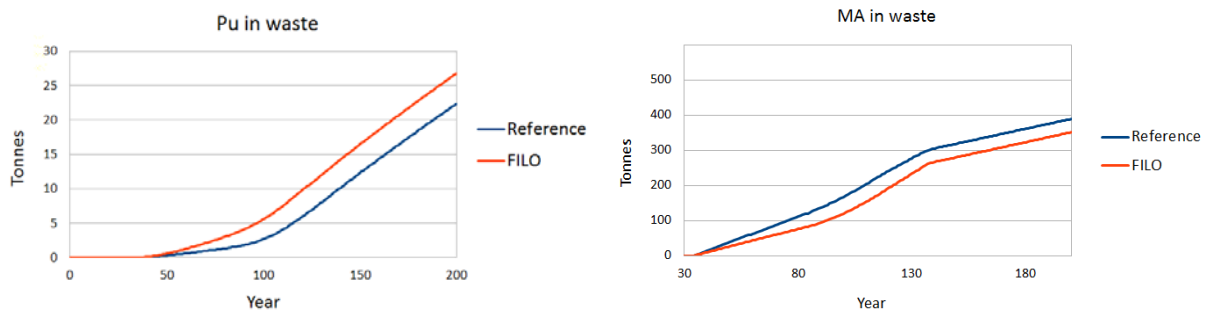
The impact of the different reprocessing strategy leads to a bigger amount of Pu in the separated Pu pool. This result is shown in Figure 7.4-3. This additional Pu includes however some ^{241}Pu that will decay to ^{241}Am . This isotope will be included likewise in the MOX fuel, as there is no isotope separation in the fabrication plants.

Figure 7.4-3: Pu interim storage sensitivity to reprocessing strategy



In the FILO strategy, the amount of Pu in final waste is increased as the decay of ^{242}Cm into ^{238}Pu takes place mainly in the waste storage, while the absence of ^{241}Pu leads to a smaller amount of ^{241}Am . When the PWR fleet is fully decommissioned and its spent fuel elements have all been reprocessed, these two parameters Pu and MA in waste evolve at the same rate as in the reference case (Figure 7.4-4).

Figure 7.4-4: Evolution of the materials in waste

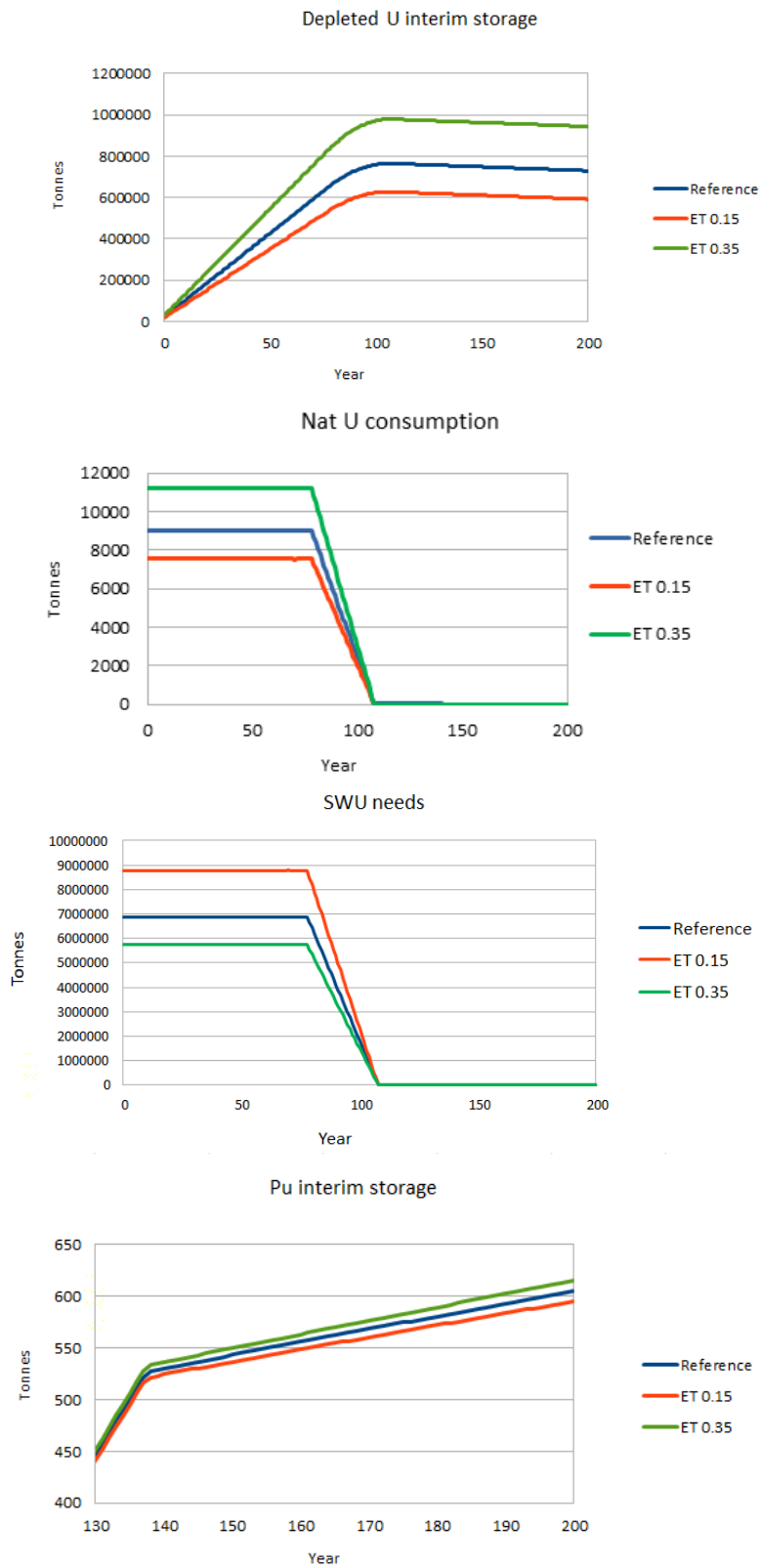


7.5. Enrichment tails

In the reference case, depleted U after UOX fabrication for the PWR fleet has a content of 0.25% in ^{235}U . Figure 7.5-1 shows how a better exploitation of this isotope allows for a reduction in the amount of depleted U stored, and vice versa. Once the PWR fleet is totally shut down (year 110) the difference remains stable until the end of the scenario, generating 13 8000 tonnes more and 214000 tonnes less for an enrichment tail (ET) of 0.35% and 0.15% respectively at the end of cycle. An analogous but opposed in sign effect takes place both in the amount of required natural U and in the required SWU units.

Additionally, the larger amount of the fissile isotope in the depleted U used for the fabrication of the FR MOX and blankets allows for savings in the Pu required for MOX fabrication. Results show a reduction of the order of 2 wt% from the moment the FR fleet is fully deployed. This can be appreciated in the mass of separated Pu (Pu interim storage) that remains in the storage at the end of the scenario, which increases when the depleted U has a bigger concentration of ^{235}U . Similarly, in the case of a smaller amount of ^{235}U in the enrichment tail the opposite effect is found, leading to the use of 2wt% more Pu in the fabrication of MOX fuel.

Figure 7.5-1: Sensitivity to enrichment tail variation



8. Summary of results and conclusions

8.1. Overview of the study

This report has documented work performed by the NEA Expert Group on Advanced Fuel Cycle Scenarios to assess and quantify the importance of input parameters in scenario analysis. The primary applications are for scenario code designers, to decide when additional capabilities should be included in their code, and scenario code users, to determine what types of sensitivity cases should be included in their analyses.

The expert group drew on their collective experience to identify 15 input parameters which we felt are the most important in modelling a fuel cycle scenario, and then assessed the impact of uncertainties in these parameters on 22 of the most important scenario outputs or indicators. The range of uncertainty for each input parameter was also based on experience as well as familiarity with the body of published work in this area.

Multiple fuel cycle scenario analysis codes are in use today, and comparison of code results on standard problems is a common way to ensure code quality. However, every code is also different, either in the basic structural approach (e.g. systems dynamics, object-oriented, etc.) and/or in the aspects or level of detail of the real world included in the code model (e.g. radioactive decay, depletion, etc.). Where prior benchmarking by this group and others has, of necessity, focused on only the common core capabilities of the different scenario analysis codes involved in the benchmark, this activity deliberately assessed some parameters or engaged some analysis capabilities only found in a subset of the codes. This has allowed this report to include a broader range of cases than that available in the benchmarks, allowing us to provide fuller information on the impacts of uncertainty.

The following sections summarise the findings contained in the detailed analyses of the previous chapters. In short, these summaries show where the impacts of uncertainty in the parameters often used to define fuel cycle scenarios were or were not significant in shaping the outcome of those scenarios.

8.2. Sensitivity analysis results

Chapter 4 described in detail two methods for consolidating the results in this report, the sensitivity table and the tornado diagram. This included the mathematical formulation and examples. These methods are both used in this section to summarise the results of the sensitivity analyses.

The sensitivity table provides a way to display the sensitivity analyses results, with all inputs and outputs together on a single table. The correlation of each input (table row) to each output (table column) is shown through colour shading of each cell in the table. The

stronger the shading in the table cell, the larger the impact of the input parameter on the output parameter. Red shading indicates a positive correlation versus blue shading for a negative correlation. In cases where the relationship is non-linear or otherwise hard to quantify into a single value, the shading may be replaced with a “?”. The reader is referred to Section 4.3 for the mathematical explanation.

Where the sensitivity table shows all of the analyses together, the tornado diagram shows the correlation of a single output parameter to changes in the input parameters, one bar per input, where the direction of the bars on the tornado diagrams show whether the correlation is positive, meaning both values move in the same direction (either both increasing or both decreasing), or negative, meaning the output value moves in the opposite direction of the input value. The length of the bars shows the magnitude of the coupling between the parameters, with longer bars indicating a larger impact on the output parameter. The ordering of longest bars on top to shorter bars below shows at a glance which inputs have the greatest impact on the output. Two bars are used for each input parameter, one representing the impact of the low input value and the other the high input value used as the bounds of the sensitivity analysis. If the bars differ in length or direction, it indicates a non-linear relationship between the input and output parameters.

Non-linear relationships occur when a part of the system moves from constrained or saturated to unconstrained or unsaturated over the range of the input parameter. Often when one part of the system becomes less constrained, it results in a new constraint appearing elsewhere in the system. The drivers of non-linear system responses are the underlying properties and interactions of the physical system. An understanding of those drivers can contribute to improving system performance and resiliency.

8.2.1. Sensitivity table analysis results

Table 8.2-1 summarises all the sensitivity indicators “S” obtained from the various input parameters (one row for each input parameter) and the various output parameters (one column for each output parameter). As those results should be considered as tendencies rather than absolute values, colours only are indicated on this table.

When a sensitivity coefficient is positive (red), this means an increase (resp. a decrease) of the input parameter induces an increase (resp. a decrease) of the output parameter. On the contrary, when a sensitivity coefficient is negative (blue), this means an increase (resp. a decrease) of the input parameter induces a decrease (resp. an increase) of the output parameter.

When a coefficient of determination “ r^2 ” is lower than 0.9, then the related sensitivity indicator is replaced by a question mark “?” in the table.

When the output parameter is not impacted by the variation of the input parameter, then the related sensitivity indicator is not available and it is replaced by a blank in the table.

Table 8.2-1: Sensitivity values “q” obtained from the various input parameters (one row for each input parameter) and the various output parameters (one column for each output parameter)

S ($r^2 \geq 0,9$)	PWR cycle				FR cycle			Reprocessing		Inventory					
	Front-end			Back-end	Front-end		Back-end	PWR	FR	Pu				MA	
	Uranium	Enrichment	Fabrication	Storage	Pu	Fabrication	Storage			Plants	Reactors	Storages	Cycle	Waste	Cycle
Reactor characteristics															
Breeding gain	?				?			?							
Reactor Lifetime							?								
MA Recycling															
MA content															
MA recuperation rate															
General Scenario Assumption															
Energetic production								?							
Minimum cooling time	PWR														
	FR														
Fabrication time	PWR														
	FR														
Introduction date of FR															
Rate of introduction															
Reprocessing															
First year of reprocessing	PWR														
	FR														
Reprocessing capacity	PWR														
	FR														
(U, Pu) Losses	?														
Enrichment tail															

The sensitivity table provides the big picture view of the analysis results. The most important input parameters from a sensitivity perspective can easily be identified by noting the rows with the strongest shading. These are the overall growth rate (energetic production) and the introduction date of the FRs, followed by the rate of FR introduction, the reactor lifetime and the breeding gain.

Growth rates impact everything in the system, including the number of reactors and associated fuel cycle facilities needed and the mass flow rates of fuel and waste materials. A small growth rate compounds over multiple years to yield a large change in the system size, which is the driver behind the strong shadings in the sensitivity table. Similarly for the assessed base scenario, the date of introduction of FRs strongly impacts the infrastructure and mass flow changes necessary to transition from an open-cycle PWR-based system to a full recycle FR-based system. The rate of FR introduction is a modifier on the introduction

date, as it compresses or stretches the transition period. Again, this impacts all of the infrastructure and mass flow changes that must occur to complete transition.

In contrast to the previous 3 items, the change in reactor lifetime impacts material flows without changing the rest of the system. The shorter life of each reactor, the more full start-up cores that must be fabricated and the more full retirement cores that must be managed. The longer the reactor life, the slower the transition can proceed in a low- or no- growth scenario, because an old reactor must retire before it can be replaced with a new reactor of a different type.

Additional input parameters with strong impacts on the scenario results are in the areas of breeding gain and reprocessing, where rows have some strongly shaded cells, but also have a number of “?” cells, indicating possible non-linear responses. Breeding gain is usually non-linear because Pu insufficiency significantly impedes transition, but once sufficient Pu is available, additional changes in breeding gain have limited consequences. The same is true of the reprocessing parameters, since reprocessing controls how much of the Pu in the system becomes available for use in FRs during transition.

The sensitivity table also reveals more localised high impacts where one or two cells are strongly shaded relative to the rest of a row. For example, the enrichment tails assay only impacts NU usage and SWU requirements, but does not impact the rest of the fuel cycle.

Shifting to the columns, only a few outputs show significant sensitivity to a majority of the inputs, and none are sensitive to all inputs. The outputs most impacted are the storage and inventory values.

8.2.2. Sensitivity analysis results with tornado diagrams

The results presented in this section are from VISION (CNL), COSI (CEA), and TR_EVOL (CIEMAT). Figures 8.2.1 to 8.2.6 show the tornado diagrams in which the input parameters that have the largest impact on a given output parameter are shown. In these diagrams the vertical axis represents the input parameters that have a q value for the given output parameter of more than 1×10^{-4} , which are sorted in descending order of the difference between the high and low values of q from top to bottom. The horizontal axis represents the value of q for the given input and output parameters. Horizontal bars are plotted to show the value of q for the low and high value of the given input parameter.

Each tornado diagram also shows how a given output parameter responds to changing an input parameter from its reference to its low value, and from its reference to its high value. This means that the following inferences can be made based on the relative values of q for the low and high input parameter value.

Similar q values (sign and magnitude): the relationship between the input parameter and output parameter is approximately linear over the given range of the input parameter.

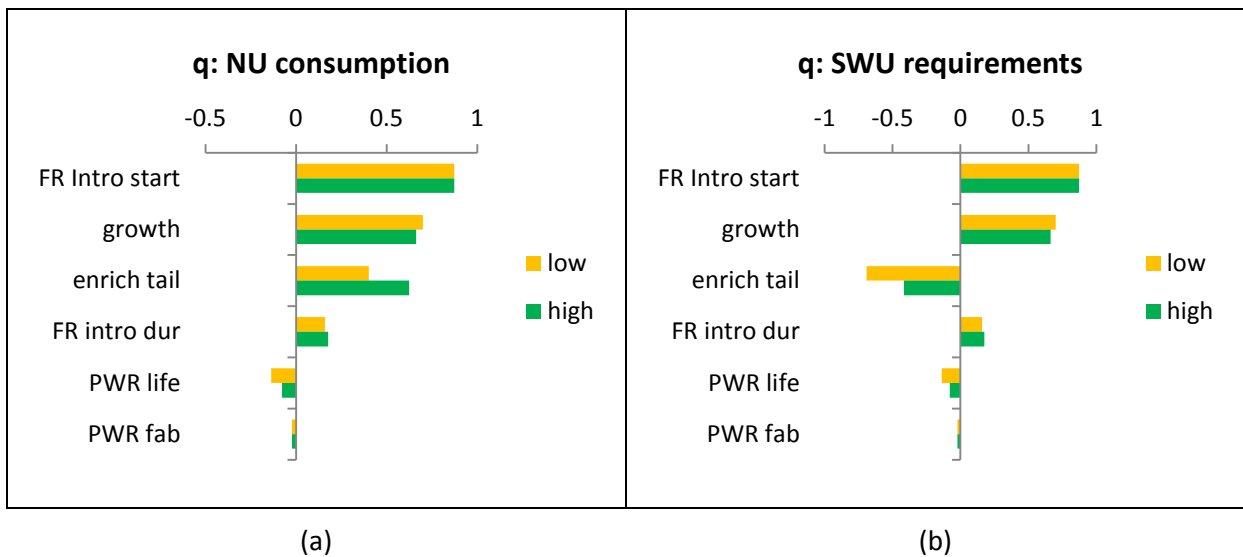
q values with the same sign but different magnitudes: the change in output parameter value per unit change in input parameter value differs when going from the reference to the low input parameter value versus going from the reference to the high input parameter value (non-linear response).

q values with different signs: the given range of the input parameter contains at least one critical value across which the relationship between the input parameter and the output parameter changes from negative to positive.

Note that the sensitivity cases that require changing more than one input parameter (eg. PWR fuel and MA fuel loading) are excluded from this analysis.

The low and high FR breeding ratio sensitivity cases correspond to 0.91 and 1.2, respectively. The low FR breeding ratio case crashes after year 161 due to a shortage of separated plutonium. Therefore the q values for the low FR breeding ratio case are calculated over the first 161 years of the scenario. The q values for the high FR breeding ratio case are calculated over the entire 200 years of the scenario.

Figure 8.2-1: The sensitivity of (a) NU consumption, and (b) SWU requirements, to each parameter



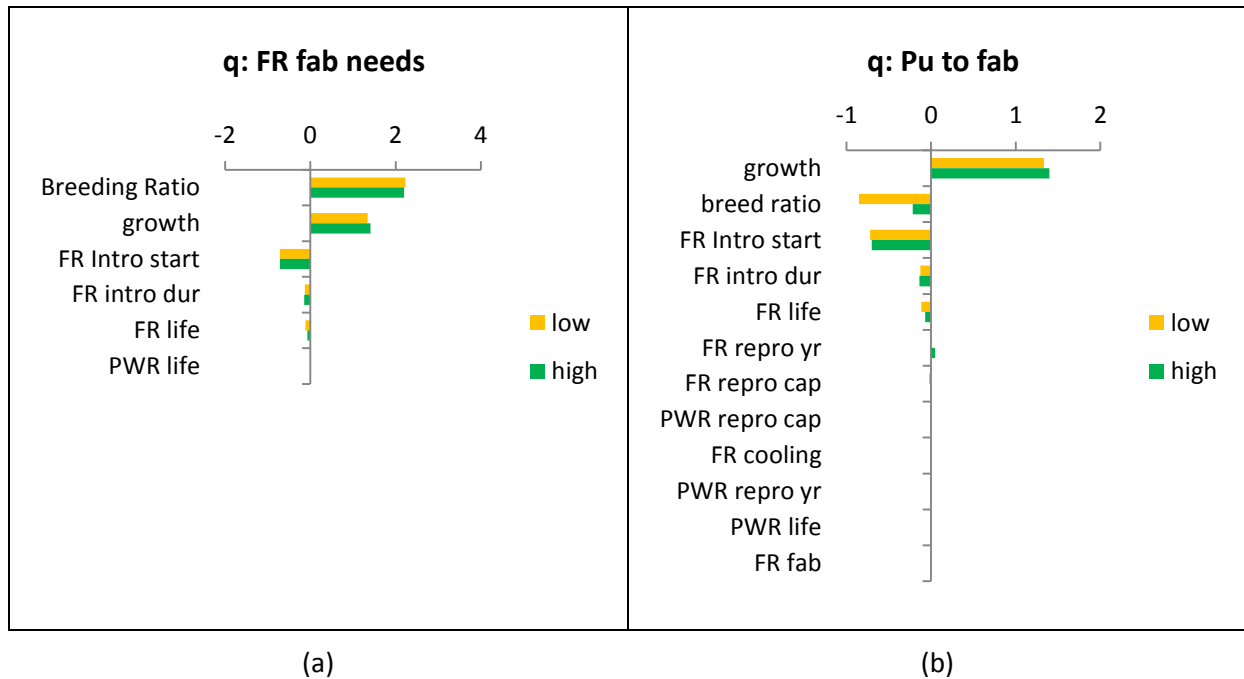
The three factors with the largest impact on both NU consumption and SWU requirements are when the FRs are introduced, the energy demand growth, and how much fissile material is extracted from each tonne of NU. The date of FR introduction directly correlates to the total reactor years of LWR operation, and therefore to the amount of NU needed. Once the FRs are in operation, they can run on the accumulated DU. The relationship is linear.

The energy demand growth results in an increase in LWR operational years, and therefore higher NU consumption and higher SWU requirements. The relationship is linear.

For tails enrichment, the relationships are nonlinear. This is due to the additional effort (SWUs) required to extract additional fissile material from lower assay tails, and the resulting larger delta between the NU and DU assay values. Since lower tails assay results in more fissile material in the LEU per tonne of NU, the relationship is positive: lower tails = less NU. Since lower tails require more separative work, the relationship to SWUs is negative: lower tails = higher SWUs.

Longer rollout of FRs results in some increase in LWR operational years, and therefore higher NU consumption and higher SWU requirements. However, the impact is smaller than a direct shift in the FR introduction date because only a portion of the reactors are impacted.

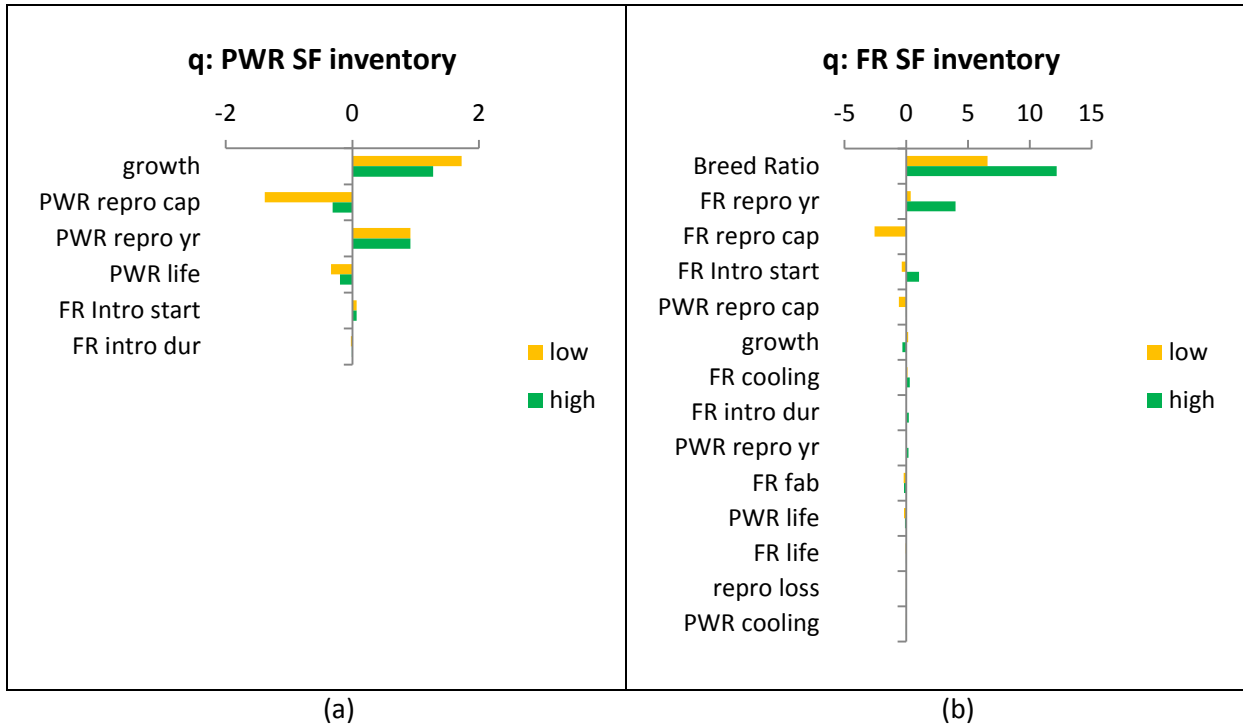
Figure 8.2-2: The sensitivity of (a) FR fuel fabrication, and (b) Pu flow through fabrication, to each parameter



FR fuel fabrication is greatly affected by the FR breeding ratio and the overall growth rate. Raising the FR breeding ratio requires a reduction in burn-up of the FR fuel, resulting in an increase in FR fuel fabrication requirements. The energy demand growth determines the total FRs needed. The FRs are more sensitive to growth than the PWRs because the total fleet size is larger later in the scenario when only FRs are present. A later introduction of FRs reduces the total FR reactor fleet period of operation (and number of units), as does a slower FR introduction (to a lesser extent). Both reduce fuel needs (negative correlation). The impact on FR life on fuel fab is also a negative correlation, as a shorter reactor life requires more start-up cores per total reactor years of operation. This impact is slightly non-linear, as it is driven by the fraction of start-up cores over reactor life. PWR life is also slightly impacted due to changes in the Pu isotopic vector.

The energy demand growth has the largest impact on Pu flow through fuel fabrication due to the increased number of operating FRs. The FR breeding ratio has a large, non-linear impact on Pu flow through fuel fabrication. This is primarily due to FRs with a breeding ratio below 1.0 (negative breeding gain) having no fertile blankets, impacting the quality and the quantity of the Pu (lower fissile content). There is a sharp transition in Pu quality when blankets are added and the breeding gain becomes positive due to the higher quality Pu produced in the blankets and recycled back into the driver fuel. As the breeding gain increases beyond 1.0, the Pu quality stabilises with only small isotopic content changes resulting in much smaller changes in the amount of Pu needed in the fresh fuel burn-up. Other factors impacting Pu flow through fuel fabrication are driven by total FR fuel needs, as described in the previous paragraph.

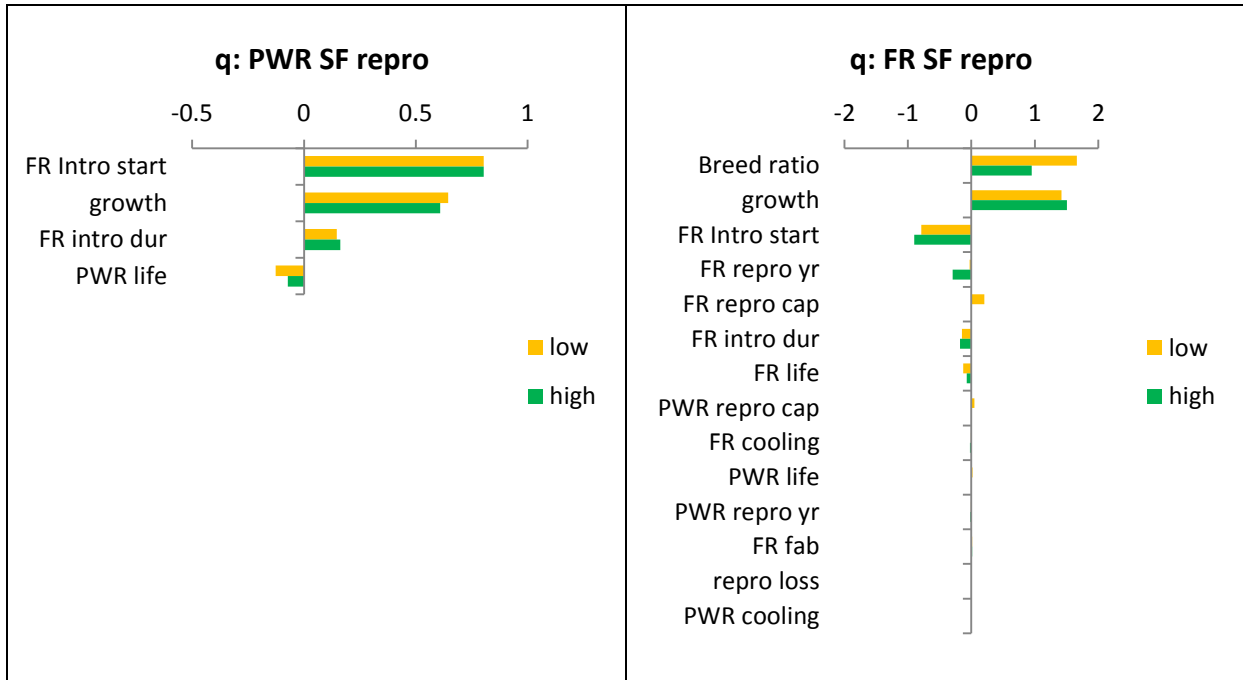
Figure 8.2-3: The sensitivity of (a) PWR UOX, and (b) FR MOX spent fuel storage, to each parameter



The largest impacts on PWR SF inventories are the rate of growth and the timing and rate of reprocessing. The inventory values used here are the maximum values (see Table 4.1-1), which results in large changes in the output parameter for unit changes in the input parameter (see the horizontal axis scale). PWR SF inventories are sensitive to reprocessing parameters because the inventories are first growing, then being depleted as the scenario proceeds and slower or delayed reprocessing allows more inventory growth and therefore a higher maximum before depletion occurs.

The largest impacts on FR SF inventories are the FR breeding ratio and the timing and rate of reprocessing. The increase in fuel consumption from increasing the breeding ratio results in a build-up of more FR SF in storage over the lifetime of the FR fleet. This property is non-linear because with increased SF production there is relatively higher amount of FR SF in cooling storage that is not yet available to be reprocessed. Low reprocessing rates can result in a net accumulation over time. A delayed start to FR SF reprocessing results in a temporary accumulation of cooled SF, while an accelerated start has limited impact because the inventory is controlled by the minimum cooling time. The growth in energy demand does not have as much of an effect on the maximum FR SF in storage as the maximum PWR SF in storage. This result is due to the balance between the extra SF produced by the larger FR fleet and the extra SF required to be reprocessed to fuel the larger fleet. The negative effect that the high case has on the maximum FR SF in storage is due to the reprocessing of additional FR SF to fuel the first cores of FRs that begin operation near the end of the scenario, when the maximum FR SF in storage occurs.

Figure 8.2-4: The sensitivity of (a) PWR UOX, and (b) FR MOX spent fuel sent to reprocessing, to each parameter

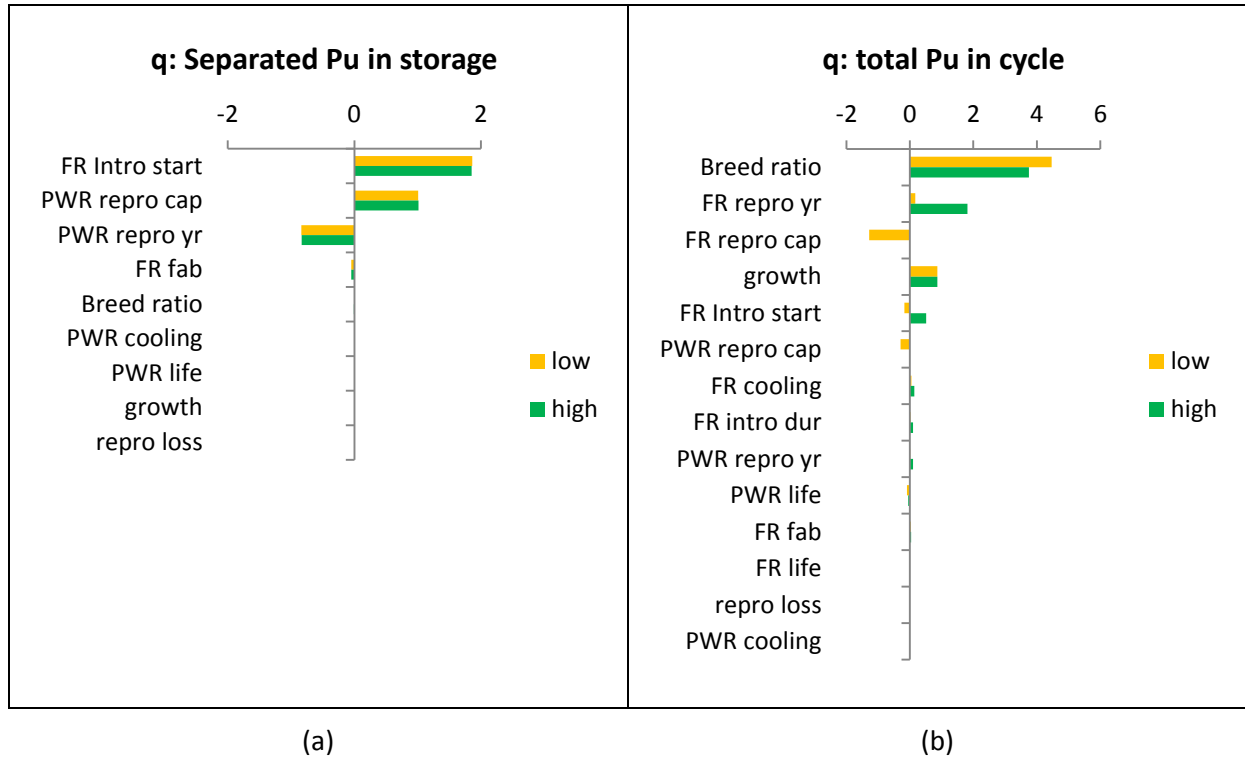


(a)

(b)

The total mass of PWR and FR SF reprocessed depends significantly on the average reactor mix over the scenario. If the introduction of the FRs is delayed or takes longer to complete, more PWR SF is produced and reprocessed, while less FR SF is produced and reprocessed. Energy demand growth increases both types of reactors, and thus more SF of both types is reprocessed. This property is non-linear because with higher growth there is relatively more SF in cooling inventory that is not yet available for reprocessing. The FR breeding ratio has the largest impact on the total FR SF reprocessed due to its effect on the FR fuel consumption and SF production. Increasing the FR breeding ratio increases the annual FR fuel consumption and SF production, with increased FR reprocessing to meet the increased fuel consumption. The impact is non-linear because the plutonium quality also varies with breeding ratio, which affects the FR reprocessing required to meet the FR fissile requirements. When reactor life is decreased, the number of final cores increases, increasing the amount of SF of that type of reactor. The FR SF total reprocessing is also impacted by when reprocessing begins, since a later start day means fewer total years of reprocessing.

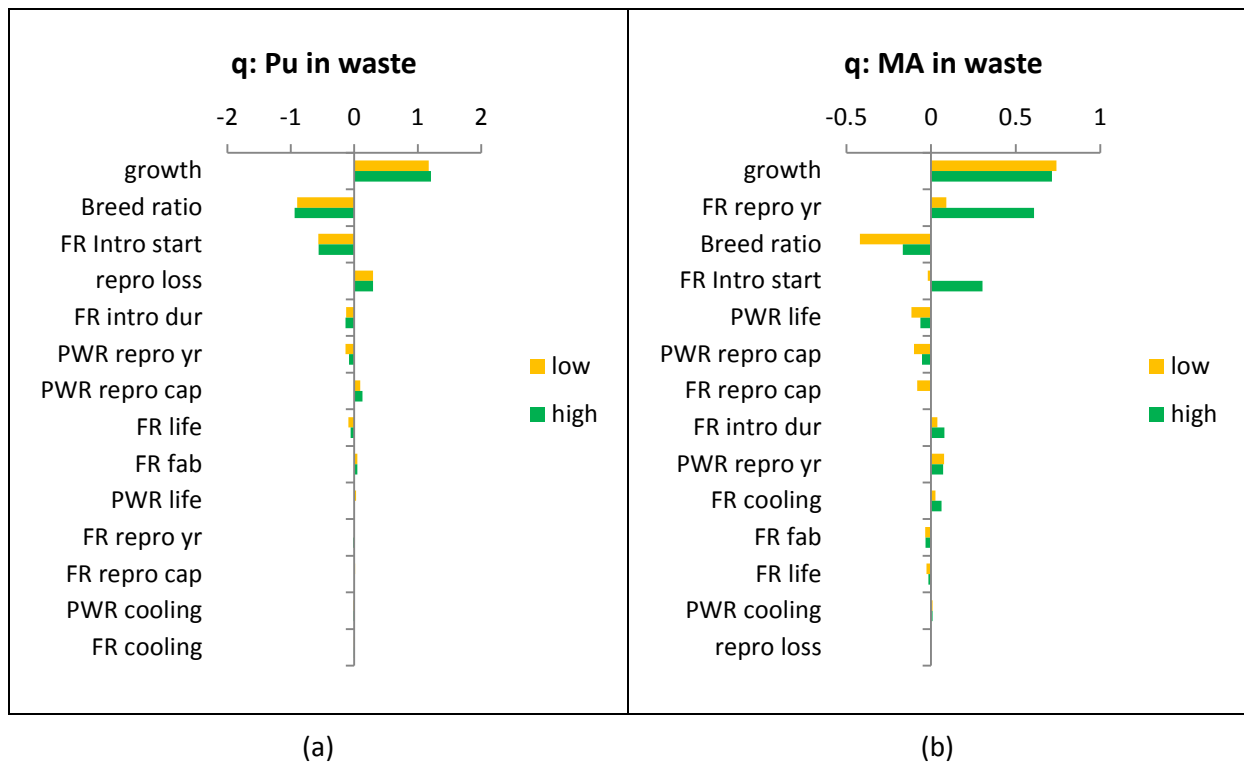
Figure 8.2-5: The sensitivity of (a) separated Pu, and (b) total Pu in the cycle, to each parameter



A later start date for FRs increases the available Pu because more is produced by the PWRs. Changing the PWR reprocessing capacity and start year change the amount of PWR-produced Pu that is separated, which impacts the Pu availability.

Total Pu in cycle is increased by having more Pu in SF storage, which in turn is impacted by breeding ratio, FR introduction, etc. A larger growth rate increases the total fleet size and total material inventory of the fleet, including the Pu in cycle. Two other significant inputs that impact Pu in cycle in a non-linear way are the FR reprocessing year and reprocessing capacity, where less total FR reprocessing either due to a later start or a lower capacity result in broken scenarios where Pu is added to the system from an external source to “fix” the scenario (see Section 4.4), with the side effect of increasing total Pu in the fuel cycle.

Figure 8.2-6: The sensitivity of (a) Pu, and (b) MA, in the waste, to each parameter



The final two tornado diagrams show the impact of input parameters on the total Pu and MA in waste. In both cases, growth in energy demand is the most significant input parameter causing an increase in the total PWR and FR SF reprocessed, which in turn increases the amount of Pu and MA in waste. The breeding ratio is another significant input parameter, with a non-linear impact where a lower breeding ratio (net negative Pu production) results in more Pu and MA in the waste. This is again due to the change in Pu quality, with a burner system needing more Pu to be separated to fuel the FRs. The burner FR SF has a higher ratio of fertile Pu, requiring higher Pu content in the FR fuels. The higher burn-up of burner FR fuel produces more MAs that end up in the waste stream. Delaying the start of the FR SF reprocessing plant to year 105 results in significantly more MA in waste due to the addition of PWR-produced Pu to make up for the lack of Pu from FR SF reprocessing. The FR SF from PWR-produced Pu, which has significantly more MAs than the equilibrium FR SF, is subsequently reprocessed with the MAs going to the waste stream.

8.3. Concluding remarks

Fuel cycle analysis is part art and part science. What to include in an analysis depends on the types of decisions to be supported; scenario definitions and code selection need to be based on this end goal. Analyses must integrate the political, economic, social and environmental constraints, intercepting the impact (and possibly the consequences) of an uncertain economics on energy futures. At the same time, scenario models must also include key phenomena of the physical systems being modelled. In the area of the nuclear fuel cycle, these phenomena can range from subatomic physics behaviour to the interaction of systems of complex facilities over many decades.

This report has attempted to approach the intersection of the art of scenario definition and the science of complex nuclear systems modelling in a systematic manner to contribute to improving both the design of scenario analysis tools and the application of scenario analysis to the advancement of nuclear energy systems.

While the content of this report is most directly applicable to fuel cycle systems analysts, it should also be of value to researchers and developers working of the different parts of the nuclear energy system by providing a systems perspective for their work. In particular, since the reactor is the intersection of front-end and back-end of the fuel cycle, the core designer should be among the end users of this report. Many of the adopted approaches, and obtained results, meaningful for the analyst of the fuel cycle, should be shared by the reactor core designer for improving conception, design and performances of the reactor core, in coherence with the design and performances of the whole fuel cycle. Designers of fuels, reprocessing systems, waste forms, disposal systems, etc. are also potential end users. All can benefit from both a systems perspective and an understanding of the impact of uncertainty on design choices. Finally, through its application by systems analysts, this report can be of significant benefit to energy policy makers and those making development and deployment decisions for nuclear energy facilities, as an improved understanding of potential system response under uncertainty can contribute to improving system performance and resiliency.

Appendix A: Specifications of alternative options and models used in the benchmark studies

An overview of alternative options and models used in the benchmark study is provided in the present Appendix with the aim of helping to understand the results and the discrepancies underlined.

The following items have been considered:

- designs of the FR considered in the study;
- differences between smoothed and unsmoothed COSI results;
- impact of the reprocessing strategy;
- impact of the FR core breeding gain.

A.1. FR designs considered in the study

A reference fast reactor design was provided (Chapter 2) for the benchmark study. The design considered, based on the European Fast Reactor (EFR) system, is the same as the one adopted in the previous OECD/NEA benchmark study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes [1].

Concerning the parametric study, new designs (ESFR, SFR-V2B, JSFR, etc.) have been considered as alternatives. The present appendix summarises the characteristics of the different cores adopted. The data provided here may help to understand the discrepancies obtained with respect to the reference case (based on EFR design).

A.1.1. SFR V2B design

Within the study, CEA has proposed to consider a new Sodium Fast Reactor (SFR) design [2,3], referenced as SFR V2B. This MOX SFR fuel is only composed of a fissile part (no fertile blanket in the basic design) with an initial equivalent ^{239}Pu content of 11.0%wt., corresponding to an initial Pu content going from 16.0%wt. to 17.0%wt. in function of the Pu isotopic composition. The reactivity coefficients are similar to those given for the EFR case (Table 2.1-2 in Chapter 2). The main characteristics of this core are listed in Table A.1.

Table A.1: Data compilation for the benchmark study

		FR (SFR V2B)
Fuels/blankets		
Fissile burn-up	GWd/tHM	100
Minimum cooling time	y	2
Fabrication time	y	2
Fresh fuel ²³⁵ U enrichment	%	0,25
Equivalent Pu content	%	11.0
Cores		
Electrical nominal power	GW	1.45
Efficiency	%	40
Load factor	-	0.786
Heavy metal masses	t	72,5
Breeding gain		≈1
Cycle length	EFPD	410
Core fraction (fuel)		1/5

A.1.2. ESFR designs: Working Horse (WH) and CONF2 designs

During the 7th EURATOM framework program CP-ESFR, a large (3600 MWth) sodium cooled fast reactor (ESFR) loaded with MOX fuel has been proposed by CEA [4]. Within the project, the proposed configuration (Working Horse - WH core) has been revised and optimised in order to reduce the positive void reactivity effect (SVRE). After the optimisation process, a new configuration (CONF2) has been proposed to the international community for safety investigations. Currently, CONF2 core was adopted in another EU project (e.g. PELGRIMM Project [5]).

The WH core layout and axial structure are presented in Figure A.1. The core is composed of 453 fuel subassemblies (SAs) subdivided into two zones in order to flatten the core power profile in the equilibrium cycle. The average Pu content (BOL) is 14.5%wt. for the inner zone and 16.9%wt. for the outer zone. The active height is 100 cm. Above the active zone, an upper axial blanket (UAB, 7.6 cm height) and upper gas plenum (UGP, 15 cm height) are placed. Just above the gas plenum, there are plugs, a 15 cm height Na plenum zone, and the upper steel structure. The lower part is composed of a lower axial blanket (LAB, 30 cm height) and a lower gas plenum (LGP, 91.3 cm height). In the reference configuration (WH core) the blankets are made of steel in order to improve the reflection of neutrons towards the core.

The average burn-up of ca. 100 GWd/tHM is reached after 2050 equivalent full power days (efpd) for a power density of 206 W/cm³. A loading strategy of 1/5 is considered. In Figure A.1, the positions of the nine Diverse Shutdown Devices (DSD, containing B₄C with 90%

of ^{10}B) and of the 24 Control and Shutdown Device (CSD, containing natural boron carbide) are also indicated.

The WH core shows positive void effects (core and core plus upper zone) as also indicated in [6]. Within the CP-ESFR project, optimisations have been carried out to reduce the void effects without changing the other core characteristics (power distribution, Doppler constant, reactivity swing). The optimised configuration (CONF2) shows the same core layout as WH core (Figure A.1) but a different axial structure (see Figure A.2 [6]).

The most effective way to reduce the sodium void reactivity effects is to increase the core leakage rate under voided conditions by modifying the region above the core by the adoption of a larger Na plenum (60 cm instead of 15 cm) with an absorber layer of B_4C above (30 cm). To enhance this effect, the Na plenum has been shifted close to the core by eliminating the UAB and reducing the UGP height (from 7.6 to 5 cm). In addition, the lower reflector is replaced by fertile material in order to reduce reflection back to the core under voided conditions. This last modification improves the Pu balance of the system as illustrated in the scenario study, as well.

The WH core has been design to work as breakeven (breeding ratio ca. 1) while CONF2 shows slightly breeder characteristics (BR ca. 1.2 due to the fertile material in the LAB). The core characteristics are summarised in Table A.2.

With respect to these two configurations, several Minor Actinides (MA) loading strategies have been considered (see Par. 6.2): 1) homogeneous loading (different MA shares) in WH and CONF2 and 2) heterogeneous strategy with different shares of MA loaded in the LAB.

Figure A.1: WH configuration: (a) core layout; (b) axial structure

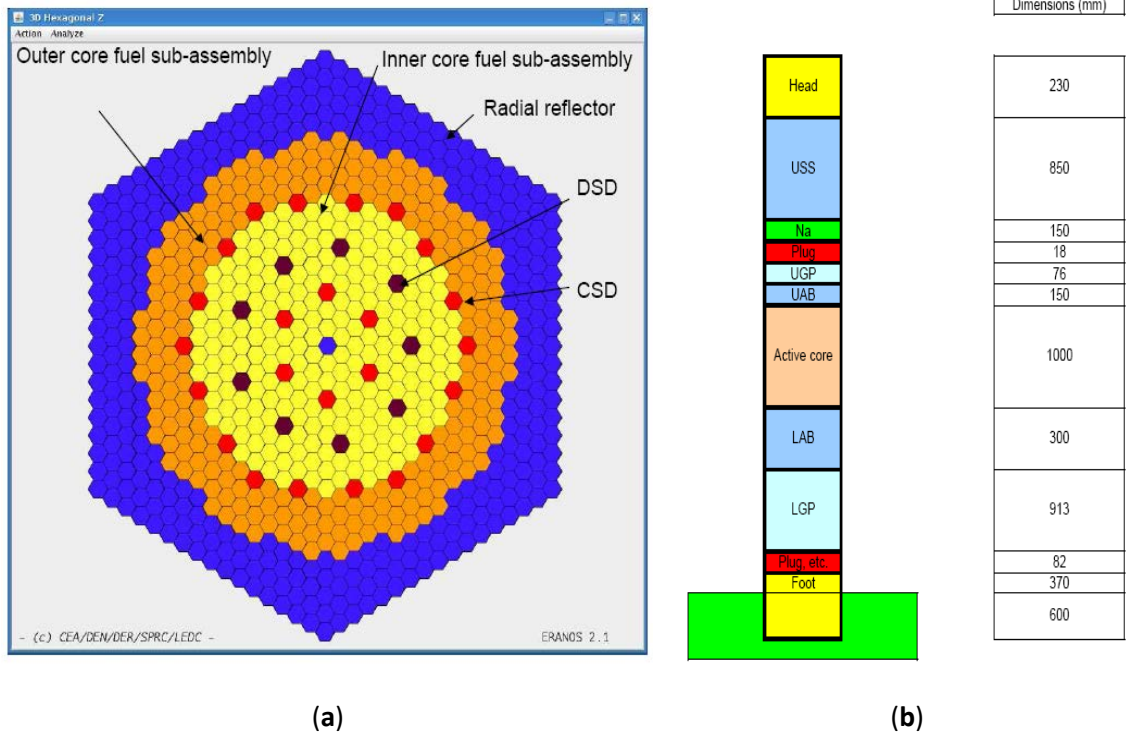


Figure A.2. CONF2 axial structure

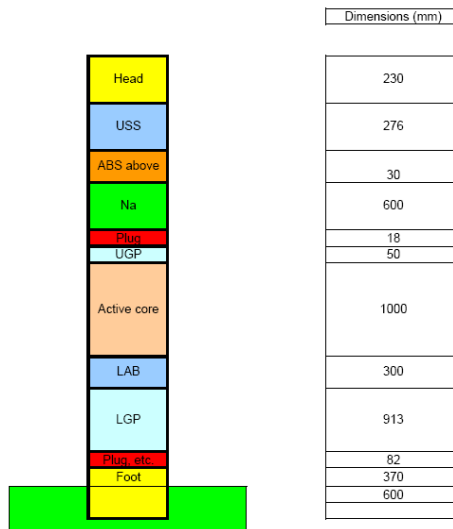


Table A.2: ESFR WH and CONF2

		ESFR WH	ESFR CONF2
Fuels / blankets			
Burn-up core	GWd/tHM	Ca. 100	Ca. 100
Burn-up LAB		-	Ca. 10
Fresh fuel ²³⁵ U enrichment	%	0.25	0.25
Equivalent Pu content	%	10.7	10.9
Cores			
Nominal power	GWth	3.6	3.6
Heavy metal in core	t	74	74
Heavy metal in LAB		-	26
Breeding ratio		≈1	≈1.2
Cycle length	EFPD	410	410
Core fraction (fuel)		1/5	1/5

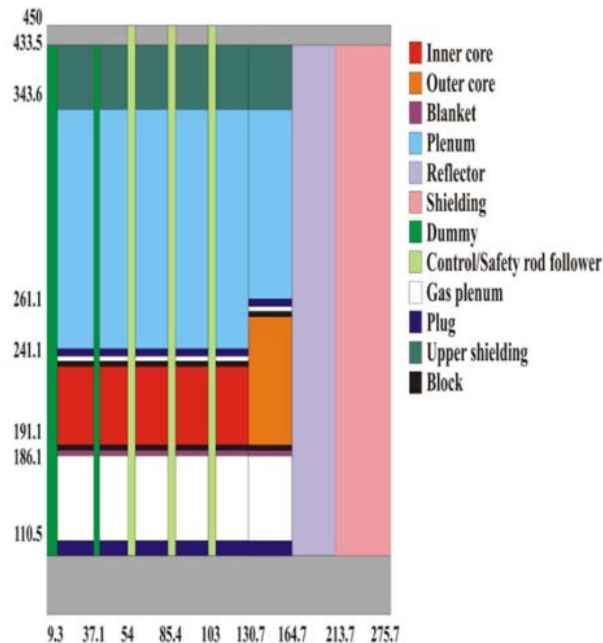
The configuration described here (WH and CONF2) has been used also by CIEMAT for providing updated cross-section libraries for the core zone as indicated in Section 6.2.3.

A.1.3. ASTRID-like: MA burner

As an alternative to MA management with homogeneous and heterogeneous loadings in ESFR cores, KIT has considered the introduction of a mixed fleet ESFR plus ASTRID-like burners (see Par. 6.2). The ASTRID-like burners are dedicated to MA management.

The ASTRID-like burner model was based on the French ASTRID CFV concept developed by CEA with support of AREVA and EDF [7,8]. The models (1200 MWth) have been assessed for achieving a conversion ratio (CR) of 0.5-0.7 (by increasing the transuranic content in fuel) without a significant power reduction as compared to the ASTRID original design (1500 MWth and CR=1). Detailed description is available in [9]. In order to compensate for the reactivity increase due to the use of large Pu content and partly the deterioration of safety parameters, the core height has been reduced by about 20% (50/70 cm height for the inner/outer core regions, respectively) and the internal blanket in the inner core has been removed. The thermal power has been reduced to 1200 MWth in order to keep the same power density as in the French ASTRID design. In addition, the height of the lower axial blanket has been reduced to 2 cm in order to decrease the breeding capability). The RZ layout is shown in Figure A.3.

Figure A.3: ASTRID-like burner 2D (RZ) model (dimension in cm) [9]



The transuranic (TRU) vector considered (Table A.3) corresponds to one used in the past in European studies associated to the inert matrix fuel in the design of the European Facility for Industrial Transmutation (EFIT), i.e. a typical MOX Spent Nuclear Fuel (SNF) reprocessed 30y after its irradiation in a PWR with a burn-up rate of about 45 MWd/kg [10]. This vector, if compared with the composition evaluated for the German SNF inventory by GRS [11] or the one corresponding to 400 TWhe PWR park with a 60 GWd/t burn-up after 50 years cooling [12], shows the worst Pu quality and the larger ^{241}Am and ^{244}Cm contents, making the choice very conservative for the aim of the study. Such a vector, indeed, can be assumed as a reasonable estimation of the average fuel composition during the fuel cycle taking into account the disappearance of ^{239}Pu and the formation of ^{241}Am during the cycle. The main characteristics of the burners are given in Table A.4. The Pu enrichment has been set to get the required CR under the condition that the systems are critical after they are loaded with fresh fuel and operate for 3 irradiation cycles.

The MA burner model considered in the benchmark is one of ASTRID-like burners developed at KIT [9]. Alternatives, e.g. oriented to Pu burning, have been already considered at KIT in scenario studies [9,11,13]. All models have been assessed with great attention on the safety and on the burning performances of the systems [9].

Table A.3: Pu and MA vectors: reference composition and alternatives [9]

Isotope	Ref. [10]	GRS [11]	IAEA [12]
	Pu vector (wt. %)		
²³⁸ Pu	3.7	2.45	4.45
²³⁹ Pu	46.4	52.49	57.17
²⁴⁰ Pu	34.1	32.19	28.49
²⁴¹ Pu	3.8	0.9	0.6
²⁴² Pu	11.9	11.97	9.29
	MAs vector (wt. %)		
²⁴¹ Am	75.5	63.8	62.09
^{242m} Am	0.3	0.1	0.04
²⁴³ Am	16.1	10.7	8.62
²³⁷ Np	3.9	24.4	28.53
²⁴³ Cm	0.1	0.0	0.03
²⁴⁴ Cm	3.0	0.5	0.30
²⁴⁵ Cm	1.1	0.5	0.33
²⁴⁶ Cm	0.1	0.0	0.05

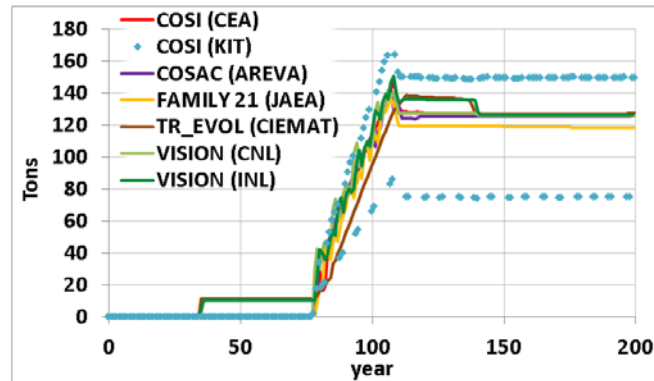
Table A.4: Main parameters of the ASTRID-like burners [9, 13]

	MA burner
Power (MWth)	1200
Cycle length (efpd), No. cycles	365, 5
MA/Pu ratio	1/2
Max burn-up Inner/outer core (MWd/kgiHM)	100/133
Initial HM (t)	18.6
Conversion Ratio (TRU)	0.55
Pu content (%)	
Inner core	23
Outer core	25
²³⁹ Pu eq.	13
MA content (%)	
Innercore (axial blanket)	11.8 (10.6)
Outer core (axial blanket)	12.8 (10.6)
Mass in Core (kg)	
Initial Pu/ Initial MA	4400/2170
Discharged Pu/ Discharged MA	4170/1450
Burning capability (kg/TWhth)	
Pu burning	-4.2
MA burning	-14.5
Main safety parameters (BOL)	
K _D (pcm)	-275
Void Core ($\Delta\rho$, \$)	5.9
Void core + plenum ($\Delta\rho$, \$)	-0.3

A.2. Difference between smooth and unsmooth COSI results

CEA, KIT and ENEA participated in the benchmark study by using the COSI (CEA) code. However, as indicated in Chapter 3 for the reference scenario, some differences in the results have been underlined. In particular for some figures-of-merit CEA has provided “smoothed results” while KIT has provided “batches-oriented” results. One example is Figure 3.3-1 (see Figure A.4 below) where the Pu inventory in plants is depicted.

Figure A.4: Base case scenario – Pu inventory in plants (Figure 3.3-1 in Chapter 3)



Both results are valid. The difference comes only from the way chosen on presenting the results.

In the following part, a small description of the smoothing procedure is included.

In the COSI environment, smoothing procedure may be used for presenting the results of a fleet modelled by means of a macro-reactor [14]. The nominal power and the fuel mass loaded in the core of the macro-reactor correspond to the sum of the nominal power and the fuel mass of each reactor of the fleet. This model is equivalent to a pool of reactors loaded and unloaded simultaneously.

If the irradiation length is longer than one year, it leads to oscillations in the material balances evaluated with physical post-processing. More precisely, an alternation of null values (or lower values) and positive values (or greater values) can be observed for material balances of facilities linked to the macro-reactor (fabrication plant, reprocessing plant ...). The null values correspond to the year without fuel loading or unloading in the macro-reactor whereas the positive values correspond to the material requirements per irradiation cycle.

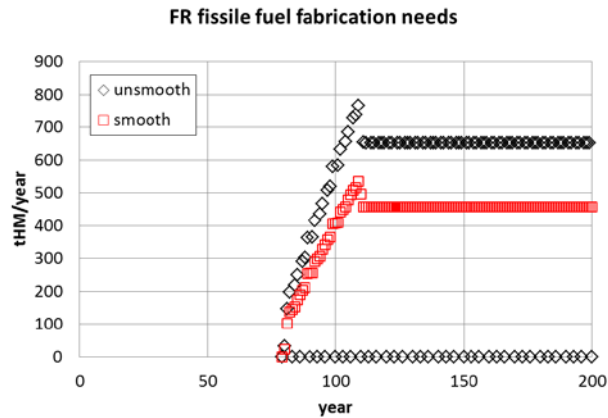
When the oscillations consist of an alternation of null values and positive values, to obtain the annual material balances, it is necessary to smooth the results by using the irradiation length (noted as L) in order to represent annual results instead of results per cycle. The irradiation length in year can be obtained as follows:

$$L(\text{year}) = L(\text{EFPD}) / (365.25 \times f(\%))$$

with L the irradiation length expressed in year or in equivalent fuel power day and f the load factor.

To smooth the results, the user has to use the values of material balances in the excel files automatically generated with results post-processing by COSI, to suppress the null values and to divide the remaining values by the irradiation length expressed in year [14].

One example is indicated in Figure A.5 concerning the fissile annual fabrication needs for ESRF fleet (cycle length > 1 year).

Figure A.5: Comparison smooth and unsmooth results: FR fissile fuel fabrication needs (ESFR cores)

When the oscillations of the result consist in an alternation of lower and greater value (spent fuel storage, Pu inventory in plants and in storages), the smoothing procedure simply consists in taking the average value.

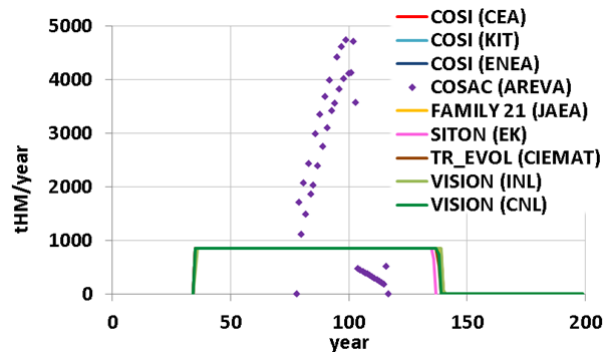
A.3. Impact of the reprocessing strategy

In a scenario study it is possible to reprocess spent fuel according to the need in Pu for the fresh fuel fabrication without any anticipation, or to impose a constant reprocessing. Due to scenario codes specificities, different strategies have been adopted. This appendix aims at describing the resulting differences on the scenario results.

A.3.1. PWR spent fuels reprocessing strategy

In the benchmark, the UOX spent fuel reprocessing is constant and fixed at 850 tHM/y. However, with COSAC, as it is not possible to fix the reprocessing at a constant value, UOX spent fuels are reprocessed according to the need in Pu for FR fuel fabrication. Thus, UOX spent fuels are reprocessed starting in year 79, to feed the FR fuel fabrication. It increases up to 4730 tHM/y in year 99 and decreases in year 103 when the first FR spent fuel is reprocessed (see Figure A.6).

Figure A.6: Base case scenario - PWR UOX spent fuel reprocessing (Figure 3.2-1 in Chapter 3)



The consequences of this strategy are the following:

- the UOX spent fuel storage increases until year 78 when it reaches 66,000tonnes;
- there is no reprocessed uranium until year 79;
- as the reprocessing is adjusted to the need in Pu, there is no stored separated Pu;
- there is no loss at reprocessing before year 79 so that nothing goes to the waste before year 79.

It is noteworthy that the absence of Pu storage between reprocessing and fabrication leads to a lower ^{241}Am content in FR fresh fuel.

The impact of a reprocessing of the UOX spent fuels according to the need in Pu is represented on Figures A.7 to A.10.

Figure A.7: Base case scenario – PWR UOX spent fuel storage (Figure 3.2-3 in Chapter 3)

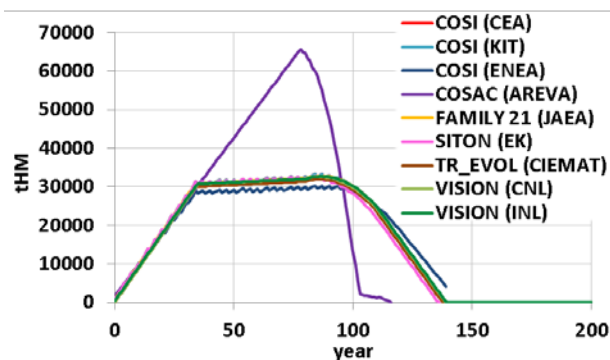


Figure A.8: Base case scenario – PWR UOX spent fuel reprocessing (Figure 3.2-1 in Chapter 3)

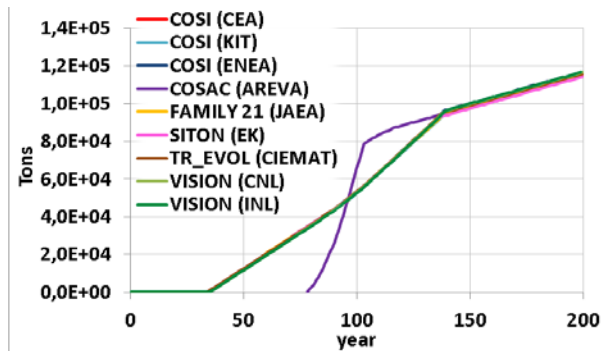


Figure A.9: Base case scenario – Pu interim storage (Figure 3.2-7 in Chapter 3)

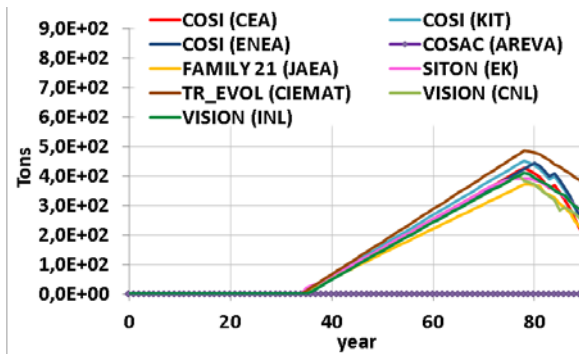
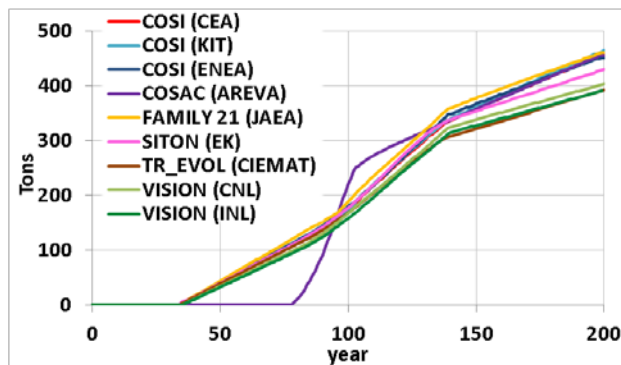


Figure A.10: Base case scenario – Reprocessed U interim storage (Figure 3.2-6 in Chapter 3)

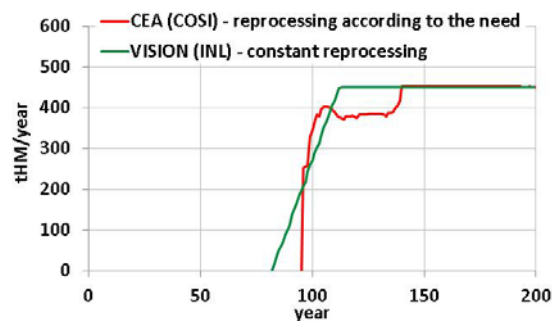


It is noteworthy that the results during the transition phase are the only ones impacted. The reprocessing of the UOX spent fuel according to the need in Pu has no impact on the results when the equilibrium is reached.

A.3.2. FR spent fuels reprocessing strategy

In the benchmark specifications, FR spent fuels are reprocessed according to the need in Pu for the fresh fuel fabrication without any anticipation. The reprocessing strategy adopted with TR_EVOL, VISION and SITON is different: the FR spent fuel reprocessing starts between years 82 and 87 (depending on the code) and is fixed at 450 tHM/y. To illustrate the impact of this reprocessing strategy, the results obtained with COSI (reprocessing according to the need) and VISION (constant reprocessing) are represented in this part of the appendix. The FR spent fuel reprocessing strategy is represented on Figure A.11; note that in both cases the FR spent fuel reprocessing is stabilised at 450 tHM from year 140.

Figure A.11: Base case scenario – FR spent fuel reprocessing



Due to a higher FR spent fuel reprocessing up to year 140, the FR spent fuel storage is reduced (see Figure A.12). In the VISION (INL) case, the FR spent fuel storage stabilises around 1010 tHM/y, which represents a 70% reduction in comparison with the COSI (CEA) results. On the other hand, there is more Pu separated at reprocessing than Pu required for the fresh fuel fabrication, which results in an interim Pu storage (see Figure A.13).

Figure A.12: Base case scenario – FR spent fuel storage

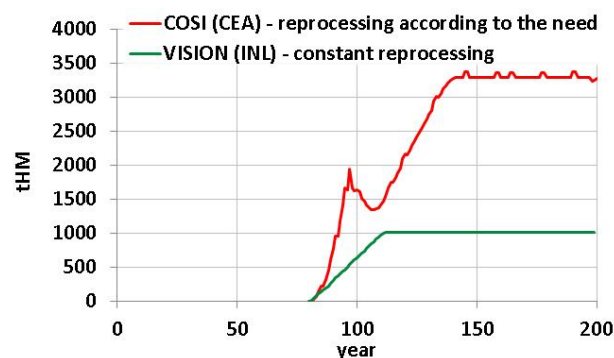
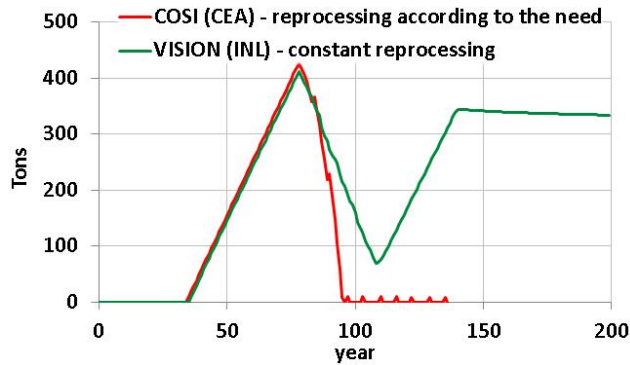


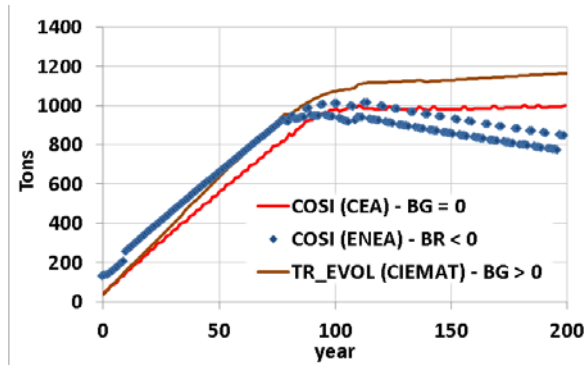
Figure A.13: Base case scenario – Pu interim storage



A.4. Impact of the FR core breeding gain

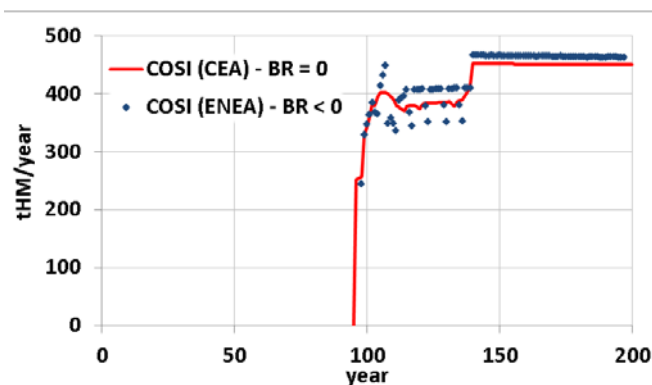
Differences on the FR reactor breeding gain can explain some discrepancies observed in the base case scenario. The impact of the breeding gain is mainly visible on the total Pu inventory (see Figure A.14). When the equilibrium of the FR fleet is reached, breakeven cores lead to the stabilisation of the Pu inventory while the Pu production in breeder cores leads to an increase in the Pu inventory and the Pu consumption in core with a negative breeding gain leads to a decrease in the Pu inventory.

Figure A.14: Base case scenario – Pu inventory in cycle (BG = Breeding Gain)



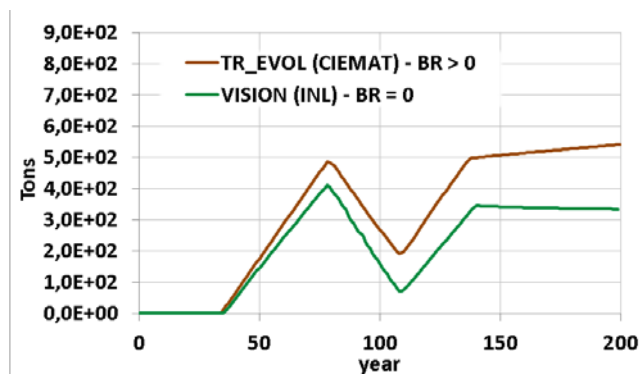
When FR spent fuels are reprocessed to meet the need in Pu for fresh fuel fabrication, the breeding gain has an impact on the reprocessing need. Indeed, for example with a negative breeding gain, the Pu consumption during irradiation leads to a smaller Pu content in the spent fuels than the one required for the fresh fuels. In that case, it is necessary to reprocess a little more spent fuel (Figure A.15) to meet the need in Pu for fresh fuel fabrication.

Figure A.15: Base case scenario – FR spent fuels reprocessing (BG = Breeding Gain)



When the FR spent fuel reprocessing is constant, the FR breeding gain impacts the amount of separated Pu in interim storage. For example, with a positive breeding gain, the Pu production during irradiation leads to a greater Pu content in spent fuels than in the fresh fuels. This surplus of Pu is stored after reprocessing (see Figure A.16).

Figure A.16: Base case scenario – Pu in interim storage (BG = Breeding Gain)



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Appendix B: Impacts of anticipation/no anticipation option about used fuel reprocessing through various scenario codes

B.1 Description of the issue

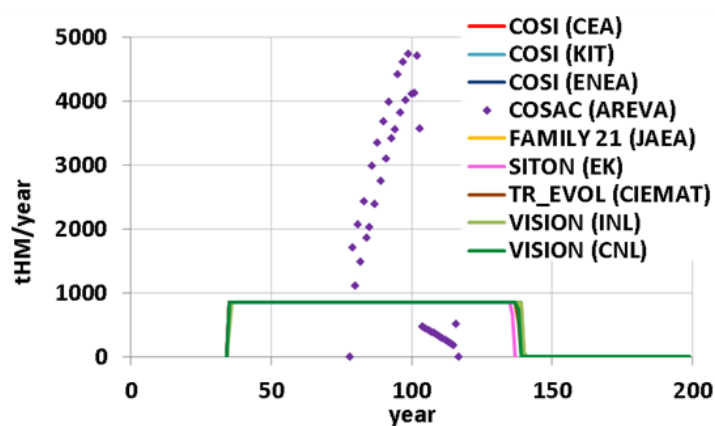
All the calculations in the present benchmark, except COSAC calculations, are representative of steady reprocessing capacities. Therefore, a fixed capacity had to be defined in this study. As to cope with the plutonium needs during Fast neutron Reactor deployment, a capacity value was fixed at 850 tonnes of heavy metal per year for all the codes except COSAC.

For COSAC, the calculations are representative of the plutonium needs year after year. In other words, COSAC doesn't anticipate the future needs in plutonium.

This difference of strategy between COSAC and the other codes can be seen on the below graph. For the other codes than COSAC, the UOX spent fuel reprocessing capacity is steady (850 tonnes per year). For COSAC, the UOX spent fuel reprocessing capacity is varying during the scenario:

- from zero ton per year before the deployment of FRs (that means there is no Pu need at FR fuel fabrication because no FR is yet in operation in the fleet);
- to a maximum value of around 4 800 tonnes per year when the FRs are being deployed (the maximum value of Pu needs at FR fuel fabrication is reached when first full cores are still required for the last newly commissioned FRs in addition to a maximum of reload batches for the almost all already commissioned FRs).

Figure B.1: PWR UOX spent fuels reprocessing



B.2. Impacts on the outputs

The difference in the way reprocessing capacities are simulated between COSAC and the other codes has impacts on several outputs. Indeed, all the outputs in relation with the operation of a spent fuel reprocessing plant will be impacted by the way reprocessing capacities are simulated in a code.

In practical terms, this means all the storages connected with a spent fuel reprocessing plant are potentially impacted:

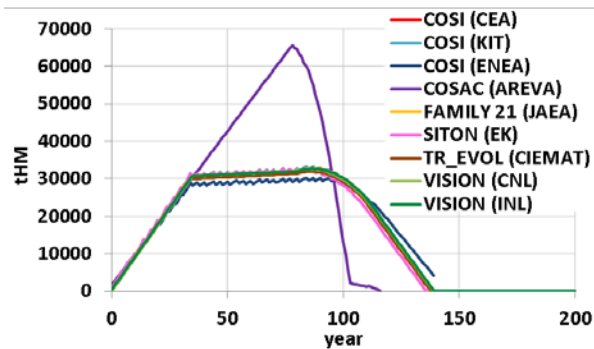
- storage from where some nuclear materials are sent into a spent fuel reprocessing plant, such as a spent fuel storage;
- storage into where some nuclear materials are sent from a spent fuel reprocessing plant, such as a reprocessed uranium interim storage and a plutonium interim storage.

B.2.1 Impact on spent fuel storage:

The next graph gives an example of how much the mass contained in a UOX spent fuel storage can be significantly different between COSAC and the other codes, since this type of storage provides some UOX spent fuel to be reprocessed by a spent fuel reprocessing plant.

- In the case of COSAC: the UOX spent fuel storage is increasing from the beginning of the scenario until 2 years before the beginning of the FR deployment in year 80 (2 years are indeed necessary to reprocessing and fabrication operations before MOX fresh fuel can be loaded into FRs). After year 80, the UOX spent fuel storage decreases quickly as it is intensively sent to the UOX spent fuel reprocessing plant to fulfill the important Pu needs required by the newly commissioned FRs. It decreases more slowly when the FR fleet is completely commissioned, that means after year 108, as 2 years being necessary for reprocessing and fabrication operations before the full core loading of the last newly commissioned FRs.
- In the case of the other codes: the UOX spent fuel storage is first increasing as quickly as for COSAC from the beginning of the scenario until the UOX spent fuel reprocessing plant begins to be operated, then it grows slowly from year 35 when the UOX spent fuel reprocessing plant begins to be operated until year 80 when the PWR fleet begins to be decommissioned and to be replaced by FRs, then it decreases as long as the PWR fleet is decommissioned and in the meantime the UOX spent fuel reprocessing plant is still operating, until the UOX spent fuel storage is completely cancelled in year 140.

Figure B.2: PWR UOX spent fuels storage

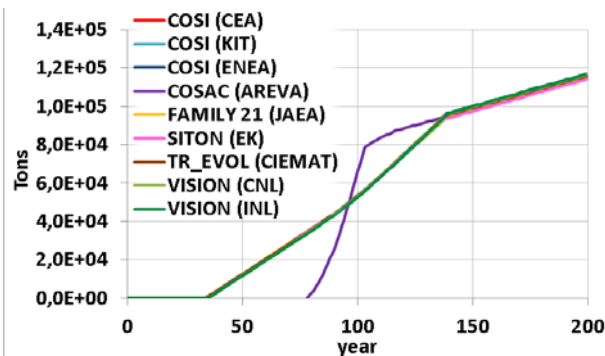


B.2.2 Impact on a reprocessed uranium interim storage:

The next graph gives an example of how much the masses contained in reprocessed uranium interim storage can be significantly different between COSAC and the other codes, since the reprocessed uranium comes from the operation of a spent fuel reprocessing plant.

- In the case of COSAC: the UOX spent fuel reprocessing plant doesn't operate from the beginning of the scenario until year 78 (i.e. 2 years before the beginning of FRs deployment), therefore there is no reprocessed uranium produced and the interim storage of reprocessed uranium stays empty until year 78. From year 78, the intensive operation of the UOX spent fuel reprocessing plant leads to the accumulation of reprocessed uranium in the reprocessed uranium interim storage. This accumulation is as much quick as the Pu needs are high and the operation of the UOX spent fuel reprocessing plant is intensive. Therefore, from year 78 to year 108, the reprocessed uranium interim storage increases quickly. After year 108, i.e. once the important Pu needs required by the FR deployment has been completely satisfied, it increases more slowly.
- In the case of the other codes: the operation of the UOX spent fuel reprocessing plant starts in year 35, so the reprocessed uranium interim storage is empty until year 35. After year 35, as the annual capacity of the UOX spent fuel reprocessing plant remains steady, the accumulation of reprocessed uranium grows regularly from year 35 to year.

Figure B.3: Reprocessed U interim storage

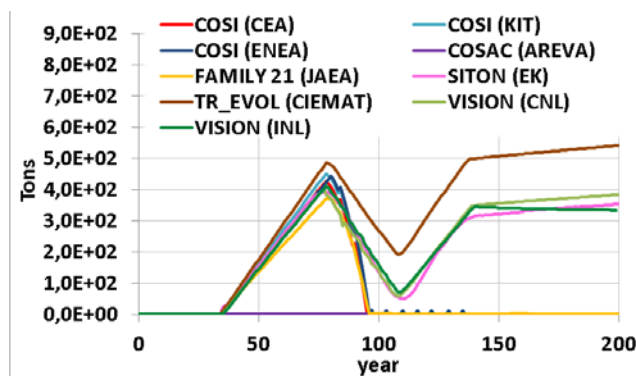


B.2.3. Impact on a plutonium interim storage

The next graph gives an example of how much the mass contained in a plutonium interim storage can be significantly different between COSAC and the other codes, since plutonium comes from the operation of a spent fuel reprocessing plant.

- In the case of COSAC: there is never any plutonium interim storage, whatever the scenario is, because the plutonium produced by a spent fuel reprocessing plant will be immediately used for manufacturing fresh MOX fuel to be loaded in a thermal- or a fast-neutron reactor. This is another manner to say the reprocessing capacities are following the Pu needs at MOX fresh fuel fabrication, without any anticipation of the future MOX fresh fuel fabrication needs.
- In the case of the other codes: the plutonium produced by a spent fuel reprocessing plant can be stored some time before being used for manufacturing MOX fresh fuel to be loaded in a thermal- or fast-neutron reactor at a later time. Therefore, there can be a plutonium interim storage resulting from a plutonium production without immediate use.

Figure B.4: Pu Interim storage



B.3. Discussion about the advantages and disadvantages of the two options

The behaviour of the two types of codes (with and without the anticipation of reprocessing) offers both advantages and drawbacks.

- For the code type without anticipation (like COSAC code): there is no possibility of extra production of plutonium, in other words the production of plutonium is continuously matching the current needs. This type of codes is worthwhile from a safeguard point of view, especially for non-proliferation concerns, as it doesn't allow the accumulation of separated plutonium. Moreover, this type of codes doesn't require any assumption from the user about *when* the reprocessing plant must start neither about its rated *capacity*: the only information that matters is the total plutonium mass contained in the used fuel and being available for reprocessing. In addition, as reprocessing is always driven by the current needs in plutonium in this type of codes, the time lag between reprocessing and fabrication then loading is shortened at its minimum value (usually 2 years) so that the decay process from ^{241}Pu (fissile isotope) to ^{241}Am (capture isotope) is quite limited after reprocessing operation. On the contrary, the main disadvantage of such type of codes is the huge variations induced in annual capacities of a reprocessing plant. These huge variations are not realistic from an economics point of view. These variations are illustrated in § 1: industrial investment could be jeopardised if the reprocessing plant had an annual capacity that would rise from 0 to 4,800 tonnes per year in a 20-year period of time, and then decline from 4,800 tonnes per year to zero in another 20-year period of time.
- For the code type with anticipation (i.e. the other codes than COSAC in this study): they offer several advantages. From an economic point of view, it looks sounder to have a steady annual capacity of reprocessing rather than huge variations of the capacity over the time. Indeed, when a reprocessing plant is built, it is sized for a given capacity. If this rated capacity is not fully used during all the operation of the plant, then the capital investment realised at the building is not properly used. Second, an anticipated vision of the future contributes to take the right decisions at the right time. For instance, if a huge amount of plutonium is expected to launch a future fleet of reactors (like Fast neutron Reactors - FRs), then this future can be prepared by building a reprocessing plant a long time before the new reactor fleet is launched. But this second advantage is double-edged because it is hardly possible to know how much time in advance the future can be forecast. If building a reprocessing plant a long time before the plutonium will actually be used allows minimising the rated capacity of the plant and also maximising the risk that future will not be the one expected. Another disadvantage of reprocessing in advance lies in the accumulation of separated plutonium during all the years before the plutonium will actually be used. This accumulation of separated plutonium does not go in the way of non-proliferation policy, and the so-produced stock of plutonium will require serious monitoring arrangements to protect it from misappropriation. A last disadvantage is about the decay process leading to the formation of ^{241}Am (poison isotope) from ^{241}Pu (fissile isotope): the earlier the used fuel is reprocessed, the longer the decay process will occur so that, when the separated plutonium will finally be used in a reactor, it might contain a significant proportion of ^{241}Am .
- Conclusion: the two types of codes – with and without the anticipation of reprocessing- give a view on the two extreme approaches to address the issue of the availability of plutonium to make possible the launch a new fleet of reactors

(like FRs) when it is necessary. The first one (without anticipation) prevents from creating any stock of separated plutonium, but the variations of its capacity during its lifetime can be a barrier for the investment because these variations may not be realistic from an economic point of view. The second one (with anticipation) creates a stock of separated plutonium and obliges the investor to take a risk in the future by investing a capital earlier than the purpose of this investment will happen, but it allows to operate a reprocessing plant at a steady and lower capacity during its lifetime, and it also allows to evaluate and to simulate a combination of *capacity* and *starting date* for the reprocessing plant. Finally, it must be noticed the two types of codes lead to very similar final results in terms of scenario calculations, and using both types of codes is a way of comforting a given study.

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