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Analysis of Transition Nuclear Fuel Cycles from LWRs to Gen-IV Reactors

Ph.D. Thesis

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Abstract

The present work is a contribution to the analysis of advanced nuclear fuel cycles and of their impact in terms of natural resources, radioactive waste management and of infrastructures that these cycles would require for specific world regions, or for a specific country.

The activity has been developed to get an overview about the challenges, implications, boundary conditions, assumptions and correlated consequences of different scenarios suggested or envisaged for the deployment of nuclear energy (mainly for electricity production) on a global or regional basis.

In this context, the implications of the LWRs utilization (in terms of resources availability, e.g. natural uranium consumption, fuel cycle facilities, e.g. plant for uranium enrichment, waste production, etc.) in particular in countries that decide a first deployment of nuclear energy and the possible transition to Fast Reactors (FRs) (in terms of Pu availability, enlargement of reprocessing capacities, etc.) have been evaluated.

In order to investigate the sustainability of the nuclear option in a country interested in developing nuclear energy in isolation, several scenarios have been compared under dynamic evolution conditions.

The analysis of the whole fuel cycle has been performed by means of the COSI6 code, a dynamic scenario and fuel cycle code developed at CEA (France).

A reference scenario has been defined in order to be representative of more complex scenarios (e.g. European scenarios). The reference scenario results formed the basis to handle more complex situations, e.g. a geographical region where different countries have different strategies (restarting, continuing, phasing-out) with respect to nuclear energy deployment (e.g. similar to the European situation).

The "once-through" strategy, where only Light Water Reactors (LWRs) and uranium fuel are deployed to cover the energy demand, has been assumed in the reference scenario for the period 2020-2200. This seems the most realistic case, in terms of technology readiness.

In order to optimize the use of the resources and to reduce drastically the radioactive wastes produced (in terms of quantity, repository capacity and potential risks), the introduction of advanced fuel cycles that envisage the transition from LWRs to FRs has been analyzed: by this approach the Pu from LWRs is considered to be a resource and is recovered by reprocessing the spent fuel to be (after refabrication) successively used as new fuel.

In order to quantify the uncertainties and validity range of the scenario study results, parametric studies have been performed where the values of some important parameters (nuclear energy demand evolution, pace of deployment of new reactors, core burn-up, irradiated fuel cooling times etc.) have been varied within realistic ranges. For each parameter, separate effects on the selected fuel cycle performance indicators (radiotoxicity in a deep geological storage, heat load at fuel discharge and in a repository, resource consumption/utilization and waste inventories) have been quantified.

Since radioactive waste management is a key issue for the acceptability of nuclear energy, scenarios that go beyond the separation and use of Pu have been examined. In fact in these scenarios, fission products and all the so-called "Minor Actinides" (MAs) are sent to a repository. To further reduce both radiotoxicity and heat load of the waste to be stored for disposal, the additional separation and transmutation (P&T) of the

MAs offers a potential for enhancing the reduction of both quantities. For this purpose, scenarios based on several advanced FRs (with different core configurations and coolants) have been compared in terms of fuel cycle performances in particular when the MAs multi-recycling in FRs would have been implemented.

The results obtained allow to underline some important features and impacts of future nuclear fuel cycles. If safety will stay an absolute priority, the sustainability in terms of resources optimization, waste reduction and proliferation risk resistance will be and remain key objectives of any nuclear energy deployment policy and this study provides some, even if modest, contribution to understand technology requirements and potentially more favorable options. Economy is also a key factor: the study did not attempt any economic evaluation at the present stage. That area is certainly a priority field for future research on the basis of prevailing legislation and cost.

The present Ph.D. activity has been developed, since 2008, in collaboration between the University of Pisa (UNIPI, Italy) and the Karlsruhe Institute of Technology (KIT, Germany).

Sommario

Nel presente lavoro sono stati condotti studi di cicli di combustibile innovativi con lo scopo di valutarne il relativo impatto in termini di uso delle risorse naturali, ammontare di rifiuti radioattivi ed infrastrutture necessarie. Queste valutazioni sono state estese a condizioni energetiche locali, regionali e globali.

Gli studi sono stati effettuati in modo tale da fornire una panoramica quanto più estesa possibile circa le sfide, le implicazioni, le condizioni al contorno, le ipotesi e le relative conseguenze associate alle analisi di scenari per lo sviluppo dell'energia nucleare su scala globale o regionale.

In particolare, sono state valutate le implicazioni riguardanti l'uso dei reattori termici (per quanto riguarda le risorse coinvolte, gli impianti del ciclo del combustibile, i rifiuti prodotti, etc.) e una possibile transizione verso l'uso dei reattori veloci (in termini di disponibilità di Pu nel ciclo, ampliamento delle capacità di riprocessamento, etc.).

Una valutazione della sostenibilità dell'opzione nucleare (attraverso l'analisi di indicatori selezionati), è stata effettuata confrontando vari scenari tra loro tenendo conto dell'evoluzione dinamica delle condizioni al contorno.

Nella prima fase di questo studio è stato usato il codice IAEA NFCSS che ha permesso di definire una metodologia semplificata per la quantificazione delle condizioni al contorno. Per la seconda parte il codice di scenario COSI6 sviluppato dal CEA-Cadarache (Francia) è stato adottato a riferimento per le analisi dinamiche dei cicli di combustibile.

Uno scenario di riferimento con precise condizioni al contorno è stato definito e analizzato. I risultati ottenuti per questo scenario di riferimento formano la base per studi più complessi quali ad esempio studi regionali con diverse strategie per quanto rigurada l'uso dell'energia nucleare. L'opzione "once-through", basata sull'uso dei soli reattori termici refrigerati ad acqua (Light Water Reactors, LWRs) e di combustibili a base di uranio, è stata presa a riferimento per il periodo 2020-2200. Questa strategia coincide con la condizione più realistica tenuto conto delle tecnologie oggi disponibili.

Per ottimizzare l'uso delle risorse naturali (uranio) e ridurre i rifiuti radioattivi prodotti (per quanto riguarda quantità, potenziale rischio e ottimizzazione della capacità del deposito), sono stati studiati cicli di combustibile innovanti che permettano una transizione dai LWRs ai reattori veloci. Attraverso questo approccio, il Pu prodotto nei LWRs viene considerato una risorsa.

Per fornire una valutazione quantitativa delle incertezze e del range di validità degli studi di scenario, è stato sviluppato uno studio parametrico. Parametri importanti quali la domanda energetica nucleare, il rateo di introduzione dei reattori, il burn-up, etc., sono stati variati entro range realistici. Per ciascun parametro, è stato valuato l'effetto sugli indicatori selezionati per analizzare parametri di sostenibilità del ciclo del combustibile (evoluzione della radiotossicità e del calore di decadimento associato al combustibile esausto, consumo/utilizzo delle risorse e quantificazione dei rifiuti).

Il trattamento dei rifiuti è uno dei punti chiave per l'accettabilità dell'energia nucleare, per questo motivo scenari alternativi alla sola separazione ed uso del Pu sono stati studiati. Negli scenari con il solo riciclo del Pu, i prodotti di fissione e tutti gli attinidi minori sono inviati al deposito. Per ridurre, la radiotossicità e il calore di decadimento associato ai rifiuti, una soluzione percorribile risulta essere la separazione e la

trasmutazione (P&T) degli attinidi minori, cioè Np, Am, e Cm. Per questo motivo, scenari basati su vari reattori veloci innovativi (con differenti refrigeranti e configurazioni del core) sono stati confrontati come anche il multi-riciclo in reattori veloci degli attinidi minori.

I risultati ottenuti permettono di evidenziare aspetti importanti dei futuri cicli del combustibile. Seppure la sicurezza rimarrà un'assoluta priorità, la sostenibilità in termini di ottimizzazione delle risorse, riduzione dei rifiuti prodotti e resistenza alla proliferazione sono obiettivi chiave per lo sviluppo dell'energia nucleare. Con il presente studio si vuole fornire un contributo all'analisi delle necessità tecnologiche associate alle opzioni più potenzialmente favorevoli. L'aspetto economico è un altro fattore chiave seppur non analizzato in dettaglio in questa fase dello studio. Questa area è certamente un aspetto importante da implementare in studi futuri.

Il presente lavodo di Dottorato è stato sviluppato, a partire dal 2008, presso l'Università of Pisa (UNIPI, Italia) in collaborazione con il Karlsruhe Institute of Technology (KIT, Germania).

Contents

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Abbreviations

UNFCCC	United Nations Framework Conventions on Climate Change
OECD	Organization for Economic Co-operation and Development
EU	European Union
EC	European Commission
IAEA	International Atomic Energy Agency
IIASA	International Institute for Applied Systems Analysis
IPCC	International Panel on Climate Change
IEA	International Energy Agency
NEA	Nuclear Energy Agency
KIT	Karlsruhe Institute of technology
MIT	Massachusetts Institute of Technology
CEA	Commissariat à l'Energie Atomique
SNE-TP	Sustainable Nuclear Energy Technology Platform
SET	Strategic Energy Technology plan
SRA	Strategic Research Agenda
WPFC	Working Party on Scientific Issues of the Fuel Cycle
DOE	Department of Energy
RD&D	Research Development and Demonstration
R&D	Research and Development
P&T	Partitioning and Transmutation
PRIS	Power Reactor Information System
INPRO	International Project on Innovative Nuclear Reactors and Fuel Cycles
NFCSS	Nuclear Fuel Cycle Simulation System
GHG	Greenhouse Gas
CCS	Carbon Capture and Storage
CSS	Capture Store and Sequestration
LCA	Life Cycle Assessment
USD	US dollar
efpd	effective full power day
NPP	Nuclear Power Plant
LWRs	Light Water Reactors
FRs	Fast Reactors
PWRs	Pressurized Water Reactors
EPR	European Pressurized reactors
BWRs	Boiling Water Reactors
LFR	Lead-cooled Fast Reactor

SFR	Sodium-cooled Fast Reactor
GFR	Gas-cooled Fast reactor
FBR	Fast Breeder Reactor
ADS	Accelerator Driven System
MYRRHA	Multi-purpose hYbrid Research Reactor for High-tech Applications
ESFR	European Sodium Fast Reactor
ELSY	European Lead cooled System
EFR	European Fast Reactor
HWR	Heavy Water Reactor
HTGR	High Temperature Gas Reactor
PHWR	Pressurized Heavy Water Reactors
MOX	Mixed Oxide
UOX	Uranium Oxide
MAs	Minor Actinides
FPs	Fission Products
TRUs	Transuranics
HM	Heavy Metal
SF	Spent Fuel
UNF	Used Nuclear Fuel
HLW	High Level Waste
ILW	Intermediate Level Waste
LLW	and Low Level Waste
RAR	Reasonably Assured Resources
IR	Inferred Resources
CR	Conversion Ratio
BR	Breeding Ratio
BG	Breeding Gain
CDT	Composite Doubling Time
DT	Doubling Time
IR	Identified Resources
CDA	Core Disruptive Accidents
DBE	Design Basic Earthquake
CMR	Controlled Material Relocation
SA	Subassembly
BOL	Beginning of Life
EOL	End of Life
EBL	End of Blanket Life
BOC	Beginning of Cycle
EOC	End of Cycle

Chapter 1

Introduction

It is recognized that the world energy system is in front of a crossroad. The global trends of the energy demand and supply as well as the economic crises of the last two years have indicated that the development of industrialized countries is not sustainable [42, 43].

The primary energy mix based on fossil fuels (81% of the total where 34% is supplied by oil) has a) limitations for long-term security of supply, and b) a potentially significant environmental impact (greenhouse effect, global warming, climate change).

These two issues, together with the foreseen increase of primary energy demand, are the driving forces toward a drastic evolution of the global energy system.

To mitigate the present trends, more sustainable solutions are required. Main actions are oriented to reduce the environmental impact (e.g. by the stabilization of the greenhouse gas emissions), to guarantee the long-term energy supply and to extend the access to electricity to one fifth of the world population currently with practically no access. A delay in the implementation of those solutions will result in higher concerns for the environmental protection and in higher investments to invert the trend during the coming years [42, 43].

The timely actions of Governments as well as common worldwide strategy agreements are the main challenges for the success of the process as indicated by the United Nations Framework Conventions on Climate Change (UNFCCC) hold in Copenhagen in 2009 for the negotiation of the post-Kyoto treaty [43].

In the last decade, OECD countries have taken some actions in order to help changing the present trends [43].

Another example is the definition of the European short-term energy strategy where several objectives for 2020 have been fixed by the European Union (EU) [44].

The effects of these measures, mainly based on the improvement of the end-user efficiency for reducing the CO_2 emissions associated to the energy sector, are encouraging but not sufficient. Moreover, the role of OECD countries and the impact of the measures taken can be marginal in terms of global trends when compared to the role of growing economies (e.g. China and India) and to the impact of the strong increase of energy demand in these areas.

According to [43], this situation can result, for the short-term, in no substantial changes of the actual trends. Only a coordinate effort of OECD and non-OECD governments for providing incentives for more effective measures can lead to the greenhouse gas (GHG) emissions reduction.

In the period 2008-2030, the average increase of world primary energy demand is expected to be 1.5% per year, where the major contribution (about 90%) comes from non-OECD countries. China and India, indeed, have overtaken the OECD countries in terms of energy needs already in 2005 [43].

Assuming no changes in the energy policy, the future energy mix will remain based on fossil fuels.

Therefore, the demand of oil, natural gas and coal is expected to grow even though the total share should be marginally reduced (from 81% to 80%) [43].

The geographical distribution of the fossil fuels does not match the geographical distribution of the demand. China and India have only the 3% of oil and the 8% of natural gas proven reserves even though they are the major drivers of the increasing energy demand. In addition, 60% of oil and 41% of natural gas are distributed in countries that have experienced in recent years some political instability (as the Middle East countries) [45].

The continuous use of fossil fuel can result both in a growing environmental impact and in higher concerns about energy security for Europe (and OECD countries in general). In fact, the present energy mix is no more sustainable also from the environmental point of view.

Assuming no substantial changes in the energy mix, the energy-related CO_2 emissions in 2030 are expected to increase by 40% in comparison to the 2007 value [43] with serious consequences on the climate evolution.

Higher GHG emissions can bring about substantially larger concentrations in atmosphere. The effect is the increase of the average temperature with strong impacts on the whole ecosystems. Under this condition, the countries that will be first affected are the most vulnerable ones. Examples are the developing countries (e.g. sub-Saharan Africa) where the enlarging of the desertification process results in increasing famine or island states affected by the sea level increase (e.g. Japan).

According to the studies performed by the International Panel on Climate Change (IPCC), the limitation of the average temperature up to +2 °C (with respect to the pre-industrial level) can mitigate the impact on climate change [3]. In order to achieve this target, the GHG concentration in air has to be maintained below the 450 parts per million of carbon-dioxide equivalent (ppm CO_2-eq.) limiting the total annual emissions.

The actual GHG concentration in atmosphere is about 455 ppm CO_2-eq. (60% of the pre-industrial level) that corresponds to +0.76 °C of average temperature increase [3].

The energy-related emissions in 2007 are estimated equal to be 28.8 Gtons (ca. 68% of the total). This value can increase to 40.2 Gtons in 2030 (reaching ca. 71% of the total) if no actions for the energy sector are undertaken. If the emissions trend is extrapolated to 2050, the concentration in air can become equal to 1000 ppm CO_2-eq [43, 3], very far above the desirable target.

For limiting the energy-related emissions, the use of nuclear energy by fission is one of the possible solutions considered [43].

At the European level, since the publication of the *Green Paper* in 2006 [46] the problem of the sustainability of the energy sector has been addressed. The European Councils have provided indications to the member states on the path to be followed and common ambitious objectives have been fixed in order to reach these goals.

In particular, by the European Council of March 2007 [44], three medium term objectives for 2020 have been decided to timely change the actual path ("invert the route").

These objectives, collected under the name of 20-20-20 package, are based on the available technologies that can be applied for changing the energy mix. They can be summarized as follows:

- A 20% reduction of the energy related CO_2 emissions compared to 1990 level;
- A 20% energy end-used efficiency improvement;
- A 20% renewable energies (hydro, photovoltaic and biomass) in the mix.

In order to achieve these ambitious goals and to realize the long-term vision of a "carbon-free" society, the European Commission (EC) published in November 2007 the *Strategic Energy Technology Plan (SET Plan)* with the aim to identify key low carbon energy technologies for the future [47]. Among them, a) renewable energy sources (hydro, wind, photo-voltaic, and biomass), b) Carbon Capture and Storage (CCS) for the CO_2 sequestration and c) nuclear energy production (by fission) are considered.

At present, nuclear energy provides 31% of the EU's electricity production avoiding ca. 900 million tons the CO_2-eq emissions [48]. This aspect together with the higher security of supply that can be potentially achieved with a better utilisation of Uranium (and/or Thorium), see Chapter 5, make nuclear energy very attractive.

The same tendency concerning nuclear energy can be found also in Asian countries (e.g. [34]) and in USA (e.g. [49]).

However, the civil society has expressed several concerns about the use of nuclear energy that make its acceptability a complex task. The recent accident at Fukushima will certainly have an impact in this respect even if difficult to predict quantitatively in the short-term.

According to some recent public surveys in Europe (before the Fukushima accident) the public opinion seems divided about nuclear energy [50, 51]. Roughly 45% of population is in favor (with 11% totally in favor and 33% fairly in favour) the rest being opposed (with 17% totally opposed).

The major concerns are related to the safety of the Nuclear Power Plants (NPPs), perceived as a risk by ca. 53% of the population [50], and to the long-term safe management of the radioactive waste.

In order to give convincing answers to these concerns, the ongoing Research Development and Demonstration (RD&D) on nuclear energy sectors is focused on improving the safety of the systems (e.g. reducing or even eliminating the routes leading to Core Disruptive Accidents, CDA [52, 53]) and implementing closed fuel cycle (e.g. activities performed within the Partitioning & Transmutation, P&T, framework). These aspects are also at the center of the discussion pointed out in the SET Plan and in the Sustainable Nuclear Energy Technology Platform (SNE-TP) [48].

In order to increase the contribution of nuclear energy production in the future energy mix, the public acceptance is an essential prerequisite. The transparency for the communication and education about waste and emergency crisis should become a primary requirement [48].

In order to pursue these goals, the Research and Development (R&D) on nuclear fission has been focused on short-term main objectives (up to 2020) and long-term objectives considering actions up to 2050.

For the short-term, the objectives can be summarized by maintaining the *competitiveness in fissions technologies, together with long-term waste management solutions* [48].

The answer of the international community to the Fukushima event could strongly impact the future of the nuclear energy sector and therefore current technical initiatives (e.g. stress tests) are of the utmost importance.

However, despite the present focus on the Fukushima accident and its impact, crucial aspects, focused on the nuclear fuel cycle sustainability, as the adoption of advanced fuel cycles oriented to the wastes minimization and to the resources optimization, are still essential issues for the future nuclear energy development [54, 49].

These studies are carried out worldwide. Studies are ongoing in Europe [55] and also in the framework of the Gen-IV activities [52]. Of course, the introduction of advanced fuel cycles has to be pursued maintaining (or increasing) the safety level and ensuring proliferation resistance.

In fact, as for oil and gas, the uranium resources are limited. The present nuclear technology uses only less than 1% of the extracted resources and therefore an optimization in their use should be foreseen.

The assessment provided by [2] has fixed a maximum limit value for the Identified Resources (IR, available at cost less than 130 USD/kgU) to approximately 16 Mtons[1]. This quantity can be increased adding uranium dispersed in phosphates (ca. 22 MTons as indicated by [56]) and a maximum limit of 38 Mtons could be reached.

[1]The IR are called also Reasonably Assured Resources, RAR, and they are equal to 59% of the total [2].

This assessment is obviously subject to readjustments on the basis of new exploration campaigns or evolutions in the uranium prices. An example is the Red Book 2009 [57] assessment that adds a new resources category (available at cost less than 260 USD/kgU) in response both to overall uranium price market increase and increased mining costs that enables to extend of 15% the resources availability (conventional ones) [57].

For a better utilization of uranium, three main actions have been investigated for the short-term: 1) the adoption of cores with high conversion ratio (CR); 2) the adoption of very high burn-up fuel in LWRs; and 3) the recycling of plutonium in LWRs (by the use of Mixed Oxide, MOX, fuel).

For the long-term, fully closed fuel cycles, based on spent fuel reprocessing and fast reactors (e.g. as investigated in Europe [48, 54, 58, 59]) are the key technologies to achieve sustainability: resources optimization, waste minimization and proliferation resistance, as stated in the Generation-IV initiative [52].

As for the closed fuel cycle, several options are investigated: recovery of Pu and higher TRUs sent to the repository together with fission products (FPs) or full TRUs recovery with successive recycling in FRs (only Fission Products, FP, and losses are sent to the repository), etc. All these options are investigated within the so-called Partitioning and Transmutation (P&T) strategies.

In this context, the development of fuels containing large quantities of Minor Actinides (MAs) to be burned in critical or sub-critical systems as well as the development of chemical separation technologies are among the main challenges for the R&D activity concerning the advanced fuel cycle [60].

The development and demonstration of the industrial feasibility of Partitioning and Transmutation together with the necessary cost assessment should allow the transition from the open fuel cycle to appropriate closed or partially closed fuel cycles based on P&T (i.e. Pu and MAs recovery) is foreseen to meet the sustainability goals fixed in Gen-IV [52], as indicated above.

The implementation of these advanced fuel cycles has a potential significant impact in terms of sustainability, of environment protection and on public opinion perception.

1.1 Motivation and aim of the present study

The major aim of these studies has been to provide a contribution to the analysis of advanced fuel cycles with special attention to the analysis of the impact in terms of resources, waste inventory and infrastructures that these cycles require for well defined world regions, or for a specific country.

The activity has been developed to get an overview about the challenges, implications, boundary conditions, assumptions and correlated consequences of different scenarios suggested or envisaged for the deployment of nuclear energy (mainly for electricity production) on a global or regional basis.

In particular, the implications of the LWRs use (resources availability, fuel cycle facilities, waste production, etc.) and the possible transition to FRs, in terms of Pu availability, enlargement of reprocessing capability, etc, have been key points of the analysis.

Scenario analysis algorithms, have been found to be a suitable tool for evaluating the impact of the advanced fuel cycles.

Moreover, the most important parameters (or hypotheses) affecting the scenario analysis have been identified.

As an example, it has been shown that scenarios associated to global trends have different input and output with respect to scenarios focused on local trends. Moreover, the analysis of a phase-out scenario implies different assumptions than a scenario oriented to a nuclear energy start-up or constant situation (both in terms of energy mix and type of cycle facilities).

Moreover the adoption of different reactor types (e.g. fast reactors with higher conversion ratio or higher burn-up fuels for LWRs) and different fuel cycle options (i.e. only Pu recycling or MAs or TRUs recycling) have been considered for the study together with hypotheses on the energy demand in different geographical areas.

All these data represent the scenario boundary conditions. The identification of the most suitable ones for the study is an important aspect to be analyzed.

Once that the general boundary conditions are defined, several other parameters (e.g. fuel burn-ups, or reactor introduction rates) can further slightly affect the results.

In order to quantify the effect of each parameter studied, a simplified case, i.e. a country with constant energy demand and with the intention to start the nuclear energy production in the coming years, has been selected and used as reference.

This study can be easily extrapolated to the European situation and used as "unit of measure" for representing a more complex scenario (e.g. an increasing nuclear energy demand scenario). For this reason, the energy demand chosen is small enough (the nuclear energy production considered is 70 TWhe/y) to make it suitable to the extrapolation and large enough to investigate in detail some hypotheses (e.g. substitution of the LWRs fleet).

The identification of the critical points has been made by several steps: preliminary scoping study oriented to boundary conditions investigation, selection of suitable tools, study of the reference scenario, parametric study oriented to LWRs and transition study toward FRs.

A preliminary scoping study has been performed adopting the IAEA Nuclear Fuel Cycle Simulation System (NFCSS code) [16]. In order to refine the analysis after the identification of the critical points associated to the scenario considered (mainly to transition scenarios) the fuel cycle code COSI6 has been used [20]. The use of this code implies the definition of suitable libraries for each reactor modeled and the assumption of more refined hypotheses.

A parametric study has then been performed for quantifying the uncertainties associated to the study.

The study has shown how the adoption of FRs and advanced fuel cycles based on P&T are essential for reaching the sustainability targets, i.e. resources optimization as well as waste minimization and radiotoxicity reduction.

Several FR concepts have been compared as well as several Pu and MAs multi-recycling strategies. Some indication about the safety characteristics of the systems considered has been given when available.

1.2 Organization of the thesis

In Chapter 2 an introductory overview of the scenario studies performed worldwide has been performed in order to point out the applicability domain of fuel cycle studies. In particular, scenarios oriented to world studies (e.g. [4, 61]) are compared with regional scenarios (e.g. [5, 62]) and with country-oriented scenarios for underlining the differences regarding objectives and hypotheses (e.g. [9, 63]) that the "geographical" scale can require.

In fact, for a world study the main focus is oriented to the investigation of the possible resources shortage, where different conditions, as different energy demand projections, play the main role.

For regional studies, e.g. [5], the focus is on the share of resources and fuel cycle installation (e.g. shared fuel fabrication and reprocessing facilities, (accelerator driven systems or critical fast reactor burners) to meet the different country's objectives within the region.

For a country-oriented scenario, the main purpose is to point out the facility needs for sustaining the fuel cycle (e.g. the fabrication and reprocessing plants capacities) as well as the inventory of material to be sent to a repository. The main driver of this kind of study is the strategy adopted for the country (e.g. future transition to FRs, adoption of Pu mono-recycling in LWRs, etc.).

In Chapter 3 the methodology adopted for setting up the boundary conditions of a scenario study is described in detail. This methodology mainly focuses on a country-oriented scenario and on how to adress the chosen strategy (e.g. for the transition to LWRs to FRs). In order to compare scenarios in terms of sustainability, the impact on different sectors (environment, economy and social sectors) have been considered. Based on the literature, the identification of several indicators, adopted for comparing the scenarios, has been done and summarized in Par 3.2. The choice of the indicators has been performed looking also to the most important aspects that can affect the social acceptability of nuclear energy production.

Some preliminary scoping studies have been performed. The results obtained with the NFCSS code [16] are summarized in Par. 3.3. In order to highlight possible differences on the choice of the scenario boundary conditions, several countries with different nuclear energy policies (at 2008) have been considered.

The preliminary scoping study has indicated how the results of the scenarios study are affected by the selection of the hypotheses. In order to deal with these systematic "uncertainties", a parametric study has been performed for one of the cases selected.

In Chapter 4, scenarios with only LWRs have been compared. The impact of several parameters has been evaluated on the scenario results (e.g. Pu availability, resources adopted, facilities needs, etc.). The parameters selected for the study are a) the adoption of higher discharge burn-ups for LWRs fuel, b) the impact of different introduction rates for the reactors and c) different energy demand hypotheses. These parameters have been selected in agreement with the hypothesis beyond the short-term objectives described in [54] in order to provide the impact that these hypotheses can have on the transition to FRs (e.g. impacting the Pu availability in the cycle).

In Chapter 5, the analysis of transition scenarios from LWRs to FRs is summarized. Several options have been considered in order to identify the advantages of each solution. The adoption of fast reactors with different core designs and breeder characteristics has been investigated. In addition several fuel cycles (with Pu and/or MAs multi-recycling) have been compared in order to underline the their respective advantages and drawbacks.

Finally in Chapter 6 conclusions and the possible future studies are reported.

Moreover, in Appendix A the description of the fuel cycle and neutronic codes adopted in the study is reported and in Appendix B a comparison about the available energy data is included. These data provide an example of the selection of suitable boundary conditions for the world study.

In Appendix C provides a discussion on important parameters as the Conversion Ratio (CR) and the Breeding Gain (BG).

In Appendix D, the description of the neutronic models for the systems considered is summarized. In particular, the ESFR and ELSY models are described in detail.

Finally, in Appendix E the data adopted for simulating the Italian scenario are included.

Chapter 2

Overview of the Existing Fuel Cycle Scenario Studies

In the present Chapter, an overview of the existing fuel cycle and scenario studies has been included in order to show the applicability range of this kind of investigations.

The key objective of this study is the long-term nuclear energy sustainability investigated as the analysis of innovative fuel cycles and reactor concepts able to guarantee the long-term security of supply in an economical and safe manner by reducing the impact on the environment.

According to the literature, the scenario studies can be subdivided in three main categories:

- studies oriented to world scenarios;
- studies oriented to regional scenarios;
- studies oriented to country-specific scenarios.

These three categories imply different level of approximation and, therefore, different types of hypotheses and boundary conditions.

For world-oriented scenarios (where the increasing energy demand is the main driving force), the long-term sustainability is mainly related to the analysis of the global availability of natural resources (uranium, and thorium). Hence, the identification of factors that can accelerate or postpone the stress on the uranium market (e.g. adoption of different fast breeder systems [4]) is one of the main goals. In addition, this kind of studies can be adopted to analyze fuel cycle needs to follow-up of an increasing energy demand (e.g. due to a change in the energy mix for limiting GHG emissions as proposed e.g. by IPCC [3]) pointing out the possible limiting factors.

In a regional-oriented scenario the sustainability of the nuclear energy production can be also related to address available industrial capacities and nuclear waste issues. Therefore, the analysis of the impact of P&T strategies becomes more important[1]. In this context, the selection of the more suitable and reliable burner system (e.g. accelerator driven systems, fusion-fission hybrid systems, critical fast burners [5, 62, 64, 65]) and strategies (e.g. homogeneous MAs multi-recycling in FRs) are key points of the studies.

Finally, for a country-oriented scenario, the sustainability is related to maintain in operation, in a safe and economic way, the implemented fuel cycle. The infrastructures needs (e.g. fabrication, enrichment, reprocessing capacities) as well as the strategy adopted (e.g. LWRs "once-through" or Pu recovered and

[1] This aspect is import in particular for the long-term sustainability of the repository, by the reduction of the potential risk source term.

recycled in FRs) are fundamental aspects to be analyzed. At the same time, the nuclear waste reduction (both inventory- and potential risk-associated) and resources optimization are additional aspects influencing the long-term sustainability and suitable choices for achieving that goal.

All the scenarios considered need suitable hypotheses and boundary conditions. A good selection helps in reducing the systematic (and unavoidable) uncertainties associated to a scenario study[2].

Before describing the existing fuel cycle studies, a short overview of the proposed fuel cycle strategies as well as of the systems is given Par. 2.1 and 2.2.

2.1 Fuel Cycle Strategies

An overview of the advanced fuel cycle schemes, alternative to "once-through" fuel cycle, has been proposed by [1]. These schemes can be classified in three families, [1]:

- 1st Family includes the fuel cycles based on current industrial technologies and possible extensions.
- 2nd Family considers partially closed fuel cycles (where Pu is multi-recycled).
- 3rd Family considers fully closed fuel cycles with all actinides continuously recycled in FRs or ADS systems.

Several kinds of reactors[3] are combined together in different ways in order to provide the 13 schemes proposed in ref. [1].

In the next section, only the most representative examples for the study are described. A complete analysis is given ref. [1]. In these examples the composition of the spent fuel at disposal assessed by a steady state approach are given[4] [1]. However, it is expected that these compositions will be different if the complete scenario study (including the transition period) is analyzed.

Family 1: "Once-Through"

The "once-through" scheme is represented in Figure 2.1. In the scheme, only PWRs of 1,450 MWe, 4.9% U235 enrichment, 34.1% thermal efficiency and 60 GWd/tHM average burn-up are considered [1].

The average Spent Fuel (SF) composition indicated in Figure 2.1 is calculated under equilibrium conditions considering 2 years of aging time and 5 years of cooling time.

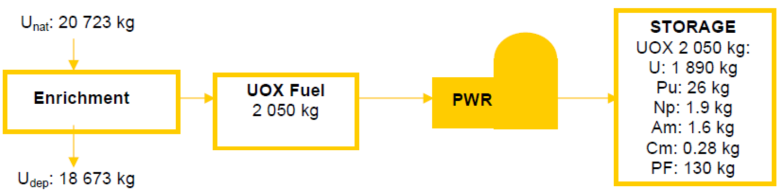

Figure 2.1: "Once-through" scheme [1]

[2] In order to quantify these uncertainties a parametric study has been developed within the present Ph.D. activity as described in Chapter 4 and Chapter 5.

[3] Mainly the Pressurized Water Reactors (PWRs), the Heavy Water Reactors (HWR-CANDU), the Fast Reactors (FRs) and the Accelerator Driven Systems (ADS)

[4] It could be useful to FRs has a larger thermal efficiency than LWRs that can affect the SF composition, i.e. by the FPs amount.

Family 1: UOX reprocessed and Pu mono-recycled in thermal reactor systems (MOX fuel)

In Figure 2.2 is represented the fuel cycle scheme for Pu mono-recycling in thermal reactor systems (only PWRs).

In particular, a PWR/UOX with the same characteristics as the "once-through" case and a PWR/MOX with 10% Pu content, 34.1% thermal efficiency and 60 GWd/tHM average burn-up. PUREX process is considered for the reprocessing with 0.1% U and Pu losses and 100% MAs sent to the repository [1].

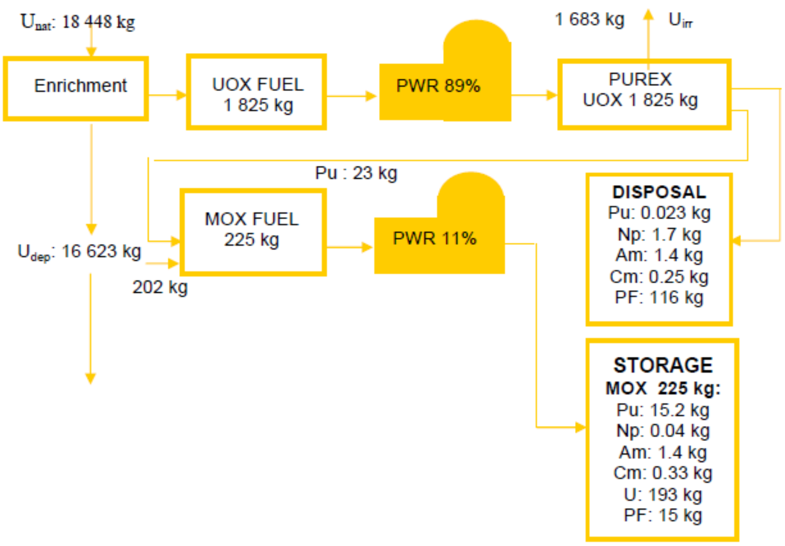

Figure 2.2: UOX reprocessed and Pu mono-recycling in MOX [1]

Family 2: UOX and MOX reprocessed, Pu multi-recycled in FRs, Cm stored and Am "once-through" transmuted in FRs

Figure 2.3 shows the fuel cycle scheme for Pu multi-recycling in FRs with Am targets transmuted in FRs, only "once-through" without multi-recycling of these targets.

The PWR/UOX system considered has the same characteristics as in the "once-through" case. The FR considered is based on the European Fast Reactor (EFR), i.e. 1450 MWe, 18% Pu content, 40.3% thermal efficiency and 140 GWd/tHM average burn-up. An advanced PUREX process is considered for reprocessing with 0.1% U, Pu and Am losses and 100% Np and Cm sent to the repository [1].

Family 3: Transuranics burning in FRs

In Figure 2.4 is represented the fuel cycle scheme for TRUs multi-recycling in FRs.

The systems considered are: 1) a PWR of 1450 MWe electric power (34.1% thermal efficiency) with 4.9% U235 enrichment and average burn-up of 60 GWd/tHM; and 2) an FR of 600 MWe electric power (fast system adopting metal, AcZr, fuel with 38.1% thermal efficiency) with 29.1% Pu content and 3.9% MAs content and average burn-up of 140 GWd/tHM. For reprocessing, the UREX process is used for thermal SF (losses 0.1% U, Pu, MAs) and the PYRO process for fast SF treatement (losses 0.1% U, Pu, MAs) [1].

The average SF composition evaluated under equilibrium conditions is indicated too. For the thermal reactor system an aging time of 2 years and cooling time of 4 years has been considered, however for fast systems an aging time of 1 year and a cooling time of 2 years have been adopted.

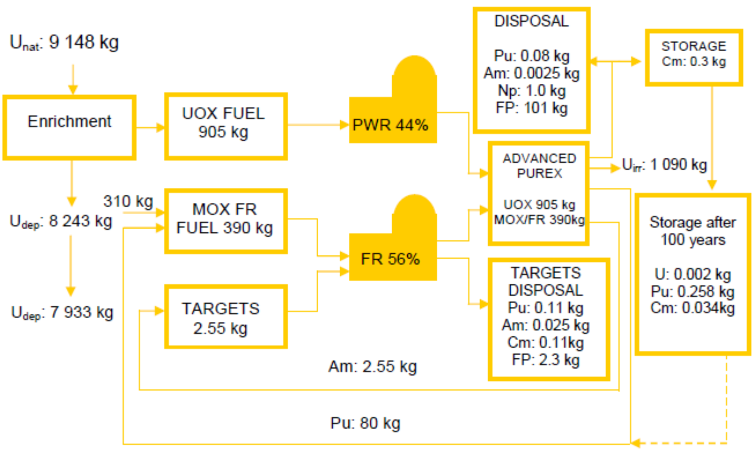

Figure 2.3: UOX and MOX reprocessed and recycled in FRs, Cm stored and Am target [1]

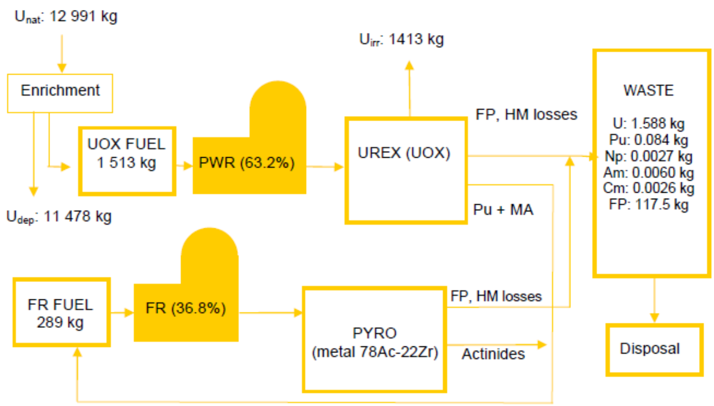

Figure 2.4: TRUs burning in FRs scheme [1]

2.2 Thermal and Fast Reactor Concepts

In order to provide a complete basis for scenario studies described in the next sections and Chapters, a short overview of the reactors considered is presented here .

At present (2011), about 440 NPPs are in operation (65 NPPs are under construction) [17], where the largest fraction belong to Gen-II PWR and BWR as indicated in Table 2.1.

NPPs In Operation			
Type		No. of Units	Total MW(e)
BWR	Boiling Water Reactor	88	81,480
FBR	Fast Breeder Reactor	1	560
GCR	Gas Cooled Reactor	18	8,949
LWGR	Light-Water-cooled Graphite-moderated Reactor	15	10,219
PHWR	Pressurized Heavy Water Reactor	47	23,042
PWR	Pressurized Water Reactor	271	250,009
Total		440	374,259
NPPs Under Construction			
Type		No. of Units	Total MW(e)
BWR	Boiling Water Reactor	4	5,250
FBR	Fast Breeder Reactor	2	1,274
LWGR	Light-Water-cooled Graphite-moderated Reactor	1	915
PHWR	Pressurized Heavy Water Reactor	3	1,952
PWR	Pressurized Water Reactor	55	53,486
Total		65	62,877
NPPs in Long-Term Shutdown			
Type		No. of Units	Total MW(e)
FBR	Fast Breeder Reactor	1	246
PHWR	Pressurized Heavy Water Reactor	4	2,530
Total		5	2,776

Table 2.1: Overview of the Existing NPPs [17]

It is expected that for the short-term these reactors will be replaced by Gen-III(+) systems (e.g. EPR developed by AREVA [66] or AP1000 developed by Westinghouse [67]) and for the long-term by the advanced fast reactord [52, 58, 68, 54].

A detailed description of all reactor types would be far beyond the aims of the present Ph.D. activity, and, therefore, is not included.

As indicated in Chapter 4 and 5 the European Pressurized Reactor has been considered as representative for Gen-III systems. The data adopted for this type of reactor, in agreement with the literature [66, 7], are included in Par. 4.1. Description of the neutronic and core design is not included in the thesis[5].

Moreover, two kinds of advanced fast reactor systems have been considered, i.e. Sodium-cooled Fast Reactor (SFR) and Lead-cooled Fast Reactor (LFR). In particular, the European Sodium Fast Reactor (ESFR) and the European Lead-cooled SYstem (ELSY) have been analyzed [58, 68]. The data adopted for assessing the neutronic models are described in Appendix D.

The advanced SFRs can be considered the first choice for Europe due to the large experience gained in the past. As possible alternatives, the lead-cooled and gas-cooled fast reactors are investigated as well [54].

[5]The EPR system has not be modeled in detail, see Chapter 4

In addition, the analysis of dedicated facilities for burning MAs is still ongoing [54, 69].

The key issue of their success is the objective of a higher and improved safety level as well as the improved economic competitiveness. Innovative design and technology features are needed to achieved these goals [70, 48, 54].

For the development of an advanced SFR, activities are oriented to enhancing safety (by the reduction or even elimination of risks and routes that can lead to severe Hypothetical Core Disruptive Accidents, CDAs), to improve economical aspects (as the reduction of capital costs) and to improve the sustainability (resources optimization, reduction of waste inventory and enhancement of proliferation resistance) [48, 54, 70].

During the present work, activities oriented to improve the safety and the sustainability of the SFR have been performed. Results obtained for the development of the ESFR concept, are shortly summarized in Appendix D and published in [31, 30, 71]).

At the European level, studies are ongoing for the design and construction of the industrial demonstrator (Advanced Technology Reactor for Industrial demonstration, ASTRID project) in France [72].

Other studies are ongoing in other parts of world as the Japanese Sodium Fast Reactor Concept (JSFR, [53]) or the KALIMER-600 developed in Korea [73].

As alternative the LFR is investigated in Europe with special attention to the material qualification, the lead technology development, the competitiveness and safety [52, 54, 70, 74, 75, 59].

Within the present work, scenarios based on lead cooled fast reactors have been analyzed (see Chapter 5). The description of the data used for assessing the neutronic model is included in Appendix D.

Other systems as ADSs [69] and Gas-cooled Fast Reactors (GFRs) [54, 70, 76, 77, 78, 79] are also under consideration in international studies of fuel cycle assessment.

2.3 World-oriented Studies

One of the main concerns for the long-term sustainability of the energy sector is the security of supply. Hence, in order to establish a sustainable energy mix for the future, the efficient use of the natural resources involved is a goal to be pursued. The reference resources considered in the different studies is uranium. However, also thorium could be considered for these purposes.

Availability of natural uranium is limited to ca. 40 Mtons (only 42% are conventional resources) as indicated by [2, 57]. This limited value and the potential increase of nuclear energy demand can lead to a peak, similar to the Hubbert peak for the oil sector, that may result, during the coming years, in stresses on the market. A brief summary of the available data on U resources is given in Par. 2.3.1.

Increasing the uranium utilization factor for the mined resources (actually less than 1%) reaching 50% (or higher values) is one of the ways to improve the long-term security of supply. Several concepts have been proposed: e.g. the adoption of LWRs high burn-up fuels [80, 22], or using high conversion ratio systems or the introduction of Fast Breeder Systems and several other options [81, 82].

The analysis of possible uranium shortage is best performed on a global basis by world scenario studies. The adoption of several technologies and energy envelops reveals the major trends for resources utilization and infrastructures needs.

The advantages of innovative fuel cycles (based on advanced reactors and/or P&T strategy), are investigated with respect to the ongoing "once-through" strategy.

A common agreement about a global solution, both for resources and for waste and for related implications, has not been reached. This is due to the heterogeneity of objectives and strategies pursued by each country as indicated in Par. 2.5..

Altogether, the world-oriented studies provide a set of possibilities in which the future trend could be embedded. In that sense, more hypotheses are investigated, more details can be provided for analyzing the different energy mix.

The most important boundary conditions are the primary energy projections and the expected nuclear energy share.

A short overview of the data available for defining future energy demands is indicated in Par. 2.3.2. More data are included in Appendix B, where the process adopted for selecting the energy envelop for the heterogeneous world study (see Par. 2.3.4) is described.

At the international level several world scenario studies are ongoing. IAEA established in 2000 an International Project on Innovative Reactors and Fuel Cycle (INPRO) with the main aim to discuss nuclear energy as an available and reliable source of energy for contributing, in sustainable manner, at the fulfillment of the energy needs in 21st century [55]. Within the project, a specific area has been devoted to provide a better understanding of the role of nuclear energy by the development of global and regional nuclear energy scenarios on the basis of a scientific-technical pathway analysis [33].

Similar and complementary studies have been proposed by NEA/OECD within the Working Party on Scientific Issues of the Fuel Cycle (WPFC) by the Expert group on Fuel Cycle Transition Scenarios Studies (WPFC/FCTS) [83].

An additional interesting study performed is the 2010 analysis of the Massachusetts Institute of Technology (MIT) [61]. In the study, the acceptable increasing uranium cost is one of the criteria adopted to affirm that there are no constraints for the present century in developing LWRs only. Implicitly, they consider that there are no urgent needs on developing FRs and associated reprocessing facilities.

In addition to these main projects other studies are worldwide developed, e.g. [84, 85], in order to contribute to the debate.

An overview of all these studies, underlying hypotheses and main results achieved, is included in the following paragraphs (Par. 2.3.3, 2.3.4).

2.3.1 Natural Resources

According the OECD/NEA and IAEA "Red Book" on Uranium resources [2], the total amount of the Identified uranium Resources is estimated to be ca. 5.5 Mtons. These resources are available at cost less than 130 USD/kgU, cost considered today acceptable by the market. This category includes the resources in known mineral deposits that can be recovered with current proven mining technology (called also Reasonably Assured Resources, RAR, and equal to 59% of the total) and the resources supposed on the basis of direct geological evidence but with estimates affected by uncertainties (called also Inferred Resources, IR, and equal to 41% of the toal).

In addition, 2.8 Mtons are associated to the prognosticated undiscovered resources, supposed on the basis of undirected geological evidence, and 7.7 Mtons are associated to the speculative undiscovered resources, supposed on the basis of undirected geological evidence and on geological extrapolations, [2].

Considering all these categories, the limit on the conventional uranium resources can be fixed to ca. 16 Mtons. This value is increased to 38 Mtons if uranium in phosphate[6] (ca. 22 Mtons as indicated by [56]) is taken into account. This value, 38 Mtons, has been considered as limit by several scenarios studies, e.g. the NEA/OECD [4] or [85]. Different approach has been adopted within the INPRO project [33], where the limit has been fixed to 20 Mtons representing a reasonably confident value evaluated on the basis of reasonable costs [33]. Other studies, indeed, consider the conventional resources (16 Mtons) as the limit, e.g. [84, 87, 88].

These different initial conditions as well as the different energy envelops considered affect the scenario strategy e.g. anticipated uranium shortage implies the early introduction of advanced fuel cycles based on

[6]Only uranium in phosphate rocks has been considered [2]. The uranium in seawater, ca. 1000 more than in terrestrial ores [86], has not be considered.

breeder systems.

The assessment of the uranium resources is subject to adjustement on the basis of new exploration campaigns or accepted higher uranium price. An example is the Red Book 2009 [57] where a high price resources category, available at a cost less than 260 USD/kgU, has been added, thus increasing the conventional resources total amount by about 15%.

However, for the short-term scenario, another interesting parameter, that can affect the market and the LWRs deployment, is the relative behavior between the annual uranium demand and the annual uranium extraction together with the existing stock pile.

In 2007, indeed, the uranium demand associated to all of the reactors in operation worldwide has been 69,110 ton, and the extraction in the same year has been 43,328 tons [89].

This means that each year, in order to cover the demand, the stock pile of uranium (collected during previous years) is driven down [2]. In fact, as presented in Figure 2.5, since 1990 the requirements of uranium exceed its production.

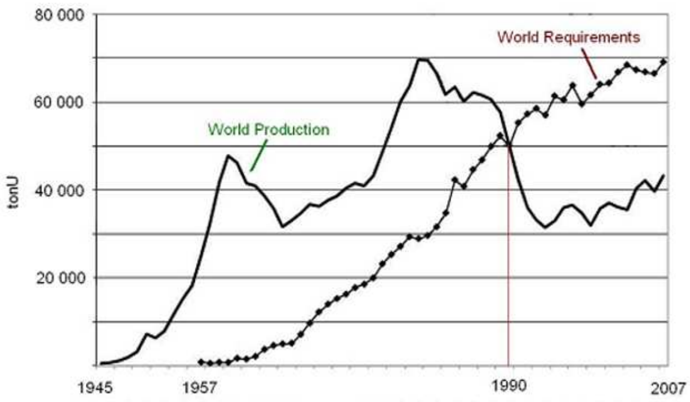

Figure 2.5: Historical data on Uranium World Requirements and Production [2]

The uranium resources can be augmented taking into account thorium [90] resources. In fact, thorium is 3 to 4 times more abundant in nature than uranium, reaching an average concentration of about 10 ppm in earth crust, and its use has practically not been exploited commercially so far. The adoption of Th based fuel cycle should be particularly interesting for countries with large thorium resources (e.g. India). Several studies are ongoing in order to investigate the feasibility of the thorium fuel cycle also from the point of view of fuel cycle back end parameters [91, 92, 90, 93, 94].

2.3.2 Energy Demand

The limit on resources availability is the main constraint for world scenario studies. This limit becomes more important when increasing energy demands are taken into account.

The definition of the energy projections implies, indeed, the analysis of several heterogeneous parameters as the expected population and economy growth rates, the energy policy choices (at international and national level), the international agreements on environment protection, etc..

In order to define these future energy projections, specific multi-sector models have been developed (e.g. the MESSAGE model developed by the IIASA, the MiniCAM model developed by the Pacific Northwest National Laboratory, PNNL, [95], etc.) and applied.

Uranium Resources below 130 USD/kgU				
Country	RAR	Inferred	Total	%
tons				
Australia	725,000	518,000	1,243,000	22.7
Kazakhstan	378,000	439,200	817,200	14.9
Canada	329,200	121,000	450,200	8.2
USA	339,000	-	339,000	6.2
South Africa	284,400	150,700	435,100	8.0
Namibia	176,400	30,900	207,300	3.8
Brazil	157,400	121,000	278,400	5.1
Niger	243,100	30,900	274,000	5.0
Russian Federation	172,400	373,300	545,700	10.0
Uzbekistan	72,400	38,600	111,000	2.0
India	48,900	24,000	72,900	1.3
China	48,800	19,100	67,900	1.2
Others	363,300	263,900	627,200	11.5
Total	3,338,300	2,130,600	5,468,900	100.0

Table 2.2: Geographical distribution of the Uranium resources [2, 32]

This analysis of ways for assessing the future energy trends as well as the more general analysis of the penetration of a specific technology (e.g. [96, 97, 98]) has not be treated in the present study because it is considered to be beyond the aims of this work.

The results of these models, i.e. the energy trends and energy mix, are used as boundary conditions for the scenario studies. Indeed, the nuclear energy projections can be extracted and used for the fuel cycle analyses.

For the mid term, more reliable data are provided by the International Atomic Energy Agency (IAEA) and by the Nuclear Energy Agency (NEA), e.g. [15, 99]. Otherwise, for the long-term, the data provided by the Intergovernmental Panel on Climate Change (IPCC), the International Institute for Applied Systems Analysis (IIASA) and the OECD International Energy Agency (OECD/IEA) [23, 3] are considered.

Some of the most interesting data for a world scenario study are reported in Appendix B.

In order to reduce the uncertainties related to the definition of global trends, IPCC and IIASA adopted a fine subdivision in sub-zones with more homogeneous initial conditions (then collapsed in four macro-regions for IPCC [3] and three for IIASA [23]).

The IPCC subdivision shown in Figure 2.6 has been adopted as reference by the NEA/WPFC groups in the framework of the world homogeneous-heterogeneous scenarios studies [4, 100], as summarized in Par. 2.3.4.

The energy demand data are important boundary conditions for the scenario studies, therefore, the differences on energy demands and future mix can largely justify possible different conclusions given by the various global studies.

In the next paragraph some relevant world energy studies are summarized.

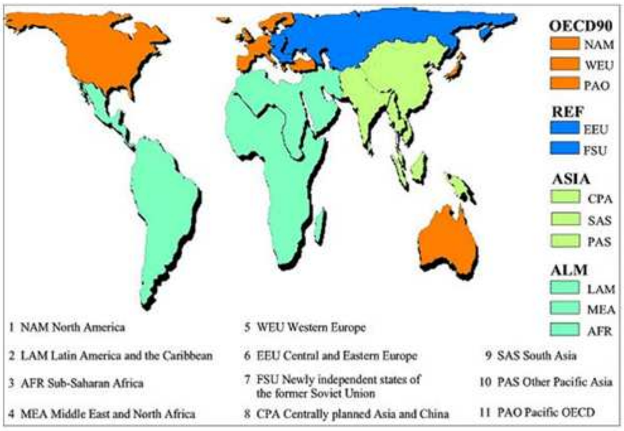

Figure 2.6: IPCC world subdivision: the 4 macro-regions considered [3]

2.3.3 The IAEA INPRO Project

The International Project on Innovative Reactors and Fuel Cycle (INPRO project) proposed by IAEA in 2000 has the main aim to investigate how nuclear energy can give its contribution for fulfilling, in a sustainable way, the energy needs of the coming centuries [55].

Several hypotheses and boundary conditions have been compared for this purpose: energy demands, type of reactors considered, natural resources (U and Th) and fuel cycle strategies.

Three nuclear energy demands (see Table 2.3) representing low, moderate and high growth rates scenarios have been compared [33]. In the three cases, a similar behavior is followed up to 2030, in agreement with the SNE-TP short-term projections where no substantial changes are expected for the next 20 years [48, 54]. After 2030 the projections start to diverge: in the LOW case the nuclear capacity installed increases to about 7 times (from 370 GWe to 2,500 GWe in 2100), value that becomes two times larger in the MODERATE scenario (5,000 GWe in 2100) and four times larger in the HIGH scenario (10,000 GWe in 2100).

Nuclear Capacity Installed (GWe)				
Year		LOW	MODERATE	HIGH
2009		370	370	370
2030		500	600	700
2050		1,000	1,500	2,000
2100		2,500	5,000	10,000

Table 2.3: World nuclear energy scenarios used within the INPRO study [33]

For refining the analysis, several world subdivisions have been considered: geographical-based or "current developing level"-based subdivisions [33]. Here, only the results for the geographical-based subdivision are presented.

The scenarios proposed focus on the period 2007-2100 [33]. During this time period, the use of thermal spectrum reactors remains predominant: in-operation reactors are expected to be substituted (during the next 20 years) by the available technologies (Gen-III, Gen-III+) and these advanced thermal reactors are expected to remain in operation at least for 60 years reactor lifetime. However, the substitution with FRs is investigated too (as expected ready from 2050 [52]).

Thermal Reactors Characteristics	
WCR	Present LWRs
WCR-M	LWRs with higher burn-up and efficiency
SMR	Small PWR with high specific uranium consumption
HTR	High temperature reactor, uranium cycle
HTR (U3)	High temperature reactor, thorium cycle
Fast Reactors Characteristics	
FBR-C	Liquid metal FR, BR ca. 1
FBR-S	Advanced Na FR, BR ca. 1.4
FBR-A	Strong breeder FR, BR ca. 1.6
FBR-A(Th)	Strong breeder with Th232 blanket, BR ca. 1.6

Table 2.4: Thermal and Fast Reactors considered within the INPRO study [33]

As indicated in Table 2.4, the adoption of both uranium and thorium fuel cycles has been considered as well as the adoption of open or closed fuel cycles. Different combinations of reactors and strategies have been considered to cope with the energy demand indicated in Table 2.3.

Additional hypotheses have been added concerning the material exchange between regions. Fuel resources (e.g. fresh and spent fuel) can freely circulate, whereas enriched uranium and reprocessed material can not circulate for proliferation resistance issues. This implies the adoption of regional fuel centers for making fuel material management economically profitable [33].

For the LOW scenario only thermal reactors have been adopted. The installed nuclear capacity projection listed in Table 2.3 has been subdivided for each region: in Europe (EU) and North America (NA) the installed capacity is expected to slightly increase reaching a value of two - three times the 2007 capacity, whereas in Asian and African countries the considered increase is much higher [33].

Under these assumptions, the natural uranium consumption in 2100 is expected to be equal to 18.5 Mtons, just above the currently estimated conventional resources limit [2]. However, the open fuel cycle strategy implies spent fuel disposal capacities very high i.e. 1.6 Mtons for the total SF whit 23,000 tons of Pu [33]. Therefore, the "once-through" strategy seems not completely sustainable (looking both to fuel cycle front-end and back-end). The introduction of FRs in closed fuel cycles (starting from 2030-2040) and of the reprocessing plants can substantially reduce the SF inventory providing beneficial contribution also regarding uranium saving (ca. 30-50%) [33].

A different situation appears for the MODERATE scenario. The increase in North America and Europe remains limited but Asia and Africa increase much more. Under this energy demand, the introduction of FRs is mandatory for guaranteeing the security of supply. In order to analyze the effect of the transition to FRs (starting from 2030), two cases have been compared in INPRO: 1) the use of a breeders system with Breeding Ratio, BR, ca. 1.4; and 2) the use of a strong breeder with BR larger than 1.6. In both cases the natural resource limit (38Mtons including uranium in phosphates) is reached before the end of the present century [33].

The implementation of this kind of scenario implies big challenges for the infrastructures, i.e. large scale reprocessing, large scale FRs fuel fabrication plants and intra-regional transfer for resources, SF, and wastes [33].

In order to cope with the energy demand defined by the HIGH scenario, the adoption of high breeder reactors (BR larger than 1.6) as well as the adoption of thorium fuel cycles are the only solutions available. The challenges associated to high demand are the same as highlighted for the moderate scenario but much more challenging. By the adoption of strong breeders together with Th cycles, the natural uranium consumption can be maintained below the limit of resources estimated reaching a value of about 20 Mtons.

The INPRO study confirms that the transition toward FRs is essential for the long-term sustainability of the energy sector in terms of resources, SF inventory, SF heat load and radiotoxicity.

2.3.4 The NEA/OECD World Homogeneous-Heterogeneous Studies

Studies complementary to the INPRO are under development at the NEA/OECD [83].

The study focuses on a limited number of parameters with respect to the INPRO study in order to point out the major trends and issues [4].

The adoption of a single energy demand scenario, selected "a priori" on the basis of available literature data analysis (see Appendix B for details) has been considered as well as uranium fuel cycle only.

The activity has been focused on the comparison between a nominal homogeneous world treatment, for determining the global trends, and a heterogeneous world scenario study, for refining and differentiate the contribution of the individual regions [4, 100]. For the heterogeneous study, the four macro-regions proposed by the IPCC (see Figure 2.6) have been considered. Each macro-region has been studied providing the specific energy trend and reactors to be considered within the region.

Concerning materials circulation, the following assumptions have been considered: 1) no shared facilities between the macro-regions, 2) only the natural uranium can freely move, and 3) within each macro-region shared facilities (enrichment, reprocessing, fabrication, disposal plants) are available.

In terms of energy envelops, a description of the procedure adopted for selecting them is reported in Appendix B.

For the homogeneous case, the nuclear energy envelop selected corresponds to one of the 40 scenarios proposed by IPCC (namely B2 MiniCAM scenario). This scenario is oriented to environment protection by the adoption of regional solutions [3, 101] and it assumes a reasonable increase of the global nuclear energy demand. In 2100, indeed, the electricity produced by nuclear power plants is assessed to be equal to ca. 19,000 TWhe (more than 6 times the present demand but quite comparable with the LOW scenario presented by INPRO, see Table 2.4).

For the heterogeneous case, the subdivision proposed by one of the IIASA scenarios (namely B scenario, the "business as usual" case) has been adopted, re-scaled in order to preserve the same total energy envelop considered for the homogeneous case [23, 3, 4].

This choice is a compromise to avoid questionable regional trends presented by the B2 MiniCAM scenario (e.g. ASIA and ALM macro-regions overtake in terms of nuclear energy production OECD90 before 2035) and to preserve a reasonable total energy envelop (in fact, in the B-IIASA scenario the nuclear energy envelop reaches in 2100 a double value with respect to the B2 - MiniCAM case [23, 3, 4]).

By this assumption, the ASIA and ALM macro-regions overtake OECD90 toward at the end of the century (2075-2090) as indicated in Figure 2.7, and the total envelope is maintained reasonable (the share of nuclear electricity production in 2100 is about 25%).

In addition, for the period 2100-2200 a slight energy demand increase (0.25%/year) has been considered as indicated by [102].

As for the INPRO study, different nuclear systems and fuel cycles have been considered for covering the energy demand. A diversification about the technology implemented in each region has been considered too. Two fuel cycles have been compared (the "once-through" fuel cycle and the closed fuel cycle with transuranics (TRUs) multi-recycling) as well as two different FRs.

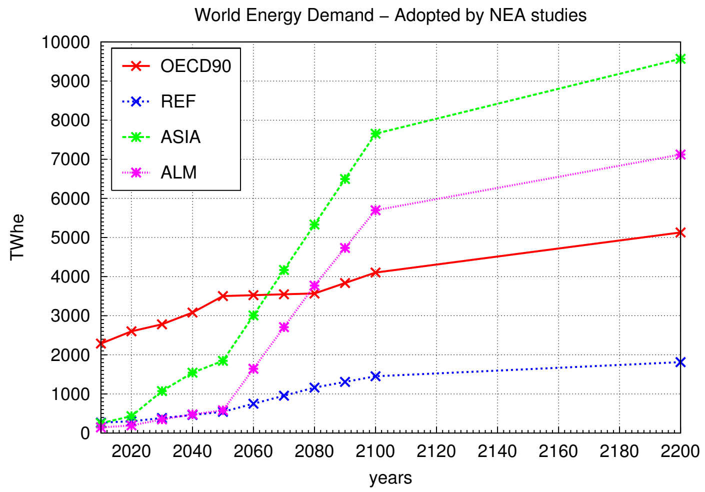

Figure 2.7: Nuclear energy demands (TWhe): regional subdivision adopted for the NEA/OECD study [4]

In particular, a self-sustaining sodium cooled fast reactor (with BR ca. 1) based on the EFR concept [103] and a sodium cooled fast breeder system with higher BR (ca. 1.5) [104], assuming the same power density, MW/tonnes of Pu equivalent, as the Superphenix core and a composite doubling time (CDT) of 17.8 years with in-pile fuel irradiation time of 1200 days and ex-core time of 5 years. In order to reduce the CDT the ex-core time has been reduced to 2 years (then the CDT ca. 11.7 years).

The homogeneous case has been oriented to the analysis of the resources involved and to the assessment of the time in which a possible resource shortage can happen [4] by the maximization of the FRs introduction (according to the Pu available in the cycle).

The adoption of self-sustaining systems has been compared with the introduction of breeders (assuming different CDTs as mentioned above). The adoption of FRs with BR ca. 1 indicates that only a slow stepwise deployment schedule is possible in agreement with the Pu available in the cycle (the Pu free circulation between macro-regions is allowed). On the contrary, the use of FRs with higher breeding characteristics, recycling their own fuel, allows more flexibility in regard to Pu management.

In agreement with [33], LWRs remain a dominating part of the nuclear world fleet until the end of the century. Depending on the system implemented, the transition can be very different. If self-sustaining systems are adopted, the full energy demand can be covered by FRs starting from 2200. If breeders with CDT of ca. 18 years are adopted the substitution happens ninety years earlier (in 2110 instead of 2200) and before the end of the century (2090) if systems with shorter doubling time are considered (CDT ca. 12 years).

The introduction of FRs has advantages from the sustainability point of view. In particular, the cumulative consumed uranium mass is significantly reduced.

The advantages of using self-sustaining FRs help in reducing by 2100 the uranium needs by ca. 14% that becomes 37% for the breeders with CDT of 18 years and 53% for the breeders with CDT of 12 years (the reduction assesssed for 2200, is respectively 41%, 81%, and 87%).

Under "once-through" fuel cycle conditions, the maximum value is exceeded around 2150, the cumulative value reached in 2100 (ca. 18 Mtons) is comparable with the INPRO study (LOW scenario) [55].

A heterogeneous approach has been developed for demonstrating more details concerning the uranium demands [4].

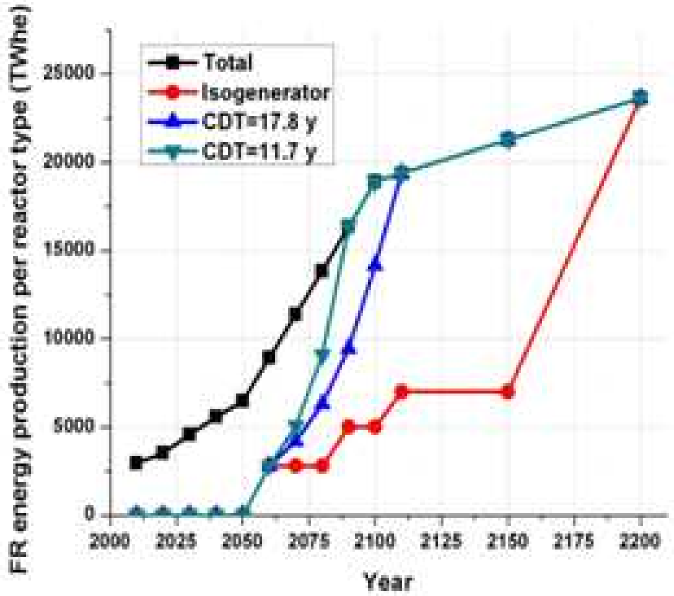

Figure 2.8: Nuclear energy production share: different FR options contribution [4]

Different breeder reactors are considered for the different regions i.e. high breeders for ASIA and ALM and self-sustaining systems for OCED90 and REF.

In order to remain well below the natural uranium limit, all regions have to develop FRs. In fact, under the assumption that only ALM remains with LWRs-based fuel cycle and the rest of the world introduces FRs, it has been noticed that conventional resources limits is exceeded in 2124.

The increasing pressure on uranium market and consequently on uranium ore price should be expected not later than at the end of the present century. Moreover, the increase of mining needs of unequally distributed resources can be a factor of uncertainty with an impact potentially more important than uranium cost considerations.

In agreement with the IAEA study, also the analysis performed by NEA points out the strong challenges associated to the transition to FRs.

In particular, suitable fuel cycle infrastructures (fabrication and reprocessing plants) especially in the world regions that presently have limited (or no) nuclear power plants are crucial points and source of uncertainties for the global development. In fact, the needed fuel fabrication and spent fuel reprocessing capacities should increase by at least one order of magnitude.

Fuel cycle facilities for uranium extraction, enrichment, fabrication, reprocessing and storage of spent fuel and retrieved fissionable material must be technologically feasible and successively built in order to manage efficiently a strongly increasing fuel supply required for rapid transition.

Additional activities are ongoing within the WPFC of NEA [100].

Other studies performed by individual associations [85, 84, 88, 87] confirm the same trends as the IAEA and the OECD studies.

2.4 Regional-oriented Studies

For a world oriented scenario as indicated in Par. 2.3, the hypothesis of shared facilities among all the macro-regions has been adopted in several studies [55, 84, 88, 87] with the major aim to maximize the transition to FRs. Other studies, e.g. [4], indeed consider shared facilities only within each macro-region in order to minimize the costs associated to the implementation of advanced fuel cycles.

In order to refine the analysis of the world average trends indicated before, scenarios can be oriented to the detailed regional studies. An overview of the available studies in Europe is reported in the next sections. In particular Europe has been selected because the diversity of the strategy adopted by each country can make quite attractive the investigation of a common strategy for fulfilling different objectives.

According to [47, 48, 105], common objectives have been fixed in order to envisage the long-term nuclear energy sustainability even though the present European situation is very heterogeneous. In fact, the approach includes ongoing nuclear energy as well as gradual phasing-out strategy.

In all these approaches a common objective is the waste inventory reduction, several solutions (Partitioning & Transmutation, fuel cycles and systems) can be considered. The most interesting one should be the adoption of shared facilities dedicated to the waste burning for minimizing the number of facilities to be installed and therefore the investment costs [106].

The idea to develop a scenario with shared facilities has been launched in 2005 by [10, 107] and developed in [5]. Preliminary studies was developed by [108].

As indicated in [10], Partitioning & Transmutation (P&T) can be developed either at national or regional levels. The adoption of regional facilities can allow the reduction of costs for the implementation of P&T and can result in complementary advantages for countries with a phasing-out strategy and countries with an ongoing nuclear energy policy.

The strategy to be adopted can be different according to the fuel cycle and facilities, e.g. adoption of dedicated MAs transmuters (e.g. Accelerator Driven Systems, ADSs) or TRUs multi-recycling in FRs. Both of them have been investigated. A short overview of the results achieved is presented in the following.

Under a regional scenario the optimization of the use of resources (e.g. by the recycling of Pu coming from phasing-out countries) and the waste minimization are objectives that can be achieved. These two objectives help in improving the long-term nuclear energy sustainability [5].

Other regional studies can be oriented to analyze the advantages (and the penetration dynamics) of the introduction of a new technology in an existing nuclear fleet [58, 88, 109, 110].

2.4.1 Regional Fuel Cycle Analysis: Europe as model

Europe can be subdivided into a few groups according to the adopted nuclear energy strategy [5]. There are essentially four groups:

- Group A: stagnant or phase-out scenario for nuclear energy. The main objective is to manage the spent fuel;
- Group B: continuation of nuclear energy. The main objective is the optimization in the use of Pu for FRs future development and the stabilization of its own MAs;
- Group C: is a subset of Group A that consider a renaissance after stagnation;
- Group D: countries that decide to include nuclear in their existing energy mix.

The studies described in the following are mainly related to the interaction between Group A and Group B. Different fast spectrum concepts are adopted both for nuclear energy production and/or waste burning.

In particular, scenarios 1 and 2 analyzed the introduction of sub-critical systems (Accelerator Driven Systems, ADSs) shared between Group A and B to manage MAs arising from the two groups. Therefore, Group B that has an ongoing energy strategy, can keep its Pu for future development. In particular, scenario 1 considers that Pu is mono-recycled in PWRs as MOX fuel for leaving sufficient amount of Pu (and of good quality) for future fast reactors deployment. Scenario 2 considers that Pu is continuously recycled in PWRs (in this scenario no fast reactor deployment is considered).

The scenario 3 considers a fast fleet development in group B where MAs coming for the two groups are homogeneously burned in FRs.

The scenario 4 considers that Group A has a renaissance deploying FRs and recycling its own TRUs.

By the results obtained, a more clear understanding of the challenges associated to the P&T implementation in Europe has been provided. In particular, the advantages connected to the introduction of ADS or FRs has been extensively analyzed [5].

Scenario 1 and 2

Scenarios 1 and 2 are oriented to analyze the influence of the ADSs (and related fuel cycle) introduction in the European context considering ADSs as shared facilities between Group A and Group B. The schematic diagram representing the study is shown in Figure 2.9.

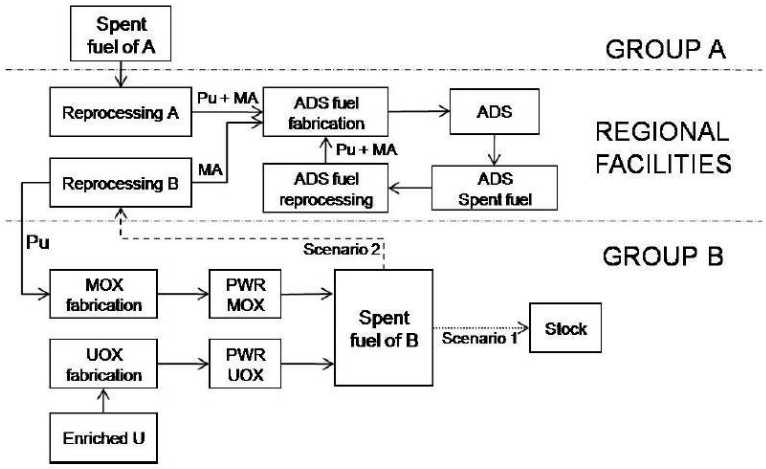

Figure 2.9: Schematic diagram of scenarios 1 and 2 of PATEROS Project [5, 6]

The main objectives are:

- decrease down to zero the stock of spent fuel in Group A by the end of the century,
- stabilize MAs in Group B,
- investigate the required number of ADSs to be developed in order to achieve the previous two objectives,
- determine the number and the capacities of the fuel cycle facilities needed (reprocessing, fabrication, etc.),
- stabilize the Pu inventory in Group B.

The ADS-type adopted is based on the European Facility for Industrial Transmutation (EFIT) [111]. It is a small size lead sub-critical (keff equal to 0.97) system (384 MWth) loading uranium free fuel for enhancing the MAs transmutation.

The fuel adopted is a CERamic-CERamic (CER-CER) fuel composed by 50% of inert matrix (MgO) and 50% of fuel (55% are MAs and 45% is Pu corresponding to a MA/Pu ratio of 1.2) [112, 113].

In order to stabilize MAs coming from Group A and Group B, 25 ADS EFIT units are required in the case of scenario 1. By this assumption, both SF stock of Group A and Group B are reduce down to zero [5] obtaining a 50% reduction of MAs in the cycle by 2200, as more clearly indicated in Figure 2.10.

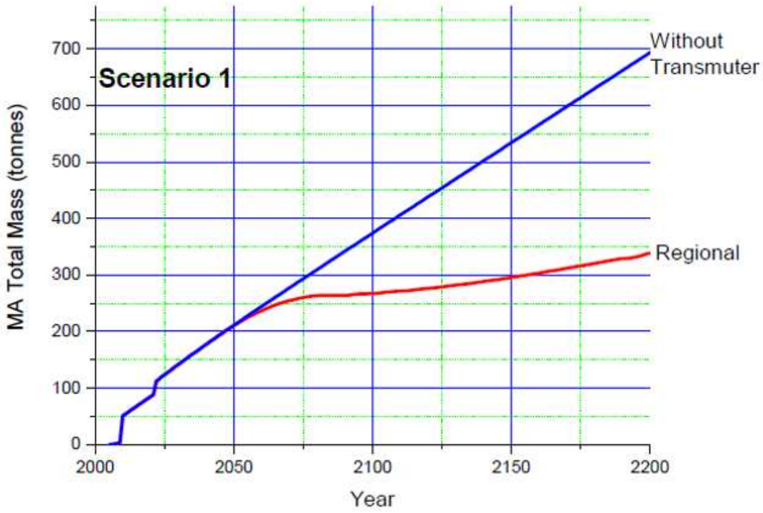

Figure 2.10: Scenario 1, total MAs cumulative mass compared with "no transmuter" case [5]

Adopting a multi-recycling of Pu in PWRs (scenario 2) the number of ADSs needed to stabilize the MAs quantity is increased to 27 units [5]. The MAs in the cycle are in 2100 equal to 265 tons (of which 13% Cm due to the Pu multi-recycling in LWRs) that corresponds to 69% of the case without transmuters [5, 6].

A different number of units has be found for stabilizing the MAs inventory of Group B instead of reducing it to zero. These additional studies have been performed at KIT [104, 62, 64, 65]. .

Scenario 3 and 4

The main objective of scenario 3 is the consumption of TRUs coming from Group A (at a reasonable time horizon) adopting homogeneous burning in fast systems with different breeding characteristics[7] [114].

Other parameters affecting the transition scenario as well as the TRUs consumption are the initial inventory considered, the data and pace of FRs deployment, the load factor, the LWRs SF burn-up and the use of MOX fuel in LWRs [5].

The schematic diagrams representing the scenario 3 is indicated in Figure 2.11.

The adoption of MAs multi-recycling in FRs is shown in Figure 2.12. The initial cumulative MAs content is reduced when FRs start and then these MAs are transmuted in FRs as soon as they are deployed (this explains the initial reduction and then stabilization).

[7] According to [114], it was noticed that a critical burner system with negative BG (-0.061 i.e. BR < 1) whatever the out-of-pile time considered can consume the TRUs of Group A within less than 100 years.

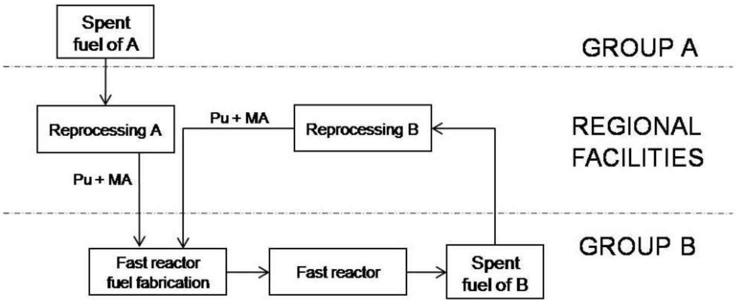

Figure 2.11: Schematic diagram of scenario 3 of PATEROS Project [5, 6]

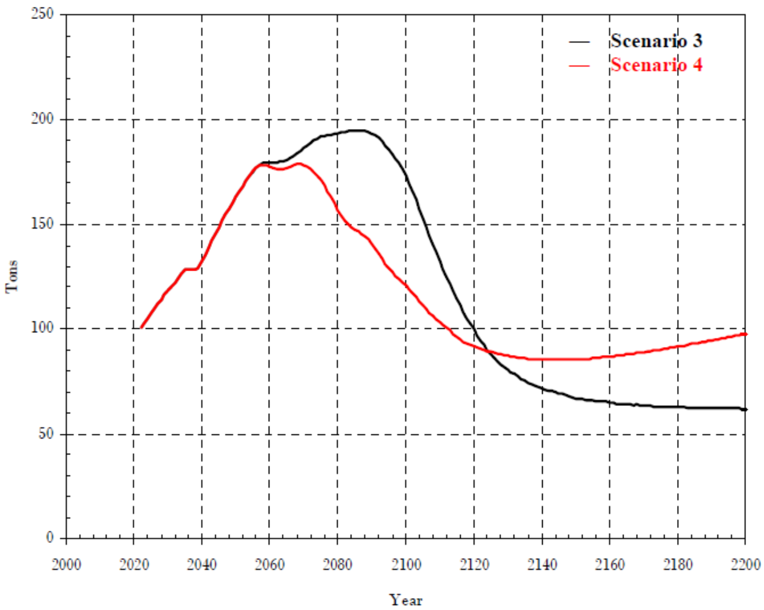

Figure 2.12: MAs inventory in the cycle: comparison between scenarios 3 and 4 [5, 6]

These kinds of study confirm the wide range of applicability of critical FRs, as proposed in previous CAPRA/CADRA studies [115, 116, 117], the reversibility from burner to breeder of FRs could be useful to manage Pu and MAs in several countries [5].

In addition, the study exemplifies the advantages of developing a common European strategy for P&T implementation.

The adoption of the ADS concept shows advantages in terms of regional scenarios. However, the development of this machine seems to be not completely reasonable for countries that want to develop P&T in isolation. In this direction the adoption of critical burners (with CR ca. 0.5 or 0.8) results more attractive as shown in complementary studies performed at KIT [104, 62, 64, 65].

The implementation of P&T at a regional level gives potential advantages to all the countries in the region despite their different policies in terms of nuclear energy. In fact, the radiotoxicity and the heat load and, hence, the source term associated to the geological disposal are substantially reduced by the P&T implementation [5, 6].

2.4.2 Comparison of Transmutation Systems capabilities

As alternative to ADSs, different kinds of burner systems can be considered as low Conversion Ratio (CR) fast reactors [104, 62, 64, 65] or Hybrid fusion-fission sub-critical systems (SABR) [64].

The results of the neutronic analysis show that the total consumption rate of the fast "burner" cores (27.1 kg/TWh for MA/Pu 0.1 fuel and 29.3 kg/TWh for MA/Pu 1.2 fuel) reaches ca. 70% of the ADS rate (41.9 kg/TWh) in agreement with the theoretical results obtained in [118].

Both systems (ADS and critical burners) show comparable results in terms of MAs stabilization as indicated in [104, 62]. Therefore, low conversion ratio critical fast reactors offer a valuable and potential alternative to U-free fueled sub-critical systems, like ADSs, in order to burn significant amounts of TRUs or MAs, thus stabilizing the total value in the region.

However, big challenges are associated to these advanced technologies. In particular, the fuel fabrication, where the high MAs content implies the fabrication under remote handling equipments in hot cell or shielded box installations due to high heat generation and gamma, alpha and neutron emissions, requiring fabrication and reprocessing facilities installed in the same place of the transmuters (i.e. avoiding material movement) [62, 64, 108] and the accelerator performances.

As additional term of comparison, the Sub-critical Advanced Burner Reactor (SABR) has been considered. It is a sub-critical fast reactor driven by a D-T fusion neutron source able to load different MAs contents (up to 100% of TRUs) in the sub-critical core [64] developed by Ga-Tech, USA [119, 120, 121].

Several studies are ongoing for further analyzing the preliminary trend obtained [65].

2.5 Country-oriented Studies

An overview of the fuel cycle studies ongoing in several countries is presented here. This overview does not want to be completed. Only some significant examples have been investigated and here described.

The regional studies described above have shown the potential of the P&T implementation, with shared objectives and facilities in a region.

This paragraph will demonstrate the complex dynamic nature associated to the infrastructures needs for the transition from open to closed fuel cycles. This kind of issue that strongly can affect the transition to FRs is country-specific if addressed in detail [8].

Several examples for the national transition scenarios are presented to provide a kind of comparison for the activity performed.

A certain scenario corresponds to a foreseen national development scenario (e.g. France case), others are only hypothetical development scenarios (e.g. Belgium). However, all those cases have here been introduced because they have been used as example of comparison for the hypothetical scenario analyzed in detail during the Ph.D. activity (results summarized in Chapter 4 and Chapter 5).

National objectives and strategies, indeed, can affect the advanced fuel cycle implementation as well as the choice of the technology to be used. An overview of some objectives and drivers are reported in Table 2.5.

Objective drivers	Means to meet the objectives	Technology requirements
Enhance proliferation resistance, facilitate waste management and disposal	Minimize and monitor flows of separated Pu239, Am241 and Tc99, control of reactor operation, surveillance of NPPs and nuclear facilities	Advanced spent fuel reprocessing, specific fuel and target forms, specialized storage/disposal media, particular detectors, monitors and devices
Reduce number and/or size of HLW repositories	Reduce heat and dose at the contact of waste packages	Same as above plus decay storage for Cs137, Sr90
Minimize environmental impact	Reduce radiotoxicity of waste, dose at the contact of the repository, reduce effluents	Same as above plus pay attention to waste streams at all fuel cycle steps, including fuel fabrication and reprocessing
Enhance security of energy supply	Increase the lifetime of natural resources	Recycling and breeding

Table 2.5: National energy policy objectives and associated technology requirements [8]

For instance, a country in phasing-out strategy can be interested in an advanced fuel cycle in order to deal with waste management, a country in stagnant nuclear energy share situation will chose advanced fuel cycles on the basis of cost-benefits analysis and countries with an ongoing energy policy will be attracted by advanced fuel cycles able to guarantee the long-term security of supply [8].

All these aspects together with the initial conditions in the country as well as the availability of the new technologies (reactors but also infrastructures as the reprocessing and fabrication plants for innovative fuel and plants, e.g. for waste vitrification) determine the choice adopted in the single country and then, it can modify the regional and global trends analyzed before. An example is the delay on implementing a strategy (e.g. transition to FRs) that can affect the availability of resources and the security of supply at world level.

The studies are presented according to three groups: 1) OECD countries with ongoing nuclear energy (e.g. France, Canada, Japan, Korean), 2) Other OECD countries (e.g. Germany) and, 3) non-OECD countries (China and India).

2.5.1 OECD countries with ongoing nuclear energy programs

A short overview of the studies ongoing in France, Japan, USA is summarized in the following parts.

France

Advanced fuel cycles are studied in France since long time with the main objective to optimize the use of resources, to envisage the partitioning of MAs for the next reactor generation maintaining proliferation resistance and economical competitiveness [8].

France has the largest nuclear energy production in Europe. The 58 PWRs are producing every year ca. 430 TWhe corresponding to 78% of the total gross electricity generated [122]. The policy considered by France, as also indicated in the Act of June 2006 [123], is to mainain the actual level of nuclear energy production reducing the MAs sent to disposal and deploying the future facilities [8].

Three steps for the French future scenario are investigated (see Figure 2.13): 1) replacement of the 50% of operating fleet by the available technology (i.e. EPR developed by AREVA [124]) for the short-term (period 2020-2040); 2) replacement of the 50% of the remaining old fleet by FRs systems (or EPRs if the technology is not available) for the medium term (period 2040-2050); and 3) replacement of the first EPRs installed as Gen-IV systems for the long-term toward equilibrium situation (2080).

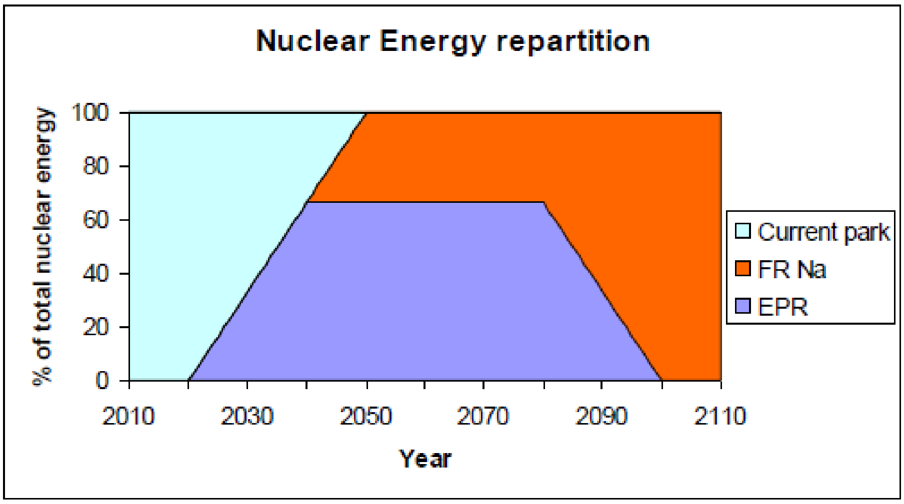

Figure 2.13: French future reactor substitution strategy [7]

The reference scenario considers the Pu mono-recycling in EPR (in agreement with the actual policy where 22 reactors are licensed for using MOX fuel [125]) and then the multi-recycling of second generation Pu (and the remaining first generation Pu) in FRs.

For a homogeneous recycling situation, the fuel contains 1.2% MAs (Am, Np, Cm) and ca. 20% Pu content (values that depend on the FRs considered as it is shown later on).MAs are then stabilized to ca. 86 tons in 2100 and natural uranium needs can be reduced by about 30-40% (in agreement, e.g. with the world study performed at KIT [4]).

These data can be affected by several scenario options (e.g. burn-up or out-of-pile time) but they represent reasonably well the average values for the transition from a 100% LWRs fleet to a FRs fleet. The same order of magnitude has been found also for the reference case studied within this Ph.D. activity as described in Chapter 5.

Several alternatives have been analyzed. Some of them are oriented to investigate the final inventory if FRs are not deployed, e.g. considering only the mono-recycling of Pu in EPR or multi-recycling of Pu in EPR. All the studies have confirmed that, in order to properly transmute MAs, the adoption of FRs is essential. The expected favorable contribution is confirmed by the radiotoxicity behavior shown in Figure 2.14.

Several studies have been developed in the past at CEA (France), some examples (concerning also the investigation of Pu and MAs burning in LWRs under the assumption that FRs will not reach the expected safety level and, therefore, its industrial application) are reported in [125, 126, 127, 128, 129, 7, 9].

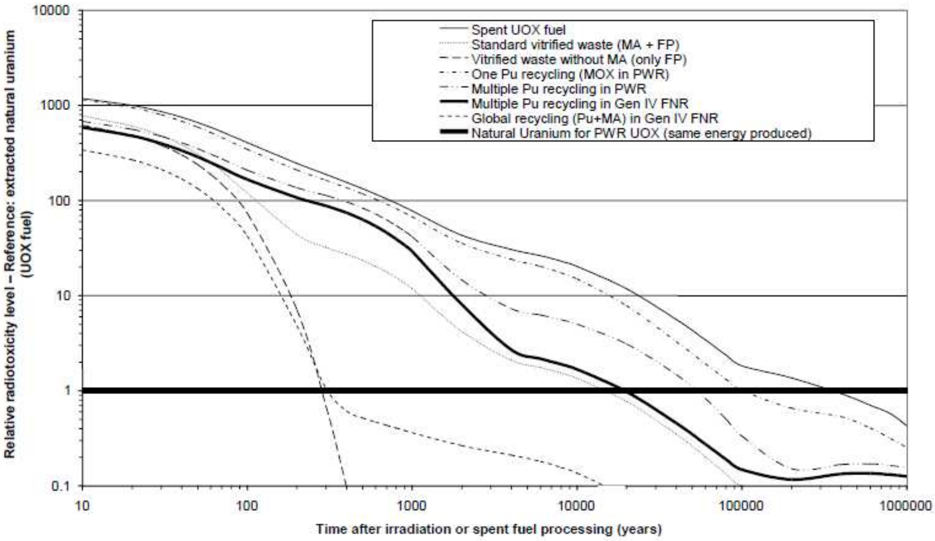

Figure 2.14: Radiotoxicity level of the TRUs disposed in the storage [8]

Additional studies have been performed considering the MAs separation and burning in ADS systems [128]. It has been concluded that the advantages of the use of an ADS in a double strata scenario compared to FRs with homogeneous MAs multi-recycling are not sufficient to justify this choice for a country with a continuous use of nuclear energy policy ongoing (as confirmed also by [130]). However, the adoption of an ADS in a regional strategy should be more feasible [5, 11].

The systems considered in the transition as well as the associated breeding capabilities can strongly impact the country-oriented scenario (and not only the world scenario as shown in Par. 2.3). According to [129], the comparison of three fast reactors has been performed in order to evaluate the ability of the systems for renewing the French fleet[8].

All systems considered in [129] are able to renew the current fleet in the period 2035-2100. Assuming Pu and MAs multi-recycling, a stabilization of TRUs can be achieved too [129].

Further studies about MAs recycling in FRs have been carried out at CEA during the last two years (e.g. [7, 9]). The homogeneous and heterogeneous multi-recycling of MAs in sodium cooled fast reactors has been further analyzed [133]. These studies have shown how Am recycling plays the major role for the radiotoxicity reduction [9].

The addition of Am to depleted uranium blanket as well as to fuel core composition has been considered as an option also within the CP-ESFR project oriented to develop an innovative industrial scale sodium fast reactor able to fulfill the Gen. IV goals [58, 31, 30]. More details are included in Chapter 5 and Appendix D of the present study.

These scenarios show that the total Pu inventory can be stabilized (after the transition) in the case with only Pu multi-recycling. Concerning MAs, only the case with MAs multi-recycling attains a stabilization of the total amount[9] (see Figure 2.15). In fact, in the case of Am multi-recycling the Cm and Np produced

[8]In the study, the systems compared are the Na-cooled European Fast Reactor (EFR) developed in the nineties and characterized by a negative core BG (-0.2) and by a high power density (300 W/cm3) [103], the He-cooled Fast Reactor (Gas-cooled Fast reactor, GFR) characterized by positive BG and lower power density (100 W/cm3) [131] and a lead-cooled FR (BREST-OD-300) with an equilibrium BG of 0.08 and a power density of 150 W/cm3 [132].

[9]Scenario 1 considers only Pu multi-recycling in FRs and MAs are sent to disposal; Scenario 2 considers Pu multi-recycling in FRs core and MAs in radial blankets; and Scenario 3 considers Pu multi-recycling in FRs core and Am in radial blankets, Cm and Np are sent to disposal [9].

in the FRs increase slightly, thus modifying the total amount. Similar results have been obtained for the reference case studied within this Ph.D. activity (see Par. 5.2).

Recent studies [125, 134] concerning the industrial research for transmutation scenarios have been developed taking into account an advanced Na-cooled fast reactor (similar configuration as the model studied within the CP-ESFR project [135, 58]). The Am and MAs transmutation advantages have been compared.

The heterogeneous recycling by the adoption of Americium Bearing Blanket sub-assemblies (AmBB) is preferable because the number of SAs to be treated is lower and therefore the requested fabrication plant capacities remain limited (limiting the cost too). However, a higher thermal power and a more difficult handling strategy are drawbacks well known for this solution [125].

Japan

Despite the last event (Fukushima) in Japan, nuclear energy will probably continue to play an important role for a country that imports ca 96% of its energy resources. The 53 nuclear power plant installed are able to cover ca. one-third of the electricity needs [8].

Since the eighties, in Japan, the development of advanced fuel cycles has promoted a better use of the resources and associated ambitious research programs have been launched (as the OMEGA program oriented to "Options Making Extra Gain from Actinides and Fission Products" to reduce the high-level radioactive wastes [8]). In particular, the recycling of MAs in ADSs and FRs has been considered.

The analysis of the possible fuel cycle options can be restricted to four main cases: 1) LWRs "once-through" case; 2) Partial reprocessing scenario where only one part of the SF is reprocessed, the remaining part is directly disposed; 3) Total reprocessing with Pu utilization in thermal reactors and Pu and MAs (Am, Np, Cm) recycled in FRs from 2050; and, 4) Interim storage scenario where FRs are deployed in 2050 after SF interim storage (no recycling in LWRs).

These cases have been compared (for more details see [8]), an example is reported in Figure 2.16 where cumulative natural uranium resources are analyzed for the several scenarios. This aspect is particularly important in country as Japan obliged to import all the material and therefore eager to guarantee the long-term security of supply.

The study performed in Japan is in agreement with the French case. It shows the more favorable behavior of a FRs-based scenarios in terms of environment burden and natural uranium demands. In order to limit the build-up of Am241 from Pu241 decay as well as the accumulation of Pu stock (improving the proliferation resistance), an integrated reprocessing technology, called Flexible Fuel Cycle Initiative (FFCI), is under development and study in Japan [136].

Korea and Canada

Several studies have been performed also in Korea (e.g. [137, 8]). Korea has a total installed capacity of 17.7 GWe supplying 39.0% of its electricity needs.

The Korean fleet is composed of 16 PWRs and 4 PHWRs installed late in the eighties [17]. Due to fairly short period of operation up to now in Korea the main objectives are presently not related to the replacement of the fleet as indicated for France (Par. 2.5.1).

The main requirements for the Korean scenarios are that: 1) the accumulated PWR spent fuel shall keep below 20ktonHM (value assessed according to the estimated capacity requirement for the repository at present), and 2) the accumulated uranium demand shall remain below 5.0% of the identified uranium resources in the world (share that corresponds to the actual share).

In order to fulfill these objectives seven different fuel cycles have been compared [137, 8].

A parametric study concerning the adoption of SFR with different conversion ratio has been recently published [137, 138].

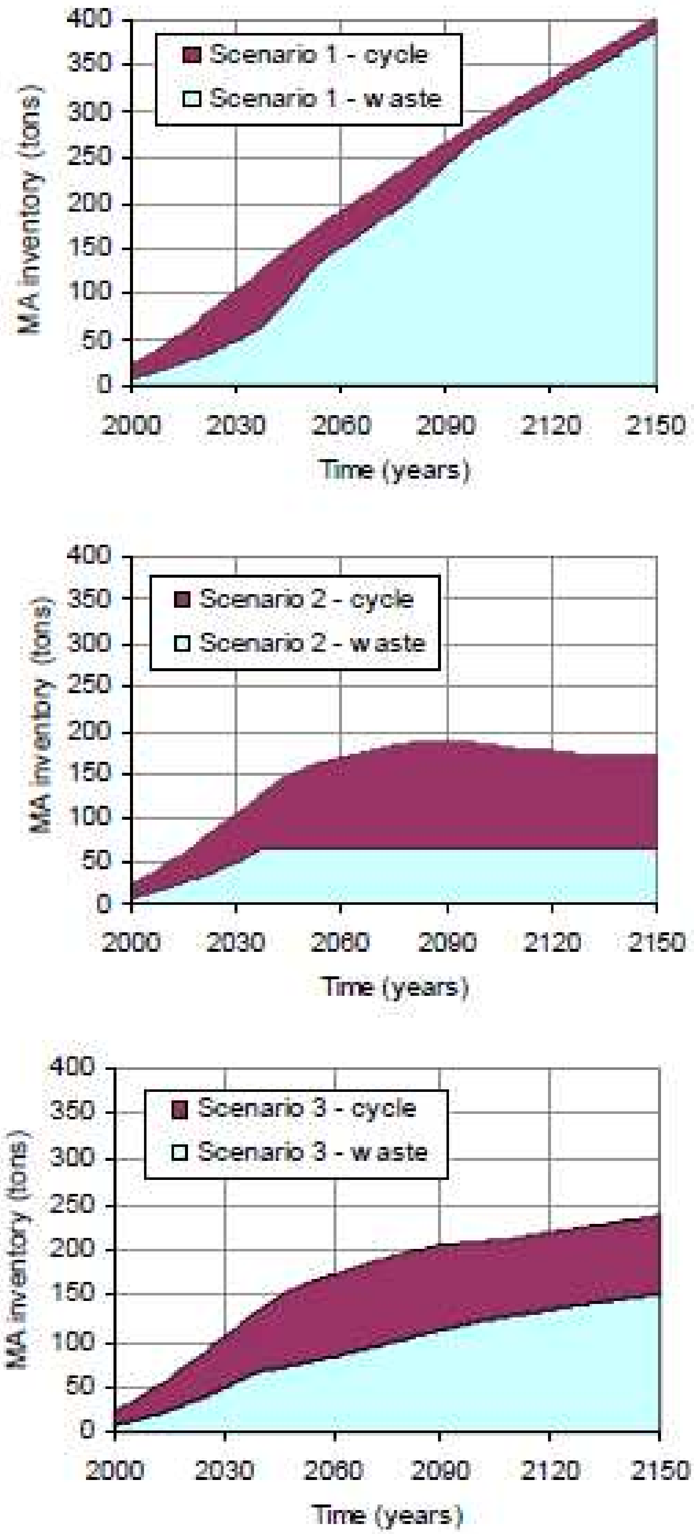

Figure 2.15: Minor Actinides inventory in cycle and in waste for the three cases considered in [9]

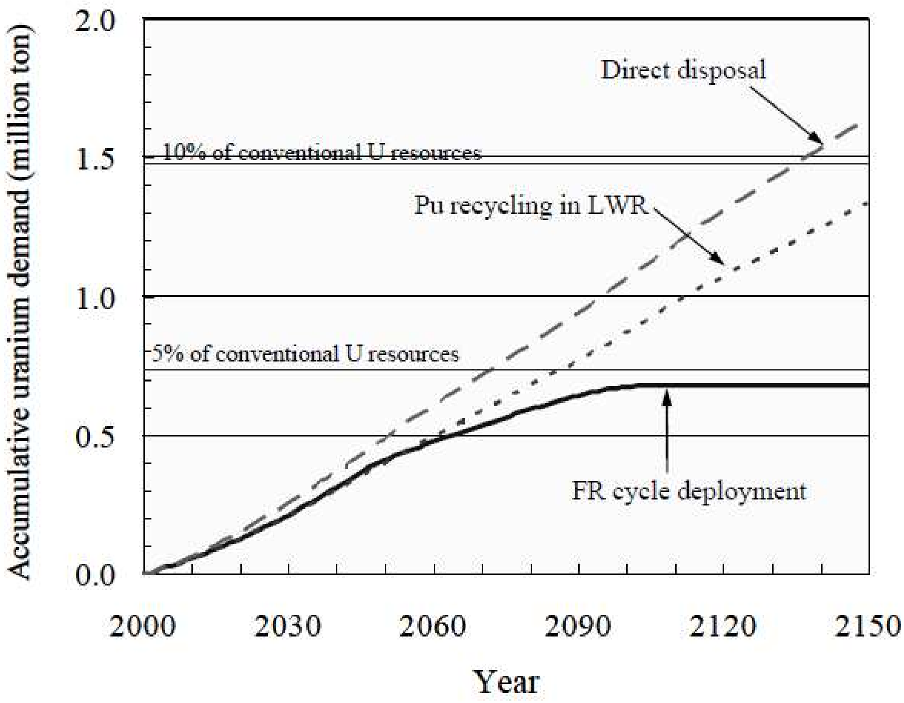

Figure 2.16: Accumulative uranium demands for Japan [8]

The study has shown that a complete substitution of the thermal fleet by FRs with CR equal to 1 (i.e. self-sustaining system), leads to 50% reduction of the natural uranium use, 44% reduction of UOX fuel fabrication and 50% reduction of total out-of-pile TRUs amount. Comparable results have also been obtained by own investigations of similar substitution scenarios. Essential findings of that particular study, forming a constituent part of the present work, are outlined more precisely in Chapter 5 and in [139].

The Canadian case is an interesting scenario because it opens the attention to other kind of systems as the CANDU technology is originally based on natural uranium and heavy water. The transition from thermal-to-fast reactors has been analyzed for Canada too [8].

The adoption of Heavy Water reactors (HWRs) facilitate the transition toward fast reactors because of their good conversion capability enabling to provide the first fissile loading.

In addition the option of combining the Th-fuel cycle with CANDU reactors enables the generation of U233 while significantly extending uranium resources [8, 140].

2.5.2 Other OECD countries

Preliminary studies are ongoing also in other OECD countries. The example of Germany is here presented.

Germany

R&D activities are ongoing in Germany related to waste management issues under phasing-out strategy.

Several studies have been performed [11, 10] in the past, mainly oriented to the implementation of burner systems for getting rid of the TRUs produced.

According to [11], in order to fulfill the goal of the inventory reduction within the present century, two ADS generations are needed considering a total number of ADSs equal to 8 units (for the first generation) and 3 for the second generation (assuming as reference the AFCI/AAA model [141]).

The effects of the ADS deployment on the time-dependent evolution of Am, Cm and Pu reduction are shown in Figure 2.17.

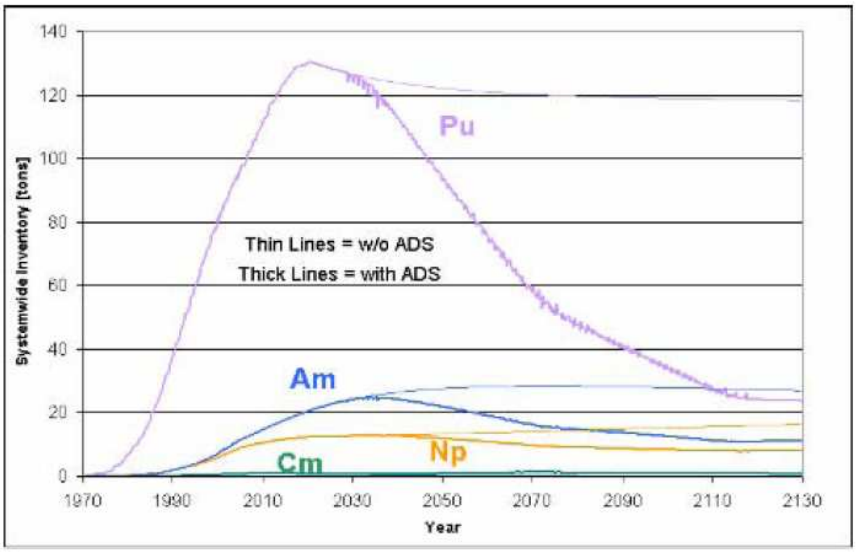

Figure 2.17: The Effect of ADS Deployment on Transuranic Inventories in the German Fleet [10, 11]

2.5.3 Non-OECD countries: China and India

China is currently the most growing economy in the world, with an increasing economy annual rate of 7-10% [34].

The energy demand is expected to grow before 2050 reaching predicted capacity of electricity production of about 1200-1500 GWe [34]. Therefore, in the Chinese energy mix nuclear energy is an option.

Some recent projections are indicated in Table 2.6 [34, 35].

Scenario	% tot el.	% primary energy	Nucl. Capacity	Approximate Scale
Low Level	10	6	120 GWe	Twice of France
Middle Level	20	12	240 GWe	Sum of US, France and RF
High Level	30	18	360 GWe	Sum all over the world

Table 2.6: Possible nuclear future development in China [34, 35]

For identifying the needs in terms of infrastructures, resources and to assess the SF produced, preliminary scenario studies are ongoing in China on the basis of the following strategy: 1) introduction of thermal reactors, 2) transition to fast reactors and 3) development of fission-fusion systems [142, 143].

In order to meet the goal fixed for the nuclear capacity in 2050 (see Table 2.6) e.g. for the Middle Level case, the natural uranium request can be of the order of 2.5 Mtons (ca. 16% of the world estimated conventional resources) [142].

Such a huge amount of uranium is not available in China. Therefore, for providing the security of supply, the adoption of a closed fuel cycle based on PWRs and fast breeder systems (FBRs) is essential. Studies are ongoing for analyzing the transition to FRs (e.g. [142]).

In addition, in order to cope with the issues associated to the waste treatment, the comparison of several strategies have been performed [143]. In particular the "once-through" case is compared with the Pu recycling in thermal reactor systems and with the Pu+MA multi-recycling in Gen-IV systems.

A reduction of 24.6% of the total SF in 2050 is achieved by adopting the Pu+MA multi-recicling. More detailed studies are actually in preparation as indicated by [34].

Together with China, also India is driving the increasing energy demand for the coming years.

The main focus of the nuclear energy study in India has been directed to improve the efficient use of its energy resources.

India has low uranium resources but large amounts of thorium. Therefore, for the long-term guarantee of the security of supply, Th-based fuel cycle options are analyzed [144, 145, 146].

The strategy proposed considers the adoption of natural uranium in Pressurized Heavy Water Reactors (PHWRs) for enhancing the fuel utilization and the conversion of Pu due to the good neutron economy of the system (first stage).

The Pu is then used for fast breeder systems in order to cover the rapid growth of the nuclear energy capacity where Th232 loaded in blankets is converted to U233 (second stage).

The last stage (third stage) indeed considers the adoption of U233 generated in the second stage, as fissile for the fast breeder systems adopting again blankets with thorium [144].

In order to sustain this strategy, several studies are underway also for assessing the fabrication capacity for stage two and three. More details can be found in [90, 147].

2.6 Summary

In the present Chapter, an overview of existing fuel cycle and scenario studies has been included in order to show the applicability range of this kind of investigations.

According to the literature, the scenario studies can be subdivided into three main categories: world-oriented scenarios, regional-oriented scenarios and country-specific scenarios.

These three categories imply different levels of approximation and, therefore, different types of hypotheses and boundary conditions.

In this chapter the main differences and the associated consequences regarding fuel cycle characteristics have been pointed out in order to provide a basis for the following parts.

Chapter 3

Preliminary Scoping Studies

The previous chapters have provided an overview of the framework of the present activity. This chapter summarizes the preliminary scoping studies performed, oriented to the investigation of the boundary conditions. The results are used as preparatory work for refining the approach adopted.

In the present study a simplified "methodology" has been developed and assessed (see Par. 3.1).

On the basis of this simplified methodology, several scenarios (based on European context) have been defined and their analysis has been performed adopting the Nuclear Fuel Cycle Simulation System (NFCSS code) developed by IAEA [16], being the fuel cycle code available at University of Pisa (Italy) at that time (2008).

For comparing the scenarios proposed, suitable indicators for the nuclear energy sector have been chosen as described in Par. 3.2.

On the basis of preliminary scoping study, a "reference" case has been selected and used to generalize the results.

Furthermore, parameters that impact the results (e.g. burn-up of LWRs, breeding ratio of FRs[148, 12]) have been further investigated as described in Chapters 4 and 5.

The selected "reference" case is a country with a constant nuclear energy production (ca. 70 TWhe/y). This value has been chosen to be small enough in order to represent the "unit of measure" for complex increasing energy scenarios and large enough to investigate in detail some hypotheses (e.g. substitution of the LWRs fleet).

This kind of scenario can easily be generalized[1], by means of energy-based scale factors, as better explained in Chapter 4.

The preliminary scoping study has shown the limits of the NFCSS code (see Par. 3.4). Therefore, a more flexible tool, the COSI6 code [20, 149, 150], has been selected and applied.

This occurred at the Karlsruhe Institute of Technology (KIT, Germany) where the second part of the Ph.D. has been developed.

3.1 Methodology: the choice of the scenario boundary conditions

The analysis of nuclear fuel cycles could be considered as a partial effort to be appropriately included within a more general analysis: the analysis of nuclear scenarios. These studies are again one part of a more extended activity concerning the energy source Life Cycle Assessment (LCA) oriented to sustainable development. In this sense, links between the different studies (e.g. adoption of average energy demand trends) can be pointed out.

[1] For instance to other emerging countries (e.g. Armenia, Poland).

The state-of-the-art summarized in Chapter 2 have underlined how the choice of boundary conditions is a critical issue for nuclear scenario studies and therefore for nuclear fuel cycle analyses. The impact of the boundary conditions on the results could be extremely high as indicated by some examples (e.g. [4, 55]).

In general, this boundary conditions or "specifications" include a complete set of data as the energy demand, the reprocessing strategy, the systems considered and transition strategies (e.g. early date and pace).

In order to define in a reasonable way these boundary conditions, a simplified "methodology" has been assessed for checking that all the important parameters are taken into account, and particular attention has been devoted to the way for comparing the scenario results. Several indicators have been selected on the basis of the common criterion adopted for the study; among them only indicators that can be quantified have been applied to the analysis. More details are included in Par. 3.2.

The criterion selected for the study is *"the sustainable development"* of the nuclear energy sector; the same criterion adopted by the Gen-IV initiative (Sustainable goals [52]) and by the SNE-TP activities in Europe [48].

The concept of sustainability is quite extended and somehow not fully defined. An extended treatment of all the implications of the sustainability has not been performed because it is beyond the aim of the study. Some Information could be found in [151, 152].

However, the sustainability applied to nuclear sector has been analyzed. In a simple way, it can be explained as the analysis of innovative fuel cycles and reactor concepts able to guarantee the long-term security and reliability of supply in an economical and safe manner by reducing the impact on the environment[2].

This implies that several heterogeneous sectors, as the economy (by security of supply), the environment and the social sector (increasing acceptability and avoiding possible future conflicts for uranium holding), are linked together for developing a sustainable (and durable) condition [151, 152].

During the present activity, the analysis of possible strategies oriented to making in practice the sustainable development, at least for a specific energy source, has been performed.

For setting up the boundary conditions, the first point to address is the analysis of the actual situation in a specific country or region, as indicated by the scheme of methodology shown in Figure 3.1.

Then, future energy envelops could be defined on the basis of annual rates available in literature (up to 2050) for nuclear and total electricity productions[3]. A comparable approach has been adopted for world scenario studies as indicated in Par. 2.3 and in Appendix B.

For the preliminary scoping study, a limited period in time, from 2008 to 2050, has been considered. This choice has been assumed in order to focus only on some specific parameters (e.g. replacement strategy of the in-operation fleet) and to avoid the adoption of long-term energy projections. By this assumption, more reliable data for the energy trends have been applied (see Appendix B).

In the second part of the study (Chapters 4 and 5), the period of interest has been enlarged (up to 2200) in order to analyze a complete transition from LWRs to FRs and the effects of advanced fuel cycles under equilibrium conditions.

The data chosen for the preliminary scoping study have been extracted from the 2008 *Energy, Electricity and Nuclear Power: Developments and Projections. 25 years Past & Future* IAEA publications [15, 99].

The IAEA data have been updated by new publications (e.g. [153]) taking into account the last two years economical crisis. The 2008 trends remain quite unchanged, as expected for an industrialized area. An example is the nuclear energy production for Western Europe in 2030 estimated equal to 570 TWhe by [15] and to 666 TWhe by [153]. These differences could result in slightly different trends of the fuel cycle

[2]The proliferation resistance issue, related to the social sector, is not explicitly taken into account because the systems considered, mainly Gen.IV, are implicitly developed in order to assure this aspect.

[3]The short-term energy projection have been extracted by the most up-to-date international publications (IAEA, IEA, IPCC, IIASA and NEA [15, 89, 3]) available in 2008 as described in Appendix B

quantities (e.g. uranium needs), however, the calculations performed during the preliminary scoping study have not been repeated[4].

As indicated in Par. 2.5, each country is characterized by a particular nuclear energy demand depending on the local aspects depending on the energy mix.

The adoption of the same trends (i.e. annual rates) for defining the energy envelop of each European country gives, of course, an approximation. In order to take into account the existing differences, additional hypotheses have been added as described in Par. 3.3.

These average trends together with the analysis of the present situation in a country, have been used in order to define possible reactor substitution strategies for the in-operation systems. These strategies are dependent on the type of reactors considered.

For the preliminary scoping study, the Gen-III+ European Pressurized Reactor (EPR) has been selected as a suitable technology for replacing the in-operation systems in Europe. From 2040, the introduction of FRs can be envisaged and the European Lead-cooled SYstem (ELSY) has been considered a possible option.

Further details about the reactor models considered for the analysis are included in Par. 3.3[5].

Figure 3.1 shows the simplified block diagram representative of the methodology adopted for defining the scenario boundary conditions.

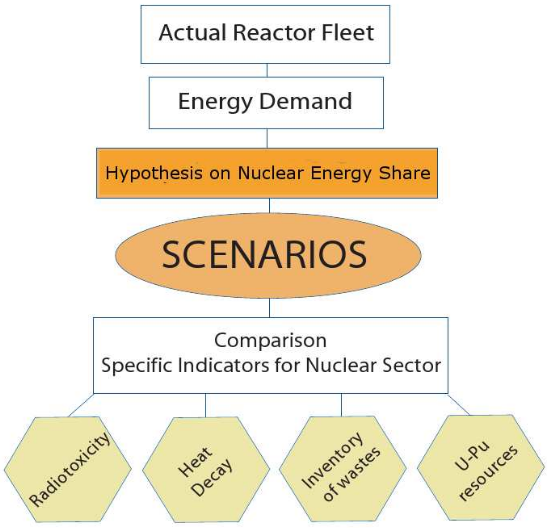

Figure 3.1: A simplified block diagram of the methodology proposed for the scenario analysis [12]

As shown in Figure 3.1, the comparison of the scenarios is done on the basis of four main indicators:

- the natural resources involved (uranium), and the Pu needed for FRs start-up;
- the total waste inventory and its isotopic composition;
- the radiotoxicity evolution of the material sent to the geological repository;
- the decay heat evolution of the material sent to the geological repository.

[4]In order to take into account this kind of aspect, the impact of the energy variation on the fuel cycle parameters has been considered in the parametric study, see Chapter 4

[5]The description of the neutronic model for the ELSY-like system is included in Appendix D

In addition to these indicators, the infrastructures needs have to be assessed. In fact, they can impact the scenario studies introducing additional limitations (e.g. reprocessing capacities underestimated can impact the dynamics of the transition) as also pointed out by other studies (e.g. [55]).

During the preliminary scoping study, the infrastructures needs have not been analyzed, but they are considered in Chapters 4 and 5.

3.2 Sustainablility Indicators

Starting from the analysis of the available indicators for the energy sector, a selection of the most suitable ones for the aims of the present activity, has been performed.

The need to quantify the sustainable development starts with the Agenda 21 [154, 152]. In order to provide a background for the decision-making process, several indicators, covering all the dimensions and aspects of the sustainable development, have been proposed by the analysts [13].

Dimension	Indicator	Unit
Economy	Production cost	c/kWh
	Fuel price sensitivity	-
	Availability (load factor)	%
	Geopolitical factors	Relative scale
	Energetic resource lifetime	Years
	Non-energetic resource consumption	kg/GWh
	Peak load response	Relative scale
Environment	CO_2 equivalent	tons/GWh
	Change in unprotected ecosystems	km^2/GWh
	Land use	m^2/GWh
	Fatalities (Severe accident)	Fatalities/GWh
	Weight (Total waste)	tons/GWh
Social	Technology job opportunities	Person-years/GWh
	Potential (proliferation)	Relative scale
	Mortality	Years of life lost/GWh
	Noise, virtual amenity	Relative scale
	Confinement time	Thousand of Years
	Number of fatalities per accident	Fatalities/accident

Table 3.1: Illustrative set of technology specific indicators for the energy and electricity sectors [13, 36]

Focusing on the energy sector, the proposed indicators can be classified in agreement with the three "pillars" of the sustainability as described in [152]. Altogether, they can provide a common view of the strategy adopted.

Studies are ongoing for developing a single aggregate indicator to quantify the sustainability (e.g. the definition of the "Ecological Footprint" of a specific technology [155]). However, a common agreement for the energy sector has not been reached yet.

In fact, a single set of indicators suitable for all the sources of energy (and the associated energy production chain) does not exist. The choice of the indicators depends on the target of the study and on what would be "measured" or highlighted.

In general, they are selected in order to underline advantages and drawbacks of the technology. Therefore, indicators for the nuclear sector are different from indicators for the renewable energy sectors [152, 13].

The most critical point is the "quantification" process, in particular for the indicators referring to environmental and social sectors.

In Table 3.1, an example of the indicators proposed for the energy sector is provided [156, 13, 152]. The selected indicators, subdivided in agreement with the three "pillars" of sustainable development, are quantified taking into account the entire energy production chain (from resources extraction to final services) [13].

In the following parts, only the most significant indicators have been described. More details could be found, e.g. in [156, 13, 36, 152].

Economical Indicators

As indicated in Table 3.1, for the economic dimension several indicators can be defined.

The more intuitive indicator is the *"the total cost of generation"* expressed as cost per kWhe generated. However, the quantification of this indicator is not so easy. In fact, the market competition leads to unavailability of fully disclosed data for the analysis. In addition, the projection of cost for plants to be commissioned in the future makes the study more complex.

An overview of the levelised lifetime cost approach for several energy chains is shown in Table 3.2. The data have been assessed considering average data (40 years of economic lifetime, 85% average load factor, 5 and 10% discount rates) [37].

According to Table 3.2, nuclear energy production remains comparable and competitive with coal costs and well below wind and gas costs.

More refined studies are ongoing for assessing the cost of kWhe generated, in particular analyzing the cost associated to the Capture Store and Sequestration (CSS) technology for coal plants (for limiting the emissions of CO_2) or to storage electricity plants developed together with renewable sources (e.g. [157]).

According to [13], other economical indicators could be considered:

- the fuel price sensitivity;
- the availability factor;
- the security of fuel supply;
- the lifetime of fuel resources;
- the use of energy resources;
- the use of non-energetic resources.

At 5% discount rate					At 10% discount rate			
Coal	Gas	Nuclear	Wind		Coal	Gas	Nuclear	Wind
22/48	39/56	23/36	35/90		28/59	43/59	31/53	45/125

Table 3.2: Ranges of electricity generation costs [Euro/MWh] considering 5% and 10% discount rates [37]

Environmental Indicators

The "environmental indicators" have been fully analyzed taking into account the whole energy production chain by the adoption of a LCA approach.

One of the most important indicators for the global environmental impact is the *"greenhouse gas (GHG) emissions"* expressed in terms of kg of CO_2 equivalent, where the warming potential of each gas is taken into account and weighted with respect to the CO_2 [13].

The emissions associated to the nuclear energy production chain are several orders of magnitude lower than the emissions from fossil fuel chains, as indicated in Figure 3.2.

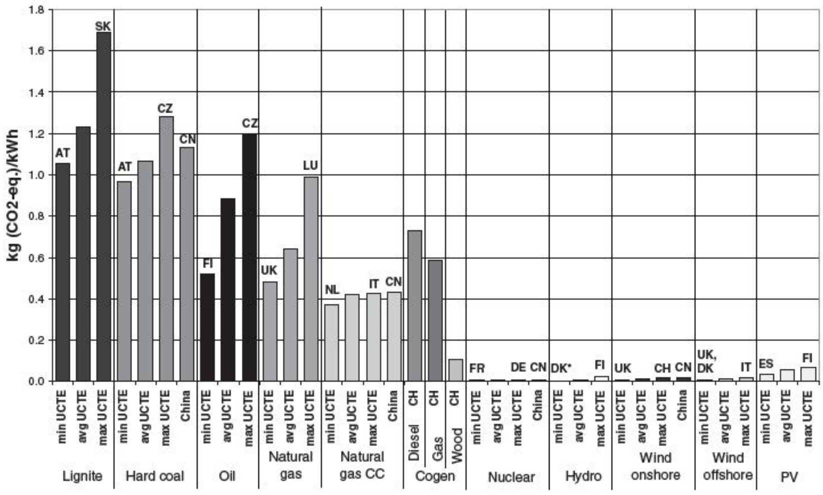

Figure 3.2: GHG emissions of selected energy chains [13]

However, the nuclear chain is the larger producer of radioactive waste as indicated in Figure 3.3, where the volume occupied by the radioactive wastes produced by each source is compared[6].

This aspect is a main central concern for the social acceptability of the nuclear energy chain[51].

The total radioactive inventory can be subdivided in sub-categories according the IAEA classification [158]: High Level Waste (HLW), Intermediate Level Waste (ILW) and Low Level Waste (LLW).

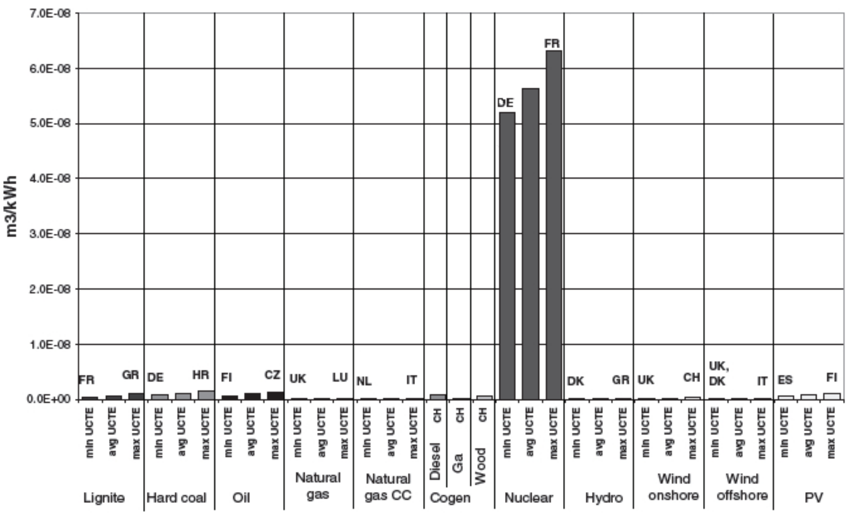

Figure 3.3: Production of radioactive waste for selected energy chains (comparison of the total volumes produced) [13]

[6] Instead of volume, other most suitable units of measure can be considered, e.g. radiotoxicity.

Social Indicators

The most critical dimension to be quantified is the social sector. In fact, a common agreement about the aspects to be included in the analysis has not been reached yet. For this purpose, a suitable European Programme, the NEEDS projects [159], has been launched.

Some criteria have been fixed "a priori" for helping in the selection of the indicators: e.g. the political stability, the social acceptable development, the impacts on settlements and landscape quality, the social component of the economic impacts and the social component of risks including physical security [13, 159]. In addition, a balance between objectivity and perception has to be taken into account.

The social indicators selected and described in [13] are then:

- Employment;
- Human health impacts from normal operation;
- Waste confinement;
- Proliferation risk;
- Risk aversion.

The *waste confinement* is expressed in terms of necessary confinement time. There is not a common definition that can be applied to all the energy chains due to the different nature of the wastes (e.g. radioactive or stable toxic wastes).

In particular, for nuclear energy, the decay time of the HLW (mainly TRUs) is extremely long, needing about $2x10^5$ year before the natural level (mine level used as term of comparison) is reached.

Other indicators are the risk of proliferation and the prevention of severe accidents. These parameters have not been studied explicitly because they are at the basis of the development of advanced nuclear power plants and fuel cycles, in agreement with the Gen-IV goals [52].

Indicators chosen for the analysis

In order to make a comparison of the nuclear scenarios proposed during the work, several indicators have been selected starting from the general energy sector indicators summarized in Par. 3.2.

The selected ones are:

- the analysis of the resources involved. Natural uranium for LWRs start-up and Pu for the transition to FRs (belonging to the set of Economical indicators),
- the analysis of the long-term radiotoxicity evolution of the material sent to the final geological repository (belonging to the set of Environmental indicators),
- the analysis of the long-term decay heat evolution of the material sent to the final geological repository (belonging to the set of Environmental and Social indicators),
- the inventory of the spent fuel and of the isotopic composition of the material sent to the final geological repository (belonging to the set of Social indicators).

In addition, the assessment of the capacities for each fuel cycle facility (mining, enrichment, fabrication and reprocessing plants) are analyzed. They, indeed, can be limiting factors for the development of nuclear energy both at country and world levels as indicated also by [33, 84].

A similar set of indicators has been adopted by NEA/OECD for the comparison of the advanced fuel cycles as described in [1]. Others studies have chosen comparable set of indicators, e.g. [160]).

The assessment of the engaged natural resources is the first parameter to be considered in order to address the problem of security of supply. This aspect, indeed, can impact, from the point of view of the sustainability, both the economic sector (as indicated by the cost of kWhe or by the fuel doubling cost impact) and the social sector (by the possible increase of the conflict for the resources holding). The last argument has been pointed out also by [161].

The other three parameters selected for the study are oriented to quantify the impacts of the advanced fuel cycles and innovative reactor concepts on the fuel cycle back-end.

In particular, the analysis of the waste inventory is important for planning the repository size. This aspect becomes critical for countries, e.g. U.S. or Korea (see Par. 2.5), that have some limitations in storage facilities and/or have found several problems for the repository construction (as the Yucca Mountain experience has indicated).

The reduction of the total inventory can be achieved by implementing advanced fuel cycles based on P&T. The systems considered in those cycles are developed, mainly, within the Gen-IV initiative. Hence, other indicators (as accident mitigation and proliferation resistance) are implicitly taken into account.

The heat load associated to the material sent to the geological repository influences the repository size itself, and, therefore, a reduction of it can help in optimizing the disposal.

In addition, the radiotoxicity is a parameter to evaluate the potential danger associated to the material disposed. In fact, the radiotoxicity is a measure of the radiobiological hazard of radioactive waste. It is related to the intake of radionuclides by the human body along with inhaled air (inhalation radiotoxicity), drinking water or food consumption (ingestion radiotoxicity) [162, 163].

Combining together the radiotoxicity evolution of waste material (that can be affected by the strategy considered) with the radiotoxicity associated to the natural uranium ore, it is possible to determine the period in which the potential danger of material in disposal becomes comparable to the natural conditions (an example is presented in Par. 2.5.1) giving a quantification also of the waste confinement indicator.

Anyway, the adoption of the radiotoxicity as sustainability indicator is not yet completely accepted as indicated by [125, 134]. A brief note about the discussion ongoing is included in Par. 3.2.1.

Other indicators have not been explicitly selected, as the cost of kWhe, but they can be considered for further studies.

The analysis of the infrastructures for sustaining the cycle have been performed as shown in detail in Chapters 4 and 5.

3.2.1 Impact on Disposal and Radiotoxicity as Indicator

Several studies have been performed during the last years with the aim to analyze the impact of P&T and advanced fuel cycles on different repository media.

The radiotoxicity level of the material sent to the repository has been adopted as an index to demonstrate the Pu and MAs transmutation efficiency (same approach adopted in the present activities for comparing the different scenarios).

This approach implies that the radiotoxic material comes into contact with the human environment in which it can follow the food-chain or dispersed in air [14].

Minor Actinides and Pu are the major contributors to the long-term radiotoxicity, however they seem to be almost stable and stationary from the point of view of the release and the diffusion processes in almost all the hosting materials considered (e.g. in volcanic tuff, crystalline or argillanceous formations, bedded or domed salt or borehole) [14].

Therefore, the possibility for Pu and MAs to impact the human environment (without considering voluntary intrusion in the repository) is very limited, in normal conditions.

The major contribution comes from other isotopes, e.g. I-129 and Zr-93. An example is represented in Figure 3.4 for the case of a thick clay layer [14, 164, 165].

From Figure 3.4 is clear that long-living fission products dominate the expected dose rate for nominal undisturbed performance for a repository in a thick saturated clay layer. In this media (but similar behaviors can be found for other media [164]) the actinide transport is hindered by the geologic environment [14].

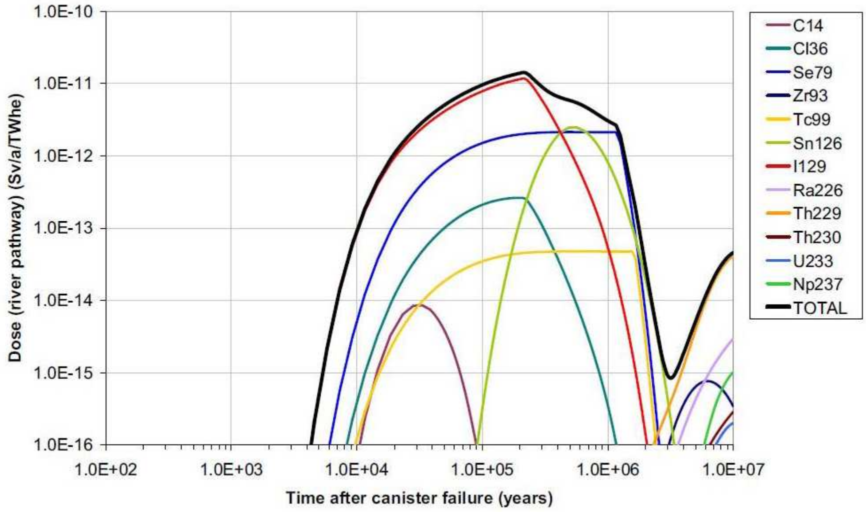

Figure 3.4: Dose rate for nominal undisturbed performance for a repository in a thick saturated clay layer [14]

All these considerations resulted in different opinions about the efficiency of P&T (and therefore of processes oriented to reduce the long-term waste radiotoxicity) from the point of view of the long-term impact in the repository.

As indicated by [125, 134], the radiotoxicity reduction and the implementation of P&T (e.g. by double strata ADS based cycles) is not the most important aspect to be pursued in order to guarantee the long-term isolation of the repository.

In addition, the implementation of advanced fuel cycles and MAs multi-recycling implies the increase of low and intermediate level waste flows as well as the adoption of fabrication and reprocessing plants with higher concerns related to the workers (due to strong α emitters).

A quite different opinion comes from [14], where the reduction of the radiotoxicity by the implementation of P&T is considered to bring advantages also for the disposal point of view helping in reducing of the "source term" associated to the disposal.

This aspect has been also pointed out within the OECD/NEA WPFC Task Force on Potential Benefits and Impacts of Advanced Fuel Cycles with Partitioning and Transmutation (P&T), (WPFC/TFPT) [166].

The approach adopted in this study is in agreement with [14, 166]. Therefore, radiotoxicity has been maintained as one of the indicators for the scenario comparisons.

3.3 Preliminary scoping study: results

As mentioned in Chapter 2, a scenario analysis could be oriented to different geographical scales: world, continental (e.g. European) or country scale.

The geographical dimension of the study implies different hypotheses (e.g. energy demands) that can impact the objectives of the study itself.

As seen in Par. 2.3, the more appropriate geographical scale for analyzing the possible resources shortage is the world scale, where general energy trends are applied at the expense of the heterogeneity.

However, the particular behavior of each country (planned on the basis of a country oriented cost-benefit analysis) should strongly change these general trends.

During the preliminary scoping study here summarized, the boundary conditions for country-oriented scenarios have been studied. In particular, the European situation has been considered in detail.

As just previously indicated, an averaged energy behavior has been applied to all the countries to compensate the lack of data. Each country has been studied separately according to the particular initial situation in order to show possible differences in the choice of the boundary conditions.

The scenarios proposed are only academic studies and, therefore, they do not aim at representing any country strategy. They, indeed, have been used for tuning the methodology and for identifying the most critical hypotheses to be further analyzed.

In order to maintain a certain level of heterogeneity, the energy policy adopted by each country has been used as discriminant aspect.

In order to reduce the heterogeneity (in agreement with the approach adopted in the PATEROS project [5]), the countries have been grouped according to the 2008 nuclear energy policies. For each group, some representative countries have been selected for further studies.

In particular four groups have been considered:

- Group I: it includes countries with an ongoing nuclear energy strategy (e.g. France) that are expected to maintain this position also in the coming years;

- Group II: it includes countries with a gradual phasing-out nuclear policy ratified by law (e.g. Belgium, Germany, Spain have respectively decided the phase-out in 2003 and 2002[7]). For these countries, according to the 2008 situation, some possibilities for changing their position about nuclear policies were considered;

- Group III: it includes countries with renaissance nuclear energy policy (e.g. Italy) where the decision to re-open (or open for the first time) to nuclear energy production was considered in 2008;

- Group IV: it includes countries that have decided to use not at all the nuclear energy production in their mix (e.g. Austria).

Different energy policies imply different needs for the country. For instance, countries with ongoing pro nuclear strategy need to replace their fleet and the guarantee of the long-term security of supply (e.g. uranium), improving safety of existing and new NPPs and finding a solution for waste management. On the contrary, countries in phasing-out strategy are focusing mainly on the optimization of the fuel back-end (e.g. waste management and repository).

During this preliminary approach only energy demand and substitution of the in-operation fleet have been considered. The choice mainly depends on the limitations of the NFCSS fuel cycle code adopted [16] and on the target of the study limited to 2050.

[7] For Germany the phasing-out strategy has been confirmed in 2011 as consequences of the Fukushima impact.

Whatever is the energy policy adopted, the first step performed is the analysis of the current situation represented by the number and the age of NPPs in-operation. Only three cases have been considered: France as representative of Group I, Belgium for Group II and Italy for Group III.

In France, 58 PWRs are in operation since 1977 [17]. In 2009, the NPPs have produced 75% of the total net electricity production of the country [38].

Assuming a reactor lifetime of 40 years, the replacement of in-operation plants is expected to start in 2018. The existing fleet can follow a shutdown behavior similar to the one reproduced in Figure 3.5 (red line). In the same figure, the behavior assuming 60 years reactor lifetime is indicated too (green line), as a possible alternative.

In Belgium, 7 PWRs are in operation since 1975 [17]. According to the 2003 Belgian law, starting from 2015 (year in which DOEL-1 reaches 40 years reactor lifetime) the systems will gradually shut down (see Figure 3.6 where plant extension life strategy has not been considered). These 7 NPPs in 2009 have provided ca. 52% of the electricity produced in Belgium [38].

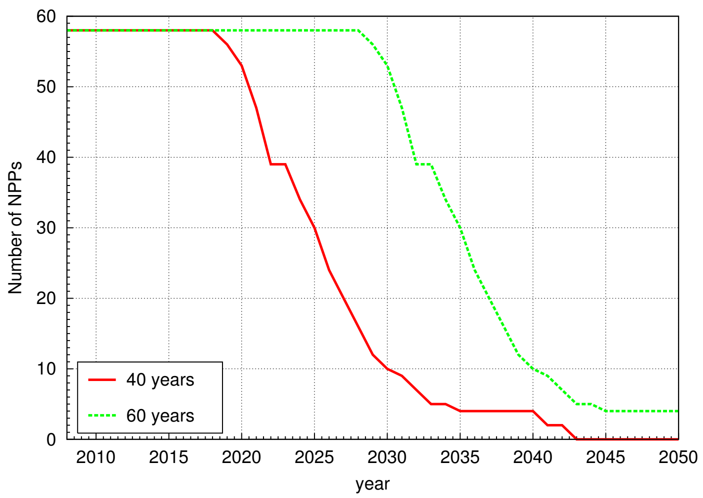

Figure 3.5: Actual France NPPs in operation, shutdown according to reactor lifetime (40 years or 60 years)

In Italy, actually no NPPs are in-operation according to the 1987 law. However, four reactors (1 gas-cooled system, 2 BWRs and 1 PWRs) were operated between 1964 and 1990 contributing to few percents of the total electricity production (max. 5% as indicated by [19]). In 2008, the government has declared the intention to restart nuclear energy production in the following decades. Therefore, the analysis of the associated fuel cycle was considered useful to technically support the decision-making process[8].

After the analysis of the initial conditions, the definition of the future energy envelops and of the share to be covered by nuclear energy is the second step to be performed.

For these preliminary studies, the energy demand considered is not country-specific. An average value, indeed, for Western Europe has been applied to all countries.

As described in Par. 2.3.2, for defining the future energy trends, only data available in literature have been considered.

[8]The moratorium of 2011 has stopped the nuclear renaissence in Italy, changing completely the boundary conditions for the country. Nevertheless, the study performed does not lose its originality because, as explained better in Chapter 4, it can easily be applied to any other country interested in nuclear energy deployment, e.g. Poland.

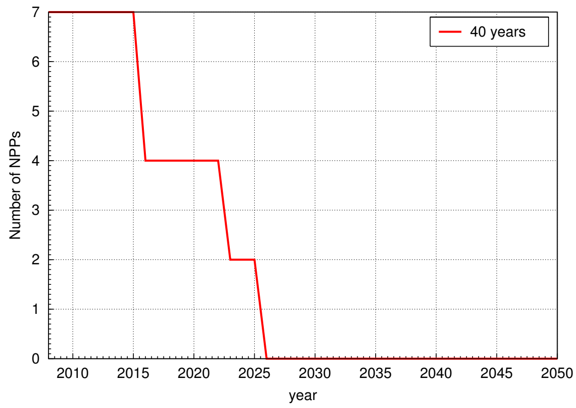

Figure 3.6: Actual Belgian NPPs in operation, shutdown according to reactor lifetime (40 years) and 2003 law

For the preliminary scoping study, a limited period (2008-2050) has been taken into account in order to determine the critical parameters affecting the transition scenarios from Gen-II to Gen-III fleet. This choice partially depends on the limitations of the fuel cycle code available (see codes comparison in Par. 3.4).

Therefore, only short and medium term energy projections have been considered [15, 89, 42]. In particular, the data are extracted from the 2008 IAEA publications [15], where nuclear and total electricity trends are provided. This solution avoids to add further hypotheses concerning the future nuclear energy share.

The data provided by IAEA are referred to a period up to 2030 [15]. Hence, an extrapolation to 2050 has been performed comparing these data with IEA up-to-date data (up to 2050) [25, 167].

For defining future trends the starting point is again the investigation of the historical behavior (regarding electricity and nuclear energy). An example for Western Europe (1980-2005 period) is indicated in Figure 3.7. This Figure shows how the energy produced by NPPs, after an initial increase (1980s) remains constant indicating that no new NPPs have been installed.

Therefore, an almost constant behavior has been taken as assumption also for the next years (taking into account both the age of reactor in-operation, ca. 20-25 years in average [17]).

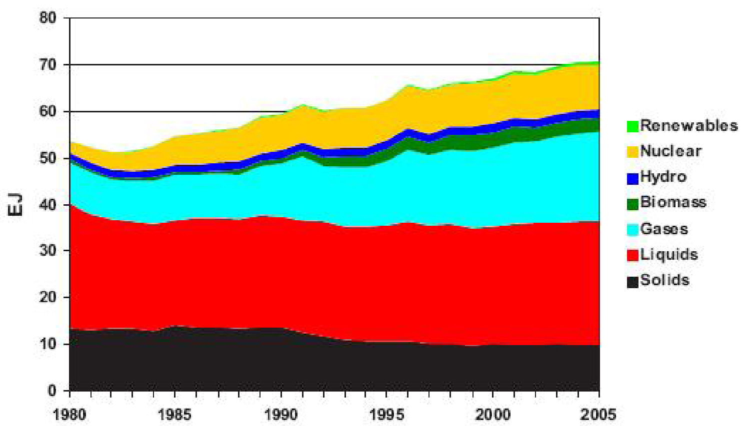

Figure 3.7: Historical data (1980-2005) for Western Europe concerning the share of primary energy sources adopted [15]

In order to have a range in which the future nuclear energy development can be inserted, two energy projections trends have been proposed by IAEA. The rates for Western Europe for the period 2007-2050 are summarized in Table 3.3. These trends are called respectively LOW and HIGH scenarios and applied as average trends for the study.

Parameters	1997-2007	
Population	0.4%	
Primary Energy	1.0%	
Electric Energy	1.8%	
Nuclear Energy	-0.1%	
Nuclear Capacity	-0.1%	
	LOW	HIGH
Parameters	2007-2050	
Population	0.2%	
Primary Energy	0.2%	0.9%
Electric Energy	1.0%	2.8%
Nuclear Energy	-1.6%	1.5%
Nuclear Capacity	-2.2%	0.9%

Table 3.3: Annual Rates for Western Europe [15]

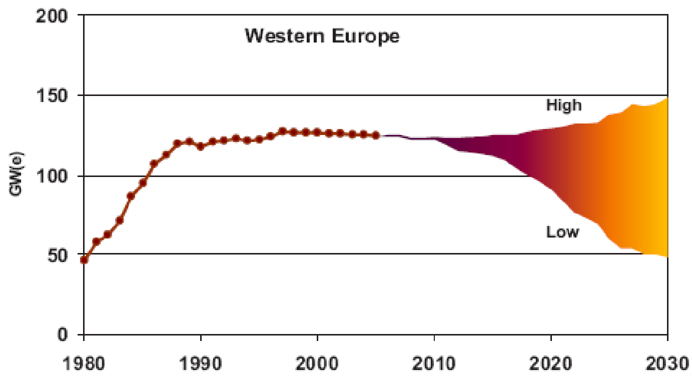

Figure 3.8: Western Europe: nuclear expansion capacity 2008-2030 [15]

From Figure 3.8, referring to the same data listed in Table 3.3, it is clear that the LOW case corresponds to the gradual NPPs shutdown and the HIGH case refers to a moderate increase of nuclear capacity.

After the definition of the energy boundary conditions, additional hypotheses about the way to cover the nuclear capacity trends have to be added in order to describe completely the scenarios.

For the preliminary approach, two types of systems have been considered: 1) EPR-like system representative of the Gen-III+ reactor and used for replacing the in-operation NPPs, and 2) lead cooled fast reactor representative of Gen-IV systems installed from 2040.

The EPR [124] has been considered in agreement with other studies (see Par. 2.5), the suitable system able to replace the existing reactors.

Among the Gen-IV facilities, the ELSY model [68], largely studied at the University of Pisa (DIMNP), has been selected.

Two examples are described in detail: the Belgium and the Italian case in the following parts. Other cases are indicated in [168, 169, 12].

The results obtained by the NFCSS code are summarized in Par. 3.3.1 and together with the analysis of the Italian case (an "edge case") are used for illustrating the limitations associated to the fuel cycle code adopted (see Par. 3.4).

The NFCSS fuel cycle code [16] is a free code developed by IAEA. The code is a web-based computer application which enables users to estimate long-term nuclear fuel cycle material and service requirements as well as material arisings [16].

Every system is characterized by few input data provided by the user. These data are the initial enrichment, the residence time of fuel in core, the load factor, the efficiency, the tail assay, the reprocessing characteristics and the average burn-up [16]. In addition, a single set of one group equivalent cross sections (capture, fission, n2n) averaged over burn-up is provided by the user. Few isotopes[9] are considered in order to evaluate the fuel composition versus burn-up. For the preliminary scoping study, the available libraries in NFCSS code have been used [16]. Respectively for EPR the PWR UOX1 library has been applied (standard PWR 4 loops with enrichment of about 4% U235) and for ELSY the LMFR library (Liquid-Metal Fast Reactor, with 15.4% Pu content) has been considered.

This approximation has been considered acceptable for this part and purpose of the activity. In fact, the major aim of the scoping study has been to define the approach to be followed and not the results themselves.

Concerning the model, the other data adopted are summarized in Table 3.4. The energy demand for each system is provided according to the transition strategy assumed.

Parameters	EPR	ELSY
Power (MWe)	1600	600
Load Factor (%)	85	80
Tails Assay (%)	0.3	0.3
Residence Time (years)	4	3
Enrichment (%)	4.5	15
Discharge burn-up (GWd/tHM)	50	67

Table 3.4: EPR and ELSY data adopted in the NFCSS simulation

The fixed structure of the NFCSS code is clearly indicated in Figure 3.9. From this figure, it appears that only Pu can be recycled and MAs sent always to disposal. Therefore, not all the fuel cycle schemes presented in Par. 2.1 can be modeled.

[9]Isotopes considered by NFCSS code: U235, U236, U238, Np237, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am242m, Am243, Cm242, Cm244

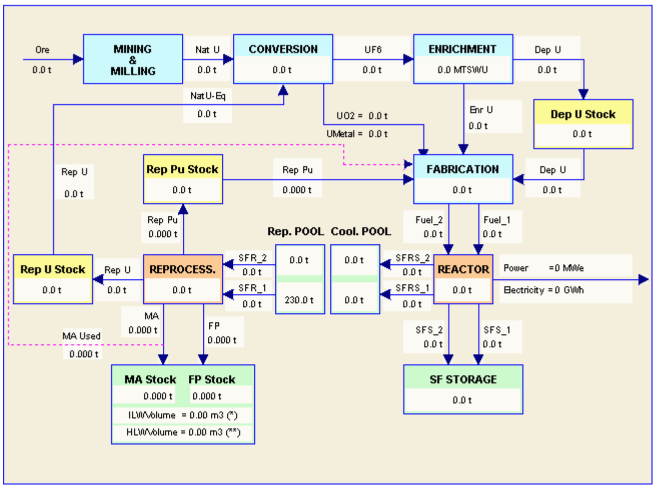

Figure 3.9: Fuel cycle scheme of the NFCSS code [16]

3.3.1 The Belgian case

In order to analyze the fuel cycle associated to the scenarios considered with the aim to point out the major sensible parameters, the case of Belgium has been analyzed.

The seven PWRs installed in Belgium are able to cover 52% of its electricity needs [38]. The capacity installed corresponds to ca. 35% of the global capacity installed in Belgium, representing the base load of the country.

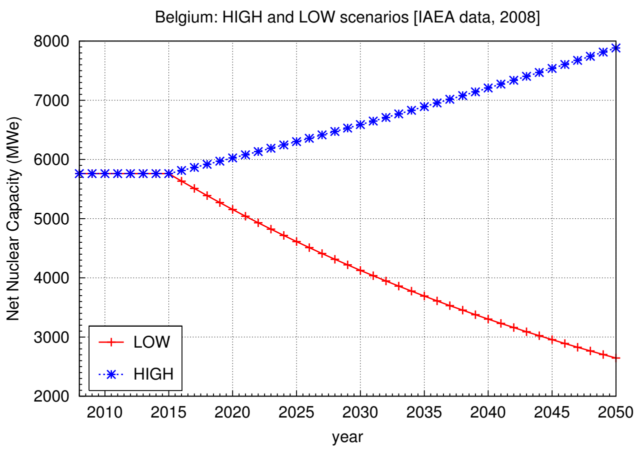

Figure 3.10: Scenario LOW (red line) and HIGH (blue line) for Belgium, period 2008-2050 [17, 15]

For the study, three cases have been simulated:

- Phasing-out of all the reactors at the end of their reactor lifetime (by 2025);
- Re-opening of the phasing-out decision, transition scenario made in order to cope with the energy trend defined by the LOW case (see Figure 3.10, [15, 170, 171]);

- Re-opening of the phasing-out decision, transition scenario made in order to cope with the energy trend defined by the HIGH case (see Figure 3.10, [15, 172, 173]);

Phasing-out

In Figure 3.11 is shown the nuclear capacity under the phasing-out strategy fixed by the 2003 Belgian law [18, 174]. The gradual phasing-out of the 7 PWRs will start with DOEL-1 in 2015 and it will end with the final shut-down of DOEL-4 and TIHANGE-3 in 2025.

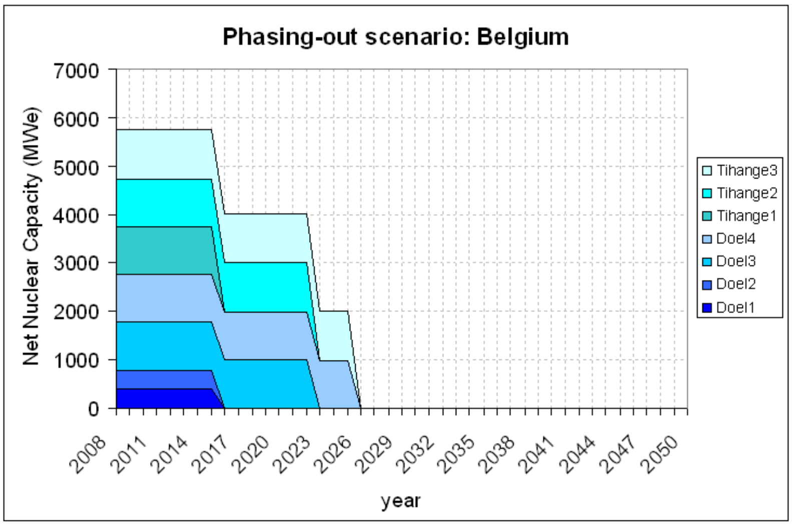

Figure 3.11: Belgium PHASING-OUT scenario: closure of the existing reactors (40 years reactor lifetime) [17, 18]

This scenario for phasing-out strategy has been simulated by means of the NFCSS code [16] in order to quantify the inventory of waste produced and the resources needed for period 2008-2025. Every reactor has been singularly simulated according to the data available in the IAEA PRIS database [17]. The data adopted for each reactor are summarized in Table 3.5.

Site	DOEL				TIHANGE		
Unit	1	2	3	4	1	2	3
Power (MWe)	392.5	392.5	1006	985	962	1008	1015
Efficiency (%)	33	33	33	33	33	33	33
Load Factor (%)	80	80	80	80	80	80	80
Tails Assay (%)	0.3	0.3	0.3	0.3	0.3	0.3	0.3
Residence Time (ys)	4	4	4	4	4	4	4
Enrichment (%)	4	4	4	3.85	4	3.8	3.8
burn-up (GWd/tHM)	45	45	50	45	45	45	45

Table 3.5: Belgian reactors data adopted in the NFCSS simulation [17]

The annual electricity produced by the 7 NPPs installed in Belgium, calculated by NFCSS, is shown in Figure 3.12. The 2010 value, ca. 40 TWhe, is in agreement with the literature data [175, 89]. In particular, the NEA reports indicate a energy produced in 2010 equal to 44.9 TWhe, in which the contribution coming from two French plants operated in collaboration is taken into account.

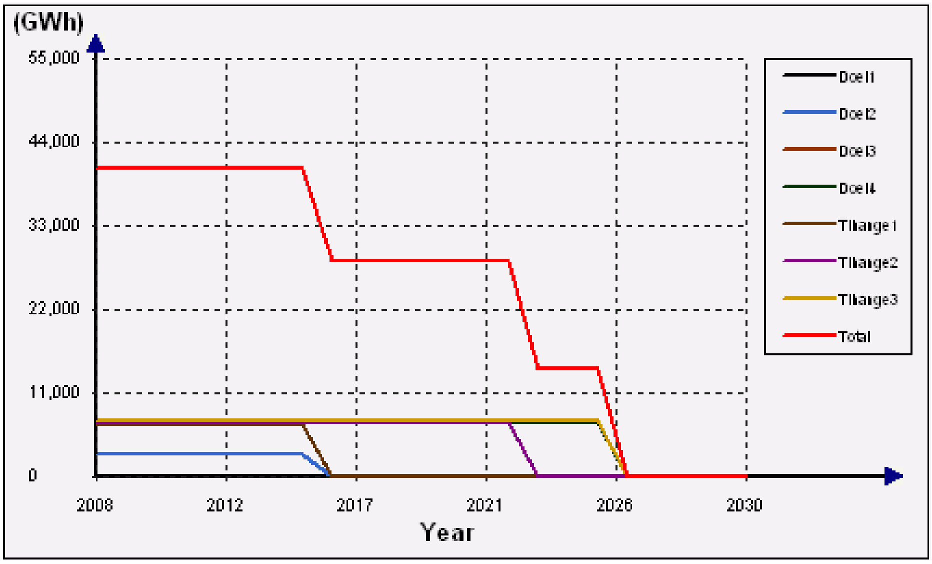

Figure 3.12: PHASING-OUT scenario: energy produced by the Belgian fleet [results by NFCSS-IAEA]

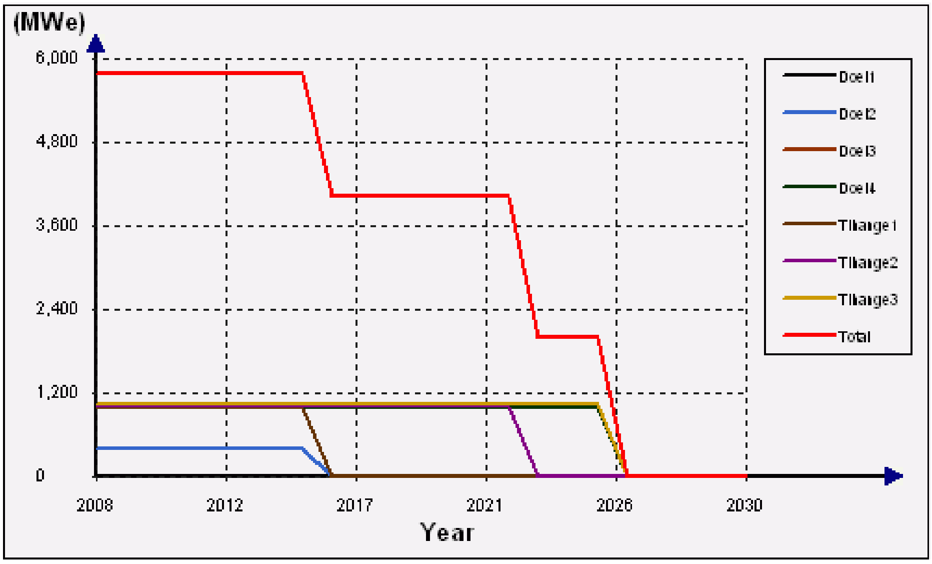

Figure 3.13: PHASING-OUT scenario: nuclear capacity installed in Belgium [results by NFCSS-IAEA]

Figure 3.13 shows the nuclear capacity installed under the phasing-out strategy as simulated in NFCSS code. As expected it is in agreement with strategy depicted in Figure 3.11.

The NFCSS simulations have been developed to tackle the mass flow between the facilities composing the cycle. An example is indicated in Figure 3.14. Figure 3.14, indeed, shows the annual SF discharged by the fleet (ca. 100 tonHM/year) where the peaks are referring to the discharge of shut-down core (respectively in 2015, 2022, 2025, as expected).

The cumulative SF inventory is shown in Figure 3.15. This value (ca. 1880 tonHM) represents only the cumulative value of the SF produced during the period 2008-2025 to which the inventory produced up to 2008 (ca. 2,480 tonsHM [175]) has to be added. In terms of SF isotopic composition, no data are provided by the code in the case of "once-through" fuel cycle. Therefore, the radiotoxicity and heat load (main indicators selected for the study) associated to the material sent to disposal has been assessed. This aspect is one of the limits of NFCSS.

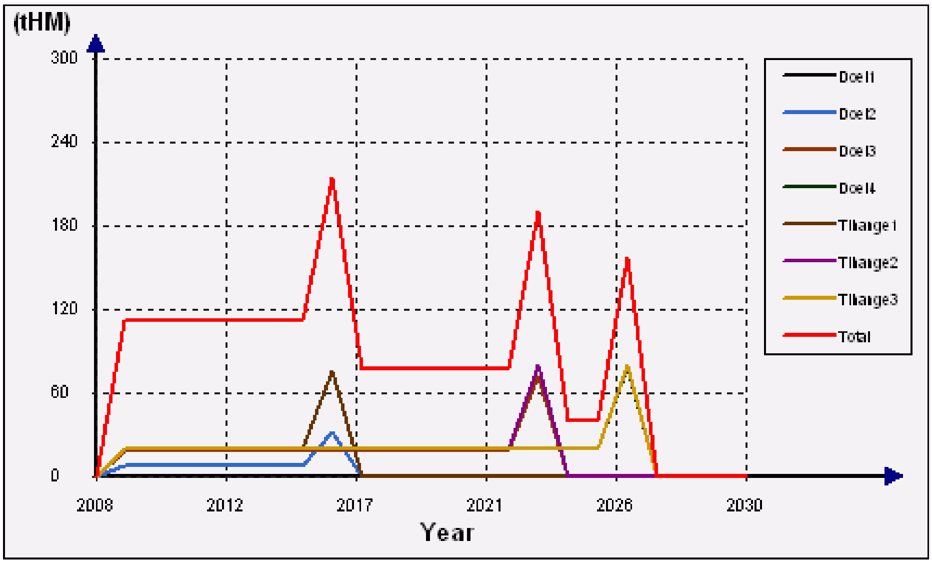

Figure 3.14: PHASING-OUT scenario: annual discharged Spent Fuel in Belgium [results by NFCSS-IAEA]

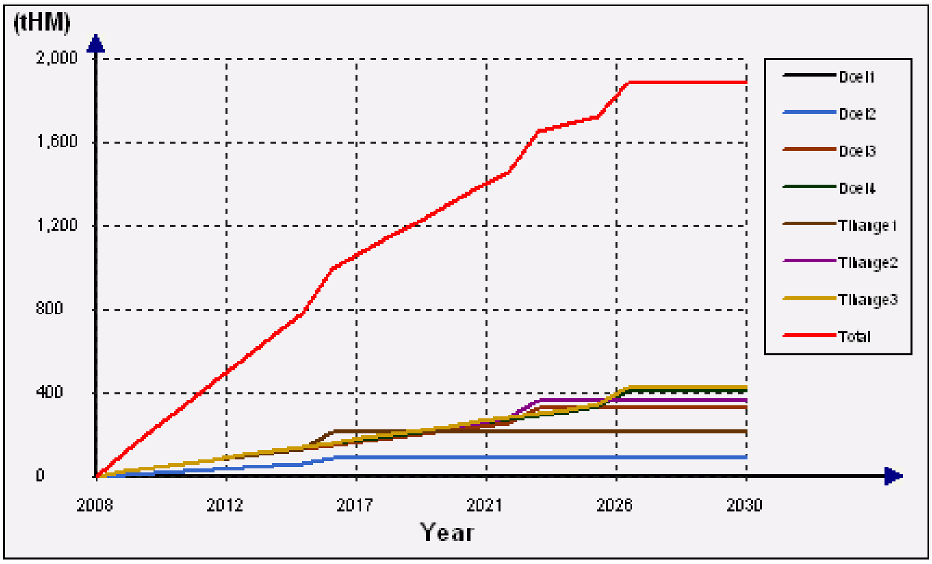

Figure 3.15: PHASING-OUT scenario: cumulative discharged Spent Fuel in Belgium [results by NFCSS-IAEA]

Scenario LOW

In order to consider alternatives to the phasing-out strategy, the LOW and HIGH trends proposed by IAEA [15] have been applied to the Belgian case.

In particular, for the LOW case, the electricity capacity installed increases by 1% per year to cope with the increased energy demand, and the nuclear capacity installed decreases by ca. 2.2% per year.

These trends have been applied starting from 2015, the year in which the first reactor (DOEL-1) is definitely shut-down (see Figure 3.10). The nuclear capacity in 2008 corresponds to 35% of the total electric capacity installed in Belgium, but considering the rates above described the share is substantially reduced to ca. 21% in 2030 and 11% in 2050. More details are included in Appendix E, Table E.5.

In order to substitute the in-operation NPPs (simulated accordingly to the data listed in Table 3.5) the introduction of EPR systems has been considered. The EPR systems have been simulated according to data listed in Table 3.4.

Assuming 40 years reactor lifetime, the gradual substitution of the 7 NPPs is performed introducing three EPRs (respectively in 2016, 2023 and 2026[10]). Considering this substitution strategy for the existing LWRs shown in Figure 3.16, the nuclear energy production in 2030 and 2050 becomes higher than the values calculated by the IAEA average trends (i.e. in 2050, the share remains equal to 20%). In fact, the substitution considered is step-wise and therefore, it can not completely match with the nuclear capacity trend evaluated with the IAEA rates (as shown in Figure 3.10).

The LOW Belgian case has been completely simulated with the NFCSS code. The nuclear capacity installed is shown in Figure 3.17. As expected, it is in good agreement with the strategy depicted in Figure 3.16. The energy produced by the systems is represented in Figure 3.18.

Under this assumption the cumulative SF produced in the period 2008-2050 is about 2,433 tonHM (ca. 30% higher than the phasing-out case).

[10]This data are in agreement with the phasing-out strategy adopted; i.e. NPPs closed in 2015, 2022 and 2025

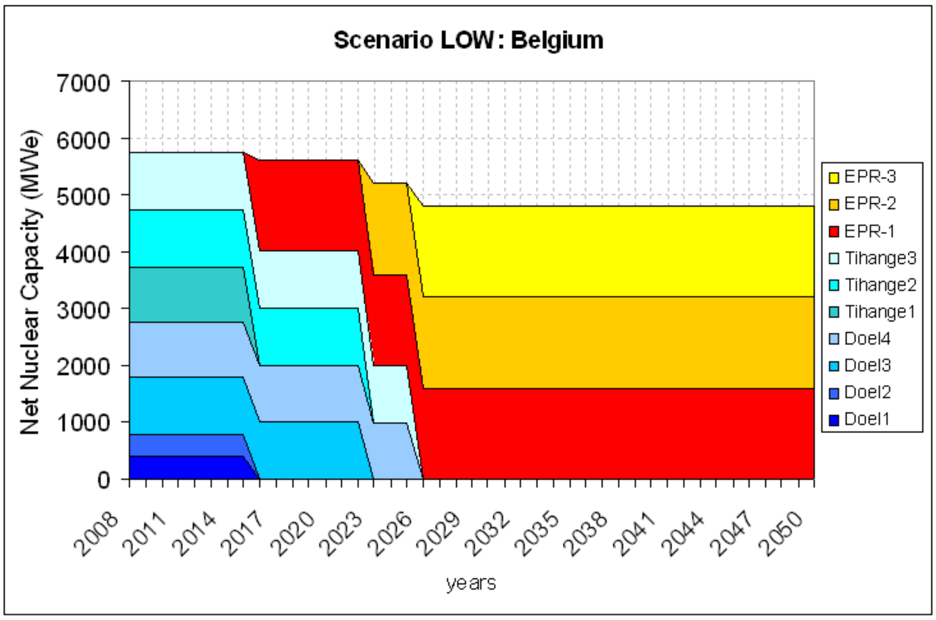

Figure 3.16: Belgium LOW scenario: substitution of the existing reactors with EPR systems

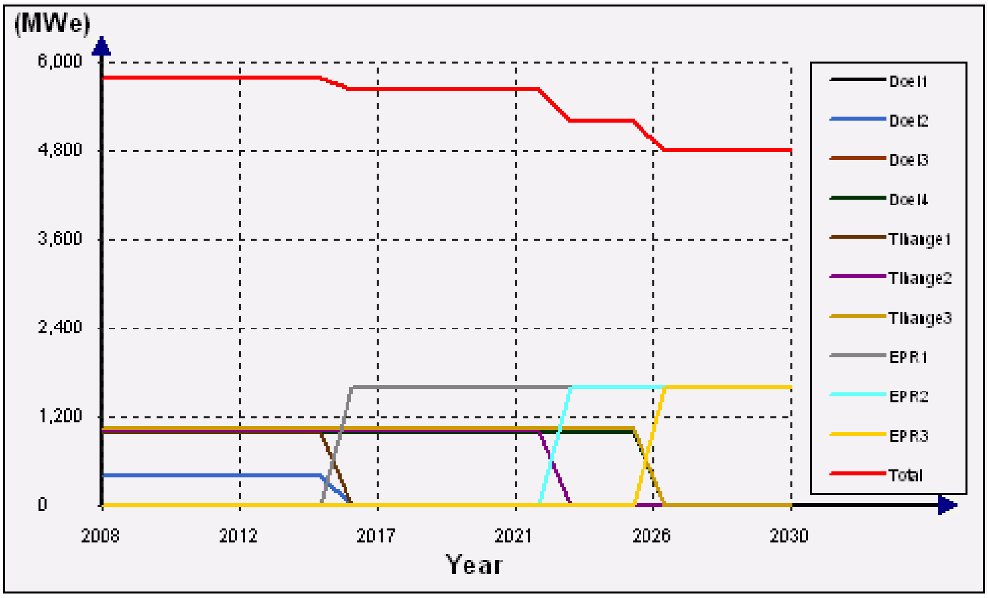

Figure 3.17: LOW scenario: nuclear capacity installed in Belgium [results by NFCSS-IAEA]

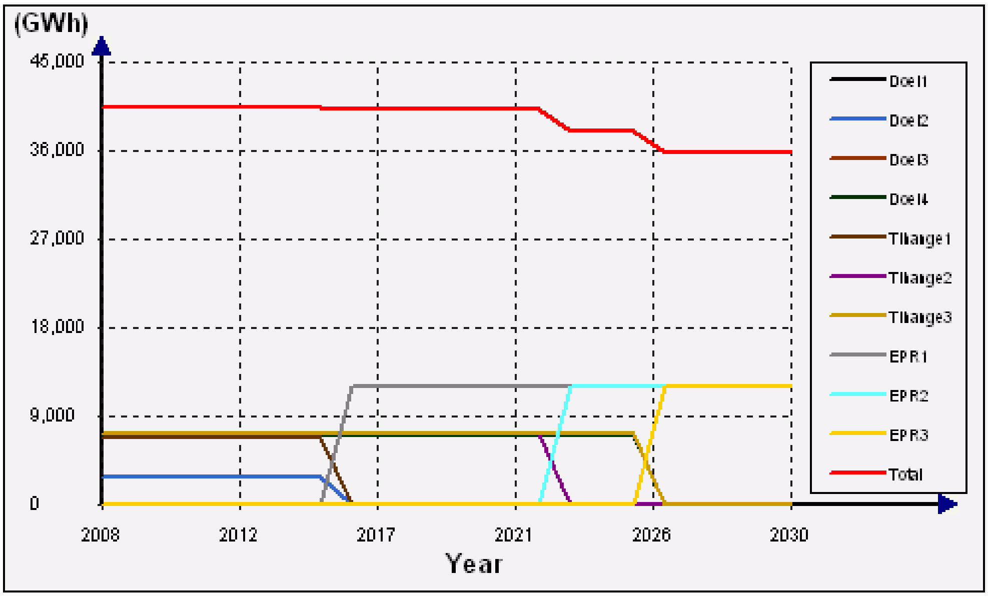

Figure 3.18: LOW scenario: nuclear energy produced in Belgium [results by NFCSS-IAEA]

Concerning the security of supply, Figure 3.19 shows the annual natural uranium demand. Three small peaks appear when the three EPRs are introduced as expected. However, a high peak appears in 2008. This peak could be explained on the basis of the simplified assumption for the scenario, where the historical behavior is not taken into account.

However, the peak in 2008 can not be eliminated except for modeling all the historical scenario. This aspect is again a drawback associated to the use of NFCSS code.

Moreover, the input of the NFCSS code does not allow to perform a more detailed analysis with a sustainable timeframe and effort.

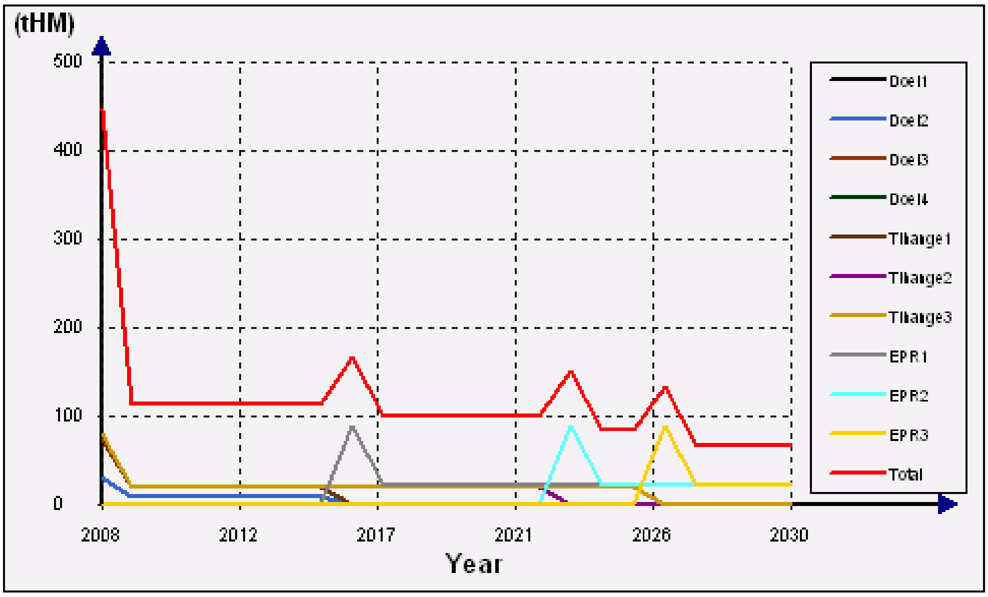

Figure 3.19: LOW scenario: nuclear fresh fuel annual request [results by NFCSS-IAEA]

The cumulative natural uranium needed for sustaining the scenario in the period 2008-2050 is about 21,965 tonU (ca. 62% more than the uranium needed for the phasing out case, 13,570 ton).

In addition, as indicated in Figure 3.20, the structure of the fuel cycle considered by the code is fixed. Standardized facility paths for the front-end and the back-end are considered. This aspect is a further limitations of the code, some fuel cycles can not be model and/or the simulation route is less flexible than other codes (e.g. COSI [20]).

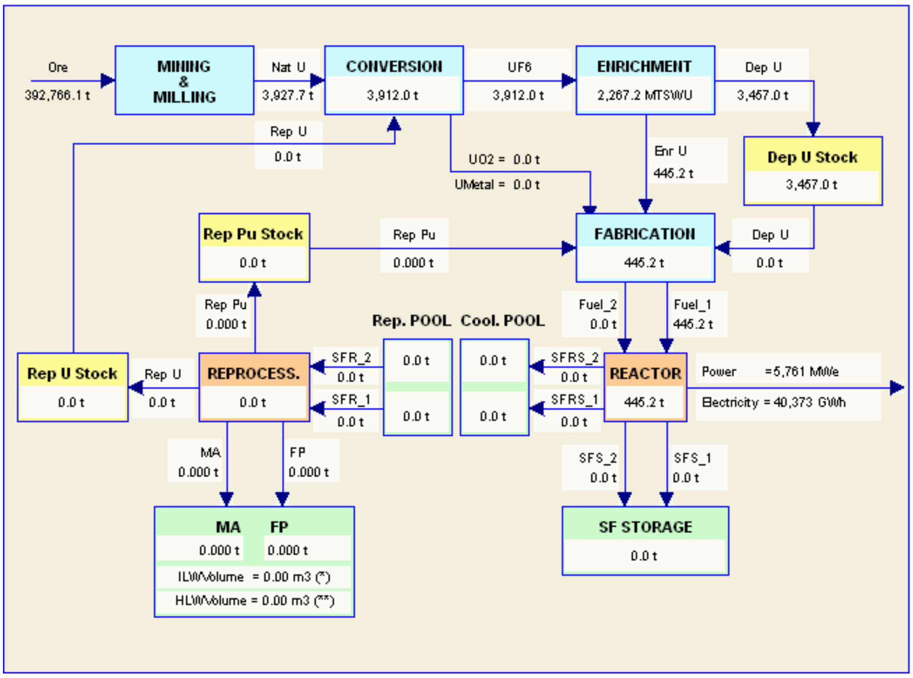

Figure 3.20: LOW scenario: fuel cycle scheme and 2008 mass flows [results by NFCSS-IAEA]

Scenario HIGH

A similar study can be repeated for the HIGH scenario assuming the following annual rates: 1) increase of the electricity capacity installed by about 2.8% per year, and 2) increase of the nuclear capacity installed by about 0.9% per year starting from 2015 (see Figure 3.10) in agreement with the IAEA data [15].

In order to follow the HIGH nuclear capacity trend (see Figure 3.10 and Appendix E for details), four EPRs have been introduced: the first EPR in 2016, two EPRs in 2023 and the last in 2026. However, they are not enough in order to cover the increase energy demand. Hence, the introduction, from 2036, of two ELSY systems, of 600 MWe each, has been considered. The systems have been introduced in a quite early date[11] in order to cope with the HIGH trend defined by the IAEA data [15].

By this approach a real "transition scenario" has not be considered because FRs are introduced in addition to LWRs. However, the results obtained can provide some indications about the capability of the NFCSS code to model scenarios with more than one type of reactors.

The substitution of the in-operation reactors is shown in Figure 3.21. As for the case LOW the substitution considered is step-wise. The resulting installed capacity differs from the value calculated by the means of average IAEA trends [15], as indicated in Figure 3.22.

[11]The early introduction date fixed to 2036 seems from a technological point of view too early. However, for the purpose methodology, the evaluation of the approach has been put forward.

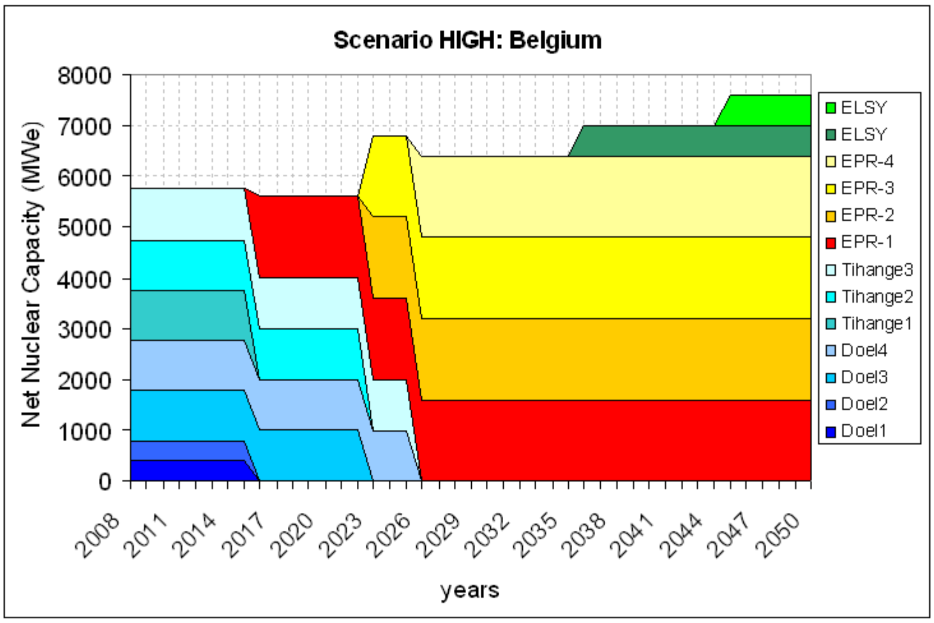

Figure 3.21: Belgium HIGH scenario: substitution of the existing reactors with EPR and ELSY systems

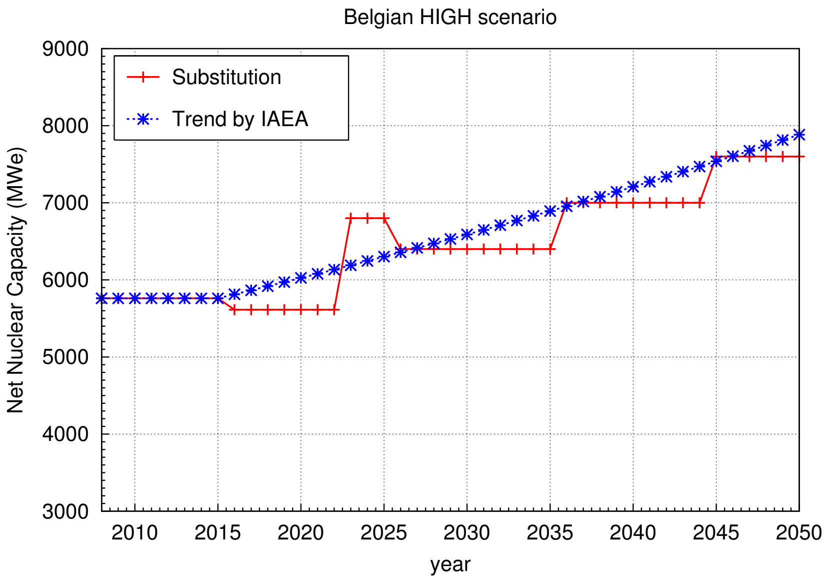

Figure 3.22: Belgium HIGH scenario: comparison between the trend evaluated by IAEA data and the substitution strategy considered

The nuclear installed capacity, as modeled in NFCSS code, is shown in Figure 3.23. It is in agreement with the substitution strategy considered (Figure 3.21) as expected. The energy produced, indeed, is indicated in Figure 3.24.

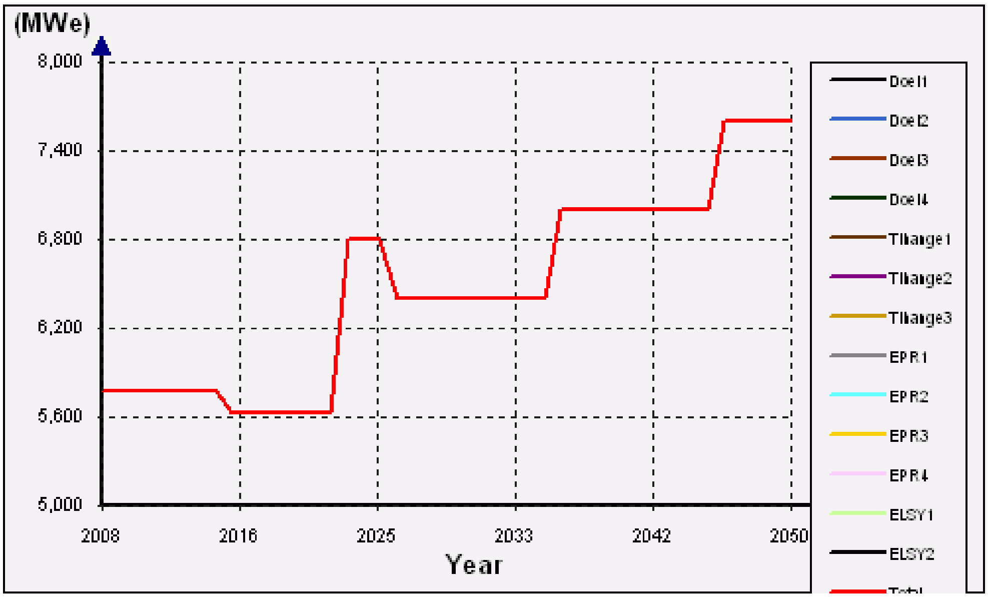

Figure 3.23: HIGH scenario: nuclear capacity installed in Belgium [results by NFCSS-IAEA]

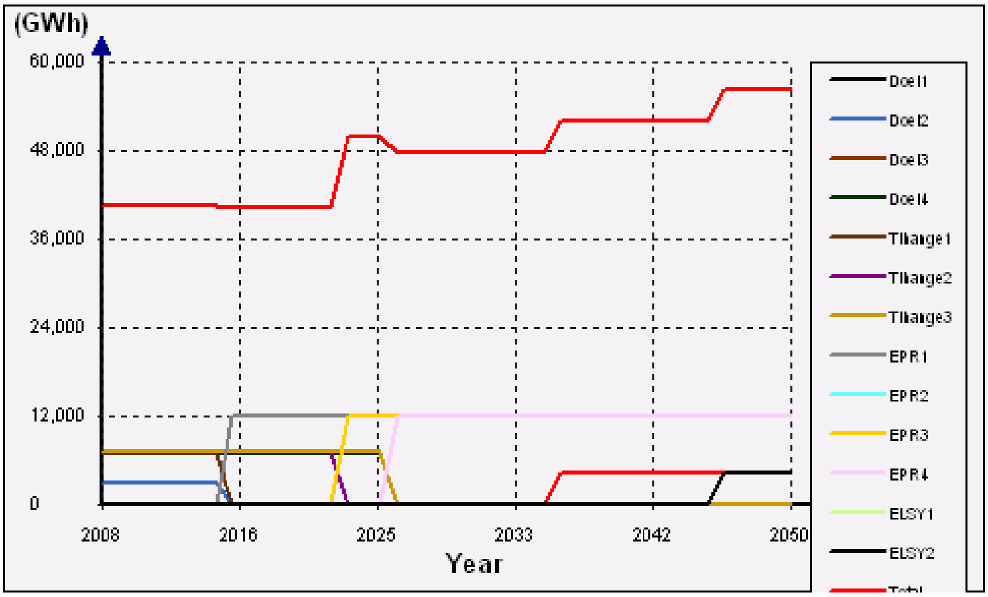

Figure 3.24: HIGH scenario: nuclear energy produced in Belgium [results by NFCSS-IAEA]

The cumulative uranium mass required is about 42,600 tons only for the period 2008-2050 (ca. 2 times the LOW case).

In Figure 3.25 the annual natural uranium requirements are indicated. The unrealistic peak at the beginning of the scenario (in 2008) appears also in this case, as pointed out for the LOW scenario.

In addition, two peaks (in 2036 and 2046) associated to the ELSY introduction appear. These peaks on uranium demand are unreasonable because the ELSY system has considered to be loaded with Pu coming from SF reprocessing and with depleted uranium and not natural uranium.

This aspect can be due by an unclear definition as Pu multi-recycling simulation by NFCSS code for fast reactors. In fact, the code is available only as a web-interface (a source files is not available) within limited information concerning the calculation schemes.

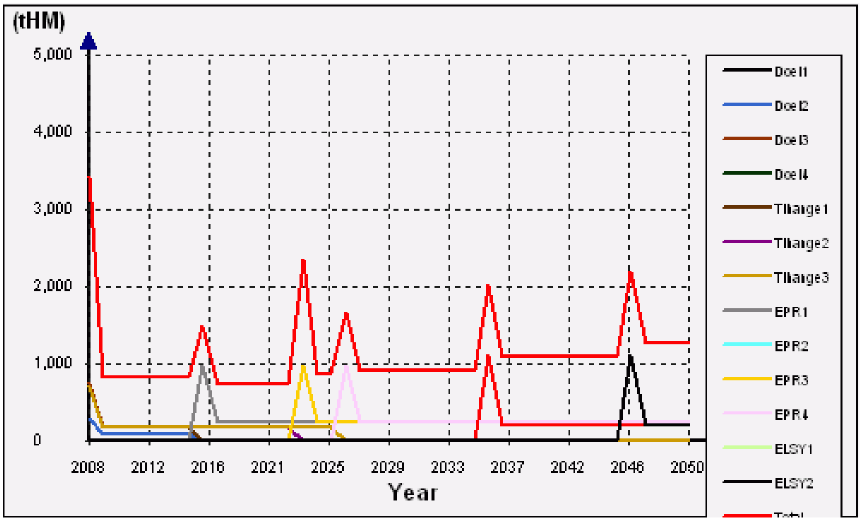

Figure 3.25: HIGH scenario: natural uranium requirements in Belgium [results by NFCSS-IAEA]

Can be assumed that the NFCSS code version available in 2008 seems to have some problems in treating the multi-recycling of Pu and MAs. This is one of the motivations for shifting toward a more flexible code.

3.3.2 The Italian case

In addition to the Belgian case, also hypothetical Italian scenarios have been analyzed.

Due to the limited number of systems involved (a prerequisite considered important when data have to be typed manually) the Italian case has been considered the "unit of measure" also for the preliminary scoping study performed.

With respect to the methodology described in Par. 3.1, the Italian case could be considered an "edge case".

In fact, Italy does presently not have nuclear power plants installed in the country in order to cover part of its energy demand. Therefore, it is impossible to treat transition scenarios in "continuity" with historical behavior.

In order to apply the methodology previously described, the only available information for setting the scenario boundary conditions considers a re-open of the nuclear option.

During this preliminary phase, several hypothetical scenarios have been compared. The introduction of EPRs and Gen-IV systems has been preliminary investigated.

Italian scenario: 1963-1990

Even in this case the starting point is the definition of the initial conditions.

The actual situation of Italy is characterized by no-nuclear energy production since 1990. Four NPPs were operated between 1964 and 1990.

In order to assess the Pu and MAs initial inventory (important parameters for the transition to FRs and for the fuel back-end management) the historical Italian scenario has been model by the NFCSS code. The data reported in the IAEA PRIS database have been considered as reference [17].

This study helps also in checking the capability of the code for simulating a "once-through" fuel cycle.

In total, the four NPPs have produced about 90 TWhe during their lifetime (see Table 3.6 for more details) operating from 1964 to 1990.

NPPs	Type	MWe net	Period	Lifetime gen. (GWhe)
Latina	GCR	153	1964-1987	25489
Garigliano	BWR	150	1964-1982	12246
Trino	PWR	260	1965-1990	24307
Caorso	BWR	860	1981-1990	27726

Table 3.6: Italian reactors lifetime energy generation (GWhe) [17]

According to the historical data provided by TERNA [19], the energy mix in the period 1963-1989 was dominated by fossil fuel. Nuclear energy contributed by less than 5% as indicated in Figure 3.26.

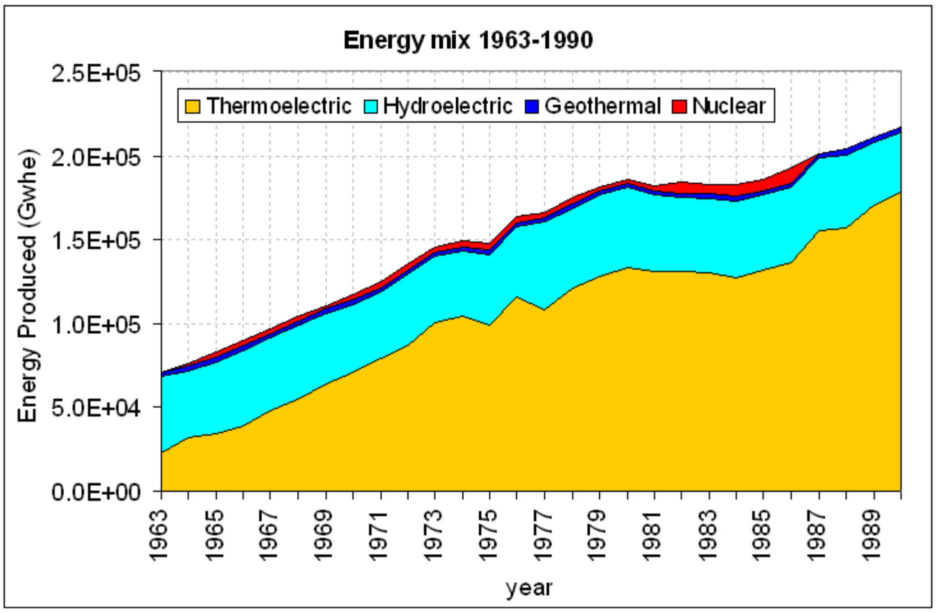

Figure 3.26: Italian energy mix in the period 1963-1990 [19]

In order to reproduce the historical scenario, the available data for each reactor (in terms of capacity, load factor, enrichment) have been adopted in the NFCSS model [17]. A summary of the data adopted for each reactor is reported in Appendix E.

In particular, the LATINA GCR reactor has been connected to the grid in 1963, the GARIGLIANO BWR reactor at the beginning of 1964, the ENRICO FERMI (TRINO) PWR reactor at the end of 1964, and the CAORSO BWR has been connected to the grid in 1978 and operated for only 10 years.

The in-operation period (1963-1990) has been carefully simulated in the NFCSS code. The variation in the annual load factor (see Figure 3.27) have been reproduced in the code. As underlined in Figure 3.27, the LATINA reactor has been operated in a quite constant manner, whereas TRINO has followed a more complicated history.

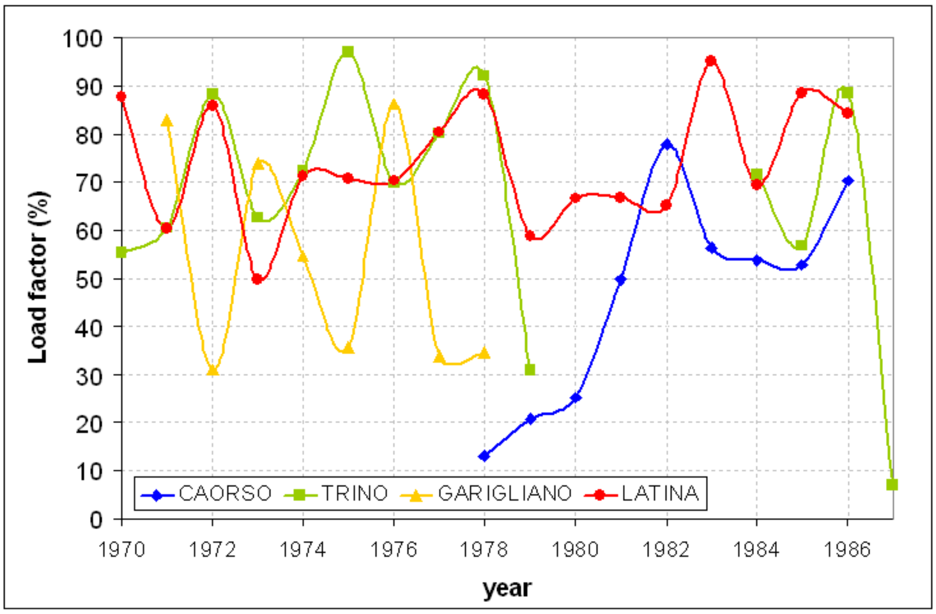

Figure 3.27: Load factor variation for the four NPPs in operation in Italy during the period 1963-1989 [17]

The load factor variation is well represented in the NFCSS simulation, as indicated by the energy produced shown in Figure 3.28. The irregular behavior is due to the long shut-down periods of the systems for maintenance or power enhancement. More detailed data are reported in Appendix E.

In Figure 3.29 is represented the nuclear capacity installed in Italy, the small variation due to the power enhancement can be noticed (e.g. CAORSO plant). The total electricity produced during the period considered as assessed by the NFCSS simulation is about 99 TWhe not far from the value indicated in the IAEA PRIS database [17]. The difference can be explained by the fact that in the IAEA database is considered the net of the electricity provided by the systems and in the simulation is considered the gross of the electricity provided.

In order to assess the SF inventory, some assumptions about the Italian reprocessing contracts have been made.

In particular, all the SF produced in LATINA (gas cooled thermal reactor) was assumed to have been completely reprocessed in Sellafield (UK) according to the information collected from the Italian Agency for nuclear power plants decommissioning, SOGIN [176]. Also for the SF discharged from TRINO and GARIGLIANO it has been assumed to be reprocessed in the AREVA plant at La Hague, France. Concerning the CAORSO plant, the assumption made has been that all the SF has been reprocessed except the last core [176].

Under these assumptions, the cumulative SF in Italy, not yet reprocessed, has been evaluated to be ca. 237 tonHM (see Figure 3.30), a value in close agreement with the data provided by SOGIN (235 tonHM that are the object of the 2007 contract with AREVA-La Hague plant [176]).

This quite good agreement has indicated that the NFCSS code is reasonably good for the simulation of "once-through" fuel cycle with proven technologies, as the comparison with the historical data has shown.

Future Italian Hypothetical Scenarios: 2008-2050

In order to define suitable boundary conditions, the IAEA electricity projections for Western Europe have been applied also to the Italian case [15].

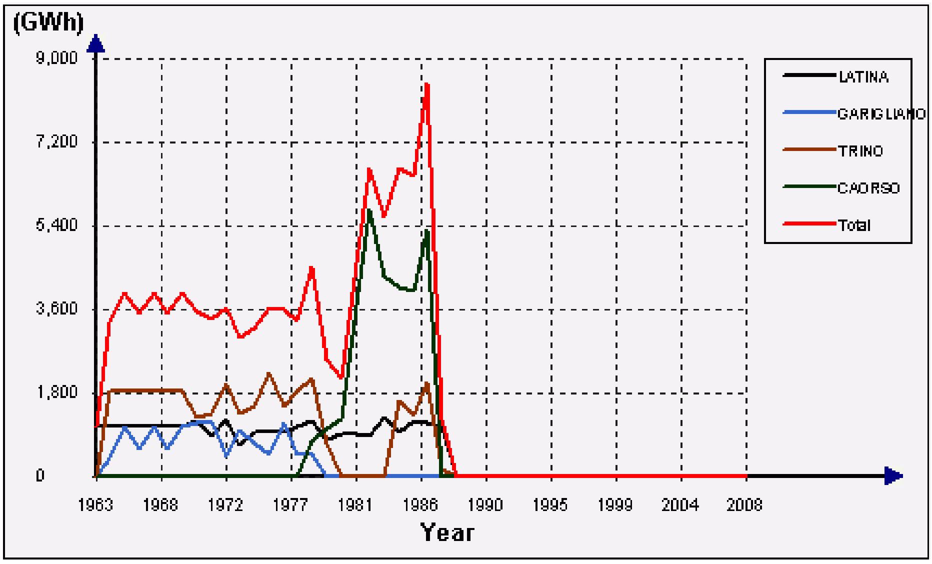

Figure 3.28: Historical Scenario: nuclear energy produced in Italy during the period 1963-1990 [results by NFCSS-IAEA]

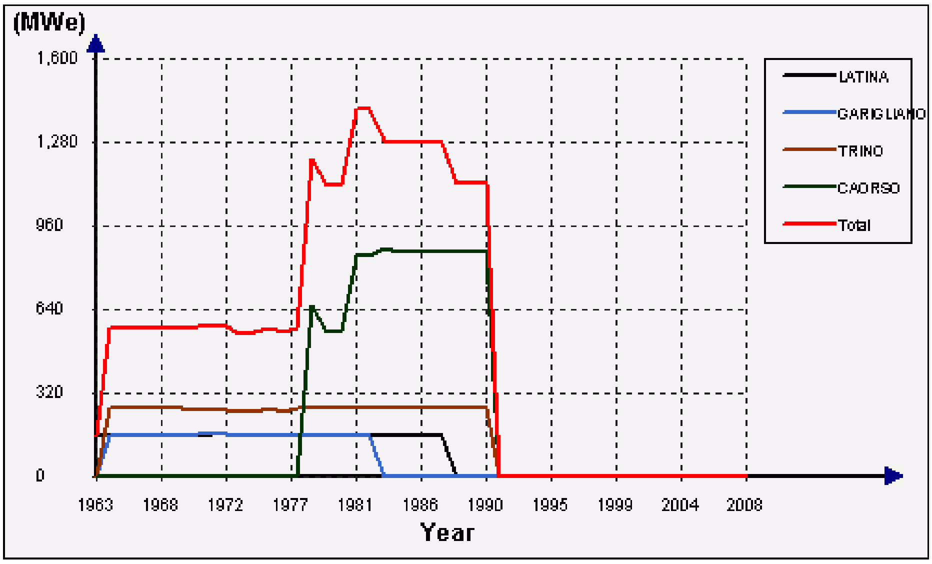

Figure 3.29: Historical Scenario: nuclear capacity installed in Italy during the period 1963-1990 [results by NFCSS-IAEA]

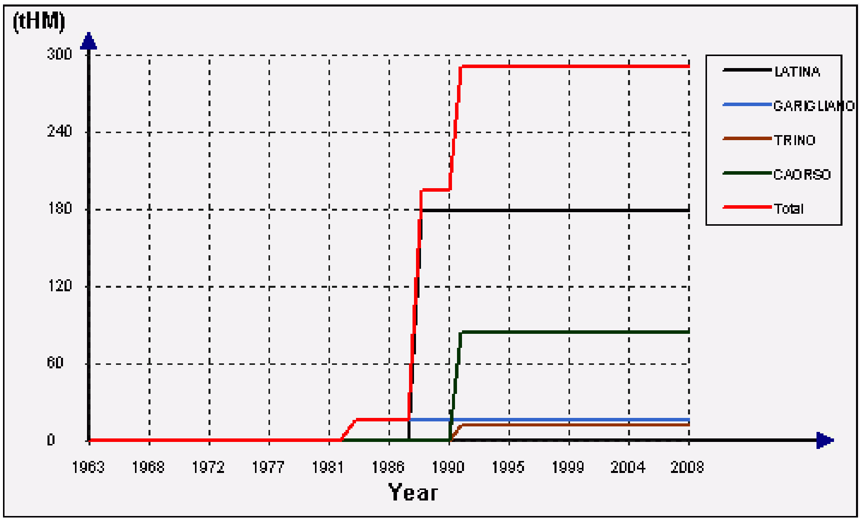

Figure 3.30: Historical Scenario: cumulative SF in Italy at 2008 coming from the period 1963-1990 [results by NFCSS-IAEA]

In addition to the LOW and HIGH rates for electricity production (Table 3.3) an average trend has been proposed (called AVEG case). Table 3.7 summarizes the electricity projections evaluated applying 1.0%, 1.9% and 2.8% annual increase rates, respectively, for the LOW, AVEG and HIGH cases.

	LOW	AVEG	HIGH
Year	TWhe		
2008	304.62		
2018	336.49	367.70	401.50
2023	353.65	460.95	403.99
2028	371.69	443.85	529.19
2038	410.58	535.77	697.50
2045	440.19	611.22	846.25
2050	462.65	671.53	971.55

Table 3.7: Italian projections of the electricity needs considering three cases: HIGH, AVEG, and LOW

No nuclear rates can be applied to this specific case because there is no continuity with the past situation. Therefore, some hypothetical scenarios, see Figure 3.31, have been proposed:

- Case ONE: Introduction of 4 EPRs in the near term (starting from 2018) loaded with UOX. MOX fuel is considered in order to partially burn the Pu produced;

- Case TWO: Introduction of 8 EPRs (4 in 2028 where recycling of MOX is considered and 4 in 2025 using only UOX fuel) and 3 LFRs starting from 2038;

- Case THREE: Introduction of 12 EPRs (in three successive slots of 4 units) and 7 LFRs starting from 2038.

The three cases proposed (shown in Figure 3.31) are compared with the increasing electricity demand for Italy assessed on the basis of the IAEA data [15]. The nuclear capacity share is indicated in Table 3.8.

Only the case THREE associated to the LOW electricity increase is able to enlarge the nuclear share from ca. 6% in 2008 to ca. 18% in 2050. The case ONE under the same electricity projections corresponds to a share equal to 5%. These shares obiously depend on the projections assumed as reference and they can be affected by the adoption of other sets of data.

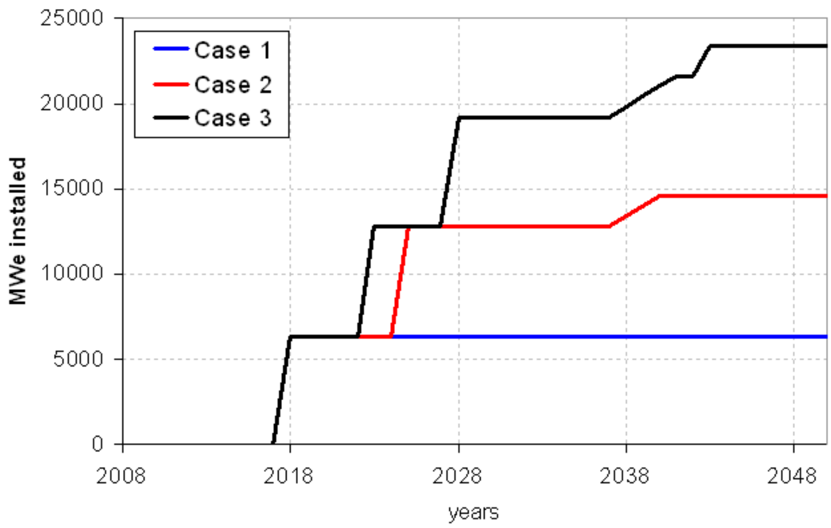

Figure 3.31: Scenarios proposed for Italy: cumulative nuclear capacity installed

2018			
	LOW	AVEG	HIGH
Case ONE	6.86%	6.28%	5.75%
Case TWO	6.86%	6.28%	5.75%
Case THREE	6.86%	6.28%	5.75%
2038			
Case ONE	5.62%	4.31%	3.31%
Case TWO	11.78%	9.02%	6.93%
Case THREE	17.40%	13.33%	10.24%
2050			
Case ONE	4.99%	3.44%	2.38%
Case TWO	11.39%	7.84%	5.42%
Case THREE	18.25%	12.57%	8.69%

Table 3.8: Nuclear capacity shares for the three scenario cases considered

The three cases have been implemented in the NFCSS code. Here only the case TWO is described in detail. More details could be found in [12, 169].

Altogether these cases (and the others shown for Belgium) have provided a basis for the refinement of the study as better indicated later on.

Case TWO

The case TWO considered here is composed of 8 EPRs introduced at two different dates: four in 2018 (UOX fuel but use of MOX allowed) and four in 2025 (only UOX fuel). From 2038, also 3 LFRs are introduced.

In Figure 3.32 is represented the annual electricity production. According to Table 3.8, it represents ca. 11.4% of total electricity needs in 2050 for the LOW energy envelop.

Just to provide examples of particular data and mass flows determined by the code, Figure 3.33 shows the annual natural uranium required for sustaining the cycle and Figure 3.34 shows the needs on enriched uranium.

The peaks around 2040 indicate that natural uranium is used to load FRs even though these systems are loaded with Pu (ca. 15% Pu content). The same problem has been underlined for the Belgian HIGH scenario case.

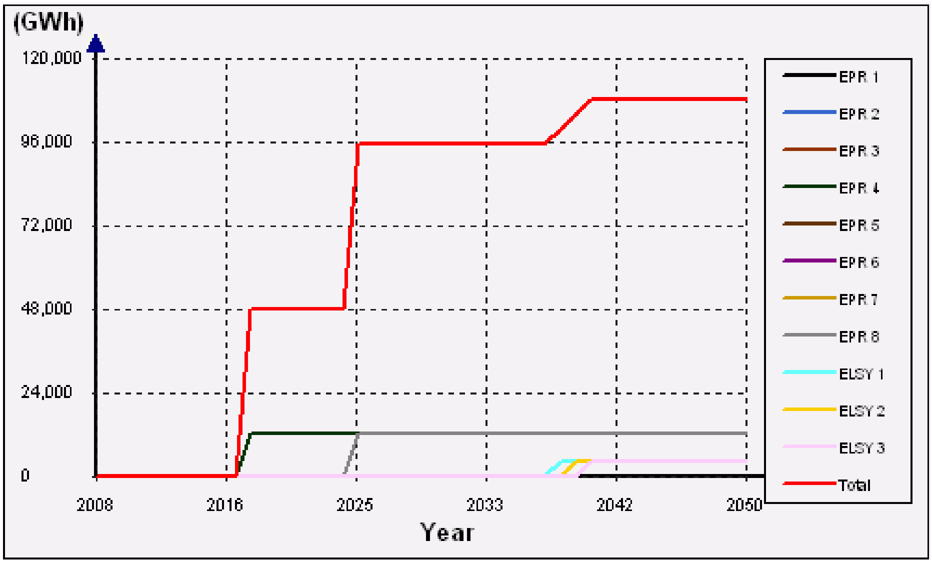

Figure 3.32: Electricity produced by nuclear energy by 8 EPRs and 3 LFRs in the period 2008-2050 [results by NFCSS-IAEA]

The cumulative SF is represented in Figure 3.35, where the minimum in the total behavior around 2038 underlines the SF reprocessing for the FRs introduction.

It has been decided to show only some typical results in order to exemplify the approach adopted. The results obtained are not significant for themselves. All these studies indeed have been used for refining the approach and for assessing the computational tools to be used.

The main focus has been directed to the choice of the hypotheses for setting the fuel cycle boundary conditions and of the indicators for comparing the results. Different approaches have to be followed depending on the objectives of the study and different parameters need to be assessed.

The outcome of the study is a general overview of the parameters that need to be treated in a scenario analysis. The check of the computational tools available at the University of Pisa is an additional outcome of the preliminary scoping study.

This preliminary study helped in gaining a certain level of confidence with this kind of investigations, forming the basis of the more detailed study proposed.

Several drawbacks related to the applicability of the NFCSS code to this Ph.D. activity have been pointed out as confirmed also by a complementary activity developed by [177].

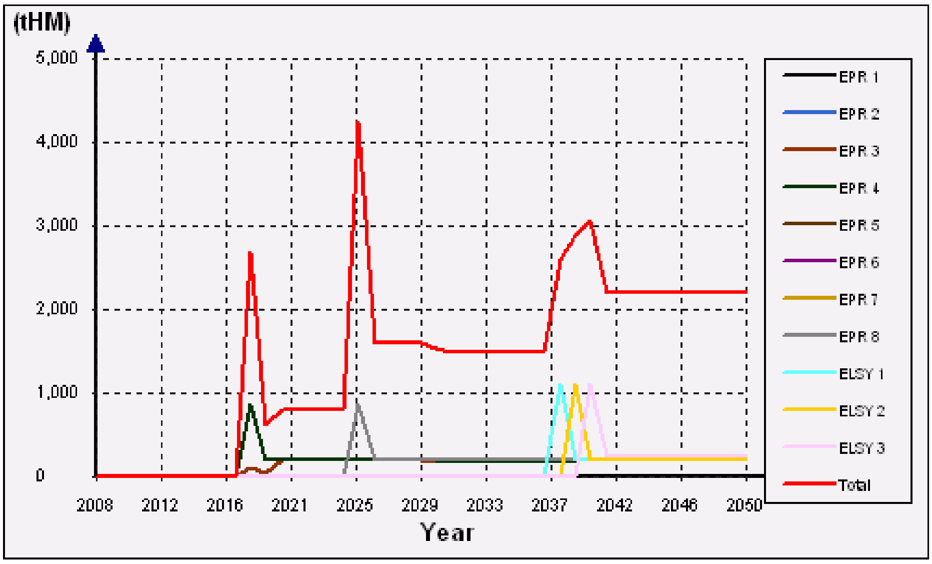

Figure 3.33: Natural uranium requirement for the case TWO. Period 2008-2050 [results by NFCSS-IAEA]

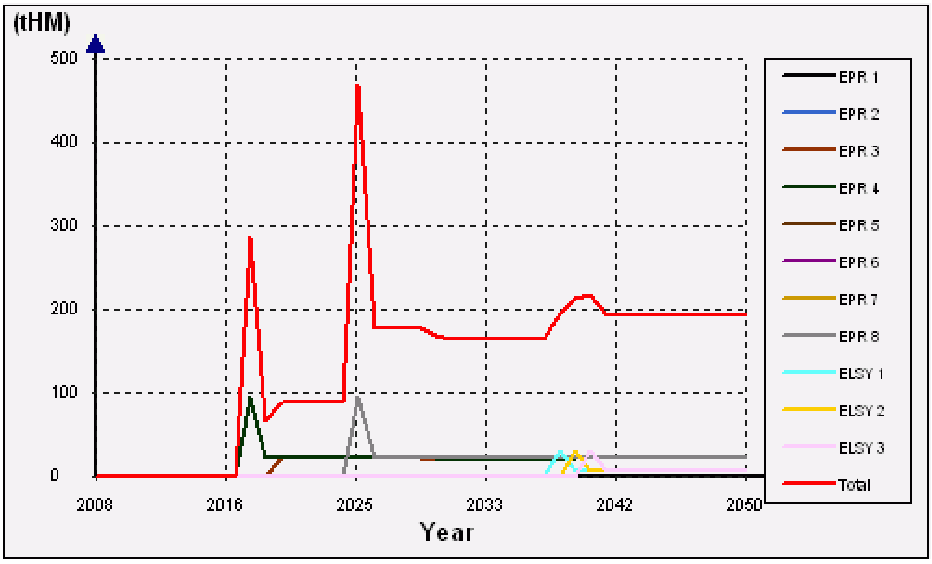

Figure 3.34: Enriched uranium requirement for the case TWO. Period 2008-2050 [results by NFCSS-IAEA]

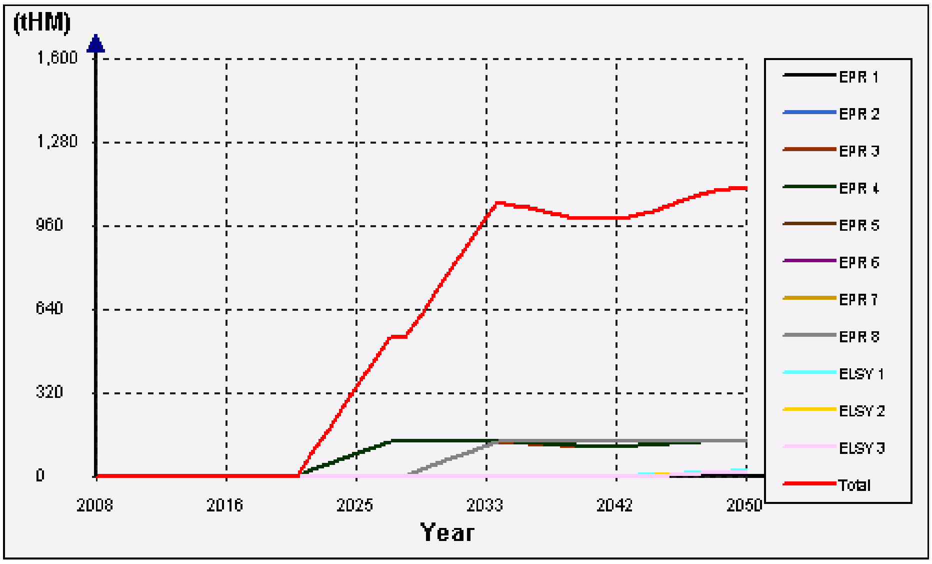

Figure 3.35: SF cumulative inventory for the case TWO. Period 2008-2050 [results by NFCSS-IAEA]

The description of the main drawbacks limiting the adoption of the NFCSS code is included in Par. 3.4.

However, it has been decided to show the preceding part of the activity because it is considered as important propaedeutics to the refined study performed during the final part of the Ph.D. activity.

3.4 Selection of the scenario code

The analyses presented before have clearly demonstrated the limitations of the use of the IAEA code for the aims of the present Ph.D. activity.

The main problem observed is related to the fixed fuel cycle structure considered both for the front-end and the back-end. By this structure the simulation of the double strata fuel cycle as well as of the transition scenario seem not to be feasible. The adoption of some procedure where the simulation of each type of facility is performed separately can be in principle applied (as indicated also by [177]). However, this procedure can be susceptible to be affected by human-errors due to the way in which the user inputs the data in the code. The time consumed in applying this procedure is not reasonable in order to fulfill the objectives of the present study, where several scenarios (somehow different only for a single parameter, e.g. loading factor) ought to be compared.

The fixed structure is clearly indicated in Figure 3.9. From this figure, it appears that only Pu can be recycled and MAs sent always to disposal. Therefore, this limitation is in contrast with the aim of the activity where advanced fuel cycle based on P&T need to be modeled (e.g. the ones indicated in Par. 2.1). In addition, the procedure for modeling transition scenario is unclear (as indicated by previous examples), due to the unavailability of the source files and further useful information.

There is no possibility to perform a rigourous check of the input data.

The NFCSS code allows to model only a single reprocessing plant and a single fabrication plant whatever the destination of the fuel is (fast or thermal reactor). At the same time it is not possible to define different stocks for differentiating the material stream (e.g. Am, Cm, MAs).

Concerning the post-processing, the selected indicators can not all be evaluated by the code (e.g. radiotoxicity and heat load). An external code has to be used for assessing the isotopic evolution in the disposal and then, by means of suitable coefficients calculate the radiotoxicity and the heat load associated to the material disposed.

All these limitations led to the decision that the NFCSS code [16] is not sufficient for the aims of the study. However, the code has been useful in order to perform the preliminary scoping study which could be considered as basis for the following activities.

Several codes have been developed worldwide in order to assess the future fuel cycle needs associated to a selected scenario. Some of them have been developed at industrial level (e.g. the TIRELIRE - STRATEGY developed by EDF-France [63] or the COSAC code developed by AREVA [178]) or by laboratories (e.g. COSI code [20] developed at CEA-CADARACHE or DANESS code [179] developed at Argonne National Lab).

Within the Working Party on Scientific Issues of the Fuel Cycle (WPFC) by the Expert group on Fuel Cycle Transition Scenarios Studies (WPFC/FCTS) also a code-to-code benchmark has been performed considering three scenarios with increasing difficulty levels (preliminary results are reported in [149, 150]).

The codes considered in the study are the ones most used worldwide as the COSI6 code developed at CEA-France, the DESAE2.1 code developed at ROSATOM-Russia, the EVOLCODE code developed at CIEMAT-Spain [180], the FAMILY21 code developed at JAEA-Japan and the VISION code developed at INL-USA. The outcome of the study has shown that general trends observed for each code are comparable. However, an increasing on the scenario complexity shows some discrepancies [149, 150].

The comparison has also demonstrated the importance of initial assumptions and the common interpretation of the hypotheses and results [149] in agreement with the approach adopted during the present Ph.D. activity.

Therefore, in order to address the objectives of the proposed studies, the adoption of a more complete and flexible code has been mandatory for fulfilling the given objectives.

The code selected is the COSI6 code [20], a code widely used in France for fuel cycle and scenario analyses [9, 133].

The adoption of this new computational tool has been possible thanks to the collaboration between University of Pisa (Italy) and the Karlsruhe Institute of Technology (KIT) in Germany where the second part of the Ph.D. activity has been developed.

The general scheme of the COSI6 code is shown in Figure 3.36. The code is more flexible than NFCSS code enabling the simulation of a fleet composed by several type of systems. The black arrows indicated in Figure 3.36 can be fixed by the user enlarging the range of applicability of the code.

In order to show the advantages of the COSI6 code with respect to the NFCSS code, the comparison performed by [21, 181] can be very useful.

According to [21], four technical functions can be distinguished for a fuel cycle code:

- Function 1: Characterize and deploy individual fuel cycle facilities and reactors;
- Function 2: Perform component and aggregate uncertainty analyses;
- Function 3: Optimize simultaneously across multiple objective functions;
- Function 4: Open and accessible code software and documentation.

These functions can be further subdivided in specific requirements. A list of them is included in Table 3.9.

As indicated in Figure 3.37, COSI6 satisfies the larger number of requirements requested for a fuel cycle code. In particular, the comparison between COSI and NFCSS codes shows the larger flexibility of COSI6 compared to NFCSS.

More details about the codes adopted are summarized in Appendix A.

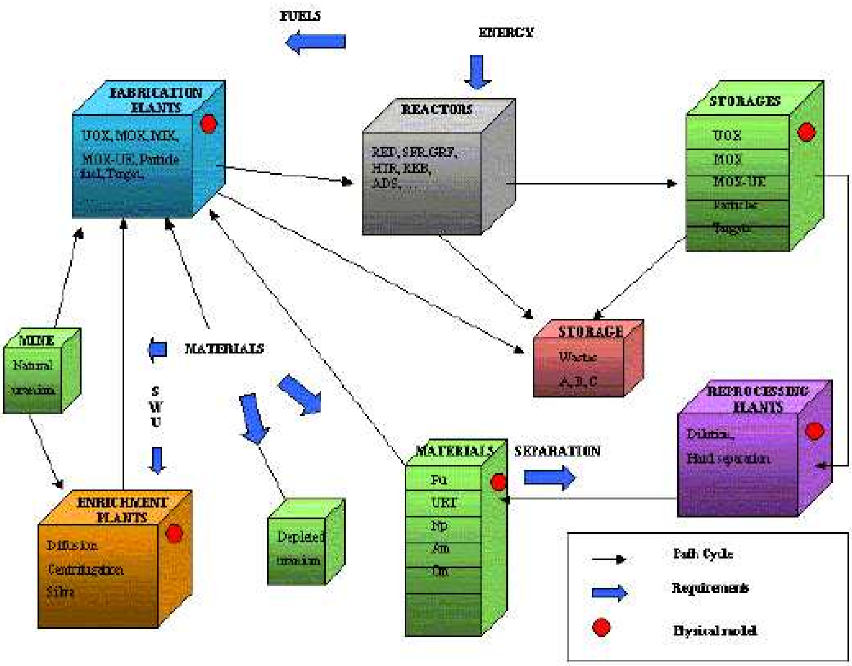

Figure 3.36: COSI6 fuel cycle general scheme [20]

Requirement	CAFCA	COSI	CEPMNFC	DANESS	NFCSim	VEGAS	VISION	NFCSS	GENIUS 1
R1.1									
R1.2		✓	✓		✓				
R1.3		✓	✓		✓				✓
R1.4		✓			✓				✓
R1.5		✓			✓				✓
R1.6		✓			✓				
R1.7									
R1.8		✓	✓		✓				
R1.9		✓	✓		✓				
R2.1									
R2.2									
R2.3									
R3.1									
R3.2									
R3.3	✓	✓		✓	✓		✓	✓	
R4.1	✓			✓		✓	✓	✓	✓
R4.2	✓						✓		✓
R4.3		✓							

Figure 3.37: Comparison between fuel cycle codes [21]

Requirement	Description
R1.1	Simulations must be able to reflect all significant design data for elements of a fuel cycle
R1.2	Track quantities of natural resources as a function of time, location, and accessibility
R1.3	Track process materials as a function of time, location, and accessibility element and isotope
R1.4	Track the operations status of each production facility as a function of time, location, and capacity
R1.5	Track the operations status of each storage facility as a function of time, location, and capacity
R1.6	Track the operations status of each disposal facility as a function of time, location, and capacity
R1.7	Track status of nuclear materials transportation as a function of time, location, and type
R1.8	Track products and by-products as a function of time, location, and type electricity, heat, hydrogen, etc.
R1.9	Track costs expenditures as a function of time, location, and type
R2.1	Capable of performing uncertainty analysis for each element of the fuel cycle
R2.2	Generate sensitivity coefficients for each element of the fuel cycle
R2.3	Capable of performing aggregate uncertainty analysis by propagating the uncertainty in each element of the fuel cycle
R3.1	Capable of running in a semiautomated mode using inputs to produce desired model outputs. e.g., nonproliferation, economic, and waste management targets
R3.2	Be able to dynamically perturb local optimum solutions to test robustness and adaptability
R3.3	Support a graphical user interface
R4.1	Maintain abstractions between data and process algorithms
R4.2	Open and accessible code architecture, source code, and documentation
R4.3	Be able to communicate with other codes through weak links or databases

Table 3.9: Fuel Cycle code requirements according to four technical functions [21]

In order to perform fuel cycle calculations adopting COSI6, suitable libraries for each system have to be generated with appropriated codes. More detailed knowledge about neutronic models and burn-up calculations are needed.

For fast reactors, the neutronic model is assessed using the ERANOS code [182] and then converted by means of the APOGENE code [183] for the final use in COSI, as indicated in Figure 3.38.

The cross-sections (33 groups burn-up dependent) and fluxes generated for the 3D neutronic model are associated to the COSI model where only the mass of the system (and not the geometry) is taken into account.

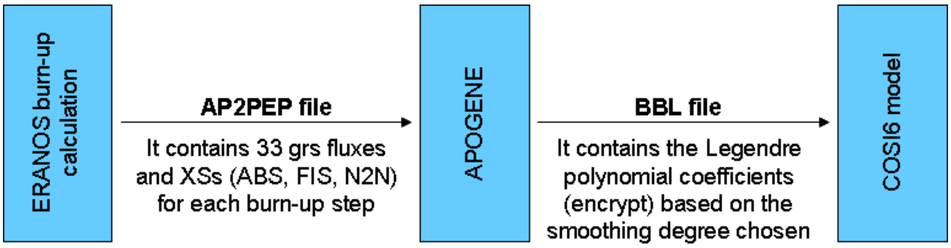

Figure 3.38: Schematic way for creating the BBL for COSI6

More details are included in Appendix D where the neutronic models of the fast reactors considered are described.

Concerning the model of EPR in COSI, the available library in the code has been adopted for the study, assuming all other parameters in agreement with CEA studies, as time by time indicated while presenting the results. According to [183], the libraries for thermal reactor systems are generated with the same approach indicated in Figure 3.38 but adopting the APOLLO2 code [184]; this code is not available for the present study.

3.5 Summary

In the present Chapter a preliminary scoping study has been described.

This preliminary study has been essentially devoted to the methodology to be followed and to the assessment of tools.

As for the definition of the boundary conditions, the features related to the initial conditions (e.g. reactors in operation and nuclear energy share), the future expected energy trends and the transition strategy have been carefully analyzed.

Moreover, in order to compare the scenarios in terms of long-term sustainability of the nuclear energy sector, four indicators (as radiotoxicity, heat load, resources and waste inventory) have been selected.

Finally, the capabilities of the available tools for fuel cycle studies have been considered.

A few relevant scenarios have been modeled first with the IAEA NFCSS fuel cycle code. The results obtained have shown that the NFCSS code, despite its simplicity, is not flexible enough to meet the objectives.

The adoption of the COSI6 code, a more flexible code (even if more complex and time consuming), has been the choice to overcome the NFCSS shortcomings.

Chapter 4

Nuclear Energy Development based on LWRs Deployment

On the basis of the preliminary scoping study described in Chapter 3, some critical items associated to the procedure adopted have been pointed out. The adoption of a more flexible fuel cycle code as well as the selection of a reference case has enabled to overcome these items.

Under these conditions a refined analysis has been developed. The results obtained are summarized in the present and next Chapters.

The choice of the hypotheses and boundary conditions is the most critical point for a scenario study. Some hypotheses, as the energy envelops and reactor systems, give a large impact on the results that can change also the main trends. Other hypotheses (e.g. reactor lifetime, burn-up) give an impact on the results but the major trends remain essentially unchanged.

In order to quantify the impact of each parameter studied separately, a parametric study (on LWRs and FRs) has been performed. The outcome is a sort of "database" containing the variations (in %) of the selected indicators (U and Pu, SF inventory, radiotoxicity and heat load) for each parameter studied. This set of data can be considered as some type of sensitivity inventory which can be extrapolated also for other studies in order to take into account the uncertainties associated to the hypotheses chosen.

In addition, the quantification process has indicated which boundary conditions are the most important ones for the scenario definition (indicating which of the hypotheses could be initially neglected).

In order to quantify these variations, a reference case has been selected.

The reference case is a country with constant nuclear energy demand small enough to be used as "unit of measure" for increasing nuclear energy demand scenarios[1] (mainly for regional areas as e.g. Europe or OECD area) and large enough to investigate in detail some hypotheses (e.g. substitution of the LWRs fleet).

The nuclear energy production considered is 70 TWhe/y (this corresponds for instance to 20% of the electricity production in a country like Italy). Therefore, the reference scenario, adopted as "unit of measure" for Europe (e.g. by assuming superposition effects), has been called the "Italian scenario".

The investigation of the cycle implications (facilities, resources and waste involved) associated to an option with re-introduction of nuclear energy been considered.

This case has been selected because it can be easily extrapolated to other countries adopting a scaling factor based on the nuclear energy demand.

[1]In order to better clarify the definition of some parameters, in the study sometimes the gross electric nuclear production of the fleet is called also "energy demand". This is related to the way in which these energy curves are used in COSI6 code. Actually, they represent an input for the code to be satisfied by the facilities in the cycle. In fact, the mass balance between the facilities is driven by energy data and in this sense it is a "demand".

Due to the systems considered (e.g. EPR and self-sustaining FRs), the results are suitable for the direct extrapolation to other local areas (e.g. Armenia, Poland, China).

In the study, issues related to both introduction of a thermal reactor fleet and transition toward a fast reactor fleet are taken into account and quantified (respectively discussed in Chapter 4 and Chapter 5).

The reference scenario selected enables to analyze two main strategies:

- the adoption of **"once-through"** fuel cycle where a thermal reactor fleet (consuming only uranium) is developed and the total SF is sent to the repository without Pu and U recovering;
- the adoption of **closed or partially-closed** fuel cycles where the LWRs fleet is gradually replaced by a FR-based fleet. The complete substitution has been analyzed too in order to investigate the equilibrium conditions.

All the simulations have been performed by means of the COSI6 code [20], a dynamic scenario analysis code originally developed at CEA (France), and now largely adopted at the European level [5, 4].

4.1 Hypotheses for the reference scenario: nuclear energy demand and reactors considered

In agreement with the approach adopted in Par. 3.1, the first step for defining a scenario is the identification of the initial conditions. For this purpose, the age of the in operation nuclear fleet (in order to take into account the reactors replacement once they have reached the reactor lifetime) and the nuclear share in the energy mix are aspects to be considered.

The same approach, as described in Chapter 3, has been followed in order to identify the initial conditions for the reference scenario.

In Par. 3.3, it has been described how the energy projections provided by international organizations (e.g. NEA, IAEA) can be used to set up scenarios for the local areas. The main criterion adopted is to maintain (or to gradually modify) the energy trends of a country (following, at least for the next 10-20 years, the development of nuclear energy in the past) of course "smoothed" on the age of the fleet and on the declared energy policy.

This continuation criterion can not be applied to the reference case (or to any other country in the same conditions). Therefore, some additional hypotheses have to be added (e.g. the hypothetical scenario proposed in Par. 3.3.2).

As additional hypothesis a constant nuclear energy demand (ca. 70 TWhe/y introduced in the period 2020-2030) has been considered as reference case. The nuclear energy production is sufficient small to represent the "unit of measure" for more complex regional scenarios and sufficiently large to represent the electricity produced by 6-7 reactor systems and to enable the investigation of several hypotheses. Under this assumption the introduction rate is expected to be much higher, ca. 7 times more, than the average nuclear energy rates proposed by IAEA [15], NEA [89] or IPCC and IIASA [3, 23] for Western Europe.

The value of 70 TWhe/y has been chosen on the basis of the indications given e.g. in 2008 by the Italian government. 70 TWhe/y corresponds to about 20% of the electricity production in Italy [122, 38]).

As additional parameter for the study, increasing energy demands (assessed on the basis of IIASA electricity projections [23]), have been considered as indicated in Par. 4.3.2.

For the reference case, the nuclear energy production is considered constant in the period 2030-2200. The data adopted are indicated in Figure 4.1 corresponding to a nuclear energy production equal to 70 TWhe per year. With this assumption, it has been implicitly taken into account that the possible increasing energy demand should be covered by alternative sources of energy.

This energy envelop has been adopted for the whole parametric study except for the case with increasing energy demand. These increasing trends have been based on the electricity projections provided by IIASA for Western Europe [23] assuming that nuclear energy will cover always the same fraction along the years (~20% or 40%). The detailed description of the energy envelops is presented later on (Par. 4.3.2).

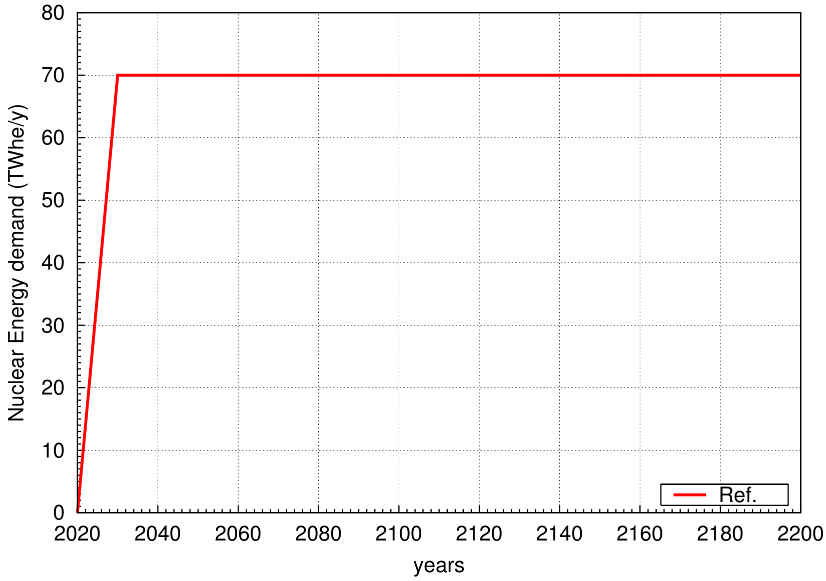

Figure 4.1: Nuclear Energy demand (TWhe/y) assumed for the reference scenario: period 2010-2200

On the basis of the energy demand depicted in Figure 4.1, the "once-through" strategy where only LWRs are deployed has been studied. The fuel cycle associated to this strategy is indicated in Figure 4.2.

Under this scheme, all the spent fuel (including Plutonium, Uranium and Minor Actinides) is directly sent to the repository without TRUs recovering. The scheme adopted is in agreement with the general scheme presented in Figure 2.1.

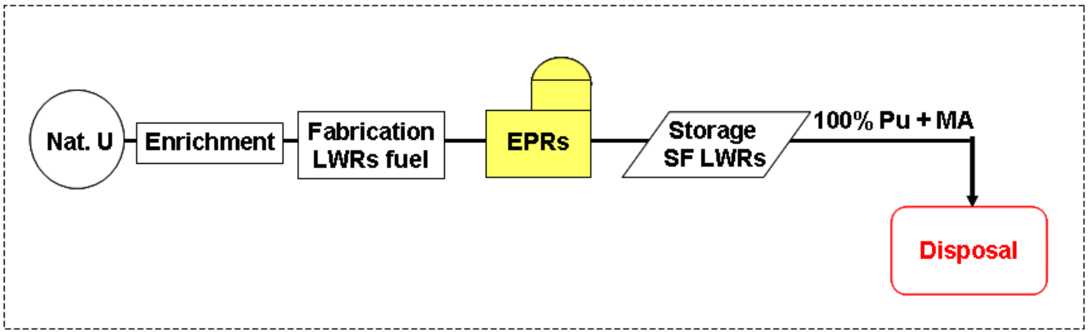

Figure 4.2: A simplified flow scheme for the reference scenarios: "once-through" strategy

In order to simplify the analysis, a single type of advanced LWRs has been considered, namely the European Pressurized Reactor (EPR-type), [124]. The model considered for the COSI6 implementation is in agreement with other recent studies, e.g. [7, 9]. The main data adopted are indicated in Table 4.1.

	EPR-like
Thermal Power (MWth)	4500
Electric Power (MWe)	1550
Load factor (%)	81.76
Cycle length (efpd)	4*366.6
Efficiency (%)	34.44
Ave. Burn-up (GWd/tHM)	55
Initial HM mass (tons)	~120
Enrichment [U235], (%)	4.6
Fuel (S/A geometry)	UOX (17x17)
COSI6 library, spectrum	UOX, 14, Thermal

Table 4.1: Thermal reactor characteristics: EPR-like [7]

In order to set up the COSI6 model, in addition to the input data summarized in Table 4.1, suitable one group burn-up dependent equivalent cross-sections are needed. For the study, the library available in the COSI6 distribution[2] has been assumed as reference, since the APOLLO2 code [184], deterministic code adopted in France for thermal reactor neutronic models, was not available.

As indicated in Par. 3.4, for the transition phase study "ad hoc" libraries have been generated for each fast system considered (see Chapter 5). Due to the availability and applicability of the ERANOS code [182], the neutronic models have been assessed (see Appendix D).

4.2 The reference case: "once-through" option

The reference case is described here in order to assess the basis of comparison for the parametric study. For this case, the LWR considered is an EPR-like system with 55 GWd/tHM burn-up and 4.6% U235 enrichment (no MOX fuel has been considered).

The nuclear energy demand considered is shown in Figure 4.1. The introduction of ca. 6 EPRs (modeled as in Table 4.1) in 10 years results in a quite high introduction rate (7 TWhe per year).

The standard way for modeling a reactor fleet in COSI6 code[3] is the adoption of an "equivalent reactor" representing the whole fleet (both in terms of energy produced and the total mass [4]). This approximation implies that all reactors have common conditions in terms of load factor and efficiency.

Refined studies, assuming a separate model for each reactor composing the fleet have been performed (see Par. 4.3.3) but no advantages in terms of major trends have been noticed. Therefore, the "equivalent reactor" approach has been considered suitable for the present activity.

Taking into account the load factor listed in Table 4.1, corresponding to the average load factor assessed for the French fleet[5] [7, 17, 150], the nuclear capacity needed to produce the energy demand chosen (70 TWhe per year) is about 9.8 GWe and it corresponds to about 6.3 EPR-like Nuclear Power Plants (NPPs). In order to consider a finite number of systems the constant level of the energy demand[6] has been slightly changed and 6 respectively 8 EPRs have been considered (see for more details Par. 4.3.2).

Based on these assumptions, the impact on front-end and back-end of the fuel cycle has been evaluated.

[2]Namely `BIBLIO-CEA14.BBL` data library.

[3]In particular when increasing energy demand is considered.

[4]The mass associated to the core of the "equivalent reactor" corresponds to the sum of the masses of each unit.

[5]The average load factor calculated over the last 20 years in France has been assumed as reference for simplifying the study. Approximation considered suitable for the analysis.

[6]See also note n.1.

The total cumulative natural uranium amount needed for sustaining the cycle of 6.3 units is in the order of 111,500 tons in 2100 and 257,700 tons in 2200 corresponding to about 1,450 tons per year. This cumulative behavior is shown in Figure 4.3 where the natural uranium extracted, the depleted uranium and the used uranium are indicated.

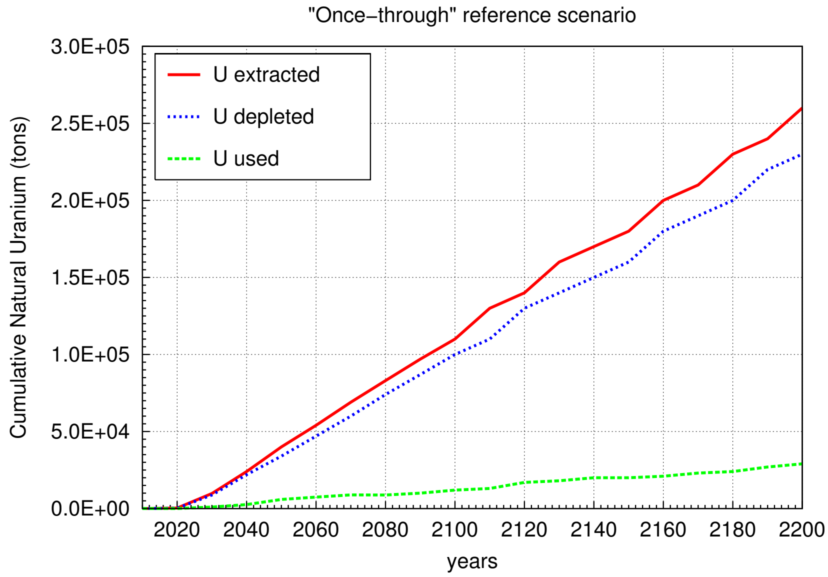

Figure 4.3: Cumulative natural uranium consumption for the reference scenario

The quantities involved for the reference scenario are very low if compared with the total resources of ca. 40 Mtons of uranium estimated [2, 57].

As expected according to the low nuclear energy demand considered, the influence of the reference scenario on the uranium market is limited, e.g. in 2100 it corresponds to ~0.3% of the total uranium resources, including phosphates [2].

In order to quantify an order of magnitude of the uranium needed to sustain this kind of scenario for 200 years, the uranium demand has been compared with the annual production of Australian mines (~8,500 tons per year [185]).

The quantity of uranium needed before the end of the next century corresponds to ca. 30 years of full capacity operation of the Australian mines. Whereas, limiting the analysis to 2100, the uranium needed corresponds to ca. 13 years of operation for the same mines [185]. The values here presented are only indicative values that are used for providing the major trends depending on the hypotheses made (e.g. on burn-up).

As indicated above, when a scenario study is oriented to a country's scale, an additional key point to be treated is the assessment of the waste inventory, the potential risk associated (e.g. by radiotoxicity evaluation) and the capacities of the facilities composing the fuel cycle. For the "once-through" case only fabrication and repository capacities need to be analyzed.

The cumulative total amount of SF sent to the repository is about 10,900 tons in 2100 and 26,400 tons in 2200. For every cycle 189 tons are discharged and sent to the repository.

The quantity of Pu in the waste is 135 tons in 2100 and 328 tons in 2200 respectively (in both cases, representative for a burn-up of 55 GWd/tHM). These values correspond to ~1% of the total SF, and they are in agreement with literature data.

In order to have a basis of comparison, the annual SF discharged is compared with the data available for the French case. The value obtained for the reference case corresponds to about 1/6 of the annual SF discharged by the French fleet (about 1,100 tons per year [89]).

This is an example of scaling of the activity because the 1/6 ratio for the annual SF (the same ratio could also be found for other parameters, e.g. fresh fuel fabricated amount) corresponds to the ratio between the energy demand assumed for the reference case (70 TWhe/y) and the one assumed for the French case (430 TWhe/y, [7, 125]).

Small differences exist and they can be justified by the reactor considered in the scenarios. For the reference case, a fleet composed by identical EPR-like reactors has been considered whereas for the French case, several operating reactors with different burn-ups and characteristics (e.g. enrichment and power [17]) are taken into account.

The same consideration can be derived from the cumulative spent fuel analysis. In fact, the cumulative SF in France in 2008 is about 12,400 tons as indicated by [89]. According to the relative behavior of the energy demand between the two countries, the expected cumulative value after 25-30 year from the introduction of the first EPR in the reference scenario is expected to be of the order of 2,000 tons (ca. 1/6 of the French stockpile).

The value in 2045 assessed by the simulation is 2,500 tons. This value is slightly larger than the expected value calculated by the scaling down of the French case. This discrepancy can be justified considering that in the France case Pu has been recovered for the use of MOX in LWRs. In the value provided by [89], part of the SF inventory has been reprocessed, whereas in the simulation for the reference case this contribution to stockpile reduction is not taken into account because the use of MOX has not be considered.

Another parameter evaluated is related to the facilities belonging to the fuel cycle. In order to sustain the Italian scenario, the annual fabrication capacity needed is around 189 tons (1/6 of the annual capacity of the currently existing French fabrication plant 1,400 tons per year [89]), that corresponds to an enrichment capacity feed material of 1,450 tons per year (where the U235 enrichment required for the target burn-up is 4.6%[7]). Also these parameters are affected by the hypotheses chosen.

A further parameter of interest is the composition of material sent to the repository and its evolution under "once-through" strategy. The mass and waste composition in terms of Pu and MAs are summarized in Table 4.2. The MAs content in 2200 in disposal is 74 tons of which 41 tons (ca. 55%) are generated by the Pu241 decay into Am241 (half-life of 14.35 years). The impact of Pu separation and multi-recycling is addresses in Chapter 5.

Cumulative values in Disposal								
year	SF	Pu		MAs		Am	Cm	Np
	tons	tons, %		tons, %		tons	tons	tons
2045	2,556	28	1.10	4.3	0.17	2.13	0.23	1.94
2100	10,900	135	1.24	28	0.26	17.5	0.56	9.53
2200	26,400	328	1.24	74	0.28	46.7	0.75	26.54

Table 4.2: SF composition in disposal

[7] According to the natural uranium feed material amount, 1,450 tons per year, the tail assay considered, 0.25% and the produced material, ca. 155 tons per year, the work necessary in the enrichment plant can be assessed to ca. 1,085 tons of Separative Work Unit (SWU). Data in agreement with [186].

4.3 The parametric study

With respect to the reference "once-through" case, the parametric study has been conducted.

In particular, the parameters investigated within this Ph.D. activities are (see also Table 4.3):

- The LWRs discharge burn-up. Several values have been assumed in order to quantify the impact on U resources and on Pu quality,
- The LWRs reactor introduction rate (it will be analyzed how the introduction rate can impact the Pu available in the cycle and the early and dynamic introduction of FRs),
- The energy demand, adopting different constant or increasing energy envelops,
- The impact of reactor lifetime, comparing 40 and 60 years lifetime.

Parameters considered	
Burn-up (GWd/tHM)	33, 50, 55, 65
Batch fraction	3, 4, 6
Energy demand (TWhe/y)	70 (ca. 6.3 EPRs, 20%) 66.6 (6 EPRs, 19.6%) 88.8 (8 EPRs, 26%) Increasing demand
Introduction rate (years)	10 (2020-2030) 20 (2020-2040)
Start-up core	With or without
Reactor lifetime (years)	40, 60

Table 4.3: LWRs parametric study: parameters considered

These parameters have been evaluated separately and presented in the next sections. A final overview is reported in Par. 4.3.4.

4.3.1 Influence of the discharge burn-up

The average discharge burn-up can influence the uranium resources involved, and the availability and quality of Pu for the transition to a fast reactors based fleet.

In order to provide an overview as extensive as possible, four different discharge burn-ups have been considered for the comparison. The range between a burn-up of 33 GWd/tHM (corresponding to the earlier LWRs burn-up) up to the maximum target burn-up fixed for EPR (65 GWd/tHM [66]) has been considered.

For each burn-up, assuming that the energy indicated in Figure 4.1 is delivered by the same type of reactor (i.e. same mass of the core and same thermal power), the relative irradiation cycle lengths (taking into account also the fraction of fuel discharged from the core during a refueling outage) and the U235 enrichments have been evaluated.

For evaluating the total effective full power days (efpd) depending on burn-up, the simple relation between total irradiation length ($efpd_{cycle} \times n_{cycle}$), *burn-up*, thermal power (*Pth*), and mass has been applied (see Eq. 4.1) keeping constant the *Inventory* (equal to ca. 120 tons) and the thermal power (Pth equal to 4,500 MWth). Therefore, for each burn-up the total residence time has been calculated (see Table 4.4).

The U235 enrichment is adjusted in agreement with the target burn-up. In this study, literature data have been considered [22] as indicated in Figure 4.4. In this figure, the relation between the initial enrichment and the average discharge burn-up for several reactor concepts [22] is shown as well as the values adopted in the study (red points). As indicated by the linear behavior of these points, the values adopted are in good agreement with the PWR case with 4 batch refuelling scheme and gadolinium poison.

$$P_{th} = \frac{burn-up \times Inventory}{efpd_{cycle} \times n_{cycle}} \tag{4.1}$$

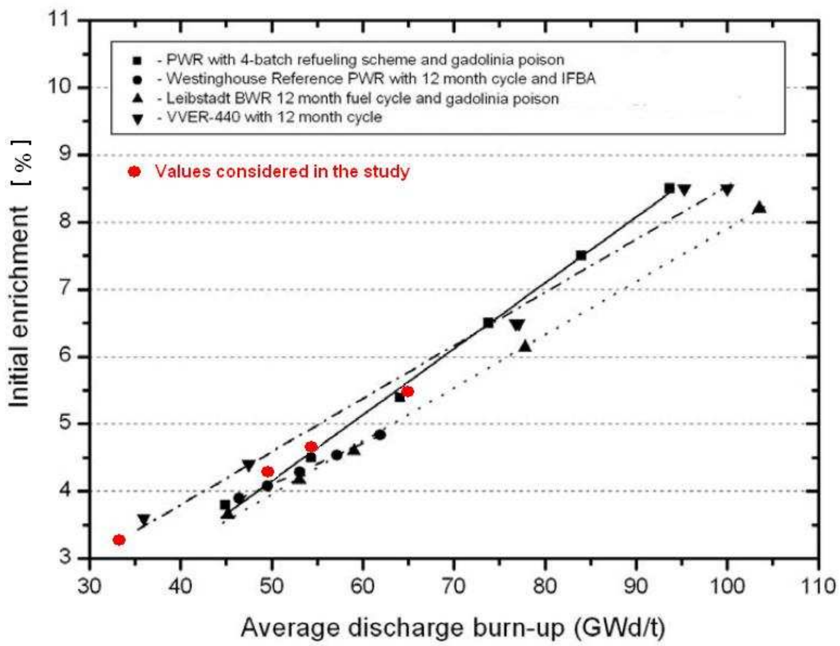

Figure 4.4: Initial enrichment versus average discharge burn-up trends [22]

In particular, the complete set of data adopted in the simulations is indicated in Table 4.4. Other parameters, like efficiency and load factor are those of Table 4.1.

For the parametric study, a single BBL library is considered. The library selected (`BIBLIO-CEA14.BBL`) has been generated in order to accept different ranges of burn-up and enrichment. The available enrichment range is between 1.5% and 10.1% U235 and the burn-up ranges between 0 GWd/tHM and 100 GWd/tHM [187]. All data chosen for the study (Table 4.4) are included in those ranges.

The impact of burn-up variation on front-end and back-end parameters has been assessed.

The first parameter investigated is the influence of the discharge burn-up on the natural uranium demand. The results obtained are summarized in Table 4.5.

As expected, the natural uranium cumulative mass extracted (expressed in tons) does not change significantly with the burn-up. Assuming 55 GWd/tHM as reference burn-up, the variation on uranium demand could be between ~13% more for 33 GWd/tHM and ~2% more for the case with 65 GWd/tHM.

This behavior, with the minimum uranium demand for a burn-up in the range of 50-55 GWd/tHM, is due to a balance between the U235 enrichment required and the amount of energy produced by the differently enriched fuels and the fraction of energy provided by fissioning of Pu. This behavior can easily be seen in Figure 4.5 where the total values of uranium demand are plotted for several years.

	Burn-up (GWd/tHM)			
	33	50	55	65
	Enrichment [%U235]			
	3.2	4.2	4.6	5.5
Batch Fraction	Irradiation length			
3	880 efpd			
4	880 efpd	1334 efpd	1466 efpd	1734 efpd
6				1734 efpd
Batch Fraction	Residence Time, including load factor			
3	1076 days			
4	1076 days	1632 days	1793 days	2120 days
6				2120 days
Composition Unloaded Fuel, COSI6 results				
U (%)	0.87	0.79	0.81	0.90
Pu (%)	0.97	1.19	1.24	1.35
Am (%)	0.07	0.12	0.14	0.17

Table 4.4: Parameters adopted for the burn-up study [8]

Burn-up (GWd/tHM)	33		50	55	65	
	Batch Fraction					
	3	4	4	4	4	6
year	tons					
2100	126,400	126,400	111,900	111,500	114,500	113,800
Share (%)	13.4	13.4	0.4	Ref.	2.7	2.1
2200	290,900	290,900	256,100	257,700	263,100	263,100
Share (%)	12.9	12.9	-0.6	Ref.	2.1	2.1

Table 4.5: Cumulative natural uranium demand versus burn-up

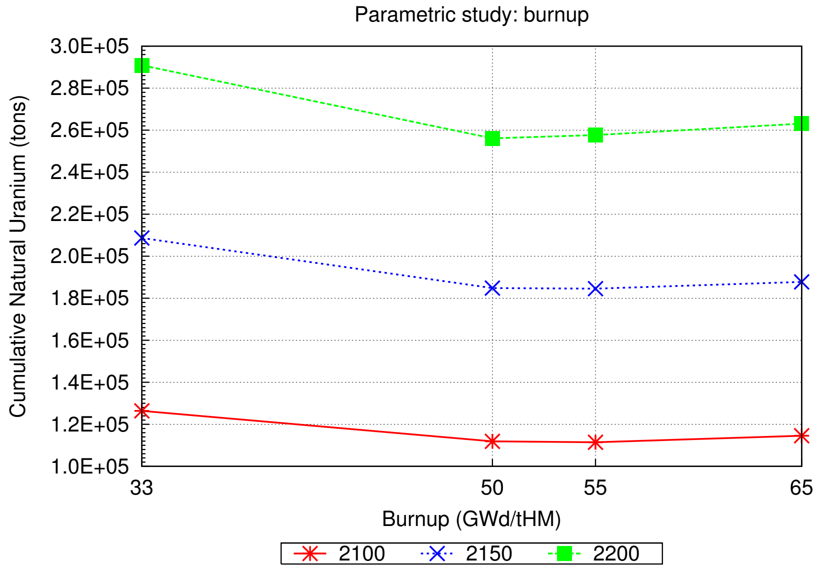

Figure 4.5: Cumulative natural uranium demand versus burn-up

As expected for higher enrichment a higher burn-up and a higher energy output can be achieved. Owing to the increased in-situ fission of Pu, its contribution to the total energy production is higher than in the case of lower enrichment.

However, at the same time an increasing fraction of neutrons is absorbed in fission products and MAs already introduced and, thus, is removed from the fission chain so that the unloaded fuel still contains a larger fraction of Pu isotopes than remaining in the unloaded fuel for lower enrichment (see Table 4.4).

This behavior could be seen in Figure 4.6, where the cumulative natural uranium demand is represented for the period 2180-2200 (period chosen only to highlight the effect).

From Figure 4.6, it is clear that the impact of the batch fraction reloading scheme (f3, f4, f6) is completely negligible. For instance, in the 33 GWd/tHM case with batch fraction equal to one-third (red line) and one-quarter (dots blue line) of the core; the two curves are comparable, small variations are only due to the different time in which the fuel is loaded in the systems and, hence, different dates for fabrication and extraction[8]. This was an expected result because the total mass loaded in the systems has been considered fixed for the six cases[9].

Concerning this point, more details are provided in the following parts where the total mass and the annual fabrication capacity are considered. Therefore, the internal reloading scheme can influence the fuel cycle only in terms of annual trends and not cumulative values.

For what concerns the cumulative spent fuel, the maximum quantity is produced with a discharge burn-up of 33 GWd/tHM (44,500 tons) and the minimum quantity for a discharge burn-up of 65 GWd/tHM (22,200 tons). These values correspond to 68% more and 16% less quantity with respect to the reference (55 GWd/tHM).

[8] In COSI code the fuel cycle facilities date of operation are calculated on the basis of the reactor fresh fuel requirements. The reactor needs drive the front-end and the back-end mass flow as better explained in Appendix A.

[9] In the study it has been considered that the batch fraction does not impact the enrichment of fresh fuel. The expected effects have been considered negligible and, therefore, the approximation valid for the study. However, the variation of enrichment with respect to the core reloading scheme can be in principle taken into account in COSI simulation.

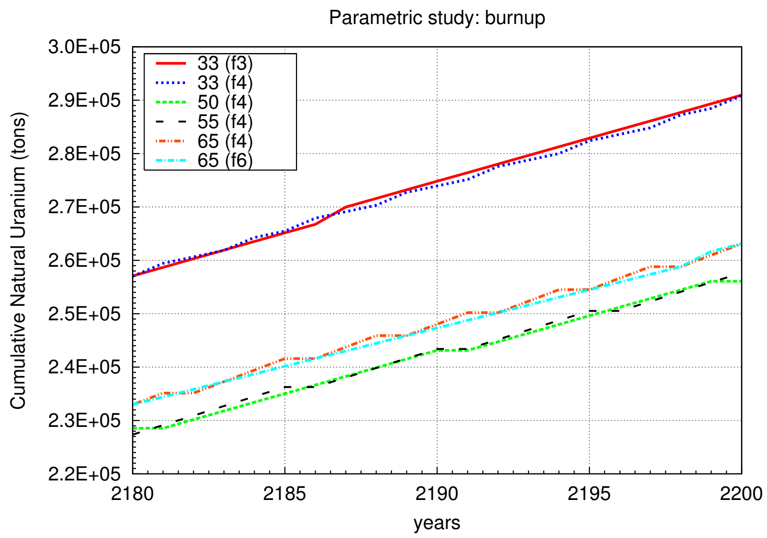

Figure 4.6: Natural uranium demand versus burn-up: period 2180-2200

This behavior depends on the number of batches loaded and unloaded from the core in the period of interest. For instance, for the case with 33 GWd/tHM and batch fraction equal to 4, the batch (corresponding to 1/4 of the core, therefore to ca. 30 tons) is unloaded every 269 days (time calculated as the full irradiation time, 880 efpd, divided 4 and divided by the load factor). For the 55 GWd/tHM with the same batch fraction, the same mass is unloaded every 448 days resulting in a smaller number of the same discharge material (147 times instead of 244 times). Therefore, the impact on SF stockpile is related to the different total residence time associated to different burn-up.

Even in this case, the contribution of the batch fraction reloading scheme is negligible (see Figure 4.7). In fact, the batch fraction influences only the cycle length and not the total residence time.

Equivalent trends have been obtained for the depleted uranium (minimum in the case of 50 GWd/tHM and equal to 228,800 tons in 2200) and for the Pu and MAs cumulative amount in the interim storage.

For what concerns the fabrication plant, different burn-ups with the same batch fraction reloading scheme (e.g. one-quarter of the core) requires the same fabrication capacity (the same core fraction is substituted every cycle) otherwise if the batch fraction is changed (e.g. from 4 to 3) the annual demand will vary accordingly (e.g. from 189 tons, one-quarter of the cores, to 253 tons, one-third of the cores). The values indicated in Figure 4.8 represent the mass of the batches calculated on the basis of the "equivalent reactor" model adopted in COSI6 code.

As indicated, the mass of batch in the case of 1/4 batch reloading strategy is 189 tons[10], which corresponds to a total mass for the equivalent core of about 756 tons. As above indicated the "equivalent reactor" correspond to 6.3 EPRs, therefore the mass of each core recalculated by the COSI6 data (756 tons divided into 6.3 units) is in agreement with the EPR heavy metal core mass adopted as reference for the study (see Table 4.1 [66]).

In Figure 4.8 it is also shown that all the systems with 1/4 batch reloading scheme need an equivalent mass of batch for every reloading (189 tons) but the time in which they are fabricated changes in agreement with the cycle length (in COSI6 the fabrication needs are correlated to the reactor batch loading needs).

[10]The average annual value corresponds to 155 tons.

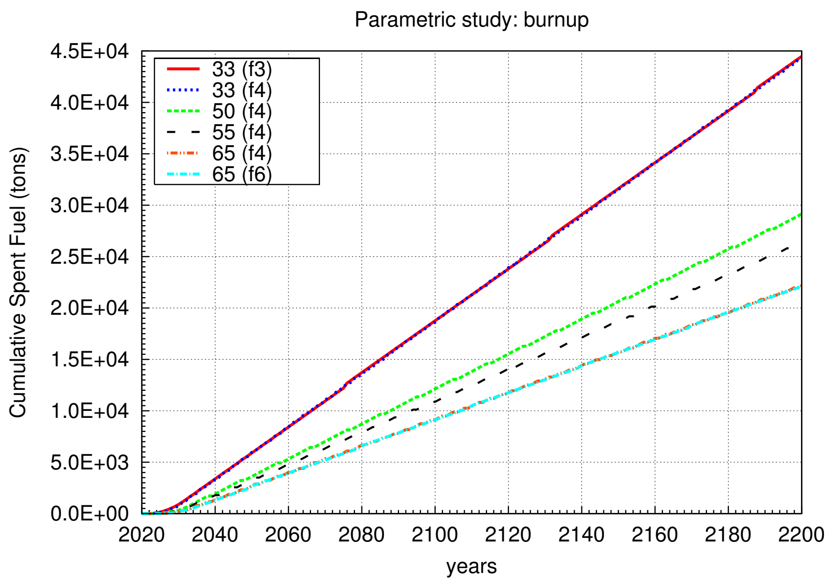

Figure 4.7: Cumulative Spent Fuel produced versus burn-up

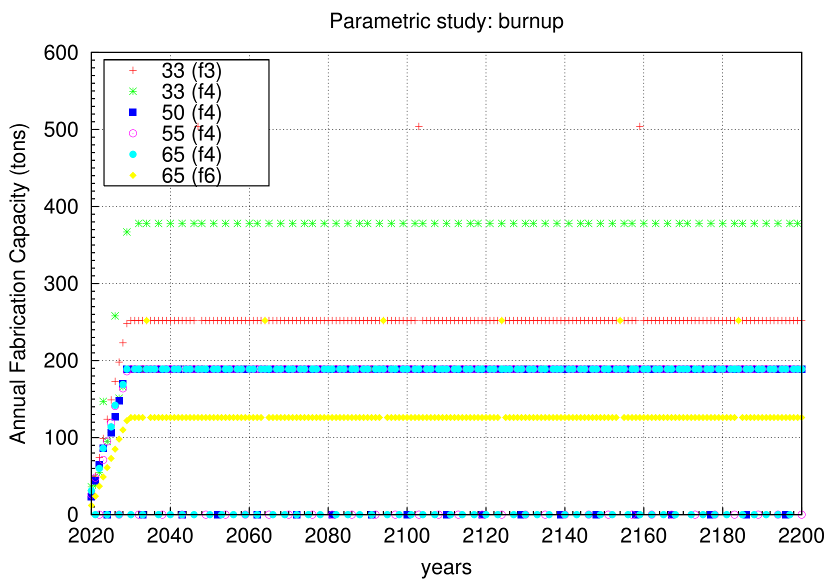

Figure 4.8: Annual Fabrication Capacity versus burn-up

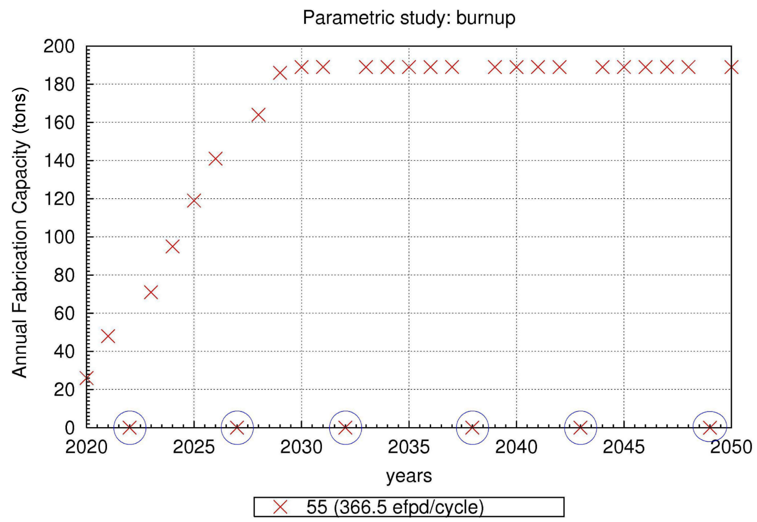

Figure 4.9: Annual Fabrication Capacity, 55 GWd/tHM case: zoom for explaining COSI6 model

In fact, as indicated in Figure 4.8 in some particular years the fabrication goes down to zero or is twice in regard to the constant value[11]. An example is indicated in Figure 4.9 when only the 55 GWd/tHM case is represented for a limited period of time in order to show more clearly the effect.

This rather unrealistic behavior is originated from the way the code calculates the fabrication needs. In fact, it calculates the time in which the fresh batch is loaded in the core on the basis of the efpd per cycle and of the load factor, but it provides fuel cycle facilities needs only every year. Therefore, it could happen that in a specific year no fresh fuel is loaded in the core, corresponding (according to the fabrication time assumed) to no requests of fabrication in a specific year. More details on this item are included in Appendix A.

A key parameter evaluated in this part, is the Pu availability in the cycle, because it affects the subsequent transition to fast systems (for 100% fleet substitution by FRs, 80-100 tons of Pu are needed, see Chapter 5 for details). This parameter varies significantly with the burn-up. Table 4.6 shows the Pu availability in 2050, 2080, 2100 and 2200 for all the burn-ups considered.

As expected, increasing the discharge burn-up (which means in-situ contribution of Pu to energy production) inevitably reduces the availability of Pu (for the future development of fast systems). In 2080, changing the burn-up from 33 to 65 GWd/tHM induces a reduction of the Pu available of ca. 30%.

In addition, high burn-ups worsen the Pu quality. Indeed as indicated in Table 4.7, the percentage of fissile material (Pu239, Pu241) decreases with increasing burn-ups. The values summarized in Table 4.7 have been derived assuming that the fuel is unloaded from the reactor and sent directly (without reprocessing or any other treatment) to the repository and the Pu vector correspond to the Pu in storage in 2050 (affected then by the Pu241 decay into Am241)[12].

[11] In the calculations performed any delay time (or buffer time) between the fabrication and loading has been considered as well as any time associated to transport of material between the fuel cycle facilities. In principle, these times can be taken into account in the COSI simulation. Further studies can be analyze the effect of these delay times.

[12] In order to clarify the apparent discrepancy between Table 4.6 and Table 4.7, it can be noticed that the value in Table 4.6 are affected by the dynamics of the scenario, i.e. the contribution coming from several batches is taken into account. Otherwise, Table 4.7

Burn-up (GWd/tHM)	33		50	55	65	
Batch Fraction	3	4	4	4	4	6
tons						
2050	57	58	43	41	37	37
2080	133	131	104	98	89	90
2100	182	181	144	135	123	123
2200	431	430	346	328	300	299

Table 4.6: Cumulative plutonium amount versus burn-up

2050						
Burn-up (GWd/tHM)	Batch Fraction	% weight				
		Pu238	Pu239	Pu240	Pu241	Pu242
33	3	1.48	56.02	23.46	13.63	5.41
	4	1.48	56.02	23.46	13.63	5.41
50	4	2.88	50.37	24.06	14.81	7.87
55	4	3.32	49.52	23.93	14.97	8.26
65	4	4.18	48.36	23.53	15.20	8.73
	6	4.18	48.36	23.53	15.20	8.73

Table 4.7: Pu vector in 2050

Another parameter changing with burn-up is the MAs content in the reactor unloaded fuel. In Table 4.8, the Pu and MAs fractions over the initial Heavy Metal (HM) loaded in the core (considering no cooling time after the discharge) have been summarized.

Burn-up (GWd/tHM)	33		50	55	65	
Batch Fraction	3	4	4	4	4	6
HM per batch (tons)	252.1	189.1	189.2	189.0	189.1	126.1
Pu/HM (%)	0.97	0.97	1.19	1.24	1.35	1.35
MA/HM (%)	0.07	0.07	0.12	0.14	0.17	0.17

Table 4.8: Pu and MAs content in the unloaded fuel versus burn-up (no cooling time after discharge has been considered)

Other parameters related to the waste produced and to the long-term impact on the repository are the SF radiotoxicity and the heat load, as described in Par. 3.2.

In Figure 4.10, the SF specific ingestion radiotoxicity i.e. normalized to the produced amount of electricity for three burn-ups (33, 55 and 65 GWd/tHM) is represented. Even if the differences due to burn-up are not so important, it can be noticed that for the time period between 300 and 100,000 years (where the Pu isotopes are the most important ones) a fuel with 33 GWd/tHM gives the higher contribution (more fissile is in waste). Whereas, after 100,000 years (where MAs and long-lived fission products are most important) it is the 65 GWd/tHM burn-up fuel that gives the higher contribution, in agreement with what was expected, pointed out also by [80].

The same conclusions can be drawn from Figure 4.11, where the SF decay heat evolution for the same three burn-ups is indicated.

refers only to a single batch, the one discharged from the core in 2050 (first batch discharged under equilibrium conditions).

In Figure 4.12 the MAs trends for different burn-up over years are given. As expected those trends are not affected by the different burn-up.

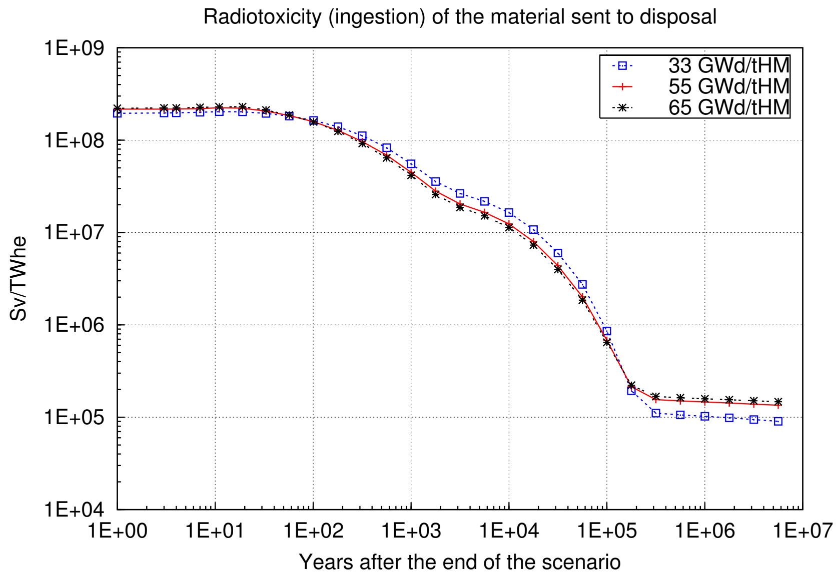

Figure 4.10: Specific radiotoxicity (ingestion) evolution versus burn-up [2200 is fixed as t=0]

In summary, the choice of the SF burn-up can impact the fuel cycle parameters as the cumulative uranium resources involved, the Pu inventory and the MAs stockpile. This impact is less important than other parameters (e.g. the nuclear energy demand). However it is important when for instance planning a transition toward a FR-based fleet. In fact, the Pu vector is changed and, as indicated in Chapter 5, it can play a role in the dynamic of the transition.

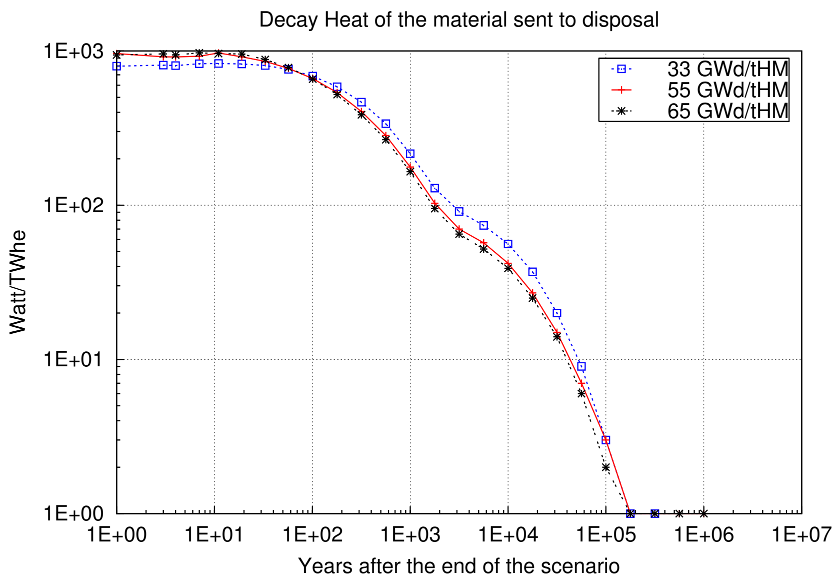

Figure 4.11: Specific decay heat evolution versus burn-up [2200 is fixed as t=0]

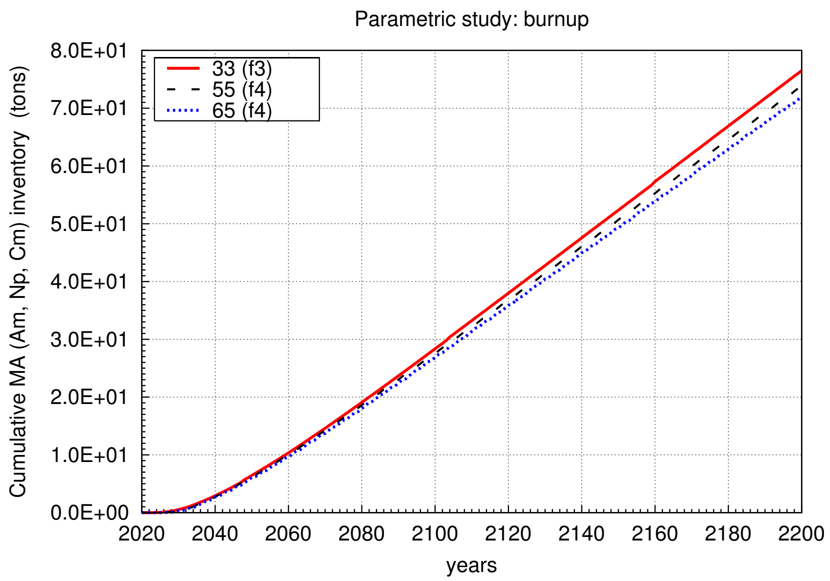

Figure 4.12: Cumulative MAs (Np, Am, Cm) in disposal versus burn-up

4.3.2 Influence of the nuclear energy demand and introduction rate

On the basis of the EPR-like system (described by the data listed in Table 4.1), it becomes clear that the energy demand previously considered (70 TWhe per year) could not be covered by an integer number of reactors. In fact, 70 TWhe/y correspond to ca. 6.3 EPR-like systems of 1.55 GWe each with 81.76% loading factor.

In order to consider a finite number of reactors installed, two cases have been addressed: 1) the introduction in 10 years of 6 EPR-like systems (55 GWd/tHM), capable to cover the 19.6% of the energy needs by the production of 66.7 TWhe per year; and 2) the introduction in 10 years of 8 EPR-like systems (55 GWd/tHM), able to cover the 26% of the energy needs by the production of 88.9 TWhe per year.

These two cases have allowed to quantify the impact of different constant nuclear energy demands (see Figure 4.13).

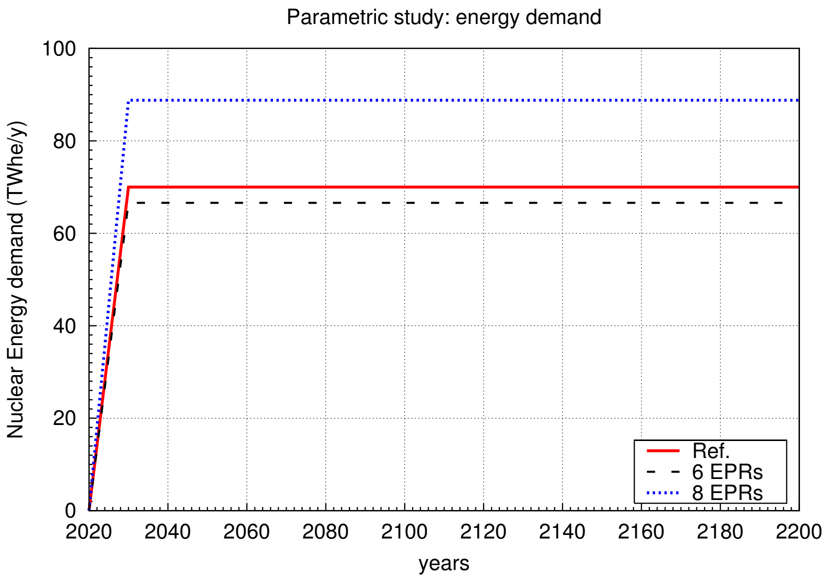

Figure 4.13: Different constant nuclear energy demands considered for the study

As expected the energy produced by 6 EPRs (66.7 TWhe/y) instead of (6.3) to produce 70 TWhe does not impact dramatically the uranium demand and the total waste produced (see Table 4.9).

Assuming that the reactor characteristics remain unchanged when dealing with a modified scenario, the variation of 1 TWhe/y corresponds to 1.42% of variation in uranium demand, (see Figure 4.14).

The same small differences are found for the facilities needs (e.g. annual fuel fabrication needs or enrichment plants). The annual fabrication capacity is reduced to 180 tons for the case of 6 EPRs (instead of 189 tons) and it becomes 240 tons for the case with 8 EPRs. The annual enrichment capacity (modeled as diffusion plants with 0.25% tails) is indicated in Figure 4.15, where the trends reflect the nuclear energy production lines.

Same trends can be found for the total spent fuel as indicated in Table 4.9.

The impact of the variation per TWhe is useful for the extrapolation of the results to more complex scenarios.

To show this potentiality, three cases (see Figure 4.16), have been compared with respect to the constant one:

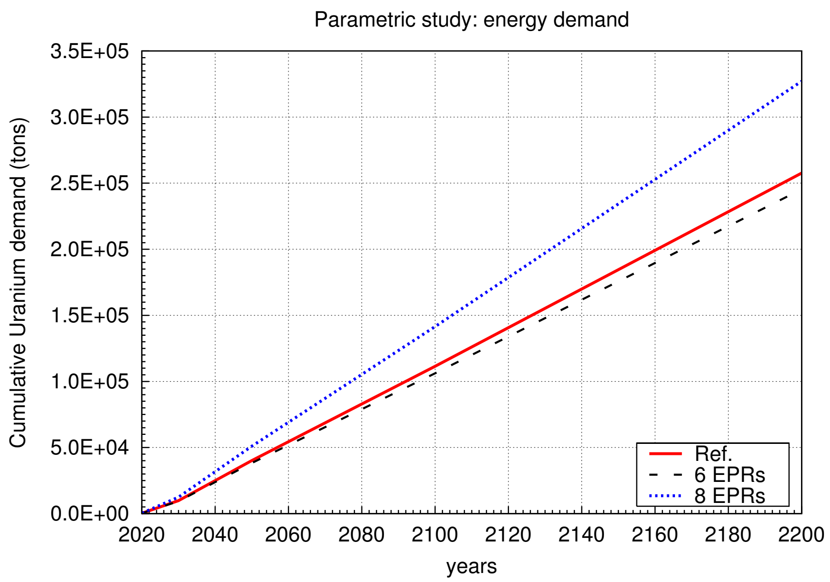

Figure 4.14: Cumulative natural uranium demand versus different levels of constant nuclear energy demand

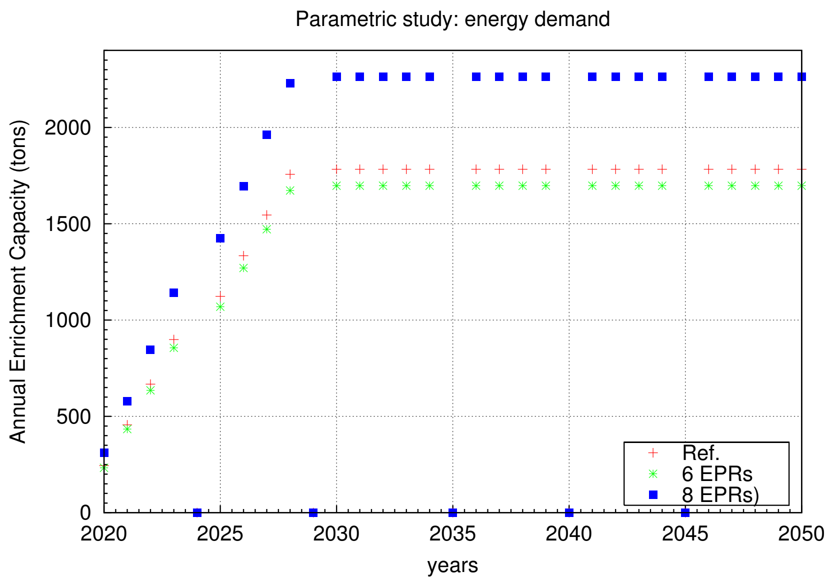

Figure 4.15: Annual Enrichment Capacity versus different levels of constant nuclear energy demand

	70 TWhe/y	6 EPRs	8 EPRs
Energy (TWhe/y)	70	66.7	88.9
Share (%)	20	19.6	26
Nat. U demand, tons			
2100	111,500	106,100	141,500
%	Ref.	-4.8	27.0
SF produced, tons			
2100	10,900	10,400	13,800
%	Ref.	-4.8	27.0

Table 4.9: Influence of the nuclear energy demand: 20% vs. 19.6% vs. 26%

- Case A: constant nuclear energy demand up to 2050 and then increasing energy demand for reaching 140 TWhe in 2200 (2 times the 2030 level).
- Case B: 20% nuclear energy share up to 2100 plus 0.1% year between 2100-2200 (as assessed on the basis of the IIASA electricity projection for Western Europe, B-IIASA scenario [4, 23]).
- Case C: the same electricity projection as Case B but with a higher nuclear share (40% instead of 20%). For the period 2100-2200 the same 0.1% per year increasing rate has been considered.

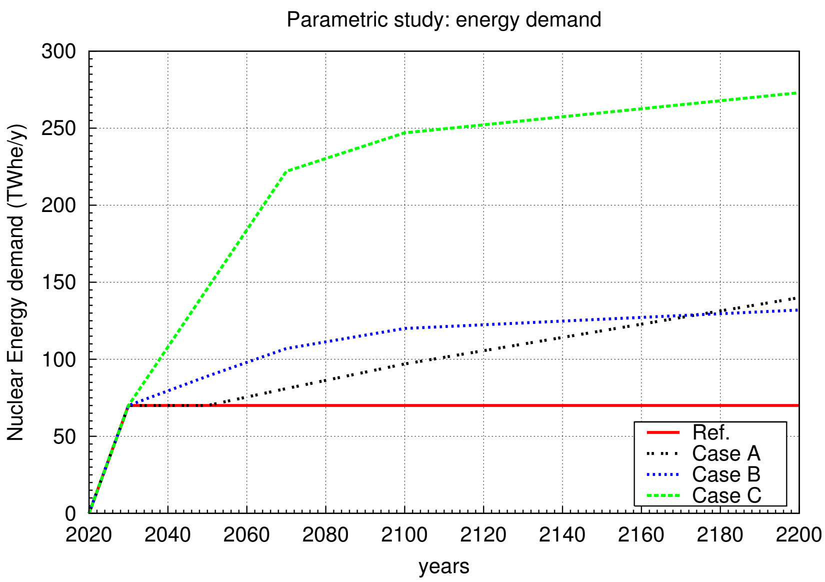

Figure 4.16: Different increasing nuclear energy demands considered for the study

The electricity projections for Western Europe and Italy up to 2100 are indicated in Figure 4.17 [23, 38]. In order to be consistent with the present situation, the 2010 value has been checked. As indicated in Figure 4.17, the value calculated for 2010 is 300 TWhe and it corresponds to actual electricity production in Italy [38]. In Table 4.10 the electricity production in EU and Italy is given.

In Figure 4.18 the uranium cumulative consumption as function of years for the four (A, B, C and reference) cases is given.

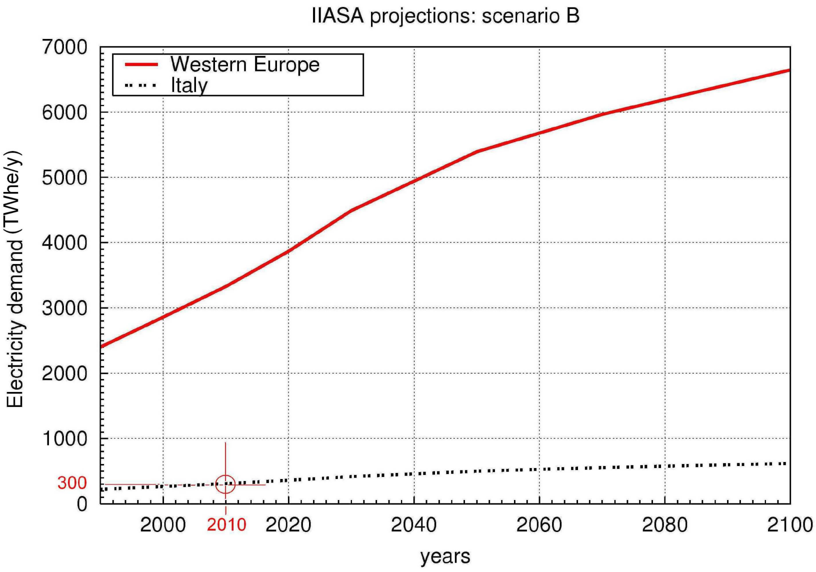

Figure 4.17: Electricity projection for Western Europe: Scenario B - IIASA [23]

Electricity production, TWhe/y						
	2003	2004	2005	2006	2007	2008
EU27	3,216,146	3,289,225	3,310,402	3,353,514	3,367,692	3,374,182
Italy	293,884	303,322	303,699	314,122	313,887	319,129
Share (%)	9.1	9.2	9.2	9.4	9.3	9.5

Table 4.10: EU27 and Italian electricity needs [38]

As shown in this figure, in 2100, the natural uranium consumption for the case C is about 2.5 times the reference value and it becomes equal to 3.2 times in 2200.

A similar trend can be noticed for the cumulative spent fuel as indicated in Table 4.11.

For the higher nuclear energy demand case (case C) the request of resources in 2100 corresponds to 0.75% of the total world resources including phosphates [57], a value considered to be negligible regarding its possible impact on the total resources availability.

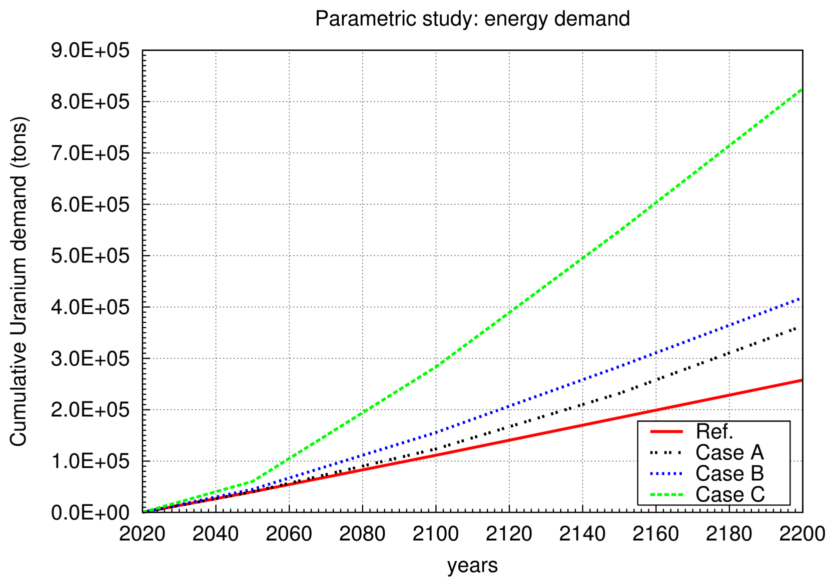

Figure 4.18: Cumulative natural uranium demand for various increasing nuclear energy demands

In order to cope with these increasing nuclear energy demands, the number of reactors and therefore the capacity of the fuel cycle facilities (e.g. fabrication and enrichment) is much higher than in the reference case.

For case A, the number of systems needed to cover the energy demand in 2200 is about 13 EPR-like systems (considering as unit of measure the systems modeled with the data in Table 4.1) where ca. 7 systems are gradually introduced in the period 2050-2200[13]. For case B, the number of systems needed to cover the energy demand in 2200 is about 12-13 EPR-like systems but they are introduced in a different way. The complete fleet is introduced before 2100 and they operate for about 100 years.

In fact, the results in SF accumulation and uranium consumption show differences between case A and B even though the number of systems in 2200 is the same.

For case C, the number of systems needed to cover the nuclear energy demand in 2100 is about 22 EPR-like systems that become 25 in 2200. Also in this case the systems are introduced mainly before 2100.

The increasing nuclear energy demand influences also the Pu available in the fuel cycle and then the transition to a FRs fleet. No detailed results have been here summarized because for the parametric study it has been decided to refer to a constant nuclear energy demand. In fact, a constant nuclear energy demand could be better extrapolated to other studies than results related to an increasing nuclear energy share based on fairly arbitrary and uncertain assumptions.

[13]In the scenario calculations it is tacitly assumed that shut down plants will be replaced immediately, i.e. COSI does not take into account explicitly the reactor lifetime and it assumes continuous operation if not explicitly instructed otherwise.

Year	Ref.	Case A	Case B	Case C
Nat. U demand, tons				
2050	40,150	40,181	45,070	60,361
share (%)	-	0.1	12.3	50.3
2100	111,465	123,619	155,713	283,360
share (%)	-	10.9	39.7	154.2
2200	257,662	363,090	418,201	824,934
share (%)	-	40.9	62.3	220.2
SF produced, tons				
2050	3,312	3,312	3,589	4,450
share (%)	-	0.0	8.4	34.4
2100	10,873	11,874	14,898	26,724
share (%)	-	9.2	37.0	145.8
2200	26,373	36,669	42,572	83,810
share (%)	-	39.0	61.4	217.8

Table 4.11: Influence of the increasing nuclear energy demand

In order to give an example of the extrapolation of the reference study, in Figure 4.19 is represented the decomposition of the Case C energy demand assuming the reference case as "unit of measure".

According to Figure 4.19, the total energy demand can be subdivided by the introduction of three full (200 years scenario) reference case, value in agreement with the results of the scenario above indicated (e.g. natural uranium consumption 3.2 times in 2200, and number of systems 22 instead of 6.3 units).

As outcome of the energy demand analysis, the case with constant energy demand produced by 6 EPR-like systems (66.7 TWhe/y) has been selected for the following part of the analysis and used as reference.

An additional important parameter, related to the energy demand, is the introduction rate of the systems. This parameter represents the impact of a transition phase from one condition of constant demand to another one.

Starting from the 6 EPR-like systems (55 GWd/tHM ave. burn-up), two introduction rates have been compared: 1) introduction in 10 years (period 2020-2030); 2) introduction in 20 years (period 2020-2040).

The results show that the differences concerning e.g. uranium demand and SF amounts are negligible (ca. 2.8% between the two cases). As example the cumulative SF is represented in Figure 4.20.

Moreover, the introduction rate of the thermal systems can influence the Pu availability for the transition to fast reactors. Table 4.12 shows the Pu available over the rest of the century for the two introduction periods considered. With a fast introduction rate the Pu inventory is higher at the beginning but after ca. 50 years the introduction rate does not show any significative difference.

	2035	2040	2050	2080	2100	2200
tons						
2020-2030	12.4	21.3	39.2	92.9	128.7	312.2
2020-2040	6.3	12.7	30.1	83.8	119.6	303.1
%	-48.9	-40.5	-23.2	-9.8	-7.1	-2.9

Table 4.12: Availability of Italian Pu versus reactor introduction rate

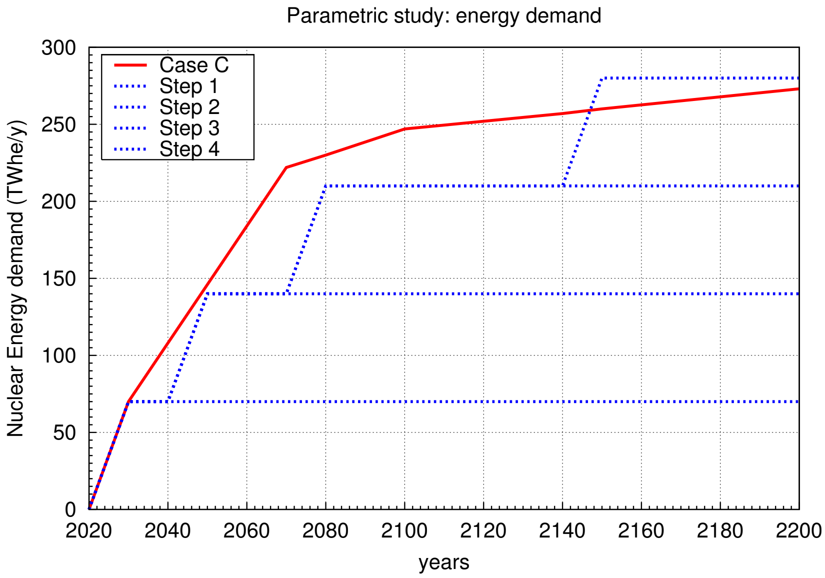

Figure 4.19: Example of superposition effects: the Case C and the decomposition according to the "reference case"

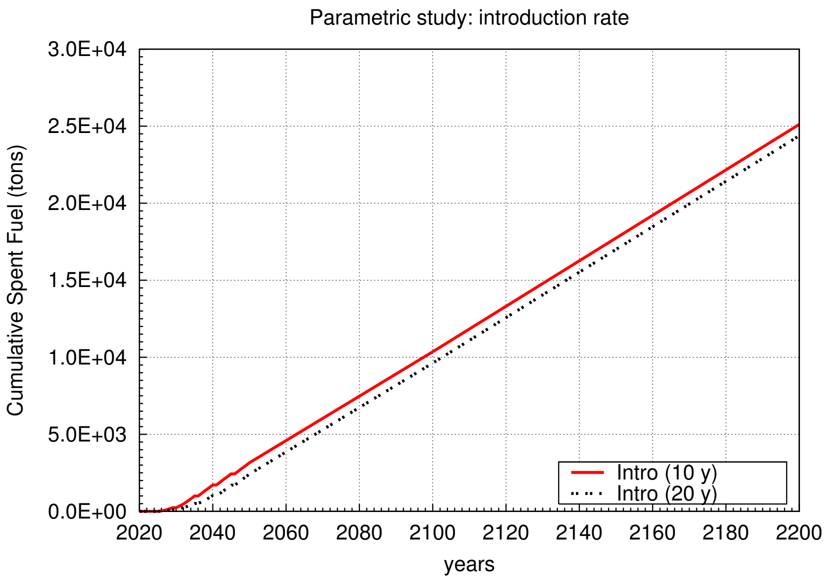

Figure 4.20: Cumulative Spent Fuel produced versus introduction rate (55 GWd/tHM case)

4.3.3 Influence of the reactor lifetime and start-up core

As already mentioned, different ways to model the reactors exist within the COSI6 code [20]. In particular, two main approaches are possible: 1) the model of the fleet by an "equivalent reactor", 2) the model of each reactor composing the fleet. The model of the start-up core can be added or not.

The results presented in the previous paragraphs have been obtained with the more common way ("equivalent reactor" without start-up core modeling). This model, indeed, provides, in a relatively simplified way, significant parameters with reasonable reliability for the scenario.

In order to evaluate the inaccuracies related to this approach, the analysis of the reference scenario (6 EPRs and constant energy demand up to 2200) has been repeated taking into account the contribution of the start-up core and the reactor lifetime.

The COSI6 code does not take into account explicitly the reactor lifetime as other codes do (e.g. TIRELIRE-STRATEGIE [63] or EVOLCODE [180]). In order to solve this point, a simplified model based on the energy demand provided by the user (see next paragraphs) has been set up and applied for the study.

Even in this analysis, two cases have been considered:

- 60 years reactor lifetime;

- 40 years reactor lifetime.

In both cases the "equivalent reactor" model (representative of the 6 EPRs) has been assumed as for the previous analyses.

Figure 4.21 shows how the reactor lifetime of 60 years has been modeled in COSI6 code. A zoom is shown in Figure 4.22 to better shown the substitution at end of life. The same substitution but considering 40 years reactor lifetime has also been studied.

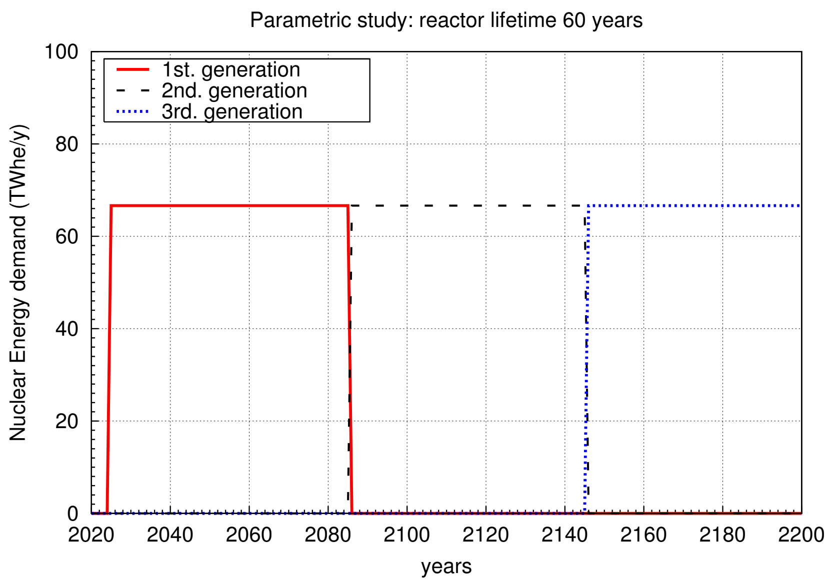

Figure 4.21: Nuclear Energy demand considering 60 years reactor lifetime

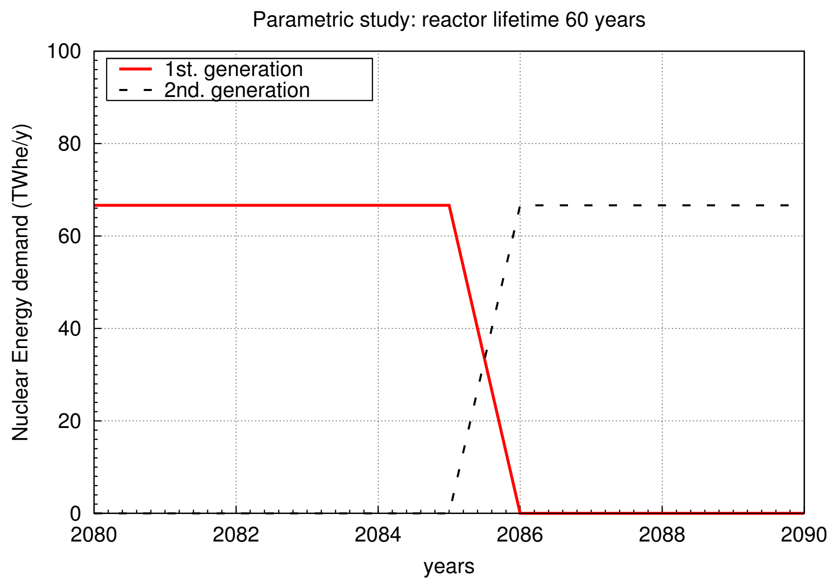

Figure 4.22: Nuclear Energy demand considering 60 years reactor lifetime: zoom to highlight the substitution

In addition, for each reactor, the core start-up (full mass of core) and the core discharged (full mass of core) have been considered, respectively, at Beginning of Life (BOL) and End of Life (EOL) for each system.

The refined model adopted can be seen in Figure 4.23, where the peaks of 720 tons (equivalent mass of 6 EPR full cores of 120 tons each) are represented. In Figure 4.24 it is clear that the two peaks (the shutdown peak of the first generation EPRs and start-up core of the second generation EPRs are overlapping). The same type of model has been adopted for the 40 years reactor lifetime.

In order to set-up this model some additional hypothesis have been added. In particular, the introduction of the 6 EPRs has been considered contemporaneously in 2025 instead of the introduction in 10 years (2020-2030), as also indicated in Figure 4.21.

Then, the start-up core for each system has been considered. In the COSI6 code (version 5.1.4) one feature for modeling the start-up core is available but the analysis of the results has shown some inconsistency. In fact, if the automatic feature is adopted, the start-up core is correctly simulated (in terms of masses and date) but not the shut-down core.

According to the implemented model[14], the batches composing the core (e.g. 4 batches) are discharged all at the same time (e.g. after 4 cycles) and not when the reactor has reached its real lifetime. Therefore, these batches are sent to the disposal almost 60 years earlier, i.e. before the expected date resulting in a wrong composition in the disposal (mainly for Pu241 and Am241).

In order to solve this point, an improved way for modeling the start-up core has been adopted. Assuming 4 batches reloading scheme, the seven different batches have been modeled in COSI6 code[15] (see Table

[14]Since no clear information are available in the COSI manual, other assumptions could also be possible.

[15]The approach adopted for modeling the start-up core does not carefully represent the reality but it can be considered a good approximation. In practice several strategy can be considered, i.e. batches with U235 enrichment content as in equilibrium but earlier unloaded or adoption of partially burnt SAs in new reactors.

4.13) and the loading list[16] has been modified in order to model the shut-down core at the correct date with the correct mass (as indicated in the Figure 4.23 around 2085).

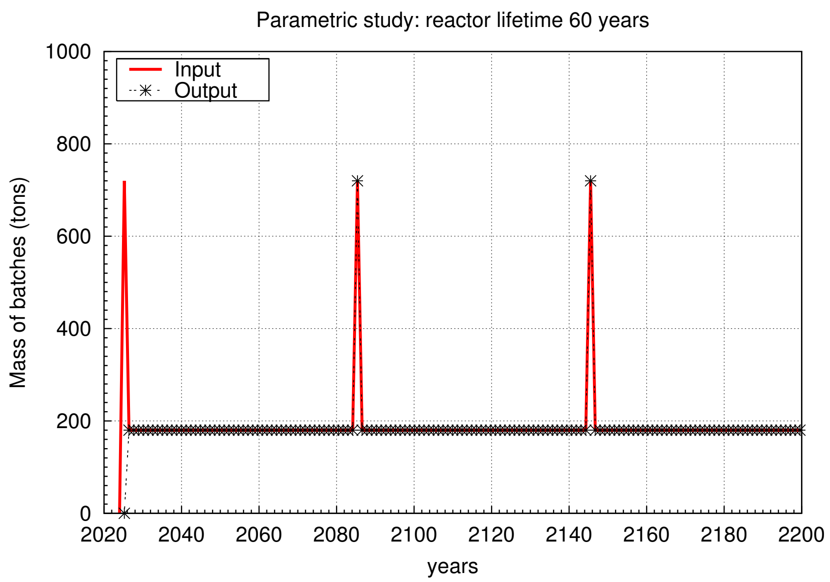

Figure 4.23: Mass of batches loaded and unloaded from the reactors (6 EPRs case)

Type of batch	Burn-up (GWd/tHM)	U235 enrichment [%]	Residence Time	For:
Batch EQ	55	4.6	4 cycles	Equilibrium
Batch ST1	3/4 (55)	4	3 cycles	Start-up
Batch ST2	1/2 (55)	3	2 cycles	Start-up
Batch ST3	1/4 (55)	2.1	1 cycle	Start-up
Batch SD1	3/4 (55)	4.6	3 cycles	Shut-down
Batch SD2	1/2 (55)	4.6	2 cycles	Shut-down
Batch SD3	1/4 (55)	4.6	1 cycle	Shut-down

Table 4.13: Batches considered for properly modeling the start-up and shut-down core

The influence of the start-up cores and the reactor lifetime on main parameters of the scenario is not so pronounced as indicated in Table 4.14.

The influence on the cumulative uranium demand (in 2200) has been evaluated to be of the order of ca. 6% for a lifetime of 60 years and ca. 10% for 40 years, as indicated also in Figure 4.25.

This small difference could be neglected for a scenario oriented to a single country (the error made is less important of the uncertainty on the nuclear energy demand) but it becomes important for global scenarios dealing with the uranium availability and very high growing nuclear energy demand (10% at global level corresponds to ca. 1,600,000 tons of natural uranium [4]).

[16]The loading list is a list containing the characteristics of fresh batches loaded every cycle in the reactor for all the period of operation. Each batch is characterized by the mass, the residence time and the type of the batch

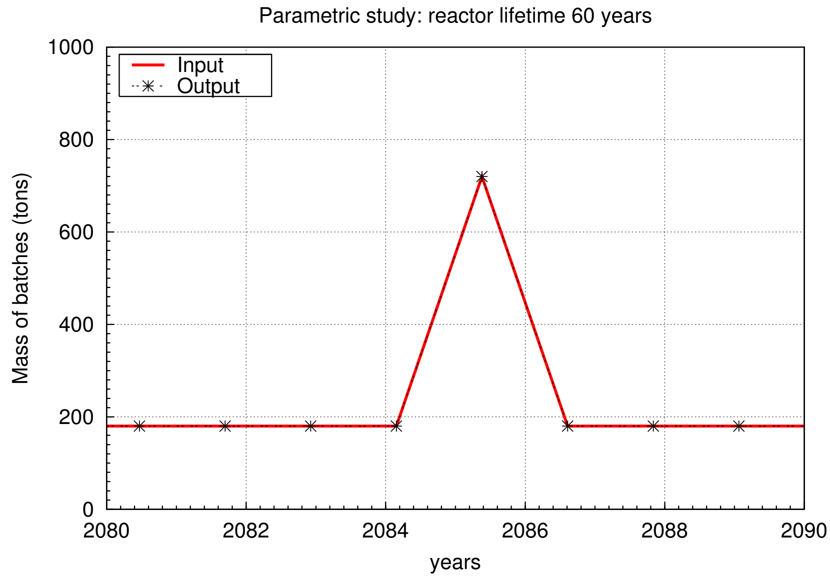

Figure 4.24: Mass of batches loaded and unloaded from the reactors (6 EPRs case): zoom to see the peaks

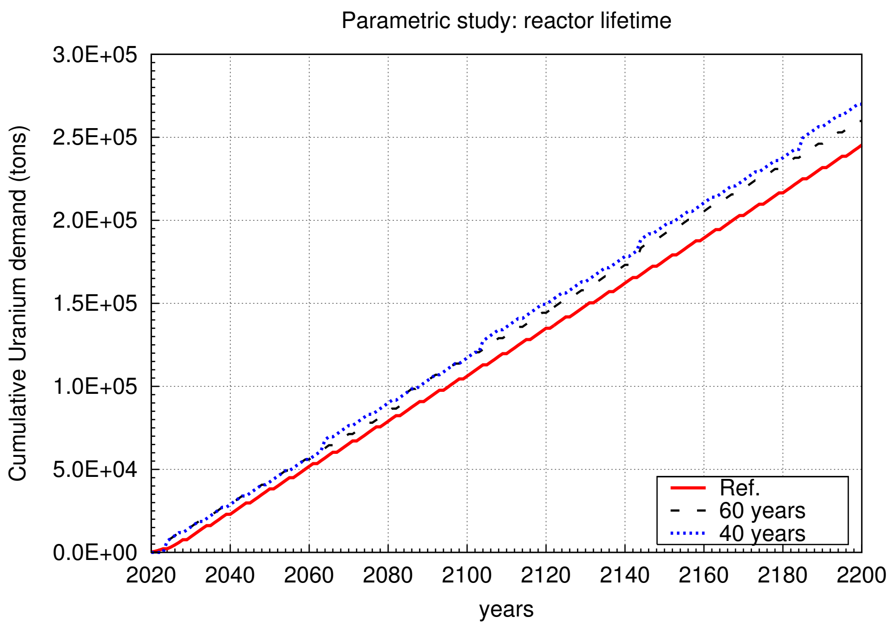

Figure 4.25: Cumulative natural uranium demand versus reactor lifetime [refined model adopted]

The reactor lifetime affects also the Pu availability in the cycle (see Table 4.15). The difference in 2080 is roughly 6 tons more in the case of 40 years lifetime and 2 tons more in the case of 60 years lifetime (as also indicated by Figure 4.26). Differences that are limited if compared with the Pu needed for the 100% fleet substitution with FRs (ca. 80-100 tons of Pu depending on the model considered, see Chapter 5 for details).

No significant influences on the radiotoxicity and the heat load behaviors can be seen. An example is indicated in Figure 4.27. Concerning the fabrication and enrichment capacities, the effect is limited, as indicated in Table 4.16.

Those values (i.e. the availability of Pu in the cycle) have relatively low impact on the fast reactor development and, as a first approximation, can be neglected.

	6 EPRs		
		Refined Model	
	Ref.	60 years	40 years
Nat. U demand, tons			
2100	106,137	117,138	117,138
share (%)	Ref.	10.4	10.4
2200	245,300	259,700	269,900
share (%)	Ref.	5.9	10.0
SF produced, tons			
2100	10,353	11,519	11,519
share (%)	Ref.	11.3	11.3
2200	25,100	26,800	27,900
%	Ref.	6.8	11.1

Table 4.14: Natural U and SF mass for different reactor lifetimes [refined model adopted]

	2050	2080	2100	2200
	tons			
6 EPRs	41.2	97.6	135.2	327.9
Refined Model				
6 EPRs (60 years)	43.4	99.4	139.3	326.8
6 EPRs (40 years)	43.4	103.5	139.3	335.0

Table 4.15: Pu availability during the scenario versus reactor lifetime [refined model adopted]

Additionally, the separate model for each reactor composing the fleet has been set up. In Figure 4.28 the introduction strategy considered for each system is shown (1 reactor every 2 years). Assuming 60 years lifetime, the uranium cumulative resources used in 2200 becoming 260,500 tons instead of 259,700 tons, the cumulative SF 25,112 tons instead of 24,379 tons. The fabrication and enrichment capacities are not changed as well as the back-end parameters (e.g. heat load evolution trend of Figure 4.29).

The results are similar when comparing them with those determined with the "equivalent reactor" model, therefore the adoption of the "equivalent reactor" has been validated and used also for the second part of this study.

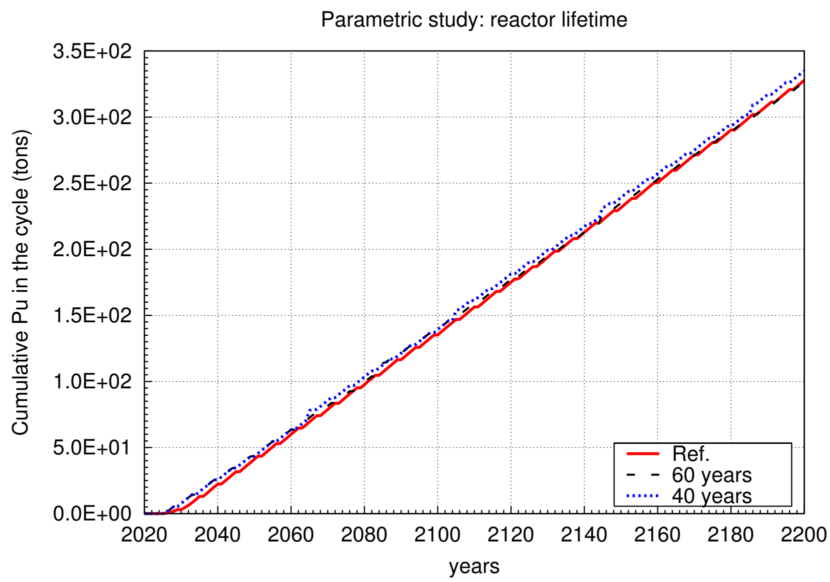

Figure 4.26: Cumulative plutonium in the cycle versus reactor lifetime [refined model adopted]

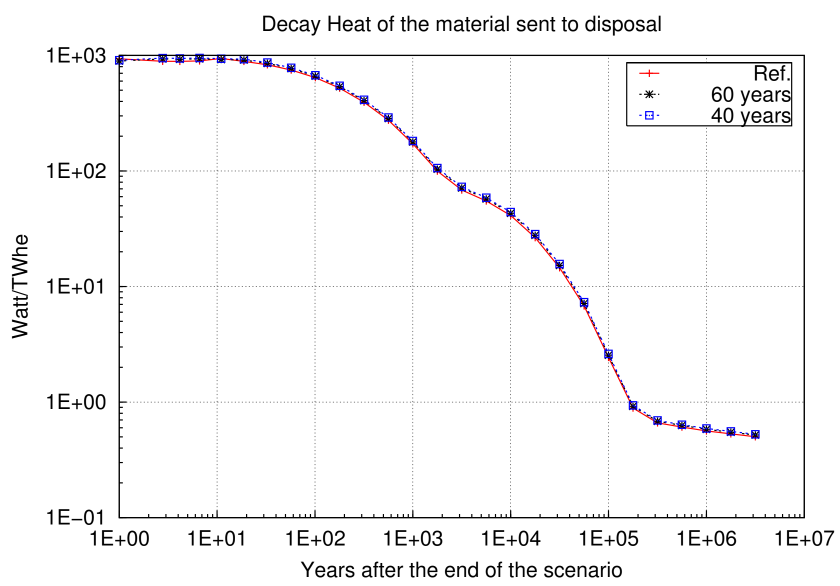

Figure 4.27: Specific decay heat (ingestion) evolution: adoption of reference model [2200 is fixed as t=0]

	6 EPRs		
		Refined Model	
	Ref.	60 years	40 years
Fabrication capacity, tons			
2050	3,873	4,500	4,500
share (%)	-	16.2	16.2
2100	11,253	12,419	12,419
share (%)	-	10.4	10.4
2200	25,832	27,539	28,618
share (%)	-	6.6	10.8
Enrichment capacity, tons			
2050	38,231	42,441	42,441
share (%)	-	11.0	11.0
2100	106,137	117,138	117,138
share (%)	-	10.4	10.4
2200	245,345	259,741	269,927
share (%)	-	5.9	10.0

Table 4.16: Fabrication and enrichment cumulative capacities versus reactor lifetime [refined model adopted]

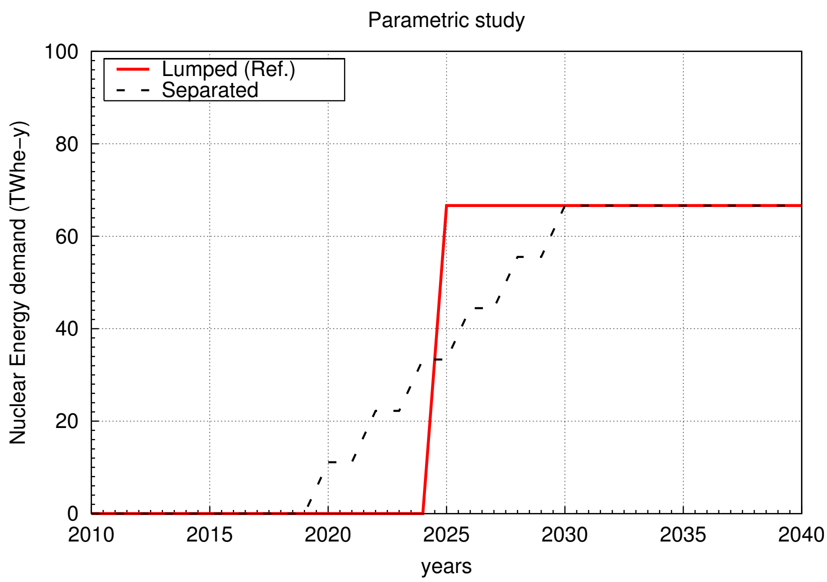

Figure 4.28: Nuclear Energy demand considering reactor-by-reactor model

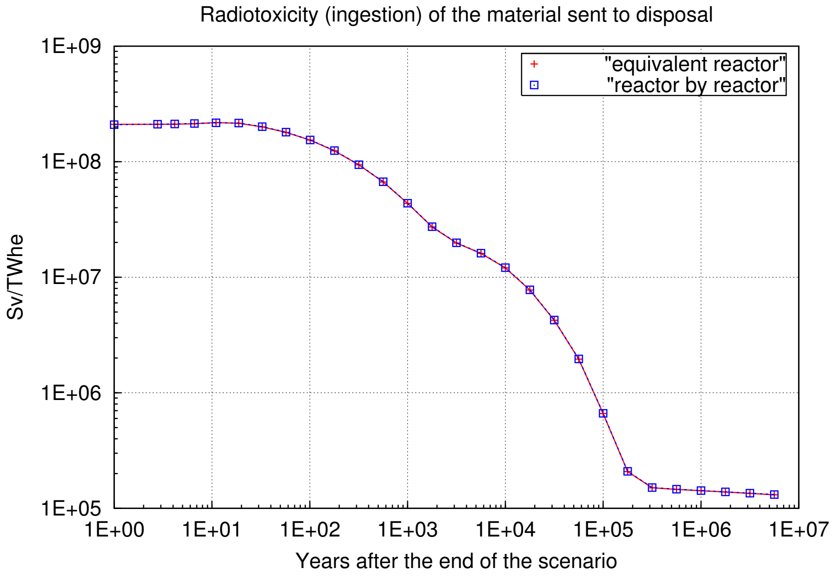

Figure 4.29: Specific radiotoxicity (ingestion) evolution: comparison COSI6 models adopted [2200 is fixed as t=0]

4.3.4 Summary of the parametric study

The impact of the investigated parameters has been quantified with respect to the selected reference case. In general, except for the nuclear energy demand, their influence is rather limited. However, as indicated in the previous section, the effect of some parameters becomes important in a more complex scenario, e.g. regional or global scenarios.

The parametric study concerning the "once-through" case has been summarized, in a qualitative way, in Table 4.17.

Indicators	Parameters			
	Burn-up	Nucl. Energy demand	Reactor Intro. rate	Reactor Lifetime
Uranium Mass	Medium	High-Medium	Low	Low
Plutonium Mass	Medium	High-Medium	Medium	Low
Spent Fuel Inventory	Medium	High-Medium	Low	Low
Radiotoxicity	Low	Low	Low	Low
Heat Load	Low	Low	Low	Low
Infrastructures needs	Medium	High-Medium	Low	Low

Table 4.17: Impact of each parameter over the indicators selected [High: impact considerably the trends; Medium: impact not drastically the trends; Low: impact negligibly the trends]

An indication about the level of the influence (low, medium, high) regarding the indicators chosen is reported too. This general overview can provide an indication of the most dominating and most important hypotheses to be studied more in detail for defining the boundary conditions of the scenario as well as it helps in defining which hypotheses can be neglected.

4.4 Summary

In the present Chapter the hypothesis of a nuclear energy development based on LWRs deployment has been analyzed in detail.

A reference case, suitable for future extrapolation at regional scale, has been defined.

The activity has been focused on the investigation of the implication on selected indicators which are natural uranium requirements, industrial infrastructure needs, waste inventory, radiotoxicity and heat load evolution in a repository.

In order to quantify the impact of the boundary conditions, a parametric study has been developed.

Selected parameters (defining the boundary conditions) have been separately investigated and quantified with respect to the reference case (70 TWhe/y constant nuclear energy demand and "once-through" strategy).

The outcome of the parametric study is a kind of "database" containing the impact on the selected indicators (U and Pu mass flows, SF inventory, radiotoxicity and heat load) of each parameter considered (i.e. nuclear energy demand, reactor introduction rate, core burn-up, etc.).

These data can be used in order to take into account the uncertainties associated to the chosen hypotheses.

Moreover, the study has provided also an indication of the most critical boundary conditions (above all the nuclear energy demand) for the definition of a fuel cycle scenario.

Chapter 5

Transition from Thermal to Fast Reactors

The parametric study concerning the nuclear energy development by LWRs deployment presented in Chapter 4 has shown how quantities like the utilization of uranium resources and the waste produced can be modified by the hypotheses and constraints considered.

The effect becomes much important if the fuel cycle strategy is changed from the actual "once-through" case to the adoption of advanced fuel cycles based on Partitioning & Transmutation (P&T).

In the present Chapter, the analysis of the transition scenarios from an LWRs based fleet to an FRs based one is described. The activity has been developed highlighting, case by case, the most critical parameters affecting the sustainability indicators selected for the study (resources, waste inventory, radiotoxicity and heat load).

The period of interest remains unchanged: 2020-2200. This period, indeed, was chosen in order to allow the full transition toward a fast reactor based fleet (if possible according to the system considered) and to point out the advantages and possible disadvantages, difficulties and drawbacks of the selected strategy under equilibrium conditions.

The reference scenario remains practically unchanged (see Par. 4.2 for details). The only difference is the constant value assumed for the nuclear energy demand. In fact, the energy produced by 6 EPR units (55 GWd/tHM average burn-up) equal to 66.7 TWhe/y has been adopted instead of 70 TWhe/y initially considered (corresponding to ca. 6.3 units). As shown in Par. 4.3.2, the differences between the two cases are negligible. The data adopted for the EPRs model are listed in Table 4.1. The "equivalent reactor" approach has been used also in this part of the activity.

In order to define the "transition scenarios", additional constraints have been fixed in terms of Pu availability and FRs start-up. The transition has been limited by the Pu available in the cycle, produced only by the thermal reactor fleet operated in the so called Italian scenario since 2030. Under this assumption, the breeding characteristics and the power density of the FRs considered play the central role in the dynamics of the transition.

This constraint has been fixed for several reasons. First of all, it has been assumed that at the time horizon in which FRs will be inserted in the scenario (toward the end of the century), the technology is mature. Moreover, it has assumed that no "spar" plutonium resource is available.

For the present activity, the FRs considered are all self-sustaining or slightly breeder systems, as described in Appendix D. This choice depends on observed technological trends in e.g. Europe [4, 84, 58].

In fact, the expected low energy demand increase (justified by the expected low population increase and by the already achieved level of industrialization) does not justify or even urge the adoption of strong breeders at the moment.

Suitable COSI6 libraries (including one group cross sections and fluxes; burn-up dependent) have been generated for the reactor systems considered in the study. The 3D core models and burn-up evolution have been treated by means of the ERANOS code [182] in agreement with the procedure described in Par. 3.4 and in Appendix D.

For the study, three fast reactor concepts have been considered:

- European Lead-cooled System: so-called ELSY-like system [74, 75, 188]. The ELSY system is a medium size (600 MWe) lead cooled critical fast reactor [68]. The model considered in the study is the HEX-Z model developed at SCK-CEN [189] characterized by three core zones with different Pu content (14.6, 15.4, 17.3 at% Pu/HM). The fuel residence time is ca. 5 years with 4 batches reloading scheme.

- European Sodium Fast Reactor: so-called ESFR-like systems [58, 31, 30]. The ESFR system is a large (1440 MWe) sodium cooled fast reactor [58]. The oxide fuel configuration has been considered in the study. The core is composed of two zones with different Pu content (14.43 and 16.78 at.%) in order to achieve a rather flat power profile. The total residence time is 2050 efpd with 5 batches reloading scheme.

- European Fast reactor: so-called EFR-like systems [103, 190, 191]. It is a large (1500 MWe) sodium cooled fast reactor. The applied model is a HEX-Z core, with three enrichment zones. In order to reach a near zero BG, axial and radial blanket zones of depleted uranium oxide have been considered. The fuel residence time is ca. 4.5 years with 5 batches reloading scheme. The same time and reloading strategy is considered for the axial blankets. For the radial blanket the total residence time is ca. 9 years.

In particular, the ELSY and ESFR neutronic models have been assessed for the Ph.D. purpose adopting the ERANOS2.2 neutronics code and JEFF3.1 data libraries [182, 192]. The relative BBL libraries for COSI6 code have been generated on the basis of the ERANOS(DARWIN)-APOGENE-COSI chain [183].

On the contrary, the EFR library for COSI6 simulations has been generated at CEA. This library has been applied in other studies (e.g. [4, 193, 150]). Some details about the EFR model can be found in [103].

For all the cases, the earliest introduction date of fast reactors is fixed at 2080, i.e. after an assumed LWR reactor lifetime[1]. A partially closed fuel cycle where only Pu is multi-recycled in FRs has been considered. The fuel cycle scheme is indicated in Figure 5.1.

5.1 Introduction of Fast Reactor

As indicated previously, for the transition to FRs only the Pu available in the cycle has been considered.

The Pu produced by the LWR fleet, the breeding characteristics, and the power density of the fast systems are the main factors for the transition strategy in a country or region that wants to develop nuclear energy in isolation as elucidated by the results of the transition scenarios performed (see Par. 5.1.1).

In the study, the spent fuel cooling time (for both thermal and fast reactor fleet) before reprocessing has been maintained fixed: 5 years for LWRs SF and 2 years for FRs SF.

Fabrication and reprocessing times have been kept unchanged irrespective of the considered scenarios and equal to 0.5 years each.

[1] This date is essentially a consequence of the shut-down and substitution of the first LWR NPP; it does not necessarily mean that the technology of metal cooled reactors and FR fuel fabrication would not already be available at an earlier time.

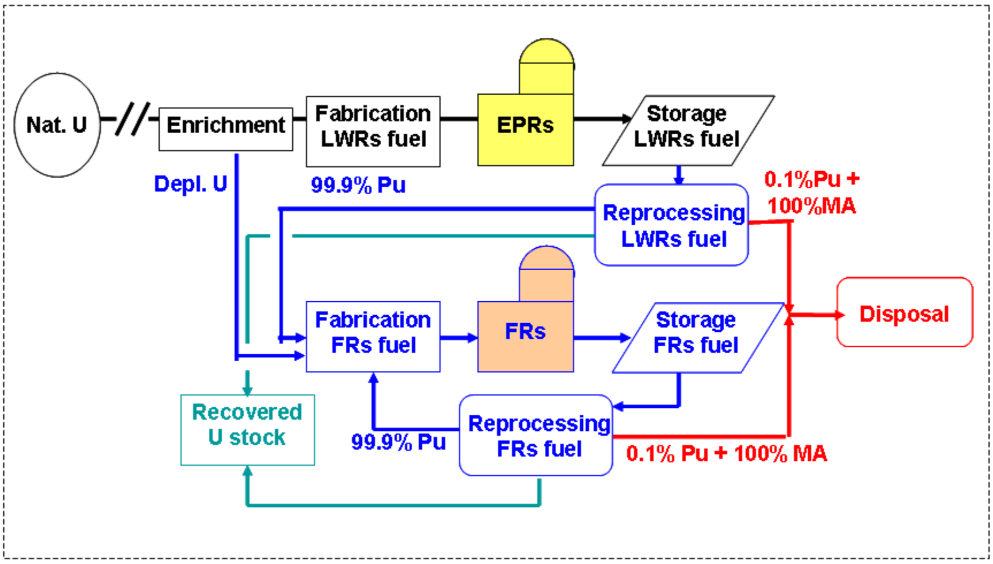

Figure 5.1: A simplified flow scheme for the reference scenarios: the partially closed fuel cycle

These hypotheses have not been changed in order to better highlight the effects of other parameters (e.g. breeding characteristics). However, it is well known that these parameters can strongly affect the dynamics of the transition (influencing the amount of Pu in the cycle) as indicated also by [55, 84, 4].

The main characteristics of the FRs considered are listed in Table 5.1. The data involved in the transition evolution (early start-up, shares, etc.) are specified case by case in accordance with the obtained results.

	ELSY	ESFR	EFR
MWth	1500	3600	3625
MWe	600	1440	1450
Coolant	Lead	Sodium	Sodium
Spectrum	Fast	Fast	Fast
Load factor (%)	85	85	85
efpd	4*456	5*410	5*340
Efficiency (%)	40	40	40
Burn-up (GWd/tHM)	60	100	136
HM mass (tons)	ca.50	ca.75	ca.45
Power density (kgHM/MWe)	83.3	52.1	31.0
Fuel Volume Fraction (%)	30.7	52.4	43.7
Fissile content (%)	16.4	15.6	21.5
BR	ca.1.01	ca.1.03	ca.1.02
Fuel (geometry)	MOX(HEX,z)	MOX(HEX,z)	MOX(HEX,z)

Table 5.1: Main characteristics of Fast Reactors

The data listed in Table 5.1 are directly used as the input in COSI6 code. The other needed data, as one group equivalent burn-up dependent cross-sections are provided by the libraries generated. The description of the neutronic models for obtaining the suitable COSI6 libraries is included in Appendix D.

5.1.1 Impact of the adoption of different FR systems

As first comparison, a fleet based on ELSY, ESFR or EFR systems has been developed to replace, starting from 2080, the thermal EPR based fleet.

In order to verify if the 100% LWRs replacement is feasible, the amount of Pu in the cycle has been calculated.

In Table 5.2, the annual energy and electricity produced by each unit (assuming 85% constant load factor; 40% thermal efficiency), the Pu content in each core, the total Pu needed in order to allow 100% replacement of an LWR fleet, are indicated. The Pu content considered for each system is in agreement with the average values available in literature [8].

Comparing the total amount of Pu in the cycle (97 tons[2]) with the Pu needed for 100% FR-based fleet (Table 5.2), it is clear that in the case of ELSY-like systems, the complete substitution can not be achieved by a single step.

A full substitution with ESFR and EFR systems in a single step starting from 2080 seems feasible.

Moreover, when practical Pu needs are taken into account (e.g. Pu needed for the batch reloading in the previous years), additional lack of material may prevent the complete transition to a fast reactor fleet (in one step). Even though the systems consist of self-sustaining reactors, it can happen that, due to cooling time of SF before reprocessing, there will not be available enough Pu for reloading the core (as pointed out for the ESFR).

	ELSY	ESFR	EFR
Electric power (MWe)	600	1440	1450
Load factor (%)	85%		
Electricity per year (TWhe/y)	66.7		
Electricity per unit (TWhe/y)	4.47	10.73	10.80
N. of reactors needed by the fleet	14.9	6.2	6.2
Pu in single core (tons)	8.5	11.5	11.1
Total Pu needed by the fleet (tons)	126.6	71.7	68.3
Pu available in 2080 (tons)	97		
replacement	×	⊕	✓

Table 5.2: Reactors Characteristics: Pu needed by the systems [× = full replacement not possible, ⊕ = full replacement possible with some delay, ✓= full replacement possible]

By the adoption of medium size lead-cooled systems only ca. 40% of the nuclear energy demand can be covered by FRs in 2080 (corresponding to 26.82 TWhe/y) and the difference to the total remains to be covered by thermal reactors (ca. 40 TWhe/y).

Whereas by employing ESFR systems, this share increases to 76% (equal to 50.64 TWhe/y) and it becomes equal to 100% by the adoption of EFR reactors.

In order to maximize the FRs shares for the scenarios based on ELSY and ESFR systems, a second introduction step has been considered based on the available Pu in the cycle produced by the remaining LWRs fleet. By taking into account this further Pu availability, a second step can become feasible at around 2130-2140.

As outcome, the replacement strategies considered for the three fast systems are summarized in Figure 5.2. In Figure 5.2, only the total nuclear energy demand and the fraction covered by FRs are represented.

Adopting ESFR-like systems the complete substitution is achievable before 2150, while for the case of

[2] It refers to the 6 EPRs (66.7 TWhe/y) case as indicated in Par. 4.3.2.

ELSY-like systems the complete substitution is not possible before the end of the next century, therefore, an optimization of the model, e.g. including fertile blankets, should be adopted[3].

The transitions depicted in Figure 5.2 include additional hypotheses. For the ESFR case, the complete substitution of the thermal fleet has been considered to occur beyond the 40 years lifetime of LWRs. For the ELSY case, a delay of 10 years with respect to ESFR case has been considered in order to maximize the share in 2200.

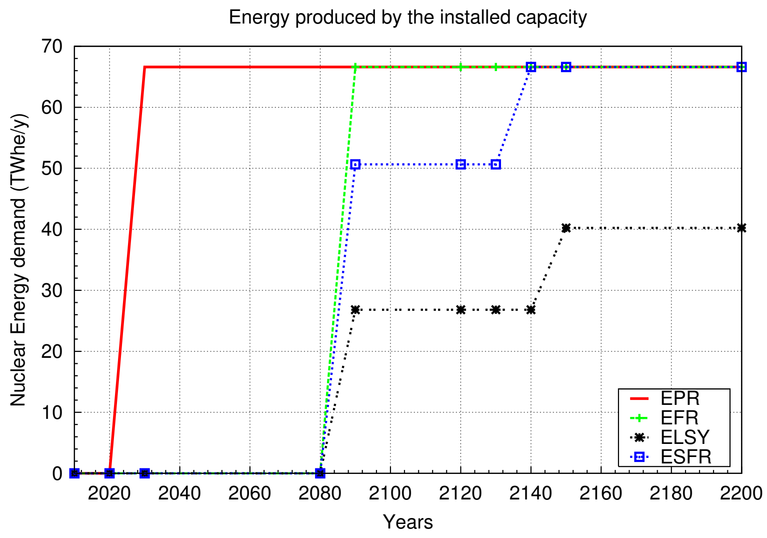

Figure 5.2: Nuclear Energy demand produced by the three FR types according to the Pu available in the cycle

An example about the maximization of the use of Pu for FRs start-up can be seen analyzing the Pu mass balance in the stock of Pu after reprocessing. In this stock, the input material is the mass of Pu from reprocessing and the output material is the mass of Pu requested by the fuel fabrication plant. An example of Pu stock for the ESFR case (energy demand shown in Figure 5.2) is reported in Figure 5.3. In this Figure, the minimums (at around 2080 and 2130) represent the mass of Pu in the stock at the time in which the two steps of introduction of ESFR systems are considered. The Pu in the stock goes to zero, therefore no Pu remains available for developing a higher fraction of FRs, i.e. the maximum possible substitution (due to the Pu available in the cycle) has been achieved. The small peaks indicated in Figure 5.3 are due to the discrete nature of the COSI6 model. An explanation of them is included in Appendix A.

In order to assess the transition strategy indicated in Figure 5.2, with the aim to maximize the FRs shares, an optimization process aiming at the most favorable use of the available Pu has been performed for each system. Here, the final results are summarized.

The contribution to natural uranium saving coming from the introduction of FRs depends on the share of FRs in the fleet, and on the time in which the FRs can cover 100% of the total[4] (impacting the cumulative values). Therefore, the uranium saving depends on the characteristics of the systems themselves.

[3]The COSI6 "equivalent reactor" approach has been adopted also for transition scenario studies. Hence, the curves represented in Figure 5.2 have been defined taking into account the Pu available in the cycle for maximizing the FR share. Therefore, the values do not always correspond to an "integer" number of FRs units.

[4]The needed uranium for FR MOX fuel fabrication comes from depleted uranium stock (see also scheme in Figure 5.1). Hence, no natural uranium is mined for sustaining the 100% FR-based fleet.

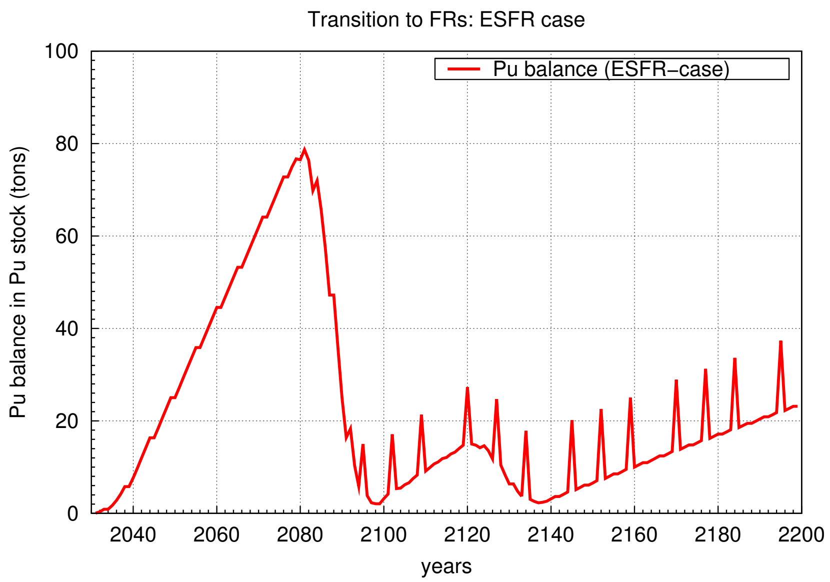

Figure 5.3: Pu mass balance in Pu stock (ESFR-case)

In order to highlight the advantages of the FRs introduction, the comparison with the "once-through" case (based on 6 EPRs, 55 GWd/tHM burn-up) is presented in Figure 5.4.

As indicated in Figure 5.4, the uranium resources saved by the introduction of FRs is not negligible: in 2150 (i.e. at the time horizon when a possible resources shortage could be expected [4, 55, 33]) the adoption of an FR-based fleet (ELSY-like reactors) allows a saving of 22.4% (fleet composed by ca. 50% EPRs and ca. 50% FRs) of the uranium resources needed.

This fraction becomes higher if ESFR and EFR systems are adopted and the 100% FRs fleet is reached (respectively, 43.3% and 52.8%, values in agreement with other studies, e.g. [137]). More details are indicated in Table 5.3 where the contribution on uranium saving is shown for several points in time during the scenario.

	EPR-based scenario	ELSY-based scenario	ESFR-based scenario	EFR-based scenario
year	Cumulative Natural Uranium, 1000 tons [U saved, %]			
2040	23.0 [-]	23.0 [-]	23.0 [-]	23.0 [-]
2080	79.0 [-]	78.9 [0.2]	78.7 [0.3]	78.7 [0.4]
2090	92.6 [-]	88.7 [4.2]	85.3 [7.9]	83.0 [10.4]
2100	106.1 [-]	96.8 [8.8]	88.5 [16.6]	83.0 [21.8]
2150	175.7 [-]	136.4 [22.4]	99.6 [43.3]	83.0 [52.8]
2200	245.3 [-]	164.0 [33.1]	99.6 [59.4]	83.0 [66.2]

Table 5.3: Influence of the FRs introduction on uranium resources

For helping in the extrapolation to other studies, the uranium saving due to the maximum introduction of FRs by a single step in 2080, has also been analyzed. The comparison is indicated in Table 5.4. In the same table (second column) the share of nuclear energy demand covered by FRs is indicated, too.

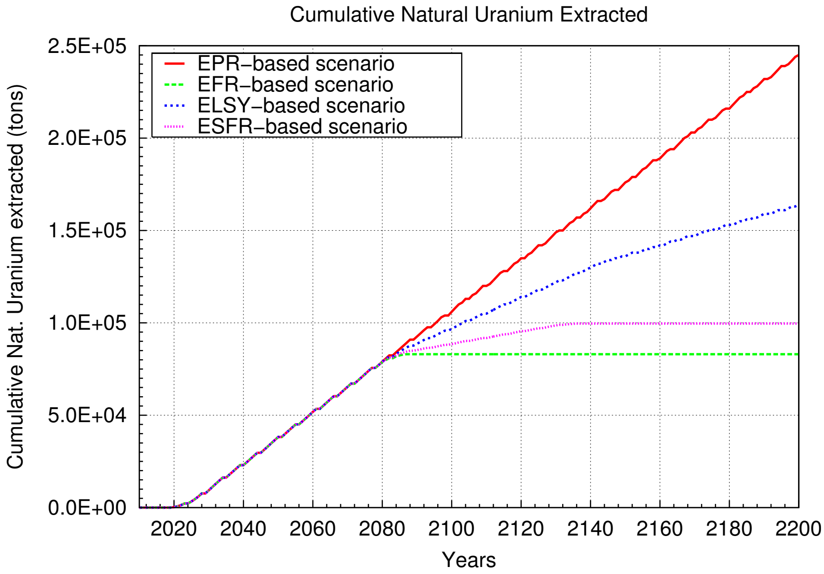

Figure 5.4: Natural U demand: influence of FRs introduction

	EPR-based scenario	ELSY-based scenario	ESFR-based scenario	EFR-based scenario
	Share of electricity cover by FRs (%)			
	Ref.	40	76	100
year	Cumulative Natural Uranium, 1000 tons [U saved, %]			
2040	23.0 [-]	23.0 [-]	23.0 [-]	23.0 [-]
2080	79.0 [-]	78.9 [0.2]	78.7 [0.3]	78.7 [0.4]
2090	92.6 [-]	88.7 [4.2]	85.3 [7.9]	83.0 [10.4]
2100	106.1 [-]	96.8 [8.8]	88.5 [16.6]	83.0 [21.8]
2150	175.7 [-]	138.4 [21.2]	105.3 [40.1]	83.0 [52.8]
2200	245.3 [-]	180.0 [26.6]	122.0 [50.3]	83.0 [66.2]

Table 5.4: Natural uranium saving assuming a single step of FRs introduction in 2080

Looking at the masses involved, the influence of FRs introduction is not so significant as absolute values for a local area because the resources involved also for the "once-through" scenario were assessed to be less than 0.3% of the total available uranium resources (value in 2100). However, if the results are extrapolated to other situations, e.g. extended local area or global scenarios, the impact of resources becomes very important for the long-term security of supply limiting the stress on the uranium market (e.g. maintaining the resources below the level of conventional resources [4]). In addition, also for a small country the transition toward a FRs fleet can be favorable for the long-term security of supply.

The adoption of Pu multi-recycling in FRs (in partially-closed fuel cycle, Figure 5.1) provides advantages also with respect to the fuel cycle back-end. In fact, only MAs (Np, Am, and Cm separated in the reprocessing plant) and 0.1% of Pu reprocessing losses are sent to the repository for the final disposal. This option limits considerably the inventory of waste to be treated and, therefore, it improves the repository capacity. In addition a reduction of the radiotoxicity and heat load associated to the material sent to the repository is achieved.

The MAs cumulative values sent to the repository are affected by the Pu multi-recycling strategy, too.

A reduction of about 30% in 2200 with respect to the "once-through" case can be obtained as indicated in Figure 5.5.

In Figure 5.5, all MAs (Np, Am, and Cm) cumulative mass sent to the repository are taken into account. The relative contribution to the total of each element is indicated in Table 5.5[5]. The quantity of Np to be stored in disposal is not negligible in particular from the proliferation resistance point of view (see e.g. [186]).

In the study, for all cases considered, the reprocessing start-up date has been fixed to 2078 (close to FRs early introduction date), as also visible by the step in Figure 5.5 around 2078. Up to this data, the SF discharged from LWRs is sent to an interim storage waiting for the final solution (reprocessing or disposal).

In 2078, the reprocessing of the oldest SF starts in order to provide enough Pu for the FRs fresh fuel fabrication. In the reprocessing plant, the separation of U, Pu and MAs is performed. Pu is sent to FRs fuel fabrication plant, U to an intermediate recovered reprocessed U storage[6] and MAs to the repository for the final disposal.

This can also be seen analyzing Table 5.5 where the MAs sent to disposal in the case of FRs based fleet remain equal to 0 up to 2078, i.e. up to this data the SF remain in the cycle without any separation. Different options concerning the start-up date for the reprocessing plants have been analyzed as indicated in Par. 5.2.3.

Even though the share of MAs produced in FRs (ca. 0.30% of the total HM discharged) is higher than in LWRs (ca. 0.14%), the cumulative MAs amount remains lower. In fact, in FRs based scenarios, Pu is separated from the MAs stream and remains in the cycle. Therefore, the contribution to the total MAs amount in disposal coming from the Pu241 decay into Am241 (ca. 55% as indicated in Par. 4.2) is not included.

The amount of Pu in the cycle has been analyzed, too as indicated in Figure 5.6.

Figure 5.6 shows that all fast systems perform in a comparable manner: a 45% inventory reduction in 2200 is achieved in the case of EFR-like scenarios, consistently with other similar studies (e.g. [137]). After 2140, the difference between the EFR and ESFR cases is reduced due to the second step of introduction for ESFR as indicated in Figure 5.2[7].

[5]For FR based scenarios, the MAs initially generated to the thermal reactor fleet are initially stored in a interim storage until the reprocessing plant start-up. Afterward, they are sent, together with the MAs produced by the FRs fleet to the repository. The values indicated in Table 5.5 include this aspect as well as the contribution of decay in disposal.

[6]In the study, the recovering of the U for the LWR fresh fuel fabrication has not be considered for simplifying the cycle. However, this aspect can be analyzed in further studies.

[7]This behavior was expected (same breeding characteristics and almost same power density) when compared under same assumption on nuclear energy demand, e.g. 100% of the fleet.

MAs content in the Repository: scenario comparison				
	EPR-based	ELSY-based	ESFR-based	EFR-based
2078				
tons				
TOTAL	16.79	0.00	0.00	0.00
%				
Am	61.3	-	-	-
Cm	2.7	-	-	-
Np	36.0	-	-	-
2150				
tons				
TOTAL	48.26	42.39	37.31	36.29
%				
Am	64.1	66.7	66.6	68.9
Cm	1.3	1.4	2.1	2.4
Np	34.6	31.9	31.3	28.7
2200				
tons				
TOTAL	70.41	62.46	50.44	47.34
%				
Am	63.1	69.0	69.1	70.2
Cm	1.0	1.0	1.5	1.6
Np	35.9	29.9	29.5	28.2

Table 5.5: Cumulative MAs and relative Np, Am, Cm content of the material sent to disposal

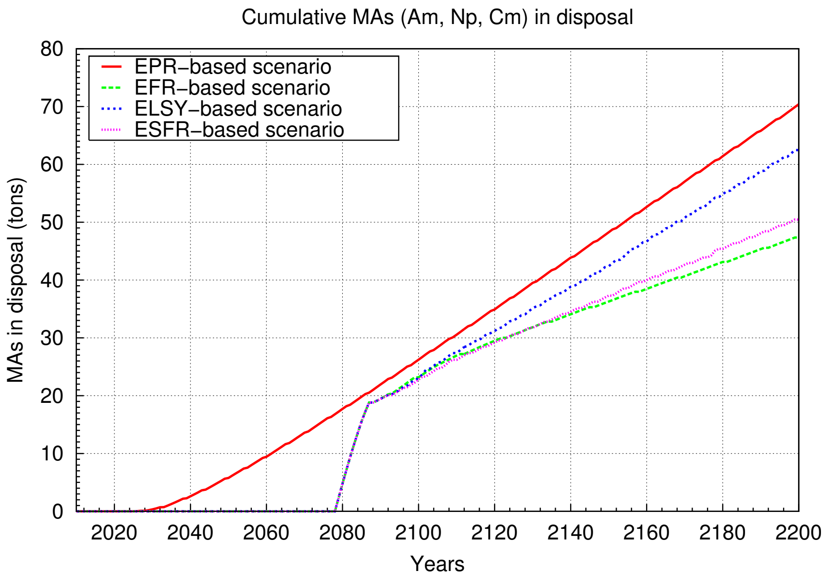

Figure 5.5: Cumulative MAs (Np, Am, Cm) sent to disposal

The reprocessing start-up date can slightly influence the Pu quality loaded in the systems, i.e. reducing the aging time of Pu stocks and therefore the build-up of Am241. However, the reprocessing options give a small effect on the results obtained and, hence, the trends remain unchanged. Also this point is described later on.

Other parameters representative of the fuel cycle back-end have been investigated, as the radiotoxicity and the heat load of the materials sent to disposal.

The three systems with the partially closed fuel cycle (only Pu multi-recycling in FRs) give comparable reduction of the radiotoxicity and heat load for the period 1,000-10,000 years where Pu isotopes are the main contributors.

In Figures 5.7 and 5.8 the radiotoxicity (evaluated on the basis of ICRP68 coefficients [39]) and the heat load of the material sent to disposal are compared with respect to the reference scenario ("once-through" strategy). The end of the scenario, 2200, is assumed as t = 0 for the disposal evolution (driven by decay only).

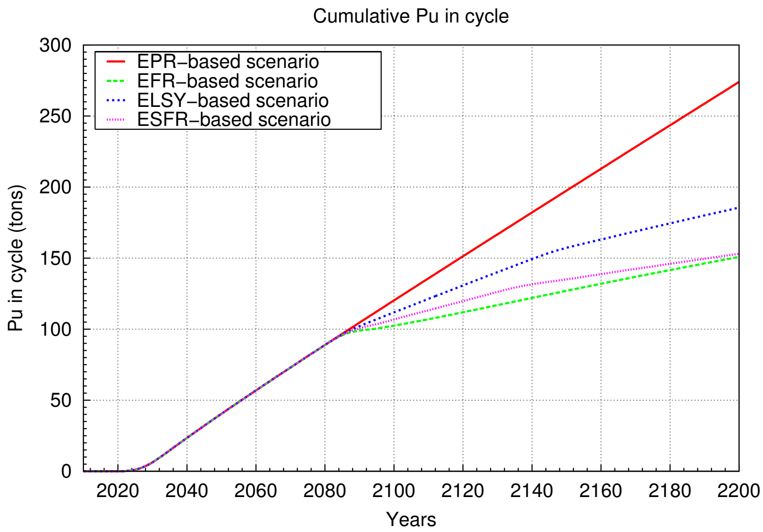

Figure 5.6: Cumulative Pu in the cycle

The contribution of Fission Products (FPs) has not been taken into account for the radiotoxicity and heat load curves. This depends on the option adopted in COSI6 code concerning the burn-up module.

In particular, two burn-up/depletion modules (namely CESAR4 and CESAR5) can be linked with the fuel cycle code. The main difference between the two modules is the treatment of FPs and the depletion chain adopted [187].

CESAR4 adopts lumped fission products to which the appropriated one group equivalent "lumped" cross-section (mainly capture cross-sections) is associated. Therefore, CESAR4 is able to provide an assessment of the total mass of FPs generated but not its isotopic composition.

On the contrary, CESAR5 adopts separated treatment for about individual 200 FPs (the remaining FPs are treated as lumped FPs [187]).

For the study, the CESAR4 module has been used as reference, [4, 62]. This choice has been taken mainly in view of the post-processing computer time required (CESAR5 needs ca. 60 times more time than CESAR4).

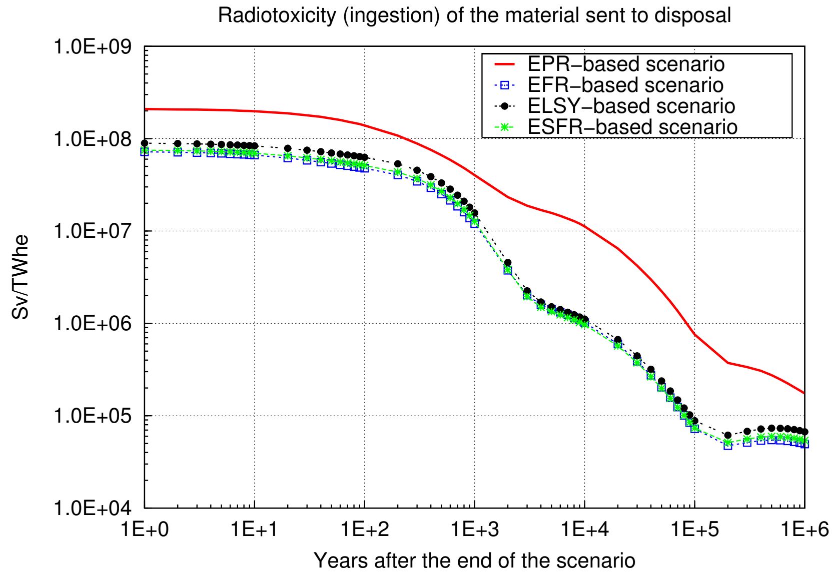

Figure 5.7: Specific radiotoxicity (ingestion) evolution of the material sent to disposal. Comparison "once-through" and Pu multi-recycling strategy [2200 is fixed as t = 0]

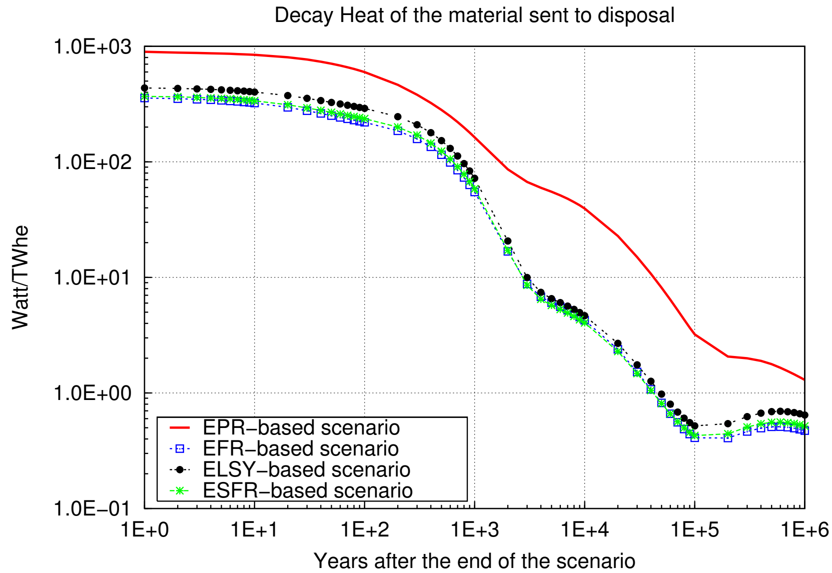

Figure 5.8: Specific decay heat evolution of the material sent to disposal. Comparison "once-through" and Pu multi-recycling strategy [2200 is fixed as t = 0]

This approximation influences the curves during the first 300 years (where FPs are the main contributors) but it does not affect the value in the period (1,000-10,000 years) in which the effect of the Pu multi-recycling is dominating (period of time more relevant for comparing fuel cycle strategies).

During the activity performed, it has been noticed that CESAR4 treats in a simplified way the long-lived isotopes with usually small concentrations in SF and waste (e.g. U234, Cm245, Cm246). This simplification can impact the radiotoxicity curves at around 100,000 years (mainly when high contents of Cm are considered). In order to solve this point, the isotopic composition in disposal in 2200 (data fixed as end of the scenario) has been extracted from the COSI simulations and provided as input to the ORIGEN2.2 depletion code [194] for decay evolution.

Applying that scheme the Figures 5.7 and 5.8 have then been obtained by adopting COSI6 code (CESAR4 module) for scenario calculations and ORIGEN2.2 for mass evolution in disposal.

Finally, the infrastructures needs have been investigated. In particular, the fabrication and reprocessing plant capacities needed for sustaining the cycle have been assessed.

As an example, the comparison between the fabrication capacities for ESFR and the reference cases is summarized in Figure 5.9. Similar considerations are valid for the other two systems, as also indicated in Table 5.6.

The complete transition to a fast reactor based fleet enables to reduce by ca. 50% the annual HM treated in fabrication plants as indicated in Figure 5.9 and Table 5.6. The results are in agreement with e.g. [137].

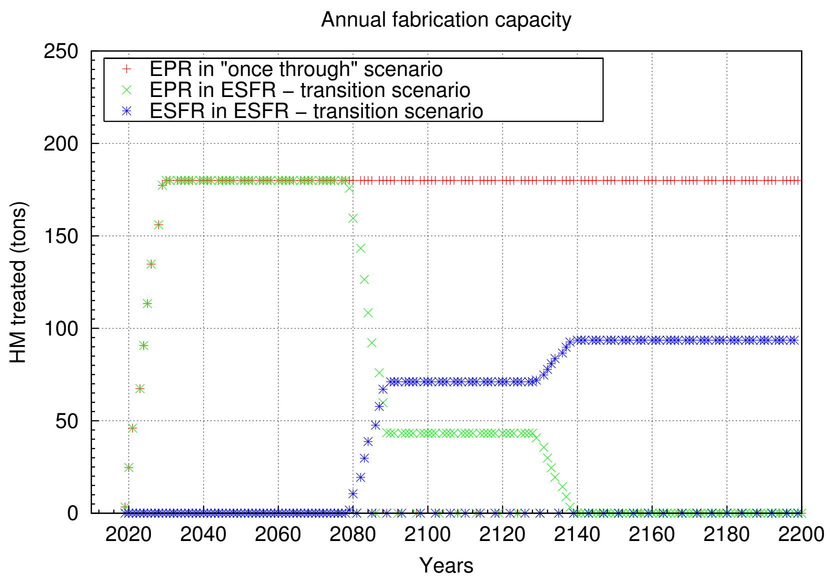

Figure 5.9: Annual fabrication capacity for LWR and FR fuel ("once-through" scenario compared with respect to ESFR-based scenario)

However, in order to sustain the cycle, additional reprocessing plants have to be introduced. For the ESFR case, the annual reprocessing capacity at equilibrium is about 100 tonHM per year (i.e. after the complete LWRs substitution) as indicated in Figure 5.10. The same behavior can be identified for EFR- and ELSY-based scenarios (only small differences due to the different dynamics of the transition have to be noticed).

		Comparison between scenarios			
		EPR-based	ELSY-based	ESFR-based	EFR-based
		tons			
2050	Fab (LWRs)	180.0	180.0	180.0	180.0
	Fab (FRs)	0.0	0.0	0.0	0.0
2100	Fab (LWRs)	180.0	107.6	43.2	0.0
	Fab (FRs)	0.0	78.2	71.1	84.4
2150	Fab (LWRs)	180.0	71.3	0.0	0.0
	Fab (FRs)	0.0	117.2	93.6	84.4
2200	Fab (LWRs)	180.0	71.3	0.0	0.0
	Fab (FRs)	0.0	117.2	93.6	84.4

Table 5.6: Fabrication capacities for the systems considered

In Figure 5.10, the contribution of LWR SF reprocessing and FR SF reprocessing plants (considered to be separated in the study) is indicated, too. From Figure 5.10 it is clear that the introduction of FRs is initially sustained only by the Pu coming from LWR SF reprocessing plant resulting in the limiting factor for the transition.

Afterward, only Pu coming from FR SF reprocessing plant is used for sustaining the cycle (under equilibrium condition this is expected due to the self-sustaining characteristics of the reactors considered, i.e. BR equal to ca. 1). However, the Pu accumulated by the remaining LWR fleet is used for sustaining the second step of introduction (ESFR and ELSY cases).

From Figure 5.10 it is also evident that during the first years of the transition the reprocessing plant works at the maximum capacity in order to reprocess enough Pu for the FRs start-up (the value considered is 850 tons similar to the La Hague plant [195]). Of course such an abrupt decrease in annual reprocessing capacity, as shown in Figure 5.10, is very unlikely. This item can be optimized, for instance, by placing on an earlier date the start-up of the reprocessing plant. However, the earlier separation of Pu implies a higher content of Am241 in the Pu stock (due to Pu241 decay) that affects the Pu content in the fresh fuel or might call an additional chemical separation treatment of the Pu taken from the stock immediately in advance of fuel fabrication. In order to evaluate this aspect, a parametric study has been performed and described in Par. 5.2.3.

Same effects can be seen, analyzing the cumulative HM mass reprocessed by thermal and fast reprocessing plants as indicated in Figure 5.11, for the ESFR case. For the other cases, further data are indicated in Table 5.7.

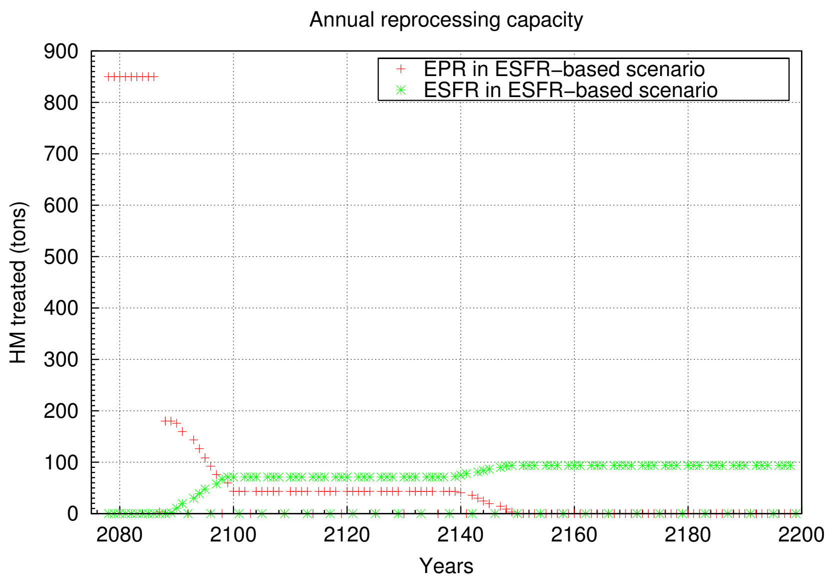

Figure 5.10: Annual reprocessing capacity for LWR and FR fuel (ESFR based scenario)

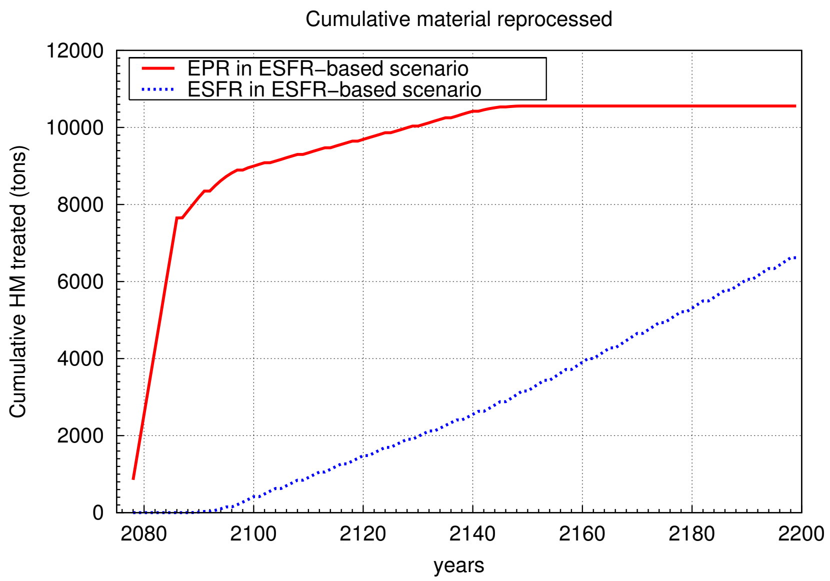

Figure 5.11: Cumulative reprocessing capacity for LWR and FR fuel (ESFR based scenario)

		Comparison between scenarios		
		ELSY-based	ESFR-based	EFR-based
		tons		
2078	Rep (LWRs)	850.0	850.0	850.0
	Rep (FRs)	0.0	0.0	0.0
2100	Rep (LWRs)	107.7	43.5	0.4
	Rep (FRs)	78.2	71.1	83.1
2150	Rep (LWRs)	106.0	0.0	0.0
	Rep (FRs)	85.6	93.6	84.4
2200	Rep (LWRs)	71.3	0.0	0.0
	Rep (FRs)	117.2	93.6	84.4

Table 5.7: Reprocessing capacities for the systems considered

5.1.2 Summary of FRs introduction

The comparison between FRs has shown that the three fast systems (assuming the same fuel cycle strategy, Pu multi-recycling) work in a relatively comparable manner. In particular, the EFR and ESFR cases give almost same results. For the ELSY case some differences have been highlighted mainly due to the reactor size and power density that affect the dynamics of the transition. This would become even more evident if the time scale would be extended beyond 2200.

For a better optimization of the resources a early and complete transition from LWRs to FRs would be reccomended.

In case of EFR, the complete transition can be achieved before 2100, while for the ESFR case, the 100% LWR substitution can be achieved around 2130. For the ELSY case, due to lack of Pu in the cycle, the complete transition can not be reached before the end of the scenario (2200). For this aspect, the ELSY system, at least in its present configuration, seems to be not optimized.

In term of back-end impact, the adoption of a partially closed fuel cycle helps in reducing of one order of magnitude the radiotoxicity of waste and heat load is well known. The three systems work in a comparable way.

However, the MAs inventory in the cycle continues to increase. A reduction of 30% (in 2200) can be obtained mainly due by the partitioning of Pu eliminating the contribution of Pu241 decay in Am241. In order to stabilize MAs inventory, fully closed fuel cycles, where all MAs are recycled in FRs, are presented in Par. 5.2.2.

In terms of infrastructures, the total fabrication needs are reduced by about 50% with respect to the "once-through" strategy. However, reprocessing plants need to be added to the cycle and FRs fuel fabrication plants.

The breeding characteristics of the systems can affect the dynamics of the transition increasing the Pu available in the cycle. In order to analyze this effect a parametric study has been carried out, see Par. 5.2.

5.2 Parametric Study concerning FRs

The implementation of specific MAs recycling strategies and improved breeder capabilities can considerably affect the scenario results [8]. In order to quantify the effects, a parametric study has been carried out.

Three cases have been studied in detail:

- Breeding characteristics studied by the adoption of a modified lead cooled system core (ELSY) with an improved Pu balance due to introduction of the depleted U radial blankets;

- Closed Fuel Cycles where Am and/or other MAs are homogeneously multi-recycled in the ESFR core;

- Reprocessing Options studied in order to analyze the influence on the Pu vector loaded in core.

The parameters considered are listed in Table 5.8.

Parameters considered	
BR, power density → systems	ESFR, ELSY, EFR
Introduction rate Energy demand	Pu availability
Advanced fuel cycles	Pu recycling Am recycling Am,Cm recycling MAs recycling
Reprocessing	Start-up date
Load Factor	85%, 76%
Cross-sections evaluation	Transport, Diffusion

Table 5.8: Parameters considered in the FRs parametric study

5.2.1 ELSY modified core: adoption of radial fertile blanket

In order to analyze the effects of a dedicated variation of the breeding capabilities on the scenario results, the ELSY model has been slightly modified including radial fertile blankets of depleted uranium oxide (0.25% U235 and 99.75% U238).

The analysis summarized in Par. 5.1 has shown how the actual configuration of the ELSY-like model is not optimized and, hence, not suitable for a complete transition to FRs. Therefore, the adoption of fertile blankets for increasing BG and BR, and thus Pu production, has been considered for improving the model.

The core configuration has been maintained unchanged (the three fuel zones are not modified) and radial steel reflector has been replaced by radial blanket (depleted UOX). In particular, two new models have been investigated by means of ERANOS code [182]. Suitable libraries for the COSI6 simulations have been generated.

The two models, described more in detail in Appendix D are:

1- ELSY-1-BLANKET-RING model: one ring of depleted uranium oxide sub-assemblies (90 SAs) has been added to the core periphery, replacing one ring of steel reflector, with respect to the reference configuration. The core layout is shown in Figure 5.12;

2- ELSY-2-BLANKET-RINGS model: two rings of depleted uranium oxide sub-assemblies (180 SAs) have been added to the core periphery, replacing two rings of steel reflector, with respect to the reference configuration. The core layout is depicted in Figure 5.12.

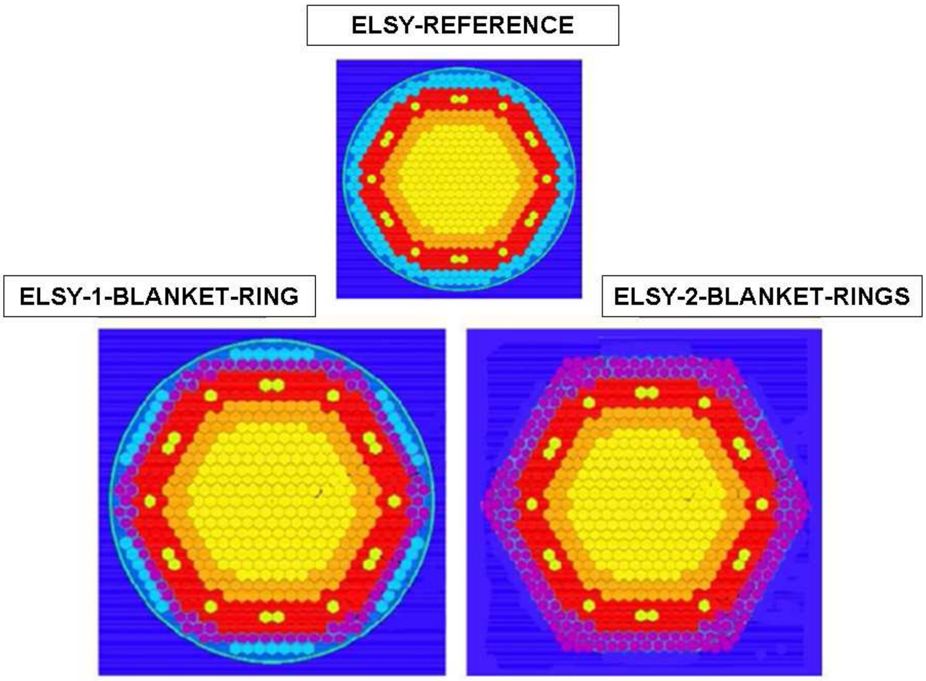

Figure 5.12: Modified ELSY models considered

The ELSY reference core has been defined in order to have a flat radial power distribution at Beginning Of Life (BOL). Therefore, the influence of the replacement of steel reflector by UOX blanket on the radial power shape in the core region and on the criticality of the reactor is expected to be only marginal (see Appendix D for more details).

The adoption of radial blanket increases the Pu balance of the system (increasing BG and BR) and improve the quality of the Pu in the cycle (increasing the average fissile content). This aspect has an effect on the dynamics of the transition scenario for the second step of introduction of ELSY systems.

The Pu vectors calculated by means of ERANOS2.2 code [182] at BOL and at EOL (after 1824 efpd) are included in Table 5.9. In the same table is also indicated the Pu average vector in the blanket at the End of Blanket Life (EBL)[8]. For the study, it has been assumed that blanket sub-assemblies remain in the core for twice the time of the core fuel assemblies (3,648 efpd instead of 1,824 efpd). This has been modeled in COSI6 assuming 1/4 batch reloading scheme for the core fuel and 1/8 for the radial blankets.

In term of masses the differences between the models are indicated in Table 5.10. The major contribution comes from the first ring of blanket as expected by neutron balance considerations. Small differences concerning the Pu content at BOL indicated in Table 5.10, are due to the adjustment made in the model in order to maintain the same criticality level (BOL reactivity of about +2000 pcm) [9].

[8]As indicated by Table 5.9, the adoption of fertile blanket generating a Pu vector with ca. 95% Pu239, has concerns for the proliferation resistance point of view [186]. For this reason, the advanced FRs, e.g. ESFR [58], have been developed to be self-sustaining without the adoption of blanket (internal BG ca. 0).

[9]The Pu content has been increased homogeneously in all the zones in order not to disturb the power profile, see Appendix D

Pu vector (%)							
Isotopes	BOL	EOL			EBL		
		1,824 efpd			3,648 efpd		
		ELSY	1-RING	2-RINGS	ELSY	1-RING	2-RINGS
Pu238	2.33	1.65	1.63	1.63	-	0.01	0.00
Pu239	56.87	58.44	58.5	58.5	-	94.51	96.36
Pu240	27.0	28.36	27.71	27.71	-	5.08	3.53
Pu241	6.10	4.5	4.38	4.38	-	0.39	0.11
Pu242	7.69	7.04	6.89	6.89	-	0.01	0.00

Table 5.9: Pu vectors loaded and unloaded from the ELSY models considered [ERANOS results]

The Pu excess (see Table 5.10) together with the better quality of Pu have an influence on the dynamics of the transition (for second step ELSY introduction). In fact, a higher nuclear energy share can be covered as indicated in Figure 5.13.

	ELSY	ELSY-1-BLANKET-RING	ELSY-2-BLANKET-RINGS
	Pu inventory at BOL (tons)		
CORE	8.74	8.78	8.78
BLANKET	-	0	0
TOTAL	8.74	8.78	8.78
	Pu inventory at EOL (tons)		
CORE	8.81	8.85	8.85
BLANKET	-	0.21	0.28
TOTAL	8.81	9.06	9.13
	Pu balance (kg)		
EOL	76	277	346

Table 5.10: Pu Mass balance [ERANOS results]

In 2150, the adoption of the ELSY reference model enables to cover 60% of the total nuclear energy demand. The adoption of blankets increases the share to 68% and 71% respectively for the case with one ring and with two rings of fertile subassemblies. As expected also in terms of scenario performance the differences remain limited but might facilitate ELSY introduction after 2200.

The impact of these variations on the fuel cycle front-end and back-end is quite low. An example is indicated in Figure 5.14 where the natural uranium saving by the adoption of modified ELSY model is represented. Similar figures can be obtained looking at the other back-end parameters.

A larger effect should be achieved when adding also axial blankets or changing the systems itself, as the comparison with EFR and ESFR has shown.

However, for the study it has been decided to restrict the changes of the breeding capabilities of the systems by considering in this investigation only the addition of radial blankets.

By the analysis of the results, the study has confirmed the good agreement between the neutronic model developed and the scenario results.

In fact, assuming the same energy demand for the three systems (Figure 5.15 where the second step, around 2140, of introduction has not be considered), the analysis of the Pu balance on the Pu stock after reprocessing shows that a large amount of Pu is available for achieving a higher energy share[10].

[10]The comparison with respect to a not optimize energy share, i.e. assuming the energy demand depicted in Figure 5.15, has been

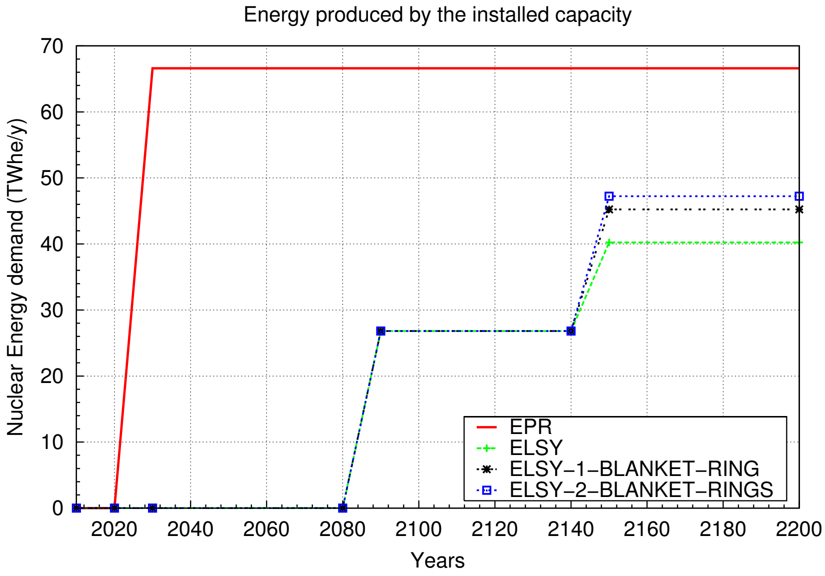

Figure 5.13: Nuclear Energy demand produced for the three ELSY models considered (according to the Pu available in the cycle)

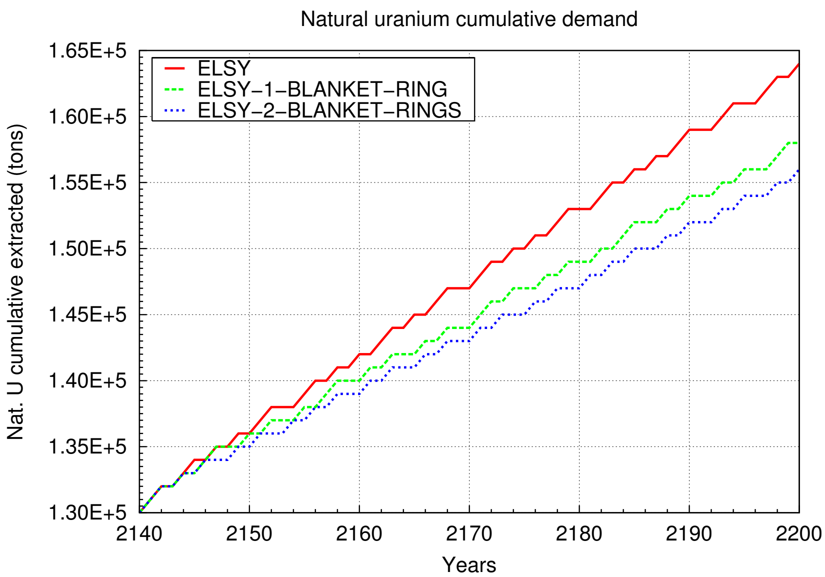

Figure 5.14: Natural uranium demand for the different ELSY models: period 2140-2200

In Figure 5.16 this comparison is shown. In particular, the red curve refers to ELSY reference model and the slope of the curve after the first introduction step (in 2100) is due to the accumulation of Pu coming from the remaining LWRs fleet (42 TWhe).

A different slope can be noticed for the ELSY-1-BLANKET-RING model (green curve), where in addition to the Pu coming for LWRs, the Pu coming from blankets is included. For the ELSY-2-BLANKET-RINGS, more Pu in the cycle is expected as confirmed by the blue curve in Figure 5.16.

These different slopes (Figure 5.16) justify and enable the different shares of substitution (for the second step) as pointed out in the previous Figure 5.13[11]. The variation between the slopes depends only on the Pu bred in the blankets (higher variation could be noticed between ELSY and ELSY-1-BLANKET-RING than between ELSY-1-BLANKET-RING and ELSY-2-BLANKET-RINGS cases, as expected).

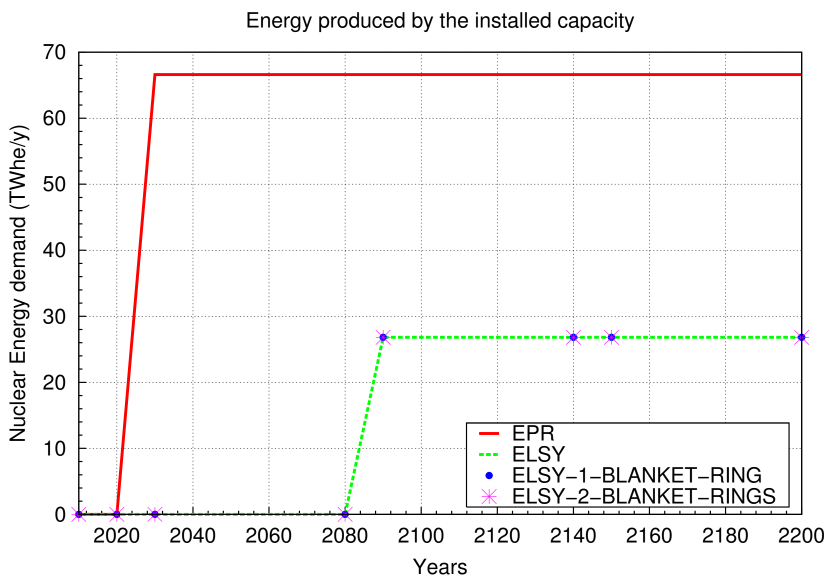

Figure 5.15: Nuclear energy demand assumed for the comparison

The results obtained with the ERANOS and COSI6 are compared in Table 5.11 and a good agreement can be observed (see last column). From the scenario point of view, the difference between the achievable energy shares is compared with the difference between the Pu balance calculated on the basis of the neutronic study. The behavior is fully comparable when the full irradiation time for the blankets (2 x fuel residence time) is taken into account. The same good agreement can be found looking to the Pu balance in Pu stock[12] (Table 5.11).

The Pu produced in blankets affects the total Pu vector by increasing the average fissile content.

This effect is represented in terms of scenario, by the variation of the Pu content loaded in the core.

The FR fresh fuel composition is evaluated by COSI6 adjusting, for each batch loaded, the Pu content in order to maintain fixed the reactivity level of the system. This procedure assumed that each isotopic composition can be represented in terms of reactivity by an equivalent composition where only U238 (fertile) and Pu239 (fissile) are taken into account.

performed only to point out the effects of blanket on the total amount of Pu int he cycle.

[11] A larger amount of Pu is available in the cycle, therefore a higher share of FRs can be introduced.

[12] For this comparison two dates have been chosen: 2090 where the Pu in the stock comes principally from FRs (Pu from LWR has been consumed for allowing the first introduction step) and 2140 time considered for the second introduction step.

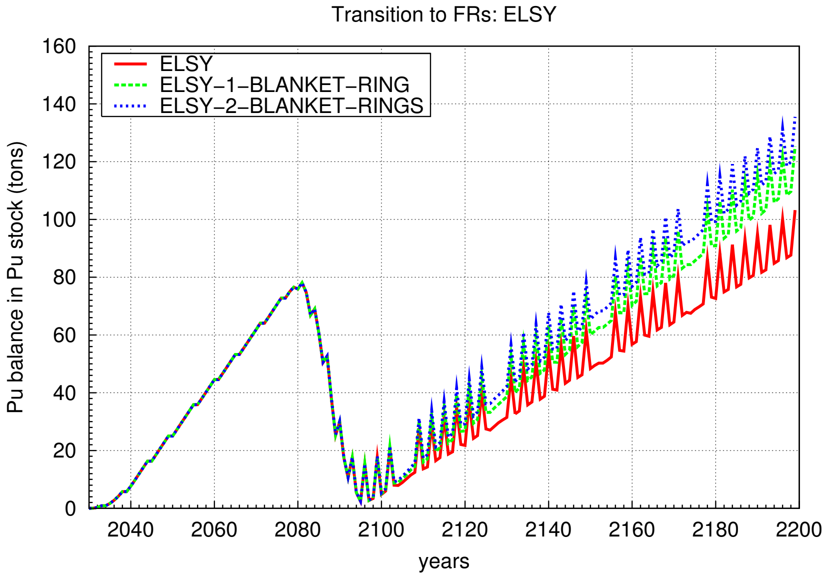

Figure 5.16: Pu stock: Pu mass balance comparison for the three models considered

	ELSY	ELSY-1-RING	ELSY-2-RINGS	B-A	C-B	(C-B)/(B-A)
	A	B	C			
Relative behavior in ERANOS code						
efpd	Pu produced in blankets (kg)					
1842	0	210	278	210	68	0.3
3648	0	389	529	389	140	**0.4**
Relative behavior in COSI code						
year	Pu balance in Pu stock (tons)					
2090	29.9	30.0	30.1	0.13	0.05	**0.4**
2140	53.9	62.9	67.3	8.9	4.4	0.5
year	Energy demand after second substitution (TWhe/y)					
2150	40.23	45.23	47.23	5	2	**0.4**
2200	40.23	45.23	47.23	5	2	**0.4**

Table 5.11: Comparison of the ERANOS and COSI relative behaviors concerning the ELSY models

In order to make this equivalence, the definition of an equivalent Pu239 content ("Pu239 equivalent" content) has been proposed by [196] and adopted in the study.

By the "Pu239 equivalent", a measure of the contribution of each isotope to the total reactivity of the system is provided. Each isotope is weighted on the basis of "reactivity coefficients", called also "ω-values", calculated with respect to a scale where Pu239 has weight equal to 1 and U238 equal to 0 [196].

Therefore, the "Pu239 equivalent" could be defined as in Eq. 5.1, where the E' is the Pu239 equivalent, E represent the effective Pu content, ω_i the reactivity coefficient associated to the isotope i defined as in Eq. 5.2 and ξ_i the isotopic fraction of the isotope i loaded in core. In Eq. 5.2, ν is the average number of neutrons produced per fission, σ_f is the microscopic fission cross-section, σ_a is the microscopic absorption cross-section and σ^+ is defined as $\nu\sigma_f - \sigma_a$.

More details about this parameter are provided in Appendix C where the analysis of breeding gain (BG) and breeding ratio (BR) concepts (based on "Pu239 equivalent") are presented.

$$E' = E\left[\sum_i \omega_i \xi_i\right] \tag{5.1}$$

$$\omega_i^n = \frac{(\nu\sigma_f - \sigma_a)_i^n - (\nu\sigma_f - \sigma_a)_8^n}{(\nu\sigma_f - \sigma_a)_9^n - (\nu\sigma_f - \sigma_a)_8^n} = \frac{\sigma_{i,n}^+ - \sigma_{8,n}^+}{\sigma_{9,n}^+ - \sigma_{8,n}^+} \tag{5.2}$$

This theory[13] is implemented in the COSI6 code in order to calculate the composition of each FR fresh fuel batch [187].

For each fast system the input data provided to COSI6 are the BOL "Pu239 equivalent" and the ω_i reactivity coefficients. A single set of omega values are used for the whole simulation[14]. On the basis of these data, taking into account the isotopic composition in the Pu stock (influenced by the presence or absence of blanket and by the options for reprocessing), the Pu content for every fresh batch is calculated in agreement with Eq. 5.1. The more fissile fraction is available in the Pu stock, the less Pu content is loaded in the core.

In order to check the influence of the modified ELSY models on the fresh fuel Pu content, the three ELSY models have been compared assuming the same nuclear energy demand (see Figure 5.15) and the same assumptions on the reprocessing plant.

The "Pu239 equivalent" calculated in input and output for the three systems is represented in Figure 5.17. Furthermore, the Pu content (%wt.) in input is shown in Figure 5.18.

As expected "Pu239 equivalent" in input to the systems (see Figure 5.17) is maintained constant for all the three cases. The "Pu239 equivalent" in input calculated for the ELSY reference model is slightly lower than the value adopted for the modified models (12.35% instead of 12.45%). In fact, in the modified models the initial content of Pu has been slightly increased to maintain the same criticality level at BOL (by the compensation of the leakage term increase due to absorption in the blanket), as indicated in Appendix D.

From the same figure (Figure 5.17) it is also evident that the composition initially loaded is not an equilibrium composition (the "Pu239 equivalent" in output changes with the scenario). However, the three systems reach fairly well equilibrium conditions (represented by the stabilization of the "Pu239 equivalent" in output) toward the end of the scenario.

In this case, where the fraction of LWRs remains significant in the scenario (ca. 60%), an equilibrium condition can not be completly reached because of the SF coming from LWRs and the Am241 accumulation in the Pu stock (as confirmed also by Figure 5.19) change the isotopic composition ot the stock and, hence, the Pu content to be loaded in the core (for each batch loaded).

[13]Theory described e.g. in detail in *Traité de Neutronique, 1978*, [196]

[14]The adoption of a single set of omega values for the whole scenario is justified by the small variation of XSs for FRs.

However, the adoption of blankets helps on reaching a condition more stable and close to the equilibrium than the reference case (as indicated in Figure 5.18).

All systems have comparable behaviors for the first batches (period 2080-2100) in which the composition in stock is only related to the SF from LWRs. The Pu content of these first batches is higher than the reference value (ca. 19%wt. instead of 18%wt.). This is mainly due to the option adopted in the reprocessing plant (start-up in 2030), where the Pu241 depletion increases the Am241 and Np237 content in the stock (an example for the ELSY reference case is indicated in Figure 5.20). These isotopes are then loaded in the fresh fuel, and the Pu content is increased in order to balance the negative contribution to the reactivity arising from Am241 and Np237 (with ω equal to -0.33 and -0.28, respectively). More details about the impact of the reprocessing options are indicated in Par. 5.2.3.

After the first batches, the composition in the Pu stock is affected by the coming SF from the remaining LWR fleet and by the Pu bred in blankets. The more Pu is bred in the blanket, the more stable remains the Pu content.

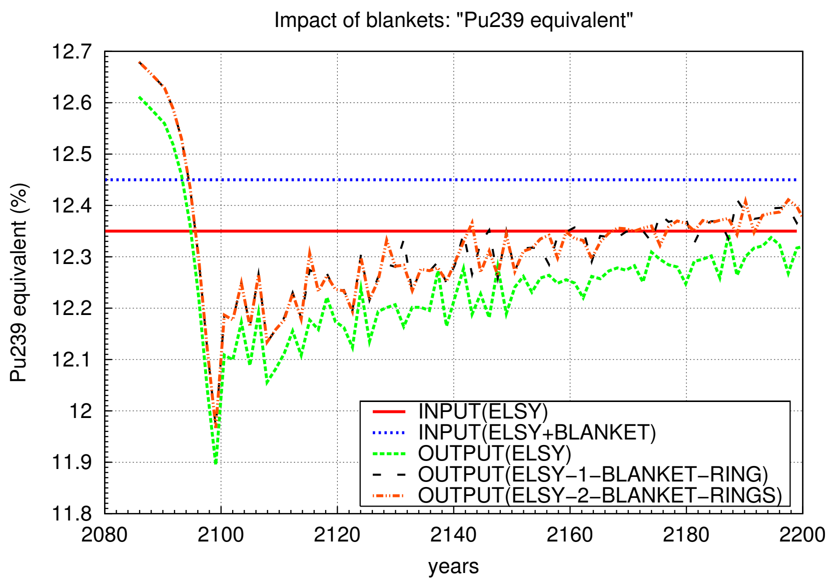

Figure 5.17: Pu239 equivalent (input and output) for the three ELSY models considered [Energy demand depicted in Figure 5.15]

The adoption of modified ELSY models can also affect the reprocessing and fabrication annual capacities. In fact, as indicated in Figure 5.13, higher fractions of the energy demand can be covered by models including blankets.

Therefore, the FRs fuel fabrication and FRs SF reprocessing capacities need to be properly extended. The contribution of blankets fabrication and reprocessing have to be taken into account in addition to the contribution due to the higher nuclear energy share covered.

Both effects can be seen in Figure 5.21 showing the required fabrication capacities. In particular, the contribution of blanket fabrication can clearly be identified in the period 2100-2140 (i.e. before the second step of FRs introduction, when the energy demand is the same for the three models), and the contribution of the different energy demand covered is also visible for the period 2160-2200.

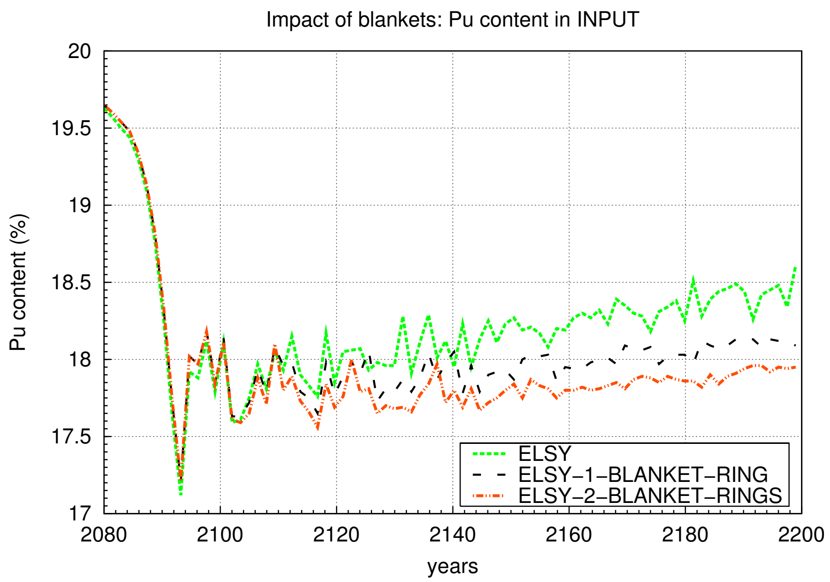

Figure 5.18: Pu content (%) in input to the systems [Energy demand depicted in Figure 5.15]

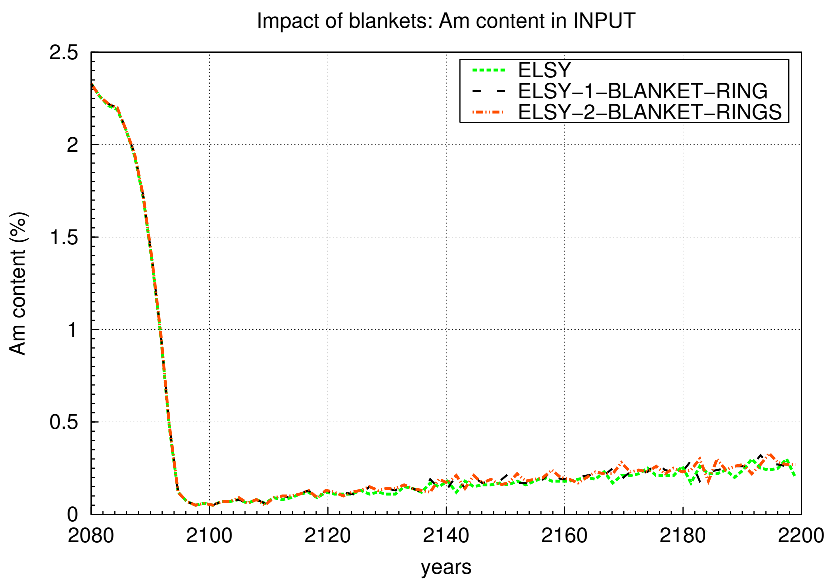

Figure 5.19: Am content (%) in input to the systems [Energy demand depicted in Figure 5.15]

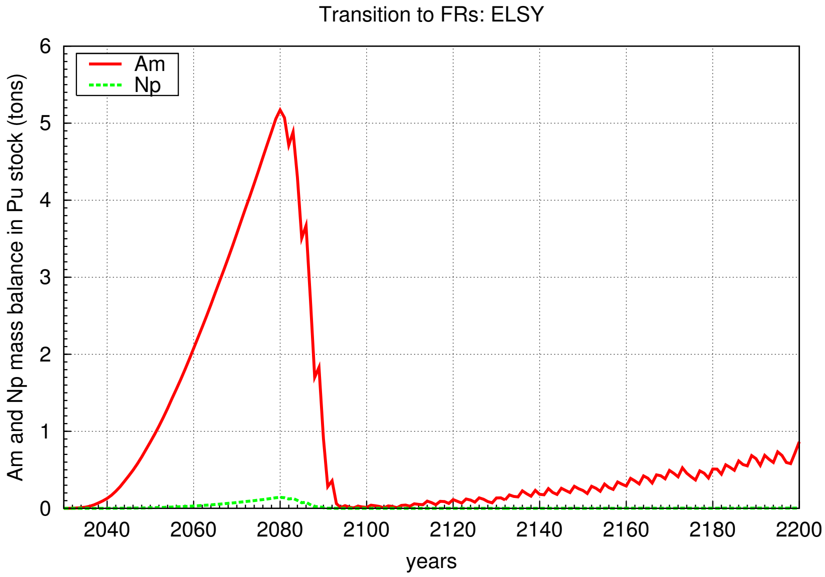

Figure 5.20: Am and Np accumulation in Pu stock [Energy demand depicted in Figure 5.15]

The differences in annual fabrication with and without blanket contribution have been checked also in regard to the ERANOS model. The results are in good agreement as indicated in Table 5.12. The small differences are related to some approximations made in the generation of the COSI6 library in terms of maximum burn-up fixed for the library (more details are indicated in Appendix A). The same behavior could be found also for the reprocessing plants.

Relative behavior in COSI code			
year	ELSY	ELSY-1-BLANKET-RING	ELSY-2-BLANKET-RINGS
Annual fabrication capacity (tons)			
2120	78.16	86.12	94.38
Dif. (%)	Ref.	8.0	8.3
Relative behavior in ERANOS code			
Mass of core (tons)			
	77.93	86.98	96.18
Dif. (%)	Ref.	9.0	9.2

Table 5.12: Comparison of the ERANOS and COSI relative behaviors concerning the ELSY models: fabrication capacity

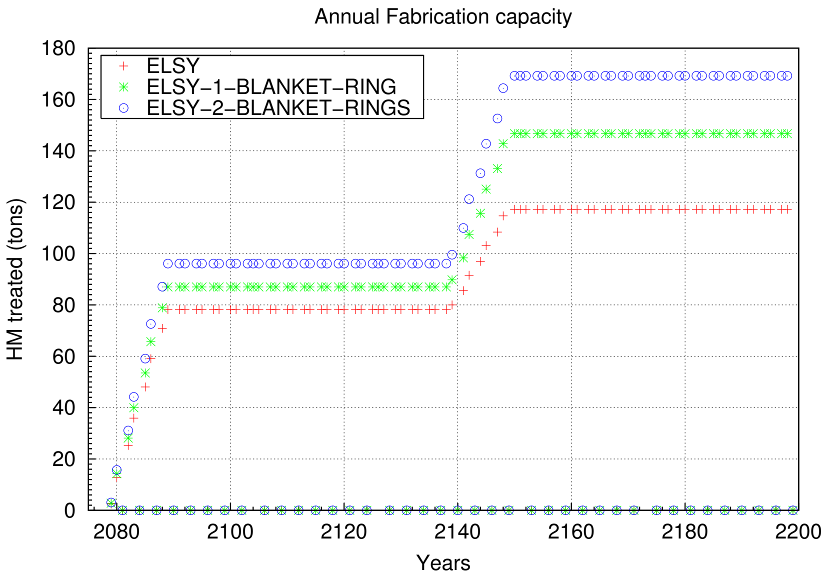

Figure 5.21: Annual Fabrication capacities for the three models considered [Energy demand depicted in Figure 5.13]

5.2.2 ESFR and MAs multi-recycling

In order to optimize the fuel cycle back-end, the adoption of closed fuel cycles has been investigated, too.

The analysis has been oriented to the ESFR case considering MAs multi-recycling in core. In particular two cases have been considered:

- a partially closed fuel cycle where Am and Pu are multi-recycled in FRs. The fuel cycle scheme considered is indicated in Figure 5.22;

- a closed fuel cycle where Pu and MAs are multi-recycled in FRs. The fuel cycle scheme considered is indicated in Figure 5.23;

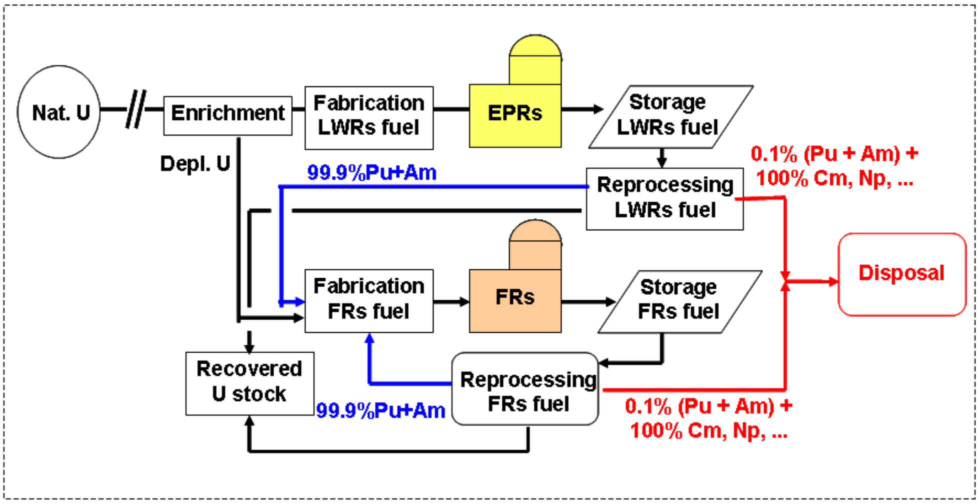

Figure 5.22: A simplified flow scheme for the Am and Pu multi-recycling in FRs

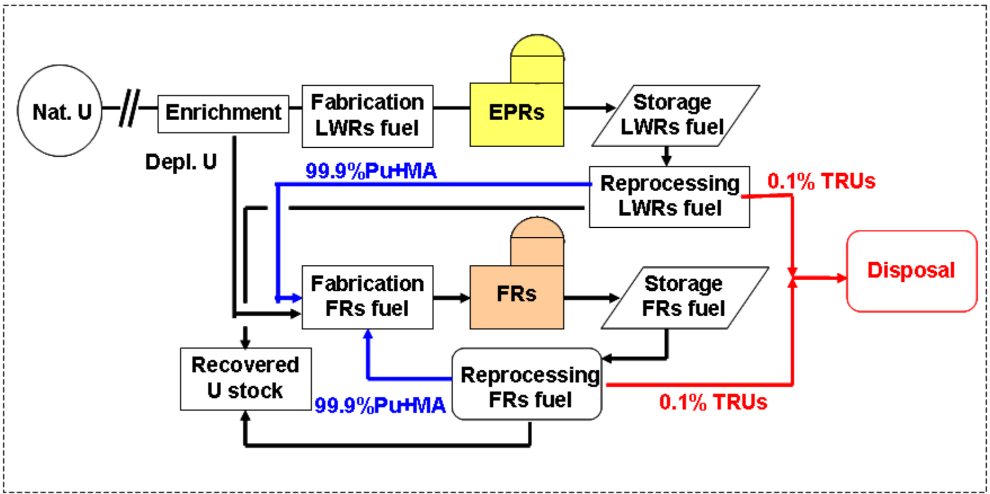

Figure 5.23: A simplified flow scheme for the MAs and Pu multi-recycling in FRs

The first case analyzed is the Am multi-recycling in FRs (americium is the main contributor to the total amount of MAs sent to disposal). The homogeneous scheme has been adopted in the study. The idea to adopt only Am multi-recycling has been considered also in other work [58], in order to improve the MAs mass balance of the system, and to reduce the reactivity swing. Preliminary results, obtained in the frame of the Ph.D. activity, in terms of safety are included in [31, 30] and briefly presented in Appendix D.

In fact, for the present scenario study, safety analyses have not be performed in details.

However, the results obtained during the study, concerning the neutronic and safety performances of the ESFR core (summarized in [31, 30]) have been considered as guidelines to fix the maximum MAs and Am content for homogeneous burning in ESFR also in the scenario studies.

In fact, when loading up to 4% of Am homogenously in core, the safety feedbacks (as void effect and Doppler) are not dramatically deteriorated (the void effect increases by ca. 350 pcm and Doppler is deteriorated of about 300 pcm, as better underlined in Appendix D and in [31, 30]). The value of 4-5% adopted is in agreement with other studies [133, 197].

Therefore, a maximum of about 4% Am in core has been considered as reference also for the present fuel cycle study. The same value has been assumed also for the MAs multi-recycling (4% Am or 4-5% MAs).

For the Am multi-recycling two cases have been compared: 1) 4% of Am homogeneously loaded in core, and 2) 2% of Am homogeneously loaded.

The effect of this option on the front-end and back-end facilities has been evaluated. Main attention has been devoted to the fuel back-end parameters.

As indicated in Figure 5.22, only 0.1% of Am and Pu are sent to the repository together with 100% of Cm and Np and higher actinides.

Assuming the same nuclear energy demand depicted in Figure 5.13 for an ESFR-like fleet, and loading 2% or 4% of Am in each batch, an error "lack of material" appears as COSI message. In fact, the Pu available (with the relative vector) in the cycle is not sufficient to compensate the negative reactivity contribution associated to Am241 and Am243 and to maintain the same "Pu239 equivalent" (for ESFR the "Pu239 equivalent" is equal to 10.71%).

The "lack of material errors" is visible also from Figure 5.24, where Pu balance in Pu stock goes to zero (both for 2% and 4% Am content). The Pu stock considered in Figure 5.24 is a stock defined for containing the Pu stream coming from reprocessing. In the study, the Pu stock incorporates Pu from LWRs ad FRs reprocessing, as indicated in Figure 5.25.

A higher Pu content to compensate the negative reactivity contribution from Am was expected by the neutronic studies performed. In the neutronic assessment, the Pu content was increased to maintain the same criticality level at BOL as for the reference model (keff = 1.00974).

As indicated in Figure 5.24, the reprocessing start-up date has been chosen close to the FRs early introduction date (in 2078 with FRs starting in 2080) in order to avoid the build-up of Am241 in the Pu stock. However, this option is not sufficient for avoiding the "lack of material" obtained.

Two options have been considered for solving the problem: 1) to reduce the first step of FRs introduction (from 53.64 TWhe/y, initially considered, to 49.64 TWhe/y), or 2) to shift the start-up of FRs from 2080 to 2085 (maintaining the same share).

Hereafter are summarized only the results associated to the second option, but the obtained results for both options are comparable.

The new nuclear energy demand adopted is indicated in Figure 5.26. A shifting of 5 years enables to eliminate the BOL lack of Pu, for compensating the negative reactivity contribution coming from Am content. This delay on the introduction of FRs implies the LWRs lifetime extension of 5 years, considered in principle feasible.

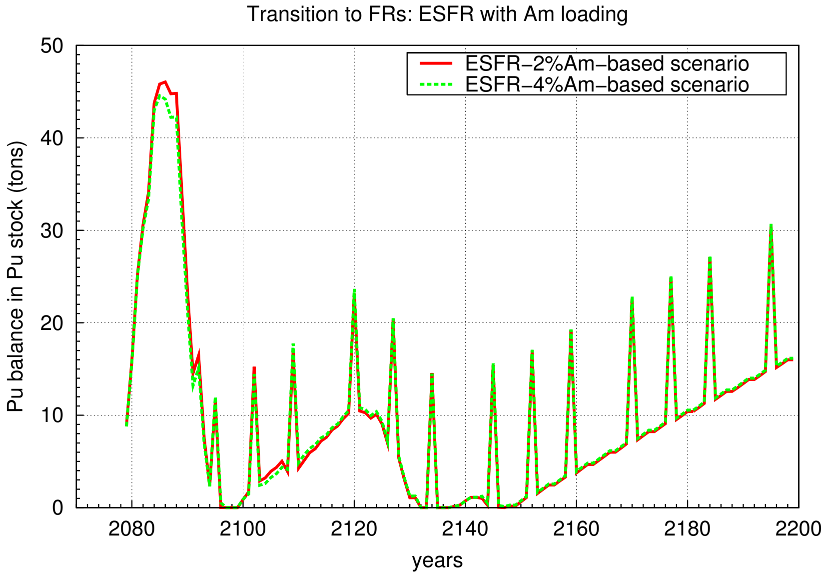

Figure 5.24: Pu stock: Pu mass balance comparison for the ESFR-Am recycling cases

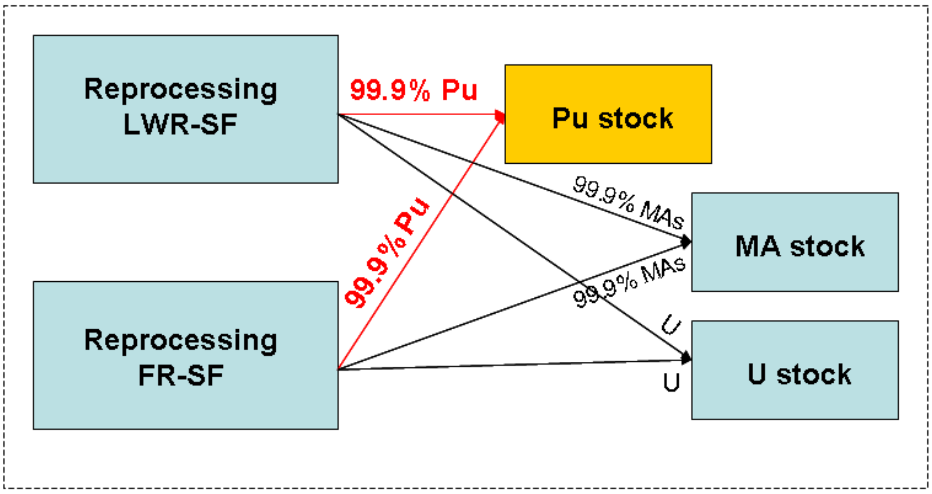

Figure 5.25: Reprocessing plant: Pu, MAs, U stocks

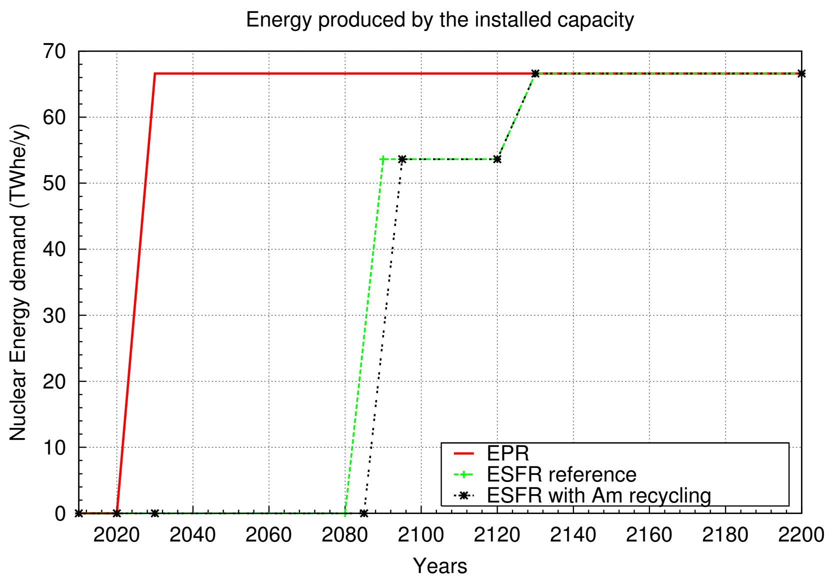

Figure 5.26: Nuclear Energy demand assumed for avoiding "lack of material error" in the simulation

The influence on the uranium resources involved is negligible; in 2200 the cumulative value of uranium resources is 98,900 tons instead of 99,600 tons. The fabrication and reprocessing annual capacities are not affected by this parameter.

As expected the most important impacts are on the fuel back-end. In particular, the total MAs in the cycle are reduced if compared with the ESFR reference case or with the "once-through" case. Figure 5.27 shows this comparison. Figure 5.27 shows also that the multi-recycling of 2% or 4% of Am gives fully comparable results for the scenario considered. The asymptotic behavior is not influenced by the Am percentage. This aspect is mainly due by the specific scenario considered where only the MAs (hence, also the Am) generated in the cycle are loaded in the core. After the first cycles, the available Am is burnt (earlier for the ESFR-4%Am case than the ESFR-2%Am, as indicated in Figure 5.27 in the period 2090-2120) and the same Am content is laoded for the following cycle in the cores.

This behavior depends on the total MAs (mainly Am) in cycle. In fact, only during the first year the amount of Am is enough to load the target content (4% or 2% of Am). This Am comes mainly by the accumulation by Pu241 decay. After few cycles, the two systems reach a comparable equilibrium condition as indicated in Figure 5.28, where the Pu and Am contents in the fresh batches are analyzed.

The reduction of MAs in the cycle is mainly due to the Am reduction. In fact, as indicated in Figure 5.29 the Am is largely reduced if compared with the standard ESFR based scenario. On the contrary, Cm, generated mainly by the irradiation of Am in FRs, is largely increased (see Figure 5.30).

The reduction of Am contributes to the medium-term radiotoxicity reduction in the period 1000-3000 years where Pu and Am are the main contributors (Figure 5.31). However, the increase of Cm stabilizes the radiotoxicity in the period 300-5,000 years (mainly due to Cm245 with $T_{1/2}$ equal to 8,500 years). The same behavior can be found for the heat load (Figure 5.32).

In the mentioned figures only the case ESFR-4%Am has been considered; the case with 2% of Am in core has not been plotted because it has the same trends as shown in the Figures 5.27, 5.28, 5.29, 5.30.

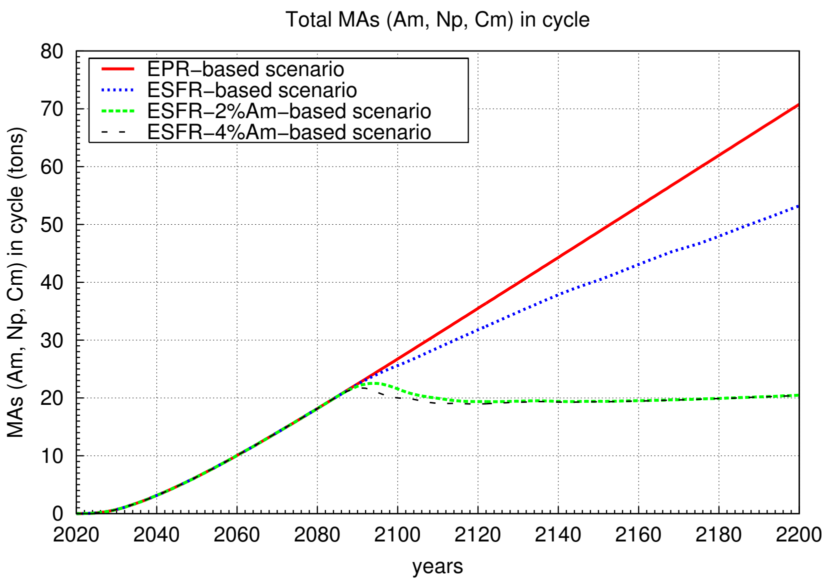

Figure 5.27: Cumulative MAs (Np, Am, Cm) in the cycle

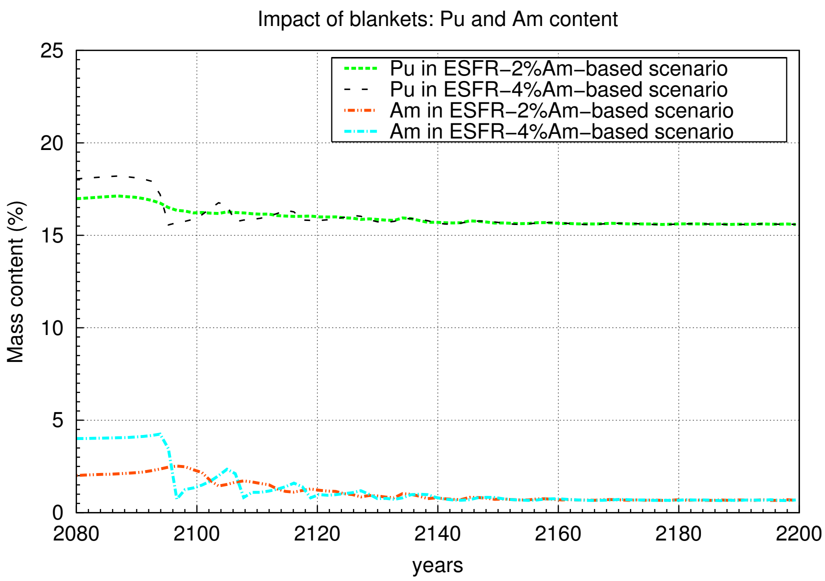

Figure 5.28: Pu and Am content (%) in input for the two ESFR models

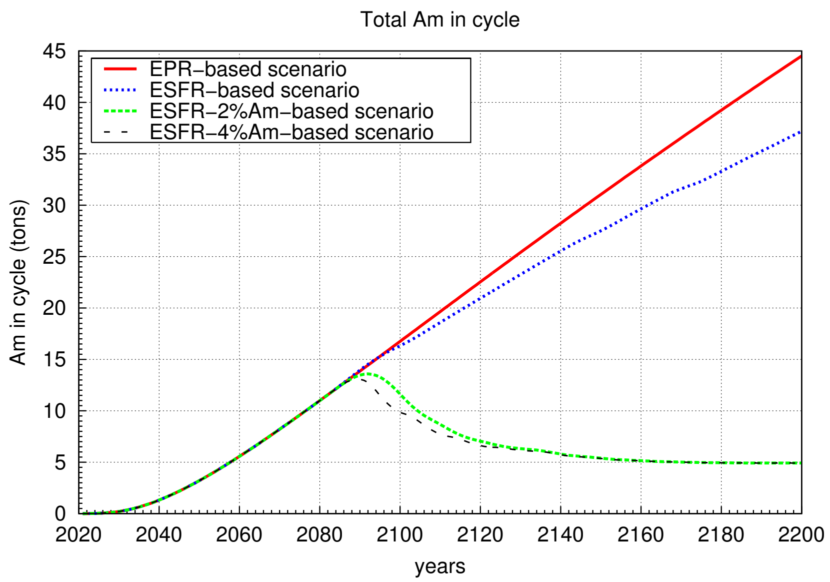

Figure 5.29: Cumulative Am content in the cycle

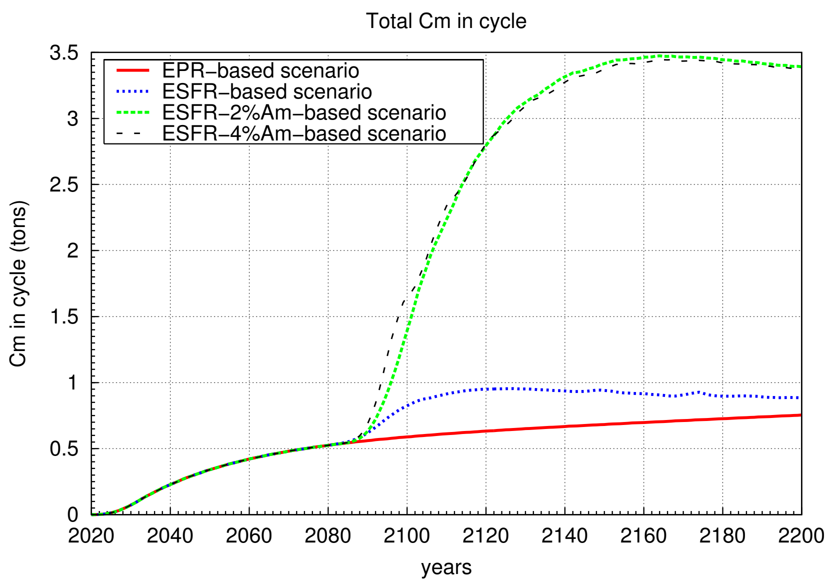

Figure 5.30: Cumulative Cm content in the cycle

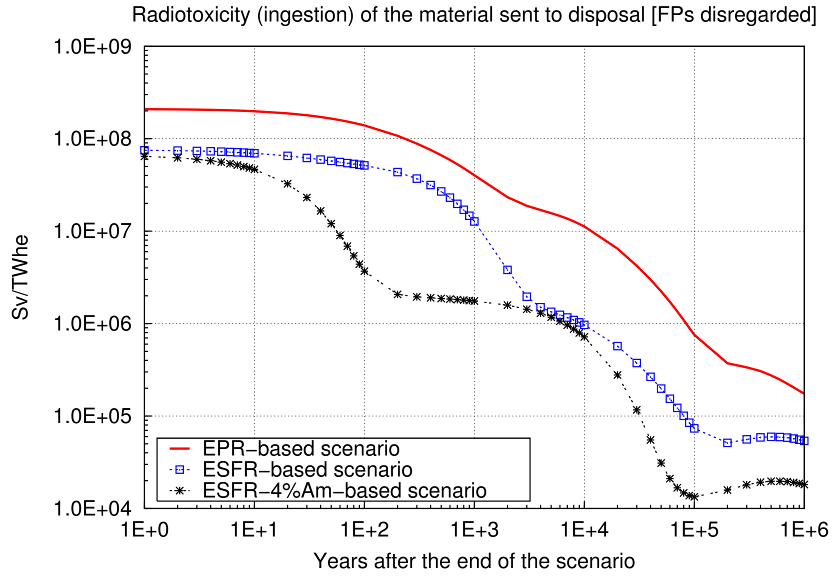

Figure 5.31: Specific Radiotoxicity (ingestion) evolution of the material sent to disposal. [2200 is fixed as t = 0]

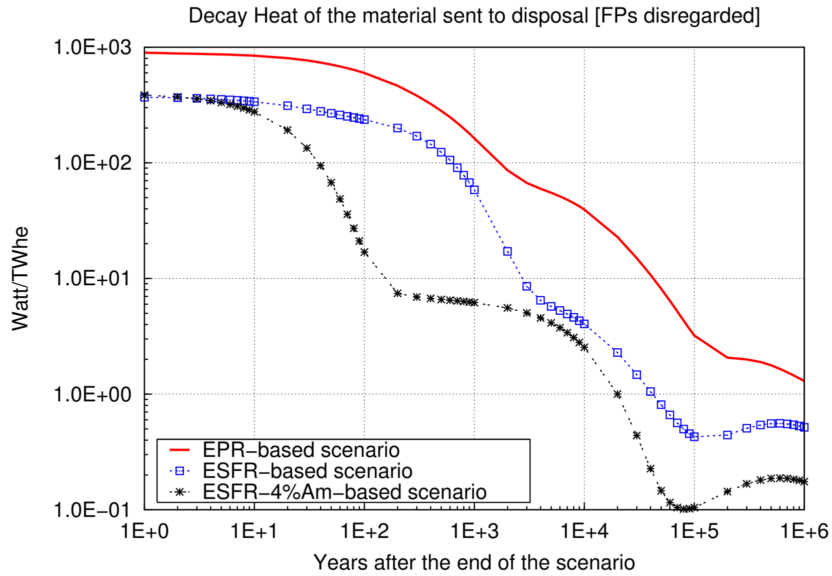

Figure 5.32: Specific Decay heat evolution of the material sent to disposal. [2200 is fixed as t = 0]

A more significant radiotoxicity reduction should be obtained by the MAs (Am, Cm and Np) multi-recycling in FRs. Two cases have been considered: 1) an homogeneous content equal to 4% of Am and Cm (maintaining the relative fraction between the two isotopes), and 2) the complete MAs multi-recycling.

These two cases have been compared with respect to the same energy demand depicted in Figure 5.26, in order to avoid lack of material (Pu) in the cycle.

However, a slightly better situation (in term of Pu balance) can be noticed for the case with Am and Cm loaded in core in regard to the case with Am only. This is an expected behavior because Cm gives a positive contribution to the reactivity as indicated by the positive ω values reported in Table 5.13[15].

Reactivity coefficients (ω)		
	BOL (T = 0)	BOC4 (T = 1230 efpd)
U234	0.01878	0.03346
U235	0.79637	0.78145
U236	-0.03172	-0.02313
U238	0.00000	0.00000
Np237	-0.30178	-0.26878
Np238	2.09327	2.06637
Np239	-0.37232	-0.33065
Pu238	0.65565	0.66422
Pu239	1.00000	1.00000
Pu240	0.12394	0.13925
Pu241	1.55104	1.51873
Pu242	0.06816	0.08140
Am241	-0.35690	-0.32456
Am242g	2.32057	2.26161
Am242m	2.26892	2.21121
Am243	-0.34197	-0.31338
CM242	0.39046	0.41006
Cm243	2.60300	2.55418
Cm244	0.07477	0.10154
Cm245	2.26464	2.20714
Cm246	0.11949	0.13560
Cm247	2.03154	2.00898
Cm248	0.17953	0.19187

Table 5.13: ESFR reactivity coefficients

MAs multi-recycling enables to stabilize the MAs in cycle at a lower level than the case with only Am multi-recycling (mainly due to the contribution of Cm reduction). The comparison is presented in Figure 5.33. A reduction of about 86% (in 2200) with respect to the "once-through" case is obtained by the multi-recycling of MAs in FRs core.

It is also interesting to analyze the material (0.1%Pu losses and MAs) sent to disposal, because they are the only contributors to the radiotoxicity. The comparison among the different fuel cycles considered is indicated in Figure 5.34.

[15]The reactivity coefficients at BOL and BOC4, i.e. representative of quasi-equilibrium conditions, are shown in Table 5.13 in order to point out that they remain practically unchanged versus burn-up.

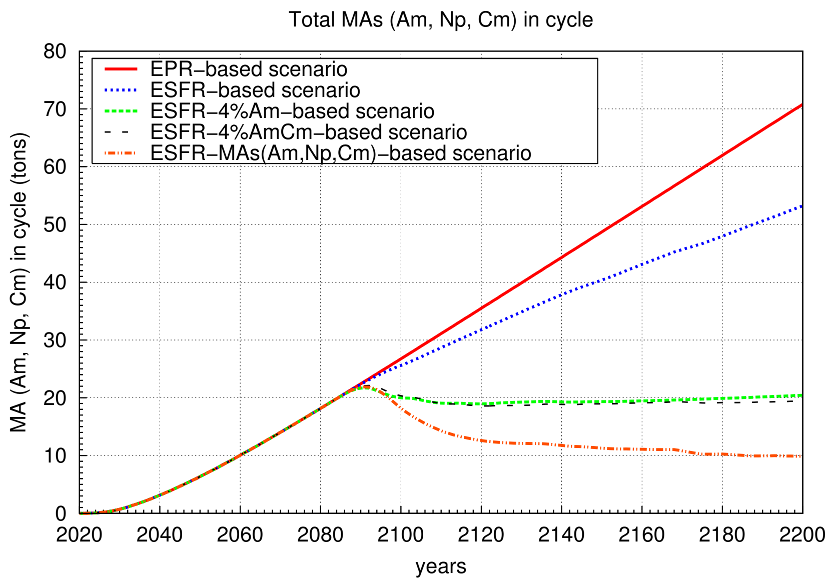

Figure 5.33: Cumulative MAs (Np, Am, Cm) in the cycle

In particular:

- for EPR and ESFR cases: all MAs are sent to the repository (plus 0.1% Pu losses);
- for ESFR-4%Am case: only Cm and Np are sent to the repository (plus 0.1% Pu and Am losses);
- for ESFR-4%AmCm case: only Np is sent to the repository (plus 0.1% Pu, Am and, Cm losses);
- for ESFR-MAs case: only losses (0.1% Pu, Np, Am, Cm) are sent to the repository.

In particular, to assess the contribution in radiotoxicity and heat load associated to the disposal, the comparison of the MAs content in 2200, i.e. at the end of scenario considered here, has been performed and summarized in Table 5.14. It is clear from Table 5.14 that in the case of ESFR loaded with Pu and Am only the build up of Cm is essentially different (three order of magnitude more than in the case with Am and Cm multi-recycling), and Np has a dominating fraction.

This amount of Cm impacts the radiotoxicity behavior in the period 1,000-10,000 years generating a kind of "plateau", in agreement with Cm245 half-life. The comparison in terms of radiotoxicity and heat load is shown in Figure 5.35 and Figure 5.36 respectively.

The implementation of P&T and MAs multi-recycling in FRs is the most important hypothesis affecting the scenario results, in particular from the back-end point of view.

The study performed, where several fuel cycles have been compared in a dynamic way, has confirmed the positive effect of P&T both in terms of total inventory and potential risk reduction (by reducing the radiotoxicity and decay heat associated to the material sent to disposal) in agreement with more general studies (e.g. [60]).

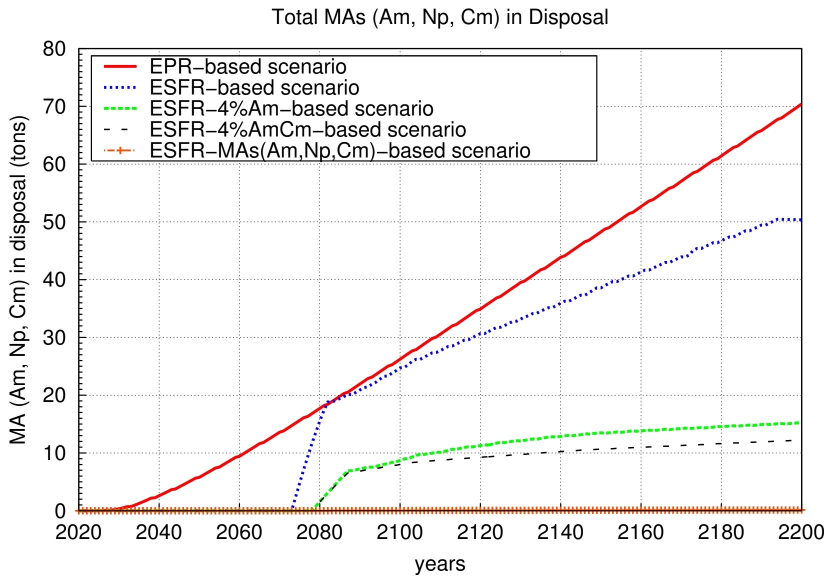

Figure 5.34: Cumulative MAs (Np, Am, Cm) in disposal

	PWR	ESFR	ESFR-4%Am	ESFR-4%AmCm	ESFR-MAs(Am,Np,Cm)
Mass in disposal at 2100 (tons)					
Total MAs	26.227	22.780	8.709	8.027	0.028
Am	16.626	14.180	0.019	0.019	0.018
Np	9.072	8.037	8.009	8.008	0.009
Cm	0.529	0.563	0.681	0.001	0.001
Share (%)					
Am	63.4	62.2	0.2	0.2	65.1
Np	34.6	35.3	92.0	99.8	32.6
Cm	2.0	2.5	7.8	0.0	2.3
Mass in disposal at 2200 (tons)					
Total MAs	70.408	50.441	15.156	12.233	0.132
Am	44.430	34.836	0.098	0.097	0.096
Np	25.268	14.860	12.146	12.129	0.029
Cm	0.711	0.744	2.912	0.008	0.007
Share (%)					
Am	63.1	69.1	0.6	0.8	72.6
Np	35.9	29.5	80.1	99.1	22.2
Cm	1.0	1.5	19.2	0.1	5.2

Table 5.14: MAs (Np, Am, Cm) content in disposal

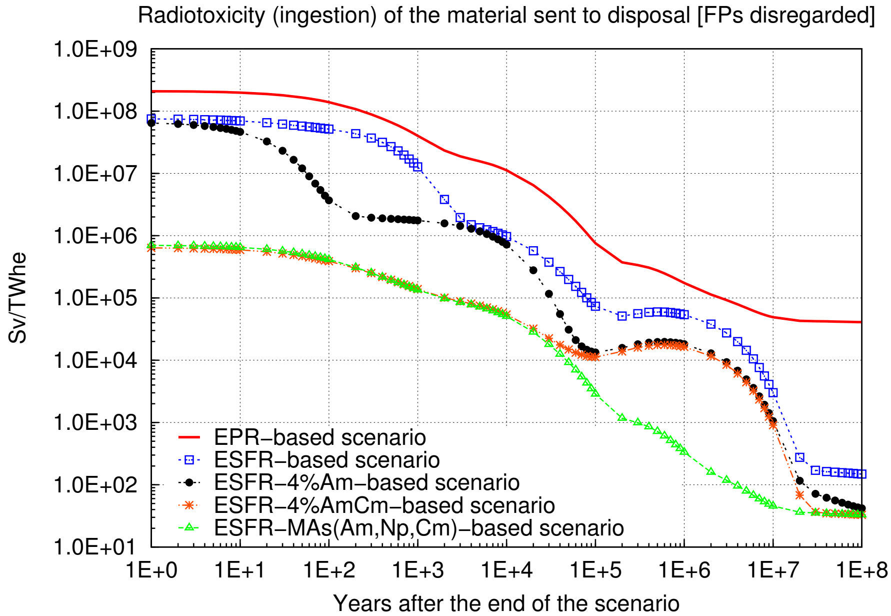

Figure 5.35: Specific Radiotoxicity (ingestion) evolution of the material sent to disposal. Comparison fuel cycle strategies [2200 is fixed as t = 0]

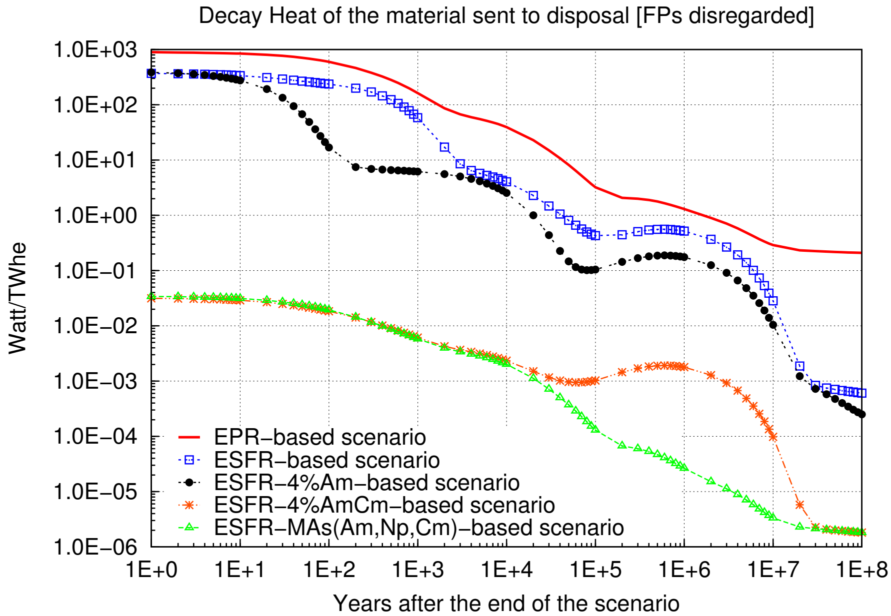

Figure 5.36: Specific decay heat evolution of the material sent to disposal. Comparison fuel cycle strategies [2200 is fixed as t = 0]

The advantages from the point of view of the fuel cycle are evident. However, in order to envisage the implementation of this solution, the analysis of the safety characteristics deterioration of the reactor systems in terms of void, Doppler, Beta effective should be checked. For the ESFR case, some preliminary studies have been performed (e.g. [31]). These studies have shown that there are margins of improvement of the systems in order to allow, with respect to the safety constraints, the MAs multi-recycling in FRs [31, 30].

Other technological aspects, as the problems related to the chemical separation of Am and Cm or to the handling of this kind of fuel at fabrication (having in mind e.g. heat production, neutron source intensity or He-release due to α-decay) have not be considered in detail.

5.2.3 Reprocessing options

In the previous paragraphs, it has been indicated that reprocessing options give an effect on the composition of the fresh fuel loaded in FRs. In order to clarify this point, the ESFR case has been considered and two different reprocessing options (related to the reprocessing start-up date) have been adopted.

The FRs start-up date is maintained fixed to 2080, but the reprocessing plant start-up date has been fixed to: 1) two years before the FRs start-up, or 2) once the LWRs spent fuel has been produced (i.e. the first SF from LWRs is discharged in 2025 and therefore, taking into account 5 years of cooling before reprocessing, the first material reprocessed would be in 2030).

Assuming the same energy demand as described in Figure 5.2, where ESFR plants are introduced for maximizing the energy share, one of the parameters affected by the reprocessing option is the Pu balance in Pu stock as indicated in Figure 5.37.

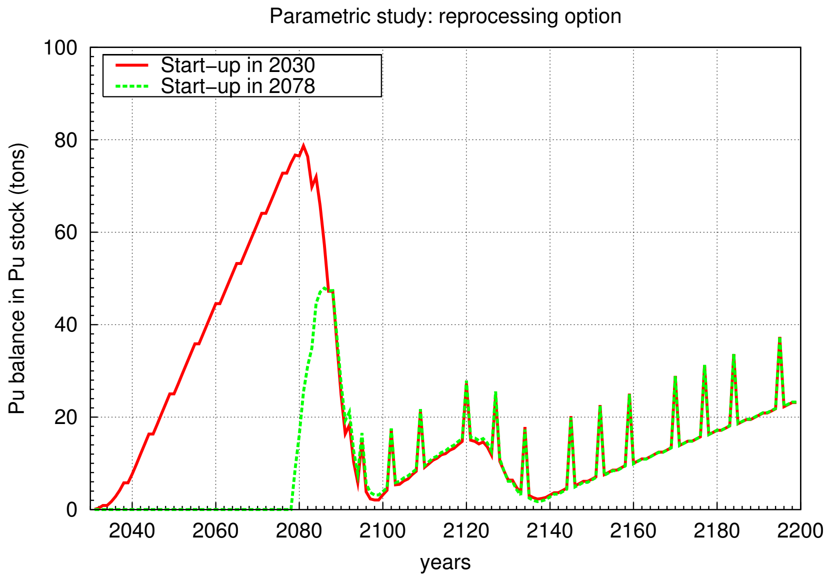

Figure 5.37: Impact of the reprocessing option on the Pu stock balance

From Figure 5.37 is clear that the Pu needed for the FRs start-up when reprocessing plant starts working close to 2080 is lower than in the case of early reprocessing start-up: the Pu that remains in Pu stock is slightly higher as indicated by Figure 5.38 (zoom of Figure 5.37 for the period 2090-2110).

In addition, the quantity of Pu in stock (represented by the peaks in 2080) is much lower in the case with reprocessing start-up close to FRs introduction. The Pu remains in stock for less time (additional favorable aspect: good also from proliferation resistance point of view) because once reprocessed it is used for the fabrication of new FR fuel. The annual mass treated by the fabrication capacity (both FRs and LWRs) is independent from the reprocessing options adopted[16] (see Figure 5.39). Only the composition of the fuel fabricated is changed on the basis of the Pu vector in Pu stock.

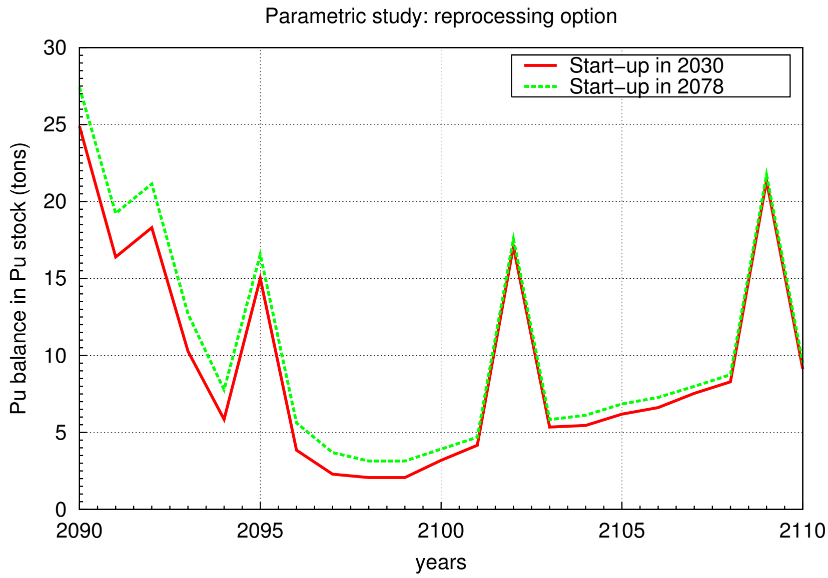

Figure 5.38: Impact of the reprocessing option on the Pu stock balance: zoom for the period 2090-2110

The behavior indicated in Figure 5.38 is related to the composition in stock (affected by Am241 and Np237 build-up) and to the way in which fresh fuel is calculated.

As indicated in Figure 5.40 and Figure 5.41, the build-up of Am241 and Np237 when is drastically reduced shifting the reprocessing start-up date close to FRs introduction date.

The differences in term of Pu241 and Am241 are respectively presented in Figure 5.42 and Figure 5.44. These behaviors substantially impact the Pu and MAs content in the fresh batches.

In fact, according to the COSI6 procedure for core reloading, the presence of Am241 and Np237 in the Pu stock is reflected in MAs loading in fresh batches and in an increased Pu content for maintaining the same criticality level (evaluated in COSI6 on the basis of the "Pu239 equivalent" as previously described).

The Pu and MAs content in the fresh fuel for the two cases is compared in Figure 5.43 and Figure 5.45. The adoption of closed reprocessing start-up date reduced the MAs content of the first batches from 2% (mainly Am) to 0.2-0.3%.

More details about the MAs and Pu content versus scenario are indicated in Table 5.15.

The adoption of a reprocessing start-up date close to the FRs introduction date gives better results in terms of Pu and MAs content loaded in core as indicated in Figure 5.43 and Figure 5.44. However it implies that reprocessing plants works at its maximal capacity during the first years in order to produce enough material for the fast reactor start-up.

[16]For the study, the fabrication plant has been modeled with "unlimited" capacity to follow the needs of the reactors. The fabrication start-up date (both for LWR and FR fuel) has been considered fixed.

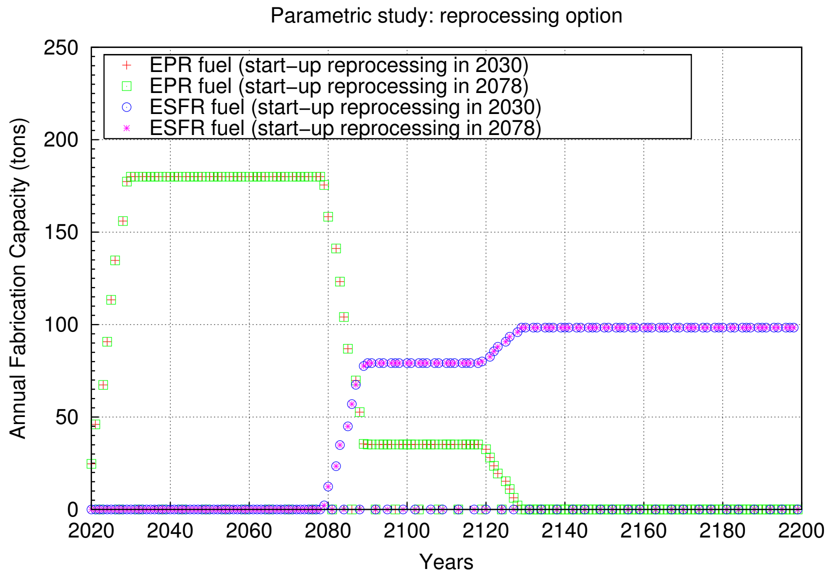

Figure 5.39: Annual fabrication capacity assuming different options for reprocessing start-up: LWR and FR fuels

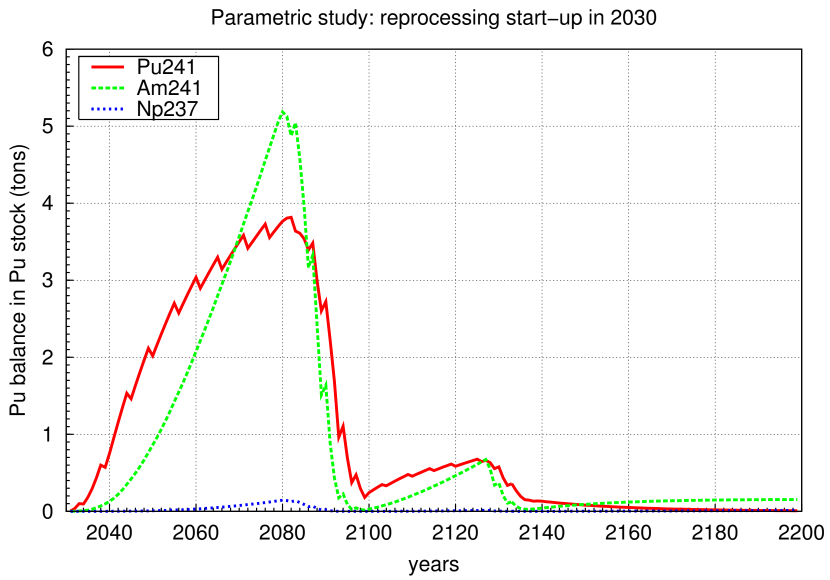

Figure 5.40: Composition in Pu stock: reprocessing start-up in 2030

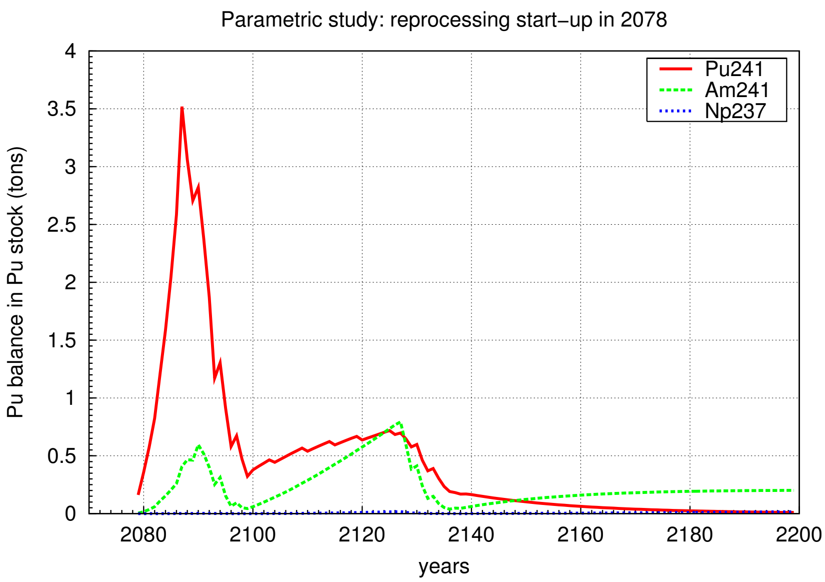

Figure 5.41: Composition in Pu stock: reprocessing start-up in 2078

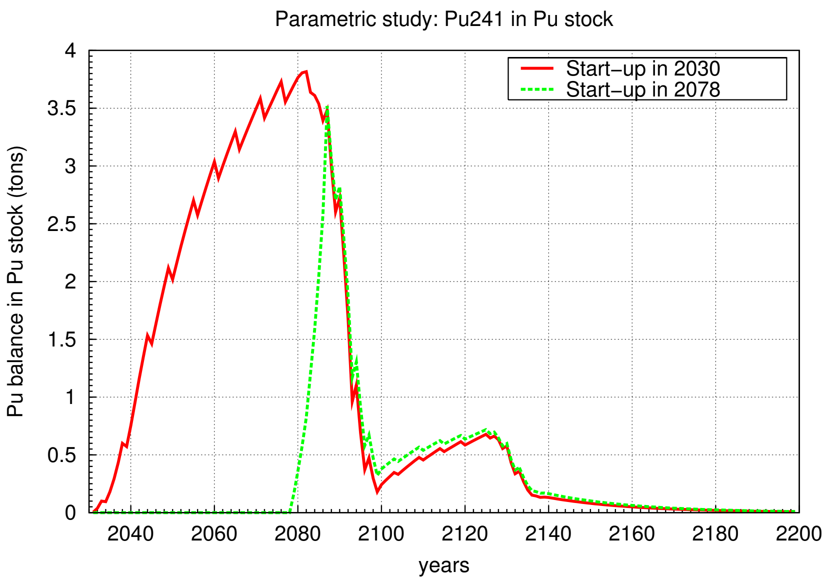

Figure 5.42: Pu241 content in Pu stock: comparison between reprocessing start-up dates

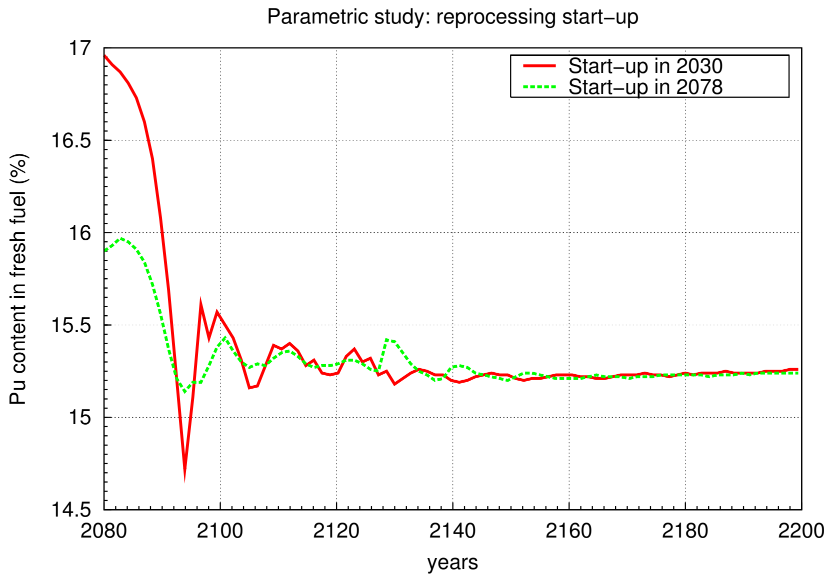

Figure 5.43: Pu content in fresh fuel assuming different reprocessing start-up options

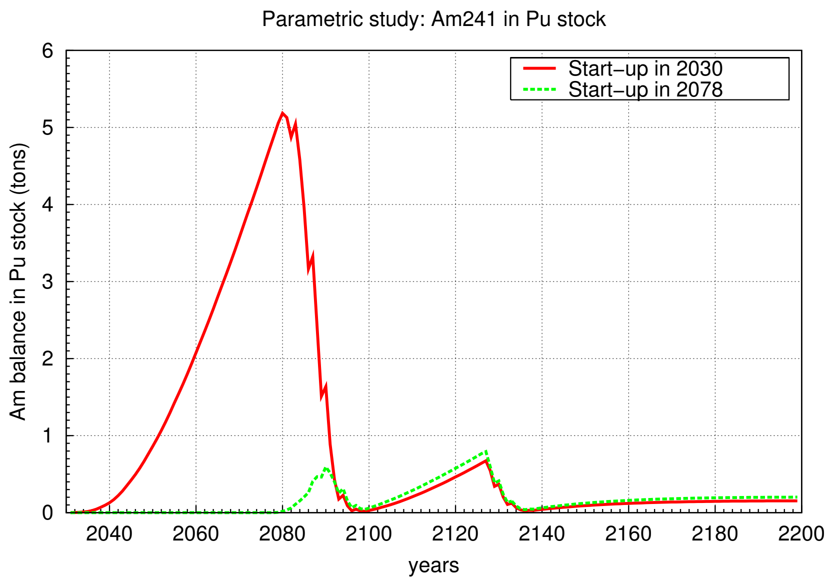

Figure 5.44: Am241 content in Pu stock: comparison between reprocessing start-up dates

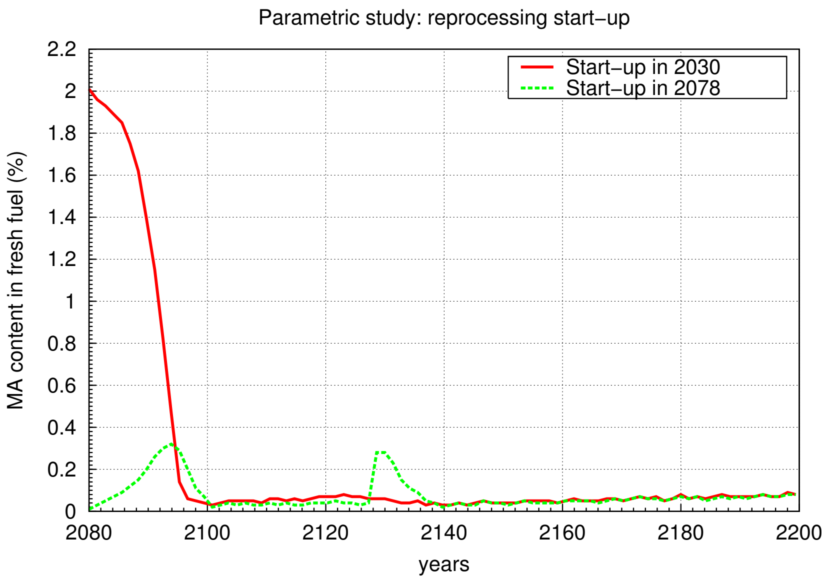

Figure 5.45: MAs content in fresh fuel assuming different reprocessing start-up options

	Start-up 2030		Start-up 2078	
year	Pu content	MAs content	Pu content	MAs content
fresh fuel				
2080	16.96%	2.01%	15.90%	0.01%
2090	15.69%	1.15%	15.37%	0.26%
2120	15.24%	0.07%	15.29%	0.04%
2150	15.21%	0.04%	15.22%	0.03%
2190	15.24%	0.07%	15.24%	0.08%
spent fuel				
2086	17.30%	1.44%	16.28%	0.43%
2096	16.56%	1.14%	16.04%	0.54%
2126	15.84%	0.39%	15.88%	0.36%
2156	15.86%	0.33%	15.84%	0.32%
2196	15.88%	0.32%	15.88%	0.32%
2206	15.90%	0.32%	15.89%	0.32%

Table 5.15: Pu and MAs content in fresh and spent fuel versus reprocessing option

Different situation appears when reprocessing operation starts once the SF is generated in LWRs. The reprocessing annual capacity follows the LWR SF inventory annually discharged behavior.

In order to compare the two approaches the contribution of the LWR and FR reprocessing has been taken into account separately. The comparison between the reprocessing annual capacity based on different reprocessing options is shown in Figure 5.46 where the behaviors above described are visible. Concerning FRs reprocessing annual capacity, the two approaches lead to results which are in very good agreement (in both cases SF coming from FRs is reprocessed once it has been created). The comparison is summarized in Figure 5.47.

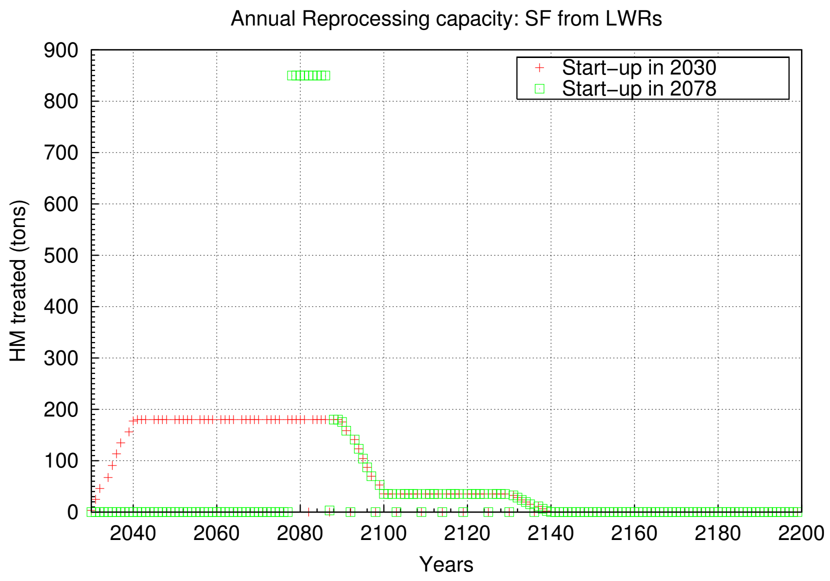

Figure 5.46: Annual reprocessing capacity assuming different options for reprocessing start-up: LWR fuel reprocessing plant

The differences are mainly related to the annual reprocessing capacity during the first years of FRs introduction. In fact the cumulative LWR SF reprocessing capacity for the two options is fully comparable, as expected and indicated in Figure 5.48.

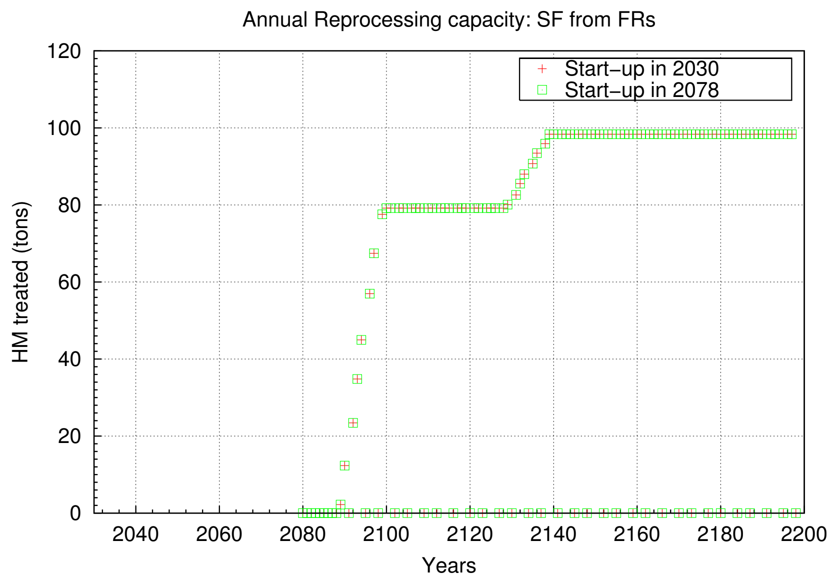

Figure 5.47: Annual reprocessing capacity assuming different options for reprocessing start-up: FR fuel reprocessing plant

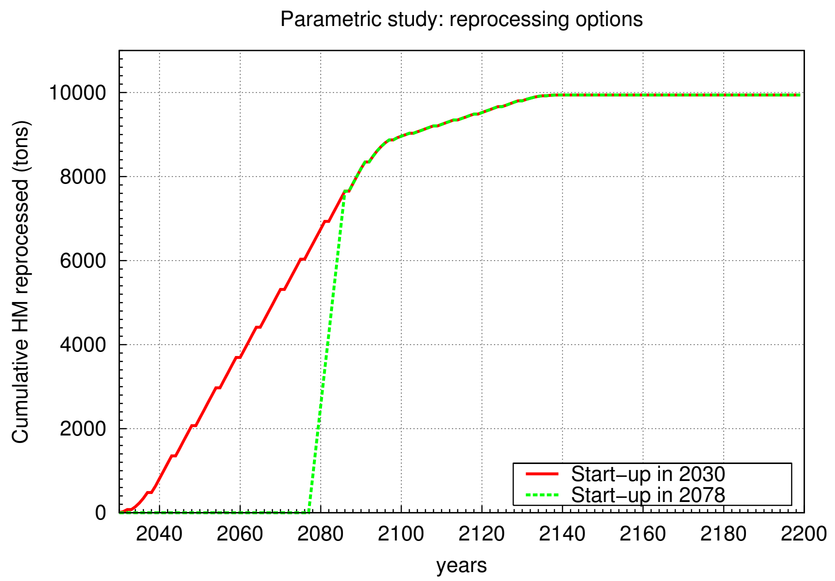

Figure 5.48: Cumulative reprocessing capacity for LWRs assuming different reprocessing options

5.2.4 Other parameters

The parameters previously described are the most important ones in a transition scenario study. However, several other parameters can be considered for the sake of completeness.

Within the study, three other parameters have been analyzed:

- different load factor (case studied: EFR);
- transport versus diffusion approximation for generating the library (case studied: ELSY);
- adoption of CESAR5 module and separated FPs treatment.

Impact of Load Factor: EFR case

In order to investigate the effect of the load factor, the EFR case has been selected. Similar results are expected when using the other systems.

The load factor adopted as reference is 85% (value adopted in previous simulations). This value has been compared with a lower load factor (76%), in agreement with the study presented in [20].

The load factor influences the energy produced every year and, therefore, the number of systems needed to cover the 66.7 TWhe/y of target energy. In particular, assuming a load factor equal to 85%, one EFR reactor produces in one year 10.8 TWhe/y resulting in ca. 6.2 EFR systems needed to cover 66.7 TWhe/y. If the load factor is reduced to 76%, the energy produced in one year is 9.7 TWhe/y and the number of units is increased to 6.9 units.

A larger number of units corresponds to a larger Pu quantity needed and it can influence the transition to FRs.

In particular, assuming the same energy demand as in Figure 5.2, when adopting a load factor equal to 76%, a "lack of material error" appears in the simulation. Therefore, with the assumption of a lower load factor it is not possible to cover 100% of the demand in a single step in a EFR scenario starting from 2080. This is also indicated by the analysis of the Pu balance in Pu stock shown[17] in Figure 5.49.

In order to avoid this aspect the start-up of FRs has been shifted by 10 years (from 2080 to 2090).

This delay on deploying a 100% FRs fleet has an influence on the natural uranium consumption as indicated in Figure 5.50. The uranium demand is increased of about 17% compared to the reference EFR case.

Same small differences can be noticed also for Pu and MAs in the cycle (e.g. MAs in the cycle shown in Figure 5.51). It depends on the energy demand (shifted by 10 years) and it gives a low effect to the total MAs.

Transport versus Diffusion approximation: ELSY case

An additional aspect analyzed is the adoption of the diffusion approximation versus the transport approach for the cross-sections definition.

Starting for the reference ELSY mode (for details see Appendix D) a new COSI6 library has been generated using diffusion approximation.

The results have shown that for a critical system the differences in terms of scenario results are negligible. An example of that is the Pu content of the fresh batches loaded in the core as shown in Figure 5.52.

This value has been chosen as example because it is affected both by the one group cross-sections calculated by transport or diffusion approximation and by the ω values (also calculated under transport or diffusion approximation).

[17] The Pu in stock goes to zero.

As indicated in Figure 5.52 and Figure 5.53[18] the difference is minor, and it has no effect on the main scenario parameters as uranium consumption, waste inventory and radiotoxicity.

CESAR-4 versus CESAR-5 (FPs treatment)

As mentioned in Par. 5.1.1, two burn-up/depletion modules are available in COSI6 [20].

For the study, the CESAR4 module has been applied. Due to this choice the contribution of FPs to the radiotoxicity and heat load associated to the waste in disposal has not been taken into account.

The approximation is considered acceptable for the current purpose because neglecting FPs contribution gives essential differences only for the time period of the first 300 years[19] of the evolution and not in the period of time in which the options on Pu and MAs management are important (1,000 - 10,000 years).

However, in order to evaluate the impact of applying different burn-up modules, a comparison between calculations based on CESAR-4 and CESAR-5 has been performed.

The scenario considered is based on the ELSY model (reference model without blanket), with a single introduction step in 2080 (reaching 26.82 TWhe/y).

The Pu and MAs content in the fresh fuel is compared in Figure 5.54 and Figure 5.55. The two modules work in a comparable manner.

Nevertheless, some differences exist. Assuming that cross sections, "Pu239 equivalent" and omega values have remained unchanged, the only differences resulting from the irradiation performed by the module for determining the composition of discharged batches[20]. Minor changes in SF isotopic composition will then propagate to reprocessed and fabricated fuel. This aspect can modify the TRUs composition disposed in 2200 but the effects are limited (see Table 5.16 for details).

Therefore, the main difference for back-end parameters is related to the treatment of FPs. In Figure 5.56 is represented the comparison for the first 500 years of the radiotoxicity (ingestion) evolution calculated with the two modules. The heat load for the same period of time is indicated in Figure 5.57.

[18] The period 2140-2160 has been considered as an example for showing the small differences underlined.

[19] Except the long-lived FPs, as Tc99 and I129.

[20] Due to the unavailability of the source files, this aspect can not be checked looking at e.g. transmutation and decay chains adopted internally to each module.

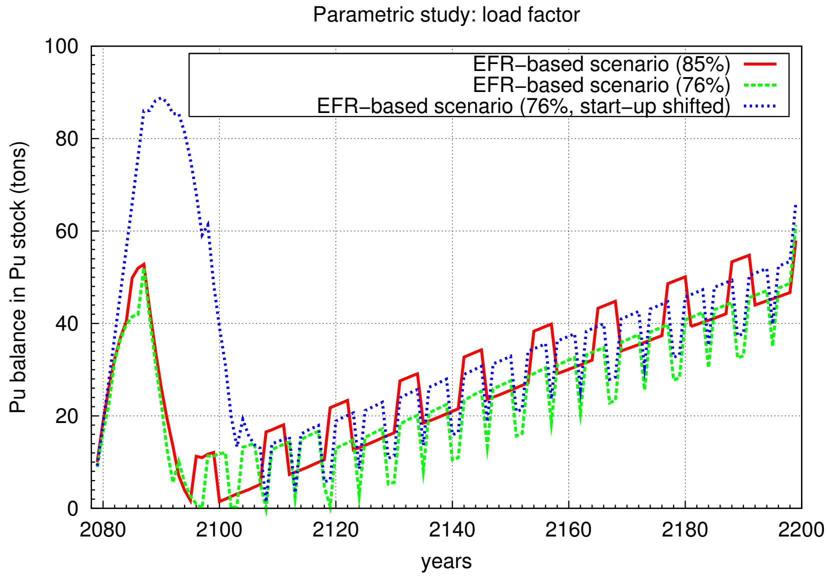

Figure 5.49: Pu stock: Pu mass balance comparison for the EFR scenario (comparison of different load factors)

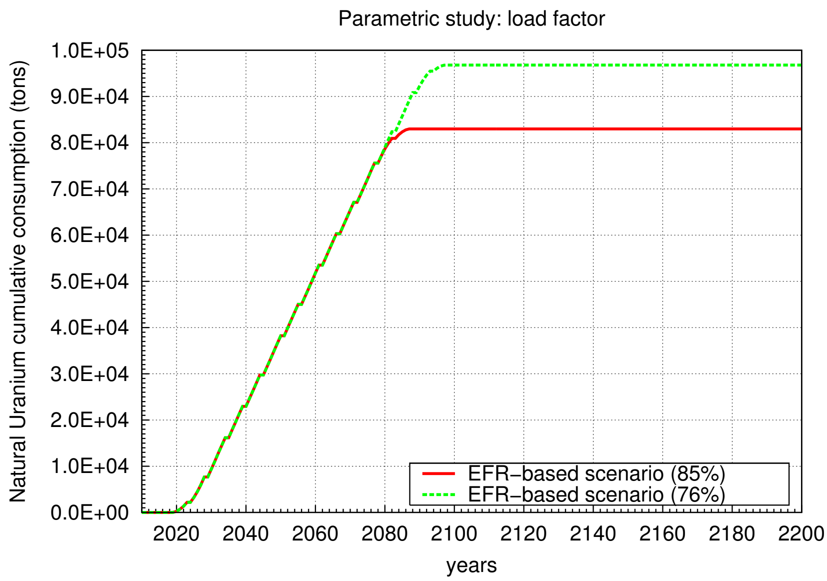

Figure 5.50: Natural uranium cumulative consumption: EFR scenario with different load factors

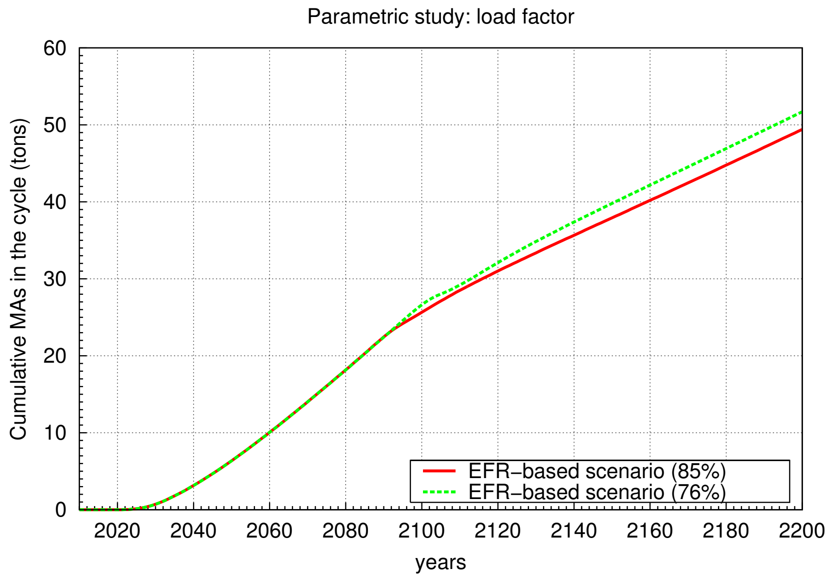

Figure 5.51: Cumulative MAs (Np, Am, Cm) in the cycle: EFR scenario with different load factors

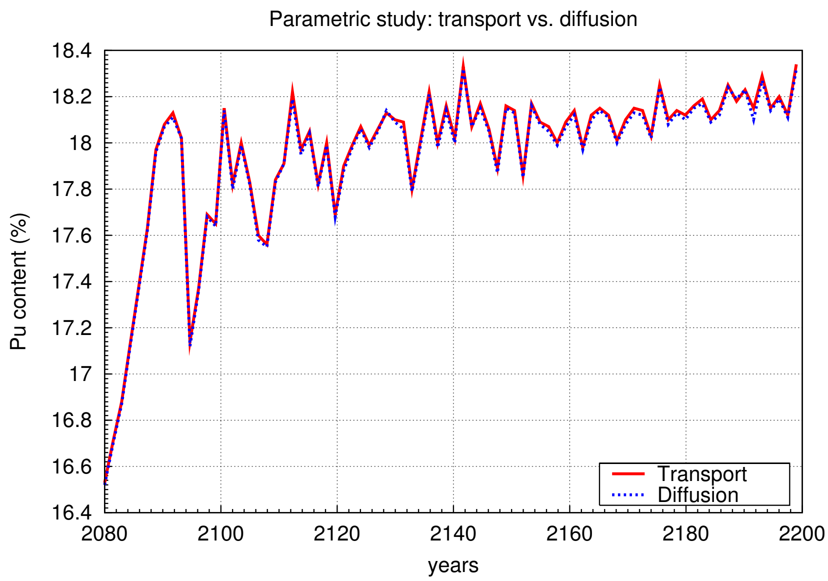

Figure 5.52: Pu content in fresh fuel assuming transport or diffusion approximation (ELSY case)

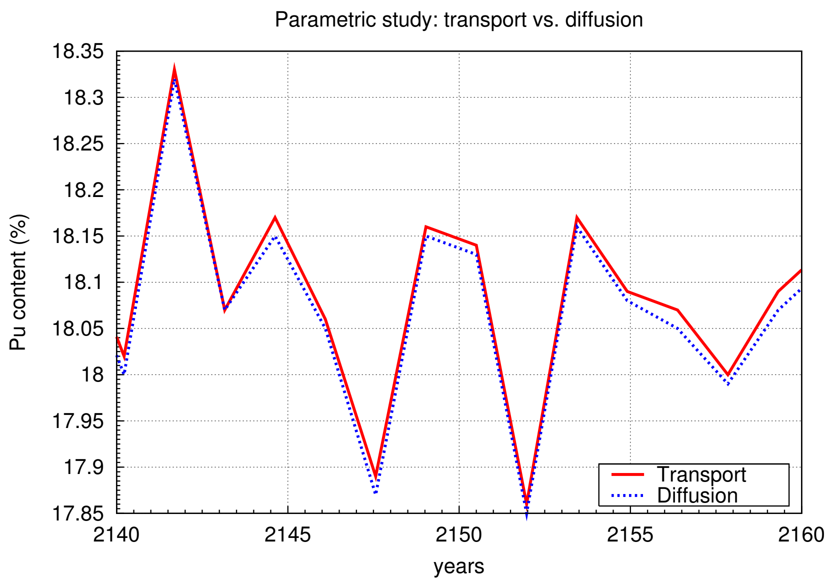

Figure 5.53: Pu content in fresh fuel assuming transport or diffusion approximation: zoom for the period 2140-2160 (ELSY case)

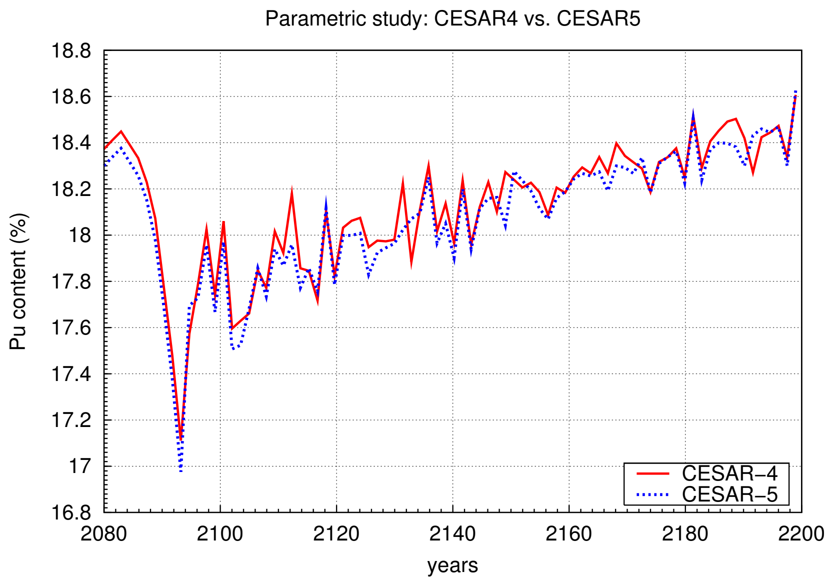

Figure 5.54: Pu content in fresh fuel: CESAR4-CESAR5 comparison

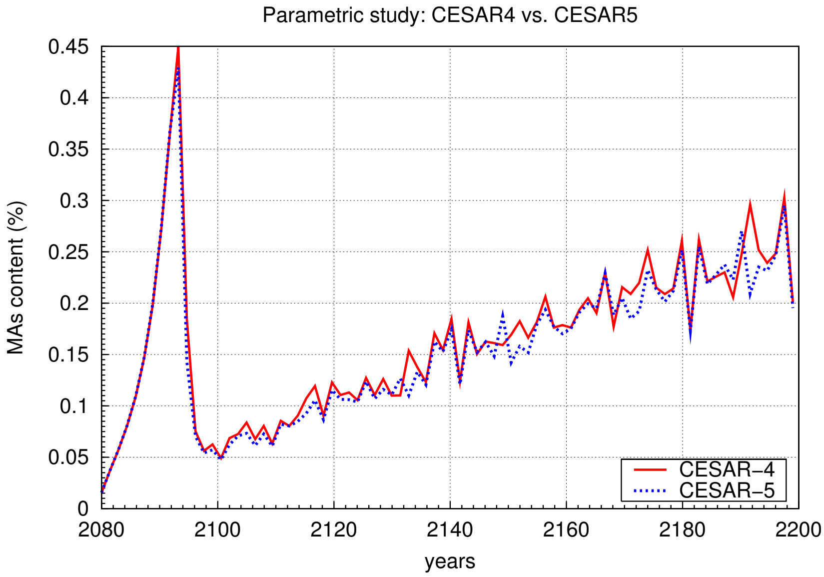

Figure 5.55: MAs content in fresh fuel: CESAR4-CESAR5 comparison

Composition Disposed in 2200		
Year	CESAR-4	CESAR-5
tons		
U234	0.013	0.033
U235	0.143	0.145
U236	0.116	0.122
U238	20.687	20.693
Pu238	0.017	0.082
Pu239	0.792	0.775
Pu240	2.216	2.065
Pu241	0.008	0.008
Pu242	0.085	0.099
Am241	30.204	29.903
Am242M	0.403	0.388
Am243	10.871	10.189
Np237	19.550	18.852
Np239	0.000	0.000
Cm242	0.001	0.001
Cm243	0.015	0.011
Cm244	0.410	0.385
Cm245	0.206	0.196
Cm246	0.021	0.020
Total	85.761	83.970

Table 5.16: Composition in disposal in 2200 evaluated by CESAR-4 and CESAR-5

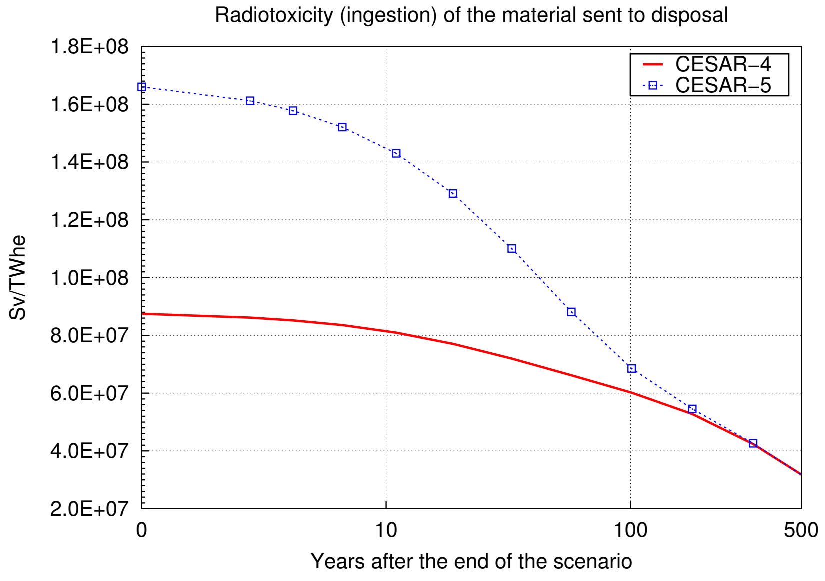

Figure 5.56: Specific Radiotoxicity (ingestion) evolution of the material sent to disposal. CESAR4-CESAR5 comparison [2200 is fixed as t = 0]

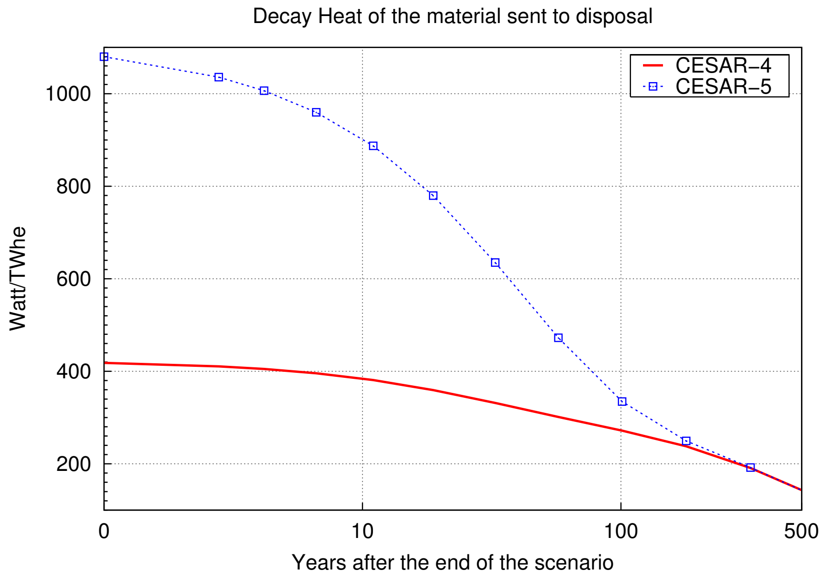

Figure 5.57: Decay heat evolution of the material sent to disposal. CESAR4-CESAR5 comparison [2200 is fixed as t = 0]

5.3 Summary

In the present Chapter, the analysis of the transition scenario from a fleet "LWRs only" to a mixed fleet "LWRs and FRs" has been described.

For the second case, advanced fuel cycles options have been compared in terms of fuel cycle front-end and back-end characteristics.

In particular, three innovative FR concepts (ELSY, a medium size lead cooled critical system, and the ESFR and EFR large size sodium cooled systems) have been considered under dynamics conditions for the time evolution of the NPP fleet.

It has been pointed out that, if one assumes that the only Pu available is the Pu produced within the cycle selected by a specific country (i.e. no "spar" Pu available), the breeding potential of the systems and the core power density are the key characteristics that drive the transition.

In particular, the adoption of the ESFR (or EFR) cores in the mixed fleet enables the complete transition toward the new fleet within the timeframe of ca. 130 years. The advantages of the advanced closed cycle with respect to the open cycle associated to the "LWRs only" case have been pointed out e.g. reduction of 50% natural uranium resources needs, reduction of 30% MAs to be sent to disposal, reduction of one order of magnitude for the radiotoxicity in the period 1,000-10,000 years.

However, adopting a Pu-only multi-recycling strategy, the MAs inventory continues to increase. To avoid that build-up that prevents the optimization of the geological disposal characteristics, a MAs multi-recycling strategy has been considered. In particular, the homogeneous burning of MAs has been adopted for the present investigation.

This option enables to stabilize the MAs in the cycle and to drastically reduce the radiotoxicity and the heat load in the final repository.

Other features of the fuel cycle (reprocessing and fuel fabrication needs) have also been quantified.

Chapter 6

Summary, Conclusions and Outlook

Nuclear energy can play an important role in the coming decades and centuries in order to meet (a) the growing world energy demand and (b) sustainability requirements using environmentally acceptable energy sources.

In the present work, the challenges, implications, boundary conditions, assumptions and correlated consequences of different scenarios suggested or envisaged for the deployment of nuclear energy (mainly for electricity production) on a global, regional or local basis have been investigated.

The activity has been focused on the analysis of the long-term sustainability of nuclear energy in terms both of resources optimization and radioactive waste minimization. In this context, advanced fuel cycles and reactor concepts have been investigated. Several scenarios based on P&T have been compared analyzing, for each choice, the effects on fuel cycle front-end and back-end.

For this purpose, resources needs (e.g. consumption of natural uranium) infrastructures deployment, waste inventory, and associated potential risk for each scenario have been quantified.

The results of the study have confirmed that, when starting on the basis of conventional or advanced LWRs, the adoption of innovative reactors (mainly fast spectrum systems) and closed fuel cycles are crucial for the long-term sustainability of the nuclear option.

In this context, several fast reactor concepts (sodium and lead cooled, self-sustaining or slightly breeders) have been compared. Several closed fuel cycle options (e.g. Pu only retrival and recycle, full TRUs retrieval and recycle, Pu and selected MAs retrieval and recycle) have been investigated.

The scenario analyses have proved to be a powerful tool for this purpose as confirmed also by several European and International projects (e.g. [5, 55]).

In order to perform scenario analyses, several boundary conditions need to be fixed. The first part of the activity has been oriented to investigate how the boundary conditions are defined and which are the most important ones for a scenario study.

Among them, the energy demand and the FRs breeding characteristics have been analyzed in detail since they can significantly affect the results changing the general trends obtained.

Other parameters (e.g. load factor or core burn-up) have been analyzed too and their influence on the general trends has been found to be marginal.

In order to quantify the impact of the initial hypotheses with regard to the main trends, a parametric study (oriented to thermal and fast reactors) has been undertaken. The outcome is a kind of "database" (including the effect of each parameter studied separately) that can be used for quantifying the potential uncertainties associated to more complex scenarios.

In our study, the implications of the LWRs long-term utilization (resources availability, fuel cycle facilities needs, waste production and impact on a geological storage etc.) and the possible transition to FRs (in terms of Pu availability, enlargement of fuel reprocessing and fabrication capacities, etc.) have been key points of the analysis.

In this study the boundary conditions were defined on the basis of a simplified methodology.

This methodology has been considered the starting point for reducing the uncertainties involved. The characterization of the initial situation, in terms of age and type of the reactor fleet in operation (if any) and of the nuclear energy deployment strategy, has been the first step to be performed. This implies the adoption of different options (also in terms of systems) if ongoing nuclear energy or phasing-out strategies are considered.

The analysis of future trends for nuclear energy demand has also been considered an important aspect for the definition of the scenario boundary conditions. In fact, the hypothesis chosen influences the selection of the systems: e.g. different increasing rates of nuclear energy demand can imply the adoption of different types of FRs, self-sustaining or breeder systems.

As part of the methodology, a matrix for the scenarios comparison has been fixed "a priori". Suitable indicators have been selected from the literature in order to point out the sustainability of nuclear energy deployment.

Among them, specific indicators for the fuel cycle front-end and the back-end have been selected, i.e. the resources involved, the inventory of waste to be disposed, the radiotoxicity and the heat load associated to the material sent to the repository (in order to evaluate the potential risk associated). These indicators have been selected looking also to the most important aspects that can affect the social acceptability of nuclear energy production. All these parameters have been evaluated for each scenario treated in the study.

In addition, the infrastructures needs to sustain the fuel cycle have been considered as an important feature to be investigated. In fact, the challenges associated to the infrastructures (e.g. reprocessing plants) can result in limitations for the scenario development.

Other parameters, like "the cost of TWhe" or other economical aspects have not been analyzed in this work. Further effort would be needed to include these indicators.

In the first part of the activity, mainly oriented to boundary conditions investigation, the NFCSS fuel cycle code (developed by IAEA) has been applied primarily during the activities at Pisa University. The results of these preliminary scenarios studies (mainly related to regional areas) have shown a limited flexibility of this code.

Indeed, the fixed structure of the code does not enable the modelling of dynamic aspects of transition scenarios and closed fuel cycles.

Therefore, the COSI6 code developed at CEA-Cadarache, has been adopted for the second part of the study mainly performed at KIT.

However, the preliminary scoping study performed with the NFCSS code has shown several important points. By the analysis of regional-based scenarios, the dominating parameters that affect appreciably the results have been identified. The impact of those parameters over the selected indicators has been successively quantified by a detailed parametric study performed with COSI6 code.

The preliminary scoping study has also indicated which characteristics are suitable for the definition of a reference scenario in order to extrapolate the results to a rather general regional context.

The main characteristics are related to the nuclear energy demand: the reference scenario should be based on a small enough energy share of nuclear to be considered the "unit of measure" for regional studies and large enough to investigate in detail some hypotheses (e.g. substitution of the LWRs fleet).

The implications of the introduction of thermal and fast systems has been studied.

As reference fuel cycle, the "once-through" strategy where only EPR-like systems are deployed, has been considered. This scenario gives first indications on the natural uranium demand, waste produced and facilities (e.g. fuel fabrication) needs to sustain the cycle up to 2200.

In terms of resources, the impact of the chosen reference share of nuclear energy production (70 TWhe/y), in 2100, is quite low (ca. 0.7%) of the total conventional resources world-wide estimated but it becomes important if a higher energy demand is considered (e.g. considering the actual nuclear energy production in Europe, the U required in 2100 could be assessed, by a scaling factor, to be 9% of total world conventional resources).

Other interesting parameters, like the total waste produced and the potential Pu availability in the cycle, have been investigated too. In particular, the analysis of the Pu availability is the most important parameter driving a potential transition toward an FRs based fleet.

By the parametric study carried out, some possible alternatives to the reference scenario have been considered in order to take into account the variation of presently unknown or uncertain parameters like a possible delay-time for the site licensing (e.g. considering different LWRs introduction rate) or the burn-up and cooling down time period of the unloaded fuel. The quantification of the effect of these parameters has been one of the goals of the study.

After the analysis of the thermal reactor and "once-through" fuel cycle scenario, it has been shown how the introduction of the advanced fuel cycles, based on fast systems and closed fuel cycle (i.e. spent fuel reprocessing) can favorably affect the cycle front-end (by resources optimization) and the back-end (by waste inventories and long-term potential reduction of heat load and radiotoxicity).

Several FRs have been modeled by the ERANOS code in order to generate suitable data sets for the COSI6 code. The different FRs considered provide comparable results as far as the same fuel cycle strategy (i.e. multi-recycling of Pu in FRs) is assumed. This result was expected due to the specific characteristics of the systems: they are all self-sustaining systems.

However, some differences mainly due to the different power density can be highlighted during the transition phase. In particular, a 50% of uranium saving (in 2150) is obtained by the total substitution of the LWRs fleet with FRs. A 45% Pu reduction in the cycle can be achieved by systems such as ESFR and EFR, even if they are rather weak breeders.

Advantages in terms of radiotoxicity and heat load reduction have been also assessed.

In order to improve the fuel cycle back end parameters, the adoption of partially closed fuel cycles respectively with Am, Am and Cm and full MAs multi-recycling have been investigated. For that specific study, the European ESFR concept has been considered as reference. The results show that it is possible to stabilize the MAs in cycle (ca. 10 tons) assuming a maximum homogeneous loading of 4-5% MAs in the core fuel.

Safety studies have also been performed in order to validate if specific kinds of fuels (i.e. with a rather high MAs content) could be adopted. These results have confirmed the validity of this option.

A parametric study concerning the breeding characteristics of the systems has been performed too. For this purpose, the original ELSY concept has been modified by adding a radial blanket. A clear indication of the required breeding value to allow long-term sustainability has been obtained.

The FRs comparative analysis has shown how the power density affects the transition toward FRs fleet. This parameter is the main limiting factor if the development of nuclear energy is considered for a country in isolation.

The ELSY system has been designed to achieve the Gen-IV sustainability goals (less MAs compared to thermal spectrum facilities and possibility to burn MAs) and, hence, it has not been optimized for a potential future LWRs substitution (the complete substitution of the LWRs fleet is not achievable before the end of the next century). The adoption of radial fertile blanket, considered to improve the ELSY model for scenario point of view, is not sufficient.

The best solution seems to be the adoption of ESFR systems because a complete substitution can be achieved within a reasonable period of time (before 2150) with beneficial influence on the fuel cycle parameters. A period of overlapping of proven technology (LWRs) and a new one (ESFR-like) can be suitable for gaining experience with this technology and the associated fuel cycle.

This option could be also suitable for the adoption of MAs multi-recycling strategies, both from the point of view of scenario results and safety parameters.

The impact of other parameters, such as the reprocessing option or the load factor value, has also been quantified.

In summary the approach used in this study has provided an overview of the aspects to be treated in any scenario analysis. The reference case considered has to be selected in order to make easy the extrapolation of the results to more complex scenarios (e.g. increasing energy demand).

The parametric studies provide the quantification of the impact of several parameters studied in isolation. This approach can then be used to give an estimation of the range of validity (or applicability) of the general trends that have been assessed. The present study has been focused on specific conditions: 1) continuous use of nuclear energy in Europe during the coming years, 2) use of the U fuel cycle and 3) adoption of the most common reactor concepts implemented or studied in Europe (PWRs and FRs with different liquid metal coolants).

However, the same approach could be applied also to other studies with different hypotheses. Different regional studies could be envisaged, based on the introduction of different type of TRUs burners or different types of LWRs, e.g. high conversion LWRs in order to delay the introduction of fast reactors.

An interesting complementary study would be the U-Th fuel cycle as a transition to a full Th-based fuel cycle, in order to investigate the potential effect on repositories, their volume, heat load or source of potential radiotoxicity as well as regarding proliferation resistance.

Finally, different strategies for phasing out the nuclear option in the future, if a new energy mix would become possible, should be investigated. That type of study could point out potential legacies of the nuclear option and how it could be possible to cope with them (e.g. in a coordinated effort between cooperating countries).

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Appendix A

Appendix: Computational tools description

In the following part, the fuel cycle codes, NFCSS and COSI6 codes, adopted in the study are shortly described.

In both cases, the code source files are not available. Therefore, a detailed analysis of the equations implemented in the codes for modeling the fuel cycle facilities can not be added here. Some indications, if available, are provided.

A.1 The NFCSS code

At IAEA, scenario studies have been developed since the International symposium on "Nuclear fuel cycle and reactors strategies: adjusting to new realities", June 1997. Different scenarios of energy consumption including nuclear electricity production have been studied [16].

IAEA developed the NFCSS code (called also VISTA code) with the aim to calculate, by year over a very long period, nuclear fuel cycle requirements for all types of reactors [16]. In the code two options for cycle back-end (representative of the world situation) have been applied: 1) adoption of a direct disposal, and 2) adoption of reprocessing & recycling strategy.

The NFCSS code enables to perform calculations for a reactor, reactor park in a country or worldwide nuclear power plant park evaluating natural uranium, conversion, enrichment and fuel fabrication quantities.

The code has been developed in order to reduce the input data to few basic parameters (e.g. enrichment of the fuel, burn-up, power of the system). This choice has been done in order to let non-nuclear fuel specialists to develop different energy scenarios.

The calculation speed has been optimized in order to enable the comparisons of different options in a considerably short time.

The time period that can be covered by NFCSS code is quite large (e.g, from the beginning of nuclear energy production to 2050 or 2100). This enables indeed to support estimations for the future, starting from the historical data in its database.

These historical data (including annual load factor and power of the systems) are stored in the IAEA database (PRIS [17]) and they are used for fixing the initial conditions[1] [16].

Future projection data can be calculated by using publications from different institutions. The "Energy, Electricity and Nuclear Power Estimates up to 2030" published by IAEA is one of the authoritative publi-

[1]The absence of clear information does not make easy the modeling of an existing fleet as pointed out in Chapter 3.

cations which is used to calculate future nuclear power projection data in NFCSS [15]. This publication, as well as the NFCSS code, have been adopted in the thesis for the preliminary scoping study (see Chapter 3).

A.1.1 Description

Nuclear Fuel Cycle Simulation System (NFCSS) is a scenario based computer model developed by IAEA for the estimation of nuclear fuel cycle material and service requirements. It focuses on the evaluation of long-term fuel cycle requirements and actinide arisings.

A simplified approach has been adopted for assessing the fuel cycle requirements. In principle, it enable to estimate the long-term fuel cycle service requirements for both open and closed fuel cycle strategies.

The input parameters are included in Figure A.1. The input data are quite simple. They can be easily understood by no-nuclear specialists. It is not required to develop a neutronic model of the systems (producing cross-sections and flux evolution versus burn-up) in order to perform the calculation because reactor models are just included in the code libraries.

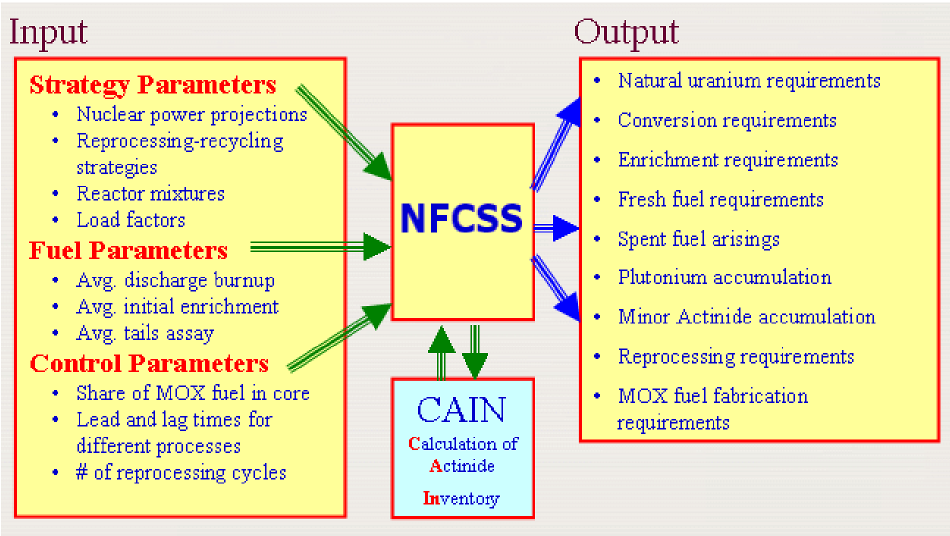

Figure A.1: NFCSS code: input and output parameters

The complete fuel cycle is analyzed, starting from the mining of unused nuclear materials from the nature and ends with the safe disposal of used nuclear material in the nature.

All the processes in the fuel cycle front-end (from uranium ore to fresh fuel) are modeled in the code. As well as the processes in the fuel back-end oriented to the treatment of the spent fuel including temporary storage, reprocessing, long-term storage, or final repository.

The basic schematic illustration of nuclear fuel cycle with recycling in thermal reactor is shown in Figure A.2.

In particular, the following facilities have been considered in the study:

- Mining & Milling: The uranium ore needs to be mined and then processed (milled) before being usable. Uranium ore is mined by open-pit or underground mining methods and the uranium is extracted from the crushed ore in processing plants or mills using chemical methods. Sometimes it is possible to pass chemical solutions to the ore beds and dissolve the uranium from the ore directly. This process is known as in-situ leaching. This is the first step in a nuclear fuel cycle. The feed for mining & milling process is uranium ore and the product is U_3O_8 compound, which is mostly called

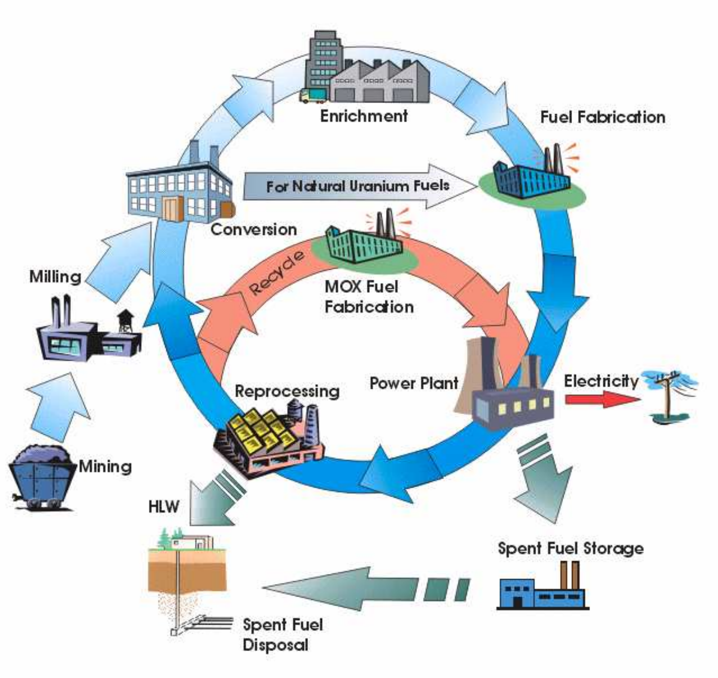

Figure A.2: The basic schematic illustration of nuclear NFCSS fuel cycle code

"yellowcake" due to its color. In the NFCIS database, the commercially operating Uranium Milling Facilities are indicated.

- Conversion: The term conversion refers to the process of purifying the uranium concentrate and converting it to the chemical form required for the next stage of the nuclear fuel cycle. There are three such forms in common usage: metal, oxide (UO_2) and uranium hexafluoride (UF6). UF6 is the predominant product at this stage of the nuclear fuel cycle since it is easily converted to a gas for the enrichment stage, as employed in worlds most common reactor types (LWRs). In the NFCIS database, the commercially operating Conversion Facilities are indicated.

- Enrichment: Uranium naturally consists of about 0.7% of U235 isotope which is the main energy source (fissile isotopes) in thermal reactors. For LWR technology which is the most common reactor type, it is impossible to build a nuclear reactor with the natural occurrence of U235, so the U235 content should be increased with a special process. This process is called enrichment. There two commercially available technologies: gaseous diffusion and centrifuge. Both techniques are based on the slightly different masses of the uranium isotopes nuclei. So the enrichment is defined as the process of increasing the amount of U235 contained in a unit quantity of uranium. The feed for this stage is Natural UF6 and the product is enriched UF_6. The other output of the process is the uranium which has lower fissile content than the natural uranium. It is known as enrichment tail or depleted uranium. In the NFCIS database, the commercially operating Uranium Enrichment Facilities are indicated.

- Fuel Fabrication: Enriched uranium in UF_6 form is converted to UO_2 powder to make fuel for LWR technology. This powder then is formed into pellets, sintered to achieve the desired density and ground to the required dimensions. Fuel pellets are loaded into tubes of Zircaloy or stainless steel, which are sealed at both ends. These fuel rods are spaced in fixed parallel arrays to form the reactor fuel assemblies. The whole process is referred as fuel fabrication. The similar procedure is adopted for natural uranium oxide fuel for some reactor types. The feed of this process is enriched or natural uranium oxide powder and the product is fuel assembly. In the NFCIS database, the commercially operating UOX Type or MOX Type Fuel Fabrication Facilities are indicated.

- Reactor: There are currently 7 types of reactors in the world (classification is based on NFCSS assumptions): PWR, BWR, PHWR, RBMK, GCR, AGR, WWER. The same 7 systems are modeled within the NFCSS code. The feed for reactor is fresh fuel containing uranium and plutonium, in case of Mixed Oxide (MOX) fuel, for existing nuclear fuel cycle options. The product is the spent fuel consisting of new nuclides such as fission products (Cs, I, Tc), minor actinides (Np, Am, Cm) and Plutonium as well as the uranium. The biggest part of the spent fuel is still uranium. The reactors worldwide in operation are included in the IAEA PRIS database.

- Reprocessing: The spent nuclear fuel still consists of significant amount of fissile material that can be used to produce energy. The considerable amount of U235 is still contained in the spent fuel and there are new fissile nuclides that were produced during normal operation of nuclear reactor such as Pu239. Some nuclear fuel cycle options consider taking out the fissile material from the spent fuel, re-fabricating it as fuel and burning in reactor. MOX fuel is the most common fuel that uses reprocessed material. Reprocessing process is based on chemical and physical processes to separate the required material from spent nuclear fuel. The feed of this process is spent fuel and the products are reusable material and High Level Wastes (HLW). In the NFCIS database, the commercially operating Spent Fuel Reprocessing Facilities are indicated.

- Storages: Several storages can be defined. The Spent Fuel Storage contains the spent fuel not reprocessed stored or in pools (wet type) or in silos (dry type). The HLW Storage contains the waste from fuel fabrication and reprocessing facilities classified as HLW and requires careful treating. The Spent Fuel Conditioning storage contains the spent fuel prepared for reprocessing or conditioned for further storage or disposal. The Spent Fuel Disposal contains the spent fuel that can be disposed in deep geological formations (after being properly conditioned). In the NFCIS database, the commercially operating Storage Facilities are indicated.

In the NFCSS code, the material flow between consecutive facilities is model in detail. NFCSS code is in principle able of simulating different nuclear fuel cycle models with different reactor and fuel types (UOx or MOX fuel) including non-existing fuel types (fuels with MAs content) with the introduction of accurate libraries and data.

Since commercially existing nuclear fuel cycle options in bulk amount are "once-through" fuel cycle and U and Pu recycling in some reactor types, these two options are illustrated in Figure A.3. For this two options the code seems to provide reasonable results. However, the fuel cycle structure is fixed limiting the applicability of the analysis.

In this fuel cycle, two fuel types are simulated. The first fuel type in the model is Uranium fuel from natural material whereas the second fuel type is the fuel using reprocessed material. The second fuel type in the system is mostly Mixed Oxide (U+Pu) fuel type since it is the only commercially available fuel from reprocessed material.

The heavy metal starts flow from its natural location (U mine) to the its end location which is spent fuel storage or HLW storage in NFCSS modeling. NFCSS does not include final repository of spent fuel in its model. Therefore, any associated indicators, as radiotoxicity and heat load, can not be evaluated.

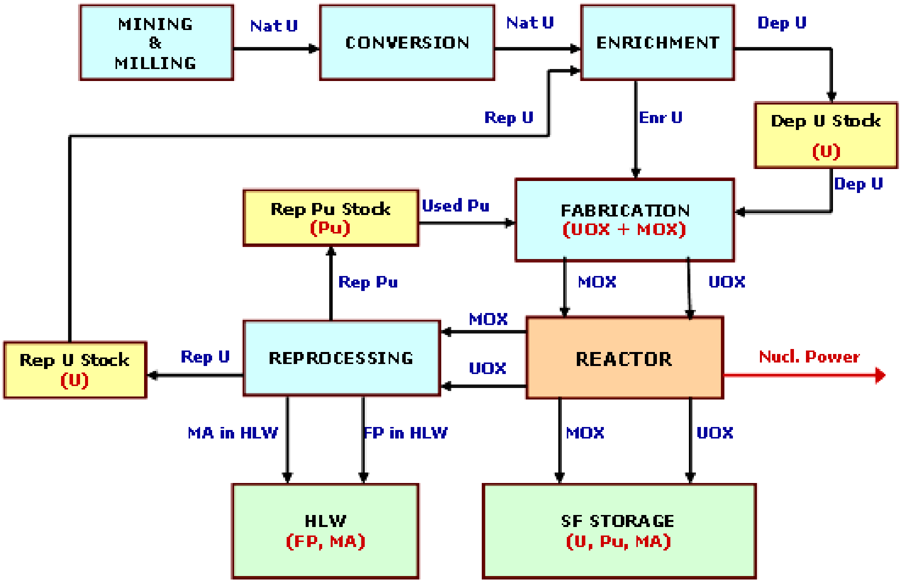

Figure A.3: NFCSS code for "once-through" model and Pu recycling in LWRs

The time associated to every facility of the cycle is taken into account. Some reference values are indicated in Table A.1.

Products or Services	Lead Times
NatU Procurement	2 years before loading
Conversion to UF6	1.5 years before loading
Enrichment	1 year before loading
UO2 fabrication	0.5 year before loading
Spent fuel storage in the reactor pool before transport	2 years after unloading
Spent UO2 or URE(*) storage before reprocessing	5 years after unloading
Spent MOX storage before reprocessing	5 years after unloading
Reprocessing (Pu and RepU availability)	1 year
MOX fuel fabrication	1 year
URE fuel fabrication	0.5 year

Table A.1: Reference time for each facility in the cycle

The NFCSS code includes a depletion module (called CAIN) that is the most important part of the nuclear fuel cycle simulation system. In fact, it calculates the inventory of spent nuclear fuel after irradiation and uit gives the isotopic contribution of the batch at end of irradiation.

As other more known codes (e.g. ORIGEN2.2), the CAIN module solves Bateman's equations for a point assembly using one group neutron cross sections.

In the NFCSS distribution, 7 different reactor types and therefore XS libraries are available: PWR, BWR, PHWR, RBMK, AGR, GCR, WWER.

The simplified depletion module considers the 28 reaction and decay chains during irradiation and 14 decay chains during cooling/storage. The nuclides are chosen on the basis of the importance on SF radiotoxicity and their nuclear characteristics. Some simplifications are applied also for the natural uranium, were U234 (abundance < 0.01%) is ignored. In addition, nuclides with short half lives (half-life < 8days) are ignored as U237, Np238, Pu243, Am242, Am244 and Am244m.

Simplifications are applied also to long half-life nuclides (half life > 400years). These isotopes are assumed as stable for the irradiation period. For example, Am241 ($t_{1/2}$ 432 years) is treated as stable during irradiation. For decay (cooling) period after discharge, all nuclides are treated by their actual decay scheme.

The chain considered in the CAIN module is represented in Figure A.4.

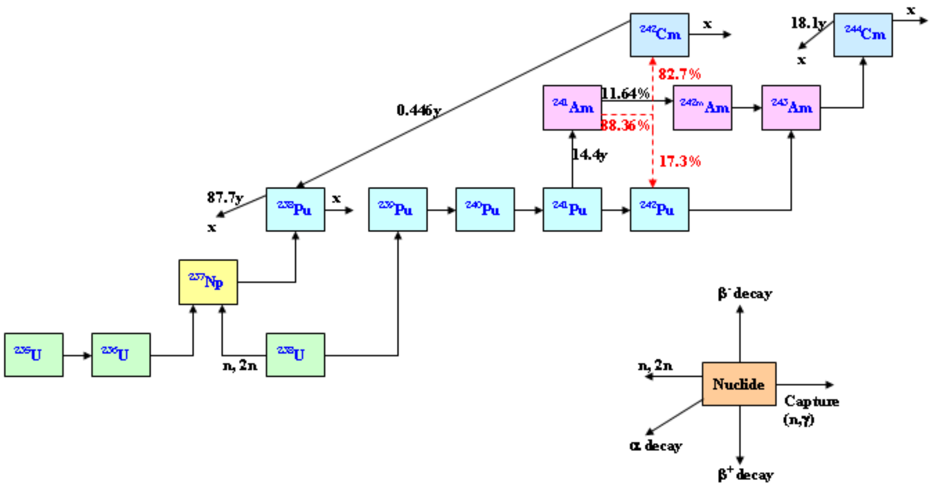

Figure A.4: Isotopic chain implemented in CAIN depletion module

The 28 reaction chains and 14 decay chains are selected to be suitable for fresh fuels containing any of the 14 nuclides of the CAIN library. Some reaction chains are neglected due to their contribution to the composition of the spent fuel, such as chains starting from decay of Am241 (432 years).

In total 14 nuclides, decays of Pu238 (87.7 years), Pu241 (14.4 years), Cm242 (0.447 years) and Cm244 (18.1 years) are considered during irradiation.

In summary, all the nuclides included in the calculations are included in Table A.2.

Element	Isotopes
Uranium	U235 U236 U238
Neptunium	Np237
Plutonium	Pu238 Pu239 Pu240 Pu241 Pu242
Americium	Am241 Am242m Am243
Curium	Cm242 Cm244

Table A.2: Nuclides considered in the NFCSS code [16]

A.2 The COSI6 code

The other fuel cycle code adopted in the study is the COSI6 code developed by CEA [187, 20].

This code has been designed to simulate the operation of a fleet of nuclear power plants and associated fuel cycle facilities.

The level of accuracy of the code and the knowledge needed to perform a simulation with this code are much extended than in the case with NFCSS code. In fact, detailed neutronic analyses are at the basis of the

COSI6 simulations. For this reason, the ELSY and ESFR systems simulated in COSI have been modeled by the ERANOS code as described in Appendix D.

In particular, COSI6 code has been developed to perform scenario studies making it possible to better understand and analyse the consequences of choices made not only in terms of the size and nature of the fleet of reactors but also in terms of the characteristics of the different fuel cycle facilities (plants, interim storage sites and final repositories) [187].

The code is able to manage several kind of materials: natural uranium, depleted uranium, reprocessed uranium, Pu, Np, Am, Cm, waste, FPs, new fuels (like inert matrix fuels) and irradiated fuels. This large range makes COSI6 code to be suitable for several kind of fuel cycles.

All the flows of materials in the cycle are determined by the requirements in nuclear-generated electricity.

The nuclear power plants generating this energy (i.e. gross energy production) govern the mass flow associated to the front-end of the cycle in terms of fuel fabrication operations, extraction & conversion and uranium enrichment.

These materials flows are also connected to the cycle back-end concerning storage of irradiated fuel and final destination. In case of partially closed fuel cycle, the model of fuel reprocessing plants (even separated for each kind of SF) is allowed by COSI code.

In order to assess the material sent to disposal, the finished warehouses governs the waste path from interim storage to disposal.

The structure is flexible as indicated in Figure A.5. The user can decide to implement several facilities in the cycle as indicated by the blank rows in Figure A.5.

Therefore, the simulation of double-strata and closed fuel cycles are feasible making the COSI6 code very actractive for the study.

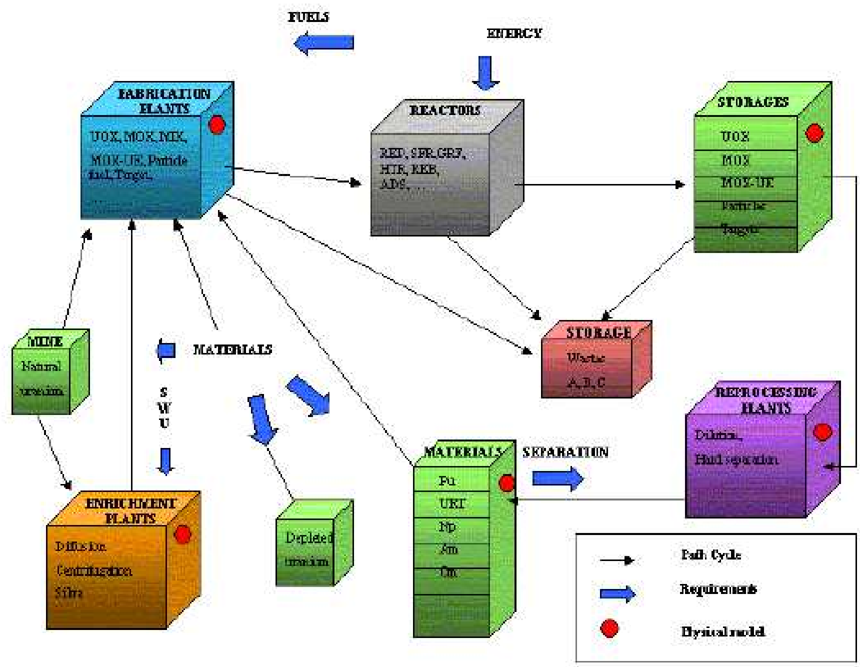

Figure A.5: COSI6 fuel cycle scheme

For the modelization, the starting point is the energy demand produced by the nuclear fleet and the systems considered for covering this energy demand.

Once the mass of the fresh fuel is determined, the fuel element requirements (UOX, MOX, FBR fuel, targets, etc.) with enrichment levels or Pu contents, that are coherent with the management strategy and burn-up of the reactor in question, are evaluated.

These data are driving the front-end plant capacity requirements in terms of natural uranium, depleted uranium, reprocessed uranium, enriched uranium and plutonium [187].

The SF extracted, according to the reloading scheme, impacts the fuel cycle back-end. The options on reprocessing contribute to the definition of the back-end material flow. These plants can operate with an objective defined by the user (case of real plants) or in relation to material requirements for fuel fabrication. Option useful for underling special objectives of the study.

The following main stages can be pointed out:

- Calculation of fuel loading dates in relation to reactor commissioning dates (e.g. beginning and end of the scenario), cycle lengths (irradiation length divided by batch fraction) and load factors (as indicated in Figure A.6).
- Calculation of fuel unloading dates (based on fuel loading dates and irradiation length associated at the specified burn-up).
- Calculation of operating dates for real reprocessing plants (taking into account the time needed for reprocessing (fixed in the study to 0.5 year) and the cooling time needed before reprocessing (i.e. 5 years for SF coming from LWRs and 2 years for SF coming from FRs).
- Execution of the scenario in chronological order of the previously calculated dates.
- The fresh fuel demand calls the fuel fabrication operation. Each fuel fabrication operation calls upon the front-end cycle: fuel fabrication, enrichment, conversion and extraction plants, not to mention material stocks: natural uranium, Pu (assuming partially closed fuel cycles or MOX), depleted uranium, reprocessed uranium, minor actinides.
- Calculation of the initial composition of fuel batches may be based on energy equivalence models, as in the case of MOX. One specific example is the fast reactor fresh fuel composition where it is maintained fixed the "Pu239 equivalent" calculated on the basis of the reactivity equivalent coefficients [196]. More details are included in Appendix C.
- Fuel irradiation is generally calculated using the simplified evolution code CESAR based on the initial fuel composition, burn-up at unloading and irradiation time. CESAR uses cross section libraries that are parameterized with an energy group.

The cross sections libraries are generated by an appropriate chain of codes. In particular, for thermal reactor they are created using APOLLO-APOGENE-COSI6 and for fast reactors are created ba ERANOS-APOGENE-COSI6 [183].

Two versions of the burn-up code, called CESAR code [187] can be used for solving the Bateman equations (Eq. A.1 [198]).

$$\begin{aligned}\frac{dN_i(\vec{r},t)}{dt} &= \sum_{j>1}^{N} l_{ij}\lambda_j N_j(\vec{r},t) - \lambda_i N_i(\vec{r},t) + \sum_{k>1}^{N}\sum_{r}\int_0^{\infty} N_j(\vec{r},t)Y_{k,r}(E)\sigma_{k,r}(E)\varphi(\vec{r},E,t)dE \\ &- \int_0^{\infty} N_j(\vec{r},t)\sigma_{j,a}(E)\varphi(\vec{r},E,t)dE, \qquad \text{(A.1)}\end{aligned}$$

where:

- N_i = number of atoms per unit volume for the isotope i;
- λ_i = decay constant of the isotope i;
- l_{ij} = fraction of the isotope j dacays in the isotope i;
- $Y_{i,r}$ = reaction rate for the isotope i for the reaction r;
- $\sigma_{i,r}$ = microscopic cross section for the isotope i for the reaction r;
- $\sigma_{i,a}$ = macroscopic absorption cross-section of the isotope i;
- φ = neutronic flux.

One version, called CESAR4, is a simplified version that takes into account major and minor actinides main isotopes. The chain is limited to few isotopes (form Th232 to Cm248). The contribution of FPs is taken into account as lumped FPs (with the associated lumped absorption cross-section).

In addition, some short-cuts are included in the chain (e.g. Pu243, Am244) in order to neglect isotopes with very short half-life. Other isotopes, as Cm245 are considered stable due to their long half-life.

This treatment seems not reasonable e.g. in the case of high Cm build-up (leading to a high contribution to the total radiotoxicity after 1,000,000 years.

In order to solve these points, and the correct evaluation of the radiotoxicity during the first 300 years, an other version of the CESAR code, called CESAR5, can be adopted.

In this version all actinides isotopes and ca. 200 fission products are taken into account for the calculation. This version needs higher computational times than the CESAR4 version.

In the present activity, the CESAR4 version has been adopted as reference. In fact, the main attention has been devoted to the fuel cycle for MAs treatment and therefore the contribution of FPs could be neglected.

However, in order to take into account the correct evolution of the isotopes in disposal (considering also the decay of isotopes as Cm245) the composition in disposal in 2200 (year considered as end of scenario) has been extracted by each simulation and let evolve by an additional depletion code. In particular, ORIGEN2.2 code has been considered [194].

In order to calculate the radiotoxicity, the same coefficients implemented in COSI and listed in Table , have been used.

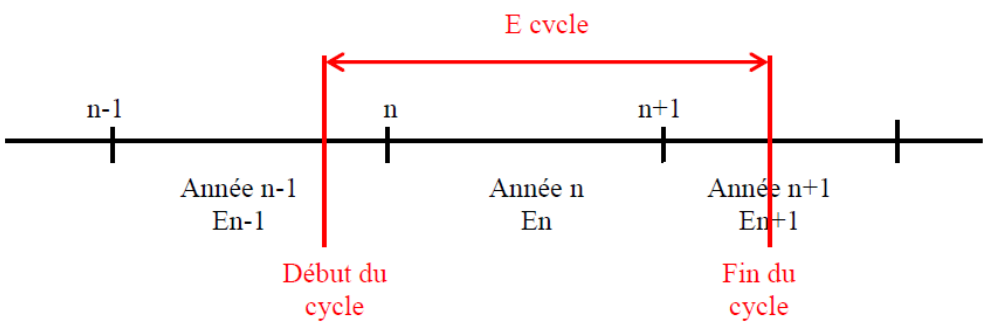

Figure A.6: Definition of loading date in COSI6

Isotope	TSv/ton	Isotope	TSv/ton	Isotope	TSv/ton	Isotope	TSv/ton
PB209	9.58E+00	PB210	1.92E+00	PB211	1.64E+02	PB212	3.03E+02
PB214	1.70E+02	BI210	5.97E+00	BI212	1.41E+02	BI213	1.43E+02
BI214	1.80E+02	PO210	3.99E+01	FR223	3.29E+03	RA223	1.89E+02
RA224	3.83E+02	RA225	1.39E+02	RA226	1.02E-02	RA228	5.81E+00
AC225	5.15E+01	AC227	2.94E+00	AC228	3.57E+01	TH227	1.01E+01
TH228	2.12E+00	TH229	3.78E-03	TH230	1.57E-04	TH231	6.69E+00
TH232	9.33E-10	TH234	2.91E+00	PA231	1.24E-03	PA232	1.14E+01
PA233	6.76E-01	PA234	3.77E+01	U232	2.30E-01	U233	1.79E-05
U234	1.13E-05	U235	3.68E-09	U236	1.10E-07	U238	5.47E-10
U240	3.77E+01	NP235	2.75E-03	NP236	8.29E-06	NP237	2.87E-06
NP238	8.73E+00	NP239	6.87E+00	NP240	3.66E+01	PU236	1.69E+00
PU238	1.46E-01	PU239	5.75E-04	PU240	2.11E-03	PU241	1.79E-02
PU242	3.39E-05	PU243	8.18E+00	PU244	1.57E-07	PU246	5.97E+00
AM241	2.54E-02	AM242M	6.83E-02	AM242	8.97E+00	AM243	1.48E-03
AM246	6.56E+01	CM242	1.59E+00	CM243	3.82E-01	CM244	4.79E-01
CM245	1.91E-03	CM246	3.30E-03	CM247	9.27E-07	CM248	1.73E-04
CM249	1.35E+01	CM250	1.92E-02	BK249	5.88E-02	BK250	2.02E+01
CF249	5.31E-02	CF250	6.47E-01	CF251	2.11E-02	CF252	1.79E+00
CF253	1.50E+00	CF254	1.26E+02	ES253	5.69E+00	ES254	1.93E+00

Table A.3: Radiotoxicity coefficients (ingestion) based on ICRP68 [39]

A.3 The ERANOS code

In order to generate suitable libraries for the COSI6 code, the detailed 3D models of the systems have to be generated with an appropriate neutronic code before the conversion by the interface code APOGENE [183].

For thermal reactor, the reference neutronic code is the APOLLO deterministic neutronic code developed at CEA [184, 183].

This APOLLO2 code is not available outside CEA, therefore, during the Ph.D. thesis only LWRs library available in COSI6 has been used.

Otherwise, for the simulation of FRs appropriate libraries have been generated by ERANOS code (the reference code for FRs).

The ERANOS code is a deterministic neutronic code developed at CEA within a European collaboration with the aim of providing a suitable basis for reliable neutronic calculations of current as well as advanced fast reactor cores [182].

The ERANOS code is a deterministic code system, so neutron physics calculations are performed in two steps: at the cell/lattice level and at the core level.

Internally to ERANOS, the ECCO cell/lattice code is used for processing the effective cross-section for the following core calculations.

The libraries that are provided to ECCO are in a direct access format in various energy meshes: 1968 groups (all-purpose), 175 groups (shielding purposes), the 172-group XMAS scheme (refined in the low energy range), and 33 groups (energy mesh generally used for core calculations) [40].

In particular four sets of libraries are available:

- JEF-2.2 obtained directly from JEF2.2 evaluations,

- ERALIB1 obtained from the JEF-2.2 libraries by a statistical fitting on integral experiments,

- JEFF-3.1 obtained directly from JEFF3.1 evaluations,
- ENDFB-VI.8 obtained directly from ENDFB-VI.8 evaluations [40].

The ECCO code solves the resonant nuclide self-shielding using the sub-group method and computing, with a collision probability method, a fine-group solution of the transport integral equation. The cross-sections can be condensed and homogenized. The resulting broad-group cross-sections, corresponding to an equivalent homogeneous cell, can then be used in core calculations [40, 182]. The geometry that can be solved are listed in Table A.4.

Dimension	Geometry treated in ECCO
0D	Homogeneous
0D	Region
1D	Plane
1D	Cylindrical
1D	Hexagonal
1D	Square
2D	Rettangular Lattice
2D	Hexagonal lattice
2D	Plate

Table A.4: Geometries treated in ECCO [40]

The core calculations are then carried out by ERANOS. The calculations include reactivity, flux, spatial power distribution, reactivity coefficients, burn-up, and control rod worth. Moreover, for very different applications (analysis of experiments, reactivity coefficients, follow-up and management of core loadings), traditional, generalized and harmonics perturbation modules are available [182, 40].

In particular for the study has been applied:

- Reference route for cross-section evaluation: the calculation involves a STEP in fine groups (1969 energy groups) and heterogeneous geometry;
- burn-up calculation: HEX-Z core geometry has been considered. For the flux evolution the TGV-VARIAN nodal method has been applied;
- ERANOS-DARWIN interface: for generating a interface file (called AP2PEP file) to be provide in input to the APOGENE code for the conversion in COSI BBL library. The AP2PEP file containing flux and cross-sections (main actinide isotopes) burn-up dependent (according to the burn-up step considered for the recalculation of the flux, ca. every 100 efpd).

Appendix B

Appendix: Data Available for the Future Energy Demand

As extensively described in the text, the choice of the energy envelops is one of the most important parameters affecting the general trends of a scenario study.

A short overview of the data available has been provided in Par. 2.3.2. In this Appendix more details are included.

In particular, the contribution provided to the activity of the NEA/OECD WPFC Expert group on transition scenario concerning the sanalysis of the energy envelops for the homogeneous-heterogeneous world scenario study is here summarized. The activity is partially summarized in [4, 199].

The available data (for different time periods) are presented and compared with the aim to define a reasonable energy demand for the world study. Some unreasonable trends have been pointed out in order to show how the choice of the energy envelops is a quite complex aspect to be treated. The influence on the major trends of the energy envelopes, adopted as boundary conditions of the study, has been pointed out in Chapters 4 and 5.

B.1 Boundary conditions selection for the Heterogeneous World Scenario Study

Analyzing the energy envelops, two kinds of data have been considered: the global trends to be applied to the homogeneous world study and the regional trends to be applied to the heterogeneous world study.

The analysis has shown that not always the same set of data suitable for the homogeneous approach one can be applied for the heterogeneous study. In order to clarify this aspect, several set of data have been compared as indicated in this Appendix.

In general, the process to define future energy demands or envelops is quite complex and it becomes more complex when only a single technology is taken into account (e.g. nuclear energy or renewable sources).

Different heterogeneous sectors (and their foreseen evolutions) are considered at the same time, e.g. the population and economy growth rates expected, the energy mix or the penetration of a new technology in the market (including the effect of governmental incentives) [3].

Therefore, in order to define possible electricity evolution trends, complex and multidisciplinary models have been created and adopted by some established International Organizations [3, 23], as the Intergovernmental Panel on Climate Change (IPCC), the International Institute for Applied Systems Analysis (IIASA), the International Energy Agency (IEA) and the International Atomic Energy Agency (IAEA).

It could be useful to know that some common points exist between models and approaches adopted. All models link together economical, political and environmental evaluations. The driving forces considered are the Gross Domestic Product (GDP), the use of land, the technological level and the population growth rate linked together in order to provide energy projections and greenhouse gas (GHG) emissions scenarios.

All models, indeed, are based on the so-called Kaya identity (Eq. B.1) with the aim to determine the level of human impact on climate change evaluated as GHG emissions [200, 3] and on the IPAT equation defined in 1970 to evaluated the human impact over the environment (Eq. B.2), [201, 3].

$$CO_2 = Population \times (GDP/Population) \times (Energy/GDP) \times (CO_2/Energy) \quad (B.1)$$

$$IPAT = Population \times Affluence \times Technology \quad (B.2)$$

The differences in the assumptions and relative "weighting factors" given to each parameter enable to obtain different scenarios with different future energy mix.

The data provided by these Organizations have been analyzed and compared in order to chose suitable energy envelops for the world scenario studies.

The analysis has shown that all models have common points for linking together economical, political and environmental evaluations. On the basis of these parameters (e.g. the Gross Domestic Product (GDP), the use of land, the technological level, etc.), they provide the projection for energy demand and for greenhouse gas (GHG) emissions.

In order to refine the analysis and to take into account different situation, a fine subdivision for the world regions can be applied.

As described in Par. 2.3, the IPCC subdivision (4 "macro-regions") has been chosen also as reference for NEA/WPFC world scenario study.

This subdivision has been adopted by IPCC in order to compare models and initial conditions [3]. In order to compare other available data (e.g. data produced by IIASA [23] and IEA [24]) the same way to collapse in four macro-regions has been adopted. In particular, the four macro-regions considered are:

- OECD90: composed by North America (NAM), Western Europe (WEU) and Pacific OECD (PAO),
- REF: composed by Central and Eastern Europe (EEU) and Newly independent states of the former Soviet Union (FSU),
- ASIA composed by Centrally planned Asia and China (CPA), South Asia (SAS) and Other Pacific Asia (PAS).
- ALM composed by Latin America and the Caribbean (LAM), Middle East and North Africa (MEA) and Sub-Saharan Africa.

A brief description of the models and data available is here summarized.

B.2 IPCC Emission Scenarios

In 2000, the Intergovernmental Panel on Climate Change (IPCC) has provided to the scientific community a series of 40 scenarios collected under the name of "Emission Scenarios" and summarized into the *"Special Report of Emission Scenarios"* , knows as SRES report [3].

The main aim of the SRES report is to investigate, on the basis of few selected driving forces and approaches, the panorama of the global future development concerning economical, environmental, and social sectors. In particular, the focus has been to establish the relative environment impact of each scenario in terms of CO_2 emissions (or other GHG emissions in general).

In the study, six different models, representative of different approaches for modeling emissions scenarios and different integrated assessment (IA) frameworks in the literature, have been adopted and compared. The models treated by [3] are:

- AIM Asian-Pacific Integrated Model from the National Institute of Environmental Studies in Japan. It is a large-scale model for scenario analyses of greenhouse gas (GHG) emissions and impacts of global warming (bottom-up model).
- ASF Atmospheric Stabilization Framework Model from ICF Consulting in the USA. It includes energy, agricultural, deforestation, GHG emissions and atmospheric models and provides emission estimates for World's regions. It is focused on four end-use sectors: residential, commercial, industrial and transportation.
- IMAGE Integrated Model to Assess the Greenhouse Effect from the National Institute for Public Health and Environmental Hygiene (RIVM). It is a three modules code (EIS, TES, AOS) linking together the evaluation of the Energy-Industry System (to compute emissions vs. regions), of the Terrestrial Environment System (to simulate global land-use and land-cover changes) and of the Atmosphere-Ocean System.
- MARIA Multi-regional Approach for Resource and Industry Allocation from the Science University of Tokyo in Japan. It is a compact integrated assessment model to assess the interrelationships between economy, energy, resources, land use and global climate change.
- MESSAGE Model for Energy Supply Strategy Alternatives and their General Environmental Impact from the International Institute of Applied Systems Analysis (IIASA). It is a modular model that combines extensive historical data about economic development and energy systems with empirically estimated equations of past economic and energy developments to determine future structural changes. The starting point is the population and the capita economic growth by region.
- MiniCAM Mini Climate Assessment Model from the Pacific Northwest National Laboratory (PNNL). It is a modular code that estimates global GHG emissions with the ERB model (a partial equilibrium model that uses prices to balance energy supply and demand for the seven major primary energy categories: coal, oil, gas, nuclear, hydro, solar, and biomass).

Every one of these models investigates the environmental impacts of the assumptions introduced. All of them could be considered based on the so called IPAT identity (Eq. B.2), equation used to evaluate the environmental impact, or on the KAYA equation (Eq. B.1), equation used to evaluate the CO_2 emissions in particular. These two equations are able to link together the main driving forces as population, GDP and technology level.

The combination of the models and specific assumptions about populations, GDP and technology level (as indicated by Eq. B.2) enables to obtain the 40 SRES scenarios (organized in 4 "families") proposed by IPCC in 2000 [3].

The simplified scheme, shown in Figure B.1, better explains the rationale at the basis of the IPCC scenarios.

The IPCC scenarios can be subdivided in four "families" as represented in Figure B.1.

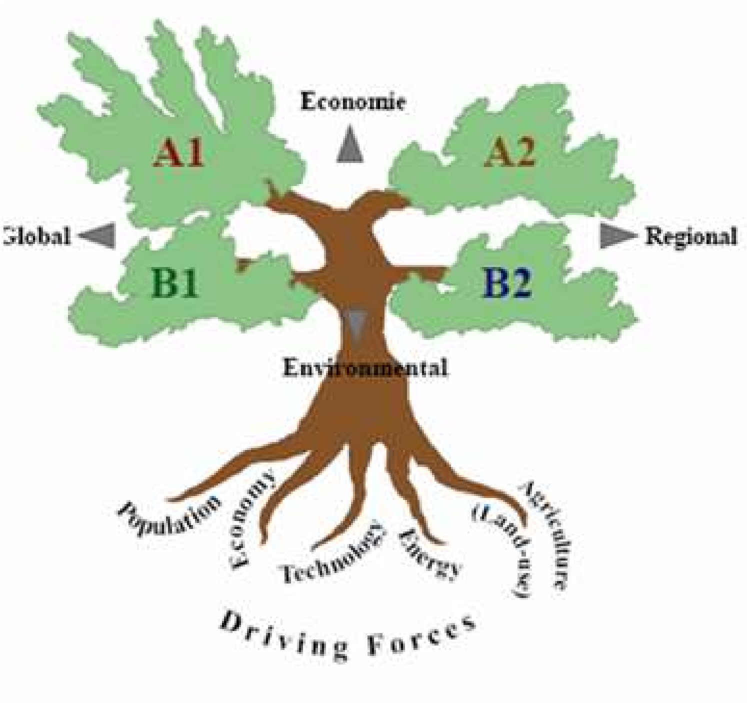

Figure B.1: IPCC scenarios "families" and driving forces [3]

In fact, the 4 "families" represents the "branches" of a two-dimensional tree, where driving force (population, economy, technology, energy, land-use) are the input (the "roots") of the analysis. Within this representation, it is possible to highlight four poles along two major axes representing the approaches: economic vs. environmental and global vs. regional [3].

Anyhow, each storyline describes a global paradigm based on prevalent social characteristics, values and attitudes that determine, for example, the extent of globalization, economic patterns and environmental resource quality [3].

In addition, within each "family", two types of scenarios can be distinguished: the harmonized scenarios (HS) and the "other scenarios" (OS).

The first type scenarios (26 scenarios) have harmonized assumptions about global population, economic growth and final energy use (as the comparison of results can underline).

The second type (14 scenarios) have alternative quantifications of the storyline adopted in order to investigate uncertainties in driving forces beyond those of the harmonized scenarios.

A more detailed subdivision of the 4 "families" is shown in Figure B.2.

In this figure, some scenarios are called "marker" scenarios. They are the ones selected for representing the "family" and they have been published and detailed described in SRES publication [3]. All other scenarios are collected in the SRES database [41].

In agreement with the aim of the present work (heterogeneous World transition scenarios investigation), only B2 family oriented to the environment and to the regional solutions, has been considered. In particular, B2-MESSAGE and B2-MiniCAM scenarios have been analyzed in detail.

The B2-MESSAGE scenario has been studied because it is the "marker" scenario for B2 family proposed by the IPCC publication [3].

Otherwise, B2-MiniCAM scenario has been studied because it has been adopted as reference for the

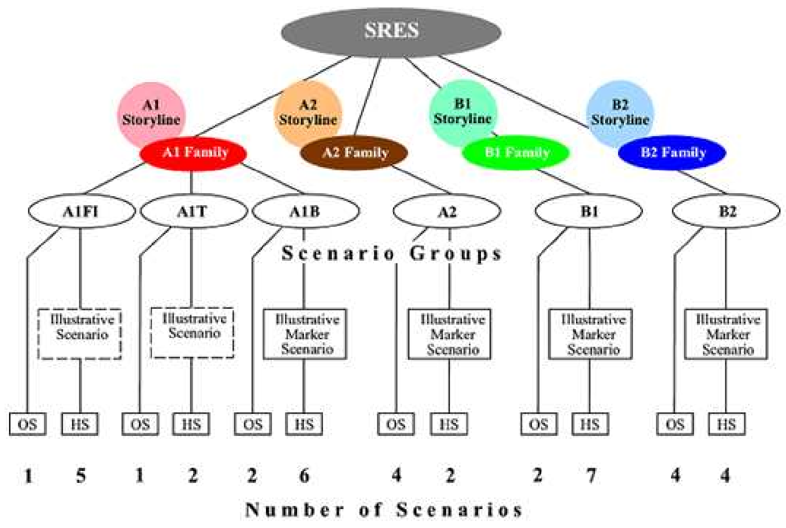

Figure B.2: Schematic illustrations of SRES scenarios: the 40 scenarios divided in HS and OS scenarios [3]

Homogeneous World Transition study [4, 199] as suggested by NEA/OECD WPFC working group [83].

As further term of comparison, the A2 "family" has been briefly analyzed considering the A2-MESSAGE and the A2-MiniCAM scenarios in order to give some examples of scenario more oriented to globalization aspects than environmental [3].

As indicated in Figure B.2, the B2-MESSAGE scenario is an harmonized scenario (HS) whereas the B2-MiniCAM scenario is an alternative scenario (OS). Therefore, the comparison is somewhat difficult due to different starting assumptions on population and economy growth rate. One example is the energy demand (at 2100) evaluated with the two models: the value reached with B2-MESSAGE is almost double the value reached with B2-MiniCAM.

Whereas, some similarities could be found between A2-MESSAGE and B2-MESSAGE (both are harmonized scenarios) or between A2-MiniCAM and B2-MiniCAM (because both are alternative scenarios).

All these aspects, are clarified in the next paragraphs.

B2-MESSAGE

The B2-MESSAGE is the marker scenario for the B2 "family". It has been generated by the MESSAGE code, one of the 5 models that constitute the integrated modeling framework of IIASA (Figure B.3).

Only the nuclear energy demand has been analyzed and presented here. Anyhow, data concerning other sources of energy are available in [41].

As indicated before, the aim of this investigation is the choice of the energy envelop for the world scenario study performed by NEA/OECD. The time period of interest is 2010-2200 [199].

Hence, for the nuclear energy demand, it has been assumed that up to 2100, data come directly from IPCC - B2-MESSAGE [41] database, and for the period 2100-2200, the nuclear capacity remain constant. This second assumption is in agreement with more general studies, e.g. [87].

For the B2-MESSAGE scenario, the contribution of each "macro-region" (OECD90, ALM, REF, ASIA) is shown in Table B.1 where nuclear data (at global and regional level) are indicated in EJ.

Assuming the data in Table B.1, OECD90 increases its nuclear capacity more than 3 times before the end of the century passing from 5.9 EJ in 1990 (that is equal to ca. 1,640 TWhe, value in agreement with

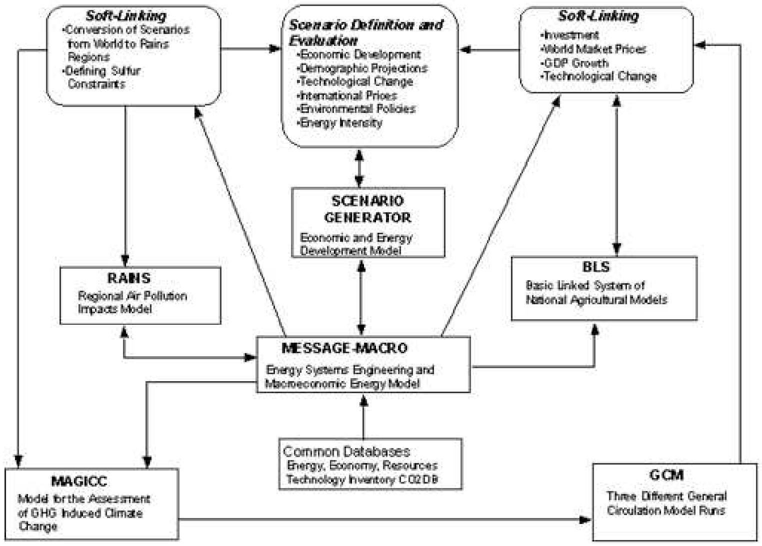

Figure B.3: MESSAGE code: integrated modeling framework [3]

B2-MESSAGE IPCC										
	1990	2000	2010	2040	2050	2070	2080	2090	2100	2200
EJ										
World	7.3	8.4	10.9	32.1	47.6	82.9	99.3	119.8	142	142
OECD90	5.9	7	8.5	15.1	16.5	20.6	20.7	25	29.3	29.3
REF	1	0.8	0.7	1.3	2.4	4.4	5.6	7.2	9.3	9.3
ASIA	0.3	0.5	1.5	12.9	20.9	38	47	55.5	64.7	64.7
ALM	0.1	0.1	0.2	2.8	7.8	19.9	26	32.1	38.7	38.7

Table B.1: B2-MESSAGE: Nuclear energy demand by regions [3, 41]

NEA data [89]) to 29.3 EJ (ca. 8,140 TWhe). REF countries increase of ca. 9 times their nuclear capacity, but the bigger increase is expected by ASIA and ALM regions.

In fact, ASIA passes from 0.3 EJ to 64.7 EJ (ca. 18,000 TWhe), overtaking OECD countries around 2045 and reaching in 2100 a value equal to 2.5 times the OECD90 (2100) level. The ALM follows a very high development as well, passing from 0.1 EJ to 38.7 EJ (ca. 10,800 TWhe) in 2100. With this scenario, ALM prevail OECD90 before the end of the century (around 2075).

In the B2-MESSAGE scenario, that is one of the IPCC scenarios oriented to environment, the World energy mix change radically from 1990 to 2100. In fact, as shown in Figure B.4, nuclear and renewable sources considerably increase their shares in the primary energy mix, whereas oil decreases drastically its influence.

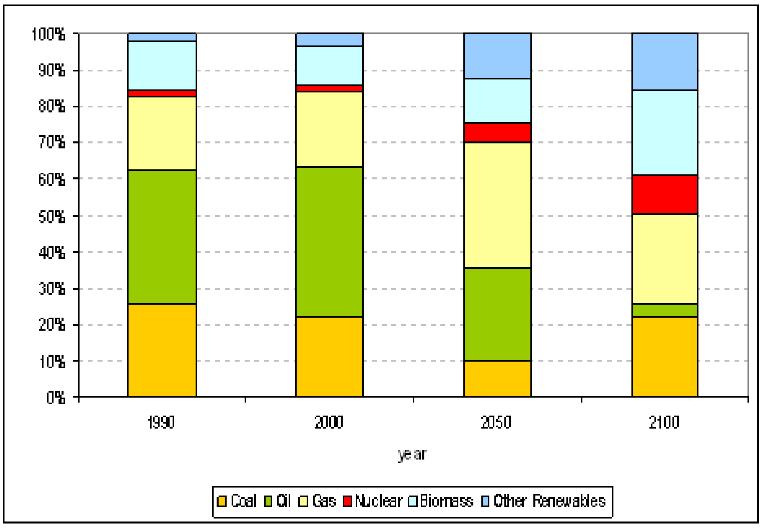

Figure B.4: B2-MESSAGE: primary energy mix variation [3]

The only "drawback" of this energy envelops is that they are very ambitious concerning the total nuclear energy demand (7.3 EJ in 1990 that becomes 20 times more in 2100) resulting in strong challenges in sustaining the cycle (e.g. availability of natural uranium resources, infrastructures needs).

The reason for this high increase could be found in the MESSAGE model adopted. In fact, the population growth rate (even though weighted by per capita economic growth and technological development level) has been considered a central driving force. Therefore, countries like China, India or Africa have a big influence.

For this reason, should be useful to analyze other possible alternatives. As for instance the B2 - MiniCAM case (as just indicated within the WPFC [102]).

B2-MiniCAM

The B2-MiniCAM scenario is an alternative scenario developed in order to investigate uncertainties in driving forces beyond those of the harmonized scenarios.

The model adopted (MiniCAM) is a simplify Integrated Assessment Model able to evaluate global GHG emission and land use. MiniCAM is composed by four modules (see Figure B.5) where the demand for energy is determined by regional population levels, levels of economic activity and the price of energy services.

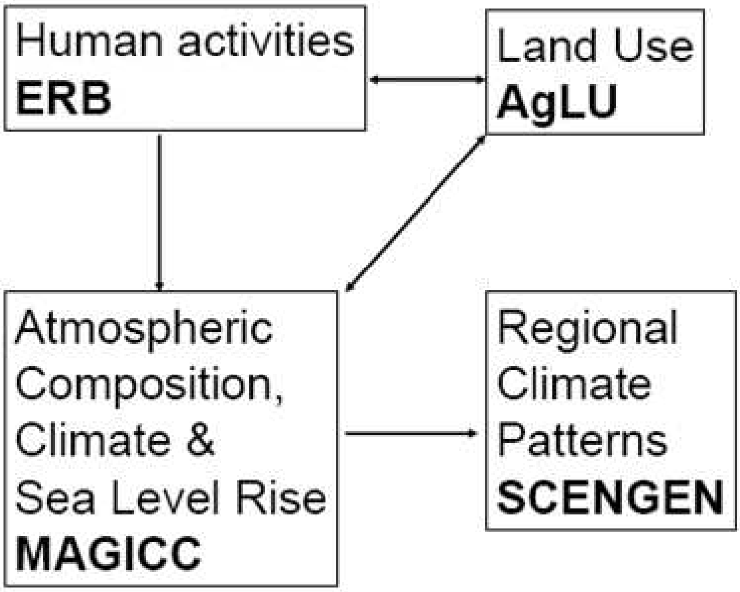

Figure B.5: MiniCAM Integrated modeling framework [3]

This scenario is less ambitious concerning nuclear energy development in the present century: the energy demand in 2100 is about 59.7 EJ corresponding to half of the B2-MESSAGE value (142 EJ). The relative regions subdivision is indicated in Table B.2.

B2-MiniCAM IPCC										
	1990	2000	2010	2040	2050	2070	2080	2090	2100	2200
EJ										
World	7.8	8.0	9.1	20.5	27.2	34.7	39.5	49.6	59.7	59.7
OECD90	6.6	4.9	3.8	3.7	4.0	4.1	4.3	5.4	6.4	6.4
REF	0.9	1.2	1.6	2.0	2.1	2.0	2.2	2.6	3.0	3.0
ASIA	0.3	1.2	2.5	9.6	13.3	16.8	18.9	23.4	27.9	27.9
ALM	0.1	0.6	1.2	5.1	7.7	11.8	14.0	18.2	22.4	22.4

Table B.2: B2-MiniCAM: Nuclear energy demand by regions [3]

Analyzing Table B.2, it is clear as the development followed by each region, and in particular by ASIA and ALM, is completely different with respect to the B2-MESSAGE case. In fact, OECD90 maintains roughly the same level of nuclear energy capacity as in 1990. Therefore, only the substitution of old reactors has been considered (neglecting any new energy programs).

ASIA, indeed, increases the nuclear energy demand of 100 times passing from 0.3 EJ to 27.9 EJ and covering more than half of the totals in 2100. Following this development, ASIA overtakes OECD countries before 2020 (i.e. constructing in less than 10 years more or less 80-100 nuclear reactors of about 1GWe - 1.2 GWe each).

These data seem no realistic at all.

ALM shows an unrealistic behavior too. In fact, according to B2-MiniCAM data, ALM prevails OECD90 around 2035 reaching at the end of the century a capacity installed equal to 3.4 times the actual OECD90 capacity and not so far from the ASIA value [3, 41]. T

Even though B2-MiniCAM scenario seems reasonable for World analysis (in terms of homogeneous trends), it has shown unrealistic behaviors for the applicability to a heterogeneous study.

In order to have a complete vision of the scenario, in Figure B.6 is presented also the primary energy mix vs. time.

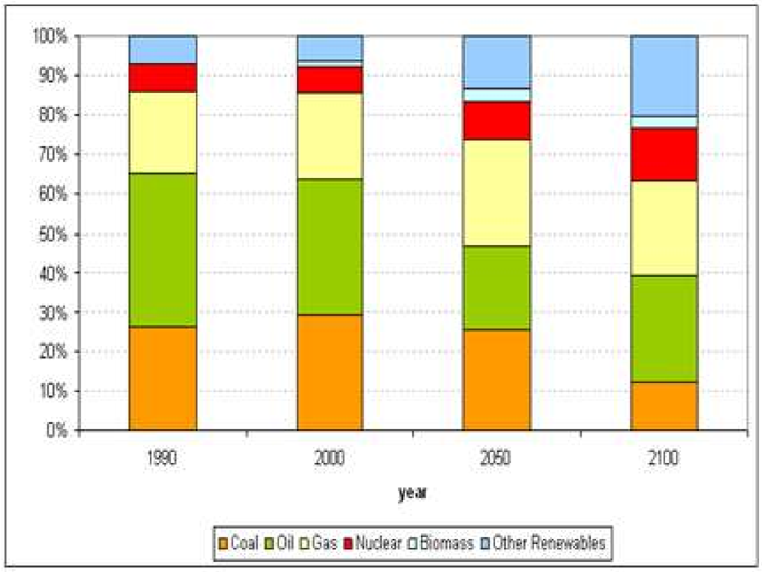

Figure B.6: B2-MiniCAM: primary energy mix variation [3]

Other scenarios: A2-MESSAGE and A2-MiniCAM

The same difference between MESSAGE and MiniCAM models could be found also analyzing the A2 "family".

The A2 "family" is oriented to economic growth with greater regional focus, and also in this case A2-MESSAGE is a harmonized scenario (HS) and A2-MiniCAM is an alternative scenario (OS).

The A2-MESSAGE gives a World nuclear energy demand equal to 2 times the value from A2-MiniCAM in agreement with the behavior inside the B2 "family".

Assuming A2-MESSAGE envelops, OECD90 remains the bigger contributor to World nuclear energy production even at the end of the present century. In this scenario, OECD90 has the same increase of ALM and ASIA, reaching an energy production equal to 10 times the actual level.

This aspect seems a little bit strange due to the fact that OECD90 is an industrialized are (with more or less stabilized electricity needs) and a higher growth rate is not expected for the coming years.

Otherwise, assuming A2-MiniCAM, it is possible to repeat the same considerations as B2-MiniCAM concerning the development of ASIA and ALM with respect to OECD90 [3, 41].

B.2.1 Comparison between IPCC scenarios

During the description of B2-MESSAGE and B2-MiniCAM some "critical" points about regional trends have been underlined. In particular for the B2-MiniCAM the behavior of ASIA and ALM with respect to OECD90 seems to be not realistic at all.

Analyzing the data, the scenarios based on MESSAGE model seem too optimistic: the value at the end of century equal to ca. 20 times the value in 1990 entailing problems related to U-Pu availability.

According to IPCC [3], the best scenario does not exist but there is a set of assumptions and scenarios built, on the basis of the state of the art before 2000, to give several possibilities for the energy development of the present century. Of course, these scenarios and assumptions needed some revisions (a process in this direction is ongoing internally to IPCC [202]) supported by studies oriented to each energy source.

B.3 IIASA Scenarios

In 1998, the International Institute for Applied Systems Analysis (IIASA) of Vienna, supplied to the scientific community a series of energy-electricity projections (up to 2100) for a World regional subdivision [23].

In this publication, a World subdivision in 11 zones has been considered. The 11 zones considered are in agreement with Figure 2.6, hence, it has been very easy to combine them in order to make a comparison with the 4 IPCC "macro-regions" scenarios.

The scenarios presented by [23] could be classified in 3 groups: "A", "B" and "C".

Case "A" presents a future of impressive technological improvements and consequent high economic growth [23].

Case "B" describes a future with less ambitious, though perhaps more realistic, technological improvements, and consequently more intermediate economic growth ("business as usual" case) [23].

Case "C" presents an ecologically driven future. It includes both substantial technological progress and unprecedented international cooperation centered explicitly on environmental protection and international equity [23].

The total set of IIASA scenarios consists of the following six scenarios:

- B Middle Course ("business as usual"),
- A1 High growth, based on oil and gas,
- A2 High growth, return to coal,
- A3 High growth, fossil fuel phaseout,
- C1 Ecologically driven, new renewable sources with nuclear phaseout,
- C2 Ecologically driven, renewable sources and new nuclear.

The model adopted by IIASA includes the MESSAGE model just described for the IPCC evaluations [3, 23]. Therefore, some similarities are expected.

Only two scenarios have been analyzed: the B scenario, because it is a middle course scenario, and C2 scenario, because it is oriented to environment (like IPCC B2 "family") and takes into account the use of nuclear energy. In additions, the C2 scenario has been also adopted by [87], for the analysis of fast reactors development under homogeneous assumptions.

Even in this case (as in MESSAGE model described above) the driving forces are the population growth rate and the economical development (opportunely weighted on the beginning technological level). Concerning the population growth rate, all IIASA scenarios have the same assumptions, hence, they differ principally in the energy mix foreseen and in the GDP projections [23].

B scenario

The scenario B is characterized by modest estimates of economic growth and technological development, and the demise of trade barriers and expansion of new arrangements facilitating international exchange.

Compared with the Case A scenarios (and the Case C scenarios), it is more "pragmatic".

Case B manages to fulfill the development aspirations of the South, but less uniformly and at a slower pace than in the other cases. For regions such as Africa, progress is painfully slow [23].

The world nuclear energy demand reaches in 2100 a level close to the B2-MESSAGE scenario (128.2 EJ vs.142 EJ of IPCC [3]). The regional subdivision is described in Table B.3.

B-IIASA									
	1990	2000	2010	2020	2030	2050	2070	2100	2200
EJ									
World	7.3	8.36	10.99	14.4	21.87	42.73	68.66	128.2	128.2
OECD90	5.91	6.83	8.65	10.62	14.87	23.12	26.16	28.85	28.85
REF	0.98	0.8	1.02	1.23	1.69	3.58	7.27	10.02	10.02
ASIA	0.33	0.49	0.69	1.76	4.32	12.19	22.46	51.41	51.41
ALM	0.08	0.24	0.63	0.79	0.99	3.83	12.76	37.9	37.9

Table B.3: B-IIASA: Nuclear energy demand by regions [23]

According to this scenario, ASIA and ALM follow a high nuclear energy increase and they exceed OECD90 respectively around 2080 and 2095. Also OECD90 increase its nuclear energy capacity of about 3 times with respect to the actual value.

The B scenario proposed by IIASA and B2-MESSAGE scenario proposed by IPCC are comparable.

However, if a choice has to be done, the B-IIASA scenario seems preferable because the total value is slightly lower than B2-MESSAGE case (as indicated, the total value so high for this scenario should be considered a critical point) and in addition ASIA and ALM overtake OECD90 more close to the end of the century (for ASIA is 2080 instead of 2045, and for ALM is 2095 instead of 2075).

C2 scenario

In order to be comparable with B2 "family" (oriented to regional solution and environment), the C2 scenario proposed by IIASA has been analyzed too [23].

The C group, indeed, is the most challenging group considered by IIASA. In fact, it is optimistic about technology and geopolitics and it assumes unprecedented progressive international cooperation focused explicitly on environmental protection and international equity. It includes substantial resource transfers from industrialized to developing countries, spurring growth in the South. In this case nuclear energy is at a crossroads.

A new generation of nuclear reactors is considered in the energy mix [23].

For this scenario, the projection of the total nuclear energy demand is less optimistic than B scenario. It is, indeed, comparable with the B2-MiniCAM scenario (at 2100, the C2 value is 60.6 EJ instead of 59.7 EJ for the B2-MiniCAM). The relative regions subdivision is represented in Table B.4.

In this case, the total value (compared to B scenario) and also the time for ASIA and ALM to exceed OECD90 seem "realistic" value (ASIA in 2060 and ALM in 2085). Nevertheless, this scenario has critical points too; the REF behavior does not respect the actual nuclear energy plans, and the OECD90 behavior seems questionable.

C2-IIASA									
	1990	2000	2010	2020	2030	2050	2070	2100	2200
EJ									
World	7.3	8.49	10.44	13.47	18.4	27.38	32.68	60.59	60.59
OECD90	5.91	6.83	8.33	9.94	12.45	15.57	12.66	8.18	8.18
REF	0.98	1	1.21	1.45	1.57	1.3	1.43	1.37	1.37
ASIA	0.33	0.49	0.7	1.76	3.93	9.68	15.91	32.78	32.78
ALM	0.08	0.15	0.21	0.3	0.45	0.82	2.68	18.25	18.25

Table B.4: C2-IIASA: Nuclear energy demand by regions [23]

For OECD90, indeed, the decrease hypothesized in the period 2050-2100 seems not plausible if the reactor lifetime (ca. 60 years) is taken into account (assuming that these reactors will not be shutdown before reaching their reactor lifetime).

Difference among the projections provided by a single organization can be found as just indicated for IPCC. Analyzing the IIASA data for the business as usual scenario (case B) and a scenario more oriented to the environment protection (case C2) a factor of ca. two in 2100 can be found both for the total electricity (ca. 75,400 TWhe and 44,600 TWhe respectively for scenario B and C2) request and the nuclear electricity production (see Figure B.7).

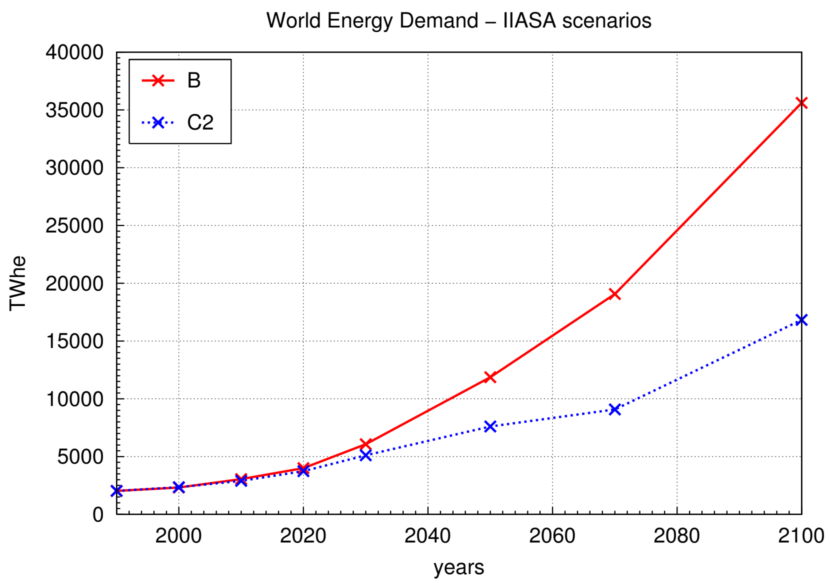

Figure B.7: IIASA scenarios: world electricity demand proposed for the "business as usual" (case B) and a scenario more oriented to the environment protection (case C2) [23]

As for the total energy demand, also the energy mix assumed in each scenario can be quite different. An example is in Table B.5 where the electricity mix in 2100 for scenario B and C2 (IIASA) is shown. Under "Other" are included all the renewable sources (solar, wind, biomass).

Electricity MIX in 2100		
Source	B	C2
%		
Coal	7.88	0.00
Oil	0.16	0.00
Natural Gas	5.24	5.60
Nuclear	47.24	37.72
Hydro	9.85	13.82
Others	29.64	37.88

Table B.5: IIASA scenarios: world electricity mix in 2100 for the business as usual case (B) and a scenario more oriented to the environment protection (C2) [23]

B.4 Summary on long-term energy projections

From the analysis presented before, it is clear that assumptions and models affect future energy envelopes, and hence, scenario studies.

According to [3], there is not a perfect scenario and the choice depends on the plausibility or not of the data referring to the aim of the work (for instance on the basis of medium-term projections or energy policy of a group of States).

For this reason, it could be useful to analyze more than one scenario, in order to have a panorama of possibilities. In addition, on the basis of critical points indicated for each scenario, the detailed nuclear fuel cycle analysis and the results obtained should be an input to scenario making groups to adjust projection and trends in each region (e.g. Figure B.8).

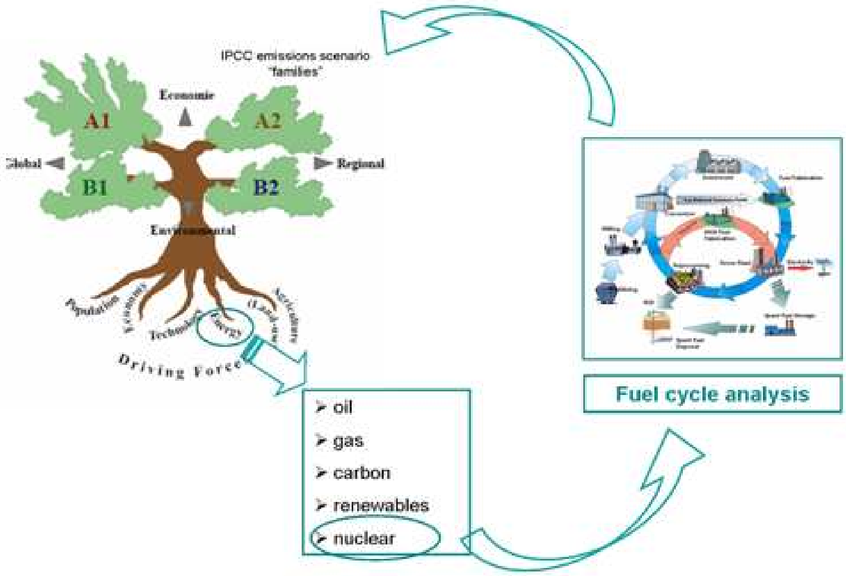

Figure B.8: Possible link between fuel cycle analysis and scenario development

Before concluding, medium-term scenarios have been analyzed too. Both IAEA and IEA publications have been considered. In the following part, only IEA data (up to 2050) will be presented.

B.5 IEA Energy Projections up to 2050

In order to consider also short-term scenario, the IEA publication have been considered. In particular the "Energy to 2050, Scenarios for a Sustainable Future", 2003 [24] and the "Energy Technology perspectives", 2008 [25] have been chosen as reference.

The regional subdivision adopted by IEA is substantially in agreement with the IPCC subdivision described before.

In general HIGH and LOW scenarios are presented in order to fix the boundaries of the analysis. These data strongly depend on the economical development of the area considered (and some times by the technological choices).

From 2003 publication [24], only the scenario "SD" (Sustainable Development) has been considered. The results are shown in Figure B.9.

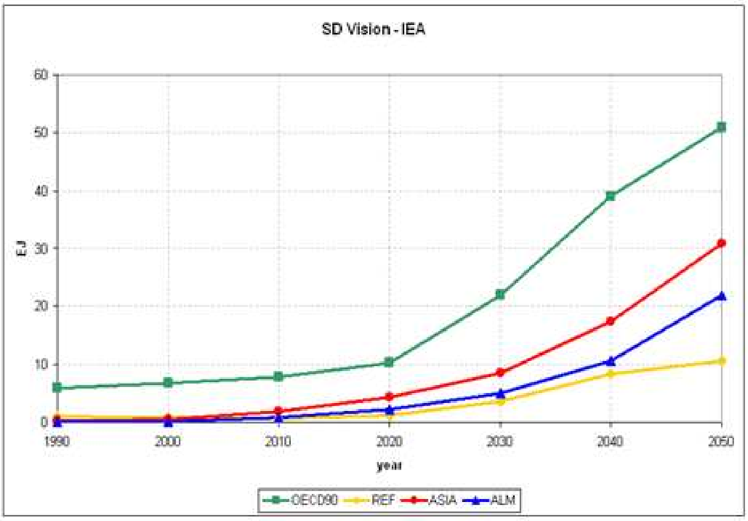

Figure B.9: SD IEA: Nuclear energy demand by regions [24]

In this figure, ASIA and ALM increase but not enough to reach OECD90 before 2050. In any case, the projection for OECD90 seems too much optimistic, considering the increase of nuclear energy of 9 times in 40 years.

This projections have been modified by most up-to-date IEA publications, as indicated in Figure B.10 (data from [25]).

B.6 Conclusions

In order to analyze nuclear transition scenarios for an heterogeneous World a detailed analysis of different energy envelopes proposed in literature has been performed.

This revision has been carried out in order to explain "strange" behaviors of some regions in the B2 - MiniCAM scenario considered as reference by NEA/OECD WPFC activities (World Homogeneous study).

Different energy envelops should be considered on the basis of models and assumptions adopted. Each scenario proposed seems to have some criticalities when applied to a specific sector (e.g. nuclear production). For instance, the world nuclear energy demand is too high (B2 - MESSAGE and B - IIASA), or not

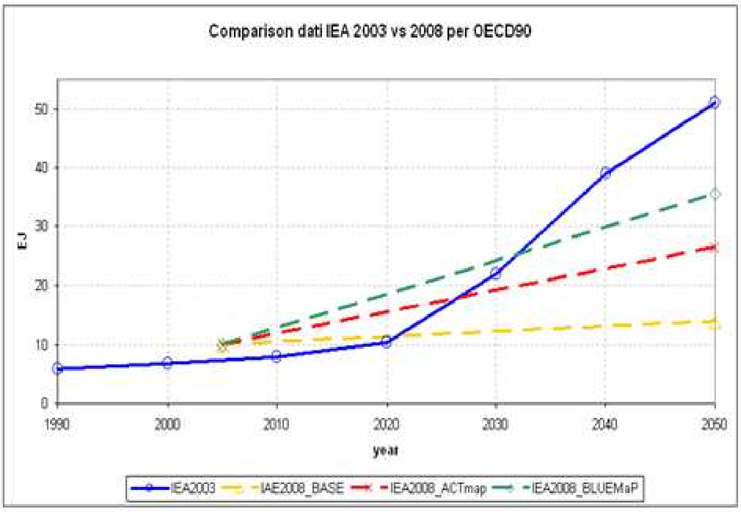

Figure B.10: Comparison between SD - IEA2003 [24] and IEA2008 [25] scenarios

realistic growth rates for ALM and ASIA (B2 - MiniCAM) can be underlined.

As possible alternatives to the B2-MiniCAM, other scenarios can be considered: 1) the B - IIASA scenario because it has a high demand compared to B2-MiniCAM but with more realistic regional trends for ASIA and ALM, and 2) the C2-IIASA because it is comparable with B2-MiniCAM even though the time for ASIA and ALM to exceed OECD90 seem more realistic.

As indicated in Chapter 2, the outcome of this study has been the selection of the boundary conditions to be applied to the NEA/OECD WPFC study concerning world scenarios.

Appendix C

Appendix: Analysis of some common parameters: breeding gain and breeding ratio

During the Ph.D. activity, the analysis of the definition of some common parameters characterizing fast reactors and their cycles has been performed.

In particular, the analysis of the available definitions of Breeding Gain (BG), Breeding Ratio (BR) and "Pu239 equivalent" has been performed.

In the following part is reported the analysis performed as summarized in the paper presented at the 11th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation (11th IEMPT conference) held by NEA/OECD in San Francisco (USA), 2010[1] [203].

This study has provide a contribution to the KIT activity on P&T applied to regional studies as indicated in Par. 2.4 [64, 65, 62].

C.1 Impact of MAs Content on Breeding Gain Definition for Innovative Fast Reactor Fuel

In the frame of Partitioning and Transmutation (P&T) strategies, where innovative fuels and systems are considered to burn transuranics (TRUs) mass discharged from conventional thermal reactors, the analysis of the applicability [203] the existing definitions to characterize the breeding or burning capability of a system could help in order to draw appropriate conclusions and assessments from fuel cycle analyses.

The aim of this work is to present an overview of the existing definitions of the Breeding Gain (BG), the Breeding Ratio (BR) and the Conversion Ratio (CR) and to analyze the impact of their application to critical fast "burner" reactors loaded with a high content of Minor Actinides (MAs).

Low CR critical burner fast reactor concepts loaded with different MA/Pu ratios are investigated at KIT [104] as possible alternatives to Accelerator Driven Systems in a so-called 'double strata' strategy developed in order to manage Spent Fuel (SF) inventories of European countries. The interest in this kind of system is to show the high flexibility of fast reactor technology to revert the core characteristics from "breeder" through "iso-generator" to "burner", as it has been shown within the CAPRA international program in the early nineties [115].

[1]The co-authors of the paper are: B. Vezzoni, F. Gabrielli, A. Schwenk-Ferrero, V. Romanello, W. Maschek, M. Salvatores

For the fuel cycle logistic, one of the main parameters characterizing a fast reactor is the Doubling Time, DT (linear in case of a single reactor or composite in case of reactor fleet) that is the time needed to produce a sufficient amount of fissile fuel (by conversion of fertile material loaded into the reactor e.g. in the blanket) to build up a new identical core. The DT is defined on the basis of BG, a parameter that gives a measure of the fuel produced in excess by the system during a so called fuel cycle. For this reason, the BG definitions have been analyzed in depth.

For defining the BG, two approaches could be distinguished: the "American approach", which takes into account integral quantities establishing a mass balance between the reactor initial and final states (i.e. Beginning of Life, BOL, and End of Life, EOL), and the "European approach", more oriented toward "point in time" quantities and the reactivity balance between the same states. Whereas in the early definitions, fuel components have been simply distinguished as "fissile" and "fertile" (e.g. for mixed oxide fuel Pu-239, Pu-241 are fissile and U-238, Pu-240 are fertile nuclides), refined definitions take additionally into account the contribution of each isotope to the reactivity of the system (by means of reactivity equivalence coefficients).

Historically the definition of BG, BR/CR and DT have been formulated for fuel containing conventional mixed oxide (MOX) where only uranium and plutonium are included (e.g. typical for the LWR spent fuel or the fuel self-recycled in Fast Reactors, FR). In advanced fuel cycles, the introduction of variable fractions of MAs (in homogeneous and heterogeneous modes) still has to be carefully analyzed. Therefore, the applicability of the existing definitions of BG, CR, DT has to be analyzed too.

An accurate neutronic assessment of fast reactors loaded with a high content of MAs (ca. 50%), shows that the computation of BG, BR, DT could be performed using basically the same formalisms as adopted in the past for MOX fuel, but one should be careful when analyzing results in order to avoid misinterpretations. All these aspects will be discussed in detail on the basis of the results obtained by their application to critical burner concepts developed at KIT. Two burners with the same CR (0.8) and, respectively, loaded with 10% of MAs (MA/Pu=0.1) and 50% of MAs (MA/Pu=1) have been chosen for this comparison.

C.2 Overview of Breeding Gain definitions

For the breeding gain definitions, two main approaches could be distinguished: an integral approach based on the mass balance between initial (e.g. BOL) and final states (e.g. EOL) and a punctual or "point in time" approach based on the reactivity balance between the same states (where reactivity equivalence coefficients are applied to take into account the contribution of each isotope to the system's reactivity). In general, two conceptual definitions for the BG could be found in literature. The first one, "standard" (Eq. C.1), considers the surplus of the fissile isotopes produced by the systems over the total destruction rate (the variant is called "British" when the fission rate is considered for the normalization [204]). The second one, called "French", considers the surplus of the fissile isotopes produced but just scaled on the Pu-239 equivalent (Eq. C.2), [196].

$$BG_{stand.} = \frac{surplus - production - rate - of - fissile}{destruction - rate - of - fissile} \tag{C.1}$$

$$BG_{French} = \frac{(production - minus - absorption - rate) - of - Pu^{239} equivalent}{fission - rate} \tag{C.2}$$

They differ only in the way how to calculate the surplus of fissile material (normalized or not on Pu-239 equivalent). The French definition seems more convenient when different fuels (with different Pu vectors) generated by thermal reactors are compared.

The approach based on mass balance is unambiguous once the fuel has been defined (fissile vs. fertile isotopes); meanwhile the approach based on reactivity balance is affected by the fuel cycle logistic considered. As indicated in [205], the reprocessing scheme and the equilibrium vs. non-equilibrium operation regimes could influence the breeding capability of a system and they have to be taken into account for the BG and DT evaluation. Therefore, only the quantities obtained in the equilibrium operation regime (which is the property of the reactor alone), give the proper basis to compare and characterize the breeding capabilities, otherwise the value obtained is strongly influenced by the composition of feed fuel from the stockpile, and it becomes a figure of merit for the fresh fuel composition [205, 206]. At the same time, the asymptotic BG (or equilibrium BG) has not a clear definition too [207].

In fact, it could be defined at least by 4 different relations (considering the reactor-park definition, the detailed single reactor calculation, the integrated continuous model for a single reactor, the isotopic breeding worth) that lead to different results. Of course these differences are not negligible in the transition scenarios from a thermal reactor to a FR fleet (where the doubling time is a crucial parameter for the strategy to adopt).

All these considerations have been underlined by the analysis of the critical burner concepts developed at KIT [104] and modeled by means of the ERANOS neutronic code [182]. In particular, mass vs. reactivity approaches are considered as well as the decay contribution of the isotopes (e.g. Np-239, $t_{1/2}$=2.3565 days, Am-242g, $t_{1/2}$=16.02 hours and Cm-242, $t_{1/2}$=162.8 days.

Another figure of merit associated to the BG is the breeding ratio (BR), see Eq. C.3. It gives a measure of the system conversion capability. Historically it is called BR when it exceeds 1 (net fissile production) and CR when it is less than 1 (net fissile concentration reduced). This equation is respected by the mass balance approach but it could result in misinterpretation if it is considered to be valid always (see below the comparison between burners).

$$BR_{stand.} = 1 + BG_{stand.} \tag{C.3}$$

The above concepts could be made explicit by adopting both the reactivity equivalence coefficients ("European approach") and the masses ("American approach"), as it will be shown in the following sections.

C.2.1 The Integral approach oriented to mass balance

The integral approach makes a mass balance between two states: BOL and EOL, but also over a single irradiation cycle (Beginning of Cycle, BOC, and End of Cycle, EOC).

Starting from the CR definition (the ratio of fissile material produced to fissile material destroyed), that is applicable to thermal and fast reactors; it is possible to define the BG as follows. According to [207, 208], the definition of the CR (or BR), applied to an equilibrium fuel cycle, could be expressed as in Eq. C.4, where FP indicates the fissile material produced per cycle and FD the fissile material destroyed per cycle, respectively, where FP has the following contributions: FD, FEOC (fissile inventory in the core and blankets at the EOC) and FBOC (fissile inventory in the core and blankets at BOC).

$$BR = \frac{FP}{FD} = \frac{FD + FEOC - FBOC}{FD} \tag{C.4}$$

$$\tag{C.5}$$

From this formulation, it is clear that the BR is averaged over a fuel cycle, where all the fissile material produced (expressed in kg or in atoms/barn*cm [207]), including the fissile material destroyed in the cycle itself, is taken into account. The BG has been defined (Eq. C.6) to evaluate the net fissile amount produced during the cycle (i.e. the fissile material available for loading another reactor). It is, indeed, the ratio of the fissile material gained per cycle (FG) over the fissile material destroyed in the cycle (FD). By this approach, the well known relation (Eq. C.3) between BG and BR is preserved.

$$BG = \frac{FEOC - FBOC}{FD} = \frac{FG}{FD} \tag{C.6}$$

Based on this approach, the fast critical burner CR has been evaluated. In particular, U-238, Pu-238 and Pu-240 are the fertile isotopes considered and U-235, Pu-239 and Pu-241 are the fissile isotopes considered. In detail, the formulation adopted (Eq. C.7), considering the capture (TCA) and the absorption (TAB) rates calculated by the ERANOS code over the cycle, takes into account also the contribution of the decay (rates calculated on the basis of the Bateman Equation).

$$CR \cong \frac{TCA_{U238} + TCA_{Pu238} + TCA_{Pu240}}{TAB_{U235} + TAB_{Pu239} + TAB_{Pu241}} \tag{C.7}$$

Applying Eq. C.7 to the two burner concepts presented in the section following the next one, a CR 0.8 has been obtained for MA/Pu=0.1 and MA/Pu=1 cases. This value (<1) confirms that the system is a burner reactor. The expected BG in both cases is (-0.2) but the values provided by ERANOS are not always in agreement with that expected value (see following paragraphs). In order to take into account different isotopes reactivity properties, an additional step (introduced in reference [209]) is to consider a breeding ratio based on Pu-239 equivalent fuel, where, for instance, the contribution of each isotope to the system reactivity, instead of simply lumping all the fissile isotopes together, is taken into account to evaluate the critical mass of the system [209, 210]. This additional approach leads to the BG formulation based on the reactivity balance.

C.2.2 The punctual or point in time approach oriented to reactivity balance

In order to take into account the contribution of each isotope to the reactivity of the system, a set of weighting coefficients (called ω-values) has been defined. They represent the weights in term of Pu-239 equivalent reactivity of each isotope contained in the fuel (historically defined for MOX fuel) evaluated by comparing the system neutronic balance at criticality (fixing the total fuel mass and the material buckling, Bm2) with a systems loaded only with Pu-239. A reactivity scale, where Pu-239 and U-238 have, respectively, ω = 1 and ω = 0, is obtained on the basis of Eq. C.8 [196, 210].

$$\omega_i^n = \frac{(\nu\sigma_f - \sigma_a)_i^n - (\nu\sigma_f - \sigma_a)_8^n}{(\nu\sigma_f - \sigma_a)_9^n - (\nu\sigma_f - \sigma_a)_8^n} = \frac{\sigma_{i,n}^{+} - \sigma_{8,n}^{+}}{\sigma_{9,n}^{+} - \sigma_{8,n}^{+}} \tag{C.8}$$

The extension to fuel containing MAs could be performed and the contribution of each isotope (in particular Cm isotopes) could be taken into account. One of the key points is that these weights enable to determine a doubling time of critical mass material independent of the specific plutonium isotopic composition [207]. Different weights can be defined in order to take into account the different contributions to the reactivity originating from fuel containing isotopes due to transition and equilibrium FR loading, as it will finally be the case in a closed FR fuel cycle. Some examples may be found in [207, 211].

The reactivity coefficients depend on the reactor zone considered (due to their dependence on the neutron spectra); therefore, the total BG is obtained by the sum of the contributions from each reactor zone (e.g. different fuel zones and/or blankets). The "standard" definition (Eq. C.1) could be expressed as in Eq. C.9, [204].

$$BG_{stand.} = \frac{\sum_n \sum_i \omega_i^n (C_{i-1}^n - A_i^n)}{\sum_n \sum_i \omega_i^n A_i^n} \tag{C.9}$$

The "French" formulation (in Eq. C.10) differs from the "standard" one only for the "cut" adopted in the burn-up chain, affecting not so significantly the total BG value for conventional MOX fuel.

$$BG_{French} = \frac{\sum_n \sum_i (C_i^n \omega_{i+1}^n - A_i^n \omega_i^n)}{\sum_n \sum_i F_i^n} = \frac{\sum_n \sum_i (C_i^n (\omega_{i+1}^n - \omega_i^n) - F_i^n \omega_i^n)}{\sum_n \sum_i F_i^n} \tag{C.10}$$

In both these formulations, "Ain", "Fin", "Cin", "ωin" indicate, respectively, the absorption (capture plus fission), the fission, and the capture rates, and the reactivity equivalent coefficients of the nuclide "i" in region "n". These expressions do not consider the contribution of the decay (they have been formulated originally for conventional MOX fuel where this contribution is not so significant at least for reasonably short cooling, reprocessing and refabrication times). A revised formulation of the "French" BG (Eq. C.11) has been implemented in the ERANOS code [182]: a conversion rate (Cin including decay) and a disappearance rate (Din including decay) are adopted instead of the capture and the absorption rates. This modification has a significant impact when fuels containing short-lived nuclides (Np-239 Pu-239, Pu-241 Am-241, Am-242g Cm-242 Pu-242) are taken into account, as it will be shown for the fast burner reactors case.

$$BG_{ERANOS} = \frac{\sum_n \sum_i (C_i^n \omega_{c(i)}^n - D_i^n \omega_i^n)}{\sum_n \sum_i F_i^n} \tag{C.11}$$

C.2.3 The characterization of the fast critical burners

In order to show the high flexibility of fast reactor technology, several 1000 MWth fast critical MAs burner cores cooled by sodium have been investigated at KIT [104]. These cores (with different CR , 0.5 and 0.8, oxide and metal) have been modeled by means of the ERANOS code [182] on the basis of previous studies at Argonne National Laboratory [212]. The nuclear library adopted is the JEF 2.2 [213].

For the analysis of the BG, two cores with CR 0.8 and loaded with (U-TRU)O2 fuels have been selected (main parameters listed in Table C.1). Respectively, two different MAs to Pu ratios (MA/Pu) have been considered, respectively MA/Pu 0.1 and 1 (isotopic compositions listed in Table C.2). These two values are representative of discharged LWR fuels (MA/Pu=0.1) and of TRUs fuels from multi-recycling Pu fuels (MA/Pu=1) [212]. Further details about the models are given in [104, 62]. Table C.3, obtained by applying the burn-up chain presented in Figure C.1, shows that the core with MA/Pu=0.1 has a TRUs consumption rate of 13.23 kg/TWh (mainly due to the reduction of Pu-239 and of Pu-241) whereas the core with MA/Pu=1 has a rate of 26.65 kg/TWh (mainly due to the reduction of Am-241 and of Am-243). This indicates that from the point of view of the mass balance both reactors are good TRUs burners [104, 62].

C.2.4 Fast critical burners: comparison of BG and CR approaches

For the two fast burner reactors previously described, a BG, of the order of (-0.2) according to Eq. C.3, was expected. For the case of MA/Pu=0.1 the BG value provided by the application of the Eq. C.11 is in agreement with the expected value (BG -0.262). On the contrary, for the case of MA/Pu=1 the BG value obtained is equal to 0.691. To explain this positive value, several definitions have been applied and the reactivity loss for each isotope has been evaluated too.

Due to the fact that there are no significant differences between the "standard" and the "French" BG definitions, only the "French" formulation has been compared with the ERANOS one in order to evaluate

Parameters	Fast Reactor Burner	
Fuel type	(U-TRU)O2	
MA/Pu	0.1	1
Conversion ratio	0.73	0.75
Cycle length (EFPD)	353	353
Average TRUs content (%)	27.1	41.2
Power (GWth)	1.0	1.0
Average discharge burn-up (GWd/tHM)	149	117
Reactivity Loss (%$\Delta\rho$) over the 1st cycle	-4.5	-0.6

Table C.1: Main design parameters for the two burner cores considered

Isotope	MA/PU=0.1	MA/PU=1
	Content (wt.%)	
Np-237	4.8	7.3
Pu-238	2.3	2.0
Pu-239	47.9	18.2
Pu-240	22.5	13.4
Pu-241	10.6	5.8
Pu-242	6.5	10.6
Am-241	3.4	18.8
Am-242m	0.0	0.1
Am-243	1.5	15.9
Cm-243	0.0	0.1
Cm-244	0.5	7.0
Cm-245	0.0	0.8
Pu	89.8	50.0
MAs	10.2	50.0

Table C.2: Isotopic compositions (wt.%) for MA/Pu=0.1 and MA/Pu=1 fuels

Isotope	MA/PU=0.1	MA/PU=1
	kg/TWh	
U-235	-0.31	-0.27
U-238	-29.44	-22.59
Pu-238	1.13	2.32
Pu-239	-8.61	-0.72
Pu-240	1.94	-2.01
Pu-241	-7.07	-6.78
Pu-242	0.53	-0.18
Am-241	-0.27	-14.83
Am-242m	0.01	1.24
Am-243	0.15	-10.29
Np-237	-2.75	-5.35
Cm-242	0.83	5.24
Cm-243	0.02	-0.06
Cm-244	0.85	4.21
Cm-245	0.02	0.55
Total TRUs	-13.23	-26.65

Table C.3: U and TRUs consumption (kg/TWh) per cycle for the MA/Pu=0.1 and MA/Pu=1 cores

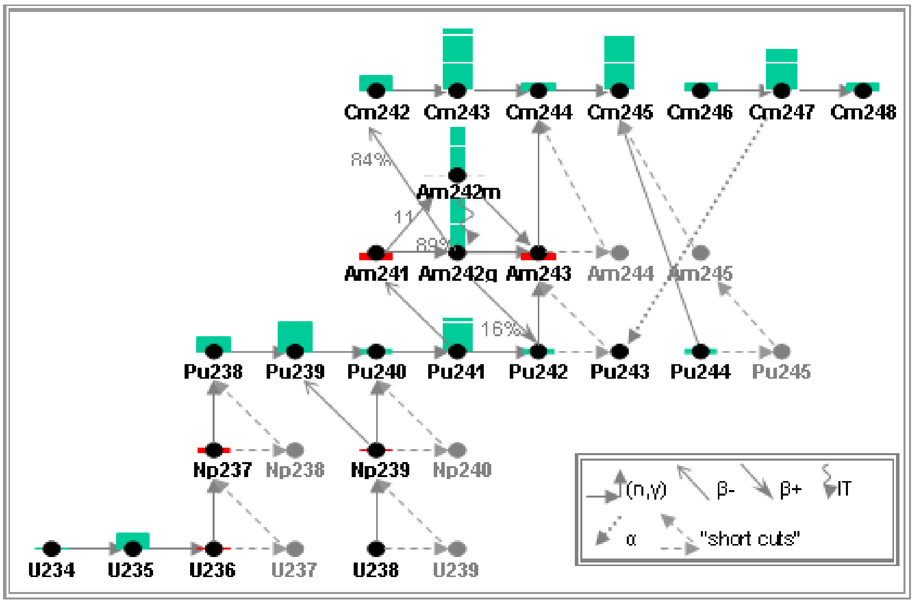

Figure C.1: Burn-up chain implemented in the ERANOS code the contribution of the decay to the BG. As indicated in Tables C.4 and C.5 (respectively showing results for the total value and the isotopic contributions in a chosen reactor zone), the impact of the decay on the BG evaluation becomes important when a high fraction of MAs is loaded into the fuel. Analyzing Tables C.4 and C.5, it is clear that the contribution of the short-lived nuclides to the total BG evaluation is significant in both cases (e.g. the decay of Np-239 into Pu-239; if it is neglected the Pu-239 positive contribution to the reactivity of the system is lost). In addition, the MAs fraction could affect significantly the BG value. In fact, for MA/Pu=0.1 the French and ERANOS formulations give negative BG that differ only for the contribution of the decay of the Np-239 (the ERANOS one is in reasonably good agreement with the value obtained by the mass balance approach). For MA/Pu=1 the two values are strongly positive (see Table C.4). A burner (as indicated in Table C.3) with a positive BG seems at first sight to provide contradictory indications.

MA/PU=0.1		MA/PU=1	
BG Treatment	BG	BG Treatment	BG
French	-0.98220	French	0.82255
ERANOS	-0.26183	ERANOS	0.69127

Table C.4: Total BG evaluated by the French (no decay) and the ERANOS formulations

	MA/PU=0.1			MA/PU=1		
Isotope	Omega	BG Treatment		Omega	BG Treatment	
		French	ERANOS		French	ERANOS
U-235	0.852	-0.00078	-0.00078	0.684	-0.00066	-0.00066
U-238	0.000	-0.03332	-0.03332	0.000	-0.00602	-0.00602
Np-237	-0.373	0.00652	0.00652	-0.132	0.01329	0.01329
Np-239	-0.455	0.00005	0.10456	-0.094	0.00002	0.06942
Pu-238	0.524	-0.00046	-0.00062	0.645	-0.00234	-0.00272
Pu-239	1.000	-0.09702	-0.09704	1.000	-0.06964	-0.06965
Pu-240	0.058	0.02205	0.02205	0.183	0.01043	0.01042
Pu-241	1.604	-0.04189	-0.05583	1.305	-0.03275	-0.04494
Pu-242	0.040	-0.00148	-0.00148	0.127	-0.00277	-0.00277
Am-241	-0.442	0.03331	0.03335	-0.218	0.21588	0.21607
Am-242g	2.405	-0.00001	-0.01208	1.963	-0.00005	-0.07197
Am-242m	2.361	-0.00031	-0.00032	1.905	-0.00290	-0.00297
Am-243	-0.424	0.00147	0.00147	-0.200	0.01954	0.01955
Cm-242	0.227	0.00019	-0.00059	0.409	0.00066	-0.01176
Cm-243	2.634	-0.00003	-0.00003	2.273	-0.00126	-0.00142
Cm-244	0.125	0.00026	0.00025	0.279	0.01122	0.00868
Cm-245	2.612	-0.00002	-0.00002	2.138	-0.00922	-0.01034
Total zone1		-0.11145	-0.03391		0.14343	0.11222

Table C.5: Contribution to the total BG of each isotope for one (zone 1) of the fuel zones composing the core

The positive BG for the MA/Pu=1 case is explained on the basis of the isotopic burning capability of the system. In fact, looking at Table 3, the MA/Pu=1 reactor burns high quantities of Am-241 and Am-243 (both isotopes give a "negative reactivity" contribution, $\omega<0$, to the system, see Table 5). These isotopes produce (by capture and decay) daughters that give a "positive reactivity" contribution, ω>0, to the system. One example is the Am-243 (ω<0) that is transmuted to Cm-244 (ω>0). Am-243 is loaded into the fuel with a fraction comparable to Pu-239 (16% vs. 18%) and it is burned for 10.29 kg/TWh (see Table 3). At BOL it gives a negative contribution to reactivity otherwise at EOC it provides by its transmutation products a positive contribution. The processes followed by the isotopes are indicated in Figure C.1, where the positive ω-values are represented in green and the negative ones in red (a column of more than 2 boxes stacked on top of each other meaning an ω-value > 2.0, see e.g. Cm-243). In the same Figure, the burn-up chain implemented in ERANOS is represented too. The global effect of the MAs transmutation in the MA/Pu=1 burner core is the very small reactivity loss at EOL with respect to the MA/Pu=0.1 case (see Table C.1).

All the results furthermore presented have been determined on the basis of a reference burn-up chain originally implemented in the ERANOS code for this study (see Figure C.1). A more refined burn-up chain (that neglects the contribution of the isotopes loaded into the fuel with very low fractions (e.g. Pu-244) and takes into account the decay contribution of all the MAs isotopes) have been applied. The results obtained differ more or less appreciably but they enable to reach the same conclusion: for MA/Pu=0.1 the BG is equal to -0.28967 instead of -0.26183 and for MA/Pu=1, it is equal to 0.52348 instead of 0.69127.

C.3 Conclusions

The detailed investigation of parameters like BG, BR and DT could help to avoid erroneous conclusions in the framework of innovative fuel cycle analyses. With this work a preliminary revision of the definitions has been performed in order to understand in detail the parameters necessary to characterize a fast burner reactor.

The definitions historically proposed may be applied also to innovative fuels (like 50% of MAs content) even though additional attention to treat them properly has to be used. In particular, the BG defined on the basis of the reactivity balance ("European approach') indicates that a burner with CR 0.8 and high content of MAs can have a positive BG, if it is calculated by means of the reactivity balance approach, since that system transmutes isotopes like Am-241 and Am-243 that have negative contributions to the reactivity (ω<0) into isotopes with positive omegas.

The two approaches presented here ("American" and "European") are apparently consistent only when limited quantities of MAs are present in the core (see MA/Pu=0.1 case), but when high contents of MAs are considered, the use of the usual correlation between BG and BR (i.e. BG=BR-1) is no more applicable. In fact, for high MAs content, the BR evaluated by the mass balance approach and BG by the reactivity balance approach can not be correlated by this simple relation as shown in the paper in particular for the MA/Pu=1 case.

Further activities for a more detailed analysis of these figures of merit will be the equilibrium cycle analysis and the evaluation of a revised reactivity scale of the ω-values for innovative fuels. For this last point, as indicated in [205], an explicit multi-group first order perturbation formulation will be applied and the decay contribution will be taken into account explicitly for the ω-values evaluation.

Appendix D

Appendix: The ELSY and ESFR Neutronic Models

In order to generate suitable libraries for the scenario calculations, the procedure described in Appendix A, based on the ERANOS-APOGENE-COSI code chain, has been applied [182, 183, 20].

In the present Appendix, the neutronic models of the systems considered in the study are described.

Four reactor systems have been considered for the scenario calculations: 1) a EPR-like as typical thermal reactor, and 2) three fast reactors, ELSY-like, ESFR-like and EFR-like.

As described in Chapters 4 and 5, only ESFR and ELSY systems have been modeled by means of ERANOS code [182].

For the EFR-like[1] and EPR-like[2] models, available libraries in COSI6 have been applied.

A description of the neutronic models is indicated in the followings.

D.1 The ELSY model

The European Lead-cooled System (ELSY) has been considered as one of the reference systems for the analysis of the transition from LWRs to FRs based fleet.

The ELSY system has been proposed by the FP6-EURATOM ELSY project [68] with the aim to develop a lead cooled fast reactor able to fulfill the Gen-IV goals (sustainability, economics, safety and reliability, proliferation resistance and physical protection) using proven technologies.

In fact, the Lead-cooled Fast Reactor (LFR) is one of the six selected options investigated within the Generation IV initiative [214].

Two subassembly (SA) options have been investigated in parallel within the project [68], leading to two core configurations.

In particular:

- a wrapper-less, open square SA, typical of a PWR, (see Figure D.1), leading to the core configuration shown in Figure D.2, has been developed mainly by ENEA (Italy) [26]. It represents the reference model for the ELSY design;

[1] Even if, in principle possible, it has been decided to do not create our own ERANOS model for EFR.

[2] The unavailability of the APOLLO2 code, used for thermal reactor models, has obliged to adopt the library present in COSI6 distribution.

- a hexagonal wrapped, closed SA, typical of a SFR (see Figure D.3), leading to the core configuration shown in Figure D.4, has been developed mainly by SCK-CEN (Belgium) [27]. It represents the backup model for the ELSY design.

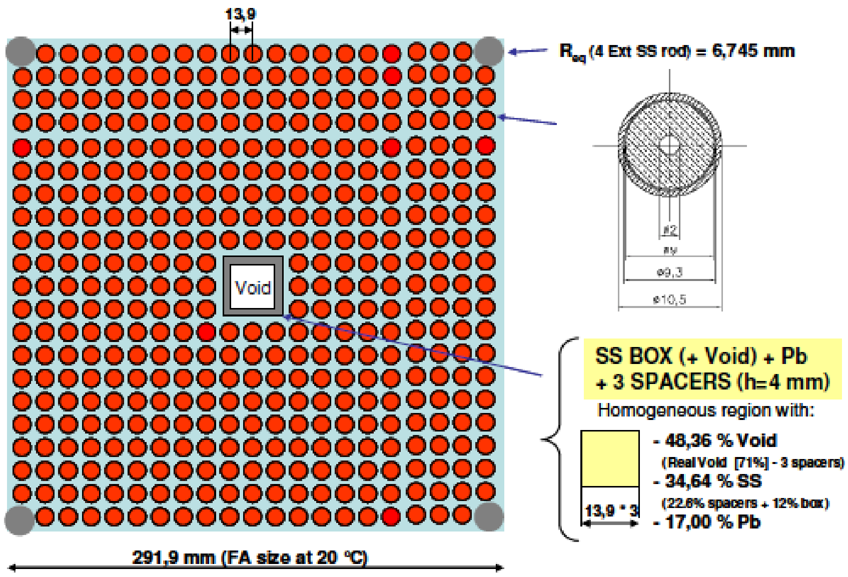

Figure D.1: ELSY reference configuration: SA geometry [26]

The two configurations (600 MWe) have been designed to demonstrate that it is possible to model a competitive and safe fast critical reactor, used also for waste transmutation, by adopting simple engineered technical features.

In particular, the ELSY fast reactor concept has been proposed to achieve the Gen-IV sustainability goals: reduction of uranium consumption, production of significantly less MAs compared to thermal spectrum facilities and, in a long-term, possibility to burn MAs.

The core composition has been designed with the sufficient reactivity reserve and small reactivity swing to assure at least two-three years of operation without fuel reloading or the core reconfiguration (important for reducing the operational cost by the reduction of the number of the intermediate reactor shutdowns for the core reshuffling) [27, 26].

For the present activity, it has been decided to adopt the backup configuration (hexagonal wrapped SA model) as described in the following Par. D.1.1.

In Table D.1, the core specifications adopted for defining fuel rod, fuel assembly and core layout for the ELSY cores are indicated [27, 26].

The reference (square) configuration has not been described in detail. Additional information can be found in [215, 26, 74, 75, 188].

D.1.1 The ERANOS model: ELSY HEX-z core

For the aim of the present Ph.D. activity, the ELSY ERANOS model has been set up. The model has been assessed on the basis of the backup core design, as indicated in the ELSY project [68, 216, 189, 27].

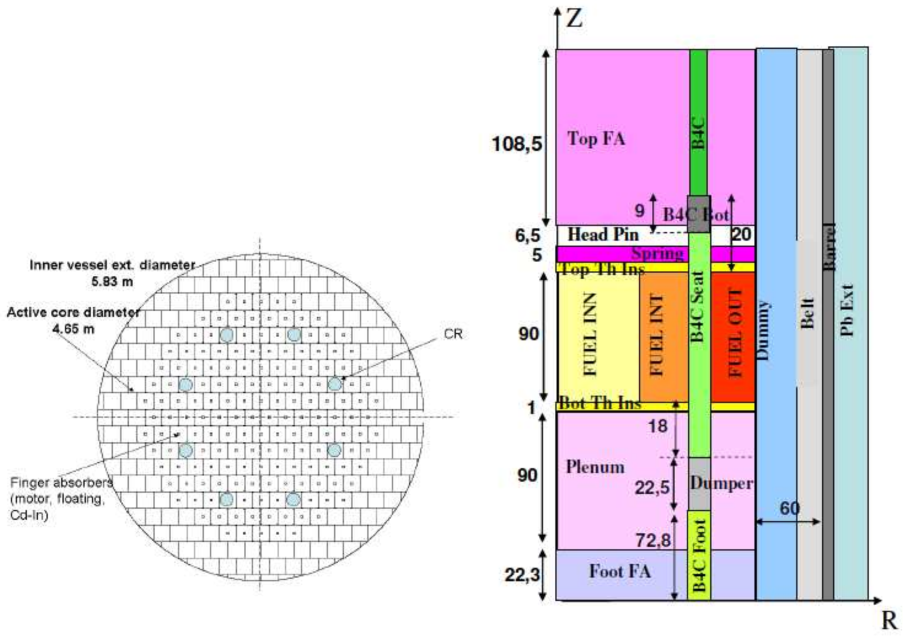

Figure D.2: ELSY reference configuration: core geometry [26]

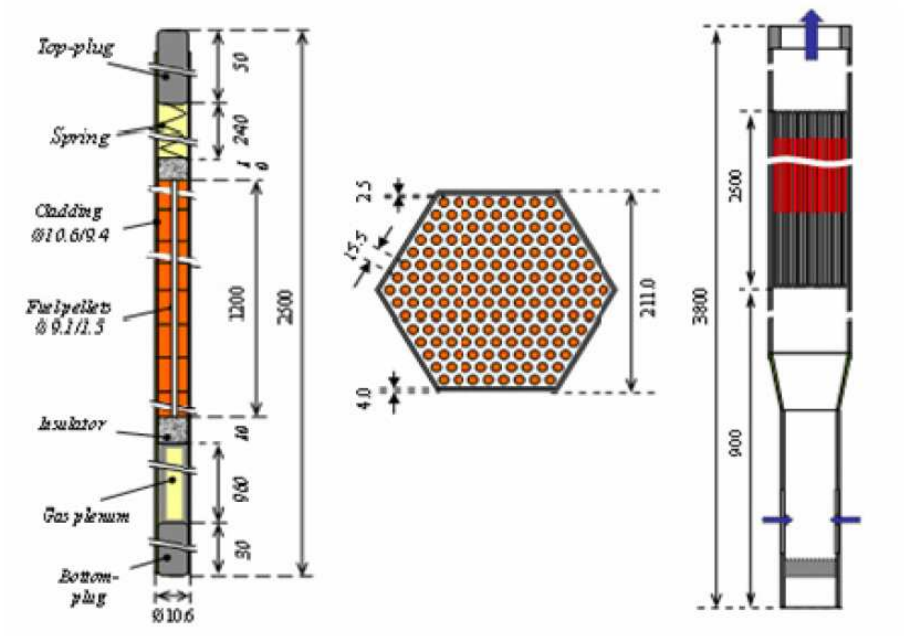

Figure D.3: ELSY backup configuration: SA geometry [27]

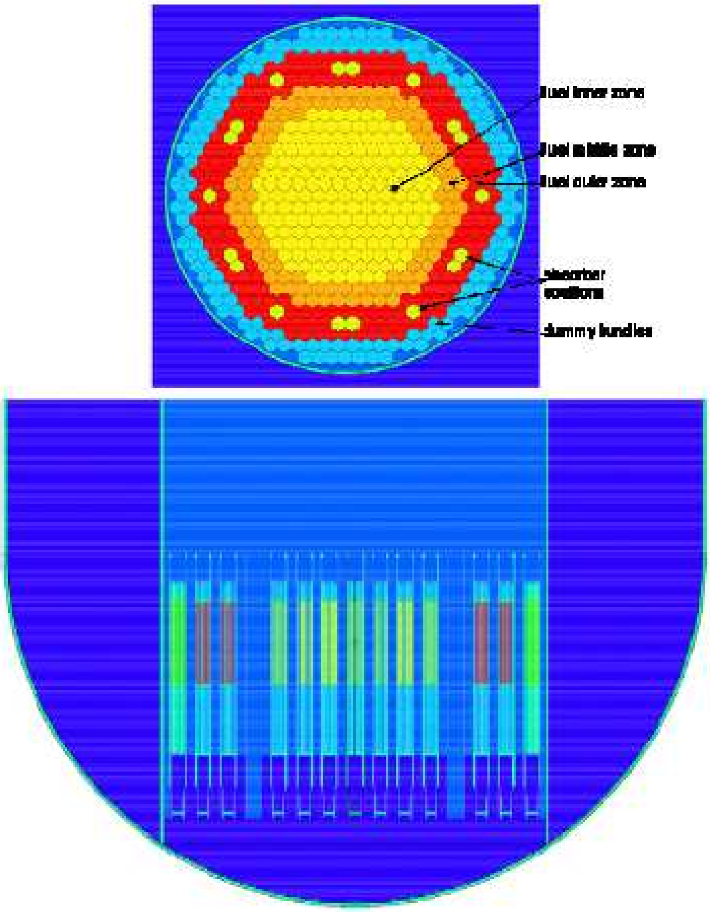

Figure D.4: ELSY backup configuration: core geometry [27]

Characteristics	Value
Thermal Power (MWth)	1500
Electric Power (MWe)	600
Coolant	Lead
Spectrum	Fast
Core BR	ca. 1
Minimum sub-cycle duration (years)	2
Fuel-residence time (years)	5
Fuel	$(Pu,U)O_{2-x}$MOX
Maximum Target Burn-up (GWd/tHM)	100
Cladding	FMS T91
Allowed clad damage (dpa)	100
Maximum clad temperature (°C)	550
Coolant inlet temperature (°C)	400
Coolant outlet temperature (°C)	480
Maximum allowed Pb bulk-velocity (m/s)	2.0

Table D.1: ELSY-600 core specifications [27, 26]

The oxide core considered consists on three radial zones with different Pu content (in order to flat the power profile) for a total of 433 fuel SA (163 SA inner zone, 102 intermediate zone and 168 outer zone). In the model 18 control and shutdown rods have been considered.

The core layout is indicated in Figure D.4.

In Figure D.5 the RZ equivalent model, as assessed with ERANOS code, is shown. Finally, Figure D.6 represents the core cross section.

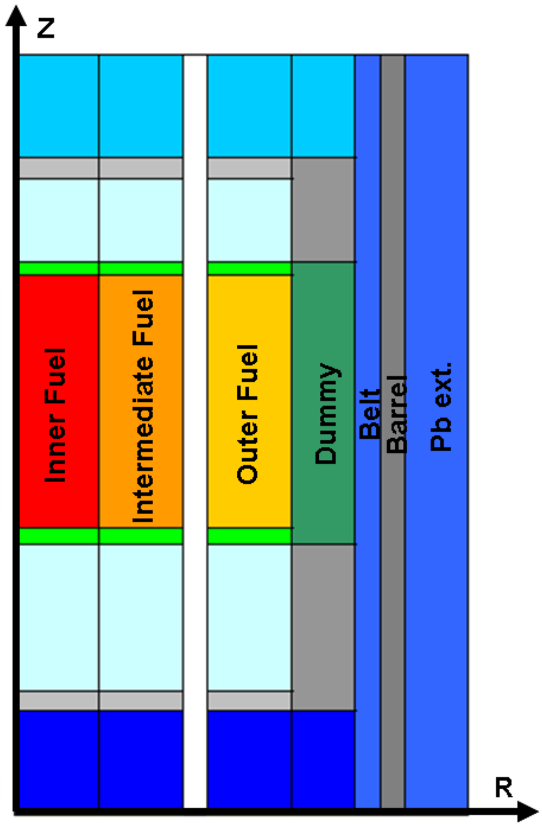

Figure D.5: ELSY core layout [ERANOS model]

The fuel composition adopted is indicated in Table D.2 and the SA dimensions in Table D.3. Other data (as material composition, control rods model and dimensions) are in agreement with [27, 26, 216, 215] and not here summarized.

The reactivity swing, CR worth and void parameters evaluated for the ELSY model are in agreement with the general results obtained in the project [68]. The reactivity swing assuming all CR extracted is indicated in Figure D.7 and CR worth and void effect in Table D.4.

Starting from the ELSY reference design, two modified models (ELSY-1-BLANKET-RING and ELSY-2-BLANKET-RINGS) have been generated.

In particular, as described in Par. 5.2.1, the following models have been assessed:

1- ELSY-1-BLANKET-RING model: one ring of depleted uranium oxide sub-assemblies (90 SAs) has been added to the core periphery, replacing one ring of steel reflector, with respect to the reference configuration. The core layout is shown in Figure 5.12;

2- ELSY-2-BLANKET-RINGS model: two rings of depleted uranium oxide sub-assemblies (180 SAs)

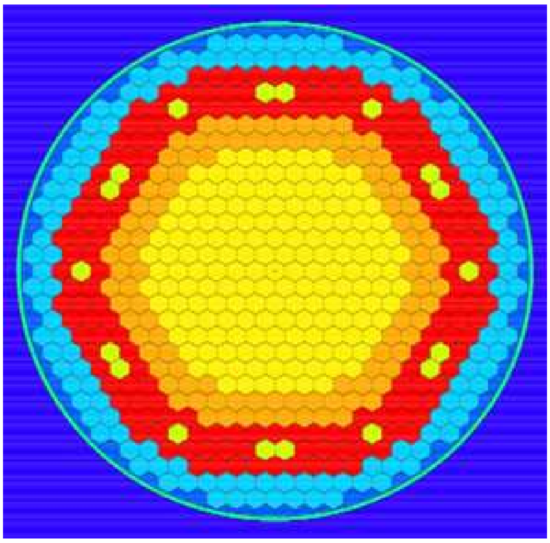

Figure D.6: ELSY core cross section [27]

Isotope	Inner Fuel	Intermediate fuel	Outer Fuel
Pu content (%at.)	15.03	15.96	19.02
	Composition at 20°C (at/cm^3)		
U234	6.090E-07	6.027E-07	5.820E-07
U235	8.166E-05	8.082E-05	7.804E-05
U236	2.013E-06	1.992E-06	1.924E-06
U238	1.987E-02	1.967E-02	1.899E-02
PU238	8.291E-05	8.805E-05	1.052E-04
PU239	2.013E-03	2.138E-03	2.555E-03
PU240	9.518E-04	1.011E-03	1.208E-03
PU241	2.143E-04	2.276E-04	2.720E-04
PU242	2.690E-04	2.856E-04	3.413E-04
O16	4.627E-02	4.630E-02	4.640E-02

Table D.2: Fuel isotopic composition adopted for the ELSY-REFERENCE model [27]

SA characteristics	
Number of SA	433
Number of CR	18
Number of pin per SA	169
Fuel pin radius (cm)	0.470
External cladding radius (cm)	0.530
HEX flat-to-flat internal (cm)	20.5
HEX flat-to-flat external (cm)	21.0
Active height (cm)	120

Table D.3: SA and core dimensions adopted (ELSY) [27]

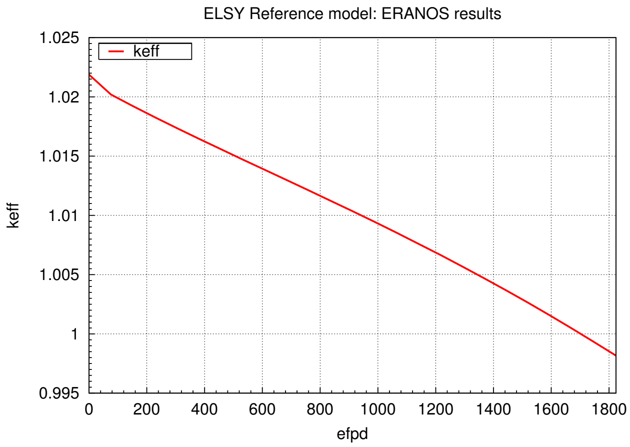

Figure D.7: Reactivity swing: ELSY reference model

Control rod status	ERANOS	ELSY project	Ref.
Insertion of 6 CR (pcm)	1405	1304	D7
Insertion of 12 SDR (pcm)	1600		
Insertion of 18 SDR (pcm)	3005	3000	D7
Void effect (pcm)	3800	4000	D8

Table D.4: Comparison between the ERANOS model and the project results [27, 26]

have been added to the core periphery, replacing two rings of steel reflector, with respect to the reference configuration. The core layout is depicted in Figure 5.12.

In order to maintain the same criticality level as the reference model, the Pu content has been homogeneously increased (in order to compensate the increased neutron leakage term) in the three fuel zones. The compositions adopted are indicated in Table D.5.

As indicated in Par. 5.2.1, the adoption of radial blanket improves the Pu balance of the systems, giving a positive contribution also to the reactivity swing (see Figure D.8).

D.2 The ESFR model

The ESFR model, a large sodium-cooled fast reactor concept, is investigated within the large integrated Collaborative Project on European Sodium Fast Reactor (CP-ESFR [58]) realized under the aegis of the EUROATOM 7th Framework Programme. The activities have been oriented to system performances, safety aspects and fuel cycle issues.

The model considered is the reference model assumed in project as provided by CEA [28]. Starting from this model and from the large experience gained internationally from studies on experimental, prototype and commercial size reactors designed and operated in the past, a further progress in exploring potential of sodium-cooled reactors with fast neutron spectrum is performed [29].

In the study, the ESFR reference design, oxide fuel, provided by [28] has been adopted as reference.

In this Appendix only the model and the main results are summarized.

The activity performed concerning the sodium void effect reduction in cores with oxide (ESFR-OXIDE

Isotope	Inner Fuel	Intermediate fuel	Outer Fuel
Pu content (%at.)	15.10	16.03	19.10
	Composition at 20°C (at/cm^3)		
U234	6.090E-07	6.027E-07	5.820E-07
U235	8.166E-05	8.082E-05	7.804E-05
U236	2.013E-06	1.992E-06	1.924E-06
U238	1.987E-02	1.967E-02	1.899E-02
PU238	8.331E-05	8.847E-05	1.057E-04
PU239	2.023E-03	2.149E-03	2.567E-03
PU240	9.564E-04	1.016E-03	1.214E-03
PU241	2.154E-04	2.287E-04	2.733E-04
PU242	2.703E-04	2.870E-04	3.430E-04
O16	4.631E-02	4.634E-02	4.645E-02

Table D.5: Fuel isotopic composition adopted for the ELSY-1-BLANKET-RING and ELSY-2-BLANKET-RINGS models

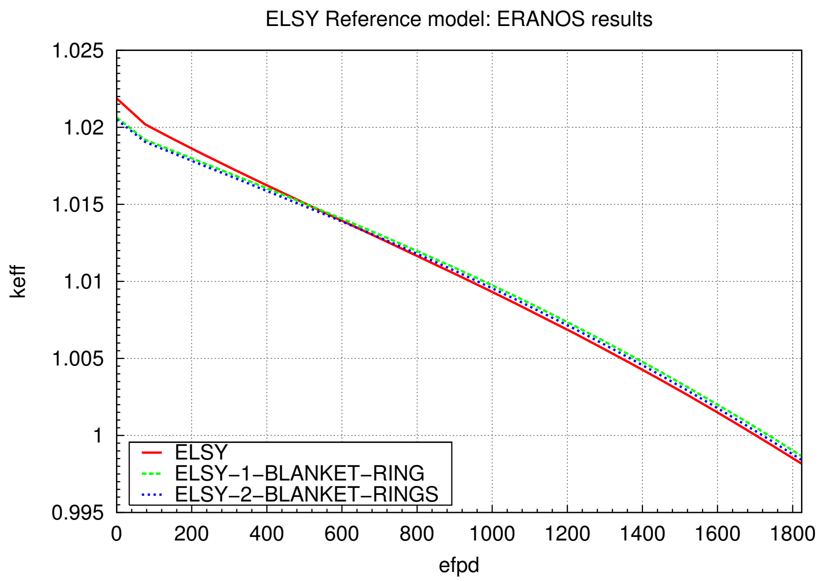

Figure D.8: Reactivity swing: comparison between models considered core) and carbide (ESFR-CARBIDE core) fuels as well as the analysis of the MAs burning has not be included because beyond the aims of the present activities. However, the main results obtained can be found in [29, 30, 31].

D.2.1 The ERANOS model: ESFR HEX-z core

The ESFR-OXIDE core reference configuration, provided by CEA in 2008, is described in detail in the "Working Horses" document [28]. In this section, main characteristics of the oxide core are highlighted.

The oxide core consists of two parts, inner and outer cores, the Pu content being 14.43 at% and 16.78% in the 225 inner and 228 outer core SAs, respectively.

The fuel residence time at EOL is assumed to be equal to 2050 equivalent full power days (efpd). The fuel reloading strategy assumes a replacement of about a 1/5 of the core after 410 efpd.

The average and maximum core burn-up values are 100 GWd/tHM and 145 GWd/tHM, respectively for the average power density of 206 W/cm3. The control system includes 9 DSDs (Diverse Shutdown Devices) and 24 CSDs (Control and Shutdown Devices). The CSD rods contain natural boron carbide (B_4C with ca. 19.9% of B10) whereas the DSD rods contain enriched boron carbide (B_4C with ca. 90% of B10). The radial reflector containing EM10 steel surrounds the core and includes three rings of SAs [28, 29].

The core cross section and the axial configuration are respectively shown in Figure D.9 and Figure D.10 [28, 29].

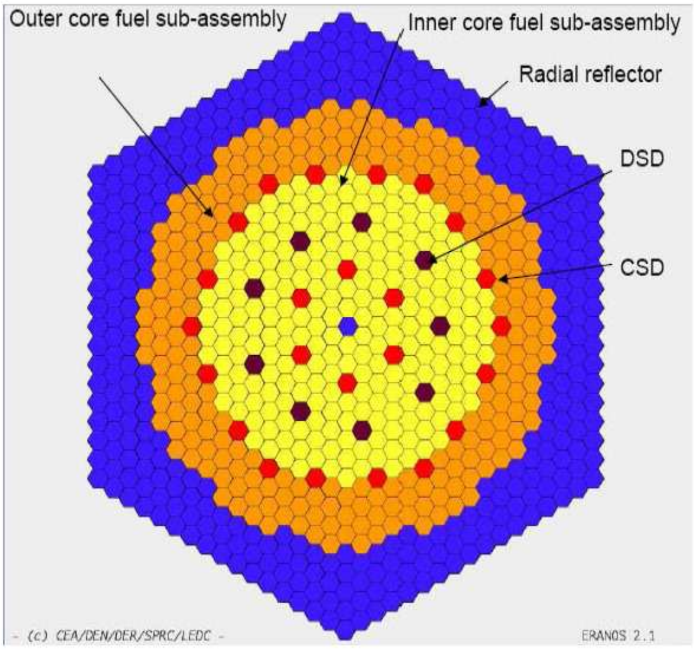

Figure D.9: ESFR core cross section [28, 29]

The fuel composition adopted is indicated in Table D.6. The other material composition, and CSD and DSD model are included in [28, 29].

The main parameters concerning the core design are included in Table D.7.

This configuration shows a positive void effect (1,400 pcm) as indicated in Table D.8 partially compensated by the Doppler effect (1,000 pcm).

The reactivity swing is represented in Figure D.11.

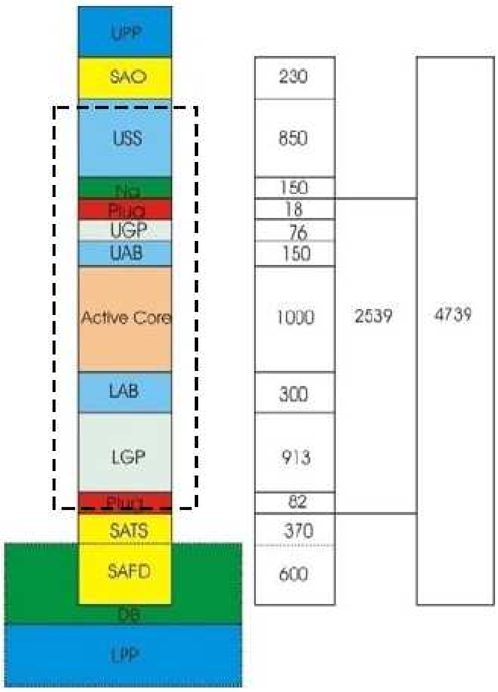

Figure D.10: ESFR axial configuration [28, 29]

ISOTOPE	Inner Fuel	Outer Fuel
Pu content (%at.)	15.10	19.10
	Composition at 20°C (at/cm^3)	
O16	4.356E-02	4.359E-02
U235	4.719E-05	4.593E-05
U238	1.859E-02	1.809E-02
PU238	1.141E-04	1.327E-04
PU239	1.508E-03	1.755E-03
PU240	9.398E-04	1.094E-03
PU241	2.597E-04	3.022E-04
PU242	3.258E-04	3.792E-04
AM241	2.461E-05	2.864E-05

Table D.6: Fuel isotopic composition adopted for the ESFR-REFERENCE model [28, 29]

SA characteristics	
Number of SA	453
Number of CR	33
Number of pin per SA	271
Fuel pin radius (cm)	0.4715
External cladding radius (cm)	0.5365
HEX flat-to-flat internal (cm)	20.63
HEX flat-to-flat external (cm)	21.08
Active height (cm)	120

Table D.7: SA and core dimensions adopted (ESFR) [29, 30, 31]

Reactivity coefficients		ESFR-OXIDE reference core
		pcm
Core void effect	Void (In + Out)	1402
	Void (In)	856
	Void (Out)	612
Extended void effect	Void (In + Out + Upper)	1014
	Void (In + Out + Gap)	1607
	Full Void	-875
	Doppler	-1062
Keff (BOL)		1.00974

Table D.8: Reactivity effect: ESFR-OXIDE Reference configuration [29, 30, 31]

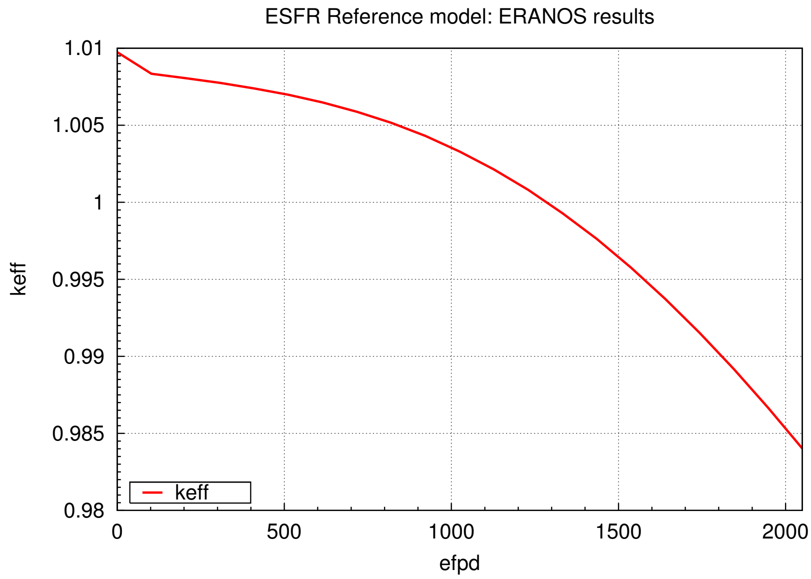

Figure D.11: Reactivity swing: ESFR reference model [29]

Additional information can be found [29, 30, 31] as well as the description of the optimized core with reduced void effect.

In fact, one of the possibilities investigated within the present activity, has been the adoption of larger Na plenum above the active zone and a lower fertile blanket with 5% vol. AmO_2[3] in order to increase the leakage term under voided conditions and to reduce the positive void worth.

By the modifications of the axial zone (upper part) the positive void worth is reduced (see Table D.9) and the core configuration remains almost unchanged as indicated by the comparison of the radial power profile (see Figure D.12).

	Reference		Optimized	
	BOL	EOC1	BOL	EOC1
	pcm			
Core void effect	+1402	+1588	+1270	+1457
Extended void effect	+1014	+1241	-243	+33
Doppler	-1062	-907	-987	-852

Table D.9: Void and Doppler effects for reference and optimized configuration [29, 30, 31]

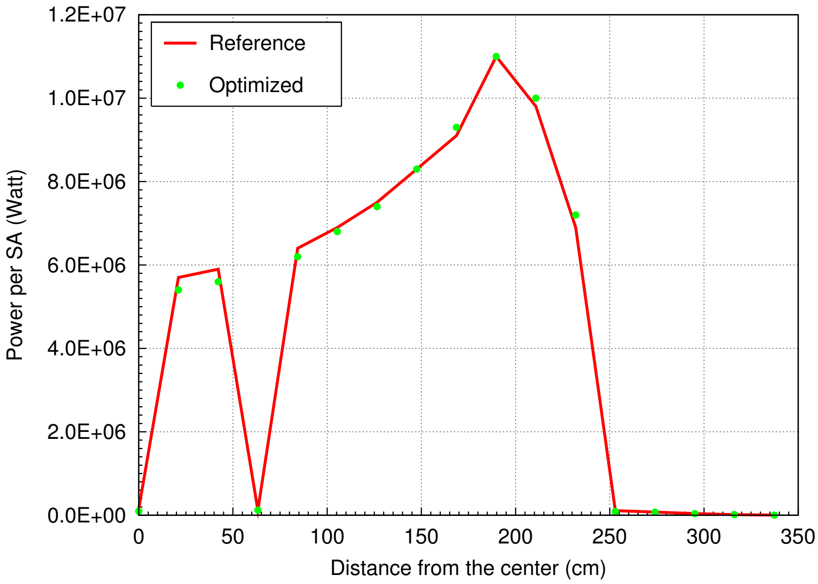

Figure D.12: Radial Power Distribution at BOL [29, 30, 31]

The low extended void reached for the optimized configuration[4] offers an opportunity for introduction of Minor Actinides (MAs), in particular Am, into the reactor.

Two cases have been analyzed: CONF-2-2% and CONF-2-4% with fertile blanket, with 2% and 4% weight of Am in the blanket and about 1.7% and 3.5% in the core, respectively [29, 30, 31].

[3] AmO_2 has been inserted for proliferation resistance issues [186]. It helps, indeed, to increase the Pu238 content (> 12%) and, hence, to reduce the Pu239 (<80%).

[4] The configuration called here "optimized" is not the final ESFR optimized configuration (project ongoing up to 2013) but it is an intermediate configuration adopted for further studies also in the project.

The reactivity swing is slightly negative, about -500 pcm between BOL and EOL, for CONF-2-2% and positive, about 1000 pcm, for CONF-2-4%.

The Pu build-up in the core at EOL is slightly smaller (due to higher Pu enrichment at BOL) as compared to REF for CONF-2-2% and slightly higher for CONF-2-4%. MAs balance in the core is definitely negative (unlike REF) at EOL, the reductions being about 300 kg and 900 kg for CONF-2-2% and CONF-2-4%, respectively.

The Pu production in the reactor is similar for CONF-2, CONF-2-2% and CONF-2-4%. The MAs mass variation is about -400 and -1200 kg at EOL for CONF-2-2% and CONF-2-4%, respectively [30, 31].

Burn-up calculations show deterioration of safety parameters with time. However, the extended SVRE goes from about +120 to +670 pcm and from about +400 to +800 pcm in CONF-2-2% and CONF-2-4%, respectively after 1230 days of irradiation (at that time the core average composition is assumed to be similar to that for the end of equilibrium cycle, EOC). The Doppler constants at EOC are smaller by about 20% compared to those at BOL, which are about -780 and -610 pcm, for CONF-2- 2% and CONF-2-4%, respectively [30, 31].

One may expect that addition of about 2% of Am (or maybe more) into the core fuel does not lead to over-criticality after Na boiling onset. This condition may not be necessary or sufficient for proving the reactor safety, but seems to be a reasonable one for making scoping design studies. With this amount of Am in the core and a several times higher content of Am in the low fertile (or absorber) blanket one can obtain a definitely negative MAs balance in the optimized ESFR. The Pu balance can be made positive or negative depending on the blanket option and Am content [30, 31].

On the basis of these studies, the Am and MAs maximum content for scenario analysis have been fixed, as indicated in Chapter 5 .

Appendix E

Appendix: Data adopted during the Preliminary Scoping Study

In this Appendix, several data adopted during the preliminary scoping study are summarized. In particular, the data used in NFCSS for modeling the Italian historical scenario and the Belgian scenarios are here included.

E.1 Historical Data for Italy

In order to simulate the Italian historical scenario, the following data have been adopted. The data assumed (e.g. load factor) for each reactor operated in the period 1963-1990 are in agreement with the IAEA PRIS database [17].

In particular, Table E.1 refers to the LATINA GCR reactor (connected to the grid in 1963), Table E.2 refers to the GARIGLIANO BWR reactor (connected to the grid at the beginning of 1964), Table E.3 refers to the ENRICO FERMI (TRINO) PWR reactor (connected to the grid at the end of 1964), and Table E.4 refers to the CAORSO BWR (connected to the grid in 1978 and operated for only 10 years).

All these data have been modeled in detail in the NFCSS; simulation results are included in Chapter 3.

E.2 Data adopted for Belgian scenarios

The electricity and nuclear capacities evolutions adopted for the Belgian study are here summarized. These envelops have been assessed on the basis of the IAEA projections as described in Chapter 3, [15].

In Table E.5, the electricity and nuclear capacities evolutions for the LOW Belgian scenario have been indicated. These data have been adopted for the preliminary scoping study (see Par. 3.3).

In Table E.6 are summarized the electricity and nuclear capacities evolutions for the HIGH Belgian scenario proposed during the preliminary scoping study (see Par. 3.3).

Year	Energy	Capacity	Load Factor (%)		Annual Time Online
	GWhe	MWe	Annual	Cumulative	hours
1963	0	153	-	-	-
1964	0	153	-	-	-
1965	0	153	-	-	-
1966	0	153	-	-	-
1967	0	153	-	-	-
1968	0	153	-	-	-
1969	0	153	-	-	-
1970	1191	155	87.72	12.66	-
1971	845.3	160	60.31	18.84	6397
1972	1204.1	160	85.67	26.53	7751
1973	654.6	150	49.82	28.79	5401
1974	953.8	153	71.16	32.61	7455
1975	948.3	153	70.75	35.77	7362
1976	946.6	153	70.43	38.42	7980
1977	1076.4	153	80.31	41.39	7653
1978	1184.9	153	88.41	44.51	8351
1979	789	153	58.87	45.4	5284
1980	894	153	66.52	46.64	5981
1981	895.4	153	66.81	47.75	6182
1982	872.2	153	65.08	48.66	5997
1983	1274	153	95.05	50.97	8641
1984	933	153	69.42	51.84	6160
1985	1186.6	153	88.53	53.5	8331
1986	1131	153	84.39	54.84	7663
1987	0	153	-	52.57	-

Table E.1: Italian historical scenario: LATINA data [17]

Year	Energy	Capacity	Load Factor (%)		Annual Time Online
	GWhe	MWe	Annual	Cumulative	hours
1964	0	150	-	-	-
1965	0	150	-	-	-
1966	0	150	-	-	-
1967	0	150	-	-	-
1968	0	150	-	-	-
1969	0	150	-	-	-
1970	0	160	-	-	-
1971	1163.7	160	83.03	12.41	7936
1972	436	160	31.02	14.84	3210
1973	970	150	73.82	21.24	7197
1974	719.6	150	54.76	24.53	5818
1975	470.7	150	35.82	25.54	4154
1976	1144.9	151	86.32	30.56	8063
1977	448.3	151	33.89	30.81	4003
1978	456.5	151	34.51	31.07	3402
1979	0	151	-	29.02	-
1980	0	151	-	27.22	-
1981	0	151	-	25.63	-
1982	0	150	-	24.23	-

Table E.2: Italian historical scenario: GARIGLIANO data [17]

Year	Energy	Capacity	Load Factor (%)		Annual Time Online
	GWhe	MWe	Annual	Cumulative	hours
1964	0	260	-	-	-
1965	0	260	-	-	-
1966	0	260	-	-	-
1967	0	260	-	-	-
1968	0	260	-	-	-
1969	0	260	-	-	-
1970	1243	256	55.43	9.12	5723
1971	1355.4	256	60.44	16.36	6000
1972	1985.4	256	88.29	25.29	8475
1973	1357	247	62.72	29.28	6324
1974	1560.1	247	72.1	33.4	7079
1975	2207.4	260	96.92	39.25	8706
1976	1514.4	247	69.8	41.72	6591
1977	1826	260	80.17	44.72	6952
1978	2094.9	260	91.98	48.14	8401
1979	705	260	30.95	46.98	3175
1980	0	260	-	44	-
1981	0	260	-	41.38	-
1982	0	260	-	39.06	-
1983	0	260	-	36.98	-
1984	1631	260	71.41	38.73	6415
1985	1295.3	260	56.87	39.6	5752
1986	2016.2	260	88.52	41.84	8417
1987	159	260	6.98	40.31	1903
1988	0	260	-	38.62	-
1989	0	260	-	37.06	-
1990	0	260	-	35.62	-

Table E.3: Italian historical scenario: TRINO data [17]

Year	Energy	Capacity	Load Factor (%)		Annual Time Online
	GWhe	MWe	Annual	Cumulative	hours
1978	459	652	13.15	-	1878
1979	1003	548	20.89	-	2708
1980	1220	548	25.34	-	1976
1981	1659.3	840	49.86	-	2450
1982	5732.5	840	77.9	77.9	7906
1983	4312	873	56.38	66.94	5858
1984	4065	860	53.81	62.54	5769
1985	3975	860	52.76	60.09	6073
1986	5300	860	70.35	62.15	6648
1987	0	860	-	51.78	-
1988	0	860	-	44.36	-
1989	0	860	-	38.81	-
1990	0	860	-	34.82	-

Table E.4: Italian historical scenario: CAORSO data [17]

Year	MWe	MWe	Share (%)		Year	MWe	MWe	Share (%)
	Nuclear	Total	Nuclear			Nuclear	Total	Nuclear
2008	5761	16183	35.6		2030	4127	20143	20.5
2009	5761	16344	35.3		2031	4036	20344	19.8
2010	5761	16508	34.9		2032	3947	20548	19.2
2011	5761	16673	34.6		2033	3860	20753	18.6
2012	5761	16840	34.2		2034	3775	20961	18.0
2013	5761	17008	33.9		2035	3692	21170	17.4
2014	5761	17178	33.5		2036	3611	21382	16.9
2015	5761	17350	33.2		2037	3531	21596	16.4
2016	5634	17523	32.2		2038	3454	21812	15.8
2017	5510	17699	31.1		2039	3378	22030	15.3
2018	5389	17876	30.2		2040	3303	22250	14.9
2019	5271	18054	29.2		2041	3231	22473	14.4
2020	5155	18235	28.3		2042	3160	22697	13.9
2021	5041	18417	27.4		2043	3090	22924	13.5
2022	4930	18601	26.5		2044	3022	23154	13.1
2023	4822	18787	25.7		2045	2956	23385	12.6
2024	4716	18975	24.9		2046	2891	23619	12.2
2025	4612	19165	24.1		2047	2827	23855	11.9
2026	4511	19357	23.3		2048	2765	24094	11.5
2027	4411	19550	22.6		2049	2704	24335	11.1
2028	4314	19746	21.9		2050	2645	24578	10.8
2029	4219	19943	21.2					

Table E.5: Scenario LOW: total and nuclear capacity installed in Belgium according to IAEA trends for the period 2008-2050 [15]

Year	MWe	MWe	Share (%)		Year	MWe	MWe	Share (%)
	Nuclear	Total	Nuclear			Nuclear	Total	Nuclear
2008	5761	16183	35.6		2030	6590	29710	22.2
2009	5761	16636	34.6		2031	6649	30541	21.8
2010	5761	17102	33.7		2032	6709	31397	21.4
2011	5761	17580	32.8		2033	6769	32276	21.0
2012	5761	18073	31.9		2034	6830	33179	20.6
2013	5761	18579	31.0		2035	6892	34108	20.2
2014	5761	19099	30.2		2036	6954	35063	19.8
2015	5761	19634	29.3		2037	7016	36045	19.5
2016	5813	20183	28.8		2038	7079	37055	19.1
2017	5865	20748	28.3		2039	7143	38092	18.8
2018	5918	21329	27.8		2040	7207	39159	18.4
2019	5971	21927	27.2		2041	7272	40255	18.1
2020	6025	22541	26.7		2042	7338	41382	17.7
2021	6079	23172	26.2		2043	7404	42541	17.4
2022	6134	23821	25.8		2044	7470	43732	17.1
2023	6189	24488	25.3		2045	7538	44957	16.8
2024	6245	25173	24.8		2046	7605	46215	16.5
2025	6301	25878	24.4		2047	7674	47509	16.2
2026	6358	26603	23.9		2048	7743	48840	15.9
2027	6415	27347	23.5		2049	7813	50207	15.6
2028	6473	28113	23.0		2050	7883	51613	15.3
2029	6531	28900	22.6					

Table E.6: Scenario HIGH: total and nuclear capacity installed in Belgium according to IAEA trends for the period 2008-2050 [15]