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ROLE OF THORIUM
TO SUPPLEMENT FUEL CYCLES
OF FUTURE NUCLEAR ENERGY SYSTEMS

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IAEA NUCLEAR ENERGY SERIES No. NF-T-2.4

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FUTURE NUCLEAR ENERGY SYSTEMS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2012

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FOREWORD

One of the IAEA's statutory objectives is to "seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world". One way this objective is achieved is through the publication of a range of technical series. Two of these are the IAEA Nuclear Energy Series and the IAEA Safety Standards Series.

According to Article III.A.6 of the IAEA Statute, the safety standards establish "standards of safety for protection of health and minimization of danger to life and property." The safety standards include the Safety Fundamentals, Safety Requirements and Safety Guides. These standards are written primarily in a regulatory style, and are binding on the IAEA for its own programmes. The principal users are the regulatory bodies in Member States and other national authorities.

The IAEA Nuclear Energy Series comprises reports designed to encourage and assist R&D on, and application of, nuclear energy for peaceful uses. This includes practical examples to be used by owners and operators of utilities in Member States, implementing organizations, academia, and government officials, among others. This information is presented in guides, reports on technology status and advances, and best practices for peaceful uses of nuclear energy based on inputs from international experts. The IAEA Nuclear Energy Series complements the IAEA Safety Standards Series.

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was established in 2001 on the basis of a resolution of the IAEA General Conference in 2000 (GC(44)/RES/21). INPRO activities have since been continuously endorsed by resolutions of IAEA General Conferences and by the General Assembly of the United Nations.

The objectives of INPRO are to: help ensure that nuclear energy is available to contribute, in a sustainable manner, to the goal of meeting the energy needs of the 21st century; and bring together technology holders and users so that they can consider jointly the international and national actions required for ensuring sustainability of nuclear energy through innovations in technology and/or institutional arrangements.

To fulfil these objectives, during its first phase INPRO developed a set of basic principles, user requirements and criteria together with an assessment method, which taken together, comprise the INPRO methodology for the evaluation of the long term sustainability of innovative nuclear energy systems. The INPRO methodology is documented in IAEA-TECDOC-1575, which comprises an overview volume and eight additional volumes covering economics, institutional measures (infrastructure), waste management, proliferation resistance, physical protection, environment (impact of stressors and availability of resources), safety of reactors, and safety of nuclear fuel cycle facilities.

In its second phase, INPRO established a series of collaborative projects that cover issues related to innovations in technology and institutional arrangements. One of these projects is 'Further Investigation of the Thorium Fuel Cycles', in which Canada, China, India, France, the Republic of Korea, the Russian Federation, Slovakia, Ukraine and the European Commission participated. In addition, institutions such as Thorium Power (USA), Thor Energy (Norway) and the Institute of Energy Research at Jülich (Germany) were involved as observers. The overall objective adopted by the participants was to examine the potential of thorium based fuel cycles (ThFC) to improve the sustainability of nuclear power.

Fuel cycles based on thorium (^{232}Th) are the options that may provide an opportunity to use vast deposits of this nuclear material to supply future large scale deployment of nuclear energy systems and enhance the sustainability of nuclear power. ^{232}Th is a fertile material that can be converted in nuclear reactors into fissile material (mainly ^{233}U), which can be used for the energy generation in a similar way to naturally occurring ^{235}U.

Different from studies of thorium conducted by the IAEA in the past, this ThFC study considers the potential role of thorium in supplementing the uranium–plutonium fuel cycle in scenarios with a significant increase in the use of nuclear energy in the world. Special attention is paid to consideration of the ThFC from the point of view of economics and proliferation resistance. The INPRO methodology is applied to assess general aspects mainly related to material flows in various ThFC options.

This is the final report of the INPRO ThFC collaborative project. The IAEA officer responsible for this publication was A. Korinny of the Division of Nuclear Power.

CONTENTS

1. INTRODUCTION

Historically, investigations of a potential application of thorium (^{232}Th) as a fuel for nuclear power reactors started in parallel with the first studies of uranium and plutonium utilization. Thorium seemed an attractive option of nuclear material mainly due to its abundance, the opportunity to reduce the need for enrichment in the fuel cycle, the high conversion ratios (to ^{233}U) achievable in a thermal neutron spectrum, and also due to other neutron and thermal physical properties studied at the early stage of nuclear power history. In spite of a rather long list of advantages and the significant rise of uranium prices during the last years, thorium is not yet augmenting the use of uranium fuel on a commercial basis, although research efforts regarding the thorium fuel cycle (ThFC) continue.

As yet, there is no commercial fabrication or reprocessing infrastructure for thorium fuel, unlike the available infrastructure for the uranium fuel cycle. In the 21st century, market conditions may change in such a way that thorium options will become commercially more attractive for nuclear power application.

1.1. CHARACTERISTICS OF THORIUM IN COMPARISON WITH URANIUM

Unlike data for uranium, the data on thorium deposits in the world are not yet well systemized [1]; nevertheless, it is generally assumed that thorium is three to four times more abundant in nature than uranium.

Thorium (^{232}Th) is a fertile material that can be used to produce the fissile isotope uranium-233 (^{233}U) in a reactor, which in turn could be used as fissile material in fuel for nuclear reactors. Compared to the only other naturally occurring fertile material uranium-238 (^{238}U), thorium has some advantageous physical properties leading to higher conversion rates, i.e. production rates of fissile material: the ratio of neutron capture of thorium to neutron loss (capture) in parasitic material (e.g. fission products, coolant) is higher for ^{232}Th than for ^{238}U.

The conversion ratio of thorium into ^{233}U depends on the type of the reactor utilizing thorium and on the mode chosen (open or closed fuel cycle). Historically, in the Shippingport 72 MW(e) light water reactor (LWR) in the USA, a breeding ratio of 1.01 was achieved [2]. According to design calculations performed later, a light water breeding reactor (LWBR) could theoretically reach a breeding ratio of 1.06. Preliminary studies have indicated that, due to better neutron efficiency in the core of a heavy water moderated reactor (HWR), a conversion ratio higher than 1 should be more easily reached than in an LWR. An HWR has long been known to have a high Th/^{233}U conversion ratio, and innovative HWR concepts can be designed to have at least (semi-self-breeding, i.e. a conversion ratio of 1.0) the possibility of a self-sustaining fuel cycle.

A thorium–uranium fuel cycle is more amenable to multiple recycling of ^{233}U than plutonium (Pu) recycling in a uranium–plutonium fuel cycle. This is caused by the lower neutron capture (n, γ) cross-section in ^{233}U than in ^{235}U and ^{239}Pu, whereas the fission cross-sections are similar, leading to lower generation of isotopes with higher masses in the thorium–uranium fuel cycle. Thus, the amount of minor actinides in spent Th-^{233}U fuel (with no ^{238}U or plutonium in fresh fuel) drops significantly.

In its oxide form, usually used for fuel pellets, thorium dioxide (ThO$_2$) is also a chemically more stable substance than uranium dioxide (UO$_2$) and possesses better thermal conductivity and also a lower thermal expansion coefficient. The melting point of thorium dioxide (3370ºC) is higher than uranium dioxide (2760ºC). These characteristics enable higher safety margins and provide opportunities for increased economy in reactor core design (e.g. extended operational limits, high burnup and high thermal efficiency).

From the point of view of technology amenability, there are fewer conversion processes required from converting mined thorium ore into fuel forms ready for first use in a reactor than with conversion of mined uranium into the (currently most used) conventional fuel form of enriched UO$_2$. The enrichment of uranium — a rather sophisticated technological process with a significant proliferation threat — is not needed in a pure ThFC; however, instead of enrichment, a reprocessing step is necessary.

In addition to the elimination of enrichment, a fuel cycle based on thorium/^{233}U has other proliferation resistance related peculiarities. To create one more barrier to potential proliferation, ^{233}U — unlike plutonium — can be mixed with the non fissile isotope ^{238}U (in practice, with depleted uranium) to create a 'reactor grade' ^{233}U/^{238}U mixture not adherent to chemical separation. The critical configuration (i.e. mass, geometry, etc. needed

for a nuclear weapon) of a mixture of 12% ^{233}U with ^{238}U approximately corresponds to a 20% enriched ^{235}U/^{238}U compound [3].

Once irradiated in a reactor, the fuel of a thorium–uranium cycle contains an admixture of ^{232}U (half-life 68.9 years) whose radioactive decay chain includes emitters (particularly ^{208}Tl) of high energy gamma radiation (2.6 MeV). This makes spent thorium fuel treatment more difficult, requires remote handling/control during reprocessing and during further fuel fabrication, but on the other hand, may be considered as an additional non-proliferation barrier.

Thorium fuel may be utilized either in a once-through manner (also called an open fuel cycle), i.e. by producing ^{233}U in a fuel element and in parallel burning it up in the same element, or in a mode with spent fuel reprocessing, i.e. in a closed fuel cycle.

Thorium and ^{233}U utilization is technically feasible in most existing and prospective reactor designs, including LWRs, HWRs, fast breeders (FBRs) and molten salt reactors. However, for the majority of thorium introduction options, only reactor physics studies have been performed to date. Additionally, there are other aspects in the use of thorium that may require more detailed investigation as well as several technological developments that are necessary for its commercial implementation. In particular, the incorporation of thorium–uranium fuel into cores of existing reactors usually requires certain modifications in engineered (safety relevant) systems, such as reactor and reactivity control devices, mainly because of the difference in the effective fractions of delayed neutrons per fission that are the basis for power control of a reactor. For ^{235}U, the fraction is ~0.0065, but for ^{233}U, it is only ~0.00266 [3].

1.2. OBJECTIVE

Different from studies of thorium conducted by the IAEA in the past [4–7] , this ThFC study considers the potential role of thorium to supplement the uranium–plutonium fuel cycle in scenarios with a significant increase in the use of nuclear energy in the world. Special attention is paid to consideration of the ThFC from the point of view of proliferation resistance. With regard to proliferation resistance, the INPRO methodology is applied to assess general aspects mainly related to material flows in various ThFC options.

In the short term, the implementation of thorium–uranium based fuel in operating reactors in a once-through (or open) mode may become technically available. However, in a closed fuel cycle, services such as reprocessing and recycled fuel fabrication demand the development of new technologies to provide the necessary economic competitiveness on a commercial scale. These new technologies will require a longer time to be developed and deployed. Taking into account a high growth of nuclear power in the future, there are some concerns within the nuclear community regarding the availability of reasonably priced nuclear fuel based on uranium–plutonium fuel cycles. Such considerations may change the priorities for nuclear power and accelerate development of such new technologies needed for thorium introduction.

Therefore, a collaborative effort by interested INPRO members to study possibilities of introducing thorium seems justified. It also generates a new perspective/clarification of the potential contribution of thorium to sustainable global growth of nuclear power in the 21st century.

This report was developed in the framework of the INPRO collaborative project Further Investigation of the Thorium Fuel Cycles, which was launched in December 2007. INPRO members participating in the project include Canada, China, the European Commission, India, the Republic of Korea, the Russian Federation, Slovakia and Ukraine. In addition, companies and institutions such as Thorium Power (USA), Thor Energy (Norway) and the Institute of Energy Research at Jülich, Germany, were involved in the project as observers. The overall objective adopted by the participants was to examine the potential of thorium based fuel cycles to improve the sustainability of nuclear power.

In this report, the sustainability of nuclear power is explored by re-examining the potential of thorium based fuel cycles to support future large scale deployment of nuclear energy systems by increasing the availability of nuclear material.

1.3. STRUCTURE

Section 2 provides an overview of the status of activities in some countries regarding introduction of thorium into nuclear fuel cycles. Section 3 presents results of R&D on the reactors potentially utilizing thorium in several Member States. Section 4 presents several global scenarios of thorium introduction into different nuclear energy systems including thermal and fast reactors. Section 5 deals with economic aspects of thorium fuel cycles (ThFCs). Section 6 deals with aspects of proliferation resistance of ThFCs. Section 7 summarizes the results of the report and provides recommendations for additional studies.

Annex I presents detailed neutronic data of reactors with uranium–plutonium and thorium fuel. Annexes II and III present the results of scenario development assuming a moderate demand for nuclear power. Annex IV presents a scenario for the thermal breeder reactor with thorium fuel. Annex V shows a comparison of a simplified nuclear energy system (NES) consisting of LWRs, advanced light water reactors (ALWRs) and advanced heavy water reactors (AHWRs). Annex VI presents the results of scenario studies performed by Canada for an HWR with thorium fuel. Annex VII presents the results of a calculation of levelized unit energy cost (LUEC) with an alternative set of economic input data. Annex VIII shows the application of the INPRO methodology in the area of proliferation resistance for ThFCs .

2. USE OF THORIUM IN THE NUCLEAR FUEL CYCLE

This section provides an overview of the status of activities in several countries anticipating the introduction of thorium into their nuclear fuel cycle.

2.1. RUSSIAN FEDERATION

The use of fossil fuel to cover increasing electricity demand is linked to several serious problems related to resource, ecological and economic issues. No doubt, in the short and even in the medium term perspective, fossil fuel will occupy a significant place in the country's energy balance. It is, however, doubtful that growing energy production can rely on fossil fuel alone. Moreover, most of the Russian Federation's power plants using fossil fuel, especially coal, are based on outdated technologies and are approaching the end of their operational life.

Under these conditions, the decision has been made that the growing demand of the Russian Federation population and economy for energy should be satisfied in the long term by nuclear power based on modern technologies and by increasing the share of nuclear electricity from a current 16–25% by 2030. An annual rate of commissioning of nuclear units of 2–3 GW(e)/year should therefore be reached in the near term, taking into account the necessary compensation of the capacities of nuclear units to be decommissioned.

2.1.1. Scenarios of development of nuclear power in the Russian Federation

The national programme of nuclear power development envisages a scenario with the transition of Russian nuclear power to utilization of ^{238}U by the mid-21st century to ensure the security of energy supply. The amount of energy that could be retrieved from Russian ^{238}U is ten times higher than energy from coal mined in the country and approximately 30 times higher than natural gas. The utilization of ^{238}U may become available as a result of transition to a New Technological Platform of nuclear power with a closed nuclear fuel cycle based on fast reactors [8, 9].

The process of transition is under consideration in the framework of development of a Strategy of Nuclear Power Development up to 2050. The basic goals of this *innovative* scenario are:

— The total installed nuclear capacity should reach 40 GW by 2020, 60 GW by 2030, and 100 GW by 2050.
— The share of fast reactors in the total nuclear capacity should reach 60% by 2050.

In the following, an alternative scenario option is compared with the 'New Technological Platform'. It is an *evolutionary* scenario and envisages the same amount of nuclear capacity within the same timeframe but comprising only a new generation of thermal reactors of WWER type and no fast reactors.

Figure 2.1 shows the result of a calculation of uranium consumption in the Russian Federation NES for the two scenarios. In the *evolutionary* scenario, by the middle of the century, the demand of natural uranium will reach ~600 kt, which corresponds to the proven deposits of natural uranium in the Russian Federation. In reality, natural uranium may be exhausted in an even shorter time due to the increasing share of nuclear fuel exported abroad. More than 1300 kt of natural uranium are required to continue the operation of thermal reactors, which would be commissioned by 2050, until the end of their designed lifetime; this number is approximately equal to the most optimistic estimation of uranium deposits in the Russian Federation.

In the *innovative* scenario (with fast reactors), the consumption of uranium is lower and reaches ~350 kt by 2050 and ~570 kt by 2090, i.e. at the end of the lifetime of thermal reactors commissioned in 2030. This demand can be covered by uranium deposits available in the Russian Federation territory; however, a shortage of uranium may still occur due to export of nuclear fuel.

There are several options that may compensate for the potential deficit of uranium, e.g. further enrichment of available uranium tails or the manufacture of export fuel using uranium supplied by customers. Another attractive possible solution is related to the application of thorium–uranium fuel via conversion of part of operating thermal reactors to use a thorium–uranium fuel cycle. In this case, ^{233}U can be produced in specialized thermal or fast breeder reactors.

2.1.2. Conversion of thorium into ^{233}U in thermal and fast reactors

A great variety of reactor types can be used for ^{233}U production, including conventional thermal reactors, fast breeders and accelerator driven systems (ADSs). In the Russian Federation, the 'evolutionary intergrowth' of a thorium cycle alongside a uranium fuel cycle is the most likely scenario. A ThFC may be introduced in thermal neutron WWER type reactors dominating in the short and medium term perspective, or in fast neutron BN type reactors in the case of their full scale deployment. In this latter case, ^{233}U can be accumulated by using breeding blankets (primarily radial ones) composed of thorium. The scale of breeding depends on the number of fast reactors in a system.

^{233}U can be produced in WWER-1000 reactors using mixed enriched uranium and thorium oxide fuel. For this purpose, high enriched uranium (for example, weapon grade uranium withdrawn from defence programmes) or medium level enriched uranium (less than 20% ^{235}U) could be used. Both options were considered in a study documented in Ref. [10]. All main indicators of the two options are comparable to the conventional (reference core design) WWER-1000, and from this viewpoint, the replacement of ^{238}U with thorium will not pose significant new
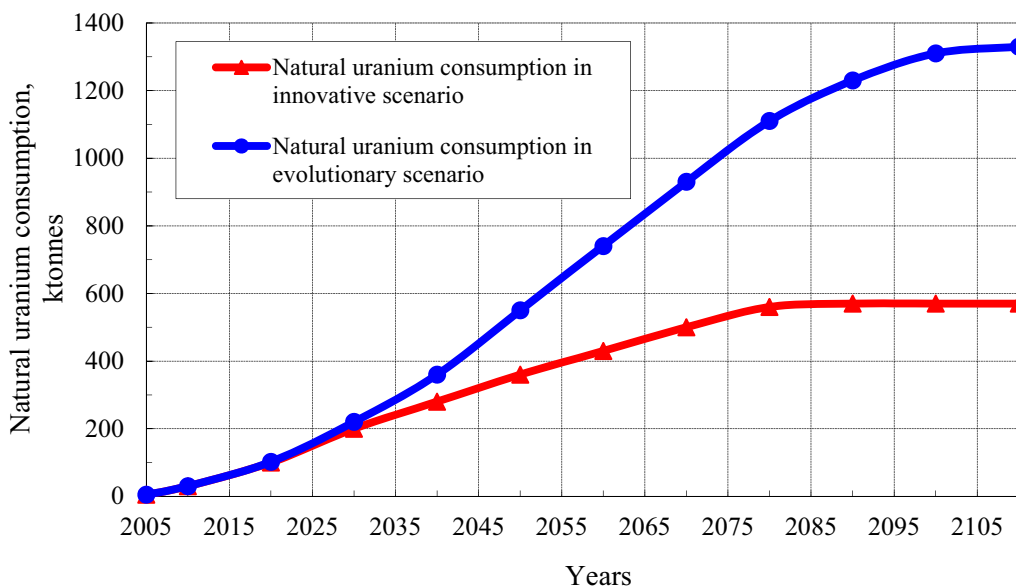

FIG. 2.1. Natural uranium consumption in the national innovative and evolutionary scenarios in the Russian Federation.

requirements for either reactor construction or reactor operation schemes. The use of mixed uranium/thorium fuel in a nuclear reactor could constitute the first experience of thorium in the nuclear fuel cycle.

The extraction of ^{233}U from spent fuel (enriched uranium and thorium) and the fabrication of new fuel assemblies based on ^{233}U is a challenge; first, the discharged fuel composition should be considered carefully. The complicated isotopic composition of heavy metal (uranium isotopes, including a great bulk of ^{238}U, plutonium isotopes, Np, other minor actinides (MAs)) with a significant amount of fissile material is a peculiarity of this kind of spent fuel. The ^{232}U fraction will reach 160 ppm, and therefore (due to the radiation field), the handling of this fuel will require special procedures.

To produce ^{233}U separately from other uranium isotopes — ^{235}U, ^{236}U, ^{237}U and ^{238}U — uranium and thorium fuel rods should be arranged separately. This idea was theoretically confirmed in the reactor concept known as Radkowsky's thorium reactor. However, this concept has no provision for spent fuel reprocessing. Proposals for its modification are outlined in Ref. [11].

Another possible approach to the production of ^{233}U in a WWER is to use a mixture of plutonium and thorium as fuel. In this case, ^{233}U produced in thorium will be free of significant admixtures of other uranium isotopes. Disposition of excess weapon grade plutonium provides another reason to study the concept of a WWER with mixed plutonium/thorium oxide fuel. Plutonium consumption and ^{233}U production characteristics of a WWER-1000 type reactor are presented in Table 2.1.

The use of plutonium in a WWER is not accompanied by the accumulation of secondary plutonium, and the ^{233}U produced is protected against non-authorized use by the high activity (primarily gamma radiation) of ^{232}U decay products. ^{232}U content in produced uranium reaches more than 3000 ppm. The investigation of a WWER type reactor with plutonium-thorium fuel is documented in Ref. [12].

The isotope ^{233}U can be also produced in the thorium blankets of a fast reactor of BN type (fast neutron reactor with sodium coolant) with MOX fuel in the core. Thorium in the blankets is introduced either in metallic or in oxide form. The metallic form is preferable from the point of view of subsequent reprocessing of spent fuel. The main advantage of a fast reactor as a ^{233}U producer is a smaller content of ^{232}U in the uranium bred in the thorium blankets (order of ~100 ppm) compared to its production in thermal reactors. The basic fuel characteristics of one of the fast reactor options with radial and lower axial blankets with metallic thorium are presented in Table 2.2.

An major indicator of the efficiency of thermal and fast reactors for ^{233}U production is the amount of ^{233}U produced per tonne of discharged heavy metal. The efficiency of the radiochemical reprocessing of spent fuel and also the volume of radioactive waste depend on this indicator. Table 2.3 demonstrates that the indicator is almost the same for both reactors under consideration. Nevertheless, it is possible to improve the indicator in a fast reactor at the cost of some decrease in ^{233}U production by reducing the size of the blanket, i.e. excluding the fuel assemblies (FAs) of the external row of radial thorium blanket that produces ^{233}U in comparatively small concentration.

TABLE 2.1. CHARACTERISTICS OF PLUTONIUM CONSUMPTION AND ^{233}U PRODUCTION IN A WWER-1000 TYPE REACTOR LOADED WITH (PuO_2-ThO_2) FUEL WITH REACTOR GRADE PLUTONIUM [12]

No.	Parameter	Unit	Value
1	Power	MW(e)	1000
2	Fuel loading	t HM	68.1
3	Annual loading	t HM	22.6
4	Fissile plutonium isotope content	%	5.4
5	Total plutonium charge	kg	5107
6	Amount of plutonium within annual feed up in equilibrium cycle	kg	1694
7	Annual ^{233}U discharge in equilibrium cycle	kg	295
8	Annual plutonium return (after recycling) in equilibrium cycle	kg	870
9	^{232}U fraction in discharged uranium	ppm	3300

TABLE 2.2. CHARACTERISTICS OF ^{233}U PRODUCTION AND PLUTONIUM CONSUMPTION IN A BN-800 TYPE FAST REACTOR WITH MOX CORE AND BLANKETS WITH METALLIC THORIUM

No.	Parameter	Unit	Value
1	Power	MW(e)	880
2	Fuel loading in core (MOX)	t HM	8.3
3	Fuel loading in blanket core (thorium)	t HM	22.4
4	Fissile plutonium isotope content	%	15.7
5	Total plutonium charge	kg	2728
6	Amount of plutonium within annual feed up in equilibrium cycle	kg	1819
7	Annual ^{233}U discharge in equilibrium cycle	kg	231
8	Annual plutonium return (after recycling) in equilibrium cycle	kg	1657
9	^{232}U fraction in discharged uranium	ppm	132

TABLE 2.3. COMPARISON OF EFFICIENCY OF WWER-1000 AND BN-800 TYPE REACTORS AS ^{233}U PRODUCERS

No.	Parameter	Unit	WWER-1000	BN-800
1	Produced ^{233}U/consumed Pu	kg/kg	0.36	1.43
2	Produced ^{233}U/discharged heavy metal	kg/t	13.1	10.5
3	^{232}U content in discharged uranium	ppm	3300	132

The ratio of produced uranium to consumed plutonium (for isotope content and burnup, see Annex I) demonstrates that the efficiency of a thermal neutron spectrum reactor is significantly lower than that of a fast neutron spectrum BN type reactor.

The concentration of ^{232}U in uranium produced in the blanket of a fast reactor is significantly lower than in the case of a WWER type reactor. This is an important issue for the selection of a radiation protection strategy for proliferation resistance, particularly at the initial stage of thorium introduction into a nuclear energy system. Nevertheless, one expects that the concentration of ^{232}U in uranium produced in the blanket of the BN-800 is still high enough to be a barrier against unauthorized proliferation of nuclear material.

The production of ^{233}U in WWER type LWRs with plutonium/thorium fuel is less effective than in BN type reactors with a MOX fuelled core and thorium blanket. The introduction of thorium/plutonium fuelled WWER reactors would lead to effective exhaustion of accumulated plutonium stock. This would reduce the potential for deployment of plutonium fuelled fast reactors planned in the Russian Federation after 2030. For fast reactors with a high ratio of plutonium production (breeding), the use of thorium as fertile material in the blankets for the production of ^{233}U to be used in thermal reactors can provide the thermal reactors with fuel long term.

2.1.3. Options for introduction of ^{233}U in thermal and fast reactors

^{233}U could be introduced in thermal reactors as a mixture of ^{233}U and thorium. However, in order to reduce non-proliferation concerns, it would be worthwhile to introduce ^{233}U into the fuel cycle as a mixture of ^{233}U and depleted uranium (U_{dep}, residuals from enrichment process) or recycled (or regenerated) uranium (U_{reg}, separated from the spent fuel of conventional LWRs). This 'denaturing' of ^{233}U with uranium strengthens the proliferation resistance of this fuel cycle and the absence of thorium in fresh fuel composition eliminates ^{232}U buildup in any significant amount and, consequently, the problems of high radioactivity caused by ^{232}U decay products in spent fuel.

Plutonium produced from ^{238}U can be recycled in thermal reactors as a mixture with ^{233}U or used in fast reactors in the form of MOX fuel. Irradiated (enriched) uranium can also be recycled again into regenerated uranium (or 'reprocessed uranium') and mixed with ^{233}U (the mixture, denaturized ^{233}U). In this case, two relatively 'dirty' types of uranium (^{233}U plus ^{232}U and ^{235}U plus ^{238}U) would be used in the same technological process that would further improve its resistance to proliferation. In addition, the option with regenerated uranium enables the demand for ^{233}U to be reduced to some extent because of the concentration fissile of ^{235}U (~1.2%) in regenerated uranium from conventional LWR spent fuel.

Neutron physics calculations, however, demonstrated a positive reactivity feedback for the water temperature at zero power conditions at the beginning of cycle of a WWER-1000 reactor fuelled with denaturized uranium. According to nuclear safety regulations of the Russian Federation, this positive feedback is inadmissible. Partial replacement of ^{235}U with plutonium leads to a negative shift of the reactivity coefficient of the coolant temperature. Therefore, the introduction of plutonium into the core is one conceivable way to meet safety requirements. In this case, both the homogeneous arrangement of plutonium and uranium in the fuel (pellets) and a separated one in individual fuel rods or even fuel assemblies become feasible. In addition, plutonium admixture allows compensation for a possible deficit of ^{233}U in the fuel cycle.

The option of ^{233}U introduction in a denaturized form together with plutonium enables a smooth switch to a ThFC with minimal modifications of reactor design, and gained experience with fresh and spent fuel management technology at the early stages. Characteristics of a WWER-1000 reactor fuelled with oxide ^{233}U–U$_{dep}$–Pu fuel regarding isotope composition at an initial stage and after the 5th recycle are shown in Table 2.4.

TABLE 2.4. FUEL CYCLE CHARACTERISTICS OF AN OPTION, THE WWER-1000 REACTOR

No.	Characteristic	Units	Initial cycle	5th recycle
1	Power	MW(e)	3000	3000
2	Fuel loading	t HM	70.2	70.2
3	Refuelling interval	EFPD	291.5	294.5
4	Discharged FAs average burnup	MW·d/kg HM	42.2	42.7
5	^{232}U fraction in discharged uranium	ppm	1.64	3.1
	Average initial fissile isotopes content in FAs			
6	^{233}U	%	1.94	2.25
7	^{235}U	%	0.19	0.16
8	Pu fissile isotopes	%	1.42	1.42
	Annual loading in the core			
9	^{233}U	kg	400.9	466.5
10	^{235}U	kg	39.8	33.0
11	Pu (including ^{241}Am)	kg	408.7	503.2
12	Pu fissile isotopes	kg	293.9	293.8
	Annual discharged amount of			
13	^{233}U (with full ^{233}Pa decay)	kg	96.4	130.7
14	^{235}U	kg	19.7	34.9
15	Pu	kg	380.4	456.3
	Annual balance of charging-discharging			
16	^{233}U (with full ^{233}Pa decay)	kg	304.5	335.8
17	^{235}U	kg	20.1	−1.9
18	Pu	kg	23.8	47.5

FIG. 2.2. Optimized scenario for introduction of thorium into the Russian Federation's nuclear energy system (fuel mass flows in fifth recycle).

Taking into account the above information, an 'optimized' scenario may be suggested for the introduction of thorium and [233]U in the Russian Federation's nuclear energy system. This scenario includes BN type fast reactors with thorium blankets for conversion of thorium into [233]U and WWER type thermal reactors utilizing this [233]U as the main component of its fuel.

Figure 2.2 illustrates the mass flow in this optimized scenario: two LWRs (conventional WWER-1000 and 'WWER-1000 ([233]U)' fuelled with [233]U-Pu-U$_{dep}$) and two BN-800 reactors with thorium blankets (BN-800-Th). The conventional WWER-1000 reactor serves as a plutonium producer for the system. The numbers in Fig. 2.2 correspond to the fifth recycle (equilibrium conditions). (The links corresponding to the dotted lines in Fig. 2.2 were not taken into account during the calculation.)

In this optimized scenario, natural uranium consumption drops down from 215 t/(GW·a) for the conventional WWER-1000 to 60 t/(GW·a) for a ThFC with fast reactors. The waste of the thorium option comprises fission products, minor actinides and irradiated uranium.

The implementation of this 'optimized' scenario can be commenced in the middle of the century, when a uranium deficit may be expected and the majority of the thermal reactors in operation will not have exceeded their design lifetime yet. It is assumed that an essential number of fast reactors will be commissioned by that time. A ThFC scenario may be further improved through the development of thorium breeders, which is a challenging task for the future.

2.1.4. Conclusion

A nuclear energy system based exclusively on thermal reactors may exhaust reasonably priced uranium resources in the Russian Federation by the middle of the century. At that time, the Russian Federation reactors — comprising a large number of WWER reactors and consuming mined uranium — must continue to operate because a significant proportion of thermal reactors will not have reached their design lifetime. A smooth transition to a denaturized ^{233}U based fuel cycle is one possible way of prolonging their operation. Denaturized uranium provides high resistance against proliferation of nuclear material.

A good option for ^{233}U production is the irradiation of thorium in blankets of fast reactors fuelled with (plutonium based) MOX fuel. The produced uranium contains ^{232}U with a concentration of 100 ppm that can be used for the fabrication of fresh fuel for conventional thermal reactors without requiring essential design modification for maintaining required safety levels. A combination of fast reactors with thorium blankets and thermal reactors consuming ^{233}U may result in a significant improvement in the economics of natural uranium use compared to a conventional thermal reactor system.

Denaturization of ^{233}U by depleted uranium or regenerated uranium in the process of fuel fabrication results in a further reduction of the ^{232}U content and, consequently, a decreased radiation burden. The combination of ^{233}U with regenerated uranium allows the use of these two relatively 'dirty' types of uranium in one technology and a lower of consumption of ^{233}U due to the significant content of ^{235}U in regenerated uranium. Negative reactivity feedback can be compensated by adding plutonium to the core.

2.2. CANADA

Most prospective options of ThFC for HWRs envisage fuel made of a homogeneous mixture of two or even more heavy metal oxides. If homogeneous ThFCs are ever to be implemented, then the issue of whether thorium fuel can be fabricated and how to fabricate it needs to be addressed. Uranium fuel used in reactors today contain only one component. Homogeneous ThFCs could also contain multiple components, namely thorium and plutonium, and potentially uranium (^{233}U). The addition of a fissile component to the fuel poses a challenge, as the components must be mixed at a microscopic level so as to avoid hot spots in the fuel.

There are two basic strategies for mixing at the microscopic level — solution blending and mechanical mixing. Both are discussed below — pellet fabrication and irradiation experience, and a summary is presented of AECL's experience with the fabrication and irradiation of thorium based (ThO$_2$) nuclear fuel [13]. The mining, refining and powder production of thorium are not covered, although sol-gel is discussed.

2.2.1. Thorium fuel fabrication for CANDU reactors

Atomic Energy of Canada Limited (AECL) has developed an extensive knowledge base with regard to various aspects of ThFCs, including pellet fabrication technology [14]. The process and equipment used to fabricate thoria pellets is the same as that used for UO$_2$ pellet fabrication. There are three notable differences:

— Pure thoria can be sintered in any atmosphere (oxidizing, reducing or inert).
— Thorium has a higher radiotoxicity than natural uranium.
— Unless pure thorium is to be used, a fissile component will have to be mixed with the thorium.

As is true for any ceramic, the production of a high density product is dependent on the quality of the starting powder, among other things.

Unlike uranium, thorium has only one oxidation state, 4+, making thorium dioxide chemically inert. This provides a processing advantage over uranium since powder oxidation is not possible and, if making pure thoria pellets, sintering can be carried out in air, reducing or inert atmosphere, compared to the reducing atmosphere required for UO$_2$. Clearly, the inert nature of thorium is also attractive from an accident and a waste management perspective.

Thoria and the solid solutions that it forms with UO_2 and PuO_2 have the same fluorite structure as UO_2. This structure is proven to have excellent resistance to (or recovery from) neutron and fission fragment damage, i.e. the fuel will not swell significantly due to these interactions.

Since thorium lacks a fissile isotope, most thoria pellet fabrication flow sheets include some process of adding a fissile component. Thoria fuel pellet fabrication experience at AECL includes the use of high enriched uranium, plutonium, and ^{233}U as the fissile additive. Thoria can be combined with a fissile component via mechanical mixing by a number of standard methods such as high intensity mixing or co-milling, and can be done wet or dry. Just as when combining any two ceramic grade powders, the required degree of dispersion of one powder in the other is largely a function of the intensity of the mixing method. Low energy or low intensity methods, such as tumbling mixers, will result in heterogeneous mixtures and hence heterogeneous sintered pellets. Since this has generally been considered undesirable, most development and production activities have focused on high energy mixing or some sort of solution blending in order to produce a solid solution of the thoria and the fissile additive in the sintered pellet. A solid solution can be achieved in the sintered pellet if the mixing method is adequate. At the same time, solution blending produces a perfectly uniform distribution and, if the fissile additive can be added during the thoria powder production stage, this method is preferable.

The sol-gel process has been used at AECL to produce (Th–U)O_2 microspheres. Microsphere development took place in the mid-1970s at Whiteshell Laboratories and in the mid-1980s at Chalk River Laboratories (CRL) [15, 16]. A sol is a liquid containing evenly distributed and stably suspended solid particles. It may be transformed to gel by the removal of the inter-particle repulsive forces. The sol-gel process can be applied to a variety of metal oxides. For the thorium sol-gel work at AECL, two approaches have been successfully applied.

For the work carried out at Whiteshell Laboratories, the hydrothermal denitration sol-gel process was used. This process is unique to thoria (not commonly used for other metal salts) and was based on a process developed at the Oak Ridge National Laboratories. To prepare the sol, a thorium nitrate solution was exposed to steam at various temperatures in a rotary calciner. The thoria powder produced from this process was combined with a uranyl-nitrate solution, and adjustments were made for concentration and pH. Spheres were formed by forcing the sol through a nozzle into an organic liquid. Under the correct conditions, the surface tension of the aqueous sol in the immiscible organic liquid causes it to form a sphere. Water is removed from the sol causing it to gel. The gelled spheres are dried and sintered to high density (>9.9 g/cm^3) at relatively low temperatures (1300°C).

Later, sol-gel work was carried out at CRL; the work was limited to pure thoria and a different technique was employed. Forced base hydrolysis of an aqueous solution of thorium nitrate was used to form the sol. Slow addition of concentrated ammonium hydroxide was used to hydrolyse the thorium nitrate. The resulting thoria was instantly dispersed in the solution provided base addition was not too rapid. Droplets were formed by pumping the sol through vibrating nozzles of various diameters. The droplets passed through counter-current flowing ammonia gas, which caused a gelled skin to form on the droplets. This skin protected the droplets from breaking up when they impacted the gelation bath of concentrated ammonium hydroxide. Gelled spheres were washed in distilled water. Green spheres (as formed) had a density of 48% of theoretical. Sintering was conducted in air, and spheres with >99% of theoretical density were achieved at a sintering temperature of 1250°C.

The conditions of solution blending of the thorium and uranium must be such that the salts precipitate together; otherwise, segregation could result. Solution blending should result in a completely homogeneous mixture of thorium and uranium since the mixing occurs at the atomic scale. Powders were prepared by co-precipitating the oxalates from a nitrate solution. The powders thus produced did not need to be milled prior to pelletizing in order to achieve high sintered densities.

A novel technique of adding a fissile component to thorium was solution impregnation. Green (pressed but not sintered) pellets of pure thorium were immersed in solutions containing enriched uranyl nitrate. The green pellet adsorbs the uranyl nitrate solution. When dried, the uranyl nitrate remains in the green pellet. Standard sintering drives off the nitrate and results in a solid solution of the uranium in the thorium, although the distribution within the pellet was not found to be uniform. The uranium seemed to bond preferentially with the surface of the thorium particles while the solution penetrated the green pellet. This resulted in the periphery of the pellet being uranium rich and the centre being almost pure thorium. Impregnation of the pellets for increasing time in the solution is shown in Fig. 2.3. Aspects of this fissile distribution are attractive from a performance perspective since the (initial) fissile material is closest to the coolant, similar to DUPLEX fuel, in which the pellet core has a different fissile content than the outer shell. The other attractive feature of this fabrication method, particularly for ^{233}U containing ^{232}U or plutonium addition, is that green pellet fabrication can take place outside shielded facility or glove box since
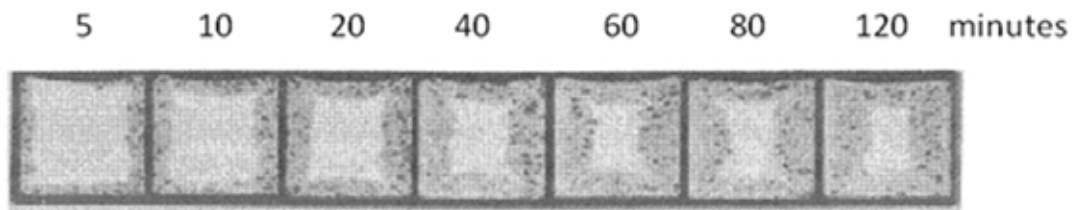
FIG. 2.3. Impregnation of pellet related to time in solution.

the green pellet has low radiotoxicity, and then the subsequent pellet processing operations involving impregnation and sintering take place in the shielded facility for ^{233}U containing ^{232}U material or glove box for plutonium bearing material. This greatly reduces the size and complexity of the containment required for pellet fabrication compared to traditional methods for fissile material addition.

Novel techniques were also applied to mechanical mixing of fuel powders. Although thorium and uranium or plutonium dioxides are mixable in all proportions, this does not ensure that a mixture of the two powders will produce an even solid solution during sintering. A solid solution can be achieved from a mechanical mixture but the powders must be dispersed at the level of individual micrometre (μm) sized particles. Sintering is a diffusion process, and since the components of the mixture are miscible, the atoms seek to disperse themselves evenly; however, since this is solid state diffusion, the distance that atoms can migrate during sintering is limited to a few microns. For this reason, very fine fissile additive particles with a high surface area to volume ratio are consumed during sintering while larger agglomerates are not. Ceramic grade powders are cohesive or 'sticky' by virtue of surface forces and their very fine particle size distribution. If the *powders are mixed by a low energy method*, *agglomerates of the original powders will remain*, and this arrangement survives into the green and sintered pellet. Discrete regions of the pure or near pure fissile additive will remain in the pellet. Their size and number will be determined by the properties of the starting powders and the mixing method applied. AECL's experience has centred on co-milling the thorium with the fissile additive.

Milling is an extremely high energy process. The same action that would serve to break up individual particles into smaller pieces also has enough energy to break up agglomerates of powder. Co-milling is a conservative approach to mixing powders in that the intimate mixing of the powders is virtually assured. There are disadvantages to co-milling, such as contamination from the milling media and vessel, and either the separation of the media from the powder (if dry milling) or the drying of the powder (if wet milling).

Some of these disadvantages can be reduced or eliminated by the use of high intensity mixing. With high intensity mixing, the energy is imparted to the powder via a high speed rotor of some sort. The tips of the rotor tines travel at great velocity and break up the agglomerates upon impact. The effectiveness of this method depends on the rotor-tine-tip speed and the nature of the powder agglomerates. Agglomerates are often roughly categorized as soft or hard. Soft agglomerates are those that are easily broken up, such as by high intensity mixing, whereas hard agglomerates require a milling method to break them up. Non- or soft agglomerated powder is strongly preferred because it allows for the use of high intensity mixing over the more laborious co-milling.

The following is a list of mechanical mixing techniques examined by AECL:

— Twin shell blender with an intensifier bar (a high speed rotor) followed by two passes through a high speed blade mill;
— Wet attrition milling;
— Dry vibration milling (several variations);
— Tubular mixer;
— High intensity mixer;
— Homogenizer (wet).

Various granular pellet structures were created and investigated. It has been observed that wet-milled thoria powders can form high density cakes if allowed to dry by evaporation. Presumably, the broad particle-size distribution produced during milling allows for efficient packing during drying. Dried cakes were broken up by forcing them through a standard sieve screen. The resulting granules made a free flowing, final press feed, and green pellets of adequate strength for handling were produced. The sintered pellet densities were high (≈ 9.6 g/cm^3),
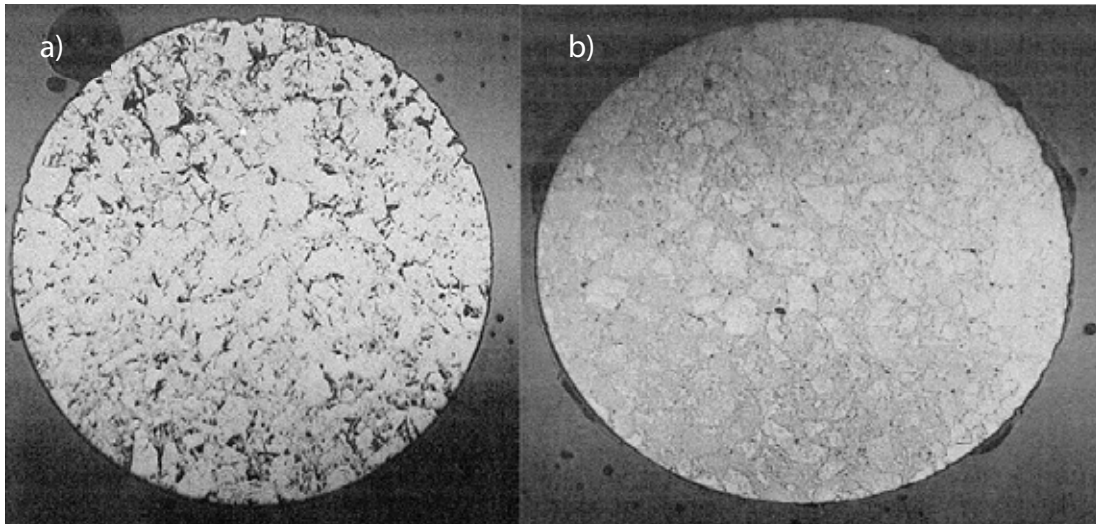
FIG. 2.4. Examples of granular pellet structure: (a) granular microstructure with large, open pores; (b) granular microstructure with low density porous regions between the high density granules.

but macroscopic examination of a cut and polished surface revealed a granular structure. Some of the dried cake granules did not break up completely during final pressing and their structure remained in the sintered pellets. The microstructure within the granules was homogeneous and very dense, but the low density (porous) regions between granules created a barrier for heat conductance out of the pellet (resulting in elevated operating temperatures) and provided a less resistive path for fission gasses to escape to the free volume of an element. The performance consequences of such a structure are discussed in detail in the Irradiation Experience section of this report. Two examples of granular microstructures observed in thorium pellets are given in Fig. 2.4.

A number of standard and novel approaches to thorium pellet fabrication have been explored at AECL; each process has advantages and deficiencies. The process of choice for fabricators will depend on the type of fuel required (HEU/Th, Pu/Th, ^{233}U/Th, pure thorium, pellets, spheres) and the conditions and constraints that they must work under. Some general observations regarding thorium pellet fabrication:

— The properties (particle size, size distribution, surface area, etc.) and sinterability of the starting thoria powder are the most important considerations when determining the appropriate process to apply to achieve the addition of a fissile component.
— Caution must be taken during choice and execution of pre-treatment processes to avoid granular microstructures.
— Thoria powders are not as amenable to the cold pellet pressing process as UO_2 due to a tendency to end cap (delamination of the ends of the green pellet).
— Pre-irradiation characterization of pellets must be thorough and always include macrostructure and microstructure examination of cut and polished pellets.

2.2.2.   Thorium irradiation experience and post-irradiation examinations

There is significant experience within AECL in thorium based fuel irradiation, with burnup of up to 47 MW·d/kg HM, and power ranging up to 77 kW/m. Irradiation is ongoing and more is planned for the future. The composition of the fuel includes natural thorium oxide (ThO_2), thorium + high enriched uranium oxide (Th-U)O_2 and a thorium-plutonium oxide fuel (Th-Pu)O_2. Pellet geometry was generally standard (length over diameter (L/d) ≈ 1.3), but several experiments examined short pellets (L/d ≈ 0.5) and a variety of pellet designs.

Early experiments showed great promise for ThO_2 based fuel, with fuel performance parameters superior to UO_2 under similar operating conditions. These results created an incentive for various experiments in the late 1970s and early 1980s. An important lesson learned during this period was the importance of uniform, non-granular structure in order to ensure superior performance from the thorium fuel. It is noted that even poorly fabricated (granular) thoria performed comparably with high quality UO_2 fuel.
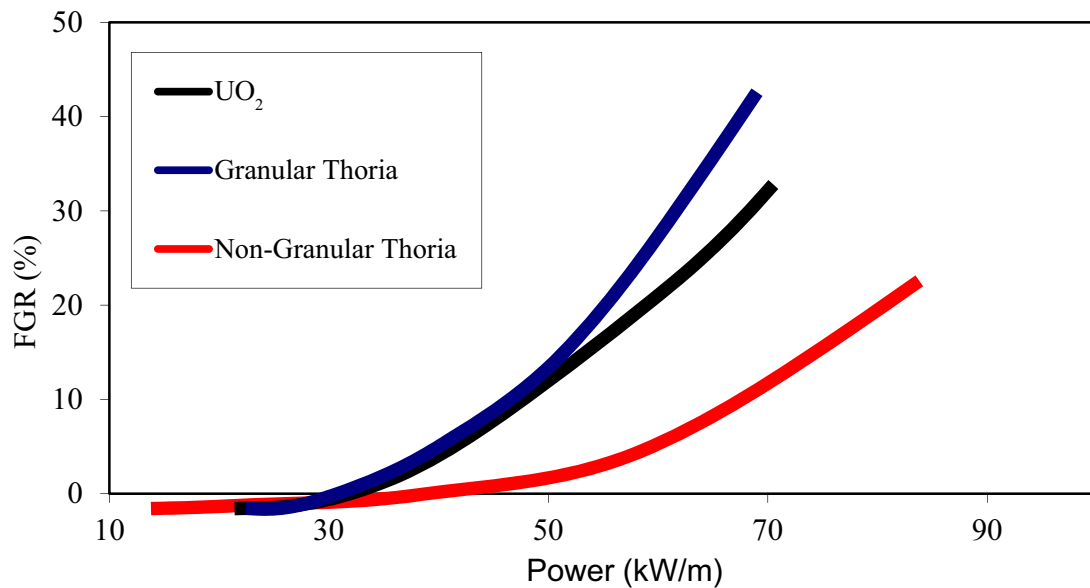
FIG. 2.5. Fission gas release (%) of UO$_2$ and thorium fuel.

Throughout these experiments, numerous fuel performance parameters were investigated and correlated, including fuel power, fuel burnup, fission gas release, pellet microstructure, sheath strain, sheath corrosion, sheath hydriding/deuteriding behaviour, CANLUB performance, power ramping, and defect performance. Some results on the fission gas release aspects of these ThO$_2$ based fuels, with emphasis on the effects of pellet microstructure, are shown here.

Fission gas release (FGR) is primarily dependent on element power (fuel temperature), with fuel burnup as a secondary variable [17]. Figure 2.5 plots FGR versus element power, and compares UO$_2$ to granular and non-granular thorium. Below ~40 kW/m fuel microstructure plays a minimal role in FGR due to the low fuel temperature. The lines drawn in the figure represent data trends, and demonstrate comparable performance between UO$_2$ and granular thorium; non-granular thorium demonstrates superior performance.

An overview of the effect of pellet microstructure on fission gas release in thorium based oxide fuel is summarized in a paper by Smith et al. [18]. This paper demonstrates that granular ThO$_2$ results in elevated central temperatures (despite the higher thermal conductivity of the ThO$_2$), which subsequently causes increased fission gas release (comparable to that of similarly operated UO$_2$).

The granular nature of the poor thoria structure fuel causes degradation in the thermal conductivity of the fuel. This results in higher operating temperatures, particularly near the pellet centre, which increases thermal diffusion of the fission gases out of the fuel grains and into the void space. The granular fuel often contains networks of tunnels that assist fission gas transport to the free void volume (Fig. 2.4).

More recent thorium irradiation experiments confirm improved FGR performance of thorium fuel over UO$_2$ of similar burnups. These recent tests are represented by the 'non-granular' line in the Fig. 2.5. The ratio of the granular FGR to non-granular FGR above 40 kW/m is approximately 2–4. Therefore, high quality non-granular thorium fuel exhibits significantly less fission gas release (two to four times less), even at higher power ratings and burnups.

Thorium based fuel has been successfully operated at high power to extended burnups, with excellent fuel performance. Higher thermal conductivity, which resulted in lower operating temperatures, resulted in low FGR (relative to UO$_2$). This improved performance is achieved only when the as-fabricated microstructure of the thorium is of high quality and contains no as-fabricated granules.

2.2.3. Summary

Fabrication development activities have resulted in several successful traditional and novel methods of fuel production, from tumble mixing and cold pressed pellets to sol-gel microspheres. One of the important observations from these investigations is the propensity of thorium powders to produce granular microstructures. Care must be

taken in any type of pre-compaction process, be it purposeful (such as pre-pressing), or coincidental (such as air drying of slurries) in order to avoid granular microstructures.

Virtually all fabrication methods investigated have been followed up with irradiation testing and post-irradiation examinations. FGR behaviour is strongly linked to pellet microstructure. Inhomogeneous or granular as-fabricated microstructures tend to release more gas to the free volume, due to reduced thermal conductivity (higher fuel operating temperatures) and the presence of a less resistive path for escape. Thoria fuel with dense, homogeneous microstructures has less gas release than UO_2 fuel under similar conditions, but even granular microstructure thoria fuel has gas release performance similar to 'good' UO_2.

2.3. NORWAY

The Norwegian company Thor Energy continues to work on the design and licensing of viable thorium fuel for LWRs. The company assesses the introduction of thorium as a fertile component for LWR MOX fuel as by far the shortest route towards deriving appreciable energy share from thorium.

According to Thor Energy, thorium fuel for LWRs fits into the current fuel cycle context because:

— Thorium fuel should be attractive to utilities when uranium/SWU (enrichment) savings become sufficiently strong imperatives. Uranium resources are secure for a long time, but prices are likely to be substantially higher at some point — probably after 2020.
— LWRs are here to stay as the nuclear power generating workhorse for the rest of the century.
— Fast reactors are meritorious, but have proven slow to license and deploy. It will be at least three decades before there are enough fast reactors to serve the nuclear industry. Thorium LWR fuel can be designed to achieve some of the actinide management and high fissile conversion goals that are expected of fast reactors, but without the difficulty of bringing into service a new reactor type.
— Thorium-MOX fuel offers a credible plutonium management option that leads to more sustainable nuclear fuel use than current modes of using UOX and uranium-MOX fuel.
— Proliferation concerns will remain, which will centre on inventories of accumulated plutonium in SNF and with the ubiquity of centrifuge enrichment technology. Thorium-MOX fuel may utilize/destroy plutonium in SNF and do not require enrichment services.

Thor Energy has two lines of work:

— The design and planning of an irradiation experiment for a candidate thorium-plutonium oxide fuel (thorium-MOX).
— The design and modelling of thorium-MOX and thorium-^{233}U fuel assemblies for BWRs.

2.3.1. Irradiation testing of prototypical thorium-plutonium mixed oxide fuel

A sophisticated data collection programme has been designed in which a number of thorium-plutonium oxide fuel pins will be irradiated in simulated LWR conditions in the fuel-testing reactor in Halden, Norway. The fuel will be prototypical of what can be fabricated commercially as a variant of today's uranium–MOX fuel. The irradiation will be performed by the Institutt for Energiteknikk (IFE) operators of the Halden Reactor.

Thor Energy is undertaking this thorium fuel test activity in the knowledge that:

— There is a paucity of data on the irradiation behaviour of thorium-plutonium oxide fuel, especially those with characteristics akin to current MOX fuel. Licensing such a fuel will call for high quality data to support the verification and modelling of its behaviour under operating conditions;
— Due to the length of the fuel licensing process, testing activities should be commenced as soon as possible, with a view to lead-test rod/assembly (LTA) testing in a commercial power reactor.

The data collection plan is summarized in Table 2.5 according to the fuel behaviour that needs to be characterized. The ultimate goal for the activity is to create a set of quality data that demonstrate that the fuel

TABLE 2.5. CHARACTERISTICS OF THORIUM FUEL TO BE DETERMINED EXPERIMENTALLY

	Data collection to quantify behaviour	
	On-line experimental measurable	Data from pre/PIE measurements
Thermal behaviour to characterize		
Thermal conductivity decay	Centreline temperature, including from a pin operating at higher temperature	Thermal conductivity of fresh pellet, Thermal conductivity of spent fuel
Thermal conduction pathway changes	Centreline temperature, cladding elongation (gap closure)	Neutron radiography, microscopy
FGR behaviour to characterize		
FGR onset	Rod pressure, centreline temperature	
FGR amount	Rod pressure, including from pin operating at higher temperature	Rod-puncture gas analysis
FGR composition		Rod-puncture gas analysis
Mechanical behaviour to characterize		
Cracking		Neutron radiography, gamma scanning
Densification	Fuel column elongation, temperature	
PCMI and swelling	Fuel column elongation, cladding elongation, centreline temperature	Microscopy, fuel rod profilometry
High burnup structure	Rod pressure, fuel column elongation, cladding elongation, temperature	Thermal conductivity, microscopy
Chemical behaviour to characterize		
Stress corrosion cracking		Microscopy and EPMA, rod puncture gas analysis, neutron radiography
Oxygen mobility and high oxygen-affinity fission products		Microscopy and EPMA, rod puncture gas analysis

ceramics operate safely in normal operating conditions. This data will support follow-on testing of thorium-MOX fuel segments in a commercial power reactor, leading to the testing of the fuel in transient conditions [19]. Collectively, these data are vital for the safety licensing of such a new fuel.

The thorium-MOX fuel to be tested in this programme is prototypical of commercial MOX fuel in its microstructure and its composition. Fuel behaviour is sensitive to the physical make-up of the fuel pellet, so it is important that a commercially oriented fuel programme tests fuel material that is closely matched to what can be produced industrially. If not, the irradiation data may be inadmissible for demonstrating the safe performance of the fuel. This is even more important for a two phase plutonium bearing fuel because there are more variability options in terms of: (i) plutonium isotope vector, (ii) plutonium homogeneity, (iii) americium content, and (iv) non-metal impurities (C, N, H).

These 'MOX parameters' exist in addition to the normal pellet properties of density, grain size and oxygen stoichiometry, which also need careful specification and for which manipulation and control are different in a thoria based ceramic. Thor Energy has defined a thorium-plutonium MOX test fuel specification taking careful account of all desired properties, control parameters and logistic restrictions relating to handling plutonium fuel. It is noteworthy that the thorium-MOX fuel for this experiment will be made using a co-milling process that gives a micro-heterogeneity of plutonium distribution that is similar to the MOX fuel produced in commercial plants in France and the UK.

Thor Energy is building an international consortium whose members will collectively steer the irradiation experiment, co-fund the undertaking and share all resulting data. The consortium is open to all interested parties. The irradiation experiment will commence in late 2011 and run for around five years.

2.3.2. Thorium fuel design for BWRs

Thor Energy has undertaken a number of sponsored studies on optimal thorium fuel designs for the BWR platform. One key design direction has been to maximize the use of plutonium in a BWR fuel assembly. The other direction has been to maximize the conversion of thorium into ^{233}U in a reduced–moderated BWR (RBWR). These optimization modes are quite distinct and invoke very different design strategies.

Thorium-plutonium BWR fuel design

The operation of thorium-MOX BWR fuel assemblies has been modelled using the Studsvik-Scandpower CASMO-5–SIMULATE-3 simulation codes — multigroup 2-D transport theory codes for performing depletion calculations and collapsing cross-sections for subsequent input to less rigorous models. The unadjusted JEFF 2.2 cross-section library was used in this study. The study looked at the effect of fuel composition, fuel geometry and moderation ratio on plutonium utilization and safety indicators (void coefficient, shutdown margin and linear heat generation rate). The main findings are listed below:

— The highest reactivity (and thus efficiency of plutonium use) for a thorium-MOX BWR fuel is reached with a moderation (H/HM) ratio that is significantly higher than those giving the highest reactivity for UOX fuel.
— Higher plutonium fractions tend to give higher efficiency of use of that plutonium — within practical limits.
— Highly moderated fuel designs, however, have a reduced total fuel mass and thus a concomitant higher relative fuel cost.
— The different design options for varying the H/HM ratio — modifying the fraction of coolant or moderator in the fuel assembly — have significantly different effects on the coolant void reactivity and moderator temperature coefficients.
— Water channel size and placement are key for determining (maximizing) the reactivity difference between hot full power and cold zero power (CZP). An optimum channel size is close to the neutron migration area at CZP.
— Generally, when the H/HM ratio exceeds the value giving the highest reactivity for the thorium-MOX fuel, reactivity coefficients become positive.

While none of these results is surprising, they provide good guidance for continuing efforts to design thorium-MOX fuel with high plutonium efficiency/consumption. This requires finding a balance between maximizing the content of both fuel and moderator (water) in the assembly.

The operation of thorium-MOX fuel in a standard 10×10 'GE14-N' assembly was also compared with that of other related oxide fuel in the same geometry [20]: LEU, uranium-MOX, thorium-^{233}U, thorium-LEU and thorium-plutonium + recycled-uranium from thoria fuel (~89% ^{233}U and 11% ^{234}U). Burnup profiles for these are shown in Fig. 2.6. The flatter depletion profile for thorium-MOX fuel is evident, which is a favourable feature. Operating safety indicators were computed, including power peaking, reactivity coefficients and control rod efficiency. Findings include the following:

— Thorium-plutonium MOX fuel is neutronically similar to corresponding uranium-MOX: power peaking factors are very similar (higher than UOX fuel at end-of-life) and control rod efficiency is very similar (lower than for UOX fuel).
— Moderator temperature coefficients and coolant void reactivity coefficients for thorium-^{233}U fuel are significantly more positive over the life of the fuel compared to other fuel types. Special design measures would be needed to ensure negative coefficients for such fuel.
— If efficient destruction of stockpile plutonium is sought, thorium-MOX is superior to uranium-MOX, but more plutonium is needed.
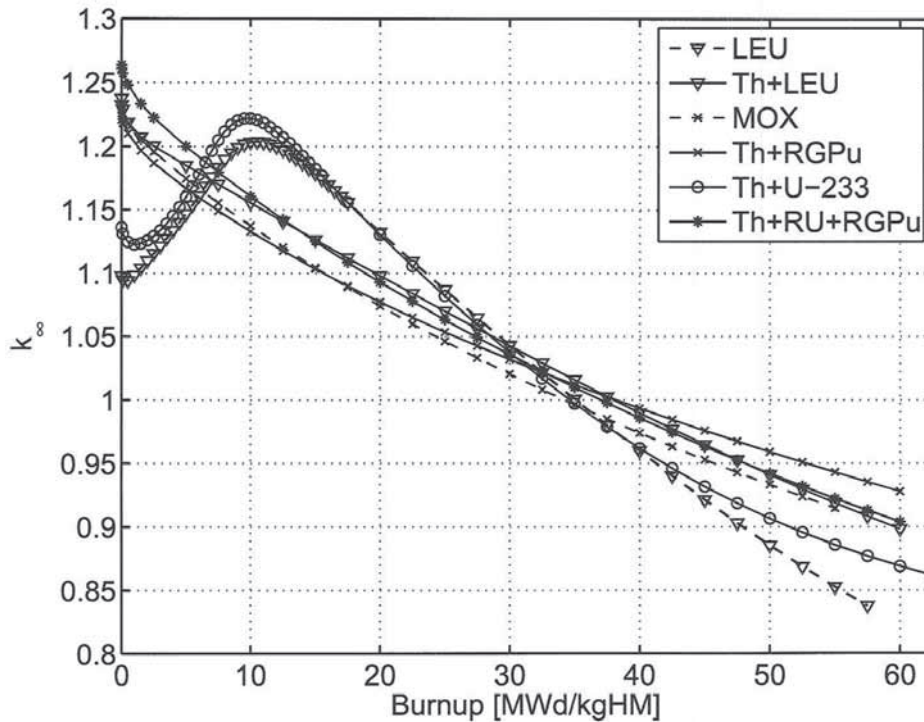
FIG. 2.6. Infinite multiplication factor (reactivity) dependence on burnup for various thorium fuels in 'GE14' 10 × 10 BWR assembly, together with MOX fuel and LEU uranium fuel for comparison.

— It is not possible to achieve uranium savings using thorium-LEU fuel.
— Useful power share of 25–30% of energy output is derived from thorium in BWR fuel containing this fertile element. This seems indicative of what can be attained in optimized designs.

These results illustrate the broad feasibility of designing and operating thorium fuel for BWRs. Due to the similarity of thorium-MOX fuel to uranium-MOX fuel, there is a reasonable basis for the design of viable thorium-MOX BWR assemblies. Features used to address control rod efficiency and power peaking factors for today's MOX fuel will be applicable for thorium-MOX fuel, but special attention will need to be paid to lowering reactivity coefficients in the design of thorium-^{233}U BWR fuel.

Thorium-^{233}U fuel for a reduced-moderation BWR

There continuing motivation to explore the limits of fissile breeding from thorium fuel in thermal spectrum systems. A self-sustaining ^{233}U-ThFC is a desirable transactinide free goal. Thor Energy undertook the exploration of thorium-^{233}U oxide fuel operation in a Gen III+ reduced-moderation BWR (RBWR) since this reactor platform should be well suited for achieving high ^{233}U conversion factors due to its hard/epithermal neutron spectrum. The 'resource-renewable BWR' concept has been developed in Japan by Hitachi Ltd. and JAEA [21], and is also referred to as the reduced moderation water reactor (RMWR), or the flexible fuel cycle LWR (FLWR). It was designed as a platform for flexible uranium–plutonium fuel in which high conversion or actinide destruction can be achieved. Physically, it is based on ABWR architecture but has a shorter, flatter pancake shaped core and a tight lattice to ensure sufficient fast neutron leakage and negative void reactivity coefficient in a LOCA scenario.

The study aimed to shed light on: the characteristics of uranium that is multi-recycled in a RBWR, specifically which uranium isotopes attain equilibrium, and the extent to which ^{233}U conversion can occur and be maximized (measured as fissile inventory ratio). Uranium from the fifth recycle was taken as the fissile feed for studies on RBWR core configurations.

Neutronic analyses were performed using the MCNP-4C transport code [22] and a 3-D model of the RBWR core developed from the open literature. A coupled code 'BGCore' gives burnup calculations and uses the JEFF-3.1

17

evaluated cross-section library. A core configuration designed for a uranium–plutonium fuel was initially used; however, this is adequate for demonstrating the evolution of the uranium vector.

The main results are as follows:

— The multi-recycled uranium vector was found to attain approximate equilibrium after six–seven cycles, although not with the content of ^{236}U and ^{238}U, which are seen to rise with each recycle stage.
— The equilibrium content of ^{232}U is 1700 ppm, which is high from the point of view of radiation protection requirements, but it is reassuring that it saturates at some level.
— Equilibrium for ^{234}U is still not attained after five cycles, but is probably reached after seven cycles at ~27 at.%, which is quite high.
— The uranium vector reactivity worth deteriorates with each successive recycle and top-up loading is needed at each stage such that the uranium content in the fuel rises from 13 to 18% after five cycles. The reactivity penalty from ^{234}U and ^{236}U is somewhat compensated for by ^{235}U production.
— The extent of ^{233}U conversion is very sensitive to core configuration. Modifying the size and number of blanket zones may improve breeding, trading off with an inherent negative impact on k_{eff} and achievable burnup.
— Maintaining a negative void coefficient in this system seems achievable but power peaking needs careful attention, especially when high enrichment is used in the seed fuel.
— The largest single positive impact on conversion ratio comes from increasing the heterogeneity of the core and matching the size of seed-blanket zones with void dependent mean neutron path lengths.
— The thorium absorption rate depends on neutron energy and is higher in blanket material in lower void regions, and in all cases it peaks 3–6 cm from the seed region.

A fissile inventory ratio and k_{eff} are shown in Figs 2.7 and 2.8 as a function of burnup for a number of different core configurations. Note that it is possible to find a configuration that shows a positive conversion over a burnup period of 1800 EFPD so that a 3-batch core with 500 days cycle seems feasible.

This study was undertaken in an infinite radial approximation and more detailed studies are warranted on such topics as thermal hydraulics, power density variations, blanket compositions, core simulations.
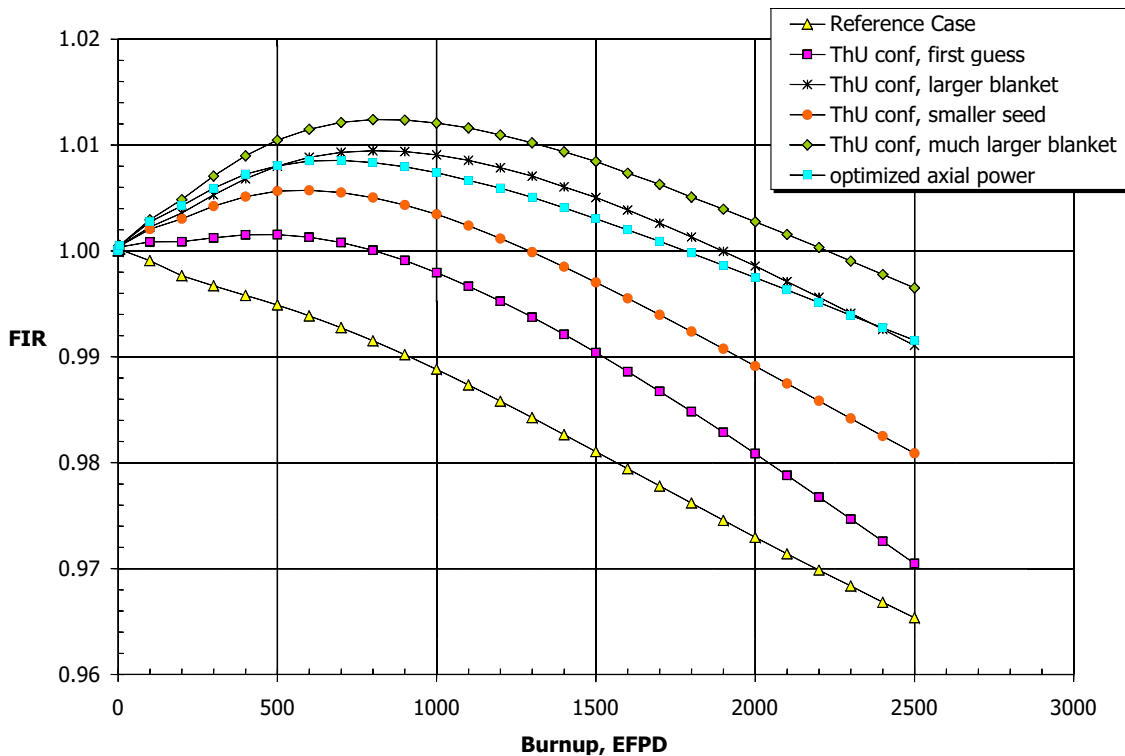

FIG. 2.7. Fissile inventory ratio (FIR) as a function of burnup (EFPD).
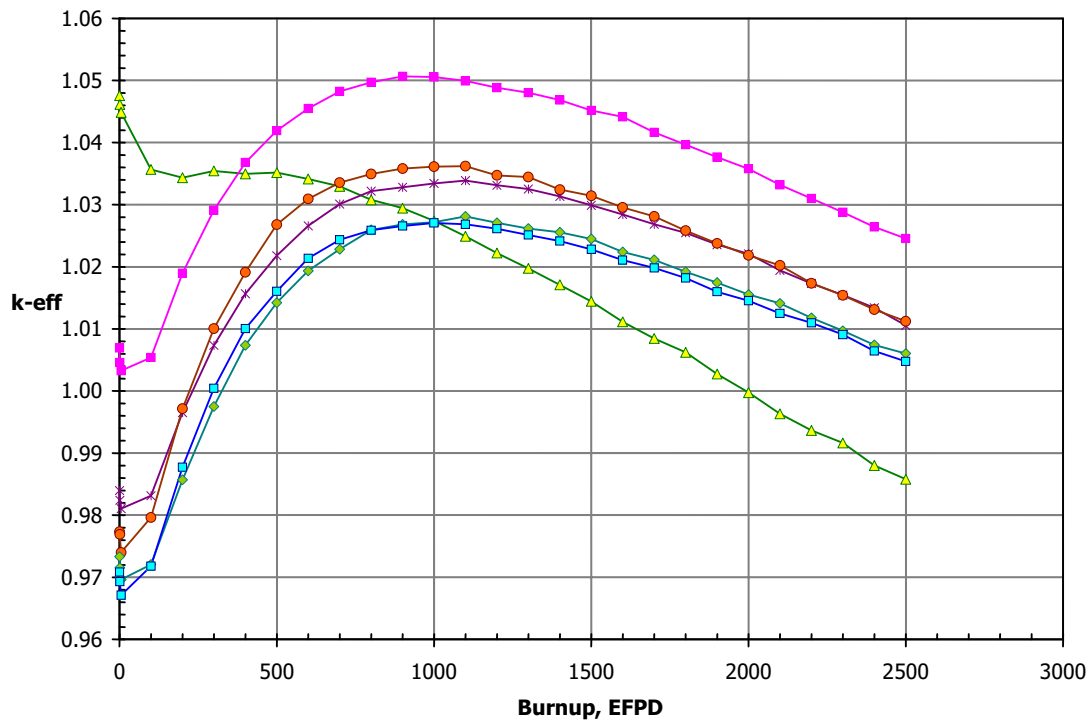
FIG. 2.8. Reactivity as function of burnup (EFPD).

3. RESEARCH AND DEVELOPMENT OF THORIUM IN NUCLEAR FUEL

This section presents results of several Member States' R&D activities related to the use of thorium in their nuclear fuel cycle.

3.1. INDIA

The AHWR currently under design in India is a 920 MW(th), vertical pressure tube type thorium based reactor cooled by boiling light water and moderated by heavy water [23–26]. The main objectives of the core design are to achieve a relatively higher fraction of power from Th/^{233}U, a negative void reactivity coefficient, minimization of initial inventory and consumption of plutonium, self-sustaining characteristics in ^{233}U and a high discharge burnup. Plutonium is used as makeup fuel to achieve high discharge burnup and self-sustaining characteristics of Th–^{233}U fuel cycle.

The basic fuel cycle is based on the goal that the AHWR core will be self-sustaining in ^{233}U. The ^{233}U required is to be bred in situ. The loss of fissile ^{233}U in the (Th–^{233}U) MOX pins is to be compensated by conversion of fertile ^{232}Th in the (thorium–plutonium) MOX pins to fissile ^{233}U. However, calculations show that there will be an annual deficit of about 22 kg of ^{233}U if the core is refuelled with only Standard D5 composite clusters having 30 (Th–^{233}U) MOX and 24 (thorium-plutonium) MOX pins. Hence, an alternative cluster was designed to generate the required ^{233}U to make the AHWR core self-sustaining in ^{233}U (the 'AHWR-Reference').

Additionally, the reactor offers enough flexibility to accommodate different kinds of fuel cycles. A few cases were studied to achieve high discharge burnup by using LEU as external feed in thorium oxide fuel (AHWR–LEU). The enriched uranium with initial configuration 19.75% ^{235}U and 80.25% ^{238}U was used together with thorium in all the 54 pins of the AHWR fuel cluster, and calculations were done for the equilibrium fuel cycle. It was found that

the case with an average LEU content of 21.3% gives a high burnup of about 64 MW·d/kg HM. However, there is better utilization of natural uranium resources. The fissile content in the uranium recovered after reprocessing the discharged fuel is about 8.0%, which is denatured and contains sufficient quantity of ^{232}U to make it further proliferation resistant. Also, the discharged fuel contains less plutonium and less minor actinides. All the reactivity coefficients are negative. The following section discusses the above results in more detail.

3.1.1. Core design optimization for achieving self-sustainability

Lattice simulations have been performed using the WIMSD code system [27] and the ITRAN code [28–31] based on transport theory. The nuclear data library used was the 69 group multigroup cross-section set based on the ENDF-B/VI.8 dataset [32]. The core calculations were made using the diffusion theory code FEMTAVG [33].

Standard D5 composite cluster

The D5 standard composite cluster consists of a circular array of 54 fuel pins [34]. The fuel assembly has a central multipurpose displacer of 36 mm OD and a 3 mm thick Zircaloy tube containing a solid Zircaloy rod in the lower half (18 mm diameter) and a steel rod in the upper half as a mild absorber. The inner and intermediate array of 12 and 18 pins contain (Th-^{233}U) MOX and the outer 24 pins contain (thorium-plutonium) MOX. The innermost array of 12 pins has a ^{233}U content of 3.0% by weight, and the middle 18 pins have 3.75% ^{233}U. The 24 (thorium-plutonium) MOX pins in the outer array have an axial gradation of 4.0% plutonium in the lower half of the active fuel and 2.5% plutonium in the upper part.

Alternative cluster design for self-sustenance

In the alternative cluster, the fuel in inner 12 pins is replaced by 4.0% (thorium-plutonium) MOX, which is the same as used in the lower half of the outer ring. The enrichment used in these pins is constant from top to bottom. Graded enrichment is employed in the outer ring pins only. Also, only a Zircaloy rod is used in the displacer (no segmentation). The cluster diagram is shown in Fig. 3.1; cluster data are given in Table 3.1 with differences indicated between standard and alternative clusters wherever they are.

The isotopic vectors of uranium and plutonium in the discharged fuel are given in Tables 3.2 and 3.3. In these calculations, the initial uranium is considered to be ^{233}U only, whereas the plutonium used is the one discharged from PHWRs at an average discharge burnup of 6.7 MW·d/kg HM having an isotopic vector of ^{239}Pu/^{240}Pu/^{241}Pu/^{242}Pu as 68.79/24.6/5.26/1.35. In the discharged fuel, ^{233}Pa content has been added to ^{233}U and
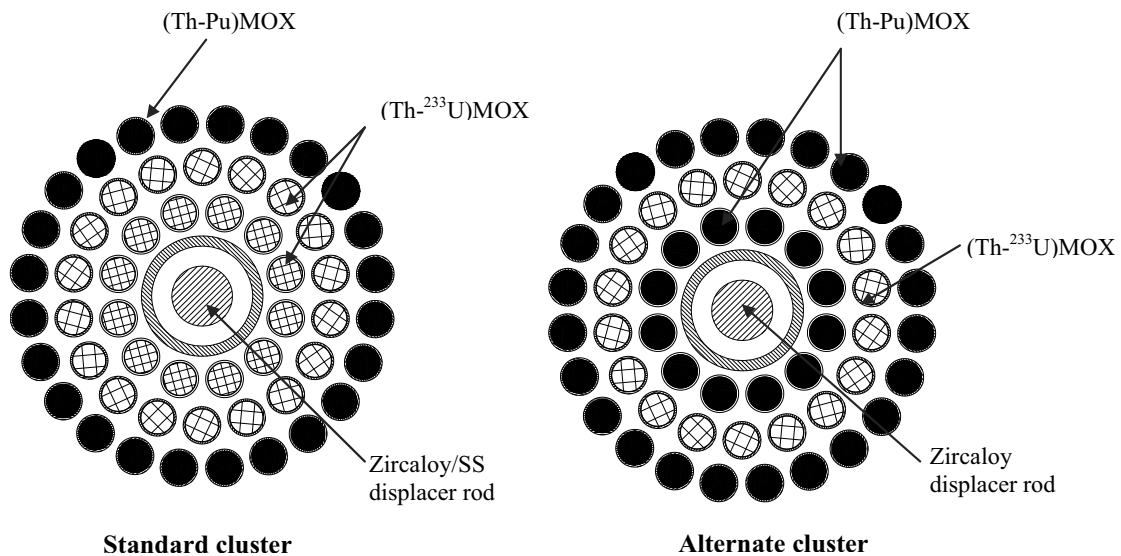
FIG. 3.1. Cross-section of fuel cluster of the AHWR.

TABLE 3.1. DESCRIPTION OF THE LATTICE OF THE AHWR FUEL CLUSTER FOR EQUILIBRIUM CORE

Description	Physical dimensions/specifications
Lattice pitch, mm	225
Fuel	
Total no. of fuel pins	54
No. of fuel pins in each ring	12/18/24
Pitch circle diameter, mm	51.4/77.4/103.7
Fuel pin OD/fuel pellet OD, mm	11.2/9.8
Fuel density, g/cc	9.3
Heavy metal, kg	116.5
AHWR composite cluster of (Th, Pu)MOX and (Th,^{233}U)MOX	
Ring 1 — standard cluster	(Th-^{233}U)MOX, 3.0%
Ring 1 — alternative cluster	(Th-Pu)MOX, 4.0%
Ring 2	(Th-^{233}U)MOX, 3.75%
Ring 3 (Two axial enrichments)	(Th-Pu)MOX, 4.0% (lower half); 2.5% (upper half)
Pressure tube (PT) material and dimensions, ID/OD, mm	Zr-2.5% Nb, 120.0/128.0
Calandria tube (CT) material and dimensions, ID/OD, mm	Zircaloy-2, 163.8/168.0
Multipurpose displacer (annular)	
Displacer tube, material/OD/thickness , mm	Zircaloy-2/36/3
Solid rod inside displacer, material/OD, mm -standard cluster	Stainless steel (upper half); Zircaloy-2(lower half)/18
Solid rod inside displacer, material/OD, mm -alternative cluster	Zircaloy-2 (throughout)/18
Coolant	
Material	Light water
Average coolant density, g/cc/temperature, C	0.45/285
Moderator	
Material	Heavy water
Average moderator density, g/cc/temperature, °C	1.089/67.5

^{239}Np content to ^{239}Pu. It is seen from the table that plutonium burns efficiently in presence of thorium and the fissile content reduces from 75 to 23%.

All the reactivity coefficients, i.e. void coefficient, fuel temperature coefficient, channel temperature coefficient and the moderator temperature coefficients for the alternative cluster, are nearly the same as those for the standard cluster. However, the burnup of this cluster is lower, which implies that the core average discharge burnup will be reduced.

TABLE 3.2. ISOTOPIC VECTORS OF URANIUM IN DISCHARGED FUEL (% OF HM)

Fuel type/discharge burnup	^{232}U	^{233}U	^{234}U	^{235}U	^{236}U	^{237}U	^{238}U
AHWR-Reference: (^{233}U-Th) and (Pu-Th) MOX (35.5 MW·d/kg HM)	0.14	83.33	13.93	2.30	0.30	0.0	0.0
AHWR-LEU: 21.3% LEU in Th (64 MW·d/kg HM)	0.02	6.53	1.27	1.52	3.25	0.0	87.41

TABLE 3.3. ISOTOPIC VECTORS OF PLUTONIUM IN DISCHARGED FUEL (% OF HM)

Fuel type/discharge burnup	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
AHWR-Reference: (^{233}U-Th) and (Pu-Th) MOX (35.5 MW·d/kg HM)	2.27	2.09	30.75	21.04	43.85
AHWR-LEU: 21.3% LEU in Th (64 MW·d/kg HM)	9.62	41.28	21.17	13.85	14.08

3.1.2. Core simulations of AHWR reference design

The AHWR is being designed to produce a total power to the coolant of 920 MW(th). The reactor core of the AHWR-Reference design consists of 513 lattice locations in a square lattice pitch of 225 mm where fuel assemblies occupy 452 locations [35]. Sixty-one locations are reserved for the reactivity control devices and the shutdown system no.1 (SDS-1), of which 37 locations are used for locating shutoff rods (SORs) and 24 for locating control rods (CRs). The reactor also has an independent secondary shutdown system based on a diverse principle, the 'shutdown system No. 2 (SDS-2)'. The core layout with the locations of the reactivity devices is shown in Fig. 3.2. During nominal conditions, the absorber rods (ARs) remain fully IN (in the core), the regulating rods (RRs) are 67% IN and the shim rods (SRs) are fully OUT (out of the core).

The calculations show that for the equilibrium core refuelled with the standard clusters only, the annual fuelling rate is 73 fuel assemblies, which requires 123 kg plutonium and results in an annual deficit of about 22 kg of ^{233}U [35]. The average discharge burnup is about 38 MW·d/kg HM. In order to achieve an equilibrium core that is self-sustaining in ^{233}U and meets all the other design objectives and safety features of AHWR, the core was loaded with 228 alternative clusters in the outer zone and 224 standard clusters in the inner zone. The core loading pattern is shown in Fig. 3.2. It is observed that the average discharge burnup of this core is about 35.5 MW·d/kg HM. The self-sufficiency in ^{233}U is achieved by employing a combination of two types of clusters and an annual feed of 39 fuel assemblies of each type is required. The annual requirement of plutonium will be about 173 kg. The change in reactivity per C due to change in temperature from any reactor state to another state is defined as temperature reactivity coefficient. The core design features and the reactivity coefficients of this core are given in Table 3.4.

Although the channel temperature coefficient is positive, it can easily be controlled by the reactor regulating system. The moderator temperature coefficient is also low and positive, but is within the range of the regulating system. The void reactivity, however, becomes less negative with burnup, but remains negative throughout the burnup range. The reactivity swing from cold critical to hot zero power is positive, but is well within the range of the regulating system, while the reactivity swing from hot zero power to full power is negative. The power coefficient is always negative.

The channel power distribution was optimized by adopting three burnup zones in the core of 28.5, 34.5 and 48 MW·d/kg HM. The average discharge burnup of this core is about 35.5 MW·d/kg HM and the maximum channel power is about 2.45 MW(th).

	Shut off Rod (1-37)		Standard clusters
N			
AR	Absorber Rod		
RR	Regulating Rod		Alternate clusters
SR	Shim Rod		
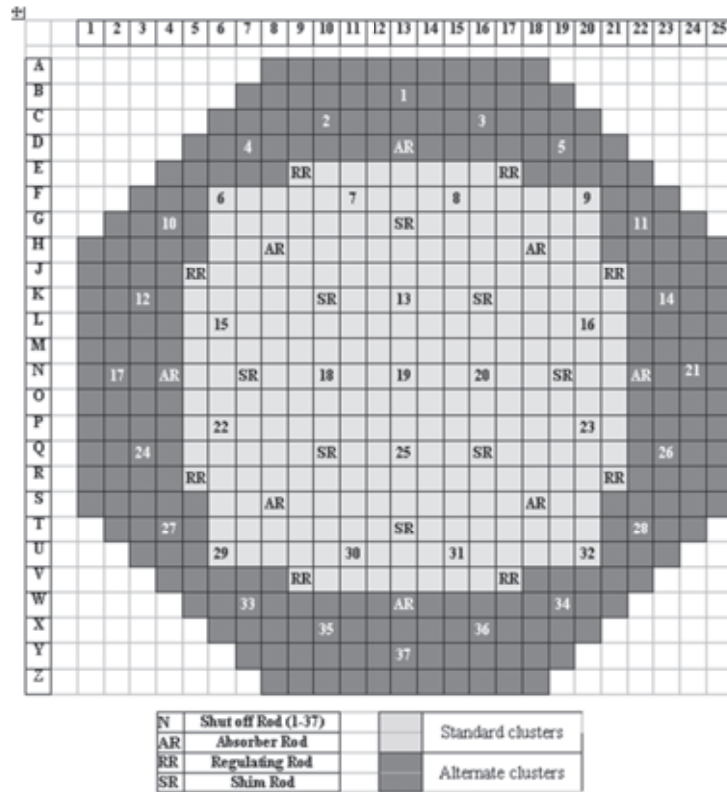
FIG. 3.2. Core loading pattern of equilibrium core of AHWR-Reference core.

TABLE 3.4.   MAIN CORE DESIGN FEATURES OF EQUILIBRIUM CORES OF AHWR-REFERENCE AND AHWR-LEU

Parameter	AHWR-Reference	AHWR-LEU
Power to the coolant, MW(th)	920	920
Total No. of channels	513	513
No. of fuel channels	452	444
Fuelling rate, annual:		
No. of channels fuelled per year	78	44
Pu, kg	173	—
^{233}U, kg	0	—
LEU (net), kg	—	1094
Peaking factors (maximum):		
Local/radial/axial/total	1.32/1.2/1.49/2.36	1.35/1.2/1.36/2.2
Average discharge burnup, MW·d/kg HM	35.5	64
Energy extracted per tonne of equivalent mined uranium, MW·d	—	7828
Average heat rating, kW/m	10.8	10.8
MCHFR at 20% over power	1.7	1.7

TABLE 3.4. MAIN CORE DESIGN FEATURES OF EQUILIBRIUM CORES OF AHWR-REFERENCE AND AHWR-LEU (cont.)

Parameter	AHWR-Reference	AHWR-LEU
Core averaged reactivity coefficients in operating range, $\Delta k/k/°C$		
Fuel temperature coefficient	-2.1×10^{-5}	-2.82×10^{-5}
Channel temperature coefficient	$+2.1 \times 10^{-5}$	-3.73×10^{-5}
Void coefficient, $\Delta k/k$ /% void	-5.9×10^{-5}	-3.09×10^{-5}
Moderator temperature coefficient	$+5.5 \times 10^{-5}$	-8.72×10^{-5}
Control and safety devices		
No. of control rods (AR/RR/SR), worth, mk	8 each 11.4/10.9/12.0	8 each 10.9/11.6/9.9
SDS-1: No. of shutoff rods	37	45
Total worth of SDS-1 (all SORs)	-70.4	-83.25
Total worth of SDS-1 when two rods of maximum worth are not available, mk	-51.9	-60.2

3.1.3. Use of LEU as an alternative fuel in the AHWR

LEU can be used as an alternative external feed in the current AHWR-LEU core design. The enrichment of LEU in the 12 pins of the innermost ring is 18% and in the 18 pins of the intermediate ring is 22%. The 24 pins in the outermost ring have axially graded enrichments of 19% and 26% in the upper and lower halves, respectively. The displacer unit has a central rod of Zircaloy-2 throughout the length. The cluster design is geometrically identical to the AHWR composite cluster.

The initial content of uranium is about 24.9 kg in a fuel assembly. With burnup, the fissile ^{235}U and the fertile ^{238}U in the (thorium, LEU) MOX deplete and the other isotopes of uranium and plutonium buildup. At the same time, the fertile thorium also gets converted into fissile ^{233}U. The power fractions from ^{235}U, ^{233}U and plutonium are shown in Fig. 3.3.

The average discharge burnup of this core is about 64 MW·d/kg HM and requires an annual feed of 44 fuel clusters. The fissile content in the uranium recovered after reprocessing the discharged fuel is about 8.0% and the fissile content in the reprocessed plutonium is about 55.13%. The isotopic vectors of uranium and plutonium for AHWR-LEU fuel are compared with those of the AHWR-Reference design in Tables 3.2 and 3.3, respectively.

3.1.4. Core design features of the AHWR with LEU

The reactor core of AHWR-LEU consists of 513 lattice locations in a square lattice pitch of 225 mm where fuel assemblies occupy 444 locations. Sixty-nine locations are reserved for the reactivity control devices and shutdown system-1 (SDS-1), of which 45 locations are used for locating shutoff rods (SORs) and 24 for locating control rods (CRs). Forty-five shutoff rods constitute the primary shutdown system, SDS-1. The core layout with the locations of the reactivity devices is shown in Fig. 3.4.

Power distribution was optimized by adopting three burnup zones 59/64/74 MW·d/kg HM. The average discharge burnup of this core is about 64 MW·d/kg HM.

The various reactivity coefficients from cold critical to hot operating condition were calculated and found to be negative for this core configuration. The void reactivity, which has safety implications, remains negative throughout the burnup range. The various reactivity coefficients and the design features of this core have been compared with those of the AHWR's self-sustaining core (AHWR-Reference) in Table 3.4.

Since all the reactivity coefficients are negative for the AHWR-LEU core configuration, the shutdown requirement becomes higher than in the AHWR-Reference case and hence the number of shutoff rods has been
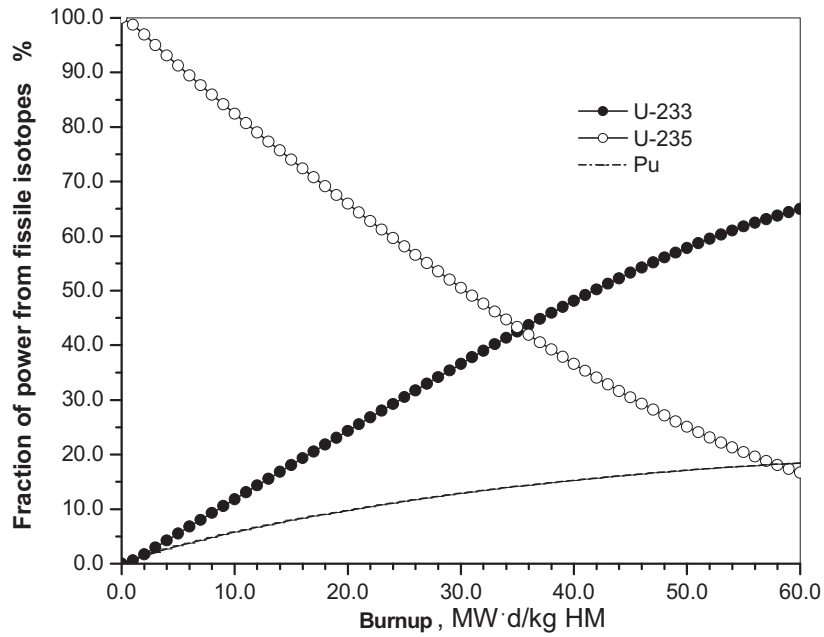
FIG. 3.3. Power fraction from ^{233}U, ^{235}U and Pu for AHWR-LEU fuel.

N	Shut off Rod (1-45)		74000 MWd.te
AR	Absorber Rod		64000 MWd/te
RR	Regulating Rod		61000 MWd/te
SR	Shim Rod		
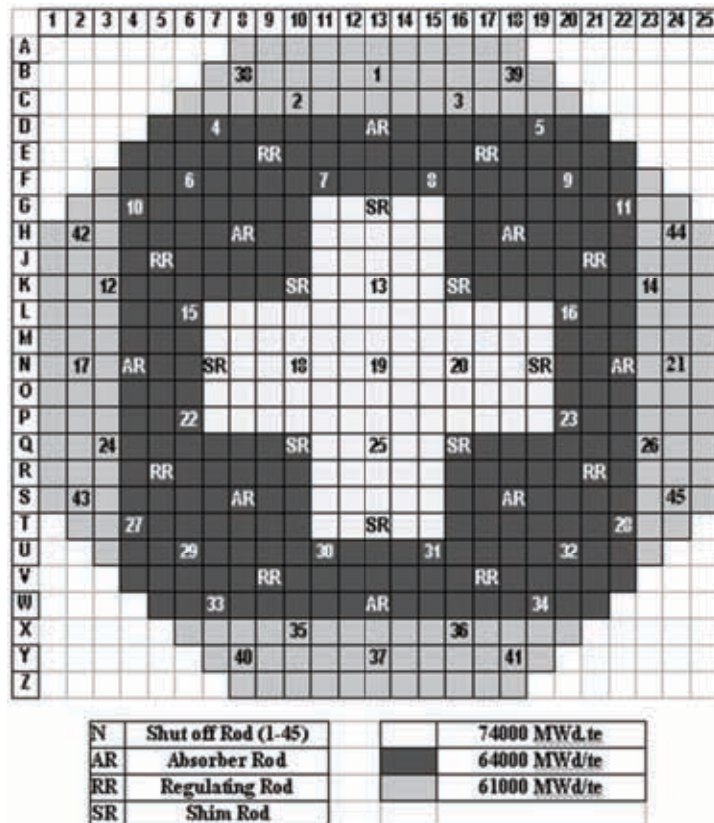
FIG. 3.4. Core loading pattern of equilibrium core of AHWR-LEU.

increased. Shutdown system-1 (SDS-1) for this core consists of 45 mechanical shutoff rods falling under gravity when actuated. The worth of SDS-1 with 45 shutoff rods (SORs) has been calculated as –83.25 mk and –60.3 mk for 43 rods (with two rods of maximum worth not available). The shutdown system also meets the shutdown requirement under operational and under accidental conditions for this core and the reactor is subcritical by more than 10 mk under all circumstances.

3.1.5. Conclusions

For the equilibrium AHWR core fuelled only with the standard clusters, the annual refuelling rate is 73 fuel assemblies, which requires about 123 kg plutonium and 163 kg of ^{233}U. Annual production of ^{233}U in this core is about 141 kg and hence the core results in an annual deficit of about 22 kg of ^{233}U.

Self-sustenance in ^{233}U is achieved by using a combination of two types of clusters (standard and alternative design) in the core, the 'AHWR reference design'. An annual feed of 39 fuel assemblies of each type is required. The annual requirement of plutonium rises to 173 kg. The 22 kg deficit of ^{233}U can be overcome by charging 50 kg extra plutonium into the core, but there is a burnup penalty of 2.5 MW·d/kg HM. The void reactivity becomes slightly more negative. The other reactivity coefficients are no different from those for the equilibrium core with standard clusters only. The worth of reactivity devices is nearly the same as in the equilibrium core with standard clusters. The shutdown system No. 1 (with two maximum worth rods not available) is able to meet the shutdown requirement under operating and accidental conditions.

The AHWR being designed for the use of thorium in a closed fuel cycle with minimum additional requirement of plutonium as fissile feed offers enough flexibility to accommodate different kinds of fuel cycles. One option is the AHWR LEU core design: LEU (UO$_2$) with an appropriate enrichment mixed with ThO$_2$ (thorium, LEU)MOX provides an alternative fuel for the AHWR, which enables high discharge burnup. For this fuel, all reactivity coefficients in the core are negative. The discharge burnup of the equilibrium core is 64 MW·d/kg HM. For this core, the energy extracted per tonne of equivalent mined uranium is about 7826 MW·d as compared to 6700 MW·d in a PHWR. The fissile content in the uranium recovered after reprocessing the discharged AHWR LEU fuel is about 8.0%, which is denaturized and contains sufficient quantity of ^{232}U to make it further proliferation resistant. Also, the discharged fuel of the AHWR LEU core contains less plutonium and less minor actinides compared to the AHWR reference core.

3.2. CANADA

3.2.1. Thorium fuel cycles for the CANDU-6 reactor

Building on previous studies [36–39], this subsection looks at several possibilities for homogeneous fuel cycle options in CANDU-6 reactors. Since thorium does not have a naturally occurring fissile isotope, an initial fissile inventory needs to be provided until enough ^{233}U can be bred from the thorium to sustain the reaction. The cases in this study are homogeneous mixtures of thorium mixed with plutonium. These studies include a low burnup of 20 MW·d/kg HM and a higher burnup of 40 MW·d/kg HM. There is significantly better thorium utilization with the higher burnup. Recycle cases for plutonium driven fuel were also investigated, where the amount of ^{233}U at the beginning and end of cycles is roughly equivalent. The current chapter addresses physics involved in lattice cell calculations of thorium fuel in CANDU-6 reactors and does not consider details of its full-core implementation.

All models for this section were lattice cell calculations performed using WIMS-AECL v.3.1.2.1 and an ENDF/B-VI data nuclear library [40, 41]. There are four different ThFC options included in this chapter:

— Low burnup (~20 MW·d/kg HM) once-through plutonium driven thorium;
— High burnup (~40 MW·d/kg HM) once-through plutonium driven thorium;
— Low burnup (~20 MW·d/kg HM) plutonium driven thorium with ^{233}U recycle;
— High burnup (~40 MW·d/kg HM) plutonium driven thorium with ^{233}U recycle.

The isotopic composition of the reactor grade driver plutonium is given in Table 3.5. For the recycle calculations, only the ^{233}U is being recycled, and not other uranium isotopes or plutonium.

The power normalization used a constant flux, which was chosen such that the average power over the total burnup averaged to around 32 W/g. Leakage and absorption by un-modelled reactor components were assumed to be worth 30 mk (dk/k = 0.30, 3%, 3000 pcm) in total, which is typical of a CANDU-6 reactor with the adjustor rods removed.

TABLE 3.5. INPUT ISOTOPIC COMPOSITION FOR PLUTONIUM DRIVER FUEL

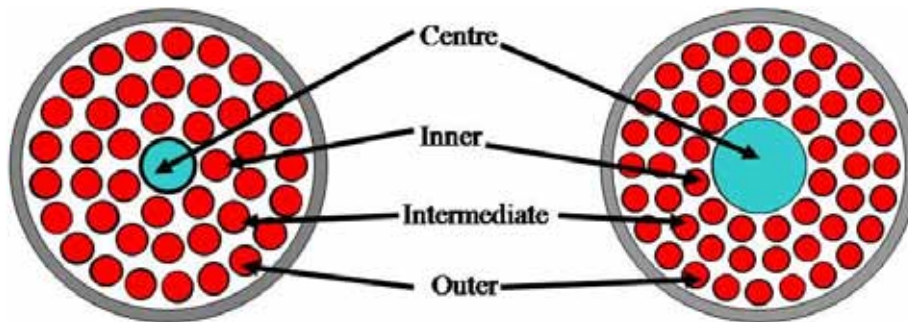| Nuclide	Mass fraction (wt% in Pu)
^{238}Pu	2.5
^{239}Pu	54.2
^{240}Pu	23.8
^{241}Pu	12.6
^{242}Pu	6.8

FIG. 3.5. Fuel bundle design for low burnup (left) and high burnup (right) cases.

To calculate the amount of energy that came from the thorium in the fuel, the reaction data from the WIMS-AECL output was extracted. The fission reaction for nuclides derived from thorium, and ^{233}U, ^{235}U and ^{232}Th was compared to that from the driver fuel, ^{239}Pu and ^{241}Pu. This reaction data, together with the power and the length of the time step, were used to calculate the amount of energy that came from thorium versus the driver fuel. It was assumed that the same amount of energy results from each fission (~200 MeV/fission), independent of which nuclide was the source of the fission. A total bundle power of 800 kW was assumed for the calculation of linear element ratings. These models were developed to maximize the amount of energy derived from the input thorium.

To calculate the fuel temperature reactivity coefficients (FTC), WIMS-AECL models were created such that the temperature of the fuel was increased and decreased by 50°C at each burnup step. The change in reactivity across this 100°C range was used to calculate the burnup-weighted average FTC.

A different fuel bundle design was used for each of the four cases studied. The bundle designs were changed to optimize for the amount of energy derived from thorium while maintaining a maximum linear element rating of 60 kW/m. Each bundle design has a centre element consisting of a tube of zirconium-filled hafnium. This configuration serves to introduce a poison to the centre of the bundle in a way that is simple and easy when creating computer models. The thickness of the hafnium layer is different in each case in order to tailor the reactivity coefficients. The composition of the centre poison can be changed later when a particular bundle design is chosen for fuel cycle development.

For the high burnup and the ^{233}U recycle cases, the fuel was graded, with more fissile content in the outer rings. This grading helps minimize the amount of poison needed in the central absorbing element, which in turn leads to better thorium utilization. However, this grading also leads to larger radial form factors. In order to decrease the linear element ratings, the size of the fuel pins was reduced and the number of fuel pins increased. This change in geometry results in around 10% less fuel in the bundle. The low burnup cases have 42 fuel elements, with 7, 14, and 21 elements in the inner, intermediate and outer rings, respectively. The high burnup cases have a larger centre pin, and 12, 18, and 24 fuel elements in the inner, intermediate, and outer rings, respectively (see Fig. 3.5). The bundle composition is given in Table 3.6.

TABLE 3.6. BUNDLE COMPOSITIONS FOR THE SIX CASES

Case	Burnup	Total No. of fuel elements	Bundle average Pu (wt%)	Bundle average ^{233}U (wt%)
Once-through	Low	42	3.5	N/A
	High	54	4.9	N/A
^{233}U recycle	Low	42	0.8	1.4
	High	54	2.1	1.4

TABLE 3.7. RESULTS FOR THE SIX HOMOGENEOUS THORIUM FUEL CYCLE CASES

Case	Burnup (MW·d/kg HM)	FTC (μk/°C)	Maximum linear element rating (kW/m)	Energy derived from Th (%)	Amount of ^{233}U at exit burnup (g/bundle)
Once-through	19.4	−3.8	56	18.9	41
	45.0	−5.0	61	29.2	50
^{233}U recycle	19.7	−7.5	49	78.1	68
	44.0	−7.3	59	65.7	68
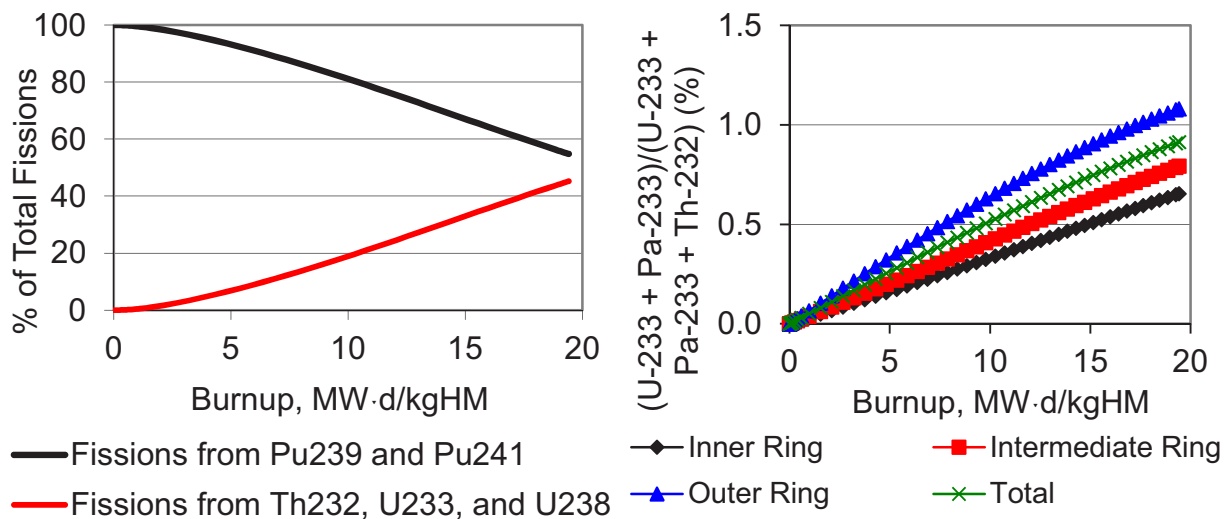
FIG. 3.6. Distribution of fissions for plutonium-driven once-through low burnup case.

FIG. 3.7. Accumulation of ^{233}U and ^{233}Pa in plutonium-driven once-through low burnup case.

The placement of the ^{233}U in the bundles in the recycle cases is not in a configuration designed to maximize the breeding of ^{233}U. The requirement for these models was to have roughly the same amount of ^{233}U input and output (or slightly more on output to allow for losses during reprocessing).

Values obtained for the burnup, fuel temperature coefficient, maximum linear element rating, and percentage of energy derived from thorium for the four cases are shown in the Table 3.7. All of the cases studied have negative fuel temperature coefficients that are comparable to those for a CANDU-6 reactor with natural uranium fuel. Figures 3.6 and 3.7 show the distribution of fissions derived from the thorium fuel vs. fissions from plutonium, and the distribution of ^{233}Pa and ^{233}U in the bundle for the low burnup once-through plutonium-driven thorium case. These graphs for the high burnup plutonium driven thorium case are in Figs 3.8 and 3.9. For the low and high burnup plutonium driven thorium with ^{233}U recycle, the graphs are found in Figs 3.10, 3.11, 3.12 and 3.13, respectively.
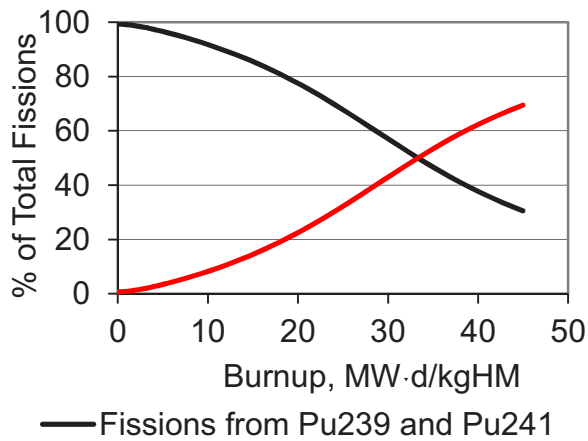
FIG. 3.8. Distribution of fissions for plutonium driven once-through high burnup case.
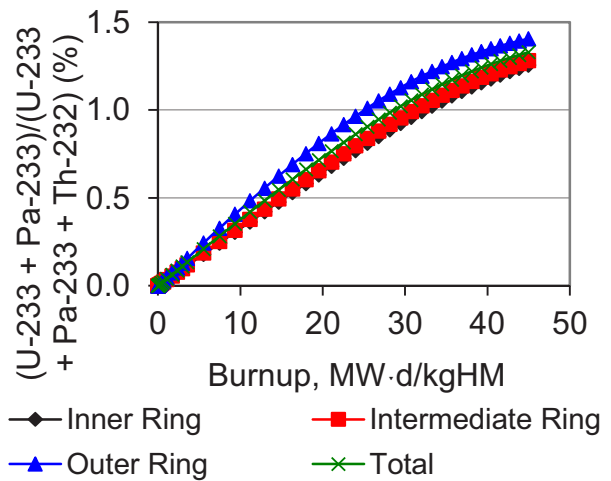

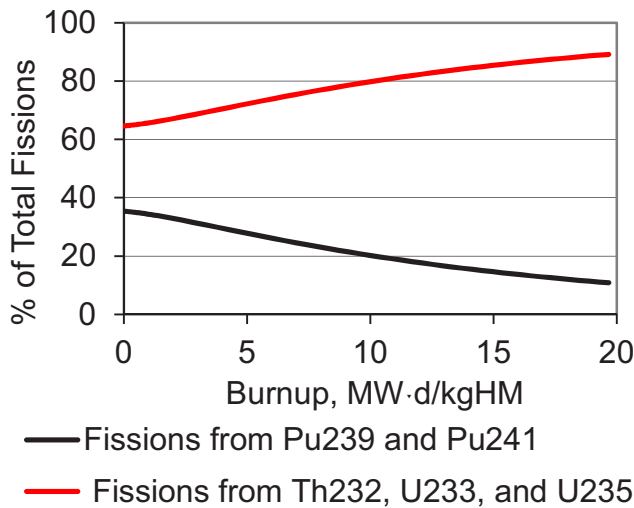*FIG. 3.9. Accumulation of ^{233}U and ^{233}Pa in plutonium driven once-through high burnup case.*

FIG. 3.10. Distribution of fissions for plutonium driven with ^{233}U recycle low burnup case.
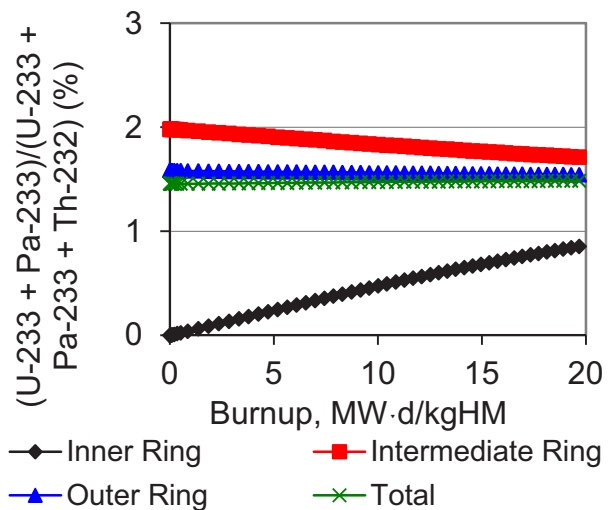
FIG. 3.11. Distribution of ^{233}U and ^{233}Pa for plutonium driven with ^{233}U recycle low burnup case.

For the once-through fuel cycles, the driver fuel is the only fissile component present at the beginning of the irradiation and thus produces all the power. As the irradiation proceeds, ^{233}U is bred in and produces an increasing fraction of the power; this effect can be seen in Figs 3.6 and 3.8. For the cases with ^{233}U recycle, the initial division of power depends primarily on the distribution of fissile material in the fresh fuel. As irradiation progresses, the amount of plutonium decreases, while the total amount of ^{233}U does not, resulting in an increasing fraction of the power being generated by ^{233}U fissions; these effects can be seen in Figs 3.10 and 3.12.

At the beginning of irradiation, in most cases the fissions originate primarily from the driver fuel. Before exit burnup, the source of the majority of fissions switches and ^{233}U becomes dominant. However, this is not the case for the once-through low burnup plutonium-driven case; plutonium remains the dominant source for the whole irradiation. The cross-over points occur at 33 MW·d/kg HM and 10 MW·d/kg HM for the high burnup plutonium-driven cases without and with recycle. The low burnup plutonium driven case with ^{233}U recycle has the majority of fissions coming from ^{233}U for the entire irradiation. This occurs because only a small amount of additional plutonium is needed to achieve the exit burnup. Thus, from the beginning of the irradiation, ^{233}U is the dominant fissile isotope. These results show that there is more benefit to using thorium in cycles with recycling of ^{233}U and/or with higher burnup.

For the once-through thorium cases, ^{233}U builds up most quickly in the outer ring, which is where the flux is highest. More neutrons are available in this region to capture onto ^{232}Th and lead to the creation of ^{233}U. For the

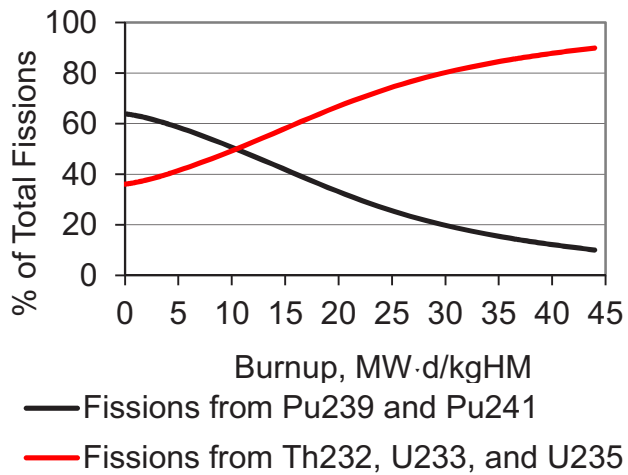FIG. 3.12. *Distribution of fissions for plutonium driven with* ^{233}*U recycle high burnup case.*
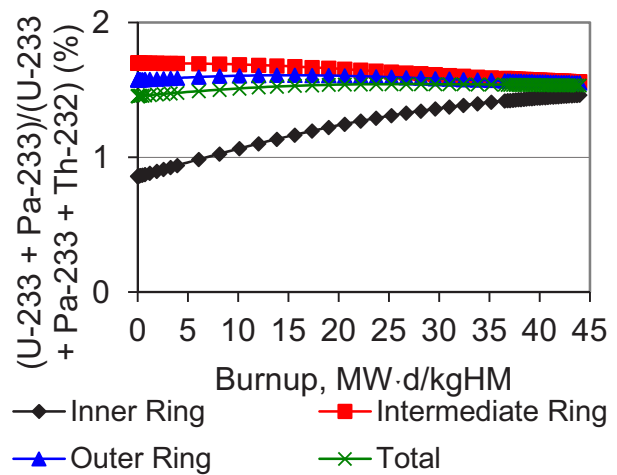
FIG. 3.13. *Distribution of* ^{233}*U and* ^{233}*Pa for plutonium driven with* ^{233}*U recycle high burnup case.*

recycle cases, the ^{233}U is initially placed in the outer and intermediate rings (Figs 3.11 and 3.13). The ^{233}U decreases from the intermediate ring, stays approximately constant in the outer ring, and grows into the inner ring. These fuel designs were chosen so that the total amount of ^{233}U in the fuel bundle stays approximately constant throughout the irradiation, i.e. the rate of ^{233}U absorption is equal to the ^{233}U production. The once-through plutonium/thorium low burnup case has a residence time of 1.7 years; therefore, the ^{233}U to supply a fresh core for one of the ^{233}U recycle cases could be generated every 2.8 years. The once-though high burnup case has a longer residence time of 4.2 years. Sufficient ^{233}U to fuel a fresh core for the ^{233}U recycle cases would be generated every five years.

The CANDU-6 reactor can efficiently exploit homogeneous thorium based fuel cycles. Fuel bundles can be designed to produce relatively high levels of burnup while providing good reactor physics operating parameters throughout their irradiation history. The low burnup plutonium-driven thorium case with ^{233}U recycle gives the best results for percentage of energy derived from thorium. For the once-through cases, the higher burnup scenarios lead to a higher percentage of energy from thorium. The percentage of energy from thorium is higher for the low burnup recycle case than the high burnup recycle case. To achieve higher burnup, more energy is proportionally required from the driver fuel in the recycle scenarios. In general, it was found that the maximum energy from thorium corresponds to a case with the minimum amount of poison in the centre pin. This explains the trend in the models, where the fissile content of the fuel is graded towards the outside of the fuel bundle, which contributes to lowering the CVR. The extent to which the fuel can be graded is constrained by the linear element ratings that the outer fuel elements can sustain.

3.2.2. Thorium fuel cycles in advanced CANDU reactors

The study [42] considers options to establish a ThFC in the advanced CANDU reactor (ACR-1000) driven by the plutonium derived from the reprocessing of the spent fuel of commercial LWR. The fissile component of this civilian grade plutonium is typically about 67%. Both the once-through cycle, which does not involve reprocessing of the spent thorium fuel, and the closed cycle, which reprocesses the spent thorium fuel and recycles the ^{233}U, are considered in this study. The option of plutonium/thorium fuel offers attractive reactor physics properties; hence, it is a practical method of introducing thorium as a fuel. It also offers an efficient way to extract the energy potential of the plutonium feedstock, together with a beneficial way to manage plutonium stockpiles. Similar to the CANDU-6 studies above, this study implements the thorium/plutonium cycles using homogeneous fuel in each fuel pin.

The unique combination of a simple fuel bundle design and on-power fuelling capability in the ACR-1000 enables easy and gradual introduction of thorium fuel bundles into the reactor [43–46]. Initially, the thorium fuel bundles will be designed to give a discharge fuel burnup of approximately 20 MW·d/kg HM and a small negative full-core CVR at the end of reactor operating lifetime. This specification leads to safety characteristics for the thorium fuelled core and for the transition core, which are within the licensing basis for the standard ACR-1000.
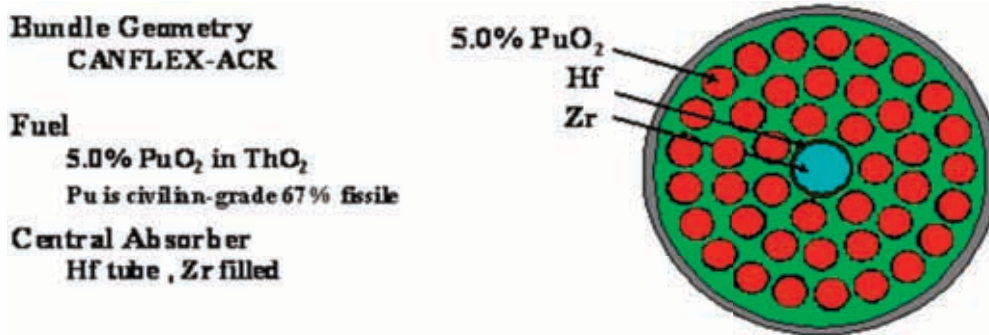
FIG. 3.14. Fuel bundle configuration for the once-through thorium fuel cycle in ACR-1000 with 21 MW·d/kg HM burnup.

FIG. 3.15. Fuel bundle configuration for the closed thorium fuel cycle in ACR-1000 with recycled ^{233}U and 21 MW·d/kg HM burnup.

Transition from a full-core LEU fuel to a full-core load of thorium fuel in the ACR-1000 is estimated to be carried out on power using the standard two bundle-shift fuelling scheme over approximately two full power years.

The once-through ThFC option that has been studied uses a uniform mixture of 5% PuO_2 and 95% natural ThO_2 in the 42 fuel elements, as shown in Fig. 3.14. Figure 3.15 shows the fuel design for the closed ThFC option giving a 21 MW·d/kg HM discharge burnup. It uses a mixture of 3.7% PuO_2 and 96.3% natural ThO_2 in the outer fuel ring (21 elements) and in the inner fuel ring (seven elements). The intermediate fuel ring, which has 14 elements, contains a mixture of 5.0% $^{233}UO_2$ and 95% natural ThO_2. The ^{233}U is obtained from the reprocessing of the spent thorium fuel. The amount of ^{233}U contained in a typical spent thorium fuel bundle is slightly more than that required for a fresh bundle. Therefore, this closed fuel cycle option is self-sufficient in ^{233}U requirement. However, it is not self-sufficient in fissile requirement and requires the addition of fresh civilian grade plutonium to each new bundle. The plutonium in the spent fuel is not recycled.

The combination of negative CVR and negative FTC ensures that the power feedback coefficient is negative for the ThFCs. The similarity of the reactivity feedback coefficients between the ThFCs and the LEU fuel cycle gives a strong assurance that implementation of the ThFCs in the ACR-1000 will not require major modifications to the basic control and safety systems, which are originally designed for the LEU fuel cycle. Furthermore, due to the similar reactivity insertion introduced by each new thorium bundle relative to an LEU bundle, the transition from the standard LEU core to the plutonium/thorium core does not change the basic core safety and control characteristics.

Table 3.8 provides the fuel compositions and the material requirements for the once-through and the closed ThFC options in the ACR-1000. All quantities are expressed in grams per bundle unless otherwise indicated. The fuelling rate is about 10.5 fresh thorium fuel bundles per full power day. This fuel cycle is then equivalent overall to the ACR-1000 LEU fuel cycle.

TABLE 3.8. FUEL COMPOSITION AND MATERIAL FLOW OF PLUTONIUM-THORIUM FUEL CYCLE OPTIONS IN ACR-1000 (20 MW·d/kg HM BURNUP)

	Once-through cycle		Closed cycle	
	Fresh fuel	Discharged fuel	Fresh fuel	Discharged fuel
^{232}Th	14 912.51	14 744.75	15 049.84	14 850.34
^{233}U	0.00	110.04	261.63	269.09
^{238}Pu	19.71	13.34	9.72	5.36
^{239}Pu	427.25	136.67	210.79	36.24
^{240}Pu	187.61	182.71	92.56	79.08
^{241}Pu	99.33	76.16	49.00	32.63
^{242}Pu	53.60	66.79	26.45	37.85
Total Pu	787.50	475.67	388.52	191.16
Total fissile Pu	526.58	212.83	259.79	68.87
Fissile Pu %	66.87	44.74	66.87	36.03
Total Pu destroyed		311.83		197.36
% of initial Pu destroyed		59.22		50.80
Civilian Pu required per year (Mg)	3.02		1.49	
Th required per year (Mg)	57.19		57.72	

Note: All quantities are expressed in g/bundle unless otherwise indicated. 10.5 bundles are required per full power day operation in ACR-1000 (for 20 MW·d/kg HM burnup).

The low fissile content removes the proliferation potential of the plutonium remaining in the spent thorium fuel. Each ACR-1000 reactor consumes about 3 t of the civilian grade plutonium per year and creates approximately 0.4 t of ^{233}U per year. Operation of the once-through ThFC for about five years will accumulate enough ^{233}U for the manufacturing of a full-core load of thorium bundles for the closed cycle option. The transition from the once-through ThFC to the closed ThFC can also be implemented using on-power fuelling. The thorium fuel bundles for both options are designed to be almost interchangeable. Consequently, no significant difficulties are anticipated during the transition period.

For a closed thorium based fuel cycle, two options have now been studied based on the reference fuel burnup of 20 MW·d/kg HM and the longer term target burnup of 40 MW·d/kg HM. In practice, according to fuelling optimization studies, any burnup between 20 and 40 MW·d/kg HM can be selected. The results below show that, on the basis of fundamental core design characteristics, the selected fuel delivering 40 MW·d/kg HM falls within the required parameter values for ACR-1000. Figure 3.16 shows the main fuel bundle parameters for the 40 MW·d/kg HM plutonium/ThFC option in ACR-1000.

Table 3.9 shows the main core characteristics based on both lattice calculations using WIMS-AECL, and 3-D full core calculations using the RFSP code [47] with fuel properties generated by WIMS-AECL. It is notable that for the range of options, including the open cycle and both closed cycles, peak fuel channel and fuel bundle powers lie within the licensing basis for ACR-1000, and core void reactivity is predicted by the 3-D model to be very close to the original design target for a small negative value. This gives confidence that as additional, more detailed design studies proceed, the core characteristics will continue to meet design expectations.

Table 3.8 shows that each fresh thorium fuel bundle for the closed cycle option contains about 262 g of ^{233}U, which is slightly less than the amount of ^{233}U contained in a typical spent fuel bundle. Therefore, this closed fuel cycle option is self-sufficient in ^{233}U once started. It requires no new source of ^{233}U, but will require additional

Bundle Geometry
    CANFLEX-ACR

Fuel
    6.5 % PuO$_2$ (Rings 4 & 2)
    5.0 % ^{233}UO$_2$ (Ring 3)
    Pu is civilian-grade, 67% fissile

Central Absorber
    Hf tube, Zr filled

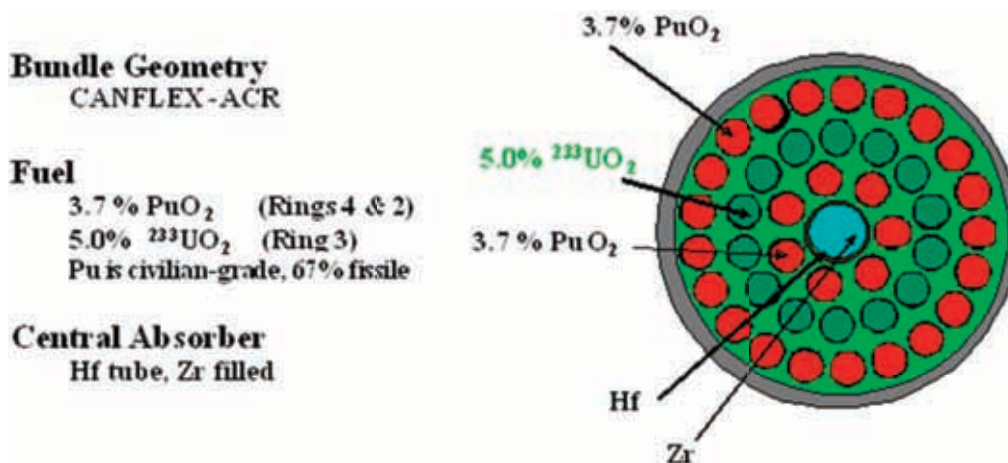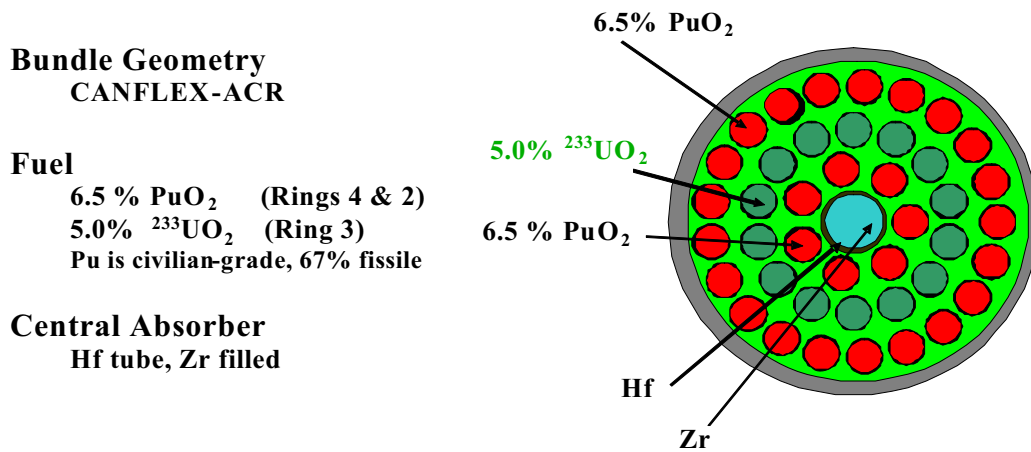*FIG. 3.16. Fuel bundle configuration for closed thorium fuel cycle in ACR-1000 with recycled ^{233}U and 40 MW·d/kg HM burnup.*

TABLE 3.9. FUEL COMPOSITION AND MATERIAL FLOW OF PLUTONIUM-THORIUM FUEL CYCLE OPTIONS IN ACR-1000

	2-D WIMS-AECL lattice calculations		3-D RFSP full-core calculations		
	Burnup MW·d/kg HM	FTC μk/°C	Burnup MW·d/kg HM	Max. channel power MW	Max. bundle power MW	Max. elem. rating kW/m
Once-through uniform, 5% Pu, 95% Th	21	–12	20.4	6.8	767	53
Closed cycle, 3.7% Pu, 5% ^{233}U, 3.7% Pu	21	–13	21.3	6.8	740	48
Closed cycle, 6.5% Pu, 5% ^{233}U, 6.5% Pu	40	–14	40.3	6.7	805	55

fissile material, i.e. plutonium, in order to achieve the desired fuel burnup target. The amount of plutonium requirement is about 2.5% on a bundle averaged basis, which is half the amount required in the once-through option. This suggests that a large fraction of the total energy generated in the fuel bundle of the closed cycle option must come from fissions derived from ^{233}U.

Figure 3.17 shows the fraction of the fissions derived from fissile plutonium and from ^{233}U over the anticipated fuel burnup range of 20 MW·d/kg HM for the closed cycle option. The plutonium fraction decreases from about 70% at the beginning to about 35% at discharge burnup because of the progressive depletion of the fissile plutonium isotopes. However, the ^{233}U fraction increases from 30% at the beginning to about 65% at discharge burnup because the depletion of ^{233}U due to fission is effectively compensated by the creation of additional ^{233}U by neutron absorption in ^{232}Th. It is estimated that about 60% of the total energy produced in this closed fuel cycle option is derived from thorium.

Further optimization of the fuel design and reactor core fuel management scheme can significantly improve the fraction of thorium derived energy beyond 60%. For example, as higher fuel burnups are demonstrated, further efficiencies in thorium utilization can be achieved. It is estimated that up to 70% of the energy could be derived from thorium by extending the discharge fuel burnup to 40 MW·d/kg HM and beyond. The data in Table 3.8 show that the fissile component of the plutonium isotopes decreases from 67% in the fresh fuel to about 36% in the spent fuel. Hence, this closed ThFC is also a good choice for reducing the proliferation potential of the stockpiled plutonium.
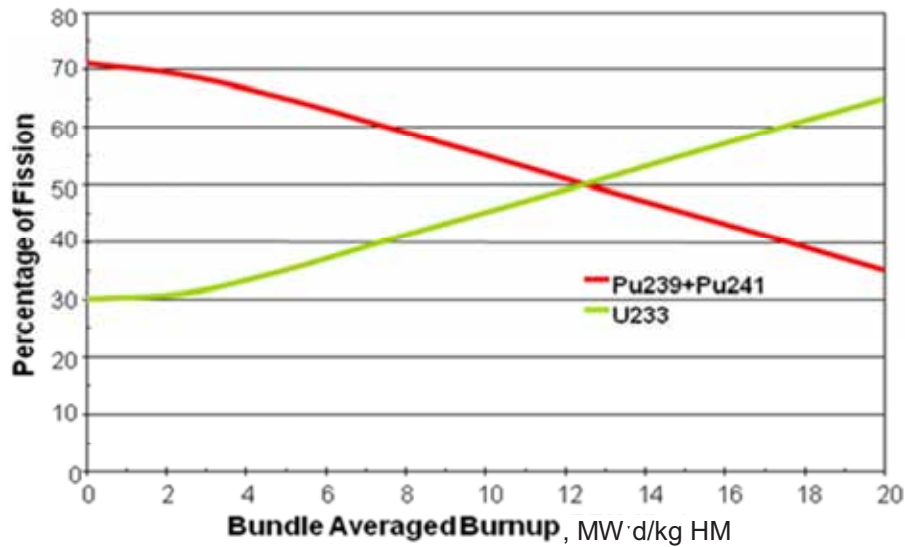
FIG. 3.17. Relative fission from plutonium and ^{233}U for closed cycle option in ACR-1000.

TABLE 3.10.   BASIC DESIGN PARAMETERS OF 900MW(e) PWR

Core design parameters		Fuel assembly data		Fuel rod data	
Installed capacity thermal, MW(th)	2775	Array	17 X 17	Pellet diameter, mm	8.05
Installed capacity electric, MW(e)	900	No. of rods	264 fuel rods, 24 GT and 1 IT	Active fuel length, cm	367.30
System pressure, bars	150	I.D. of GT/IT, mm	11.42	I.D. of clad	8.236
Core average coolant temperature, °C	309.9	O.D. of GT/IT, mm	12.26	O.D. of clad	9.518
Inlet coolant temperature, °C	291.7	Material of GT/IT	Zircaloy-4	Material of clad	Zircaloy-4
Average coolant heating, °C	35.7	Fuel rod pitch at hot state, cm	1.266		
No. of fuel assemblies	157				
Assembly pitch at hot state, cm	21.607				

*   GT — guide tube; IT — instrumentation tube.

3.3.   REPUBLIC OF KOREA

The subject of a national study was the investigation of isotopic composition characteristics of a thorium fuel loaded core of a currently existing typical 900 MW(e) PWR. It was assumed that original basic design parameters of PWR operating uranium fuel (see Table 3.10) should be kept constant, at least at the first stage of thorium fuel introduction.

The chart of fuel assembly has been kept unchanged also, as shown in Fig. 3.18.
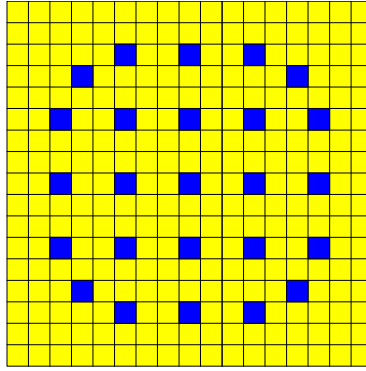
FIG. 3.18. Fuel assembly cross-section chart.

TABLE 3.11. ISOTOPIC COMPOSITIONS OF PLUTONIUM

	Fractions of isotopes, %				
	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
Reactor grade plutonium (rgPu)	1.8	59.0	23.0	12.2	4.0
Weapon grade plutonium (wgPu)	0.0	94.0	6.0	0.0	0.0

Three types of thorium based fuel were considered. Thorium fuel form was assumed as a homogenized mixture of ThO_2 and UO_2 (or PuO_2) powders sintered providing a percentage of theoretical density customary for uranium fuel.

Fresh fuel composition of the first type ($ThO_2 + UO_2$) consisted of a mixture of 76% ThO_2 and 24% UO_2, and uranium consisted of 80% ^{238}U and 20% ^{235}U. This composition corresponds to equilibrium conditions for reactor reloading. This first fuel type can be applied either in combination with consecutive spent fuel reprocessing or in a once-through mode.

The second thorium fuel option involves the use of reactor grade plutonium ($ThO_2+rgPuO_2$ fuel). The fuel matrix is composed of thorium (91% of total heavy metal mass), plutonium (7.5% of total heavy metal mass) and minor amounts of other isotopes, mainly ^{238}U. The share of fissile isotopes of plutonium reached 5.34% of heavy metal mass.

The use of weapon grade plutonium has been envisaged in a third fuel option ($ThO_2 + wgPuO_2$). It contained 94% of thorium, 5% of plutonium and admixtures of other isotopes. The share of fissile isotopes in plutonium reached 94%, i.e. 4.7% of total heavy metal mass; plutonium vectors associated with the two last options are shown in Table 3.11.

Complementary requirements and limitations were applied to the fuel management scheme as follows. In equilibrium conditions, every cycle of the reactor should be equal or exceed one year. The number of fuel assemblies to be reloaded annually should be equal or lower than 64. The concentration of burnable absorber (Gd_2O_3) in burnable poison rods should be equal to 4%, and the number of these rods in an assembly could be 0, 4 or 8 rods per assembly.

The reactor core pattern and fuel reshuffling schemes were optimized to satisfy actual requirements for uranium (MOX) fuel except for the burnup level permitted, assuming conservatively that demands on reactivity coefficients, peaking factors, control rods efficiency, etc., remain the same at least until new operation data are compiled and possible justifications and conclusions become available. The fuel loading pattern common for $ThO_2 + UO_2$ and $ThO_2 + rgPuO_2$ is presented in Fig. 3.19, and general characteristics of the reactor core are presented in Table 3.12.

The equilibrium conditions for each fuel type considered were attained after three cores fully loaded with thorium fuel. Due attention was paid to the possible neutron economy, and efforts were made to apply a low leakage loading pattern. Reactor lifetime was assumed to be 60 years in every fuel option case.

2 | 1 | F-8Gd | 2 | 2 | F-4Gd | 1 | F
1 | 1 | 1 | 2 | F-8Gd | 2 | F | 1
F-8Gd | 1 | 1 | F-8Gd | 1 | F-4Gd | F
2 | 2 | F-8Gd | 2 | 1 | F | 1
2 | F-8Gd | 1 | 1 | F | 1
F-4Gd | 2 | F-4Gd | F | 1
1 | F | F | 1
F | 1

1 — Once Burnt Fuel

2 — Twice Burnt Fuel

F — Fresh Fuel

F-4Gd — Fresh Fuel with 4Gd

F-8Gd — Fresh Fuel with 8Gd
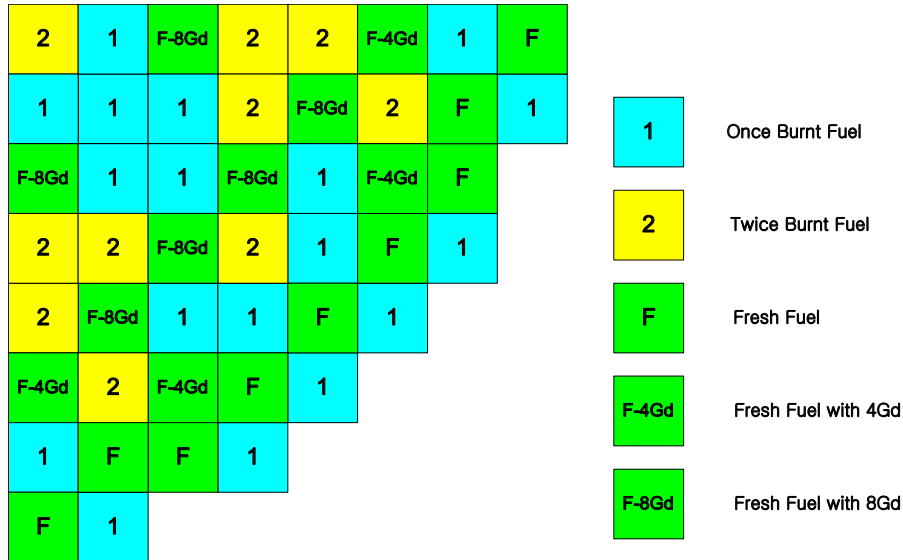
FIG. 3.19. Pattern of equilibrium core of 900 MW(e) PWR loaded with $ThO_2 + UO_2$ or $ThO_2 + rgPuO_2$ fuel assemblies.

TABLE 3.12. CORE CHARACTERISTICS

Core characteristics	Fuel type		
	$ThO_2 + UO_2$	$ThO_2 + rgPuO_2$	$ThO_2 + wgPuO_2$
No. of feed FA total	64	64	64
No. of feed FA with 0 rods containing Gd	32	32	36
No. of feed FA with 4 rods containing Gd	12	12	0
No. of feed FA with 8 rods containing Gd	20	20	28
Content of U/Pu in fresh fuel, %HM	23 (U)	7.5 (Pu)	5 (Pu)
Fissile isotopes content in fresh fuel, %HM	4.6 (^{235}U 20%)	5.34 (odd Pu)	4.7 (odd Pu)
Equilibrium cycle length, EFPD	378	401	361
Average burnup, MW·d/kg HM	38.2	40.5	36.4
Average fuel residence time, EFPD	927.3	983.7	885.6
Mass of heavy metal in the core, tHM	67.05	67.01	67.03

For the purpose of analysing the fuel composition of the thorium loaded core, the HELIOS/MASTER code system was applied. The HELIOS 1.4 code is based on a 34 energy group's ENDF/B-VI neutron cross-section library and was used for the generation of group constants. The nodal core simulator MASTER was used for burnup imitation. MASTER is the code developed by KAERI that allows performing 2-D or 3-D calculations of isotopic composition, and includes extended nuclide decay chains to treat ^{232}Th and its neighbour nuclides (e.g. ^{233}Pa, ^{233}U, ^{234}U). Fuel isotope inventory change between loading and discharging is represented in Tables 3.13 and 3.14.

TABLE 3.13. FUEL ISOTOPE INVENTORY — RELOADING FEED FUEL COMPOSITION

Parameter or isotope	ThO$_2$ + UO$_2$		ThO$_2$ + rgPuO$_2$		ThO$_2$ + wgPuO$_2$	
	mass, tHM	fraction	mass, tHM	fraction	mass, tHM	fraction
Reloading mass	27.65	1	27.64	1	27.64	1
^{232}Th	21.0264	0.7604	25.2591	0.9138	25.9142	0.9377
^{235}U	1.2654	0.0458	0.0059	0.0002	0.0063	0.0002
^{238}U	5.3599	0.1938	0.3215	0.0116	0.3463	0.0125
^{238}Pu	0.0000	0.0000	0.0368	0.0013	0.0000	0.0000
^{239}Pu	0.0000	0.0000	1.2101	0.0438	1.2867	0.0466
^{240}Pu	0.0000	0.0000	0.4737	0.0171	0.70822	0.0030
^{241}Pu	0.0000	0.0000	0.2523	0.0091	0.0000	0.0000
^{242}Pu	0.0000	0.0000	0.0831	0.0030	0.0000	0.0000

TABLE 3.14. FUEL ISOTOPE INVENTORY — UNLOADED SPENT FUEL COMPOSITION

Parameter or isotope	ThO$_2$ + UO$_2$		ThO$_2$ + rgPuO$_2$		ThO$_2$ + wgPuO$_2$	
	mass, t HM	fraction	mass, t HM	fraction	mass, t HM	fraction
Spent fuel mass	26.60	1	26.54	1	26.65	1
^{232}Th	20.4309	0.7681	24.6683	0.9296	25.3201	0.9500
^{233}U	0.3436	0.0129	0.3940	0.0148	0.3888	0.0146
^{234}U	0.0354	0.0013	0.0316	0.0012	0.0332	0.0012
^{235}U	0.4025	0.0151	0.0091	0.0003	0.0093	0.0003
^{236}U	0.1537	0.0058	0.0010	0.0000	0.0011	0.0000
^{238}U	5.1206	0.1925	0.3127	0.0118	0.3377	0.0127
^{238}Pu	0.0036	0.0001	0.0320	0.0012	0.0031	0.0001
^{239}Pu	0.0554	0.0021	0.2860	0.0108	0.2190	0.0082
^{240}Pu	0.0184	0.0007	0.3564	0.0134	0.1853	0.0070
^{241}Pu	0.0154	0.0006	0.2400	0.0090	0.1096	0.0041
^{242}Pu	0.0054	0.0002	0.1391	0.0052	0.0316	0.0012
^{241}Am	0.0004	0.0000	0.0185	0.0007	0.0051	0.0002
^{242m}Am	0.0000	0.0000	0.0005	0.0000	0.0001	0.0000
^{243}Am	0.0011	0.0000	0.0276	0.0010	0.0061	0.0002
^{242}Cm	0.0002	0.0000	0.0050	0.0002	0.0016	0.0001
^{244}Cm	0.0003	0.0000	0.0144	0.0005	0.0020	0.0001
^{237}Np	0.0106	0.0004	0.0001	0.0000	0.0001	0.0000

Several scenarios constructed with these data on reactor and material flows are considered in Section 4.

The introduction of thorium fuel in a core causes a drop of burnup stipulated by the need to keep reactor criticality at the defined power level. To keep the fuel cycle length unchanged, the fissile inventory in the core should be raised, e.g. by increasing enrichment of uranium fuel up to 20%. In this case, uranium spent fuel will contain a significant fissile inventory after discharge that can be recycled in a closed fuel cycle. In a once-through fuel cycle, the fissile content of spent fuel will be lost and these losses are commensurate with the gains provided by the fertile material used, i.e. thorium, so the task of optimization remains. The optimization of thorium–uranium shares to improve competitiveness of thorium options could be the subject of further studies to be performed.

3.4. EUROPEAN COMMISSION

ThFCs have been investigated with varying intensity for many different reactor types in the past [4, 49–53]. This was motivated by the vast abundance of thorium and its ability to be used as a fertile material in most reactor types. Thorium oxide fuel possesses favourable neutronic, thermal and chemical properties that could enable higher fuel utilization, lower minor actinide production, and improved proliferation resistance.

In LWRs, ThFCs have better neutronic characteristics than the conventional uranium–plutonium fuel cycles, which could enable higher fuel utilization in combination with spent fuel recycling or high burnup operations.

In the European Union, the use of ThFCs has been studied in France [54] and Germany [55]. In France, various studies on ThFCs have been performed by the Commissariat à l'Énergie Atomique (CEA) and Electricité de France (EDF) since 1969. ThFCs in a PWR have been assessed in the framework of waste reduction, resource saving, and plutonium burning. In Germany, most of the research on ThFCs was related to high temperature reactors. The use of ThFCs in a PWR was assessed in German–Brazilian cooperation between 1979 and 1988. Both the French and German studies on ThFC in a PWR aimed at the use of thorium fuel without significant modification of the considered PWR core and assembly designs. The studies concluded that ThFC are, in general, comparable to the conventional uranium–plutonium fuel cycle from the standpoint of overall technical feasibility in current commercial PWRs. In particular, the use of ThFC for waste reduction and plutonium recycle in an extended burnup once-through fuel cycle was found to be promising.

The use of thorium fuel in a current PWR with a once-through fuel cycle remains an attractive option due to potential advantages, such high conversion ratio in connection with extended burnup and low initial fissile material inventory. Furthermore, ThFCs could have the potential to improve waste generation issues, operational safety, economics, and proliferation resistance. The practical implementation of ThFCs requires significant research and development to meet the challenges needed to achieve high burnup for efficient fuel use, economic fuel fabrication and reprocessing, core design and fuel cycle optimization for improved safety and economics.

The following sections present results of neutronic assessments of the use of thorium fuel in current PWR without modification of core and assembly characteristics performed within the current INPRO ThFC study. The assessments are based on a French 3400 MW(th) PWR assuming a once-through fuel cycle for the initial core. Four thorium fuel types were considered and compared with the standard UO_2 fuel. Following current practice, the fissile enrichment of the considered fuel was set to 5 wt%.

The results of these analyses show that three of the considered thorium fuels have sufficient reactivity to reach a burnup level of 45 MW·d/kg HM. Furthermore, the thorium fuel with [233]U as a major fissile component could operate beyond the burnup level of 45 MW·d/kg HM. With thorium as a main fertile component, the minor actinide generation is generally lower. The total plutonium production in thorium fuel is lower than in the standard oxide UO_2 fuel.

3.4.1. Pressurized water reactor assembly model

The French 3400 MW(th) PWR core design contains 192 assemblies. The fuel pins are arranged in 17×17 square lattices. Further characteristics of the assemblies are given in Table 3.15. The assembly model contains 27 (empty) fuel pins filled with moderator fluid. The use of burnable poison for reactivity control is not considered in this assessment. The cladding material is Zircaloy. Uniform fuel enrichment has been assumed.

TABLE 3.15. CHARACTERISTICS OF CONSIDERED PWR SUBASSEMBLY

Parameter	Dimension	Value
Active height	cm	365.8
SA pitch size	cm	21.5
Fuel rod pitch size	cm	1.26
Pellet radius	cm	0.41
Clad inner radius	cm	0.418
Clad outer radius	cm	0.475
Pins per assembly	—	262

3.4.2. Fuel types

The purpose of the current assessment has been to analyse the use of thorium fuel in current PWRs without a need for core and subassembly design modification. Therefore, following current practice, the fissile enrichment of the studied fuel types is set to about 5 wt%. This enrichment level limits the length of the refuelling cycle and the discharge burnup level.

Four thorium fuel types were considered:

— Thorium with low enriched uranium (enrichment < 20% ^{235}U) (thorium–LEU fuel);
— Thorium with reactor grade plutonium (thorium–plutonium fuel);
— Thorium with ^{233}U (Th-^{233}U fuel);
— Thorium with ^{235}U (Th-^{235}U fuel).

The performance of thorium fuel is compared to standard UO_2 fuel. Assuming a once-through cycle, the neutronic performance of the fuel for average burnup up to 45 MW·d/kg HM is analysed.

3.4.3. Pressurized water reactor neutronic characteristics with thorium fuel

In the Monte Carlo neutronic analysis, a 3-D geometry model for the assembly has been set up. The model reproduces the physical conditions in the axial direction while in the radial direction reflecting boundary conditions are assumed. Further, core average thermal-hydraulics conditions have been assumed.

Continuous nuclear cross-section data were used based on the Joint Evaluated Fission and Fusion File (JEFF3.1). All neutronic analyses were performed at steady state for hot full power conditions. For the burnup analyses, the core is assumed to deplete from initial fresh core up to a burnup of 45 MW·d/kg HM without refuelling considerations. In the burnup calculations, a burnup interval of 120 effective full power days (EFPDs) was defined.

Table 3.16 presents calculated neutronic parameters of the cores with the different thorium fuels and the standard UO_2 fuel. At the same level of fissile enrichment of 5 wt% in all fuels, the excess reactivity at the beginning of life (BOL) is the highest for the Th-^{233}U fuel, reflecting the ability of this fuel type for extended burnup operations. The UO_2 fuel achieves the next highest excess reactivity at BOL. At the assumed enrichment level, the excess reactivity of thorium-plutonium fuel core is not sufficient to achieve a realistic cycle length. It is well known that for this type of fuel, a higher enrichment level, in the range of 15 to 20 wt%, is required to achieve realistic cycle length.

The effective delayed neutron fraction (eff. β) in the thorium-LEU, Th-^{235}U and UO_2 fuels is at the same level due to the same major fissile nuclide, ^{235}U. The effective delayed neutron fraction of a fuel is determined by its fissile nuclides. The effective delayed neutron fraction of the Th-^{233}U and thorium-plutonium fuels is significantly lower than that of the other fuels. This is because the delayed neutron fraction of ^{233}U and ^{239}Pu is half the value

TABLE 3.16. NEUTRONIC CHARACTERISTICS OF PWR CORES WITH DIFFERENT FUELS

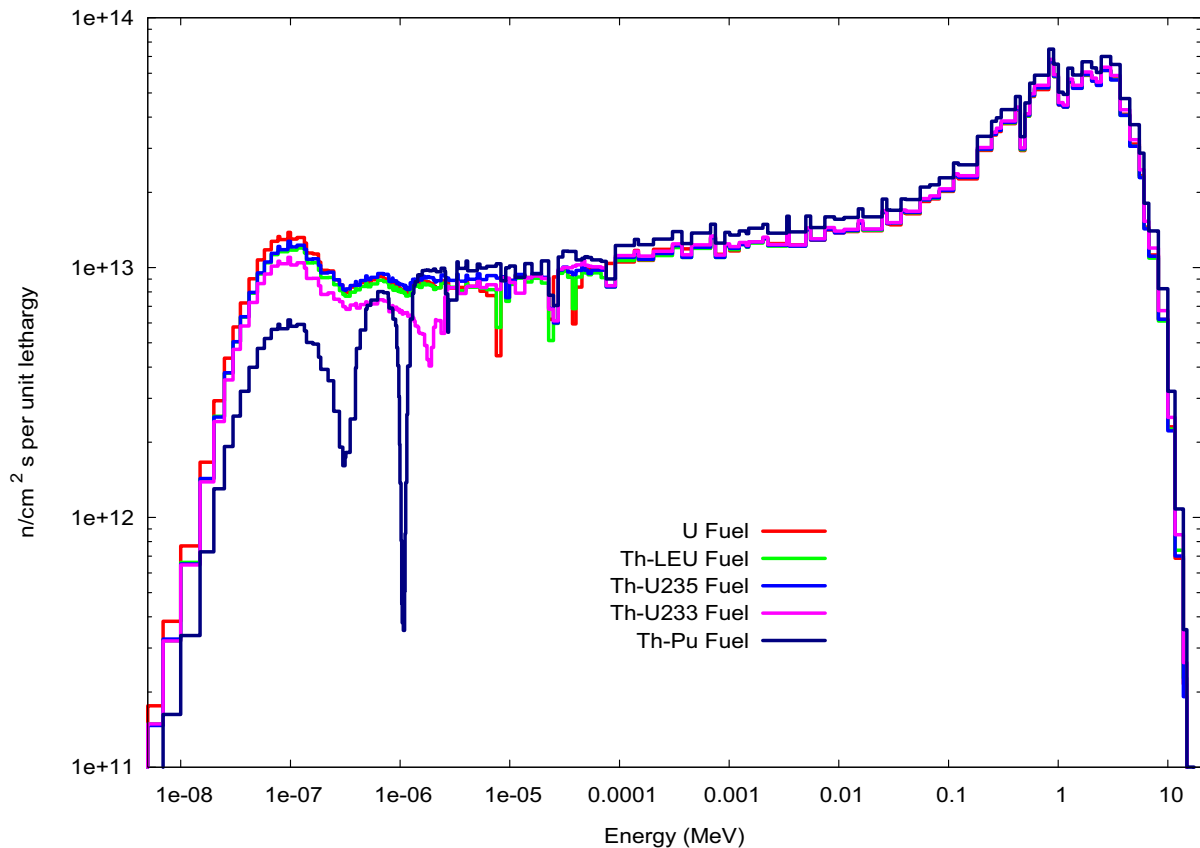| Fuel types	Th-LEU	Th-Pu	Th-U233	Th-U235	UO$_2$
Reactivity (pcm)	21 160	13 638	35 460	24 706	30 317
Eff. β (pcm)	711	259	384	676	774
ν	2.44	2.88	2.49	2.44	2.46
η	1.45	1.41	1.61	1.39	1.38
Flux (n/cm^{2}s)	2.518×10^{14}	2.965×10^{14}	2.074×10^{14}	2.438×10^{14}	2.255×10^{14}

FIG. 3.20. Subassembly average neutron energy spectrum in PWR cores with different fuel types.

compared to ^{235}U. As the burnup of the cores proceeds and the composition of the fuel changes, the effective delayed neutron fraction decreases due to the generation of the fissile nuclides ^{233}U and ^{239}Pu from the fertile nuclide ^{232}Th and ^{238}U, respectively.

The average number of neutrons per fission (ν) and the average number of neutrons per absorption (η) are at a similar level in both the thorium and the standard oxide fuel.

The neutron spectrum in the fresh core with different fuel, calculated in 172 energy group structure, is shown in Fig. 3.20. The neutron spectrum in a core with thorium–plutonium fuel is significantly harder than in the other cores. The shift to a fast spectrum is mainly due to the higher absorption cross section of plutonium in the thermal energy range. The neutron spectrum in the core with Th-^{233}U fuel is softer than in the other cores.

The mass balance of heavy metals (HM) in the assembly with the different fuel types is given in Tables 3.17–3.21 for BOL, 17 MW·d/kg HM (450 EFPD) and 45 MW·d/kg HM (1200 EFPD). The HW loading of the cores with different fuels is at a level of 26.4 kg/MW. The fissile material consumption of the cores is about 0.8 g/MW·d. The ^{233}U production of the cores with thorium–LEU, thorium–plutonium and Th–^{235}U fuels is

TABLE 3.17. THORIUM–LEU FUEL HM INVENTORY IN AN AVERAGE ASSEMBLY

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{232}Th	346400	74.19	342 400	74.6946	4000	334900	75.4109	11500
^{233}Th	0	—	0.2005	4.4E-05	−0.2005	0.2354	5.3E-05	−0.2354
^{232}Pa	0	—	0.03709	8.1E-06	−0.03709	0.08321	1.8E-05	−0.08321
^{233}Pa	0	—	336.8	0.07347	−336.8	395.4	0.08903	−395.4
^{233}U	0	—	2672	0.58289	−2672	5235	1.17878	−5235
^{235}U	23510	5.035	14 900	3.25044	8610	6125	1.37919	17 385
^{236}U	0	—	1582	0.34511	−1582	2975	0.66989	−2975
^{237}U	0	—	2.749	0.0006	−2.749	5.025	0.00113	−5.025
^{238}U	97 040	20.78	95350	20.8006	1690	92 220	20.7655	4820
^{237}Np	0	—	63.1	0.01376	−63.1	271.4	0.06111	−271.4
^{238}Pu	0	—	6.831	0.00149	−6.831	83.42	0.01878	−83.42
^{239}Pu	0	—	888.1	0.19374	−888.1	1151	0.25917	−1151
^{240}Pu	0	—	130.1	0.02838	−130.1	308.5	0.06946	−308.5
^{241}Pu	0	—	81.19	0.01771	−81.19	307.2	0.06917	−307.2
^{242}Pu	0	—	7.848	0.00171	−7.848	108.9	0.02452	−108.9
Total	466 900	—	458 400	—	8500	444 100	—	22 800

TABLE 3.18. THORIUM–PLUTONIUM FUEL HM INVENTORY IN AN AVERAGE ASSEMBLY

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{232}Th	441100	94.49	436 400	95.18	4700	427 000	96.1062	14 100
^{233}Th	0	—	0.2421	5.2E-05	−0.2421	0.3017	6.8E-05	−0.3017
^{232}Pa	0	—	0.0648	1.4E-05	−0.06481	0.1461	3.3E-05	−0.1461
^{233}Pa	0	—	404	0.08813	−404	504.9	0.11364	−504.9
^{233}U	0	—	3262	0.71145	−3262	6608	1.48728	−6608
^{235}U	0	—	26.73	0.00583	−26.73	249.8	0.05622	−249.8
^{236}U	0	—	1.35	0.00029	−1.35	22.33	0.00502	−22.33
^{237}U	0	—	0.0034	7.3E-07	−0.00337	0.05711	1.3E-05	−0.05711
^{238}U	0	—	0.0037	8.0E-07	−0.00367	0.02989	6.7E-06	−0.02989
^{237}Np	0	—	0.1845	4.0E-05	−0.1845	1.792	0.00040	−1.792
^{238}Pu	256.4	0.05	223.7	0.04879	32.7	262.9	0.05917	−6.5

TABLE 3.18. THORIUM–PLUTONIUM FUEL HM INVENTORY IN AN AVERAGE ASSEMBLY (cont.)

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{239}Pu	15130	3.24	7112	1.55114	8018	986	0.22192	14144
^{240}Pu	5897	1.26	5624	1.22660	273	2890	0.65046	3007
^{241}Pu	3077	0.66	3304	0.72061	−227	2199	0.49494	878
^{242}Pu	1282	0.27	1599	0.34875	−317	2322	0.52262	−1040
^{243}Pu	0	—	0.2646	5.7E-05	−0.2646	0.3679	8.3E-05	−0.3679
^{244}Pu	0	—	0.0365	7.9E-06	−0.03646	0.1503	3.4E-05	−0.1503
^{241}Am	0	—	127.9	0.02789	−127.9	126.6	0.02849	−126.6
^{242m}Am	0	—	1.475	0.00032	−1.475	1.327	0.00029	−1.327
^{243}Am	0	—	256.4	0.05592	−256.4	535	0.12041	−535
^{244}Am	0	—	0.1903	4.1E-05	−0.1903	0.4193	9.4E-05	−0.4193
^{242}Cm	0	—	28.04	0.00617	−28.04	70.81	0.01593	−70.81
^{243}Cm	0	—	0.4082	8.9E-05	−0.4082	2.627	0.00059	−2.627
^{244}Cm	0	—	92.79	0.02024	−92.79	485.9	0.10936	−485.9
^{245}Cm	0	—	5.076	0.00110	−5.076	45.02	0.01013	−45.02
^{246}Cm	0	—	0.1791	3.9E-05	−0.1791	7.654	0.00172	−7.654
Total	466 800	99.99	458 500	99.993	8300	444 300	100.005	22500

TABLE 3.19. Th-^{233}U FUEL HM INVENTORY IN AVERAGE ASSEMBLY

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{229}Th	0	—	0.0919	2.0E-05	−0.0919	0.1617	3.6E-05	−0.1617
^{230}Th	0	—	0.1942	4.2E-05	−0.1942	0.5148	0.00012	−0.5148
^{232}Th	442 400	94.81	438 400	95.846	4000	429 900	97.1086	12 500
^{233}Th	0	—	0.208	4.5E-05	−0.208	0.2816	6.4E-05	−0.2816
^{231}Pa	0	—	20	0.00437	−20	38.7	0.00874	−38.7
^{232}Pa	0	—	0.0346	7.6E-06	−0.035	0.0930	2.1E-05	−0.0930
^{233}Pa	0	—	350.4	0.07660	−350.4	473.9	0.10705	−473.9
^{232}U	0	—	3.855	0.00084	−3.855	23.12	0.00522	−23.12
^{233}U	24250	5.19	18530	4.05116	5720	11 600	2.62028	12650
^{235}U	0	—	107.3	0.02346	−107.3	584.3	0.13199	−584.3
^{236}U	0	—	2.852	0.00062	−2.852	59.32	0.0134	−59.32
total	466 600	100.	457 400	100.00	9200	442 700	99.995	23 900

TABLE 3.20. Th–^{235}U FUEL HM INVENTORY IN AN AVERAGE ASSEMBLY

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{232}Th	442 300	94.79	437 600	95.5249	4700	427 900	96.4608	14 400
^{233}Th	0	—	0.2452	5.3E-05	–0.245	0.3144	7.1E-05	–0.314
^{232}Pa	0	0	0.04597	1E-05	–0.046	0.1055	2.4E-05	–0.105
^{233}Pa	0	—	410.8	0.08967	–410.8	526.6	0.11871	–526.6
^{233}U	0	—	3171	0.69220	–3171	6122	1.38007	–6122
^{235}U	24 280	5.20	15 170	3.31150	9110	5460	1.23083	18 820
^{236}U	0	—	1682	0.36716	–1682	3206	0.72272	–3206
^{237}U	0	—	2.828	0.00061	–2.828	5.525	0.00124	–5.525
^{237}Np	0	0	63.47	0.01385	–63.47	281.5	0.06346	–281.5
^{238}Pu	0	—	6.793	0.00148	–6.793	84.85	0.01913	–84.85
^{239}Pu	0	—	0.4283	9.3E-05	–0.428	13	0.00293	–13
Total	466 600	99.99	458 100	100.001	8500	443 600	99.9999	23 000

TABLE 3.21. UO$_2$ FUEL HM INVENTORY IN AN AVERAGE ASSEMBLY

Nuclide	BOL		17 MW·d/kg HM			45 MW·d/kg HM		
	m[g]	Wt%	m[g]	Wt%	Δm	m[g]	Wt%	Δm
^{235}U	23540	5.03	15 160	3.298	8380	5818	1.305	17 722
^{236}U	0	0	1525	0.331	–1525	2966	0.665	–2966
^{237}U	0	—	2.853	0.001	–2.853	5.704	0.001	–5.704
^{238}U	444 500	94.97	440 200	95.75	4300	431 600	96.81	12900
^{237}Np	0	0	70.19	0.015	–70.19	293.2	0.066	–293.2
^{238}Pu	0	—	7.846	0.002	–7.846	97.28	0.022	–97.28
^{239}Pu	0	—	2236	0.486	–2236	3066	0.687	–3066
^{240}Pu	0	0	322	0.070	–322	1008	0.226	–1008
^{241}Pu	0	—	164.4	0.036	–164.4	674.8	0.151	–674.8
^{242}Pu	0	—	14.98	0.003	–14.98	237.3	0.053	–237.3
Total	468 000	100.0	459 700	100.00	8300	445 800	99.99	22 200

0.24 g/MW·d, 0.3 g/MW·d, and 0.28 g/MW·d, respectively. For comparison, the plutonium production rate in the core with the UO$_2$ fuel is 0.18 g/MW·d. The total plutonium production in the thorium-LEU fuel is lower than in the UO$_2$ fuel by a factor of 2.6. For the burnup rates considered here, there is no significant shift in the plutonium vector of the two fuel types.
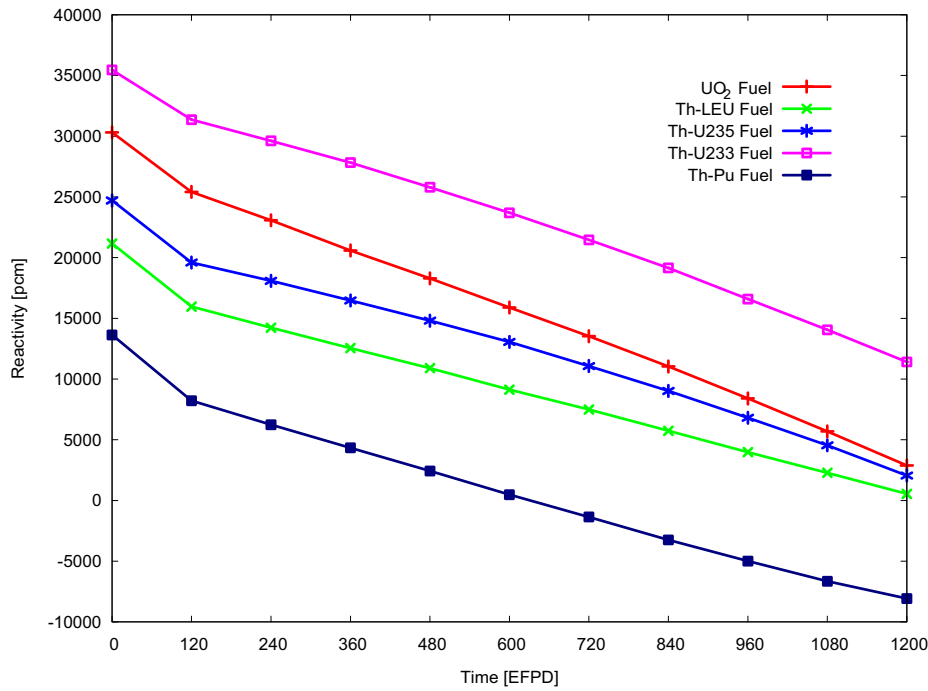
FIG. 3.21. Reactivity swing of PWR core with different fuel types.

For the discharge burnup considered in the current study, the production of minor actinides in the fuel types is significant only if the fuel contains plutonium or ^{238}U. Therefore, a significant amount of MAs is produced only in the thorium-plutonium fuel and in the conventional UO$_2$ fuel. The minor actinide production in the thorium-LEU fuel is barely observable.

The reactivity swing of the cores with different fuel types as a function of EFPD is shown in Fig. 3.21. At the assumed level of fissile enrichment of 5 wt%, the thorium-plutonium fuel is able to sustain criticality only up to 22 MW·d/kg HM. All other fuel types are able to sustain criticality up to 45 MW·d/kg HM. The Th-^{233}U fuel has a considerable reactivity reserve, which could allow extended operation with the current enrichment level of 5 wt%. Comparable extended burnup cores with UO$_2$ fuel require significantly higher enrichment and additional measures for reactivity control.

Figure 3.22 shows the conversion rate of PWR cores with different fuel types as a function of burnup level. The conversion rate of the cores with thorium fuel is significantly higher than with the conventional oxide fuel. This high conversion rate of thorium fuel cores could enable long cycles and high burnup rates, but it could also be used to design cores with lower enrichment levels and less HW consumption compared to cores with standard UO$_2$ fuel. In particular, the high conversion rate in the Th-^{233}U fuel enables the core to maintain a less pronounced reactivity swing profile, which could ease the measures needed for reactivity control.

3.4.4. Waste characteristics

One of the incentives for the use of ThFC in a PWR is its potential to reduce minor actinide generation in spent fuel and the related level of radiotoxicity. The use of thorium as a fertile material decreases the minor actinide production in the considered PWR cores. As can be seen from Tables 3.17–3.21, the minor actinide inventory of thorium fuel cores is significantly lower than in a core with conventional oxide fuel. Furthermore, at the assumed burnup level, the minor actinide production becomes observable only when the fuel involves plutonium or ^{238}U.

Figure 3.23 shows the ingestion dose of the different fuels after a burnup of 45 MW·d/kg HM as a function of cooling time. The thorium-plutonium fuel shows higher radiotoxicity up to 10 000 years' cooling times, whereas the Th-^{233}U fuel shows the lowest. At cooling times above 10 000 years, the ingestion dose is at the same level in all fuel types.
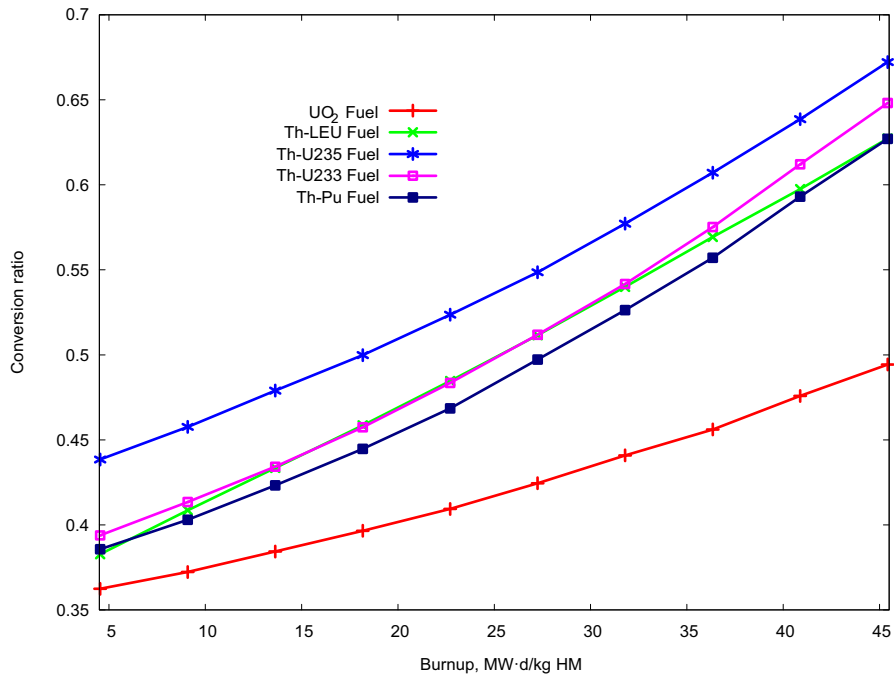
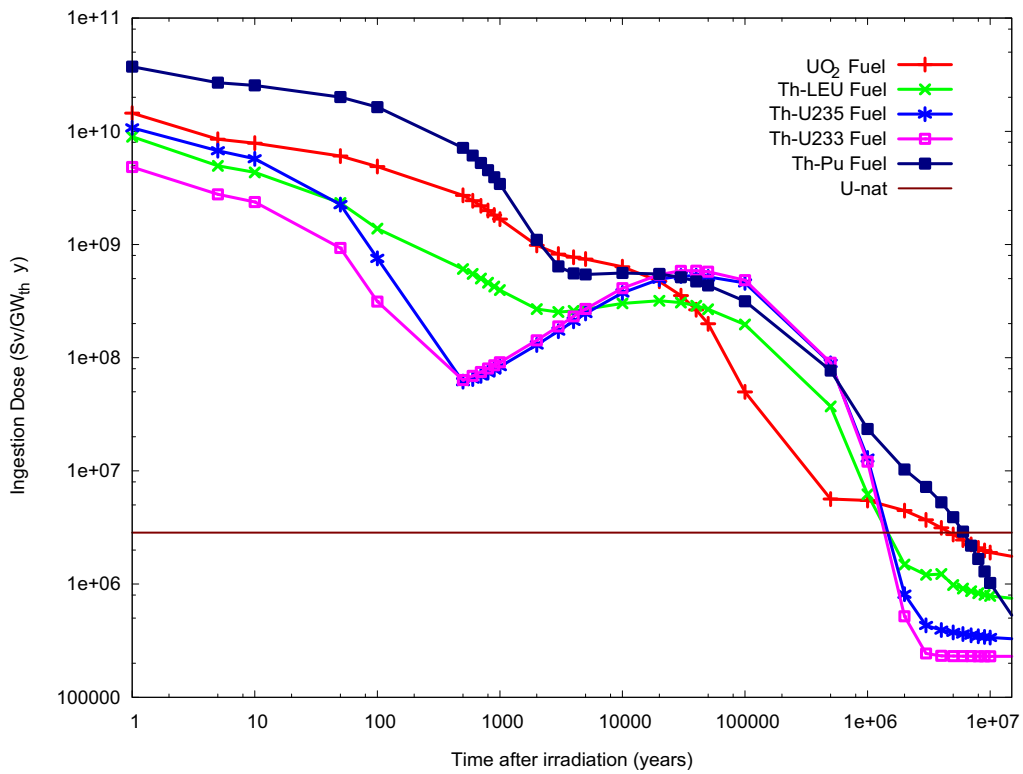*FIG. 3.22. Conversion rate of PWR core with different fuel types.*

FIG. 3.23. Ingestion dose of spent fuel after burnup of 45 MW·d/kg HM.

Figures 3.24 and 3.25 show the specific activity and the decay heat output, respectively, as a function of the cooling time after a burnup of 45 MW·d/kg HM. Both the specific activity and decay heat output are slightly higher for the thorium-plutonium and UO$_2$ fuel types for cooling times up to 1000 years. However, both values decrease
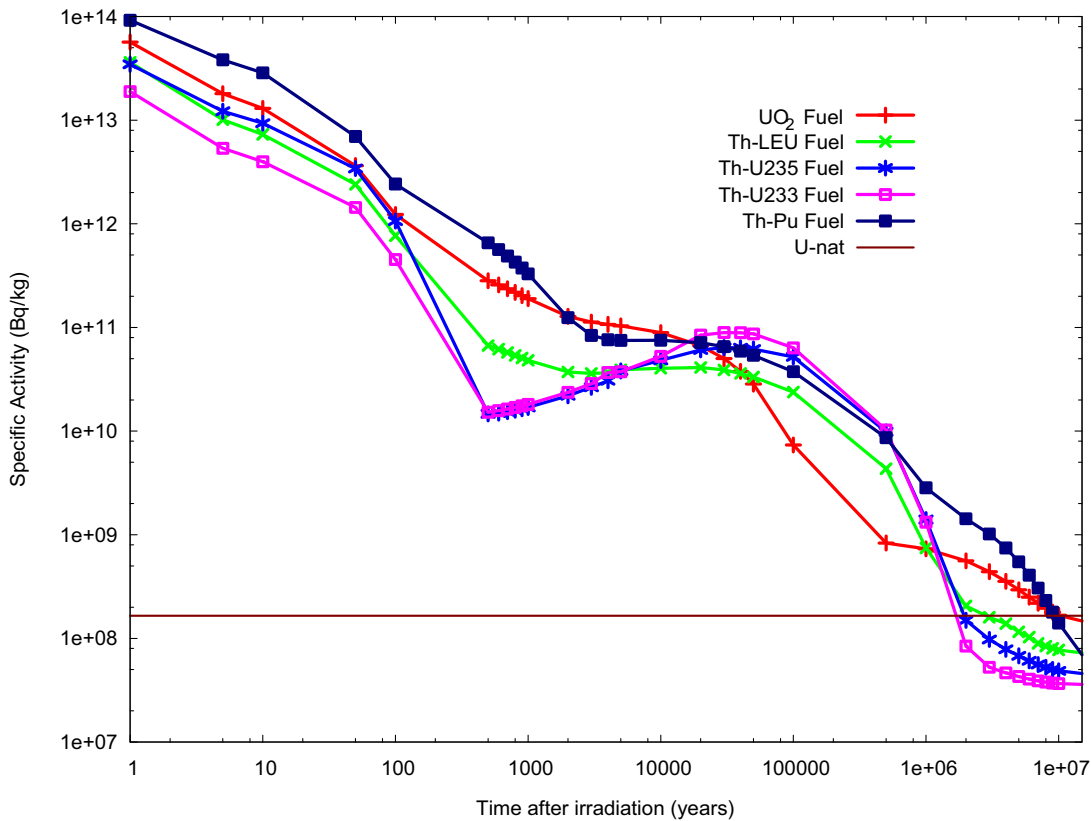
FIG. 3.24. Specific activity of spent fuel after burnup of 45 MW·d/kg HM.

significantly to a lower level in all fuels within cooling times up to 1000 years. Both the specific activity and the decay heat output increase for cooling times up to 100 000 years in the other thorium fuels. For cooling times beyond 10 000 years, the decay behaviour of all fuel types is similar.

3.4.5. Proliferation considerations

The use of ThFC in a PWR is also motivated by considerations of proliferation resistance. In general, plutonium production can be reduced by the use of ^{232}Th instead of ^{238}U as a fertile material. The potential of a fuel type for better proliferation resistance may be characterized by the spontaneous fission neutron generation, the decay heat output, and surface gamma dose rate. Table 3.22 shows these quantities for the studied fuel types after irradiation to a burnup level of 45 MW·d/kg HM.

The spontaneous fission neutron generation in Th–^{233}U and Th–^{235}U fuel is significantly lower than in the other fuel both at shutdown and after a one year cooling period. The decay heat generation in thorium fuel is almost at the same level as in the UO$_2$ fuel at shutdown. After a longer cooling time of up to 10 000 years, the thorium-plutonium fuel shows higher decay heat generation than the other fuel types. The dose rates given in the table are for infinite plane source irradiated material.

3.4.6. Conclusions

The use of thorium fuel in a current PWR without core modifications was considered by the EU in the past. The current neutronic assessments performed within this INPRO ThFC study indicate that the ThFC could supplement the current uranium–plutonium fuel cycle to improve operational performance and waste reduction in a current PWR.
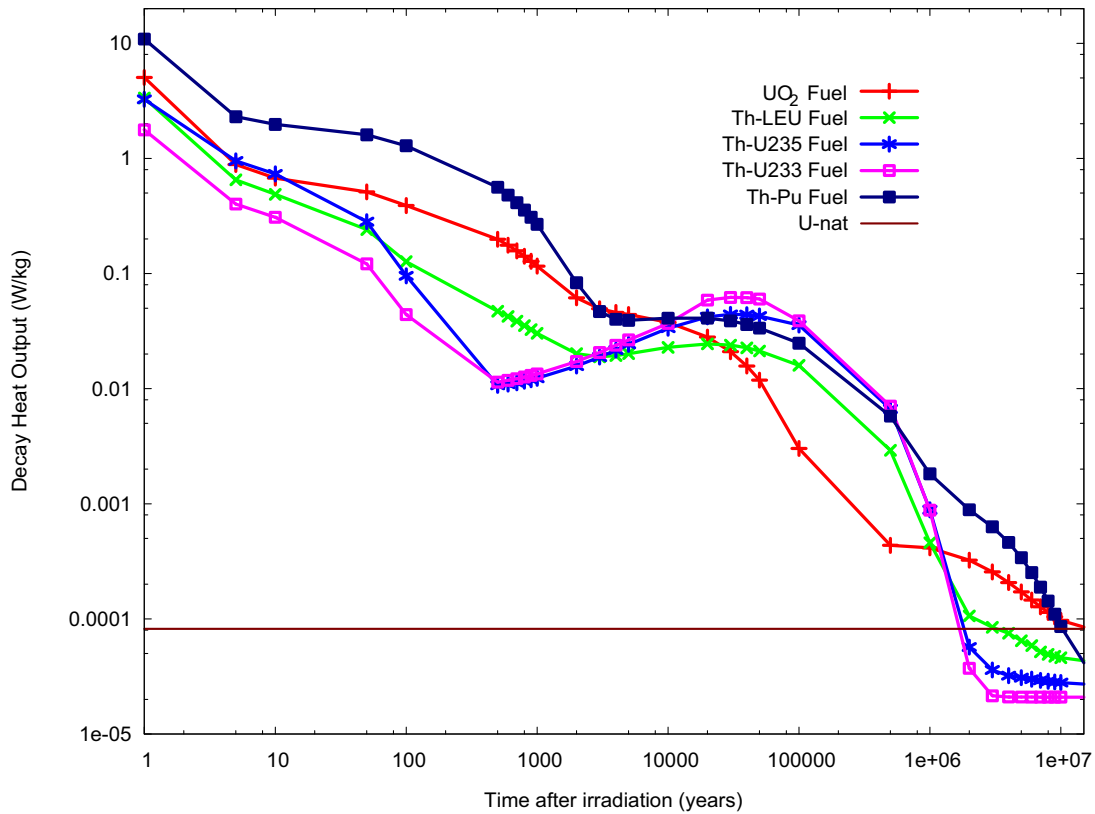
FIG. 3.25. Decay heat of spent fuel after burnup of 45 MW·d/kg HM.

TABLE 3.22. PROLIFERATION RESISTANCE RELEVANT QUANTITIES FOR DIFFERENT FUELS AFTER A BURNUP OF 45 MW·d/kg HM

Parameter	Cooling time [a]	Th-LEU	Th-Pu	Th-^{233}U	Th-^{235}U	UO$_2$
Spontaneous fission (n/kg)	0	40 763.9	5 588 350	0.0188	13.5012	201 148
	1	11 629.8	3 323 250	0.018725	13.2959	577 61.7
Specific activity (Bq/kg)	0	3.3E15	4.017E15	2.207E15	2.54E15	4.74E15
	1	3.6E13	9.216E13	1.888E13	2.1E13	5.67E13
Decay heat (kw/kg)	0	0.70786	0.858194	0.795171	0.91357	0.98695
	1	0.00338	0.010950	0.003301	0.00370	0.00526
Dose (Sv/h) plane source	0	432984	510 309	267 095	297 382	536 650
	1	576.042	1041.56	314.058	346.923	785.005

The thorium fuel types studied show significantly higher conversion rates in comparison to conventional oxide fuel. This could enable efficient fuel utilization in a PWR through high burnups in the fertile thorium. To achieve this goal and for practical implementation of ThFC, further research and development in all aspects of fuel development, core design and fuel cycle optimization are needed.

Further, ThFC have the potential to improve issues related to waste management in a current PWR. The use of thorium as a fertile material can reduce minor actinide generation and the radiotoxicity of spent fuel.

In considerations related to proliferation resistance, the results of the current analyses show no significant difference between the studied thorium fuel and the standard oxide fuel for the assumed characteristics and burnup levels.

4. GLOBAL SCENARIOS WITH THE INTRODUCTION OF THORIUM

4.1. SCOPE OF CONSIDERATION

In 2007, INPRO established a collaborative project on Global Architecture of Innovative Nuclear Systems Based on Thermal and Fast Reactors Including Closed Fuel Cycle (GAINS) to investigate possible benefits from the introduction of innovative nuclear systems. INPRO's collaborative project Further Investigation of the Thorium Fuel Cycles and GAINS are symbiotic and were developed in close cooperation. The scenarios of global nuclear capacity growth were developed in GAINS and are used for the considerations in this report together with principles for distribution among the types of reactors selected. Detailed analysis of possible thorium options was performed in the current study, and the results have been delivered to GAINS for generic considerations.

To determine the potential of a given global nuclear energy system (NES) to become a sustainable energy supply option in the 21st century, several calculations were carried out simulating various possible scenarios of global nuclear energy demand and corresponding nuclear generation capacity growth. In the framework of every scenario, a number of options are considered to satisfy the total expected nuclear capacity needed.

The study presented in this report is an attempt to apply a holistic approach to the following issues of nuclear fuel cycles in order to:

— Investigate potential for reduction in ^{235}U enrichment requirements and natural uranium requirements in the case of thorium utilization;
— Analyse the reduction of long-lived radioactive waste inventories by diminishing the production of plutonium and minor actinides;
— Assess the advantages from increasing the world's fissile resources by breeding ^{233}U from thorium;
— Estimate fabrication and reprocessing capacities necessary for the commercial utilization of thorium fuel and fuel cycles.

4.2. INTRODUCTION TO THE TOOL APPLIED

Material flow calculations for the nuclear energy system considered were performed with the tool Model for Energy Supply System Alternatives and their General Environmental impacts (MESSAGE) [56], which is the IAEA's large scale dynamic system engineering, economic optimization model used for the development of medium and long term energy scenario and policy analysis.

MESSAGE is commonly used to formulate and evaluate alternative energy supply strategies for user defined constraints on, for example, new investment, market penetration rates for new technologies, fuel availability and trade. The tool is flexible enough to be used for analysis of NES, including those involving thorium fuel. Nuclear power processes can be taken into account: for example, changes of the isotopic composition of spent fuel during the cooling time in storage at the nuclear power plant and reprocessing lag time ('external fuel cycle time') because of radioactive decay of unstable isotopes (half-life time of ^{238}Pu: 87.7 years, ^{241}Pu: 14.4 years, ^{242}Cm: 0.447 years and ^{244}Cm: 18.1 years). Nonetheless, in MESSAGE some minor limitations do exist, e.g. the decay of plutonium and minor actinides in stock cannot be taken into account.

The results of NES modelling were compared with various tools, including MESSAGE, in the framework of the INPRO project, and good convergence of the results was confirmed.

The user manual's extension for modelling of nuclear energy system scenarios with MESSAGE is currently under development. The manual will provide step by step guidance for creating mathematical models representing nuclear energy systems to the required level of detail and will present several demonstration cases, including the modelling of a nuclear energy system based on thermal and fast reactors with a fully closed fuel cycle.

Special models were created in the MESSAGE code for ThFC introduction options, and calculations of scenarios with the introduction of thorium and ^{233}U fuel in thermal reactors (LWRs and HWRs) were performed. Further, calculations of scenarios were made with introduction of breeder reactors using thorium in the prospective nuclear energy system with a closed fuel cycle. The calculations included optimization of material flows and economic considerations. In the back end of thorium–uranium fuel cycle, there are isotopes such as ^{231}Pa, ^{229}Th and ^{230}U, which may have long term radiological impact. These isotopes were not considered in the present study and should be considered in following publications.

MESSAGE output parameters selected as indicators to compare fuel cycle options are:

— Distribution of total nuclear generation capacity among reactor types constituting the NES as the result of the optimization process;
— Cumulative consumption of natural uranium in the system;
— Necessary services of uranium enrichment, fuel fabrication, spent fuel reprocessing;
— Spent fuel and minor actinides accumulated in the NES;
— Annual discharge and consumption of plutonium, ^{233}U and minor actinides.

4.3. APPROACH USED IN THE STUDY

High and moderate growth scenario

In GAINS, two nuclear energy demand scenarios were selected for analysis. The high case scenario corresponds to the medium expectation for growth of nuclear capacity within the IPCC/SRES scenarios (Intergovernmental Panel on Climate Change/Special Report on Emission Scenarios [57]). In this high case scenario, global nuclear power installation is assumed to reach 1500 GW(e) by the the mid-21st century and 5000 GW(e) by 2100. In the moderate case scenario considered in GAINS and in this study, 1000 GW(e) of world nuclear generation is assumed to be reached by the middle of the 21st century and 2500 GW(e) by 2100.

Non-geographical groups of countries

To analyse these scenarios, the world is divided into three non-geographical groups of countries specified as:

— G1 — countries that are most involved in the development and deployment of NES and consequently, able to incorporate them as soon as commercially available;
— G2 — countries with significant experience in the use of nuclear energy and possessing most of the necessary infrastructure, but which are not yet ready to incorporate the most advanced nuclear energy systems;
— G3 — countries supposed to incorporate nuclear energy in their energy supply mix as newcomers.

In accordance with GAINS, this study assumed, for the three non-geographical groups, that the share of power demand by 2100 is G1:G2:G3 = 40%:40%:20%.

G3's power share is assumed to increase linearly from 0% in 2008 to 20% by 2100. An important characteristic of the G3 group is that for these countries, the installation of exclusively thermal reactors is assumed. In G2 countries, no fast reactors will be installed, and in G1 countries, all types of reactors are assumed to be installed.

The growth of nuclear power demand of each group for the *high case* and *moderate case* scenario is shown in Tables 4.1 and 4.2.

TABLE 4.1. DISTRIBUTION OF NUCLEAR ELECTRICITY DEMAND AMONG NON-GEOGRAPHICAL GROUPS IN THE HIGH CASE SCENARIO OF GLOBAL NUCLEAR GROWTH (5000 GW(e) BY 2100)

Year	Capacity, GW(e)			
	G1	G2	G3	Total
2008	149	149	0	298
2030	333	333	33	700
2050	682	682	137	1500
2100	2000	2000	1000	5000

TABLE 4.2. DISTRIBUTION OF THE NUCLEAR ELECTRICITY DEMAND AMONG NON-GEOGRAPHICAL GROUPS IN THE MODERATE CASE SCENARIO OF GLOBAL NUCLEAR GROWTH (2500 GW(e) BY 2100)

Year	Capacity, GW(e)			
	G1	G2	G3	Total
2008	149	149	0	298
2030	286	286	29	600
2050	454	454	91	1000
2100	1000	1000	500	2500
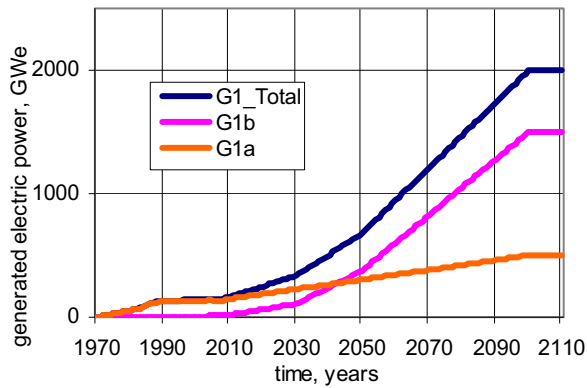
FIG. 4.1a. Division of group G1 into subgroups G1a and G1b — high case.
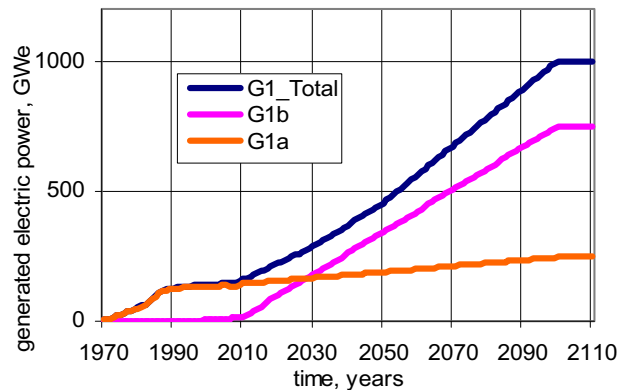
FIG. 4.1b. Division of group G1 into subgroups G1a and G1b — moderate case.

Additionally, G1 is divided in two subgroups in order to incorporate in the model a difference between two groups of technology developing countries, i.e. those that do not anticipate a rapid nuclear growth (innovative reactors are necessary to keep nuclear as the best economic option) and those that do anticipate rapid growth of nuclear power (innovative reactors with high breeding rates are needed). Figure 4.1 shows the fractions of subgroups G1a (rapid growth) and G1b (modest growth) in both scenarios (high and moderate case).

Assumptions on features of reactors and fuel cycles

The approach in GAINS makes several assumptions about reactor and fuel cycle features that are also reflected in this study:
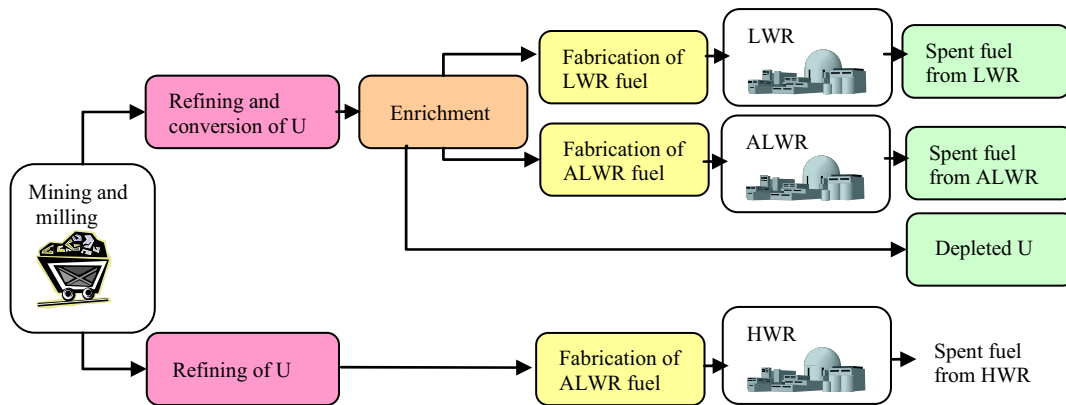
FIG. 4.2. NES based on thermal reactors with once-through fuel cycle (assuming HWR is fuelled without enrichment).

— The lifetime of existing plants (in operation in 2008) (LWR and HWR) is 40 years.
— The lifetime of advanced and innovative plants is 60 years.
— Uranium enrichment tails assay is equal to 0.3% until 2015 and to 0.2% after 2015.
— The time period for thermal reactor spent fuel cooling is five years, reprocessing time is one year.
— The time period for fast reactor spent fuel cooling and reprocessing is three years.
— Fuel is reprocessed without loss of heavy metal isotopes.
— Spent fuel from HWRs can be reprocessed for scenarios involving thermal reactors, but it cannot be reprocessed in the case of introduction of fast reactors.
— The time from nuclear material mining until fuel assemblies loading is not taken into account in material flow calculations.

Another assumption that can be very important from the point of view of thorium introduction is the postulated share of heavy water moderated reactors. In all GAINS scenarios, this fraction is defined at a level of 6% of total nuclear capacity. The results shown below indicate that there may be increased positive effects of thorium implementation in heavy water moderated thorium breeders. An attempt to study ThFC introduction in HWRs by increasing its share limit beyond 6% is presented in Annex VI. However, a thorough consideration of an extension of the HWR share in the global NES is currently beyond the scope of this study and should be investigated in upcoming projects.

Once-through and closed fuel cycles

GAINS subsequently considers two types of nuclear energy system. In 'business as usual' (BAU) cases, the nuclear energy system includes *thermal reactors* of LWR (including advanced LWR) and HWR design with a *once-through fuel cycle* (i.e. without spent fuel reprocessing). The scheme of this fuel cycle is presented in Fig. 4.2.

Two methods for the introduction of ThFCs are presented in this study. One will correspond exactly to Fig. 4.2 with thermal reactors in a once-through fuel cycle, and the other one will address thermal reactors with spent fuel recycling and possible thermal breeding as an option.

A second type of nuclear energy system in GAINS is based on thermal and fast reactors with MOX fuel, and the use of plutonium multi-recycling in fast reactors. Options such as molten salt reactors or accelerator driven systems could also be considered in the future. In addition to GAINS, several thorium options are considered in the current report. NES based on a *closed fuel cycle* is depicted in Fig. 4.3.

Global uranium and thorium resources

The available global uranium resources data were taken from the OECD/NEA–IAEA Red Book [58] and divided in five grades: a, b, c, d and e. Grades a, b and c refer to identified and undiscovered resources of various cost that comprise 16 million tonnes of natural uranium as shown in the Table 4.3. Grade d is associated with uranium in phosphates and has a deposit of 22 000 kt of uranium with recovery cost in the range > \$350/kgU.
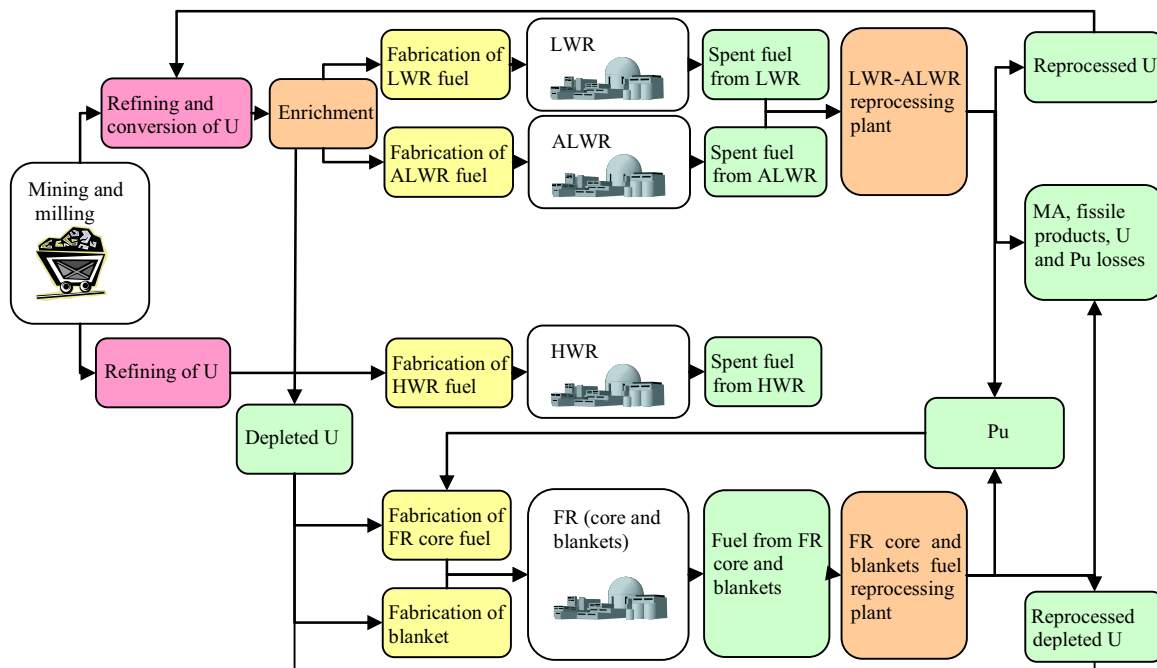
FIG. 4.3. Nuclear energy system based on thermal and fast reactors with inclusion of plutonium multi-recycling in fast reactors with MOX fuel (assuming HWR is fuelled without enrichment).

TABLE 4.3. URANIUM RESOURCES

Cost of recovery, $/kg U	Identified resources		Undiscovered resources		Phosphates
	Reasonably assured resources (RAR)	Inferred resources	Prognosticated resources	Speculative resources (SR)	
<40 **a**	1 766 400	1 203 600			
40–80 **b**	1 191 600	654 800	1 900 000		22 000 000 **d**
80–130 **c**	380	277 600	900 000	4 800 000	
Cost range unassigned				2 978 600	
Total	5 474 300		10 578 600		
	16 052 900				
	38 052 900				

Natural uranium resources are limited by 38 million tonnes for the sum of a, b, c and d grades. Grade e resources are associated with uranium in sea water. Grade e is assumed to include practically unlimited resources with cost of recovery higher than $350/kgU (see Table 4.4).

Information on thorium resources was published in Red Books, typically using the current terminology for uranium resources (e.g. reasonably assured resources and estimated additional resources I and II, which are now termed inferred and prognosticated resources, respectively). Most of the largest identified thorium resources were discovered during the exploration of carbonates and alkaline igneous bodies for uranium, rare earth elements, niobium, phosphate, and titanium. Now, thorium is recovered mainly from the mineral monazite as a by-product of processing heavy mineral sand deposits for titanium, zirconium, or tin bearing minerals.

Monazite is a phosphate mineral normally occurring in small isolated crystals. It contains rare earth metals and is an important ore for thorium, lanthanum and cerium. Many regions in the world have large deposits of monazite sands. At the beginning of 2008, more than 37 kt of thorium were produced worldwide [52].

TABLE 4.4. URANIUM RESOURCES DISTRIBUTION IN GRADES

Grade	Resources type/cost	Resources value, kt	Total amount, kt
a	Identified and undiscovered/ $40/kg	2970	
b	Identified and undiscovered/ $80/kg	3746	16 053
c	Identified and undiscovered/ $130/kg	9337	
d	Phosphates/>$350/kg	22 000	Including phosphates: 38 053
e	Sea water/>$350/kg	Unlimited	

TABLE 4.5. REACTOR TYPES USING UOX/MOX AND THORIUM FUEL

No.	Reactor index	Fuel type	Data provider
	Reactor types using UOX/MOX fuel		
1	HWR	Nat U	IAEA (NFCSS)
2	LWR	UOX	IAEA (NFCSS)
3	ALWR	UOX	France
4	LWRM	MOX	France
5	FR (BR~1)	MOX, depleted U	Russian Federation
6	FR12 (BR~1.2)	MOX, depleted U	India
	Reactor types using thorium fuel		
7	LWR0	UO2, Th	Rep. of Korea
8	LWR1	Pu, Th	Rep. of Korea
9	LWR2	Pu, U233, depleted U	Russian Federation
10	LWR3	Th, U233	Shippingport type
11	HWR1	Pu, Th	Canada
12	HWR2	Pu, U-233, Th	Canada
13	AHWR	Pu, U-233, Th	India
14	HTR	U-3, Th	Russian Federation
15	FRTh	Pu, depleted U, Th in blankets	Russian Federation

Thorium resource data were taken from the same Red Book [58] as for uranium. Worldwide total thorium resources are estimated at about 6.08 million tonnes, including undiscovered resources, with cost of recovery of less than $80/kg [59].

4.4. REACTOR DATA

Data of the reactors (with the exception of standard HWRs and LWRs) considered in this study were provided by the Member States participating in the ThFC project. Data of reactors using thorium fuel are presented in more detail in Annex I.

Uranium–plutonium fuel cycle data for nuclear energy system compilation and comparison of the variants were taken from a databank compiled within the GAINS project. Reactors using a uranium–plutonium and/or a Th–^{233}U fuel cycle chosen for scenario simulation are listed in Table 4.5.

General input data for the reactors listed above are presented in Table 4.6 to evaluate the scenarios based exclusively on thermal reactors including the BAU case (LWR3 is considered separately in Annex IV).

Input data for the fast reactors considered in the study are shown in Table 4.7. Note that the ALWR type is a hypothetical reactor that was proposed by Member States in the framework of GAINS as a level of light water

TABLE 4.6. GENERAL CHARACTERISTICS OF THERMAL LIGHT AND HEAVY WATER COOLED REACTORS AND HTRs

Parameter	LWR	ALWR	LWRM	LWR0	LWR1	LWR2	HWR	HWR1	HWR2	HTR
Fuel type	UOX	UOX	U-Pu	UOX-Th	Th-Pu	Pu-U3	UOX	Pu-Th	Pu-Th-U3	Th-U3
Electric capacity, MW	1000	1500	1500	900	900	1000	600	668	668	270
Thermal efficiency	0.33	0.34	0.34	0.32	0.32	0.33	0.30	0.32	0.32	0.45
Load factor	0.80	0.86	0.86	0.8	0.8	0.8	0.8	0.95	0.95	1
Life time, years	40	60	60	60	60	60	40	60	60	60
Average burnup, MW·d/kg HM	45	60	60	38.2	40.5	42.7	7	20.3	20	
Construction time, years	5	5	5	5	5	5	5	5	5	5
Uranium enrichment, %	4	4.9	Depl. U	20	—	Depl. U	0.711	—	—	—
Cooling time, years	5	5	5	5	5	5	5	5	5	5
Fuel residence time, EFPD	1168	1760	1760	927	983	885	292	825	810	833
Mass of the core, tHM	78.7	129.4	128.9	67	67	70.3	83.4	71.4	71.4	6
Pu content in fresh fuel	—	—	0.119	—	0.074	0.024	—	0.038	0.011	—
Th content in fresh fuel	—	—	—	0.76	0.913	—	—	0.096	0.915	0.92
U3 content in fresh fuel	—	—	—	—	—	0.018	—	—	0.014	0.08

Note: U3 refers to ^{233}U, depl. U refers to depleted U.

TABLE 4.7. GENERAL CHARACTERISTICS OF FAST REACTORS USED IN SCENARIO CALCULATION

Parameter	FR (BR~1)		FRTh		FR12 (BR~1.2)	
Core fuel type	Pu-depleted U		Pu-depleted U		Pu-depleted U	
Blanket fuel type	Depleted U		Th		Depleted U	
Electric capacity, MW	870		880		500	
Thermal efficiency, %	42		42		40	
Load factor, %	85		85		75	
Life time, years	60		60		40*	
Core fuel burnup, MW·d/kg HM	65.9		72		78	
Conversion ratio	1		1		1.2	
	Core	Depleted U blanket	Core	Thorium blanket	Core	Depleted U blanket
Fuel residence time, EFPD	420	457	441	441	540	762
Mass of the fuel, HMt	12.6	11.7	12.5	33	8.1	19.2
Pu content in fresh fuel	0.22	—	0.218	—	0.24	—

* As in GAINS, the lifetime of all innovative reactors in scenario calculations is assumed to be 60 years.
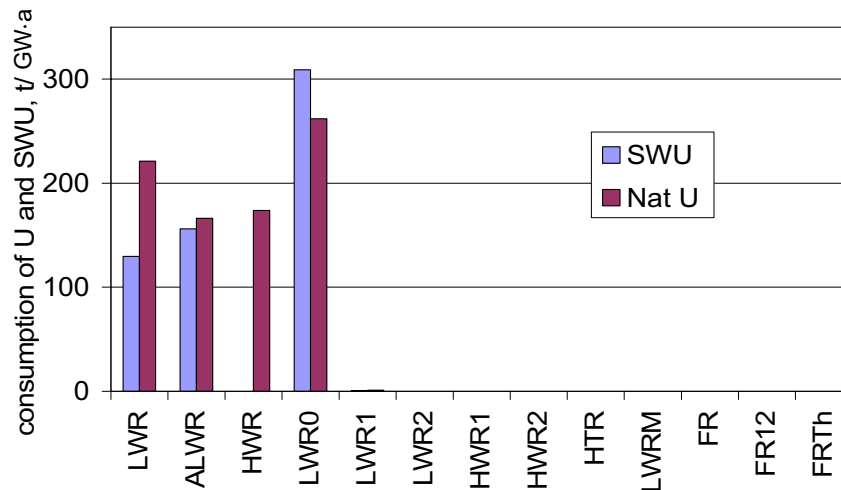
FIG. 4.4. Specific annual consumption of natural uranium and SWU requirements.

technology that could be attained and will become dominant during the 21st century. The characteristics of an ALWR are quite challenging, e.g. its specific uranium consumption parameters are better than those of modern HWRs. There are still some concerns of its feasibility, and the maturity of ALWR design is far lower than of other reactors operated in once-through fuel cycle.

The general characteristics of fast reactors with a conversion rate ~1, based on the uranium–plutonium fuel cycle and those using thorium in blankets are quite similar, apparently, because the thorium blanket has been designed as an option for a fast reactor with a blanket with depleted uranium.

Important characteristics of reactor types selected

Preliminary evaluation of results of MESSAGE is useful for an in-depth understanding of the scenario simulation results by identifying cause-effect links. Here, several basic characteristics of reactor types included in the scenario studies will be compared: consumption of natural uranium, enrichment requirements, mass of fresh fuel, inventory/consumption/discharge of plutonium, and amount of minor actinides discharged.

A comparison of amount of natural uranium consumed per unit of produced energy is shown in Fig. 4.4 together with corresponding specific uranium enrichment requirements. (For data of reactors, see Table 4.5.)

Figure 4.4 shows that the reactor called LWR0 (see also Table 4.6) using thorium and enriched uranium has the highest consumption of natural uranium and the highest requirements for enrichment. The lowest consumption of natural uranium and the lowest requirements for enrichment are achieved in the reactor LWR1 using a mixture of thorium and plutonium as fuel.

Specific mass of fresh fuel consumed (equals the value of specific mass of spent fuel to be discharged) per unit of energy produced is provided in Fig. 4.5 for all reactors. In the case of the fast reactor with thorium blanket the mass of the core fuel consumed and mass of the blanket are provided separately.

The data on plutonium consumption and its production (accumulated in the spent fuel) are presented in Fig. 4.6. The figure also shows the 'inventory' of plutonium in a reactor type, i.e. the difference between plutonium production and its consumption. Negative inventories indicate that this type of reactor must be supplied with plutonium from outside.

Production of minor actinides is shown in Fig. 4.7. All thorium utilizing reactors, with the exception of the HTR, produce minor actinides because of the presence of plutonium and ^{238}U in their cores (see Table 4.6).

The basic characteristics of reactors presented above and the share of these different reactors in an NES are the main cause for the results produced within the scenario studies presented later in this report.
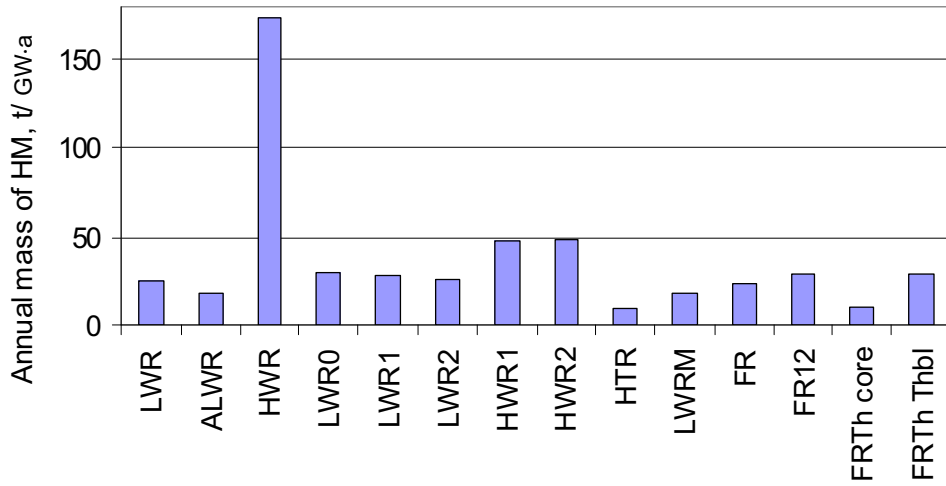
FIG. 4.5. Specific annual consumption of fresh fuel (discharge of SF).
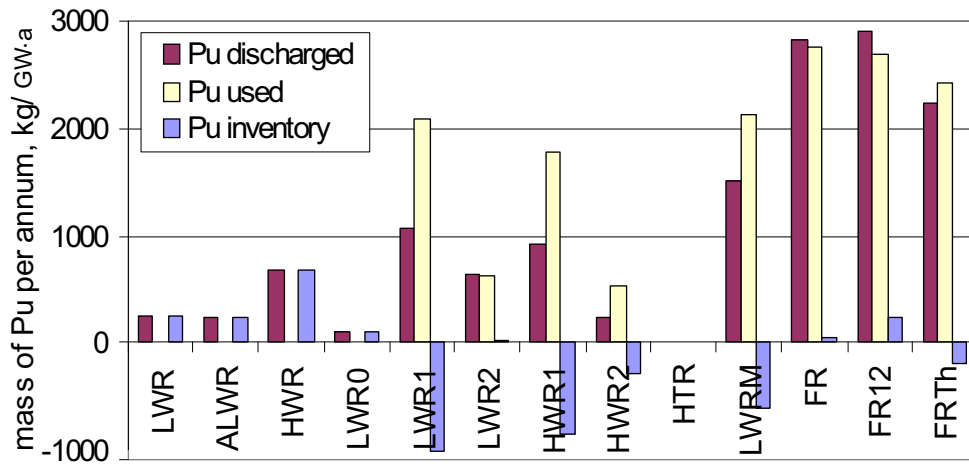

FIG. 4.6. Balance of plutonium: specific annual consumption of plutonium (in 'fresh' fuel) and its production (in spent fuel).
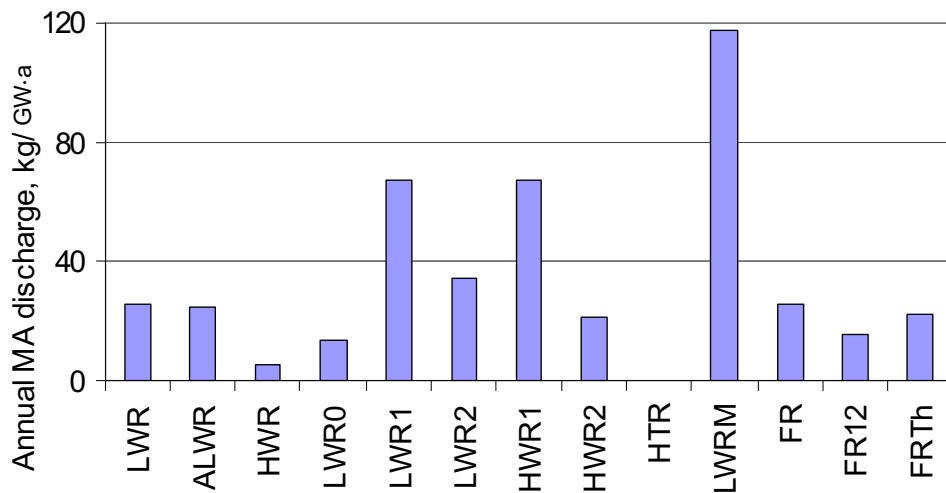

FIG. 4.7. Specific annual amount of minor actinides to be discharged from reactors in their spent fuel.
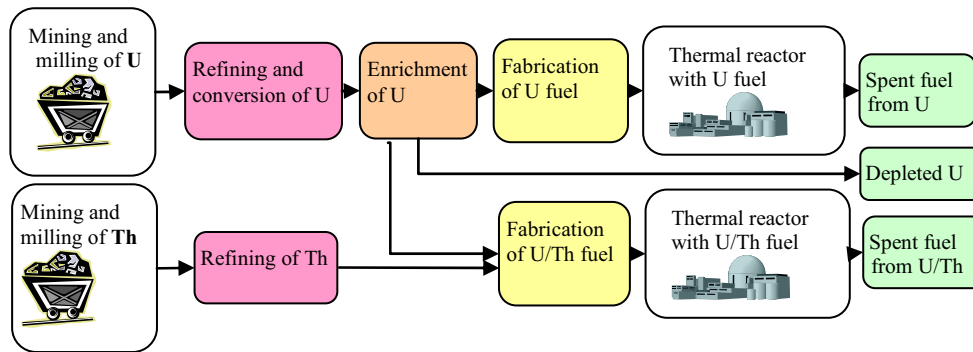
FIG. 4.8. Once-through fuel cycle based on water cooled thermal reactors using thorium without spent fuel reprocessing.

4.5. BASIC CHARACTERISTICS OF FUEL CYCLES WITH THORIUM UTILIZATION

Schemes and characteristics of the fuel cycles available for simulation and results of the MESSAGE simulation depend strongly on the reactor data and material flow parameters compiled. Pursuant to the reactor data presented in Section 4.4, three variants, some of which are divided in several options, of thorium fuel introduction are considered in the scenario study:

— Once-through fuel cycle based on thermal reactors using thorium without spent fuel reprocessing;
— Closed fuel cycle based on thermal reactors using thorium and/or ^{233}U with spent fuel reprocessing and ^{233}U (as well as plutonium) recycling;
— Closed fuel cycle based on thermal and fast reactors using thorium and/or ^{233}U with spent fuel reprocessing and recycling of ^{233}U and plutonium.

A once-through fuel cycle based on thermal reactors using uranium fuel (BAU case in GAINS) and thermal reactors with Th/(^{235}U+^{238}U) fuel is depicted in Fig. 4.8. The nuclear energy systems with once-through ThFCs considered in the scenario study of this report usually include existing and advanced LWR type reactors using UOX fuel, existing HWR using UOX fuel together with advanced LWRs using Th/(^{235}U+^{238}U) fuel. The back end consists of intermediate storage of spent fuel and depleted uranium.

One variant of a closed nuclear fuel cycle with uranium and thorium spent fuel reprocessing (shown in Fig. 4.9) includes existing and advanced LWR type reactors using UOX fuel, existing HWRs with UOX fuel, advanced LWRs utilizing Pu/Th/^{233}U fuel and HWRs also utilizing Pu/Th/^{233}U fuel. Plutonium from reprocessed spent fuel is being used as the fissile driver in fresh fuel for advanced thermal reactors and also ^{233}U is being recycled for Pu/Th/^{233}U fuel production.

Figure 4.10 depicts a closed fuel cycle based on thermal and fast reactors with thorium in radial blankets, and multi-recycling of plutonium and ^{233}U in thermal (both plutonium and ^{233}U) and fast (only plutonium) reactors.

Thorium use in fast reactors is usually limited to its insertion into blankets. In this case, if the fast reactor (as it is usually the case) doesn't use ^{233}U in its core, it becomes unsustainable from the point of view of plutonium consumption and production, i.e. it consumes more plutonium than it produces and needs some external feed, unless it is used as a thorium to ^{233}U converter to produce ^{233}U fuel to be used in LWRs and HWRs.

List of options considered in the study

As shown in Table 4.8 for the high case scenario of nuclear energy demand (see Table 4.1), in total, eight nuclear energy systems consisting of combinations of ten types of thermal reactors are considered in once-through and closed fuel cycles. General reactor data and material flow parameters were provided in the Section 4.4 and presented in more detail in Annex I.

Seven nuclear energy systems with combinations of 12 types of thermal and fast reactors with closed fuel cycles are considered as shown in Table 4.9 for the high case scenario (see Table 4.1) of energy demand.
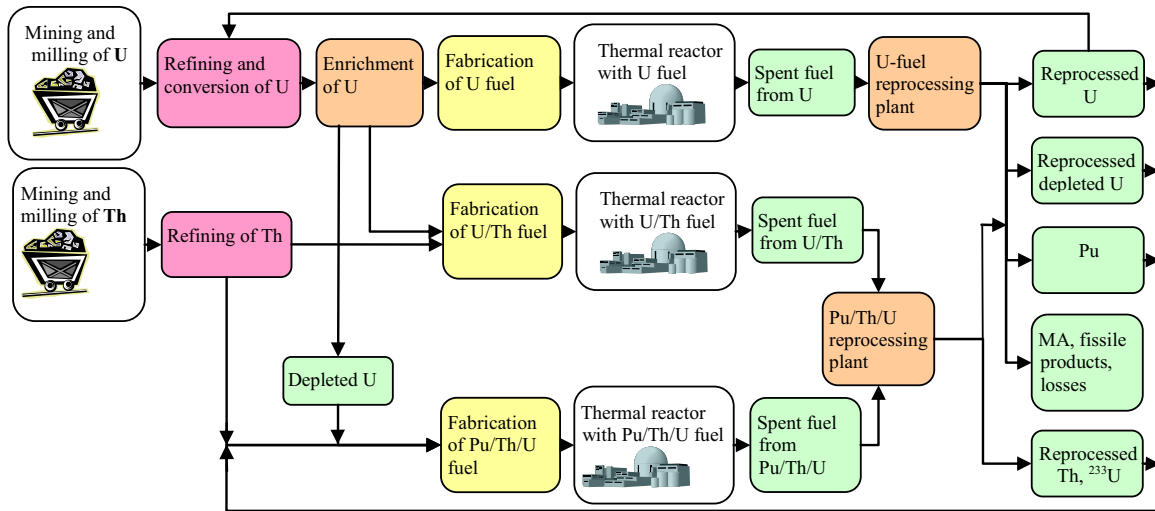
FIG. 4.9. Closed fuel cycle based on water cooled thermal reactors using thorium and/or ^{233}U (with spent fuel reprocessing and recycling of plutonium and ^{233}U).
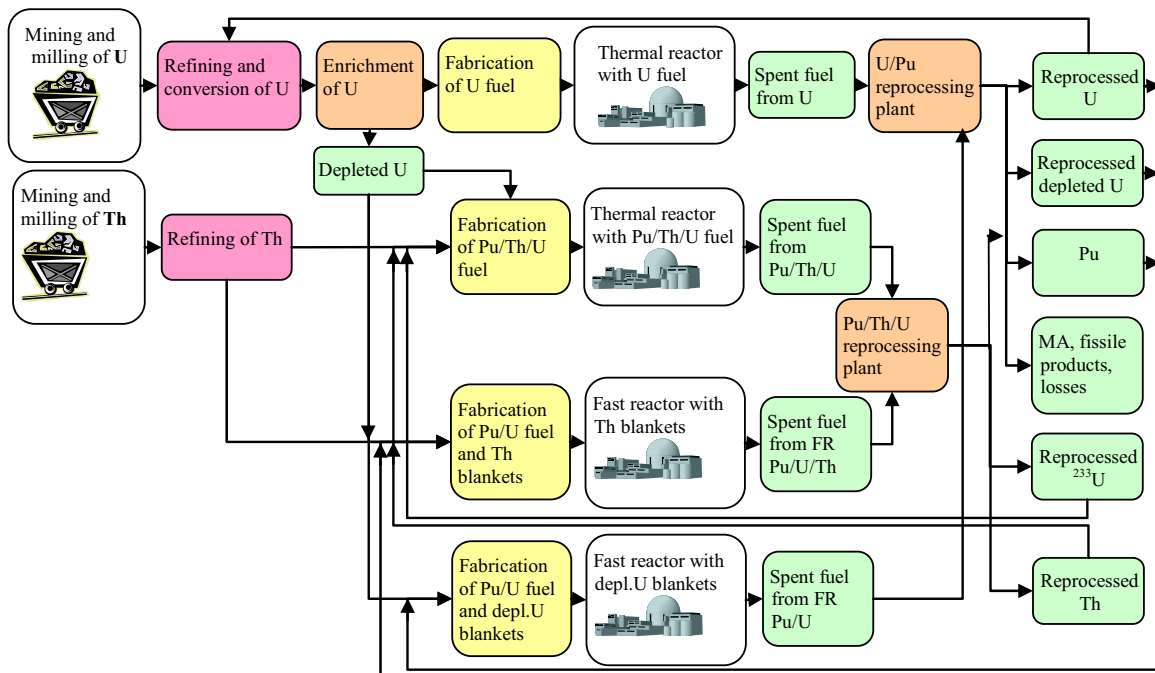

FIG. 4.10. Closed fuel cycle based on thermal and fast reactors using thorium and/or ^{233}U (with spent fuel reprocessing and recycling of ^{233}U and plutonium).

Some additional scenarios were evaluated and are documented in the annexes to this report:

— For the moderate case scenario, some of the nuclear energy system based on thermal reactors only (BAU, LWRM, LWR12/HWR12) and some for the NES with combinations of thermal and fast reactors (BAU, FR, FRTh/LWR12/HWR12) are considered in Annexes II and III, respectively.

— Results for the high case scenario with the nuclear energy system FRth/LWR3 are presented in Annex IV. The LWR3 reactor is based on the thermal breeder reactor of Shippingport type [2].

— Simplified scenarios and nuclear energy system consisting of only LWR, AHWR or ALWR reactors are considered in Annex V in order to compare characteristics and effects of thorium once-through mode in the heavy water reactor domain.

TABLE 4.8. LIST OF NUCLEAR ENERGY SYSTEM OPTIONS BASED ON THERMAL REACTORS CONSIDERED IN THE HIGH CASE SCENARIO

Reactors	Options considered							
	BAU	SLWRM	LWR1/LWR2	HWR1/HWR2	LWR12/HWR12	LWR1/HTR	SLWR0	LWR0/HTR
HWR	x	x	x	x	x	X	x	x
LWR	x	x	x	x	x	X	x	x
ALWR	x	x	x	x	x	X	x	x
LWR0							x	x
LWR1			x		x	X		
LWR2			x		x			
LWRM		x						
HWR1				x	x			
HWR2				x	x			
HTR						X		x

TABLE 4.9. LIST OF NUCLEAR ENERGY SYSTEM OPTIONS BASED ON THERMAL AND FAST REACTORS CONSIDERED IN THE HIGH CASE SCENARIO

Reactors	Titles of the options considered						
	SFR	FRTh/LWR2	FRTh/HWR2	FRTh/LWR12/HWR12	FRTh/HTR	SFR12	FRTh/LWR3
HWR	x	x	x	x	x	x	x
LWR	x	x	x	x	x	x	x
ALWR	x	x	x	x	x	x	x
LWR1				x			
LWR2		x		x			
HWR1				x			
HWR2			x	x			
HTR						x	
FR	x	x	x	x	x		x
FRTh		x	x	x	x		x
FR12						x	
LWR3							x

In the following sections, results of nuclear energy systems in the high case scenario are compared either for a specific non-geographical group of countries (see definition of groups in Section 4.3) or on a global basis.

4.6. COMPARISON OF THREE OPTIONS FOR THE HIGH CASE SCENARIO IN THE G3 GROUP

The G3 group consists of countries that are to incorporate nuclear energy as newcomers and to install only thermal reactors. The high case scenario requires that, at the end of the 21st century, the installed nuclear capacity in these countries will reach 1000 MW(e) (see Table 4.1). Three different NES were selected for comparison, i.e. the BAU, SLWR0 and the LWR0/HTR (see Table 4.8).

According to the approach adopted in GAINS, which was also chosen in this report, the reference NES and its fuel cycle to compare with is the BAU option. The nuclear energy system in the BAU option consists of conventional thermal reactors (LWR, ALWR and HWR) using enriched and natural uranium in a once-through fuel cycle.

The nuclear energy system of the SLWR0 option includes the same types of reactors as the BAU option and adds a reactor type LWR0, i.e. it considers the introduction of thorium in conventional currently operating LWR type reactors. Fresh fuel of the LWR0 reactor consists of uranium enriched up to 20% and thorium. Like the BAU case, the SLWR0 option is an example of a once-through fuel cycle, i.e. it does not envisage reprocessing of the spent fuel.

The third option, LWR0/HTR, comprises all reactors of the SLWR0 option and additionally HTR type reactors to use ^{233}U produced in thorium fuel. Unlike the first two options, it requires spent fuel reprocessing from both reactor types LWR0 and HTR to recycle ^{233}U. Plutonium recycling is not envisaged in all three options (BAU, SLWR0, and LWR0/HTR).

This nuclear generation structure, i.e. the development of the share of each reactor type in the nuclear energy system, for the BAU, SLWR0 and LWR0/HTR cases is shown in Figs 4.11, 4.12 and 4.13. The structure is the result of drivers and constraints defined in the MESSAGE model. The main constraint is the availability of necessary nuclear material in a fuel cycle, i.e. primarily ^{233}U, but also natural and enriched uranium, and thorium. The main driver is the goal to maximize the introduction of thorium in nuclear reactors (in reactors that use thorium). As a
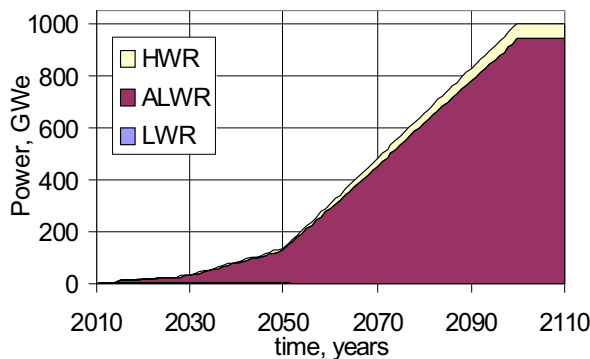
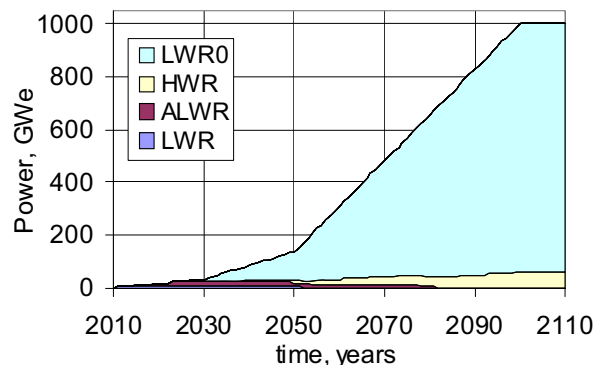

FIG. 4.11. Nuclear power structure. BAU, high demand, G3.

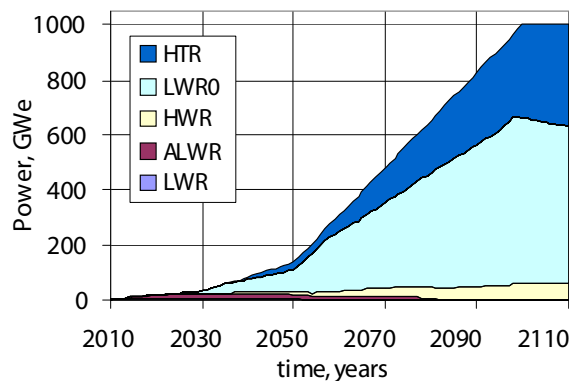FIG. 4.12. Nuclear power structure. SLWR0, high demand, G3.

FIG. 4.13. Nuclear power structure. LWR0/HTR, high demand, G3.

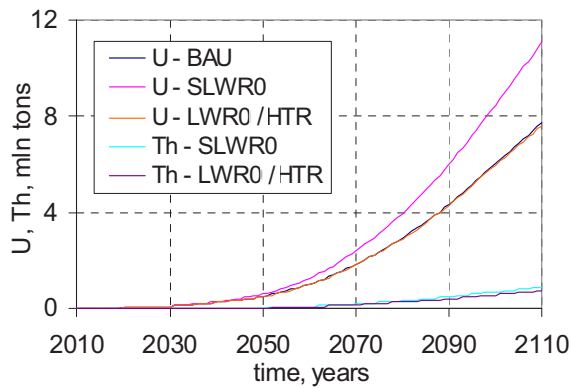**Note:** 'High demand' means that the high case scenario is applied.

FIG. 4.14. Cumulative natural uranium, thorium requirements. BAU, SLWR0 and LWR0/HTR. High demand, G3.
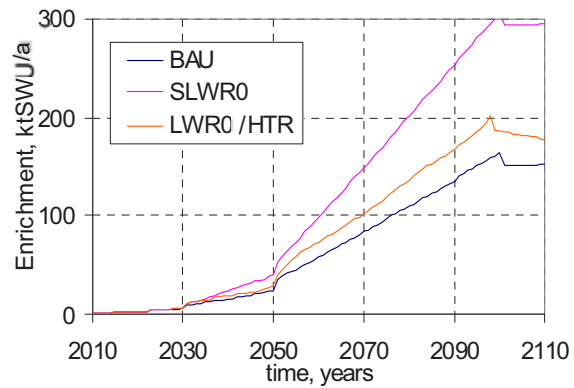
FIG. 4.15. Enrichment requirements. BAU, SLWR0 and LWR0/HTR. High demand, G3.

result of the defined drivers and constraints, by 2100 in both the SLWR0 and LWR0/HTR case, the share of thorium based nuclear power plants reaches ~94% of total nuclear generation capacity of 1000 GW(e).

Cumulative requirements of natural uranium and thorium for all three nuclear energy system options are presented in Fig. 4.14. By the end of the century, the total mass of consumed natural uranium would reach approximately 6 million tonnes for BAU and LWR0/HTR cases. In the SLWR0 case, uranium consumption rises to ~8 million tonnes, i.e. it becomes 25% higher than in the BAU and the LWR0/HTR scenarios. This can be explained by a higher specific uranium consumption of the reactor LWR0 compared with the other reactors (see Fig. 4.5).

Figure 4.15 presents the result of the annual separation work requirements for uranium enrichment. By 2100, the separation work necessary for both advanced nuclear energy system options, i.e. LWR0/HTR and SLWR0, would be ~30% and ~100% higher than in the referent BAU scenario. The reason for this significant increase is the high specific SWU requirement of the LWR0 reactor type (see Fig. 4.5) caused by the required high enrichment (20%) of the uranium part of fuel in its core and by the comparatively short fuel residence time and corresponding not very high level of burnup.

Annual fuel fabrication requirements for all three nuclear energy system options are presented in Figs 4.16, 4.17 and 4.18. The SLWR0 case (Fig. 4.17) shows the highest requirement for fresh fuel production of all three cases. This is because the specific (per energy unit produced) mass of fuel consumed in a LWR0 reactor type and corresponding fabrication requirements — due to the lower average burnup — are higher than for a advanced ALWR reactor type. The HTR reactor in the LWR0/HTR case reduces the amount of necessary fresh fuel to almost the same level as for the BAU case.

The mass of spent fuel to be reprocessed annually in the LWR0/HTR case by reactor types is shown in Fig. 4.19. Since ^{233}U is the only fissile material recycled in this scenario, 100% of total reprocessing activities are performed on spent fuel containing thorium (i.e. fuel from the LWR0 and HTR reactor). The mass of spent fuel to be reprocessed annually by 2100 is 20 kt HM and the share of heavy metal of HTR spent fuel is ~14% of the total annual heavy metal mass; the rest (~86%) is discharged from the LWR0 reactor.

Figures 4.20, 4.21 and 4.22 show the accumulated amount of spent fuel for every reactor type in the three nuclear energy system options under consideration. Comparing the two nuclear energy system options with a once-through fuel cycle, i.e. the BAU and SLWR0 case, it is noted that a total amount of 826 kt HM and 1142 kt HM, respectively, of spent nuclear fuel has been accumulated by the end of the century. Thus, the latter option, SLWR0, shows no advantage over BAU. Faster accumulation of spent fuel in the SLWR0 case corresponds to the specific data of the LWR0 reactor regarding spent fuel accumulation (shown in Fig. 4.6). The thorium fuel reprocessing based option LWR0/HTR would accumulate only 441 kt HM of spent fuel utilizing almost all available ^{233}U in the HTR.

The amount of plutonium and minor actinides discharged is shown in Figs 4.23 and 4.24. Both results are in full concordance with the reactor data in Fig. 4.7, bearing in mind that there is no recycling of plutonium in these three options considered. The lowest masses of discharged plutonium and minor actinides are attained in the LWR0/HTR scenario, and the highest ones in the BAU case. Compared to the BAU case, the annual discharge of plutonium drops by a factor of ~1.9 for the SLWR0 case, and by a factor of ~2.5 for the LWR0/HTR case. Annual minor actinide discharge of the two advanced vases in comparison to BAU drops by factors of ~1.9 and ~2.8, respectively.
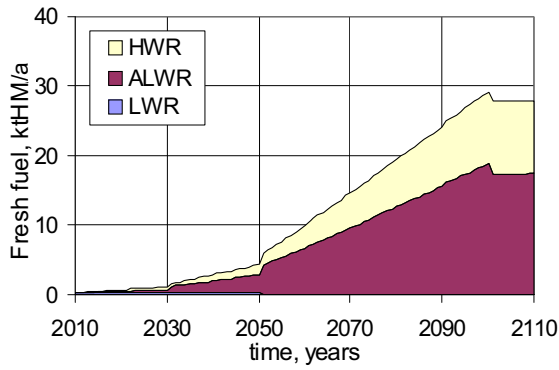
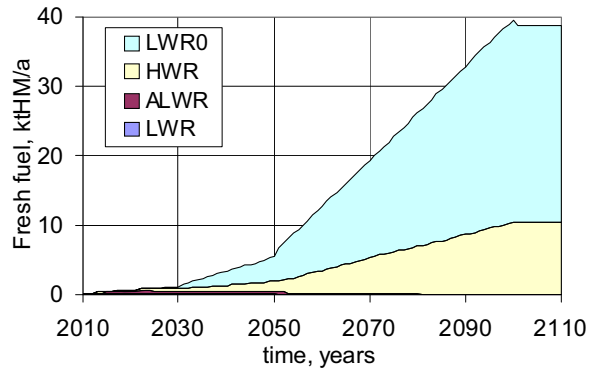FIG. 4.16. Annual fresh fuel requirements, BAU, high demand, G3.

FIG. 4.17. Annual fresh fuel requirements, SLWR0, high demand, G3.
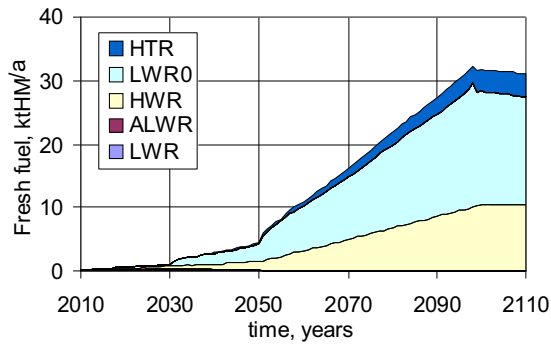

FIG. 4.18. Fresh fuel requirements, LWR0/HTR, high demand, G3.
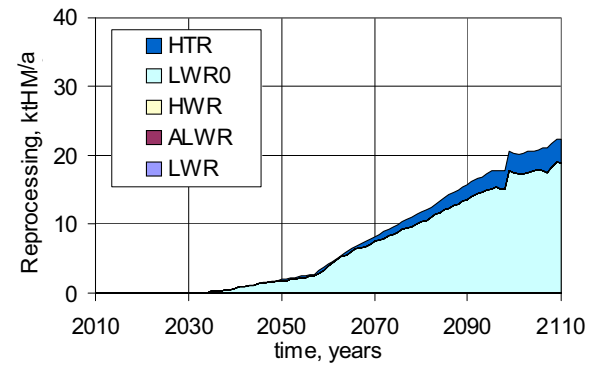

Fig. 4.19. Reprocessing requirements, LWR0/HTR, high demand, G3.
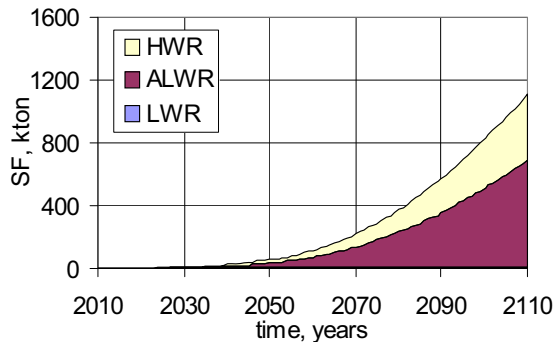

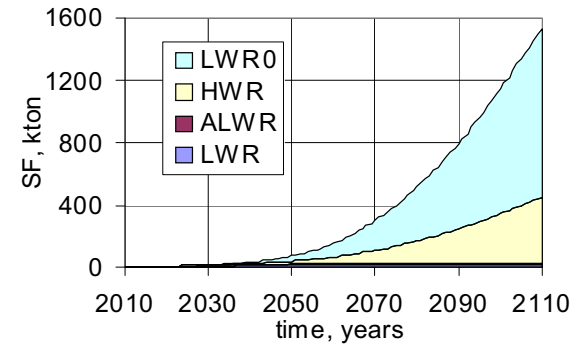FIG. 4.20. SF storage. BAU, high demand, G3.

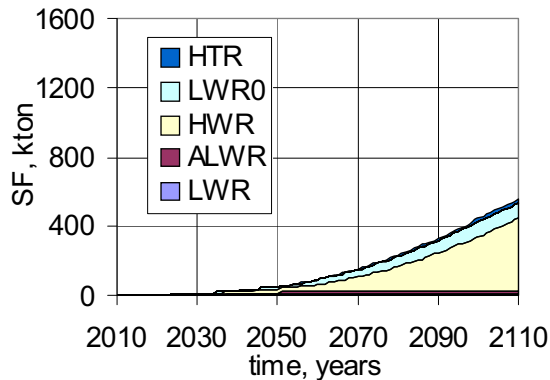FIG. 4.21. SF storage. SLWR0, high demand, G3.
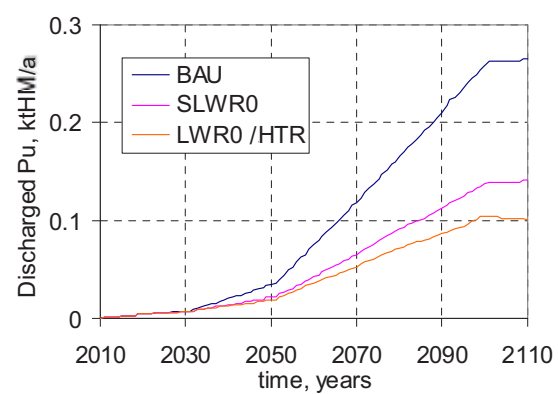

FIG. 4.22. SF storages. LWR0/HTR, high demand, G3.

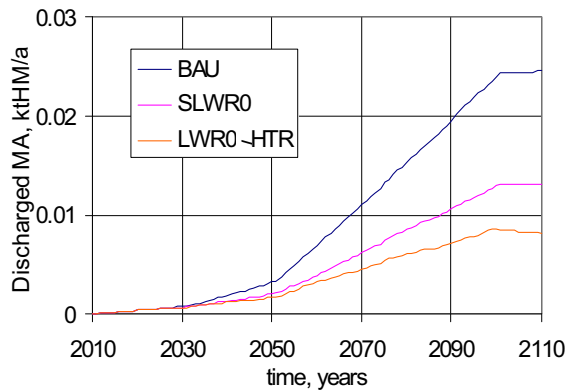FIG. 4.23. plutonium discharge, BAU, SLWR0, LWR0/HTR, high demand, G3.

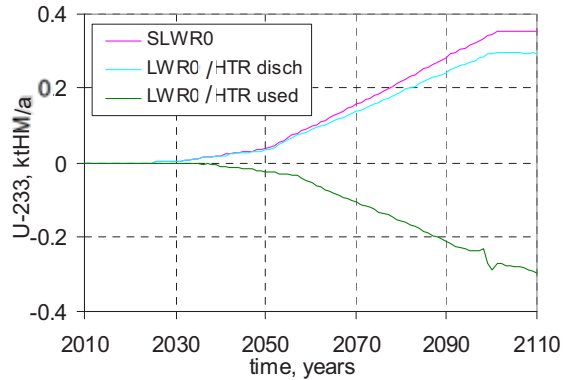FIG. 4.24. Minor actinide discharge, BAU, SLWR0, LWR0/HTR, high demand, G3.

FIG. 4.25. ^{233}U production and consumption, SLWR0, LWRO/HTR, high demand G3..

Note: LWR0/HTRds refers to discharged ^{233}U, and LWR0/HTR used refers to used ^{233}U.

Annual mass of ^{233}U discharged in spent fuel of SLWR0 and LWR0/HTR and used in fresh fuel for LWR0/HTR is presented in Fig. 4.25. In the LWR0/HTR case, due to the cooling period of spent fuel, almost all reprocessed ^{233}U is used in the HTR and annual discharge of ^{233}U is only slightly lower than in the SLWR0 scenario.

Summary of results for the comparison of the BAU, SLWR0 and LWR0/HTR scenarios

Three scenarios have been studied assuming the high case demand of nuclear power for the G3 group of countries (newcomers). Only thermal reactors were included in the nuclear energy systems of the three scenarios.

Adding an innovative LWR ('LWR0') that uses thorium in addition to enriched uranium to a conventional nuclear energy system ('BAU') based on an open fuel UO_2 cycle consisting of water cooled reactors (LWR, ALWR and HWR) worsens the efficiency of such an advanced nuclear energy system ('SLWR0') by significantly increasing the specific (per unit of energy produced) consumption of natural uranium, the need for enrichment and mass of fresh fuel, and mass of spent fuel to be stored. The only advantage of the advanced nuclear energy system (SLWR0) is a reduction of discharged plutonium and minor actinides in the spent fuel.

Adding a reactor that uses Th/^{233}U fuel (HTR) and introducing reprocessing of spent fuel that contains ^{233}U to this advanced nuclear energy system (SLWR0) creates a further advanced nuclear energy system (LWR0/HTR). In this NES, the disadvantages at the front end of the fuel cycle (U_{nat} consumption, fresh fuel and enrichment requirements) of introducing thorium in comparison to a conventional nuclear energy system (BAU) are mostly compensated. There is significant reduction of spent fuel to be put in storage, and plutonium and minor actinides are caused primarily by the introduction of reprocessing into this nuclear energy system.

As stated before, all the results of the three scenarios are caused mainly by the characteristics of the reactors (discussed in Section 4.5) and the development of the share of these reactors included in the three nuclear energy system.

4.7. COMPARISON OF THREE OPTIONS BASED ON THERMAL REACTORS IN THE HIGH CASE GLOBAL SCENARIO

The results presented in this section are for the global high case scenario, i.e. for all three non-geographical groups of countries combined.

Unlike the BAU, SLWR0 and LWR0/HTR cases discussed in Section 4.6 for the G3 group, two scenarios were selected here — one for LWRs with MOX (SLWRM) and one for LWR12/HWR12 (see Tables 4.6 and 4.8), which demonstrate utilization of recycled plutonium in thermal reactors without or with Th/^{233}U fuel together with recycling of ^{233}U for a ThFC.

In addition to conventional LWRs and HWRs, the SLWRM option considers the use of MOX fuel consisting of depleted uranium and plutonium in advanced LWR type reactors.
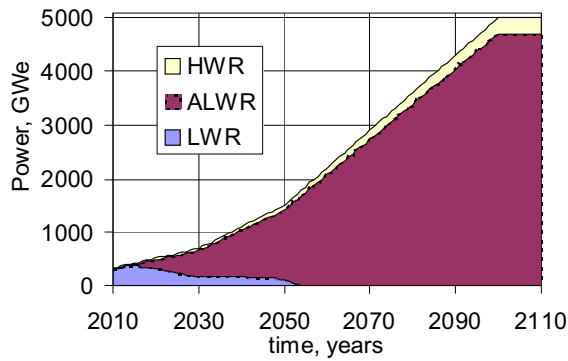
FIG. 4.26. Nuclear power demand structure, BAU, global high case scenario.
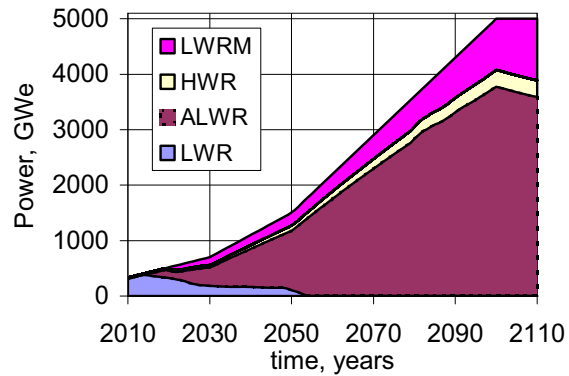

FIG. 4.27. Nuclear power demand structure, SLWRM, global high case scenario.

In addition to conventional LWRs and HWRs, the LWR12/HWR12 option envisages thorium utilization in LWR1 type reactors using plutonium/thorium fuel and, consequently, utilization of [233]U in LWR2 reactors using Pu/[233]U/depleted uranium fuel. Corresponding reactors of type HWR1 use plutonium/thorium fuel and HWR2 uses Pu/[233]U/Th fuel.

All spent fuel is reprocessed in both the SLWRM and LWR12/HWR12 scenario. In the case of a lack of plutonium produced in the reactors of a given option, it is assumed that the transition to thorium could occur through the incineration of civilian grade plutonium and by achieving a reduction of the existing SF stockpiles. Therefore, the lack of plutonium is not a constraint for the introduction of thorium. In both cases, as stated above, the total share of HWR is kept at a 6% level.

Figure 4.26 shows the trends of power generation for each reactor type in the BAU scenario. ALWRs are introduced in 2015 and gradually replace the LWR.

SLWRM (see Fig. 4.27) is developed as an extension of the BAU case through introducing the recycling of plutonium in thermal reactors (type LWRM). SLWRM is used here mainly to compare the shares of NPPs based on recycled fuel with thorium and plutonium based options and also to distinguish effects of thorium introduction from those caused by reprocessing.

The reactors under consideration in the LWR12/HWR12 scenario that are using a ThFC are also plutonium burners, and use plutonium/thorium fuel and Pu/[233]U/Th fuel. The process of material flow analysis has been divided in two steps. In the first step, the fractions of reactor types were determined in this scenario based on [233]U availability only. The result of this first step (0-point of optimization) is shown in Fig. 4.28.

As shown in this figure, by 2100 about half of the reactors are of the ALWR type, and reactors (using thorium, [233]U and plutonium fuel) of type LWR1 and LWR2 amount to about 45%; the rest are HWR2 reactor types. This preliminary result of thorium introduction in the LWR12/HWR12 scenario was used as a 'zero-point' for the next step of simulation, taking into account also availability of plutonium.

As shown in Fig. 4.29, this adjustment of the 'zero-point' scenario decreases the share of thorium utilizing reactors significantly because of limited resources of reprocessed plutonium. Taking the availability of plutonium into account, the share of thorium based nuclear power plants reaches only ~23% of total nuclear generation by 2100 compared to about 50% shown in Fig. 4.28 (plutonium availability is neglected); the dominant reactor type is clearly the ALWR.

Cumulative natural uranium consumption is one of the major concerns when contemplating fuel cycle options, including those with thorium. Fig. 4.30 shows the cumulative natural uranium demand for BAU, SLWRM and LWR12/HWR12. In the BAU case, total annual natural uranium demand reaches approximately 40 million tonnes by 2100. The situation regarding uranium consumption improves through the introduction of either a MOX fuel cycle (in the SLWRM case), and additionally, a ThFC (in the LWR12/HWR12 case). In both cases, the cumulative consumption of uranium is ~20% lower by the end of century compared to the BAU case.

Figure 4.31 presents the annual separation work needed for uranium enrichment. In comparison to the BAU scenario, the separation work necessary is 24% lower by 2100 for both recycling based thermal options (LWR1/HWR12 and SLWRM).
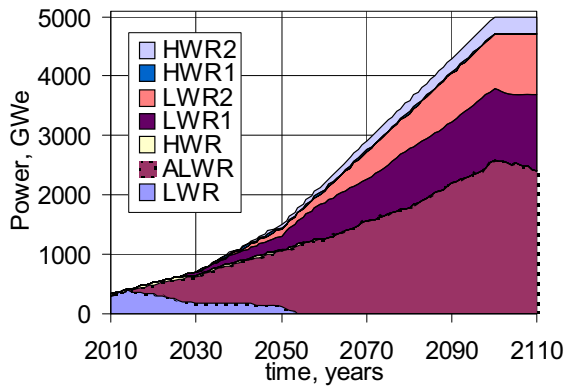
FIG. 4.28. Nuclear power demand structure, LWR12/HWR1.2, 0-point of optimization (only availability of ^{233}U taken into account), global high case scenario.
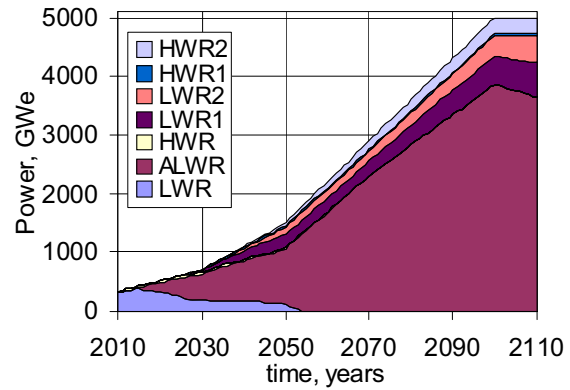
FIG. 4.29. Nuclear power demand structure, LWR12/HWR1.2, global (plutonium and ^{233}U availability taken into account).
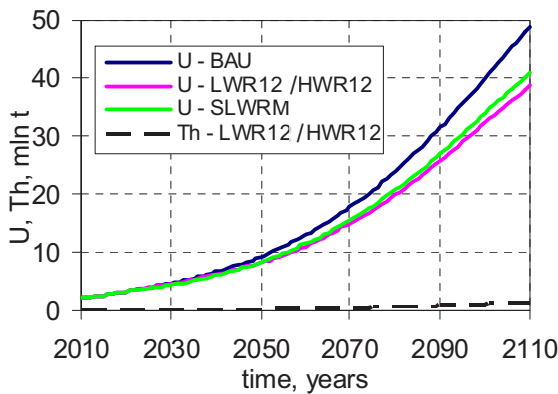
FIG. 4.30. Cumulative natural uranium and thorium demand, BAU, SLWRM, LWR12/HWR12, global high case scenario.
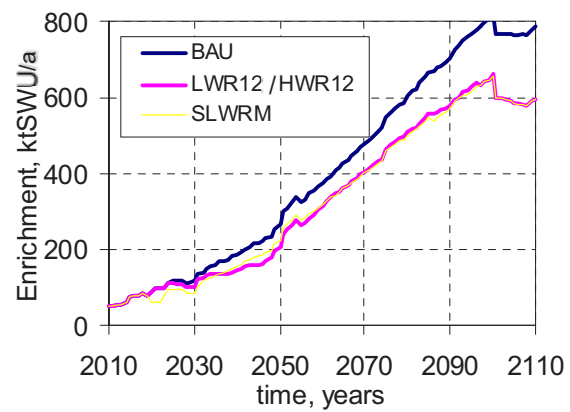
FIG. 4.31. Annual enrichment requirements, BAU, SLWRM, LWR12/HWR12, global high case scenario.
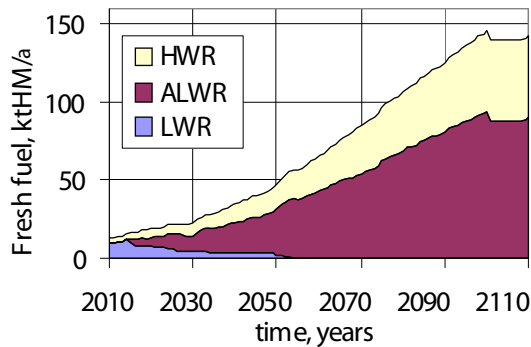
FIG. 4.32. Fresh fuel requirements, BAU case, global high case scenario.
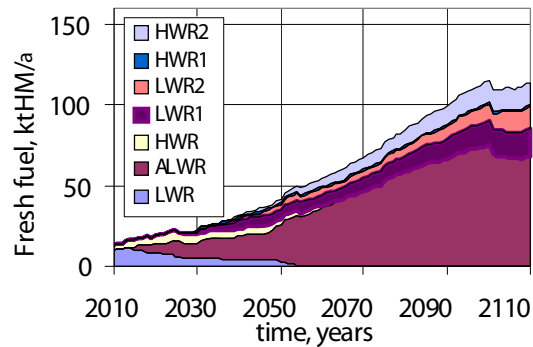
FIG. 4.33. Fresh fuel requirements, LWR12/HWR12 case, global high case scenario.

Annual requirements for the fuel fabrication in different reactors in the BAU scenario are shown in Fig. 4.32. The reactors of LWR type (LWR+ALWR) need ~94 kt HM/a of fabricated fuel and the HWRs need ~52 kt HM/a by 2100. (The calculation of fresh fuel parameters for the SLWRM option gives roughly the same numbers.)

In the case of the thorium utilizing option LWR1/HWR12, the fuel fabrication facilities should provide ~74 kt HM/a for ALWRs and ~42 kt HM/a for thorium based reactors by 2100 (see Fig. 4.33), which is about 30 kt HM/a less than for the BAU case (shown in Fig. 4.32). Fuel fabrication capacities for ThFC reactors reach ~36% of total capacity.
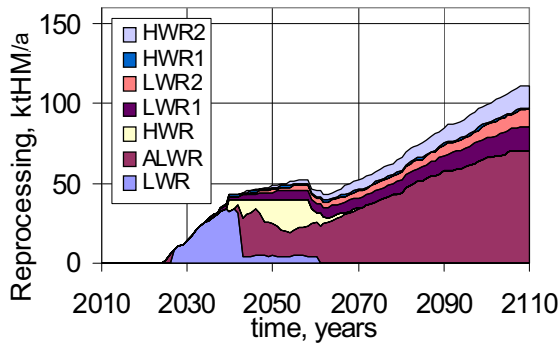
FIG. 4.34. Reprocessing requirements, LWR12/HWR12 case.
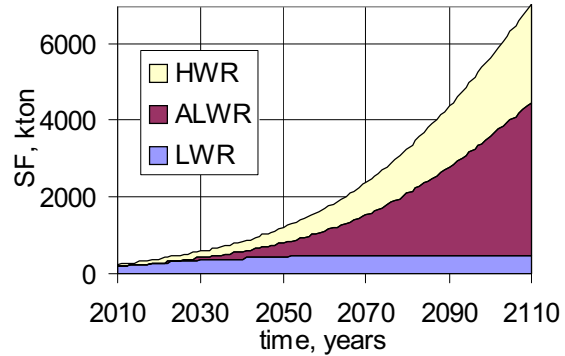
FIG. 4.35. SF storage requirement, BAU.
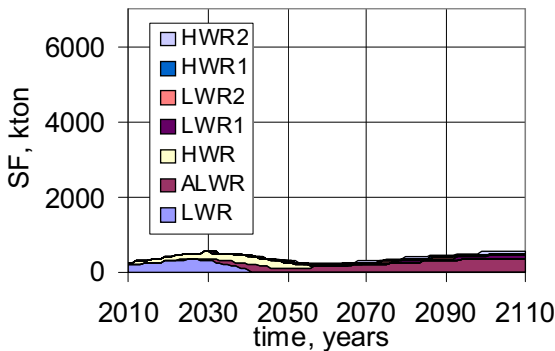
FIG. 4.36. SF storage requirement, LWR12/HWR12 case.
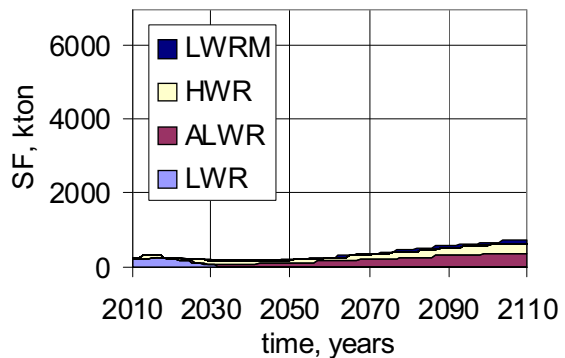
FIG. 4.37. SF storage requirement, SLWRM case.

In the LWR1/HWR12 and SLWRM options, spent fuel unloaded from all reactor types — including HWRs — is to be reprocessed. Reprocessing required by different reactor types in the thorium applying case LWR1/HWR12 is presented in Fig. 4.34. By 2100, the annual requirement for reprocessing of the spent fuel from reactors using Th/^{233}U fuel together with other fuel types reaches 36 kt HM/a, i.e. approximately 34% of total reprocessing capacity. This means that thorium reprocessing related infrastructure, including a threefold process of separation of uranium, plutonium and thorium from spent (Th, Pu)O$_2$ fuel, should be developed and deployed at a similar level to that for the uranium–plutonium case.

The total amount of spent fuel accumulated by 2100 in the BAU scenario reaches 5.8 million tonnes of HM (see Fig. 4.35). This number corresponds to a capacity of more than 80 Yucca Mountain repositories (reported designed capacity of Yucca Mountain — 70 000 tonnes) to be built all over the world in the case of realization of the BAU high case scenario.

This problem becomes greater in relation to the plutonium accumulated, as considered below. Introduction of Th/^{233}U fuel and continuous recycling of ^{233}U and plutonium as defined for the LWR1/HWR12 case (see Fig. 4.36) causes a significant reduction in spent fuel accumulation from ~5800 kt for the BAU case (Fig. 4.35) to ~560 kt (mainly because of reprocessing). Although the waste (HLW) treatment would still be an issue, the fissile isotopes (plutonium and ^{233}U) would be used in reactors.

SLWRM (see Fig. 4.37) shows a similar effect of reducing accumulation of spent fuel in comparison to BAU (presented in Fig 4.35) with the exception of ^{233}U recycling.

The amount of plutonium and minor actinides to be discharged in all three options is shown in Figs 4.38 and 4.39. The lowest amount of plutonium and minor actinides is discharged in the BAU case, although all of this plutonium would have to be accumulated in storage facilities or depositories, since it is not used in a reactor. There is no plutonium accumulation in the SLWRM and LWR12/HWR12 cases because plutonium consumption corresponds to its production with only a short delay. The amount of plutonium handled continuously in the fuel cycle is lower for the LWR12/HWR12 case than for SLWRM; by 2100, the consumption (use) of plutonium reaches 1.5 kt/a in the LWR12/HWR12 case and 2.2 for the SLWRM. Annual discharge of minor actinides in the LWR12/HWR12 scenario rises by 25% compared to the BAU scenario (from 0.12 to 0.15 kt/a).
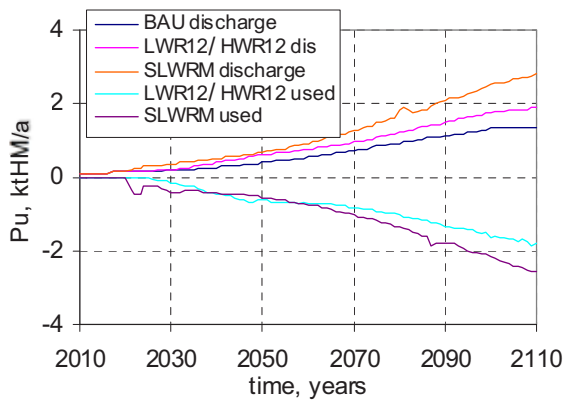
FIG. 4.38. Plutonium discharged/used annually, BAU, SLWRM, LWR12/HWR12.
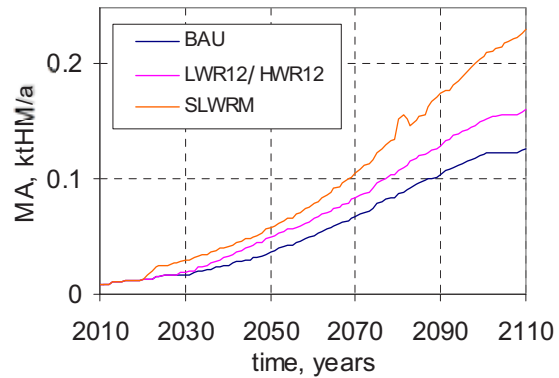
FIG. 4.39. Minor actinide annual discharge, BAU, SLWRM, LWR12/HWR12.
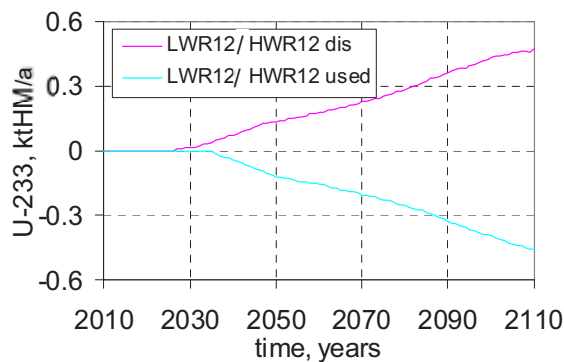
FIG. 4.40. U233 discharged and used, LWR12/HWR12.

According to the boundary conditions of the scenarios, all minor actinides are kept in spent fuel waste without recycling (see Fig. 4.39). The results of minor actinide production in the three cases are similar to the results for plutonium production.

^{233}U balance (discharged and used) for the LWR12/HWR12 option is shown in Fig. 4.40. Almost all reprocessed ^{233}U is used with a short delay (five years), and by 2100, annual discharge of ^{233}U is ~0.43 kt/a.

An attempt to consider ThFC economic issues is presented in Section 5. The results of such consideration have been applied to the LWR12/HWR12 and SLWRM options to get economically reasonable options for comparison. For this purpose, standard algorithms provided by the MESSAGE tool (see Section 4.2) were used. MESSAGE provides the opportunity to optimize an NES by minimizing the total system cost subject to constraints such as those concerning the satisfaction of demand, the capacity of facilities, the balances of energy flow, the availability of resources, and the bounds on capacity and activity.

An optimized structure of nuclear generation was obtained on the basis of input data for availability and pricing of resources provided in Tables 4.3 and 4.4, and for reactors and fuel cycles provided in Section 5. The resulting optimized structures of global nuclear energy systems for LWR12/HWR12 and SLWRM scenarios are shown in Figs 4.41 and 4.42.

Thorium based HWRs produce electricity cheaper than LWRs using thorium; the cheapest thorium reactor is HWR1 that is admitted to be commissioned starting from 2025 and becomes competitive with ALWR when cheap uranium of grade 'a' ($40/kg) is exhausted by 2030. LWR1 reactors could be commissioned starting from 2065 when uranium of grade 'c' ($130/kg) is exhausted by 2072. HWR2 and LWR2 are commissioned in 2070 and 2080, respectively. Spent fuel of an LWR1 reactor type has the highest concentration of ^{233}U and produces the main amount of spent fuel to be reprocessed. By 2100, the share of NPPs using Th/^{233}U could reach 15% of total nuclear generation capacity.
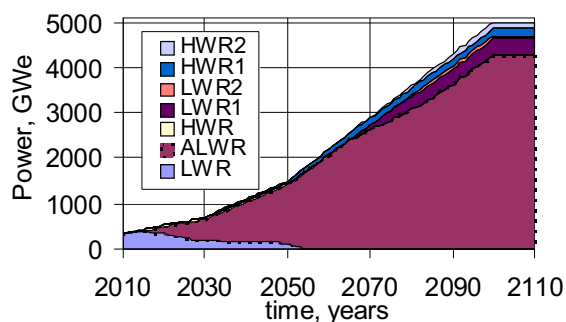
FIG. 4.41. Optimized nuclear power demand structure, LWR12/HWR12, global high case scenario.
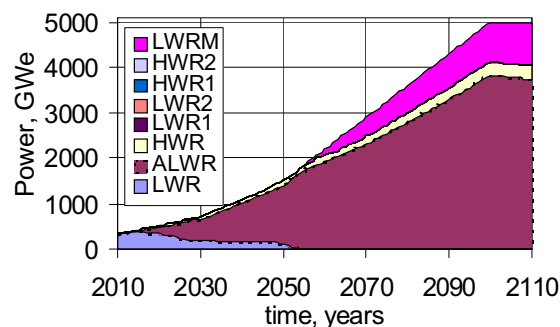
FIG. 4.42. Optimized structure of nuclear power demand, SLWRM, global high case scenario.

Summary of results for the comparison of BAU, SLWRM and LWR12/HWR12 scenarios

Three global (all three groups of countries combined) scenarios have been studied, assuming the high case demand of nuclear power in the 21st century. As in the previous section, only thermal reactors were included into the nuclear energy systems of these three scenarios. The reference scenario BAU includes water cooled reactors using only UO_2 fuel in an open fuel cycle.

The two advanced nuclear energy systems ('SLWRM' and 'LWR12/HWR12') are based on a closed fuel cycle reprocessing all spent fuel and include reactors with Th/Pu, Pu/U and Pu/Th/^{233}U fuel.

In comparison to the BAU case, the advanced nuclear energy systems demonstrate a reduction of U_{nat} consumption, enrichment and fresh fuel mass requirements. The strong reduction of spent fuel storage requirements for the advanced NES is a consequence of complete reprocessing of all spent fuel.

In this section, additional constraints were taken into account for developing a scenario with advanced nuclear energy systems. The first additional constraint was the availability of plutonium from spent fuel only (i.e. no outside source of plutonium). The second constraint was using economic considerations, i.e. choosing the more economic reactor over its competitors. Taking into account these constraints of scenario development had a significant effect on the structure of nuclear plants installed in the 21st century: the number of reactors that use very advanced types of fuel such as Th/Pu and Pu/Th/^{233}U in a closed fuel cycle was remarkably reduced and replaced by LWR with a low enriched uranium fuel.

4.8. COMPARISON OF OPTIONS USING FAST REACTORS IN THE HIGH CASE GLOBAL SCENARIO

In this section, two NES options that include fast reactors, namely SFR and FRTh/LWR12/HWR12 (see Tables 4.7 and 4.9) were evaluated in comparison to the BAU case.

The SFR option illustrates a possible scenario of introduction of fast reactors operating with a uranium–plutonium closed fuel cycle in thermal LWRs and HWRs.

The FRTh/LWR12/HWR12 option comprises uranium–plutonium fast reactors, fast reactors with thorium blankets, LWRs and HWRs both using thorium, uranium and plutonium. This option is an extension of the SFR option by including ThFCs and also an extension of the LWR12/HWR12 option (discussed in Section 4.7) by adding fast reactors.

All spent fuel is allowed to be reprocessed in both the SFR and FRTh/LWR12/HWR12 scenarios with the exception of spent fuel from HWR type reactors, which is not reprocessed in the SFR scenario according to the GAINS' approach.

Similar to the LWR12/HWR12 option, in the case of lack of plutonium produced in the reactors of a given option, it is assumed that the transition to thorium could be effected through the incineration of civilian grade plutonium and achieving a reduction of existing SF stockpiles. As mentioned before, the share of HWRs is kept at a 6% level of total thermal reactor electricity production.

The development of the nuclear generation structure for the SFR case is shown in Fig. 4.43. By 2100, the share of fast reactors can reach ~47% of global nuclear energy production. Figure 4.44 presents the structure of the nuclear energy systems of the FRTh/LWR12/HWR12 scenario optimized only according to ^{233}U availability ('zero-point' as it was introduced in Section 4.7).
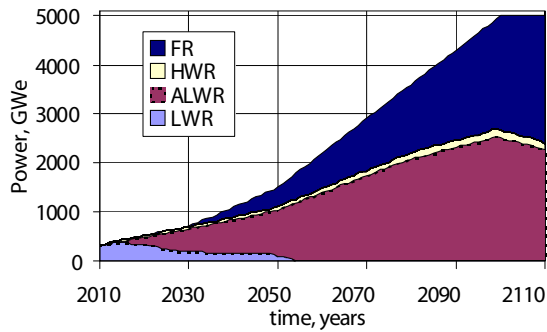
FIG. 4.43. Nuclear power demand structure, SFR.
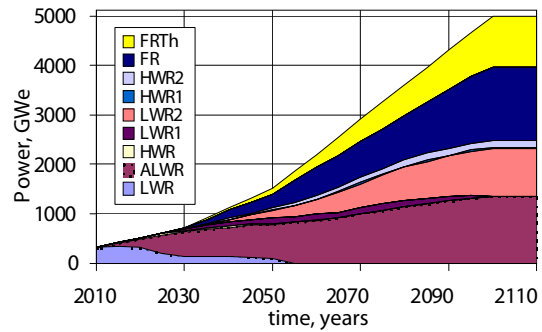
FIG. 4.44. Nuclear power demand structure, FRTh/LWR12/ HWR12, 0-point of optimization.
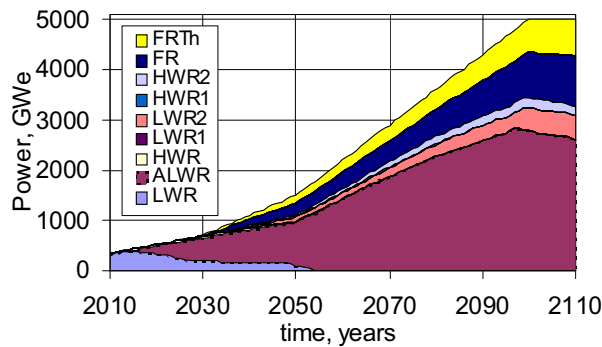
FIG. 4.45. Nuclear power demand structure, FRTh/LWR12/HWR12, taking availability of plutonium and ^{233}U into account.
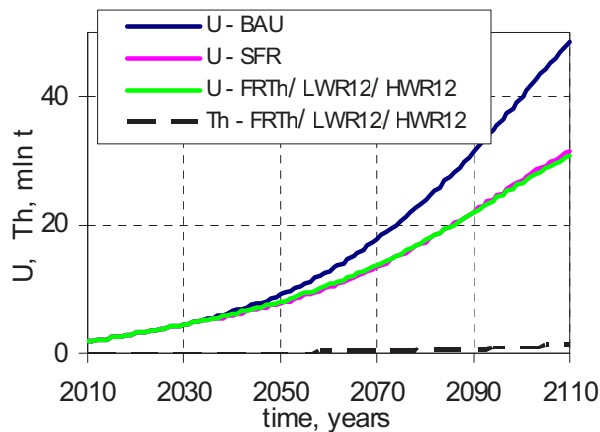
FIG. 4.46. Cumulative natural uranium and thorium demand.
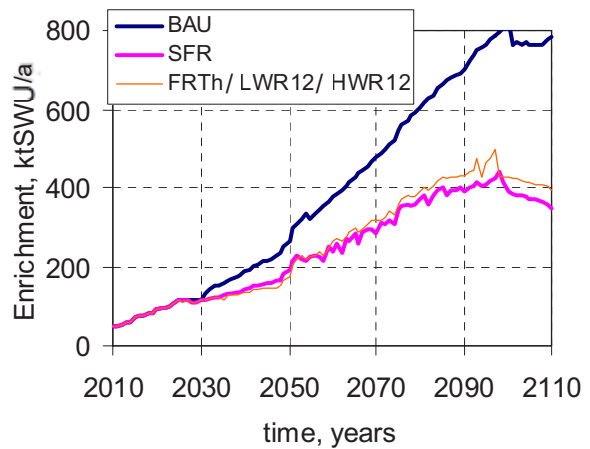
FIG. 4.47. Annual enrichment requirements.

The adjustment to take into account that the availability of plutonium decreases the share of thorium utilizing reactors from 47% to 27% by 2100 is shown in Fig. 4.45. The remaining share is divided approximately 50/50 between fast and thermal reactors. The adjustment was performed only for material flows to ensure availability of necessary nuclear material (plutonium in addition to ^{233}U) in the fuel cycle, but economic parameters were not considered.

Figure 4.46 demonstrates cumulative natural uranium consumption for BAU, SFR and FRTh/LWR12/HWR12 cases. At the end of the century, the cumulative consumption of uranium drops from ~40 million tonnes in the BAU case to ~27 million tonnes in the SFR and FRTh/LWR12/HWR12 cases. Transition to NES comprising fast reactors, either with thorium or without it, produces a positive effect not only on uranium consumption, but also on the amount of separation work necessary. The estimated requirements of enrichment capacities for both advanced scenarios are approximately equal and, by 2100, approximately two times lower than the corresponding annual separation work in the BAU case (see Fig. 4.47).
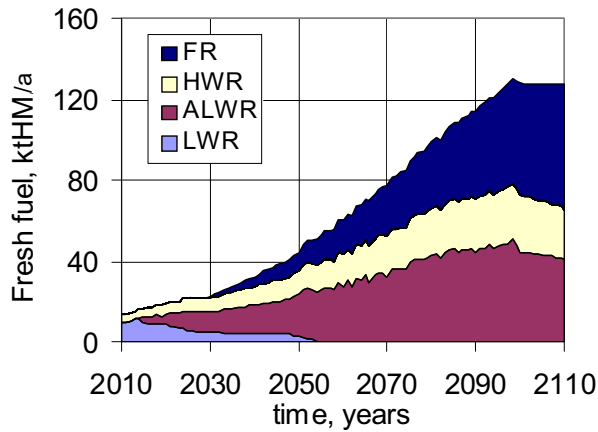
FIG. 4.48. Fresh fuel requirements, SFR case.
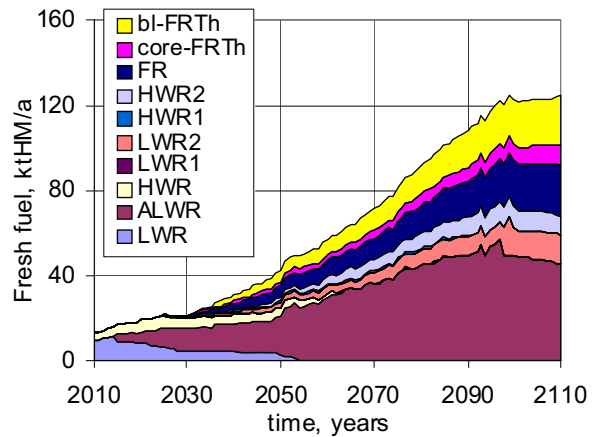
FIG. 4.49. Fresh fuel requirements, FRTh/LWR12/HWR12 case.
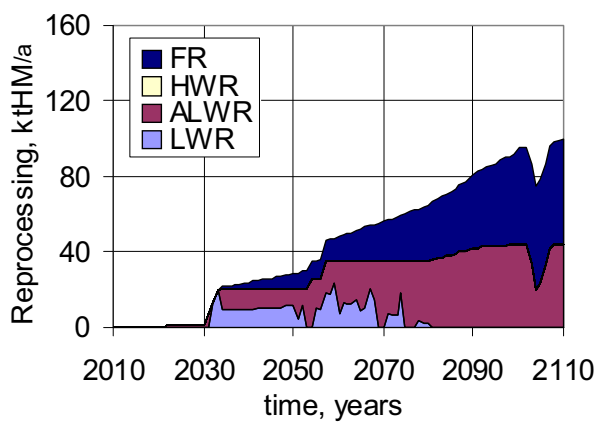
FIG. 4.50. Reprocessing annual requirements, SFR case.
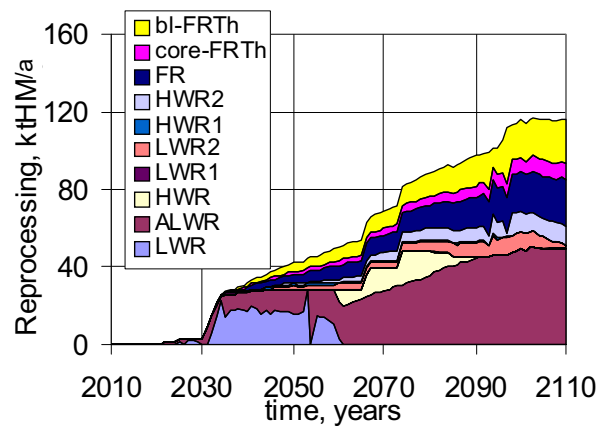
FIG. 4.51. Reprocessing annual requirements, FRTh/LWR12/HWR12 case.

Annual requirements for fuel fabrication in the SFR scenario are shown in Fig. 4.48. The amount of fuel to be fabricated by 2100 drops from 146 kt HM/a (LWRs — 94 ktHM/a and HWRs — 52 kt HM/a) in the BAU case down to 128 kt HM/a (MOX and blanket fuel — 56 kt HM and UOX fuel — 72 kt HM) in the SFR case. The estimation of the total fresh fuel volume for the FRTh/LWR12/HWR12 option gives roughly the same number (see Fig. 4.49). In the case of the thorium utilizing option, the reactors that use Th/^{233}U fuel (together with uranium–plutonium fuel) will consume 42% of fabricated fuel producing only 27% of electricity (see above).

Both advanced scenarios, SFR and FRTh/LWR12/HWR12, envisage large scale deployment of spent fuel reprocessing capacities (see Figs 4.50 and 4.51). By the end of century, annual reprocessing will comprise 94 kt HM/a according to the FR0 scenario, and 116 kt HM/a according to the FRTh/LWR12/HWR12 option. In the first case, the share of reprocessed spent fuel from fast reactors reaches 54% (plutonium assumed to be 'cheap' reprocessing) and in the second case, the share of reprocessed spent fuel from Th/^{233}U utilizing reactors is 40% (^{233}U assumed to be more expensive reprocessing).

Owing to large reprocessing activities by the end of the century, the total cumulative amount of spent nuclear fuel to be stored drops from 5.8 million tonnes for the BAU case to 1.7 million tonnes for the SFR case (herein spent fuel from HWRs is 80%) as is presented in Fig. 4.52. The FRTh/LWR12/HWR12 scenario requires reprocessing of all spent fuel (including that from HWRs, unlike in the SFR case) and only 0.5 million tonnes would have to be stored by 2100 because of the need to decrease residual heating. Although the waste treatment would be still an issue, the fissile isotopes (plutonium and ^{233}U) in spent fuel would be used in reactors.

A comparison of the annually discharged plutonium for all three options is provided in Fig. 4.54. In the SFR option, all plutonium unloaded from reactors — save HWRs — will be used immediately after reprocessing. In the FRTh/LWR12/HWR12 option, all plutonium — including that from HWRs — will also be recycled without
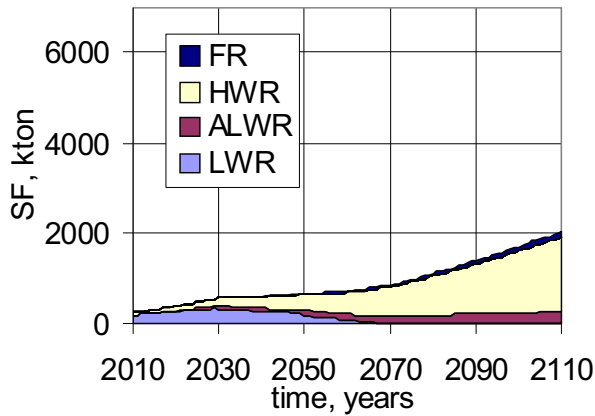
FIG. 4.52. SF storages, SFR case.
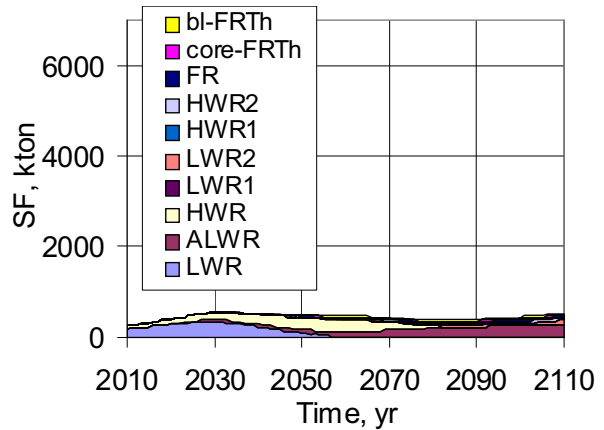
FIG. 4.53. SF storages, FRTh/LWR12/HWR12 case.
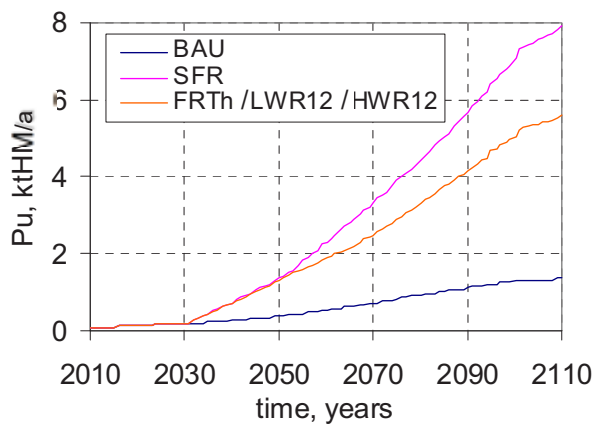
FIG. 4.54. Plutonium discharged annually.
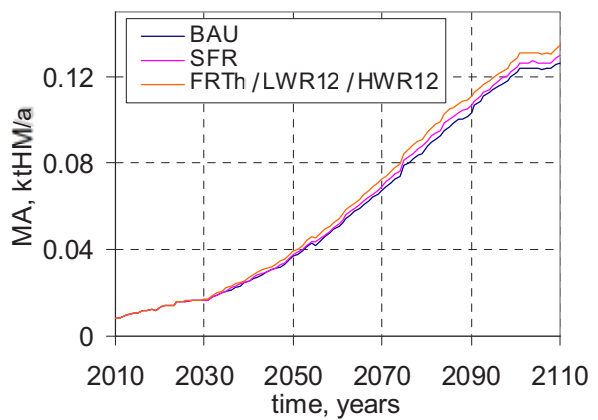
FIG. 4.55. Minor actinides discharged annually.
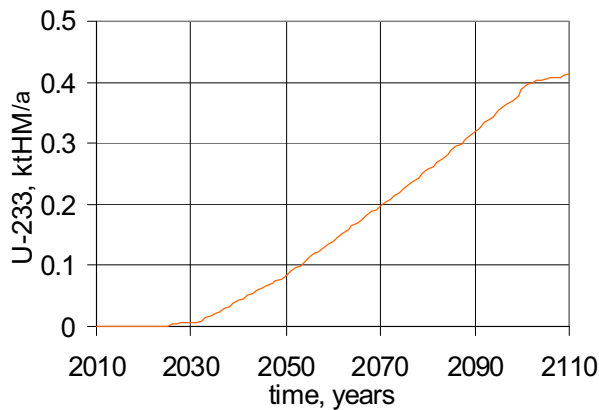
FIG. 4.56. [233]U discharged annually. FRTh/LWR12/HWR12 case.

accumulation. By 2100, the discharge of plutonium in the FRTh/LWR12/HWR12 case is lower than in the SFR case by a factor of 1.4, and the corresponding amount of plutonium handled in the fuel cycle is also lower.

Data on annual discharge of minor actinides are shown in Fig. 4.55. In all three scenarios, they are approximately identical (0.12 kt/a by 2100). According to the scenario conditions, all minor actinides are kept in spent fuel without recycling.

[233]U discharge for the FRTh/LWR12/HWR12 scenario (see Fig. 4.56) is very close to the one presented in Fig. 4.40 for the LWR12/HWR12 case. In a similar way, all reprocessed [233]U is used and, by 2100, the annual discharge of [233]U reaches ~0.40 kt/a.
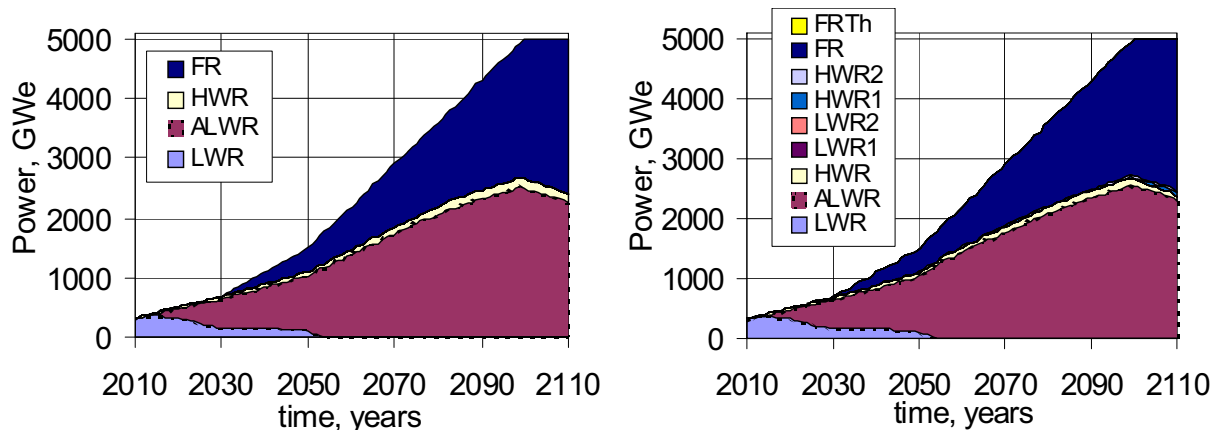
FIG. 4.57. Optimized nuclear power demand structure taking economics into account, SFR, high case.

FIG. 4.58. Optimized nuclear power demand structure taking economics into account, FRTh/LWR12/HWR12, high case.

Economic parameters have been optimized in the same way as for the LWR12/HWR12 and SLWRM options at the end of Section 4.7. Using the method presented in Section 5 for the SFR and FRTh/LWR12/HWR12 scenarios to achieve economically reasonable options produced the results shown in Figs 4.57 and 4.58. The standard algorithms of the MESSAGE tool were used. Using the economic considerations presented in Section 5, the cost of electricity provided by the thorium utilizing reactors considered in the FRTh/LWR12/HWR12 scenario would not be competitive compared to ALWR and fast reactor electricity cost and, consequently, the thorium option would be completely eliminated from the light water and fast reactor market domain. However, ThFC utilizing reactors may become a significant or even dominant constituent of the heavy water reactor domain.

Summary of results for the comparison of BAU, SFR and FRTh/LWR12/HWR12 scenario

Three global (i.e. all three groups of countries combined) scenarios have been studied, assuming the high case demand for nuclear power in the 21st century. In the advanced scenario, SFRs, in addition to conventional thermal reactors (the BAU scenario), a fast reactor with uranium–plutonium fuel (but without thorium) has been included in the nuclear energy system. In the case of the advanced FRTh/LWR12/HWR12 scenario, a fast reactor with a thorium blanket was added to the conventional NES and, additionally, the same types of advanced thermal reactors (using Pu/^{233}U/Th fuel) as studied in Section 4.7 that use the ^{233}U produced in the blanket of the fast reactor.

Adding fast reactors and reprocessing into a nuclear energy system significantly reduces (>40%) the consumption of natural uranium and the need for enrichment in comparison to the reference scenario BAU. Also, the amount (mass) of spent fuel to be put in storage is reduced by a factor of 3 and 10, respectively, for the nuclear energy system including a fast reactor (FR) with a depleted uranium blanket and for the NES including a fast reactor (FRTh) with a thorium blanket plus advanced thermal reactors.

If the availability of plutonium produced in the nuclear energy system is taken into account in the development of the structure (share of reactor types) of the nuclear energy system until the end of the century, the number of fast reactors that can be installed is significantly (by about 50%) reduced. Taking into account also economic criteria results in a complete elimination of fast and thermal reactors that use thorium or ^{233}U, leading to a mixture of fast and thermal reactors using the uranium–plutonium fuel cycle.

4.9. COMPARISON OF URANIUM CONSUMPTION IN VARIOUS OPTIONS IN THE HIGH CASE SCENARIO IN THE G1B GROUP

The concern about uranium shortages in the future and the resulting need for an extension of available nuclear fuel resources are among the main driving forces for the debate on the introduction of a ThFC. However, in the scenario study in Sections 4.6–4.8, the results reveal only minor advantages in uranium consumption in thorium utilizing scenarios compared to corresponding innovations in the uranium–plutonium route.
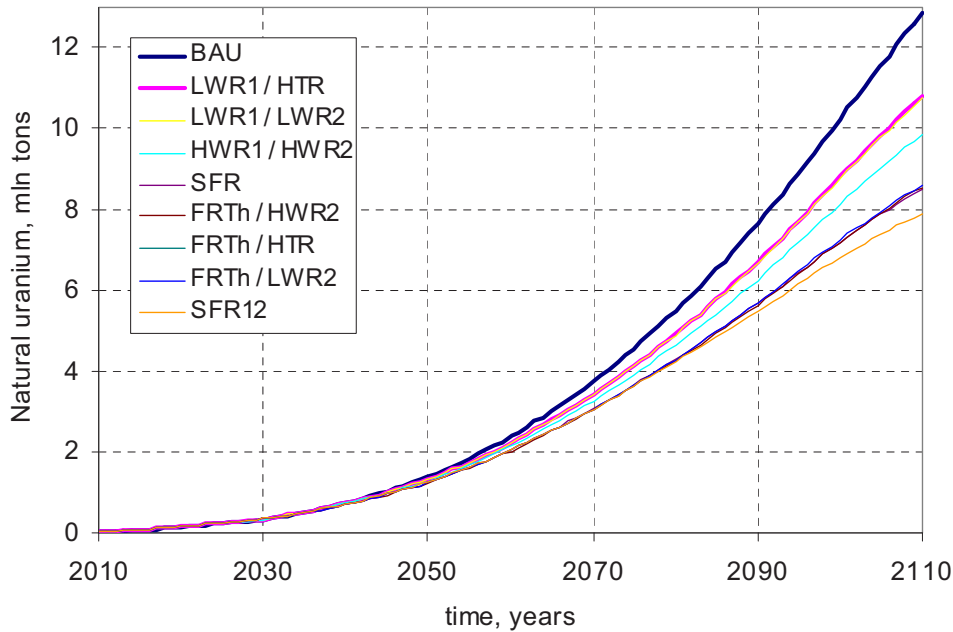
FIG. 4.59. Cumulative natural uranium requirements, G1b,

TABLE 4.10. CUMULATIVE URANIUM CONSUMPTION BY 2100

	Scenarios without FR				Scenarios with FR			
	BAU	LWR1/HTR	LWR1/LWR2	HWR1/HWR2	SFR	FRTh/LWR2	FRTh/HWR2	FRTh/HTR	SFR12
Consumption of natural U, million tonnes	10.2	8.8	8.8	8.1	7.2	7.2	7.2	7.1	6.8

Therefore, in addition to the scenarios discussed in the other sections (including those in the annexes), several additional options are evaluated in this section searching for the option with least uranium consumption: LWR1/HTR, LWR1/LWR2, HWR1/HWR2, FRTh/HWR2, FRTh/HTR, FRTh/LWR2 and SFR12 (see Tables 4.8 and 4.9).

The group of countries that are most appropriate for such considerations is the G1b group: countries supposed to be candidates for the introduction of fast reactors and closed fuel cycle with a high growth rate of nuclear power in the 21st century (see Section 4.2 and Fig. 4.1(a)).

Figure 4.59 shows the calculated cumulative natural uranium consumption for the selected various reactor combinations in the G1b group of countries. BAU and SFR cases are added for comparison.

The data on cumulative consumption of uranium of the G1b group of countries by 2100 are also presented in Table 4.10, in descending order.

As expected, the BAU option has the greatest uranium consumption and is followed by options with thermal reactors that involve reprocessing, which are followed by options that include fast reactors. The least natural uranium is consumed in the SFR12 scenario, which includes fast breeder reactors with a breeding rate of 1.2 but without a thorium blanket.

In the scenarios with only advanced thermal reactors using Pu/^{233}U/Th fuel, the option with HWR type reactors shows lower consumption of natural uranium than the option with LWR type reactors. In the fast reactor variants, the consumption of uranium depends mainly on the share of LWRs remaining in the scenario; the share of LWRs is the result of balances of plutonium and ^{233}U production and consumption in the NES.

When comparing the SFR and SFR12 options (which both include fast reactors without thorium) to options that include the FRTh type reactor (Table 4.10), it can be concluded that introduction of thorium in blankets of fast reactors does not result in savings of natural uranium in a nuclear energy system.

4.10. CONCLUSIONS AND OUTLOOK

Many reactor types, including LWR, HWR, FR, HTR and MSBR, can use thorium or ^{233}U as a fuel with different efficiency. Twelve scenarios of ThFC introduction were considered in comparison with four other scenarios of a 'traditional' uranium–plutonium fuel cycle (see Tables 4.8 and 4.9).

The scenarios studied and the results achieved do not claim ultimate completeness but, rather, highlight points of concern and should provide an incentive for further considerations and development.

According to the information presented in Chapter 4, the introduction of thorium in a once-through fuel cycle option in advanced LWRs may cause an increase of uranium consumption, and necessary enrichment and fuel manufacturing capacity, and also growth of the amount of spent fuel to be unloaded. The advantage of such a scenario would be limited to a decrease of plutonium and minor actinide content in spent fuel. Nevertheless, further optimization of the thorium–uranium ratio in advanced LWR cores may lead to improvement of once-through scenario results, an issue that may be subject of further consideration.

The introduction of thorium in a once-through fuel cycle of an AHWR may provide more advantages even compared to the advanced LWR (ALWR) and is deemed worthy of detailed further investigation.

In the thermal reactor options with spent fuel reprocessing, the introduction of a Th/^{233}U fuel route achieves approximately the same benefits — by decreasing uranium consumption, enrichment requirements, fuel manufacturing and spent fuel accumulation — as the customary uranium–plutonium MOX route. In the thorium case, the fuel manufacturing requirements for thorium utilizing reactors rise disproportionally to the contribution in electricity generation, implying that a ThFC related infrastructure should be developed on a significant scale. Nevertheless, compared to MOX option characteristics, the thorium based options show advantages in the smaller amount of plutonium (partly substituted by ^{233}U) circulating in the system and the lower number of minor actinides accumulated.

The options that comprise FBs together with thermal reactors demonstrate similar rates of consumption of uranium, enrichment, fuel manufacturing and minor actinide accumulation, regardless of thorium use. Traditionally, thorium allows minimization of plutonium handled in the system but leads to a significant rise in reprocessing.

The thermal light water breeder (LWBR) utilizing option, the results of which are presented in Annex IV, have a number of outstanding advantages and also formidable hurdles impeding its implementation. The introduction of the LWBR decreases uranium consumption down to the lowest level of all scenarios considered. Uranium consumption becomes 15–20% lower than in the case of a uranium–plutonium fuel cycle scenario with fast reactors. The separation work necessary for the enrichment of uranium also drops as does the the total number of minor actinides produced in the reactors. Nevertheless, the mass of fuel to be manufactured grows by 65%, and the mass of reprocessed spent fuel almost doubles. Although a 15% decrease of plutonium circulating in the system could be achieved, the corresponding amount of ^{233}U compensates this effect. Again, due to their better neutron efficiency, HWRs could provide the better option of thorium application here.

With regard to the analysis of waste produced in ThFCs , only a limited number of spent fuel isotopes (^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{237}Np and ^{231}Pa) were considered in this study. Some long living radioactive isotopes that may be important for the analysis of waste produced in a ThFC were not considered here and could be the subject of further consideration.

Another extension of the scenario study could be made through elimination of the defined 6% share limit for HWRs in a global nuclear energy system. Although the consideration of LWBR (Annex IV), AHWR (Annex V) and CANDU (Annex VI) does not take this limit into account, further investigations of possible variations of the LWR/HWR share in the scenarios should be a valuable effort because our preliminary results show that the highest benefit of a ThFC could be achieved in HWRs.

5. ECONOMIC CONSIDERATIONS OF THORIUM UTILIZATION

This section presents a short description of the NESA economic support tool (NEST) developed within INPRO, a set of input data needed for economic considerations of nuclear fuel cycles, and results of using the defined input data in NEST to calculate the levelized unit energy cost (LUEC) of different reactors with different fuel cycles.

5.1. TOOL DESCRIPTION

INPRO methodology [60] is a tool that provides the opportunity to make a comprehensive assessment of sustainability of nuclear energy systems (NES). In economics, the methodology requires calculation of a scope of economic parameters for a NES:

— The LUEC, which is the ratio of total lifetime expenses to total expected power output, expressed in terms of present value equivalent);
— Total investments;
— Economic figures of merit such as internal rate of return and return of investment.

These economic parameters are to be compared against every other possible energy supply option.

NEST is an MS Excel and Visual Basic for Applications (VBA) based spreadsheet developed by the INPRO group as the part of the NESA support package. It is destined for calculation of all parameters of economics envisaged in the methodology of INPRO and also for sensitivity analysis. The calculations can be made for any type of the power plant (e.g. nuclear power plants, fossil and hydropower plants), and calculation of the LUEC can be performed for arbitrary combinations of up to 22 000 power plants.

NEST comprises four different methods of calculation to provide the opportunity to choose the most convenient one for every specific task and also for possible analysis of method-input-result dependence. The four methods are:

— INPRO methodology for an open nuclear fuel cycle in the area of economics [60];
— ["the method described in a"?] Massachusetts Institute of Technology (MIT) study [61];
— ["the method described in a"?] Harvard University study [62];
— Extending the INPRO methodology to a closed fuel cycle by applying the main ideas of [62].

The NEST spreadsheet contains a description including all final equations for every method used in the calculations. Advanced users have the possibility to adapt algorithms to specific situations, i.e. to replace isotopes to be recovered from spent nuclear fuel or to transform fossil power plant into a hydro plant.

In this report, only the extended INPRO methodology was used. The main equation for LUEC in this method is:

$$LUEC = \frac{ONC + IDC + \dfrac{FE_{firstcore}}{\eta \cdot \delta_{th}}}{8760 \cdot Lf} \cdot \left(\frac{1 - \left(\dfrac{1}{1+r}\right)}{1 - \left(\dfrac{1}{1+r}\right)^{t_{LIFE}}} \right) + \frac{\left(1 - \dfrac{1}{t_{LIFE}}\right) \cdot FE_{reload} + BE}{Q \cdot \eta} + LBF + LD + LOM$$

where:

ONC   is the total overnight cost (per unit of installed capacity), including contingency and owner cost;
IDC   is the interest paid during construction per unit of installed electrical capacity;

FE   is the levelized fuel front end cost per kg of heavy metal. To define this parameter one has to know corresponding data on heavy metal mass flow;

BE   is the fuel back end cost of considered type of reactor represented in US $ per kilogram of heavy metal of spent fuel;

Lf   is the average load factor;

Q   is the average burnup of unloaded fuel;

r   is the real discount rate;

t_{LIFE} is the lifetime of the plant;

η is the net thermal efficiency of the plant;

δ_{th} is the average power density in the reactor core at full power (during the first reactor cycle);

LBF   is the levelized back fitting cost (exists only if the plant design envisages lifetime extension);

LD   is the levelized decommissioning cost per unit of produced energy;

LOM   is the levelized unit lifecycle operation and maintenance cost including refurbishment cost.

LUEC   is equivalent to the average real price that would have to be paid by consumers to exactly repay the capital, operation and maintenance (O/M) and fuel cost with a proper discount rate (and without profit) throughout the entire life of the system (plant).

For the four methods described above, NEST provides additional combinations of options based on common approaches used in economic analysis, namely:

— Accounting of the depreciation tax benefit;
— Exclusion of contingency cost at interest during construction (IDC) calculation;
— Definition of date of the investment (beginning of the year, middle of the year, other date) to be determined every year during construction.

5.2. INPUT DATA FOR ECONOMIC ANALYSIS

Input data compilation is a major problem of economic analysis of nuclear energy systems, not only because of the great uncertainty in the values of published parameters, but also sometimes because of the very different value authors assume for the same parameters. For example, contingency cost and owner's cost may be included in overnight cost or may be treated separately; spent fuel reprocessing cost and MOX fabrication cost depend on the assumptions made in their definition (some assumptions are analysed in [61]) and should not be treated independently. Moreover, economic data of nuclear installations are not only technology dependant, but also country specific [63], e.g. the overnight cost of pressurized reactors may vary from $1556/kW(e) published in the Republic of Korea to $5863/kW(e) in Switzerland at the same time. In addition, however, some of the published economic values lack the appropriate justification [59, 64] that would allow estimating the reliability of these data, e.g. the cost of MOX fuel fabrication (as discussed below).

The scope of this report comprises two different reactor groups using thorium. The first group consists of reactors of 'customary' types such as LWRs, HWRs and FRs consuming [233]U/Th fuel and having similar analogues among reactors using uranium–plutonium fuel. The second group of thorium fuelled reactors, which do not have well established analogues in the uranium–plutonium fuel cycle, consists of HTRs and LWBRs. Several other reactor types (e.g. MSR, ADS) are omitted and could be the subject of upcoming studies.

The estimation of economic parameters is performed only for the first reactor group (LWRs, HWRs and FRs) because of the lack of reliable input data published in open sources for other reactors. For each of these three reactor types, the difference between uranium–plutonium and [233]U–Th fuelled reactors of the same type in capital cost and in cost of operation and maintenance is assumed to be low enough to be negligible. Also, roughly following the estimated ratio in the example given in Ref. [60], the capital cost of an HWR is assumed to be 10% higher than that of LWRs. The capital cost of FR is 25% higher than of LWRs, as was agreed in the framework of the GAINS project.

The cost of operation and maintenance depends on many factors, including reactor type and national infrastructure development, but its modest contribution to the final cost of electricity allows it to be assessed at a similar level for all three types of reactors. Reactor decommissioning cost is to be included in the fixed O/M cost.

TABLE 5.1. REACTOR CAPITAL AND O/M COSTS [61, 62, 64–68])

Item	Unit	Reactor type	Range, [61,62, 64-68]	Reference value
Capital cost	$/kW(e)	LWR	1200–4000	2000
		LWR /MOX	1200–2300	2000
		HWR	1200–2800	2200
		FR /MOX	—	2500
Fixed O/M cost	$/kW/a	LWR, LWR /MOX	49–63	55
		HWR	55–63	60
		FR /MOX	80*	60*
Variable O/M cost	mill/kW·h	LWR, LWR /MOX	0.47–0.90	0.5
		HWR	0.47–0.90	0.5
		FR /MOX	—	0.5

* The fixed O/M cost value of FR is estimated in Ref. [62] as equal to that of LWR and HWR, and the variable O/M cost is assumed to be zero.

Front end and back end cost of the fuel cycle are included in the cost of electricity via fuel cost, which aggregates cost of material and services at every stage of nuclear fuel production, storage, reprocessing (if any) and waste disposal.

The main economic parameters of reactors of the first group are presented in Table 5.1. (An alternative set of economic input data is defined in Annex 7.)

The fuel cycle cost, with certain exceptions, usually has a rather moderate impact on the cost of electricity; otherwise, the continuous and insurmountable lack of reliable input data would become a formidable hurdle for the analysis of nuclear fuel cycle options. The authors of a comprehensive study published in Ref. [64] compiling data on the cost of various fuel cycle options claim that the only documents found that presented a uniform costing methodology for all fuel types were prepared nearly 30 years ago by ORNL for the International Nuclear Fuel Cycle Evaluation (INFCE) effort. However, the efforts to define the origin of all of the cost data used in INFCE were also not fully successful.

The parameters of uranium–plutonium fuel cycle front end are relatively more transparent, and the values of mining and milling cost, conversion cost, enrichment and UOX fuel fabrication cost either have very little influence or converge to certain values with acceptable discrepancies caused by differences in conditions (e.g. vendors, time). However, back end fuel cycle cost, i.e. reprocessing, MOX, waste management, and related cost data, are far less reliable.

The above example of MOX fabrication cost will be evaluated in more detail here because its approach is sometimes used as an example for estimation of cost of innovative fuel cycles [59]. According to the published data, the cost of MOX fabrication is $1100/kg, and the same parameter for UOX fuel is $275/kg (in both cases, the cost of nuclear material is not included), i.e a ratio of 4 to 1. The cost of MOX fuel fabrication published in the last 10 years, e.g. presented in Refs [4, 59, 66, 67, 69, 70], usually refer to the same source of input taken from a chain of OECD/NEA reports [59, 71–72], each referring to a previous one. The final explanation of the MOX/UOX cost ratio as 4 to 1 is given in an 1989 OECD/NEA report documented in Ref. [72], whose authors provided a justification of this ratio substantiating that glove-boxes should be obligatory for MOX fabrication. At that time, UOX fuel was often fabricated without glove-boxes but today they are also commonly used for UOX fabrication [59]. Since today, glove-boxes are usually used for both MOX and UOX fabrication, the ratio 4 to 1 may be obsolete, although possible distinctions in ventilation requirements, container designs, radiation shielding, etc, may still affect (increase) the cost of MOX fabrication.

Detailed cost data for process steps of ThFCs are not yet available, and particularly in the backend and ine reprocessed fuel utilization, the data sources [4, 59, 66, 67, 69, 70] still have a large degree of uncertainty. According to Ref. [4], in the area of non-irradiated fuel, the cost of front end parts of ThO_2-UO_2 and conventional

uranium cycles are similar, e.g. the cost of UO2 fabrication ($250/kgHM) and ThO$_2$-UO$_2$ fabrication ($300/kgHM) differ by about 20%. The estimation published in Ref. [70] also shows that the cost of thorium based fuel can deviate by up to 10% (up or down) from the cost of conventional uranium nuclear fuel.

The results of economic analysis documented in Ref. [66] show that fuel costs for a ThO$_2$-UO$_2$ LWR core, which is designed to be operated up to an average burnup of 72 MW·d/kg HM, are about 10% higher than for an all-uranium core operated up to the conventional burnup of 45 MW·d/kg HM based on a price of uranium and thorium valid in 2001.

The Russian Federation Kurchatov Institute and the corporation Thorium Power [64] claim that their once-through ThFCs (fuel component of the cost of electricity) are at least 20% cheaper than the conventional UO$_2$ fuel cycle. Although fuel fabrication costs in a thorium cycle are not cheaper than in a uranium fuel cycle, the thorium cycle is shown to have favourable economics based on high burnup and long core residence time (up to nine years) of the fuel assemblies. Economics of UO$_2$ fuel cycle could, however, also be improved by increasing residence time and burnup. Moreover, since economic effects of the thorium fuel introduction and the improved fuel claddings (for high burnup and long residence time) still to be developed are not taken into account; thus, the cost advantage of the thorium cycle over UO$_2$ may be lower.

In an early Generation IV International Forum (GIF) publication [67], the time lags and some cost data (e.g. reactor construction time, licensing time, overnight cost) for the reactor operating in an open ThFC are assumed to be approximately the same as those for a conventional LWR. In addition, some of the fuel cost parameters such as the fresh fuel fabrication cost, the spent fuel cooling and storage time, and the disposal cost (including the shipping cost) are assumed to be the same as the corresponding LWR cost.

The task of estimating economic competitiveness of innovative reactors is also complicated by the need to assess possible trends of cost constituents. For example, the electricity that could be produced by fast reactors of current design would be several times (from 40% to three times according to the estimation in Ref. [73]) more expensive than that from thermal reactors or from coal fired power plants. However, the necessary improvements in the FR design are integrated into current R&D programmes to make their cost of electricity competitive. Fast reactor developers optimistically assume that cost of energy supplied from fast reactors will equal cost of energy from advanced LWRs.

Taking into account the high uncertainty of data available on the fuel cycle cost discussed above, and only for the purpose of this report, a list of economic input data of a fuel cycle was established and presented in Table 5.2. Compared to the current status of nuclear fuel cycle systems, the data in Table 5.2 are biased in favour of innovative reactors, although the ratios of LWR2/LWR1 and HWR2/HWR1 costs are kept much higher than LWR1/HWR1. This corresponds to the aim of this report to assess the impact of economic parameters on the reactor type fractions (structure) in the system. It is fully recognized that cost is subject to change in the future and should be updated when new information is available and evaluated. To estimate the level of influence of specific parameters an alternative set of economic input data is considered in Annex 7.

In the economic estimations in this report, depleted uranium long term storage, reprocessed uranium storage and the separated fissile product storage are not taken into account. The cost of disposal of spent nuclear fuel and HLW gives a modest influence on the total energy cost [48] and may be estimated via linear extrapolation. The cost of direct disposal of spent fuel for LWRs was taken from Ref. [74] and the same parameters for other reactor types, including HWR, were estimated as:

$$c_{SNFDD}\left[\$/kgHM\right]=10\times B\left[MW\cdot d/kgHM\right],$$

where c_{SNFDD} is the cost of direct disposal, and B is the average burnup of spent fuel.

This estimation is based on published data on the cost of direct disposal of 7 MW·d/kg HM burned HWR fuel at $73/kg HM and the LWR data mentioned above. The cost of disposal of the fissile products after reprocessing of spent fuel is assumed to be equal to the cost of direct disposal of spent nuclear fuel [62].

Currently, economics is one of the main driving forces for innovation in nuclear energy production. The quality of an economic estimation depends mainly on the authenticity of data on the cost of facilities, material and services. The compilation and preliminary assessment of reliability of such data are perfectly suited to the objectives of the INPRO project. Therefore, it is recommended to consider the introduction of such an activity into the INPRO action plan.

TABLE 5.2. URANIUM–PLUTONIUM AND THORIUM FUEL CYCLE COSTS (DATA SOURCES OF THE 'RANGE' ARE IN REFS [4, 59, 61, 62, 64–70, 74])

Fuel cycle step	Units	U-Pu fuel cycle			Th fuel cycle		
		Type	Range	Ref. value	Type	Range	Ref. value
Conversion	$/kg U	LWR, HWR	3–12	8			
Enrichment	$/kg SWU	LWR UOX	80–164	110			
Fuel fabrication	$/kg (HM)	LWR UOX	200–300	275	LWR 0	200–300	275
		HWR UOX	65–135	85	LWR 1	—	325
		LWR MOX	1000–1500	325	LWR 2	1000–1500	1500
		FR-MOX	650–2500	350	HWR 1	—	100
		FR bl U	350–700	350	HWR-2	—	500
					FR-MOX	650–2500	350
					FR bl Th	350–700	350
Reprocessing	$/kg (HM)	UOX	700–900	800	UOX	700–900	800
		MOX	700–1000	800	MOX	700–1000	800
		FR-MOX	1000–2500	1000	FR-MOX	1000–2500	1000
		FR bl U	900–2500	800	FR bl Th	1000–2500	1200
					Th/HEU	*	2000
					Th/Pu		2000
					Th/Pu/U3		2000
SNF direct disposal	$/kg (HM)	LWR	600	600	LWR (Th)	—	600
		HWR	73	Variable[**]	HWR (Th)	—	Variable[**]

Note: 'FR bl. U' or 'FR bl. Th' refers to U or Th blanket in a fast reactor.

[*] A range of 6000 to 20 000 was presented in Ref. [59], where it is used for the cost of reprocessing of the fuel from ADS and FR systems designed for MA burning. For the study, a more optimistic value of 2000 was chosen. (The results for the reprocessing cost 6000 are given in Annex VII.)

[**] The cost of direct disposal of HWR spent fuel may depend on the fuel composition and are roughly proportional to burnup by a factor of 10.

5.3. RESULTS OF ECONOMIC ANALYSIS

The LUEC was calculated on the basis of the technical and economic input presented above. The real discount rate was assumed to be 0.04, which is a rather low value that increases the importance of fuel cost compared to capital cost and probably disguises some deficiencies of reactors with high capital costs but relatively low fuel expenses, such as FRs.

The calculation was performed using NEST with some assumptions made to simplify the analysis. The construction period was assumed to be five years for every reactor type and investments during construction of every reactor were distributed evenly (0.2 a^{-1}). Potential losses of nuclear material in the fuel cycle (e.g. during conversion, reprocessing) are assumed to be zero. The cost of final disposal of fissile products separated during reprocessing is assumed to be equal to the cost of direct disposal of spent nuclear fuel [62]. The cost of plutonium

retrieved from spent fuel of a reactor operated in once-through mode and consumed in different reactor types was estimated as a difference:

$$c\left(^{total}\mathrm{Pu}\right) = c_{repr} + c_{FPdisp} - c_{SFdisp},$$

where $c(^{total}Pu)$ is the cost of plutonium unit, c_{repr} is the cost of reprocessing of spent fuel necessary to produce plutonium unit, c_{SFdisp} is the cost of direct disposal of the same amount of spent fuel, and c_{FPdisp} is the cost of disposal of fissile products separated at reprocessing.

It is conservatively assumed here that the cost of ^{233}U is related to the cost of plutonium as follows:

$$c\left(^{233}\mathrm{U}\right) = \frac{m\left(^{total}\mathrm{Pu}\right)}{m\left(^{239}\mathrm{Pu} + {}^{241}\mathrm{Pu}\right)} \times c\left(^{total}\mathrm{Pu}\right).$$

The results of the calculations are presented in Table 5.3 for 11 reactors and several values of the cost of natural uranium. (Results of calculations performed with an alternative set of input are presented in Annex 7.)

LUEC as a function of natural uranium cost results is also shown in Fig. 5.1.

Discussion of results (Table 5.3 and Fig. 5.1)

The low cost values, especially, of fast reactors are caused by the rather optimistic economic assumptions taken over from the GAINS project, such as FR capital cost (only 25% higher than LWR) and a moderate discount rate (4%).

Energy costs of all reactors using reprocessed fuel (LWR1, LWR2, HWR1, HWR2, FR and FRTh) are evidently not dependent on natural uranium cost; therefore, they are represented by horizontal (dashed) lines in Fig. 5.1.

A FR with a blanket consisting of depleted or reprocessed uranium has the lowest electricity generation cost (28.8 mills/kW·h) of all fast reactors considered. Using a thorium blanket in a fast reactor slightly increases its cost of electricity generation (32.0 mills/kW·h) as part of its needed plutonium has to be reprocessed from ALWR spent fuel. Cost of electricity (34.2 mills/kW·h) generated by an FR consuming plutonium recycled from ALWR spent fuel during its first six years in operation is 15% higher than in the case of consumption of its own plutonium (32.0 mills/kW·h). This is caused by the much higher reprocessing effort needed to produce fuel for FR from ALWR spent fuel instead of from FR spent fuel with a much higher plutonium content. Therefore, it is to be expected that at the initial stage of a fast reactor programme, all new FRs will pass through a stage of rather expensive fuel; consequently, additional constraints, i.e. not only the availability of plutonium, but also its cost should be considered in upcoming scenario studies. The same effect of higher plutonium cost causing higher electricity cost occurs at the initial stage of commercial deployment of fast reactors with a thorium blanket (FRTh).

TABLE 5.3. LUEC DEPENDING ON THE NATURAL URANIUM COST (U_{nat} \$/kg U)

U_{nat} \$/kg U	HWR	LWR	ALWR	LWR0	LWR1[1]	LWR2[2]	LWR2[3]	HWR1[1]	HWR2[4]	HWR2[5]	AHWR[6]	FRTh[9]	FR[7]	FR[8]
50	30.1	29.7	27.3	31.2	40.2	37.2	35.6	36.4	40.9	38.9	27.6	32.0	29.8	34.2
150	32.1	32.7	29.7	37.2							32.6			
300	35.3	37.2	33.2	46.2							40.0			
1000	49.8	58.3	49.9	88.4							74.7			

Notes:

[1] Pu from ALWR spent fuel.
[2] ^{233}U from LWR1, Pu from ALWR.
[3] ^{233}U from LWR1, Pu from LWR1.
[4] ^{233}U from HWR2, Pu from ALWR.
[5] ^{233}U from HWR2, Pu from LWR1.
[6] Once-through mode (Annex V).
[7] Pu from FR.
[8] For the first six years of operation Pu is taken from ALWR.
[9] Pu from FRTh plus a small amount from ALWR.
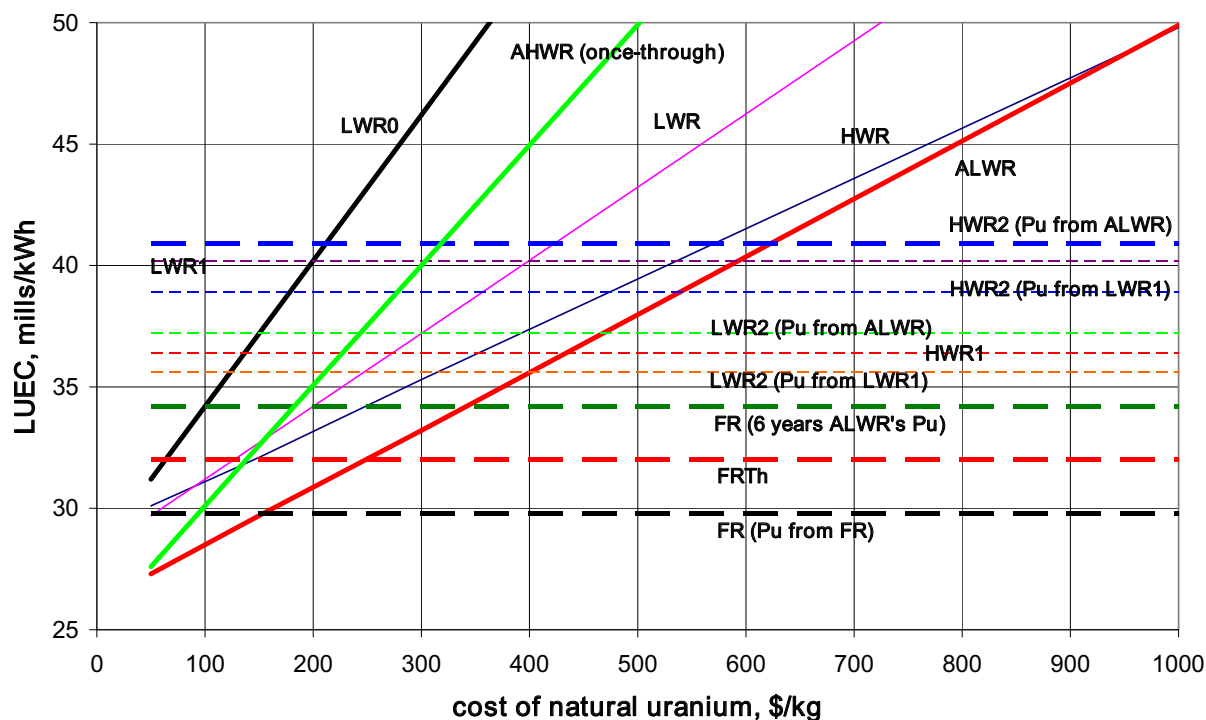
FIG. 5.1. LUEC depending on natural uranium cost. (For characteristics of reactors, see Tables 4.6 and 4.7 and Annex I.)

Thermal reactors with a closed fuel cycle show similar effects as discussed above for fast reactors: the use of plutonium from spent fuel of a different reactor increases its generation cost for electricity.

Among thermal reactors with a once-through fuel cycle, the AHWR using thorium generates some of the cheapest electricity at low uranium cost and is shown to be competitive against the ALWR below a uranium price of ~$50/kg and against conventional water cooled reactors (LWR, HWR) up to a cost of $150 per kilogram of natural uranium. This is stipulated by the AHWR's high capacity factor (0.9) and also by the combination of its long service life (100 years) and low discount rate assumed (0.04), and not directly due to the fuel type used, although high capacity factor and reactor lifetime are achieved in the design of thorium–uranium fuelled reactor. However, in the once-through fuel cycle, the AHWR has one of the steepest growths of energy cost by an increase of uranium cost due to the high enrichment of the uranium fraction in fresh fuel and the relatively high share of ^{235}U in spent fuel. At a higher cost of uranium, AHWR reactors — originally designed to work in an open fuel cycle — might be compelled to introduce reprocessing. Such reactors using thorium and envisaging reprocessing may become competitive with the traditional thermal reactors operated in once-through fuel cycle at a cost of natural uranium at a level of $400/kg.

The economic input parameters defined above in Tables 5.1 and 5.2 were used as input for the economic considerations in Sections 4.7 and 4.8 in combination with the data provided in Table 4.4 of uranium resources and prices.

However, it is recognized that the levelized costs of electricity calculated in this chapter are only preliminary estimations and should be subject to further consideration and update in the future when new input data and advanced economic models become available.

General economic aspects of thorium introduction

Historically, the currently existing uranium–plutonium fuel cycle has been driven by the investment made in the past for military applications, which created a certain economic barrier against the introduction of other innovative fuel types. To be implemented in a short and medium time scale, the ThFC has to overcome gaps in infrastructure development by demonstration of strong economic and non-proliferation advantages, and by maintaining the required levels of safety and security compared with other options. For thermal reactors,

particularly for heavy water reactors and for the once-through fuel cycle, advantages of thorium introduction can be expected.

Unlike the advantages for thermal reactors, those for an application of thorium as fertile material in the blanket of fast breeder reactors are less obvious. An FR operating with a uranium–plutonium fuel cycle and a conversion ratio equal or more than one (breeding) is a self-sufficient system that consumes only depleted uranium, which is a practically unlimited resource with material cost close to zero. Further, the introduction of thorium blankets in an FR leads to the necessity of:

— Installation and maintenance of corresponding reprocessing and fabrication facilities of thorium fuel;
— Introduction of coupled reactors in the nuclear energy system, i.e. reactors that will consume ^{233}U produced in an FR blanket;
— Reactors that will produce plutonium necessary for FR fresh fuel.

Such nuclear energy systems will consume not only depleted uranium, but also thorium, with costs in the same order as natural uranium. The size of a thorium blanket in the reactor types considered in this study can be several times the size of a uranium blanket, which leads to an increase of reprocessing demands and cost of produced fissile material. Such a system does not usually lead to improvement of the achievable level of conversion rate. From the point of view of proliferation resistance (discussed in Section 6), handling of ^{233}U separated from a thorium blanket hardly has any significant advantages over the handling of plutonium from uranium blankets.

6. PROLIFERATION RESISTANCE CONSIDERATIONS OF THORIUM FUEL CYCLES

In this section, a uranium–plutonium fuel cycle is compared with a thorium–uranium fuel cycle concerning proliferation resistance (PR), applying an analysis method developed within INPRO. (A more detailed description of the comparison is provided in Annex VIII.)

6.1. GENERAL FEATURES OF THE METHODOLOGY FOR PROLIFERATION RESISTANCE

The proliferation resistance analysis method developed within INPRO (documented in Vol. 5 of Reference [60] looks at the intrinsic features of a nuclear energy system and extrinsic measures in the country running the nuclear energy system. Intrinsic features are part of the technical design of a nuclear energy system, including those that facilitate the implementation of extrinsic measures. They are included in the design to ensure that a long time and great effort would be required for modifications necessary to use a civilian nuclear energy system for a weapon production programme, thereby increasing the probability of early detection of such activities. Extrinsic measures result from States' decisions and undertakings related to a nuclear energy system, e.g. establishing appropriate safeguards agreements with the IAEA that enable safeguards to ensure that all nuclear installations of a nuclear energy system are used for civilian purposes only.

In the comparative assessment (performed in this report) of proliferation resistance of different nuclear fuel cycles, only intrinsic features are considered. Examples of such intrinsic features are: isotopic content of the nuclear material (NM) used in the fuel cycle; the chemical form of the NM; its radiation field; the heat it generates; its spontaneous neutron generation rate; its mass and bulk; and the design features that limit access to it. Depending on the design of these features, the INPRO methodology defines a range of proliferation resistance values from very weak to very high.

6.2. COMPARISON OF PROLIFERATION RESISTANCE OF ThFCs AGAINST URANIUM FUEL CYCLES

Two pairs of nuclear fuel cycle (NFC) systems are chosen for the comparison:

— Uranium–plutonium once-through NFC against thorium–uranium once-through NFC;
— Uranium–plutonium closed NFC against thorium–uranium closed NFC.

The uranium–plutonium once-through NFC uses enriched uranium in fresh fuel and accumulates some plutonium in spent fuel during reactor operation. The thorium–uranium once-through NFC uses in its fresh fuel, in addition to enriched uranium, also some fuel rods (or separate fuel elements) that contain only thorium. Once-through NFCs are assumed to be used in thermal reactors (i.e. the reactors with thermal neutron spectrum: LWR or HWR) and do not envisage any reprocessing of spent fuel.

In the fresh fuel of the uranium–plutonium closed NFC, plutonium is used as fissile material together with uranium (recycled or depleted). In the thorium–uranium closed NFC, fresh fuel contains ^{233}U as fissile material together with thorium as fertile material. The uranium–plutonium closed NFC is based on fast breeder reactors and the thorium–uranium closed NFC is based on thermal breeding reactors (LWRs or HWRs).

6.3. RESULTS OF COMPARISON OF PROLIFERATION RESISTANCE FOR ONCE-THROUGH FUEL CYCLES

The comparison in Annex VIII shows that in the once-through variant, thorium fuel can provide certain advantages regarding proliferation resistance, but the need for enriched uranium (usually up to 20% of ^{235}U/U) in the core and the corresponding composition of spent fuel compensates for such benefits. Therefore, no advantage of a thorium–uranium open NFC was found in comparison to a uranium–plutonium open NFC.

6.4. RESULTS OF COMPARISON OF PROLIFERATION RESISTANCE FOR CLOSED NUCLEAR FUEL CYCLES

To illustrate the detailed comparison documented in Annex VIII, examples of the results of proliferation resistance assessment using the INPRO methodology of the two closed NFC options are shown in Tables 6.1 and 6.2 (at the end of the section).

Tables 6.1 and 6.2 present the proliferation resistance assessment of several intrinsic features of the closed NFC that relate to the attractiveness of the nuclear material in the NFC for use in a weapons programme. The first column in both tables lists the intrinsic features (parameters) to be examined. The following columns (evaluation scale) show the possible score of proliferation resistance depending on the value of the parameter. The last four columns contain the score of proliferation resistance level for the steps of the closed NFC, i.e. fuel fabrication, reactor operation, spent fuel reprocessing and waste disposal.

For both cases all possible scores (from very weak to very strong proliferation resistance) were found for different parameters. Therefore, to enable an easier comparison of the detailed results, an aggregation was performed as follows:

— All scores with the same level of proliferation resistance for both NFC were eliminated from further considerations.
— Only scores of proliferation resistance level that were different for the two NFC assessed were kept and listed together in Table 6.3.

The results in Table 6.3 are listed according to the following scheme: the score for the uranium–plutonium closed NFC is shown in the upper part of each box, and for thorium–uranium in the lower part of the box.

TABLE 6.1. ASSESSMENT OF PROLIFERATION RESISTANCE OF A URANIUM–PLUTONIUM CLOSED NFC USING THE INPRO METHODOLOGY —ATTRACTIVENESS OF NM CONSISTING OF URANIUM–PLUTONIUM

Group of parameters	Parameter	Evaluation scale					Scores			
		VW[1]	W[1]	M[1]	S[1]	VS[1]	FF[1]	RO[1]	SNFR[1]	WD[1]
Material quality	1. Isotopic composition: ^{239}Pu/Pu (wt%)		>50		<50		W[2]	W[2]	W[2]	W[2]
	2. Heat generation characteristic: ^{238}Pu/Pu (wt%)		<20		>20		W[3]	W[3]	W[3]	W[3]
	3. Spontaneous neutron generation rate characteristic: $(^{240}$Pu+ ^{242}Pu) /Pu (wt%)	In comparison to Th/U closed fuel cycle[4]				VS[1]	S[4]	S[4]	S[4]	S[4]
	4. Material type/category	UDU[1]	IDU[1]	LEU[1]	NU[1]	DU[1]	W	W	W	W
	5. Radiation field: dose (mGy/h) at 1 mr	<150	150–350	350–1000	1000–10 000	>10 000	VW[5]	M[6]	VW[5]	VS[7]
Material quantity	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	VW[8]	M[8]	VW[8]	W[8]
	7. No. of items for SQ	1	1~10	10~50	50~100	>100	VS[9]	VW[9]	VS[9]	S
	8. Mass of bulk material for SQ (dilution) (kg)	10	10~100	100~500	500~1000	>1000	W	N/A	W	VS
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VW	VW	VW	VW
Material form	10. Chemical/physical form	Metal	Oxide/solution	Compounds	Spent fuel	Waste	W	S	W	VS

Notes:

1. FF = fuel fabrication, RO = reactor operation, SNFR = spent nuclear fuel reprocessing, WD = waste disposal; VW = very weak PR, W = weak PR, M = medium PR, S = strong PR, VS = very strong PR; UDU = un-irradiated direct weapon usable material, IDU = irradiated weapon usable material, LEU = low enriched U (<20%), NU = natural U, DU = depleted U.

2. According to the fast reactor data considered in scenario simulations, the average ratio of ^{239}Pu/Pu is about 60 wt%; this value for blanket plutonium is even higher. At the waste disposal stage, 1% of total Pu of the same isotopic composition is assumed to be lost during reprocessing.

3. The average ratio ^{238}Pu/Pu does not exceed 1 wt% at 5 years after discharge).

4. The average ratio (^{240}Pu+^{242}Pu)/Pu is in a range 28–31 wt% at 5 years after discharge and the value for blanket plutonium is much lower. Majors efforts are still necessary, such as the introduction of obligatory combined reprocessing of core and blanket fuel. Since the evaluation scale for this parameter is not yet well established, in the framework of this study, higher scores are given to U/Pu fuel cycle to stress higher spontaneous neutron generation rate of plutonium only.

5. The judgment is based on the experience of MOX fuel fabrication. The radiation field from MOX fuel depends on the stage of its cleaning. In general, well cleaned MOX fuel requires modest special measures during fuel fabrication in addition to glove boxes, which are a casual practise in LEU fuel fabrication.

6. Because of higher burnup and higher concentration of MA, the core fuel has a higher radiation field than LEU fuel, but the blanket fuel will possess lower radioactivity. Taking into account the level of prospective burnup of blanket fuel, a radiation field can be estimated as more than the order of magnitude lower than in the case of PWR spent fuel.

7. High concentration of MA in wastes and their long half-lives were taken into account.

8. At the stages of FF and SNFR, it is possible to operate with fuel pellets; at the RO stage, it is possible to operate with fuel assemblies that are apparently lighter than modern PWR assemblies; at the WD stage, one operates with vitrified waste items of limited mass to provide necessary cooling.

9. Following the previous reference, at the FF and SNFR stages, 'item' is assumed to be 'pellet' (otherwise, the score should be changed correspondingly in item 6) and assemblies are considered at the RO stage, where the concentration of Pu in the fresh fuel is assumed to be 20 wt%, SQ (Pu)=8 kg, and the mass of the fuel in assembly is over 40 kg.

TABLE 6.2. ASSESSMENT OF PROLIFERATION RESISTANCE OF A THORIUM–URANIUM CLOSED NFC USING THE INPRO METHODOLOGY — ATTRACTIVENESS OF NM CONSISTING OF TH–^{233}U

Group of parameters	Parameter	Evaluation scale					Scores[1]			
		VW	W	M	S		FF[1]	RO[1]	SNFR[1]	WD[1]
	1. Isotopic composition: highest value of ^{233}U/U (wt%) in pin		>12		< 12		W[2]	W[2]	W[2]	W[2]
	2. Heat generation characteristic: decay heat emission (watt/critical mass)		< 600		>600		W	W	W	W
Material quality	3. Spontaneous neutron generation rate		In comparison to U/Pu closed fuel cycle			VS	W[3]	W[3]	W[3]	W[3]
	4. Material type/category	UDU[1]	IDU[1]	LEU[1]	NU[1]	DU[1]	W	W	W	W
	5. Radiation field: ^{232}U contamination in ^{233}U (ppm)	< 400[4]	400–1000[4]	1000–2500[4]	2500–25 000[4]	>25 000[4]	M	M	M	M
	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	VW[5]	S[5]	VW[5]	W
Material quantity	7. No. of items for SQ	1	1~10	10~50	50~100	>100	VS	VW[6]	VS	VS[7]
	8. Mass of bulk material for SQ (dilution) (kg)	10	10~100	100~500	500~1000	>1000	M	N/A	M	VS
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VW	VW	VW	M[8]
Material form	10. Chemical/physical form	Metal	Oxide/solution	Compounds	Spent fuel	Waste	W	S	W	VS

Notes:

[1] FF=fuel fabrication, RO=reactor operation, SNFR=spent fuel reprocessing, WD = waste disposal; VW = very weak PR, W = weak PR, M = medium PR, S = strong PR, VS = very strong PR; UDU = unirradiated direct weapon usable material, IDU = irradiated weapon usable material, LEU = low enriched U (<20%), NU = natural U, DU = depleted U.

[2] The share of ^{233}U is 98% approximately.

[3] Spontaneous fission rate of ^{233}U is 0.5 (s·kg)$^{-1}$. (The same parameter for ^{239}Pu (containing 6% of Pu-240) is $2.5*10^4$ (sec·kg)$^{-1}$).

[4] The scale of ^{232}U contamination has been elaborated in the INPRO methodology [60]; it differs from the scale applied for the radiation field of the U/Pu fuel cycle because of the complimentary factors considered, e.g. the difference of half-lives of the main γ-radioactive isotopes (including predecessors), low concentration of γ-radioactive isotopes (^{224}Ra, ^{220}Rn, ^{212}Bi, ^{208}Tl) in the purified ^{233}U/^{232}U mixture, etc.

[5] The same mass as in U/Pu and Th/U one-through fuel cycles (and SNFR = FF).

[6] The concentration of ^{233}U is up to 5%, SQ(^{233}U) = 8 kg.

[7] Estimated with the same assumptions as for U/Pu closed fuel cycle.

[8] From 10 GW(e) nuclear energy system per annum.
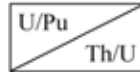
TABLE 6.3. AGGREGATED RESULTS FOR CLOSED FUEL CYCLE OPTIONS (URANIUM–PLUTONIUM–THORIUM–URANIUM)

No.	Parameter	FF	RO	SFNR	WD
3	Spontaneous neutron generation rate	S / W	S / W	S / W	S / W
5	Radiation field	VW / M		VW / M	VS / M
6	Mass of an item		M / S		
7	No. of items for SQ				S / VS
8	Mass of bulk material for SQ	W / M		W / M	
9	No. of SQ				VW / M
20	Detectability of radiation signature*	W / S		W / S	
25	Accessibility of NM to inspectors*	S / W	S / W	S / W	
26	Extent of automation*	S / VS			

Also, as shown in Table 6.3, in the closed fuel cycle options, benefits and deficiencies with regard to proliferation resistance are divided 50/50; therefore, thorium can hardly contribute substantially to an increase in the PR of a closed fuel cycle.

However, the options considered here are not the only variants of ThFC introduction, and other methods of comparison using different inputs are also possible that could lead to different conclusions.

7. CONCLUSIONS AND OUTLOOK

7.1. CONCLUSIONS

The natural abundance of thorium in comparison to uranium, its chemically inert nature, superior thermal conductivity of ThO_2 over UO_2 and advanced neutron characteristics make thorium based fuel cycles attractive. The investments in R&D activities related to thorium use continue, and designers have amassed considerable knowledge as a result. While thorium fuel fabrication and irradiation experience cannot yet be characterized as commercially 'mature', there are sufficient knowledge and experience today for a technically feasible implementation of a once-through ThFC .

Experiments and analytical studies demonstrate the following:

— The heavy water reactors can efficiently exploit thorium based fuel cycles for breeding and burning [233]U, and for application of thorium in once-through mode without recycling. For the once-through cases, higher burnup scenarios lead to a higher percentage of energy from thorium fuel. For fuel cycles with recycling, percentage of energy gained from thorium is higher for the low burnup than the high burnup case.

— Introduction of thorium fuel in an open fuel cycle using LWRs by replacing part of their (enriched) uranium fuel with thorium apparently requires significant modification of fuel management strategy (e.g. super high burnup of thorium assemblies); otherwise, it generally increases the consumption of natural uranium or plutonium. Another possible role of LWRs with a ThFC may be associated with burnup of ^{233}U produced in other reactor types.

— Use of thorium in thermal reactors with recycling can provide approximately the same reduction of uranium, enrichment and fuel manufacturing efforts as can be achieved by the introduction of uranium–plutonium MOX fuel. The decrease of MA accumulation in spent fuel via thorium use is possible only if designers succeed in avoiding plutonium use in the thorium based fresh fuel; otherwise, the production of MA grows.

Economic considerations show that:

— An optimized design (100 year lifetime, 90% availability) of a thorium reactor operating in a once-through fuel cycle may be competitive against uranium–plutonium reactors, depending on the cost of natural uranium;

— The ThFC with reprocessing may become competitive against uranium–plutonium once-through fuel cycle if the cost of natural uranium is higher than ~$400/kg, depending on the cost of reprocessing;

— Taking into account some national conditions, the application of thorium may be considered as a complementary option for a NES with fast reactors, but the former can hardly be competitive against successfully deployed energy system based on fast reactors without thorium.

The proliferation resistance advantages of the ThFC are not as strong as could be first assumed. Strong proliferation resistance features of thorium application are quite balanced by corresponding deficiencies and potential advantages can probably be realized only through the development of specific designs.

7.2. OUTLOOK AND RECOMMENDATIONS FOR UPCOMING STUDIES

The scenario study presented in this report has comprised various combinations and a wide scope of reactor options designed for thorium based fuel use; nevertheless, broadening the list of available reactor types, fuel compositions and strategies (e.g. LWR with super high burnup of thorium fuel) may provide novel information on ThFC benefits.

The majority of the scenario simulations (all except for three) in this report were bonded with the assumption of a 6% share of nuclear generation by HWRs. Because of their high neutronic efficiency, the HWR reactors would probably benefit from thorium more than LWRs. A variation of the ratio of HWR/LWR generation share may be studied in follow-up projects.

Several reactor types that apparently have good prospects for thorium application were out of the focus of the current study. Examples of such reactors are:

— Small transportable nuclear reactors that may potentially benefit from thorium due to very long fuel cycles (breeding and burning of ^{233}U in the core), possible decrease of initial enrichment of the fuel, and strengthening of physical protection;

— Molten salt reactors that may also eventually adopt thorium to improve performance parameters, e.g. to increase their conversion ratio.

Follow-up projects may focus on thorium implementation in reactors of innovative design.

Only a limited number of isotopes traditionally used in uranium–plutonium spent fuel consideration (minor actinides) were analysed as waste constituents in this study. Other long lived radioactive isotopes that may be important for waste analysis of ThFC are not covered here and may be investigated in follow-up projects.

Economic considerations are an important part of a vision defining process, but are heavily dependent on the reliability and availability of input data. In addition to its availability, the price of plutonium depending on its source should also be taken into account in future studies. Follow-up projects may focus on the compilation and preliminary assessment of accuracy of data on NPP cost constituents and fuel cycle facility service cost.

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Annex I

DATA ON THORIUM UTILIZING REACTORS FOR SCENARIO SIMULATION

I–1. PRESSURIZED WATER REACTOR DATA PROVIDED BY THE REPUBLIC OF KOREA

The scope of PWR data comprises three options:

- UO_2+ThO_2 — the reactor utilizing thorium fuel and UO_2 seed fuel (20% enriched ^{235}U), 'LWR0' in the study;
- $ThO_2+Pu(RG)O_2$ — the reactor utilizing thorium fuel and reactor grade plutonium seed fuel, 'LWR1' in the study;
- $ThO_2+Pu(WG)O_2$ — the reactor utilizing thorium fuel and weapon grade plutonium seed fuel.

See Tables I–1 to I–16.

TABLE I–1. GENERAL DATA ON PWRs

Parameter	Units	UO_2+ThO_2	$ThO_2+Pu(RG)O_2$	$ThO_2+Pu(WG)O_2$
Net electric output	MW	900	900	900
Thermal power	MW	2775	2775	2775
Thermal efficiency	—	0.3243	0.3243	0.3243
Load factor	—	0.8	0.8	0.8
Operating cycle	Days	378	401	361
Reactor lifetime	Years	60	60	60
Average burnup	MW·d/kg HM	38.2	40.5	36.4
Averagefuel residence time	D	927.3	983.7	885.6
Mass of the core	t HM	67.05	67.01	67.03
Reloading mass	t HM	27.65	27.64	27.64
First core mass	t HM	67.05	67.01	67.03
Annual reloading mass	t HM	27.65	27.64	27.64
Annual discharge mass	t HM	26.60	26.54	26.65

TABLE I–2. ISOTOPIC COMPOSITION OF PWR FRESH FUEL

Parameter	Units	UO_2+ThO_2	$ThO_2+Pu(RG)O_2$	$ThO_2+Pu(WG)O_2$
^{232}Th-	t HM	21.0264	25.2591	25.9142
^{233}U	t HM	0.0000	0.0000	0.0000
^{234}U	t HM	0.0000	0.0000	0.0000
^{235}U	t HM	1.2654	0.0059	0.0063
^{236}U	t HM	0.0000	0.0000	0.0000
^{238}U	t HM	5.3599	0.3215	0.3463
^{238}Pu	t HM	0.0000	0.0368	0.0000
^{239}Pu	t HM	0.0000	1.2101	1.2867
^{240}Pu	t HM	0.0000	0.4737	0.0822
^{241}Pu	t HM	0.0000	0.2523	0.0000
^{242}Pu	t HM	0.0000	0.0831	0.0000
^{241}Am	t HM	0.0000	0.0000	0.0000

TABLE I–3. ISOTOPIC COMPOSITION OF PWR SPENT FUEL

Parameter	Units	UO_2+ThO_2	$ThO_2+Pu(RG)O_2$	$ThO_2+Pu(WG)O_2$
^{232}Th	t HM	20.4309	24.6683	25.3201
^{233}U	t HM	0.3436	0.3940	0.3888
^{234}U	t HM	0.0354	0.0316	0.0332
^{235}U	t HM	0.4025	0.0091	0.0093
^{236}U	t HM	0.1537	0.0010	0.0011
^{238}U	t HM	5.1206	0.3127	0.3377
^{238}Pu	t HM	0.0036	0.0320	0.0031
^{239}Pu	t HM	0.0554	0.2860	0.2190
^{240}Pu	t HM	0.0184	0.3564	0.1853
^{241}Pu	t HM	0.0154	0.2400	0.1096
^{242}Pu	t HM	0.0054	0.1391	0.0316
^{241}Am	t HM	0.0004	0.0185	0.0051
^{242m}Am	t HM	0.0000	0.0005	0.0001
^{243}Am	t HM	0.0011	0.0276	0.0061
^{242}Cm	t HM	0.0002	0.0050	0.0016
^{244}Cm	t HM	0.0003	0.0144	0.0020
^{237}Np	t HM	0.0106	0.0001	0.0001

I–2.   AHWR REACTOR DATA PROVIDED BY INDIA

The scope of AHWR data comprises three options:

(1)   Th+^{233}U+Pu (standard cluster, st.cl.) and (2) Th+^{233}U+Pu (alternative cluster, alt.cl.) — the AHWR reactors utilizing thorium fuel cycle with reprocessing;
(2)   Th+LEU – the AHWR reactor utilizing ThFC in once-through mode.

TABLE I–4. GENERAL DATA ON AHWRs

Parameter	Units	Th+U3+Pu (st.cl.)	Th+U3+Pu (alt.cl)	Th+LEU
Net electric output	MW	300	300	300
Thermal power	MW	920	920	920
Thermal efficiency	—	0.33	0.33	0.33
Load factor	—	0.9	0.9	0.9
Operating cycle	Days	810	810	810
Reactor lifetime	Years	100	100	100
Average burnup	MW·d/kg HM	38	35.5	64
Power density	MW/m^3	12.23	12.23	12.5
Mass of the core	t HM	26.096	26.562	51.725
Annual reloading mass	t HM	4.5435	4.5435	5.05

TABLE I–5. ISOTOPIC COMPOSITION OF AHWR FRESH FUEL

Parameter	Units	Th+U3+Pu (st.cl.)	Th+U3+Pu (alt.cl)	Th+LEU
^{232}Th-	t HM	4.39079	4.38069	4.03007
^{233}U	t HM	0.08708	0.05679	—
^{235}U	t HM	—	—	0.21541
^{238}U	t HM	—	—	0.88051
^{239}Pu	t HM	0.04515	0.07293	—
^{240}Pu	t HM	0.01614	0.02608	—
^{241}Pu	t HM	0.00345	0.00558	—
^{242}Pu	t HM	0.00089	0.00143	—

TABLE I–6.   ISOTOPIC COMPOSITION OF AHWR SPENT FUEL

Parameter	Units	Th+U3+Pu (st.cl.)	Th+U3+Pu (alt.cl)	Th+LEU
^{232}Th	t HM	4.25518	4.26202	3.82310
^{232}U	t HM	0.00013	0.00012	0.00021
^{233}U	t HM	0.07579	0.07028	0.06132
^{234}U	t HM	0.01285	0.00970	0.01194
^{235}U	t HM	0.00211	0.00152	0.01440
^{236}U	t HM	0.00026	0.00016	0.03064
^{238}U	t HM	0.00000	0.00000	0.82179
^{238}Pu	t HM	0.00028	0.00046	0.00147
^{239}Pu	t HM	0.00019	0.00193	0.00628
^{240}Pu	t HM	0.00376	0.01222	0.00320
^{241}Pu	t HM	0.00257	0.00643	0.00211
^{242}Pu	t HM	0.00529	0.00736	0.00212
^{241}Am	t HM	0.00025	0.00025	0.00014
^{242m}Am	t HM	0.00000	0.00000	0.00000
^{243}Am	t HM	0.00164	0.00164	0.00062
^{242}Cm	t HM	0.00008	0.00008	0.00003
^{243}Cm	t HM	0.00000	0.00000	0.00000
^{244}Cm	t HM	0.00080	0.00080	0.00032
^{237}Np	t HM	0.00001	0.00001	0.00302
^{231}Pa	t HM	0.00018	0.00018	0.00019

I–3.   BWR DATA PROVIDED BY NORWAY (THOR ENERGY)

The BWR options are currently under development and only limited scope of material flow data are available for publication. These data are represented here although they were not used in scenario calculations due to the lack of necessary parameters (spent fuel composition):

— Th+Pu – the reactor and fuel cycle without reprocessing of thorium fuel;
— Th+^{233}U+Pu — the reactor and fuel cycle with reprocessing of thorium fuel and recycling of ^{233}U.

TABLE I–7. GENERAL DATA ON BWRs

Parameter	Units	Th+Pu	Th+^{233}U+Pu
Net electric output	MW	1190	1190
Thermal power	MW	3300	3300
Thermal efficiency	—	0.360606	0.3606061
Load factor	—	0.905556	0.9055556
Operating cycle	Days	316.9444	316.94444
Reactor lifetime	Years	60	60
Average burnup	MW·d/kg HM	55	55
Average fuel residence time	D	1901.667	1901.6667
Power density	MW/t	28.922	28.921998
Mass of the core	t HM	114.1	114.1
First core mass	t HM	114.1	114.1
Reloading mass	t HM	19.01667	19.016667

TABLE I–8. ISOTOPIC COMPOSITION OF BWR FRESH FUEL

Parameter	Units	Th+Pu	Th+^{233}U+Pu
^{232}Th	t HM	17.19667	17.675
^{233}U	t HM	—	0.3383333
^{234}U	t HM	—	0.0418165
^{238}Pu	t HM	0.0455	0.0242083
^{239}Pu	t HM	0.98644	0.5248367
^{240}Pu	t HM	0.43316	0.2304633
^{241}Pu	t HM	0.22932	0.12201
^{242}Pu	t HM	0.12558	0.066815

TABLE I–9. ISOTOPIC COMPOSITION OF PLUTONIUM IN SPENT FUEL OF BWR

Parameter	Units	Th+Pu	Th+^{233}U+Pu
^{238}Pu	t HM	0.021292	0.007875
^{239}Pu	t HM	0.461603	0.17073
^{240}Pu	t HM	0.202697	0.07497
^{241}Pu	t HM	0.10731	0.03969
^{242}Pu	t HM	0.058765	0.021735

I–4. HEAVY WATER REACTOR (CANDU TYPE) DATA PROVIDED BY CANADA

Two options of CANDU reactors were considered in the scenario study:

— Th+Pu — the reactors utilizing thorium fuel and breeding ^{233}U. This reactor can be used both in the fuel cycle without ^{233}U recycling and for production of ^{233}U to be consumed in other reactor type, 'HWR1';
— Th+^{233}U+Pu — the reactor utilizing thorium fuel and recycled ^{233}U (in a self-sustainable mode) and Pu, called 'HWR2'.

TABLE I–10. GENERAL DATA ON CANDU REACTORS

Parameter	Units	Th+Pu	Th+U3+Pu
Net electric output	MW	668	668
Thermal power	MW	2064	2064
Thermal efficiency	—	0.32	0.32
Load factor	—	0.95	0.95
Operating cycle	Days	825	810
Reactor lifetime	Years	60	60
Average burnup	MW·d/kg HM	20.292	19.85
Av. Fuel residence time	d	825	810
Power density	MW/t	24.6	24.51
Mass of the core	t HM	71.397	71.397
Reloading mass	t HM	71.397	71.359

TABLE I–11. ISOTOPIC COMPOSITION OF CANDU FRESH FUEL

Parameter	Units	Th+Pu	Th+U3+Pu
^{232}Th	t HM	68.707	69.57898
^{233}U	t HM	—	1.031254
^{238}Pu	t HM	0.067159	0.019401
^{239}Pu	t HM	1.456002	0.420623
^{240}Pu	t HM	0.639351	0.184702
^{241}Pu	t HM	0.33848	0.097783
^{242}Pu	t HM	0.182672	0.052772

TABLE I–12. ISOTOPIC COMPOSITION OF CANDU SPENT FUEL

Parameter	Units	Th+Pu	Th+U3+Pu
^{232}Th	t HM	67.766922	68.39363
^{233}U	t HM	0.6210774	1.031282
^{234}U	t HM	0.0360566	0.108079
^{235}U	t HM	0.0027752	0.010705
^{236}U	t HM	0.0001527	0.000904
^{238}U	t HM	3.12E-06	9.89E-07
^{238}Pu	t HM	0.0401457	0.010766
^{239}Pu	t HM	0.3105844	0.055048
^{240}Pu	t HM	0.6229421	0.155835
^{241}Pu	t HM	0.1558441	0.043947
^{242}Pu	t HM	0.264036	0.083448
^{241}Am	t HM	0.055212	0.015316
^{242m}Am	t HM	0.055212	3.29E-05
^{243}Am	t HM	0.0001192	0.014168
^{242}Cm	t HM	0.0395702	4.89E-07
^{244}Cm	t HM	1.552E-06	0.002321
^{245}Cm 5	t HM	0.0064581	0
^{237}Np	t HM	0.0002981	9.71E-05

I–5. PWR, FBR AND HTR DATA PROVIDED BY THE RUSSIAN FEDERATION (INSTITUTE OF PHYSICS AND POWER ENGINEERING)

The water moderated water cooled reactor (WWER) is a PWR type reactor of the Russian design that was considered in the scenario study as well as the fast breeder reactor of BN-800 type and the high temperature reactor:

— WWER-1000 — PWR type reactor utilizing ^{233}U+Pu+depleted uranium fuel, 'LWR2' in the study;
— FBR-A — BN-800 type fast reactor utilizing ^{235}U+^{238}U +Pu fuel in a core and thorium in a blanket, 'FRTh' in the study;
— HTR — reactor using thorium fuel and recycled ^{233}U (in a self-sustainable mode) and plutonium.

TABLE I–13. GENERAL DATA ON WWERs, FBRs AND HTRs

Parameter	Units	WWER	FBR-A	HTR
Net electric output	MW	1000	880	270
Thermal power	MW	3000	2100	600
Thermal efficiency	—	0.33	0.42	0.45
Load factor	—	0.8	0.8	1
Operating cycle	Days	295	147	833
Reactor lifetime	Years	60	60	60
Average burnup	MW·d/kg HM	42.7	72.2	—
Average fuel residence time	d	885	441	833
Power density	kW/l	—	430	—
Mass of the core	t HM	70.3	12.5	6
Reloading mass of the core (annual)	t HM	—	8.3	—
Mass of the blanket	t HM	—	33	—
Reloading mass of the blanket (annual)	t HM	—	22	—

TABLE I–14. ISOTOPIC COMPOSITION OF FIRST CORE FRESH FUEL

Parameter	Units	WWER	FBR-A		HTR
			Core	Blanket	
^{232}Th	t HM	—	—	33	5.52
^{233}U	t HM	1.379	—	—	0.48
^{234}U	t HM	0.0414	—	—	—
^{235}U	t HM	0.1362	0.0437	—	—
^{238}U	t HM	67.2899	9.717	—	—
^{238}Pu	t HM	0.0134	0.01364	—	—
^{239}Pu	t HM	0.9148	1.636	—	—
^{240}Pu	t HM	0.3299	0.67	—	—
^{241}Pu	t HM	0.1636	0.2973	—	—
^{242}Pu	t HM	0.0616	0.1118	—	—
^{241}Am	t HM	0.016	—	—	—

TABLE I–15. ISOTOPIC COMPOSITION OF REGULAR RELOADING FRESH FUEL

Parameter	Units	WWER	FBR-A		HTR
			Core	Blanket	
^{232}Th	t HM	—	—	22	5.52
^{232}U	t HM	0.0000391	—	—	—
^{233}U	t HM	0.3794	—	—	0.48
^{234}U	t HM	0.01134	—	—	—
^{235}U	t HM	0.04004	0.02914	—	—
^{238}U	t HM	19.779	6.478	—	—
^{238}Pu	t HM	0.004441	0.009094	—	—
^{239}Pu	t HM	0.3043	1.091	—	—
^{240}Pu	t HM	0.1097	0.4465	—	—
^{241}Pu	t HM	0.05436	0.1982	—	—
^{242}Pu	t HM	0.02045	0.07456	—	—
^{241}Am	t HM	0.005486	—	—	—

TABLE I–16. ISOTOPIC COMPOSITION OF SPENT FUEL

Parameter	Units	WWER	FBR-A		HTR
			Core	Blanket	
^{232}Th	t HM	—	—	21.77	5.72994
^{230}Th	t HM	—	—	2.47E-04	1.3E-06
^{232}U	t HM	2.17E-05	1.73E-08	3.08E-05	2.6E-07
^{233}U	t HM	0.05628	4.51E-09	0.2308	0.14354
^{234}U	t HM	0.03179	2.47E-04	2.11E-03	0.05566
^{235}U	t HM	0.01967	1.76E-02	2.81E-05	0.01025
^{236}U	t HM	0.005917	2.43E-03	2.43E-06	0.00275
^{238}U	t HM	19.153	6.04E+00	—	5.3E-06
^{238}Pu	t HM	0.007949	7.47E-03	—	3.2E-05
^{239}Pu	t HM	0.231	9.85E-01	—	2.9E-06
^{240}Pu	t HM	0.138	4.63E-01	—	1.9E-06
^{241}Pu	t HM	0.08711	1.22E-01	—	7.3E-07
^{242}Pu	t HM	0.04108	7.82E-02	—	4.4E-07
^{241}Am	t HM	0.005849	2.69E-02	—	2.9E-09

TABLE I–16. ISOTOPIC COMPOSITION OF SPENT FUEL (cont.)

Parameter	Units	WWER	FBR-A		HTR
			Core	Blanket	
^{242m}Am	t HM	9.46E-05	1.01E-04	—	—
^{243}Am	t HM	0.01052	5.96E-03	—	—
^{242}Cm	t HM	0.001781	6.27E-06	—	—
^{244}Cm	t HM	0.0055	5.93E-04	—	5E-09
^{245}Cm	t HM	6.00E-04	4.10E-05	—	—
^{237}Np	t HM	0.003	1.84E-03	—	8.8E-05

**COMPARISON OF THREE OPTIONS BASED ON THERMAL REACTORS FOR
MODERATE GLOBAL DEMAND**

This scenario uses the same reactors and conditions as in the case of the high global demand in Section 4.6 and only the rates of growth of nuclear generation differ. As a consequence of a slower growth rate, the limitations on available material differ and may theoretically disproportionally influence results. The figures below contain an evaluation of such possible effects. The scenario for LWRs with MOX (SLWRM) option is used here mainly to compare the shares of NPPs based on recycled fuel with the thorium based option and also to distinguish effects of thorium introduction from those due to reprocessing.

Figure II–1 shows the development trends of power generation for each reactor type for the BAU scenario. ALWRs are introduced in 2015 and gradually replace the conventional LWRs. The SLWRM scenario is constructed as the extension of BAU case through the introduction of recycling plutonium in thermal reactors (see Fig. II–2).

Fractions of reactor types in the LWR12/HWR12 scenario were determined according to availability of only ^{233}U, and the preliminary result of such a thorium introduction ('zero-point of optimization') is shown in Fig. II–3. By 2100, the fraction of thorium based NPPs could amount to ~53% of total nuclear generation. The adjustment of this scenario to plutonium availability decreases the share of thorium utilizing reactors due to the scarcity of reprocessed plutonium resources. By 2100, the share of thorium based NPPs reaches only ~24% of total nuclear generation, as shown in Fig. II–4.

Figure II–5 gives cumulative natural uranium consumption for BAU, SLWRM and LWR12/HWR12 cases. In the BAU case, total annual natural uranium demand reaches approximately 24 million tonnes by 2100. The situation regarding uranium consumption improves through introduction of either MOX (SLWRM) or ThFC

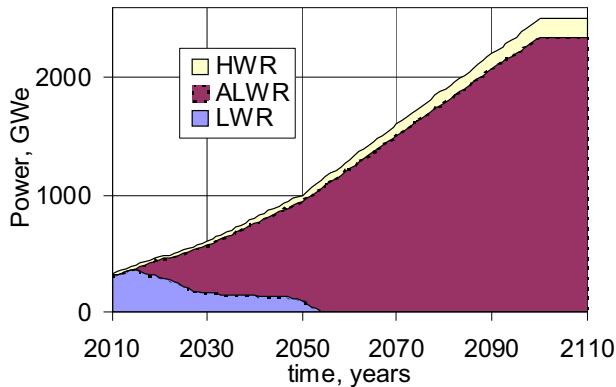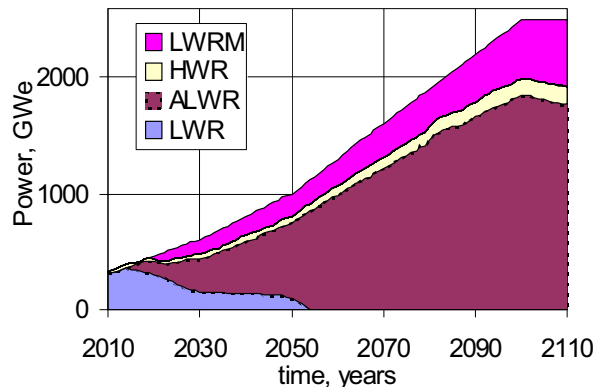*FIG. II–1. Nuclear power demand structure, BAU.*

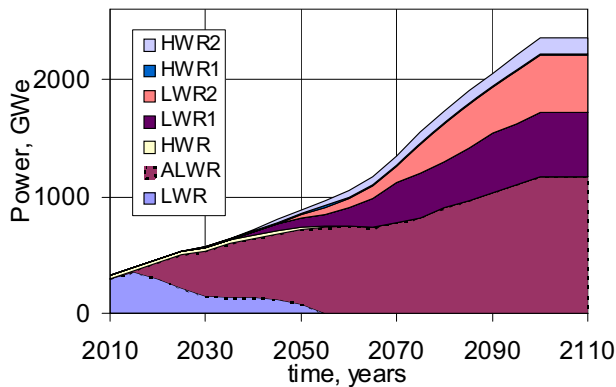*FIG. II–2. Nuclear power demand structure, SLWRM.*

FIG. II–3. Nuclear power demand structure, 0-point of optimization, LWR12/HWR12.
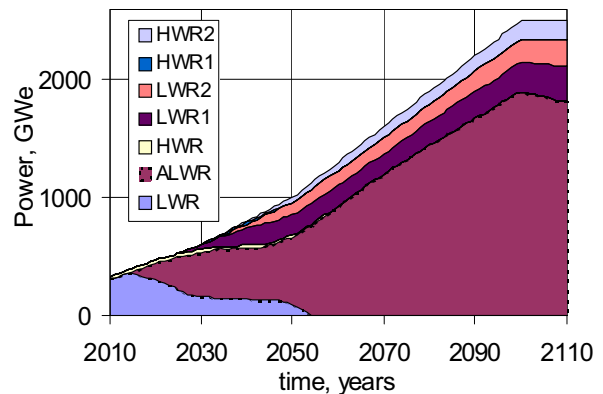

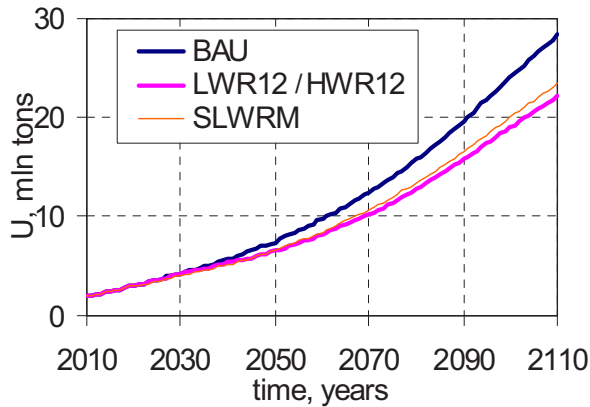*FIG. II–4. Nuclear power demand structure, LWR12/HWR12.*

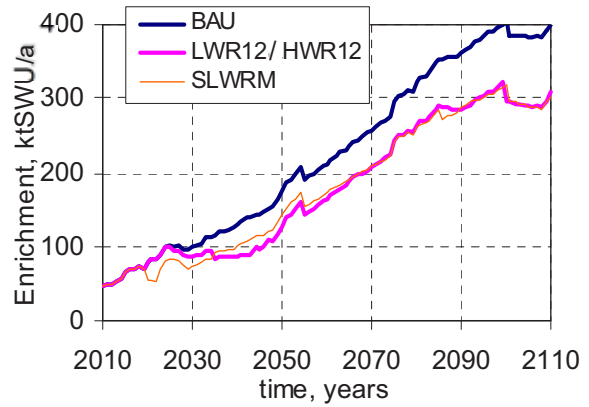FIG. II–5. Cumulative natural uranium requirements.

FIG. II–6. Annual enrichment requirements.
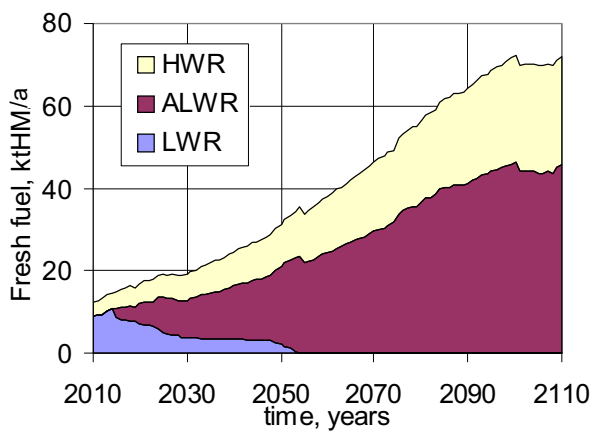
FIG. II–7. Fresh fuel requirements, BAU case.
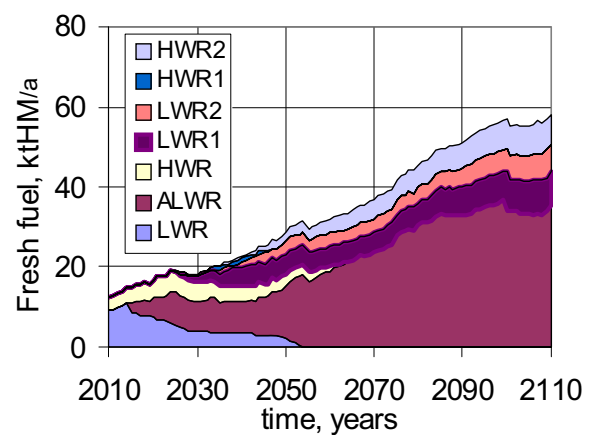
FIG. II–8. Fresh fuel requirements, LWR12/HWR12 case.

thermal reactors (LWR1/HWR12). In both cases, the cumulative consumption of uranium becomes ~20% lower by the end of the century.

Figure II–6 shows the annual separation work to be conducted for uranium enrichment. The separation work necessary for both recycling based thermal options (LWR1/HWR12 and SLWRM) is about 25% lower than in the BAU scenario by 2100 .

Annual requirements for fuel fabrication in the BAU scenario are shown in Fig. II–7. The reactors of LWR type need ~46 kt HM/a of fabricated fuel and HWRs need ~26 kt HM/a by 2100. The estimation of fresh fuel parameters for the SLWRM option gives roughly the same numbers.

In the case of the thorium utilizing option, the fuel fabrication facilities should provide ~34 kt HM/a for customary ALWRs and ~23 kt HM/a for thorium based reactors by 2100 (see Fig. II–8). Fuel fabrication capacities for ThFC reactors come to ~40% of total value.

In both the LWR1/HWR12 and SLWRM option, the spent fuel unloaded from all reactor types including HWRs is to be reprocessed. The estimation of reprocessing needs distributed among the spent fuel of reactor types in the thorium using case LWR12/HWR12 is presented in Fig. II–9. By 2100, the annual requirement for reprocessing of spent fuel from the reactors utilizing Th/^{233}U fuel together with other fuel types attains 20 kt HM/a, i.e. approximately 38% of total value.

The total amount of spent fuel accumulated by 2100 in the BAU scenario reaches 3.4 million t HM (see Fig. II–10). Introduction of Th/^{233}U fuel and continuous recycling of ^{233}U and plutonium in the LWR1/HWR12 (see Fig. II–12) case causes a drop of spent fuel accumulation from ~3400 to ~260 kt (mainly due to reprocessing); nevertheless, high level waste treatment would be still an issue, but at least the fissile isotopes (plutonium and ^{233}U) would be used in reactors. The SLWRM scenario has the same effects, with the exception of ^{233}U recycling (see Fig. II–11).
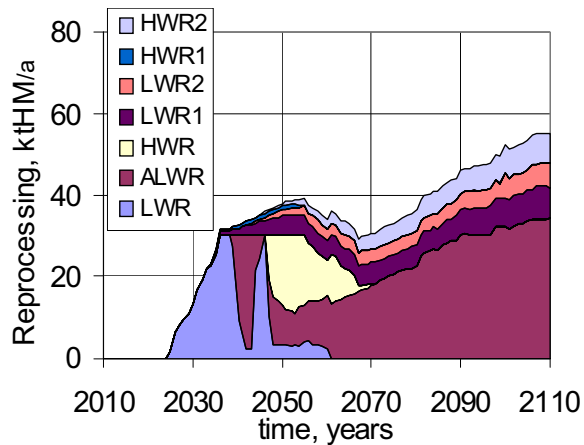
FIG. II–9. Reprocessing requirements, LWR12/HWR12 case.
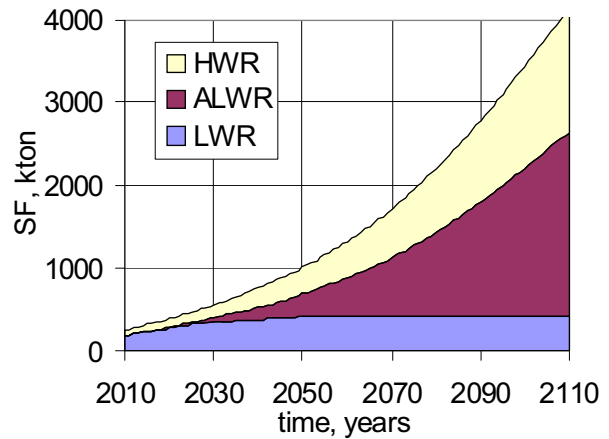
FIG. II–10. Spent fuel storage, BAU.
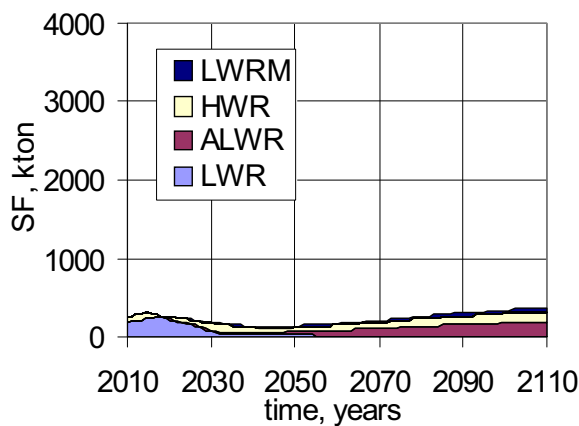
FIG. II–11. Spent fuel storage, SLWRM case.
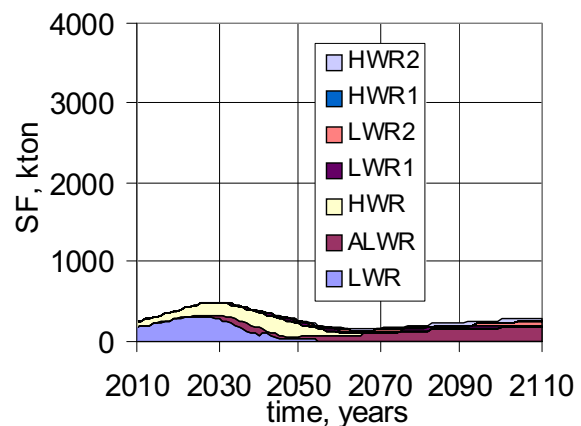
FIG. II–12. Spent fuel storage, LWR12/HWR12 case.

The amount of plutonium and minor actinides to be discharged in every option is shown in Figs II–13 and II–14. The lowest amount of plutonium and minor actinides is to be discharged in the BAU case, although all plutonium produced would be accumulated in the storage facilities or depositories. There is no plutonium accumulation in the SLWRM and LWR12/HWR12 cases because plutonium consumption corresponds with a short delay to its production. The amount of plutonium handled continuously in the fuel cycle is lower for the LWR12/HWR12 case, and by 2100, the consumption of plutonium reaches 0.9 kt/a. Annual discharge of minor actinides in the LWR12/HWR12 scenario rises by 23% in comparison to the BAU scenario (from 0.061 to 0.075 kt/a). According to the defined boundary conditions, all minor actinides are kept in spent fuel without recycling.

^{233}U balance (discharged/used) for the LWR12/HWR12 option is shown in Fig. II–15. Almost all reprocessed ^{233}U is used with a short delay (five year), and by 2100, annual discharge of ^{233}U is ~0.22 kt/a.

The economic results of Section 5 have also been applied to the LWR12/HWR12 option here to obtain a reasonable economic scenario. An optimized structure of nuclear generation was obtained on the basis of input data (availability and price) for resources presented in Table 4.4 and for reactors and fuel cycles (reactor and fuel costs) presented in Section 5. The resulting structure of global nuclear energy systems for the LWR12/HWR12 scenario is shown in Fig. II–16. By 2100, the share of thorium based NPPs reaches around 8% of the total nuclear generation mainly because of HWRs (6%).
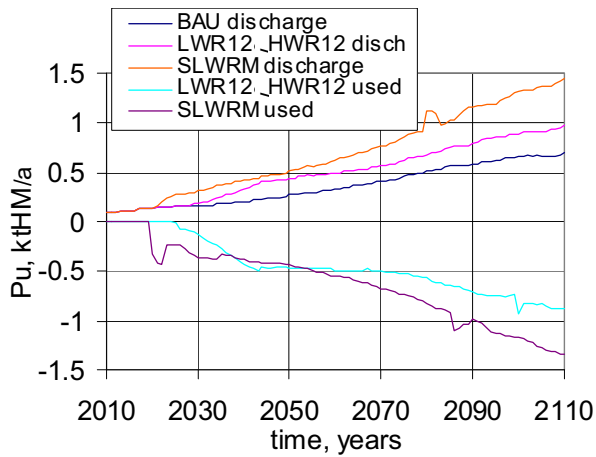
FIG. II–13. Plutonium discharged/used annually.
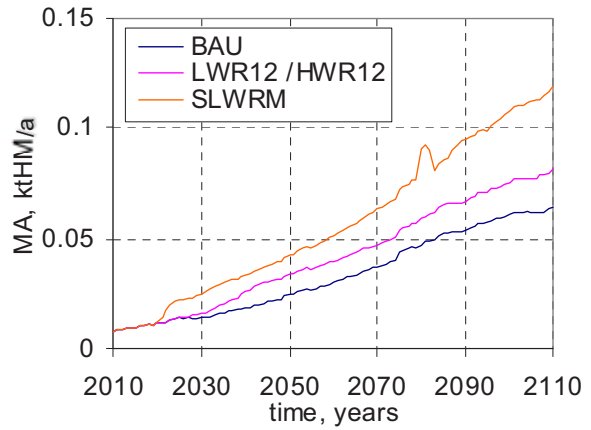
FIG. II–14. Minor actinide annual discharge.
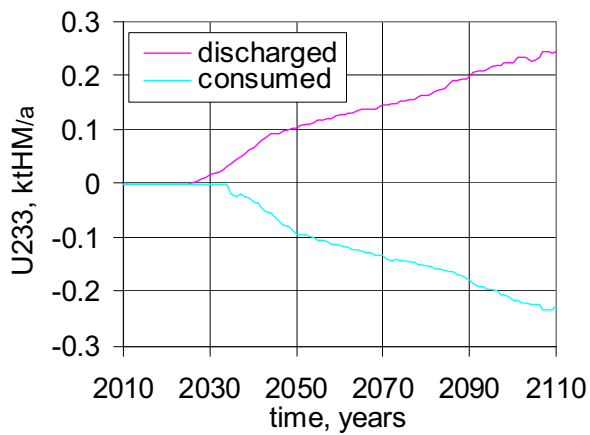
FIG. II–15. U^{233} balance. LWR12/HWR12 case.
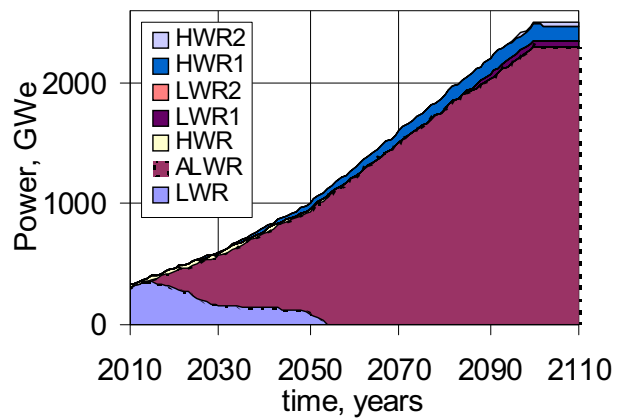
FIG. II–16. Nuclear power demand structure.

COMPARISON OF THE MODERATE TO THE HIGH CASE SCENARIO

The high case scenario for the SLWRM and LWR12/HWR12 case was discussed in Section 4.6. A comparison with the results for the moderate scenario presented above shows no significant influence of total capacity reached at the end of the 21st century.

Annex III

COMPARISON OF OPTIONS USING FAST REACTORS FOR MODERATE GLOBAL DEMAND

Two options that were considered in Section 4.8 for a high global demand scenario (SFR and FRTh/LWR12/HWR12) are now evaluated in a moderate forecast of electricity demand. The rate of growth of nuclear generation is slower in the moderate scenario, and the variation of nuclear material available for reprocessing may theoretically disproportionally influence the final result.

According to conditions defined in the GAINS project:

— All spent fuel is allowed to be reprocessed in SFR and FRTh/LWR12/HWR12 scenarios with the exception of the spent fuel from HWRs, which is not reprocessed in the SFR scenario.
— In cases of a lack of plutonium produced in the reactors of a given option, the transition to thorium can be assumed to be made through the incineration of civilian grade plutonium and achieving reduction of existing spent fuel stockpiles.
— The HWR share is traditionally kept at the 6% level of the total thermal reactor electricity production.

Nuclear generation structure for the SFR case is shown in Fig. III–1. By 2100, the share of fast reactors can attain ~58% of global nuclear energy capacity. Figure III–2 represents the structure of the nuclear energy system in the FRTh/LWR12/HWR12 scenario optimized only according to ^{233}U availability (called 'zero-point of optimization', as in Section 4.8).

An adjustment considering additionally plutonium availability decreases the share of thorium utilizing reactors to 27% by 2100, as shown in Fig. III–3; this share is divided approximately 50/50 between fast and thermal reactors. The structure adjustment was performed only for material flows to ensure availability of necessary nuclear material in the fuel cycle, and economic parameters were not considered.

Figure III–4 demonstrates cumulative natural uranium consumption for BAU, SFR and FRTh/LWR12/HWR12 cases. At the end of the century, the cumulative consumption of uranium drops from ~24 million tonnes in the BAU case to ~16 million tonnes in the SFR and FRTh/LWR12/HWR12 cases. Also, transition to the NES comprising fast reactors either with thorium or without has a positive effect on the amount of separation work necessary. The estimated requirements of enrichment capacities for the SFR scenario drops by a factor of 2.7 compared to the BAU case; in the case of FRTh/LWR12/HWR12, it drops by a factor of 1.9 (see Fig. III–5).

Annual requirements for fuel fabrication in the SFR scenario are shown in Fig. III–6. The amount of fuel to be fabricated by 2100 drops from 72 kt HM/a (LWRs, 46 kt HM/a and HWRs, 26 kt HM/a) in the BAU case to 62 kt HM/a (MOX and blanket fuel35 kt HM and UOX fuel, 27 kt HM) in the SFR case. The calculation of the total fresh fuel volume for the FRTh/LWR12/HWR12 option gives roughly the same value (see Fig. III–7). In the case of the thorium utilizing option, FRTh/LWR12/HWR12, the reactors that use Th/^{233}U fuel (together with
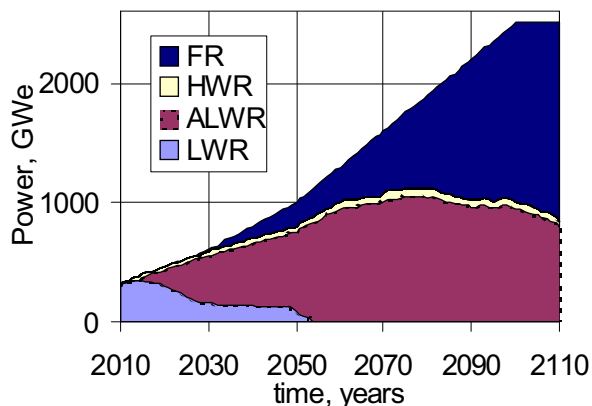
FIG. III–1. Nuclear power structure, SFR.
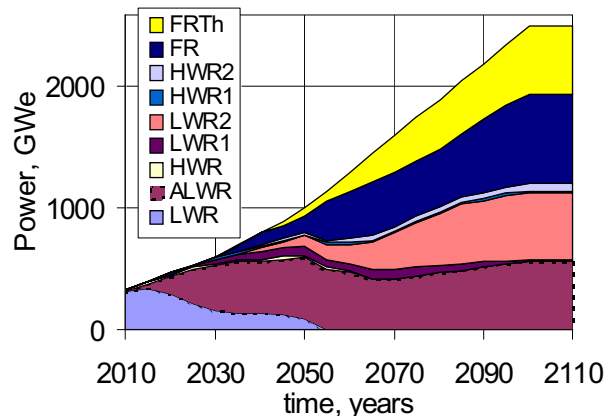
FIG. III–2. Nuclear power demand structure, 0-point of optimization, FRTh/LWR12/HWR12.
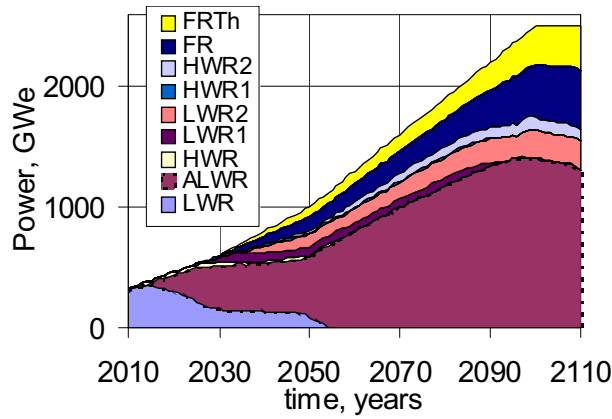
FIG. III–3. Nuclear power demand structure, FRTh/LWR12/HWR12.

FIG. III–4. Cumulative natural uranium requirements.
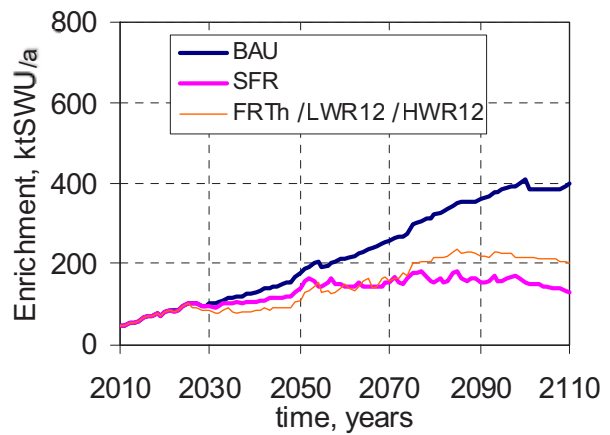
FIG. III–5. Annual enrichment requirements.

FIG. III–6. Fresh fuel requirements, SFR case.

FIG. III–7. Fresh fuel requirements, FRTh/LWR12/HWR12 case.

uranium–plutonium fuel) will consume 42% of the fabricated fuel producing only 27% of the electricity (see above).

Both advanced scenarios, SFR and FRTh/LWR12/HWR12, envisage large scale deployment of spent fuel reprocessing (see Figs III–8, III–9). By the end of the century, annual reprocessing should comprise 62 ktHM/a according to the SFR scenario and 59 kt HM/a according to the FRTh/LWR12/HWR12 option. In the first case, the share of reprocessed spent fuel from fast reactors reaches 52% and in the second case, the share of reprocessed spent fuel from Th/^{233}U utilizing reactors amounts to 41%.
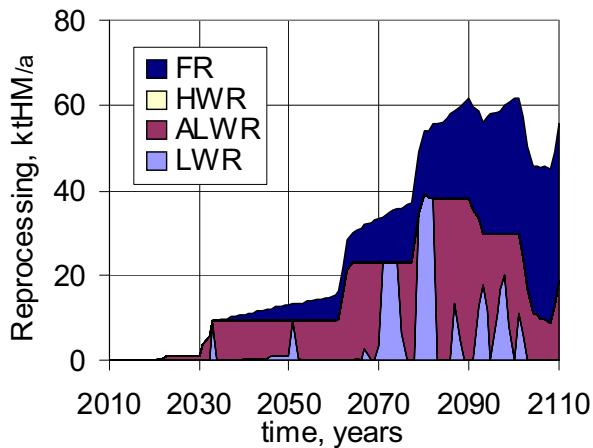
FIG. III–8. Reprocessing requirements, SFR case.
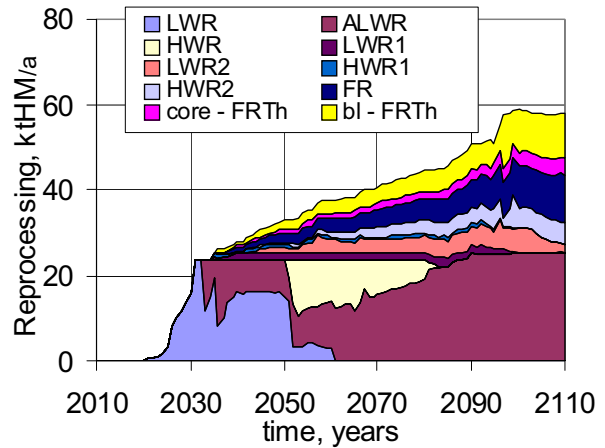
FIG. III–9. Reprocessing requirements, FRTh/LWR12/HWR12 case.
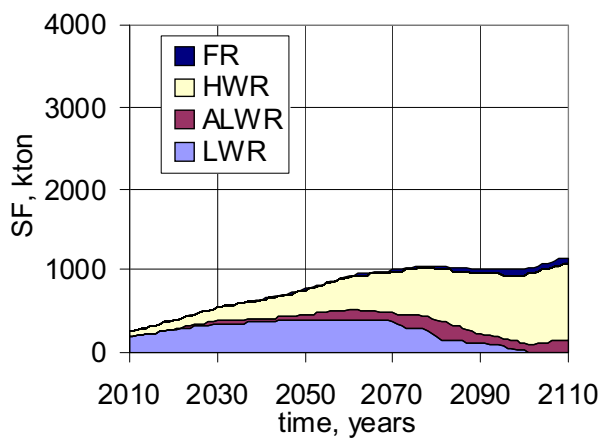
FIG. III–10. SF storages, SFR case.
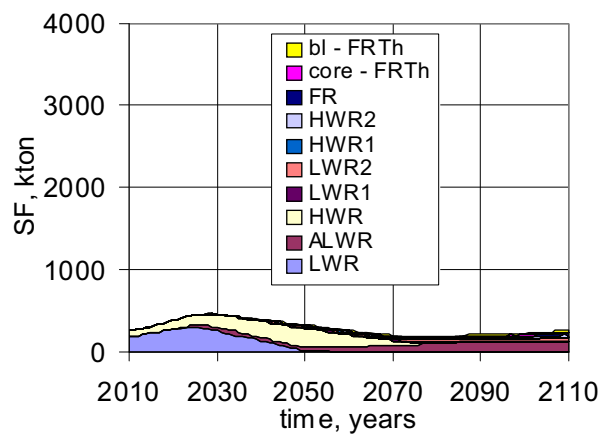
FIG. III–11. Spent fuel storage, FRTh/LWR12/HWR12 case.

Due to very active reprocessing by the end of the century, the total cumulative amount of spent nuclear fuel drops from 3.4 million tonnes for the BAU case to about one million tonnes for the SFR case (of which spent fuel from HWRs is about 80%), as presented in Fig. III–10. The FRTh/LWR12/HWR12 scenario requires total reprocessing of all spent fuel (including that from HWRs unlike in the SFR case), and only 0.23 million tonnes would be stored by 2100 due to the need to decrease residual heating (see Fig. III–11); although high level waste treatment would still be an issue, at least the fissile isotopes (plutonium and ^{233}U) would be used in reactors.

The discharged and used plutonium for all three options is compared in Fig. III–12. In the SFR option, all plutonium unloaded from reactors save HWRs will be used immediately after reprocessing. In the FRTh/LWR12/HWR12 option, all plutonium — including that from HWRs — will be also recycled without accumulation. By 2100, the discharge of plutonium in the FRTh/LWR12/HWR12 case is lower than in the SFR case by a factor of 1.8.

Data on annual discharge of minor actinides are shown in Fig. III–13. In all three scenarios, they are approximately identical (about 0.06 kt/a by 2100). According to the defined condition for all scenarios, all minor actinides are kept in spent fuel without recycling. ^{233}U balance for FRTh/LWR12/HWR12 scenario (see Fig. III–14) assumes that all reprocessed ^{233}U is to be utilized immediately, and by 2100, annual discharge of ^{233}U reaches ~0.19 kt/a.

The optimization of the structure of reactors through the consideration of economic parameters has been achieved in the same way as for the options at the end of Sections 4.7, 4.8 and in Annex II of this report. The economic results of Section 5 in the SFR and FRTh/LWR12/HWR12 moderate scenarios are reflected in Fig. III–15. In case the cost of NPPs and fuel cycles correspond to the values introduced in Chapter 5 the cost of
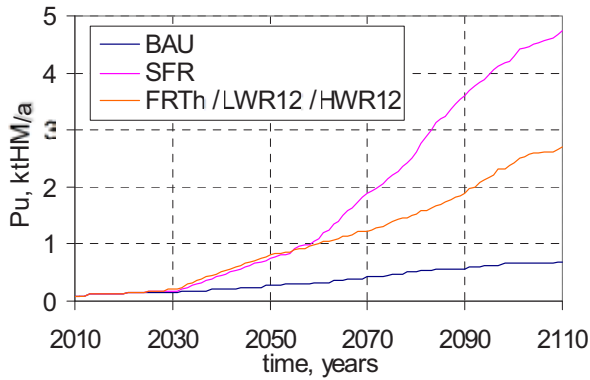
FIG. III–12. Plutonium discharged annually.
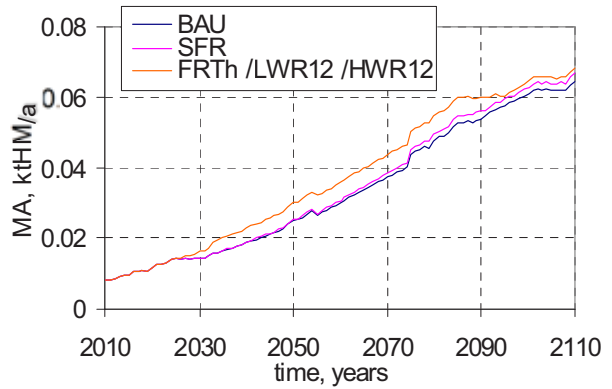
FIG. III–13. Minor actinide discharged annually.
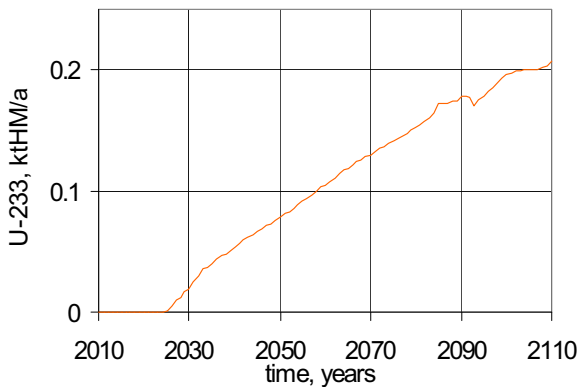
FIG. III–14. ^{233}U discharge, FRTh/LWR12/HWR12 case.
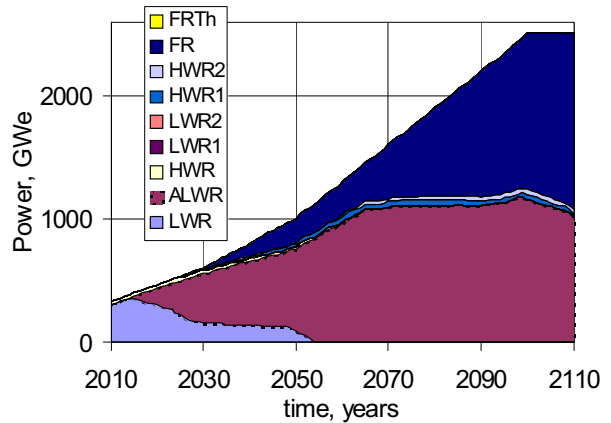
FIG. III–15. Nuclear power demand structure, economics optimized, FRTh/LWR12/HWR12.

electricity in a domain of light water and fast reactors provided by the thorium utilizing reactors (as considered in FRTh/LWR12/HWR12 scenario) would not be competitive with ALWR and fast reactor electricity cost; therefore, thorium using reactors would be eliminated by the end of the century. In the domain of HWRs, thorium utilizing reactors would dominate.

COMPARISON OF MODERATE TO HIGH CASE SCENARIO RESULTS

Results of the high case scenario for the SFR and FRTh/LWR12/HWR12 case were presented in Section 4.8. In comparing these results with those of the moderate scenario above, no significant influence of the demand for nuclear capacity on the results could be found.

Annex IV

THORIUM THERMAL BREEDER INTRODUCTION SCENARIO, FRTh/LWR3 OPTION FOR HIGH GLOBAL DEMAND

The Shippingport Atomic Power Station in Pennsylvania was one of the first commercially operated NPPs in the world. Originally, the reactor had been designed for the use of uranium fuel and the application of the reactor could be twofold: either production of electricity or as a propulsion engine for naval vessels. Later, this reactor was transformed into the light water breeder reactor (LWBR).

The Shippingport LWBR was developed by the Pittsburgh Naval Reactors Office and operated at Shippingport Atomic Power Station from 1977 to 1982. This prototype reactor was developed to prove the concept of a pressurized water breeder reactor. The fuel consisted of binary ceramic pellets with 1 to 5% uranium in a thoria matrix. The uranium contained more than 98% of ^{233}U, which was the only uranium isotope believed to enable breeding in a LWR [IV–, IV–2].

The calculations based on the data published in Ref. [IV–2] reveal that the Shippingport NPP had a thermal efficiency at a level of only 16%. Assuming that this is not linked to the type of fuel cycle used, a thermal efficiency is assumed at the level of other modern LWRs (32%) for the purpose of this study.

An important feature of the LWBR is a huge specific core mass (the mass of the core per unit of installed capacity). For most modern PWRs. this value lies at a level of up to 0.07 kg HM/kW(e), and in Shippingport's case, it is more than four times higher and reaches 0.3 kg HM/kW(e) even with the amended thermal efficiency.

TABLE IV–1. LWR3 TECHNICAL PARAMETERS

| No.	Parameter	Value
1	Electric capacity, MW	139.6
2	Thermal efficiency, %	32*
3	Load factor, %	80
4	Average burnup, MW·d/kg HM	12.4
5	Fuel residence time, EFPD	1208
6	Mass of the core, t HM	42.5
7	Annual reloading, t HM/a	10.3
8	Pu content in fresh fuel	—
9	Th content in fresh fuel	0.9982
10	^{233}U content in fresh fuel (average)	0.0118
11	Pu content in spent fuel, kg/GW·a	—
12	MA content in spent fuel, kg/GW·a	—
13	^{233}U content in spent fuel, kg/Gw·a	1084
14	Natural U consumption, t/Gw·a	—
15	SWU requirements, tSWU/Gw·a	—
16	FF requirements, kg/Gw·a	92 000
17	Pu used, kg/Gw·a	—
18	^{233}U used, kg/Gw·a	1084
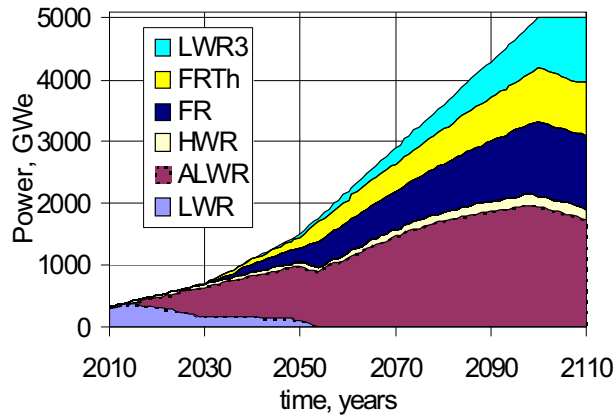
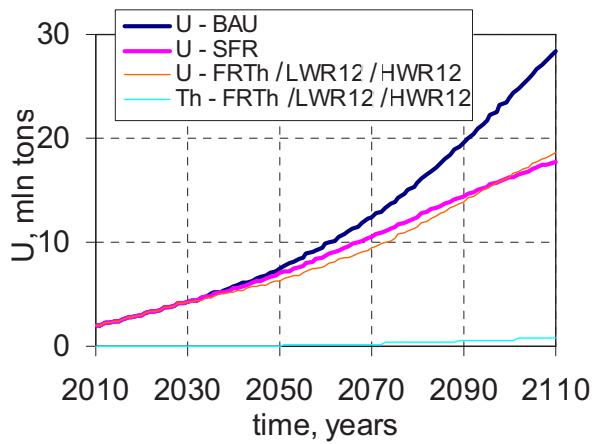*FIG. IV–1. Nuclear power demand structure, FRTh/LWR3.*
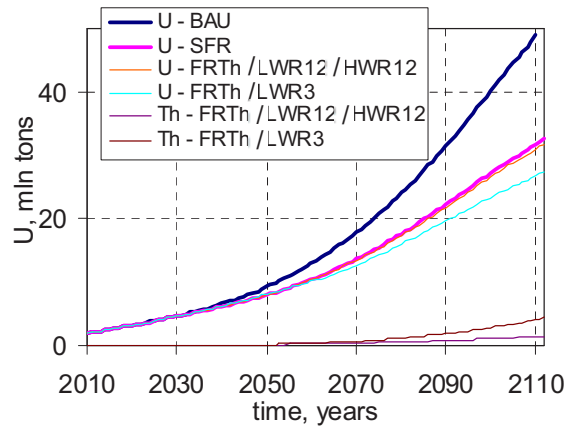
FIG. IV–2. Cumulative natural uranium requirements.
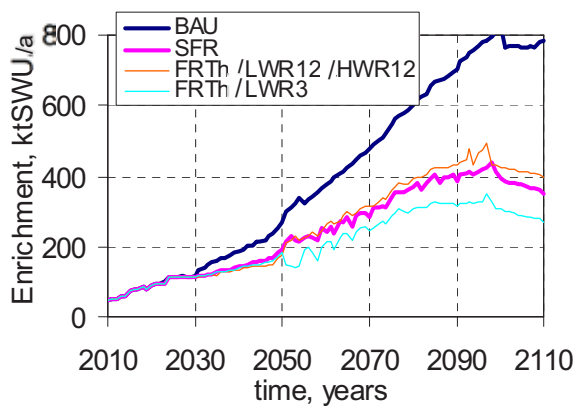
FIG. IV–3. Annual enrichment requirements.
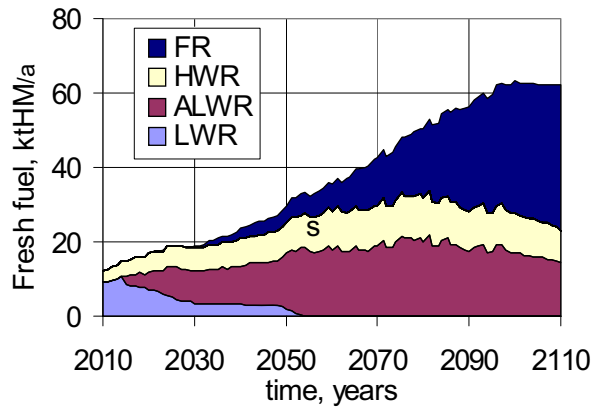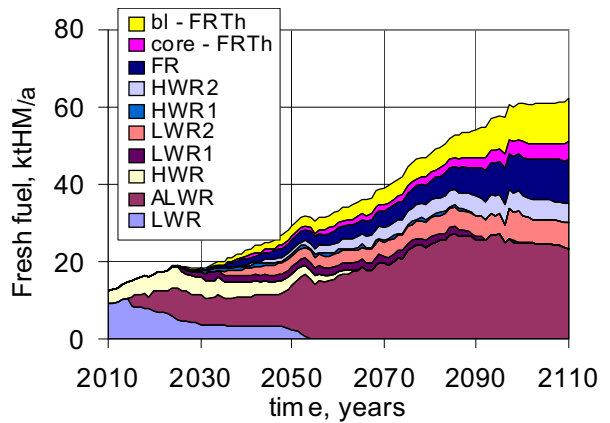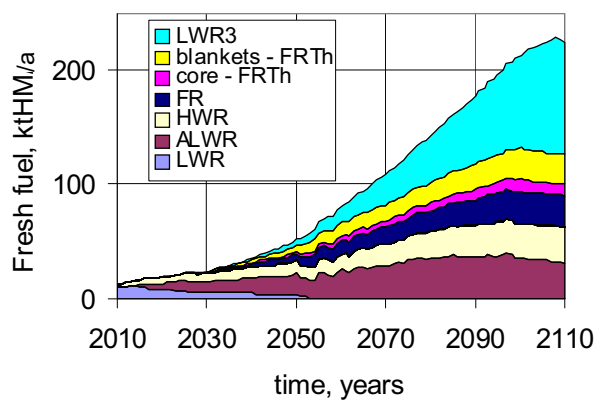

FIG. IV–4. Fresh fuel requirements, FRTh/LWR3 case.

It was reported that the LWBR had achieved a conversion rate of 1.01 (i.e. breeding) and had been virtually self-sufficient in steady state from the point of view of ^{233}U consumption and production. Nonetheless, in a scenario, it requires ^{233}U to be supplied to ensure necessary growth of capacity of electricity generation and therefore the fast reactor with thorium blanket as a possible source of ^{233}U was chosen. All spent fuel is allowed to be reprocessed, and plutonium is available without limits, i.e. in the case of a lack of plutonium produced in the reactors of a given option, the incineration of civilian grade plutonium, achieving reduction of existing spent fuel stockpiles is assumed. HWR's share is kept at 6% level of total thermal reactor's electricity production.

Development of the nuclear power generation structure in the FRTh/LWR3 case is presented in Fig. IV–1. At this stage, the optimization of the system structure was performed considering material flow only (^{233}U and plutonium availability) but without any economic considerations. By 2100, the share of thorium fuel utilizing NPPs reaches about 50% of total nuclear generation. The cumulative consumption of natural uranium of the BAU, SFR, FRTh/LWR12/HWR12, and FRTh/LWR3 is shown in Fig. IV–2. It shows a drop in demand from 40 million tonnes in the BAU case to 23 million tonnes for the FRTh/LWR3 case by the end of the century. This drop is even more significant than that achieved by fast breeders (SFR) without thorium blanket (down to 27 million tonnes).

Figure IV–3 shows the annual separation work necessary for uranium enrichment. In both the SFR and the FRTh/LWR12/HWR12 scenario by 2100, the enrichment requirements are at a level of 400 kt SWU/a, i.e. two times lower than in the BAU case. In the FRTh/LWR3 case, this value is even lower and amounts to only 300 kt SWU/a.

By 2100, in the FRTh/LWR3 scenario, the annual fuel production (see Fig. IV–4) would reach 214 ktHM/a (including 110 ktHM/a for reactors usng thorium), which is the highest value among all scenarios considered (BAU — 146 kt HM/a, SFR — 128 kt HM/a, FRTh/LWR12/HWR12 — 123 kt HM/a).
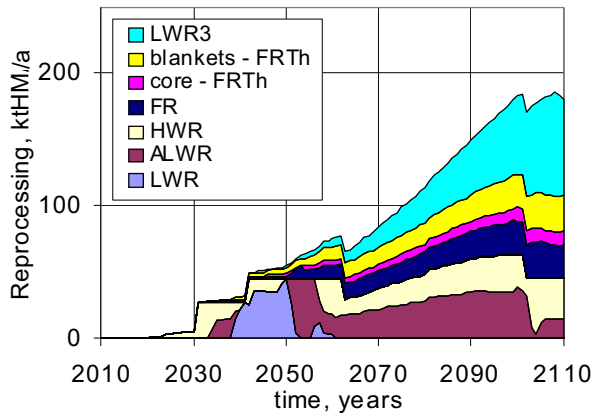
FIG. IV–5. Reprocessing requirements, FRTh/LWR3 case.
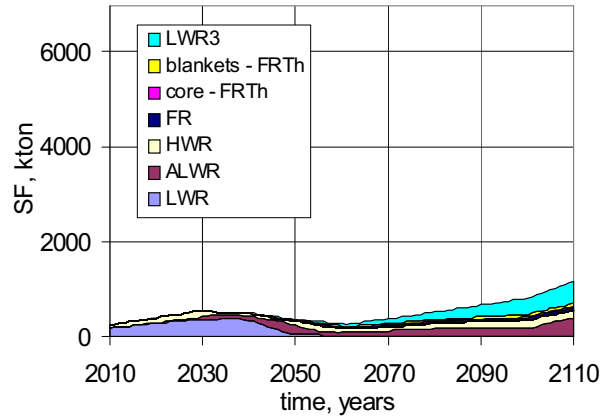
FIG. IV–6. Spent fuel storage, FRTh/LWR3 case.
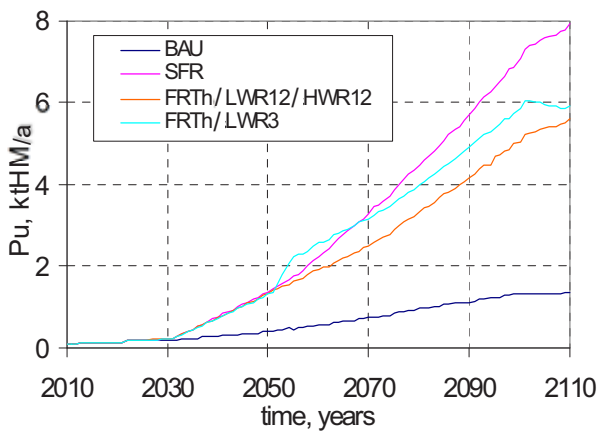
FIG. IV–7. Plutonium discharged annually.
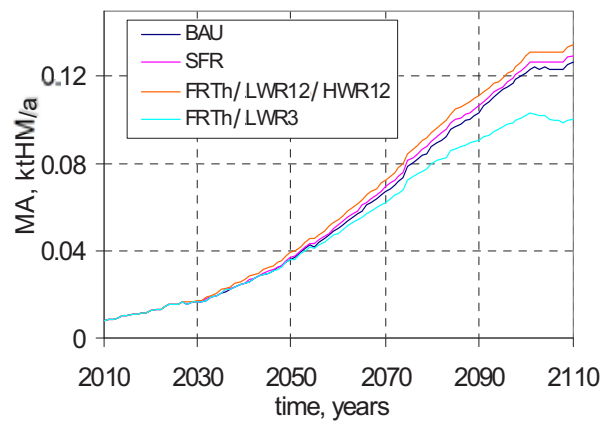
FIG. IV–8. Minor actinides discharged annually.

The amount of spent fuel that should be reprocessed annually from every reactor type considered is given in Fig. IV–5. By 2100 in the FRTh/LWR3 scenario, 183 kt HM/a of spent fuel (the fraction of fuel from Th/^{233}U utilizing reactors is 52%) should be reprocessed annually, which is also the highest value among all scenarios considered (SFR — 94 kt HM/a, FRTh/LWR12/HWR12 — 116 kt HM/a). As a result of total reprocessing in the FRTh/LWR3 case, the accumulation of spent fuel is not significant and by 2100 reaches 0.8 million tonnes (see Fig. IV–6).

Figure IV–7 demonstrates the amount of plutonium to be unloaded from the reactors. All reprocessed plutonium should be recycled immediately without accumulation. By 2100 in the FRTh/LWR3 scenario, the annual discharge of plutonium would reach 6 kt/a. The annual discharge of minor actinides (see Fig. IV–8) in the FRTh/LWR3 case is 0.1 kt/a, which is lower than 0.12 kt/a achieved by the BAU, SFR, and FRTh/LWR12/HWR12 scenario.

By the end of the century, the annual discharge and utilization of ^{233}U rises to 1.2 kt/a in the FRTh/LWR3 case and is three times higher than the same parameter in the FRTh/LWR12/HWR12 case (see Fig. IV–9).
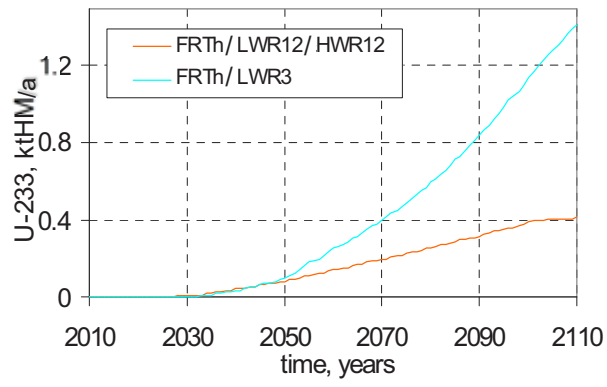
FIG. IV–9. U²³³ discharged annually.

COMPARISON OF LIGHT WATER REACTOR, ADVANCED LIGHT WATER REACTOR AND ADVANCED HEAVY WATER REACTOR IN ONCE-THROUGH OPTIONS

Thorium is a fertile material that needs neutrons to be converted to fissile ^{233}U. Since heavy water reactors are highly neutron effective systems, they could be a promising option to host thorium fuel. However, the current heavy water constituent of nuclear generation is rather low and there is no indication on an optimistic prognosis for the future. Nevertheless, in anticipation of the share of HWRs possible growing beyond today's 6%, the benefits of thorium in a once-through heavy water option will be considered. The AHWR provides a series of options of thorium fuel utilization including well-considered a once-through option without reprocessing.

Two types of LWRs have been selected for comparison: LWR is the reactor with parameters that correspond to an averaged modern status of the type and ALWR that was proposed in the framework of GAINS as a tentative level of light water technology for the next hundred years. As stated in Section 4.4, the characteristics of the ALWR are not properly proven. Although there are still concerns about its feasibility and the ALWR design is less mature than that of the AHWR, the ALWR is selected as the best possible traditional once-through option.

All three options are considered in a scenario regime, i.e. as a transition from current status of nuclear generation structure and following development until the end of the century to meet the demand curve. The GAINS group of countries G1b has been chosen as an example. It is assumed that 6% heavy water domain will be filled with AHWRs a priori and only the remaining 94% is subject to consideration. In the study, only material flow parameters were taken into account and any economic characteristics were not considered.

Nuclear generation structures for LWR, ALWR and AHWR cases are shown in Figs V–1 to V–3. There are no obstacles to any option becoming the sole supplier of electricity in the group considered. Cumulative requirements
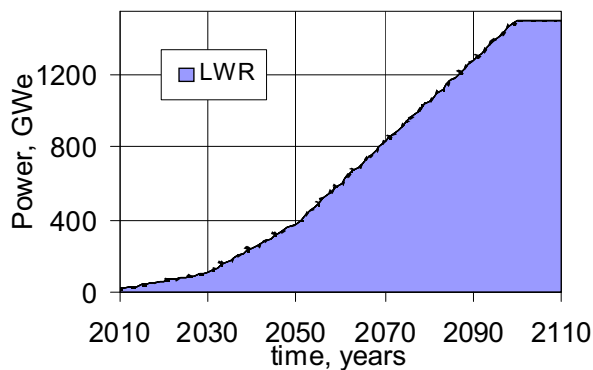
FIG. V–1. Nuclear generation structure LWR.
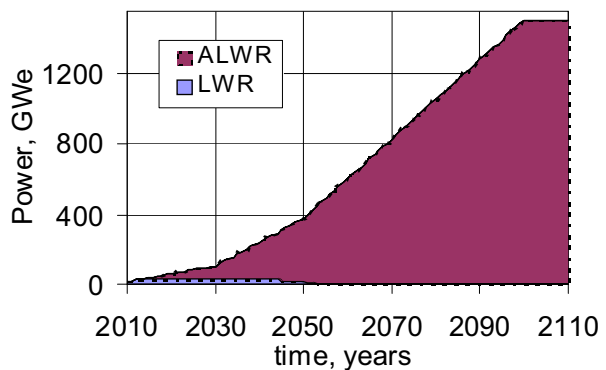
FIG. V–2. Nuclear generation structure ALWR.
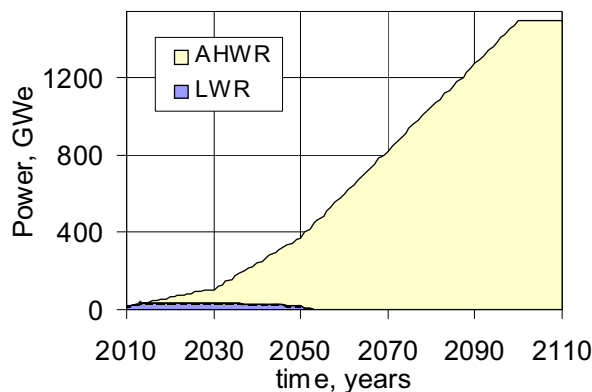
FIG. V–3. Nuclear generation structure AHWR.
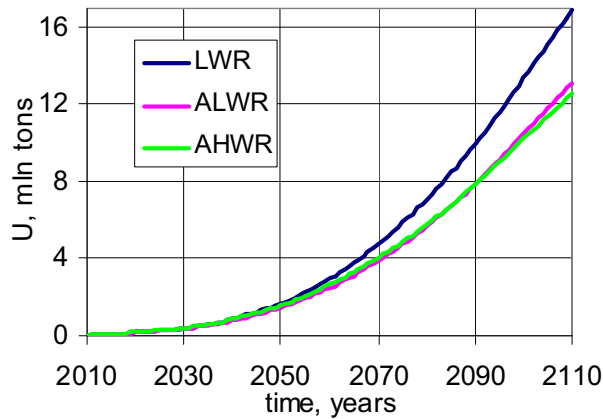
FIG. V–4. Cumulative uranium consumption.
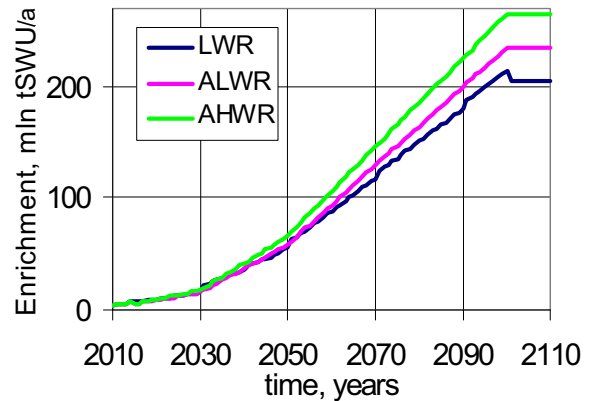
FIG. V–5. Enrichment requirements.
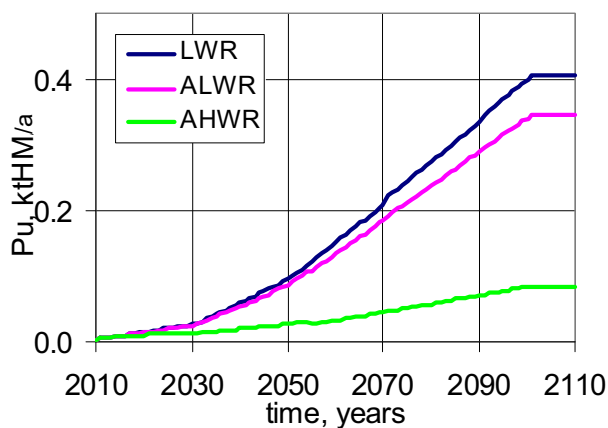
FIG. V–6. Unload of plutonium.
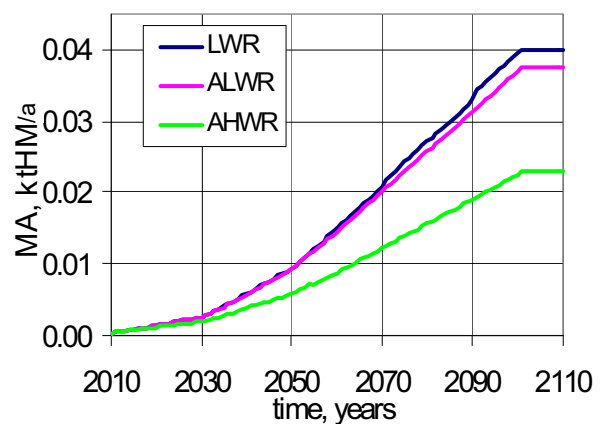
FIG. V–7. Unload of minor actinides.

of natural uranium for every system option are shown in Fig. V–4. By the end of the century, the total mass of consumed natural uranium would reach approximately 13.4 million tonnes for the LWR case, 10.5 million tonnes for the ALWR case and 10.25 million tonnes for the AHWR case.

Figure V–5 gives the result of estimation of the annual separation work requirements for uranium enrichment. By 2100, the separation work necessary for both innovative options, i.e. ALWR and AHWR, would be ~9% and ~24% higher respectively than in the LWR scenario. The amounts of plutonium and minor actinides to be discharged in every scenario are shown in Figs V–6 and V–7. The lowest masses of discharged plutonium and minor actinides are attained in the AHWR scenario, and the biggest ones in the LWR case. Compared to the LWR case, annual discharge of plutonium drops by a factor of ~1.17 for the ALWR case and ~4.8 for the AHWR case. Annual minor actinide discharge drops by factors of ~1.07 and ~1.76, respectively. The ^{233}U balance (discharged/used) for the ALWR option is demonstrated in Fig. V–8. In this case all reprocessed ^{233}U is left in spent fuel without reprocessing and annual discharge of ^{233}U attains 0.34 kt HM/a by 2100.
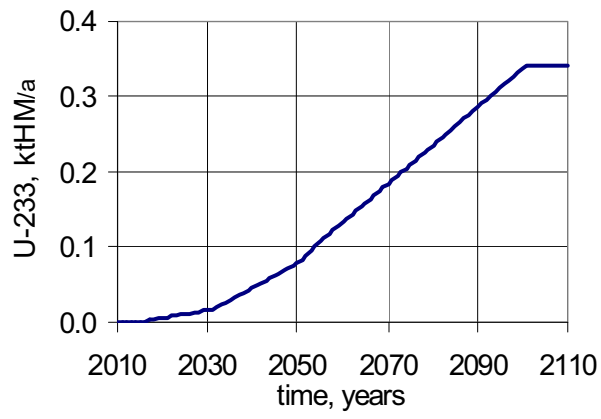
FIG. V–8. Unload of [233]*U from AHWR.*

Annex VI

**SCENARIOS FOR THE THORIUM NUCLEAR FUEL CYCLE
IN HEAVY WATER REACTORS (CANADIAN STUDY)**

VI–1. DISCUSSION OF SCENARIO OPTIONS

This scenario study is based on the results achieved in AECL for HWR type of reactors [see Refs 13, 36–38, 42, 75 in the main text]. The CANDU reactor has a high degree of fuel cycle flexibility as a result of its high neutron economy, fuel channel design, on-power refuelling, and simple fuel bundle design. These features exist in both the traditional CANDU reactor designs, such as the CANDU-6 reactor, and in the Generation III+ design, the Advanced CANDU Reactor (ACR-1000) [76]. These features facilitate the implementation and exploitation of ThFCs.

One potential implementation of a ThFC is through a mixture of plutonium and thorium dioxide in CANDU fuel bundles. Since thorium has no fissile isotope, plutonium must be added as a driver fuel until enough ^{233}U is bred into the fuel to sustain the reaction. Enriched uranium could also potentially be used as a driver fuel, but the studies here consider plutonium only as a driver to simplify the reprocessing. Separating ^{233}U from plutonium/thorium fuel requires only chemical separation, whereas a homogeneous fuel of LEU/Th would contain other uranium isotopes that would dilute the ^{233}U and would significantly complicate analysis and fuel cycle.

The once-through thorium (OTT) fuel cycle provides an evolutionary approach to the implementation of ThFCs [77]. This fuel cycle can take advantage of the energy potential of thorium without requiring recycling of the spent fuel. A reserve of the valuable fissile ^{233}U can be produced before the facilities to reprocess the thorium are available, creating a resource that is ready to be exploited when the technology becomes commercially available.

The OTT fuel cycle is the easiest to implement and can be readily accomplished prior to the implementation of spent fuel reprocessing and recycle. The primary objectives of this first pass thorium fuel are to:

— Consume the current stockpile of civilian plutonium and use it for electricity production.
— Remove the proliferation potential of the civilian plutonium stockpile.
— Develop experience in the use of thorium fuel in power reactors.
— Create a reservoir of spent fuel containing ^{233}U that can be used to initiate the closed cycle following the implementation of reprocessing.

To study the long term impact of implementing ThFCs , system scenario assessments have been performed. In this case, the differences between maintaining the current reactors and fuel cycles to 2100 (i.e. mostly LWRs, with a smaller number of HWRs, all fuelled with uranium) is compared with scenarios in which plutonium driven thorium CANDU-6 reactors both with and without ^{233}U recycle are added into the reactor combination. The effects on natural uranium consumption and on decay heat from actinides are also presented. Three thorium fuelled reactor scenarios were investigated to determine how they affected resource availability over the next hundred years. The scenarios and reactors described here are similar to those appearing in chapter 4. The nuclear scenario evaluation was carried out using the systems analysis code DESAE 2.2 [78, 79].

VI–2. BUSINESS AS USUAL CASE

The scenarios in national study comprise a BAU case (moderate demand) and two variations on it in which thorium fuelled HWR replaced natural uranium fuelled HWRs and LEU fuelled LWRs. The BAU case (as it was introduced in Section 4.2) was simulated with a couple of insignificant simplifications: all reactors in the world are generic LWRs and generic HWRs (see Table VI–1) with capacity factors of 85% and reactor lifetimes of 60 years; enrichment tails are 0.3% ^{235}U/U throughout the scenario. The natural uranium requirements of the BAU case are shown in Fig. VI–2 (annual) and VI–3 (cumulative).

TABLE VI–1. GENERIC LWR AND HWR CHARACTERISTICS

Parameters	Units	LWR		HWR	
Thermal power	MW(th)	3030.3		2000	
Electrical power	MWe	1000		600	
Thermal efficiency	%	33		30	
Fuel residence time in core	EFPD	1168		292	
Average burnup	MW·d/kg HM	45		7	
Fuel loading (equilibrium)	t	78.653		83.429	
Fuel composition (equilibrium)	(kg)	Reload	Discharge	Reload	Discharge
^{235}U		786.50	156.60	593.20	198.20
^{236}U			102.00		59.33
^{238}U		18 880.00	18 270.00	82 840.00	82 250.00
^{237}Np			13.65		2.16
^{238}Pu			5.04		0.28
^{239}Pu			106.30		221.80
^{240}Pu			41.33		79.84
^{241}Pu			36.45		15.13
^{242}Pu			15.38		3.28
^{241}Am			1.23		0.12
^{242m}Am			0.03		—
^{243}Am			3.60		0.10
^{242}Cm			0.43		0.04
^{244}Cm			1.26		0.01
Total FP			912.20		602.10
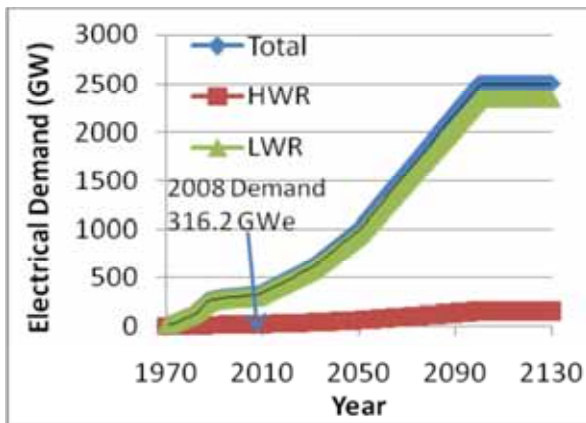
FIG. VI–1. BAU nuclear electricity demand.
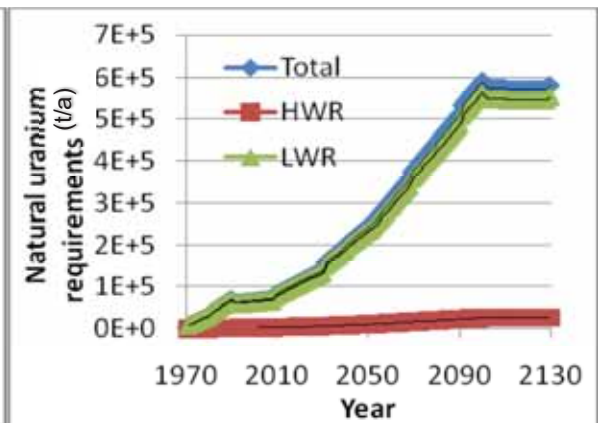
FIG. VI–2. Annual uranium requirements (BAU).
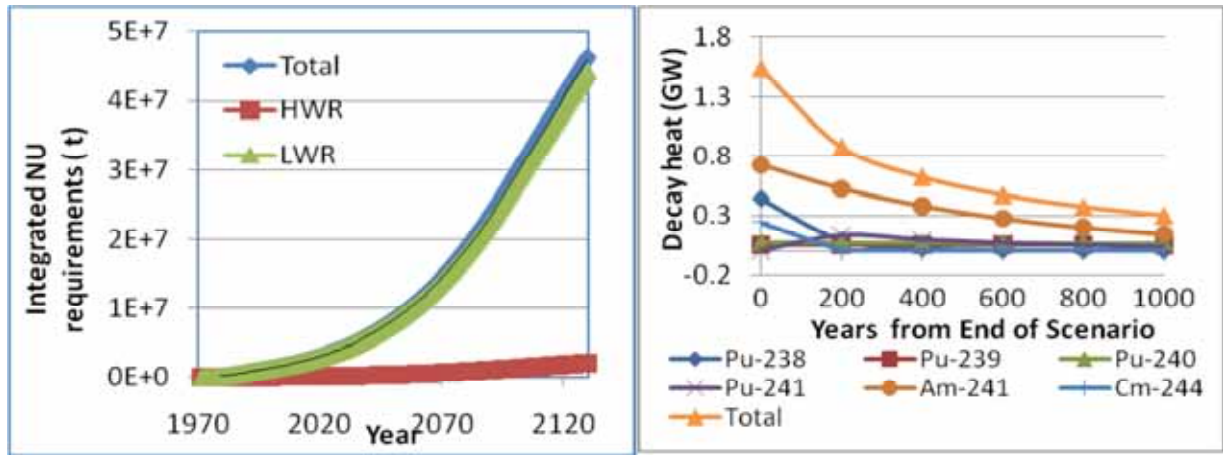
FIG. VI–3. Cumulative uranium requirements (BAU).   *FIG. VI–4. Decay heat from actinides (total spent fuel) after 2130 (BAU).*

The spent fuel actinide decay heat up to 1000 years after the end of the scenario was considered, because heat output on these time scales may be a useful indicator of capacity of a geological repository. The actinides included in the calculation were ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am and ^{244}Cm, but the uranium isotopes were suppressed (see Fig. VI–4) as their decay heat is negligible. Decay heat was calculated by running stand-alone ORIGEN-S [80] cases for the decay of 1 tonne of each isotope in the list and compiling the output for thermal power at 0, 200, 400, 600, 800 and 1000 years. A CANDU specific library, distributed with SCALE 5.1 [80], was used. DESAE was then rerun with reprocessing applied to all reactors so that all leftover spent fuel would be broken down into isotopes in the recycling facility. The tonnage of each isotope was then multiplied by the ORIGEN-S supplied thermal power per tonne factor to obtain the decay heat of the fuel projected past 2130. Figure VI–4 shows this decay heat for the BAU case, the various lines corresponding to the decay heat from each isotope and its daughters. All spent fuel is considered, including HWR fuel. Decay heat at 1000 years is approximately ~0.304 GW total.

VI–3. THORIUM FUEL CYCLE SCENARIOS

Two types of thorium fuelled HWR were introduced in BAU scenarios: the first is a thorium/plutonium reactor (ThPu) in the OTT cycle; the second is a thorium/plutonium/^{233}U fuelled reactor (ThPuR) whose fuel is recycled. Parameters of both reactors are outlined in Annex 1, A1.4. In the current study, these reactors had 85% capacity factors, 60 year lifetimes, and six year spent fuel cooling times. The two variant scenarios considered:

— After 2008, decommissioned HWRs were replaced by a combination of ThPu and uranium fuelled HWRs; the number of ThPu reactors was maximized on the basis of available stocks of ^{233}U and ^{239}Pu, with the intent that all uranium HWRs would eventually be replaced by ThPu HWR.
— Uranium HWRs are decommissioned as in the previous case, ThPu reactors are constructed and their fuel is recycled into stocks of ^{233}U, which allow ThPuR reactors to be built (as the stocks of ^{233}U and ^{239}Pu permit) during the second half of the century.

For the second variant, all of the fuel from the ThPuR is recycled, i.e. all the isotopes of uranium and plutonium. This is due to a restriction in DESAE that does not allow for selective recycle of only certain elements or isotopes. An self-sufficient equilibrium thorium (SSET) reactor that runs on only ^{233}U mixed with thorium, without the plutonium top-up of the ThPuR, is a natural extension of this analysis, but was not modelled explicitly here.

The division of electrical capacity among the various reactor types in Scenario 1 is shown in Fig. VI–5. The cumulative natural uranium requirements (Fig. VI–6) for this scenario are reduced by ~10% (Fig. VI–7) over the
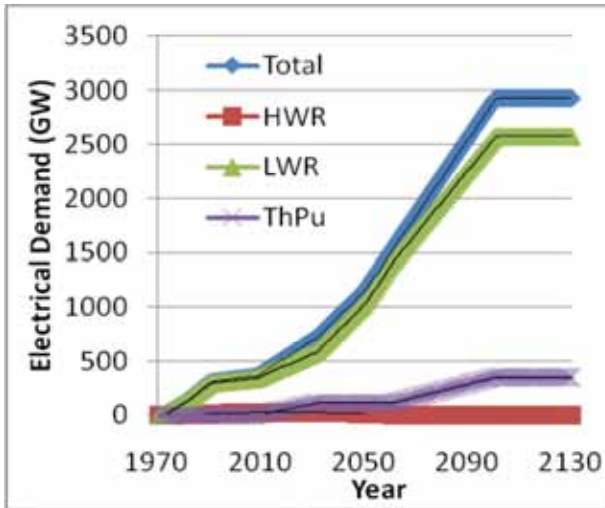
FIG. VI–5. Electrical capacity for variant with ThPu.
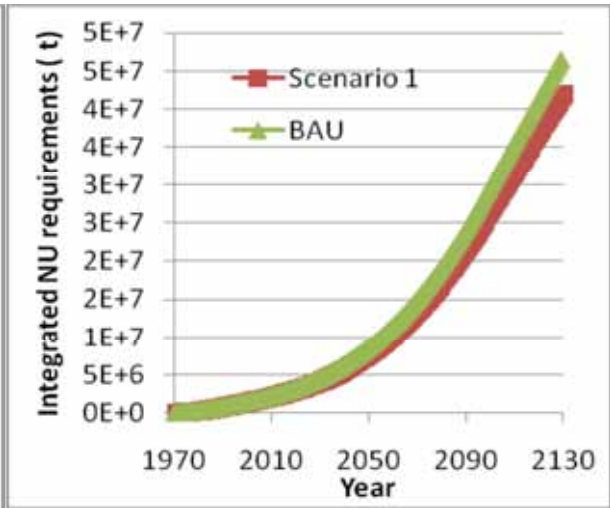
FIG. VI–6. Cumulative uranium requirements for scenario with ThPu vs. BAU.
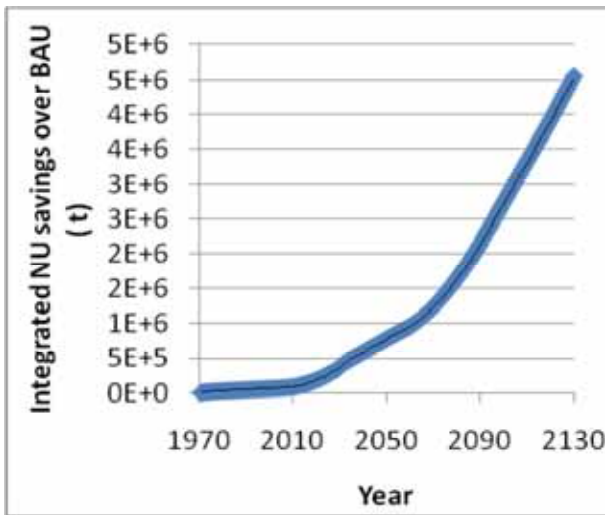
FIG. VI–7. Integrated natural uranium savings (scenario with ThPu).
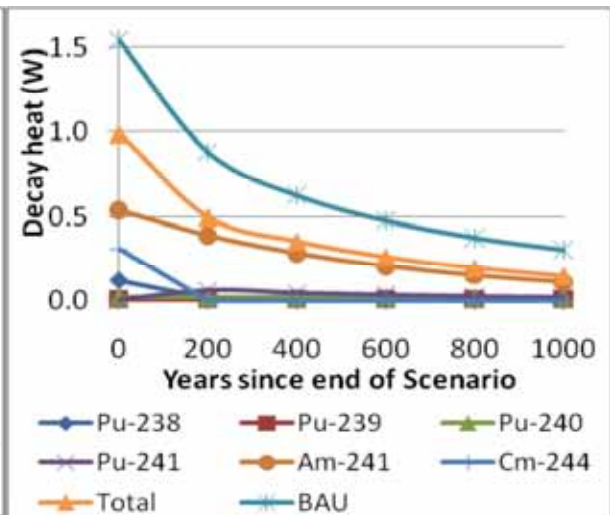
FIG. VI–8. Decay heat from actinides (total spent fuel) after 2130 for scenario with ThPu.

BAU case. Actinide decay heat (Fig. VI–8) at 1000 years has been reduced to 0.147 GW, slightly less than half of the BAU case.

In the second scenario (with ThPuR reactors), the ThPu reactors are used to build up ^{233}U in the reprocessing plants, and the ^{233}U is used, as available, to construct as many ThPuR reactors as possible. It was found that a 730 GW(e) capacity can be supported, amounting to 24.8% of total capacity (Fig. VI–9). At 2130, there are also 140 GW(e) capacity of ThPu reactors (required to burn up excess plutonium from the LWRs), or about 4.8% of total capacity, i.e. thorium-burning HWRs amount to nearly 30% of all capacity. Integrated natural uranium savings (Fig. VI–10 are approximately ~1E7 tonnes, or about ~22% of the BAU requirements (Fig. VI–11). Actinide decay heat (Fig. VI–12) is reduced by ~70% from the BAU scenario to 0.088 GW at 1000 years.
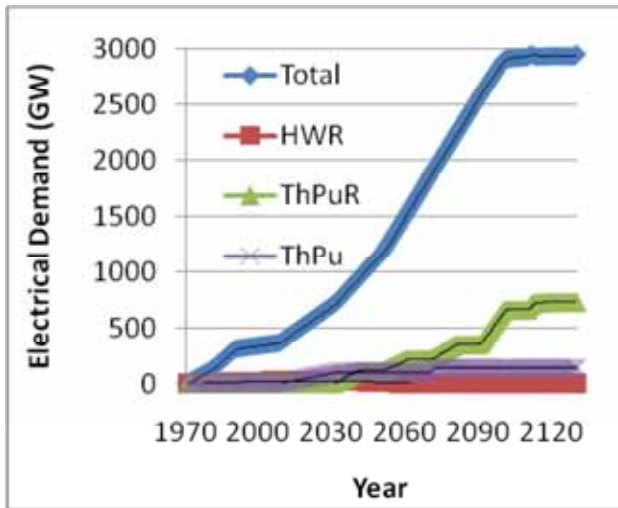
FIG. VI–9. Electrical capacity (scenario with ThPuR).
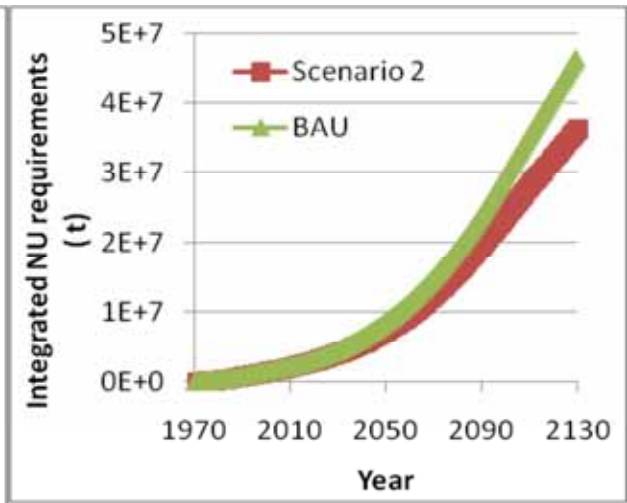
FIG. VI–10. Integrated natural uranium requirements (scenario with ThPuR).
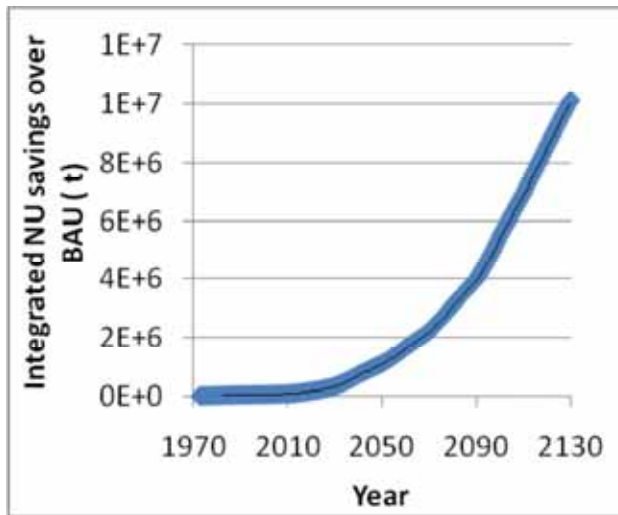
FIG. VI–11. Integrated natural uranium savings over BAU (scenario with ThPuR).
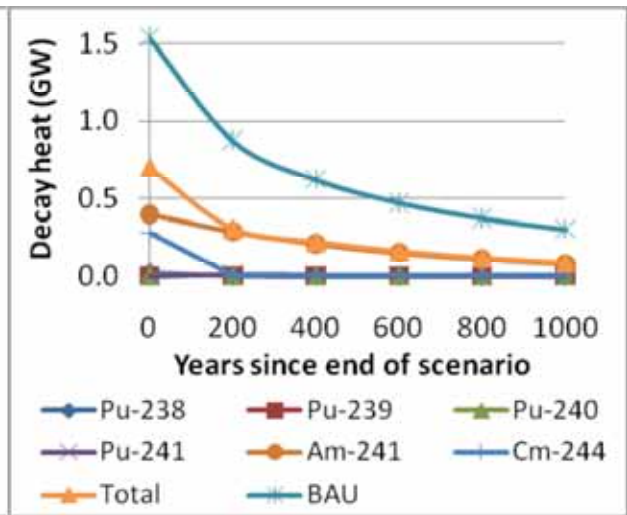
FIG. VI–12. Decay heat from actinides (total spent fuel) after 2130 (scenario with ThPuR).

LEVELIZED UNIT ENERGY COST CALCULATIONS BASED ON ALTERNATIVE INPUT DATA

The objective of Section 5 was only to compare different fuel cycles rather than to give a quantitative assessment of the cost of nuclear power. This comparison was based on some optimistic expectations of the reactor and fuel cycle data for the 21st century. Some of these data fall out of the range published in the economic studies referred to, which usually demonstrate that the capital costs of nuclear power plants have significantly increased in recent times, but that the O/M costs and fuel costs have not greatly increased.

Therefore, the updated input data presented in Section 5 (Tables 5.1 and 5.2) were based on the 2009 US dollar and complemented with some additional recent values from Refs [81–83]. Rounded average values are included in the Tables VII–1 and VII–2. To determine the prices of uranium conversion and enrichment, an average five-year stock market value was used [84]. For the construction costs, the 10% increase for HWRs and 25% increase for FRs defined in the GAINS project were kept.

TABLE VII–1. REACTOR CAPITAL AND O/M COSTS (ALTERNATIVE)

Item	Unit	Reactor type	Cost
Capital cost	$/kWe	LWR	3800
		LWR /MOX	3800
		HWR	4180
		FR /MOX	4750
Fixed O/M cost	$/kW/a	LWR, LWR /MOX	70
		HWR	75
		FR /MOX	75
Variable O/M cost	mill/kWh	LWR, LWR /MOX	0.7
		HWR	0.7
		FR /MOX	0.7

TABLE VII–2. URANIUM–PLUTONIUM AND THORIUM FUEL CYCLE COSTS (ALTERNATIVE)

Fuel cycle step	Units	U-Pu fuel cycle		Th fuel cycle	
		Type	Ref. value	Type	Ref. value
Conversion	$/kg U	LWR, HWR	10		
Enrichment	$/kg SWU	LWR UOX	150		
Fuel fabrication	$/kg(HM)	LWR UOX	275	LWR 0	275
		HWR UOX	100	LWR 1	325
		LWR MOX	325	LWR 2	1500
		FR-MOX	350	HWR 1	125
		FR bl U	350	HWR-2	625
				FR-MOX	350
				FR bl Th	350

TABLE VII–2. URANIUM–PLUTONIUM AND THORIUM FUEL CYCLE COSTS (ALTERNATIVE) (cont.)

Fuel cycle step	Units	U-Pu fuel cycle		Th fuel cycle	
		Type	Ref. value	Type	Ref. value
Reprocessing	$/kg (HM)	UOX	800	UOX	800
		MOX	800	MOX	800
		FR-MOX	1000	FR-MOX	1000
		FR bl U	800	FR bl Th	3000
				Th/HEU	6000
				Th/Pu	6000
				Th/Pu/U3	6000
SNF direct disposal	$/kg (HM)	LWR	600	LWR (Th)	600
		HWR	Variable	HWR (Th)	Variable

TABLE VII–3. LUEC DEPENDING ON NATURAL URANIUM COST

$/kgU	HWR	LWR	ALWR	LWR0	LWR1[1]	LWR2[2]	LWR2[3]	HWR1[1]	HWR2[4]	HWR2[5]	AHWR[6]	FRTh[1]	FR[7]	FR[8]
50	47.6	46.3	42.4	47.4	69.9	54.4	56.4	64.4	78.9	81.4	42.1	54.0	46.9	52.3
150	49.7	49.3	44.8	53.4							47.0			
300	52.8	53.8	48.3	62.5							54.4			
1000	67.4	75.0	65.0	104.6							89.1			

Notes:
[1] Pu from ALWR spent fuel
[2] 233U from LWR1, Pu from ALWR
[3] 233U from LWR1, Pu from LWR1
[4] 233U from HWR2, Pu from ALWR
[5] 233U from HWR2, Pu from LWR1
[6] Once-through mode
[7] Pu from FR
[8] For the first six years of operation, Pu is taken from ALWR

The main difference of these alternative set of input data (Table VII–1 compared to Table 5.1 in Section 5) is an approximate doubling of the capital cost of all reactor types and tripling of the reprocessing cost of thorium containing spent fuel.

In comparison to Table 5.2, Table VII–3 shows that the significant increase in construction costs causes an overall increase in electricity generation costs, and slightly changes its dependence on uranium cost for several reactors. The increase in the reprocessing cost of thorium based spent fuel significantly influenced the cost of energy produced by reactors operating in a closed ThFC with relatively low burnup (HWR2). Other thorium based reactors are less vulnerable to the cost of reprocessing.

The once-through application of thorium in AHWRs remains competitive against conventional water cooled reactors up to the cost of $250 per kg of natural uranium. The cost of uranium required to make the introduction of a closed ThFC (LWR2) competitive against traditional thermal reactors (LWRs, HWRs) operated in a once-through fuel cycle depends on the type of reactor using thorium and starts at a value of ~$400/kg.
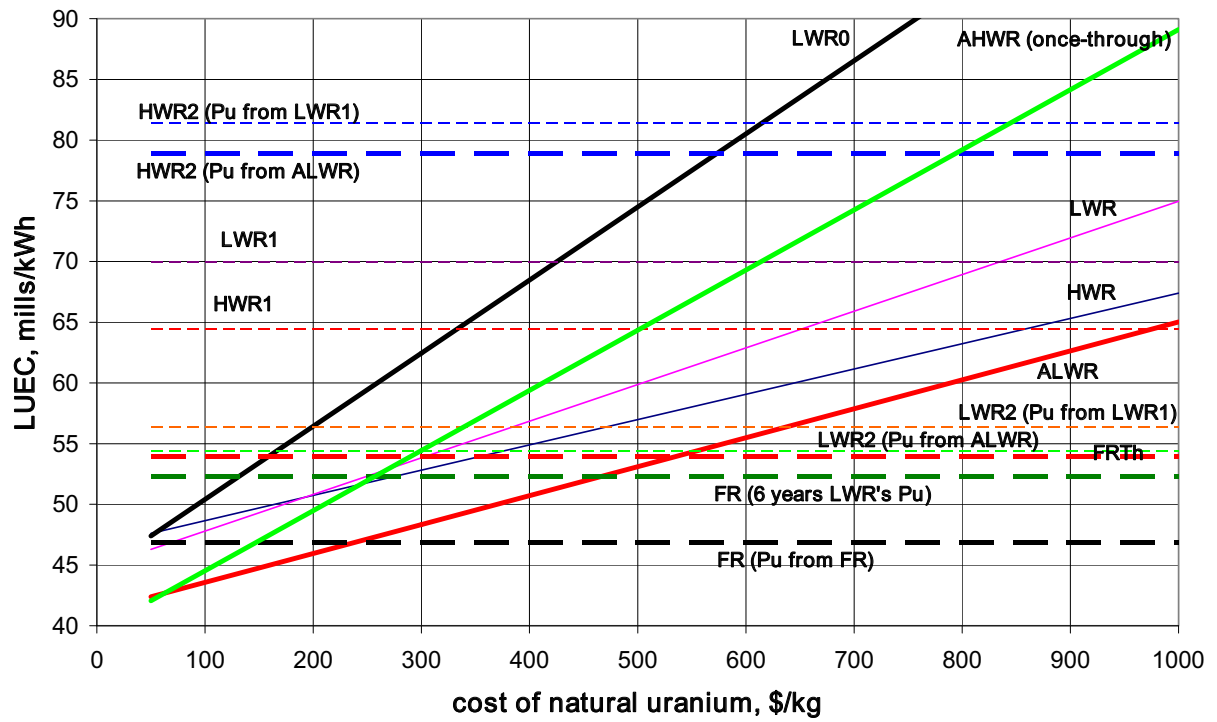
FIG. VII–1. Levelized cost of electricity depending on natural uranium cost.

Annex VIII

USE OF INPRO METHODOLOGY FOR EVALUATION OF PROLIFERATION RESISTANCE

This section describes the INPRO methodology for proliferation resistance and its application to a nuclear energy system with a ThFC in comparison to a uranium–plutonium fuel cycle (both open and closed).

VIII–1. GENERAL FEATURES OF INPRO METHODOLOGY

First, general features of the INPRO methodology will be presented. Second, specific features for proliferation resistance will be described. Finally, the analysis method of INPRO for proliferation resistance will be discussed.

In 2008, INPRO completed the development of its holistic and comprehensive methodology [60] for the assessment of nuclear energy systems (NES) in regard to long term sustainability. The methodology covers all areas of NES development or deployment, i.e. economics, infrastructure (institutional measures), waste management, proliferation resistance, physical protection, environment (impact of stressors and availability of resources) and safety. The methodology is documented in nine volumes of an IAEA report [60]. Volume 1 provides an overview of the methodology; the other volumes cover each area mentioned above.

According to the INPRO methodology, all fuel cycle steps should be assessed: mining and milling, conversion, enrichment (if any), fuel fabrication, reactor operation (including short term spent fuel storage), intermediate spent fuel storage (if any), spent fuel reprocessing (if any), final waste disposal. Also, a once-through fuel cycle and a closed fuel cycle for both thorium–uranium and uranium–plutonium cases could be considered. In addition, variations in the fuel cycle such as ^{233}U denaturization are taken into account at every stage of the fuel cycle.

The INPRO methodology provides a set of requirements for the assessment of a NES and a method to reveal the status of this system with regard to each area, e.g. economics and safety. The requirements are developed in a hierarchical structure where at the top level, one basic principle is defined and at the second level there are five user requirements, which are then linked to 12 criteria at the third level. Every criterion comprises an indicator and a corresponding acceptance limit.

Within INPRO, a *basic principle* is a statement of a general goal that is to be achieved in a NES to be sustainable and provides broad guidance for the development of a NES or a design feature.

User requirements are the conditions that should be met to achieve acceptance of a given NES by all stakeholders in nuclear energy, i.e. they define what the designer, the operator, industry and the government should do to achieve the goal set out in the basic principle.

Criteria enable the INPRO assessor to determine whether and how well a given user requirement is being met by a given NES, e.g. whether the designer of the NES has performed the requested action. As stated above, an INPRO criterion consists of an *indicator* and an *acceptance limit*.

There are two types of *indicators* — numerical and logical. A *numerical indicator* may be based on a measured or a calculated value that reflects a property of a NES (or component thereof). A *logical indicator* is formulated in the form of a question and is usually associated with some necessary feature of a NES.

Acceptance limits are the targets, either qualitative or quantitative, against which the value of the indicators can be compared by the INPRO assessor leading to a judgment of acceptability (pass/fail, good /bad, better/poorer). Corresponding to the two types of indicators, there are also two types of acceptance limits, numerical (for quantitative targets) and logical (for qualitative targets, i.e. 'yes' or 'no' as the answer to the question raised).

For some criteria, corresponding *evaluation parameters* were introduced to assist the INPRO assessor in determining whether the acceptance limit for an indicator has been met.

The methodology ensures fulfilment of all criteria associated with a specific user requirement. This means that this user requirement has been met, and also that fulfilment of all user requirements related to a basic principle means that this basic principle has been met by the NES assessed. Conversely, in the case of at least one failed criterion, the corresponding user requirement and basic principle are not fulfilled, i.e. a 'gap' in the NES has been identified. In such a case, the INPRO methodology requires the assessor to describe follow-up actions to close the gap in order to achieve a sustainable NES.

VIII–2. METHODOLOGY FOR ASSESSMENT OF PROLIFERATION RESISTANCE

Volume 5 of the INPRO methodology [60] is devoted to non-proliferation and defines proliferation resistance as:

"the characteristic of a nuclear energy system that impedes the diversion or undeclared production of nuclear material, or misuse of technology, by States intent on acquiring nuclear weapons or other nuclear explosive devices."

The proliferation resistance area of the INPRO methodology is limited to proliferation by States, i.e. it is focused on the prevention of a possible contribution of an NES to a national nuclear weapon programme in a given State.[1]

The basic principle of proliferation resistance in the INPRO methodology is formulated as follows:

"Proliferation resistance intrinsic features and extrinsic measures shall be implemented throughout the full life cycle for innovative nuclear energy systems to help ensure that NES will continue to be an unattractive means to acquire fissile material for a nuclear weapon programme. Both intrinsic features and extrinsic measures are essential, and neither shall be considered sufficient by itself."

Intrinsic features are part of the technical design of an NES, including those that facilitate the implementation of extrinsic measures [85]. Intrinsic features ensure the high complexity of and a long time required for the modifications necessary to use a civilian NES for a weapon production programme, which require specific skills, expertise, and knowledge to divert or produce nuclear material and convert it into a weapon useable form. Extrinsic measures result from States' decisions and undertakings related to an NES, e.g. signing appropriate safeguard agreements with the IAEA.

According to the INPRO methodology, the structure of requirements for proliferation resistance is shown in Fig. VIII–1.

As with proliferation resistance intrinsic features and extrinsic measures, the total set of INPRO proliferation resistance requirements is divided into country specific demands (corresponding to 'extrinsic measures', grey cells in Fig. VIII–1) and technology related demands (corresponding to 'intrinsic features', clear cells in Fig. VIII–1).

VIII–3. ANALYSIS METHOD TO DETERMINE LEVELS OF PROLIFERATION RESISTANCE

Historically, INPRO developed a *proliferation resistance analysis method* first [86]. Later, an INPRO *proliferation resistance assessment method* was derived from this analysis method. The proliferation resistance assessment method is presented above (Fig. VIII–1). In this section, some features of the proliferation resistance analysis method are described.

The proliferation resistance analysis method defines either two or several levels of proliferation resistance for each evaluation parameter (of each criterion). For requirements addressing a government or for logical criteria, only two possible levels of proliferation resistance — strong or weak — are defined. In particular for criteria associated with requirements addressing the designer or operator, a range of proliferation resistance values was developed for each evaluation parameter, e.g. very weak to very strong proliferation resistance.

Examples of such evaluation parameters (intrinsic features) are: isotopic content of nuclear material, its chemical form, its radiation field, heat generated by nuclear material, its spontaneous neutron generation rate, its mass and bulk, and design features that limit access to nuclear material.

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[1] The issue of protection against the misuse of fissile material by subnational groups or the sabotage of nuclear installations or transport systems is covered in Vol. 6, Physical Protection. It is assumed here that physical protection measures are the responsibility of the government and that they can be efficiently introduced regardless of the type of the fuel cycle implemented.

User Requirements Limits	Indicator	Acceptance
State commitments: States' commitments, obligations and policies regarding non-proliferation and its implementation should be adequate to fulfil international standards in the non-proliferation regime — legal framework — States' commitments, obligations and policies regarding non-proliferation established? (Yes, in accordance with international standards)	Party to international non-proliferation treaty NPT.	Whether the State intending to deploy a NES is party to NPT.
	Party to regional non-proliferation regimes.	Whether the State is party to such a treaty. INPRO encourages, but does not require, a State to be a party of such regimes where they exist.
	Comprehensive safeguards agreements in force.	Whether in the State such a CSA is in place.
	Additional protocol in force.	Whether in the State such an additional protocol is in place. INPRO recommends such agreement in addition to a comprehensive safeguards agreement, INFCIRC/66 type agreement, or voluntary offer agreement.
	INFCIRC/66-type safeguards agreement in force.	Whether in a State (not party to the NPT), intending to deploy a NES, at least such an agreement is in place for all its nuclear facilities for peaceful use of nuclear energy.
	Export control policies of NM and nuclear technology.	Whether in the State export controls are in place.
	SSAC or RSAC in force.	Whether a SSAC or RSAC is established in the country of the NES to be deployed.
	Regulatory body, designated in national legislation for implementing and applying safeguards agreements.	Whether the national authority dealing with safeguards is established in accordance with the requirements in Section 12.4 of Handbook on Nuclear Law.
Attractiveness of NM and technology: The attractiveness of NM and nuclear technology in an NES for a nuclear weapon programme should be low. This includes the attractiveness of undeclared nuclear material that could credibly be produced or processed in the NES.	Attractiveness of NM — Material quality; material quantity; material form.	Whether the attractiveness of NM considered in design of NES and found acceptable low.
	Attractiveness of technology — Nuclear technology	Whether the attractiveness of technology is considered in the design of NES and found acceptable low.
Difficulty and detectability of diversion: The diversion of NM should be reasonably difficult and detectable. Diversion includes the use of an NES facility for the production or processing of undeclared material.	Quality of measurement — Accountability of NM	Whether the accountability is equal or better than existing designs, meeting international state of practice.
	C/S measures and monitoring — Amenability	Whether the amenability is equal or better than existing designs, meeting international best practice.
	Detectability — Detectability of NM	Whether the detectability of NM is equal or better than existing facilities.
	difficulty of modification and misuse — Difficulty to: modify process; modify facility design; misuse technology or facilities.	Whether the difficulties are equal or better than existing designs, meeting international best practice.
Multiple features: Innovative nuclear energy systems should incorporate multiple PR features and measures.	Defence in depth — The extent by which the NES is covered by multiple intrinsic features and extrinsic measures	Whether all plausible acquisition paths are (can be) covered by extrinsic measures on the facility or State level and by intrinsic features which are compatible with other design requirements.
	Robustness of PR barriers — Robustness of barriers covering each acquisition path.	Whether the robustness of barriers is sufficient based on expert judgment.
Optimization of design: Combination of intrinsic features and extrinsic measures, compatible with other design considerations, should be optimized (in design/ engineering phase) to provide cost efficient PR.	Inclusion of PR in NES design — Whether PR has been taken into account as early as possible in the design and development of the NES	Yes
	Cost of PR features and measures — The cost of incorporating into an NES those intrinsic features and extrinsic measures, which are required to provide or improve proliferation resistance	Whether the total cost of the intrinsic PR features and extrinsic PR measures over the life cycle of the NES is acceptable.
	Verification approach — Verification approach with a level of extrinsic measures agreed to between the State and verification authority (e.g. the IAEA, regional SG organization)?	Yes

FIG. VIII–1. Structure of proliferation resistance requirements in INPRO methodology (assessment method).

The general sequence of actions during the application of the analysis method of the INPRO methodology is shown in the table below.

	Action	Objective
Step 1:	Application of the method of PR analysis, first part — defining the level of PR for all evaluation parameters.	To get a detailed level of PR of features of the NES and to identify potentially 'weak points'.
Step 2:	Application of the method of PR analysis, second part — defining compensatory safeguards measures for every 'weak point' revealed.	To demonstrate that safeguarding the evaluated NES is possible in an effective and efficient way at a reasonable cost.

The INPRO *proliferation resistance assessment method* should be started after the *proliferation resistance analysis* has been performed. Its objective is to confirm that all weak points have been successfully compensated by safeguards measures or to define necessary follow-up actions to close gaps found (weak points not compensated by safeguards measures) in the design of the NES.

VIII–4. EVALUATION METHOD OF PROLIFERATION RESISTANCE

It is not possible to assess all proliferation resistance related criteria of the INPRO methodology in this report at such an early stage of the NES development, especially in the case of a ThFC, since there are no design details yet. However, country specific requirements do not depend on the type of fuel cycle to be implemented and could/should be considered at later stages of the NES implementation.

Consequently, particularly, user requirements and criteria related to technology (intrinsic features) and the design of the NES and not country specific requirements (extrinsic measures) will be considered as follows:

— *Attractiveness of nuclear material* defined by material quality, material quantity, and material form for a nuclear weapon programme (to be found acceptably low);
— *Attractiveness of a given technology* for a nuclear weapon programme considered in the design of NES (to be found acceptably low);
— *Accountability* and *quality of measurement* (to be equal or better than existing designs);
— *Amenability to C/S measures* and *monitoring* (to be equal or better than existing designs);
— *Detectability of nuclear material* (to be equal or better than existing designs);
— *Difficulty to modify process*, to *modify facility design*, to *misuse technology* or *facilities* (to be equal or better than existing designs);
— All plausible *acquisition paths* are (can be) *covered by multiple intrinsic features*, which are compatible with other design requirements;
— *Robustness* of proliferation resistance barriers covering each acquisition path is *sufficient*;
— *Total cost of* incorporating into an NES those *intrinsic features and extrinsic measures* required to provide or improve proliferation resistance should be *minimal* over the life cycle of the NES.

The last three bullets above cannot be evaluated without an acquisition path analysis that depends strongly on NES facility design and the specific country, and cannot be reported here, considering the early stage of NES development. For the time being, it is assumed that all necessary measures can be implemented in all fuel cycle options and at every stage of the fuel cycle. In the particular case considered below, the requirements formulated in the last three bullets above cannot yield information to distinguish options, so they do not participate in further discussion.

At such an early stage of development of NES, Step 2 (definition of compensatory measures of weak proliferation resistance features) described above in Section 6.1.3 cannot be performed completely and, therefore, the following analysis, referring to Section 6.2, deals mainly with Step 1.

VIII–5. COMPARATIVE PROLIFERATION RESISTANCE ANALYSIS OF A THORIUM FUEL CYCLE WITH OTHER NUCLEAR FUEL CYCLE OPTIONS

Two pairs of nuclear fuel cycle (NFC) systems are chosen for the comparative proliferation resistance analysis:

— Uranium–plutonium once-through fuel cycle against thorium–uranium once-through fuel cycle;
— Uranium–plutonium closed fuel cycle against thorium–uranium closed fuel cycle.

The uranium–plutonium once-through cycle uses enriched uranium in fresh fuel and accumulates some plutonium during reactor operation in spent fuel. In addition to enriched uranium, the thorium–uranium once-through cycle uses thorium (in pellets, in separate fuel rods or in separate fuel elements) in its fresh fuel. Once-through fuel cycles are assumed to be used in thermal reactors (i.e. those with thermal neutron spectrum: LWR or HWR) and do not envisage any reprocessing of spent fuel. The comparison of reprocessing features regarding proliferation resistance can be derived from the closed fuel cycle variants if needed.

In fresh fuel of the uranium–plutonium closed fuel cycle, plutonium is used as fissile material together with uranium (recycled or depleted). In the thorium–uranium closed fuel cycle, fresh fuel contains ^{233}U as fissile material together with thorium as fertile material; options of nuclear fuel cycles that, in addition to ^{233}U, also use $^{235}U/^{238}U$ will be discussed. Moreover, closed fuel cycles are assumed not to envisage any enrichment of uranium. The uranium–plutonium closed fuel cycle is based on fast breeding reactors, and thorium–uranium closed fuel cycle is based on thermal breeding reactors (LWRs or HWRs). The comparison of enrichment features can be derived from the once-through fuel cycle variants if needed. Fast reactors with uranium–plutonium core fuel and thorium blankets can be considered an option compiling the results represented below.

In closed fuel cycles (Fig. VIII–3), mining, milling and refining stages are not considered. A comparison of these stages can also be derived from the once-through fuel cycle variants if needed. Also, $^{238}U+20\%$ Pu oxide fuel utilization in a core (in the uranium–plutonium case) and consequent minor actinide separation from uranium and plutonium at the fuel reprocessing stage is assumed. In the case of a thorium fuel cycle, assumed reactor parameters (save thermal efficiency and capacity) correspond to the Shipping Port breeding reactor [2].

Schemes of the fuel cycles considered are represented in Figs VIII–2 and VIII–3.
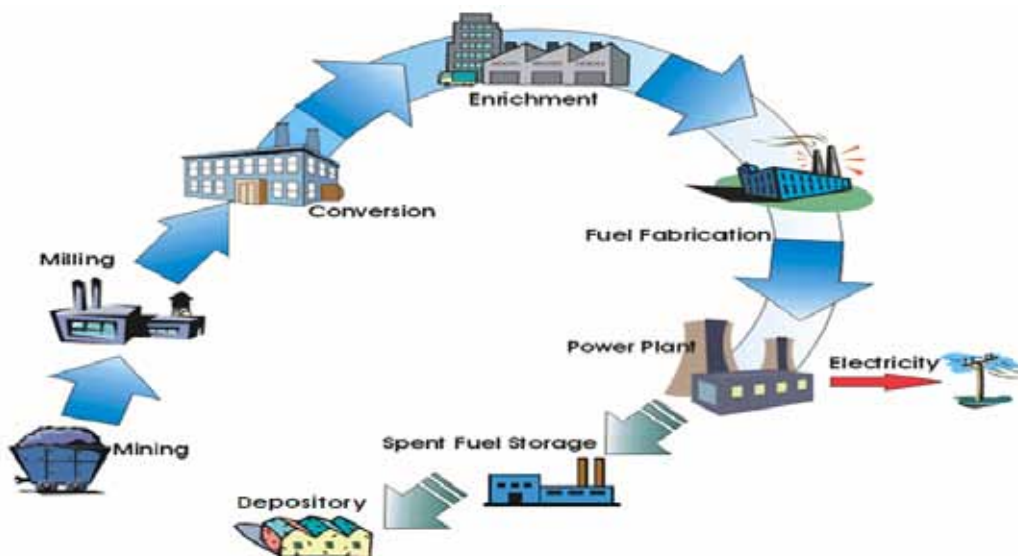
FIG. VIII–2. Once-through fuel cycle.
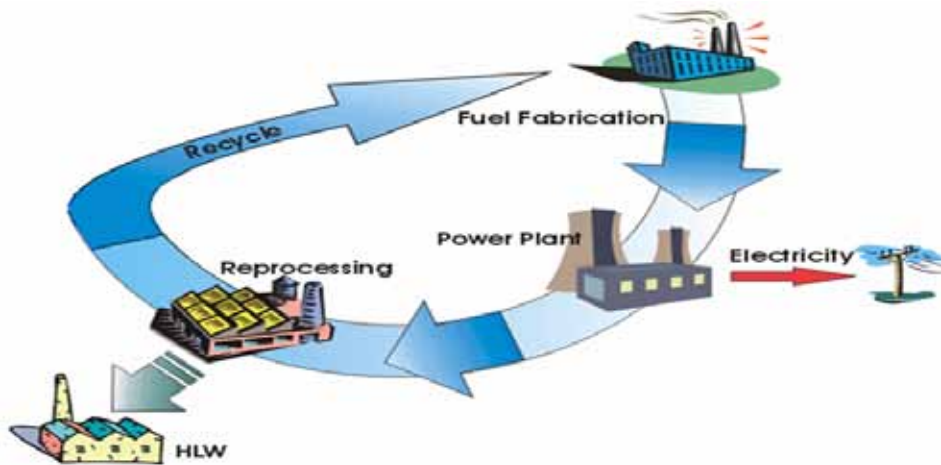
FIG. VIII–3. Closed fuel cycle.

The fuel cycle options assessed below are not the only possible variants of thorium implementation, nor are the reactors that have been chosen for comparison. The choice has been made to demonstrate the application of proliferation resistance analysis method of the INPRO methodology. Appropriate justification of proliferation resistance advantages or drawbacks of a ThFC would require consideration of all possible fuel designs, which is beyond the scope of this report.

In the following, the *proliferation resistance analysis method* is applied in accordance with Appendix A of Volume 5 of the INPRO manual [60]. The evaluation scale proposed in this appendix is also used in this report. The scores of proliferation resistance are: VW — very weak; W — weak; M — medium; S — strong; and VS — very strong.

VIII–5.1. Uranium–plutonium once-through fuel cycle

The scheme of this fuel cycle is shown in Fig. VIII–2.

ATTRACTIVENESS OF NUCLEAR MATERIAL

Group of parameters	Parameter	Evaluation scale VW	W	M	S	VS	Scores M/M[1]	C/R[1]	E[1]	FF[1]	RO[1]	SFS[1]	SFD[1]
Material quality	1. Isotopic composition: ^{239}Pu/Pu (wt%)		>50		<50		N/A[2]	N/A	N/A	N/A	W[4]	W[4]	W[4]
	2. Heat generation characteristic: ^{238}Pu/Pu (wt%)		<20		>20		N/A	N/A	N/A	N/A	W[5]	W[5]	W[5]
	3. Spontaneous neutron generation rate characteristic: $(^{240}$Pu+^{242}Pu)/Pu (wt%)	In comparison to Th/U once-through fuel cycle					N/A	N/A	N/A	N/A	S	S	S
	4. Material type/category	UDU[3]	IDU[3]	LEU	NU	DU	S	S	M	M	W	W	W
	5. Radiation field: dose (mGy/hr) at 1 metre	<150	150–350	350–1000	1000–10 000	>10 000	VW	VW	VW	VW	VS [86]	S [86]	S
Material quantity	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	N/A	N/A	N/A	VW[6]	S[6]	S[6]	S[6]
	7. No. of items for SQ	1	1~10	10~50	50~100	>100	N/A	N/A	N/A	W[6]	W[6]	W[6]	W[6]
	8. Mass of bulk material for SQ (dilution) (kg)	10	10~100	100~500	500~1000	>1000	VS	VS	VS[7]	VS[7]	N/A	N/A	N/A
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VW[8]	VW[8]	VW[8]	VW[8]	VW	VW	VW

ATTRACTIVENESS OF NUCLEAR MATERIAL (cont.)

Group of parameters	Parameter	Evaluation scale					Scores						
		W			S	Waste	M/M[1]	C/R[1]	E[1]	FF[1]	RO[1]	SFS[1]	SFD[1]
		Metal	Oxide/solution	compounds	Spent fuel								
Material form	10. Chemical/physical form						W[9]	W[9]	M	W[9]	W	S	VS

Notes:

[1] Mining and milling (M/M); conversion and refining (C/R); enrichment (E); fuel fabrication (FF); reactor operation including short term storage (RO); Spent fuel storage (SFS); Spent fuel depository (SFD).

[2] Not applicable.

[3] Un-irradiated direct use material (UDU); irradiated direct use material (IDU).

[4] The ratio ^{239}Pu/Pu depends on the burnup of the fuel (the lower burnup, the higher ratio ^{239}Pu/Pu). In LWR type fuel with burnup at 45 MW·d/kg HM, the ratio ^{239}Pu/Pu can be estimated at 50–55 wt% at 5 years after discharge (the result is achieved in GAINS vith the help of the NFCSS code [87]). Various fuel burnups are permanently available in a reactor core. Low burned assemblies are also available in spent fuel, e.g. discharged from the first and second reactor core or among leaked fuel.

[5] In LWR type fuel with burnup at 45 MW·d/kg HM, the ratio ^{238}Pu/Pu can be estimated at 2–4 wt% at 5 years after discharge (GAINS, NFCSS code [87]).

[6] The weight of fresh assembly is assumed to be 700 kg including 400 kg of HM. At the stage of fuel fabrication the lowest weight item is fuel pellet (its mass is less than 10 kg).

[7] 5% LEU in the fuel and corresponding enrichment up to 5% is assumed.

[8] It is also assumed that annually consumed stock of nuclear material can be accumulated at any stage in the frontend facility's storage. Assuming NES power equal to 10 GW(e), annual reloading ~20 tLEU/GW(e), annual consumption is 133 SQ.

[9] It is assumed that at some stages of the fuel cycle (mining and milling, conversion, etc.), the nuclear material can exist in various forms (compounds, oxides), but 'oxide' is a conservative option for PR assessment.

ATTRACTIVENESS OF A GIVEN TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores						
		W	S	M/M	C/R	E	FF	RO	SFS	SFD
Nuclear technology	11. Enrichment	Yes	No	S	S	W	S	S	S	S
	12. Extraction of fissile material (other than enrichment)	Yes	No	S	S	S	S	S	S	S
	13. Irradiation capability of undeclared fertile material	Yes	No	S	S	S	S	W	S	S

ACCOUNTABILITY, QUALITY OF MEASUREMENT AND AMENABILITY TO C/S MEASURES AND MONITORING

Group of parameters	Parameter	Evaluation scale					Scores						
		VW	W	M	S	VS	M/M	C/R	E	FF	RO	SFS	SFD
Accountability	14. σMUF/SQ (for Pu or ^{233}U; ^{235}U with HEU; ^{235}U with LEU)	>2	2~1	1~0.5	0.5~0.1	<0.1	N/A	N/A	VS[1]	VS[1]	VS[2]	VS[2]	VS[2]
	15. Inspectors measurement capability	IC only	DA only	Comb NDA/NA	NDA active	NDA passive	W[3]	W[3]	W[3]	M	VS	VW[4]	VW[4]
			W		S								
Amenability for C/S and monitoring systems	16. Amenability of containment measures		No		Yes		W	W	W	W	S	S	S
	17. Amenability of surveillance measures		No		Yes		W	W	W	W[5]	S	S	W
	18. Amenability of other monitoring systems		No		Yes		W	S	S	S	S	S	W

Notes:

1 In the case considered (LEU, ^{235}U – 5%), SQ = 1500 kg.

2 Although book inventory of NM is always defined in weight units, the physical inventory of NM is accounted in items (i.e. assemblies or fuel rods) at these stages of one-through fuel cycle. Any disassembling is not envisaged further (and could be detected if any), so the deviations in NM accounting cannot be caused by the low accuracy of measurement.

3 Representative determination of ^{235}U concentration in a mix could be done only in laboratory conditions.

4 At these stages of the fuel cycle, only casks are available directly for accountancy.

5 At the fuel fabrication stage, surveillance measures are available at the latest steps of the process (assemblies).

DETECTABILITY OF NUCLEAR MATERIAL, DIFFICULTY IN MODIFYING THE PROCESS, DIFFICULTY IN MODIFYING FACILITY DESIGN, TO MISUSE TECHNOLOGY

Group of parameters	Parameter	Evaluation scale					Scores						
		VW	W	M	S	VS	M/M	C/R	E	FF	RO	SFS	SFD
Detectability of nuclear material (NM)	19. Possibility to identify NM by NDA		No		Yes		S	S	S	S	S	S	S
	20. Detectability of radiation signature		No reliable signature		Reliable signature		W	W	W	W	S	S	W[1]
Detectability of misuse of technology or facilities	21. Possibility to detect misuse of the technologies and the NES facilities for processing of undeclared NM		No		Yes		W	W	W	W	S	S	S
Difficulty in modifying the facility design	22. Verifiability of facility design by inspectors (transparency of design)		No		Yes		W[2]	W[2]	W[2]	W[2]	S	S	S
	23. Availability of data for inspectors		NRTA not active		NRTA active		W	S	S	S	S	S	S[3]
	24. Transparency of process		No		Yes		W	W	W	S	S	S	W
Difficulty the modifying the process	25. Accessibility of NM to inspectors		No		Yes		S	S	S	S	S	W[4]	W
		VW	W	M	S	VS							
	26. Extent of automation	N/A	Manual operation		Partial automation	Full automation	W	VS	VS	S	W	W	W

Notes:

[1] A long period of consideration is assumed.

[2] In the framework of this report, it is assumed that, at the fuel cycle stages with the bulk form of NM, the quantitative verifiability of facility design by inspectors is complicated.

[3] Near real time accounting is assumed to be feasible during the active life (loading of spent fuel canisters) of the depository.

[4] Dry container storage of the spent fuel is assumed.

VIII–5.2. Th/^{233}U fuel in once-through fuel cycle

The fresh fuel of such a fuel cycle consists of thorium and a certain part of enriched uranium. Thorium fuel is assumed to be located in separate fuel rods or fuel elements, i.e. thorium is never blended (during fabrication) with uranium–plutonium composition. During reactor operation, small quantities of ^{233}U will be produced in uranium–plutonium fuel in a 'natural' way. However, all reactors that would use thorium in the core in a once-through fuel cycle have to also use enriched uranium, with an even higher average core enrichment in ^{235}U compared to a core with pure uranium–plutonium fuel, if the reactivity of the core containing thorium is to be kept equivalent to the reactivity of a pure uranium–plutonium core. This means that all weak points regarding proliferation resistance of the uranium–plutonium once-through fuel cycle will occur also in the thorium–uranium cycle, but maybe even at a higher scale due to higher enrichment.

The analysis below is focused only on nuclear material consisting of Th/^{233}U in separated fuel rods (or elements) as part of the core in a thorium–uranium once-through fuel cycle and aims to show qualitatively whether the Th/^{233}U fuel introduction improves proliferation resistance features of once-through fuel cycle.

ATTRACTIVENESS OF NM

Group of parameters	Parameter No.	Evaluation scale					Scores					
		VW	W	M	S	VS	M/M	C/R	FF	RO	SFS	SFD
	1. Isotopic composition[1]: highest value of ^{233}U/U (wt%) in pin		>12[1]		<12[1]		N/A	N/A	N/A	W[2]	W[2]	W[2]
	2. Heat generation characteristic: decay heat emission (watt/crit.mass)		<600[3]		>600[3]		N/A	N/A	N/A	W[3]	W[3]	W[3]
Material quality	3. Spontaneous neutron generation rate	In comparison to U/Pu one-through fuel cycle					N/A	N/A	N/A	W[4]	W[4]	W[4]
	4. Material type/category	UDU	IDU	LEU	NU	DU				W	W	W
	5. Radiation field: ^{232}U contamination for ^{233}U (ppm)	<400[6]	400–1000[6]	1000–2500[6]	2500–25000[6]	>25,000[6]	VS[5]	VS[5]	VS[5]	M[3]	M[3]	M[3]
Material quantity	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	VW	VW	VW[7]	S[8]	S[8]	S[8]
	7. No. of items for SQ	1	1~10	10~50	50~100	>100	N/A	N/A	VS[7]	W[8]	W[8]	W[8]
	8. Mass of bulk material for Q (dilution) (kg)	10	10~100	100~500	500~1000	>1000	VS	VS	VS	N/A	N/A	N/A
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VS	VS	VS	VW	VW	VW
Material form	10. Chemical/physical form	Metal	Oxide/solution	compounds	Spent fuel	Waste	W	W	W	W	S	VS

Notes:

1 This parameter differs from one in Ref. [60]. It was modified to keep the logical link of requirements of U/Pu fuel cycle and Th/U one. Contamination with ^{232}U, related to the radiation field, is considered among PR barriers (intrinsic features)(see parameter No. 5).

2 Heterogeneous composition of uranium and thorium in the fuel bundles is assumed.

3 To estimate evaluation scale it was assumed that the ratio ^{238}Pu/Pu = 3% corresponds to 90 watts/crit.mass decay heat emission of fissile element [3].

4 According to Ref. [3], spontaneous fission rate of ^{233}U is 0.5 $(\text{s·kg})^{-1}$. The same parameter for ^{239}Pu (containing 6% of Pu-240) is $2.5\cdot10^4$ $(\text{s·kg})^{-1}$.

5 Thorium is a fertile material ranked in the same category as ^{238}U.

6 The scale of ^{232}U contamination has been elaborated in the INPRO methodology [60], it differs from the scale applied for radiation field of U/Pu fuel cycle because of the complimentary factors considered, e.g. the difference of half-lives of the main γ-radioactive isotopes (including predecessors) and low concentration of γ-radioactive isotopes (^{224}Ra, ^{220}Rn, ^{212}Bi, ^{208}Tl) in the purified ^{233}U/^{232}U mixture, etc.

7 It is taken into account that thorium fuel bundles contain only fertile material.

8 The same as in U/Pu one-through fuel cycle.

ATTRACTIVENESS OF A GIVEN TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores					
		W	S	M/M	C/R	FF	RO	SFS	SFD
Nuclear technology	11. Enrichment	Yes	No	S	S	S	S	S	S
	12. Extraction of fissile material (other than enrichment)	Yes	No	S	S	S	S	S	S
	13. Irradiation capability of undeclared fertile material	Yes	No	S	S	S	W	S	S

ACCOUNTABILITY, QUALITY OF MEASUREMENT AND AMENABILITY TO C/S MEASURES AND MONITORING;

Group of parameters	Parameter	Evaluation scale					Scores					
		VW	W	M	S	VS	M/M	C/R	FF	RO	SFS	SFD
Accountability	14. σMUF/SQ (for Pu or ^{233}U; ^{235}U with HEU; ^{235}U with LEU)	>2	2~1	1~0.5	0.5~0.1	<0.1	N/A	N/A	N/A[1]	VS[2]	VS[2]	VS[2]
	15. Inspectors measurement capability	IC only	DA only	Comb NDA/NA	NDA active	NDA passive	N/A[3]	N/A[3]	N/A[3]	VS	VW[4]	VW[4]
			W		S							
Amenability for C/S and monitoring systems	16. Amenability of containment measures		No		Yes		N/A[3]	N/A[3]	N/A[3]	S	S	S
	17. Amenability of surveillance measures		No		Yes		N/A[3]	N/A[3]	N/A[3]	S	S	W
	18. Amenability of other monitoring systems		No		Yes		N/A[3]	N/A[3]	N/A[3]	S	S	W

Notes:

[1] This criterion is not applicable to pure thorium fuel, but it should be borne in mind that in the case considered, the uranium fuel used in the feed and breed scheme (LEU, ^{235}U – 20%) is SQ = 375 kg.

[2] Although book inventory of NM is always defined in weight units, the physical inventory of NM is accounted in items (i.e. assemblies or fuel rods) at these stages of one-through fuel cycle. No further disassembling is envisaged (and could be detected if any) so the deviations in NM accounting cannot be caused by the low accuracy of measurement.

[3] One acknowledges that the measurements of Th-232, and also containment, surveillance and monitoring at these stages are of low importantance for PR control. These four parameters of proliferation resistance of the Th/U one-through fuel cycle can be assessed as 'N/A', considered here the highest rank of PR for these four items.

[4] At these stages of the fuel cycle, only casks are available directly for accountancy.

DETECTABILITY OF NM AND DIFFICULTY TO MODIFY PROCESS, TO MODIFY FACILITY DESIGN, TO MISUSE TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores					
		W	S	M/M	C/R	FF	RO	SFS	SFD
Detectability of NM	19. Possibility to identify NM by NDA	No	Yes	S	S	S	S	S	S
	20. Detectability of radiation signature	No reliable signature	Reliable signature	W	W	W	S	S	W
Detectability of misuse of technology or facilities	21. Possibility to detect misuse of the technologies and the NES facilities for processing of undeclared NM	No	Yes	W	W	W	S	S	S
Difficulty to modify facility design	22. Verifiability of facility design by inspectors (transparency of design)	No	Yes	W¹	W¹	W¹	S	S	S
	23. Availability of data for inspectors	NRTA not active	NRTA active	W	S	S	S	S	S²
	24. Transparency of process	No	Yes	W	W	S	S	S	W
Difficulty to modify the process	25. Accessibility of NM to inspectors	No	Yes	S	S	S	S	W³	W

	VW	W	M	S	VS						
26. Extent of automation	N/A	Manual operation	N/A	Partial automation	Full automation	W	VS	S	W	W	W

Notes:

¹ In the framework of this report it is assumed that at the fuel cycle stages with the bulk form of NM, the quantitative verifiability of facility design by inspectors is complicated.

² Near real time accounting is assumed to be feasible during the active life (loading of spent fuel canisters) of the depository.

³ Dry container storage of the spent fuel is assumed.

It is to be emphasized that the assessment performed above focused on the proliferation resistance of fuel in the core that contains thorium.

VIII–5.3. Uranium–plutonium closed fuel cycle

The scheme of this fuel cycle is shown in Fig. VIII–3.

ATTRACTIVENESS OF NM

Group of parameters	Parameter	Evaluation scale					Scores			
		VW	W	M	S	VS	FF	RO	SNFR[1]	WD[1]
Material quality	1. Isotopic composition: ^{239}Pu/Pu (wt%)		>50		<50		W[2]	W[2]	W[2]	W[2]
	2. Heat generation characteristic: ^{238}Pu/Pu (wt%)		<20		>20		W[3]	W[3]	W[3]	W[3]
	3. Spontaneous neutron generation rate characteristic: $(^{240}$Pu$+ ^{242}$Pu$)$ /Pu (wt%)	In comparison to Th/U closed fuel cycle[4]					S[4]	S[4]	S[4]	S[4]
	4. Material type/category	UDU[3]	IDU[3]	LEU	NU	DU	W	W	W	W
	5. Radiation field: dose (mGy/hr) at 1 meter	<150	150–350	350–1000	1000–10000	>10,000	VW[5]	M[6]	VW[5]	VS
Material quantity	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	VW[7]	M[7]	VW[7]	W[7]
	7. No. of items for SQ	1	1~10	10~50	50~100	>100	VS[8]	VW[8]	VS[8]	S
	8. Mass of bulk material for SQ (dilution) (kg)	10	10~100	100~500	500~1000	>1000	W	N/A	W	VS
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VW	VW	VW	VW
Material form	10. Chemical/physical form	Metal	Oxide/solution	Compounds	Spent fuel	Waste	W	S	W	VS

Notes:

1 Spent nuclear fuel reprocessing (SNFR); waste disposal (WD).

2 According to the fast reactor data considered in scenario simulations (NFCSS calculations in the framework of GAINS project), the average ratio of ^{239}Pu/Pu is about 60 wt%, and this value for blanket plutonium is even higher. At the waste disposal stage, 1% of total Pu of the same isotopic composition is assumed to be lost during reprocessing.

3 The average ratio ^{238}Pu/Pu does not exceed 1 wt% at 5 years after discharge (NFCSS calculations in GAINS).

4 The average ratio $(^{240}$Pu $+ ^{242}$Pu$)$/Pu is in a range of 28–31 wt% at 5 years after discharge (NFCSS calculations in GAINS) and the value for blanket plutonium is much lower. Major efforts, such as the introduction of obligatory combined reprocessing of core and blanket fuel [88] are still necessary in this area. Since the evaluation scale for this parameter is not yet well established in the framework of this study, higher scores are given to the U/Pu fuel cycle in order to stress a higher spontaneous neutron generation rate of plutonium only.

5 The judgment is based on the experience of MOX fuel fabrication. The radiation field from MOX fuel depends on the stage of its cleaning. In general, well cleaned MOX fuel does not require any special measures during fuel fabrication safe glove boxes. which is a casual practice in LEU fuel fabrication.

6 Taking into account higher burnups and higher concentrations of MA the core fuel has a higher radiation field than LEU fuel, but the blanket fuel will possess lower radioactivity. Taking into account the level of prospective burnups of blanket fuel, the radiation field can be estimated as being more than the order of magnitude lower than in the case of PWR spent fuel.

7 At the stages of FF and SNFR, one is able to operate with fuel pellets; at RO stage one operates with fuel assemblies, which are apparently lighter than modern PWR assemblies [89, 90]; at the WD stage, one operates with vitrified waste items of limited mass to provide necessary cooling.

8 Following the previous reference, it is assumed that at FF and SNFR stages, 'item' refers to 'pellet' (otherwise, the score could be changed correspondingly in item 6) and assemblies are considered at the stage of RO. At the stage of RO, it is assumed that concentration of Pu in the fresh fuel is 20 wt%, SQ(Pu) = 8 kg, and the mass of the fuel in assembly is more than 40 kg.

ATTRACTIVENESS OF A GIVEN TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores			
		W	S	RO	FF	SNFR	WD
Nuclear technology	11. Enrichment	Yes	No	S	S	S	S
	12. Extraction of fissile material (other than enrichment)	Yes	No	S	S	W	S
	13. Irradiation capability of undeclared fertile material	Yes	No	W	S	S	S

ACCOUNTABILITY, QUALITY OF MEASUREMENT AND AMENABILITY TO C/S MEASURES AND MONITORING

Group of parameters	Parameter	Evaluation scale					Scores			
		VW	W	M	S	VS	FF	RO	SNFR	WD
Accountability	14. σMUF/SQ (for Pu or ^{233}U; ^{235}U with HEU; ^{235}U with LEU)	>2	2~1	1~0.5	0.5~0.1	<0.1	VW^1	VW^2	VW^1	S^3
	15. Inspectors measurement capability	IC only	DA only	Comb NDA/NA	NDA active	NDA passive	M	VW	M	M
			W		S					
Amenability for C/S and monitoring systems	16. Amenability of containment measures		No		Yes		W	S	W	S
	17. Amenability of surveillance measures		No		Yes		W^4	S	S	W
	18. Amenability of other monitoring systems		No		Yes		S	S	S	W

Notes:

1 A standard one year period between two consequent physical inventories of NM is assumed. Assuming 0.1% accuracy of Pu concentration determination (i.e. 0.5 relative %), SQ of Pu can be achieved in 8t of fast reactor core fuel. Annual consumption of the core fuel in the 10GW(e) NES can be estimated at a level of 100 t/a.

2 Unlike the one-through fuel cycle in the case of a closed fuel cycle, a disassembling is envisaged further; the isotope composition definition accuracy in irradiated fuel bundles is far lower than 0.5%; the deviations in NM accounting can be caused by the low accuracy of measurement.

3 The approach is similar to the first reference.

4 At the fuel fabrication stage, surveillance measures are available mainly at the latest steps of the process (assemblies).

Group of parameters	Parameter	Evaluation scale (W)	Evaluation scale (S)	FF	RO	SNFR	WD
Detectability of NM	19. Possibility of identifying NM by NDA	No	Yes	S	S	S	S
	20. Detectability of radiation signature	No reliable signature	Reliable signature	W	S	W	W[1]
Detectability of misuse of technology or facilities	21. Possibility of detecting the misuse of he technologies and the NES facilities for processing of undeclared NM	No	Yes	W	S	W	S
Difficulty to modify facility design	22. Verifiability of facility design by inspectors (transparency of design)	No	Yes	W[2]	S	W[2]	S
	23. Availability of data for inspectors	NRTA not active	NRTA active	S	S	S	S[3]
Difficulty to modify the process	24. Transparency of process	No	Yes	S	S	W	W
	25. Accessibility of NM to inspectors	No	Yes	S	S	S	W

	VW	W	M	S	VS				
26. Extent of automation	N/A	Manual operation	N/A	Partial automation	Full automation	S	W	VS	W

Notes:

1 A long period of consideration is assumed.

2 In the framework of this report, it is assumed that at the fuel cycle stages with the bulk form of NM, the quantitative verifiability of facility design by inspectors is complicated.

3 Near real time accounting is assumed to be feasible during the active life of the depository.

4 Dry container storage of the spent fuel is assumed.

VIII–5.4. Thorium–uranium closed fuel cycle

The fuel of such a fuel cycle may consist of thorium and ^{233}U only (LWBR), or it may include plutonium or ^{235}U/^{238}U of certain enrichment. In the case of involvement of plutonium or enriched uranium, the weak points regarding proliferation resistance of a uranium–plutonium closed fuel cycle will occur also in a thorium–uranium cycle but maybe even at a higher scale. The analysis below is focused on only the Th/^{233}U part of thorium–uranium closed fuel cycle. It is assumed that thorium fuel is placed in separate fuel rods and is never blended (during fabrication) with uranium–plutonium. During reactor operation, small quantities of ^{233}U will be produced in uranium–plutonium fuel in a 'natural' way.

ATTRACTIVENESS OF NM

Group of parameters	Parameter	Evaluation scale					Scores			
		VW	W	M	S	VS	FF	RO	SNFR	WD
	1. Isotopic composition[1]: highest value of ^{233}U/U (wt%) in pin		>12[1]		<12[1]		W[2]	W[2]	W[2]	W[2]
	2. Heat generation characteristic: decay heat emission (watt/crit.mass)		<600[1]		>600[1]		W[3]	W[3]	W[3]	W[3]
Material quality	3. Spontaneous neutron generation rate	In comparison to U/Pu closed fuel cycle[1]					W[3]	W[3]	W[3]	W[3]
	4. Material type/category	UDU	IDU	LEU	NU	DU	W	W	W	W
	5. Radiation field[1]: ^{232}U contamin, for ^{233}U (ppm)	<400[4]	400–1000[4]	1000–2500[4]	2500–25000[4]	>25,000[4]	M[3]	M[3]	M[3]	M[3]
	6. Mass of an item (kg)	10	10~100	100~500	500~1000	>1000	VW[5]	S[5]	VW[5]	W
Material quantity	7. No. of items for SQ	1	1~10	10~50	50~100	>100	VS	VW[6]	VS	VS[7]
	8. Mass of bulk material for SQ (dilution) (kg)	10	10~100	100~500	500~1000	>1000	M	N/A	M	M[8]
	9. No. of SQ (material stock or flow)	>100	50~100	10~50	10~1	<1	VW	VW	VW	VW
Material form	10. Chemical/physical form	Metal	Oxide/Solution	compounds	Spent fuel	Waste	W	S	W	VS

Notes:

1 See corresponding table in Section VIII–5.2.

2 The share of ^{233}U is approximately 98% [5].

3 According to [3] spontaneous fission rate of ^{233}U is 0.5 (s·kg)$^{-1}$. The same parameter for ^{239}Pu (containing 6% of Pu-240) is 2.5×10^4 (s·kg)$^{-1}$.

4 The scale of ^{232}U contamination has been elaborated in the INPRO methodology [60]; it differs from the scale applied for radiation field of U/Pu fuel cycle because of the complimentary factors considered, e.g. the difference of half-lives of the main γ-radioactive isotopes (including predecessors), low concentration of γ-radioactive isotopes (^{224}Ra, ^{220}Rn, ^{212}Bi, ^{208}Tl) in the purified ^{233}U/^{232}U mixture, etc.

5 The same mass as in U/Pu and Th/U one-through fuel cycles (and SNFR=FF).

6 The concentration of ^{233}U is up to 5%, SQ(^{233}U) = 8 kg.

7 Estimated with the same assumptions as for U/Pu closed fuel cycle.

8 From 10 GW(e) NES per annum.

ATTRACTIVENESS OF A GIVEN TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores				
		W	S	RO	FF	RO	SNFR	WD
Nuclear technology	11. Enrichment	Yes	No	S	S	S	S	S
	12. Extraction of fissile material (other than enrichment)	Yes	No	S	S	S	W	S
	13. Irradiation capability of undeclared fertile material	Yes	No	W	S	W	S	S

ACCOUNTABILITY, QUALITY OF MEASUREMENT AND AMENABILITY TO C/S MEASURES AND MONITORING

Group of parameters	Parameter	Evaluation scale					Scores			
		VW	W	M	S	VS	FF	RO	SNFR	WD
Accountability	14. σMUF/SQ (for Pu or ^{233}U; ^{235}U with HEU; ^{235}U with LEU)	>2	2~1	1~0.5	0.5~0.1	<0.1	VW[1]	VW[2]	VW[1]	S[3]
	15. Inspectors measurement capability	IC only	DA only	Comb NDA/NA	NDA active	NDA passive	M	VW	M	M
			W	S	S					
Amenability for C/S and monitoring systems	16. Amenability of containment measures		No		Yes		W	S	W	S
	17. Amenability of surveillance measures		No		Yes		W[4]	S	S	W
	18. Amenability of other monitoring systems		No		Yes		S	S	S	W

Notes:

1. A standard one year period between two consequent physical inventories of NM is assumed. Annual consumption of ^{233}U in the 10 GW(e) NES can be estimated at a level 10 t/a. Assuming 0.05% accuracy of ^{233}U concentration determination (i.e. 1.0 relative %), the result is σMUF=100 kg.

2. Unlike a once-through fuel cycle, in a closed fuel cycle, further disassembling is envisaged. Isotope composition definition accuracy in irradiated fuel bundles is far lower than 1% and deviations in NM accounting can be caused by the low accuracy of measurement.

3. The approach is similar to the first reference.

4. At the fuel fabrication stage, surveillance measures are available mainly at the latest steps of the process (assemblies).

DETECTABILITY OF NM AND DIFFICULTY TO MODIFY PROCESS, TO MODIFY FACILITY DESIGN, TO MISUSE TECHNOLOGY

Group of parameters	Parameter	Evaluation scale		Scores			
		W	S	FF	RO	SNFR	WD
Detectability of NM	19. Possibility to identify NM by NDA	No	Yes	S	S	S	S
	20. Detectability of radiation signature	No reliable signature	Reliable signature	S	S	S	W[1]
Detectability of misuse of technology or facilities	21. Possibility to detect misuse of the technologies and the NES facilities for processing of undeclared NM	No	Yes	W	S	W	S
Difficulty to modify facility design	22. Verifiability of facility design by inspectors (transparency of design)	No	Yes	W[2]	S	W[2]	S
	23. Availability of data for inspectors	NRTA not active	NRTA active	S	S	S	S[3]
Difficulty to modify the process	24. Transparency of process	No	Yes	S	S	W	W
	25. Accessibility of NM to inspectors	No	Yes	W	W	W	W

		VW	W	M	S	VS	FF	RO	SNFR	WD
	26. Extent of automation	N/A	Manual operation	N/A	Partial automation	Full automation	VS	W	VS	W

Notes:

1 A long period of consideration is assumed, which takes into account the rather short life time of strong γ-irradiators produced in the ThFC.

2 In the framework of this report, it is assumed that at the fuel cycle stages with the bulk form of NM, the quantitative verifiability of facility design by inspectors is complicated.

3 Near real time accounting is assumed to be feasible during active life of depository.

4 Dry container storage of the spent fuel is assumed.

VIII–6. AGGREGATION OF PROLIFERATION RESISTANCE ANALYSIS RESULTS

The aggregation of the scores achieved is performed in two steps. In the first step, the scores of both fuel cycle options are compiled in two tables. Table VIII–1 compares the once-through uranium–plutonium cycle with the once-through thorium–uranium cycle, and Table VIII–2 compares the closed fuel cycle options.

All scores regarding proliferation resistance, i.e. very weak to very strong or not applicable (N/A), are collected in the following two tables from Sections VIII–5.1 to VIII–5.4 above and are represented below in the following order:
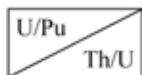
Cells in Tables VIII–1 and VIII–2 showing the same score for both options are coloured in grey and will be eliminated in the following step of aggregation. This approach is stipulated by the goal of our considerations, i.e. a preliminary comparison of options, which differs from an in depth assessment of every separate option.

TABLE VIII–1. COMPARISON OF THE RESULTS FOR ONCE-THROUGH FEUL CYCLE OPTIONS (URANIUM–PLUTONIUM–THORIUM–URANIUM)

No.	Parameter	Fuel cycle stage												
		M/M		C/R		E		FF		RO		SFS		SFD	
1	Isotopic composition	N/A	N/A	N/A	N/A	N/A	-	N/A	N/A	W	W	W	W	W	W
2	Heat generation characteristic	N/A	N/A	N/A	N/A	N/A	-	N/A	N/A	W	W	W	W	W	W
3	Spontaneous .neutron generation rate	N/A	N/A	N/A	N/A	N/A	-	N/A	N/A	S	W	S	W	S	W
4	Material type/category	S	VS	S	VS	M	-	M	VS	W	W	W	W	W	W
5	Radiation field	VW	VW	VW	VW	VW	-	VW	VW	VS	M	S	M	S	M
6	Mass of an item	N/A	N/A	N/A	N/A	N/A	-	VW	VW	S	S	S	S	S	S
7	No. of items for SQ	N/A	N/A	N/A	N/A	N/A	-	W	VS	W	W	W	W	W	W
8	Mass of bulk material for SQ	VS	VS	VS	VS	VS	-	VS	VS	N/A	N/A	N/A	N/A	N/A	N/A
9	No. of SQ	VW	VS	VW	VS	VW	-	VW	VS	VW	VW	VW	VW	VW	VW
10	Chemical/physical form	W	W	W	W	M	-	W	W	W	W	S	S	VS	VS
11	Enrichment	S	S	S	S	W	-	S	S	S	S	S	S	S	S
12	Extraction of fissile material	S	S	S	S	S	-	S	S	S	S	S	S	S	S
13	Irradiation capability of undeclared fertile material	S	S	S	S	S	-	S	S	W	W	S	S	S	S
14	σMUF/SQ	N/A	N/A	N/A	N/A	VS	-	VS	VS	VS	VS	VS	VS	VS	VS
15	Inspectors measurement capability	W	N/A	W	N/A	W	-	M	N/A	VS	VS	VW	VW	VW	VW
16	Amenability of containment measures	W	N/A	W	N/A	W	-	W	N/A	S	S	S	S	S	S
17	Amenability of surveillance measures	W	N/A	W	N/A	W	-	W	N/A	S	S	S	S	W	W
18	Amenability of other monitoring systems	W	N/A	S	N/A	S	-	S	N/A	S	S	S	S	W	W
19	Possibility of identifying NM by NDA	S	S	S	S	S	-	S	S	S	S	S	S	S	S
20	Detectability of radiation signature	W	W	W	W	W	-	W	W	S	S	S	S	W	W
21	Possibility of detecting the misuse of the technologies and the NES facilities	W	W	W	W	W	-	W	W	S	S	S	S	S	S
22	Verifiability of facility design by inspectors (transparency of design)	W	W	W	W	W	-	W	W	S	S	S	S	S	S
23	Availability of data for inspectors	W	W	S	S	S	-	S	S	S	S	S	S	S	S
24	Transparency of process	W	W	W	W	W	-	S	S	S	S	S	S	W	W
25	Accessibility of NM to inspectors	S	S	S	S	S	-	S	S	S	S	W	W	W	W
26	Extent of automation	W	W	VS	VS	VS	-	S	S	W	W	W	W	W	W

TABLE VIII–2. COMPARISON OF THE RESULTS FOR CLOSED FUEL CYCLE OPTIONS (URANIUM–PLUTONIUM–THORIUM–URANIUM)

No.	Parameter	Fuel cycle stage			
		FF	RO	SNFR	WD
1	Isotopic composition	W / W	W / W	W / W	W / W
2	Heat generation characteristic	W / W	W / W	W / W	W / W
3	Spontaneous neutron generation rate	S / W	S / W	S / W	S / W
4	Material type/category	W / W	W / W	W / W	W / W
5	Radiation field	VW / M	M / M	VW / M	VS / M
6	Mass of an item	VW / VW	M / S	VW / VW	W / W
7	No. of items for SQ	VS / VS	VW / VW	VS / VS	S / VS
8	Mass of bulk material for SQ	W / M	N/A / N/A	W / M	VS / VS
9	No. of SQ	VW / VW	VW / VW	VW / VW	VW / M
10	Chemical/physical form	W / W	S / S	W / W	VS / VS
11	Enrichment	S / S	S / S	S / S	S / S
12	Extraction of fissile material	S / S	S / S	W / W	S / S
13	Irradiation capability of undeclared fertile material	S / S	W / W	S / S	S / S
14	σMUF/SQ	VW / VW	VW / VW	VW / VW	S / S
15	Inspectors measurement capability	M / M	VW / VW	M / M	M / M
16	Amenability of containment measures	W / W	S / S	W / W	S / S
17	Amenability of surveillance measures	W / W	S / S	S / S	W / W
18	Amenability of other monitoring systems	S / S	S / S	S / S	W / W
19	Possibility to identify NM by NDA	S / S	S / S	S / S	S / S
20	Detectability of radiation signature	W / S	S / S	W / S	W / W
21	Possibility of detecting the misuse of the technologies and the NES facilities	W / W	S / S	W / W	S / S
22	Verifiability of facility design by inspectors (transparency of design)	W / W	S / S	W / W	S / S
23	Availability of data for inspectors	S / S	S / S	S / S	S / S
24	Transparency of process	S / S	S / S	W / W	W / W
25	Accessibility of NM to inspectors	S / W	S / W	S / W	W / W
26	Extent of automation	S / VS	W / W	VS / VS	W / W

An accurate comparison of the fuel cycle options should consider all scores that do not coincide and all coinciding scores that indicate weak or very weak proliferation resistance for a particular item. Additionally, as stated above in Section VIII–6, compensatory measures must be defined to improve all very weak (or weak) points regarding proliferation resistance found in the analysis. For such items with low proliferation resistance, the defined compensatory measures should be compared in the fuel cycles from the point of view of their efficiency and cost.

In our preliminary assessment here, we assume that the cost and efficiency of compensatory measures are similar for the items with coinciding scores of options under comparison. The comparison of these options roughly reflects main trends because thorium application usually implies the usage of (enriched) uranium or plutonium (see Sections 6.2.2 and 6.2.4).

VIII–7. DISCUSSION OF RESULTS OF THORIUM FUEL CYCLE PROLIFERATION RESISTANCE ANALYSIS

To facilitate the comparison of the fuel cycles, data presented in Tables VIII–1 and VIII–2 were further reduced: items (coloured in grey) were eliminated where the scores coincided regardless of whether they were positive or negative (strong or weak proliferation resistance). This is justified because the goal of the study was not a comprehensive assessment of a single option, but a comparison of options. In a case of equally negative scores (weak proliferation resistance), the possible difference in necessary follow-up actions between options (to compensate weak proliferation resistance features) may be significant from a financial and technical point of view, but this difference has not yet been taken into account. The result of this reduction of data is shown in Tables VIII–3 and VIII–4.

The results for uranium–plutonium and thorium–uranium fuel cycles are presented in the same format as above:

At the stage of enrichment (absent in Table VIII–3), the scores achieved in Section VIII–6 are, apparently, in favour of thorium. In the case of once-through fuel cycle options (Table VIII–3), differences between these scores were not taken into account because the reactors that would use thorium in the fuel cycle without reprocessing would use enriched uranium as driving fuel (see Section VIII–5.2). Although the introduction of thorium may
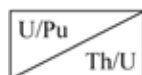
U/Pu
Th/U

TABLE VIII–3. AGGREGATED RESULTS FOR ONCE-THROUGH FUEL CYCLE OPTIONS (URANIUM–PLUTONIUM–THORIUM–URANIUM)

No.	Parameter	M/M	C/R	FF	RO	SFS	SFD
3	Spontaneous .neutron generation rate				S / W	S / W	S / W
4	Material type/category	S / VS	S / VS	M / VS			
5	Radiation field				VS / M	S / M	S / M
7	No. of items for SQ			W / VS			
9	No. of SQ	VW / VS	VW / VS	VW / VS			
15	Inspectors measurement capability	W / NN	W / NN	M / NN			
16	Amenability of containment measures	W / NN	W / NN	W / NN			
17	Amenability of surveillance measures	W / NN	W / NN	W / NN			
18	Amenability of other monitoring systems	W / NN					

TABLE VIII–4. AGGREGATED RESULTS FOR CLOSED FUEL CYCLE OPTIONS (URANIUM–PLUTONIUM–THORIUM–URANIUM)

No.	Parameter	FF	RO	SNFR	WD
3	Spontaneous neutron generation rate	S / W	S / W	S / W	S / W
5	Radiation field	VW / M		VW / M	VS / M
6	Mass of an item		M / S		
7	No. of items for SQ				S / VS
8	Mass of bulk material for SQ	W / M		W / M	
9	No. of SQ				VW / M
20	Detectability of radiation signature	W / S		W / S	
25	Accessibility of NM to inspectors	S / W	S / W	S / W	
26	Extent of automation	S / VS			

potentially reduce the requirements for enrichment, an essential reduction does not seems to be realistic (see Section 4.5), and without recycling it, definitely cannot fully eliminate this process. Effects of enrichment reduction related to thorium introduction are design specific and may be carefully investigated in upcoming projects with an increased number of reactor options.

In the once-through variant, thorium fuel can provide certain advantages regarding proliferation resistance, but the necessity for enriched uranium (usually up to 20% of ^{235}U/U) as a driving fuel in the core and the corresponding content of spent fuel eliminate such benefits.

The proliferation resistance related features of thorium closed fuel cycles differ from features in traditional uranium/Pu closed cycles; nevertheless, a general conclusion cannot be drawn on the superiority of any options considered.

In the closed fuel cycle variant, since the benefits and deficiencies divided 50/50, thorium can hardly contribute substantially to an increase of proliferation resistance of a closed fuel cycle.

As mentioned in Ref. [3], additional barriers provided by thorium are not efficient enough in the case of a country with developed nuclear infrastructure (e.g. with reprocessing, hot cells). However, it must be owned that, to date, globally, no nuclear weapon has been based on the fissile ^{233}U produced in thorium. The options considered here are not the only possible variants of ThFC introduction, and other comparisons based on different approaches, inputs and models are also possible. The study in this section is more a demonstration of the method (i.e. of INPRO methodology application) than substantiation of any kind of firm conclusion.

ACRONYMS AND ABBREVIATIONS

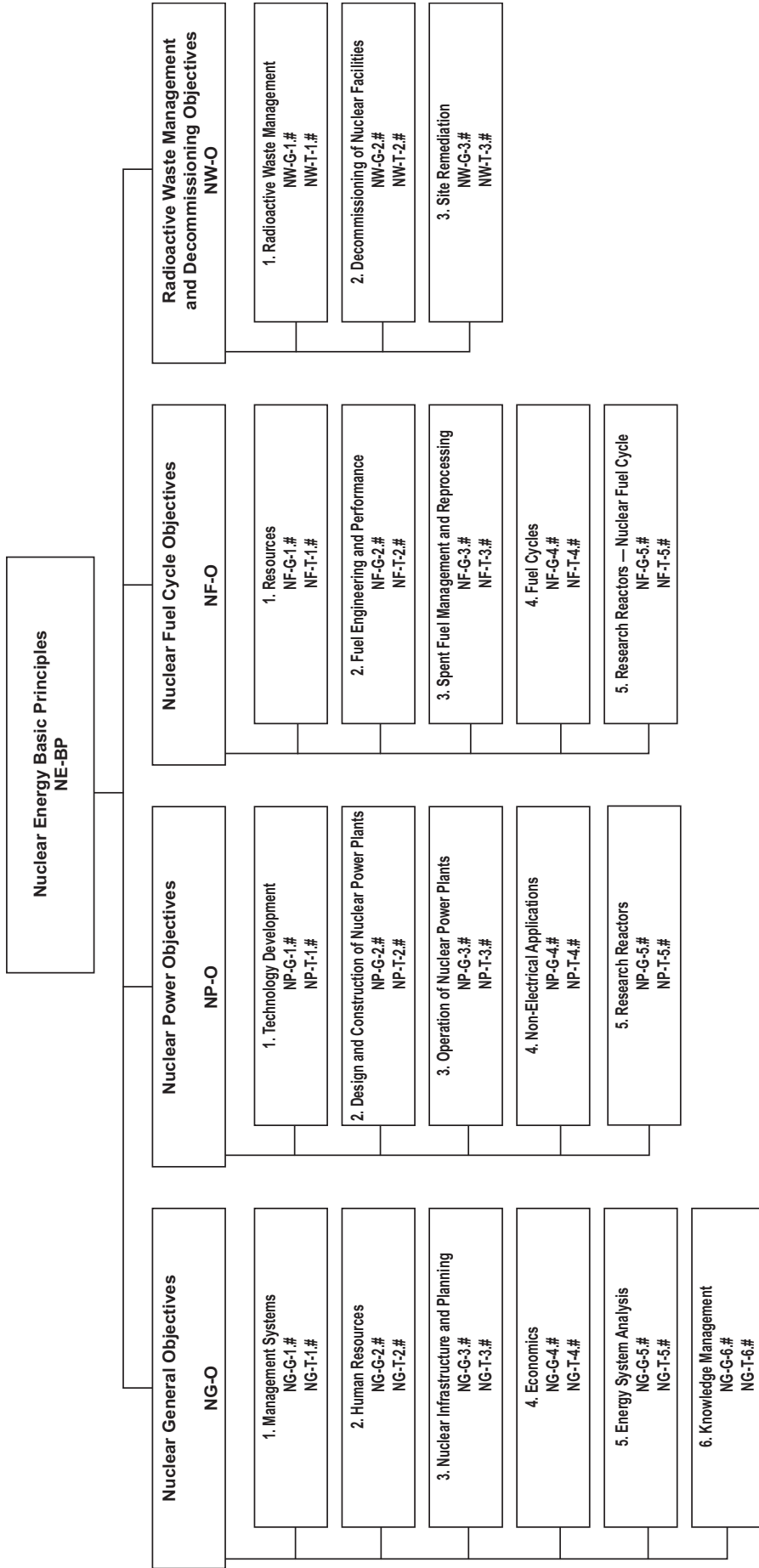|	
ACR	Advanced CANDU Reactor
ADS	accelerator driven system
AECL	Atomic Energy Canada Limited
AHWR	advanced heavy water reactor with plutonium, ^{233}U, thorium fuel
ALWR	advanced light water reactor with UOX fuel
BAU	business as usual
BN	fast reactor (Russian design)
BOL	beginning of life
BR	breeding ratio
BWR	boiling water reactor
CEA	Commissariat à l'Énergie Atomique (France)
CR	conversion rate
CRL	Chalk River Laboratories, Canada
Cm	curium
CSA	comprehensive safeguards agreement
CVR	void reactivity coefficient
CZP	cold zero power
EFPD	effective full power day
FGR	fission gas release
FP	fission products
FR	fast reactor with MOX fuel (CR ~1), depleted uranium in blankets
FRTh	fast reactor with plutonium, depleted uranium fuel, and thorium in blankets
FTC	fuel temperature reactivity coefficient
GAINS	global architecture of innovative nuclear systems (INPRO collaborative project)
GIF	Generation IV International Forum
GW	gigawatt
HTR	high temperature reactor with ^{233}U thorium fuel
HM	heavy metal
HW	mass balance of heavy metals
HWR	heavy water reactor with natural uranium fuel
HWR1	heavy water reactor with plutonium thorium fuel
HWR2	heavy water reactor with plutonium, Th, ^{233}U fuel
ID	inner diameter
IHE	initial heavy elements
INPRO	International project on innovative nuclear reactors and fuel cycles
IPCC	Intergovernmental Panel on Climate Change
KAERI	Korea Atomic Energy Research Institute
kt	kilotons
LEU	low enriched uranium
LUEC	levelized unit energy cost
LWBR	light water breeding reactor
LWR	light water reactor with UOX fuel
LWR0	light water reactor with UOX Th fuel
LWR1	light water reactor with plutonium Th fuel
LWR2	light water reactor with plutonium, ^{233}U, depleted uranium fuel
LWR3	light water reactor with Th ^{233}U fuel
LWRM	light water reactor with MOX fuel
MA	minor actinides
MOX	mixed oxide fuel uranium–plutonium
MSBR	molten salt breeder reactor

MSR	molten salt reactor
NEA	Nuclear Energy Agency (OECD)
NES	nuclear energy system
NESA	nuclear energy system assessment (using the INPRO methodology)
NEST	NESA economic support tool
NM	nuclear material
Np	neptunium
NPP	nuclear power plant
OD	outer diameter
O/M	operation and maintenance
ORNL	Oak Ridge National Laboratory (USA)
OTT	once-through thorium fuel cycle
PCMI	pellet cladding mechanical interaction
PHWR	pressurized heavy water reactor
PR	proliferation resistance
Pu	plutonium
PWR	pressurized water reactor
RBWR	reduced moderation BWR
RG	reactor grade (plutonium)
RSAC	regional system of accounting of nuclear material
SA	subassembly
SF	spent fuel
SLWRM	scenario for light water reactors with MOX
SNF	spent nuclear fuel
SSAC	state system of accounting of nuclear material
SWU	separative work unit (enrichment)
Th	thorium
ThFC	thorium fuel cycle (INPRO collaborative project)
U	uranium
UOX	uranium oxide
UR	User Requirement (INPRO)
WWER	pressurized water reactor of Russian design

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