#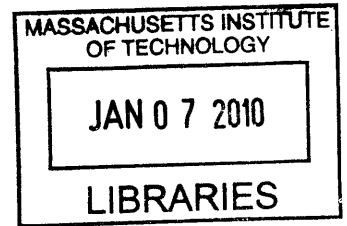 Impact of Alternative Nuclear Fuel Cycle Options on Infrastructure and Fuel Requirements, Actinide and Waste Inventories, and Economics.

by

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Impact of Alternative Nuclear Fuel Cycle Options on Infrastructure and Fuel Requirements, Actinide and Waste Inventories, and Economics.

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Laurent Guérin

Abstract:

The nuclear fuel once-through cycle (OTC) scheme currently practiced in the U.S. leads to accumulation of uranium, transuranic (TRU) and fission product inventories in the spent nuclear fuel. Various separation and recycling options can be envisioned in order to reduce these inventories while extracting additional energy and sending the ultimate waste to a repository. Choosing one of these options has direct implications for the infrastructure requirements, natural uranium consumption, actinide inventories in the system, waste repository needs and costs. In order to account for the complexity of the nuclear enterprise, a fuel cycle simulation code has been developed using system dynamics (CAFCA). An economic module was added using spreadsheets.

Four main advanced fuel cycle schemes are assessed here within the context of the US market: 1) the twice-through cycle scheme (TTC): single-pass plutonium recycling in thermal spectrum LWRs using Mixed OXide (MOX) fuel; 2) Multi-recycling of TRU in sodium-cooled fast spectrum burner cores, characterized by a fissile conversion ratio lower than 1 (FBu); 3) Multi-recycling of TRU in sodium-cooled fast breeders with a conversion ratio of 1.23 (FBr); and 4) A two-tier scenario: a TTC scheme is practiced as a transition scheme to fast reactors. The base case scenario assumes annual nuclear energy demand growth rate of 2.5% from 2020 on. The technologies for plutonium separation as well as MOX fuel fabrication are assumed to be available in 2025 while the first commercial fast reactors, as well as the possibility to recycle their spent fuel, are assumed to be available in 2040. For fast reactors, the cores are assumed to be TRU fueled, and the technology to separate the minor actinides is supposed to be available at the latest 5 years before deployment of fast reactors. Limits are applied on the building rate of reprocessing plants, which are also subject to a 80% minimum life-time loading factor requirement.

It is found that, despite its higher cost, at the end of the century, the TTC scheme (single Pu-MOX recycle) does not lead to large improvements in terms of natural uranium consumption (16%), repository needs (considering both fission products and MA from reprocessing facilities, and spent MOX fuel) and TRU inventory reduction (although some shifting of TRU from storage to reactors occurs). This is especially significant

because it is the only advanced fuel cycle option that can be deployed in large scale in the next few decades. However, if the primary reason for introduction of the more expensive fast reactors is resource enhancement and/or control of TRU in the nuclear waste, thermal reactor recycling allows the introduction of fast reactors to be delayed by 20-25 years. Moreover, once fast reactors are introduced, their deployment is accelerated compared to a 1-tier FR scenario. However, the two-tier scheme is the most expensive scheme as it combines the requirements of both the MOX technology and the FR technology.

Sensitivity analyses were performed in order to assess the impact of secondary parameters. It is found that whatever the growth rate assumed, LWRs remain a significant part of the system at the end of the century, decades after fast breeders are introduced. The reason is the fissile materials required for fabrication of start-up cores considerably affect the rate at which fast reactors can be deployed. As a result, the choice of the core design (compact core vs. large core) may be as significant as the choice of the conversion ratio. For example, the breeder scenario (CR=1.23) may lead to the same cumulative natural uranium consumption reduction (by 2100) as the self-sustaining reactors (CR=1.0) while leading to larger TRU inventory in the system and requiring greater fast reactor fuel reprocessing capacity. Allowing fast reactors to start with uranium only cores was not considered, as it will likely limit resource enhancement benefits of fast reactors. Still, in general, the higher the conversion ratio, the greater the fast reactor installed capacity, hence the greater the savings in natural uranium. Conversely, the best reduction in TRU from the OTC amount is obtained by the lower conversion ratio (45% for a pure burner with conversion ratio 0.0 by 2100). Doubling the minimum cooling time before reprocessing for all fuel types from 5 years to 10 years slows down the deployment of the fast reactors and therefore reduces their share in the total installed capacity. This is almost equivalent to replacing breeders with fast reactors with a conversion ratio of 0.75. Finally, the results show that starting the separation of the TRU 10 years prior to introduction of the fast reactors instead of 5 years provides a mid-term advantage (faster initial deployment) that vanishes within 25 years.

In the long term, the fast reactor penetration results are insensitive to the assumed industrial capacity to build reprocessing facilities for the base case or at lower nuclear energy growth rates. However, the assumed industrial capacity can be a real constraint if the nuclear energy growth rates are 4% or higher.

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Acknowledgements

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Nomenclature

$\%_w$	Weight percentage
$	U.S. dollars
ABR	Advanced Burner Reactor or Actinide Burner Reactor
ALMR	Advanced Liquid Metal-cooled Reactor
B, bn	Billion
CAFCA	Code for Advanced Fuel Cycle Assessment
CF	Capacity Factor
COE	Cost of Electricity
CONFU	Combined Oxide Non-Fertile and UO_2 fuel
CR	Conversion Ratio
DLCC	Dynamic Levelized Cost of Capital
DLCOE	Dynamic Levelized Cost of Electricity
DLOMC	Dynamic Levelized O&M Cost
DOE	Department Of Energy
dmnl	Dimensionless
DU	Depleted Uranium
EFPD	Effective Full Power Day
FBr	Fast Breeder
FBu	Fast Burner
FFF	Fertile-Free Fuel
FP	Fission Product
FR	Fast Reactor
FRP	Fast Reprocessing Plant
GFR	Gas-Cooled Fast Reactor
GWe	Gigawatt electric
GWe-yr	Gigawatt electric year
HEU	Highly-Enriched Uranium
HLW	High Level Waste
HM	Heavy Metal
kgHM	Kilogram of Heavy Metal
kgIHM	Kilogram of Initial Heavy Metal
kWe	Kilowatt electric
kWh	Kilowatt hour
LCOE	Levelized Cost of Electricity
LEU	Low-Enriched Uranium
LWR	Light Water Reactor
LWRmf	LWR licensed to be loaded with MOX fuel
LWR2	LWR of Generation II
LWR2mf	LWR2 licensed to be loaded with MOX fuel
LWR3	LWR of Generation III
MA	Minor Actinides
mills	One thousandth of a dollar
MLF	Minimum Loading Factor

MT	Metric Tons
MTHM	Metric Tons of Heavy Metal
MTIHM	Metruc Tons of Initial Heavy Metal
MWd	Megawatt day
MWe	Megawatt electric
MWt	Megawatt thermal
MOX	Mixed Oxide Fuel
MOX2	MOX for LWR2
MOX3	MOX for LWR3
NWF	Nuclear Waste Fund
NWPA	Nuclear Waste Policy Act
O&M	Operation & Maintenance
OTC	Once-Through Cycle
PUREX	Plutonium and Uranium Refining by EXtraction
PWR	Pressurized Water Reactor
R&D	Research & Development
RU	Recovered Uranium
SF	Spent (Nuclear) Fuel
SWU	Separation Work Unit
tHM	Metric Tons of Heavy Metal
tIHM	Metric Tons of Initial Heavy Metal
ThRP	Thermal Reprocessing Plant
TRU	Transuranics
TTC	Twice-Through Cycle
UOX	Uranium Oxide
WACC	Weighted Average Cost of Capital
YM	Yucca Mountain

Chapter 1

1. Introduction

1.1 Motivation

After a few decades of stagnation, not to say a quasi-moratorium in the US, nuclear power is increasingly considered as an important energy option by the government as well as public opinion, to a point that the expression "nuclear renaissance" is now often heard. Several trends have led to such a shift, among which are the concerns over climate change, the lack of stability or reliability of most of the oil/gas producer countries, the upward trend of the price of natural gas as well as its volatility, and above all the ever-growing electricity demand.

However, the nuclear energy industry entails some specific characteristics that call for appropriate policies:

- The relatively slow R&D pace, the long lifetime as well as the large capital costs of the facilities result in time scales comparable to the human generation time.

- Nuclear proliferation and waste issues are both irreversible features. Indeed, an option that has been chosen, developed, and used in the U.S. is implicitly considered acceptable for promotion and thus very likely to be adopted all over the world. But, fundamentally, transuranics and fission products created in a reactor must be either recycled or disposed of as waste, as there is currently no technology to make them inert in a short time.

- Safety aspects also entail some irreversibility, as a single large accident can be detrimental for the whole industry.

Those three externalities naturally explain the strictness of the nuclear regulatory framework, which in turn reinforces the relative rigidity and inertia that characterize the sector. The entire fuel cycle actually consists of multiple steps, each of them being an industry with its own limits in terms of flexibility and capacity, and its own uncertainties.

As a result, when it comes to nuclear power, policies cannot be based on a short-term approach, or "trial-and-error" approach, but must reflect well-coordinated, long-term strategies, with a time horizon of at least 50 years. Decisions taken in this field have large implications and are often irreversible. Beside technical and financial aspects, the rigidities and uncertainties stemming from regulatory concerns make even more obvious the necessity to anticipate developments decades in advances, as evidenced by the Yucca Mountain project case.

1.2 Background

Decision parameters are numerous. Of first importance is the choice of fuel cycle. At the basic level the choice is between a once-through fuel use or the application of recycling of the useful contents. This can be expanded due the possibility of recycling in thermal or fast spectrum reactors into four main schemes: 1) The once-through cycle, as currently practiced in the US: the UO_2 fuel is burnt only once in thermal reactors and then sent to storage until disposal becomes possible. 2) The twice-through cycle: the plutonium contained in spent UO_2 fuel is separated in reprocessing plants and then recycled as Pu-U mixed Oxide (MOX) in thermal reactors. The separated uranium can also be recycled. The spent MOX fuel is finally sent to a repository. 3) The fast, closed fuel cycle: the spent UO_2 fuel discharged from thermal reactors is reprocessed; the TRU extracted is then burnt in fast reactors. The used fuel discharged from fast reactors is reprocessed in turn and recycled in fast reactors, and so on. 4) An hybrid scenario between 2 and 3 is a two-tier scenario in which the spent MOX fuel is recycled in fast reactors instead of being disposed of.

Many secondary parameters exist. To begin with, the design of fast reactors as well as their conversion ratios, from 0 (pure burners) to 1 (self-sustaining reactors) or even more than 1 (breeders). Such a choice has strong policy implications: deploying burners may reflect a policy focused on minimizing the overall transuranic inventory due to proliferation concerns while the choice of breeders aims at extracting as much energy as possible from natural resources. Other parameters include various introduction dates for the different technologies, and various sizes of the plants, fuel burn-ups, and reactor designs. The system is also subject to external constraints such as desired nuclear power

demand, industrial capacity for the building of the facilities, and level of economic profitability (minimum loading factors must be ensured for both fast reactors and reprocessing plants).

Outputs of main concern are the consumption rate of natural uranium, the size of spent fuel inventories (especially the actinides that are contained in them and represent. depending on the viewpoint, a potential source of energy, a proliferation concern or a long-term waste burden), the industrial needs (enrichment, reprocessing, fabrication), the inventories of various types of wastes, and finally the costs.

Due to the complexity of choices, nuclear policy makers must be provided with tools that enable them to clearly see the consequences of present decisions decades from now, and build realistic scenarios. As for any simulation models, such tools help formulate policy recommendations *a priori*, support them once decisions are taken following those recommendations, and eventually allow anticipation of R&D and industrial needs years in advance.

Steady-state models, assuming equilibrium states and maturity of the technologies, are indispensable and abundant. However, they by nature do not take into account real-world initial conditions of a technology as well as the time-dependent nature of the nuclear systems. As mentioned earlier, the nuclear industry entails long time scales, which means the state of equilibrium assumed in the static studies may actually require decades to be reached. Dynamic features such as building rate limits, economies of scale and spent fuel legacy might be reflected in such models, but in a very indirect way that leave too much latitude to the user: they generally propose a range of numbers (low/medium/high estimations) for each variable, which can be chosen by the user -involuntarily or regardless of overall consistency and hence realism. In addition, decay heat and transmutations pose additional time-dependant problems that cannot be captured in a static model. For instance, if plutonium is separated from the spent UO_2 fuel too early, the build-up of Americium over time may make necessary a second, costly reprocessing of this plutonium prior to its loading into a fast reactor.

Therefore, a complex, dynamic code has been developed at MIT. This Code for Advanced Fuel Cycle Assessment (CAFCA) is not unique, and appears to reflect a trend

of developing simulation tools for the fuel cycle in many research and industrial outfits. For example, the *Argonne National Laboratory* has been developing the *Dynamic Analysis of Nuclear Energy System Strategy* (DANESS) code, the *United States Department of Energy Advanced Fuel Cycle Initiative* (DOE-AFCI) is developing the *Verifiable Fuel Cycle Simulation Code* (VISION) while the French Atomic Energy Commission (CEA) is developing the *Commelini-Sicard* (COSI) Code. A benchmark assessment of CAFCA and these three codes was recently performed [L. Guérin et al., 2009].

Eventually, the purpose of such dynamic codes is naturally not to find a unique, "optimal" strategy but to provide insight into some of the processes and problems that may characterize the course of nuclear power over the next decades. Non-linear phenomena, path-dependencies and cumulative effects are among the aspects that cannot be accurately understood in the absence of such a model. Effects of timing of introduction of new technology and the size (capacity) of the introduced facilities can be better studied by dynamic system models.

1.3 Scope of the Work

The purpose of this work was to study the impact of different fuel cycle schemes for the next 100 years, through performing simulation of a set of scenarios with associated sensitivity analyses. The outcomes mentioned in the previous section are analyzed. Chapter 2 first describes the various fuel cycle schemes studied in this work and provides the data for the designs taken as references. The perceived advantages and drawbacks of each of the schemes are also noted.

Chapter 3 presents a simple model that provides some results for steady state and steady-growth scenarios. These results will provide guidance for the interpretation of the results provided by CAFCA (presented in Chapters 6 and 7) and will eventually underline the necessity of having such a code. Chapter 4 presents the CAFCA code, its capabilities as well as the methods, approximations and assumptions that underlie its implementation.

Chapter 5 presents the modeling approach used for the new capabilities of the code (various conversion ratios for the fast reactors, MOX fuel utilization, 2-tier scenarios, utilization of recovered uranium, U-price model). This chapter is to be read by whoever is

interested in the code in itself and may therefore be skipped by others. Chapter 6 deals with the economic module of CAFCA. The methods as well as the inputs used in the study are presented. Chapter 7 shows the results for the base case scenarios. The behavior of the nuclear system is analyzed under a growth rate of 2.5%/year. Four main scenarios are studied: the OTC scenario, a TTC scenario (Single Pu-U MOX recycle), a fast burner scenario and a fast breeder scenario, both using TRU fuel. Some sensitivity analyses are performed in Chapter 8. The effects of varying some of the key parameters with respect to the base case scenarios are studied. The parameters include the growth rate (1%/year, 4%/year), the conversion rate of the fast reactors (from CR=0.0 to CR=1.23) the cooling time (10 years vs. 5 years), the period of time between the introduction date of the thermal reprocessing and that the fast reactors (10 years vs. 5 years). Finally, the 2-tier scenario is analyzed and compared with both the 1-tier FR scenario and the TTC scenario. Conclusions and recommendations are given in Chapter 9.

Chapter 2

2. Fuel Cycle Options

2.1 Introduction

In addition to the Once-Through Cycle (OTC), four types of advanced fuel cycle schemes are explored in this study (1) Pu recycling in Light-Water Reactors ("MOX scheme"), (2) TRU recycling in fast metal cores of an Advanced Burner Reactor (ABR) of various conversion ratios from 0 to 1, (3) TRU recycling in fast metal cores of breeders, of which conversion ratio is higher than 1, (4) Two-tier scenarios, which are a combination of the MOX scheme and the FR scheme. We restrain the study of the MOX scheme to a "twice-through cycle", which means that the plutonium is recycled only once as MOX in LWRs. In this chapter, the various schemes are explained in detail, including assumptions about the timing of the fuel cycles and reactors (all normalized at 1GWe for comparison purposes), and the equilibrium properties of both fresh and spent fuels are presented. The assumptions provided in this chapter are those used over the entire study.

2.2 One-Through Cycle Scheme

2.2.1 Description of the OTC scheme

The one-through cycle scheme (denoted OTC) is the fuel cycle currently practiced in the U.S. and is considered as the base case. In this scheme, UO_2 assemblies are loaded in thermal reactors, irradiated, discharged and left in "cooling storage" (typically in reactor pools) for a few years ("minimum cooling time"). Finally, the spent fuel is sent either to interim storage or to a repository.

The fabrication of UO_2 fuel out of natural uranium requires a chain of front-end services. After mining, natural uranium is milled, resulting in a concentrate called yellowcake, containing about 80%$_w$ of uranium oxide (U_3O_8). This concentrate is then purified and converted into uranium fluoride (UF_6), which is the form used in both of the enrichment methods (gaseous diffusion and centrifugation). After enrichment from the natural level

(0.711 %$_w$ of ^{235}U) to a "low-enrichment level" (LEU, typically 4%$_w$), the uranium is finally sent to UO$_2$ pin fabrication plants. A byproduct of enrichment, depleted uranium (DU, typically containing 0.25%$_w$ of ^{235}U) is also generated in significant amounts (typically 9 kgHM of DU for 1 kg of LEU) but is not used in the once-through cycle scheme.

Figure 2.1 shows a representation of the one-through cycle scheme.
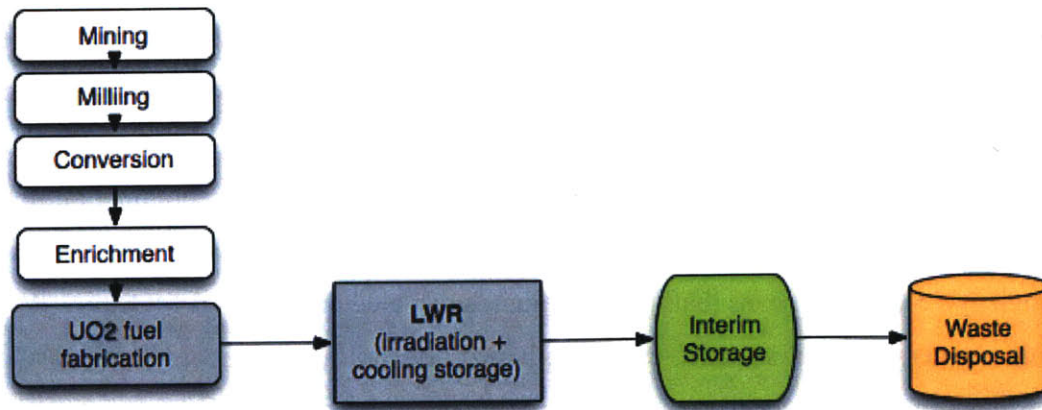
Figure 2.1 - Once-through cycle scheme

2.2.2 Data for the OTC scheme

For the sake of simplicity, we use a single model of an LWR and assume a unique set of parameters for the fuel cycle. Data are taken from [Hoffman et al., 2005]. In reality, there are many sizes of LWRs, and their fuel cycles also differ according to their fuel management.

Table 2.1 summarizes the characteristics of interest of the model considered (scaled to a 1000 MWe unit).

Table 2.1 - LWR: Plant and cycle descriptions

Thermal Power	2,966 MWt
Thermal efficiency	33.71%
Electrical output	1000 MWe
Cycle Length	500 EFPD
Number of Batches	3
Irradiation time	1,500 EFPD
Discharge Burn-up	50 MWd/kgHM

Table 2.2 shows the average fuel composition, at loading and after 5 years of cooling after discharge.

Table 2.2 - LWR fuel composition for a 100% UO$_2$ core

Compositions in %w of the Initial Heavy Metal load		
	Load	After Cooling
U	100% (4.23%$_w$ of ^{235}U)	93.56% (0.82%$_w$ of ^{235}U)
Pu	0	1.15%
MA	0	0.13%
TRU	0	1.28%
FP	0	5.16%

We make the approximation that the results are linearly scalable for any capacity factor within a narrow range. Table 2.3 summarizes the fuel mass flows for a capacity factor of 90% and a resulting residence time of 1,667 calendar days (nearly 4.5 years).

Table 2.3 - Fuel mass flows for a 100% UO_2 LWR core (with CF=90%)

Core mass at BOC (MTHM)	87.77	
Mass balance MTHM / GWe / Calendar year		
	Load	After Cooling
HM	19.500	18.494
U	19.500 (^{235}U : 0.825)	18.244 (^{235}U : 0.150)
Pu	0	0.225
MA	0	0.025
TRU	0	0.250
FP	0	1.006

2.2.3 Perceived advantages and drawbacks of the OTC scheme

- Perceived advantages:
  - The OTC is currently the least expensive scheme.
  - The OTC benefits from large commercial experience. The technology is already available, proven safe, and well known.
  - The risk from operations involved in OTC is minimal, as there is no separation of constituents and little handling of irradiated materials.

- Perceived drawbacks:
  - The OTC produces both short-lived wastes (dominated by fission products, in particular ^{137}Cs, ^{90}Sr) and long-lived wastes (dominated by actinides, in particular ^{241}Am, ^{240}Pu and ^{239}Pu, but also some fission products such as ^{99}Tc and ^{129}I). The existence of these nuclear wastes pose technical and institutional challenges as well as public acceptance issues.
  - In the OTC scheme, the final wastes still have considerable energy content

(only 5% or less of the energy content of the fuel is used, corresponding to less than 1% of the mined uranium), which can be criticized on either economical or ethical grounds.

2.3 Twice-Through Cycle Scheme (single pass MOX in thermal reactors)

2.3.1 Description of the TTC scheme

As an alternative to the traditional UO_2 assemblies, thermal reactors may be loaded with Mixed Oxide (MOX) assemblies. MOX is a mixture of Plutonium/Americium[1] (PuO_2/AmO_2) and depleted (or natural) uranium (UO_2).

Unlike uranium, plutonium can only be found in trace quantities in nature, but is formed in reactors. Although over half of this plutonium is fissioned (typically contributing to about one third of the energy produced over the irradiation of a UO_2 batch) or decayed *in situ*, significant amounts (typically over 1%$_w$, of which 60-70%$_w$ is fissile) remain in the discharged spent UO_2 fuel.

Hence the twice-through cycle (denoted TTC) is intrinsically a limited recycling strategy. After a minimum cooling time, the UO_2 fuel discharged from thermal reactors is sent to reprocessing plants where both the uranium (which typically constitutes 95%$_w$ of the used UO_2 fuel) and the plutonium are extracted. The minor actinides are sent along with the fission products to interim storage or disposal.

The plutonium is then sent to MOX fabrication plants (possibly collocated with the reprocessing plant) for MOX pin fabrication.

MOX assemblies are then loaded in thermal reactors for electricity production. Depending on the capability of the reactor and the policy choice, the core can be fully loaded with MOX assemblies, or only partially (typically 30% or 50%). In the latter case, the remainder is constituted of traditional UO_2 assemblies. As of today, very few of the existing U.S reactors, so-called Generation II reactors, are licensed to be loaded with MOX assemblies.

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[1] The –undesirable- presence of Americium is due to the decay of Pu241 into Am 241 (half-life of about 14.4 years).

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Due to buildup of non-fissile (even numbered) plutonium isotopes that would require larger plutonium enrichments negatively affecting reactivity feedbacks, the plutonium is generally recycled only once in *thermal* reactors, which is our assumption in this study. Therefore the spent MOX fuel is sent to interim storage after a minimum cooling time. In the case of a 1-tier scenario, this spent MOX fuel is to be sent to disposal eventually. Figure 2.2 shows a representation of the twice-through cycle scheme.
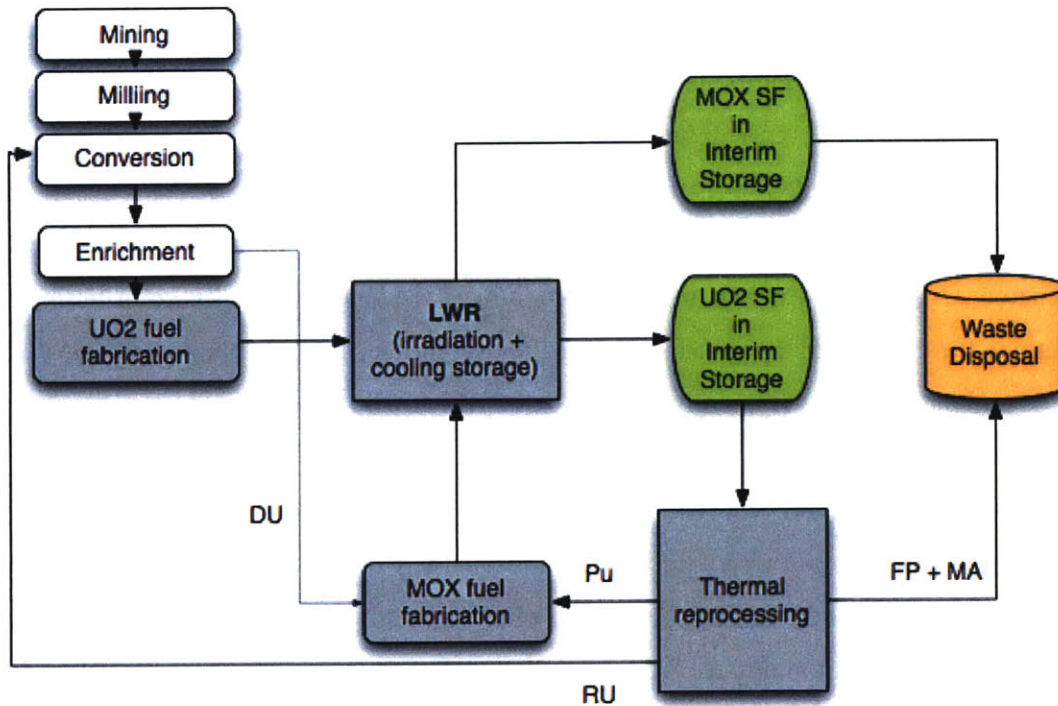
Figure 2.2 - Twice-through cycle scheme

2.3.2 Data for the TTC scheme

We use in this study the data for a typical PWR core loaded with about $30\%_w$ of MOX, as modeled in [de Roo et al., 2009].

The isotopic vector of the Pu/Am mix used as a make up feed for the MOX pin fabrication corresponds to typical spent UO_2 fuel with $4.5\%_w$ initial enrichment, 50 MWd/kgHM discharge burn up, decayed over 5 years in cooling storage. The plutonium is then extracted and decayed over 2 years (transit time in reprocessing plants plus fuel

fabrication time). This is slightly inconsistent with the data that we actually use for the all-UO$_2$ cores in LWRs (same discharge burnup but a lower (4.23%$_w$) initial enrichment).

The uranium mixed with the Pu/Am oxide is depleted uranium (0.25%$_w$ enrichment). The average discharge burnup is 50.3 MWd/kgHM for both UO$_2$ and MOX. After discharge, both MOX and UO$_2$ assemblies are cooled for 5 years[2].

Table 2.4 summarizes the compositions of the MOX and UO$_2$ assemblies.

Table 2.4 - MOX/UO$_2$ fuel compositions for a MOX core

Fuel Compositions in %w of the Initial Heavy Metal load				
	Load		After cooling	
	MOX	UO$_2$	MOX	UO$_2$
U	91.27% (DU)	100% (4.5%$_w$ U$_{235}$)	88.16%	93.57%
Pu	8.59%	0	6.00%	1.14%
MA	0.14%	0	0.70%	0.14%
TRU	8.73%	0	6.70%	1.28%
FP	0	0	5.14%	5.15%

---

[2] [de Roo et al., 2009]'s calculations actually assumed 7 years of cooling for the MOX spent fuel but we prefer to use 5 years for comparison purposes, while keeping the same data. [NEA, 2009] assumes only 3 years of cooling for spent MOX fuel burnt at 45 MWd/kgHM while [NEA, 2002] assumes 7 years of cooling (including reprocessing) for spent MOX fuel burnt at 50 MWd/kgHM.

The thermal efficiency of the LWR is still assumed to be 33.71%. Table 2.5 summarizes the mass flows, normalized for a unit plant of 1GWe and scaled up for a capacity factor of 90% (the original values were 1.150 GWe and 84.5% respectively).

Table 2.5 - MOX/UO₂ mass flows for a 30% MOX core (with CF=90%)

Mass balance MTHM / GWe / Calendar year						
	Load			After cooling		
	MOX	UO2	Total	MOX	UO2	Total
HM	5.719	13.667	19.386	5.425	12.964	18.389
U	5.220	13.667	18.887	5.041	12.788	17.829
Pu	0.491	0	0.491	0.343	0.157	0.500
MA	0.008	0	0	0.040	0.020	0.60
TRU	0.499	0	0.491	0.383	0.177	0.560
FP	0	0	0	0.293	0.703	0.996

The spent UO₂ fuel reprocessing and the MOX fuel fabrication are assumed to take 1 year[3] each.

2.3.3 Perceived advantages and drawbacks of the TTC scheme

- Perceived advantages:
  - The mass of the HLW is considerably reduced (factor of 20), as well as its volume (a factor of 2 at most if wastes are placed in glass).
  - The removal of plutonium from spent UOX fuel (which also avoids build-up of [241]Am and ultimately [237]Np if done shortly after discharge from reactors) reduces the long-term radioactivity and heat load.
  - The use of MOX as a substitute for UOX reduces the consumption of natural uranium to a small extent (less than 20%), which may enhance energy security for countries that import it.
  - Recycling of both recovered plutonium and uranium increases the energy

---

[3] [Bunn et al., 2003] also assumes 1 year of reprocessing for the spent UO₂ fuel burnt at 50 MWd/kgHM; [NEA, 2009] assumes only 0.5 year of reprocessing and 0.5 year of MOX fuel fabrication; [NEA, 2002] assumes 2 years of storage of the fresh fuel (including fabrication); [de Roo and Parsons, 2009] assumes 1 year of reprocessing, 0.5 year of fuel fabrication + 0.5 year of shipping and storage.

content that is extracted from a given amount of natural uranium before it is disposed of.

- About 1/3 of the original plutonium is burnt, thus reducing the plutonium inventory in the system, the existence of which potentially poses proliferation issues.
- The utilization of MOX fuel does not require new types of reactors. Hence, the TTC scheme minimizes implementation risks and allows for more flexibility than the FR schemes.
- The technology already benefits from commercial experience, notably in France.
- The "deliberate waiting" strategy (spent fuel is stored in dry casks, or in a retrievable repository, for a few decades) as well as the repository strategy (spent fuel is disposed of) may be rationally (technically and economically) sound, but could imply to the public that there is actually no solution for the management of the nuclear waste. Reprocessing may mitigate public anxiety about nuclear waste even if the MOX spent fuel as well as the fission products must eventually be disposed of.

- Perceived drawbacks:
    - Both reprocessing and MOX fuel fabrication are expensive with respect to the current price of uranium and its manufacturing. As a result, economic studies generally show that the utilization of MOX does not make economic sense (e.g. [Ansolabehere et al., 2003]).
    - The fabrication of MOX fuel pins implies that plutonium may be in a separated form first (e.g. PUREX process) for some time. The mere existence of separated plutonium, even temporarily, poses proliferation issues because of the risk of diversion, especially if the reprocessing plant and the MOX fuel fabrication plant are not co-located. Moreover, the existence of civilian enrichment and reprocessing as a commercial technology may provide a cover for spread of the technological know-how, making undesirable military programs more likely.
    - The TTC scheme does not eliminate the need for a repository, since fission products, minor actinides and spent MOX fuel must be disposed of. Yet, it

requires new fuel cycle services (MOX fuel fabrication, spent UO_2 fuel reprocessing) that currently do not exist on a commercial scale in the U.S.

2.4 Fast Burners Scheme

2.4.1 Description of the fast burner scheme

A "fast reactor" is a nuclear reactor in which born neutrons are not moderated, and most fissions occur due to neutrons with energies above 1 keV. On the one hand, fast neutrons are less likely to induce fissions of transuranic isotopes than slow neutrons (much smaller fission cross sections), which leads to higher enrichment requirements. But on the other hand, the probability of fission relative to sterile neutron capture is much higher for most of the TRU isotopes, in particular the even-mass-number ones, which are virtually not fissionable in a thermal neutron spectrum. Hence the burning of plutonium in fast reactors would eventually generate much fewer higher-mass transuranics isotopes than in thermal reactors. In short, fast reactors can perform a relatively uniform destruction of the TRU isotopes. If they are designed to do so, they are called "burners" and are characterized by their conversion ratios[4] (CR), from CR=0.0 (fertile-free) to CR=1.0 (break-even, or "self-sustaining").

The fast burner strategy is initiated by reprocessing of the spent UO_2 fuel discharged from the LWRs and decayed over a minimum cooling time. Transuranics are separated from the fission products, which are sent to disposal, in thermal reprocessing plants (uranium is also recovered in the process). These transuranics are mixed with depleted uranium to fabricate fast reactor fuel pins. Fuel assemblies are loaded into fast reactors, irradiated, discharged and decayed over a minimum cooling time. They are then reprocessed in turn in fast reprocessing plants in order to be recycled back into the fast burner. The mix of uranium and transuranics is separated from the fission products (sent to disposal) and used to fabricate fresh pins for the fast reactor. Thus, feed materials for fast reactors come from two sources: external supply (TRU separated from spent UO_2

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[4] The conversion ratio is defined as the ratio of the rate of production of fissile materials to the rate of destruction of the existing fissile materials, approximated by "the ratio of the macroscopic cross section of U-238 capture to that of TRU fission" [Hoffman et al., 2008]. This last definition neglects secondary sources of fissile materials (e.g. the decay of Pu-240 into Pu-241).

fuel) and self-recycling (U-TRU mix separated from spent FR fuel). As the scheme allows for multi-recycling, it is also known as the closed-fuel cycle: only unusable fission products are eventually sent to disposal.

Figure 2.3 shows a representation of the fast reactor fuel cycle, which also applies to the fast breeder scheme (see Section 2.5).
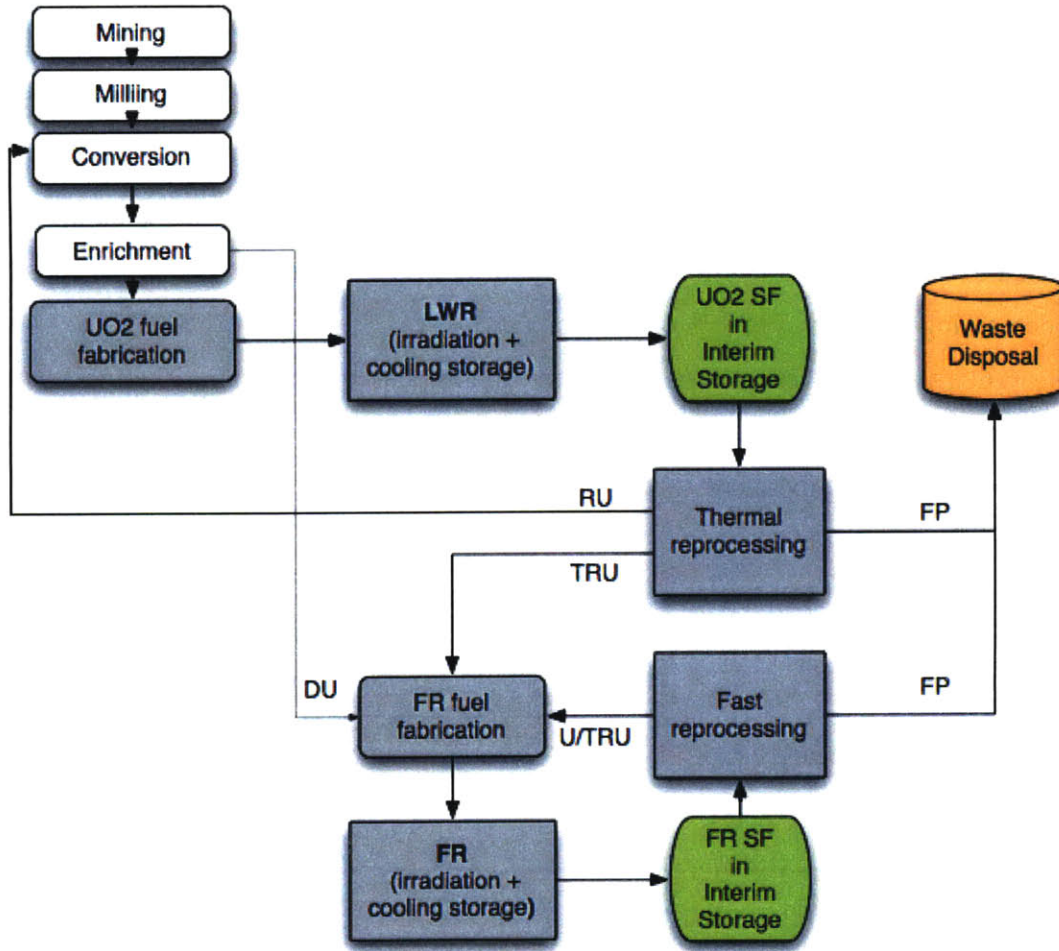
Figure 2.3 - Fast reactor scheme diagram

2.4.2 Data for the fast burner scheme

We use for this study the Advanced Burner Reactor (ABR) designs developed by [Hoffman et al., 2006]. These sodium-cooled reactors designs are still theoretical as of today. In particular, the use of the highly enriched fuels required for the low conversion

ratio burners has never been experienced. Moreover, safety analysis for the burner reactor has not been fully performed to date. Nevertheless, these core models were developed to generate representative fuel cycle mass flows for systems studies of fast burners as a part of the GNEP, which is also our purpose here.

Designs considered in this study are for metal cores with conversion ratios from breakeven (CR=1.0) to fertile-free (CR=0.0), including the intermediates CR=0.75 and CR=0.5. The data used are those for the equilibrium system in which the U-TRU mix extracted from the spent FR fuel produced over a previous cycle was recycled back into the reactor, together with the makeup TRU that was recovered from the spent UO_2 fuel from LWRs, itself irradiated to 50 MWd/kgHM and stored for five years prior to reprocessing (the very design that we are using for our LWRs). The makeup uranium was assumed to be depleted uranium. The amount of materials coming from external sources is a decreasing function of the FR conversion ratio.

Table 2.6 gives some characteristics of the plants and the cycles. All four designs considered were 969 MWt plants. Assuming a thermal efficiency of 0.38, Data were scaled for 1000 MWe unit plant.

Table 2.6 - FR: plant and cycle descriptions

Thermal Power	2,632 MWt			
Thermal efficiency	38%			
Electrical Output	1,000 MWe			
Conversion Ratio	0.0	0.5	0.75	1.0
Cycle Length	132 EFPD	221 EFPD	232 EFPD	370 EFPD
Average number of batches	8.33	5.82	5.95	3.41
Average irradiation time	1,099 EFPD	1,286 EFPD	1,380 EFPD	1,262 EFPD
Discharge Burn up (MWd/kgHM)	293.9	131.9	99.6	73.0

Table 2.7 summarizes the compositions of the FR fuel assemblies for various conversion ratios. One can see that the required enrichment increases as the conversion ratio decreases. After discharge, the spent FR fuel is cooled over 297 days. For comparison purposes, the minimum cooling time is assumed to be 5 years instead[5], while keeping the same fuel composition data.

Table 2.7 - FR metal fuel compositions for various conversion ratios (equilibrium cycle)

Compositions in %w of the Initial Heavy Metal loading								
Conversion Ratio	0.0		0.5		0.75		1.0	
	load	After cooling	load	After cooling	load	After cooling	Load	After cooling
TRU	98.59%	67.13%	33.32%	27.07%	21.21%	19.20%	13.86%	14.04%
U	1.41%	1.44%	66.68%	58.88%	78.79%	70.12%	86.14%	78.30%
FP	0	31.43%	0	14.05%	0	10.68%	0	7.66%

A capacity factor of 0.85 was assumed, as this advanced technology may encounter a period of operation troubles before matching the LWR record of 90% capacity factor. Table 2.8 summarizes the resulting final mass flows (linearly scaled up from [Hoffman et al., 2006] to obtain a 1000 MWe reactor).

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[5] We deem 5 years to be more realistic. [Bunn et al., 2003] implicitly assumes only 1 year of cooling before reprocessing; [NEA, 2009] assumes 4 years of cooling before reprocessing; [NEA, 2002] assumes 2 years of cooling, including reprocessing; [de Roo and Parsons, 2009] assumes 5 years of cooling before reprocessing.

Table 2.8 - FR metal fuel mass flows for various conversion ratios (CF=0.85, equilibrium cycle)

Conversion ratio	0.0		0.5		0.75		1.0	
Core Mass at BOC (MTHM)	9.84		25.66		36.47		45.50	
TRU content in core at BOC (MTHM)	9.70		8.55		7.74		6.31	
Mass balance in MTHM/GWe/calendar year								
	Load	After cooling	Load	After cooling	Load	After cooling	Load	After cooling
HM	2.780	1.906	6.194	5.324	8.203	7.327	11.192	10.335
TRU	2.741	1.866	2.064	1.677	1.740	1.575	1.552	1.571
TRU net destruction rate	875 kg/GWe/year		387 kg/GWe/year		165 kg/GWe/year		-19 kg/GWe/year	
U	0.039	0.040	4.130	3.647	6.463	5.752	9.640	8.763
FP	0	0.874	0	0.870	0	0.876	0	0.857

The spent FR fuel reprocessing and the FR fuel fabrication (including shipping and storage at reactor site) are assumed to take 1 year[6] each.

2.4.3 Perceived advantages and drawbacks of the fast burner scheme

- Perceived advantages:

    - The FR scheme is an actinide multi-recycling scheme, which means that no actinides are supposed to be left in the final wastes, except the TRU separation and fabrication losses (about 0.1%). The long-term heat load as well as the radio-toxicity of the HLW is therefore dramatically reduced.

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[6] [Bunn et al., 2003] also assumes 1 year of reprocessing and 0.5 year of fuel fabrication + 0.5 year of storage of the fresh fuel; [NEA, 2009] assumes only 0.5 year of reprocessing and 0.5 year of fuel fabrication; [NEA, 2002] assumes 2 years of storage of the fresh fuel (including fabrication); [de Roo and Parsons, 2009] assumes 1 year of reprocessing, 0.5 year of fuel fabrication + 0.5 year of shipping and storage.

- The use of FRs as a substitute for LWRs reduces the consumption of natural uranium, which may enhance energy security for countries that import uranium and slow down the depletion of natural uranium reserves. The reduction of the number of LWRs avoids additional production of TRU.
- Theoretically, the energy content of the natural uranium is to be more fully used (assuming an infinite time horizon).
- No fissile materials are to be left in the final wastes, which solves long-term proliferation issues.
- As in the TTC scheme, spent fuel recycling may mitigate public anxiety about nuclear waste, even if the fission products must eventually be disposed of. However, commercial FRs would be available for a few decades from now on.
- Unlike the TTC/MOX scheme, the FR scheme keeps minor actinides together with plutonium, which considerably reduces the risk of its diversion for military programs.

- Perceived drawbacks:
  - Fast reactors, FR fuel fabrication and metal fuel reprocessing are all technologies that are not mature yet. Requires R&D expenses and uncertainties could hinder the transition to a fast reactor regime.
  - As of today, the FR capital costs are deemed to be greater than those of the LWRs (about +20%).
  - As in the TTC scheme, the existence of reprocessing as a commercial technology may provide a cover for undesirable military programs, as civilian enrichment programs do, and make the spread of the technological know-how more likely.
  - Fast Burners with conversion ratio lower than 1 are not self-sufficient. As they have a TRU net destruction rate, LWRs are still necessary to provide TRU feeds.

2.5 Fast Breeders Scheme

2.5.1 Description of the fast breeder scheme

When introducing the fast burners, we did not mention the fact that the number of neutrons emitted per neutron absorbed in the fuel is largest for fast neutrons. This excess of neutrons can be used to generate new fissile materials through the use of fertile blankets (unlike the burner design in which limited fertile materials are included and no blankets are used). This method results in breeding ratios higher than one (compared to about 0.6 for a typical LWR), meaning that the irradiation of the core eventually generates more fissile materials than it consumes.

Having made this distinction, the fast breeder scheme is otherwise exactly the same as the fast burner scheme. Differences are quantitative: while a burner (especially of low conversion ratio) will continuously need an external source of TRU (e.g. separated from spent UO_2 fuel) to complete the supply from self-recycling, a fast breeder actually becomes a net source of fissionable materials, as TRU production is in excess of its own needs.

A representation of the fast reactor fuel cycle was shown on Figure 2.3 (see Section 2.4.1).

2.5.2 Data for the fast breeder scheme

We use for this study the advanced liquid metal-cooled nuclear reactor (ALMR), which has a breeding ratio of 1.23, as a generic breeder. Table 2.9 summarizes the main characteristics of the reactor (scaled up to 1000 MWe from a 319 MWe unit).

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Table 2.9 - Plant and cycle descriptions (with CF=85%)

Thermal power	2,632 MWt
Thermal efficiency	38%
Electrical output	1000 MWe
Cycle Length	700 calendar days
Number of batches	3 (+ blankets)
Irradiation time	1,785 EFPD (2380 EFPD for the blankets)
Discharge burn-up	103.23 MWd/kgHM

Table 2.10 shows the average fuel composition, at loading and after discharge.

Table 2.10 - FR fuel (including blankets) compositions for breeder

Compositions in %$_w$ of the Initial Heavy Metal load		
	Load	After discharge
U	91.10%	84.03%
Pu	8.67%	10.15%
MA	0.23%	0.23%
TRU	8.90%	10.38%
FP	0	5.60%

A capacity factor of 0.85% is assumed. Table 2.9 summarizes the resulting mass flows. It is noticeable that the TRU loading in the fast breeder core is between those of the FR CR=0.5 and the FR CR=0.0.

Table 2.11 - Mass flows for a breeder, including blankets (with CF=0.85%)

Core mass at BOC (including blankets, in MTHM)	97.13	
TRU content in core at BOC (MTHM)	8.64	
Mass balance MTHM / GWe / Calendar year		
	Load	After discharge
HM	14.843	14.014
U	13.521	12.473
Pu	1.287	1.507
MA	0.034	0.034
TRU	1.321	1.541
TRU net destruction rate	-220 kg/GWe/year	
FP	0	0.831

2.5.3 Perceived advantages and drawbacks of the fast breeder scheme

- Perceived advantages:

  - Breeders are by definition a net source of fissile materials, which leads to even more natural uranium consumption reduction compared to the fast burner schemes.

  - Unlike the net burners, fast breeders are more than self-sufficient and may therefore eventually eliminate the need for LWRs and enriched uranium.

  - No fissile materials are to be left in final wastes, which solves the long-term proliferation issues.

  - As in the TTC scheme, spent fuel recycling may mitigate public anxiety about nuclear waste even if the fission products must eventually be disposed of. However, commercial FRs would be available from now on.

  - Unlike the TTC/MOX scheme, the FR scheme keeps minor actinides together with plutonium, which considerably reduces the risk of its diversion for military programs.

- Perceived drawbacks:

- Fast reactors, FR fuel fabrication and metal fuel reprocessing are all technologies that are not mature yet. Commercial fast reactors should be available in the 2030s at the earliest. Required R&D expenses and uncertainties hinder the transition to a fast reactor scheme.

- As of today, the FR capital costs are deemed to be greater than those of the LWRs (about +20%).

- As in the TTC scheme, the existence of reprocessing as a commercial technology may provide a cover for undesirable military programs, and make the spread of the technological know-how more likely.

- The plutonium bred in the blankets is generally of high quality and could be diverted for military purposes.[7]

2.6 Two-tier scheme: MOX / Fast Reactors

2.6.1 Description of the two-tier scheme

This strategy is a combination of the MOX scheme (see Section 2.3) and the fast reactor scheme (see Sections 2.4 and 2.5), the latter succeeding the former in the scenario. Unlike the MOX scheme described in Section 2.3, the scenario assumes that, in addition to plutonium, minor actinides are also separated from the fission products during the reprocessing of the spent UO_2 fuel. These minor actinides are stored while the plutonium is used to fabricate MOX pins, loaded into thermal reactors, irradiated, discharged, cooled over a minimum cooling time, and finally processed. The transuranics extracted from the spent MOX fuel are finally blended with the minor actinides to provide makeup feed for the fast reactors. As in the fast reactors scheme, the spent FR fuel is recycled in fast reprocessing plants after a few years of decay, providing U-TRU for new FR fuel fabrication.

Figure 2.4 shows a representation of the two-tier scheme.

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[7] However, [Stauff et al., 2009] developed a scheme to prevent blankets from breeding weapon-grade plutonium.

Figure 2.4 - Two-tier scheme diagram

2.6.2 Data for the two-tier scheme

It is assumed that the TRU separated from spent UO_2 fuel and that separated from spent MOX fuel are about of the same quality (for utilization in FR), therefore data given in Section 2.3, 2.4 and 2.5 are also used for this scheme.

2.6.3 Advantage and drawbacks of the two-tier scheme

- Perceived advantages, compared to the TTC scheme:
    - As the spent MOX fuel is recycled, no fissile materials would theoretically be left in the final wastes, which solves the long-term proliferation issues.

- Unlike the TTC scheme that leaves both the minor actinides and the spent MOX fuel in the final wastes, all the TRU are recycled in the 2-tier scheme (except the 0.1% TRU losses). The long-term heat load as well as the radio-toxicity of the HLW is therefore dramatically reduced.

- Perceived advantages, compared to the FR scheme:
    - As commercial FRs would be available only from now on, a MOX transition allows starting reprocessing much earlier, thus limiting the spent UO_2 fuel inventory and possibly mitigating public anxiety about nuclear wastes.

    - TRU are about 5.2 times more concentrated in the spent MOX fuel than in the spent UO_2 fuel. As a result, a reprocessing plant will produce 5.2 times more TRU if spent MOX fuel is reprocessed rather than spent UO2 fuel. Finally, going through a MOX phase prior to the introduction of the FRs should accelerate the deployment of the latter, as more TRU should be available for the same thermal reprocessing capacity, with respect to the FR case.

    - A MOX phase allows starting reprocessing activities decades prior to the introduction of FRs. As a result, there would be already an important reprocessing capacity available as the FRs are introduced, which allows a faster deployment.

- Perceived drawbacks
    - Those of the TTC and the FR schemes (Section 2.3.3, 2.4.3 and 2.5.3)

2.7 Conclusion

This chapter presented, in addition to the Once-Through Cycle (OTC), the advanced fuel cycle schemes explored in this study: the twice-through cycle (MOX single pass in thermal reactors, the fast burner scheme, the fast breeder scheme and finally the two-tier scheme that combines the recycling in thermal reactors (MOX) and recycling in fast reactors.

For each of these fuel cycle schemes, the equilibrium properties of both fresh and spent fuels were given. Many other fuel cycle schemes exist (e.g. Pu or TRU multi-recycling in thermal reactors, high burnup fuels in thermal reactors) but they are not considered in this study.

Chapter 3

3. A Simple Model for Nuclear Energy Systems at Equilibrium

3.1 Introduction

One of the most important outputs of any nuclear fuel cycle simulation is the composition of the nuclear energy portfolio. Indeed, the share of fast reactors or LWRs-MOX eventually determines how much TRU is burnt or bred, and how much uranium is saved compared to the once-through cycle scenario[8]. This chapter presents a very simple model to obtain preliminary results for the system at equilibrium and eventually emphasizes the need for a more complex code.

With the exception of the once-through strategy, all other scenarios entail the coexistence of the combination LWR/UO$_2$ with an advanced technology that requires a continuous feeding of Pu (MOX fuel) or TRU (fast reactors). Yet the only source of Pu for fabricating MOX is the reprocessing of spent UO$_2$ fuel (in the twice-through scheme). The TRU needed for FR fuel fabrication must be extracted either from spent UO$_2$ fuel or spent FR fuel. However, some reactors are pure producers of Pu/TRU (LWR/UO$_2$), some are pure consumers (LWR/MOX), and some are both (FR). Therefore, there may be a nuclear energy portfolio such that the total net production (or consumption) of Pu/TRU is zero. If such a portfolio exists, we call it the "equilibrium portfolio".

3.2 Steady-state equilibrium

The steady-state equilibrium is generally a situation assumed in economics studies, when the total electricity output is assumed to be steady. The only assumption made in this Chapter is that the total output is constant over time and that neither the reprocessing capacity nor the fuel fabrication capacity are constraining.

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[8] The utilization of the recovered uranium allows even further reduction in the natural uranium consumption.

Let TRU_i^I be the annual fissile materials (Pu or TRU) mass inflow in the reactor/fuel of type I, TRU_i^O be the reactor's annual fissile materials mass outflow, F_i the installed effective[9] capacity of reactor/fuel i, and λ_i the share (in terms of electricity generation capacity) of the reactor/fuel i in the total nuclear energy portfolio:

$$\lambda_i = \frac{F_i}{\sum_i F_i} \tag{3.1}$$

The coexistence of only 2 technologies is considered (1=LWR/UO2, 2=FR or MOX). At equilibrium, the total net production of TRU is to be zero i.e.

$$\begin{cases} \lambda_1 (TRU_1^O - TRU_1^I) + \lambda_2 (TRU_2^O - TRU_2^I) = 0 \\ \lambda_1 + \lambda_2 = 1 \end{cases} \tag{3.2}$$

An obvious result is that the system has a positive solution (λ_1, λ_2) if and only if $(TRU_1^O - TRU_1^I)$ and $(TRU_2^O - TRU_2^I)$ are of opposite signs, i.e if one of the reactors is a net producer of TRU $(TRU_1^O > TRU_1^I)$ and the other reactor is a net consumer of TRU $(TRU_2^O < TRU_2^I)$. As in all our scenarios 1 = LWR/UO2, which is a pure producer of TRU, and there cannot be equilibrium if 2 is also a producer of TRU, namely a breeder. On the other hand, we know that there is an equilibrium portfolio if 2 is a net burner of TRU (MOX or any fast reactor with a conversion ratio <1). This equilibrium state is given by the equation:

$$\lambda_2 = \frac{1}{1 - \dfrac{TRU_2^O - TRU_2^I}{TRU_1^O - TRU_1^I}} \tag{3.3}$$

Actually, fabrication and reprocessing losses exist, displacing the threshold at which the equilibrium no longer exists (allowing net producers of TRU to be at equilibrium with the LWR/UO2). Let l_f and l_r be the losses for the fuel fabrication and reprocessing, respectively. In this case, the TRU actually converted into fuel ("TRU apparent", denoted by $TRU_i^{O, app}$) is given by:

$$TRU_i^{O, app} = TRU_i^O \cdot (1 - l_f) \cdot (1 - l_r) \tag{3.4}$$

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[9] The "effective capacity" takes into account the capacity factor. A 1GWe-reactor with a capacity factor of 90% has an effective capacity of 0.9 GWe.

Equilibrium exists if and only if $TRU_2^O(1-l_f)(1-l_r) < TRU_2^I$ (assuming that we still have

$TRU_1^{O,app} > TRU_1^I$, which is always the case if 1=LWR/UO2), and:

$$\lambda_2 = \cfrac{1}{1 - \cfrac{TRU_2^O(1-l_f)(1-l_r) - TRU_2^I}{TRU_1^O(1-l_f)(1-l_r) - TRU_1^I}} \qquad (3.5)$$

Assuming that 1=LWR/UO2 and using $l_f = l_r = 0.1\%$, we obtain the equilibrium ratios shown in Table 3.1. Note that for the MOX case, only the plutonium (including non-fissile Pu) is considered and, as the spent MOX fuel is not recycled, its Pu content is assumed to be zero for our model.

Table 3.1 - Equilibrium portfolio at steady-state for various system

i	1	2					
	LWR/ UO2	LWR/ MOX	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
TRU_i^I (in kgHM/ GWe eff/yr)	0	1,848	3,225	2,428	2,047	1,826	1,554
TRU_i^O (in kgHM/ GWe eff/yr)	278 (250 if 2=MOX)	0	2,195	1,973	1,853	1,848	1,813
λ_2	X	11.89%	21.15%	37.7%	58.39%	Does not exist	Does not exist

As expected, the higher the conversion ratio, the higher the share of fast reactors in the nuclear energy portfolio. The share of MOX is even lower than that of the pure burner (CR=0.0) because the spent MOX fuel is not recycled (twice-through cycle).

3.3 Steady growth model

Starting from the previous model, we now assume that the system is not in a steady state, but instead undergoes a constant growth g (in terms of total effective capacity), i.e.

$$\sum_i F_i(t+1) = (1+g)\sum_i F_i(t) \tag{3.6}$$

In this case, it is important to know the time between the discharge of the TRU and its reloading in reactors. This time T_s is the sum of the cooling time T_c, the reprocessing time T_r and the fuel fabrication time T_f.

$$T_s = T_c + T_r + T_f \tag{3.7}$$

Hence the TRU ready to be reloaded in reactors is the TRU that was discharged from reactors T_s years before. Hence the following system:

$$\begin{cases} F_1(t-T_s)\cdot TRU_1^O \cdot (1-l_r)\cdot(1-l_f) - F_1(t)\cdot TRU_1^I + \mathrm{K} \\ \quad \mathrm{K}\, F_2(t-T_s)\cdot TRU_2^O \cdot (1-l_r)\cdot(1-l_f) - F_2(t)\cdot TRU_2^I = 0 \quad \textit{for any } t, T_s \\ \\ F_1(t+T) + F_2(t+T) = (1+g)^T \cdot (F_1(t) + F_2(t)) \quad \textit{for any } t, T \end{cases} \tag{3.8}$$

Plugging the second equation into the first (with T= -T$_s$) and dividing by $(F_1(t) + F_2(t))$, we obtain

$$\begin{cases} \lambda_1 \cdot (1+g)^{-T_s} \cdot TRU_1^O \cdot (1-l_r)\cdot(1-l_f) - \lambda_1 \cdot TRU_1^I + \mathrm{K} \\ \quad \mathrm{K}\, \lambda_2 \cdot (1+g)^{-T_s} \cdot TRU_2^O \cdot (1-l_r)\cdot(1-l_f) - \lambda_2 \cdot TRU_2^I = 0 \quad \textit{for any } t \\ \lambda_1 + \lambda_2 = 1 \end{cases} \tag{3.9}$$

Hence the result, constant over time:

$$\lambda_2 = \cfrac{1}{1 - \cfrac{TRU_2^O(1-l_f)(1-l_r)(1+g)^{-T_s} - TRU_2^I}{TRU_1^O(1-l_f)(1-l_r)(1+g)^{-T_s} - TRU_1^I}} \tag{3.10}$$

We assume for all cases $T_f = T_r = 1$ year but test two values for the cooling time, $T_c = 5$ years or $T_c = 10$ years (resulting in $T_s = 7$ years or 12 years). Furthermore, we test 3 growth rates: g = 1.0%, 2.5% or 4.0% (the case of g=0 was already done). Table 3.2 shows the resulting equilibrium ratios.

Table 3.2 - Equilibrium portfolios for a steady-growth system (values of λ_2)

Growth rate	Cooling Time (years)	1=LWR/UO2, 2 =					
		LWR/ MOX	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
g=1.0%	5	11.18%	17.96%	30.44%	44.55%	70.98%	Does not exist
	10	10.70%	16.12%	26.57%	37.76%	56.54%	Does not exist
g=2.5%	5	10.20%	14.45%	23.23%	32.21%	45.96%	88.00%
	10	9.12%	11.45%	17.63%	23.49%	31.21%	49.72%
g=4.0%	5	9.31%	11.90%	18.45%	24.73%	33.19%	54.08%
	10	7.78%	8.54%	12.64%	16.27%	20.45%	29.02%

It can be seen that the higher the growth rate/cooling time, the lower is the share of MOX or fast reactors in the nuclear energy portfolio. In particular, a growth rate of 1% allows for equilibrium between LWRs/UO2 and self-sufficient FRs, and a growth rate of 2.5% allows for equilibrium between LWRs/UO2 and breeders (which implies that a 100% breeder park is not possible for a 2.5% growth rate, in the conditions of the model).

It should be noted that the higher the conversion ratio, the larger is the ratio λ_2 sensitivity to the growth rate/cooling time. The ratio of MOX in the system in particular is little affected because the spent MOX fuel is not recycled.

Figure 3.1 shows the values of λ_2 for various growth rates (cooling time of 5 years).
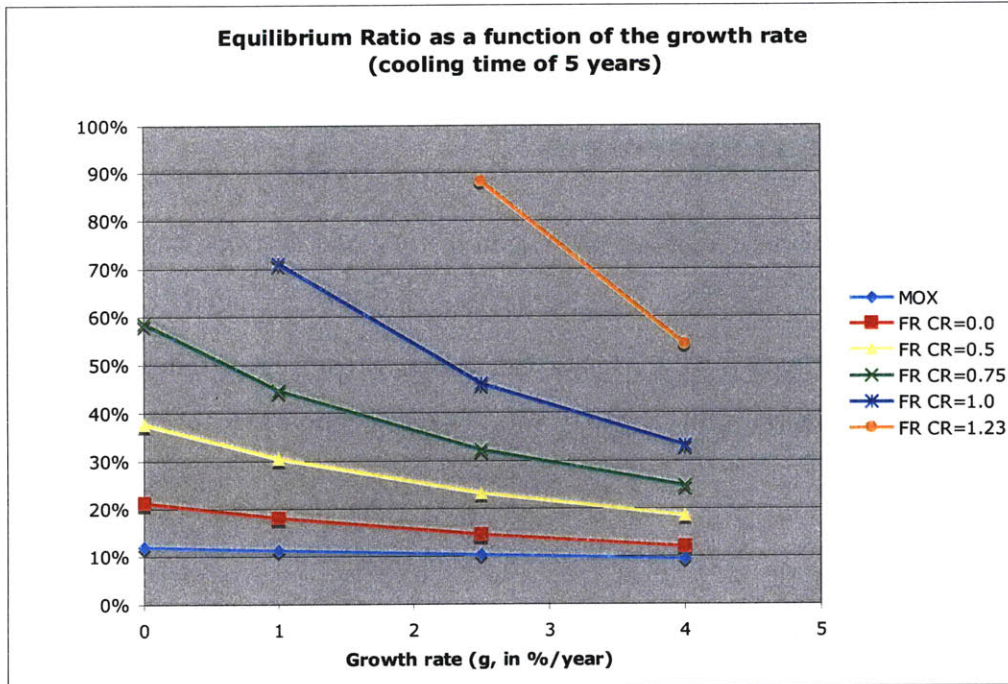
Figure 3.1 - Equilibrium portfolios for various growth rates (cooling time of 5 years)

Figure 3.2 shows the values of λ_2 for various growth rates (cooling time of 10 years).
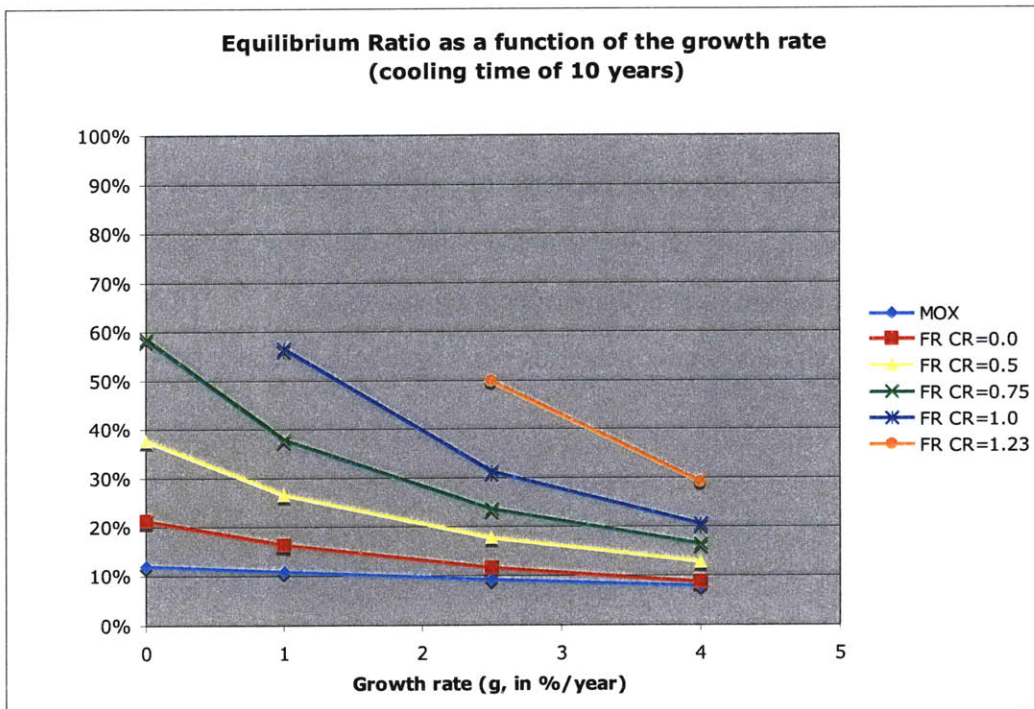
Figure 3.2 - Equilibrium portfolios for various growth rates (cooling time of 10 years)

3.4 Reprocessing requirements

We can also calculate the requirements of both thermal and fast reprocessing capacities for these different scenarios. Let $P^i_{TRU,O}$ be the heavy metal fraction of TRU (or Pu) in the spent fuel of reactor type i. As the spent fuel undergo reprocessing after the minimum cooling storage, the reprocessing capacity $R_i(t)$ required at time t is:

$$R_i(t) = \frac{F_i(t - T_c) \cdot TRU^O_i}{P^{TRU,O}_i} \tag{3.11}$$

For i=1,2 :

$$R_i(t) = \frac{\lambda_i \cdot (1+g)^{t-T_c} \cdot (F_1(0) + F_2(0)) \cdot TRU^O_i}{P^{TRU,O}_i} \tag{3.12}$$

The ratio μ_2 of the fast reprocessing capacity needed over the thermal reprocessing capacity needed is also obtained.

$$\mu_2 = \frac{1}{1 + \dfrac{P^{TRU,O}_2 \cdot \lambda_1 \cdot TRU^O_1}{P^{TRU,O}_1 \cdot \lambda_2 \cdot TRU^O_2}} \tag{3.13}$$

It can be seen that this ratio is constant over time, as λ_i is.

Assuming an initial total effective capacity of $F_1(0) + F_2(0) = 90$ GWe eff, the total thermal reprocessing capacity can be calculated at any time. Table 3.3 shows some of the results for the 2.5% growth rate, 5 years of cooling storage ($P^{TRU,O}_1 = 1.28\%$, 1.15% in the MOX case), at different times in the scenario (years 0, 25, 50, 75 and 100).

Table 3.3 - Reprocessing requirements for the base case scenario

(rep. capacity in tHM/year)		1=LWR/UO2, 2 =					
		LWR/ MOX	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
t (year)	$P_2^{TRU,0}$	0	67.13%	27.07%	19.20%	14.04%	10.38%
0 *total: 90 GWe)*	Effective capacity	9.2 GWe	13 GWe	21 GWe	29 GWe	41 GWe	79 GWe
	UO2 rep. capacity	1,553	1,478	1,326	1,171	934	207
	FR rep. capacity	0	38	135	247	481	1,223
25 *total: 167 GWe*	Effective capacity	17 GWe	24 GWe	39 GWe	54 GWe	77 GWe	147 GWe
	UO2 rep. capacity	2,879	2,740	2,459	2,171	1,731	384
	FR rep. capacity	0	70	250	458	892	2,267
50 *(total: 309 GWe*	Effective capacity	32 GWe	45 GWe	72 GWe	100 GWe	142 GWe	272 GWe
	UO2 rep. capacity	5,337	5,080	4,559	4,026	3,209	713
	FR rep. capacity	0	129	463	850	1,654	4,202
75 *(total: 573 GWe*	Effective capacity	58 GWe	83 GWe	133 GWe	185 GWe	264 GWe	505 GWe
	UO2 rep. capacity	9895	9418	8452	7463	5949	1321
	FR rep. capacity	0	239	858	1576	3,066	7,791
100 *(total: 1,063 GWe*	Effective capacity	108 GWe	154 GWe	247 GWe	342 GWe	489 GWe	936 GWe
	UO2 rep. capacity	18,346	17,461	15,670	13,836	11,030	2,450
	FR rep. capacity	0	444	1,591	2,921	5,685	14,444

3.5 Limits

This simple model presented above does not take into account many constraints and time-related aspects that would apply in reality:

1) A start-up core (with a TRU content greater than the average annual loading) is needed for any reactor at commissioning. At the other end, an entire core is unloaded from a reactor at the end of its life, releasing extra amounts of TRU.

Consequently, the lifetimes of the reactors play a role in the dynamic of the mass flows. As the lifetime of a reactor (60 years) is long relative to the time horizon of our simulation (100 years), the effect of the start-up core loading would prevail, leading to a decrease in the number of fast reactors compared to that calculated.

2) Introduction date: advanced technologies are not necessarily available from time 0. For example, the first commercial fast reactors may appear in 2040 at the earliest.

3) The increase in reprocessing capacity supply is incremental and not continuous. In order to benefit from economies of scale, a thermal reprocessing plant has a capacity of about 1000 tHM/year.

4) Due to limits on the industrial capacities, the rate of building of the reprocessing plants is capped. Both limits 3) and 4) leads to a reduced TRU supply, which decrease the FR/MOX installed capacity compared to our simple model.

5) The combination of limits 2), 3) and 4) results in accumulations of spent fuel. The reprocessing of this legacy, when possible, leads to an increase in the FR/MOX installed capacity compared to what was calculated in our simple model, until the legacy is depleted.

6) Reprocessing plants are built only provided that their capacity will be fully used (or used at least at a certain minimum level) over their economic lifetimes. This constraint slows down the rate at which the legacy mentioned in limit 5) is depleted.

7) Constraints 3) and 4) also apply to the reactors and to the other fuel cycle services (uranium mining, conversion, enrichment, fuel fabrication). 7) will be ignored in this study.

3.6 Conclusion

This chapter presented a very simple model which takes into account the mass flows related to the various types of reactor/fuels, the fuel fabrication and reprocessing losses, the energy demand growth, the cooling, reprocessing and manufacturing times of the spent/fresh fuels, allowing the calculation of the composition of the nuclear energy reactor portfolio as well as the required reprocessing capacities.

Then, a set of constraints that underlines the limits of this simple model was identified. The existence of these constraints shows the necessity of having a more complex, dynamic model (CAFCA)

However, the results obtained in this chapter will be used as a benchmark and will provide some guidance for the interpretation of the CAFCA results. Any departure from these results will necessarily be caused by the combination of some of the constraints that were identified.

Chapter 4

4. The CAFCA model: overview

4.1 Introduction

CAFCA (Code for Advanced Fuel Cycles Assessment) is a nuclear fuel cycle analysis code that has been developed at MIT over the last 5 years. Three versions of CAFCA (CAFCA I, II and III) have been developed prior to the one currently used (CAFCA-SD), which is coded in system dynamics and was created by [Busquim et al., 2008]. New options have been added and modifications have been brought to CAFCA-SD ever since; they are described in chapters 4, 5 and 6 and Appendix 4.A.1. It is strongly recommended to whoever is interested in the building of the code (especially future CAFCA developers) to read [Busquim et al., 2008] prior to this study.

4.2 CAFCA presentation

4.2.1 CAFCA: brief history
- CAFCA I, II and III

Prior to the current version (CAFCA-SD), three versions were developed in the MATLAB simulation environment. The first version was used to simulate the deployment of two technologies: transuranics (TRU) recycling in combined UO_2 and fertile-free fuel LWR assemblies (CONFU) and TRU recycling in fertile free actinide burning reactors (ABRs). The second version of CAFCA introduced an additional TRU recycling scheme in fast self-sustaining reactors (conversion ratio of 1.0), and the option for a minimum prescribed loading factor for recycling plants. The third version of CAFCA introduced a simplified way of tracking the isotopic composition through the fuel cycle in order to assess the radioisotope decay in the system.

- CAFCA-SD

The current version of CAFCA, designated CAFCA-SD, is coded in System Dynamics, using the software VENSIM as a platform, with potential interactions with MSExcel

spreadsheets and C++ scripts. System Dynamics indeed appeared as a very appropriate approach to model and understand the behavior of a complex system such as the nuclear enterprise, which can be modeled as a set of stocks and flows, non-linearly connected because of the presence of internal feedback loops (e.g. more TRU available $\prod$ more FR $\prod$ less LWR $\prod$ less TRU produced $\prod$ less TRU available, and so on) and time delays (e.g. cooling time).

On the programmer side, the graphical interface of VENSIM makes the understanding and modification of the code easier than a more traditional programming language. This relative ease of building of the code and customizing it is an essential feature as the code is to be shared and open to anyone interested in studying the fuel cycle.

On the user side, an excel spreadsheet interface was developed, which notably allows for cash-flow analyses (see Chapter 6).

The first version of CAFCA coded in System Dynamics ("CAFCA-SD") resulted from the work of [Busquim et al., 2008]. In addition to the once-through cycle scheme, the following fuel cycle options were implemented: TRU multi-recycling in LWRs (CONFU scheme), the self-sustaining Gas-cooled Fast Reactor (GFR) and the Actinide Burner Reactor (ABR). All these options/designs are not considered in this study.

The following modifications (i.e implying changes from the previous version of the code) and additions have been made ever since (they are described in details in Chapter 5 and Chapter 6):

- Main modifications to the existing model:
  - The GFR and ABR models have been replaced by a generic FR model that can be applied to any conversion ratio, including breeding ratios (see Section 5.3).
  - The economic module has been rebuilt from scratch. The use of spreadsheet allows for transparent and easy-to-tune cash-flow analysis. New accounting schemes are available to deal with the reprocessing costs and disposal costs (see Chapter 6).

- The Pu (resp. TRU) extracted from spent LWR fuel in thermal reprocessing plants can be accumulated in storage until the MOX (resp. FR) fuel fabrication starts.

- Main additional modifications
  - LWR of second generation (in practice, the existing reactors, denoted LWR2) and LWR of third generation (denoted LWR3) are distinguished, which allows one to give them different features. Likewise, the LWRs that can be loaded with MOX fuel (noted LWRmf) are separately tracked. All the LWR3s are LWRmf, as well as some of the LWR2 (currently 17% of the U.S. reactors).
  - The LWRmfs can be loaded with MOX fuel assemblies. The LWR2s and LWR3s can be loaded with different cores. In practice, the LWR2s are loaded with 30% MOX cores while LWR3s can be loaded with 100% or below MOX cores (see Section 5.2. This option is not exercised in this study as all the LWRs are loaded with 30% MOX).
  - 2-tier scenarios, in which the spent MOX fuel is recycled in FRs, are possible. In such scenarios, the minor actinides separated from the spent UO_2 fuel are directly burnt in the FRs along with the TRU separated from the spent MOX fuel (see Section 5.2.3 and 5.4).
  - Fast reactors of diverse conversion ratios (including a breeder) are available (see Section 5.4).
  - The uranium recovered from the spent UO_2 fuel reprocessing can be used as a substitute for natural uranium to make UO_2 fresh fuel for LWRs. The use of recovered uranium has an impact on the need for natural uranium and enrichment capacity, and ultimately affects the fuel cycle cost (see Section 5.5.)
  - A U-price/reserve model has been implemented. This model assesses at any time in the simulation the worldwide reserves of natural uranium for a given cost/price (see Section 6.4.3.1)
  - The HLW (depending on the scenario: spent fuel, various losses, FP, MA) are tracked (see Section 5.6).

- High Burn-up fuel can be introduced to the once-through cycle via LWR2 uprates or construction of LWR3s (see [Feng et al., 2008] or Appendix 4.A).

4.2.2 Capabilities

CAFCA automatically deploys reactors and reprocessing facilities over time based on a nuclear energy demand curve specified by the user. Different types of reactors and fuel may be used simultaneously in CAFCA, including once-through LWRs, LWRs that recycle CONFU or MOX and fast reactors with various conversion ratios (0.0, 0.5, 0.75, 1.0 and 1.2) and of different design (2 self-sustaining FRs and 2 pure burners available). CAFCA calculates the market share of each technology involved in the scenario so as to deplete the stockpiles of spent fuel waiting in interim storage, under a set of constraints and characteristics provided by the user. These user inputs include introduction dates for advanced technologies as well as capacities of various facilities, their lifetimes, and maximum building rates of new industrial units. A parameter can be set to deploy used fuel reprocessing facilities in such a way that a minimum capacity factor is maintained over their lifetimes to ensure their efficient and cost-effective operation. The methodology used in CAFCA is described in Section 4.2.3.

The model focuses on tracking the masses of uranium and transuranics (TRU) through the fuel cycle. However, Pu and minor actinides are tracked separately if the MOX option is used in one of the scenarios chosen. At this time, reactor cores are assumed to be at equilibrium and CAFCA does not account for isotopic decay. CAFCA also provides numerous outputs including uranium ore consumption, SWU as well as thermal and fast reprocessing capacity requirements, and repository capacity needed to store waste for a given scenario.

CAFCA eventually generates economic outputs including the annual system-averaged cost of electricity and the levelized cost of electricity (see Chapter 6).

4.2.3 General Methods

System Dynamics allows the modeling of material stocks and flows as well as informational fluxes, and the use of mathematical operators, including integration, derivation and time delays. Some of the variables represented in CAFCA are real outputs, such as reactors and facilities fleets; others are internal variables that set the decision rules applied to the system. Thus CAFCA is segmented into single-input single-output (SISO) subsystems that interact, each of them modeling both a physical structure (e.g. the LWR fleet) and the performance of those structures (e.g. to order building and decommissioning of LWRs).

The model is discrete: rules are first applied to an initial system defined by the user, based on the state of this system. As a result, the state of the system at the next time step of the simulation is changed. Rules are applied to the system based on this new state, and so on. The time step currently used in CAFCA is 0.125 years (1.5 months).

Overall structure ("High-Level Structure Diagram")

Figure 4.1 represents the high-level structure diagram of the code, giving an overall view of the stocks and flows in CAFCA (both materials inventories and fleets of facilities are modeled as stocks). Variables modeling the decision rules are not shown in this figure.

Among all the facilities required over the fuel cycle, only reactors and reprocessing facilities are actually represented in CAFCA (in bold in the diagram). The code implicitly assumes that the other markets (mining, milling, conversion, enrichment, fuel fabrication) are not constraining, offering at any time a supply capacity that exactly matches the demand.
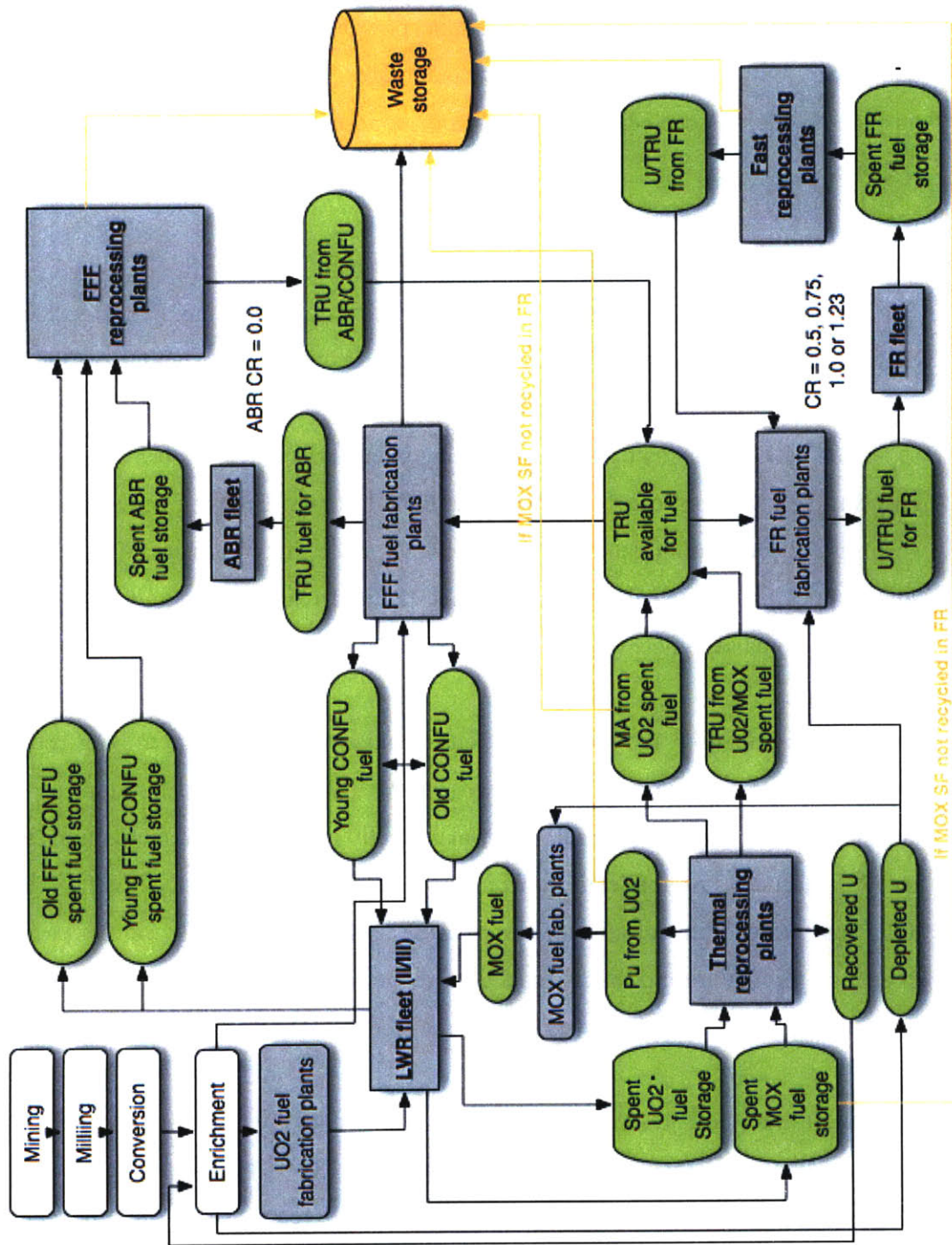
Figure 4.1 - CAFCA high-level structure diagram: Inventories (rounded rectangles), mass flows (arrows) and facilities (grey rectangles), front-end steps (white rectangles). The plants actually modeled in CAFCA are in bold.

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Reactors and fuels are black-boxes for CAFCA: their characteristics - such as core mass, masses loaded and discharged, cycle length, power and capacity factor, composition of both fresh and spent fuel – are inputs externally produced and provided by the user through an Excel spreadsheet. It is therefore up to the user to make sure that those data are consistent.

Structures performance ("Structure-Policy Diagrams")

The first driver of the simulation is the nuclear electricity demand. CAFCA calculates at each time step the "optimal" number of fast reactors (or depending on the scenario chosen, the number of LWRs loaded with MOX etc.) in such a way that the stockpiles of separated TRU produced by the reprocessing plants are reduced as fast as possible, while ensuring that reactors can be fueled over their lifetime. Building orders are calculated accordingly, depending on the existing fleet (under the constraint that the electricity supply does not exceed the demand). Then thermal reactors are built to meet the total electricity demand, if necessary.

UO_2 fuel fabrication and other front-end requirements (mining, milling, conversion, enrichment) are calculated solely based on the LWR fleet consumption, as no other constraint is imposed. In general, materials directly usable for fuel fabrication (Pu, TRU, U-TRU mix) are systematically utilized. As a result, there is no stock of Pu, TRU or U-TRU but instead inventories of ready-to-be-used fuel (U-TRU for FR, MOX). An exception is the period between the introduction of thermal reprocessing and the introduction of MOX or FR, over which time stockpiles of Pu or TRU are accumulated.

The building of thermal reprocessing plants is driven by the inventories and mass inflows of thermal spent fuel, rather than any demand profile (which is precisely calculated so as to avoid any shortage of fuel supply). To start fast reactors, an "optimal" number of reprocessing plants is calculated in such a way that the inventory of LWR spent fuel is depleted as fast as possible while ensuring that the reprocessing plant will operate over its lifetime at a minimum loading factor. Eventually, a cap may be imposed on the building rate, reflecting industrial capacity limits. The same scheme is applied to calculate the number of fast reprocessing plants based on the spent FR fuel inventory.

Practically, there are five main types of structure-policy diagrams in CAFCA, characterized by their input/output and internal methods:

- The *LWR structure-policy diagram* for construction and decommissioning of LWRs is driven by the demand for nuclear energy (system-input), and provides as a system-output the number of LWRs under commercial operation as well as their ages. See Section 2.4.1 in [Busquim et al., 2008] for more details.

- The *FR structure policy diagram* for construction and decommissioning of FRs is driven by the mass of fissile materials from both thermal and fast reprocessing plants, and provide as a system-output the number of FRs under commercial operation as well as their ages. This diagram applies to:
  - the FR CR=0.0 ("pure burner")
  - the FR CR=0.5
  - the FR CR=0.75
  - the FR CR=1.0 ("self-sustaining" or "breakeven")
  - the FR CR=1.2 ("breeder")
  - the GFR
  - the ABR

  See Section 5.3.2 for more details.

- The *RP structure policy diagram* for construction and decommissioning of RP (reprocessing plants) is driven by the mass available for partitioning (spent fuel inventories). The system-output is the number of reprocessing plants under commercial operation. This diagram applies to:
  - the thermal reprocessing plants ("ThRP").
  - the fast reprocessing plants ("FRP")
  - the FFF reprocessing plants ("FFF RP")

  See Sections 2.4.2.1 and 2.4.2.5 in [Busquim et al., 2008] and Section 5.2.3 in this study for more details.

- The *CONFU fuel structure-policy diagram.* The system-input is the TRU available for FFF fabrication (from both thermal and FFF reprocessing plants). The system-output is the number of LWR loaded with CONFU. See Section 2.4.4 in [Busquim et al., 2008] for more details.

- The *MOX fuel structure-policy diagram.* The system-input is the Pu available for MOX fuel fabrication (separated from spent UO_2 fuel in reprocessing plants). The system-output is the number of LWRs loaded with MOX fuel. There are two main differences between this diagram and the CONFU fuel structure-policy diagram. First, unlike CONFU, MOX fuel is burnt only once in LWRs ("one-pass"). Second, there are actually two different types of MOX fuel (characterized by different plutonium content) as some LWR (all the new LWR as well as some of the existing ones) can be loaded with 100% MOX cores, while the others only accept 30% MOX cores. See Section 5.2.2 fore more details.

These five main structure-policy diagrams are completed with various secondary modules:

- The *Front-end module*: the main input is the number of reactors of each type and the number of reactors starting commercial operation). For the LWRs, outputs include the natural uranium, SWU, conversion and UO_2 fuel fabrication requirements. In scenarios in which recovered uranium is used as a substitute for natural uranium, the amount of recovered uranium available is also an input. For the LWRs loaded with MOX fuel and FRs, the output is the consumption of depleted uranium. See Section 2.4.3 in [Busquim et al., 2008] and Section 5.4 in this study for more details.

- The *Waste management module:* inputs include the number of reactors under commercial operation (and the number of reactors being decommissioned) and the throughputs of the fuel cycle facilities (enrichment plants, fuel fabrication plants, reprocessing plants). The output is an itemized inventory of the high-level wastes (fission products, minor actinides, spent fuel of various types and TRU content).

A rough estimation of the repository requirement is also provided (see Section 5.5).

- The *TRU tracking module* tracks at any time the total amount of TRU in the system as well as its locations (see Section 5.6)

- the *Economics module* takes as inputs the number of reactors of various types as well as their ages, and the different fuel utilization rates. Outputs include the cost of electricity and the levelized cost of electricity (see Chapter 6).

4.2.4 Notable approximations and assumptions

We recall in this section the main assumptions made in CAFCA:

- CAFCA is a "continuous-flow" code in the sense that it does not track actual fuel batches but assumes continuous mass inflows. For any combination reactor/fuel, the annual mass flow M (yearly averaged throughput for the reactor at equilibrium) is used as an input. For a given reactor, a fictive fuel batch is consumed at each time step of the simulation (1.5 months). The mass m_b of this fictive batch is the product of the annual mass flow and the time step (in years). An exception is the first year of the lifetime of the reactor over which the annual mass flow is replaced by the total mass of the core. Hence the loading of the first core is spread over one entire year. In the case of the LWR (UO_2 fuel assemblies, cycle length of 1.5 years, 3 batches), this method leads to an extra use of fuel of 0.6% over the entire lifetime. Both real and CAFCA mass loading for the LWR are shown in Figure 4.2.
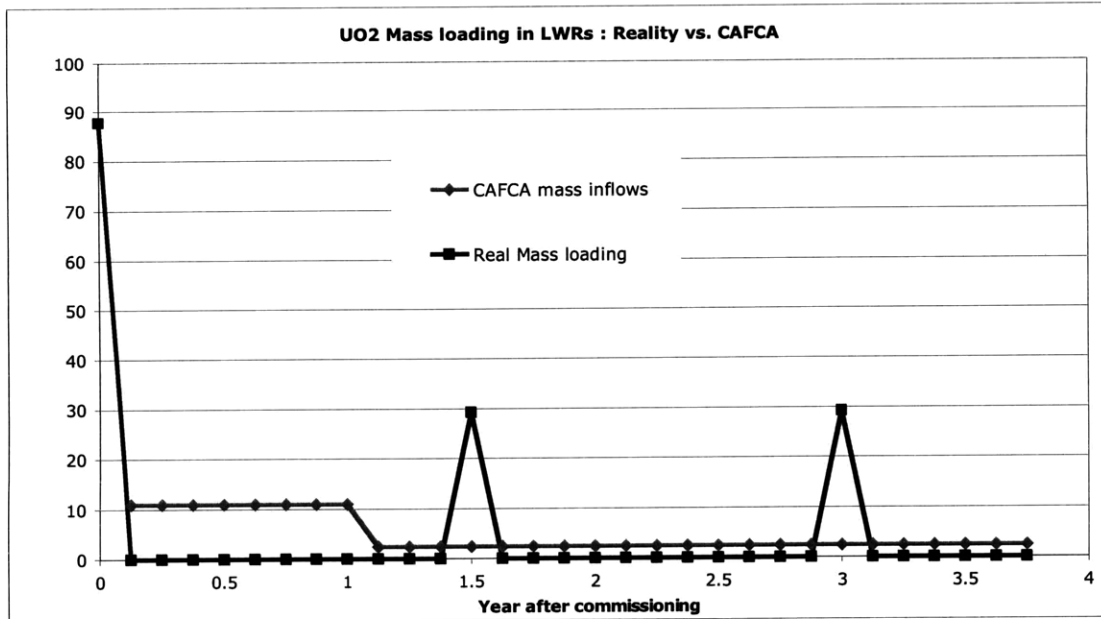
Figure 4.2 - UO₂ Mass loading in LWRs: CAFCA vs. reality

- For every type of reactor/fuel, transient regimes are ignored and equilibrium fresh fuel only is used. In particular, the data used for the fast reactors are those of an equilibrium cycle in which TRU feeds come from both spent FR fuel reprocessing and spent UO_2 fuel reprocessing (the ratio being the "equilibrium ratio" for a steady-state system, see Chapter 3).

- The spent UO_2 fuel discharged after fewer than 3 cycles (which is the case for two of the three batches that are loaded in the first core, when the reactor is commissioned, as well as two of three batches that are loaded in the last core, when the reactor is decommissioned) is assumed to have the same composition as the spent UO_2 fuel discharged after 3 cycles. [Aquien et al., 2006] estimates the relative error to be less than 5% over the 60-year lifetime of the reactor. The same assumption is made for all the fuels.

- Radioactive decay occurring in the spent fuel is ignored. As a result, both the weight percentage of TRU in the spent fuel and its quality are assumed to be constant over time. As long as the TRU are considered as a lump, these two

assumptions are reasonable because of the relatively long half-life of most of the major transuranic elements (^{237}Np237, ^{238}Pu, ^{239}Pu, ^{240}Pu240, ^{242}Pu) compared to the scale of the simulation (100 years). An exception is the ^{241}Pu, which decays into ^{241}Am with a half-life of only 14.4 years. But, since ^{241}Am is relatively stable (half-life of 432 years) and is also fissionable in fast spectra (which defines the quality of the TRU vectors), the ^{241}Am gain compensates for the ^{241}Pu loss and the two assumptions mentioned above still hold.

As a consequence, the spent fuel is treated as a homogeneous lump and its aging is not tracked. When a batch of spent fuel is reprocessed, the code does not "know" how old this batch is.

However, a source of concern is the scenarios involving the use of MOX fuel, which entails that Pu is separated from the minor actinides and thereby TRU are no longer treated as a lump. In this case, the older the spent fuel, the lower the fissile Pu content because of the decay of ^{241}Pu into ^{241}Am. Hence the assumption of ignoring the decay of the spent fuel needs to be revisited for this case, which is done in Section 5.2.1).

- For the sake of simplicity, the existing nuclear reactor park is represented by a fleet of 1 GWe-reactors of a unique design, generically designated as "LWR". The characteristics of the initial fleet, including a list of the LWR2s licensed for MOX utilization (LWR2), are provided in Appendix A.4.2.

- Likewise, the current UO$_2$ spent fuel legacy is assumed to have the composition of a 5-year cooled UO$_2$ spent fuel discharged at 50 MWd/kgHM. In reality, the average age of the spent fuel in the U.S. is about 15 years, and its average burn up 36.4 MWd/kgHM[10].

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[10] The most recent data on the U.S spent fuel legacy are provided on the website of the Energy Information Adminstration [EIA, 2004]. As of 2002, the total spent fuel legacy amounted to 47,023.4 tIHM, with an average discharge burn-up of 33.6 MWd/kgHM. Assuming that from 2003 to 2008, each of the 100 1GWe-LWRs generated 19.50 tIHM of spent fuel burnt at 50 MWd/kgHM (no reactor has been decommissioned over this period), we end up with a total spent fuel legacy of 56,800 tIHM of spent UO$_2$ fuel, with an average discharge burn up of 36.4 MWd/kgHM, and an average age of 15 years.

- We assume in CAFCA that the spent fuel discharged from reactors before 2002 had been irradiated to the same burnup as practiced today, namely 50 MWd/kgHM. In reality, the burnup was generally lower as reactors had been experiencing delays (reflected by mediocre capacity factors), resulting in an average burnup of 33.6 MWd/kgHM as of 2002. As the composition of the spent fuel is directly affected by the burnup, our approximation of the burn-up of the spent fuel may lead to an overestimation of the fissile material legacy. The fissile Pu content (^{239}Pu + ^{241}Pu, in kg per kgIHM) of the spent UO_2 fuel is 0.672% for a discharge burnup of 33 MWd/kgHM and 0.790% for a discharge burn-up of 53 MWd/kgHM [Bunn et al., 2003]. Assuming that the fissile content in the spent fuel is a linear function of the burn-up, we thus overestimate the fissile Pu content of the legacy (as of 2002) by about 15-20%. Since this legacy amounts to 47,023.4 tHM, this overestimation (assuming a conservative 20%) is equivalent to 85 tHM of plutonium, which is the Pu content of the spent UO_2 fuel released by the generation of about 360 GWe-year (assuming a burnup of 50 MWd/kgHM). This amount of plutonium could be used to make 970 tHM of MOX fuel, which could generate 51 GWe-year.

4.3 Summary

This chapter presented the fuel cycle simulation code developed at MIT, CAFCA (Code for Advanced Fuel Cycle Assessment), including its history, its capabilities as well as the methods used in its implementation. The underlying assumptions and approximations are also recalled; it is recommended that they be considered in the interpretation of the results.

Chapter 5

5. Modeling Strategies - Update

5.1 Introduction

This chapter describes in detail the modifications made in CAFCA mentioned in 4.2.1 and is to be read by whoever is interested in the implementation of the code (e.g. future developers of CAFCA). However, it may be necessary to read Sections 5.2.1, 5.4 and 5.5 for a full understanding of the results presented in Chapters 6 and 7.

5.2 Thermal recycling: MOX assemblies in LWRs

5.2.1 Assumptions

- It is assumed that current licenses allow for a maximum of 30% of MOX assemblies in the core (licensed Generation II LWRs are denoted LWR2mf). Furthermore, it is assumed that all the reactors that will be built in the future (so-called Generation III reactors, denoted LWR3) are licensed for any rate of MOX assemblies. CAFCA allows the implementation of one core configuration for the LWR2 (e.g. 30% MOX fuel), and one core configuration for the LWR3 (e.g. 100% MOX fuel). As the average composition of the fresh MOX fuel depends on the core configuration, two types of MOX fuel must be distinguished: the MOX fuel for the LWR2 (denoted MOX2) and the MOX fuel for the LWR3 (denoted MOX3).

- Isotope decay is not taken into account in the current model and plutonium is considered as a whole, without any distinction among its isotopes. This is consistent with the CAFCA code, which is intended for scoping studies of large systems and treats plutonium and minor actinide streams as lumped, without tracking individual isotopes and their decay. As mentioned in Section 4.2.4, this approximation is reasonable as long as only TRU are tracked. If plutonium is

separated as in the MOX case, a concern is the decay of fissile ^{241}Pu into ^{241}Am. The impact of our approximation can be assessed. The main fissile isotopes of the plutonium present in the spent UO$_2$ fuel are ^{239}Pu and ^{241}Pu. Figure 5.1 shows the ^{241}Pu and Fissile Pu (^{241}Pu+^{239}Pu) contents in spent UO$_2$ fuel over time. Their values at 5 years of cooling, used in CAFCA for any type of fuel, are also shown.
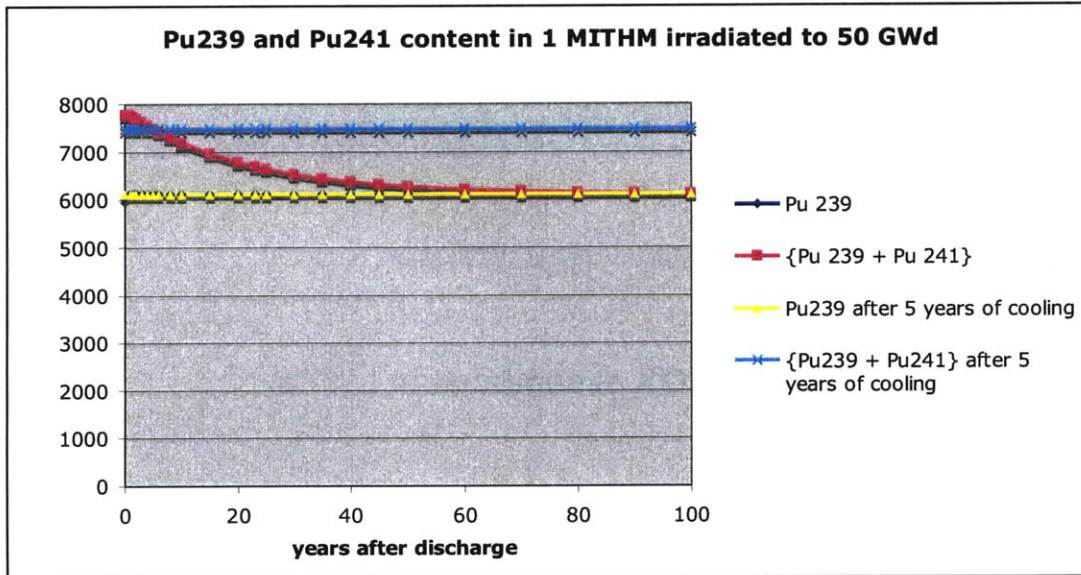
Figure 5.1 - Fissile Pu content in spent UO$_2$ fuel over time

Between 5 years and 15 years of cooling (the average age of the U.S. spent fuel legacy is 15 years), the Pu fissile content in spent UO$_2$ fuel (and therefore the amount of MOX fuel that one can make) is reduced by 7% (from 0.75%$_w$ to 0.70%$_w$).

Between 5 years and 30 years of cooling (the average age of the current U.S spent fuel will be about 30 years in 2025), the Pu fissile content is reduced by 13% (from 0.75%$_w$ to 0.65%$_w$).

Between 5 years and 100 years of cooling, the Pu fissile content is reduced by 18% (from 0.75%$_w$ to 0.61%$_w$). Actually there is virtually no ^{241}Pu left then.

- As for any other fuel in CAFCA, the MOX fuel fabrication plants are not considered as distinct industrial facilities. There is no constraint on the supply side for the fabrication of fuel. This approximation is reasonable as there is

already a constraint on the reprocessing plants' building rate, which limits the downstream supply of plutonium and TRU fuel.

- In case the number of LWRs loaded with MOX is limited to the point that some of the Pu available is not needed, this Pu is not used to make MOX. Depending on considerations related to economics and proliferation resistance, it would be more desirable to use it though.[11]

- The code does not track the LWRs individually: it can be known at any time how many LWRs are loaded with all-UO_2 fuel, MOX fuel, and young or old CONFU fuel, but not precisely which ones, and how old they are. In particular, it is not possible to know what a LWR that is in its first year or last year of operation is loaded with. Hence, for the sake of simplification, it is assumed that those specific LWRs are all loaded with all-UO_2 fuel. In general, for a given reactor, the transition regime between an all-UO_2 fuel core and a MOX fuel core is ignored.

5.2.2 MOX technology structure-policy diagram

This part uses the same terminology as in [Busquim et Al., 2008], briefly recalled in Chapter 2. The *Front-end module* is also modified.

The system-inputs of the MOX structure-policy diagram are:

- The *separated Pu* available for MOX fuel fabrication, PU_{MOX}^{FUEL}, produced in thermal reprocessing plants.
- The number of LWRmf in operation, $F_{LWR2mf}(t)$ and $F_{LWR3}(t)$.

The main system-output is the number of LWR2 and LWR3 reactors loaded with MOX fuel, $F_{MOX2}^{LWR2}(t)$ and $F_{MOX3}^{LWR3}(t)$ respectively. Secondary inputs include the costs associated with the MOX fuel cycle and the rates of generation of spent MOX fuel (after cooling storage) $SF_{MOX2}(t)$, and $SF_{MOX3}(t)$.

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[11] Actually, the worldwide fabrication of MOX fuel has not kept pace with the production of separated Pu so far, due to economic and political reasons. As a result there are today more than 200 metric tons of separated plutonium in storage around the world.

The model is described in figure 5.2, assuming the precedence order of loading fuel first to LWR2mf units and then to LWR3s.
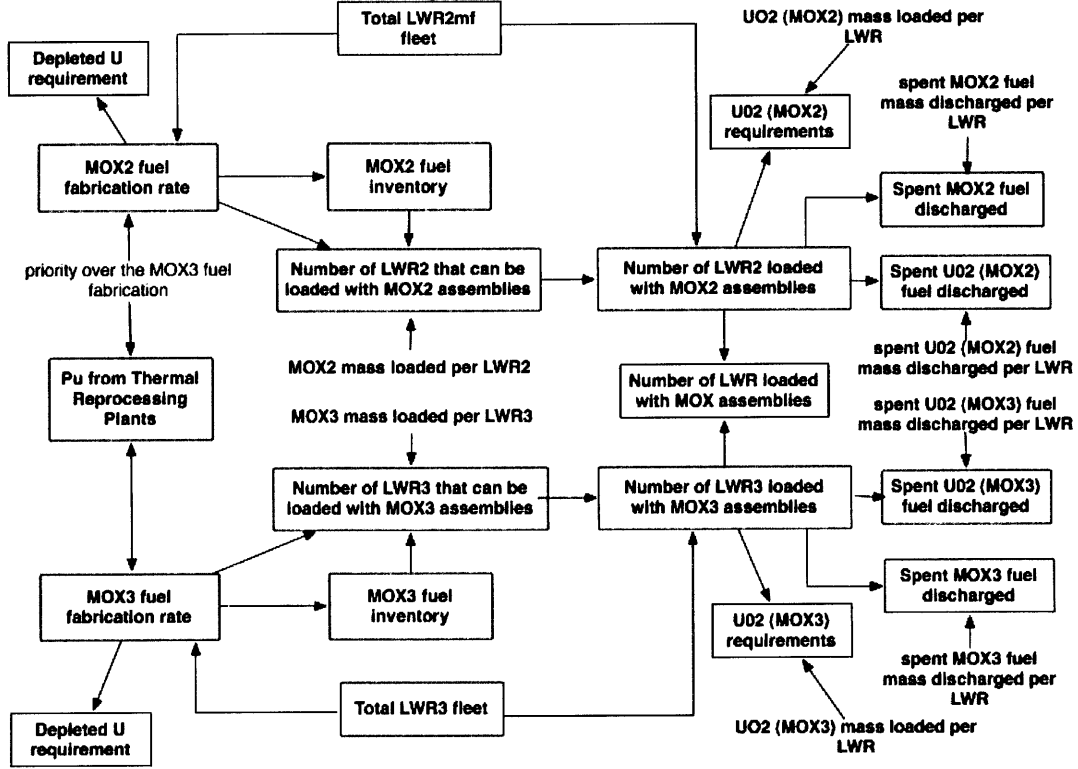
Figure 5.2 - MOX technology structure-policy diagram of the system

The structure-policy diagram is driven by the *mass of Pu available for MOX fuel fabrication per year*, $PU_{MOX}(t)$, equal to 0 before the *MOX introduction date* and then modeled as:

$$PU_{MOX}(t) = Delay(P^{Pu} \cdot F_{ThRP}(t) \cdot NC_{ThRP} \cdot \langle P_{Pu} \rangle, T_{ThRP}) + R_{UR}^{Pu_{Inv}}(t) \qquad (5.1)$$

$$\text{with } \langle P_{Pu} \rangle = \frac{P_{Pu}^{UO2_{All-UO2}} \cdot SF_{UO2_{All-UO2}}(t) + P_{Pu}^{UO2_{MOX2}} \cdot SF_{UO2_{MOX2}}(t) + P_{Pu}^{UO2_{MOX3}} \cdot SF_{UO2_{MOX3}}(t)}{SF_{UO2}(t)} \qquad (5.2)$$

where

- T_{ThRP} is the transit time in thermal reprocessing plants,

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- P^{Pu} is a user input that defines the *ratio of reprocessed spent fuel that went through the PUREX[12] process*,

- $P_{Pu}^{UO2_{All-UO2}}, P_{Pu}^{UO2_{MOX2}}$ and $P_{Pu}^{UO2_{MOX3}}$ are the Pu fraction in the spent UO_2 fuel discharged from All-UO2 LWRs, LWRs-MOX2, and LWRs-MOX3, respectively,

- $SF_{UO2_{All-UO2}}(t)$ is the amount of *all-U0$_2$ fuel discharged per year, after cooling storage* (sent to separation plants),

- $SF_{UO2_{MOX2}}(t)$ is the amount of *U0$_2$ (MOX2) spent fuel discharged per year, after cooling storage*,

- $SF_{UO2_{MOX3}}(t)$ is the amount of *U0$_2$ (MOX3) spent fuel discharged per year after cooling storage*,

- $SF_{UO2}(t)$ is *the total amount of spent U0$_2$ fuel discharged per year, after cooling storage* and sent to the thermal reprocessing plants i.e:

$$SF_{UO2}(t) = SF_{UO2_{All-UO2}}(t) + SF_{UO2_{MOX2}}(t) + SF_{UO2_{MOX3}}(t)$$ (5.3)

- NC_{SP} is the *Thermal Reprocessing Plants Nominal Capacity*.

- $F_{ThRP}(t)$ is the *Total Fleet of Thermal Reprocessing Plants*.

- $R_{UR}^{PU_{Inv}}(t)$ is the rate of utilization of the inventory of separated plutonium $S_{Pu}(t)$ that has been cumulated over the period between the *Pu separation introduction date* and the *MOX introduction date*. It is generally assumed that this inventory is converted into MOX fresh fuel within one time step so that

$$R_{UR}^{PU_{Inv}}(t) = \begin{cases} S_{Pu}(t) \text{ when t} = \text{MOX introduction date} \\ 0 \text{ } else \end{cases}$$ (5.4)

- **MOX2 fuel assemblies**

It is assumed that the plutonium available is used to the largest extent possible to make MOX, either MOX2 or MOX3. First, the *Pu available for MOX2 fuel fabrication* is

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[12] PUREX: both plutonium and uranium are separated from the spent fuel but the minor actinides are left with the fission products (High-Level Waste), as opposed to pyroprocessing, in which both the plutonium and the minor actinides ("TRU") would be incorporated in fresh fuel (CONFU or fast reactor fuel).

capped so as not to fabricate too much MOX2 fuel with respect to the number of LWR2mf:

$$PU_{MOX2}(t) = \text{MIN} \left(F_{LWR2mf}(t) \cdot \overline{M}_{MOX2} \cdot \overline{P}_{Pu}^{MOX2}, PU_{MOX}(t) \right) \tag{5.5}$$

where $\overline{P}_{Pu}^{MOX2}$ is the *fraction of Pu in the MOX2 fuel* and $\overline{M}_{MOX2}$ is the *mass of MOX2 loaded per year per LWR*. All this Pu is used to make MOX2 fuel. Hence, the *MOX2 inflow rate*, $R_{IR}^{MOX2}(t)$, is equal to

$$R_{IR}^{MOX2}(t) = Delay \left(\frac{(1 - L_{MOX2}) \cdot PU_{MOX2}(t)}{\overline{P}_{Pu}^{MOX2}}, T_{MOX\ fab} \right) \tag{5.6}$$

where L_{MOX2} are the *MOX2 fuel fabrication losses* and $T_{MOX\ fab}$ is the MOX fuel fabrication time.

Next, the *MOX2 inventory,* $S^{MOX2}(t)$, is modeled as

$$\frac{dS^{MOX2}(t)}{dt} = R_{IR}^{MOX2}(t) - R_{UR}^{MOX2}(t), \ S_0^{MOX2} \tag{5.7}$$

where $S_0^{MOX2} = S^{MOX2}(0) = 0$, and $R_{UR}^{MOX2}(t)$ is the *MOX2 utilization rate*. $R_{UR}^{MOX2}(t)$ is modeled as

$$R_{UR}^{MOX2}(t) = F_{LWR2}^{MOX2} \cdot \overline{M}_{MOX2} \tag{5.8}$$

where F_{LWR2}^{MOX2} is the actual *number of LWR2 loaded with MOX2*. This output still needs to be determined: it is actually the maximum between the *maximum number of LWRs loaded with MOX2 assemblies,* $F_{MAX}^{MOX2}(t)$, permitted by the MOX2 fuel inventory as well as its inflow rate, modeled as

$$F_{MAX}^{MOX2}(t) = \frac{S^{MOX2}(t) + R_{IR}^{MOX2}(t)}{\overline{M}_{MOX2}} \tag{5.9}$$

and the *number of LWR2mf* F_{LWR2mf}.

Thus:

$$F_{LWR2}^{MOX2}(t) = \text{Max} \left(F_{MAX}^{MOX2}(t), \; F_{LWR2mf}(t) \right) \tag{5.10}$$

Finally, the *mass of MOX2 fuel discharged per year from reactors* (expressed in MTIHM) is modeled as:

$$M_{MOX2}^{SF} = F_{LWR2}^{MOX2} \cdot \overline{M}_{MOX2} \tag{5.11}$$

A minimum cooling time, denoted CT_{MOX2}, is applied to this variable, resulting in the *mass of MOX2 fuel discharged per year from reactors after cooling storage* SF_{MOX2}:

$$SF_{MOX2} = Delay(M_{MOX2}^{SF}, CT_{MOX2}) \tag{5.12}$$

- **MOX3 fuel assemblies**

The calculations are the same as those for the MOX2, except that the amount of Pu available per year may be reduced, as LWR2s have priority for MOX loading. Hence the *Pu available for MOX3 fuel fabrication* is modeled as:

$$PU_{MOX3}(t) = PU_{MOX}(t) - PU_{MOX2}(t) \tag{5.13}$$

Taking this variable as the main input, the methods used to calculate the *MOX3 fuel inflow rate* $R_{IR}^{MOX3}(t)$, the *number of LWRs loaded with MOX3 fuel* F_{LWR3}^{MOX3}, the *MOX3 fuel inventory* $S^{MOX3}(t)$, and the *mass of MOX3 fuel discharged per year from reactors after cooling storage* SF_{MOX3} are exactly the same as in the MOX2 case.

o **Depleted Uranium**

MOX fuel pins are a blending of plutonium/americium oxide and uranium oxide. To make the MOX fuel, the uranium used is depleted (0.25%$_w$ enriched).

The *depleted uranium utilization rate for MOX fuel fabrication*, $R_{UR}^{DU\,for\,MOX}(t)$, is modeled as

$$R_{UR}^{DU\,for\,MOX}(t) = \frac{PU_{MOX2}(t)}{\overline{P}_{Pu}^{MOX2}} \cdot (1 - \overline{P}_{Pu}^{MOX2}) + \frac{PU_{MOX3}(t)}{\overline{P}_{Pu}^{MOX3}} \cdot (1 - \overline{P}_{Pu}^{MOX3}) \qquad (5.14)$$

- **UO₂ fuel assemblies**

As a complement to the MOX assemblies, the core may be loaded with UO₂ assemblies (non-100% MOX core configurations. The *Front-end module* is applied (with the parameters specific to the MOX cases, see 2.3.2) to calculate the front-end requirements (natural uranium, conversion, enrichment and fuel fabrication capacity). These requirements are aggregated with those for the fabrication of UO₂ for all- UO₂ fuel cores, young CONFU fuel cores and old CONFU fuel cores.

The main input is the *UO₂ fuel utilization rate for LWR loaded with MOX i* $M_{LWR}^{UO2\,MOXi}$ (i=2,3):

$$M_{LWR}^{UO2\,MOXi} = F_{LWRi}^{MOX\,i} \cdot \overline{M}_{LWR}^{UO2\,MOXi} \qquad (5.15)$$

where $\overline{M}_{LWR}^{UO2\,MOXi}$ is the *mass of UO₂ loaded per year per LWR that is loaded with a MOX i core configuration* (i=2,3).

The *mass of UO2_{MOX i} fuel discharged per year from reactors* (expressed in MTIHM) is modeled as (i=2,3):

$$M_{UO2_{MOXi}}^{SF} = F_{LWRi}^{MOX\,i} \cdot \overline{M}_{LWR}^{UO2\,MOXi} \qquad (5.16)$$

A minimum cooling time, denoted $CT_{UO2_{MOXi}}$, is applied to this variable, resulting in the *mass of MOX i fuel discharged per year from reactors after cooling storage* $SF_{UO2_{MOXi}}$ (i=2,3):

$$SF_{UO2_{MOXi}} = Delay(M_{UO2_{MOXi}}^{SF}, CT_{UO2_{MOXi}}) \qquad (5.17)$$

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5.2.3 ThRP structure-policy diagram: update

The thermal reprocessing plant (ThRP) structure-policy diagram has been modified in order to integrate the possibility of separating the plutonium (and the minor actinides) from the spent UO2 fuel and the possibility of reprocessing the spent MOX fuel (for recycling in fast reactor cores). It is indeed assumed in CAFCA that the same plants perform the reprocessing of the spent UO_2 fuel (generating either TRU/U/FP streams or Pu/MA/U/FP streams) and of the spent MOX fuel (generating TRU/U/FP streams).

The algorithm applied to compute the number of thermal reprocessing plants in operation has remained unchanged (see Section 2.4.2.1 in [Busquim et al., 2008]).

- Plutonium/MA separation

Unlike the other advanced fuel cycle schemes implemented in CAFCA (CONFU, fast reactors), which require the extraction of the TRU from the spent UO_2 fuel, the MOX option only requires the separation of plutonium[13]. The management of the minor actinides depends upon whether fast-reactor recycling is part of the scenario (2-tier scenario). If that is the case, the minor actinides are extracted at the same time as the plutonium from the spent UO_2 fuel and stored until loading in the fast reactors' cores, along with the TRU separated from the spent MOX fuel. In the other case (twice-through cycle scenario), minor actinides are sent to disposal along with the fission products.

The implementation of the separation of Pu from spent LWR fuel in CAFCA follows exactly the same methods as the separation of TRU (see Section 2.4.2.1 in [Busquim et al., 2008]).

- Spent MOX fuel reprocessing

In the 2-tier scenarios (see Section 2.6), the spent MOX fuel is sent to thermal reprocessing plants after a period of cooling. As it is assumed that the same plants reprocess both spent UO_2 fuel and spent MOX fuel, the latter must be aggregated in the main input for the Thermal Reprocessing Plant Structure-Policy Diagram (see Section

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[13] The current standard method is PUREX (standing for Plutonium and Uranium Recovery by EXtraction). The PUREX method results in the extraction of both uranium and plutonium, independent of each other, from the fission products.

2.4.2.1 in [Busquim et al., 2008]). Hence the input to calculate the *number of thermal reprocessing plants permitted from inventory* is

$$S_{Th} = S_{UO2\,SF} + S_{MOX\,SF} \tag{5.18}$$

where $S_{UO2\,SF}$ and $S_{MOX\,SF}$ are the inventories of spent UO_2 fuel and spent MOX fuel (after cooling storage), respectively.

The input to calculate the *number of thermal reprocessing plants permitted from spent fuel rate* is

$$SF_{Th} = SF_{UO2} + SF_{MOX} \tag{5.19}$$

The second modification refers to the allocation of the reprocessing capacity between the spent MOX fuel and the spent UO_2 fuel. Using [Busquim et al, 2008]'s terminology, the *desired spent UO_2 fuel reprocessing rate* $R_{DR}^{ThRP_{UO2}}(t)$ and the *desired spent MOX fuel reprocessing rate* $R_{DR}^{ThRP_{MOX}}(t)$ are modeled as:

$$R_{DR}^{ThRP_{UO2}} = R_{DR}^{ThRP} \cdot \frac{S_{UO2\,SF}}{S_{Th}} \tag{5.20}$$

$$\text{and } R_{DR}^{ThRP_{MOX}} = R_{DR}^{ThRP} \cdot \frac{S_{MOX\,SF}}{S_{Th}} \tag{5.21}$$

where $R_{DR}^{ThRP}(t)$ is the *desired spent LWR fuel reprocessing rate*, defined as

$$R_{DR}^{ThRP}(t) = NC_{ThRP} \cdot F_{ThRP} \tag{5.22}$$

where NC_{ThRP} is the *nominal capacity* of the thermal reprocessing plants and F_{ThRP} is the *number of thermal reprocessing plants*.

Finally, the actual *spent UO_2 fuel reprocessing rate* $R_{RR}^{ThRP_{UO2}}(t)$ is modeled as the minimum between the *desired spent UO_2 fuel reprocessing rate* $R_{DR}^{ThRP_{UO2}}(t)$ and *the maximum spent fuel utilization rate* due to inventory $R_{PR}^{ThRP_{UO2}}(t)$:

$$R_{RR}^{thRP_{UO2}}(t) = Min(R_{DR}^{thRP_{UO2}}, R_{PR}^{thRP_{UO2}}) \qquad (5.23)$$

$R_{PR}^{thRP_{UO2}}(t)$ is the ratio between the inventory of spent UO_2 fuel and the simulation time step:

$$R_{PR}^{ThRP_{UO2}}(t) = \frac{S_{UO2\,SF}}{TimeStep} \qquad (5.24)$$

Eventually, the *spent UO2 fuel inventory* $S_{UO2\,SF}(t)$ is modeled as:

$$\frac{dS_{UO2\,SF}(t)}{dt} = SF_{UO2} - R_{RR}^{ThRP_{UO2}} \qquad (5.25)$$

The same methods are applied to calculate the spent MOX fuel reprocessing rate $R_{RR}^{ThRP_{MOX}}(t)$ and the spent MOX fuel inventory $S_{MOX\,SF}$.

5.3 Fast Reactors

5.3.1 Introduction

Two types of Fast Reactors were implemented in the previous version of CAFCA, as described in [Busquim et al., 2008]. The GFR (conversion ratio CR=1.0) has the characteristics to be self-sufficient, while the ABR (conversion CR=0.0) is loaded with pure TRU fuel in non-fertile host material (zirconium) in the entire load of assemblies. Therefore, neither of the two structure-policy diagrams associated with the GFR and the ABR could be applied to a fast reactor of another conversion ratio: a burner of conversion ratio lower than 1.0 needs an external source of fissile materials over its entire life (unlike the GFR) as well as a supply of uranium (unlike the ABR) while a breeder (conversion ratio higher than one) provides fissile materials for other reactors (unlike both the ABR and the GFR).

A generic module (simple extension of the GFR model) was implemented to model the front-end part of the FR fuel cycle for any conversion ratio. An advantage of this generic module is a simplification of the code: the ABR structure-policy diagram and the GFR structure-policy diagram were replaced by a unique FR structure-policy diagram. The utilization of a vector allows the implementation of an unlimited number of designs.

The principle of the model is simple. Two sources of fissile materials are available for FR

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fuel fabrication: the TRU separated from the spent LWR fuel in thermal reprocessing plants and the U-TRU mix separated from the spent FR fuel in fast reprocessing plants. The TRU must be mixed with depleted uranium while the U-TRU mix must be mixed with either depleted uranium or TRU in proportions depending upon the conversion ratio. The resulting streams of U-TRU fuel are lumped together in a U-TRU fuel inventory. It is therefore assumed that there is no difference between the fuel made out of TRU coming from external sources and the fuel coming from self-recycling[14].

5.3.2 FR structure-policy diagram

The system-input to the FR structure-policy diagram is *the TRU available for FR fuel fabrication per year* $TRU_{FR}^{fuel}(t)$ and the *reprocessed FR U-TRU per year* $FR_{FRP}^{U-TRU}(t)$ coming from fast reprocessing plants. The system-output is the number of FRs under commercial operation.

P_U^{TRU} is the fraction of uranium in the fuel made from separated TRU (from ThRP). Thus:

$$P_U^{TRU} = 1 - P_{TRU}^{FR\,fuel} \qquad (5.26)$$

where $P_{TRU}^{FR\,fuel}$ is the TRU fraction in FR fuel.

P_U^{U-TRU} is the *fraction of added U* (uranium that had to be added to the U-TRU mix in order to obtain the desired concentration for the fresh fuel) in the fuel made from separated U-TRU (from FRP).

$$P_U^{U-TRU} = Max(0, 1 - \frac{P_{TRU}^{FR\,fuel}}{P_{TRU}^{U-TRU\,from\,FRP}}) \qquad (5.27)$$

where $P_{TRU}^{U-TRU\,from\,FRP}$ is the TRU fraction in the spent FR fuel after cooling storage (expressed in % of the initial heavy metal mass).

P_{TRU}^{U-TRU} is the final *fraction of added TRU* (TRU that had to be added to the U-TRU mix

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[14] The [Hoffman et al., 2006] study, from which the data used in this study are taken, assumes the "equilibrium" ratio between the two sources of TRU.

in order to obtain the desired concentration for the fresh fuel) in the fuel made with separated U-TRU.

$$P_{TRU}^{U-TRU} = Max(0, 1 - \frac{P_{TRU}^{FR\,fuel} - P_{TRU}^{U-TRU\,from\,FRP}}{1 - P_{TRU}^{U-TRU\,from\,FRP}})$$ 　(5.28)

Hence the amount of *TRU mixed with the separated U-TRU per year* $TRU_{FR}^{Old\,fuel}(t)$:

$$TRU_{FR}^{Old\,fuel}(t) = P_{TRU}^{U-TRU} \frac{FR_{FRP}^{U-TRU}}{1 - P_{TRU}^{U-TRU} - P_{U}^{U-TRU}}$$ 　(5.29)

Which leaves the *TRU available for young FR fuel per year* $TRU_{FR}^{Young\,fuel}(t)$:

$$TRU_{FR}^{Young\,fuel}(t) = TRU_{FR}^{fuel}(t) - TRU_{FR}^{Old\,fuel}(t)$$ 　(5.30)

Finally the FR fuel fabrication rate is modeled as:

$$R_{IR}^{FR\,fuel}(t) = Delay(\frac{TRU_{FR}^{Young\,fuel}(t)}{1 - \overline{P}_{U}^{TRU}} + \frac{FR_{FRP}^{U-TRU}(t)}{(1 - \overline{P}_{U}^{U-TRU} - \overline{P}_{TRU}^{U-TRU})}, T_{FR\,fuel\,fab})$$ 　(5.31)

where $T_{FR\,fuel\,fab}(t)$ is the *FR fuel fabrication time.*

Table 5.1 shows the values $P_{TRU}^{FR\,fuel}, P_{U}^{TRU}, P_{TRU}^{U-TRU\,from\,FRP}, P_{U}^{U-TRU}$ and P_{TRU}^{U-TRU}. Recall that these fractions are for the FR fresh fuel made from self-recycling.

Table 5.1 - Parameters for the front-end part of the FR structure-policy diagram

(fraction of heavy metal masses)					
Conversion Ratio	0.0	0.5	0.75	1.0	1.23
TRU fraction in FR fuel as loaded	98.59%	33.32%	21.21%	13.86%	8.90%
U fraction in FR fuel	1.41%	66.68%	78.79%	86.14%	91.10%
TRU fraction in U-TRU mix from FRP[15]	97.90%	31.50%	21.50%	15.20%	10.99%
U that had to be added: fraction in fresh FR fuel	0	0	1.35%	8.79%	19.05%
TRU that had to be added: fraction in fresh FR fuel	32.76%	2.66%	0	0	0

o **Depleted Uranium**

MOX fuel pins are a blending of plutonium/americium oxide and uranium oxide. The data used for the MOX fuel assumes that the uranium used is depleted ($0.25\%_w$ enriched). The *depleted uranium utilization rate for FR fuel fabrication*, $R_{UR}^{DU\ for\ FR}(t)$, is modeled as

$$R_{UR}^{DU\ for\ FR}(t) = P_U^{TRU}\frac{TRU_{FR}^{Young\ fuel}(t)}{P_{TRU}^{FR\ fuel}} + P_U^{U-TRU}\frac{FR_{FRP}^{U-TRU}(t)}{1 - P_U^{U-TRU} - P_{TRU}^{U-TRU}} \qquad (5.32)$$

The rest of the structure-policy diagram is identical to the GFR's (see Section 2.4.2.4 in [Busquim et al., 2008]).

5.4 Recovered uranium utilization

Even though the primary purpose of reprocessing is separating and recycling of fissile materials (TRU), the uranium contained in the spent UO_2 fuel (about $95\%_w$) is also recovered in the process.

This uranium has been downgraded while being irradiated, from about $4.5\%_w$ enrichment to less than $1\%_w$. But the spent fuel content in fissile ^{235}U may be greater than that of natural uranium ($0.711\%_w$), which makes recovered uranium a possible source of ^{235}U for the fabrication of UO_2 fuel.

However, in order to compensate for the presence of various undesirable isotopes (notably ^{236}U, a neutron absorber) in recovered uranium, the required enrichment in ^{235}U

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[15] defined as the weight percentage of the TRU contained in the U-TRU mix produced by the fast reprocessing plants.

is greater than if natural uranium were used. [Bunn et al., 2003] uses the following formula[16] to determine that required enrichment x_{rp}:

$$x_{rp} = \frac{x_p}{1 - 0.21\dfrac{x_{236}}{x_{rf}}}$$

(5.33)

where:

x_{236} = ^{236}U concentration in recovered U

x_{rf} = ^{235}U concentration in recovered U

x_p = product ^{235}U enrichment when the source is natural uranium : 4.23 % $_w$ in the model of LWR considered in this study (see Section 2.2.2)

The concentrations of ^{235}U and ^{236}U in recovered uranium are dependant on the fuel burnup. For the level of 50 MWd/kg, they are : x_{236} = 0.60 % $_w$, x_{rf} = 0.82 % $_w$ and x_p= 4.23 % $_w$. The result obtained is x_{rp} = 5.00 % $_w$.

Assuming that the enrichment process leaves the tails with 0.25 % $_w$ of U_{235}, the production of 1 kgHM of UO_2 fuel requires 8.38 kg of recovered U or 8.70 kg of natural U. Therefore, the recovery of 1 kg of uranium out of the spent UO_2 fuel allows us to save 1.04 kg of natural uranium.

CAFCA-SD's *Front-end module* has been modified to include the possibility of using recovered uranium as fuel in LWRs. If the option is activated by the user, the recovered uranium produced in thermal reprocessing plants immediately (no transit time) goes through the successive front-end services (conversion, enrichment, fuel fabrication). The UO_2 made from recovered uranium is then mixed with the UO_2 made from natural uranium.

Front-end service requirements for all types of UO_2 fuel (UO_2 fuel made from natural uranium for all-UO_2 cores, UO_2 fuel made from recovered uranium for all-UO_2 cores, UO_2 fuel for cores loaded with MOX2, UO_2 fuel for cores loaded with MOX3, UO_2 fuel for CONFU cores) are finally aggregated.

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[16] Coming from *Plutonium Fuel: An Assessment* (p158), Nuclear Energy Agency, Paris: Organization for Economic Development and Cooperation, 1989.

5.5 Waste management

The notion of waste is not intrinsic; materials are waste only if they are considered as such, which depends on the scenario chosen. CAFCA only tracks the high-level wastes (HLW), indicated by the grey cells in table 5.2, for the 4 scenarios considered (NA = Not Applicable).

Table 5.2 - HLW in the four scenarios

HLW \ Scenario	Once-Through Cycle	Twice-Through Cycle (MOX)	Fast Reactors Strategy	Two-Tier Strategy (MOX→FR)
Spent UO$_2$ fuel	▓			
FP in spent UO$_2$ fuel	NA	▓	▓	▓
MA in spent UO$_2$ fuel	NA	▓		
MOX fuel fabrication losses	NA	▓		▓
Spent UO$_2$/MOX fuel reprocessing losses	NA	▓	▓	▓
Spent MOX fuel	NA	▓	NA	
FR fuel fabrication losses	NA	NA	▓	▓
FP in spent FR fuel	NA	NA	▓	▓
FR spent fuel reprocessing losses	NA	NA	▓	▓
Spent FR fuel	NA	NA		

As high-level wastes have various levels of decay heat and radiotoxicity, both varying over time, one cannot simply aggregate their masses to make comparisons between the scenarios. Moreover, CAFCA does not track the isotopes and hence ignores their decay, which prevents a thorough study of the waste management. However, one of the two following approaches can still be adopted:

1) CAFCA can provide separate inventories of the various waste types (fission products, minor actinides, spent fuel, TRU content) which allows whoever is familiar with the properties of these wastes to form a rough assessment of the advantages of each cycle (qualitative assessment).

2) "Densification factors" may be used to aggregate the different types of waste in order to compare the total repository requirements. A definition of the densification factor can be found in [BCG, 2006]: "the quality of HLW or used

fuel that can be disposed per unit length of Yucca Mountain is […] referred to as the "*drift loading factor*" and is expressed in MTHM/m$_{YM}$. […] *The densification factor is the ratio of the drift loading factor of HLW to the drift loading factor of used fuel*[17]". Two potential constraints are considered: volume and heat (taking the waste package into account). In all studies, a total cooling time of 25 years prior to disposal in repository is assumed, and the conditions of the repository are those of Yucca Mountain (the repository is assumed to be ventilated for 75 years after repository closure). The values of the densification factors are also sensitive to the assumptions about the burnup, the cooling time before reprocessing (e.g the build-up of ^{241}Am from ^{241}Pu decay drives up long-term heat) and far above all to the amounts of TRU, Cesium and Strontium remaining in the spent fuel, as shown by Figure 5.2 (Source: [Wigeland, 2006]).
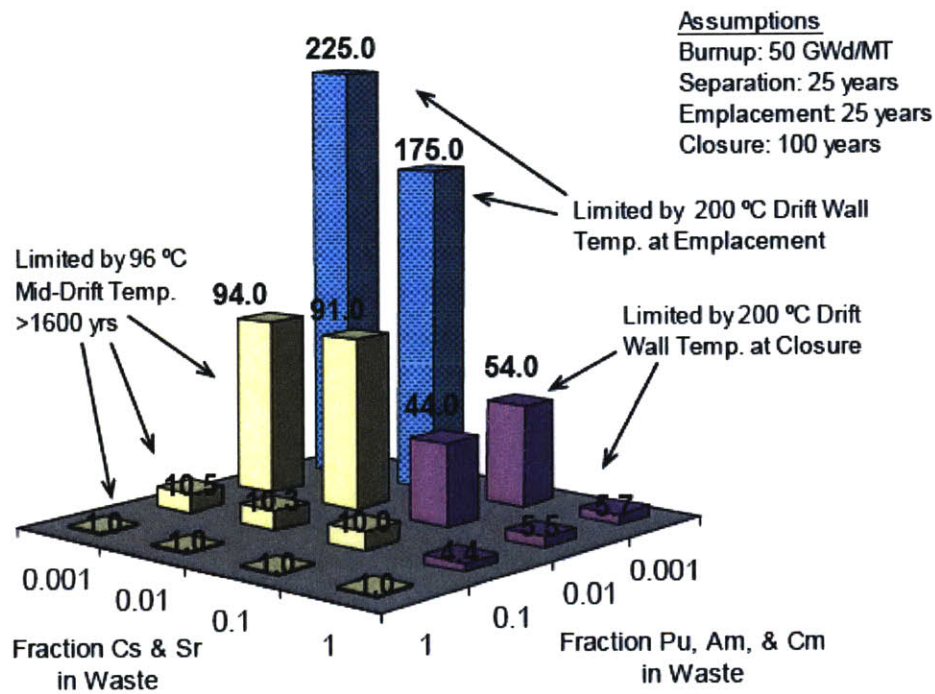
Figure 5.3 - Densification factors as a function of the composition of the HLW [Wigeland, 2006].

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[17] The "used fuel" taken as a reference (densification factor of 1) is a spent UO$_2$ fuel burnt to 50 MWd/kgHM.

Recall that it is assumed in CAFCA that 99.9%$_w$ of the Pu or TRU (depending on the scenario) is removed from spent fuel over the reprocessing process[18].

The densification factor is generally directly translated into repository costs.[19]

    a. The densification factor found in [BCG, 2006] for the FP/MA mix resulting from spent UO_2 fuel reprocessing in the TTC scenario is ~ 4.

    b. The [Shropshire et al., 2009] study suggests a densification factor from 2 to 10 for the FPs alone (separated from spent UO_2 fuel), with a nominal value of 2.5. This value is more pessimistic than the [BCG, 2006] estimation, as removing the minor actinides should lead to a higher densification factor. However, [Wigeland and Bauer, 2004] find that, with 99.9% removal of plutonium and americium, the densification factor can be a factor of 5-6[20]. Finally, there is no difference in treatment between the fission products that are separated from UO_2 spent fuel and those separated from spent MOX fuel (in 2-tier scenarios) and spent FR fuel.

    c. The densification factor found in [BCG, 2006] for the spent MOX fuel is ~0.15, a low number due to its very high heat content, caused by greater quantities of americium and curium.

Table 5.3 summarizes the densification factors used in our study. Note that the densification factor has a different definition when it comes to spent MOX fuel. Indeed, in the case of the FP (and it is the same principle for FP/MA), a densification factor of 5

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[18] This performance has been obtained only on small-scale extractions so far. Only 99% may be feasible for large-scale facilities; 99.9% is therefore assumed to be a target for the future. Note that the choice between theses two assumptions has little incidence on the densification factor when Cs and Sr are not removed (5.5 for 99% vs. 5.7 for 99.9% [Wigeland, 2006]). The 99.9% assumption is deemed to be realistic in [NEA, 2002]: *"The value of 0.1% for the reprocessing losses is an extrapolation from the current technology to a technology which can be expected to work at a time when transmutation systems could be introduced on a larger scale. The extrapolation is based on expected and partly at laboratory scale proven advances in the wet and dry reprocessing technology. The assumptions are comparable to assumptions which have been made in other national and international transmutation studies".* [Wigeland and Bauer, 2004] makes the same assumption (not including Curium and Neptunium though): *"Plutonium and americium are]* [assumed to be] *separated from the spent PWR fuel with an efficiency of 99.9% to address the repository heat load issue ".*

[19] The [Shropshire et al., 2009] study directly uses the densification factor as a cost ratio.

[20] [Wigeland and Bauer, 2004] also shows that removing cesium and strontium (they would be sent to a short-term separate repository or another facility) would lead to a densification factor of 40-50, but this option is not considered in our scenarios.

means that the FP extracted from 1kgIHM of spent UO_2 fuel (i.e. *51.6 g of FP*) have a repository requirement five times lower than that of 1 kgIHM of spent UO_2 fuel. In the case of the spent MOX fuel, a densification factor of 0.15 means that *1 kgIHM of spent MOX fuel* has a larger repository requirement (1/0.15 or 6.7 greater) than that of 1 kgIHM of spent UO_2 fuel.

Table 5.3 - Densification factors for different types of wastes

HLW type	Densification factor
Spent UO_2 fuel	1
Spent MOX fuel	0.15
FP/MA mix	4
FP	5

5.6 TRU tracking module

The *TRU tracking module* tracks at any time in the simulation the total amount of TRU in the system as well as its locations. There are several locations for TRU: fresh fuel inventories, fuel in reactor cores, spent fuel in cooling storages, spent fuel in interim storages, spent fuel in transit in reprocessing plants, separated TRU in storage, wastes in interim storage, and wastes in repositories.

As CAFCA does not employ any depletion calculations, it is assumed that the fuel located in reactor cores has the same TRU content as the spent TRU fuel. This assumption overestimates the TRU content in the UO_2 cores and breeder cores, and underestimates it in the MOX cores and fast burner cores.

5.7 Summary

This chapter presented in detail the modifications brought to CAFCA in order to implement some of the options and outputs considered in this study. Namely: the FR structure-policy diagram (usable for any conversion ratio), the MOX structure-policy

diagram, the thermal reprocessing ThRP structure-policy diagram (with the capability to reprocess both spent UOX fuel and spent MOX fuel, and to separate either only plutonium or TRU), the recovered uranium module, the waste assessment module, and the TRU tracking module.

Chapter 6

6. Economics in CAFCA

6.1 Introduction and definitions

6.1.1 Introduction

CAFCA includes a separate module for economic analysis of the scenarios considered. This module provides final outputs that have no impact on the rest of the simulation. This notably implies that CAFCA simulations are not driven by any cost-minimizing scheme. However, it does not mean that the economics were completely ignored in building the physical part of the code: for purposes of realism, both fast reactors and reprocessing plants are built only if guaranteed to be fed fuel at a minimum level over their entire lifetime. This minimum level is called the "minimum loading factor" and is a user choice. In reality, such guaranty could materialize by contracts for fuel supply and reprocessing demand, or legal requirements for utilities to reprocess their spent fuel, as practiced in some European countries. Even in the absence of contracts, investors would not finance a reprocessing plant if they were not expecting some minimum utilization.

Many nuclear services are involved in the whole fuel cycle, at the front-end (ore purchase, conversion, enrichment, fuel fabrication) as well as at the back-end (spent fuel interim storage, reprocessing, waste disposal). With the notable exceptions of both reprocessing and disposal (see Sections 6.4.3.5 and 6.4.3.6), all these industries are assumed to be mature and market-driven as soon as they are introduced in the scenario; the supply is assumed to match the demand whatever its fluctuations. Costs of production are considered rather than spot prices, and unit costs are assumed to be constant (in real dollars) and hence independent from the demand (with the exception of the natural uranium price with a U price model that can be optionally used, see Section 6.4.3.1).

Furthermore, it is assumed that:

- Costs of development (including research, development and demonstration facilities) are not explicitly taken into account. They are reflected in the capital costs of the plants.

- Even if times between reactor refuelings are different from 1 year (e.g.18 months for the LWR), yearly fuel loadings are assumed for numerical purposes.

- Costs are those of nth-of-a-kind facilities, provided that such data are available (operational improvements and cost reduction are generally observed between the first-of-a-kind and nth-of-a-kind facility).

- An overall tax rate (composite marginal corporate income tax) is assumed. Unlike return on equity, interest on debt is tax-deductible expenses. Property taxes are neglected.

- Insurance costs are neglected

- Inflation is not considered in the model (real costs, expressed in 2008 dollars).

- As market supplies and demands are always assumed to be balanced, as technical and practical improvements may compensate for increasing costs in uncertain proportions, we assume no escalation in unit costs (constant prices for services). One exception to this assumption is the price of uranium, which may vary when the U price model is used (see Section 6.4.3.1).

- All expenses are accounted for at the beginning of the year in which they occur while revenues are collected at the end of the year in which the electricity is produced.

- Reactors and reprocessing plants have negligible salvage value.

- Disposal costs for LLW are not accounted for explicitly. They are part of the Operations and Maintenance costs of reactors, and part of the service prices at fuel recycling facilities.

Economics outputs include:

    - NPP owners' total outlays (cash flows in B$)

    - The Cost of Electricity (in mills/kWh),

    - The (Dynamic) Levelized Cost Of Electricity (in mills/kWh),

These items are calculated using discounted cash-flow analysis.

6.1.2 Financial Parameters

It is assumed that most of the investments that occur in the nuclear industry have similar financial risks and therefore the same discount rate is applied for all the cash flows associated with these investments.

The discount rate used is the Weighted Average Cost of Capital (WACC). In the U.S, most nuclear facilities are privately owned and financed through a combination of bonds and equity (only common stocks) with various rates of return. Taxes on net income (which are not applied to bond dividends) are also taken into account. The financial numbers used in CAFCA are shown in Table 6.1:

Table 6.1 - Financial Assumptions for the U.S. Nuclear Enterprise

Percentage of equity (common stock) f_e	50%
Percentage of debt f_b	50%
Required Return on Equity R_e (real value)	12.0%
Cost of Debt R_b (real value)	5.0%
Bond Term	20 years
Tax Rate τ_{tax}	37%
WACC $i = f_e R_e + (1 - \tau_{tax}) f_b R_b$	7.58%
Risk-free rate (secure funds) R_f	2%

Note that the risk-free rate may be used instead of the discount rate in cases for which the money required for a given expense is collected ahead of that expense. This notably applies to the decommissioning costs.

6.1.3 Cash Outflows

Cash outflows are calculated at any time and are the sum of the outlays associated with the amortization of the capital costs, the O&M expenses and the fuel cycle costs.

At a given time t, the cash flows associated with the fuel cycle correspond to the mass flows actually calculated in CAFCA at time t. This mass flow does not necessarily occur

at the "ideal" time[21]. Note that the outlays occurring at a given time are not necessarily associated with the energy produced at that very time.

6.1.4 Cost of Electricity

At any time, the Cost of Electricity COE is defined as the ratio of the reactors owners' outlays, which are also the revenues required to cover the expenses, to the amount of energy produced meanwhile. The COE is actually, at any time step, the ratio of the total cash flows occurring over this time step to the total amount of energy produced meanwhile.

It includes both capital costs (construction costs amortized over the economic lifetime of the plant, decommissioning costs, incremental capital costs) and operating costs (O&M costs, fuel cycle costs). The COE is expressed in mills/kWh (or equivalent $/MWh) and is the actual minimum pricing imposed on the ratepayers at any time (in an unregulated system).

6.1.5 Levelized Cost of Electricity

The Levelized Cost of Electricity LCOE is defined for a given system (e.g. a reactor, or a fleet of reactors) as the constant amount of money that need be charged per unit of energy produced over the lifetime of this system to cover the total costs, including cost of money charges. The LCOE is a metric calculated to make easier comparisons among alternative sources of energy and is not used for actual pricing. It is also expressed in mills/kWh (or $/MWh).

6.2 Capital costs

Capital costs are by far the main component of the total cost of the nuclear electricity. They not only include the construction and decommissioning costs of the plant as well as

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[21] e.g., in the calculation of the fuel cycle costs associated with the production of 1 kWh by a fast reactor, it is assumed that this spent fuel is reprocessed a certain number of years, say N, after the discharge of this spent fuel (see Section 5.4). Actually (in the physical model), this reprocessing may occur M>N years after discharge, due for example to a temporary shortage of reprocessing capacity. The cash outflows calculation will consider M as an input while the levelized cost calculation will consider the "ideal" N as an input.

the incremental capital costs that occur over the lifetime of the reactor, but also the financial charges associated with them, as both construction and decommissioning take years and are financed through a combination of bonds and equity.

6.2.1 Construction Costs

For each reactor, construction capital costs are calculated in an input spreadsheet as follows:

The first step is to calculate the total construction cost, which is the sum of overnight construction costs and financial charges:

- The overnight cost of a reactor C_{ON} is defined as the cost that would be charged for the construction of one reactor if no interest was incurred during construction (or if the reactor was built instantaneously, which is equivalent). The following central values are assumed (Table 6.2, the LWR value is taken from [Du and Parsons, 2009]):

Table 6.2 - Overnight Construction Costs

Reactor type	Overnight Cost
LWR	$C_{ON}^{LWR} = 4000$ \$/kWe
FR	$C_{ON}^{FR} = 4800$ \$/kWe

- CAFCA allows for any construction schedule of 8 years or less. The schedule assumed in this study is shown in Table 6.3, peaking at mid-construction (operation of the reactor starting at the beginning of year 1):

Table 6.3 - Construction schedule

Time	Construction expenses fraction S_{cons} (%)
Year -4	9.5%
Year -3	25%
Year -2	31%
Year -1	25%
Year 0	9.5%

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Using the financial parameters defined in Table 6.1, the total construction cost C_{Cons} at time t=0 is obtained:

$$C_{Cons} = \sum_{n=0}^{4} \frac{C_{ON} \cdot S_{Cons}(year_{-n})}{(1+i)^{-n}}$$ (6.1)

with $\sum_{n=0}^{\infty} S_{Cons}(year_{-n}) = 1$ (6.2)

Even though moderate, the impact of varying the construction schedule on the total construction cost is important, as capital costs are the main contributor to the total cost of nuclear electricity. One can compare the total construction cost obtained with the building schedule assumed in Table 6.3 with a building schedule spread over 7 years, with the same sinusoidal profile; and a construction schedule spread over 5 years, but using a uniform profile. Numbers are shown in Table 6.4. Spreading the building schedule over 7 years increases the total construction cost by 8.1% while using a uniform profile only increases it by 0.2%.

Table 6.4 - Impact of the construction schedule on the total construction cost (assuming a 7.58% discount rate)

Construction total time		5 years (sinusoidal distribution)	7 years (sinusoidal distribution)	5 years (uniform distribution)
Year -6		0	7.6%	0
Year -5		0	14.1%	0
Year -4		9.5%	18.4%	20%
Year -3		25%	19.9%	20%
Year -2		31%	18.4%	20%
Year -1		25%	14.1%	20%
Year 0		9.5%	7.6%	20%
Total construction cost	**LWR**	**4,645**	**5,023 (+8.1%)**	**4,654 (+0.2%)**
($/kWe)	**FR**	**5,573**	**6,027**	**5,584**

This total construction cost is to be paid off over a certain period (bond term) starting the first year of operation.

The revenues required each year to cover these costs over a specified period are calculated, given that the interest on debt is tax deductible, as well as the cost of

construction (depreciation allowances). CAFCA allows the use of any depreciation schedule and any bond term, but it is assumed that the structure of the financing remains the same over time, which means in our case that the principal remaining to be paid is always composed of 50% debt and 50% equity.

Generalizing [Boscher et al., 2005]'s calculations to any depreciation schedule (with L_D the depreciation life):

$$D(year\, y) = C_{ON} \cdot S_D(year\, y) \tag{6.3}$$

$$\text{with } \sum_{y=1}^{L_D} S_D(y) = 1 \tag{6.4}$$

and any bond term N, the yearly payments for the construction costs (per kWe, LWR or FR) are obtained, starting Year 1 (first year of operation of the reactor):

$$YC_{Cons}\,(year\, y) = \begin{cases} \dfrac{1}{1-\tau_{tax}} \left[\dfrac{C_{Cons}\, i(1+i)^N}{(1+i)^N - 1} - D(y) \cdot \tau_{tax} \right] & \text{if } y \le N \\[2ex] \dfrac{-D(year\, y) \cdot \tau_{tax}}{1-\tau_{tax}} & \text{else} \end{cases} \tag{6.5}$$

The standard case assumes N= 20 years and the depreciation schedule shown in Table 6.5 (nuclear qualifies for the 15-year MACRS[22], as indicated by the U.S. income tax code). Table 6.5 also shows the payments required to cover the total construction costs, assuming a discount rate of i = 7.58% and a tax rate τ_{tax} = 37%.

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[22] Modified Accelerated Cost Recovery System, IRS Asset class 49.12 (Electric utility nuclear production plant).

Table 6.5 - Total construction costs payments over the lifetime of reactors (including depreciation deductions)

Year	Depreciation rate S_D (%)	Revenues required (M$/GWe)	
		LWR	FR
1	5.00%	610	732
2	9.50%	504	605
3	8.55%	526	632
4	7.70%	546	656
5	6.93%	564	677
6	6.23%	581	697
7	5.90%	589	706
8	5.90%	589	706
9	5.91%	589	706
10	5.90%	589	706
11	5.91%	589	706
12	5.90%	589	706
13	5.91%	589	706
14	5.90%	589	706
15	5.91%	589	706
16	2.95%	658	790
17	0	727	873
18	0	727	873
19	0	727	873
20	0	727	873
21...60	0	0	0

6.2.2 Decommissioning Costs

Reactors are decommissioned at the end of their lifetime. The lifetime of both thermal and fast reactors is assumed to be L = 60 years. Decommissioning costs are significant but as they occur 60 years from now, their discounted value is very small compared to construction costs.

It is assumed that the money required for paying the total decommissioning cost is collected over the lifetime of the reactor and deposited in a secure fund. For the tax

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calculations, this expense is considered as an operating cost.

A first step is to calculate the total decommissioning costs, which is the sum of the overnight decommissioning costs and the (negative) associated charges. Then this total decommissioning cost is spread over constant annual payments during the lifetime of the reactor, taking account of the time value of money. As the money is deposited in secure funds, the discount rate used in these calculations is the real risk-free rate of 2.00%. The following central values are assumed (Table 6.6):

Table 6.6 - Overnight Decommissioning Costs

Reactor type	Overnight Cost
LWR	$C_{Dec\ ON}^{LWR} = 450\ \$/kWe$
FR	$C_{Dec\ ON}^{FR} = 600\ \$/kWe$

A uniform decommissioning schedule over 5 years (S_{Dec} (year y) = 20% for y=L+1 to L+5) is assumed. The total cost of decommissioning at the year 0 C_{Dec}, is calculated as follows:

$$C_{Dec} = \sum_{y=L+1}^{L+5} \frac{C_{Dec\ ON} \cdot S_{Dec}(year_y)}{(1+R_f)^y} \qquad (6.6)$$

$$\text{with } \sum_{y=1}^{5} S_{Dec}(year_y) = 1 \qquad (6.7)$$

Then the yearly payment required to cover this total cost of decommissioning (per kWe, LWR of FR) is finally obtained (classical ordinary annuity formula, recall that L is the reactor lifetime):

$$YC_{Dec} = C_{Dec} \cdot R_f \cdot (1 - \frac{1}{(1+R_f)^L}) \qquad (6.8)$$

Table 6.7 shows the annual payments required to cover the total decommissioning cost.

Table 6.7 - Annual payments required to cover the total decommissioning costs

Reactor type	Yearly Payments
LWR	$YC_{Dec}^{LWR} = 2.9$ $/kWe/Year
FR	$YC_{Dec}^{FR} = 3.3$ $/kWe/Year

6.2.3 Incremental Capital Expenditures

Incremental capital costs occurring every year over the lifetime of a reactor are considered as operating expenses (in reality, they are added to the depreciable asset base). Table 6.8 shows the annual incremental capital costs for both types of reactor:

Table 6.8 - Annual incremental capital costs

Reactor type	Costs
LWR	$YC_{Inc}^{LWR} = 40$ $/kWe/Year
FR	$YC_{Inc}^{FR} = 48$ $/kWe/Year

6.2.4 Total capital costs

For a given reactor and a given year, the payment $YC_{capital}$ (per kWe) required to cover the total capital cost is the sum of the payments required to cover the total construction cost, the total decommissioning cost, and the incremental capital costs.

$$YC_{capital}(year y) = YC_{Cons}(y) + YC_{Dec}(y) + YC_{Inc}(y) \tag{6.9}$$

For the whole system (fleets of reactors), the total capital cost $\Sigma YC_{capital}$ is obtained by computing the sum of all the capital costs associated with each of the reactors of the system.

$$\Sigma YC_{capital} = YC_{Capital}^{LWR} \cdot P^{LWR} + YC_{Capital}^{FR} \cdot P^{FR} \tag{6.10}$$

where P^{LWR} is the total LWR installed capacity and P^{FR} the total FR installed capacity.

6.3 O&M costs

Beside capital costs, operating costs are the second component of the total cost of nuclear energy. Operating costs aggregate Operation and Maintenance costs (O&M) and fuel cycle costs. The tax rate does not matter for the operating costs, as the revenues are

assumed to occur at the same time as the expenses.

O&M costs have a fixed component (expressed in $/kWe) and a much smaller variable component (expressed in $/kWh). For simplicity, O&M costs are assumed to be all fixed (assuming a capacity factor of 90%).

Table 6.9 shows the assumed O&M costs (fixed costs) for both types of reactors:

Table 6.9: Operation and Maintenance costs

Reactor type	O&M Costs
LWR	$YC_{O\&M}^{LWR} = 70$ $/kWe/Year
FR	$YC_{O\&M}^{FR} = 70$ $/kWe/Year

For the whole system, the total O&M costs $\Sigma YC_{O\&M}$ is obtained by computing the sum of all the O&M costs associated with each of the reactors of the system.

$$\Sigma YC_{O\&M} = \Sigma YC_{O\&M}^{LWR} + \Sigma YC_{O\&M}^{FR} = YC_{O\&M}^{LWR} \cdot P^{LWR} + YC_{O\&M}^{FR} \cdot P^{FR} \qquad (6.11)$$

6.4 Economics of the Fuel Cycle

6.4.1 Method

Fuel cycle costs are the last component of the cost of nuclear energy and, unlike the capital costs and O&M costs, are not related to the reactor. Fuel cycle service prices are expressed in $ per unit of mass produced or treated. In order to calculate the total fuel cycle cost associated with the generation of a certain amount of energy, all the steps occurring in the fabrication of the required fuel as well as the management of the spent fuel are accounted for.

The money required to cover the total fuel cycle associated with the production of a given amount of energy is assumed to be collected at the mid-irradiation point of the corresponding fuel, from the sale of electricity. Therefore, expenses for the front-end steps of the fuel cycle (ore purchase, fuel fabrication etc.) occur before the money is collected while the money needed for back-end expenses (spent fuel storage, disposal costs, etc.) is collected in advance. The fuel cycle costs calculated with this method will

be used to calculate the levelized cost of electricity, and differ from the instantaneous fuel cycle costs occurring at a time t, used for the calculation of the cash flows (and the "cost of electricity").

- **fuel cycle cost formula**

Finally, each cost component "I" is dependent on three types of parameters:

- The **unit cost** C_i of the service

- The **mass flow** M_i corresponding to the production of the amount of energy considered (e.g. the production of 1 MT of UO_2 fuel requires about 10 MT of natural uranium).

- The **lead time** Δ_i, defined as the interval of time between the mid-irradiation point and the actual transaction (negative for a front-end step).

The formula used to calculate the revenue required to cover the fuel cycle costs is as follows (demonstration in [Boscher et al., 2005])

$$\Sigma YC_{FC} = \sum_i M_i C_i \left[\frac{(1 - \tau_{tax}) f_b e^{r_b \Delta_i} + f_e (e^{r_e \Delta_i} - \tau_{tax})}{1 - \tau_{tax}} \right] \tag{6.12}$$

As Δ_i is small, the following approximation can be found in the literature:

$$\Sigma YC_{FC} \approx M_i C_i + M_i C_i \phi \Delta_i \tag{6.13}$$

with the carrying charge factor

$$\phi = \frac{f_e R_e + f_b R_b (1 - \tau_{tax})}{1 - \tau_{tax}} = \frac{x}{1 - \tau_{tax}} \tag{6.14}$$

The exact formula is used in CAFCA.

By selecting the appropriate mass flows (see Section 6.4.4), CAFCA can also segment the total fuel cycle cost into the UO_2 fuel cycle costs, MOX fuel cycle costs, CONFU fuel cycle costs and FR fuel cycle costs. The allocation of the costs partially depends on the approach chosen ("waste-based" or "energy-based", see Section 6.4.2).

- **Assumptions regarding the lead times and the mass flows**

At every time step, the mass flows taken as inputs in the calculation of the total fuel cycle

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cost must correspond to the production of the specific amount of energy considered.

All the mass flows are calculated in CAFCA, but they cannot be used as such due to the following limitations:

- Just as for other variables, the calculations of the costs at time t take as inputs the state of the variables at the previous time step. Therefore, the value of these variables in the future is not accessible at the time the total fuel cycle cost is calculated. In particular, the back-end mass flows calculated by CAFCA at a given time step cannot be associated with the specific amount of energy generated during the time step.

- Batches of fuel or separated materials are not individually tracked in CAFCA. Rather, they are lumped (e.g. "inventory of UO_2 spent fuel"). This implies that the code does not "know" when a given batch of spent fuel is recycled (or, seen from a different perspective, when a currently reprocessed batch was discharged from the reactor), or when a given batch of fresh fuel or TRU is utilized. Therefore, actual lead times are not always known.

Instead, we assume for the fuel cycle cost calculation fixed lead times that do not necessarily reflect the actual timing of the transaction. Thus discrepancies between the outputs of the physical simulation and the inputs to the fuel cycle cost calculation module may appear for the following variables:

- Reprocessing of spent fuel (in particular when reprocessing plants are not introduced yet).
- Fabrication of fresh fuel (which can occur months before its actual utilization)
- Disposal of HLW (Spent fuel in OTC scenarios, TRU losses and Fission products in scenarios involving reprocessing).

For each type of fuel/reactor, *the mass of fuel loaded per year in reactors* (or *"fuel utilization rate"*) is taken as a reference. The other mass flows are all inferred from the

value of this variable using technical assumptions.

6.4.2 Treatment of the reprocessing costs

Different interpretations exist as to the main purpose of reprocessing and thereby the allocation of its costs. The two extreme positions are the following:

- The "waste-based approach": "Reprocessing, just as for the repository, is a way of handling spent fuel. In this view, reprocessing is a back-end fuel cycle step. One considers fuel reprocessing as a waste management policy. In that case, their costs are to be paid by the electricity that produces the wastes. The money is thus collected before the conduct of operations and has time to build interest" [Boscher et al., 2005]

- The "energy-based approach": "Reprocessing, as enrichment, provides raw material input for pin fabrication plants. In this view, reprocessing is a front-end fuel cycle step. As a consequence, the costs of reprocessing and separation operations are assumed to be paid from the revenues generated by the electricity produced from the recycled materials[23] and sold in the future. Money needs to be borrowed for generating the revenue and so the time value of money increases the cost of the recycle operations" [Boscher et al., 2005].

These two accounting schemes are available in CAFCA. Depending on the approach chosen, reprocessing costs will be integrated in the costs associated with the cycle that generates the spent fuel or with the cycle that uses the fissile materials extracted from this spent fuel. Mass flows and lead times are determined accordingly.

The "waste-based" scheme must be applied to countries where reprocessing is compulsory, regardless of the capability of the utility to make MOX (Japan, France, Germany until 1994). However, in the absence of such legal requirement or public

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[23] As the recovery of uranium alone would not motivate the reprocessing of spent fuel, the "recycled materials" only refer to the transuranics here.

incentives, the "waste-based" scheme is unlikely to be realistic. Indeed, as reprocessing is very likely to remain more expensive than repository or interim storage (from the sole viewpoint of the UO_2 spent fuel producer), the "waste-based" scheme would always make this producer worse-off, and therefore not motivated to send spent fuel to reprocessing.

6.4.3 Fuel cycle service prices

A summary of the fuel cycle service prices used in this study can be found in Appendix A.6.1.

6.4.3.1 Natural U price model

This part, and in particular the U-price and U-reserve models, is mostly derived from [Matthews and Driscoll, 2009]. More details about the model can be found in this reference as well as in a forthcoming report on an interdisciplinary study at MIT of "The Future of the Nuclear Fuel Cycle".

Unlike the other services, such as conversion, enrichment, fabrication and reprocessing, the natural uranium supply is by definition resource-limited. The availability of uranium as well as its price are important factors not only in comparing nuclear energy with alternative sources[24] but also in comparing the various fuel cycle options, which is the focus of this study. The implementation of a U price model in CAFCA, as imperfect as it might be, is motivated by two facts:

- The uncertainty that has the most important impact on the total fuel cycle is that of the price of natural uranium.

- Some antinuclear organizations claim that nuclear energy is not a sustainable option because uranium resources are scarce and will run out or be very expensive in the near future. The impact of this assertion needs to be explored, even though it is based on a misperception.

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[24] As natural uranium is currently a small component (~ 4%) of the total cost of nuclear energy (which is dominated by the capital costs), only an impressive rise in its price would significantly change the outcome of any comparative economic study between nuclear energy and its alternatives. The security of supply is a different issue, however not of concern currently given the high level of reliability of the suppliers, among which are several free-market democracies.

All other things being equal, a significant increase in the natural uranium price would make the reprocessing option and the use of breeders economically more interesting, since they would reduce the needs for natural uranium. The possibility of using recovered uranium instead of natural uranium would also work in favor of the recycling option. The uranium price is also a driver for secondary technical choices such as the burn-up (and accordingly the U^{235} enrichment) and the enrichment tails assay.

The uranium spot price has been extremely volatile over the last few years, ramping up from a low of 17 $\$_{2007}$/kgU in 2001 to a peak at 350 $\$_{2007}$/kgU in 2007. As of April 2009, this price is about 110 $\$_{2007}$/kgU. A similar peak last occurred in the late seventies [OECD, 2005].

The uranium market has been unbalanced over the last decade. Currently production from world uranium mines (about 70,000 tons/year) only meets about 62% of the consumption by nuclear plants, because of the existence of alternative sources. The market was also affected by a transient loss in supply capacity due to unexpected incidents (notably the flooding of the large "Cigar Lake" mine in 2006 and 2008). The secondary sources result from the drawdown of various inventories held by both governments and utilities. The main contribution by far comes from the recycling of highly enriched uranium originally fabricated for military purposes. By March 2008, 325 MT of Russian HEU had been used in U.S. nuclear plants after being diluted and converted into LEU (This translates to an equivalent 60,000 tons of natural uranium). However, with the end of the "Megatons to Megawatts" program in 2013 and more generally the shrinking of military stockpiles, as well as the recent increase in exploration and investments, the market should get closer to equilibrium within 10 or 20 years[25].

Thus, the U-price model presented below and provided as assumption in CAFCA forecasts long-term costs of production. It is assumed that the supply continuously matches the demand (adjusting times and delays due to environmental or legal

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[25] However, it has to be noted that a recent agreement allows U.S. companies to buy directly up to 500 tons of HEU from Russia.

requirements are neglected) and that the ore price is only related to the cost of its extraction. Estimates of the price of uranium require on the one hand some forecasts of the total demand and on the other hand an assessment of the world reserves recoverable as a function of the extraction cost.

- Demand forecast

Natural uranium is exclusively used for electricity generation and military purposes, therefore its future demand is relatively easy to predict, compared to other resources with a broader set of uses.

The U.S. (or whichever area studied) demand is a CAFCA output that depends on scenario assumptions. However, assumptions also have to be made as to the natural uranium consumption by the rest of the world. The user can for example assume that this additional consumption will be a multiple of the U.S consumption.

- Reserves Assessment

Since its first volume in 1956, the "Red Book" has been providing up-to-date projections for the reserves accessible at different categories of price (however, the upper limit to these categories is a constant 130$/kgU, in *nominal* dollars). Table 6.10 shows the projections made in 2007 for the world resources at the upper limit cost.

Table 6.10 - Worldwide Uranium Resources at < 130 $_{2006}$/kg (Source: [Red book, 2007])

Country	Identified* Resources <130 $/kg 10^6 MT	+ Undiscovered** Resources <130 $/kg 10^6 MT	= TOTAL Resources <130 $/kg 10^6 MT
Australia	1.243	Not reported	>1.243
Kazakhstan	0.817	0.800	1.617
Canada	0.423	0.850	1.273
Russia	0.546	0.991	1.537
South Africa	0.435	0.110	0.545
USA	0.339	2.131	2.470
All Others	1.666 (37 countries)	2.685 (28 countries)	4.351
TOTAL	5.469	7.567	> 13.036

*Identified = Reasonably Assured Resources (RAR) + Inferred

**Undiscovered = Prognosticated + Speculative

As a rough guide, $12 \cdot 10^6$ MT of uranium represents the consumption of about 1000 LWRs over their lifetime (assuming a consumption of 200 MT/GWe-year and a reactor lifetime of 60 years).

The approach adopted by [Matthews and Driscoll, 2009] to make long-term predictions combines a model for the ore grade elasticity of cumulative resources (Deffeyes' model) with economies of scale and learning curve correlations to find a relation between cost (C) and cumulative U consumption (G).

- The Deffeyes' model, which applies to mineable reserves with an ore grade comprised in the range of interest[26] 10^2-10^4 ppm, allows calculation of the size of the worldwide recoverable reserves for a given ore grade. Defining cumulative reserves for a given ore grade x (ppm U) as the worldwide recoverable reserves for which the ore grade is equal to or above x:

$$s = \frac{\% \text{ increase in cumulative reserves}}{\% \text{ decrease in ore grade}} \approx \frac{\sqrt{\pi}}{2\sigma^2} \left[\ln x - v + \frac{\sigma}{\sqrt{2}} \right] \qquad (6.15)$$

where v is the mean value of $\ln x = 2.48$,

and σ the standard deviation of $\ln x = 1.51$.

S is positive, which means that the supply is predicted to increase when the ore grade decreases.

With U = cumulative reserves, one also has:

$$\frac{U}{U_r} = \left(\frac{x}{x_r} \right)^{-s} \qquad (6.16)$$

where r refers to the reference case (initial time of the simulation)

- Economies of scale are obtained when the total mass processed increases, which happens when the ore grade decreases:

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[26] Extracting ore with lower grades would currently require an amount of energy comparable to that recoverable by irradiation of fuel in LWRs.

$$\frac{C}{C_r} = \left(\frac{x_r}{x}\right)^n \qquad (6.17)$$

where C = cost for extracting 1kgU and n = scale exponent (typically 0.7)

Combining the two equations, a relation between cost and cumulative consumption is obtained (assuming that higher-grade reserves are exploited first):

$$\frac{C}{C_r} = \left(\frac{U}{U_r}\right)^{\frac{n}{s}} \qquad (6.18)$$

- Finally, a correction factor is integrated to account for learning effects:

$$\frac{C}{C_r} = \left(\frac{U}{U_r}\right)^{-\alpha} \qquad (6.19)$$

$$\text{where } \alpha = -\left(\frac{\ln(f/100)}{\ln 2}\right) \qquad (6.20)$$

in which f is the progress rate (typically f=85%, hence α =0.23).

Finally :

$$\frac{C}{C_r} = \left(\frac{U}{U_r}\right)^{\theta} \text{ where } \theta = \frac{n}{s} - \alpha \qquad (6.21)$$

Parameters n, s, and f are each assumed to be uniformly distributed over the following ranges:

$$\begin{array}{l} 0.5 \leq n \leq 1.0 \\ 1.5 \leq s \leq 3 \\ 70\% \leq f \leq 100\% \end{array} \qquad (6.22)$$

Using the Monte-Carlo method, a frequency distribution for θ is obtained. 15% of the outcomes fall below θ=-0.10, 50% below θ=0.11, 85% below θ=0.29. It is therefore suggested to the CAFCA user to choose the values of θ shown in Table 6.11, depending on the level of optimism wanted:

Table 6.11 - Recommended values for θ in U-price model

θ	Natural U cost forecasts
-0.10	Optimistic
0.11	Median case
0.29	Conservative

Table 6.12 shows the reference values (initial conditions):

Table 6.12 - Initial values in U-price model

	Initial value (2005)
C_r	100 \$/kgU
U_r	2,000,000 MT

- This U price model also provides an assessment of the remaining worldwide reserves recoverable at a given cost C (for positive values of theta), calculated by the following formula (G is the cumulative consumption since the beginning of the simulation, G(0)=0):

$$U_{remaining}(t) = U_r \left(\frac{C}{C_r} \right)^{\frac{1}{\theta}} - G(t) \tag{6.23}$$

For example: for C = 130 \$/kgU, using the median value θ=0.11, the remaining recoverable reserves obtained is $U_{remaining}$ =19.7 10^6 MT in 2009. Using the pessimistic θ=0.29, one finds $U_{remaining}$ = 2.9 10^6 MT. These values are to be compared to 13.04 10^6 MT deemed to be the total reserves in the Red Book, corresponding to θ=0.130.

- Finally, when the U-price model is not activated, the default value for the price of natural uranium is 100 \$/kgU.

6.4.3.2 Depleted U utilization

"Depleted uranium" is the name given to the byproduct of enrichment. The production of 1 MT of LEU generates about 9 MT of tailings, with a U_{235} content of about 0.25%w. Depleted uranium demand for non-nuclear purposes does exist[27] but remains far below

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[27] Depleted uranium can be useful in industry particularly because of its very high density. Uses include counterweights in aircraft and boats, radiography equipment, containers for radioactive materials transportation and armor-piercing projectiles.

the supply. This demand is neglected in CAFCA.

The cost of storage of the depleted uranium is assumed to be included in the enrichment cost.

As an inexpensive fertile material, depleted uranium may be used for fabricating fuel for fast reactors and MOX fuel for thermal reactors (as an alternative to natural uranium and recovered uranium). The depleted uranium has to be converted from uranium hexafluoride UF_6 for metal fuel fabrication, for a cost of $6/kgU in 2003 [Bunn, 2003]. A price of $10/kgU (2008 dollars) is assumed in CAFCA.

6.4.3.3 Conversion, Enrichment, Fuel Fabrication costs

Table 6.13 shows the front-end fuel cycle services costs for the UO_2 fuel (assuming natural uranium is used), MOX and FR fuels.

These costs include all the transportation costs (including shipping to the reactor) and the LLW disposal costs, but not the disposal of the tails (only their temporary storage).

Table 6.13 - Conversion, enrichment and fuel fabrication costs

Front-end Service		Price
Conversion (nat. U)		10 $/kgHM
Enrichment (nat. U)		160 $/kgSWU
UO_2 fuel fabrication (nat. U)		250 $/kgHM
MOX fuel fabrication		2,000 $/kgHM
FR U-TRU fuel fabrication	CR=1.23	2,000 $/kgHM
	CR = 1.0	2,000 $/kgHM
	CR=0.75	2,125 $/kgHM
	CR=0.5	2,250 $/kgHM
	CR=0.0	2,500 $/kgHM
FR CR=1.2		2,000 $/kgHM

6.4.3.4 Reprocessing costs

The reactors and the reprocessing plants are the only facilities that are modeled in CAFCA. This choice results from the fact that, unlike most of the other services, the industry of reprocessing is subject to significant economies of scale[28], which leads to an

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[28] According to [Haire, 2003], the capital costs of reprocessing plants are proportional to the nth power of capacity, with n=0.1 for small plants (less than 300 MT/year) and n=0.9 for capacities ranging from 2000 MT/year to 7000 MT/year. The reason is that the minimum equipment is too large for very small capacities

average plant capacity factor relatively important with respect to needs. For example, the THORP plant in Sellafield, U.K., has the theoretical capacity (600 tHM/year) to process the output of about 40 GWe of LWRs (assuming a burn-up of 50 GWd/tHM). The size of the plants, combined with the economic requirement to avoid overcapacity ("minimum loading factor"), introduces some rigidity (delays) into the fuel cycle. Finally, modeling individual reprocessing plants (and giving the user the choice of their capacity) allows this trade-off between economies of scale and agility of the supply to be studied.

In order to assess the reprocessing costs, the same method as for the reactors is applied, using spreadsheets. The construction schedule as well as the decommissioning schedule are assumed to be the same as for the reactors. However, a shorter lifetime for the reprocessing plants is assumed: L_{RP} = 40 years. This assumption is for cost recovery calculations purposes only: the actual lifetime of the plant can be extended by the user to 50 years or more. Furthermore, a government financing is assumed (100% bonds, interest rate of 4%, no tax, bond term = 30 years). If the plant were to be financed by the private sector, the same financial scheme and the same depreciation schedule for tax calculations as for the reactors would be used. In reality, the THORP and La Hague plants were financed by the utilities through pay ahead contracts, with no interest or returns to investors required. But, "Financing with pay-ahead contracts and without requirements for return on investment was possible only because the reprocessors' customers were legally obliged by their governments to enter into reprocessing contracts and BNFL and Cogema were the only firms offering the service; this seller's market for reprocessing services will not occur again." [Bunn et al., 2003].

Beside these assumptions common to all the reprocessing plants, the final levelized cost primarily depends on the construction costs and the operating costs.

• Construction Costs

[Aquien et al., 2006] semi-empirically assessed the capital costs of the reprocessing

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(100 MT/year) but can only be duplicated for high capacities, as each equipment module reaches its maximum capacity. [Haire, 2003] recommends 2,500 MT/year as an optimal capacity.

plants as a function of their capacity, for various types of fuel[29]. This approach is maintained, with small modifications, as described below.

The capital costs of UP2 and UP3 at La Hague (each 800MT/year), THORP (600 MT/year) and Rokkasho-Mura (800 MT/year) are taken as a reference. The [Shropshire et al., 2008] report provides overnight costs in 2005 dollars, that were adjusted to 2008 dollars using the US GDP deflator index[30] (factor of 1.0517), to obtain for the capital costs UP2 = \$8.5bn, UP3 = \$7.2bn, THORP = \$5.9bn, Rokkasho-Mura = ~ \$21bn. As Rokkasho-Mura clearly stands out, we consider it a special case that should lie beyond this database. Taking the average value, we finally assume a construction cost of \$7.5bn for an 800 MT/year reprocessing plant.

Then the assumptions made by [Haire, 2003] about the effect of the economies of scale are used to evaluate the capital costs for plants of various capacities. We assume here the same factors that [Aquien et al., 2006] derived from [Haire, 2003].

Costs are also affected by the type of fuel reprocessed. The higher the fissile content in spent fuel, the higher the costs because of the need for more features to prevent criticality and the lower mass content of one single assembly. [Haire, 2003] makes assumptions as to the impact of the type of fuel processed on the capital costs (capital cost ratio: spent MOX fuel = 1.1, spent Breeder fuel= 1.5, spent UO_2 fuel being the reference). Additional assumptions for other conversion ratios need to be made. "There are virtually no capital cost differences in reprocessing LWR and other fuels at throughput lower than 300 MT/year" [Haire, 2003]. As FFF reprocessing plants are of small size (100 MT/year is the capacity needed to reprocess the output of about 16 GWe of ABRs), their capital costs are assumed to be the same as those of the spent UO_2 fuel reprocessing plants. However, the reprocessing plants for the FR CR=0.5 and CR=0.75 spent fuel may be larger than 300 MT/year. In these cases we assume a factor of 2 for both of them. Finally, the reprocessing costs for the spent fuel discharged from the breeders (including blankets) are assumed to be the same as for the self-sustaining reactors. These assumptions lead to the capital costs shown in Table 6.14.

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[30] It would be however more appropriate to use an escalation factor specific to the industry.

Table 6.14 - Capital Costs for Reprocessing Plants

Capacity (tIHM/year)	Spent UO2 fuel, in $bn (equivalent in $/kg/year)	Spent MOX fuel, in $bn (equivalent in $/kg/year)	Breakeven FR spent fuel (GFR, FR CR=1.0), Breeder (FR CR=1.23), in $bn (equivalent in $/kg/year)	FFF spent fuel (CONFU, ABR, FR CR=0.0), in $bn (equivalent in $/kg/year)	FR CR=0.5 spent fuel, in $bn (equivalent in $/kg/year)	FR CR=0.75 spent fuel, in $bn (equivalent in $/kg/year)
50	-	-	-	5.4 (108,000)	-	-
100	5.4 (54,000)	5.4 (54,000)	5.4 (54,000)	5.4 (54,000)	5.4 (54,000)	5.4 (54,000)
200	5.4 (27,000)	5.4 (27,000)	5.4 (27,000)	5.4 (27,000)	5.4 (27,000)	5.4 (27,000)
500	6.2 (12,400)	6,8 (13,600)	10.2 (20,400)	-	12.4 (24,800)	12.4 (24,800)
1,000	7.8 (7,800)	8.6 (8,600)	11.7 (11,700)	-	15.6 (15,600)	15.6 (15,600)
2,000	8,9 (4,500)	9.8 (4,900)	13.3 (6,700)	-	-	-
7,000	18.9 (2,700)	20.8 (3,000)	28.3 (4,000)	-	-	-

These numbers (for UO_2 reprocessing) are significantly higher than the [Shropshire et al., 2008] results ($3.8bn, $4.8bn and $6.7bn for the 500, 1,000, and 2,000 MT/year UO2 spent fuel reprocessing plants, respectively), which are based not only on the costs for the reprocessing plants cited above, but also on those of the U.S. plants built in the seventies (Barnwell, Dupont, GE Morris etc.), which may have not been designed and built to today's standards. Furthermore, the large swing in cost of construction materials, such as steel and concrete in the last three years leave us with a large uncertainty as to what the eventual costs turn out to be when constructions occurs.

- **Operating Costs**

Operating costs include the packaging and temporary storage of the products (recovered

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materials, wastes), property taxes (if any), insurance and decommissioning. However, wastes and byproducts are eventually returned to the customer.

[Haire, 2003] assumes that the operating costs for spent UO_2 fuel reprocessing are about 6% of the total capital cost (considering a 1,500tHM/year plant). We assume for the 1000-tHM/year UO_2 spent fuel reprocessing plant a construction cost of 7,800 \$/kgHM/year. Applying Haire's factor (6%) to this number, we obtain an operating cost of 468 \$/kg.

This number is significantly lower than the forecast that BNFL made prior to the operation of THORP (735 \$/kgHM) [Bunn et al., 2003]. This latter number was used as a reference by [Bunn et al., 2003], but deemed to be optimistic. [Bunn et al., 2003] adds \$100/kgHM for the allowances for refurbishment and decommissioning, and \$70/kgHM to account for the start-up costs.

Finally, we assume for the reprocessing of UO_2 spent fuel an operating cost of 900 \$/kgHM, independent from the plant output. This value would be consistent with the price currently practiced by La Hague (900 \$/kgHM, given in [INL 2008]), which only reflects the operating costs (the plant has already been amortized) and a low profit margin (because of the competition between La Hague and THORP).

For the same reasons as given in the capital costs part, we assume factors of 1.1, 1.5 and 2 for the reprocessing of MOX spent fuel, FR CR=1.0 spent fuel and FR CR=0.5/0.75, respectively. In order to fit the [NEA, 2002]'s costs assessment for the reprocessing of HTGR-TRU spent fuel (deemed to be similar to FFF spent fuel reprocessing), [Aquien et al., 2006] uses a factor of 5 for the operating costs associated with the reprocessing of FFF spent fuel. We take the same number for the reprocessing of FR CR=0.0 spent fuel.

• **Reprocessing prices**

The total cost of reprocessing is the sum of the levelized capital costs and the operating costs. Table 6.15 shows the final reprocessing price if the profit margin is assumed to be zero as a function of the unit capacity and fuel type.

Table 6.15 - Spent fuel reprocessing prices for zero profit margin

Capacity (tIHM/year)	UO2 spent fuel ($/kgIHM)	MOX spent fuel ($/kgIHM)	Breakeven FR spent fuel (GFR, FR CR=1.0), Breeder (FR CR=1.23) ($/kgIHM)	FFF spent fuel (CONFU, ABR, FR CR=0.0) ($/kgIHM)	FR CR=0.5 spent fuel ($/kgIHM)	FR CR=0.75 spent fuel ($/kgIHM)
50	-	-	-	10,408	-	-
100	3,854	3,944	4,314	7,454	4,754	4,754
200	-	-	-	5,977	-	-
500	1,578	1,734	2,466	-	3,157	3,157
1,000	1,327	1,460	1,990	-	2,653	2,653
2,000	1,146	1,258	1,716	-	-	-
7,000	1,048	1,154	1,569	-	-	-

These prices, based on the available data on the existing reprocessing plants, include the costs of the temporary storage of the recovered uranium, fission products and separated TRU, of transportation, and of disposal of some low-level waste.

Several factors could offset the learning effect and eventually lead to an increase in the prices in the future:

- The regulatory framework could become more stringent, especially in the U.S, leading to additional costs (e.g. reduction of the effluent allowances).

- In CAFCA, reprocessing plants separate the uranium, the plutonium and finally the minor actinides from the fission products, whereas existing plants (notably La Hague, THORP, and Rokkasho-Mura), from which costs were taken as reference, only separate plutonium (PUREX).

- The existing facilities cited above benefited from advantageous financial schemes as their construction was directly financed by utilities, which are under government requirement to reprocess their spent fuel. This might not be the case if reprocessing

were to become optional for the utilities. Funds might then have to be raised on private capital markets; and taxes and insurances paid (this possibility is not very probable though).

6.4.3.5 Disposal costs

It is assumed in this study that repository is to be part of a disposal service, whatever the scenario. In order to reflect the various types of wastes that will occur (as opposed to the "traditional" spent UO_2 fuel generated so far), an economic model that accounts for potential future impact on repository needs of various fuel cycles has been developed.

We call "repository" any permanent storage (typically a geological repository such as Yucca Mountain)[31], as opposed to interim storage.

Utilities are currently not accountable or responsible for the long-term management of the waste they generate (spent fuel is considered as waste in the Once-Through Cycle scheme). Instead, they are required by the government (Nuclear Waste Policy Act of 1982, NWPA) to pay into a Nuclear Waste Fund (NWF) through a constant fee of 1 mill/kWh generated (but in nominal dollars, which means that the real value of the payment has been decreasing over time). The NWF has been collecting interest and has only spent a small fraction on Yucca Mountain site development and testing, as allocated by congress. Therefore, the NWF has grown over time, and is supposed to fully cover the costs of one (or two) repository. The NWPA requires the DOE to reassess the amount of the fee every year and, if necessary, to adjust it (which has never happened so far).

The original version of the Act (1982) foresaw a first repository in the mid-1990s, but the realization of this first repository has been repeatedly delayed, due mainly to legal and political reasons. The last "Civilian Radioactive Waste Management Fee Adequacy Assessment Report" released in July 2008 [DOE, 2008] assumes that the first repository will be operational in 2017. However, as the current administration recently expressed the intent to phase out the Yucca Mountain project, this prediction seems very unlikely.

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[31] Excerpt from the Nuclear Waste Policy Act of 1982: *"The term "repository" means any system licensed by the Commission that is intended to be used for, or may be used for, the permanent deep geologic disposal of high-level radioactive waste and spent nuclear fuel, whether or not such system is designed to permit the recovery, for a limited period during initial operation, of any materials placed in such system. Such term includes both surface and subsurface areas at which high-level radioactive waste and spent nuclear fuel handling activities are conducted."*

The 1-mill/kWh fee is deemed by the DOE to be appropriate for current practices (UO_2 fuel irradiated to about 50 MWD/kgHM) [DOE, 2008] but would be probably redefined if those practices were to change. Therefore, using a unique waste fee expressed per kWh whatever the scenario would not be appropriate. A method available in CAFCA (and used in this study as soon as reprocessing is introduced in the scenario) is to infer from the value of the current fee a price for the repository expressed in $/kgHM of spent UO_2 fuel (burnup of 50 MWd/kgHM). The service price for other types of fuel is then obtained using densification factors (see Section 5.5).

Once reprocessing is introduced in the scenario, it is seen as a back-end service occurring when the spent fuel/HLW has been cooled down in wet storage and is therefore to be sent away from reactors. At a discount rate of 7.58 %, assuming for the spent UO_2 fuel (50 MWd/kgHM burn-up) a residency time of 4.5 years and a cooling time of 5 years (hence a total of 7.25 years between the irradiation mid-point and the time at which the spent fuel is removed from cooling storages), the 1 mill/kWh translates into a unit cost of disposal of 687 $/kgIHM, paid 5 years after discharge from reactors.

In the scenarios involving reprocessing and recycling of fuel, only fission products and various unrecovered actinides (losses) are sent to disposal over the period of the simulation, with the exception of the TTC scenario, in which both separated fission products and minor actinides as well as MOX spent fuel are disposed of. The number calculated for the disposal of the UO_2 spent fuel cannot be applied to the disposal of fission products, minor actinides or MOX spent fuel, because the volume, heat generation (both initial heat load and integrated long-term heat output) and radiotoxicity rather than the mass drive the total disposal costs. These differences are accounted for through the use of the densification factor, detailed in Section 5.5.

Table 6.16 shows the final costs that we obtain using the assumptions of the densification factors as well as the composition of the spent fuels. These costs would not be so low if interest were not accrued over the years following the collection of the fee. Therefore, they must not be considered as the price for the service.

Table 6.16 - Disposal costs for spent fuel and HLW 5 years after fuel discharge (Thus, less is paid over irradiation)

Waste type	Densification factor	Cost in $/kgIHM	Spent fuel composition (UO_2, 50 MWD/kgHM)	Cost in $/kgHM
spent UO_2 fuel	1	687	100 %$_w$	687 $/kgIHM
Minor Actinides + Fission Products	4	172	5.29 %$_w$	3,251 $/kgFPMA
Fission Products	5	137	5.16 %$_w$	2,663 $/kgFP
MOX spent fuel (1-tier scenario)	0.15	4,580	-	4,580 $/kgIHM

6.4.3.6 Storage costs

According to the NPWA:

> *"The persons owning and operating civilian nuclear power reactors have the primary responsibility for providing interim storage of spent nuclear fuel from such reactors, by maximizing, to the extent practical, the effective use of existing storage facilities at the site of each civilian nuclear power reactor, and by adding new onsite storage capacity in a timely manner where practical".*

After being discharged from reactors, the spent fuel is directly sent to wet cooling storage. Reactor pools are used for this temporary storage so we assume that the related costs are already integrated into the O&M costs of the reactors (monitoring, security).

We assume in CAFCA that the spent fuel is then transferred to interim storage. In practice, the spent fuel has generally been left in the reactor pools for decades. However, these pools are increasingly filled up, making the construction of new pools, or the transfer to unsaturated pools, necessary. Dry interim storage is also a solution that is more and more favored, for safety and security perspectives.

One could argue that it is redundant to add the interim storage costs to the repository costs. Indeed, it is written in the NPWA that:

> *"in return for the payment of the fees established by this section [Nuclear Waste Fund], the Secretary, beginning not later than January 31, 1998, will dispose of the high-level radioactive waste or spent nuclear fuel involved as provided in this*

subtitle. ".

Therefore utilities are not supposed to bear the interim storage costs (which have become necessary since the government failed to open a repository) in addition to the NWF fees. It is true that they may get compensation from the government through litigation. But the intent of CAFCA is to provide real costs independent of an arbitrary legal framework that is very likely to change within a few years. As we made the choice in CAFCA to treat disposal as an option for which the price reflects actual costs; we should also include interim storage, which is likely to become an essential part of the spent fuel management (it is relatively cheap and provide much flexibility). Therefore, we always include interim storage costs in CAFCA whatever the fuel cycle scenario is. In an OTC scenario, this conservative assumption is likely to be verified over the next decade, as the Yucca Mountain project is liable to be phased out soon. In the scenarios in which the spent fuel is to be reprocessed, the interim storage provides some flexibility when the reprocessing capacity supply does not match the demand.

We assume a unique price of 200 \$/kgHM for interim storage, regardless of the number of years the spent fuel stays there, paid at the time the fuel is discharge from cooling storage (e.g. 5 years after discharge from reactors for the spent UO_2 fuel). In the case of the UO_2 fuel, using the assumptions made in Section 2.1, this payment is equivalent to a cost of 0.29 mills per kWh produced.

6.4.3.7 Recovered uranium use and unit costs

Unit costs for conversion, enrichment and UO_2 fuel fabrication are greater when recovered uranium is used in the fuel in lieu of natural uranium. Table 6.17 shows the assumed premium for these services, compared to natural uranium use (taken from [de Roo and Parsons, 2009]:

Table 6.17 - Premium on front-end services for recovered U use

	Premium	Service cost
Conversion	200%	20 \$/kg
Enrichment	10%	176 \$/SWU
Fuel fabrication	7%	268 \$/kg

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6.4.4 Mass flows for fuel cycle costs calculation

For each type of fuel, we take as a reference the *mass of fuel loaded in reactors per year* (or *"fuel utilization rate"*). This is an approximation, as in reality fuel batches are not loaded at every time step (e.g. only once every 18 months for the LWRs).

Then the other mass flows are derived from this reference using the following technical assumptions: U235, Pu or TRU enrichments, tails assay, and various losses. For some of these mass flows (ore purchase, conversion, enrichment), calculations are already made in the physical model at time t so we directly use the results. Note that $C_{\%_w \, X/Y}$ means the weight fraction of X in Y.

- **UO$_2$ fuel in LWRs cycle (all-UO$_2$ cores, UO$_2$/MOX cores, CONFU fuel cores):**

$M_{Ore\ purchase}$ = as calculated at time t *("Natural uranium utilization rate")*   (6.24)

$M_{Conversion\ NU}$ = as calculated at time t *("total mass rate of natural uranium feed for traditional fuel per year")*   (6.25)

$M_{enrichment\ NU}$ = as calculated at time t ("SWU consumption for Nat U per year")   (6.26)

$M_{fabrication\ NU}$ = as calculated at time t *("total UO$_2$ fuel (Nat U) utilization rate")*   (6.27)

$M_{interim\ storage} = M_{fabrication}$ (6.28)

- Specific to non-OTC scenarios (waste-based approach):

$M_{reprocessing} = M_{fabrication}$ (6.29)

$M_{FP\ disposal} = C_{\%_w \, FP/UO_2\ SF} \, M_{fabrication}$ (6.30)

- Specific to UO$_2$/MOX scenarios:

$M_{MA\ disposal}$ (in the OTC scenario) $= M_{reprocessing} \, C_{\%w\ MA/UO2\ SF}$ (6.31)

- Specific to OTC scenarios:

$M_{disposal}$ (in the OTC scenario) $= M_{fabrication}$ (6.32)

- Specific to scenarios with recovered uranium utilization:

$M_{Conversion\ RU}$ = as calculated at time t *("mass rate of recovered uranium feed for traditional fuel per*

$year$") (6.33)

$M_{enrichment\ RU}$ = as calculated at time t ("*SWU consumption for Rec U per year*") (6.34)

$M_{fabrication\ RU}$ = as calculated at time t *("total UO_2 fuel (Rec U) utilization rate")* (6.35)

- ## • CONFU fuel in LWRs' cycle (different types of CONFU fuel are treated separately):

$M_{fabrication}$ = as calculated at time t *("Young/Old CONFU fuel utilization rate")* (6.36)

$M_{interim\ storage} = M_{fabrication}$ (6.37)

- Specific to the energy-based approach:
  - o Specific to Young CONFU:

$$M_{UO2\ SF\ reprocessing} = \frac{M_{fabrication}}{(1 - L_{\%_w\ UO_2\ SF\ reprocess.})(1 - L_{\%_w\ CONFU\ fab.})C_{\%_w\ TRU/UO_2\ SF}}$$ (6.38)

$$M_{FP\ (from\ UO2\ SF)\ disposal} = M_{UO2\ SF\ reprocessing}\ C_{\%w\ FP/UO2\ SF}$$ (6.39)

  - o Specific to Old CONFU:

Making the approximation that all the FFF SF is "young":

$$M_{FFF\ SF\ reprocessing} = \frac{M_{fabrication}}{(1 - L_{\%_w\ FFF\ SF\ reprocess.})(1 - L_{\%_w\ CONFU\ fab.})C_{\%_w\ TRU/FFF\ SF}}$$ (6.40)

$$M_{FP\ (from\ FFF\ SF)\ disposal} = M_{fabrication}C_{\%w\ FP/FFF\ SF}$$ (6.41)

- Specific to the waste-based approach:

$M_{FFF\ SF\ reprocessing} = M_{fabrication}$ (6.42)

$M_{FP\ (from\ FFF\ SF)\ disposal} = M_{fabrication}C_{\%w\ FP/FFF\ SF}$ (6.43)

- ## • MOX fuel in LWRs cycle (different types of MOX fuel are treated separately):

$$M_{depleted\ U\ (or\ Natural\ U)\ purchase} = \frac{(1 - C_{\%_w\ Pu/MOX})M_{fabrication}}{(1 - L_{\%_w\ UO_2\ fab.})}$$ (6.44)

$M_{fabrication}$ = as calculated at time t *("MOX fuel utilization rate")* (6.45)

$M_{interim\ storage} = M_{fabrication}$ (6.46)

- Specific to the Energy-based approach:

$$M_{UO2\,SF\,reprocessing} = \frac{M_{fabrication}C_{\%_w\,Pu/MOX}}{(1 - L_{\%_w\,UO_2\,reprocess.})(1 - L_{\%_w\,MOX\,fab.})C_{\%_w\,Pu/UO_2\,SF}} \qquad (6.47)$$

$$M_{FP\,(from\,UO2\,SF)\,disposal} = M_{UO2\,SF\,reprocessing}\,C_{\%w\,FP/UO2\,SF} \qquad (6.48)$$

- Specific to UO$_2$/MOX/FR scenarios (waste-based approach):

$$M_{MOX\,SF\,reprocessing} = M_{fabrication} \qquad (6.49)$$

$$M_{FP\,(from\,MOX\,SF)\,disposal} = M_{fabrication}C_{\%_w\,FP/UO_2\,SF} \qquad (6.50)$$

- Specific to UO$_2$/MOX scenarios (no recycling in FR and hence no use of the separated minor actinides)(energy-based approach):

$$M_{MA\,disposal}\,(in\,the\,OTC\,scenario) = M_{UO2\,SF\,reprocessing}\,C_{\%w\,MA/UO2\,SF} \qquad (6.51)$$

- **FR fuel cycle (different types of FR fuel are treated separately):**

$$M_{fabrication} = as\,calculated\,at\,time\,t\,("FR\,fuel\,utilization\,rate") \qquad (6.52)$$

$$M_{interim\,storage} = M_{fabrication} \qquad (6.53)$$

- Specific to the Energy-based approach:

In CAFCA, the FR fuel made with TRU separated from spent UO$_2$ fuel, the FR fuel made with TRU separated from MOX spent fuel and the FR fuel made with U-TRU separated from FR spent fuel are lumped together. Therefore, the code does not "know" the origin of the TRU used to fabricate the mass of FR fuel considered. Assumptions have hence to be made about these origins for the calculation of the fuel cycle cost.

We assume for the origins of this TRU the respective proportions of the three TRU streams (separated from spent UO$_2$ fuel, separated from MOX spent fuel, separated from FR spent fuel) n years beforehand, with n the number of years required for the fabrication of the FR fuel.

With $P_{\%w\,LWR\,RP/RP}^{TRU}$ the proportion of the TRU that comes from thermal reprocessing plants

(n years beforehand)[32] and $P^{TRU}_{\%_w\, UO_2\, SF/UO_2\, +MOX\, SF}$ the proportion of the TRU separated from spent UO_2 fuel (as opposed to the TRU separated from UO2 spent fuel):

$$M_{UO2\, SF\, reprocessing} = \frac{M_{fabrication} C_{\%_w\, TRU\,/\,FR\, fuel}\, P^{TRU}_{\%_w\, LWR\, RP/RP}\, P^{TRU}_{\%_w\, UO_2\, SF/UO_2\, +MOX\, SF}}{(1 - L_{\%_w\, UO_2\, reprocess.})(1 - L_{\%_w\, FR\, fuel\, fab.}) C_{\%_w\, TRU/UO_2\, SF}} \qquad (6.54)$$

$$M_{FP\,(from\, UO2\, SF)\, disposal} = M_{UO2\, SF\, reprocessing}\, C_{\%w\, FP/UO2\, SF}$$

- Specific to UO₂/MOX/FR scenarios (waste-based approach):

$$M_{MOX\, SF\, reprocessing} = \frac{M_{fabrication} C_{\%_w\, TRU/FR\, fuel}\, P^{TRU}_{\%_w\, LWR\, RP/RP}\, (1 - P^{TRU}_{\%_w\, UO_2\, SF/UO_2\, +MOX\, SF})}{(1 - L_{\%_w\, MOX\, reprocess.})(1 - L_{\%_w\, FR\, fuel\, fab.}) C_{\%_w\, TRU/MOX\, SF}} \qquad (6.55)$$

$$M_{FP\,(from\, MOX\, SF)\, disposal} = M_{MOX\, SF\, reprocessing}\, C_{\%_w\, FP/MOX\, SF} \qquad (6.56)$$

- Specific to UO₂/TTC scenarios (no recycling in FR and hence no use of the separated minor actinides) (energy-based approach):

$$M_{MA\, disposal}\ (in\ the\ OTC\ scenario) = M_{UO2\, SF\, reprocessing}\, C_{\%w\, MA/UO2\, SF} \qquad (6.57)$$

- In every cases, depleted U is needed:
  o to be blended with the TRU separated from the UO₂/MOX spent fuel.
  o to be blended with the U-TRU separated from the FR spent fuel.

Hence:

$$M_{depleted\, U\, purchase} =$$

$$\frac{\left[P^{TRU}_{\%_w\, LWR\, RP/RP}(1 - C_{\%_w\, TRU/FR\, fuel}) + (1 - P^{TRU}_{\%_w\, LWR\, RP/RP}) C_{\%_w\, added\, U/FR\, fuel} \right] M_{fabrication}}{(1 - L_{\%_w\, FR\, fuel\, fab.})} \qquad (6.58)$$

6.4.5 Lead times

Tables 6.16 through 6.23 show the timeline and associated lead times for the thermal UO₂ fuel cycle, the thermal MOX fuel cycle, the FR CR=0.0 fuel cycle, the FR CR=0.5 fuel cycle, the FR CR=0.75 fuel cycle, the FR CR=1.0 fuel cycle, and the FR CR=1.2 fuel cycle, respectively. The lead times for the CONFU fuel cycle, the ABR and the GFR can be found in Appendix A.6.2.

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[32] For practical reasons, we actually used the stream of U-TRU in the code. Proportions are the same.

Cells are black for one-pass schemes (no recycling of MOX spent fuel or UO$_2$ spent fuel). If the spent fuel is reprocessed: cells in dark grey are only accounted for in the "waste-based approach" (except in the Once-Through Cycle strategy) while cells in light grey are only considered in the "energy-based approach".

As one can see in these tables, lead time calculations simply result from the assumptions made for the transit time in the various facilities. Moreover, we assume that:

- Interim costs are accounted for even in scenarios where the spent fuel is reprocessed (conservative assumption). Indeed spent fuel reprocessing may actually occur years after the fuel has been cooled down beyond the minimum cooling time in wet storage, due to reprocessing capacity limitations. In such cases, interim storage may be required if wet storage (reactor pools) is saturated.
- For accounting purposes, disposal and reprocessing are assumed to occur right after the minimum cooling time. This conservative assumption may lead to an overestimate of the real costs (especially over the first decades of the simulation, when both reprocessing and disposal are not available yet).

Table 6.16 – Thermal UO$_2$ fuel cycle lead times (in years)

Fuel Cycle Stage and Duration	Ore	Conv.	Enr.	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
(year)	...	0.5	0.5	0.5	0.5	2.25	2.25	5	...	1	...
Ore Purchase				4.25							
Conversion					3.75						
Enrichment						3.25					
Fabrication							2.75				
Interim St.									-7.25		
UO2 spent fuel Reprocessing									-7.25		
Disposal (OTC)									-7.25		
HLW Disposal (Rep)								-8.25			

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Table 6.17 – Thermal MOX fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO₂ rep.	HLW Disp.	DU	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	2.25	2.25	5	...	1	...
UO₂ spent fuel reprocessing	4.25										
HLW disposal		3.25									
Depleted U purchase			3.25								
Fabrication				2.75							
Interim St.								-7.25			
MOX spent fuel reprocessing								-7.25			
Disposal								-7.25			
HLW Disposal (Rep)								-8.25			

Table 6.18 – FR CR=0.0 fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO₂/FR rep.	HLW Disp.	DU	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	1.77	1.77	5	...	1	...
UO₂/FR spent fuel reprocessing	3.77										
HLW disposal		2.77									
Depleted U purchase			2.77								
Fabrication				2.27							
Interim St.								-6.77			
FR spent fuel reprocessing								-6.77			
HLW Disposal (Rep)								-7.77			

Table 6.19 – FR CR=0.5 fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	2.07	2.07	5	...	1	...
UO$_2$/FR spent fuel reprocessing				4.07							
HLW disposal				3.07							
Depleted U purchase					3.07						
Fabrication						2.57					
Interim St.								-7.07			
FR spent fuel reprocessing								-7.07			
HLW Disposal (Rep)								-8.07			

Table 6.20 – FR CR=0.75 fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	2.22	2.22	5	...	1	...
UO$_2$/FR spent fuel reprocessing				4.22							
HLW disposal				3.22							
Depleted U purchase					3.22						
Fabrication						2.72					
Interim St.								-7.22			
FR spent fuel reprocessing								-7.22			
HLW Disposal (Rep)								-8.22			

Table 6.21 – FR CR=1.0 fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	2.03	2.03	5	...	1	...
UO$_2$/FR spent fuel reprocessing	4.03										
HLW disposal		4.03									
Depleted U purchase			3.03								
Fabrication				2.53							
Interim St.								-7.03			
FR spent fuel reprocessing								-7.03			
HLW Disposal (Rep)								-8.03			

Table 6.22 – FR CR=1.23 fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	2.875	2.875	5	...	1	...
UO$_2$/FR spent fuel reprocessing	4.875										
HLW disposal		3.875									
Depleted U purchase			3.875								
Fabrication				3.375							
Interim St.								-7.875			
FR spent fuel reprocessing								-7.875			
HLW Disposal (Rep)								-8.875			

6.5 Cash Flows

At any time t, the cash flows ΣYCF are the sum of the outlays associated with the

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amortization of the reactors $\Sigma YC_{Capital}$ (calculated in Section 6.2), the O&M expenses $\Sigma YC_{O\&M}$ (calculation in Section 6.3) and the fuel cycle expenses ΣYC_{CF}.

The fuel cycle expenses ΣYC_{CF} are the sum of the payments for each of the fuel cycle services used at time t. With C_i the unit cost of the fuel cycle service "i" (provided in Section 6.4.3) and M_i^{t}[33] the mass flow processed through this service at time t:

$$\Sigma YCF(t) = \sum_i M_i^{t}(t)C_i(t) \tag{6.59}$$

The mass flows M_i^{t} are those calculated in the physical model at time t.

6.6 Cost of Electricity

It is assumed that the total nuclear enterprise will share the cost of nuclear electricity production. Thus, The Cost of Electricity is defined at any time as the ratio of the total cash flows ΣYCF per time step (calculated in Section 6.5) to the total amount of energy produced per time step. With CF^{LWR} and CF^{FR} the capacity factors of the LWR and the FR, respectively:

$$COE(t) = \frac{\Sigma YCF(t)}{(P^{LWR}CF^{LWR} + P^{FR}CF^{FR}) \cdot 365.25 \cdot 24} \tag{6.60}$$

6.7 Levelized Costs

The Levelized Cost Of Electricity LCOE is defined for a given system (e.g. a reactor) as the constant amount of money that should be charged over the lifetime of this system to cover the total costs. The LCOE is a metric calculated to make easier comparisons with alternative sources of energy and is not used for actual pricing. It is also expressed in mills/KWh (or \$/MWh). As CAFCA does not consider steady-state scenarios but rather long-term simulations (with a typical time horizon of 100 years), calculating a unique

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[33] M_i^{t} is generally different from the mass flow M_i calculated in Section 6.4.4 for the fuel cycle costs calculation and corresponding to the production of the amount of energy generated at time t.

LCOE for the entire scenario would make little sense.

Instead, CAFCA calculates a "Dynamic Levelized Cost" DLCOE, which is, at any time, the average value of the levelized costs associated with each of the reactors present in the system. This is not exactly the definition of a levelized cost (which should be constant) but rather a levelized cost continuously updated over the simulation (at anytime, the DLCOE, as defined in CAFCA, indicates what constant price should be charged per year to cover the total expenses, until either the decommissioning or the building of a new reactor occurs).

- Levelized capital costs for one single reactor LC_C:

The first step is to calculate the levelized capital costs for a single reactor (LWR or FR), expressed in $ per kWe, which is directly done in the input spreadsheet. By definition (annual compounding):

$$LYC_{capital} = \frac{i \cdot \sum_{y=1}^{L} \frac{YC_{Capital}(year\ y)}{(1+i)^y}}{\left(1 - \frac{1}{(1+i)^L}\right)}$$ (6.61)

(recall that L is the lifetime of the reactor).

- "Dynamic Levelized Cost of Electricity":

The DLCOE is defined as the sum of the weighted average value of the levelized capital costs (the weights being the share of the LWR energy output and the share of the FR energy output), the weighted average value of the O&M levelized cost (already levelized for a single reactor, as it is constant), and the fuel cycle costs per kWh. (The fuel cycle costs are more complicated to levelize, especially as different schemes can be applied to account the reprocessing costs. For the sake of simplicity, the fuel cycle cost is not levelized in CAFCA). Finally, the DLCOE is at any time t the present cost corresponding to the energy produced at that time t.

$$DLCOE = \frac{(LYC_{Capital}^{LWR} + YC_{O\&M}^{LWR}) \cdot \frac{P^{LWR}}{P^{LWR} + P^{FR}} + (LYC_{Capital}^{FR} + YC_{O\&M}^{FR}) \cdot \frac{P^{FR}}{P^{LWR} + P^{FR}} + \Sigma C_{FC}}{E}$$

with E the energy produced over the year:

$$E = (P^{LWR}CF^{LWR} + P^{FR}CF^{FR}) \cdot 365.25 \cdot 24 \qquad (6.63)$$

Chapter 7

7. Base Case Options for the U.S.

7.1 Introduction

This chapter presents the CAFCA results for the base case scenarios, for nuclear energy growth in the U.S. at 2.5% from 2020 to 2108.
The implications of the various nuclear fuel cycle options for uranium resources, amounts of actinides in the system, amounts of wastes, repository requirements, cost of electricity, and other aspects are investigated in the U.S context.

7.2 Framework and Assumptions

The four scenarios studied in this chapter as fuel cycle options are: once-through cycle (OTC), one time Pu recycling in LWRs (twice-through cycle, TTC), TRU multi-recycling in fast burners (FR CR=0.5, denoted Fbu), TRU multi-recycling in fast breeders (FR CR=1.23, denoted Fbr).
The four scenarios assume a nuclear energy demand growth rate of 2.5%/year from 2020 on (following a slower increase from 100 GWe in 2008 to 120 GWe in 2020) and take place in the U.S context (initial installed capacity of 100 GWe at the start of 2008, spent UO_2 fuel legacy of 56,800 tHM). The minimum cooling time is 5 years for all types of fuel. In the twice-through scenario, the first thermal reprocessing plant starts operation in 2025, and the separated plutonium is immediately used to make MOX fuel. In the scenarios involving fast reactors, the first thermal reprocessing plant starts in 2035, 5 years prior to the introduction of the fast reactors in 2040.
The simple model presented in Chapter 3 (see Table 3.3) provides insight into the fast reprocessing capacities needed in the different scenarios (however, this simple model is much less appropriate to assess the needs of thermal reprocessing capacities because it ignores the spent UO_2 fuel legacy). In an attempt to find optimal choices that trade off

between economies of scale and modularity, we assume for the capacity of the fast reprocessing units the values shown in table 7.1.

Table 7.1 - Fast reprocessing plant unit capacity

Scenario	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.2
Fast rep. unit capacity (tHM/year)	100	200	200	500	500

As for the thermal reprocessing plants, a single 1000 tHM/year unit is assumed in all the scenarios.

Another parameter is the industrial capacity. It is assumed for the thermal reprocessing plants an industrial capacity of 4 years/plant, which means that only one plant can start commercial operation every four years[34]. This industrial capacity is doubled in 2050. As for the fast reprocessing plants, we assume an initial (year 2040) industrial capacity of 2 years/plant, doubled in 2065. Finally, a minimum loading factor of 80% is generally ensured for the reprocessing plants, meaning that they are always used at a minimum of 80% of their capacity. However, some exceptions have been allowed; they will be made explicit.

7.3 Impact on Infrastructure Requirements

7.3.1 Reactors

Figure 7.1 shows the total LWR-UO_2 installed capacity in the four scenarios, while Figure 7.2 shows the capacity provided by the MOX fuel irradiation in LWRs (twice-through scenario), the fast burners and the fast breeders. Recall that the MOX technology is introduced in 2025 vs. 2040 for the fast reactors. As the capacity factor of the LWR and that of the FRs differ (90% vs. 85%), the total installed capacity may vary from one scenario to another, but the total energy produced per year does not.

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[34] But it does not say anything about the construction time.

LWR-UO2 - Installed Capacities (Base Case)
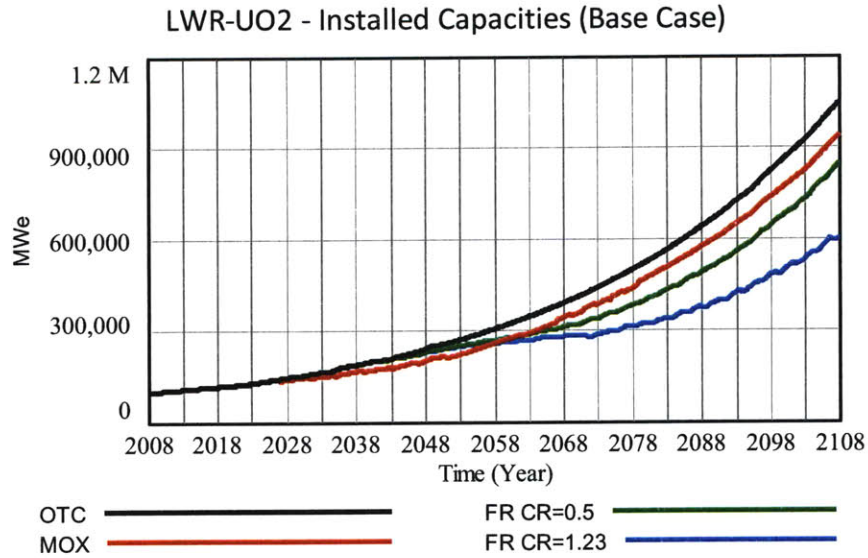
Figure 7.1 – LWR-UO2 installed capacity (base case)

Recycling Technologies - Installed Capacities (Base Case)
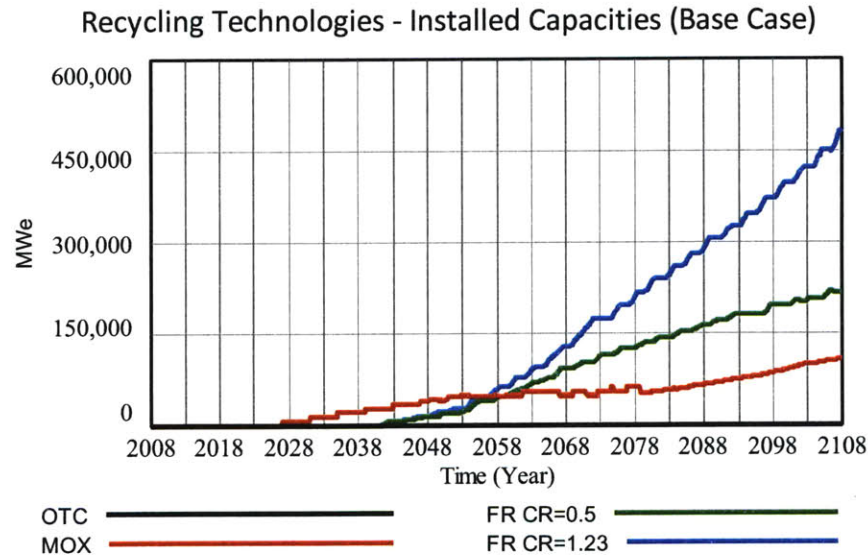
Figure 7.2 – Recycling Technologies (MOX, FR) installed capacity (base case)

As expected from the simple model, the breeder installed capacity in the nuclear energy portfolio is finally greater than that of burners, itself greater than that of LWRs-MOX. In 2108, the ratio of MOX installed capacity to the total installed capacity is 10.24% vs. 10.20% in the equilibrium model. This ratio is shown in Figure 7.3 for the three scenarios involving advanced technologies, as well as the equilibrium ratios calculated in chapter 3.

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In the MOX case, one can see that the equilibrium is reached in 2079, a few years after the spent UO2 fuel accumulated in interim storage has fallen to about 1,000 tHM (see Figure 7.5). Until then, the share of MOX capacity is above the equilibrium level thanks to the extra amounts of plutonium provided by the reprocessing of the spent fuel legacy.

Share of Advanced Technology Installed Capacity (base case)

MOX
Simple model value for MOX
FR CR=0.5
Simple model value for FR CR=0.5
FR CR=1.23
Simple model value for FR CR=1.23
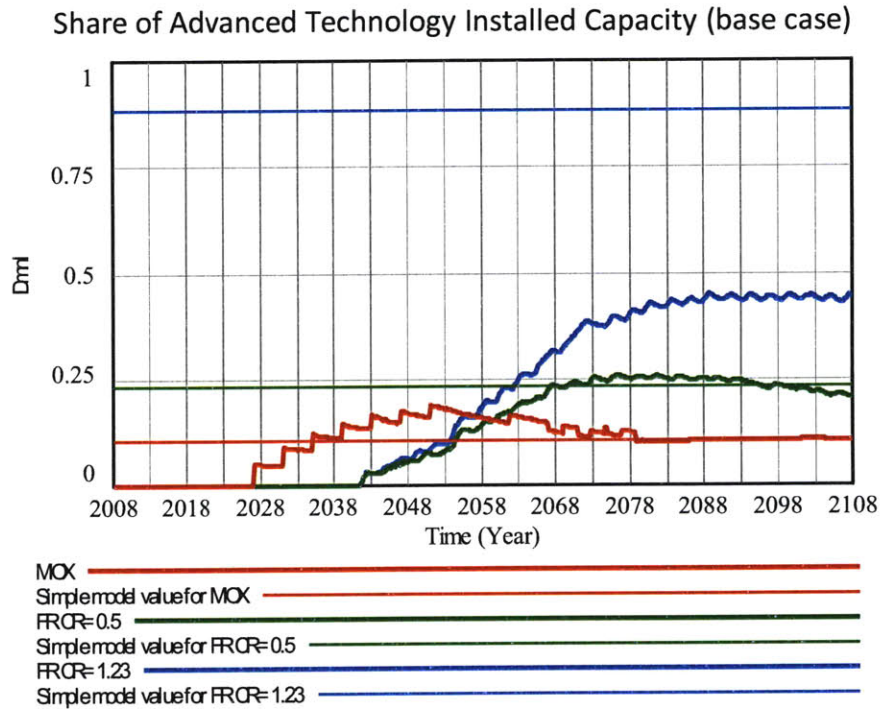
Figure 7.3 – Simple Model and CAFCA predicted share of the advanced technologies in the total installed capacity

The predictions of the simple model also seem to be verified for the burners since in 2108, their share of the total capacity is 20.43% vs. 23.23% in the model. As in the MOX case, we can observe in Figure 7.3 a slight overshoot in the period 2070s-2080s permitted by the reprocessing of the spent fuel legacy. However, the burner share should keep decreasing after 2108 until it reaches its real equilibrium level. Indeed, we observe in the results a permanent, residual depot of spent UO_2 fuel (equivalent to about 0.4 years of reprocessing activity), spent FR fuel (equivalent to about 0.2 year of reprocessing activity) and FR fresh fuel (equivalent to about 1 year of FR consumption). These residual depots are not an effect of the cap on the industrial capacity but result from the constraints on the minimum loading factors. If 1.6 years are added to the cooling time in our simple model, a new equilibrium level of 21.17% is obtained (hence a difference of

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0.74%, or a relative difference of 3.5%). Finally, the constraint #1 mentioned in Section 3.5 (start-up cores) should lead to an even lower equilibrium level.

However, the share of breeders in the total installed capacity (44.61% in 2108) remains very far from the predicted value (88.00%). Again, one can observe a residual depot of spent UO2 fuel (equivalent to about 0.6 years of reprocessing activity), spent FR fuel (equivalent to about 0.2 years of reprocessing activity) and of FR fresh fuel (equivalent to about 1.5 years of FR consumption). Adding 2.3 years to the cooling time in our simple model, we get a new equilibrium level of 65.55%. Finally, the 21.08% (or a relative 32%) difference between the observed and the final expected share is attributable to the extra need for TRU for the start-up cores. This is very important compared to what we observed in the burner case. However, the TRU content of one start-up breeder core represents 6.56 times the annual TRU loading vs. 4.06 for the burner (hence an *additional* consumption factor of 5.56 for the breeder and 3.06 for the burner). Moreover, there are 2.02 as many breeders as burners in 2108. Therefore, we obtain a "start-up factor" of 2.02 × 5.56=11.23 for the breeder vs. 3.06 for the burner. Finally, assuming that the difference between the result provided by the simple model and the CAFCA result is due to the start-up core, we should roughly expect it to be 11.23/3.06 ~4 times more important in the breeder scenario than in a the burner scenario. What we observed was a factor of 33/3.5 ~ 9.

Finally, table 7.2 shows the installed capacity for the four scenarios in 2050 and 2100.

Table 7.2 - LWR-UO₂/MOX/FR installed capacities in 2050 and 2100 (base case)

	Date	OTC	MOX	FR CR=0.5	FR CR=1.23
LWR-UO$_2$ installed capacity (GWe)	2050	251	211	238	235
	2100	859	769	677	495
MOX/FR installed capacity (GWe)	2050	0	41	16	20
	2100	0	90	196	396

7.3.2 Reprocessing plants

Figure 7.4 shows the development of the thermal reprocessing capacities in the four scenarios. Recall that the unit capacity is 1000 tHM/year and that thermal reprocessing is

introduced in 2025 in the TTC scenario vs. 2040 in the FR scenarios. Figure 7.5 shows the amount of spent UO2 fuel in interim storage (available for reprocessing).

As long as no reprocessing plant is introduced, the spent UO_2 fuel discharged from LWRs is accumulated in interim storage. A few years after the introduction of the first thermal reprocessing plant, this pile reaches a peak, as the reprocessing rate overtakes the inflow rate. In the TTC scenario, the stock of spent UO_2 fuel peaks at 91,000 tHM in 2033 (8 years after the introduction of the first reprocessing plant). In both FR scenarios, the peak occurs at 127,000 tHM and occurs in 2050 (15 years after the introduction of the first plant).

As seen in Figure 7.4, construction of the thermal reprocessing plants is mainly driven by the spent fuel legacy (but is slowed down by the limit on the industrial capacity), which is why the building schedule is exactly the same in both fast reactor scenarios until 2076. The legacy is then depleted and the spent UO_2 fuel available is directly proportional to the LWRs- UO_2 capacity. Hence, as there are more LWRs in the burner scenario than in the breeder scenario, there are also more reprocessing plants (about 11 vs. 9 in the 2090s). We also notice that, as the pile of spent fuel gets depleted in the 2070s, the number of plants is stabilized (the loading factor falls to 80% meanwhile). As the number of LWRs- UO_2 keeps increasing, the building of new plants finally restarts (and the loading factor goes back to 100%) in the early 2070s in the TTC scenario, in the late 2080s in the burner scenario, and in the 2090s in the breeder scenario. As expected from our simple model (table 3.3) the TTC scenario is the most demanding in terms of thermal reprocessing capacity (because there are more LWRs- UO_2): in 2108 there are 19 plants in the TTC scenario vs. 14 (resp. 12) in the burner (resp. breeder) scenario.

thermal reprocessing capacity (base case)
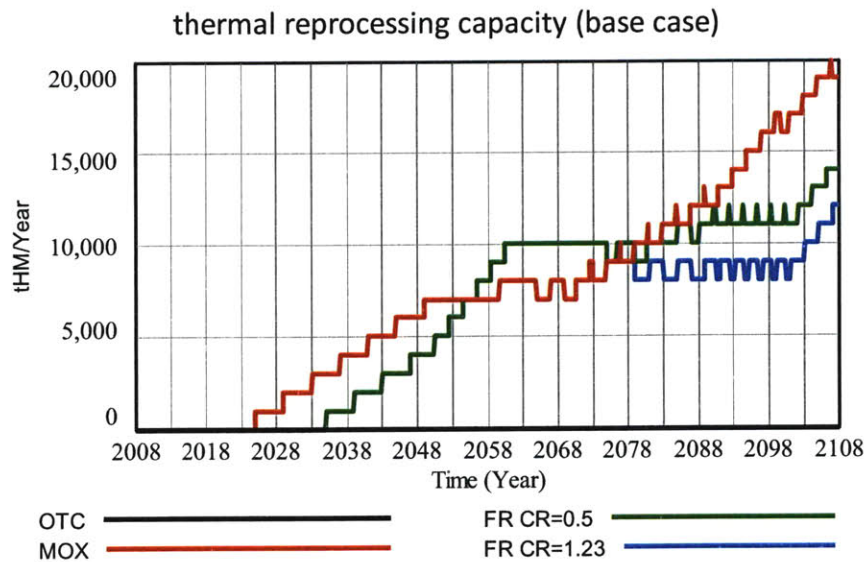
Figure 7.4 - Thermal reprocessing capacity (base case)

Spent UO2 fuel in interim storage and repository (base case)
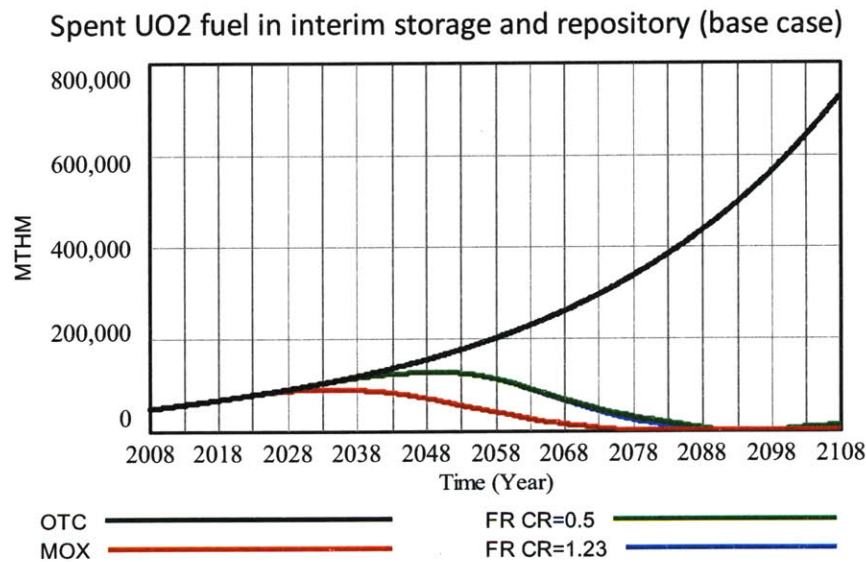
Figure 7.5 - Spent UO$_2$ fuel in interim storage and repository (base case)

Figure 7.6 shows the development of the fast reprocessing capacity. Recall that the unit capacity is 200 tHM/year in the burner case, and 500 tHM/year in the breeder scenario. In both FR scenarios, the first plant starts in 2051, 10 years after the construction of the first FR. The reprocessing capacity needed is much higher (factor of 4.6 in 2108) in the breeder case than in the burner case (6,500 tHM/year vs. 1,400 tHM/year in 2108) for

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two reasons. First, there are about twice as many breeders than burners. Second, the annual heavy metal consumption rate of a breeder is 2.4 times that of a burner (14,843 vs. 6,194 tHM/year/GWe). However, the TRU content is about the same (1541 tHM of TRU are discharged every year from a 1GWe-burner vs. 1677 tHM for the 1GWe-breeder).
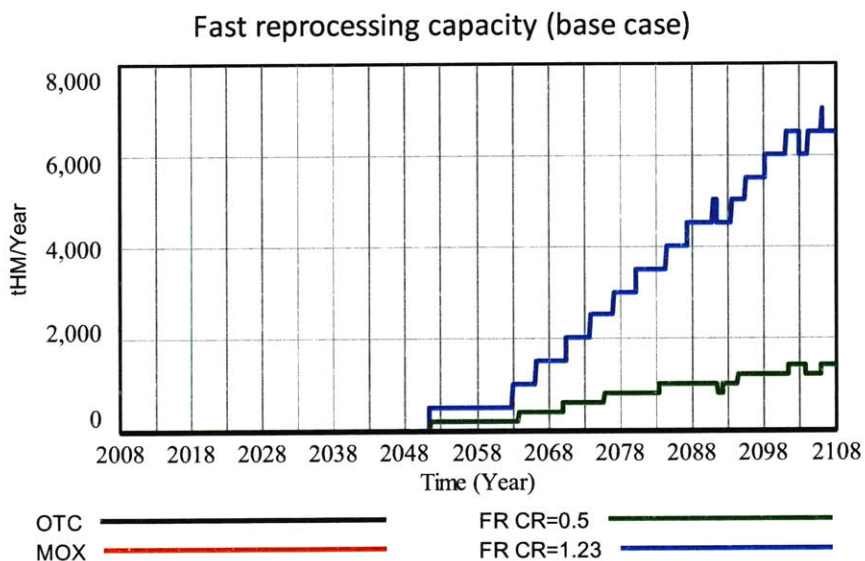
Figure 7.6 - Fast reprocessing capacity (base case)

Figure 7.7 shows the average mass loading factor in the fast reprocessing plants[35]. A limitation of CAFCA is that only one unit capacity can be chosen by the user for the entire scenario. As the nuclear reactor fleet increases exponentially, it would be more appropriate to enable the code to increase the unit capacity over time. As a result of this limitation, the first plant remains underused (loading factor under 80%) for 5 to 6 years, the second plant starting only in 2063. This overcapacity would have been avoided if the capacity of the first plant was 100 tHM/year in the burner scenario and 200 tHM/year in the breeder scenario. Using these lower unit capacities over the entire scenario would have been possible, but the opportunity to benefit from economies of scale would have been lost in the second half of the century. Actually, additional economies of scale could have been obtained from the 2060s on by building 1000 tHM/year plants in the breeder scenario, and tHM 500/year plants in the burner scenario. Finally, our choices for the unit

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[35] erratum : the y-axis unit on Figure 7.7 is "dimensionless" and not MWe

capacities result from a trade-off between economies of scale and modularity, given the limitations of the code.

<figure>
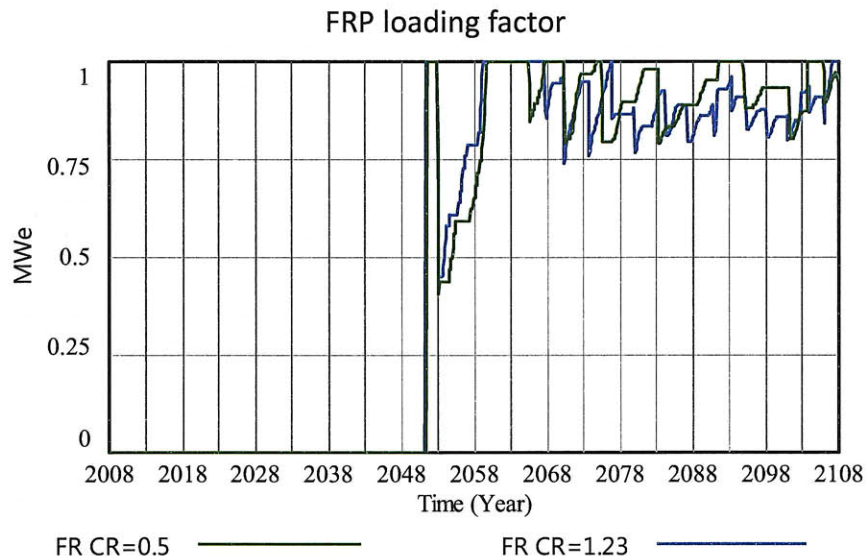
FRP loading factor

Figure 7.7 - FRP mass loading factor (base case)

</figure>

7.4 Impact on natural uranium requirements and price

7.4.1 Impact on U.S natural uranium consumption

One motivation for the introduction of advanced technologies is a perceived scarcity of economic the natural uranium resources in the long-term, which would lead to a large rise in price. For some countries such as India, the limitation of natural uranium needs in the long-term is more a matter of national security. In both cases, it is important to assess the impact of the various strategies on uranium consumption.

By replacing LWRs- UO_2 with other types of reactors/fuel (fed with transuranics and depleted uranium), the introduction of advanced technologies automatically reduces the needs for natural uranium. Moreover, the utilization of recovered uranium (from thermal reprocessing plants) as a substitute for natural uranium allows extra savings. Figure 7.8 shows the natural uranium utilization rate in the four scenarios.

Natural U utilization rate (base case)
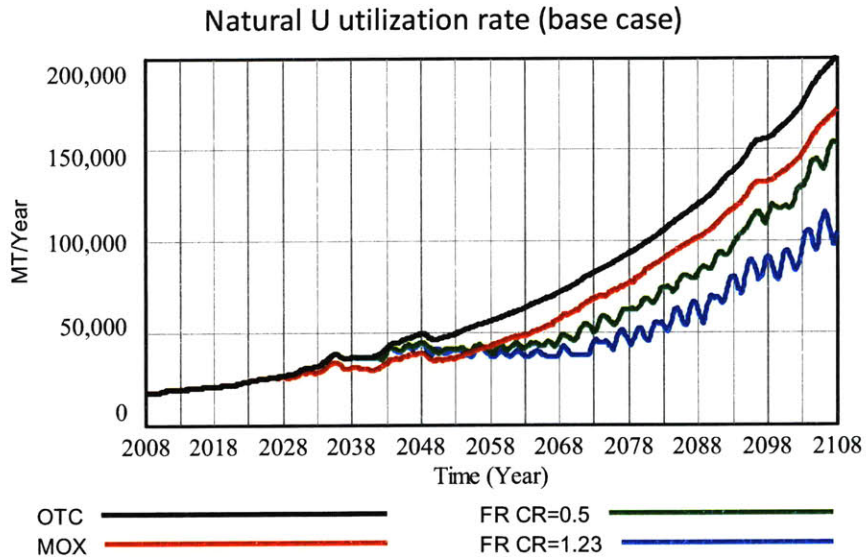
Figure 7.8 - Natural uranium utilization rate (base case)

The oscillations that one can observe in the two FR cases reflect variations in the number of LWRs, with peaks occuring when one or more new LWR is commissionned, as each requires the loading of an entire core at once. This is particularly obvious in the 2030s, as most of today's existing reactors will be decommissionned then, and hence replaced by new units.

Table 7.3 shows some values of the natural uranium utilization rate (average value over 3 years) in the various scenarios.

Table 7.3 - Natural uranium utilization rate (base case)

Scenario	Date	OTC	MOX	FR CR=0.5	FR CR=1.23
If the recovered uranium is recycled, after re-enrichment (tons/year)	2050	X	~ 35,000 *(-24.0%)*	~ 40,000 *(-13.0%)*	~ 40,000 *(-13.0%)*
	2100	X	~ 135,000 *(-16.0%)*	~ 118,000 *(-26.5%)*	~ 86,000 *(-46.5%)*
If the recovered uranium is not recycled (tons/year)	2050	~ 46,000	~ 40,000 *(-13.0%)*	~ 43,000 *(-6.5%)*	~ 43,000 *(-6.5%)*
	2100	~ 161, 000	~ 148,000 *(-8.0%)*	~ 128,000 *(-20.5%)*	~ 95,000 *(-41.0%)*

Unlike the 2100 results, the 2050 results are not a good indicator of the long-term trends as the system is still very far from equilibrium (the spent fuel legacy is not depleted yet,

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providing extra amounts of recovered uranium, and the fast reactors are just starting to be deployed). We can see in 2100 that, as expected, the breeder strategy yields the best results in terms of natural uranium savings, reducing its consumption by half (-46.5%). The burner strategy is half as efficient (-20.5%) while the MOX strategy only yields very modest results (-16.0%, of which half is allowed by the utilization of recovered uranium). The more FRs (or LWR-MOX), the fewer LWRs, and finally the less spent UO_2 fuel generated, and hence the less uranium recovered. This is why the reduction in natural uranium consumption allowed by the use of recovered uranium is larger in the TTC scenario (13,000 tons in 2108), than in the burner case (10,000 tons) and the breeder case (9,000 tons).

Figure 7.9 shows the cumulative natural uranium consumption from 2008 on (when recovered uranium is used). This result is interesting as it takes into account the history of the entire century and, unlike the consumption rate, is little affected by temporary variations. In terms of cumulative uranium savings, the breeder scenario catches up with the TTC scenario in 2068, and with the burner scenario 5 years later in 2073.
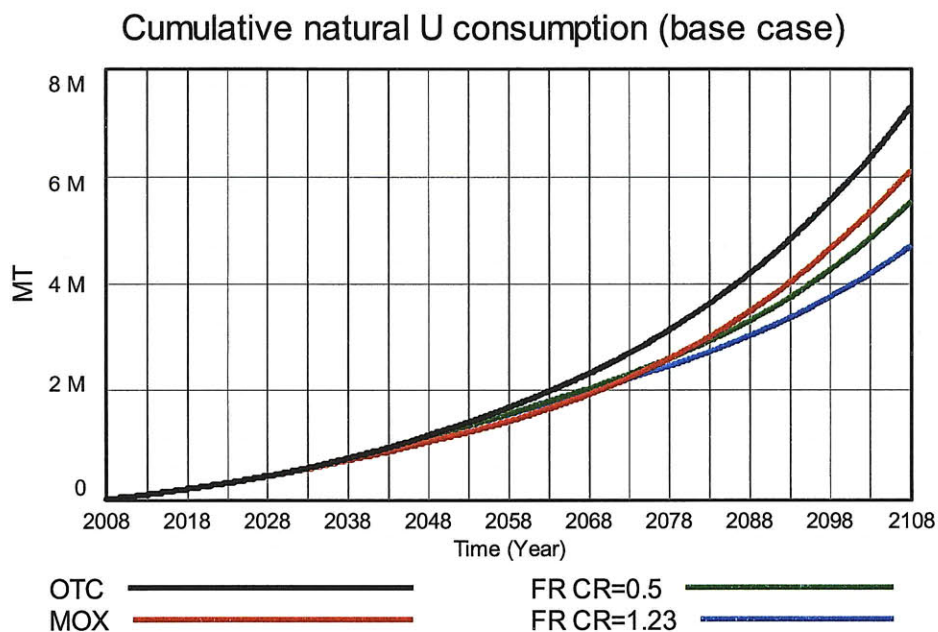
Figure 7.9 – Cumulative natural uranium consumption (base case)

Table 7.4 shows the cumulative natural uranium consumption in 2050 and 2100 in the various scenarios (in millions of tons). We note that the impact of the utilization of MOX is modest (-16.5% compared to the OTC scenario 75 years after its introduction). Despite its later introduction, the impact of the breeders is much more significant (-35.5% compared to the OTC scenario 60 years after its introduction).

Table 7.4 - Cumulative natural U consumption and its magnitude relative to the OTC case (base case)

Scenario	OTC	MOX	FR CR=0.5	FR CR=1.23
2050	1.26 Mtons	1.12 Mtons *(-11%)*	1.22 Mtons *(-3%)*	1.22 Mtons *(-3%)*
2100	5.86 Mtons	4.90 Mtons *(-16%)*	4.42 Mtons *(-24%)*	3.78 Mtons *(-35%)*

7.4.2 Impact on World reserves and Uranium Price

The numbers shown in the previous section are of little meaning if they are not set against the assessed world reserves in natural uranium. The model developed by [Matthews and Driscoll, 2009] and described in Section 6.4.3.1 can be helpful here[36]. We assume in all the cases that the world consumption is 3.5 times that of the U.S.

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[36] Note that this model does not address the uranium contained in phosphates (low production volumes) and in the seawater (technology not mature and still too expensive).

Figure 7.10 shows the resources available at less than 150 $/kg (nominal value, 2008 dollars) in the medium case ($\Theta = 0.11$). In this medium case, the initial world reserve (accessible at 150 $/kg) is of 78 million tons. This initial value is extremely sensitive to the value of Θ (e.g = 0.12 leads to an initial reserve of 57 million tons). In 2108, the OTC (resp. MOX, burner, breeder) scenario has consumed 42% (resp 36%, 32%, 27%) of the world reserves, leaving 45 (resp 50, 53, 57) million tons in reserves (with recycling of recovered uranium). In 2108, the breeder scenario has consumed 21 million tons worldwide; this level is reached in 2091 in the OTC scenario, in 2098 in the TTC scenario and in 2101 in the burner scenario.
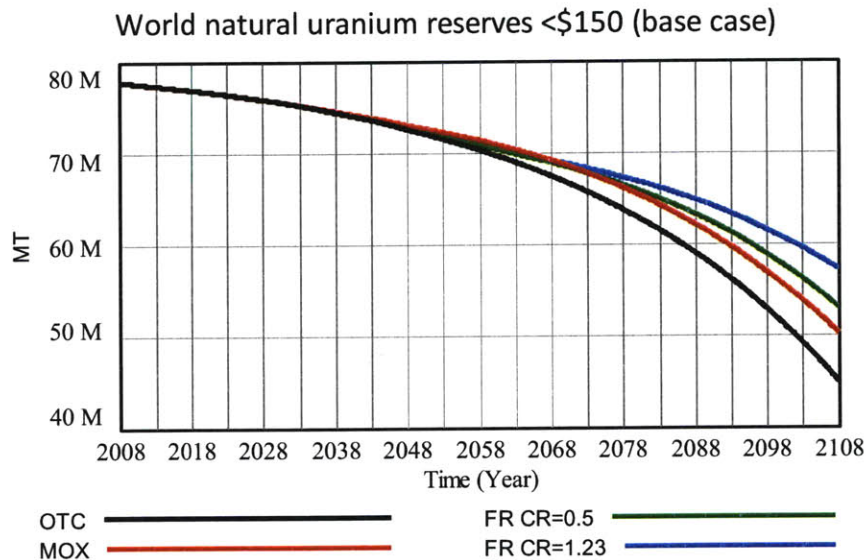
Figure 7.10 - World reserves (less than 150 $/kg) in the medium case (base case)

As the reserves accessible at less than 150 $/kg are not depleted in 2108, the price of uranium should logically remain under this level over the entire simulation whatever the scenario. Figure 7.11 shows the price of natural uranium over time. As one can see, this price is barely affected by the option chosen. Starting from a reference price of 100 $/kg in 2108, we end up with a price of 137 $/kg in the OTC scenario vs. 134 $/kg, 133 $/kg and 131 $/kg in the MOX, burner and breeder scenario, respectively (all values are real and given in 2008 dollars).
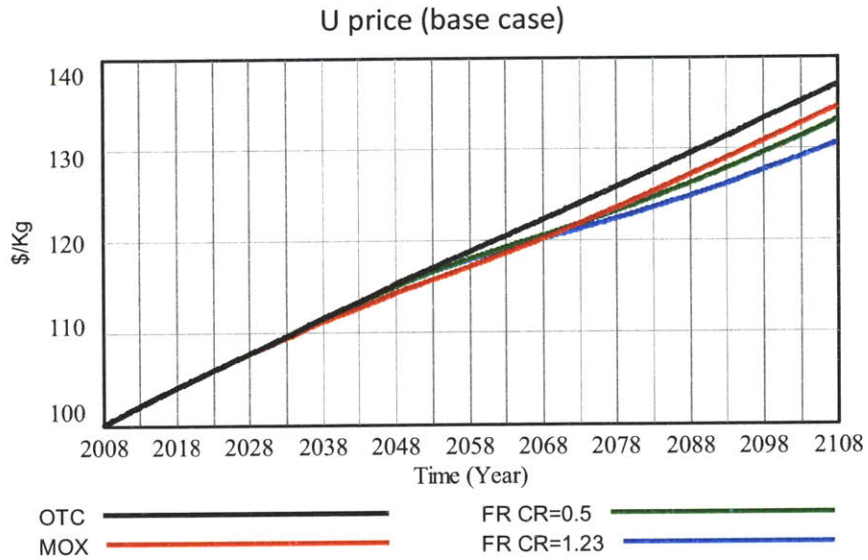
Figure 7.11 – U price in the medium case (base case)

The pessimistic case was also explored, assuming a pessimistic $\Theta = 0.29$. Figure 7.12 shows the resulting U price. Even in this pessimistic framework, the price of uranium has only doubled in 100 years, from 100 $/kg in 2008 to 229 $/kg in 2108 in the OTC case vs. 202 $/kg in the breeder case.
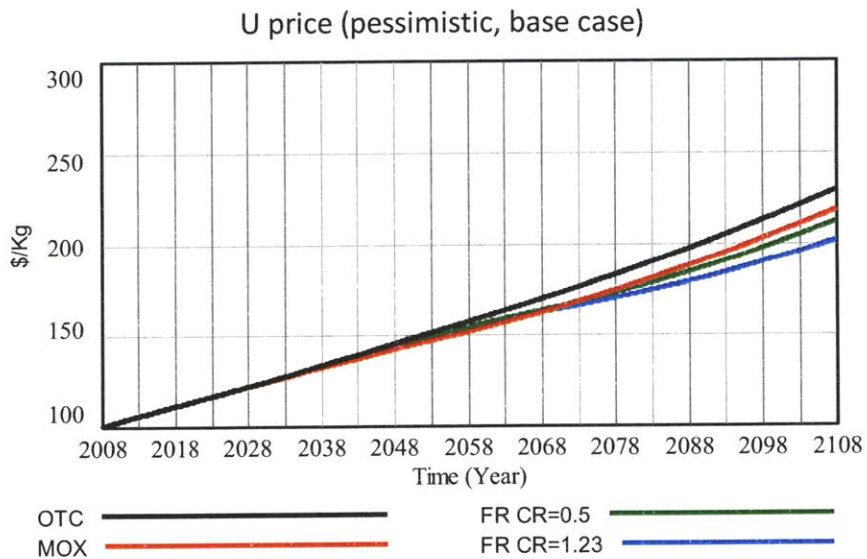
Figure 7.12 - U price in the pessimistic case (base case)

A consequence is that the reserves of uranium accessible at less than 150 $/kg are depleted sometime over the century. Figure 7.13 shows the level of these reserves for the four options. The initial world reserve of 6.1 million tons is depleted in 2052 in the OTC scenario and 2053 in both FR scenarios. The early introduction of thermal reprocessing pushes back this depletion time to 2056 in the TTC scenario, which is negligable.
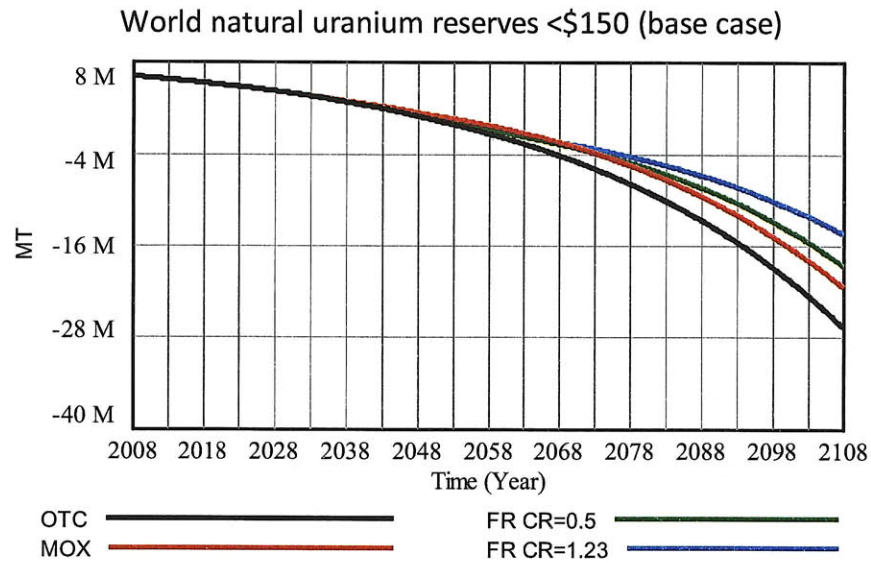
Figure 7.13 - U reserves (<150 $/kg) in the pessimistic case (base case)

Finally, figure 7.14 shows the evolution of the price of uranium in the optimistic case (Θ = -0.10). Unlike the medium and the pessimistic case, the price of natural uranium decreases as it is consumed, ending up at less than 80 $/kg whatever the scenario. The reserves (< 150 $kg) keep increasing meanwhile.

U price (optimistic, base case)

OTC   FR CR=0.5
MOX   FR CR=1.23
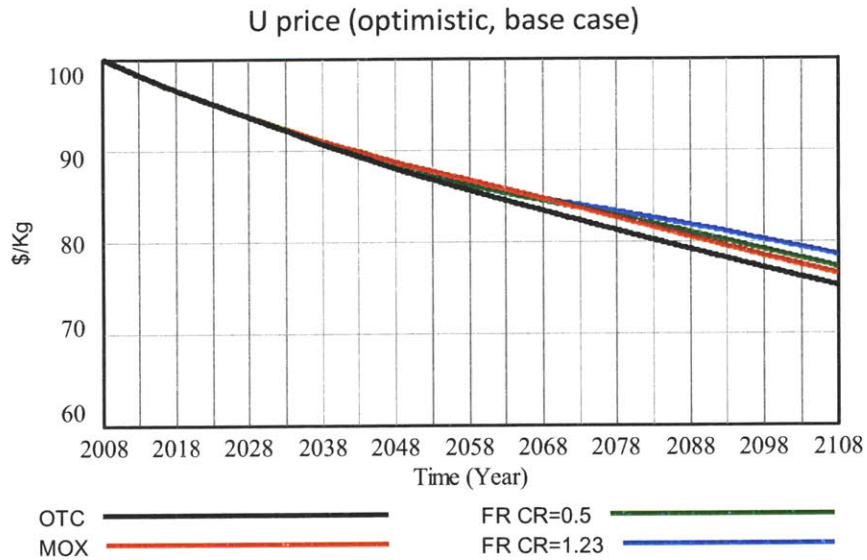
Figure 7.14 - U price in the optimistic case (base case)

7.5 Impact on actinide inventories

Another aspect of the nuclear fuel cycle is the transuranics inventories (either separated or mixed with other products), which can be seen as a source of fuel (MOX, fast reactors) rather than a waste, but their mere existence potentially poses a proliferation concern. The rationale behind the burning strategies is to reduce the long-term inventories of plutonium and minor actinides while producing energy. If the vision of the transuranics as a substitute for enriched uranium outweighs proliferation concerns, the breeder option will be preferred.

Figure 7.15 shows for the four scenarios the total amount of TRU in the system regardless of their location (LWR cores, FR cores, fuel fabrication plants, cooling storages, interim storages, reprocessing plants, wastes).

TRU: total mass in the system (base case)
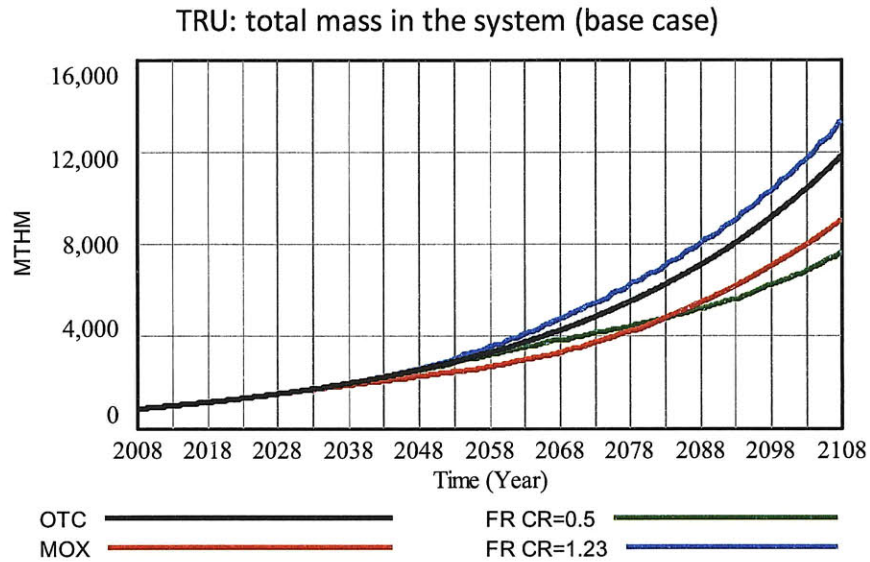
Figure 7.15 – Total amount of TRU in the system (base case)

TRU are continuously produced in LWR cores, the number of which keeps increasing (see Figure 7.1). It is therefore expected to observe an upward trend in the total mass of TRU. However, the difference between the OTC and the advanced options is not necessarily intuitive. Indeed, LWRs- UO_2, which are burners of fissile materials, are also net producers of TRU, and introducing fast reactors reduces their numbers. Therefore, there are two effects:

- Complementary in the burner schemes: TRU producers (LWR) are replaced by TRU burners (MOX/Fast burners).

- Antagonistic in the breeder scheme: TRU producers (LWR) are replaced by TRU producers (Fast Breeders).

Starting from a total TRU inventory of 840 tHM (600 tHM in interim storages, 125 tHM in cooling storage, and 115 tHM in LWR cores), we have in 2108 a total of 11,785 tHM in the OTC scenario, 9,010 tHM in the TTC scenario, 7,590 in the fast burner scenario, and 13,310 in the breeder scenario. In terms of TRU reduction, the fast burner scenario catches up with the TTC scenario in 2083.

For each of the scenarios, the locations of the TRU are shown in composite graphs in Appendix A.7.1. In the fast reactor cases, we can see that, as soon as the legacy is

depleted, the TRU is mainly located firstly in the cooling storages (at-reactor pools), secondly in the reactor cores, which may be considered more secure than interim storages. In the twice-through scenario, the TRU is mainly accumulated in the wastes (minor actinides mixed with fission products, or in spent MOX fuel). Of course, TRU are mainly accumulated in the spent UO_2 fuel inventories (in repository) in the OTC scenario.

7.6 Impact on Repository Needs

Nuclear waste is one of the main causes for public anxiety about nuclear energy. Their reduction could therefore have a significant impact on the public acceptance of nuclear energy. Although recycling options dramatically reduce the total mass of wastes ($95\%_w$ of the spent UO2 fuel is recovered and recycled), they do not eliminate the necessity of a deep repository, as fission products and unrecoverable TRU amounts (losses) still have to be disposed of.

Figure 7.16 shows the total mass of HLW destined to a repository in the various scenarios. Recall that HLW is assumed to be sent to disposal 25 years after it is generated (it is cooled in surface storage meanwhile). We assume that the repository opens in 20 years, i.e. in 2028. For each scenario, the composition of the HLW is shown in appendix A.7.2.

HLW in repository (base case)
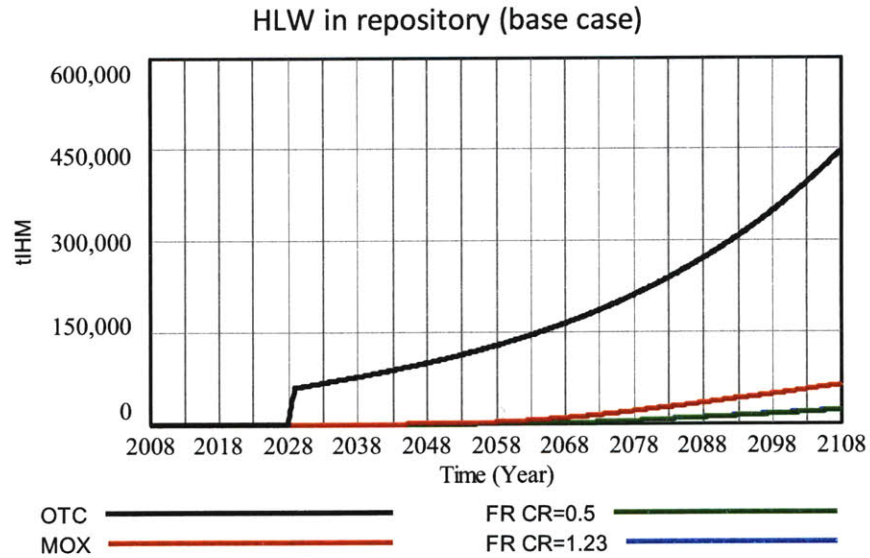
Figure 7.16 – Total amount of HLW in repository to open in 2028 (base case).

As expected, the OTC generates the greatest amount of HLW in terms of mass. The current spent fuel legacy (56,800 tHM) is transferred to the repository in 2028. The accumulation of spent UO2 fuel in repository reaches 444,000 tIHM in 2108, equivalent to "6.3 YM" (YM standing for "Yucca Mountain", assuming its legal capacity). About two thirds as much as this amount will be waiting in interim storage: 293,000 tIHM in 2108.

For comparison, the HLW in repository rises to only 63,000 tIHM in 2108 in the TTC scenario, plus about two thirds of this amount in interim storage (43,000 tIHM). When comparing these data, one must keep in mind that the recycling options may also delay the transfer to repository of some wastes (at least in our model). The same fission products as contained in the UO_2 spent fuel will be sent to repository at t+25 in the OTC scenario (t being the time when the spent fuel is discharged from the LWRs). With T being the period of time between the transfer of spent UO_2 fuel to interim storage (after a minimum period of cooling) and its reprocessing, the same fission products are sent to disposal at t+25+T in the TTC scenario.

In both FR scenarios, HLW are essentially fission products, with which some TRU losses are mixed. The amount of fission products generated being roughly proportional to the energy produced, the FR scenarios produce about the same amount of HLW, whatever

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the conversion ratio: about 22,000 tIHM in repository + 14,000 tIHM in interim storage in both scenarios.

However, the mass is not an appropriate metric to compare the different scenarios. Wastes vary in decay heat, volume (including packages) and radio-toxicity. Figure 7.17 shows the aggregated amount of wastes using the densification factors described in Section 5.5.
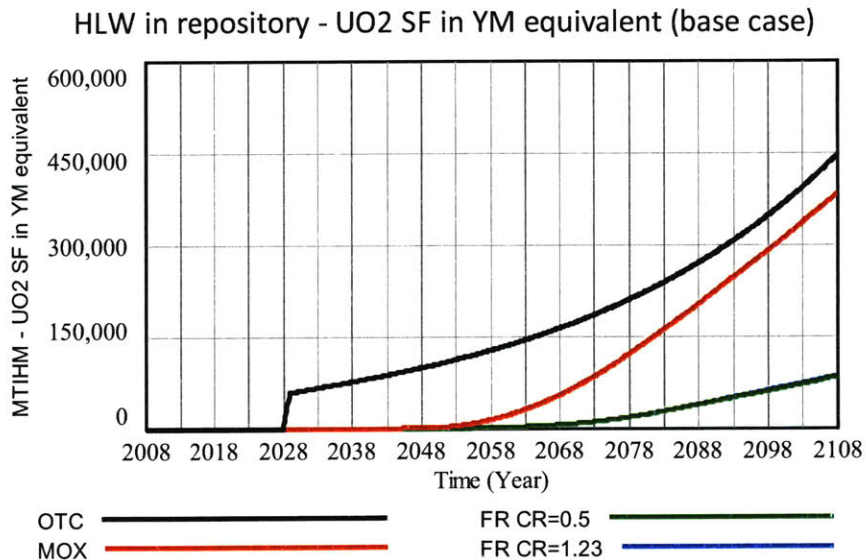
Figure 7.17 – HLW in repository in YM equivalent of spent fuel (base case)

The comparative advantage of the recycling options is reduced. In 2108, the repository requirements of the twice-through scenario are equivalent to those of 382,000 tIHM of spent UO_2 fuel (5.5 YM, or a gain of only 13% with respect to the OTC). In both FR cases, the repository requirements in 2108 are equivalent to those of 84,000 tIHM of spent UO_2 fuel (1.2 YM, or a gain of 81% with respect to the OTC).

Finally, Figure 7.18 shows the amount of TRU in repository. In the FR scenarios, the 20 tHM of TRU present in wastes in 2108 are diluted in about 38,000 tIHM of fission products and therefore do not pose any proliferation concern. The twice-through scenario reduces the TRU content in wastes, in 2108, by only 40% (5.6 tHM vs. 9.3 tHM). Recall

that in this scenario, the minor actinides are left with the fission products over the reprocessing of the spent UO_2 fuel and the spent MOX fuel (which still contains about $7\%_w$ of TRU) is disposed of.
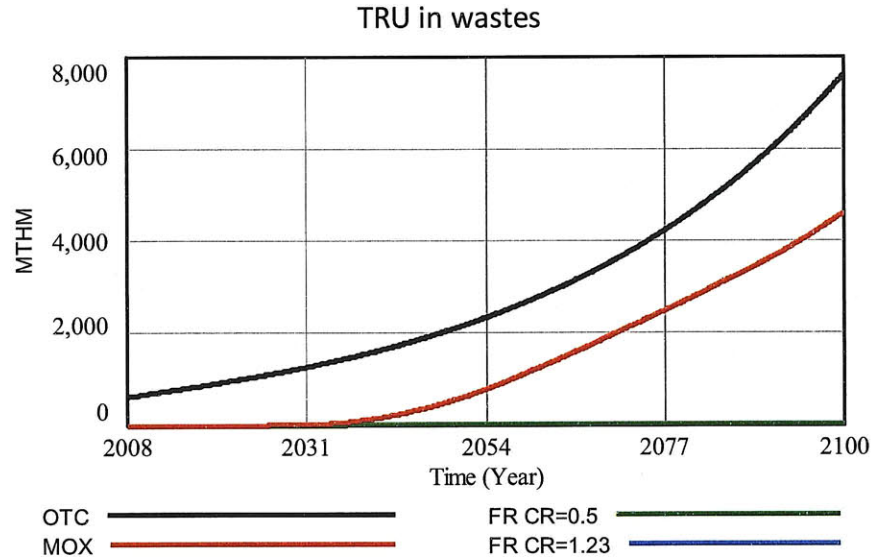
TRU in wastes

Figure 7.18 – TRU content in wastes (base case)

7.7 Impact on Economics

7.7.1 Introduction

As rational agents, utilities are supposed to make their decisions solely based on economics. It is therefore of primary importance to consider the impact of the fuel cycle choices on the total cost of electricity production.

It is well known that the OTC scheme is the least expensive fuel cycle given the current data. However, the external advantages brought by the other options, such as waste burden reduction, reduced need for natural uranium, reduced inventory of stored TRU (which may be a matter of national security, or public acceptance), may be internalized in the future, through subsidies or legal requirements. An economic assessment of the other options thus remains necessary, in order to evaluate the potential costs of such measures. Reprocessing and disposal costs are treated differently over time depending on whether recycling technologies are introduced or not. The use of a hybrid economic scheme is due to the necessity to account for the current U.S. legal system, namely the enforcement of

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the nuclear waste policy act (see Section 6.4.3.5), which is very likely to prevail in the near future. For all fuel cycle scenarios, utilities are assumed to pay a 1-mill/kWh fee until reprocessing is introduced (in 2025 in the TTC scenario, in 2040 in the FR scenarios). Then, reprocessing and disposal are considered as fuel cycle services and paid as such, as described in Section 6.4.1 (see Section 6.4.3.4 and 6.4.3.5 for the unit costs and Section 6.4.5 for the lead times). Two different approaches are used for the second phase of the scenario: an "energy-based" accounting scheme (meaning that reprocessing and HLW disposal costs are accounted for as front-end expenses for the post-processing cycle, which uses the separated fissile materials to produce energy) and a "waste-based" accounting scheme (meaning that reprocessing and HLW disposal costs are accounted for as back-end expenses for the pre-processing cycle that generates the spent fuel).

Note that this model needs to be refined. Indeed:

- If *the "energy-based" accounting scheme* is used: potential draw on the nuclear waste fund (resulting from the collection of the waste fees plus the accrued interest) is not accounted for in the model. In effect, utilities (or nuclear electricity consumers) are paying twice for the management of the spent fuel generated prior to the introduction of reprocessing (thus our present model could be an *overestimation of the costs*).

- If the *"waste-based" accounting scheme* is used: the nuclear waste fund that utilities are feeding over the first phase would be appropriate for paying the cost of processing, but may not be sufficient to cover, in the future, the costs of the reprocessing (and associated HLW disposal) of the spent fuel generated prior to the introduction of recycling. However, this may not be the case if recycling is introduced late as interest on the funds is accrued over the long time period (thus our present model is *potentially an underestimation of the costs of the cycles requiring reprocessing*).

7.7.2 Dynamic Levelized Cost of Electricity

Figure 7.19 shows the total fuel cycle costs for the four scenarios, using the 'energy-based" accounting scheme. The OTC fuel cycle cost increases by about 30%, from about 9.6 mills/kWh in 2008 to 12.2 mills/kWh in 2108. This increase is due to the rise of the price of uranium from $100/kg to 137 $/kg (the contribution of the uranium purchase to the OTC fuel cycle cost goes from 31% to 39%, and is even more important once costs are discounted).

Fuel Cycle Costs (energy-based accounting scheme, base case)
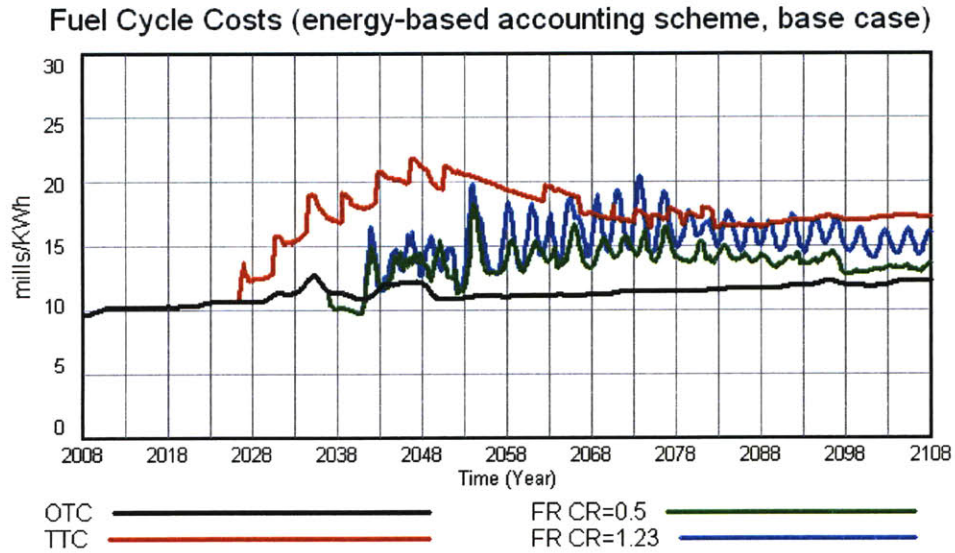
Figure 7.19 – Fuel Cycle Costs (Energy-based accounting scheme, base case)

The TTC cost takes off in 2025 as the first batches of spent UO_2 fuel are reprocessed. The maximum (22 mills/kWh) reached in the late 2040s occurs when the share of MOX in the total installed capacity is at its maximum (see Figure 7.3). When equilibrium is reached (from 2078 on), the difference between the OTC cost and the MOX fuel cycle cost is stabilized at about 5 mills/kWh (+42%).

One may be surprised that the FR fuel cycle costs are lower than the MOX fuel cycle cost. This difference is due to the spent fuel management costs. In the TTC scenario, the spent MOX fuel is sent to disposal, which is relatively expensive because of the concentration of "hot" minor actinides (densification factor of 0.15, contributing to a system-averaged disposal cost of 2.4 mills/kWh); whereas in the FR scenarios, the reprocessing of the spent fuel is accounted as a front-end expense in the *next* cycles that recycle its TRU content.

While the composition of the nuclear system is stabilized over the last decade, the difference of cost between the OTC and the burner scenario is about ~1.5 mills/kWh (+13%); the difference between the OTC and the breeder scenario is about ~ 3.5 mills/kWh (+29%). The breeder scenario is more costly than the burner scenario mainly because it implies more reprocessing costs (lower unit cost but much higher throughput, see Figure 6.6).

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Figure 7.20 shows the dynamic levelized O&M costs (DLOMC), which is the weighted average levelized O&M costs, the weights being the installed LWR capacity and the installed FR capacity (divided by the total installed capacity). The O&M costs are 8.9 mills/kWh for the LWRs (assuming a 90% capacity factor) and 9.4 mills/kWh for the FRs (assuming a 85% capacity factor). In 2108, the DLOMC is 9 mills/kWh in the burner scenario, and 9.1 mills/kWh in the breeder scenario, simply reflecting the greater share of FRs in the reactor park.
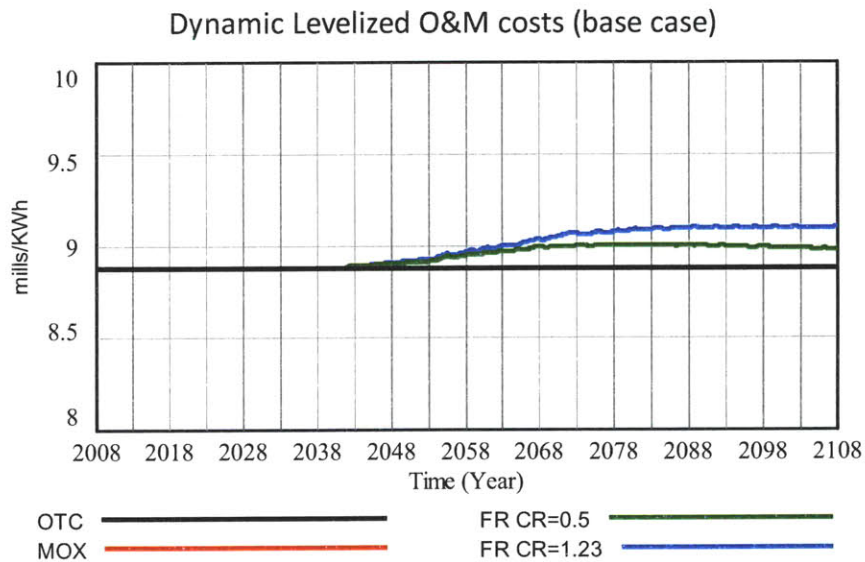
Figure 7.20 – Dynamic Levelized O&M costs (base case)

Figure 7.21 shows the dynamic levelized capital costs (DLCC), calculated similarly to the DLOM. The capital costs, or "reactor costs" include construction costs, incremental capital costs, decommissioning costs, and associated charges. The total capital cost is 65 mills/kWh for the LWR and 81 mills/kWh for the fast reactor. As the proportion of fast reactors increase, the DLCC slowly increases from 65 mills/kWh to 69 mills/kWh in the burner scenario and 72 mills/kWh in the breeder scenario. The TTC scenario has no impact on the capital costs, as MOX fuel is loaded into LWRs.

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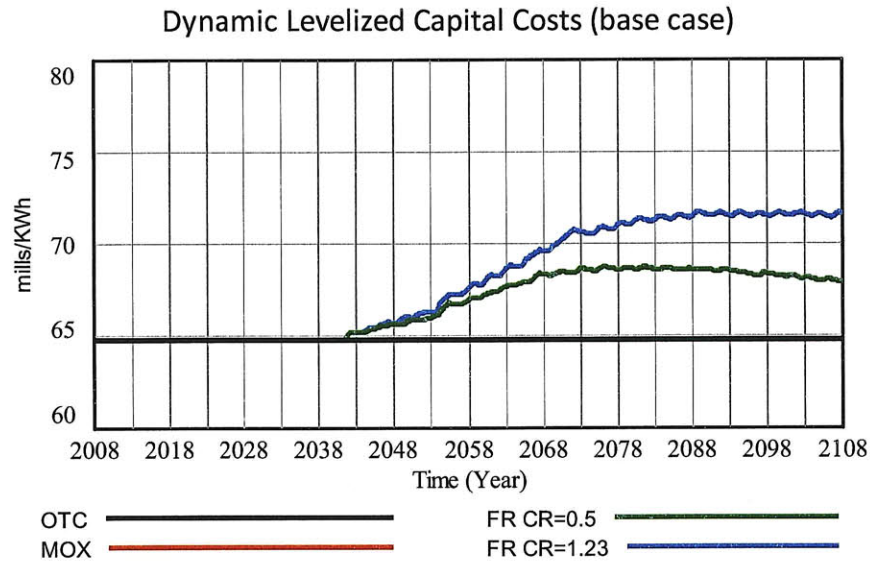
Figure 7.21 – Dynamic Levelized Capital Costs (base case)

Figure 7.22 shows the total dynamic levelized cost of electricity (DLCOE), which is the sum of the fuel cycle costs, DLCC and the DLOM costs (8.9 mills/kWh for the LWRs, 9.4 mills/kWh for the FRs). In all the scenarios, the DLCOE starts at 82 mills/kWh in 2008. The OTC ends up at 85 mills/kWh in 2108, owing to the increase in natural U price. In the burner scenario as well as the MOX scenario, the DLCOE ends up at 90 mills/kWh (+6% compared to the OTC scenario) and remains in the range 82-93 mills/kWh over the entire simulation. In the breeder scenario, the DLCOE ends up at 96 mills/kWh (+13%) and remains in the range 82-99 mills/kWh.

DLCOE (energy-based accounting scheme, base case)
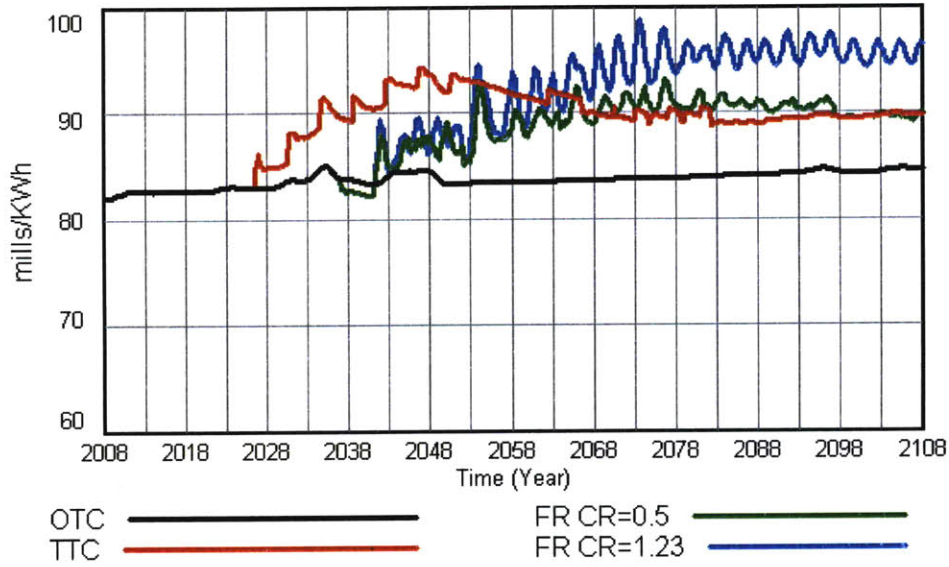
Figure 7.22 – Dynamic Levelized Cost of Electricity (energy-based accounnting scheme, base case)

The results are significantly different if a "waste-based accounting scheme" is applied, in which the reprocessing costs are paid for by the cycle that generates the spent fuel. Figure 7.23 shows the resulting fuel cycle costs.

Fuel Cycle Costs (waste-based accounting scheme, base case)
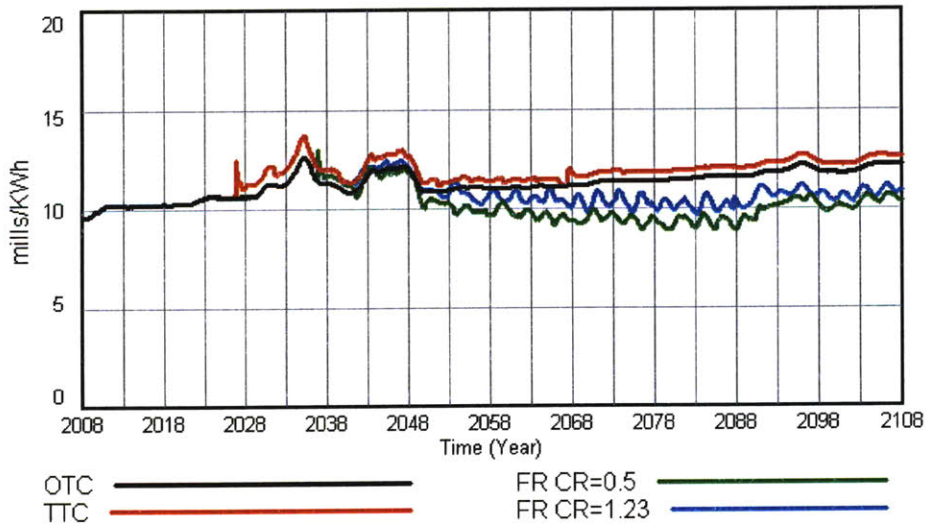
Figure 7.23 – Fuel cycle costs (waste-based accounting scheme, base case)

The difference between the various scenarios is much smaller than in the "energy-based scheme" because reprocessing costs are now accounted for as back-end expenses and discounted as such. Furthermore, the costs of reprocessing of the spent fuel legacy do not occur in this graph as they are assumed to have been paid prior to 2008. In 2008, the average fuel cycle costs still amounts to 9.6 mills/kWh in all the scenario. Once the equilibrium is reached (from the late 2070s on), the difference between the OTC scenario (12.2 mills/kWh in 2108) and the TTC scenario (12.6 mills/kWh in 2108) is about 0.4 mills/kWh (vs. 5 mills/kWh in the "energy-based" view).

It is surprising to see that, unlike in the energy-based accounting scheme, fuel cycle costs get lower in the FR scenarios than in the OTC scenario once the FR are introduced. Front-end services (natural U purchase, enrichment, UO_2 fuel fabrication) are paid ahead in the OTC scenario (positive lead times), which generates additional financial charges. In the FR scenarios, the most expensive of the front-end steps, namely the reprocessing of the spent UO_2 fuel and spent FR fuel that provide the fissile materials, is now accounted for in the costs of the previous cycle, which generate these spent fuels. As the reprocessing of spent fuel occurs years after its generation, the payment required at the time of this generation is reduced compared to the nominal cost because interest will be accrued over time. Finally, the combination of the time value of money and the accounting of the reprocessing cost as a back-end step makes the fuel cycle costs less expensive in the FR scenarios than in the OTC scenario once the FRs are introduced. In 2108, the difference between the OTC scenario (12.6 mills/kWh) and the FR scenarios (~11 mills/kWh) remains between 1 and 2 mills/kWh.

Finally, Figure 7.24 shows the dynamic levelized cost of electricity (sum of the DLCC, DLOM and fuel cycle costs), using the waste-based accounting scheme for the reprocessing costs. Overall, the higher FR capital costs have the dominant impact. The DLCOE still starts at 82 mills/kWh. In 2108, the DLCOE is about 84.4 mills/kWh in the OTC scenario, 0.4 mills/kWh more in the TTC scenario, ~87 mills/kWh in the burner scenario, ~91 mills/kWh in the breeder scenario.

DLCOE (waste-based accounting scheme, base case)
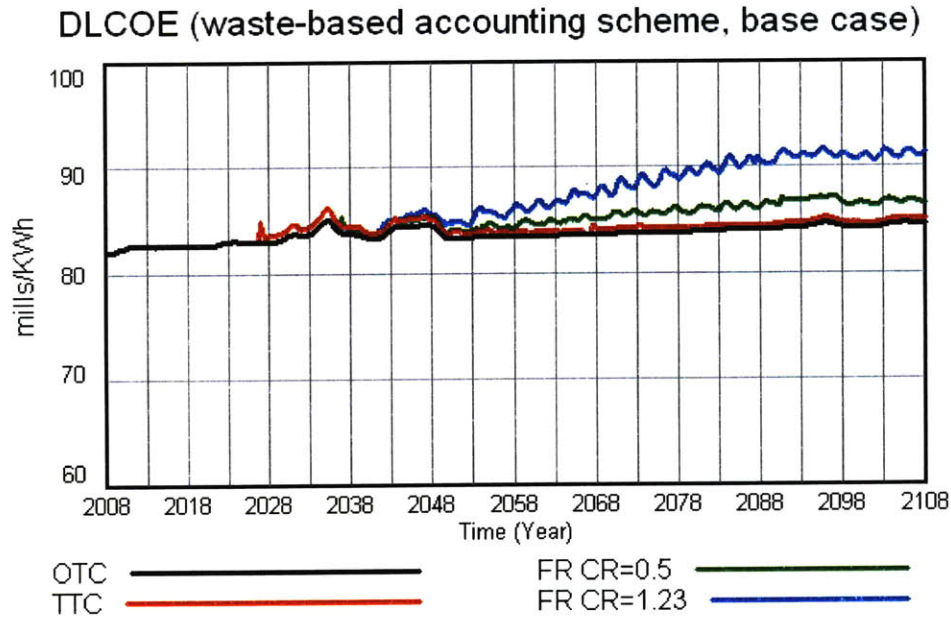
Figure 7.24 – Dynamic Levelized Cost of Electricity (waste-based accounting scheme, base case)

7.7.3 Cost of Electricity (Cash Flows per kWh)

We saw in the previous section that the accounting scheme chosen had a significant impact on the fuel cycle costs, to the point that the year-by-year comparisons may be reversed.

Figure 7.25 shows the cash flows associated with the fuel cycle; payments are made at the time the service is used (therefore neither the time value of money nor the account scheme chosen has an impact here). Note that these services are not associated with the electricity produced at the same time, unlike the fuel cycle costs calculated in the previous section. Finally, this "cash flow per kWh" shows year-by-year the cost of electricity that the utilities must cover as part of their annual expenses.

The fuel cycle cash flows starts in 2008 at 7.2 mills/kWh in all the scenarios. In 2108, the OTC fuel cycle cash flow per kWh is 8.9 mills/kWh vs. 11.3 mills/kWh for the TTC, 9.3 mills/kWh for the burner scenario, and 10.2 mills/kWh for the breeder scenario. It is overall observed that, contrary to what the "waste-based" results may let one think, the recycling options always imply more expenses than the OTC, at any time.

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Fuel Cycle Cash Flows per kWh (2 years late, base case)
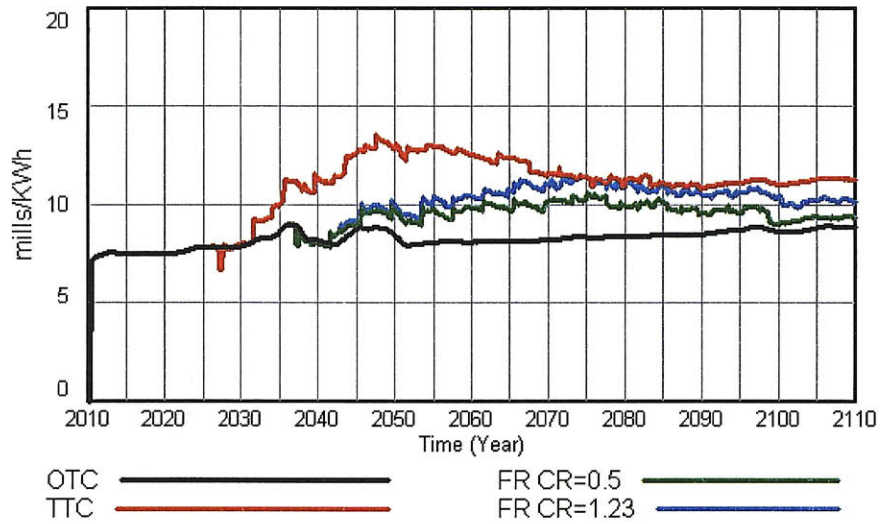
Figure 7.25 – Fuel Cycle Cash Flows per kWh

Figure 7.26 through 7.29 show for each scenario the three ways to calculate the fuel cycle costs. In general, accounting for present expenses in the future ("fuel cycle cost" vs. "fuel cycle cash flows" and "energy-based accounting scheme" vs. "waste-based accounting scheme") leads to higher values as expected. [37]

Fuel Cycle Costs (OTC, base case)
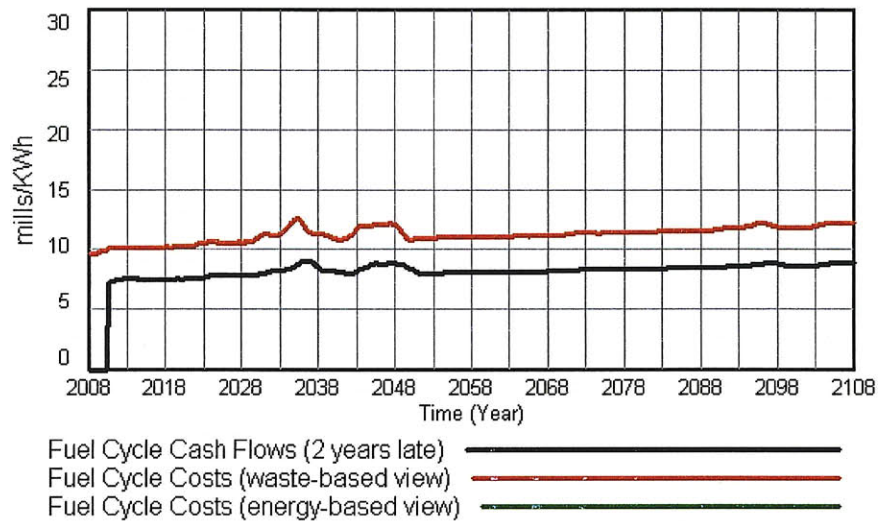
Figure 7.26 – Fuel cycle expenses/costs in the OTC scenario

[37] The mention "2 years late" just means that the graph must be shifted 2 years in the past (this is due to a technical issue occurring over the production of the graphs by VENSIM)

Fuel Cycle Costs (TTC, base case)

Fuel Cycle Cash Flows (2 years late) ——
Fuel Cycle Costs (waste-based view) ——
Fuel Cycle Costs (energy-based view) ——
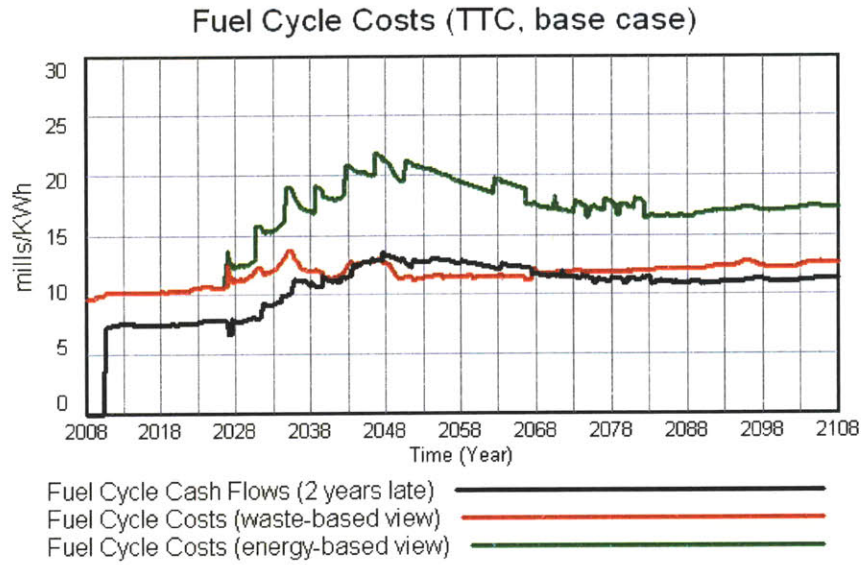
Figure 7.27 – Fuel cycle expenses/costs in the TTC scenario

Fuel Cycle Costs (FR CR=0.5, base case)

Fuel Cycle Cash Flows (2 years late) ——
Fuel Cycle Costs (waste-based view) ——
Fuel Cycle Costs (energy-based view) ——
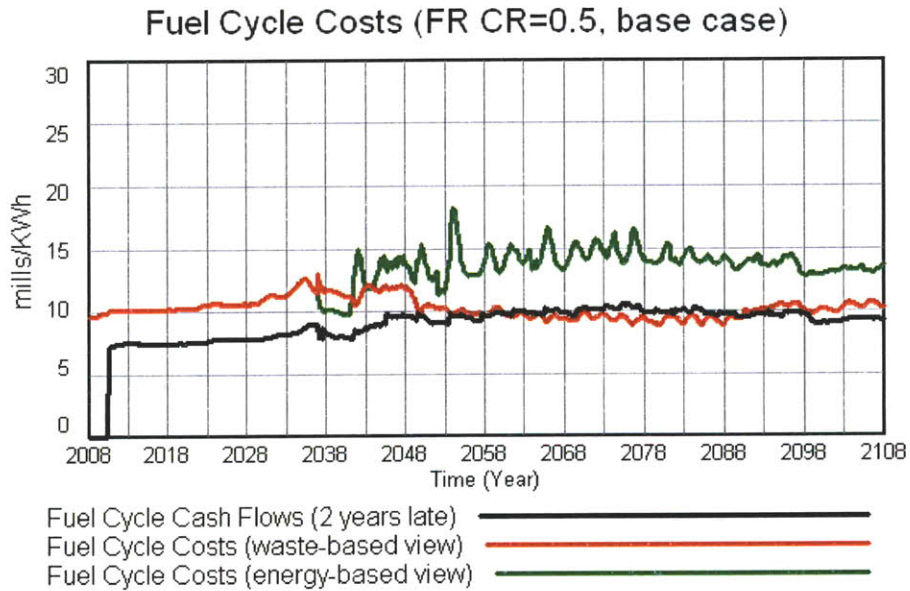
Figure 7.28 – Fuel cycle expenses/costs in the FR CR=0.5 case

Fuel Cycle Costs (FR CR=1.23, base case)

Fuel Cycle Cash Flows (2 years late) ————
Fuel Cycle Costs (waste-based view) ————
Fuel Cycle Costs (energy-based view) ————
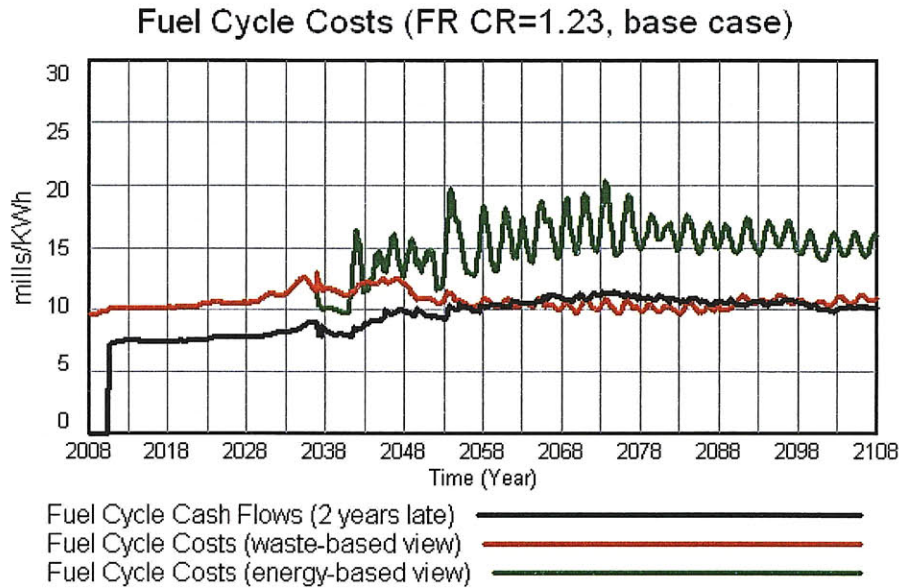
Figure 7.29 – Fuel cycle expenses/costs in the FR CR=1.23 case

Figure 7.30 shows the payments of the capital costs in the four scenarios. As MOX fuel is loaded into LWRs, there is no difference between the OTC scenario and the TTC scenario. As most of the existing reactors are already amortized (they were built more than 20 years ago), the average annual payment is at its lowest at the beginning of the simulation (about 20 mills/kWh). As new reactors are built, this value starts increasing, reaching a peak in 2050 at 75 mills/kWh in the LWR scenarios and 77 mills/kWh in the FR scenarios. In particular, the sharp increase in the late 2030s corresponds to the replacement of most of the existing reactors by new ones. After the 2050s, the average age of the LWR fleet decreases leading to a decrease in the average capital costs payments. This inverse effect is observed from the late 2060s. In order to meet the exponential growth of the electricity demand, more and more reactors are built making the fleet younger and younger, which result in higher average capital cost payments.

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Capital Costs payments - CF per kWh (2 years late, base case)
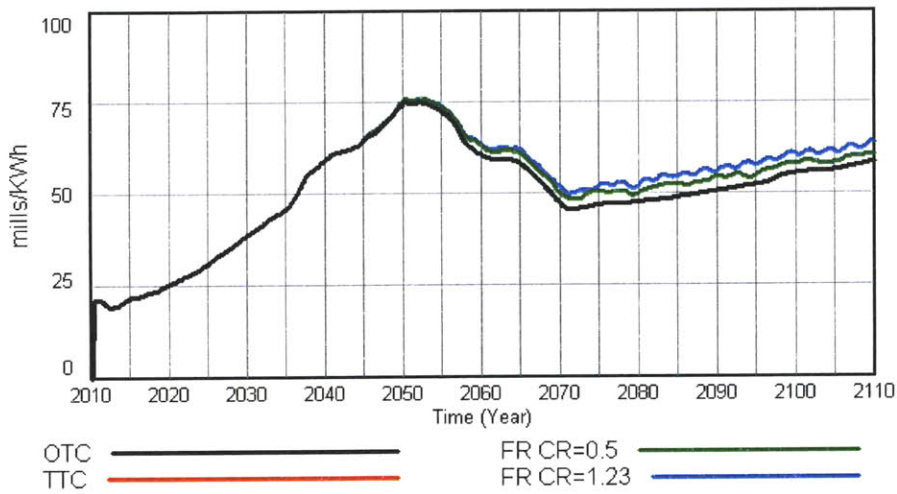
Figure 7.30 – Capital Costs Payments per kWh (base case)

Finally, Figure 7.31 shows the total cost of electricy, which is the addition of capital cost payments (see Figure 7.30), the fuel cycle payments (see Figure 7.24), and the O&M expenses[38] (see Figure 7.120). This cost starts at about 28 mills/kWh in all the scenarios; reaches a peak in 2050 at about 83 mills/kWh in the OTC scenario, 87 mills/kWh in the TTC scenario and 86 mills/kWh in the FR scenarios. In 2108, the system-averaged payments per kWh produced is about 66 mills/kWh in the OTC scenario, 69 mills/kWh in both MOX and FR CR=0.5 scenarios, and 72 mills/kWh in the FR CR=1.23 scenario.

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[38] The "O&M cash flows per kWh" and "O&M dynamic levelized costs" are exactly the sae, as O&M costs are constant, annual costs.

Total cash flows per kWh (2 years late, base case)

OTC	FR CR=0.5
MOX	FR CR=1.23
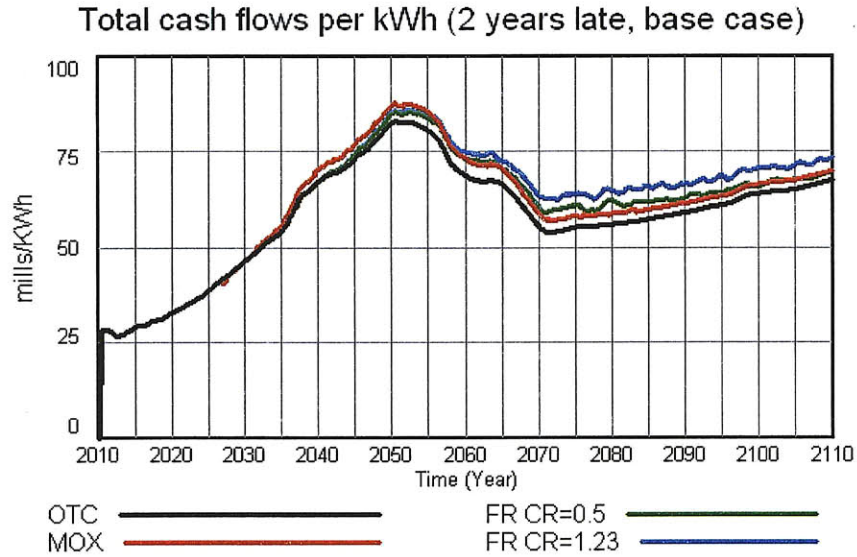
Figure 7.31 – Total cash flows per kWh (base case)

7.7.4 Total Cash Flows

It is also interesting to examine the total cash flows to provide insight about the future financial needs of the nuclear enterprise.

Figure 7.32 and 7.33 show the annual payments for the fuel cycle services in the four scenarios. $5.6bn are required in 2008 in all the scenarios. The reprocessing of spent fuel legacy and the fabrication of MOX fuel adds considerable expenses (+51% in 2058, from $20.4bn in the OTC scenario to $30.8bn in the TTC scenario). In 2108 (at equilibrium), the TTC scenario increases the fuel cycle expenses by 28% from $72.3bn to $92.5bn.

In the FR scenarios, the reprocessing of spent UO_2 fuel starts only in 2035. In 2108, the fuel cycle expenses are increased by 17.4% in the breeder scenario ($84.9bn) and 5.8% in the burner scenario ($76.5bn).

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Fuel Cycle Cash Flows (2 years late, base case)
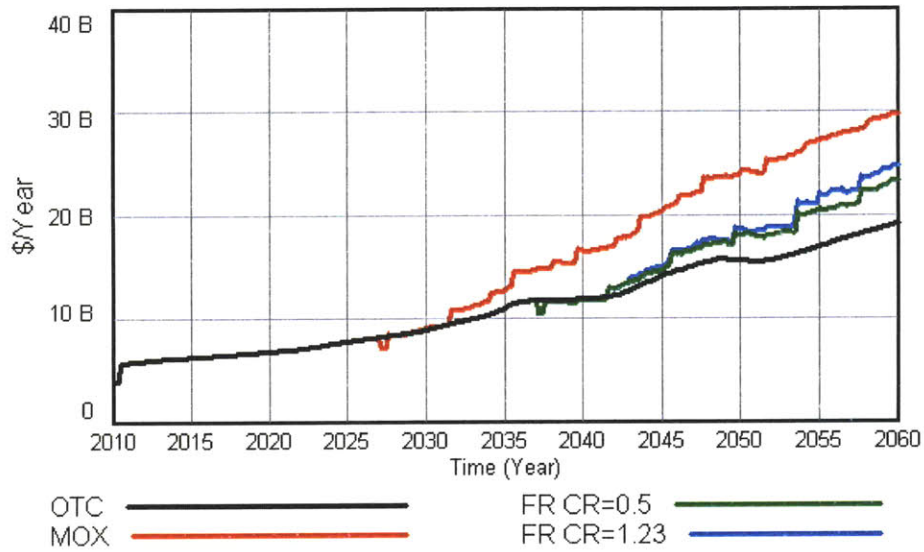
Figure 7.32 – Total cash flows for the fuel cycle services (base case) 2008-2058

Fuel Cycle Cash Flows (2 years late, base case)
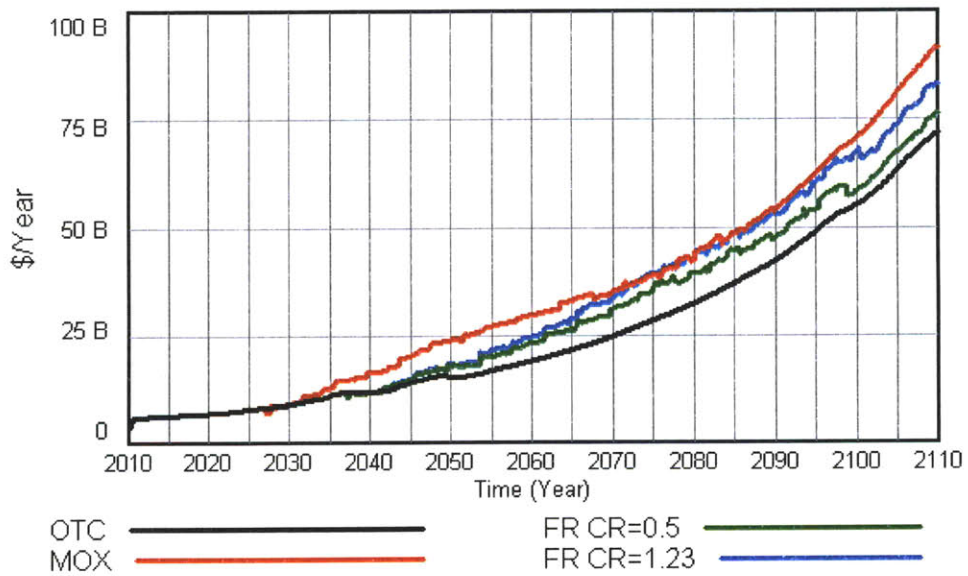
Figure 7.33 – Total cash flows for the fuel cycle services (base case) 2008-2108

Finally, Figure 7.34 and 7.35 show the total cash flows (including capital payments, O&M and fuel cycle expenses). Starting from $24.1bn in 2008 in all the scenarios, the

total cash flows reach in 2058 about $163.7bn in the OTC scenario, $174.2bn in the TTC scenario (+6.4%), $174.4bn in the burner scenario (+6.5%) and $181.5bn in the breeder scenario (+10.9%),

In 2108, these cash flows are more than tripled : $558bn in the OTC scenario, $578bn in the TTC scenario (+3.6%), $581bn in the burner scenario (+4.1%), and $617bn in the breeder scenario (+10.6%).

Total Cash Flows (2 years late, base case)
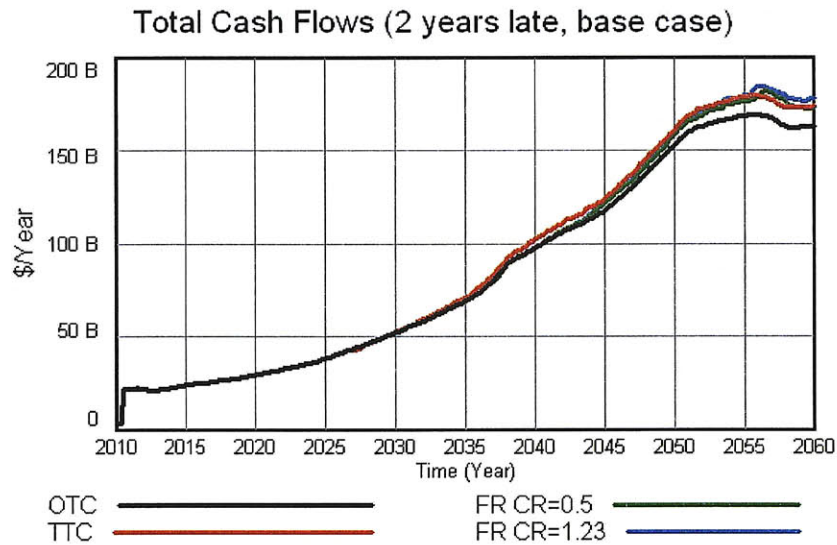
Figure 7.34 – Total cash flows (base case) 2008-2058

Total Cash Flows (2 years late, base case)
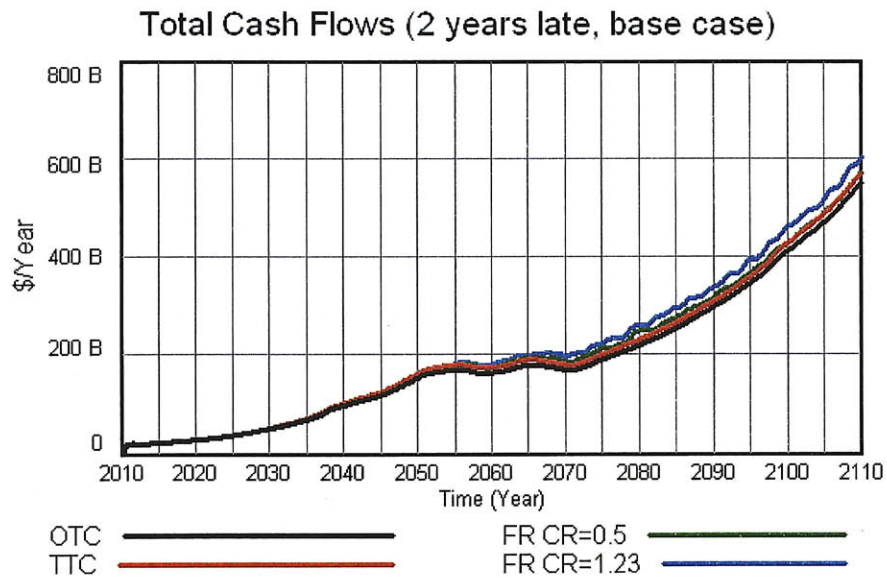
Figure 7.35 – Total cash flows (base case) 2008-2108

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Figures in Appendix A.7.3 show the composition of the total cash flows in the four scenarios.

7.8 Summary

This chapter presented the results for the base case scenarios, which all assume annual US nuclear energy growth rate of 2.5% from 2020 on, following a slower increase in capacity from 100 GWe in 2008 to 120 GWe in 2020. The initial spent UO_2 fuel legacy was assumed to be 56,800 tHM. Four fuel cycle options were studied: (1) Once-Through Cycle (OTC), (2) Twice-through cycle, using MOX (TTC) (3) Fast burners (FBu) with a conversion ratio of 0.5 (4) Fast breeders (FBr) with a conversion ratio of 1.23. It is assumed that spent fuel reprocessing starts in 2025 and commercial fast reactors are introduced in 2040. The other assumptions can be found in Section 7.2.

Results show that the share of FBr in the total installed capacity remains less than 50%, well below the value predicted by the "equilibrium" model presented in Chapter 3 (88%), despite the spent fuel legacy that makes available considerable amounts of TRU. This discrepancy is due to the needs of a start-up core, which requires 8.64 MTHM of TRU vs. an annual production rate of 0.22 MTHM/GWe/year. On the contrary, the share of FBu (more compact core) reaches its expected value of 23% in 2070 (30 years after its introduction); the share of MOX reaches its expected value of 10% in 2030, and even overshoots it until 2080 because of the spent fuel legacy.

As a result, the effect of the introduction of the FBr on the cumulative natural uranium consumption is below expectations: -35% in 2100 (vs. -24% for the FBu and -16% for the TTC). Assuming that the worldwide natural uranium market remains free-market dominated, these savings have a very modest effect on its price (and hence its availability), which, in the medium case, ends up at $131/kg in the FBr scenario vs. $137/kg in the OTC scenario. Even in a pessimistic framework about resource availaibity, the advantage remains modest: $202/kg in the FBr scenario vs. $229/kg in the OTC scenario.

All the recycling scenarios deplete the spent fuel legacy by the end of the 2080s (10 years before in the MOX case) and maintain stored fuel level below 130,000 tHM (The OTC scenario results in a final inventory of 600,000 tHM of spent fuel in 2100). This result requires the building of 8,000 to 10,000 tHM/year of thermal reprocessing capacity by 2060. This requirement remains stable in the FR scenarios until the end of the century but must meanwhile be doubled in the TTC scenario. As for the fast reactor fuel reprocessing capacity (metal fuel), it is recommended to start with a 250 tHM/year plant in the FBr scenario and a 100 tHM/year in the FBu scenario as a good trade-off between modularity and economies of scale. Then the unit capacity must be gradually increased. Because of the blankets, the FBr scenario requires much more reprocessing capacity than the FBu scenario: 6,000 tHM/year vs. 1,200 tHM/year in 2100.

The FBr scenario increases the total amount of TRU in the system by only 13% compared to the OTC in 2100. The FBu scenario ends up doing better than the TTC scenario at the end of the century, in terms of TRU reduction: -23% vs. -35% respectively, compared to the OTC. In the FBu scenarios, the remaining TRU are essentially located in the cores, cooling storage, reprocessing plants and fuel fabrication plants, whereas in the TTC scenario, the majority of TRU is contained in the spent MOX fuel, which is considered as waste.

All the recycling scenarios allow a dramatic reduction in the mass of HLWs compared to the OTC scenario, from a factor of 7 in the TTC scenario to a factor of 20 in the FR scenarios in 2100. However, if the heat and volume of the wastes are considered rather than masses (using the "densification factors"), these factors may go down to only 1.6 for the TTC scenario and 5.3 in the FR scenarios.

Finally, economic results show that at a given time t, the fuel cycle costs is strongly affected by the accounting scheme chosen. Because of the time value of money, the energy-based scheme always leads to higher costs than the waste-based scheme as soon as the recycling starts. However, the contrary may be observed if the waste-based scheme is used. However, it is important to be aware that the choice of the accounting scheme only affects the allocation of the expenses over time and not the expenses themselves. Actually, the present value of the total costs should ideally be independent from the accounting scheme chosen. An observation of the actual cash flows shows that the recycling schemes are always more expensive than the OTC.

The TTC scheme implies higher average fuel cycle costs than the FR schemes but this drawback is compensated for by the fact that the TTC does not require the building of new types of reactors with higher capital costs. The average cost of electricity is higher for high conversion ratios because the share of FRs (capital costs and O&M costs of which are higher than for the LWRs: 81 mills/kWh vs. 65 mills/kWh and 8.9 vs. 9.4 mills/kWh, respectively) in the total installed capacity is higher too.

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Chapter 8

8. Sensitivity Analysis: alternative assumptions

8.1 Introduction

This chapter analyzes the impact of the variation in the values assumed for some of the key parameters of the fuel cycle scenarios presented in the previous Chapter. In all the cases analyzed in this chapter, only one parameter is changed with respect to the base case.

Results for low-growth rate and high-growth rate scenarios are presented first. In a context of medium growth rate, sensitivity analyses are then performed on the conversion ratios of fast reactors; the minimum cooling time of the spent fuel; the introduction date of the thermal reprocessing in the FR scenarios; the introduction date of recycling technologies. Finally, results for a two-tier scenario (recycling of the spent MOX fuel in fast reactors) are analyzed.

8.2 Sensitivity to the nuclear energy demand growth rate

8.2.1 Low growth rate scenarios

We study in this section the same scenarios as in Chapter 7 except that the nuclear energy demand grows at a 1.0%/year rate from 2020 on.

8.2.1.1 Impact on infrastructure requirements

Figure 8.1 shows the total LWR-UO2 installed capacity in the four scenarios, while Figure 8.2 shows the capacity provided by the MOX fuel irradiation in LWRs (twice-through scenario), the fast burners and the fast breeders. The number of breeders significantly differs from the number of burners only from 2084 on. We observe in both FR scenarios stabilization of the number of FR after a few decades of progression. This stabilization corresponds to the depletion of the spent UO_2 fuel legacy (see Figure 8.6).

LWR-UO2 - Installed Capacities (1.0% growth rate)
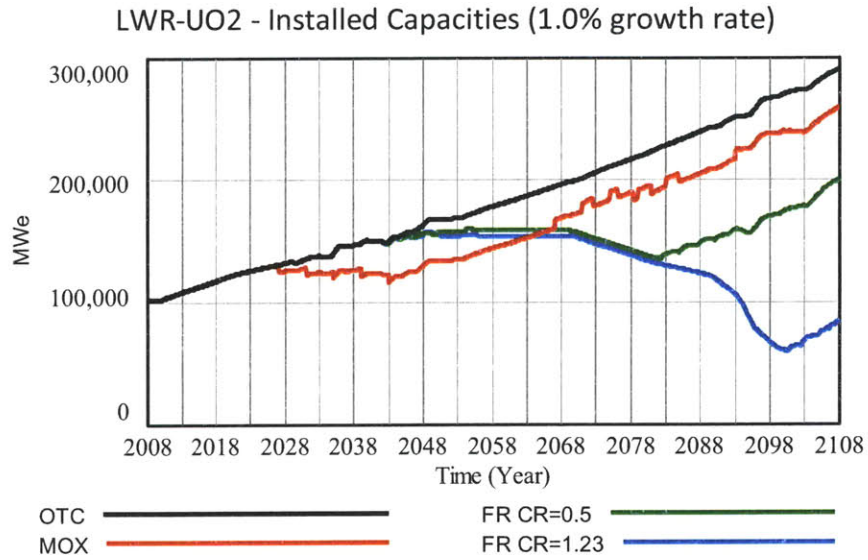
Figure 8.1 – LWR-UO2 installed capacity (1.0% growth rate)

Advanced Technologies - Installed Capacities (1.0% growth rate)
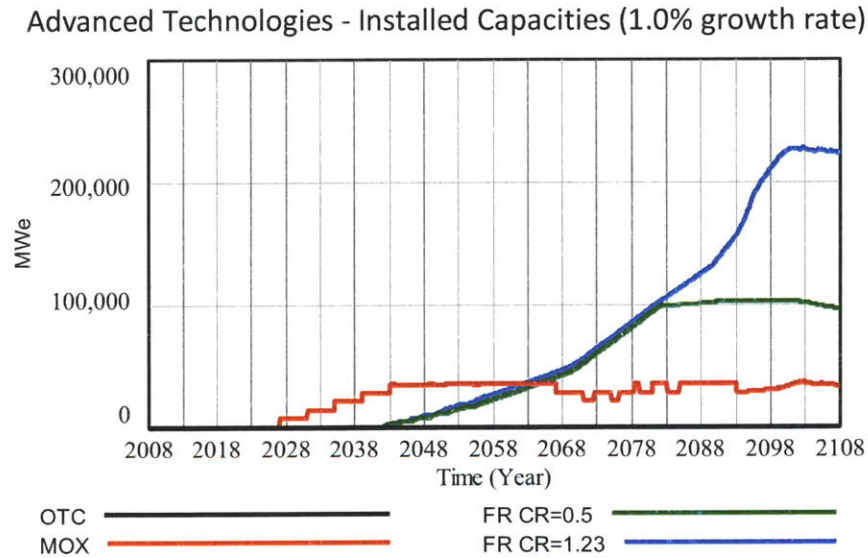
Figure 8.2 – Recycling technologies: installed capacity (1.0% growth rate)

Figure 8.3 shows the "years of FR fresh fuel available", which is the inventory of TRU available for FR fresh fuel fabrication divided by the annual consumption of the FR fleet. Unlike in the medium growth rate case, we here observe an accumulation of fissile materials over several decades. Until the early 2070s (resp. early 2080s), there are enough

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fissile materials to fuel the fast burner (resp. fast breeder) fleet for more than 5 years. These inventories of FR fresh fuel are eventually depleted, but FRs could have been deployed at a faster pace. Observation of Figure 8.1 reveals that the limiting constraint is the total energy demand. Until the late 2060s, most of the energy demand is met by the LWRs built before introduction of the FRs in 2040. FRs can only fill up the difference between the current electricity demand and the supply brought by the existing LWR fleet (CAFCA does not allow the shutdown of LWRs that are less than 60 years old). In 2068, some LWRs start being decommissioned, giving more room to FRs, of which building rate is accelerated (see Figure 8.2). This phenomenon is specific to low-growth scenarios.
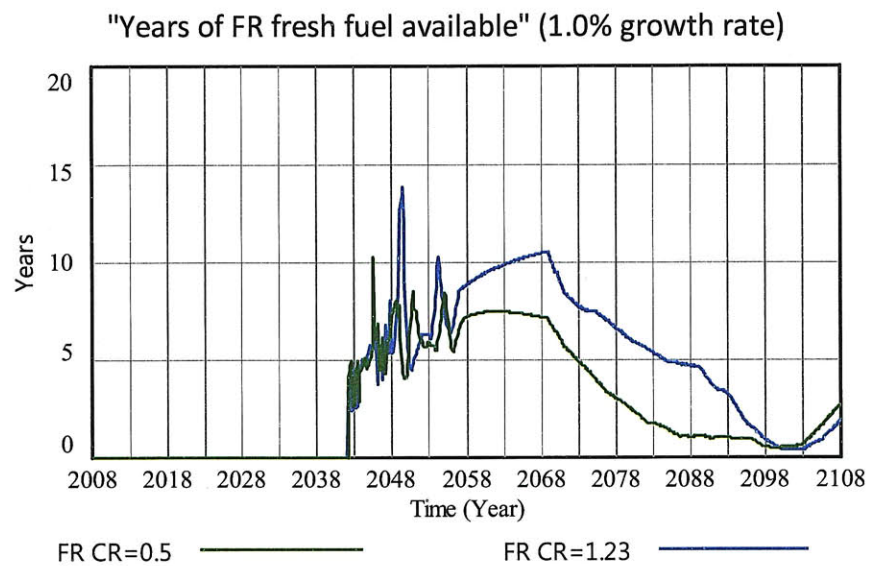
Figure 8.3 – Years of FR fresh fuel available (1.0% growth rate)

As we did in the base case, let us compare the results predicted by the simple model (see Chapter 3) and the outcomes of the dynamic simulations. Figure 8.4 shows for each of the three recycling scenarios the fraction of the advanced technology in the total installed capacity along with the values predicted by the simple model. As in the base case, the MOX fraction reaches its predicted value (11.38%) as the spent UO_2 fuel legacy is near depletion (see Figure 8.6), i.e. in the 2070s.

In the fast burner scenario, the constraint on the building of fast reactors that we observed limits its fraction below the predicted level (30.44%) until 2074. Then the recycling of the spent fuel legacy allows a slight overshoot (peak at 42% in 2082), which gradually

disappears. The spent fuel legacy is not totally depleted in 2108 (see Figure 8.6), so the fraction of FR CR=0.5 should keep decreasing after 2108, reaching an equilibrium value that is expected to be lower than the number provided by the simple model, because of the loading of start-up cores.

As for the fast breeder scenario, there is theoretically no equilibrium ratio for a 1.0% growth rate. We expect the breeders to replace all LWRs and, in the presence of a constraint on the total energy demand, fissile materials should even be accumulated. Despite the presence of a spent UO$_2$ fuel legacy, the predicted scenario does not happen because of the start-up cores' requirements, which were neglected in the simple model. The maximum breeder fraction observed is 79% in 2101. These simulations were done without any constraint on the reprocessing capacity.
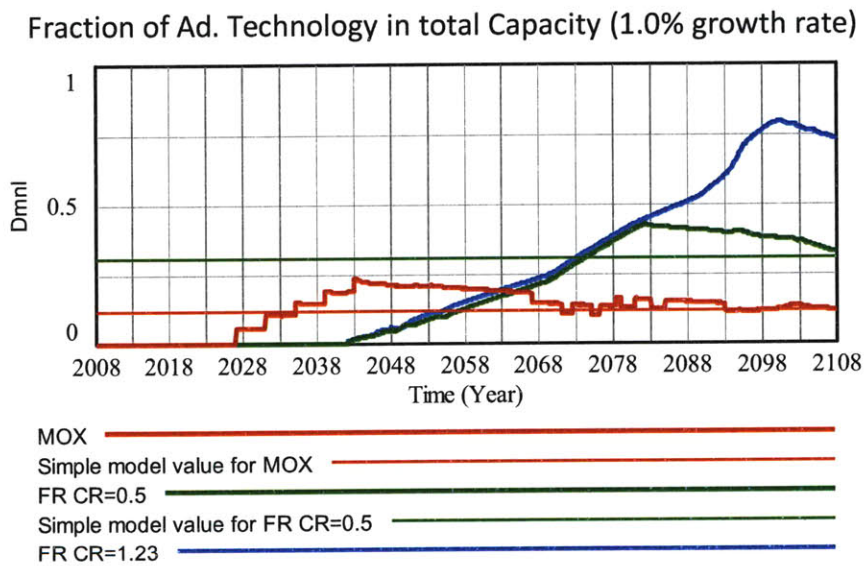
Figure 8.4 – Years of FR fresh fuel available (1.0% growth rate)

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Finally, Table 8.1 shows the installed capacity for the four scenarios in 2050 and 2100.

Table 8.1 - LWR-UO₂/MOX/FR installed capacities in 2050 and 2100 (1.0% growth rate)

	Date	OTC	MOX	FR CR=0.5	FR CR=1.23
LWR-UO2 installed capacity (GWe)	2050	166	132	154	154
	2100	269	238	172	60
MOX/FR installed capacity (GWe)	2050	X	34	11	11
	2100	X	31	102	223

Figure 8.5 shows the thermal reprocessing capacity for the three recycling scenarios, stabilized between 4,000 and 6,000 tHM/year starting in the 2050s. Figure 8.6 shows the spent UO_2 fuel in interim storage (and repository in the OTC scenario). In the recycling scenarios, reprocessing plants gradually deplete the inventory of spent fuel, which reaches a peak at 90,000 tHM in the TTC scenario and 117,000 tHM in the FR scenarios. In the OTC scenario, the total amount of spent UO_2 fuel reaches 394,000 tHM in 2108. One may be surprised that the spent UO_2 fuel legacy is not depleted by 2108 in the FR scenarios. Indeed, the constraint on the building rate of reprocessing plants is clearly not limiting as new reprocessing plants could be built from the 2060s on (see Figure 8.5). This is actually due to the constraint on the loading factor: if more reprocessing plants were built, we would end up, once the legacy is depleted, with an overcapacity that is economically undesirable. The impact of this requirement on the result is studied in 8.2.1.5.

thermal reprocessing capacity (1.0% growth rate)
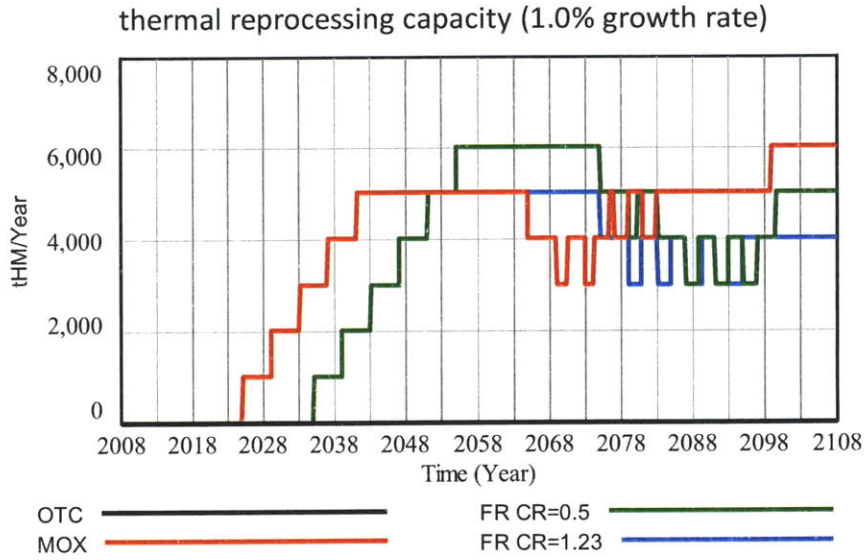
Figure 8.5 – Thermal reprocessing capacity (1.0% growth rate)

Spent UO2 fuel in interim storage and repository (1.0% growth rate)
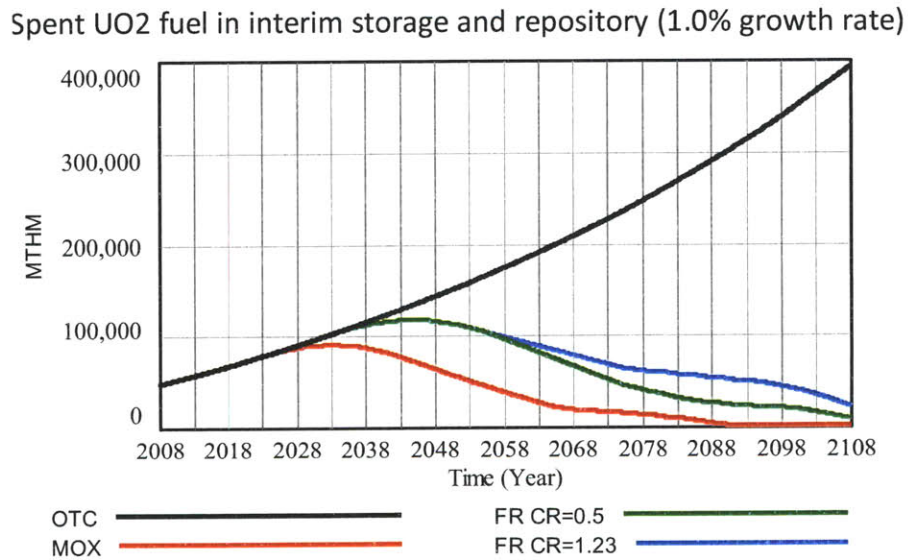
Figure 8.6 – Spent UO₂ fuel in interim storage and repository (1.0% growth rate)

Figure 8.7 shows the fast reprocessing capacity in both FR scenarios while Figure 8.8 shows their loading factors[39]. Recall that the single unit capacity assumed in the breeder scenario is 500 tHM/year and is 200 tHM/year in the burner scenario. In both FR

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[39] Erratum: the y-axis unit on Figure 8.8 is "dimensionless" and not MWe.

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scenarios, the second plant is built about 20 years after the first one. Like the base case, it would have been more economical to start with a 200-tHM/year plant in the breeder case and a 100-tHM/year plant in the burner case (loading factor lower than 80% over more than 10 years). In 2108, there are 4 fast reprocessing plants (total capacity of 800 tHM/year) in the burner case vs. 7 in the breeder case (total capacity of 3,500 tHM/year).
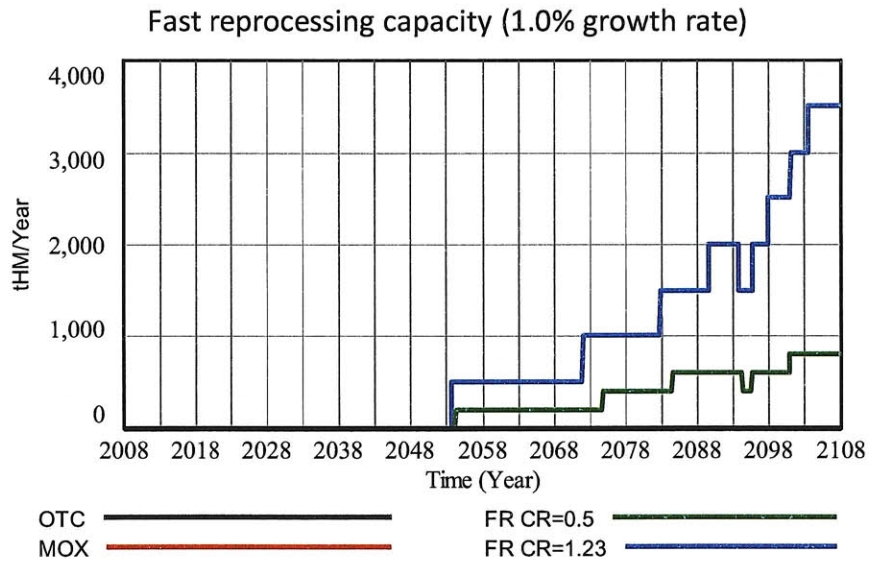
Figure 8.7 – Fast reprocessing capacity (1.0% growth rate)
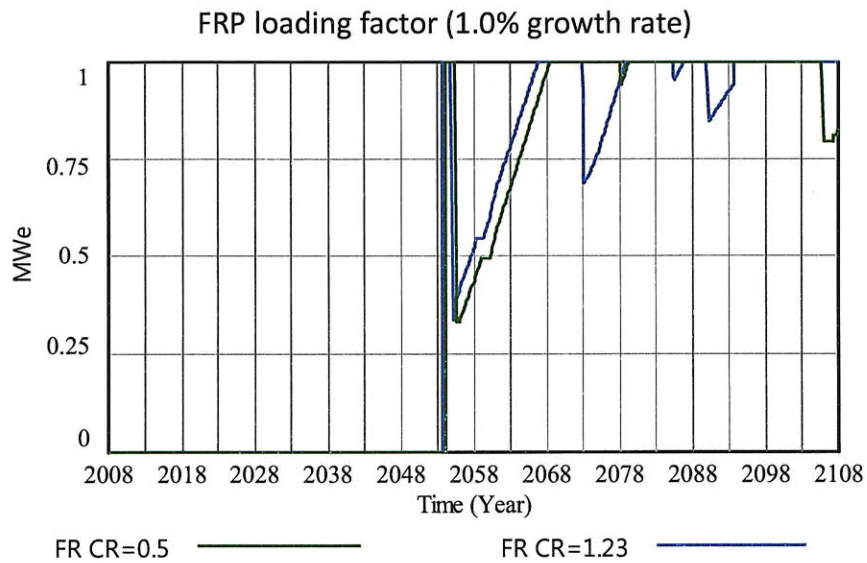
Figure 8.8 – Fast reprocessing plants mass loading factor (1.0% growth rate)

8.2.1.2 Impact on U.S. natural uranium consumption

Figure 8.9 shows the impact of the introduction of advanced technologies on the uranium utilization rate (as in every scenario in this study, the uranium recovered from spent fuel reprocessing is recycled after re-enrichment). As expected, the greater the fraction of advanced technologies in total installed capacity, the more the natural uranium consumption is reduced. The MOX option has its greatest impact in the late 2030s (natural U consumption reduced to 18,500 tHM/year in 2040), as the MOX utilization is at its maximum (in percentage). The burner scenario maintains about this level, which is the best that the TTC scenario ever does, until 2082; the breeder scenario does even better until the end of the simulation.
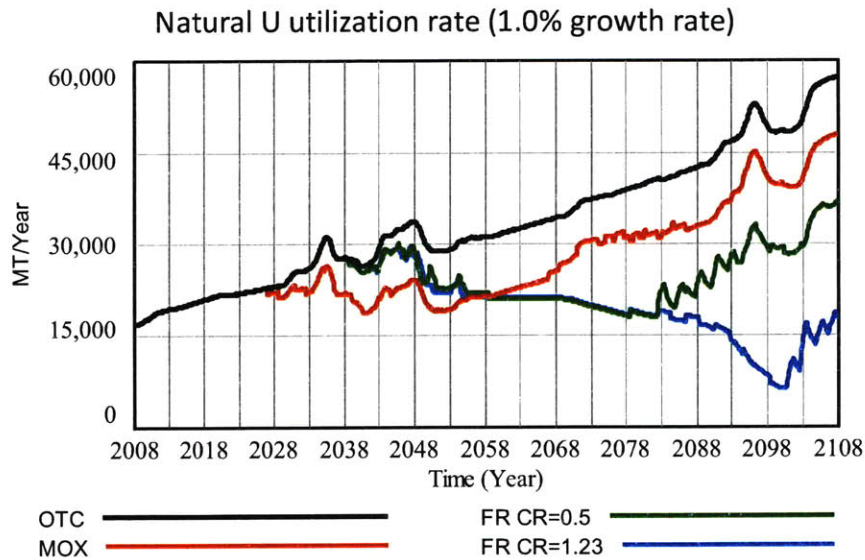
Figure 8.9 – Natural U utilization rate (1.0% growth rate)

Table 8.2 shows the natural uranium utilization rate in 2050 and 2100 in the four scenarios.

Table 8.2 - Natural uranium utilization rate (1.0% growth rate)

Date	OTC	MOX	FR CR=0.5	FR CR=1.23
2050	~ 29,000	~ 19,000 (-34.5%)	~ 24,000 (-17.0%)	~ 23,000 (-20.5%)
2100	~ 48,000	~ 40,000 (-16.5%)	~ 29,000 (-39.5%)	~ 6,000 (-88.0%)

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Figure 8.10 shows the resulting cumulative natural uranium consumption in the four scenarios. The FR scenarios catch up with the TTC scenario about 40 years after the introduction of the first fast reactor.

Cumulative natural U consumption (1.0% growth rate)
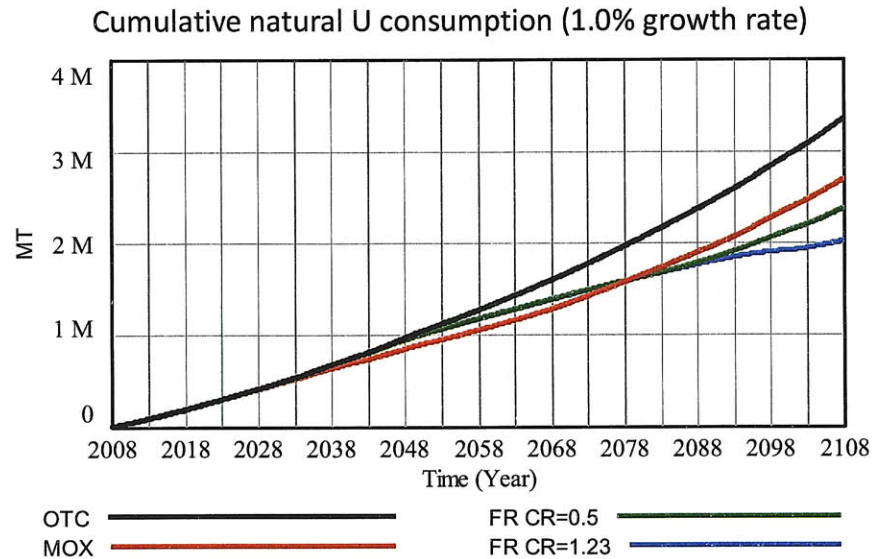
Figure 8.10 – Cumulative natural U consumption (1.0% growth rate)

Table 8.3 shows the cumulative natural uranium consumption in 2050 and 2100.

Table 8.3 - Cumulative natural U consumption (1.0% growth rate)

Scenario	OTC	MOX	FR CR=0.5	FR CR=1.23
2050	1.030 Mtons	0.893 Mtons (-13.3%)	0.995 Mtons (-3.4%)	0.996 Mtons (-3.3%)
2100	2.929 Mtons	2.351 Mtons (-19.7%)	2.114 Mtons (-27.8%)	1.912 Mtons (-34.7%)

8.2.1.3 Impact on TRU inventories

Figure 8.11 shows the total mass of TRU in the entire system (plants, reactor cores, storages, and repository). In terms of TRU reduction, the fast burner scenario catches up with the TTC scenario more than 50 years after the introduction of the first fast reactor. In 2108, the TRU in the entire system amounts to 5,755 tHM in the OTC scenario, 4,236 tHM in the TTC scenario, 3,381 tHM in the fast burner scenario and 6,421 tHM in the fast breeder scenario.

TRU: total mass in the system (1.0% growth rate)
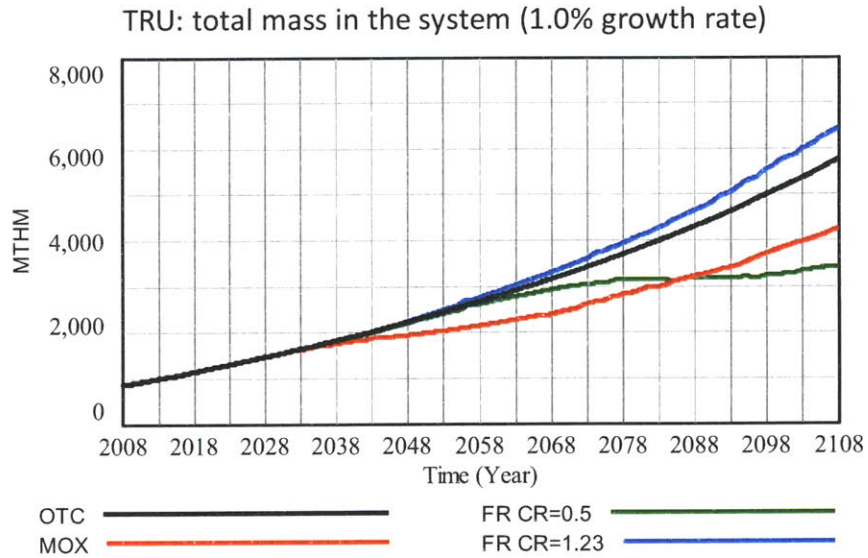
Figure 8.11 – TRU: total mass in the system (1.0% growth rate)

8.2.1.4 Impact on HLW inventories

Figure 8.12 shows the HLW in the repository, assuming that HLW is sent to repository 25 years after it is generated, and that the first repository opens in 2028. In 2108, there are 300,000 tHM of HLW (spent UO_2 fuel) in the OTC scenario, 43,000 tHM in the TTC scenario (Mix of Fission products and minor actinides + spent MOX fuel) and 13,000 tHM in both FR scenarios (fission products + TRU losses).

HLW in repository (high growth rate)
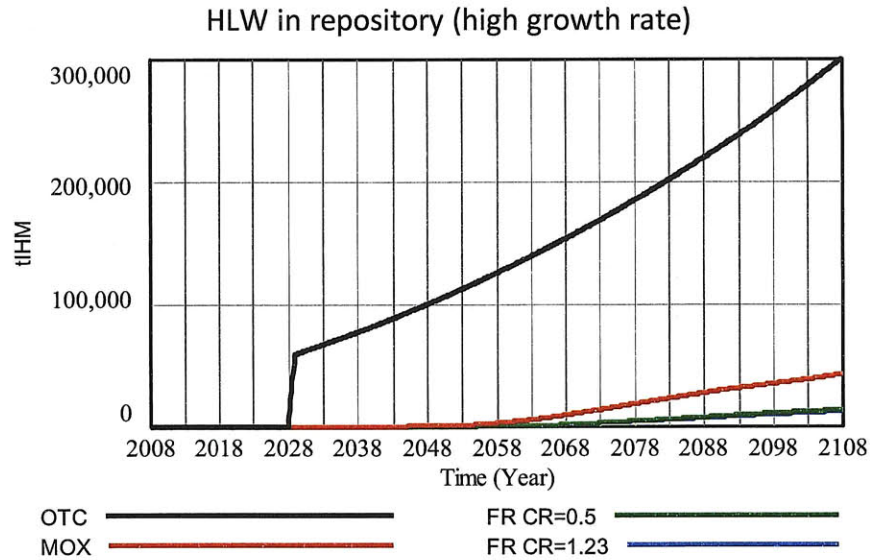
Figure 8.12 – HLW in repository (1.0% growth rate)

Finally, Figure 8.13 shows the TRU content in the HLW (interim storage + repository). The high TRU content in the spent MOX fuel reduces the apparent advantage of the twice-through scenarios in terms of waste reduction. In 2108, there are 5,035 tHM of TRU in HLWs in the OTC scenario vs. 3,234 tHM in the TTC scenario, 9 tHM in the burner scenario and 10 tHM in the breeder scenario.

TRU in wastes (1.0% growth rate)
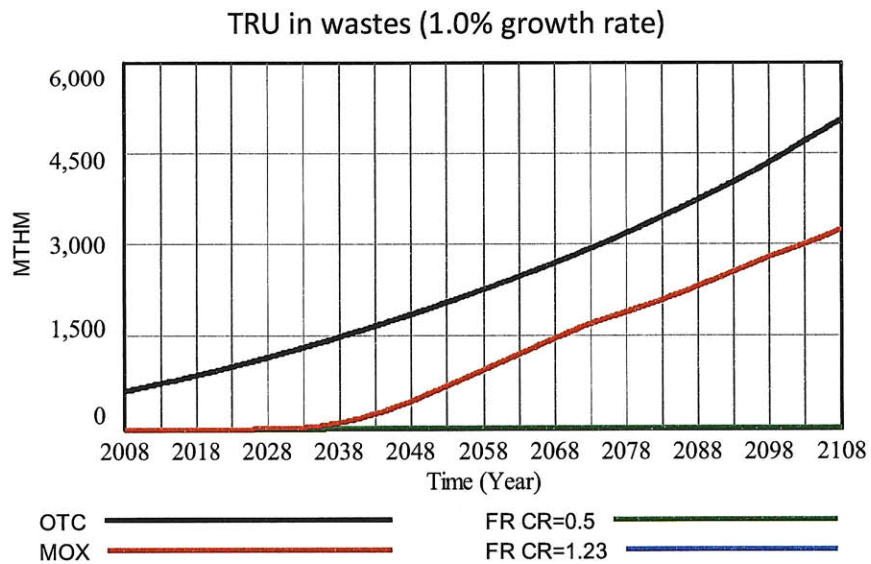
Figure 8.13 – TRU in wastes (1.0% growth rate)

8.2.1.5 Impact of the minimum loading factor value on the breeder scenario

It was noticed in 8.2.1.1 that the constraint on the loading factor of the reprocessing plants (minimum of 80%) delayed the reprocessing of some of the spent UO_2 fuel available.

The fast breeder scenario was run for the low-growth case with no constraint on the loading factor (the cap on the industrial capacity still applied though). Figure 8.14 shows the fraction of fast breeders in the total installed capacity for both minimum loading factors (MLF=80% and 0). One observe that lowering the minimum loading factor has a modest effect until 2108 (slight increase), because the deployment of FRs is limited by the total energy demand, mostly met by existing LWRs, as noted in 8.2.1.1. However, results diverge from each other in 2101: instead of starting decreasing as in the previous case, the share of breeders keeps increasing until it reaches 100% in 2113.
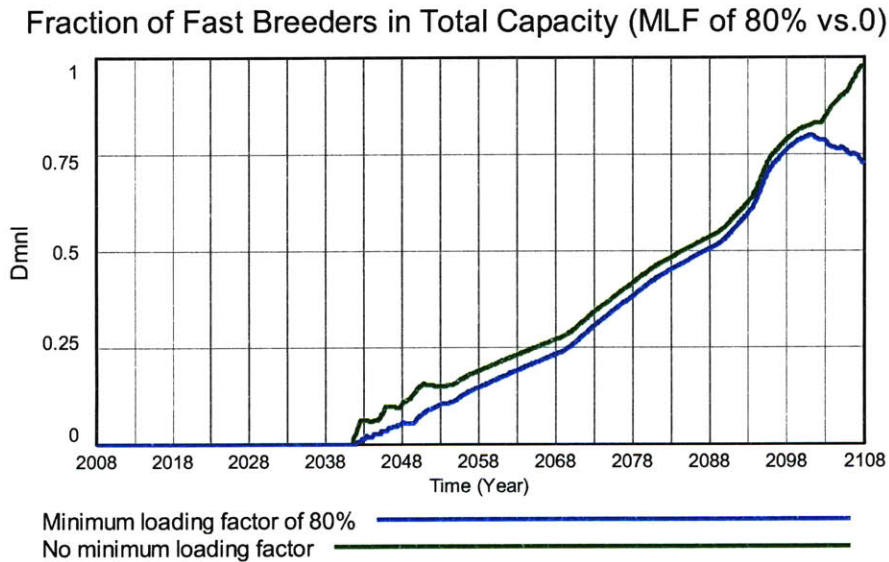
Figure 8.14 – Fraction of fast breeders in the total installed capacity (MLF=80% vs. 0)

The price to pay for this late result is that, in the absence of LWRs, thermal reprocessing plants become idle. For both scenarios, Figure 8.15 shows the thermal reprocessing capacity while Figure 8.16 shows their mass loading factors. It can be seen that without any constraint on the loading factor, one more plant is built in 2056 but, as a result of the

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replacement of the entire LWR fleet with FRs, all the thermal reprocessing plants become idle in 2109; their ages are 16, 19 and 25 then.

thermal reprocessing capacity (high growth rate)

Minimum Loading Factor of 80%
No Minimum Loading Factor
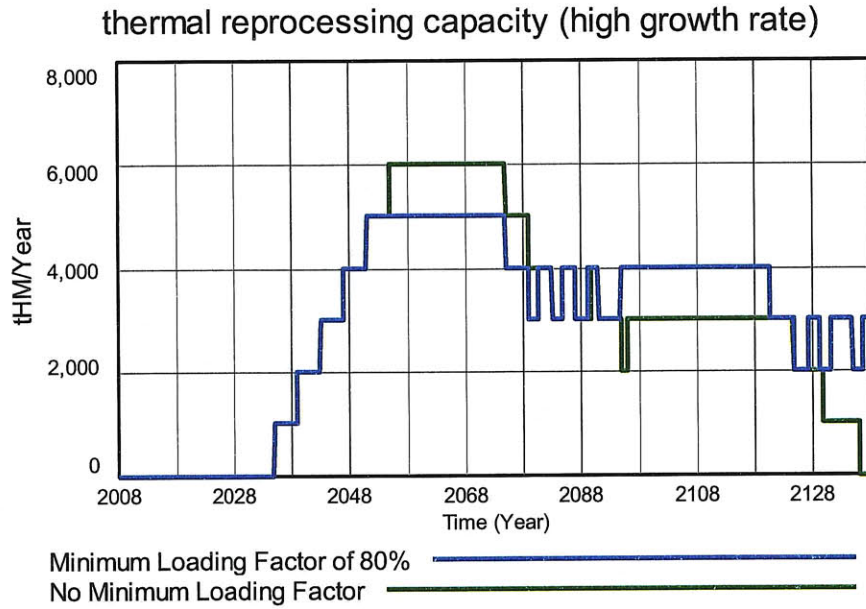
Figure 8.15 – Thermal Reprocessing Capacity (MLF=80% vs. 0)

ThRP loading factor (MLF of 80% vs. 0)

Minimum Loading Factor of 80%
No minimum Loading Factor
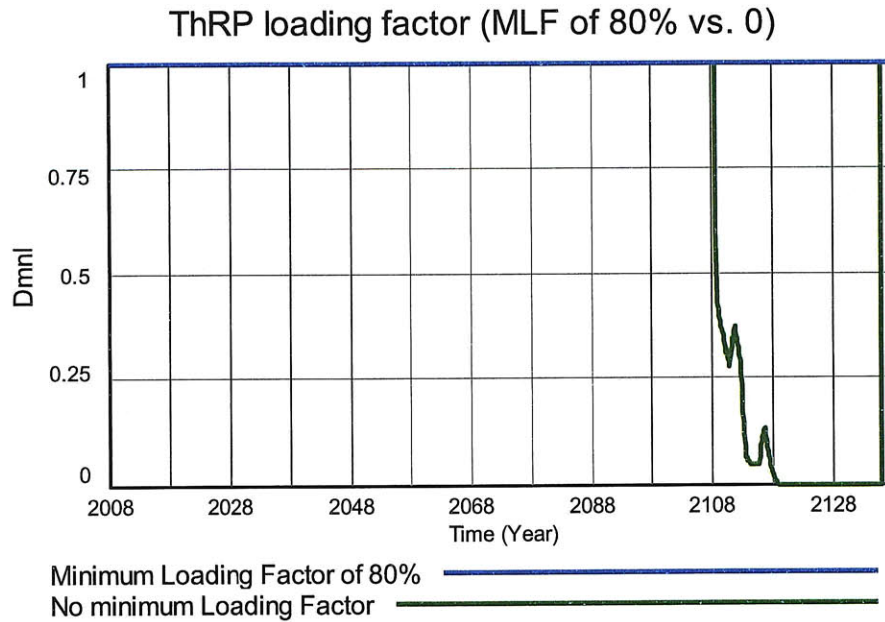
Figure 8.16 – ThRP Mass Loading Factors (MLF=80% vs. 0)

As a conclusion, the necessity to amortize the reprocessing plants capital costs slows down the deployment of the fast breeders, preventing the total phase out of the thermal reactors within 100 years. If the phase out of the LWRs were to be a political priority, reprocessing services prices would have to be increased in order to compensate for the premature shutdown of some of the plants. Note that this observation is specific to the low-growth rate case.

8.2.2 High growth rate scenario

In order to be realistic, the 4.0% growth rate is not maintained over the entire period. Thus, after 30 years of 4.0%/year growth (from 2020 to 2050), the growth rate is slowed down to 2.5%/year until the total energy production of 1000 GWe-yr per year is reached (which happens in 2088), after what the nuclear energy demand is stabilized. This unsteady growth profile certainly makes analysis of the results less conclusive than in the previous cases.

8.2.2.1 Impact on infrastructure requirements

Figure 8.17 shows the total LWR-UO$_2$ installed capacity in the four scenarios, while Figure 8.18 shows the capacity provided by the MOX fuel irradiation in LWRs (twice-through scenario), the fast burners and the fast breeders.

As expected by the simple model, the lower the growth rate, the greater the fraction of recycling technologies in the total installed capacities. Thus, as the growth rate goes down from 2.5% to 0, the fraction of MOX and fast reactors increase (see Figure 8.17 from 2088 on).

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LWR-UO2 - Installed Capacities (high growth rate)
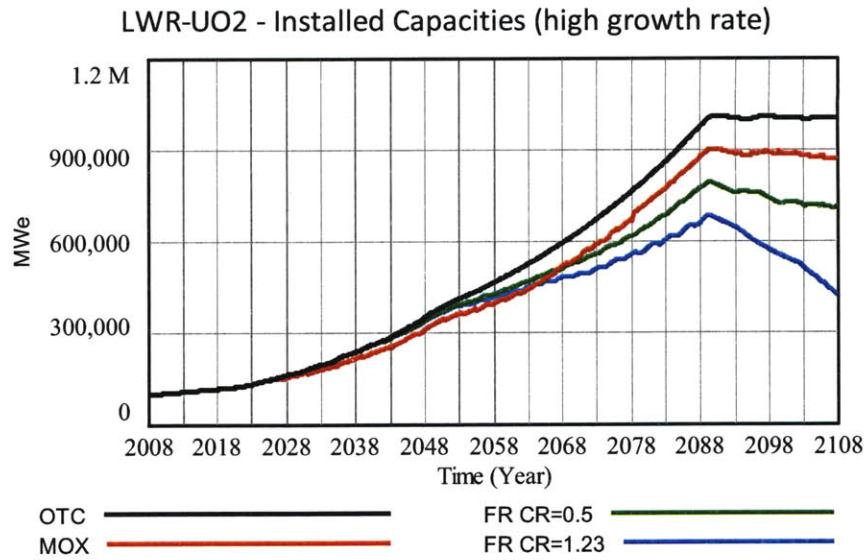
Figure 8.17 – LWR-UO2 installed capacity (high growth rate)

Advanced Technologies - Installed Capacities (high growth rate)
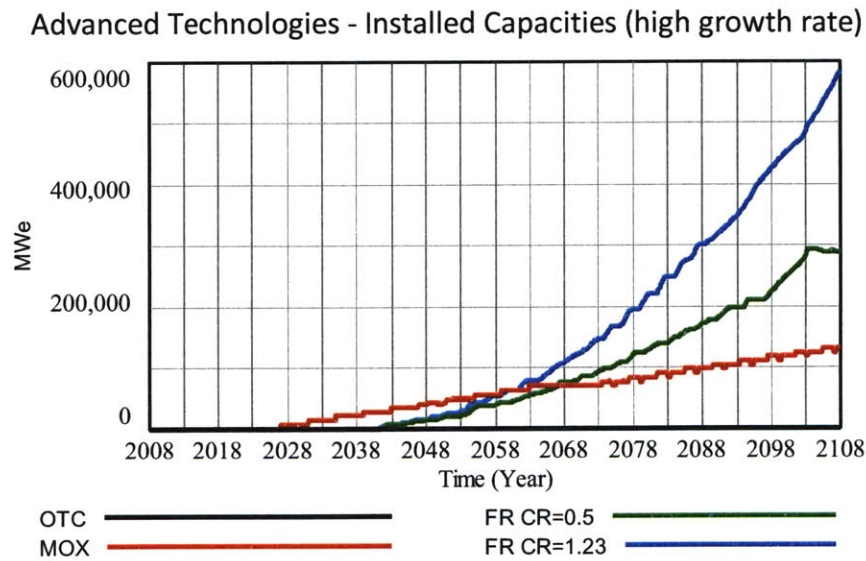
Figure 8.18 – Advanced Technology installed capacity (high growth rate)

Table 8.4 shows the installed capacity for each technology in 2050, 2089 and 2100. Note that in this higher growth scenario, the fast reactors are unable to reach even a 50% share of electricity generation. Yet, breeders are needed in this case to limit the depletion of the uranium resource.

Table 8.4 - LWR-UO2/MOX/FR installed capacities in 2050, 2080 and 2100 (high growth rate)

	Date	OTC	MOX	FR CR=0.5	FR CR=1.23
LWR-UO2 installed capacity (GWe)	2050	377	336	362	360
	2089	993	897	834	712
	2100	1,001	884	769	577
MOX/FR installed capacity (GWe)	2050	X	41	16	20
	2089	X	96	174	305
	2100	X	117	245	447

Figure 8.19 shows the thermal reprocessing capacity while Figure 8.20 shows the amount of spent UO_2 fuel in interim storage. Recall that the unit capacity is 1000 tHM/year. The loading factor remains 100% in the four scenarios, over the entire simulation. From 2070 on, the industrial capacity is doubled from 4 years/plant to 2 years/plant. However, the first plants start being decommissioned in 2065 in the TTC scenario and 2075 in the FR scenarios. The replacement of these plants absorbs the additional industrial capacity. As a result, the thermal reprocessing capacity increases at a steady state over the entire simulation. In 2108, there are 20 plants in the TTC scenario, 19 plants in the fast burner scenario and 15 plants in the fast breeder scenario. Actually, a unit capacity of 2,000 tHM/year or even more could be assumed in this scenario, thus reducing the reprocessing costs through economies of scale.

As can be seen in Figure 8.20, the reprocessing capacity is not sufficient to deplete the spent UO2 fuel inventory in interim storage. Thus, unlike in the base case and the low-growth case, the industrial capacity is a limiting constraint in this scenario.

In the OTC scenario, 985,000 tHM of spent UO_2 fuel are accumulated by 2108, which is 14 times the legal capacity of the Yucca Mountain repository. The inventory is maintained below 93,000 tHM in the TTC scenario and 166,000 tHM in the FR scenarios.

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thermal reprocessing capacity (high growth rate)
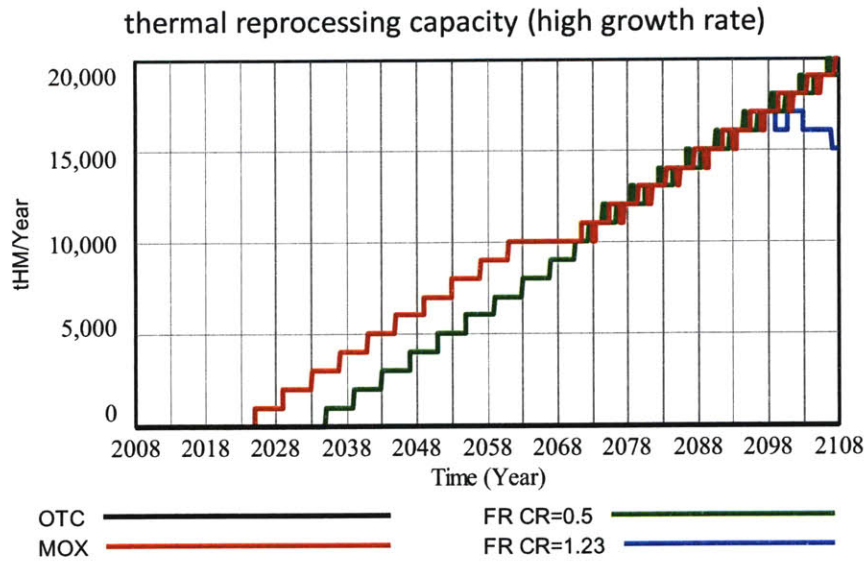
Figure 8.19 – Thermal reprocessing capacity (high growth rate)

Spent UO2 fuel in interim storage and repository (high growth rate)
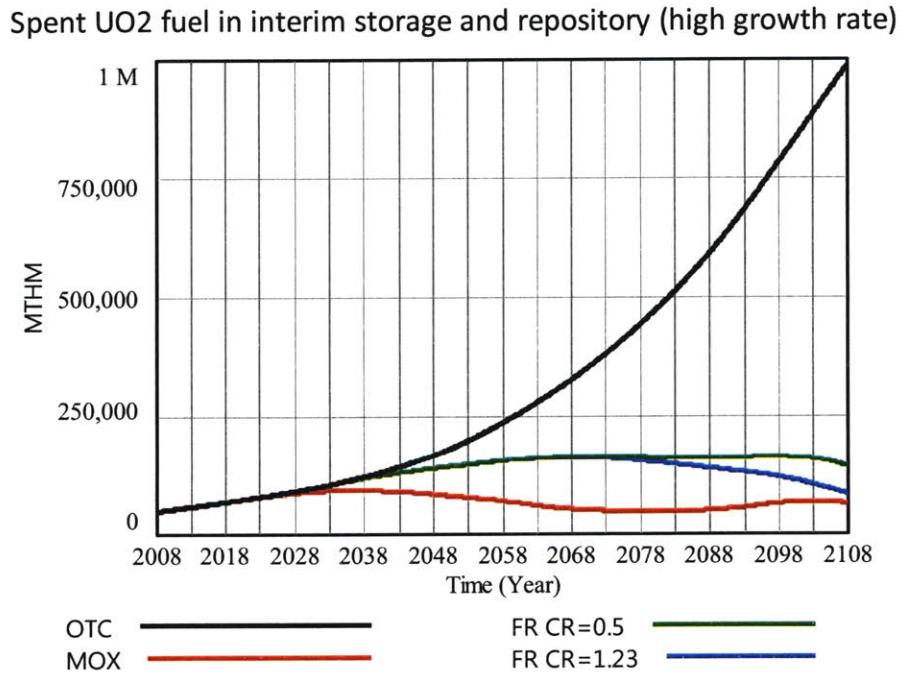
Figure 8.20 –Spent UO2 fuel in interim storage and repository (high growth rate)

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Figure 8.21 shows the fast reprocessing capacity in the two FR scenarios while Figure 8.22[40] shows their loading factors. Recall that the unit capacity is 500 tHM/year in the breeder case and 200 tHM/year in the burner case. As expected, requirements are much higher in the breeder case not only because the breeders have a higher annual mass loading than the burners, but also because there are also twice as many breeders as burners (see Figure 8.18). As in the previous cases, the second reprocessing plant is built at least 10 years after the first one, which is in overcapacity. It would be more economical to build a first plant of lower capacity: 250 tHM/year in the breeder scenario and 100 tHM/year in the burner case. Conversely, we could benefit from economies of scale from the 2070s by building 1000 tHM/year plants in the breeder scenario, and 500 tHM/year plants in the burner scenario. In 2108, the total fast reprocessing capacity is 7,500 tHM/year in the breeder case vs. 1,600 tHM/year in the burner case. There is virtually no accumulation of spent FR fuel in interim storage, which means that the industrial limit on the FRP building is not constraining.
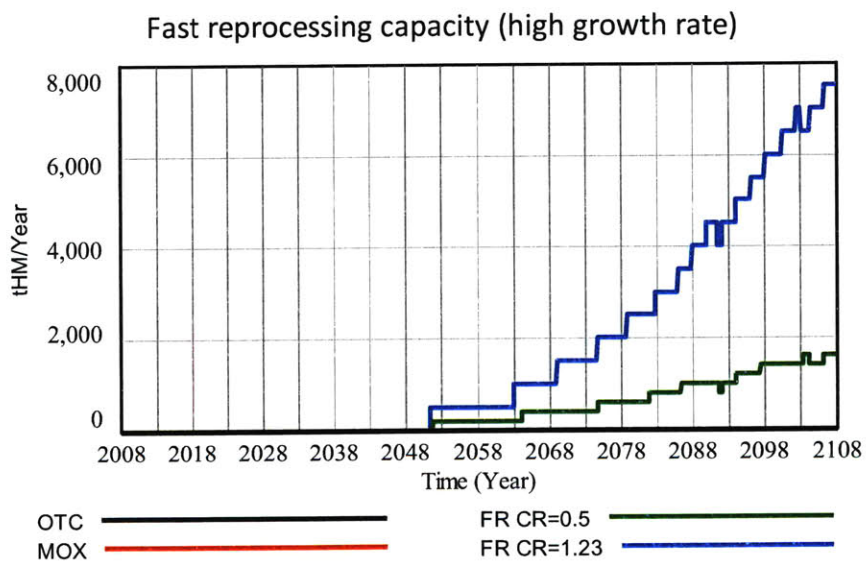
Figure 8.21 – Fast reprocessing capacity (high growth rate)

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[40] Erratum: the y-axis unit on Figure 8.22 is "dimensionless" and not MWe

FRP loading factor (high growth rate)
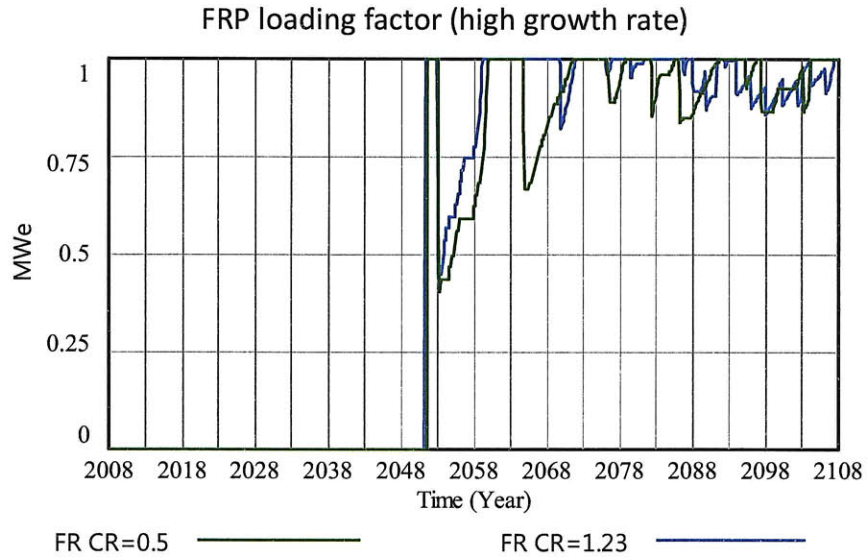
Figure 8.22 – Fast Reprocessing Plants mass loading factor (high growth rate)

8.2.2.2 Impact on the U.S. natural uranium consumption

Figure 8.23 shows the natural uranium utilization rate. Note that the breeder has the largest effect on the demand for U, which still grows to a peak in 2088.

Natural U utilization rate (high growth rate)
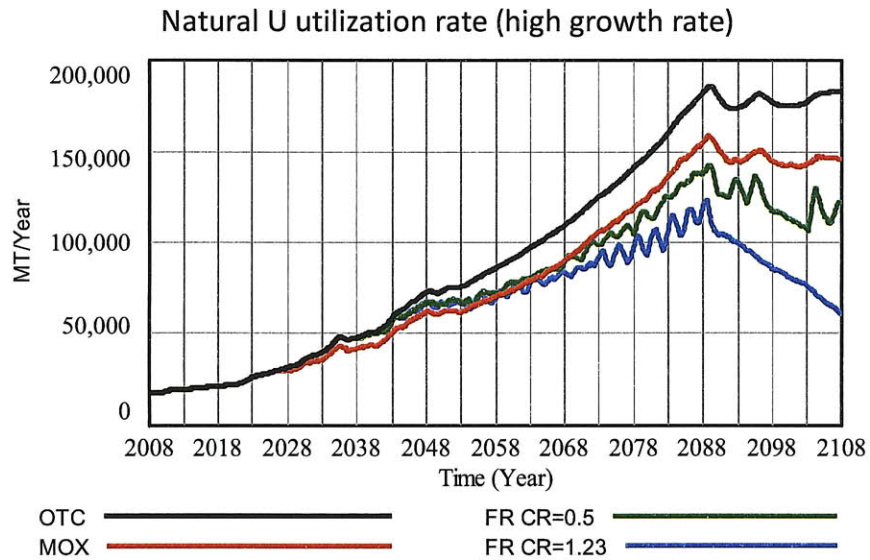
Figure 8.23 – Natural U utilization rate (high growth rate)

Table 8.5 shows the natural uranium utilization rate in 2050, 2090 and 2100 in the four scenarios.

Table 8.5 - Natural uranium utilization rate (high growth rate)

Date	OTC	MOX	FR CR=0.5	FR CR=1.23
2050	~ 72,000	~ 60,000 (-16.5%)	~ 66,000 (-8.5%)	~ 66,000 (-8.5%)
2089	~ 185,000 (-34.5%)	~ 159,000 (-14.0%)	~ 142,000 (-23.0%)	~ 119,000 (-35.5%)
2100	~ 175,000	~ 142,000 (-17.5%)	~ 115,000 (-34.5%)	~ 82,000 (-53.0%)

Figure 8.24 shows the cumulative natural U consumption in the four scenarios. Note that the fast burner scenario does little better than the TTC scenario.
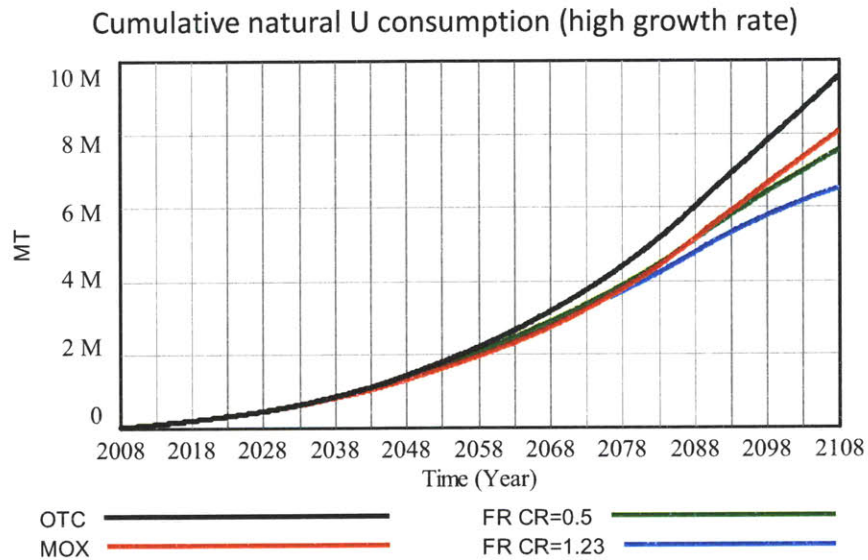
Figure 8.24 – Cumulative Natural U consumption (high growth rate)

Table 8.6 shows the cumulative natural uranium consumption in 2050 and 2100. Note that in this high-energy demand case, the fast reactors provide a more limited reduction in the demand for U. due to some extent to the limits on industrial capacity for thermal processing of spent fuel.

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Table 8.6 - Cumulative natural U consumption (1.0% growth rate)

Scenario	OTC	MOX	FR CR=0.5	FR CR=1.23
2050	1.558 Mtons	1.416 Mtons (-9.1%)	1.519 Mtons (-2.5%)	1.515 Mtons (-2.8%)
2100	8.119 Mtons	6.886 Mtons (-15.2%)	6.621 Mtons (-24.9%)	5.916 Mtons (-27.1%)

8.2.2.3 Impact on the TRU inventories

Figure 8.25 shows the total mass of TRU in the entire system (plants, reactor cores, storages, and repository). In terms of TRU reduction, the fast burner scenario catches up with the TTC scenario more than 50 years after the introduction of the first fast reactor. In 2108, the TRU in the entire system amounts to 16,869 tHM in the fast breeder scenario, 11,555 tHM in the TTC scenario, 10,705 tHM in the fast burner scenario and 15,094 tHM in the OTC scenario.
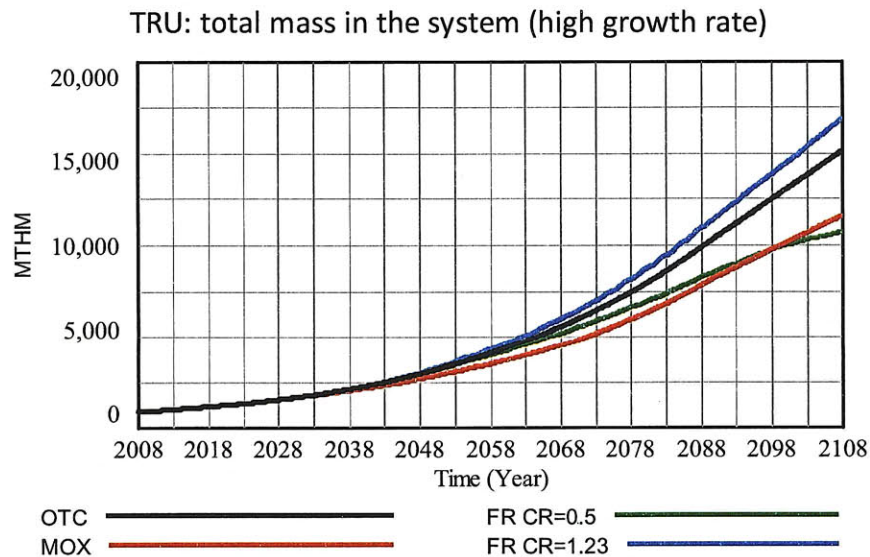
Figure 8.25 – TRU: total mass in the system (high growth rate)

8.2.2.4 Impact on the HLW inventories

Figure 8.26 shows the HLW in repository, assuming that the HLW is sent to repository 25 years after it is generated, and that the first repository opens in 2028. In 2108, there are 600,000 tHM of HLW (spent UO2 fuel) in the OTC scenario, which is almost 9 times

the legal capacity of Yucca Mountain. There are 77,000 tHM of HLW in repository in the TTC scenario (Mix of fission products and minor actinides + spent MOX fuel) vs. 22,000 tHM in the fast burner scenario and 23,000 tHM in the fast breeder scenario (fission products + TRU losses).
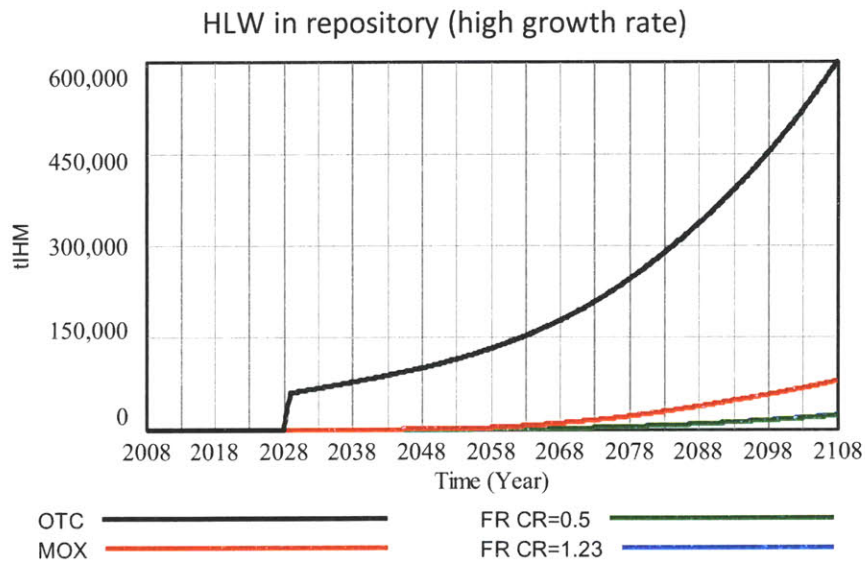
Figure 8.26 – HLW in repository (high growth rate)

Finally, Figure 8.27 shows the TRU content in the HLWs. In 2108, this content amounts to 12,631 tHM in the OTC scenario, 7,219 tHM in the twice-through scenario and only 18 tHM and 24 tHM (diluted in fission products) in the burner scenario and the breeder scenario, respectively.

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TRU in wastes (high growth rate)
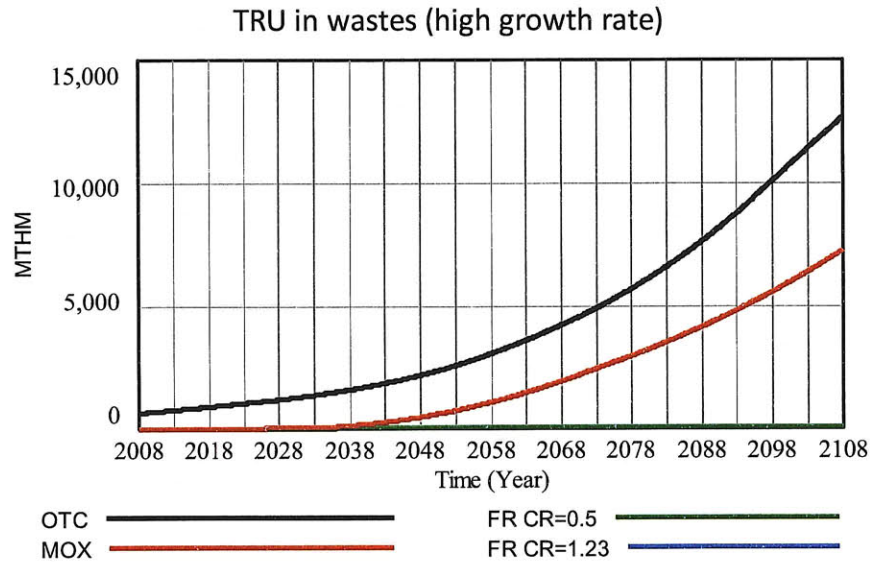
Figure 8.27 – TRU content in HLW (high growth rate)

8.3 Effect of Conversion Ratios of the Fast Reactors

We study in this section the impact of the conversion ratios of the fast reactors (CR=0.0, 0.5, 0.75, 1.0 and 1.23). For each of the scenarios, associated with one given conversion ratio, the framework is exactly the same as in the base case (Chapter 7, FR scenarios). After a slight growth leading to an installed capacity of 120 GWe in 2020, the nuclear energy demand increases by 2.5% per year. Thermal reprocessing plants are introduced in 2035; fast reactors in 2040. As in all our scenarios, the recovered uranium is recycled in reactors.

8.3.1 Reactors

Figure 8.28 shows the FR installed capacity in the five scenarios, while Figure 8.29 shows the resulting LWR-UO$_2$ installed capacity. In general, the higher the conversion ratio, the more fissile materials are available and therefore the more FR are built. However, one can notice that there are, over the first few decades, more self-sustaining FRs (CR= 1.0) than breeders (CR=1.23). The number of LWR-UO$_2$ becomes for the first

time lower in the breeder scenario than in the self-sustaining FR scenario only in 2080, i.e. 40 years after the introduction of the first fast reactor.
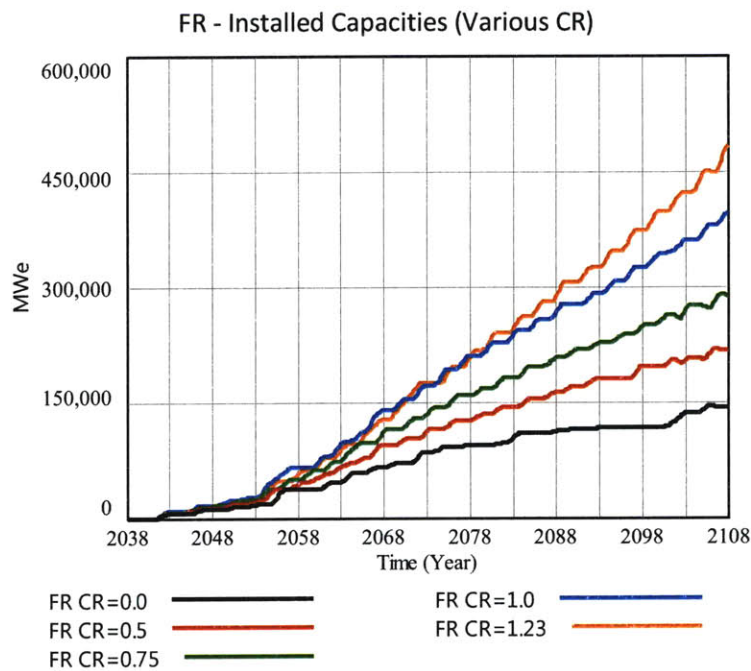
Figure 8.28 – FR installed capacities (various conversion ratios)
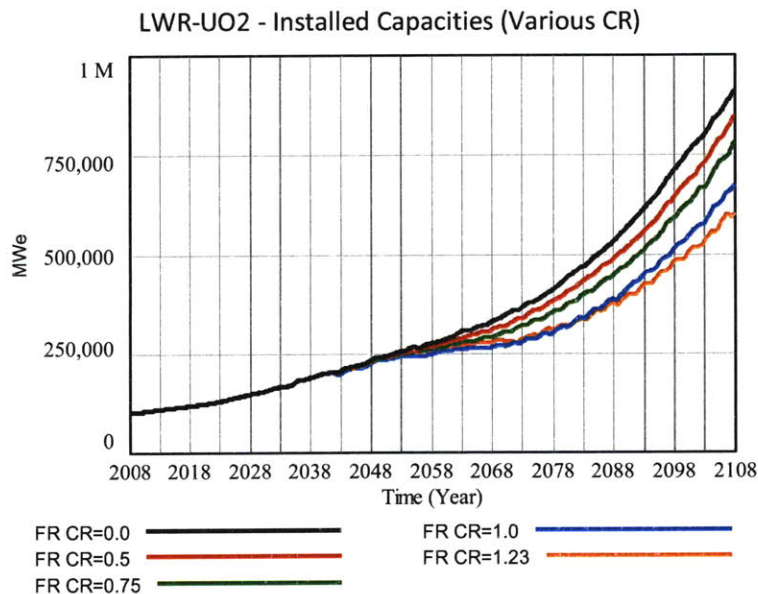
Figure 8.29 – LWR-UO2 installed capacities (various conversion ratios)

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Finally, Table 8.7 shows the installed capacity for the four scenarios in 2050 and 2100.

Table 8.7 - LWR-UO2//FR installed capacities in 2050 and 2100 (various conversion ratios)

	Date	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
LWR-UO2 installed capacity (GWe)	2050	240	238	237	231	235
	2100	748	677	623	536	495
FR installed capacity (GWe)	2050	13	16	18	23	20
	2100	118	196	254	341	396

8.3.2 Natural U consumption

Until 2080, there is more LWR installed capacity than FR installed capacity, therefore more consumption of natural uranium, In the breeder scenario than in the self-sustaining FR scenario. Even though the breeders are then deployed at a faster pace than the self-sustaining FRs, this reversal occurs too late to significantly affect the cumulative natural U consumption, as shown in Figure 8.30, by the end of the century.

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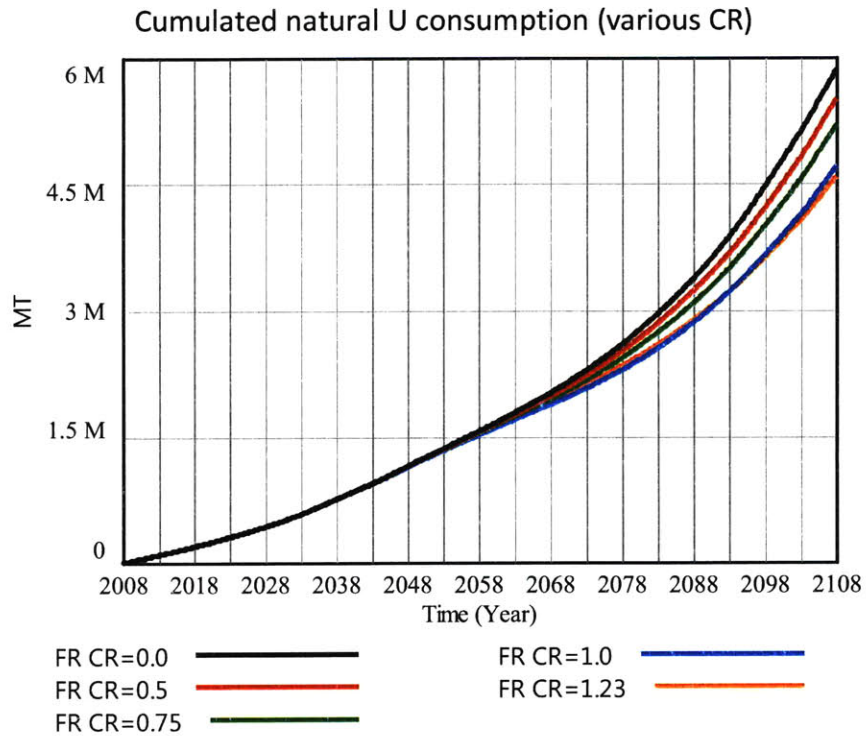
Figure 8.30 – Cumulative natural U consumption (various CR)

Table 8.8 shows the cumulative natural U consumption for the various conversion ratios, in 2050 and 2100. The OTC results are recalled as a reference. The introduction of breeders brings only an advantage of 0.8% compared to the self-sustaining FRs by 2100.

Table 8.8 - Cumulative natural U consumption (various CR)

Scenario	OTC	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
2050	1.26 Mtons	1.22 Mtons *(-3.2%)*	1.22 Mtons *(-3.2%)*	1.22 Mtons *(-3.2%)*	1.21 Mtons *(-4.0%)*	1.22 Mtons *(-3.2%)*
2100	5.855 Mtons	4.701 Mtons *(-19.7%)*	4.425 Mtons *(-24.4%)*	4.195 Mtons *(-28.4%)*	3.825 Mtons *(-34.7%)*	3.778 Mtons *(-35.5%)*

We observe that in both FR scenarios (self-sustaining and breeder), the reprocessing capacity is not constraining, as one can observe in the results that the spent fuel is always reprocessed within 2 months after it has been cooled down in cooling storage. Therefore

the only reason that FRs are not built at a faster pace in the breeder scenario is that breeders need more fissile materials than the self-sustaining reactors for their start-up cores: 8.64 MT TRU/GWe for the breeder vs. 6.31 MT TRU/GWe for the self-sustaining reactor (relative difference of 37%). This difference of 2.33 MT TRU (37%) is the equivalent of 9.3 years of production of the 1GWe LWR-UO_2 or 10.6 years of production of the 1 GWe breeder.

However, one cannot make definitive conclusions about the comparison between the self-sustaining reactor and the breeder. Indeed, the self-sustaining sodium cooled reactor as designed [Hoffman et al., 2006] has a very compact core and a short cycle length (370 EFPD) which may be beyond practicability. Figure 8.31 shows the results of the same simulation with a Gas-cooled Fast Reactor (GFR), described in [Busquim et al., 2008]. The GFR also has a conversion ratio of 1.0 but its start-up core contains 9.7 MT TRU/GWe (at a thermal efficiency of 47%). Using the same thermal efficiency as we used for the [Hoffman et al., 2006] reactors (38%), we obtain an even greater TRU loading of 12.0 MT TRU/GWe for the start-up core.

It is observed that (1) there are always more breeders than GFRs and (2) the number of GFRs significantly departs from the number of FR CR=0.75 (Hoffman design) only from the early 2080s.
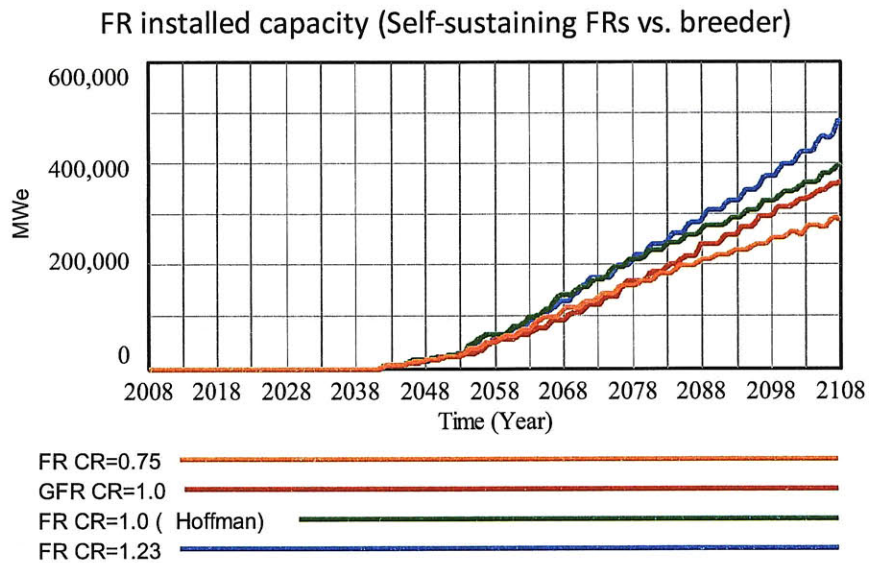
Figure 8.31 – FR installed capacities (self-sustaining FRs vs. breeder)

Figure 8.32 shows the corresponding cumulative natural U consumption (The OTC scenario is given as a reference). One can see that at the end of the century, the advantage provided by the introduction of the GFR is closer to that of the FR CR=0.75 than that of the FR CR=1.0 (Hoffman designs). Finally, if we use the GFR design rather the Hoffman design as a representative for the self-sustaining reactor, we find an advantage (by the end of the century) in the introduction of breeders vs. self-sustaining reactors, in terms of cumulative natural U consumption. By 2100, the latter amounts to 5,805 million MT in the OTC scenario, 4,195 (-27.7%) in the FR CR=0.75 scenario, 4,104 (-29.3%) in the GFR CR=1.0 scenario, 3,825 (-34.1%) in the FR CR1.0 scenario (Hoffman design), 3,749 (-35.4%) in the FR CR=1.23 scenario.
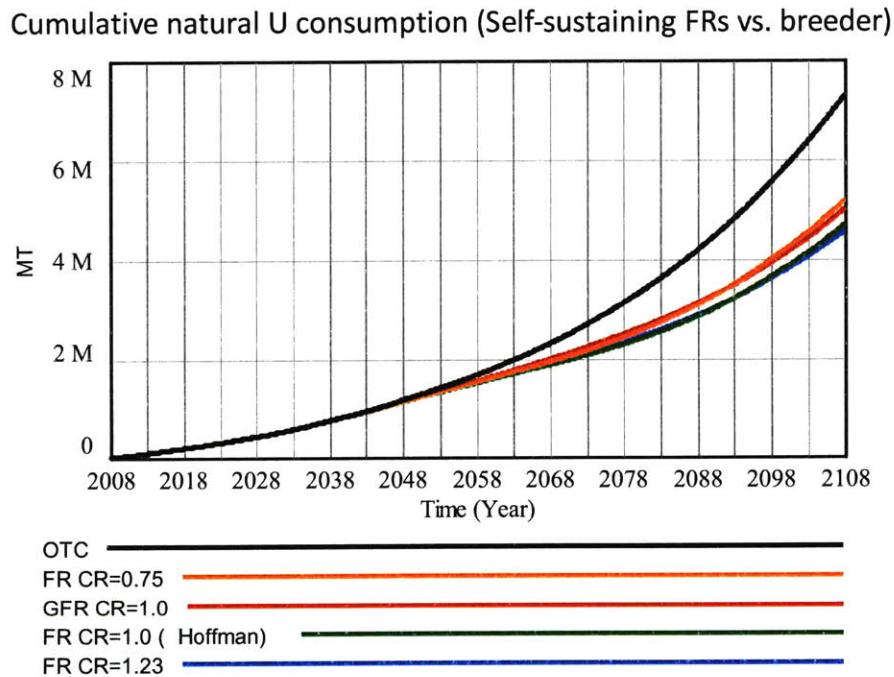
Figure 8.32 - Cumulative natural U consumption (self-sustaining FRs vs. breeder)

8.3.3 Reprocessing plants needed

Figure 8.33 shows the fast reprocessing capacity for the five scenarios. The unit capacity of the Fast reprocessing plants (FRP) is 100 tHM/year for the FR CR=0.0, 200 tHM/year

for the FR CR=0.5 and the FR CR=0.75, 500 tHM/year for the FR CR=1.0 and the FR CR=1.23.

As expected from the results of the simple model (see Table 3.3), the higher the conversion ratio, the higher the required reprocessing capacity. This results from the combination of two facts: first, the annual loading in reactors increases with the conversion ratio (2.780 tHM/year for the FR CR=0.0 to 14.843 tHM/year for the FR CR=1.23); second, the higher the conversion ratio, the more FRs are built (in general). In 2108, the fast reprocessing capacity is 400 tHM/year in the FR CR=0.0 scenario, 1400 tHM/year in the FR CR=0.5 scenario, 2600 tHM/year in the FR CR=0.75 scenario, 4500 tHM/year in the FR CR=1.0 scenario, and 6500 tHM/year in the FR CR=1.0 scenario
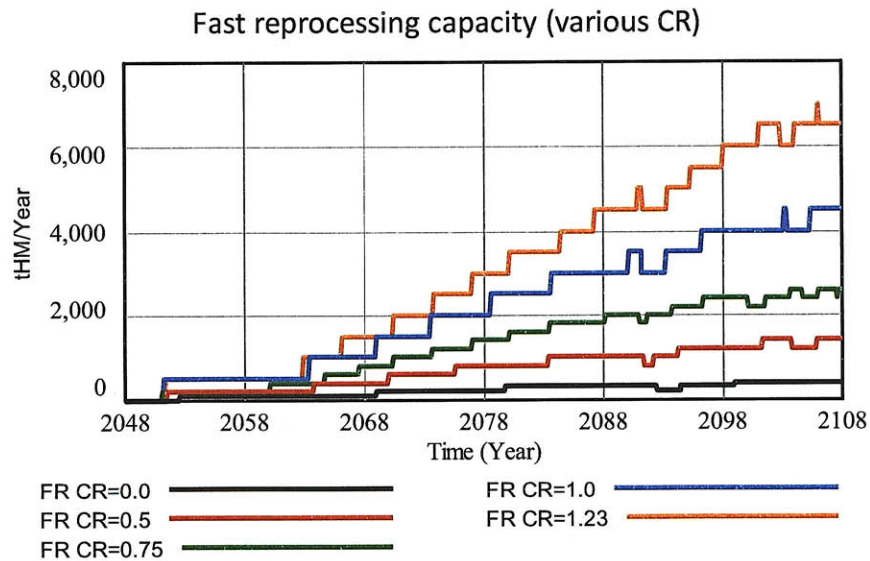
Figure 8.33 – Fast reprocessing capacity (various CR)

Figure 8.34 shows the mass loading factor of the fast reprocessing plants. As shown by Figure 8.33, the first FRP is built about 10 years after the first reactor and the second one 10 years to 20 years later. With the exception of the FR CR=0.75 case, the first plant remains underused (loading factor lower than 80%) for 10 to 15 years. Choosing a lower unit capacity for the first plant would have allowed starting fast reprocessing a few years earlier and running the plant at full capacity. However, economies of scale would have been lost.

FRP mass loading factor (various CR)
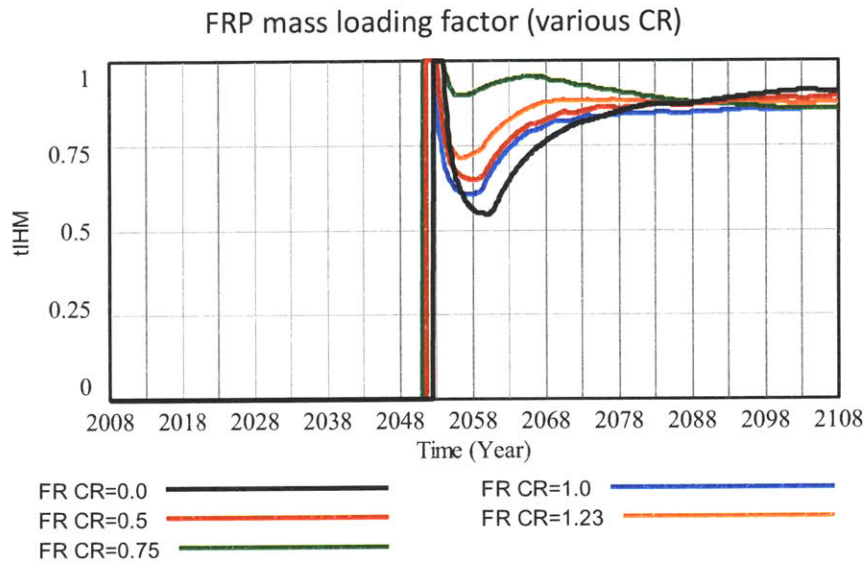
Figure 8.34 – Fast Reprocessing Plants mass loading factor (various CR)

8.3.4 TRU balance

Figure 8.35 shows the total mass of TRU in the system for the five FR scenarios, as well as the OTC scheme as a reference. One notes that the breeder scenario is the only one to generate more TRU than the OTC scheme (even the self-sustaining FR scenario leads to a reduction of the total amount of TRU). In terms of TRU reduction, the pure burner (CR=0.0) scenario does not do significantly better (+33%) than the FR CR=0.5 scenario despite additional costs due to the high TRU content of the pins.
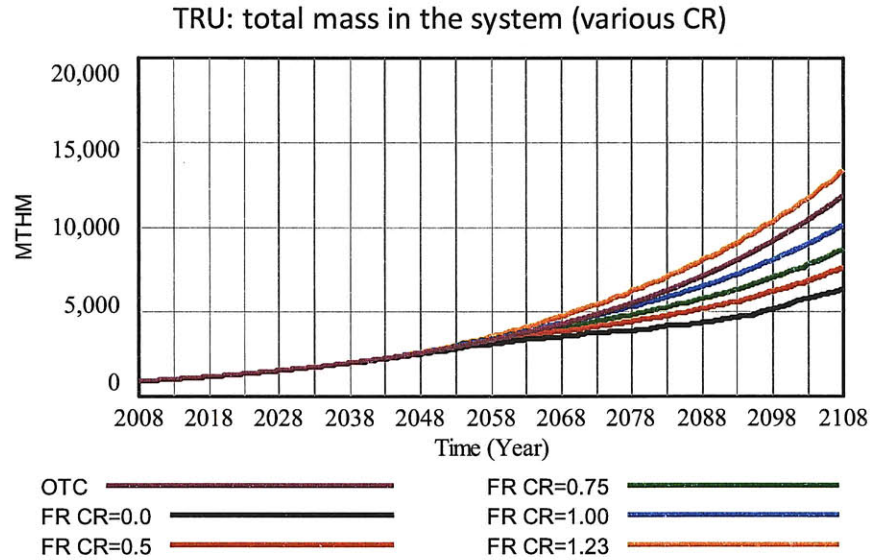
Figure 8.35 – TRU: total mass in the system (various CR)

Table 8.9 shows the total mass of TRU in the system in 2100. The best reduction in TRU from the OTC amount is obtained by the lowest conversion ratio FR. With a true burner (CR=0.0), the reduction amounts to nearly 45% by the end of the century.

Table 8.9 - Total amount of TRU in the system (various CR) in 2100

Scenario	OTC	FR CR=0.0	FR CR=0.5	FR CR=0.75	FR CR=1.0	FR CR=1.23
Total TRU in tHM	9,607	5,337 *(-44.4%)*	6.404 *(-33.3%)*	7,269 *(-24.3%)*	8,393 *(-12.6%)*	10,825 *(+12.7%)*

8.3.5 HLW

Finally, Figure 8.36 shows the HLW in repository, which consists in every FR scenario of fission products plus some diluted TRU losses. It is observed that the conversion ratio has virtually no impact on the mass of HLW. The mass of fission products is indeed roughly proportional to the amount of energy produced (whatever the type of reactor or fuel), which is the same in all the scenarios. There are in 2100 about 17,000 tIHM of fission products in repository.

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HLW in repository (various CR)

FR CR=0.0	FR CR=1.0
FR CR=0.5	FR CR=1.23
FR CR=0.75	
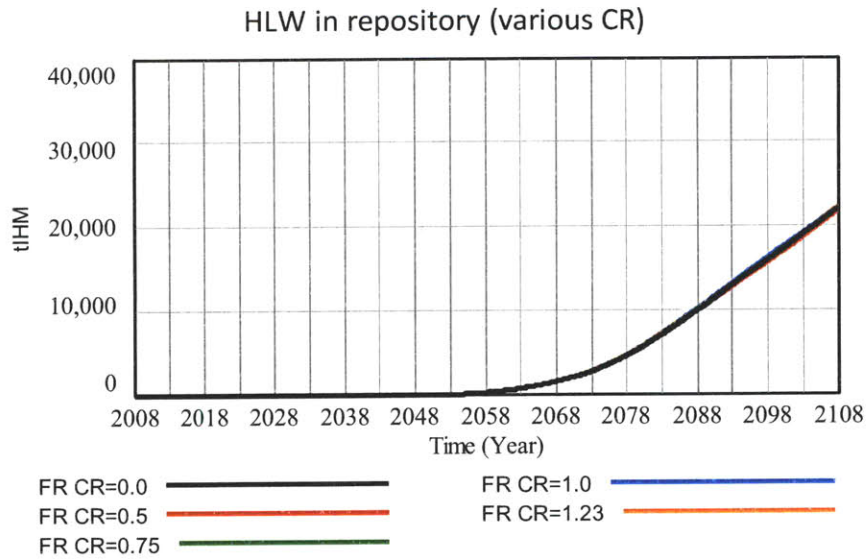
Figure 8.36 – TRU: total mass in the system (various CR)

Conclusions for this entire section are given in Section 8.7 ("Conclusion").

8.4 Variation in the cooling times

In this subsection, we study the effect of using a longer minimum cooling time of 10 years for all types for fuel, instead of 5 years as in the base case. We only look at the impact on the nuclear energy portfolio (fraction of advanced technologies in the total installed capacity), which ultimately affects the consumption of natural uranium as well the requirements in reprocessing capacity.

Figure 8.37 shows the MOX installed capacity (twice-through cycle scenario) for the two minimum cooling times. We observe that changing the minimum cooling time does not make any difference until 2062 (37 years after the introduction of MOX). This is due to the spent fuel legacy, which has already been cooled more than 10 years. Doubling the cooling time from 5 years to 10 years eventually reduces the MOX installed capacity by about 11% (the simple model presented in Chapter 3 forecasts a reduction of 10.6%).

MOX - Installed Capacities (cooling time 5 years vs. 10 years)

Cooling time = 5 years ———————
cooling time = 10 years ———————
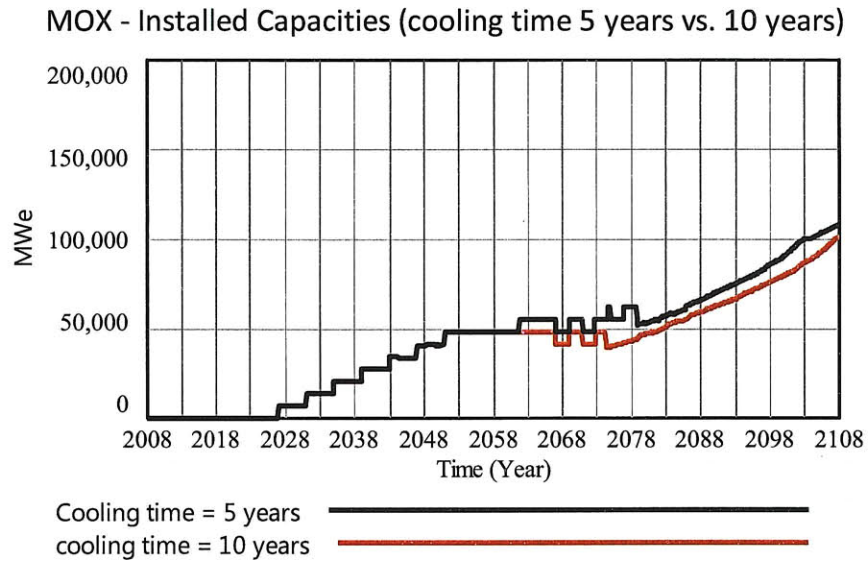
Figure 8.37 – MOX installed capacity (cooling time 5 years vs. 10 years)

Figure 8.38 shows the FR CR=0.5 installed capacity for the two minimum cooling times. In 2100, the FR CR=0.5 installed capacity represents 22.5% of the total installed capacity if the cooling time is 5 years vs. 16.7 % if the cooling time is 10 years (relative reduction of 25.5%, which is close to the result expected from the simple model : 24.1%).

FR CR=0.5 - Installed Capacities (cooling time 5 years vs. 10 years)

Cooling time = 5 years ———————
cooling time = 10 years ———————
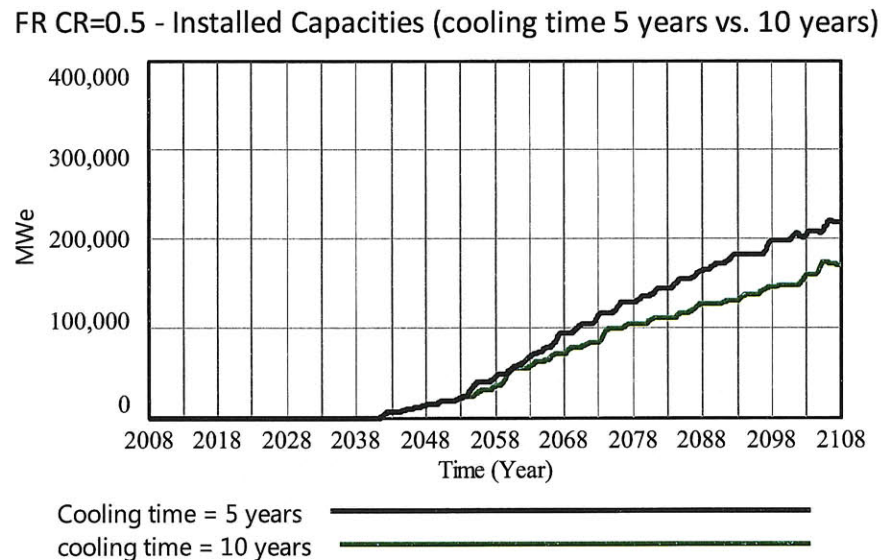
Figure 8.38 – FR CR=0.5 installed capacity (cooling time 5 years vs. 10 years)

Figure 8.39 shows the FR CR=1.23 installed capacity for the two minimum cooling times. In 2100, the FR CR=1.23 installed capacity represents 44.5% of the total installed capacity if the cooling time is 5 years vs. 30.6 % if the cooling time is 10 years, corresponding to a relative reduction of 31.2%. This reduction is much lower than the value expected from the simple model (43.5%). Once again, this is due to the importance of the loading of the start-up core, which is more pronounced when more FRs are built. That is why the CAFCA results are closer to those of the simple model presented in Chapter 3 when the minimum cooling time is extended.

Eventually, extending the minimum cooling time from 5 years to 10 years is almost equivalent to replacing the FRs CR=1.23 by FRs CR=0.75 (28.30% of the total installed capacity, see Table 8.7), in terms of nuclear energy portfolio and therefore of natural uranium consumption.

FR CR=1.23 - Installed Capacities (cooling time 5 years vs. 10 years)
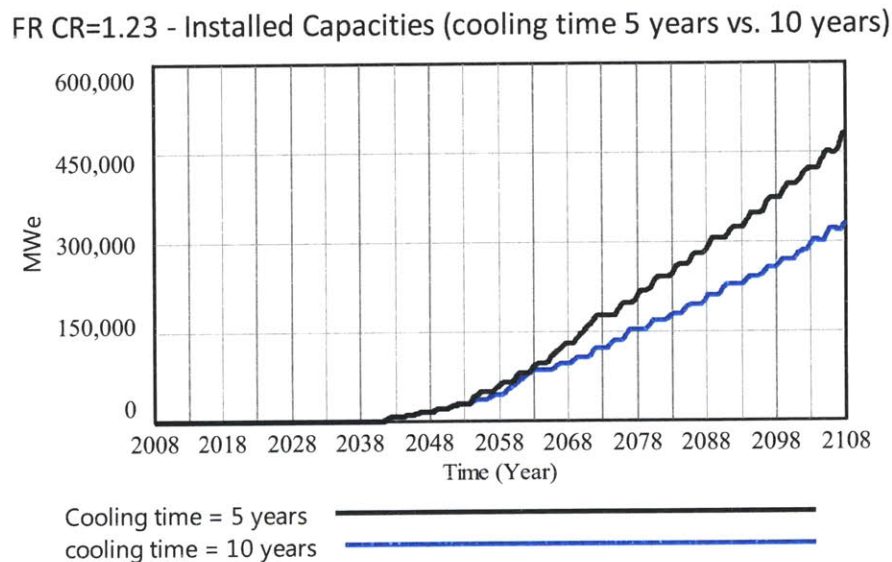
Figure 8.39 – FR CR=1.23 installed capacity (cooling time 5 years vs. 10 years)

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8.5 Variation of the thermal reprocessing introduction date (FR scenarios)

We study in this section the impact of introducing the thermal reprocessing plants in 2030 (scheme A) instead of 2035 (scheme B) as in the base case. Fast reactors are still introduced in 2040. Figure 8.40 and Figure 8.41 show the results for both introduction dates in the FR CR=0.5 and the FR CR=1.23 scenarios. The accumulation of fissile materials prior to 2040 (150 MT TRU if the ThRPs are introduced in 2030 vs. 40 MT TRU if they are introduced in 2035) allows a faster deployment of FRs over the first few years (e.g. 57 FRs vs. 43 FRs in 2058 in the FR CR=0.5 scenario). However, the advantage permanently disappears over the 2070s. We even observe a reversal in trend, especially in the FR CR=0.5 scenario. As a result, the cumulative consumption of natural uranium in 2100 remains virtually unchanged (difference of 0.2%).
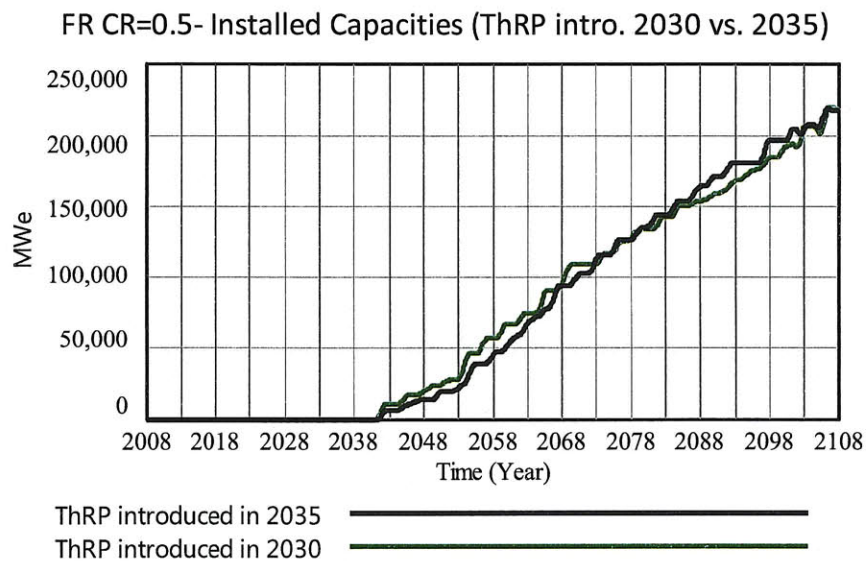
Figure 8.40 – FR CR=0.5 installed capacity (ThRP intro. 2030 vs. 2035)

200

FR CR=1.23- Installed Capacities (ThRP intro. 2030 vs. 2035)

ThRP introduced in 2035
ThRP introduced in 2030
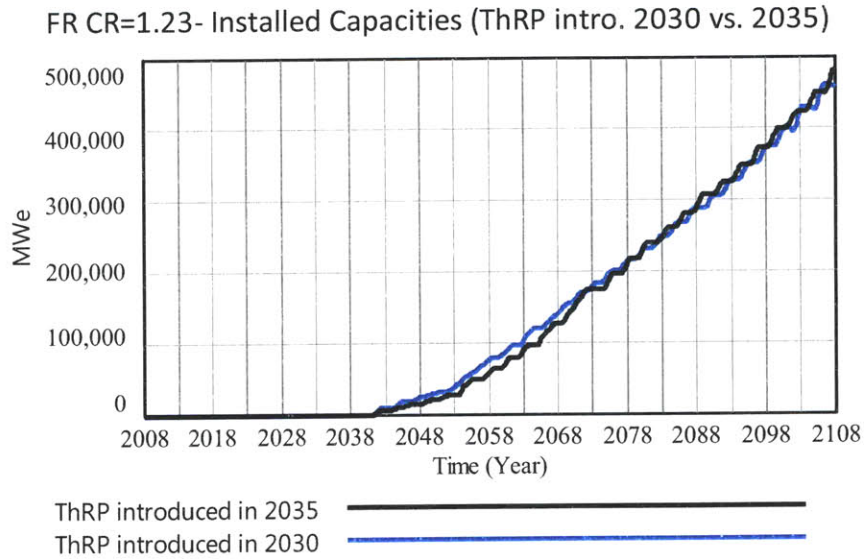
Figure 8.41 – FR CR=1.23 installed capacity (ThRP intro. 2030 vs. 2035)

Figure 8.40 shows the thermal reprocessing capacity in the FR CR=0.5 scenario. We can see that, on average, introducing the reprocessing plants in 2030 instead of 2035 provides an extra reprocessing capacity of 1,200 tHM/year until 2058. As a result, there is in 2058 33,600 tHM more of spent UO_2 fuel that has been reprocessed in scheme A than in scheme B. However, this does not change the total amount of spent UO_2 fuel to be reprocessed (actually it even decreases it as there are fewer LWRs-UO_2), which is why the number of thermal reprocessing plants is stabilized first in scheme A.

FR CR=0.5 - Th. rep. capacity (ThRP intro 2030 vs. 2035)

ThRP introduced in 2035

ThRP introduced in 2030
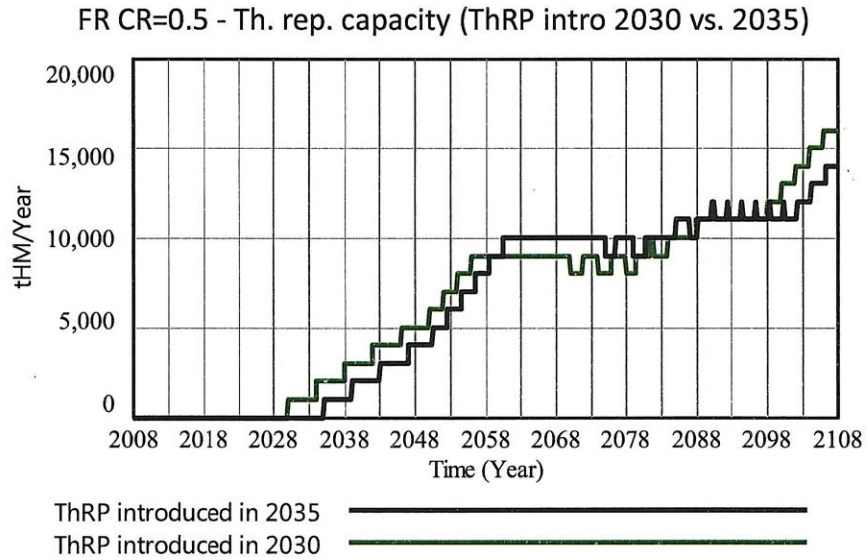
Figure 8.40 – Thermal reprocessing capacity (FR CR=0.5 case, ThRP intro. 2030 vs. 2035)

Finally, Figure 8.43 shows the total amount of spent UO_2 fuel in interim storage. An advantage of scheme A is to reduce the peak reached by the total amount of UO_2 spent fuel in storage: 127 tHM in scheme A vs. 107 tHM in scheme B.

UO2 SF in interim storage (ThRP intro. 2030 vs. 2035, CR=0.5)

ThRP introduced in 2035

ThRP introduced in 2030
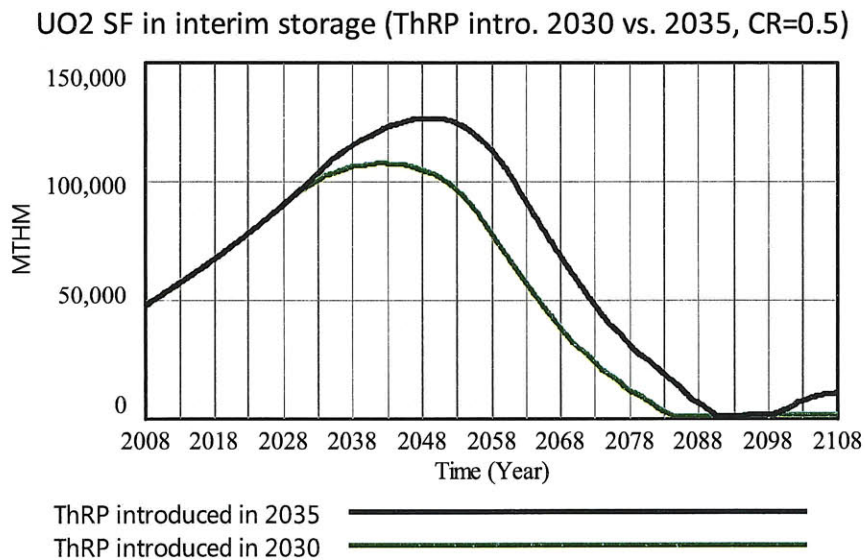
Figure 8.43 – Spent UO2 fuel in interim storage (FR CR=0.5 case, ThRP intro. 2030 vs. 2035)

8.6 Two-tier scenario: MOX in 2025, FR in 2060

We study in this section the scheme described in Section 2.7. In this 2-tier scenario, reprocessing of the spent UO_2 fuel as well as utilization of the MOX fuel is introduced in 2025. In 2055, the thermal reprocessing plants stop producing separated plutonium. The entire capacity is then employed to separate the TRUs from both the spent UO_2 fuel and the spent MOX fuel. The first FRs are finally introduced in 2060.

In order to see the advantages/drawbacks of such a scheme, a "FR scenario" (1-tier scheme), in which thermal reprocessing is introduced in 2055 and FRs start in 2060, was also run. The results of the twice-through scenario (called "MOX" and presented in Chapter 6) are also shown for comparison purposes.

The conversion ratio chosen for the fast reactor is 1.23 (i.e a breeder fast reactor).

Figure 8.44 shows the MOX installed capacities (electricity generated by the burning of MOX fuel batches) in the 2-tier scenario and the TTC scenario. There is virtually no difference until 2055, the date at which the separation of plutonium, and therefore the fabrication of MOX pins, is stopped in the 2-tier scenario.
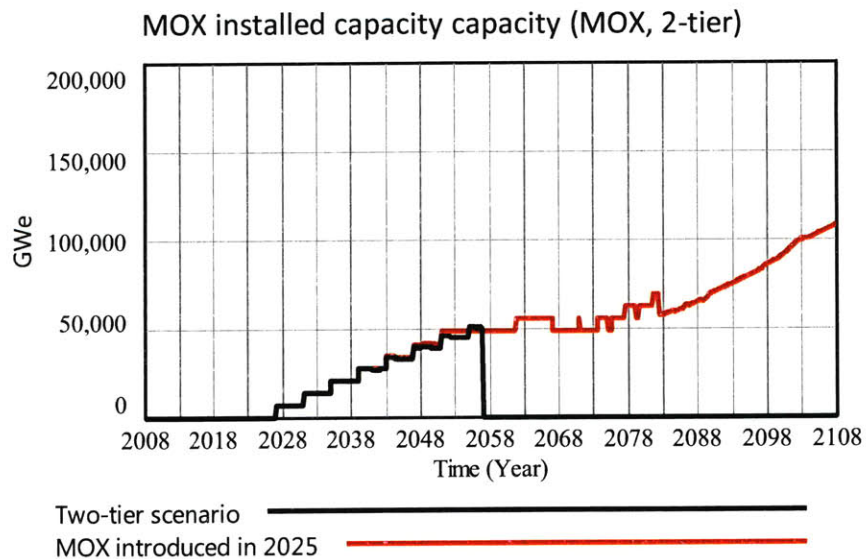
Figure 8.44 – MOX installed capacity (TTC scenario, 2-tier scenario)

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Figure 8.45 shows the FR installed capacities in the FR CR=1.23 scenario and the 2-tier scenario. A striking difference is the initial rate of building of the reactors. There are in 2075 (15 years after the introduction of the first reactor) 163 GWe of FR in the 2-tier scenario (more than 10 reactors built per year) vs. 40 GWe in the FR scenario. The reason for this difference is twofold. First, the installed ThRP capacity is already significant (8,000 tHM/year) in 2055, when the separation of TRU for FR fuel fabrication starts. Whereas the second 1000tHM/year plant is built only in 2059 in the FR scenario, and the capacity of 8,000 tHM/year is reached only in 2077. Second, about 5% of the spent fuel reprocessed in the 2-tier scenario is spent MOX fuel, which has a TRU content 5.23 times higher than the spent UO_2 fuel. Therefore the final TRU output is increased by 21% with respect to the situation where 100% of the spent fuel reprocessed is spent UO_2 fuel

In 2075, the building of new reactors stops in the 2-tier scenario (from 2077 to 2088) whereas it is accelerated in the FR scenario. As a result, the difference in the installed FR capacities between the two scenarios is dramatically reduced by the end of century (337 GWe in the 2-tier scenario vs. 289 GWe in the FR scenario in 2100). The stabilization of the number of FR in the 2-tier scenario is due to a shortage of fissile materials.
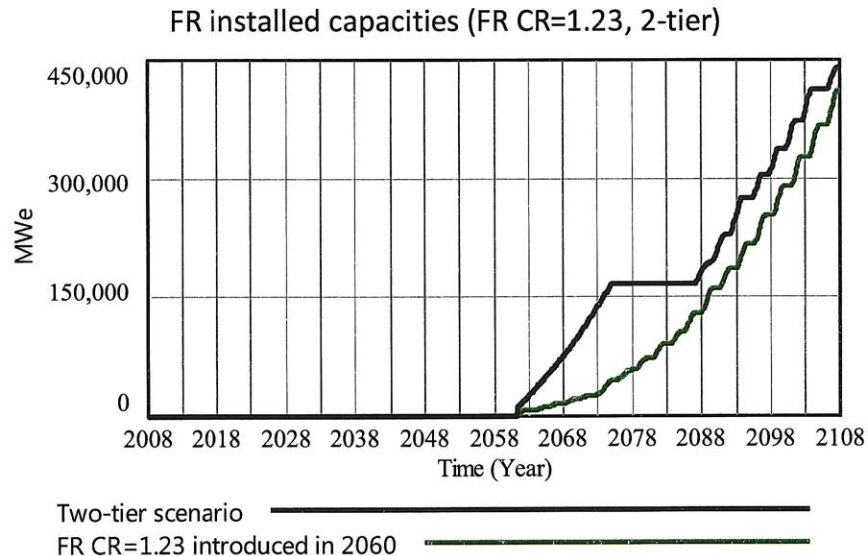
Figure 8.45 – FR CR=1.23 installed capacity (FR scenario, 2-tier scenario)

Figure 8.46 shows the resulting LWR-UO₂ installed capacities in the three scenarios. Apart from a short period of about 5-10 years following the end of utilization of MOX fuel, the best option to reduce the need for UO₂ fuel is the 2-tier scenario. Indeed, this scenario does as good as the TTC scenario until 2055 and then does better than the FR scenario (which is much more efficient as seen in Chapter 6) until the end of the century.
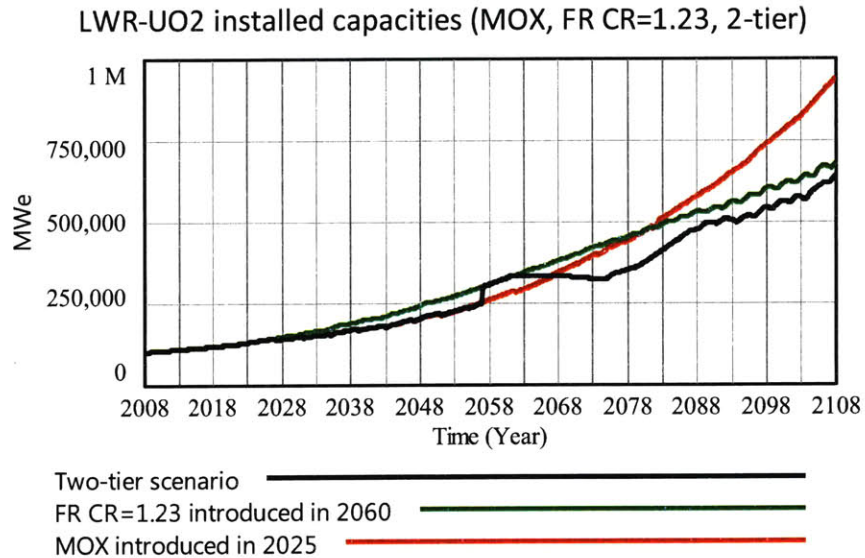
Figure 8.46 – LWR-UO2 installed capacity (FR scenario, TTC scenario, 2-tier scenario)

Table 8.10 shows the installed capacities for the three technologies in 2050, 2075 and 2100.

Table 8.10 - Installed capacities in 2050, 2075 and 2100 (MOX, FR and 2-tier scenarios)

Date	Scenario / Installed capacity (GWe)	OTC	MOX in 2025	FR CR=1.23 in 2060	2-tier
2050	LWR-UO2	252	211	252	213
	LWR-MOX	0	41	0	39
	FR	0	0	0	0
2075	LWR-UO2	463	408	330	320
	LWR-MOX	0	55	0	0
	FR	0	0	40	163
2100	LWR-UO2	859	769	602	557
	LWR-MOX	0	90	0	0
	FR	0	0	289	337

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As expected from the previous results, the 2-tier scenario is the most efficient in terms of natural uranium consumption reduction, at virtually any time of the simulation. Figure 8.47 shows the cumulative natural uranium consumption in the three scenarios.
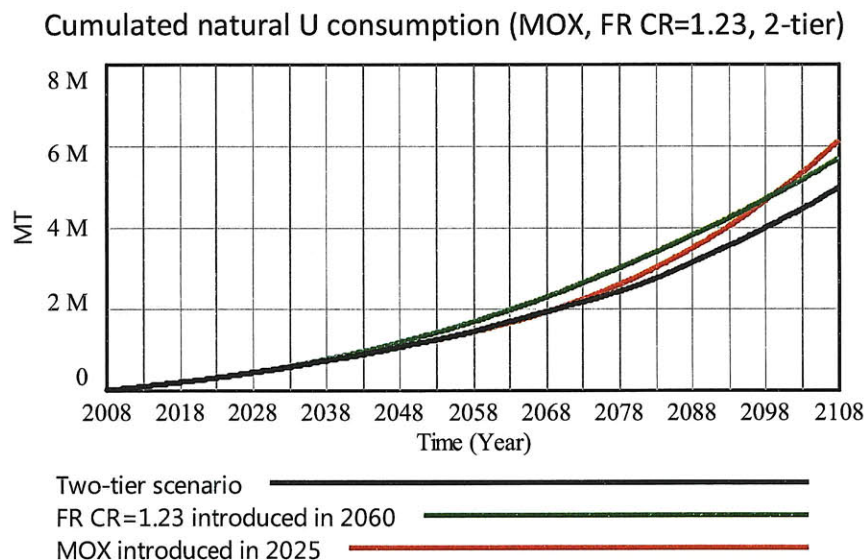
Figure 8.47 – Cumulative natural uranium consumption (FR, MOX and 2-tier scenarios)

Table 8.11 shows the cumulative natural uranium consumption in 2050, 2075 and 2100. Numbers for the OTC scenario are recalled as a reference. As of 2108, the 2-tier scenario is 71% more efficient than both the FR scenario (introduced in 2060) and the TTC scenario. However, it should be recalled that introducing the fast reactors earlier (in 2040), with a CR of 1.0 or higher) leads to a similar reduction near 30%. Thus, if the primary reason for introduction of the fast reactors is resource enhancement, introduction of thermal reactor recycling allows the introduction of fast reactors to be delayed by 20-25 years.

Table 8.11 - Cumulative natural U consumption (MOX, FR, 2-tier scenario)

Scenario	OTC	MOX	FR CR=1.23	2-tier
2050	1.26 Mtons	1.12 Mtons *(-11%)*	1.26 Mtons *(0%)*	1.12 Mtons *(-11%)*
2075	2.85 Mtons	2.36 Mtons *(-17%)*	2.76 Mtons *(-3%)*	2.25 Mtons *(-21%)*
2100	5.86 Mtons	4.89 Mtons *(-17%)*	4.85 Mtons *(-17%)*	4.15 Mtons *(-29%)*

For the three scenarios, Figure 8.48 shows the thermal reprocessing capacity while Figure 8.45 shows the fast reprocessing capacity. Until its total depletion in the late 2070s (see Figure 8.46), the reprocessing of the spent MOX fuel inventory requires on average about 5% of the total thermal reprocessing capacity, with a maximum of 13%. Therefore, little difference in the thermal reprocessing capacity is expected between the TTC scenario and the 2-tier scenario, until the generation rates of spent UO_2 fuel (after minimum cooling) significantly differ from each other.

Then, there are less ThRPs in the 2-tier scenario than in the TTC scenario (total capacity of 13,000 tHM/year vs. 16,000 tHM/year in 2100), but this difference is countered by the need for fast reprocessing plants (3,500 tHM/year in 2100).

However, there is in 2100 a difference of 4,000 tHM in the thermal reprocessing capacity (13,000 tHM/year in the 2-tier scenario vs. 19,000 tHM in the FR scenario) while the fast reprocessing capacities differ only by 1,500 tHM/year. The reprocessing activity is obviously spread over a longer period of time in the 2-tier scenario than in the FR scenario, resulting in lower requirements at a given time.
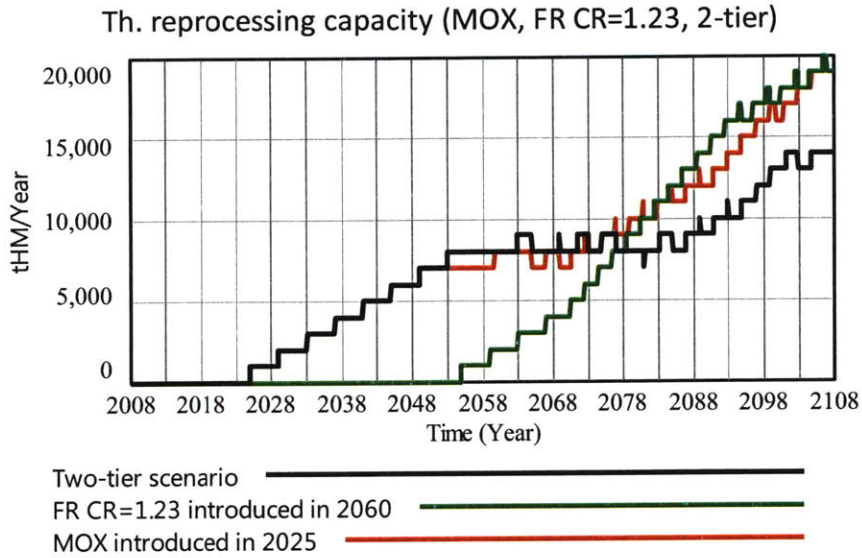
Figure 8.48 – Thermal reprocessing activity (FR, MOX and 2-tier scenarios)

One can see in Figure 8.49 that fast reprocessing plants are built as fast as allowed by the industrial capacity, which means that the latter is a limiting constraint. This constraint is actually only temporary – and has therefore limited effect in the medium-term, as the inventory of spent FR fuel is depleted in the late 2080s, which explains the temporary stabilization of the number of FRP in 2090.
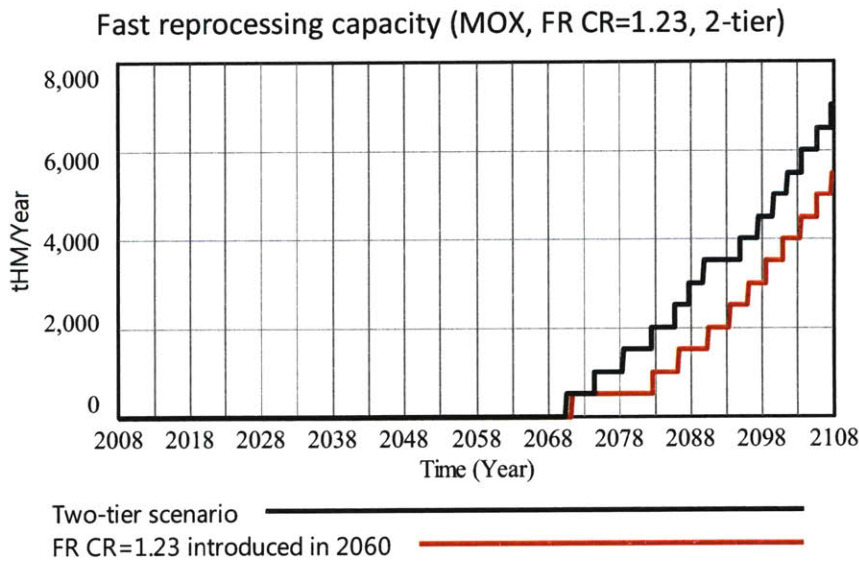
Figure 8.49 – Fast reprocessing capacity (FR scenario, 2-tier scenarios)

The management of spent fuel is a major advantage of the 2-tier scheme with respect to the FR scheme. Figure 8.50 shows the amount of spent UO_2 fuel in interim storage. In the FR scenario, the spent UO_2 fuel is accumulated until the FR technology becomes available in the 2060s. The inventory thus reaches a peak at 249,000 tHM in 2077, which represents 560 tHM per LWR under commercial operation at that time. In 2108, there are still more than 100,000 tHM of spent UO_2 fuel in interim storage.

In comparison, the inventory of spent UO_2 fuel reaches a peak at 91,000 tHM in 2033 in the 2-tier scenario and is depleted by 2085.

UO2 SF in interim storage (MOX, FR CR=1.23, 2-tier)

Two-tier scenario
MOX introduced in 2025
FR CR=1.23 introduced in 2060
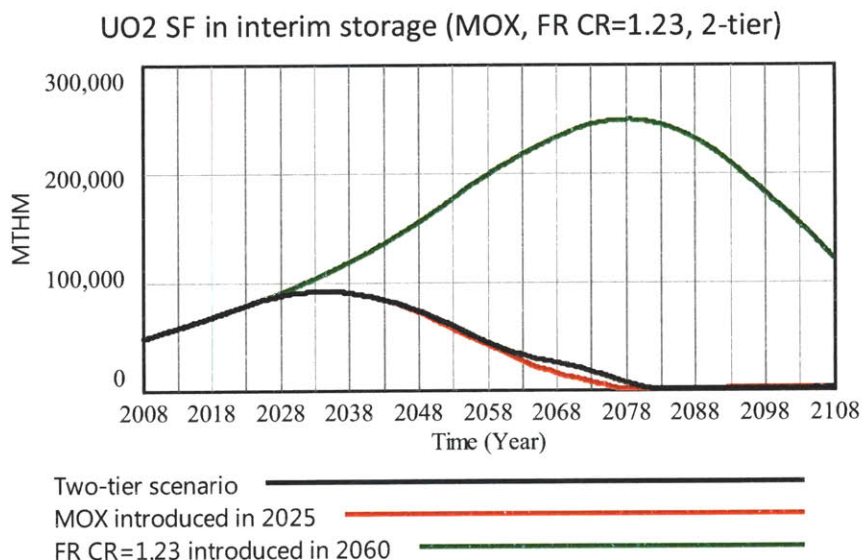
Figure 8.50 – Spent UO₂ in interim storage (FR, MOX and 2-tier scenarios)

An obvious advantage of the 2-tier scenario with respect to the twice-through cycle scenario is the recycling of the spent MOX fuel. Figure 8.51 shows the inventory of spent MOX fuel in the TTC scenario, which is destined to be sent to the repository, and in the 2-tier scenario, in which it is recycled and finally depleted.

MOX SF in interim storage/repository (MOX, 2-tier)

Two-tier scenario
MOX introduced in 2025
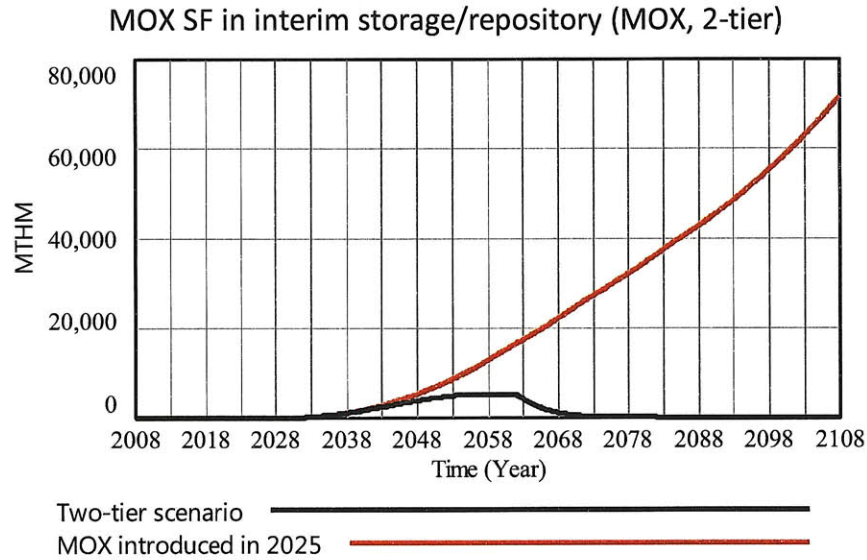
Figure 8.51 – Spent MOX fuel in interim storage/repository (MOX and 2-tier scenarios)

Finally, the fuel cycle cost (using the "energy-based scheme") and the dynamic levelized cost of electricity are presented in Figure 8.52 and Figure 8.53 respectively. As a composite scenario, the 2-tier scheme has a somewhat chaotic behavior. Until the separation of Pu stops in 2055, there is little difference between the 2-tier scheme and the MOX scheme: the reprocessing of the spent UO_2 fuel and the MOX fuel fabrication generates additional fuel cycle costs that are not offset by the savings in uranium. As all the reactors are still LWRs, capital costs remain unchanged. In 2050, the total levelized cost of electricity is increased by about 91 mills/kWh in both scenarios involving MOX. Between 2056 and 2062, only UO_2 fuel is utilized, which explains the dip in the fuel cycle costs. In the following 15 years, FRs are built at a fast rate, which dramatically increases their proportion in the nuclear energy portfolio. As a result, the average capital cost reaches its maximum in 2075 at 69 mills/kWh. Meanwhile, important amounts of FR fuel are used, making the fuel cycle cost soar. From the late 2080s, the fraction of FRs goes back to an equilibrium level, leading to stabilization of both the fuel cycle cost and the capital cost, and finally the total cost of electricity. Despite a higher FR installed capacity, the 2-tier scenario has over the last few decades a fuel cycle cost slightly lower than that of the FR scenario (difference of about 2.5 mills/kWh). This is due to the fact

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that extracting a given amount of TRU from spent UO_2 fuel is more expensive than extracting the same amount from the spent fuel of the FR ("self-recycling"). And as observed in Figure 8.45 and Figure 8.46, the TRU utilized for FR fuel fabrication comes more from self-recycling in the 2-tier scenario than in the FR scenario from the early 2080s on.

Fuel cycle cost (TTC, FR CR=1.23, 2-tier)

Two-tier scenario
FR CR=1.23 introduced in 2060
MOX introduced in 2025
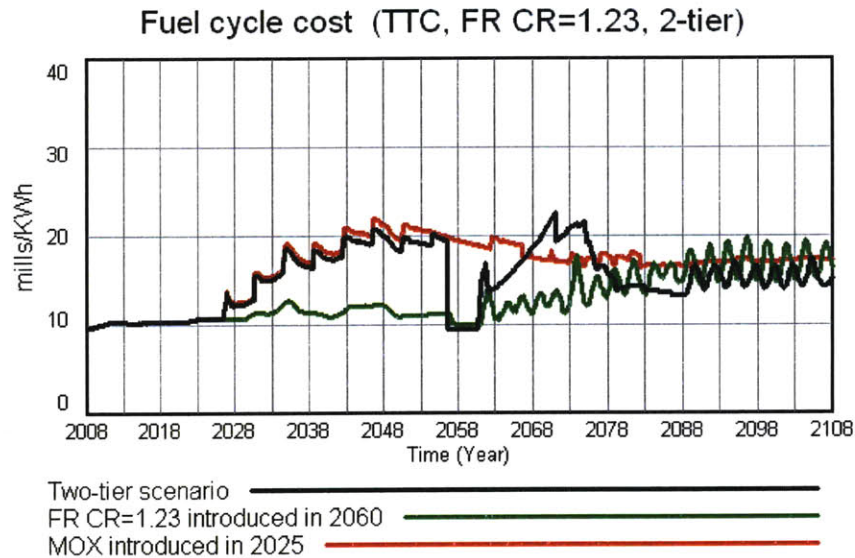
Figure 8.52 – Fuel cycle cost (MOX, FR and 2-tier scenarios)

Dynamic Levelized Cost of Electricity (TTC, FR CR=1.23, 2-tier)

Two-tier scenario
FR CR=1.23 introduced in 2060
MOX introduced in 2025
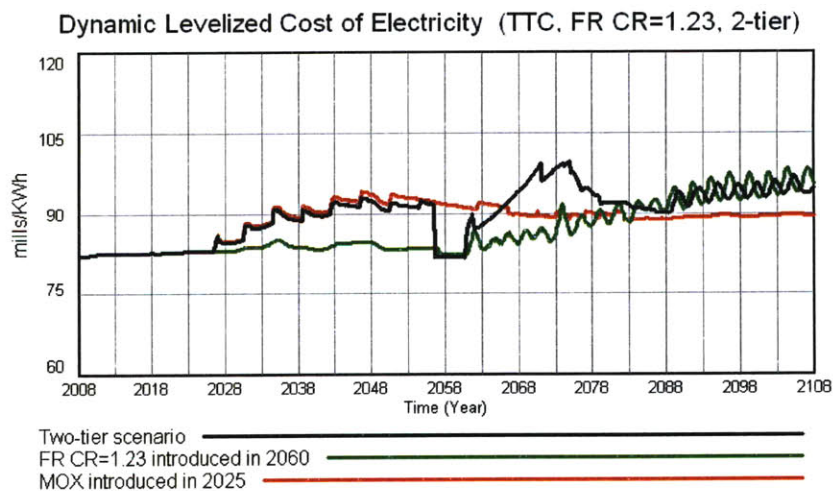
Figure 8.53 – Dynamic Levelized Cost of Electricity (MOX, FR and 2-tier scenarios)

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8.7 Summary

This chapter analyzes the impact of variation in the values assumed for some of the key parameters of the fuel cycle scenarios that were studied in the previous Chapter. In all the cases analyzed in this chapter, only one parameter is changed at one time with respect to the base case. Some notable observations are given below:

- In the low-growth rate scenario (1%/year), the share of fast breeder (FBr) in the total installed capacity cannot reach 100% in a sustainable way but is at a significantly higher level than in the medium growth-rate scenario (2.5%/year): 78% vs. 44% by 2100. The reason for this difference is threefold: 1) a lower growth rate always causes a lower "equilibrium" ratio as shown in Chapter 3; 2) a low-growth rate increases the relative impact of the initial spent fuel legacy; and 3) since a lower growth rate requires fewer FRs (absolute value), the impact of the start-up core loading is reduced. As a result, the breeder scenario brings the natural uranium consumption rate below its initial level at about 2080. The number of fast burner (FBu) starts differs from the number of FBr only in 2084 but ends up at less than half its level. It is noted that, in the three recycling scenarios, there is an excess of fissile materials that is eliminated only at the end of the century. Accelerating the utilization of these fissile materials would lead to undesirable low loading factors for the fast reactors and reprocessing plants (some reprocessing plants would become idle and some fast reactors would have to be shut down).

- Unlike in other cases, the limit on the industrial capacity for building of the thermal reprocessing plants is constraining in the high-growth rate case. The inventory of spent UO_2 fuel is thereby stabilized but not depleted. Moreover, the spent fuel legacy has relatively less impact than in the medium-growth rate case. As a result, the impact of the recycling options on the uranium consumption is more limited than in the lower growth rate scenarios.

- As expected, the higher the conversion ratio, the greater the share of fast reactors in the total installed capacity (from 14% in 2100 in the FR CR=0.0 scenario to 44% in the FR CR=1.23 scenario). However, it takes about 40 years for the FBr to do better than the self-sustaining FR (CR=1.0), due to differences in the startup core requirements (the core of the FR CR=1.0 is relatively compact). As a result, increasing the CR from 1.0 to 1.23 has virtually no impact on the cumulative natural uranium consumption as of 2100, while it creates more TRU in the system and requires higher fast reprocessing capacities. However, considering another design for the self-sustaining reactor (e.g. the Gas-Cooled Fast Reactor) changes this conclusion. The GFR indeed provides results similar to those of the FR CR=0.75 designed by [Hoffman et al., 2005] in terms of energy portfolio and uranium consumption, which relatively enhances the benefits of using breeders. Finally, the study reveals the importance of the design considered for a given conversion ratio (a small core is an advantage from a mass flow point of view but might be impractical in reality) on the final outputs. Besides, the study underscores the fact that the choice of the conversion ratio has virtually no impact on the HLW inventory, since the amount of fission products is roughly proportional to the energy produced. Furthermore, the pure burner (CR=0.0) is the best option to reduce the actinide total inventory (in 2100: -45% compared to the OTC scenario).

- As expected from the "equilibrium" model in chapter 3, extending the minimum cooling time from 5 years to 10 years for all fuel types reduces the share of fast reactors in the total installed capacity. In effect, the impact on the fast breeder scenario is almost equivalent to replacing the FBr by FRs having CR=0.75, in terms of energy portfolio and natural uranium consumption. By contrast, the extension of the cooling time has a small impact on the TTC scenario, which appears only from the late 2050s on. This is because the TTC recycles the spent UO_2 fuel only once, while most of the legacy is already more than 10 years old.

- Starting the reprocessing of the spent UO_2 fuel 10 years prior to the introduction of the fast reactors instead of 5 has a small impact in the short-term (the first FRs

are commissioned at a higher rate) and almost no impact on the mid-term (25 years later).

- The 2-tier scheme (assuming 35 years between the introduction date for the MOX and the introduction date for the FRs) dramatically accelerates the building of the first fast reactors for about 15 years. This is because introduction of MOX comes with elongating the period of LWR spent fuel production. If the primary reason for introduction of the fast reactors is resource enhancement, introduction of thermal reactor recycling allows the introduction of fast reactors to be delayed by 20-25 years (introducing MOX in 2025 and FRs in 2060 is equivalent to introducing FRs in 2040). The 2-tier scenario also allows maintaining the stock of spent UO_2 fuel in interim storage under 100,000 tHM (vs. 250,000 tHM in a pure FR scheme) and to finally deplete it by the end of the century. The 2-tier scenario is however the most expensive, as it combines the additional costs specific to the MOX fuel cycle (spent UO_2 reprocessing, MOX fuel fabrication) and those specific to the FR (in particular, the greater proportion of FRs, with respect to the FR scenario, results in higher capital costs).

Chapter 9

9. Summary of Conclusions and Recommendations

9.1 Summary of Conclusions

The system dynamics code CAFCA-SD has undergone further development and is used in this study to assess five alternative fuel cycle scenarios for nuclear power generation in the US in the 100 years to come. The performance of these scenarios was evaluated in terms of infrastructure needs, uranium consumption, actinide inventories, nuclear high-level waste inventories and costs. The five main fuel cycle options considered were:

- Once-Through Cycle (OTC): once-through uranium cycle for LWRs.
- Twice-Through Cycle (TTC): single-pass recycling of plutonium by burning MOX fuel pins in LWRs.
- TRU multi-recycling in sodium-cooled fast burners (conversion ratio less than 1)
- TRU multi-recycling in sodium-cooled fast breeders (conversion ratio exceeds 1)
- Two-tier scenarios: combination of the TTC scenario and Fast Reactors scenarios (the spent MOX fuel is recycled in fast reactors).

The various scenarios for the U.S., are influenced by the existence of a sizeable spent fuel legacy from operating LWRs for more than 40 years, and an initial installed capacity of 100 GWe. The technologies for separating plutonium and manufacturing MOX fuel pins are assumed to be available for market deployment in 2025; while the technologies for the reprocessing of the spent metal FR fuel are assumed to be introduced to the market in 2040, at the same time as commercial fast reactors; On the other hand the technology for separating the minor actinides along with plutonium is assumed to be available at least 5 years before introduction of fast reactors.

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The main observations can be summarized as follows:

- Comparisons were made between the results from CAFCA and the results from a steady-state (i.e. equilibrium) model that does not take into account the loading of the start-up cores, neglects the existence of spent fuel legacy, and assumes no constraint on the reprocessing capacity. In the fast burner scenarios (and a medium nuclear energy demand annual growth of 2.5 %), the existence of the spent fuel legacy leads to an initial overshoot of the share of fast reactors in the installed nuclear capacity over 30 years, with respect to the level predicted by the simple model (23% for the fast burner CR=0.5). On the contrary, the share of fast breeders remains far below the level provided by the simple model (less than 50% vs. 88% respectively), because fabrication of the first cores requires much TRU material.

- As a result, in a medium growth scenario, increasing the conversion ratio from 1.0 (self-sustaining reactors) to 1.23 (breeders) has virtually no impact on the natural uranium consumption by the end of the century, while it creates more TRU in the system and requires higher fast reprocessing capacity. However, considering a different design for the self-sustaining reactor (namely the gas-cooled fast reactor, the core of which is less compact than that of the sodium-cooled fast reactor taken as a reference) gives an advantage to the breeder. The larger GFR core need for TRU inventory leads it (with a CR = 1) to be only as good as the SFR with a conversion ratio of 0.75 in terms of reduction of cumulative natural uranium consumption. Thus, the study reveals that the choice of the design for a given core size may be as important as the choice of the conversion ratio in terms of reactors deployment and finally natural uranium consumption.

- In the medium growth case (and assuming that the worldwide natural uranium consumption rate is 3.5 times that of the U.S.), the introduction of fast breeders or self-sustaining reactors in 2040 reduces the cumulative natural uranium consumption by 35% in 2100 (vs. 16% for the twice-through cycle). However, the U price model indicates that these savings have a very modest impact on the price

of uranium, even in the pessimistic scenario ($200/kgU vs. $230/kgU in the OTC scenario in 2100). Overall, the U price model forecasts justify none of the recycling technologies for the next few decades, as the resulting savings do not offset the additional fuel cycle costs and capital costs. A reduction of $30/kgU in the U price leads to a decrease in the cost of electricity by less than 3 mills/kWh while the FR scenario adds on average between 5 and 11 mills/kWh to the OTC cost (compared to a total cost of about 85 mills/kWh).

- In the medium growth case, all recycling scenarios deplete the spent fuel inventory by the end of the century and maintain its level below 130,000 tHM (compared to 600,000 tHM in 2100 in the OTC scenario). This result requires the building of 8,000 to 10,000 tHM/year of thermal reprocessing capacity by 2060. The reduction in spent fuel inventory seems to be the only advantage of the twice-through cycle scenario, which is costly and does not bring a significant improvement in terms of repository requirements: the mass of spent fuel inventory is indeed reduced by a factor of 7, but the fission products and americium contained in the spent MOX fuel makes it so hot that the repository impact may go down to less than 2 once heat is taken into account for repository requirements.

- The 2-tier scenario enables delaying the introduction of fast reactors but dramatically accelerates the building of the first fast reactors, with respect to the 1-tier FR scenarios (the earlier the introduction of MOX utilization, the larger the effect). If the primary reason for introduction of the fast reactors is resource enhancement, introduction of thermal reactor recycling allows the introduction of fast reactors to be delayed by 20-25 years (introducing MOX in 2025 and FRs in 2060 is equivalent to introducing FRs in 2040). Moreover, the MOX transition allows a start on reducing the spent fuel legacy earlier than in the FR scenarios. Furthermore, only fission products are left in the final HLW, unlike in the twice-through cycle scenario, which eliminates the long-term proliferation issue but has the same short term proliferation concerns like the FR scenario. The two-tier case significantly reduces the repository requirements (by a factor of 5 accounting for

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the decay heat). However, the 2-tier scenario increases the additional costs associated with the MOX technology and the FRs alone, and is thereby the most expensive scenario.

- In a low-growth rate scenario (1%/year), available fissile materials are in excess of needs until the end of the century. Yet, the share of fast breeders in the total installed capacity cannot reach 100% in a sustainable way, even though it is much higher than in the medium growth rate scenario. Accelerating the utilization of the available fissile materials would lead to undesirable loading factors of fast reactors and reprocessing plants (some reprocessing plants would become idle while some fast reactors would have to be shut down). Eventually, LWRs will still be present in the system at the end of the century in all the scenarios considered.

- The higher the conversion ratio, the greater the fast reactors installed capacity (from 14% for the pure burner CR=0.0 to 44% for the breeder CR=1.23 in 2100, in the medium growth case) and hence the greater the savings in natural uranium. Besides, the best reduction in TRU from the OTC amount is obtained by the reactors with lower conversion ratio. With a true burner (CR=0.0), the reduction amounts to nearly 45% by the end of the century.

- Extending the minimum cooling time from 5 to 10 years for all types of spent fuel considerably reduces the share of advanced technology in the total installed capacity. Thus, in the fast breeder scenario it is almost equivalent to replacing the breeders by FRs with a conversion ratio of 0.75 (reducing installed capacity from 44% to 29%, respectively).

- Starting separation of the transuranics from the spent UO_2 fuel 10 years prior to introduction of the fast reactors instead of 5 years has a short-term impact (FRs are initially commissioned at a higher rate) but almost no impact in the mid-term (25 years later). It however implies additional costs because of the time value of money.

9.2 Recommendations

The following recommendations are made for the future development of CAFCA:

- CAFCA is still modest as a tool for nuclear waste assessment. The use of densification factors with regards to repository capability was a rough attempt to take into account the heat and the volume of the wastes, beside their masses. A more accurate analysis of the impact of the alternative fuel cycle options on the nuclear waste management would require CAFCA to account for radioactive decay. Such a feature would probably require deep changes in the code. Yet, the benchmark between CAFCA and other codes that account for isotopic decay (see [L.Guérin et al., 2009]) showed that the impact of the latter on the outcomes was not large and even in some cases not obvious. Eventually, any attempt to enable CAFCA to account for radioactive decay should be motivated by strong and explicit needs related to the nuclear waste assessment. In the case such a capability is implemented, a new benchmark against other dynamic codes like VISION, DANESS and COSI would be useful.

- The current economic module takes the fuel utilization rate as a basis for the fuel cycle cost calculations (in order to calculate the "dynamic levelized cost of electricity"). This is not correct when it comes to the loading of the start-up core because its associated fuel cycle cost exceeds that associated with the energy actually produced during the first year. This approximation results in peaks in the fuel cycle costs. The latter should be levelized for one given reactor.

- The hybrid model currently used to account for the reprocessing costs and disposal costs needs to be refined. In particular, a model must be developed to

allocate the money accumulated in the Nuclear Waste Fund to the future payments of reprocessing (and HLW disposal) costs.

- One interesting aspect of the dynamic codes is the possibility to study the trade-off that must be made between economies of scale and modularity for the reprocessing plants. Therefore, the user should not be constrained to use a unique unit plant capacity for the entire scenario. CAFCA could for example calculate the appropriate size for the reprocessing plant at any time of the simulation (the simple model presented in Chapter 3 could be helpful here).

- Instead of TRU, ^{235}U can be used to make the start-up core for fast breeders. Such a scheme is interesting for countries with a small spent fuel legacy or to accelerate the transition to a regime dominated by fast reactors. This capability could be added to CAFCA.

- The study did not consider the Very High Temperature Reactor (VHTR), which is one of the 6 designs retained by the Generation IV forum and is characterized by a very high thermal efficiency (nearly 45%). As an application of the VHTRs is the production of hydrogen, they could have a significant share in the nuclear energy portfolio in the future.

- A user-friendly interface for CAFCA has been developed but the access to the results is still tedious for a user not familiar with VENSIM. The use of macros on MSExcel could be combined with the export options available in VENSIM so that the user would easily obtain a series of graphs showing the main outputs. Moreover, the use of Excel would allow to smoothen these graphs (e.g. values would be averaged over 3 years), thus enhancing their readability and therefore their usefulness.

The following are recommendations for further study using the system simulation models (CAFCA or one of the other codes):

- It would be interesting to estimate the impact on the needed advanced reactor and fuel recycling infrastructure of advanced technologies with very different characteristics from what was studied here. Two important examples would be a very high conversion ratio LWR (for example the HITACHI designed RBWR) and a very long fuel cycle sodium cooled fast reactor (such as the traveling wave reactor). Such options may provide a disruptive approach to resource enhancement than has been assumed thus far.

- The thorium cycles are absent in these studies. However, it is possible that certain regions in the world move to thorium based fuel cycles due to the abundance of thorium locally, and to avoid high risks of dependence on global markets. This would imply a lower rate for the global demand for U after the regional introduction of thorium cycles. The extent that such new cycles would change the demand for U should be assessed.

- It is important to assess an alternative approach to recycling vs disposing of minor actinides. In the present study it is assumed that minor actinides along with plutonium will be placed in the recycled fuel. This requires considerable fuel development, and an alternative approach worth studying is disposal of minor actinides as wastes, while Pu is recycled. This approach is the more traditional approach in the closed cycles designed in the 1970s, but the disposal options of the minor actinides were not widely explored

- The assessment of US infrastructure needs ignored any impact of future fuel export and take back agreements with other countries. It is expected that such demands would remain small and dominate the US needs. But, an assessment of their impact on the dynamics of demand for fuel cycle services should be estimated.

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Appendix A.4.1 - Simplified US LWR Reactor Park Evolution

The U.S. LWR park for is simplified for CAFCA, assuming PWRs only, each having the same 1000MWe power rating and total installed capacity at 2008 of 100GWe. Figure A.4.1.1 plots capacity of currently operating U.S. LWRs. It is assumed that all currently operating reactors will have lifetime extended to 60 years. Corresponding shutdown profile is given in Figure A.4.1.2. It can be seen that the largest number of current LWRs is shutdown in 2034. The last reactor to be shutdown is in 2055. Numerical values are given in Table A.4.1.1. The initial capability of LWRmfs (licensed for MOX utilization) is assumed to be 17GWe.
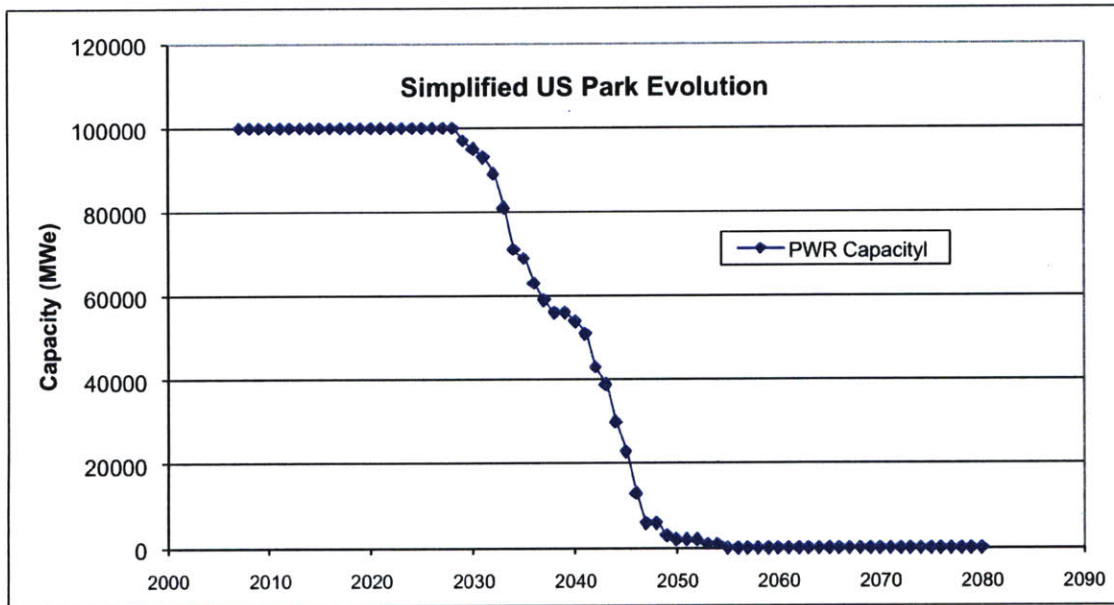
Figure A.4.1.1 - Simplified US LWR Park Evolution

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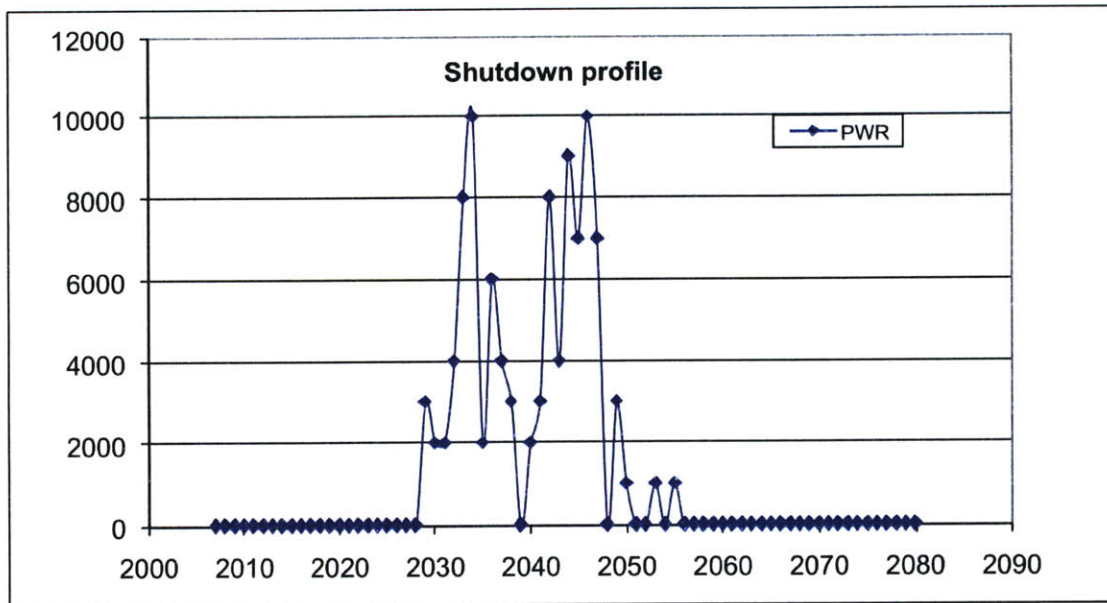
Figure A.4.1.2 - U.S. LWR Park Shutdown profile

Table A.4.1.1 - Simplified US LWR Park Shutdown Profile (MWe)

Year	Total shutdown	PWR UOX	PWR MOX	Total capacity	PWR UOX	PWR MOX
2007	0	0	0	100000	83000	17000
2008	0	0	0	100000	83000	17000
2009	0	0	0	100000	83000	17000
2010	0	0	0	100000	83000	17000
2011	0	0	0	100000	83000	17000
2012	0	0	0	100000	83000	17000
2013	0	0	0	100000	83000	17000
2014	0	0	0	100000	83000	17000
2015	0	0	0	100000	83000	17000
2016	0	0	0	100000	83000	17000
2017	0	0	0	100000	83000	17000
2018	0	0	0	100000	83000	17000
2019	0	0	0	100000	83000	17000
2020	0	0	0	100000	83000	17000
2021	0	0	0	100000	83000	17000
2022	0	0	0	100000	83000	17000
2023	0	0	0	100000	83000	17000
2024	0	0	0	100000	83000	17000
2025	0	0	0	100000	83000	17000
2026	0	0	0	100000	83000	17000
2027	0	0	0	100000	83000	17000
2028	0	0	0	100000	83000	17000
2029	3000	3000	0	97000	80000	17000
2030	2000	2000	0	95000	78000	17000

2031	2000	2000	0	93000	76000	17000
2032	4000	4000	0	89000	72000	17000
2033	8000	8000	0	81000	64000	17000
2034	10000	10000	0	71000	54000	17000
2035	2000	2000	0	69000	52000	17000
2036	6000	6000	0	63000	46000	17000
2037	4000	4000	0	59000	42000	17000
2038	3000	3000	0	56000	39000	17000
2039	0	0	0	56000	39000	17000
2040	2000	1000	1000	54000	38000	16000
2041	3000	2000	1000	51000	36000	15000
2042	8000	8000	0	43000	28000	15000
2043	4000	3000	1000	39000	25000	14000
2044	9000	6000	3000	30000	19000	11000
2045	7000	4000	3000	23000	15000	8000
2046	10000	8000	2000	13000	7000	6000
2047	7000	4000	3000	6000	3000	3000
2048	0	0	0	6000	3000	3000
2049	3000	1000	2000	3000	2000	1000
2050	1000	0	1000	2000	2000	0
2051	0	0	0	2000	2000	0
2052	0	0	0	2000	2000	0
2053	1000	1000	0	1000	1000	0
2054	0	0	0	1000	1000	0
2055	1000	1000	0	0	0	0
2056	0	0	0	0	0	0
2057	0	0	0	0	0	0
2058	0	0	0	0	0	0
2059	0	0	0	0	0	0
2060	0	0	0	0	0	0
2061	0	0	0	0	0	0
2062	0	0	0	0	0	0
2063	0	0	0	0	0	0
2064	0	0	0	0	0	0
2065	0	0	0	0	0	0
2066	0	0	0	0	0	0
2067	0	0	0	0	0	0
2068	0	0	0	0	0	0

Appendix A.4.2 - CAFCA Modifications for Uprated Reactors

This appendix presents the modifications made in CAFCA in order to model uprated LWRs using annular fuel (ULWR). This is actually an excerpt of [Feng et al., 2008], however edited. The numbers, which are those used by Feng, are not necessarily the same as those provided in this study. It is recommended to read [Feng et al., 2008] for more information about annular fuel and to see the results and conclusions about it drawn from CAFCA runs.

A.4.2.1 Introduction

The existing version of CAFCA SD was unable to model two types of LWRs simultaneously, thus the source code was modified in order to simulate both LWRs and 50% uprated LWRs using annular fuel (UWLR) simultaneously: a new High-level Structure Diagram (HLSD) was created to represent the flow of ULWRs, the Mass Flow HLSD was modified to include the fresh and spent fuel compositions of the ULWR fuel, and the corresponding fuel and capital costs were updated in the Economics HLSD. Only the LWR and ULWR HLSD modifications are described in detail since their fundamental structures had to be changed while the Mass Flow and Economics HLSDs were simply updated to include new fuel and reactor types. Only basic details about the original code will be mentioned, so it is recommended to refer to Chapter 2 of [Busquim et. al, 2008] for a fuller understanding.

A.4.2.2 Economics

The economic analysis for IXAF was done by Westinghouse [Kazimi et al., 2006] to determine the additional fuel manufacturing cost as well as overall economic incentive for utilizing annular fuel. For the manufacturing cost effects, the baseline constraints and assumptions used in the evaluation were the permitting and operational constraints at the Westinghouse Nuclear Fuels plant in Columbia, South Carolina. This study did not take into account any enrichment costs so any additional cost resulting from this study would only result in additional fabrication costs. The manufacturing changes in fabricating annular fuel were found to be insignificant except for the additional cost of the zirconium inner cladding. This increased the fabrication cost from \$0.005/kWhr(e) to \$0.00502/kWhr(e).

To increase the power density of the annular fuel, Westinghouse decided to pursue increasing the mass of the fuel by extending the rod length from 12 to 14 feet and increasing the density from 95.5 to 97% theoretical density instead of increasing the U-235 enrichment. The reason the enrichment method was dropped was because it had an undesirable effect on all the wet process areas of the fabrication plant.

The three options of interest analyzed by [Kazimi et al., 2006] are: 1) new construction of a 1117 MWe solid fuel PWR, 2) uprate of a 1000 MWe solid fuel PWR to a 1800 MWe annular fuel PWR, and 3) new construction of a 1717 MWe annular fuel

PWR. Option 2 was analyzed including costs for replacement power and un-utilized fuel value (Option 2a) and with only costs for new installed equipment (Option 2b). The summary and overall costs for the three of the options considered in this report are shown in Table A.4.2 with the financial assumptions for the comparison in Table A.4.1:

Table A.4.1 - Financial assumptions for comparison of annular and pellet PWR fuels (from Xu et al., 2004)

Price of Electricity	0.05 $/kWhr(e)]
Discount Rate	11 %/yr
Inflation Rate for Power Cost	1 %/yr
Inflation Rate for Fuel and O&M Costs	2 %/yr
Operating Rate	95 %/yr
Operating Time	20 years
Fuel Enrichment	5 %U235
Burnup	67 MWd/kg U
Plant Efficiency	35%

Table A.4.2 - Summary and overall costs of three considered options (from Kazimi et al., 2006)

Opt.	Description	Size (MW(e))	Fuel Cost ($/kWh(e))	O&M Cost ($/kWh(e))	Capital Cost ($/kW(e))	Total Capital Cost ($)	ROR (%)
1	New Solid Fuel PWR	1117	0.005	0.01	1313	1,466,853,397	6.9
2a	PWR Uprate all costs included	600 added to 1200, total 1800	0.00502	.005 on added 600 MW(e)	1817	1,090,200,000	6.3
2b	PWR Uprate component costs only	600 added to 1200, total 1800	0.00502	.005 on added 600 MW(e)	1381	828,600,000	11.6
3	New Annular Fuel PWR	1717	0.00502	0.0066	1103	1,893,299,218	11.5

A 3-year construction period is assumed for the new plant or components but for the uprate case (Options 2a and 2b), the plant was assumed to be offline for 12 months. Notice that if replacement power and unused fuel costs are included, the uprate option would have the lowest Rate of Return (ROR) as seen in option 2a. However, if these

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costs are ignored and only component costs are included (option 2b), then it would have the highest ROR. The Rate of Return figures are the return on investments beyond what can be obtained in passive deposition of funds at banks. For a total return figure, one should add to this ROR the interest rate assumed, which for the Westinghouse study was 11%.

It was demonstrated by [Beccherle, 2007] that by using a fuel management scheme that gradually replaces the solid fuel with annular fuel one batch at a time before the uprate construction, the solid fuel will be fully utilized and will not contribute towards the capital cost of the uprate. The savings from fully utilizing the solid fuel before the uprate was calculated to be $71,000,000 [Lahoda et al., 2007] for the PWR. In addition, it was pointed out that by timing the uprate construction with a scheduled steam generator replacement the additional cost of the steam generators, estimated to be $150,000,000 [Lahoda et al., 2007], can be subtracted from the capital cost as well.

In [Lahoda et al., 2007], the total overnight cost of option 2a was actually $1,230,589,281 (2051 $/kWe) for a 1200 MWe reactor, with a replacement power cost of $224,694,000. However this overnight cost was then normalized from a 1200 MWe reactor to the AP1117 (option 1) to obtain 1817 $/kWe.

If the replacement power was assumed for a period of 1 year, then the cost of electricity assumed by Westinghouse can be calculated using Equation A.4.1:

$$C_{rep} = \frac{C_R}{P_E \cdot CF \cdot T_{off}} \cdot$$

(A.4.1)

where C_{rep} is the cost of replacement electricity, C_R is the total cost of replacement power ($224,694,000), P_E is the nominal electric power (1,200,000 kWe), CF is the capacity factor (0.95), and T_{off} is the offline period in hours (365 days x 24 hours/day= 8760 hours). Given these assumptions, the C_{rep} used by Westinghouse was $0.0225/kWhe.

A.4.2.3 ULWR from new construction

There are two ways in which a ULWR is created: 1) through new construction and 2) by uprating an existing LWR. Although both pathways result in the same type of reactor, the uprated ULWR will enter the fleet with the LWR age maintained. The creation of ULWRs through the uprate method is more complex, so the construction of new ULWRs will be discussed first.

The ULWR Construction HLSD, which at this point does not include the uprate feature, is heavily based on the LWR Construction HLSD. The major difference is that construction of ULWR does not begin until the *introduction date for ULWR, D_{ULWR}* which is a variable designated by the user. It should be noted that CAFCA first calculates a fractional number of reactors needed to fulfill the electricity demand but the reactors are only constructed in integer values. For example, although the *fractional ULWR construction order rate, $R_{CO}^{ULWR}(t)$* , at a given time may be 2.1 reactors per timestep, the *ULWR fulfilled order rate, $R_{FO}^{ULWR}(t)$*, is only 2 reactors per timestep. To implement the introduction date for ULWR construction, $R_{CO}^{ULWR}(t)$ was set to zero until the year D_{ULWR}. Afterwards , it is:

$$R_{CO}^{ULWR}(t) = Maximum\left[0, ADJ_{ULWR}(t) + R_{DR}^{ULWR}(t) + \frac{R_{DR}^{LWR}(t)}{UR}\right] \qquad (A.4.2)$$

where $R_{DR}^{ULWR}(t)$ and $R_{DR}^{LWR}(t)$ are the *ULWR decommissioning rate* and *LWR decommissioning rate*, respectively, *UR* is the *average uprate* (1.5 by default so 1 ULWR is ordered for every 1.5 LWRs decommissioned), and $ADJ_{ULWR}(t)$ is the *adjustment for fleet of ULWR* or difference between the number of reactors needed to fulfill the power demand and the current number of reactors evaluated as:

$$ADJ_{ULWR}(t) = \frac{F_{EST}^{ULWR}(t) - F_{ULWR}(t)}{\tau_{ULWR}} \qquad (A.4.3)$$

where $F_{ULWR}(t)$ is the *ULWR fleet under commercial operation*, τ_{ULWR} is the *ULWR fleet adjustment time*, the time constant in which the discrepancy would be corrected (default of 1 year), and $F_{EST}^{ULWR}(t)$ is the *forecasted ULWR fleet* evaluated as:

$$F_{EST}^{ULWR}(t) = \frac{P_N(t) - \sum_r F_r(t) \cdot P_r(t) \cdot CF_r}{CF_{ULWR} \cdot P_{ULWR}(t)} \qquad (A.4.4)$$

where $P_N(t)$ is the *nuclear power demand* and the terms in the summation $F_r(t)$, $P_r(t)$, and CF_r represent the number of reactors, net electrical output per reactor, and capacity factor, respectively, for the fleets of non-ULWR reactors (ABR, GFR, and LWR).

A major assumption made in the model is that once the annular fuel technology is implemented, all new light water reactors will be constructed as ULWRs since this is the most economic option as shown by Table A.4.2. Thus the *fractional LWR construction order rate* $R_{CO}^{LWR}(t)$ will be zero after the year D_{ULWR}.

A.4.2.4 ULWR from uprate

If the uprate construction for LWRs is timed with that of a scheduled steam generator replacement, then the cost of the new steam generator would not be assigned to the capital cost of the uprate. In addition, the replacement power for the 3-month offline period for steam generator replacement is assumed to be covered by the operations and maintenance costs as well. If the steam generator is assumed to be replaced every 20 years, then the model assumes that the *available number of LWRs for uprate*, $A_{Ui}(t)$ at each time step is only non-zero for $i = 20$ or 40, where i is the coflow number (from 1 to 60) and the value $A_{Ui}(t)$ is equal to the number of LWRs under commercial operation that are i years old. However, not all of the LWRs available for uprates need to be uprated; the user can determine the *percent of available LWRs actually uprated* P_{AU}.

When an uprate occurs, a 20 or 40 year-old LWR is removed from the LWR fleet and added as a 21 or 41 year-old ULWR to the ULWR fleet during the same year. The off-line period of 1 year is not actually simulated; the power demand is assumed to be satisfied instantly. If this delay were to be modeled, additional ULWRs would be constructed during that period to try to make up for the demand. However, this off-line period is still accounted for in the Economics HLSD in terms of replacement power cost.

An *uprate rate* $R_{Ui}(t)$ is introduced as the outflow of the 20 and 40 year-old uprated LWRs from the ULWR fleet which is evaluated as a fixed delay with a delay of 1 year:

$$R_{Ui}(t) = Integer\left(R_{UAi}(t) \cdot P_{AU}\right) \tag{A.4.5}$$

where $R_{UAi}(t)$ is the *uprate availability rate*, which is equal to the rate at which 20 and 40 year-old LWRs become available. Notice that this is in units of [fractional reactors per time step] and different from $A_{Ui}(t)$ which is a measure of the discrete number of LWRs available. $R_{UAi}(t)$ is zero until D_{ULWR} and afterwards it is simply:

$$R_{UAi}(t) = R_{CR\ i}^{TRA}(t) \tag{A.4.6}$$

where $R_{CR\ i}^{TRA}(t)$ is the *transition LWR construction rate* modeled as:

$$R_{CR\ i}^{TRA}(t) = R_{FO}^{LWR}(t) \quad \text{if} \quad i = 1 \tag{A.4.7}$$

$$R_{CR\ i}^{TRA}(t) = R_{DR\ i-1}^{TRA}(t) \quad \text{if} \quad i = 2\ to\ 60 \tag{A.4.8}$$

where $R_{FO}^{LWR}(t)$ is the *LWR fulfilled order rate*. When $R_{Ui}(t)$ is greater than zero, the *transition LWR decommissioning rate* $R_{DR\ i}^{TRA}(t)$ must be adjusted so that the losses from the LWR fleet are not double counted:

$$R_{DR\ i}^{TRA}(t) = MAX\left(0, R_{IN\ i}^{TRA} + R_{LWRi}^{TRA} - R_{Ui}(t)\right) \tag{A.4.9}$$

where $R_{IN\ i}^{TRA}$ is the *transition rate for initial number of LWRs* and R_{LWRi}^{TRA} is the *transition rate for LWR* which is just a fixed delay of 1 year for $R_{CR\ i}^{TRA}(t)$.

To avoid building or uprating an excessive amount of ULWRs, the *fractional ULWR construction order rate*, $R_{CO}^{ULWR}(t)$ from Equation A.4.1 must be updated so that the fluctuations in the *adjustment for fleet of ULWR* $ADJ_{ULWR}(t)$ are minimized. This was done by ensuring that new ULWRs are only constructed based on the ULWR demand after all the uprates have occurred, thus giving the uprates precedence, even though it is the more costly option based on Table A.4.2. To have new ULWR construction as the only option, the user can simply set P_{AU} equal to zero. So now Equation 4.1 becomes:

$$\text{if} \quad \frac{A_{Ui}(t) \cdot P_{AU}}{3} < ADJ_{ULWR}(t),$$

$$R_{CO}^{ULWR}(t) = Maximum\left[0, ADJ_{ULWR}(t) - \frac{A_{Ui}(t) \cdot P_{AU}}{3} + R_{DR}^{ULWR}(t) + \frac{R_{DR}^{LWR}(t)}{UR}\right] \tag{A.4.10}$$

$$\text{if} \quad \frac{A_{Ui}(t) \cdot P_{AU}}{3} > ADJ_{ULWR}(t)$$

$$R_{CO}^{ULWR}(t) = R_{DR}^{ULWR}(t) + \frac{R_{DR}^{LWR}(t)}{UR} \tag{A.4.11}$$

Notice that the contribution of the uprates to fulfilling the ULWR adjustment is correctly modeled as $A_{Ui}(t) \cdot P_{AU}/3$ because for each uprate, 1.5 GWe is added by the ULWR and 1

GWe is removed due to removing an LWR from the fleet. This gives a net gain of 500 MWe for each uprate so 3 uprates are needed to satisfy the demand for a single ULWR from new construction.

If $R_{CO}^{ULWR}(t)$ was not modified to properly account for the ULWR fleet adjustment, then the $ADJ_{ULWR}(t)$ curve would fluctuate sinusoidally before finally converging to the exponential growth of the reactor demand. Although the adjustment does not affect the overall economics, the adjustment curves should match the power demand curve for results which are consistent with the model.

To account for uprates in the ULWR HLSD, the *transition ULWR construction rate* $RU_{CR\ i}^{TRA}(t)$ is set equal to the *uprate rate* of the LWRs that underwent the uprate:

$$RU_{CR\ i}^{TRA}(t) = R_{Ui-1}(t) \tag{A.4.12}$$

For example, the transition construction rate for ULWRs that are 21 years old is equal to the uprate rate of LWRs that are 20 years old. Thus, the age of the reactor is maintained through the fleet transfer.

Lastly, to account for the additional number of fresh cores needed per year due to uprated ULWRs, the *number of ULWR starting commercial operation per year* $F_N^{ULWR}(t)$ is now modeled as:

$$\frac{dF_N^{ULWR}(t)}{dt} = R_{FO}^{ULWR}(t) - O_{ULWR}(t) + \sum_i R_{Ui}(t) \tag{A.4.13}$$

where $R_{FO}^{ULWR}(t)$ is the *ULWR fulfilled order rate* and $O_{ULWR}(t)$ is a one year fixed delay applied to the fulfilled order rate.

A.4.2.5 Mass flow modification

Due to the annular fuel higher enrichment, lower mass loading, and different spent fuel composition due to higher burnup, the mass flow for ULWR fuel had to be tracked separately from that of LWR fuel. The modifications made to the Mass Flow HLSD to account for ULWR fuel are simply duplicates of all the stocks and flows in the original HLSD which accounted for LWR fuel. The only differences are the values in Table A.4.3 and the fact that all ULWRs are only loaded with traditional UO_2 fuel as opposed to the option of both UO_2 and CONFU for LWRs. CONFU fuel has not yet been proven to be adaptable to the different conditions of ULWRs so it was not adopted in the modifications.

Table A.4.3 - Fuel properties for LWR and ULWR

	LWR	ULWR
Enrichment [wt%]	4.51	8.7
Feed Enrichment [wt%]	0.711	0.711
Tails Enrichment [wt%]	0.25	0.25
Mass Loading [MT/y]	17.15	15.54
Discharge Burnup [MWd/kg]	~50	83.4
Spent Fuel Composition		
U [wt%]	92.54	90.51
Pu [wt%]	1.32	1.96
FP [wt%]	5.83	7.29
MA [wt%]	0.31	0.24
U-235 [wt% U]	2.5	1.96
Np [wt% MA]	-	72.7
Am [wt% MA]	-	19.0
Cm [wt% MA]	-	8.3

The ULWR fuel enrichment was found by averaging over all the fresh assemblies in Table 2-1 (see [Feng et al., 2008]) and the ULWR mass loading per year was determined by multiplying the LWR mass loading, which is 17.15 MT/y for 1000 MWe LWRs [Busquim et al., 2008], by the ULWR/LWR total core mass ratio (0.905). The discharge burnups were calculated from the equilibrium WS and XU core descriptions from [Xu et al, 2004], so the 1 GWe LWR discharge burnup is approximated as that of a 1150 MWe PWR. The spent fuel composition for ULWR was calculated from an annular pin model (8.7 wt% enrichment, depleted up to 83 MWd/kg) using MCODE (an MCNP-ORIGEN Depletion Program) [Xu et al., 2006]. The fuel burnup is currently not an input for CAFCA but future modifications can use burnup as well as other inputs to correlations that calculate the spent fuel vectors. For example, Table A.4.4 shows a correlation to calculate the weight fractions of the plutonium vector developed by [Xu, 2003].

Table A.4.4 - Approximate isotopic composition correlations for current PWR lattices with hydrogen-to-heavy metal ratios ~ 3.4 (from Xu, 2003)

For total plutonium:

$$\text{(kg Pu)/(kg initial heavy metal)} = 0.0193\left(\frac{B}{100}\right)^{0.565}$$

$$\text{(kg Pu)/(GWe*yr)} = \frac{70.45}{\eta\left(\dfrac{B}{100}\right)^{0.435}}$$

where   B = burnup, MWd/kg;

η = thermodynamic efficiency.

For individual isotopes:

$$\frac{X48}{X49} = \left(\frac{0.0125}{X25(0)}\right)\left(\frac{B}{100}\right)^{2.47}$$

$$\frac{X48}{X42} = 9.73\left(\frac{X25(0)}{1 - X25(0)}\right)\left(\frac{B}{100}\right)^{0.0339}$$

$$\frac{X40}{X49} = \frac{0.0354}{X25(0)}\left(\frac{B}{100}\right)^{1.01}$$

$$\frac{X41}{X49} = 1.47(1 - X25(0))\left(\frac{B}{100}\right)^{0.152} - 1$$

where   X4z = weight fraction of plutonium isotope of mass number ending in "z";

X25(0) = U-235 weight fraction in initial heavy metal.

The higher discharge burnup of the spent ULWR fuel results in a lower weight percent of uranium and subsequently a higher weight percent of fission products due to more fissions occurring during the fuel's in-core lifetime. The total Pu inventory also increases due to transmutation of U-238. Figure A.4.1 shows the mass inventory of plutonium isotopes in UO_2 fuel as a function of burnup [Xu, 2003].
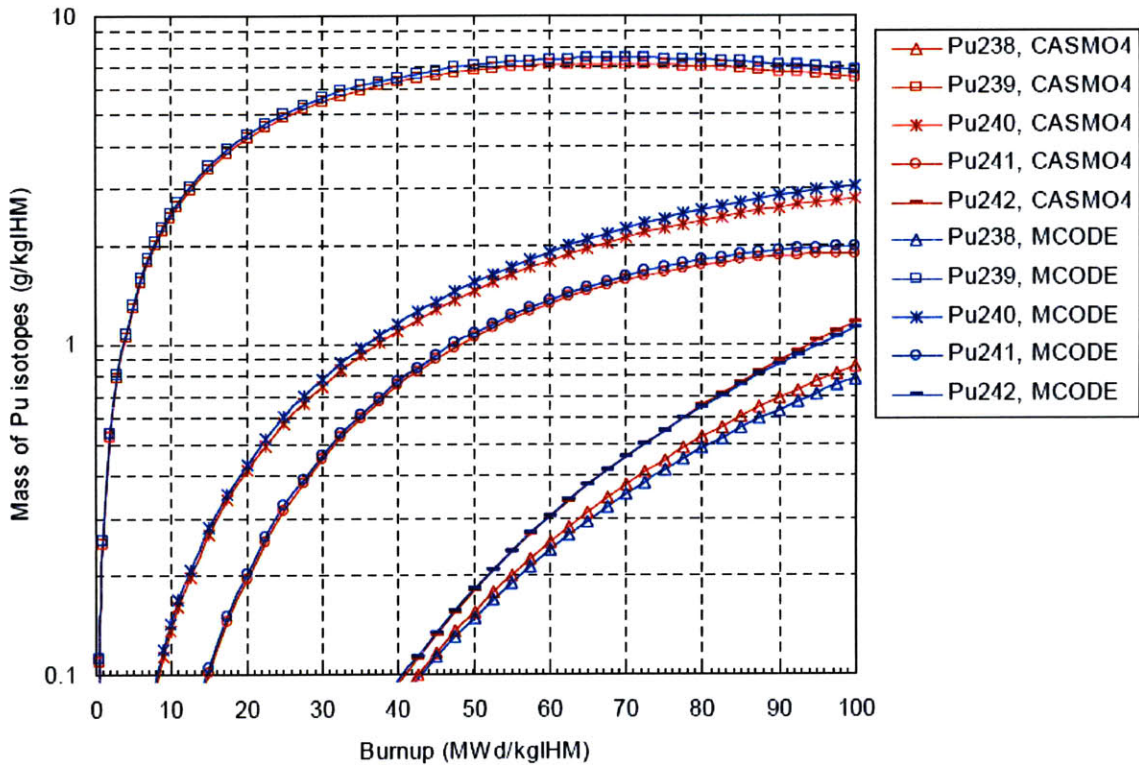
Figure A.4.1 - Plutonium vector as a function of burnup calculated by CASMO-4 and MCODE (from Xu, 2003)

A.4.2.6 Economic Update

Now that the reactor construction and mass flow HLSDs can account for ULWRs, the capital and fuel costs must also include the ULWR contributions. There are two different capital costs for ULWRs depending on how they were created, either through new ULWR construction or by uprating LWRs. The capital costs of these two pathways were based on the capital costs of options 2 and 3 calculated by Westinghouse in Table A.4.2.

A.4.2.6.1 ULWR Construction Capital Cost

The overnight construction cost (the cost of a hypothetical instantaneous construction) for a 1000 MWe LWR in as applied in the last work using CAFCA was 1700 $/kWe [Busquim et al., 2008] while in Table A.4.2, it is 1313 $/kWe (option 1) [Kazimi et al, 2006]. Thus, to calculate the overnight construction cost for ULWRs, the capital cost of option 3 (1103 $/kWe) was simply multiplied by the ratio between the costs from CAFCA and Table A.4.2 (1700/1313). This yields a value of 1428 $/kWe (approximately 1430 $/kWe) for new ULWR construction. This cost does not need to be scaled down even though the uprate in CAFCA is 500 MWe as opposed to the 600 MWe for the reference PWRs since it is in units of $/kWe. Table A.4.6 compares the values

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used in the Westinghouse study [Kazimi et al, 2006] and in CAFCA [Busquim et al., 2008].

A.4.2.6.2 ULWR uprate Capital Cost

To calculate the overnight uprate cost, option 2b from Table A.4.2 was used as the base cost of the structural components. The top number in Table A.4.5 (which is this base cost) does not include the cost of replacement power and the unused fuel so it can be designated as the 'component uprate cost'. The component uprate cost includes the costs for larger steam generators, new recirculation pumps, a new pressurizer, and additional balance of plant to accommodate the additional power. Once this component uprate cost for Westinghouse PWRs is converted into a component uprate cost used for LWRs in CAFCA, additional costs and savings are factored in to produce the total capital cost for uprates. For the aforementioned calculation (detailed in Table A.4.5), the following assumptions were made:

1.) The component uprate total cost ($828,600,000) is scaled down proportionately as the power uprate changes from 600 MWe for Westinghouse PWRs to the 500 MWe for LWRs in CAFCA.

2.) The component uprate total cost is then multiplied by the LWR cost ratio, described in Section 4.1.4.1, to obtain correct costs for use in CAFCA.

3.) The uprate construction is timed to coincide with a scheduled steam generator replacement so the cost of the steam generators ($150,000,000) can be deducted from the uprate construction total cost [Beccherle, 2007].

4.) By using a transitional refueling scheme when converting from solid to annular fuel, all of the remaining solid fuel is used before the uprate construction thus there is no extra cost for the unused fuel [Beccherle, 2007].

5.) The replacement power cost is only calculated for 9 out of the 12 months that the reactor is offline. 3 months of replacement power have already been considered for the scheduled steam generator replacement and do not contribute to the cost of the uprate [Beccherle, 2007].

6.) The cost of electricity for the replacement power was assumed to be $0.035/kWhe, approximately the average cost of electricity calculated by CAFCA in the once-through cycle for the next 30 years.

The replacement power cost was calculated as follows:

$$C_{rp} = C_{Elec} \cdot (\tau_{offline} - 2160) \cdot CF \cdot P \qquad (A.4.14)$$

where C_{Elec} is the cost of electricity ($0.035/kWhe) assumed by [Xu et al, 2004], $\tau_{offline}$ is the offline period (1 year or 8640 hours), CF is the capacity factor (0.9) assumed in CAFCA [Busquim et al., 2008], P is the electric power rating (10^6 kWe), and C_{rp} is the total cost of the replacement power, calculated to be $204.12 M. As previously mentioned, the offline period was subtracted by 2160 hours (3 months).

The ULWR uprate total capital cost in Table A.4.5 is divided by the 500 MWe of added power to obtain the ULWR uprate capital cost of 1896 $/kWe (approximated as 1900 $/kWe) in Table 4-3. This is comparable to the 1817 $/kWe as proposed by Westinghouse [Kazimi et al, 2006] even though the replacement power was calculated at a price of $0.035/kWhe compared to Westinghouse's $0.019/kWhe. The increase in cost of replacement power was almost entirely offset by using an optimized solid to annular fuel transition scheme (Beccherle, 2007) and timing the uprate with a scheduled steam generator replacement which, together, yielded savings from the cost of steam generators, unused fuel, and the 3 month replacement power credit.

Table A.4.5 - Calculation of uprate construction cost for CAFCA

$828,600,000	Component Uprate Cost (Westinghouse)
x 500/600	Westinghouse to CAFCA added power ratio
$690,500,000	
x 1700/1313	Westinghouse to CAFCA cost ratio
$894,021,325.21	Component Uprate Cost (CAFCA)
- 150,000,000	Cost of steam generators
$744,021,325.21	
+ $0	Unused fuel cost
$744,021,325	
+ $204,120,000	Nine months of replacement power (at $0.035/kWhe)
$948,131,325	**ULWR uprate total capital cost**

Table A.4.6 - Westinghouse and CAFCA capital costs in [$/kWe]

	Westinghouse	CAFCA [Busquim et al., 2008]
LWR Construction Cost	1313	1700
ULWR Construction Cost	1103	1430
ULWR Uprate Cost	1817*	1900**

* On additional 600 MWe only
** On additional 500 MWe only

A.4.2.6.3 Up to date costs

Although the costs used in CAFCA are more recent than those used in the Westinghouse study, they still require more recent updates to more accurately reflect costs in the year 2008. As of February 2008, the overnight construction cost for a Gen. III LWR is around $3,000/kWe while the overnight decommissioning costs are closer to $500/kWe for all reactors (around 40% higher than in the previous CAFCA reference values). This increase is mainly due to the increased cost of concrete and steel. The

construction costs of the reactors from Table 4-3 along with the construction costs of advanced reactors were multiplied by the new LWR cost ratio (3000/1700) and are shown in Table 4-5. The decommissioning costs were simply changed from $350 to $500/kWe.

The uprate cost, however, was calculated a little differently: only the component uprate cost was multiplied by the ratio instead of the entire uprate cost which also includes replacement power and steam generator costs. The replacement power cost was added and the steam generator costs (also multiplied by the 3000/1700 cost ratio) were deducted from this updated uprate component cost to yield a total uprate capital cost of $1.5171B as shown in Table A.4.7. Dividing this value by the power added (500 MWe) yielded approximately 3035 $/kWe as shown in Table A.4.8.

Table A.4.7 - Calculation of updated uprate construction cost for CAFCA

$828,600,000	Component Uprate Cost (Westinghouse)
x 500/600	Westinghouse to CAFCA added power ratio
$690,500,000	
x 1700/1313	Westinghouse to CAFCA cost ratio
$894,021,325.21	Component Uprate Cost (CAFCA)
x 3000/1700	Cost update
$1,577,684,692	
- $264,705,882	Cost of steam generators ($150 M x 3000/1700)
$1,312,978,810	
+ $0	Unused fuel cost
$1,312,978,810	
+ $204,120,000	Nine months of replacement power (at $0.035/kWhe)
$1,517,098,810	**ULWR uprate total capital cost**

Table A.4.8 - Cost updates for CAFCA (from Kazimi, 2008)

cost updates		
Overnight Construction Costs [$/KWe]		
	old	updated
LWR	1700	3000
ULWR (new)	1430	2520
ABR	2500	4400
FBR	2500	4400
Overnight Uprate Cost [$/KWe]		
ULWR (uprated)	1900	3035*
Overnight Decommissioning Costs [$/KWe]		
	old	new
LWR	350	500
ULWR	350	500
ABR	350	500
FBR	350	500

* On the additional 500 MWe only

A.4.2.6.4 Economic Updates in CAFCA

In updating the Capital Cost HLSD, three new cost calculation features were implemented for: 1) new ULWRs, 2) uprated ULWRs, and 3) existing LWRs that are still paying annuities. To calculate the capital cost, the **overnight construction cost** $C^{constr}_{overnight}$ (from the new costs in Table A.4.8) was multiplied by the electric power generated by reactors under 20 years old since $C^{constr}_{overnight}$ is paid during the amortization period of the plant (20 years) through annual payments (annuities) of Y_{constr}, given an effective discount rate r and tax rate on equity, τ

$$Y_{constr} = C^{constr}_{overnight} \cdot \left(\frac{e^{r \cdot T_{constr}} - 1}{r \cdot T_{constr}} \right) \cdot \frac{1}{(1-\tau)} \cdot \left(\frac{e^{r \cdot L_e} \cdot (e^r - 1)}{e^{r \cdot L_e} - 1} - \frac{\tau}{L_e} \right) \quad (A.4.15)$$

where L_e is the *amortization period*, T_{constr} is the *plant construction time*. To account for the capital costs of constructing and uprating ULWRs (features 1 and 2), the annuity for each case was calculated from the values in Table A.4.9 and multiplied by the electricity produced from the additional electric power added to the fleet.

The third implementation was necessary due to the capital cost updates in Table A.4.8; the new LWR overnight construction costs should only be applied to new reactors (presumably Gen. III) constructed after 2007. However, CAFCA SD did not make the distinction between existing and newly constructed LWRs so all existing LWRs under 20 years old would have been paying the updated construction cost annuity. To fix this discrepancy, a separate fleet of LWRs was created for the existing Generation II reactors under 20 years old. These LWRs would pay a lower construction cost annuity based on the old LWR overnight construction cost in A.4.8.

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Table A.4.9 - Capital cost values for new reactor types

Reactor Type	Overnight Construction Cost [$/kWe]	Construction Time [years]	Tax Rate	Discount Rate	Amortization Period [years]
LWR (Gen II)	1700	4	38%	7.55%	20
LWR (Gen III)	3000	4	38%	7.55%	20
ULWR (new)	2520	4	38%	7.55%	20
ULWR (uprate)	3035	1	38%	7.55%	20

A major correction made to the original CAFCA SD version was the calculation of the cost of fuel enrichment. Originally, the cost of fuel enrichment was the product between the total enriched fuel mass per year [kg/y] and the cost of enrichment [$/kg]. This has been correctly changed to the product between the *separative work unit* (SWU) per year [kg/y] and the cost of enrichment where SWU requirements are evaluated as

$$SWU_{LWR}(t) = P_{UO_2}(t) \cdot V(x_P^{LWR}) + T_{UO_2}(t) \cdot V(x_T^{LWR}) - F_U(t) \cdot V(x_F^{LWR}) \qquad (A.4.16)$$

where $P_{UO_2}(t)$ is the *mass of enriched uranium for UO₂ per year*, x_P^{LWR} is the *enrichment of the product for UO₂*, $T_{UO_2}(t)$ is the *mass of the UO₂ tails*, x_T^{LWR} is the *enrichment of the tails*, $F_U(t)$ is the *mass rate of natural uranium feed enrichment for traditional fuel per year*, x_F^{LWR} is the *enrichment of the feed for UO₂*, and *V(x)* is defined as:

$$V(x) = (2 \cdot x - 1) \cdot \ln\left(\frac{x}{1-x}\right) \qquad (A.4.17)$$

The *Uranium mass feeding the conversion process per year*, $M_{CON}(t)$, and the *Uranium mass feeding the milling process per year*, $M_{MIL}(t)$ are evaluated as

$$M_{CON}(t) = \frac{F_U(t)}{(1-L_C)} \qquad (A.4.18)$$

$$M_{MIL}(t) = \frac{M_{CON}(t)}{(1-L_M)} \qquad (A.4.19)$$

where L_M is the Uranium milling process losses, and L_C is the Uranium conversion process losses. In addition, the *mining mass rate*, $M_{MIN}(t)$, is considered as equal to $M_{MIL}(t)$.

The *cumulative demand for natural Uranium* is represented by one stock, $S_D^U(t)$. The inflow for this stock is the sum of the *mining mass rate for traditional, young CONFU*, and *old CONFU* fuels, $M_{MIN}(t) + M_{MIN}^{Young}(t) + M_{MIN}^{Old}(t)$. $S_{Do}^U(t)$ is the initial demand at time $t = 0$ [Busquim e Silva, 2008]:

$$\frac{dS_D^U(t)}{dt} = M_{MIN}(t) + M_{MIN}^{Young}(t) + M_{MIN}^{Old}(t), S_{Do}^U(t) \qquad (A.4.20)$$

A.4.2.7 Future Work

For CAFCA, it is recommended to that power uprate of PWRs be made a variable value in CAFCA from 0 to 50%, since not all plants may be interested in the 50% uprate. This would entail automatically modifying the spent fuel composition, fuel enrichment, and economics, which may require extensive reprogramming of the current code structure. Correlations may be employed to determine spent fuel vectors rather than coupling with a depletion code.

Another recommendation is to allow variation of the age at which LWRs are available for uprate. Currently it can be varied while changing a few parameters but would be more user-friendly if only one variable had to be changed.

References

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[Feng et al., 2006] D. Feng, M.S. Kazimi and P. Hejzlar, *Innovative Fuel Designs for High Power Density Pressurized Water Reactor*, MIT-NFC-TR-075, Center for Advanced Nuclear Energy Systems, MIT, 2006.

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[Xu, 2003] Z. Xu, *Design Strategies for Optimizing High Burnup Fuel in Pressurized Water Reactors*, PhD Thesis, Center for Advanced Nuclear Energy Systems, MIT, 2003.

[Xu et al., 2006] Z. Xu, P. Hejzlar, and M. S. Kazimi, *MCODE, Version 2.2-an MCNP-ORIGEN Depletion Program*, Center for Advanced Nuclear Energy Systems, MIT, 2006.

Appendix A.6.1 – Fuel cycle Service Prices Summary

Table A.6.1 Fuel Cycle Service Prices Summary

Fuel cycle service		Price
Natural Uranium Ore		100 $/kgU
Depleted Uranium		10 $/kg
Conversion (natural uranium, recovered uranium)		10 $/kg, 20 $/kg
Enrichment (natural uranium, recovered uranium)		160 $/kgSWU, 176 $/kgSWU
UO_2 fuel fabrication (natural uranium, recovered uranium)		250 $/kgHM, 268 $/kgHM
MOX fuel fabrication		2,000 $/kgHM
FFF fuel fabrication		11,600 $/kgHM
FR U-TRU fuel fabrication	CR = 1.23	2,000 $/kgHM
	CR = 1.0	2,000 $/kgHM
	CR = 0.75	2,000 $/kgHM
	CR = 0.5	2,000 $/kgHM
	CR = 0.0	2,500 $/kgHM
Spent Fuel Reprocessing		Depends on plant size and fuel type, see Table 6.15
Disposal Cost	Spent UO2 fuel	687 $/kgIHM
	Spent MOX fuel	4,580 $/kgIHM
	Fission Products	2,663 $/kgIHM
	FP/MA mix	3,251 $/kgIHM
Interim Storage		200 $/kgIHM

Appendix A.6.2 – Lead times for the CONFU cycle, ABR fuel cycle and GFR fuel cycle.

Table A.6.2.1 – Young CONFU cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$ rep.	HLW Disp.	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	0.5	0.5	2.25	2.25	5	...	1	...
UO$_2$ spent fuel reprocessing	4.25									
HLW disposal		3.25								
Fabrication			2.75							
Interim St.							-7.25			
spent CONFU reprocessing							-7.25			
HLW Disposal (Rep)							-8.25			

Table A.6.2.2 – Old CONFU cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO$_2$ rep.	HLW Disp.	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	0.5	0.5	2.25	2.25	5	...	1	...
spent Young FFF reprocessing	4.25									
HLW disposal		3.25								
Fabrication			2.75							
Interim St.							-7.25			
spent CONFU reprocessing							-7.25			
HLW Disposal (Rep)							-8.25			

Table A.6.2.3– ABR fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO₂/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	1.2	1.2	5	...	1	...
UO₂/FR spent fuel reprocessing				3.2							
HLW disposal					2.2						
Depleted U purchase					2.2						
Fabrication						1.7					
Interim St.								-6.2			
FR spent fuel reprocessing								-6.2			
HLW Disposal (Rep)								-6.2			

Table A.6.2.4 – GFR fuel cycle lead times (in years)

Fuel Cycle Stage and Duration (year)	UO₂/FR rep.	HLW Disp.	Depl. U	Fuel Fab.	(Transp.)	Irradiation		Cooling storage	Interim Storage	Rep.	Disp
	1	...	...	0.5	0.5	3.75	3.75	3	...	1	...
UO₂/FR spent fuel reprocessing				5.75							
HLW disposal					4.75						
Depleted U purchase					4.75						
Fabrication						4.25					
Interim St.								-5.75			
FR spent fuel reprocessing								-6.75			
HLW Disposal (Rep)								-6.07			

Appendix A.7.1 - TRU locations in the base case scenarios.

Total amount of TRU in the system (OTC, base case)

TRU in LWR cores ···
TRU in FR cores ···
TRU in fuel fabrication plants ·······················
TRU in cooling storages ·······························
TRU in interim storage and reprocessing plants ·······
TRU in wastes ···

Figure A.7.1.1 – Total amount of TRU in the system (OTC, base case) 2008-2108

Total amount of TRU in the system (OTC, base case)

TRU in LWR cores ···
TRU in FR cores ···
TRU in fuel fabrication plants ·······················
TRU in cooling storages ·······························
TRU in interim storage and reprocessing plants ·······
TRU in wastes ···

Figure A.7.1.2 – Total amount of TRU in the system (OTC, base case) 2008-2058
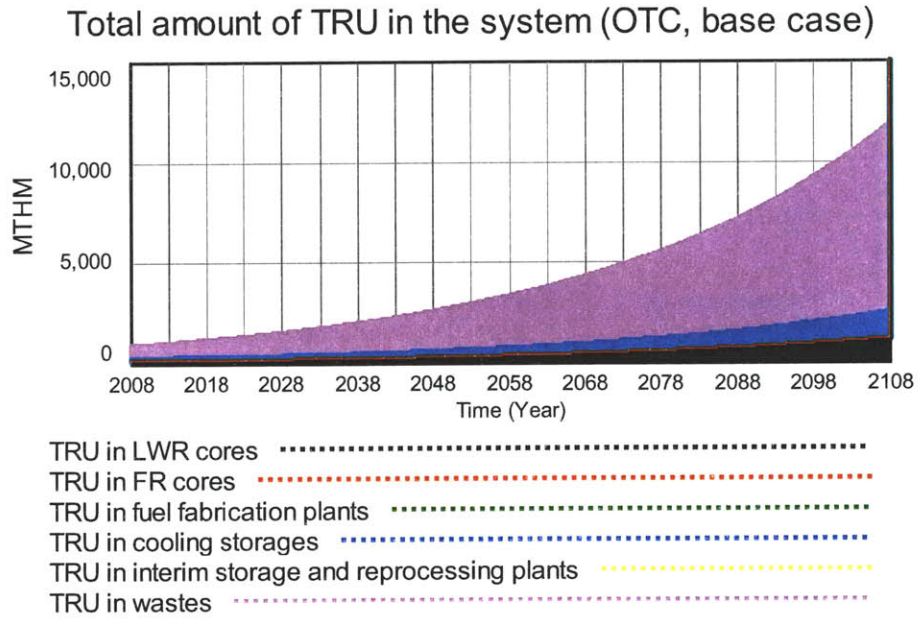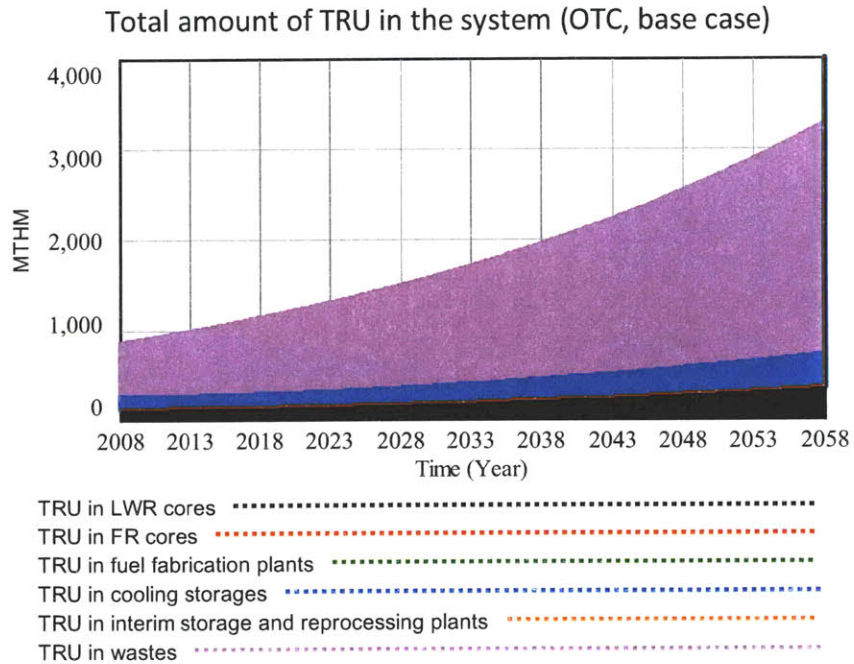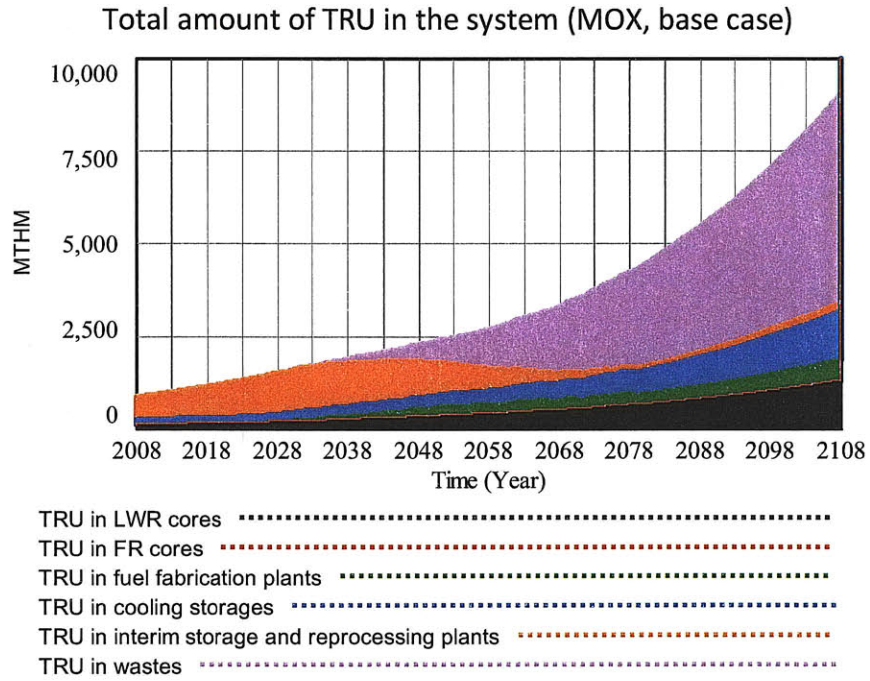
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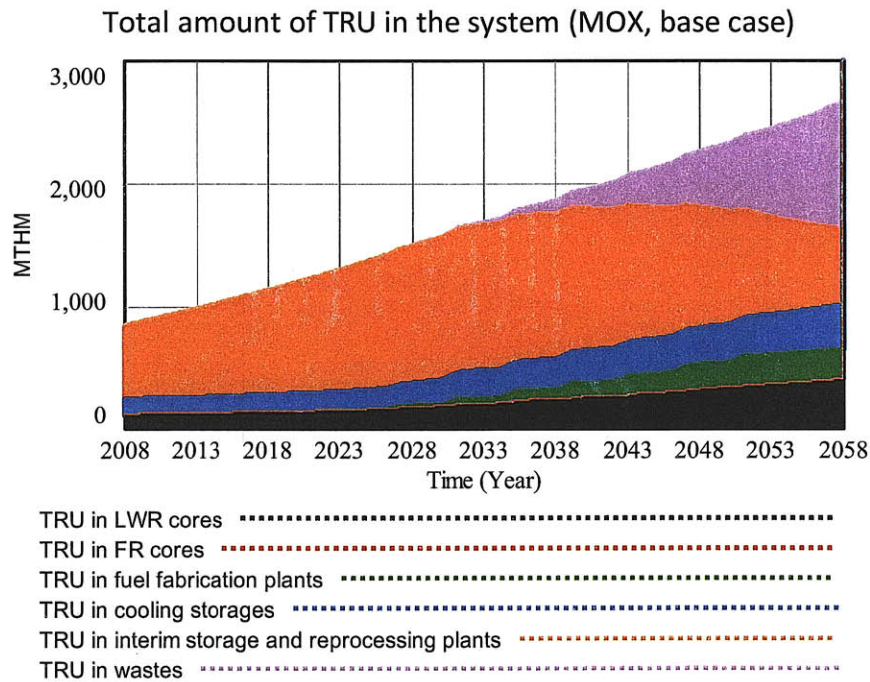**Figure A.7.1.3 – Total amount of TRU in the system (MOX, base case) 2008-2108**

Figure A.7.1.4 – Total amount of TRU in the system (MOX, base case) 2008-2058
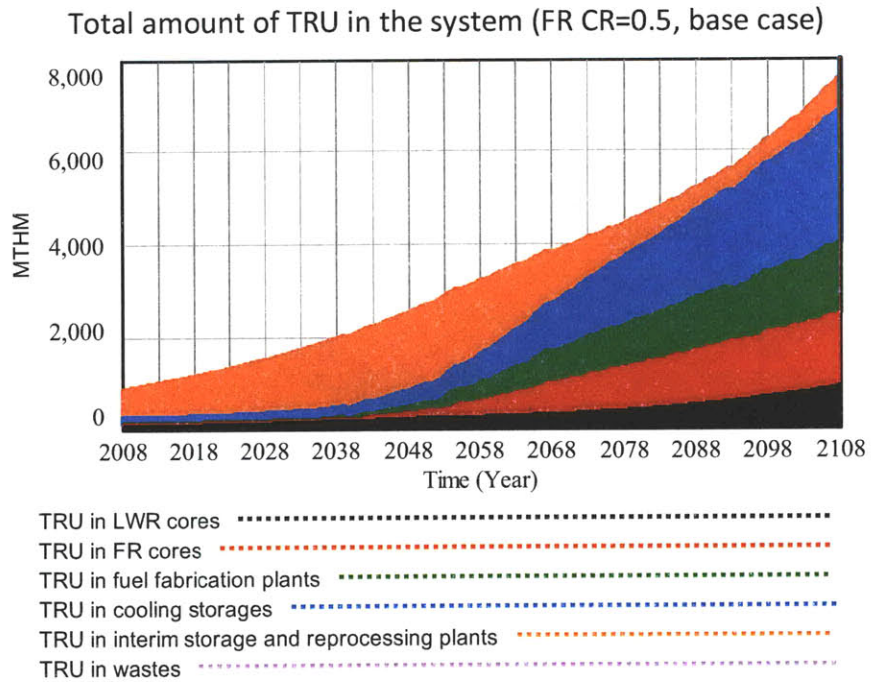
Figure A.7.1.5 – Total amount of TRU in the system (FR CR=0.5, base case) 2008-2108
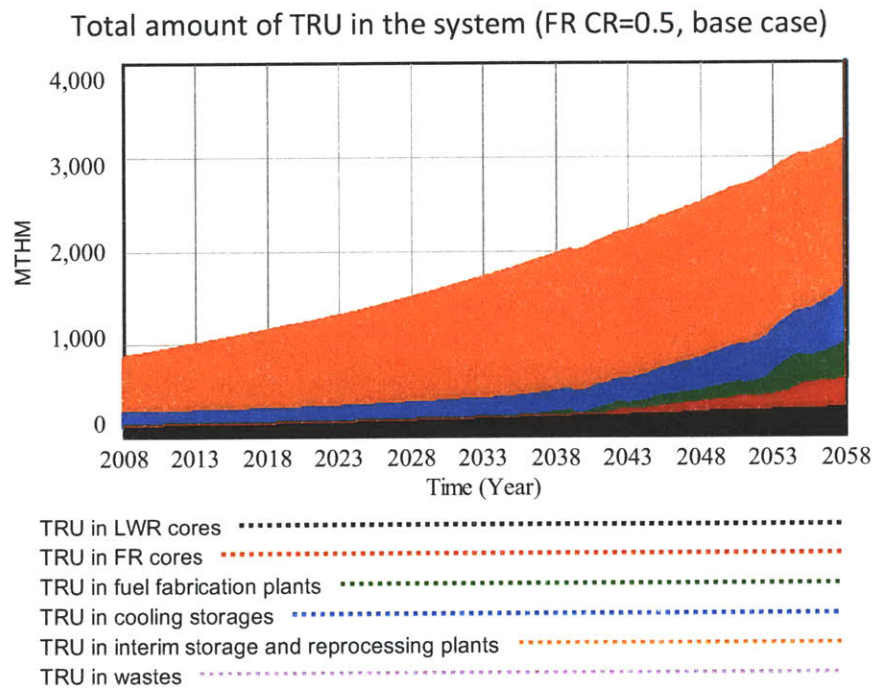

Figure A.7.1.6 – Total amount of TRU in the system (FR CR=0.5, base case) 2008-2058
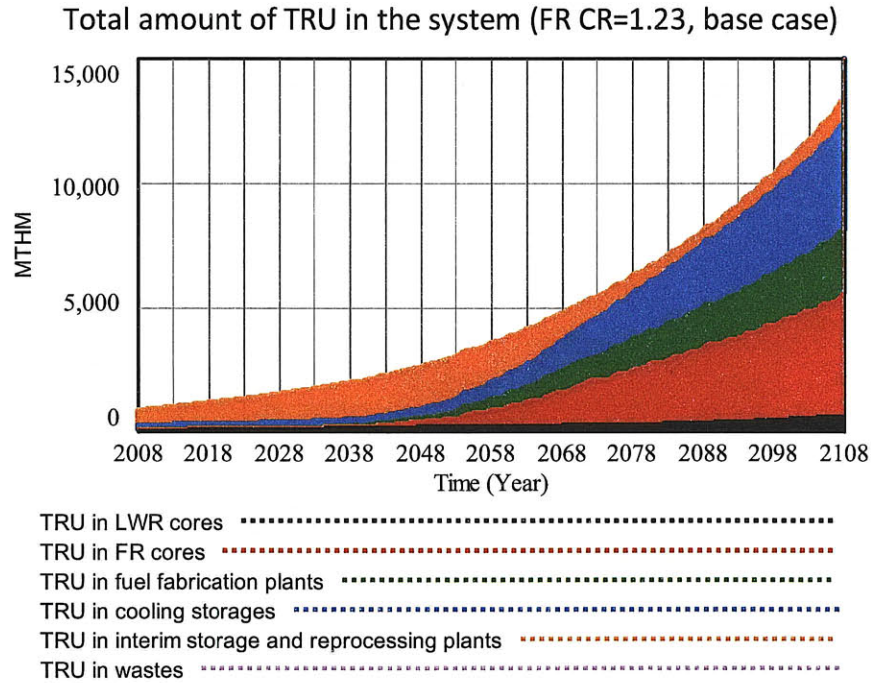
Total amount of TRU in the system (FR CR=1.23, base case)

TRU in LWR cores ··
TRU in FR cores ··
TRU in fuel fabrication plants ··
TRU in cooling storages ··
TRU in interim storage and reprocessing plants ··
TRU in wastes ··

Figure A.7.1.7 – Total amount of TRU in the system (FR CR=1.23, base case) 2008-2108

Total amount of TRU in the system (FR CR=1.23, base case)

TRU in LWR cores ··
TRU in FR cores ··
TRU in fuel fabrication plants ··
TRU in cooling storages ··
TRU in interim storage and reprocessing plants ··
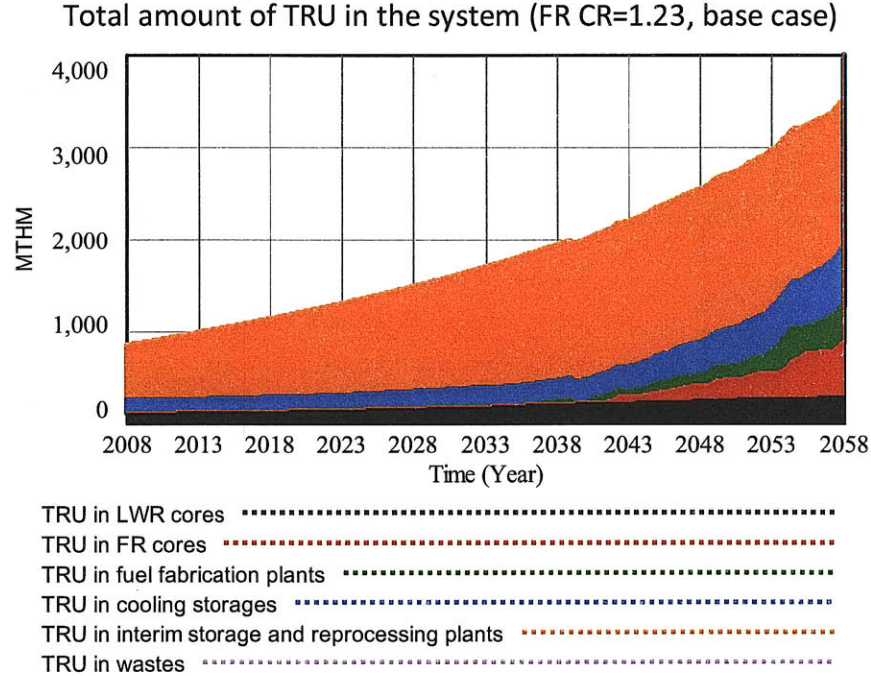TRU in wastes ··

Figure A.7.1.8 – Total amount of TRU in the system (FR CR=1.23, base case) 2008-2058
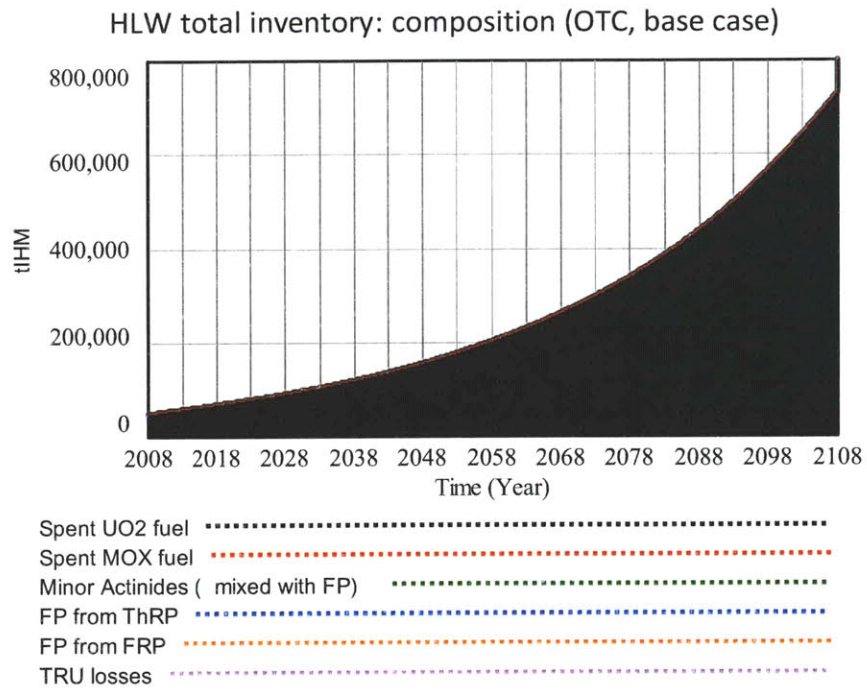
Appendix A.7.2 - HLW inventories, composition

HLW total inventory: composition (OTC, base case)

Spent UO2 fuel ●●●●●●●●●●●●●●●●●●●●●●●●●●●●
Spent MOX fuel ────────────────────────
Minor Actinides (mixed with FP) ●●●●●●●●●●●●●●●●●●
FP from ThRP ●●●●●●●●●●●●●●●●●●●●●●●●●●
FP from FRP ●●●●●●●●●●●●●●●●●●●●●●●●●●
TRU losses ·····················

Figure A.7.2.1 – HLW inventories and composition (OTC, base case) 2008-2108

HLW total inventory: composition (OTC, base case)

Spent UO2 fuel ●●●●●●●●●●●●●●●●●●●●●●●●●●●●
Spent MOX fuel ────────────────────────
Minor Actinides (mixed with FP) ●●●●●●●●●●●●●●●●●●
FP from ThRP ●●●●●●●●●●●●●●●●●●●●●●●●●●
FP from FRP ●●●●●●●●●●●●●●●●●●●●●●●●●●
TRU losses ·····················

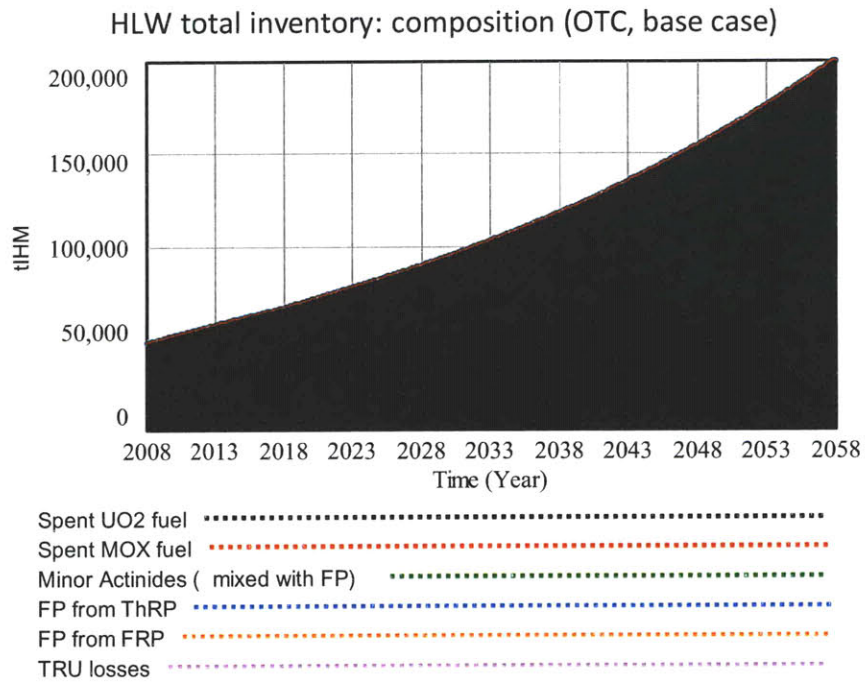*Figure A.7.2.2 – HLW inventories and composition (OTC, base case) 2008-2058*

HLW total inventory: composition (MOX, base case)

Figure A.7.2.3 – HLW inventories and composition (MOX, base case) 2008-2108

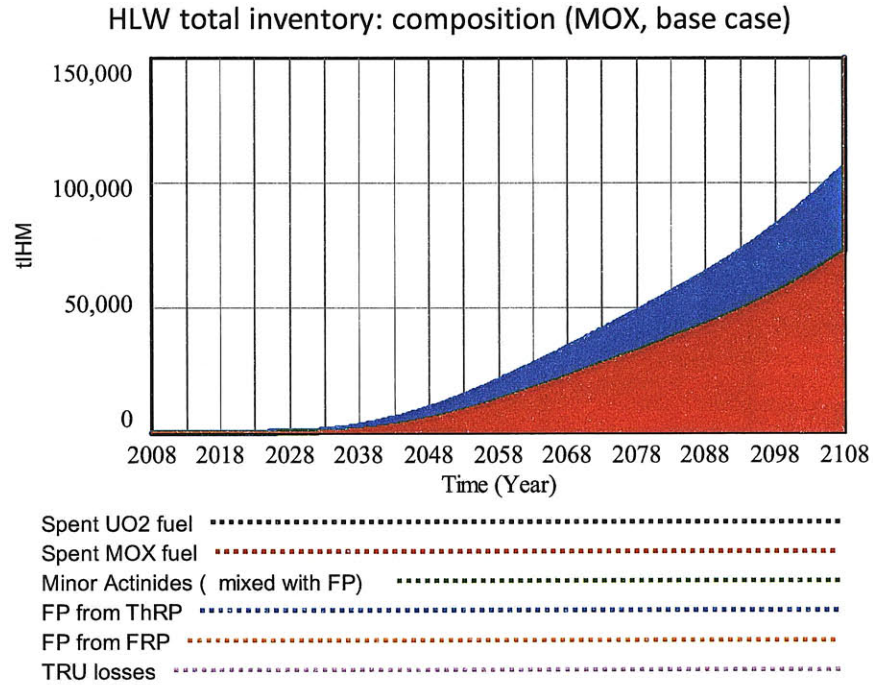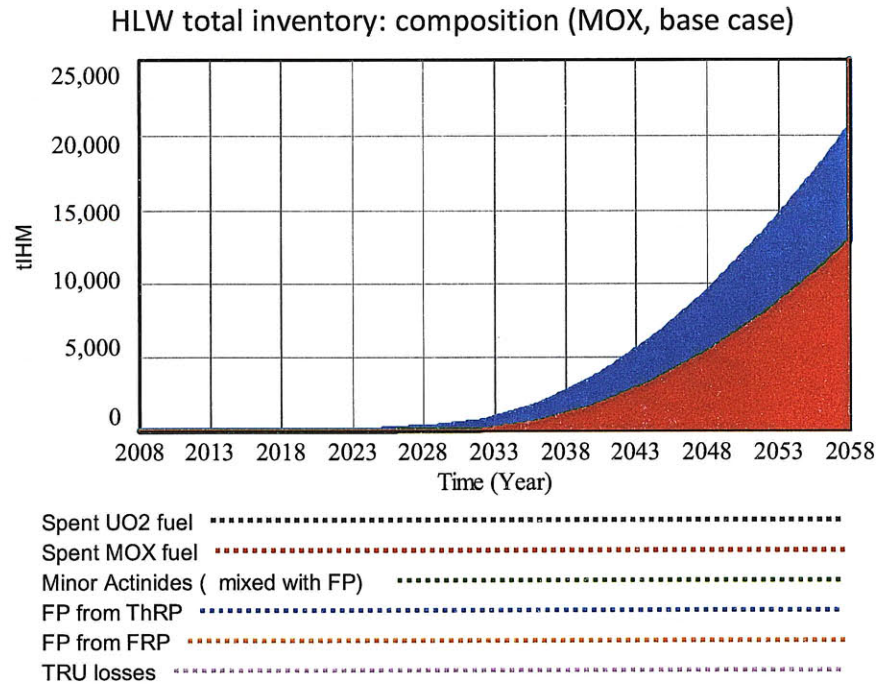HLW total inventory: composition (MOX, base case)

Figure A.7.2.4 – HLW inventories and composition (MOX, base case) 2008-2058

HLW total inventory: composition (FR CR=0.5, base case)

Spent UO2 fuel
Spent MOX fuel
Minor Actinides (mixed with FP)
FP from ThRP
FP from FRP
TRU losses

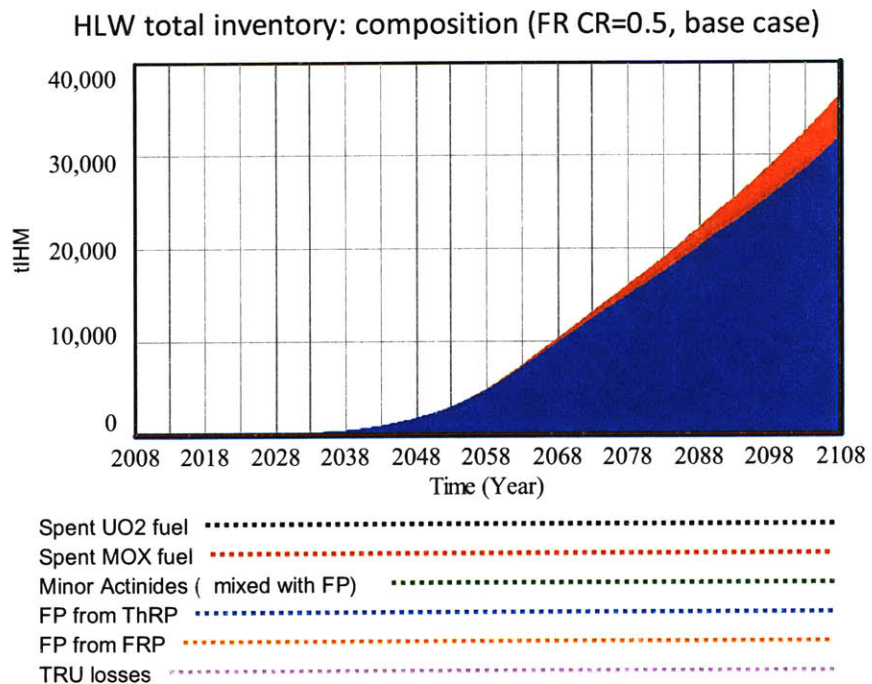*Figure A.7.2.5 – HLW inventories and composition (FR CR=0.5, base case) 2008-2108*

HLW total inventory: composition (FR CR=0.5, base case)

Spent UO2 fuel
Spent MOX fuel
Minor Actinides (mixed with FP)
FP from ThRP
FP from FRP
TRU losses

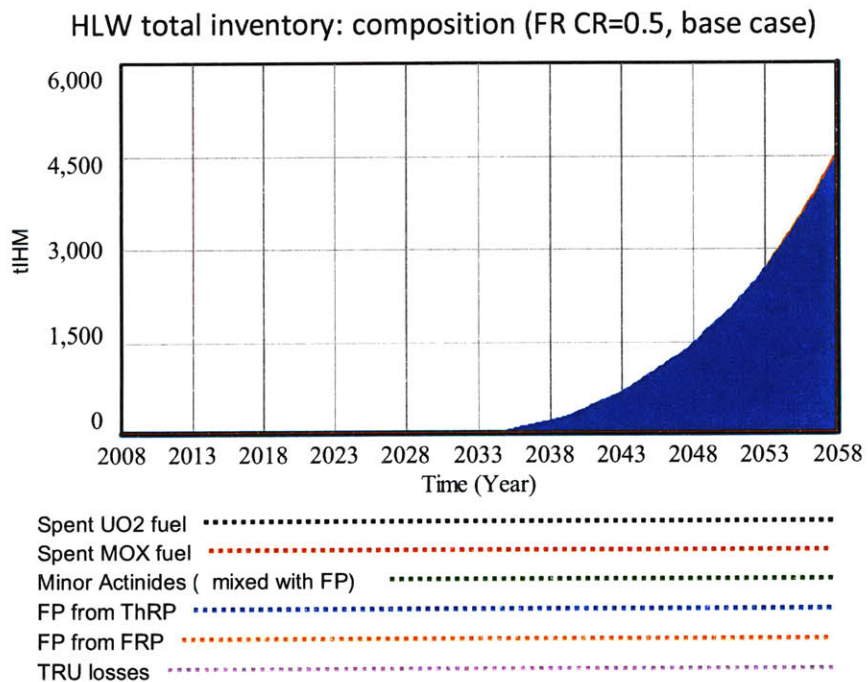*Figure A.7.2.6 – HLW inventories and composition (FR CR=0.5, base case) 2008-2058*

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HLW total inventory: composition (FR CR=1.23, base case)

Spent UO2 fuel ·········
Spent MOX fuel ---------
Minor Actinides (mixed with FP) ·········
FP from ThRP ··········
FP from FRP ---------
TRU losses ···········

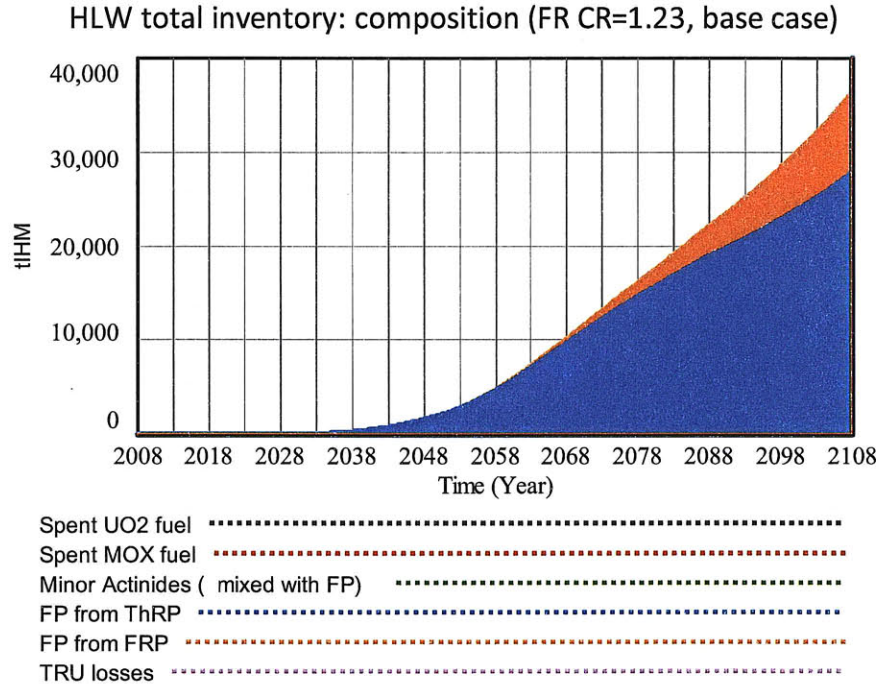*Figure A.7.2.7 – HLW inventories and composition (FR CR=1.23, base case) 2008-2108*

HLW total inventory: composition (FR CR=1.23, base case)

Spent UO2 fuel ·········
Spent MOX fuel ---------
Minor Actinides (mixed with FP) ·········
FP from ThRP ··········
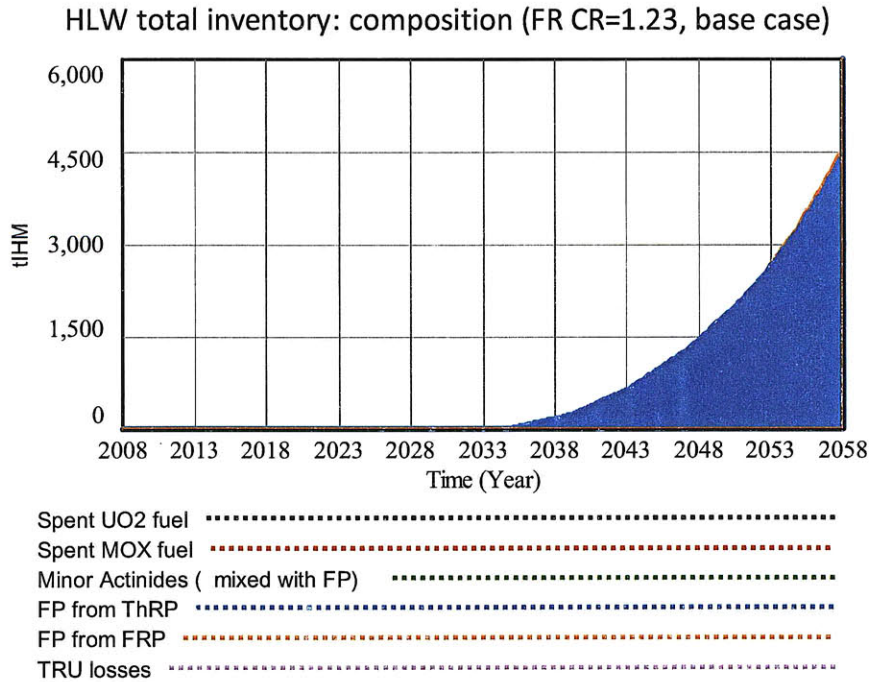FP from FRP ---------
TRU losses ···········

Figure A.7.2.8 – HLW inventories and composition (FR CR=1.23, base case) 2008-2058

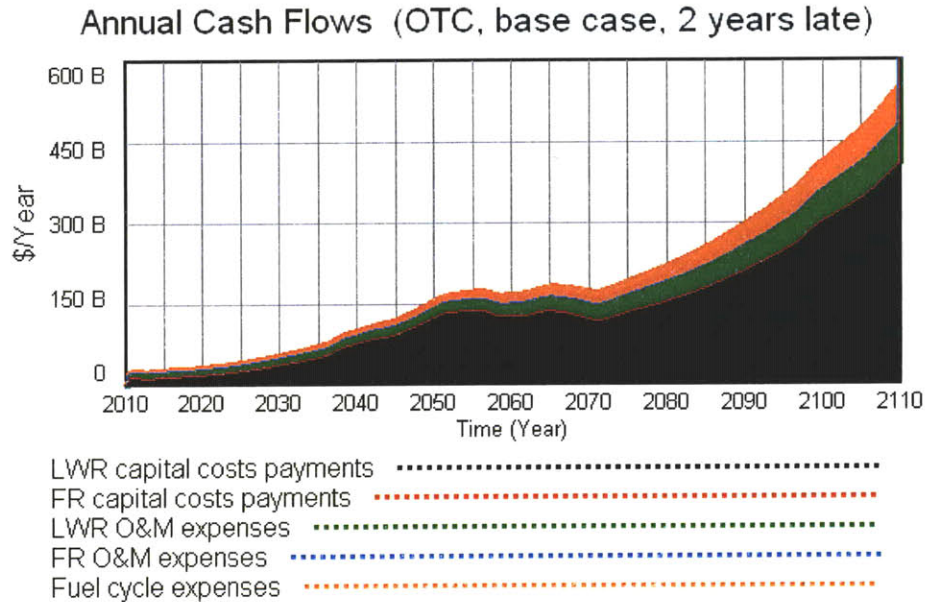## Appendix A.7.3 - Annual Cash Flows in the base case scenarios

Annual Cash Flows (OTC, base case, 2 years late)

LWR capital costs payments   ••••••••••••••••••••••••••••••
FR capital costs payments   •••••••••••••••••••••••••••••
LWR O&M expenses   ••••••••••••••••••••••••••••••
FR O&M expenses   ••••••••••••••••••••••••••••••
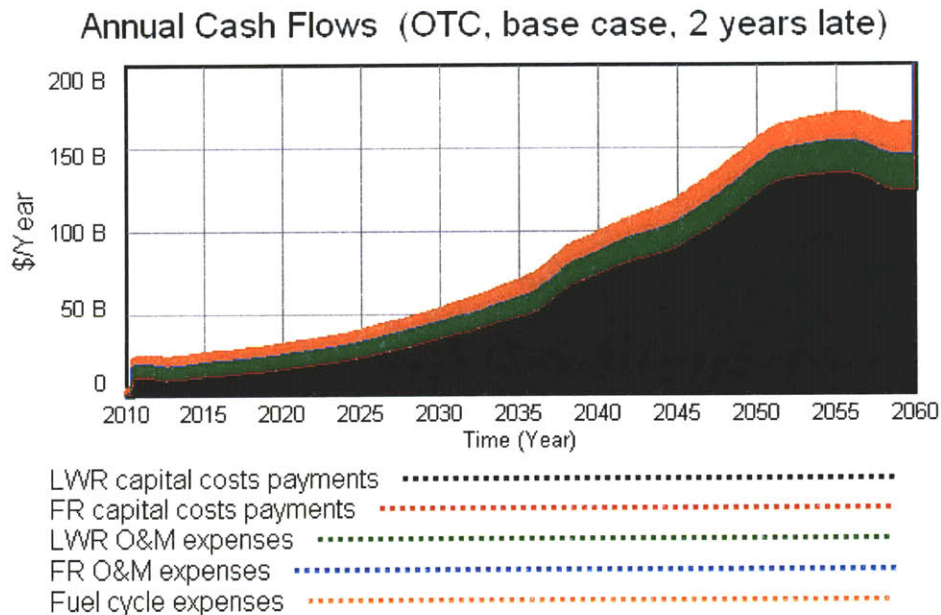Fuel cycle expenses   ••••••••••••••••••••••••••••••

Figure A.7.3.1 – Composition of the total cash flow (OTC, base case) 2008-2108

Annual Cash Flows (OTC, base case, 2 years late)

LWR capital costs payments   ••••••••••••••••••••••••••••••
FR capital costs payments   •••••••••••••••••••••••••••••
LWR O&M expenses   ••••••••••••••••••••••••••••••
FR O&M expenses   ••••••••••••••••••••••••••••••
Fuel cycle expenses   ••••••••••••••••••••••••••••••

Figure A.7.3.2 – Composition of the total cash flow (OTC, base case) 2008-2058

Annual Cash Flows (TTC, base case, 2 years late)

LWR capital costs payments
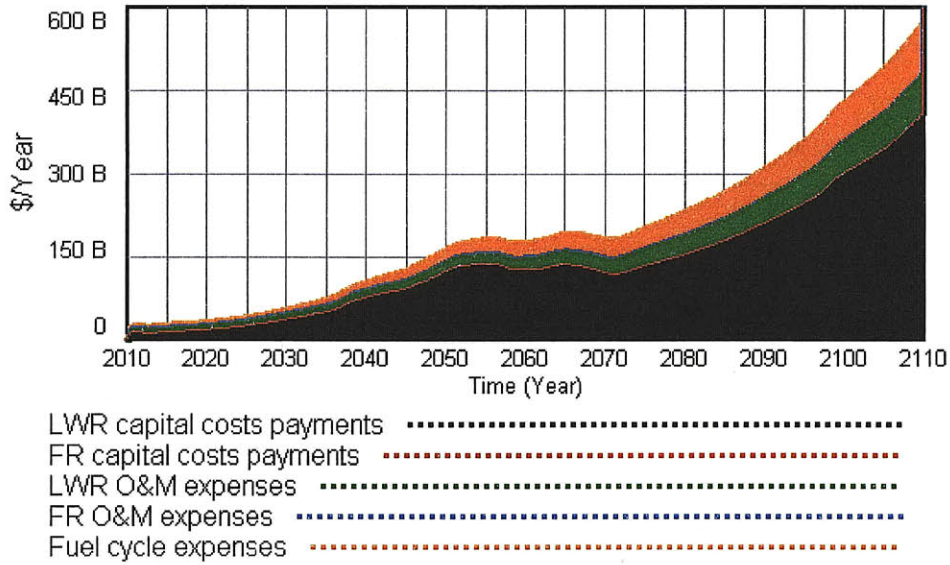FR capital costs payments
LWR O&M expenses
FR O&M expenses
Fuel cycle expenses

Figure A.7.3.3 – Composition of the total cash flow (MOX, base case) 2008-2108

Annual Cash Flows (TTC, base case, 2 years late)

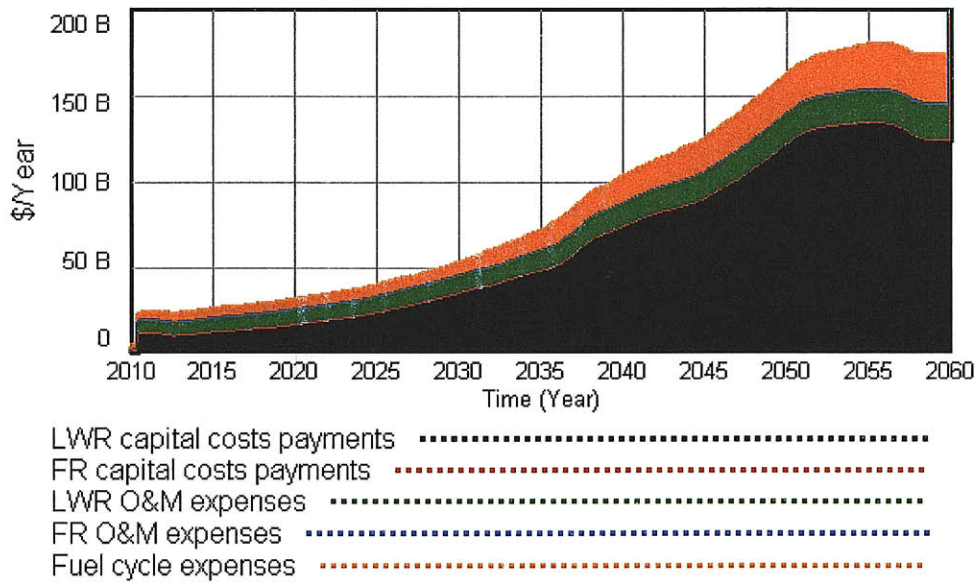LWR capital costs payments
FR capital costs payments
LWR O&M expenses
FR O&M expenses
Fuel cycle expenses

Figure A.7.3.4 – Composition of the total cash flow (MOX, base case) 2008-2058

Annual Cash Flows (FR CR=0.5, base case, 2 years late)

LWR capital costs payments ·······················
FR capital costs payments ·······················
LWR O&M expenses ·······················
FR O&M expenses ·······················
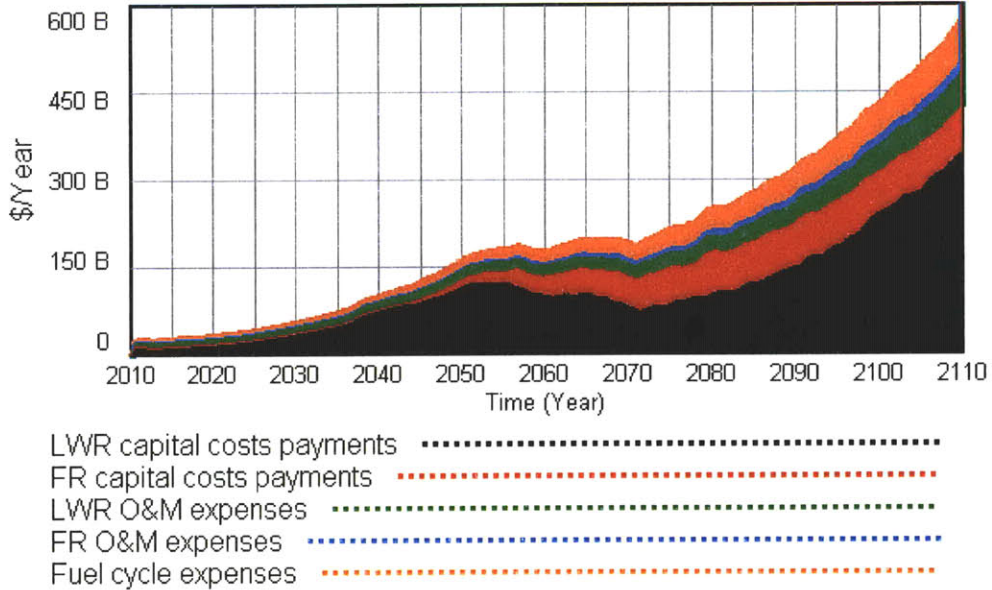Fuel cycle expenses ·······················

Figure A.7.3.5 – Composition of the total cash flow (FR CR=0.5, base case) 2008-2108

Annual Cash Flows (FR CR=0.5, base case, 2 years late)

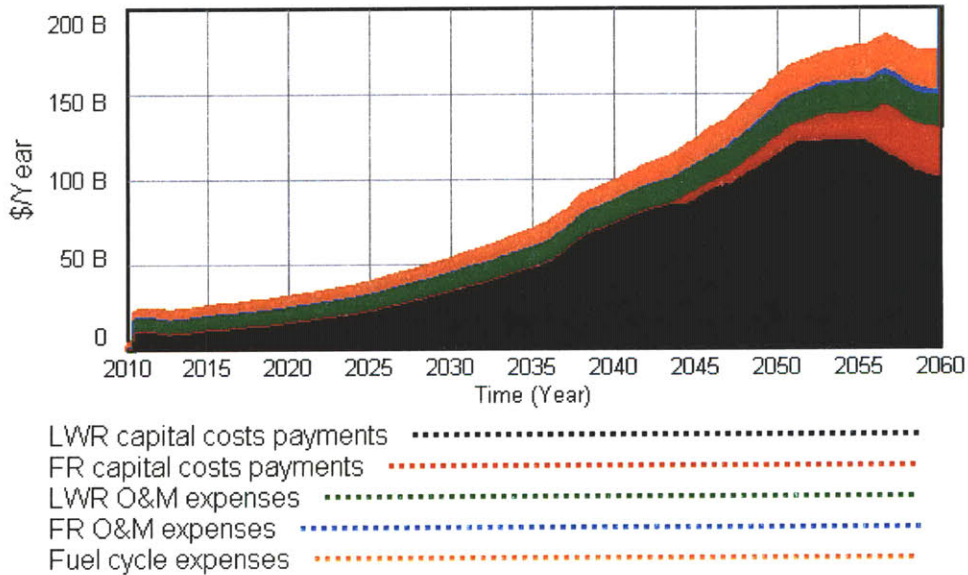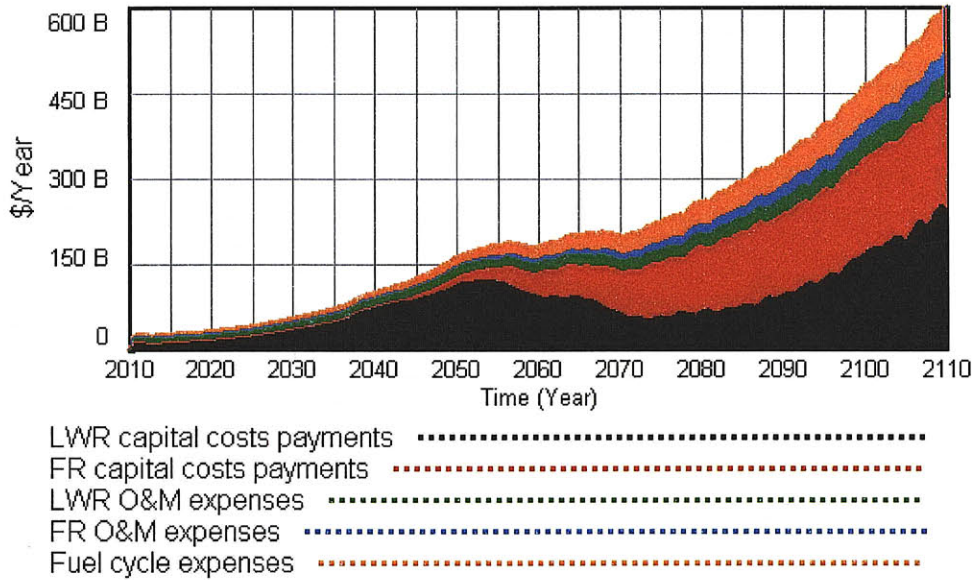LWR capital costs payments ·······················
FR capital costs payments ·······················
LWR O&M expenses ·······················
FR O&M expenses ·······················
Fuel cycle expenses ·······················

Figure A.7.3.6 – Composition of the total cash flow (FR CR=0.5, base case) 2008-2058

Figure A.7.3.7 – Composition of the total cash flow (FR CR=1.23, base case) 2008-2108
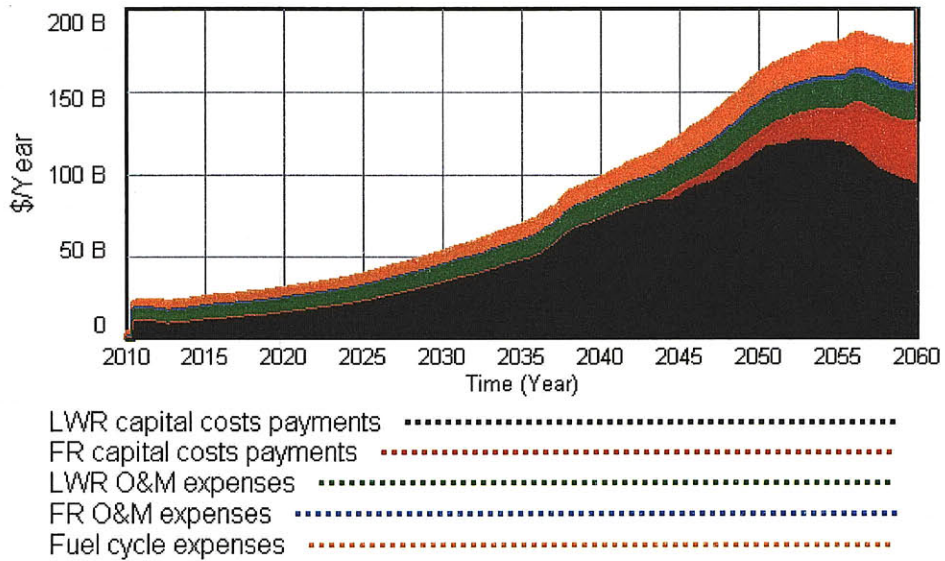

Figure A.7.3.8 – Composition of the total cash flow (FR CR=1.23, base case) 2008-2058