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Nuclear Fuel Cycle Transition Scenario Studies

Status Report

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NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

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FOREWORD

Under the auspices of the NEA Nuclear Science Committee (NSC), the Working Party on Scientific Issues of the Fuel Cycle (WPFC) was established to co-ordinate scientific activities regarding various existing and advanced nuclear fuel cycles, including advanced reactor systems, associated chemistry and flow sheets, development and performance of fuel and materials, and accelerators and spallation targets. The WPFC has different expert groups that cover the wide range of scientific fields in the nuclear fuel cycle.

The Expert Group on Fuel Cycle Transition Scenarios Studies was created in 2003 to consider R&D needs and relevant technology for an efficient transition from current to future advanced reactor fuel cycles. The objectives of the expert group are: i) to assemble and to organise institutional, technical and economic information critical to the understanding of the issues involved in transitioning from current fuel cycles to long-term sustainable fuel cycles or a phase-out of the nuclear enterprise; ii) to provide a framework for assessing specific national needs related to that transition.

This report discusses issues related to future fuel cycles, and gives an overview of possible transition scenarios for Belgium, Canada, France, Germany, Japan, the Republic of Korea, Spain, the United Kingdom and the United States, at the time of writing for each. The key issues and technologies which are crucial to the deployment of advanced fuel cycles are also identified.

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EXECUTIVE SUMMARY

Past studies on the implementation of partitioning and transmutation (P&T) performed within the NEA Nuclear Science Committee have mostly concentrated on equilibrium mode scenarios, wherein the global infrastructure is fixed and mass flows of materials are constant. These studies have resulted in a fairly comprehensive understanding of the potential of P&T to address nuclear waste issues, and have indicated the infrastructure requirements for several key technical approaches. While these studies have proven extremely valuable, several countries have also recognised the complex dynamic nature of the infrastructure problem: severe new issues arise when attempting to transition from current open or partially closed cycles to a final equilibrium or burn-down mode. While the issues are country specific when addressed in detail, it is believed that there exists a series of generic issues related only to the current situation and to the desired end point. Specific examples include:

- time lag to reach equilibrium, which can take decades to centuries;
- wide range of transmutation performance for the various technologies involved;
- accumulation of stockpiles of materials during either a transition phase or a growth period;
- very significant, and possibly prohibitive, investments required to reach equilibrium;
- complex interactions with final waste disposal paths.

These issues are critical to implementing a sustainable nuclear energy infrastructure. The work of the Expert Group activity has thus been devoted to:

- defining the key issues by collecting, comparing and organising information available from experts in member states;
- assembling information on the technologies available for the transition period;
- developing and assessing generic scenarios that are representative of the paths envisaged by member countries;
- evaluating for each generic scenario the major findings that will help guide country policy makers.

The first phase of the Expert Group's activity was focused on:

- definition of key issues;
- assessment of technologies;
- national scenario assessment.

As for the identification of key issues, a number of constraints have been raised that must be addressed:

- *Time lines*. The speed with which appropriate technologies can be developed and implemented will be tempered by factors such as investments required, penetration times for new technologies, regulatory requirements, etc.

- *Materials inventory effects*. At a minimum interim, or "lag" storage, capacities will probably be required under most, if not all, fuel cycle transition scenarios.

- *Materials management associated with implementation and operation of fuel cycle transition*. Appropriate material inventories must be available to provide the fuel sources needed to achieve fuel cycle performance goals.

- *Material dynamics impact on fuel cycle system performance requirements*. Since complete equilibrium will most likely not be achieved in envisioned fuel cycle transitions, the design and performance assessment of technological systems must take dynamic effects into consideration.

- *Economic*. Advanced nuclear systems need to compete with alternatives, nuclear and non-nuclear, in most countries in deregulated markets. On the one hand, government policies should recognise the value of security of supply and actinide management, but on the other hand the added cost of advanced fuel cycles should be as low as feasible. The key issue for policy makers is to make the right trade-offs in their strategy choices to reflect economics and social benefits associated with enhanced security of energy supply in the long term and reduced volumes and radiotoxicity of nuclear waste.

As concerns technology assessments, the following areas were identified as crucial with regard to the implementation of advanced fuel cycles:

- fuels for LWR recycle (from standard Pu recycle to TRU recycle);

- fuels for HTGR recycle (from U fuels to deep Pu burners);

- fuels for fast reactor recycle (fuels for homogeneous or targets for heterogeneous TRU recycle, dedicated fuels, *e.g.* for MA consumption);

- separations technologies (both with aqueous and pyro-processes);

- advanced reactors (critical or subcritical) and related technologies (*e.g.* specific coolant technology, materials).

As for national transition scenarios towards advanced fuel cycles, participants provided in some cases foreseen national development scenarios and in some cases hypothetical development scenarios based on consistent data (*e.g.* on available spent fuel stocks).

The findings of the group on all these topics are documented in the present report in separate chapters, together with some conclusions. Much of this report was completed over a year ago, and thus represents a snapshot of possible transition scenarios under consideration at that point in time.

While the Expert Group was actively undertaking its work, the interest of regional approaches to the implementation of future fuel cycles was pointed out, and it was decided to devote a second phase

of study to some specific scenarios for the implementation of innovative fuel cycles, for which some member countries were ready to supply relevant input data. The regional approach, and its available and foreseen applications, is discussed in Appendix 1.

Finally, it was decided to conduct a benchmark exercise to compare available scenario codes, to consolidate the results obtained with these codes for time-dependent cases. A benchmark has been defined and results will also be part of the outcome of the second phase activity.

Chapter 1

ISSUES ASSOCIATED WITH THE TRANSITION TO FUTURE NUCLEAR FUEL CYCLE TECHNOLOGIES AND STRUCTURES

The next several decades could witness sizable changes in nuclear fuel cycles implemented in various countries and regions throughout the world. The transition from current open or partially closed fuel cycles to ones offering long-term nuclear energy sustainability on the one hand or to phase-out of nuclear energy on the other will most likely involve the set of issues discussed in this paper. The issues potentially involved in fuel cycle transitions have seen relatively little focus, as most studies of nuclear fuel cycles have been made under equilibrium operation and mass flow assumptions. While fuel cycle transition issues are in the end country-specific, a set of generic issues can be identified that provide a general framework for further technical analyses. Such issues produce a set of overarching conditions and constraints that overlay results obtained from purely technology-based analyses.

1.1 National objectives in implementing advanced fuel cycles

Different countries will have different strategic reasons for adopting an advanced nuclear fuel cycle. These differing objectives can impact technology choices and the performance expected from such systems. The following table provides examples of choices, the drivers for making them, and general technology requirements. Such factors are also discussed later in Section 1.6.2.

1.2 Economic and sustainable development issues

As nuclear energy is competing with alternatives in deregulated markets, the implementation of advanced nuclear fuel cycles should take into account economics in order to avoid affecting the competitiveness of the nuclear option. A key issue in this regard is the recognition by policy makers of external costs associated with insecurity of energy supply, global climate change and long-term stewardship of high-level radioactive waste.

Analysts and policy makers recognise that external costs, supported by society as a whole rather than by consumers directly, are preventing market mechanisms to provide the right price signals. However, for various reasons, many externalities remain in present regulatory frameworks of most OECD countries. All national energy policies include security of energy supply as a central goal but market prices do not integrate the cost associated with energy independence or assurance of resource availability in the long term. Similarly, in spite of the efforts made, in the European Union in particular, to allocate a cost to carbon emissions, the establishment of a market price for those emissions has not yet been achieved.

Table 1.1. National energy policy objectives and associated technology requirements

Objective/drivers	Means to meet the objectives	Technology requirements
Enhance proliferation resistance, facilitate waste management and disposal	Minimise and monitor flows of separated ^{239}Pu, ^{231}Am and ^{99}Tc	Advanced spent fuel reprocessing, specific fuel and target forms, specialised storage/disposal media
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 ^{137}Cs, ^{90}Sr
Minimise 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

Regarding advanced fuel cycles, externalities are relevant in two ways:

- The internalisation of external costs associated with security of supply and/or carbon emissions increases the competitive margin of nuclear electricity and thereby facilitates the implementation of advanced cycles that may be more expensive than the once-through option.

- The recognition of the value of actinide burning, as a service to society through alleviating long-term stewardship of high-level radioactive waste, would reduce the cost barrier that may prevent choice in favour of advanced fuel cycles.

1.3 Advanced fuel cycles and nuclear development scenarios

The incentive to implement advanced fuel cycle options and their benefits depends on the evolution of nuclear capacity and electricity generation. Depending on the country considered, the role of nuclear energy in national supply may increase, remain stable or decrease towards an eventual phase-out in the coming decades.

In scenarios leading to eventual phase-out, the implementation of advanced fuel cycle schemes requiring new investments and some infrastructure building, even if the country relies on import of services, is not highly relevant. However, burning actinides may be an attractive option in countries where waste management and disposal is a social issue.

In scenarios with stable nuclear capacity, the choice of fuel cycle options will be based on cost benefit analyses as well as environmental and social concerns, and the outcome will vary from country to country depending on many factors. In such cases, transition scenarios will require careful crafting in order to monitor that material flows are adequate for fuelling advanced systems.

Obviously, the most favourable context for the development of advanced fuel cycles is a scenario of nuclear capacity growth where systems based on fast neutron reactors offer unique opportunities for ensuring long-term security of the nuclear fuel supply.

1.4 Issues arising from non-technical impacts on fuel cycle implementation

Any fuel cycle system, whether associated with nuclear or other energy systems, has associated with it substantial investments in supporting infrastructures. Infrastructure requirements and associated costs will accompany any direction in nuclear fuel cycle development and implementation, whether

under nuclear sustainability or phase-out conditions. The level of such investments may cause decision makers to weigh expanded nuclear fuel cycle implementation against other options for fuel and materials management, for example long-term storage in interim facilities or direct disposal in geologic facilities, assuming their feasibility and availability in the country or region.

Closely associated with cost and investment issues is the time required to implement required fuel cycle systems. If new, beyond evolutionary, technologies are involved or needed, then appropriate timelines must include significant periods devoted to development and demonstration up through prototype-level facilities. At the same time new regulatory requirements and associated rules and implementation infrastructure must be created. Ideally such regulatory process development would occur in parallel to technology development and demonstration. However it is more likely that regulatory process activities will occur sequentially after a period of technical development, thus adding to the period required for advanced fuel cycle implementation.

Finally, the existing status of the nuclear infrastructure in a given country will have a significant impact on implementation times. If key elements of the infrastructure are lacking or need substantial development or required levels of expertise are not available, then timelines will be drawn out. The health, vitality and completeness of a nation's nuclear infrastructure can be a deciding factor not only as concerns the time associated with advanced fuel cycle implementation, but its overall feasibility as well.

National decisions on fuel cycle implementation are also subject to requirements and guidelines associated with international nuclear non-proliferation norms. Inspection regimes by regional or international agencies will lead to design and implementation requirements on fuel cycle systems in areas such as transparency and materials accountability.

The regulatory environment and philosophy present in a specific country will create major impacts on new fuel cycle development requirements and timelines. Regulatory issues associated with waste disposal will depend heavily on whether legal requirements are defined in relative terms (toxicity of disposed product compared with natural uranium) or in absolute terms. As an example of the latter, regulations surrounding dose release from the proposed Yucca Mountain site in the United States require that the dose measured at the site boundary be some small fraction of the source term, no matter whether the source term of geologically disposed materials has been reduced significantly in quantity or in the characteristics of disposed products.

Finally the most potentially complicated factor having an impact on nuclear fuel cycles and their transition is that of prevailing public interest and opinion in the country of implementation. Factors related to public acceptance will strongly impact investment as well as timelines associated with implementation.

1.5 Technical issues associated with, and impacting, fuel cycle transition

For the Expert Group's efforts, general issues arising from technology and technological system choices are most amenable to analyses. As described previously, principal factors driving nuclear fuel cycle transitions will relate to nuclear materials management – ranging from ensuring long-term sustainability to final disposition of wastes associated with a drawdown or close out of the nuclear option.

1.5.1 Performance

Technologies developed and implemented in advanced fuel cycle strategies must meet certain performance objectives. For example in Table 1, objectives associated with increasing repository

performance or avoidance of additional repositories require that overall decontamination factors associated with the actinide content of disposed materials be 99 to 99.9%. This places demanding levels of performance on both separations and fuel fabrication portions of an advanced fuel cycle system in terms of waste production, losses, etc.

A second performance issue relates to costs required to achieve a desired or required level of performance. For example performance goals for losses occurring in reprocessing or fuel fabrication can be theoretically (and practically) met from a purely technology perspective. However their overall cost may limit the practicality of large-scale implementation. Advanced fuel cycle costs can (and will) be compared to other strategies such as direct disposal of spent fuel (costing of the order of $500 to $1 000 per kilogramme of heavy metal) versus reprocessing and transmutation. These comparisons can serve to set limits on overall costs that can reasonably be incurred to achieve a stated set of objectives.

A final performance issue relates to the ability of a given technology to scale up to levels required for full fuel cycle implementation. Such scale-up issues will also include the ability to function effectively (and at required capacity factors) under "industrial scale" systems where maintenance, equipment operational constraints, capital and operational outlays become deciding factors in technology choices.

1.5.2 National objectives and their impact on technology choices

The first set of issues concerns the overall objectives of the nuclear fuel cycle transition as introduced in Section 1.2. If in an environment of sustained or growing nuclear energy, stabilisation in the overall fuel cycle of radionuclide inventories, particularly plutonium, is a key objective; in such cases candidate timelines can drive, or at least greatly influence, technology decisions. For example when significant nuclear energy capacity exists in the form of thermal reactors, then plutonium-containing mixed-oxide fuels can augment standard low-enriched uranium fuels for the purpose of overall management of separated plutonium inventories. Likewise in environments where the phase-out of nuclear energy represents national policy, specific timelines for implementation and operation of burn-down systems may be specified by policy makers, in addition to overall requirements for final material residues and inventories. Finally timelines associated with transitions to nuclear systems sustainable over the long term (breeders) can be uncertain because of factors associated with new technology penetration (displacement of currently operating reactor fleets) along with externalities related to uranium supplies (price, surety, etc.).

Fuel cycle transition decisions made based on one primary objective, say plutonium inventory management, can have important implications for other nuclear materials areas. In the example alluded to previously wherein thermal reactors are used for plutonium management, the creation of larger inventories of higher actinides will occur as a consequence of successive thermal neutron capture on plutonium and higher actinides. On the other hand fast reactors would consume both plutonium and higher actinides efficiently but require significant investment in new systems and associated infrastructure. Very advanced systems such as an accelerator-driven higher actinide burner could be implemented further out in time as compared with reactor-based systems. Such machines would be specialised, aimed at consumption of higher actinides and residual plutonium from fast reactor consumption. Feed materials could be stored until the relatively small number (resulting from high support ratios) of such systems became available.

The above example indicates that fuel cycle technology choices will impact streams of materials destined for final disposal in geologic repositories. A fuel cycle consisting of thermal reactors optimised for plutonium consumption would send higher actinides such as americium to high-level waste.

Americium is a significant contributor to long-term heat management issues in repository environments. If higher actinides are separated from spent fuel then choices arise related to whether consumption in nuclear systems is desired versus long-term (centuries) decay storage. Curium represents such an example. Other choices, particularly those associated with separation and above-ground decay storage of fission products can be effective in dealing with intermediate-term heat management challenges associated with high-level waste or spent fuel disposal. These examples illustrate that fuel cycle choices should lead to analyses that focus on understanding potentially complex interactions of discharged materials with final disposal environments.

Material inventories, either from legacy production of nuclear energy or ongoing, perhaps rapidly increasing, nuclear power generation are an important issue in any fuel cycle transition scenario. Temporary, interim storage of spent fuel and possibly separated materials will be necessary under any fuel cycle transition. Start-up of materials management systems (and particularly breeder reactors in transitions to sustainability), will likely be (partially) fuelled using plutonium obtained from thermal reactor spent fuel. The availability of material inventories needed for nuclear systems is a key factor impacting time-dependent studies of fuel cycle transition.

In fuel cycle transition scenarios the production of low-enriched uranium will continue, as even with respect to a movement towards sustainable nuclear fuel cycles, a large fraction, or even the majority, of reactors will continue to be thermal-neutron-based. In such systems the movement towards higher burn-up fuel will most likely continue to occur in countries committed to long-term nuclear energy production. Such trends could require higher levels of enrichment to achieve higher burn-ups, which in turn would increase enrichment capacities needed during fuel cycle transitions.

Technology choices and performance features of chosen technologies will have direct impacts on times required to reach material equilibrium. Fast spectrum systems that have favourable cross-sections for materials management also require large (as compared with thermal systems) inventories of materials residing in the system. For environments where transition to a sustainable nuclear fuel cycle is a primary objective, the time required to reach equilibrium will take a number of decades at the very least, and potentially much longer (centuries). Conversely, fuel cycle systems implemented to burn down materials in phase-out scenarios are not designed to reach equilibrium. Thus for most, if not all, transition scenarios reaching nuclear equilibrium will not occur. Fuel cycle strategies and technologies will have to contend with continuing time-dependence of certain material inventories, at least for the foreseeable future.

Finally, any technology developed and implemented under the transition of fuel cycles will have to meet safety and regulatory requirements at least as high as those associated with today's nuclear power producers. Developing appropriate databases for technology components of advanced fuel cycles could introduce significant time lags into fuel cycle implementation. A particularly relevant example involves fuels that would need to be employed for purposes such as actinide management. The qualification and certification of such fuels could involve a decade-long period to meet current and future performance, safety and regulatory requirements.

1.6 Other considerations

Achieving nuclear materials management objectives may lead countries to pool facilities and other technology resources. A country lacking in full fuel cycle facilities may pursue co-operative agreements with a neighbouring country having more extensive nuclear fuel cycle capabilities. Such arrangements, although politically challenging, could lead to more cost-effective fuel cycle approaches for both countries involved in such a partnership.

1.7 The impact of general fuel cycle issues on the activities of the Expert Group

The identification and discussion of generic issues in this paper lay out a number of constraints that must be addressed in follow-up analyses. The summary below indicates associated impacts on the activities of the Expert Group.

- *Time lines*. More ideal assumptions associated with the speed with which appropriate technologies can be developed and implemented will be tempered by factors such as investments required, penetration times for new technologies, regulatory requirements, etc.

- *Materials inventory effects*. At a minimum interim, or "lag" storage, capacities will be probably required under most, if not all, fuel cycle transition scenarios.

- *Materials management associated with implementation and operation of fuel cycle transition*. Appropriate material inventories must be available to provide fuel sources needed to achieve fuel cycle performance goals.

- *Material dynamics impact on fuel cycle system performance requirements*. Since complete equilibrium will most likely not be achieved in envisioned fuel cycle transitions, the design and performance assessment of technological systems must take dynamic effects into consideration.

- *Economics*. Advanced nuclear systems have to compete in market environments with current nuclear systems and other energy sources. The economic impact of implementing fuel cycles aiming at enhancing security of energy supply, facilitating radioactive waste management and disposal, and increasing proliferation resistance will depend on the degree of internalisation of external cost in national energy policies. In this regard, it should be noted that consultation with all stakeholders in civil society is a prerequisite for the successful internalisation of such social costs.

These issues may be emphasised to varying degrees under country-specific scenarios but they have served as overall guidance to the Expert Group's further analysis efforts.

Chapter 2

OVERVIEW OF NATIONAL TRANSITION SCENARIOS

National transition scenarios as provided by Belgium, Canada, France, Korea, Japan, Spain and the United States are presented in this chapter. A short preliminary contribution from BNFL is also available in Appendix 2.

In some cases (Canada, France, Korea, Japan, United States), the national scenarios presented are potential development scenarios towards innovative fuel cycles which have been discussed and are the object of consensus at a wider national level. In other cases (Belgium, Spain), the transition scenarios are more hypothetical, and essentially correspond to the points of view of the authors and the organisms they represent.

2.1 The Belgian implementation scenario

At the end of 2002 the total installed electric power in Belgium was 16 200 MWe, of which 40% (6 485 MWe) corresponds to the seven nuclear power plants installed on the two Belgian sites of Doel (four power plants) and Tihange (three power plants) and 25% participation in the two French Units B1 and B2 at Chooz on the Belgian-French border. Installed nuclear power in Belgium corresponds to 5 800 MWe (see Table 2.1). In 2003 the government decided to progressively phase out nuclear energy, determining to close down Belgian NPPs after 40 years of operation. First-generation units (Doel 1, Doel 2, Tihange 1) will be closed in 2015 and the remaining NPPs in 2022-2025. Nevertheless, this phase-out is subject to certain conditions, namely:

- the guarantee of energy independence should not be affected;
- the engagement to respect the Kyoto agreement (reducing CO_2 production by 7.5% in 2010 as compared to 1990 levels).

If these conditions are not met, the phase-out decision may be reconsidered.

2.1.1 Present fuel type

The real status of the Belgian cycle is rather complex. Four types of units are to be taken into account:

- three types of UO_2 assemblies: 14 × 14, 15 × 15 and 17 × 17;
- three types of UO_2-Gd_2O_3 assemblies (burnable poisons): 8, 12 and 16 UO_2-Gd_2O_3 pins;
- three different active fuel lengths: 2.44 m for 14 × 14 assemblies, 3.66 m for 15 × 15 and some 17 × 17 assemblies and 4.27 m for some 17 × 17 assemblies;

- from 12 to 18 month cycles;
- from 33 GWd/tHM to 55 GWd/tHM final burn-up;
- mixed UO_2-MOX cycles in Doel 3 and Tihange 2 (from 1995 to end-2005);
- time and unit dependant load factors: from 0.75 to 0.98.

For Belgian cycle modelling, only three basic fuel cycles are considered (see Table 2.1):

1. Short cycle (12 months) for Doel 1, Doel 2 and Tihange 1:
   - final UO_2 average burn-up of 33 GWd/tHM;
   - 1/3 core loading replacement every 12 months.
2. Long cycle (18 months) for Doel 3, Doel 4, Tihange 2 and Tihange 3:
   - final UO_2 average burn-up of 50 GWd/tHM;
   - 1/3 core loading replacement every 18 months.
3. Mixed UO_2-MOX cycle in Doel 3 and Tihange 2:
   - limited to 66.4 tHM resulting from UO_2 reprocessing;
   - between 1995 and 2005 (last MOX cycle at end-2005);
   - final MOX average burn-up of 45 GWd/tHM;
   - final UO_2 average burn-up of 50 GWd/tHM.

Table 2.1. Belgian nuclear power plants: model of present situation

NPP	BOL	EOL	P_e [MWe]	P_{th} [MWth]	Eth [Gwd/y]	Fuel	Burn-up [GWd/t]	Fuel cycle	Total/y [t/y]	Total [t]
Doel 1	1975	2015	393[a]	1 192	370[b]	UO_2	33[d]	3 × 1.0 y[f]	11.2	448
Doel 2	1975	2015	433[a]	1 311	407[b]	UO_2	33[d]	3 × 1.0 y[f]	12.3	493
Doel 3	1982	2022	1 008[a]	3 054	948[b]	UO_2[c]	50[d,e]	3 × 1.5 y[f]	19.0	758
Doel 4	1985	2025	986[a]	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	742
Tihange 1	1975	2015	945[a]	2 865	889[b]	UO_2	33[d]	3 × 1.0 y[f]	26.9	1 077
Tihange 2	1982	2022	1 008[a]	3 054	948[b]	UO_2[c]	50[d,e]	3× 1.5 y[f]	19.0	758
Tihange 3	1985	2025	986[a]	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	742
Doel	1975	2025	2 820	8 545	2 651				48.8[g]	2 441
Tihange	1975	2025	2 939	8 907	2 763				51.5[g]	2 577
Total	1975	2025	5 759	17 452	5 414				100.4[g]	5 018

[a] Thermodynamic efficiency is assumed to be 0.33.

[b] Load factor is assumed to be 0.85.

[c] Mixed UO_2-MOX cycle (about 1/5 of MOX) between 1995 and 2005.

[d] This burn-up does not necessarily correspond to the real burn-up. This is only the "model burn-up" considered for the calculations.

[e] The average MOX burn-up is 45 GWd/tHM.

[f] This cycle does not necessarily correspond to the real cycle. This is only the "model cycle" considered for the calculations.

[g] Averaged over 50 years.

We consider a typical loading scheme of n fuel zones with an average burn-up increment of the fuel in each zone of b [GWd/tHM] per reactor cycle. At each cycle, $1/n$ of the fuel (the fuel which reached a burn-up of $B = n.b$) is replaced by fresh fuel. The reactor fuel loading *Mcore* [tHM], the fuel going out each cycle from the reactor *Mout* [tHM] and the fuel yearly consumption *My* [tHM] are then given by:

$$M_{core} = \frac{n \cdot c \cdot E}{B}$$

$$M_{out} = \frac{c \cdot E}{B}$$

$$M_{y} = \frac{E}{B}$$

with:

$$c = \frac{\text{cycle duration [y]}}{1\,[\text{y}]}$$

$$E = f \cdot P_{th}[\text{GW}] \cdot 365\,[\text{d}]$$

where E is the thermal energy effectively produced in one year and f is the load factor. With a load factor of 0.85, the estimated spent fuel in 2025 is about 5 000 tHM. Apart from 670 t (UO_2 fuel) which has been reprocessed, no further reprocessing is foreseen. Considering the growth of electricity demand during the last decade (3% per year), the limited availability of other resources and the conditions imposed by the nuclear phase-out, one can foresee a power shortage in the future if no appropriate measures are taken. In order to compensate for the possible shortage, it is reasonable to consider that Belgium may not be renouncing nuclear energy, depending on the reigning political climate. Future deployment of nuclear reactors should thus not be ruled out.

2.1.2 Transition fuel cycle

A realistic park deployment could be envisaged as follows (see Tables 2.2 and 2.3):

- The shutdown in 2015 of the three oldest units (Doel 1 Doel 2, Tihange 1) corresponding to a net capacity of about 1 800 MWe and replacing them with an EPR (1 800 MWe), perhaps decided upon toward 2010 and put in service for 2015.

- The second-generation PWRs (Doel 3, Doel 4, Tihange 2, Tihange 3) lifetimes could easily be extended (PLEX) from the present (political) determination of 40 years up to 60 years, meaning that these reactors would be taken out of service in 2042-2045.

- At this date one can consider that the Gen-IV fast reactors will be ready for deployment and would take care of their own long-lived waste. Generation IV fast reactors could then replace the second-generation PWRs.

- The "dirty" Pu (3 t) resulting from the second recycled MOX in PWRs as well as the accumulated MAs (15 t) would be absorbed in one of several accelerator-driven systems (ADS) (600 MWth). A realistic start-up date for these industrial ADS could be foreseen in 2045. The ADS power will be adapted to the total stockpile of MAs and dirty Pu. It is not necessary for a large scale ADS to be installed in Belgium.

- Following this scenario, the total installed power is assumed to remain constant.

Table 2.2. Chronology of the Belgian scenario

Year	Event	P_e [MW]	P_{th} [MW]
1975	Start Doel-1 (400 MWe), start Doel-2 (400 MWe), start Tihange-3 (1 000 MWe)	1 771	5 368
1982	Start Doel-3 (1 000 MWe), start Tihange-2 (1 000 MWe)	3 787	11 476
1985	Start Doel-4 (1 000 MWe), start Tihange-3 (1 000 MWe)	5 759	17 452
1988	Begin interim storage in Doel-1, Doel-2 and Tihange-1 (no reprocessing)	5 759	17 452
2015	Stop Doel-1 (400 MWe), stop Doel-2 (400 MWe), stop Tihange-3 (1 000 MWe), start EPR (1 800 MWe)	3 988 5 759	12 084 17 452
2022	PLEX-20y Doel-3 (1 000 MWe), PLEX-20y Tihange-2 (1 000 MWe)	5 759	17 452
2025	PLEX-20y Doel-4 (1 000 MWe), PLEX-20y Tihange-3 (1 000 MWe)	5 759	17 452
2042	Stop Doel-3 (1 000 MWe), stop Tihange-2 (1 000 MWe) start self-burning FR [SFR, LFR] (2 × 1 000 MWe)	3 744 3 744	11 344 17 405
2045	Stop Doel-4 (1 000 MWe), stop Tihange-3 (1 000 MWe), start self-burning FR [SFR, LFR] (2 × 1 000 MWe), start ADS (3 × 600 MWth)	3 771 3 771 3 771	11 429 17 489 19 289
2075	Stop EPR (1 800 MWe), start self-burning FR [SFR, LFR] (2 × 1 000 MWe)	4 000 6 000	13 921 18 982
2085	Stop ADS (3 × 600 MWth)	6 000	18 182

Table 2.3. Belgian nuclear power plants: model of future situation

NPP	BOL	EOL	P_e [MWe]	P_{th} [MWth]	Eth [Gwd/y]	Fuel	Burn-up [GWd/t]	Fuel cycle	Total/y [t/y]	Total [t]
Doel 1	1975	2015	393[a]	1 192	370[b]	UO_2	33[d]	3 × 1.0 y[f]	11.2	448
Doel 2	1975	2015	433[a]	1 311	407[b]	UO_2	33[d]	3 × 1.0 y[f]	12.3	493
Doel 3	1982	2022	1 008[a]	3 054	948[b]	UO_2[c]	50[d,e]	3 × 1.5 y[f]	19.0	758
PLEX D3	2022	2042	1 008	3 054	948[b]	UO_2	50[d]	3 × 1.5 y[f]	19.0	379
Doel 4	1985	2025	986[a]	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	742
PLEX D4	2025	2045	986	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	371
Tihange 1	1975	2015	945[a]	2 865	889[b]	UO_2	33[d]	3 × 1.0 y[f]	26.9	1 077
Tihange 2	1982	2022	1 008[a]	3 054	948[b]	UO_2[c]	50[d,e]	3 × 1.5 y[f]	19.0	758
PLEX T2	2022	2042	1 008[a]	3 054	948[b]	UO_2	50[d]	3 × 1.5 y[f]	19.0	379
Tihange 3	1985	2025	986[a]	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	742
PLEX T3	2025	2045	986[a]	2 988	927[b]	UO_2	50[d]	3 × 1.5 y[f]	18.5	371
EPR	2015	2075	1 771	5 368	1 665	UO_2	50[d]	3 × 1.5 y[f]	33.3	1 999
Total	1975	2025				UO_2			85.2[g]	8 516

[a] Thermodynamic efficiency of 0.33 is assumed.

[b] Load factor of 0.85 is assumed.

[c] Mixed UO_2-MOX cycle (about 1/5 of MOX) between 1995 and 2005.

[d] This burn-up does not necessarily correspond to the real burn-up. This is only the "model burn-up" considered for the calculations.

[e] The average MOX burn-up is 45 GWd/tHM.

[f] This cycle does not necessarily correspond to the real cycle. This is only the "model cycle" considered for the calculations.

[g] Averaged over 50 years.

2.1.3 Calculations

Due to the simplified representation of the Belgian cycle adopted, simple models are also employed for fuel evolution.

PWR modelling

- Three types of fuel are considered:
  - UO_2 3.3% 33 GWd/tHM short cycle for D1, D2 and T1;
  - UO_2 4.3% 50 GWd/tHM long cycle for D3, D4, T2,T3 and EPR;
  - MOX 7.7% 45 GWd/tHM long cycle for D3 and T2 (1995-2005).
- Fuel cell determined to be in an infinite lattice.
- Lattice pitch chosen to conserve the moderation ratio of the assembly (1.31 cm in place of the typical 1.26 cm).
- The error introduced by these simplifications is maximum 15% (FP) with respect to a multi-assembly calculation for MOX evolution.
- Recalculation of the neutron energy spectrum each step of 1 GWd/tHM.
- Cycle in equilibrium.
- The first 670 t already reprocessed are not taken into account in the study. All evaluations given below do not include this already reprocessed waste.
- Load factor of 0.85 for all installations.

ADS modelling

The following assumptions are made:

- An industrial ADS which operates between 2045 and 2085 at a constant power of 600 MWth with an average fuel power density of 1 kW/cm^3. This corresponds to a fuel loading of 3.6 tonnes (2.2 t HM: 1.3 t MA and 0.9 t Pu).
- An average cycle of two years (660 effective full power days, 180 GWd/tHM) followed by a decay period of 10 years (this period is the time needed for fuel cooling and re-fabrication).
- Homogeneous core loading.
- "Reasonable" burn time is considered to be four years effective full power. Indeed, calculations show that the effective multiplication factor begins to increase, reaches a maximum (reactivity increase of about 6 000 pcm) and then decreases to about the same initial effective multiplication factor four years effective full power later.

- Only the second-generation Pu (3.3 tonnes) and all accumulated MA (20.4 tonnes) is burned.
- The proton source is 600 MeV.
- MgO (40%) + Pu (24%) + MA (36% = 15% Am, 15% Np, 6% Cm) inert matrix loading is used.
- ρ_{fuel} = 6.1 g/cm^3, ρ_{HM} = 3.7 g/cm^3.
- The neutron spectrum is taken from the ADS prototype MYRRHA [8] (central channel at midplane) with the same energy of the proton source (600 MeV).
- MCNPX-2.5.0 [4] calculation.
- Time-independent neutron spectrum.

Calculation code

The code used for all calculations is ALEPH [1-3], a Monte Carlo activation and burn-up C++ interface code using any version of MCNP(X) [4] for particle transport, ORIGEN 2.2 [5] for evolution calculations (slightly modified) and NJOY 99.90 [6] for the nuclear data processing of the original ENDF files. ALEPH is currently under development at SCK•CEN in collaboration with Ghent University in the framework of the MYRRHA project. The main idea behind ALEPH was to create a general purpose continuous energy Monte Carlo burn-up and activation code that is efficient, flexible and easy to use:

- *Efficient*: A method that allows accelerating the calculation in an optimal way has been identified. However, it has been proven that, all other things being equal (*i.e.* no hardware modifications and the same precision), the acceleration factor reaches 95% of the theoretically maximum possible one (*i.e.* when the CPU time needed to perform the burn-up calculation equals the time needed to evaluate only the effective multiplication factor), while ensuring exactly the same accuracy. Using this method, reductions in calculation time by factors of 30 to 100 have been observed.
- *Flexible*: ALEPH uses direct access to the original ENDF data files for its needs in nuclear data. ALEPH is the first burn-up code allowing multi-particle calculations (can take into account the coupling between the proton source and the core in an ADS). ALEPH allows variable geometry (simulation of boron concentration, temperature effects, core reshuffling, etc.) and variable materials (simulation of control rod movement for example).
- *Easy to use*: Only minor modifications to the MCNP(X) input files are needed. Neither ORIGEN nor NJOY input files are required.

ALEPH has been successfully tested against APOLLO2, WIMS8a and experimental ARIANE data [7].

2.1.4 Result

PWR

- With phase-out, the accumulated waste between 1975 (first PWR) and 2025 (last PWR) is estimated at 4 658 tonnes:
  - 4 380 t of U;
  - 49 t of first-generation Pu;
  - 3 t of second-generation Pu;
  - 9 t of MA;
  - 217 t of FP.
- Without phase-out, the accumulated waste between 1975 (first PWR) and 2075 (last EPR) is estimated at 7 825 tonnes:
  - 7 340 t of U;
  - 81 t of first-generation Pu;
  - 3 t of second-generation Pu;
  - 20 t of MA;
  - 381 t of FP.
- Belgium should retain its first-generation Pu for start-up of the self-burning FR. Indeed, the Pu needed to start the self-burning FR is evaluated between 60 t and 90 t (based on 10 to 15 t per GWe).

The evolution of HM and FP inventory in interim storage is given in Figures 2.1 to 2.5.

ADS

- 54% of the MA loaded in one ADS (taking into account the natural decay of the same waste in storage) are burned in four years effective full power:
  - 59% of the Np are burned (Cm [4 y effective full power]/Cm [natural evolution] = 0.41);
  - 19% of the Pu are burned (Pu [4 y effective full power]/Pu [natural evolution] = 0.81);
  - 53% of the Am are burned (Am [4 y effective full power]/Am [natural evolution] = 0.47);
  - 27% of the Cm are burned (Cm [4 y effective full power]/Cm [natural evolution] = 0.73).
- ADS transmutation capabilities (in a homogeneous scheme) are comparable to those of the FR (for-example, the recycling of Am and Pu reduces the Am-Cm content in the cycle by a factor of two). The use of an inhomogeneous scheme should increase ADS transmutation capabilities.
- The remaining MA waste could be incinerated in FR.

- If MA decrease globally, certain isotopes increase:
  - ^{242m}Am content is increased by a factor of 32;
  - ^{242}Cm content (major contribution to long-term residual power and to neutronic emission by spontaneous fission) is increased by a factor of 30 (because of the increase in ^{242m}Am);
  - ^{238}Pu content [major contribution to neutronic emission through (*a*,*n*) reactions] is increased by a factor of 13;
  - ^{241}Pu content is increased by a factor of 2.8;
  - ^{244}Pu content is increased by a factor of 2.4.

- There is not enough second-generation Pu to "maintain" the effective multiplication factor. Indeed, the Pu needed to burn 1 t of MA is about 0.7 t Pu/t MA. The Pu needed to burn all accumulated MA is therefore about 14 t, more than four times that available. Countries that have decided to bring a halt to nuclear energy production could provide the required Pu to keep the ADS running.

- If the required Pu is available and if the MA composition of the inert matrix is adapted to the Belgian MA vector (whose inert matrix contains too much Cm and not enough Am), three ADS (about 10% of the installed thermal power) should be necessary to reduce by a factor of two in 24 years the entire MA accumulated between 1975 and 2075.

2.1.5 Conclusions

- The evaluated stockpile of waste in Belgium (with no increase in electricity demand) resulting from the thermal reactor park is 4 380 tonnes (52 t Pu, 9 t MA, 217 t FP) with phase-out (*i.e.* between 1975, first PWR and 2025, last PWR) and 7 825 tonnes (84 t Pu, 20 t MA, 381 t FP) without phase-out (*i.e.* between 1975, first PWR and 2075, last EPR).

- According to the present study, Belgium should maintain all of its first-generation Pu for the eventual start-up of the self-burning FR. Indeed, the Pu required to start the self-burning FR is evaluated between 60 t and 90 t (based on 10 t to 15 t per GWe).

- Elimination of 54% of the MA could be accomplished in 24 years with three 600-MWth industrial ADS (corresponding to about 10% of the nuclear installed thermal power) if enough dirty Pu is available. Countries that have stopped nuclear energy production could provide the required Pu to keep the ADS running.

- ADS (if considered as a "burner") should therefore be envisaged only in regional scenarios and complementary to FR.

- More elaborate burning schemes (inhomogeneous burning) must be considered if higher elimination rates, say 90%, are desired. However, the time to reach equilibrium will be much longer.

- Full scale (industrial ADS) burn-up calculations with ALEPH (as has already been done for the prototype MYRRHA) are planned. More accurate results about the burning capabilities of industrial ADS will be obtained, leading to more reliable data for decision making.

Figure 2.1. Reference scenario: Total inventory per element in interim storage

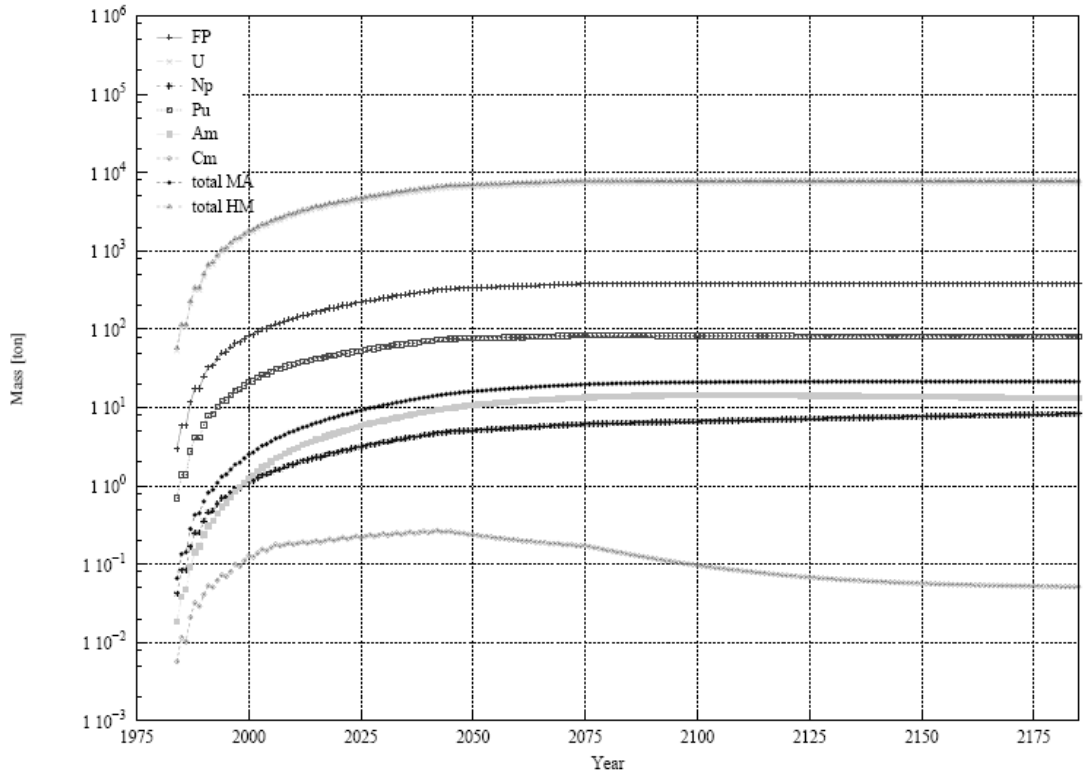


Figure 2.2. Reference scenario: MA inventory per element in interim storage

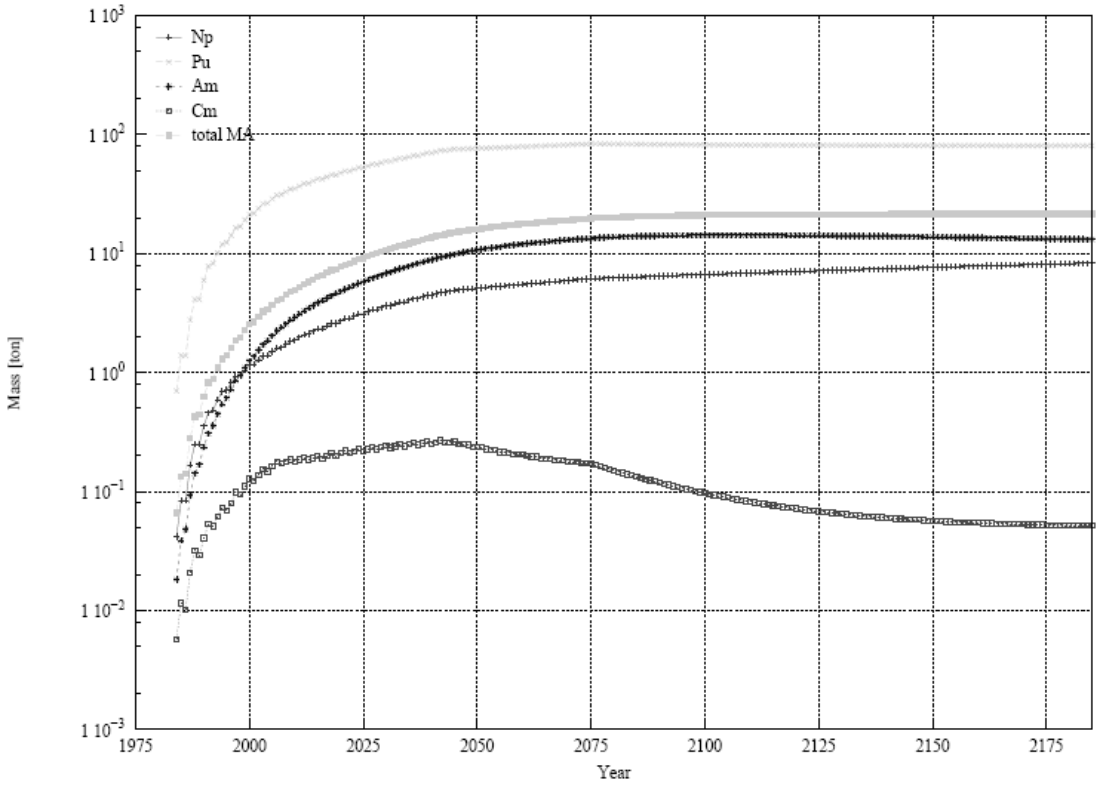


Figure 2.3. Reference scenario: MA inventory per isotope in interim storage

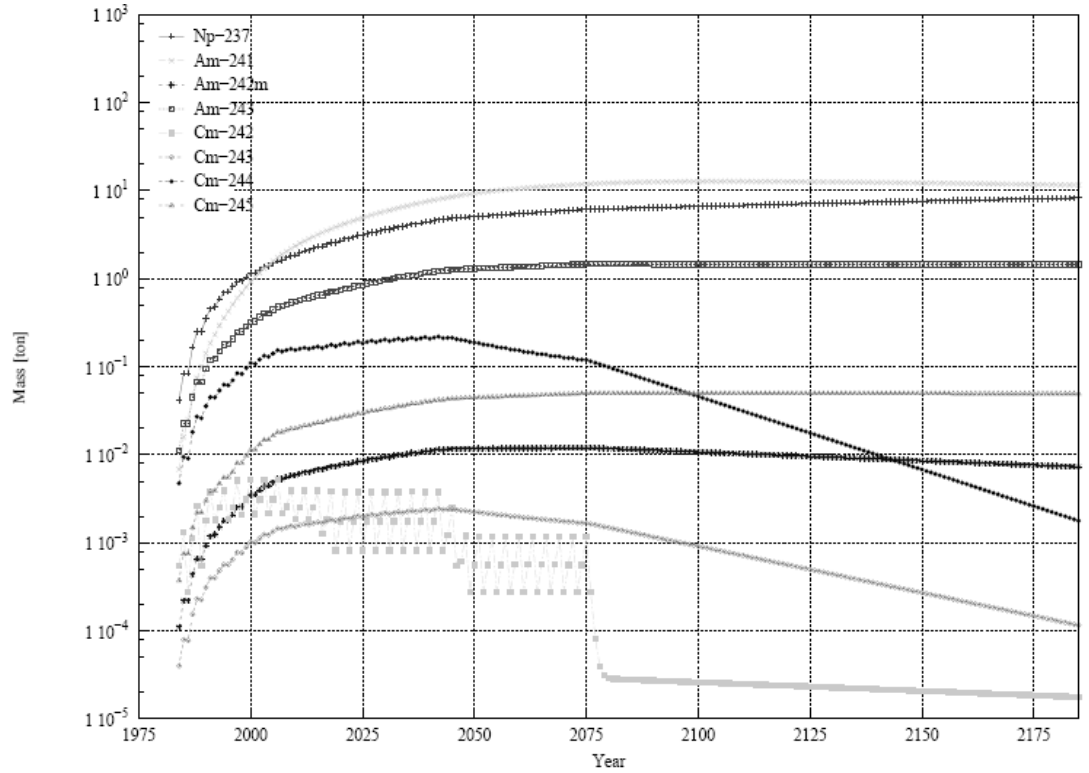


Figure 2.4. Reference scenario: Pu inventory per isotope in interim storage

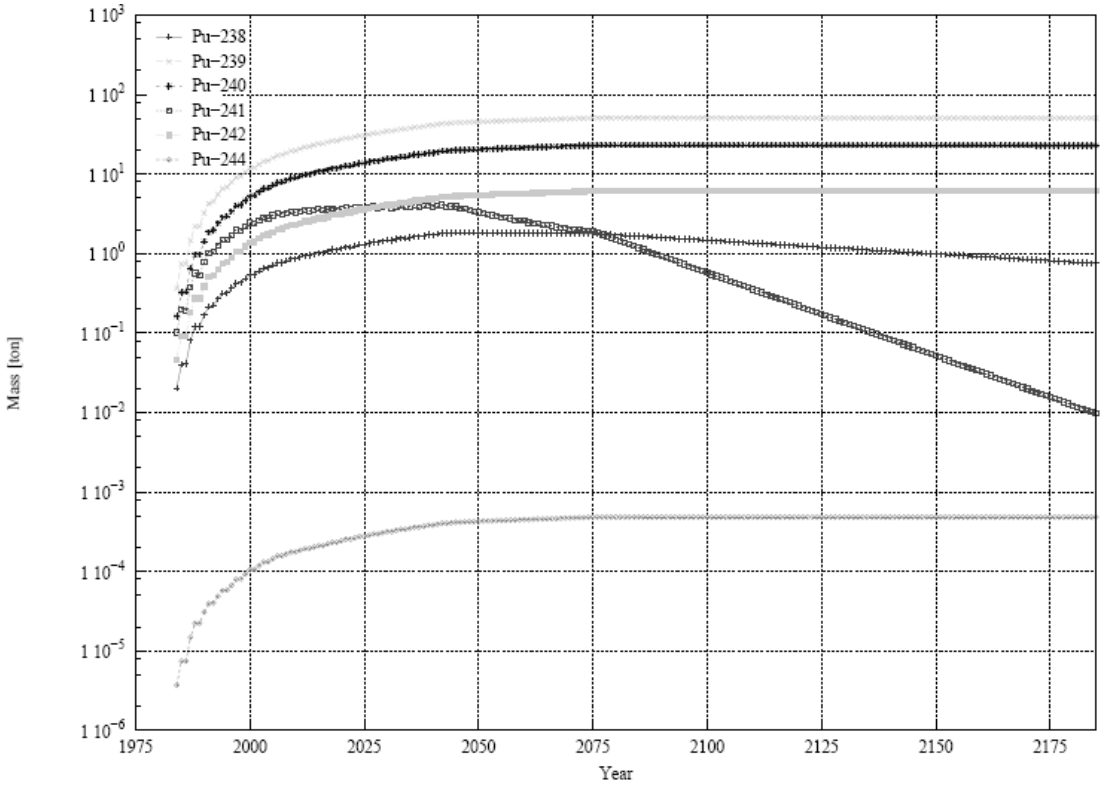
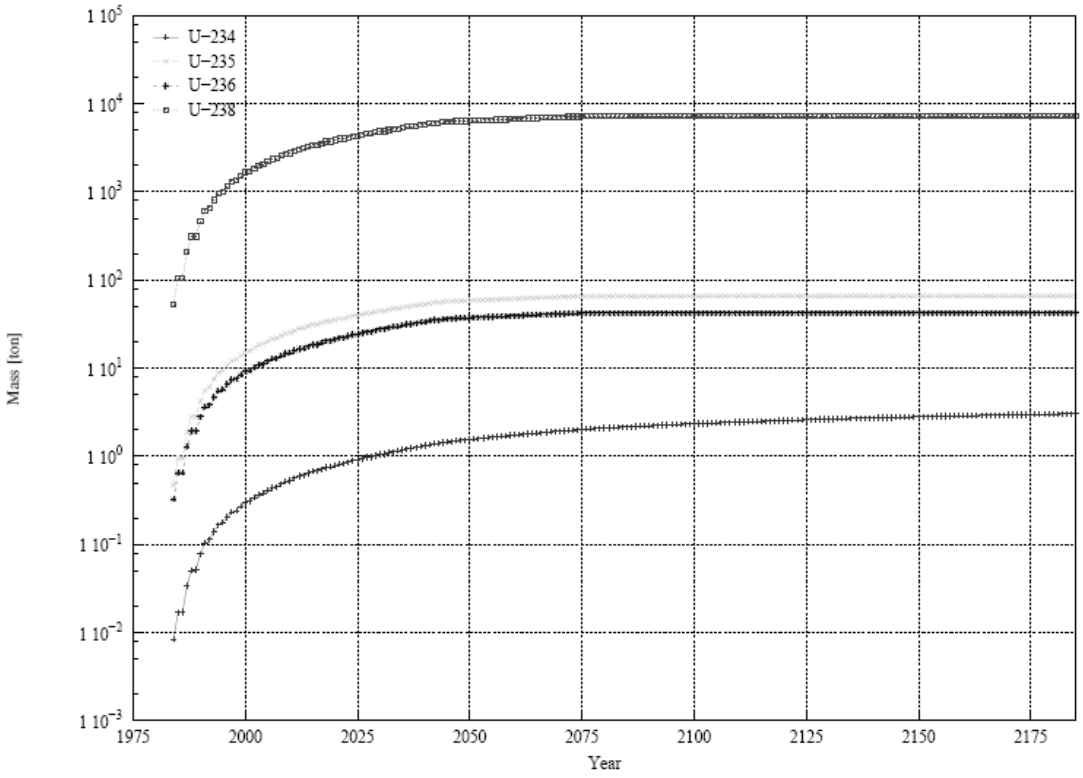


Figure 2.5. Reference scenario: U inventory per isotope in interim storage



2.2 Canadian work on transition scenarios

The Canadian nuclear power programme is based on CANDU® technology,[1] which provides unequalled flexibility for the use of different fuel cycles. Its inherent high neutron economy, fuel channel design, on-power refuelling capability and simple fuel bundle design allow for the optimisation of an assortment of different nuclear fuel cycles.

Atomic Energy of Canada Limited (AECL) is actively examining CANDU fuel cycles that exploit synergies between heavy-water-moderated CANDU reactors (HWRs) and light-water reactors (LWRs), as well as fast reactors. Optimisation of thermal-to-fast reactor transition scenarios involves the exploitation of these synergies.

Canadian research has shown that there are unique and valuable roles for heavy water reactors in thermal-to-fast reactor transition scenarios. Heavy water reactors could be used to match the size of the reactor fleet to electricity demands, make efficient use of fissile resources and to manage the minor actinide inventory in the fuel cycle.

2.2.1 Transition to fast reactors with low breeding ratios

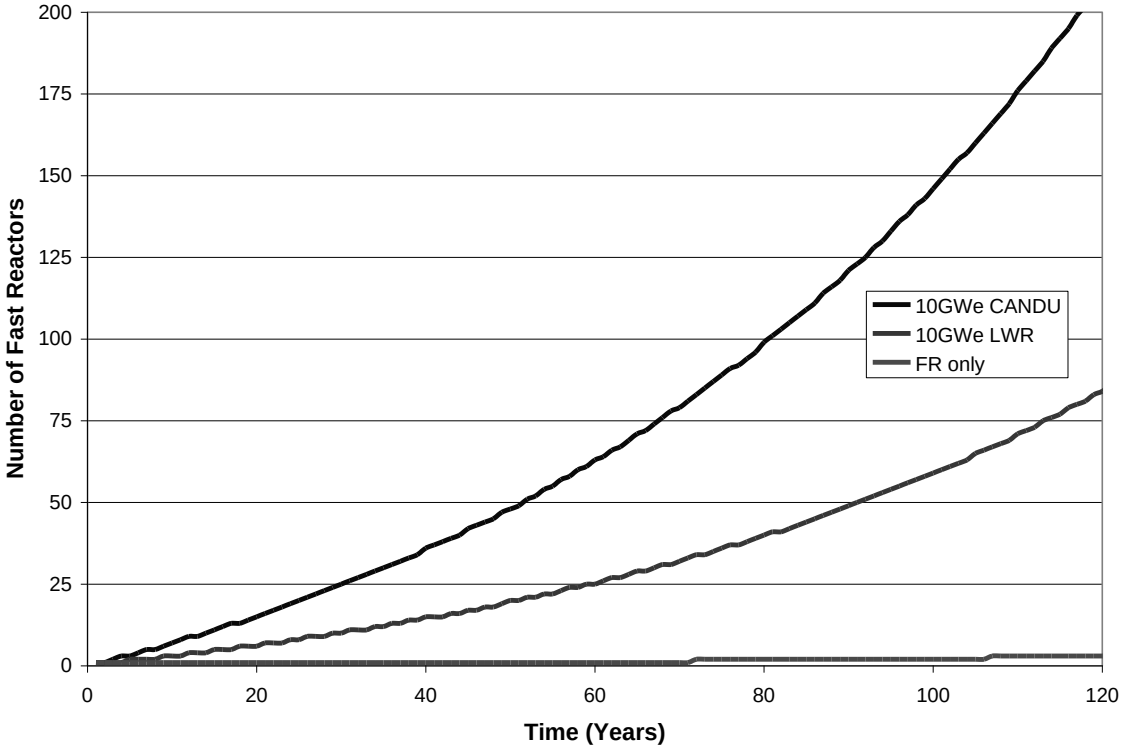
Heavy water reactors can efficiently supply fissile material for a fast reactor fleet. In a transition scenario where there is a limited supply of available fissile material, and where the fast reactors have low breeding ratios, the rate at which the fast reactor fleet can be increased is limited by the large

[1] CANDU® (CANada Deuterium Uranium) is a registered trademark of Atomic Energy of Canada Limited.

fissile requirement for the initial fast reactor core load. This would make it difficult to increase the size of the fast reactor fleet to match increasing demand for electricity. In these scenarios, a small fleet of HWRs would be the most resource-efficient way to convert natural uranium into fissile material for use in the initial core load for next-generation fast reactors. In scenarios where a supply of plutonium comes from reprocessing spent LWR fuel, the addition of a small number of HWRs would allow the reprocessed uranium from the LWR spent fuel to be converted to both fissile plutonium and depleted uranium for use in the fast reactors, while generating valuable electricity.

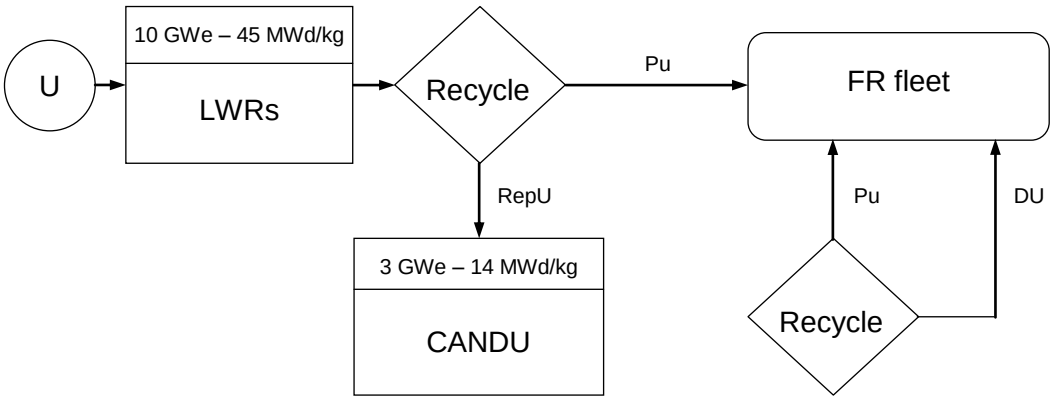
As an example, a nominal, low-breeding-ratio fast reactor could have a doubling time (the time required to produce enough fissile material to start another fast reactor) as high as 70 years. In this case, the increase in the fast reactor fleet would be extremely slow. The addition of fissile material from recycling of spent fuel from a small (10 GWe) fleet of either LWRs or HWRs would allow the fast reactor fleet to be increased much more quickly. Three idealised scenarios are illustrated in Figure 2.6, in which the spent fuel from LWRs or HWRs is reprocessed and the Pu used in the initial core of FRs. Additionally, the natural uranium resources required for 10 GWe of HWRs would be lower than for 10 GWe of LWRs.

Figure 2.6. Growth of a fast reactor fleet



As mentioned earlier, a combination of LWRs and HWRs could provide an extremely efficient supply of both fissile material and depleted uranium by exploiting the low fissile requirements of HWRs. An example of such a fuel cycle is shown in Figure 2.7. This type of fuel cycle would take maximum advantage of existing thermal reactor technology.

Figure 2.7. Use of thermal reactors to generate fissile material for fast reactors

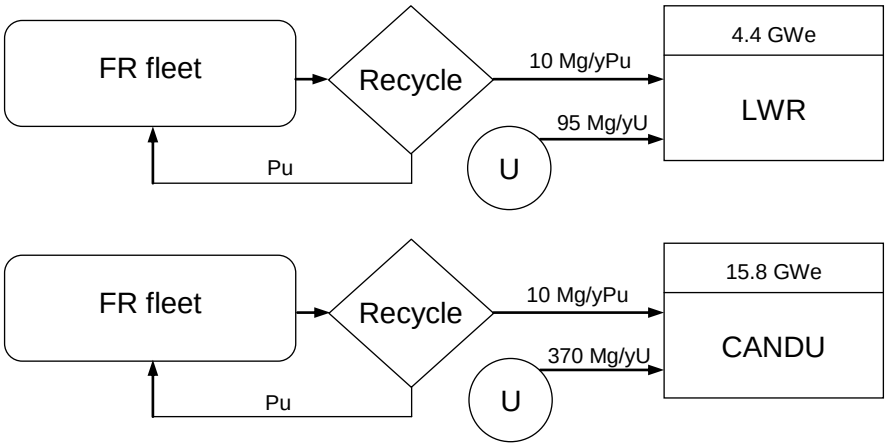


2.2.2 Transition to fast reactors with high breeding ratios

The high neutron economy of HWRs allows them to produce a large amount of energy from a small amount of fissile material. In fuel cycle scenarios involving fast reactors with high breeding ratios, net plutonium production would exceed the demand for increases in the size of the fast reactor fleet. Here, an HWR could efficiently convert the excess plutonium production to electricity with minimal impact on uranium resource utilisation through either a plutonium-uranium MOX fuel cycle, or a plutonium-thorium fuel cycle. The introduction of ^{233}U recycle in a plutonium-thorium fuel cycle would significantly increase the amount of energy produced from the initial plutonium feed. In these fuel cycles, HWRs would make much more efficient use of plutonium, uranium and thorium resources than LWRs and, in the extreme, an HWR-based thorium fuel cycle with ^{233}U recycle could produce a large amount of energy from a very small amount of plutonium input.

Figure 2.8 shows a comparison of a simple uranium-plutonium, mixed-oxide (MOX) fuel cycle implemented with LWRs or HWRs. The mass flows are based on a comparison of plutonium burning in LWRs and HWRs [9] and assumes that both reactor types are capable of running with a full core load of MOX fuel. If the LWRs were capable of running with only, for example, a one-third core load of MOX, this would increase the LWR fleet of a factor of three, but would require a dramatic increase in the natural uranium requirements to produce enriched uranium fuel for the remaining two-thirds core load.

Figure 2.8. Comparison of LWRs and HWRs used to burn excess plutonium



2.2.3 Management of minor actinides

There may also be instances where the transition to a nuclear fleet containing fast reactors is driven by a desire to reduce the requirements for spent nuclear fuel disposal capacity. Reducing the requirements for spent nuclear fuel disposal involves the reduction in decay heat from the spent fuel, and in particular, the reduction of the minor actinide content of the spent fuel. The high thermal flux of an HWR makes it an effective platform for reducing the minor actinide content of the spent fuel before a large fleet of fast reactors is available for this purpose [10]. Dedicating an HWR fleet to minor actinide burning would reduce the eventual number of fast reactors required for actinide burning, and also reduce the risks associated with the need to bring a new reactor technology on-line. Including an HWR intermediate burner stage between the LWR and fast reactor fleets to reduce the minor actinide flow to the fast reactors would further reduce the number of fast reactors required to manage the minor actinide inventory in the fuel cycle.

2.2.4 Summary

The fundamental design features of heavy-water-moderated reactors give them unparalleled fuel cycle flexibility. This flexibility, in turn, allows heavy water reactors to play unique roles in the transition from a nuclear fleet consisting only of light water reactors to one that includes fast reactors.

The ultimate success of these transition scenarios may depend on making optimum use of our existing technology and capital investments. Making optimum use of existing technology will involve taking maximum advantage of the abilities of the different reactor types available and exploiting synergies between the various reactor technologies.

2.3 Scenario analysis of Gen-II to Gen-IV systems transition: The French fleet

The current management of spent uranium fuel in the LWR fleet includes direct disposal, temporary storage or processing and recycling of plutonium in the form of MOX fuel. The latter option allows to reduce required storage capacity for the spent fuels for the short term.

In order to eliminate main actinides (plutonium and minor actinides) that represent the long-life radiotoxic component of today’s ultimate wastes (direct disposal or not), a basic and physically optimal scenario (system: reactor and fuel cycle facilities) is proposed, which foresees the optimal use of natural resources and partitioning of MA in the fourth-generation fast neutron reactors, maintaining proliferation resistance and economical competitiveness.

Following a physical analysis of the respective potential of the fast neutron or thermal neutron spectra for transmutation and natural resources use, we analyse scenarios cases from the current fuel cycle of PWRs to a full fourth-generation systems scenario, including recycling stages for all of the actinides: uranium, plutonium and minor actinides.

This section presents a preliminary analysis of the various scenario cases for France, taking into account constraints and inventories in all installations of the fuel cycle (fabrication and processing), including reactors and final disposal.

The fast neutron systems allow global recycling of actinides or optimum use of natural resources by plutonium recycling based on their intrinsic physical characteristics, minimising impacts on the fuel cycle facilities and improving global fuel cycle performances by removing all front-end facilities, this being strongly related to the uranium cost and availability.

2.3.1 Transition scenarios: Proposal for a reference for the future

Objectives

The objectives of the reference scenario and of the alternative scenarios for managing the actinides in the French context can be summarised as follows:

- to reduce the actinide fraction in vitrified waste to minimise the potential radiotoxicity and thermal load, which drives the size of the deep geological repository;
- to use current facilities and installations to their best advantage up to the time of their planned replacement (2030-2040), and to prepare the deployment of future facilities (2040-2100), whether using current technologies or not;
- to prepare for the introduction of fourth-generation FRs (GFR or SFR) systems.

Key steps

To meet these objectives, the following steps were identified as the most important:

- In the frame 2020-2030:
  - *Start of the renewal of 50% of the fleet with EPR reactors*; this renewal relates to the end of the service life of the first PWR plants introduced between 1975-1985 and is carried out, depending on EDF prospects, at the rate of 2 GWe a year.
  - For alternative scenarios to the reference scenario (see later):
    - *Implementation of advanced partitioning* and production of so-called "light" glass matrices, independently of the scenario that is later deployed; creation of a temporary storage solution for minor actinides (Am, Np and Cm, in a mix or separately, depending on the scenario). This implementation can occur at an industrially by adding a workshop to the existing processing facility at La Hague after 2025 or 2040. The date for this study (2020) was chosen before the analysis of industrial optimisation which led to 2025 at the earliest.
    - *Implementation of the advanced processing* of spent MOX fuel to perform a second recycling of plutonium in PWRs, by temporarily storing the minor actinides for later recycling in Gen-IV systems.
- In the frame 2035-2040:
  - *Start of renewal* of the remaining 50% reactors of the previous generation:
    - by fourth-generation fast neutron systems;
    - by EPRs if the fourth-generation systems are not industrially mature by that date.
  - *Implementation of the advanced processing* of spent MOX fuel to recycle the plutonium and minor actinides in the fourth-generation fast neutron systems.
- In 2080:
  - *Start of renewal* of the EPRs which were first introduced in 2020 by fourth-generation FRs.

Analysis of the results of each scenario

- Reference scenario: one Pu recycle in PWR-EPR then recycling in fourth-generation fast neutrons systems:
  - The fuel for a homogeneous recycling situation at equilibrium contains approximately 1.2% of MA (Np + Am + Cm) in the fuel with 20% Pu. However, the absorption of the stock accumulated during the transitional period can be envisioned, with a maximum fraction of the order of 2.5-3% MA in a large SFR up to 5% for a small one (such as Phénix). The introduction of GFR systems capable of accepting a 5% fraction limit would enable increasing the consumption of minor actinides and therefore reducing the inventory in 2100 at a lower level, compared to SFR.
  - The minor actinide inventory is down in 2100 to a level of about 86 tonnes (64 for GFR).
  - The ratio between plutonium and minor actinide inventories starts dropping in 2050. The minor actinide inventories in 2100, after the 100% FR fleet has been put into operation for five years, come very close to the inventories in 2035, when the fourth-generation fast neutrons are first introduced to replace 50% of the fleet over the period 2035-2080.
  - The natural uranium needs are 30-40% less than in the other scenarios.
  - The specific facilities for the cycle of the fourth-generation systems to be introduced as follows:
    - in 2030, for the fuel manufacture;
    - in 2040, for the reprocessing of the fuel in a shielded chain.

A modular reprocessing facility with hydro-metallurgic processes would enable, starting in 2040, to process both the spent UO_2 fuels from the PWRs and the fuels from the fourth-generation systems, and would enable grouped management of the actinides. The current process would be transformed into a GANEX-type process, after partial reduction of the flow of uranium materials. GANEX, still at a prospective stage, could be envisaged using recent results on MA partitioning obtained at ATALANTE in Marcoule. The resulting products would in this case be a set of uranium and transuranium elements for re-use in the manufacture of fuel assemblies to be recycled in the fourth-generation (FR) systems. This modular design is based on the GANEX process which is the topic of a programme of research and experiments.

- *Alternative 1*: One Pu recycling in PWR
  - The Pu and MA (771 t Pu + 264 t MA in 2100) continue to grow continuously, due to the decay of the ^{241}Pu in the ^{241}Am and to the production of minor actinides in the MOX fuel.
  - In the case of a recovery in 2070 of the TRU from the spent fuels available for reprocessing and their introduction into the fourth-generation (GFR or SFR) systems in 2080, the average MA fraction in the GFR (or SFR) fuel is close to 3.4%, which remains below or compatible with allowable content in the FR cores (5% for GFR, 2.5% for a large size SFR, 5% for a small one).

- *Alternative 2*: Multiple recycling of the Pu in EPRs
  - The need for an enriched uranium support for MOX fuel (associated with the degradation of the isotopic vector and the limit of 12% for the fraction of Pu in the fuel) is effective at the third recycling (support with ~1.8% ^{235}U), starting in 2045-2055. Prior to 2040, a support of U_{dep} or U_{nat} type is sufficient.

– The Am and Np inventories increase and differ little in the open cycle, one Pu recycling and multiple Pu recycling options, demonstrating the importance of the ^{241}Pu decay for the production of ^{241}Am.

– In the case of a recovery in 2070 of TRU from the irradiated fuels available for reprocessing and their loading into the fourth-generation GFR systems in 2080, the average MA fraction in the GFR (or SFR) fuel is 2.9%, which remains below or compatible with the allowable content for FR cores (5% for GFR, 2.5% for a large SFR, 5% for a small one).

– The plutonium inventory, stabilised at 2050, will not allow introducing 60 GWe of FR in 2110. Therefore, either EPR reactors will be in the fleet up to 2170, or the Pu recycling has to be stopped in 2060 and UOX burn-up reduced to 42 GWd/tHM from 2060 to 2080 leading to an increased use of natural uranium resources compared to Alternative 1.

Reference scenario vs. alternatives

The partitioning/transmutation scenario implemented in Gen-IV FR in (2025-2040) also allows:

- to minimise the mass (weight) disposed in the final waste at the end of the century, by a factor of 40-50 or more compared to the once-through cycle and by a factor close to 10 compared to a plutonium recycling (in PWR or FR) without minor actinide recycling;

- to minimise the thermal output of the final wastes, allowing a strong and rapid decrease of power with time (Figure 2.9);

- to minimise the potential radiotoxicity inventory (and radioactivity) in the final disposal (Figure 2.10);

- to save natural uranium resources by 40%.

As for the two last items, after several hundred years (300 years), waste activity is below that of the natural uranium extracted to produce the same energy and using the PWR once-through cycle, and the decay heat represents few W/g of waste disposed.

However, the impact of this reduction must still be related to the volume reduction and to the potential increase in capacity of the final waste disposal. This work is still underway and closely linked to the final waste repository design and the site type for the disposal (granite, clay, salt, etc.).

2.3.2 Conclusions

Various recycling modes can be envisioned for the PWRs (EPR) to temporarily stabilise the plutonium inventory, but the fourth-generation fast neutron systems, whose physical characteristics are optimum for transmutation, are essential over the longer term if all the actinides produced by the water reactors have to be managed and recycled.

The prospect of deploying a first series of fourth-generation systems in 2035 bolsters the objective of implementing towards 2020-2030 a system to manage the back end of the PWR cycle with partitioning (and temporary storage) of the minor actinides. If the deployment of the fourth-generation

Figure 2.9. Decay power of the final wastes (actinides + FP)

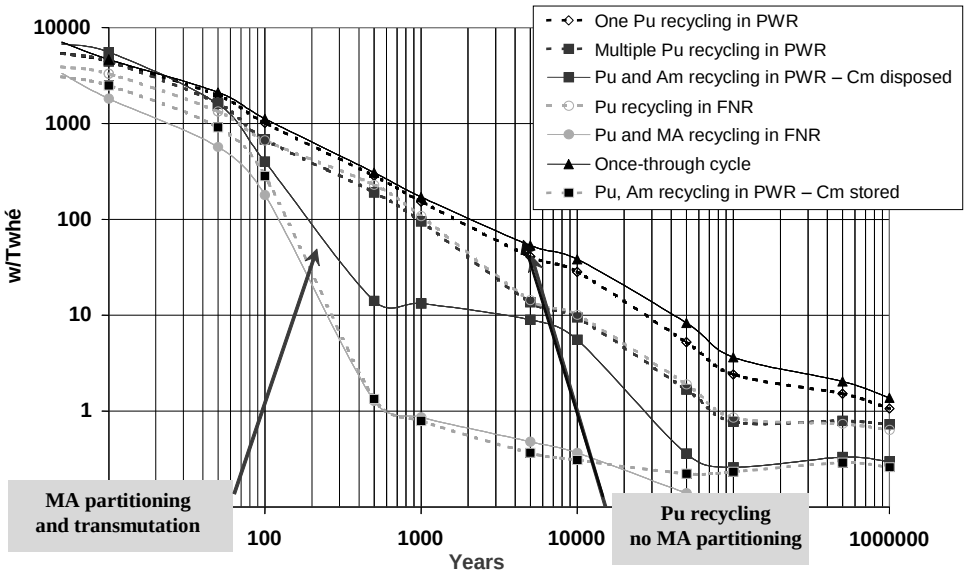
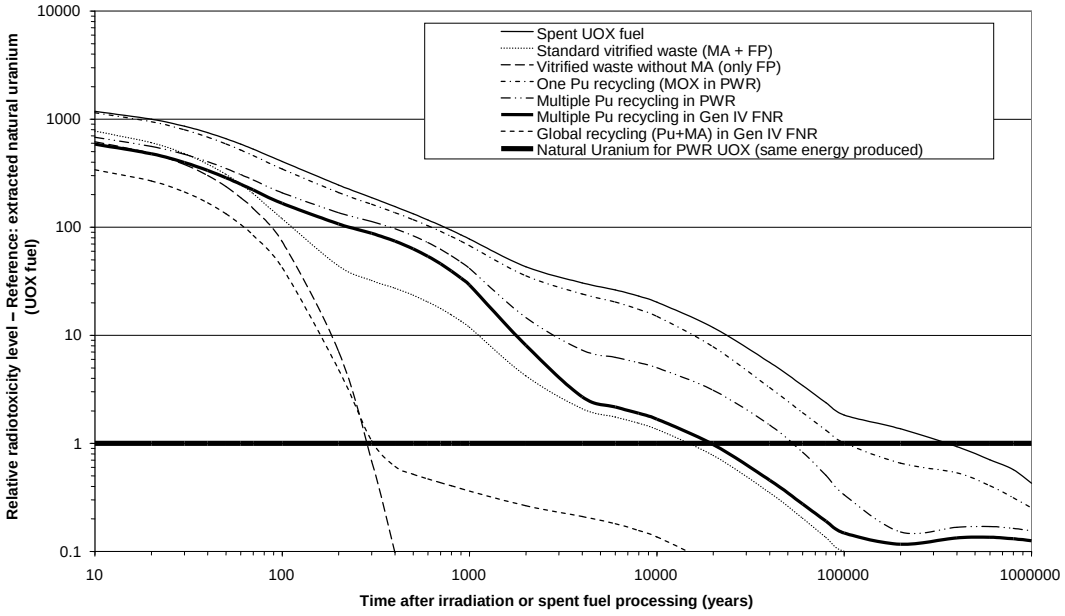


Figure 2.10. Radiotoxicity level of the TRU disposed in the storage



Spent UOX fuel: Direct disposal of the irradiated fuel.

Standard vitrified waste: Glasses with MA and PF from the UOX spent fuel processing (as produced today at La Hague facility).

Vitrified waste without MA: Standard vitrified waste (see upper) but without any MA (only FP from the UOX spent fuel processing).

One Pu recycling: All TRU after single Pu recycling in PWR.

Multiple Pu recycling in PWR: MA and FP from the UOX and MOX spent fuel processing in case of a scenario with multiple Pu recycling in PWR.

Multiple Pu recycling in FR: MA and FP from the FR spent fuel processing in case of a scenario with multiple Pu recycling in FR.

Global recycling (Pu+MA) in Gen-IV FR: FP from the FR spent fuel processing in case of a scenario with multiple Pu and MA recycling in FR.

systems is delayed, the preceding strategy would still be possible and would offer all the same advantages, because of the capability of the fast neutron systems to eventually recycle the transuranium elements produced by the PWRs through the end of the 21st century (with, however, increasing restrictions relating to the accumulation of minor actinides due to the aging of the nuclear materials and possible multiple recycling processes in the PWRs).

The increasing difficulty involved in recycling plutonium and efficiently burning up all the minor actinides in the PWRs under quite realistic economical and industrial conditions, should favour the deployment, around the middle of the 21st century, of a first series of fast neutron systems to manage the actinides produced by the PWR fleet.

FRs can also allow saving up to 40% of the consumed natural uranium during the 21st century in the French context and would not require any use of uranium enrichment technologies at the end of the century.

Table 2.4. Inventories in the fuel cycle for scenarios with PWRs

Inventories (t)	One recycling Pu (MOX) *Alternative 1*			Multiple recycling Pu (MOX-EU) *Alternative 2*			Once-through cycle (UOX)			
	2035	**2050**	**2070**	**2035**	**2050**	**2070**	**2035**	**2050**	**2070**	**2100**
Natural U (annual values/ aggregates)	7 400/ 410×10^3	7 500/ 520×10^3	7 500/ 670×10^3	7 160/ 410×10^3	7 100/ 520×10^3	7 000/ 660×10^3	8 360/ 420×10^3	8 360/ 550×10^3	8 360/ 720×10^3	8 360/ 970×10^3
UTS (annual, M SWU/yr)	5.8	5.8	5.8	5.3	5.1	5.1	6.4	6.4	6.4	6.4
Pu (Total)	396	479	596	373	398	400	474	612	793	1062
MA (Total)	76	120	178	76	125	191	99	138	191	271
% fuel with TRU in fleet	12%	10%	10%	23%	26%	33%	0%	0%	0%	0%
TRU in storage	389	529	703	128	175	240	573	750	984	1333

The inventory values in this table have been rounded up or down to the first significant figure, after summation, except for Cm, rounded up or down to the nearest decimal.

Table 2.5. Inventories in the fuel cycle for scenarios with FRs

Inventories	One recycling of MOX in PWR and Pu recycling in fourth-generation FR system (SFR), MA disposed in storage				One recycling of MOX in PWR and global multiple recycling (Pu, Np, Am, Cm,…) in fourth-generation FR system (SFR)				One recycling of MOX in PWR and global multiple recycling (Pu, Np, Am, Cm,…) in fourth-generation FN (GFR) system			
	2035	**2050**	**2070**	**2100**	**2035**	**2050**	**2070**	**2100**	**2035**	**2050**	**2070**	**2100**
Natural U (annual values/ aggregates)	7 850/ 430×10^3	4 200/ 510×10^3	4 200/ 600×10^3	0/ 660×10^3	7 850/ 430×10^3	4 200/ 515×10^3	4 200/ 600×10^3	0/ 660×10^3	7 850/ 430×10^3	4 200/ 515×10^3	4 200/ 600×10^3	0/ 660×10^3
UTS (annual, M SWU/yr)	5.9	3.3	3.2	0	6	3.2	3.2	0	6	3.2	3.2	0
Pu (total)	450	566	672	802	455	576	685	848	455	577	698	815
MA (total)	70	106	149	205	76	96	105	86	76	89	76	64
% fuel with TRU in fleet	0%	50%	50%	100%	0%	50%	50%	100%	0%	50%	50%	350
TRU in storage	65	103	149	208	27	28	29	30	27	28	29	30

2.4 German strategies for transmutation of nuclear fuel legacy to reduce the impact on deep repository[2]

2.4.1 Nuclear power in Germany: Background and current status

In 2005 German electricity demand totalled 576 TWh. Three national nuclear power companies RWE, E.ON (created with the fusion between VEBA and VIAG) and EnBW operated 19 nuclear power plants. These 19 units produced a total of 29% of German electric power. Nuclear power thus remains the most important energy source, followed by brown coal (26%) and hard coal (21%). Due to the phase-out decision of the German government and the shutdown schedule agreed upon with the German utilities, the nuclear power plants at Stade and Obrigheim were to be turned off on 14 November 2003 and 11 May 2005, respectively. The plants' dismantling was scheduled, however, to begin in 2007. No externality pertained to the economics of German nuclear power since it became cost effective (no further subsidies by German government as was the case in the past). At present, the key externality that may pertain to rethinking of nuclear energy growth is a necessity to reduce fossil fuel consumption and the implementation at the national level of carbon dioxide emission controls that had been agreed upon during the 1998 world climate conference in Kyoto.

Siemens AG (the third largest German company) produced all 19 German NPPs and has provided security upgrades since then. Today, the German reactor fleet consists of 11 pressurised water reactors (PWRs) and 6 boiling water reactors (BWRs). The fleet is subject to the German "Nuclear Phase-Out Law", and is thus slated to retire by 2021. The Consensus Agreement between the utilities and the government is based on calculations which assume a 32-year average operating lifetime for each NPP. The agreement specifies a target energy production for each power plant to reach before shutdown. The Consensus Agreement permits, however, a flexibility on residuals which can be redistributed between nuclear power plants in operation (but in principle only older to more modern units). Up to now, two of the power utilities (RWE and EnBW) have applied for lifetime extensions for two NPPs.

Transport and reprocessing of spent nuclear fuel ceased in 2005. Decentralised interim storage facilities were constructed at the sites of German NPPs to store spent fuel elements until final disposal. Between the year 2000 and the time at which of the use of nuclear power in Germany is fully terminated, an additional 8 000 t of irradiated fuel elements will be discharged from the various NPPs. This amount includes the respective final core loads. Of roughly 17 000 t of irradiated fuel elements, about 57% were reprocessed, while 43% will have to be put into final storage as spent fuel elements. Vitrified high-active waste from the reprocessing of German SFE has to be returned to Germany from abroad. In the spring of 2001, there were 9 CASTOR casks holding 28 vitrified waste canisters, each located in the Gorleben transport cask store. When all contracts with COGEMA and BNFL are fulfilled and the HLW from reprocessing of the WAK facility is vitrified, a total of 305 CASTOR casks holding 28 vitrified waste canisters will have to be put into interim storage and eventually – after several decades of radioactive decay – into a repository [11].

2.4.2 National scenario studies: Rationale and objectives

Long-lived radionuclides of spent nuclear fuel and the question whether it can be ensured over the long term that no release of radioactive substances disposed in underground repository will occur, for instance under an intrusion scenario assumption, motivate national R&D studies searching for alternatives. The most promising option is partitioning and transmutation (P&T), which however

[2] Portions of this section were performed in collaboration with Massimo Salvatores (CEA), Erich Schneider (LANL) and H.W. Wiese (FZK). The NFCSim code developed at LANL was used to simulate the fuel cycles.

requires the separation of some of the high-level radioactive and long-lived transuranic (TRU) isotopes (high-level waste – HLW) from the spent nuclear fuel and converting them into stable or short-lived fission products. A similar strategy could be applied to long-lived and radiotoxic fission products. For this purpose, dedicated facilities must be deployed in which separated isotopes could be converted by neutron-induced reactions (fission, capture) reducing their long-term hazard [12]. In the early 90s, accelerator-driven subcritical transmuters (ADS) were proposed as systems potentially suitable for very efficient transformation of TRU such as plutonium and the minor actinides (neptunium, americium and curium).

The benefit of a particular P&T strategy can only be assessed by performing extensive scenario studies on the entire fuel cycle. Given the strongly time-dependent nature of the national nuclear economy, it is often desirable to look beyond a static or quasi-equilibrium paradigm when considering the course that might be taken in the future. While steady-state analyses of nuclear fuel cycles can provide vital policy guidance in that they can show whether the mature state of a proposed nuclear economy is a desirable one, they cannot take into account real-world initial conditions or time-dependent variations in deployment strategies, nor do they take into account the time required to move from the current reactor fleet configuration to the equilibrium state. In fact, in many cases this time interval is so great that the eventual, equilibrium reactor fleet configuration is itself immaterial to short-term policy decisions.

The present analysis, then, focuses upon a suite of scenarios that are evidently poorly portrayed by a steady-state analysis. The modelling tool deployed to analyse these scenarios, NFCSim [13], was developed at Los Alamos National Laboratory (LANL). This software tool tracks nuclear materials from mining to disposal, incorporating elemental and isotopic transformations following from decay or irradiation. In addition to depicting the evolving stockpile of nuclear materials, NFCSim computes quantities such as the time-dependent location and mass throughput, radiotoxicity, and decay heat production rate of nuclear materials. These are chosen based upon their relevance to the economics, proliferation resistance, resource utilisation and ease of waste disposal for a fuel cycle.

The first of the time-dependent scenarios studied with NFCSim addresses the German reactor fleet, with the aim of characterising the final spent nuclear fuel (SNF) inventory when the nuclear fleet is retired. Where available, historical data from public sources was used to define Germany's 19 reactors. Where data was not available, for instance regarding the time-dependent mixed-oxide (MOX) core fraction employed by MOX-capable reactors, estimates that led to accurate reproduction of known SNF inventories were employed. The performance of the fleet from the present day through the retirement of Germany's final reactor in 2021 was estimated based upon present trends in the United States and Germany.

In the second scenario, then, it was postulated that Germany address its SNF inventory by pursuing an accelerator-driven system (ADS) based on a partitioning and transmutation strategy. The initial conditions used for this scenario were those generated for the final German SNF inventory. This R&D programme is expected to yield substantial reductions in the medium- and long-term decay heat production rate of nuclear material, even if it might offer nearly zero short-term benefit when compared to allowing natural decay to take its course.

The above strategy suggests that Germany follow an independent path in resolving its respective waste issues of a growing stockpile of stored MA and an inventory of SNF for which no disposal facility currently exists, respectively. This dedicated facility would employ ADSs to transmute the TRU feed stream. The feed streams are especially amenable to ADS transmutation since accelerator-driven systems operate best (highest availability, greatest per-pass transmutation rate, least number of facilities required) when their feed is constituted of roughly half plutonium and half MA. Indeed, the ADS is not

the only tool that can fulfil the scenario goals; options including, for instance, LWR-based transmutation in traditional or inert matrices and/or use of a Generation IV FR in place of the ADS, might be explored in the future.This document outlines the results of scenario studies conducted at Forschungszentrum Karlsruhe (FZK) using the NFCSim nuclear fuel cycle simulation software [14]. NFCSim tracks the progress of nuclear materials through the fuel cycle. Its embedded burn-up and criticality engines, ORIGEN 2.2 and LACE, respectively, support a diverse suite of reactor technologies and fuel cycle strategies; in this study, for instance, mixed-oxide (MOX) burning BWRs and PWRs as well as accelerator-driven systems (ADS) were closely studied.

The first objective of the study was to characterise, in an approximate fashion, the size and content of the spent fuel (SF) inventory that will ultimately be produced by the German reactor fleet. Given the published retirement schedule, the behaviour of the fleet from the present day through decommissioning of the final reactor can be estimated based upon extrapolation of current trends.

To lend consistency of methodology to the analysis, the historical characterisation of the German fleet was also carried out using NFCSim. Hence, the entire simulation, from the first delivery of electric power from Obrigheim in 1969 through the decommissioning of Neckar-2 in 2021 was carried out in a single calculation. Rather than undertaking to re-create each individual cycle for every reactor – for which supporting data were scanty and difficult to locate – key parameters such as load factors, fuel discharge burn-ups and cycle times were treated such that their fleet-average values approximated published realities.

The treatment of MOX fuel loading in German reactors also presented a challenge. Data regarding MOX loadings – the fraction of reloaded assemblies that were MOX and the plutonium content of that MOX, for instance – for individual cycles was not available. Hence, given that the time intervals during which specific reactors burned MOX was available [15], as was the licensed MOX fraction for each reactor, MOX use was estimated in the spirit described above. This estimate was guided by published data regarding the amount of German SF that had been reprocessed at facilities in France.

Given that the central result of the work is an isotopic-level characterisation of the German SF, a logical follow up to this work might address incorporation of this SF into a next-generation fuel cycle. Waste management strategies, for instance those making use of partitioning/transmutation technologies, imply the development of new dedicated installations for the fuel cycle, thus, the second objective of this work is to illustrate the degree to which ADS could contribute to mitigating the burden of SNF disposal.

2.4.3 Case I: Assessment of German spent fuel legacy

The primary aim of this work is to approximately characterise the isotopic content of all SNF discharged from the German reactor fleet. This includes historical arisings, *i.e.* fuel that has already been discharged. Hence, the analysis performed with the NFCSim code commences with the first criticality of the Obrigheim reactor in 1969. In 2005 Germany possessed 19 power reactors; of these, two (Obrigheim and Stade) have recently ceased operation. MOX fuel has been used in Germany since 1980, though its prevalence has not reached the level observed in France.

Characterisation of German reactor fleet

Under the current German phase-out law,[3] all reprocessing of SNF must cease by 2005. The law also commits Germany to phasing out nuclear power; the decommissioning schedule to be followed by the reactor park is specified. The study is carried out under the assumption that Germany will proceed with this phase-out, decommissioning its final reactor, Neckar-2, by 2021.

NFCSim groups fuel batches by type; batches of a given fuel type are subject to the same rules governing fuel cycle decisions such as reprocessing. Four fuel types are used here: PWR-UOX, PWR-MOX, BWR-UOX and BWR-MOX. As already mentioned, it was decided to simulate the fleet for the entire time period, 1969 through 2021, rather than commencing from the present day. There are two reasons for this. First, although some published data regarding current German SNF inventories exist, this data is not comprehensive: it does not offer sufficient detail regarding isotopic composition, nor does it adequately discriminate between fuel types. Second, since the data that is available concern aggregate SNF inventories, simulating the historical behaviour of the reactor fleet with the aim of reproducing these inventories affords a good opportunity for benchmarking of the reactor fleet parameterisation used in NFCSim.

The characterisation of the reactor fleet requires that a set of top level parameters be gathered for each facility. While some data (*e.g.* thermal power, core inventory, start-up and planned shutdown dates, periods during which MOX capable reactors burn MOX) are available in full, other information (discharge burn-ups, up and down times for each cycle, the core fraction of MOX employed by the MOX-capable reactors) is not.

Assumptions

Source data for all facilities has been compiled from the literature and is given in Table 2.6. The data shown in the table duplicates the NFCSim input file used for the analysis. Much of this data – start-up and shutdown dates, power, core inventory, the number of batches per core, the enrichment of the uranium matrix used when fabricating MOX, the dates of MOX utilisation – is straightforward to obtain. Even these simple data contain some subtleties, however. Given the lack of comprehensive burn-up data, MOX parity was assumed throughout.

Where data is missing, assumptions or approximations are made. Some of these assumptions are embedded in the data of Table 2.6. Perhaps the most significant of these involves the utilisation of MOX fuel. Data concerning MOX use was drawn from Ref. [13]. The information provided included the intervals during which reactors burned MOX as well as the maximum MOX core fraction for which each facility was rated. The enrichment of the uranium carrier – natural uranium or depleted uranium with 0.25% ^{235}U content – for the MOX was also provided. While the plutonium fraction in MOX as of 2000 was given for each reactor, historical and present-day information concerning the number of MOX FAs that were in fact loaded was not provided. Given that 4 000 tIHM of German UOX SNF was reprocessed by 2000, it is easy to show that the MOX burning reactors could not have been operating at their full, licensed MOX fractions.

[3] For a summary of the 2002 Bundestag Act see: Vorwer, A., "The 2002 Amendment to the German Atomic Energy Act Concerning the Phase-Out of Nuclear Power", *IAEA Nuclear Law Bulletin*, 69.

Table 2.6. The German reactor fleet: Input parameters

Name	Type	Power [MWt MWe]		Start-up	Shut-down	Inventory [tonne IHM]	Burn-up* in 1990 [MWd/kg][a]	Load factor* in 1990[b]	Batches/core UOX	MOX	MOX matrix[c]	Max.[d] MOX frac. [%]	MOX use [Time period/ MOX fraction][e]
BIBLIS-A	PWR	3 517	1 146	2/75	3/07	102.7	31.5	72	3				
BIBLIS-B	PWR	3 752	1 240	1/77	2/09	102.7	32.9	75	3				
BROKDORF	PWR	3 989	1 370	12/86	12/19	103.7	32.2	83	4	4	NU	17	88-05/17
BRUNSBUETTEL	BWR	2 292	771	2/77	2/09	91.5	27.5	75	6				
EMSLAND	PWR[f]	3 962	1 290	7/88	6/20	102.9	32.2	85	4				
GRAFENRHEINFELD	PWR	3 899	1 275	6/82	6/14	103.6	34.1	78	4	4	DU	33	85-00/20; 00-06/33
GROHNDE	PWR	3 961	1 360	2/85	2/17	103.5	34.0	85	4	4	NU	33	88-05/20
GUNDREMMINGEN-B	BWR	3 941	1 284	7/84	8/16	136.4	30.0	80	6	4	NU	38	97-00/19; 00-05/38
GUNDREMMINGEN-C	BWR	3 941	1 288	1/85	2/17	136.4	30.0	80	6	4	NU	38	96-00/19; 00-05/38
ISAR-1	BWR	2 575	870	3/79	3/11	103.0	27.8	83	4				
ISAR-2	PWR[f]	3 782	1 285	4/88	4/20	101.4	32.2	82	3	3	DU	40	99-06/20
KRUEMMEL	BWR	3 690	1 260	3/84	3/16	156.0	35.0	75	4				
NECKAR-1	PWR	2 510	810	12/76	11/08	63.1	31.0	83	3	3	NU	9	82-92/9; 98-05/9
NECKAR-2	PWR[f]	3 765	1 230	4/89	4/21	103.0	35.0	85	3	3	NU	37	82-92/20; 98-05/30
OBRIGHEIM	PWR	1 050	340	4/69	12/03	34.0	30.0	82	3	3	NU	26	80-91/15; 98-05/26
PHILIPPSBURG-1	BWR	2 575	864	2/80	6/12	115.0	27.0	81	4				
PHILIPPSBURG-2	PWR	3 765	1 268	4/85	5/17	103.0	34.0	84	3	3	DU	50	89-05/20
STADE	PWR	1 900	630	5/72	5/04	56.2	31.5	80	3				
UNTERWESER	PWR	3 733	1 230	9/79	9/11	103.4	31.5	75	3	3	DU	50	84-02/20; 02-05/35

* These quantities evolve. The 1990 values only are shown. See text for discussion.

[a] MOX parity assumed.

[b] Obtained by averaging three annually-reported load factors.

[c] DU = depleted uranium, NU = natural uranium.

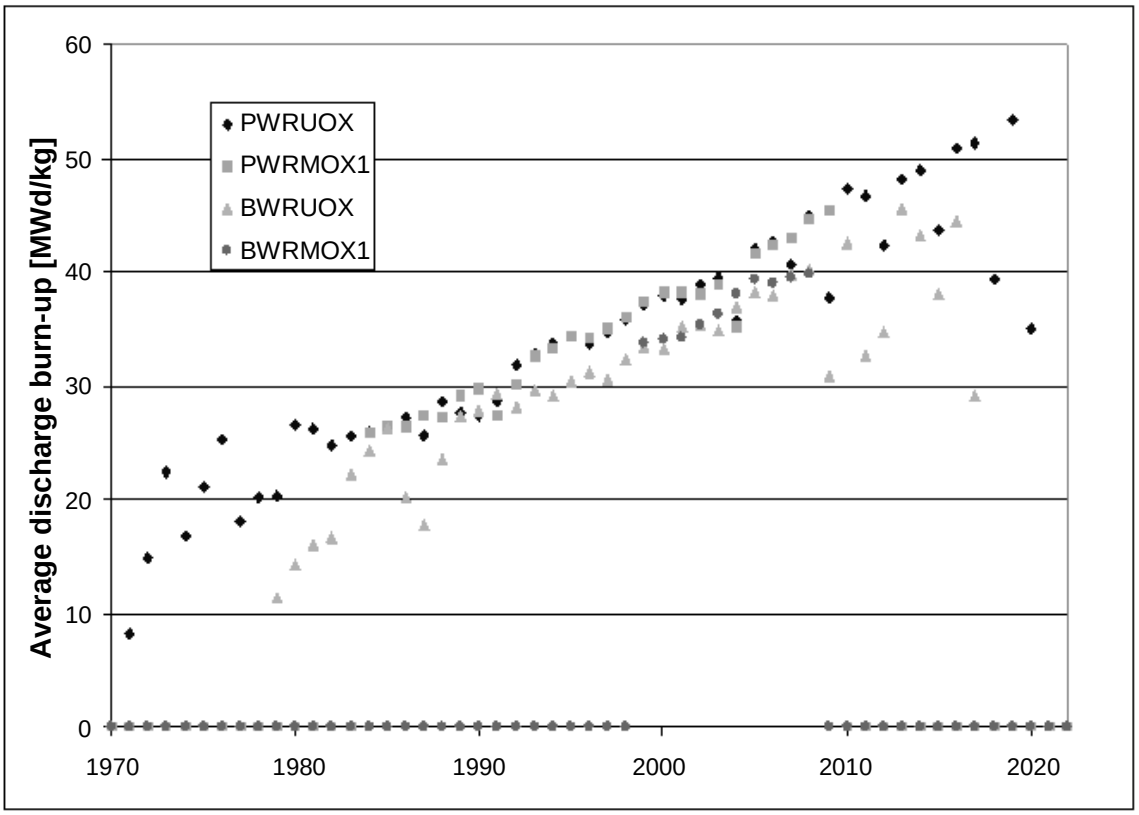
[d] This is the maximum licensed MOX fraction when available; when not, it is the maximum observed in practice.

[e] Defined as the MOX fraction by mass of reloads occurring during this time.

[f] PWR of Convoy type.

Time-dependent fuel burn-ups and residence times, along with reactor availabilities, constitute another important set of inputs. The burn-up data therein were used as reference values; however it was noted that they seemed low (the reference gave fleet averaged burn-ups for PWRs as 33 MWd/kg and BWRs as 28 MWd/kg. The comparable United States values for 1992, obtained from the Energy Information Administration (EIA), were 38 and 31 MWd/kg. Load factors were similarly lower than prevailing United States figures at this time. The burn-up trajectory, which is an input to the model, is shown in Figure 2.11. Note that the averages shown in the figure include "transient" discharges – those associated with reactor start-up or shutdown. Further on, in this simulation, it was assumed that discharge burn-ups increase by 9% every five years after 2000, in keeping with historical trends. After 2000, the refuelling outage time was allowed to decrease by 5% every five years.

Figure 2.11. Average discharge burn-up for NFCSim German reactor fleet model



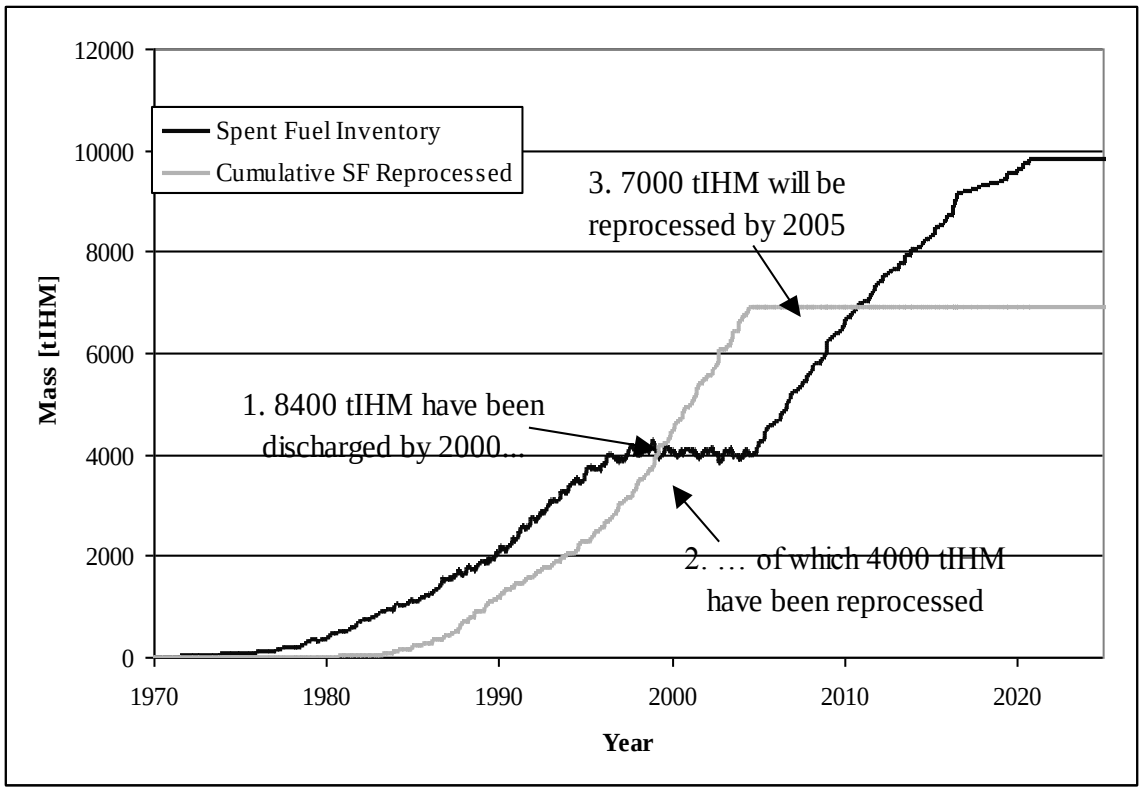
Results

Three temporal data points describing the performance of the German reactor system must be matched by the NFCSim results. These are:

1. The total amount of SF discharged from the fleet by 2000 was 8 400 tIHM.
2. Of this, 4 000 tIHM had been reprocessed.
3. At the time reprocessing ceases in 2005, 7 000 tIHM will have been reprocessed.

As described in the previous section, MOX utilisation by individual reactors is adjusted so that these stipulations are met. Figure 2.12 shows the aggregate inventory of discharged unreprocessed SF as well as the cumulative amount of UOX fuel reprocessed. The three conditions mentioned above are indicated in the figure.

Figure 2.12. Spent fuel inventory and integrated reprocessing throughput for German fleet



It can be seen that, given the current trajectory of MOX use and reactor retirement, the final SF inventory in 2022 will be 9 840 tIHM. A detailed breakdown by fuel type of the composition of this SF is given in Table 2.7. It can be seen that the SF will contain 127 tonnes of plutonium at that time; note that this compositional data reflects the decay of each fuel batch for the appropriate amount of time following its discharge. HLW having been vitrified is also given in the table. Fission products constitute the bulk – 96.5% – of this waste. The trace actinides present follow from the assumed 99.8% recovery efficiency of all transuranics. Uranium is recovered in a separate stream, composition not shown here, at 99.99% efficiency.

In addition to the individual and aggregated material balances presented above, NFCSim derives a number of quantities related to the disposability, proliferation resistance and radiotoxicity of the various waste forms. These are presented in Table 2.8; both totals and per-tonne values are given.

The German reactor fleet is thus characterised in an approximate sense. Subsequent sections of this report address a transmuting fuel cycle as applied to German SNF inventories and waste arising.

Table 2.7. Inventories (tonnes) of German SNF and HLW as of 1 January 2022

Quantity	PWR-UOX	PWR-MOX	BWR-UOX	BWR-MOX	**Tot. SF**	HLW
Total	5.35E+03	7.73E+02	3.47E+03	2.46E+02	**9.84E+03**	2.15E+02
U	5.06E+03	7.02E+02	3.31E+03	2.27E+02	**9.29E+03**	6.64E-01
Pu	5.17E+01	3.43E+01	3.29E+01	7.95E+00	**1.27E+02**	2.01E-01
Np	3.60E+00	2.34E-01	2.16E+00	4.97E-02	**6.04E+00**	2.94E+00
Am	4.60E+00	4.96E+00	3.48E+00	1.17E+00	**1.42E+01**	3.63E+00
Cm	2.30E-01	2.26E-01	1.48E-01	6.44E-02	**6.69E-01**	7.36E-02
FP	2.34E+02	3.04E+01	1.29E+02	9.69E+00	**4.03E+02**	2.08E+02
^{234}U	1.47E-01	1.45E-01	1.20E-01	3.30E-02	**4.45E-01**	3.86E-04
^{235}U	4.56E+01	1.85E+00	3.29E+01	7.84E-01	**8.12E+01**	5.22E-03
^{236}U	2.59E+01	3.67E-01	1.43E+01	1.79E-01	**4.07E+01**	2.43E-03
^{238}U	4.99E+03	7.00E+02	3.26E+03	2.26E+02	**9.17E+03**	6.56E-01
^{238}Pu	1.26E+00	7.56E-01	8.32E-01	2.00E-01	**3.04E+00**	2.25E-03
^{239}Pu	2.94E+01	1.62E+01	1.99E+01	2.95E+00	**6.85E+01**	7.64E-02
^{240}Pu	1.32E+01	1.15E+01	7.83E+00	3.08E+00	**3.56E+01**	1.12E-01
^{241}Pu	4.30E+00	2.34E+00	2.39E+00	6.56E-01	**9.69E+00**	4.01E-03
^{242}Pu	3.48E+00	3.46E+00	1.94E+00	1.06E+00	**9.94E+00**	6.42E-03
^{237}Np	3.60E+00	2.34E-01	2.16E+00	4.97E-02	**6.04E+00**	2.94E+00
^{241}Am	3.74E+00	4.27E+00	2.94E+00	9.28E-01	**1.19E+01**	2.97E+00
^{242m}Am	5.31E-03	1.03E-02	1.04E-02	1.58E-03	**2.76E-02**	8.25E-03
^{243}Am	8.58E-01	6.79E-01	5.35E-01	2.46E-01	**2.32E+00**	6.47E-01
^{242}Cm	3.49E-04	2.50E-05	2.58E-05	3.85E-06	**4.04E-04**	2.00E-05
^{243}Cm	2.57E-03	2.37E-03	1.76E-03	6.60E-04	**7.35E-03**	1.39E-03
^{244}Cm	2.09E-01	1.81E-01	1.32E-01	5.62E-02	**5.78E-01**	6.31E-02
^{245}Cm	1.59E-02	4.05E-02	1.26E-02	6.63E-03	**7.57E-02**	8.21E-03
^{246}Cm	2.12E-03	2.35E-03	1.57E-03	9.11E-04	**6.95E-03**	8.20E-04
^{135}Cs	2.72E+00	6.06E-01	1.86E+00	1.25E-01	**5.31E+00**	2.16E+00
^{137}Cs	6.09E+00	6.72E-01	3.08E+00	2.33E-01	**1.01E+01**	3.53E+00
^{90}Sr	2.65E+00	1.49E-01	1.30E+00	5.37E-02	**4.16E+00**	1.46E+00
^{99}Tc	5.21E+00	6.92E-01	2.85E+00	2.18E-01	**8.97E+00**	4.74E+00
^{129}I	1.23E+00	2.16E-01	6.86E-01	6.53E-02	**2.20E+00**	1.14E+00

Table 2.8. Properties of German SNF and HLW

Evaluated at the beginning of 2022 unless otherwise noted

	PWR-UOX (total/*per tonne*)	**PWR-MOX (total/*per tonne*)**	**BWR-UOX (total/*per tonne*)**	**BWR-MOX (total/*per tonne*)**	**All SNF (total/*per tonne*)**	**HLW (total/*per tonne*)**
Alpha activity [MCi]	49.9/*9.34E-03*	46.3/*6.00E-02*	38.5/*1.11E-02*	12.2/*4.95E-02*	**147.0/*1.49E-02***	15.7/*7.29E-02*
Gamma decay power [MW]	2.14/*4.00E-04*	0.23/*3.03E-04*	1.27/*3.66E-04*	0.08/*3.33E-04*	**3.72/*3.79E-04***	1.22/*5.67E-03*
Spont. fission neutrons [$\times 10^9$ n/s]	1974/*3.69E-01*	2055/*2.66E+00*	1502/*4.33E-01*	639/*2.60E+00*	**6170/*6.27E-01***	711/*3.31E+00*
Decay power in 2026* [MW]	6.95/*1.30E-03*	1.92/*2.48E-03*	3.67/*1.06E-03*	0.54/*2.19E-03*	**13.07/*1.33E-03***	3.13/*1.46E-02*
Decay power in 2122 [MW]	1.76/*3.29E-04*	1.01/*1.31E-03*	1.08/*3.12E-04*	0.25/*1.03E-03*	**4.11/*4.18E-04***	0.57/*2.66E-03*
Decay heat integral** [MW-yr]	849/*1.59E-01*	649/*8.40E-01*	548/*1.58E-01*	158/*6.43E-01*	**2205/*2.24E-01***	203/*9.43E-01*
Inhalation radiotoxicity*** [m^3 air to dilute to RCG]	4.14E+19/*7.74E+15*	2.88E+19/*3.72E+16*	2.65E+19/*7.64E+15*	6.70E+18/*2.72E+16*	**1.03E+20/*1.05E+16***	9.30E+17/*4.33E+15*
Ingestion radiotoxicity*** [m^3 water to dilute to RCG]	5.17E+11/*9.67E+07*	3.59E+11/*4.65E+08*	3.30E+11/*9.52E+07*	8.50E+10/*3.46E+08*	**1.29E+12/*1.31E+08***	2.88E+10/*1.34E+08*

* Evaluated at 2026 rather than 2022 to allow short-lived nuclides from recently discharged batches to decay.

** Integral of decay power over 1 900 year period commencing in 2122.

*** Long-term radiotoxicities: evaluated from concentrations following 10 000 year decay.

2.4.4 Case II: Partitioning and ADS-based transmutation of German spent fuel

The purpose of this scenario is to illustrate the degree to which accelerator-driven systems could contribute to mitigating the burden of SNF disposal for the German fleet. We wish to emphasise that this scenario is hypothetical and can be generalised to other nations with nuclear economies broadly similar to that assumed for this study.

For this scenario, then, an ADS park is deployed beginning in 2030. The ADS park is sized such that all *German* LWR SNF is reprocessed during the 40-year lifetimes of the ADS. Subsequently, a smaller fleet of "second-generation" ADS are deployed following the retirement of the first-generation facilities. Hence, the simulation commences in 2030 and extends approximately 100 years, covering two facility lifetimes. The progress made in reducing actinide inventories in 2100 as well as upon retirement of this second generation is assessed.

Assumptions

The ADS is a Na-cooled, metal-fuelled facility with an LBE target, the same design as was used in earlier AFCI/AAA scoping studies [16]. Table 2.9 provides a summary of parameters used for this facility and its associated fuel cycle. Note that facility availability is assumed to be 85%. The actinide to zirconium ratio in the fuel was adjusted to achieve the desired k_{eff} at BOC. The non-leakage probability was treated as a calibration parameter; it was adjusted such that the model arrived at Ac:Zr ratios in line with results presented in Refs. [16] and [17]. ADS fleet size is determined by the amount of material available for transmutation: the fleet must be of sufficient size to take up, as nearly as possible, the entire SNF inventory during the lifetimes of the first generation of transmuters. Hence, eight 840 MWt facilities were deployed in the first generation and three in the second.

Table 2.9. Top-level ADS design parameters

Target k_{eff}	0.97 (BOC); 0.94 (EOC)
Core inventory	3 000 kgIHM
Thermal power	840 MWt
Discharge burn-up	200 MWd/kg
Fuel management	5 batches/core
Cycle time	168 days (142.9 efpd)

Within each of the four spent fuel types produced by the German fleet (PWR-UOX and MOX, BWR-UOX and MOX), an oldest-first reprocessing strategy was pursued. The MOX fuel was recycled first, for two reasons. First, spent MOX yields about six times more TRU per kgIHM reprocessed, reducing the mass flow through the reprocessing facility in the early years of the transmutation programme. Second, the higher MA content of spent MOX represents a better quality feed stream for the ADS than that of spent UOX.

Note that MA arising from reprocessing of UOX fuel have been assumed to be vitrified, rather than stored for future transmutation; hence no MA top-up is available. This aggravates a difficulty inherent in this strategy: since plutonium constitutes ~85% of the TRU contained in SNF, the ADS used to transmute that TRU must necessarily employ relatively short cycles. In fact, it was found that the relatively steep burn-up reactivity gradient resulting from use of the TRU inventory limited the ADS cycle burn-up to 40 MWd/kg (with a reactivity swing $\Delta k_{eff} = 0.03$) and cycle time to slightly less than half a year.

Results

As mentioned above, the ADS deployment schedule is a result created by the scenario assumptions. If the scenario objective is to incorporate, to the extent possible given that transmuters are built in discrete increments of 840 MWt, all SNF into the transmuting fuel cycle within one facility lifetime, the power density of the transmuting system largely determines the required size of the fleet. With 40-year facility lifetimes assumed, this was found to be eight transmuters. Similarly, the second and subsequent generations of transmuters take the final discharges of the previous generation as their feed. Since just over half of the transuranic content of *German* SNF was transmuted by the first generation of eight facilities, the remaining TRU support another generation of three transmuters. Deployment scheduling was not optimised in this study; rather, members of the first generation of transmuters were deployed every 18 months (see Figure 2.13). This deployment rate is in line with that pursued in the only available time-dependent study of ADS park deployment [8].

Figure 2.14 shows time-dependent SNF inventories, and Figure 2.15 illustrates reprocessing throughput. All German SNF is reprocessed during the lifetimes of the first generation of eight ADS, in the order described earlier. The second generation of ADS obtains its feed exclusively from the final discharges of the first ADS generation. The sharp peak in oxide reprocessing throughput in the late 2030s follows from exhaustion of the relatively high-yield MOX SNF. In the NFCSim model, a just-in-time reprocessing strategy was pursued. Realistically, reprocessing of SNF assemblies could be scheduled so that actual throughput would be limited to 200 tIHM/year.

Figure 2.13. ADS deployment schedule for transmutation of German SNF

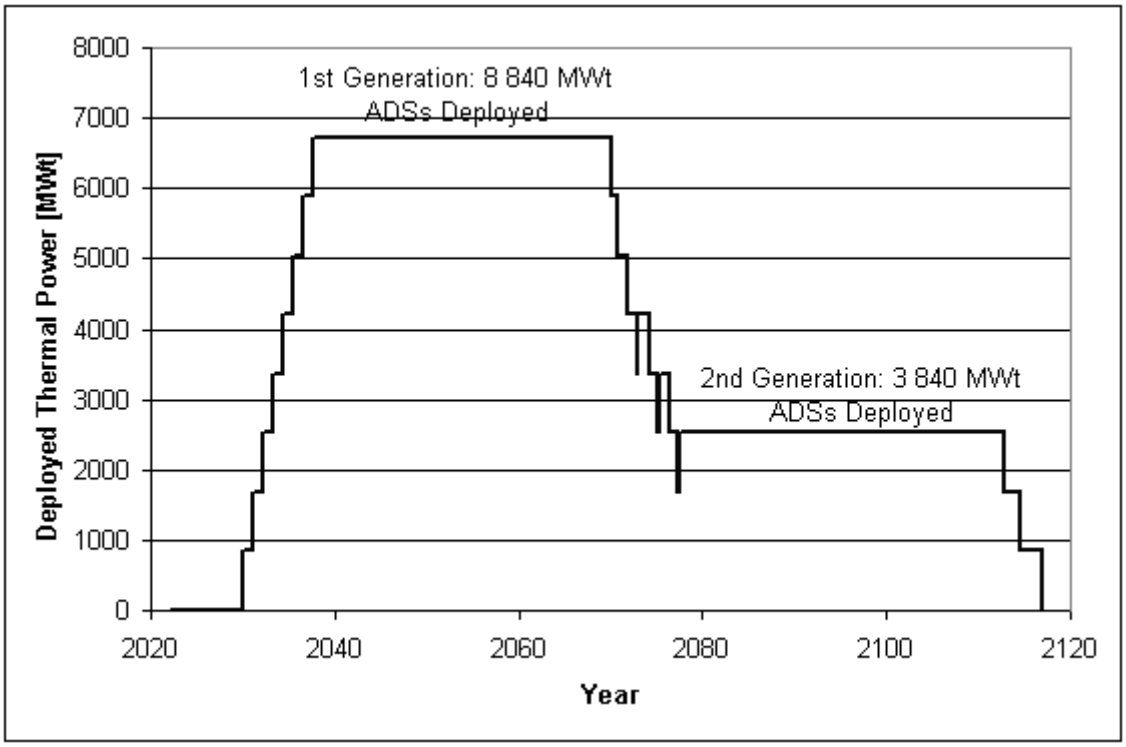


Figure 2.14. German spent fuel inventory showing just-in-time reprocessing over a 45-year period

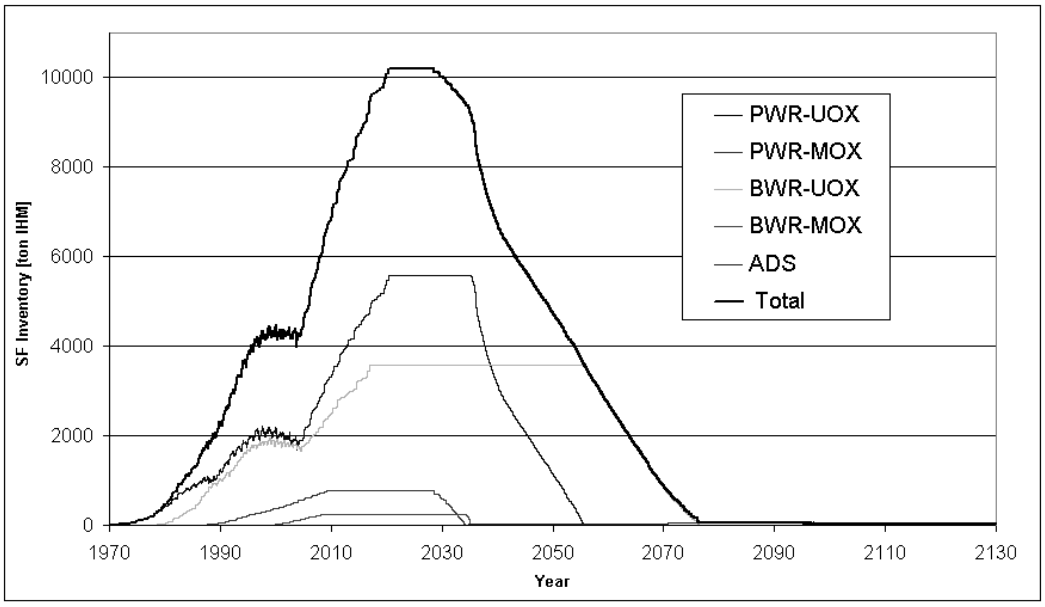
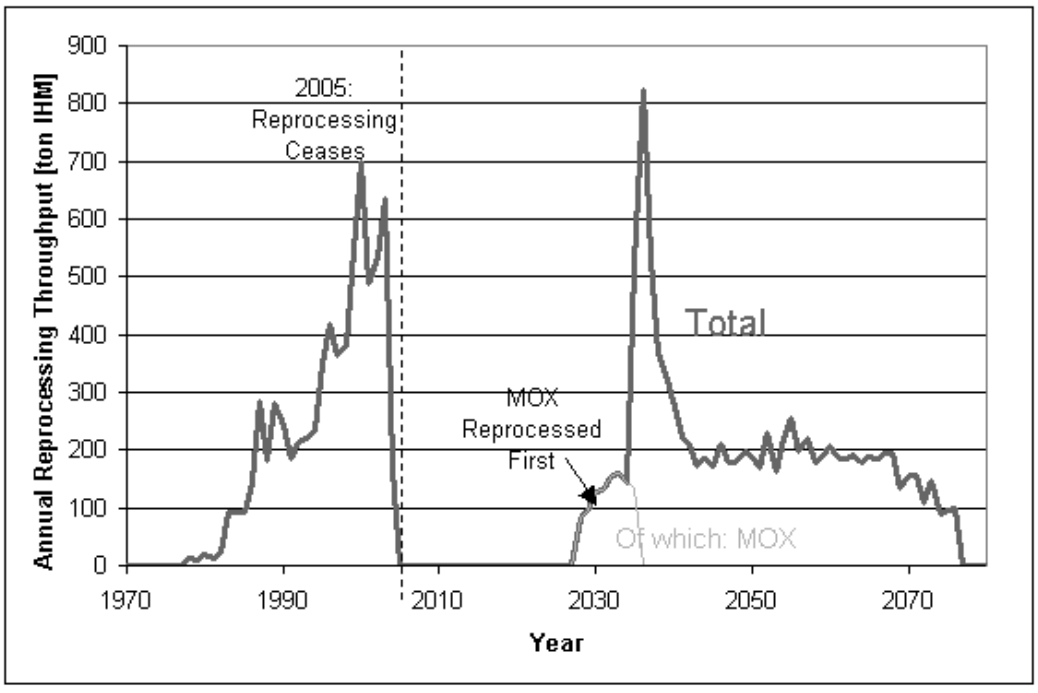
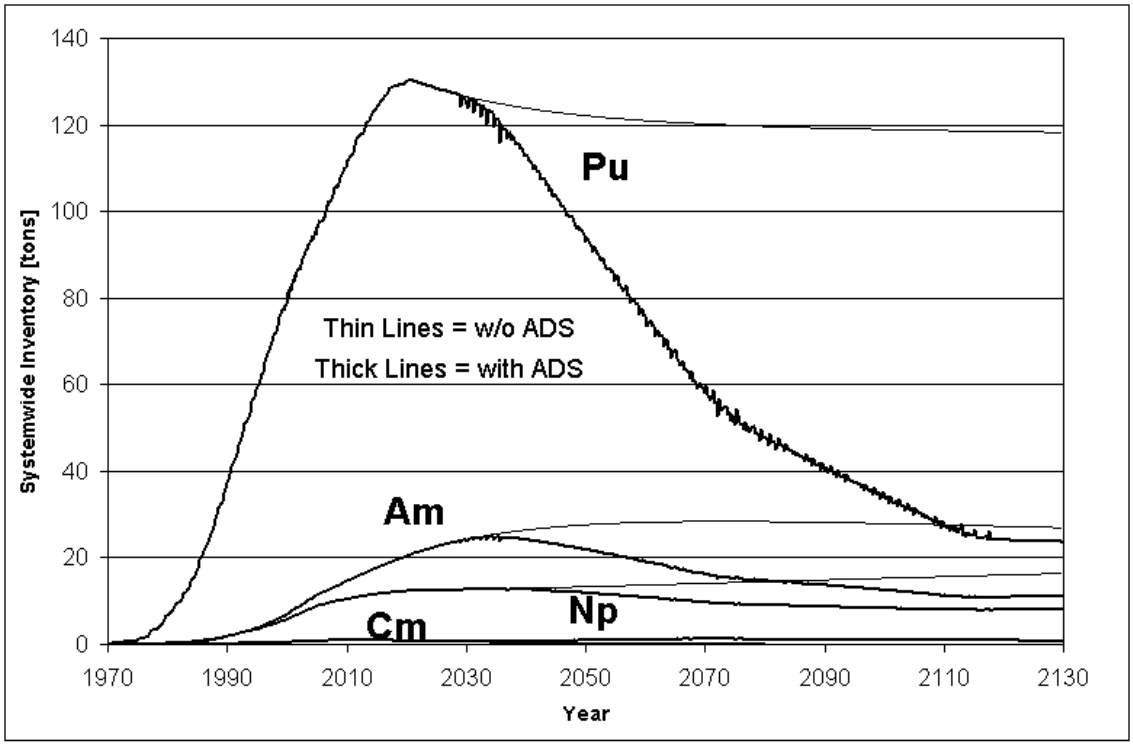


Figure 2.15. Annual oxide fuel reprocessing throughput, following oldest-first, just-in-time reprocessing



In addition to reducing SNF volumes, Figure 2.16 shows that this strategy results in a five-fold reduction in plutonium inventories over two generations of ADS operation. The reductions in MA inventories are not as great; note, however, that those shown in the figure represent system-wide inventories, including MA that were vitrified prior to the cessation of reprocessing in 2005.

Figure 2.16. The effect of ADS deployment on transuranic inventories



To further quantify the implications of this strategy on disposal options, the decay power of all nuclear material in the system was evaluated at several points in time. At any given time, this evaluation is carried out based upon all materials that have been out of pile for greater than five years. Younger SNF is discounted because the presence of very short-lived nuclides would render the results difficult to interpret. The instantaneous decay power of the SNF and vitrified HLW is shown in Figure 2.17. The current strategy, that incorporating ADS transmutation, diverges from the reference case in 2030. Increases in the decay power associated with the transmutation strategy after 2070 and 2110 are associated with the shutdown of ADS and discharge of their final cores. It is of interest to observe that the short-term heat release rate of the oxide SNF (were it allowed to decay) is approximately the same as that of the HLW and spent metal fuel discharged from the ADS. The bulk of the decline in the heat production rate of the oxide SNF during this time period is ascribed to the decay of ^{90}Sr and ^{137}Cs. ADS transmutation would seem to offer little benefit in the very short term simply because ^{90}Sr and ^{137}Cs are continuously being created during the operation of the ADS fleet. This new influx of high heat release fission products offsets, in the very short term, the benefit gained from fissioning the transuranics.

The benefits of the transmutation strategy become apparent when one examines heat production in the longer term. The decay power of stored nuclear material following 100 years of cooling is shown in Figure 2.18. In this figure, the value given at, say, 2020 reflects the heat production rate of all material that is out of pile in 2020 evaluated at 2120. This figure is meant to be relevant to long-term interim storage needs or to the early phases of repository operation, depending on the disposal strategy pursued. The benefits of transmutation are still partially offset by ongoing production of fresh high heat release nuclides, but to a lesser extent than was the case for the short-term decay heat. It is seen that two generations of transmutation reduce this medium-term heat load burden by roughly a factor of two.

Figure 2.17. Decay power of stored nuclear material at date shown

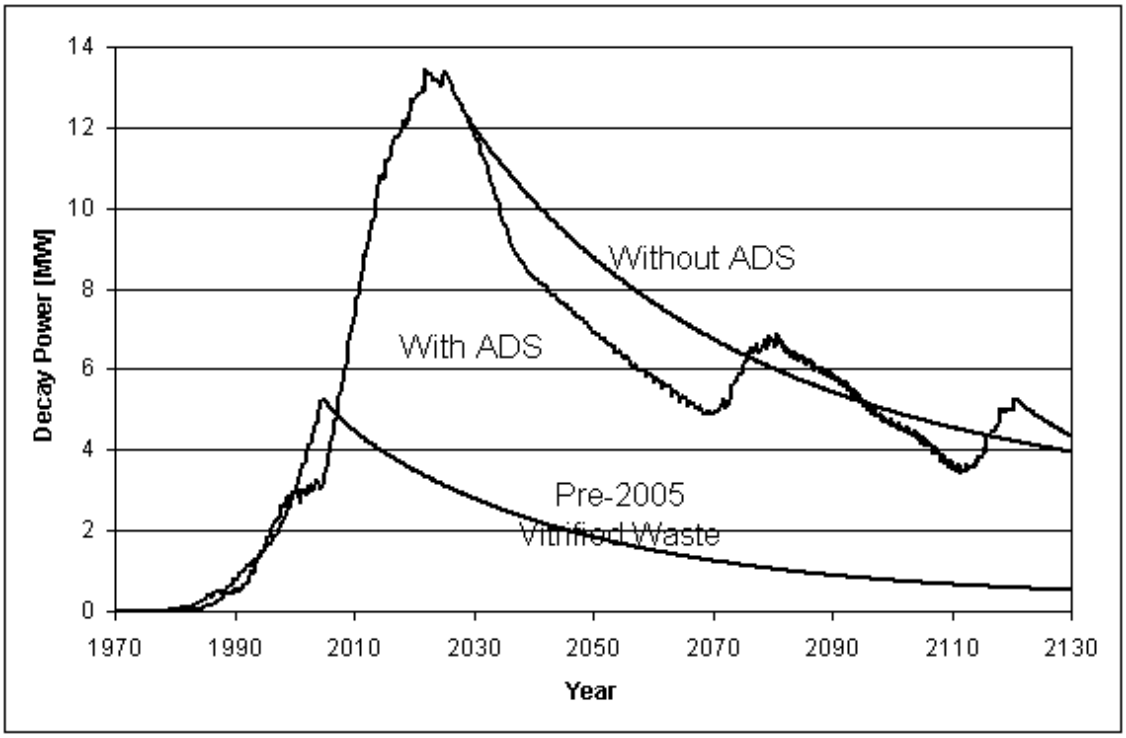
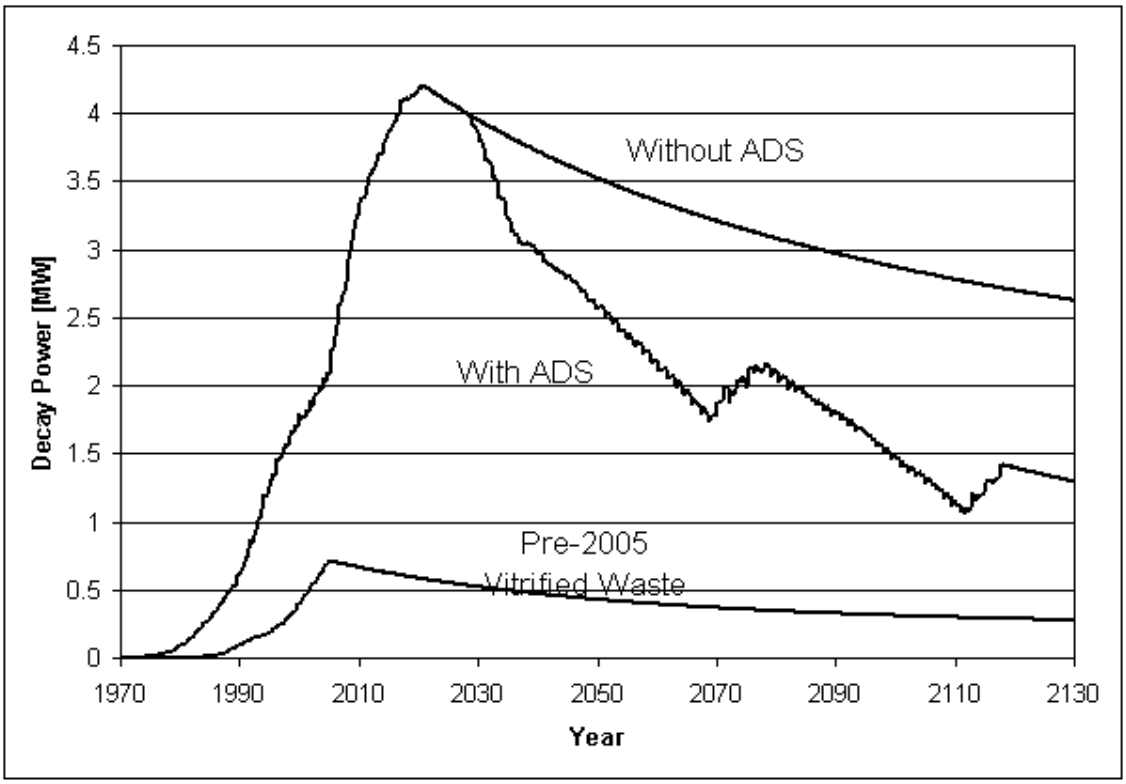
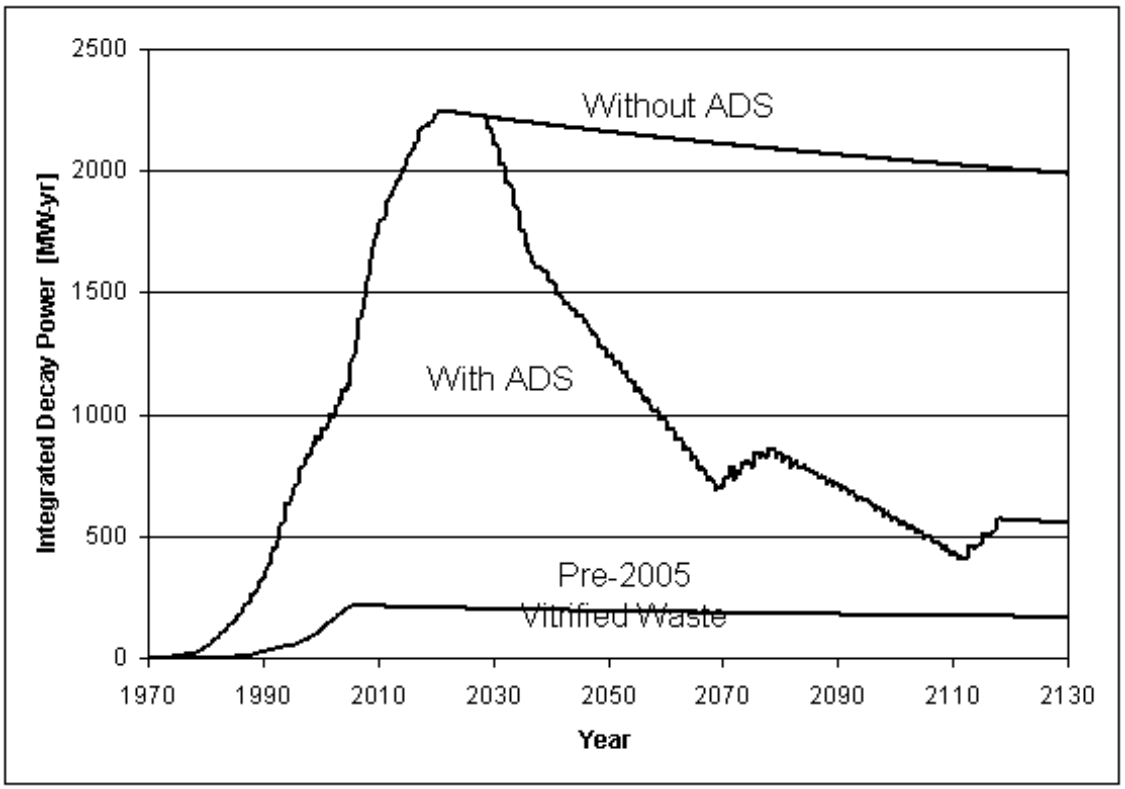


Figure 2.18. Decay power of stored nuclear material 100 years after date shown



The long-term decay heat – the decay power integrated over a period extending from 100 to 2 000 years in the future – is shown in Figure 2.19. Since transuranics, particularly ^{241}Am and ^{238}Pu, dominate heat production during this time scale, destruction of most of these isotopes via transmutation is seen to offer a substantial benefit: the decay heat production is reduced by a factor of four following two generations of ADS operation.

Figure 2.19. Decay power of stored nuclear material, integrated over period from 100 to 2 000 years after date shown



A transmuting fleet consisting of accelerator driven-systems can thus significantly alter, and by most metrics reduce, the burden of spent fuel and waste disposal. In view of the large investment required, it is questionable whether a nation possessing a relatively small nuclear infrastructure and inventory of SNF and HLW can independently afford the deployment of an ADS park.

Since the ADS deployed in all three cases have a zero conversion ratio and thus transmute at the same rate when they are at power all SNF was to be reprocessed by 2080.

Table 2.10. Facility deployment impacts of transmutation strategies

	First generation of ADS transmuters
Maximum no. of 840 MWt piles deployed	8 (2040-2070)
Integrated capacity deployed [GW_t-yr]	332
Integrated electrical generation [GW_e-yr]	282

Summary

Under Case II, following two generations of ADS deployment Germany transmutes 82% of the 129 tonnes of Pu and 45% of the 35.8 tonnes of MA it possessed in 2022. In fact, since Germany had already vitrified 7.4 tonnes of MA prior to 2005, it might be better to state that Germany disposed of 57% of the 28.4 tonnes of MA present in its SNF in 2022 and thus available for transmutation.

As compared to the no-action alternative, these accomplishments reduce the medium- and long-term heat production of the waste inventory by 50% and 72% respectively, as was shown in Figures 2.18 and 2.19. NFCSim results also showed large reductions in long-term radiotoxicity: the inhalation toxicity after 10 000 years was reduced by 79% and the ingestion toxicity by 76%.

Additionally, the evolved composition of the plutonium present in SNF better fulfils the criteria of non-proliferation after two generations of transmutation. Table 2.11 shows the isotopic content of all plutonium present in SNF in 2022 and 2122. For simplicity, oxide SNF – PWR and BWR, UOX and MOX – is lumped. In 2022, the SNF is the mixture of UOX and MOX presented under the heading *Results* on pg. 44. In 2122, only ADS SNF is present. It is clear that the plutonium resident in ADS SNF is of little value for weaponisation, even ignoring the substantial intrinsic radiation barrier to separations posed by the SNF itself.

Table 2.11. Proliferation-relevant attributes of German plutonium vectors averaged over all SNF at dates given

	238 (%)	239 (%)	240 (%)	241 (%)	242 (%)	Decay heat [W/kg]	Spont. fission neutrons [#/kg/s]	Bare sphere critical mass [kg]
SNF in 2002	2.4	54.4	28.1	7.5	7.7	16.9	450 000	14.7
SNF in 2122	10.5	13.9	52.6	4.2	18.7	63.9	1 080 000	22.0

Against these gains must be set the cost associated with deployment of 11 ADS plus oxide fuel reprocessing and dedicated metal fuel fabrication/reprocessing infrastructure. This scenario can be too high a burden, as it would be any other strategy (including the use of IMF), since specific installations should be deployed, including dedicated fuel fabrication, on a scale that is substantial for a nation with a limited nuclear infrastructure. Moreover, although serious design proposal will be made of an IMF fuel handling both Pu and MA in the framework of the EU STREP Project "LWR-Deputy", this premature option cannot presently be considered as a viable water-reactor-based path to complete fuel cycle closure.

2.5 Japanese transition scenario study

2.5.1 Current status

Japan imports most of energy resources (approximately 96%) from overseas. The Japanese energy supply structure is fragile. To improve this situation, Japan has developed nuclear power for the last fifty years based on the principle of peaceful use, and 53 nuclear power plants are now in commercial operation with a total install capacity of about 47 GWe at 2005. Nuclear power is an extremely stable energy supply and generates 16% of the primary energy supply in Japan. Nuclear power supplied one-third of electricity and the dependence rate of energy resource import is improved to 80% if nuclear power is considered as domestic energy resources. Nuclear power generation is an important main power supply system in Japan and contributes to stabilisation of domestic total energy supply and discharge restraint of greenhouse gas.

In addition, Japan has promoted the development of the nuclear fuel cycle to enhance the efficient use of uranium resources and to reduce high-level radioactive wastes (HLWs) as a national policy. Progress has been achieved in some fields, including uranium enrichment and nuclear waste management. A 1 050 t-SWU enrichment plant and a low-level radioactive waste disposal facility are in operation. The Rokkasho reprocessing plant with annual throughput of 800 tHM has started the uranium test and its commercial operation is scheduled to begin in 2007. The construction of a mixed-oxide (MOX) fuel fabrication plant is also in progress at the Rokkasho site. Plutonium extracted from the reprocessing of spent fuel will be recycled into LWRs as MOX fuel. The legal framework of the disposal of HLWs was promulgated in 2000. Potential sites are now being surveyed in accordance with the law, and construction and operation of facilities are planned to commence by the late 2030s [19].

2.5.2 Basic plans for TRU management

Japanese basic policy is that spent fuels are reprocessed and all high-level wastes are vitrified and disposed of in geological repositories. On the other hand, the "Options Making Extra Gains from Actinides and Fission Products" project (OMEGA Project) started in 1988 under the aegis of the Atomic Energy Commission of Japan in an effort to seek further efficiency and rationalisation of final disposal, aggressive improvement of safety, and efficient utilisation of resources. In the OMEGA project, the Japan Atomic Energy Agency [JAEA: created as a result of the fusion of the Japan Atomic Energy Research Institute (JAERI) and the Japan Nuclear Cycle Development Institute (JNC)] and Central Research Institute of Electric Power Industry (CRIEPI) have been developing partitioning and transmutation technologies. With regard to the partitioning process, technology for separation of transuranium elements (TRU), Tc-platinum group elements, Sr-Cs group elements, and other elements from high-level waste has been developed. There are currently two options with regard to a transmutation system. The former JNC has developed a TRU transmutation system using a fast reactor, and the former JAERI has researched and developed an accelerator-driven system (ADS).

JAEA (mainly former JNC) and the Japan Atomic Power Company (JAPC) started the feasibility study on a commercialised fast reactor (FR) cycle system in 1999 and are estimating several promising FR cycle concepts in co-operation with CRIEPI and the former JAERI. During Phase 2 of the feasibility study (FS), which started in 2001 under a five-year plan, several promising FR cycle concepts will be selected considering comprehensive examination results from the viewpoints of safety, economics, efficient utilisation of resources, reduction of environmental burden, nuclear non-proliferation, technical realisation and social acceptability. Figure 2.20 shows the concept of the FR cycle system pursued in the FS.

In the FS, TRU is defined not to be "waste" and most of the TRU is recovered from LWR and FR spent fuels and burned and transmuted in FR. The basic strategy is a shift from the phase of Pu recycling in LWR to a phase of TRU recycling in FR. MA in LWR spent fuels will be recovered after in a second reprocessing plant (near the Rokkasho plant) and 99.9% of MA in FR spent fuels will recycled in our own FR cycle in homogeneous mode.

2.5.3 FR cycle deployment scenario study

Basic nuclear energy scenarios

Japanese basic nuclear energy scenarios adopted in deliberation of a long-term programme of research, development and utilisation of atomic energy under the Atomic Energy Commission of Japan (AEC) are shown in Table 2.12. The nuclear energy scenarios are classified roughly into four

Figure 2.20. Concept of FR cycle system

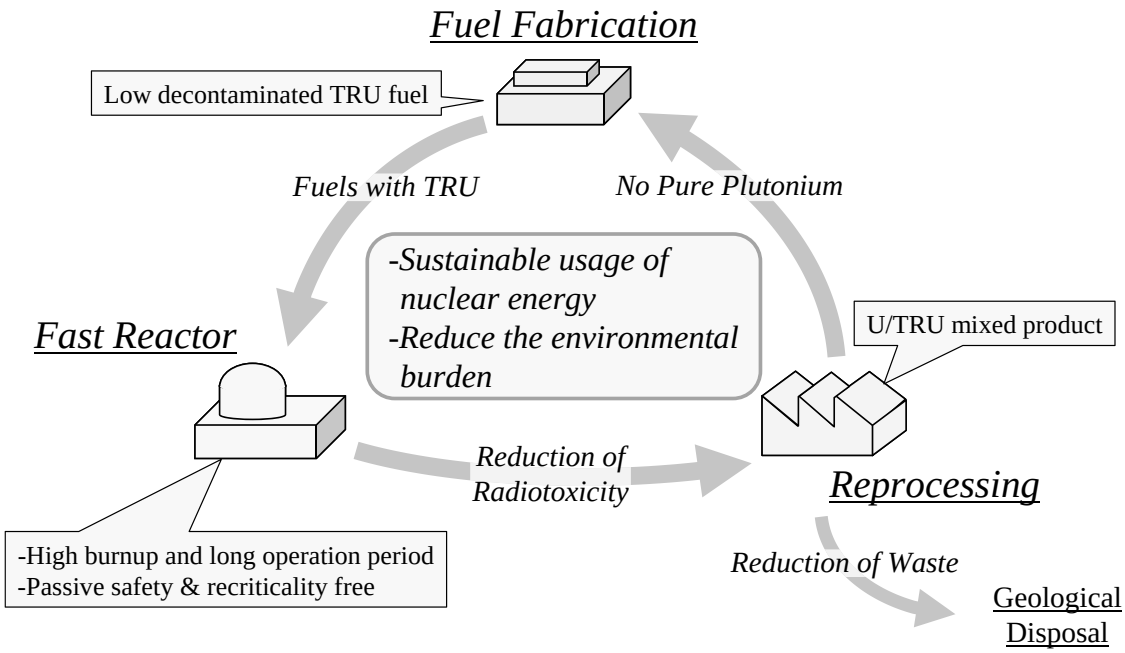


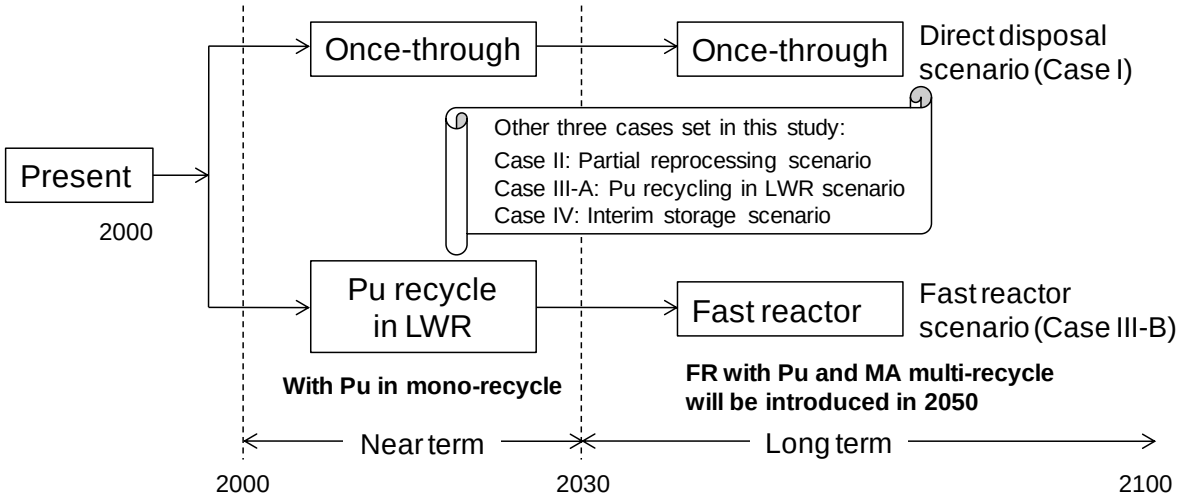
Table 2.12. Japanese nuclear energy scenarios

Case		Note
I. Direct disposal scenario		LWR once-through scenario (direct disposal of all spent fuels)
II. Partial reprocessing scenario		Reprocessing of a part of spent fuels and directly disposing of the remainders (Rokkasyo LWR reprocessing will terminate in 2047)
III. Reprocessing of all spent fuels		Continuance of nuclear fuel cycle policy
	(A) Pu recycling in LWR scenario	Continuation of LWR cycle by plutonium thermal utilisation in LWR
	(B) FR cycle deployment scenario	FR cycle will be deployed after 2050 with minor actinide (MA; Np, Am, Cm) recycling
IV. Interim storage scenario		FR cycle will be deployed in 2050 after interim storage

cases (Case I: direct disposal scenario; Case II: partial reprocessing scenario; Case III: reprocessing all spent fuels scenario; Case IV: interim storage scenario) from the viewpoint of disposal policy of spent fuel. Case I is a policy change to the direct disposal option with a prompt freeze of operation plan of Rokkasho Reprocessing Plant (hereafter RRP). Case II is a policy change to a direct disposal option after design lifetime of RRP. Cases I and II are considered to be one of the LWR once-through scenarios, though the deployment capacities of Pu recycling in LWR are different. Case III is divided into two cases according to the reactor types for Pu utilisation. The Pu recycling in LWR is assumed in Case 3-A and FR cycle deployment is assumed in Case III-B. Both Case III-B and Case IV are FR cycle deployment scenarios, but in Case IV the operation plan of RRP is put on ice and Pu utilisation will be resumed after 2050.

The analyses of the necessity of FR cycle deployment in Japan from a long-term viewpoint are carried out, by comparing "FR scenario (Case III-B)" with "LWR direct disposal scenario (Case I)" and "Pu recycling in LWR scenario (Case III-A)", from the viewpoints of efficient utilisation of uranium resource and reduction of environmental burden, such as cumulative uranium demand, spent fuel storage, radioactive waste arising, etc. Scenario studies are performed using the simulation code "FAMILY" developed by former JNC. Figure 2.21 shows the outline of this scenario study.

Figure 2.21. Outline of scenario study



Main assumption

In the future, Japan will face growing problems related to a decrease in the work force and a hollowing out of the industrial structures through declining birth rates and a growing proportion of elderly people. In addition, the deregulation of the energy industry renders long-term energy supply and demand perspectives complicated. The Japanese future as regards energy supply and demand is expected to evolve as follows:

- Energy demand and electricity demand will grow slowly. (Final energy demand is expected to decrease in the future because of the offset of a steady increase of energy demand in the residential sector and a decrease in population. On the other hand, electricity demand could be saturate at some point.)

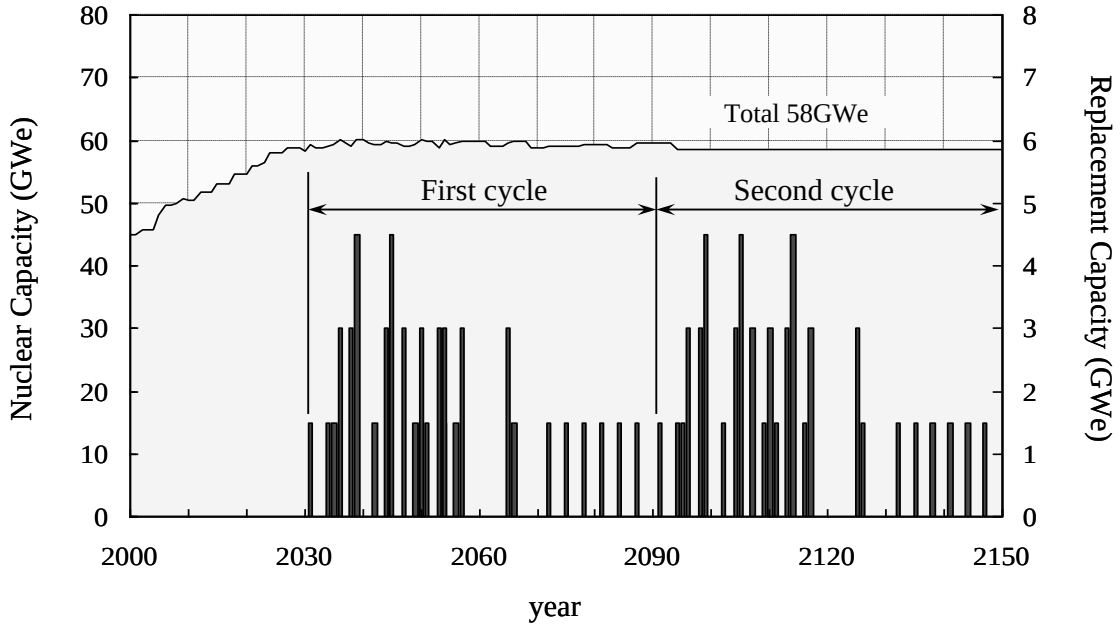
- Promotion of nuclear energy remains necessary as a means to break away from a weak energy supply structure to improve energy security.

- One of the primary roles of nuclear energy, which scarcely releases CO_2, as a basic power supply system is its importance as a means of observing the Kyoto Protocol and contributing to a global warming prevention policy.

- Similarly, energy conservation and renewable energy concerns arise from the viewpoint of global warming prevention measures.

A nuclear power generation capacity adopted in this scenario study is shown in Figure 2.22.

Figure 2.22. Assumption of nuclear power generation capacity in Japan



In 2030, nuclear power generation capacity is expected to increase to 58 GWe from the present 46 GWe, reducing Japanese CO_2 emissions to 1990s levels. The prediction of nuclear power generation capacity is based on the reference case of the interim report *Long-Term Outlook for Energy Supply and Demand* (October 2004), which was produced by the Energy Supply and Demand Subcommittee in the Advisory Committee for Natural Resources and Energy of the Ministry of Economy, Trade and Industry [20].

In order to analyse the influence of the various spent fuel disposal options and of the Pu recycling process in terms of long-term mass flow, the nuclear power generation capacity is assumed as 58 GWe, which is constant from 2030. The main assumptions concerning characteristic data of reactor and fuel cycle systems are shown in Table 2.13. On the basis of the technical summary of FR and its fuel cycle concepts in the preliminary evaluation of the FS Phase 2, the sodium-cooled FR with the advanced aqueous process and simplified palletising seems to be the most promising FR cycle concept, due to its technical advancement and conformity to the development target in the FS. Therefore, the sodium-cooled FR with the advanced aqueous process and simplified palletising concept is adopted in this scenario study.

The fuel burn-ups of LWR and FR are assumed to be 45-60 GWd/t and 150 GWd/t (core fuel), respectively. The FR breeding ratios are about 1.03 and about 1.10, and there will be a switchover to low breeding type core according to the Pu balance. The lifetime for each type of reactor is assumed to be 60 years. The ex-core time periods have been assumed to be four years for the LWR cycle and five years for the FR cycle (including three or four years storage at the reactor site in each cycle). The loss factor of the entire fuel cycle is 1.1% for the LWR cycle and 0.2% for the FR cycle. The tails assay in enrichment plant is assumed to be 0.3%. It was assumed that MA recovered from the high-level radioactive waste fluid in LWR reprocessing plants next to RRP was used in FR fuel. The upper limit for the MA density of FR fuels is 5%.

Table 2.13. Assumption of main system characteristic data

Item		Assumption
Reactor system	LWR	BWR, PWR: Burn-up 40 GWd/t, for reactor which will be deployed by 2019 Load factor 80% BWR, PWR: Burn-up 60 GWd/t, for reactor which will be deployed after 2020 Load factor about 90%
	FR	Na-MOX: Sodium-cooled type reactor with mixed-oxide fuel Breeding ratio 1.1 (breeding type core), 1.03 (break-even type core) Load factor about 95%, MA content 5% (upper limit)
	Lifetime	60 years for both LWR and FR
Ex-core time	LWR	Four years (cooling time three years, reprocessing 0.5 years, fabrication & transportation 0.5 years) (irradiation period about 4-6 years)
	FR	Five years (cooling time four years, reprocessing 0.5 years, fabrication & transportation 0.5 years) (irradiation period about 8-11 years)
Loss factor	LWR	Conversion 0.5%, fabrication 0.1%, reprocessing U 0.4%, Pu 0.5%, MA 0.1%
	FR	Fabrication 0.1%, reprocessing 0.1%
Reprocessing plant	LWR	JAEA's Tokai: 2001-2005, 40 tonnes/year Rokkasyo: 2005-2010, plan value, 2011-2046; 800 tonnes/year, abolished in 2047, 2047- , 800 tonnesHM/year (with MA recovery process)
	FR	Primary plants introduce 50 tonnes/year, and are expanded at unit of 200 tonnes/year depending on FR deployment capacity appropriately.
	Lifetime	40 years for both LWR and FR
Other		The uranium recovered from spent fuel is re-enriched

In the FS, four main fuel cycle concepts have been examined, namely advanced aqueous process with simplified pelletising, advanced aqueous process with sphere-packing, oxide electrolysis with vibro-packing, and metal electrorefining with injection casting. Preliminary evaluation results of the fuel cycle concepts are as described below.

The main process flow of the advanced aqueous process with simplified pelletising is shown in Figure 2.23. The advanced aqueous process consists of a simplified process with the addition of a uranium crystallisation step, a single cycle co-extraction step of U, Pu and Np, and a MA recovery step. The crystallisation step removes most of the bulk heavy metal and eliminates it from downstream processing. The purification step of U and Pu in the conventional process is eliminated, and U/Pu is co-extracted with Np. The simplified pelletising process is rationalised by eliminating the powder blending step and the granulation step from the conventional MOX pellet process. The perspective of technical feasibility toward the commercialisation of this concept would be relatively high as a result of many years research at JNC-Tokai. Recovery of U/TRU was estimated to be greater than 99%. The key technical issues for the commercialisation of the advanced aqueous process are scale-ups of the additional steps. Further, it is important to demonstrate the production of MOX pellets containing MA and trace amounts of fission products in a hot cell facility, which is remotely operated and maintained.

Results of scenario study

The calculation results of the long-term mass flow analyses until 2150 on nuclear scenarios are described here. Nuclear power generation capacity for each reactor type in a direct disposal scenario (Case I) is shown in Figure 2.24. Although Case I is basically LWR once-through, the maximum capacity for LWR with Pu recycling to use Pu returned from reprocessing plants in foreign countries will reach about 6 GWe.

Figure 2.23. Main process flow of advanced aqueous process and simplified pelletising

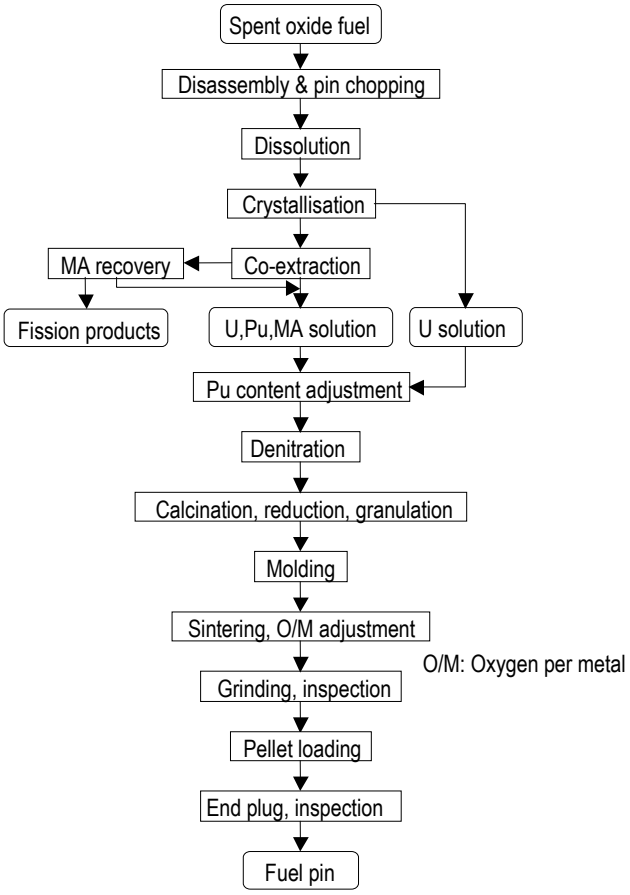
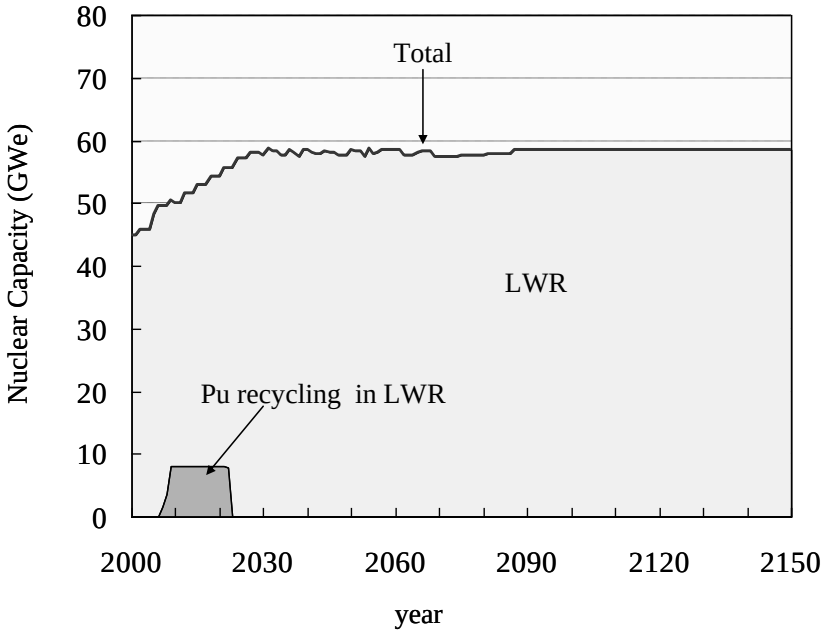
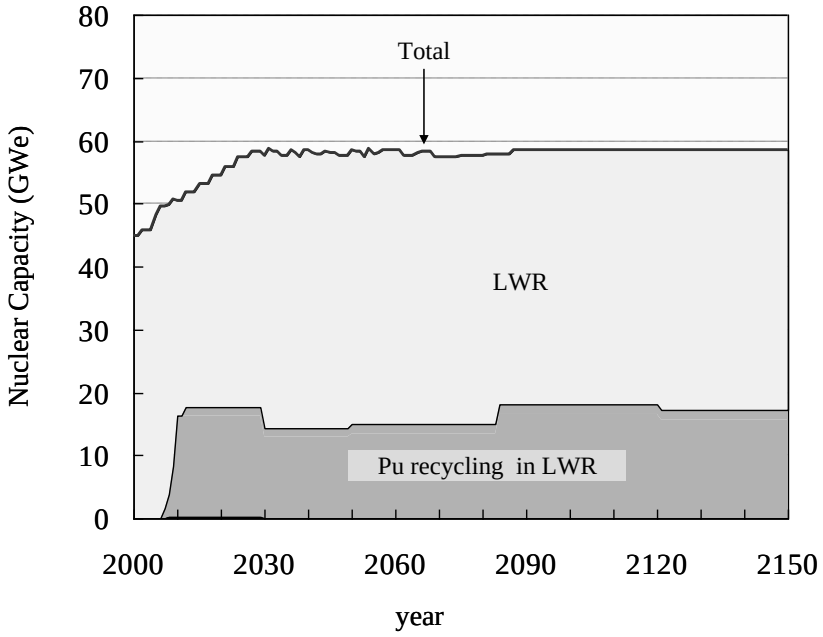


Figure 2.24. Capacity for each reactor of type Case I (direct disposal scenario)



The capacity for LWR with Pu recycling in Case III-A, Pu recycling in a LWR scenario is estimated to be about 30% of the whole LWR, as is shown in Figure 2.25. LWR capacity for Pu recycling will be restricted based on the Pu balances, with the average capacity being about 17 GWe after 2050. The second LWR reprocessing plant capacity will increase from 800 to 1 000 tonnes/year in accordance with the amount of spent fuel storage. Pu multi-recycling in LWRs utilises reprocessing plants for processing both MOX and UOX (MOX:UOX = 1:7). In this case, the amount of the reprocessing of MOX spent fuel becomes about 50 tonnes/year.

Figure 2.25. Capacity for each reactor of type Case III-A (Pu recycling in LWR scenario)



Nuclear power generation capacity of FR cycle deployment scenario (Case III-B) is shown in Figure 2.26. In this case, a premeditated restriction of Pu recycling by LWR is necessary to save Pu used for fabricating FR initial loading core fuel. In the calculation of Case III-B, the end of Pu recycling by LWR is 2045. After 2050, LWRs of about 1 GWe will be replaced by FRs every year, and the switchover to FRs will be almost complete at the beginning of the 22nd century. In addition, the maximum reprocessing capacity for processing LWR spent fuel and FR spent fuel in a FR cycle deployment scenario (Case III-B) is estimated to be about 1 400 tonnes/year as is shown in Figure 2.27. FR reprocessing plants of 50 tonnes/year unit or 200 tonnes/year unit will be introduced based on the amount of spent fuel storage. Even if the FR and FR reprocessing plants of 200 tonnes/year are introduced almost at the same time, a high load factor will be expected by reprocessing the MOX fuel of LWR in FR reprocessing plants. Reprocessing of LWR spent fuel would be complete in about 2120.

Figure 2.28 shows the accumulative natural uranium demands. The accumulative natural uranium demands for Case I (direct disposal scenario) continue to increase at a rate of about 10 000 tonnes/year, and will reach about 1.6 million tonnes U in 2150. The natural uranium demand per one year of Case III-A (Pu recycling in LWR scenario) is less than that of Case I by about 15%, but accumulative natural uranium demands will increase continuously until 1.3 million tonnes U in 2150. In addition, accumulative natural uranium demands of Case III-B will be saturated with about 5% of conventional uranium resources (14.8 million tonnes U [21]) at the beginning of the 22nd century and it is not necessary to import natural uranium from foreign countries after the saturation. Case III-B is less likely than any other scenario because of the FR cycle deployment.

Figure 2.26. Capacity for each reactor of type Case III-B (FR cycle deployment scenario)

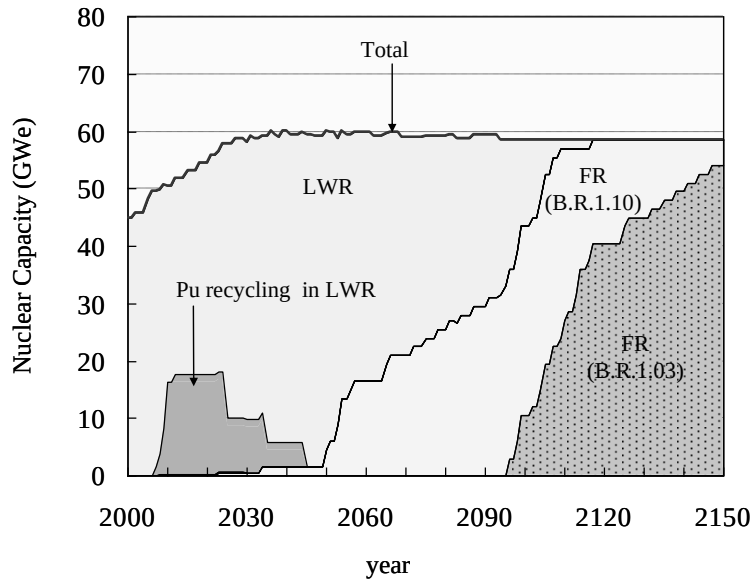


Figure 2.27. Capacity for reprocessing plants of Case III-B (FR cycle deployment scenario)

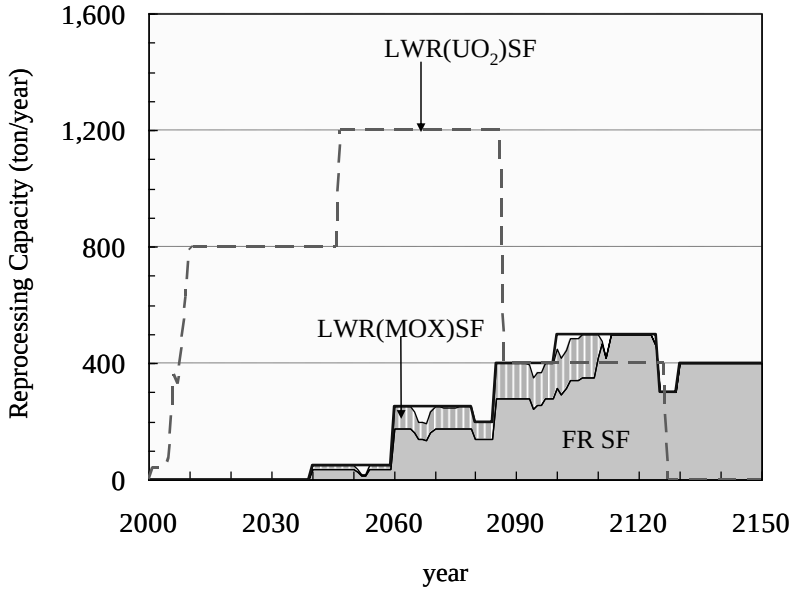


Figure 2.29 shows spent fuel storage in the three scenarios. The spent fuel storage is defined as the spent fuels from LWRs and FRs stored in reactor sites (three or four years cooling storage) and interim storage sites except for the spent fuels in direct disposal sites. In Case I (direct disposal scenario), an interim storage capacity of about 50 000 tonnes would be needed. On the other hand, approximately 20 000 tonnes capacity would be sufficient for Case III (reprocessing of all spent fuels).

Pu accumulation in high-level radioactive wastes which are disposed of in final disposal sites is shown in Figure 2.30. Pu accumulation in Case I is about 900 tonnes in 2150. The quantity of Pu for Case III-B, however, is less than 1 tonne. Most of the Pu is recovered from spent fuels and is recycled for the FR cycle.

Figure 2.28. Accumulative uranium demands of three scenarios

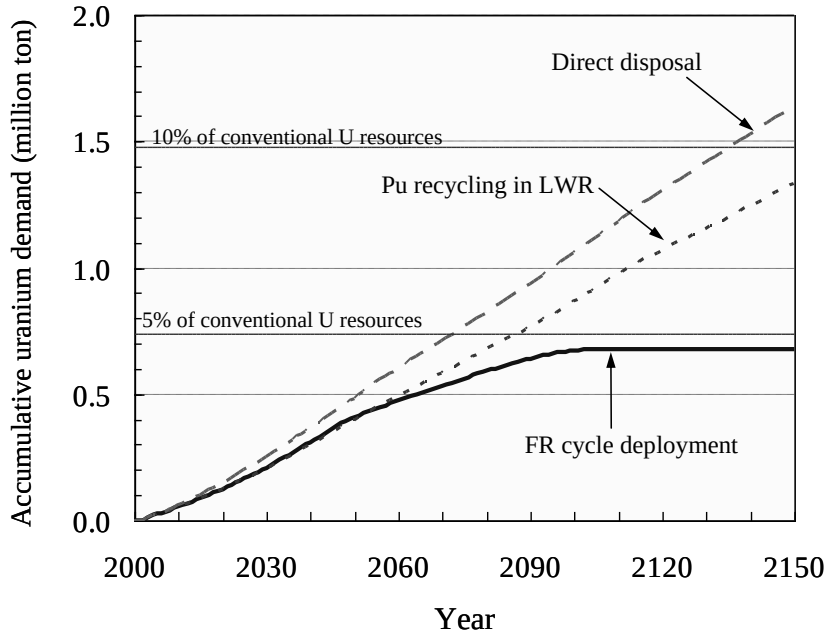
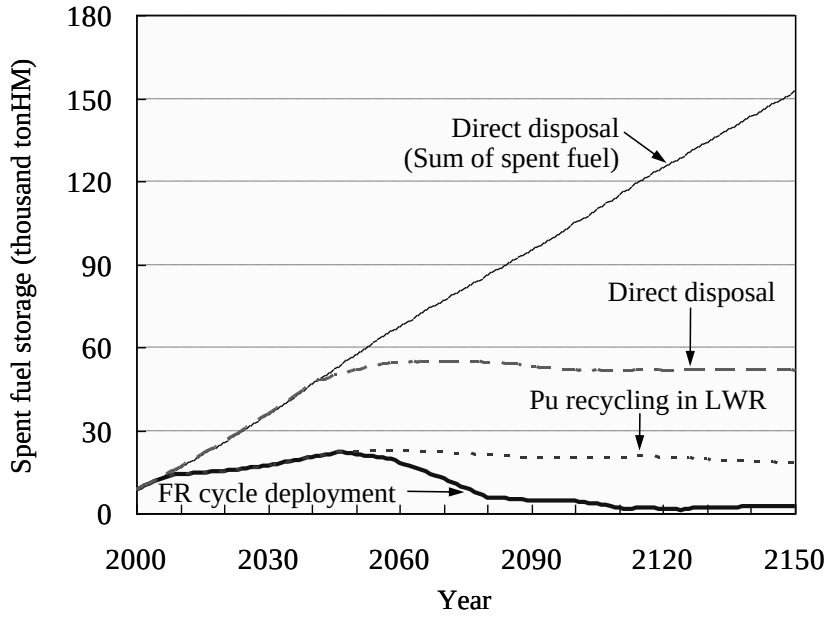


Figure 2.29. Spent fuel storage of all scenarios



MA accumulation in high-level radioactive wastes (including spent fuel for direct disposal) which are transferred to a final disposal site is shown in Figure 2.31. By 2150, MA accumulation in Case I and Case III-A is estimated to be about 220 tonnes, for Case III-B about 80 tonnes. Direct disposal of LWR spent fuel (both UO_2 and MOX) will increase the MA accumulation. In the FR cycle deployment scenario (Case III-B), increase of MA accumulation will cease after 2100 because MA will be recovered after the second LWR reprocessing plants are deployed in 2047.

Figure 2.30. Plutonium in LWR spent fuel and vitrified waste after disposal

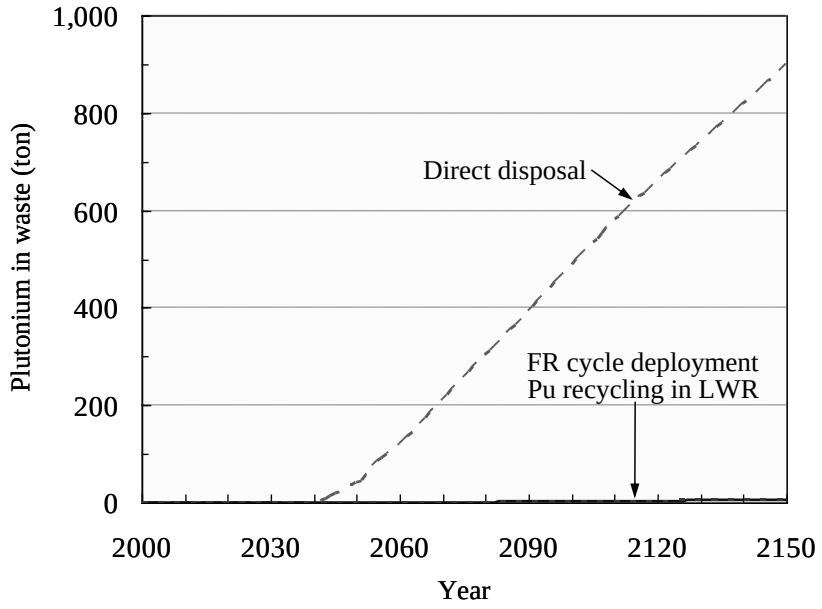


Figure 2.31. Minor actinides in LWR spent fuel and vitrified waste after disposal

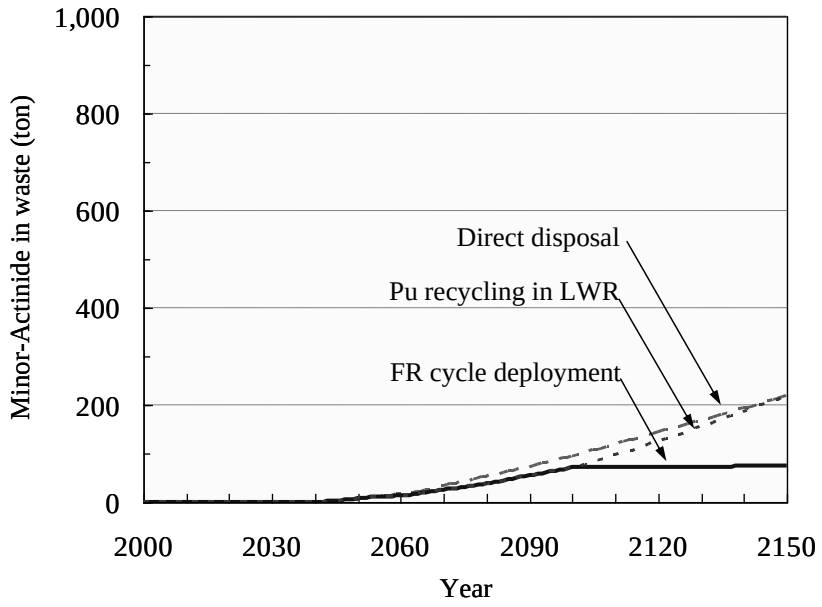
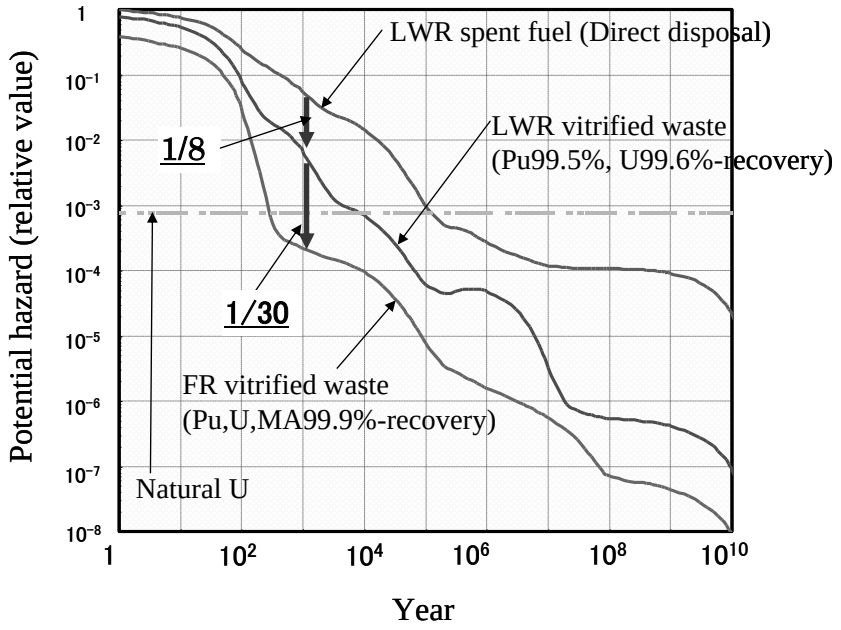


Figure 2.32 illustrates the potential radioactive hazard of high-level wastes per electricity unit. The vertical value does not signify real risk, but rather potential hazard of high-level wastes out of consideration of the barrier between human and wastes. In the direct disposal scenario, spent fuels including all nuclides (uranium, plutonium, minor actinides and fission products) become high-level wastes. On the other hand, as vitrified wastes after reprocessing include fission products and a little uranium, plutonium and minor actinides, the potential hazard is small. The potential hazard of LWR vitrified waste at one thousand years after discharge is one-eighth of LWR spent fuel under the direct disposal scenario. The potential hazard of FR vitrified waste is one-thirtieth of LWR vitrified waste because of the high recovery rate (99.9%) for uranium, plutonium and minor actinides with the FR reprocessing process.

Figure 2.32. Radioactive potential hazard of high-level wastes



2.5.4 Conclusions

The nuclear power generation capacity of Japan was assumed to be 58 GWe in the future, and with this figure in mind, long-term mass flow analyses for representative nuclear scenarios were carried out. From the viewpoint of reduction of environmental burden, a large decrease of actinides (U, Pu, MA) in high-level radioactive waste is expected under the FR cycle deployment scenario. Most actinides can be managed within the FR cycle. The FR cycle deployment scenario is superior to any other as concerns the reduction of the environmental burden and natural uranium demands. Therefore, this choice is considered to contribute to the preservation of the environment and sustainable utilisation of nuclear power.

2.6 Reactor deployment strategy with SFR introduction for spent fuel reuse in Korea

The present domestic nuclear fleet is composed of 16 PWRs and 4 PHWRs with a total capacity of 17.7 GWe in Korea. More than 700 tonnes of spent fuel is annually discharged from the present nuclear fleet. The spent fuel arisings are temporarily stored at each nuclear power site and await their final waste disposal. The accumulation of PWR spent fuel already amounts to about 9 000 tonnes. With the continuous expansion of nuclear power capacity, overall PWR spent fuel storage capacity is foreseen to be saturated by 2016, even taking into account the expansion of spent fuel storage pools at each nuclear power site. In addition, it is difficult to determine the location of a waste disposal site from the viewpoint of public acceptance. The disposal of radioactive waste is an impending challenge in Korea.

The sodium-cooled fast reactor (SFR)/PWR coupled scenario study has already shown that SFRs can substantiate the domestic waste management claims in Korea by reducing the amount of spent fuel and the environmental burden by decreasing the radiotoxicity of high-level waste through transmutation [22]. SFRs are designed to recycle transuranics (TRU) through the reuse of PWR spent fuel, which is also of benefit in terms of efficient use of natural uranium, thus contributing to sustainable

development. With innovations for reductions in capital cost, waste management can be extended to electricity production, given the proven capability of SFRs to utilise almost all of the energy in natural uranium. From this viewpoint, SFRs designed for an integral recycling of all actinides (uranium and TRU), appear to be one of the Generation IV (Gen-IV) candidate nuclear energy systems.

The Gen-IV SFR is expected to be commercialised by around 2030, well before other Gen-IV reactor systems. In this context, according to the Nuclear Technology Roadmap established in Korea in 2005, a SFR was chosen as one of the most promising future types of reactors which could be deployable by 2030. The SFR Basic Key Technologies Development Project for the development of a conceptual design of a Gen-IV SFR is being conducted by KAERI under the third national mid- and long-term nuclear R&D programme, newly launched as a ten-year programme in 2007.

Korea's share in the world reactor-related uranium requirement was 5.1% in 2005 [23]. Its share by the year 2015 is projected to be 5-7%. The role of nuclear power in electricity generation is expected to become more important in Korea in the years to come due to increasing electricity demand and poor natural resources. Concerning the security of the uranium supply, however, difficulty is expected in obtaining a supply of uranium over 5% in the global uranium market, in light of the projection that nuclear capacity will more than double in the coming era of nuclear renaissance, particularly in several Asian countries.

Efficient reactor deployment scenarios including SFRs are sought to optimise the SFR deployment strategy for replacing the existing nuclear fleet mainly composed of PWRs, with a view toward spent fuel reduction and the efficient utilisation of uranium through its reuse. An accelerator-driven subcritical (ADS) system, the Hybrid Power Extraction Reactor (HYPER), currently being developed as a possible nuclear option, is not included in the future nuclear fleet, as it is still at the stage of fundamental research.

2.6.1 Scenarios and evaluation

Description of scenarios and assumptions

Description of scenarios

Deployment scenarios are simulated for the period of 2005-2100. Seven deployment scenarios for reactor strategy are considered to evaluate the total amount of uranium demand and spent fuel accumulated with different SFR missions and mixing ratios in the future nuclear fleet:

- *Case 1*: PWR once-through cycle (OTC), direct disposal of spent fuel without treatment;
- *Case 2*: Breeder (BR) only with all of decommissioned PWRs being replaced with BRs;
- *Case 3*: Burner (BN) only with mix ratio of SFRs in 2100 being 30 ~ 40%;
- *Case 4*: Breakeven (BK) reactor only with mix ratio of SFRs in 2100 being 30 ~ 40%;
- *Case 5*: (BK + BN) with mix ratio of SFRs in 2100 being 30~40%;
- *Case 6*: (BN + BK) with mix ratio of SFRs in 2100 being 30~40%;
- *Case 7*: (BN + BK) with mix ratio of SFRs in 2100 being ~50%.

In cases of SFR deployment (Case 2-7), a demonstration SFR will be introduced in 2030, with commercial SFRs being deployed from 2040 in accordance with the corresponding SFR type deployment scheme.

This scenario study aims to find an efficient reactor deployment scenario which can meet the following requirements:

1. The amount of accumulated PWR spent fuel arising shall be kept below 20 ktHM, which is an estimated capacity requirement for repository at present.

2. The amount of uranium demand accumulated shall be below 5.0% of identified uranium resources in the world.

Long-term nuclear power generation projection

In 2007, 16 PWRs (6 OPRs) and 4 PHWRs are in operation. The nuclear electricity generation installed capacity in 2006 was 17.7 GWe, supplying 39.0% of the total electricity. According to the "Third Basic Plan for Long-Term Electricity Supply and Demand", the nuclear installed capacity will become 27.3 GWe in 2020 and the nuclear share will be 43.4% of the total electricity generation [24].

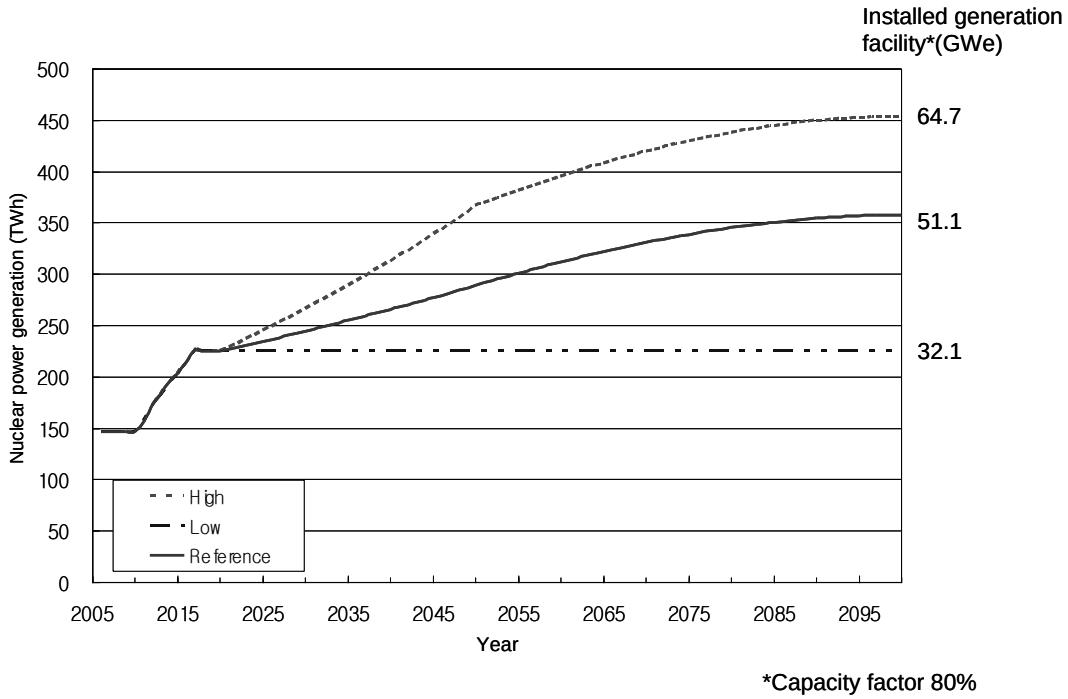
With the basic assumption that nuclear power is maintained as a major electric power source, three scenarios (high, reference and low) for total and nuclear power generation differentiated by either annual growth rates or nuclear shares are considered in this SFR introduction scenario study. Total and nuclear electricity generation for three scenarios by the year 2020 are given by the same data, according to the "Third Basic Plan for Long-Term Electricity Supply and Demand". From 2020-2050, total electricity generation for the reference scenario is projected to have an annual growth rate of 1.0%; after 2050 a gradual decrease is projected its value to reach 0% in 2100. In the reference scenario, the nuclear share 43.4% planned as of 2020 is kept until 2100. In the high scenario, the nuclear share gradually increases to 55.0% until 2050 and since then it is maintained until 2100. On the other hand, the low scenario assumes that nuclear power generation 225 TWh as of 2020 is kept until 2100.

Figure 2.33 shows long-term nuclear power generation projections estimated by three nuclear power generation scenarios: high, reference and low. The reference scenario was used to begin the SFR introduction scenario study. In the reference scenario, the total nuclear installed capacity is projected to increase to 51.1 GWe in 2100, which corresponds to 350 TWh/yr of nuclear electricity generation estimated by the capacity factor 80%.

Assumptions

The lifetime of existing nuclear power plants is extended up to 60 years, the same as that of SFRs. Commercial SFRs are introduced into the power grid as of 2040, following the introduction of a demonstration SFR in 2030. CANDU (PHWR) reactors will no longer be constructed, and will be retired around 2050. Three types of SFRs [breeder (BR, breeding ratio 1.22), breakeven reactor (BK, breeding ratio 1.0) and burner (BN, conversion ratio 0.61)] are considered for SFR deployment. Power capacities of PWRs and SFRs are 1 000 MWe and 600 MWe, respectively. Input data for BN and BK reactors were prepared based on the Korea Advanced Liquid Metal Reactor (KALIMER)-600 designs [25,26].

Figure 2.33. Long-term nuclear power projection



Existing SFR fuel is supplied by pyroprocessing of spent fuels. All TRUs (Pu and MA) produced from PWRs and SFRs are recycled and transmuted in SFRs. Recycling of CANDU (PHWR) spent fuel is not considered in the study. It is assumed that a reasonable amount of PWR spent fuel should be maintained for supplying SFR fuel without interruption even after 2100.

Details concerning the annual fuel mass balance for a PWR-SFR coupled equilibrium fuel cycle are schematically diagrammed in Figure 2.34. The start-up fuel for SFRs is composed of recovered PWR discharged TRU and depleted uranium. The isotopic compositions of PWR TRU are given, based on a typical five-year-cooled 50 000 MWD/t burnt PWR spent fuel discharged from domestic nuclear power plants. By forming a closed fuel cycle, remaining and newly red fissile material is recovered and recycled together with long-lived radiotoxic nuclides. The comparison of TRU mass balances indicates that the burner will be more efficient for reducing the accumulated PWR spent fuel arisings.

2.6.2 Results and discussions

Results for the reference scenario

The main results obtained from the scenario analyses are given in Table 2.14. In this table, the results for first seven cases (Cases 1-7) were obtained until 2100 based on the reference scenario. From the synthetic comparison of the results obtained for the reference scenario (Cases 1-7), Case 6 (BN+BK), where BNs are deployed prior to BKs, is selected as the most appropriate SFR deployment scenario. The results of last three cases (Cases 8-10) will be discussed later.

Figure 2.34. Annual fuel mass balance

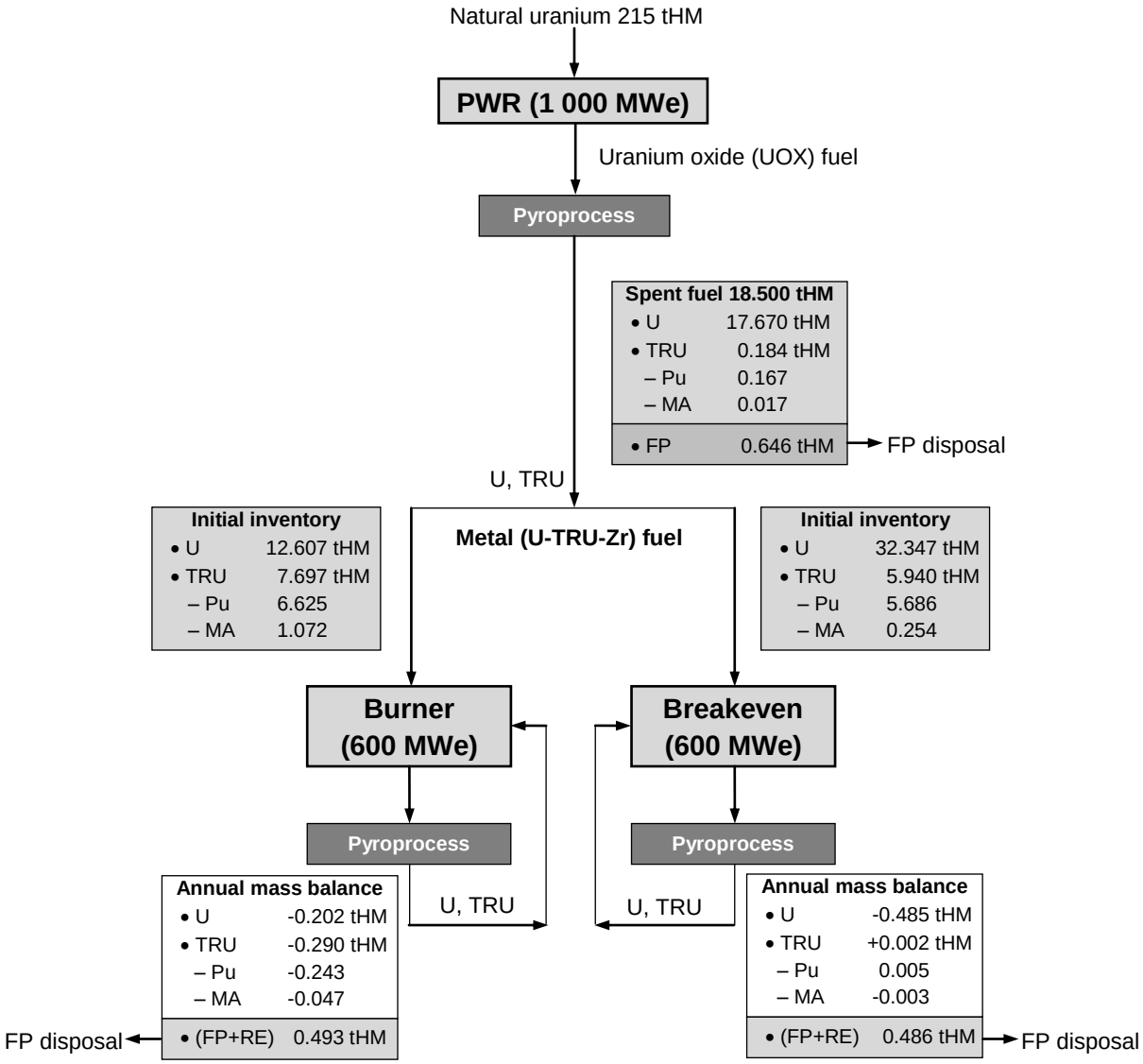


Table 2.14. Main results of scenario studies (as of the end of the year 2100)

Scenarios		Reference (first investigation)							High	Reference	Low
		1	2	3	4	5	6	7	8	9	10
		PWR-OTC	BR only	BN only	BK only	BK+BN	BN+BK	BN+BK	BN+BK	BN+BK	BN+BK
Uranium resource	Accumulated demand (ktU)	885	509	717	727	728	723	685	537	445	335
	Savings (ktU)	0	375	158	159	157	162	200	143	115	86
	Accum. domestic demand/Identified resources*(%)	6.0	3.4	4.9	4.9	4.9	4.9	4.9	3.6	3.0	2.3
Spent fuel	Accumulated (ktHM)	83.2	41.0	1.0	50.2	22.0	15.1	1.2	1.0	2.0	6.7
	Savings (ktHM)	0.0	40.1	74.4	33.2	57.6	66.1	82.0	82.0	64.6	46.8
MA	Accumulated (t)	77.9	38.4	0.9	44.9	23.5	14.1	1.1	1.0	2.8	6.3
	Savings (t)	0.0	37.5	69.6	31.1	75.7	61.9	78.8	75.5	60.5	43.8
Reactor	SFR mix ratio (%)	–	100.0	41.6	35.0	37.2	35.0	50.4	39.0	38.0	45.0
Remark			Does not satisfy Req. (1) in Sec. 2.6.1	Insufficient fuel supply is expected after 2100	Does not satisfy Req. (1) in Sec. 2.6.1	Does not satisfy Req. (1) in Sec. 2.6.1	Satisfies Reqs. (1) and (2) in Sec. 2.6.1	Insufficient fuel supply is expected after 2100	Satisfies Reqs. (1) and (2) in Sec. 2.6.1		

BR: Breeder, BN: Burner, BK: Breakeven.

* 14.80 million tU [OECD/NEA-IAEA, *Uranium 2005: Resources, Production and Demand* (2006)].

Figure 2.35 shows the accumulation of annual PWR spent fuel arisings for several SFR deployment cases, compared with the PWR once-through (PWR-OTC) strategy with no reprocessing (Case 1). The PWR spent fuel accumulation is greatly reduced at the SFR introduction due to the substantial amount of spent fuel being used for the start-up core of SFRs. SFRs are to be deployed in support of substantial reduction of PWR spent fuel at the first stage of deployment. The continuous deployment of burners effectively reduces the amount of PWR spent fuel accumulation below 20 ktHM in 30 years after the introduction of commercial SFRs, thus lightening the burden for PWR spent fuel management.

Figure 2.35. Accumulated spent fuel arisings (reference scenario)

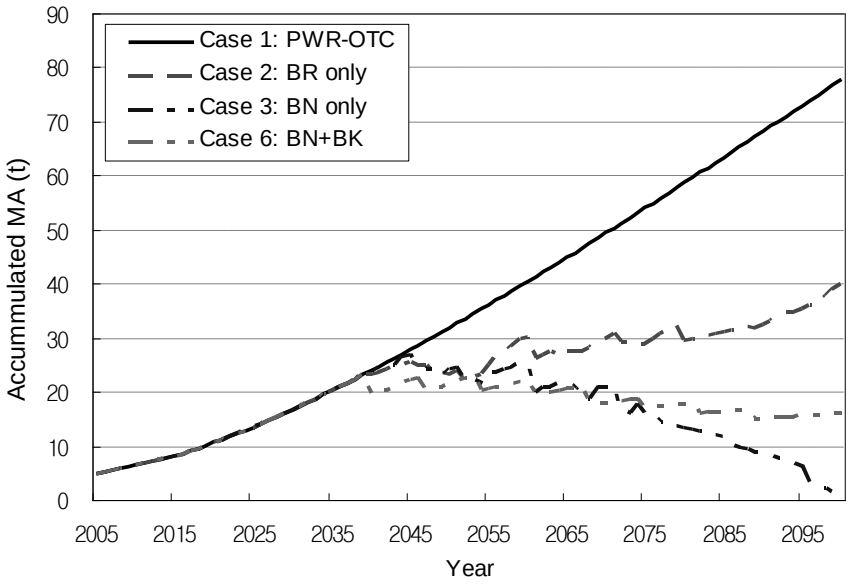
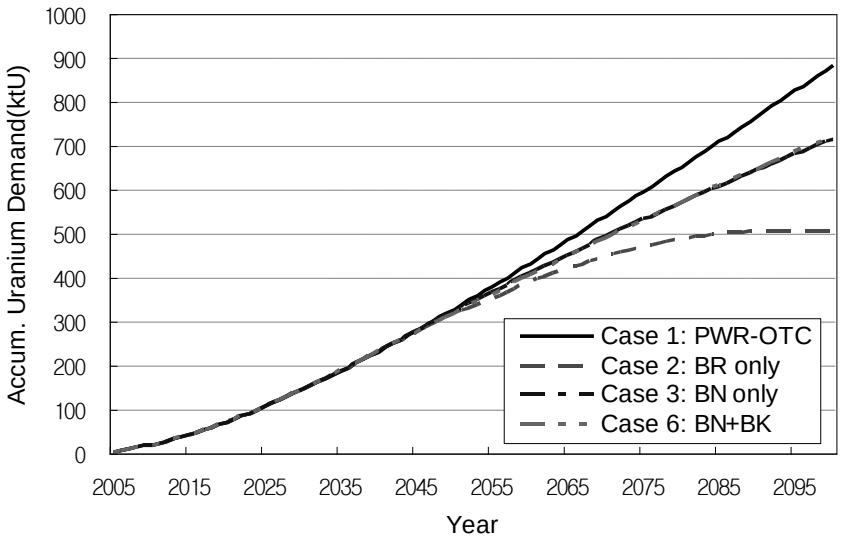


Figure 2.36 illustrates accumulated uranium demands for various SFR deployment strategies in comparison with the PWR once-through (PWR-OTC) strategy with no reprocessing. It can be seen that the introduction of SFRs, where TRUs are recycled by the reuse of PWR spent fuel, substantially reduces uranium demand. The introduction of breeders (BRs) effectively reduces uranium demand through producing excess TRU during the operation. This leads to the efficient use of natural uranium, thus contributing to a sustainable nuclear power development. Accumulated uranium demand is estimated to be less than 740 ktU, 5% of the amount of identified uranium resources 14.8 million tU [24], for all cases with the SFR deployment. The uranium savings generated due to SFR deployment is estimated to be more than 158 ktU.

The amount of installed capacity and the deployment rates for burners are limited by the amount of TRU or plutonium available for feeding the start-up fuel at the burner introduction. TRU availability strongly depends on the amount of PWR spent fuel accumulated from achievement of nuclear power plant operations as well as the spent fuel arisings from existing nuclear power plants. It is noted that the continuous deployment of burners only (Case 3) could effectively exhaust all PWR spent fuel accumulation before 2100. In this case, scenario solutions are sought subject to the requirement that a reasonable amount of PWR spent fuel accumulation should be maintained.

Figure 2.36. Accumulated uranium demand (reference scenario)



Applicability to different nuclear power development environments

The SFR deployment scenario (Case 6) selected as the most appropriate, is applied to the other two cases, *i.e.* high and low cases (corresponding to Cases 8 and 9 for analysis, respectively), with the view toward investigating its applicability to various nuclear power development environments. In this investigation, spent fuel is assumed to be produced only from PWRs.

The results obtained from the analyses of the last three cases (Cases 8-10) show that the SFR deployment strategy (Case 6) is applicable to various nuclear power development environments even with no additional nuclear installed capacity to the existing nuclear fleet after 2020 (Case 10). From the comparison of the results for these three cases (Cases 8-10), Case 9 (BN+BK) is finally chosen as the most appropriate SFR deployment scenario.

In case of the most appropriate deployment scenario (Case 9), where BKs are deployed from 2068 after the deployment of BNs starting from 2040, PWR spent fuel accumulation is reduced to a certain amount below 20 ktHM. This is illustrated in Figure 2.37. In Figure 2.38 the accumulated uranium demand for PWRs until 2100 is estimated to be 445 ktU, which indicates 115 ktU of uranium savings subsequent to the introduction of SFRs. The accumulated uranium demand occupies 3.0% of identified uranium resources, 14.8 million tU, which implies a secure purchase in the global uranium market. PWR spent fuel disposal is reduced by 64.6 ktHM and the SFR mix ratio in the nuclear fleet is estimated to be 38.0% around 2100. From these results, it is conjectured that an appropriate SFR mix ratio in the nuclear fleet around 2100 is 35.0-40.0% in the long-term nuclear power projection that corresponds to the reference and high scenarios.

Figure 2.39 illustrates reactorwise generation capacities within the total nuclear power demand for Case 9, where the SFR mix ratio in the nuclear fleet in 2100 is 38.0%. Figures 2.40 and 2.41 show the reactorwise generation capacities for Cases 8 and 10, respectively. As can be seen in Figure 2.41, where the reactor mixing strategy is sought for Case 10 based on the low scenario, the relative importance of BNs in the SFR mix is smallest compared with that for the other two scenarios. In other words, the relative importance of BNs in the SFR deployment would be increased with more emphasis on nuclear power expansion by employing PWRs as a main nuclear power system. The role of BNs for waste management would become more important at the early SFR deployment stage.

Figure 2.37. Accumulated PWR spent fuel arisings

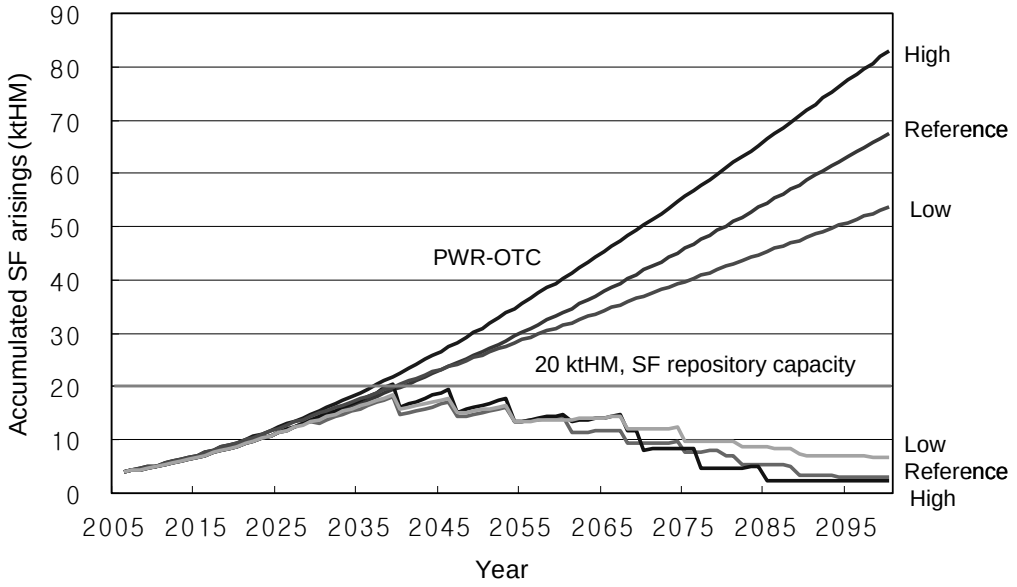


Figure 2.38. Accumulated uranium demand for PWRs

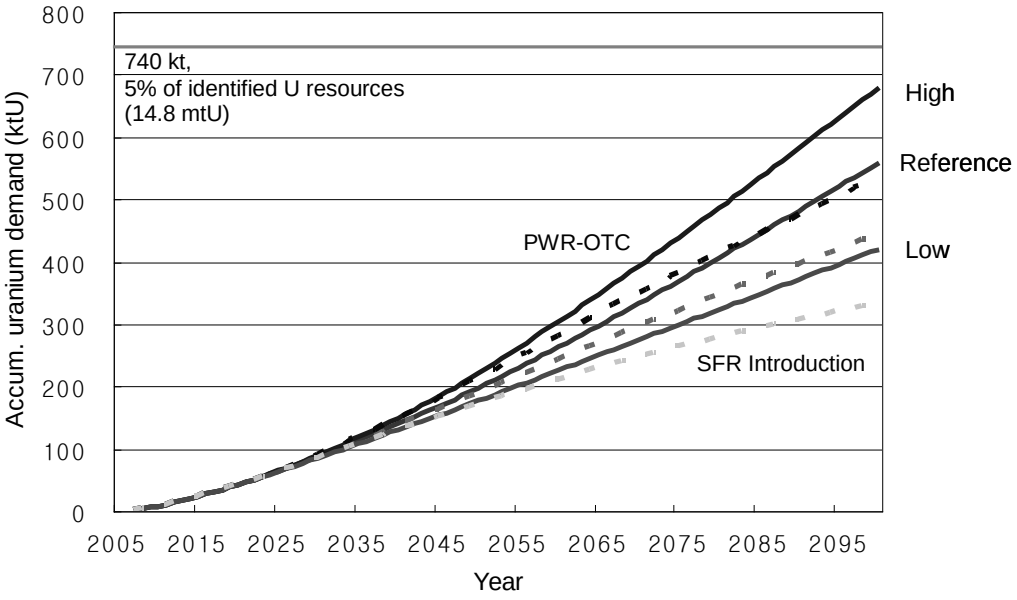


Figure 2.39. Reactorwise nuclear capacities (Case 9; reference scenario)

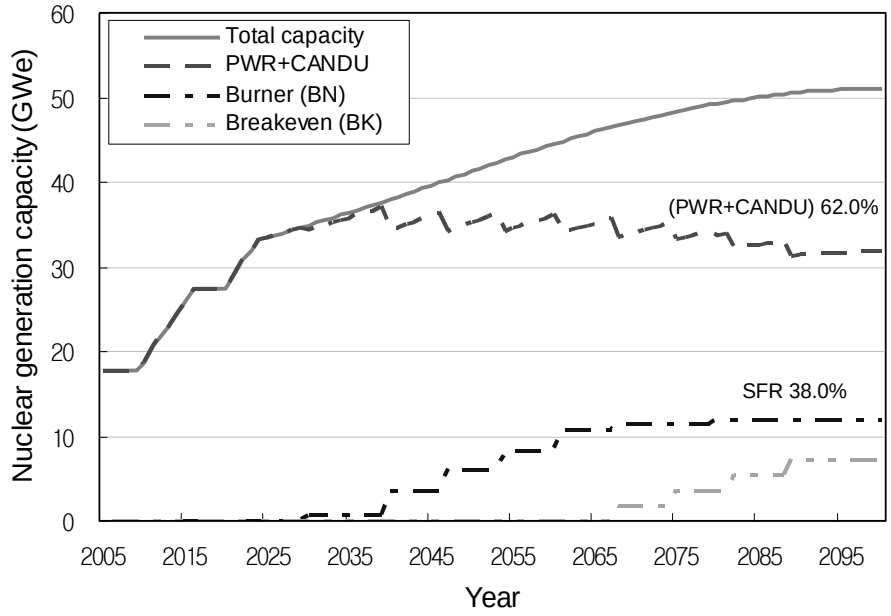


Figure 2.40. Reactorwise nuclear capacities (Case 8; high scenario)

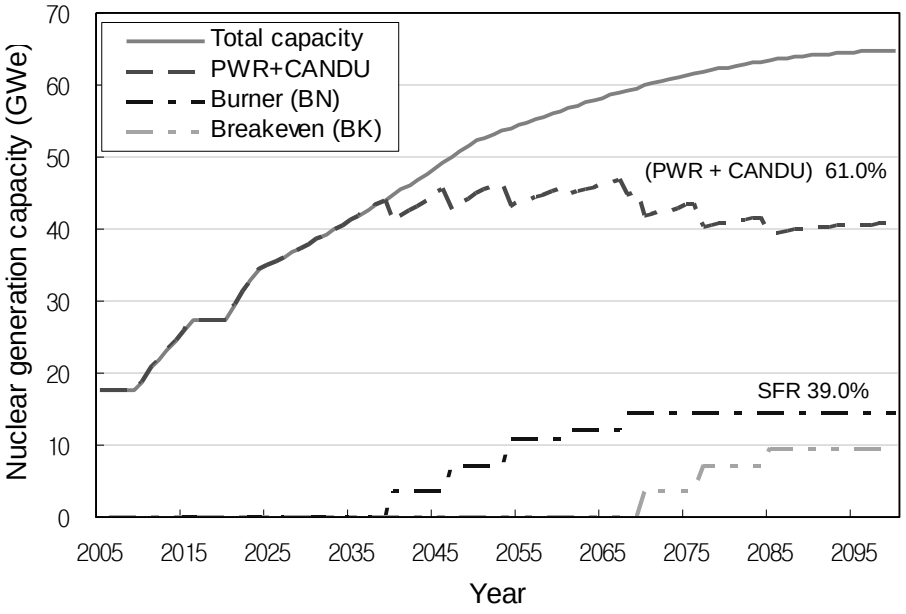
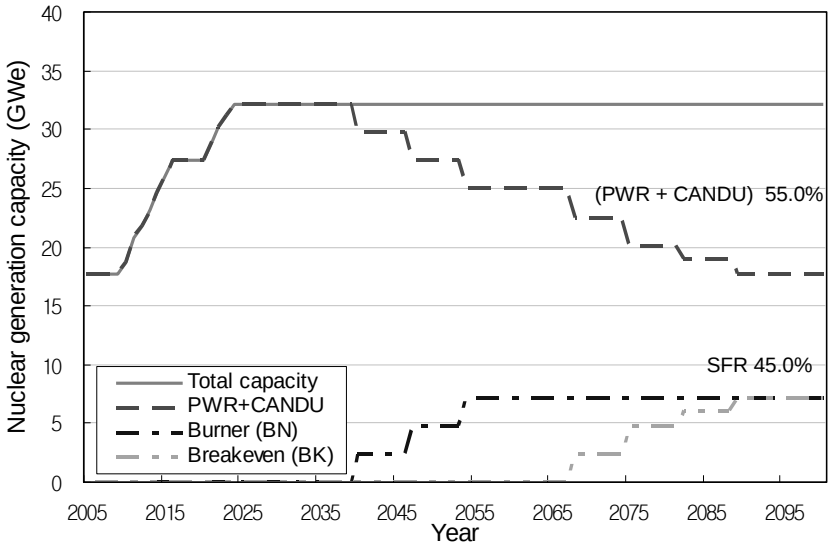


Figure 2.41. Reactorwise nuclear capacities (Case 10; low scenario)



From the viewpoint of nuclear reactor evolution up to 2100, drawn based on the most appropriate SFR deployment scenario (Case 9), an appropriate SFR mix ratio around 2100 is estimated to be 35.0-40.0% in the long-term nuclear power projection. SFRs are to be deployed in support of substantial reduction of PWR spent fuel at the first stage of deployment. From the viewpoint of spent fuel management, it would be desirable to continuously deploy SFRs in the nuclear fleet even after 2100 so as to build a symbiotic nuclear power system consisting of PWRs and SFRs, in which PWRs fuel SFRs.

2.6.3 Conclusion

An efficient reactor deployment strategy with SFR introduction starting in 2040 is drawn, based on the most appropriate SFR deployment scenario where burners are deployed prior to breakeven reactors in order to substantially reduce PWR spent fuel at early deployment stage. The SFR mixing ratio in the nuclear fleet around 2100 is estimated to be about 35-40%. PWRs will remain as a main power reactor type till 2100 and SFRs will be in support of waste minimisation and fuel utilisation.

The use of SFRs and recycling of TRUs by reusing PWR spent fuel leads to the substantial reduction of the amount of PWR spent fuel and environmental burden by decreasing radiotoxicity of high-level waste, and a significant improvement on the natural uranium resources utilisation.

2.7 Reducing phase-out time in Spain through the exchange of equivalent TRUs with a plutonium-utilising country*

The management of high-level nuclear wastes, produced mainly as spent fuel in nuclear power plants dedicated to electricity production, is a matter of continuing concern in many countries. Phase-out of electricity production from nuclear fission remains one possible option for countries such as Spain. In this case, one solution proposed for the management of the high-level wastes is the use of partitioning and transmutation (using an ADS in this study) to minimise the transuranium (TRU) inventory in the final storage and eventually to simplify that final storage.

* It should be noted that this study does not reflect any particular strategy proposed by the Spanish authorities.

The objective of the studies that should be undertaken are to evaluate the possible reductions on TRU mass, the time required to achieve that reductions, the need and time profile of resources (new ADS or reactors, reprocessing capacity, fabrication capacity, etc.) and finally the financial implications.

In previous studies [27-29], two phase-out scenarios have been conceptually discussed:

- direct TRU transmutation on fast inert matrix ADS (with a pseudo-equilibrium fuel);
- one pass of Pu on MOX followed by TRU transmutation on ADS.

In both studies, the phase-out was undertaken independently by a country employing its own facilities and TRUs. These studies indicated the need for very long periods to substantially reduce the amount of TRUs (150 years to reduce them by a factor of 25).

The present study explores the possibility of reducing the phase-out period by employing the facilities of, and exchanging "equivalent TRUs" with, a country utilising plutonium for energy production with a closed fuel cycle. The main objective of this scenario is to reduce the phase-out time, while respecting reasonable hypotheses on the deployment of the facilities.

In the previous studies, after fixing the ADS design and the choice of a pseudo-equilibrium fuel, the main constraints on the phase-out duration were:

1) the peak LWR reprocessing capacity;
2) the delay introduced in the availability of TRU from the ADS reprocessing;
3) the progressive reduction of ADS installed power needed to reach large reduction factors (as a consequence of the remaining last cores of each ADS).

In the new proposal, the regional collaboration between a country in phase-out (Ph) and another country with a large nuclear power park installed, user of advance reprocessing for Pu utilisation (PUC), presents some advantages:

1) The reprocessing of the LWR spent fuel of the phase-out country can be performed in the PUC facilities (paying for the service).
2) Constraints 2 and 3 are eliminated by exchanging equivalent amounts of TRUs between Ph and PUC.

The present study evaluates only the technical possibilities of the proposal, however the large leg and political difficulties should be evaluated somewhere else. There could also be non-negligible difficulties associated with the transport of sensible materials between different components of the scenario.

2.7.1 Scenario hypotheses

The main hypothesis of the scenario evaluated is the principle of TRU equivalence. In soft form, this implies that the TRU contained in the LWR spent fuel of both countries (Ph and PUC), irradiated under similar conditions (similar reactor and burn-up), can be exchanged (different time periods). In strong form, the principle of TRU equivalence implies that even TRU from different reactors and

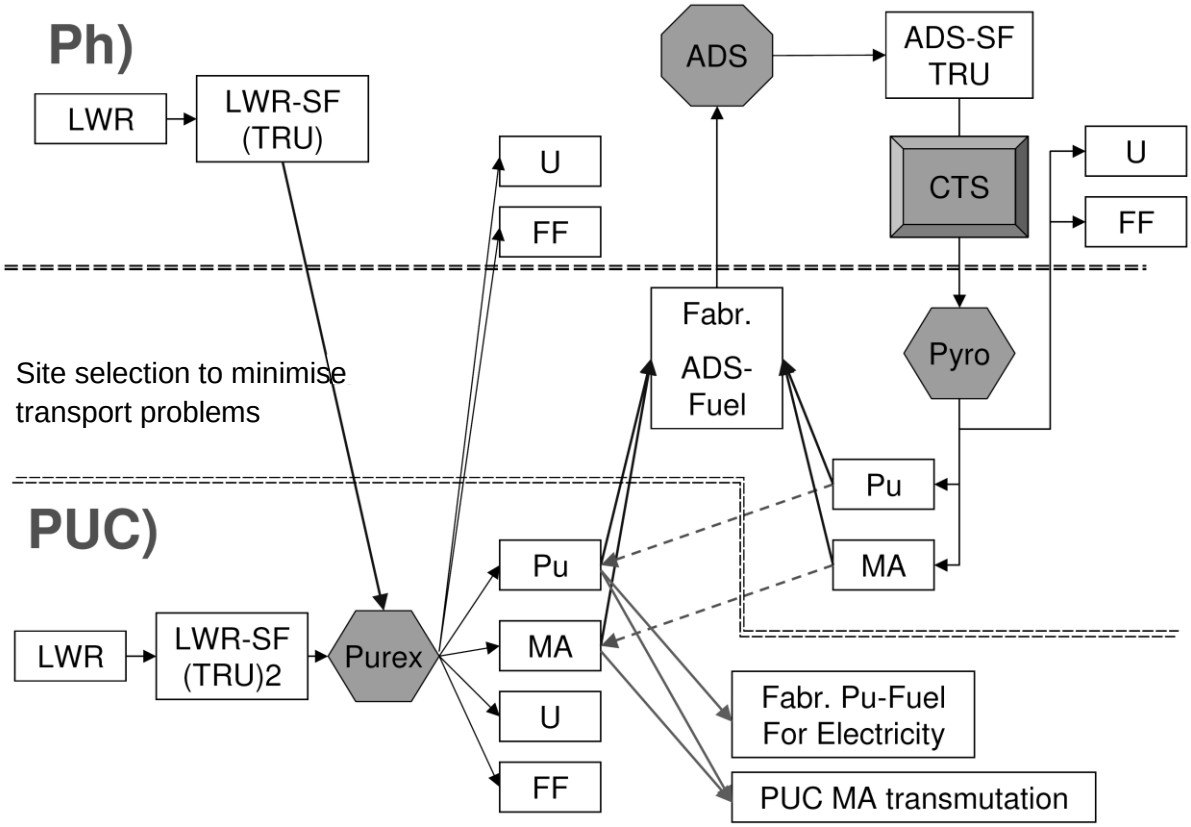
burn-ups (LWR and ADS), having different isotopic content, can be exchanged respecting the total mass, if both countries can profit from the exchange and there is some kind of correspondence in the quality of the TRU. In the present study, both the strong and soft TRU equivalences are assumed:

1) In the first stages of the phase-out, the use of TRU from the PUC spent fuel is authorised, as if it were from Ph, just after the decision to reprocess and without the need to wait for the actual reprocessing of the Ph spent fuel. Even more minor actinides (MA) from the PUC spent fuel than that contained in the Ph spent fuel are used to complete the first loads of the ADS (see Figure 2.42).

2) In the middle of the phase-out period, MA from the PUC are used to complete the reloads of the ADS. At the same time some Ph Pu is returned to PUC.

3) At the end of the phase-out the Pu and MA contained in the last transmutation ADS cores are returned to the PUC.

Figure 2.42. Details of the proposed scenario



The phase-out is finished when the total amount of TRU converted in fission fragments reaches the amount of TRU from Ph LWR. Globally, MA from the PUC LWR are exchanged for a mixture of Ph LWR Pu, and Pu and MA from the ADS recycling and last cores.

The proposed data for the scenario are:

- A total amount of 100 tonnes of TRUs, produced from a total installed power of 23.5 GWth during 50 years equivalent with an average load factor of 80% and a final average burn-up of 40 GWd/THM. The LWR power decreases linearly to 0 during the last 20 years (40 years of constant LWR installed power and 20 years of linearly decreasing power, with a start date of linear reduction ~2030), as shown in Figure 2.43.

- Use of a fixed ADS design with an initial pseudo-equilibrium inert matrix fuel (60/40 for Pu/MA, TRU MOX on ZrO_2) and with the characteristics shown in Table 2.15. The isotopic composition of the TRUs at BOL and EOL is shown in Table 2.16.

- The ADS installed power is chosen taking into respecting:
  - as high installed power as possible by other constraints;
  - a transmutation ADS plant lifetime close to 60 years;
  - a continuous progressive reduction of the total nuclear installed power.

Figure 2.43. Total power installed in the scenario

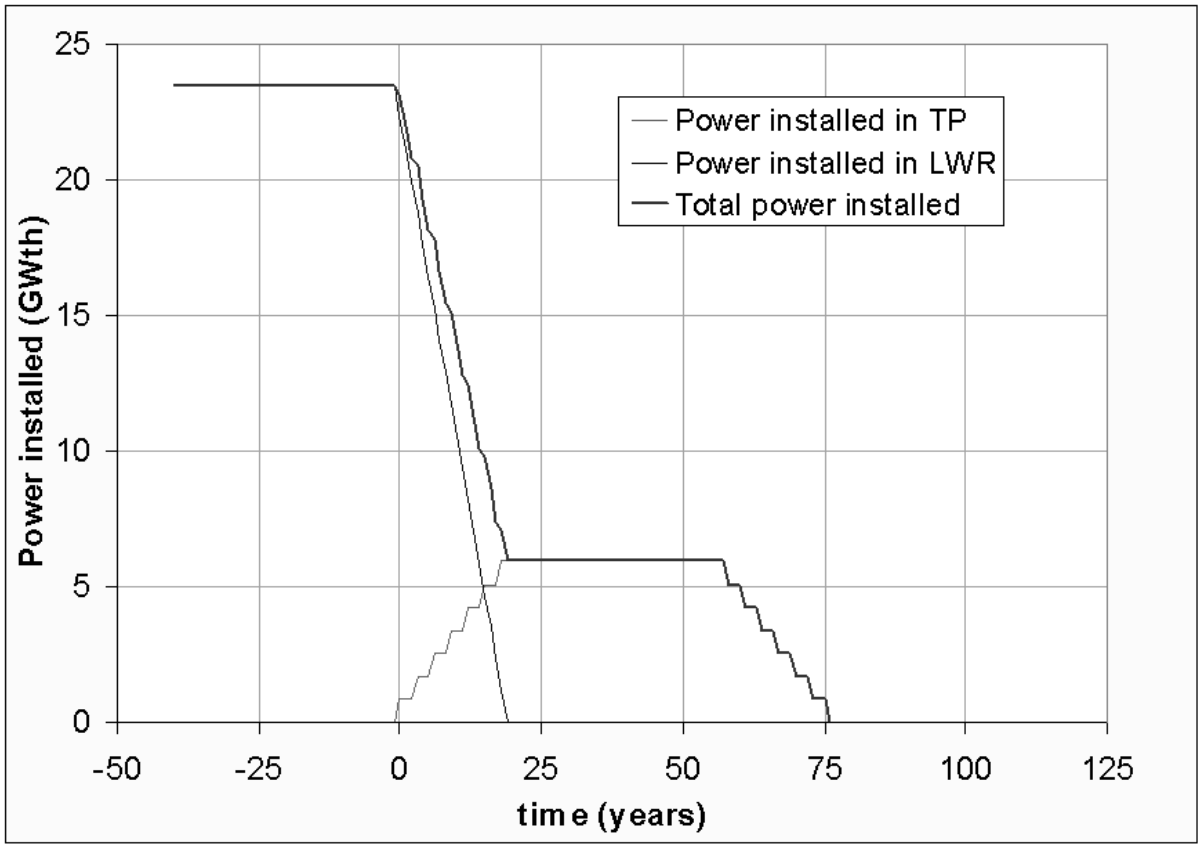


Table 2.15. ADS characteristics

Transmutation plant (TP) power	850 MWth
Initial HM = TRU fuel mass per TP	~3 tonnes
TP burn-up per cycle	150 GWd/THM
TP cycle length	1.75 years
TP load factor	80%
BOL k_{eff}	0.96-0.97

Table 2.16. Isotopic composition of the TRUs at charge and discharge of the ADS

Isotopes	Mass fraction %	Element fraction %	Mass fraction %	Element fraction %	Mass fraction %	Element fraction %
	TRU-LWR		TRU-BOL		TRU-EOL	
^{234}U					0.044	
^{235}U					0.0035	
^{236}U					0.0027	
^{238}U					0.00001	0.05
^{237}Np	5.61	5.61	16.31	16.31	13.36	13.36
^{238}Pu	1.96		1.36		7.21	
^{239}Pu	50.92		35.42		28.15	
^{240}Pu	22.34		15.54		18.84	
^{241}Pu	5.88		4.09		4.14	
^{242}Pu	5.15		3.59		4.84	
^{244}Pu		86.24	0.00025	60.00	0.0007	63.18
^{241}Am	6.59		19.16		15.59	
^{242m}Am	0.021		0.061		0.739	
^{243}Am	1.25	7.86	3.62	22.84	3.46	19.79
^{242}Cm	0.00005		0.00015		1.61	
^{243}Cm	0.004		0.011		0.097	
^{244}Cm	0.266		0.774		1.66	
^{245}Cm	0.020		0.059		0.229	
^{246}Cm	0.003		0.0079		0.019	
^{247}Cm	0.00003		0.00009		0.0007	
^{248}Cm		0.29	0.00001	0.85	0.00003	3.62

Figure 2.42 shows the details of the proposed scenario. There is an intermediate zone between Ph and PUC. It will have to be decided (or negotiated) where to build the pyroreprocessing and ADS fuel fabrication facilities in order to minimise transport problems. The U and FF wastes from PUREX going to Ph corresponds to those generated in the spent fuel including the initial 100 tonnes of Ph TRUs.

Finally, the number of ADS transmutation plants is fixed at seven. For each ADS there are three interleaved fuel core sets.

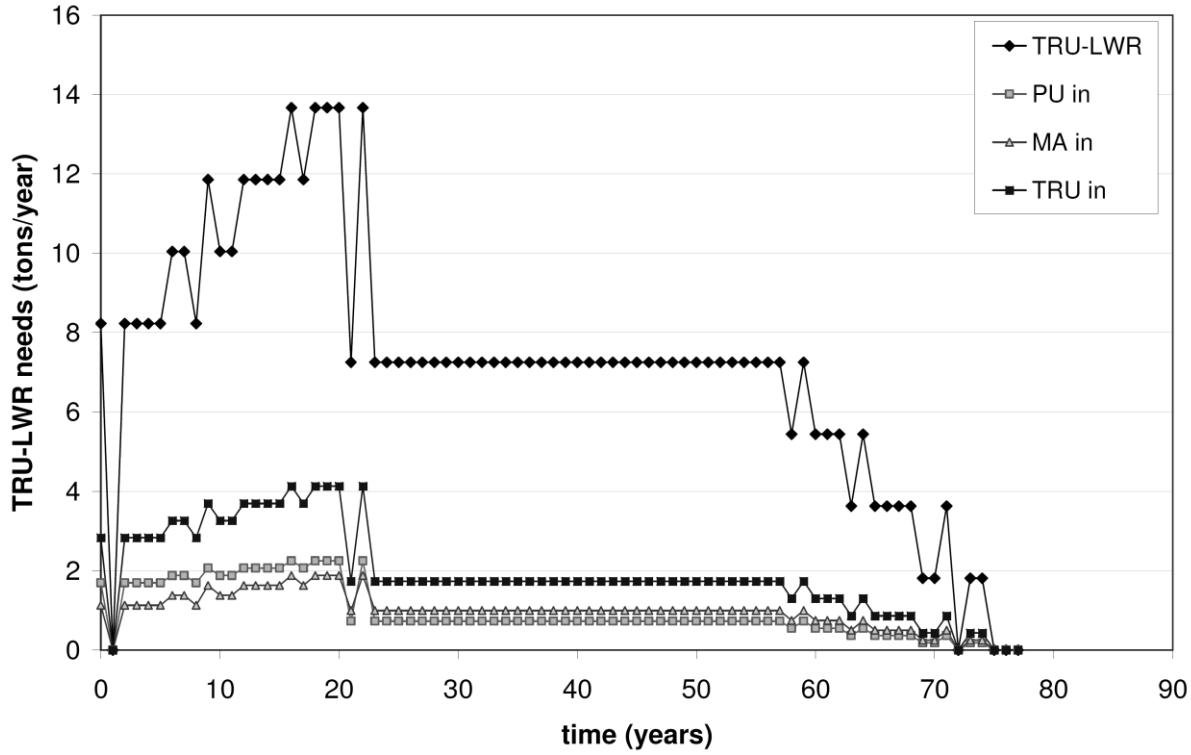
2.7.2 Results

To transmute 100 tonnes of LWR TRUs using seven ADS transmutation plants with three interleaved cores each, the scenario employs a total of 33 ADS cycles, corresponding to 58-year ADS lifetime. Each new ADS core is delayed 1.75 years (cycle length) and each new ADS transmutation plant begins operation every three years. The result is a total phase-out duration of 78 years.

According to these results and the fixed ADS characteristics, it can be extrapolated that the maximum ADS installed power of this scenario is 25% of the maximum LWR installed power, as shown in Figure 2.43. For this and the following figures, year zero is considered to be the year when the first transmutation plant begins operation (~2030).

Figure 2.44 displays the TRU needs by year to load in the ADS transmutation plants. This result is shown as "TRU in" in the figure and it is the addition of the amounts of Pu and MA, displayed in the figure as "Pu in" and "MA in", respectively. This figure also shows the total amount of TRU-LWR necessary to extract the TRU needed to upload the ADS ("TRU-LWR" line). This value is greater than the total TRU in the ADS fuel because the Pu/MA ratio in the LWR spent fuel (86/14) is greater than in the ADS fuel (60/40).

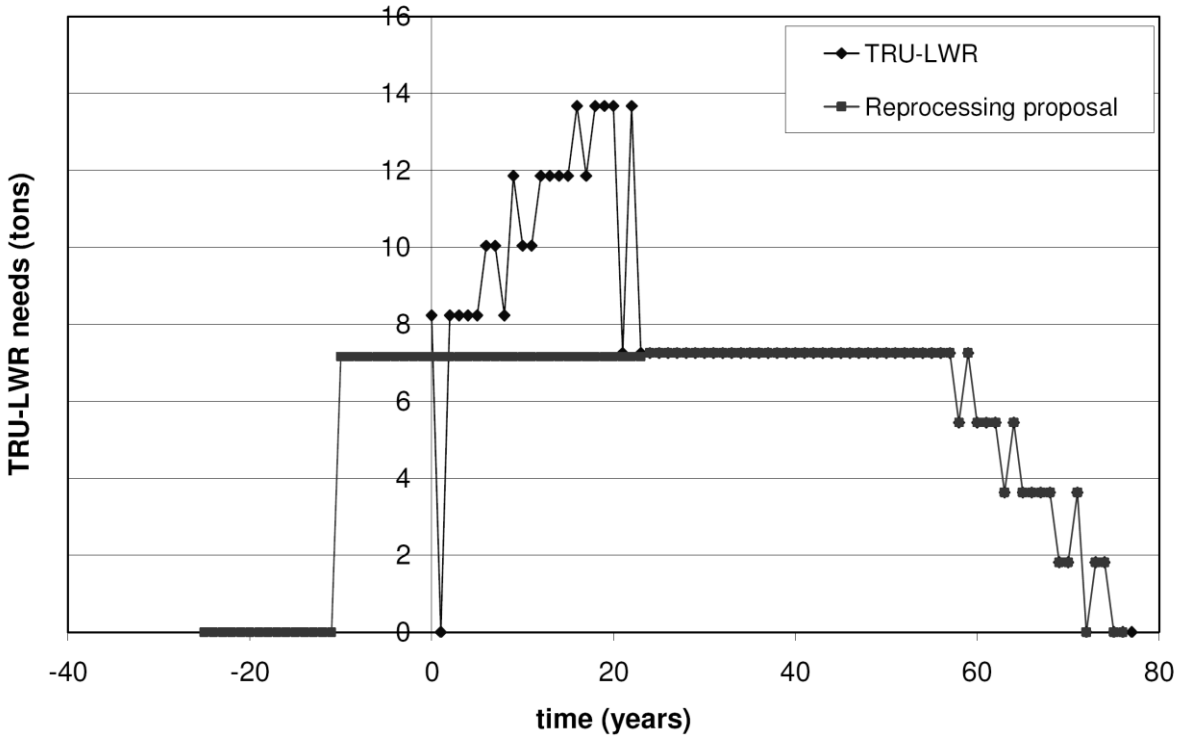
Figure 2.44. TRU-LWR needs by year to load in ADS



In the first 20 years approximately, there is a larger need for Pu in the fuel, mainly because of the greater ratio of Pu (60%) in the first cores. After this period of time, without new first cores, there is only TRU need for refuelling and the ratio of MA to refuel is larger (the ADS consumes more MA than Pu as shown in Table 2.16).

Figure 2.45 displays a reprocessing proposal to avoid the peaks in the TRU-LWR needs. If advanced PUREX reprocessing is started 11 years before the start of the first plant, the required LWR reprocessing capacity is limited and maintained constant at nearly 7.2 tonnes of TRU/year, which is smaller than the present La Hague plant yearly capability. A similar TRU mass pyroprocessing capacity of 9.6 tonnes/year is needed, although corresponding to a large difference in spent fuel mass to be reprocessed.

Figure 2.45. Reprocessing proposal for the TRU-LWR needs



To produce the initial cores and all the top-ups of the reloads as previously described, a total of 74.3 tonnes of Pu and 76.2 tonnes of MA must be obtained from the LWR spent fuels. This corresponds to a total of 553.7 TRU tonnes extracted (477.5 tonnes of Pu and 76.2 tonnes of MA), divided in 100 tonnes from Ph and 453.7 from PUC. This latter value means that the minimum installed power in PUC must be 4-5 times the Ph installed power.

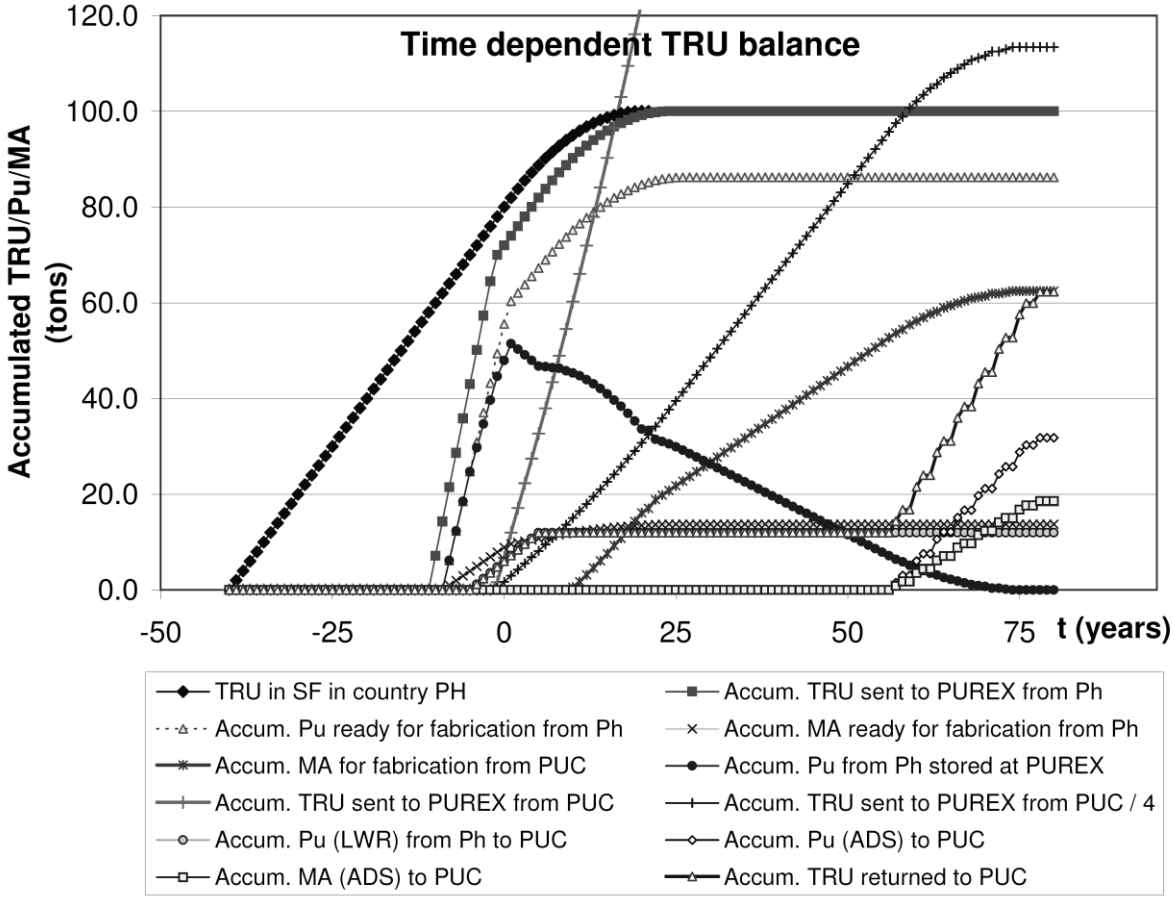
Figure 2.46 displays the time evolution of the TRU balance, including several groups of lines. The first group of lines shows the time generation of the TRUs in the Ph LWR: the total accumulated amount of TRUs generated (during the 40 years of constant LWR installed power and 20 years of linearly decreasing power) and the accumulated amount of TRU, Pu and MA sent to PUREX from Ph (finally 100 tonnes of TRUs, with 86.24% of Pu and 13.76% of MA). To calculate the TRUs sent to PUREX from Ph, the reprocessing proposal shown in Figure 2.45 has been employed, therefore the delivery of TRUs begins 11 years before year zero. Another consequence of the reprocessing proposal employed is the accumulation of separated Pu, stored at the PUREX plants while awaiting fuel fabrication. The accumulated quantity of separated Pu is also shown in this figure.

The second group of lines contains the information on the accumulated amount of TRUs sent to PUREX from PUC. It is necessary to use the PUC TRUs two years before the start of the first plant so as to provide the required quantity of TRUs for ADS fuel fabrication. Two lines show this information, one of them with the prior value and other with this accumulated amount of TRUs divided by four, with the purpose of showing its evolution in the same figure scales as the other lines. As mentioned earlier, the total value of TRUs sent to PUREX from PUC is 453.7 tonnes. The accumulated amount of MA sent to fabrication from PUC is also displayed. In this sense, an accumulated total of 62.4 tonnes of LWR MA are borrowed from the PUC LWR to produce the ADS fuel. The third group of lines shows the accumulated quantities of TRUs (Pu and MA) returned to PUC. They are returned as:

- 12.0 tonnes of Pu from Ph LWR spent fuel;
- 31.8 tonnes of Pu from the last cores of the ADS transmutation plants;
- 18.5 tonnes of MA from the last cores of the ADS transmutation plants.

The figure also shows the total amount of TRUs returned, which, at the end of the phase-out, is equal to the total quantity of MA sent to fabrication from PUC (and borrowed).

Figure 2.46. Time evolution of the TRU balance



From the 477.5 tonnes of Pu separated in the PUC LWR reprocessing, 391.2 tonnes of Pu have no use for Ph. This rest and the Pu returned (a total addition of 435 tonnes of Pu) can be used by PUC for electricity production. According to these results, only a 10.1% of the Pu employable by PUC is under the applicability of the principle of TRU equivalence in its strong form.

2.7.3 Conclusions

The regional collaboration of a country performing phase-out and a country with sustainable nuclear energy and Pu utilisation could provide interesting advantages. If the principle of TRU equivalence is accepted:

- The possible TRU mass reductions can be above a factor 100 in less than 80 years, depending on the efficiency on partitioning.

- The maximum ADS installed power proposed is 25% of the maximum LWR installed power.

- The minimum installed power of the country with a large nuclear power park installed, PUC, must be four to five times larger than the installed power of the country in phase-out, Ph.

- A limited advance PUREX capacity is needed, being comparable (but smaller) to the present La Hague plant yearly capabilities.

- Limited pyroreprocessing capacity requirements.

The principle of TRU equivalence, in its strong form, applies to 10.1% of the Pu employable by PUC and also implies a reduction of a factor larger than three (with change in the isotopic composition) in the amount of PUC MA.

Non-negligible legal and political difficulties need to be resolved before implementing this type of collaboration. In addition, minimisation of transport of separated materials requires particular attention when selecting the sites of different facilities.

2.8 Scenarios for transition in the United States nuclear fuel cycle

The United States is currently storing spent commercial reactor fuel that contained approximately 52 000 metric tonnes of heavy metal (MTHM) prior to irradiation. Almost all of that fuel is UO_2 fuel, initially enriched to <5 wt/o in ^{235}U and now stored at the reactor site. The quantity of stored spent fuel is increasing by about 2 000 MTHM per year.

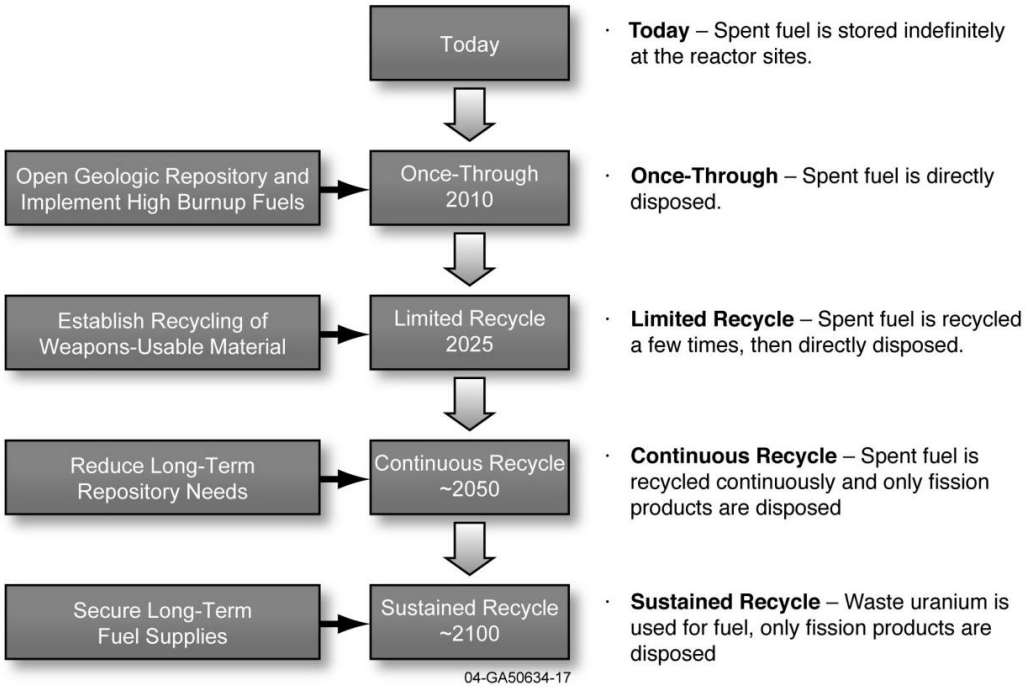
The United States reactor fleet consists of 103 light water reactors, capable of generating 98.8 GWe. Thirty of the 103 reactors have received 20-year license extensions, an additional 14 have applied for license extensions and 25 more are expected to apply within the next six years. In addition to these license extensions, the US Nuclear Regulatory Commission has granted power uprates totalling 4 183 MWe, with an additional 12 uprates totalling 990 MWe now pending before the NRC and 26 additional uprates totalling 1 548 MWe expected from licensees as of July 2005. Thus the generating capability of the United States fleet can be expected to remain near 100 GWe for the next two decades. Since the capacity factor of United States reactors has been near or greater than 90% for the past five years, little increase in that parameter can be expected in the coming years.

In considering the overall United States fuel cycle the most telling aspect is the time lag between the making of a decision and any impact of that decision on the inventory and disposition of spent fuel. There is no facility for the reprocessing of commercial spent fuel in the United States today. The earliest reasonable date such a facility could be operating is 2025. The earliest possible date for the emplacement of spent fuel in the Yucca Mountain geological repository is 2012. Once emplacement in the repository begins, the process will continue for at least 25 years. A key near-term decision point is a determination by the US Secretary of Energy, required by the Nuclear Waste Policy Act of 1982, as to whether a second geological repository will be needed. In the interim, spent fuel is being transferred from spent fuel storage pools to dry storage casks, usually at the same reactor site. As of 2005, about 60% of United States reactors have filled their spent fuel pools to capacity and must move older spent fuel to dry storage to continue operating.

2.8.1 Possible transition scenarios

The various stages in the development of a long-term fuel cycle for the United States are shown in Figure 2.47. As noted above, spent fuel is being stored today at reactor sites, in anticipation that it will be moved to a geological repository in the future, after about 2010. At that time a "once-through" fuel cycle will be in operation, with the only option of increasing the capacity of the geological repository being the use of high burn-up fuels.

Figure 2.47. Potential fuel cycle strategies



In order to decrease amounts of weapons-usable materials in existence, the "limited recycle" stage of fuel cycle development is foreseen between 2010 and 2025. During this stage, the plutonium and some of the minor actinides would be recycled a few times, either as mixed-oxide fuels or as inert matrix fuels, in existing LWRs. During these few recycles, the weapons-usability of the actinides will be greatly diminished through its accumulation of ^{238}Pu, ^{240}Pu and ^{242}Cm.

2.8.2 Basis for comparing repository needs of various fuel cycle strategies

The capacity of a geological repository is limited by a range of factors, from the short-term heat load imposed by the fission products to the long-term toxicity of actinides such as ^{237}Np (half-life of 2.14 million years). For a dry repository the most severe limitation appears to be the heat load of the shorter-lived actinides, particularly ^{241}Am (half-life of 431 years). Specifically, repository capacity in the United States is limited by the maximum temperature midway between the drifts. The temperature is limited to 96°C, the boiling point of groundwater at the elevation of the site. This temperature limitation is imposed to prevent the formation of a perched water table above the emplaced spent fuel. If such a perched water table should develop, groundwater would collect above the fuel and then flood the fuel when the decay heat decreases through a critical value, usually about 1 300-1 500 years after closure. Thus the metric for comparing repository capacity is the integrated heat load from 100 years (at which time the forced ventilation is presumed to be turned off) to 1 500 years after discharge

2.8.3 Impact on eventual repository needs

Obviously, the need for future geological repositories rests both on the number of reactors in operation and on the nuclear fuel cycle being used. Figure 2.48 is a graphical representation of the implications of those two choices. The columns represent different numbers of reactors in operation in the United States and the rows represent various fuel cycle strategies.

Figure 2.48. Impact of different fuel management approaches on eventual repository needs under different nuclear futures, through 2100

Nuclear Futures		Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share Generation	Growing Market Share Generation
Cumulative discharged fuel in 2100 (metric ton)		100,000	120,000	250,000	600,000	1,400,000
		Existing Reactors Only ⟷		Existing and New Reactors		
Fuel Management Approach		Number of Repositories Needed at 70,000 Metric Ton Each				
No Recycle	Once-Through	2	2	4	9	20
	Once-Through, High Burnup Fuels	2	2	3	7	17
Reprocess & Recycle	Limited Recycle, High Burnup Fuels	Recycle not applicable		2	5	10
	Continuous Recycle			1	1	1
	Sustained Recycle			1	1	1

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The columns range on the left from operating existing reactors only through the completion of their existing licenses through a continuation of the present level of nuclear-electric generation (~900 TW-hr/yr) to a growing market share for nuclear electricity. The details of each of these scenarios are shown in Table 2.17.

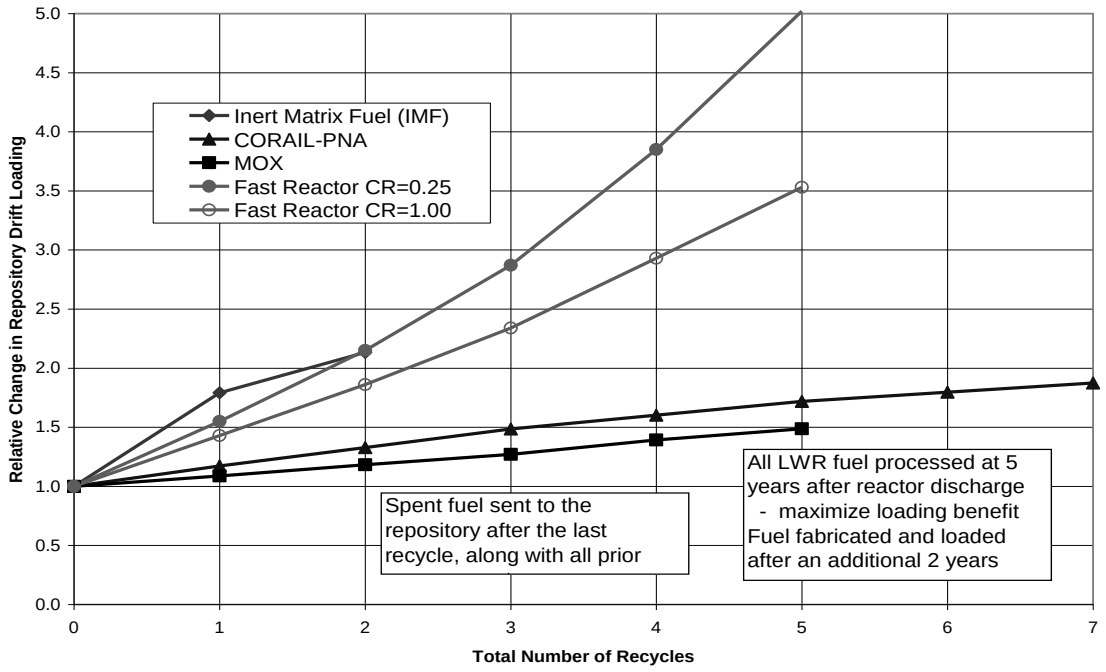
The rows in Figure 2.48 range from a once-through fuel cycle, as is currently being practiced in the United States, through continuous recycle, using thermal reactors and sustained recycle, using both thermal and fast reactors. The metric used to evaluate each of the fuel cycle/nuclear future intersections is the number of geological repositories, each with a 70 000 metric tonne capacity, needed now through 2100. Because only small amounts of the long-lived actinides are placed in a repository when continuous recycling or sustained recycling is practiced, those two fuel management approaches would require only one repository through 2100.

The increase in repository capacity, as defined by the maximum temperature between drifts, as described earlier, is shown for three thermal and two fast fuel management approaches in Figure 2.49. The MOX approach has the smallest increase in repository benefit per fuel cycle, while a fast reactor with a conversion ratio (CR) of 0.25 has the highest impact on repository space utilisation. The inert matrix fuel approach has the highest impact after one and two cycles, but cannot be pursued further because the fissile content of the fuel become severely depleted and, in contrast to the MOX, Corail and fast reactor approaches, no additional fissile material can be bred from the inert matrix.

Table 2.17. Details of potential future energy scenarios

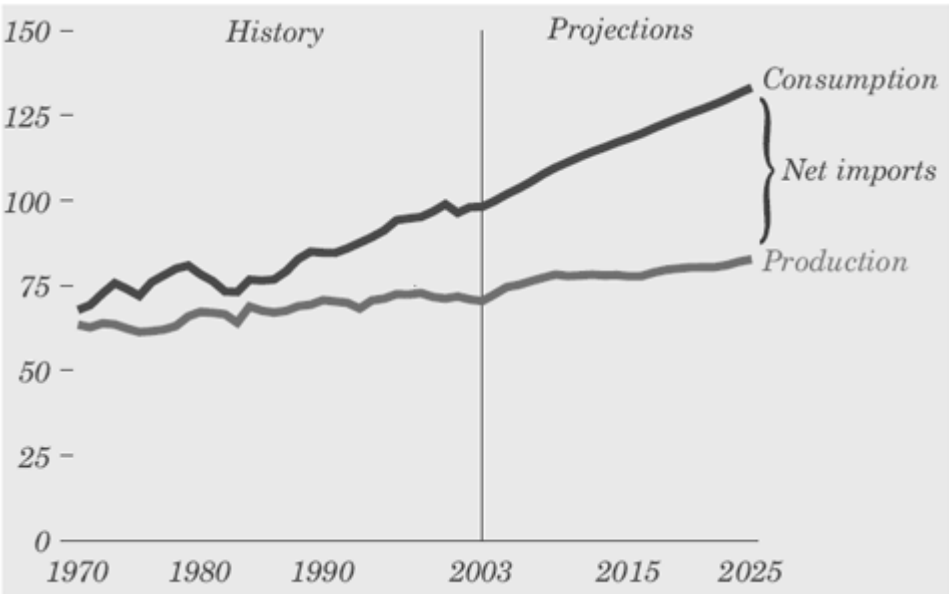
Future energy scenario	Total discharged fuel (MT = metric tonnes = 1 000 kg or 2 200 pounds)
1. Legislative limit	70 000 MT = based on the legal capacity of the first repository per the Nuclear Waste Policy Act (63 000 MT of initial heavy metal for commercial waste, 7 000 MT for defence waste).
2. Existing license completion	100 000 MT = based on existing spent fuel inventories plus a plant-by-plant extrapolation of future discharges developed using current discharge rates until the end of each operating license, including known license extensions as of 10/2003 – result rounded.
3. Extended license completion	120 000 MT = based on existing spent fuel inventories plus a plant-by-plant extrapolation of future discharges assuming on all operating plants having one 20-year extension, result rounded.
4. Continuing level energy generation	250 000 MT = based on extension of the current average annual spent fuel discharge rate of 2 100 MT/yr through the year 2100. No growth in nuclear power compared to today.
5. Continuing market share generation	600 000 MT = Extension of the current average annual spent fuel discharge rate through 2100 with 1.8% compounded market growth starting in 2004. Steady electricity market share for nuclear power compared to today.
6. Growing market share generation	1 500 000 MT = Extension of current average annual spent fuel discharge through 2100 with 3.2% growth in nuclear power. Expands nuclear power market share, including potential entry into transportation market via hydrogen generation.

Figure 2.49. Potential increase in repository space utilisation with limited recycle



The calculations shown in Figure 2.50 assume that the increase in drift loading corresponds to stopping recycle after a given number of recycles. Disposal of the process waste, while continuing recycle, allows for a drift loading to be increased up to a factor of 100 over the once-through approach based on heat load considerations alone.

Figure 2.50. Total energy production and consumption in the United States, 1970-2025 ($EJ_{thermal}$)



2.8.4 Factors potentially leading to annual nuclear growth of more than 1.8%

The "Nuclear Futures" heading shown in Figure 2.48 assumes that reactors remain primarily used for the generation of electricity. However, the pressure of fossil fuel imports on the OECD economies, particular importers of petroleum, may stimulate growth of nuclear power at rates >1.8%/yr. Stimulation for more rapid growth in nuclear power may come through limitations on the emissions of greenhouse gases as well. The cost and insecurity of petroleum imports will result in the increased use of natural gas to supplant or synthesise liquid transportation fuels, reducing its use for electricity generation. In addition, the nuclear production of hydrogen will enable the upgrading of low-quality crude oil and possibly the direct use of hydrogen as a substitute for gasoline and diesel fuel. Figure 2.50 shows that the projected net imports of fuels to the United States will be 55 $EJ_{thermal}$ in 2025, of which 41 $EJ_{thermal}$ will be petroleum imports. For comparison, the 2004 total thermal output of the 103 United States reactors was about 8 $EJ_{thermal}$.

2.8.5 Conclusions

The present United States fleet of 103 LWRs will remain the dominant force in the country's nuclear energy make-up at least until 2025, through license extension and continuing high burn-up fuel development. Nuclear energy in the United States is no longer declining, as it was 10 years ago. Based on limitations on the maximum temperature between the drifts, the planned 70 000 tonne geological repository would be nearly filled with the presently existing spent nuclear fuel and that which will be produced by existing plants, even in the absence of license extension.

The driving event in the next decade in the United States will be the decision on the need for a second repository. Such a decision is to be made between 2007 and 2010, according to the Nuclear Waste Policy Act of 1982. Therefore, the primary goal of the United States fuel cycle this century will be to conserve repository capacity the fuel recycling strategies.

REFERENCES

[1] Haeck, W. and B. Verboomen, *ALEPH – A Monte Carlo Burn-up Code*, SCK•CEN (2005).

[2] Haeck, W. and B. Verboomen, "ALEPH – A Monte Carlo Burn-up Code", *International Conference on Emerging Nuclear Energy Systems (ICENES'2005)*, Brussels, 21-26 August, SCK•CEN (2005).

[3] Haeck, W. and B. Verboomen, "An Optimum Approach to Monte Carlo Burn-up", *Nuclear Science and Engineering*, American Nuclear Society (June 2007).

[4] *MCNPX User's Manual, Version 2.5.0*, Denise B. Pelowitz (ed.), LA-CP-05-0369, Los Alamos National Laboratory, USA (2005).

[5] Croff, G., *A User's Manual for the ORIGEN2 Computer Code*, ORNL/TM-7175, Oak Ridge National Laboratory, Oak Ridge, USA (1980).

[6] MacFarlane, R.E., D.W. Muir, *The NJOY Nuclear Data Processing System Version 91*, LA-12470-M, Los Alamos National Laboratory, USA (1994).

[7] Pilate, S., B. Lance, H. Gabaieff, B. Verboomen, W. Haeck, J. Kuijper and D. da Cruz, *VALMOX: Validation of Nuclear Data for High Burn-up MOX Fuels*, Final Report, Contract No. FIKS-CT-00191 (2005).

[8] Malambu, E., Th. Aoust, W. Haeck, N. Messaoudi and G. Van den Eynde, "Sub-critical Core Neutronics Design Calculations", in *MYRRHA ADS Pre-Design*, Draft 2, SCK•CEN (2005).

[9] Taiwo, T.W., T.K. Kim, F.J. Szakaly, R.N. Hill, W.S. Yang, G.R. Dyck, B. Hyland and G.W.R. Edwards, "Comparative Study of Plutonium Burning in Heavy and Light Water Reactors", *Proceedings of ICAPP 2007*, Nice, France, 13-18 May 2007.

[10] Hyland, B. and G.R. Dyck, "Actinide Burning in CANDU Reactors", *GLOBAL 2007*, Boise, Idaho, USA, October 2007.

[11] GRS Jahresbericht (2002/2001).

[12] Schwenk-Ferrero A., W. Tromm, "Potential Fuel Cycle Strategies for Transmutation of German Nuclear Fuel Legacy", *JK2006*, Aachen, Germany (2006).

[13] Schneider, E., *et al*., "NFCSim: A Dynamic Fuel Burn-up and Fuel Cycle Simulation Tool", *Nucl. Techn*., Vol. 151, No. 1, July 2005, pp. 35-50.

[14] Schneider, E., M. Salvatores, A. Schwenk-Ferrero, *et al*., *NFCSim Scenario Studies of German and European Reactor Fleets*, LA-UR-04-4911.

[15] Porsch, D., *et al*., "Plutonium Recycling in LWRs at Framatome ANP – Status and Trends", *ANFM 2003*, Hilton Head Island, USA, 2-8 October 2003.

[16] Van Tuyle, G.J., *et al*., *Candidate Approaches for an Integrated Nuclear Waste Management Strategy – Scoping Evaluations*, Los Alamos National Laboratory Report LA-UR-01-5572 (2001).

[17] OECD Nuclear Energy Agency, *Accelerator-Driven Systems (ADS) and Fast Reactors (FR) in Advanced Nuclear Fuel Cycles*, NEA-3109-ADS (2002).

[18] Smith, R.I., *et al*., *Estimated Cost of an ATW System*, Pacific Northwest National Laboratory Report PNNL-13018 (1999).

[19] Maeda, H., "Nuclear Energy in Japan – Current Status and Future", *Int. Conf. on Fifty Years of Nuclear Power – the Next Fifty Years in Russia*, IAEA, Keynote Speech (2005).

[20] ACNRE/METI, "Long-Term Outlook for Energy Supply and Demand", pp. 21-28 (October 2004) (in Japanese).

[21] OECD/NEA/IAEA, *Uranium 2005: Resources, Production and Demand*, Paris, OECD, pp. 13-22 (2005).

[22] Lee, K.B., J.W. Jang, Y.I. Kim and D.H. Hahn, "Reduction of Spent Fuel Storage by Coupling Strategy of PWR and KALIMER", *KNS Spring Meeting* (2005).

[23] OECD/NEA-IAEA, *Uranium 2005: Resources, Production and Demand* (2006).

[24] *The Third Basic Plan of Long Term Electricity Supply and Demand (2004-2020)*, Ministry of Commerce, Industry and Energy (MOCIE) announcement 2006-349, 12 December 2006.

[25] Hahn, D.H., Y.I Kim, S.O. Kim, *et al*., *KALIMER-600 Conceptual Design Report*, KAERI/ TR-3381/2007, Korea Atomic Energy Research Institute (Feb. 2007).

[26] Hong, S.G., S.J. Kim and D.H. Hahn, *Establishment of Gen IV Sodium Cooled Transmutation Nuclear Reactor Concept*, KAERI/RR-2679/2006, Korea Atomic Energy Research Institute (August 2006).

[27] Gonzalez, E., *et al.*, "TRU Transmutation Studies for Phase-out Scenarios Based on Fast Neutron ADS Systems", presented at ADTTA'01, Reno, Nevada, USA (2001).

[28] González, E. and M. Embid-Segura, "Detailed Phase-Out TRU Transmutation Scenarios Studies Based on Fast Neutron ADS Systems", *7th International Exchange Meeting on Partitioning and Transmutation*, NEA/OCDE, Jeju, Korea, October (2002). Also presented to the WPPT meeting, Paris (2003).

[29] Pérez-Parra, A., *et al*., *Transuranic Transmutation on Partially Fertile (U-Zr) Matrix Lead-Bismuth Cooled ADS*, CIEMAT-DFN/TR01/PC01 (2001). Presented by M. Embid, *et al*. at Global 2001, Paris (2001).

Chapter 3

KEY TECHNOLOGIES

The following are identified as crucial areas towards the implementation of advanced fuel cycles:

- fuels for LWR recycle (from standard Pu recycle to TRU recycle. This last option, probably impractical, see below);
- fuels for fast reactor recycle (fuels for homogeneous or targets for heterogeneous TRU recycle, dedicated fuels, *e.g.* for MA consumption);
- fuels for HTGR recycle (from U fuels to deep Pu burners);
- separation technologies (both aqueous and pyroprocesses);
- advanced systems (critical or subcritical), and related technologies (*e.g.* specific coolant technology, materials).

The following tables summarise for each potential option within each area, perceived advantages, development needs and estimated time to implementation, together with the indication of the countries interested in a specific technology. Some comments are also included, when appropriate.

Table 3.1. Fuels for LWR recycle

Fuel type	Perceived advantage	Countries interested	Development needs	Time to implementation	Comments
U oxide	Current industrial practice; potential for decreasing waste; impact large scale commercial deployment.	Most countries.	Increased burn-up and acceptable reliability at high burn-up.	2015-2020	Standard burn-up fuel (>70 GWd/t) is available now.
U-Pu oxide	Considerable industrial experience; way to reduce Pu stockpiles.	Belgium, France, Germany, India, Korea, Russia, Switzerland and USA.	Improved remote fabrication methods.	2025	
U-Pu-Am oxide	Reduced attractiveness of recycled material; some level of management of MA stockpiles.	No one at this time.	Chemical method for separation of Am from Cm; development of remote fabrication methods; complete fuel qualification testing programme; special plant needed/	2030-2040	May meet with resistance from utility operators; benefits for minor actinide management limited.
U-TRU oxide	Reduced attractiveness of recycled material; only losses at reprocessing sent to repository.	Research mode only.	Development of remote fabrication methods; complete fuel qualification testing programme.	2035	Very high neutron dose from fuel assembly will require remote handling at all times.
Pu oxide inert matrix fuel	Efficient consumption of Pu, essentially to get rid of fissile Pu.	Switzerland (paper study), some studies sponsored by EU countries. One fuel irradiation study underway.	Development of inert matrix material and of reprocessing methods; development of fabrication methods.	2030	Very limited irradiation performance data for inert matrix fuel
TRU oxide inert matrix fuel	High burn-up capability.	Research mode only.	Development of inert matrix material and of reprocessing methods; development of fabrication methods.	2045	Very limited irradiation performance data for inert matrix fuel. Build-up of higher mass actinide. Neutron dose from fuel assembly will require remote handling at all times. Only calculations, practically no work.

Table 3.2. Fuels for fast reactor recycle*

Fuel type	Perceived advantage	Countries interested	Development needs	Time to implementation	Comments
Oxide (U-Pu)	Already industrialised technology; current industrial practice.	China, France, India, Japan, Korea, Russia and UK.	Known.	2025	Not on critical path; availability of fast irradiation facility (20 years) more limiting.
Oxide (U-TRU oxide)	Highest level of technological maturity.	France, Japan, UK and USA. Gen-IV irradiation project in MONJU.	Validation of ceramic properties with minor actinide content (fabrication issue); fast reactor irradiation of minor actinide bearing fuels. Irradiation facilities availability.	2030	Homogeneous TRU recycle. MA content depends on reactor size and coolant technology (3-10%). Neutron dose increase at fuel fabrication.
Metal (U-TRU-Zr)	High level of technological maturity; highly favourable safety characteristics in SFR application.	France, Japan, Korea and USA.	Demonstration of fabricability of minor actinide (*i.e.* Am) bearing fuels; fast reactor irradiation of minor actinide bearing fuels. Irradiation facilities availability.	2030	Homogeneous TRU recycle. MA content depends on reactor size and coolant technology (3-10%). Utilisation in lead-cooled reactor would require use of different thermal bonding material and confirmation of chemical compatibility with fuel. Know how to do Na bonding – not Pb or Pb-Bi. Neutron dose increase at fuel fabrication.
Nitride (UN-TRU N-ZrN)	Complete solubility of actinide nitrides; irradiation stability of fuel at normal operating temperatures; amenable to aqueous or non-aqueous reprocessing.	Russia.	Development of efficient fabrication methods; fast reactor irradiation testing. Irradiation facilities availability.	2040	Potential issue with dissociation of nitrides at accident temperatures. Might require ^{15}N enrichment. Neutron dose increase at fuel fabrication.

* Deployment is limited by lack of fast reactor testing capability at the scale required.

Table 3.2. Fuels for fast reactor recycle *(cont.)*

Fuel type	Perceived advantage	Countries interested	Development needs	Time to implementation	Comments
Carbide (UC-TRU C-SiC)	High-temperature capability.	France.	Development of new fuel forms and efficient fabrication methods; fast reactor irradiation testing. Irradiation facilities availability.	2040	Homogeneous TRU recycle. MA content depends on reactor size and coolant technology (3-10%). If used for GFR, new fuel forms are possible: advanced fuel particles, cellular plate fuel concept, advanced pin fuel concept. Neutron dose increase at fuel fabrication.
Targets for heterogeneous MA recycling	Separation (in the reactor core and in the fuel cycle) of "standard" Pu-bearing fuel and (high concentration) MA-bearing fuel. Potentially, only a fraction of the fast reactor to be deployed should be loaded with MA targets in special fuel subassembly.	France.	Development of appropriate matrix: inert or uranium. Fabricability in presence of high content of MA (Cm). Need for irradiation tests. Irradiation facilities availability.	2035-2040	Potential difficulties related to high thermal power (both at beginning and end of irradiation), and high He production. A larger part of the fast reactor fleet to be loaded with MA targets, if MA content should be limited.
Dedicated fuels for MA transmutation	Can be used for MA transmutation in a separate stratum of the fuel cycle. If ADS are used, practically any MA/Pu ratio can be envisaged. Dedicated fuels can in principle be oxide, metal, nitride or carbide.	Belgium, France, Germany, Korea, Japan, Russia, Spain, Sweden, and Russia.	Development of appropriate matrix: inert or uranium. Fabricability in presence of high content of MA (Cm). Need for irradiation tests. Irradiation facilities availability.	2035-2040	If U-free fuel, inert matrix choice should accommodate fabrication, spent fuel processing and core constraints. U matrix can allow up to 80% of maximum theoretical MA consumption.

Table 3.3. Fuels for HTGR recycle

Fuel type	Perceived advantage	Development needs	Time to implementation	Comments
TRISO UO_2	Prior experience with this fuel type in the Germany and the US.	Development of fuel fabrication technology; irradiation testing to confirm fuel integrity. Determination of fuel behaviour in repository in case of direct disposal.	2017	May be prone to kernel migration during irradiation to high burn-up.
TRISO UCO	Similarity to TRISO UO_2 fuel; resistance to kernel migration.	Development of fuel fabrication technology; irradiation testing to confirm fuel integrity. Determination of fuel behaviour in repository in case of direct disposal.	2022	More complex kernel preparation method required (essentially a mixture of UC_2 and UO_2).
TRISO PuO_2	Potential high burn-up capability.	Development of fuel fabrication technology; irradiation testing to confirm fuel integrity. Determination of fuel behaviour in repository in case of direct disposal.	2025	Plutonium consumption application.
TRISO U/TRU oxycarbide	Deep burn concept.	Development of fuel fabrication technology; irradiation testing to confirm fuel integrity. Determination of fuel behaviour in repository in case of direct disposal. Development of reprocessing technology for two-pass case.	2030	Validation of core physics analysis required. Potential high build-up of higher mass actinides.

Table 3.4. Separation technologies

Technology type	Perceived advantage	Development needs	Time to implementation	Comments
PUREX	Extensive industrial experience base. Possible minimum-cost approaches for U-Pu MOX recycle fuel.	Continuous optimisation and waste reduction. Np and Tc recovery.	Under way	Not acceptable for US applications.
Extended PUREX	Continuity with PUREX process.	Demonstration on a few tens of kilogrammes of spent fuel performed by CEA at CBP facility in ATALANTE in 2005. Need to deploy a facility to process ~1 tonne spent fuel.	2015	Step 1 (DIAMEX): partitioning of the actinides (Am +Cm) and lanthanides from the fission products. Step 2 (SANEX): partitioning actinides (Am +Cm) from lanthanides. Step 3: partitioning Am from Cm.
NEXT	Removal of excessive uranium to reduce process solution for economical advantage by crystallisation and co-recovery of remaining U, Pu and Np by simplified solvent extraction. Recovery of Am, Cm from high active waste by extraction chromatography.	Confirmation of chemical flow sheet at chemical process facility in 2003-2006. Pilot-scale demonstration for process and engineering scale equipment validation in TOKAI site.	2015	This process has an advantage of economic, environmental burden and non-proliferation in comparison with PUREX.
GANEX	Optimum strategy for not-separated TRU recovery.	Demonstration in hot Lab at ATALANTE. Micro-pilot installation to be developed at La Hague.	2008-2012 2015-2020	International experiment (GACID) in the framework of Gen-IV.
UREX+1	No separation of plutonium; group extraction of the TRU.	Pilot-scale demonstration for process validation.	2030	TRU are stored pending a decision on fast or thermal recycle.
UREX+2	Pu+Np product is readily amenable to fuel fabrication without requiring remote handling of the fabrication facility.	Pilot-scale demonstration for process validation.	2025	Am+Cm are co-recovered and stored with lanthanide fission products pending the availability of fast reactors for burning.

Table 3.4. Separation technologies *(cont.)*

Technology type	Perceived advantage	Development needs	Time to implementation	Comments
UREX+3	Pu+Np product is readily amenable to fuel fabrication without requiring remote handling in the fabrication facility.	Pilot-scale demonstration for process validation.	2025	Am+Cm are co-recovered and stored (after removal of lanthanide fission products) pending the availability of fast reactors for burning.
UREX+4	Pu+Np product is readily amenable to fuel fabrication without requiring remote handling in the fabrication facility.	Pilot-scale demonstration for process validation. Development of process for separation of Am from Cm.	2030	Cm is recovered separately and is stored for decay. The Am is also recovered separately and can be stored or added to the Pu+Np product to reduce material attractiveness.
Grind/Leach[a]	Technical feasibility established; capable of efficient actinide recovery.	Pilot-scale demonstration of economic and environmental viability.	2030	Problem with disposal of large quantities of carbon (including ^{14}C) persists.
METROX[b]	Pyrochemical alternative to aqueous processing.	Process development and verification; pilot-scale demonstration.	2035	At a very early stage of concept development.
PYROX[c]	Pyrochemical alternative to aqueous processing.	Laboratory tests with hot fuel to assess the ability of the process to handle the presence of fission products and minor actinides; pilot-scale demonstration if warranted.	2025*	Because it does not separate individual TRU, and because it may not have a satisfactory decontamination factor for lanthanide fission products, the process is probably not suitable for the thermal reactor recycle. In the fast reactor recycle mission, it has good potential for deployment in small-scale plants. Ability to process LWR spent fuel on a large scale is in question. May be more appropriate for fast reactor oxide fuel or as part of an aqueous/ pyrochemical hybrid process for treatment of LWR spent fuel.

* Only if proven technically and economically feasible through demonstration with actual spent fuel.

a. Aqueous process with mechanical head-end; application to coated-particle (TRISO) fuel.

b. Pyrochemical process; application to coated-particle (TRISO) fuel.

c. Pyrochemical process; application to oxide fuel.

Table 3.4. Separation technologies *(cont.)*

Technology type	Perceived advantage	Development needs	Time to implementation	Comments
Pyro metal[d]	Parts of process demonstrated over the course of conditioning EBR-II spent fuel.	Development and demonstration of TRU recovery step.	2035[g]	Recycle of Am in metal fuel must be demonstrated at larger scale.
Pyro nitride[d]	Pyrochemical alternative to aqueous processing.	Process verification with irradiated fuel; pilot-scale demonstration.	2035[g]	May be useful if recovery of ^{15}N is required.
Pyro carbide[d]	Pyrochemical alternative to aqueous processing.	Concept validation, laboratory-scale tests with hot fuel, pilot-scale demonstration.	2035[g]	At an early stage of concept development.
Fluoride volatility[e]	Potential for efficient extraction of uranium.	Laboratory-scale and pilot scale technology demonstrations.	2040	Process control is difficult, off-gas handling requirements are overwhelming, and product purity may be difficult to ensure. Might be useful for TRISO fuel processing.
DDP[f]	Compact process for FR oxide fuel treatment. Extensive experience with irradiated fuel processing.	Improvement of product purity, improved efficiency of recovery of minor actinides.	2025[g]	Russian technology. Would require extensive verification if it were to be applied in the US.

d. Pyrochemical process; application to metal, nitride or carbide fast reactor fuel.

e. Pyrochemical process; application to various fuel types.

f. Dimitrovgrad Dry Process; application to fast reactor oxide fuel treatment.

g. Introduction depends on timing of deployment of fast reactors.

Table 3.5. Advanced systems

Technology type	Perceived advantages	Countries interested	Development needs	Time to implementation	Comments
LWR	They exist.	Most countries.	Life extension-related material issues.		Potential for Pu multi-recycle; very limited potential for MA recycle.
ALWR (beyond AP600/1000)		France, Japan and USA.	If new needs are pointed out during deployment of Gen-III LWRs.	Under way	100 % MOX core.
HTGR/VHTR	Process heat and high temperature hydrogen production.	China, France, Korea and USA.	R&D on the He technology and components; innovative IHX[a] design; high and very high temperature materials; corrosion by impure He of cooling systems structural materials; irradiation damage and corrosion in graphite, SiC, carbon, composites and other new generation ceramic materials; graphite oxidation if air ingress.	2030	Strong potential for Pu transmutation. MA transmutation more questionable and needs to be demonstrated.
SFR	Mature technology.	China, France, India, Japan, Korea, Russia and USA.	Cost reduction; simplification (elimination) of secondary cooling system; compatibility of CO_2 with Na; improved structural materials for high burn-up; corrosion behaviour of F/M ODS steels in Na; in-service inspection; Na void reactivity coefficient reduction; safety behaviour when TRU loaded core.	2030-2035	Only available fast technology today.

a. Intermediate heat exchanger.

Table 3.5. Advanced systems *(cont.)*

Technology type	Perceived advantages	Countries interested	Development needs	Time to implementation	Comments
LFR	Higher operating temperature, no interaction between lead and air/water.	EU and Russia. Only paper studies by Japan, Korea and USA.	Corrosion control technologies; thermodynamic and physical-chemical properties of lead and lead alloys; compatibility of structural materials with coolant (corrosion, LME[a], fatigue, creep, etc.); new material coatings; design concepts for cost reduction; safety behaviour when TRU loaded core; in-service inspection; experimental reactor for technology demonstration.	2040-2045	Very little experience; difficult corrosion issues.
GFR	Still higher operating temperature; possible synergy with HTR/VHTR development; past EU experience with thermal gas-cooled reactors.	France and UK.	Fuel technology to be developed; DHR[b] system strategy and design; safety case needs to be demonstrated; high temperature structural materials; corrosion by impure He of cooling systems structural materials; experimental reactor for technology demonstration (ETDR[c]).	2040-2045	Major technological gaps (fuel); safety development (DHR).
ADS	Specific stratum in fuel cycle dedicated to burning minor actinides and some long-lived fission products; can be accepted fuels with practically any MA content.	Belgium, France, Germany, Japan, Korea, Russia, Spain and Sweden.	Spallation target technology (window vs. windowless concept); reliability of high-power proton accelerator; Pb-Bi technology and associated material issues (see LFR); need for an experimental reactor for demonstration.	2050-2055	Major technological developments needed.

a. Liquid metal embrittlement.

b. Decay heat removal.

c. Experimental test and demonstration reactor.

Chapter 4

CONCLUSIONS

Advanced fuel cycles allow optimising the use of natural resources, to minimise radioactive wastes and to increase proliferation resistance. These fuel cycles imply the transmutation of TRU or of MA.

There is wide international consensus that the best approach to the transmutation of TRU or of MA is the use of fast neutron spectrum reactors (critical or subcritical). The transmutation of minor actinides in conventional light water reactors, although possible from a reactor core physics point of view, is probably not a practical approach:

- The very high capture-to-fission cross-section ratios for most actinides in a thermal neutron spectrum (generally much higher than the corresponding ratios in a fast neutron spectrum), favour the build-up of higher mass actinides during irradiation of TRU fuels. The Cm and Cf build-up is responsible of an increase of ~10^4 of the neutron dose at fuel re-fabrication with respect to standard MOX fuel fabrication.

- Due to the less favourable neutron economy of a thermal neutron reactor, a very high over-enrichment is necessary, *e.g.* to maintain the same burn-up as compared to the case without minor actinides. For instance, in the case of MOX with enriched uranium support and 1% americium, the fissile enrichment has to be increased by ~1%.

- The implementation of a strategy of Pu and Am-only transmutation in an LWR fleet would imply around 50% of the fleet, (in the case of continuous recycling of plutonium and americium) and would necessitate dedicated facilities, including shielded hot cells, to develop and qualify the fabrication of transmutation fuel. A challenging chemical process of separation of Am from Cm would be needed, as would special dedicated facilities for the storage of curium with particular consideration for criticality and heat generation issues.

As far as the practical implementation of an advanced fuel cycle based on fast reactors, the following options can be considered:

- homogeneous recycling of not-separated TRUs in a critical fast reactor;

- heterogeneous recycling of MA as targets in specific SA, *e.g.* at the periphery of the core of a critical fast reactor;

- high MA content fuel in dedicated ADS facilities.

An advanced fuel cycle (*i.e.* that allows to optimise the use of natural resources, minimise radioactive wastes and to increase proliferation resistance) based on relatively conventional technologies (*e.g.* transuranics fuel multi-recycled in sodium-cooled fast reactors) will take about 20 years for implementation in countries where the technologies have not yet been deployed; 30 years for advanced technologies (transuranics fuel in other types of fast reactors or accelerator-driven systems, ADS).

If we move from the present thermal (light water) reactor economy to a reactor economy that is sustainable in the long term, it is vital that we preserve accumulating stocks of plutonium as generated by LWRs in order to fuel the initial group of fast reactors. The initial loading (including fuel in fabrication) is approximately 10-15 t of reactor grade (RG) Pu per GWe of fast reactor capacity. In that case, minor actinide management might have to be performed in dedicated facilities.

- Every transition fuel cycle aims to burn or stabilise the plutonium inventory. However, in case of transmutation of plutonium by ADS or other burner reactor, the amount of plutonium may not be sufficient to feed the fast reactor.

- Some minor actinides can be produced by decay from another element.

In the case of significant growth, the transition to a fast reactor fleet will be slowed by the availability of RG plutonium from the existing LWR fleet. For small or no growth, the transition can be relatively rapid if sufficient separation capabilities are implemented. This conclusion is supported by the analysis of the Japanese, Korean and French situations.

- For the US, the currently accumulated Pu inventory (non-separated) coming from LWR operations amounts to over 500 tonnes.

- Roughly 10 tonnes of plutonium is needed to start a fast reactor with a generating capacity of 1 GWe (start-up core and first reload).

- If a need is identified for doubling the current generating capability and for deploying a sufficient number of fast reactors in terms of sustainable development, plutonium availability will be a major factor in terms of being able to start a necessary number of fast reactors in a timely manner.

- For a small or no growth situation, the amount of Pu available at present would be sufficient for satisfying the need. For the US case, where the existing fleet of LWRs will be replaced in 2025-30, there is a need for a massive reprocessing capacity by about 2030. For the French case, where the first fleet of LWRs will have been replaced with EPRs by about 2020 and no large backlog of spent fuel exists, the reprocessing capacity needs for LWR spent fuel remain of the same order of magnitude as the current capacity, thanks to the relatively high content of Pu in the MOX spent fuel as compared to UOX. Additional capacity should be added for fast reactor spent fuel.

- In some countries, the most pressing major issue concerning Pu management is burning as much as possible of the plutonium. However, a certain amount of Pu needs to be reserved for its potential use for future fast reactor deployment.

For small nuclear infrastructures the prospect of sharing can be of high relevance, not only with regard to facilities (*e.g.* reprocessing plants, fuel fabrication plants, dedicated burner reactors such as ADS and even repositories), but also as concerns regional borrowing of fissile material. The limitation on plutonium for initial operation of fast reactors may require the trading of plutonium in exchange for other fuels or the storage of separated plutonium in regional facilities. Countries with small nuclear infrastructures may also have different timeframes for their transitions to fast reactors, in which case the shared use of reprocessing facilities can flatten temporary peaks in reprocessing or fuel fabrication needs. A regional approach to advanced fuel cycles has been developed (see Appendix 1) and has been applied as part of the activity of this Expert Group.

For countries that started their nuclear fuel cycles early and want to continue their use of nuclear energy, stocks of TRU and/or MA can be stabilised by the end of the century. Countries that want to diminish their dependence on nuclear energy can only partially reduce their inventories during this century, unless they act in a regional context.

Countries that will be undertaking new nuclear fuel cycles, for example a FR cycle, for Pu and MA recycle later in this century (by around 2050), can still stabilise the MA inventory over the entire nuclear fuel cycle during this century. In case minor actinide inventory reduction would be required to meet fuel cycle acceptability criteria, more time would be needed. MA inventory is related to FR deployment pace and a long period is necessary to replace all LWRs by FRs because of restrictions concerning Pu balance. To avoid any growth in MA inventory, the FR cycle should be deployed as early and as quickly as possible. In this context there can also be incentives: economy, availability of resources, safety (use of best practices and internationally recognised technologies) and non-proliferation (strict international control over transport flaws and a very limited non-proliferation number of jointly operated sites) to develop a "regional" approach.

More efficient use of geological repositories can be achieved through advanced fuel cycles. However, as indicated above, advanced fuel cycles need to be started early to have an impact.

Metrics for "more efficient use" of a repository can be defined as:

- radioactive element inventory – in mass and volume;
- potential source of radiotoxicity;
- dose;
- heat load.

More efficient repository use depends on the conditions of the groundwater, ventilation, etc., and on repository type:

- host geological strata – tuff, clay, granite, etc.;
- presumed duration of ventilation;
- local natural resources – salt, natural gas, minerals, etc.;
- exposure scenarios – water wells, intrusion.

The impact of advanced fuel cycles on repositories can be evaluated, defining and comparing appropriate scenarios in terms of, *e.g.* inventories sent to the repository:

- once-through fuel cycle – all spent fuel;
- limited recycle – fission products, processing losses and final-cycle spent fuel assemblies;
- continuing recycle – fission products and processing losses only.

A recently completed NEA study, *Advanced Nuclear Fuel Cycles and Radioactive Waste Management*, OECD/NEA (2006), has quantified the impact of selected advanced fuel cycles on different types of repositories.

Timing the implementation of advanced reprocessing technologies is critical to more efficient use of repository capacity. Countries with operating PUREX plants, will continue to separate and retain the minor actinides with the fission products. Once the HLW is vitrified, later separation of the minor actinides will be difficult and expensive. Countries or regional compacts that have not yet implemented reprocessing, must address the twin issues of cost minimisation and dose reduction, requiring that separation occur in one stage and that the minor actinides be separated at that time. Therefore, it is preferable that spent LWR fuel not be reprocessed until shortly before the LWR plutonium is needed for the initial loading of fast reactors.

Most of these scenarios assume little or no growth in the demand for nuclear energy. If there is a significant need for the upgrading of petroleum or the synthesis of transportation fuels, the growth in the nuclear reactor fleet could be much more rapid. For example, if the US were to use coal hydrogenation to produce hydrocarbon fuels equal to present petroleum imports, approximately 600 GWth of new nuclear capacity, about twice the size of the present US LWR fleet, would be needed. This transition from imported hydrocarbon fuels to the synthesis of transportation fuels would require the simultaneous construction of new reactors, construction of fuel processing plants, renewed exploration for uranium deposits and opening new uranium mines. Each of these endeavours is a 20-30 year task. Depending of the success of exploration efforts, reprocessing of the existing stocks of spent LWR fuel and the construction of a generation of fast reactors may be necessary, both of which are also 20-30 year tasks.

The thorium cycle is not a short-term solution to the resource or repository limitations. Thorium technology is not ready to be used, though thorium resources are available in large amounts. In fact:

- The use of thorium fuel does not reduce the demand for natural uranium in the short term. Because thorium has no fissile isotopes, initial core loadings will require enriched uranium. The natural uranium needed for a thorium-uranium core is approximately equal to the natural uranium required for a UO_2 core.

- Plutonium fuel is needed for "starting" the fleet, one consequence being a reduction in the amount of plutonium available for fast reactors.

- The generation of additional unwanted actinides, such as ^{232}U and ^{231}Pa, is a result.

- A toxicity reduction associated with the thorium fuel cycle with respect to the uranium cycle can be expected in the short and medium terms. Radiotoxicity is higher, though, in the long term (*i.e.* beyond ~10^4 years).

Appendix 1

IMPROVED RESOURCE UTILISATION, WASTE MINIMISATION AND PROLIFERATION RESISTANCE IN A REGIONAL CONTEXT

Abstract

Regional centres for the nuclear fuel cycles are "an old and new idea". This potential of this concept is being investigated, as are possible implementation issues in the context of advanced fuel cycles. In particular, scenarios have been worked out and quantified wherein countries with different policies with respect to nuclear energy development attempt to determine a common approach with the aim of minimising wastes and optimising the use of resources. These objectives can potentially be tackled with the implementation of shared facilities. The first attempt was an application of the regional approach to the case of two countries, one committed to the further development of nuclear energy, while the other one plans a nuclear phase-out. Successively, a "user/provider" scenario has also been studied. The results have been found encouraging, and a further application is underway in support of the development of a European roadmap towards the implementation of a European strategy for P&T, within a co-ordinated action of the EU 6th Framework Programme.

Introduction

"Regional approaches" to the fuel cycle have previously been the subject of discussion, even before the ElBaradei proposal [1,2], mostly for non-proliferation reasons [3-5].

McCombie [6], mainly dealing with the especially contentious area of final disposal in geological repositories, recently arrived at the conclusion that "the time is ripe to consider again the global benefits of nuclear fuel cycle centres for both front end and back end activities."

Some examples of regional studies

We developed and worked out [7] an original "regional approach" involving two European countries with the purpose to support the deployment of P&T strategies aiming at waste minimisation. In fact, to benefit from the recognised potential of these strategies, it is necessary to develop sophisticated technologies for the fuel cycle and to develop new facilities for fuel reprocessing and fabrication and innovative reactor systems. It does not seem realistic for most countries to cope with this major endeavour in isolation.

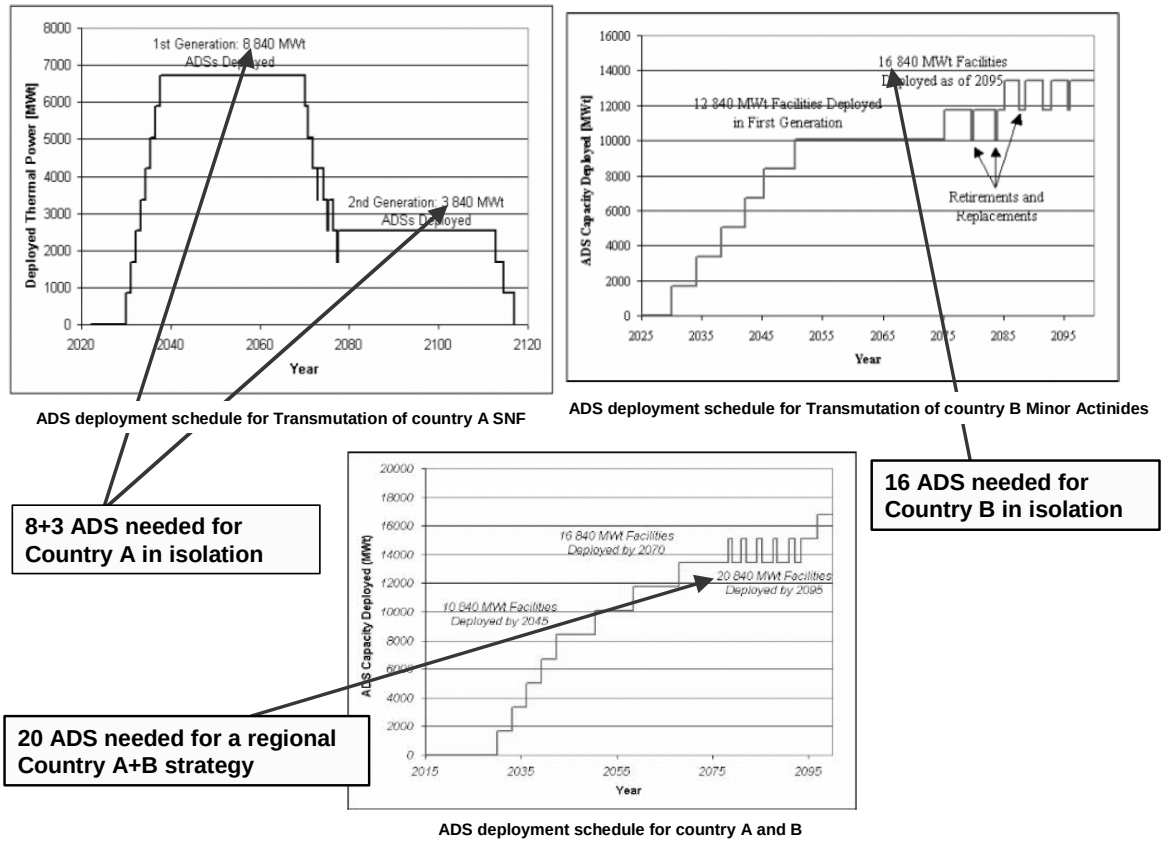
In Ref. [7], we considered both ADS-based transmutation and critical fast-reactor-based transmutation. Some of the most significant results are summarised, in order to highlight the potential benefits of a regional approach, and the potential for application to a more general case.

The first scenario considered in Ref. [7] was related to the deployment of a number of ADS shared by the two countries. In this case, the ADS uses the plutonium of Country A and transmutes the

minor actinides of the two countries. The plutonium of Country B is continuously recycled in PWRs. The main objective of this scenario is to decrease the stock of spent fuel of Country A down to ~0 at the end of the century, and to stabilise the Pu and MA inventories of Country B.

As an example of the results, Figure 1 shows the comparison of the number and pace of deployment of the ADS in the regional approach and in the case of ADS deployment by Country A and Country B in isolation. The results shown in Figure 1 indicate the significant benefits of the regional approach.

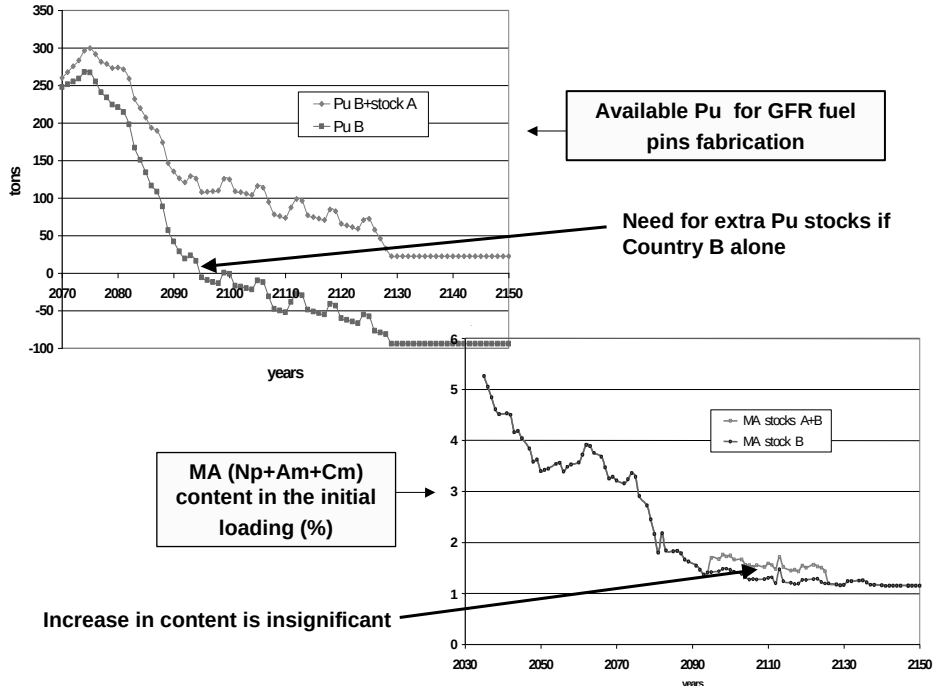
Figure 1. Results of scenarios of ADS deployment [7]



The second scenario considered the deployment of fast reactors in Country B. These fast reactors are deployed with the plutonium of the two countries and recycle all the minor actinides. The main objective of this scenario is to decrease the stock of spent fuel of Country A down to 0 at the end of the century and to introduce Gen-IV fast reactors in Country B, starting *e.g.* in 2035.

As a demonstration of the results, Figure 2 shows that the deployment of fast reactors in Country B is not jeopardised by a shortage of plutonium if the TRU inventory in the spent fuel of Country A is reprocessed and used. Moreover, Figure 2 shows that the increase in minor actinide content in the fast reactor fuel, due to the higher minor actinide content in the spent fuel of Country A, has no significant impact on the feasibility of the fast reactors in Country B.

Figure 2. Impact of a regional approach on the deployment of fast reactors in a selected country [7]



A further study of a "user/supplier" scenario has also been performed. The scenario involves two types of countries:

- Countries A (*e.g.* with small grid systems) decide to implement small (~50 MWe) reactors as transportable cartridges (*e.g.* SMFR [8], with ~30 years lifetime, passive safety, compact and robust technology and high proliferation resistance). These countries are designated "user" countries.

- Country B with reprocessing and fuel fabrication capabilities, with its own nuclear power fleet, acts as the "supplier" country.

The scenario is represented in Figure 3.

The objective of the study is to quantify fabrication/reprocessing/material transport needs, potential constraints, etc.

Results are shown in Figure 4. These results correspond to the following hypothesis:

- PWR UOX (BU 50 GWd/t, 10 y cooling) in Country B.

- 20 SMFRs adapted to Pu+MA fuel, with 30 years of operation in Countries A.

- After 30 years, the fuels are sent back for reprocessing and used in country B for Gen-IV reactors.

Figure 3. A "user/supplier" scenario

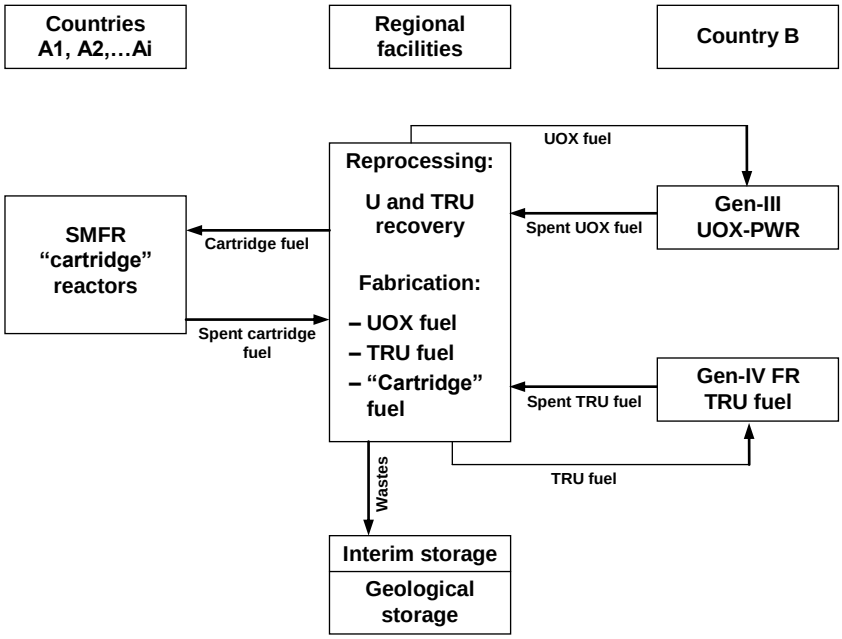
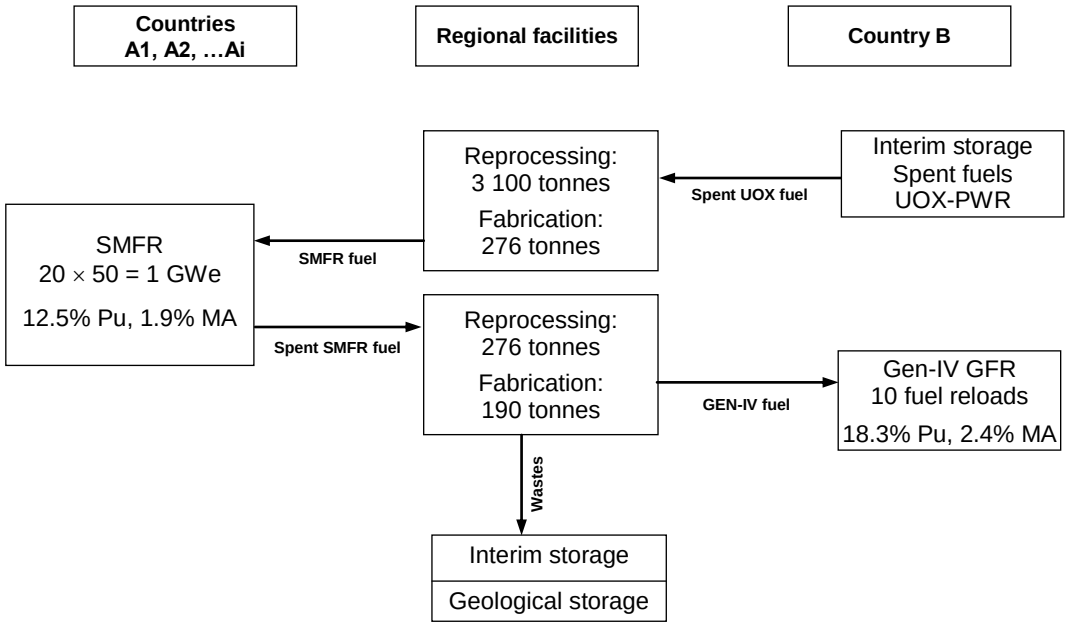


Figure 4. Results for the "user/supplier" scenario



Further analysis, *e.g.* to establish the rate of penetration of the SMFRs, would require further specification of the policy of Country B. For example:

- If Country B stores irradiated UOX fuel (*e.g.* as is the case of the USA), the 3 100 t UOX needed will be available at any time.

- If Country B undertakes reprocessing and makes use of Pu (*e.g.* France), how and when the UOX could be "diverted" and made available should be determined.

- The data allow figuring out the size of the reprocessing and fabrication facilities, according to the SMFRs penetration rate foreseen.

The reprocessing as shown in the scheme considers not-separated TRU. Other schemes can be envisaged.

A regional approach for the implementation of P&T in Europe

A more comprehensive study will involve a larger number of countries (Belgium, France, Germany, Spain, Sweden and Switzerland) and a wider number of scenarios.

The regional approach should help to outline a strategy on how to share facilities and fuel inventories to optimise the use of resources and investments in an enhanced proliferation-resistant environment.

The scenarios will consider several groups of countries:

- Group A is in a phase-out (or stagnant) scenario for nuclear energy and has to manage its spent fuel, especially the plutonium and the minor actinides.

- Group B is in a continuation scenario and has to optimise its resources in plutonium for the future deployment of fast reactors or ADS.

- Group C, after stagnation, envisages a nuclear "renaissance".

- Group D, initially with no NPP, decides to go nuclear.

Different scenarios will be studied and are being defined. Examples being examined include:

- Scenarios which consider the deployment of a group of ADS shared by several countries, *e.g.* the ADS will use the minor actinides of Group B and will transmute the TRU of the other groups. The plutonium of Group B is mono- or continuously recycled in PWRs.

  The main objective of these scenarios is to decrease the stock of spent fuel of Countries A and C down to 0 by the end of the century. The result of the study will be the pace of deployment and the number of ADS necessary to eliminate the stocks of Group A and to stabilise/decrease the MA stocks of Group B; fuel cycle facilities needed and time horizon for deployment; masses and heat load in a repository.

- Scenarios which consider the deployment of fast reactors in Group B countries. These fast reactors are deployed with the plutonium of all groups of countries and recycle all the minor actinides. The main objective of this scenario is to decrease the stock of spent fuel of Countries A and C down to 0 by the end of the century and to introduce Gen-IV fast reactors in Group B, starting *e.g.* in 2035.

  The result of the study will be the number and feasibility (e.g. allowable MA content) of fast reactors to be deployed in Group B which will have the mission both to produce electricity and to eliminate the stock of spent fuel of Countries A and C by the end of the century. Other results will be the number and characteristics of the fuel cycle facilities; masses and heat load in a shared repository.

- Scenarios where countries of Group C (and/or D) decide, after a certain period of time, to restart nuclear energy with fast reactors which recycle all their own TRU. Variants can be envisaged, according to the policy of Group B, *e.g.* mono-recycling of Pu and successive use of fast reactors or use of fast reactors at an early date. In these scenarios, one can make the hypothesis that the spent fuel of the other countries of Group A is used to facilitate the deployment of fast reactors in Group C.

  The result of the scenario study will be the maximum level of electricity production achievable at equilibrium for Group C. This result will depend on the amount of plutonium available and on the pace of deployment of the fast reactors. Here again, fuel cycle facilities characteristics and parameters related to the repository will be obtained.

At present, as indicated above, six countries have made their spent fuel inventories and isotopic compositions available (at various dates): Belgium, France, Germany, Spain, Sweden and Switzerland.

Detailed scenarios are presently being discussed. Hypotheses on parameters such as energy demand, cooling times, etc. and on characteristics such as type of fast reactor and ADS, etc., will be agreed upon shortly.

Preliminary results (mostly obtained with the COSI code [9]) are expected at the end of 2007.

Conclusions

Regional approaches to the nuclear fuel cycles have been proposed in various frameworks.

In the case of Europe, it is interesting to develop such scenarios to investigate opportunities for enhanced collaboration, in particular in the perspective of advanced fuel cycles.

First results have been obtained, which confirm the potential interest of regional approaches to the fuel cycle. More results are expected in the very near.

However, to make these scenarios more realistic, a number of rather involved institutional (*e.g.* shared repository) and practical (*e.g.* material transports) issues should be tackled and discussed in-depth.

REFERENCES

[1] ElBaradei, M. (2003), "Towards a Safer World", *The Economist*, 16 October 2003.

[2] ElBaradei, M. (2004), "Nuclear Non-Proliferation: Global Security in a Rapidly Changing World", *Carnegie International Non-Proliferation Conference*, 30 June 2004.

[3] IAEA (2004), *Developing and Implementing Multinational Repositories: Infrastructural Framework and Scenarios of Co-operation* (draft).

[4] Meckoni, V., R.J. Catlin and L. Bennett (1977), *Regional Nuclear Fuel Cycle Centres: IAEA Study Project*, IAEA-CN-36/487, Vienna.

[5] McCombie, C. and N. Chapman (2004), "Siting Multinational Facilities: A Bottom-Up Approach", *WM'04 Conference*, Tucson, Arizona, 29 February-4 March.

[6] McCombie, C. and N. Chapman (2004), "Nuclear Fuel Centres – An Old and New Idea", *World Nuclear Association Annual Symposium*.

[7] Salvatores, M., *et al*. (2004), "Partitioning and Transmutation Potential for Waste Minimization in a Regional Context.", *8th NEA Information Exchange Meeting on Actinide and Fission Product P&T*, University of Nevada, Las Vegas, 9-11 November.

[8] Smith, C., D. Crawford, M. Cappiello, A. Minato, J. Herczeg (2004), "The Small Modular Liquid Metal Cooled Reactor: A New Approach to Proliferation Risk Management", *14th Pacific Basin Nuclear Conference, "New Technologies for a New Era"*, Honolulu, Hawaii, 21-25 March.

[9] Grouiller, J.P., *et al*. (1991), "COSI: A Code for Simulating a System of Nuclear Power Reactors and Fuel Cycle Plants", *Proc.FR'91*, Kyoto, Japan, 28 October-1 November.

Appendix 2

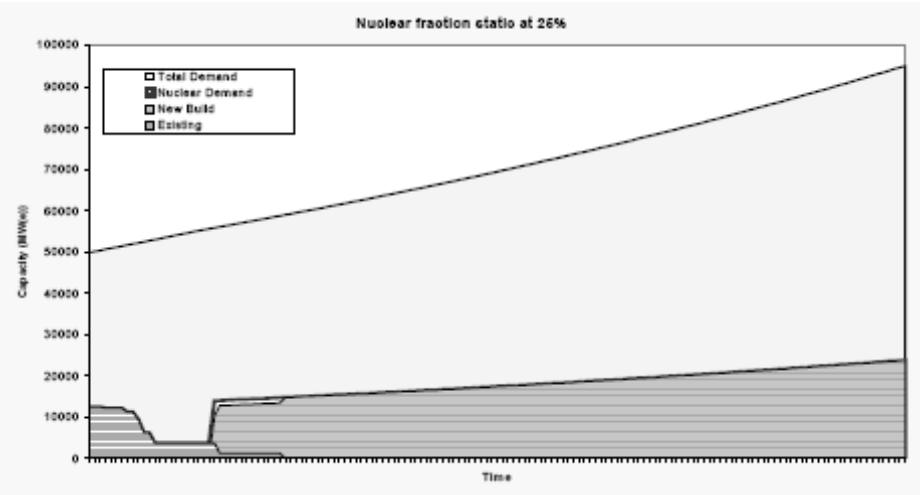
SUMMARY OF UK ADVANCED FUEL CYCLE SCENARIOS

Current UK nuclear development

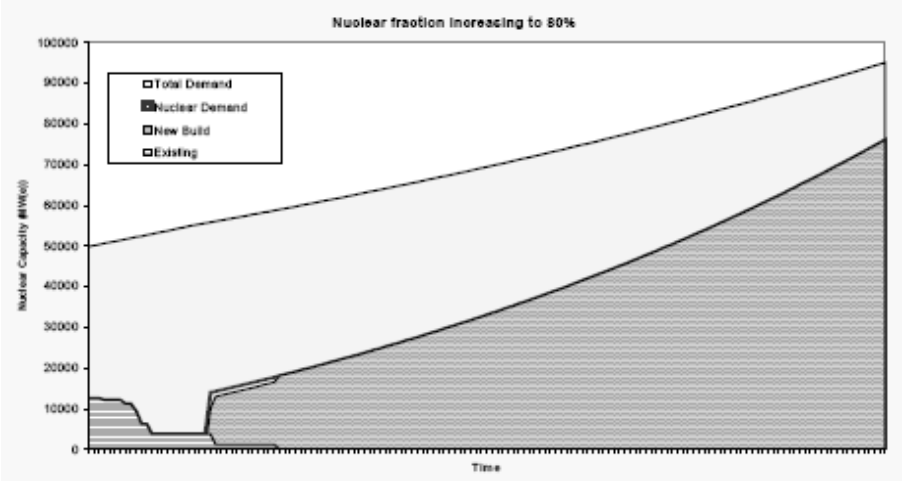
- Total UK electricity demand was 350 TWh in 1999, of which 25% was nuclear.
- Expected demand is 387 TWh by 2020 with a growth thereafter of 0.42% per annum.
- Without rebuild the nuclear fraction will fall to -18% by 2010:
  - 7% by 2020;
  - 0% by 2035.
- Assume replacement capacity required.

Nuclear rebuild (not official UK policy)

- *Scenario 1*:
  - re-establish 25% nuclear beginning 2020;
  - nuclear runs until 2150 at which point evaluation made for future options:
    - coast down due to replacement technologies or continuation.



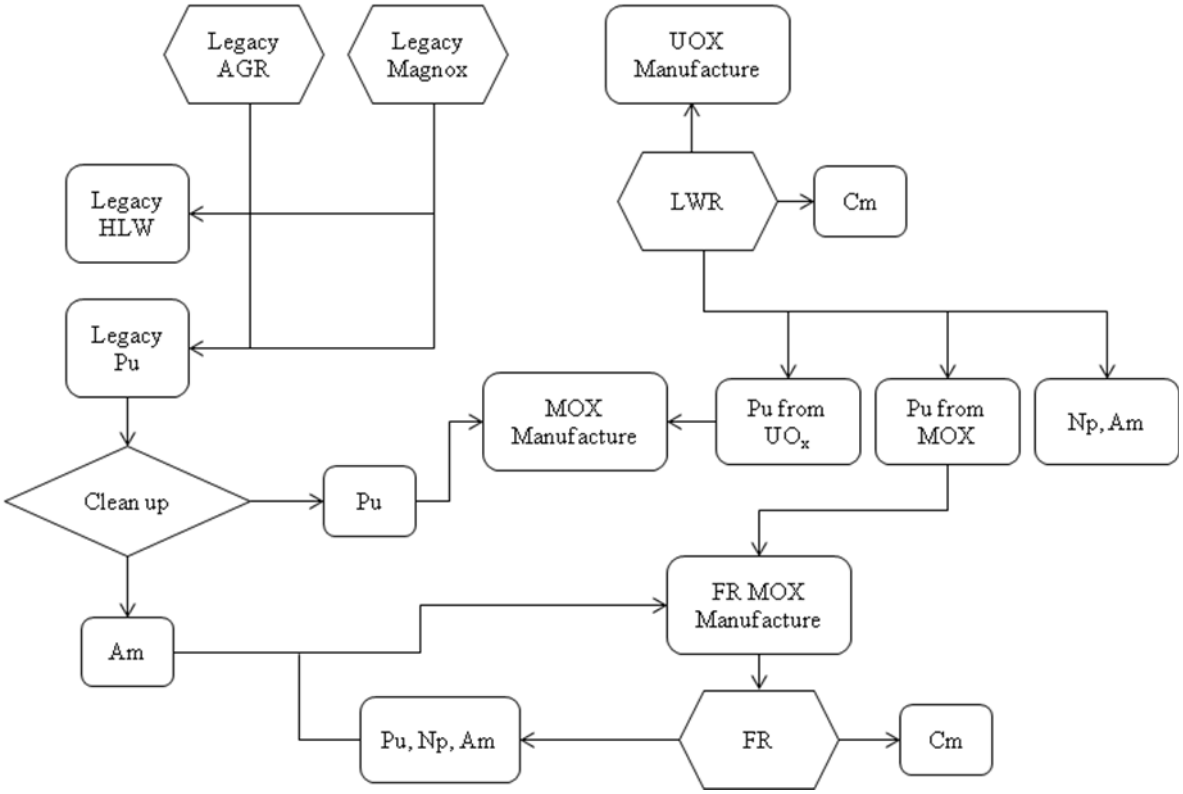
- *Scenario 2*:
  - re-establish 25% nuclear beginning 2020, escalating to 80% by 2150;
  - evaluation point at 2150 with coast down option.



Installed reactors

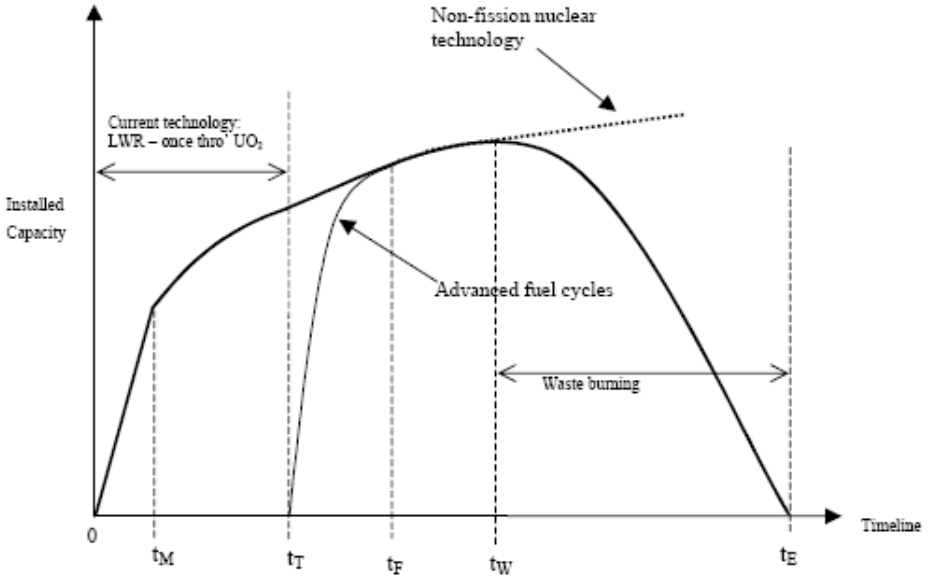
- UK scenarios need to consider legacy reactors. Current reactor deployment consists of:
  - Magnox (metal fuel, gas-cooled, low burn-up);
  - AGR (oxide fuel, gas-cooled, intermediate burn-up);
  - LWR (Sizewell B PWR).
- Gas-cooled reactor fuel is reprocessed. Pu is separated and stored; HLW is vitrified and deemed non-recoverable.
- LWR current decision is for on-site storage of spent fuel prior to ultimate processing or disposal.
- New reactors assumed to be mixture of:
  - LWR burning UOX and MOX;
  - MO fraction a free-variable depending on scenario specifics;
  - fast reactors with low breeding ratio for burning Pu, Np and Am.
- Legacy Pu is cleaned prior to use to remove Am.
- Cm is stored and not recycled due to handling and processing difficulties in the short term.

Generalised mass flow



Reactor scenarios

- *Reactor scenario 1*:
  - LWR introduced in 2020 to burn UOX;
  - FR introduced in 2080 to begin Pu and MA burning.
- *Reactor scenario 2*:
  - LWR introduced in 2020 to burn UOX and MOX;
  - The actual fraction of MOX may also be a scenario parameter;
  - FR introduced in 2080 to extend Pu burning and introduces MA burning.
- *Reactor scenario 3*:
  - LWR introduced in 2020 to burn UOX and MOX;
  - FR co-introduced in 2020 to extend Pu burning and introduces MA burning.


Non-fission nuclear technology
Current technology: LWR – once thro' UO_2
Installed Capacity
Advanced fuel cycles
Waste burning
Timeline
0
t_M
t_T
t_F
t_W
t_E

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