3. Benchmark specification [1]

3.1. Depletion part

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

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

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

3.1.1. PWR UOX fuel composition (wt%)

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

Table 10: Initial composition for UOX fuel

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

3.1.2. Irradiation history for PWR UOX fuel

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

3.1.3. PWR MOX fuel composition (wt%)

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

Table 11: Initial composition for MOX fuel

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

3.1.4. Irradiation history for PWR MOX fuel

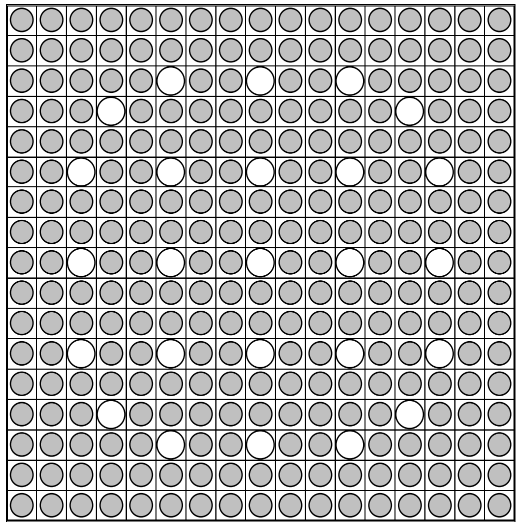
The calculation is made in one step for a burn-up of 60 GWd/t (1 760 EFPD) and a cooling time of 5 years.

3.1.5. Other hypotheses for UOX and MOX in PWR

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

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

Figure 18: Scheme of the fuel assembly



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

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

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

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

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

3.1.6. MOX FR fuel composition (wt%)

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

Table 12: Initial composition for MOX Na-FR fuel

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

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

The composition of the fuel assembly is the following:

Table 13: Composition for MOX Na-FR fuel

	Composition (volume)
Fuel	37.51
Na	32.94
Structure	23.59

3.1.7. Irradiation history for MOX Na-FR fuel

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

3.2. Transition scenarios

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

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

The common hypotheses are:

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

Table 14: Data compilation for the benchmark study

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

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

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

3.2.1. Scenario 1 - Open cycle

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

Figure 19: Scenario 1 – Flow chart

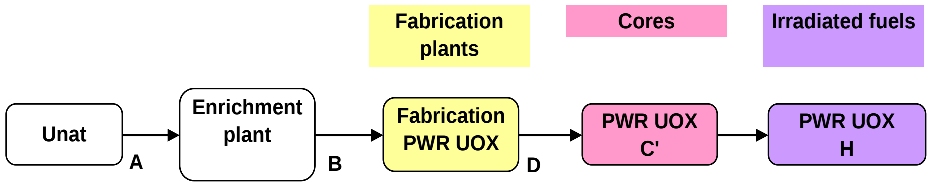
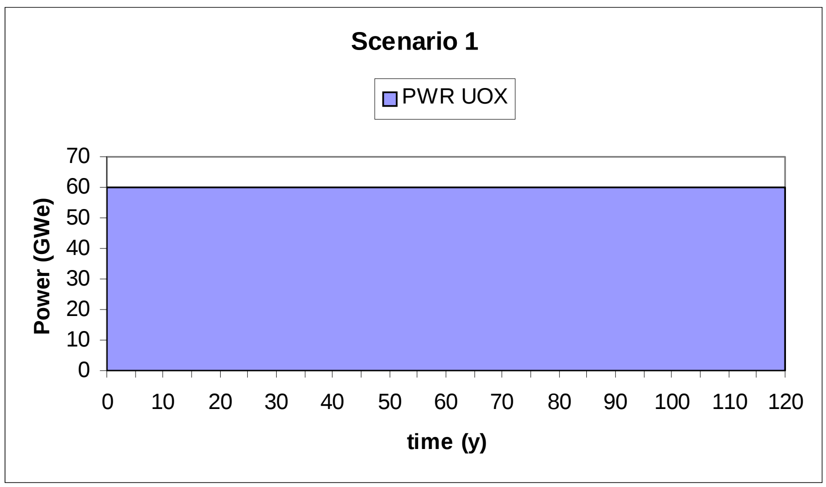


Figure 20: Scenario 1 – Installed capacity



3.2.2. Scenario 2 - Monorecycling of plutonium in PWRs

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

Figure 21: Scenario 2 – Flow chart

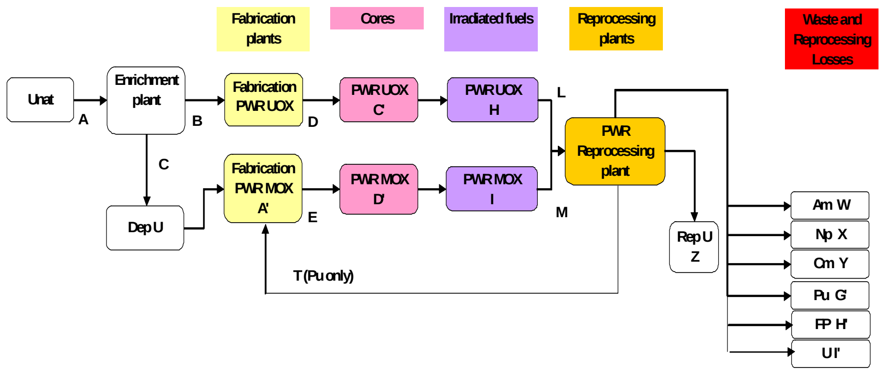


Figure 22: Scenario 2 – Installed capacity

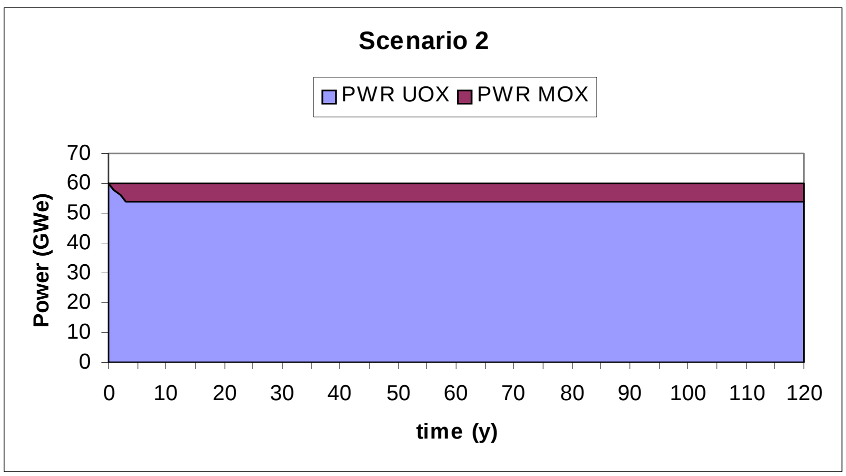


Table 16: Scenario 2 – Installed capacity

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

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

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

Figure 23: Scenario 3 – Flow chart

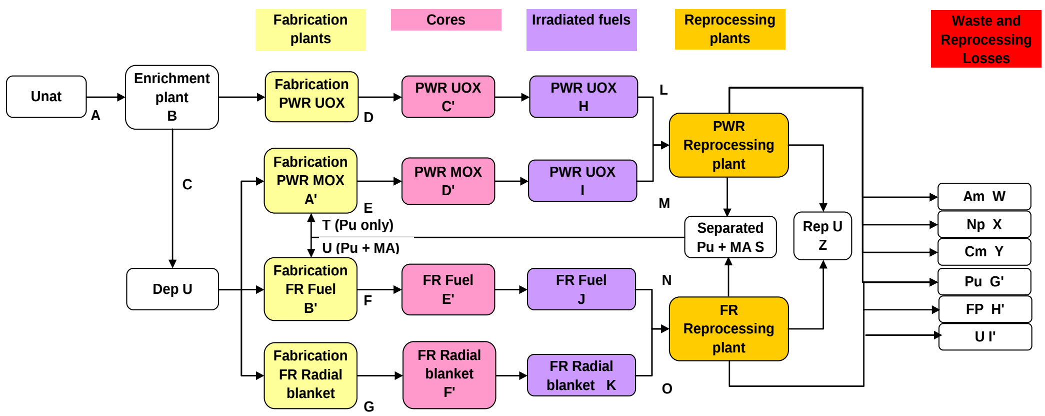


Figure 24: Scenario 3 – Installed capacity

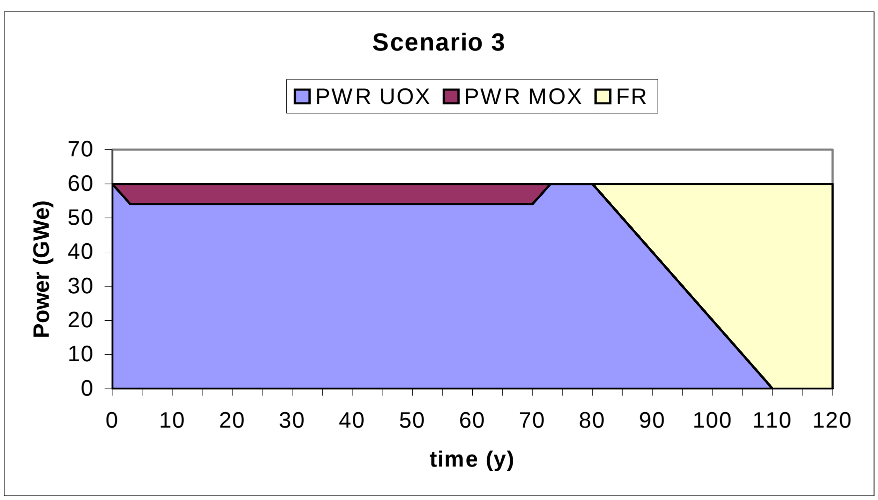


Table 17: Scenario 3 – Installed capacity

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

3.2.4. Other specifications

Reactors

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

First core

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

Enrichment plant

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

Fabrication plant

- scenarios 1, 2 and 3

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

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

Table 18: Reactivity coefficients

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

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

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

Reprocessing plants

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

Table 19: Reprocessing plant assumptions

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

Spent fuel

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

3.2.5. Results expected to be reported

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

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

Reference

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