Program on Technology Innovation: Advanced Fuel Cycles—Impact on High-Level Waste Disposal

Analysis of Deployment Scenarios of Fast Burner Reactors in the U.S. Nuclear Fleet

1016643

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Technical Update, September 2008

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REPORT SUMMARY

This report presents the results of a dynamic simulation analysis for deployment of advanced light water reactors (LWRs) and fast burner reactors, as proposed by the Global Nuclear Energy Partnership (GNEP) program. Conditions for the analysis were selected for their potential to challenge the nuclear fuel simulation codes that were used, due to the large variations in nuclear fuel composition for the burner reactors before equilibrium conditions are approached. The analysis was performed in a U.S. context assuming 1) a zero-growth scenario with regard to installed nuclear capacity, and 2) timing for fleet renewal driven by the age of existing nuclear power plants.

Background

Current civil uses of nuclear power in the United States are based on a once-through fuel cycle involving the irradiation of low-enriched uranium fuel in LWRs and the subsequent storage and eventual disposal of the spent fuel. However, continued use of nuclear power may be predicated on improved economics and sustainability, especially when it is assumed that applications of nuclear technology may expand beyond production of electricity to areas such as the production of hydrogen for industrial and transportation applications. Such developments may require adoption of a different fuel cycle. Past and more recent findings—published by EPRI, the U.S. electric utility industry's Advanced Reactor Corporation, National Academy of Sciences, and Massachusetts Institute of Technology—have been in general agreement with regard to support for the present U.S. policy relying on the once-through fuel cycle because of its simplicity, economic advantages, and non-proliferation benefits. However, there is also broad agreement that R&D should be conducted on selected topics to support the safe and cost-effective future application of commercial spent fuel reprocessing and recycling.

Objectives

- To model a transition scenario from the current U.S nuclear fleet (entirely composed of LWRs) to a future fleet that consists of evolutionary LWRs and fast burner reactors, using EDF's nuclear fuel cycle simulation code, TIRELIRE-STRATEGIE.
- To assess the capabilities of the nuclear fuel simulation code, Dynamic Analysis of Nuclear Energy System Strategies (DANESS).

Approach

First, the research team collected the data necessary to model the deployment of the U.S. nuclear fleet from 1967 to 2000. Next, a zero-growth nuclear scenario relying on a mixed fleet of LWRs and fast burner reactors was selected and compared to a once-through, LWR-only scenario. To provide additional validation of the results, the fractions of LWRs and fast burner reactors at equilibrium conditions were chosen to be those predicted by ERANOS, a state-of-the-art neutronic reference code for fast-spectrum system analysis. For the fast burner reactor design, the research team relied on the CAPRA (French acronym for "Enhanced Plutonium Burning in Fast Reactors") design developed by the Commissariat à l'Énergie Atomique (Atomic Energy Commission or CEA).

The team conducted simulations of the deployment scenarios using two different codes: EDF R&D's TIRELIRE-STRATEGIE and DANESS. Specifically, calculations were performed to obtain annual mass fluxes and inventories (plutonium, minor actinides) between reactors and associated fuel cycle facilities.

Results

On the basis of CEA studies of the CAPRA fast burner design, the transuranic (TRU) burner model implemented in the TIRELIRE-STRATEGIE code turned out to be very accurate when compared to reference calculations performed with the ERANOS code. This is because the TIRELIRE-STRATEGIE code's physical models allow accurate calculation of fresh fuel content and used fuel compositions. However, a satisfactory simulation of scenarios involving fast burner reactors by DANESS was not obtained. The deployment of CAPRA fast burner reactors allows a quasi-immediate stabilization of TRU nuclides, with the total TRU inventory being reduced by a factor of two by the end of the century compared to a once-through fuel cycle scenario.

Dynamic simulations of GNEP-type scenarios, consisting of the deployment of a significant number of fast burner reactors, are challenging due to substantial variations in the burner fresh fuel composition during the transient period before equilibrium conditions are approached. Simpler calculations using data giving fixed compositions for fresh and used fuels introduce errors of the order of 10%–20%. An important insight derived from this work is the need for a powerful technical environment, such as benchmarking provided by the ERANOS code, for validating the proper use of nuclear fuel simulation codes, such as TIRELIRE-STRATEGIE and DANESS.

EPRI Perspective

New strategies may be required to balance the needs for 1) sustainability—particularly, the shift to a plutonium economy and reduction in high level waste (HLW) burden on permanent geologic repositories, 2) operational efficiencies, and 3) diversion resistance of plutonium-based fuel cycles. Such strategies would rely on interim storage of spent fuel as well as partitioning and transmutation of plutonium and minor actinides before final HLW disposal in a permanent geologic repository. Although equilibrium system analysis gives indications on the end states of any transition between current and future nuclear energy systems, closing the fuel cycle introduces complex dynamic feedback effects with regard to mass flows, inventories, and isotopic fuel and HLW compositions. Integrated process models simulating nuclear energy systems—from uranium mining to final waste disposal—are needed to properly conduct comprehensive assessments of nuclear energy system strategies and to select the most promising development paths.

Keywords

Nuclear Fuel Cycles
Reprocessing
Recycling
Minor Actinides
Fast Burner Reactors
Waste Management

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1 INTRODUCTION

This document provides the results of an investigation conducted under the EDF-EPRI agreement entitled "Analysis of deployment scenarios of fast burner reactors in the U.S. nuclear fleet" [1].

Chapter 2 details the results of the performance of the CAPRA burner design in the French fleet under equilibrium conditions calculated with the neutronic code ERANOS.

Chapter 3 details the scenario codes used in this study, TIRELIRE-STRATEGIE and DANESS.

Chapter 4 details the main hypotheses concerning the dynamic scenarios of evolution of the U.S. nuclear fleet analyzed in this study, as well as the development of a burner model for the scenario codes.

Chapter 5 summarizes and discusses the results of the dynamic scenarios obtained with TIRELIRE-STRATEGIE.

Chapter 6 presents the results obtained with the DANESS code and compares them to those obtained with TIRELIRE-STRATEGIE.

Appendix A provides an analysis of the performance of the CAPRA design under equilibrium conditions in a fleet composed of PWR and CAPRA reactors to supplement the information contained in an earlier EPRI report.

2
SCENARIO CODES USED IN STUDY

2.1 TIRELIRE-STRATEGIE

TIRELIRE-STRATEGIE [5] is a calculation code aimed at simulating the operation of a nuclear fleet and the associated fuel cycle facilities over a long period of time (decades, even centuries). It is used to analyze the consequences of strategic choices related to the nuclear fleet composition (reactors and fuels) and other fuel cycle facilities' features. A template nuclear fuel cycle modeled in TIRELIRE-STRATEGIE is shown on Figure 2-1.

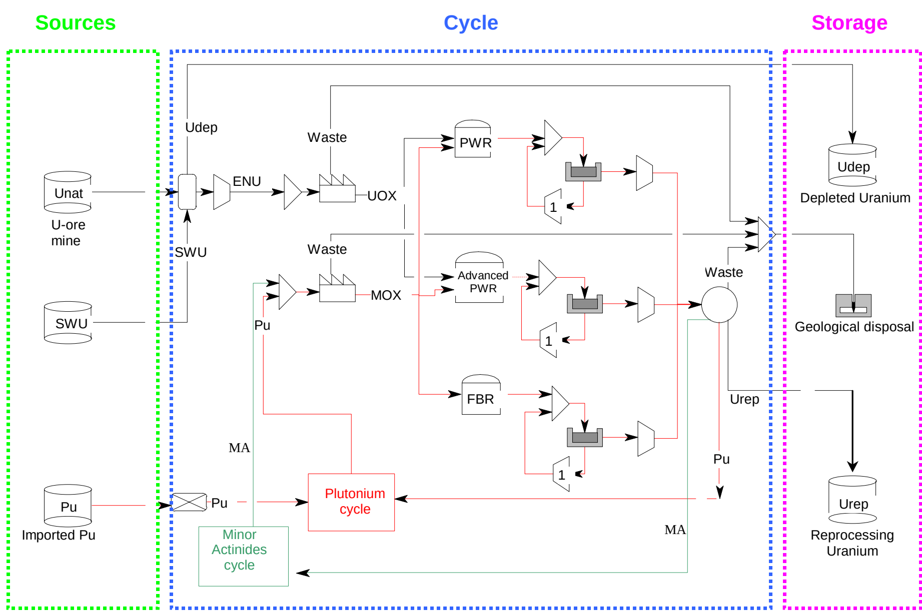

Figure 2-1
Template of the Nuclear Fuel Cycle in TIRELIRE-STRATEGIE

TIRELIRE-STRATEGIE allows nuclear scenarios simulation to:

1. Comply with industrial requirements [such as spent uranium oxide (UOX) and mixed oxide (MOX) fuels reprocessing capacity limitation, interim storage capacity, cooling time before reprocessing, delay for fresh fuel fabrication, losses during reprocessing or fabrication, number and characteristics of reactors being reloaded for each year], and
2. Take into account strategic choices (i.e., type of reactors and fuel management used for the nuclear fleet renewal, minor actinide incineration rates, and interim storage management).

2.1.1 General Features

The main parameters defining the dynamics, year per year, of a nuclear scenario in TIRELIRE-STRATEGIE are basically:

- Nuclear fleet installed capacity (in GWe), which is related to the electricity production via the average fleet load factor.
- Installed capacity of each nuclear system type [i.e., pressurized water reactors (PWRs), fast breeder reactors (FBRs), high-temperature gas reactors (HTGRs), etc.].
- Priority level associated to the deployment of each nuclear system type.
- Maximum deployment rate for the total fleet and for each reactor type.
- Minor actinides [MAs (Np, Am and Cm)] fuel fabrication rates and reprocessing losses for all actinides from each fuel type.
- Reprocessing rate for each fuel type.
- Spent nuclear fuel (SNF) cooling time before reprocessing and delay before fresh fuel fabrication, for each reactor type.

Each reactor type is characterized by:

- Maximum lifetime.
- Core heavy metals (HM) mass and HM mass reload (taking into account its reload batch size).
- Fuel type that can be loaded in each reactor type (i.e., UOX, MOX, plutonium on thorium support for advanced PWRs).
- Fuel irradiation time.
- Parameters specifying the models for the calculation of the discharged fuel isotopic composition (*evolution model*) and the Pu content for fresh MOX fuel (*equivalence model*). The *evolution* and *equivalence* models are different for PWR and FBR (Figure 2-2), and will be presented in the subsections dealing with PWR and FBR modeling.

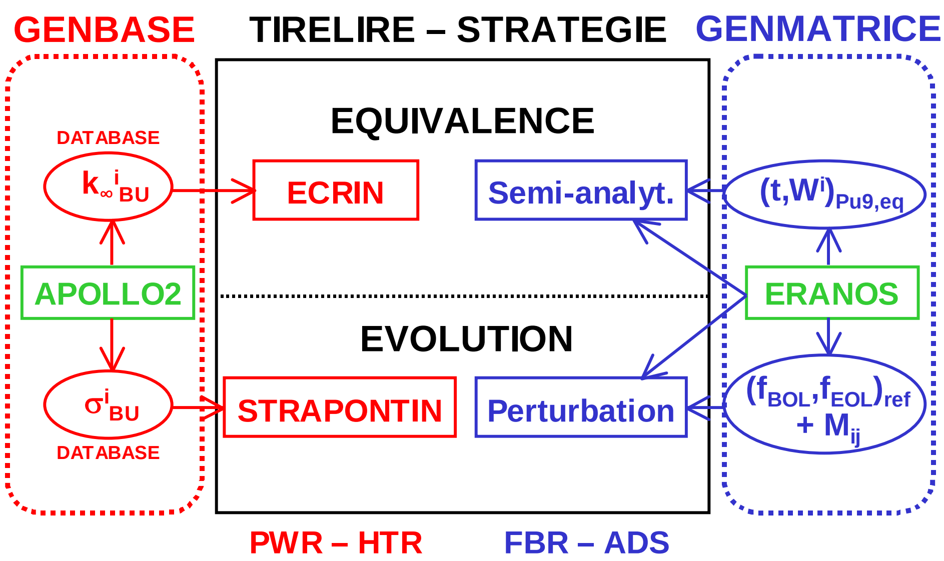

Figure 2-2
Scheme of TIRELIRE-STRATEGIE and of the Associated Procedures GENBASE and GENMATRICE for Neutronics Database Creation

The TIRELIRE-STRATEGIE code calculates the power capacity to be installed every year, on the basis of the total power demand and the number of decommissioned units, if any. Power demand will be satisfied by the reactor types selected in the analyzed scenario, according to their priority level and their maximum introduction rate, i.e., if the priority is 1 for PWR, 2 for FBR, and 3 for HTGR, then PWR will be deployed up to their maximum deployable power. When this latter limit is reached, the next reactor type in the priority order is considered (in this case FBR), and so on.

Resource availability depends on fuel type. If uranium availability is assumed to be unlimited, the plutonium used for MOX fuel fabrication is clearly a finite resource. This requires a special treatment for MOX-fueled reactor types such as fast breeder reactors (FBRs). Thus, if a shortfall in Pu inventory develops when considering fresh MOX fuel fabrication requirements in a given year of the simulation, two cases are possible, depending on which kind of fuel is allowed for fueling the FBRs:

1. If FBRs can be fueled with only MOX fuel, the simulation will go back to the year corresponding to the last FBR starting up and FBR deployment capacity will be lowered by one unit, the corresponding capacity being replaced by the next reactor type in the priority order. If shortfalls in Pu inventory appear again later in the simulation, the process repeats itself. As a result, the code automatically calculates deployment rate and maximum FBR installed capacity compatible with the Pu inventory available over time in the simulated fuel cycle scenario;

2. If FBRs can be started up with UOX fuel, deployment of FBRs will be enabled by using UOX fuel and a first amount of Pu will be obtained from reprocessing the UOX SNF. In the subsequent operating period, the FBR will be fed with a mixed UOX/MOX fuel. Then, when a sufficient Pu stock is available to keep the core almost self-breeder or breeder, the core will be converted to only MOX fuel. The performance of this particular strategy is analyzed in more details in References [6] and [7].

Once the nuclear fleet composition is determined, the discharged fuel composition is assessed by means of the *evolution* models detailed in the following section. After taking SNF cooling into consideration, reprocessing is modeled considering:

- The corresponding mass flow rates for each fuel type;
- The fuel management strategy for each reactor type.

As a result of SNF reprocessing, fission products (FPs) are sent to a permanent geologic repository, whereas Pu and MAs, depending on the fuel management strategy, either are sent to a permanent repository or feed the corresponding in-cycle stocks for further in-cycle handling.

The main results of a simulation (whose calculation time is about half a minute for a mixed PWR-FBR scenario over one century on a 1280 MHz UltraSPARC IIIi processor) are:

- Natural uranium consumption;
- Required separation work units (SWU);
- Mass flows for each reactor type, fuel cycle facility, interim storage, and final repository.

The fuel physical composition is represented by a simplified actinides chain of 18 nuclides from ^{232}Th to ^{245}Cm (all Pu isotopes, whereas for MAs only ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm and ^{245}Cm are explicitly considered; other isotopes, like ^{242}Am or ^{242}Cm, for examples, are being taken into account implicitly in the decay chain). With regard to the fission products, only their total mass is calculated.

2.1.2 PWR Modeling

Different PWR designs are modeled, including:

- Current generation PWR fed with UOX fuel (with different ^{235}U enrichment corresponding to different discharge burn-up values and reload batch size).
- Current generation PWR fed with standard MOX fuel (Pu with depleted uranium as support).

Additionally, the following Advanced PWRs are available, all derived from the evolutionary pressurized reactor (EPR) design:

- EPR fed with standard UOX or MOX fuel (Pu with depleted uranium as support).
- EPR fed with MIX fuel (Pu with enriched uranium as support).
- EPR fed with MOX-Ue fuel (Pu with enriched uranium as support in an over-moderated MOX sub-assembly with 36 extra water holes).

- EPR fed with Pu on thorium as support.

As a general feature, the plutonium content of a standard MOX fuel (depleted uranium as support) and the ^{235}U enrichment of the uranium support for MIX and MOX-Ue fuels vary along a scenario, as a consequence of the plutonium isotopic vector variation, itself changing with time (because of β^- decay of ^{241}Pu to ^{241}Am), and according to the fuel management strategy for each reactor type. The calculation of the Pu content (or the ^{235}U enrichment for MIX and MOX-Ue fuels) allowing to meet the targeted cycle length for a given reactor type is referred to as the *equivalence* between different MOX fuels.

This problem is solved by the ECRIN code, which is coupled to TIRELIRE-STRATEGIE. The equivalence between different MOX fuels in ECRIN is based on energy release over the fuel cycle, expressed by a targeted burn-up and with a nearly zero reactivity value at End of Cycle, EOC. The calculation is performed in two steps:

1. Burn-Up Calculation With The Transport Cell Code APOLLO2 [8] – Using The CEA93 99 Energy Groups Library[1] (Based On JEF2.2 Evaluations) – On A Large Number Of MOX Isotopic Vectors (About One To Two Thousands) And Creation Of A Library For Each MOX Fuel Type (I.E., Standard MOX, MIX, MOX-Ue). Each Library Contains The K_∞ Values At Each Burn-Up Step For Each Isotopic Vector. The Library Is Created Once For All By Means Of A Procedure Called GENBASE (Before The Execution Of Any TIRELIRE-STRATEGIE Calculation) And Constitutes Input Data To ECRIN.
2. Interpolation by ECRIN in the pre-calculated library to obtain the plutonium content associated to the given Pu isotopic vector, corresponding to the requested value for the EOC reactivity. In case of an enriched ^{235}U support, the plutonium content is fixed and the unknown is the ^{235}U enrichment.

The performance of ECRIN is very satisfactory: the error on k_{eff} with respect to APOLLO2 is only a few tens of pcm (1 pcm = 10^{-5} $\Delta k/k$), with a quasi-instantaneous calculation time.

The calculation of the fuel isotopic composition at discharge is carried out by STRAPONTIN [9], the EDF code for burn-up and decay heat power computation. The Bateman equations are solved by means of a Runge-Kutta method with input data (one-group cross sections) provided by interpolation from pre-calculated libraries. These libraries are obtained from APOLLO2 burn-up calculations at 99 or 172 energy groups (as previously described for ECRIN data libraries, cross-sections are computed at different burn-up steps for a large number of fuel isotopic vectors) with consideration of 26 actinides and 154 FPs and their decay chains. The precision of STRAPONTIN is excellent (the relative error on EOC isotopic masses is less than 1% with respect to APOLLO2), and the calculation is quasi-instantaneous.

Thus, PWRs are very accurately modeled in TIRELIRE-STRATEGIE, with very short calculation times, by means of the utilization of ECRIN and STRAPONTIN, both based on APOLLO2 neutron transport calculation results.

[1] The library name is «CEA 93» whereas «99» refers to the number of energy groups

2.1.3 FBR Modeling

Several FBRs are available: besides the Na-cooled European fast reactor (EFR), more innovative concepts have been modeled, including a gas-cooled fast reactor (GFR), a lead-cooled fast reactor (LFR), and an accelerator-driven system (ADS) [10].

Two methods are available for calculating the *equivalence* and the isotopic *evolution* of the MOX fuel:

1. Direct coupling of TIRELIRE-STRATEGIE with ERANOS [11], a deterministic modular code system for fast reactor neutronics, developed by the French Commissariat à l'Energie Atomique (CEA). ERANOS utilizes the transport code ECCO for the lattice calculation, with ERALIB-1 nuclear data (based on JEF2.2 data libraries adjusted on approximately 350 integral experiments), to execute a 33 energy groups diffusion or transport core calculation, with a burn-up and decay actinides chain from ^{231}Pa to ^{248}Cm. For each fuel reload, ERANOS calculates both the Pu content in the fresh MOX fuel to achieve a nearly zero reactivity value at EOC (*equivalence* calculation) and to calculate the isotopic composition of the SNF at discharge (*evolution* calculation). This method is highly time-consuming; therefore, it is mainly employed as a reference calculation, in order to validate the simplified method described below.
2. An analytical approach based on first-order perturbation method around a specified fuel composition, considered as the reference one. This method is also based on ERANOS calculations, which are executed once for all by the execution of the GENMATRICE procedure (with fixed values of both the reload batch size and the fuel burn-up) to calculate, for each FBR concept:
   - A set of equivalent-^{239}Pu weights, allowing to calculate the Pu content [or, generally speaking, if MAs are multi-recycled, the transuranics (TRUs) content, noted t_{TRU}] of the fresh MOX fuel as follows:

     $$t^{TRU} = \frac{t^{equiv,239Pu} - W^{U}}{W^{TRU} - W^{U}} \qquad \textbf{Eq. 2-1}$$

     where W^{U} and W^{TRU} are the total equivalent-^{239}Pu weights of the uranium support and of the transuranics contained in the MOX fuel, respectively, and $t^{equiv,239Pu}$ represents the equivalent-^{239}Pu content for the reference fuel composition. The equivalent-^{239}Pu weights set is calculated (as for the previous method and as for the equivalence calculated by ECRIN in case of PWRs) to ensure that the cycle length is the same, even when the Pu or MAs contents in the fresh fuel evolve over time in the analyzed scenario;

- An evolution perturbation matrix, $\hat{M}$, around the reference fuel vector, allowing to calculate the discharged fuel isotopic composition for the fuel vector "*i*", noted f_i^{EOC}, as a function of the fuel vector at BOC, f_i^{BOC}, and the fuel vector at BOC and at EOC for the reference fuel composition – noted f_{ref}^{BOC} and f_{ref}^{EOC} – by a first-order perturbation expression:

$$f_i^{EOC} = f_{ref}^{EOC} + \hat{M}\cdot\left(f_i^{BOC} - f_{ref}^{BOC}\right) \qquad \textbf{Eq. 2-2}$$

The analytical method, despite the fact that its accuracy is subjected to the validity of the first-order perturbation hypothesis, shows excellent performance, the calculation time being negligible. Its validation against ERANOS shows that the error on EOC k_{eff} value is only a few tens of pcm, whereas the relative error on EOC mass is rarely above 1% for the Pu isotopes. Thus, the accuracy of FBR modeling with the analytical method is of the same order of magnitude as the accuracy of PWR modeling based on the ECRIN and STRAPONTIN codes.

2.1.4 Validation

The validation of TIRELIRE-STRATEGIE was carried out by the following code-to-code comparisons:

- APOLLO2, for PWR-only scenarios;
- ERANOS, for FBR-only scenarios;
- COSI [12], the CEA code for fuel cycle studies, on different scenarios including both PWRs and FBRs.

The agreement has proven to be very satisfactory. However, the presentation of the validation results is beyond the scope of the present document and will not be presented here. The reader is referred to References [13] and [14] for some validation elements of the physical models (*equivalence* and *evolution*) for both fast- and thermal-spectrum systems.

2.2 DANESS

The DANESS ("Dynamic Analysis of Nuclear Energy System Strategies") code [15] is an integrated nuclear process model developed by Argonne National Laboratory (ANL). It is intended for the dynamic analysis of today's and future nuclear energy systems on a fuel batch, reactor, and country, regional, or even worldwide level. The model allows simulating up to 10 different reactor types and up to 10 different fuel types in one simulation. The fuel cycle consists of 21 steps in the fuel cycle chain where several fuel cycle facility technologies can be characterized in the model.

Starting from today's nuclear reactor fleet and fuel cycle situation, DANESS simulates energy-demand-driven nuclear energy system scenarios over time and allows the simulation of changing nuclear reactor fleets and fuel cycle options. The energy demand is hereby given as an energy-demand scenario for electricity production. New reactors are introduced based on the energy demand and the economic and technological ability to build new reactors. The technological development of reactors and fuel cycle facilities is modeled to simulate the delays in technology availability.

Levelized fuel cycle costs are calculated for each nuclear fuel batch for each type of reactor over time and are combined with capital cost models to arrive at bus-bar costs per reactor and, by aggregation, into a cost of energy for the whole nuclear energy system. More detailed cost analyses are performed to give an evolution of expenses for utilities, taking into account taxes, depreciation policies, average cost of capital, and others.

A utility sector and government policy model may be activated to simulate the decision-making process for new generating assets and new fuel cycle options. The government policy model allows simulating different actions that governments may exert through, for instance, tax rates, regulation, R&D-funding and others. Extension to life-cycle analysis data, non-proliferation metrics and ecological impact for the system as a whole and/or sub-elements of the system is foreseen in future versions of DANESS.

The use of DANESS is focused on scenario analysis of different development paths for nuclear energy systems. The evolution viewpoint may be from a governmental, utility, or R&D perspective. Scenario simulation with DANESS is not aimed at predicting the future, but rather at helping with projecting and analyzing, in a consistent way, the longer-term outcomes from selecting alternative nuclear energy development paths.

DANESS is implemented on standard PC/Mac platforms. A typical full-scale DANESS simulation covering a time span of 100 years calculated in time steps of one month takes about 15 to 30 minutes on a modern PC or Mac. DANESS is currently implemented using the *Ithink-Analyst* software of high performance systems [16].

DANESS Version 3.5 was used in this investigation.

2.2.1 General Features

Figure 2-3 provides a schematic breakdown of DANESS.

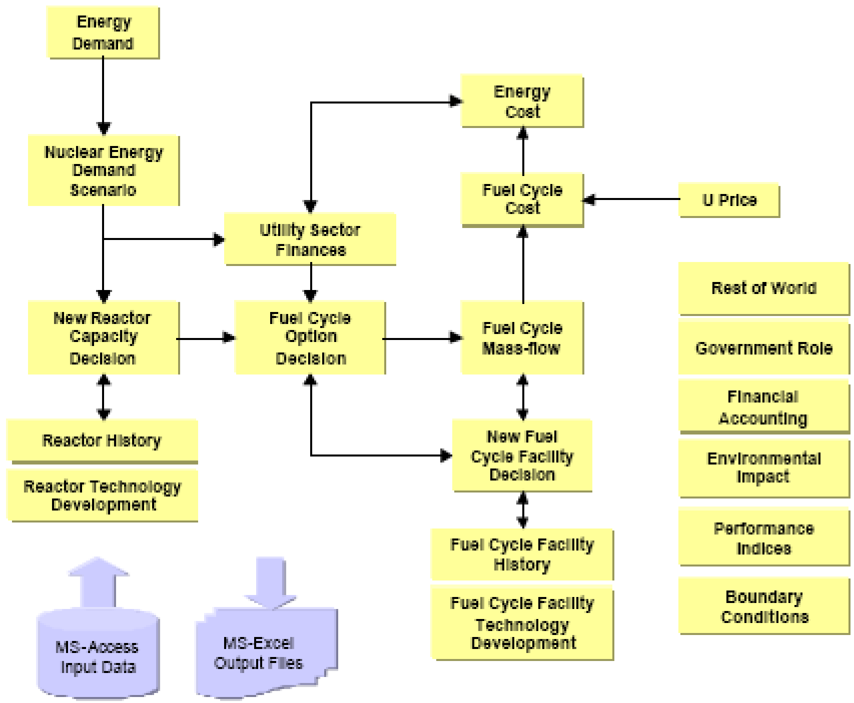

Figure 2-3
Schematic Breakdown of DANESS

DANESS is an energy-demand-driven model based on an exogenously defined energy-demand scenario. This energy-demand scenario may be inputted graphically, using the DANESS user interface as a function or as tabled values. Energy-demand scenarios may cover a country, region or world, but may also be set as a fixed value if the user wants to simulate one reactor or a non-expanding reactor park. The DANESS model will use the energy-demand data as historic data to forecast the energy demand within a certain planning horizon. DANESS will order new reactors to match this energy-demand forecast based on the forecasted operational reactor capacity, the expected energy demand, and the margin for improvement of the average capacity factor of the operating reactors.

A DANESS simulation may start from an existing reactor fleet. The data on the existing reactor fleet (composition, initial fuel cycle stock, etc.) can be fixed by the user in the input sheet. Based on the shutdown schedules of the existing reactors and the forecasted energy demand (fixed by the user either directly in the input sheet or with the DANESS user-interface), DANESS will aim to match this demand by ordering new reactors depending on:

- Expected energy shortage in the planning horizon. The planning horizon is defined by the user or set by the economic decision making sub-model (if this model is activated). If the expected energy shortage can not be met by changing the average capacity factor of the reactors (if allowed), new reactors will be considered for ordering.
- Technological readiness of the reactor type. For common simulations, reactor technology is generally supposed to be immediately available, by putting in the input sheet the technological readiness level at its highest value.

The type and amount of reactors to be ordered is also based on constraints:

- The user may give a preferred reactor fleet composition, or the model will apply the economic decision making sub-model (if activated), which will distribute the reactors ordered as a function of their bus-bar cost.
- The availability of fissile material to fuel new reactors. In case of shortage of fissile material (for instance, plutonium to feed fast reactors), the model will order new fuel cycle facilities (if the user has allowed this option) in case the fuel shortage is due to a limitation of the fuel cycle facilities capacity, or will order a limited number of reactors according to the availability of fissile material. If a reactor type uses two or more fuel types, e.g., a light-water reactor (LWR) partially MOX-loaded, or a fast reactor (FR) with a conversion ratio equal to one, or CR = 1, the model will check the availability of all these fuel types and will possibly limit the ordering of reactor types accordingly.

Once the reactor is ordered, its life cycle will be followed, i.e., licensing, construction, operation, shutdown, and finally decommissioning. Reactors that were ordered, but that are short of fuel at the time they could be ready for start-up, are kept “on-hold” until enough fresh fuel has been fabricated. The same applies for operating reactor capacity that may be set in “stand-by” mode if not enough fresh reload fuel can be fabricated.

The most extensive DANESS sub-model is the fuel cycle mass-flows model, the diagram of which is shown as Figure 2-4.

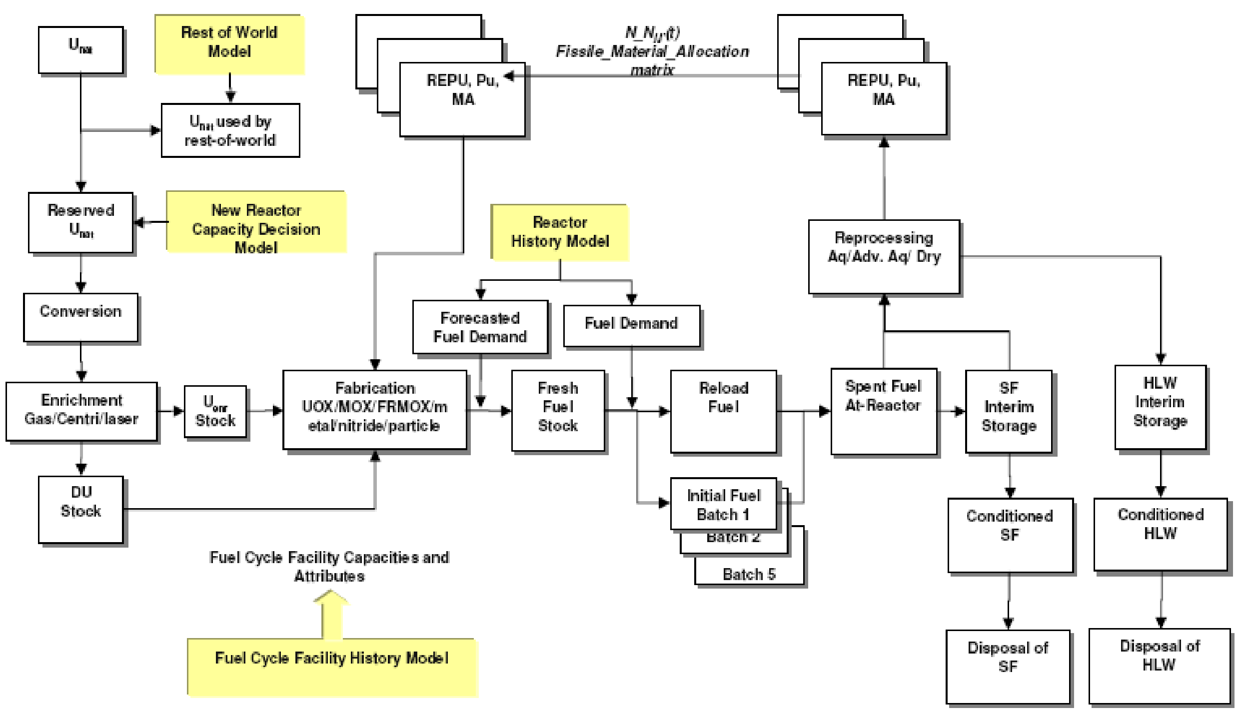

Figure 2-4
DANESS Fuel-Cycle Mass-Flows Model

This sub-model calculates for each fuel type the mass flows and mass balances throughout the 21 fuel cycle steps taken into account, from U-mining to geological disposal. The allocation of fuels to reactors or to fuel cycle facilities is made by using, respectively, a reactor-fuel and a fuel-facility combination matrix.

2.2.2 Reactor and Fuel Modeling

Reactor and fuel modeling is simple. The user has to give the value for the few data required for the calculation, as shown on the two following tables extracted from the DANESS user interface (Figure 2-5).

Reactor Data

Main Screen

U | Reactor 1

Reactorsdata[Reac 1,Pth]	2647
Reactorsdata[Reac 1,Pe]	900
Reactorsdata[Reac 1,Effth]	34
Reactorsdata[Reac 1,fLoad]	90
Reactorsdata[Reac 1,Cycle]	12
Reactorsdata[Reac 1,Batches]	5
Reactorsdata[Reac 1,Fuel Use Flag]	0
Reactorsdata[Reac 1,Total Eq Fuel Mass need per year]	17.403
Reactorsdata[Reac 1,Reactor Group]	1
Reactorsdata[Reac 1,LicensingTime]	2
Reactorsdata[Reac 1,ConstructionTime]	4
Reactorsdata[Reac 1,Lifetime]	50
Reactorsdata[Reac 1,ConstructionCost]	1.44
Reactorsdata[Reac 1,OtherCapitalCost]	0
Reactorsdata[Reac 1,DecomCost]	0
Reactorsdata[Reac 1,Contingencies]	0
Reactorsdata[Reac 1,OMCost]	15
Reactorsdata[Reac 1,InitialTRL]	9

Fuel Data

Main Screen

U | Fuel 1 Data

Fueldata[Fuel 1,BU]	50
Fueldata[Fuel 1,CycleLength]	12
Fueldata[Fuel 1,NumberBatches]	5
Fueldata[Fuel 1,Initial U]	1
Fueldata[Fuel 1,Initial REPU]	0
Fueldata[Fuel 1,Initial DU]	0
Fueldata[Fuel 1,Initial Enr]	4.2
Fueldata[Fuel 1,Initial Pu]	0
Fueldata[Fuel 1,Initial MA]	0
Fueldata[Fuel 1,Initial Np]	0
Fueldata[Fuel 1,Initial Am]	0
Fueldata[Fuel 1,Initial Cm]	0
Fueldata[Fuel 1,Initial SLFP]	0
Fueldata[Fuel 1,Initial LLFP]	0
Fueldata[Fuel 1,Spent U]	0.93545
Fueldata[Fuel 1,Spent Enr]	0.82
Fueldata[Fuel 1,Spent Pu]	0.012
Fueldata[Fuel 1,Spent MA]	0.00125
Fueldata[Fuel 1,Spent Np]	0
Fueldata[Fuel 1,Spent Am]	0
Fueldata[Fuel 1,Spent Cm]	0
Fueldata[Fuel 1,Spent FP]	0.0513
Fueldata[Fuel 1,Spent SLFP]	0
Fueldata[Fuel 1,Spent LLFP]	0
Fueldata[Fuel 1,Spent Activity]	0
Fueldata[Fuel 1,Spent Heat]	0

Figure 2-5
DANESS User Interface: Reactor Data and Fuel Data Menus

For each type of reactor (10 different types at most), the user has to provide the following main data: thermal and electrical power (in MW), conversion efficiency, and average capacity factor. Other data are relative to reactor lifetime, licensing and construction times, or are economic data (construction, capital, O&M, decommissioning costs) used in the economic model for energy cost calculation.

For each type of fuel (10 at most), the data are used fuel burn-up (in GWd/t), fuel management data (duration of cycle and number of batches), fresh fuel (at beginning of cycle) and used fuel (at end of cycle) composition (uranium enrichment, plutonium and minor actinides content, fission products content).

Contrary to TIRELIRE-STRATEGIE, DANESS does not dispose of an equivalence model and does not calculate the fuel composition evolution during the irradiation. Such models are under development but had not been implemented in the code at the time of this study. The lack of these two physical models is one of the weaknesses of DANESS, especially for the calculation of scenarios with actinide recycling, and for the simulation of transient periods like the replacement of a light-water reactor fleet by a fast neutron reactor fleet, which are characterized by a strong variation in fuel composition before reaching a more stable state. Since the fuel isotopic evolution under irradiation is not calculated by the code, this may require the user to provide a different set of isotopic compositions for each type of used and fresh fuel at different times in the scenario.

Also, DANESS is supposed to have the capability to calculate the evolution of the actinides in the fuel out-of-pile, which allows to take into account the decay, for examples, of ^{241}Pu and ^{244}Cm (with half life respectively of ~14 and ~18 years). However, the model did not work satisfactorily in the DANESS Version 3.5 that was tested in the framework of this study.[2]

2.2.3 Validation

According to [15], DANESS has been extensively verified with other calculations of nuclear energy systems, and this verification has indicated error margins inferior to a few percent depending on the quality and detail of data. DANESS is currently used by several laboratories, universities and nuclear R&D companies, such as NRG (The Netherlands) [17], Argonne National Laboratory [15, 18], and the University of Tennessee in the United States [19].

[2] See Section 5.

3
DESCRIPTION OF THE DYNAMIC SCENARIOS OF EVOLUTION OF THE U.S. NUCLEAR FLEET

3.1 Modeling of the Start-Up of the U.S. Fleet from 1967 to 1999

The start-up of the U.S. nuclear fleet was modeled in TIRELIRE- STRATEGIE during the period from 1967 to 1999 with:

- Input data on installed nuclear capacity and net nuclear electricity generation, taken from Reference [20].
- Input data relative to annual spent fuel discharges and burn-up (period 1968 to 1999) from Reference [21].

Nevertheless, in order to have the same input data set as in DANESS, the installed capacity in 1990 was fixed at 100 GWe and kept constant afterwards, and the load factor was adapted in the period 1990 to 1999 in order to match the actual electricity generation of Reference [20]. Actual and simulated deployments are showed in Figure 3-1.

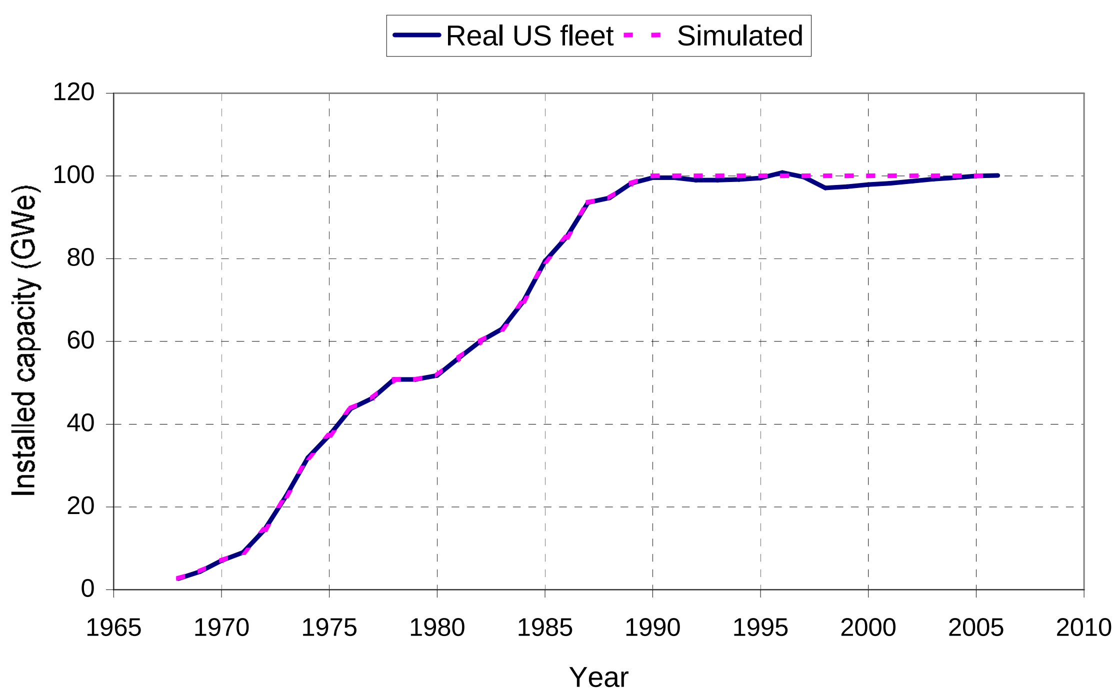

Figure 3-1
Start-Up of the U.S. Nuclear Power Fleet

In the case of the DANESS code, only a simplified approach was applied, with the simulation beginning in 2010 with the same installed capacity of 100 GWe. Consequently, the spent nuclear fuel (SNF) stock value in 2000 was initialized to the corresponding value of Reference [20].

3.2 Spent Nuclear Fuel Inventory in 2000

The total SNF inventory in early 2000 (hence, the cumulated SNF inventory in the period 1968 to 1999, without taking into account the SNF discharged in 2000) is equal to 40,464 MTU (metric tonnes of uranium).

3.3 Hypotheses and Basic Data for the Period After 2000

3.3.1 Nuclear Fleet

Starting with 2000, the installed capacity is equal to 100 GWe and generates 780 TWhe/yr, corresponding to a mean load factor of 89%. These values are kept constant all along the scenarios considered after 2000 (hence, only zero-growth scenarios are considered in this study).

3.3.2 Nuclear Reactors

The U.S. nuclear fleet is composed of PWRs and BWRs. In 2002, the average discharge burn-up was 40 GWd/MTU for BWRs and 45.7 GWd/MTU for PWRs.

In the TIRELIRE-STRATEGIE code, only model data relative to PWR are available (both in terms of sub-assembly and core geometries and isotopic concentrations). The problem associated with lack of models for BWRs was solved by a "mean" PWR model, with a mean discharge burn-up of 43.9 GWd/MTU (weighted on the discharged SNF mass from BWRs and PWRs in order to respect the total mass inventories).

Hence, three types of nuclear reactors were considered in this study:

1. Gen. II PWR with average discharge burn-up of 43.9 GWd/MTU.
2. Gen. III PWR with average discharge burn-up of 60 GWd/MTU.
3. CAPRA TRU burner.

3.3.2.1 Gen. II PWR with average discharge burn-up of 43.9 GWd/MTU

The initial ^{235}U enrichment is 3.7%. The isotopic concentrations corresponding to the average discharge of 43.9 GWd/MTU are issued from a calculation by the EDF depletion code STRAPONTIN [9], with microscopic cross-sections of a typical French-type PWR geometry.

The thermodynamic yield is 32.4% and the load factor, equal to the total fleet average load factor, is 89%. The reactor lifetime is 55 years. This value is also equal to the lifetime of the U.S. fleet (the lifetime has been assumed to be the same for all Gen. II units).

3.3.2.2 Gen. III PWR with average discharge burn-up of 60 GWd/MTU

The initial ^{235}U enrichment is 4.95%. The SNF isotopic concentrations at 60 GWd/MTU (after a SNF cooling time of 5 years), again issued from a STRAPONTIN calculation, are shown in Table 3.1.

Table 3-1
Isotopic Vector of a PWR Fuel Irradiated at 60 GWd/tHM (at EOC + 5 Years Cooling Time)

Isotope	Mass Fraction (%)
Np237	6.34
Pu238	3.48
Pu239	44.50
Pu240	21.52
Pu241	10.79
Pu242	7.24
Am241	3.41
Am243	1.88
Cm244	0.77
Cm245	0.07
Total	**100.000**

The thermodynamic yield is 32.4%, and the load factor, equal to the total fleet average load factor, is 89%. The TRU production at EOL + 5 yrs cooling time is 30.8 kg/TWhe. The core irradiation time is five years. The reactor lifetime is 60 years, but the latter is not factored in this study.

3.3.2.3 CAPRA TRU Burner

The CAPRA (French acronym for "Consommation Accrue du Plutonium dans les réacteurs RApides", Enhanced Plutonium Burning in Fast Reactors) fast burner reactor concept chosen for this study was developed in the framework of the CAPRA-CADRA European collaborative program [2], whose aim was to investigate a broad range of possible options for plutonium and radioactive waste management. CAPRA was thus originally developed as a plutonium burner, based on the EFR (European Fast Reactor) design turned into a burner by removing the fertile blankets and enhancing the plutonium content from ~20% to ~40%.

More recently, a TRU (transuranic elements, i.e., plutonium and minor actinides) burner version of CAPRA was also studied, the minor actinides being homogeneously mixed with plutonium in the CAPRA fresh fuel. This design of TRU burner was modeled for this study thanks to input data received from CEA [3].

The core power is 1450 MWe, the thermodynamic yield is 40.0% and the load factor is equal to the total fleet average load factor, 89%. The MOX fuel residence time is about four years (1,320 EFPD) corresponding to an average discharge burn-up of about 180 GWd/MT. The conversion ratio of the CAPRA TRU-burner is about 0.5.[3]

At equilibrium, the Pu mass content in the CAPRA reactor at BOC is 38.7% and the MA (minor actinides) is 6.3% (see the fuel composition in Table 3-2 issued from ERANOS calculation).

Table 3-2
CAPRA Fuel Composition at Equilibrium (ERANOS Calculations)

	Mass Fraction (%)	
	BOC	**EOC + 5 yrs**
U	55.0	46.7
Np	1.1	0.5
Pu	38.7	30.1
Am	3.5	2.8
Cm	1.7	1.7
FPs	0.0	18.2
Total	**100.0**	**100.0**

3.4 Dynamic Scenarios during the Period 2000 – 2150

3.4.1 Scenario A: PWR Once-Through

This scenario is characterized by the complete absence of SNF reprocessing. The Gen. II U.S. fleet is renewed by a new fleet of PWR 60 GWd/MTU[4], with the same installed capacity and electricity generation, still operated in the once-through strategy. It clearly follows that the SNF inventory steadily increases all along the scenario.

The lifetime being the same for all the Gen. II reactors, 55 years, it follows that the renewal of the Gen. II fleet by a Gen. III fleet is characterized by the same kinetics, shown in Figure 3-1, during the period from 2022 to 2044.

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[3] The Conversion Ratio is defined as the complement to 1 of the ratio of the production of fissile isotopes (N_{fiss}) to the variation of Heavy Nuclides between BOC and EOC (which is equal to the number of Fission Products):

$$CR = 1 - \frac{N_{fiss}(EOC) - N_{fiss}(BOC)}{N_{HN}(EOC) - N_{HN}(BOC)}$$

Referring to the values shown in Table 3-2, the CR for the CAPRA reactor calculated by ERANOS is equal to:

1 – [(1.1 + 38.7 + 3.5 + 1.7) – (0.5 + 30.1 + 2.8 + 1.7)]/18.2 = 1 – 9.9/18.2 = 0.46

[4] In the remainder of the document, "Gen. II PWR" designates the PWRs with an average discharge burn-up of 43.9 GWd/MTU, and "Gen. III PWR" designates the PWRs with an average discharge burn-up of 60.0 GWd/MTU.

This once-through scenario is analyzed with the purpose of providing a comparison with the burner scenario presented hereafter.

3.4.2 Scenario B: Deployment of CAPRA Burners + Gen. III PWRs

In this scenario, the Gen. II fleet is renewed by a mixed fleet of CAPRA TRU burners and Gen. III PWRs, the renewal occurring over the same time period as in Scenario A.

The chosen fraction of CAPRA units deployed has been chosen to be the one corresponding to the equilibrium conditions in a mixed CAPRA + Gen. III PWR fleet, with 35% of CAPRA burners. This allows carrying out a supplementary validation by checking if the equilibrium conditions obtained by the ERANOS calculations of Appendix A are effectively realized in the long-term behavior of dynamic scenarios calculated with TIRELIRE- STRATEGIE and DANESS.

For simplicity, the first 65 GWe of the Gen. II fleet are replaced by Gen. III PWRs during the period from 2022 to 2038, and the remaining 35 GWe are replaced by CAPRA reactors during the period from 2038 to 2044 (in 2038, both reactor types are started up, the last 2.3 GWe of Gen. III PWRs and the first 4.4 GWe of CAPRA burners).

3.5 Fuel-Cycle Facilities

3.5.1 Reprocessing

In Scenario A, no reprocessing facilities are modeled.

In Scenario B, the modeled reprocessing process, common to both PWR UOX and CAPRA MOX spent fuel, is the global actinides extraction process (GANEX); however, the yearly capacities are specific to each SNF type:

1. Gen. II SNF Is Partially Reprocessed, In Order To Provide The TRU Inventory Necessary For The Deployment Of CAPRA Burners. Ten Years Of Reprocessing At A Rate Of 5,200 MT/Yr Between 2034 And 2043 Are Sufficient[5].
2. Reprocessing of Gen. III SNF begins in 2030. The total available amount of SNF is reprocessed every year, in order to provide TRU to feed the CAPRA burners. At equilibrium, the SNF reprocessed amounts to 1,192 MT/yr (for 65 GWe installed).
3. Reprocessing of CAPRA SNF begins in 2047. The total available amount of SNF is reprocessed every year. At equilibrium, this amounts to about 155 MT/yr (for 35 GWe installed).

The loss rate at SNF reprocessing is assumed to be 0.1% for all actinides and for all types of fuel in the scenario. The SNF cooling time before reprocessing is five years for all fuel types.

---

[5] This rate is very high and somewhat unrealistic; however, the simulation result would be identical if a reduced rate was implemented over a correspondingly longer reprocessing period. It is to be noted that the cooling time was averaged over the inventory of spent Gen. II fuel; the isotopic composition of reprocessed SNF at date “X” is an average of all SNF discharged from 1966 to “X-5”, given that SNF has to cool for a minimum of 5 years before reprocessing.

3.5.2 Fabrication

The fabrication facilities are driven by the needs of installed reactors in both scenarios. At equilibrium, the values of fresh fuel fabricated every year are clearly the same as the values of SNF reprocessed every year (except for the reprocessing losses that are small).

The aging time before fuel irradiation, equal for all fuel types, is two years.

4
RESULTS WITH TIRELIRE-STRATEGIE

In this section, only results obtained with TIRELIRE-STRATEGIE are presented and discussed. The analysis focuses on the comparison between Scenarios A and B, and on the sensitivity of the Scenario B results to the use of different physical models for simulating the CAPRA burner.

4.1 Scenario A

The renewal of the Gen. II fleet occurs during the period from 2023 to 2044 (see Figure 4-1). The increased burn-up after 2023 (60 GWd/MTU with Gen. III PWR instead of 43.9 GWd/MTU with Gen. II PWR) allows a small reduction in the SNF production rate, but the SNF amount is still steadily increasing as no reprocessing is implemented.

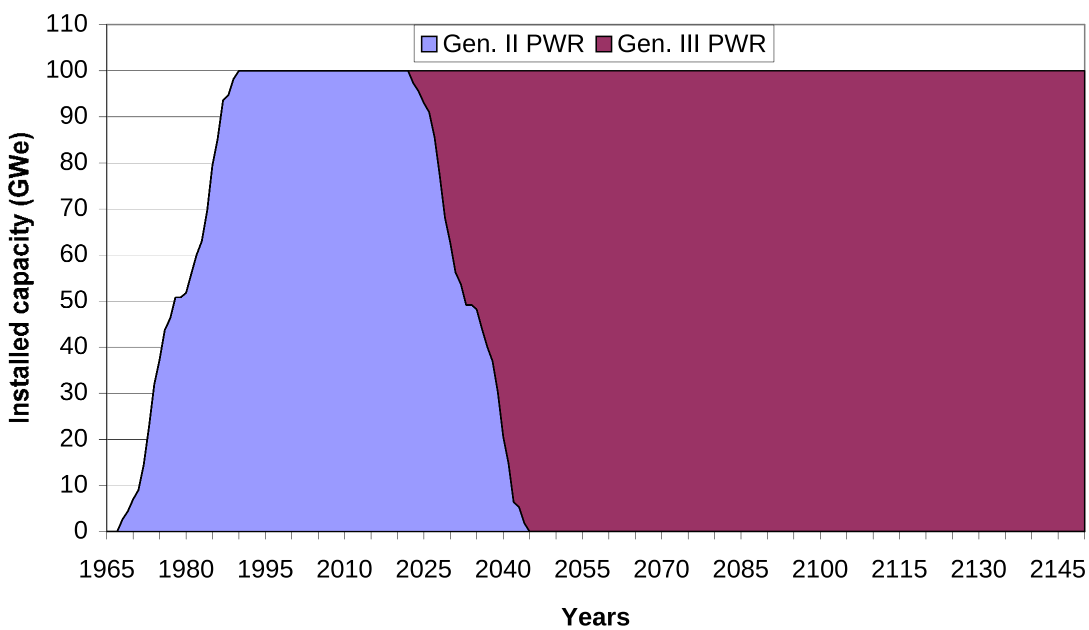

Figure 4-1
Evolution of the Installed Capacity in Scenario A

The total SNF produced by the Gen. II fleet is 126,712 MTU (this value is constant after 2044), as shown in Figure 4-2. If the production by the Gen. III fleet is taken into account, the total SNF inventory to be disposed in a geological repository by 2100 is 239,295 MTU; the latter inventory contains 3,030 MT of TRU consisting of 2,439 MT of Pu and 591 MT of minor actinides.

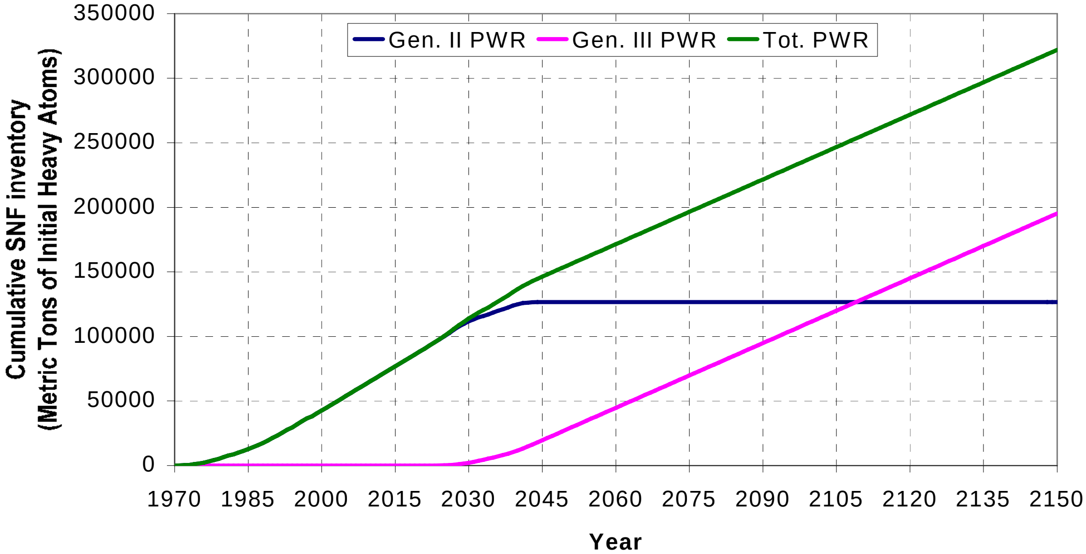

Figure 4-2
Cumulative SNF Inventory in Scenario A

The total TRU inventories over the fuel cycle (including nuclear reactors and fabrication facilities in addition to stored SNF) are shown in Figure 4-3. In 2100, the total TRU inventory is about 3,130 MT consisting of 2,533 MT of Pu and 597 MT of MAs.

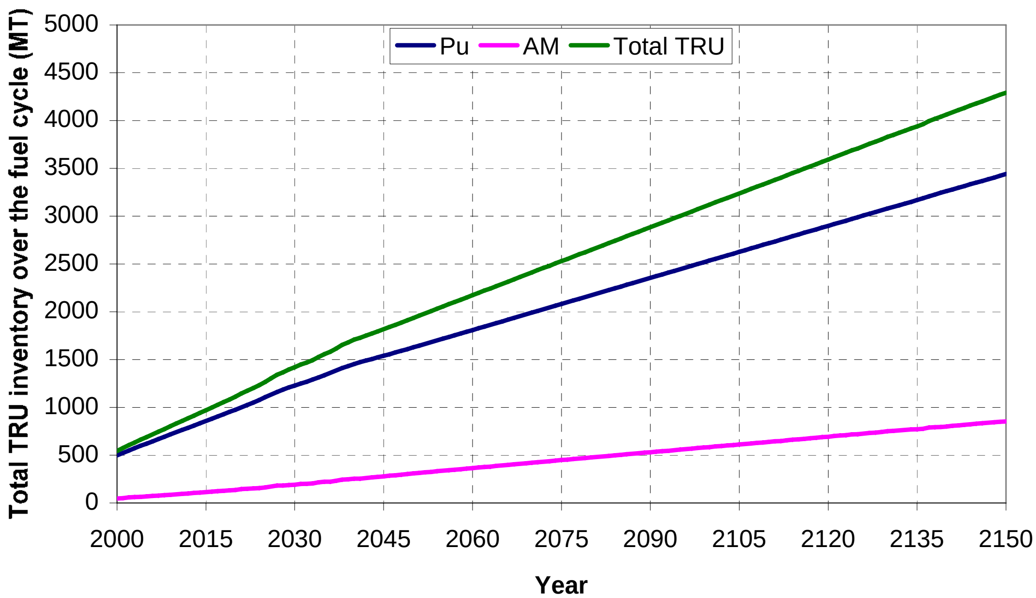

Figure 4-3
Total TRU Inventories over the Fuel Cycle in Scenario A

4.2 Scenario B

In this scenario, the deployed capacity of the CAPRA burners is 35 GWe during the period from 2038 to 2044 (see Figure 4-4).

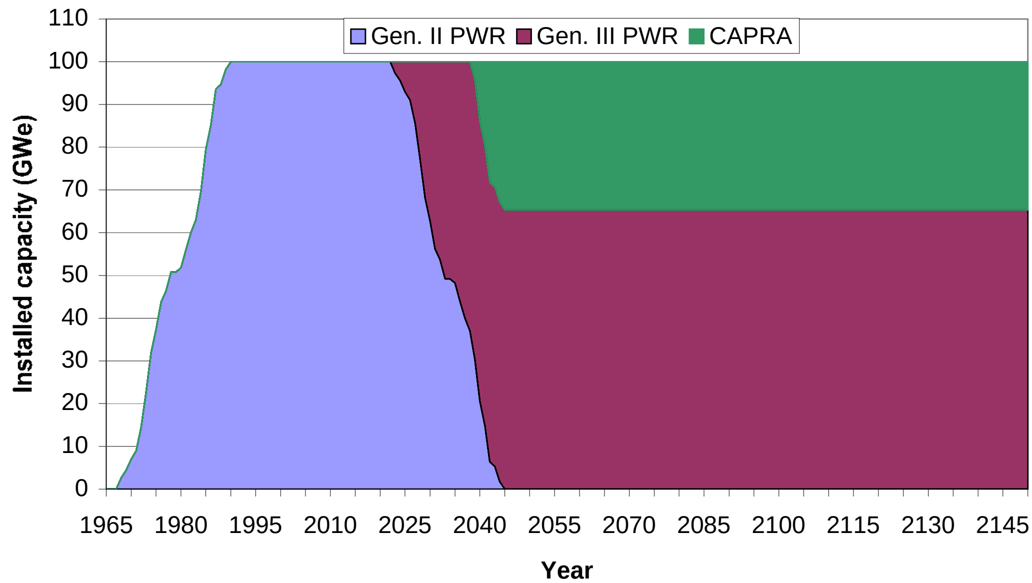

Figure 4-4
Evolution of the Installed Capacity for Each Reactor Type in Scenario B

It should be noted that this particular scenario was chosen in order to carry out a cross-comparison with the ERANOS equilibrium calculations. Hence, the following results will be highlighted:

1. Comparison Of The TIRELIRE-STRATEGIE Long-Term Dynamic Scenario With The ERANOS Equilibrium Calculations, I.E., Whether Or Not 35 Gwe Of CAPRA Burners Correspond To The Equilibrium Fraction Of TRU Burners Allowing To Burn The TRU Production From The 65 Gwe Of Gen. III PWR Installed.
2. Performance of the CAPRA reactors in terms of TRU burning, which consists in comparing the results from Scenario B to those of Scenario A in terms of TRU inventory stabilization resulting from the adoption of reprocessing following by burning in the CAPRA reactors.
3. Sensitivity of the TIRELIRE-STRATEGIE results to the physical models of the CAPRA burner.
4. Finally, a few comments will be devoted to the development of a very simplified model for the simulation of CAPRA burners, very similar to the one required for DANESS, in order to carry out a comparison between the two codes.

4.2.1 Comparison between TIRELIRE-STRATEGIE's Long-Term Predictions and ERANOS' Equilibrium Calculations

The fraction of CAPRA burners allowing to equilibrate the TRU production from the remaining Gen. III PWR stratum is about 35.1% (rounded at 35% in the rest of the document) in the TIRELIRE-STRATEGIE simulation, slightly less than the value of 35.8% obtained by the ERANOS equilibrium calculations (Appendix A). This slight difference can be explained by some differences in the hypotheses (actinide isotopic chain, modeling of the global actinide extraction at SNF reprocessing, etc.) between the two calculations on one hand, and the description of a real transition scenario from Gen. II to a mixed CAPRA + Gen. III fleet as carried out with TIRELIRE-STRATEGIE on the other hand.

The TRU contents are shown in Table 4-1. Although slight differences are apparent, the coherence between the two codes is quite good and the agreement is judged to be very satisfactory.

Table 4-1
Equilibrium Mass Balances in the CAPRA Burner Calculated with ERANOS and TIRELIRE-STRATEGIE (shown as "TIRESTRAT)

	Mass Fraction (%)			
	BOC		EOC	
	ERANOS	TIRSTRAT	ERANOS	TIRSTRAT
U	55.1	54.9	46.6	46.5
Np	1.1	1.1	0.5	0.4
Pu	38.7	38.7	30.4	30.7
Am	3.5	3.7	2.3	2.4
Cm	1.7	1.6	2.1	1.9
FPs	0.0	0.0	18.1	18.1
Total	**100.0**	**100.0**	**100.0**	**100.0**

4.2.2 Performance of the CAPRA Reactors in Terms of TRU Burning: Comparison between Scenarios A and B

The deployment of the CAPRA burners allows a quasi-immediate stabilization of TRU inventory over the fuel cycle: Pu stabilizes at around 1,400 MT, and MAs at about 260 MT just a few years after the deployment of the CAPRA burners. The total TRU inventory is reduced by about a factor 2 in 2100 and not far from a factor 3 by 2150 (Figure 4-5).

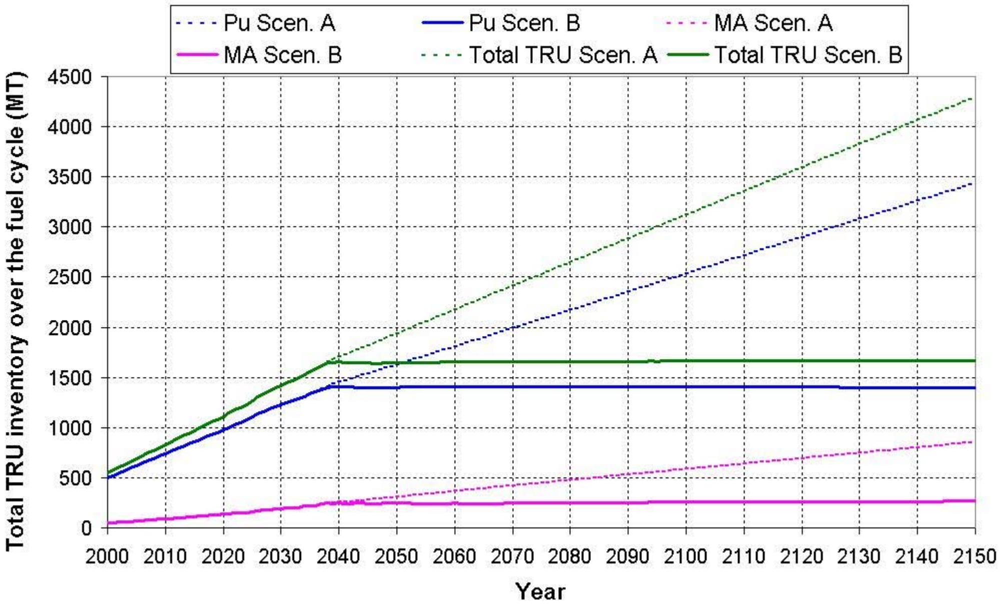

Figure 4-5
Total TRU Inventories over the Fuel Cycle: Comparison between Scenarios A and B

The first fuel loading for 35 GWe of CAPRA burners requires about 230 MT of TRUs. This inventory, plus the inventories required for fueling the CAPRA burners during the transient period before the equilibrium situation in which the external feed comes only from the spent Gen. III PWRs, comes from reprocessing 52,000 MT of Gen. II PWR SNF. Hence, as long as the Gen. II SNF inventory is concerned, 74,712 MT (that is, 126,712 minus 52,000 MT) of un-reprocessed SNF are present in Scenario B after 2043.

To absorb any extra-amount of Gen. II SNF, the deployed fraction of CAPRA burners would have to be higher than the equilibrium fraction of 35%, at least for a limited period of time.

4.2.3 Sensitivity of the TIRELIRE-STRATEGIE Results to the Physical Models of the CAPRA Burners

As explained in Subsection 2.1.3, two cases are possible concerning the physical models used to calculate the *equivalence* and the *evolution* for FBRs in TIRELIRE-STRATEGIE:

5. Equivalence calculation by the equivalent-^{239}Pu weights and evolution by direct coupling, at every year of the scenario, of TIRELIRE-STRATEGIE with ERANOS. This approach provides the best-estimate results for TIRELIRE-STRATEGIE, and it is the reference calculation for this study. The calculation time for this approach is between 10 and 20 hours;
6. Equivalence calculation by the equivalent-^{239}Pu weights and evolution by the perturbation method. This approach is the standard model adopted for FBRs, the calculation time being negligible (a few tens of seconds).

A comparison of the results obtained with these two calculation routes is performed, focusing on the following parameters:

- Pu and MA content at the time of fresh fuel fabrication for the CAPRA reactors;
- Evolution of the Pu inventory available for fresh fuel fabrication.

A new, very basic description of CAPRA reactors, very similar to the one allowing modeling FBRs in the DANESS code, was developed for the purpose of the present study, in order to allow a comparison between TIRELIRE-STRATEGIE and DANESS with comparable models for FBR. This development will be further described in Subsection 4.2.4.

The direct coupling of TIRELIRE-STRATEGIE with ERANOS allows to access, year after year, the reactivity values in the CAPRA reactors calculated by ERANOS. The reactivity values at beginning-of-cycle (BOC) and at end-of-cycle (EOC) are shown in Figure 4-6. It can be clearly seen that, even if the BOC varies (because the fuel isotopic composition of the CAPRA reactors changes along the scenario, so does the conversion ratio and so does the BOC reactivity), the EOC reactivity is rather constant all along the scenario, close to the requested value of 450 ± 50 pcm (this positive margin at EOC is due to the fact that the control rods are not modeled in this simplified ERANOS evolution scheme).

Hence, this confirms that, for any given year in the scenario, the Pu and MA mass contents in the fresh fuel are correctly calculated by the TIRELIRE-STRATEGIE model to ensure that the campaign length of the CAPRA reactors (the EOC reactivity) remains the same all along the scenario. Thus, the direct coupling of TIRELIRE –STRATEGIE with ERANOS allows an indirect validation of the semi-analytical model for the equivalence calculation developed for the CAPRA burners in the framework of the present study.

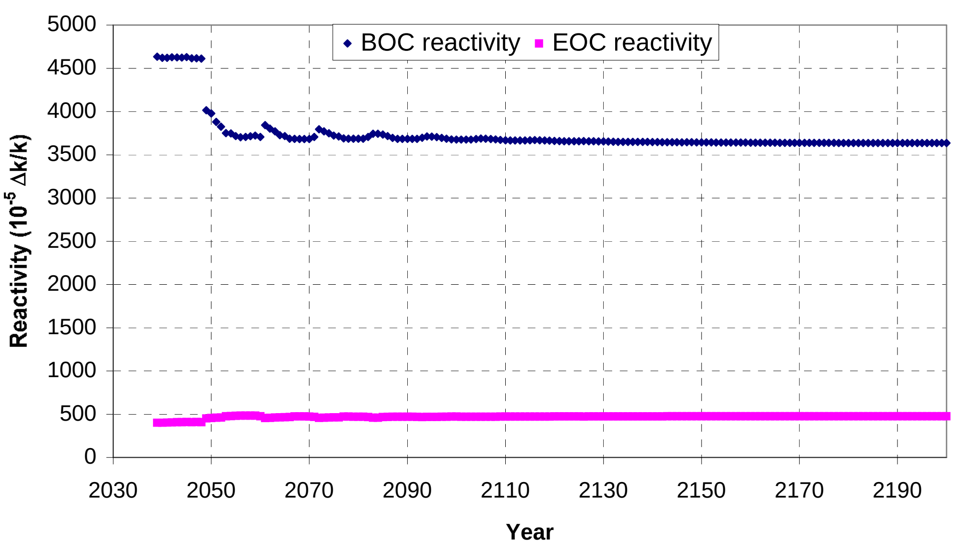

Figure 4-6
Evolution of BOC and EOC Reactivities over the Cycle Length in the CAPRA Burners

The comparison between reference (evolution by direct coupling with ERANOS) and perturbation method calculations for the evolution of the Pu inventory available for new fresh MOX fuel fabrication is shown in Figure 4-7. In this case, the scenarios have been analyzed to a very long term (until 2400) in order to check very precisely the establishment of the equilibrium conditions in the mixed 65-GWe PWR + 35-GWe CAPRA fleet.

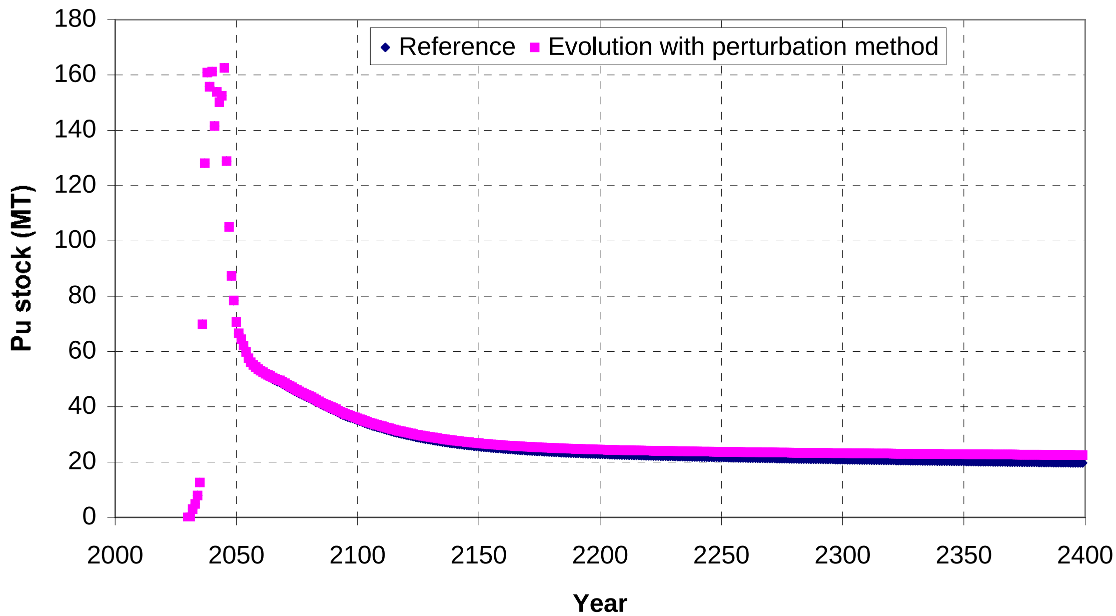

Figure 4-7
Evolution of Pu Inventory Available for Fresh MOX Fuel Fabrication as a Function of the Physical Models for CAPRA Burner Evolution Calculation

The analysis of Figure 4-7 clearly shows the followings:

1. The Two Curves Are Nearly Superposed All Along The Scenario. The Error Of The Simplified Method With Respect To The Direct Coupling With ERANOS Is An Over-Estimation Of Less Than 3 MT Of Pu In 2400. This Very Small Error Shows, Once Again, The Very Good Performance Of The Simplified Model Based On The Perturbation Method For The Evolution Calculation. This Very Positive Conclusion Is Even More Clearly Evident If We Consider That The Calculation Times Are, Respectively, ~30 Hours For The Reference Calculation And ~30 S For The Simplified Method!
2. Both curves have reached their asymptotic value. This confirms that 35% is the equilibrium fraction of CAPRA burners allowing burning the TRU production from the remaining 65% PWR.

The excellent coherence between the reference calculation and the one computing the evolution by means of the perturbation method is confirmed by the TRU mass content at fabrication given in Figure 4-8. At equilibrium, the Pu content is 38.79% in the reference calculation (38.75% with the simplified evolution model) and the MA content is 6.38% (6.42% with the simplified evolution model).

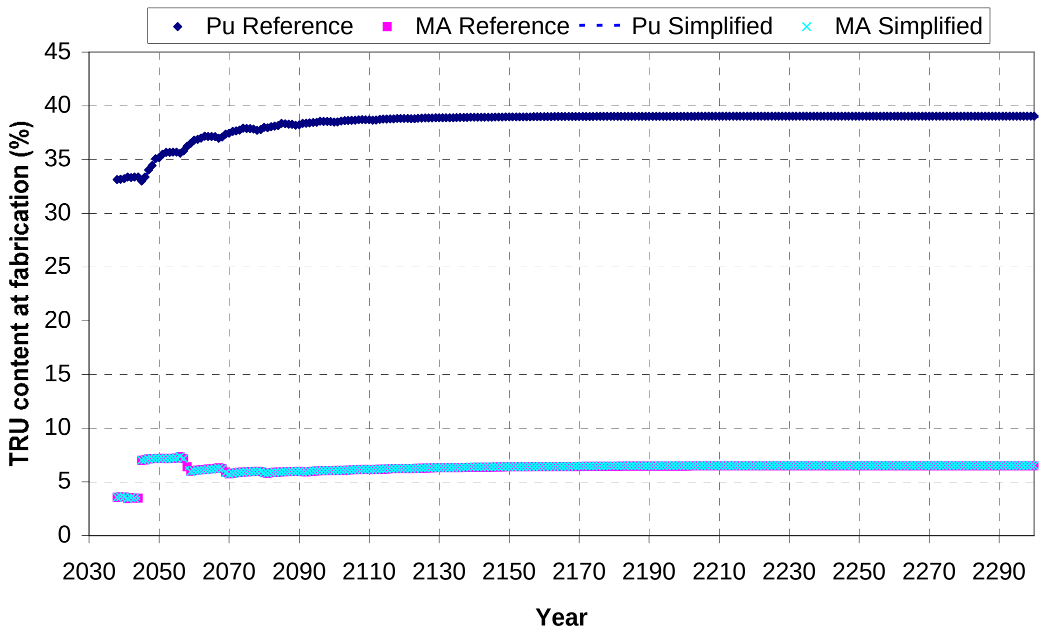

Figure 4-8
TRU Mass Content at CAPRA Fresh Fuel Fabrication as a Function of the Physical Model for the CAPRA Evolution Calculation

4.2.4 Development of a Simplified Model Similar to the One Required for DANESS

In the DANESS code, no physical models are implemented to treat the equivalence or the in-pile (under neutron flux) evolution processes. Each reactor type is described by an input file giving the fuel compositions at BOC and at EOC, which are constant all along the scenario, and generally chosen as the fuel composition at equilibrium state. Of course, this very basic description does not allow taking into account the changes in isotopic composition of TRUs all along the scenario, which result in a different TRU mass content in order to satisfy the given cycle length for FBRs, in this case the CAPRA burners. It follows that there is no control at all that the TRU contents are compatible with operation of the different reactors inside a pre-defined *physical* domain (for example, the simulated reactor could be even sub-critical or largely over-critical). It follows that a comparison between the results obtained by TIRELIRE-STRATEGIE and DANESS would be strongly affected by the absence of such physical models in DANESS.

Hence, in order to isolate this source of discrepancy between the codes, and with the objective of carrying out a calculation with TIRELIRE-STRATEGIE as much as possible compatible with DANESS, a basic FBR treatment was implemented in TIRELIRE-STRATEGIE.

In this section, after a quick description of the developed model, a comparison of this simplified TIRELIRE-STRATEGIE calculation with the two calculations paths shown in the previous section is presented.

The mass balances used for the simplified calculation are taken from the long-term TIRELIRE-STRATEGIE situation, which is quite similar to the ERANOS equilibrium state results. The Pu content in the fresh fuel of CAPRA burners is 38.8% and the MA content is 6.4% (1.1% for Np, 3.7% for Am and 1.6% for Cm, with reference to Table 4-1). These values, constant all along the scenario since the first fuel loading of the CAPRA reactors in 2038, are very different from those calculated by the equivalence model of TIRELIRE-STRATEGIE in the deployment phase of the CAPRA burners. For example, in 2038 the Pu content is only 33.1% and the MA content is 3.6%.

In terms of results, the higher value of the TRU mass content of the basic model (taken from the equilibrium situation in which effectively the TRU mass content is much higher than at the beginning of the scenario) compared to the TIRELIRE-STRATEGIE calculation with the equivalence/evolution model leads to the following observations:

- The same amount of TRU allowing to deploy 35 GWe of CAPRA in the TIRELIRE-STRATEGIE calculations using fine physical models only allows deploying about 31 GWe of CAPRA using the basic DANESS-like models for the CAPRA burners, thus resulting in an under-estimation of 11%.

As the 31-GWe capacity is lower than the equilibrium fraction, it follows that this results in an accumulation of TRUs in the Gen. III PWR SNF with respect to the reference TIRELIRE-STRATEGIE calculation, when the calculation is run with no new CAPRA burners being started-up after the complete renewal of the Gen. II fleet in 2044. This is shown in Figure 4-9.

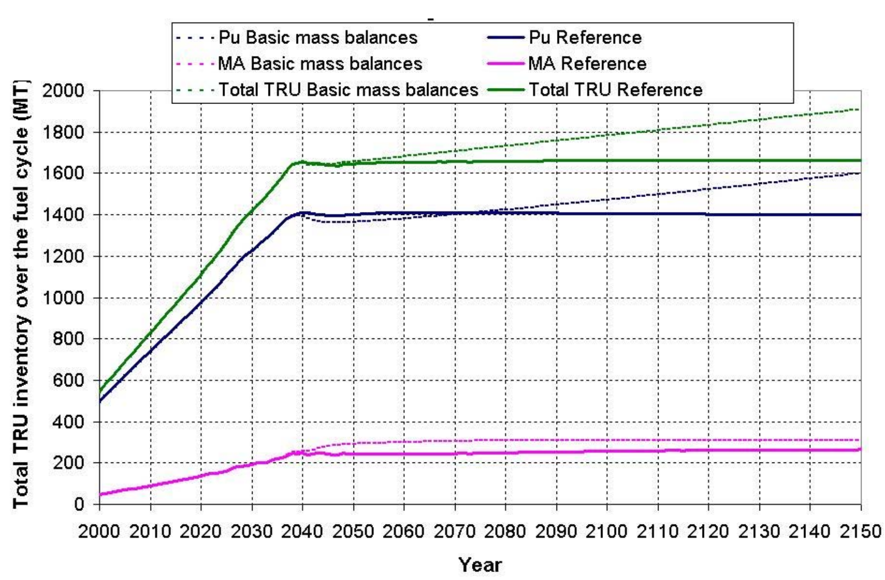

Figure 4-9
Total TRU Inventory over the Fuel Cycle with Different Modeling of CAPRA Burners in TIRELIRE-STRATEGIE: the Reference Modeling (Indicated as "Reference") and the DANESS-Like Basic Modeling ("Basic Mass Balances")

Figure 4-9 shows that:

1. The MA inventory is stabilized with the basic DANESS-like model simulation, but at a level which is 20% higher than the one obtained with the reference TIRELIRE-STRATEGIE calculation (about 310 MT instead of 260 MT).
2. The Pu inventory is not stabilized and increases all along the scenario; in 2150, the total value is 15% higher than in the reference TIRELIRE-STRATEGIE calculation, 1600 MT instead of 1400 MT.
3. As the Gen. III PWR fraction is higher than the equilibrium level, it follows that some SNF is also accumulating at a rate of ~6% of the annual SNF discharged from the Gen. III PWRs [corresponding to the excess of four PWRs (69 instead of 65) divided by 65].

Conversely, in order to deploy, in the DANESS-like TIRELIRE-STRATEGIE calculation, the same capacity of CAPRA reactors as in the reference case, i.e., 35 GWe, an additional amount of Gen. II SNF should be reprocessed, equal to 11% of the 52,000 MT of Gen. II SNF reprocessed in the reference case (see Subsection 4.2.2). It follows that, in this case, the remaining Gen. II SNF at the end of the simulation would be 11% less than in the reference TIRELIRE-STRATEGIE calculation.

Therefore, the development in TIRELIRE-STRATEGIE of the DANESS-like model for the modeling of the CAPRA burners allows:

1. Carrying out a simplified calculation with TIRELIRE-STRATEGIE providing results that are very close to those that would have been obtained with DANESS, the model being the same.
2. Carrying out a comparison between this simplified DANESS-like TIRELIRE-STRATEGIE calculation with the reference TIRELIRE-STRATEGIE calculation.

This comparison shows that the adoption of a simplified DANESS-like model for the CAPRA burners (with constant values for the Pu and MA mass content at fresh fuel fabrication taken from the equilibrium mass balances) underestimates by 11% the deployable capacity of CAPRA burners (31 GWe instead of 35 GWe) and overestimates the MA and Pu inventories over the fuel cycle (+20% for MAs, +15% for Pu in 2150, with this value increasing with time beyond 2150). Conversely, in order to deploy the same capacity of CAPRA reactors as in the reference case, the error would be about 11% on the residual Gen. II PWR SNF inventory at the end of the scenario.

5

RESULTS WITH DANESS AND COMPARISON WITH TIRELIRE-STRATEGIE

Only Scenario A was successfully studied with the DANESS code.

The comparison between the TIRELIRE-STRATEGIE results (already presented in Subsection 4.1) and those obtained with DANESS will focus on the out-of-pile TRU inventory (see Figure 5-1), which is, in this case, strictly equal to the TRU inventory contained in the SNF.

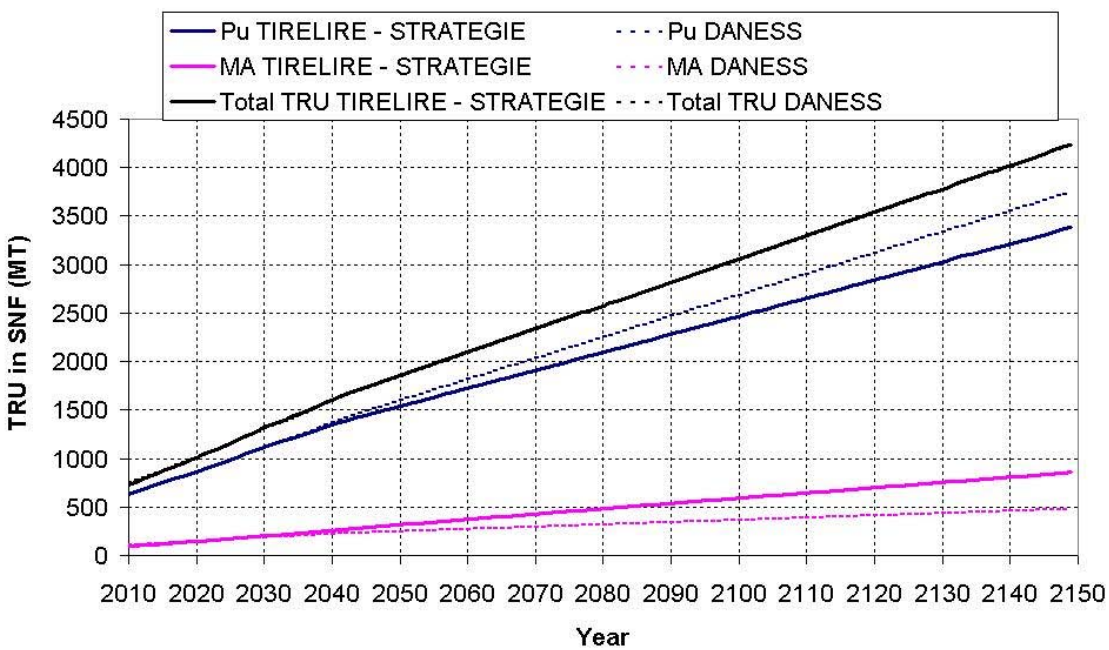

Figure 5-1
TRU Inventory Contained in the SNF Out-of-Pile

The two following conclusions can be drawn from Figure 5-1:

1. The total TRU inventory calculated by the two codes is nearly the same.
2. Nevertheless, as time passes, differences appear between the Pu and MA contents, and these differences are increasing with time. DANESS over-estimates the Pu inventory and, at the same time, under-estimates the MA inventory. This is due to the fact that DANESS' out-of-pile evolution model does not work properly. In particular, the β^- decay of ^{241}Pu into ^{241}Am (whose half life is about 14.4 years) is not properly accounted for during used fuel cooling. This explains why DANESS over-estimates the Pu inventory and under-estimates the MA inventory compared to TIRELIRE-STRATEGIE.

Hence, in spite of a very good agreement on total TRU, a more detailed comparison of Pu and MA inventories would have little interest. DANESS' simulation of Scenario B, for which the decay of ^{241}Pu into ^{241}Am cannot be neglected in order to correctly estimate the deployable capacity of CAPRA burners and the evolution of Pu and MA mass flows and inventories, gave non-physical results when compared to TIRELIRE-STRATEGIE's simulation. The failure of DANESS out-of-pile evolution model has been reported to the DANESS development team.

6
CONCLUSION

Performing an accurate simulation of a GNEP-type scenario, with the deployment of a significant amount of advanced burner reactors, is not a trivial task due to the burner fresh fuel composition's strong variation during the transient period before sufficiently nearing equilibrium conditions.

On the basis of CEA studies of the CAPRA fast burner design, the TRU burner model implemented in TIRELIRE-STRATEGIE turned out to be very accurate when compared to reference calculations performed with the ERANOS code, thanks to its physical models allowing to calculate accurately fresh fuel content and used fuel compositions [respectively, through its equivalence and "in-pile" (under irradiation) evolution models].

A simpler calculation with no equivalence and no in-pile fuel evolution models, using only a data file giving fixed compositions for fresh and used fuel, can only result in a degradation of the accuracy of the simulation. The loss of accuracy was estimated by introducing the same type of approximation in TIRELIRE-STRATEGIE. Although significant, it can, however, be considered as acceptable, the error being of the order of 10-20% for the number of burners to be deployed or for the size of TRU inventory. Nevertheless, DANESS would be significantly improved by implementing both equivalence and "in-pile" evolution models in the code.

A satisfactory simulation of the "burner scenario" by DANESS was not possible due to the failure of DANESS' out-of-pile fuel evolution model; the latter is necessary to take into account, among many other radioactive decays, the rapid ^{241}Pu decay resulting in ^{241}Am build-up during used fuel cooling and fresh fuel aging.

Eventually, it is worth pointing out the difficulty in properly using such tools as TIRELIRE-STRATEGIE and DANESS. The potential for errors without being able to notice them is very high. TIRELIRE-STRATEGIE benefits from a powerful technical environment, since its use can be validated thanks to the coupling with ERANOS.

Concerning DANESS, on the basis of this study, it is recommended to:

- Improve the code by implementing more physical models (equivalence and in-pile fuel evolution models); by correcting the out-of-pile decay model; and by improving the user interface.
- Provide an enlarged DANESS validation, which turned out to be insufficient.

Enhance the robustness and reliability of DANESS. Calculation shortcomings were noticed only thanks to more detailed, comparative analyses with TIRELIRE-STRATEGIE, and complete mastering of the tool can be expected, at least at this time, only by the experts who developed the code.

7
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A
EQUILIBRIUM ANALYSIS: PERFORMANCE OF THE CAPRA DESIGN IN THE FRENCH FLEET UNDER EQUILIBRIUM CONDITIONS

The CAPRA design [3] was modeled with the ERANOS CEA code (Subsection 2.1.3) under equilibrium conditions in a fleet composed of PWRs and CAPRA reactors. PWRs are fed with UOX fuel, and the Spent Nuclear Fuel (SNF) from PWRs is reprocessed, TRUs are extracted and used to fabricate the fresh fuel for the CAPRA reactors (the support is depleted uranium). At equilibrium, the TRUs are multi-recycled in the CAPRA reactors.

A.1 Hypotheses

For the hypotheses and basis data relative to the CAPRA and PWR reactors, see Subsection 3.3.2. The TRU production from PWRs at an average discharge burn-up of 60 GWd/MT and with a thermodynamic yield of 34.2% is 30.8 kg/TWhe (respectively 87.5% plutonium and 12.5% minor actinides).

A.2 Results

A.2.1 Fleet Composition

At equilibrium conditions, the mass contents in the CAPRA fresh fuel are 38.7% for Pu and 6.3% for MA. The TRU consumption in the CAPRA burners is 55.3 kg/TWhe. It follows that the equilibrium composition of a mixed PWR + CAPRA fleet (in which the TRU mass flow coming from PWRs equalizes the input mass flows into the CAPRA reactors) is 35.8% of the electricity production by CAPRA reactors and 64.2% by PWRs.

A.2.2 TRU Waste and Fuel Cycle Inventory

The TRU amounts sent per year to the HLW repository (assuming 0.1% losses at SNF reprocessing) and the fuel cycle inventories (assuming two years for fresh fuel ageing and five years for SNF cooling before reprocessing) are shown in Table A-1. Results are normalized to a total production of 8.76 TWhe/yr (hence, 3.14 TWhe/yr by CAPRA and 5.62 by PWR), i.e., the energy produced by one GWe during one year. For more details on the hypotheses behind this normalization value, see Reference [4].

Table A-1
TRU Content Going to HLW Repository Per Year and Fuel-Cycle Inventory in the Mixed 64.2% PWR + 35.8% CAPRA Fleet at Equilibrium Conditions, Normalized to 1 GWe

TRU content going to HLW repository assuming 0.1% loss in separation process (kg/year)	
Pu	0.682
Np	0.020
Am	0.059
Cm	0.031
Total TRU	0.792
Cycle inventory (reactors + fabrication + reprocessing) (kg)	
Pu	9094
Np	256
Am	803
Cm	413

A.3 Relation to Equilibrium Conditions Results Reported in EPRI 1015129[6-7]

Table 5-4 of EPRI Report 1015129 can be augmented with the results given in this Appendix, as shown in Table A-2.

[6] Program on Technology Innovation: Advanced Fuel Cycles – Impact on High-level Waste Disposal: 2007 Progress Report. EPRI, Palo Alto, CA: 2007. 1015129.

[7] This section was added by the EPRI project manager, A. Machiels, to supplement information contained in the referred-to EPRI report.

Table A-2
Fuel-Cycle TRU Inventories and Annual TRU Inventories Going to Direct Waste Disposal – Amounts Normalized to 8.76 TWhe

Fuel Cycle →	Once-through in PWRs	Mono-recycling of Pu in PWRs	Multi-recycling of Pu in PWRs	TRU Burning in Fast breeder Reactors	TRU Burning in Fast burner Reactors
TRU Content Going to HLW Repository Assuming 0.1% Loss in Separation Processes [kg/year]					
Pu	230	153	0.37	1.25	0.682
Np	16.2	16.6	14.45	0.0066	0.020
Am	***6.35***	16.2	39.4	0.055	***0.059***
Cm	3.3	8.1	19.7	0.013	0.031
Cycle Inventory (Reactors + fabrication + reprocessing) [kg]					
Pu	767	3,285	4,818	17,520	9,094
Np	53	131	116	88	256
Am	***22***	88	307	701	***803***
Cm	11	44	158	175	413

An evaluation similar to the one performed in Report 1015129 and illustrated in Figure 5-3 in that report is now performed.

The adoption of multi-recycling leads to a reduction in the mass of TRU wastes, but this reduction is obtained by maintaining a substantial TRU inventory in the fuel cycle (i.e., in fuel facilities, reactors, storage, transport). Comparing the once-through fuel cycle to TRU burning in fast burner reactors with a conversion ratio of ~0.5, the amount in americium-bearing waste going to the geologic repository is reduced by a factor of 6.35/0.059, i.e., by a factor greater than 100[8]. However, assuming shutdown of the technology, the residual americium-bearing inventory in the fuel cycle would be 22 kg and 803 kg for the once-through and TRU-burning fuel cycle, respectively. To balance for this difference in Am inventory at shutdown, equal to 803 – 22 = 781 kg of Am wastes, equilibrium conditions would have to have been maintained for 124 years, which is obtained by dividing 781 by (6.35 - 0.059); the latter term represents the net annual mass flow of Am-bearing wastes going to the repository when compared once-through to multi-recycling of TRUs in fast burner reactors. Figure A-2 shows the number of years required to yield additional Am-bearing waste reduction levels. In conclusion, the benefits of minor actinide multi-recycling are mostly relevant in the context of reliance on the technology lasting for very long times (hundreds to thousands of years).

[8] Assuming 0.1% loss in separation processes (scenario involving fast burner reactors).

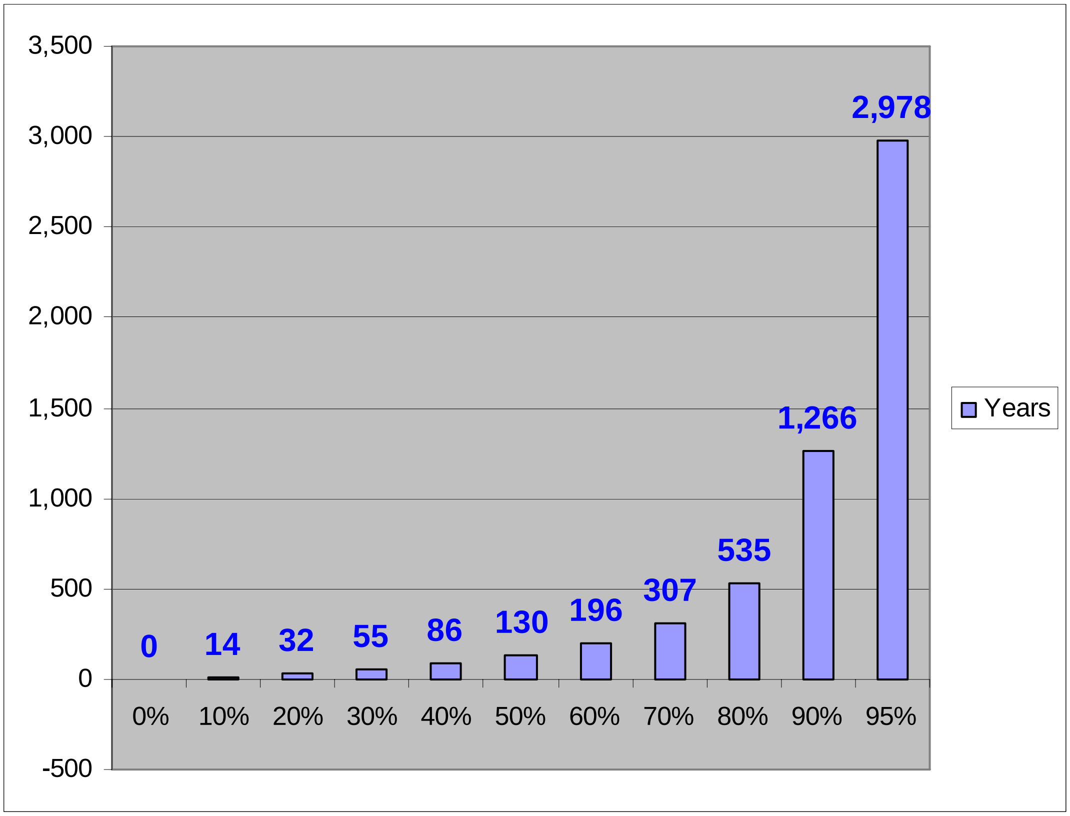

Figure A-1
Years Required for Achieving Specified Reduction Factors (from 10% to 95%) for Americium-Bearing Wastes (Multi-Recycling in Fast Burner Reactors with CR = 0.5 vs. Once-Through, Assuming 0.1% in Reprocessing Losses)

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