Fissile Material Management Strategies for Sustainable Nuclear Energy

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FISSILE MATERIAL MANAGEMENT STRATEGIES FOR SUSTAINABLE NUCLEAR ENERGY

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FOREWORD

There was a pivotal point in the development of nuclear energy, which took place around 1990. Up to this time most Member States that had implemented nuclear energy programmes envisaged development of a closed nuclear fuel cycle by employment of fast breeder reactors (FBRs) to efficiently use plutonium as an energy source. Owing to the substantial reduction in FBR development programmes after 1990, new realities in nuclear energy have emerged, which have entailed an increase in the inventory of separated plutonium. In these new circumstances, a number of issues and challenges have appeared, such as diverse strands of public opinion regarding nuclear energy, the balance of supply and demand in the uranium market, the continued increase in the plutonium inventory, whether separated or not, the growing inventories of spent fuel and waste, environmental protection issues, safety concerns and proliferation risks. Member States with nuclear energy programmes are addressing these issues through new technologies, installation of new facilities and expansion of the capacities of existing facilities.

Under such circumstances, nuclear energy is approaching a new pivotal point, where fissile material management will play an essential role in the nuclear fuel cycle. Accordingly, the IAEA organized a technical meeting on fissile material management strategies for sustainable nuclear energy in order to provide essential information to Member States for their policy making and strategic planning needs. The meeting was held in Vienna from 12 to 15 September 2005. The purposes of the meeting were:

(a) To identify fissile material management strategies for different nuclear fuel cycle options;
(b) To clarify the issues and challenges existing in fissile material management;
(c) To seek possible solutions for these issues and challenges, focusing, in particular, on the sustainability of nuclear power in different fuel cycle options.

Coverage of these activities is focused on the technical aspects of fissile material management rather than on institutional aspects or physical protection.

Prior to the technical meeting, three working groups composed of experts from ten countries were organized to prepare key issue papers on three topics that were selected as the principal themes for the meeting. The front end fuel cycle group prepared a paper entitled ‘Uranium demand and supply up to 2050’. The back end fuel cycle group prepared a paper entitled ‘Fissile material

management strategies for sustainable nuclear energy: Back end fuel cycle options'. The future fuel cycle technologies group prepared a paper entitled 'Sustainable nuclear energy development beyond 2050: Cross-cutting issues'. In addition to these three key issue papers, 32 papers, prepared by the working group members and invited participants, were presented at the technical meeting.

The information contained in these proceedings should provide a basis for policy makers and the public to discuss and explore different options for the nuclear fuel cycle and contribute to strategic planning by Member States.

The IAEA wishes to thank all of the members of the working groups, in particular the chairpersons of the groups, J. McMurray (United States of America), H. Bairiot (Belgium) and L. Koch (Germany), and all those who contributed to the success of the technical meeting.

The IAEA officer responsible for this publication was K. Koyama of the Division of Nuclear Fuel Cycle and Waste Technology.

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INTRODUCTION

1. BACKGROUND

A technical meeting, Fissile Material Management Strategies for Sustainable Nuclear Energy, was held in Vienna, from 12 to 15 September 2005. The meeting was organized by the IAEA to:

(a) Identify fissile material management strategies for different nuclear fuel cycle options;
(b) Clarify issues and challenges existing in fissile material management;
(c) Seek possible solutions for these issues and challenges, focusing, in particular, on the sustainability of nuclear power in different fuel cycle options.

Prior to the technical meeting, three working groups composed of experts from ten countries prepared key issue papers on three topics that were selected as the principal themes for the technical meeting.

The front end fuel cycle group (Working Group 1) prepared a key issue paper, 'Uranium demand and supply up to 2050' (Paper 1.1) that presents assessments of the adequacy of uranium resources to meet uranium requirements related to reactors for a range of demand projections.

The back end fuel cycle group (Working Group 2) prepared a key issue paper, 'Fissile material management strategies for sustainable nuclear energy: Back end fuel cycle options' (Paper 2.1) that discusses and assesses possible scenarios up to 2050, for the best and most timely utilization of fissile material resources available in spent nuclear fuel and in already separated plutonium and uranium, taking into account existing and developable industrial infrastructures, as well as the effects on the environment, on personal exposure, on waste characteristics and quantities, and on fuel cycle costs.

The future fuel cycle technologies group (Working Group 3) prepared a key issue paper, 'Sustainable nuclear energy development beyond 2050: Cross-cutting issues' (Paper 3.1) that assesses and evaluates future sustainable fuel cycle technology options beyond 2050, in particular:

(a) An open fuel cycle using all possible uranium resources;
(b) Plutonium breeding in fast neutron energy reactors;
(c) Thorium fuel cycles.

This paper also compares the performance of these options from various points of view, such as cross-cutting fuel cycle issues of developments in technology, economic potential, environmental impact of waste and proliferation resistance.

In addition to the above three key issue papers, 32 papers, prepared by working group members and invited participants, were presented during the three technical sessions of the meeting:

(1) In the session on front end fuel cycle strategies, nine papers (Papers 1.2–1.10) that address issues related to the front end of the fuel cycle;
(2) In the session on back end fuel cycle strategies, thirteen papers (Papers 2.2–2.14) that address issues related to the back end of the fuel cycle;
(3) In the session on future fuel cycle technology options, ten papers (Papers 3.2–3.11) that address issues related to future fuel cycle technologies.

2. SESSION 1: FRONT END FUEL CYCLE STRATEGIES

The primary objective of the session on the front end of the nuclear fuel cycle was to assess the adequacy of uranium resources and production capacity to meet reactor related uranium requirements up to 2050.

Three demand projections were established against which to measure the adequacy of uranium resources and production capacity to meet demand up to 2050 (Paper 1.1). The reference and high demand cases project average annual increases in uranium requirements up to 2050 of 1.6 and 2.5%, respectively. The low demand case envisions a gradual phasing out of nuclear power by 2100. Although uranium is clearly the main focus of the session on the front end of the nuclear fuel cycle, for completeness a paper on thorium resources was also included (Paper 1.9). Thorium resources are projected to far exceed demand for the foreseeable future.

Uranium supply is divided into two broad categories: firstly, newly mined and processed uranium, and secondly, primary supply and secondary supply, which includes highly enriched uranium (HEU), inventory drawdown, mixed oxide (MOX), reprocessed uranium (RepU) and re-enrichment of tails. Projected annual availability of secondary supply is subtracted from the total reactor related uranium requirements to project annual primary supply requirements.

In 2005, primary supply and secondary supply are expected to cover 57 and 43% of total demand, respectively. To offset uncertainty regarding availability of secondary supply, a conservative projection was used, which assumes that the availability of secondary supply will continue to decrease and will

account for only about 8% of total uranium requirements by 2050 in the reference demand case.

A very strong case can, however, be made that additional sources of secondary supply could become available that would be adequate to cover up to 25% of total requirements annually up to 2050 (Paper 1.2). The largest contribution will probably come from reduction in enrichment tails assays. Additional excess military fissile material may also be made available for civilian use. Increased availability of secondary supply will reduce demand for primary supply; the magnitude of the reduction will depend on the economics of both supply sources. Secondary supply has been fully integrated into the market; in the near-term it will be important for filling the gap between primary supply and uranium requirements. In the longer term, secondary supply will continue to be an important supplement to primary supply in ensuring a balance between supply and demand.

Uranium resources, on which primary supply is based, are classified according to confidence level and production cost. Reasonably assured resources (RARs), the resources in which confidence is highest, are mostly attributable to known deposits. Lower confidence resource totals, including inferred, prognosticated and speculative resources, are reported by Member States for publication in the Red Book. To provide an economic context to the balance between supply and demand, uranium resources are assigned production cost categories. Determining when a given cost category will be needed to balance supply and demand provides an indirect indication of uranium market price trends.

Uranium resources are only one part of the supply side of the supply–demand equation. The ability of the industry to develop resources and then deliver them to the marketplace can, in the near-term, be more important as a supply issue than the resources in the ground. As demand grows so too will the need to develop new production capacity (Paper 1.6). Because low grade, mainly low capacity, deposits dominate the resource base, there may need to be an exponential growth in the number of mines needed to satisfy future demand. As the number of mines increases so too will the need to recruit engineers, geologists and miners to plan and operate them. As is already the case, however, the uranium industry will continue to face competition in the recruitment of new technical and operating personnel (Paper 1.7), presenting a significant challenge to the needed expansion.

Reasonably assured resources, the highest confidence resource category, are projected to be adequate to satisfy primary supply requirements up to 2038 (Paper 1.1). By comparison, in the high demand case, RARs are adequate to satisfy demand only up to 2031. The supply–demand analysis was based on the assumption that enrichment tails assays will remain at approximately 0.3% ^{235}U

up to 2050. Lowering the tails assay, which is increasingly a realistic assumption in the present economic environment (Paper 1.2), will result in a reduction in primary supply requirements, such that RARs will be adequate to balance supply and demand in the reference demand case, but not in the high demand case.

Without savings from reduced tails assays, inferred resources will be required to balance supply and demand in the reference case. Addition of the next lowest confidence resource category, prognosticated resources, will be required to balance supply and demand in the high demand case. These lower confidence resources will, however, require a great deal of additional drilling and engineering evaluation to upgrade them to high confidence status.

There are two remaining resource categories that can be called upon to offset potential supply shortfalls. Speculative resources totalling 4 400 000 t of uranium provide a buffer to offset future supply shortfalls. Speculative resources are, however, based on indirect evidence and extrapolations; a large percentage of speculative resources have not yet had the benefit of a single hole being drilled. In addition to speculative resources, if the market price of uranium continues to rise, production from unconventional resources could once again become economically viable. For example, by-product recovery of uranium during phosphoric acid production could be renewed if the uranium market price exceeds US $115/kg U. Uranium resources associated with phosphorite deposits are estimated to total 9 000 000 t of uranium (Papers 1.1 and 1.9).

The recent tripling of the spot market price is attributable to a number of factors that occurred in rapid succession, including supply disruptions at large production facilities, uncertainty regarding the availability of HEU, potential mine closures and weakness of the US dollar relative to the currencies of many producing countries. The real or perceived loss of a number of supply sources resulted in re-evaluation of strategic inventory levels, which in turn led to an increase in market demand at a time when it was not readily available at then existing market prices. The recent increase in market price is the result of concern about the ability of the industry to deliver resources to the marketplace in a timely manner, not concern about adequacy of resources.

Higher prices typically stimulate additional uranium supply, either from expansion of existing production centres or development of new mines. These new or expanded supply sources have the potential to displace higher cost sources. The primary supply response to higher prices may take a number of years to fully develop as new mines are committed for construction. Once there is market consensus that future supplies are adequate, the market price could return to the levels projected by the supply–demand model. Even if primary

supply concerns continue, lower tails assays will probably help moderate the pace of price increases by reducing the demand for primary supply.

Expansion of existing mines and development of new mines is the key to assuring an adequate future supply of uranium. There are, however, two obstacles that have the potential to disrupt or delay new mine development (Paper 1.10). Opposition to uranium mining has already delayed new mine development in Australia and the United States of America. These delays have affected deposits that are projected to have relatively low production costs, which will probably put upward pressure on uranium prices. Of equal concern to the uranium industry is the emergence of a regulatory climate in which unreasonable environmental constraints impair project viability by increasing costs and even lead to loss of licences to explore for and develop new primary supplies. In some jurisdictions, regulatory constraints are being imposed that benefit neither the environment nor the uranium production industry (Paper 1.10).

While this study emphasizes uranium resources and production capacity, it also addresses conversion and enrichment capacity as key activities at the front end of the nuclear fuel cycle. There will almost certainly be a need to construct new conversion and enrichment capacity as the availability of secondary supply diminishes. Vertical integration within the industry, whereby major uranium producers are also involved in conversion and enrichment will help ensure adequate development of these services.

The future of nuclear power depends on an adequate supply of uranium and fuel services to meet future demand. High confidence RARs are projected to be adequate to satisfy uranium requirements up to 2037 in the reference demand case. Addition of lower confidence inferred resources will balance the reference case up to 2050. Still lower confidence prognosticated resources will be required to balance the high demand case. A large speculative resource base remains available to satisfy the even higher demand projections that have been published (Paper 1.3).

The uranium industry is well positioned to meet future primary supply requirements. Recent increases in the uranium market price have stimulated exploration targeted at expanding the high confidence resource base. Exploration could well double compared with current levels by 2010 (Paper 1.6). Increased exploration will, however, require corresponding manpower increases at a time when the industry is facing a shortage of experienced personnel and contractors that provide drilling and geophysical services (Paper 1.7).

Higher prices have also stimulated development of new mines and expansion of existing operations in Canada (1.5), Kazakhstan (1.4, 1.8), Namibia and South Africa. Exploration and mine development are taking

place on a global scale, which will help ensure a broadly diversified uranium production industry. However, because of potential opposition to development of new mines, adequate lead times must be planned as buffers against unforeseen delays. The nuclear industry cannot afford to risk a tight balance between supply and demand in which there is no margin for error.

3. SESSION 2: BACK END FUEL CYCLE STRATEGIES

There are currently three options being followed for the management of spent nuclear fuel (SNF). These options can be described as:

(1) Reprocessing and recycling of valuable fissile materials (closed fuel cycle);
(2) The 'once-through' cycle (open fuel cycle);
(3) The 'wait and see' option.

Whilst the wait and see option must ultimately evolve into reprocessing and recycling or the once-through option, it is at present a valid management strategy that simply focuses on the long term storage of spent fuel.

For the three strategic options, the key factors centre on how the fissile material present in the SNF can best be managed up to 2050 to extend the sustainability of nuclear energy as follows:

(a) The quantities and qualities of the fissile material present in the various types of SNF and their evolution;
(b) The availability of the technologies and their degree of maturity in terms of significant industrial deployment;
(c) The further deployment rates needed to meet demand;
(d) The implications of utilizing fissile material sooner or later;
(e) The economic implications;
(f) The criteria influencing the choice of back end strategy.

At the end of 2003, there were 439 operating nuclear reactors worldwide; the majority of these were light water reactors (LWRs) (213 pressurized water reactors (PWRs), 50 WWERs and 92 boiling water reactors (BWRs)), accounting for almost 90% of the installed nuclear capacity. Even though the rate of building new reactors has slowed dramatically, the contribution of nuclear power has continued to grow. This growth has been achieved primarily through increased performance, including lifetime extension and power uprates of the existing reactors. In Paper 2.1, quantitative predictions have

been undertaken using the VISTA code (Paper 2.2) to assess three reactor deployment scenarios up to 2050, reaching 400 GW(e) (low), 565 GW(e) (medium) and 730 GW(e) (high), three reprocessing scenarios and three MOX fuel utilization scenarios. These 27 case studies result in a forecast of fissile material inventories in the back end of the fuel cycle (Paper 2.3). Since the range of future inventories forecasted is large, an IAEA paper (Paper 2.4) advocates establishment of an international database of spent fuel inventories and management for transparency of information, as is already the case for plutonium management reporting by nine Member States under INFCIRC/549 [1].

The quality of fissile material in SNF, i.e. the isotopic compositions of uranium (Paper 2.8) and plutonium (Papers 2.1 and 2.7), has deteriorated over the years as a result of continuously increasing discharge burnups. This tendency will persist in the future, but not indefinitely, as higher discharge burnups require higher enrichments of the fresh fuel, and industrial and regulatory infrastructures are limited to 5% uranium enrichment. As plutonium is the most radiotoxic constituent of spent fuel, separating and burning it as MOX fuel reduces the long term environmental legacy of back end fuel cycle wastes. The recycling of plutonium in LWRs has a good safeguards record and is the only industrially available option for plutonium utilization. However, recycling the plutonium in LWRs increases the levels of americium, which is the next most long term radiotoxic element in fuel waste, and curium, a short term high heat emitter in fuel waste. Additionally, LWRs cannot fully consume the plutonium as its isotopic characteristics deteriorate over time (Papers 2.1, 2.6 and 2.7).

The three strategic options require extended storage of SNF and/or high level waste from reprocessing. Industrial experience has been amassed, which demonstrates the viability of the storage technologies. Large industrial plants, serving the international market, are currently available in France and the United Kingdom (UK) for reprocessing of spent fuel, and additional facilities are in operation or under construction in Japan, India, the Russian Federation and China. Plutonium and uranium separated by reprocessing have been recycled on an industrial scale for many years, and there is considerable experience of all the activities required to support the recycling option. Deep geological disposal of spent fuel and/or high level waste, which is ultimately required for all three options, is technically developed to a point that generates adequate confidence in its deliverability in the future, but this has not yet been industrially implemented.

Meeting future demands is not a problem for the industrially established technologies (storage, reprocessing and recycling of plutonium and of reprocessed uranium), except for a timely expansion of the capacities of the

corresponding facilities, for which the timeliness of deployment should not be underestimated (Paper 2.1). Geological disposal, not yet industrially deployed, might need to be optimized to reduce the quantities and long term radioactivities of the packages. Strategies, benefits and challenges are being investigated to evaluate reprocessing, P&T of actinides and long lived fission products (Papers 2.10 and 2.11). In particular, recycling plutonium and americium reduces the environmental impact of final disposal and increases repository loading density (Paper 2.10). Recycling of plutonium and reprocessed uranium already provides significant relief (Paper 2.1). Both long term storage and final disposal have to cope with safety relevant aspects that might have an impact on the sustainability of these practices (Paper 2.9).

Recycling of plutonium should take into account the ageing effect of plutonium. Owing to radioactive decay, the fissile value of plutonium deteriorates irreversibly (Papers 2.1 and 2.7) between the time spent fuel is discharged from the reactor and MOX fuel is recycled into an LWR or fast neutron reactor (FR). Although recycling in LWRs cannot fully consume the plutonium, as its isotopic characteristics deteriorate with recycling (Paper 2.1, Annex 2), long term storage of spent fuel or separated plutonium until such a time that fast reactors might be deployed is not the optimum technical solution. Moreover, it is imperative to pursue reprocessing and MOX fuel management on an industrial scale for the technology to be available and well mastered by the time deployment of FRs will become a strategic necessity. On the timescale that FRs may be required, the use of plutonium whose characteristics have deteriorated due to previous recycling in LWRs, or with an originally low fissile content due to the increased discharge burnup foreseen by 2050, will not be a problem as FRs are less sensitive to the isotopic composition of plutonium than LWRs. Fast neutron reactors can also eliminate americium to a major extent if reprocessing technology is properly adapted to separate it, and fabrication and demonstration of americium targets has progressed.

Unlike plutonium, the fissile composition of reprocessed uranium does not vary with age so that reprocessed uranium can be kept as a strategic fissile resource almost indefinitely. Significant industrial recycling of reprocessed uranium in LWRs, RBMKs and (in the UK) advanced gas cooled reactors (AGRs) has demonstrated the value of reprocessed uranium as a strategic resource. Reprocessed uranium is also highly attractive for recycling in heavy water reactors (HWRs), whose neutron flux spectrum is less sensitive to the poisoning effect of ^{236}U than other reactor systems.

The economics of the fuel cycle are affected by recycling of plutonium and/or uranium. Recovery of these fissile materials will probably not become economically feasible for HWR, RBMK, AGR and GCR systems. It is applicable only to spent fuel from LWRs and FRs. The cost disadvantage of the

closed fuel cycle will persist, even if potential variations in reprocessing prices and other fuel cycle costs and parameters are taken into account (Paper 2.5). This disadvantage affects the overall generating cost, but not to the extent that it could influence the future sustainability of nuclear energy (Paper 2.1). In the future, uranium price and geological disposal costs are the most likely unit costs to increase, reducing the closed cycle economic disadvantage (Paper 2.1). Economic considerations will, however, not be a sufficient incentive for deploying by 2050 large numbers of new reactors better suited to utilizing plutonium and reprocessed uranium (Papers 2.12 and 2.13), although FRs might well start to play a role by 2050. The DUPIC technology could effectively use in a single recycle both fissile materials if the reactor park contains an appropriate proportion of LWRs and HWRs (Paper 2.14).

Several criteria have been taken into account in considering the advantages and disadvantages of the different fuel cycle back end options (Paper 2.1):

(a) *Use of fissile resources*: In this respect, plutonium recycling without delay provides for the best utilization of fissile material resources.
(b) *Availability of technology*: Only geological disposal, which is ultimately required for the three back end strategies, is not yet deployed on an industrial scale. The closed fuel cycle is the least affected.
(c) *Economics*: As already indicated, economics have only a minor impact.
(d) *Environmental impact*: Recycling strategies reduce the long term radiotoxicity legacy, at the expense of slightly increasing the current occupational dose exposure. The total collective dose remains, however, at the same order of magnitude, i.e. very small as compared with radioactive exposures from other sources unrelated to nuclear power production.
(e) *Public acceptance*: Provided the public is adequately informed, any strategy would be acceptable.
(f) *Proliferation concerns*: The increased discharge burnup of spent LWR fuel and, even more so, the plutonium recycling renders plutonium progressively easier to detect (simplifying safeguards implementation) and less attractive as a material for manufacturing nuclear explosives.

In conclusion, considering the requirements of sustainable development of nuclear energy, fissile material in spent fuel constitutes a valuable resource, with established technology for its utilization. In the period up to 2050, this fissile material will continue to be utilized industrially. It is essential to maintain the expertise and infrastructure that will be needed to support deployment of contemplated new reactor systems and scenarios for sustainable nuclear

energy, most of which assume larger scale utilization of the resource circa 2050 and beyond.

4. SESSION 3: FUTURE FUEL CYCLE TECHNOLOGY OPTIONS

During the third session of the technical meeting, information was presented that will aid decision makers in elaborating a future nuclear fuel management policy beyond 2050. Before a widespread deployment of any new nuclear fuel cycle concept could start, a lead time of 30 years or more might be required. In order to obtain sustainable public support, the new concepts must meet concerns about environmental compliance and proliferation. Together with other cross-cutting issues of sustainability in resources, economics and energy supply, the future nuclear fuel cycle options are discussed and compared.

Sustainable fuel management can build on three main options:

(1) Direct fission of ^{235}U, the only fissile nuclide in sufficient quantities still on the earth;
(2) Breeding fissile plutonium from already abundant and fertile ^{238}U;
(3) Breeding fissile ^{233}U from equally abundant and fertile ^{232}Th.

The once-through uranium fuelled thermal reactor strategy is at present preferred, although the option for disposition of the spent fuel is not universally agreed upon. Since plutonium remains in spent fuel, the repositories could serve as 'plutonium mines', presenting a proliferation concern for future generations. To continue to supply uranium for centuries in an open fuel cycle strategy will eventually require exploitation of unconventional supply resources. Speculative uranium resources are, however, expected to be adequate up to 2050 (Papers 1.1 and 3.1) and perhaps for a much longer time (Paper 3.2). In addition, uranium in sea water is a potential source for the distant future (Paper 3.2), but it will probably not be needed during this century.

Current SNF management in a closed fuel cycle consists of PUREX processing and MOX in some 35 LWRs; this total will increase in the future. Thermal plutonium recycling reduces high level waste volume for geological storage and the uranium resource requirements by 15% or more. Removing plutonium compensates for the increase of heat producing and highly radiotoxic minor actinides (MAs) in the thermal recycling mode. Burning plutonium in an LWR increases the ^{240}Pu content, which makes the plutonium less usable as a nuclear explosive material. The preferable option for the future

is multi-recycling of all transuranic elements in FRs. The transition from LWRs to FRs and consequently the transmutation of the accumulated transuranic elements strongly depend on the recycling mode and the introduction of suitable nuclear reactors (Paper 3.6). The present obstacles to introducing FRs are the higher investment cost for the fuel cycle facilities, the processing of the radiotoxic plutonium and the proliferation risk. With rising uranium prices and final disposal expenses, the cost threshold from LWRs to FRs (with minor actinide transmutation) will eventually be crossed. Beyond 2050 the technology for recycling of all transuranic elements in Generation IV reactors is being considered. By then fast reactors will be favoured anyway since they are the only ones to offer the necessary extra neutrons (Papers 3.4–3.6). One envisaged scenario for Japan provides an example of a future closed fuel cycle with transuranic element management (Paper 3.5).

There has been renewed interest in thorium fuels and fuel cycles because of the favourable properties of thorium oxide as a matrix for high burnup fuels, the possibility of self-sustaining ^{232}Th–^{233}U in several thermal and fast reactor systems, and the attractive features of thorium related to the Accelerated Driven System (ADS) (Paper 3.8). The former Molten Salt Breeder Reactor (MSBR) has been re-evaluated and, with a breeding ratio of one, it is also possible to avoid continuous reprocessing and the separation of ^{233}Pa (Paper 3.9).

Environmental impact management benefits from minimizing long lived radiotoxicity by partitioning and transmutation (P&T) processes and from limiting land consumption and groundwater contamination, which are achieved by fuel cycle strategies requiring less uranium mining and geological repositories.

Partitioning of all the transuranic elements can be accomplished by pyrochemical or aqueous processes (Paper 3.3), like the group extraction of actinides, GANEX, under development in France (Paper 3.6), which co-separates the transuranic elements. Previously developed aqueous partitioning processes often separate individual elements, usually following plutonium recovery (Paper 3.3).

The transmutation strategy will destroy MAs in Generation IV reactors, which will allow for global recycling of all actinides with optimum use of natural resources. Furthermore, this global recycling will favourably impact on the uranium cost and availability as well as on proliferation concerns about ^{235}U enrichment. Global recycling is considered for both thermal and fast reactor cores. Fast reactors are favoured for Generation IV general objectives (Papers 3.1–3.7). Transmutation schemes for MAs are largely classified into homogeneous recycling, heterogeneous recycling and recycling confined in a dedicated system. Optionally the long life fission products such as ^{99}Tc and ^{129}I

could also be transmuted, in order to achieve further reduction of long term radioactivity release risk from a nuclear waste repository. Research and development (R&D) on P&T have so far accumulated data for evaluating the technical feasibility of transmuting minor actinides and long lived fission products in either a dedicated recycling scheme for an accelerator driven system or in homogeneous or heterogeneous recycling in FRs. However, the technology for handling these highly radioactive materials in the fuel cycle is still at an early stage. Significant development is required to produce fuel or targets containing MAs for heterogeneous transmutation; there is less development effort necessary for achieving homogeneous transmutation in FRs. Some fundamental information, such as nuclear data on americium and curium isotopes, has yet to be expanded (Paper 3.4).

The concern about nuclear weapon proliferation is twofold: the misuse of technology by a non-weapon State or of weapon usable material by subnational groups, for example, terrorists. Any fuel management strategy requiring access to the two major genuine weapons technologies, uranium enrichment and PUREX reprocessing of spent fuels, makes such a strategy vulnerable to misuse. Heavy water reactors (with closed lid and natural uranium fuelling) and partitioning of the transuranic elements by pyrochemical or advanced aqueous processes comply with these non-proliferation objectives. Partitioning and transmutation results in highly radioactive FR fuels, which are difficult to use as nuclear explosives; they lead to nuclear pre-ignition or degradation of the chemical explosives in detonators. Such highly radioactive FR fuels emitting fatal radiation doses would deter terrorists from handling the material (Papers 3.1, 3.2 and 3.5). Thorium fuels have a similar intrinsic proliferation resistance — especially the amount of ^{232}U decaying into ^{208}Tl with a 2.6 MeV gamma ray, which also makes it easy to control any ^{233}U movement (Papers 3.8 and 3.10).

'Protected plutonium production' (P^3) has been proposed to improve the proliferation resistance of plutonium by increasing the fraction of ^{238}Pu through the transmutation of the MAs ^{237}Np and ^{241}Am. The decay heat and the high spontaneous fission neutron rate would cause the quality of the nuclear explosive to deteriorate (Paper 3.10). However, the effect is limited by the achievable ^{238}Pu content. Further studies have to prove the feasibility of the concept. Instead the presence of ^{244}Cm, ^{246}Cm and possibly ^{242}Cm in fuels resulting from a P&T scheme will give a ten times higher neutron background and lead to prompt ignition of a nuclear explosive device as soon as criticality is achieved, which makes a transuranic mix ineffective for use as a nuclear explosive (Paper 3.1).

Progress in nuclear material safeguards will result from current and near-future developments in implementation methods, particularly integrated

safeguards, and from worldwide forums for developing and evaluating future designs of nuclear energy systems, which will deal with proliferation resistance and safeguardability concepts from the earliest design stages (3.11). Such safeguard progress may supplement implementation of intrinsically resistant fuels, in order to reduce fuel cycle costs and radioactive exposure to workers and the public.

In the context of non-proliferation, a fissile material management strategy relying on uranium enrichment or current PUREX reprocessing of spent fuel has an inherent risk of proliferation (Paper 3.1).

Sustainability in resources, in economics and energy supply can be achieved using different fuel cycle options. Though the most economic at present, the expense of the open fuel cycle will rise for fuel deriving from unconventional resources and for geological repositories. Sustainability in uranium resources and consequently in energy supply by thermal fission reactors is assured. Developed and proven reactors are already in existence (Paper 3.1). The operation of the closed fuel cycle with a P&T strategy exceeds the present cost for the once-through strategy but offers savings in lower repository expenditure and future fuel cost. Since the need for uranium is considerably less for the closed fuel cycle, security of uranium supply is easier to achieve for States with limited indigenous uranium resources. The main technological challenges lie in actinide group partitioning from spent FR fuel by advanced processing, in remote MA containing fuel fabrication and in reprocessing for the thorium cycle (Papers 3.1–3.6). For Generation IV FRs the following technological challenges are being addressed (Papers 3.1, 3.5 and 3.7):

(a) Utilization of coolants other than sodium;
(b) Fission product retention capabilities;
(c) Improved core behaviour under severe accident conditions;
(d) Decay heat removal.

In conclusion, the prospects for resources, fuel cycle cost and technological advances will allow for several optional nuclear fuel management strategies beyond 2050. Social criteria, non-proliferation and environmental friendliness — as perceived by a State — as well as safety and economic considerations, will ultimately be decisive in choosing a sustainable strategy for the future (Paper 3.1).

5. CONCLUDING REMARKS

To be sustainable, a fissile material management strategy must meet certain criteria. Firstly, it must draw upon a resource of fissile materials adequate to allow for continued and expanded use for centuries. Secondly, it must embody technical measures and implementation arrangements that will gain and maintain public and political acceptance, including considerations of safety, non-proliferation, environmental protection and economic viability.

REFERENCE TO THE INTRODUCTION

[1] Communication Received from Certain Member States Concerning their Policies Regarding the Management of Plutonium, INFCIRC/549, IAEA, Vienna (1998).

FRONT END FUEL CYCLE STRATEGIES

(Session 1)

Key Issue Paper of Working Group 1

URANIUM DEMAND AND SUPPLY UP TO 2050

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Abstract

The projected growth in nuclear power up to 2050 will have to be accompanied by expansions in uranium production capacity and in conversion and enrichment capacity. Each of these expansions faces challenges. While the total uranium resource base may be adequate to satisfy future demand, confidence in the reliability of uranium resources will have to be increased by additional detailed exploration and development drilling, and by detailed engineering studies. Existing mines will have to be expanded and new mines will have to be developed to replace mine closures that result from resource depletion. Similarly, new conversion and enrichment capacity will have to be built to keep pace with expanded uranium demand. These new expansions, whether they involve new mines or conversion and enrichment facilities, will take place in an uncertain environment of public acceptance for nuclear fuel cycle activities.

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Environmental opposition has delayed development of some new, low cost, production centres, and the environment for approving new mines and new conversion and enrichment plants remains uncertain. For these industry expansions to take place, producers must be confident that prices for their products and services will be adequate to justify the risks inherent in new project development. Even where there is general public acceptance of fuel cycle activities, development of these activities will have to pass rigorous environmental approval procedures that can take between three and ten years or even longer. Therefore, for new facilities, producers and operators will need to build in adequate lead times for granting of permits and for construction, to make sure that the new plants will be available so as to ensure a continued balance between supply and demand.

1. INTRODUCTION

The role that nuclear power will play in satisfying future world energy requirements will depend on a combination of factors, including assurances that there will be adequate uranium resources and fuel cycle services to sustain the nominal growth rate projected for nuclear energy. In 2001, the IAEA published a study entitled 'Analysis of Uranium Supply to 2050' [1], which analysed uranium supply–demand issues up to 2050. The research on which that study was based was completed in 1999. Since that research was completed and the study was published, the dynamics of the industry have continued to change; these changes include:

(a) A tripling of the uranium market price since the 2050 report research was completed. That increase has, however, been partially offset by the weakness of the US dollar relative to the currencies of the major producing countries.
(b) Temporary supply disruptions at key production centres.
(c) Delays in new project developments resulting from a combination of the long depressed market price and environmental activism.
(d) The possible closure of a major production centre as early as 2009.
(e) Uncertainty surrounding the future supply of Russian highly enriched uranium (HEU).

These events are among the concerns that have led experienced industry analysts to recently raise questions about the adequacy of uranium supply and production capacity to meet near- to mid-term demand projections.

Because of the importance of uranium supply to the future of the nuclear fuel cycle, the IAEA has included an evaluation of uranium supply and

demand as part of a broader forum on the nuclear fuel cycle. This study includes a projection of nuclear power up to 2050 and the uranium resources that will be required to support those projections. Uranium resources are, however, only one part of the overall supply question. The capability of the industry to deliver those resources to the marketplace is of equal importance as are assurances that there will be adequate conversion and enrichment capacity to move the material along the supply chain. Accordingly, the capability of the industry to meet increasing annual uranium production, conversion and enrichment requirements is also addressed in this study. Re-evaluation of uranium supply and production capability is a particularly timely issue. In 2005, secondary supply sources, including inventory drawdown and HEU from decommissioning of nuclear weapons and excess government inventories, are expected to supply about 43% of reactor uranium requirements (demand). By 2020, however, the contribution of secondary supply is expected to have declined to 23% of total demand, with a further decline to as low as 10% by 2050, all of which puts increasing pressure on development of new production, conversion and enrichment capacity.

The ability of the industry to satisfy the increasing demand for primary supply is the main theme of this paper. We will examine a range of demand projections, availability of uranium resources and adequacy of production capability to meet these projections, as well as other issues that will have to be addressed to ensure a long term balance between uranium supply and demand.

2. DEMAND

Modelling of the nuclear fuel cycle to project future demand for uranium must take into account a broad range of interrelated technical and political uncertainties. Numbers and types of reactors, load and burnup factors and length of reloading cycles are some of the technical variables that can affect demand for uranium and fuel cycle services. From the political and policy standpoints, public perception of the safety, environmental and economic benefits of nuclear power will influence its growth potential and its requirements for resources and services. Resources and services requirements can also be affected by whether governments elect to utilize closed or open fuel cycles. Projections of nuclear generating capacity and related uranium demand and requirements for fuel cycle services such as conversion and enrichment must, therefore, account for a wide range of variables. Because of the inherent uncertainties in long range forecasting, these variables can be best accommodated by presenting a range of demand projections.

There are a number of organizations that publish projections of reactor uranium requirements. As can be seen in Fig. 1, there is a considerable diversity of opinion regarding the future of nuclear power, which in turn has led to a broad range of projections of future uranium requirements. The divergence in these projections is relatively small in the early years, but it broadens significantly to reflect increasing flexibility to change nuclear policy over time. The consultants that participated in this study selected projections developed by the World Nuclear Association (WNA) [2] and the IAEA [3] as the basic framework to develop a projection of uranium demand up to 2050. Both sets of projections were developed by experts on all aspects of the nuclear fuel cycle, and both provide a range of projections. Figure 2 compares the projection up to 2050 developed for this study with those of the WNA and IAEA, which extend up to 2025 and 2030, respectively. In developing the demand projections for this study, it has been assumed that the current percentages of power produced by light water reactors (LWRs) (87%) and heavy water reactors (HWRs) (6%) will remain unchanged up to 2050, proportionally to each another. Each 1% increase in HWR generated power with a proportionate decrease in LWR generated power would result in a 0.29% decrease in uranium demand and vice versa.

Table 1 compares various parameters for the range of projections on which this study is based. The reference case forecasts an average annual growth rate of 1.6%, while the high case envisions an annual growth of 2.6%.

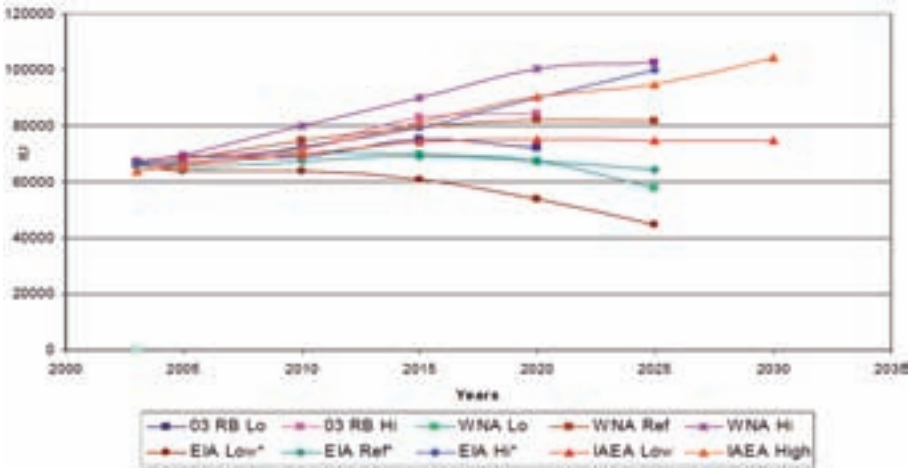

FIG. 1. Published projections (t U) of annual uranium requirements up to 2030.

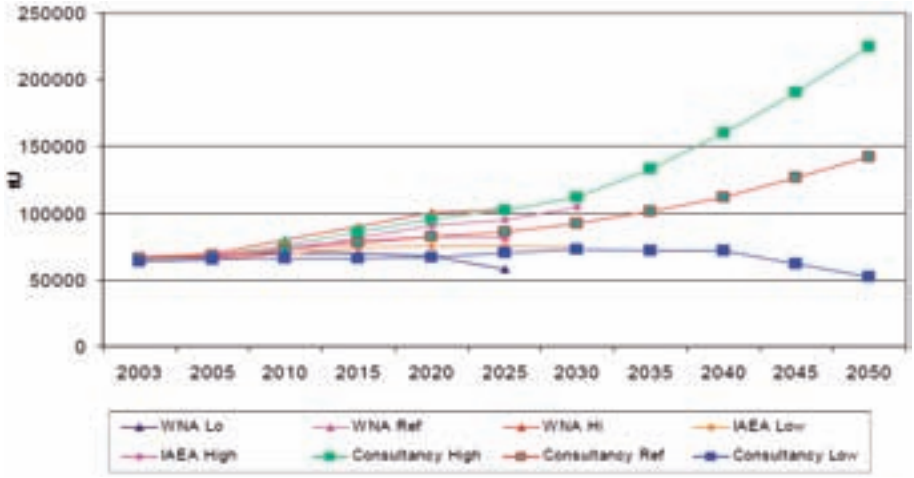

FIG. 2. Consultancy demand projection (t U) up to 2050 compared with WNA and IAEA projections.

The low demand case, which assumes a gradual phase-out of nuclear power by 2100, is relatively flat up to about 2040 before beginning a 3–4% annual decline in uranium demand. The cumulative demand for uranium in the high case exceeds that of the reference case by nearly 30%, while demand in the low case is about 30% less than the reference case. The task facing the uranium industry is underlined by comparing the reference and high demand case projections of annual demand in 2005 and 2050. The ability of the nuclear industry to meet the two- to threefold growth of annual demand in 2050 compared with 2005 will be the subject of the remainder of this paper.

TABLE 1. OVERVIEW OF DEMAND PROJECTIONS

Consultancy projections	Cumulative demand (t U)	Average annual growth rate (%)	Projected demand in 2005 (t U)	Projected demand in 2050 (t U)
Low case	3 178 580	–0.04	67 180	52 000
Reference case	4 449 196	1.6	68 360	142 000
High case	5 716 273	2.6	68 360	225 000

The demand projections in Table 1 (reactor related uranium requirements) will support the following nuclear generating capacities:

(a) Low case — 380 GW(e) in 2002 and 295 GW(e) by 2050;
(b) Reference case — 387 GW(e) in 2002 and 804 GW(e) by 2050;
(c) High case — 387 GW(e) in 2002 and 1274 GW(e) by 2050.

The projections of uranium reactor requirements in this study assume that enrichment tails assays, the percentage of ^{235}U that is not recovered during enrichment, will remain constant up to 2050 at 0.3% for Western enrichment facilities and 0.1% for Russian Federation facilities. Increased uranium prices could, however, lead to lowering Western tails assays, with an accompanying reduction in uranium demand. For example, reducing the tails assays from 0.3% to 0.25% would reduce Western uranium demand by about 10%. However, as a measure of the interrelated nature of activities within the nuclear fuel cycle, reducing the tails assay will result in an increase in enrichment requirements (measured in terms of separative work units (SWUs)) by a percentage approximately comparable to the reduction in uranium demand. The implications of decreasing (or increasing) the tails assays is only one of the variables that are accommodated within the range of demand projections.

3. SUPPLY

3.1. SUPPLY OVERVIEW

Uranium supply in this study is divided into two broad categories: newly mined and processed uranium or primary supply, and secondary supply, which includes downblending of weapons grade HEU, inventory drawdown, mixed oxide fuel (MOX), reprocessed spent nuclear fuel (RepU) and re-enrichment of depleted uranium (tails).

Figure 3 shows the historical relationship between uranium demand and primary supply. Prior to 1990, primary supply exceeded demand, with the balance being held as inventory. After 1990, however, declining worldwide production was no longer adequate to meet demand and secondary supply became increasingly important in ensuring a balance between supply and demand. As shown in Fig. 3, in 1994 primary supply declined to 32 000 tonnes of uranium (t U), or about 55% of 1994 uranium demand. Subsequent increases in annual uranium output were matched by growing demand so that

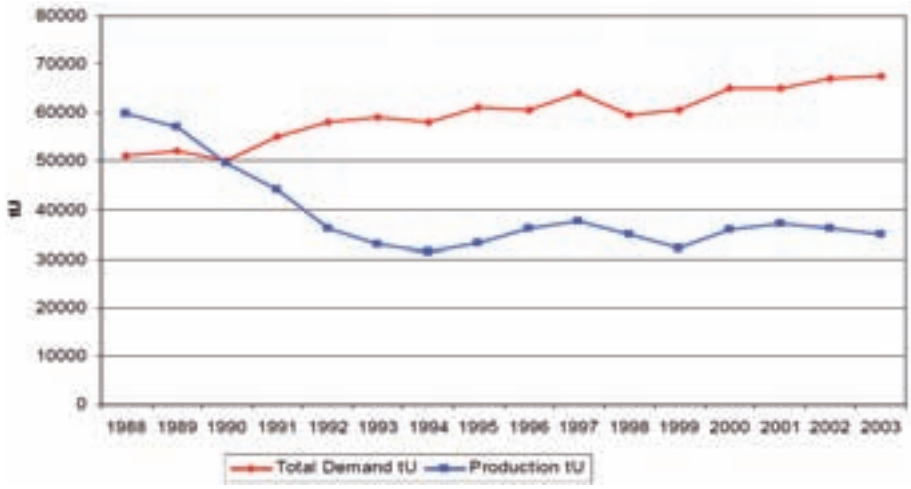

FIG. 3. Historical relationship between worldwide uranium production and total requirements for reactor uranium (demand).

in 2003 uranium demand was satisfied by approximately equal contributions from primary and secondary supplies.

3.2. SECONDARY SUPPLY

The importance of secondary supply during the past decade has led to increasingly reliable estimates of its annual availability and total quantities. Though they are not without uncertainty, these estimates allow us to project the relative contributions that secondary supply will make towards meeting future reactor uranium requirements up to 2050. Figure 4 shows a projection of the role that secondary supply will make in satisfying future demand in the reference case. It is evident from Fig. 4 that availability of secondary supply will decline significantly as a percentage of future annual demand. In 2005, secondary supply is expected to cover about 43% of total demand; by 2020 that total is expected to decline to 23% of demand. Annual demand for uranium is expected to increase steadily up to 2050 in both the reference and high demand cases. By contrast, the availability of secondary supply will continue to decrease so that it could account for as little as 10% of total uranium requirements in 2050 in the reference demand case. At the same time, however, there is also the potential that additional secondary supplies could become available that will reduce the demand for primary supply relative to that depicted in Fig. 4. For

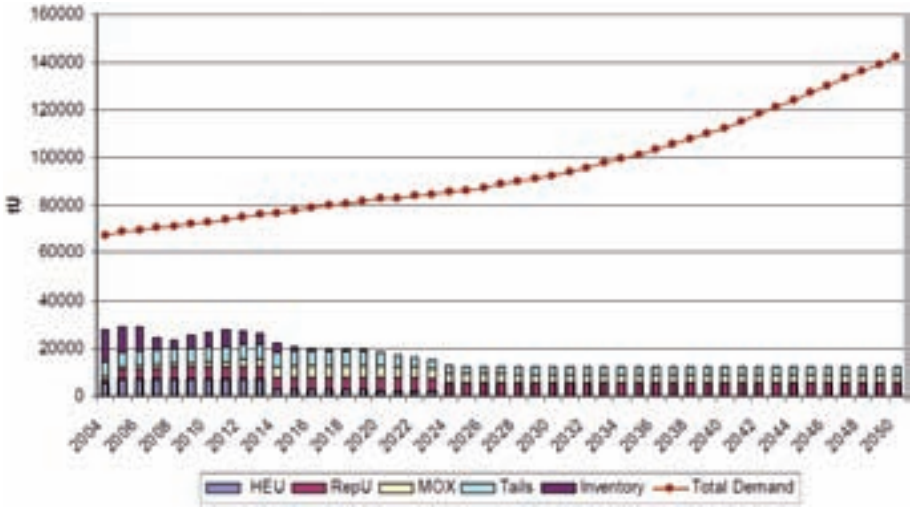

FIG. 4. Projected role of secondary supply in filling demand in the reference case.

example, additional military fissile material could be made available for civilian use. Similarly, concerns about the availability of primary supply could stimulate increased use of recycled material from spent fuel, which would also reduce primary supply requirements.

It is important to emphasize that although secondary supply sources are reasonably well understood their annual availability remains an estimate. There is, for example, a great deal of uncertainty surrounding the future availability of Russian HEU, particularly after 2013 when the US–Russian agreement on HEU is set to expire. The scenario depicted in Fig. 4 assumes that Russian HEU will no longer be available after 2013, which seems an unlikely case. While Russian HEU may not be available to Western markets after 2013, it will probably still be available to help satisfy the domestic requirements of the Russian Federation and those of its fuel cycle customers in Asia and in central and eastern Europe. Increased availability of Russian HEU within its own markets will lessen the need for the Russian Federation to compete for supplies elsewhere.

3.3. PRIMARY SUPPLY

Evaluating the adequacy of the primary supply to meet future reactor uranium requirements is a key objective of this study. Two independent sources have been used to estimate worldwide uranium resources — the 2003

Red Book [4] and information contributed by the consultants who participated in this study. Worldwide uranium resources are ranked in the Red Book according to confidence level and projected production costs within broad cost ranges. Similar conventions have been adopted in this study, including the use of the following resource classifications (listed in order of decreasing confidence):

(a) Reasonably assured resources (RARs)[1];
(b) Inferred resources (formerly EAR-I)[1];
(c) Prognosticated resources (formerly EAR-II);
(d) Speculative resources.

Narrower ranges of production costs than those used in the Red Book are utilized in this study to provide more definitive market price analysis. The cost categories and their ranges (in US $/kg U) used in this study are as follows:

— Low cost: <34;
— Low–medium cost: 34–52;
— Medium–high cost: 52–78;
— High cost: 78–130;
— Very high cost: >130.

The collective experience of the consultants who participated in this study is the basis for the highest confidence resources — termed 'study RARs'. These are resources that the consultants have attributed to uranium deposits known to them, including details of their geology, resources, ore grade, mining method, production cost range and potential development/production schedule. Though a considerable amount of information is available for the study RARs, not all of the projects have been subjected to rigorous feasibility studies and they do not all qualify as 'proven reserves' under standard mining industry nomenclature. Study RARs were compared with the RARs published in the 2003 Red Book on the basis of country by country. If Red Book RARs exceeded study RARs in a given country, the difference was assigned to the next lower confidence category 'incremental RARs'. Information about specific deposits was considered to have sufficient credibility to give study RARs precedence if they exceed Red Book RARs for a given country. The production cost ranges listed in the Red Book were adjusted for incremental RARs to match the cost ranges used in this study. Incremental RARs were assumed to be similar to

[1] RAR and inferred resources are collectively termed 'identified resources'.

study RARs as far as production capacities and development/production timetables.

Information on quantities and production cost ranges for the next two lower confidence categories — inferred resources and prognosticated resources — was taken directly from the 2003 Red Book. Broad assumptions based on knowledge regarding higher confidence resources in each country were used to project production capacities and development schedules for these two resource categories.

3.4. SUPPLY–DEMAND METHODOLOGY

We have now broadly defined the key elements of a uranium supply–demand analysis: annual demand (reactor uranium requirements) and total uranium resources (primary and secondary supplies). These individual factors must then be integrated into a projection of how demand will be satisfied and the market price required to ensure the availability of adequate annual production capacity. Annual reactor uranium requirements have been projected for three demand cases (Fig. 2). Similarly, the annual contributions of secondary supply have been projected up to 2050 (Fig. 4). Annual primary supply requirements are estimated by subtracting annual secondary supply from total annual demand.

The supply–demand model used in this study is based on the assumption that the worldwide uranium production industry is controlled by market based economics. On the basis of a combination of published information and the judgment of the consultants that participated in the study, production centres have been grouped by confidence category (RARs, inferred resources, etc.) and by production cost range. Estimates have also been made as to the first year that each production centre could start operations based on projected permit and development timetables. The supply–demand model is structured such that the lowest cost producer in the highest confidence category will fill the first increment of demand. The remaining demand will be filled by successively higher cost production centres until annual demand is satisfied. No project is allowed to contribute output prior to the practical startup date dictated by permit and development/construction schedules. Sufficient flexibility has been built into the model to accommodate production centres that are not strictly controlled by market based economics, including those in which issues of national security may pre-empt market economics.

One of the main objectives of this study is to evaluate the adequacy of the different confidence levels of uranium resources to meet annual demand. Accordingly, we have begun by analysing the adequacy of the highest

confidence resources to satisfy demand. Successively lower confidence resources are added to meet demand that exceeds the total production capacity of higher confidence resources. Table 2 shows the mechanics of the supply–demand model during a hypothetical nine year period. Production from low cost resources is adequate in this example to satisfy demand up to 2006 after which 400 t U of output from Project F (40% of capacity), a low–medium cost project, will be required to fill the remaining increment of demand in 2007.

Similarly a combination of output from low and low–medium cost projects will be adequate to satisfy requirements up to 2008; production from medium–high cost Project K will be required to balance supply and demand in 2009. In 2013, there is a projected shortfall of 300 t U between demand and available supply. Figure 5 graphically illustrates the gradual addition of production centres required to satisfy demand.

Although the example shown in Table 2 and Fig. 5 is a hypothetical case, it does reflect the reality of the future of the uranium industry. Increasing numbers of projects will be needed to satisfy steadily growing demand. In Fig. 5, five production centres will be able to satisfy demand in 2006, but the combined capacities of 16 projects will not be adequate to meet demand in 2013. Because some deposit types have inherently smaller production capacities, the total number of operating production centres will need to grow steadily over time to satisfy increasing demand. This expansion scenario will have implications, for example for permit agencies, availability of experienced professional and operating personnel and the infrastructure of nearby communities.

The supply–demand model also provides an indirect measure of market price trends. In the example in Table 2, capacity derived from low cost production (<US $34/kg U) will be adequate up to 2006, after which low–medium cost output will have to be added. This suggests that, beginning in 2007, the market price for the scenario described in Fig. 5 will have to increase to between US $34/kg U and US $52/kg U, the range for low–medium cost production centres. The price level within that range will depend on the production cost of the highest cost producer needed to satisfy demand. By 2009 the price will have to rise above US $52/kg U, the floor price for medium–high cost production. The scenario depicted in Fig. 5 is accelerated to show the mechanics of the model, but, over a longer time frame, similar trends can be expected for the industry.

TABLE 2. EXAMPLE OF A SUPPLY–DEMAND MODEL

Year			2005	2006	2007	2008	2009	2010	2011	2012	2013
Primary supply requirements (t U)			9000	9200	9600	12000	13000	13500	14000	14500	16500
Cost category	Project name	Production capacity (t U)									
Low	Project A	5000	5000	5000	5000	5000	5000	5000	5000	5000	5000
Low	Project B	2000	2000	2000	2000	2000	2000	2000	2000	2000	2000
Low	Project C	1000	1000	1000	1000	1000	1000	1000	1000	1000	1000
Low	Project D	800	800	800	800	800	800	800	800	800	800
Low	Project E	400	200	400	400	400	400	400	400	400	400
Low–medium	Project F	1000			400	1000	1000	1000	1000	1000	1000
Low–medium	Project G	800				800	800	800	800	800	800
Low–medium	Project H	400				400	400	400	400	400	400
Low–medium	Project I	400				400	400	400	400	400	400
Low–medium	Project J	400				200	400	400	400	400	400
Medium–high	Project K	800					800	800	800	800	800
Medium–high	Project L	1000						500	1000	1000	1000
Medium–high	Project M	400								400	400
Medium–high	Project N	400								100	400
High	Project O	1000									1000
High	Project P	400									400
Total production			9000	9200	9600	12000	13000	13500	14000	14500	16200

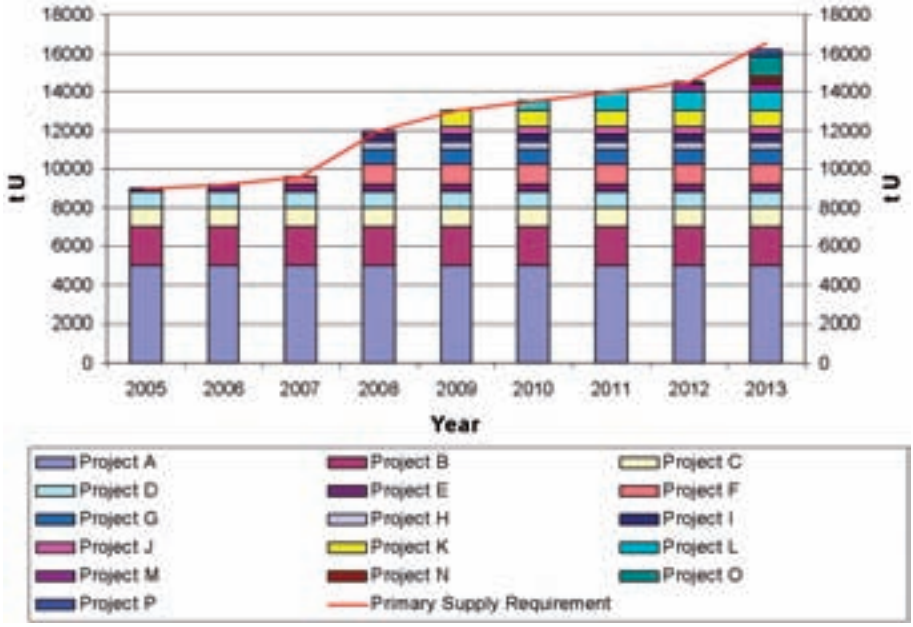

FIG. 5. Example of a supply–demand model.

3.5. SUPPLY–DEMAND ANALYSIS

Using the methodology just described the adequacy of uranium resources and production capacities based on the different confidence levels of resources are measured against the three demand scenarios. Figure 6 depicts the adequacy of production derived from study RARs, the highest confidence resource category, to satisfy primary supply requirements for the reference demand case. As indicated in Fig. 6, production from low and low–medium cost production derived from study RARs (i.e. <US $52/kg U) will, if all projects are brought on-line as soon as technically feasible, be adequate to satisfy production up to 2022, after which output from higher cost production centres will be required.

A pertinent question is why a study of this nature predicts a current market price considerably lower than that the actual market price today. Essentially, an exercise of this nature produces gradual shifts in both supply and demand. However, in reality it is possible for both to change rapidly, and potentially in opposite directions.

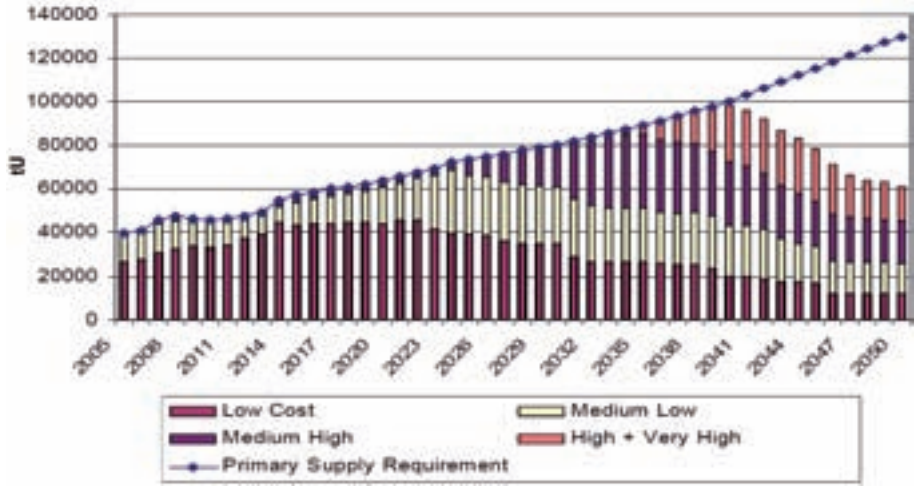

FIG. 6. Relative contributions by cost category of study RARs available to fill the reference demand case.

A number of key events have occurred in the past two years which have reduced available low cost supplies while increasing demand. Some of these events include:

(a) Depreciation of the US dollar versus the currencies of many uranium producing countries;
(b) Production disruptions at the McArthur River and Olympic Dam mines resulting in reduced supply and additional market demand;
(c) A reduction of available inventories, resulting in less flexible contract supply terms and the need for new production sources at prices that must reflect a sufficient return on applied capital;
(d) The potential closure of the Rössing mine as early as 2009 due to increased projected future costs in US dollars;
(e) Further producer consolidation (previous Rio Tinto acquisition of Energy Resources of Australia Ltd (ERA));
(f) The HEU feed material dispute between TENEX and Global Nuclear Services and Supply Ltd (GNSS).

The real or perceived loss of a number of supply sources resulted in a re-evaluation of strategic inventory levels, leading to an increase in market demand at a time when uranium was not readily available at existing market prices.

Higher market prices, however, stimulate or accelerate additional uranium supply from new or existing low cost sources. The sources incorporated into Fig. 6 include:

(a) Cigar Lake production commencing in 2007 and reaching over 6000 t U by 2010;
(b) Kazakhstan in situ leach (ISL) production reaching 6200 t U in 2010 and at least 10 000 t U by 2014;
(c) Expanded Olympic Dam production, reaching as much as 11 500 t U by 2015;
(d) Assuming a lifting of various mining restrictions in Australia, resulting in production from up to four new sources by 2015;
(e) Additional planned supply from countries of the former Union of Soviet Socialist Republics, referred to as the Commonwealth of Independent States (CIS), in addition to Kazakhstan, totalling up to 4800 t U per annum by 2016;
(f) A doubling of US ISL production to 1500 t U by 2010 and as much as 3600 t U by 2015.

These sources have the potential to displace higher cost sources in order to then reduce supply costs below US $52/kg U for all market price sensitive production.

The primary supply response to higher prices may take a number of years to fully eventuate as new mines are committed for construction. Once there is a market consensus that future assured supplies are adequate, price relief may occur quickly to return the market price to model equilibrium. However, until the market is convinced that new supply sources will be adequate to satisfy demand, the market price will probably remain at or near current levels in order to ensure producers an adequate return on investment that will stimulate development of new supplies.

It is important to note that the model relies on the assumption that projects will come on-line within the time frame dictated by permit and construction schedules when needed to satisfy demand. Disruptions in obtaining permission and starting construction have the potential to create supply shortfalls, which underline the need for careful planning within the supply industry.

As noted in Fig. 6, study RAR based production in all cost categories will be adequate to satisfy the reference demand case up to 2035. Starting in 2036 the annual supply shortfall based on production derived from study RARs will increase annually, reaching a cumulative total of 686 500 t U by 2050. The addition of incremental RARs, the next lower confidence level, extends the

TABLE 3. COMPARISON OF REFERENCE AND HIGH DEMAND CASES SUPPLY–DEMAND PARAMETERS BASED ON TOTAL RARs (STUDY RARs + INCREMENTAL RARs)

Demand case	First year of deficit between supply and demand	Cumulative production (t U)	Cumulative deficit between supply and demand (t U)	First year high cost production needed
Reference	2038	3 144 500	520 600	2033
High	2032	3 472 300	1 461 800	2028

adequacy of total RARs by two years up to 2037 and reduces the cumulative shortfall up to 2050 from 686 500 to 520 600 t U for the reference demand case. Table 3 compares several parameters that relate total RAR based production (study RARs + inferred RARs) to the reference and high demand cases. Study RARs are adequate to satisfy the low demand case so there is no need to add incremental RARs to the low demand case supply–demand analysis.

The high demand case results in slightly more efficient utilization of resources, which explains the increase in cumulative production. The cumulative deficit between supply and demand in the high demand case is, however, nearly three times that of the reference case. It is evident from Table 3 that total RARs are not adequate to fill either the reference or high demand cases and that lower confidence resources will be needed to ensure a balance between supply and demand. Table 4 shows the net effect of the addition of inferred resources to overall supply. Figure 7 shows these relationships graphically.

TABLE 4. COMPARISON OF REFERENCE AND HIGH DEMAND CASES SUPPLY–DEMAND PARAMETERS BASED ON IDENTIFIED RESOURCES (TOTAL RARs + INFERRED RESOURCES)

Demand case	First year of deficit between supply and demand	Cumulative production (t U)	Cumulative deficit (t U)	First year high cost production needed
Reference	2050	3 365 730	6 450	2045
High	2042	4 360 600	573 600	2038

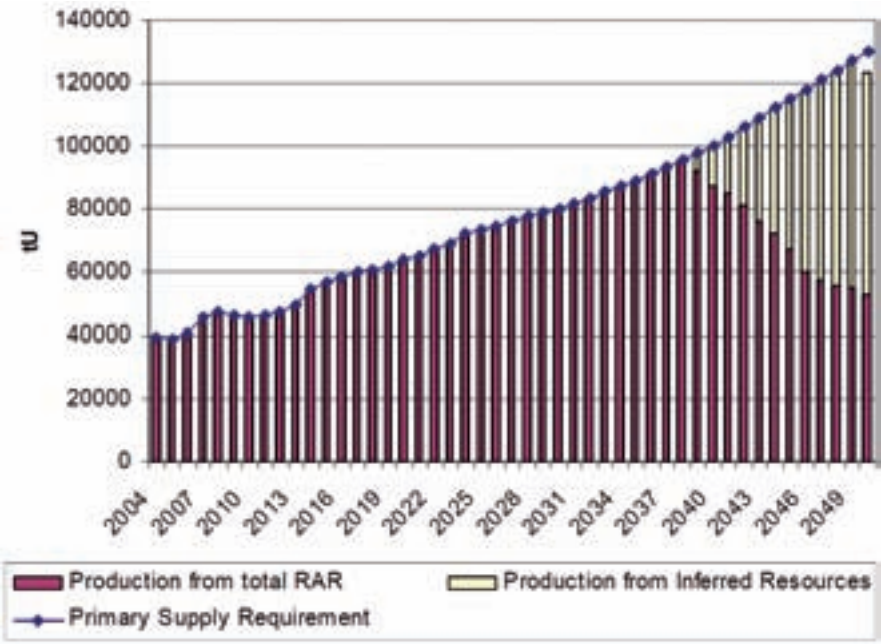

FIG. 7. Relative contributions of RARs and inferred resources to the reference demand case.

As noted in Table 4 and in Fig. 7, production derived from identified resources is projected to be adequate to satisfy demand up to 2049 in the reference case, while cumulative production falls short of satisfying demand by only 6450 t U. To put this deficit into perspective, annual production in 2050 based on output derived from indicated resources is projected to be about 123 600 t U, so that the cumulative deficit is projected to equal about 5% of a single year's production. This is well within the limits of accuracy of the model and, for practical purposes, identified resources are adequate to satisfy the reference demand case. High cost resources will not be required to balance demand until 2045 compared with 2033 when output is limited to production derived from total RARs (Table 3).

In the high demand case, however, cumulative supply derived from identified resources falls short of demand by 573 600 t U. Production based on identified resources in the high demand case will fall short of satisfying demand starting in 2042. Therefore, the next lower confidence level of resources will have to be developed to ensure a balance between supply and demand in the high demand case. Figure 8 shows the relative contributions of all resource cost categories, including indicated and prognosticated resources, which are

projected to be available to satisfy the high demand case. As shown in Fig. 8, with the addition of prognosticated resources, annual output will be adequate to satisfy demand up to 2045, with annual shortfalls ranging from about 10 000 t U in 2046 to 66 000 t U in 2050. The cumulative deficit is reduced from 573 600 t U if production is limited to identified resources to 162 000 t U with the inclusion of prognosticated resources, which is the equivalent of just over one year of production at the level of output forecast for 2050 in the high demand case. In other words, the addition of prognosticated resources essentially balances supply and demand in the high demand case within the limits of accuracy of the model.

This scenario assumes that everything will proceed as projected. We need to remember, however, that events are being projected 45 years into the future, and the probability that everything will take place exactly as planned is minimal. Even with the addition of the prognosticated resources, there will still be a gap between supply and demand, starting in 2045 in the high demand case. Therefore, this gap needs to be kept in focus and prepared for. There is one final confidence category of resources, speculative resources, to be called upon to fill the projected shortfall in the final years of the study. Speculative resources recoverable at less than US $130/kg U are estimated to total 4 440 000 t U [4]; there are an additional 3 100 000 t U of speculative resources for which no cost range has been assigned. As is the case with prognosticated resources, speculative resources have not yet been discovered. They are based on ‘indirect evidence and extrapolations’. As their name implies, they are

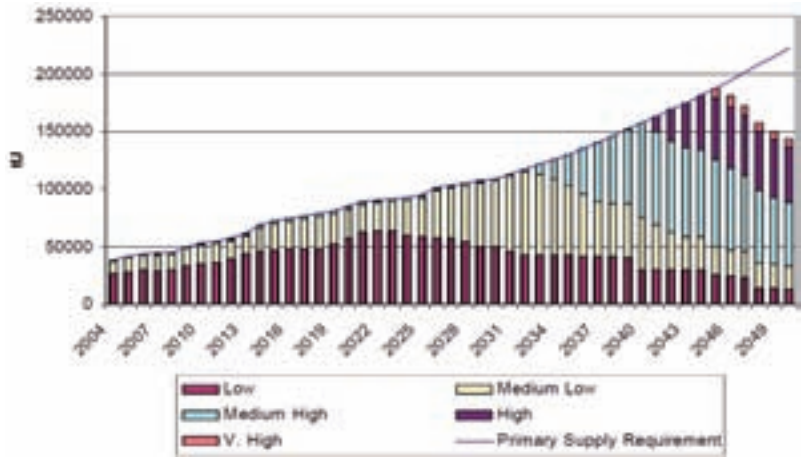

FIG. 8. Resource cost categories available to balance supply and demand: the high demand case based on identified and prognosticated resources.

speculative. They provide a cushion that we can look to for the future, but it needs to be remembered that a large percentage of speculative resources have not had the benefit of a single hole being drilled: they are truly speculative.

In each of the supply–demand scenarios that were evaluated, there were resources that were not utilized by the end of 2050. Unutilized resources resulted either because the capacity of individual production centres was not adequate to exhaust their resources within the study period or because many of the higher cost projects will not be needed to satisfy demand until late in the study period so that there will not be sufficient time to exhaust their resources by 2050. The Olympic Dam mine is a good example of unutilized resources. Because of its large resource base (nearly 1 200 000 t U) even by doubling or tripling its production capacity, Olympic Dam's resources will not be depleted by 2050. Table 5 summarizes unutilized resources for each of the demand scenarios and combinations of resources. These totals are based on relatively high confidence estimates of production capacities for deposits in the RAR category of the study, but on more speculative estimates in the case of lower confidence resources. Increases in production capacity will result in more complete utilization of resources to satisfy demand. Conversely, unanticipated constraints on production capacity will result in higher unutilized resource totals and greater deficits between supply and demand. Section 4 provides more details on the importance of individual and cumulative production capacities.

On the basis of the foregoing discussion, the following conclusions can be drawn regarding the adequacy of uranium resources to meet demand:

(a) Study RARs are adequate to satisfy the low demand case.
(b) Identified resources are adequate to satisfy the reference demand case.
(c) Identified + prognosticated resources are adequate to satisfy the high demand case.

TABLE 5. UNUTILIZED RESOURCES (t U) FOR THE THREE DEMAND CASES

Demand case	Study RARs	Total RARs	Identified resources	Prognosticated resources
Low	1 563 700	*		
Reference	997 300	1 557 300	2 534 200	*
High		1 343 700	1 829 900	3 697 000

* Resources of higher confidence categories adequate to balance supply and demand.

These conclusions, however, represent only the beginning of the analysis. They portray the relationship between supply and demand based on a number of assumptions, which, if varied, can significantly change these conclusions. In addition, they do not address many of the complexities of the uranium supply industry.

4. PRODUCTION CAPACITY

Uranium resources are only one part of the supply side of the supply–demand equation. As the recent price increase has proven, perceptions regarding the ability of the industry to develop and deliver resources to the marketplace, as measured by annual production capacity (t U/a), can be more important than resources in the ground. As shown in the simplified example in Fig. 5, as demand increases so too does the number of production centres required to fill that demand. The rate at which the number of new production centres will have to increase depends on the rate of expansion of demand and the production capacities of the next highest cost producers needed to fill that demand. As previously noted, increases or decreases in production capacities compared with the estimates used in this study will have a direct bearing on annual and overall availability of resources from individual production centres.

In 2003, 40 production centres in 16 countries accounted for worldwide output totalling 35 385 t U. The production centres operating in 2003 ranged from the McArthur River mine with a capacity of 7200 t U to small operations in developing countries that produced between 40 and 200 t U. Except for study RARs, we are not able to assign production capacities to individual production centres. Instead, we have extrapolated the geology and deposit sizes of the study RARs in a given country to lower confidence resources in the same country. There is, however, no reliable way to determine the actual number of deposits that host the lower confidence resources. As we look to the future, we can say with relative confidence that there are no known deposits, which are not currently developed, that will support a very high capacity operation except those at Cigar Lake, which will nearly equal McArthur River's capacity when it reaches nameplate capacity in about 2010. There are plans at Olympic Dam mine in Australia to at least double its current 3880 t U capacity, but the timing of that expansion is uncertain.

Production centres such as McArthur River, Cigar Lake and an expanded Olympic Dam are truly exceptions with respect to annual capacity. No other projects are currently planned with comparable capacities. In fact, there are relatively few deposits that even have projected production capacities ranging

between 2000 and 3000 t U that are likely to be developed in the near-term. Instead, because of their smaller initial capital costs and shorter development time frames, except for the Cigar Lake mine, the most likely candidates for near-term development are ISL projects, which typically have a capacity potential of between 400 and 1200 t U. Accordingly, it could take from five to as many as 15 ISL projects to equal the output of the McArthur River mine. Annual primary production requirements in the reference case are projected to expand by about 60% between 2005 and 2020. Under this scenario, by 2020 the number of production centres could increase from the current level of 40 to between 70 and 100 mines as demand for primary production increases, and that number will continue to grow up to 2050. The exact number of production centres will depend on the types of deposits and extraction methods that will be developed in the future.

The growth in the uranium production industry will require a corresponding expansion in the numbers of experienced miners, engineers and geologists needed to implement the expansion of the industry. Other natural resources industries, including oil and gas, base and precious metals, and coal, are also facing shortages of experienced technical personnel. For the uranium industry, ensuring an adequate workforce will have to start by reversing the current shortage of experienced technical personnel and the fact that a large percentage of the most experienced personnel are approaching retirement age. The industry needs to initiate and support programmes in universities and technical schools that will ensure the availability of the miners, engineers, geologists and environmental scientists who will be needed to implement the expansion of the industry in the future.

5. DEVELOPMENT AND PRODUCTION SCHEDULES

Every effort has been made to use published information to project the availability of currently undeveloped projects. Production startup dates for projects that have not published development plans have been based on estimated approval, development and construction schedules. Startup schedules can be projected with reasonable accuracy for study RARs for which information on geology, mining method and production capacity is available. Such projections, however, become increasingly speculative for lower confidence resources that cannot be attributed to specific deposits. There is even uncertainty about when the undeveloped deposits in the study RAR category could begin production. For example, companies that control large resource bases which occur in several deposits may have to delay even low cost

projects as they prioritize projects to accommodate the availability of capital and experienced personnel.

The time required for project permission and licensing has increasingly become a key component of the time frame leading ultimately to the start of production, with the time required to approve a project ranging between two and ten or more years. Permit requirements are different in different jurisdictions. Regulatory agencies operate differently in different countries and even between states or provinces in the same country. Permit schedules can be affected by agency staffing levels, the size and complexity of a project, public awareness and attitudes towards uranium mining as well as proximity to towns and cities to name a few of the variables. Among these variables, the public attitude towards uranium mining deserves special attention.

6. OPPOSITION TO URANIUM MINING

The history of the uranium industry is full of well documented and anecdotal examples of delays in project development because of public opposition to uranium mining. In some cases the opposition is based on specific concerns, while in other cases it is based on a broad anti-nuclear philosophy. This issue is of concern to the uranium industry because of its potential to disrupt development and to jeopardize availability of supply. Whether a project can be developed and, if so, when can become more of a political decision than an economic one. While there are many examples of delayed development, a few are of particular importance because of the magnitude of the resources affected.

6.1. AUSTRALIA

Australia has had a history of opposition to uranium production. In 1983, the Australian Labor Party (ALP) implemented its 'three mines policy', which limited production to only three mines — Nabarlek, Ranger and Olympic Dam. When Nabarlek ceased operations in 1988, the policy effectively became a 'no new mines policy', as the ALP refused to issue export licences to new mines, thus eliminating any incentive to develop new projects. The ALP was replaced by a pro-business coalition Government in 1996, which rescinded the no new mines policy, placing development decisions back into the realm of marketplace economics. This decision at the Commonwealth level did not,

however, end opposition to uranium mining. Instead, it moved back to the local and state levels, where it had always had its roots.

At the state level, there is diversity of policy as far as uranium mining is concerned. South Australia, which hosts the Olympic Dam and Beverley mines, promotes uranium exploration, development and mining. At the other end of the spectrum, the anti-uranium mining ALP Government won re-election in Western Australia in 2005, and has vowed to continue its opposition to uranium production. This policy has had an impact on three projects — Kintyre, Yeelirrie and Manyingee — with combined resources and production capacities of 80 000 t U and 3700 t U, respectively. There are other resources that could be affected by the ALP policies, which also have a negative impact on uranium exploration in Western Australia.

At the level of local opposition perhaps no other project has received more publicity than Jabiluka in the Northern Territory. Resources and production capacity at Jabiluka total 132 000 t U and 2290 t U, respectively. Jabiluka, which was discovered in 1971, has faced a series of development delays, brought on by a combination of market forces and opposition from Aboriginal owners, with no real end to the delays in sight. The Aboriginal owners of the project site currently hold a veto over development at Jabiluka. Local and state opposition to mining at Kongarra also exists. Likewise, all uranium mining in Queensland is opposed by the state.

6.2. NEW MEXICO, UNITED STATES OF AMERICA

Western Australia is by no means alone in its opposition to uranium mining. The state of New Mexico, which was once the leading US uranium producing state, now has a well organized and well financed anti-uranium mining coalition, which includes environmental advocates and Native American tribes. This coalition has effectively blocked development of the Church Rock and Crownpoint ISL projects, despite expenditure of approximately 20 million US dollars in permit and licensing activities by the owner of the projects. Between them, these projects have resources totalling 40 300 t U and production capacities of 1400 t U. The Navajo tribe, which controls land in the Crownpoint and Church Rock areas, has passed legislation that would effectively place a 20 year moratorium on uranium mining on lands that it controls. There is some uncertainty as to whether this moratorium will apply to lands in the Crownpoint area, but if it does it could effectively ban development of at least part of the Crownpoint resources until 2025.

6.3. OPPOSITION TO URANIUM MINING AND EXPLORATION EXPENDITURES

Opposition to uranium mining hangs like the sword of Damocles over the industry, not only in Western Australia and New Mexico, but also in other parts of the world. Figure 9 compares low cost uranium resources (identified resources recoverable at less than US $40/kg U) and 2002 exploration expenditures in Australia with those of Canada, which ranks third in low cost identified resources. Australia, with 38% of worldwide low cost identified resources attracted only 3% of worldwide exploration capital in 2002. Canada, on the other hand, with its potential for discovery of large, high grade, unconformity related deposits and less opposition to uranium mining attracted 24% of 2002 exploration expenditures compared with 15% of low cost identified resources.

Australia has potential for discovery of both unconformity related deposits (e.g. Jabiluka and Ranger) and deposits amenable to ISL (Beverley and Manyingee). We should not necessarily expect an exact one-to-one relationship between resources and exploration expenditures, but the question should be posed of why the world's leader in low cost resources has attracted so little in the way of recent exploration expenditures. The likely answer is that the risk is too high that discovery of low cost resources does not guarantee

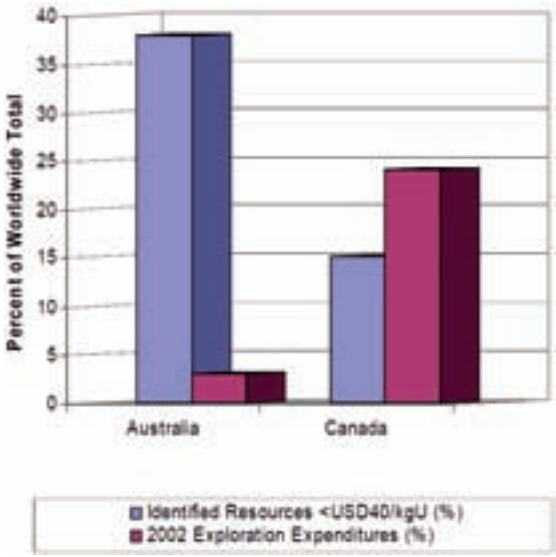

FIG. 9. Comparison of low cost resources and exploration expenditures.

timely development of those resources. The same situation holds true in some parts of the United States of America (USA). Well organized opposition to uranium mining has virtually eliminated exploration in some resource-rich states, including New Mexico. Whether or not there is a formal moratorium on uranium mining may not matter. Uncertainty can create a de facto moratorium, as no company will risk exploration capital where a discovery may not be rewarded by the opportunity to develop a mine.

6.4. OPPOSITION TO MINING: A WORLDWIDE CONCERN

We can point to current examples where opposition to uranium mining has stymied development and discouraged exploration. There may, however, be an even broader concern related to the asymmetry between uranium demand and production worldwide. Figure 10 compares uranium requirements related to reactors in 2003 with production in the same year for the four countries with the highest uranium demand. These four countries accounted for more than 60% of demand but collectively contributed only about 12% of 2003 output. For comparison purposes, Table 6 shows the percentage of worldwide uranium output contributed by the four leading uranium producing countries in 2003. A comparison of Fig. 10 and Table 6 shows that the leading producers of uranium are not the leading users.

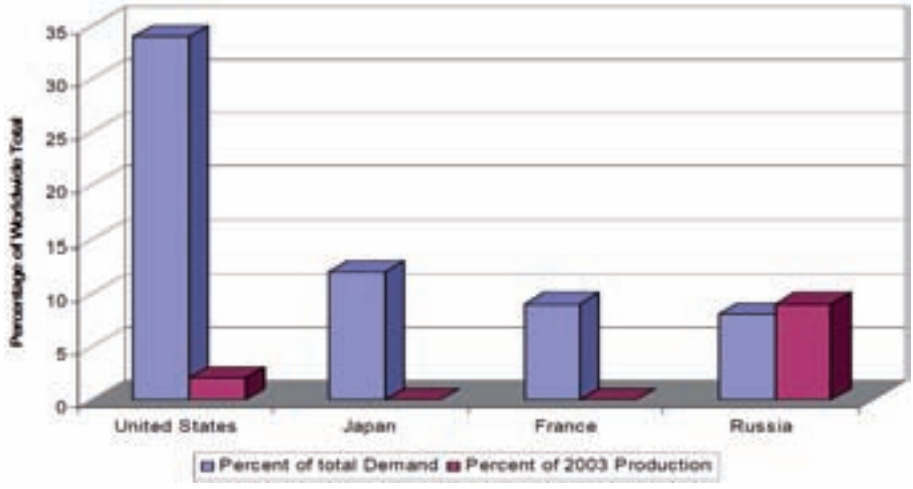

FIG. 10. Comparison of percentage of 2003 uranium reactor requirements and production.

TABLE 6. LEADING URANIUM PRODUCING COUNTRIES IN 2003

Producing country	Percentage of 2003 production
Canada	30
Australia	22
Niger	9
Kazakhstan	8

Does this asymmetry between uranium producing and consuming countries potentially pose a future threat to the industries of some of the major producing countries? Australia, Kazakhstan and Niger do not currently have civilian nuclear power industries, so all of their uranium production is available for export. Uranium is an important source of hard currency for Kazakhstan, so its industry is at present insulated from broad philosophical concerns related to uranium production. Similarly, with higher uranium prices, revenue from uranium exports is again important to Niger's economy, and its uranium operations are relatively insulated from organized opposition to uranium mining. Can it be guaranteed, however, that the same holds true for Australia, where the anti-uranium ALP is waiting for a return to national power and possible reimposition of the no new mines policy? Could the time come when the Australian people are no longer willing to be exposed to the risks (perceived or real) of producing uranium to satisfy demand in countries with highly developed nuclear power programmes but with limited resources or uranium production industries?

These are broad philosophical questions, but in a study that projects uranium supply and demand up to 2050, they cannot be entirely ignored, particularly when the future prospects of large, low cost, resources and production capacities are finely balanced. There is no evidence that Australia or any other country is moving towards a formal national ban on uranium production. Nevertheless, a globally well diversified uranium production industry is critical for offsetting such risks. At the same time, the distribution of low cost uranium resources is an accident of geology. Therefore, loss of availability of even a small percentage of low cost production capacity would require development of higher cost resources, with an accompanying impact on market price.

7. SENSITIVITY ANALYSES

The supply–demand model used in this analysis and described in Table 2 and Fig. 5 is a tool for approximating how the balance between uranium supply and demand will be maintained up to 2050. By changing the model input we can assess the adequacy of different confidence levels of resources to satisfy a range of demand projections. We can project when production from projects of a given production cost range will be required to balance growing demand as a means of projecting market price trends. It is, however, important to remember that we are dealing with a model and that the projections based on the model are only as good as the input to the model.

Nearly all of the model input is based on estimates and assumptions, the accuracy of which decreases the further into the future we consider. The present uranium industry can be modelled with a high degree of accuracy using data reported to organizations such as the IAEA, WNA, Organisation for Economic Co-operation and Development and various government agencies. Predicting how the industry will look 20, 30 and 40 years from the present, however, becomes increasingly speculative. To partially accommodate these uncertainties we have used three demand cases against which to measure the adequacy of uranium supply. In addition, we can use sensitivity analyses to determine how changes in different input parameters will affect the adequacy of supply or market price projections. The following sections provide examples of how changes in certain parameters could affect the future balance between supply and demand.

7.1. AVAILABILITY OF RUSSIAN HIGHLY ENRICHED URANIUM

The reference demand case for primary supply assumes that the availability of Russian HEU under the current US–Russian HEU agreement will end when the agreement expires in 2013. It has already been noted that while the availability of Russian HEU to Western markets may indeed end in 2013, a portion of that material will probably still be available to satisfy Russian markets and obligations to its fuel cycle customers. This likelihood has been reflected in the secondary supply projections. The potential that the United States Department of Energy will also make additional HEU available to the market has also been provided for in these projections.

7.2. CHANGING THE ENRICHMENT TAILS ASSAY

It has already been noted that lowering the enrichment tails assay reduces the demand for natural uranium. To evaluate the impact of such a reduction on the balance between supply and demand we assumed that the average Western enrichment tails assay will be reduced from 0.30% to 0.25% starting in 2008 and that this reduction will result in a 10% reduction in Western uranium demand. The WNA [2] estimates that annual Western uranium requirements account for an average of about 90% of worldwide primary supply requirements up to 2025. Table 7 shows the effect of reducing Western primary supply requirements by 10% annually between 2008 and 2050 as a result of reducing the enrichment tails assay. As shown in Table 7, a reduction in the tails assay from 0.3 to 0.25% would result in a balance between supply and demand in the reference case with production derived from total RARs. The cumulative deficit, 14 830 t U, is well within the limits of accuracy of the model. Although tails assay reduction has already been implemented by some Western utilities, there is no assurance that uranium prices will justify the reduction in average tails assays over the long term.

We cannot evaluate the effect of lowering (or increasing) enrichment tails assays as an isolated strategy. Instead, the implications of such a reduction relative to other fuel cycle activities must also be considered. Just as there are questions about the adequacy of uranium resources and production capacity, so too are there similar questions regarding availability of conversion and enrichment services. Therefore, the broader implications of changes in one stage of the fuel cycle for other related activities need to be examined.

TABLE 7. REFERENCE CASE TOTAL RARs WITH AND WITHOUT REDUCTION IN ENRICHMENT TAILS ASSAY

Tails assay scenario	First year of deficit between supply and demand	Cumulative deficit (t U)
Without tails assay reduction	2038	520 060
With tails assay reduction	2048	14 830

8. CONVERSION

All reactors that use enriched uranium, which is the majority of reactors worldwide, require conversion of natural uranium concentrate (U_3O_8) to gaseous uranium hexafluoride (UF_6). Many of the same factors that affect uranium demand — capacity factors, tails assays optimization, etc. — also affect UF_6 conversion requirements. Therefore, demand for UF_6 conversion closely parallels primary uranium supply requirements. Similarly, many of the same secondary supply sources that displace primary uranium supply — HEU, inventory drawdown, RepU, tails re-enrichment and optimization of tails assays — also displace UF_6 conversion requirements.

Heavy water reactors, mainly those of the CANDU design, do not use enriched uranium. Instead UO_2 produced from the direct conversion of U_3O_8 is used in the fabrication of fuel for HWRs.

8.1. UF_6 CONVERSION REQUIREMENTS

When averaged over the period to 2050, UF_6 conversion requirements will approximately equal uranium requirements, with a downward adjustment of about 7% for reactors that do not use enriched uranium. On the basis of this relationship, Table 8 lists current worldwide UF_6 conversion capacity.

TABLE 8. WORLDWIDE UF_6 CONVERSION CAPACITY
(source: WNA [2])

Country	Facility/owner	Capacity[a] (t U/a)
Brazil		90
Canada	Port Hope/Cameco	12 500
China	CNNC[b]	1 000
France	COMURHEX/AREVA	14 000
Russian Federation	Rosatom	15 000
United Kingdom (UK)	BNFL	6 000
USA	Metropolis/ConverDyn	14 000
Total		62 590

[a] Nameplate capacities. There is no assurance that these capacities can be achieved or maintained throughout the study period.
[b] CNNC: Chinese National Nuclear Corporation.

In addition to the conversion capacities listed in Table 8, the WNA [2] estimates that the Western inventories of UF_6 total approximately 50 000 t U. The Russian Federation also holds inventories of UF_6, which contain uranium in various forms. The Russian UF_6 inventory is thought to be about half that held in the West, but the exact quantity is uncertain.

Cameco and BNFL recently announced a ten year toll conversion agreement, ensuring that the Springfields conversion facility (Lancashire, UK) will remain in operation to at least 2016. Cameco is currently studying conversion options after this.

AREVA is currently envisaging the building of new conversion facilities at its existing sites with an operating capacity in the range of 15 000–20 000 t/UF_6, which will progressively replace the old plants from 2010 to 2020.

ConverDyn has also announced its intention to increase its capacity when market conditions justify such an increase.

In the Russian Federation, Rosatom has also indicated its intention to increase its concentrates purification capacity when market conditions are judged favourable.

With good planning and proper economic incentives, future UF_6 conversion capacity will be adequate to meet demand. This may be accomplished by expansion of existing facilities or construction of new plants. Whereas a uranium mine must be built where the orebody is located, a new conversion facility can be constructed in many favourable jurisdictions regardless of geography. Some of the major uranium producers are also participants in the conversion and enrichment industries. This vertical integration of the industry will help to ensure a future balance between uranium production and the downstream services required to balance supply and demand at all stages of the front end of the nuclear fuel cycle.

8.2. UO_2 CONVERSION

Table 9 lists worldwide UO_2 conversion capacity. Demand for UO_2 conversion in the reference case is expected to increase from about 2870 t U in 2005 to 5480 and 9730 t U in 2025 and 2050, respectively, assuming that HWRs continue to account for about 7% of total reactor uranium requirements. Although there will be an apparent shortfall of UO_2 conversion capacity by 2025, capacity increases at existing facilities are expected to keep pace with increases in UO_2 demand throughout the study period. Cameco is the sole market price based supplier of UO_2 for HWRs and maintains additional standby production facilities at Port Hope, Ontario.

TABLE 9. UO_2 CONVERSION CAPACITY
(source: WNA [2])

Country	Capacity (t U/a)
Argentina	150
Canada	2800
India	250
Romania	100
Total	3300

Cameco is currently developing a new fuel called slightly enriched uranium for use in the Bruce Power HWRs. This fuel is approximately 1.1% ^{235}U. Depending upon successful introduction of this fuel at Bruce Power, and perhaps other HWRs, UO_2 conversion requirements will marginally decrease, with a consequent slight increase in demand for UF_6 conversion and enrichment.

9. ENRICHMENT CAPACITY

As noted in the previous section, most of the world's reactors use enriched uranium, which involves the use of either gaseous diffusion or gaseous centrifuge separation to enrich the concentration of ^{235}U from its naturally occurring level of 0.71% to between 3 and 5%, to produce low enriched uranium (LEU) for use in nuclear fuel. Both processes, which use gaseous UF_6 as their starting feed, are measured in terms of SWUs. An SWU measures the quantity of separative work performed to enrich a given amount of uranium by a certain amount[2]. Separative work units indirectly measure the energy used in enrichment. Enrichment processes transform natural uranium into two products, enriched uranium with a higher percentage of ^{235}U than naturally occurring uranium and depleted uranium (tails) with a lower ^{235}U content.

Many of the factors that control uranium demand — secondary supply, enrichment tails assays, load factors, cycle lengths, etc. — also affect the

[2] Example: One kilogram of uranium enriched to 3% ^{235}U requires 3.8 SWU if the plant is operated at a tails assay of 0.25%, or 5 SWU if the tails assay is 0.15%, thereby requiring 5.1 kg instead of 6.0 kg of natural feed.

demand for enrichment services, although the relationships may work differently. For example, lower enrichment tails assays result in lower uranium demand; conversely, they result in higher enrichment requirements.

Table 10 lists the current suppliers of uranium enrichment by location and enrichment process. The relationship between uranium demand and enrichment requirement is not linear, and it is beyond the scope of this paper to undertake a rigorous assessment of the enrichment capacity that will be required to ensure a balance between supply and demand up to 2050. We can, however, report on nearer term issues in the enrichment business and make projections for these issues into the future.

The WNA [2] has projected that enrichment requirements in its reference demand case will increase from about 41 million SWU in 2005 to 51.5 million SWU by 2025. An even greater increase to 65 million SWU is projected for the WNA high demand case. Therefore, the current enrichment capacity of nearly 51 million SWU is adequate to satisfy the WNA reference demand case up to 2025, which is within 5% of the reference case developed for this paper (Fig. 2) for the same period. Current capacity is, however, projected to fall short of satisfying the high demand case by nearly 14 million SWU by 2025. There are reasons to be more concerned about enrichment supplies that are not readily apparent from a simple comparison of current enrichment capacity and future demand. The ageing diffusion plants in France and the USA need to be

TABLE 10. WORLDWIDE ENRICHMENT PLANT CAPACITY (NAMEPLATE CAPACITY 1000 SWU)
(source: WNA [2])

Country	Owner	Gaseous diffusion	Centrifuge
China	CNNC	300	1 000
France	Eurodif	10 800	
Germany	Urenco		1 850
Japan	JNFL		1 050
Netherlands	Urenco		1 500
Russian Federation	Minatom		20 000
UK	Urenco		2 500
USA	USEC	11 300	
Others*			300
Total		22 400	28 200

* Argentina, Brazil, India and Pakistan.

replaced with more modern and cost effective centrifuge capacity. The first replacement facilities will be in place in France and the USA by the early to middle part of the next decade. In addition, the process of gaining approval is underway for a consortium of European and US companies to build an enrichment plant in New Mexico, which could add about 3 million SWU to US capacity. The future of that plant is, however, not certain. While there is generally strong support for the facility at the local level there is opposition at the state level that could slow the permit process.

The near-term adequacy of enrichment capacity seems reasonably well assured. However, while new centrifuge plants are being constructed to replace diffusion capacity, the timing and magnitude of this replacement capacity remains uncertain. Improvements in centrifuge technology will probably allow existing centrifuge plants to increase their capacities, partially offsetting the uncertainty surrounding the replacement schedules for the diffusion plants. For the longer term, however, it is evident that new enrichment capacity will have to be added between 2025 and 2050. Modular expansion of existing capacity will probably be sufficient to add additional capacity, but new plants will also have to be built to ensure an adequate supply of enrichment capacity up to 2050. Given the potentially long lead times for approvals and construction, planning for these plants needs to be started in the near future.

10. UNCONVENTIONAL SUPPLY

The balance between supply and demand described in Section 3.5 was limited to uranium produced by what have historically been referred to as 'conventional' extraction methods: open pit and underground mining, conventional milling and in situ leaching. As market prices continue to increase, 'unconventional' production methods could once again become economically viable. In particular, interest in recovery of uranium as a by-product of phosphoric acid production could be renewed if an increase in the current uranium market price to US $100/kg U takes place, which is considered to be sustainable.

Recovery of uranium from marine phosphorite deposits employs well established technology, which has been used in Belgium, Canada and the USA as recently as 1999. In addition, uranium was recovered from processing organic phosphate deposits in Kazakhstan up until 1994. Estimates of the uranium resources associated with marine and organic phosphorite deposits total approximately 9 million t U, with four countries — Jordan, Mexico,

Morocco and the USA — accounting for about 90% of these estimated resources [1].

Production capacity constraints will far outweigh total resources as far as determining the future importance of recovering uranium from phosphorite deposits. Cumulative production from US operations totalled nearly 18 000 t U. At their peak, the US operations produced about 1000 t U annually. The report on uranium supply to 2050 [1] projected that uranium recovery could total 3700 t U/a, with the total output depending more on the phosphate fertilizer market than on uranium demand. To offset the uncertainty related to fertilizer economics, and to take a full measure of the potential impact on supply and demand, we chose a more optimistic outlook for production of uranium from phosphorite deposits and assumed a 5000 t U worldwide capacity. While the addition of this capacity to the supply–demand model reduced cumulative deficits, it did not change the general conclusions regarding when supply deficits would first occur and when prices would have to rise to the next production cost category to ensure timely development of new production centres.

The addition of uranium recovered as a by-product of manufacturing phosphate fertilizers does not significantly alter conclusions regarding the future balance between supply and demand. This fact does not, however, diminish the potential importance of these unconventional resources in meeting future needs. Uranium recovery from phosphorite deposits is based on well tested technology. The processes are well understood as are the economics. The timing as to when these resources enter the market is simply a matter of market price and uranium demand.

There are also other unconventional supply sources that could be developed should very high cost resources be required in the future. Black shale deposits are estimated to host between four million and five million tonnes of uranium, with most of the resources being located in Germany, Sweden and the USA. While a potential source of uranium for the future, black shale deposits are very low grade and would require huge open pit or underground mines and conventional mills for their development. The probable environmental 'footprint' of black shale mines and mills, as well as their high extraction costs, will make their development unlikely before 2050.

Any discussion regarding unconventional uranium resources would not be complete without mentioning the resource potential of sea water, which is estimated to contain as much as 4×10^9 t U. Even with the recent price increase, however, the estimated cost of extraction of uranium from sea water of US $300/kg U is still more than four times the current long term market price. Sea water may represent a long term resource, but the technology for its extraction has not progressed beyond the laboratory or pilot scale. This is not

to say that research into technology for extracting uranium from sea water is not a worthy endeavor. At the same time, the long term potential of extracting uranium from sea water should not be a distraction from the more pressing issues surrounding near-term to midterm supply and demand.

11. THORIUM

In the early 1970s, orders for high temperature reactors that can run on the thorium fuel cycle prompted assessment of worldwide thorium resources. Interest in the thorium fuel cycle did not, however, develop and as a consequence there has never been a large commercial market for thorium. Nevertheless, with renewed interest in the thorium fuel cycle, particularly in India, a brief summary of thorium resources is appropriate in a discussion about resource availability for the nuclear fuel cycle. Because thorium has had a limited market, there has been little incentive to explore for new thorium deposits or to develop detailed information about currently known thorium deposits. Accordingly, most of the information on thorium resources dates from the 1960s and early 1970s.

Table 11 summarizes worldwide thorium resources estimated to be recoverable at less than US $80/kg Th.

In addition to the resources listed in Table 11, there are significant low grade thorium deposits that host resources with unspecified production costs.

12. CONCLUSIONS

The overall aim of this paper is to assess the adequacy of uranium resources and production capacity to meet reactor uranium requirements and to ensure a long term balance between supply and demand. While the adequacy of uranium resources was the main emphasis of the research for this report, summary evaluations of the availability of enrichment and conversion services to meet future demand were also undertaken.

As a lead-in to this summary of our findings, perhaps we should note that there is neither cause for alarm nor reason for complacency as we look to the future. There is reason for concern because there is the potential for supply to fall short of demand in the later years of the period under consideration. At the same time, the shortfalls, if they occur, are projected to be far enough into the future that actions can begin to be taken now to ensure that they do not occur. In the following

TABLE 11. WORLDWIDE THORIUM RESOURCES: RARs AND INFERRED RESOURCES (1000 t Th) [5]

Country/region	Reasonably assured resources	Estimated additional resources	Host rocks
Brazil	171	50	Monazite beach sands; alkalic igneous rocks
Canada	n.a.[a]	44	Quartz pebble conglomerate; Elliott Lake mill tailings
Egypt	n.a.	100	Monazite beach sands
Denmark (Greenland)	54	n.a.	Metasomatic deposits
India	319	n.a.	Monazite beach sands
South Africa	18	n.a.	Monazite placer sands, Karoo formation; vein deposits
Turkey	344	n.a.	Volcanic breccia
USA	122	278	Vein deposits; monazite placers
Russian Federation	75	n.a.	Alkaline complexes
Australia	13	<1	
Venezuela	n.a.	300	
Norway	n.a.	132	Alkaline complexes
Others	23	10	
Total	1139	914	

[a] n.a.: not applicable.

sections are the conclusions that have been reached about supply–demand relationships for each of the sectors at the front end of the nuclear fuel cycle.

12.1. URANIUM SUPPLY AND DEMAND

We have partially offset the vagaries of long term forecasting by establishing three demand cases:

(1) The low demand case, with an average annual growth rate of –0.04%;
(2) The middle or reference demand case, with a projected average annual growth rate of 1.6%;
(3) The high demand case, with a projected annual growth rate of 2.6%.

The uranium supply is divided into secondary supply — for example, HEU, inventory drawdown and MOX — and primary supply or newly mined and processed uranium. The demand for primary supply has been determined by subtracting the secondary supply projected to be available on an annual basis from the total annual demand.

Uranium resources have been subdivided by confidence category using Red Book terminology and by production cost categories. Estimates were made as to when deposits or resources could be developed considering the permit and development time frames. These parameters were then integrated to determine the adequacy of different confidence categories of resources to meet varying demand scenarios. The supply–demand model was predicated on the assumption that the uranium production industry is market based and that the lowest cost producer will fill the first increment of demand, followed by the next highest cost producer until demand for a given year is satisfied.

The following conclusions have been reached as to the adequacy of different confidence levels of resources to satisfy a range of demand projections:

(a) High confidence study resources (i.e. those resources that are related to known deposits) are adequate to satisfy the low demand case up to 2050.
(b) Identified resources (RARs and inferred resources) are adequate to satisfy the reference demand case. Resources with projected production costs of US $78–130 could be required to meet demand as early as 2036.
(c) Identified resources and prognosticated resources are adequate to satisfy the high demand case.

Increasingly lower confidence resources will be required to meet the higher demand cases, including inferred resources in the reference case and prognosticated resources in the high demand case. Although there is a geological basis for these lower confidence resources, their name alone sounds a warning. They will require further exploration to be upgraded to higher confidence categories, and this exploration must take place years in advance of when these more speculative resources will be needed to ensure that there will be time for their permission/licensing and development.

Despite these reservations, the industry can draw some comfort from the fact that resources through the prognosticated category are projected to be adequate to satisfy the high demand case. That still leaves speculative resources, the lowest confidence resource category, to draw from to satisfy unexpected shortfalls. Speculative resources recoverable at less than US $130/kg U are estimated to total 4 440 000 t U [4], which is a substantial buffer against uncertainty. However, the caution expressed previously for other low confidence

resources applies even more emphatically for speculative resources. It can be a long and tortuous path between resources in the ground and 'yellow cake in the can', particularly when starting with the uncertainties surrounding speculative resources. The industry must now begin to develop a strategy to ensure that low confidence resources are advanced to higher confidence categories on a timely basis. New generations of geologists and engineers must be trained to advance this effort. The earlier this programme is implemented, the more likely it will be that the experienced scientists and engineers that have carried the industry through its first 65 years will still be able to train the next generation.

While uranium resources are adequate to fill the demand for primary supply in the three demand cases, additional challenges must be met to ensure that the industry is able to deliver those resources to the marketplace. Careful planning will be required to make certain that adequate time is allowed for extended permit (two to ten or more years) and construction (one to ten years) timetables. Community relations programmes need to be put in place to ensure public support for project development. Even then, well organized public opposition to a project may preclude its development on a timely basis. These uncertainties require that the balance between supply and demand is not close. Instead, excess supply should be advanced to higher confidence resource categories to accommodate projects that cannot be developed on a timely basis because of local or institutional opposition.

12.2. MARKET PRICE PROJECTIONS

When research for this paper was being carried out, the market price for uranium was about US $78/kg U, near the high end of the medium–high cost range established for this study. These price levels have been brought about by a combination of events, including interruptions in production at two key production centres, possible closure of a large mine, uncertainty about availability of Russian HEU and delays in project development because of local opposition to uranium mining. None of these 'events' could have been forecast, and their true long term impact on the balance between supply and demand has yet to be determined. Nevertheless, the current price levels cannot be ignored; they exceed those projected by the supply–demand model, which indicated that prices at these levels would not have been anticipated until about 2020, depending on the demand case.

The model, in other words, suggests that the market has overreacted to a short term set of circumstances. The market, on the other hand, is:

(a) Not confident that the projections from the model regarding production capacity can be met;
(b) Not convinced that secondary supply projections are sustainable; or
(c) Not confident about a combination of 1 and 2.

In other words, the market is leaving its options open by pushing prices to levels it believes are needed to develop new capacity, reduce average tails assays or release more supply from whatever source. Whether the market is correct or the model is correct remains to be seen. The market is determined in part by uncertainty and emotion. The model evaluates a set of assumptions and balances supply and demand then projects price trends accordingly — without regard for uncertainty or emotion. The model assumes that profitable supply opportunities translate into mine construction leading to target output levels.

How high the price will have to rise to ensure a balance between supply and demand will be determined by a combination of factors including:

(a) The level of confidence that the market has regarding the adequacy of production capacity to meet demand;
(b) The highest cost producer needed to fill demand under a given scenario.

Market confidence in production capacity can only be established by performance — the industry must demonstrate that it can satisfy projected primary demand. With regard to secondary supply, the impact on the market of the potential release of additional government inventories including HEU has not been entirely resolved. Depressed uranium prices during the past 20 years have been responsible for declining production and industry consolidation, and secondary supply has been required to supplement primary supply. The single factor that had the most impact on keeping prices at depressed levels during the 1990s was the availability of Russian HEU and of other government inventories. The recent rapid rise in the uranium market price is caused in part by continued uncertainty regarding the availability of government inventories. This uncertainty is a disincentive to the development of the new production capacity that will be needed to balance supply and demand in the future. Market stability will be well served by governments making their intentions known regarding future availability of their nuclear stockpiles. Likewise, countries such as Australia must make their policies on uranium mining clear and lasting in order to allow proper supply planning elsewhere if necessary.

12.3. CONVERSION CAPACITY

Current UF_6 conversion capacity, if supplemented by currently held inventories, will probably be adequate to satisfy demand up to 2025. In the period between 2025 and 2050, however, there will be a need to approximately double UF_6 conversion capacity in the reference and high demand cases. Expansion of existing UO_2 conversion capacity will probably be adequate to cover demand during the study period. Suppliers are sensitive to market needs and are expected to respond with increased conversion capacity as required.

12.4. ENRICHMENT CAPACITY

Conclusions regarding the adequacy of enrichment capacity to meet future demand essentially mirror those described above for conversion. The enrichment industry is in transition as ageing gaseous diffusion plants are due to be replaced by more cost effective centrifuge capacity. Although plans are in place for these replacement plants, the industry needs to keep their construction schedules under review to ensure that no imbalance occurs between enrichment capacity and requirements. In the longer term, enrichment capacity will need to approximately double between 2025 and 2050 in both the reference and demand cases. This increase will require the building of new capacity, which needs to be anticipated well in advance of when it will be needed in order to accommodate potentially long timetables for permissions to be obtained.

12.5. THE FUTURE

Primary supply has been the main focus of this paper, with emphasis having been placed on the adequacy of resources and the production capacity needed to move those resources to the marketplace. We have determined that uranium resources of varying confidence levels are adequate to meet a broad range of demand scenarios. This conclusion, however, relies on credible resource estimates and on the ability of the industry to continually upgrade its confidence in resources to offset depletion of identified resources in the latter part of the study period.

The industry must also demonstrate that resources can be developed and delivered to the marketplace when they are needed. The challenge to the uranium industry is clear; annual primary supply requirements are expected to increase threefold between 2005 and 2050 in the high demand case. The

uranium industry must keep this goal in focus as it plans for development of new production capacity.

The future of nuclear power depends on an adequate supply of uranium and fuel services to meet future demand. While we can model the industry using different sets of assumptions to forecast demand, we must not be uncritical of our models. Sensitivity analyses can be made to help bracket risks and uncertainty, but uncertainty is inevitable: the road to 2050 will not be easy. The path between 'resources in the ground and yellow cake in the can' is unpredictable. Therefore, adequate lead times in planning will be required as a buffer against unforeseen delays, whether in the development of new mines or the construction of new conversion and enrichment capacity. The nuclear industry can ill afford supply and demand to be in close balance and for there to be no margin for error.

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URANIUM MARKET SECONDARY SUPPLIES: AVAILABILITY AND APPRAISAL OF MARKET IMPACT

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Abstract

In addition to the uranium concentrates produced at mine sites, it appears that very significant quantities of fissile material are contributing to an increase in the supply side of the nuclear fuel supply and demand equation. In order to name and characterize these additional sources of fissile material, the term 'secondary supplies' is used in a very broad way that includes the following sources: commercial uranium inventories, excess high enriched uranium (HEU), excess non-HEU uranium government inventories, excess military plutonium recycled as mixed oxide fuel (MOX), civilian plutonium from spent fuel reprocessing recycled as MOX, reprocessed uranium (RepU) from spent fuel reprocessing recycled as RepU fuel and re-enriched depleted uranium (tails). The paper discusses in detail the characteristics of each of these secondary supplies, their future availability and likely annual contribution to reactor requirements. It must be underlined that secondary supplies will contribute very differently to market balance in the future from the current situation. Some, such as excess commercial inventories, are likely to end soon, even shifting to a 'negative contribution', for strategic inventory replenishing purposes, for example. Others, such as spent fuel fissile material recycling, are likely to last for the entire period and beyond, even if the pathways that can be envisaged might differ, long term, from current routes. Previous studies on these topics were written in a period of lackluster perspectives for nuclear power growth. The perception of the future of nuclear power has now fundamentally changed, at a time when the availability of excess inventories appears in question. Therefore, the prospects of secondary supplies have clearly changed, as have their potential market impact.

1. INTRODUCTION; DEFINITION AND LIST OF SECONDARY SUPPLIES; RECENT CHANGES IN TRENDS

In addition to uranium concentrates produced at mine sites, otherwise termed 'primary supplies', very significant quantities of fissile material are

contributing to the supply side of the nuclear fuel supply and demand equation (Fig. 1).

In order to name and characterize these additional sources of fissile material, the term 'secondary supplies' is widely used as a very broad term that includes the following sources:

(a) Commercial uranium inventories in all forms (from uranium concentrates to possibly fabricated fuel);
(b) Excess highly enriched uranium (HEU);
(c) Excess non-HEU uranium government inventories;
(d) Excess military plutonium recycled as mixed oxide fuel (MOX);
(e) Civilian plutonium from spent fuel reprocessing recycled as MOX;
(f) Reprocessed uranium (RepU) from spent fuel reprocessing recycled as RepU fuel;
(g) Re-enrichment of depleted uranium (tails).

The meanings of these terms are discussed in detail in Section 3, along with the availability of each of these secondary supplies. Finally, the availability figures are added, to establish the total annual market contributions from all secondary supplies and their potential market impacts (Section 4).

It is important to emphasize here that these secondary supplies will contribute to market balance very differently in the future from the way they have in the past. Some are likely to come to an end soon, even shifting to 'negative contributions', for strategic inventory replenishing purposes,

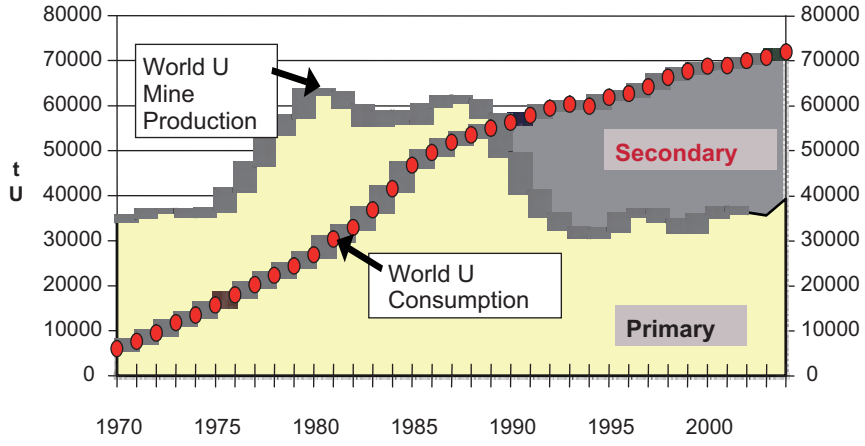

FIG. 1. Historical uranium secondary sources.

for example. Some are likely to last for the entire period under review and beyond, even if under a different path, such as spent fuel fissile material recycling.

It is also important to recall that a previous study, entitled 'Analysis of Uranium Supply to 2050' [1], even if not so old, was written in a period when the uranium market was depressed. The current perception of secondary supplies is significantly modified, and commercial inventories are now being reluctantly utilized 'to bridge the gap', and no longer to raise cash.

2. URANIUM MARKET EVOLUTION AND GENERAL TRENDS IN SECONDARY SUPPLIES

The uranium market is going through its first major change for two decades. The most important change, apart from prices, is a shift from a secondary sources contribution exceeding 30 or even 40% of total requirements to lower numbers (Fig. 1). This is primarily due to the long awaited depletion of excess commercial inventories built up during the late 1970s and early 1980s from Western production, and even a decade more from the former Eastern bloc production.

A few production accidents contributed to the acceleration of the depletion process of these 'historical inventories' and the decline in their market impact. As a result, the uranium price rose significantly and many believe it is likely to follow this trend until a new equilibrium is reached between increased production and actual natural uranium requirements.

Taking into account the rather stable enrichment prices, the first adjustment should be through the change in enrichment tails assays, a normal adjustment variable based upon the economic ratio of feed cost to separative work unit (SWU) cost. Its effect is not strictly that of a true secondary supply except when the actual tails assays depart widely from the median tails assays as is the case for Russian Federation enrichment plants, leading to a uranium stripping contribution to secondary supply.

Also linked to isotopic enrichment operations is re-enrichment of depleted uranium. While this secondary supply source is currently supplying significant quantities of uranium to the market, it remains of second order. The two above mentioned sources should not be aggregated with the normal tails assay optimization effect.

Another major secondary source, excess HEU, is now approaching the halfway point of its life, at least for quantities publicly known as available for the market. In 2013, the US–Russian HEU agreement will come to an end, and even if one speculates about its extension, significant uncertainties remain

about the precise date when it will end, not so far from now, and already within the current time frame for new contracting.

3. DETAILED DESCRIPTION OF MAJOR SECONDARY SUPPLIES AND THEIR TRENDS

3.1. Commercial uranium inventories

3.1.1. General description and data

It should be stressed that, in general, numbers about commercial inventories are not available to the public, and that estimates necessarily carry significant uncertainties. In addition, departing from a previous IAEA study [1], the inventory held by the Russian Federation is not considered here as falling into commercial inventories but instead into non-HEU material held by the Government of the Russian Federation.

Entities holding commercial inventories are very different in kind. They are primarily nuclear power plant operators, otherwise the said uranium end users, as they need to have ownership not only of 'pipeline material' but also of strategic inventories and of additional quantities beyond strategic needs. The related material covers uranium in all its forms, from concentrates to fabricated fuel:

(a) Uranium producers own concentrate inventories awaiting shipment at mine sites or at conversion sites, mainly for risk coverage.
(b) Processors, especially enrichers, must keep significant inventories at their sites, especially as natural UF_6 feed material or enriched uranium product (EUP) material. However, these inventories generally belong to utility customers, with some minor exceptions.
(c) Finally, traders (and more recently, speculators) hold limited and variable inventories for commercial purposes.

Owing to the major differences between justifications for inventory keeping, resulting policies differ greatly. Basically, utilities aim particularly at minimizing supply disruptions, while producers aim to offset risks when guaranteeing delivery schedules, and some utilities speculate about trading income.

In addition to 'pipeline inventory', which is hardly available for the market, another essential distinction is between strategic inventory and discretionary inventory:

(a) Pipeline inventory is set by the annual requirements at each fuel cycle step, regardless of who is the owner of the material.
(b) Strategic inventory is set to be permanently held, with some periodic adjustments to deal with updates of risk evaluation.
(c) Discretionary inventory allows opportunistic arbitration in the market and can or should be sold, traded or otherwise disposed of. It contains what is sometimes referred as 'excess inventory'.

In some cases, inventory policies are a matter of public record, but more typically they are commercial confidential information.

However, some estimates can be found and when combined with published data they allow access to a broad but meaningful picture (Table 1).

From the total inventory given in Table 1, only a small share, if any, is believed to be excess inventory. One can remark on the recent increase in the inventories of US utilities (Fig. 2), and if the optimum level has not yet been reached elsewhere, the excess inventory, on a net global scale, is probably at most 20 000 t U for utilities and less than 10 000 t U for suppliers.

Normally, the full commercial inventory of the United States Department of Energy (USDOE) is potentially available throughout the period. Mainly stockpiled as an HEU feed source, its availability remains partly in question.

TABLE 1. PUBLISHED DATA AND ESTIMATES OF COMMERCIAL INVENTORIES
(from Uranium Institute/World Nuclear Association (WNA) [2–4] and USDOE Energy Information Administration [5])

Entity	Year end inventory (t U)		
	1997	2002	2004
Utilities	113 000	110 500	114 000
Of which, US utilities	25 300	20 600	About 21 680
Uranium producers	20 000	15 000	10 000
United States Enrichment Corporation (USEC)	30 000	< 18 700	< 15 000
USDOE commercial inventory	5 500	20 000	< 20 000 (estimate)

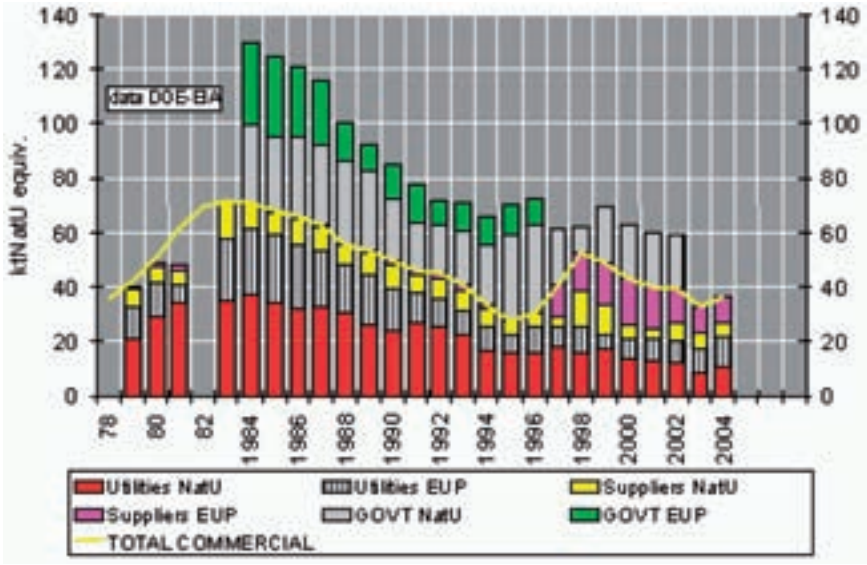

FIG. 2. Commercial stockpiles trend; example of US data.

3.1.2. Scenario for use of commercial inventories: Shifting from a supply (drawdown) scenario to a demand (inventory replenishment) scenario

As just explained, and contrasting with the situation that prevailed when the previous report [1] was drafted, commercial inventories are now basically depleted of any sizeable excess. Therefore, future contributions will be limited to the positive or negative adjustments required to balance the fuel cycle demand, and/or to rebuild excessively depleted stockpiles.

Assumptions for market modelling purposes:

(a) *Pipeline and utilities strategic inventories*: If it is correct that these are at the optimum level, then they will have to be adjusted annually in proportion to the evolution of requirements (an assumption of a change of 100% in annual requirements is proposed). Some of the strategic inventories will probably be used to balance supply and demand, and must be replenished afterwards when possible.

(b) *Government strategic inventories*: Their trend is in general hardly predictable. The USDOE commercial inventory is considered to fully reach the market. Regarding Russian Federation inventory, the published contribution is certainly the best assumption to make.

(c) *Producers' inventories*: These are tied to sales commitments. They will have to remain in aggregate at levels equal to 50–66% of the previous year's production.
(d) *Fuel cycle producers (converters, enrichers and fabricators)*: Although they hold most of the physical inventories, they usually own limited or even no inventories. It can be assumed that this situation will not change.

3.2. Excess non-HEU uranium government inventories

It is believed that, with the noticeable exceptions of the Russian Federation and of the USA, no significant uranium inventories of commercial grade (from natural uranium to 5% ^{235}U) are currently held by governmental entities.

There are neither official figures of Russia's total non-military inventory nor reliable estimates.

Regarding USDOE commercial material, data were available until 2002, but this is no longer the case. Therefore, analyst estimates and the subjectivity associated with these characterize drawdown projections for these types of inventory.

3.2.1. The Russian inventory

Uranium production in the former Union of Soviet Socialist Republics (USSR) and Eastern Europe far exceeded military and civilian requirements, resulting in the buildup of a large stockpile of nuclear material. The total extent and availability for civilian use of the commercial grade part of this stockpiled material is uncertain. What is known, however, is that only a limited amount of the material conforms to international specifications and is thus suitable for immediate use in reactors. The remainder of the material would require considerable additional reprocessing to make it suitable for reactor fuel, and in fact some will probably never be commercially useful.

The Uranium Institute/World Nuclear Association (WNA) [2–4] estimated that the Russian inventory of potentially available commercial grade material at the end of 1997 totalled approximately 58 000 t U. According to official Russian statements and publications [6], the priority for the Russian Federation's non-military inventory will be to satisfy its internal reactor requirements as well as supply commitments to Russian built reactors in other Commonwealth of Independent States (CIS) and Eastern European countries, and as a supply incentive to purchase newly exported Russian reactors. Some charts were published indicating at what rate inventory will be made available for civilian use to around 2020; unfortunately it seems there is a mix between

various types of uranium forms and sources. Our interpretation is that only limited amounts are now available, *in the range of 10 000 t U*, and that these will be consumed *during the next 10 years*. The other part of the flow, further into the future, will be an HEU related flow and must be identified as such.

3.2.2. The USDOE inventory

As mentioned above, increased uncertainties now characterize USDOE data. As a result, the USDOE Energy Information Administration (EIA) decided to withdraw the related data from its publications. A contribution of around 20 000 t U is tentatively considered throughout the period 2007–2020 (about 500–2000 t U/a).

3.2.3. Other government held uranium inventories

These are believed to be limited and, according to current market conditions, are likely to be either sold when they belong to non-nuclear power equipped countries (a recent example for small quantities is Portugal) or to remain stored for local requirements (e.g. China and the Czech Republic).

3.3. Excess highly enriched uranium

After commercial inventories, this is the most prominent part of total secondary supplies. Very large quantities of uranium have been enriched to HEU level (beyond 20% ^{235}U), mainly in the countries of the former USSR and in the USA for military purposes. A dominant share of this material was enriched to or beyond the 90% ^{235}U level.

As a consequence of nuclear arsenal reduction treaties, excess HEU has been identified in both the Russian Federation and the USA [7].

At present, 500 t of Russian HEU and 174 t of US HEU have been declared surplus and included in programmes to market their uranium and other components. It is assumed that the potential HEU contributions, if any, from other countries, will be negligible from the viewpoint of their impact on the uranium market.

3.3.1. Market impact of the US–Russian HEU agreement

Many documents are available on this subject; see, for more details, Ref. [8]. To summarize, 500 t of weapons grade HEU are set to be down blended to LEU grade material in the form of UF_6, ready for use at US fuel fabrication plants from 1995 to 2013.

This material currently represents an annual flow of about 910 t of LEU equivalent to 9100 t NatU as UF_6 and 5.5 MSWU.

In order to blend down the HEU, an annual backflow of 2600 t of uranium as natural UF_6 is transported to the Russian Federation. As a result, the apparent net balance for the world market is 6500 t of natural uranium. Owing to the recovery of additional uranium, achieved through a 'tails stripping process' at Russian enrichment plants, we believe that the ultimate rounded figure for the annual net balance is closer to 7000 t U, and this hypothesis is retained for our market impact evaluation (Fig. 3). The SWU content is purchased and subsequently marketed by the US enricher, the United States Enrichment Corporation (USEC). The feed content (the 'Russian HEU feed') is purchased and then marketed by Cameco, AREVA and RWE NUKEM, on the US side and TENEX on the Russian side.

A system of quotas is in place to limit the impact of this feed material on the US market (according to the 'Suspension Agreement' of a past anti-dumping action). However, there is no formal limitation for exports to Europe and Asia. As a result, the above mentioned net balance is considered as equivalent to the actual quantity reaching the market annually.

Note that a share of the feed material (initially about 11 000 t U as UF_6) was acquired by the US Government and is now managed as a USDOE commercial stockpile, with some restrictions on the scheduling of its market availability. In our scenarios, the fate of this material will be discussed within the USDOE commercial stockpile disposition programmes.

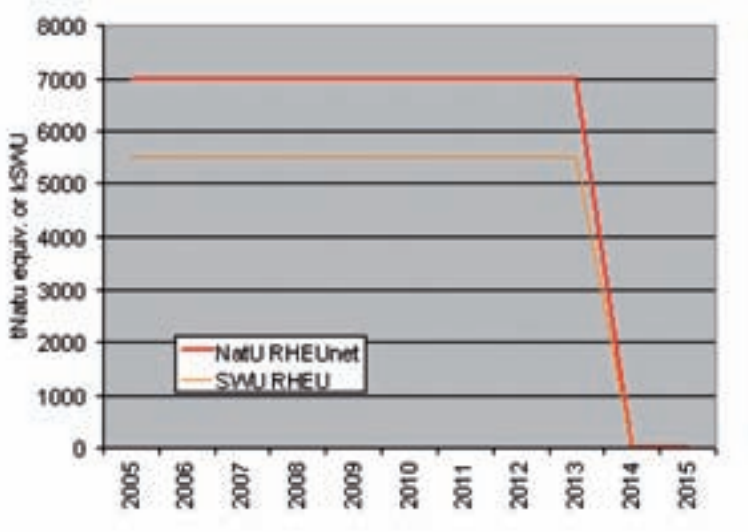

FIG. 3. Impact of the US–Russian HEU agreement.

Another share of this feed was stockpiled in the Russian Federation (the 'monitored inventory') and is either usable for blending down or submitted to similar schedule restrictions as its US counterpart. We believe that, at most, only a small amount of feed will remain available for the market at the end of the agreement.

From the mid-1990s until mid-2003, the US–Russian HEU agreement had a rather clear impact, through its forecast effect on the market balance and pushing down of market prices. The impact of this major secondary supply has now been fully assimilated by the market, and one might even say that the reverse is now likely. The impact on the market caused by the normal end of the agreement will be that of a steep reduction in supply, and a premature ending would have an even greater effect. The important date is thus 2013.

There is a great deal of speculation about the likelihood of a continuation of the agreement, often quoted as HEU-2. The Russians themselves have made clear that the existing agreement was concluded soon after the collapse of the former USSR, a moment when the Russian Federation was desperately seeking hard currency resources. The situation no longer pertains, and, even if additional HEU remains, its use is likely to be directed to fuel Russian nuclear power plant (NPP) programmes for both internal requirements and exports.

3.3.2. Market impact of other Russian HEU

It is well known [8] that some irradiated HEU from Russian Navy propulsion reactors has been recovered and used to upgrade reprocessed uranium. The product is of commercial grade and enters into the fuel fabrication process at the TVEL Elektrostal fabrication plant. The resulting fuel elements are either for Russian RBMK reactors or light water reactor (LWR) fuels for Framatome ANP European customers.

Because this flow is now essentially allowed by HEU availability, it is counted 100% an HEU linked secondary supply. It represents about 100 t HM (heavy metal) [9] for LWR fuel (about 800–1000 t NatU equivalent) and about half of the RBMK fuel elements, i.e. around 250 t HM (about 1200 t NatU for Russian tails assays).

3.3.3. Market impact of US HEU

At present, the technically recoverable portion of the 174 t of declared excess US HEU are being brought to the market by two main routes [8]:

(1) Quantities transferred by USDOE to USEC: These are now accounted for as commercial inventories, and delivered to the market through normal commercial transactions.
(2) Out of specification HEU transferred to the Tennessee Valley Authority (TVA) after blending will be fabricated and loaded in the federal utility TVA reactors after 2005. This material will probably displace the equivalent of 5000–10000 t NatU on the market.

Other routes for limited quantities are being studied, including possible incentives to help orders for new reactors in the USA.

3.4. Excess military plutonium recycled as MOX fuel

In order to eliminate weapons grade plutonium inventories declared excess by the governments involved, a twin programme was set up between the USA and the Russian Federation with the participation of other partners, primarily the European Union and France for financing and for the MOX fabrication technology. The agreement calls for a roughly parallel evolution of both projects, at least for the first steps, even if the actual implementation is progressing at different speeds.

About 50 t of Pu has been declared excess in the USA and about 100 t in the Russian Federation. For various reasons, primarily the isotopic and chemical/physical compositions, smaller quantities are contemplated for a first recycling programme as MOX fuels to be loaded into civilian NPPs. Changed over the years, the quantities considered here are about 34 t Pu for each country (Table 2).

For estimating the market impact of these programmes on natural uranium equivalent savings, a factor of 9 is suggested as a rounded figure (and 6.5 for the SWU savings), taking into account a theoretical tails assay of 0.25% ^{235}U.

Officially the US full scale programme is set to start in 2009. A one year delay is considered here (Table 3, Fig. 4). The first plans were for parallel programmes. However, the Russian programme seems to be less advanced, and a five year lag is added here. Additional quantities for a six year prolongation (15–16 t Pu each) are considered in the ‘high’ case, and there is still a potential for more.

TABLE 2. MOX FUEL FABRICATION PROGRAMMES FROM EXCESS MILITARY PLUTONIUM

Programme		Period				
		2006–2010	2011–2015	2016–2020	2021–2025	2026–2050
US MOX capacity (t HM)	US-MFFF* (for 34 t Pu)	0 (Starting 2010?)	60	60 (Ending 2022?)	Additional quantities?	None?
Annual natural uranium savings (t NatU)	US-MFFF (for 13 years)	0	540	540	Potential for additional quantities	
Annual SWU savings (kSWU)	US-MFFF	0	390	390		
Russian MOX capacity (t HM)	R-MFFF (for 34 t Pu)	0 (Starting 2015?)	60	60 (Ending 2027?)	Additional quantities?	Additional quantities?
Annual natural uranium savings (t NatU)	R-MFFF (for 13 years)	0	540	540	Potential for additional quantities	
Annual SWU savings (kSWU)	R-MFFF	0	390	390		

* MFFF: mixed oxide fuel fabrication facility.

TABLE 3. NATURAL URANIUM DISPLACED BY MOX FUEL FROM EXCESS MILITARY PLUTONIUM

	2010	2011	2012	2013	2014	2015	2016	2017	2018	2019	2020
US (t NatU)	540	540	540	540	540	540	540	540	540	540	540
Russian (t NatU)						540	540	540	540	540	540
Total (t NatU)	540	540	540	540	540	1080	1080	1080	1080	1080	1080

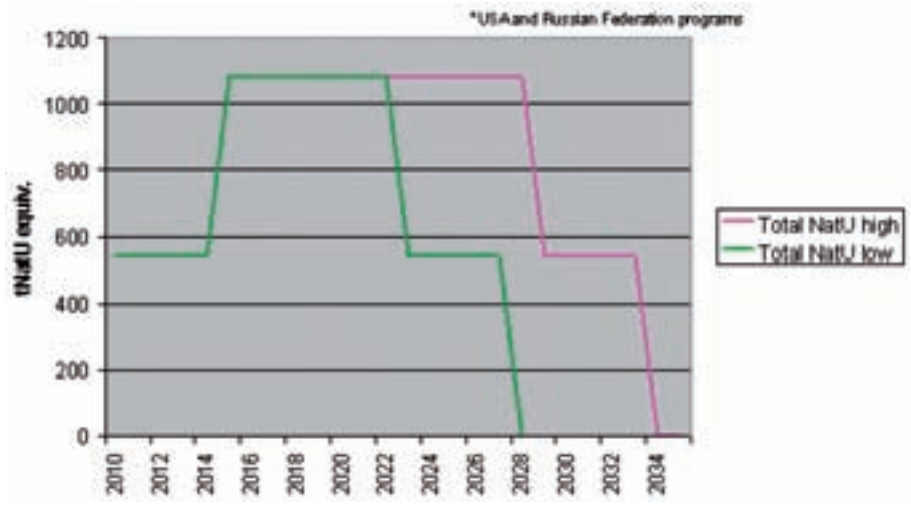

FIG. 4. Savings in natural uranium from weapons grade plutonium recycling programmes.

3.5. Recycling of fissile material recovered from spent fuel reprocessing

3.5.1. Spent fuel reprocessing: Status and trends

Spent nuclear fuel can be reprocessed to separate the remaining uranium and plutonium formed during irradiation from waste products. Uranium and plutonium recovered during reprocessing can be recycled and used in new fuel assemblies, and, therefore, they become a secondary supply source and can effectively displace equivalent amounts of primary supply.

At present, five countries have established reprocessing/recycling programmes: Belgium, France, Germany, Japan and Switzerland. Sweden is also considering recycling of its separated plutonium. However, in three of these countries, current nuclear policy is having an impact on MOX recycling, and there are some uncertainties about the future use of this type of fuel.

Four countries currently have commercial reprocessing facilities operating, under startup or planned: France, Japan, the Russian Federation and the UK. China is constructing a small facility and is studying the building of a larger one around 2020. These facilities currently define the foreseeable upper limit for fissile material recycling. However, several countries, such as the Russian Federation constantly state that future requirements will be partly covered by enhanced recourse to recycling fissile material from spent fuels. In the future, another driver for reprocessing will be separation of minor actinides

to allow their transmutation, probably in fast neutron reactors, thus allowing recovery of more uranium and plutonium, separately or mixed, and their subsequent recycling. This will imply the building of large new reprocessing plants.

In order to study the impact of a more limited reprocessing flow, a low reprocessing case is proposed. This will imply increased primary uranium requirements (Table 4).

3.5.2. Civilian plutonium from spent fuel reprocessing recycled as MOX

Plutonium from reprocessing is used to manufacture fuels that contain a mixture of plutonium and uranium dioxides, hence the name mixed oxide (MOX) fuel. Fissile plutonium replaces ^{235}U as the major source of energy in MOX fuels, which can be loaded in most reactors in place of conventional enriched uranium fuel.

Figure 5 shows the projected uranium equivalent that would be displaced by use of MOX fuels. Only the reference case which projects MOX use up to 2050 and possibly beyond, is shown here. A low case, which assumes that MOX use will fall progressively to terminate by 2025, is tentatively described.

The reference case (Fig. 5) assumes steady growth of MOX fuel use up to 2013, after which usage stabilizes up to 2050 at 3700 t U equivalent, which approximately equals the capacity of the plants currently in operation or under construction (no significant change from our previous report). Refurbishment of existing plants (Table 5) and/or investment in new plants will be required to sustain this projection. It is unlikely that MOX usage will exceed that considered in the reference case unless the fast breeder regains its former popularity as an alternative to very high uranium prices.

TABLE 4. TWO SCENARIOS FOR WORLD SPENT FUEL REPROCESSING

	Period					
	2005	2006–2010	2011–2015	2016–2020	2021–2025	2026–2050
Total reference reprocessing capacity (t HM)	2700	3000	3800	3800	4300	4500
Total low reprocessing capacity (t HM)	2700	2700	3000	3000	3000	2000

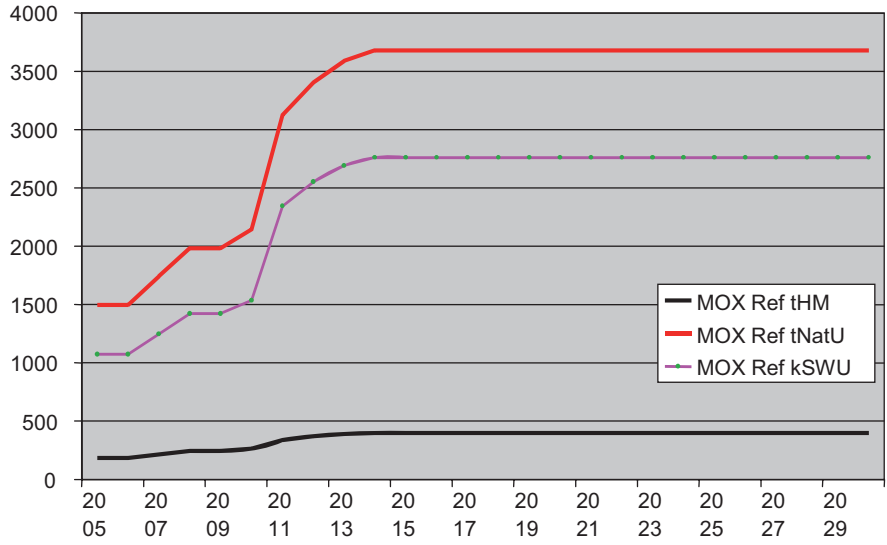

FIG. 5. Impact on the natural uranium market of civilian plutonium recycling as MOX.

The low case assumes that MOX usage will be phased out by 2025 in response to ageing facilities requiring massive investments for upgrades within a continued opposition from anti-nuclear and/or anti-plutonium campaigners, and taking into account hardened measures dealing with non-proliferation issues. We believe this case has a low probability, but it is the most challenging scenario for a study on availability of uranium supplies.

TABLE 5. MOX FUEL FABRICATION COMMERCIAL FACILITIES
(capacities in t HM)

Plants	2005 Capacity	Scheduled capacity	Planned startup date
MELOX (AREVA)	145	195	2009? New licence required
P0 DESSEL (Belgonucléaire)	40	Closure after 2007 if no orders?	
SMP (BNFL)	Under startup	120 or 60?	2008?
J-MOX (JNFL)*	Under construction	120	2010?
Total	185	375 min.– 475 max.	

* JNFL: Japan Nuclear Fuel Limited.

3.6. Reprocessed uranium from spent fuel reprocessing recycled as RepU fuel

Reprocessed uranium can be used as a direct substitute for newly produced uranium in reactor fuel fabrication. Consequently, a utility's decision about whether to use RepU is generally driven by the comparative cost of fuel manufactured using the two different sources of uranium. Therefore, projections of RepU use are directly tied to uranium market price projections; as market price increases, RepU becomes more competitive. Figure 6 shows the projected uranium equivalent that would be displaced by use of RepU.

Separated RepU inventories currently exist, mainly in France, the Russian Federation and the UK. However, to simplify the scenario, their influence over the long term has been neglected, and the main factor of influence retained is the available reprocessing capacity.

The RepU base case scenario, (Fig. 6) shows a gradual stepwise increase, which is capped at 4000 t U equivalent after 2020 for the remainder of the study period. This base case assumes a slowdown or even a ceiling for current reactor burnup increase and sufficient spent fuel feed flow at reprocessing plants (large stockpiles of relatively low burnup spent fuel exist in the world and are constantly being increased through annual unloadings).

The RepU low case (Fig. 6) shows RepU progressively falling to zero by 2025. The main reasons could be, primarily, an extension of aversion to reprocessing and, secondarily, a continuation of the current trend towards higher burnup. In this case, reprocessing plants will lack economically attractive spent

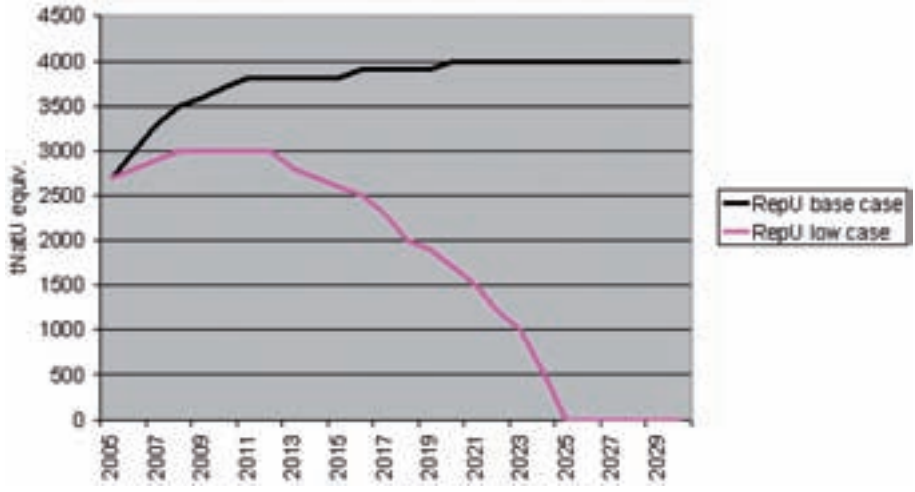

FIG. 6. Market impact of RepU recycling.

fuel from countries using reprocessing by around 2020. This scenario should at present be considered to have a low probability. It is retained to study the availability of primary uranium production. It should be added that the civilian MOX and RepU scenarios are closely tied to, and dependent upon, reprocessing policies.

3.7. Impact of isotopic enrichment on uranium supply: Tails assays adjustment impact and re-enrichment of depleted uranium (tails)

3.7.1. Background

Nuclear power is mainly produced in reactors fuelled with enriched uranium, and the trend is to extend this further to heavy water reactors which are currently fuelled with natural uranium. In the enrichment process, the fissile isotope ^{235}U is preferentially extracted from natural uranium to produce enriched uranium. Depending on the extent to which the isotopic separation is achieved, uranium requirements can vary by more than 20% under already experienced cost conditions. The first and major impact of enrichment is on the range of that variation.

To give orders of magnitude, for each kilogram of enriched uranium produced, an average of 8 kg of depleted uranium (range: 5–10 kg) is produced from 9 kg of natural uranium feed. Consequently, about 80% of the total amount of uranium devoted to fuelling reactors is now stored in the form of depleted uranium (otherwise called enrichment tails). Significant quantities (about 80% of uranium deliveries to enrichment plants) are added annually to the stockpiles. These tails contain a large quantity of total uranium potentially available for future use. Some of them have a relatively high residual ^{235}U content and are re-enriched, thus allowing a secondary separation of additional usable uranium.

3.7.2. Economic optimum enrichment tails assay: Effect of lowering the tails assays and the resulting savings

Until this year, many reports, including an IAEA analysis of uranium supply to 2050 [1], were using an enrichment tails assay of 0.3% ^{235}U, or close to this value. At the same time, for economic optimization reasons, a majority of utilities were asking for tails assays between 0.3 and 0.35%. However, for various reasons (e.g. excess marginal capacity and the availability of low cost uranium), enrichment plants were operating at a lower global average, resulting in an actual average close to the 0.3% assumption.

With the increase of uranium prices, and the relative SWU price stability, utilities notify lower tails assays and enrichers tend to adjust, thus pushing the average assay significantly below 0.3%.

In the long term, it is believed that the general trend would be a uranium price increase for all scenarios except a nuclear power decline. In the same time frame, the prices of isotopic enrichment services are likely to remain steady or even decrease due to improvements in technology. Hence, a declining trend for tails assays is foreseen. The resulting uranium savings are not strictly falling into the definition of secondary supplies, except in the case of economic conditions departing significantly from average market values. This is currently the case for enrichment in the Russian Federation, reportedly operating at tails assays below 0.15% ^{235}U.

If needed, it is also possible to forecast uranium savings beyond assumed optimum tails assays for the long term in order to bridge a potential uranium gap through the use of additional enrichment capacity, which is easier to build than finding new uranium deposits.

Figure 7 and Table 6 illustrate the major impact on uranium demand resulting from a change in enrichment tails assays. These savings are up to 20% in the case of uranium prices rising from 10 US \$/lb U_3O_8 (22 US \$/kg) to 40 US \$/lb U_3O_8 (88 US \$/kg). Obviously this kind of change will have an

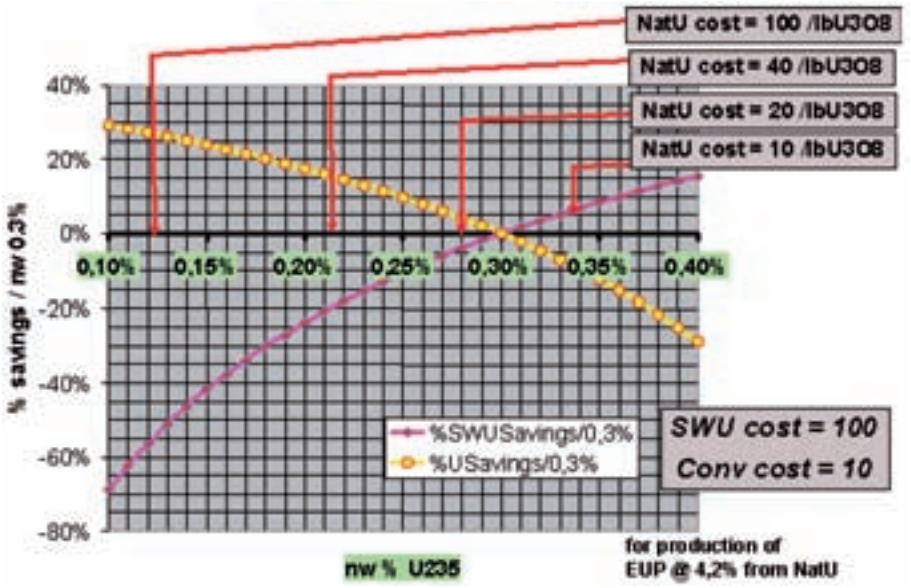

FIG. 7. Market impact from adjustment of tails assays (nw: norm of waste).

TABLE 6. URANIUM FEED REQUIREMENTS FOR 1 kg OF 4.2% ^{235}U LEU AT VARIOUS TAILS ASSAYS

(optimum tails assays are assuming a fixed conversion cost of 10 US $/kg U and a fixed SWU cost of 100 US $/SWU)

NatU cost (US $/kg U)	26	40	80	130	260
NatU cost (US $/lb U_3O_8)	10	15.4	30.8	50	100
Optimum tails assays (%)	0.338	0.303	0.240	0.193	0.122
Uranium requirements at optimum tails assays	10.36 (and 5.28 SWU)	9.55 (and 5.63 SWU)	8.41 (and 6.38 SWU)	7.73 (and 7.13 SWU)	6.93 (and 8.78 SWU)
Uranium requirements at fixed tails assays of 0.30%	9.49 (and 5.66 SWU)	9.49 (and 5.66 SWU)	9.49 (and 5.66 SWU)	9.49 (and 5.66 SWU)	9.49 (and 5.66 SWU)
Uranium requirements at fixed tails assays of 0.25%	8.57 (and 6.25 SWU)	8.57 (and 6.25 SWU)	8.57 (and 6.25 SWU)	8.57 (and 6.25 SWU)	8.57 (and 6.25 SWU)
Uranium requirements at fixed tails assays of 0.15%	7.22 (and 8.02 SWU)	7.22 (and 8.02 SWU)	7.22 (and 8.02 SWU)	7.22 (and 8.02 SWU)	7.22 (and 8.02 SWU)

impact only on reactor requirements on the basis of EUP based fuels (and will arithmetically lower the savings from MOX and RepU fuels). It is, however, important to emphasize that these savings imply a correlated increase of SWU consumption in a similar proportion.

Tables 6 and 7 show that, on the basis of current SWU and conversion market prices, the optimum tails assay for the current uranium price of US $80/kg U (US $30/lb U_3O_8 (US $66/kg)), is now tending to around 0.24% ^{235}U (owing to the effect of long term uranium contracts, a multi-year lag is observed for full tails assays adjustment). However, because of the rather flat curve of the optimum, one can see that the economic incentive to depart from a 0.30% tails assays is limited (less than 2% of the EUP cost).

TABLE 7. COST OF 1 kg OF 4.2% ^{235}U LEU AT VARIOUS TAILS ASSAYS AND URANIUM COSTS ASSUMING A FIXED CONVERSION COST OF 10 US $/kg U AND A FIXED SWU COST OF 100 US $/SWU
(cost expressed in US $/kg U)

NatU cost (US $/kg U)	26	40	80	130	260
NatU cost (US $/lb U_3O_8)	10	15.4	30.8	50	100
At optimum tails assays	901	1040	1395	1795	2748
At fixed tails assays of 0.30%	907	1040	1420	1894	3128
At fixed tails assays of 0.25%	933	1053	1396	1824	2938
At fixed tails assays of 0.15%	1062	1163	1452	1813	2752

Tables 7 and 8 also show that use of a tails assay below 0.15% ^{235}U is justified only in the case of very costly uranium (about 100 US $/lb U_3O_8 (220 US $/kg), i.e. more than three times the current level), or for the current price, in the case of very low cost SWU, typically below 50 US $/SWU, such as marginal excess centrifuge SWUs.

This paper must remain consistent with the updated IAEA study published in these proceedings ([10], Paper 1.1). In this study, uranium requirements were calculated using a 0.3% tails assay for Western plants, and 0.1% for Russian supplied reactors. Therefore, additional uranium savings between 0.3% and future optimum tails assays required by incremental marginal resources cost will be calculated and evaluated as 'secondary supplies'. We will do the same calculation to evaluate the savings already achieved through Russian enrichment operations.

TABLE 8. COST OF 1 kg OF 4.2% ^{235}U LEU AT VARIOUS TAILS ASSAYS AND SWU COSTS ASSUMING A FIXED CONVERSION COST OF 10 US $/kg U AND A FIXED NatU FEED COST OF 66 US $/kg U_3O_8
(cost expressed in US $/kg U)

SWU cost (US $/kg U)	100	80	60	40	30
Optimum tails assay	0.240	0.216	0.185	0.142	0.111
Uranium requirements	8.41	8.05	7.64	7.13	6.82
SWU requirements	6.38	6.74	7.27	8.23	9.15
EUP cost at optimum tails assays	1395	1263	1123	971	888

Since the last study, prices have risen from less than a level of 26 US $/kg U to around the 80 US $/kg U level, i.e. the lower limit of our high cost project. We believe this price spike is still disconnected from 'the last marginal production cost', and mostly reflects disequilibrium between supply and demand. When analysing the updated scenarios it appears that the current price level is probably a sound long term reference. As a result, considering tails assays in the range of 0.25–0.20% ^{235}U for the period until 2025, and of 0.15–0.20% until 2050 is certainly a reasonable assumption for the reference case.

The formulas show that lowering the tails assay from 0.30% (the demand assumption) to 0.225% results in a savings of 140 t U per 1000 t U of demand, and that lowering it from 0.30 to 0.175% results in a savings of 210 t U per 1000 t U of demand. Therefore, in the reference case, the savings from the average tails assays reduction represent about 8000 t U annually. In addition, owing to the lower tails assays achieved at Russian enrichment facilities, a further 15% reduction in the requirements must be applied to the reactors served by the Russian enricher. In total, for the period from 2005 to 2025, the impact of the tails assays is likely to represent 105 000 t U. Similarly, for the period 2025–2050, the savings amount to 438 000 t U and for the entire period up to 2050 would total 543 000 t U (they have been evaluated as 590 000 t U in the Key Issue paper ([10], Paper 1.1).

3.7.3. The depleted uranium (tails) potential

3.7.3.1. The depleted uranium status debate

Whether the depleted uranium stockpiles represent a valuable energy source or a waste to be disposed of has been debated for three decades. The answer to this question has evolved over time, and will probably continue to change according to changes in national policies, the uranium market, isotopic enrichment costs and fission reactor technologies. In the 1970s and 1980s, the answer was clearly that depleted uranium is potentially a valuable energy source for the future. At that time, uranium prices were high, development of fast breeders was considered by many to be unavoidable within one or two decades and transformation of fertile ^{238}U into fissile plutonium was considered the appropriate answer to the lack of uranium. During the 1990s, the answer was less certain and the issue of depleted uranium management started to be controversial. Low uranium prices and the economic burden of tails management have altered the equation so that depleted uranium was more often considered to be a waste. At present, recent market trends and the perspectives of a nuclear renaissance are likely to change the conclusion again, and there is a renewed interest in depleted uranium re-enrichment.

3.7.3.2. Current uses of depleted uranium

Depleted uranium can be used as a raw material for fuelling reactors, assuming re-enrichment in the case of sufficient ^{235}U residual content, or mixing with other fissile material (e.g. Pu for MOX and HEU for dilution). Non-fuel uses in the reactor cores involve small amounts of depleted uranium, mainly for radiological shielding. Uses of depleted uranium for fuelling reactors include:

(a) *Re-enrichment*: From a purely economic point of view, depleted uranium can be reused as feed for a further enrichment step if the ratio between the enrichment unit cost and the natural uranium price allows such a recovery. This is currently the case for limited but growing quantities.
(b) *MOX matrices*: The quantities involved are small but still constitute about 94% of MOX heavy metal content.
(c) *HEU dilution*: The quantities of depleted uranium tails at present being used for dilution of HEU are reported to be significant as a result of the Russian HEU agreement. They are already counted in the HEU impact figures, and should be deducted from the tails stockpile totals.
(d) *Core blankets*: Pellets made with depleted uranium are quite often used peripherally to the reactor core as neutron shielding. Currently this use is very limited in LWRs and CANDUs (5–10 t/a). However, assuming the development of fast breeder programmes, this use can become very significant, multiplying the duration of uranium resources use by a factor of at least 50. It should be noted that the majority of the so-called Gen-4 list of future reactors belong to the fast breeder type.

Since depleted uranium storage does not represent a significant hazard when de-converted to a stable form such as U_3O_8, storage costs are likely to remain low, thus ensuring their availability for future needs. This is of particular interest for the tails already having a very low residual ^{235}U content, whether it is the result of a low initial tails assay or of a secondary process (tails of tails).

3.7.4. Existing stockpiles and flows of depleted uranium

A detailed updated estimate of the quantities of depleted uranium tails is provided in Table 9, the total being at approximately 1.5 million tons of depleted uranium at the end of 2004 (more than an OECD Nuclear Energy Agency estimate [11] and to be compared with a cumulative world uranium production exceeding two million tonnes of uranium). Table 9 also provides an estimate of the chemical form of these tails (depleted uranium hexafluoride

TABLE 9. DEPLETED URANIUM STOCKPILES ESTIMATE AT THE END OF 2004 *(updated from Ref. [1])*

Enricher	Total depleted uranium (t U)	U/DUF_6 (t U)	Estimate of U/UF_6 with U-235 content over 0.3%
USDOE–USEC	535 000	535 000	120 000
Eurodif	225 000	35 000	15 000
Urenco	61 000	61 000	40 000
BNFL	30 000	30 000	25 000
Rosatom	565 000	565 000	80 000
China	27 000	27 000	10 000
Other	15 000	15 000	10 000
Total	1 458 000	1 268 000	300 000

(DUF_6) versus defluorinated oxide forms) and the quantities that are believed to have a ^{235}U content in excess of 0.3% and are therefore considered as potential sources for our study.

Table 10 provides an estimate of the recoverable natural uranium equivalent content of tails having a ^{235}U content in excess of 0.3%. For the calculation, an average of 0.35% is utilized; we considered this to be a high figure. A secondary tails assay of 0.15% is considered; we consider this to be an economically achievable figure in the period up to 2025. The result shows that the readily available natural uranium equivalent content of tails stockpiles worldwide is limited compared with some published reports [11].

In addition, there is some uncertainty about the fate of part of the USDOE and USEC tails, in principle set to be defluorinated and sent to a repository. Some of these are currently identified for re-enrichment at the USEC plant [12].

Under the above assumptions (considered to be optimistic), the total could provide about 110 000 t of natural uranium equivalent, comparable to the resources of a large deposit, but would imply a significant SWU consumption.

3.7.4.1. Likely trend for depleted uranium use (2050 and beyond)

With the exception of limited quantities such as its use as MOX matrices or axial blankets in certain types of reactor, the uses of depleted uranium must be considered from now as an ‘on-line’ process. During the last decade,

TABLE 10. RECOVERABLE NATURAL URANIUM EQUIVALENT FROM 'RICH' TAILS

Enricher	Estimate of U/UF_6 with U-235 content over 0.3% (aver. 0.35)	Recoverable NatU (t U) at nw2 = 0.15%	kSWU required at nw2 = 0.15%
UDOE–USEC	120 000	43 200*	35 690
Eurodif	15 000	5 400	4 461
Urenco	40 000	14 400	11 897
BNFL	25 000	9 000	7 435
Rosatom	80 000	28 800	23 793
China	10 000	3 600	2 974
Other	10 000	3 600	2 974
Total	300 000	108 000	89 225

* According to Ref. [13] only about 15 500 t U are likely to be recovered from US tails, but this was evaluated before the increase in price of natural uranium.

stockpiled depleted uranium has been utilized for HEU blending down purposes in the Russian Federation. Owing to the limited availability of quantities having a high residual ^{235}U content, this use is declining and is being progressively replaced by the use of on-line tails and imported tails that are already fed to the enrichment cascades under normal commercial programmes.

Because of these practices, and of the low residual tails assays currently in use in the Russian Federation, the ^{235}U content of the world's depleted uranium stockpiles is permanently decreasing. In many places, it is also assumed that keeping these depleted uranium inventories as UF_6 is now worthless with regard to the schedule for their potential future use, and programmes are ongoing or planned to convert the inventories to an oxide form through defluorination, thus allowing safe and efficient long term storage.

In addition to the availability of ^{235}U rich-enough tails, commercial re-enrichment of depleted uranium depends mainly upon the availability of relatively low cost unutilized (marginal) SWU capacity.

Figure 8 illustrates some aspects of the economics of tails re-enrichment. At the level reached by the uranium market price in the middle of 2005, rich tails are clearly becoming attractive when enrichment capacity is available.

The US and West European gaseous diffusion plants have relatively high marginal SWU costs, but as there is free capacity at the US plant, limited re-enrichment operations are currently starting, owing to the changes in the

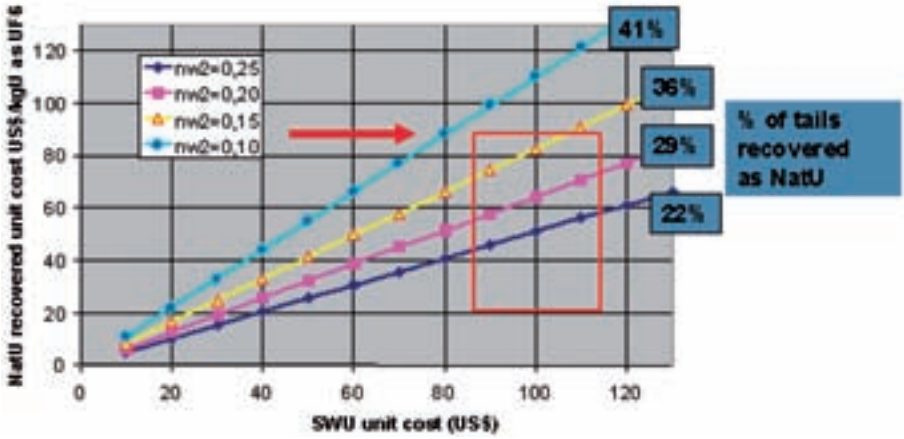

FIG. 8. The economics of tails re-enrichment of 0.35% ^{235}U tails to natural uranium for four secondary tails assays (nw2) versus SWU unit cost (assuming no feed related cost or credit).

uranium market. The capacities at West European centrifuge plants are currently fully committed to normal enrichment contracts. Future expansion of these plants could release some capacity for tails re-enrichment but they will be primarily devoted to normal enrichment operations, probably at lower tails assays.

In the near-term, the largest contribution to the re-enrichment of tails is coming from Russian centrifuge plants, which reportedly have large available marginal capacities, and thus can offer fuel contracts on a marginal cost basis. Total enrichment capacity exceeded 20 MSWU/a in 2005, and to supply Soviet design reactors, the Russian Federation has to supply fuels with a content of 4.4 MSWU if calculated for a 0.3% tails assay. Because the Russian enrichment plants are operated at tails assays below 0.15%, an additional SWU consumption of about 50% is needed, increasing the requirements to about 6.6 MSWU.

Table 11 projects the allocation of Russian enrichment capacity potentially available for tails re-enrichment.

TABLE 11. RUSSIAN ENRICHMENT CAPACITY AVAILABLE FOR TAILS RE-ENRICHMENT (kSWU/a)

	Year 2005	Year 2010
Supply from Soviet design reactors	6 600	8 000
Tails re-enrichment for HEU dilution	3 500	3 500
Russian LEU exports	3 600	4 500
Enrichment capacity	21 000	26 000
Available for tails re-enrichment	7 300	10 000

4. APPRAISAL OF IMPACT OF SECONDARY SUPPLIES ON THE URANIUM MARKET

In order to summarize the potential of all the secondary supplies identified in this paper, Table 12 gathers together the cumulative data and

TABLE 12. CUMULATIVE SECONDARY SUPPLIES FROM 2005 TO 2050 (REFERENCE CASE)

Secondary supply	NatU (kt)	Percentage of total	Percentage of requirements
Excess commercial stocks worldwide	40	3.6	0.9
USDOE commercial stocks	20	1.8	0.5
US + Russian HEU	56	5.0	1.3
Other HEU US + RF	40	3.6	0.9
MOX/weapons grade Pu US + RF	20	1.8	0.5
MOX/civilian Pu	150	13.5	3.5
RepU	134	12.1	3.1
Tails re-enrichment	110	9.9	2.5
Tails assays reduction effect	540	48.6	12.5
Total secondary supplies	1 110	100.0	25.7
Cumulative requirements	4 316		100.0
Mining production/RARs	3 067		71.1
Shortfall	139	12.5	3.2

compares them with the total requirements and expected production from mining of identified resources (Red Book reasonably assured resources (RARs)).

The largest contribution by far is likely to come from tails assays reduction. The second and third largest contributions come from recycled material flows, MOX and RepU from spent fuel reprocessing. Finally, the fourth largest supply is linked to tails re-enrichment, and the fifth to total HEU (US + Russian) already declared as excess. The later is likely to represent more, but probably not beyond a doubling of the quantities concerned.

On average during the period to 2050, the contribution of secondary supplies to the total supply needed is expected to be about 25%. This is obviously less than the current situation, but remains very significant.

5. CONCLUSION: A CHANGED PERCEPTION OF SECONDARY SUPPLIES

During the first civilian uranium 'boom' of the mid-1970s, secondary supplies were absent. At that time, the forecasts for nuclear power development were very high and the then known uranium resources relatively limited. As a result, spent fuel reprocessing and fissile material recycling were considered as the solution, along with fast breeder reactors.

Then a period of energy abundance, of energy savings and of lower than expected economic development in many areas of the world followed. In addition, two major events, the Three Mile Island accident and the Chernobyl accident, had a chilling effect on the development of nuclear power. As a result, the previously needed recycled fissile materials became 'secondary supplies', often accused of competing unfairly against fresh mining production in an increasingly difficult market. Progressively in this glutted market, huge uranium inventories were accumulated to levels of hundreds of thousands of tons, leading to the buildup of the largest single source of secondary supplies: commercial inventories.

Almost in coincidence with the turn of the century, and departing from the previous period, the global energy supply scene has greatly changed, along with global warming and other environmental concerns. Amongst the solutions to the new challenges, a renaissance of nuclear power appears to be needed to an increasing number of decision makers.

This brief history allows us to consider there to be two eras for the impact of secondary supplies on the market:

(1) A first era with a large contribution from secondary supplies having a strong suppressive effect on market prices. This era is currently vanishing in a deeply and rapidly changing market.
(2) A second era is likely to develop from now onwards. Uranium savings will be achieved through lower enrichment tails assays. Additional excess military fissile material will be made available for civilian use. Finally, recycled fissile material will again be seen as one of the responses to higher uranium prices. In the short to medium term, these supplies will be constrained by available plant capacities and other technical factors. Ultimately, increased secondary supplies from recycling are likely to appear in the case of the growth of fissile material requirements and continued tension about uranium. Tension can result from delays in finding and starting new mines, especially from 'speculative resources'. Recycled material from spent fuels can be considered as a type of insurance, because, regardless of the economic and administrative issues, its availability is constrained only by the construction of new factories.

For the future, using the energy potential of all types of available fissile and fertile materials, including spent fuel, is certainly better than burying them, not to mention the advantages related to waste management issues.

It can be concluded that all secondary supplies will help to make possible the required nuclear renaissance. In the short term they will bridge the gap between mining production and consumption, in the longer term they will contribute to supplies, especially in the case of accelerated nuclear power development scenarios, through the optimized use of fissile and fertile materials.

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WORLD NUCLEAR CAPACITY PROJECTIONS UP TO 2050

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Abstract

Nuclear power is a proven technology which has maintained a steady 16% contribution to world electricity supply for more than a decade. Nuclear's constant share of global generation has been maintained not so much through new capacity as through substantially improved load factors, the result of improved management enforced by competitive markets, and capacity upratings often in connection with licence renewals. However, the future prospects of nuclear power have changed notably since 2000. First, the enormous demand for energy and electricity to fuel economic development in populous developing countries has until recently been underrated. Second, the period of excess generating capacity in several OECD countries is nearing an end. Third, the entry into force of the Kyoto Protocol has begun to affect the economics of many utilities because it places a value on the environmental benefits of nuclear power. Fourth, energy supply security and reliability are again important concerns in many countries. Fifth, while fossil fuel price volatility is normal, the recent elevated price levels are unlikely to return to previous levels given current demand prospects and the geographic distribution of oil and gas fields. The medium and long term nuclear capacity projections periodically produced by the IAEA are not predictions but rather scenarios of plausible ranges of future nuclear power generation capacities reflecting a variety of economic, social and environmental driving factors. Both the low and high projections are derived from a bottom-up approach based on a review of nuclear power programmes and the plans of IAEA Member States up to 2030. The projections until 2050 are extrapolations of the IAEA low and high projections based on the growth dynamics of nuclear power for a set of long term energy scenarios developed by the Intergovernmental Panel on Climate Change (IPCC).

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1. TRENDS IN ENERGY AND ELECTRICITY DEVELOPMENT

Energy and electricity are essential for economic development and enhancement of the quality of life. Energy provides services, such as heating, cooling, lighting, information exchange and motive power, to which the world's population aspires. Figure 1 presents the historical pattern of development of the global primary energy mix. During the period 1980–2004, the growth rate of world energy demand averaged about 2% per annum, from 274 to 451 EJ [1]; electricity demand grew consistently faster than total primary energy, at a rate of 3% per annum (Fig. 2) [2]. Over the last three decades, liquids have dominated in the total primary energy mix with a share of about 35%, followed by solids (essentially coal) and gases with 27 and 23%, respectively. The share of nuclear showed steady growth and is now fifth largest, at about 6%.

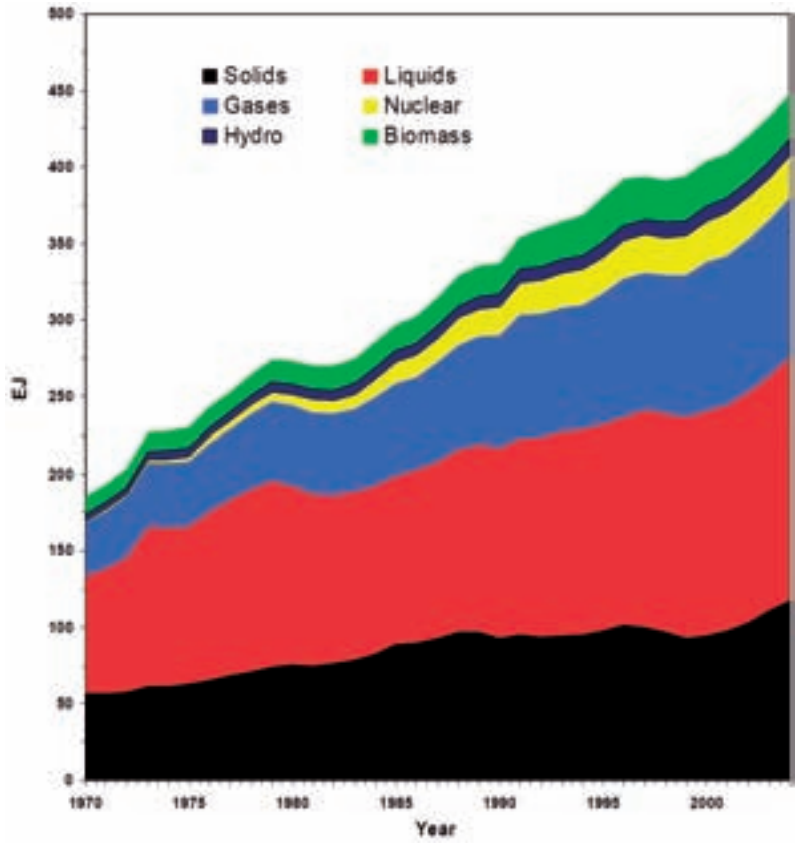

FIG. 1. Global primary energy mix, 1970–2004. Source: IAEA [1].

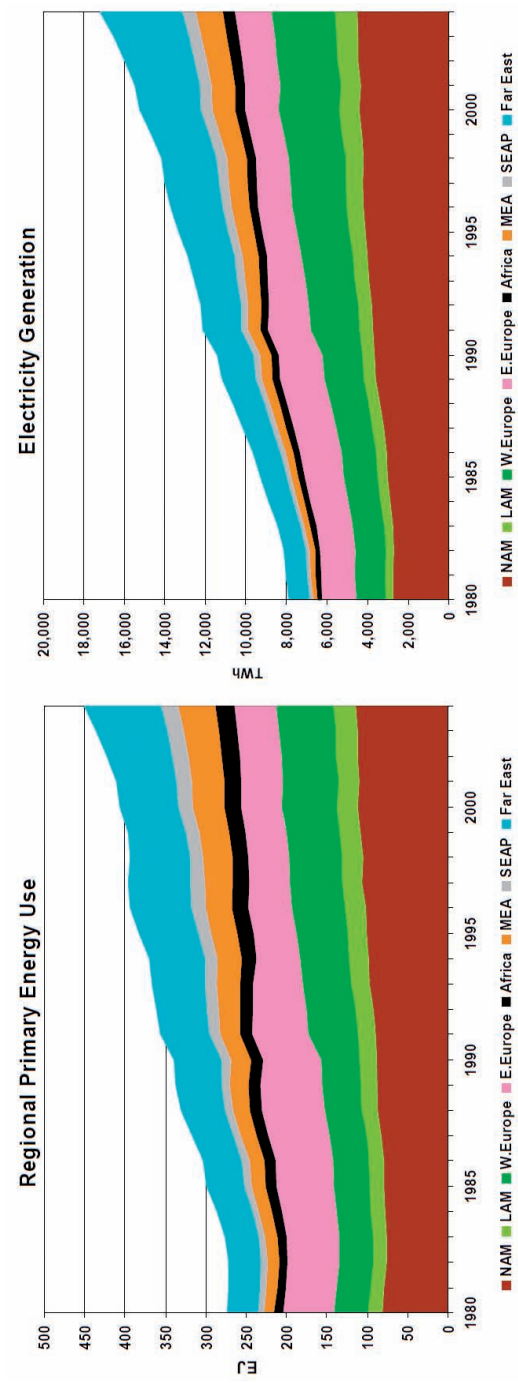

FIG. 2. World total energy consumption and electricity generation by region, 1980–2004. Source: IAEA [1].

By region, North America remains the largest energy consumer, accounting for about 25% of total energy demand (Fig. 2), while Eastern Europe has shown a decreasing share since the 1990s. Strong growth has occurred in the Far East, the Middle East and South Asia. Future energy demand will depend on a number of factors but population growth, economic development and technology change will continue to be the main driving forces for continued growth in demand, even though changes in lifestyles and technological progress may eventually lead to less energy intensive societies. However, the sheer number of persons (currently 1.6 billion) not at present having access to electricity will translate into growing electricity needs regardless of energy efficiency and energy intensity improvements.

As shown in Fig. 3, per capita electricity generation in the world increased from 1.7 MW·h in 1980 to 2.6 MW·h in 2004, with most of the growth occurring in the developed countries of North America and Western Europe [1]. However, developing countries will account for 95% of the world's population growth between 1993 and 2030, and by 2030 more than 85% of the world's population will be in the developing world [3], where the average electricity consumption per capita is now less than one tenth of that in the developed industrialized countries (Fig. 4). Such a rapid population expansion in developing countries, combined with urbanization and the need to foster economic and industrial development, will lead to a drastic increase in energy and electricity supply. Technical breakthroughs and innovations are expected to improve the efficiency of end use appliances, reducing the supply requirements for delivering the same quality of energy services. However, such productivity gains are not infinite and structural changes will have only a limited impact on global demand.

A range of electricity generation technologies based on fossil fuel combustion, renewables and nuclear power, is available or under development. Factors that will influence strategies for expansion of electricity systems include economics, security of supply and environmental impacts. An increasing awareness of the potential health and environmental impacts of human activities has already induced development of cleaner and more efficient technologies. Abatement measures and increased efficiency of power plants have substantially reduced emissions and other residuals from fossil fuel chains. Unless carbon capture and storage from the use of fossil fuels is technically feasible on a substantial scale and commercially viable, environmental sustainability, especially significant reductions of greenhouse gas (GHG) emissions, can only be achieved by a broader deployment of non-fossil energy sources for electricity generation. Most renewable energy sources other than hydropower still require considerable research and development before they will be ready

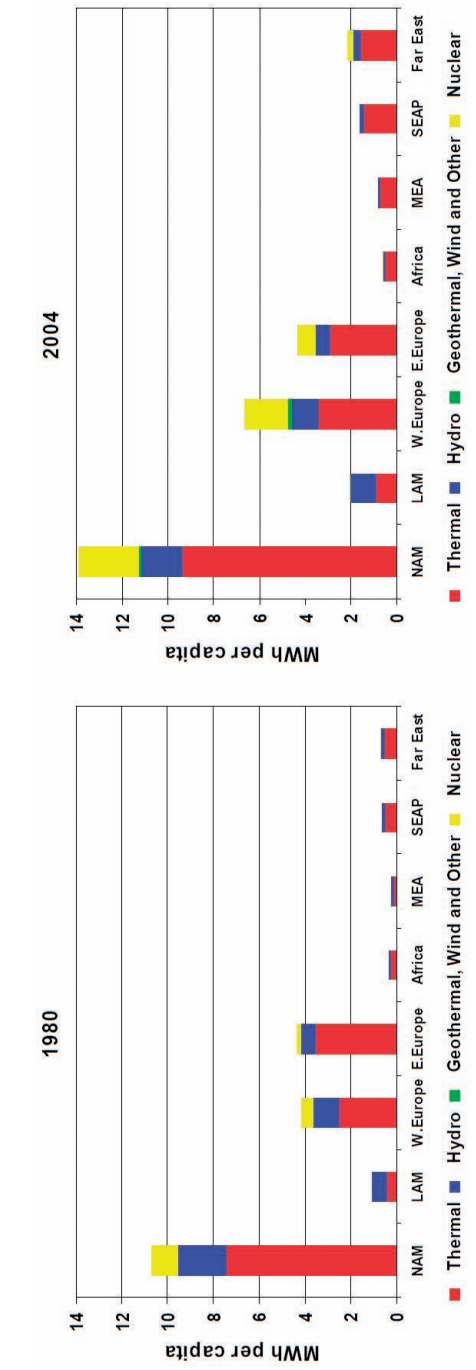

FIG. 3. Per capita electricity generation by region and fuel, 1980 and 2004. Source: IAEA [1].

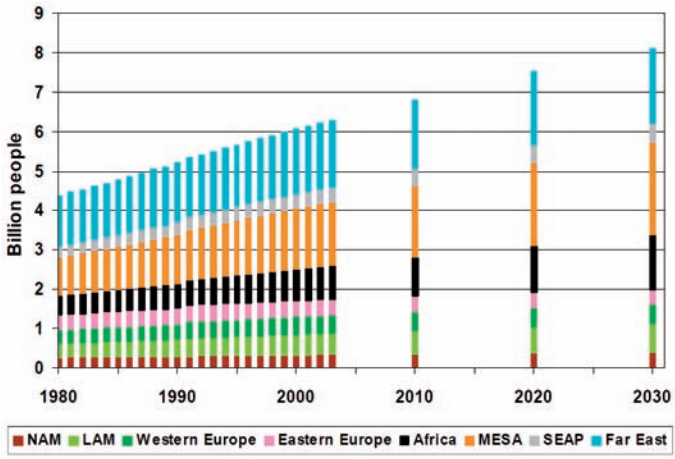

FIG. 4. World population growth by region, 1980–2030. Source: United Nations [3].

for large scale deployment or be economically competitive for baseload electricity generation. Therefore, nuclear power is the only non-fossil technology that can be deployed, in the short term, on a sufficiently larger scale to achieve successful implementation of environmentally benign and competitive electricity supplies (Fig. 5) [4].

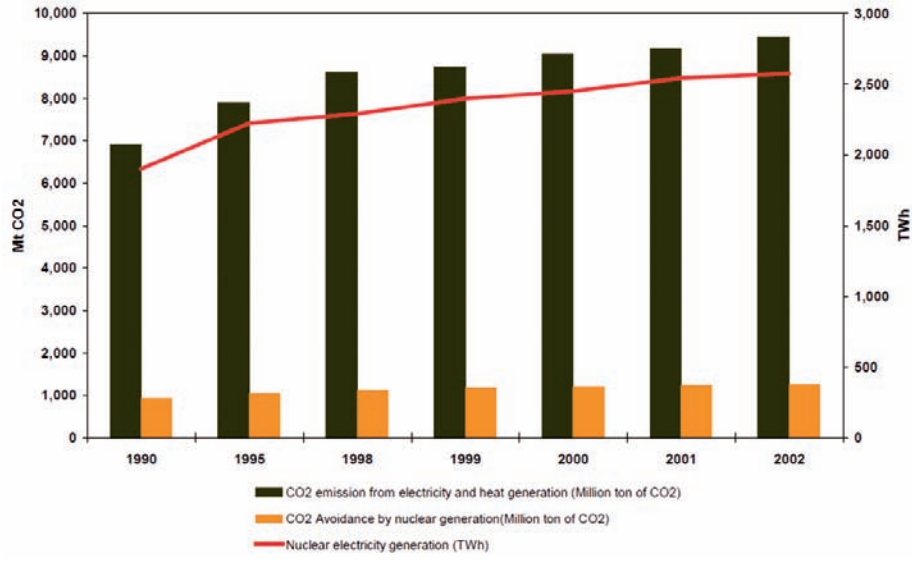

FIG. 5. Estimates of CO_2 emissions avoided by nuclear electricity generation, 1980–2002. Source: OECD International Energy Agency (OECD/IEA) [4].

2. NUCLEAR DEVELOPMENT TRENDS

According to data in the IAEA's Power Reactor Information System (PRIS), at the end of 2004 there were 440 nuclear power reactors in operation, with a total installed nuclear power generating capacity of 367 496 MW(e), as shown in Table 1 [2]. There were also 26 power reactors under construction, with a total generating capacity of 20 826 MW(e). Worldwide in 2004, 33 countries were operating or building nuclear power plants (NPPs), with an accumulated total nuclear reactor operating experience of more than 11 500 reactor-years [5].

Figure 6 shows the growth of nuclear electricity generation since 1970. There was rapid growth in the 1970s, with an expansion of almost ninefold to reach 692 TW · h by 1980. By then, nuclear power contributed 8.4% of total world electricity [1]. Growth then slowed in the early 1980s because of lower electricity demand due to slower economic growth and because of cost escalation due to high inflation.

Since 1986, average nuclear production growth has kept pace with average annual global electricity growth of about 2.5%. This modest growth has kept nuclear power's share of global electricity production at a stable 16%, i.e. in the period 1990–2004, nuclear electricity generation increased from 1735 to 2619 TW · h [1].

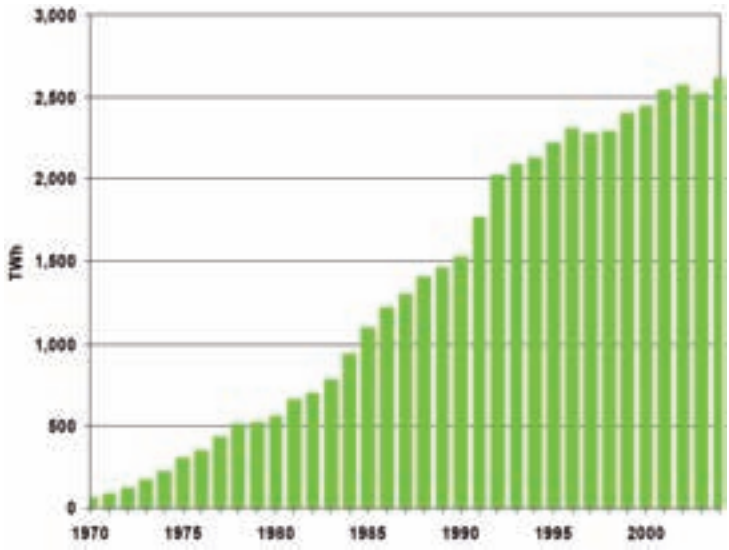

FIG. 6. World total nuclear electricity generation, 1970–2004 (TW · h). Source: IAEA [1].

TABLE 1. NUCLEAR POWER REACTORS IN THE WORLD (END OF 2004)
(source: IAEA [2])

Group and country	In operation		Under construction		Electricity supplied by nuclear power reactors in 2004	
	Number of units	Total MW(e)	Number of units	Total MW(e)	TW·h	Total electricity (%)
North America						
Canada	17	12 113			85.3	15.0
United States of America	104	99 210			788.6	19.9
Latin America						
Argentina	2	935	1	692	7.3	8.2
Brazil	2	1 901			11.5	3.0
Mexico	2	1 310			10.6	5.2
Western Europe						
Belgium	7	5 801			44.9	55.1
Finland	4	2 656			21.8	26.6
France	59	63 363			426.8	78.1
Germany	18	20 679			158.4	32.1
Netherlands	1	449			3.6	3.8
Spain	9	7 585			60.9	22.9
Sweden	11	9 469			75.0	51.8
Switzerland	5	3 220			25.4	40.0
United Kingdom	23	11 852			73.7	19.4
Eastern Europe						
Armenia	1	376			2.2	38.8
Bulgaria	4	2 722			15.6	41.6
Czech Republic	6	3 548			26.3	31.2
Hungary	4	1 755			11.2	33.8
Lithuania[a]	2	2 370			13.9	72.1
Romania	1	655	1	655	5.1	10.1
Russian Federation	31	21 743	4	3 775	133.0	15.6

TABLE 1. NUCLEAR POWER REACTORS IN THE WORLD (END OF 2004) (cont.)
(source: IAEA [2])

Group and country	In operation		Under construction		Electricity supplied by nuclear power reactors in 2004	
	Number of units	Total MW(e)	Number of units	Total MW(e)	TW·h	Total electricity (%)
Slovakia	6	2 442			15.6	55.2
Slovenia	1	656			5.2	38.8
Ukraine	15	13 107	2	1 900	81.8	51.1
Africa						
South Africa	2	1 800			14.3	6.6
Middle East and South Asia						
India	14	2 550	9	4 092	15.0	2.8
Iran, Islamic Republic of			1	915	0.0	
Pakistan	2	425			1.9	2.4
Far East						
China	9	6 602	2	2 000	47.8	2.2
Japan	54	45 468	3	3 237	273.8	29.3
Korea, Republic of	19	15 850	1	960	124.0	37.9
World total[b]	441	367 496	26	20 826	2 619	16.0

[a] Including the following data in Taiwan, China:
— Six units in operation with a total capacity of 4884 MW(e) and two units under construction with a total capacity of 2600 MW(e);
— 37.9 TW · h of nuclear electricity generation, representing 20.9% of the total electricity generated.
[b] One unit was shut down on 31 December 2004.

Figure 7 shows the annual incremental capacity changes and the annual electricity production for the period 1966–2004 [1]. The peak of nuclear power expansion occurred in the mid-1980s; thereafter new construction of plants slowed down to almost a standstill by the end of the 1990s. At the beginning of

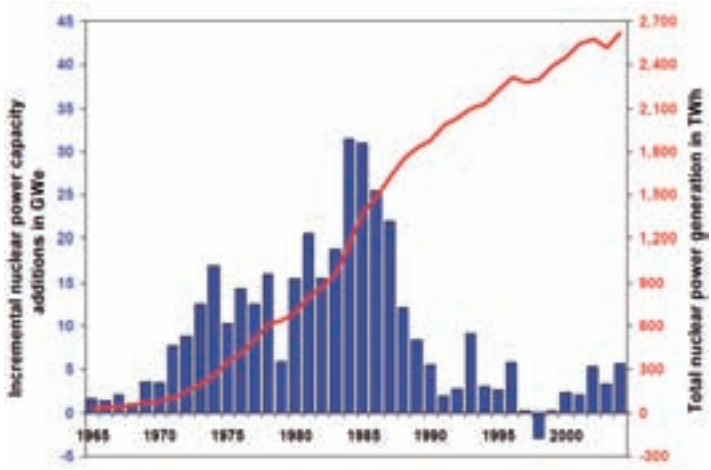

FIG. 7. Incremental changes in nuclear capacities and total nuclear electricity generation. Source: IAEA [1].

the new century, capacity addition picked up momentum again — almost exclusively in Asia. Twenty-two of the last 31 NPPs connected to the world's energy grid have been built in Asia, driven by the pressures of economic growth, scarcity of natural resources and population increase. Of the new NPPs at present under construction, 16 of the 26 are located in Asia, while construction has virtually halted in Western European and North American countries with long-standing nuclear power programmes [5]. Figure 7 also depicts the continued growth in nuclear generation after the late 1980s despite the fall in construction of new plants. Here competitive pressures brought about by market liberalization and an end to the 'cost plus' mentality in many countries resulted in improved plant availability. In fact, the increase in the availability of the global fleet of reactors translates into the virtual construction of more than thirty 1000 MW NPPs.

Figure 8 shows the nuclear share of electricity generation by country [2]. Worldwide, 16 out of 31 countries (including Taiwan, China) rely upon NPPs to supply at least 26% of their total electricity needs. France has the largest share of nuclear electricity, with about 78% of its total production, followed by Lithuania with about 72% and Slovakia with about 55%. Eastern European countries show a high dependence on nuclear power generation; with the exception of the Russian Federation and Romania, which produce 15.6 and 10.1%, respectively, of their electricity from NPPs, the countries in the region have a nuclear share of electricity generation higher than 30%. Slovakia has the highest share, with 55.2% of its electricity generated by nuclear reactors, followed by Ukraine with 51.1%, Bulgaria with 41.6%, Slovenia with 38.8%,

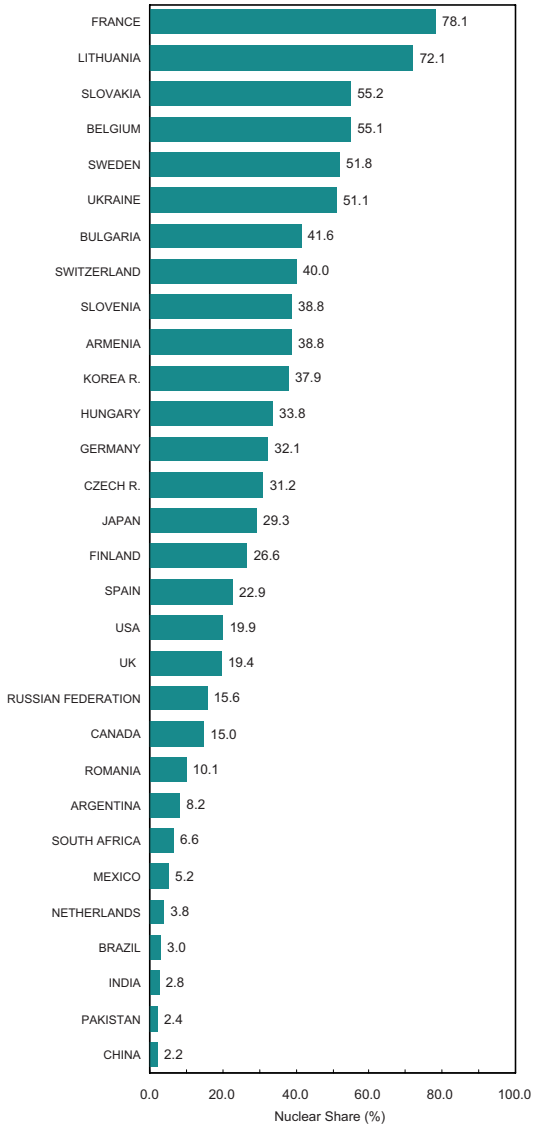

FIG. 8. Nuclear share of total electricity generation worldwide in 2004. Source: IAEA [1].

Armenia with 38.8%, Hungary with 33.8% and the Czech Republic with 31.2%.

In the short term, the prospects for nuclear power development are fairly clear, since the units ordered by the turn of the century are already under construction. Most of the additional nuclear capacity will be brought into

operation in Asia and in Eastern Europe. Nuclear power programmes have slowed down in many countries in Western Europe and North America over recent years, where lower than expected growth of demand has resulted in limited need for additional baseload capacity, and because of public concern about nuclear safety and radioactive waste disposal (Fig. 9). In the other regions of the world, nuclear power will remain a relatively small contributor to electricity supply (Table 2) [5].

In the medium and long term, a broad range of factors will influence the prospects of nuclear power. Above all, economics will be the decisive factor (ultimately everything can be translated into costs). The technologies for reactors and fuel cycle facilities, as well as fuel resources, are available to sustain large scale deployment of nuclear power worldwide. The barriers to nuclear power development other than its high initial investment costs are more institutional and organizational than technological. The main prerequisites for the revival of nuclear power are the alleviation of public and political concerns about safety and waste management, proliferation and physical security, and the establishment of adequate mechanisms for technology adaptation and transfer, as well as financing to facilitate the implementation of nuclear power programmes in developing countries where the need for electricity is greatest.

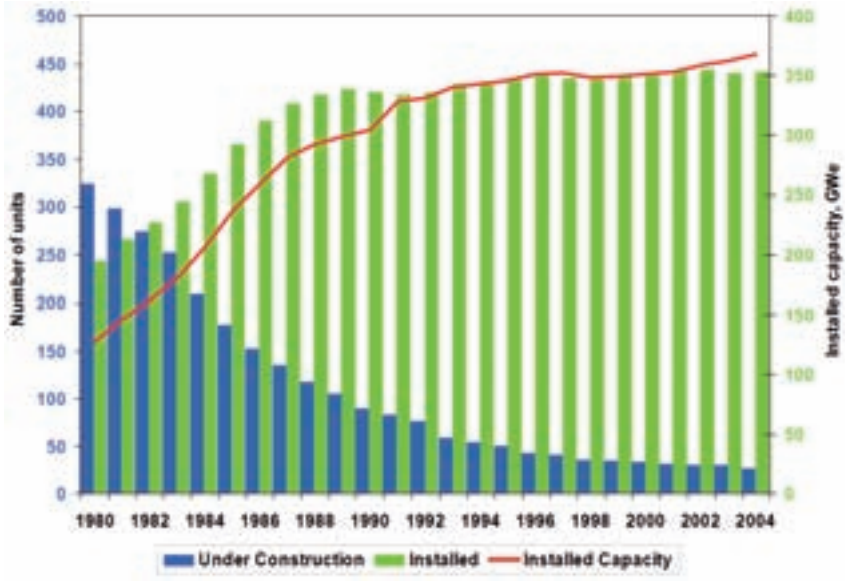

FIG. 9. Worldwide installed nuclear capacity and the number of nuclear reactors that are operational or under construction. Source: IAEA [1].

TABLE 2. NUCLEAR POWER REACTORS IN OPERATION AND UNDER CONSTRUCTION WORLDWIDE (AS OF 31 DEC. 2004)
(Source: IAEA [2])

Country	Reactors in operation		Reactors under construction		Nuclear electricity supplied in 2002		Total operating experience to Dec. 2003	
	No of units	Total MW(e)	No of units	Total MW(e)	TW · h	Per cent of total	Years	Months
Argentina	2	935	1	692	7.31	8.24	52	7
Armenia	1	376			2.21	38.83	37	3
Belgium	7	5 801			44.86	55.13	198	7
Brazil	2	1 901			11.54	3.00	27	3
Bulgaria	4	2 722			15.60	41.58	133	3
Canada	17	12 113			85.27	15.02	509	7
China	9	6 602	2	2 000	47.80	2.19	47	11
Czech Republic	6	3 548			26.32	31.22	80	10
Finland	4	2 656			21.78	26.59	103	4
France	59	63 363			426.80	78.08	1 405	2
Germany	18	20 679			158.39	32.11	666	0
Hungary	4	1 755			11.21	33.83	78	2
India	14	2 550	9	4 092	15.04	2.82	237	5

TABLE 2. NUCLEAR POWER REACTORS IN OPERATION AND UNDER CONSTRUCTION WORLDWIDE (AS OF 31 DEC. 2004) (cont.)
(Source: IAEA [2])

Country	Reactors in operation		Reactors under construction		Nuclear electricity supplied in 2002		Total operating experience to Dec. 2003	
	No of units	Total MW(e)	No of units	Total MW(e)	TW · h	Per cent of total	Years	Months
Iran, Islamic Republic of			1	915			0	0
Japan	54	45 468	3	3 237	273.81	29.31	1 176	4
Korea, Republic of	19	15 850	1	960	123.97	37.95	239	8
Lithuania	1	1 185			13.92	72.11	38	6
Mexico	2	1 310			10.58	5.20	25	11
Netherlands	1	449			3.61	3.79	60	0
Pakistan	2	425			1.93	2.37	37	10
Romania	1	655	1	655	5.14	10.08	8	6
Russian Federation	31	21 743	4	3 775	133.02	15.61	791	5
Slovakia	2	1 800			14.28	6.61	40	3
Slovenia	6	2 442			15.62	55.18	106	6
South Africa	1	656			5.20	38.85	23	3
Spain	9	7 585			60.89	22.86	228	2

TABLE 2. NUCLEAR POWER REACTORS IN OPERATION AND UNDER CONSTRUCTION WORLDWIDE (AS OF 31 DEC. 2004) (cont.)
(Source: IAEA [2])

Country	Reactors in operation		Reactors under construction		Nuclear electricity supplied in 2002		Total operating experience to Dec. 2003	
	No of units	Total MW(e)	No of units	Total MW(e)	TW · h	Per cent of total	Years	Months
Sweden	11	9 451			75.04	51.82	322	1
Switzerland	5	3 220			25.43	40.04	148	10
Ukraine	23	11 852			73.68	19.43	1 354	8
United Kingdom	15	13 107	2	1 900	81.81	51.11	293	6
United States of America	104	99 210			788.56	19.95	2 975	8
Total	440	366 293	26	20 826	2 618.56		11 588	6

Note: The total includes the following data in Taiwan, China:
— Six units, 4884 MW(e) in operation; 2 units, under construction;
— 33.94 TW · h of nuclear electricity generation, representing 20.93% of the total electricity generated in 2004;
— 140 years and 1 month of total operating experience.

The NPPs of the present generation already ensure a high level of safety, based upon built-in redundancy and reliance on decades of experience with proven technology and engineering. Advanced reactors are being designed and developed with new approaches to address the challenge of increasingly demanding safety requirements by, inter alia, utilizing more passive safety systems in order to reduce probabilistic risks of accident. Furthermore, a consensus on international practices and standards for nuclear safety has emerged aimed at achieving and maintaining a high level of nuclear safety worldwide through national measures and international cooperation. All these positive elements plus the successful safe and economic operation of the current fleet of NPPs are the ingredients for future public and political acceptance of nuclear power.

The cost of electricity generation will remain a cornerstone for the assessment and choice of options in electricity system expansion strategies. Recent studies show that, in many countries, nuclear power is among the cheapest sources for baseload electricity generation, especially where solid fuels are not accessible at low production costs. However, the high initial cost presents an economic risk in a liberalized market environment. Designers of advanced NPPs, therefore, aim to reduce capital costs through streamlining reactor systems and reducing the amount of material required for their construction. Investment costs may also be reduced by shortening construction lead times through the use of more components prefabricated off-site. Rising costs of non-renewable natural resources and the enforcement of more stringent atmospheric pollution standards may also increase the cost of electricity generated by fossil fuelled power plants, thereby making nuclear power even more attractive economically. Nonetheless, financing the high capital costs of nuclear power will remain a key issue, especially in developing countries contemplating the implementation of nuclear programmes and also in energy markets that favour short term maximization of share holder value.

3. THE PROSPECTS OF NUCLEAR POWER

Projecting nuclear power development in the medium and long term is a difficult exercise, because a large number of driving factors cannot be assessed with any high degree of certainty. The scenarios developed by the IAEA are not meant to be a prediction of the likely evolution of nuclear power generation but rather are intended only to illustrate some plausible future possibilities. The medium term scenarios, up to 2030, are derived from a bottom-up approach based upon a review of nuclear power programmes and plans in Member States of the IAEA. The low and high estimates of nuclear

capacity (Table 3) correspond to a set of contrasting but not extreme assumptions with regard to the parameters which will influence the implementation of nuclear programmes.

The low estimates are based on assumptions that reflect a continuation of present trends: public opposition in some countries, low economic growth in industrialized countries, institutional and sociopolitical uncertainties in economies in transition and lack of funding in developing countries. In this case, the nuclear units under construction will be completed but only those countries where nuclear programmes are already firmly committed to will continue to order new units. In some countries, nuclear units will not be replaced at the end of their lifetimes and the total installed nuclear capacity in these countries will decrease before 2030. The projected nuclear capacity worldwide, in the low case, would be about 418 GW(e) in 2030, and the share of nuclear power in electricity generation worldwide would be 13%.

The high estimates assume a moderate revival of nuclear power development that could occur in the light of a more comprehensive assessment of the macroeconomic and environmental aspects of the different options available for electricity generation. This revival is assumed to occur mainly in Western Europe and to a lesser extent in North America. In Eastern Europe, nuclear power programmes are assumed to be implemented according to the present plans. In the Far East, nuclear power is assumed to be developed in line with rapid growth of electricity demand. In the high case, total nuclear capacity is projected to reach about 640 GW(e) in 2030, which will allow the share of nuclear power in electricity generation to be just about 12% [2].

In both sets of projections, nuclear growth starts to lag behind global electricity growth. Therefore, nuclear power's share of global electricity decreases. In the latest low projection it drops from the current 16% to 13% in 2030. In the high projection it drops further, to 12%. That may seem counter-intuitive. What is happening is that, in the high projection, not only is nuclear electricity growing faster than in the low projection but so also is overall electricity use. In fact, overall electricity use is growing sufficiently fast that the differential between it and nuclear power is growing more than in the low projection, and thus the nuclear share drops more.

Assuming a nuclear revival is not unrealistic; 2004 already showed several promising developments, including upward revisions in specific expansion plans and actions in a number of countries, consequently higher medium term nuclear projections, increased media attention to the potential benefits of nuclear power including its very low GHG emissions, and more favourable ratings for nuclear power in a number of public opinion polls. It should be noted, however, that while the entry into force of the Kyoto Protocol could be

TABLE 3. ESTIMATES OF TOTAL ELECTRICITY CAPACITY AND THE CONTRIBUTION OF NUCLEAR POWER
(Source: IAEA [2])

Country group		2004			2010[a]			2020[a]			2030[a]		
		Total elect. GW(e)	Nuclear		Total elect. GW(e)	Nuclear		Total elect. GW(e)	Nuclear		Total elect. GW(e)	Nuclear	
			GW(e)	%		GW(e)	%		GW(e)	%		GW(e)	%
North America		1055	111.3	10.6	1099	116	11	1194	118	10	1318	115	8.7
					1155	117	10	1279	128	10	1422	145	10
Latin America		264	4.1	1.6	303	4.1	1.4	383	6.1	1.6	483	5.8	1.2
					350	4.1	1.2	543	6.1	1.1	828	15	1.8
Western Europe		724	125.1	17.3	762	119	16	842	97	11	940	79	8.5
					816	125	15	951	130	14	1118	145	13
Eastern Europe		466	49.4	10.6	469	48	10	505	64	13	543	66	12
					496	51	10	605	78	13	736	97	13
Africa		105	1.8	1.7	115	1.8	1.6	143	2.1	1.5	181	2.1	1.2
					135	1.8	1.3	207	4.1	2.0	316	9.3	3.0
Middle East and South Asia		284	3.0	1.0	331	9	2.8	430	15	3.6	556	18	3.2
					370	10	2.8	555	27	4.9	811	43	5.3
South East Asia and the Pacific		143			169			213	0.9	0.4	264	0.9	0.3
					184			270	0.9	0.3	391	3.0	0.8
Far East		651	72.8	11.2	685	82	12	804	113	14	937	131	14
					840	85	10	1167	142	12	1589	183	11
World total	Low estimate	3693	367.5	10.0	3934	380	10	4515	416	9.2	5223	418	8.0
	High estimate				4347	395	9.1	5576	516	9.3	7210	640	8.9

[a] Nuclear capacity estimates take into account the scheduled decommissioning of the older units at the end of their lives.

important for the future development of nuclear power, its immediate impacts on nuclear power are indirect, and significant impacts are uncertain and long term.

Most straightforward were the upward revisions in new near-term nuclear projections released in 2005 by the IAEA [1]. The low projection was adjusted upwards for the fourth year in a row, reflecting an increasingly optimistic outlook for nuclear. It now projects 416 GW(e) of nuclear capacity in 2020, the equivalent of 116 more 1000 MW(e) nuclear plants than projected just four years earlier. In the high projection there has been less change, and a less consistent pattern of change, from year to year. However, the overall pattern is consistent with an industry with good prospects. The list of reasonable medium term projects at the high end is fairly stable, and each year more of them are promoted from promising prospects to actual projects being developed. Figures 10 and 11 show historical nuclear capacity growth together with the evolution of the IAEA projections in the last four years. The IAEA's latest high projection shows an 82% increase in nuclear electricity production between 2004 and 2030.

For a regional breakdown of the low projection until 2030, the two most distinctive features are the contraction of nuclear capacity in Western Europe and the expansion in the Far East (Fig. 12). The capacity in the Middle East and

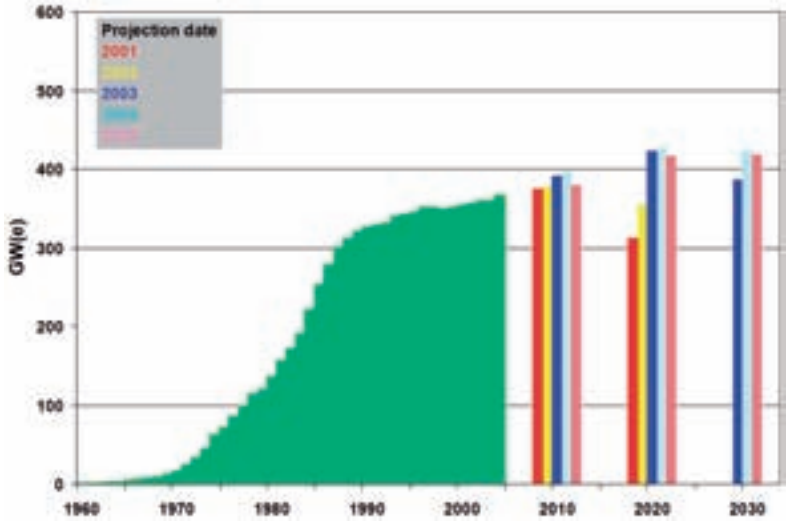

FIG. 10. Historical growth in world nuclear capacity and the IAEA's low projections for the years 2000–2005.

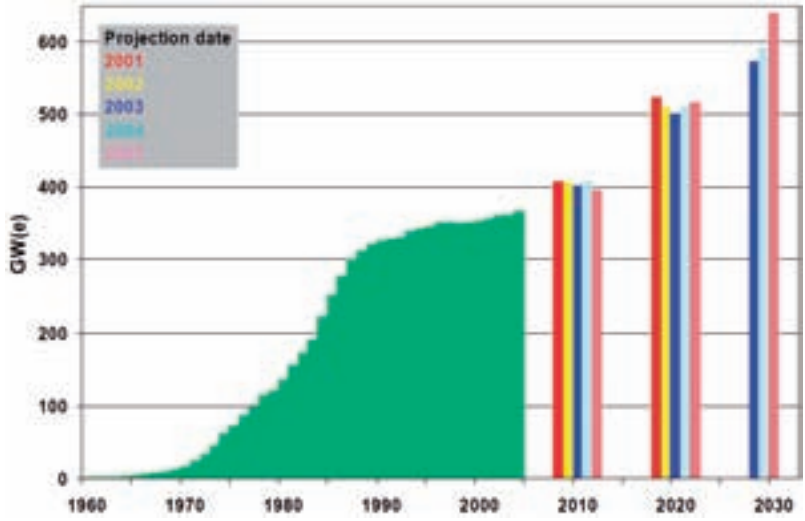

FIG. 11. Historical growth in world nuclear capacity and the IAEA's high projections for the years 2000–2005.

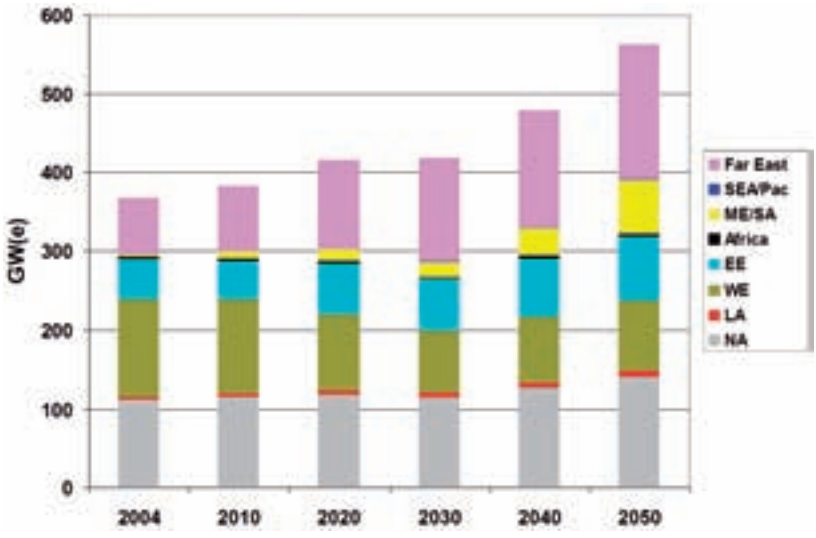

FIG. 12. Regional distribution of world nuclear capacity in the IAEA's low projection until 2030 when extended to 2050 using approximately one third of the median nuclear expansion rates of the Special Report on Emissions Scenarios (SRES) of the Intergovernmental Panel on Climate Change (IPCC) (NA: North America; LA: Latin America; WE: Western Europe; EE: Eastern Europe; ME/SA: Middle East and South Asia; SEA/Pac: South East Asia and Pacific).

South Asia region actually grows by a factor of six in this case, although from a small base. There is some small net growth in Eastern Europe and basically no change in North America.

In the high projection until 2030 (Fig. 13), there is capacity growth in all regions, but the Far East still leads with 100 GW(e) of net new capacity by 2030 (by 'net' we mean capacity additions beyond any construction to replace retiring nuclear plants with new nuclear plants). By 2030, 45% of the world's additional capacity will be in the Far East. While the Far East leads in net capacity additions, the Middle East and South Asia region has the most impressive growth rate, adding 31 GW(e) to increase capacity by a factor of ten, equal to average growth of 9% per year. Eastern Europe adds 40 GW(e) net. More salient national developments are summarized below.

The Chinese Government has authorized 7 GW(e) of new capacity, including the construction of four more units, bringing the total number of authorized units up to nine, with two additional units already under construction. The Government plans to raise total installed nuclear electricity generating capacity from the current 6587 MW(e) to between 32 000 and 40 000 MW(e) by 2020.

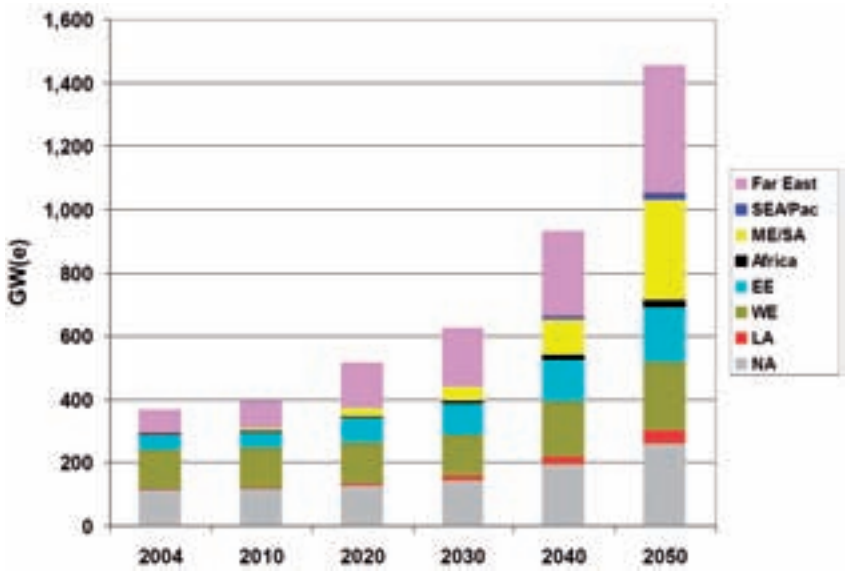

FIG. 13. Regional distribution of world nuclear capacity in the IAEA's high projection until 2030 when extended to 2050 using approximately one third of the median nuclear expansion rates of the IPCC's SRES (NA: North America; LA: Latin America; WE: Western Europe; EE: Eastern Europe; ME/SA: Middle East and South Asia; SEA/PAC: South East Asia and Pacific).

India, which has the most new nuclear construction of any country in the world (nine plants), plans a 100-fold increase in nuclear capacity by mid-century, and an increase from 3% of electricity generation to more than 25%. A 100-fold increase seems enormous, but works out at an average of 9.2% per year. This is well below the pace of global nuclear capacity growth in the 1970s of 21%, but above the average in the 1980s of 8.7%. It is comparable to the 33 year global average growth rate from 1970 to 2003 of 9.4%.

The prospects for expansion of nuclear power also appear to be gaining momentum in other parts of the world, although less dramatically than in China and India. The Russian Federation has two more plants under construction and plans to more than double capacity from the current 22 GW(e) to 53 GW(e) in 2020. The new European Union accession countries and other Eastern European countries with nuclear power have shown a clear determination to retain and expand the nuclear option. Even in Poland, where nuclear development was halted by a parliamentary decision in 1990, the Council of Ministers approved a draft energy policy in early 2005 that explicitly includes nuclear power.

In Western Europe excavation work began in 2004 for Olkiluoto 3 in Finland, a European pressurized water reactor (EPR) that will be the first construction start in the region since 1991. In France, Electricité de France selected a site for a demonstration EPR, with construction expected to begin in 2007.

In the United States of America (USA), the Nuclear Regulatory Commission approved seven more licence extensions of 20 years each (for a total licensed life of 60 years for each plant), bringing the total number of approved licence extensions to 30 by the end of 2004. Some three quarters of the 104 US NPPs have either received, applied for or stated their intention to apply for such licence extensions. The possibility of new construction was further enhanced when the US Department of Energy (USDOE) approved financial assistance to two industrial consortia for NPP licensing demonstration projects, taking advantage of the NRC's new combined construction and operating licence procedure.

Public opinion polls in 2004 also appear to have shifted positively for nuclear power, although details are different in different countries. In Finland, the only Western European country with new nuclear construction, a November 2004 poll showed 46% supporting the use of nuclear energy, compared with only 36% in 2002. Twenty-five per cent of respondents had a negative opinion and 29% were neutral. In Sweden, which is phasing out nuclear power and will close its second reactor in accordance with that policy in 2005, an October 2004 poll showed support for nuclear power at 82%, up slightly from previous years. The proportion of respondents supporting an

expansion of nuclear power, not just the replacement of existing reactors, increased markedly to 21%. An October 2004 poll in the USA found a new record high of 67% of Americans in favour of the use of nuclear energy, compared with 26% of respondents who opposed the use of nuclear energy and 7% who had no opinion. The percentage of respondents who thought the USA should 'definitely' build new nuclear plants in the future increased to 60%, compared with 54% in the most recent poll earlier in 2004.

Most long term studies of global energy demand and supply consistently project an increasing share of nuclear energy, especially for the period after 2030, irrespective of the presence of policies targeted at climate protection. Short term energy outlooks usually show a much lower share of nuclear electricity. With the entry into force of the Kyoto Protocol, short term projections have already been revised upwards.

However, even with upward revisions in short and medium term projections, the gap between short and long term projections still remains. Figure 14 offers one illustration, comparing the IAEA's latest low and high projections (the blue and red bars, respectively) and the projections of the OECD/IEA (grey bars) [6] with the median value of nuclear expansion (the green line) in the scenarios in IPCC's SRES [7].

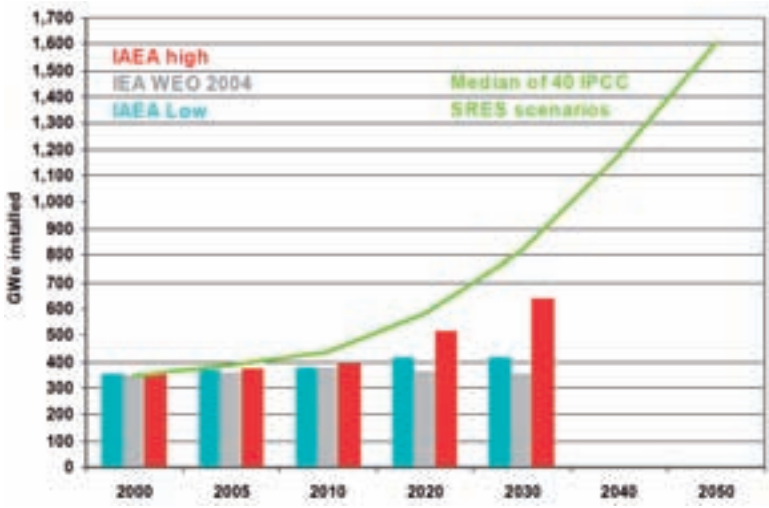

FIG. 14. Nuclear capacity projection: (red bars: IAEA high projection; blue bars: IAEA low projection; grey bars: OECD/IEA projection; olive green line: median values for the SRES scenarios).

The SRES long term scenarios have a planning horizon up to 2100. There will be significant depletion of low cost fossil resources (and low cost uranium) over such a long time horizon, and a resulting continual increase in fossil fuel costs. The anticipation of higher fossil fuel costs translates into larger shares for nuclear power. The effect is evident even early in the scenarios as the optimization models used for SRES, taking into account the fact that sharp increases in nuclear capacity cannot take place quickly, include early investments in nuclear power so that industry is ready when needed to provide the required capacity.

The medium term projections are based largely on current designs at current costs, while the long term scenarios assume continuing technological innovation to both raise performance and lower cost. The SRES scenarios include, of course, innovations as well as performance and cost improvements for all technologies, not just nuclear power. Thus, it is not clear how much of the projection gap may be due to innovations. However, even if the contribution is small, the importance of innovation must not be overlooked.

Moreover, in the models used to create the SRES scenarios, investments are essentially risk-free. They benefit from the model's 'perfect foresight'. In the real world, there are many risks that potential investors need to factor into their investment choices. These risks will not be the determining factor in every decision but they will move some decisions away from nuclear power, and thus lead to fewer investments in nuclear power than calculated in risk-free scenarios. These risks are mainly the risk of regulatory delays that raise costs at the front end of a plant's life cycle, and the risk of longer term low demand or low prices that reduce the revenue stream in the out-years. Reducing such risks, through mechanisms such as the partial regulatory insurance provided in the new US Energy Policy Act, would reduce the projection gap.

Figures 12 and 13 depict nuclear capacity projections by region until 2050. For the period to 2030 these projections correspond to the IAEA's global low and high projections, shown in Figs 12 and 13, respectively. These projections were extended using the median nuclear expansion rates of the IPCC SRES cases.

In the extended low projection, global nuclear generating capacity reaches 563 GW(e) by 2050. The growth rate of only 1% per year from the present to the middle of the twenty-first century implies a continuous decline of the nuclear share in the electricity mix. Even in the extended high scenario, where total installed nuclear generating capacity amounts to 1454 GW(e) by 2050 or a 3.9-fold increase over current capacity, the annual growth rate of 3.03% per year means that the nuclear share will barely be able to remain at the present value of 16%.

4. CONCLUSIONS

If the challenges of economic competitiveness, public acceptance and appropriate safety requirements can be met, nuclear energy can play a more important role in the future than it does at present in supplying the world population with energy. The desire to diversify from the present reliance on fossil fuels, the commitment to reduce CO_2 emissions and the limited prospects of large scale use of renewable sources tend to emphasize the potential contribution of nuclear power.

These points suggest several policy directions for governments that would like to encourage nuclear expansion. First, to the extent that policies can accelerate technological advances in the nuclear field, with associated cost reductions, they would encourage nuclear expansion. Second, to the extent that they can reduce political opposition to nuclear power in some countries and allow decisions more on the basis of economics, they will benefit nuclear power. And, third, to the extent they can reduce the financial and regulatory risks associated with large nuclear investments, they will also encourage nuclear expansion.

The incentives for the use of nuclear power are strong, especially in countries with growing populations, aspirations for economic development and improved quality of life, or concerns about environmental protection. If the objectives of advanced nuclear power development programmes are met, nuclear power could provide a long term, safe and economical energy supply as an integral part of the future energy system.

ACKNOWLEDGEMENTS

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URANIUM PRODUCTION CAPABILITY IN THE CIS COUNTRIES

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Abstract

Uranium production began in the former USSR in 1946, with annual production increasing gradually up to about 15 500 t U in 1986. Eight uranium producing centres were in operation: two in what is now the Russian Federation, two in Kazakhstan, and one each in Ukraine, Uzbekistan, Tajikistan and Kyrgyzstan. The total known uranium resources in the USSR were the largest in the world and amounted to 1 877 000 t U. Uranium production sharply decreased after the disintegration of the USSR. The stagnation in the mining industry of the Commonwealth of Independent States (CIS) ended only in the late 1990s. The current CIS primary uranium production, together with the supply from secondary sources in the Russian Federation, covers about 35% of world annual nuclear fuel cycle requirements. Primary uranium production in the CIS countries comprised 24% of total world production in 2004. Only the Russian Federation and Ukraine have nuclear power plants. All the uranium produced in Kazakhstan and Uzbekistan is available for export. Kazakhstan has the best potential for development of uranium production. The total annual uranium supply from the CIS countries, together with the Russian secondary supply, is planned to reach the equivalent of 30 000 t U by 2020. About 30% of this amount will be used by Russian and Ukrainian nuclear power plants and 20 000 t U will be available for the world market.

1. HISTORICAL REVIEW

The history of uranium mining in the USSR covers about 60 years. The necessity of establishing a uranium industry in the USSR was initially dictated by the strategic task of creating national nuclear weapons.

Uranium production in the USSR started in 1942. The uranium resources at that time amounted to only about 500 t U and related to five small deposits in the central Asian region. Comprehensive uranium exploration and technological research began in 1945, and 27 uranium deposits with total uranium

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resources of 28 000 t U were prepared by 1955 for development. Most of them were discovered in what are now the Commonwealth of Independent States (CIS) countries of Uzbekistan, Kazakhstan and Kyrgyzstan.

The first uranium producing centre in the USSR was constructed in Tajikistan, in 1945. In the 1950s, four more uranium mining enterprises were formed in the Russian Federation, Ukraine, Kyrgyzstan and Kazakhstan, and the amount of uranium produced increased by almost 30 times. By the end of the 1960s, eight uranium producing centres were already in operation. Starting in the 1970s and through the late 1980s, the total mining and processing of uranium gradually increased.

In the mid-1980s, the USSR was the world leader in uranium production and resources [1]. In 1988 its uranium industry reached a production peak of about 15 500 t U. Figure 1 shows the historical development of uranium production in the USSR and its successor republics. Eight uranium mining centres were in operation: Priargunsky Mining and Chemical Combine in the Chita region of the Russian Federation, Lermontov Mining Utility (now ALMAZ) in the North Caucasus of the Russian Federation, Tselinny Mining and Chemical Combine in northern Kazakhstan, Pricaspian Mining and Metallurgical Combine in the Mangyshlak peninsula of Kazakhstan, Vostochny Integrated Mining and Concentrating Plant (VostGOK) in Ukraine, Navoi Mining and Metallurgical Combine in Uzbekistan, Leninabad Mining Combine

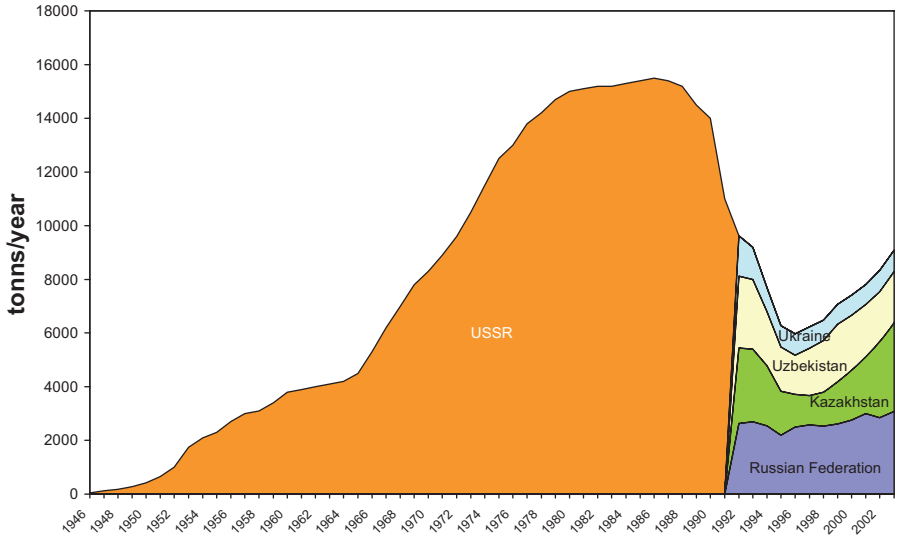

FIG. 1. Historical uranium production in the USSR and the countries of the CIS.

in Tajikistan, and Karabalty Mining Ore Combine in Kyrgyzstan. These operations were closely interrelated, and ore mined in one republic, or even in the countries of Eastern Europe, could be processed in another republic. The distribution of the cumulative uranium production among enterprises is shown in Fig. 2.

The uranium resources in the former Union of Soviet Socialist Republics amounted to 1 187 000 t U, and fully satisfied the needs of the atomic industry. Annual exploration expenditures at the end of the 1980s were about 280 million US dollars, and up to 5000 km of exploration drilling was performed annually (Fig. 3).

In 1990, a decision of the Soviet Government about conversion of the military industry resulted in a sharp decrease of national demand for uranium. As a result, uranium production sharply decreased in all republics, dropping to 6000 t U in 1996 [2] (Fig. 1). Some mines and uranium processing plants in Kazakhstan, the Russian Federation, Tajikistan and Kyrgyzstan were closed, placed on standby or had their uranium production significantly reduced. Funding for uranium exploration also decreased sharply, and no new uranium deposits were discovered during this period. The CIS mining industry entered a period of stagnation which ended only in the late 1990s. The situation was aggravated by unfavourable market conditions.

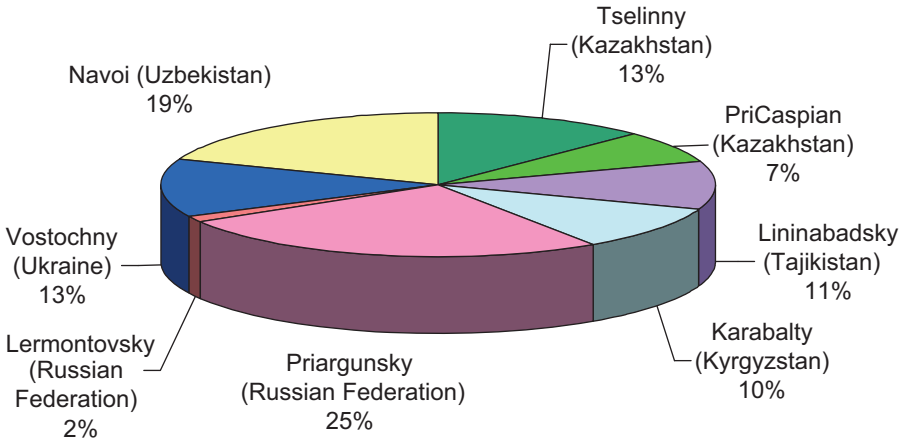

FIG. 2. Distribution among producing centres of uranium production in the USSR.

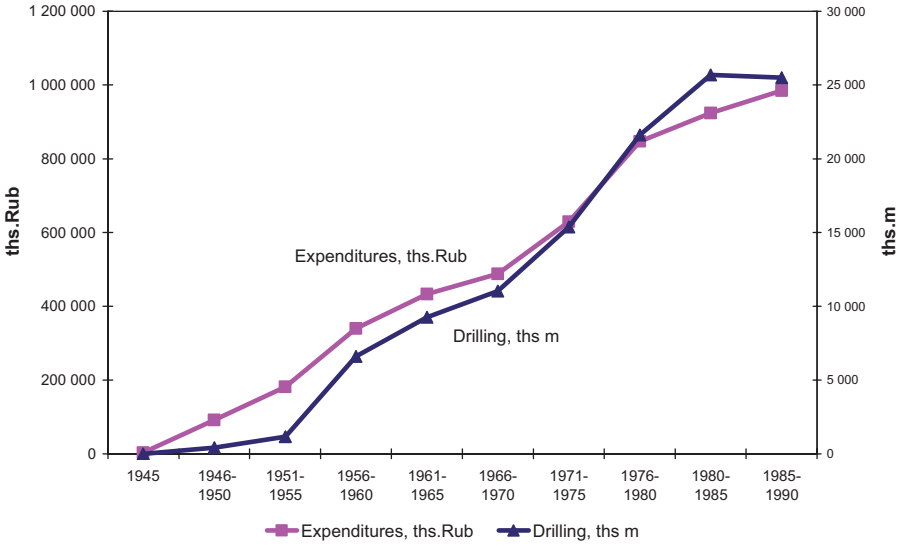

FIG. 3. Historical five year exploration expenditures and drilling efforts in the USSR.

2. STATUS AND DEVELOPMENT OF URANIUM PRODUCTION IN THE CIS COUNTRIES

The current uranium supply from the CIS countries covers about 35% of world annual nuclear fuel cycle requirements. About 60% of this amount comes from Russian secondary sources and 40% from primary uranium production. Secondary supply consists of blended, highly enriched uranium (HEU) (weapons grade) delivered to the United States of America (USA), inventory drawdown and reprocessed uranium (RepU) (spent fuel). The US–Russian agreement on HEU will expire in 2013, and the majority of Russian stockpiles will be depleted by 2020. The role of primary uranium from CIS countries will expand in the future as the contribution from secondary supply diminishes.

Only the Russian Federation and Ukraine have nuclear power plants. Russian nuclear fuel cycle demand consists of about equal requirements from domestic reactors and fuel assemblage exports, including exports to Ukraine. Uranium produced in Ukraine is shipped to the Russian Federation but Ukrainian requirements are three times greater than domestic production. All uranium produced in Kazakhstan and Uzbekistan is available for export.

Table 1 gives data on the uranium resources of the CIS countries. Most of these resources were discovered prior to the disintegration of the USSR. Regarding the amount of identified resources (reasonably assured resources (RARs) and estimated additional resources – category I (EAR-I)/inferred resources), Kazakhstan is the leader and the Russian Federation second. About 70% of CIS known resources recoverable at a cost below US $40/kg U are tributaries to existing and committed production centres. Low cost uranium resources in Kazakhstan and Uzbekistan belong to sandstone type uranium deposits which are amenable to in situ leach (ISL) mining. Most Russian and Ukrainian resources require conventional underground mining.

Primary uranium production in the CIS countries in 2004 comprised about 9900 t U, which was 24% of total world production: Kazakhstan produced 3720 t U, the Russian Federation 3280 t U, Uzbekistan 2100 t U and Ukraine 800 t U.

2.1. Uranium production in Kazakhstan

Currently, Kazakhstan ranks third in world uranium production and second in low cost resources. All uranium related activities in Kazakhstan are

TABLE 1. URANIUM RESOURCES (TONNES URANIUM) IN CIS COUNTRIES [2]
(as of 1 Jan. 2003, with costs in US dollars)

Country	RAR			EAR-I			Total
	<$40/kg	<$80/kg	<$130/kg	<$40/kg	<$80/kg	<$130/kg	
Russian Federation	52 610	124 050	143 020	15 860	34 260	121 220	264 220[a]
Kazakhstan	280 620	384 625	530 460	113 220	237 780	317 160	847 620
Uzbekistan	61 510	61 510	79 620	31 760	31 760	38 840	118 460
Ukraine	15 380	34 630	64 660	900	4 735	11 410	76 070[b]
Total	410 120	604 815	817 760	161 740	308 535	488 630	1306 370

[a] Besides the explored resources presented in this table, the Russian Federation has about 350 000 tonnes of EARs of a cost category that has not been defined yet, which belong to the standby deposits in the Elkon uranium ore region of the Sakha (Yakutia) republic. They were opened, explored, and technically and economically assessed in the 1960s and 1970s. A complex technical and economic re-assessment of these deposits is under way at present.

[b] In Ukraine, the resources of the Novokonstantinovskoe deposit are not taken into account.

under the supervision of the National Atomic Company Kazatomprom (NAC Kazatomprom). It incorporates the JSC mining company (responsible for ISL uranium mining), the JSC Ulba metallurgical plant (production of nuclear fuel pellets) and JSC Volkov Geology (uranium exploration and drilling). The JSC mining company includes three uranium mining utilities: Stepnoe, Central and Mine #6. Two joint ventures, Inkai with the Canadian company Cameco and Katko with the French company COGEMA, started pilot ISL production in 2000 [2].

In recent years, uranium output in Kazakhstan has increased more than threefold: from 1090 t U in 1997 to 3720 t U in 2004. Production in 2004 by sulphuric acid ISL mining technology totalled 3604 t U; the remaining production (116 t U) was by underground mining at the Stepnogorsk Mining and Chemical Combine, which was handed over to Kazatomprom in 2004. The JV Betpakdala concern produced 647 t U at the Akdala deposit. During pilot operations, JV Katko produced 57 t U and JV Inkai produced 183 t U. Construction work is being carried out at JV Zarechnoe. The net cost of uranium produced by ISL technology is estimated to be US $15–20/kg of uranium.

According to the basic development scenario, uranium production in Kazakhstan will reach 9300 t U in 2015 and 15 000 t U in 2010, in the 'aggressive' case [3]. Available known low cost resources amenable to in situ leach mining are quite sufficient to reach these plans, and most of the known deposits are already under or planned for development. To achieve this goal, it is planned to develop existing and committed mines to their nominal capacities and to construct seven new mines in new areas. The rated annual output of each mine is about 1000 t U. NAC Kazatomprom plans to start the development of new deposits, mainly with the help of foreign investors. The programme also envisages the construction of mines at the Central Moinkum, West Mynkuduk, Inkai, Kharasan, Budenovskoe and other deposits. In addition, increased output can be achieved by using advanced uranium mining and processing technologies, together with efficient foreign equipment, as well as by increased production on the part of the joint ventures.

In April 2004, NAC Kazatomprom signed an agreement with the French company AREVA (COGEMA) to launch a joint project for uranium mining in Kazakhstan. After operation for three years of an experimental plant at the Moinkum deposit in Southern Kazakhstan between 2000 and 2003, JV Katko began the design and construction of an industrial plant. Initial production is expected in late 2005, with a growth of output to 1500 t U per year [4].

In 1998, NAC Kazatomprom and Cameco set up a joint venture for uranium mining and processing — JV Inkai at the Inkai deposit. Kazatomprom's share is 40%, and that of Cameco is 60%. Initially the

enterprise design capacity was about 1000 t U/a, but after completion of feasibility studies it was decided to begin commercial uranium production in 2007, followed by a three year buildup to the full capacity of 2000 t U by 2010 [5]. Experimental leach tests conducted between 2000 and 2004 confirmed the economic viability of the project and suggested the major ISL parameters.

In late 2001, the Russian–Kyrgyzstan–Kazakhstan Zarechnoe joint venture was established to mine the Zarechnoe deposit in southern Kazakhstan. It is planned that the enterprise will start production of uranium in late 2006, and in 2008 the project will reach a nominal capacity of 500 t U/a. Negotiations between Kazakhstan and China are underway to set up a joint venture for the development of the Zhalpak deposit and part of the Kharasan deposit. The Japanese Itochu Corporation and NAC Kazatomprom established JV Irkol in 2004 (NAC Kazatomprom has a 74% share and Itochu Corporation has 26%) for joint development of the Irkol deposit.

Taking into consideration the existing and planned production capacities in relationship to resources, it was accepted that Kazakhstan's annual output will reach 15 000 t U by 2020.

2.2. Uranium production in the Russian Federation

The Russian Federation has a well developed nuclear power industry. Its annual nuclear fuel cycle demand amounts to some 15 000 t U, which consists of the about equal requirements of domestic reactors, nuclear fuel assemblage exports (supplied by the TVEL Corporation) and low enriched uranium exports (supplied by the TENEX Corporation).

Russian uranium mining enterprises belong to the state TVEL Corporation, which produces nuclear fuel for 76 power reactors worldwide, including all Russian nuclear power plants. The total uranium production of the Russian Federation comprised 3280 t U in 2004. At present, TVEL is modernizing existing mines and developing new deposits. The Russian strategy is based on development of its domestic uranium production. It is planned to increase annual uranium production from the current 3200 t U to 10 000 t U in 2020: about 5500 t U will be produced at existing and developed mines, and the remaining 4500 t U will be produced at new mines, which will be constructed after 2010 [6].

In 2004 the state programme 'Russian Uranium' was developed. It envisages implementation of the following most important long term tasks:

(a) Development of uranium production at mining enterprises currently operating and those whose construction is committed;

(b) A feasibility study and re-evaluation of previously discovered Russian uranium deposits, to select the most favourable deposits and prepare them for mining;
(c) Discovery of new uranium resources for construction of new uranium producing centres.

At present, uranium is produced in the Russian Federation at three centres: the existing conventional mining centre at Priargunsky and two new ISL mines at Dalur and Khiagda.

The Priargunsky Mining–Chemical Production Association (PPGHO) was the only active uranium production centre in the Russian Federation during the last decade. Production is derived from the Streltsovsk district deposits, where the total resources exceed 150 000 t U, at an average grade of about 0.17% uranium. Operations started in 1968, with two open pits (both are now depleted) and three underground mines (mines Nos 1 and 2 are active, and mine No. 4 is on standby). Annual production in the last five years was 2500–3100 t U. More than 90% of production comes from underground mining and conventional milling, and the remainder is produced from low grade ores by heap leaching and ISL methods. The following main projects are planned at Priargunsky to provide sustainable uranium production at about 3000–3500 t/a for the next 20–25 years: modernization and reconstruction of operating mining complexes, introduction of block and heap leaching mining methods for low grade ores, new sulphuric acid plant construction, new mine development and operation, reconstruction of the hydrometallurgical processing plant and construction of a new radiometric sorting mill. Expansion of production is related to a new mine (No. 6), which is planned to come into operation after 2010. Reserves are hosted in carbonate rocks, and their processing requires construction of an alkaline leaching circuit.

Commercial ISL uranium production began at Dalur in 2002 on the Dalmatovskoe basal channel type deposit. In 2005, the main processing plant will come into operation, and annual production will gradually increase from 180 t U in 2004 to 500 t U in 2008. New developments are planned for the Khohlovskoe deposit, situated 85 km to the south-east of Dalmatovskoe. Multiwell ISL sulphuric acid tests and geological exploration will take place at Khohlovskoe during 2005–2007, and commercial operation will begin in 2008. Productive solutions will be processed at local sorption plants, and saturated resin transported to the main plant. The total annual uranium production from both deposits will increase to 750 t U in 2010 and to 1000 t U in 2012. The uranium resources in the district total about 20 000 t U.

Pilot ISL production has been in progress at Khiagda since 1999, and a feasibility study will be developed in 2005. In 2006 the main construction will

start, with a goal of reaching the nominal annual capacity of 1000 t U in 2011. Data obtained to date show favourable technical and economic parameters for ISL mining in spite of very low temperatures of the leaching solutions. Production will come from seven similar basal channel type deposits located within the Khiagda uranium district with resources estimated preliminarily at about 40 000 t U.

The projected uranium production shortfall in the Russian Federation will be partly covered from new mines to be developed at previously discovered deposits. Numerous uranium deposits were discovered in the Russian Federation during the 1960s–1980s that were unfavourable for production at that time. They will be re-evaluated according to current economic conditions.

During the next three years, feasibility studies will be carried out to evaluate the metasomatite type uranium deposits of the Elkon district with resources of more than 250 000 t U containing about 0.14% uranium. In addition to uranium, the ore contains gold, molybdenum, silver and some vanadium. Despite large uranium resources, these deposits may require a market price of over US $70 per kilogramme uranium to be profitable. However, a preliminary estimate shows that production costs may be reduced by recalculation of resources using a higher cut-off grade and improving mining, enrichment and processing technologies. Preliminary feasibility studies relating to the Elkon uranium deposit suggest that the output of the enterprise may reach 2500 t of uranium per year, provided that the necessary investments of about 400–500 million US dollars are allocated.

Another objective of the Russian programme is evaluation and selection of the most favourable deposits in the Transbaikalia district, which can be effectively mined using in situ, heap or in block leaching methods. The deposits with the best prospects are at Gornoe and Olovskoe. The resources of the Gornoe deposit total about 17 000 t U, with an average grade of 0.25% uranium. Pilot radiometric sorting and processing by heap and block leaching methods performed in the late 1970s yielded good results. The main task now is to prepare a feasibility report in 2006–2007 for a uranium mine with an annual output of about 400 t U. A feasibility study for the Olovskoe deposit is scheduled for the near future, to evaluate the profitability of block and heap leaching mining on the basis of an annual output of up to 500 t U.

Realization of development plans for uranium production and resources during 2005–2010 should create a stable base for sustainable development of the Russian nuclear fuel cycle during the next 20 years.

Exploration activities in the Russian Federation during the last eight years were focused primarily on sandstone basal channel deposits amenable for ISL mining and on unconformity related deposits with high grade ores. The

potential for discovery of new low cost unconformity and sandstone type deposits is rather high. The uranium exploration budget was doubled in 2005 and reached about 17 million US dollars.

2.3. Uranium production in Uzbekistan

Uranium in Uzbekistan is produced by the Navoi Mining and Metallurgical Combine (NGMK) using sulphuric acid ISL technology at three mining divisions: Northern (in Uchkuduk), Southern (Nurabad) and Division No. 5 (Zafarabad). Seven deposits are currently in operation, and pilot tests are being carried out at one deposit. Over the last three years, annual output has stabilized at around 2000 t U, totalling 2087 t U in 2004. Uzbekistan exports uranium to the USA under a single long term contract with the RWE Nukem company. The uranium development programme envisages the improvement of weak acid uranium extraction technology, modernization of production stages, new designs of wells and more efficient methods for extracting and treating productive solutions.

The volume of uranium production for the next two–three years is planned at the same level as for 2004, but, depending on world uranium prices, estimated figures can be revised upwards. The above programme development will depend on the volume of foreign investment in the uranium industry. For the next 5–10 years, production growth will be ensured by the deposits currently operated. Mining of the Northern Kenimekh deposit is planned for 2006.

Taking into account their considerable depletion, the annual mining output up to 2010 at existing mines will not exceed 2300 tons. Development of new ISL mines with an annual capacity of up to 500 t U each may increase uranium production to 3000 t U by 2015 [2].

2.4. Uranium production in Ukraine

The only uranium producing centre in Ukraine is the state owned VostGOK. The annual uranium production level remains at about 800 t U. Current production capacities are supported by the resources of the Vatutinskoe and Michurinskoe deposits. Production development at existing mines is limited due to depletion of resources. An increase in uranium mining and processing is primarily associated with the mining of the new Novokonstantinovskoe deposit and construction of a mining enterprise on the Severinskoe deposit.

Development at the Novokonstantinovskoe deposit is limited to three shafts down to a depth of 300 m. The Severinskoe deposit is now at the stage of

wet preservation. Fulfilment of production plans for these projects is impossible without significant investment.

According to official information, annual uranium production is planned to double to 2000 t U [2], but taking into account the depletion of existing mines it is more likely that annual production will not exceed 1500 t U up to 2015. The potential for new mines by application of the ISL extraction method has not been determined yet. Pilot ISL tests are planned for the Safonovskoe deposit.

2.5. Uranium production in Kyrgyzstan

Uranium mining in Kyrgyzstan started in 1955, when a mill with a nominal annual capacity of 2500 t U was built to process ore from Kyrgyzstan, Kazakhstan and the Russian Federation. The operator was originally Yuzhpolymetal Mining and Metallurgical Combine; the successor is Kara Balty Ore Processing Combine. Uranium production lasted until 1989, when conventional mining in south-east Kazakhstan was abandoned. Subsequently, yellow cake slurries from ISL operations in southern Kazakhstan were processed from 1994 at a rate of 1000 t U/a of concentrate. In recent years, the circuit has been partly reconfigured to treat commodities other than uranium, including gold ores. However, uranium production could be resumed.

For the present, Kyrgyzstan has rejected plans for mining uranium again in its territory. Uranium deposits explored in the USSR period have been depleted, while there are no monetary means or investors to fund explorations for new deposits. Kyrgyzstan has no projects for development of new uranium deposits.

2.6. Uranium production in Tajikistan

The first uranium producing centre in the former USSR was built in Tajikistan in 1945. Its current name is Vostochny Rare Metal Industrial Association (Vostokredmet). Originally it was known as Combine No. 6, later as the Leninabad Mining–Chemical Combine. Ore processing ceased in the early 1990s. Production during the early years was based on ore mined at two small local deposits. Annual capacity later reached 2000 t U, with mill feed derived from conventional ores and ISL slurries from Uzbekistan. Since 1993, the mill has been transformed to treat Pb–Zn–Ag ores [2]. Currently, there are no uranium resources in Tajikistan. The favourability of reported uranium occurrences is not high [7]. No production of uranium is planned.

2.7. Uranium production in Turkmenistan

Discovered in 1952, the Sernoye deposit was mined from 1952 to 1967 by open pit and underground methods. After enrichment by radiometric ore sorting on-site, ore concentrates were shipped to Kazakhstan for yellow cake production. Total production from the Sernoye deposit is estimated to have been between 5000 and 7000 t U. Currently, Turkmenistan has no reported mineable uranium resources [2].

3. PROJECTIONS OF URANIUM SUPPLY–DEMAND IN CIS COUNTRIES

Summarizing officially published information and according to expert estimations, uranium production in the CIS countries is expected to increase threefold during the next 15 years: from the current 9600 t U to 27 500 t U in 2020 (Fig. 4). Kazakhstan will be the key world producer in the future, having large low-cost resources and developing its uranium production. To cover the requirements of domestic nuclear reactors and export supplies, and taking into account exhausted stockpiles, the Russian Federation plans to more than double uranium production by 2020.

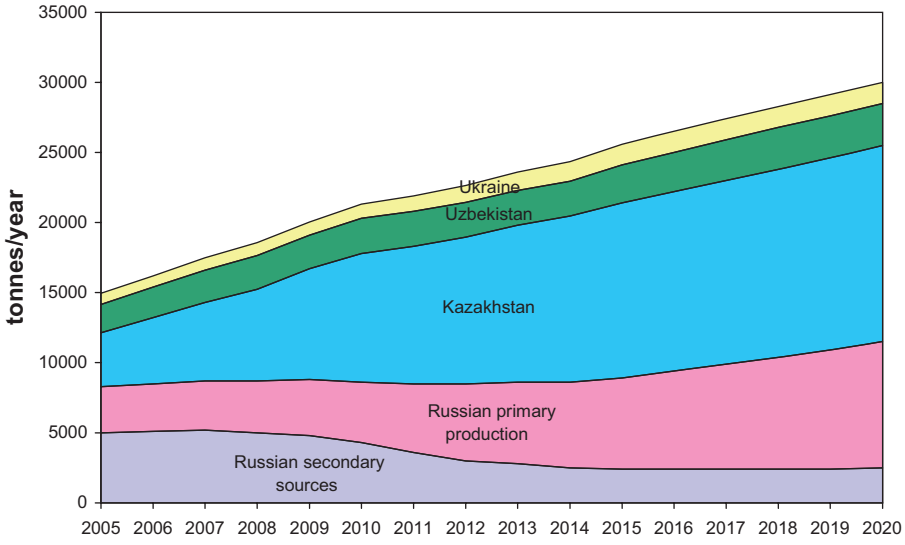

FIG. 4. Projections of uranium supply from CIS countries up to 2020.

The uranium supply–demand relationship in CIS countries shows that the total annual uranium supply from these countries, including Russian secondary supply, is planned to reach 30 000 t U by 2020. Less than 10 000 t U of this amount will be used by Russian and Ukrainian nuclear power plants. This means that about 20 000 t U produced in the CIS countries will be available annually for the world market after 2020 (Fig. 5). The cumulative CIS market supply for the 15 year period (from 2005 to 2020) will reach 232 300 t U.

The development of uranium production in the CIS after 2020 will depend on the availability of resources sufficient for construction of new mines. While the total uranium resource base may be adequate to satisfy future demand, confidence in the reliability of uranium resources will have to be increased by more detailed exploration. Taking into consideration the fact that that the lead time between exploration and start of operation is usually more than 15 years, significant intensification of exploration is required. New mines will have to be developed to replace mines closed as a result of resource depletion.

The potential for discovery of new uranium deposits in CIS countries is rather high, taking into account the historical results of previous explorations, rising uranium prices, and favourable geographical and geological diversity. Recognition criteria for the different types of uranium mineralization have

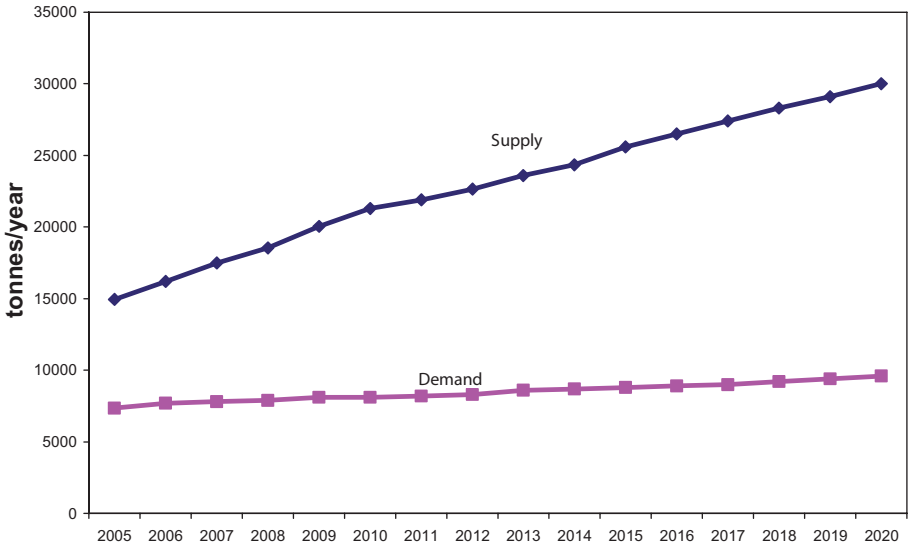

FIG. 5. Projections of the relationship of uranium supply to demand from CIS countries to 2020.

been applied to different geological provinces. However, discovery of the new uranium deposits requires broader international cooperation among exploration, mining and research organizations.

4. CONCLUSIONS

The following conclusions can be drawn:

(a) A powerful system of natural uranium geological surveying, mining and processing was created in the former USSR.
(b) Uranium mining will develop dynamically in most CIS countries. Kazakhstan is recognized as the most promising country for significant increases in uranium production. The Russian Federation has a good potential for production increases by development of existing centres and construction of new mines at previously discovered deposits.
(c) The total supply from CIS countries, including secondary supply, is planned to reach 30 000 t U by 2020. Less than 10 000 t U will be used by CIS nuclear power plants and more than 20 000 t U will be available annually for the world market.
(d) Development of uranium production after 2020 will require discovery of additional uranium resources and therefore a significant intensification of exploration efforts. The potential for new deposits to be discovered on the territory of CIS countries is significant.

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URANIUM PRODUCTION CAPABILITY: NORTH AMERICA, AUSTRALIA, ASIA AND AFRICA

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Abstract

With the apparent need for additional primary uranium supply, the paper discusses necessary preconditions for mine construction and then reviews the supply potential in North America, Australia, Africa and Asia. Canadian uranium supply will increase from 12 000 t U to over 16 000 t U by 2011 with the commencement of production at the Cigar Lake mine and the eventual mining of the Midwest Lake deposit. Canadian production levels will then begin a slow and steady decline from 2014 onwards as existing economic deposits become exhausted. Production in the United States of America will continue to be dominated by in situ leech mines but certain conventional mines will provide some additional production. Total US production may reach 4000 t U by 2010. Future Australian production levels are clouded in uncertainty. Restrictive Government policies are once again having an impact on uranium prices. A potential tripling of production at Olympic Dam, however, will have a dramatic impact on uranium supplies if a decision to expand production is indeed made. Depending upon market forces, Government and corporate decisions, uranium production in Australia could vary from as little as 4500 t U to as much as 22 000 t U by 2020. After 30 years of production, Phase 1 mining at the Rössing mine in Namibia is nearly complete and the mine is now faced with a series of difficult decisions in order to remain in operation beyond 2009. Steady output from existing mines in Niger is expected for at least a further ten years, and additional mining opportunities have been identified. Other potential African production sources are also discussed. Mongolia has the potential to become a new modest sized producer, depending upon the outcome of further studies. Production levels in India and China are expected to increase modestly in order to help satisfy domestic needs.

1. INTRODUCTION

This paper discusses the future uranium supply potential for North America, Australia, Africa and Asia. The paper has been written from the perspective of a mining engineer who has been involved in recent uranium

mine assessments, construction and operation. Although this paper discusses current and future supply potential based upon cost, it also discusses many of the non-economic issues that must be successfully dealt with in order to create a viable mining operation. A discussion of these issues is presented first, followed by an assessment of the future supply situation for the continents discussed. Emphasis is on the extraction potential of known deposits and not the exploration potential of new deposits.

This paper utilizes the IAEA conventions of reasonably assured resources (RARs), estimated additional resources (EAR-Is and EAR-IIs) and speculative resources (SRs) to describe uranium contained in individual deposits where possible. Information sources are referenced where applicable.

This paper was written in mid-2005 when prevailing long term contract prices were US $75/kg U. The analysis and opinions expressed in this paper are those of the author and do not reflect those of Cameco Corporation, the IAEA or the Government of Canada.

2. IMPORTANT PRECONDITIONS FOR MINE DEVELOPMENT

With an increased awareness that new mines will be necessary to meet demand, it is instructive to review some of the factors that have an impact on development decisions. The price at which a given deposit is deemed economically feasible to exploit is the key determinant as to whether a market price based project will proceed to production or not. However, there is a multitude of other factors that must also be satisfied. These factors are largely regulatory, social, environmental and legacy issues.

The uranium oversupply situation prevailing in the 1980s and 1990s led to an exodus of producers from the uranium mining business. A current uranium property holder may have acquired the property through a merger or acquisition but may have no desire or means to mine uranium if deemed economic to do so. A similar situation arises when a company has too many potential economic opportunities available in their suite of properties. In this case, it may not be logistically possible to exploit all opportunities, although profitable to do so.

Numerous deposits, particularly in the United States of America (USA), have complex ownership, royalty and tax structures such that the current economics of exploitation may be greatly altered by previous commitments.

The period of oversupply led, in many cases, to changes in the regulatory and social frameworks surrounding uranium mining. Perhaps the best known example of many is Australia's previous 'three mines' policy. When a company perceives that there is a high likelihood that permits to mine a deposit will not

be received, there is little incentive to study the feasibility of its extraction. Delays in acquiring permits also have an impact on the economics of deposit extraction, since the economic payback period is increased and the return on investment is lowered.

Providing a local jurisdiction is amenable to uranium mining, it is then necessary in many jurisdictions to receive a social licence to do so. Typically, this entails entering into a formal or informal socioeconomic agreement with residents in the area surrounding the planned operation. It is also necessary to ensure that the environmental impact of a project is acceptable to the public and to regulatory agencies. In general, the level of permissible environmental impact has continued to decrease markedly in many jurisdictions. The net result is to increase an operation's capital and operating costs as more expensive measures are implemented. Examples include mine water discharge and tailings disposal standards.

An entity considering uranium mining may have a limited pool of capital available and more financially attractive non-uranium mining opportunities available at the time. Project execution may, therefore, not be based solely upon satisfying certain investment criteria.

Reasonable assumptions with respect to future uranium prices must also be made. These assumptions may bear no relationship to current prices unless a significant portion of planned output is pre-sold in long term contracts.

Owing to the numerous uncertainties in projects discussed above, and a potentially long pre-production lead time, it is necessary to include a considerable risk premium into the economic analysis of a uranium deposit. In an oversupplied market, it may be possible for a utility to obtain uranium at the average or marginal unit cash operating cost of a mine attempting to remain in business. However, when new primary production is required to meet demand, capital costs, operating costs and risk adjusted profit margin must be adequately satisfied by anticipated commodity prices for construction to be warranted. Publicly listed companies do not enter the uranium mining business with the aim of only breaking even.

Contrary to the arguments presented above, however, some new production may be price insensitive if security of supply concerns override supply cost concerns.

3. URANIUM MINING IN CANADA

All Canadian uranium production is located in the Athabasca Basin of Saskatchewan. There are currently three production centres: McClean Lake, Rabbit Lake and Key Lake/McArthur River. On the basis of RARs only, Fig. 1

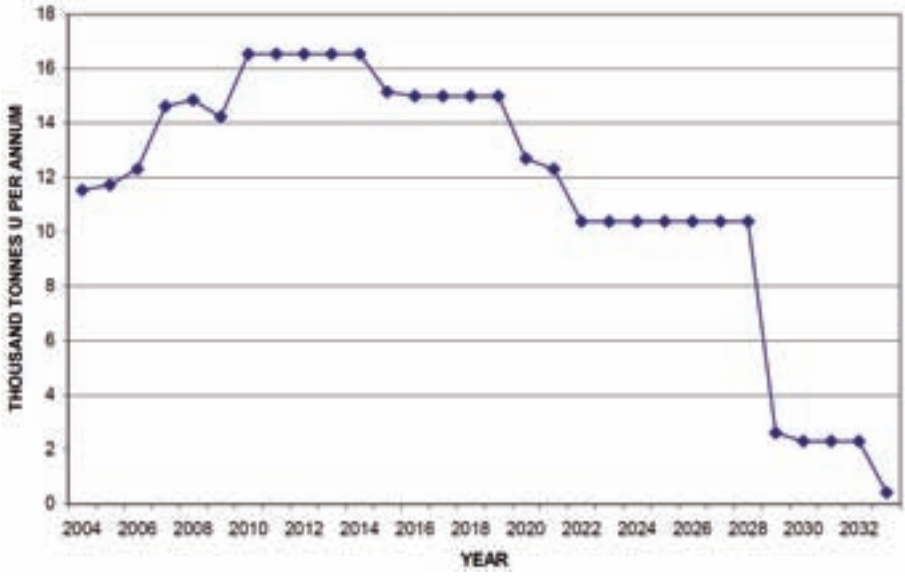

FIG. 1. Estimated Canadian uranium production from current and planned mines.

illustrates the life of mine production schedules for current and planned mines as estimated by the author.

3.1. The McClean Lake joint venture

The McClean Lake mine consists of a mill licensed to produce up to 3 000 t U/a but currently operating at 2310 t U/a. The mine exploits a number of unconformity type uranium deposits. Production commenced in 1999. Feed to date for the mill has come from the Jeb and Sue C open pits. Milling of ore from these pits will be complete by 2006 when it is expected that additional ore from future Sue A, B and E pits will be processed. Ore from these pits totals approximately 3200 t U of RARs [1].

Beginning in 2007, ore from the Cigar Lake deposit is expected to be milled at McClean Lake, following a mill expansion to 9200 t U/a [2]. Cigar Lake is expected to take three years to ramp up production to 7000 t U/a, and is discussed in greater detail below.

When processing of ore from the Sue A, B and E pits is complete, ore from the Midwest deposit is expected to be processed. Tentative plans, subject to regulatory approval, are to open pit mine approximately 13 500 t U from the

Midwest deposit by 2010 and recover 1500 t U/a in the McClean Lake mill for approximately eight years [1].

The expanded McClean Lake mill will have the capability of leaching up to 9200 t U but only packaging approximately 4600 t U. Surplus milling capacity at the Rabbit Lake mill will be utilized by transporting uranium rich solution from McClean Lake to Rabbit Lake and packaging up to 4600 t U there [2].

Although some additional small pods of mineralization are present in the McClean Lake area, no conclusive extraction plans have been formulated [3].

3.2. The Cigar Lake joint venture

On 21 Dec. 2004, the Cigar Lake joint venture partners announced plans to construct the Cigar Lake mine in order to commence production in 2007. A three year production rate rampup is anticipated. The Cigar Lake deposit is a high grade, flat lying, clay rich deposit containing RARs of 89 000 t U and EAR-Is of 29 500 t U [4]. The RARs are sufficient to sustain the planned production rate of 7000 t U to 2019 (Phase1). The novel jet bore mining method will be used from underground workings following extensive area ground freezing. The orebody depth is approximately 450 m.

As illustrated in Fig. 2, a series of bored tunnels is developed below the orebody. The lower tunnels are utilized to freeze the ground above, including the orebody and clay rich cap. Production tunnels are then bored through the frozen ground. Jet boring is then used to extract the high grade, clay rich, ore.

EAR-Is are present to the south of currently planned mining (Phase 2) sites. These resources require additional exploration drilling. Mining of these resources at a rate of 1500–2300 t U/a may be possible providing Phase 1 mining has demonstrated the economic and technical attractiveness of doing so. This resource has the potential to extend the life of the mine at Cigar Lake beyond 2030, albeit at a greatly reduced annual production rate.

3.3. The Rabbit Lake mine

The Rabbit Lake mine has been in operation since 1975. Originally, a series of open pits was mined. Underground mining commenced in 1992 at Eagle Point, and this deposit is the sole source of ore for the mill today. Blasthole stoping is utilized, with recent mining grades being approximately 1% U. Although the mill is capable of processing in excess of 6000 t U/a, mining constraints in recent years have limited annual production to between 2000 and 2300 t U. Reasonably assured resources of 5460 t U at the end of 2004 are sufficient to maintain a production rate in this range to mid-2007 [4]. An aggressive exploration programme has been under way since 2002 to find

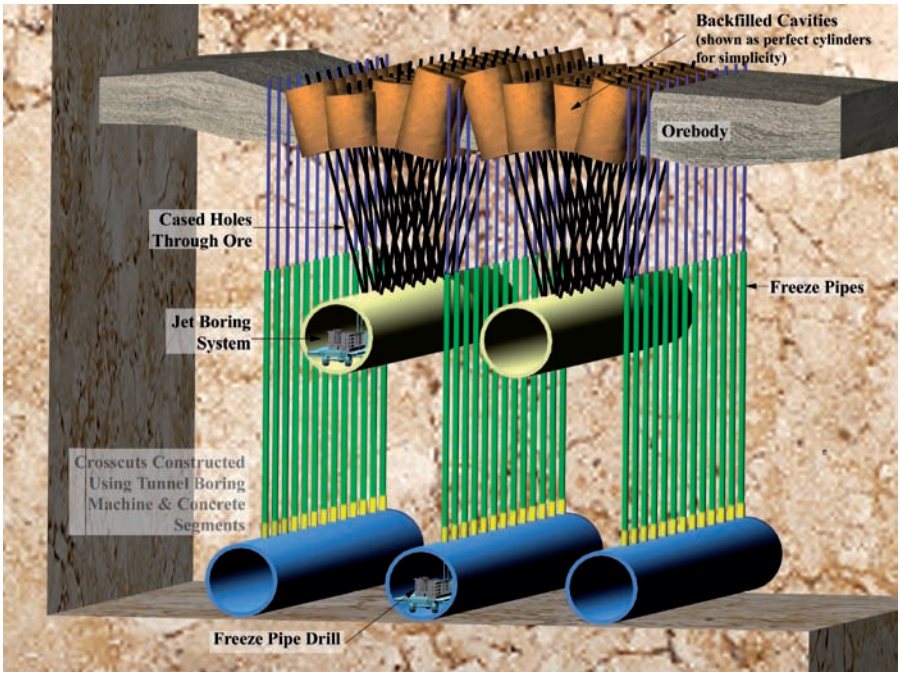

FIG. 2. Layout of jet boring mining method at Cigar Lake.

additional ore accessible from existing workings. This programme has been successful in maintaining the reserve base, and it is anticipated that further exploration will have some additional success. Exploration potential remains in the area and Cameco hopes to extend the mine life beyond 2007.

As noted above, the Rabbit Lake mill will be utilized to process a portion of the Cigar Lake ore. Critical site infrastructure will therefore remain in place long after the mine at Eagle Point has been mined out. It is conceivable, therefore, that the mill could be used to process further uranium ore discovered in the area over the next twenty years.

3.4. The Key Lake/McArthur river operation

McArthur River is the world's largest uranium mine in terms of annual output. Typical mining grades are 14% U. A novel raise bore mining method is used in combination with ground freezing to limit the potential for groundwater to enter mine workings. Ore is transported 70 km to the Key Lake mill for processing where it is diluted down to 3.7% U, in order to reduce

worker radiation exposures in the milling process. Production in 2004 totalled 7200 t U, which represents the current annual regulatory limit. A total of four zones of mineralization have been drill defined by underground exploration programmes, and this ore constitutes RARs of 161 300 t U and EAR-Is of 6650 t U as of 1 Jan. 2005 [4]. These zones are illustrated in Fig. 3. Current plans are to increase production by the end of 2005 or early 2006 to an annualized rate of 8100 t U.

At McArthur River, all mining to date has occurred in Zone 2. At least one additional zone is expected to be in production by 2007 in order to meet and sustain the increased production rate planned. Reasonably assured resources are sufficient to maintain the planned production rate of 8100 t U beyond 2025.

Surface drilling has identified two additional zones of mineralization, with EAR-Is of 40 500 t U being estimated from this drilling [4]. Underground exploration programmes in the next five years will attempt to better define this mineralization.

Although the main ore bearing P2 fault structure extends considerably beyond the known reserve and resource base, there are currently no indications

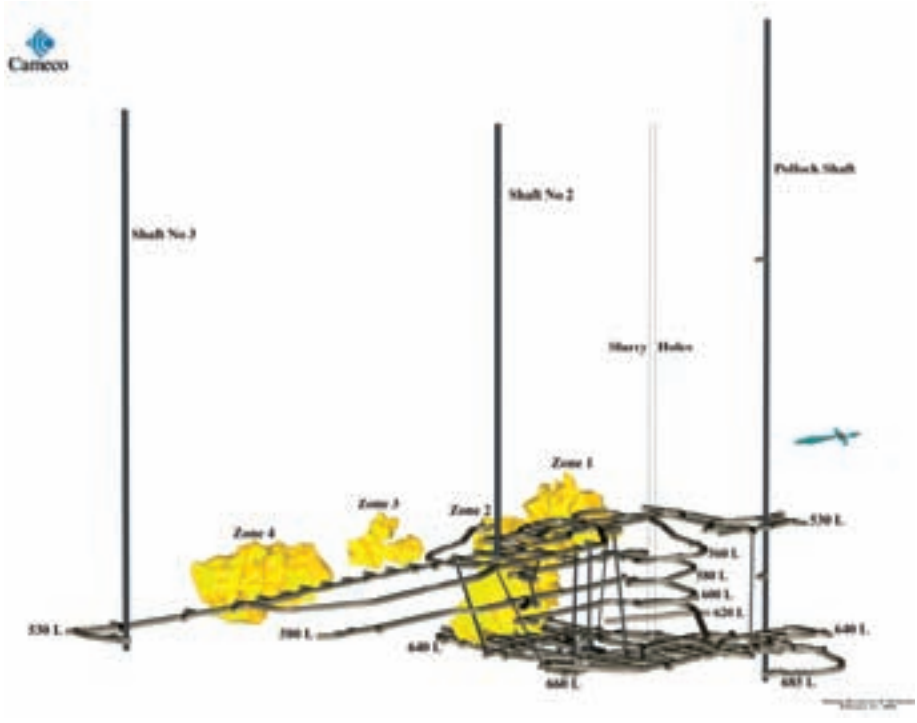

FIG. 3. Isometric view of underground workings at McArthur River.

that other attractive mining targets exist. Additionally, a further expansion of uranium production rate at Key Lake and McArthur River is complicated by milling and mining rate bottlenecks.

3.5. Millennium deposit

Cameco recently published resource figures for a new uranium deposit located 35 km north of Key Lake: RARs total 547 000 t with a grade of 2.44% U for 13 300 t U and EAR-Is total 293 000 t with a grade of 2.07% U for 6000 t U [4]. The Millennium deposit is located on the Cree Extension property. This property is a joint venture between Cameco, AREVA and Japan–Canada Uranium Co. (JCU).

The deposit is located at a depth of 600–750 m. It is largely a basement rock hosted uranium deposit with some similarities to Cameco's Eagle Point mine. Exploration drilling is still in progress and no joint venture studies have been undertaken to determine mining potential.

3.6. Uranium mining in Saskatchewan

The Cigar Lake and Midwest Lake deposits represent the last of the significant economic deposits discovered by previous concerted exploration efforts in the 1970s and 1980s. New discoveries, such as the Millennium deposit, are necessary to refill the project pipeline. A 1990s exploration cycle was missed owing to the uranium prices then prevailing.

Mining has progressed from near surface open pit mining to deeper underground targets. Underground mining often requires high unit capital and operating cost methods, on a per tonne ore basis. Illustrated in Table 1 is the relationship between resource size, depth and the time taken from discovery to production. Deposits at or very near the surface have generally been mined irrespective of size. However, the correct combination of depth and size must be present for deeper open pit mining or underground mining to occur. Hydrogeological and geotechnical factors play important rolls.

Exploration has typically focused on identifying graphite conductors in basement rocks below sandstone cover. Uranium, if present, may be located at or near the intersection of the conductor and the sandstone above. Exploration drilling has progressed from near surface conductors to depths currently approaching 800 m, as likely host environments have been systematically eliminated.

As deeper deposits are exploited, the high cost of shaft sinking or overburden removal dictates that a large resource base is necessary to produce an acceptable return on investment. The often incompetent nature of the ore

TABLE 1. URANIUM EXPLORATION AND MINING HISTORY IN SASKATCHEWAN

Deposit	Discovery year	First production	Elapsed time (a)	Mining method	Resources (t U)[a]	Avg. ore depth (m)
Key Lake						
Gaertner	1975	1983	8	OP[b]	23 000	60
Dielmann	1976	1986	10	OP	45 000	120
Cluff Lake	1969	1980	11	OP/UG[c]	29 000	120
Cigar Lake, Ph. 1	1981	2007	26	UG	89 000	430
Midwest Lake	1978	2009?	31	OP	13 500	230
Dawn Lake	1978	Unmined		?	5 000	140
Rabbit Lake						
Pit	1968	1975	7	OP	19 200	80
B zone	1977	1985	8	OP	11 300	50
A zone	1971	1996	25	OP	6 500	40
D zone	1979	1996	17	OP	2 150	40
Eagle Point	1980	1993	13	UG	25 000	250
McClean Lake						
Jeb	1982	1999	17	OP	2 100	90
Sue A–E	1988–91	2002	14	OP	17 300	60–100
McClean North	1979	Unmined		?	2 700	160
McArthur River	1988	1999	11	UG	215 000	530
Millennium	2000	Unmined		UG	19 300	650
Anne/Collette	1992	Unmined		UG	15 000	750

[a] Author's estimate of previously recovered or current remaining RARs/EARs.
[b] OP: open pit.
[c] UG: underground.

and surrounding waste rock means that conventional ground support systems may not be suitable in underground mines. In addition, the presence of groundwater under potentially high static pressures dictates that costly dewatering or isolation methods are often required. Ore grades may also

dictate extraordinary radiation protection measures. These factors are largely the reasons why deposits such as Dawn Lake remain unmined to date.

Typically, it requires three to six years to adequately define a potentially mineable resource, another three to seven years to determine its economic potential and receive necessary regulatory approvals for mining, and a further two to five years to construct the necessary infrastructure in order to commence mining. Assuming the economic incentive exists to proceed, an extraordinarily fast tracked project would take eight years from discovery to production but more typically 11–15 years are required, particularly if shaft sinking through water bearing sandstone is required. Significant economic discoveries must therefore be made in the next five to eight years to sustain the current Saskatchewan production rates illustrated in Fig. 1.

3.7. Other Canadian deposits

The area with perhaps the best future mining potential is the Thelon Basin in Nunavut. Reasonably assured resources of over 40 000 t U are reported in three deposits; Kiggavik, Andrews Lake and End Grid. These deposits are within 15 km of each other and so could benefit from operating synergies. Average ore grade is approximately 0.42% U and a portion of resources are amenable to open pit mining [5]. Orebodies are hosted in basement rocks very similar to those of the Athabasca Basin. The deposits appear capable of supporting a production rate in the 2000–3000 t U/a range.

Like most new mineral discoveries in Canada's north, mine construction in the Thelon Basin would be logistically challenging. Production prior to 2015 is unlikely due to Canada's regulatory processes and current limited apparent activity by the property owners.

The Elliot Lake area still contains vast uranium resources [6]. Mining commenced in this area in the 1950s, with 12 mines in operation. Most operations were short lived. Mining boomed once again in the mid-1970s with production largely tied to price insensitive contracts. Uranium mineralization is present in quartz pebble conglomerates such as brannerite. The previous ore grades mined averaged 0.09% U. Mining and milling costs were high due to the nature of the mineralization and the abrasiveness of the ore. All mines and mills are now decommissioned with rehabilitation essentially complete. Similarly, some very high cost resources remain at the decommissioned Cluff Lake, Agnew Lake and Pronto mines [7, 8].

The Kitts and Michelin deposits in central Labrador were advanced towards production in the early 1980s, but mine construction was shelved due to weakening uranium prices and an unfavourable regulatory assessment [9].

Exploration potential in the area appears good based upon recent findings. Some potential may therefore exist for future uranium production in the area.

3.8. Summary of Canadian supplies

As illustrated in Fig. 1, it is likely that the RAR in the Athabasca Basin will be largely exhausted by around 2030, and only concerted exploration will prevent this. The Thelon Basin has good potential as a future production centre. Redevelopment of decommissioned mines elsewhere in Canada would probably require very high uranium prices. The exploration potential in Canada remains good, but, based upon recent experience, new discoveries may require in excess of ten years to put into production.

4. URANIUM MINING IN AUSTRALIA

Uranium production in Australia is currently derived from three production centres: Olympic Dam, Ranger and Beverley. The future uranium supply situation is largely shrouded in the uncertainties related to future production rates at Olympic Dam, the potential for a transition of mining from the Ranger deposit to the neighbouring Jabiluka deposit, and the potential longevity of the Beverley in situ leech (ISL) mine. Matters are further complicated by varying local, State and Federal attitudes towards uranium mining.

Details of the more significant uranium deposits in Australia are illustrated in Table 2.

4.1. The Olympic Dam mine

There is great interest in the BHP Billiton concern and their plans for an expansion of the Olympic Dam deposit in South Australia. The mine produced a total of 3730 t U in 2004, along with 224 731 t Cu and by-product gold and silver. A total of 8.9 million tonnes of ore have been milled. The ore grade milled was 2.3% Cu and 0.055% U [10].

The mine currently employs underground blasthole stoping and is analysing the merits of expanding annual production up to 40 million tonnes of ore, albeit at considerably lower ore grades. The net result may be production up to 11 500 t U/a. Open pit mining will probably be used. This will entail the removal of over 350 vertical metres of waste rock above the orebody before significant ore is exposed for mining. This translates into approximately one billion tonnes of waste rock before stripping. Mining equipment procurement

TABLE 2. SIGNIFICANT AUSTRALIAN URANIUM DEPOSITS

Deposit	State	Uranium (%)	Uranium (t)	Classification	Ref.
Angela/Pamela	NT[a]	0.10	5 600	RAR	[12]
Ben Lomond	Q[b]	0.21	4 000	RAR	[13]
Beverley	SA[c]	0.15	10 000	RAR	[14]
Bigrlyi	NT	0.29	2 300	RAR	[15]
Crocker Well	SA	0.04	4 200	RAR	[13]
Honeymoon Well	SA	0.10	5 700	RAR	[16]
Jabiluka	NT	0.45	138 000	RAR	[17]
Kintyre	WA[d]	0.38	22 000	RAR	[14]
Koongarra No. 1	NT	0.68	12 000	RAR	[13]
Manyingee	WA	0.10	6 500	RAR/EAR-I	[18]
Maureen	Q	0.11	2 500	RAR	[13]
Mt Painter	SA	0.08	6 200	RAR	[19]
Mulga Rock	WA	0.10	11 000	RAR	[20]
Olympic Dam	SA	0.034	1 200 000	RAR	[10]
(including only proven reserves)		0.042	310 000	RAR	[10]
Oobagooma	WA	0.10	8 000	RAR	[18]
Ranger	NT	0.20	37 200	RAR	[21]
Valhalla	Q	0.10	44 000	RAR/EAR-I	[22]
Westmoreland	Q	0.12	18 000	RAR	[23]
Yeelirrie	WA	0.18	43 000	RAR	[24]
Yilgarn Block	WA	0.08	5 800	RAR	[25]

[a] NT: Northern Territory.
[b] Q: Queensland.
[c] SA: South Australia.
[d] WA: Western Australia.

and pre-production waste removal costs have been estimated at almost $A2 billion [11].

A fast track approach could result in additional ore starting to be processed in 2010, with 2012 possibly being the first year of fully expanded production. In the meantime, efforts are being made to raise the capacity of current infrastructure to a production rate of 12 million tonnes of ore per annum to modestly increase annual copper and uranium production.

Without the benefit of completed expansion studies, total project capital cost has been independently estimated at $A5 billion, with a range of $A4.3–$A6.3 billion [11]. This does not include identified capital items for sustaining the existing production rate.

At the time of writing, BHP Billiton was completing the takeover of Western Mining Corporation, previous owners of Olympic Dam. This takeover may have an impact on the decision to expand, or at least on the timing and extent of, this expansion. The decision to expand or not could be the single most important factor in uranium supply during the coming decade. Excluding unforeseen events, this mine has sufficient resources to still be producing uranium fifty years from now and it could be the world's dominant producer as early as 2012.

4.2. The Ranger/Jabiluka mine

Energy Resources of Australia's (ERA) Ranger mine recently announced a modest increase in mineable reserves to 37 200 t U [21]. This open pit operation located in the Northern Territory processes approximately two million tonnes of ore per annum with a grade of 0.25% U to produce 4350 t U/a after mill losses (Fig. 4).

Energy Resources of Australia indicates that mining will cease in 2008 but stockpile milling will continue to at least 2011 depending upon annual milling rates. Applying last year's mill performance, for instance, results in mill closure in mid-2012. Reports indicate that EAR-I resources amount to an additional 23 800 t U at a grade of 0.16% U. Depending on the economics prevailing, a fraction of these resources may be upgraded to RARs to allow economic recovery, but this may entail more expensive underground mining.

In order to sustain production after ore exhaustion at Ranger, underground mining of the nearby Jabiluka deposit is necessary. Owing to higher ore grades, approximately one million tonnes of ore per annum (at a head grade of 0.48% U) is necessary to maintain current uranium output. If development is approved, it may take three years to ramp production up to current mill output levels.

Energy Resources of Australia recently agreed to give Aboriginal traditional owners the right to veto mining of Jabiluka. Commencing in 2006,

FIG. 4. Open pit mining at Ranger (courtesy of Cameco Corporation).

ERA may approach the traditional owners every four years seeking approval to mine this deposit [26]. There are no assurances, therefore, that this orebody will be mined in the foreseeable future.

4.3. The Beverley ISL mine

General Atomics operates Australia's only ISL mine. The mine, located in South Australia, produced approximately 920 t U in 2004. Mine performance has improved due to commencement of mining in Central Beverley and "a more focused approach to specific issues relating to each Wellfield" (see Ref. [27]).

Exploration has recommenced at Beverley to potentially increase the resource. Since General Atomics is a private company, few data are available concerning resource recovery rates, operational issues and costs. Previous resource figures suggest that production at current rates is sustainable until approximately 2016. Exploration efforts and higher than anticipated recovery rates may extend mine life.

4.4. Significant unexploited deposits

Currently, deposits located in Queensland and Western Australia cannot be mined due to the positions of the State Governments. Both Governments have been recently re-elected. If State approval for uranium mining is eventually received it will, in most cases, still be necessary to establish Native Title agreements with local Aboriginal communities as well as to satisfy all licensing matters.

Perhaps the two best known deposits in Western Australia are Kintyre and Yeelirrie. Both deposits could support a lower cost production rate of the order of 2000 t U/a using open pit methods. The Yeelirrie deposit is an extensive near surface free digging calcrete deposit containing RARs in excess of 40 000 t U [24]. The ore is only 5–10 m deep and extends for over 9 km laterally.

Rio Tinto has previously assessed the Kintyre deposit. Resources (RARs and EAR-Is) total in excess of 21 150 t U (with regional figures as high as 30 000 t U) [14]. Although a number of potential mining and milling options were assessed, one production scenario envisioned constructing a 1700 t U/a mine and mill which would operate for at least six years. Ore from the Kintyre and East Whale pits would be mined and radiometrically upgraded prior to milling. The disseminated nature of the mineralization creates a number of mining dilution challenges.

The Koongarra deposit, located in Kakadu National Park, had a five year mining moratorium placed upon it in 2000 by the Aboriginal traditional owners. This expired in April 2005, but no efforts are currently underway to once again advance this project. Development has been stalled since the early 1980s. Previously, a low cost 1150 t U/a open pit uranium mine was proposed with an approximate life of ten years. Resources for the Koongarra No. 1 mine are stated as 12 300 t U, with an ore grade of 0.67% U [13].

The Westmoreland deposit in north-west Queensland is actually a series of shallow, largely sandstone hosted, flat lying zones of mineralization that would be readily amenable to open pit mining [23]. Mineralogy varies considerably from zone to zone, which may have an impact on recovery of mills. Little work has been conducted on this property since the 1980s.

Underground mining methods would probably be necessary to exploit the majority of the Valhalla deposit located in north-west Queensland. Relatively high milling costs are likely since the uranium is contained in often difficult to leach brannerite. Acid consuming carbonate and phosphate minerals are also present [28].

Perhaps the only deposit with immediate exploitation potential is Honeymoon Well in South Australia, where ISL mining is planned. Recent

studies have determined that a unit operating cost of US $32/kg U is possible but a commodity price of US $65–80/kg U is probably necessary due to the high unit capital costs envisaged [29]. A mine life of six to eight years with a plant capacity of 350 t U/a has been assessed. This adequately demonstrates a key issue with many unexploited uranium deposits. Although low unit operating costs are possible, high unit capital costs, due to limited resource size, prevent construction.

Table 2 also reveals that although there are numerous unmined deposits in Australia, many lack sufficient resources for low unit capital costs to result and many lack sufficient grade for low unit operating costs to result.

4.5. Summary of Australian supplies

Figure 5 illustrates three possible production scenarios for Australia. The low production case illustrated assumes that the Olympic Dam mine does not expand and that the Jabiluka mine does not proceed. The middle production case assumes that there is a smooth transition in production from the Ranger mine to the Jabiluka mine and that the capacity of the Olympic Dam mine doubles. Finally, the high production case assumes that there is a smooth transition from the Ranger mine to the Jabiluka mine, that the capacity of the

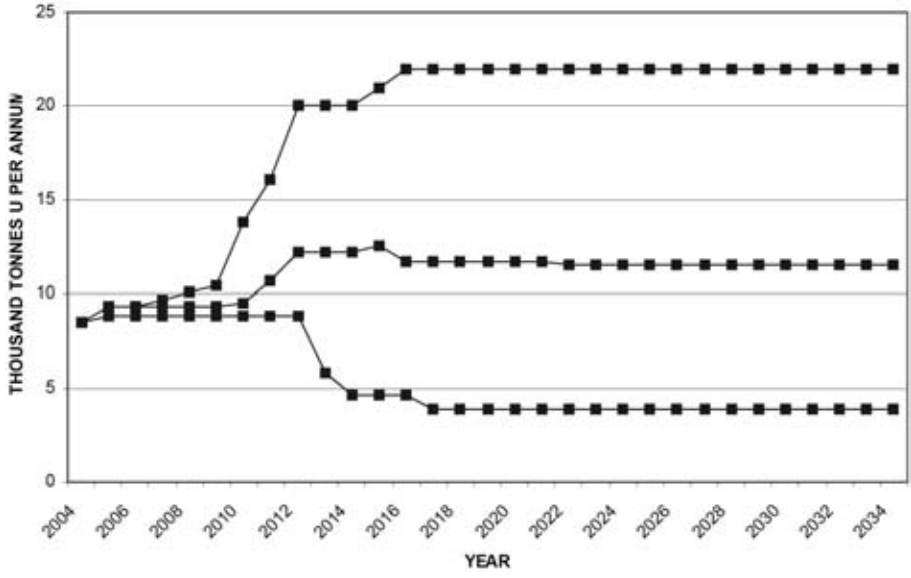

FIG. 5. The low, middle and high Australian uranium production cases.

Olympic Dam mine triples and that remaining deposits are brought on-line in a systematic fashion.

As illustrated, the wide range of potential outcomes confounds those attempting to make reasonable estimates of future Australian uranium supplies. However, it is this puzzle that may determine future supply price. The approach taken in Australia towards uranium mine development creates uncertainty for utilities and suppliers alike. Mines elsewhere may not be developed if there is a real or perceived threat of new low cost supplies from Australia due to a change in policy.

5. URANIUM MINING IN THE USA

The 2003 Red Book reports the resources listed in Table 3 for the USA.

Very little exploration and deposit assessment work has been conducted in the USA for over twenty years. It is often difficult, therefore, to determine the current economic potential of known deposits. Only RARs under US $80 kg/U are of commercial interest at this time. The net result is to greatly narrow the list of immediate production opportunities to less than 5% of the resources listed in Table 3.

Currently, production in the USA is predominantly from ISL mines in Wyoming, Nebraska and Texas. Conventional mining on the Colorado Plateau has also been recommenced recently.

In the early 1980s, the USA was the largest uranium producer in the Western world. There were 44 open pit mines, 200 underground mines, 11 solution mines, 23 conventional mills, six wet process phosphoric acid plants, two copper leach plants and one tailings pond leach operation [31]. Production in 1980 was 16 800 t U. The average grade of the ore mined was 0.1% U. The demise of the industry, largely for competitive reasons in an oversupplied world market, came quickly in the early 1980s. Total production in 2004 for the USA was 885 t U.

TABLE 3. 2003 RED BOOK URANIUM RESOURCES FOR THE USA [30]

Category (t U)	<US $80/kg U	<US $130/kg U
RARs	≈102 000	≈345 000
EARs	839 000	1 273 000
Speculative resources		858 000

A fundamental question to ask therefore is under what price scenario this production could be re-established. The answer is complicated by a number of factors including lack of, or poorly located, milling capacity, the impact of twenty years of inflation, increased regulatory and public scrutiny, exhaustion of lower cost resources and poor availability of skilled personnel, particularly underground miners.

The US consumer price index factor from 1982 to 2004 was 1.96. Therefore, if a mine was closed in 1982 because it could not cover its cash operating costs in a US $50/kg U environment, all other factors being unchanged, this mine would be uneconomic today in a US $100/kg U environment. Infrastructure would have to be re-established in most cases.

Significant previous production was largely located in the Wyoming Basin, the Colorado Plateau (New Mexico, Utah, Colorado and Arizona) and the Gulf Coastal Plain (Texas). The production potential of each area is discussed in greater detail below.

5.1. Wyoming Basin

Past production in Wyoming has totalled over 77 000 t U. Large open pit mines once predominated in Wyoming. Attempts at underground mining had limited success due in large part to poor ground conditions at some locations [32].

Cameco operates the Highland/Smith Ranch ISL mine near Casper. In 2004 this mine produced 470 t U. Reasonably assured resources are sufficient for another twenty years production at this rate. However, EAR-Is are also roughly equivalent to RARs. Additional ISL amenable deposits occur at Gas Hills, the Shirley Basin and the Great Divide Basin. For instance, Cameco reports RARs of 5500 t U for Gas Hills after factoring in relatively poor ISL recovery rates of 65% [4]. Cameco has recently announced their intention to double ISL production in Wyoming, probably by 2010 [4].

Typical ISL cash operating costs are in the US $30–35/kg U range. Construction costs for greenfield deposits are generally in the range US $5–9 million per 100 t U of annual production. At current spot prices, the economic rationale therefore exists for further production increases.

Conventional mining interests are largely centred upon deposits in the Great Divide Basin near Rawlins, but additional exploration drilling and updated studies are probably required so that extraction potential and costs can be fully understood.

5.2. Colorado Plateau: Colorado and Utah

Previous mining in Utah and Colorado centred upon the Uravan mineral belt which straddles the Utah/Colorado border. The remaining deposits are largely controlled by the Cotter Corporation and the International Uranium Corporation (IUC). Cotter reports RARs of 7700 t U, and IUC reports 2110 t U [33, 34]. Deposits are usually mined by labour intensive random room and pillar methods with ramp access from the surface. Typical historic ore grades are 0.18% U and 1.2% V_2O_5.

In 2004, the Cotter Corporation refurbished the mill at Canon City, Colorado, to receive uranium and vanadium ores from their Western Slope mines. Up to seven small mines will be in operation in 2005 (JD-6, JD-7, JD-8, JD-9, SM-18, LP-21 and SR-11). Cotter expects to produce 72 500 t of ore in 2005 and 140 000 t of ore in 2006 at average grades of 0.34% U_3O_8 and 1.84% V_2O_5 containing 200 and 400 t U, respectively [35].

The IUC White Mesa 1800 t/d mill at Blanding, Utah, restarted operation in mid-2005 to process feed sourced largely from Cameco's conversion operations. Combined, the White Mesa and Canon City mills could theoretically produce in excess of 1200 t U/a from the Uravan mines. Total production, however, will probably not exceed 600 t U/a for the foreseeable future due to mining constraints.

The IUC has recently acquired the Tony M mine which is adjacent to their undeveloped Bullfrog deposit. The process of gaining a permit is in progress. No production details have been announced yet.

5.3. Colorado Plateau: New Mexico

Over 115 000 t U have been mined from the prolific uranium deposits in the Grants, New Mexico, area. Mining commenced in the mid-1950s at the large open pit Jackpile and Paguate mines. Mining progressively expanded over the next 30 years as orebodies were discovered and developed over a 150 km long corridor from the outskirts of Albuquerque to the Churchrock area. In 1980, total area milling capacity exceeded 18 000 t/d of ore [31]. A number of factors conspired then to lead to total cessation of production by 1990. Although the collapse in the uranium price was the primary reason, the reduction in ore grade in time combined with exploitation of deeper, more challenging, deposits also played significant roles.

A large resource base remains in the Grants area but the following issues confront resource companies and engineers:

(a) Public acceptance — Local opposition to uranium mining exists. Land access is difficult in many instances.
(b) Lack of area milling capacity — All previous mills have been decommissioned.
(c) Technical risk — Many of the remaining unmined deposits may suffer from high rock temperatures, high groundwater inflow rates and poor ground conditions [36, 37];
(d) Environmental impact mitigation — Tailings disposal methods and the impact of mine dewatering upon the regional groundwater table are two key issues [38, 39].

Deposits under review, partially developed or partially exploited at the onset of the uranium price collapse in the early 1980s include Nose Rock, Mount Taylor, La Jara Mesa, Marquez, Roca Honda, Dalton Pass, Bernabe Montano, Rio Puerco, Churchrock and Crownpoint. These deposits generally have grades from 0.10 to 0.45% U, are from 200 to 1000 m below surface, and contain RARs between 2000 and 40 000 t U.

Uranium Resources Incorporated (URI) has been assessing the ISL potential of the Churchrock and Crownpoint area deposits since the late 1980s but has faced numerous delays in acquiring permits [40].

5.4. Colorado Plateau: The Arizona Strip

Energy Fuels Nuclear Inc. (EFN) began explorations in northern Arizona during 1979 and in the subsequent seven years discovered in excess of 40 breccia pipes containing uranium mineralization. Uranium mineralization is typically located 300–500 m below surface, necessitating underground mining methods. A total of 7300 t U was mined from seven pipes in the 1980s. The ore grade is typically 0.5% U [41].

The IUC, which later acquired EFN's assets, lists 'mineralized material' as 271 000 t with a grade of 0.56% U, (approximately 1500 t U) for the Arizona 1, Canyon and Pinenut mines [34]. Energy Fuels Nuclear Inc. previously stated that an additional 12 pipes were of mining interest. Uranium prices higher than US $80/kg U are thought necessary by the author to justify new feasibility studies.

5.5. Gulf Coastal Plain: Texas

Uranium Resources Incorporated has commenced ISL mining at the Vasquez property. Recoverable RARs are stated as 1000 t U as of March 2004. In addition, the nearby Kingsville Dome property has recoverable RARs of

86 t U as of March 2004 [42]. The extraction rate planned is 230 t U/a, suggesting a further four to five year mine life as of January 2005. Exploration potential is considered good by URI.

In situ leach mining of the Alta Mesa property is reportedly under investigation, with a potential startup to production in 2005. Reasonably assured resources have been variously reported in the past as ranging from 1500 to 3800 t U.

5.6. Nebraska

Cameco operates the Crow Butte ISL mine in Nebraska near Crawford. This mine has been in continuous operation since 1991. Typically, the annual production level is 320 t U. Cameco reports RARs of 2880 t U after recovery losses [4]. This is sufficient for nine years of operation at current rates. Additional resources are located along the Crow Butte trend. Cameco reports area EAR-Is of 5300 t U. Good potential seems to exist, therefore, to extend the life of the mine and expand the annual production rate.

5.7. Other areas of potential economic attractiveness

A partial list of areas with some possible future extraction potential include the McDermitt Caldera in Oregon, the phosphate deposits in Florida and Louisiana, the Northern Piedmont area, and additional areas in the States discussed above.

5.8. Summary of US supplies

In situ leach mining will continue to provide most of the uranium production in the USA. An economic justification exists for additional ISL mining in Wyoming, Texas, Nebraska and New Mexico. Annual ISL production rates approaching 2500–3000 t U/a by 2010 are economically conceivable at current uranium prices.

Output from conventional mining may be limited to 500–1000 t U from the Uravan belt in the immediate future, unless further uranium price increases make other Colorado Plateau deposits and perhaps certain Wyoming ones economically attractive.

In relation to other countries, it can be argued that the USA has been overexplored: the large number of known uranium occurrences do not translate into a large number of future uranium mines.

6. URANIUM MINING IN AFRICA

The African production of uranium in 2004 is estimated at 7100 t U. The Arlit and Akouta mines in Niger accounted for 3282 t U of this total. The Rössing mine produced 3040 t U, and the remaining output was recovered as a by-product of South African gold mining.

6.1. Mining in Namibia

The ultimate fate of the Rössing mine in Namibia remains uncertain. This large open pit mine has been in operation since 1976 and relies upon economies of scale to remain competitive. Ore grades are in the range of 0.03–0.04% U. Rössing currently plays the role of a classic 'swing producer' and supplies approximately 8% of world primary production. The performance of this mine should therefore be closely monitored since its situation and size provide it with certain indirect price setting powers. In other words, should the mine close, the current ability of other suppliers to quickly fill the resulting shortfall would be limited. The uranium price may therefore have to be attractive enough to ensure future mining at Rössing remains economic until such time as sufficient additional lower cost supplies can be developed to replace its output.

Plans for Phase 2 mining, which would have extended the mine life to 2016 were rejected as uneconomic in late 2004 when long term uranium prices were approximately US $50/kg U. Up to a hundred million US dollars of capital expenditures were necessary for mill modifications and additional mining equipment. The mining grade was also expected to decrease by 20%, while the amount of waste stripping needed annually was expected to double [43].

The Namibian dollar has appreciated by approximately 100% since 2002. Since revenues are received in US dollars, the impact of the recent uranium price rise has been largely negated by the strengthening of the Namibian dollar.

Current plans are to continue mining until 2009 with a revised Phase 1 operating plan: "The Phase 1WNT10 plan will serve as an interim mining plan while the Phase 3 mine planning studies continue" [44]. In order to continue mining beyond 2009, "a major challenge is the processing of high calc ore, which will most likely use large volumes of expensive sulphuric acid to leach out the uranium" [44]. Segregation of acid consuming marble from the uranium bearing alaskite ore is necessary, and a solution must be found by 2007 to allow time for mill modifications if necessary.

Paladin Resources has recently completed a feasibility study for the Langer Heinrich surfacial calcrete uranium deposit located 80 km east of Walvis Bay, south of the Rössing mine. Current plans entail a mine processing 1 500 000 t/a of ore at an average grade of 0.074% U, to produce 1000 t U/a for

eleven years and 340 t U/a for a subsequent four years [45]. Ore will be mined by open pit methods. A conventional alkaline leach mill will be constructed.

Recently released findings from the feasibility study list capital costs of 92 million US dollars, and average mine life operating costs of US $37/kg U. At a 225 ppm U cut-off grade, RARs are 14 500 t U and EAR-Is are 13 300 t U [46]. If and when EAR-Is are upgraded to RARs, a longer mine life, higher annual production rate or both may result. Paladin states that production may commence in September 2006; however, financing remains outstanding at the time of writing.

6.2. Mining in Niger

In 2003, Niger was the world's third largest producer of uranium. The first major exploration programme since the 1950s has been initiated recently. In 2006, Niger expects to pass 100 000 t of cumulative uranium production.

Production commenced in 1970 at the Arlit mill constructed by Somair. Production is sourced from several open pits mined to a depth of about 70 m. The uranium is flat lying and sandstone hosted with an average ore grade of 0.3% U. Reasonably assured resources totalled 13 489 t U as of 31 Dec. 2004, which at current production rates of 1125 t U/a are sufficient for over 12 years of production [47].

The neighbouring Akouta mill commenced production in 1978 and was constructed by Cominak. The Akouta mine provides access to several adjacent orebodies at approximately 250 m depth. These are high grade underground uranium orebodies (0.4–0.5% U) extracted predominately by the room and pillar mining method. Reasonably assured resources of 23 626 t U are sufficient for 12 years of production at current production rates of 2000 t U/a [47].

Extensive resources exist in the area, but further drilling and technical assessment are necessary to upgrade these to mineable reserves. For instance, AREVA, the operator of both the Arlit and Akouta mills reports 22 200 t U of RARs for the Arlit concession [47]. The ultimate life of the existing mills is therefore difficult to assess since this is based largely upon decisions about future cut-off grades and the outcome of current exploration efforts. Existing mines are competitive in today's uranium price environment and they benefit from the extensive infrastructure constructed over 30 years ago.

AREVA is also assessing the ISL potential of the Imouraren sandstone hosted deposit in Niger. The mineralization is related to fluvial sandstones of the Jurassic age and is similar to other roll front type deposits. AREVA reports recoverable resources of approximately 80 000 t U, assuming a 70% recovery

rate [48]. These resources are contained in three horizontal zones containing over 130 million tonnes of ore with a grade of 0.11% U for 143 600 t U [49].

6.3. Mining in South Africa

Previously, South Africa was a major supplier of uranium, largely derived as a by-product of gold mining. In 1980, 19 uranium plants treated 76.6 million tonnes of ore to extract 6146 t U, ranking South Africa as the world's second largest producer [31]. Average ore grades were in the 0.015% U range. By 2002, production had fallen to 824 t U from one operation.

AngloGold produces 800–900 t U/a by-product at its Vaal River gold operations. The Great Noligwa mine processes ore with grades of approximately 10 g/t Au and 0.04% U. On the basis of published resources, RARs are sufficient for an additional eight years of uranium production [50].

AngloGold uranium output is expected to increase substantially as mining ramps up at the Moab Khotsong operation. This mine is expected to produce 15.6 t Au annually by 2008 [50]. On the basis of the relationship of gold to uranium grade reported, this may result in new production of up to 600 t U/a for at least 12 years [51].

The price at which other gold producers would be induced to recover uranium as a by-product is somewhat uncertain. Factors include ore grade, the prevailing rand to US dollar exchange rate, the prices of gold and uranium, and the impact of inflation on operating costs. Apart from Vaal River, all other uranium processing plants have been decommissioned, and most gold operations no longer report uranium resources. In addition, owing to appreciation of the rand, capital expenditure constraints are in place at many operations.

The South African RAR and EAR-I resources prior to the recent appreciation in the rand are presented in Table 4.

TABLE 4. 2003 RED BOOK RESOURCES FOR SOUTH AFRICA [30]

Production method	<US $40/kg U RARs	<US $80/kg U RARs	<US $40/kg U EAR-Is	<US $80/kg U EAR-Is
By-product	101 534	165 090	44 433	53 888
Open pit	1 643	22 543	2 974	7 376
Unspecified	19 259	47 283	2 906	5 676
Total	122 436	234 916	49 313	66 940

There appears to be an economic justification for some selective additional uranium by-product production.

The Aflease concern is assessing the viability of mining extensive low grade uranium resources at their Dominion and Rietkuil mines. Audited resources include RARs of 7.5 million tonnes of ore, with a grade of 1.35 g/t Au and 0.052% U (3940 t U), and EAR-Is of 98.8 million tonnes of ore, with a grade of 0.73 g/t Au and 0.048% U (47 750 t U) [52]. Resources are contained in shallow dipping narrow reefs extending from the surface to a depth of over 2000 m. The lack of significant amounts of gold will have a great impact on the economics of the project in comparison with, for instance, Great Noligwa.

Some interest in extracting uranium from gold tailings dumps has also been expressed recently. The ERGO project for extracting gold from tailings previously also extracted uranium. This project will close in 2005, however [50]. For viability of uranium extraction, it is necessary to identify tailings that have not been extensively diluted by non-uranium bearing tailings.

6.4. Mining in Malawi

Paladin Resources has commenced a feasibility study for the Kayelekera deposit in Malawi. This orebody was delineated from 1984 to 1989. A feasibility study was completed in 1990, but the result was negative in the prevailing low uranium price environment. A production rate of up to 400 t U for 17 years was proposed by utilizing open pit mining methods and conventional milling to extract approximately 5800 t U. Global resources are 9840 t U with a grade of 0.13% U using a 0.042% U cut-off [18]. Much of the resource was therefore outside the original limits of the pit feasibility study.

Paladin Resources will be assessing the economics of an annual production rate of 850 t U for ten years.

6.5. Summary of African supply

The presence of extensive uranium resources in Africa is undisputed. Future extraction rates are therefore dependent upon cut-off grade decisions directly related to uranium price. Long term supply contracts to provide mine operators with the necessary sustainable economic return in order to commit to greenfield projects or mine life extensions appear necessary.

7. URANIUM MINING IN ASIA

7.1. Uranium mining in Mongolia

Mongolia possesses considerable uranium resources. Deposits with ISL potential have been discovered as well as mineralization that would probably be mined by conventional open pit and underground methods [53].

7.1.1. In situ leach potential

The IUC's Gurvan–Saihan joint venture project 250 km south of Ulan Batar contains over 8500 t U of RARs with a grade of 0.044% U below the water table [34]. This is considered amenable to ISL mining. Additional resources are present above the water table, and sites with good exploration potential remain.

Exploration in the Choir Depression commenced in the 1950s. The Haraat deposit was identified in the 1970s, and delineation drilling in the 1990s was sufficient to allow resource calculation. Preliminary ISL field studies were conducted for this deposit in 1994.

The nearby Hairhan deposit was discovered in 1996 and constitutes the majority of identified resources. Mineralization is found in paleochannels and alluvial–fluvial systems. Ore depth ranges from 10 m to over 100 m. Preliminary ISL testing was conducted for this deposit in 1998. Further testing is necessary to confirm operating parameters and costs.

7.1.2. Conventional mining potential

The Dornod deposits are located 120 km north of the city of Choibalsan in Dornod Province. Numerous assessments of the Dornod No. 2 and No. 7 deposits have been conducted previously. The No. 2 orebody is an open pit resource with approximately 2900 t U remaining [54]. Mining of this pit commenced in 1988 with almost one million tonnes of ore with a grade of 0.092% U extracted. Most of this ore was transported by rail to the hydrometallurgical plant at Krasnokamensk. Heap leaching of the remaining resources has been investigated recently.

The Dornod No. 7 deposit was previously extensively explored with approximately 20 km of underground workings. The main orebody is flat lying, approximately 30 m thick and lies 400 m below surface. Three access shafts have been completed.

Recent studies have indicated that the No. 7 orebody has recoverable resources of 7400 t U, with a grade of 0.30% U [54]. Mining of this resource

over a nine year period has been proposed previously. Updated studies would be necessary to determine the costs of extraction.

At the nearby Gurvanbulag deposits, previous Russian exploration programmes outlined RARs of over 15 000 t U with a grade of 0.152% U [55]. Over ten kilometers of underground development were completed. On the basis of available information, the inclined room and pillar technique would be used as the primary mining method. Combined with the nearby Dornod deposits, therefore, the area may possess sufficient resources to support conventional mining operations, perhaps feeding a centralized mill at 1000–1500 t U/a. Without updated studies and an understanding of underground mining costs in Mongolia, however, it is difficult to comment upon likely production costs from this area. Considerable potential exists for area exploration.

7.2. Mining in India

Uranium production in India is used to supply the national programme and is therefore not sensitive to market price. Indian uranium resources are listed in Table 5.

Uranium production commenced in 1968 by the Uranium Corporation of India. A total of four underground mines are currently in operation in Singhbhum East district of Jharkhand: Jaduguda, Bhatin, Narwapahar and Turamdih. Locations are illustrated in Fig. 6. Ore from these four mines is treated at Jaduguda, where the milling capacity is 2100 t/d. The ore grades average less than 0.1% U. A new 3000 t/d mill at Turamdih is under construction to process ores from the Turamdih deposit and the planned open pit Banduhurang deposit.

Two additional deposits are planned for mining; one deposit is located at Lambapur-Peddagattu, in Andhra Pradesh, and one at Domiasiat in Maghalaya. Two mills, with capacities of 1250 and 1500 t/d, respectively, will be built.

TABLE 5. URANIUM RESOURCES IN INDIA [56]

Resource category	Resource (t U)
RARs	54 600
EAR-Is	25 300
EAR-IIs	15 500
Speculative resources	17 000

FIG. 6. Important Proterozoic and Phanerozoic basins of India as target areas for uranium exploration [56].

The World Nuclear Association estimates recent annual production levels for India of 200–230 t U [57].

7.3. Mining in China

China's growing reliance upon nuclear energy has meant steadily increasing exploration activity. Deposits amenable to ISL are the primary exploration target.

Uranium resources in China total 77 000 t U [30]. Chinese RARs and EAR-Is are listed in Tables 6 and 7, respectively. An additional 7700 t U is in the EAR-II and SR categories. The majority of resources are located in Jiangxi province, Guangdong province and Xinjiang autonomous region.

Currently, China has five active production centres, as listed in Table 8. Production at a new mine, in the Fuzhou area, is scheduled to commence in 2006 or 2007. The capacity of this new mine will be 200 t U/a.

In situ leach pilot plant testing on the Shihongtan and Dongsheng deposits is planned or ongoing. This may result in new production centres.

TABLE 6. REASONABLY ASSURED RESOURCES IN CHINA (TONNES URANIUM OF IN SITU RESOURCES) [30]

Production method	<US $40/kg U	<US $80/kg U	<US $130/kg U
Underground mining	10 050	12 050	12 050
ISL	3 000	7 000	7 000
Heap leaching	23 450	29 750	29 750
In place leaching	400	400	400
Total	36 900	49 200	49 200

TABLE 7. ESTIMATED ADDITIONAL RESOURCES (CATEGORY I) IN CHINA (TONNES URANIUM OF IN SITU RESOURCES) [30]

Production method	<US 440/kg U	<US $80/kg U	<US $130/kg U
Underground mining	3 400	7 400	7 400
ISL	0	3 000	3 000
Heap leaching	2 600	7 700	7 700
In place leaching	2 000	2 000	2 000
Total	8 000	20 100	20 100

TABLE 8. CURRENT URANIUM PRODUCTION CENTRES IN CHINA [30]

Production centre	Fuzhou	Chongyi	Yining	Lantain	Benxi
Mining method	UG[a]	UG	ISL	UG	UG
Deposit type	Volcanic	Granite	Sandstone	Granite	Granite
Size (t/d ore)	700	350	n.a.	200	100
Capacity (t U/a)	300	120	200[b]	100	120

[a] UG: underground.
[b] Expansion to 300 t U/a planned.

7.4. Summary of Asian supplies

The potential exists for Mongolia to become a significant producer of uranium should ISL and conventional mining projects currently under assessment prove economic. Both India and China will expand uranium production for domestic requirements.

8. SUMMARY OF MARKET SUPPLIES

The key variables to determining primary uranium supply on a 10–15 year horizon include:

— Whether Olympic Dam will expand or not;
— Whether or when mining will occur at Jabiluka;
— Whether additional mines will be permitted to enter production in Australia;
— The success of the Cigar Lake production ramp-up;
— Future mining plans in Africa at numerous locations;
— The ramp-up and ultimate steady state rate of production in Kazakhstan and the Russian Federation (not discussed in this paper);
— The ability to establish Mongolia as a significant producer.

Attempting to determine future uranium prices largely entails accurately predicting the outcomes of the above key variables. Since future supply levels are largely dictated by the decisions and actions of a few suppliers and related governments, this is not a simple exercise to complete.

Beyond the 15 year horizon, there is a significant impact on the supply of uranium from:

— The success of exploration in the Athabasca Basin and elsewhere;
— The cost and ability to re-establish mine production in previous centres such as New Mexico and Elliot Lake;
— By-product production costs at phosphate and South African gold mines;
— The ability to develop additional mines in Australia.

Numerous lower cost production centres will exhaust known resources in the next three to twenty years. Concerted exploration will be required to replace these resources. Higher cost sources will be required if exploration is not successful and access to known potentially low cost deposits remains restricted:

"No engineer can approach the prospective value of a mine with optimism, yet the mining industry would be non-existent today were it approached with pessimism. Any value assessed must be a matter of judgement based on geological evidence. Geology is not a mathematical science, and to attach a money equivalence to forecasts based on such evidence is the most difficult task set for the mining engineer" (see Ref. [58]).

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URANIUM MARKET IN THE CONTEXT OF EXPLORATION AND PRODUCTION ACTIVITIES

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Abstract

Increasing quantities of uranium fuel are required in order to meet the requirements of an expanding nuclear industry. These needs cannot be met, however, without a corresponding increase in the price of uranium. The market is complex, however, as newly produced natural uranium must compete with a variety of secondary sources such as inventories, converted nuclear weapons, reprocessed fuel, enrichment tails and mixed oxide fuel. During recent years, this competition has forced prices to record lows from which a substantial recovery is now underway. Market prices for uranium have a direct impact on both exploration and production, the activity in which is proportional to price. Exploration will occur only when the value of a potential discovery is perceived to be greater than the cost of exploration. Such perceptions are unlikely during periods of sustained low prices. High prices, however, can cause an exploration boom such as that seen in the late 1970s. Production, in the current market driven uranium industry, will occur only when revenues are seen to exceed the cost of production. As prices rise and fall, both production and the list of producing deposits will increase and decrease, respectively. This process gives rise to the 'marginal producer' concept of market analysis and forecasting. Through this concept, deposits that might contribute to future production in a rising market can be identified and their development sequence predicted.

1. THE MARKET FOR URANIUM

Uranium is currently sold only for use as nuclear fuel. This is in contrast to most other metals which have, typically, multiple uses and multiple markets. The market for uranium is small; a total of perhaps 100 buyers worldwide. However, it is a very important market as nuclear power accounts for almost 20% of the world's electricity supply.

Figure 1 illustrates the total value of the market for uranium in terms of both production and consumption.

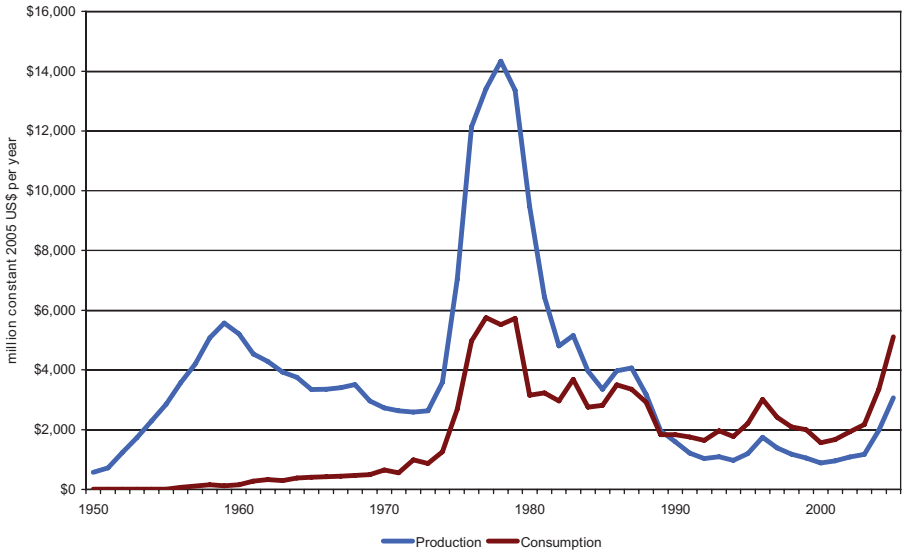

FIG. 1. Uranium market value.

In characterizing the uranium market, it is necessary to review the balance between demand and supply, which, in turn, dictates prices, which, in turn, provide the impetus for exploration and production.

1.1. Demand for uranium

Demand for uranium comprises the fuel needs of approximately 440 nuclear reactors worldwide. A 1000 MW reactor, for example, will require some 500 000 pounds (225 000 kg) of natural uranium per year. Current world uranium requirements are approximately 175 million pounds (79.75 million kg) U_3O_8 per year.

Consumption of uranium is increasing both because of the increasing number of reactors and because of an increasing reactor utilization factor. Such utilization factors were typically in the range of 60–80% only a decade ago, but the profit incentive provided by deregulation has now increased most utilization factors into the 85–95% range.

Figure 2 illustrates both past and projected consumptions of uranium.

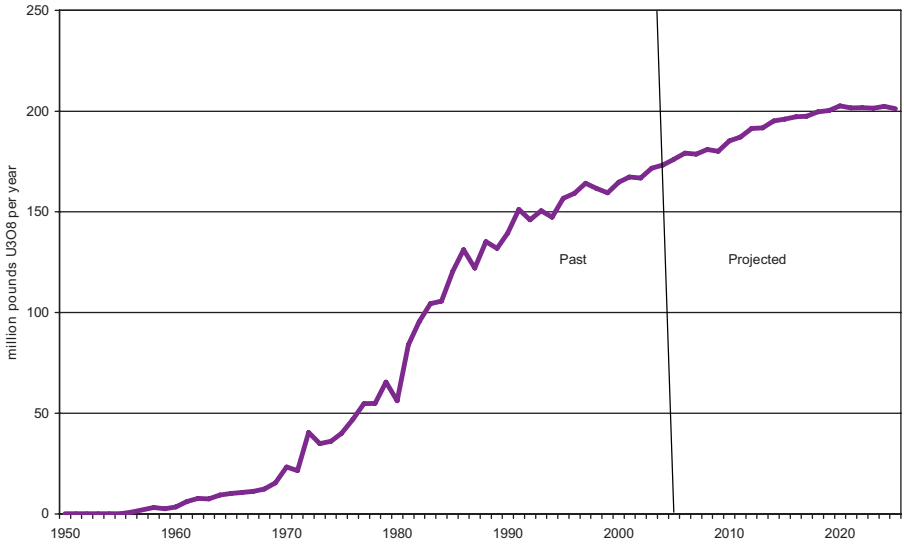

FIG. 2. Uranium consumption: past and projected.

1.2. Supply of uranium

The supply of uranium has progressed in just a few years from a simple country by country oriented primary production and inventory basis to a complex system of international trade in interrelated fuel components. This transition derived mainly from the dissolution of the former Union of Soviet Socialist Republics (USSR) and a subsequent agreement between the Russian Federation and the United States of America (USA) for blending down Russian nuclear weapons into nuclear fuel.

Figure 3 shows historical uranium production in comparison with consumption.

Overproduction in the early years of the atomic age produced a very large inventory of material that is still being drawn down. This inventory constitutes the basis for most current secondary supplies.

1.3. Primary uranium production

Primary production of uranium currently includes 42 production facilities in 18 countries. Of these facilities, 20 are conventional mine–mill operations, 18 produce uranium by means of in situ leaching (ISL) and four produce uranium as a by-product. Current primary production is about 100 million pounds

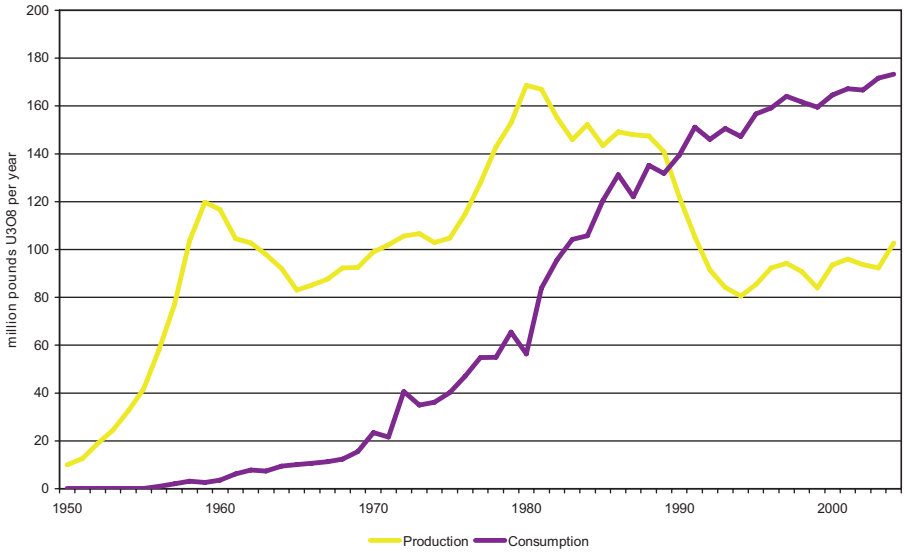

FIG. 3. Uranium production and consumption: historical.

(45 million kg) U_3O_8 per year; just under 60% of the world's total current requirements.

Difficulties attend the process of bringing new uranium mines into production. Financing has been difficult to obtain not only because of recent low prices but also because of the controversy surrounding any endeavour in which radioactivity is involved. Environmental controls continue to become more stringent. Anti-nuclear activism remains a serious impediment to uranium development, but the increasingly good record of nuclear power in general and its ability, in particular, to reduce greenhouse gases are making the task of anti-nuclear activists more difficult.

1.4. Secondary uranium supply

Primary uranium producers have faced significant competition from secondary supplies for more than two decades. Government, utility and supplier inventories of previously produced uranium have been a major feature of the market in recent years, but are now in rapid decline. Current secondary supply features nuclear fuel blended down from Russian nuclear weapons. In 1993, the Governments of the Russian Federation and the USA reached agreement for the conversion of approximately 40% of the Russian nuclear

weapons arsenal to nuclear fuel. By 2004, this process had removed some 8000 warheads from the Russian arsenal and was supplying the world with over 20 million pounds (9 million kg) U_3O_8 equivalent per year, some 12% of the world's fuel needs. The agreement expires in 2013, but it is anticipated that the process will continue if only for the fuel needs of the Russian Federation itself. The progress made by the USA in this regard is slow, but it is to be hoped that the example of Russian disarmament will prove to be compelling.

Other notable sources of secondary supply include reprocessing of enrichment tails, reprocessing of spent fuel and utilization of the plutonium component of spent fuel in mixed oxide (MOX) fuel.

Most nuclear reactors require an enriched uranium fuel of 3–5% ^{235}U. Natural uranium contains only 0.711% of ^{235}U. Current enrichment processes can efficiently recover only about 60% of the ^{235}U contained in natural uranium. When enrichment costs are quite low, as is the case currently in the Russian Federation, a higher percentage can be recovered economically. Spent fuel contains both unutilized uranium and a fission product of potentially usable plutonium. While reprocessing of spent fuel is costly, it does avoid the substantial costs associated with disposal.

Over time, the balance between primary production and secondary supply has changed markedly and has had a major impact on prices. Figure 4 shows this balance.

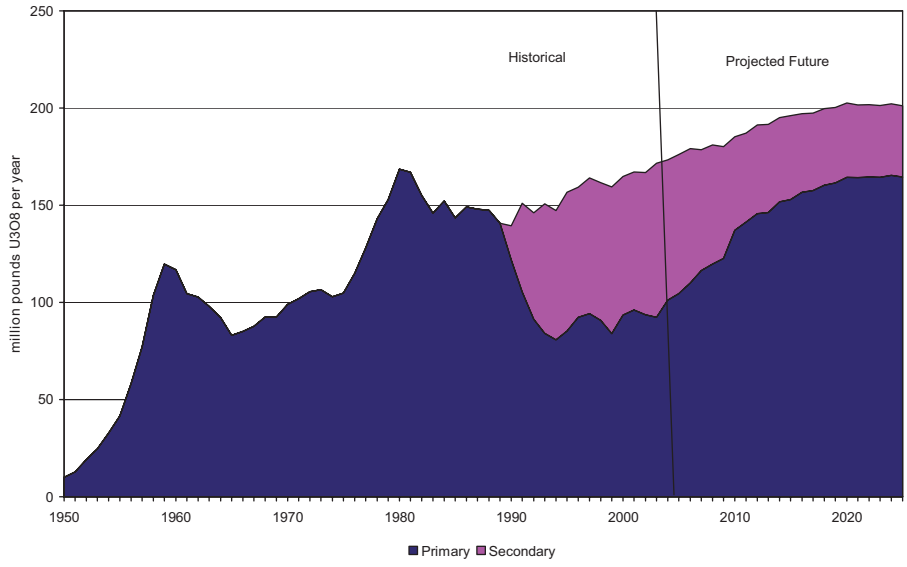

FIG. 4. Primary and secondary uranium supplies.

2. URANIUM PRICES

Uranium is typically priced and sold on a US dollar per pound U_3O_8 basis. Historical prices have been relatively volatile, as shown in Fig. 5.

Early prices in the late 1940s and the 1950s were driven by cold war needs for nuclear weapons. As those needs were more than fulfilled by the early 1960s, prices declined until the rapidly escalating need for nuclear fuel pushed prices up once more. Note, however, that US Government policy regarding enrichment scheduling in the mid- to late-1970s was a major factor in the record high prices during that era.

Most uranium is sold under term contracts of three to five years. Prices, however, are set mainly by spot market sales of which there may be only a few per month and of which not all may be reported accurately. Thus, uranium prices are not fully transparent, a disconcerting element to many market participants. These prices, nevertheless, are closely followed and are the controlling factor in most plans for exploration and production.

2.1. Uranium exploration

Exploration for uranium, as is the case for most minerals, occurs when the opportunity for profit is seen to exceed the cost. This situation, not surprisingly,

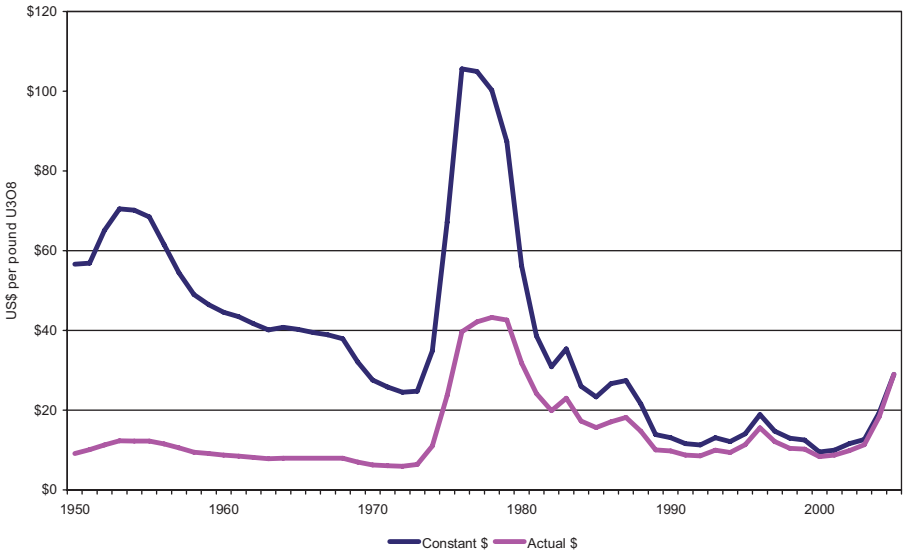

FIG. 5. Historical uranium prices.

occurs mainly during periods of high prices. However, several factors in addition to price influence exploration decision making including: geological, political and environmental considerations. Even in a period of relatively high prices, uranium exploration may be precluded in certain areas by these factors.

2.2. Geological factors

Uranium exploration was driven initially by the Governments of the USA, the former USSR, United Kingdom and France — the 'big four' nuclear powers. Much of the world was explored as a result of programmes sponsored or underwritten by these Governments. Much terrain was explored and many discoveries were made and catalogued. These programmes identified a number of geological situations favourable for the accumulation of uranium. Such situations are not, however, evenly distributed throughout the world. Canada, Australia, central Asia, central and southern Africa, the eastern Russian Federation and the western USA seem to have most of the better uranium resources. From a geological standpoint, these areas remain prime targets for additional exploration.

Figure 6 provides a summary of world uranium resources. These resources amount to almost a 40 year supply at the current rate of

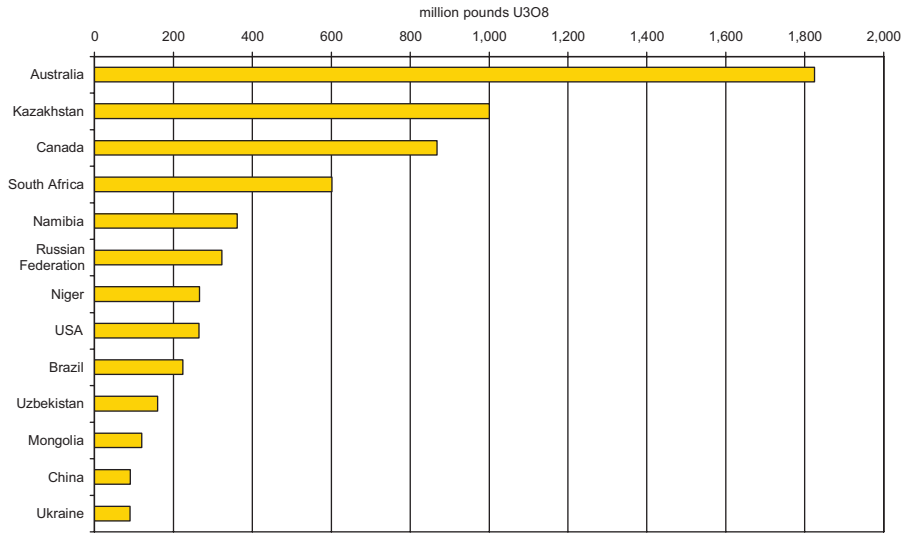

FIG. 6. Uranium resources: reasonably assured resources (RARs), IAEA 2003: <US $30/lb U_3O_8 (US $66/kg)

consumption. Even so, substantial portions of these resources may be unavailable due to political factors, environmental concerns or low prices.

2.3. Political factors

Not all identified resources are available for near term development and production. Much of Australia's rather large resource base may be precluded from development by anti-nuclear State Governments that can effectively forestall the permitting process. Australia's Federal Government when controlled by the Labour Party limited uranium mining to just three specific mines. Brazil has large resources, but most are reserved for domestic use. Both British Columbia and Newfoundland in Canada have banned uranium mining and/or exploration in the past. Similar bans were enacted in the US states of Montana and Virginia. Germany also prohibits uranium mining.

2.4. Environmental factors

While few uranium deposits and few highly promising uranium terrains are located in areas of high population density, several uranium deposits are located in environmentally sensitive areas that may preclude their development. Jabiluka and Koongarra are both large and rich but are surrounded by Kakadu National Park in Australia's Northern Territory. Potential development of Jabiluka has been highly contentious not only because of the park but also because of anti-nuclear sentiment among a portion of the local Aborigines; a sentiment that has been exploited by non-local interveners and activists.

Notwithstanding the fact that a historic uranium mine is located only a few hundred metres from the headquarters of Grand Canyon National Park, development of many high grade breccia pipes in northern Arizona, even if outside the park boundaries, is unlikely due to anti-nuclear activism in the region.

2.5. Market impact on exploration

What is the impact of market forces on exploration for uranium? Figures 7 and 8 provide potent evidence of the direct relationship between uranium prices and uranium exploration activity. Higher prices promote increased exploration.

In both cases, it is notable that the response of exploration to price was relatively prompt and seems to have lagged less as time progressed.

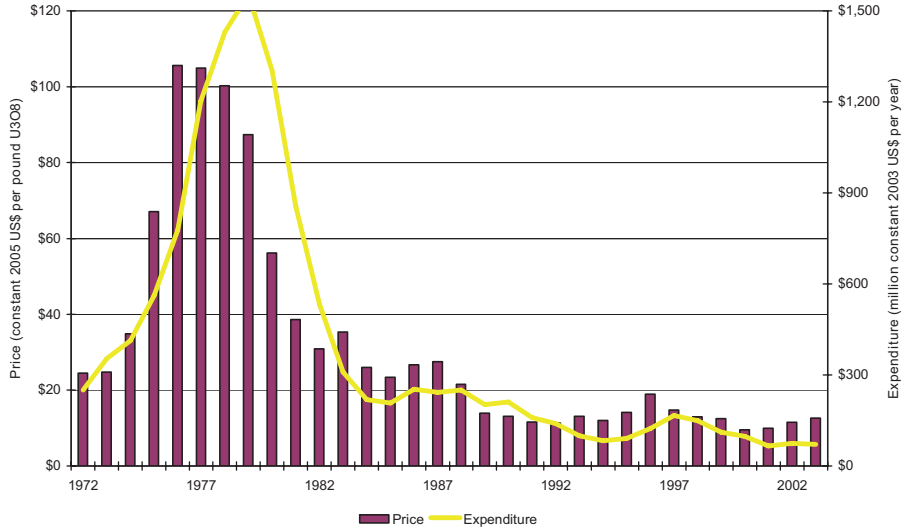

FIG. 7. Relationship between U_3O_8 price and expenditure on exploration for market economy countries.

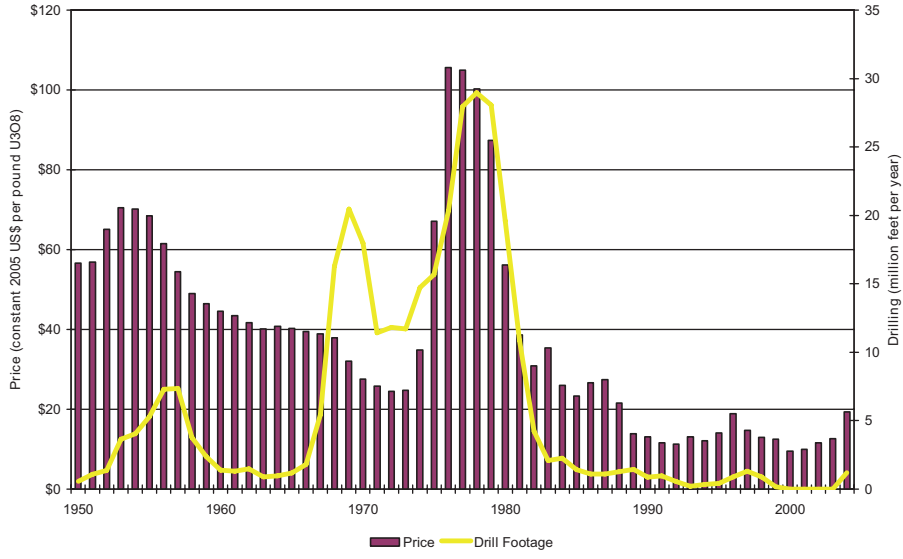

FIG. 8. Relationship between U_3O_8 price and amount of US uranium exploration drilling.

Anomalous drilling activity in the late 1960s and early 1970s was a result of legislation allowing private ownership of uranium.

Exploration expenditures in Australia have also been closely related to prices, as shown in Fig. 9.

Note the time lags in both Figs 8 and 9. This delay in correspondence is due to the fact that most uranium sales are based on long term contracts of three to seven years duration.

Uranium prices are currently quoted and published in the USA in US dollars and have a direct effect only on US activities. The impact of price on activity in other countries ebbs and flows with exchange rates. Recent uranium price increases and major swings in exchange rates versus the US dollar have had different impacts in different countries. These differences are illustrated in Fig. 10.

Thus, while uranium prices in US dollars have moved up by almost 50%, Canadian dollar prices have moved up by only 22%, Australian dollar prices show a net gain of 4% and South African rand prices have dropped by almost 20%. The net result of all these changes will be to strongly encourage US exploration while offering much less encouragement in other countries.

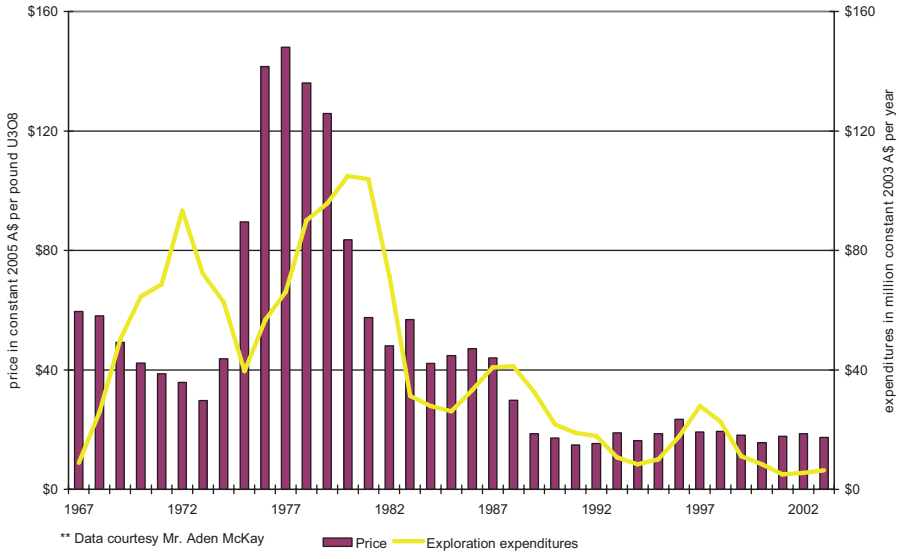

FIG. 9. Relationship between U_3O_8 price and Australian uranium exploration expenditure.

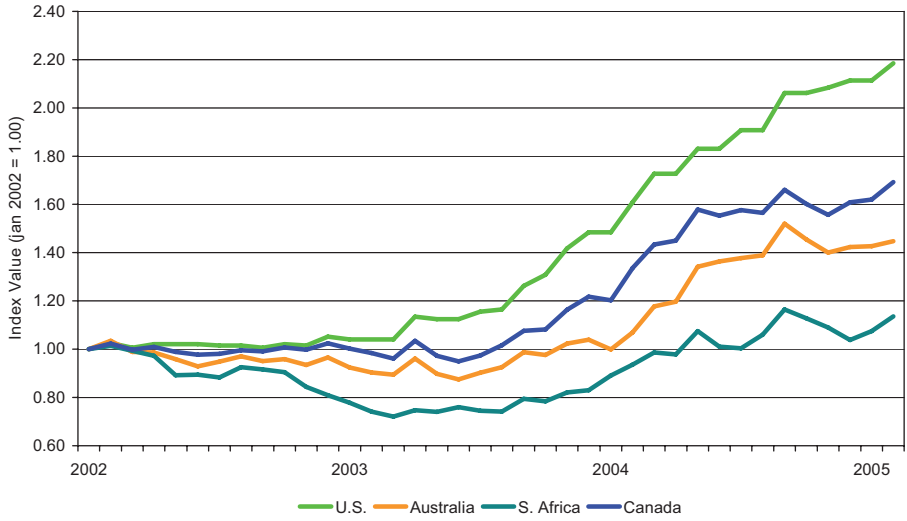

FIG. 10. Uranium prices in the currencies of selected countries.

3. URANIUM PRODUCTION

Uranium production is afflicted by the same political and environmental constraints as uranium exploration, only to a greater degree. Nuclear power, elements of the nuclear fuel cycle including uranium production, and radioactivity in general are complex technical topics where the typical level of public understanding is quite low, and where the same lack of understanding by the media has generated and continues to support a baseless fear of all aspects of radioactivity. When one compares, for example, the total number of fatalities attributable to commercial nuclear power reactor accidents in the world for all time, 41, with the number of fatalities from pneumoconiosis in the USA alone for the period from 1968 to 1999, in excess of 68 000, the fallacy of public and media concern about radiation can be seen clearly.

Much of the political and environmental concern with regard to uranium production relates to low level radioactive wastes such as mill tailings that are produced from uranium ores mined by both open pit and underground methods. In situ leach production of uranium results in only very minor solid wastes, but concerns about groundwater contamination come to the fore in this case.

Historical world uranium production and recent production by method are presented in Figs 11 and 12, respectively.

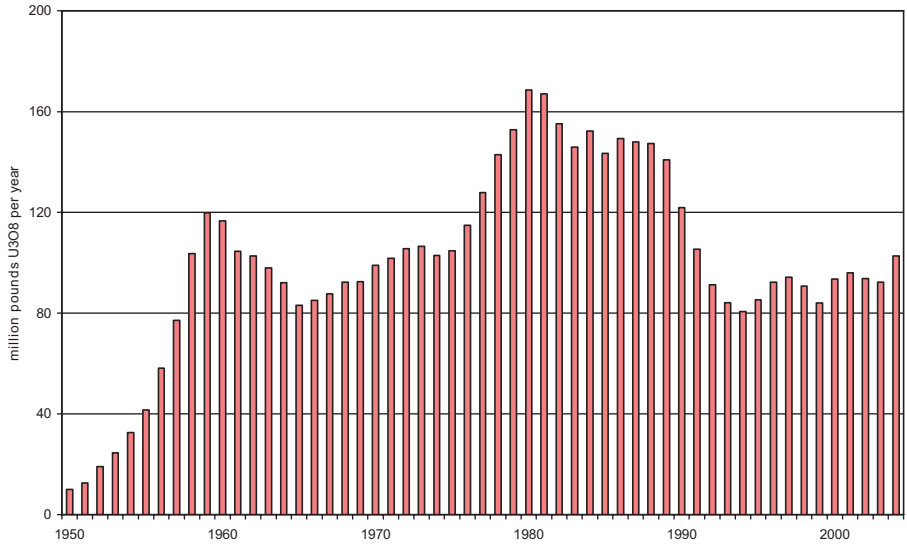

FIG. 11. Historical world uranium production.

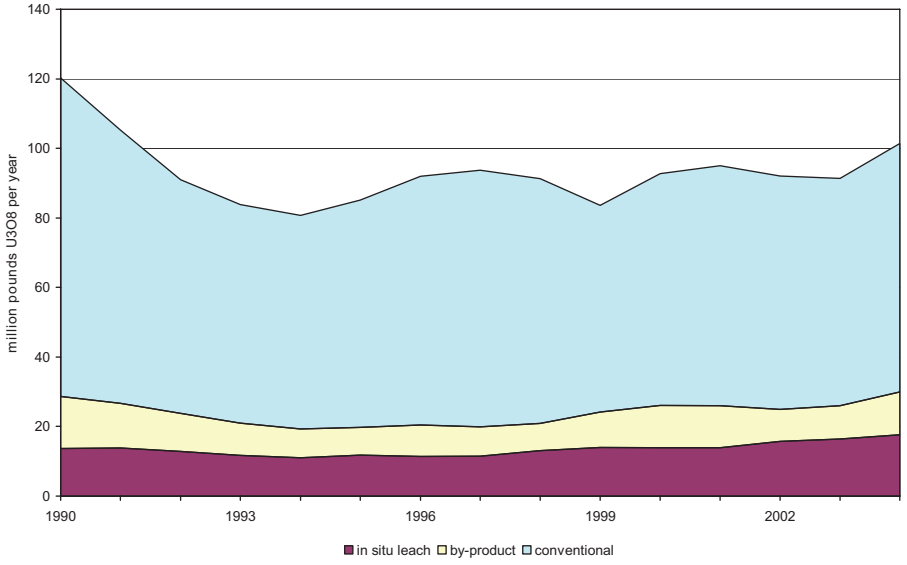

FIG. 12. Recent uranium production by method.

3.1. Impact of market on production

Ore grade is the dominating factor in the cost of uranium production. Higher grade ores have an intrinsic economic advantage over lower grade ores, since most production costs relate much more to moving and processing a given quantity (US tons) of ore than to the amount of uranium contained within that quantity. High grade ore deposits are quite rare. In fact, one anticipates that the distribution of ore deposit grades might well be log-normal; i.e. an abundance of low grade deposits and a constantly decreasing number of progressively higher grade deposits. It follows, therefore, that as prices rise, an increasing number of lower grade deposits will become economically exploitable.

This thesis is borne out by Fig. 13, which relates historical US uranium production with historical uranium prices.

It is interesting to note the lag time between higher prices and production for most of the period. Lag times can vary substantially, from as little as 6–12 months for projects on standby, to 1–3 years for expansions, 3–5 years for development of known deposits and 10–15 years for new discoveries.

This theme is echoed by both the production and the number of uranium producers in South Africa, as shown in Figs 14 and 15.

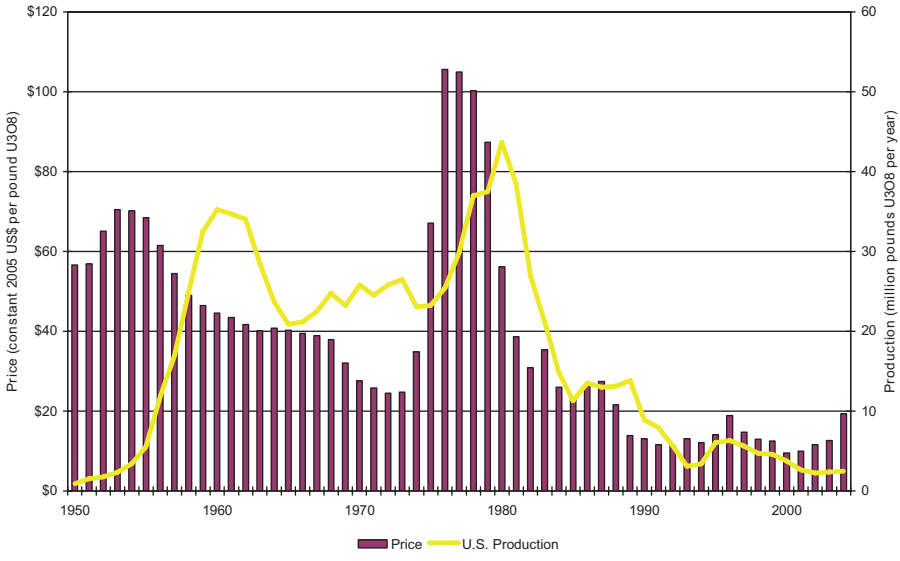

FIG. 13. Relationship between US uranium production and price.

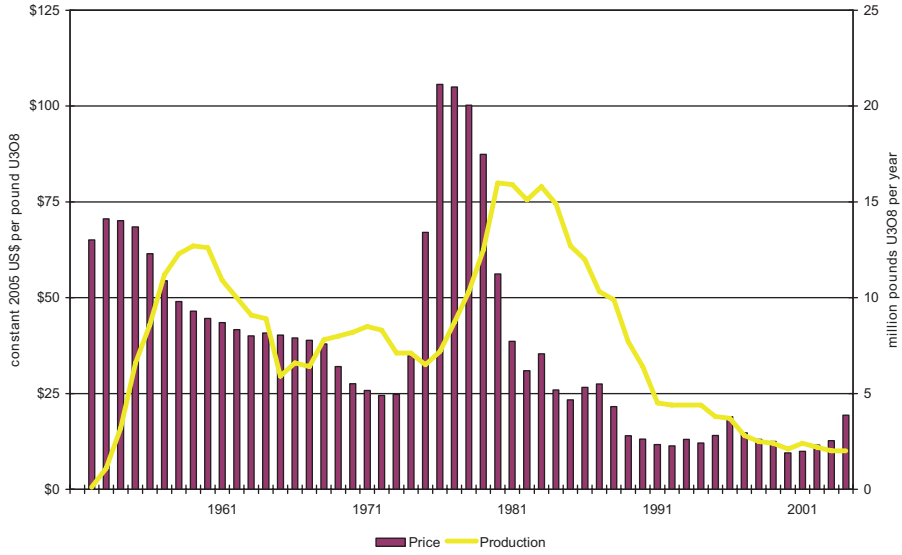

FIG. 14. Relationship between South African uranium production and price.

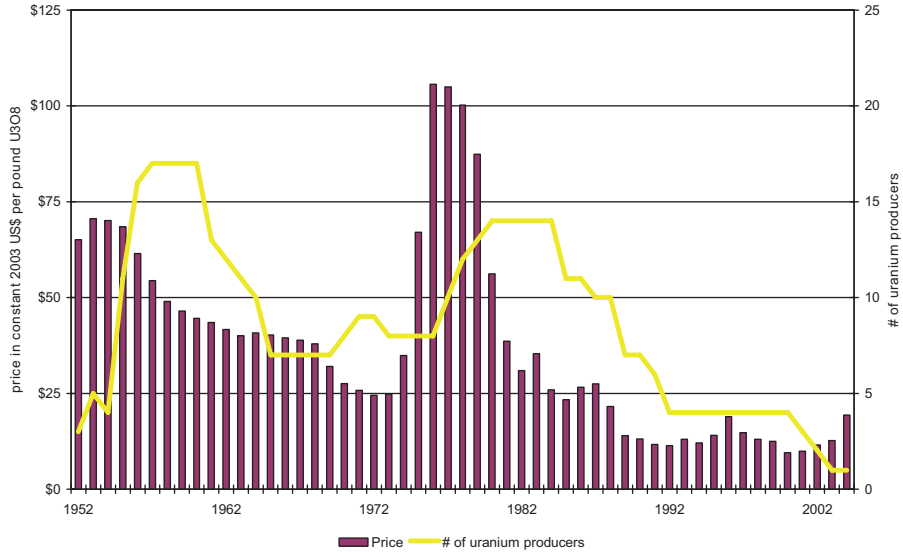

FIG. 15. Relationship between the number of South African uranium producers and price.

4. OUTLOOK

In assessing the future of the uranium industry, it is necessary to consider production capability versus cost of production and projections for future prices.

4.1. Production capability versus cost

The correspondence between price and production allows compilation of an assessment of future production capability as indicated in Fig. 16, where each point on the curve represents an estimated production cost for an actual deposit.

In this particular analysis, cost and price are considered to be synonymous. Thus, at a sustained price of US $20 per pound (US $44 per kg) U_3O_8, a production of 150 million pounds (68 million kg) U_3O_8 per year is seen to be achievable.

4.2. Future uranium prices

International Nuclear Inc. (INI) prepares annual assessments of future uranium prices based on the balance between supply and demand in addition to considerations of production capabilities and costs, as shown in Fig. 16.

The projection made by INI in 2004 for future uranium prices for the base case scenario of supply and demand is shown in Fig. 17.

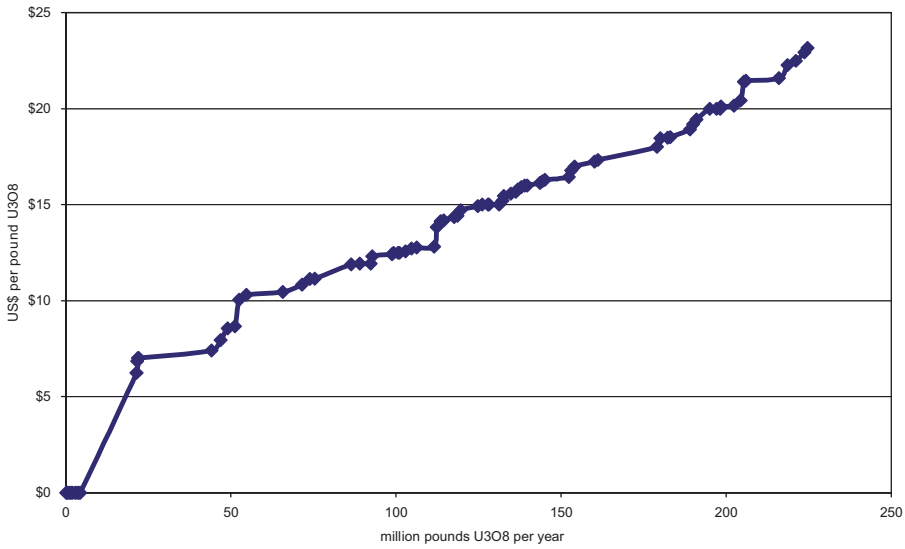

FIG. 16. Relationship between production capability and cost.

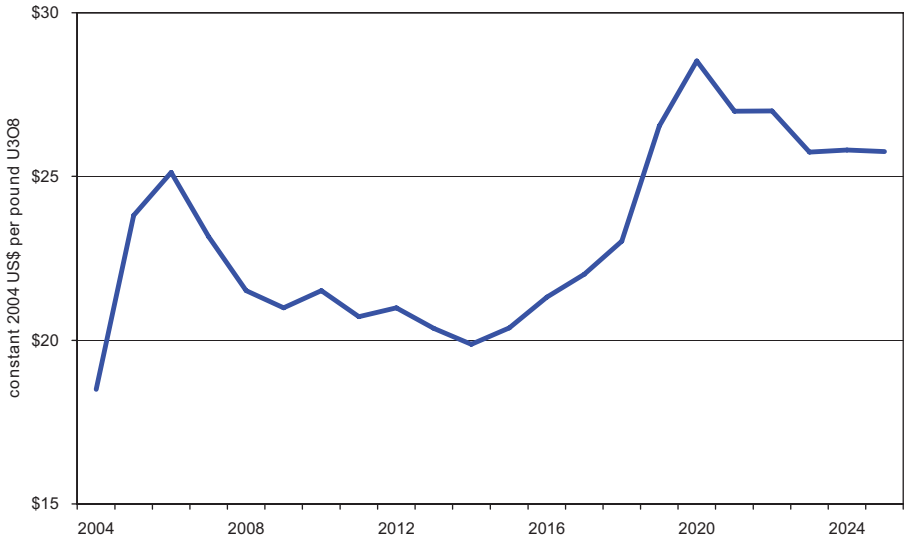

FIG. 17. The forecast made by INI in 2004 for uranium prices.

4.3. Uranium exploration

Increasing prices for uranium portend an increase in uranium exploration. On the basis of the correlation between historical prices and historical exploration expenditures, future expenditures, worldwide, can be expected to increase, from less than US $100 million at present to over US $230 million by 2010 and over US $350 million by 2020, as shown in Fig. 18.

4.4. Uranium production

As with exploration, uranium production can be expected to increase with increasing prices. An increase in price is needed in order to provide the incentive for new production as the availability of secondary supply wanes. Figure 19 provides an illustration of world uranium production, both historical and projected.

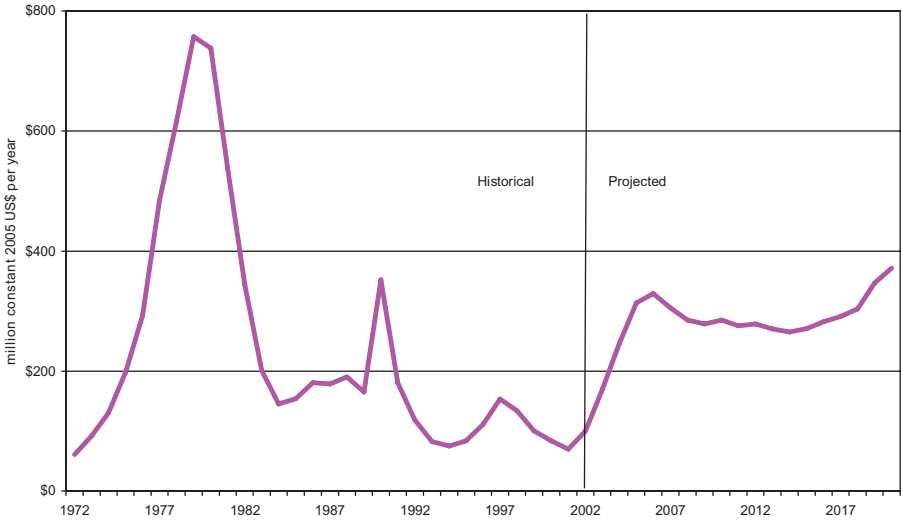

FIG. 18. World uranium exploration expenditures: historical and projected values. Note that the expenditure in the former USSR is not included.

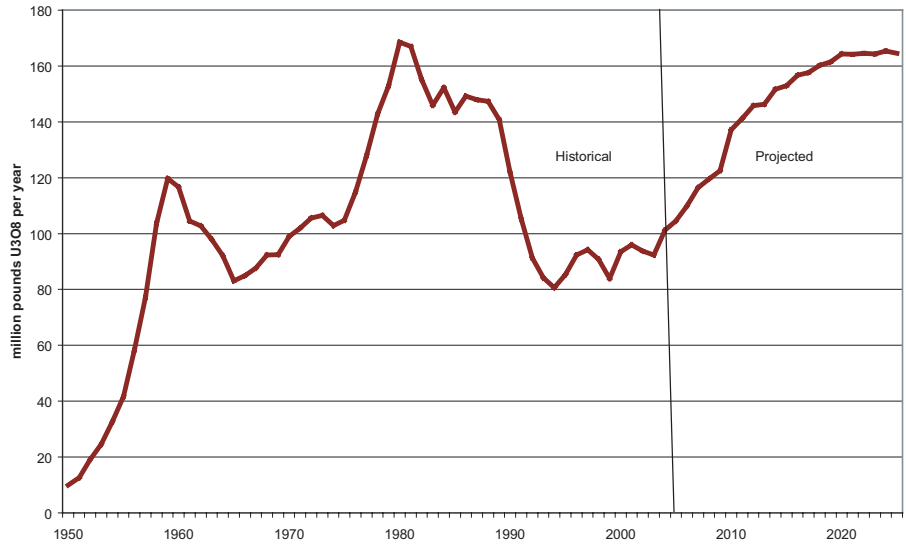

FIG. 19. World uranium production: historical and projected amounts.

5. CONCLUSIONS

Uranium exploration and production show a distinct relationship with uranium prices. As prices rise and fall, so do exploration and production. Prices are at a higher level than in the recent past and can be expected to remain relatively strong for some time. As a consequence, both exploration and production are also expected to remain at higher levels.

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WORLDWIDE URANIUM EXPLORATION AND MINING
Status and related challenges

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Abstract

Since early 2003, following two decades of depressed market conditions, there has been a revival in the uranium industry. The need for nuclear power as a crucial component for future sustainable development is regaining international acceptance. The steady uranium price increase has triggered a worldwide boom in exploration, with major companies redeploying and intensifying their exploration efforts. In addition, smaller companies have entered the market because of the opportunity to easily finance their exploration activities through stock markets. The most spectacular example of this is to be found in the Athabasca Basin in Canada, where four companies were actively exploring in 2003 but now there are over 50. This renaissance in uranium exploration is welcome and necessary. World production has been stable for many years. However, most current supply sources will come to an end by between 2015 and 2030 or continue to decline. They need to be replaced, and new resources will have to enter production to meet forecasted demand. After 20 years of recession and related low key exploration activities, the uranium industry finds itself with an ageing workforce and loss of expertise, staff and know-how. Therefore, recruitment, education, training, R&D and consolidation, as well as the availability of contractors to provide drilling and geophysical services, are serious issues that must be addressed in the very near future. Meanwhile, competition for qualified personnel in the uranium industry is increasing because other sectors within the mining and energy industries (e.g. gold, base metals, oil and gas) are experiencing the same boom and related challenges. The boom has also created more speculative activity. However, speculators are unlikely to survive for very long, because exploration is a long term effort requiring investment over many years.

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1. INTRODUCTION

Since early 2003, following two decades of depressed market conditions, there has been a revival of the uranium industry that has become progressively stronger. The need for nuclear power as a crucial component for future sustainable development is regaining international acceptance, becoming a major issue among politicians and the public.

The steady uranium price increase has triggered a worldwide boom in exploration, with major companies redeploying and intensifying their exploration efforts. In addition, smaller companies have entered the market because of the opportunity to easily finance their exploration activities through the stock market. The most spectacular example of this is to be found in the Athabasca Basin, where only four companies were actively exploring in 2003, but there are now over 50. The situation is quite similar in other parts of the world but perhaps not as extreme.

This renaissance in uranium exploration is, of course, welcome and necessary. World production has been stable for many years. However, most current supply sources will come to an end between 2015 and 2030 or continue to decline. They need to be replaced, and new resources will have to enter into production to meet the forecasted demand increase for the nuclear energy sector.

After 20 years of recession and related low key exploration activities, the uranium industry finds itself with an ageing workforce and loss of expertise, staff and know-how. Therefore, recruitment, education, training, R&D and consolidation, as well as the availability of contractor companies providing drilling and geophysical services, are serious issues that must be addressed in the very near future if the challenges ahead are to be met. Meanwhile, competition for recruitment and retention of qualified personnel in the uranium industry is increasing because other sectors within the mining industry and energy industries (e.g. gold, base metals, oil and gas) are experiencing the same boom and related challenges.

The boom has also created more speculative activity. However, speculators are unlikely to survive for very long, because exploration is a long term effort requiring investment over many years.

The purpose of this paper is to highlight and discuss the main trends and challenges that have been observed in the uranium industry during the past few years. It will also focus on the challenges and opportunities that will arise in the coming years.

The paper will neither review nor analyse in detail data regarding uranium exploration and production. Such exercises have been carried out in other papers and will be published in the forthcoming 2005 Red Book.

2. EXPLORATION STATUS AND CHALLENGES

2.1. Claim staking activity and exploration spending

The renewed interest in uranium exploration first started in early 2004, with a sharp increase in claim staking activity. This was followed by a phase of fund raising on stock markets and more recently with some 'in the ground spending'. In most cases we are still experiencing claim staking and fund-raising. The 'in the ground spending' remains at an early stage and must be confirmed for companies new to this business. It is expected that many of these will remain speculators and not become true explorers. Many will disappear within a few years.

The trend in exploration activities and spending since 2004 is similar in countries with good uranium potentials. The situation will not be reviewed country by country because no official data are available at present. However, a few illustrative examples of what has been observed in different parts of the world will be given.

One of the most spectacular examples of the renewed interest in uranium exploration is found in the Athabasca Basin, Saskatchewan, Canada, which hosts the Cigar Lake and McArthur River deposits. During the late 1990s and early 2000, only four companies were actively exploring there for uranium. At present, more than 50 companies are involved in uranium exploration in the basin. Figures 1 and 2 illustrate this spectacular evolution. Similar trends are observed in other parts of Canada (i.e. the Northwest Territories, Nunavut, Quebec, Labrador and Ontario) where the resurgence in uranium exploration is gaining in strength, with more than 40 companies actively involved in uranium exploration.

Australia is another part of the world where there is intense renewed interest in uranium exploration. Activity is mainly located within South Australia and in the Northern Territory but other states may become involved in the near future. In 2003, nine companies held claims related to uranium exploration and four were actively conducting exploration. In 2004, 12 companies held land positions dedicated to uranium exploration and nine were actively conducting exploration. At present, 42 companies hold claims for uranium exploration and 21 are more or less actively conducting exploration. The level of expenditure has tripled during the same time period and is now around 15 million Australian dollars (Table 1 and Fig. 3).

Kazakhstan, Mongolia, Niger and the Russian Federation are other countries where an increasing interest in uranium exploration is observed.

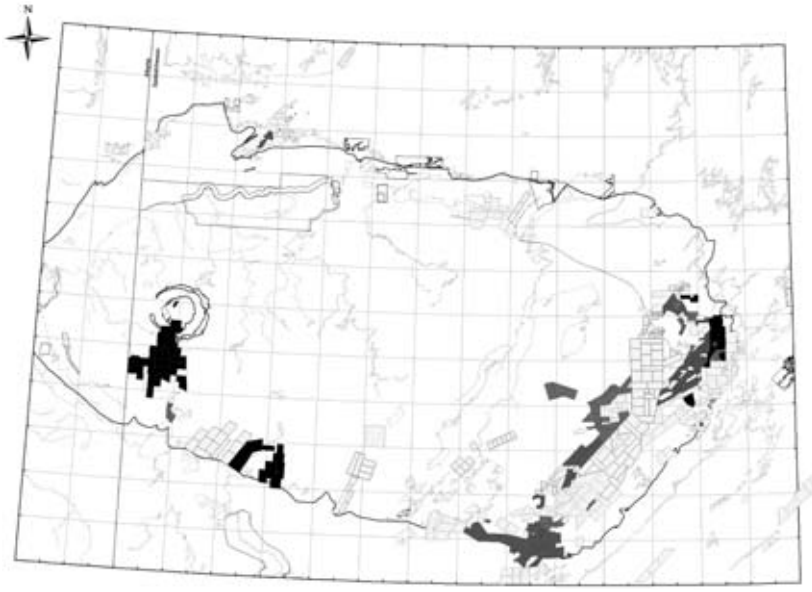

FIG. 1. Exploration permits in the Athabasca Basin in 2003.

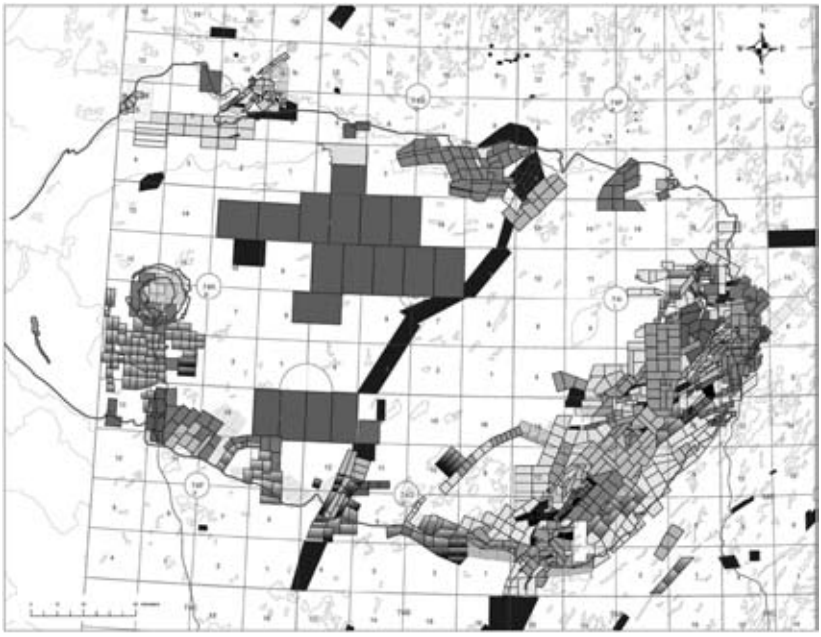

FIG. 2. Exploration permits in the Athabasca Basin in 2005.

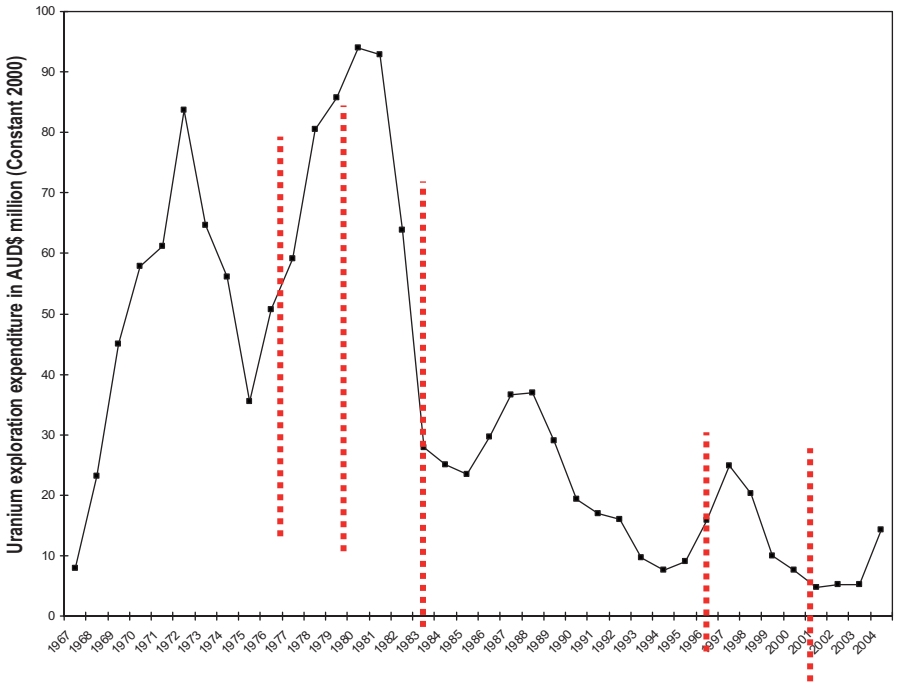

FIG. 3. Exploration expenditure in Australia.

TABLE 1. COMPANIES INVOLVED IN URANIUM EXPLORATION IN AUSTRALIA

Financial year	Australian public companies	Foreign public companies	Private companies	Number active
2002/2003	5	2	2	4
2003/2004	7	3	2	9
2004/2005	32	5	5	21

In the Athabasca Basin of Canada, the expenditures for uranium exploration (in millions of Canadian dollars) were 15.5 in 2002, 13 in 2003 and 26 in 2004, with 66 predicted for 2005. During the same period, exploration expenditures for all commodities in the basin (in millions of Canadian dollars) were 30.1 in 2002, 31.3 in 2003 and 53.0 in 2004, with 119.8 predicted for 2005, showing that it is not only uranium exploration that is experiencing a boom.

2.2. Challenges

The sharp increase in exploration expenditure is very encouraging. It has been eagerly awaited by explorers for many years. The arrival of new players is stimulating for the industry in general and has resulted in increased competition. Next, the short and long term challenges lying ahead need to be addressed. Some of the major challenges facing the uranium industry are now discussed.

2.2.1. Risk assessment

Risk assessments are an common issue for all serious exploration and mining companies who want to conduct their business in a sustainable world. Risks related to prospectivity (geological potential and economics), exploration (technology), country, environment, mining and reputation are parameters that are taken into consideration in the decision making process. At present, most of the major players have realized the necessity to integrate these factors into their everyday way of thinking and acting, and they are in the process of successfully managing these risks, although there is still scope for improvements.

2.2.2. Mining legislation

Progressive mining legislation, and licensing and permitting processes must be created by governments and regulatory agencies in order to guarantee long term and stable business environments for the exploration/mining and energy sectors. The uranium industry requires the following:

(a) Well balanced long term legislation and efficient application of the regulations, including international harmonization where and when possible;
(b) Serious but fair competition (speculators could slow down or even discourage serious competitors);
(c) Access to skilled technical and scientific personnel;
(d) Incentives for the mining industry.

Legislation, regulations and permitting processes are becoming increasingly complex and time consuming. They often delay exploration activities and the startup of mining activities. This is often due to authorities that do not have adequate human resources to process the increasing workload and complexity

generated by the resurgence of exploration activities. Clear laws and regulations are needed, and also an efficient and fair system.

International harmonization of legislation and regulations would benefit all players. However, different histories and cultures will have to be respected so that this process will require time and patience. Countries with well proven working legislation and regulations should assist the more actively emerging mining nations in setting up their own legislation and regulations.

2.2.3. Research and development

Supporting R&D efforts to obtain access to equipment and new technologies is an important issue facing the uranium industry. National institutions and the industry will have to significantly increase their efforts and investments.

2.2.4. Community consultation

The uranium mining industry will have to make further improvements in its relations and communications with the communities affected by the industry. At present, various types of impact benefit agreement are negotiated whereby the community members share the economic benefits of a project through the jobs and business development opportunities that are created. Responsible companies operating within a sustainable model are able to provide these economic benefits while respecting the health and safety of workers and the environment. Ethics, environment, safety, security and sustainable development are the main concerns and objectives of all serious parties.

The attitude of the new companies and speculators entering the business is largely unkown regarding some of these issues. Some will certainly have difficulties in coping with most of the forthcoming challenges. In some cases this is likely to be damaging to serious players, who have a long term commitment.

2.2.5. Market entry of speculators

As is well recognized, finance is a necessity for, but not a guarantee of, success. Access to short term finance is, as we have observed with the new entrants raising money on the stock markets, the easiest requirement to achieve. On the other hand, insufficient long term finance could tempt these minor players into taking short cuts that could prove catastrophic for the entire mining industry. Such short cuts have been experienced in the past with regards

to the environment, safety and shareholders, and have resulted in a loss of credibility for the entire industry.

2.3. Skill shortages

Access to skilled personnel and contractors is a major issue in the short term. The two main reasons for this are:

(1) The depressed uranium exploration and mining sector has for a very long time produced very little turnover or renewal of personnel, resulting in a loss of knowledge and know-how.
(2) There is competition for recruitment and retention of personnel from other sectors (i.e. gold, base metals, oil and gas) that are experiencing a similar and simultaneous boom.

The most important issue is the delay that the uranium industry will have to face due to the shortage of skilled and specialized technicians, scientists and engineers. In addition, there is the need for the uranium mining industry to make itself more attractive to young scientists and engineers.

To educate young people takes time, but the time it will take for them to acquire the skills and experience required by the industry is going to be much longer. To achieve a good university education, five to eight years are needed. To acquire a minimum of professional experience, three to five years are required, and to reach the level of an experienced generalist or specialist, eight to thirteeen years is what is normally necessary.

If this is true for the exploration/mining industry, it is more prevalent within the academic world which, due to a lack of demand and funding from the industry, has reoriented its academic programmes and research away from the uranium industry towards other areas.

Access to skilled contractors is another limitation. They are too few in number. Furthermore, the competition from other sectors experiencing the same boom as the uranium sector will certainly delay parts of the exploration activities.

The increased competition, and the overall skill and capacity shortage, is going to have an impact on the cost of exploration activities, as the demand for skilled labour is greater than the supply.

2.4. Time factor

During the renaissance of exploration activities, time will be the major challenge lying ahead. In addition to issues related to human resources, and to

geological, geographical and political contexts, an abundance of time and patience will be needed to find new uranium deposits. Time cannot very often be compressed or compensated simply by provision of more finance.

The typical timescale from exploration to mining is:

(a) Area selection — permanent activity for major players;
(b) Exploration — ten years;
(c) Feasibility — five years;
(d) Licensing and construction of mines — five to ten years;
(e) Total timescale — 20–25 years,

This timescale emphasizes the importance of the long term policies that the industries require to be able to motivate their shareholders during this period, as well as to be able to meet the community's expectations for the sustainable development of the mining and energy sectors.

3. MINING STATUS AND CHALLENGES

Uranium production will increase only slightly during the next ten years. Increasing further the capacity of existing production centres, where possible, will take time and a great deal of effort. Bringing new uranium deposits into production will take even longer, as the deposits will have to be discovered first.

The quickest increase in production could come from the in situ leach production centres (within five to ten years), and the largest could come from the operations at Olympic Dam (in 10 years time) and from Jabiluka if a change of policy occurs in Australia.

Most of the challenges for exploration activities mentioned earlier are valid for mining activities. Time and skill shortages are the most crucial.

4. OPPORTUNITIES

It is important for the uranium industry to realize that with every challenge come opportunities. The arrival of ambitious entrants to the industry will result in interesting projects, new approaches and creativity, leading to discoveries and new partnerships.

Emergent mining nations with excellent potential for uranium are opening up their borders and will become sources of future production centres.

The energy debate is back on the agenda in most nations and has a high priority internationally. It has lead to a more balanced view regarding the nuclear industry. All available energy sources will have to be utilized to ensure the welfare of future generations in a sustainable environment.

The opportunities emerging are overwhelming for the uranium industry, but, if the challenges can be met, the industry will undoubtedly play an important role in tomorrow's world.

URANIUM SUPPLY AND PRODUCTION SCENARIOS IN KAZAKHSTAN

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Abstract

Developments in the nuclear market point to a future shortage of uranium. Kazatomprom has the potential to contribute significantly to the supply of uranium for nuclear energy needs. Thousands of tons of uranium will be required in addition to what suppliers report now. Kazatomprom controls some of the worldwide largest uranium resources. The processing capacities of Kazatomprom provide for production of uranium concentrates to any specification, from ASTM international standards to nuclear purity concentrates suitable for direct fluoridation, to uranium dioxide powder for CANDU fuel pellets. Kazatomprom was established in 1997 as the national organization for export and import of uranium and uranium compounds. Kazatomprom today is a conglomerate including geological, mining, metallurgical and energy companies, and has its own research and development centre. Since 1998 Kazatomprom has gradually been increasing uranium production. Such stable development is provided by our own investment programme in support of production growth and replacement of older sites. The increase covers corporate contractual obligations, with an expected annual surplus of 100–300 tonnes of uranium for sale. Further increases in uranium production in Kazakhstan are dependent on investments in new mines. With the price stable at around US $21/lb U_3O_8 (46 US $/kg) and customers willing to buy, Kazatomprom could reach a production level of 8300 t U/a by 2010 and up to 15 000 t U/a by 2015. There are several possibilities for cooperation with Kazakhstan in preventing a future uranium deficit: (a) direct investments with repayment from sales of uranium production; (b) setting up of a joint venture for development of a specific deposit for the needs of a specific customer; (c) long term contracts with a floor to the price to ensure guaranteed sales.

1. INTRODUCTION

Kazakhstan is located in the centre of the Eurasian continent. Regarding the facts favourable for the development of the uranium industry, the following can be noted:

(a) Political stability and well developed legislation on subsoil use and ecology;
(b) Rather good environmental conditions;
(c) Low population density in combination with a highly qualified labour force;
(d) Accessible transport communications and closeness to some of the main world conversion and separation capacities in the Russian Federation and China.

In the territory of the Republic of Kazakhstan, mining is being carried out by the national atomic company Kazatomprom, its subsidiaries and affiliates [1]. In 2005, the uranium output will amount to 4200 tons, and at the same time Kazakhstan will remain the world's third largest producer.

2. PROVINCES WITH URANIUM ORE DEPOSITS

There are 129 uranium deposits and ore occurrences in six provinces in the territory of Kazakhstan. At these deposits, the estimated and governmentally approved reasonably assured resource (RAR) ($B + C_1$) and estimated additional resource (EAR-1) (C_2) category reserves are about 940 000 tons. These data are given in the 'Red Book' of 2003, a joint report of the OECD Nuclear Energy Agency and the IAEA [2].

There are more reserves that have actually been explored but not approved of formally by the Government. According to our estimates, they exceed 1 500 000 tons, 1 200 000 tons of which are suitable for the acid in situ leaching (ISL) method of production, which has the lowest cost and is environmentally sound.

At present, commercial production is being carried out at 11 deposits in the Chu-Saryssuiskaia, Syrdaryinskaia and North Kazakhstani provinces.

On the whole, all the required factors, both objective and subjective, for a significant increase of natural uranium production are present in the territory of Kazakhstan.

Currently, Kazatomprom has already formulated a concept for its further development. This concept is dependent on the following preconditions:

(a) Production by the ISL method is profitable at current prices, but the profitability of underground and open pit mining is doubtful. Therefore, the development plans are directed only towards the ISL method, with the exception of one operating mine each at the Vostochnoye and Zvezdnoye deposits.

(b) The available explored reserves for ISL are sufficient to make reasonable plans for production activity.
(c) Rates of increase in production are limited by organizational factors. For maximum simplicity of the organizational questions, a standard mine conception has been accepted.

A standardized approach to the deposit development suggests dividing a deposit into blocks with reserves of 30 000–40 000 tons, followed by construction of mines with a production capacity of 1000 tons each. Construction of the first such mine will be completed at Mynkuduk deposit in November 2005.

On the basis of this concept, Kazatomprom has approved a development programme up to 2010, pursuant to which production in the territory of Kazakhstan will be increased by up to 15 000 tons.

2.1. Groups of uranium deposits

To describe the programme for increase of uranium output by the year 2010 and outline the scenarios of further production development, Kazakhstani deposits may be divided into five groups:

(1) The first group includes those deposits where mining is taking place at present.
(2) The second group includes those deposits where commercial facilities are under construction.
(3) The third group includes those deposits for which the starting dates of development have been determined in the plans of Kazatomprom and its subsidiaries, the future mining operators of these deposits.
(4) The fourth group includes those deposits which are provisional, subject to confirmation of the available forecasts on price increases, but for which Kazatomprom has not yet identified a starting date for development.
(5) The fifth group includes the remaining deposits for which there are not enough grounds for development starting.

While dividing the deposits into groups, we will at the same time outline optimistic (upper) and basic (lower) scenarios for the development of the deposits considered. Here, the basic scenario will show the plans under existing economic conditions, and the optimistic scenario will reflect the possible expansion of mining while economic conditions are changing. For simplicity, it is assumed that the basic scenario is in agreement with a U_3O_4 price of US $30/lb (US $66/kg) and the optimistic one with US $40/lb (US $88/lb).

The Kanzhugan, South Moinkum, Uvanas, Vostochnyi Mynkuduk, Severnyi and Yuzhnyi Karamurun and Akdala deposits located in Chu-Sarysuiskaia and Syrdaryinskaia provinces belong to the first group of deposits. These deposits have been developed by ISL. The Vostochnoye deposit situated in North Kazakhstan, which has been developed by open cut mining, also belongs to this group.

2.2. Production

Mining at the first group of deposits will be reduced starting from 2010 under the basic scenario and from 2015 under the optimistic scenario because of depletion of reserves (Table 1). The optimistic scenario supposes an increase in the reserves at the Kanzhugan and Uvanas deposits found by means of supplementary exploration and involving operations that are not a convenient part of those at the Yuzhny Moinkum reserves.

The Zarechnoye, Inkai, Centralnyi Mynkuduk and Tortluduk deposits belong to the second group. Here, the joint ventures formed in partnership with well known companies such as TENEX (Russian Federation), Cameco (Canada) and AREVA (France) have already started the construction of commercial facilities. Even under the basic scenario, these deposits would make possible a doubling of production in the Republic of Kazakhstan by 2015 (Table 2). The optimistic scenario supposes a twofold increase in production at the Inkai and Moinkum deposits.

The Severnyi Kharasan, Irkol, Yuzhnyi Inkai, the central part of the Budenovskoye, and the Vostochnyi and Zapadnyi Mynkuduk deposits belong

TABLE 1. GROUP I DEPOSITS PRODUCTION FORECAST (TONS)

Scenario	2010	2015	2020	2025	2030	2035
Optimistic	4953	5870	5170	2270	460	166
Basic	4768	4592	3637	1630	400	166

TABLE 2. GROUP II DEPOSITS PRODUCTION FORECAST (TONS)

Scenario	2010	2015	2020	2025	2030	2035	2040	2045	2050
Optimistic	2555	5810	6852	6205	5560	4220	3700	1250	80
Basic	3158	4500	4392	3705	3260	2540	2500	1500	2300

to the third group. For development of these deposits, subsidiaries of Kazatomprom have already been established. According to the plan, by 2010, new mines for development of the third group of deposits will be built.

In the optimistic scenario, the capacities of the mines at Severnyi Kharasan, Yuzhnyi Inkai and Centralny Munkuduk will be doubled (Table 3).

The north and south parts of the Budenovskoye, Severnyi Kharasan, Semizbai, Zvezdnoye, Suluchinskoye, Sholak-Espe and Zhalpak deposits are considered to be within the fourth group. Since no production plans for this group of deposits have been approved, for the purposes of this paper it has been conditionally assumed that these deposits will be evenly involved in development after 2015. The basic scenario assumes development only of a part of the Budenovskoye deposit, explored in terms of the C_2 category (Table 4). The optimistic scenario assumes development of the entire Budenovskoye deposit and other deposits included in this group.

We did not prepare any forecast of production with respect to the remaining deposits attributed conditionally to the fifth group, because for the time being it is difficult to make any reasonable assumptions concerning the dates of their involvement in development. However, at the same time, it should be borne in mind that a number of deposits in the North Kazakhstan province are suitable for underground and open pit mining, and might be rather promising at U_3O_8 prices higher than US $40/lb (US $88/kg).

Note that the value of 9500 tons, in the 2010 column, is the average predicted production for the period 2005–2010.

In accordance with the basic scenario, Kazakhstan will reach a production peak of 15 000 tons in 2010, after which the production will remain at the same level for the next ten years (Table 5). Under the optimistic scenario, production will approach its maximum level of 22 000 tons in 2015 and remain at this level

TABLE 3. GROUP III DEPOSITS PRODUCTION FORECAST (TONS)

Scenario	2010	2015	2020	2025	2030	2035	2040	2045	2050
Optimistic	2042	9270	10 000	9410	5091	992	0	0	0
Basic	1950	6460	6 500	6460	4810	3620	2800	2000	1440

TABLE 4. GROUP IV DEPOSITS PRODUCTION FORECAST (TONS)

Scenario	2020	2025	2030	2035	2040	2045	2050
Optimistic	562	4332	8682	8657	6096	4298	3744
Basic	440	1880	2000	1960	617	0	0

TABLE 5. ALL URANIUM DEPOSITS PRODUCTION FORECAST (000 TONS)

Scenario	2010	2015	2020	2025	2030	2035	2040	2045	2050
Optimistic	9.5	21.0	22.6	22.2	19.8	14.0	9.8	5.5	3.8
Basic	9.2	15.6	15.0	13.7	10.5	8.3	5.9	4.5	3.7

for almost 15 years. During the period under review, 440 000 tons would be produced under the basic scenario and almost half as much again under the optimistic scenario.

3. CONCLUSION

The rates of increase in production in the territory of Kazakhstan could be higher if the world community were to attend to the main problem of the uranium market — instability of the spot price. Without going into the particularities of the economic mechanism, the creation of a stabilizing fund to buy uranium when the price is low and sell it when the price is high might be beneficial. The activity of such a stabilizing fund would reduce the uncertainty that sometimes prevents development of production capabilities.

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THORIUM AND UNCONVENTIONAL URANIUM RESOURCES

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Abstract

Three decades ago, thorium was considered as an alternative to uranium. Resource estimates for thorium were reported by the OECD Nuclear Energy Agency and the IAEA from 1965 to 1986. When developments of nuclear power slowed down, interest in thorium fell. Total world thorium resources are estimated at about six million tonnes, of which about two million tonnes are regarded as known (=identified) resources. Economically interesting are resources in coastal sands (placers), mined for the rare earth mineral monazite. Thorium can be extracted as a by-product of Ce, La and Y. Other resources, for example, those in vein deposits, may become interesting once requirements arise. Unconventional uranium resources are found in low grade deposits, or are recoverable as a by-product. By-product resources are of interest in the case that conventional resources are insufficient. In by-product recovery, the greatest portion of costs are borne by the main product(s). Examples are marine phosphate deposits containing more than 100 g/t U. Uranium was extracted in phosphoric acid plants in the United States of America and Belgium until the end of the 1990s. World resources of uranium as a by-product of the use of such phosphate are being studied; however, the production of uranium is limited to phosphoric acid facilities. Low grade uranium deposits in black shales, lignites, carbonatites and granites were expected to be potential sources in the past. With very few exceptions none is at present economic. The enormous land requirements for mining operations pose ecological barriers. Sea water was also considered as a potential source. Extraction has been proved at laboratory scale but at costs well above current market prices.

1. INTRODUCTION

Application of nuclear energy for the production of electricity has a short history compared with that of fossil energy. It took about ten years from the end of World War II for the first commercial nuclear power stations to produce electricity. More nuclear power stations were installed in the following years; however, requirements for fuel remained at levels of a few thousand tonnes of uranium annually during the initial phase of civil nuclear power. Procurement

of nuclear fuel was not a problem in the initial years due to the existing production facilities built for military requirements.

It took about another ten years before, in 1965, a group of independent national experts was called by the Organisation for Economic Co-operation and Development (OECD) to compile data on world resources of uranium and thorium. This first worldwide attempt of experts from government organizations to provide an independent and unbiased compilation of resource data was published in 1965 by the OECD Nuclear Energy Agency (OECD/NEA) as 'World Uranium and Thorium Resources'. This was a thin booklet with information from 16 countries, restricted to the 'Western World'. This exercise was repeated every second year, and until 2003 twenty editions of what is known as the Red Book have been published jointly with the IAEA, some of them containing information about almost 50 countries and with up to 400 pages [1].

In the 1970s and 1980s, growth of nuclear power generation was expected in view of possible shortages of fossil fuel, namely oil. Predictions by experts of a rapid increase in the number of nuclear power stations raised concerns about the availability of uranium resources worldwide.

The anticipated growth of nuclear capacities called for future increases in uranium requirements. During this period, information on resources outside WOCA (the world outside centrally planned economies area (the former Eastern bloc)) was not available. Resource estimations and demand/requirement forecasts for nuclear fuel were thus limited.

The situation at that time called not only for complete analysis of resource availability for the conventional[1] type of uranium resources but also for uranium which could possibly be made available from unconventional types of deposits (e.g. from phosphates, black shales, granites and sea water), as well as for thorium resources as an obvious fuel option for uranium. It has been postulated that these resources might be required in the future.

Resource figures for thorium have been collected for the Red Books of 1965–1986 for WOCA on a country by country basis. Detailed articles on thorium deposits were published in 1991 and 1992 [2, 3]. Accumulated figures for resources of thorium in the Commonwealth of Independent States (CIS) were published in 1997 [4]. The United States Geological Survey continuously reports thorium resources in its annual Mineral Commodity Summaries; however, resource data have not changed significantly during the period covered.

[1] Conventional: uranium resources in producing deposits or of well established characteristics.

Unconventional uranium resources were reported by countries in the Red Books between 1977 and 1993, and as summaries in the years following. A study on uranium as a by-product of phosphate, published in 1979 [5], contains detailed information for the Western World.

2. THORIUM

2.1. Thorium: A nuclear fuel

The average content of thorium in the upper crust of the earth reaches 6–10 ppm, about three times the content of uranium. Thorium is widely distributed in rocks and minerals, usually associated with uranium, elements of the rare earth group and niobium and tantalum in oxides, silicates and phosphates. In vein deposits, thorium may be present as thorium silicate (thorite) or thorium oxide (thorianite). The content of 8–10% Th in the Ce–La–Y–phosphate mineral monazite is commercially interesting.

The isotope ^{232}Th itself is not fissile; however, interacting with neutrons the fissile isotope ^{233}U is formed. Because no ^{239}Pu is generated, thorium may be preferable to uranium under certain conditions.

Thorium based fuel cycles have been investigated in Germany, India, Japan, the Russian Federation, the United Kingdom (UK) and the United States of America (USA) [6, 7]. Reactors with thorium are mainly high temperature, gas cooled reactors (HTGRs) and pebble bed reactors (THTRs (thorium high temperature reactors)), developed and built in the 1960s and 1970s in Germany (Jülich and Schmehausen) and in the USA (Peach Bottom and Fort St. Vrain). Experimental reactors were operated in the UK (Dragon at Winfrith) and in India. In high temperature reactors, helium is used as cooling gas. Temperatures as high as 800–1000°C are used for generation of electricity and for chemical processes (e.g. gasification of coal). However, the high costs of fuel fabrication and unsolved technical problems have slowed down further developments.

Publications in the 1970s indicated that a 1000 MW reactor needs initial loadings of ≈40 t thorium and ≈10 t of highly enriched uranium (≈90% ^{235}U) (HEU) and annual reloadings of ≈10 t Th.

In the late 1980s both Germany and the USA decided to shut down their thorium reactors.

For countries having insufficient access to uranium resources, thorium fuelled reactors are still an option. Research for advanced types continues in some countries.

Recently, thorium has been tested in the thorium–molten salt concept, for example in Japan. In India, owing to large resources in coastal placer deposits, thorium is used in the Kakrapar reactors. If research on high temperature reactors (HTRs) proves successful, thorium may be used with HEU and plutonium [8].

2.2. Major types of thorium deposit

The classification of major types of thorium deposit is based on their genesis and their descriptive characteristics. In general, thorium deposits may be divided into those that are associated with the magmatic cycle (endogenous deposits) and those of the sedimentary cycle (exogenous deposits) [2, 3].

The endogenous deposits include (examples are given in parentheses):

(a) Granites and pegmatites (Jos, Nigeria; Bancroft, Canada);
(b) Alkaline rocks, for example, nepheline syenite (Ilimaussaq, Greenland; Langesundfjord, Norway; Ulug Tansek, Russian Federation);
(c) Hydrothermal veins (Wet Mountains, Powderhorn, USA; Steenkampskaal, South Africa);
(d) Carbonatites (Araxa, Brazil; Mountain Pass, USA; Phalaborwa, South Africa).

The exogenous deposits include:

(1) Alluvial placers in river valleys (North and South Carolina, USA);
(2) Coastal placers (Kerala, India; Bahia, Brazil);
(3) Ancient metamorphosed placers (Blind River, Canada).

The economic importance of deposit types may be demonstrated by the shares of each type.

Total world thorium resources, irrespective of economic availability, are at present estimated at about 6 078 000 t Th.

In a simplified subdivision according to major deposit characteristics, a ranking may be given (Table 1).

Placer deposits may have economic relevance (see below).

2.3. Countries with major thorium resources

The following ranking of countries may be subjective due to limited knowledge. Some data are more than 20 years old. In the past, when thorium was regarded as an alternative or addition to uranium, official resource

TABLE 1. MAJOR THORIUM DEPOSIT TYPES AND THEIR RESOURCES

Major deposit type	Resources (1000 t Th)	Percentage
Carbonatite	1900	31.3
Placer	1500	24.7
Vein type	1300	21.4
Alkaline rocks	1120	18.4
Others	258	4.2
Total	6078	100.0

assessments of thorium were made by independent experts [1]. After 1986 statistics on thorium resources have no longer been reported. An overview of thorium deposits and resources was published in 1991 [2]. The World Nuclear Association [6] refers to the reserves given by the US Geological Survey in its Mineral Commodity Summaries as a world total of 1 200 000 t Th. The same data are reported in Refs [9, 10], with additional data, however, for the reserve base of 1 400 000 t Th. The thorium resources in the CIS reported in 1997 in a publication of the IAEA [4] totalled 1 700 000 t Th.

An overview of total world resources (the confidence of estimates and economic attractiveness are not considered) is given in Table 2.

About 53 000 t Th in the CIS are 'off-grade' [4]. Deposits of grades between 0.1 and 2% Th, reported as 75 000 t Th, are economically interesting and estimated by the author as reasonably assured resources (RARs) (≈reserves).

Estimates on the economic significance of thorium deposits have been made in the Red Books [1]. Similarly to uranium, thorium resources are classified into RARs, estimated additional resources, category I (EAR-I), and estimated additional resources, category II (EAR-II). In 1983, EARs were separated into EAR-I and EAR-II. Reasonably assured resources and EAR-I are called 'known resources' or 'identified resources' due to the degree of confidence in the estimates. Thorium resources were classified according to costs of recovery (in 1986 recoverable at costs of up to US $80/kg Th).

Most thorium resources have been discovered and evaluated during exploration for deposits of uranium, rare earth elements (e.g. monazite in coastal placer deposits) or elements such as tantalum and niobium.

With the evolution of knowledge, resource data changed on both the national and world scales between 1965 and 1986.

In Table 3 resource figures are presented as reported in different sources.

TABLE 2. WORLD TOTAL OF UNSPECIFIED THORIUM RESOURCES

Country	Total thorium resources (1000 t Th)
Brazil	1306
Turkey	880
USA	432
Australia	340
India	319
Egypt	295
Norway	180
Canada	173
South Africa	115
CIS[a]	1650[b]
Others	388
Total	6078

[a] Not separated into data for individual States.
[b] Adjusted for 'off-grade' resources.

World total thorium resources estimated in the categories RAR, EAR-I (identified resources) and EAR-II (prognosticated resources) amount to 4.1–4.6 million t Th, about 67% of the world's unspecified thorium resources (Table 2). Differences in both estimates are the result of the approaches used. Total unspecified resources are higher because resources beyond those classified for the Red Books are included, such as resources recoverable at costs higher than US $80/kg Th and resources in categories of lesser degree of assurance than EAR-II. Therefore, resources in Australia, Brazil, South Africa and the CIS are much higher, mainly due to resources in placer deposits and other deposits not considered in the Red Book estimates.

Known (identified) resources amount to about two million t Th.

2.4. Considerations for the recovery of thorium

Thorium often occurs in minerals that are mined for another commodity, thorium being a co-product or by-product. The costs of mining and milling would then generally be accounted for in the costs of the principal commodity/ies. Recovery from ores where thorium is the principal or single beneficial

TABLE 3. WORLD THORIUM RESOURCES, CLASSIFIED INTO CATEGORIES (in 1000 t Th)

Country	RARs recoverable < US $80/kg Th	EAR-I (inferred) recoverable < US $80/kg Th	Known resources (identified)	EAR-II[a] (prognosticated)
Turkey	344	n.a.[b]	344	400–500
India	319	n.a.	319	n.a.
Brazil	171	50	221	329–700
USA	122	278	400	274[c]
Russian Federation[d]	75	n.a.	75	n.a.
Greenland	54	n.a.	54	32[c]
South Africa	18	n.a.	18	130[e]
Australia	13	<1	13	300[c]
Venezuela	n.a.	300	300	n.a.
Norway	n.a.	132	132	132
Egypt	n.a.	100	100	280[c]
Canada	n.a.	44	44	128[c]
Others	23	10	33	81
Total	1139	914	2053	2086–2557

[a] Costs of recovery are not available.
[b] n.a.: not applicable.
[c] Earlier estimate.
[d] Estimated by author after Ref. [4].
[e] Preliminary estimate in 1983.

element would generally be restricted to specific circumstances, for example, high demand or very rich/high grade deposits.

For rising thorium demand, economic interest would primarily be concentrated on those deposits from which thorium can be easily extracted as a co-product or by-product in the quantities desired.

The principal sources of thorium are deposits of the placer type, i.e. concentrations of heavy (rare earth) minerals in coastal sands, from which monazite and other thorium bearing minerals are recovered. Rare earth ores are recovered for their content of Ce, La, Y and other elements used in, for example, catalysts, ceramics, television sets and the computer industry. Placer

deposits have varying grades of valuable minerals, and, in general, those having concentrations of several per cent are of economic interest.

Monazite production can be used as a theoretical measure for thorium availability, due to contents of about 8–10% Th. World monazite production over the last five years has fluctuated between 5500 and 6500 t annually [11, 12], from which theoretically between 300 and 600 t Th could be recovered annually.

At present India has the highest monazite output in the world, accounting for about 90% of the total annually. India is investing in the thorium fuel cycle as part of its major interests in its domestic resources [6].

Brazil and Malaysia are other monazite producers, while Sri Lanka was a producer in the past. Production has also been reported from China, Indonesia, Nigeria and the CIS.

Deposits in carbonatites and alkaline syenites can be mined for niobium and tantalum. They often contain thorium concentrations of 0.*x*–*x*%. A typical example for carbonatite is Araxa in Brazil (producing) and for alkaline syenite Ulug Tansek in the Russian Federation [4]. Worldwide annually a few tens to a few hundred tonnes of thorium could theoretically be recovered.

Vein deposits of thorium may be profitable if other commodities can be recovered.

In the 1970s, vein deposits in the USA were investigated for their economic significance. The largest low cost reserves of about 100 000 t Th are located in the Lemhi Pass deposit (Idaho and Montana) and in the Wet Mountains (Colorado). In 1979 the costs of their recovery were estimated to be less than US $40/kg Th [2, 3].

At the current low demand for thorium, annually a few hundred tonnes, supply shortages would not arise. Industry covers its demands either from stockpiles or material obtainable as a by-product. Details are not disclosed. However, if the nuclear industry were to require substantial quantities, they could be obtained as a by-product in existing or newly installed facilities.

3. UNCONVENTIONAL URANIUM RESOURCES

In the period of growing nuclear power generation, estimates of the future availability of uranium were made by a number of governmental organizations, utilities and commercial companies. Shareholders and stakeholders were also asking whether nuclear power generation could satisfy the need for sustainability, namely sufficient uranium supply for the expected future demand. Estimates of cumulative requirements in the 1970s and 1980s raised doubts about the supply by conventional resources being steady. At that time,

little was known about uranium resources and demand in the former Eastern bloc, and future nuclear planning had to rely mainly on the resources of the 'free market'. Inquiries were made about uranium resources beyond conventional resources and estimates were made, both by experts for the Red Book as well as by others, and published in various articles [13–15]. Some publications were of a speculative nature. The author prefers to refer to estimates based on official unbiased sources.

The 2003 Red Book summarized only a little information — about uranium from phosphate deposits (≈22 million t U) and from sea water (~4000 million t U) [16]. Uranium was recovered from phosphate through production of phosphoric acid in the USA and Belgium for a number of years, but these operations terminated in the early years of the current decade because of high extraction costs. Recovery of uranium from sea water has been investigated on a laboratory scale in a number of countries in the past, but all have now stopped with the exception of Japan. Very low amounts of uranium per unit and very high cost (of the order of US $300/kg U) were the limitations of this process [16].

Reports on unconventional uranium resources for the Red Book are used to compile what is known on unconventional uranium resources. In many cases, information is based on extensions of knowledge obtained for conventional resources, for example for low grade deposits or as a by-product. In addition, other publications are considered.

3.1. Igneous rocks

Acidic (granites) and alkaline magmatic rocks (nepheline syenites) may have higher uranium contents, sometimes up to 100 ppm and more. An exception is the alaskitic granite at Rössing in Namibia with 300–400 ppm U and resources of more than 100 000 t U. This deposit has been mined since 1976 and was included under conventional resources. Research in the USA (e.g. Conway granite) and in other countries on the potential of granites has shown that none of the Rössing type were found. Even although total resources might be of the order of several tens of thousands of tonnes or a few hundred thousand tonnes, grades are too low to be considered economic. The immense size of operations for low grade deposits would have, apart from the economic considerations, environmental implications, and mining would become impossible.

Carbonatites are examples where uranium might be extracted as a by-product of Nb–Ta ores and sometimes of thorium. The Phalaborwa carbonatite deposit in South Africa is mined for copper; uranium is recovered as a by-product. The deposit is included under conventional deposits.

Brazil and Finland report 13 000 and 2500 t U, respectively, in carbonatite. Resources in other countries, for example, eastern Africa and the Russian Federation (Kiev carbonatite), are not reported.

Total world resources of uranium in carbonatites may reach several tens of thousands of tonnes.

3.2. Non-ferrous ores

Uranium may occur in deposits of base metals and other ores, for example, copper deposits in Chile, India, Mexico, Peru and the USA. By-product recovery of uranium from a copper deposit in Utah took place for several years. By-product uranium resources from the quartz pebble gold deposits in South Africa are reported as conventional resources, as well as the uranium recovered as a by-product at the Au–Cu deposit at Olympic Dam in South Australia.

Total resources, with the exception of the quartz pebble deposits in South Africa and Olympic Dam, are estimated at several ten thousand tonnes.

3.3. Black shale and lignites

Many marine black shales, rich in inorganic matter, contain uranium in considerable quantities (about 10–80 ppm). The alum shale of Ranstad in southern Sweden contains around 300 ppm and has been mined on a limited scale. Its total resources of about 300 000 t U are classified as high cost conventional resources. The deposit is not available for mining for environmental reasons.

The black shale resources of 10 000–14 000 t U in the Republic of Korea are listed as high cost conventional resources. In Finland about 3000–9000 t U are estimated as unconventional resources. The Chattanooga Shale in the USA has been regarded as a potential unconventional source; however, at present ecological reasons do not allow any mining.

Coals and lignites generally contain a few ppm of uranium; however, some coal deposits can have a high enough content to make extraction of uranium feasible. Lignites in Dakota (0.*x*% U) were exploited for a short period, leaving a remaining reserve of 2000 t U. In eastern Germany, the Permian coal of Freital near Dresden (0.086% U) yielded 3700 t U, extracted during 1947–1955 and 1986–1989. Lignites in Spain have been reported as high cost conventional resources (40 000 t U RARs and 62 000 t U EARs). Uraniferous coal has been found in South Africa, in the northern Transvaal.

3.4. Phosphates

Marine phosphate deposits are known to contain uranium due to the ability of the calcium phosphate mineral apatite to capture uranium in its lattice. The uranium content in phosphates varies from a few tens of ppm to more than 100 ppm.

Interestingly, magmatic apatite is generally of lower uranium content than marine apatite. Consequently, marine phosphates have the potential for economic recovery of uranium as a by-product in the wet phosphoric acid process.

Industrial extraction of uranium from phosphates took place in the USA (>10 000 t U, Florida phosphate), Belgium (about 700 t U (phosphate imported from Morocco)) and in the former Union of Soviet Socialist Republics (about 6000 t U, fish-bone deposits near the Caspian Sea); however, all were terminated because of high extraction costs.

In the past a number of countries planned uranium extraction in their fertilizer industries, but nothing was accomplished [15].

Estimates of world resources of uranium recoverable from phosphate deposits were provided by official organizations for the Red Books from 1965 to 1993; the last totalled about 7 million t U (in situ).

In the late 1970s, the Uranium Institute in London reported about 15 million t U in phosphates [14]. At that time, all existing phosphoric acid facilities in the 'free world' were estimated to have a theoretical annual capacity of 5000–10 000 t U.

The uranium potential of world phosphate resources is of a speculative nature because only those deposits mined at present can theoretically be regarded as sources for uranium.

The US Geological Survey Mineral Commodity Summaries of Jan. 2005 estimates world reserves of about 18 billion t of phosphate rock [17]. This figure includes low grade non-marine phosphates. Uranium contents are not reported here. Thus the potential for uranium cannot be estimated from these data.

To illustrate the theoretical potential, the following calculation may be interesting. The annual world production in 2004 was 138 million t phosphate. Assuming an average grade of 60–80 ppm U contained in the phosphates mined, between 8000 and 11 000 t U is available.

In 1979 De Voto and Stevens [5] investigated the uranium content and the technology of extraction from the phosphate resources of the 'free world' and found about 22.7 million t U to be extractable. However, the amount of recoverable uranium, in particular in the USA, appears to be much higher than

the estimated total of about 7 million t U, published in the 1993 Red Book, where the latest information by country is available.

In 1978 the world production of in excess of 24 million t P_2O_5 had a content of about 7000 t U.

At present no uranium production from phosphoric acid plants is reported; however, with the rise in prices (mid-2005: US $30/lb U_3O_8) it may become viable again to reactivate mothballed facilities.

An increase in the market price of 15–20% above the current long term price (>US $70–80/kg U) could initiate renewed recovery of uranium as a by-product of phosphoric acid production [18].

From the environmental standpoint, it might be desirable to remove all uranium from products derived from uraniferous phosphate, creating additional supply sources.

3.5. Sea water

Sea water has been regarded as a possible supply source due to its almost inexhaustible quantity of about 4 billion t U. However, the concentration of only 3 ppb is very low. It would be necessary to treat huge volumes of sea water (about 350 000 t water for 1 kg U) and use large amounts of absorber.

Research has been carried out in Germany, Italy, Japan, the UK and the USA but this was given up in the 1970s or 1980s with the exception of Japan. The major reasons for terminating recovery from sea water were the high extraction costs and low quantities of uranium obtained per unit.

In Japan laboratory work has indicated extraction costs of US $300/ kg U, which is three to four times higher than the present spot market price [16].

Of academic interest perhaps is the idea to couple sea water desalination with the extraction of metals, including uranium.

4. ESTIMATE FOR URANIUM REQUIREMENT AND ITS AVAILABILITY

4.1. Estimates of requirements

In the 1970s and 1980s, estimates of future nuclear power developments and related uranium requirements raised the issue of uranium availability.

The 1982 Red Book assumed an annual requirement for WOCA of 60 000–160 000 t U for 2005, which compares with the actual requirements for that area of about 59 000 t U and the total world requirements for 70 000 t U in 2005. Similarly, a study by the OECD/NEA and the IAEA in 1987 [19]

estimated annual requirements for 2005 of 65 000–161 000 t U for the world total. Cumulative requirements for WOCA were estimated in 1982 by the Red Book to reach between 1.5 and 2.1 million t U until 2005 and up to between 2.5 and 7.6 million t U until 2025.

A study by the OECD/NEA and the IAEA in 1987 estimated cumulative requirements to 2005 of between 1.1 and 2 million t U and to 2025 of between 2.5 and 7.3 million t U. More recently, a study of the IAEA on the cumulative uranium requirements to 2050 resulted in an estimate of 5.4 million t U between 2000 and 2050 for a 'middle case' [20]. From estimates given in the 2003 Red Book, cumulative uranium requirements between 2005 and 2020 are calculated as approximately 1.2 million t U [16].

Industry estimates of current consumption are about 69 000 t U [21]. If growth continues as assumed at present, consumption could increase to 79 000–86 500 t U by 2015 and to 82 700 or even 103 800 t U by 2024. In this case, cumulative consumption could increase by 2024 to as much as 1 538 000 or even 1 769 000 t U [21].

4.2. Resource estimates of known resources

A comparison of resource estimates and production over a period of 20 years is shown in Table 4.

For the 2003 Red Book, countries producing uranium or planning to produce in the near future reported that nearly 2.2 million t U of known resources, recoverable at less than US $80/kg U, might become available.

If the cumulative requirements estimated in 1982 for 2005 had materialized and no resources had been added, all RARs < US $80/kg U, of about 1.7 million t U, would have been used up.

Three major facts have changed the basis for past assumptions:

(1) Growth of nuclear power is less than expected.
(2) Successful exploration has added new resources.
(3) Resources data for the CIS, Eastern Europe and China are now available.

In 2003, RARs recoverable at less than US $80/kg U amounted to 2.5 million t U, sufficient to cover the at present estimated requirements beyond 2020. The total known resources of 4.6 million t U would be short by about 0.8 million t U according to the estimates of the IAEA for 2050 [20]. The estimates are based on currently known resources, not considering successful exploration. If uranium prices are at an encouraging level, exploration expenditures will increase [22], consequently resulting in additional resources.

TABLE 4. URANIUM RESOURCE ESTIMATES FOR DIFFERENT PERIODS (in 1000 t U)

Year of estimate	RARs recoverable <US $80/kg U	RARs recoverable <US 130/kg U	Known resources* <US $80/kg U	Known resources* recoverable <US $130/kg U
1982 (WOCA)	1700	2300	2600	3500
1987 (WOCA)	1600	2200	2400	3500
2003 (World)	2500	3200	3500	4600
Resource growth 1982–2003	+800	+900	+900	+1100
World production 1982–2003 ≈1000				
Total growth, including production 1982–2003	≈1800	≈1900	≈1900	≈2100

* Known resources: RARs + EAR-Is.

4.3. Undiscovered resources

Beyond known resources (the sum of RARs and EAR-Is (=inferred resources)), an additional supply potential can be seen in the so-called undiscovered resources, i.e. EAR-IIs (prognosticated) and speculative resources (SRs). These resources are not well known but they are expected to occur. Industry in general is not interested in undiscovered resources; however, they are considered in government and other long term planning. Therefore, the Red Books have collected information on undiscovered resources. Not all countries have provided data. The data available for undiscovered uranium resources in the 2003 Red Book are shown in Table 5.

Conservatively estimated, more than 20 million t U are assumed to exist in undiscovered resources, a considerable potential in addition to known resources of about 4.6 million t U.

Assuming future exploration successes similar to those in the past, increasing future demand can be covered by transferring undiscovered

TABLE 5. UNDISCOVERED URANIUM RESOURCES AS OF 2003 (in 1000 t U)

EAR-IIs recoverable <US $80/kg U (prognosticated)	EAR-IIs recoverable <US $130/kg U (prognosticated)	SRs recoverable <US $130/kg U	SR total
1500	2300	4400	7500
			up to 25 000, estimated 1983 for IUREP

resources into known resources once the secondary supplies from stocks and military material have been used up [22].

5. CONCLUSIONS

Past predictions about the evolution of annual and cumulative requirements have been higher than what was subsequently observed. Some interpretations about the availability of uranium have been based on low cost uranium resources only. However, recovery costs should not be a limitation. Known resources as of 2003, including material recoverable at less than US $130/kg U and EAR-Is, are around 4.6 million t U.

If the prices of currently about US $80/kg U continue to grow, resources recoverable above US $80/kg U will become of economic interest. It has to be kept in mind that the cost of natural uranium is a small fraction of the overall cost of nuclear electricity.

If requirements were to grow beyond the known conventional resources (about 4.6 million t U as identified resources and about an additional 10 million t U as undiscovered resources), an additional potential would exist for unconventional resources, namely as a by-product from phosphates. The annual amounts of uranium from phosphates may be limited now to the capacity of existing facilities (of the order of 10 000 t U per annum), but if demand were to rise, additional facilities might be installed.

In the case that nuclear power stations using thorium are an option, more than 1 million t Th is estimated as RARs. Most probably by-product extraction of thorium from monazite could serve as a source. Around 1.5 million t Th is estimated worldwide in monazite placer deposits. From current monazite production, between 300 and 600 t Th would theoretically be extractable annually.

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IMPACT OF EMERGING ENVIRONMENTAL CONSTRAINTS ON URANIUM SUPPLY

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Abstract

Supply in the long term of uranium fuel is critical to the nuclear power industry. The general expectation at present is that future supply will be generated in a manner that meets current rigorous environmental standards. These standards, as defined by, for example, regulators, industry, international lending organizations and the IAEA, require minimal impact to the environment using a sustainable development approach that responsibly balances environmental, social and economic factors. The environmental factors that are associated with modern developments fall into two groupings: (1) conventional environmental issues, such as land use, effluent discharges and impact on biota; (2) emerging regulatory issues based upon evolving risk assessment approaches and determinations. It is the latter, in our experience, that poses the greatest risk to the long term uranium supply. Modern experience in Canada indicates that mines can be developed in a manner that satisfies conventional environmental assessment criteria. As such, there should be minimal environmental constraints to developing new supplies when they are developed using modern assessment and regulatory standards. However, when a strong precautionary, environmentally protectionist, view of development is utilized, to the exclusion of social and economic considerations, the perception of risk can become tied too closely to possible effects rather than actual impacts. This protectionist perception of risk to the environment has the potential to threaten the development of new projects through the imposition of prohibitively expensive requirements or restrictions with questionable environmental benefits, which in turn threatens the reliability of the long term global uranium supply. It is only through a balanced and systematic approach to sustainable development that both uranium supply and environmental protection will be assured. The paper briefly discusses the more conventional environmental and socioeconomic constraints to supply, the emerging environmental threats to supply and the possible approaches to overcoming these potential barriers to production.

1. INTRODUCTION

Cameco Corporation's (Cameco) vision is to be a leading nuclear energy company producing uranium fuel and generating clean electricity. In many

respects, this vision can be applied to the whole industry as we embark on a nuclear renaissance. To achieve this vision, Cameco, and the rest of the industry, require access to a steady and reliable supply of uranium from a variety of global sources. Cameco recognizes that in order for uranium fuel supplies to be a clean and sustainable counterpart to the electrical generation end of the fuel cycle, reasonable constraints are necessary to ensure environmental protection and prevent unreasonable harm as a result of our mining, milling, conversion and refining operations. Such a supply can only be accessed if it is unencumbered by overly restrictive environmental constraints.

A number of authors have discussed the potential future limitations on available supply and the potential economic barriers to those supplies. For instance, Beattie [1] notes that many lower cost uranium production centres will exhaust their known reserves in the next 3–20 years. Concentrated exploration on both greenfield and brownfield sites will be required to replace these reserves. If exploration is not successful in identifying new reserves, or too many constraints are placed upon the development of lower cost deposits, the cost of future supplies will increase and their availability will be less assured.

The slow pace of uranium exploration over the last twenty years has compounded this problem by limiting the number of deposits that have been properly and fully evaluated, and which can be considered viable reserves. Even with the short lived price rise in the mid-1990s, which stimulated short term interest in uranium exploration, no significant new deposits have been discovered for some time. In response to the relatively recent increases in the price of uranium, there has been a significant upsurge in exploration activity. Whether this activity can identify the replacement reserves required to supply future demand remains to be seen. For instance, the current exploration boom in Saskatchewan needs to replace thousands of tons of production prior to 2019 when currently easily accessible high grade reserves will start to diminish [1]. What is clear is that there are a number of constraints that could have a potential impact on the timely development of those replacement reserves, should they be found.

Given the lead time required for a newly identified uranium deposit to work its way through the chain of discovery, delineation, regulatory approval, licensing, development and start of production, about 8–10 years, or more, is required to establish a new uranium mine and bring it to full production in Canada. New deposits, such as the recently discovered Millennium deposit in northern Saskatchewan [1], may require a full decade to reach production and will only be able to supplement increasing future fuel requirements.

Jarrell and White [2], in their paper on uranium licensing and permitting, note that, as an environmental tool, a strong environmental assessment (EA)

and licensing process is conducive to sustainable development and demonstrate the overall positive environmental impact associated with nuclear power. This process in turn helps to remove constraints by illustrating how primary uranium supply can be acquired in a demonstrably responsible manner, and thereby remove unreasonable and unnecessary development constraints.

This paper will discuss both the conventional environmental constraints currently being addressed, some of which are beneficial, and some of the emerging constraints that have the potential:

(a) To have an impact on access to exploration and development;
(b) To impose onerous conditions on development and decommissioning;
(c) To restrict fuel transportation.

Most of the discussion here will be focused on constraints to primary uranium production, which include exploration for primary uranium deposits, development of those deposits and production of uranium oxide. It is in this area that the main environmental constraints appear to be arising. However, some discussion is required about the environmental constraints related to the transportation side of the fuel production cycle, as this is another emerging area of concern.

It is presumed as the basis for this paper that most uranium companies are currently on a path of pursuing exploration, development and operation in a responsible manner that is based on sustainable practices and protection of the environment. If nuclear energy is to be solidly established and publicly accepted as a safe and clean form of baseload energy, then all stages of the fuel supply chain must be developed in a sustainable manner. Some non-governmental organizations (NGOs) have, reacting largely to legacy issues and the results of some unscrupulous mining companies, become solid anti-mining activists. The net effect is that these NGOs work against most mining projects, mostly in developing countries, and especially those that do not conform to sustainable criteria. We argue that not acting in a responsible and sustainable manner poses a greater risk of potential new constraints due to the withdrawal of the public's acceptance, and thus, our social licence. Acting in a responsible and sustainable manner necessarily includes going beyond regulatory compliance with international safeguards treaties and other sustainable mining regulations to which mining companies subscribe.

The governing political climate is a major factor in the promotion of constraints. In discussions with industry practitioners, political risk is mentioned as the single largest factor in their consideration of the accessibility of potential reserves. Political risk can come in the form of:

(a) Excess regulation;
(b) Artificial access restrictions;
(c) Too little regulation;
(d) Barriers to trade and commerce;
(e) Bribery and corruption;
(f) Unstable investment climate;
(g) Lack of civil rights;
(h) Lack of law and order, and/or due process;
(i) Threat of nationalization;
(j) Conflict and insurgency.

This paper will only briefly deal with general political risk, but, fundamentally, constraints exist because of the reigning political climate in a given region.

It should also be noted that most of the constraints that the primary fuel cycle and transportation of that material have to deal with in assuring a steady future supply of fuel are not unique to the nuclear fuel cycle but reflect environmental issues that are common throughout the mining and energy sectors globally. The unique and perennial concerns regarding radiation, nuclear security and waste storage, however, tend to concentrate social and political pressure on the uranium industry, often imposing environmental constraints on this industry faster than on others.

The International Organization for Standardization defines the environment as comprising air, water, land, natural resources, flora, fauna, humans and their interrelations [3]. Acknowledging the interrelationships between these elements recognizes environmental, social and economic factors as being fundamental to a discussion of environmental constraints, as these three factors are the cornerstones for a balanced and sustainable approach to uranium mining and milling developments.

2. ENVIRONMENTAL CONSTRAINTS

Before discussing the specific constraints that might affect the future primary supply of uranium, it is worth defining what we mean by the term constraint. Dictionaries generally agree that constraint is synonymous with force, so constraints can be defined as the forces that act upon the primary fuel cycle and promote or restrict its development. All industry and development operate under varying levels of restraint that may exert a beneficial or negative effect, including:

(a) Environmental regulations;
(b) Health and safety regulations;
(c) Nuclear safety regulations;
(d) Robust environmental assessment, licensing and permitting processes;
(e) Social policy related to the rights of indigenous peoples and a consideration of their knowledge;
(f) International treaties, guidelines and standards (e.g. those of the IAEA and World Bank);
(g) The governing political environment.

These constraints generally impose a beneficial effect when they ensure reasonable protection of the public and environment during exploration, development, mining, and decommissioning and reclamation activities. Within the uranium mining industry such constraints are applied to the whole life cycle, manifesting themselves as the following types of conventional constraint:

(a) Regulatory controls over access and conduct during exploration;
(b) Environmental assessment, environmental protection, health and safety, and radiation protection regulations during all phases;
(c) During operation:
  (i) Waste rock handling and disposal;
  (ii) Tailings management;
  (iii) Source water protection and management (e.g. protection of surface water and groundwater and effluent treatment);
  (iv) Social issues (local jobs, training and benefits);
(d) Requirements for decommissioning and reclamation upon closure;
(e) Institutional control and long term monitoring following abandonment.

That these constraints vary somewhat globally does not lessen their significance.

Some constraints are necessary for the responsible and sustainable development of uranium resources. However, they do add to the costs of an operation, accounting for what have traditionally been viewed as external (and often ignored) costs of production. The current trend in the Western World, based upon the Canadian and US experiences, is in the direction of an increasing number of controls, which push the unit costs of primary supplies steadily upwards, and which may eventually tip the scales into the prohibitive cost range, depending on the demand–supply ratio and uranium market prices.

The sources of supply in the future will come from primary source mine deposits (in situ leaching (ISL), open pit or underground operations) or from some form of secondary supply (i.e. spent fuel reprocessing, blending down,

tailings or phosphate) as a by-product source. Again, most of the constraints for these sources are known and can be accounted for in the EA and licensing process for life cycle development.

Some constraints are inherent and unavoidable, due simply to the fact that they are tied to the physical location in which the primary supply is found. These include: the location of potential uranium deposits, local/jurisdictional laws and regulations, and, arguably, the price of uranium on the spot market. These constraints, while unavoidable, still play a large role in the economics of supply and its development. For instance, a potential deposit in an isolated remote location will bear substantial costs due to the lack of infrastructure and supporting population. In northern Saskatchewan, it is the relative richness of the deposits that allowed development and mining even during times of low commodity prices.

The main negative outcomes arising from unreasonable environmental constraints are substantially increased costs that impair the viability of extracting primary supply, and the loss of social licence to explore and develop new resources. A loss of social licence reflects a spiralling effect where loss of social licence begets further loss of social licence, locally or globally, as has been the case with the 'no uranium development' policies in parts of Australia and Canada. Negative environmental constraints initially increase costs by increasing the requirements for environmental assessment, licensing, social commitments and pollution control measures. The secondary outcomes of these negative environmental constraints include:

(a) Production delays;
(b) Restricted access to land for exploration;
(c) Political interference (e.g. social instability, corruption, conflict, nationalization or predatory taxation);
(d) Unattainable or unreasonable standards (i.e. zero impact);
(e) Limitations on movement of fuels;
(f) Moratoriums on nuclear energy and fuel development;
(g) Reduced public, international and regulatory support;
or, at worst:
(h) An outright ban on uranium development.

Examples in Canada of some of these constraints will be discussed below. The impact of these outcomes may be a limitation on uranium supply in the future due to a failure to access supplies or delays in bringing replacement reserves to the market. While the market may create forces that partially offset these negative consequences (e.g. higher spot prices), significant primary supply could be affected.

Most of the regulatory constraints appear to be cumulative, and become negative or unduly restrictive through an evolutionary process, often based on good intentions rather than good policy that meets the test of reasonableness. Examples are the progressive lowering of dose limits and contaminant concentrations in effluents without there being a large scale consensus as to the efficacy or reasonableness of these measures. Other constraints will arise due to some acute triggering event (e.g. Chernobyl, Three Mile Island or some conflict) which results in strong reactions from the public, special interest groups and government, and inevitably leads to more constraints.

Too few constraints can also pose a potential problem, where operators who do not subscribe to principles of sustainability and corporate social responsibility develop primary supply in an irresponsible manner. The supply is acquired, but at some cost to the reputation of the industry. We have only to look at some of the legacy environmental issues generated by the industry in many parts of the world. These legacy sites (e.g. the Atlas Moab tailings in Utah, Wismut in the former German Democratic Republic and the Uranium City area in Saskatchewan) have tarnished the reputation of uranium mining generally, especially since they were developed, at least initially, to supply uranium to the nuclear weapons programmes of the cold war superpowers. The mining and waste disposal problems arising from these legacy sites need not be repeated with responsible management and development.

3. DRIVERS OF CONSTRAINTS

What turns a generally beneficial constraint into a negative constraint? If the drift from reasonable to unreasonable is in general evolutionary rather than episodic, then what pressures are driving such processes? In our experience, it is some or all of the following factors that play some part in this trend to increased constraints:

(a) Valuing regulatory process over desired outcomes;
(b) Valuing public perception rather than measured effects;
(c) Applying absolute standards rather than relative or incremental standards;
(d) Using command and control regulation instead of responsible self-regulation and adaptive management;
(e) Requiring zero risk (or impact) rather than balanced and measured risk;
(f) Applying the precautionary principle in an overly conservative way;
(g) Interpreting ‘adverse impact’ or ‘reasonable’ measures inappropriately.

All of these factors are leading to increased restrictions on development, especially for extraction industries. The root cause, in our opinion, is a lack of appropriate risk informed, or risk based, decision making. This manifests itself as a fixation on the worst case scenario or the consequences arising from an activity or incident, with no regard for its likelihood of occurrence, the mitigatory measures in place to limit the consequences, or the benefits to society as a whole which that activity may provide.

Much of the risk misperception related to the uranium mining industry stems from a historical legacy of unsustainable mining practices, as well as the abhorrent misuse and application of nuclear technology in weapons production. While these sins of the past have strongly influenced current public attitudes, the reality is that mining can operate in a sustainable manner, supplying the fuel for a power sector that has a controlled, and relatively small, impact on the environment in comparison with the social and economic benefits it provides. For instance, Environment Canada, in putting forward its ruling on whether effluents from uranium mining facilities in northern Saskatchewan are toxic under the Canadian Environmental Protection Act (CEPA) [4], found that modern uranium mining sites, such as Cameco's McArthur River and Areva's McClean Lake sites, operating under sustainable mining practices and with current pollution control technology do not impose a toxic effect on the surrounding local environment [5].

4. CONSTRAINTS: THE CANADIAN EXPERIENCE OF THE URANIUM INDUSTRY

In Canada, the constraints on a timely licensing process include:

(a) A cumbersome environmental assessment process, under the Canadian Environmental Assessment Act (CEAA), primarily focused only on environmental outcomes;
(b) The additional Canadian Nuclear Safety Commission (CNSC) licensing and approvals process sequential to the CEAA process;
(c) Lack of compliance with Government of Canada policy and the Smart Regulation initiative requiring consideration of economic issues;
(d) Excessive conservatism and a precautionary attitude in decision making, with a lack of available arbitration on scientific issues;
(e) The expectations held with respect to institutional care.

Each of these constraints will be discussed below, with examples provided from Cameco's recent experiences. These examples are provided to show that

when a well intentioned, very precautionary and environmentally protectionist view of development is utilized to the exclusion of social and economic considerations, the perception of risk can become tied too closely to possible effects rather than to actual impacts. To be fair, many of the issues discussed in the following are recognized by the respective agencies, and some discussion and effort is being made to address these issues. However, at the time of writing of this paper, significant progress had not been made and, since most of the changes reside within the legislative process, rapid progress is not expected.

4.1. Environmental assessment

In Canada, the EA process for primary uranium supply is an unduly complicated process, notable for its lack of focus and lengthy time for completion. Comprehensive generic guidelines for EA production have led to a situation where all EAs must address a myriad of issues, regardless of their relevance. The notion of focused site-specific EAs has become lost in an apparent need to pacify and placate all stakeholders or from a fear of litigation. While there are many legitimate environmental issues associated with modern uranium developments, these obstacles are far from insurmountable, and much is known about their impacts and how to control them. The CEAA EA process has not kept pace with the current state of the art, reflecting, in many respects, the regulatory needs of the mid-1970s when environmental controls were less sophisticated.

Given all the issues that can be covered in an EA and the scope of work that is required to address these issues, it is very important for industry and the regulator to reach a reasonable agreement on the scope of the EA at a fairly detailed level early on in the project [2]. Such site specific guidance would add value to the decision making process, rather than divert scarce resources to satisfy regulatory process requirements on environmental issues that present a low risk. The problem lies in the additional time required to address the myriad of identified high and low risk issues through the regulatory channels, more so than with the content of the regulatory negotiations themselves. By having a ‘one size fits all’ EA approach, the issues of most concern are at risk of being overshadowed by efforts directed at relatively less important concerns. Hence, a balance is needed, with regulatory effort commensurate with the level of environmental risk. The EA process should prioritize environmental concerns based on a risk informed process that adds value and that assesses the sustainability (environmental, social and economic) of the project, rather than simply reiterating existing knowledge and information.

By carefully focusing the EA on the identified priority issues, all stakeholders will benefit by being able to concentrate on those potential issues that

present the highest risk. The EA must document the effort, as part of the environmental management plan, to control and prevent releases during routine operation, major environmental accidents and post-decommissioning. All three present predictive challenges, especially for new projects, where lack of monitoring data prevents verification using the observational method and instead modelling predictions alone must be resorted to. In the case of operational phase releases where limited monitoring data may be available, one must decide on which biological end points to test for, such as whether to test to the cellular, individual organism or population level. Furthermore, accident prevention, evaluation methods and design objectives must be clearly defined — such as probabilistic analysis and acceptance criteria versus adopting a more deterministic approach. For long term liability management issues, the end state decommissioning objectives need to be determined, such as long term passive control versus active institutional control, leachate control objectives and in situ treatment options versus relocation options for wastes [2].

CEPA [4], the governing Federal legislation in Canada, makes the following statement:

> "respecting pollution prevention and the protection of the environment and human health in order to contribute to sustainable development…[by endeavouring] to remove threats to biological diversity through pollution prevention, the control and management of the risk of any adverse effects of the use and release of toxic substances, pollutants and wastes, and the virtual elimination of persistent and bioaccumulative toxic substances."

CEPA is also premised on the precautionary principle, which states that: "lack of full scientific certainty shall not be used as a reason for postponing cost-effective measures to prevent environmental degradation" [4]. The corollary to this would be: lack of full scientific certainty shall not be used to invoke unreasonable measures where the potential risks associated with the activity are minimal compared with the benefits to society that are generated from the activity. Yet it appears that the emphasis has changed from a lack of consideration for environmental risk in decision making to a process that overcompensates for environmental risk without considering the full benefits of an activity. From our perspective, what is needed is a full consideration of the relevant social, environmental and economic factors to complete a truly sustainable assessment. Admittedly this becomes a difficult task when one must decide on the scope of an assessment, on what benefits and adverse effects will be considered and what to omit or where to limit one's

considerations. In the case of the assessment of radionuclides from nuclear facilities and their impact on non-human biota, known as the Priority Substance List (PSL) Assessment, the regulator (a joint responsibility shared by Environment Canada and Health Canada) has relied on a purely natural science assessment approach that invokes a limited consideration of all of the possible immediate and long term, near and long range impacts [5]. What it fails to do is recognize the full global benefits of a fuel that provides emissions-free power, which in turn contributes to global social development.

As an EA is developed, one is invariably faced with data gaps that hamper estimates of environmental effects. Lack of site-specific data, chemical speciation effects or the interaction between various chemical species in the effluent to mediate their potential toxicity are compounding factors. The regulatory response to such uncertainties is to apply the precautionary principle. However, this principle can be excessively applied to the point where it may go beyond all reasonable limits in bounding an assessment. In these cases, lack of full certainty should not be used as a justification to stop or delay projects that have demonstrated clear social and economic benefits. As such, an adaptive management approach should be taken utilizing operating experience and environmental research to inform ongoing decisions. However, while environmental research opportunities arise from these uncertainties, and usefully inform the overall adaptive environmental management programme, one cannot wait for basic research before an EA is completed, since research is most often a journey, not an end point [2]. Instead, since the operational influence has an effect on the ongoing research results, it is imperative that both be allowed to proceed in parallel.

A balance must be maintained between management of the environmental aspects of radionuclides and management of other metallic constituents of the ore. Naturally, we are expected to demonstrate good control of radionuclides, and there are few allied industry practices upon which our performance can be benchmarked. This is not the case in the management of non-nuclear substances in the ore. In northern Saskatchewan, the contaminants of concern are nickel, arsenic, molybdenum and selenium. There are allied industry benchmarks within other mining sectors, but the regulatory framework, with specialist nuclear regulation, often restricts the extent to which these external benchmarks can be applied [2].

4.2. Administration of the Canadian Environmental Assessment Act under the Canadian Nuclear Safety Commission

The CEAA (CEAA legislation) is designed to prevent significant adverse effects to the environment, while recognizing that the promotion of sustainable

development is also necessary in order to maintain a healthy environment and economy [6]. In our view the CEAA is designed to screen new, or greenfield, projects, albeit in a cumbersome manner, but it does not effectively deal with brownfield sites or modifications to existing facilities. For instance, a modification that maintains or improves environmental performance must undergo the same assessment process as any other type of project. As such, for an existing modern uranium facility in Canada, one that has been duly assessed and approved previously, virtually any modification or change will trigger the full EA process. As practiced, CEAA, in not providing incentives for improvements to environmental protection, can be a tremendous barrier or disincentive to continual improvement, innovation and enhanced environmental protection [7].

The reason for this is that CEAA legislation allows the primary regulating body, the CNSC in the case of the nuclear industry, to manage the EA process for screening level assessments. To date, screening level assessments have covered the vast majority of all assessments done by the CNSC. Comprehensive EAs are required by the CEAA if the proposed expansion would result in a production capacity increase of more than 35% [8]. In such cases, the management of comprehensive studies is performed by the CEAA agency itself. A comprehensive EA incorporates an enhanced process, in theory requiring more information than a screening level assessment and commensurate with the potential increase in risk to the environment. In practice, the CNSC conducts 'enhanced' screening level assessments that are comprehensive EAs in all but name for most assessments. In our experience, the EA process, with minor exceptions, has become an indiscriminate 'one size fits all' comprehensive process.

The process is complicated by the narrowness of the CEAA legislation, which does not allow social and economic factors to be considered at the screening and comprehensive EA levels unless there is a direct connection to an environmental impact. The use of the screening process allows the CNSC full discretion in deciding which impacts are potentially significant and adverse under the CEAA, and unreasonable under their own legislation. Thus, a re-examination of a project under the CEAA can conflict directly with historical decisions, such as those of the Joint Federal Provincial Panel on Uranium Developments in northern Saskatchewan and the resultant Government approvals, where social and economic considerations were considered. The CEAA does allow consideration of all three issues when a panel is convened to examine a project, but this is a longer, more involved, process, and is not often applied.

In practice, the process reopens prior assessments and allows the CNSC to hinge licensing decisions on the correction of historical or cumulative issues

based on prior decisions by others. This has had the effect on Cameco of delaying projects for increased uranium production and acceptance of recyclable material (with high uranium content), which is currently shipped to Utah, at our Key Lake and McArthur River operations, where the EAs indicate that there will be no material change in the effluent due to these changes. This inability to separate current from ongoing issues has already had an impact on timeliness and threatens to continue to do so. The issues in the instance of the Key Lake operation revolve around selenium and molybdenum in the treated effluents, even though the current level of selenium impact was predicted, and accepted in the original Key Lake environmental impact assessment process (project EIS) [9].

Downstream of the operation, there has been a buildup of selenium in fish flesh to levels where the hazard quotient (defined below), based upon the flesh values and literature information, indicates there may be an impact on fish reproduction. The selenium discharge at Key Lake has been relatively constant, with an average concentration over time of between 0.02 and 0.04 mg/L Se. Ecological risk assessment is driven by hazard quotients, which are based on the estimated concentration levels divided by a literature based effects level for the organism in question. The effects levels are comprised of a variety of values, all broadly representative of the no observable effects level or the lowest observable effects level with the addition of a safety factor. The safety factor, often a factor of 10 or 100, is added to account for uncertainty. In theory, the selenium levels in the Key Lake discharge per se should not be cause for concern; however, there is a buildup in downstream sediments (Delta Lake) where selenium is entering the food chain and accumulating in fish flesh, resulting in reduced reproduction levels. Laboratory studies, using fish from the receiving environment, have indicated an increase in larval fish abnormalities, and this may result in a slight decrease in fish populations. However, ongoing field monitoring programmes suggest that both healthy fish populations and benthic invertebrate communities are locally present, and the overall effects may be quite limited.

For molybdenum, the Key Lake operator is being asked to implement expensive molybdenum abatement measures to reduce the current level of molybdenum in the discharge from approximately 1.0 mg/L to 0.3 mg/L in the absence of any sign of harm downstream. Studies at Key Lake indicate that the mammalian populations, including those considered at risk from molybdenum, are healthy. However, as AREVA's McClean Lake operator treats their water to 0.3 mg/L, the Key Lake operator is expected to do the same. Thus, a precedent at one mine becomes a necessary standard for all others to follow, with the attendant increase in costs.

Dealing with these two issues has delayed both the production increase and the processing of recycling product by from one to three years, depending on the point of view taken. In our opinion, the risk to the environment is not clear, and may, in fact, not exist except theoretically. The potential impacts are certainly spatially and temporarily restricted. Clearly, this is an issue that requires careful monitoring and study, and, in a properly risk informed model, could be put into an adaptive management regime. Such a regime would combine, in agreement with the regulatory agencies, research with monitoring, to determine whether the impacts were indeed significant or are limited spatially and temporally as we believe. This would allow a separation of the legacy issues from current period requirements and allow the project to move forward. Placing the molybdenum and selenium issues into an adaptive management programme would neither pose an unreasonable risk to the environment nor result in a significant adverse impact.

4.3. Sustainability

In a balanced EA and licensing process, environmental, social and economic impacts should be effectively considered in its decision making. Within the Canadian context, economic considerations are given little effective weight, and in many instances are dismissed as simply self-serving to the proponents, who are perceived as having a purely mercenary desire to develop. While the need to have profitable companies is important, economic considerations are much broader and encompass issues such as cost effectiveness, benefits to Government and a sustainable economy. Despite the importance of such considerations, they are not part of the EA equation unless the proposed venture goes to a panel hearing.

A recent report commissioned by the Government of Canada [10] addresses the lack of connection between the Federal regulatory system and the current realities of most sectors of the Canadian economy. The Smart Regulation principles are a strategic objective designed to direct all Government decision making while supporting investments in natural resources. As its main tenet, they emphasize informed decision making through cooperative measures that require closer relations between Government, industry, citizens and other stakeholders. Smart Regulation initiatives recognize the public interest as involving social, economic and environmental objectives, as these, in balance, are the pillars of sustainable development.

As discussed previously, our experience with regulatory authorities is such that they often pursue only one objective (most often the environment) to the exclusion of other objectives, or even one objective to the detriment of the others (most often social and economic) because their governing legislation is

often written to address or emphasize only one of these elements of the public interest. Currently, Smart Regulation exists only as policy recommendations, and, as such, little attention has been paid to it, or it is ignored completely in the interpretion of departmental and agency mandates. For example, the CNSC's mandate of radiological protection precludes it from considering the national objective of promoting social and economic well-being. We have to question whether it truly is in the public interest to have a regulatory authority with such a narrow mandate without some opportunity for national oversight or review. From our perspective, this arrangement has certainly increased EA and licensing requirements to the detriment of timeliness, with little or no substantive environmental protection benefits being realized. The Government of Canada is currently in the process of elevating the Smart Regulation policy to a directive, and it is our hope this will require the CNSC and Environment Canada to be more fully accountable for complying with the Smart Regulation principles.

Once the EA process has been completed, proponents must seek the formal licensing approvals required under the Nuclear Safety and Control Act and associated regulations, which is the governing legislation for the CNSC. This sequential process results in an even longer process as the proponent works through the formal CNSC licensing submissions. It is understood that the regulatory agencies do not like licensing and EA proceeding concurrently, because they do not want to appear to be pre-approving a project. However, since a positive EA decision is just one part of the licensing decision, we do not see this concern as a valid one. While the CNSC has been willing to discuss the issue of EA and licensing, the concessions made have been limited, due in part to the constraints of their legislation and licensing process.

We believe that the weight of process for both an EA and licensing should be risk informed. As practiced elsewhere within the Federal Government, screening level EAs can be very limited in scope and time if the environmental risk is limited or an improvement is proposed. In addition, since the EA and licensing groups within the CNSC are separate, there should be no conflicts in licensing and EA proceeding in parallel. We will continue the dialogue with the CNSC in the hope of improving the process over time. Other initiatives, such as ecological risk assessments, may also help in speeding up processes.

It should be possible to demonstrate that as an industry and a company we have a proven track record in assessing and developing new projects such that subsequent increases to production and new developments do not need such a lengthy and complex regulatory process. If Cameco is uncertain about its ability to economically go through the regulatory approvals process, we ask what is the likelihood of a poorly funded junior company with a new discovery

in Canada doing so? This uncertainty has and will continue to have an impact on capital allocation decisions.

To place the Canadian regulatory process in some perspective, Paladin Resources, a junior resource company, submitted an EA on the Langer Heinrich deposit in Namibia in 2004 and have recently received permission to develop the mine [11], with an expected startup in September 2006. While one might argue about the level of detail in parts of the EA, they appear on the surface to have conformed to international requirements in producing a relatively comprehensive EA, addressing those issues that should be addressed, including sustainability. Two years from EA submission to production is extremely fast by Canadian standards, but if the environment is protected and social considerations properly addressed, this should be possible even in Canada.

4.4. Conservatism and precaution

In environmental assessment there is always a degree of uncertainty as to the severity of the consequences, and the likelihood of their occurrence. While this can be addressed in a number of ways, conservatism is commonly used to account for these uncertainties. In this form, conservatism represents an effort to err on the side of caution where not all outcomes can be fully known. Unfortunately, in the regulatory arena there has been a tendency to compound conservatisms, which has the effect of requiring a very high level of protection, beyond that necessary to meet the objectives of ALARA (as low as reasonably achievable). The result is unnecessary increases in mitigation costs and delays in the approval process.

A recent CNSC draft guideline on the assessment criteria for waste facilities [12] requires conservatism in several areas and compliance with a reduced public dose objective of 0.14 mSv/a. With compounding conservatisms, it is questionable whether any waste facility will even meet the dose criteria or, if so, only after extreme effort. We question whether it is necessary to set an apparently arbitrary standard that is not internationally recognized. This is an example of the ongoing misuse of the precautionary principle in the area of dose limits, where there is pressure to drive dose limits lower and lower, even in the absence of evidence of harm.

Regardless of whether one is a proponent of the linear non-threshold dose response or the non-linear threshold model, doses at the low end of the spectrum, where most primary uranium producers operate, can be expected to do no harm. Recent studies on Saskatchewan uranium miners concluded that there was no evidence of health effects from radiation exposure in the modern context. That is, for modern mines with modern practices of radiation

protection, workers are protected. Furthermore, this finding supports the view that members of the public are protected at a dose limit of 1.0 mSv/a. Why then add additional costs to a project to achieve dose targets that are far below that required for the protection of people or the environment?

Dose to the environment does not appear to be a major hazard in modern uranium developments in Saskatchewan. The ecological risk assessment carried out for McArthur River, a modern mine, determined that radiation from the radionuclides projected to be released from the project was unlikely to have a negative impact [13]. Even current revisions to the risk assessment for the project, including the CEPA priority substances list determination [5], have not changed this conclusion (although they have highlighted several metallic toxicity issues that previously went unrecognized, including molybdenum and selenium, as previously discussed). The radiotoxicity of uranium (and its daughter nuclides) was determined not to be a 'serious threat', as risk quotient calculations for ionizing radiation indicated limited potential for detrimental effects on the environment.

In older operations, dose to the environment has been raised as an issue. For Key Lake and Rabbit Lake, CEPA determined that harm to the environment from effluent only occurs in the near field environment at the mine and milling facilities, and that some of the harm is due to past releases. Part of their determination relied on assessing the 'significance' of the effect, including both short and long term effects. In addition, although the indicators of risk were fairly low, it was

> "nevertheless believed that an increase in environmental concentrations of radionuclides could significantly increase risks to both aquatic and terrestrial organisms, particularly at uranium mines and mills and stand-alone waste management facilities" (see Ref. [5]).

The decision by Environment Canada to declare older mines and mills 'toxic' reflects a position that "no level of damage to individual organisms by ionising radiation is tolerable" [5]. Environment Canada is saying that no risk is acceptable, and yet it is known that there is virtually no human activity that does not carry some level of risk, no matter how small. Indeed, a balance of the benefits of uranium from the safe production of abundant amounts of electricity for developed and emerging communities globally would strongly suggest that the benefits far outweigh the apparent risks of these projects.

The duties of the Canadian Federal Government, including the CNSC, Canadian Environmental Assessment Agency and Environment Canada, under both Smart Regulation and the CEPA, are to take preventive and remedial measures to protect, enhance and restore the environment, although

short and long term human and ecological benefits, positive economic impacts and any other benefits must also be considered [14]. Policy has been developed at the CNSC for considering any relevant information on costs and benefits in relation to decisions involving a licence or orders, and while such cost and benefit information may be quantitative or qualitative in nature, it cannot displace other valid regulatory considerations. On the basis of the responsibilities of both the CNSC and the CEAA as Federal Government agencies, we believe that they have the scope to include cost–benefit analyses in their evaluations.

The net effect within the nuclear industry is that projects in the absence of appropriate cost–benefit analyses can be unduly delayed on single points of scientific disagreement. That disagreement usually centres on the level of uncertainty regarding a potential impact. For instance, at Cameco's Cigar Lake project the EA process stalled regarding a fundamental disagreement on the potential impacts of molybdenum in the projected effluent. The CNSC claimed that the proposed levels (3.0 mg/L) posed an unreasonable risk to the environment, particularly to terrestrial animals in the near field. Cameco countered that molybdenum treatment was not necessary because of the low densities of animals in the area (ungulates, for instance, have very large territories) and that the levels of molybdenum in the effluent were protective of the aquatic environment. Cameco proposed an adaptive management programme, but this option was rejected. The net result was the construction of a molybdenum treatment circuit for approximately two million dollars (plus annual operating costs of approximately 400 000 dollars) to ostensibly save, over the life of the project, one moose, and prevent breeding failure in one family of muskrats, a common northern rodent. Arguably, the potential risk to the environment is insignificant relative to the substantial economic and social benefits from the development. However, the regulatory decisions are fully consistent with CNSC's view of no level of risk to an individual organism, apparently at any cost.

Despite the best arguments of specialists, Cameco was placed in the position of agreeing to the treatment or suffer a substantial delay in EA approval prior to licensing. We feel that this is a type of dispute that would have benefited from some form of arbitration or impartial review process. That is, it could have been removed from the EA review process, and made subject to an independent decision. In our minds, this would have removed elements of bias from the decision making process, and allowed the proponent and the regulators to concentrate on the issues of EA approval and licensing. Such a scientific arbitration procedure does not exist within the Canadian regulatory process.

4.5. Decommissioning, reclamation and abandonment

Additional barriers to the development of new deposits, or even to the reworking of old ones, are among a number of developing trends related to the lack of decommissioning criteria, and the potential inability to abandon lands upon completion. It is always the industry's goal to leave an area in a safe non-polluting condition with a land use similar to that which existed before mining, usually wildlife habitat. As legacy sites approach decommissioning, we are finding that the regulatory requirements are increasing, with a tendency towards increasingly more stringent requirements to return sites to their previous states.

In the absence of decommissioning standards, the tendency is to demand absolute decommissioning, aimed at zero risk as opposed to 'managed' risk. For instance, at the Cameco operated Beaverlodge decommissioning project in northern Saskatchewan, numerous rounds of decommissioning and reclamation work have been performed on properties due for licence removal, often on aspects that had not previously been considered significant. This has been further compounded by the fact that the Federal and provincial regulators had differing requirements. The lack of clear, consistent, reclamation standards left the field open to the regulators (often on what appeared to be a whim) asking for anything they determined was necessary to bring the properties to the point of regulatory release. While many of the remediation projects had a relatively small cost, with the limited seasonal access to the area and lack of infrastructure, the ability to advance these projects to some conclusion has been impaired for years, with an attendant increase in costs.

The recent guidance from the CNSC on assessing waste management facilities [12] indicates that the hurdles for constructing a facility and preparing for its ultimate closure will become increasingly more challenging, not easier. The only practical way of designing effectively to the CNSC criteria may be to construct purpose-built engineered tailings or disposal facilities, even if an open pit mine is available for the disposal of tailings. The net result for providing this assurance, with no available standard for what is reasonable, will probably be very expensive and may have a severe impact on the economics of a project.

Perpetual care and long term institutional care issues have recently come to the forefront in several jurisdictions in Canada due to the recent applications to release properties from licence (e.g. portions of Cameco's Beaverlodge project). For example, in Saskatchewan, a proposed institutional control management strategy [15], with a supporting act of legislation, has just been issued for comment. The draft of this strategy indicates that any returned land must be safe and non-polluting; however, should conditions at the facility change, the original owners of the land would be required to remedy the

situation. In keeping with IAEA guidance, it contemplates long term institutional control and monitoring for facilities that retain significant quantities of radionuclides. At the Federal level, the CNSC requires any property with greater than 10^{15} Bq contained activity to remain under licence in perpetuity. This has been interpreted very literally, and poses unnecessary restrictions to release where there are quantities of waste rock with trace levels of natural uranium that exceed this threshold amount. This is an area where the ability to risk inform the decision would benefit both the regulator and the regulated. While a de minimis criterion for natural uranium in waste rock would be helpful, the ability to develop a risk analysis based upon the physical and contaminant transport properties would be more useful.

For properties that cannot be released to the Crown, perpetual monitoring will be required, and the original owner will be responsible for ensuring that financial resources are available to fund this. The CNSC and the province of Saskatchewan are considering mechanisms whereby the province might assume ownership, with ongoing institutional control being paid for by a fund set up by the original landowner. The obstacle at the moment appears to be that the province itself does not want to become the CNSC licence holder and be bound by their regulatory requirements.

The regulatory requirements associated with decommissioning, reclamation and abandonment need to be documented as policy level statements by the regulators so that the full costs of development can be accounted for at the start of the process. Furthermore, there needs to be some mechanism to ensure that projects that complied with the best practices in existence during their operation are not unduly penalized by changing requirements later on. Without reasonable limits, the potential outcomes of these policies will be to discourage prospective investors, since the full costs of the project cannot be determined. Uncertainty always increases the business risk and associated cost of an investment, and, for mining developments to remain financially viable, these issues need to be resolved in order to reduce the costs associated with these uncertainties. Only by clearly defining the standards for release from regulatory control can the uncertainties be removed.

5. POLITICAL ENVIRONMENT

The above description of constraints in the context of the Canadian nuclear industry illustrates a portion of the political spectrum that controls constraints, one that is more typically representative of the Western World. Arguably, it is an example of too much regulation, or at least too much process.

The net risk to future supply in Canada is largely in delays to the development of projects, which will have an impact on the timeliness of supply.

Even in countries with highly regulated environments, such as Canada and Australia, there are differences in constraints. For instance, within Canada, regional political situations further limit uranium mining and the potential development of reserves. Both the territory of Nunavut, host to the Thelon Basin deposits of Kiggavik, Andrews Lake and End Grid deposits [1], and the province of British Columbia have unofficial and official moratoriums, respectively, on uranium mining. Similarly, Queensland and Western Australia have uranium mining bans in place that affect a number of potential deposits (e.g. the Kintyre and Yeelirrie deposits), while uranium development in Australia's Northern Territory is heavily restricted by Aboriginal land entitlement and the Kakadu National Park. Recent announcements by the Australian Federal Government, signalling a potential shift in that country's political climate, indicate plans to open up the Northern Territory to uranium development [16].

These are examples of constraints on the potential exploration and development of future uranium supply in countries that are considered politically stable. Many of the potential areas of exploration and development lie in areas that are less politically stable. Political stability is certainly essential in providing the security of person, process and investment required to support investment in future supply. Stable government is also needed to ensure that companies that wish to explore and develop in a responsible manner can do so.

Cameco's investment and development in the Kumtor gold mine in Kyrgyzstan provides a model for successful development in less developed areas. The mine, built and operated on Western principles for the protection of the environment and workers, has been a huge success. In developing the mine, Cameco set the default requirements for Kumtor at the most stringent of Kyrgyz, Saskatchewan/Canadian and World Bank standards. Cameco's experience working with northern and Aboriginal groups in northern Saskatchewan, both from a business development and employment standpoint, also proved to be a large part of the Kumtor success, as the mine now employs about 90% nationals in its workforce. We believe our success at Kumtor, which we are now applying to the Inkai ISL development in Kazakhstan, provides an example of responsible development that can be applied to primary uranium supply.

6. TRANSPORTATION

Historically, the transportation of radioactive materials has always come under public and regulatory scrutiny, and as a result, there exists, in Canada

and internationally, excellent legislation and standards for determining the appropriate method for their safe shipment. Unfortunately, we have recently seen international shipping companies become increasingly nervous about the potential for small amounts of radioactivity to be left in shipping containers previously used for shipping drums of yellow cake. It appears that any amount is too much in their view, owing to a general ignorance of the risks involved. Although the levels of radiation are often below our cleanup criteria, these occurrences generate uncertainty and doubt with shippers and consigners over the safety of the non-uranium products being transported and their liability in the event of any complaint. While there are solutions to this (e.g. industry dedicated containers for the shipment of Class 7 low specific activity radioactive materials), they are all more expensive, and will generate delays in shipment. As a result, for some time now, there have been indications that shippers do not want to be encumbered with the potential work involved in, and liability related to, the shipping of uranium oxides in their containers, given that these do not represent a large percentage of their revenues. It is suggested that clearer international de minimis standards for cleanup criteria and radiation levels would help put the risk in perspective for shippers. For shipping companies, as with most of the general public, the scientific basis for what is safe, and what is not, when related to nuclear materials is not yet widely understood or accepted. This indicates a general need for education broadly of the general public and specifically of the transportation sector.

The fear is that gradually shipping companies will decide not to ship or receive uranium oxide or hexafluoride internationally. For instance, many Australian ports refuse to handle radioactive materials. To combat this trend, consignors and receivers of low activity fuel materials need to be aware of the growing threats to using containers previously used to ship radioactive materials, and to ensure that all shipments are clean and properly packaged, and that all spilled materials are thoroughly cleaned up.

Recently, the Port of Rotterdam has started to subject all containers to a screening for radioactivity, and to require a transport licence for containers having more than 10 kBq of activity. This translates into less than a half a gram of natural uranium dust distributed throughout a container, and represents an effective cleanup level of 0.07 Bq/cm^2. This is well below the normal cleanup criteria for the free release of materials from a site or for surface contamination on a package (0.4 Bq/cm^2). In practice, a cleanup criterion of 0.1 Bq/cm^2 should alleviate the situation, but this will require a willing and vigilant industry. It should be noted that these limits at Rotterdam, which has a large number of nuclear shipments, do not apply to properly documented shipments of Class 7 material.

7. DISCUSSION

Environmental constraints are a necessary component of responsible and sustainable primary supply cycle developments. They provide the boundaries within which we operate through the full life cycle of the uranium supply side. The problems for our industry arise when seemingly reasonable constraints to protect the environment evolve into unreasonable requirements, especially when there are no attendant significant benefits to the environment. This trend appears to have arisen from a strong precautionary and environmentally protectionist view of uranium development, to the exclusion of compelling social and economic considerations. It is represented by the protection of the individual biota, be that a large ungulate or a cellular level animal. As such, the perception of risk in isolation to an effective socioeconomic cost–benefit consideration becomes tied too closely to possible, or even imagined, effects rather than actual impacts. The net effect for the uranium industry in Canada has been that EAs cannot be accepted until it can be demonstrated that there is effectively zero impact (when all compounding conservatisms are included). The uranium industry as a whole strives for ALARA impacts, but given the limited nature of the current impacts, both temporally and spatially, the current regulatory trend seems to be driven by philosophical pursuits, rather than a balanced, risk informed, management approach.

While we have focused on the burgeoning constraints imposed by the Canadian regulatory framework, as we are most familiar with them, we believe that the trends are illustrative of what is generally occurring in the Western world. The CNSC believes it is "one of the most rigorous and modern nuclear regulatory regimens in the world" [17], and we would not disagree. However, we believe that the Canadian Nuclear Safety Control Act, in combination with the CEAA legislation, is the largest single constraint to the timely development of new primary supply in Canada, because the regulatory process is not effectively balanced to include social, environmental and economic considerations, despite it being a policy directive of the Government of Canada. The net effect is that the process, while ostensibly risk informed, is not risk informed in a sustainable manner. Until the proponent has effective means of appeal in EA decisions, the process will not be fair, effective or timely.

Maxey, a noted professor of bioethics, has stated: "Demands for safety by any constituency that disdains, or remains oblivious to, costs is ethically indefensible" [18]. While Canadian policy requires consideration of sustainable development principles, recognizing that economic development is as important as environmental protection, this policy is not currently binding on Government departments. The net effect is that regulators with a narrow regulatory focus can ignore the broader principles needed for a sustainable

economy. By ignoring Government policy, reasonable constraints become unreasonable. The state of the art in mining should mean that there are fewer barriers to sustainable and responsible development of primary supply, not more, and in the presence of a well managed regulatory environment, exploration and development should proceed in a timely manner. Merely outlawing hypothetical risks takes little courage on the part of the regulator and produces only hypothetical benefits to the environment [18].

Globally, the development of primary supply will take place in a wide range of political environments, with a wide range in the level of regulation and security afforded by the developer. Even as the nuclear renaissance is gaining momentum, regulatory constraints are increasing with benefit to neither the environment nor industry. Whether the political environment creates a situation where too much regulation and process exist or the security of investment is uncertain, the ability to explore and develop is being constrained. In many respects, the regulatory environment in Canada may be as uncertain to investors as that in a developing country, because of the burgeoning process of regulation and the resultant lengthy periods between discovery and operation. Indeed, Cameco's newest operation is the Inkai project in Kazakhstan, where uranium development is supported by the Government, and has timely approvals, but not at the expense of the environment.

If the nuclear objective is to provide clean electric power from fuel that has been produced in as sustainable a manner as possible, then there are responsibilities on governments, developers and global organizations to realize this objective. Governments must work to ensure that regulatory approvals and development proceed in a timely manner and that the environment and human health are reasonably protected. While arguably the Langer Heinrich deposit in Namibia has moved very quickly though the EA process, only time will tell if that level of effort was sufficient; the lengthy process afforded Canadian operations represents the other extreme. A balanced approach to assessing risk in the context of sustainable development would better balance the objectives of environmental protection and social and economic development, and go part of the way to facilitating the timely introduction of new supplies to the market.

In developing countries, where the political and investment climates may be uncertain, governments need to develop both the legislation and infrastructure to allow timely development. In the absence of regulatory guidance, developers need to adopt appropriate standards for environmental protection created by global institutions such as the IAEA and the World Bank. The responsible development of reserves based upon these international standards will ensure that all developments are environmentally, socially and economically viable. Failure to practice 'safe mining and development' will invariably

lead to increased international constraints. The industry as a whole does not need to be the target of a 'dirty uranium' campaign modelled on past gold or diamond campaigns. To avoid this, the industry needs to be self-regulating, eschewing development in countries that do not support socially balanced development. Furthermore, fuel producers should boycott primary supplies coming from such sources.

Only by promoting sustainable uranium development in the manner of mining initiatives such as the International Council on Mining and Metal's Global Mining Initiative [19], the sustainability principles of the National Mining Association in the United States of America [20] and the Towards Sustainable Mining programme of the Mining Association of Canada [21] can increasing regulatory constraints for excessive environmental controls be averted. The Australian Government also has an excellent series of guidance documents on the best environmental practices in mining that have a very strong sustainability message [22]. All of these initiatives promote development of mining, milling and refining projects in a sustainable manner and set standards, especially in the areas of greatest potential risk to the environment. A track record of responsible mining will help counter the anti-mining campaigns, and primary uranium producers must meet their responsibilities and produce sustainable projects with regard to the full life cycle of the endeavour.

International agencies also have to come to the fore and generate reasonable risk-based criteria for uranium development to facilitate commerce. The IAEA, for instance, should be a strong advocate of de minimis standards for commodities of trade, waste rock, decommissioning, materials for disposal, reuse or recycling, and abandonment. Such standards would bring some measure of certainty to the life cycle of primary supply, much as they have done to the transportation of radioactive materials. Clear standards would improve the ability to communicate risk to the public and provide assurance that the level of risk was acceptably low. This has application throughout the life cycle of primary supply projects, but it also has utility in reducing waste through increased reuse and recycling. Possibly, with clear guidelines the Port of Rotterdam would not feel obliged to set its own guidelines, which in the absence of clear standards ended up being well below any reasonable level of risk.

Cameco intends to continue exploring for, and developing, new sources of primary uranium supply. Cameco has evolved its operations in ways that embody the principles of sustainable development at all of its operations (northern Saskatchewan, US ISL sites, the Kumtor gold mine, Kyrgyzstan, and the Inkai project in Kazakhstan). What Cameco has not done well to date is to monitor and document its commitment to sustainability. It is embarking on that

route with the issuance of its first sustainability report later this year. The report, while reflecting our commitment to sustainable development, is also a public record of our efforts to continually improve in this regard. Combined with recent actions to integrate formalized environmental, health and safety, and quality management systems, Cameco feels that it is in a strong leadership position within the nuclear industry, especially regarding the development and production of primary uranium supplies.

8. CONCLUSIONS

If nuclear power is to be regarded as a clean, safe and dependable energy source, then the primary uranium supply industry, through its full life cycle, must ensure that development proceeds in a sustainable manner. Because of the added scrutiny the nuclear industry attracts, primary uranium producers must be leaders in sustainable exploration, development, mining, milling, refining and conversion practices. This means going beyond regulatory compliance.

The emerging constraints that threaten the timely development of primary uranium supply tend to be increases in the requirements for environmental controls. At the root of the problem is legislation that is focused in a narrow area of interest to the exclusion of balanced sustainable considerations of the environmental, social and economic aspects of a project. This is expressed through decisions made solely on the basis of environmental considerations, to the exclusion of economic, and to a lesser extent social, considerations. The manifestations of this are, for example, lengthy processes, increased costs, uncertainties of process, potential restrictions on transportation and lack of access, all of which have an impact on the ability to replace primary supplies in the future.

In an increasingly complex global economy, the threat of additional constraints on the exploration and development of primary uranium supply is likely to increase rather than decrease. The nuclear industry can mitigate the imposition of additional restraints to some extent by choosing to explore and develop in a responsible manner. Whether it is considered a sustainable development approach or one of corporate social responsibility, the balance between social, environmental and economic needs must be maintained to remain viable.

To support the industry's development, international bodies need to set clear guidance as to what is deemed acceptable in the broadest context possible and what is not, especially with respect to providing definitive, risk informed, de minimis guidelines to facilitate decommissioning and abandonment,

transportation, and the reuse and recycling of materials. Such strong guidance may also help lift the uranium moratoriums in Canada and Australia. While zero risk is a noble objective, no activity is devoid of risk. It is the management of risk and a full consideration of the social, economic and environmental costs and benefits that determines whether a risk is acceptable or not.

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BACK END FUEL CYCLE STRATEGIES

(Session 2)

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FISSILE MATERIALS MANAGEMENT STRATEGIES FOR SUSTAINABLE NUCLEAR ENERGY: BACK END FUEL CYCLE OPTIONS

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Abstract

Most back end technologies have been industrially applied for many years without problems for the two fuel cycle options (direct disposal of the spent fuel or recycling of fissile material) or developed to the stage of assured feasibility (final disposal). The evolution scenarios of installed nuclear capacity, reactor types and fuel characteristics are predictable up to 2050. Since sustainability of the nuclear power option intrinsically implies the potential of future utilization of the fissile material still present in spent fuel, plutonium and uranium, the conditions and timeliness of their reuse must be considered. Separating and burning plutonium as mixed oxide fuel reduces the long term legacy of fuel cycle back end waste. The recycling in light water reactors (LWRs) is currently the only possibility and is implemented industrially. However, this increases the americium and curium inventories and cannot consume the plutonium extensively. Long term storage of spent fuel or separated plutonium until the time has come to deploy fast reactors is not the best solution, as the fissile worth of plutonium degrades with storage time. When the deployment of fast reactors becomes a necessity, the use of plutonium with deteriorated characteristics due to previous recycling in LWRs will not be a problem, as fast reactors are less sensitive to the isotopic composition of the plutonium and can, to a major extent, eliminate americium. On the contrary, reprocessed uranium practically does not age at all and can be kept as a stockpiled fissile resource almost indefinitely. Significant industrial recycling of reprocessed uranium in LWRs and RBMKs has demonstrated its value. The economics of the fuel cycle are slightly affected by recycling of plutonium and/or uranium, at current U_3O_8 and other front end prices. However, this is only one of the criteria that must be taken into consideration in the management of fissile resources.

1. INTRODUCTION

Worldwide roughly 10 000 t HM of spent nuclear fuel (SNF) are discharged annually from nuclear power plants (NPPs). While some 15% of this is being reprocessed, a once-through cycle is currently selected for the majority of the SNF, either because the residual fissile content is too low to justify recuperation, as a result of political decisions or for economic considerations. When the choice is open, a 'wait and see' policy is applied. Implementation of the latter two policies requires interim, perhaps long term, storage.

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Since the fissile material in SNF is a fuel resource that will be required sooner or later in a perspective of sustainability of the nuclear option, the question is when to recuperate this fissile material and for which reactor type. Timeliness of reprocessing and of reuse of the recuperated plutonium and uranium must take into consideration ageing characteristics, economics and policy selection criteria.

In France, 75% of the annual arisings of SNF is being reprocessed in a medium and long term nuclear perspective [1]. All the plutonium is recycled as mixed oxide (MOX) fuel in pressurized water reactors (PWRs) and produces 8–10% of French nuclear electricity [2]. Only part of the recuperated uranium (RepU) is re-enriched and recycled in PWRs. Most of it is stored. If all of the RepU were to be recycled in PWRs, it could cover up to 10% of the fuel resources necessary for feeding NPPs [2].

However, if recycled in fast reactors, this plutonium and RepU could generate 100% of the production. If operated as breeders, more fissile material could be generated than produced, while the plutonium and RepU are degraded by recycling in light water reactors (LWRs). However, French policy is based on the belief that fast breeder reactors (FBRs) will have to be deployed in an uncertain future in the period 2040–2080.

Given such a paradox, what are the factors to be taken into account? How can preparations best be made for a sustainable future?

Quantitative evaluations of likely scenarios of increasing nuclear generation capacity and of evolution of the fuel characteristics can be made only for the first part of this century. Up to 2050, current fuel cycle technologies and their evolutionary improvements will dominate the scene, owing to the timescales required to deploy novel technologies industrially. It is in this framework that the assessment reported hereafter has been elaborated.

2. FUEL CYCLE OPTIONS

2.1. OPTIONS

There are currently three options being followed for the management of spent nuclear fuel. These options can be described as:

(1) Reprocessing and recycle of valuable fissile materials (closed fuel cycles);
(2) Once-through cycles (open fuel cycles);
(3) The wait and see option.

Whilst the wait and see option must ultimately evolve into reprocessing and recycle or the once-through option, at present it is a valid management strategy that simply focuses on the long term storage of spent fuel.

Taking each of these management strategies in turn, the rationale for choosing such a strategy and its evolution over time will be discussed for different countries. Not all national strategies will be described, rather examples that illustrate the evolving drivers for each strategy will be discussed.

2.1.1. Reprocessing and recycle

During the early years of nuclear power, the reprocessing and recycle option was chosen primarily for two reasons:

(1) The first reason was the technical availability of the option. Reprocessing was a clearly understood technology and had already reached a considerable level of maturity by the time of increased civil use of nuclear power. For certain early fuel designs, such as the metallic Magnox fuel in the United Kingdom (UK), reprocessing was the only technically viable option for managing spent fuel.
(2) The second reason for countries opting for the reprocessing and recycle option arose from the energy value of the residual uranium and plutonium separated from the wastes during reprocessing. This energy value could be multiplied many times if recycled in fast reactors and, during the early years of nuclear power development, the goal of recycle in fast reactors was pursued in France, Germany, Japan, the USSR, the UK and the United States of America (USA). China, France, India, Japan and the Russian Federation are all still domestically pursuing the development of fast reactor systems to increase the value of the energy independence conferred by reprocessing and recycle.

Other countries, such as Belgium, Italy, the Netherlands and Switzerland, for example, have chosen to reprocess their spent fuel arisings internationally in the past, on account of the availability of reprocessing services being offered on the international market and the economies of scale offered compared with developing indigenous capabilities.

Some countries are considering, as potential options, alternatives to the standard technology. The Republic of Korea is actively developing the DUPIC (Direct Use of PWR In CANDU) concept whereby spent PWR fuel is recycled (without separation of fission products and reusable U/Pu). The Russian Federation is developing electrorefining of the SNF and vibrationally packed (vipac) refabrication.

2.1.2. The once-through option

The once-through option has tended to be followed as a strategy for the management of spent fuel for a number of disparate reasons. These reasons have included countries having an abundance of natural resources and having therefore no interest in longer term recycling, questions of economics, concerns about the proliferation issues associated with reprocessing and/or anti-nuclear pressure as a route to phasing out nuclear generation as a whole. There is no one reason that countries have decided to follow this particular management option.

A number of countries have switched from the reprocessing and recycle option wholly to the once-through option; countries such as Canada, Germany, Sweden and the USA are at the forefront of this trend. The USA is, however, revisiting this option [3].

2.1.3. The wait and see option

This option is currently being followed by an increasing number of countries and individual utilities. The option in itself is simply long term storage with no commitment to either direct disposal or reprocessing of spent fuel. The reasons utilities or countries are committing to this strategy are economics, the long timescales involved in securing appropriate disposal facilities, uncertainties about the future energy requirements or simply lack of political guidance in overall national strategy.

Some countries, such as Belgium and Switzerland, have switched from the reprocessing and recycle option to the wait and see option.

Many countries are following dual strategies, for instance all countries that are committed to a reprocessing and recycle strategy are also committed to a wait and see strategy for a proportion of their spent fuel arisings. For example, in the UK, not all advanced gas cooled reactor (AGR) fuel will be reprocessed, and no decision has yet been made on the management option for Sizewell B PWR fuel. In France, not all spent fuel arisings are being reprocessed; spent MOX fuel and some UO_2 fuel are currently being stored.

2.2. OUTLOOK AND CHALLENGES FOR FISSILE MATERIAL MANAGEMENT IN THE FUTURE FUEL CYCLES

Around 1990, countries implementing nuclear power programmes substantially reduced or even cancelled fast breeder reactor (FBR) development, which had aimed at closing the fuel cycle with FBRs to utilize

plutonium as a fuel material. In these new realities, however, sustainability of nuclear energy seems to be currently at a pivotal point, for example, seeking a breakthrough by exploring innovative technologies in the nuclear fuel cycle or by phasing out nuclear energy.

The OECD Nuclear Energy Agency (OECD/NEA) [4] has made a comprehensive review of the nuclear fuel cycle taking account of high temperature reactors and LWRs, with various plutonium utilization options, to show that plutonium strategies in the medium term (the period after existing LWRs have closed and the startup of long term sustainable reactors) can be consistent with long term strategies ranging from a gradual phase-out of nuclear energy to the introduction of sustainable systems. The long term plutonium management strategies reviewed are:

(a) Spent fuel interim storage with deferred reprocessing;
(b) Prompt reprocessing with no recycle;
(c) Prompt reprocessing with single recycle in LWRs;
(d) Multiple recycle in LWRs;
(e) Prompt reprocessing with recycle in high moderation LWRs;
(f) Prompt reprocessing with recycle in low moderation LWRs;
(g) Phase-out of nuclear power.

The principal advantages of spent fuel interim storage with deferred reprocessing are relatively low storage cost and protected storage of plutonium in spent fuel. However, it is disadvantageous that the energy content of plutonium is unused (and degrades) during the interim storage period. The prompt reprocessing with single recycle using MOX fuel in LWRs has been successfully conducted. The VISTA calculations [5] are based on this strategy. As is indicated, about 7500–12 000 t HM are added annually over 50 years to the inventory in storage, which may bring about serious concerns about the capacities of storage and disposal facilities if appropriate measures are not taken. It is concluded that much of the advanced fuel cycle mentioned above has the potential to deliver benefits over the single MOX recycle approach.

The United States Department of Energy (USDOE) sent a report to Congress [6] about the significance of spent fuel treatment and transmutation, which is designated the Advanced Fuel Cycle Initiative (AFCI). The AFCI aims at finding technologies to reduce spent fuel volume and to separate out long lived, highly toxic elements. The increasing volume of plutonium in spent fuel was another concern expressed. In the case of the USA, which generates 2000 t HM of additional spent fuel annually, about 25% of the world total, the statutory limit for the planned geological repository, 63 000 t of civil nuclear spent fuel, will be reached by 2015. It is emphasized in this paper that the

quantity of spent fuel produced by NPPs may become a long term challenge to the possibility of building new NPPs.

Therefore, against the backdrop of such circumstances, which emphasize the need for development of innovative technologies, an OECD/NEA report [7], which reviewed trends in the nuclear fuel cycle, proposes challenges for future development such as:

(a) Multilateral or international cooperation R&D programmes should be conducted, since any advanced fuel cycle with an innovative reactor system needs a lengthy and expensive process.
(b) Final disposal facilities should be operated, to demonstrate to the public that the industry is responsible in managing its waste and disposal systems.
(c) The nuclear fuel cycle should help to minimize the potential for diversion of nuclear materials for weapons purposes, i.e. by the disposition of ex-weapons materials, reducing plutonium inventory and other measures.

3. NUCLEAR REACTOR SCENARIOS

The use of nuclear power for producing electricity, in operation since the late 1950s, has matured and continues to expand with an increasing output of electricity. Current NPPs are larger and better than the plants of a few decades ago, and exhibit higher efficiencies, longer lives and improved safety in operation.

Safety is paramount in the operation of an NPP. The nuclear industry, around the world, is enhancing the safety of plants, aware that sustainable nuclear energy is based on a prerequisite of safe design, construction and operation.

In the early days, there were a large number of competing technologies to produce nuclear electricity. Three basic concepts have emerged:

(1) Light water reactors (PWRs, boiling water reactors (BWRs) and WWERs);
(2) Heavy water reactors (HWRs);
(3) Gas cooled reactors (GCRs and AGRs).

Currently, 88% of installed capacity comes from LWRs (56% PWRs, 22% BWRs and 9% WWERs), 5% from HWRs (mainly CANDUs), 3% from RBMKs, 2% from AGRs and 0.7% from GCRs. Less than 0.6% of capacity

comes from other designs in operation, essentially fast reactors (liquid metal cooled fast reactors (LMFRs)), operated as breeders or more recently as burners.

One of the prime considerations in the life of an NPP is the structural integrity of the reactor vessel. Initially, this concern resulted in countries adopting a licensing lifetime of ten years, which evolved to 40 years with reactor experience. During 40 years of operation, reactor vessels were and continue to be closely monitored and tested for embrittlement. Since it has been demonstrated that reactor vessels can withstand the most adverse operating conditions with an adequate level of safety, reactor operations are being extended to 60 years and perhaps beyond. Annex I provides a country by country summary of the licensed life and national policy on the use of nuclear energy. As would be expected with a maturing industry, most countries are gradually moving towards a 60 year lifetime. Currently, reactor vendors design and construct reactors for a nominal lifetime of 60 years rather than the previous 40 years.

In addition to operating beyond the initial 40 years, there is interest in producing more electricity from existing power plant. There are two principal means of achieving additional power.

The first is by better operation of the plant with fewer shutdowns, greater equipment reliability and shorter refuelling outages, which all result in a higher capacity factor. The best example of improving plant performance is the USA. In 1980, the capacity factor was only 55%; in 2003, it was 89.4%. This is equivalent to 19 new 1000 MW(e) power plants. Around the world, half of the reactors are operating at or below 80% capacity, while in some countries the average is close to 90%. Therefore, considerable additional production can be expected.

The second approach is to evaluate the equipment limitations by reviewing the design of the plant and each system, to identify those pieces of equipment that are the limiting factors for power production and to replace them with equipment that provides for additional production. For instance, Table 1 shows what has been submitted for regulatory approval in the USA over the 13 months to the end of 2004. It represents an increase of installed capacity of almost 1100 MW(e). In Belgium, five of the seven PWRs have been uprated by 8–12%, providing for a 10% increase of national nuclear capacity. While there is no defined value for the amount of additional capacity that can be been achieved through uprates, experts in this field believe that a 10% addition across the industry would be reasonable.

In summary, it can be concluded that current reactor designs will dominate operations up to 2050. New plants that will be introduced between now and 2050 will be primarily evolutionary designs of those, a few

TABLE 1. POWER UPRATE SUBMITTALS CURRENTLY UNDER NUCLEAR REGULATORY COMMISSION STAFF REVIEW

Plant	Thermal power		Submittal date	Completion date	Electric power (MW(e))[a]
	(%)	(MW(th))			
Vermont Yankee	20	319	10 Sep. 03	Jan. 05	110
Waterford 3	8	275	13 Nov. 03	Jan. 05	90
Seabrook	5.2	176	17 Mar. 04	Feb. 05	59
Indian Point	4.85	148.6	03 Jun. 04	?	47
Browns Ferry 2	15	494	24 Jun. 04	?	160
Browns Ferry 3	15	494	25 Jun. 04	?	160
Browns Ferry 1	20	659	28 Jun. 04	?	220
Palo Verde 1	2.94	114	09 Jul. 04	Jun. 05	38
Palo Verde 3	2.94	114	09 Jul. 04	Jun. 05	38
Beaver Valley 1	8	211	04 Oct. 04	?	69
Beaver Valley 2	8	211	04 Oct. 04	?	69

[a] Approximately.

Generation III plants and prototype Generation IV plants. The analyses in this report are based on this conclusion.

4. SPENT NUCLEAR FUEL STORAGE

4.1. STORAGE FACILITIES

The issues concerning spent nuclear fuel are key to discussions about nuclear energy and to the protection and security of the environment. At the beginning of 2002, more than 150 000 t HM of spent fuel were stored in various storage facilities. Most of this fuel is under water, but dry storage is becoming a widely used technology, with more than 12 000 t HM currently stored in this manner in various countries [8]. In the future, the requirement for spent fuel capacity will continue to increase and some fuel will have to be stored for 50 years or more, before reprocessing or final disposal can take place. Currently, spent fuel is mainly stored wet, either in at-reactor (AR) facilities or in away-from-reactor (AFR) facilities. The capacity and inventory of various

storage facilities in different regions of the world, as of the end of 2003, are listed in Table 2. The following information can be seen from Table 2: more than 70% of the capacity of the AR pool facilities in North and South America and Eastern Europe are filled with spent fuel. Those in Africa, Asia and Western Europe are relatively empty:

(a) The AFR wet facilities in Eastern Europe have almost no available space.
(b) The capacities of AFR dry facilities in Africa, Asia and Eastern Europe are small.
(c) The world capacity of all types of storage facility is 285 733 t HM, of which 62% is filled with spent fuel.

Mixed oxide spent fuel storage is listed in Table 3 [9]. Currently, the stored quantity of spent MOX fuel from LWRs is more than 700 t HM. This storage is taking place in countries both with and without a reprocessing or recycling programme. Some countries are following a policy of phasing out nuclear power and therefore only hold storage of spent MOX fuel from previous nuclear programmes. Such storage may also include storage of spent MOX fuel from a foreign country.

Such a deal had been negotiated by Sweden in the wake of a tripartite agreement with France and Germany. There is still some separated plutonium in foreign reprocessing plants, which is to be returned to the owner countries in the form of either MOX or plutonium oxide. Germany is following a policy of phasing out nuclear power, but is still using reprocessing services in France and the UK until the year 2005, and the recovered plutonium is being used as MOX fuel in German NPPs. The UK has in store some tonnage of foreign spent MOX fuel. France stores the largest amount of spent MOX fuel, which has been discharged from their MOX loaded LWRs. Switzerland still applies temporarily a recycling policy and uses MOX fuel. Belgium has stopped its policy of recycling, but the implementation of reprocessing contracts concluded in 1976 and 1978 has been continued until their expiry date. The recovered plutonium has been recycled as MOX fuel in two Belgian PWRs. Table 3 shows the main contributors to spent LWR MOX fuel storage accumulated to the present. Spent MOX fuel assemblies are commonly stored in AR and AFR pools.

TABLE 2. REGIONAL RECORDS OF SPENT FUEL STORAGE IN VARIOUS STORAGE FACILITIES
(as of the end of 2003 [8])

Status by region	Cumulative discharge (t HM)	AR pool		AFR (wet)		AFR (dry)		Reproc. amount (t HM)	Stored amount (t HM)
		Capacity (t HM)	Inventory (t HM)	Capacity (t HM)	Inventory (t HM)	Capacity (t HM)	Inventory (t HM)		
North and South America	85 756	98 131	72 267	3 950	777	22 918	12 482	230	85 526
Africa and Asia	35 126	35 037	23 782	5 510	1 924	1 778	1 579	7 841	27 285
East Europe	33 270	14 119	10 171	17 856	17 479	5 193	1 750	3 870	29 400
West Europe	111 055	29 994	13 346	40 279	20 905	10 968	1 653	75 151	35 904
Global total	265 207	177 281	119 566	67 595	41 085	40 857	17 464	87 092	178 115

TABLE 3. WORLD STATUS OF SPENT LWR MOX FUEL STORAGE IN THE YEARS 2001–2002 [9]
(in t HM)

Country	Storage started	Total amount of spent MOX fuel	Facility at reactor, AR	Facility away from reactor, AFR
Belgium	1998	26	26	0
France	1990	420	225	195
Germany	1970	204	120	84
Sweden	1986	24	<1	<24
Switzerland	1978	61	61	0
Total		735	<433	<303

4.2. LONG TERM PROSPECTS FOR SPENT FUEL STORAGE

4.2.1. Annual spent fuel discharge

The long term prospects were calculated with the VISTA code [10]. The annual amounts of SNF discharged from NPPs worldwide (arisings), with three different nuclear power scenarios and an assumption of a constant reprocessing ratio (30% of the total amount of spent fuel discharged from LWRs, designated as the case R1, Section 6.3) and of a medium MOX refuelling ratio (designated as the case M2, Section 8), are illustrated in Fig. 1.

In the high nuclear capacity case (designated as P2 in Fig. 1), where the nuclear power capacity increases from 353 GW(e) in 2000 to 730 GW(e) in 2050, the arisings increase steadily from 10 890 t HM in 2000 to 13 920 t HM in 2050. These arisings are not proportional to the growth in nuclear power. In the middle capacity case (designated as P1), where nuclear capacity grows to 565 GW(e) in 2050, there is a small peak of 11 520 t HM in 2015, then followed by a slight decrease to 10 960 t HM up to 2050 even though the nuclear capacity in this scenario increases up to 2050. For the arisings in the low capacity case (designated as P0), where capacity increases slightly to 400 GW(e), there is a similar change to that seen in the P1 scenario, with a small peak in 2020 (11 390 t HM), then declining to 8000 t HM in 2050. Such changes in arisings, which are inconsistent with changes in nuclear capacity, are mainly influenced by the increasing average burnup in the reactor mix. The anticipated shutdown of GCRs in the coming decade will reduce arisings of low burnup spent fuel from GCRs without a significant drop of world nuclear capacity.

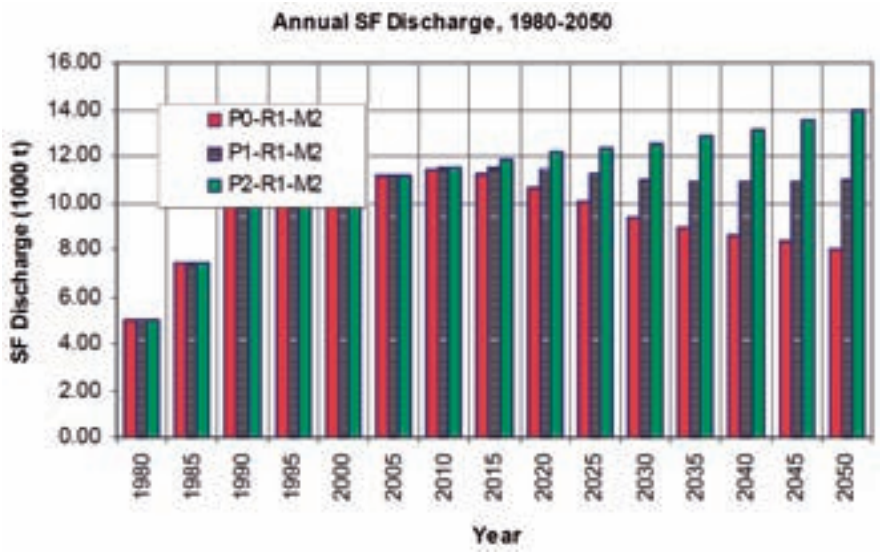

FIG. 1. Annual discharges of spent fuel.

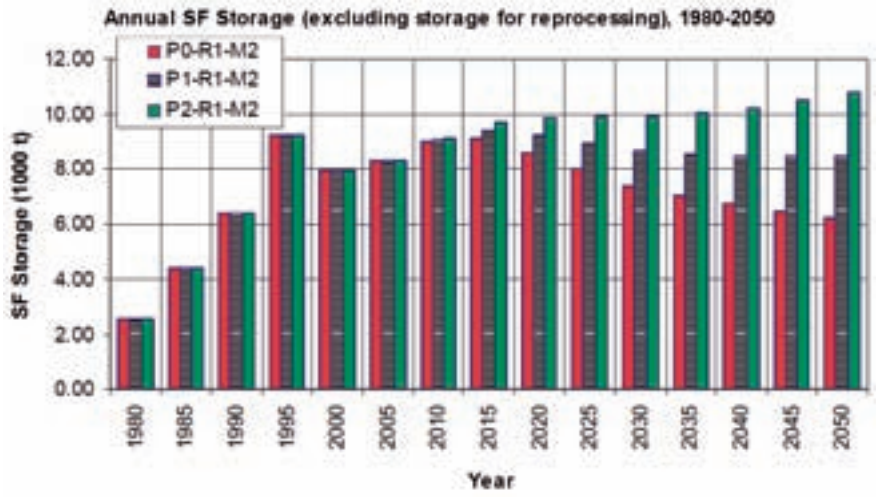

FIG. 2. Annual spent fuel storage.

4.2.2. Annual spent fuel storage

Annual changes of spent fuel storage calculated by VISTA exhibit a similar tendency to nuclear electricity growth if the 30% reprocessing ratio is assumed (Fig. 2).

There is a quantitative difference between the electric power capacity and the spent fuel storage. In the case of the high capacity scenario (P2), nuclear electricity production in 2050 is almost double that in 2000, whilst spent fuel storage only increases about 1.5 times. The medium capacity scenario (P1) presents a slight decrease of spent fuel storage after a peak in 2015 in spite of a steady increase of the electricity capacity. The low capacity scenario (P0) also indicates a similar trend. This difference is due to the increase of average burnup of spent fuel and to the ending of the discharge of low burnup, large volume GCR fuel due to the shutdown of GCRs within the coming decade.

4.2.3. Challenges

Whatever the scenario and the back end options might be, the storage technologies outlined in Section 3.1 are able to cope with the storage requirements. The designs and operation specifications can be adapted to the increasing discharge burnups in due time. The construction of storage pools or dry facilities and the fabrication of dry storage casks can meet the demand without delay. The only effect is on utilities that have opted for dry storage. They might have to increase their interim wet storage facility to cope with the longer cooling delay required for higher burnup SNF.

The accumulation of large amounts of stored SNF and the multiplication of AFR storage sites might become a public acceptance problem and a concern for proliferation resistance. The wait and see and open cycle, back end options are worse in this respect than the reprocessing option.

5. REPROCESSING AND RECYCLE OF PLUTONIUM

5.1. REPROCESSING FACILITIES

The reprocessing [11] currently conducted involves shearing, dissolution and recovery of uranium and plutonium by liquid–liquid extraction using the PUREX process. In the history of reprocessing, Oak Ridge National Laboratory developed the REDOX process to extract plutonium and uranium from liquid waste in the 1940s, which became the basis of the PUREX process. This process was adopted for the military reprocessing facilities (e.g. ICPP in Idaho, 1953; SRP in Savannah River, 1954; HPP in Hanford, 1956). Using their operational experience, civil facilities were deployed in the USA after the 1960s. The West Valley facility was the world's first civil reprocessing facility,

being put into operation in 1966. In Europe, Sellafield B204 in the UK (1952), replaced later by B205 (1964), and Marcoule UP-1 in France (1958) also started their operations for military purposes. At the same time, almost all the OECD member countries decided to build and operate together a multipurpose civil plant to demonstrate successful reprocessing of fuels differing widely in their fissile material content, chemical composition, burnup, shape, dimension and cladding material. This resulted in the Eurochemic plant, located in Belgium, which started operation in 1966. It became the basis of restricted purpose facilities, such as La Hague UP-2 (1966), Sellafield B205 (1964) and WAK in Germany (1971). In the past 15 years, additional industrial scale facilities, Sellafield THORP (1995) and UP-3 (1989), have been put into operation. New processes connected to PUREX, which can partition the MAs, are under development.

Dry processes are conceived for innovative nuclear fuel cycles. One of these is the pyrochemical process, using molten salt for reprocessing of nitride, metallic and oxide fuels. Another is DUPIC (Direct Use of spent PWR fuel In CANDU), which is aimed at reusing the fissile material contained in spent PWR fuel in CANDU reactors, after removing only volatile and semivolatile products from the spent fuel. Details of this process are given in Ref. [12].

There is considerable experience in the civil reprocessing of irradiated fuel on an industrial scale in several countries. Table 4 gives the civil reprocessing capacities for irradiated fuels currently available and foreseen, and Table 5 gives the cumulative amounts of civil spent fuel reprocessed. At the present time, the nominal total reprocessing capacity available is about 5000 t HM/a. Activities range from the small scale reprocessing of fuel from research or experimental reactors to large scale industrial plants offering an international service for standard oxide fuel from LWRs, WWERs, AGRs and GCRs. The total available reprocessing capacity by 2020 is foreseen to increase, with the introduction of new reprocessing plants currently under construction or planned in China, Japan and the Russian Federation. France is successfully operating reprocessing plants for oxide fuels with a total capacity of 1600 t HM/a and has already reprocessed about 18 000 t HM of LWR spent fuel (Table 5), while the UK's THORP plant, now fully operational, has a potential capacity of up to 1200 t HM/a and has reprocessed about 3800 t HM of AGR and LWR fuel. The Russian RT-1 plant has a capacity of 400 t HM/a and, as of the end of March 2002, some 3500 t HM of WWER fuel has been reprocessed. Reprocessing experience in India and Japan is equally relevant, although the installed plant capacity is not so large. The Japanese Tokai reprocessing plant has reprocessed about 1000 t HM of oxide fuel. Commissioning of the large scale reprocessing plant in Rokkasho is planned for 2006.

TABLE 4. PAST, CURRENT AND PLANNED REPROCESSING CAPACITIES (t HM/a) IN THE WORLD [11]

Country	Site	Plant	Fuel type	Operation		Capacity	
				Start	Shutdown	Present	Future
Belgium	MOL	Eurochemic	LWR	1966	1975		
China	Jiuquan	RPP	LWR	?			25
	Lanzhou		LWR	2020			800
France	Marcoule	APM	FBR	1988	1996		
	Marcoule	UP1	GCR	1958	1997		
	La Hague	UP2	LWR	1967		800	800
	La Hague	UP3	LWR	1990		800	800
Germany	Karlsruhe	WAK	LWR	1971	1990		
India	Trombay	PP	Research	1964		60	60
	Tarapur	PREFRE 1	PHWR	1974		100	100
	Kalpakkam	PREFRE 2	PHWR	1998		100	100
	Kalpakkam	PREFRE 3A	PHWR	2005			150
	Tarapur	PREFRE 3B	PHWR	2005			150
Japan	Tokai-mura	PNC TRP	LWR	1977		90	90
	Rokkasho	RRP	LWR	2005			800
Russian Federation	Chelyabinsk	RT1	WWER -440	1978		400	400
	Krasnoyarsk	RT2	WWER -1000	2020[a]			[b]
UK	Sellafield	B205	GCR	1967	2012	1500	
	Sellafield	Thorp	LWR/ AGR	1994		900	900
	Dounreay	UKAEA RP	FBR	1980	2001		
USA	West Valley	NFS	LWR	1966	1972		
	Hanford	Rockwell	U metal	1956	1989		
	Savannah River	SR	U metal	1954	1989		
	Idaho Falls	R	U–Al alloy	1959	1992		
Total capacity						4860	6845

[a] Not earlier than 2020.
[b] Yet to be determined.

TABLE 5. CUMULATIVE AMOUNT OF CIVIL REPROCESSED SPENT FUEL [11]
(t HM as of end of 2003)

Country	Site	Plant	Fuel type				
			GCR	LWR	FBR	MOX	Total
Belgium	Mol	Eurochemic[a]	19[b]	86			105
France	Marcoule	UP1	18 000[c]				18 000
	La Hague	UP2/UP3		19 000	10	9.6	19 020
Germany	Karlsruhe	WAK[a]		180			180
India	Trombay	PP					
	Tarapur	PREFRE-1					
Japan	Tokai-mura	TRP		1 000 +18[d]			1 018
Russian Federation	Chelyabinsk	RT-1		3 500			3 500
UK	Sellafield	B205	42 000[e]				42 000
	Sellafield	Thorp		5 800[f]			5 800
	Dounreay	UKAEA RP			14		14
USA	West Valley	NFS[a]		194			194
Total			60 019	29 780	24	10	89 830

[a] Closed facility.
[b] CANDU, GCR and others.
[c] UNGG.
[d] Spent fuel from Fugen.
[e] Magnox.
[f] LWR/AGR.

5.2. LONG TERM PROSPECTS OF REPROCESSING AND MOX UTILIZATION

5.2.1. Annual amounts of spent fuel reprocessing

The annual amount of spent fuel reprocessed, which has been calculated for the three scenarios, is presented in Fig. 3. Since spent GCR fuel (Magnox) cannot be stored for a long time due to corrosion in the Magnox cladding and therefore requires prompt reprocessing after discharge, shutdown of the Magnox GCRs will cause a significant fall of the reprocessing amounts from 2010.

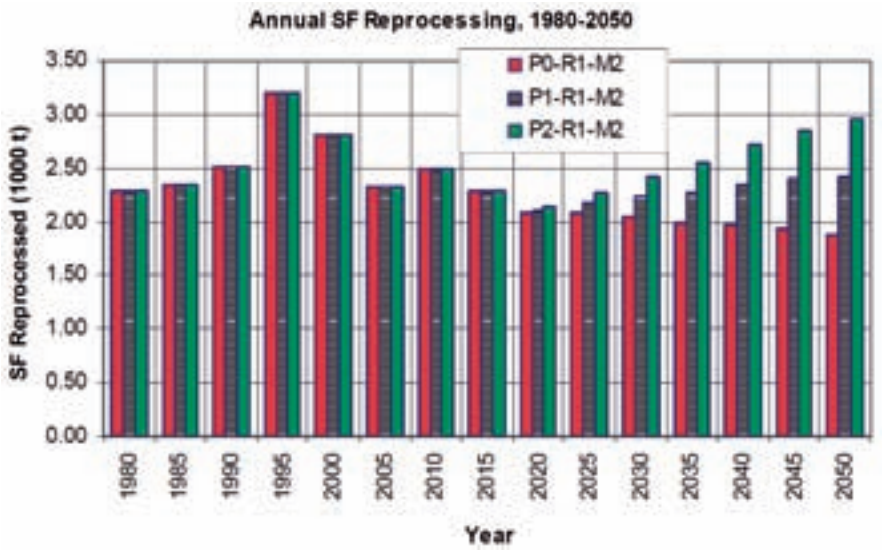

FIG. 3. Annual amount of spent fuel reprocessed.

5.2.2. MOX fuel requirements

The MOX fuel requirements for the different nuclear capacity scenarios calculated by VISTA (Fig. 4) indicate a similar trend to the growth of nuclear power capacity, but are quantitatively different. Annual MOX fuel production (Table 6) in the period 1998–2000 was actually 180 t HM, while 160 t HM was predicted in 2000 by the VISTA calculations. The calculated MOX requirement in 2000 (160 t HM) and the actual MOX fuel production (180 t HM/a) are close to the nominal MOX fuel production capacity, roughly 200 t HM (Table 7). When SMP, Sellafield, UK, is put into full operation and the MOX fuel fabrication facility (FFF) in Rokkasho, Japan, begins operating, the situation of supply and demand in the MOX fuel market will change dramatically.

5.2.3. MOX fuel fabrication

The IAEA has collected information on MOX fuel production, including capacity and future planning, through various IAEA meetings [13–15]. The present status of MOX fuel production worldwide (Table 6) was reviewed by Bairiot et al. [16, 17]. Current and projected plants for MOX fuel production are listed in Table 7. In the UK, a large scale MOX production plant, SMP, has been constructed and delivered its first fuel in 2005. The Russian small scale

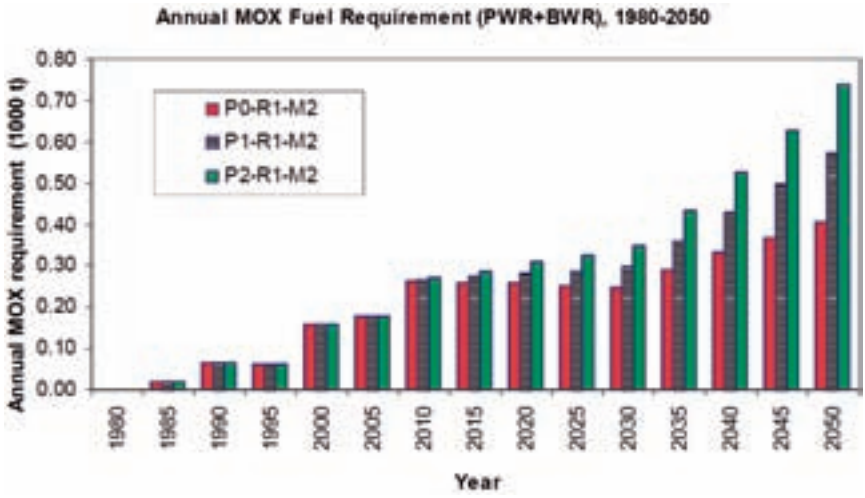

FIG. 4. Annual MOX fuel requirements (PWR + BWR).

TABLE 6. FUEL FABRICATION RECORDS FOR LWR MOX UP TO THE END OF 2000 [16] AND UP TO THE END OF 2004 [17]

Country	Facility	2000				2004
		t HM	t Pu[a]	FRs	FAs	t HM
Belgium	Belgonucléaire	467	26	246 000	1420[b]	610
Germany	Siemens/Hanau	158	6.4	77 000	380	158
France	CFCa	248	16	126 000	480[b]	343
France	MELOX	455	24	254 000	1050[c]	875
UK	MDF[d]	14[e]	1[e]	7 300[e]	36[e]	14
India	BARC	3	0.1	800	23	3
Japan	PFFF (ATR fuel)	120	1.9	22 500	750	120
Total (rounded figures)		1500	80	730 000	4100	2100

[a] Contained in the delivered fuel.
[b] Mainly manufactured at FBFC.
[c] Includes 92 FAs incorporating pellets and FRs fabricated at CFCa.
[d] The philosophy was to make different fuel designs to support SMP.
[e] Out of which, respectively 3.9 t HM, 0.3 t Pu, 2112 FRs and 8 FAs were later not accepted by the customer.

TABLE 7. MOX FUEL FABRICATION CAPACITY [16, 18]
(in t HM/a)

Country	Site	Plant	1998	2000	2005	2010
Belgium	Dessel	Belgonucléaire	35	40	40	(40)
France	Cadarache	CFCa	35	40	0	0
	Marcoule	MELOX	100	100	145	(180)
India	Tarapur	AFFF	5	10	10	10
Japan	Tokai	PFPF + PFFF	15[a]	15[a]	5[b]	5[b]
	Rokkasho	MOX FFF				(100)[c]
Russian Federation	Dimitrovgrad	NIIAR			(1)[d]	1[d]
	Seversk	DEMOX (RMFFF)				(60)[e]
UK	Sellafield	MDF	8	0	0	0
	Sellafield	SMP		0	(2)	(40)
USA	Savannah River	MFFF				(60)[e]
Total			198	205	200[f]	310[f]

[a] For ATR Fugen and FBR Monju.
[b] For Monju FBR.
[c] Date is not fixed.
[d] For the BN-600 FBR.
[e] For WPu.
[f] Evaluating impact of the uncertainties.

facilities have produced MOX fuel for FBR demonstration programmes since the 1970s, and the NIIAR MOX research complex is being upgraded into a pilot plant with a capacity of about 1 t HM/a. Further developments are considered primarily in connection with the construction of BN-800 at Beloyarsk and with the excess weapons grade plutonium (WPu) utilization project.

5.3. BALANCE OF DEMAND AND SUPPLY [17]

The reprocessing and MOX fuel fabrication technologies outlined in Sections 5.1 and 5.2.3 have exhibited maturity through decades of industrial operation. They are providing solid foundations to the timely support of technological evolutions and their industrial applications.

Reprocessing is adapting and will have to be adapted further for treating spent fuel of increasing burnup. While MOX fuel is now reprocessed in dilution

with uranium fuel, higher proportions of MOX fuel will have to be reprocessed when FBRs start to be deployed for achieving sustainability of nuclear energy. Increasing minor actinide (MA) inventories will justify separating some of those actinides, mainly americium, for transmutation into shorter life fission products, and the reprocessing facilities will have to be adapted to incorporate the MA separation technologies currently being developed and tested [19]. However, the real challenge will be to deploy the reprocessing capacity at a rate meeting the demand. The past has shown that it takes 10–20 years from the decision to build to operation at rated capacity if the plant employs existing technology. It takes over 20 years if advanced technology has to be implemented or industrial experience is inadequate. The prerequisites for capacity to meet demand in the future are forward planning the installation of additional capacity and maintaining experience by current industrial reprocessing. Both prerequisites are not simple to meet in an environment driven by short term economic considerations.

The designs and operation specifications of MOX fabrication plants can be adapted in due time to the higher plutonium contents of MOX fuel resulting from increasing discharge burnups. The real problem will be expansion of the plant capacities and construction of new plants to meet the demand without a delay. It takes at least ten years between a decision to invest and plant operation at rated capacity. Since customers are not in a position to underwrite long term commitments, the industry will be reluctant to invest and the result might well be a shortage of fabrication capacity, as was the case in the 1980s and 1990s. A prerequisite here is also to pursue industrial operation at a significant level, in order to maintain and adapt the technology in due time.

6. FINAL DISPOSAL

6.1. FINAL DISPOSAL FACILITIES

Final disposal facilities for spent nuclear fuel and high level waste (HLW) are the subject of various investigations and projects around the world, but no such facility for spent fuel is yet being operated. Deep geological repositories are the expected sites for final disposal. Table 8, taken from Ref. [20], summarizes the status of the siting of some deep repository facilities together with AFR storage facilities. The plans in Finland and the USA are particularly noteworthy. In May 2001, Finland became the first country to approve plans for a geological repository. The Finnish waste disposal company Posiva Oy will

research possible sites and intends to start building the repository in 2010 [21]. The application for an operating licence will be submitted to the regulatory authorities around 2020. In the USA, the Yucca Mountain Resolution was approved by Congress and signed by the President in July 2002, which enabled further steps to be taken in the Yucca Mountain project. The licence application is scheduled for submission to the Nuclear Regulatory Commission (NRC) in 2006, and receipt of spent nuclear fuel is scheduled to start around 2012 [22]. The Nuclear Waste Policy Act (NWPA) limits the amount of spent nuclear fuel and high level radioactive waste that can be emplaced in the first US geological repository to 70 000 t HM, after which a second repository needs to be in operation. The radioactive materials to be disposed of at Yucca Mountain include about 63 000 t HM of commercial spent nuclear fuel. Liquid waste forms would not be accepted for disposal [23]. In Belgium, a deep underground laboratory, HADES, has been operating since 1980 to quantify the final disposal characteristics of a thick clay layer, the only host formation available in the country. The facility has been expanded since 1999 and serves as a pilot and reference for various other countries interested in clay formations for their future repositories.

6.2. LONG TERM PROSPECTS OF SPENT FUEL DISPOSAL

The amount of spent fuel that will be potentially disposed of in repositories are the cumulative quantities of spent fuel discharged from the reactors, minus the quantities already reprocessed and those destined for reprocessing. Figure 5 shows the result of calculations carried out for the three nuclear power scenarios described in Section 4. It should be noted, however, that these estimates express the maximum potential amount for disposal for a given scenario, since the quantity of spent fuel stored at a given time includes the fuel stored for an extended period of time in a wait and see perspective, part of which will ultimately be reprocessed. This maximum quantity of fuel to be conceivably disposed of in a final repository is 176 000 t HM worldwide in 2005, which would require the availability of three repositories such as Yucca Mountain. By 2050, the amount of spent fuel potentially to be disposed of will increase to between 523 000 and 621 000 t HM, depending on the nuclear power growth scenario, implying that eight or nine repositories with a capacity equivalent to Yucca Mountain will be required for disposal of the total amount of spent fuel worldwide.

TABLE 8. STATUS OF FACILITIES FOR AFR STORAGE AND REPOSITORIES AS OF JANUARY 2003

Country	AFR SNF/HLW storage	Deep repository siting status
Belgium	SNF is stored at the reactor sites (AR).	A wide deep clay formation, tested since 1974, will host the repository at a yet undefined location. An underground laboratory collects the database for safety evaluation and licensing.
Canada	AFR SNF/HLW storage is not now planned.	Despite extensive generic investigations in the past, there are currently no plans for disposal of SNF. All SNF is held in interim storage pending a Government decision on what long term management method to implement.
China	Construction of an AFR storage facility started in 1994. The initial stage will have a capacity of 550 t.	In 1985, China initiated a four step programme for deep geological disposal of HLW. The goal of the second phase (1996–2010) is to select one area for further investigation. Chinese authorities have already identified one potential siting region at Beishan. The goal is to have an operational repository by 2040.
Finland	AFR SNF/HLW storage is not planned.	In 1999, Posiva Oy proposed to site the disposal facility for SNF at Olkiluoto in Eurajoki. The application was approved by the municipality of Eurajoki in Jan. 2000. Then, the Finnish Government made a decision-in-principle in Dec. 2000 and the Parliament endorsed it in May 2001. It is scheduled that the licence application will be submitted by the end of 2010, and the operating licence will be applied for around the year 2020.
France	SNF is in principle stored at the reactor sites (AR) until reprocessing.	In 1987, ANDRA initiated activities to site a geological repository; however, this was not successful due to substantial protests from the public. A new law was adopted keeping options open, and volunteer sites were sought for a deep underground laboratory that could eventually lead to a repository. One such laboratory in clay at Bure is now in operation, but siting of a second laboratory was unsuccessful.

TABLE 8. STATUS OF FACILITIES FOR AFR STORAGE AND REPOSITORIES AS OF JANUARY 2003 (cont.)

Country	AFR SNF/HLW storage	Deep repository siting status
Germany	There are two designated operational AFR storage facilities.	Preparation for two repositories, one for heat generating waste (Gorleben) and one for non-heat-generating waste (Konrad), progressed far towards completion. The Konrad facility has been licensed, although it is not yet commissioned to start operation. Although criteria for the new siting process have been worked out, the new siting process itself has not started.
Japan	Although storage capacity at some reactor sites is becoming scarce, there are as yet no designated AFR storage facilities in Japan.	Government policy specifies that HLW arising from reprocessing shall be disposed of by geological disposal. Vitrified HLW shall be emplaced in a stable geological formation at a depth of more than 300 m, following 30–50 years of interim storage for the cooling process.
Republic of Korea	Korea is aiming to construct an AFR interim storage facility with 2000 t at the first stage by 2016.	An R&D programme on the deep geological repository for SNF/HLW was launched in 1997 to establish a reference repository system and assess the feasibility of a deep geological repository.
Russian Federation	There is an AFR (wet) storage facility for SNF from WWER-1000 at the Mining and Chemical Combine (K-26). The storage capacity is 9000 t. In general, however, SNF is stored at the plant sites (AR) until sent to Mayak for reprocessing.	The Russian policy is to dispose of HLW in deep geological repositories. At present, four facilities in geological formations are considered for storage and disposal of solid radioactive waste or SNF. According to the current plans, geological disposal will not begin until 2025–2030.

TABLE 8. STATUS OF FACILITIES FOR AFR STORAGE AND REPOSITORIES AS OF JANUARY 2003 (cont.)

Country	AFR SNF/HLW storage	Deep repository siting status
Sweden	Sweden has an operating AFR (wet) storage facility for SNF. The storage facility is currently being expanded to be able to store all SNF from the Swedish nuclear power programme.	Sweden plans to build one deep geological repository for the disposal of SNF. Feasibility studies for siting of the deep repository were carried out in eight municipalities; three were proposed as candidate sites.
Switzerland	Switzerland has an AFR interim (dry) storage facility for SNF and vitrified HLW, which has been operating since 2001.	The Swiss plans for a national repository call for issuing the required licences to allow the repository to commence operation between 2040 and 2050. The ongoing siting work is based on a three phase strategy: Phase I for regional studies by borehole data and surface measurements, Phase II for more extensive investigations, and Phase III for deep underground exploration and full characterization of a candidate site. The preferred siting region has been suggested by Nagra.
UK	There are no designated AFR storage facilities in the UK.	The UK abandoned its HLW siting programme due to public opposition. After this programme failed, the UK resolved to reassess all options for long term waste management.
USA	The USA tried to site centralized monitored retrievable storage (MRS) facilities for commercial SNF through a volunteering process, but no volunteers came forward. One AFR is operational and a second one is in the licensing process. The existing AFR is not accepting additional SNF, while the AFR in the licensing process is expected to be operational by 2010.	A formal siting process for a repository was developed under the USDOE siting guidelines, as required by the NWPA. The Amended NWPA selected Yucca Mountain as the only site to be characterized. A site characterization plan was developed for the Yucca Mountain site in 1988, and a very extensive characterization programme has been carried out. On 23 Jul. 2002, after the Congressional vote of approval, the President signed the Yucca Mountain Resolution, approving the Yucca Mountain site. A licence application is now being prepared.

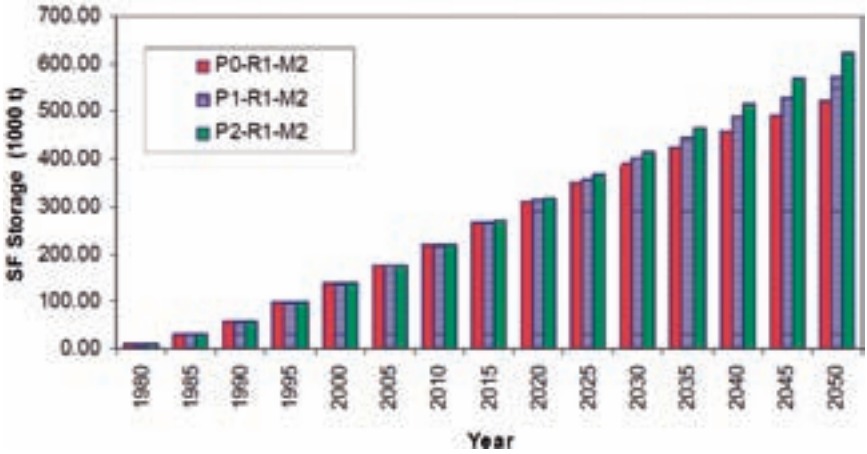

FIG. 5. Cumulative quantities of spent fuel not destined for reprocessing.

6.3. CHALLENGE FOR SPENT FUEL DISPOSAL IN THE LONG TERM

Expanding the quantities of spent fuel to be disposed in geological repositories might be a problem in the long term, as selection and opening of repository sites is meeting with legal and public acceptance problems. So, for the long term, the challenge is how to minimize the quantity of spent fuel for disposal in geological repositories.

In the USA, 44 000 t HM of spent fuel are currently stored and 2000 t HM of spent fuel are added to storage every year. At this rate, the statutory limit for the planned geological repository, 63 000 t HM of spent fuel, will be reached by 2015 (Fig. 6) and a second repository will be required, which in turn would be full by around 2050 if the current open fuel cycle option is pursued. The USDOE estimated how many geological repositories equivalent to Yucca Mountain are required with various fuel cycle scenarios up to 2100 [22]. Taking the maximum case from this study, 20 facilities of Yucca Mountain equivalent repositories will be required using the open cycle scenario if nuclear electricity generation expands as expected. If recycle of nuclear materials is undertaken, the number of repositories required can be dramatically reduced. In the same case of growing nuclear electricity generation, if fast reactors are deployed, only one repository equivalent to Yucca Mountain is required to accommodate spent fuel on this timescale. Reduction of the number of required repositories,

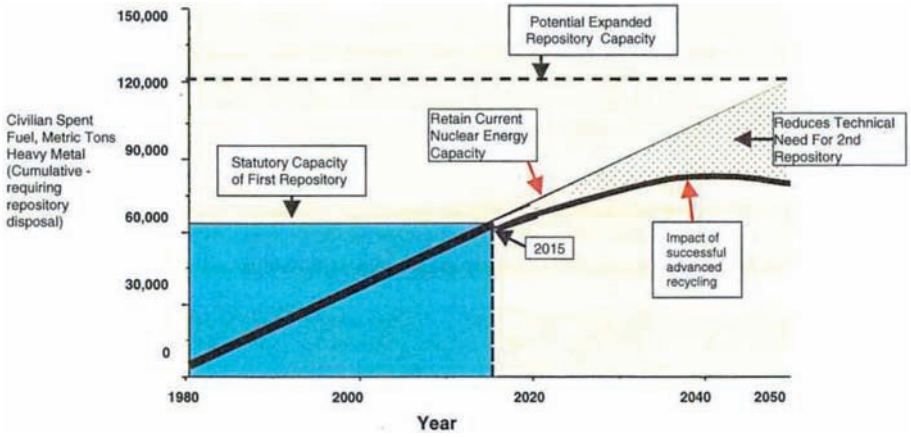

FIG. 6. Spent fuel for disposal and the capacity of the Yucca Mountain repository [6].

by reducing the quantity of spent fuel, has become a strong incentive for setting up advanced fuel cycles in the USA [6]. The USA recycle concept is designated as the Advanced Fuel Cycle Initiative (AFCI), which consists of two series. Series One aims at reducing the liability of spent fuel disposal by reducing the volume of HLW and reducing the long term proliferation threat of plutonium contained in spent fuel. Series Two is to reduce the liability of spent fuel disposal by reducing the long term radiotoxicity and heat load by supporting the development of advanced, next generation, nuclear fuel cycles. These aspects are dealt with in Ref. [24].

Another example of meeting the challenge of spent fuel disposal is provided by France, which has implemented a recycle strategy of using plutonium as MOX fuel in PWRs. The total capacity of spent fuel pool storage facilities at NPPs and La Hague, about 19 000 t HM, is adequate to accommodate spent nuclear fuel discharged from NPPs for several decades (Fig. 7), since the quantity of spent nuclear fuel stored in these facilities does not increase substantially due to reprocessing. However, if reprocessing were stopped in 2004, it is anticipated that the storage facilities would reach their storage limit by 2012. If such a situation occurred, France would have to expand its capacity for spent nuclear fuel storage.

With respect to spent fuel and HLW treatment, a French law passed in December 1991 assesses the long lived nuclear HLW along three major lines [25, 26]:

(1) Feasibility of partitioning and transmutation (P&T) of long lived radioactive elements;

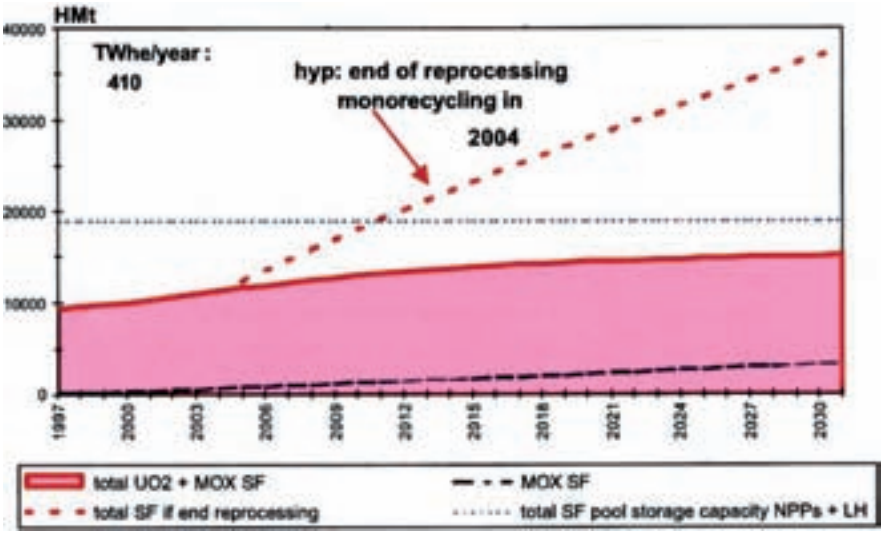

FIG. 7. Spent nuclear fuel inventory scenario in France [25].

(2) Deep geological disposal for HLW, with an underground laboratory to be implemented at Bure in a clay geological formation and studies on final disposal of HLW, such as in glass canisters, and direct disposal of spent fuel.
(3) Long term interim storage of HLW glass canisters or spent fuel.

On the basis of these directives, the Commissariat à l'Énergie Atomique (CEA), taking account of natural resource limits and long term waste management, presented scenarios and solutions for the near term, mid-term and long term fuel cycle [25]. The full management of transuranic elements strongly reduces the repository space and radiotoxicity requirements. Accordingly, the French long term solution to minimize radiotoxicity of final disposal is to keep recycling, to increase reprocessing efficiency and to incorporate transmutation of MAs in an advanced fuel cycle with FBRs acting as MA burners.

Calculations with the VISTA code predict that reprocessing limited to part of the LWR SNF may not significantly reduce the quantities of spent fuel to be disposed. Figure 8 presents the influence of reprocessing LWR and WWER fuels on the accumulation of spent fuel stored, excluding the fuel in interim storage (taken as six years) before reprocessing. The reprocessing scenarios are described in detail in paper 2.5 and are illustrated in Fig. 9. In broad terms, for the situation after 2030, R0 assumes no further reprocessing of

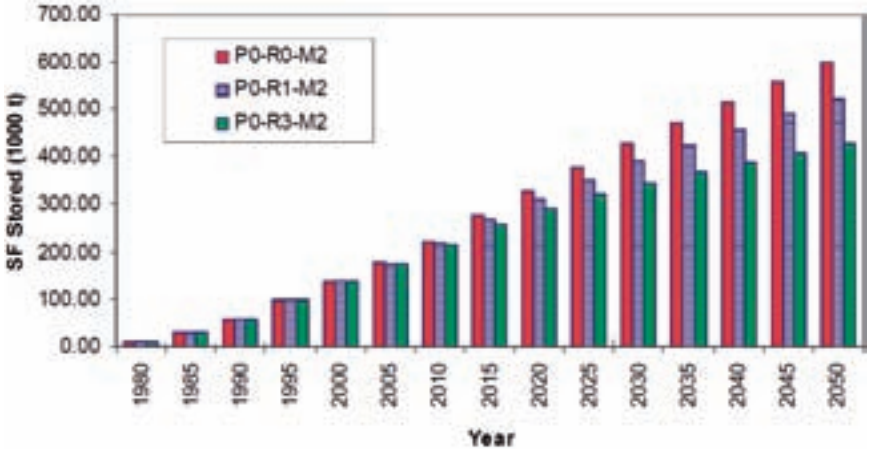

FIG. 8. Cumulative spent fuel storage excluding storage for reprocessing in different reprocessing scenarios.

spent fuel, R1 assumes 30% LWR (PWRs and BWRs) and 25% WWER reprocessing ratios, which is the current situation, and R3 models assume 70% LWR and 60% WWER reprocessing ratios. The nuclear power scenario used in Fig. 8 is the lowest growth in generation capacity.

The difference between the spent fuel quantities from scenarios R0 and R3 is 172 000 t HM in 2050. This relatively small difference results from the non-reprocessing of lower burnup fuels (CANDU, RBMK and AGR). However, the final disposal of these fuels is less challenging, since it requires relatively smaller repository volumes compared with the repository volumes for PWR and BWR fuels. It should be noted that the spent fuel inventory contains more MOX fuel with increasing reprocessing ratios, which calls for more sophisticated approaches for disposition of plutonium and MAs [27]. In such a case of mono-recycle into MOX fuel with final disposal being the destiny of spent MOX fuel, the final volume of conditioned waste is only reduced by a factor of 2 (Section 8), affecting exclusively the percentage of SNF reprocessed.

If spent uranium fuel were reprocessed to feed the separated plutonium into LMFRs and not LWRs, the reduction factor would be 4 [28, 29]. By separating and destroying MAs, the factor would rise to 5, which is the ultimate in reducing the final disposal liability.

If spent MOX is reprocessed in dilution with uranium fuel and the separated plutonium mix recycled in LMFRs, the required repository capacities for HLW are increased by 8% as compared with separating the

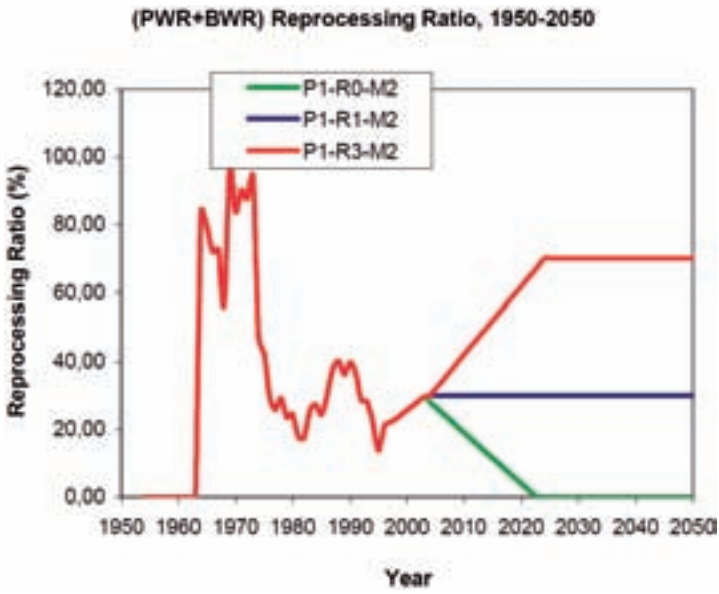

FIG. 9. The three reprocessing scenarios.

plutonium from spent uranium fuel for feeding the LMFRs [2]. Consequently, recycling plutonium in LWRs anyway reduces the final disposal liability to a large extent, whether LMFRs are later deployed or not.

In conclusion, plutonium recycling has a major impact on the final disposal problem.

7. FUEL CHARACTERISTICS AND EVOLUTION SCENARIOS

7.1. EXTENSION OF DISCHARGE BURNUP

Burnup affects the fuel cycle back end whether the SNF is recycled or sent to final disposal. In the case of recycle, the plutonium isotopic composition and the amount of fission products and MAs in the SNF will influence utilization of the fissile material [27]. In the open cycle, heat generation and radiation will affect the final disposal constraints. Fast breeder reactor fuels will not be deployed massively up to 2050 (Section 3 and Annex I), and commercial

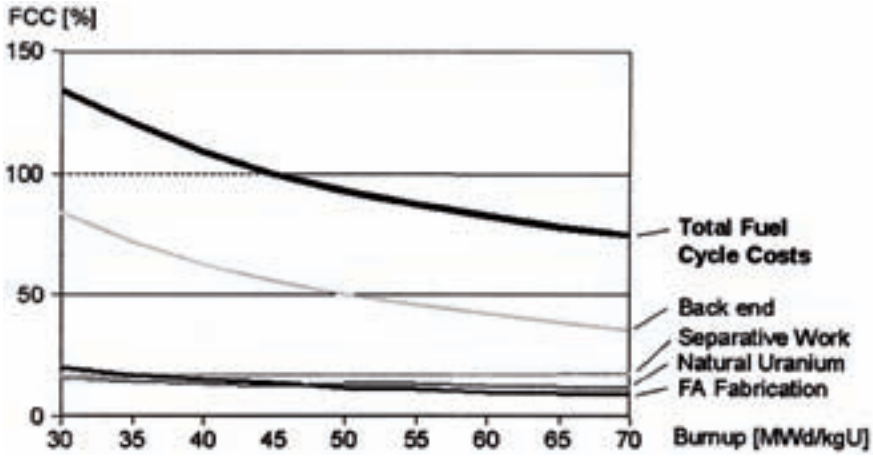

FIG. 10. Effect of discharge burnup on individual fuel cost cycle components.

utilization of thorium fuels is planned only in India. Utilization of those fuels will have a negligible effect on the global fissile material balance up to 2050.

Discharge burnups were extended significantly in the past as a result of development efforts in fuel assembly design and in-core management, which are still being pursued quite intensively. When discharge burnup is increased, fuel enrichment must also be increased (Annex II) and reload batch sizes become smaller. As a result, the amount of SNF decreases. As back end cost is related to the amount of SNF, burnup extension results in high economic benefits. Figure 10 shows fuel cycle cost as a function of discharge burnup in the range 30–70 GW·d/kg U, with 100% corresponding to the burnup reached at present on average by Framatome ANP PWR fuel assemblies [30]. In 2000, most European PWRs were using 3.7–4.2% enrichment with a 12–18 month cycle length, while the European PWRs other than those of Électricité de France (EDF) operate at the limit of their respective authorized enrichments (4.5–4.7%) and 40% of US PWRs operate on 20–24 month cycles using assemblies enriched between 4.6 and 4.95%. The historical rise in average discharge burnup over the period 1980–2000 [31] is illustrated by world region for PWRs in Fig. 11 and for BWRs in Fig. 12.

At present, practically worldwide, a licensed enrichment limit of 5% is applicable to nuclear fuel cycle facilities. This imposes some limit to burnup extension, typically an average discharge burnup of approximately 67 GW·d/t U in an annual PWR cycle [32].

For Framatome-ANP, which is providing 40% of the LWR fuel throughout the Americas, Asia, Europe and South Africa, more than 830 PWR

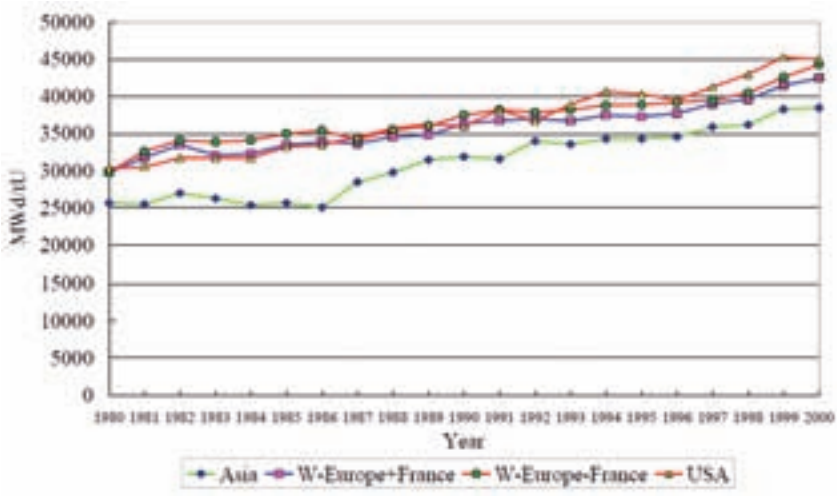

FIG. 11. Regional discharge burnup comparison for PWRs over the period 1980–2000.

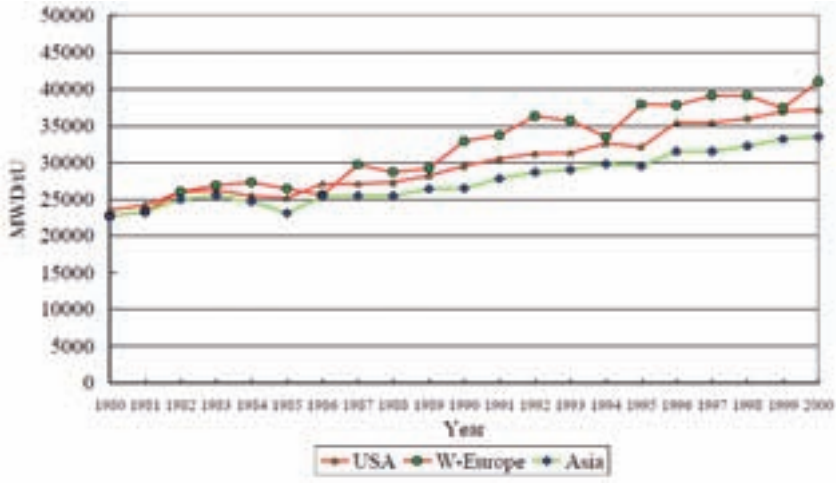

FIG. 12. Regional discharge burnup comparison for BWRs over the period 1980–2000.

fuel assemblies have reached burnups between 50 and 65 GW·d/t U. For BWRs, the average burnup has been increased from 25 GW·d/t U in the mid-1980s to more than 40 GW·d/t U at the end of 2000. In France, the maximum discharge burnup of uranium fuel reached close to the authorized limit of 52 GW·d/t U, namely 51 GW·d/t U for the 900 MW reactor series and 49 GW·d/t U for the 1300 MW reactor series, with an average burnup of about 45 GW·d/t U [33]. In

Japan, burnup of BWR fuels was increased in three steps. The first step was from 29 to 34 GW·d/t, the second from 34 to 39.5 GW·d/t and the third from 39.5 to 45 GW·d/t.

Figures 13 and 14 show projected burnups up to 2015 for PWR and BWR fuels, respectively [34].

The VISTA code [35] was used to calculate the amounts of fissile material for each reactor type with average discharge burnups (Fig. 15) and average enrichments (Fig. 16), prepared after consultation with experts and utilization of in-house knowledge and outside resources.

The cumulative amounts of uranium and plutonium in the spent fuels for each reactor type are shown in Figs 17 and 18, respectively. The contribution ratio of each reactor type to annual generation of the uranium and plutonium in 2003 and 2050 is shown in the adjacent pie diagrams. The total cumulative amounts of uranium and plutonium contained in SNF were, respectively, about 216 000 and 1400 t HM in 2003 and will reach about 743 000 and 6900 t HM in

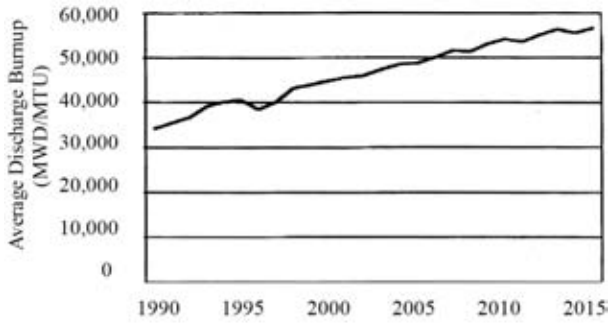

FIG. 13. Historical and projected discharge burnup trend for PWRs [34].

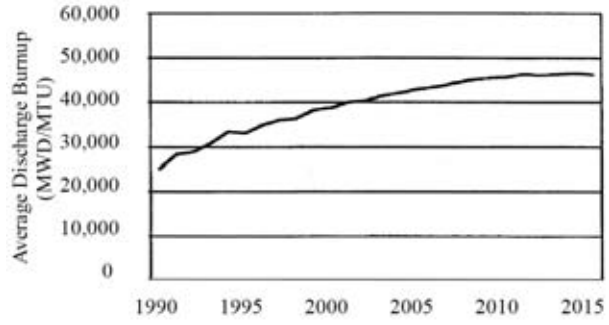

FIG. 14. Historical and projected discharge burnup trend for BWRs [34].

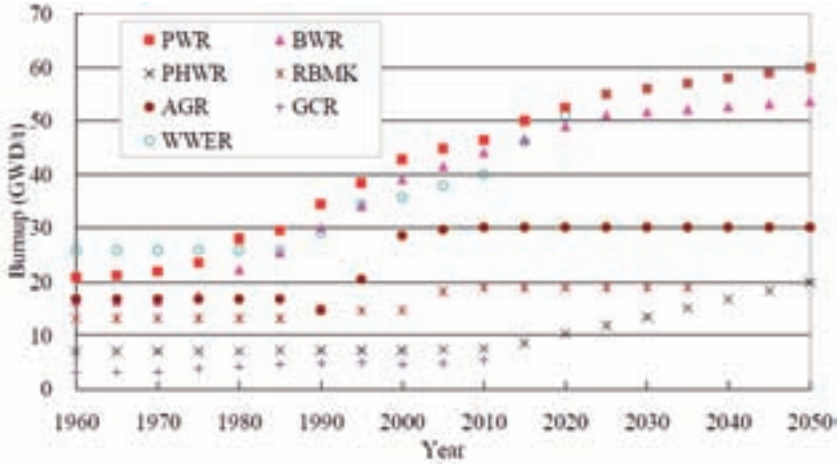

FIG. 15. Projected average discharge burnups.

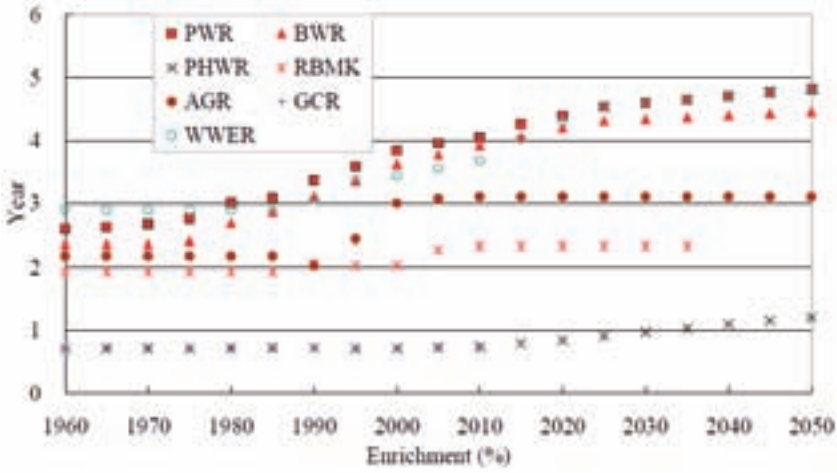

FIG. 16. Projected average initial enrichments.

2050. The changes in contribution of each reactor type to annual generation of the uranium and plutonium from 2003 to 2050 reflect the assumptions made for projection of the installed nuclear capacities (Section 3).

The evolutions of the uranium and plutonium fissile ratios in the spent fuels are shown in Figs 19 and 20, respectively. The residual ^{235}U content in the BWR and PWR SNF was about 1% in the 1970s, then decreases by 2010 to about 0.7%, which is the same as natural uranium but with a lower fissile value due to the presence of ^{236}U [36], and will go down to 0.55% by 2050. The

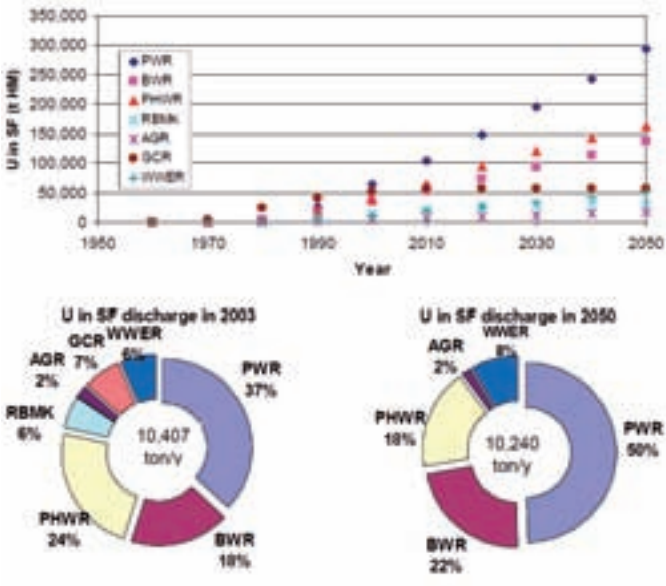

FIG. 17. Cumulative discharged uranium in SNF.

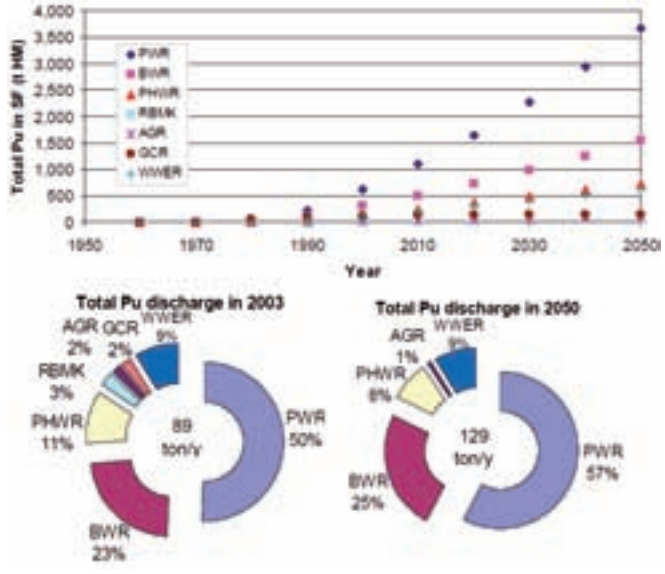

FIG. 18. Cumulative total plutonium discharged in SNF.

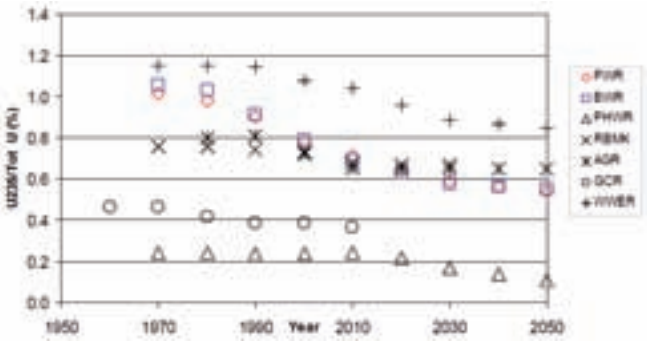

FIG. 19. Residual ^{235}U by reactor type.

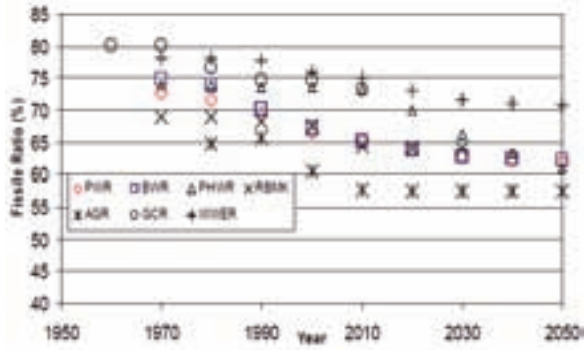

FIG. 20. Fissile plutonium ratio in SNF.

plutonium fissile content will decrease from about 75% in the 1980s to 60% in 2050, with consequences outlined in Section 9 and in Ref. [27].

7.2. UTILIZATION OF MOX FUELS IN LWRS

Fabricating MOX assemblies for LWRs has already been established on a commercial basis for over two decades (Section 5.2.3). In Europe, 37 PWRs and two BWRs currently operate with a partial MOX loading (Table 9), and some additional reactors are licensed to do so when the need arises. Smaller size applications were performed before this period in the USA, Sweden, the Netherlands and Italy.

Usage of MOX fuel in BWRs will probably increase in the future, when the Japanese programme will have the possibility to start. There is at present no

TABLE 9. RELOADS WITH MOX IN COMMERCIAL LWRs FROM 1981 TO THE END OF 2000 [16]

Country, reactor type (FA type)	Number of reactors licensed for MOX	Number of MOX fuel assemblies reloaded	Maximum FA average $Pu_{tot}(Pu_{fiss})$ (%)/carrier material	Maximum FA burnup at end of cycle (GW·d(t HM))[a]
Belgium				
PWR (17 × 17 – 24)	2	96	7.5 (4.9)/U_{tails}	47.7
France				
PWR (17 × 17 – 24)	20	1400	6.7 (4.5)/U_{tails}	40.0
Germany				
PWR (18 × 18 – 24)	2	72	6.9 (4.6)/U_{tails}	43.4
PWR (16 × 16 – 20)	5	504	6.3 (4.2)/U_{tails}	49.0
PWR (15 × 15 – 20)	1	32	4.3 (3.0)/U_{nat}	42.0
PWR (14 × 14 – 16)	1	45	5.6 (3.8)/U_{nat}	37.0
BWR (9 × 9 – 1), 10 × 10 – 9Q)	2	212	5.4 (3.6)/U_{tails}	50.5
India				
BWR (6 × 6 – 1)	2	10		16.0
Switzerland				
PWR (15 × 15 – 20)	1	68	7.3 (4.8)/U_{tails}	51.0
PWR (14 × 14 – 17)	2	152	6.2 (4.1)/U_{tails}	40.0

[a] Maximum MOX assembly burnup of regular reloads (not lead test assemblies).

experience of MOX utilization in WWERs, although a multilateral programme to irradiate weapons grade plutonium is under way [4].

As for MOX fabrication, the industrial experience of MOX loaded LWRs has demonstrated that the technology has been mastered, although delays are being experienced in some countries in licensing MOX fuel to increase discharge burnups.

8. PLUTONIUM ISSUES

A detailed treatment of some plutonium issues is provided in two topical papers in this proceedings [5, 27]. Only some aspects will be highlighted here.

Large international industrial facilities are currently available in France and the UK for the reprocessing of spent fuel, and further facilities are in operation or under construction in China, India, Japan and the Russian Federation. Conditioning of the wastes (HLW, fuel assembly structural materials and technological waste) arising from reprocessing has been improved considerably and is now operated industrially. Their final disposal is technically simpler than that of unreprocessed SNF.

Plutonium separated by reprocessing has been recycled on an industrial scale for many years (Table 9), and there is considerable experience for all activities required to support the recycle option: MOX fabrication (Table 6), spent MOX fuel storage (Table 3) and MOX reprocessing (Table 5). The recycling of plutonium in LWRs has a 100% safeguards record and is currently the only industrially available option for plutonium utilization.

The overview of the plutonium management status in the major nuclear countries indicates that the separated plutonium held in the nine Member States that report under INFCIRC 549 represent most of the world inventory, plausibly more than 95%. The calculated results of the separated plutonium inventory from 1980 to 2050 coincide well with the values reported in INFCIRC 549 up to the year 2003, indicating that the VISTA model is reasonably predictive. In the case of a low reprocessing ratio (0% after 2022), the VISTA model predicts a rapid decrease of separated plutonium holdings. Similarly, the level of holdings with 30% reprocessing of LWR fuels decreases after a peak in 2012. Only the case of 70% reprocessing indicates a continuous increase, unless MOX fabrication (and recycling in LWRs) is in due course increased over the current and short term planned capacities (Table 7). For the 30% reprocessing scenario, Fig. 21 represents the evolution of the separated plutonium inventory for three MOX refuelling scenarios (Table 10) out of the five that have been modelled, varying linearly between 2004, 2010, 2030 and 2050 [35]. Even in the 70% recycling scenario, the separated plutonium inventory can be reduced to zero by 2020, provided the fabrication (and MOX loading LWRs) capacity is adequate. In reality, separated plutonium holdings will never decrease to zero, as a working inventory roughly approximate to two years of MOX fabrication will be required to continue to support industrial operations.

As plutonium is the most radiotoxic constituent of spent fuel, separating and burning it as MOX fuel reduces the long term environmental legacy of back end wastes (Section 6.3). However, recycling the plutonium in LWRs increases the levels of americium, which is the next most long term radiotoxic element in fuel waste and curium, a high short term heat emitter. Additionally, LWRs cannot fully consume the plutonium, as its isotopic characteristics

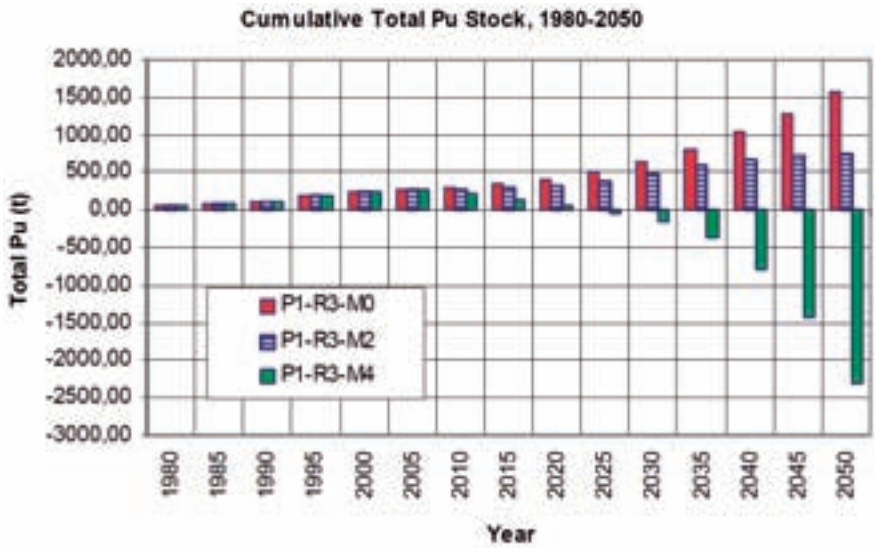

FIG. 21. Separated plutonium inventory for three MOX/U refuelling ratios in LWRs.

deteriorate over time. Even so, whatever options there might be in the future, recycling plutonium reduces the required final disposal liability and repository volumes (Table 11).

In repositories limited by temperatures at the surface of waste packages (essentially Yucca Mountain in the USA and clay formations as in Belgium, France and Switzerland), the benefit in repository volumes is somewhat smaller than the gain in conditioned waste volumes. This is essentially due to a 2.8 times higher curium inventory in HLW from spent LWR (U + MOX) fuel, as compared with HLW from spent uranium fuel.

The economic benefit in final disposal liability is, furthermore, not proportional to the required repository capacity, since some cost items are independent or only weakly dependent on volumes.

TABLE 10. MOX REFUELLING RATIOS
(% MOX fuel in a reload)

Percentage MOX/reload	M0	M2	M4
Up to 2004	Historical	Historical	Historical
2010	3	4	7
2030	3	4	10
2050	0	7	30

TABLE 11. VOLUMES OF WASTE CONDITIONED FOR FINAL DISPOSAL
(m^3/t HM discharged from an LWR)

Cycle option	
Open cycle (U fuel)	2.0
Mono-recycling in LWRs	1.0
Recycling in FBRs: Pu from LWR U fuel	0.5
Recycling in FBRs: Pu from LWR (U + MOX) fuel	0.5[a]
Recycling in FBRs: Pu and Am from LWR fuel	0.4

[a] Only 2% more.

The combined effect of heat release and repository volume dependence on financial liability levels can best be illustrated by the reserves to be set aside by the spent fuel producers and calculated by the waste disposal authorities, ANDRA (France) and NIRAS-ONDRAF (Belgium). The data are retrieved from a report of the French 'Cour des Comptes' and from the Belgian report SAPHIR 2 (Table 12).

Long term storage of spent fuel or separated plutonium until such a time that fast reactors might be deployed is not the optimum technical solution. On the timescale that fast reactors may be required, the use of plutonium with deteriorated characteristics due to ageing and/or previous recycling in LWRs (Annex II), or with originally poor isotopic characteristics due to the increased discharge burnups foreseen by 2050 (Fig. 20), will not be a problem, as fast reactors are less sensitive to the isotopic composition of the plutonium than LWRs [27]. As an example, Table 13 illustrates the 'plutonium equivalent' values (which is the value of plutonium as fissile material), under the assumption that

TABLE 12. FINANCIAL RESERVES TO BE SET ASIDE FOR HLW DISPOSAL (RELATIVE TO THE OPEN CYCLE)

Back end option	ANDRA	NIRAS
Open cycle	100	100
Reprocessing U SNF (but not MOX SNF) until 2010	78–79	—
Open cycle thereafter	55–57	—
Reprocessing U SNF, but open cycle for MOX SNF, permanently	33–36	39–49
Reprocessing (U + MOX) SNF		

TABLE 13. PLUTONIUM EQUIVALENT VALUE OF AGED PLUTONIUM IF RECYCLED IN A PWR OR AN LMFR (kG Pu^{eq}/kG Pu^{tot})

Plutonium separated from:	Age (a)	PWRs [a]	PWRs [b]	LMFRs [a]
PWR U fuel 55 GW·d/t	0	0.55	0.55	n.a.[c]
	10	0.32	0.40	n.a.[c]
	30	0.19	0.35	0.54
	70	0.13	0.33	0.53
PWR MOX fuel 45 GW·d/t	0	0.45	0.45	0.73
	10	0.22	0.36	0.60
	30	Negative	0.27	0.47
	70	Negative	0.22	0.39

[a] Pu and Am recycled in the same reactor.
[b] Pu only (Am built up by ageing is separated before MOX fuel fabrication).
[c] n.a.: not applicable.

americium resulting from ^{241}Pu decay is managed together with the plutonium, not being separated from it before fabrication into MOX or being conditioned into americium targets for incineration in the same reactor. If americium is separated from the plutonium before fabrication into PWR MOX fuel, the loss of fissile value by ageing is less dramatic, but still significant. Furthermore, disposal of the separated americium increases the radiotoxicity of HLW.

Moreover, it is imperative to pursue reprocessing and MOX fuel management on an industrial scale, and for the technology to be available and well mastered when the deployment of fast reactors will become a strategic necessity. Fast reactors are the best suited type to eliminate americium, provided that reprocessing technology is properly adapted to separate americium, and that fabrication and demonstration of americium targets has progressed.

Non-proliferation is sometimes invoked as an issue affecting plutonium management. National and international safeguards, with increasing sophistication, have been developed and are in place to cope with this concern. Weapons grade plutonium (some of it in excess of defence requirements), separated plutonium from low burnup spent fuels and low burnup spent fuel inventories are practically the only attractive plutonium types for weapons purposes. Plutonium separated from current burnup LWR fuels could only be elaborated into a nuclear explosive (not a weapon) with the help of heavy and sophisticated technologies. Degraded plutonium from high burnup uranium fuel or from spent MOX fuel is unattractive as material for nuclear explosives (Section 11). In this respect, there is an advantage to recycle the plutonium into MOX fuel as long as the safeguards records are efficient, rather than to store it,

as separated plutonium or plutonium in spent fuel, with the expectation that current safeguards will be kept equally operational over a long or very long term.

The economics of the fuel cycle are affected by recycling of plutonium and/or uranium, but not to a prohibitive extent. Generic analyses show a small cost advantage for the once-through option, which is likely to be insignificant in overall generating costs (Section 10).

The timescale to develop and implement industrial facilities of the closed fuel cycle runs to several years, as has been demonstrated by experience in France, Japan, the UK and elsewhere. It is essential to run such facilities on a representative level, to have the technology updated and the trained personnel available (Section 5.3) when the demand will expand (Fig. 4). This is the real challenge, as pressure of economic considerations or political decisions currently favour the wait and see option.

9. RepU ISSUES

A detailed treatment of some reprocessed uranium issues is provided in a topical paper [36].

The residual isotopic content of reprocessed ^{235}U (RepU) depends on the reactor type and discharge burnup of the fuel. Its ^{236}U content is another factor that determines its value as a fissile material to fuel various types of reactor. Two other isotopes, ^{232}U and ^{234}U, present in larger quantities in RepU than in virgin U, result in an increased radiological source term. All these factors have an impact on the management, processing and recycling of RepU.

Reprocessing in several countries has generated large quantities of RepU. Its purification and conditioning for storage, re-enrichment and/or direct utilization is now a routine operation. Since, unlike plutonium, the fissile value of RepU does not deteriorate with time, it can be stored as an indigenous fissile material reserve for an unlimited period of time. If economic conditions or uncertainties about the future dictate that it be recycled immediately, this can be and is being done on an industrial scale.

The re-enrichment can be, and has been, performed by centrifugation and by blending with higher enrichment low enriched uranium (LEU), medium enriched uranium (MEU) or highly enriched uranium (HEU). Experience has shown that the resulting enriched RepU (ERU) meets the standards of feed material for fabrication into fuel and loading in LWRs, AGRs, RBMKs and HWRs.

Fuel fabrication experience with ERU is significant, as well as its utilization in NPPs. Some countries are recycling all their RepU, in order not to accumulate a stockpile, which might become a liability; others are recycling only part of the arisings, to establish and maintain the technology and their competence in it.

In the longer term, the isotopic composition of RepU will become less attractive (Fig. 19 and Annex II) for re-enriching it into LWR fuel. Furthermore, the MEU and HEU resources will become exhausted. As a consequence, only uranium separated from SNF discharged during the first two decennia under consideration will be utilized as a fissile resource to fuel LWRs and WWERs. The balance is, however, a valuable resource for HWRs and LMFRs (or any other type of fast reactor).

10. ECONOMICS

Fuel cycle costs (FCCs) must be kept reasonably low to facilitate the competitiveness of nuclear power. However, they constitute only 22–31% of the operation and maintenance (O&M) cost [37], which is the generating cost of fully amortized NPPs, and 11–20% of the generating cost for not yet amortized plants [7, 38].

An international working group under the auspices of the OECD/NEA performed a generic cost analysis of nuclear fuel cycle steps for different back end options [39]. More recently, additional analyses of such data and of possible trends have been reported, providing alternative views [7, 40–42]. Given the international consensus approach of the OECD/NEA studies, it is the basis that is most commonly utilized (Table 14).

These cost assumptions must be compared with the current market situation and with some data from additional publications:

(a) The natural uranium price reached 80 USD/kg U in 2005 with ceilings said to be now above 100 USD/kg U [43]. The tendency is for further increases due to exhaustion of utility stocks, insufficient primary uranium production, limited availability of known secondary supply sources with appropriate specifications and, starting from 2012, the completion of the HEU–LEU agreement for Russian weapons grade HEU [44].
(b) The spot uranium conversion cost in 2005 reached 12 USD/kg U [43] and is vulnerable due to the limited number of relevant facilities.
(c) The current uranium enrichment cost shows no tendency to increase, and the current bounds of 90–110 USD/SWU could be reasonable for the long

TABLE 14. LOWER AND UPPER BOUNDS TO UNIT COSTS FOR LWR FUEL CYCLE

(Some values have been converted from euros to USD using a conversion rate of 1 EUR = 1 USD.)

Cost item	USD 2000	Lower	Upper
Natural uranium (U_3O_8)	USD/kg U	20	80
Conversion	USD/kg U	3	8
Enrichment	USD/SWU[a]	80	120
UOX fuel fabrication	USD/kg U	200	300
MOX fuel fabrication	USD/kg HM	1000	1500
UOX fuel reprocessing	USD/kg HM	500	900
MOX fuel reprocessing	USD/kg HM	500	900
UOX fuel interim storage (50 years)	USD/kg U	100	300
UOX fuel geological disposal	USD/kg U	300	600
HLW geological disposal	USD/kg U	80	200
LMFR MOX fuel fabrication	USD/kg HM	1200	2000
LMFR MOX fuel reprocessing	USD/kg HM	1000	2000
SNF transportation	USD/kg HM	40	60

[a] SWU: separative work unit.

term. The existing plans to shut down gaseous diffusion plants in the USA and France with deployment of new gas centrifuge facilities and a stable SWU supply from the Russian Federation leave no room for cost increases for at least the next 20 years.

(d) The LEU fuel fabrication cost is well known for large scale production facilities currently in operation in several countries. For reprocessed uranium enrichment and fuel fabrication, the costs need to be increased between 10 and 25% due to the additional expenses for radiation protection and waste management.

(e) Fabrication of plutonium containing pelletized fuels (MOX) for thermal and fast reactors is (and is expected to be in the future also) relatively expensive (1000–2000 USD/kg HM), primarily due to the need for highly automated and expensive production lines and for plutonium containing waste management. Nevertheless, R&D for new fuel types (such as vipac) shows the possibility for reductions in cost (~1000 USD/kg HM for LMFR MOX). It is difficult to predict the cost for AMOX type fuels, with

high content of MAs, and of americium targets, contemplated for future nuclear P&T technologies.

(f) Spent fuel reprocessing cost is heavily scale, country and technology dependent. It was shown that for simplified PUREX-type processing the cost might be decreased by about 25%. The cost range 400–600 USD/kg HM for future reprocessing plants involving less complicated and more compact or non-aqueous technologies might be achievable. There is some, but only limited, experience and relevant cost data for decommissioning industrial reprocessing facilities. These expenses should be taken into account when reprocessing cost is reconsidered [45].

(g) For high burnup SNF and spent MOX fuel, the possible need for additional interim cooling time and/or for conditioning of these fuels may result in additional expenses.

(h) The 300–600 USD/kg HM cost range for SNF disposal is based on detailed assessments of projected expenses in several countries. Disposal costs around 500 USD/kg HM (interim storage cost not included) were reported for projects of widely differing capacities in Finland (2600 t HM) and in the USA (83 500 t HM), while the Spanish project is evaluated at more than 800 USD/kg HM for a 7000 t HM project [46].

(i) In some countries, HLW disposal cost may be relatively high; for example, the Japanese project for an HLW repository of 50 GW(e) NPP capacity will cost around 25 000 MUSD, while the US project share for civil SNF from 100 GW(e) NPP capacity is estimated as 42 000 MUSD.

(j) Taking into account the absence of industrial experience of geological disposal of SNF (UOX and MOX) and HLW, the costly delays in deployment of permanent disposal repositories and the probable need for implementation of additional confinement and engineered barriers, cost ranges of 600–1000 USD/kg HM for UOX SNF geological isolation and 230–360 USD/kg HM for HLW disposal might be more reasonable (Section 8). Unlike reprocessing costs, deep disposal costs might be revised upwards considerably in the future. For instance, in 2005, EDF had to take into account a 46% increase of estimated future costs for deep disposal, but only a 3.6% increase for the total cost of reprocessing [47].

(k) Over the last two years, interest rates have dropped to historic lows and seem set to stay at low levels for many years to come. Such interest rates cannot justify anymore the high discount rates adopted for calculating the FCC. This biases comparison between closed cycles and open cycles, which are more long term cost loaded.

All these new cost realities invite consideration with extreme prudence of any FCC based comparison between different fuel cycle options. Whatever the

option, the total fuel cost for LWRs will probably be in the range 2400–3000 USD/kg HM, roughly equally subdivided between front end and back end.

Insufficient industrial experience exists to provide equally reliable cost fundamentals for fast reactor fuels with their high initial fissile (plutonium or ^{233}U) contents. It should be kept in mind that fast reactor fuels with high burnups (up to 20%) will contain many more fission products and MAs than LWR spent fuel and that the HLW disposal cost should be adjusted to take this into account. The additional option for recycling MAs as plutonium + MA fuels or as MA targets will complicate the combined reprocessing–refabrication facility design and operation of the reprocessing facilities. These considerations should be taken into account in evaluating a wait and see option that would be based on plutonium in future fast breeders being better valorized and the interim americium accumulation being solved by incinerating it in those reactors.

11. ASSESSMENT OF CRITERIA FOR COMPARISON OF NUCLEAR FUEL CYCLE OPTIONS

This section addresses assessment criteria for and a rough comparison of nuclear fuel cycle options on the basis of qualitative judgments made by experts.

Assessment criteria have been elaborated for years because suitable criteria are very important for comparison of fuel cycles. Subjectivity in the choice of criteria, changes in the hierarchy of criteria and the purpose of an assessment can often lead to different, if not contradictory, conclusions. Criteria for nuclear energy as a sustainable energy resource and the justification of their selection are outlined in detail in Ref. [7].

Six of these criteria (use of fissile material resources, availability of technology, economic considerations, environmental impact, public acceptance and proliferation resistance) are utilized in this section for comparison of one open fuel cycle and two closed fuel cycles. The open fuel cycle uses LEU in LWRs. The second cycle is a thermal recycle option in which spent LWR fuels are wet reprocessed, with the recovered plutonium being recycled as MOX fuel (by dilution with some depleted uranium generated in the enrichment plants) and the recovered uranium being recycled as ERU. The third cycle is fast recycle in which spent fast reactor fuels are reprocessed and multiply recycled. Two fast recycle options are considered: a wet reprocessing and recycle of the plutonium into MOX fuel, and a dry reprocessing generating metallic fuel through pyroprocessing.

11.1. USE OF FISSILE MATERIAL RESOURCES

The fissile material inventory (mainly plutonium) has been increasing in the past and will continue to increase (Figs 17 and 18), even if some is being recycled [5]. In addition, the end of the cold war has created a legacy of surplus fissile materials in excess of defence requirements (primarily weapons grade plutonium and highly enriched uranium) in the USA and the States of the former USSR. If these fissile material resources are used in the nuclear fuel cycle, the natural uranium utilization is reduced and the sustainability of nuclear power generation is prolonged at whatever time horizon is considered.

In this respect, the open cycle has the largest fissile material exhaustion rate. It can be reduced by 20% by mono-recycling plutonium and RepU in LWRs (Section 8) and by a factor of 2–10 by recycling in fast reactors.

11.2. AVAILABILITY OF TECHNOLOGY

The availability of technology can be expressed as the expected number of years of necessary R&D investment until the specific technology is fully available for commercialization on an industrial scale. This can be long after the end of the research period, as commercialization can take years of additional R&D and demonstration (RD&D). It is usual in the nuclear field for a technology available as a result of laboratory scale testing to require an additional 10–20 years for demo-scale and/or prototype operation before it can be commercially implemented, because any change in application of nuclear technology has to be demonstrated to be safe in the view of the regulatory authorities. Achieving regulatory approvals can take another 10 years.

The technology for the open fuel cycle and for the reprocessing/recycling option is fully operational, except for final disposal of spent fuels and of HLW, which has not yet been implemented commercially. The delay in the implementation of final disposal is not due to availability of technology but to social interference, such as site definition, local opposition and/or licensing delays. The fast recycle option still needs large investments in RD&D and requires time. In practice, it is at the stage of laboratory/demo/prototype development for LMFRs with MOX fuel, of basic R&D for fast recycle with metal fuel and of conceptual design for most Generation IV systems.

11.3. ECONOMIC CONSIDERATIONS

There have been continued efforts to compare the economic competitiveness of various nuclear fuel cycle options [39]. The core of the subject lies in forecasting the future price of uranium. As mentioned in Section 10, the closed cycle (thermal MOX concept) is a little more expensive than the open cycle, but the future evolution of cost constituents is highly unpredictable.

For any industrial product (chemicals, glass, metal, paper, plastic, etc.), recycle is always more expensive than 'throwaway'. So it is normal for minimizing plutonium in the final waste by recycling to be slightly more expensive than the once-through option. However, reducing the radiotoxicity of the waste by transmuting MA and long lived fission products will increase the generating cost to an extent, depending on how far it is taken. A cost–benefit evaluation taking into account economic considerations, exposure of workers and stage of industrial development should be the basis for deciding how far to go in attempting to reduce waste radiotoxicity.

11.4. ENVIRONMENTAL IMPACT

Nuclear power has a number of environmental impacts that are similar to those caused by fossil fuels, such as thermal pollution, non-radioactive and radioactive emissions, and occupational risks. However, the principal concern about nuclear fuel cycle operations has been with radioactivity. Waste volume and human health effects can be taken as representative environmental indicators.

Waste volume estimates have been carried out for various nuclear fuel cycles [48, 49]. Significant volumes of waste come from the mining process, so-called mill tailings. Reprocessing/recycling reduces mill tailings and also (Sections 6.3 and 8) the high level waste to be disposed of. This means that the more recycling is performed, the less waste volume is produced.

A comparative study on the radiological effects of two reference PWR based fuel cycles (i.e. open and closed) in 1996 [50], on the basis of actual data of radioactive releases from reference facilities, indicated that the collective radiological impacts of the two fuel cycle options considered are about the same order of magnitude for both the general public and the fuel cycle facility workers (Table 15).

TABLE 15. COMPARISON OF TOTAL COLLECTIVE DOSE (man-Sv/GW(e)·a) [50]

To workers		To population	
One-through	Recycle	Once-through	Recycle
3.43	3.37	1.65	2.97

Note: Conditioning of spent fuel is before final disposal and, for both options, long term storage and final disposal have not been accounted for.

It is important to note that the collective doses are composed of very small doses to a large number of people over a long period of time. Each contribution is negligible compared with the level of natural background radiation. However, promoted by the as low as reasonably achievable (ALARA) philosophy, further measures are implemented to continuously evaluate and reduce health impacts.

The environmental impact of plutonium and its decay product americium in a final disposal, owing to their long half-lives, is an argument for consuming the plutonium and not letting part of it decay into americium, especially if one takes into account the difficulties in locating final repositories.

For when, how many and how far to transmute MAs and long life fission products, the assessment must take into account trading a reduction of long term radiotoxicity risks of the waste against increased collective doses to current inhabitants. For instance, as mentioned in Section 7, this remark applies obviously to managing the curium inventory [27].

11.5. PUBLIC ACCEPTANCE

The nuclear power option is impeded in many countries by public concerns over safety and environmental consequences, namely the fear that humans will receive radiation doses and that large land areas will be contaminated with radioactive material and will thereafter be unusable. In this respect, public acceptance can be addressed in terms of risk aversion, which applies also to fuel cycle facilities.

Even though public acceptance of nuclear fuel cycles varies from country to country, the transport of spent fuel, the reprocessing plants and the final repositories have been the focus of public opposition. It can be inferred that the recycle option could be faced with stronger public opposition, because the various recycle options involve more facilities and transport requirements. In

particular, for recycle in fast reactors, the public feels more at risk because the great operational experience with the other NPP types is lacking for fast reactors, and hence safety has not yet been demonstrated to the same significant confidence level.

11.6. PROLIFERATION RESISTANCE

All inventories of nuclear materials (plutonium in SNF, separated plutonium, WPu and HEU, whether in weapons or in excess to defence requirements) need to be physically protected and safeguarded in order not to constitute a risk to national and international security. While protective measures are efficient at present, there can be no guarantee that this efficiency will be maintained for decades or centuries. A timely use of these materials that degrades their attractiveness or even eliminates their usefulness for weapons usage should be encouraged. It is important to note that a nuclear fuel cycle with more consumption of these fissile material resources is of benefit from both the non-proliferation and environmental points of view.

Recently, consideration of proliferation resistance or vulnerability has been a topic of renewed interest in the context of Generation IV and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). In both programmes, proliferation resistance is being considered as one of the most important factors in finding advanced and innovative nuclear systems. There has been been an attempt to find an appropriate methodology to evaluate quantitatively several nuclear energy system alternatives. However, a reasonable methodology for integrated analysis of proliferation resistance has not yet emerged.

In general, every nuclear fuel cycle entails potential proliferation risks. Among the various fuel cycle steps, however, it is recognized that centrifuge enrichment and reprocessing are the most sensitive proliferation issues. The exploitation of these technologies is not a problem in countries with solid credentials as to their peaceful only endeavours and in ‘weapon states’ that accept international safeguards on their activities. For these countries, to foreclose increasing their, or installing an, enrichment and/or reprocessing capacity will not result in any non-proliferation benefit. The technology of these processes is known and could be (and has been) utilized by any technically advanced country intending to develop an indigenous (legal or illegal) activity. As for bacteriological and chemical threats, the only solution is to establish and maintain strong international non-proliferation barriers and detection systems. In this last respect, illegal reprocessing activities are much easier to detect than enrichment activities, and at an earlier stage.

Intrinsic features of proliferation resistance are often assessed regardless of time frame. For their evaluation, however, it should be noted that the degree of proliferation resistance, which is a relative value, could be different with the lapse of time. For example, direct disposal is generally quoted as the best option in this respect, but it might well be the most vulnerable due to the so-called 'plutonium mine' effect, especially since almost all repository concepts are now based on retrievability or even reversibility and since the delay in operating final repositories results in long term interim storage of SNF.

A 100 years after discharge, thermal and decay heat will have rendered the reactor grade plutonium contained in SNF an easily accessible and very attractive material for potential proliferators, and it will become increasingly so for over 20 000 years [51]. Indeed after 300 years the radiation levels of SNF become tolerably low and may no longer be lethal for human contact without heavy shielding. Both the plutonium (half-life 88 years) and gamma radiation, deleterious for weapon applications, will have decreased significantly. On the contrary, fresh plutonium from high burnup uranium fuel [27] and particularly plutonium from spent MOX fuel [51] is inappropriate for weapons utilization [52, 53]. Therefore, recycling plutonium in LWRs can be rated as improving proliferation resistance.

To summarize, the degree of proliferation resistance of the closed fuel cycle is higher in the long term, since the plutonium is degraded and the cycle consumes plutonium, reducing the plutonium inventory. However, wet reprocessing is the most vulnerable in the short term because pure plutonium is separated during the process, and appropriate physical protection and safeguards measures must be implemented. Confidence is strong that such measures can be maintained in the short term. It is less evident that they would continue to be operational over the long term (100–20 000 years!).

11.7. SELECTION OF A FUEL CYCLE OPTION

Political decisions play the leading role when utilities select a fuel cycle option.

If the legal authorities dictate that nuclear is not a sustainable option for electricity generation and that NPPs must be decommissioned, the utility will not recycle its fuel or will discontinue doing so. In particular, the longer cooling times required for MOX spent fuel would unduly prolong the decommissioning period and the nuclear liability of the utility.

If the law does not limit future nuclear electricity generation, the legislator may put more emphasis on one or more criteria and dictate the fuel cycle policy on this basis.

In countries not imposing an option on their utilities, the utilities themselves will rank the criteria. Some will place their main emphasis on financial criteria and probably select 'wait and see' as their current option for the back end of the nuclear fuel cycle. Others will emphasize minimization of uncertainty in long term liabilities and select reprocessing and immediate recycling of the separated plutonium and RepU. Large utilities, such as EDF, plan the management of spent fuel in a long term perspective, up to the time fast breeders will need to be deployed [2].

12. CONCLUSIONS

Evolution scenarios of installed nuclear capacity, reactor types and fuel characteristics are predictable up to 2050. The sustainability of the nuclear power option relies heavily on the back end of the fuel cycle, more so, during this period, than on the front end.

For the three basic options, the open cycle (once-through policy), reprocessing and recycling, and wait and see, the technologies have been developed and industrially deployed and applied for many years without problems. The only exception is final disposal, ultimately common to all three options, which has only up to now been developed to the stage of assured feasibility, but which is confronted with licensing and public acceptance difficulties. In this respect, those options that minimize the number and volumes of geological repositories are assets for the sustainability of nuclear power generation.

Since sustainability also intrinsically implies the potential for future utilization of the fissile material still present in spent fuel, plutonium and uranium, the conditions and timeliness of their reuse must be considered.

Separating and burning plutonium as MOX fuel reduces the long term legacy of fuel cycle back end waste. For economic reasons (low plutonium content in the SNF), reprocessing cannot be applied to low burnup fuels (CANDU and RBMK). It must be, however, and is being applied for technical reasons to GCR fuels. The recycling of separated plutonium in LWRs is at present the only possibility and is being implemented industrially. However, this increases the americium and curium inventories and cannot consume the plutonium extensively. Long term storage of spent fuel or separated plutonium until the time has come to deploy fast reactors is not the best solution, as the fissile worth of plutonium degrades and americium inventories increase with storage time. This effect is, however, only dramatic for plutonium inventories from higher burnup fuels (PWR, BWR and WWER) and less so for plutonium inventories with low ^{241}Pu contents (GCR, CANDU and RBMK). When the

deployment of fast reactors becomes a necessity, the use of plutonium with deteriorated characteristics due to previous recycling in LWRs will not be a problem, as fast reactors are less sensitive to the isotopic composition of plutonium and can eliminate americium to a major extent.

On the contrary, reprocessed uranium practically does not age and can be kept as a stockpiled fissile resource almost indefinitely. Significant industrial recycling of reprocessed uranium in LWRs and RBMKs has demonstrated its value.

The economics of the fuel cycle are only slightly affected by recycling of plutonium and/or uranium at current U_3O_8 and other front end prices. However, future price changes may modify the economic differential between the back end options.

Besides best utilization of fissile resources and economics, other criteria must be taken into consideration: technology availability and further development, environmental impacts, public acceptance and proliferation concerns. These all affect the back end policy that a nuclear utility can select or is authorized to implement.

However, to provide security of supply in the future, all three back end options should be pursued on an industrially significant scale in order that the technology and expertise be maintained and further improved.

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Annex I

PROJECTION UP TO 2050 OF INSTALLED CAPACITIES AND REACTOR TYPES

I–1. PRESENT STATUS OF NUCLEAR REACTORS WORLDWIDE

At the end of 2003, a total of 439 power reactors were operating worldwide. The pressurized water reactor (PWR) is the most prominent type, with 263 units, represented by 213 PWRs and 50 WWERs. The boiling water reactor (BWR) is next with 92 units, including two advanced boiling water reactors (ABWRs). These are followed by 38 CANDU and other pressurized heavy water reactors (PHWRs), 17 light-water graphite-moderated reactors (RBMKs), 14 advanced gas cooled reactors (AGRs), 12 Magnox gas cooled reactors (GCRs) and three fast breeder reactors (LMFRs). Pressurized water reactors and BWRs are widely deployed in East Asia, North America and Western Europe. WWERs and/or RBMKs are mainly deployed in the Russian Federation, Eastern Europe and India. The PHWRs are deployed in, for example, Canada, China, India and the Republic of Korea. Gas cooled reactors are solely deployed in the UK.

Eighty per cent of operating reactors started up during the two decades from 1970 to 1990 (Fig. I–1). The gross reactor capacity increased annually to take advantage of the scale factor (Fig. I–2).

After this period, construction of new reactors slowed down. In recent years, construction of new reactors has started to increase, especially in East and South Asia, but has remained slow in Western countries. Various reasons, such as anti-nuclear sentiment, a highly competitive power generation market and better utilization of existing capacity, are contributing to this trend.

I–1.1. SITUATION OF EACH COUNTRY

Argentina. There are two reactors operating in Argentina. The Government would like to expand the use of nuclear energy and is currently considering the financial aspects of doing so.

Armenia. According to official statistics, the nuclear power station at Metamora generated two billion kW·h of electricity or almost 40% of Armenia's aggregate power output in 2004. The Armenian Government's decision in 2003 to grant the Russian Federation financial control over the nuclear power station at Metamora has proved of benefit to the Soviet built

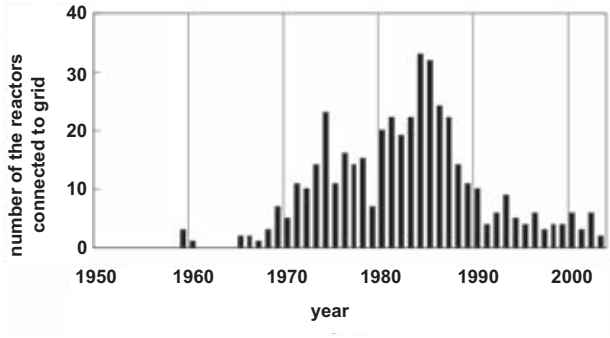

FIG. I–1. Year when connected to grid.

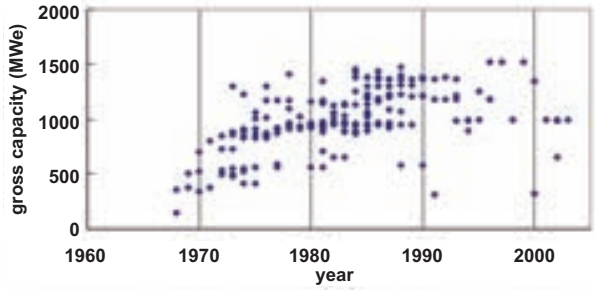

FIG. I–2. Connected PWRs: year and capacity.

facility. While pressure has been applied by the European Union (EU) to shut down the plant, the Armenian Government's position is that the plant is safe enough to continue to operate for years to come.

Belgium. A 2003 law imposes that each reactor must be shut down 40 calendar years after the start of commercial operation. Accordingly, the first reactor, Doel-1, should be shut down in Feb. 2015 and the last one, Tihange-3, in Sep. 2025. Article 9 of the law is a force majeure clause, stating that the phase-out obligation could be revised in the case of a threat to the security of the electricity supply. Given that 55% of electricity in Belgium is from nuclear power and the Kyoto targets (compulsory in Europe) will be extended to cover additional years, it is expected that the force majeure clause will have to be exercised. The Belgian utility Electrabel is confident that the NPPs are capable of pursuing operation for at least 60 years after the start of their commercial operation.

Brazil. There are two operating reactors in Brazil. The Government is interested in expanding the use of nuclear energy and is reviewing the cost associated with this expansion.

Bulgaria has agreed with the EU that it will close Kozloduy Units 3 and 4 by 31 Dec. 2006. At the same time, they are considering new NPPs as replacements for these units.

Canada has no reactor life limitations in its licences. Plants are reviewed on a five year basis and operations can continue while addressing any concerns identified in this review. Currently efforts are ongoing to change the review cycle to ten years. From the point of view of financial return, plants are expected to operate for at least 30 years. The current thinking of utilities is that plants should operate for at least 40 years. Operation after this will depend on the condition of plants and their competitiveness in the market. There is no programme or plans to terminate the nuclear power programme.

China is embarking on an ambitious nuclear power programme. There are currently nine plants in operation, providing over 7000 MW and representing 1.8% of installed capacity. Construction of two 1000 MW plants will begin before the end of 2005, and there are plans for as many as 32 additional 1000 MW plants on-line by 2020. China is following a 40 year licensing programme; however, the plants are being designed and built to be leased for 60 years of operation.

Czech Republic. There are currently six nuclear units operational, which provide most of the electricity for the country. The units have come under criticism from the EU and neighbouring countries concerning their safety. Where there has been pressure applied to close down the plants the Government has resisted.

Finland. There are currently four operating units and a fifth one under construction. Nuclear energy provides a major portion of the electricity for the country. Opinion is mixed about nuclear power; about half the population favour it and half oppose it. The majority believe that five reactors are sufficient for the needs of the country and that additional plants should not be built.

France. The next reactor will be built for EDF at Flamanville. The plant, scheduled to operate in 2012, will be a Framatome ANP 1600 MW advanced PWR model, the EPR. It is seen as the first in a series to replace EDF's existing reactors, starting in the 2020s. The first 900 MW(e) PWR is scheduled for 48 years of operation; however, all of the other plants are scheduled for 60 years of operation.

Germany. A law passed in 2001 requires the decommissioning of all 19 nuclear plants by about 2023. The Stade PWR was the first plant closed in this framework. The Obrigheim PWR is the next plant to be scheduled for

closure, by the end of 2005. This law is being questioned, as reducing Germany's carbon dioxide emissions by 40% by 2020, to meet the Kyoto targets, cannot be achieved by replacing the nuclear capacity by fossil fuel plants as they are insufficiently non-emitting sources.

Hungary. The 1866 MW nuclear capacity supplies 40% of Hungary's electricity. It is licensed into 2012. The operator is considering investment of 150 billion forint (US $750 million), to extend the lives of the station's four VVER-440s by 20 years.

India has 14 NPPs in operation and is constructing nine more. The Karachi Nuclear Power Complex (Kanupp) has been engaged in an extensive upgrading process with regard to its outdated plant control system and related instruments to qualify for another 15 years' lease of the design life. It has operated for over 30 years. India intends to have at least 20 000 MW of nuclear power on-line by 2020.

Islamic Republic of Iran. The Government has approved the construction and operation of 20 000 MW of nuclear power, specifically via construction of ten additional nuclear power plants. Issues continue to be raised by foreign governments about the construction of the NPPs and the need for a uranium enrichment facility.

Japan. There are no reactor life limitations in licences. In 1996, ageing effects on plant performance were widely examined by the utilities for plants approximately 30 years old. This was conducted under a Government initiative. In this examination, an appropriate maintenance programme was initiated assuming a 60 year period of plant operation. For each plant, the maintenance programme will be reviewed every ten years. This practice will be applied to all plants as they near 30 years of operation. There is no national policy to terminate nuclear power. Japan has 54 NPPs operational and is constructing three more.

Lithuania is considering asking the EU to allow it to extend the timetable for the closure of its ageing Soviet built RBMK plant at Ignalina, consisting of two 1300 MW reactors. At the EU's insistence, Lithuania agreed to shut down the first unit in 2004 and close the plant completely by 2009. This NPP supplies about 80% of all the energy produced in the Baltic state.

Mexico. There are currently two NPPs operating in Mexico. The Government, while it realizes the need for additional electricity supplies, has not been able to come to any decision about the need for additional nuclear plants.

Netherlands. There is currently only one operating reactor in the country. There are no indications that this situation will change in the near future.

Pakistan. There are two operating plants. The Government would like to expand this capacity but is also evaluating the alternatives to increasing the use of nuclear energy.

Republic of Korea. At the present time, there are also no reactor life limitations in licences. Korean NPPs have a design life of 40 years, except for Kori No. 1 and Wolsung No. 1 plants, which have design lives of 30 years. However, the utility is considering an extension of the life of Kori No. 1 plant. There are no plans to terminate the nuclear power programme. The Republic of Korea has 20 units operating and has plans to construct eight more up to 2015.

Romania has one NPP operational. While the country is interested in additional NPPs, the issue of funding the next plant is considerable.

Russian Federation. Plans are under way to extend the life of some of its old RBMKs by at least 15 years. The reactors are approaching their 30 year limit, but through modernization an additional 15 years is feasible. Under existing regulations they would need to be shut down in the next few years. Most of the RBMK units were built in the 1970s.

Slovakia. There is a nuclear power station at Jaslovské-Bohunice and another one at Mochovce. Between the two stations there are six units in operation and another two under construction. Nuclear energy provides about 55% of the electricity in the country.

Slovenia has one NPP operating. While the Government is supportive of nuclear energy, it is having difficulties identifying the funding necessary for construction of additional plants.

South Africa has two plants operational and is considering new technology for the additional nuclear plants that are under consideration. South Africa, the biggest carbon dioxide emitter on the African continent, may be told to reduce its CO_2 emissions at the next international meeting of Kyoto Protocol signatories.

Spain has nine plants in operation. While the Government remains supportive of the use of nuclear energy, the conservative parties in the country have been very vocal in their opposition to the addition of new nuclear plants and would like to have the existing plants closed.

Sweden. There are eleven operating plants in the country. Several years ago, the Government passed legislation that no additional plants were to be built and that the existing plants were to close at the end of their useful lives. As the pressures of energy and atmospheric pollution continue to mount, this position is being considered for revision.

Switzerland. There are five operating plants in the country. The policies of Switzerland are very similar to those of Sweden. The way in which Sweden addresses the issue is likely to have an impact on Swiss policy.

Ukraine. After full commissioning of the four plants under construction, no additional plants will be built in the country. Ukraine will complete the No. 3 power unit of the Khmelnytskyy NPP and build the No. 5 power unit of the Rivne NPP.

United Kingdom. There is a mix of activity in the UK nuclear programme. Of the 30 operating reactors in the country almost half have been in operation for over 40 years. Without further life extensions or new construction, the UK will have only one operating reactor within 20 years. There are currently no plans for building new reactors in the UK.

United States of America. Twenty six of its 104 reactors have already had their operation extended by 20 years, for a total of 60 years. Moreover, applications to extend the lives of 42 others have also been made or it has been indicated that the licence for these reactors will also be extended to 60 years. The current expectation is that the vast majority of the remaining 36 reactors will also have their operation extended to 60 years. There is discussion about possible operation beyond 60 years. There are no plans to terminate the nuclear power programme.

I–1.2. OPERATION IMPROVEMENTS

Various attempts have been made to realize maximum utilization of existing nuclear reactors.

The typical capacity factor for a nuclear plant has been less then 80%. This has been improving, but much more production can be achieved with improved operation of plants. For example, in the USA in 1980 the capacity factor was only 55%, while in 2003 it was 89.6% (Fig. I–3) [I–1]. This is an over 60% improvement in performance. Another way to consider this is to look at output. In 1990, US NPPs produced 610 billion kW·h. In 2003, this same group of plants produced 763 billion kW·h. This is equivalent to 19 new 1000 MW(e) power plants. Other countries are also working to improve the operation of their plants and increase productivity; however, there remains the opportunity for greater production.

The next step to improvement is to increase the output of existing plant through upgrades. Typically, NPPs have equipment limitations that prevent greater power production even though they can produce more power safely. As companies work through continuous improvement programmes, bottlenecks to power production have been identified and eliminated. These improvements can take the form of steam generator replacement, a new rotor in a turbogenerator or as simple as a faster acting valve. The increase in the upgrading activities in recent years is shown in Fig. I–4 [I–2]. Spain has a programme to

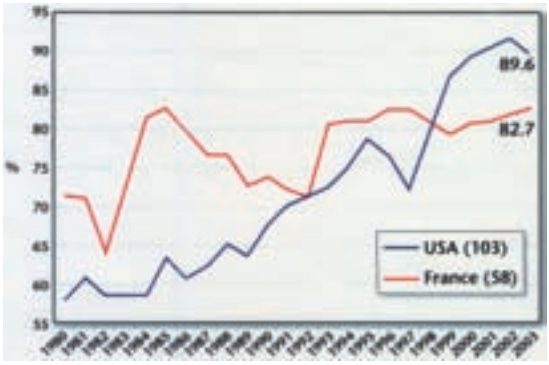

FIG. I–3. Capability factors (Kd) of the French and US NPP fleets.

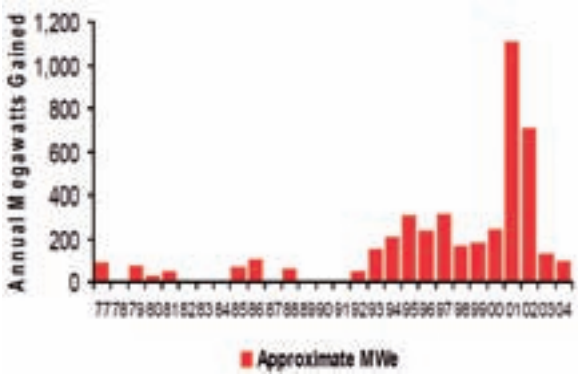

FIG. I–4. Annual power capacity increases (1977–2004) by the nuclear industry.

add 810 MW(e) (11%) to its nuclear capacity through upgrading of its nine reactors. Some 519 MW(e) of the increase is already in place. In Finland, the Olkiluoto plant recently had its capacity increased by 23%, now generating 1680 MW(e) with two reactors, and the Loviisa plant has been uprated by almost 100 MW(e) (11%). In Switzerland, the capacity of its five reactors has been increased by 12.3%. Belgium also increased its total installed capacity by over 7% [I–3]. There is no defined number for the amount of additional capacity that can be achieved, but experts in this area believe that a 10% addition would not be unreasonable.

Approximately one third of the reactors in the world have been operating for 25 years or more. Since the design lives of nuclear reactors are generally 30–40 years, ageing management is an interest common to many countries. Plant

lifetime management is being considered in many countries, not only for assuring safe operation of the plant but also for seeking the possibility of plant life extension beyond the initially expected life. In the USA, more than 15 reactors have obtained licence renewals from the NRC, which extends their operating lives from the original 40 years to 60 years; most others are expected to apply for similar extensions [I–3]. In the Russian Federation, the designated service life of an NPP is 30 years, and almost half of their NPPs will reach 30 years before 2010. The Russian government extended the operating lives of the country's 12 oldest reactors by 15 years after a comprehensive work programme to ensure extension [I–4]. In France, no limited licensing period exists but the integrity of NPPs is confirmed by a complete review and reassessment. Proactive ageing management is being implemented by EDF as an integral part of the operation and maintenance programme. Exceptional maintenance and a complete review every ten years are parts of this programme. The EDF NPPs should easily last more than 40 years [I–5]. Many other European countries and some Asian countries, such as Japan and the Republic of Korea, are applying similar plant lifetime management programmes. For example in Japan, 30 year old nuclear reactors were examined in the late 1990s for the possibility of extending their lives up to 60 years, with positive results. In the UK, the oldest GCRs, which were originally expected to have useful lives of 20–25 years, have been authorized for 50 years but, for economic reasons, are closing earlier. Most other Magnox plants are licensed for 40 year lifetimes [I–3].

As discussed previously, a number of countries have national policies that have stopped the expansion of nuclear energy and some are working towards closing down their operating plants. Most of the other existing nuclear power countries do not have any policy limiting its application, but no country has a policy promoting the use of nuclear energy over any other form of electric power production. In the energy sector, various countries provide incentives for the production of electricity from environmentally friendly sources. However, except in very few countries, nuclear power is not considered to be part of an environmentally friendly mix. This is likely to change. As countries implement the Kyoto Protocol, they are beginning to realize the true environmental benefit of electricity produced by nuclear power.

I–2. PROJECTION OF SPENT FUEL CHARACTERISTICS AND QUANTITIES UP TO 2050

The IAEA's projection [I–6] of nuclear capacity was used as a basis. Projections are derived from this country-by-country bottom-up approach,

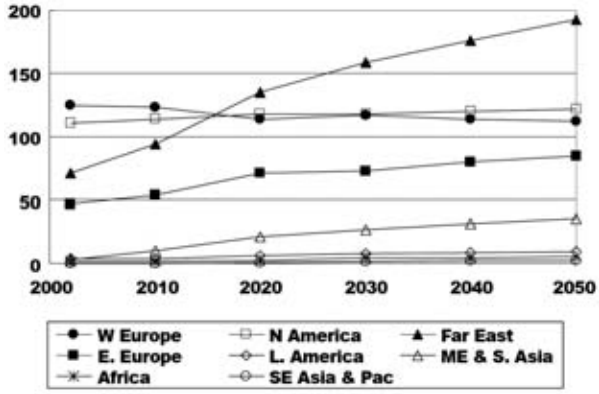

FIG. I–5. Capacity transition in each region.

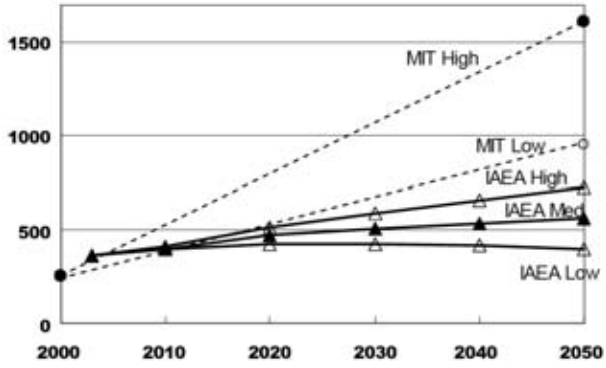

FIG. I–6. Projected nuclear capacity.

utilizing the IAEA's own continuously carried out estimates and the estimates of other international organizations. In this projection, high and low nuclear capacities are given for each region up to 2030. The average of high and low values were employed and, for 2040 and 2050, simply extrapolated values were used. While the nuclear capacities of East Asian countries are expected to increase threefold in 50 years, the nuclear capacities of North American and Western European countries do not show much change. Increases are also foreseen in Eastern Europe, the Middle East and South Asia (Fig. I–5).

Compared with the IAEA median value, the nuclear capacities in 2050 assumed in the Massachusetts Institute of Technology (MIT) report [I–7] are 50% higher in the low case and 150% higher in the high case. This is mainly because of the higher nuclear ratio estimated in the MIT report (Fig. I–6).

At present, seven types of reactor are operating in the world: PWR, BWR, WWER, PHWR, RBMK, AGR and Magnox. Even if the lifetime of the reactors is extended to 60 years, none of them can survive until 2050. Therefore, they should be replaced at some point before 2050. There are two possibilities.

The first is replacement by Generation III reactors, which are greatly improved from existing Generation II reactors. Generation III reactors have a standardized design for each type to expedite licensing and reduce capital cost and construction time, and increase availability and operating life, typically to 60 years. Fuel use and amount of spent fuel discharge are reduced by higher burnup, and many incorporate passive or inherent safety features. There are many types of Generation III reactor that are ready for application. For example, in the USA, ABWR, System 80+, AP-600 and AP-1000 are developed, and their designs have been certified by the NRC. Two ABWRs are already in operation in Japan, where a large advanced PWR has also been developed, with construction of two being planned. South Korea has developed APR-1400. Framatome ANP has developed a large European pressurized water reactor (EPR), which has the flexibility to follow loads, fuel burnup of 65 GW·d/t and the highest thermal efficiency of any light water reactor, at 36%. Availability is expected to be 92% over a 60 year service life. The first unit is about to be built at Olkiluoto in Finland and the second at Flamanville in France. Framatome ANP is also developing SWR1000, a 1000–1290 MW(e) BWR. The Russian Federation has developed V-392 (an advanced WWER-1000), WWER-91 and WWER-1500 V448. Canada has developed CANDU-9 and CANDU-NG based on its reliable CANDU-6. CAMDU-9 has flexible fuel requirements, ranging from natural uranium through slightly enriched uranium, direct use of spent PWR fuel to thorium. Canada has also developed ACR-700 [I–8].

Another possibility is direct transition to Generation IV reactors. The Generation IV International Forum is a US led group of ten countries, which identified six reactor concepts for further investigation, with a view to commercial deployment by 2030. The six reactors are:

(1) Gas cooled fast reactors;
(2) Lead cooled fast reactors;
(3) Molten salt reactors;
(4) Sodium cooled fast reactors;
(5) Supercritical water cooled reactors;
(6) Very high temperature gas reactors.

Many countries, such as France, Japan and the USA, are considering a move to Generation IV reactors. There are two ways, one is direct transition and the other is a transition through advanced LWRs. With an assumed 60 year

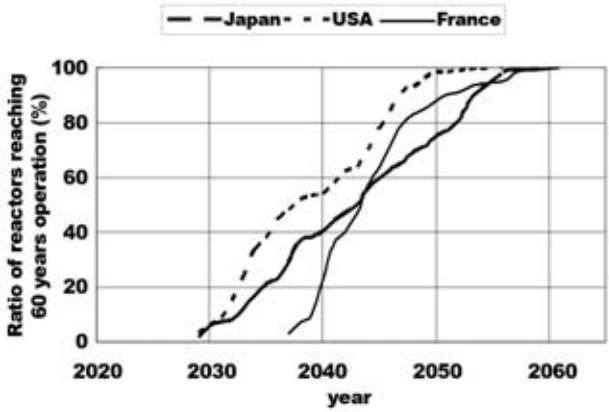

FIG. I–7. Replacement period after 60 years of service.

life, Japan and the USA need to start introduction of new reactors in the late 2020s and France needs to start in the late 2030s (Fig. I–7).

If the Generation IV reactors were ready to start in 2030 as expected, direct transition looks feasible for those countries. But there are two uncertain points. One is whether the lives of existing reactors can be extended to 60 years and the other is whether Generation IV reactors will be ready by 2030. For life extension, it seems that many countries are considering that 60 years is feasible, but there are no 100% guarantees and, because of this, most countries are applying lifetime management programmes including reviews every ten years. For example, if the life was shortened by ten years from 60 years, the USA would need to replace half of its retiring reactors with advanced LWRs. It is difficult to predict the availability of Generation IV reactors in 2030. However, the introduction of new technology normally needs long lead times and little experience is available for Generation IV reactors.

The EPRs ordered by TVO in 2003 and EDF in 2004 are the outcome of evolutionary improvements beginning with Yankee Rowe (commercial operation of which started in 1961). They are advanced PWRs, resulting from Framatome and Siemens combining in 1989 (i.e. some 15 years earlier) their APWR designs, based on lessons learned, respectively, from Chooz and Civaux, and from KKI-2, KKE and GKN-2. The TVO Olkiluoto-3 is scheduled to come on-stream in 2009 and is designed for a service life of 60 years. The Flammanville-3 EDF is scheduled to start operation in 2012, and EDF has no plans to order its next EPRs before 2020. Therefore, the deployment of EPR generating capacity at EDF will start around 2030 and be spread over at least ten years thereafter. With their 60 years life, they will only need to be replaced by other reactors at the end of the century.

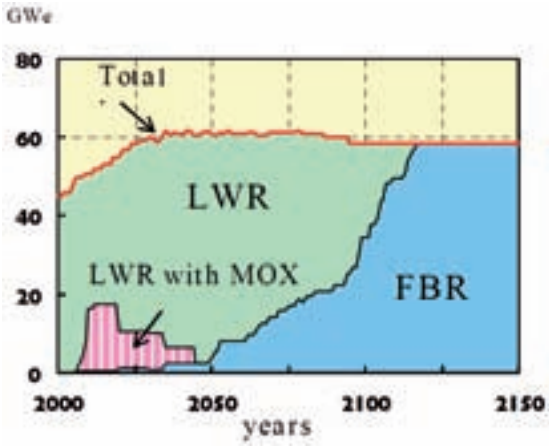

FIG. I–8. One example scenario of transition through advanced LWRs.

The experimental FBRs (DFR, Phénix and Joyo) started operation, respectively, in 1962, 1973 and 1977. Commercial operation of the corresponding demonstration FBRs occurred with a delay increasing with the years, probably due to more restrictive licensing attitudes: PFR did start in 1976, Superphénix was reasonably ready to start in 1992 and, as of the end of 2004, Monju has not yet started commercial operation. It could be said that the availability of Generation IV reactors in 2030 contains some uncertainty. In addition, when countries decide to introduce Generation IV reactors on a massive scale, they want to have some confidence in their long term reliability. However, since there are no plans for introduction on a commercial scale of Generation IV reactors foreseen at present, such confidence may not be available in 2030.

Recently, the Japanese Government announced that existing reactors will be replaced by large sized advanced LWRs first at around 2030, and then after their completion of 60 years life they will be replaced by FBRs. One of the scenarios presented at the Japanese Atomic Energy Committee review meeting illustrates this idea (Fig. I–8).

The USA is planning to introduce Generation IV reactors from 2030, but scenarios are not presented. Since most Generation IV reactors use a closed cycle, they need fuel cycle facilities. Thus, countries that do not have such facilities are likely to select a transition through advanced LWRs.

Therefore, it can be assumed that LWRs will play a major role up to 2050.

The installed capacities for each reactor type up to 2050 were projected under the following assumptions:

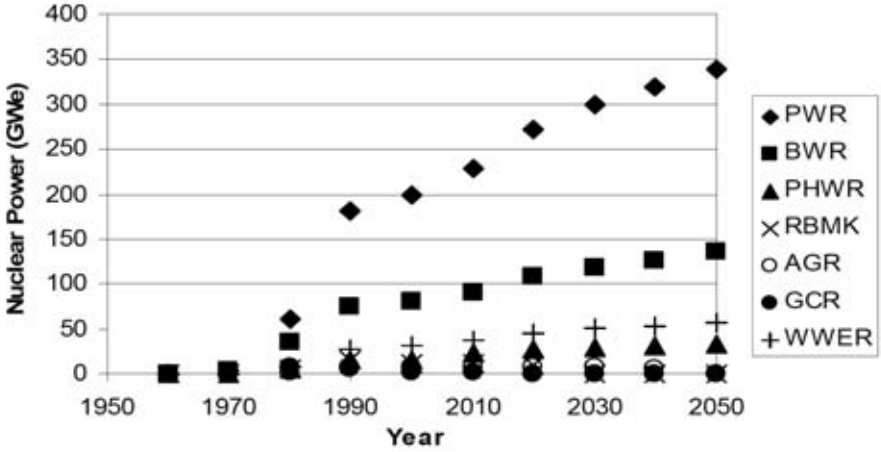

FIG. I–9. Projected installed nuclear capacities for each reactor type.

(a) PWRs, BWRs, WWERs and PHWRs will be deployed as replacements for existing reactors and as new installations until 2050, as needed by the modelled scenario.
(b) Existing reactors will be operated for 60 years, except that RBMKs are to be shut down after 40 years and GCRs will be shut down as scheduled (the last one to shut down in 2010).
(c) Projected nuclear capacity will be maintained by adding new capacity or shutting down existing reactors as needed.
(d) Reactors replacing the retiring reactors or newly installed reactors will be built in the ratio of: PWR 60%, BWR 24%, WWER 10% and PHWR 6%, i.e. in the same proportions as at present.

On the basis of all this, Fig. I–9 shows the projected installed nuclear capacities of each reactor type up to 2050.

Table I–1 presents the annual discharges of spent nuclear fuel (SNF), the contained uranium and plutonium, and their fissile ratios, derived from each type of reactor calculated by the VISTA code based on the installed nuclear capacity projection, taking into account the projected trend in average load factors, which is also loaded into VISTA. In this case, it was assumed that about 30% of the discharged UOX SNF, except that for PHWRs and RBMKs, is reprocessed and that about 4–7% of the new fuel for BWRs and PWRs will be MOX fuel. Mixed oxide SNF is included in these data.

TABLE I–1. SPENT NUCLEAR FUEL, URANIUM, PLUTONIUM AND THEIR FISSILE RATIOS FROM EACH REACTOR TYPE

(UOX + MOX for PWR and BWR types)

		1960	1970	1980	1990	2000	2003	2010	2020	2030	2040	2050
PWR	SF (tHM/y)	0.00	19.20	1,112.58	3,659.10	4,197.36	4,079.30	4,429.24	4,806.96	5,056.97	5,304.81	5,510.95
	U (tHM/y)	0.00	18.61	1,074.89	3,509.86	3,980.30	3,852.11	4,169.15	4,492.22	4,698.33	4,913.85	5,087.46
	U235 (%)	0.00	1.02	0.98	0.90	0.77	0.72	0.69	0.62	0.57	0.54	0.52
	Pu (tHM/y)	0.00	0.15	9.31	33.47	42.80	45.06	50.05	58.35	63.42	69.71	77.56
	Fissile Pu (%)	0.00	72.78	71.55	69.21	66.11	64.64	63.92	62.20	61.08	60.32	59.39
BWR	SF (tHM/y)	0.00	15.60	893.52	1,807.71	1,995.02	1,987.70	1,934.38	2,079.07	2,190.16	2,329.69	2,461.37
	U (tHM/y)	0.00	15.22	870.32	1,743.25	1,902.51	1,887.16	1,827.66	1,951.35	2,044.83	2,170.98	2,288.30
	U235 (%)	0.00	1.05	1.03	0.91	0.78	0.73	0.68	0.61	0.56	0.54	0.53
	Pu (tHM/y)	0.00	0.11	6.42	15.23	19.05	20.75	20.88	24.18	26.33	29.32	33.21
	Fissile Pu (%)	0.00	74.94	74.04	70.05	66.60	64.94	63.83	62.11	61.01	60.38	59.56
PHWR	SF (tHM/y)	0.00	21.88	799.64	1,784.48	2,213.71	2,474.30	3,069.15	2,850.09	2,412.41	2,119.54	1,882.45
	U (tHM/y)	0.00	21.64	790.78	1,764.23	2,188.60	2,446.56	3,032.55	2,807.98	2,367.15	2,071.85	1,832.95
	U235 (%)	0.00	0.24	0.24	0.24	0.24	0.24	0.24	0.22	0.17	0.14	0.11
	Pu (tHM/y)	0.00	0.08	3.07	6.94	8.61	9.55	12.44	12.87	12.22	11.60	10.95
	Fissile Pu (%)	0.00	73.95	73.95	73.56	73.56	73.81	73.22	69.96	66.17	63.39	61.00
RBMK	SF (tHM/y)	0.00	11.39	224.40	867.24	597.71	616.68	601.73	553.77	160.09	0.00	0.00
	U (tHM/y)	0.00	11.19	220.44	851.12	586.11	604.71	587.35	540.29	156.19	0.00	0.00
	U235 (%)	0.00	0.75	0.75	0.74	0.73	0.73	0.68	0.67	0.67	0.00	0.00
	Pu (tHM/y)	0.00	0.05	0.89	3.54	2.49	2.57	2.76	2.56	0.74	0.00	0.00
	Fissile Pu (%)	0.00	68.95	68.95	68.16	67.50	67.50	64.37	64.09	64.09	0.00	0.00
AGR	SF (tHM/y)	0.00	0.00	13.76	112.39	301.45	233.50	259.46	261.50	265.00	209.90	174.20
	U (tHM/y)	0.00	0.00	13.46	110.03	292.27	225.22	249.88	251.79	255.17	202.11	167.73
	U235 (%)	0.00	0.00	0.80	0.81	0.72	0.67	0.65	0.65	0.65	0.65	0.65
	Pu (tHM/y)	0.00	0.00	0.07	0.53	1.65	1.35	1.51	1.53	1.55	1.23	1.02
	Fissile Pu (%)	0.00	0.00	64.79	65.66	60.33	58.24	57.70	57.62	57.62	57.62	57.62
GCR	SF (tHM/y)	9.27	1,002.39	1,854.55	1,493.16	918.03	724.42	344.59	0.00	0.00	0.00	0.00
	U (tHM/y)	9.22	996.80	1,841.77	1,481.60	910.92	718.81	341.68	0.00	0.00	0.00	0.00
	U235 (%)	0.46	0.46	0.42	0.39	0.39	0.39	0.36	0.00	0.00	0.00	0.00
	Pu (tHM/y)	0.02	2.27	4.90	4.28	2.63	2.08	1.05	0.00	0.00	0.00	0.00
	Fissile Pu (%)	80.15	80.15	76.74	74.76	74.76	74.76	73.27	0.00	0.00	0.00	0.00
WWER	SF (tHM/y)	0.00	1.61	151.10	671.11	662.35	708.76	818.78	857.71	861.77	904.77	930.80
	U (tHM/y)	0.00	1.55	145.44	645.17	630.62	674.24	776.02	804.04	800.74	838.68	860.74
	U235 (%)	0.00	1.15	1.15	1.14	1.08	1.07	1.04	0.96	0.88	0.86	0.84
	Pu (tHM/y)	0.00	0.02	1.55	6.98	7.41	7.97	9.40	10.32	10.66	11.27	11.66
	Fissile Pu (%)	0.00	78.06	78.06	77.80	75.90	75.74	75.06	73.09	71.57	71.16	70.75

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Annex II

SPENT FUEL CHARACTERISTICS AS A FUNCTION OF DISCHARGE BURNUP AND STORAGE TIME

The CESAR code developed by the French CEA has been used to calculate some examples of typical spent fuel characteristics as functions of discharge burnup and storage time.

UOX1 represents the characteristics of a previous generation of French PWR fuel (enrichment level of about 3.5% and discharge burnup at 33 GW·d/t), while UOX2 represents current French PWR fuel (enrichment level of about 3.7% and discharge burnup at 45 GW·d/t). Finally, it was thought to be important to illustrate the characteristics of a potential future fuel by examining a high discharge burnup fuel, UOX3 (enrichment level of about 4.5% and a discharge burnup at 60 GW·d/t).

For each of these fuels, the spent fuel characteristics were calculated for three different cooling times, namely 10, 20 and 30 years.

The results given in Tables II–1 and II–2 correspond to the characteristics of the fissile materials three years after reprocessing. This represents approximately the minimal time for refabrication of the recovered plutonium and the recovered uranium (RepU) into the recycle fuel, taking into account transportation and management of the process inventory.

Tables II–3 and II–4 represent the isotopic compositions of fissile materials separated from spent ERU (enriched reprocessed uranium) fuels fabricated from the material shown in Table II–1. It was assumed that re-enrichment was achieved by blending MEU (20% ^{235}U) to RepU; the resulting ERU enrichment to reach the target burnup is indicated in Tables II–3 and II–4.

Since RepU practically does not age [II–1], the cooling time has very little effect on the isotopic composition of the reprocessed uranium, as can also be seen in Table II–1. For this reason, no calculations were performed for cooling times of 20 and 30 years.

Table II–4 shows, among other things, the significant proportion of fissile material, specifically ^{239}Pu and ^{241}Pu, in the recovered plutonium, which is therefore amenable to further recycling, even if the spent fuel reaches a high discharge burnup.

Table II–5 represents the isotopic compositions of plutonium separated from spent MOX fuels fabricated from the material shown in Table II–1 and depleted uranium (enrichment tails). For the left hand side of Table II–5 (the first two columns), the MATADOR code, also developed by the CEA, was used to calculate the initial plutonium content necessary in order to obtain the discharge burnups shown in column 3.

TABLE II–1. URANIUM ISOTOPIC COMPOSITIONS RESULTING FROM DIFFERENT DISCHARGE BURNUPS AND COOLING TIMES

	UOX fuel characteristics					Fissile material characteristics			U content
U type	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	U 234 (%)	U 235 (%)	U 236 (%)	(Kg/tHM)
U1-10	UOX1	3,5	33	10	3	0,0226	1,070	0,438	956
U2-10	UOX2	3,7	45	10	3	0,0215	0,715	0,523	941
U3-10	UOX3	4,5	60	10	3	0,0247	0,622	0,679	924

	UOX fuel characteristics					Fissile material characteristics			U content
U type	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	U 234 (%)	U 235 (%)	U 236 (%)	(Kg/tHM)
U1-20	UOX1	3,5	33	20	3	0,0237	1,070	0,439	956
U2-20	UOX2	3,7	45	20	3	0,0237	0,715	0,523	941
U3-20	UOX3	4,5	60	20	3	0,0286	0,622	0,680	924

	UOX fuel characteristics					Fissile material characteristics			U content
U type	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	U 234 (%)	U 235 (%)	U 236 (%)	(Kg/tHM)
U1-30	UOX1	3,5	33	30	3	0,0247	1,070	0,439	956
U2-30	UOX2	3,7	45	30	3	0,0258	0,715	0,523	941
U3-30	UOX3	4,5	60	30	3	0,0322	0,622	0,680	924

TABLE II–2. PLUTONIUM ISOTOPIC COMPOSITIONS RESULTING FROM DIFFERENT DISCHARGE BURNUPS AND COOLING TIMES

	UOX fuel characteristics					Fissile material characteristics						Pu+Am content
Pu type	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	Pu 238 (%)	Pu 239 (%)	Pu 240 (%)	Pu 241 (%)	Pu 242 (%)	Am 241 (%)	(Kg/tMLi)
Pu1-10	UOX1	3,5	33	10	3	1,52	61,00	23,70	7,51	5,05	1,17	9,25
Pu2-10	UOX2	3,7	45	10	3	2,61	54,20	25,70	8,20	8,02	1,27	10,80
Pu3-10	UOX3	4,5	60	10	3	3,94	50,10	26,10	8,46	10,20	1,31	12,30

	UOX fuel characteristics					Fissile material characteristics						Pu+Am content
type Pu	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	Pu 238 (%)	Pu 239 (%)	Pu 240 (%)	Pu 241 (%)	Pu 242 (%)	Am 241 (%)	(Kg/tMLi)
Pu1-20	UOX1	3,5	33	20	3	1,45	63,20	24,60	4,80	5,22	0,75	8,93
Pu2-20	UOX2	3,7	45	20	3	2,50	56,20	26,90	5,27	8,33	0,82	10,40
Pu3-20	UOX3	4,5	60	20	3	3,78	52,00	27,40	5,43	10,60	0,84	11,80

	UOX fuel characteristics					Fissile material characteristics						Pu+Am content
type Pu	UOX type	Enrichment U5 (%)	Burn-up (GWj/t)	Cooling time (years)	Aging (years) after treatment	Pu 238 (%)	Pu 239 (%)	Pu 240 (%)	Pu 241 (%)	Pu 242 (%)	Am 241 (%)	(Kg/tMLi)
Pu1-30	UOX1	3,5	33	30	3	1,37	64,60	25,20	3,04	5,34	0,47	8,73
Pu2-30	UOX2	3,7	45	30	3	2,37	57,60	27,60	3,34	8,53	0,52	10,10
Pu3-30	UOX3	4,5	60	30	3	3,58	53,30	28,30	3,44	10,80	0,54	11,50

TABLE II–3. ISOTOPIC COMPOSITION OF URANIUM RECOVERED FROM SPENT ERU FUEL

	Target characteristics of spent fuel				Uranium isotopic composition		
RepU type	Processed U	Enrichment U5 (%)	Burn-up (GWj/t)	Final Ucontent (Kg/tHM)	U 234 (%)	U 235 (%)	U 236 (%)
RepU 1-10	U1-10	3,61	33	955	0,0305	1,15	0,817
RepU 2-10a	U2-10	3,63	33	955	0,0320	1,17	0,880
RepU 2-10b	U2-10	3,83	45	941	0,0284	0,80	0,963
RepU 3-10a	U3-10	3,87	45	941	0,0307	0,82	1,083
RepU 3-10b	U3-10	4,66	60	923	0,0304	0,72	1,224

TABLE II–4. ISOTOPIC COMPOSITION OF PLUTONIUM RECOVERED FROM SPENT ERU FUEL

	Target characteristics of spent fuel				Plutonium isotopic composition					
RepU type	Processed U	Enrichment U5 (%)	Burn-up (GWj/t)	Final Pu+Am content(Kg/tHM)	Pu 238 (%)	Pu 239 (%)	Pu 240 (%)	Pu 241 (%)	Pu 242 (%)	Am 241 (%)
RepU 1-10	U1-10	3,61	33	9,97	2,09	58,1	21,8	12,5	4,5	1,01
RepU 2-10a	U2-10	3,63	33	9,98	2,25	58,2	21,7	12,4	4,4	1,01
RepU 2-10b	U2-10	3,83	45	11,76	3,41	51,5	23,3	13,6	7,0	1,19
RepU 3-10a	U3-10	3,87	45	11,80	3,79	51,5	23,1	13,5	6,9	1,18
RepU 3-10b	U3-10	4,66	60	13,57	4,94	47,7	23,3	14,1	8,8	1,26

TABLE II–5. PLUTONIUM ISOTOPIC COMPOSITIONS RESULTING FROM DIFFERENT MOX FUEL TYPES

	Spent fuel characteristics (MOX base depleted U)					Plutonium isotopic composition					
MOX type	Processed Pu	Raw fissPu content(%)	Raw totPu content(%)	Burn-up (GWj/t)	Final Pu content (Kg/tMLi)	Pu 238 (%)	Pu 239 (%)	Pu 240 (%)	Pu 241 (%)	Pu 242 (%)	Am 241 (%)
MOX 1-10	Pu1-10	3,86	5,54	33	41,84	2,69	42,47	28,02	16,13	8,44	2,26
MOX 2-10a	Pu2-10	4,50	7,07	33	52,56	3,01	41,05	29,08	15,39	8,84	2,63
MOX 2-10b	Pu2-10	5,41	8,50	45	62,05	3,77	36,03	30,69	15,43	11,45	2,63
MOX 3-10a	Pu3-10	5,89	9,83	45	69,17	4,17	39,28	27,85	15,58	10,42	2,70
MOX 3-10b	Pu3-10	7,00	11,70	60	74,60	5,54	33,56	29,94	15,40	12,04	3,51

It can be seen from Table II–5 that MOX fuel that were to be manufactured from second generation plutonium should have a very high plutonium content, owing to the degraded isotopic composition of this plutonium. Such plutonium cannot be fabricated into LWR MOX nor be used in LWRs [II–1]. For this reason, the calculation was not repeated for first generation plutonium recovered from spent fuel cooled for more than ten years, as such first generation plutonium already has less favourable isotopic compositions (Table II–2).

In practice, however, spent MOX fuel is reprocessed industrially in dilution with spent uranium fuel, and the resulting separated plutonium, intermediate between first generation plutonium (Table II–2) and second generation plutonium (Table II–5), could be fabricated into MOX fuel for loading into LWRs if LMFR deployment were delayed beyond the expectation.

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MODELLING FOR NUCLEAR MATERIAL FLOWS IN THE NUCLEAR FUEL CYCLE

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Abstract

Tracking the actinide inventory of nuclear fuel at each step of the nuclear fuel cycle is very important, owing to the physical characteristics of actinides. The long term radiotoxicity and proliferation resistance of the spent nuclear fuel are among the most important challenges of nuclear technology. The major contribution to the long term radiotoxicity of spent nuclear fuel comes from its actinide content. In order to handle radiotoxicity and proliferation challenges, it is very important to know the actinide inventory in the spent nuclear fuel. The Nuclear Fuel Cycle Simulation System (VISTA) was developed to calculate fuel cycle material and service requirements to prepare one of the key issue papers of an international symposium, Nuclear Fuel Cycle and Reactor Strategies: Adjusting to New Realities, in 1997. Later, the model was expanded to enable actinide tracking. A simplified isotopic composition calculation program (CAIN) was added to the system. By adding CAIN, VISTA became capable of calculating the isotopic composition of spent fuel for any existing reactor type, for given fresh fuel compositions and for given discharge burnups.

1. INTRODUCTION

As early as the 1970s, the IAEA perceived the need to be able to estimate the nuclear energy needs and correlated requirements for uranium and services related to different nuclear fuel cycle strategies. The first computer tool to support such requirements was developed in order to support the International Nuclear Fuel Cycle Evaluation (INFCE) in 1977 [1]. This early tool was able to provide estimates on uranium and fuel cycle service requirements, but was limited to the open nuclear fuel cycle strategy.

It was an international symposium, Nuclear Fuel Cycle and Reactor Strategies: Adjusting to New Realities, held in 1997, which required additional estimates [2]. Those were based on different reactor and fuel cycle strategies, the most important of which was the inclusion of reprocessing–recycling strategies. A new model was developed and the Nuclear Fuel Cycle Simulation System (VISTA) was established.

These models and computer tools, analysing the nuclear fuel cycle in its different strategies and options, share the same basic information: the isotopic composition of spent nuclear fuels. The VISTA model uses the isotopic composition of spent nuclear fuel calculated by use of the CAIN (Calculation of Actinide INventory) computer code as its built-in module.

The VISTA program calculates, by year over a long period, the nuclear fuel cycle requirements for all types of reactor. Calculations could be performed for a reactor, the reactor park in a country or the worldwide nuclear reactor park. Natural uranium, conversion, enrichment and fuel fabrication quantities can be estimated. Furthermore, the quantities and qualities (isotopic composition) of unloaded fuels can be evaluated to let the user apply a recycling strategy if desired.

Data input is reduced to a few basic data in order to let non-nuclear fuel specialists develop different energy scenarios. The calculation speed of the system is quick enough to enable comparisons to be made of different options in a quite short time. Therefore, the new simulation system is designed to be an optimum mixture of accuracy, simplicity and speed.

Calculations using VISTA can cover the period ranging from the beginning of nuclear energy production to 2050 or 2100. The progam stores historical data in its database to make historical evaluations. In order to make estimates for the future, the historical data are supported by a number of projections that actually reflect the trends in the world nuclear industry.

The possible use of VISTA might be:

(a) To estimate actinide accumulations in spent nuclear fuel;
(b) To calculate the nuclear fuel cycle material and service requirements for selected scenarios;
(c) To compare different options for future nuclear fuel cycle development.

This paper gives details of the VISTA model and its built-in reactor model CAIN. The paper then gives a selected example scenario and the results for that scenario.

2. THE VISTA CODE

2.1. General comments

There are a number of models and computer tools available for calculating uranium and fuel cycle service requirements [3]. These models are based on sophisticated databases that include information on each nuclear

power reactor in the world. Such databases are useful for portraying the history and short term nuclear power projections. It is, however, very difficult to build such databases with a view towards the far future such as 30–100 years. The incentive for developing the new scenario based model was to simplify long term estimates [4].

The main assumption in the model is that it is possible to simulate the nuclear fuel cycle by taking into account the evolution of different types of reactor over the years, without the precision of using a reactor by reactor database. The reactor types taken into consideration in VISTA are pressurized water reactors (PWRs), boiling water reactors (BWRs), pressurized heavy water reactors (PHWRs), gas cooled reactors (GCRs), advanced gas cooled reactors (AGRs), Soviet design pressurized water reactors (WWERs) and Soviet design light-water-cooled graphite-moderated reactors (RBMKs).

2.2. Nuclear fuel cycle

The nuclear fuel cycle can be defined as the set of processes to make use of nuclear material and to return it to its normal state. The cycle starts with mining of unused nuclear material from nature and ends with safe disposal of used nuclear material in nature. Figure 1 is a simplified nuclear fuel cycle diagram showing the main processes in a recycle mode simulated in VISTA. The main steps in this mode are mining, milling, conversion, enrichment, fuel fabrication, use in a reactor, spent fuel storage, reprocessing and high level waste (HLW) storage.

2.3. Material flow model

Overall material flow for a nuclear fuel cycle can be sketched by tracking the nuclear material in each of the processes in the nuclear fuel cycle. The VISTA code is capable of simulating different nuclear fuel cycle models with different reactor and fuel types. For the purpose of this study, only existing nuclear fuel cycle options with commercially existing reactor and fuel types were simulated. The simulated fuel cycle options are once-through fuel cycles as well as uranium and plutonium recycling in some reactor types. Figure 2 shows the overall material flow diagram of the nuclear fuel cycle that is simulated in VISTA.

The first fuel type used in this simulation is uranium fuel from natural material, whereas the second fuel type is mixed oxide (MOX) fuel using reprocessed material. The second fuel type is limited to MOX containing uranium and plutonium in this study, since it is the only commercially available fuel from reprocessed material.

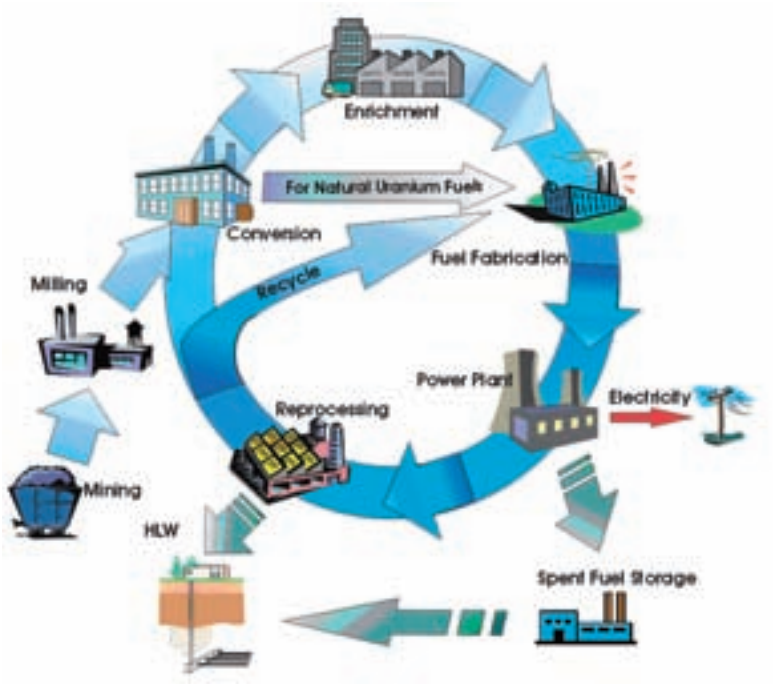

FIG. 1. Simplified diagram of the nuclear fuel cycle in recycle mode.

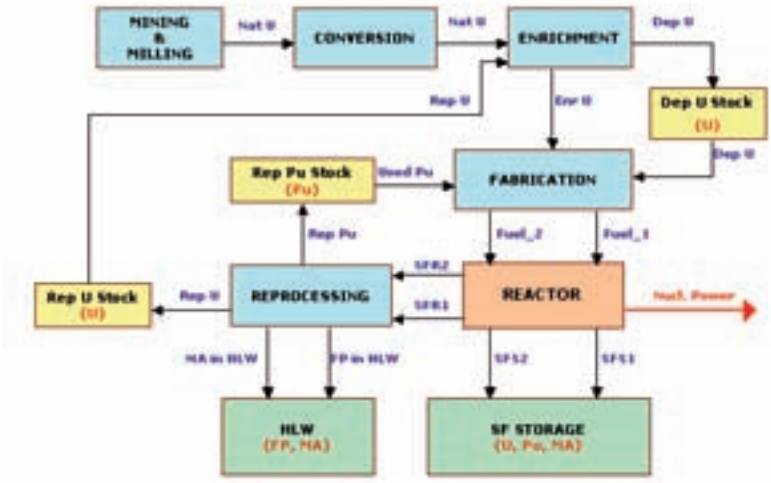

FIG. 2. Overall material flow diagram in VISTA.

2.4. VISTA information flow

The information flow in the VISTA simulation system is shown in Fig. 3. This figure displays only the main parameters. The input parameters are categorized into three groups. These input parameters can be entered by users freely, and then results can be compared for different values for any of the parameters.

2.5. Calculation model

The model is designed to estimate nuclear material and fuel cycle service requirements on an annual basis. The model also calculates, whenever relevant, cumulative requirements by adding the annual requirements. The calculation process for the case of once-through material flow is shown in Fig. 4. First, the annual fresh fuel requirement is calculated from the given electricity production and the other input parameters. Then, the calculation is performed for the front end and back end of the nuclear fuel cycle in parallel. The calculation scheme for the recycling case is similar, except for the inclusion of separation and the use of reprocessed material.

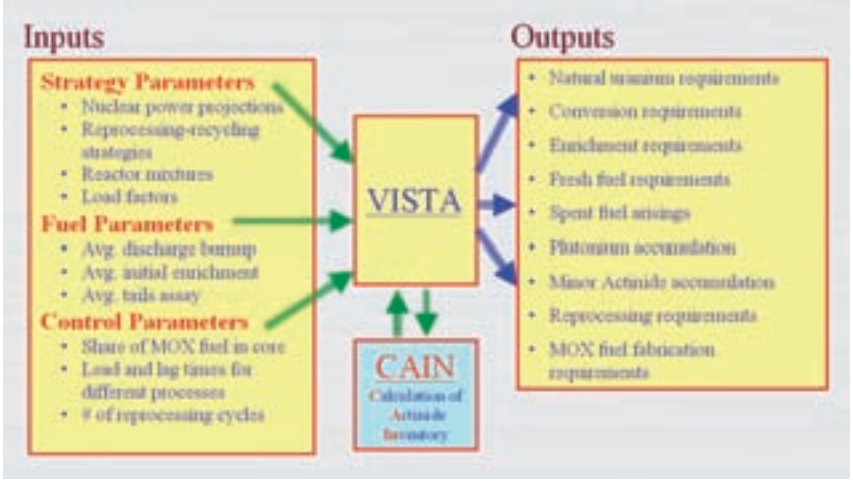

FIG. 3. VISTA information flow diagram.

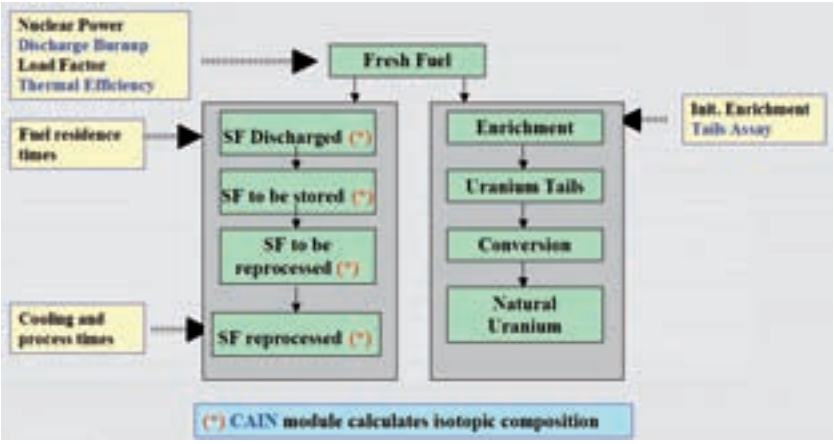

FIG. 4. Calculation process for the once-through case.

3. REACTOR MODEL

3.1. Need to track actinides

The actinide group includes the elements from Th (Z = 90) to Lr (Z = 103); however, major interest is given to the different isotopes of U and Pu (major components of fresh and spent nuclear fuel) and to the so-called minor actinides (MAs: Np, Am and Cm) with extremely long half-lives and high alpha (with alpha particle energies of 4–6 MeV, and therefore major contributors to the residual heat of spent fuel) and gamma radioactivities.

Assessment of the inventories of these elements/isotopes in spent fuel is important due to non-proliferation issues and the radiotoxicity of long lived isotopes. The latter relates to the open fuel cycle, when the safety of spent fuel storage and further final disposal should be justified and guaranteed, and to the closed fuel cycle, when the same considerations are applicable to immobilized HLWs containing MAs.

In the selection of actinides to be included in this calculation, some assumptions are made to simplify the chains to be calculated, in order to increase the speed of the calculation. Although natural uranium includes ^{234}U (<0.01%), this nuclide is ignored, because the transmutation from ^{234}U to ^{235}U is small. Short lived nuclides (half-lives less than 8 days) are omitted. That is, ^{237}U (7 d), ^{238}Np (2 d), ^{243}Pu (5 h), ^{242}Am (16 h), ^{244}Am (10 h) and ^{244m}Am (26 min) are assumed to decay and go to the next nuclide simultaneously. On the other hand, long lived nuclides (half-lives more than 400 years) are assumed to

be stable for the irradiation period. For example, ^{241}Am (432 a) is treated as stable. For the decay (cooling) period after discharge, all nuclides are treated by their actual decay scheme.

The transmutation chain is simplified according to the above assumptions. Fourteen nuclides are selected on the basis of the chain shown in Fig. 5. Among these 14 nuclides, only decays of ^{238}Pu (87.7 a), ^{241}Pu (14.4 a), ^{242}Cm (0.447 a) and ^{244}Cm (18.1 a) are considered during the irradiation period.

The characteristics of the major actinides (half-life, mode of decay, presence of gamma radiation, and total and alpha activities) are presented in Table 1. The concentrations of actinides and their specific activities are given for UO_2 fuel with an initial enrichment of 3.2% after irradiation in a 900 MW(e) PWR until burnup of 33 MW·d/kg U and five years storage.

It can be seen from Table 1 that about 90% of the alpha radioactivity of fresh spent fuel or HLW in the case of reprocessing, i.e. potential radiotoxicity, is contributed by ^{238}Pu, ^{241}Am and ^{244}Cm (spent fuel) or by ^{241}Am and ^{244}Cm (HLW).

The potential radiotoxicity of actinides may be defined, without taking into account any barriers, by calculating a 'source term' obtained by weighting the activity of each radionuclide by its specific toxicity coefficient (by ingestion or inhalation) and then by summing the resulting values. Table 2 shows the evolution of the potential radiotoxicity of spent fuel and the contribution of each long lived actinide and the total fission products (FPs, in sieverts per TW·h(e) produced). It can be seen that after the decay of highly radioactive FPs (^{137}Cs and ^{90}Sr) over about 300 years, the major contribution comes from

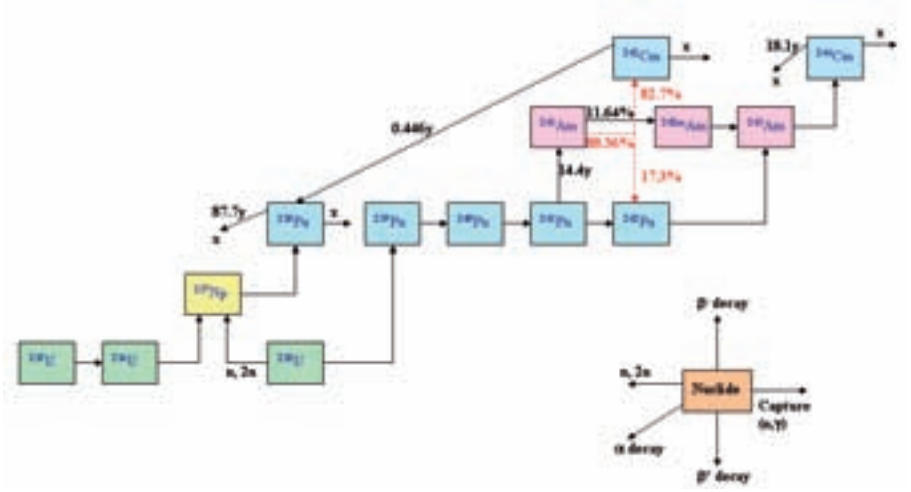

FIG. 5. The simplified transmutation chain used in VISTA.

TABLE 1. CHARACTERISTICS OF THE MAJOR ACTINIDES [5–8]

Nuclide	Half-life (a)	Mode of decay (%)	Gamma radiation	Content in spent fuel (g/t HM)	Total activity (Ci/t HM)	Alpha activity (Ci/t HM)
^{235}U	7.04×10^{8}	α	Yes	8 250	1.79×10^{-2}	
^{236}U	2.34×10^{7}	α	Yes	4 050	0.262	
^{238}U	4.47×10^{9}	α	Yes	943 000	0.317	
^{237}Np	2.14×10^{6}	α	Yes	437	0.308	0.15
^{238}Pu	87.7	α, [a]SF = 1.8×10^{-7}	Yes	140	2.39×10^{3}	2360, other Pu = 853
^{239}Pu	2.41×10^{4}	α	Yes	5 470	340	
^{240}Pu	6.56×10^{3}	α, SF = 5.7×10^{-6}	Yes	2 230	509	
^{241}Pu	14.4	$\alpha = 2.4 \times 10^{-3}$, $\beta = 99$	Yes	956	9.85×10^{4}	
^{242}Pu	3.75×10^{5}	α, SF = 5.5×10^{-4}	Yes	486	1.86	
^{241}Am	432.7	α	Yes	296	1×10^{3}	1310
^{242m}Am	141.1	[b]IT = 99.55 $\alpha = 0.45$	Yes			
^{243}Am	7.37×10^{3}	α, SF = 2.2×10^{-8}	Yes	83.8	16.7	
^{242}Cm	0.44	α, SF = 6.8×10^{-6}	Yes	6.2		
^{244}Cm	18.1	α, SF = 1.3×10^{-4}	Yes	24		1490

[a] SF: spontaneous fission.
[b] IT: isomeric transition.

Pu, for up to 10^6 years. Americium has a predominant position among the MAs between 10^2 and 10^5 years, Np after 10^5 years and Cm before 10^4 years [5]. Uranium is predominant after 10^6 years.

For long term storage or disposal of spent fuel or HLW, the potential release of activity to the biosphere can be evaluated as a consequence of degradation of engineering barriers and transfer of radionuclides through the geological environment. With account taken of the solubility of the actinides, the permissible concentrations in water and solids, the actinide hazard factors in spent fuel/HLW may be placed in the following order [5]:

$$^{241}Am \rightarrow {}^{240}Pu \rightarrow {}^{239}Pu \rightarrow {}^{243}Am \rightarrow {}^{242}Pu \rightarrow {}^{237}Np \rightarrow Cm \text{ (in spent fuel)}$$

$$^{241}Am \rightarrow {}^{243}Am \rightarrow {}^{240}Pu \rightarrow {}^{239}Pu \rightarrow {}^{237}Np \rightarrow Cm \text{ (in HLW)}$$

TABLE 2. SOURCE TERM (ACTINIDES AND LONG LIVED FISSION PRODUCTS) AND ITS COMPONENTS: VARIATION WITH TIME (Sv/TW·h(e)) [5]

Time (a)	10^2	10^3	10^4	10^5	10^6	10^7
Total (Sv/TW·h(e))	1.1×10^9	3.1×10^8	7.7×10^7	4.2×10^6	5.2×10^5	1.4×10^5
Components (%)						
RepU	—	—	0.1	6	29	79
Pu	85	90	97	88	50	17
Np	—	—	—	1.3	13	3
Am	10	9.2	2.5	2.7	6.8	1.4
Cm	0.4	0.3	0.4	—	—	—
FP	4.2	6×10^{-4}	2.4×10^{-3}	3.2×10^{-2}	9.6×10^{-2}	1.4×10^{-1}

In the late 1970s, vast R&D programmes to increase fuel burnup were initiated in several countries in order to reduce uranium and separative work requirements, fuel cycle cost and the amount of discharged spent fuel. At present, the reduction in the amount of discharged spent fuel and the nuclear fuel cycle costs is seen as a major result of burnup extension programmes. The historical design batch average burnup of LWRs, around 30 MW·d/kg U, has been increased now up to 45–50 MW·d/kg U for PWRs and 40–45 MW·d/kg U for BWRs. Economic incentives exist for extending burnup even further, to at least a batch average of about 60 MW·d/kg U.

However, an increase in burnup has also resulted in an increase in specific actinide content and alpha activity in spent fuel or MAs in HLW. The growth of actinide's alpha radioactivity is faster than linear, also the contribution of each actinide changes with increase of burnup (Table 3). This should be taken into

TABLE 3. ALPHA ACTIVITY OF SPENT FUEL FROM THERMAL REACTORS (Ci/t HM) AS A FUNCTION OF BURNUP [6]

Burnup (MW·d/kg HM)	Pu total	^{238}Pu	^{237}Np	^{241}Am	^{244}Cm	Total
33.00	3213	2360	0.15	1310	1 490	6 050
40.00	4430	3480	0.40	1500	2 700	8 750
50.00	6720	5740	0.51	1950	5 700	14 300
60.00	9270	8220	0.64	2430	11 600	23 500

account in the safety assessment of spent fuel, HLW storage or disposal facilities.

3.2. Requirements for the reactor model

The requirements for the reactor model can be summarized as follows:

(a) Since the objective of VISTA is to predict the global amount of actinides in the fuels discharged from all civilian power reactors, it is not required to evaluate the actinide inventory for each individual reactor. Therefore, all power reactors are grouped into seven reactor types. These are PWRs, BWRs, PHWRs, GCRs, AGRs, WWERs and RBMKs.
(b) Considering the future capability to adapt the burnup model to the IAEA's code, the reactor model should be simple. However, it should be accurate enough compared with other verified codes.
(c) Experts, studying the global nuclear materials flows, want to have on their personal computers a user-friendly tool to quickly test their hypotheses.
(d) The model must be in accordance with the principles of nuclear physics and reflect the realities in the nuclear industry. Data to be inputted by users should be kept to a minimum, unlike for the more sophisticated models.
(e) Formulas giving the isotopic composition of the different nuclear fuels, both before and after irradiation, must be incorporated into the program. This will allow the size of the program to be reduced, hence speeding up the calculation and also limiting the amount of data to be inputted in order to evaluate the nuclear material flows and isotopic compositions.

A new computer program has been developed using Microsoft Excel software to meet the above requirements. The selection of MS Excel is based on its having several reporting options, such as tables and charts, as a built-in feature. The results can easily be used to make a statistical study for comparison of different options. The new program is called Calculation of Actinide INventory (CAIN).

3.3. CAIN

3.3.1. The Bateman equation

The CAIN code uses the Bateman equation to make fuel depletion calculations, due to its sufficient accuracy and simplicity. It is a theoretical burnup solution of a point reactor with one group neutron energy.

Simple estimates of transmutation and decay rates of nuclear materials can be made by solving the Bateman equation for actinides, subject to a neutron flux. The Bateman equation represents an analytical solution to the general transmutation and radioactive decay problem. A simplified version of the Bateman equation for concentration N_i of isotope i in a material subject to a particular flux can be written as:

$$\frac{dN_i}{dt} = -\sum_{j \neq i} \left[\lambda_{ji}^{d} + \bar{\sigma}_{ji}^{tr} \bar{\varphi} \right] N_i + \sum_{j \neq i} \left[\lambda_{ij}^{d} + \bar{\sigma}_{ij}^{tr} \bar{\varphi} \right] N_j$$

where N_i is the concentration of the isotope i, λ is the decay constant, σ is the cross-section and φ is the neutron flux.

3.4. Cross-sections used and reactor characteristics

The CAIN code requires various inputs to evaluate the actinide inventory in the discharged fuel. The most important data sets are the neutron cross-sections and reactor constants such as specific power or neutron flux for the seven reactor types selected. The sources of the cross-sections are described below:

(a) PWR, BWR and PHWR cross-section data were taken from the ORIGEN code library [9].
(b) RBMK cross-sections were calculated by a Japanese consultant using a similar code to WIMS.
(c) AGR cross-sections were calculated by a Canadian consultant using the WIMS code. The Am and Cm cross-sections were copied from RBMK data, because their neutron spectra are similar.
(d) The GCR (0.71% enrichment) cross-sections were linearly extrapolated from the above AGR cross-sections of 1.6 and 2.6% enrichment at a range of 0–4 GW·d/t. The Am and Cm cross-sections were copied from PHWR data, because both types of reactor are using natural uranium fuel.

The reactor types that are included in VISTA have some characteristics which affect the calculation of the isotopic composition of spent fuel. These parameters differ from reactor to reactor, but for the purpose of this study average standard values for each type were used. These parameters and the values of them used in VISTA are given in Table 4.

TABLE 4. REACTOR CHARACTERISTICS

Reactor type	Specific power (kW/kg)	Reference neutron flux ($n/cm^2 \cdot s$)	Reference enrichment (%)
PWR	37.50	2.990×10^{14}	4.00
BWR	25.90	1.997×10^{14}	4.00
PHWR	18.80	1.850×10^{14}	0.71
RBMK	15.75	1.480×10^{14}	1.80
AGR	10.90	0.875×10^{14}	2.60
GCR	3.33	2.660×10^{14}	0.71
WWER	45.80	3.747×10^{14}	4.36

3.5. Validation and benchmarking

The CAIN code has been validated against three different data sets which are available. Nuclide densities calculated by CAIN are compared with those given by the ORIGEN code results (Section 3.5.1) and with available measurements (Section 3.5.2). The CAIN calculation results are also compared with the results of the WIMS neutronic design code [10] for the PHWR, RBMK, AGR and GCR reactor types.

3.5.1. Comparison of CAIN with ORIGEN

The CAIN burnup calculation was compared with the ORIGEN code results for three reactor types: PWRs, BWRs and PHWRs. Since the cross-sections and neutron fluxes are identical for both CAIN and ORIGEN, these two results should be identical except for the small influence due to the assumptions made in Section 3.1.

Table 5 gives details of a comparison between the two models. The CAIN code agrees very well with the ORIGEN code, with the errors being 1–2% for the PWR and BWR cases. For the PHWR case, the CAIN code also agrees very well with the ORIGEN code within a 1–3% error, except for ^{238}Pu. The ^{238}Pu concentration calculated by ORIGEN is 14% smaller, which may be due to an inappropriately large ^{237}U cross-section in ORIGEN.

On the basis of these results, it is concluded that the CAIN model gives essentially identical results to the ORIGEN model when the same cross-section set is used.

TABLE 5. COMPARISON BETWEEN CAIN AND ORIGEN

Isotope	PWR with 4.0% at 45 GW·d/t			BWR with 4.0% at 45 GW·d/t			PHWR with 4.0% at 45 GW·d/t		
	CAIN (wt%)	ORIGEN (wt%)	Error (%)	CAIN (wt%)	ORIGEN (wt%)	Error (%)	CAIN (wt%)	ORIGEN (wt%)	Error (%)
^{235}U	0.6798	0.6798	0.0	0.6251	0.6251	0.0	0.2323	0.2326	–0.1
^{236}U	0.5243	0.5244	0.0	0.5276	0.5277	0.0	0.0717	0.07183	–0.2
^{238}U	93.0500	93.05	0.0	93.12	93.12	0.0	98.57	98.57	0.0
^{238}Pu	0.0245	0.02404	2.0	0.02698	0.02663	1.3	0.000343	0.000302	13.3
^{239}Pu	0.5114	0.5183	–1.3	0.4855	0.4916	–1.2	0.2671	0.2701	–1.1
^{240}Pu	0.2635	0.2651	–0.6	0.2553	0.2568	–0.6	0.09781	0.09776	0.1
^{241}Pu	0.1506	0.1513	–0.5	0.1462	0.1468	–0.4	0.01871	0.01867	0.2
^{242}Pu	0.0642	0.06511	–1.4	0.06656	0.06785	–1.9	0.004138	0.004132	0.1
^{237}Np	0.0687	0.06859	0.2	0.07025	0.07018	0.1	0.002652	0.002576	3.0
^{241}Am	0.004765	0.004772	–0.1	0.007045	0.007037	0.1	0.0001878	0.0001871	0.4
^{243}Am	0.0144	0.01453	–0.9	0.01536	0.01559	–1.5	0.0001305	0.0001297	0.6
^{242}Cm	0.001808	0.001801	0.4	0.002023	0.002015	0.4	0.0000568	0.0000551	3.1
^{244}Cm	0.004771	0.004832	–1.3	0.005194	0.005264	–1.3	0.000011	0.0000111	–0.9
Total	95.36	95.37	0.0	95.35	95.37	0.0	99.27	99.27	0.0

3.5.2. Comparison of CAIN with measurements

There is one report [11] that compiles the results of open data on actinide measurements of discharged fuels from both PWRs and BWRs. This report contains Post-Irradiation Experiment (PIE) data for the spent fuels from seven PWRs and six BWRs. Figure 6 presents charts for a comparison of CAIN results with measured data for ^{235}U and total Pu. The results for the individual Pu nuclides were also compared with measured values and found to be quite satisfactory.

In order to verify the CAIN model, two calculations were performed for the 3% enrichment of PWR and BWR fuel. On the basis of the report [11], the actual enrichments vary from 2.5 to 3.4%, which would affect the result slightly. However, in general, the CAIN model agrees very well with the measurements. As can be seen, there is very little difference between PWRs and BWRs. Because of the similar neutron spectra for both reactors, the isotopic compositions would become similar. Usually, PWRs use smaller fuel rods, and this will cause a slightly higher amount of plutonium due to the larger resonance absorption for the same enrichment.

Regarding the other actinides such as Np, Am and Cm, there is another report available [12]. The comparison between the CAIN model and the PIE data for ^{241}Am and ^{243}Am is shown in Fig. 7. The CAIN model also agrees with measurements for these actinides. The ^{237}Np, ^{242}Cm and ^{244}Cm nuclides were also compared with the results from the reported measurements, and sufficient accuracy was obtained.

As for the content of ^{241}Pu or ^{242}Cm, the measurement may include the effects of the decay; meanwhile the calculation does not include the effect of the decay of ^{241}Pu of half-life 14.4 years and of ^{242}Cm of half-life 0.446 years. This means that the CAIN calculation assumes a zero cooling time. If the PIE results have been obtained after a short cooling time, this would not affect the results.

4. INPUT PARAMETERS AND SCENARIOS

The VISTA input parameters contain data sets that are a combination of actual historical data and estimates for future projections. The simulation system database contains historical data from the beginning of commercial nuclear activities. The historical data have been retrieved from the actual reported data wherever possible. The data sources are usually well known databases such as the IAEA PRIS database [13]. The other data sources include reports that are generated by consultant companies such as

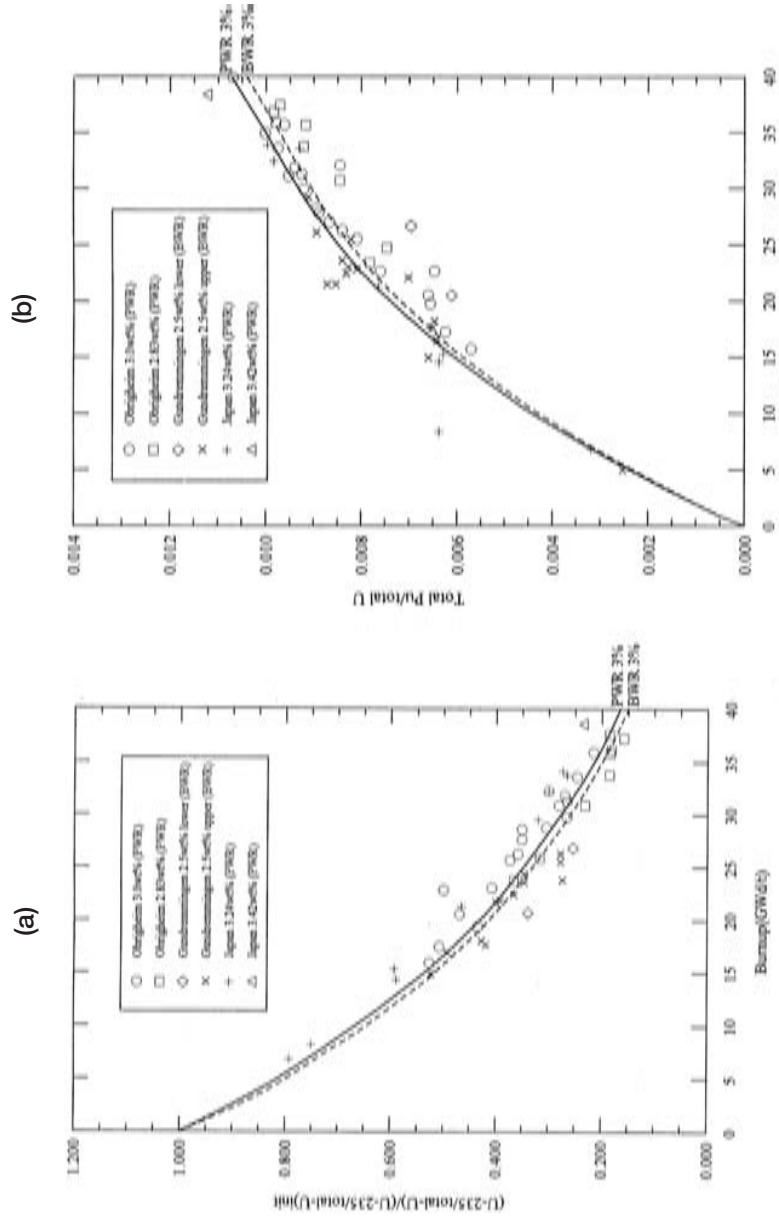

FIG. 6. Comparison of CAIN results with measurements: (a) ^{235}U, (b) total Pu.

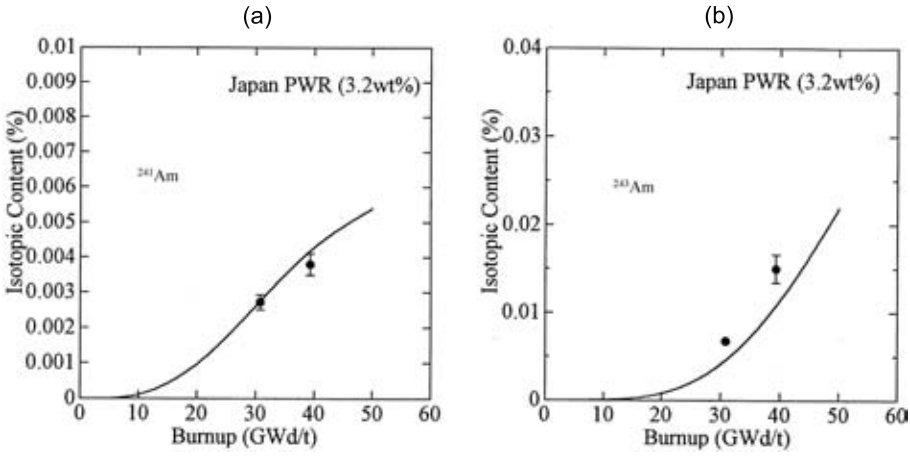

FIG. 7. Comparison of CAIN results with measurements: (a) ^{241}Am, (b) ^{243}Am.

NAC International's Nuclear Industry Status Report [14]. The data reported at IAEA conferences or in IAEA publications are also used in some cases.

The different reactor and fuel cycle characteristics and their evolution over the period up to 2050 were used in this simulation system. Although the raw database has reactor by reactor data for most of the input parameters, all the data sets used in this study are realistic mean values for each reactor type worldwide. They do not reflect any country specific characteristics, or the performance of specific reactors and fuel cycle facilities.

4.1. Nuclear power

The historical nuclear power data come from the IAEA PRIS database [13]. PRIS is one of the most comprehensive and commonly used databases on nuclear power plants and the experience of operating them worldwide. Future nuclear power projections are based on the IAEA Energy, Electricity and Nuclear Power Estimates for the Period up to 2030, 2004 Edition [15]. The estimates from this publication are based on many factors and reflect the most realistic situation for the period that it covers. The publication provides estimates until the year 2030. VISTA then extrapolates the estimates up to the year 2050 by the linear extrapolation technique.

The publication gives two variants for the evolution of nuclear power. One is the low variant and the other is the high variant. VISTA uses these two variants as well as their arithmetic averages. In VISTA, the low nuclear power

capacity case is called P0, the medium case is called P1 and the high case is called P2. The variation of the parameters is displayed in Fig. 8.

4.2. Reactor type mix

At present, there are a number of commercially available reactor types in the world. All these types are actually grouped into seven main groups based on their nuclear characteristics in order to reduce the data requirements in VISTA, as discussed in Section 3.2.

The share of each reactor type in the total nuclear reactor park is represented in this data set. The historical data come from the IAEA's PRIS database [13], and the future projection is based on the recommendations of consultants. At present, about 57% of the operating reactors are PWRs, 22% are BWRs, 5% are PHWRs, 3% are RBMKs, 2% are AGRs, 1% are GCRs and 9% are WWERs. Figure 9 displays the evolution of the shares of the different reactor types in the worldwide total.

4.3. LWR reprocessing scenario

Different countries have chosen different alternative fuel cycle options based on their specific policies and goals, taking into account the balance between their domestic energy resources and industrial capabilities. The choice also depends on the growth of nuclear electricity generation. For some

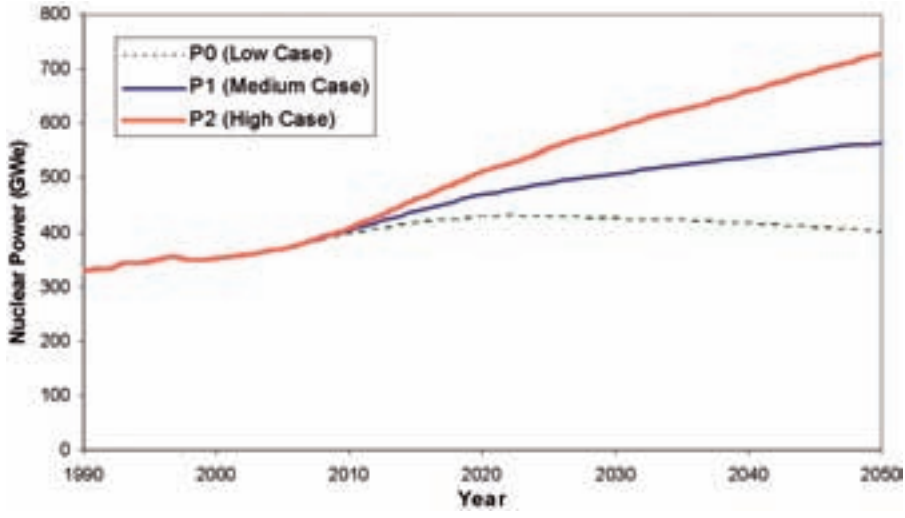

FIG. 8. The installed nuclear capacity in the world.

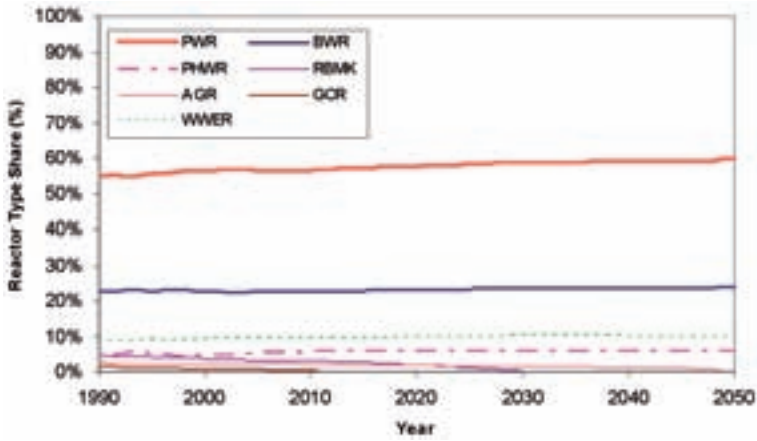

FIG. 9. Shares of reactor types in the worldwide total.

countries, there would be incentives to reprocess spent fuel to recover and recycle fissile nuclear materials as a means of reducing their requirement for natural uranium. The recycling option can also be chosen as a means to address issues related to management of the back end of the fuel cycle.

In view of the existing alternative fuel cycle options, VISTA has an input parameter to determine the amount of reprocessed material for each reactor type. The reprocessing ratio is defined as the ratio of the spent fuel to be reprocessed after a period of cooling time to the total spent fuel discharged in a given year. VISTA assumes that all GCR fuel is to be reprocessed. VISTA also assumes that the reprocessing ratios for the PWR and BWR types are the same, owing to the lack of data for the individual types. This study assumes only uranium fuel from natural material is reprocessed, although VISTA is capable of having multicycling options with the introduction of proper data.

VISTA has four different reprocessing scenarios in its database. The first scenario assumes that reprocessing will decrease in time and that there will be no reprocessing after the year 2030 (called the R0 case). The second scenario assumes that the current reprocessing ratio will be kept at a steady value throughout the period of calculation (called the R1 case). The third and the fourth scenarios assume that more of the spent fuel will be reprocessed in the future (called the R2 and R3 cases). The scenarios are presented in Fig. 10.

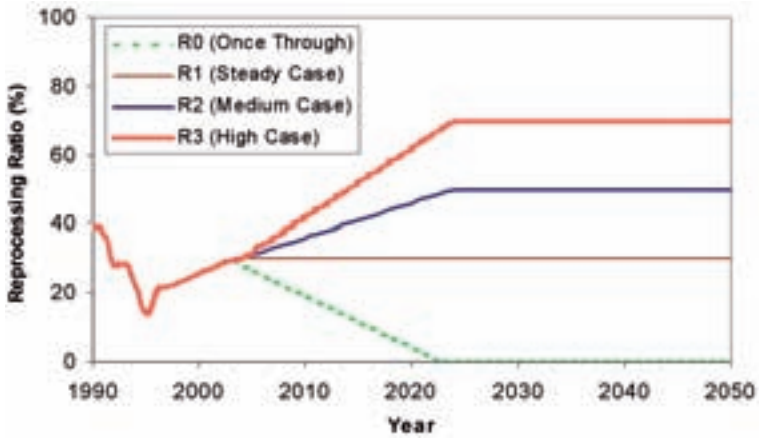

FIG. 10. Reprocessing ratios for PWRs and BWRs.

4.4. LWR MOX use scenarios

Reprocessed material has already been used by some countries in the form of MOX fuel since the 1980s. VISTA assumes that only PWR and BWR type reactors are using and will use the MOX fuel type. The MOX use ratio input parameter is defined as the ratio of the MOX fuel amount to the total fuel amount used in the reactor type to generate electricity. This assumption is made on the basis of a lack of data for individual types. The historical data are derived from the reported values of MOX fuel fabrication amounts in the fuel fabrication facilities.

VISTA has five MOX use scenarios in its database, starting from the low scenario (called the M0 case) to the very high scenario (called the M4 case). The variations in the MOX use ratios are displayed in Fig. 11.

4.5. Other input parameters

The average enrichment tails assay is another parameter that can be selected from different options. VISTA has three scenarios for average tails assay values. The high scenario is called the T0 case and assumes that the long term average value will be kept constant at its 2000 value. The case called T1 assumes that the average tails assay will drop to 0.25% in 2025 and then be constant at that level afterwards. The case called T2 assumes that the average

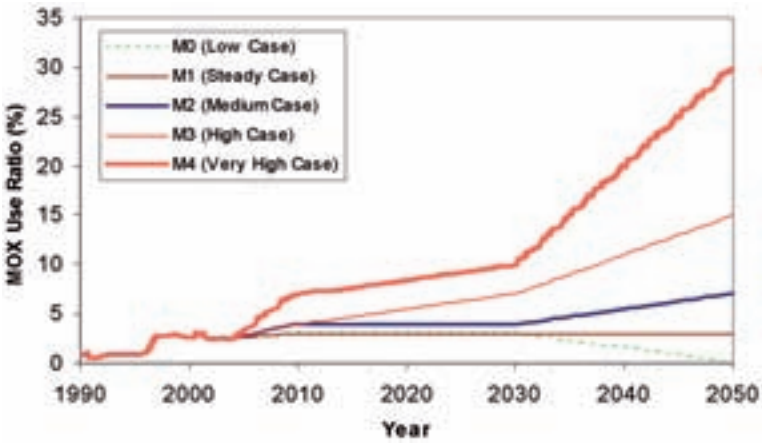

FIG. 11. Light water reactor MOX use ratios.

tails assay will drop to 0.20% in 2025 and then be constant afterwards. This reflects the latest rapid increase in natural uranium prices.

VISTA has other parameters, some of which are set to predefined values and can only be changed by changing the data. Average load factors, average thermal efficiencies, average discharge burnup levels, average initial fuel enrichments, average fuel residence times in the core, and process lead and lag times are among these parameters.

The load factor is defined as the ratio of the electricity generated in a given year to the total electricity to be generated if the facility is operated for a whole year at 100% capacity. Historical load factor values come from the IAEA PRIS database [13], and the future projection is based on the trends in operating experience and performance of the current reactor park. Figure 12 shows the average annual load factors and average enrichment tails assay values.

Average discharge burnup values are calculated from the reactor by reactor data that are provided in NAC International reports [14]. The burnup projection is based on the recommendations of consultants and reflects the latest developments in fuel performance and power reactor operating experience. Average initial enrichment values are calculated using the relation between discharge burnup and initial enrichment. This assumption is valid if the reactor fuel is used effectively. In some cases, for various reasons, the fuel is discharged before its nominal burnup value has been reached, but these are not significant amounts and do not change worldwide averages much. Discharge burnup and initial enrichment values are displayed in Fig. 13.

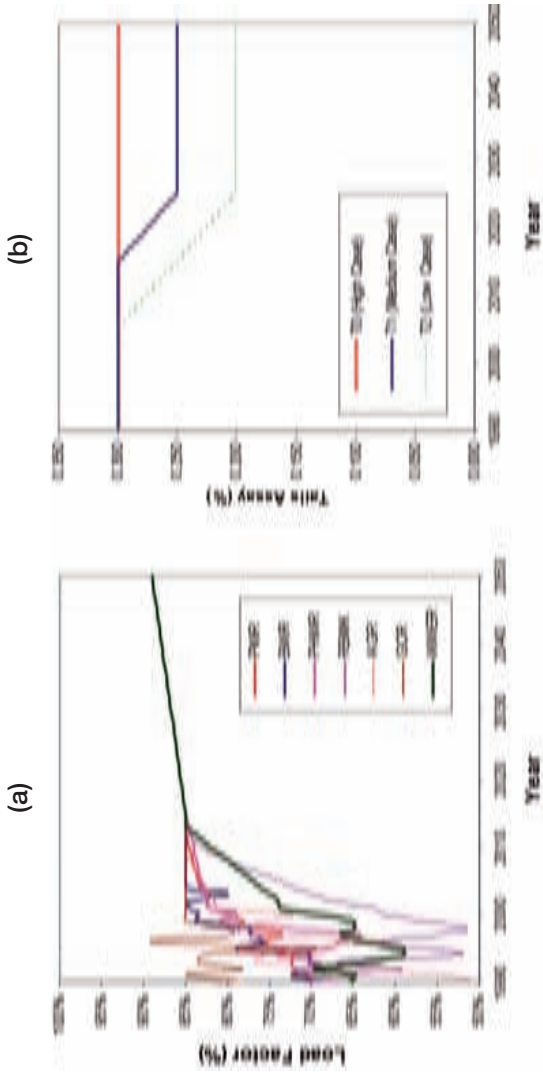

FIG. 12. (a) Average annual load factors and (b) average enrichment tails assay values.

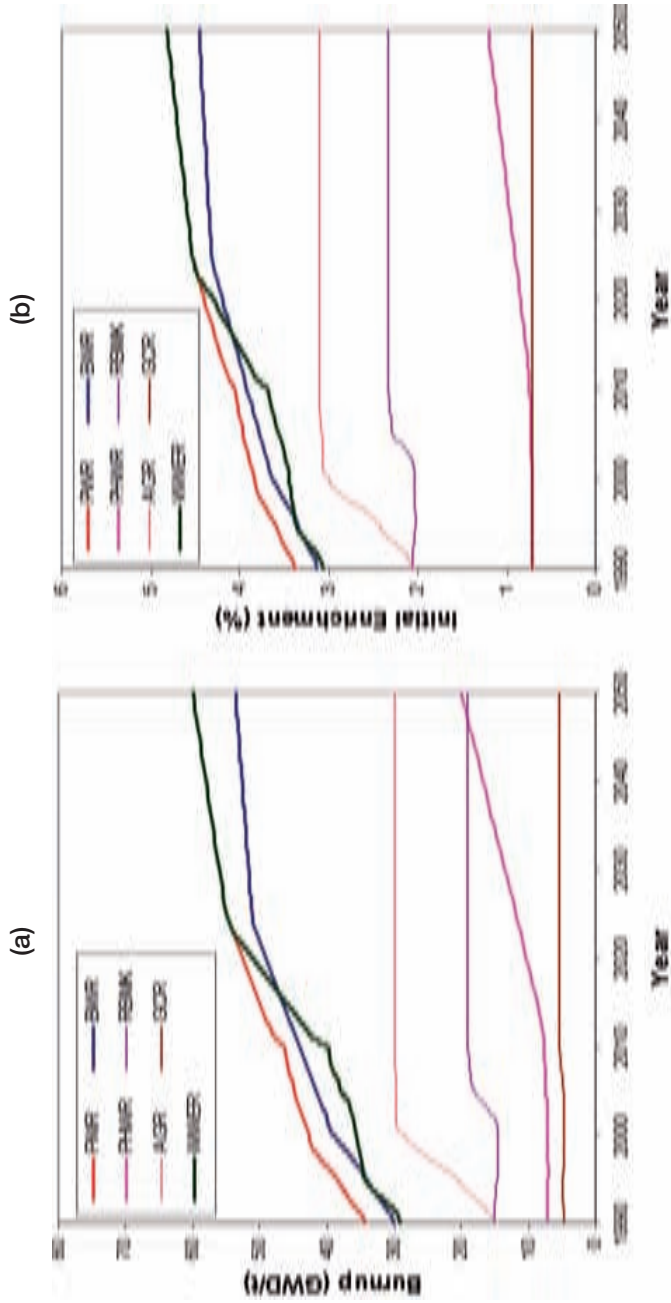

FIG. 13. (a) Average discharge burnups and (b) associated values of fuel initial enrichment.

In reality, all processes have some material losses. VISTA has input parameters for conversion, enrichment, fuel fabrication and reprocessing loss coefficients. However, owing to the lack of actual data, this study assumes that the process loss coefficients are zero (zero loss) for all processes.

All processes are assumed to occur in real time with no lags, except for the cooling period between time of discharge from the reactor and time of reprocessing and manufacturing of MOX fuel. For PWR and BWR spent uranium fuel, the cooling time is assumed to be six years and the MOX fuel manufacturing time is assumed to be one year.

5. EXAMPLE

5.1. Scenario selection

In order to illustrate the use of the VISTA simulation system, an example scenario data set has been selected. An attempt has been made to select realistic cases for each of the parameters in the scenario category. The selection of the example scenario is only based on the consultants' recommendations and does not reflect any official declaration from the IAEA or its Member States. It should also be noted that this is a theoretical calculation and might not reflect technically or practically possible options.

The selected example scenario data set is as follows:

(a) *Nuclear power:* The medium case has been selected for the example scenario (P1 in Fig. 8).
(b) *LWR reprocessing ratio:* The steady case has been selected for this example (R1 in Fig. 10).
(c) *LWR MOX use ratio:* The steady case has been selected for the example study (M1 in Fig. 11).
(d) *Enrichment tails assay:* The low case has been selected (T2 in Fig. 12).
(e) All the other parameters use the basic data set that is described in Section 4 of this paper.

5.2. Scenario results

Several results have been selected out of a complete result set from the VISTA simulation system. These results are believed to provide a good overview of the capabilities of VISTA. The results presented below focus on the material and service requirements for the main stages of the nuclear fuel

cycle and the total plutonium discharge from the reactors. More results from VISTA for different scenarios were presented at this conference.

Nuclear electricity generation will increase more than the installed nuclear capacity in the near future, due to the better performance of power plants. Average annual load factors have been increasing for almost all of the reactor types in the world for several years. This trend is expected to continue. The result is shown in Fig. 14.

Estimated annual and cumulative natural uranium requirements are shown in Fig. 15. The annual natural uranium requirement will increase until 2010 and then reach a plateau of about 70 000 t U. After 2030 the requirement will start increasing again.

The conversion requirement shows the same progress as the natural uranium requirement. The results show all the conversion requirements from natural uranium to UF_6, to UO_2 or uranium metal for different types of reactor fuels. Figure 16 shows the results for annual and cumulative conversion requirements.

Enrichment is required to increase the fissile content of UF_6. The annual enrichment requirement is expected to increase significantly until 2025 due to the increasing share of the reactor types that use enriched uranium for their fuel (mainly LWRs). After 2025, this increase will be slower because the share of LWRs will approach its maximum value. The result is shown in Fig. 17.

The annual fresh fuel requirement is expected to increase until 2010 and then decrease (Fig. 18). This decrease is explained by the increasing discharge

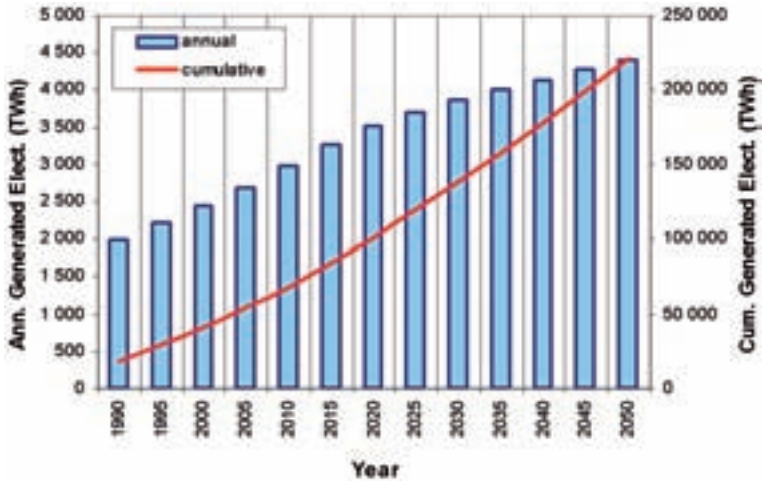

FIG. 14. Annual and cumulative nuclear electricity production (the P1–R1–M1 case).

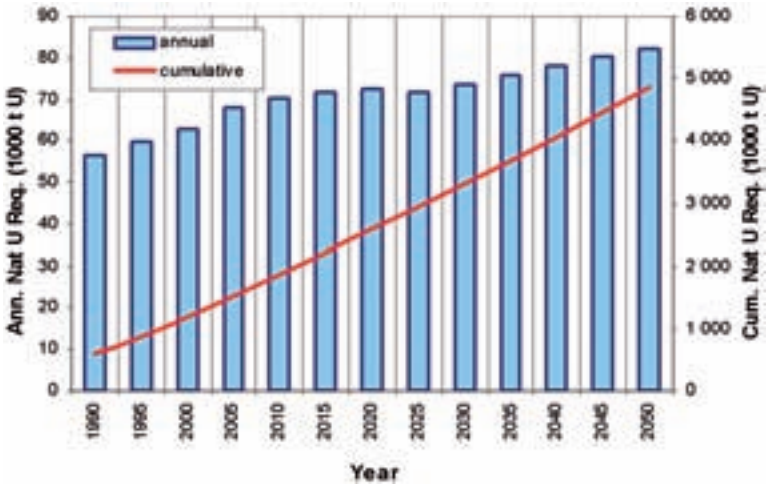

FIG. 15. Annual and cumulative natural uranium requirements (the P1–R1–M1 case).

burnup value and the decreasing share of GCRs, AGRs and RBMKs, which basically require more fuel in terms of tonnage than the PWR, BWR and WWER types.

The amount of spent fuel discharged is similar to the fresh fuel requirement, with some delay. The result is displayed in Fig. 19.

Reprocessing requirements are heavily dependent on the recycling strategy adopted by a country. For this example, the annual reprocessing will

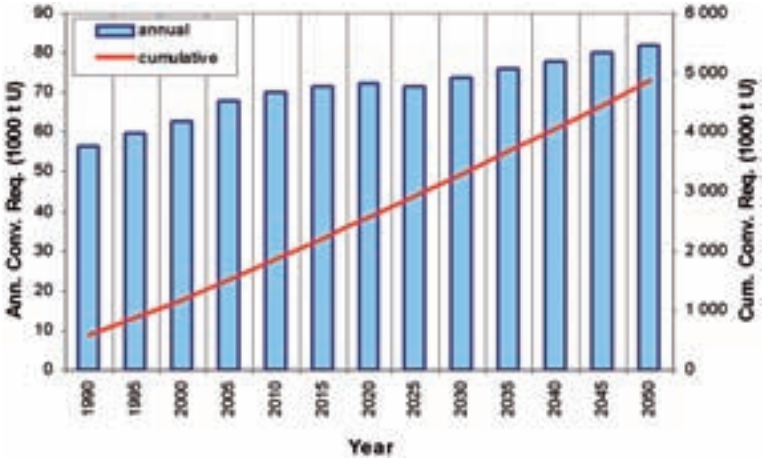

FIG. 16. Annual and cumulative conversion requirements (the P1–R1–M1 case).

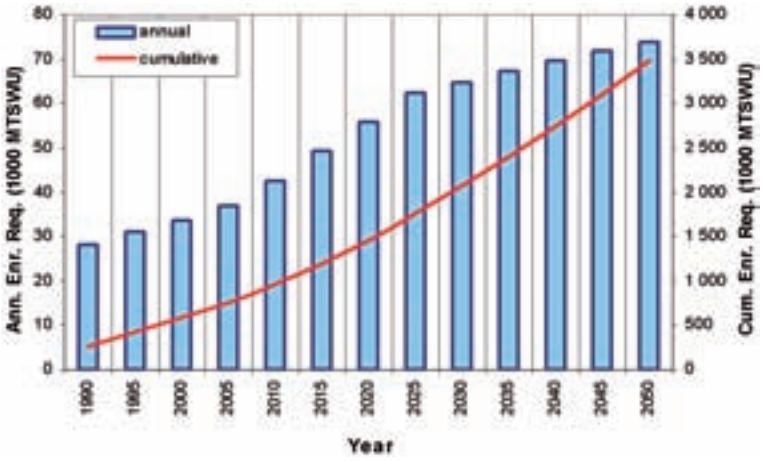

FIG. 17. Annual and cumulative enrichment requirements (the P1–R1–M1 case).

decrease until the year 2020 and then steadily increase (Fig. 20). This is explained by the phase-out of the GCR and AGR reactors in time.

The amount of spent fuel stored in interim storage facilities or reactor pools is displayed in Fig. 21. The annual spent fuel storage requirements will increase until 2015 due to the reduced amount of reprocessing. But after that, with the recovery of the reprocessed amount and due to the decreasing amount

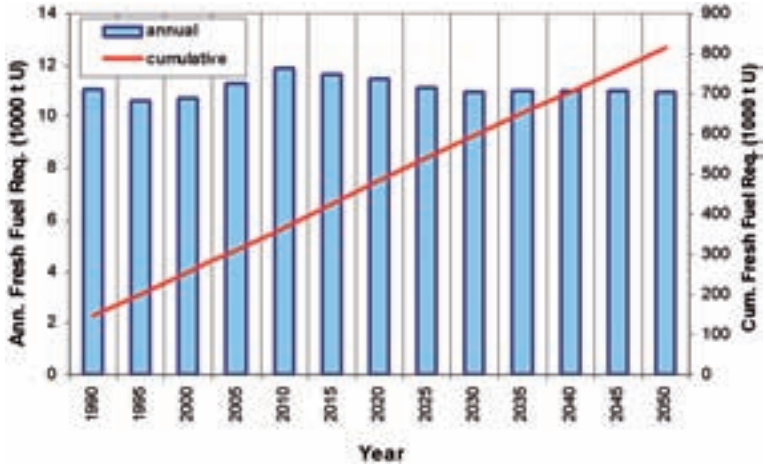

FIG. 18. Annual and cumulative fresh fuel requirements (the P1–R1–M1 case).

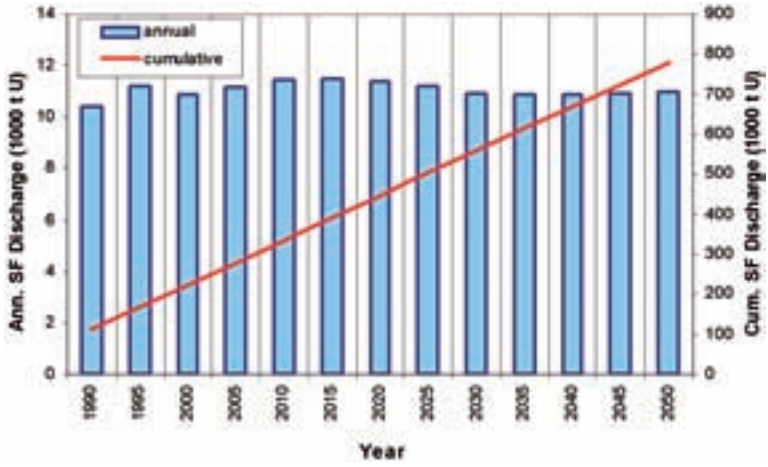

FIG. 19. Annual and cumulative spent fuel discharge amounts (the P1–R1–M1 case).

of spent fuel discharged, the annual spent fuel storage requirement will decrease slightly.

Figure 22 shows the amount of total plutonium that will be discharged from commercial nuclear power plants. The content of plutonium spent fuel increases with the increasing discharge burnup. This effect can be seen in the figure. Although the amount of spent fuel discharged annually does not increase after 2015, the amount of plutonium in discharged spent fuel will continue to increase with the increasing burnup level.

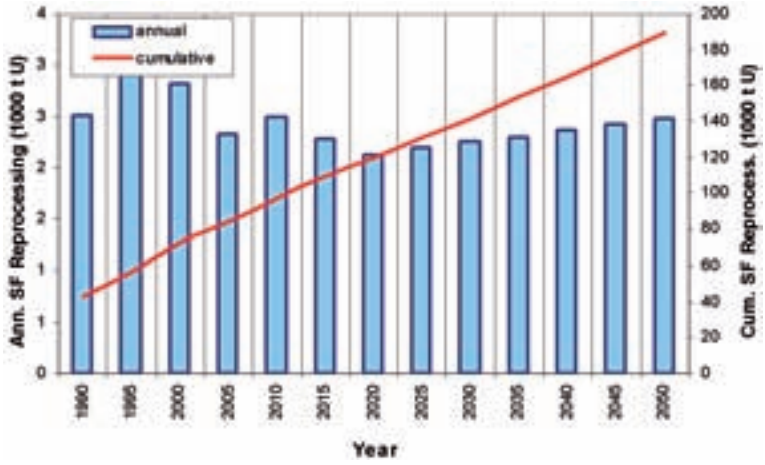

FIG. 20. Annual and cumulative reprocessed spent fuel amounts (the P1–R1–M1 case).

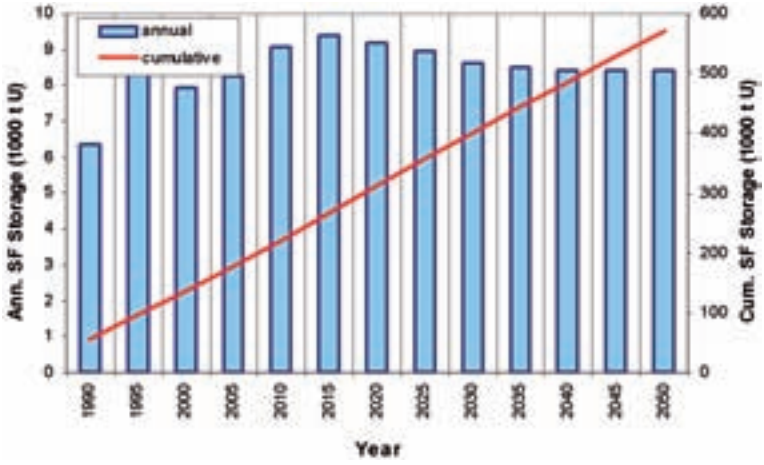

FIG. 21. Annual and cumulative stored spent fuel amounts (excluding storage for reprocessing) (the P1–R1–M1 case).

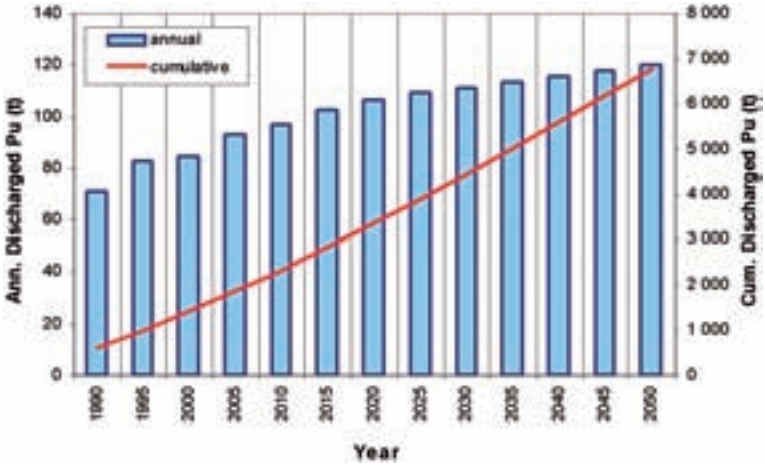

FIG. 22. Annual and cumulative total plutonium discharges from reactors (the P1–R1–M1 case).

6. SUMMARY AND RECOMMENDATIONS

In general, we can summarize VISTA as:

(a) A series of algorithms linking electricity generation and nuclear fuel operating data to nuclear material requirements and spent fuel arisings.
(b) A simplification to modelling of nuclear fuel cycle requirements, by not reflecting the actual fuelling pattern reactor by reactor. VISTA does not take into account commissioning and shutdown schedules for each reactor.
(c) A method to calculate the average requirements of different reactor types. These correspond to a given level of electricity generation. The results include estimates of uranium and fuel cycle service requirements up to the year 2050 for alternative fuel cycle strategies.
(d) A system capable of simulating the evolution of key fuel cycle features for given parameters, for example, load factors, fuel burnup and enrichment, tails assays and reprocessing fractions. All these can be changed from reactor type to reactor type, and from time period to time period.
(e) A tool to track the individual actinide content at any stage of the nuclear fuel cycle, including in the discharged and stored spent fuel even after a period of time. Therefore, it can be used to evaluate different fuel cycle options in terms of their effectiveness in reducing the radiotoxicity of spent fuel or reducing the natural nuclear material requirements.

In its current status, VISTA is capable of evaluating reactor parks with commercially existing reactor and fuel types. If the study is to be done for about 30 years or less, then this is acceptable. In order to make an analysis beyond 30 years, new fuel and reactor types should be introduced into the system. This work is actually in progress: the inclusion of the most probable fuel and reactor types is being investigated.

One of the purposes of VISTA is to evaluate the radiotoxic impacts of the different nuclear fuel cycle options. Currently, radiotoxicity can be calculated using isotopic contents from VISTA and their individual radiotoxic contributions. A direct calculation of the radiotoxicity of fuel cycles is not available in VISTA. One of the improvements of the system will be to introduce the calculation of the radioactivity and radiotoxicity of the spent fuel and HLWs from different fuel cycle options directly in VISTA.

An economic analysis and a further environmental analysis are outside the scope of the project at present. However, there could be additions to the system to also include such analyses in future.

A limited version of the VISTA simulation system is available on the Integrated Nuclear Fuel Cycle Information Systems web site (http://www-nfcis.iaea.org/).

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FORECAST OF FISSILE MATERIAL INVENTORIES IN THE BACK END OF THE NUCLEAR FUEL CYCLE

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Abstract

Under the assumption that the current plutonium recycle in light water reactors continues up to 2050, the accumulation of transuranic materials (plutonium and minor actinides (MAs)) is predicted by the IAEA computer model, VISTA, with three key parameters: nuclear power capacity, the reprocessing ratio of spent fuel and mixed oxide fuel utilization. The related calculations have clarified the stockpile of separated plutonium, plutonium contained in stored spent fuel and MA generation up to 2050. The calculated results have been compared with the actual amounts of plutonium accumulation reported by nine IAEA Member States. Plutonium, as well as MAs contained in any form, will increasingly accumulate in the wake of nuclear fuel cycle activities worldwide, which will require effective measures to reduce this accumulation. Finally, an overview is given of the status of the programmes of major countries for the disposition of transuranic material.

1. INTRODUCTION

Following substantial cancellations in the development of the closed nuclear fuel cycle incorporating fast breeder reactors (FBRs) around 1990, the nuclear fuel cycle strategy was separated into three options:

(1) Plutonium use in light water reactors (LWRs);

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(2) No decision on the fuel cycle strategy, the so-called 'wait and see' policy;
(3) Direct disposal of spent fuel.

In the new realities that emerged after this time, the inventory of separated plutonium has increased in the wake of continued reprocessing activities. As of the end of 2003, the separated plutonium inventory has been estimated as approximately 234 t HM worldwide. Meanwhile, as development of the FBR-Pu recycle was cancelled for the time being, plutonium has been consumed exclusively in LWRs. Plutonium recycle in LWRs (the LWR-Pu recycle) currently carried out in several countries is a process such that plutonium is recycled only once (single recycle) as mixed oxide (MOX) fuel, which is, after irradiation, either subject to interim storage or reprocessed without further recycle. Here, LWRs, which use MOX fuel, are conventional power reactors such as pressurized water reactors (PWRs), plutonium boiling water reactors (BWRs) and WWERs. There are several other innovative concepts that use advanced LWRs, such as high moderation LWRs and low moderation LWRs [1]. As far as the ongoing LWR-Pu recycle is continued, it is obvious that the amounts of fissile materials in spent fuel discharged from nuclear power reactors will increase continuously. Such accumulation of nuclear material entails various issues damaging to the environmental soundness, non-proliferation, safety and economy of nuclear energy. Therefore, the world nuclear energy sector seeks the ultimate solution for sustainability of nuclear energy by development of advanced technologies [2–4]. This paper provides information for policy making and strategy planning with regard to fissile material (or transuranic materials) stockpiles in the back end of the nuclear fuel cycle, which are generated in the LWR-Pu recycle by conventional nuclear fuel cycle activities worldwide in the past, present and future. For this purpose, the amounts of transuranic materials born in spent fuel and separated in the period up to 2050 were calculated by the IAEA VISTA code [5]. These results were compared with the practical amounts reported in the IAEA Guidelines for the Management of Plutonium [6] and national programmes showing how to address the problem of increasing amounts of transuranic materials.

2. CALCULATION OF FISSILE MATERIALS WITH THE VISTA CODE

2.1. Outline of the VISTA code

The Nuclear Fuel Cycle Simulation System, the so-called VISTA model, was developed for the International Symposium on Nuclear Fuel Cycle and

Reactor Strategy: Adjusting to New Realities [7] held in 1997 in order to estimate the long term nuclear fuel cycle service and material requirements as well as annual arisings worldwide. Later, the VISTA model was updated and modified by one of the present authors [8] to calculate the requirements for two back end of the fuel cycle options (direct disposal and recycling).

A detailed description of the VISTA code is given by Ceyhan in the preceding paper of these proceedings [9]: the significant points of this code are briefly as follows. This model calculates the equilibrium core with two different fuel types, in which all the input and output parameters are on an annual basis. The inputs consist of strategy parameters, and fuel and process parameters. The strategy parameters are ones such as nuclear power capacity, reactor type, MOX use ratio, reprocessing ratio over all discharged spent fuel, and average load factors. The fuel and process parameters are average discharge burnup, initial enrichment, average tail assay, process lag time, process loss coefficient and thermal efficiency. The VISTA model is connected with a reactor physics model, CAIN, which was developed by the IAEA to calculate the isotopic composition of spent fuel discharged from individual reactors of seven types: PWRs, BWRs, pressurized heavy water reactors (PHWRs), RBMKs, advanced gas cooled reactors (AGRs), gas cooled reactors (GCRs) and WWERs. It has 14 decay chains and cross-sections for ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm and ^{244}Cm. The CAIN model uses one-group cross-sections for fresh fuel, mainly from the ORIGEN library. The output of the VISTA calculation provides various requirements for natural uranium, conversion, enrichment, fuel fabrication, reprocessing, and tracking of spent fuel arisings and individual nuclides.

2.2. Input parameters and their calculation

Three key input parameters, such as the nuclear power change, reprocessing ratio and fraction of power reactors using MOX fuel, were selected for calculation. With respect to the nuclear power change, the IAEA proposed two scenarios for nuclear power capacities in the period up to 2030 [10], which are designated the low power case (P0) and the high power case (P2). A medium power case (P1) was created by interpolating those two nuclear capacities. The nuclear power capacity scenarios up to 2050 are presented in Fig. 1, where the capacities were extrapolated up to 2050. The nuclear power capacity of P0 varies from 353 GW(e) in 2000 to 400 GW(e) in 2050 (low case), showing a small peak around 423 GW(e) in 2030, the P1 case changes from 353 GW(e) in 2000 to 565 GW(e) in 2050 (medium case), and the P2 case increases from 353 GW(e) in 2000 through 592 GW(e) in 2030 up to 730 GW(e) in 2050 (high case).

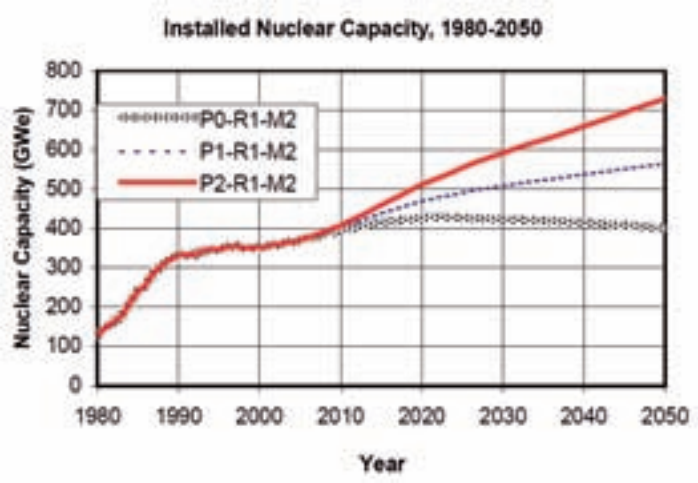

FIG. 1. Scenarios for nuclear power capacity in the period up to 2050 [10].

The reprocessing ratio is defined as the ratio of the amount of spent fuel to be reprocessed after a cooling period to the total amount of spent fuel discharged from nuclear power plants. As shown in Fig. 2, three variants of the reprocessing ratio were selected; the ratio before 2003 was based on the NAC International data [11], and thereafter they vary, one reaches 0% in 2022

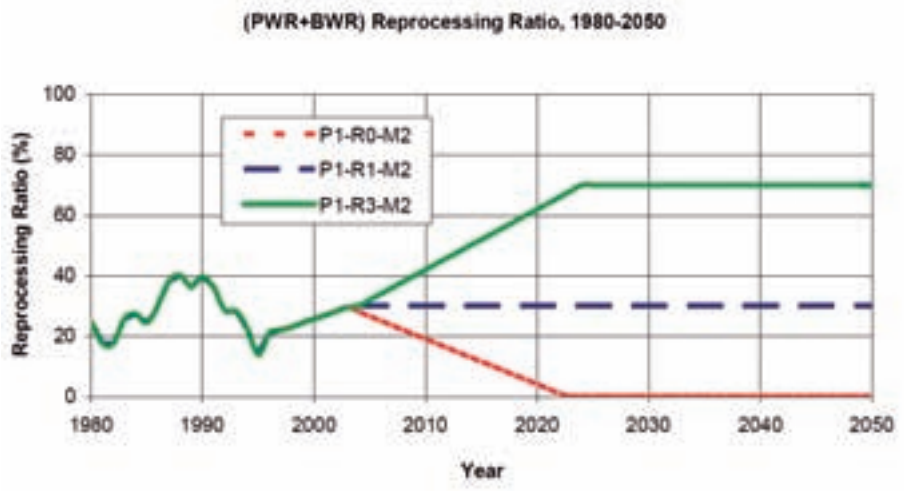

FIG. 2. The three variants of the reprocessing ratio used in calculations.

(simulating no reprocessing in the future, case R0), another, 30% (simulating the same situation as the present case, R1) and the other, 70%. The MOX fuel ratio is defined as the capacity of electricity generation by burning MOX fuel to the total capacity of nuclear power generation. The ratio exclusively used in this paper (designated M2) is characterized as follows: before 2003, it was cited from the NAC data (for instance, it was 2.60% in 2003). Thereafter, it increases to 4% in 2010 and then maintains 4% until 2030. In the period 2030–2050, it increases continuously up to 7%.

Other input parameters, such as load factor, reactor share, burnup change and fuel enrichment of individual reactors in the reactor mix, are described in detail in the report by Ceyhan [9].

3. IAEA GUIDELINES FOR MANAGEMENT OF PLUTONIUM

Following informal meetings attended by nine IAEA Member States (Belgium, China, France, Germany, Japan, the Russian Federation, Switzerland, the United Kingdom (UK) and the United States of America (USA)) in 1992–1993, these Member States agreed to establish guidelines for the management of plutonium. These guidelines are intended to provide an internationally accepted framework for the responsible management of plutonium by the Governments in all peaceful nuclear activities. Their key objective is to bring transparency to each individual State's management of plutonium. Pursuant to these guidelines, these Member States have reported, each year, their policies regarding management of plutonium, including their holdings of separated civil plutonium and quantities of plutonium in fuel forms as well as their fuel cycle policy in the interests of transparency, and these have been published by the IAEA [6]. This publication, INFCIRC/549, has shed light on civil plutonium management in the nine Member States from the end of 1996.

The INFCIRC/549 series provided information on the plutonium inventory included in spent fuel and separated plutonium inventories through the period from 1996 to 2003. The reported plutonium inventories are listed in Table 1. The separated plutonium inventory, which increased from about 160 t HM to 234 t HM in this period, neither includes the ex-military plutonium inventory (the USA voluntarily transferred some ex-military plutonium as part of safeguards) nor any plutonium stockpiles in countries other than these Member States.

TABLE 1. PLUTONIUM HOLDINGS IN THE NINE COUNTRIES REPORTED IN INFCIRC/549

Country	Separated plutonium (or in unirradiated MOX)								Plutonium in spent fuel							
	1996	1997	1998	1999	2000	2001	2002	2003	1996	1997	1998	1999	2000	2001	2002	2003
Belgium	2.7	2.8	3.8	3.9	2.7	2.9	3.4	3.5	12	14	16	17	18	20	22	23.4
France	65.4	72.3	75.9	81.2	82.7	80.5	79.9	78.6	153	156	158.8	159.8	164.4	173.2	181.9	191.1
Germany	4.9	6	6.56	7.19	9.12	10.9	11.1	11.1	28	33	37.4	42.65	46.71	51.77	54.3	57.3
Japan	5	5	4.9	5.2	5.3	5.6	5.3	5.4	49	56	64	73	81.4	90	97	105
Russian Federation	28.13	27	30.3	32	33.4	35.2	37.2	38.2	70	69	67	71	74.6	81	83	88
Switzerland	0.1	0.7	0	0.6	0.6	0.6	0.8	0.8	7	6	6	7	8	8	12	15
United Kingdom	54.8	60.1	69.1	72.5	78.1	82.4	90.8	96.2	47.4	47.2	45.9	46	43	41	38	37
United States of America	40.4	40.4	40.4	40.4	40.4	40.4	40.4	40.4	287.2	294.5	319.5	326.5	355	375	395	415
China																
Total *	161	173.9	190.6	202.6	211.9	218.1	228.5	233.8	653.6	675.7	714.6	743	791.1	840	883.2	931.8

* Separated plutonium of the USA is not included in the total.

4. PLUTONIUM INVENTORY FORECAST

The long term forecast of separated plutonium stockpile worldwide in the period from 1980 to 2050 is presented in Fig. 3, together with the INFCIRC/549 data. The overview of the plutonium management status in the major countries shows that the separated plutonium amounts in the nine Member States represent most of the world inventory of separated plutonium, plausibly more than 95%. The calculated results for the separated plutonium inventory are closely comparable with the values reported in INFCIRC/549 up to the year 2003, indicating that the VISTA model is reasonably predictive of world fuel cycle activity. In the case of the low reprocessing ratio (0% after 2015), VISTA predicts a rapid decrease of separated plutonium stockpile. Similarly, the stockpile with 30% reprocessing ratio begins to decrease after a peak in 2012. Only the case of 70% reprocessing ratio indicates a continuous increase. Since a large amount of separated plutonium stockpile increases the risk to security and the risk of proliferation, minimization of the equilibrium amount of separated plutonium is highly desirable. A balance between separated plutonium consumption and separation of plutonium in reprocessing is important.

As listed in Table 1 (INFCIRC/549), the plutonium contained in spent fuel has increased from 654 t HM to 932 t HM in the period 1996–2003,

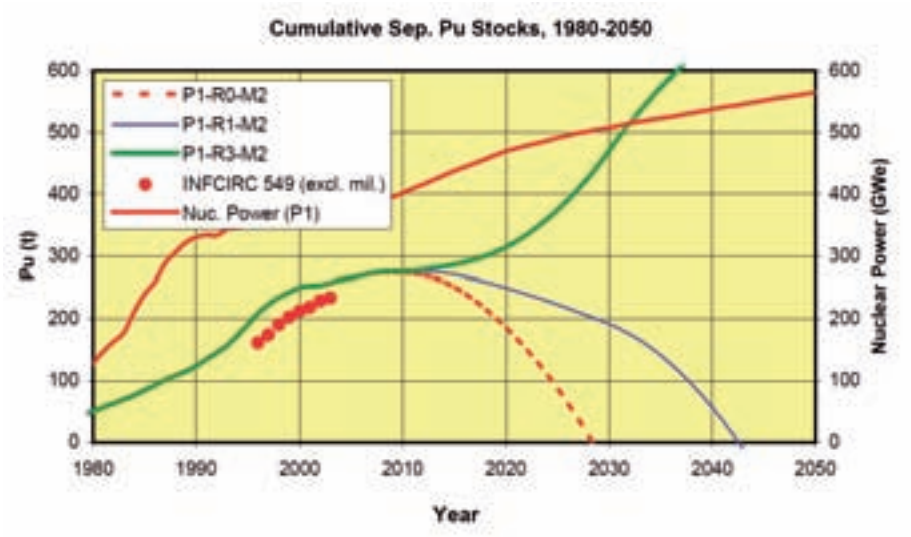

FIG. 3. Predicted cumulative separated plutonium stockpile up to 2050 with three different scenarios for the reprocessing ratio.

although these figures are obviously underestimates to the total amounts of plutonium worldwide, since the plutonium amounts in spent fuel stored in other major countries with nuclear energy programmes, such as Canada, India and the Republic of Korea, are not counted. Comparing the nuclear electricity generation in these nine countries and that worldwide, these figures were about 75% of the total in 2003. The VISTA calculations show that the total worldwide plutonium amount in spent fuel will increase annually at a rate of from 90 t HM to 150 t HM throughout the period from 2000 to 2050.

The total amount of plutonium in stored spent fuel increases steadily in all the scenarios, becoming 4670 t HM in the lowest case and 6300 t HM in the highest case in 2050 (Fig. 4), as long as LWR-Pu recycle continues. Since the plutonium stockpile represents a proliferation concern, even if it is contained in a highly radioactive spent fuel [12], it is essential to realize an equilibrium cycle in plutonium utilization worldwide that reduces the plutonium inventory in spent fuel by reprocessing, and to control the amount of separated plutonium appropriately.

5. MINOR ACTINIDE FORECAST

Minor actinides (MAs) are always generated as by-products of uranium and plutonium by neutron absorption or by decay in nuclear fuel irradiation in

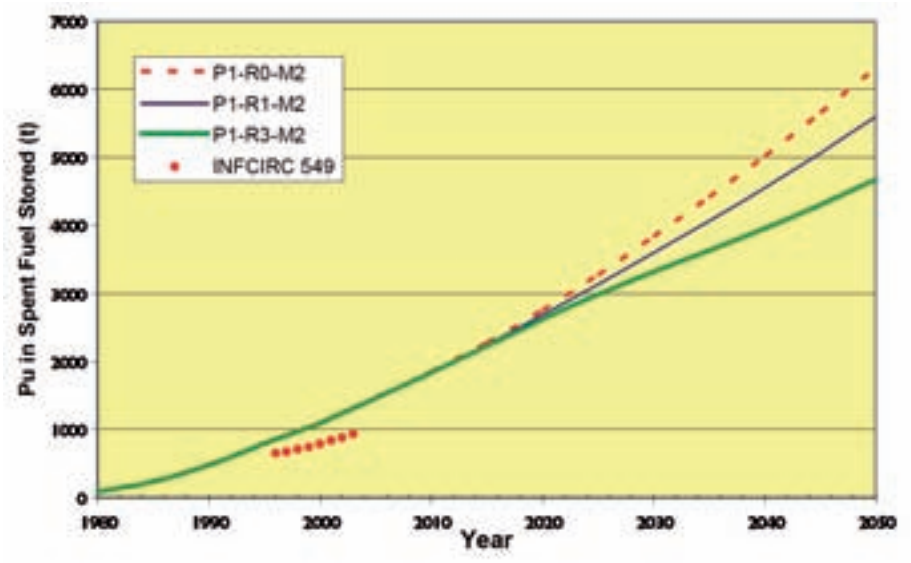

FIG. 4. Total plutonium inventory contained in stored spent fuel.

LWRs. It is essential to forecast MA generation from the viewpoint of fissile material management, particularly in relation to plutonium generation. Increasing amounts of MAs in the wake of the LWR-Pu recycle activities is a crucial issue for the environment and proliferation.

Table 2 summarizes the nuclear properties of some selected MA nuclides [13]. Neptunium-237, which can be separated easily on an industrial scale by adoption of the PUREX process, produces very little radiotoxicity but a very long term radiological dose [14]. With respect to criticality, the ^{237}Np critical mass is much larger than that of ^{239}Pu. However, ^{237}Np is of potential weapons grade; because of its high compressibility, its critical mass can be reduced by adoption of sophisticated technologies [15]. Recently, separation methods of Am and Cm have been developed that allow the isolation of the Am–Cm fraction. The separated Am–Cm is very difficult to handle in the nuclear fuel cycle, due to the high gamma, alpha and neutron emissions, coupled with a high decay heat (Table 2) as well as their chemical properties. Decay will lead to the formation of a mixed ^{240}Am–^{240}Pu component, which could ultimately be recycled. With americium, there is also potentially a danger of a nuclear explosion, with its fizzle yield. Curium can be ruled out as a weapons utilizable material because of its high spontaneous fission [15]. When Am–Cm is added to the plutonium cycle, the proliferation resistance would be increased due to the presence of ^{238}Pu and ^{244}Cm.

In the VISTA model, MAs exist either in spent fuel or in HLW and plutonium; however, separation of MAs in reprocessing is not considered. As is shown in Fig. 5, VISTA estimated that about 130 t HM of MAs (Np, Am and

TABLE 2. NUCLEAR PROPERTIES OF TRANSURANIC MATERIALS

Isotope	Half-life (a)	Neutrons ($s^{-1} \cdot kg^{-1}$)	Decay heat (W/kg)	Critical mass (kg)
^{237}Np	2.1×10^6	0.139	0.021	59
^{238}Pu	88	2.67×10^6	560	10
^{239}Pu	2.4×10^4	21.8	2.0	10.2
^{240}Pu	6.54×10^3	1.03×10^6	7.0	36.8
^{241}Pu	14.7	49.3	6.4	12.9
^{242}Pu	3.76×10^5	1.73×10^6	0.12	89
^{241}Am	433	1540	115	57
^{243}Am	7.38×10^3	900	6.4	155
^{244}Cm	18.1	1.1×10^{10}	2.8×10^3	28
^{245}Cm	8.5×10^3	1.47×10^5	5.7	13

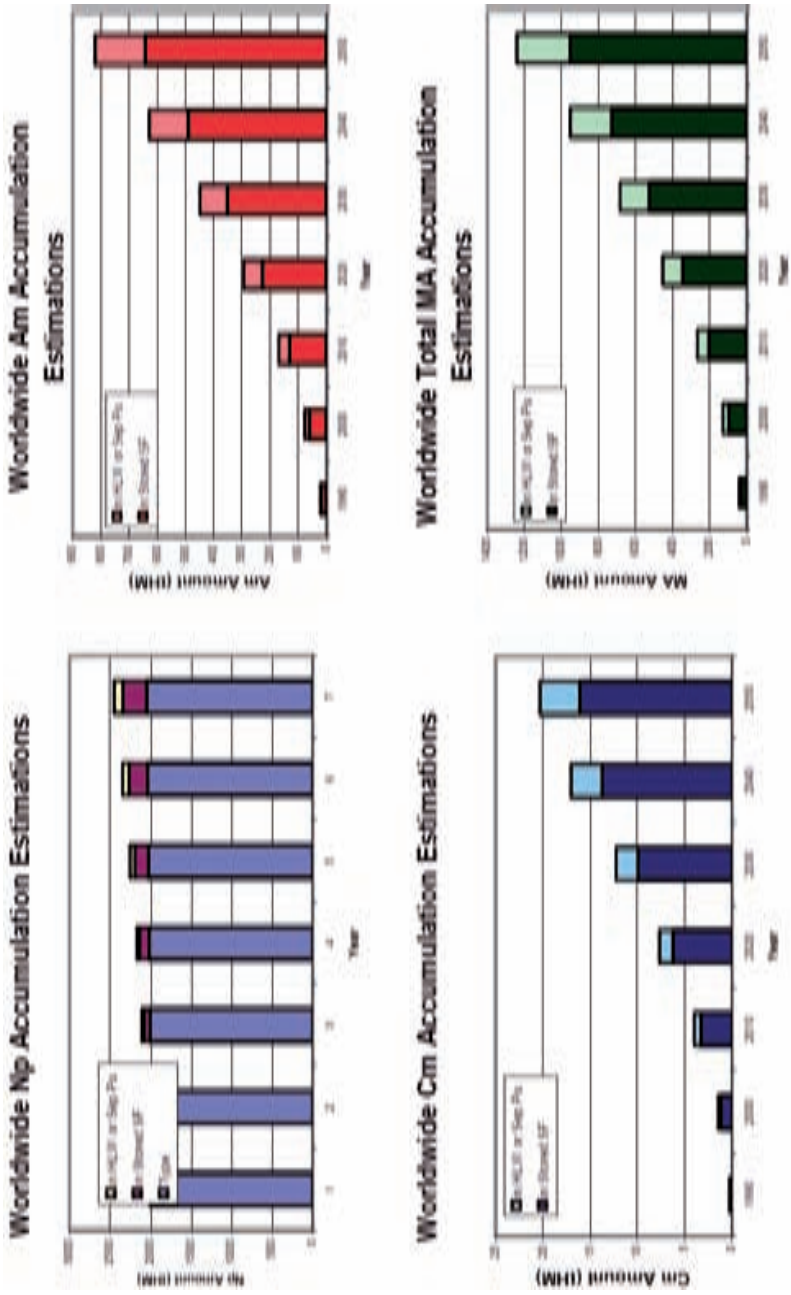

FIG. 5. Minor actinide inventories worldwide in HLW and separated plutonium, and in stored spent fuel.

Cm) were generated up to the year 2000, of which about 21% were existing in HLW and in separated plutonium. The remaining MAs are contained in spent fuel. The largest amount of MA isotope is ^{241}Am and the next is ^{237}Np, together constituting 93% of total MA generation (Fig. 6). Irrespective of the reprocessing ratio, MA generation increases at a parabolic rate up to 2050 due to increasing fuel burnup, which is different from the plutonium accumulation rate. Assuming that the present LWR-Pu recycle continues up to 2050, it is predicted that about 1240 t HM of total MAs are generated, of which 286 t HM are contained in HLW and plutonium and 953 t HM are contained in stored spent fuel at a reprocessing ratio of 30%.

Since ^{237}Np is a long term radiological hazard and has diversion potential, increasing MA amounts present serious concerns to the nuclear energy sector worldwide, regardless of whether use is once-through or recycle. Since the long term hazard of MAs continues over 40 000 years, predominantly due to ^{237}Np, the ultimate solution to cope with increasing amounts of MAs is only by partitioning and transmutation (P&T), and is a long term challenge. Although P&T mitigates the long term hazard, it requires sophisticated facilities for MA handling, and does not eliminate completely the necessity for geological disposal [16].

6. PERSPECTIVES OF TRANSURANIC MATERIAL DISPOSITION

Transuranic material disposition, particularly for plutonium, which is ongoing as MOX fuel in LWR-Pu recycle, is the key challenge in nuclear energy

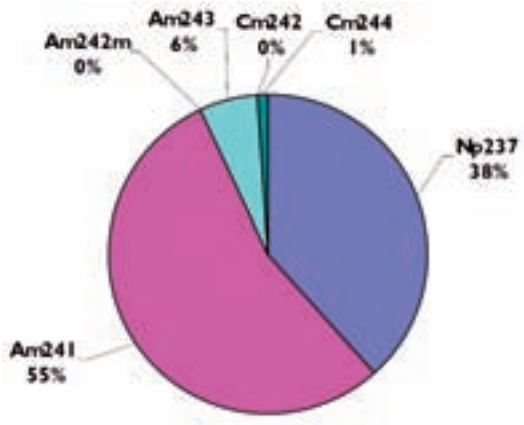

FIG. 6. Composition of MAs accumulated in 2003.

applications from the near to long term up to 2050. Within the current LWR-Pu recycle, it is essential to establish a self-sustaining equilibrium fuel cycle with the transuranic materials for sustainable development of nuclear energy. In addition, it is crucial to address the question of how to eliminate the total plutonium contained in spent fuel generated in the nuclear fuel cycle activity. Transuranic material disposition presents another challenge, which will be dealt with in the medium and long terms. Many countries implementing nuclear energy production are interested in, and pursue, disposition activities, details of which are now given.

The Belgian advanced fuel cycle programme [17] is within the framework of the European Union P&T project. Many projects are being implemented in Belgium, some of which are the synthesis of an extractant able to extract selectively Am and Cm from HLW in advanced aqueous reprocessing, a feasibility study of fuels such as the oxide components (Pu, Am)O_2 and (Pu, Am, Zr)O_2 for transmutation, and research into Am incineration.

The Chinese programme [18] has been conducted on the basis of the following policies and strategies on the back end of the fuel cycle: spent fuel is a resource, therefore the closed fuel cycle is the basic principle. The strategies are to develop techniques at pilot reprocessing plants and then to construct a commercial reprocessing plant.

The French programme [19] is focused on disposition of transuranic materials, covering plutonium and MAs. In the near term, the programme elaborates on enhancement of PUREX performance, with various objectives such as reprocessing of new fuels and simplification of the extraction process. In the medium term, development of partitioning of long term radionuclides in the aqueous process and development of new fuels, one of which is HTR fuel, will take place. In the long term, the French programme is based on the Generation IV system with an integrated cycle aimed at full actinide recycling, waste minimization, and integration of fuel treatment and fuel fabrication by using Generation IV fuel recycling.

The German programme [20] is characterized by a phase-out policy for nuclear energy, based on the new Atomic Energy Act, which came into force in April 2002. Germany owns about 32 t of plutonium, with about 10 t being in the form of fabricated MOX and the rest being stored as nitrate or oxide. The coalition Government has started a discussion about the extent to which the plutonium surplus could be reduced by burning MOX, and about ways of immobilizing and disposing of the rest of the surplus.

The Indian programme [21] is to develop the thorium fuel cycle, since India has a huge amount of thorium resources. Its strategy is ultimately to develop the self-sustaining equilibrium thorium cycle with (Th, ^{233}U)O_2 fuel

(SSET). As the intermediate step of the programme, a Pu–Th fuel cycle is under development using a PHWR and two BWRs.

The Japanese programme [22] on plutonium utilization in NPPs has been delayed due to problems with MOX pellet fabrication; however, KEPCO submitted a licensing application for MOX. It is expected to begin with MOX fuel in 2007. In its medium and long term plans, Japan is conducting a programme on development of advanced fuel cycle technologies in the framework of development of a commercialized fast reactor cycle system. These technologies comprise three candidate systems, MOX aqueous recycle, MOX dry recycle and metal dry recycle, with the objectives to develop a simplified and compact process, ensure capability to recover more than 99.9% MA from HLW and enhance proliferation resistance.

The programme of the Republic of Korea [23] has involved development of the advanced fuel cycle with basic strategies, which investigate, in terms of sustainability, technical options such as effective utilization of uranium resources and reduction of radwaste generation, as well as enhance proliferation resistance. The advanced fuel cycle development is implemented by several approaches: DUPIC (Direct Use of spent PWR fuel in CANDU reactor), ACP (Advanced spent fuel Conditioning Process), the pyro-process for partitioning and ADS (Accelerator Driven subcritical System) for transmutation. In the DUPIC programme, the instrumental irradiation test of DUPIC fuel was carried out at HANARO from 2000 to 2004.

The Russian programme [24] is concentrated on development of the closed fuel cycle incorporating the BREST fast reactor and dry reprocessing with ^{238}U–Pu nitride fuel. The main specific features of the closed fuel cycle with BREST are that plutonium will always circulate as a mixture with ^{238}U, pure plutonium will not be separated in the chemical treatment and MA generation is much less than in other fuel recycle systems.

The UK programme [25] is based on the separation of plutonium from spent fuel. In the UK to date, the preferred management option has been the indefinite storage of separated plutonium in the form of oxide powder. Until the late 1990s, there was little domestic or international pressure on the UK to develop long term plans for dealing with its plutonium stockpile. In such circumstances, a policy debate has recently opened.

The US programme [26] conducts the Advanced Fuel Cycle Initiative (AFCI) with strategies to develop proliferation resistant spent nuclear fuel treatment and transmutation technology to enable a transition from the current once-through fuel cycle to a future sustainable closed nuclear fuel cycle. In the medium term programme in the period around 2020–2030, the AFCI will develop proliferation resistant recycling of Pu (+Np/Am) in thermal reactors (the Advanced Thermal Reactor Fuel Cycle) and, in the long term after around

2040, fast transuranics burners will be developed to establish a self-sustaining closed fuel cycle.

The overview of the programme on transuranic material disposition in countries positive to nuclear energy has clarified specific features for development of the advanced fuel cycle, namely proliferation resistance, waste minimization and effective use of resources. It seems that one of the ultimate solutions to reducing amounts of transuranic materials is a self-sustaining closed fuel cycle having these features. Table 3 itemizes national programmes

TABLE 3. NATIONAL INCENTIVES TO DEVELOPMENT OF INNOVATIVE TECHNOLOGIES TO UTILIZE FISSILE MATERIALS

	Country	Programme for advanced technology	Feature that is incentive to innovation		
			Energy resources	Waste minimization	Proliferation resistance
Countries with strong incentive to innovation	France	Advanced fast reactor	◎[a]	◎	
	India	Thorium recycle	◎		
	Japan	Advanced fast reactor	◎		
	Republic of Korea	DUPIC	◎		◎
	Russian Federation	BREST	◎		◎
	USA	AFCI		◎	◎
No programme for innovation	Belgium	Support innovative technology development	–	–	–
	China	To begin plutonium use	–	–	–
	UK	Decision pending	–	–	–
Phase-out	Germany	Disposition of plutonium		–	–

[a] The symbol ◎ denotes a particularly strong incentive to innovative technology development, although all of these technologies comply with all three incentives.

on the development of innovative technologies for disposition of fissile materials, together with the incentives for this development.

7. CONCLUSIONS

This paper concludes that, if the present LWR-Pu recycle is continued for several decades, the plutonium inventory in spent fuel will continuously increase up to the range from 4670 to 6300 t HM, and MAs in any form will amount to 1240 t HM in 2050, although the stockpile of separated plutonium will strongly depend on future strategies. If such large amounts of transuranic materials are disposed of in geological sites, this gives rise to many concerns about, for example, environmental destruction, proliferation risks and economic losses. Therefore, it is desirable to realize a self-sustaining equilibrium fuel cycle with the transuranic materials for the sustainability of nuclear energy. In the near term, it is significant to develop advanced technologies in the LWR-Pu recycle such as the use of plutonium fuel assemblies (MOX/EUS, APA and PLUTON) for PWRs, as discussed by the Organization for Economic Co-operation and Development [1], and in the medium and long terms the innovative fuel cycle technologies of the P&T system incorporating FBRs and ADS, for example, the Advanced Fuel Cycle Initiative proposed by the United States Department of Energy [12, 26].

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INTERNATIONAL DATABASE OF SPENT FUEL INVENTORIES AND MANAGEMENT

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Abstract

A number of countries and organizations, especially those involved in radioactive waste management, are operating, or are trying to establish, for their own purposes, a database including spent fuel inventories. There is, in general, a global trend towards greater transparency of information, which may require more information to be made public in the case of nuclear issues such as spent fuel and waste management, including data on storage inventory or transportation, as well as on the end points of spent fuel management such as reprocessing or disposal. In the paper, methods for data collection and compilation of spent fuel inventories and management on the global level are reviewed, with a view to setting up an international database that will be both reliable and accessible to prospective users. With good cooperation and planning, duplication of effort can be avoided and the overall effectiveness of the effort can be enhanced. The Joint Convention on the Safety of Radioactive Waste Management and on the Safety of Spent Fuel Management could be a channel for compiling the relevant data through its triennial review processes, while the Power Reactor Information System (PRIS) database would be able to provide annual updates from an annual questionnaire about spent fuel arisings and storage.

1. INTRODUCTION

Whereas reliable statistics on spent fuel management would be essential for the global nuclear community for any possible approach to international cooperation and for national needs, compilation of data on large amounts of spent fuel dispersed at various nuclear facilities around the world is a challenge. Just as in a population census, it is not a trivial exercise to conduct regular collection and compilation of spent fuel inventory data, which are subject to frequent changes.

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Spent fuel inventory data are important for various national and international spent fuel management activities, especially for planning and regulatory activities. Recently, the security issue has begun to be an additional factor to be considered in the information management associated with spent fuel or radioactive waste. The specific need for spent fuel inventory data varies depending on the ultimate purpose:

(a) International level — Compilation on a gross tonnage basis mainly for statistical purposes and global trend analysis for use by the IAEA and at the request of Member States;
(b) National level — Compilation for industry and regulatory purposes on either a gross tonnage or individual assembly basis to assist in planning and public awareness;
(c) Operator level — Origination and maintenance of detailed data on individual assemblies by the utility for operational needs or to meet regulatory requirements.

There is in general a global trend towards greater sharing of information with the general public, which may require more information to be made public about spent fuel management, including data on storage inventories or transportation. With the increase in the commercialization of the nuclear industry, the trend is away from national government operation of nuclear activities. This results in a spread of information about spent fuel as it is not concentrated in a vertical manner at the government level, but is instead held by various organizations in the private sector in a more horizontal manner.

The management of information on spent fuel is also affected by national policy on spent fuel management. Some countries are pursuing reprocessing of spent fuel, which is regarded as a resource, in contrast to others adopting a once-through (direct disposal) policy declaring spent fuel as waste. In the former case, information about spent fuel is in general managed separately from radioactive waste either by the pertinent national organization or by the commercial entity involved in the reprocessing business, while, in the latter case, spent fuel data are managed in the framework of radioactive waste information management.

For the proposed global compilation of spent fuel inventory data, at a minimum on a national level, the following information would be useful for statistical status and trend analyses. Where available, the information can be collected at the facility or site level.

The spent fuel inventory should be collected individually for:

(a) At reactor (AR) and away from reactor (AFR) storage facilities;

(b) Pool (wet) storage and dry storage systems (vault and metal/concrete cask).

Such distinctions about storage type and method could be useful in compilation of trends in spent fuel management.

The accounting boundary for the inventory of spent fuel begins with the discharge from the reactor and is interfaced with spent fuel management options, including reprocessing, disposal or other future options such as direct refabrication (e.g. DUPIC[1]). The identity of a spent fuel assembly can be lost, as in reprocessing, which will be at the end of assembly life. At this time, the accounting method must be changed to a suitable one (e.g. bulk accounting) in order to maintain the balance as the assembly no longer exists.

As most off-site shipments were or are still delivered to reprocessing facilities (either planned or operating), reliable data on the quantity of material reprocessed, as well as the quantity of material in buffer storage at reprocessing facilities, are important in order to enhance the global data on spent fuel inventories and management.

Information on spent fuel may also have to be considered in the context of preservation of nuclear knowledge. In addition to current inventories, historical and projected data are important for some purposes such as consistency analysis.

2. IAEA ACTIVITIES ON SPENT FUEL DATA

The IAEA is in a good position for the collection, compilation and dissemination of information including spent fuel inventory data, as an international organization for nuclear cooperation with a large number of Member States.

2.1. INFCE and EG-ISFM

The significance of global data on the nuclear fuel cycle was recognized a few decades ago through the International Nuclear Fuel Cycle Evaluation (INFCE) and in parallel by the Expert Group on International Spent Fuel Management (EG-ISFM), which addressed the issues associated with international management of spent fuel, and specifically the data management problem [1]:

[1] DUPIC: Direct Use of spent PWR fuel In CANDU reactors.

> "The experience of Task 1 in constructing a database on spent fuel arisings and storage capacity has reconfirmed a lesson learned by INFCE Working Group 6 (Spent Fuel Management) — namely, that it is difficult and time consuming to produce an accurate database.
> "Even in the case where an adequate database already exists, it is difficult to keep it up to date.
> "Before a commitment to establishment of a permanent database on spent fuel storage requirements could be made, it would be necessary to determine the need for such a database and the uses to which it might be put."
>
> (From the Final Report of the EG-ISFM, July 1982)

The report of the EG-SFM did in fact indicate the differences between data sources (i.e. INFCE Working Group 6, EG-ISFM and other available data sources) projected for the period 1980–2000 at intervals of five years. The difference ranged from a few per cent in 1980 to a projected maximum of 22% in 2000 due to the assumptions made about the nuclear power plans of various countries [2].

Aside from this tradition of EG-ISFM, which became the basis for the activities of the IAEA in spent fuel management, there are several other lines of activity related to spent fuel inventory data.

2.2. RAG-SFM and SFMN

One of the IAEA projects on spent fuel management in this period has been collection of information on spent fuel arisings and capacity requirements. The Regular Advisory Group on Spent Fuel Management (RAG-SFM), which was established in the early 1980s on the recommendation of the EG-ISFM, was a channel for collecting information on spent fuel management in the Member States, including information on inventories of spent fuel, through its meetings held biannually until it was merged into the TWG-NFCO&SFM (Technical Working Group on Nuclear Fuel Cycle Options and Spent Fuel Management) in 2001 [3–5]. Since 2002, the TWG-NFCO&SFM has met annually and serves to collect national information on spent fuel management, but in a limited way due to the limited number of participating countries and to the lack of reporting on spent fuel inventory data by some of the participating countries.

The RAG-SFM had to be complemented by the Spent Fuel Management Newsletter (SFMN), which was intended to alternate with the RAG-SFM in providing the relevant information, and several issues were actually produced in the first half of the 1990s [6, 7]. An effort was made to revive the old SFMN

initiative (at a consultancy meeting held in 2000), and a questionnaire was issued to Member States producing nuclear power. This new initiative was soon stopped, however, due to the poor response from Member States, together with a shortage of resources, with a recommendation that it may be more efficient to integrate such efforts into the overall IAEA initiative to strengthen its web based system.

2.3. NFCIS

On the other hand, the Nuclear Fuel Cycle Information System (NFCIS) was also established in the 1980s by the Nuclear Fuel Cycle and Materials (NFC&M) Section of the IAEA as a database to provide comprehensive information on the nuclear fuel cycle facilities around the world, and thus should contain those inventory data related to spent fuel management facilities. Updating the relevant data in the NFCIS database was initiated in 2003 with a questionnaire on spent fuel inventories. The result of this initiative was dismaying, however, due to the insufficient response to the questionnaire.

There is also a tool, developed and maintained in the NFC&M Section, called VISTA[2] that can be used in conjunction with NFCIS to estimate the amount of spent fuel discharged and stored [5].

2.4. NEWMDB

The Net-Enabled Waste Management Database (NEWMDB) of the Waste Technology Section of the IAEA was launched on the Internet in January 2004. It provides information on national radioactive waste management programmes, plans and activities, relevant laws and regulations, policies and radioactive waste inventories.

The principal objectives for developing the NEWMDB were to:

(a) Support the routine reporting of status and trends in radioactive waste management on the basis of, to the greatest extent practicable, quantitative data rather than anecdotal information;
(b) Support the compilation of the inventory of radioactive waste in IAEA Member States on the basis of a unified waste classification scheme;
(c) Support the development, implementation and use of an indicator of sustainable development for radioactive waste management;

[2] Nuclear Fuel Cycle Simulation System (VISTA).

(d) Provide the means to assess the development and implementation of national systems for radioactive waste management in IAEA Member States;
(e) Conform, to the greatest extent practicable, with the reporting requirements of the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (Joint Convention).

The NEWMDB is a major database on radioactive waste inventories, and is available and managed on the web. However, it does not contain spent fuel inventories, although spent fuel facilities can be defined. In the context of the NEWMDB, the need for strengthening international cooperation on radioactive waste management information systems was emphasized [9, 10].

2.5. PRIS

The Power Reactor Information System (PRIS) is a major database for compilation of technical information on the status of global nuclear power reactors, updated through a questionnaire on an annual basis. Established in 1970, the PRIS database contains detailed information on nuclear power plants worldwide, including design characteristics, general specifications of power reactors and data on operating experience with nuclear power plants. This latter category of PRIS data lends itself to collection of spent fuel discharge and storage data from power reactors with the simple addition of a few additional questions to the PRIS questionnaire that has been conducted on an annual basis.

The possibility of adding several items to the questionnaire (e.g., amount of spent fuel discharged during a year and amount of spent fuel that is in storage at AR/AFR facilities) was posed in 2000, when there was a plan for a newsletter on spent fuel management. While a majority of the advisory group members of PRIS were supportive, a few members questioned the justification of such additional efforts.

The PRIS database is considered to be one of the best instruments for collecting spent fuel inventory and discharge data on an annual basis. Moreover, the scope of the PRIS database has recently been widened to include non-electrical applications, decommissioning and delayed projects [11].

2.6. The Joint Convention

"The Joint Convention[3] applies to spent fuel and radioactive waste resulting from civilian nuclear reactors and applications and to spent fuel and radioactive waste from military or defence programmes if and when such materials are transferred permanently to and managed within exclusively civilian programmes, or when declared as spent fuel or radioactive waste for the purpose of the Convention by the Contracting Party" (see Ref. [12]).

Article 32 of the Convention includes a requirement for reporting:

"a list of spent fuel management facilities, their location, main purpose, and essential features;
"an inventory of spent fuel that is being held in storage and that has been disposed. The inventory shall contain a description of the material and, if available, give information on its mass and total activity."

Most of the national reports that were submitted to the review meeting of 2003 contained sufficient inventory data, but they were not presented in a consistent manner. In addition, several countries with large reactor programmes are not parties to the Joint Convention[4]. In order for the data from the Joint Convention to be useful in this context, a consistent format must be used. In addition, the inventory data are only collected at intervals of three years, at the time of review meetings[5].

2.7. Data from IAEA safeguards inspection

By the Safeguards Agreement between the IAEA and Member States, the information acquired by safeguards inspectors is considered safeguards confidential and cannot be released for general use in spent fuel management.

[3] The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management was opened for signature on 29 September 1997 and entered into force on 21 June 2001.

[4] Notably China, India, the Russian Federation and Taiwan.

[5] Of which the first was held in 2002 and the second is planned for 2005.

3. OTHER SOURCES OF DATA ON SPENT FUEL MANAGEMENT

The management of data is an important part of the activities associated with spent fuel management. For this and other reasons, some Member States or other international organizations have compiled international data on spent fuel management as well as their own national data[6].

3.1. National data

Many countries have developed and maintained national databases on spent fuel inventories. Because of the difference in institutional arrangements and practices for spent fuel management in the Member States, these databases are not always consistent or available, so they cannot be easily used to compile a global spent fuel inventory. There is a trend, however, towards electronic data reporting in these various systems, which could eventually lead to a simplified data collection method using an interconnected data exchange network.

Some examples of national databases are:

(a) In the United States of America (USA) the OCRWM[7] has published an updated database containing inventories of commercial spent fuel in the USA up to the end of 2002 [12]. The Central Internet Database [13] provided by the USDOE is a very comprehensive and up-to-date database on the status of USDOE radioactive waste management, including spent fuel.
(b) In the UK, the CoRWM[8] issued a report on the inventory of radioactive waste, including spent fuel, being managed at UK nuclear sites [14].
(c) In the Republic of Korea, the regulatory institution (KINS[9]) has established a database called WACID[10] to provide radioactive waste management information to the public via the Internet [15]. Unfortunately, WACID is provided only in the Korean language.

[6] The Energy Information Administration (EIA) of the USDOE has collected and provided information on nuclear power and fuel cycles, including the global status of spent fuel management as well as US national data.

[7] OCRWM: Office of Civilian Radioactive Waste Management.

[8] CoRWM: Committee on Radioactive Waste Management (commissioned by the UK Government).

[9] KINS: Korea Institute of Nuclear Safety.

[10] WACID: Waste Comprehensive Information Database. WACID was put into operation in Jan. 2005 after a period of demonstration operation of six months.

3.2. International organizations

In addition to the IAEA, there are several international organizations that have compiled international data on spent fuel management for their own purposes:

(a) The EU used to collect data on spent fuel management status in Eastern European countries under the EU support programme to Eastern European countries. Unfortunately these data are not collected continuously and therefore are generally not always up-to-date. An overview of radioactive waste management in the EU and the enlarged EU was provided in several documents, but spent fuel quantities are mentioned only in the margin and not systematically. A recent example in this regard is the SAPPIERR report published by European Commission in 2004 [16]. It is a publication from the EC sixth Framework Programme for a pilot study called SAPIERR on the regional approach to waste disposal, and provides data collected by the SAPIERR Member States (Austria, Belgium, Bulgaria, Croatia, Czech Republic, Hungary, Italy, Latvia, Lithuania, Netherlands, Romania, Slovakia, Slovenia and Switzerland), excluding non-Member States (Finland, France, Germany, Spain, Sweden and the UK). A comprehensive database on spent nuclear fuel inventories probably does not exist within the EU administration.
(b) The Nuclear Energy Agency (NEA) of the OECD compiled data on radioactive waste management, including the spent fuel of its Member States [17].

3.3. Commercial sources

There are several sources of commercial information on spent fuel management compiled and sold by commercial companies, such as:

(a) NAC International's Fuel-Trac database is a commercial source of information on the nuclear fuel cycle, including spent fuel discharges, shipments and reprocessing of spent fuel, as well as storage [18]. The fuel cycle plan includes reload startup and shutdown dates, assemblies loaded and discharged, initial enrichment and weight of fuel assemblies, discharge burnup and enrichment, and plutonium production during irradiation. The inventories of spent fuel in the Fuel-Trac database are calculated from historic discharges, shipments, re-inserts and reprocessing. Future inventories on a site level basis are projected on the basis of expected discharges for each reactor, and planned or projected

shipments and reprocessing. These data are reconciled on an annual basis and are compiled from a variety of sources, including official government reports, industry publications and commercial information.

(b) The company New York Nuclear publishes a set of bulletins on the nuclear fuel cycle, including one on spent fuel, on a commercial basis [19].

3.4. Non-Governmental organizations

There are a number of non-governmental organizations (NGOs) that provide information on nuclear issues, including management of radioactive waste and spent fuel, nuclear safety and non-proliferation. In some cases, information on inventories and statistics can be quite extensive, although the accuracy is difficult to verify.

A good example of an NGO study on nuclear materials which includes spent fuel is that by the Stockholm International Peace Research Institute (SIPRI), which published a book on world inventories based on work by SPRU (Science and Technology Policy Research at the University of Sussex, UK) [20].

With the spreading use of the Internet, a number of web sites run by NGO groups provide information often touching on nuclear material or waste inventories. These data are mostly cited from other sources such as national or international organizations.

4. DATA ON SPENT FUEL MANAGEMENT COMPILED BY THE IAEA

An update of the global data was conducted in the late 1990s on a country level and last year on a site level, of which global figures were published in 2003 [21]. The next series of this conference will be held in June 2006, which will be an opportunity to collect updated data. Existing data to the end of 2003 were given on the basis of various sources:

(a) Data presented at TWG-NFCO&SFM and WATEC (most recently held in May and June 2005, respectively);
(b) Contacts with Member States;
(c) Professional literature (including commercial sources);
(d) Internal estimates and reference (such as NFCIS and VISTA).

5. ISSUES IN COMPILATION OF GLOBAL DATA FOR A SPENT FUEL INVENTORY

For global statistics, the primary data of interest in management of spent fuel on an international basis have been:

(a) How much has been discharged and how much reprocessed?
(b) How much remains in storage in AR and AFR facilities?
(c) How much is stored in dry storage with respect to wet storage among the AFR facilities?

With no disposal of spent fuel implemented anywhere in the world, the equation of spent fuel material balance is quite simple:

Discharged = Reprocessed + Storage (AR + AFR)

As the nuclear fuel is handled in individual assemblies or in a basket containing a multiple number of assemblies, the quantification of spent fuel is usually done in number of assemblies for fuels of uniform design, and accounting into other common units like tonnage is not complex[11]. When it comes to a variety of different designs or even worse of dealing with spent fuel materials that are not identifiable as separate items, as in the case of reprocessing facilities, accounting in a common unit like weight becomes inevitable[12].

5.1. Spent fuel inventories in storage

With the problems encountered at an increasing number of reactor sites for additional storage of spent fuel by the AFR type of facility, the provision of extra capacities for spent fuel storage has become a concern in the statistics compiled by the IAEA. The significance of such statistical data has begun to change recently with the increasing introduction of dry storage systems. The amount of spent fuel in dry storage versus conventional wet storage has also

[11] As there are a number of different fuel designs, it is often more appropriate to use weight in tonnage of heavy metals (MTHM) rather than in number of fuel assemblies for comparative means on an international basis.

[12] For accountability purposes, similar issues of numerical conversion can arise. Even if it were allowed to use data collected from safeguards inspection, mainly based on item counting, it would require additional effort to convert into a common basis for spent fuel management purposes such as tonnage.

been significant for a number of decades. Such new trends towards dry storage at an increasing number of facilities (especially of cask types) signify that the provision of extra storage capacities is not as meaningful as it used to be for pool storage facilities. While the pool facilities need to be initially built at full capacity, the modular type of dry facility can be added as needed, to the advantage of minimizing idle capacities and thereby advance financial penalties. Together with some other advantages of dry storage systems, casks are currently the most popular option in the spent fuel storage market. For expedited installation, casks can be purchased competitively in the market, on the assumption that the necessary licence from the authority has been acquired and in the absence of barriers from the local community.

5.2. Data on spent fuel reprocessing

Much global reprocessing in the early period of nuclear power development was mixed with military activities, and quantitative data associated with reprocessing of metal fuels from gas cooled reactors (GCRs) were not widely known. While information on the commercial reprocessing of spent light water reactor (LWR) fuel has been much more available in the public domain, the large amount of spent metal fuel from GCR power reactors in France and the UK (excluding those production reactors in the USA and Russian Federation) is one of the major sources of uncertainty in the compilation of global data because of its sheer quantity[13].

There is a need for interpretation of 'the one third of global spent fuel arisings reprocessed' in the global statistics, because the bulk amount of this reprocessed figure is metal fuel, and therefore the actual amount of spent oxide fuel from LWRs reprocessed represents only a fraction of the globally discharged amount. It would be more meaningful to specify the amount of LWR fuel reprocessed along with the total spent LWR fuel discharged. It should be noted in this context that spent CANDU fuel or RBMK fuel has not been reprocessed because of low recoverable fissile contents and the fact that they represent a relatively large portion of the global inventory due to their relatively low enrichment content and burnups.

In the compilation of global data on spent fuel reprocessed and stored for buffer at reprocessing facilities, attention need not be paid to double counting in the data reported by some countries exporting spent fuel for reprocessing

[13] France released a figure of 18 400 t HM, whereas the UK released a figure of 'more than 40 000 t HM', which would be equivalent together to an amount of more than double the spent LWR fuel reprocessed up to now.

services (e.g. several Western and Eastern European countries and Japan)[14]. A distinction also needs to be made from the spent fuel inventory in the reprocessing facilities of the reprocessor countries.

5.3. Projection of spent fuel inventories

The quantity of spent fuel arisings from a power reactor is inversely proportional to the thermal power generated by fuel burnup, which is in its turn a function of uranium enrichment. The actual spent fuel arisings are therefore a complex function of operational parameters of the reactor as well as of the fuel design.

The most significant parameter of spent fuel arisings is the burnup. The tendency to higher burnup will result in a smaller amount of spent fuel arisings in the future, which is not tantamount to a reduction in future burdens in the fuel cycle back end, because higher burnup will bring a proportional increase in heat and radioactivity loadings[15].

Physical size and configuration will also have some significance in spent fuel management due to the operations required for standard conditioning for storage, reprocessing or disposal[16].

6. CONCLUSION AND RECOMMENDATIONS

There is an increasing number of sources available that provide information on spent fuel inventories, including on the Internet. Many of these are not primary sources, but are instead based on the compilation of other sources of information, potentially without a consistent basis. The accuracy of this information is not always apparent, and it requires additional analysis and verification to compile a reliable global inventory.

A number of countries and organizations, especially those involved in radioactive waste management, are operating or trying to establish a database including spent fuel inventories for their own purposes. With good cooperation

[14] Those exported amounts might be only partly reprocessed, and the remaining portion may still be in buffer storage at the reprocessing plants.

[15] This also signifies that the large amount of spent fuel arisings from lower burnup reactors such as CANDU, RBMK and Magnox will have correspondingly lower heat and radioactivity loadings in comparison with higher burnup fuel.

[16] This is the case with RBMK spent fuel, assemblies of which need to be cut in half to fit inside the storage containers.

and planning, duplication of effort can be spared and the overall effectiveness of the effort made can be enhanced.

From this review of the current status of this subject, it appears that it is possible to develop an adequate database on global spent fuel inventories. Unfortunately, none of the current data collection methods and options will provide a complete worldwide compilation. Using a combination of an enhanced annual PRIS questionnaire and standardized inventory reporting under the Joint Convention every three years, reliable data could be collected for most countries with nuclear power plants.

To supplement the IAEA data collection instruments, a variety of sources can be used. Freely available information from government or non-government organizations as well as commercially available data can help to fill in the gaps in the information collected by the IAEA. These data can also be used to verify and check the data submitted to the IAEA through PRIS and the Joint Convention.

The following recommendations should be considered to improve spent fuel inventory data collection:

(a) The PRIS questionnaire should be expanded with a few additional questions on spent fuel inventories, to provide inventories for both AR and AFR storage as well as wet versus dry storage.
(b) A standardized reporting guideline for the Country Reports submitted under the Joint Convention should be developed. For spent fuel inventories, it would be desirable to report on a site level the inventory levels for both AR and AFR storage, as well as wet versus dry storage.
(c) Cooperation should be enhanced with countries and organizations trying to establish similar databases of spent fuel inventories, to minimize duplication of effort and enhance the overall product.
(d) To supplement and improve the information collected by the IAEA, all available information should be considered and utilized. A comprehensive search of available sources of information should be performed and catalogued to provide the IAEA with a list of sources of spent fuel inventory information.

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THE ECONOMICS OF REPROCESSING VERSUS DIRECT DISPOSAL OF SPENT NUCLEAR FUEL

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Abstract

The economics of reprocessing versus direct disposal of spent nuclear fuel are assessed. The break-even uranium price at which reprocessing spent nuclear fuel from existing light water reactors (LWRs) and recycling the resulting plutonium and uranium in LWRs would become economic is estimated for a wide range of reprocessing prices and other fuel cycle costs and parameters. The contribution of each fuel cycle option to the cost of electricity is also estimated. A similar analysis is performed for the break-even uranium price at which deploying fast neutron reactors (FRs) would become competitive compared with a once-through fuel cycle in LWRs, for a range of differences in capital cost between LWRs and FRs. Available information about reprocessing prices and various other fuel cycle costs and input parameters are reviewed, as well as the quantities of uranium likely to be recoverable worldwide at a range of different possible future prices. It is concluded that the once-through fuel cycle is likely to remain significantly cheaper than reprocessing and recycling in either LWRs or FRs for at least the next 50 years. Finally, there is a discussion of how scarce and expensive repository space would have to become before separation and transmutation would be economically attractive.

1. INTRODUCTION

Whether it is better to dispose of spent fuel from nuclear power reactors directly in geological repositories or reprocess it to recover and recycle the plutonium and uranium has been debated for decades. These debates have become more salient as increasing accumulations of both spent nuclear fuel and separated plutonium from reprocessing generate concern worldwide. While many other factors affect decisions on reprocessing or direct disposal of spent fuel, cost is an important element in this debate, particularly in a nuclear industry facing an increasingly competitive environment, and where fuel cycle costs are among the few expenses that reactor operators can control.

There is general agreement that at the current low uranium and enrichment prices, reprocessing and recycling is more expensive than direct disposal of spent fuel [1–3]. The debate is over the magnitude of the difference and how long it is likely to persist. Advocates of reprocessing argue that the premium is small today and will soon disappear as uranium becomes scarce and increases in price [4]. Here, we argue that the margin is wide and likely to persist for many decades to come.

These issues are increasingly important, as a number of countries face major decisions about future management of their spent fuel. In the United States of America, in particular, the US Department of Energy (USDOE) plans to spend several hundred million dollars over the next few years on research and development related to reprocessing in the Advanced Fuel Cycle Initiative [5].

We proceed as follows. Firstly, we compare the costs of direct disposal versus reprocessing and recycling in light water reactors (LWRs) by calculating the 'break-even' uranium price — the price of uranium at which the cost of electricity would be the same for both options — for various reprocessing prices and other fuel cycle prices and parameters. We focus on the break-even uranium price because the prospect that rising uranium prices would make reprocessing economic has been such a prominent focus of the arguments made by advocates of reprocessing. We also perform a sensitivity analysis and calculate the contribution of these fuel cycle options to the cost of electricity.

Secondly, we repeat this analysis to compare the costs of direct disposal with LWRs to reprocessing and recycling in fast neutron reactors (FRs). Thirdly, we review the history of uranium prices, estimates of uranium resources recoverable at a given price, and scenarios of uranium consumption under the direct disposal option, to assess when reprocessing and recycling in LWRs or FRs might become economically attractive. Finally, we discuss the impact of fuel cycle choices on repository requirements.

Where possible, we base our estimates on historical market prices for fuel cycle services. Where markets are not well developed, as is the case for reprocessing and mixed oxide (MOX) fuel fabrication, our estimates are based on the best available information on facility construction and operation costs. Unless otherwise noted, prices and costs have been converted to 2003 US dollars using market exchange rates and US gross domestic product deflators.

2. DIRECT DISPOSAL VERSUS REPROCESSING IN LWRs

We adopt the viewpoint of an LWR operator who has discharged spent fuel and is deciding which option is less expensive: direct disposal or reprocessing. With direct disposal, the reactor operator would have to pay the costs of (a) interim storage of the spent fuel and (b) transport to a repository site and disposal of the spent fuel (possibly including conditioning prior to disposal). With the reprocessing option, the reactor operator would have to pay the costs of (a) transport to the reprocessing plant and reprocessing of the spent fuel and (b) disposal of reprocessing wastes.[1] The plutonium and uranium recovered during reprocessing can be used to fabricate MOX fuel, reducing requirements for fresh low enriched uranium (LEU) fuel.

The value of the recovered plutonium and uranium is the value of the fuels that can be made from these materials minus the costs of fuel fabrication. Because fuels made with recovered plutonium and uranium would substitute for LEU fuels, their value is determined by the price of LEU fuel with the same design burnup, which in turn depends on the price of uranium. The uranium price at which the net present cost of the two fuel cycles is equal is the 'break-even' price, given notionally by

$$\begin{bmatrix}\text{cost of interim storage}\\ +\text{ disposal of spent fuel}\end{bmatrix}=\begin{bmatrix}\text{cost of reprocessing}\\ +\text{ disposal of wastes}\end{bmatrix}+\begin{bmatrix}\text{cost of producing LWR fuel}\\ \text{using recovered Pu, U}\end{bmatrix}-\begin{bmatrix}\text{cost of equivalent}\\ \text{LEU fuel}\end{bmatrix} \quad (1)$$

[1] There may be additional costs associated with storing, safeguarding and transporting plutonium and MOX fuel, licensing MOX use in reactors, and changes in fuel management. We ignore these additional costs, an assumption favourable to the recycle approach.

Of course, many factors enter into a complete calculation — carrying charges on the cost of the material during its processing and use, fuel burnup, the isotopic composition of the recovered uranium and plutonium, and the resulting plutonium concentrations or uranium enrichment levels required to achieve a given design burnup, the amount of uranium and enrichment work used to produce a kilogram of LEU at a given uranium price, and so on. The equations we have used to calculate the break-even uranium price and the cost of electricity, which take these and other factors into account, are fully documented in Ref. [6] and have been implemented in spreadsheets that we have made publicly available [7].

2.1. Break-even prices and difference in cost of electricity

Figure 1 shows the break-even uranium price as a function of the price of reprocessing (including transportation of fuel to the reprocessing plant, short term storage of spent fuel and plutonium, treatment and disposal of low level waste (LLW) and intermediate level waste (ILW), and interim storage of high level waste (HLW)). Table 1 gives central estimates of the various parameters used in this calculation, as well as estimates that reflect best and worst cases for reprocessing. These estimates are discussed in more detail below.

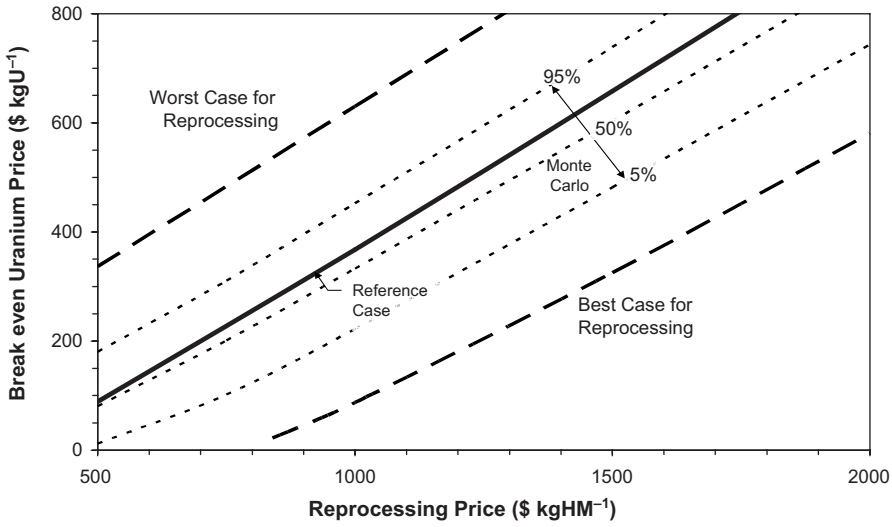

FIG. 1. Break-even uranium price as a function of reprocessing price, for various sets of assumptions about other fuel cycle prices and parameters (see Table 1).

TABLE 1. ESTIMATES OF FUEL CYCLE COSTS AND OTHER PARAMETERS, AND SENSITIVITY ANALYSIS FOR THE BREAK-EVEN URANIUM PRICE FOR DIRECT DISPOSAL VERSUS REPROCESSING AND RECYCLING IN LWRs, FOR A REPROCESSING PRICE OF $1000 kgHM^{-1}

Parameter	Parameter value[a]			Break-even U price (central = $368 kgU^{-1})		Change compared with central value
	Low	Central	High	Low	High	
Disposal cost difference ($ kgHM^{-1})	300	200	100	298	438	±70
MOX fuel fabrication ($ kgHM^{-1})	700	1500	2300	302	434	±66
Interim fuel storage ($ kgHM^{-1})	300	200	100	310	425	±57
Enrichment ($ SWU^{-1})	150	100	50	338	404	–29 +36
Spent fuel burnup (MW · d · kgHM^{-1})	33	43	43	313	368	–54
Fresh fuel burnup (MW · d · kgHM^{-1})	53	43	43	350	368	–18
Laser enrichment	Yes	No	No	329	368	–39
Discount rate (% year, real)	8	5	2	353	380	–15 +13
LEU fuel fabrication ($ kgHM^{-1})	350	250	150	359	376	±8
Premium for recovered uranium:						
Conversion ($ kgU^{-1})	5	15	25	362	373	±5
Enrichment ($ SWU^{-1})[b]	0	5	10	364	371	±3
Fuel fabrication ($ kgHM^{-1})	0	10	20	367	369	±1
Conversion ($ kgU^{-1})	8	6	4	367	369	±1

[a] Low: best case for reprocessing; high: worst case for reprocessing.
[b] SWU: separative work unit.

The solid central line in Fig. 1 shows the break-even uranium price using the central estimates given in Table 1 for other fuel cycle prices and parameters. The dotted lines in Fig. 1 labelled 'Monte Carlo' show the result of a calculation in which the values of other parameters are selected randomly from independent normal distributions, with the 5th and 95th percentiles defined by the low and high values given in Table 1. The outer dashed lines represent the result of setting *all* of the parameters equal to those we selected as either the best or the worst case for reprocessing.

For a reprocessing price of $1000 per kilogram of heavy metal (kgHM) (in this paper, dollars ($) always refers to US dollars), the break-even uranium price is about $370 kgU^{-1} for central estimates of the other parameters. This is roughly eight times the current uranium price, and a level at which the available uranium resources would probably be sufficient to sustain a once-through fuel cycle for 100 years or more, even with substantial growth (see below). Even the lower boundary of the Monte Carlo calculation represents a break-even uranium price of about $220 kgU^{-1} for a $1000 $kgHM^{-1}$ reprocessing price. The reason that uranium prices must increase so much to reach break-even is that the cost of purchasing uranium is a small fraction of the overall fuel cost in the once-through fuel cycle.

Table 2 shows the results of break-even calculations for selected cost parameters, holding the uranium price at $50 kgU^{-1} and setting other costs equal to the central values listed in Table 1. If the uranium price is $50 kgU^{-1}, the reprocessing price would have to be reduced to below $420 $kgHM^{-1}$ in order for reprocessing to be cost effective. Achieving such a low reprocessing price would be an extraordinary challenge, particularly for privately owned facilities that must pay both taxes and higher costs of money on invested capital.

TABLE 2. BREAK-EVEN PRICES OF SELECTED PARAMETERS FOR DIRECT DISPOSAL VERSUS REPROCESSING AND RECYCLING IN LWRs, ASSUMING A URANIUM PRICE OF $50 kgU^{-1} AND CENTRAL VALUES FOR THE OTHER PARAMETERS

Parameter	Central estimate	Break-even value	Break-even / Central
Disposal cost difference ($ $kgHM^{-1}$)	200	630	3.2
Interim spent fuel storage ($ $kgHM^{-1}$)	200	780	3.9
Enrichment ($ SWU^{-1})	100	1200	12
Reprocessing ($ $kgHM^{-1}$)	1000	420	0.42
Uranium ($ kgU^{-1})	50	370	7.4

Table 1 also gives the change in the break-even uranium price when each of the parameters is varied from our central estimate to the worst and best case estimates. The parameters that have the largest impact on the break-even uranium price are reprocessing price, difference between the disposal costs for spent fuel and HLW, MOX fuel fabrication price and cost of interim storage of spent fuel.

Figure 2 shows the additional electricity cost associated with reprocessing and recycling, compared with direct disposal of spent fuel, as a function of uranium price, for several reprocessing prices, with other fuel cycle cost parameters set at their central estimates. At a reprocessing price of \$1000 $kgHM^{-1}$ and a uranium price of \$50 kgU^{-1}, reprocessing increases the cost of electricity by 1.3 mill · $(kW \cdot h)^{-1}$, or about \$10 million per year for a typical 1 GW(e) LWR. If the reprocessing price is \$1500 $kgHM^{-1}$, the cost penalty would rise to about 2.4 mill · $(kW \cdot h)^{-1}$.

2.2. Reprocessing price

Unlike markets for uranium and enrichment services, for which published prices are widely available, virtually all aspects of the economics of reprocessing are considered proprietary information. Our estimates are therefore based on the limited information that is available from the reprocessors, other

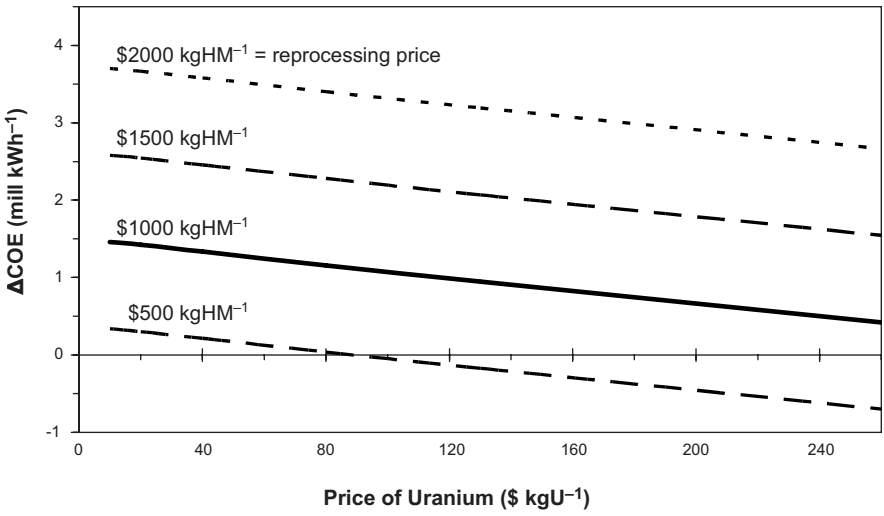

FIG. 2. The additional cost of electricity (ΔCOE, mill · $(kW \cdot h)^{-1}$) for the reprocessing–recycle option, for reprocessing prices of \$500, \$1000, \$1500 and \$2000 $kgHM^{-1}$, compared with the cost of electricity for the direct disposal option, as a function of the price of uranium (\$ kgU^{-1}).

studies and press reports. Only two companies outside the former Soviet Union operate large commercial reprocessing plants at present: COGEMA, now part of the AREVA group, which operates the UP2 and UP3 plants in France; and British Nuclear Fuels Limited (BNFL), which operates the Thermal Oxide Reprocessing Plant (THORP) in the United Kingdom. More is known about the costs at THORP because of the extended debates which have surrounded that facility.

THORP cost some $5.9 billion to build [8]. While there has been considerable controversy over its reprocessing capacity (arising from its frequent failure to meet targets), we will assume a figure of 800 tonnes of heavy metal per year (tHM a^{-1}). BNFL has not disclosed THORP's operating costs, but stated that a similar plant would cost some $560 million per year to operate [2]. BNFL subsequently asked for additional payments from customers to cover higher than expected capital and operating costs [9]. Nevertheless, to be conservative, we will rely on this early BNFL estimate.

Both the THORP and UP3 plants were built with very favourable financing arrangements — pay-ahead contracts from utility customers paid essentially the entire capital cost over a 10 year 'baseload' period. Recovering a capital cost of $5.9 billion over 10 years (without interest) would contribute $740 $kgHM^{-1}$ to the reprocessing cost. Including operational costs of $700 $kgHM^{-1}$, startup costs equal to one year of operational costs, and refurbishment and decommissioning costs of $100 $kgHM^{-1}$, the total reprocessing cost is about $1800 $kgHM^{-1}$. Indeed, BNFL figures (adjusted for inflation) indicate that baseload contracts amounted to about $2300 $kgHM^{-1}$ [8], which is consistent with expected costs plus a fee of about 25%.

The cost of reprocessing at new facilities with capital and operating costs comparable to THORP would depend crucially on how they were financed. Using the financing assumptions given in Ref. [10], a government owned facility would have a total reprocessing cost of about $1350 $kgHM^{-1}$; a private facility with a guaranteed rate of return like that which pertains to regulated utilities would have a cost of roughly $2000 $kgHM^{-1}$; and a private facility with no guaranteed rate of return would have a cost of over $3100 $kgHM^{-1}$ — all for the same capital and operating costs estimated for THORP.

Costs and baseload contract prices for the UP3 plant, built at roughly the same time to meet essentially the same market, have been reported to be generally similar to those for THORP, although much less detail is available. Costs for the most recent large reprocessing plant, the Rokkasho-Mura plant nearing completion in Japan, have been much higher. The capital cost of the Rokkasho-Mura plant is now expected to be roughly $18 billion, and the operating cost is expected to be over $1.4 billion per year [11] — both about three times the THORP costs.

Post-baseload contracts for THORP and UP3 were reportedly concluded in 1989–1990 at prices in the range of \$1000–\$1500 kgHM^{-1} [2, 3, 8, 12, 13]. More recently, prices offered for new reprocessing contracts have reportedly fallen to \$600–900 kgHM^{-1} [13], representing the operational cost plus a small profit. These low prices are only possible because recovery of capital is no longer included, and therefore they do not represent sustainable prices for reprocessing services.

To summarize, the \$1000 kgHM^{-1} reprocessing price we have used as our central estimate is quite conservative. For facilities with capital and operating costs comparable to those of THORP, costs in this range could only be achieved for facilities whose capital cost has already been paid off, or for those that are government financed. If, as seems likely, finance for future plants would have to be raised on private capital markets, a price of \$1000 kgHM^{-1} would require a more than 50% reduction in the capital and operating costs even for entities with a guaranteed government regulated rate of return.

Can the cost of reprocessing be reduced substantially? The Plutonium Redox Extraction (PUREX) process used in existing facilities has been perfected over more than five decades. While refinements are possible (and ongoing), it seems unlikely that dramatic cost reductions could be achieved using this or similar technologies. Although some argue that costs could be reduced using the experience gained from existing plants, very substantial reductions would be needed just to get to our assumed \$1000 kgHM^{-1} cost, even for government financed facilities and especially for the more likely future case of privately financed facilities. Moreover, increasingly stringent environmental and safety regulations will place countervailing pressures on costs. According to a recent report to the French Government, building a new plant similar to UP3 would cost \$6 billion — the same as the original plant [14].

A wide range of alternative chemical separation processes have been proposed over the years. Recently, attention has focused on electrometallurgical processing or 'pyroprocessing'. A 1996 review by a committee of the US National Academy of Sciences, however, concluded that the cost estimates provided in studies of the processes in the mid-1990s were "inexplicably low", that "it is by no means certain that pyroprocessing will prove more economical than aqueous processing", and that the costs of current plants such as THORP and UP3 "provide the most reliable basis for estimating the costs of future plants" [10]. More recently, official reviews have concluded that such techniques are likely to be substantially more expensive than traditional aqueous reprocessing, with a nominal estimate of \$2000 kgHM^{-1} (2.5 times higher than their nominal estimate of \$800 kgHM^{-1} for traditional reprocessing) in two of the most recent analyses [15, 16].

To summarize, while future technological developments hold some promise, it does not appear likely that within the next few decades the cost of reprocessing, including payback of the capital costs of facilities (probably at commercial interest rates), will be reduced to prices that would allow reprocessing to compete economically with uranium at the prices likely to pertain for most of this century. Indeed, it is possible that costs could increase — as suggested by the remarkable increase in the cost of the Rokkasho-Mura plant compared with THORP and UP3 — driven by the costs of meeting more stringent environmental and safety requirements.

2.3. Waste disposal cost difference

The next most important parameter is the savings resulting from treatment and disposal of reprocessing wastes, as compared with direct disposal of spent fuel. Permanent geological disposal of spent fuel and HLW has not been demonstrated, and approaches to waste disposal vary considerably from country to country, making cost estimates highly uncertain [17].

The US geological disposal programme has prepared the most detailed public analyses of any programme in the world. The most recent cost estimate for the US repository programme is $57.5 billion ($2000), of which $41.8 billion is for the disposal of 83 800 tHM of civilian spent fuel [18]. This is financed by charging utilities a fee of 1 mill · (kW · h)$^{-1}$, which is equivalent to about $370 kgHM^{-1} at the time of discharge.[2] With interest, this fee is expected to be sufficient to fund the full costs of transport to the repository, encapsulation and disposal of the spent fuel, including all future repository construction and operation costs [19].

Cost estimates produced by other countries for the disposal of spent fuel are roughly comparable. Sweden, for example, released a cost estimate in 1998 of $300–$350 kgHM^{-1} [20]. While it remains possible that these total cost estimates will continue to grow in the future, $400 kgHM^{-1} at the time of discharge is a reasonable benchmark for the total disposal cost of spent fuel. Thus, our central estimate of $200 kgHM^{-1} for the cost difference implies that reprocessing would reduce waste disposal costs by 50%.

Spent fuel and HLW differ in a number of ways that could affect disposal costs. The most important characteristics are the heat, volume and mass of the waste, and the number of waste packages to be handled.

[2] In 2003 dollars, assuming a burnup of 43 000 MW · d · tHM^{-1}, a net efficiency of 33%, a core residence time of four years and discounting at a real rate of 0.05 a^{-1}.

The heat output from waste packages determines how close to each other they can be placed while remaining within the repository's design temperature constraints. Thirty years after discharge, the heat output from the vitrified HLW is about 70% of the heat output of the original spent fuel — and the heat output of the HLW declines more rapidly than that of the spent fuel thereafter [10, 17]. This reduction in heat output at 30 years may offer even greater packing efficiencies, as the spaces between HLW packages could be left empty at first, while additional canisters were emplaced for the next 60 years, during which time another fourfold reduction in heat output would take place. New waste packages could then be emplaced between the first canisters, while remaining within the original heat limits. Although a similar strategy could be pursued with spent fuel, it does not offer as dramatic a benefit because spent fuel cools more slowly than HLW.

Waste volume and mass affect waste packaging and transportation costs. The volume of vitrified HLW waste is roughly one quarter the volume of the original spent fuel; including packaging for geological disposal, the total volume per kilogram of the original heavy metal ranges upwards from roughly about half that for the HLW. Hence, reprocessing might reduce volume related costs by as much as 50%.

Some costs increase with the number of items handled, for example, fuel assemblies or HLW canisters to be loaded into waste packages, and waste packages to be emplaced. A NIREX study estimated that each HLW waste package would hold two canisters of HLW, each containing HLW from the reprocessing of 1.2 tHM of spent fuel [21]. Thus, there would be 0.8 HLW canisters and 0.4 waste packages per tHM for the reprocessing option, compared with 2.2 fuel assemblies and 0.54 waste packages per tHM for direct disposal, for an overall reduction in item related costs of about 30% [22].

We can obtain a rough idea of how much reprocessing might reduce waste disposal cost by dividing costs into components that are affected by various waste disposal characteristics and assigning notional reduction factors for the disposal of HLW rather than spent fuel. In the case of the US Yucca Mountain repository, heat related costs (construction of repository and drip shield) amount to 19% of total programme costs; those related to volume, mass or number of items (repository emplacement operations and monitoring, waste package fabrication and transportation) are 53%, and other costs (siting, licensing, design and engineering) contribute 28% [18]. We assign a fourfold reduction factor for heat related costs and costs not related to waste form (corresponding to a potential fourfold increase in the amount of fuel that could be emplaced in the repository), and a twofold reduction factor for costs related to volume, mass or number of items.

The above discussion does not include the management and disposal of ILW and LLW from reprocessing. BNFL has permission from the UK Government to address the cost of LLW disposal through 'substitution' — adding a small amount of HLW to the amounts sent back to customers, instead of returning the LLW. BNFL hopes to receive similar permission for ILW, and if this was granted, the total amount of HLW returned to each customer would be roughly 20% higher than the amount generated by the reprocessing of the spent fuel of that customer [6]. If reprocessors are required to return all ILW and LLW, the costs of management of these wastes would be higher. We therefore assume that total disposal costs are 20% greater than the cost of HLW disposal alone. Applying this and the factors listed above results in an overall cost reduction of 55% due to disposal of reprocessing wastes rather than spent fuel, which corresponds well with our central estimate of \$200 kgHM^{-1} for the cost savings due to reprocessing. Given the large uncertainties in such estimates, we have used a range of \$100–\$300 kgHM^{-1} for the cost savings.

A 1993 OECD Nuclear Energy Agency (OECD/NEA) study compared the estimated repository costs for many countries (considering only encapsulation and disposal costs) and found that the weighted average cost was 57% less for disposal of HLW compared with spent fuel [17]. A recent French study offers substantially lower figures for disposal costs (\$80 kgHM^{-1} for HLW and \$130 kgHM^{-1} for spent LEU fuel) [1], but the percentage reduction for reprocessing (40%) is roughly in line with our central estimate (50%). A recent review of future fuel cycle options by a group advising the USDOE estimated a cost of \$200 kgHM^{-1} for disposal of HLW compared with \$300 kgHM^{-1} for spent fuel [16], consistent with the low end of our range for the cost difference. An OECD/NEA review of transmutation technologies also provided estimates that are consistent with the low end of our range.[3]

We have assumed that spent MOX fuel is not reprocessed and that the disposal costs are equal for spent MOX and LEU fuels of equal burnups. Most countries that now recycle plutonium do so only once, because of the buildup of undesirable isotopes in spent MOX fuel. The heat output of spent MOX fuel is much higher than that of spent LEU fuel — 2.2 versus 0.7 W kgHM^{-1} 50 years after discharge, for a burnup of 43 MW · d · kgHM^{-1} [23]. The greater heat

[3] The central estimates in Ref. [15] were \$400 000 m^{-3} for HLW conditioning and disposal and \$210 000 m^{-3} for spent fuel. Converting these to tons of original spent fuel using a relatively low estimate of 0.8 m^3 tHM^{-1} for HLW and a relatively high estimate of 2 m^3 tHM^{-1} for spent fuel, we have \$320 kgHM^{-1} for HLW and \$420 kgHM^{-1} for spent fuel, or a disposal cost difference of \$100 kgHM^{-1}.

output of spent MOX fuel should result in substantially higher disposal costs. If, for example, disposal of spent MOX cost \$400 $kgHM^{-1}$ more than spent LEU (twice the central value of \$400 $kgHM^{-1}$ for LEU), the break-even uranium price would increase by \$26 kgU^{-1}. If, on the other hand, spent MOX fuel is reprocessed and the recovered plutonium used in a 'self-generated recycle' mode, the total heat output from the HLW from that fuel cycle is higher, per unit of electricity generated, than that of the once-through cycle for the first 50 years after discharge from the reactor [24], negating much of the cost advantage for disposal of HLW compared with spent fuel.

2.4. MOX fuel fabrication price

The principal cost in using recovered plutonium is the price of fuel fabrication. Like reprocessing, fabricating MOX fuel is expensive because it requires large capital-intensive facilities and highly trained personnel. It is substantially more expensive than fabricating LEU fuel, primarily because of the safety requirements resulting from the much higher radiotoxicity of plutonium, and also because of the greater safeguards and security requirements required when handling weapons-usable material. As with reprocessing, the industry is dominated by a small number of firms (COGEMA, BNFL and Belgonucléaire), and virtually no official information on costs and prices is publicly available.

Again, because of the public controversies surrounding it, most is known about BNFL's Sellafield MOX plant (SMP), designed for a capacity of 120 $tHM \cdot a^{-1}$. SMP is officially estimated to have cost \$540 million [25]; when the cost of financing over the prolonged construction period and the subsequent delays in gaining approval are included, the cost increases to about \$750 million [8]. Similarly, Siemens' 120 $tHM \cdot a^{-1}$ plant at Hanau, Germany, which was built but never operated, reportedly cost roughly \$750 million [26]. In 1993, the USDOE estimated that the overnight cost of building a facility with a capacity of 100 $tHM \cdot a^{-1}$ in the USA would be \$440 million, or about \$550 million in 2003 dollars [27].

Current estimates for new plants in Japan and the USA are substantially higher. The overnight cost of building a MOX plant in the USA for disposition of excess weapons plutonium is currently estimated at over \$1 billion (not counting over \$300 million in R&D and pre-capital expenses, and another \$500 million for contingencies) [28]. A portion of the cost of this facility will go to removing gallium and other impurities from weapons plutonium before it is fabricated into MOX fuel, but even if this represented 30% of the total, the remaining overnight cost would be \$700 million. Similarly, the Rokkasho MOX

plant (RMP) in Japan, with a planned capacity of 130 tHM · a^{-1}, is expected to cost roughly $1 billion.

The operating costs of existing MOX plants have not been published. One group has estimated the operating costs of SMP at roughly $50 million · a^{-1} [29]. This is consistent with an analysis which concluded that operations costs in a facility of this kind would amount to $560 $kgHM^{-1}$ [30]; with the low end of an OECD/NEA estimate that the operating costs of such facilities are in the range of 10–25% of their capital costs [15]; and with annual operating costs (including an annuity for decommissioning) of $76 million · a^{-1} estimated in the 1993 USDOE study [27]. The operating costs for the planned US MOX plant are expected to be about $100 million · a^{-1} [28], which would be consistent with the earlier USDOE estimate if 30% of the operating cost goes to purification of weapons plutonium.

If a plant with a reported capital cost of SMP and a $560 $kgHM^{-1}$ operating cost succeeded in producing 100 tHM · a^{-1} throughout a 30 year life, the fabrication cost (with assumptions similar to those made above for spent fuel reprocessing plants) for a government financed facility would be about $1000 $kgHM^{-1}$; for a regulated private facility with a guaranteed rate of return, $1500 $kgHM^{-1}$; and for a private facility with no guaranteed rate of return, $2100 $kgHM^{-1}$. Transport of MOX fuel is a significant extra cost that must be added to these figures [30].

These costs apply for large fabrication campaigns of fuel of the same design. When a customer needs only a modest amount of MOX fuel, using different design parameters from those used by other customers, throughput suffers and per kilogram costs increase substantially. Per kilogram costs also increase if demand is not sufficient to keep a plant fully booked.

MOX fabrication prices, like costs, are not publicly divulged. For essentially all of the 1980s and 1990s, demand was higher than supply and prices were higher than one would expect on the basis of the costs given above. One review indicates that in the 1980s prices were $1900–$2400 $kgHM^{-1}$, while in the 1990s they were $2100–$2700 $kgHM^{-1}$ [13]. A USDOE survey of fabricators in 1993 reported a range of offers centring around $1850 $kgHM^{-1}$ [27]. Électricité de France enjoys lower prices of about $1200 $kgHM^{-1}$, as it buys very large quantities of a standard product and has a special relationship with COGEMA and its MELOX plant [1, 31]. German and Swiss utilities, on the other hand, report much higher prices, in the range of $3000–4000 $kgHM^{-1}$, which reflect their smaller purchases and the fact that much of their fuel has been fabricated in smaller, less automated, plants [3, 32]. With SMP now open and the supply of MOX fabrication services probably outstripping demand, prices may fall significantly — although MOX fabrication firms will still have substantial leverage to demand high prices, because the only alternative for

utilities with separated plutonium is to pay for plutonium storage at rates determined by the same firms.

MOX fuel fabrication is less mature than is PUREX reprocessing, leaving more scope for further technical improvement and cost reduction in the future. As one recent review [15] expressed it, "new plants would benefit greatly from the extensive experience gained during the last decades, thereby allowing them to simplify the plants, decrease their size, and reduce maintenance requirements." If, however, the focus remains on pellet based fuels, manufacturing each pellet to stringent standards will continue to be an expensive process, and there may be limits to the scope for cost reductions. Modern MOX fabrication facilities are already highly automated and designed to minimize maintenance. Moreover, as with reprocessing, there may be trends that would increase per kilogram costs over time — including not only increasing demands for more stringent safety and security precautions (a substantial factor driving the cost of the planned US MOX plant), but also customer demands to fabricate fuels with a higher design burnup, using plutonium recovered from higher burnup spent fuel or plutonium that has been stored for long periods and therefore has a higher americium content.

There may also be opportunities for new technologies that could simplify plutonium fuel fabrication and reduce cost, such as 'vibropak' fuels, in which the plutonium and uranium powders are packed into the fuel pins by vibration, with no pellet manufacturing required. Further development is required to determine whether such approaches can offer substantial MOX fuel cost reductions, and whether they can be used in existing light water reactors or only in plants designed for their use.

Overall, our central estimate of \$1500 $kgHM^{-1}$ is low with respect to recent prices, but reasonable for a future world in which supply and demand are balanced and prices more closely reflect production costs. Our \$700 $kgHM^{-1}$ lower bound would require either very substantial technological innovation or sales from facilities whose capital costs are already amortized, and which therefore do not reflect a long run sustainable cost for providing the service. The \$2300 $kgHM^{-1}$ upper bound is in the range of prices already charged at existing facilities and could reflect future prices if societal and customer demands drive costs higher in the future.

In many cases, there are additional costs to a reactor operator associated with using MOX rather than LEU fuel, which, to be conservative, we have not included in this analysis. Firstly, MOX fuel is often licensed to lower burnups than LEU fuel, which would require reactor operators to shut down for refuelling more often. Secondly, because fresh MOX fuel contains weapons-usable plutonium, it requires more security than would fresh LEU fuel, often imposing additional costs. (In some cases, fresh MOX fuel is simply placed with

spent fuel at the reactor site, without any additional facilities or security arrangements, on the assumption that it would be difficult and dangerous for attackers to remove it from the pool.) Thirdly, in a number of countries there are substantial political concerns over the use of MOX and additional licensing requirements for reactors for which the operators wish to use both MOX and LEU fuels.

Hence, the value of MOX fuel (if there were an open market allowing utilities to choose their fuels) would not be equal to that of LEU fuel of equal design burnup, as is assumed here. In the case of the US programme for disposition of excess weapons plutonium, for example, convincing US utilities to use the MOX fuel required it to be offered at a price some 40% below the price of LEU fuel of equivalent energy value [33] — equivalent to increasing the net fabrication price of the MOX fuel by several hundred dollars per kilogram.

Fourthly, we have assumed a reprocessing and recycling system that is operating efficiently and in balance, so that there are no charges for storing plutonium or for removing americium. Commercial rates for these services are estimated at \$1000–\$2000 $kg^{-1} \cdot a^{-1}$ for storage and \$10 000–\$28 000 kg^{-1} for americium removal. Including several years of plutonium storage and one round of americium removal would increase the effective cost of MOX fabrication by \$1000–3000 $kgHM^{-1}$, and would increase the break-even uranium price by \$80–250 kgU^{-1}.

2.5. Cost of interim spent fuel storage

For reactor operators who choose reprocessing, interim storage of spent fuel for decades is not required. Interim storage generally is required for direct disposal, however, as repositories are not expected to be available for several decades. We have therefore included interim storage as an extra cost for the direct disposal fuel cycle, although new reactors are being built with pools able to accommodate storage of all the fuel they will generate in their lifetime, reducing or eliminating this extra storage cost. Costs of interim storage can vary significantly depending on, for example, the technology chosen, whether fuel is to be transported to a centralized site or kept at reactor sites (and, if at a reactor site, whether the reactor is operating), or whether taxes or other payments must be made to local, regional or national governments.

Storage in dry casks is a well established technology for storing spent fuel for decades with minimal operating costs. In the USA, total in-advance costs to establish a new dry storage facility at a reactor site (which are largely fixed, regardless of the amount of spent fuel to be stored) are estimated at roughly \$10 million [34, 35]. Costs to purchase and load the casks — including labour,

consumables and decommissioning — are estimated at \$70–\$90 kgHM^{-1} [34]. The principal operating costs are providing the security and safety monitoring needed to maintain the Nuclear Regulatory Commission licence for the facility. For storage sites co-located with operating reactors, many of these costs can be charged to the reactor operation, and the net additional operating costs are estimated to be about \$800 000 a^{-1} (largely independent of the amount of spent fuel to be stored) [34]. Total costs for 40 years of dry cask storage of 1000 tHM at an operating reactor site in the USA would be in the range of \$100–120 kgHM^{-1} (with operational costs discounted at 3% a^{-1}).

For storage at shut down reactors or independent sites, the costs of maintaining the licence, including security and safety personnel, must be attributed entirely to the storage facility, making its operational cost substantially higher. For shut down reactors with all their spent fuel in dry storage, operating costs are estimated to be \$3.3–4.4 million per year [34, 35]. Total cost for forty years of storage in this case would range from \$150 to \$200 kgHM^{-1}. A large centralized facility could spread these operations costs over a larger amount of spent fuel, but there would be additional in-advance costs for transportation to the centralized site.

Somewhat higher costs have been estimated in Japan; in a 1998 study, total discounted costs for 40 years of storage in a 5000 t centralized dry cask facility were estimated at \$280 kgHM^{-1} [36]. These costs do not include benefits that may be paid to the local community to build public acceptance and gain government approval, which could in some cases be substantial.

We have chosen \$200 kgHM^{-1} as our central estimate of interim storage costs, which is comparable to the discounted cost of independent dry cask storage in the USA at small facilities. The lower estimate of \$100 kgHM^{-1} is close to the current cost of at-reactor dry cask storage in the USA, while the upper limit of \$300 kgHM^{-1} may represent the cost at independent facilities, including payments to nearby communities.

2.6. Other fuel cycle prices and parameters

Other factors — enrichment and LEU fuel fabrication prices, premiums for the use of recovered uranium, fuel burnup and discount rate — are less important when comparing the economics of direct disposal with those of reprocessing and recycling in thermal reactors.

Long term contract prices for enrichment services fell from earlier levels of over \$100 SWU^{-1} (in then-year dollars) to \$85 SWU^{-1} by late 1999, only to increase back to some \$110 SWU^{-1} in 2001 [37]. The gap between long term and spot SWU prices has declined substantially; in the first half of 2004, the spot price in the USA was about \$110 SWU^{-1} [38]. One projection in mid-2003

suggested that SWU prices in long term contracts would probably remain in the range of \$105 SWU^{-1} for a few years, and then rise slightly towards the end of the decade [39]. Production costs of gas centrifuge enrichment are below \$80 SWU^{-1}, and can be expected to decrease as the next generation of centrifuges is installed [13]. The OECD/NEA has estimated that enrichment prices in the short to medium term will be in the range of \$80–\$120 SWU^{-1}; over the longer term, the NEA reports that new facilities using advanced processes might reduce costs to \$50 SWU^{-1} [40]. We have chosen a central estimate of \$100 SWU^{-1}, with a high of \$150 SWU^{-1} and a low of \$50 SWU^{-1}, allowing a somewhat broader range of possibilities.

The OCED/NEA projects LEU fabrication prices in the short to medium term at \$200–\$300 $kgHM^{-1}$ [40]. A previous survey by a US National Academy of Sciences committee chose a central estimate of \$250 $kgHM^{-1}$ [27]. This central estimate is somewhat higher than recent prices in the US market, but somewhat lower than most prices in the European market [41]. We have chosen a central estimate of \$250 $kgHM^{-1}$, with a low of \$150 $kgHM^{-1}$ and a high of \$350 $kgHM^{-1}$, again allowing a somewhat broader range of possibilities than the NEA projections. The technology of LEU fuel fabrication is mature and the safety and health impacts modest, so it appears unlikely that this price will change substantially in the future.

Uranium recovered from reprocessing contains undesirable isotopes such as ^{232}U (whose radioactive daughter products emit penetrating gamma rays) and ^{236}U (which is a neutron absorber, increasing the enrichment required to achieve a given design burnup). Because of the higher radioactivity of recovered uranium, firms charge higher prices for its conversion, enrichment and fabrication. If natural uranium is cheap, recovered uranium has no value at all. Indeed, most utilities have not found it worthwhile to recycle recovered uranium, and the vast majority of all the uranium recovered from the reprocessing of LWR fuel remains in storage. Market estimates of the relevant premiums are therefore somewhat uncertain [13]. We have chosen central estimates of \$15 kgU^{-1} for conversion, \$5 kgU^{-1} for enrichment and \$10 kgU^{-1} for fuel fabrication [15]. Recovered uranium would become more valuable if laser isotope enrichment is commercialized, because laser enrichment would remove the undesirable isotopes present.

Conversion of uranium from U_3O_8 to UF_6 for enrichment is a minor cost element. We have chosen a central estimate of \$6 kgU^{-1}, with a range of \$4–\$8 kgU^{-1}. The OECD/NEA projects conversion prices in the short to medium term in the range of \$3–\$8 kgU^{-1}, nearly identical to our range [40].

Recycle becomes less attractive economically as the burnup of the reprocessed spent fuel increases, because the isotopic quality of the recovered plutonium and uranium declines [42]. On the other hand, increased design

burnup of the fresh fuel makes recycle more attractive, because the additional enrichment required makes LEU relatively more expensive [27]. We have taken, as our best case for reprocessing, the fabrication of MOX with a design burnup of 53 MW · d · kgHM^{-1} using plutonium recovered from spent fuel with a burnup of 33 MW · d · kgHM^{-1}. Our central and worst case estimates have spent and fresh fuel burnups of 43 MW · d · kgHM^{-1}.

All fuel cycle services are discounted to the time of fuel discharge. We use a central value of 0.05 a^{-1} for the real discount rate, which is roughly the debt rate available to a regulated utility with a guaranteed rate of return. We adopt a range from 0.02 to 0.08 a^{-1}, which has a modest effect on our calculations. The geological disposal cost difference is the net present value at the time of fuel discharge.

3. DIRECT DISPOSAL IN LWRs VERSUS RECYCLE IN FAST REACTORS

From the dawn of the nuclear age, the nuclear industry believed that uranium was relatively scarce and that the number of reactors would grow rapidly, leading to rapidly rising uranium prices. Hence, the industry projected that there would be a rapid transition from LWRs, which rely heavily on fissioning the rare ^{235}U isotope, to fast reactors (FRs), which more efficiently transform ^{238}U into plutonium, which is either fissioned in place or recycled via fuel reprocessing. The recycling of plutonium in LWRs was seen only as a temporary expedient until the transition to primary reliance on FRs began.

The transition to FRs has taken much longer than was once expected. Uranium has turned out to be abundant and cheap, nuclear energy has grown much more slowly than expected, and FRs have been more expensive and problematic than anticipated. As a result, only the Russian Federation, India and Japan still have near-term plans for commercializing FRs. The Russian Federation is the only country that operates a commercial scale FR (the BN-600); construction of a slightly larger plant, the BN-800, has recently resumed after having been largely on hold since the 1980s. France, Germany, the UK, the USA and other countries have terminated FR commercialization efforts, although in a number of countries longer term R&D continues. More recently, as part of efforts to develop advanced systems for a possible future resurgence of nuclear energy, FRs have again received increased attention as a long term option [43].

3.1. Break-even prices and difference in cost of electricity

At what uranium price would recycling in FRs become economic? To answer this question we must account not only for differences in fuel cycle costs but also for the fact that the capital costs of FRs and LWRs may be different. (We have assumed for the sake of simplicity that the non-fuel operations and maintenance costs of LWRs and FRs would be the same; this is a generous assumption, as studies have suggested that FRs would have higher non-fuel O&M costs [44, 45]). The estimated capital costs of sodium cooled FRs have typically been up to 50% higher than those of LWRs. As with reprocessing and MOX fuel fabrication plants, we explore three different financing arrangements for this additional capital cost, appropriate for facilities owned by: a government, a regulated utility and an unregulated electricity producer.

Figure 3 shows the break-even uranium price as a function of the difference in capital cost between LWRs and FRs for the three financing arrangements. The characteristics of the generic FRs are given in Table 3. Table 4 gives our central, low and high estimates for the various cost parameters used to produce these graphs, along with the sensitivity of the outcome to changes in each parameter. The dotted lines in Fig. 3 represent the results of a Monte

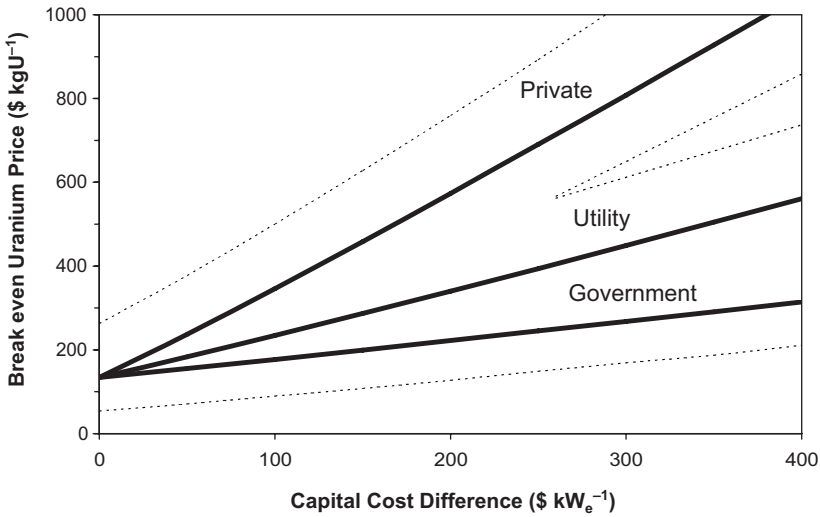

FIG. 3. Break-even uranium price for LWRs with direct disposal and for FRs, as a function of capital cost difference, for reactors financed by government, a regulated utility and a private electricity producer, for central values of other parameters (Table 4).

TABLE 3. CHARACTERISTICS OF THE GENERIC FAST REACTOR [16]

Parameter	Low	Central	High
Breeding ratio	1.0	1.12	1.25
Annual blanket loading (kgHM · MW(e)$^{-1}$ · a^{-1})	19.0	25.5	31.9
Annual core loading (kgHM · MW(e)$^{-1}$ · a^{-1})		11.5	
Residence time of core elements (a)		3.0	
Residence time of blanket elements (a)		3.2	
Plutonium fraction in core		0.246	
Make-up fraction in blanket		0.024	
Efficiency (net MW(e) · MW(th)$^{-1}$)		0.38	

Carlo calculation in which these parameters are selected randomly from independent normal distributions, with the 5th and 95th percentiles defined by the low and high values given in Table 4.

We have chosen the reactor owned by a regulated utility with a guaranteed rate of return as the reference case for the sensitivity analysis in Table 4. This may be a generous assumption, given the global trend towards increased reliance on privatized power plants operating in competitive electricity markets. While there remain some major countries where power plants are built and operated by a government owned monopoly, this is not likely to be the case in most countries that will have to consider the choice between once-through LWRs and FRs with recycling.

As shown in Fig. 3 and Table 4 for the case of a utility owned reactor, if the capital cost of FRs is \$200 kW(e)$^{-1}$ greater than that of LWRs, and other parameters are held at their central values, FRs with recycling would not be economic unless the price of uranium rose to over \$340 kgU^{-1} — similar to our central estimate of the break-even price for recycle in LWRs. Differences in capital cost between FRs and LWRs are less important for government owned facilities and more important for private ventures; for a capital cost difference of \$200 kW(e)$^{-1}$, the break-even uranium price ranges from \$220 kgU^{-1} for the former to \$570 kgU^{-1} for the latter. Even if the capital cost of FRs is equal to that of LWRs (in which case the type of financing is irrelevant to the comparison), the break-even uranium price under the same assumptions is \$130 kgU^{-1} — a price that is unlikely to be seen for decades.

One assumption we have made in these calculations should be noted. Because there are currently hundreds of tonnes of separated plutonium in storage, we have assigned zero cost to the plutonium needed for the initial FR

TABLE 4. ESTIMATES OF FUEL CYCLE COSTS AND OTHER PARAMETERS, AND SENSITIVITY ANALYSIS FOR THE BREAK-EVEN URANIUM PRICE FOR DIRECT DISPOSAL IN LWRs VERSUS REPROCESSING AND RECYCLING IN FRs, FOR REACTORS OWNED BY A REGULATED UTILITY

Parameter	Parameter value			Break-even U price (central = \$340 kgU^{-1})		Change compared with central
	Low	Central	High	Low	High	
Capital cost difference (\$ kW(e))	0	200	400	134	560	–205 +221
Reactor owner	Govt	Utility	Private	222	574	–118 +234
Reprocessing cost (\$ $kgHM^{-1}$)	500	1000	2000	255	516	–85 +176
Enrichment (\$ SWU^{-1})	150	100	50	282	415	–58 +75
FR core fabrication (\$ $kgHM^{-1}$)	700	1500	2300	286	394	±54
FR breeding ratio	1.0	1.12	1.25	294	386	±46
Geological disposal cost difference (\$ $kgHM^{-1}$)	300	200	100	322	358	±18
LEU burnup ($MW \cdot d \cdot kgHM^{-1}$)	43	53	53	322	340	–17
Construction time (a)	3	6	9	326	355	±15
FR blanket fabrication (\$ $kgHM^{-1}$)	150	250	350	325	355	±15
LEU fuel fabrication (\$ $kgHM^{-1}$)	350	250	150	327	353	±13
Capacity factor (%)	90	85	80	328	353	±13
Pre-operational, contingency costs (%)	5	10	15	330	350	±10
Interim spent fuel storage (\$ $kgHM^{-1}$)	300	200	100	332	348	±8
Conversion (\$ kgU^{-1})	8	6	4	338	342	±2
Depleted uranium (\$ kg)	6	6	U price	340	341	+1

core. Past analyses have assumed that the cost of reprocessing LWR fuel to recover plutonium for the initial core would be charged to the cost of the FR, with the cost capitalized over the life of the reactor [46, 47]. This assumption may be more accurate, because if FRs are deployed in numbers large enough to make a substantial contribution to world electricity demand, existing stockpiles of separated plutonium will not be sufficient to start them up, and reprocessing of spent LWR fuel to provide the necessary plutonium would be needed. If the cost of reprocessing LWR fuel was \$1000 $kgHM^{-1}$ and each kilogram of LWR fuel provided roughly 10 g of plutonium, the cost of startup plutonium would be \$100 000 kg^{-1}. Accounting for savings in interim spent fuel storage and waste disposal costs (\$200 $kgHM^{-1}$ each) and the value of the recovered uranium (of the order of \$300 kgU^{-1} by the time FRs might be competitive), the net cost would be of the order of \$30 000 kg^{-1}. In that case, the plutonium for the startup fuel (the initial core plus one-third core for the first refuelling) would add \$340 $kW(e)^{-1}$ to the cost of the FR (highly enriched uranium (HEU) could be used for the initial core but the cost would be even higher).[4] The cost of the startup plutonium could be offset somewhat by the sale of excess plutonium generated during the operation of the reactor; this would reduce the net plutonium cost to about \$200 $kW(e)^{-1}$. [5] Thus, even if other FR capital costs are reduced to those of LWRs, the uranium break-even price would still be at our central estimate of about \$340 kgU^{-1}, for our central values of other parameters.

Table 5 gives the break-even values of several other price parameters for the case of a regulated utility owner, assuming a uranium price of \$50 kgU^{-1} and central values for other parameters. Note that reductions in the price of reprocessing alone cannot make FRs economic so long as the FRs remain \$200 $kW(e)^{-1}$ more expensive than LWRs.

Figure 4 shows the difference between the cost of electricity from FRs with recycling and LWRs operating on a once-through cycle, as a function of the price of uranium, for differences in capital cost ranging from \$0 to \$400 $kW(e)^{-1}$, assuming utility owned reactors and the other parameters set at

[4] The startup core and initial reload would require 46 kg $kW(e)^{-1}$ of HEU with an enrichment of about 25% ^{235}U. Assuming uranium, conversion and enrichment prices of \$50 kgU^{-1}, \$6 kgU^{-1} and \$100 SWU, respectively, the cost would be \$8300 kg^{-1} of HEU, equivalent to \$380 kW(e). Using the break-even price of uranium in our reference case (\$340 kgU^{-1}) would increase these costs to \$22 000 kg^{-1} and \$1000 $kW(e)^{-1}$.

[5] With a breeding ratio of 1.25, the FR produces surplus plutonium at a rate of 0.3 kg $\cdot$ $MW(e)^{-1} \cdot a^{-1}$; assuming a value of \$30 000 kg^{-1} and a discount rate of 0.05 a^{-1} over 30 a, and taking into account the plutonium recovered from the final core, the net present value at startup of the surplus plutonium is \$130 $kW(e)^{-1}$.

TABLE 5. BREAK-EVEN PRICES OF SELECTED PARAMETERS FOR DIRECT DISPOSAL IN LWRs VERSUS REPROCESSING AND RECYCLING IN FRs, ASSUMING A REGULATED UTILITY OWNER, A URANIUM PRICE OF \$50 kgU^{-1}, AND CENTRAL VALUES FOR THE OTHER PARAMETERS

Parameter	Central estimate	Break-even value	Break-even central value
Capital cost difference (\$ $kW(e)^{-1}$)	200	–95	
Disposal cost difference (\$ $kgHM^{-1}$)	200	3400	17
Interim spent fuel storage (\$ $kgHM^{-1}$)	200	4100	21
Enrichment (\$ SWU^{-1})	100	570	5.7
Reprocessing (\$ $kgHM^{-1}$)	1000	< 0	
Uranium (\$ kgU^{-1})	50	340	6.8

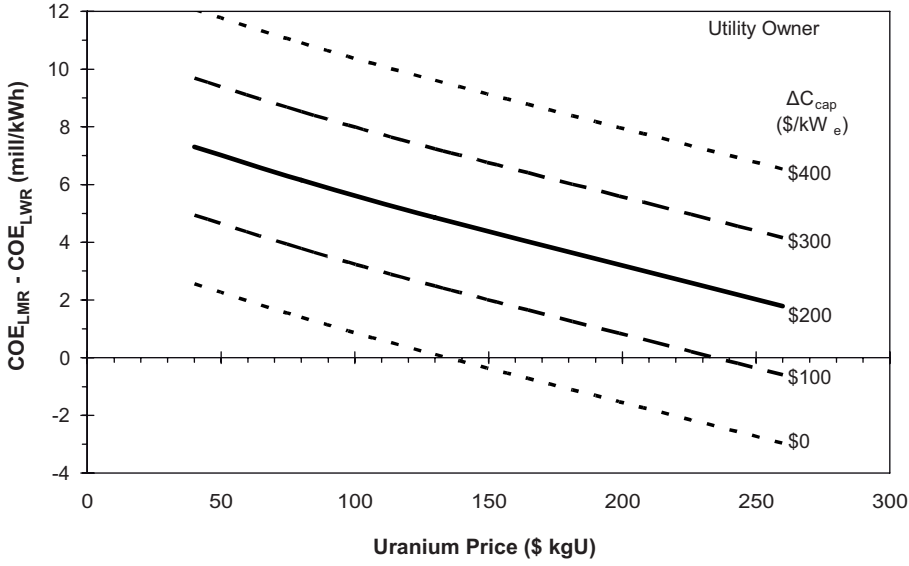

FIG. 4. Difference in the cost of electricity between an FR with recycling and an LWR with direct disposal as a function of the price of uranium, for differences in the initial capital cost of \$0, \$100, \$200, \$300 and \$400 $kW(e)^{-1}$, assuming utility ownership.

their central values. The electricity price for FRs will remain significantly higher than that for LWRs operating on a once-through cycle until the uranium price increases to at least several times its current level — a development that is not likely to occur for many decades to come.

This overall finding is broadly consistent with those of other recent studies. An OECD/NEA assessment found that the cost of electricity from FRs with recycle of plutonium and minor actinides would be 50% higher than that from LWRs operating on a once-through cycle [15]. The Generation IV Fuel Cycle Crosscut Group examined the fuel cycle contribution to electricity costs for different types of nuclear energy mixes throughout the twenty-first century, during which time they projected uranium prices to increase dramatically. Despite those projected increases (and despite looking only at fuel cycle costs, and therefore not including any increased capital cost of FRs), the costs for all the mixes that included FRs remained higher throughout the century than the price for electricity from once-through LWRs [16]. Similarly, a mid-1990s study by a committee of the US National Academy of Sciences concluded that the electricity cost of FRs would be substantially higher than that of once-through LWRs until uranium reached a price of well over \$250 kgU^{-1} (1992 dollars), even if reprocessing costs for LWR fuel and FR fuel were at the lower bounds given here [10].

3.2. Capital cost difference and related factors

The most sensitive parameters in this analysis are the difference in capital cost between FRs and LWRs, and the financing arrangements for capital costs. We have assumed a central value of \$200 $kW(e)^{-1}$ for the difference in capital cost, with a range from \$0 to \$400 $kW(e)^{-1}$. This range reflects past experience and peer reviewed estimates for the additional capital cost of FRs, and the expectation that there would be further progress in reducing FR costs.

The most recent FR designs in the USA and Western Europe were expected to be significantly more expensive than LWRs. The capital cost of the US Advanced Liquid Metal Reactor (ALMR) was estimated in the mid-1990s, shortly before the programme's termination, to be 20–30% higher than that of advanced LWRs (a difference of \$500–\$740 $kW(e)^{-1}$ in 2003 dollars) [27]. Similarly, the European Fast Reactor (EFR), after major reductions in various elements of capital cost compared with those of earlier designs, was expected to have a capital cost in series production of 20–30% higher than that of a comparable LWR [45, 48]. The Minister of Atomic Energy of the Russian Federation recently acknowledged that "life has proved that a VVER-1000 reactor [a modern Russian LWR] is one and a half times cheaper than a BN

[fast neutron] reactor; [LWRs] are cheaper, safer, and economically more viable" [49].

Some FR designers argue, however, that recent developments would make it possible to build FRs at a cost no higher than that of LWRs [50], and the Japanese FR programme, among others, has set capital cost equality with LWRs as an explicit goal [51, 52]. New FR concepts, such as lead cooled and gas cooled systems, are hoped by their advocates to have lower capital costs than those of traditional sodium cooled FRs [43]. The economic features of these concepts remain undemonstrated, however, and new thermal reactors are hoped by *their* advocates to have significantly lower capital costs than those of traditional LWRs.

Recent estimates of the cost of LWRs cover a broad range. Those based on actual experience tend to be over \$2000 kW(e)$^{-1}$ [53]. Estimates for future construction from independent assessments are in the range of \$1500–\$2000 kW(e)$^{-1}$ [15, 54], while reactor vendors project overnight capital costs of \$1000–1500 kW(e)$^{-1}$ [55]. If the LWRs that would compete with future FRs had a capital cost of \$1500 kW(e)$^{-1}$, a capital cost difference of \$0–\$400 kW(e)$^{-1}$ would correspond to a 0–27% premium for FRs — the high end comparable to that estimated in the most recently designed commercial systems, and the low end representing success in efforts to equalize capital costs. Our range is substantially more generous to FRs than that adopted in the most recent OECD/NEA assessment, whose nominal estimate for future FRs was \$400 kW(e)$^{-1}$ higher than that for future LWRs, with a range from \$150 to \$900 kW(e)$^{-1}$ higher [15].

As noted above, we used three different sets of assumptions about the financing for capital costs, corresponding to facilities owned by government, a regulated utility and a private venture. Our financing assumptions are identical to those in Ref. [10], and result in fixed charge rates of 0.058, 0.123 and 0.208 a^{-1}, respectively. Construction time (which is assumed to be the same for both types of reactor) enters into the calculation due to the interest paid on capital during construction; we use a central value of 6 a with a range of 3–9 a, and real average costs of money of 0.04, 0.064 and 0.139 a^{-1} for governments, utilities and private ventures, respectively. Finally, pre-operating costs and contingency funds are usually proportional to capital cost; we have assumed central values equal to 10% of overnight capital cost for both pre-operating costs and contingency funds, for both types of reactor, with a range from 5 to 15%.

3.3. Reprocessing and fuel fabrication prices

The break-even uranium price is also sensitive to the reprocessing price for FR fuel. For simplicity, we have chosen a central estimate for both the core

and blanket fuel of \$1000 kgHM^{-1}, with a range of \$500–\$2000 kgHM^{-1} — the same as that for reprocessing LWR fuel. This is a generous assumption, as reprocessing costs for higher burnup FR fuels with much higher plutonium content will generally be significantly higher. The recent OECD/NEA review, for example, posited a range of \$1000–2000–2500 kgHM^{-1} for core fuel and \$900–1500–2500 kgHM^{-1} for blanket fuel reprocessing (low–central–high values) [15]. The \$500 kgHM^{-1} lower bound of our range is intended to cover the possibility of substantial technological advance in the future. Our high value of \$2000 kgHM^{-1} is by no means an upper bound on the price of FR reprocessing, but if reprocessing turns out to be more expensive then there is little hope that uranium will reach the corresponding break-even price in the foreseeable future.

We have assumed a central estimate for the FR core fuel fabrication price of \$1500 kgHM^{-1}, with a range of \$700–\$2300 kgHM^{-1}. As with reprocessing, this is the same as for the MOX fuel fabrication price in the LWR recycling case. This also is generous, since FR core fuel will have much higher plutonium content and design burnup, which generally implies a higher fabrication cost. This price range is approximately equal to that employed in the recent OECD/NEA analysis for FRs using MOX fuels [15]. For metal fuels, where the OECD/NEA study assumed minor actinides would also be recycled with the plutonium, it was envisaged that core fuel fabrication would be more expensive (because of the extra cost of handling the more radioactive minor actinides), with a range of \$1400–2600–5000 kgHM^{-1}.

We assume that the price of blanket fuel fabrication is about the same as the price of LEU fuel fabrication for LWRs — a central value of \$250 kgHM^{-1}, with a range of \$150–\$350 kgHM^{-1}. This range appears again to be generous to the FR, as it is a factor of two lower than that used in the recent OECD/NEA assessment [15].

Future FR systems, such as some of those envisaged in the Generation IV initiative, might involve substantially different fuelling approaches, such as liquid fuels that would not require fabrication. Such approaches could have lower costs, but an accurate assessment will have to await further development of these technologies.

3.4. Other prices and parameters

We assume a central value of \$200 kgHM^{-1} for the difference between the disposal costs of spent LWR fuel and HLW resulting from the reprocessing of FR fuel, with a range of \$100–300 kgHM^{-1}. This is the same range used in Section 2, which again is generous to FRs, as one would expect that HLW from higher burnup FR core fuel would have higher activity and volume, increasing

disposal costs. (This factor is compensated for, however, by the fact that we have chosen the same cost of disposal for wastes from reprocessing the blanket fuel, which will have low burnup, and the core fuel, which will have high burnup.)

We assume a nominal FR breeding ratio of 1.125, with a range of 1.0–1.25. Electricity price increases with breeding ratio in our model, because more blanket material must be reprocessed each year. This result is an artefact of assigning a zero cost to the initial core fuel — and to excess plutonium produced by FRs. If startup fuel were assigned a substantial value, then higher breeding ratios could be more economical (but, as explained above, FRs would be less competitive with once-through LWRs).

After the initial core and first reload, FRs would only require depleted uranium to replace uranium that fissioned, was transformed into plutonium or was lost in processing. Many thousands of tonnes of depleted uranium (DU) already exist in the stored waste from uranium enrichment plants. As long as uranium demand is driven by LWRs, there will be little use for this DU, and its price will be low. We therefore assume a central DU price of \$6 kgU^{-1} — the price of converting the material from uranium hexafluoride. However, once uranium prices increase to the point that FRs become competitive, those holding stocks of DU may begin to assign a significant value to it. When demand for uranium begins to be dominated by FRs and stocks of DU begin to be drawn down, the price of DU should approach the price of natural uranium, because DU and natural uranium are almost perfect substitutes for use in breeder blankets. Even with such a high upper bound, the DU price has virtually no effect on the economics of FRs.

4. URANIUM PRICES AND RESOURCES

In the above analysis we have calculated the break-even uranium price — the price that would make reprocessing and recycling in LWRs or FRs economically competitive with LWRs operating on a once-through fuel cycle. In this section, we review past and estimated future uranium prices, estimates of the amount of uranium that is ultimately recoverable at a given price, and scenarios of uranium consumption during the next century. We conclude that the uranium price will probably remain below the break-even prices calculated in our reference cases above for the next 100 years, and that reprocessing and recycling in both LWRs and FRs will remain uneconomic for the foreseeable future, unless there are dramatic reductions in the price of reprocessing and fabrication of plutonium fuels, and, in the case of FRs, of capital cost.

Figure 5 shows selected uranium prices over the last 30 years. The real price paid by US reactor operators (the weighted average of deliveries under long and short term contracts) fell from a high of \$190 kgU^{-1} in 1982 to about \$28 kgU^{-1} in 2002 (in \$2003) [56]; prices in Europe were somewhat higher [57]. The spot market price for uranium has been considerably more volatile, falling from a high of \$300 kgU^{-1} in 1977 to a low of \$20 kgU^{-1} in 2000; in recent months, the spot price has seen a remarkable spike, as a result of the increasing realization that secondary supplies will not be able to meet the gap between production and demand indefinitely, coupled with unexpected events such as the flooding of a mine and the Russian decision to use more of their uranium internally. The spot price of over \$78 kgU^{-1} in August 2005 had doubled in less than two years, and was nearly twice the previous peak levels of the past 15 years [38].

The nuclear enthusiasm of the 1960s and 1970s, together with the rapid growth in electricity demand that was expected at that time, led utilities to order large numbers of reactors; expectations of a correspondingly rapid increase in uranium consumption led to a large price spike in the late 1970s. But the lower growth of electricity demand following the oil price shocks of the 1970s, coupled with the increase in nuclear costs and controversies following the Three Mile Island accident in 1979, led to the cancellation of most of these reactor orders, greatly reducing projected uranium demand and bringing the

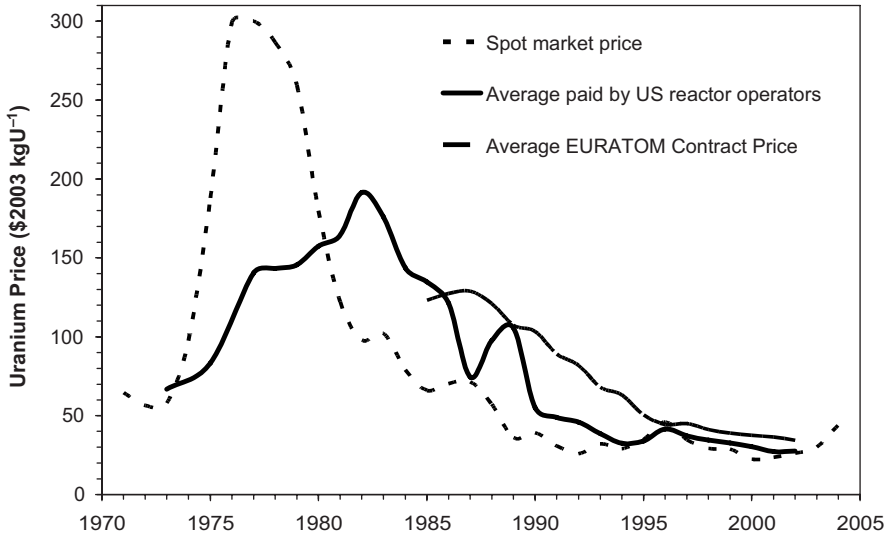

FIG. 5. Uranium prices, 1971–2004 [38, 56, 57].

price back down. During much of the 1990s, world uranium production was well below world consumption, as governments and utilities reduced their inventories (because of their increased confidence in the availability of uranium when they needed it); this additional supply from inventory sales (including the US–Russian HEU purchase agreement) reduced prices to a level below that necessary for production to equal consumption. In the last few years, however, there have been concerns about when these inventory supplies would become exhausted and whether mine production could increase quickly enough to meet demand. As a result, uranium prices have increased significantly, as just noted. Given the availability of the large quantities of uranium recoverable (once the relevant mines are brought on-line) at prices in the range of \$40–\$50 kgU^{-1}, it appears unlikely that the price would rise above this level for any sustained period over the next few decades (although temporary fluctuations such as the present one, when new mines have not yet come on-line to meet increased demand, are to be expected).

Longer term price predictions are notoriously difficult. Classical economic theory suggests that the price of non-renewable resources should rise steadily over time, as the fixed available stock grows scarcer and more costly resources have to be used. However, this model fails to take into account the ongoing discovery of additional resources and the development of improved technologies for identifying and extracting resources [58]. The amount of a mineral that can be recovered at a given cost of extraction can increase if technological improvements and discoveries of additional resources outpace the depletion of known deposits. This has, in fact, been the pattern throughout the last century for most minerals: real prices have fallen even while rates of consumption have increased. The history of copper production is illustrative: as a result of improved technology, the real price declined by a factor of 3.8 from 1900 to 2000, despite a 25-fold increase in demand [59] and a decline in the average ore grade from 2 to 0.85% [60]. There is little reason to expect that uranium prices, which have been following a similar trend, will reverse course and begin increasing steadily until far more of the resources available have been consumed than will be the case in the next few decades.

The most commonly cited estimates of uranium resources are those in the 'Red Book' [61]. The 2001 Red Book estimates that the total world 'conventional' resources available at less than \$130 kgU^{-1} amount to 16.2 MtU (the sum of 'reasonably assured resources', 'estimated additional resources' and 'speculative resources'). If inventories that have already been mined are included, the total rises to 17.1 MtU [62]. An international meeting sponsored by the IAEA in 2000 concluded that the total resources available in this category probably amount to 20 MtU [63].

Several points should be made about the Red Book estimates. Firstly, many countries do not report resources in the lower confidence and higher cost categories. For example, Australia, which has some of the world's largest uranium resources, does not estimate 'speculative' resources because its better characterized resources are so large already.

Secondly, this estimate is limited to 'conventional' resources, i.e. deposits where the uranium ore is rich enough to justify mining at the indicated price. In some cases, however, it may be attractive to produce uranium as a by-product. For example, ores with uranium concentrations as low as 4.5 ppm — less than twice the average abundance in the earth's crust — have been recovered as by-products from copper mines, at costs of less than \$50 kgU^{-1} [64]. An additional 22 MtU are estimated to be available in phosphate deposits worldwide (although at very low concentrations), and a significant fraction of this may ultimately be recovered as global demand for fertilizer continues to rise.

Thirdly, low uranium prices over the last two decades have virtually eliminated incentives for uranium exploration. Consequently, there are almost certainly large quantities of still undiscovered uranium that are not included in the Red Book estimates — particularly in the higher cost categories. Modest investments have led in recent years to dramatic increases in estimates of available resources. In early 2001, for example, the Canadian firm Cameco increased its estimate of the uranium available at its McArthur River mine by more than 50% [65]. To summarize, despite the inclusion of 'speculative resources' in the 17.1 MtU figure, there is a very high probability that the amount of uranium that will ultimately prove recoverable at or below \$130 kgU^{-1} will be significantly greater.

Another way to approach the problem is to estimate the shape of the supply curve as a function of price. On the basis of geological relationships, which indicate that exponentially larger resources are available at lower ore grades, it seems likely that the relationship between price and resources is roughly exponential. According to one industry observer [66], "a doubling of price from present levels could be expected to create about a tenfold increase in measured resources." If we assume, very conservatively, that the 2.1 MtU of known resources reported in the 2001 Red Book as recoverable at \$40 kgU^{-1} represent all the resources that will ever be recoverable at that price, then the total uranium resource R (MtU) recoverable at price p (\$ kgU^{-1}) is given by:

$$R = 2.1\left(\frac{p}{40}\right)^{\varepsilon} \tag{2}$$

where ε is the long term price elasticity of supply. If a doubling of price leads to a tenfold increase in resources, $\varepsilon = 3.32$. By this crude estimate, over 100 MtU would be available at \$130 kgU^{-1}. If the amount of uranium available at \$40 kgU^{-1} is greater than 2.1 MtU, as seems very likely, then estimates of resource availability at higher prices would be proportionately greater as well.

One of the few serious attempts to estimate how much uranium is likely to be available worldwide concluded that a tenfold reduction in ore concentration is associated with a 300-fold increase in available resources [67]. If a doubling in price results in a tenfold increase in supply, this implies that a doubling in price would make economical the exploitation of ores with uranium concentrations 2.5 times lower. This seems plausible, because not all extraction costs scale in direct proportion to the amount of ore mined and processed per tonne of uranium recovered. If, at the other extreme, we assume that costs are inversely proportional to ore grade (as might be true at very low concentrations, when total costs become dominated by the amount of material mined and processed), then $\varepsilon = 2.48$ and about 40 MtU would be available at \$130 kgU^{-1}. More recently, the Generation IV Fuel Cycle Crosscut Group judged that ε might be as low as 2.35 [16], which would give 34 MtU available at \$130 kgU^{-1}. Extrapolating to still higher prices, 170–500 MtU would be available at \$260 kgU^{-1}. These estimates are summarized in Table 6.

At the extreme of low grade resources is the 4500 MtU dissolved in the world's oceans at a concentration of 3 ppb. The recovery of this uranium has been demonstrated using adsorbents. Early approaches involved pumping sea water through the adsorbent; a pilot plant operated in Japan for two years, but the pumping required more energy than would be provided by the recovered uranium and this approach was abandoned [68]. More recent approaches rely

TABLE 6. ESTIMATES OF URANIUM RESOURCES ULTIMATELY RECOVERABLE AT \$80, \$130 AND \$260 kgU^{-1}, ASSUMING 2.1 MtU ULTIMATELY RECOVERABLE AT \$40 kgU^{-1}

Source	Long term elasticity of supply, ε	R (MtU) for p less than or equal to		
		\$80 kgU^{-1}	\$130 kgU^{-1}	\$260 kgU^{-1}
Uranium Information Centre [66]	3.32	21	105	500
Deffeyes and MacGregor [67]	2.48	12	40	220
Generation IV Group [16]	2.35	11	34	170

on ocean currents to move sea water through fixed arrays of adsorbents, with a ship collecting the uranium-bearing adsorbents for on-board processing or delivery to a shore based processing facility. Rough cost estimates have varied from \$100 kgU^{-1} to over \$1000 kgU^{-1}; the 2001 Red Book chose \$300 kgU^{-1} as representative of current thinking. If uranium could be recovered from sea water at costs below the break-even cost for reprocessing and recycling, the use of plutonium fuels could be deferred for many centuries.

Setting aside the question of seawater uranium, if the above estimates of terrestrial resource availability are matched to estimates of future uranium consumption on a once-through fuel cycle, it is clear that uranium prices will not rise anywhere close to our central estimates of the break-even price for reprocessing and recycling in LWRs or FRs for many decades to come. In a study in 1998 of future energy scenarios, the World Energy Council (WEC) and the International Institute for Applied Systems Analysis (IIASA) outlined six scenarios for future energy supply, covering a wide range of assumptions about population and economic growth, resources and technology [69]. Figure 6 shows the cumulative uranium consumption in these scenarios, assuming that nuclear electricity is produced entirely by LWRs operating on a once-through

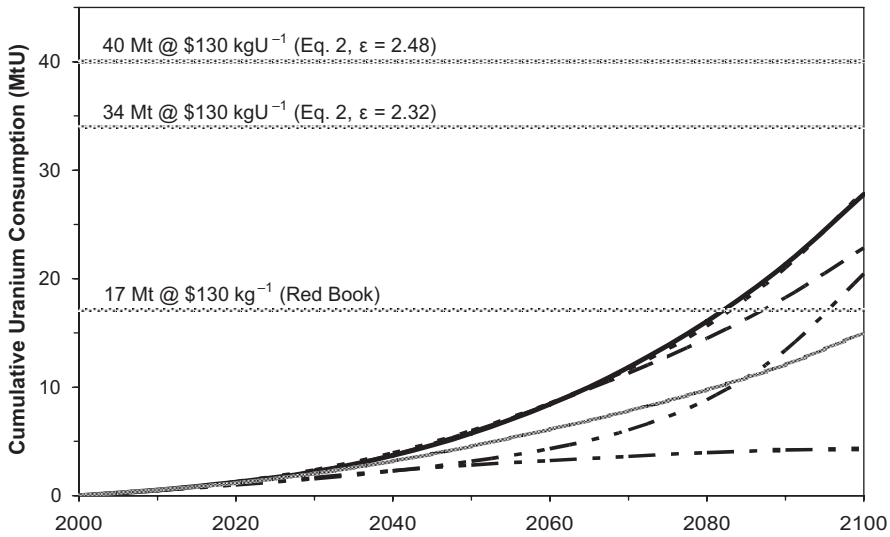

FIG. 6. Scenarios for cumulative uranium consumption, assuming a once-through fuel cycle with an average uranium requirement of 19 tU TW · h^{-1}, and estimates of ultimately recoverable uranium resources at \$130 kgU^{-1}. The scenarios for nuclear electricity production are taken from Ref. [69], normalized to 2434 TW · h in 2000.

cycle with an average uranium requirement of 19 tU TW · h^{-1}.[6] Also shown are estimates of the uranium resources available at prices of \$130 kgU^{-1} or less. On the basis of these scenarios, it seems very likely that uranium resources will continue to be available at substantially below the break-even price for reprocessing at \$1000 $kgHM^{-1}$ throughout the twenty-first century.

5. RECYCLING AND REPOSITORY SPACE PRICE

In recent years, some have argued that repository space is the most pressing constraint on the expansion of nuclear power. This argument is one of the principal drivers of the US Advanced Nuclear Fuel Cycle Initiative. The USDOE argues that existing US reactors, discharging nearly 2000 tHM of spent fuel per year, will fill the 63 000 tHM legislative capacity limit for the Yucca Mountain repository by 2015 and the 'theoretical maximum' capacity of 120 000 tHM by roughly 2050 (if the current level of nuclear capacity were retained) [70]. Reprocessing the fuel and separating and transmuting the heat generating radionuclides, it is argued, could make a second repository unnecessary, even if US nuclear energy generation grows substantially in the future. In other words, what will become scarce and expensive in the future is not uranium but repository space.

Several points should be made concerning this argument. Firstly, it applies only to the USA. Only the USA has chosen a repository site with fixed boundaries, whose capacity cannot be increased indefinitely by digging more tunnels. Other countries are examining sites in huge areas of rock, clay or salt, where the waste from centuries of nuclear electricity generation could be emplaced at a single site.

Secondly, traditional approaches to reprocessing and recycling do not lead to reductions in the amount of repository space required per unit of electricity generated. As discussed above, the required repository volume is determined by the heat output of the wastes, and if plutonium is recycled in existing LWRs, the resulting buildup of heat generating minor actinides means

[6] Assumes an average burnup of 50 MW · d · $kgHM^{-1}$, a net efficiency of 35%, and fuel enrichment and tails assays of 4.2% and 0.2% ^{235}U, respectively. (A tails assay of 0.2% would minimize cost when the uranium price is about 1.3 times the enrichment price (e.g. \$130 kgU^{-1} for \$100 SWU)). The use of higher burnups, lower tails assays and other reactor systems could reduce uranium consumption in a once-through cycle to as little as 12 tU TW · h^{-1} (e.g., a pebble-bed reactor with a burnup of 100 MW · d · $kgHM^{-1}$, an efficiency of 46%, and enrichment and tails assays of 8% and 0.1%, respectively).

that the total waste heat per unit of electricity generated is similar to or higher than that for direct disposal of spent nuclear fuel [71]. To avoid the need for an additional US repository, it would be necessary to separate, recycle as fuel, and transmute all the major long lived heat-generating radionuclides. If we assume, as recent international reviews do, a higher reprocessing cost for these kinds of separations than the central estimate for traditional reprocessing used in the text, a higher fabrication cost for the fuel (given the need for remote handling) and a transmutation reactor or accelerator capital cost \$200 $kW(e)^{-1}$ higher than that of comparably advanced once-through systems, then separations and transmutation would not be economic until the cost of spent fuel disposal reached some \$3000 $kgHM^{-1}$, nearly ten times current estimates [6].

Thirdly, the argument is based on the assumption that it would be less difficult to gain public acceptance and licensing approval for complex and expensive spent fuel separation and transmutation facilities than for a second repository. This assumption is probably wrong. Reprocessing of spent fuel has been fiercely opposed by a substantial section of the interested public in the USA for decades. The health and safety risks to current generations from a separations and transmutation approach would be greater than those associated with direct geological disposal of spent fuel.

Fourthly, the argument is also based on the assumption that, many decades in the future, when repository space has become scarce and reactor operators are willing to pay a significant price for it, it will still not be possible to ship spent fuel from one country to another for disposal. If, in fact, repository capacity does become scarce in the future, reactor operators will probably be willing to pay a price for spent fuel disposal well above the cost of providing the service. It seems likely that, if the willingness to pay reaches anything resembling \$3000 $kgHM^{-1}$, then the opportunity for profit will motivate some country with an indefinitely expandable repository to overcome the political obstacles that have blocked international storage and disposal of spent fuel in the past.

The total cost of pursuing a separation and transmutation strategy could be very high. In a 1996 report, a committee of the US National Academy of Sciences concluded that the excess cost for a separation and transmutation system over once-through disposal would be “no less than \$50 billion and easily could be over \$100 billion” (1992 dollars, \$60–\$120 billion in 2003 dollars) for 62 000 tons of spent fuel (the current legislated limit on Yucca Mountain) [10]. This conclusion still remains valid; there have been no technical breakthroughs or dramatic cost reductions in either separation or transmutation technologies.

To summarize, premature decisions based on early estimates of unproven technology can be very costly. Given the availability of proven, low cost, dry

cask storage technology that can store spent fuel safely for decades, there is no need to resolve these debates quickly.

6. CONCLUSIONS

At a reprocessing price of \$1000 $kgHM^{-1}$, and with our other central estimates for the key fuel cycle parameters, reprocessing and recycling plutonium in existing light water reactors (LWRs) will be more expensive than direct disposal of spent fuel until the uranium price reaches over \$370 kgU^{-1} — a price that is not likely to be seen for many decades, if then.

At a uranium price of \$50 kgU^{-1} (somewhat higher than current prices), reprocessing and recycling at a reprocessing price of \$1000 $kgHM^{-1}$ would increase the cost of nuclear electricity by 1.3 mill $\cdot$ $(kW \cdot h)^{-1}$. Since the total back end cost for the direct disposal is in the range of 1.5 mill $\cdot$ $(kW \cdot h)^{-1}$, this represents more than an 80% increase in the costs attributable to spent fuel management (after taking account of appropriate credits or charges for recovered plutonium and uranium from reprocessing).

These figures for the break-even uranium price and the contribution to the cost of electricity are conservative. The central estimate of the reprocessing price, \$1000 $kgHM^{-1}$, is substantially below the cost that would pertain in privately financed facilities with costs and capacities identical to the large (and predominantly not privately financed) commercial facilities now in operation. The central estimate of the MOX fuel fabrication price, \$1500 $kgHM^{-1}$, is significantly below the price that was actually offered to most utilities in the 1980s and 1990s. No charges were included for storage of separated plutonium or removal of americium, or for additional security, licensing or shutdown expenses for the use of plutonium fuels in existing reactors. A full charge for 40 years of interim storage in dry casks was included for all fuel going to direct disposal, even though new reactors are being built with storage capacity for their lifetime fuel generation. The costs of geological disposal of spent MOX fuel were assumed to be equal to those of spent LEU fuel, despite the substantially higher heat output of spent MOX fuel.

Reprocessing and recycling plutonium in FRs with an additional capital cost of \$200 kW(e) compared with new LWRs will not be economically competitive with a once-through cycle in LWRs until the price of uranium reaches some \$340 kgU^{-1}, given our central estimates of the other parameters. Even if the capital cost of new FRs could be reduced to equal that of new LWRs, recycling in FRs would not be economic until the uranium price reached \$140 kgU^{-1}.

At a uranium price of \$50 kgU^{-1}, electricity from a plutonium recycling FR with an additional capital cost of \$200 kW(e), and with our central estimates of the other parameters, would cost more than 7 mill · $(kW \cdot h)^{-1}$ more than electricity from a once-through LWR. Even if the additional capital cost could be eliminated, the extra electricity cost would be over 2 mill · $(kW \cdot h)^{-1}$.

As with reprocessing and recycling in LWRs, these estimates are conservative. We have assumed no cost for startup plutonium, no additional cost for reprocessing or fabricating higher-plutonium-content FR fuel (compared with LWR fuel), and no additional operations and maintenance costs for FRs compared with LWRs. Costs for the more complex chemical separation processes and more difficult fuel fabrication processes needed for more complete separation and transmutation of nuclear wastes have been estimated in recent studies to be substantially higher than those estimated here for traditional reprocessing and recycling. The additional cost of electricity would be even higher if these approaches were pursued.

The world resources of uranium likely to be economically recoverable in future decades at prices below the break-even uranium price amount to several tens or even hundreds of millions of tonnes, enough to fuel a growing nuclear enterprise using a once-through fuel cycle throughout the century.

Limits on repository space are not a persuasive reason to pursue reprocessing. Traditional approaches to reprocessing and recycling would not help, in any case; a complex of separation and transmutation facilities would be necessary. It is unlikely to be easier to gain approval and acceptance for building separation and transmutation facilities rather than for repository expansion or building a new repository. Reactor operators probably would be willing to pay substantially more for direct disposal of spent fuel in order to avoid expensive separation and transmutation, which would increase incentives for States or other countries to accept the spent fuel.

ACKNOWLEDGEMENTS

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THE BENEFIT OF REPROCESSING/RECYCLING FOR SUSTAINABLE NUCLEAR ENERGY: THE FRENCH VIEW

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Abstract

The French nuclear industry and research community share the vision that nuclear energy, because it is economically competitive and carbon-free, should prove to be an important, indeed vital, option for meeting the future energy needs of Europe and the world. Sustainability is viewed to be an essential condition for enhancing the future role of nuclear energy systems and for realizing the global nuclear growth scenario envisaged by experts. In France, the closed fuel cycle is viewed as an attractive and necessary element of a growth scenario that meets the conditions for sustainable nuclear development. Furthermore, meeting these conditions requires the setting of challenging technological innovation goals for the next (fourth) generation of nuclear systems. These fourth generation nuclear systems will not be ready for large scale deployment before the years 2030 or even 2040. In the meantime, nuclear growth will rely on light water reactors (LWRs), which will continue to play an essential role during this century. Reprocessing and mixed oxide (MOX) recycling in LWRs is considered to be a robust and flexible option that will best accommodate these aims since: (a) it ensures the long term (>10 000 years) safe conditioning of high level, long lived waste by vitrification; (ii) it reduces, by factors ≈4 in volume and ≈10 in radiotoxicity, the conditioned final waste requiring geological disposal; (iii) it enables part of its energy content to be burned as MOX; (iv) it concentrates the remaining material, of degraded isotopy, in proliferation-resistant MOX spent fuel for future use, through reprocessing and recycling in fourth

generation nuclear systems (possibly with minor actinides); (vi) it drastically reduces the need for spent fuel interim storage capacity.

1. VISION OF A GROWING RELIANCE ON NUCLEAR ENERGY IN THE WORLD SUSTAINABLE ENERGY MIX OF THE TWENTY-FIRST CENTURY

A growing number of prospective studies envisage that nuclear energy, because it is carbon-free, will play an important, possibly essential, role in the world energy mix of the twenty-first century, were humankind to succeed in embarking on a sustainable development path (Fig. 1).

The work performed to analyse global energy and the environment in the long term and to inform strategic decisions on these matters is considerable and has produced many contrasting scenarios for the 2050 time horizon. The large majority of these scenarios envisage a large increase of world primary energy needs over the next 50 years under the combined effects of demographic growth (experts predict world population will increase by 50–70%, to up to 9–10 billion, over this period), a massive industrialization of some heavily

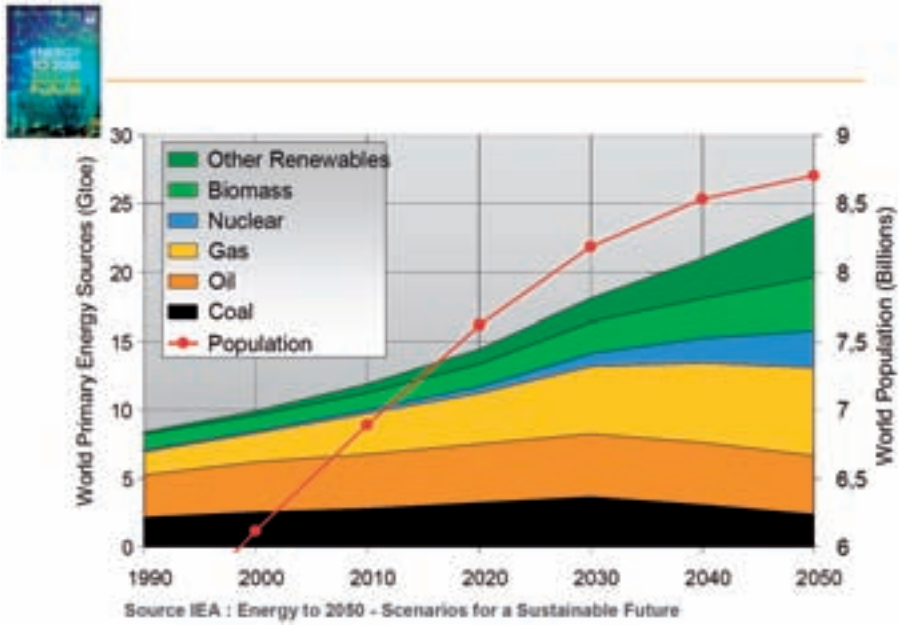

FIG. 1. Sustainable development scenario (Courtesy: IEA 2003).

populated regions (China and India) and more generally the development of trade driven by globalization.

Among these scenarios, IIASA scenario B (Fig. 2) (IIASA: International Institute for Applied Systems Analysis) is illustrative of a 'middle course' evolution, which appears reasonable in terms of both primary energy consumption level (≈20 Gtoe/a or twice the current level, i.e. an increase by 20% of the average consumption per person over the period 2000–2050) and balanced recourse to resources (oil, 4 Gtoe/a; coal, 4.1 Gtoe/a; gas, 4.5 Gtoe/a; nuclear, 3.2 Gtoe/a; renewable, 3.9 Gtoe/a).

More sustainable scenarios are also proposed that are constructed to meet three main goals:

(1) Control of CO_2 emissions (most of the international scientific community agrees that the sustainable threshold of anthropic greenhouse gas emissions should not significantly exceed 3 Gt of carbon equivalent per year, which means cutting by a factor of three the current emissions, corresponding to ≈4 Gtoe/a of fossil energy in the form of coal, oil and gas);
(2) Security of energy supply (which militates for a preferential recourse to domestic resources and takes into account, for example, the physical

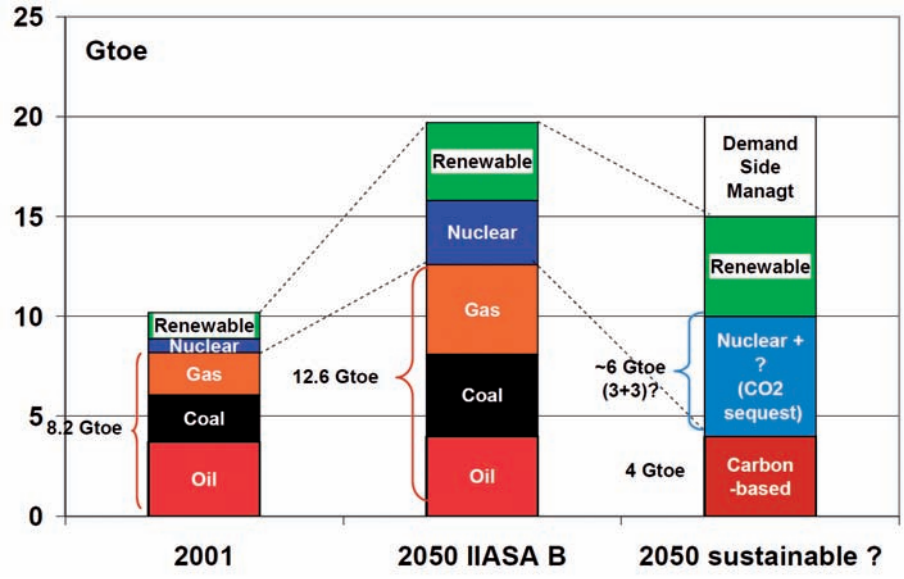

FIG. 2. World primary energy needs in 2050: illustration of the scenarios discussed.

availability of fossil energies, cartelization risks and transport infrastructure weaknesses);
(3) Access to energy by underprivileged populations.

These scenarios converge to show that achieving these goals will require strong policies combining increased efforts on demand side management (which could lead to a global savings equivalent to 5 Gtoe/a) and a determined development of renewable energies (which may reach 5 Gtoe/a). With the recourse to fossil fuels without CO_2 capture/sequestration being limited to 4 Gtoe by climate change mitigation constraints, the balance of energy requirements (≈6 Gtoe, Fig. 2) will have to be provided by other carbon-free energies, thus requiring a large expansion of (sustainable) nuclear energy (and possible development of CO_2 capture/sequestration).

Thus, were humankind to succeed in embarking on a sustainable development path, a fourfold increase of world nuclear energy generation from the current 0.7 Gtoe/a to 2.5–3 Gtoe/a at the 2050 time horizon appears a plausible assumption.

2. CONDITIONS FOR ENABLING GLOBAL NUCLEAR EXPANSION SCENARIOS

For enhancing the future role of nuclear energy and enabling such global nuclear expansion scenarios to occur, new nuclear systems will be needed, requiring:

(a) Near term deployment of third generation nuclear plants;
(b) Significant R&D on next (fourth) generation systems that meet the conditions of sustainable long term development.

The world nuclear power fleet currently consists of 440 reactors for a total installed capacity of 360 GW(e). The large majority of these units are light water reactors (LWRs) of the so-called second generation, built between 1970 and 2000. This technological preponderance is a major asset, owing to a cumulative operating experience of over 10 000 reactor-years. It has been taken advantage of to develop a third generation, which is currently at the industrial stage and to which the European Pressurized Water Reactor (EPR), in particular, belongs.

Second and third generation models are capable of meeting electronuclear generation needs for at least the next three decades, and much more at the current worldwide level of nuclear generation. During this period, about

half of the world fleet should be renewed, and the growth prospects, in particular in Asia and in the United States of America, should assure an at least equivalent market. Thus, it seems already assured that the number of third generation reactors that will be built will be at least equal to, and more likely be higher than, the number of second generation reactors currently in operation. Owing to the design lives of third generation reactors, a number will still be in operation at the end of the century. Their main characteristic is to use only the fissile ^{235}U isotope, which has a natural abundance of 0.7% of the total amount of uranium.

It remains true that, if some of the global nuclear growth scenarios envisaged by experts were to occur (e.g. a fourfold increase), nuclear energy will be seriously confronted with the issue of sustainable development due, in particular, to uranium resources. Thus, while some of the sustainability requirements are already being satisfied by third generation nuclear systems, a fourth generation of nuclear systems (reactors and their associated fuel cycle) will need in due course to be developed and made available that exhibits all the essential features required for an economic and sustainable long term development of nuclear energy on the basis of the full use of uranium resources.

2.1. Waste management

Our understanding of public attitudes towards nuclear energy makes us regard good management of the ultimate wastes and the ALARA (as low as reasonably achievable) minimization of their quantity and radiotoxicity as essential features of sustainability.

All the opinion polls taken during the last ten years in France have shown the radioactive waste issue to be the most important concern of the public as far as nuclear energy is concerned. More precisely, it is the management of high level, long lived (HLLL) radioactive waste that is the concern: disposal of low activity, short lived wastes is operating without any problems and is well accepted by local populations.

Thus, gaining the level of public acceptance needed for global nuclear growth scenarios to materialize requires that we implement a good management of high level waste, which implies studying reasonably practical and globally efficient options to reduce the quantity and radiotoxicity of the ultimate waste produced by the use of nuclear energy in an ALARA perspective (as required by the French 1991 High Level Waste Act). By recovering from spent fuel and recycling products that are still valuable (that still contain recoverable energy), we reduce by a large amount the long lived element content of the ultimate waste. In some ways, this is the same

sustainability strategy as for other wastes: selection of what is reasonably reusable and recycling it.

Plutonium is one valuable resource, which already contributes a third to nuclear production by spontaneous capture and fission in LWRs, and is also an important contributor to the long term radiotoxicity of spent fuels. With the present reprocessing technology, less than one part per thousand of spent fuel is ending up as the ultimate waste. Now, we are also studying the possibility of extracting other actinides and burning them, called partitioning and transmutation, and the impact of this on reactors and fuel cycle facilities. If it proves both industrially efficient and valuable, this option could reduce the long term issue of ultimate wastes, bringing their radiotoxicity back to the level of natural uranium in a few hundred years (Fig. 3), and contribute to further optimization of the geological repository design concept and space requirements. Such an improvement can be realized with the fourth generation of nuclear production systems using fast reactors, facilitating public acceptance of its deployment, an essential feature for sustainable development of nuclear energy.

Today in France, industrial solutions do exist and are operational for disposal of low level and short lived wastes that will have recovered after 300 hundred years to the same level as natural radioactivity. Medium and high level and long lived wastes are currently stored in industrial facilities and are conditioned in a safe manner so that there is no impact on either the environment or humans.

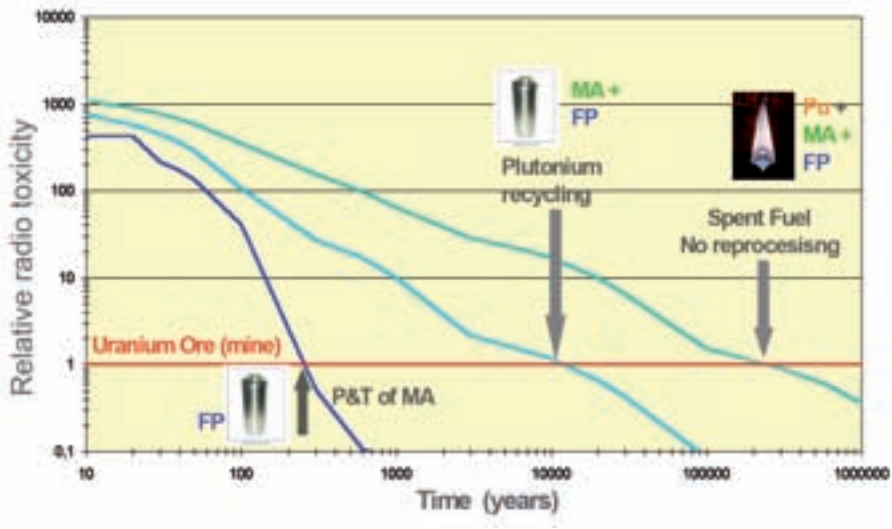

FIG. 3. Relative radiotoxicity of ultimate waste as a function of time (FP: fission products; MA: minor actinides; P&T: partitioning and transmutation).

Although French policy was already to process spent fuel, condition HLLL waste (fission products plus minor actinides) in glass, and recycle plutonium as MOX fuel, our country was confronted in 1990 with the impossibility of progress in the search for a deep geological repository for HLLL waste. This situation led the French Parliament to pass in 1991 an Act instituting, in particular, a 15 year period for completing research activities before deciding the final fate of these HLLL wastes. In order to open all the possible options for their long term management and to be sure that all possibilities have been technically explored, this Act requests in particular the study of solutions and processes for:

(a) Minimization of the quantity and radiotoxicity of waste, via partitioning and transmutation;
(b) Either reversible or irreversible disposal of waste in deep geological formations;
(c) Waste conditioning and long term interim storage.

When approaching this deadline, the deep geological repository solution appears more than ever confirmed through both the work carried out in France and in experiments in other countries. Quite convincing demonstrations have been given of the long term behaviour of the glass matrix that contains these HLLL wastes after spent fuel processing and of the retention of minor actinides. Research work confirms the interest in recycling plutonium and provides an incentive to burn the other actinides to strongly reduce the radiotoxicity and long term residual heat of the ultimate waste, thus optimizing the management of waste packages in deep repositories.

For simple physical reasons, full destruction of actinides is possible only in fast neutron reactors. This further reduction in terms of volume and radiotoxicity, which could complete the reduction already achieved on the industrial scale, by the treatment and recycling of plutonium as MOX fuel, is already considered in the design of future fourth generation nuclear energy systems (Fig. 4). And as far as proliferation resistance is concerned, global actinide management (integral recycling of actinides, and thus without separation/purification of plutonium) would provide an intrinsic protection of materials, comparable to that of spent fuel.

Of course, such an option would have to be fully assessed through a cost–benefit analysis addressing its global impact on reactors and fuel cycle facilities, its benefits, industrial efficiency and cost effectiveness, before making any great industrial investments.

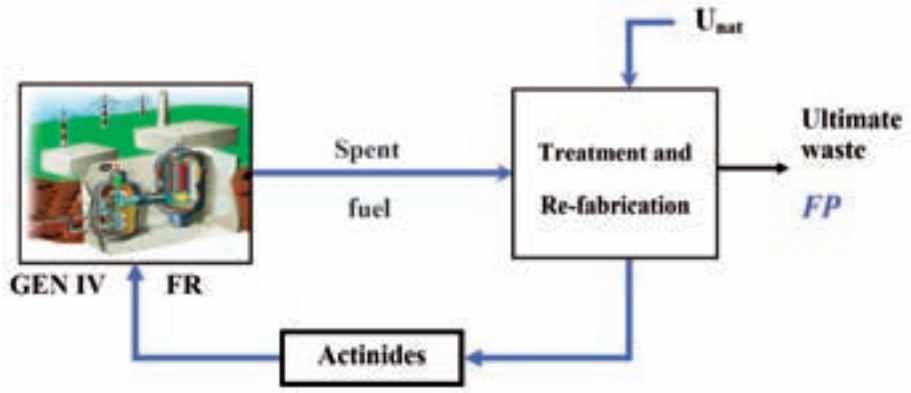

FIG. 4. Illustration of the Generation IV nuclear energy system with a fast neutron reactor and the associated fuel cycle with full actinide recycling.

2.2. Optimization of the use of natural resources

Optimization of the use of natural resources is another essential sustainability feature, as well as economic competitiveness, safety and resistance to proliferation.

There are quite large quantities of natural uranium, several billion tonnes if we include unconventional resources such as the uranium diluted in sea water or in phosphate deposits. So, in principle, there is no limitation to the use of nuclear energy, even with the present technology of LWRs, which recovers less than 1% of the energy content of the natural uranium processed for fuel fabrication. However, uranium resources that can be extracted at a reasonable cost are much more limited, which means that, to keep nuclear energy economically competitive, we shall move, sooner or later, to fast neutron reactors, which are able to exploit natural uranium fully and which are quite insensitive to its price, even if that increases by a factor of ten or more.

The ultimate uranium resources are currently estimated at about 15 Mt when taking into account both reasonably assured, additional estimated and speculative conventional resources[1]. Historically, the price of uranium (Fig. 5) underwent a very strong rise in the years 1974–1980, while a great number of nuclear projects were being launched worldwide, then a large and sudden fall

[1] On the basis of the OECD/IAEA Red Book (2003), known plus estimated additional resources recoverable at costs of less than US $130/kg (or €3/TW · h(e)), plus speculative resources.

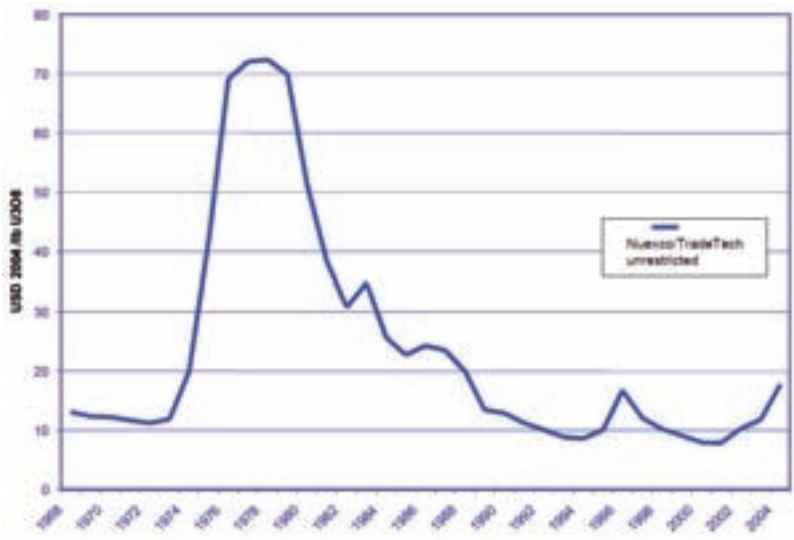

FIG. 5. Price history of uranium (Nuexco/TradeTech: unrestricted, spot market index, in US $(2004)/lb U_3O_8).

following the cancellation of some of these. After a period of low prices, there is currently an increase of the uranium price (a doubling over one year) due to a structural adjustment between production and demand, in addition to a decrease of the stocks put on the market. The period that is beginning should end up on a new equilibrium based on an adjustment of the supply (new uranium production capacities) and demand (evolution of nuclear production as a function of its economic competitiveness with other electricity generation technologies).

At the current rate of exploitation of nuclear energy, with a world uranium consumption corresponding to about 65 000 t/a,[2] these ultimate resources (15 MMt) represent at least 200 years of resources that can be used in LWRs. A uranium price that is affordable to utilities remains the main feasibility criterion for the mining of these resources.

Quantifying matters, we assume that there is a sustainable world energy mix at the 2050 time horizon, including a recourse to nuclear energy of the order of 2.5 Gtoe/a, which would be the cumulative uranium consumption

[2] Under the following assumptions: a 360 GW(e) operating fleet generating 2700 TW · h(e) (kp: 85%), and 150–200 t U_{nat} for 1 GW(e) (22–32 t U_{nat} per TW · h(e)).

committed at that time for the operation of the corresponding world nuclear power fleet, assuming it consists exclusively of LWRs. This committed consumption represents the overall amount of uranium that could be de facto consumed or pre-empted by nuclear utilities to ensure their operation, over their lives.

Under the following simplified assumptions — a 60 year reactor operating lifetime, 160 t/GW(e) annual consumption (typical of current LWRs), world nuclear capacity stable until 2020 (360 GW(e)) and then increasing linearly until 2050 — the committed uranium resources at the 2050 time horizon, as shown in Table 1, could be equivalent to the speculative resources currently estimated.

This prospect could lead at that time horizon to great and persistent pressures on ^{235}U resources. The prospect of these economic tensions would act as a brake on the large scale expansion of nuclear energy envisaged in sustainable world energy scenarios, were new fourth generation nuclear energy systems capable of fully exploiting natural uranium resources not to be mature for industrial deployment at that time: i.e. fast neutron reactors and the associated fuel cycle based on plutonium multi-recycling and the closed uranium–plutonium cycle. Obviously, no robust 'planning for growth' strategy can be built only on the hope that new uranium resources at economically attractive prices will be discovered and exploited in time to meet the growth demand scenario with present technology LWRs.

2.3. The challenging technological goals for timely development of fourth generation nuclear energy systems with closed fuel cycles

The above considerations on waste management and conservation of uranium resources lead us to view the closed cycle as an attractive and necessary element of a growth scenario that meets the conditions of a sustainable nuclear development. Meeting these conditions requires setting

TABLE 1. CONSUMED AND COMMITTED URANIUM RESOURCES IN 2050

Nuclear energy in 2050 (Gtoe/a)	0.7	2	3	4
Installed nuclear capacity in 2050 (1000 GW(e))	0.4	1.0	1.6	2.1
U_{nat} consumed and committed in 2050 (Mt)	6	12	17	22

challenging technological innovation goals for the next (fourth) generation of nuclear systems. We therefore view the need to engage the corresponding developments in a timely manner so that these fourth generation systems can be brought to industrial maturity in due time during this current half-century and enable the materialization of ambitious global nuclear growth scenarios.

This vision is shared by the members of the Generation IV International Forum: thus, a consensus was found very quickly within this framework on criteria for these fourth generation nuclear systems, and while further improvement of economics, safety and reliability remain as always important goals, sustainability features as well as proliferation resistance and physical protection are now top priorities (sustainability, as discussed above, meaning an effective use of natural resources, and management and optimization of nuclear waste).

Thus, among the six concepts selected in the Gen IV exercise (Fig. 6), most are for fast neutron and closed cycle systems[3]. This is directly linked to the fact that:

(a) They are top ranked in sustainability because of their optimum use of natural resources and of their ability to reduce the radiotoxicity and amount of wastes. On the volume criteria alone, in the nuclear growth scenario considered here, the closed cycle would avoid the prospect of needing to commit at the global level the commissioning of repository space equivalent to one 'Yucca Mountain' every two to three years at the 2050 time horizon.
(b) They are rated good not only in proliferation resistance but also in safety, economics and physical protection.

Thus, this is why we naturally consider Generation IV fast reactors as the best option, complemented by a full actinide recycling strategy if this is assessed as both efficient and valuable, and therefore a leading R&D priority, to meet the sustainability criteria and to achieve an as low as reasonably achievable (ALARA) minimization of ultimate waste and conservation of uranium resources, while continuing to guarantee a high level of proliferation

[3] Another challenge in enhancing the future role of nuclear energy, and the second main R&D orientation after sustainability, relates to meeting the emerging non-electricity needs of the twenty-first century, notably the production of hydrogen. Thus, the only concept selected in the Gen IV exercise that does not rely on a closed fuel cycle is the VHTR, a concept designed for hydrogen generation and other high temperature applications (e.g. process heat).

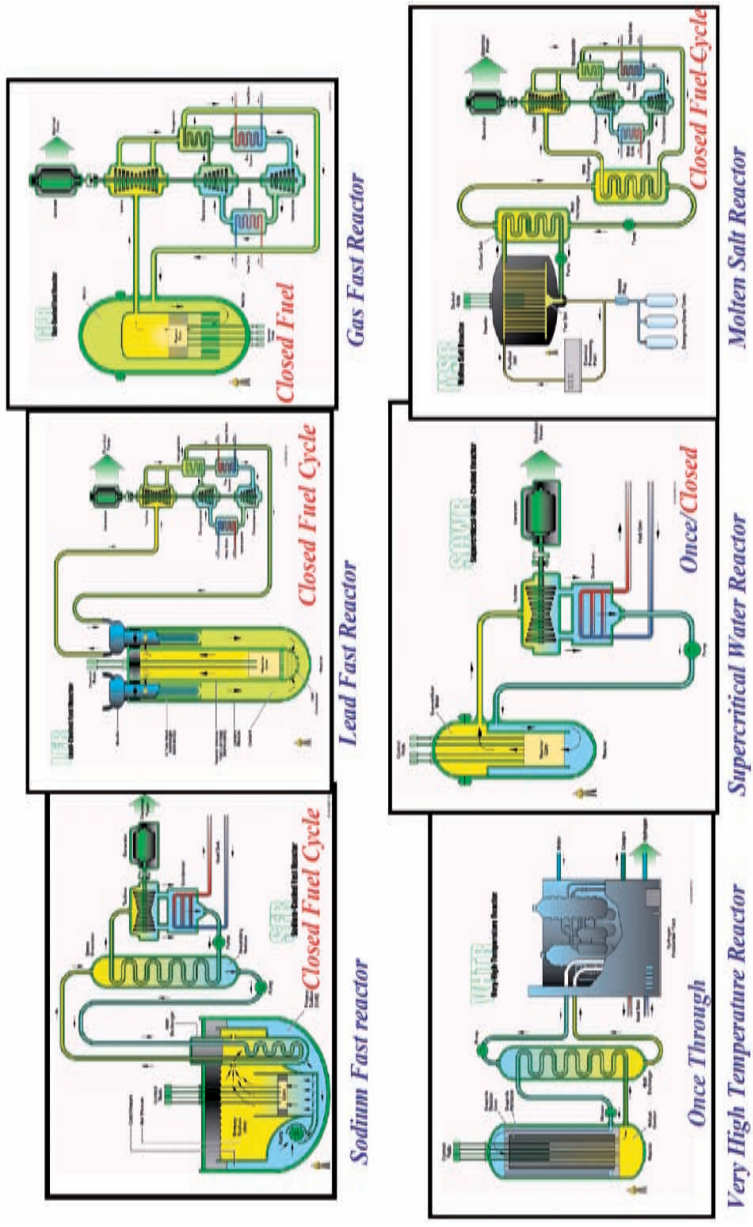

FIG. 6. The six concepts selected in the Gen IV exercise.

resistance. This means recycling fissile and fertile materials that produce energy, and recycling the minor actinides that form the long term component of the vitrified waste as produced in France at the present time, to transmute them into fuel for fast reactors. Removing minor actinides and putting them back into the reactor to be burned, if proven efficient and valuable, is effectively a means of transforming high activity and long life waste into high activity waste and medium life waste, using a packaging with a demonstrated capacity to last for very many years.

One of the most important technological processes to be developed is the grouped extraction of actinides; two main technological routes have been identified for this process: hydrometallurgy and pyrochemistry. On the basis of well established know-how on the hydrometallurgical method, the Commissariat à l'Énergie Atomique (CEA) has proposed a study of the so-called GANEX process (Fig. 7). This process is directly derived from studies carried out for separation of americium and curium from fission products by the DIAMEX and SANEX processes, in the context of R&D carried out in France, under the 1991 Act, on separation of minor actinides to reduce the radiotoxicity of vitrified waste. In detail, it is based on two separation steps so as to be able to separate part of the uranium before proceeding further. This partial separation is imposed by the design of fast neutron reactor cores, in which it is

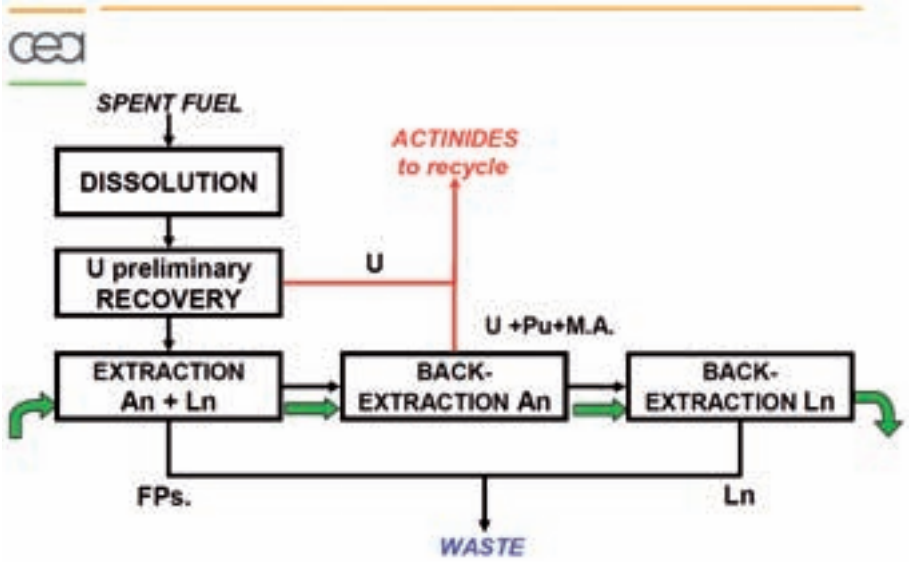

FIG. 7. Grouped actinide extraction by the GANEX process (An: actinides; Ln: lanthanides).

necessary to be able to zone the core for different U/Pu ratios so as to have a better distribution of power in the radial plane, and also the DIAMEX process itself, which does not tolerate high amounts of uranium. The remainder of the uranium and other actinides are then separated from the fission products. The fact that thorough separation of minor actinides and grouped separation could be based on the same principles suggests the possibility of grouping the two functions using the same facility, which would result in much better flexibility, even if the concentrations are not the same.

3. HLLL WASTE CONDITIONING AND PLUTONIUM RECYCLING AS MOX FUEL IN LWRs AS A ROBUST AND FLEXIBLE OPTION FOR THE PRESENT

These fourth generation nuclear systems will certainly not be ready for large scale deployment before the year 2030 or even 2040. Until then, nuclear growth will rely on LWRs, which will continue to play an essential role during the twenty-first century, and we therefore view reprocessing, HLLL waste conditioning and plutonium recycling as MOX fuel in LWRs (the current fuel cycle strategy in France) as a robust and flexible option to best accommodate this period, as it already makes a major contribution to sustainable development with existing facilities operating in the long term.

Indeed,

(a) It enables:
  (i) Assurance of a very long term (>10 000 years) safe conditioning of HLLL waste (fission products and minor actinides) by vitrification, well adapted to their lasting industrial storage and their subsequent geological disposal;
  (ii) A reduction by a factor of about four in the volume, and by one of about ten in the radiotoxicity, of the conditioned final waste requiring geological disposal.

(b) It drastically reduces the needs for spent fuel interim storage capacities (seven UO_2 spent fuel assemblies, after processing, lead to one MOX spent fuel assembly plus two vitrified HLLL waste packages (0.4 m^3)); combined with discharge burnup increases, this enables control on a long term basis of the amount of spent fuel (UO_2 and MOX) stored in existing pools.

(c) It enables:
  (i) Use of part of the energy content of the plutonium recovered from spent fuel by burning it promptly as MOX fuel in LWRs (in France,

the recycling of plutonium, currently performed in 20 900 MW(e) units[4]: 30% of the core generates 30–40 TW · h/a, i.e. 8–10% of electronuclear generation);

(ii) Concentration of the resulting plutonium, of degraded isotopic composition, in more proliferation-resistant MOX spent fuel to preserve its future usability, through reprocessing and recycling, possibly together with minor actinides, in (fast neutron) fourth generation nuclear systems capable of more fully recovering the energy content of natural uranium (the uranium–plutonium fuel cycle);

(iii) Preservation of up to 25% of natural uranium resources.

Economic studies on the cost assessment of the fuel cycle back end, based on currently operating industrial facilities, show the economic affordability of these benefits, in terms of sustainability, achieved by the reprocessing, conditioning and plutonium recycling strategy, which results in an increase in the total kW · h cost ranging from 2% to a maximum of 6% (depending on the assessment method) compared with the spent fuel direct disposal option. In more concrete terms, given the projected operating life of the amortized current fuel cycle facilities and the accumulated financial provisions, the current economic calculation carried out by the utility Électricité de France (EDF) shows an accounting value of less than 0.15 c €/(kW · h) for future discounted back end liabilities, including both spent fuel interim storage and reprocessing, and high level waste storage and final disposal. This evaluation does not take into account any further optimization of the whole back end process that could be expected in the future.

Regarding the consequences, in terms of HLLL waste management, that would be induced by the eventual reprocessing of spent MOX fuel[5], which exhibits, compared with spent UO_2 fuel, a higher thermal load (×4) and an increased proportion of minor actinides (americium and curium, ×8), the current thinking, subject to confirmation by ongoing R&D, is that, at the dilution rates (with UO_2) envisaged for the reprocessing of MOX (between 15 and 30% or more), there should be no significant impact either on the number

[4] The present capacity to recycle plutonium in MOX loaded reactors limits the amount of spent fuel that can be reprocessed, the basic principle being to limit the separated plutonium inventory to the level needed to dynamically manage the industrial recycling process. Thus, 850 t_{HM} of UO_2 fuel are reprocessed out of the 1100 t_{HM} of spent fuel discharged from EDF's reactor each year, including 100 t_{HM} of MOX fuel).

[5] For supplying the first plutonium load of fourth generation nuclear systems.

of waste packages or on the size of the geological repository, all other factors being equal. The industrial reprocessing campaign, performed without any significant problems, on 10 t of spent MOX fuel in the La Hague UP2 facility in late 2004, contributed to the confirmation of these assumptions.

4. CONCLUSIONS

Nuclear energy, because it is carbon-free, should prove a vital option for enabling humankind to embark on a sustainable development path. An energy policy should therefore plan, and aim to ensure the conditions for, an enhanced role for nuclear energy and a global expansion of nuclear energy to the quite large extent likely to be needed to satisfy world energy requirements along this path.

We view sustainability as an essential condition for this, and resource conservation and waste management and optimization as essential sustainability features of fourth generation nuclear systems, as well as economics, safety and proliferation resistance. Fourth generation fast neutron reactors associated with a closed fuel cycle, with recycling of fissile and fertile materials, and ideally of minor actinides if deemed efficient and valuable, is an attractive and necessary element of a nuclear growth scenario that meets the conditions of a sustainable nuclear development. Making such systems available for large scale deployment in due time during the first half of the twenty-first century and taking up the associated technological innovation challenges requires starting the corresponding research and development immediately.

These fourth generation nuclear systems will not be ready for large scale deployment before the year 2030 or even 2040. In the meantime, nuclear growth will rely on LWRs, which will continue to play an essential role during the twenty-first century, and we therefore view reprocessing, HLLL waste conditioning and plutonium recycling as MOX fuel in LWRs (the current fuel cycle strategy in France), as a robust and flexible option to best accommodate this period as it already makes a major contribution to sustainable development.

SEPARATED PLUTONIUM MANAGEMENT

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Abstract

The paper outlines the experience gained in relation to the Thorp store at Sellafield with storage, transport and safeguards as applied to separated plutonium. It considers the different management options and issues associated with separated plutonium, including the effects of delayed utilization of fissile materials. Projections of future levels of separated plutonium are described, and the implications for sustainability with respect to the longer term introduction of fast reactors on the timescale of up to 2050 are also considered.

1. BACKGROUND

Reprocessing of irradiated fuel on an industrial scale has been carried out since the 1950s. In the United Kingdom (UK), a dedicated commercial facility commenced reprocessing of irradiated Magnox fuel from civil reactors in the early 1960s. Since the start of these activities, there has been a requirement to manage the products and wastes resulting from reprocessing.

Over the 50 years of management of separated plutonium, the designs for the assets required for handling this material and the associated operational, safety, security and safeguards procedures have developed significantly with the aim of continuing to prevent diversion of fissile material for non-peaceful uses. No material stored under international safeguards has ever been diverted for non-peaceful uses.

Increasing attention is being given to the management of separated plutonium, as reuse has for a number of reasons not maintained pace with separation. This paper will concentrate primarily on the different management options for separated plutonium, with particular emphasis on the issues associated with the utilization of plutonium as a fissile resource.

2. STORAGE

Storage facilities are designed to avoid criticality, provide physical containment and protection, and also to take account of cooling requirements on account of the release of heat from the plutonium.

Modern facilities provided for the storage of separated plutonium are remotely operated within a fully automated operating regime. At the Thorp plutonium store at Sellafield, plutonium dioxide is packed in triple layered containers. These containers are stored in criticality-safe designed channels within secure vaults. The vaults are designed to protect the material against major events such as seismic movement or aircraft impact, and also to prevent diversion of the material. Forced cooling is provided to the store, although the tall racks generate their own chimney effect, which has been demonstrated to be sufficient to cool the store should the forced cooling system ever fail. For plutonium handling areas, strict access controls as well as monitoring and surveillance arrangements are in place. Countries with such plutonium storage facilities typically employ armed guards to ensure their security. Plutonium has been stored safely and securely in purpose built facilities for over 50 years. These facilities are operated and conform to international safeguards and security requirements.

To eliminate the risks associated with moving plutonium between facilities, the UK adopted an approach where the mixed oxide (MOX) fuel fabrication facility was built directly on to the reprocessing plant such that plutonium transfers are carried out in effect within the same building envelope.

3. TRANSPORTATION

Plutonium, including MOX fuel, has been safely transported both nationally and internationally for over 40 years. These operations have been carried out in accordance with national and international legislation. During this time, considerable experience in maintaining the safety and integrity of materials in relation to all kinds of package types, modes and routes has been gained.

Plutonium has historically been transported as plutonium dioxide, although an increasing number of movements are in the form of MOX fuel.

These movements have taken place using a variety of different modes, with each movement requiring the use of approved assets and transport plans.

As with all radioactive materials, the transportation of plutonium is regulated by the IAEA Transport Regulations. These regulations ensure that

the package selected satisfies the criteria laid down for containment, criticality and shielding.

INFCIRC/225 dictates the requirements for all forms of transport of MOX and plutonium. From these, each country develops its own security standards for the physical protection of nuclear material in use, transit and storage. These regulations lay down the requirements for each of the three categories of material and the different security requirements needed. Plutonium is allocated to the most onerous classification, Category I.

Plutonium transport has been carried out by European companies (British Nuclear Group, COGEMA Logistics, Transnubel and their predecessors) since 1952. These movements have been carried out without a single incident resulting in release of material to the environment, and in accordance with the relevant regulations. Physical security is assured by strict adherence to international conventions. The transport companies are committed to the continuation of maintaining such transportation in a safe and secure manner.

3.1. Rail transport

Rail transport of plutonium has been carried out in the Russian Federation. Although rail transport of Category I material is technically viable within Europe, it is not normally pursued because of security considerations.

3.2. Air transport

Air transport has been used in the past for both MOX fuel and plutonium dioxide. Whilst the significantly reduced transit times provide security benefits, several countries specifically prohibit transport of plutonium by air over their territory.

The 1996 IAEA Transport Regulations state that a type C package must be used for transportation of plutonium by air. The stringent requirements for type C packages have almost precluded air as a viable mode of transport.

3.3. Road transport

High security vehicles are utilized across Europe for the transport of plutonium products. These vehicles require approval by each security authority for every country they transit and are constructed to provide an effective barrier to attempts to steal or damage the package contents.

3.4. Sea transport

PNTL, a subsidiary company owned by BNFL, COGEMA and the Japanese utilities, operates a fleet of purpose built ships constructed to meet INF3, the highest level of the International Maritime Organization (IMO) code for the safe carriage of irradiated nuclear fuel, plutonium and high level radioactive wastes in flasks on-board ships. This code is complementary to the IAEA Transport Regulations and became mandatory in May 1999. The provisions of the IMO code predominantly cover ship design, construction and equipment. These ships have operated, predominantly between Europe and Japan, for over 25 years and have transported irradiated fuel, vitrified residues, MOX fuel and plutonium powder.

Two of the PNTL fleet of ships have been modified to comply with the security requirements of transporting Category I materials. In addition, a British owned vessel has also been modified for this use.

A worldwide emergency response system is operated, including a 24 hour response team and salvage cover. In the unlikely event of the loss of a ship, the emergency team is equipped with a sonar search system for locating the vessel. The vessels are fitted with an automatic sonar location system that is capable of operating at depths in excess of 6000 metres.

4. SAFEGUARDS

A number of international, regional and bilateral agreements exist in order to ensure that civil nuclear materials and equipment, such as plutonium and plutonium handling equipment and facilities, are not illicitly used for the manufacture of nuclear weapons or explosive devices. Most countries with civilian nuclear power programmes also have their own safeguards system. Civilian plutonium production, transport storage and use are therefore under strict government control and international surveillance.

Reprocessing and MOX fuel fabrication plants, as well as civilian plutonium storage facilities, can be satisfactorily safeguarded. Although plutonium handling facilities exist in several countries, large commercial reprocessing and MOX fuel fabrication plants are located within the territory of the European Union; the IAEA, as well as EURATOM, has gained substantial experience over the years in safeguarding these parts of the fuel cycle. The procedures used are comprehensive and well established. Contractually, there are provisions that require reprocessing customers to be able to demonstrate that plutonium is required for peaceful purposes, prior to government permission being given for its export from the storage facility.

5. PHYSICAL PROTECTION

The risks and hazards associated with separated plutonium handling, storage, transport and use are well understood, and the technology is available to conduct these activities safely.

Civilian plutonium facilities have a very good operational and safety record and provide high standards of protection for the environment and public health. Plutonium confinement and storage and plutonium transportation in either solution or solid form follow well established practices and international guidelines.

Various national security arrangements (e.g. use of barriers, detection measures and guards) are aimed at physically protecting isolated plutonium and preventing its diversion for non-peaceful uses (IAEA INFCIRC/225 (Rev. 2), guidelines). The responsibility for the establishment, implementation and maintenance of civil security arrangements is the sovereign responsibility of a country's national government.

6. OPTIONS FOR PLUTONIUM MANAGEMENT

Three options are under consideration for future management of separated plutonium stocks:

(1) Continued storage for future use;
(2) Immobilization as waste for long term and/or indefinite storage, then disposal;
(3) Recycle as MOX fuel in reactors followed by spent fuel management.

In respect of storage, this approach does not foreclose any options and follows the continued use of established methods. However, some stakeholders regard storage as deferring the decision and as a potential proliferation risk. There is also concern expressed in some quarters regarding the passivity of separated plutonium storage, as passive storage of nuclear materials is viewed as intrinsically safer than active systems.

A number of different immobilization options have been considered. These consider converting the plutonium into a form more suitable for disposal, with the option of providing greater perceived security by surrounding the waste form with a radiation barrier such as vitrified high level waste or spent fuel. Whilst the use of a radiation barrier makes the plutonium less accessible for diversion in the short to medium term for non-peaceful uses,

it raises safeguards issues with respect to the re-verification that underpins the existing system.

Considerable work has been undertaken examining the feasibility of disposal of plutonium in deep underground repositories. Various schemes have been suggested for the disposition route. A list of potential disposition options considered by the UK Plutonium Stakeholder Dialogue and initial assessments of their feasibility is detailed in Appendix I. There are issues that need to be resolved with all of these possibilities, such as safeguardability, acceptance criteria for deep geological disposal, physical protection and criticality control, and the economics of industrialization. These same issues will have to be resolved for the disposal of spent fuel in the longer term, i.e. when the intrinsic radiation barrier of spent fuel is no longer as effective.

Many States have adopted the MOX recycle route as the preferred option for plutonium disposition, and the remainder of this paper will focus on the issues associated with this option. Experience of the use of MOX fuel by several utilities has shown little difference between the use of MOX fuel and the use of UO_2 fuel in terms of overall generating cost, owing to the small component attributable to fuel costs. MOX fuel represents a mature technology and therefore provides greater cost certainty than the immobilization technologies under investigation.

7. EFFECT OF DELAYED FISSILE MATERIAL UTILIZATION

Delayed utilization of the residual uranium in spent nuclear fuel (SNF) or of uranium separated from spent fuel (RepU) does not affect to any significant extent the radiotoxicity of waste and the sustainability aspect of fissile material management [1]. But this is not the case for plutonium.

Plutonium can be defined under three separate accountancy units:

(1) Put, Pu total;
(2) Puf, Pu fissile;
(3) Pue, Pu equivalent.

A description of these different approaches is given in Appendix II.

Although no constant or universal relationship exists between Put, Puf and Pue, thirty years of industrial utilization of MOX fuel have resulted in a uniform acceptance by industry, national authorities and international organizations of the need to cope with this triple bookkeeping.

7.1. Plutonium isotopic compositions and radioactive decay

The isotopic composition of plutonium is affected by a number of different factors such as reactor environment (e.g., pressurized water reactor (PWR), boiling water reactor (BWR) or advanced gas cooled reactor (AGR)), discharge burnup, storage time before reprocessing, storage time after reprocessing and other secondary factors. The isotopic compositions of different types of fuel, given in Table 1, will be used to illustrate the considerations outlined in this section.

In Table 1, the total of Pu-241 and Am-241 is quoted as this is the only way to maintain constant any given quantity of Pu over periods during which Am is not removed, namely during storage of SNF or of separated Pu and during MOX fabrication. Indeed, stripping away the Am takes place only during reprocessing of the SNF, during purification of aged separated Pu or during wet recycling of scrap material. Americium results from the beta decay of Pu-241, with a half-life of 14.4 years. This decay must be taken into account in the bookkeeping of quantities of Pu, 38% of the Pu-241 being converted into Am after ten years, 76% after 30 years and 96% after 70 years. This results in a loss of between 4 and 19% of Pu depending upon material type and irradiation, with the same quantity of Am being generated over these storage periods.

In typical LWR Pu, the *alpha* activity is mainly due to Pu-238 (half-life 88 years), Am-241 (half-life 438 years) and Pu-240 (half-life 6500 years). The main inconvenience is the resulting heat generation, which impacts on storage, transportation and ultimately disposal. With ageing of the Pu, the predominant source of heat becomes Am.

The energy distributions of the *gamma* activities of the various Pu isotopes and of Am are quite discrete and are taken as a basis for safeguards

TABLE 1. TYPICAL ISOTOPIC COMPOSITION OF PLUTONIUM (rounded wt%) [2]

NPP	WPu	Magnox	AGR	PWR	PWR	BWR	PWR	PWR
SNF	U	U	UOX	UOX	UOX	UOX	MOX	MOX
GW · d/tU	Low	5–6	18–24	33	55	30	33	60
Pu-238	0.0	0.3	0.6	1.6	3.3	2.8	2.7	5.5
Pu-239	94	69	54	60	50	55	42	34
Pu-240	5.5	25	31	24	27	23	28	30
Pu-241 + Am-241	0.5	4.2	10	9	10	14	18	19
Pu-242	0.02	1.1	5	5.3	9	5	8	12

controls, through gamma spectroscopy. Typically, the main contributors to the gamma activity of Pu are Am-241, Pu-236 and Pu-240. This radioactivity does not play any role in Pu weight balances, but it influences Pu management, mainly in MOX fuel fabrication and SNF disposal. After the Pu isotopes, Am is the main source of long term radiotoxicity of SNF. Thus, recycling Pu before Am builds up further is an effective way to reduce the radiotoxicity of nuclear waste.

The neutron activity of Pu is the basis of the quantitative safeguards verification of Pu quantities by passive neutron counting. This activity influences mainly transportation, MOX fuel fabrication and SNF disposal. In particular, it is the main contributor to exposure of personnel in the last stages of manufacture of MOX fuel assemblies. The neutron activity results typically in roughly equal amounts from (α, n) reactions, i.e. the reaction of the alphas with the oxygen of the oxide, and from spontaneous fissions that originate from the decays of Pu-240, Pu-238, Pu-242 and Am-241. It is therefore aggravated by the ageing of Pu and the resulting increase of Am content.

7.2. MOX fuel fabrication

Even the most modern facilities cannot process Pu containing more than about 3–4.5% Am; thus, the time period during which separated Pu should ideally be stored before being fabricated into MOX fuel is constrained depending upon the source of the material.

One solution is to blend such aged Pu with freshly separated Pu or with ex-WPu (ex-weapons Pu) or gas cooled reactor (GCR) Pu, so that the blended Pu meets the maximum Am content specification. Such a blending operation is apparently inexpensive. However, ex-WPu is likely to be dispositioned separately into MOX fuel for political reasons, which will probably persist long into the future, if not indefinitely. Large inventories of GCR Pu exist and will continue to expand if there are no plans for utilization. However, the owner of aged Pu may be under political pressure not to acquire such GCR Pu, since it increases the quantity of Pu to be recycled. An exchange of part of the aged Pu for GCR Pu is hardly acceptable for the owner of the latter, who has to manage a Pu stockpile for an indefinite length of time. So, blending with freshly separated LWR Pu is the most likely solution to avoid purification of aged Pu.

A further solution is to purify it. However, the stripped-out Am needs to be managed, either by adding it to the waste earmarked for final disposal (increasing the radiotoxicity of that waste) or by incinerating it as a target in liquid metal fast breeder reactors (LMFBRs) or other devices still to be developed (at the expense of increasing very significantly the fuel cycle cost).

Fabrication plants specifically designed and licensed to operate with higher Am content are a possibility [3] at some point. However, such operation is unlikely to be incorporated into fabrication plants for LMFBR fuel, given the high Pu content of those fuels and the preference for conditioning Am in targets to be located in moderated zones of the LMFBR core, providing more efficient incineration of the Am [4].

7.3. MOX fuel performance in LWRs

For LWRs, MOX fuel is originally designed to be equivalent to a reference U fuel for the reactivity lifetime. The design of MOX fuel takes into account the facts that:

(a) Pu-239 and Pu-241 are fissile, Pu-241 being the best.
(b) Pu-240 is an absorber, but at the same time a fertile material producing Pu-241.
(c) Pu-242 and Am-241, the decay product of Pu-241, are absorbers, Am being the worst.

To take into account the Pu isotopic composition, Belgonucléaire initiated a Pu equivalence formulation, enabling it to correct the Pu contents of each MOX fuel reload, as a function of the actual isotopic composition of the Pu [5]. A linear formula was selected:

$$Pue = (a \times Pu8) + (b \times Pu9) + (c \times Pu0) + (d \times Pu1) + (e \times Pu2) + (f \times Am)$$

where Pu8,…, Am are the weight per cents of the Pu isotopes and Am in (Put + Am) and a,…, f are factors defined by calculating the MOX fuel for Pu isotopic compositions departing, within reasonable limits, from the reference isotopic composition. These factors are not universal. They depend on the reactor type, fuel management, reactor operation (effect of decaying Pu-241), nominal Pu isotopic composition, nominal Pu content, etc.

Similar equivalence formulations, not all of them linear, have been adopted by almost all the MOX fuel designers; however, there are differences in the methodologies used to derive the formula.

A range of factors from calculations carried out for different LWR cases up to now is given in Table 2, for the linear formula given above.

To illustrate the effect of Pu ageing on fissile material management in a sustainability perspective, applying the equivalence factors taken as an example in this table to the Pu issued from 55 GW · d/tU fuel detailed in Table 1:

TABLE 2. REACTIVITY EQUIVALENCE FACTORS FOR LWR FUEL

Isotope	Range	Example
Pu-238	– 0.11 to – 0.8	– 0.51
Pu-239	1.0	1.0
Pu-240	– 0.19 to – 0.56	– 0.36
Pu-241	1.1 to 1.3	1.3
Pu-242	– 0.7 to –1.4	– 0.7
Am-241	–1.6 to –2.2	–2.0

(a) Separated Pu still containing its Am has lost 42% of its equivalence value (i.e. its technical value as fissile material) after 10 years of storage and 66% after 30 years.
(b) Aged Pu purified from its Am immediately before utilization (or separated from stored SNF) has lost 27% of its equivalence value after 10 years of storage, 37% after 30 years and 42% after 70 years.

The losses are even larger for Pu derived from BWR U fuel and more so from MOX fuel, but, of course, smaller for the other types of Pu (much smaller for Magnox Pu and negligible for WPu).

However, in all cases, delaying Pu utilization constitutes a waste of fissile material resources.

Plutonium separated from high burnup U fuel or from MOX fuel would require high Pu contents in the MOX fuels fabricated from it to provide equivalent reactivity. The current generation of MOX fabrication plants are not equipped to manufacture such fuel, and the use of the highly enriched MOX fuels in LWRs would result in unacceptable positive void coefficients. On a stand-alone basis, the number of times Pu separated from MOX/high burnup fuel may be recycled is constrained. In practice, however, these U and MOX spent fuels are reprocessed in dilution with lower burnup fuel, and the resulting Pu mix can be recycled several times in LWRs.

Ultimately, however, the isotopic composition will deteriorate too much, and its only possible use will be in fast reactors.

7.4. MOX fuel performance in LMFBRs

In LMFBRs, all the Pu isotopes are fissile. Here, the values of equivalence reactivity factors depend on the type and size of the LMFBR and on whether the core is annular or not. On the basis of calculations performed for the SNR-300, Super-Phénix and EFR reactors [6], the range of factors is given in Table 3.

Americium in LMFBRs is less of a nuclear poison than in LWRs. As indicated in Section 7.2, it should be isolated from the Pu and conditioned in targets, for efficient incineration in LMFBRs. The effect on reactivity, and hence on Pu equivalence, is, however, practically the same as if it were left incorporated in the Pu.

To illustrate the effect of Pu ageing on fissile material management in a sustainability perspective, applying the equivalence factors in Table 3 taken as an example to the Pu issued from the fuel mentioned in Table 1:

(a) Plutonium from 55 GW · d/tU fuel will lose 10% of its equivalence value (i.e. its technical value as fissile material) after 10 years of storage, 22% after 30 years and 26% after 70 years.
(b) Plutonium from 33 GW · d/t MOX fuel (i.e. the majority of the current spent MOX inventory) will lose 17% of its equivalence value after 10 years of storage, 36% after 30 years and 42% after 70 years.

In comparing these figures with the ones quoted in Section 7.3, it is clear that equivalent fissile material losses due to Pu ageing are slower for use in LMFBRs than in LWRs.

TABLE 3. REACTIVITY EQUIVALENCE FACTORS FOR LMFBR FUEL

Isotope	Range	Example
Pu-238	0.43 to 0.65	0.45
Pu-239	1.0	1.0
Pu-240	0.13 to 0.15	0.14
Pu-241	1.5	1.5
Pu-242	0.03 to 0.05	0.04
Am-241	– 0.28 to – 0.35	– 0.32

As all isotopes are more or less fissile in LMFBRs, these reactors offer an effective solution to the issues involved in managing inventories of aged (long stored) or degraded (by irradiation in LWRs) Pu. In this respect, their superiority, as a Pu burner or self-sufficient breeder, has been clearly demonstrated in several studies and full scale experiments. Firstly, it is possible theoretically to increase the Pu content of the fuel up to 100%, but more realistically to 45%, without unduly affecting the safety parameters of the reactor. Secondly, the Pu consumption, even if the Pu grade is poor, is accompanied by a limited production of minor actinides.

7.5. Plutonium ageing as a sustainability concern in fissile material management

Not utilizing the Pu generated in the NPPs is, of course, a waste of fissile resources. Delaying its utilization also results in a reduction of fissile material resources and, in most cases, increases the Am liability. Adopting a wait and see policy, in expectation of better Pu utilization in LMFBRs to be deployed in an undefined future, leads to a net loss of fissile material resources. While the Pu ageing losses are already very important for current LWR SNF, the increasing discharge burnups currently observed and planned for the future will aggravate the ageing effect.

In France, Électricité de France (EDF) has defined its long term Pu management strategy from the perspective of coherence with deployment of LMFBRs in an uncertain future [7, 8]. Recycling Pu in PWRs to the extent currently implemented in France has been shown to provide for the best utilization of the Pu resource, whether deployment of LMFBRs occurs in the medium (2035–2050) or longer (2080–2100) term.

From the perspective of fissile material utilization, plutonium should be recycled through thermal reactors as soon as is practical. This early recycle will not compromise its future utilization if and/or when LMFBRs are deployed.

The loss of fissile material inventory by delayed Pu utilization is not the only consideration to be taken into account in defining a back end policy. Issues such as political perspectives, public acceptance, adequate MOX industrial infrastructure, economics and safeguards will all play a part in defining strategy.

7.6. Inventory of separated plutonium

Since 1996 the inventory of separated plutonium has been reported to the IAEA as part of INFCIRC/549. Nine countries in total report their holdings, which represent the vast majority of total separated plutonium. Those countries that do not regularly report are estimated to have holdings in the

range of 3–5 t. The INFCIRC data show that since 1996 the levels of separated plutonium have increased from some 160 t to over 233 t.

The increase in inventory has been due to a number of different factors. Insufficient MOX fabrication capacity due to delays in startup of some facilities, ramp-up or authorization for increased licensed capacity. This impact has been somewhat ameliorated by the reduced throughput achieved by reprocessing plants. The largest contribution to the increase has come from the separation of plutonium without a near-term end use. The position in the UK is that the Nuclear Decommissioning Authority is currently assessing options for dealing with the UK holding of separated plutonium.

Recent analysis undertaken using the IAEA VISTA code [9] has defined a long range forecast of separated Pu holdings worldwide for the period 1980–2050. The calculated results of the inventory are broadly comparable with those reported in INFCIRC/549 up to 2003, indicating that the VISTA model can provide a reasonable prediction of world fuel cycle inventory (Fig. 1). The model has assumed a low increase in generation, three different reprocessing scenarios (0, 30 and 70% of arisings) and a medium MOX utilization scenario

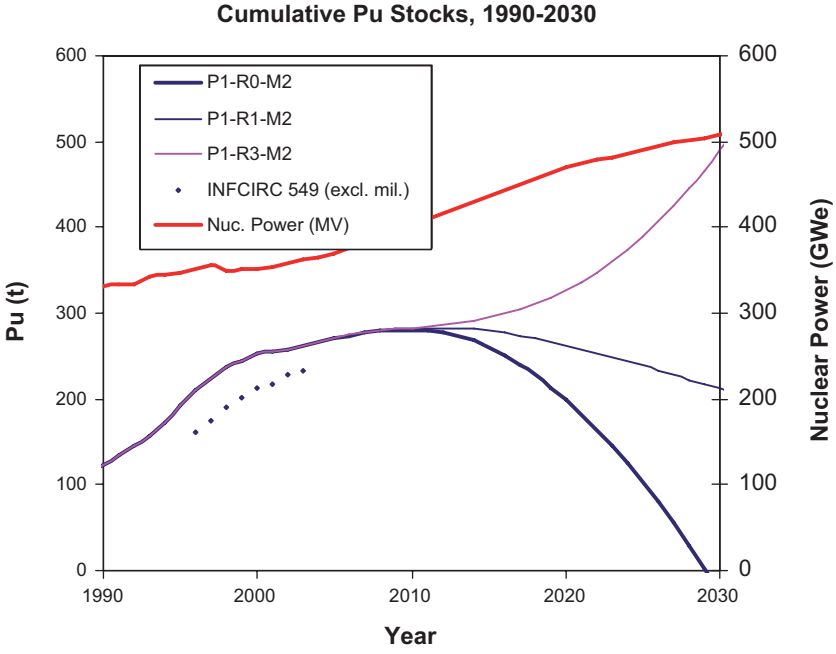

FIG. 1. VISTA modelling of separated Pu inventory [9].

based on historic utilization rates up to 2004, 4% of LWR power generated from MOX from 2010 to 2030, rising thereafter to 7% in 2050.

In the low reprocessing scenario (0% after 2015: designated the R0 case in Fig. 1), VISTA predicts a rapid decrease in separated Pu holdings, falling to zero by 2030. In addition, the level of holdings assuming a 30% reprocessing ratio (designated the R1 case in Fig. 1) starts to decrease after a peak in 2012. Continuing the trend leads to a zero holding circa 2040–2050. The 70% reprocessing scenario (designated the R3 case in Fig. 1) indicates an increasing inventory. This analysis indicates the close relationship between the rate of separation of plutonium and the rate of utilization in MOX fuel. It also indicates that the stockpile of separated plutonium currently held can be managed by the further industrialization of MOX. In reality, the separated plutonium holdings will never fall to zero if there is an ongoing programme of recycle, as a reserve of separated plutonium equating to approximately two years throughput needs to be held by the MOX fabricators for operational purposes.

The VISTA code analysis provides a useful trend based on global assumptions; however, the levels of separated plutonium held by specific countries who have yet to commit to an active management strategy will mean that the levels will not reach the minimum required as a production reserve.

8. CONCLUSIONS

Considerable experience has been gained over the last fifty years of storage, transportation, utilization, physical protection and safeguarding of separated plutonium.

Modern storage facilities are designed to withstand seismic events, with strict safeguards monitoring and surveillance and robust physical security features built in.

In the past, separated plutonium has been transported by road, rail, air and sea; however, in the future, plutonium will be internationally transported predominantly as MOX fuel.

There are a number of international, regional and bilateral agreements in existence which ensure that civil nuclear materials and equipment are not used for non-peaceful purposes. There is no record of material stored under international safeguards ever being diverted.

The responsibility for the establishment, implementation and maintenance of civil security arrangements is the sovereign responsibility of a country's national government.

There are three management options for dealing with separated plutonium:

(1) Continued storage (wait and see);
(2) Immobilization and disposal;
(3) Recycle in reactors.

The continued storage option is well understood and industrialized, having been followed for the last 50 years. The recycle option is currently followed in many countries and is industrially mature. The immobilization option is new and has not been fully developed.

The effect of delay in the utilization of fissile materials is well understood in terms of the reactivity equivalence of MOX fuel in both thermal and fast reactor systems.

From a fissile material utilization perspective, plutonium should be recycled through thermal reactors as soon as is practical. This early recycle will not compromise its future utilization in fast reactors if they are deployed.

Recent analysis using the IAEA's VISTA computer code has projected credible scenarios which show that the inventory of separated plutonium is capable of being reduced to a minimal level with a low MOX utilization rate.

Appendix I

UNITED KINGDOM STAKEHOLDER DIALOGUE: PLUTONIUM DISPOSITION OPTIONS

TABLE 4. IMMOBILIZATION OPTIONS

No.	Option	Process	Initial assessment
1	Ceramic: New build	PuO_2 converted to ceramic pucks (95 mm diameter, 16 mm thick) using a purpose designed matrix material. Pucks loaded into Pu cans. New purpose built facility required. Cans are held in interim store pending ultimate disposal.	Worth pursuing further
2	Ceramic: SMP[a] Mod.	SMP modified at the end of its MOX fuel producing lifetime to produce ceramic pucks (95 mm diameter, 16 mm thick) using PuO_2 and a purpose designed matrix material as in Option 1 above. Pucks loaded into Pu cans. Cans are held in interim storage pending ultimate disposal.	Worth pursuing further

For footnotes see end of table.

No.	Option	Process	Initial assessment
3	Low specification MOX: SMP Mod.	SMP converted at the end of its MOX fuel producing life to produce low specification MOX pellets, which would be loaded into sealed rods within the plant. These rods would then be transferred to a suitable facility for interim storage, pending ultimate disposal.	For later assessment. PuWG[b] asked for this option to be included along with the initial options selected in Apr. 2001.
4	Vitrification: New build	New vitrification facility. PuO_2 powder is vitrified in a suitable glass. Poured into Pu cans for interim storage pending ultimate disposal.	Worth pursuing further
5	Ceramic + VHLW[c] barrier: New build	PuO_2 converted to ceramic pucks (95 mm diameter, 16 mm thick) using purpose designed matrix material and loaded into Pu cans. Cans are loaded into VPS[d] style canisters and surrounded by a VHLW barrier. Canisters are stored in VPS style facility pending ultimate disposal.	Worth pursuing further. PuWG asked in Apr. 2001 that a variant of this option using the existing vitrification plant be pursued.
6	Ceramic + barrier: New build	PuO_2 mixed with some form of HLW, formed into ceramic pucks using a purpose designed matrix material. Pucks are of the same dimensions as in Option 1 above. Pucks are loaded into Pu cans. Cans are loaded into VPS style canisters for interim storage pending ultimate disposal.	Considerable engineering difficulties in mixing α and β plants. Product not perceived to bring a benefit over can-in-canister. No further work at this time.
7	Ceramic + VHLW barrier: SMP Mod.	SMP modified at the end of its MOX fuel producing lifetime to produce ceramic pucks (95 mm diameter, 16 mm thick) using PuO_2 and a purpose designed matrix material as in Option 1 above. Pucks loaded into Pu cans. Cans then loaded into VPS style canisters and surrounded by a VHLW barrier. Canisters are stored in a VPS style facility pending ultimate disposal.	Worth pursuing further

No.	Option	Process	Initial assessment
8	Low specification MOX + VHLW barrier: SMP Mod.	As Option 7 but instead of ceramic pucks, low specification MOX pellets would be produced and loaded into Pu cans. Cans are then loaded into VPS style canisters and surrounded by a VHLW barrier. Canisters are stored in a VPS style facility pending ultimate disposal.	Because of similarity to Option 7, further assessment has been deferred until later.
9	Low specification MOX + spent fuel barrier: SMP Mod.	As Option 3 but the sealed rods would then be transferred to a suitable facility for loading into appropriate spent fuel assemblies. Spent fuel rods would be removed and the low specification MOX rods inserted such that the remaining irradiated rods would act as a radiation barrier.	This option has not been pursued further since it raised a number of questions, e.g. dose uptake with respect to spent fuel handling required and fate of the irradiated rods removed from the assemblies. PuWG asked in Apr. 2001 that the alternative option of building complete assemblies with low specification MOX fuel rods and then storing these with spent fuel in storage ponds on-site be assessed.
10	Vitrification + barrier: New build	New vitrification facility. PuO_2 is vitrified in glass. Poured into Pu cans. Cans are loaded into VPS style canisters and surrounded by VHLW. Canisters are stored in a VPS style facility pending ultimate disposal.	Worth pursuing further

[a] SMP: Sellafield MOX plant.
[b] PuWG: Plutonium Working Group.
[c] VHLW: Vitrified high level waste.
[d] VPS: Vitrified product store.

Appendix II

DESCRIPTION OF APPROACH TO PLUTONIUM ACCOUNTANCY UNITS

Total Pu, Put, defined either as the sum of all the Pu isotopes (a quantity varying with time, since Pu-241 decays into Am-241) or as the sum of all the Pu isotopes plus Am (a more stable quantity, although also varying, whenever reprocessing or Pu purification takes place). It is the traditional unit for safeguards purposes, adopted both by the IAEA and by the Euratom Safeguards Office (ESO) and the European Supply Agency (ESA). As a result, it has been implemented in the Guidelines for the Management of Plutonium (INFCIRC/549), under which some IAEA Member States annually report holdings of civil unirradiated Pu [10]. It is also the commonly used unit in regulations and licences for storage, transportation and manufacturing of material containing Pu. Very often, however, the decay of Pu-241 is not taken into account, for convenience, so that the 'Put' is actually Put + Am, a quantity remaining constant with time.

Fissile Pu, Puf, defined as the sum of Pu-239 and Pu-241, is also a quantity that varies with time. It is the traditional unit incorporated in reprocessing contracts. As such, it has been adopted by customs and by the ESA whenever borders of the European Union are crossed.

Equivalent Pu, Pue, is defined as the quantity of Pu providing the same usage value to a MOX fuel assembly as the reference uranium fuel assembly taken as the basis of the nuclear design. It should be used to satisfy the utility that their fuel meets the reload requirements (Section 7.3). In some cases, however, the authorities express reload licensing limits in Put or in Puf, and such units have then to prevail over Pue.

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REPROCESSED URANIUM ISSUES

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Abstract

The residual ^{235}U isotopic content of reprocessed U (RepU) depends on the reactor type and discharge burnup of the fuel. Its ^{236}U content is another factor that determines its value as fissile material to fuel various types of reactor. Two other isotopes, ^{232}U and ^{234}U, present in larger quantities in RepU than in virgin U, result in an increased radiological source term. All these factors impact on the management, processing and recycling of RepU. Reprocessing in several countries has generated large quantities of RepU. Its purification and conditioning for storage, re-enrichment and/or direct utilization are now routine operations. Since, unlike Pu, the fissile value of RepU does not deteriorate with time, it can be stored as an indigenous fissile material reserve for an unlimited period. If economic conditions or future uncertainties dictate recycling it immediately, this can be and is being done on an industrial scale. The re-enrichment can be and has been performed by centrifugation and by blending with higher enrichment LEU, with MEU or with HEU. Experience has shown that the resulting enriched RepU (ERU) meets the standards for feed material for fabrication into fuel and loading in LWRs, AGRs, RBMKs and HWRs. Experience with ERU fuel fabrication is significant, as well as its utilization in NPPs. Some countries are recycling all their RepU, in order not to accumulate a stockpile, which might become a liability; others are recycling only part of their arisings, to establish and maintain the technology and their competence in it. The evolutions of RepU arisings and of their isotopic quality up to 2050 have been modelled for various power generation and reprocessing rate scenarios.

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1. INTRODUCTION

Utilization of reprocessed uranium (RepU) in LWRs began in the German nuclear reactor at Obrigheim in 1983, which was followed in the late 1980s by France and Belgium. At present, nine countries, including India, Japan, the Netherlands, the Russian Federation, Sweden and Switzerland have experience of recycling of RepU in NPPs.

In the mid-1970s, when the price of uranium was high, RepU from Magnox reprocessing was re-enriched and fabricated into fuel for the AGRs. Over 15 000 tU of Magnox RepU was recycled to produce over 1500 tU of AGR fuel. The recycle ceased as it became economically unattractive due to a combination of factors such as the low ^{235}U content of Magnox RepU and falling natural uranium prices.

The RepU inventory was calculated by the IAEA VISTA code to provide useful information on RepU utilization. This calculation includes annual RepU arisings, the amount of ^{235}U in RepU and the cumulative amount of RepU up to 2050.

For the use of RepU as nuclear fuel, there are two processes; one is blending the RepU with enriched uranium and another is re-enrichment of the RepU to increase the ^{235}U assay. In both cases, it is designated as enriched RepU (ERU).

2. CHARACTERISTICS OF RepU

Reprocessed U contains several minor uranium isotopes in addition to three naturally occurring ones (^{234}U, ^{235}U and ^{238}U). The isotopes that impact on RepU utilization are as follows [1, 2]:

(a) Uranium-232 is created via neutron absorption and decay steps. Because of the long and complicated generation process, ^{232}U appears in very small quantities (1 ng ^{232}U per gram U). The nuclide ^{232}U is not itself a major radiological hazard, but its daughter products, ^{212}Pb, ^{212}Bi and ^{208}Tl, emit intense beta and gamma radiation. In particular, ^{208}Tl gives off a strong gamma ray, with an average energy of 3.4 MeV. The hazard from ^{232}U daughter products is dependent on time elapsed since reprocessing. Shortly after reprocessing, there is no need for personal radiation protection, but gamma radiation from the daughter isotopes increases with time, reaching a peak approximately ten years after reprocessing.

(b) Uranium-234 is a naturally occurring isotope. The concentration of ^{234}U in natural uranium is typically 54 μg ^{234}U per gram of uranium. The ^{234}U

in RepU is predominantly a personal protection issue due to its moderate half-life and strong alpha emission. This nuclide is a neutron absorber, which reduces energy production, but this absorption generates ^{235}U, which, with another neutron, fissions to release an average of 2.4 neutrons, so that net neutron impact is small.

(c) Uranium-236 is created in reactors when ^{235}U absorbs a neutron. The concentration of ^{236}U varies with burnup, but is typically around 0.5 wt%. It is a long lived alpha emitter, which decays to the even longer lived ^{232}Th. Therefore, ^{236}U is not a significant radiological hazard. However, it absorbs a neutron to create ^{237}U, which then decays to the long lived nuclide ^{237}Np. This process removes neutrons to a significant extent from the chain reaction, reducing fission events in fuel. The presence of ^{236}U in fuel must therefore be offset by increasing the concentration of ^{235}U in fuel.

If not utilized, RepU constitutes a liability consisting of the costs of conditioning, interim storage and final disposal. It has, however, a fissile value, which is negatively affected by the quite strong reactivity losses (the 'poisoning effect') of ^{234}U in LWRs. In the neutron spectra of PHWRs, RBMKs, AGRs and LMFRs, this poisoning effect is much lower, and RepU should find an optimum utilization in these types of reactor. However, these types of reactor are not numerous enough to utilize all the RepU arisings, and this situation will persist for many decades into the future (Section 1). Moreover, except for RBMKs, the current price level of natural U prevents RepU from being competitive. Applied in the past, RepU utilization has been discontinued in AGRs for this reason.

As a result, in countries where there is political uncertainty about the future of nuclear power generation (Section 4), such as Belgium, Germany, the Netherlands and Switzerland, utilities are recycling RepU in their LWRs. In other countries, such as France and Japan, it is applied on a reduced scale to ensure the availability of the technology.

3. SEPARATION OF RepU BY REPROCESSING

Spent fuel from NPPs contains 94–96 wt% of uranium, depending on burnup. The remaining 4–6 wt% is composed of plutonium, minor actinides and fission products. Reprocessing of spent fuel separates the uranium (RepU), which can thus be processed or stored for further use. The U content of SNF from current NPPs is 93–95 wt% of the U initially contained in the fresh fuel.

The evolution of RepU arisings therefore almost enables a quantitative image of the reprocessed SNF to be formed (Table 1).

By the end of 2003, the total RepU quantity extracted by reprocessing was about 78 000 tU, and about 28 000 t RepU of this were derived from spent commercial LWR fuel, mainly in La Hague (France) and THORP (United Kingdom (UK), which offer commercial reprocessing services to other countries. The breakdown of this quantity by owner of the reprocessed spent fuel is shown in Fig. 1. The bulk belongs to French and Japanese companies (primarily utilities), accounting for more than half of the total. German companies rank third in terms of RepU received from the reprocessing of their spent LWR fuel [1, 3].

Again, according to preliminary statistics, from 2004 to the end of 2010, another 10 000 t RepU are currently expected to arise from the reprocessing of spent LWR fuel (Section 11). Almost two thirds of this total will come from reprocessing fuel from Électricité de France (EDF) [1, 3].

TABLE 1. RepU PRODUCTION [1, 3]

Country	Reprocessing plant	Period	Reactor type	Cumulative amount reprocessed (tHM)	Cumulative RepU extracted (tHM)
Belgium	Eurochemic	1966–1974	LWR RR	95 86	170, of which 31 for MTR
France	Marcoule (UP1)	1958–1976 Up to 1998	GCR LWR	14 000	13 000
	La Hague (UP2 and UP3)	1966–1987 To end 2005	GCR LWR	5 000 25 300	4 600 23 000
Germany	WAK	1970–1990	LWR, RR	150	(140)
India	TPP Prefre-1 KARP		RR PHWR PHWR	55 250 230	? ? ?
Japan	Tokai Rokkasho	1978–2004 From 2007	ATR, LWR LWR	1 000 —	(940) —
Russian Federation	RF1 (Mayak)	1977–2004	WWER, RR, BN-600	4 300	4 100
United Kingdom	B205 THORP	1964–2012 1995–2004	Magnox AGR, LWR	41 000 5 100	39 000 4 800
United States of America	NFS (W. Valley)	1966–1972	LWR	430	(400)

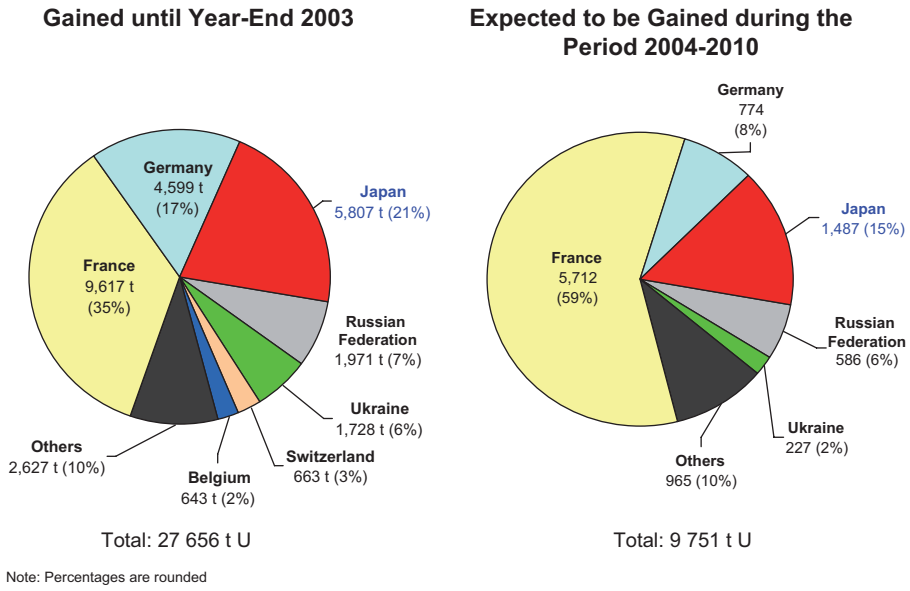

FIG. 1. Reprocessed uranium gained from reprocessing of spent commercial LWR fuel by owner of spent fuel [1, 3].

4. ENRICHMENT (ERU)

For enrichment of RepU by blending with higher enriched U, centrifugation is practically the only enrichment technology applicable, because it has very few delays and therefore lower sensibility to pollution by even isotopes of uranium. Moreover, unlike diffusion plants, it is possible to isolate some of their centrifuges and dedicate them to enrichment of reprocessed uranium. Table 2 provides data of enrichment facilities worldwide [3, 4].

5. RepU FUEL FABRICATION

COGEMA, France, sent a questionnaire to EDF (France), TRACTEBEL (Belgium) and KEPCO (Japan), to survey the status of ERU utilization. The major conclusions from experience with ERU are as follows [4, 5]:

(a) The cumulative experience of re-enriched RepU fuel fabrication has now passed 200 tonnes of re-enriched RepU. This corresponds to approximately

TABLE 2. ENRICHMENT FACILITIES WORLDWIDE [3, 4]

Country	Location	Operator	Capacity (kSWU/a)	Enrichment process	First operation
China	Lanzhou (shut down)	CNNC	450	Gaseous diffusion	1980
	Lanzhou	CNNC	500	Centrifuge	1997
	Shaanxi	CNNC	500	Centrifuge	2003
France	Tricastin	Eurodif	10 800	Gaseous diffusion	1979
Germany	Gronau	Urenco	1 300	Centrifuge	1985
Japan	Rokkasho-Mura 1	JNFL	600	Centrifuge	1992
	Rokkasho-Mura 2	JNFL	450	Centrifuge	1997
Netherlands	Almelo	Urenco	1 500	Centrifuge (RepU)	1973
Russian Federation	Angarsk	Minatom	1 400	Centrifuge	1954
	Ekaterinburg	Minatom	10 000	Centrifuge	1949
	Krasnoyarsk	Minatom	5 700	Centrifuge	1964
	Seversk	Minatom	2 900	Centrifuge (RepU)	1950
UK	Capenhurst	Urenco	2 000	Centrifuge	1976
United States of America	Paducah (shut down)	USDOE/USEC	11 300	Gaseous diffusion	1954
	Portsmouth	USDOE/USEC	7 400	Gaseous diffusion	1956
Total			45 000, of which 4300 operate partly on RepU		

1250 tonnes U in ‘raw’ RepU. All of this experience has been gained with PWRs.

(b) There are several reactor core situations in terms of the re-enriched RepU isotope composition or of loading patterns.

(c) The same burnup was achieved with re-enriched RepU and raw RepU.

In addition to these countries and utilities, RepU fuel has been fabricated in Germany, India and the Russian Federation. In particular, OAO MSZ (Mashino-Stroitelny Zavod), Russian Federation, has provided ERU fuel for

TABLE 3. RepU FUEL FABRICATION

Country	Fabricator	Year	Licensed capacity	Fabricated	Reactor type
Belgium	FBFC International	1989	Case by case	43 t ERU	PWR
France	FBFC, Romans	1993–2002	150 t ERU/a	350 t ERU	PWR
Germany	ANF, Lingen	2003–present	?	?	PWR BWR
India	?	1980–2004	?	57 t ERU	PHWR
Japan	MNF Tokai NFI Kumatori NFI Tokai		22 t ERU/a 30 t ERU/a 22 t ERU/a		PWR PWR BWR
Russian Federation	OAO MSZ Elektrostal	1996–2003	?	424 t ERU/a	RBMK WWER LWR

RBMKs for many years and, more recently on a demonstration scale, for WWERs [6]. In addition, it manufactures ERU fuel for LWRs for foreign utilities under an agreement with Siemens/Framatome ANP. Table 3 provides data for RepU fuel fabrication, on the basis of miscellaneous sources of information.

6. BURNUP EFFECT

As discharge burnups are being increased, the uranium to be recycled is issued from fuel discharged at a lower burnup than the new fuel to be fabricated. This is particularly the case because spent fuel is stored on average for 10–17 years before reprocessing, and the RepU is sometimes not utilized immediately to make new fuel.

Table 4 relates to RepU recovered from SNF initially at 3.5% ^{235}U, irradiated to 38 GW·d/tU, stored for ten years before reprocessing and to be centrifuge re-enriched and refabricated into enriched reprocessed fuel (ERU), equivalent to natural U (NatU) enriched to 4%. Enriched NatU will be designated hereafter by ENU.

The enrichment requirements are greatly influenced by the discharge burnup of the fuel from which the RepU is recovered. As an example, Table 5 provides data for RepU from spent fuel initially enriched at 3.5% ^{235}U and to be re-enriched and fabricated into ERU fuel equivalent to 4.4% ENU. The tails assay of enrichment is supposed to be 0.3% ^{235}U.

TABLE 4. EXAMPLE OF ERU EQUIVALENT TO 4% U-235 ENU

Isotope	RepU	ERU
U-232	0.07 ppb	0.77 ppb
U-234	0.021%	0.152%
U-235	0.84%	4.59%
U-236	0.49%	1.77%
U-238	98.65%	93.49%

TABLE 5. ENRICHMENT REQUIREMENTS

Type	Original U			Enriched U	Enrichment requirements	
	GW·d/tU	% U-235	% U-236	% U-235	Feed	SWU
NatU	0	0.71	Negligible	4.40	10.0	6.0
RepU	32	1.02	0.43	4.85	6.4	5.1
	36	0.84	0.45	5.00	8.7	6.3
	40	0.69	0.47	5.18	12.5	7.7
	44	0.56	0.48	5.45	19.7	9.6
	48	0.45	0.49	5.80	36.4	12.1

7. FRESH FUEL COSTS FOR CENTRIFUGE ENRICHED ERU

In the cost comparison, we will assume that the RepU resource is cost-free, i.e. has no commercial value. This is not correct, because RepU not utilized as a resource would need to be dispositioned as waste and the avoided waste disposal cost should be recorded as a bonus, i.e. a cost reduction.

The cost constituents are:

(a) Conversion of uranyl nitrate hexahydrate into UF_6 or, for extended interim storage, first into U_3O_8 (at COGEMA) or UO_3 (at BNFL) and, after storage, this oxide into UF_6 (per kgU to be converted, the cost is six times higher for RepU than for NatU);
(b) Enrichment into UF_6 of equivalent reactivity;
(c) Fabrication into an ERU fuel assembly, assumed hereafter to be for a European PWR.

TABLE 6. COST FOR RepU ISSUED FROM 32 GW·d/tU (euro/kgU)

Uranium type	ENU 4%	ERU 4.41%	
Uranium	280	—	—
Conversion into oxide	—	—	90
Conversion into UF_6	60	290	290
Enrichment	500	420	420
Fabrication	370	450	450
Fuel assembly cost	1210	1160	1250

TABLE 7. COST FOR RepU ISSUED FROM 40 GW·d/tU (euro/kgU)

Uranium type	ENU 4%	ERU 4.71%	
Uranium	280	—	—
Conversion into oxide	—	—	200
Conversion into UF_6	60	550	550
Enrichment	500	660	660
Fabrication	370	450	450
Fuel assembly cost	1210	1660	1860

TABLE 8. COST FOR RepU ISSUED FROM 44 GW·d/tU (euro/kgU)

Uranium type	ENU 4%	ERU 4.95%	
Uranium	280	—	—
Conversion into oxide	—	—	310
Conversion into UF_6	60	860	860
Enrichment	500	830	830
Fabrication	370	450	450
Fuel assembly cost	1210	2140	2450

Tables 6–8 show, for the RepU qualities outlined in Table 5 and to be refabricated into ERU fuel equivalent to ENU with 4.0% ^{235}U, the impact of the burnup from which the RepU is issued, as well as the influence of interim storage of the RepU before it is recycled.

These tables illustrate the economic impact of the fuel burnup from which the RepU is issued. The prices result from contractual negotiations, in which the quantities and other commercial considerations play a major role. However, competition is not very strong, since the enrichment services are only available from two plants:

(a) The Siberian Chemical Plant (Seversk, Tomsk region), which offers conversion into UF_6 and then re-enrichment;
(b) The Urenco Almelo plant (Netherlands), which offers only the re-enrichment service. The prior conversion into UF_6 is to be contracted to the COMURHEX plant (Pierrelatte, France), BNFL having decided to close their Springfields conversion plant, in principle also capable of operating on RepU.

8. ENRICHMENT BY BLENDING WITH HEU

The data provided in Section 6 indicate that the 5% ^{235}U limit, to which almost all the fuel industry infrastructure is designed and licensed for, is quickly reached, as the target burnup is now increasing up to 60 GW·d/tU. This would mean that the ERU fuel would soon not be able to be equivalent to standard U fuel and this would be a major fuel cycle cost penalty. Furthermore, as seen in Section 7, the fuel cost is increasing excessively when the burnup from which the RepU is originating increases.

Both effects are due to the increase in ^{236}U content by centrifuge enrichment of RepU. This inconvenience can be overcome by blending HEU with RepU, rather than enriching it by centrifuge. Table 9 compares the two ERU compositions for the fuel case illustrated in Table 4, i.e. RepU resulting from 3.5% ^{235}U fuel irradiated to 38 GW·d/tU, stored ten years before reprocessing and enriched to be equivalent to 4.0% ^{235}U ENU fuel. The example given in this table assumes that the HEU is weapons grade, in which case the blend consists of 96.3% RepU and 3.7% HEU, to achieve enrichment equivalence.

TABLE 9. INFLUENCE OF ENRICHMENT TECHNIQUE ON ERU COMPOSITION

Isotope	Feed material		ERU	
	RepU	HEU	Centrifuge	Blending
U-232	0.07 ppb	—	0.77 ppb	0.07 ppb
U-234	0.021%	0.93%	0.15%	0.05%
U-235	0.84%	90%	4.59%	4.16%
U-236	0.49%	—	1.77%	0.47%
U-238	98.65%	9.07%	93.49%	95.32%

The lower enrichment of ERU obtained by blending allows recycling of lower grade RepU and/or making higher burnup ERU fuel before being limited by the 5% ^{235}U ceiling. The lower ^{232}U content results in lower radioactivity levels in enrichment, transportation and fabrication steps. However, the total quantity of radioactive waste generated by recycling the RepU remains the same, since the total quantity of RepU to be treated remains the same.

The spent fuel quality of ERU fuel obtained by blending is intermediate between that of spent ERU fuel obtained by centrifuge enrichment and that of spent ENU fuel. Table 10 provides an example of spent fuel compositions for a PWR fuel of 4.0% ^{235}U equivalent ENU irradiated to 51 GW·d/tU and stored for five years. Taking into account the facts that 5.0% ^{235}U is the maximum admissible enrichment, for any fuel including ERU, and that the equivalent enrichment is the ERU enrichment minus typically 0.27–0.34 times its ^{236}U content, the following conclusions can be drawn for the possibilities of recycling the RepU originating from the three spent fuels, details of which are given in Table 10:

(a) If re-enrichment is performed by centrifuge, the highest achievable equivalent enrichment is 4.0% for the RepU ex-ENU, 2.5% for the RepU from ERU initially enriched by centrifuge (which means it can no longer be utilized) and 3.5% for the RepU from ERU initially enriched by blending (which means it can only be reutilized with a quite large reactivity penalty). Thus, a second recycle of RepU (i.e. recycle of RepU from a first ERU fuel) in a PWR cannot be considered if centrifuging is the only enrichment option.

TABLE 10. COMPOSITION OF SPENT FUELS

Fuel	ENU	ERU by centrifuge	ERU by blending
gU/kgU initial	935	933	934
ppb U-232	0.30	1.27	0.70
% U-234	0.02	0.09	0.03
% U-235	0.64	1.00	0.77
% U-236	0.58	2.30	1.06
% U-238	98.76	96.63	98.14
gPu/kgU initial	11.1	11.9	11.8
% Pu-238	3.24	7.9	4.05
% Pu-239	51.2	51.5	52.6
% Pu-240	25.2	22.7	23.7
% Pu-241	11.9	11.3	12.1
% Pu-242	8.4	6.58	7.6

(b) If re-enrichment is performed by blending, the three types of RepU can be recycled into PWR ERU fuel. The quantity of HEU to be blended is 4.0% for the RepU ex-NatU, 4.2% for the RepU from ERU initially enriched by centrifuge and 4.0% for the RepU from ERU initially enriched by blending. Thus, RepU can be recycled indefinitely in a PWR as long as HEU is available to blend it with, even if some intermediate recycles have been performed by centrifuge enrichment (provided centrifuge enrichment is not used in two successive recycling steps).

No cost reduction is obtained in this blending approach, on the contrary: for centrifuge enrichment of the fuels considered in Table 9, the feed factor is ten, which means that only one tenth of the RepU quantity is available as ERU and needs to be converted and fabricated, with the associated cost increases. For blending enrichment, 1.04 times the RepU quantity is available as ERU and needs to be converted and fabricated. Although the cost increase is lower, it applies to ten times more ERU fuel. Additionally, HEU is very expensive.

Blending RepU with such HEU is only possible in 'Weapons States', since access to this grade of HEU is restricted for non-proliferation reasons. In practice, only the Russian Federation and the United States of America (USA) could offer such enrichment services on a large scale. For the USA to do so is hardly conceivable, given their policy of not supporting reprocessing activities abroad. For the Russian Federation, almost all their weapons grade HEU from dismantled nuclear weapons is tied up in down-blending under the 1993 US–Russian HEU Purchase Agreement (Section 10). Any small surplus of HEU not tied up in this agreement is kept for market purposes. The demand is indeed high from customers still authorized to run their research reactors on HEU, and the production of HEU has been discontinued. Price levels are therefore high.

Production of ERU by blending RepU batches issued from spent HEU and LEU fuel has taken place in France and the Russian Federation on a commercial basis [7].

The Russian reprocessing complex RT-1, Production Association Mayak, reprocesses spent fuel from WWER-440, as well as spent fuel from the BN-600 breeder reactor and from naval reactors and spent HEU fuel from research reactors. Their fresh fuel enrichment levels vary from 40% (propulsion reactors) to more than 90% (research reactors).

The French company, COGEMA, is offering a reprocessing service for spent research reactor fuels at the La Hague plant. The spent research reactor fuel will be co-processed with the spent LEU fuels from power reactors. In this process HEU and LEU contained in the fuels are blended to produce ERU of about 1% enrichment.

9. ENRICHMENT BY BLENDING WITH MEU

The fuel often called 'HEU', utilized by TVEL, since 1981, for recycling WWER RepU into RBMK ERU fuel, arises from reprocessing spent fuel from the propulsion reactors of submarines and ice-breakers, as well as from fast and research reactors. The U recovered by the reprocessing operation is in fact MEU, having medium enrichment levels: 17–19% (Section 10 for more details). Part of this Russian MEU and the RepU from WWERs are blended to produce about 150 t ERU per year that is used to fabricate fuel for Russian RBMKs. The enrichment is currently 2.6% ^{235}U, but would increase to 2.8–3.0% after 2010, when it is planned to increase the RBMK discharge burnup to 30 GW·d/tU.

After the ordered shutdown of the Siemens fuel fabrication plant in Hanau, Siemens entered, in 1994, into negotiation with AO Elektrostalskiy Mashinostroitelniy Zavod (OAO MSZ), a subsidiary of the Joint Stock Company (JSC) TVEL, for the fabrication of RepU fuel. Not only did OAO MSZ offer a good substitute for the Hanau fabrication, which had taken place from 1982 to 1987 on a lead test assemblies level, but TVEL also offered re-enrichment by blending. After qualification and pilot fabrication in 1995, commercial fabrication at MSZ increased in quantity every year, for an increasing number of European utilities (customers of Framatome ANP GmbH). Initially limited to PWR fuel assemblies, it was extended to U–Gd pellets and BWR fuel assemblies in 2000. Currently, 200 t RepU is sent by European utilities to the Russian Federation for re-enrichment and ERU fabrication under these conditions. Russian MEU is used for blending with the European RepU (0.75% assay) to produce 35 t ERU per year with an assay of 4.0% equivalent.

In the commercial fabrication of European LWR fuel at MSZ Elektrostal from LWR RepU, TVEL utilizes MEU from propulsion reactors (14–17% ^{235}U residual enrichment level). The ^{236}U content of this MEU is not disclosed. However, the ^{236}U content is probably 1.1–1.4%. On this basis, Table 11 provides the isotopic composition range of ERU for the RepU case illustrated in Table 9.

By comparing Tables 9 and 11, it is clear that ERU obtained by blending with MEU is in any case of much better quality than ERU obtained by centrifuge enrichment.

As far as could be found from an investigation of the contractual conditions, the European customer has no right to know which MEU has been utilized to blend to the RepU. The customer receives back ERU fuel with the same quantity of ^{235}U as the RepU that was sent to the Russian Federation. To give an order of magnitude, this means that, for each 100 t RepU sent to the

TABLE 11. INFLUENCE OF ENRICHMENT BY MEU ON ERU COMPOSITION

Isotope	Feed material		Calculated ERU composition
	RepU	MEU	
U-232	0.07 ppb	—	0.05 ppb
U-234	0.021%	0.2–0.3(?)%	0.07–0.08(?)%
U-235	0.84%	14–17%	4.22–4.25%
U-236	0.49%	1.1–1.4%	0.62–0.73%
U-238	98.65%	82–85%	95%

Russian Federation, the customer receives back 20 t ERU, for the feed conditions illustrated in Table 11.

The agreement is favourable for both parties in that:

(a) The European utility recovers a quantity of ERU fuel equal to one fifth or one quarter of the RepU, which is much lower than the 104% mentioned in Section 8 as an inconvenience of the HEU dilution option. Moreover, the ^{232}U and ^{236}U contents are lower than the ERU resulting from centrifuge re-enrichment.

(b) MSZ can use the excess (80 t for each 100 t RepU supplied by the European customer) to blend it with 9 t MEU and fabricate 90 t ERU more RBMK fuel. Out of the feed necessary for RBMKs (450 t/a), 130 t is from WWER RepU, enriched by blending to 2.6–2.8% ^{235}U (the current RBMK fuel enrichment, since Gd is now used in those RBMK fuels and their discharge burnup is now 30 GW·d/tU). The quantity of WWER RepU produced annually is insufficient to feed all the RBMKs up to 50% ERU fuel, which is their current licence limit, a limit that could easily be extended to higher percentages. Utilization of ERU is interesting in these reactors, because the poisoning effect of ^{236}U is much lower in their neutron spectrum (as is also the case for CANDUs) than in the neutron spectra of LWRs. The over-enrichment compensation factor in ERU is only 5% ^{235}U/^{236}U for RBMKs, against 27–34% for LWRs and 25% for WWERs.

(c) No objection can be raised under the European Commission (EC) policy that forbids exporting radioactive waste outside the European Union (EU), as the RepU remaining in the Russian Federation is used as a fuel and is not a waste.

The various agreements between the USA and the Russian Federation do not impose limits on the latter for utilizing its MEU. However, the RepU must be Code N or P material. Sending Code A (USA), C (Canada) or S (Australia) to the Russian Federation for blending and processing would require special prior consent and result, at best, in administrative delays and restrictive conditions. In particular, the consenting country would most certainly require that the RBMK fuel fabricated with the excess RepU would bear the same obligation code, a condition unacceptable to the Russian Federation.

Therefore, this RepU recycling strategy is meeting with increasing commercial success in Europe (where most commercial fuel is or has acquired, by exchange, a Code N or P status). This opportunity has been utilized up to now mainly by Dutch, German, Swedish and Swiss utilities.

10. AVAILABILITY OF THE RUSSIAN RE-ENRICHMENT OPTION

It is uncertain how long this Russian re-enrichment option will remain available. There is strong international pressure to shut down the RBMKs. If and when this were to happen, the excess European RepU remaining in the Russian Federation would have to be converted into WWER ERU, of higher equivalent enrichment than RBMK ERU and thus requiring more HEU or MEU for blending. Demonstration ERU fuel assemblies have already been loaded in a WWER-440 plant and licensing submittals are being introduced to reload the WWER-440s and WWER-1000s with ERU fuel.

In the shorter term, however, the source of WWER-400 RepU will diminish, since the Armenian, Finnish, Hungarian and Slovakian WWER operators have decided not to sign reprocessing contracts. The only spent fuel deliveries to RT-1 now come from six Russian reactors (Kola-1, 2, 3 and 4 and Novovoronezh-3 and 4), two Bulgarian reactors (Kozloduy-3 and 4, to be shut down in 2006 [8]) and two Ukrainian reactors (Rovno-1 and 2), i.e. in total 90 and later 70 t RepU annually. Therefore, after the backlog of yet unreprocessed spent fuel that had accumulated in the RT-1 pools up to 1996 has been reprocessed, the WWER RepU will only be sufficient to fabricate approximately 100 and later 80 t ERU fuel for RBMKs. Since an additional 130 and later 150 t ERU would be desirable (Section 9), surplus LWR RepU from re-enrichment contracts would be welcome. On these bases, a contract level for a total of 170 t LWR ERU per year is justified. However, more is obviously considered better, probably for three reasons:

(1) There is hope of increasing the 50% licensed reload limit of RBMKs.
(2) Licensing submittals for loading ERU fuel in WWERs are in progress.

(3) Hard currency incomes from European contracts still have a high priority. For the European utilities, blending re-enrichment offers definitive advantages as outlined in Section 9.

The source of MEU will decrease when all the backlog of spent fuel from decommissioned submarines and ice-breakers has been exhausted:

(a) By the late 1990s, 184 retired nuclear powered Russian submarines had been decommissioned. How many of them had been dismantled, or at least unloaded of their fuel, prior to 2000 is unknown, probably 40–50. In the three years from 2001 to 2003, a total of 35 submarines were dismantled. Currently about 100–110 nuclear submarines are awaiting unloading of their fuel and decommissioning.
(b) A submarine core contains an average of 1 t MEU. If only submarine MEU were used to re-enrich LWR and WWER RepU, this source of MEU would be exhausted by 2008. However, since spent fuel from ice-breakers, BN-600 and research reactors is also contributing to the re-enrichment programme, one can assume that the MEU from dismantled submarines will only be exhausted by 2012–2013.

At the time of the Clinton administration, a senior US Department of Energy non-proliferation official proposed that, in as far as it would not perturb the US–Russia HEU Purchase Agreement, the Russian Federation could substitute MEU by material of equivalent value from its HEU stockpile, if the submarine fuel was in short supply for the Russian blending business. However, this has never been endorsed by the administration of George W. Bush. In any event, this resource is not infinite. In 1993, when the START 2 armament reduction treaty was signed between the USA and the Russian Federation, the total inventory of HEU was estimated at 1000 t in the Russian Federation and 650 t in the USA. As a result of this treaty:

(a) In March 1995, the USA declared 174 t as surplus to its defence needs and undertook to convert it progressively into LEU by blending initially with NatU and later with slightly enriched LEU.
(b) In 1993, the USA and Russian Federation signed the so-called US–Russia HEU–LEU Agreement under which the USA purchases (under bargain conditions) LEU derived from 500 t of Russian HEU to be blended in the Russian Federation with slightly enriched (1.5%) uranium. The first deliveries took place in 1995 and the last deliveries are scheduled for 2013. The approximately 300 t HEU still to be utilized out of the initial 500 t will be committed to fulfilling the obligations of the agreement.

Part of the initial 1000 t Russian HEU not demilitarized under the START 2 treaty might be released for civil use later on. In 2003, the SORT negotiations envisaged dividing by three the maximum tolerated number of warheads that each party was allowed to keep under the START 2 treaty. If the outcome of SORT were to be favourable, an additional quantity (≈300 t) of Russian HEU would become available, and it is unlikely that the Russian Federation would commit itself to an extension of the US–Russia HEU–LEU Agreement. Such an additional 300 t HEU could then be used to re-enrich over 8000 t RepU, enabling TVEL to offer re-enrichment by blending for more than 30 additional years, i.e. until the middle of the present century.

11. PERSPECTIVES OF RepU PRODUCTION UP TO 2050

Arisings of RepU worldwide up to 2050 have been calculated by VISTA simulations for different scenarios described in another paper in these proceedings [9]. Scenario P1 illustrated in Fig. 2 assumes that the nuclear power capacity grows from 370 GW(e) in 2005 to 565 GW(e) in 2050. The three scenarios R0, R1 and R3 represent the fraction of LWR spent fuel reprocessed to the total amount of spent fuel discharged from the reactors:

(1) R0: From 30% in 2005 to 0% in 2023, and no reprocessing thereafter.
(2) R1: 30% in 2005 is maintained until 2050.
(3) R3: 30% in 2005, increasing up to 70% in 2023, and 70% is maintained until 2050.

Figure 2(a) indicates that annual RepU arisings with the R0 and R1 scenarios decline, while those with the R3 scenario increase over the period. Declining arisings with R0 are obviously caused by no reprocessing after 2023. Annual arisings with R1, which slightly decrease in spite of growing nuclear power capacity, reflect the effect of increasing fuel discharge burnup. Annual arisings with the R3 scenario more than double by 2050 as compared with the arisings in 2005.

The annual ^{235}U amounts in RepU, illustrated in Fig. 2(b), exhibit the same tendency as the annual RepU arisings and are directly influenced by fuel burnup. The decline of RepU isotopic quality is visible on comparing Figs 2(a) and (b). As explained in Section 6, it reduces the attractiveness of RepU to refuel LWRs (including WWERs) and, to a lesser extent, RBMKs (which will anyway have disappeared by 2050) and HWRs. However, such degraded RepU

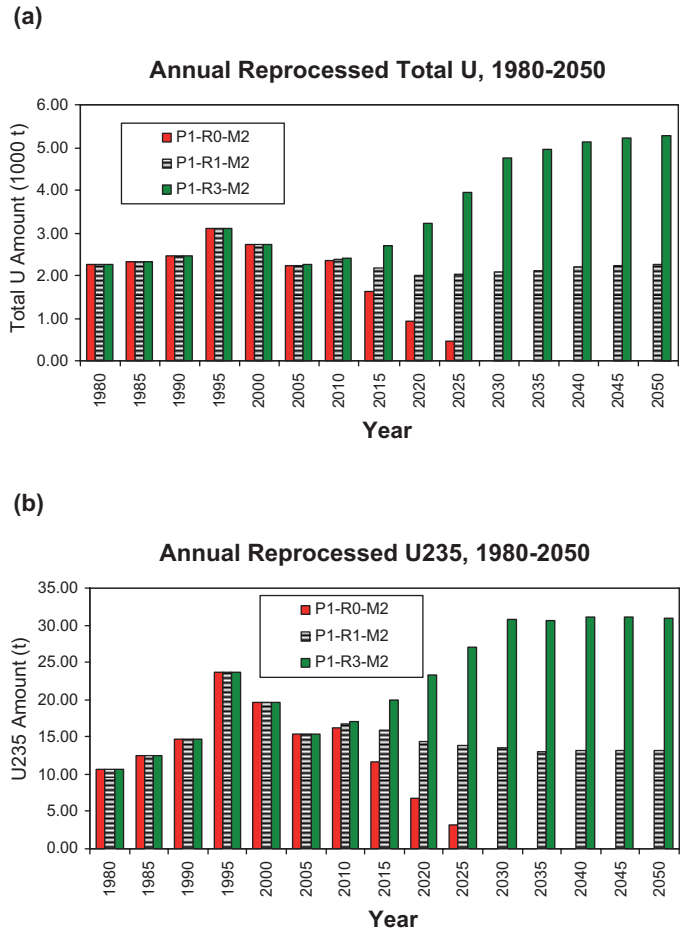

FIG. 2. (a) Annual RepU arisings and (b) amount of ^{235}U in RepU.

remains an almost undiminished resource for fuelling fast breeder reactors, which are likely to be deployed by or after the middle of the century.

In Fig. 3, the consumption of RepU for reuse as fuel is not included. As indicated in Section 3, about 78 000 t RepU were actually extracted up to 2003. The VISTA calculation results coincide fairly well with this fact.

On the basis of the VISTA scenarios, if no RepU at all were reused, the inventory would increase to between 114 000 and 265 000 t in 2050.

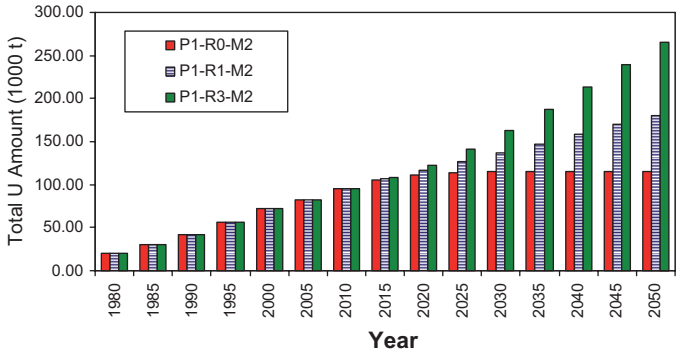

FIG. 3. Cumulative RepU arisings.

12. CONCLUSIONS

At current spent fuel discharge burnups, the RepU recuperated is a valuable resource for fuelling thermal reactors.

The re-enrichment of RepU can be and has been performed by centrifugation and by blending with higher enrichment LEU, with MEU or with HEU. Experience has shown that the resulting ERU meets the standards of feed material for fabrication into fuel and loading in LWRs, AGRs, RBMKs and HWRs.

Experience with ERU fuel fabrication is significant, as well as its utilization in NPPs. Some countries are recycling all their RepU, in order not to accumulate a stockpile, which might become a liability; others are recycling only part of their arisings, to establish and maintain the technology and their competence in it.

As discharge burnups increase, the attractiveness of RepU for fuelling thermal reactors decreases, but the RepU inventory remains a valuable resource for fuelling fast reactors.

Reprocessed U can be kept indefinitely as a strategic fissile material resource, as it does not deteriorate with ageing, unlike Pu.

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LONG TERM STORAGE AND DISPOSAL: COMPETING OR COMPLEMENTARY STRATEGIES FOR MANAGEMENT OF RADIOACTIVE WASTE?

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Abstract

The IAEA is engaged in two topics that are important for the sustainability of nuclear energy: long term storage of radioactive waste and disposal of radioactive waste. They are sometimes viewed as competing strategies for long term management of radioactive waste, and both appear frequently in discussions concerned with the sustainability of nuclear energy. The paper will contrast safety relevant aspects of long term storage and disposal, and provide some insights into the sustainability of these practices.

1. INTRODUCTION

Long term storage (LTS) is one of the radioactive waste management alternatives gaining more and more interest, especially in countries where there is strong public opposition to the final disposal of radioactive waste. A major goal of LTS is to ease the ultimate disposal of nuclear waste and spent fuel by waiting for a reduction of dose emissions and heat rejection. Volume reduction of low and intermediate level wastes could also result from a complete decay of their short lived isotopes after LTS.

Long term storage is also a means of 'buying time' for countries that have not yet made a decision as to whether spent nuclear fuel (SNF) is waste or an asset. The development of nuclear energy in future decades and depletion of uranium resources are important inputs for their decisions.

A major drawback of LTS is that it gives the impression that the problems are solved for a century or more, and that it does not make sense to worry about what will have to be done 100 years in the future.

Disposal promises to provide containment and isolation of radioactive waste from the human environment for the very long periods required to ensure safety, without the need for active controls. Near surface disposal has already been widely implemented and is accepted as a safe means for disposal

of short lived low level radioactive waste. The consensus of waste management specialists internationally, for example of the OECD Nuclear Energy Agency's Waste Management Committee, is that disposal in deep underground engineered facilities is the best option that is currently available or likely to be available in the foreseeable future for disposal of SNF and high level waste.

Final disposal of the vitrified waste from reprocessing raises no criticality problem and few or no safeguard concerns, and features much faster heat and activity decays than spent fuel. In addition, vitrified waste is not expected to ever have any economic value. The need for LTS to delay the final disposal of vitrified waste is thus far less justified than for SNF.

For public acceptance or operational reasons, a disposal facility may allow a significant period during which the waste will be kept retrievable. This could lead to so-called 'blended LTS' if this retrieval period is significant in comparison with the usual buffer or interim storage.

Alhough several disposal technologies are already available for most, if not all, waste types, there is still much scope for optimization, especially from an economic point of view. Emerging technologies such as plasma torches, partitioning and transmutation (P&T), with reactors or accelerator driven systems, and more generally with spent fuel treatment, have not yet shown their full potential. The risk of making wrong decisions by starting disposal too early is thus not negligible. By offering a more flexible decisions schedule, LTS leaves open many interesting options.

2. OBJECTIVES

The IAEA conference on the Safety of Radioactive Waste Management held in 2000 in Córdoba, Spain, produced the IAEA's first action plan for waste safety. The action plan included an action to assess the safety implications of the extended storage of radioactive waste and of any future reconditioning that may be necessary. In partial fulfilment of this action, the IAEA published a position paper entitled 'The Long Term Storage of Radioactive Waste: Safety and Sustainability' [1]. The position paper was written for a non-technical audience and examined the philosophical aspects of the issue, with much of the discussion at a policy level. Practical issues for the safety of LTS were not dealt with in any detail.

To build on the position paper, the IAEA held a technical meeting (23–27 May 2005), attended by participants from 16 countries, to draft an IAEA Safety Report to elaborate upon the more practical aspects of safety for LTS for all forms of radioactive waste. The aim of the IAEA Safety Report was to discuss

safety issues for LTS, with a view to extending the scope of the IAEA Safety Guide on storage of radioactive waste [2].

The present paper describes the findings of the technical meeting of May 2005, and hence reports on work in progress. Further development of the safety report will be addressed at subsequent specialist meetings and through the review process carried out by the IAEA standing committees for safety standards. The present paper will, as appropriate, compare LTS with disposal for the specific issues raised here.

3. DEFINITIONS

3.1. Types of storage

To place the issue of long term storage in context, it is important to keep in mind the various types of storage practiced:

(a) Decay storage — storage to allow radiation to decay to levels allowing onward handling or clearance. This would include short lived wastes and spent fuel.
(b) Buffer storage — storage to provide stock for ongoing processing. This might include unconditioned wastes and liquids waiting for conditioning.
(c) Interim storage — temporary storage whilst onward handling is being arranged. This might include the conditioned wastes waiting for disposal.
(d) Strategic storage — storage of material that may be a resource in the future, i.e. not declared as a waste. This might include spent fuel, plutonium or valuable materials such as irradiated zirconium.
(e) Legacy storage — These are old stores that did not incorporate retrievability criteria and now need to be retrieved due to their condition. These wastes would need to be re-characterized, retrieved, conditioned and then either stored for a further period or disposed of. In some cases, it may be possible to dispose of them in situ.

3.2. Duration of storage

To define safety issues for LTS, it was found to be useful to define storage practice according to duration of storage. The working definitions adopted at the May 2005 technical meeting were:

(a) Short and medium term storage — storage that could last up to fifty years. The fifty year time frame would appear to be reasonable because it covers:
   — The regulatory licensing periods of present day installations;
   — The time to produce material to make it economic to process;
   — A period over which buffer stocks are held to allow treatment and conditioning plants to be developed;
   — A period over which there is reasonable confidence that the operator will have sufficient funds to continue operating.

   This period would also cover the period to accumulate SNF for an input buffer store for a reprocessing plant.

(b) Extended or long term storage — applies to storage beyond fifty years, but with a defined end point for LTS. Fifty to one hundred years is chosen as this should allow adequate time to:
   — Substantiate disposal arguments;
   — Develop feasibility and technology studies;
   — Construct a disposal facility;
   — Gain regulatory authorizations.

   This could include decay storage, interim storage and strategic storage, although these might be for different lengths of time depending on the benefit offered by the storage. Long term storage may require justification for continuation of the practice, and it would require continuous review of the condition of the facility (facility, waste packages and waste forms), ageing management and life extensions.

(c) Indefinite storage — applies to storage for which an end point has *not* been defined.

4. REGULATORY ISSUES

This section summarizes regulatory issues that are anticipated to be of particular concern for LTS.

4.1. Ownership and operation

Owing to the long period of operation of an LTS facility, only governments can have a reasonable probability to meet the financial obligations and liability assurances for the intended period of storage. They will also have to make sure that a disposal facility will be ready at the closure of the LTS facility. An LTS facility should thus be owned by a government at the time of its commissioning or shortly thereafter.

4.2. Periodicity of reviews and licence renewals

The most likely initial licensing would be for a relatively short period (e.g. 20 years or less), with renewal options in place. Determination of licensing period and renewal periods would be based on experience, existing data and technical justification. It would be acknowledged that the facility would have a finite life and that the technical safety basis would have to be addressed at each renewal.

4.3. Long term compliance with current regulations

There is a need to ensure that an LTS facility remains in compliance with regulations, which may change over the lifetime of the facility. Governments would need to have a programme in place to address changing regulations over the lifetime of an LTS facility.

4.4. Waste package licensing for LTS

The technical safety requirements for the storage package (dry storage) may be different for LTS, and consideration should be given to the following:

(a) The licensing of the package may be separate or integral to the facility.
(b) The quality assurance for LTS packages (including record retention and documentation).
(c) The licensing of packages for LTS may differ appreciably from those for short term storage.

4.5. Records management

It is important to maintain detailed and updated documentation for the facility and the stored waste (e.g., an inventory, waste characteristics, and facility and equipment drawings). These need to cover characterization of the inventory and waste forms. Permanent records need to include a detailed record of the waste and any supporting documentation to support any claims made about the waste. Two types of record need to be maintained:

(1) Those connected directly with the waste, such as the inventory;
(2) Operational records from supporting data such as monitoring results.

Management of records of package content at the time of packaging, and maintenance of these records, would probably be similar for both short term

storage and LTS, with the exception of preservation of the form of those records, to ensure that the information can be read and understood by future generations. They should be located in different safe places and on different reliable information supports, preferably drafted in a widely spoken language to be effective for the intended duration of the storage. This may lead to the use of paper as opposed to computer records, which may not be interpretable if the appropriate software is not available.

Maintaining historical records of events/incidents occurring to packages and the facility is required for future decommissioning or disposal, and would probably be the same for both short term storage and LTS.

Once a waste package has been disposed of, updating of records for it is no longer required.

4.6. Inspection and monitoring

The design of the package and the facility should allow monitoring and/or inspection of the integrity of packages and their contents, when considered for LTS. The design of a facility to allow external monitoring and inspection should include radiation monitoring, inspection for leakage, inspection of inaccessible areas and inspection for corrosion (age related effects). Provision for routine inspections of facilities and fabricators would also need to be considered, as well as retention of inspection records for the long term.

4.7. Ageing management programme

Having a programme in place to manage the effects of ageing on packages and facilities, and having corrective action programmes in place to deal with age related degradation, would be a consideration for LTS. For example, it might include a programme for periodic sampling of spent fuel.

4.8. Funding requirements and financial guarantees

Assuming that a government is the owner and is responsible for the operation of an LTS facility, it would carry the financial responsibility for the waste stored, and the responsibility for its safety. Even if a government accepts this responsibility, this does not preclude the use of an adequate financing system to place the financial burden on those who benefited from the nuclear power.

The net present value (NPV) method can be used to evaluate the amount of money that should be saved in the present to enable the government to fund future activities. Great attention should, however, be paid to choosing a

discount rate that makes sense in this context. A 30 year risk-free interest rate can be used to give an estimate of the return of a risk-free bond after 30 years. In parallel to interest rate risk, the price evolution of commodities such as steel, cement and staff costs present another form of financial risk. The discount rate should be the difference between the interest rate and this escalation rate. It can be small or even negative, especially if conservative assumptions must be made to take into account large uncertainties in the long term economics.

A negative discount rate could be taken as an indication that it is better to act earlier if disposal is to be done within a fixed budget, which in turn favours implementation of disposal in the near term rather than the distant future.

4.9. Safety assessment

For LTS, safety and environmental assessments would need to be carried out for initial licensing, and then at some periodic interval they would need to be updated on the basis of existing regulatory requirements. In disposal, these assessments would be made at the time of approval, and then only again when the disposal facility is closed; there would not be a requirement for periodic submittals.

For LTS, the content of the safety assessment would vary depending upon the period of storage (see design criteria). Normal operating conditions and hypothetical accident conditions would be different for short term storage and LTS. For example, LTS assessments might have to assign weight to the consequences of loss of institutional control. For LTS, there is a need for an ongoing programme of safety case review and renewal of the safety case after a fixed time. A revised safety case would be required to justify an extension of store life. Revisions to the safety case would also need to take account of new or revised regulations, which in turn may lead to upgrades of safety systems.

The safety consequences of a temporary period of abandonment of an LTS facility should be assessed. The duration of this period of abandonment within which consequences remain acceptable is a good indicator of the robustness of a facility.

5. DESIGN SAFETY

This section is focused on design safety for LTS, drawing attention to long term safety aspects of package and facility design.

5.1. Waste stability

Waste stability has to be ensured for the anticipated storage period, for example, by a correct choice of the package material and gas inerting. Packaging must be replaceable in case it becomes defective. Waste such as liquids, organic substances and materials containing products that may produce gases by radiolysis or fermentation should not be candidates for LTS. A programme of waste surveillance (monitoring), as indicated below, should be implemented to confirm the stability of a waste over time. It may be necessary to take periodic samples to monitor the long term behaviour of fuels. Technical means (hot cells), analysis methods and the necessary expertise have to be maintained for the duration of LTS. The required know-how and the means to implement corrective measures must also be maintained.

5.2. Degree of passiveness

Systems should be designed, as far as practicable, to be passively safe. Long term storage systems should be designed to operate without active systems, i.e. without relying on the power supply (fans, pumps, valves and vent actuators) or on the actions of operators. This does not mean that the systems must operate passively all the time. Loading or unloading storage systems, for instance, requires all possible means to be safely achieved. Power should also be kept available to enable monitoring and surveillance. Active ventilation can possibly be implemented as well to improve the reaction to accidental or exceptional meteorological conditions.

5.3. Siting

The timescales for LTS are such that siting decisions should give increasing attention to external events that can occur on longer timescales, such as groundwater level changes, earthquakes and erosion. A centralized storage with a clear segregation of all types of waste would probably minimize economic burdens, despite the initial transportation costs. The location of an LTS facility should preferably be near a suitable site for disposal, to minimize additional transportation costs. Management of environmental conditions is also a consideration in order to reduce the need for environmental control systems. Factors to be considered include: humidity, temperature and siting of stores in a ‘non-corrosive’ environment (e.g. in a rural area).

5.4. Transition from storage to disposal

To effectively ease this transition, reliable and rather detailed information on the disposal facility should be available. It might then be possible to design packages that are compatible for storage and disposal as well. With disposal far into the future, it is difficult to imagine how sufficient information could be known so far in advance of disposal to permit design of a dual purpose package. In the case of LTS, measures for stabilizing the waste are the only ones that are certain to contribute to a smooth transition.

5.5. Design criteria

The criteria used to design an LTS facility (e.g. reference earthquake levels, extreme meteorological data, gas generation and corrosion margins) must be more conservative than those for facilities designed for around 50 years, because the uncertainties are higher and the probability of experiencing exceptional events is also higher. Therefore, design standards should be based upon events over a sufficiently long period of time. This will create a problem for the indefinite storage option, as it will not be clear which criteria to design against.

Facilities should be designed to:

(a) Allow waste to be safely retrieved;
(b) Have buffer storage space to allow for relocation or rearrangement of waste packages;
(c) Incorporate condition monitoring/inspection of the package;
(d) Minimize dose to the workforce and the public;
(e) Incorporate instrumentation to monitor process/environmental conditions;
(f) Allow simple maintenance — key equipment should be accessible without removal of the wastes;
(g) Use well proven and well documented materials;
(h) Minimize locations that are inaccessible or that could create their own adverse microclimates, i.e. geometries where there is no circulation of air or cooling media, dead zones, etc.

5.6. Remediation plans

Remediation plans might be required as barriers may fail. This can be due to:

(a) Deterioration of ingress barriers, i.e. increased ingress of water into a store;
(b) Failure of containment to stop leaks — i.e. leaks from ponds or from sources;
(c) The waste form no longer being suitable for the intended length of storage.

This may lead to the need to:

(a) Relocate wastes to new facilities;
(b) Retrieve and repackage wastes, then emplace them back in storage or into a new facility;
(c) Dispose of wastes.

Different fault scenarios should be examined and exit strategies developed and approved. These may be in response to emergencies or anticipated deterioration of the barriers provided by the store or package (e.g. leaks in tanks or structures beyond their design lives).

5.7. Safety margins

If the facility is located adjacent to other nuclear facilities then it can utilize the safety systems/management organization of these other facilities. If the other facilities were to close, then there would have to be plans for the store to take over the systems it previously relied upon. There should be formalized plans for incorporating these functions within the store management systems.

5.8. End of store life

Plans, as well as equipment, need to be in place to allow retrieval. Wastes could be transferred to a conditioning facility, another storage facility or a disposal facility. The timing of these activities would need to be planned on the basis of the national waste management strategy, safety implications and regulations. The store can then be released for other uses once all of the waste has been removed and residual activity reduced to a level that does not pose a safety hazard.

5.9. Monitoring

A graded approach to monitoring would be required. The monitoring regime would be determined by the hazard presented by the waste and the

period over which the store is intended to operate — the greater the hazard, and the longer the storage period, the greater the degree of monitoring required. Generic considerations for monitoring include:

(a) Recording of trends to identify when action is required;
(b) Deciding where to place monitoring equipment;
(c) Means of collecting signals/management of false signals;
(d) Monitoring of random packages or monitoring of specific packages;
(e) Monitoring of random or specific locations.

6. SECURITY CONSIDERATIONS: ON-SURFACE OR BELOW GROUND STORAGE

The comparison between on-surface LTS and below surface LTS is a concern in terms of physical protection and safety. A below surface facility differs from an underground disposal facility in the fact that its depth aims at protecting against external events, and not at providing a geological barrier.

Issues that should be considered before making a choice between these alternatives are:

(a) Activity protects most spent fuel assemblies and vitrified waste canisters against falling into unauthorized possession. In this regard, below surface facilities do not provide a significant improvement in comparison with surface facilities. It should, however, be noted that the repelling effect of the dose emission decreases with time and loses its efficiency, especially for low burnup fuel.
(b) Some waste that is attractive for misuse is not self-protecting and could benefit from a less accessible below surface facility.
(c) Below surface storage facilities provide an excellent response to many hazards, such as earthquakes, intrusion, explosions, terrorist attacks, aircraft crashes and their associated fires.
(d) Below surface storage facilities are vulnerable to water saturation and flooding, possibly resulting in radionuclide migration. They are also most vulnerable to sabotage of the access shafts and ventilation ducts.
(e) Below surface storage facilities could more easily be unduly considered as a disposal facility by future generations.
(f) Below surface storage facilities could possibly better resist a loss of institutional control.

Depth should not compromise retrievability and maintenance of LTS facilities.

7. OVERALL MANAGEMENT CONSIDERATIONS

7.1. Maintenance of regulatory and operational ability

If a store is expected to last for hundreds of years then there is no guarantee that the current societal structures will remain. Scenarios to consider could include loss of the skills base, the operator no longer having the required funding to continue operations, and large scale natural disruptive events. Regulatory bodies should have contingency plans in place to manage such changes and events.

7.2. International obligations

The location and operation of plants or facilities should take into account the impact on neighbouring countries. This will need to include sharing of information about the facilities with those who could be affected, and complying with international treaties. The safety consequences of an accident or of a lack of maintenance in an LTS facility could obviously be worse than in a disposal facility. This makes continuity of control by the competent authority very important. In the case of a loss of control, an international organization should have sufficient knowledge to alert national and international authorities to the problem.

7.3. Blended facility concept

It is technically possible to use a disposal facility as an LTS facility, at least as long as it keeps the waste retrievable. The disposal phase will simply start once retrievability becomes impossible. This so-called 'blended' LTS concept may be quite expensive because it requires early investments and early technical choices. That nullifies most of the advantages of LTS mentioned in the introduction (Section 1).

7.4. Intergenerational equity

Undue burdens on future generations — economic burdens (see above) and the environmental burden of maintaining an LTS facility will be passed on. The burden of decision of when to convert to disposal will also be passed on —

LTS may be a fair decision only if ultimate disposal continues to be pursued as the final solution.

Knowledge transfer and technical capability would need to be maintained to operate and maintain an LTS facility, and there may be costs associated with maintaining technical competencies of specialists, technologists, etc. (these costs will depend upon whether there is a continuing nuclear power programme). The burden of maintaining technical capability should be recognized in financial planning.

8. SUMMARY

Long term storage is an interim solution for management of radioactive waste. Regardless of the reason for choosing LTS (e.g. societal acceptance, cost), the decision to implement disposal is posed. As discussed in Sections 4–7, LTS carries with it additional burdens — regulatory, financial and technical — to ensure safety; additional burdens, that is, compared with short term storage followed by disposal. Long term storage also carries with it long term risks for safety, possibly due to loss of institutional control and the increased vulnerability of materials in storage. Decision makers must weigh these risks against the potential benefits of future alternatives for management of radioactive waste.

Several countries have programmes for disposal of SNF that are nearing implementation; disposal solutions for other types of radioactive waste have already been implemented. Regardless of the treatment technologies for radioactive waste that may emerge, both types of disposal solutions available at present — near surface and geological — will be required. In other words, disposal and emerging treatment technologies are complementary. Long term storage without a commitment to disposal is not a competitive strategy for disposal — it is an incomplete strategy. Long term storage is complementary to both disposal and future treatment technologies only if the LTS strategy includes these end points.

As mentioned previously, this paper reports on work in progress. The IAEA welcomes any suggestions for improvements and suggestions for further development of the issues raised here.

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WASTE MANAGEMENT ASPECTS OF VARIOUS FUEL CYCLE OPTIONS*

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Abstract

In the United States of America, a repository at Yucca Mountain is being proposed for the disposal of spent nuclear fuel and high level waste. Under the Advanced Fuel Cycle Initiative of the United States Department of Energy (USDOE), studies have been conducted of spent nuclear fuel reprocessing and recycling strategies to evaluate the effects on nuclear waste characteristics and the impact on geological disposal. The first area of emphasis has been on increasing the utilization of space in the repository, where the loading of the repository is limited by the decay heat of the emplaced materials. It has been shown that it is essential to remove plutonium and americium for recycling, since these two elements are the dominant source of the decay heat limiting the loading density of spent nuclear fuel. Once these elements are removed, additional removal of caesium and strontium for separate storage allows further increases in loading density. Alternative strategies involving delay of placement and extended operation of the repository have also been evaluated, since some of the important isotopes have relatively short half-lives and can decay significantly during extended storage. Fuel cycle options, including limited actinide recycle in thermal reactors and continuous, or repeated, actinide recycle in thermal or fast reactors have been studied to quantify the practical increases in utilization of repository space. Limited recycle results in relatively little benefit, while continuous recycle provides the means to achieve large increases in repository loading density. However, with such increases comes the potential to increase the dose rates associated with releases from the repository. Preliminary results indicate that the peak dose rates with increased repository utilization are at most comparable to those for direct disposal of spent fuel.

* Work performed for the USDOE Advanced Fuel Cycle Initiative (AFCI) programme.

1. INTRODUCTION

The use of nuclear power results in the creation of highly radioactive spent fuel that is potentially hazardous for hundreds of thousands of years. To address this issue, one possibility is to develop a geological repository, such as the one being envisaged for Yucca Mountain in the United States of America (USA), which will safely isolate the spent fuel and other nuclear waste from humans and the environment for extended periods of time. However, the stringent isolation requirements cause the development and licensing of such a repository to be an expensive and lengthy process. Thus, the available space in a geological repository is likely to be a scarce and valuable resource, and provides the incentive to make the best use of repository space. Under the Advanced Fuel Cycle Initiative (AFCI) of the United States Department of Energy (USDOE), studies have been conducted of spent nuclear fuel reprocessing and recycling strategies to evaluate the effects on nuclear waste characteristics and to quantify the impact on geological disposal.

To demonstrate that the goal of successfully isolating the hazardous nuclear materials from humans and the environment for hundreds of thousands of years is likely to be achieved, it is essential to be able to reliably estimate the behaviour of the geological repository over such periods of time. As part of the repository design process, repository technical requirements are being developed that will increase the reliability of the predictions of long term behaviour, especially with regard to potential licensing issues; for example, that the peak dose rates associated with such releases are within the limits specified by regulations.

The long term behaviour of the emplaced spent fuel and nuclear waste is determined by the corrosion, degradation and radionuclide release characteristics of these materials. To ensure that acceptable long term behaviour will be realized, specifications for repository operation have been developed based on the tested corrosion, degradation and release characteristics of the relevant repository materials for ranges of environmental conditions, including temperature and water chemistry.

In the example of a repository at Yucca Mountain, for the waste package materials currently being considered, the specifications are in the form of a number of temperature limits that have been imposed. These limits in turn determine the maximum decay heat for each waste package at the time of placement, and the maximum average linear heat rate for the array of waste packages in a repository drift (tunnel) [1].

1.1. Relationship of dose rate and thermal limits

The dose rate is determined by the rate at which radionuclides can enter the environment and be conveyed to individuals. Controlling the dose rate is accomplished both by limiting the radionuclide inventory and by constructing a robust engineered system in a suitable geological location to limit the degradation of the emplaced materials and the transport of any released materials to the environment.

For a repository at Yucca Mountain, this approach led to the plans for a repository system with specifications on waste package materials and design, along with loading and operating conditions to ensure that the repository system is capable of satisfying the regulatory dose rate limits [2, 3]. As mentioned in the previous section, specifications of operating conditions are in the form of temperature limits for the repository environment. For example, the requirements for the current reference operating mode of a repository at Yucca Mountain, the high temperature operating mode (HTOM) of the 'cold' repository, include a temperature limit that specifies that the rock temperature midway between drifts must always remain below 96°C.

This specification ensures that any water flowing downwards through the mountain will be able to move through the repository at all times, preventing the retention of a large volume of water above the repository that could flood the repository as it cools. A temperature limit of 200°C is also imposed on the rock at all times to prevent alteration of its crystalline structure. These temperature limits provide greater certainty about the conditions in a repository at Yucca Mountain, and increase the reliability of the assessments of repository performance. Other temperature limits for a repository at Yucca Mountain apply to the emplaced materials and the waste packages, to limit the degradation rates that lead to releases of radioactive materials.

Temperature limits act as constraints on the design and operation of a geological repository. Meeting these limits can be accomplished by a variety of methods, including controlling the amount of decay heat generated by the waste in any given area of the repository and actively cooling the repository for an extended period of time. In the case of a repository at Yucca Mountain, the design requirements were met by setting the emplacement drifts 81 m apart, while specifying the allowable peak and average linear heat loads in the drift, and by having the repository drifts cooled by forced ventilation for a period of at least 50 years after completion of waste placement. Since a limit of 70 000 tHM (metric tons of heavy metal) is legally mandatory during the first phase of operation (prior to the opening of a second repository), this amount of spent nuclear fuel and waste will cover approximately 1150 acres, with the

required area ultimately being determined by the decay heat characteristics of the emplaced waste.

1.2. Approaches to improve utilization of space in a geological repository

Since reprocessing the spent fuel allows for partitioning and recovery of the hazardous elements, and the subsequent transmutation of these elements in nuclear reactors can substantially alter the decay heat characteristics of the nuclear waste emplaced in the repository, there is the possibility of improving the utilization of space in a geological repository by this approach. In this paper, the potential for partitioning and transmutation to increase utilization of geological repository space is quantified for several of the spent fuel reprocessing and recycling strategies being studied in the AFCI programme, including both limited and continuous, or repeated, recycle. The heat generation characteristics that control the loading of the repository are described, and the chemical elements identified that need to be removed from the spent fuel and recycled. The effect of recovery efficiency of the hazardous radionuclides on the potential benefit to utilization of repository space is quantified. In addition, the changes in the estimated peak dose rates for releases from the repository that result from the alteration of the radionuclide inventory are discussed, both for the waste from the initially planned 70 000 tHM and for the increased utilization permitted by the altered radionuclide inventory in process waste.

1.3. Decay heat characteristics of commercial spent nuclear fuel

The decay heat generated by commercial spent nuclear fuel (CSNF) is determined by the mass and isotopic composition of the discharged fuel. In turn, the mass and isotopic composition depend on the discharge burnup of the fuel. For the example of a repository at Yucca Mountain, many types of CSNF need to be considered in planning the disposal strategy, including both PWR and BWR fuels. For PWR fuels, which constitute about 55% of the total tHM destined for a repository at Yucca Mountain, the average fuel is irradiated to about 41.2 GW·d/tIHM (gigawatt-days per tonne of initial heavy metal) and has an average age of about 23.1 years since discharge from the reactor. In this paper, a reference case using an average burnup of 50 GW·d/tIHM is assumed as an estimate of current and planned PWR fuel irradiation, with placement 25 years after discharge.

Figure 1 shows the decay heat generated by spent PWR fuel at 50 GW·d/tIHM discharge burnup. It is important to note that the decay heat is mainly generated by the decay of fission products for the first 60 years, with the contribution dominated by barium and yttrium as decay products of caesium and

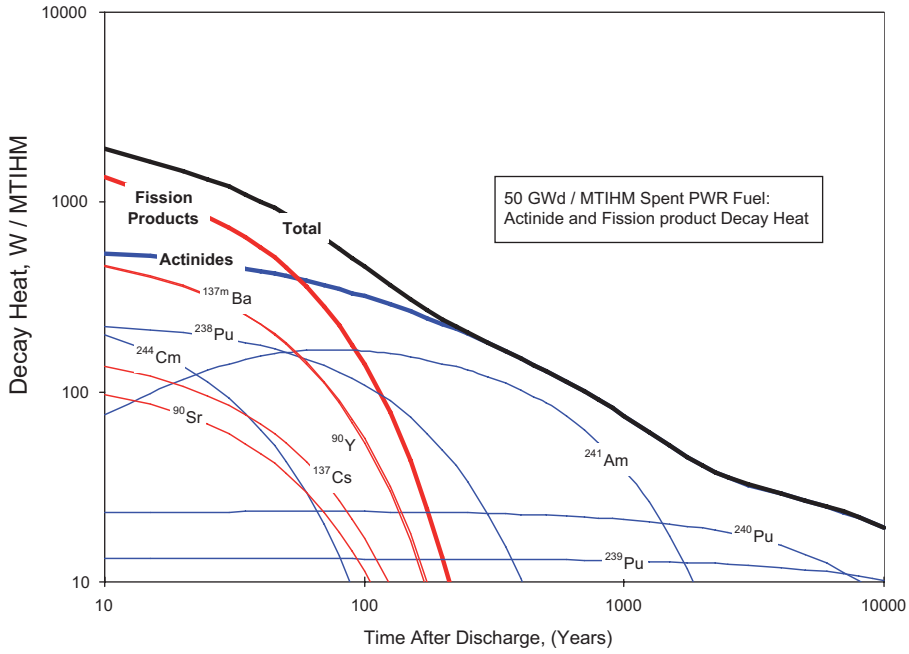

FIG. 1. Dominant decay heat contributors in spent PWR fuel irradiated to 50 GW·d/tIHM.

strontium. After 60 years, the decay heat is mostly from actinide elements, with the important actinide elements being plutonium and americium, and beyond about 200 years, the decay heat is caused entirely by isotopes of the actinide elements plutonium and americium, up to at least 10 000 years. The slow decrease of the decay heat with time is due to the relatively long half-life of the isotopes ^{241}Am, ^{238}Pu, ^{239}Pu and ^{240}Pu, as plotted in Fig. 1. In the analyses that follow, these decay heat characteristics are used with a detailed thermal model of a repository at Yucca Mountain, to demonstrate the transient thermal performance of the repository as different chemical elements are removed and the resulting waste placed in the repository.

1.4. Model of a repository at Yucca Mountain

Yucca Mountain is a mountainous ridge built up from layers of volcanic rocks (tuffs). The current natural environment is very dry. At the repository level, 324 m below the surface and 344 m above the water table on average, the rock layer is fractured, typically about 14% porous, and about 80% saturated with water. The repository within the mountain is an array of parallel storage

drifts, each about 1 km long and 5.5 m in diameter, located in an approximately horizontal plane. As stated above, the separation between parallel drifts is 81 m for the HTOM of the repository. Current plans are for the emplaced waste to be cooled by forced ventilation for at least 50 years after final waste placement. Airflow enters the drifts from both ends and exits at the centre [1].

The thermal model of a repository at Yucca Mountain developed for these analyses simulates the geometry and operation of the repository both prior to and after closure of the repository. Major heat transfer modes of conduction, forced ventilation, free convection and thermal radiation effects are included. Vertically, the model extends from the surface of Yucca Mountain to the water table, explicitly representing all of the rock layers with the corresponding thermophysical properties. Laterally, the model is defined by adiabatic boundaries that extend to one half the drift separation on either side of the drift, so that the computed temperatures would be representative of values at locations near the repository centre, where temperatures are expected to be highest.

Within the various rock layers present in the mountain, conduction is the principal heat transfer mode modelled. Details include specific saturation-dependent thermal properties for each layer. Thermal convection from surface water infiltration into the mountain through the porous rock is also included. For simplicity in the model, boiling within the rock near heated drifts is modelled by a reversible phase change of water contained in rock fractures or pores, rather than using more detailed hydrological calculations. While this simplification has little impact on the rise to the peak temperature in the rock adjacent to the drift, it does tend to slow the subsequent fall in temperature and extend the period of time when the rock is above the boiling point of water.

In support of the present approach, a previously reported comprehensive study of heat driven hydrothermal flow in Yucca Mountain [4] concluded that the high temperature boiling conditions anticipated in the Yucca Mountain repository may be severe enough to strongly perturb hydrological flow, but not severe enough for those hydrological perturbations to significantly alter subsequent thermal development. Reported results from the more complex thermal hydrological models were compared with those from the present model for reference Yucca Mountain conditions and for cases with extended ventilation.

2. THERMAL PERFORMANCE OF A REPOSITORY AT YUCCA MOUNTAIN

As discussed above, the design and loading of a repository at Yucca Mountain is governed by the transient thermal behaviour and the appropriate temperature limits needed to ensure adequate performance of the repository. The following discussion begins with the current reference design, and then proceeds with identification of the chemical element(s) whose decay heat causes the temperature limits to be met. Subsequent sections illustrate the repository performance as groups of chemical elements are removed, and quantify the potential increase in drift loading, i.e. reduction in repository size, that would be possible with such a strategy.

2.1. Reference case for direct disposal of CSNF in a repository at Yucca Mountain

The reference repository conditions for these analyses were based on the current HTOM operating plans for a repository at Yucca Mountain, as described in detail in Ref. [1]. Drifts are ventilated at a rate of 15 m^3/s for a period of 75 years. As stated above, CSNF having an average burnup of 50 GW·d/tIHM is placed in the repository 25 years after reactor discharge. For this case, the transient temperature behaviour of the repository was calculated, with the results shown in Fig. 2. As Fig. 2 illustrates, in order to remain below all of the imposed temperature limits, the maximum allowable linear loading of the repository drift was 1.10 tIHM/m, with essentially no allowance for a margin to the controlling temperature limit of 96ºC midway between adjacent drifts. The decay heat is shown as the corresponding linear heat rate in the drift, and is equivalent to the data in Fig. 1.

Figure 2 shows the waste package surface temperature initially below the boiling point of water while the drift is ventilated, but once forced ventilation is stopped (assumed to occur at 75 years after waste placement), the surface temperature increases rapidly to a peak of about 140°C. The temperature then slowly drops over time, falling below the boiling point of water after about 3000 years. The drift wall temperature exhibits the same trends. The temperature midway between adjacent drifts responds much more slowly, since the only mechanism for heating the rock in this location is conduction from hotter areas around the drifts. As Fig. 2 shows, this temperature peaks just below 96°C, one of the temperature criteria for the HTOM in a repository at Yucca Mountain.

It should be noted that none of the temperatures in and around the drift are close to any of the relevant temperature limits. It is concluded that the

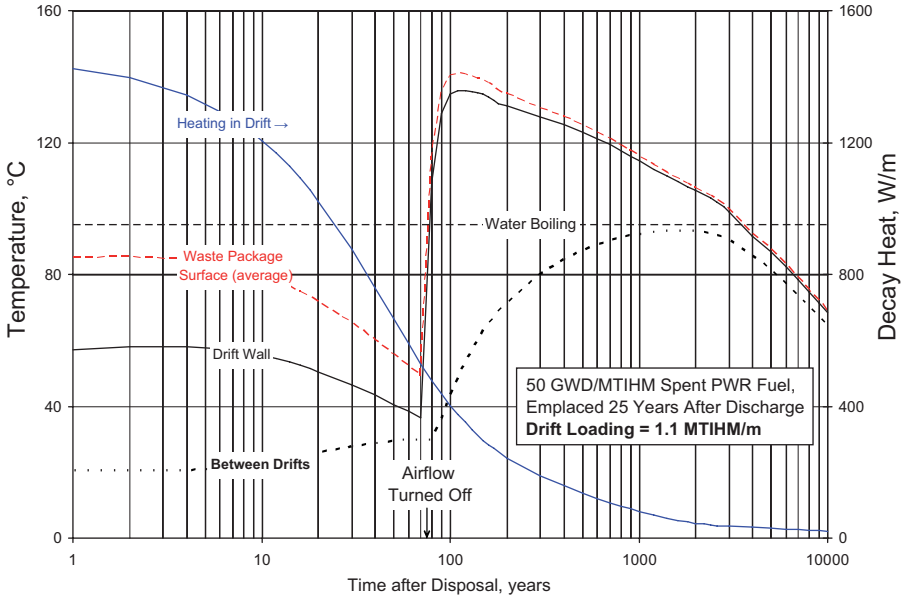

FIG. 2. Transient thermal response of a repository at Yucca Mountain for reference loading conditions of spent PWR fuel and 75 years of forced ventilation at 15 m^3/s.

temperature limit midway between adjacent drifts is the controlling limit for the reference case, and that the peak temperature at this location occurs at about 1500–2000 years after waste placement. Also, owing to the extended time frame for heating this region of the repository, the temperature peak must be the result of the integrated decay heat over the time since the placement of the waste, rather than the value of decay heat at any particular time. Since Fig. 2 shows that substantial heating of the interior of the mountain does not commence until after the forced ventilation has stopped, the responsible chemical element(s) would logically be those that have the highest integrated decay heat from the time when ventilation ceases until about 1500–2000 years.

Considering the decay heat data shown in Fig. 1, the decay heat is dominated by actinide heating for all times after about 60 years, indicating that one or more of the actinide elements is responsible for the decay heat leading to the temperature peak midway between the drifts. Examining the contributions from each of the actinide elements during the time period from 75 up to 2000 years, it is observed that the dominant contribution is from the isotope ^{241}Am. Since the isotopic composition of CSNF at the time of discharge shows very little ^{241}Am, the ^{241}Am content must be caused mainly by the radioactive decay of ^{241}Pu, with a small contribution from the decay of ^{245}Cm.

The ^{241}Am content peaks at about 80–100 years after discharge from the reactor, due to the 14.4 year half-life of ^{241}Pu, but the 433 year half-life of ^{241}Am and its energetic decay are the reasons for this isotope's dominance of decay heat generation. During the time from 75 up to 2000 years, the other contributors are the plutonium isotopes, ^{238}Pu, ^{239}Pu and ^{240}Pu. Fission product heating has essentially ceased by 300 years. The conclusion is that it would appear that the current plans for the design and loading of a repository at Yucca Mountain are constrained by the integrated decay heat mainly from ^{241}Am, which arises from decay of ^{241}Pu, with the remainder of the decay heat coming from other plutonium isotopes.

2.2. Repository behaviour when plutonium and americium are removed from CSNF

With the identification of the plutonium and americium inventory as the cause for one of the temperature limits being reached in a repository at Yucca Mountain, it is possible to examine the effect on the transient repository behaviour of removal of these two elements. While the plutonium and americium must be treated in some manner, the discussion of the impact of the disposition of the removed plutonium and americium is deferred until later in this paper.

As an example, the case is presented where the CSNF is processed with a removal efficiency for plutonium and americium of 99.9%. Figure 3 shows the transient temperatures in the repository for this case, and it is apparent that the overall behaviour is quite different from that observed for the reference case. The removal of plutonium and americium has made it possible to increase the linear loading in the drift from the reference value of 1.1 tIHM of spent PWR fuel per metre up to the process waste from 5.9 tIHM of spent PWR fuel being stored per metre. (It is likely that in the processing of spent PWR fuel, the uranium would be removed. The drift loading is for the waste associated with metric tons of initial heavy metal of the original unprocessed fuel.) The loading increase can be interpreted as an increase in the area loading of the repository by a factor of about 5.4, which would allow a reduction in repository area by the same amount for a given capacity.

In the example shown in Fig. 3, it was assumed that the recovery efficiency of the plutonium and americium was 99.9%. In studying cases with lower recovery efficiencies, it has been determined that the allowable increase in drift loading is a factor of 5.3 at 99%, and 4.3 at 90%. The relative insensitivity of the increase in drift loading to the recovery efficiency of plutonium and americium is caused by a change in the nature of the transient thermal

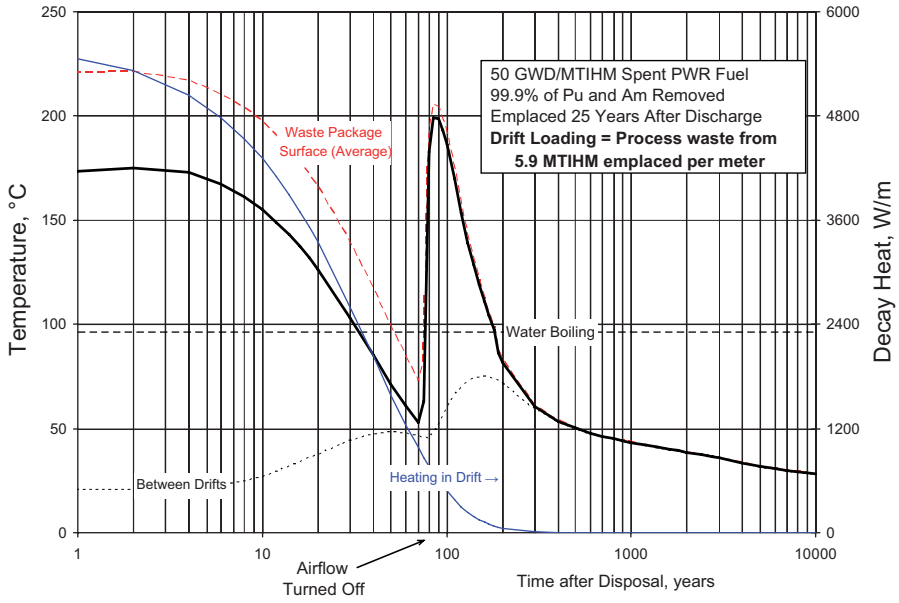

FIG. 3. Transient thermal response of a repository at Yucca Mountain with removal of plutonium and americium from spent PWR fuel with increased drift loading.

performance of the repository, and recovery efficiencies above 90% or so may not be needed unless other separations are conducted.

Referring to Fig. 3, it is seen that the drift loading is no longer limited by a peak temperature midway between the drifts, since this temperature peaks at only 75°C about 150 years after waste placement. Instead, the temperature limit constraining drift loading is the 200°C allowed for the rock, and occurs at the drift wall. The peak temperature occurs at about 90 years, shortly after the end of the forced ventilation period at 75 years, when the repository is closed. This change in the nature of the transient thermal behaviour of the repository indicates that the source of the decay heat responsible for reaching the temperature limit is not another actinide element, but is caused by shorter lived fission products, as shown in Fig. 1. Since the peak occurs rapidly after closure, this implies that it is caused by the integrated decay heat from 75 to 90 years. Examining the remaining contributors to the decay heat during this time, it is clear that, in the absence of plutonium and americium, the decay heat is due almost entirely to caesium and strontium, and to their decay products, barium and yttrium. These fission products provide almost the entire fission product portion of the total decay heat for times greater than ten years after discharge

of spent PWR fuel from the reactor, which is not placed into the repository until 25 years after discharge.

2.3. Repository behaviour with subsequent removal of caesium and strontium from CSNF

In considering the fission products caesium and strontium, along with their decay products barium and yttrium, it is noted that the half-life of barium at 2.5 minutes, and the half-lives of yttrium at either 3.2 hours or 2.7 days, are very short compared with the 30 year half-lives of both caesium and strontium. This large difference in half-lives allows a separation strategy where only caesium and strontium are removed, since any barium and yttrium will rapidly decay away. As a result, the next step was to quantify the impact of removing the caesium and strontium, analysed for various separation efficiencies as with the removal of plutonium and americium. Sample analysis results are shown in Fig. 4, where the plutonium and americium have been removed with 99.9% efficiency, and the caesium and strontium were subsequently removed, also with 99.9% efficiency. The drift loading has been further increased to being

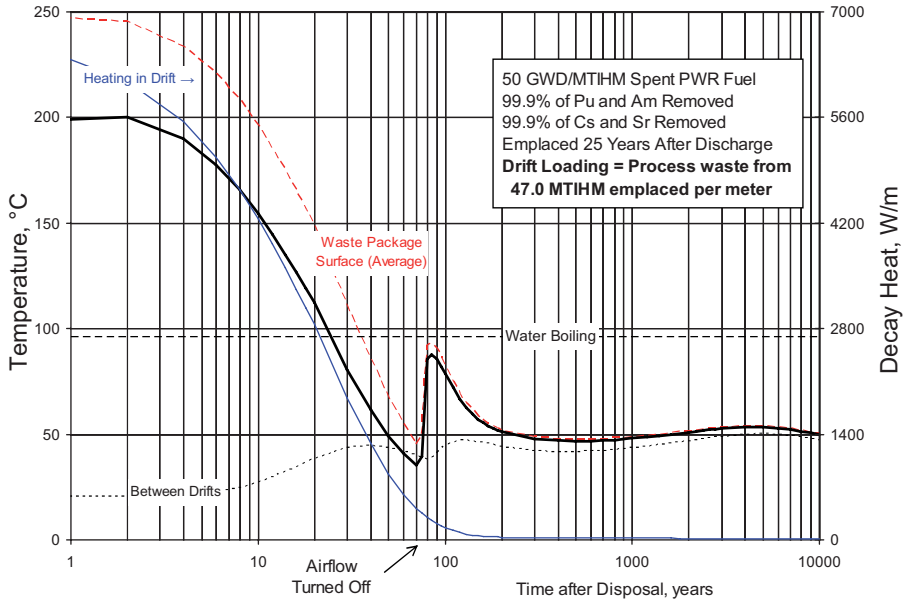

FIG. 4. Transient thermal response of a repository at Yucca Mountain with removal of plutonium and americium, followed by removal of caesium and strontium from spent PWR fuel with increased drift loading.

able to load the process waste from 47.0 tIHM of spent PWR fuel per metre of drift, a factor of 42.7 higher than for the reference case. As the figure demonstrates, the transient behaviour of the repository has changed again, where in this case the drift loading is limited by the peak temperature of the drift that occurs immediately after placement of the waste, even with the forced ventilation of the repository. The decay heat causing the temperature peak is the result of some very short lived actinide and fission product elements, but mainly of curium.

With lower recovery efficiencies of the caesium and strontium, for the same case of 99.9% removal efficiency of plutonium and americium, the drift loading can only be increased to loading the process waste from 44.0 tIHM of spent PWR fuel per metre for 99% removal of caesium and strontium, and to the process waste from 29.5 tIHM of spent PWR fuel per metre for 90%. The sensitivity of the results to recovery efficiency shows that the growing importance of the very short lived isotopes occurs with 99% removal of caesium and strontium; higher removal efficiencies of these fission products are not necessary unless other separations are carried out.

Owing to the short half-lives of the caesium, strontium, barium and yttrium, it is possible to consider the use of interim storage as the solution for these elements. After 200–300 years, the activity and decay heat have been reduced to the point where the remaining materials can be safely disposed either in the repository or at another suitable location without having an impact on the repository performance. It is useful to note that partial removal of short lived fission products can be accomplished without processing by simply delaying disposal, although to achieve a significant benefit would require delay times in excess of 150 years. As shown in Fig. 1, every 30 year period of delay reduces the heat load from caesium and strontium by about 50%. By contrast, the long half-lives of the key actinide elements shown in Fig. 1 also indicate that delay of direct disposal alone is not a practical strategy for removing actinide heat sources.

2.4. Potential increase in utilization of repository space

As a result of processing CSNF to remove the chemical elements mostly responsible for the decay heat that controls the utilization of space in the repository, it is possible to greatly increase the loading of a repository at Yucca Mountain. The results are summarized in Fig. 5 for various separation efficiencies of the higher actinide elements (including curium), caesium and strontium.

It is important at this point to consider the effects of realistic recycling scenarios for separated plutonium and americium, and storage strategies for

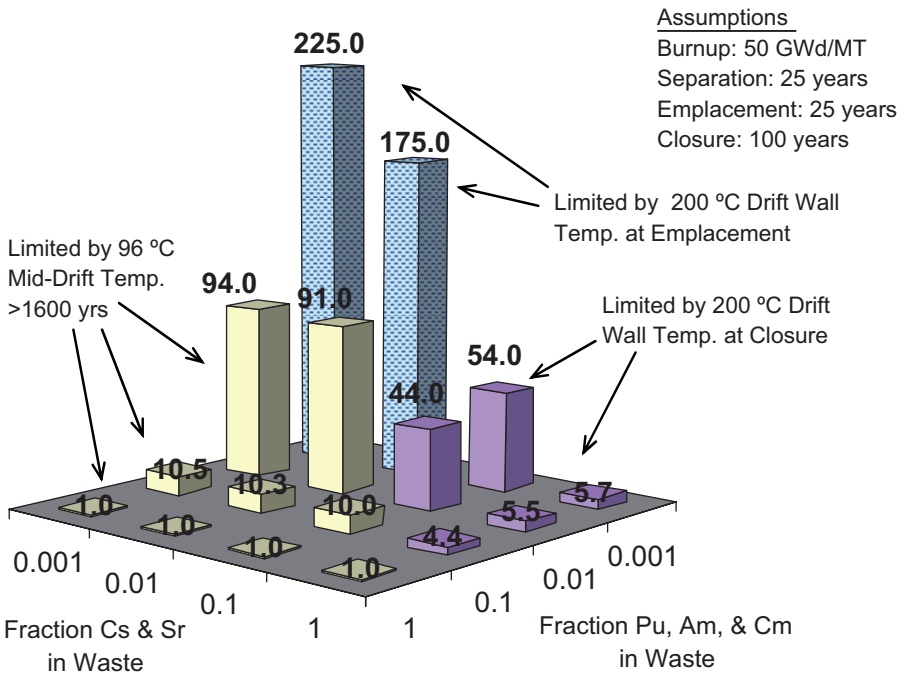

FIG. 5. Potential increase in repository drift loading as a function of separation efficiency for plutonium, americium, curium, caesium and strontium.

caesium and strontium. Such scenarios have the potential to introduce more plutonium and americium into the waste stream, and could make further reductions in short term decay heat irrelevant to repository loading constraints.

3. RECYCLING AND TRANSMUTATION OF RECOVERED ACTINIDE ELEMENTS

The disposition of the actinide elements recovered from processing CSNF must be considered when examining the potential impact of processing CSNF to achieve benefits for a waste management strategy. Some of the elements, such as plutonium, americium and neptunium, must be recycled in nuclear reactors and transmuted or fissioned into other, less hazardous, elements. If curium is recovered, it is possible to consider extended storage, as the relatively short half-life of the curium isotopes would allow decay into plutonium and americium isotopes, since thermal spectrum irradiation of curium leads to the formation of higher actinide elements that are very radioactive, such as

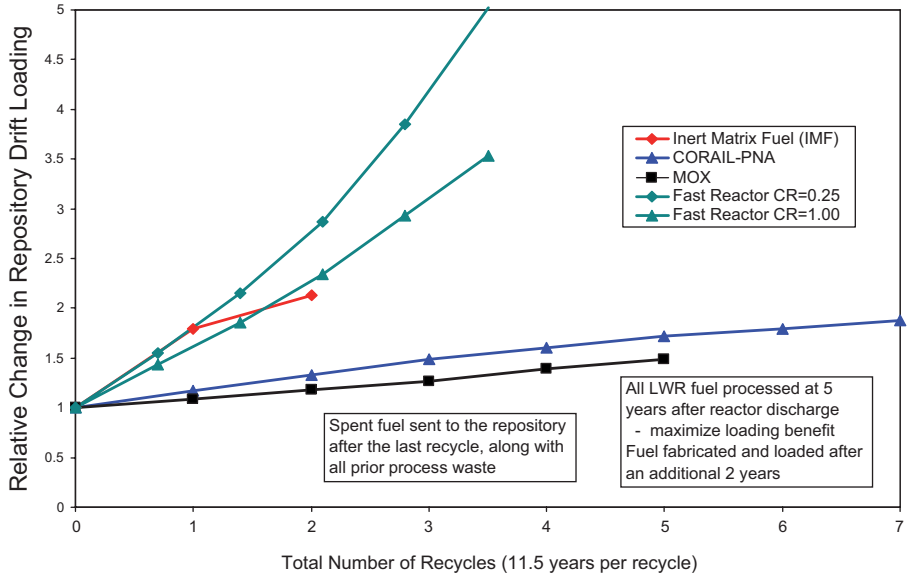

FIG. 6. Relative change in repository drift loading as a function of the total number of recycles for MOX, CORAIL-PNA, IMF and fast reactor recycling of plutonium, americium and neptunium.

californium. Alternatively, all of the actinides can be transmuted or fissioned in fast neutron reactors. Analyses have been performed for both thermal and fast reactor recycling strategies, with recycling of plutonium, americium and neptunium.

For thermal reactors, the following approaches have been investigated for both single recycles and multiple recycles, with the result for each approach summarized in Fig. 6:

(a) Mixed oxide fuel (MOX) — In this approach, the separated plutonium, americium and neptunium are used to fabricate new fuel assemblies, in a fuel matrix of recovered uranium (enriched slightly above natural uranium), all elements being present as oxides. To obtain sufficient fissile material for the first generation of MOX, provided mostly by plutonium, it is necessary to use materials recovered from a number of spent PWR fuel assemblies to fabricate one MOX assembly. Subsequent recycling with MOX uses the recovered plutonium, americium and neptunium from the current MOX generation to fabricate the next generation of MOX assemblies. The use of MOX fuel allows a steady increase in

repository drift loading with each recycle of plutonium, americium and neptunium, reaching a factor of 1.5 after five recycles of MOX fuel.

(b) CORAIL-PNA — This concept uses heterogeneous assemblies, where some of the fuel pins (about one third) are fabricated from the separated plutonium, americium and neptunium, in a uranium matrix, and the remaining fuel pins (about two thirds) are fabricated from new enriched uranium, with all the elements present as oxides. In the CORAIL-PNA case, the recovered plutonium, americium and neptunium from one spent PWR assembly are used to make the one third of the fuel pins for one assembly in the first CORAIL-PNA generation that contains these materials. After irradiation, the entire spent CORAIL-PNA assembly is processed to recover the plutonium, americium and neptunium for one third of the fuel pins in a single assembly of the next CORAIL-PNA generation. The remaining two thirds of the fuel pins are again fabricated from new enriched uranium, with the enrichment increasing with each CORAIL-PNA generation. The use of CORAIL-PNA also allows a steady increase in drift loading with each recycle at a faster rate than that for MOX, reaching a factor of 2.0 after seven recycles of CORAIL-PNA fuel due to the favourable impact of using enriched uranium to provide fissile content rather than relying entirely on the recovered plutonium, americium and neptunium.

(c) Inert matrix fuel (IMF) — This approach is similar to MOX, but the fuel matrix is an inert material, zirconia, instead of uranium oxide. The recovered plutonium, americium and neptunium from several spent PWR assemblies are used to make a single assembly in the first generation of IMF so that sufficient fissile material is provided. Subsequent generations of IMF also use several IMF assemblies of the previous generation to obtain sufficient fissile material. The use of inert matrix fuel provides a factor of 1.8 for the increase in drift loading after the first recycling, and a factor of 2.1 after the second recycling. Further recycle of IMF is hindered by the rapid depletion of fissile material with each subsequent irradiation (especially for ^{239}Pu), making it impossible to perform a third recycle to the same integrated energy for the assembly.

Approaches using fast reactor recycling have also been evaluated. The results, as a function of the total number of recycles, are shown in Fig. 6. As can be seen from this figure, use of a fast reactor allows a continuing increase in benefit with the number of recycles that is greater than that obtainable with thermal reactor recycling. In addition, there is no limitation on the number of recycles that can be used, so that, in principle, it would be possible to achieve the loading benefits shown in Fig. 5. While this may also be possible in a

thermal spectrum with the CORAIL-PNA approach, the increasing use of higher uranium enrichment makes this option problematic for future use when resources of ^{235}U are expected to become scarcer. It should be emphasized that for each of these cases, the great benefit shown in Fig. 5 is always obtained for the process waste, depending on the separation efficiency. However, once the processing has been stopped, it is the disposal of the remaining spent fuel assemblies that greatly reduces the overall benefit to the repository. As a consequence, to achieve much greater utilization, in practice, of repository space, direct disposal of any spent fuel must be avoided.

4. DOSE RATE IMPLICATIONS FOR INCREASED UTILIZATION OF REPOSITORY SPACE

The preceding results have shown that it is possible to greatly increase the utilization of repository space by processing the CSNF and removing those elements responsible for most of the decay heat. The resulting consolidation of the process waste causes the radionuclide inventory in the repository to be substantially different from that for direct disposal of CSNF.

For example, with 99% removal of plutonium, americium, neptunium and curium, along with 99% removal of caesium and strontium, it is possible to densify the remaining materials by a factor of 91 in comparison with direct disposal of CSNF. While this brings the inventory of the actinides, caesium and strontium almost back to the values for direct disposal, the inventory of elements that were not removed has increased by almost a factor of 100. This can be especially important for the potentially hazardous radionuclides that were present in small quantities in CSNF. Examples are technetium and iodine, which are not necessarily important to the peak dose rates expected for releases from a Yucca Mountain repository, but could be if their inventories were greatly increased relative to those of the actinide elements.

It is very difficult to reliably make estimates of dose rates from a geological repository over a period of hundreds of thousands, or even millions, of years, owing to the high level of uncertainty associated with the relevant physical processes and the anticipated environmental conditions. For this reason, a statistical approach has been adopted by the developers of the Yucca Mountain repository, called the Total System Performance Assessment [1].

A simplified version of this model for the site recommendation phase of the Yucca Mountain project has been used to make preliminary assessments of the potential impact on dose rate associated with the processing of CSNF and the increased utilization of the repository space. For reference, all results have been normalized to the peak dose rate estimated for the direct disposal of

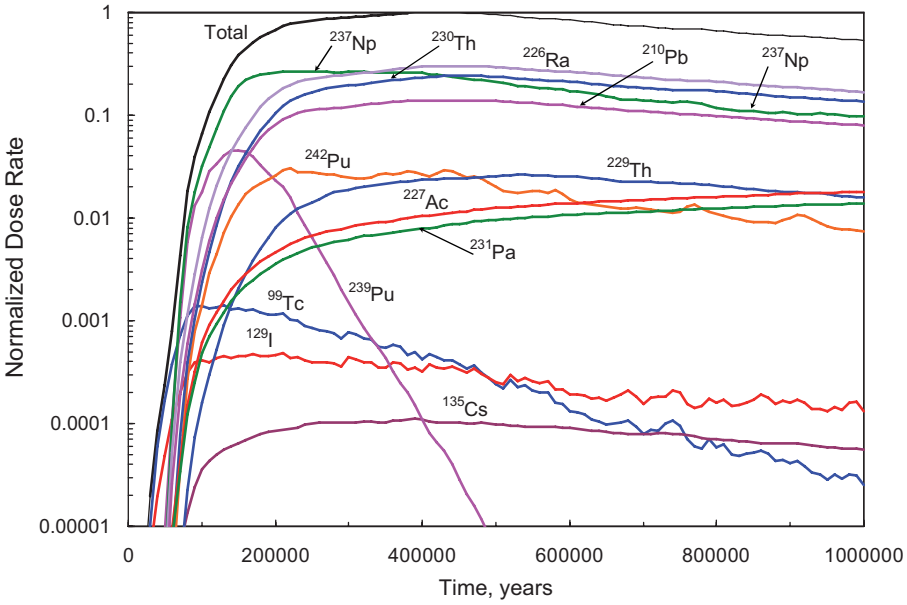

FIG. 7. Estimate of the dose rate for releases from a Yucca Mountain repository using a simplified performance assessment model for the site recommendation.

CSNF, as shown in Fig. 7. The dominant contribution from actinide elements and their radioactive decay products is clearly shown.

Since the processing of CSNF will result in the creation of waste forms for the process waste, it was assumed in this study that the waste form will be a vitrified waste form using borosilicate glass, similar to what has been used for encapsulating the wastes from defence related operations in the USA. Since the waste form can alter the release of radionuclides to the environment, a calculation was performed where the chemical element inventory of CSNF was placed in glass instead of the spent fuel cladding. The results of this calculation are shown in Fig. 8, where it can be seen that the peak dose rate is only slightly higher than that for CSNF, although the order of importance of some isotopes has changed.

Further calculations were performed where the CSNF was processed, and the actinide elements were recovered with an efficiency of 99.9%. The caesium and strontium were also removed with an efficiency of 99.9%, so that the potential increase in utilization of repository space is about a factor of 225. However, to keep the estimated peak dose rate the same as that for direct disposal of CSNF, an increase of a factor of 100 was used, with the results shown in Fig. 9.

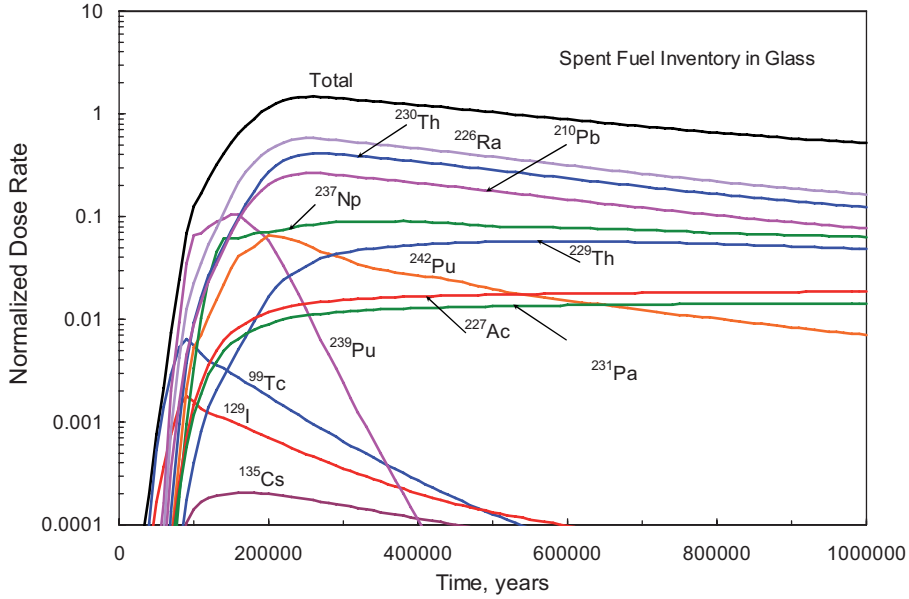

FIG. 8. Estimate of the dose rate for releases from the Yucca Mountain repository using a simplified performance assessment model for the site recommendation with the inventory of CSNF placed in a glass waste form.

As can be seen in Fig. 9, the dose rate is dominated by technetium and iodine at the time of peak dose rate. The remaining actinide elements are still important, however, controlling the dose rate after about 350 000 years. Removal and treatment of the technetium and iodine would be necessary to reduce the dose rate for the process waste in this case, although it should be emphasized that the estimates of dose rates for releases from the Yucca Mountain repository are subject to changes as the science and modelling of the repository continue to evolve.

5. SUMMARY AND CONCLUSIONS

The study of the response of a proposed repository at Yucca Mountain to the planned disposal of CSNF has identified opportunities for increasing the drift loading of the repository, while adhering to temperature limits that permit greater certainty about repository conditions and increased reliability of repository performance assessments. The analyses on the thermal response of the repository have resulted in the following separation criteria:

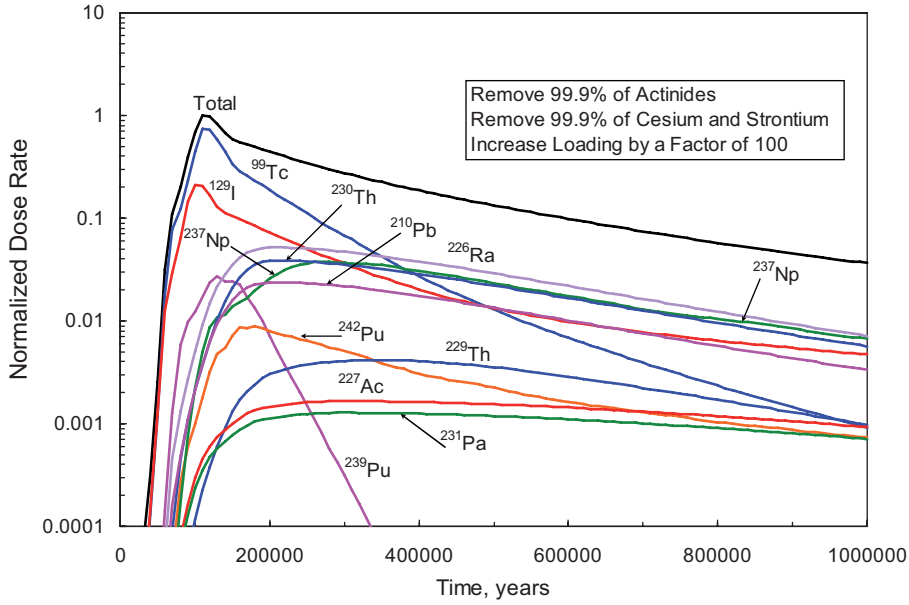

FIG. 9. Estimate of the dose rate for releases from a Yucca Mountain repository using a simplified performance assessment model for the site recommendation with processing of CSNF to remove 99.9% of the actinides, caesium and strontium, and the resulting waste densified by a factor of 100 and placed in a glass waste form.

(a) The dominant contributors to the thermal load of emplaced spent PWR fuel in a repository at Yucca Mountain that lead to reaching one or more of the temperature limits are plutonium and americium. Removal of these elements, and recycling them to reduce the hazard, is essential to increasing the drift loading of the repository. The benefit ranges from a factor of 4.3 to one of 5.4 on increasing the drift loading (or decreasing the repository size for a given capacity), depending on the separation efficiency.

(b) After the plutonium and americium have been removed, the next elements that need to be considered are caesium and strontium. Removing these elements, and sequestering them in a separate area of the repository or another facility, would allow a further substantial increase in the drift loading of the repository, up to a factor of 42.7 greater than the direct disposal case for 99.9% removal of plutonium, americium, caesium and strontium. Removal of caesium and strontium without prior removal of plutonium and americium is ineffective. Removal of plutonium, americium, caesium and strontium with 99% efficiency

provides almost the same benefit as using 99.9% separation efficiency, as other elements begin to dominate the process waste decay heat.

(c) The next most important element is curium. Assuming that curium can be removed along with the other actinide elements, with a similar efficiency, even greater increases in utilization of repository space can be realized, as shown in Fig. 5. In this case, the use of 99.9% separation efficiencies for all of the removed chemical elements results in a potential loading increase of 225. In addition, there is now a substantial difference between separation efficiencies, 99 and 99.9%, indicating that the effects of the other elements in the waste stream are still relatively unimportant in generating decay heat. It is also useful to note that with the addition of curium removal, the factor for increasing loading rises from 39 to 91 at 99% efficiency, highlighting the fact that there are several possibilities for achieving a given increase in repository loading.

(d) In considering a realistic fast reactor recycling scenario for plutonium and americium, it is observed that the drift loading can be limited by the losses of these two elements from the processing of the fast reactor fuel. In the case where 99.9% of the plutonium, americium, caesium and strontium are removed from spent PWR fuel, the 1% loss assumed in the processing of fast reactor fuel reduces the potential increase in drift loading from a factor of 42.7 to 20.5. This emphasizes the need to reduce losses of plutonium and americium at every processing step to maximize the potential benefit to the repository.

(e) When the process waste is densified to take advantage of the greater possible loading, estimates of the peak dose rate show that the resulting dose rate is similar to that for direct disposal of CSNF. This allows far greater utilization of repository space, while continuing to meet the applicable environmental regulations.

To summarize, it has been shown that removal of plutonium and americium alone from spent PWR fuel has the potential for either increasing the drift loading or reducing the size of a repository of a given capacity at Yucca Mountain by a factor of 4.3 to 5.4. Combining this with removal of caesium and strontium allows for much greater reductions in size, upwards of a factor of 40. Further separation of curium would provide for even greater reductions, up to a factor of 225 compared with direct disposal. Studies of realistic recycling options for plutonium and americium emphasize the need to have very low losses for processing recycled fuel, and would require the availability of waste forms that could be densely loaded with the remaining waste materials.

ACKNOWLEDGEMENTS

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ADVANCED FUEL CYCLE STUDIES IN THE OECD NUCLEAR ENERGY AGENCY

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Abstract

Nuclear fuel cycle studies are one example of a horizontal activity within the OECD Nuclear Energy Agency, and involve several divisions and committees. A series of system level fuel cycle studies has been performed covering various aspects such as sustainability, research needs and economics. In addition, scientific issues of the fuel cycles have been studied, especially plutonium recycling.

1. INTRODUCTION

Nuclear fuel cycle studies, especially on partitioning and transmutation (P&T), are one example of a horizontal activity within the OECD Nuclear Energy Agency (OECD/NEA), involving several divisions and committees. Apart from the Committee for Technical and Economic Studies on Nuclear Energy Development and Fuel Cycle (NDC) and the Nuclear Science Committee (NSC), the Radioactive Waste Management Committee (RWMC) is also involved in these studies.

2. EARLIER STUDIES

Since 1996, a series of studies on advanced fuel cycles has been conducted under the auspices of the OECD/NEA. The first study [1] focused on a review of the progress made in separation of long lived actinides and fission products, the options for their transmutation and the benefit for the management of the waste. Specific fuel cycle schemes were discussed, covering plutonium recycling

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and the additional burning of minor actinides (MAs) in dedicated systems. However, the study did not address transmutation strategies with fully closed fuel cycles, or the technology of accelerator driven systems (ADSs), including the specific role of ADSs in such closed fuel cycles.

The second study [2] complemented the first study. It aimed at clarifying the roles and relative merits of critical and subcritical fast spectrum systems in closed fuel cycles with the help of a set of representative 'fuel cycle schemes'. It also assessed the development status of ADSs, with the emphasis on reactor technology and safety, fuel cycle technology, cost–benefit issues and general feasibility. The target values for waste mass and radiotoxicity reduction, to be achieved by an effective transmutation strategy, were defined, and the incentives for closed fuel cycles were discussed, including the role of fast spectrum systems in such fuel cycles.

One recent study [3] paid special attention to the non-technical aspects, especially sustainability, of advanced fuel cycles. This report attempted, by taking an approach broader than the economic and technical aspects, to lay the basis for a more comprehensive decision aiding technique, i.e. multicriteria analysis, that can be used in decision making concerning nuclear energy and other energy options. Representatives of different parts of society (stakeholders) could use this technique to assess the sustainability characteristics of a particular project. Multicriteria analysis is based on the identification of the criteria associated with a sustainable development approach and of their key quantifying indicators.

3. MAIN RESULTS OF EARLIER STUDIES

Earlier studies have concluded that partitioning facilities for actinides (such as plutonium, curium and neptunium) and some long lived fission products could be designed and constructed as extensions to existing reprocessing plants. However, much work has still to be performed in order to make these extensions compatible with industrial reprocessing practices. Studies have constantly re-demonstrated that fast neutron spectrum devices (dedicated fast reactor (FR) or ADS facilities) are more efficient than current LWRs for recycling and transmuting long lived radionuclides.

Fast reactors and ADSs have been found to have similar performance with respect to criteria regarding environmental friendliness. However, from the point of view of the maturity of the technologies and of safety, they differ considerably. Being a subcritical system, the advantage of the ADS reactor concept is that it has fewer limitations on the fuel composition compared with

reactors operated in critical mode. On the other hand, the coupling between a reactor and an accelerator presents a particular technological challenge [2].

Studies of fission product transmutation have shown that transmutation rates are in most cases insufficient to significantly reduce the heat generation and the mass of disposed fission products. Some of the long lived fission products, such as ^{129}I and ^{99}Tc, could theoretically be transmuted by the excess neutrons available in FRs and ADSs. However, this demands an isotopic separation and, in addition, a very challenging preparation of the target.

The study on sustainability [3] of the advanced fuel cycle shows that developments in these cycles rely on new reactor concepts, which will require substantial long term R&D effort and will probably take decades to implement. However, the high temperature gas cooled reactor (HTGR) fuel cycle is essentially a revival of a well advanced earlier concept, which may reach industrial maturity within the next ten to twenty years. In addition, the P&T fuel cycles may need operation for decades or even centuries to reach equilibrium and to really achieve the claimed significant reduction of the radiotoxicity of the waste.

The OECD/NEA has also studied the scientific issues related to fuel cycles, especially plutonium recycling [4–10]. These studies show that single recycle of plutonium as MOX in LWRs is scientifically well established and already used on a commercial scale. Other options, especially those involving non-oxide fuels and multiple plutonium recycling, represent a greater technical challenge, and will require extended research and development programmes to be established on a commercial scale. However, many of these more advanced options have the potential to deliver significant benefits over the single MOX recycle approach.

4. ONGOING STUDIES

The aim of the current system level study is to analyse a range of future fuel cycle options from the perspective of their effect on waste management policies. The main focus is first to evaluate the characteristics of radioactive waste generated by advanced nuclear fuel cycles. Thirteen fuel cycle schemes are defined, to illustrate the differences between various technologies but not to represent foreseeable future fuel cycles. The effects of advanced fuel cycles on the management of waste are presented relative to the current technologies, using such tools as repository performance analysis and cost studies.

The current study extends the analysis performed in previous studies and assesses the fuel cycles as a whole, including all possible radioactive waste generated at each step of the cycles. Comparisons, more qualitative in nature,

with current technologies are used as the basic method, as the waste generation data, especially those for secondary waste, for the advanced technologies are mainly based on estimates by experts.

A novelty of this study is the repository performance assessment performed for the high level waste (HLW) repositories. Two aspects are analysed; firstly, the effect of different HLW isotopic compositions on the repository performance, and, secondly, the effect on repository capacity, though current repository concepts have not been optimized for the new type of waste. The final report of this study is planned to be published at the beginning of 2006.

In parallel to system level studies, the OECD/NEA continues to perform studies related to scientific issues on nuclear fuel cycles. At present, several aspects of separation technologies are being studied. In collaboration with the system level studies mentioned above, a report describing the details and assumptions of the individual separation process flowsheets is under preparation. This work has been extended to define the criteria to be applied to the design and selection of chemical separation processes for the treatment of spent nuclear fuel. The main issues are:

(a) Permissible level of fission product content (especially lanthanides) in LWR thermal recycle of Pu;
(b) Heat load reduction benefits for repository loading (near and long term);
(c) Development of objective recovery efficiency criteria for U, Pu and MAs;
(d) Methodology in general — bases for criteria (e.g. radiotoxicity, heat generation, dose rate and waste volume).

Earlier fuel cycle studies have analysed only cycles in equilibrium states. A new study aims to identify important issues involved in making the transition from current fuel cycles to long term sustainable fuel cycles, providing a framework for assessing specific national needs related to that transition.

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TECHNICAL AND ECONOMIC DRIVING FACTORS FOR ADVANCED FUEL CYCLE DESIGNS INDUSTRIALLY APPLICABLE BY 2030

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Abstract

In recent years, evidence has been accumulating which indicates that global climate change is a reality. Even with measures to improve energy efficiency, global energy demand will surely rise well above the current level. Globally, renewable sources are unlikely to be able to meet the increased demand and there must be a heavier emphasis on carbon-free nuclear power to meet the shortfall. Both the Generation IV Initiative and the IAEA's INPRO project can be cited as evidence that many countries are taking positive steps to address increased demand for nuclear power. The first of the Generation IV plants might well be operational by 2030, and the question of the technical and economic driving factors is one that the international Generation IV research effort will need to address. With engineering lives of up to 60 years, light water reactors (LWRs) currently operational and those currently under construction or planned can be expected to be still operational well into the second half of this century. Generation IV systems will therefore need to operate alongside these LWRs. This provides a point of reference that can be used to define some of the technical and economic driving factors that the Generation IV systems will need to meet. Light water reactor fuels have demonstrated a very high degree of reliability, and Generation IV fuels will at the very least be expected to match this. Utilities are used to the fuel cycle costs of their LWRs and will not want to pay substantially more for Generation IV fuel cycles. The paper develops this theme as a means of defining the technical and economic constraints that Generation IV systems will need to meet.

1. INTRODUCTION

It is difficult to envision a world in 2030 that does not have a completely different outlook on energy from that today. In recent years, more and more evidence has been accumulating which indicates that global climate change is a reality. However, even with measures to improve energy efficiency, global energy demand will surely rise well above the current level. Globally, renewable sources are unlikely to be able to meet the increased demand, and

there must be a heavier emphasis on carbon-free nuclear power to meet the shortfall.

There are signs that countries which in recent years have not promoted nuclear energy are now reconsidering their energy policies. In the United Kingdom (UK), the Government's target for 10% renewables by 2010 has led to an acceleration of construction of large wind farms, both onshore and offshore, which has been achieved through heavy subsidization. However, there are signs of difficulties with this policy, with opposition to large wind generators and the power lines needed to deliver the electricity to the centres of consumption in the south of England. Conservation groups are proposing small wind turbines in small groups, with sensitive choices of siting to meet local demand. Where long distance transmission is needed, they would prefer undersea cables in place of overhead pylons. If these demands are met, the total contribution of renewable sources is likely to be restricted, and their cost is likely to be increased even further. In response to the evident demand for electricity and the likely failure of renewable sources to deliver, there are signs that the UK Government is reconsidering nuclear power as a means of meeting its Kyoto obligations.

It is probable that many other countries will soon be in the same situation, and there is a real prospect of a revival of nuclear construction. Any new build programme prior to 2030 is likely to be dominated by light water reactors (LWRs), and these will define the economic and technical constraints for commercial deployment of Generation IV systems, which could begin about this time, and also for intermediate generation systems such as high temperature reactors (HTRs), which might have made inroads into the commercial market by that date. From the point of view of a commercial power generation company, any new systems will have to be competitive with LWRs, which can be used to define the economic and technical benchmarks that the new systems will need to equal and preferably exceed.

The stated Generation IV goals are to develop systems that will be beneficial with respect to safety, economics, sustainability, non-proliferation and waste minimization. These new systems will need to perform better in these respects than LWRs in 2030. This is particularly the case for their economics, because no commercial utility would be willing to choose a Generation IV system in place of an LWR if its overall economics were unfavourable. This paper develops the theme of LWRs as a baseline to a means of developing quantitative targets that Generation IV systems will need to meet or at least approach, concentrating on economics, sustainability and waste minimization.

2. FUEL CYCLE ECONOMICS BASELINE

From the perspective of a commercial utility, the economic baseline is probably the most important one; if the advanced systems are not economically competitive with LWRs, they are unlikely to penetrate the market. It is the overall generating cost of a system that is the main determining factor, comprising capital, operating and maintenance, and fuel costs, together with the cost of making provisions for decommissioning and waste disposal. While it might be possible to trade off different components (e.g., to accept a higher fuel cycle cost if the capital cost is much lower than that of LWRs), a utility might well be uncomfortable if such a trade-off led to an extreme imbalance relative to that of LWRs. Thus, a situation whereby the fuel cycle cost of a Generation IV system was substantially higher than that of LWRs might be considered undesirable by utilities, and a sensible baseline for the fuel cycle cost would be parity or near parity with that of LWRs. To a utility, the fuel cycle cost represents a substantial ongoing outlay that would possibly expose it to market volatility and make planning more difficult in the long term. With this premise, which is difficult to justify rigorously but nevertheless seems correct, it is possible to define a baseline for fuel cycle economics as described in the following.

While the fuel cycle economics of current LWRs is well understood, it needs to be projected to 2030 for the present purpose. Between now and 2030, LWR fuel technology can be expected to have advanced considerably, as it has in the past 25 years. In particular, average discharge burnups may have evolved well beyond the 50 GW·d/t current standard. Precisely where this evolution will have reached by 2030 is very difficult to foresee. This very question is presently one that is being addressed by an OECD Nuclear Energy Agency (OECD/NEA) Expert Group instigated by the Nuclear Science Committee [1]. It is reviewing the technical requirements for obtaining LWR discharge burnup between 60 and 100 GW·d/t. The Expert Group has already concluded that, while technologically such burnups are at least feasible, it is not at present possible to project where burnup evolution is likely to progress beyond 2015 or so.

Fortunately, this is not a significant impediment to defining the fuel cycle economics baseline. Reference [1] has already established that the fuel cycle levelized cost (i.e. measured in $/MW·h) is not very sensitive to burnup for average discharge burnups greater than 50 GW·d/t. Reference [1] strongly suggests that the levelized cost reaches a shallow minimum somewhere in the range 50–60 GW·d/t and thereafter increases with burnup, albeit very slowly. The minimum levelized fuel cycle cost for LWRs is ideal to use as the baseline for advanced systems for 2030. It decouples the choice of baseline from any

projections as to what burnup LWRs will achieve by then. If LWR utilities choose to go to higher burnups than the fuel cycle economic optimum, it will be because of operational gains that they can make elsewhere (such as achieving longer fuel cycles or reducing spent fuel volumes). This would slightly raise the LWR baseline, a relatively inconsequential change.

Having established the principle of the fuel cycle economics baseline, it is possible to quantify it. Table 1 provides a range of estimates for the minimum LWR levelized fuel cycle costs. The units used are $/MW·h, which is numerically the same as the more commonly used mill/kW·h (where a mill is 0.001 US dollars). A range of values is quoted to account for sensitivity to the market price of uranium ore and for uncertain spent fuel management costs. It is argued below that these are the main sensitivities that need to be included. For each case, the discharge burnup was varied, to establish the point at which the fuel cycle cost is a minimum, and this is indicated in Table 1. A discount rate of 10% was assumed in all the calculations.

The main uncertainty on LWR fuel cycle cost is in the projected market price for uranium ore in 2030. For many years now the uranium ore market price has been remarkably static at around 25 $/kg. There is a view that uranium prices have been held artificially low during this time [2] and that they may therefore rise over the next few years. The current market price of 25 $/kg therefore defines the lower limit of a sensitivity analysis, while the upper limit will be set at 50 $/kg purely for the purposes of establishing sensitivity.

With respect to the remaining front end fuel cycle cost items, uranium conversion, enrichment and fabrication, it can be argued that there are no major sensitivities. Market prices for uranium conversion and enrichment have historically been relatively static, with a trend to a gradual reduction in the price of separative work units (SWUs) due to steady improvements in enrichment technology. The conversion price has therefore been set at 7 $/kgU and enrichment at 90 $/SWU, in line with the current market. Similarly, it is not considered very likely that fuel fabrication prices will be sufficiently volatile to significantly affect the overall fuel cycle economics (especially since fabrication represents only about 10% of the total), and fabrication has been set at 300 $/kgU.

The sensitivity range for spent fuel management costs is more difficult to estimate. The historic trend has been that underlying costs gradually increase due to increasingly stringent safety and environmental requirements. While this trend might be expected to continue, technological improvements might reasonably be expected to reduce intrinsic costs. To establish the sensitivity, a third case has been included in Table 1 in which back end costs are assumed to scale proportionally to the average discharge burnup.

TABLE 1. COST ESTIMATES FOR THE LWR BASELINE FUEL CYCLE

Case	Minimum fuel cycle levelized cost ($/MW·h)	Discharge burnup for minimum fuel cycle cost (GW·d/t)
Nominal ore price 25 $/kgU; nominal spent fuel management costs	4.04	55
High ore price 50 $/kgU; nominal spent fuel management costs	4.88	55
High ore price 50 $/kgU; high spent fuel management costs	5.00	45

For the baseline case (25 $/kgU ore cost) and the high ore price case (50 $/kgU), the minimum fuel cycle cost occurs at approximately 55 GW·d/t average discharge burnup. The second sensitivity case, in which back end costs are assumed to increase with burnup, shows a monotonic increase in fuel cycle cost with burnup, so the lowest burnup case (45 GW·d/t) is the minimum. The sensitivity to uranium ore price is significant (≈0.8 $/MW·h for a 25 $/kgU price increase) and represents an important uncertainty in LWR baseline costs in 2030. Figure 1 shows the variation of fuel cycle levelized cost versus burnup in more detail. The more the uranium ore price increases, the more favourable the situation will be for Generation IV systems.

The baseline fuel cycle costs in Table 1, ranging from 4.04 to 5.00 $/MW·h, delineate a very specific target range for HTR and Generation IV systems for 2030. High temperature reactors have a higher thermal efficiency than LWRs, a positive factor that will help to reduce the levelized fuel cycle cost below the LWR baseline. However, HTRs do suffer a disadvantage in that HTR cores are somewhat undermoderated and therefore have a slightly higher initial enrichment requirement than LWRs. This to some extent offsets the thermal efficiency benefit. The relatively high uranium ore and SWU requirement will constrain HTR fuel fabrication costs if they are to be competitive.

Most of the Generation IV systems now being pursued as part of the Generation IV Initiative [3] make use of a fully closed fuel cycle to meet the requirement for long term sustainability. A closed fuel cycle is not dependent on the market price of uranium ore, as it is self-sufficient in fissile material. The fuel cycle cost for a closed fuel cycle is determined by the cost of reprocessing spent fuel, refabricating the fissile component into new fuel, and managing and

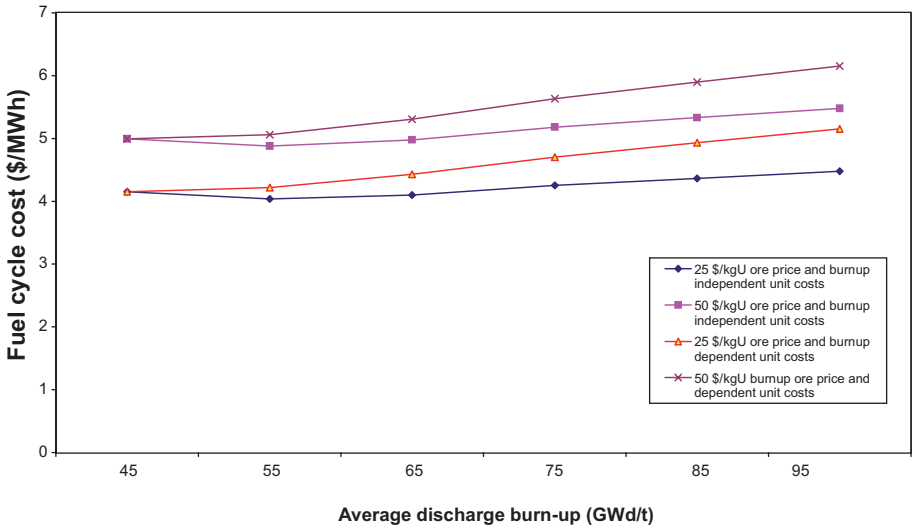

FIG. 1. Variation of fuel cycle levelized cost for a PWR versus average discharge burnup and uranium ore price.

eventual disposal of the waste. The total cost for all three components should meet the target range identified.

Like HTRs, Generation IV systems generally have a higher thermal efficiency than LWRs that reduces the levelized fuel cycle cost. However, the need to process high burnup fuels, probably with high concentrations of fission products, plutonium and minor actinides, may demand new, cheaper, reprocessing technologies. This is exacerbated by the need to minimize the spent fuel cooling time if the working inventory of fissile material is to be minimized. An additional factor is that fuel fabrication may well demand remote techniques because of the high neutron output and gamma activity that can be expected. These points have been recognized by the partners in the Generation IV Initiative and cross-cutting working groups have been identified to address these generic questions. Light water reactors have proven themselves to be remarkably efficient, and the technology is very mature. Matching the LWR fuel cycle cost baseline will be a major technological challenge for Generation IV systems.

3. SUSTAINABILITY BASELINE

It is well known that the resource sustainability (i.e. the uranium ore requirement) of once-through systems such as LWRs and HTRs is relatively

poor. Under current market conditions this is not considered a concern, but by 2030 a different view may prevail. One of the main goals of Generation IV systems is to achieve a significant improvement in resource sustainability, and the expectation is that by 2030 market conditions will have changed to emphasize this aspect.

The uranium ore requirement for LWRs varies slightly with burnup, and Ref. [1] shows that it reaches a minimum at an average discharge burnup of 55 GW·d/t of approximately 7.9 t/TW·h thermal, with a slight dependence on the fuel management scheme used. With LWR thermal efficiencies of ≈33%, this is equivalent to 23.8 t/TW(e)·h. This defines the baseline for other systems.

This raises the question of whether LWR and HTR resource sustainability could be improved. In principle, there is scope for improvement by reducing the enrichment tails assay below the 0.3 wt% that prevails at present. The tails assay is determined by optimizing between the cost of uranium ore and SWUs. With uranium ore relatively cheap, the current market favours a high tails assay. This is an adverse factor for sustainability, as can be seen by calculating the isotopic cut for ^{235}U. This is the fraction of ^{235}U atoms that carry over into the enriched product from the enrichment cascade. A high tails assay means that a significant fraction of the ^{235}U atoms resides in the tails; for current LWR enrichments this is about 40%, corresponding to a cut of approximately 60%.

This inefficiency could be reduced with a lower tails assay, with a cut of perhaps 80% achievable, but with an enrichment cost penalty. If uranium ore prices are higher by 2030, the optimum tails assay will fall below 0.3 wt%. This is natural market forces, which will have a benefit by increasing the enrichment cut and lowering the LWR baseline to a practical minimum of about 15.8 t/TW(e)·h.

Despite the slightly higher initial enrichment requirement of HTRs, their high thermal efficiency should enable them to achieve a modest improvement over LWRs, with a range of uranium requirements from 12.6 to 18.0 t/TW(e)·h, depending on the tails assay. This modest improvement contrasts with the majority of Generation IV systems, which would operate with self-sustaining breeding cycles. In principle, breeding systems can generate 100–200 times more energy per kilogram of uranium ore, so that they can be operated indefinitely. Historical fast reactor development programmes pursued by France, Japan, the Russian Federation, the UK and the United States of America focused on achieving high breeding ratios as one of their key design objectives. For the scenarios then envisaged, breeding ratios considerably in excess of 1.0 were thought to be essential to accommodate losses in the fuel cycle and to enable fast reactors to be taken on-line with only limited supplies of separated plutonium.

Both axial and radial breeder blankets were needed to achieve the desired breeding ratios, and sometimes fuel assemblies designed for optimal breeding were not the most economic ones. Since one of the objectives of Generation IV systems is to minimize proliferation risk, the use of radial breeder assemblies is undesirable, and this will make high breeding ratios more difficult to achieve.

However, is it actually necessary for Generation IV systems to operate as true breeders with a conversion ratio in excess of 1 and would it perhaps be preferable (and more practical) to accept a lower target of, say, 0.8 or 0.9? It could be argued that if the present 400 GW(e) global nuclear capacity was to continue to 2030 and beyond, current uranium reserves would continue to be sufficient to meet demand even with LWRs.

In a more optimistic scenario, where world nuclear capacity for 2030 and beyond was to expand significantly, such an argument would be more difficult to sustain. However, if the Generation IV breeding ratio was adjusted so that demand for uranium ore extraction was maintained at around present levels, then would this not satisfy the Generation IV sustainability objective? In this case, a factor of 5–10 increase in global nuclear capacity (which is consistent with the scenarios being considered under the IAEA INPRO activity [4]) could be accommodated with a conversion ratio ranging from 0.8 to 0.9, with a uranium ore requirement between 2.1 and 4.2 t/TW(e)·h. This would avoid the need for radial breeders.

Flexibility in the fuel cycles, to use low enriched uranium (LEU) for a small fraction of the fuel requirement, would allow the Generation IV systems to operate indefinitely. The precise conversion ratio (and therefore the LEU fraction) could be adjusted as needed to balance uranium demand and supply.

4. BASELINE FOR WASTE ARISINGS

Since LWRs will still be the predominant systems in use in 2030, they can be used to define the baseline in terms of waste arisings. This is a complicated area, with many different measures of waste arisings that could be used. For the once-through LWR fuel cycle, the simplest measure would be the mass or volume of spent fuel per TW(e)·h, which for current LWR burnups are approximately 3 tHM/TW(e)·h and 1.2 m^3/TW(e)·h, respectively. Both mass and volume arisings are inversely proportional to burnup, so the adoption of very high burnups would seem to reduce the LWR baseline according to these measures.

For the reprocessing case, the situation is less straightforward. The mass of fission products produced is independent of burnup. If it was possible to

assume that the incorporation rate of fission products in the high level waste matrix was constant, then this would imply a high level waste volume that is independent of burnup. However, there may be constraints on fission product incorporation due to neutron output that may cause the high level waste volume to increase with burnup. Therefore, with reprocessing, there is at best no reduction in high level waste volume with burnup, and there may even be an increase.

Perhaps a more relevant measure is the decay heat output of LWR spent fuel or high level waste. The capacity of a deep geological repository is limited principally by heat output and not by the physical mass or volume of spent fuel or high level waste packages. Irrespective of whether a once-through or reprocessing cycle applies, the same decay heat output from fission products and minor actinides needs to be accommodated within the repository. Reference [1] shows that as average burnups increase, there is a corresponding increase in decay heat output measured in kW/tHM. Indeed, the decay heat output at cooling times relevant to the time of emplacement of spent fuel or high level waste containers in the repository (taken to be 50 years) increases by a factor of ≈4 when the burnup is doubled from 45 to 90 GW·d/t. This implies a possible reduction of capacity in the geological repository unless the heat producing wastes are held longer in interim storage.

Decay heat output can therefore be regarded as the limiting factor for the back end of the fuel cycle. The preceding discussion points to the decay heat output after 50 years of cooling of LWRs at current burnups as being the logical baseline for back end waste arisings, since any move to higher burnups would only result in a higher value. At current LWR burnups, the decay heat commitment at a cooling time of 50 years equates to approximately 2 kW/TW(e)·h. Perhaps this is a suitable baseline for HTRs and Generation IV systems. Although the high thermal efficiency of HTRs is beneficial in respect of this measure, the fact that HTR burnups are higher than those of LWRs is an adverse factor. The precise balance between these competing effects is not clear and needs to be examined, but it would appear that the performance of HTRs measured against decay output will be comparable to that of LWRs.

A similar analysis will also be needed for Generation IV systems. However, those Generation IV systems with fast neutron spectra are likely to benefit both from their high thermal efficiency and the fact that fast systems tend to accumulate fewer heat producing minor actinides.

Another measure for waste arisings that might be considered is the radiotoxicity in sieverts per TW(e)·h. Light water reactors can again be used to set the baseline for radiotoxicity. Reference [1] shows that the radiotoxicity of LWR fuel, measured in sieverts per TW(e)·h, shows relatively little sensitivity to average discharge burnup, so that the precise burnup of the LWR fleet in

2030 does not substantially affect the baseline. As the various radiotoxic nuclides decay, total radiotoxicity decreases, and the timescale on which radiotoxicity should be measured becomes important. Timescales of 50 years or less are irrelevant, as repository emplacement is not expected within this time. Following emplacement and subsequent closure of the repository, it seems reasonable to assume that physical containment and institutional safeguards will be sufficient to isolate the radiotoxic nuclides for at least several hundred years. This suggests that a reference time frame for the baseline should coincide with the decay of the bulk of the fission products. Since this occurs after between 500 and 1000 years, it is suggested that this is the most relevant cooling time to specify for the baseline.

Choosing a cooling time of 1000 years, the radiotoxicity baseline of LWRs is approximately 5×10^7 Sv/TW(e)·h, assuming a once-through cycle. High temperature reactors should be able to achieve a slight reduction in radiotoxicity on account of their higher thermal efficiency. However, Generation IV systems with fast spectra should be able to improve considerably on this baseline through a combination of high thermal efficiency, lower rate of accumulation of minor actinides and recycle of plutonium.

5. CONCLUSIONS

This paper has considered the economic and technical baselines that LWR fuel cycles might define in 2030. The basic premise is that LWRs will still be dominant in 2030, with HTRs having become firmly established and Generation IV systems just starting to be deployed. If HTRs and Generation IV systems are to gain acceptance, they will need to match or improve on the LWR baselines, and this is the theme that has been explored in this paper. Despite the uncertainties as to what LWR fuel cycles will look like in 2030 (especially uncertainty about average discharge burnups), it has been possible to define quite precise criteria for fuel cycle economics, with a fuel cycle cost baseline of 4–5 $/MW(e)·h. The uranium consumption baseline is 15.8 t/TW(e)·h, and an argument is put forward that a conversion ratio in the range 0.8–0.9 should be sufficient to meet the Generation IV goals, even if a substantial increase in global capacity is expected beyond 2030. At the back end of the fuel cycle, the key limiting factors identified are the decay heat output at the time of emplacement of waste in the geological repository and the radiotoxicity at a cooling time of the order of 1000 years, the corresponding baselines for which are 2 kW/TW(e)·h and 5×10^7 Sv/TW(e)·h, respectively. It is hoped that the arguments presented here in favour of these baselines will help to guide the Generation IV Initiative research and development effort.

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STATUS AND PROSPECTS OF INNOVATIVE NUCLEAR FUEL CYCLE TECHNOLOGY

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Abstract

In the light of ever increasing energy and electricity demands worldwide and the growing concerns about greenhouse gas emissions leading to global warming, nuclear energy stands out as a viable emission-free option for the twenty-first century. At present, two international initiatives are under way, namely, the IAEA initiated International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the United States led Generation IV International Forum (GIF). Both programmes address some common issues, namely, the economics of nuclear electricity in the free market, the use of nuclear process heat for desalination of sea water and production of hydrogen for the transportation sector, the inherent or passive safety features of nuclear reactors, waste management and protection of the environment, proliferation resistant reactor technology, public acceptance and long term sustainability of nuclear power. Several countries are evolving their national nuclear fuel cycle strategies to meet the rising expectations of nuclear power in the twenty-first century. Some of the major national programmes are as follows: the Advanced Fuel Cycle Initiative (AFCI) of the United States of America, the closed fuel cycle programmes in Japan and the Russian Federation involving multiple recycling of plutonium with minor actinides in fast reactors, the French programme of monorecycling of plutonium in pressurized light water reactors, followed by multiple recycling of plutonium in fast reactors, and the three stage nuclear power programme in India linking the natural uranium fuel cycle in pressurized heavy water reactors, with the ^{238}U–^{239}Pu fuel cycle in fast reactors using thorium blankets and the self-sustaining ^{232}Th–^{233}U fuel cycle in thermal reactors. The paper summarizes the status and prospects of innovative nuclear fuels and fuel cycle technologies, highlighting the conventional and advanced fuels for water, sodium and gas cooled reactors in addition to conventional and advanced methods for reprocessing spent fuels.

1. INTRODUCTION

Nuclear power reactors, utilizing the heat energy of nuclear fission for generation of electricity, were first introduced in France, the Russian Federation, the United Kingdom (UK) and the United States of America (USA) in the mid-1950s. During the last 50 years, nuclear power reactors and their associated fuel cycles have progressively developed into industrial activities. At the end of 2004, some 441 nuclear power reactors were in operation in 30 countries, with a total installed capacity of 368 GW(e), and a further 26 power reactors were under construction, with a total capacity of 20.8 GW(e). The share of global electricity production held by nuclear power has remained steady at ≈16% for more than a decade, and the expectations are rising for nuclear power to supply, in the twenty-first century, clean energy and electricity at an affordable price, in a sustainable manner without degradation of the environment. The high and low predictions of the IAEA for nuclear power in the year 2030 are 640 and 418 GW(e), respectively [1].

Nuclear power supports sustainable development by providing much needed energy with a relatively low burden on the atmosphere, on water and on land use. Furthermore, deployment of nuclear power could help to alleviate the environmental burden caused by other forms of energy production, particularly the burning of carbon based fossil fuels, which emit greenhouse gases, leading to global warming. The future role of nuclear energy is centred around issues such as economic competitiveness, safety, waste, proliferation resistance and physical protection, and last, but not least, sustainability and environmental protection. Hence, there is a need for innovative nuclear energy systems (INSs) that will be superior to existing plants and comprise not only electricity generating plants but also supply of:

(a) High temperature process heat for economic production of hydrogen for the transportation sector as a substitute for carbon based fuels;
(b) Process heat for district heating and seawater desalination.

Innovative nuclear systems will be deployed in both developed and developing countries to meet the ever increasing demand for primary energy worldwide, without degradation of the environment, and will be most effective in large and highly populated developing countries like China and India with fast growing economies. However, the potential adverse effects that the various components of the nuclear fuel cycle may have on the environment must be prevented or mitigated effectively to make nuclear energy sustainable in the long term. Both radiological and non-radiological effects must be considered.

Uranium and thorium, the two heaviest elements occurring in nature, are the basic raw materials for nuclear fuels. Natural uranium contains 99.3% of the fertile isotope ^{238}U and only 0.7% of ^{235}U, which is the only fissile isotope occurring in nature. Natural thorium is three times more abundant in the earth's crust but does not contain any fissile isotope. It is made up of fertile ^{232}Th only. Neutron capture by ^{238}U and ^{232}Th in reactors leads to the production of the human-made fissile isotopes ^{239}Pu and ^{233}U, respectively. In addition, the neutron capture reactions of ^{238}U and ^{239}Pu lead to the formation of several fissile and fertile isotopes of plutonium and minor actinides (MAs), as listed in Table 1 [2]. Nuclear materials consisting of uranium, plutonium, thorium and MAs are radioactive and hazardous to health to varying degrees as well as being dual use materials that can be diverted for non-peaceful applications. Hence, proliferation resistance is of paramount importance in ensuring the peaceful application of nuclear fission energy.

TABLE 1. NUCLEAR PROPERTIES OF FISSILE AND FERTILE NUCLEAR MATERIALS

Isotope	Half-life (a)	Neutrons ($n \cdot s^{-1} \cdot kg^{-1}$)	Decay heat (W/kg)	Critical mass (kg)
^{231}Pa	32.8×10^3	Nil	1.3	162
^{232}Th	14.1×10^9	Nil	Nil	Infinite
^{233}U	159×10^3	1.23	0.281	16.4
^{235}U	700×10^6	0.364	6×10^{-5}	47.9
^{238}U	4.5×10^9	0.11	8×10^{-6}	Infinite
^{237}Np	2.1×10^6	0.139	0.021	59
^{238}Pu	88	2.67×10^6	560	10
^{239}Pu	24×10^3	21.8	2.0	10.2
^{240}Pu	6.54×10^3	1.03×10^6	7.0	36.8
^{241}Pu	14.7	49.3	6.4	12.9
^{242}Pu	376×10^3	1.73×10^6	0.12	89
^{241}Am	433	1540	115	57
^{243}Am	7.38×10^3	900	6.4	155
^{244}Cm	18.1	11×10^9	2.8×10^3	28
^{245}Cm	8.5×10^3	147×10^3	5.7	13
^{246}Cm	4.7×10^3	9×10^9	10	84
^{247}Bk	1.4×10^3	Nil	36	10
^{251}Cf	898	Nil	56	9

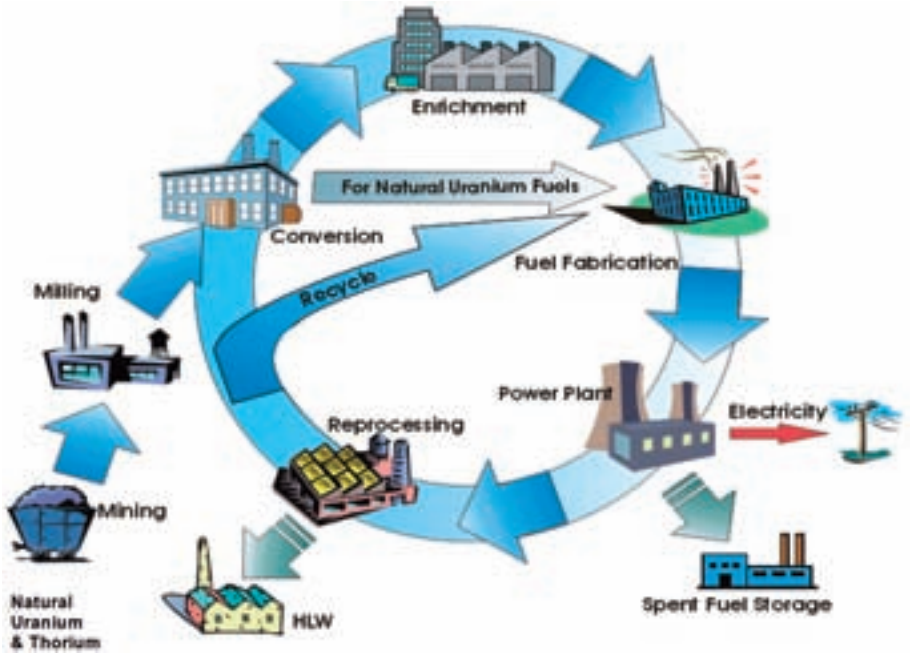

FIG. 1. Nuclear fuel cycle activities.

Nuclear fuels are made of the fissile isotopes ^{235}U, ^{239}Pu or ^{233}U, mostly with a judicious combination of the fertile isotopes ^{238}U or ^{232}Th. The fuels for the operating nuclear power reactors in the world are mostly used in the form of uranium oxide pellets and, to a limited extent, mixed uranium plutonium oxide (MOX) pellets. Metallic uranium alloy fuel, cast as long cylindrical pins, has so far been used commercially only in Magnox gas cooled reactors, which are in operation only in the UK. Non-oxide ceramic (carbides and nitrides) pellets, oxide and non-oxide ceramic microspheres, cermet and cercer of the above fissile and fertile materials have been used as fuels to a very limited extent in demonstration and prototype power reactors and test fuel pins. Thorium based fuels are not being used commercially anywhere in the world, but are likely to play an important role in the long term nuclear power programmes of some countries such as India, which has abundant thorium reserves.

Nuclear fuel cycle activities, shown in Fig. 1, encompass uranium and thorium exploration, mining and milling, purification and conversion, enrichment (for uranium only for use in light water reactors (LWRs) and advanced gas cooled reactors (AGRs)) and fuel fabrication in the front end.

All activities related to the management of spent nuclear fuel (SNF), including storage, reprocessing, refabrication of fuel and waste management, form part of the back end activities. In the once-through open ended fuel cycle, spent uranium fuel is subjected to underwater wet storage initially, followed by long term away-from-reactor (AFR) wet/dry storage before permanent disposal in a repository. In the once-through fuel cycle, only ≈1% of uranium resources are utilized. In the 'closed' fuel cycle, SNF is considered as a source of energy. The SNF is reprocessed to recover fertile and fissile materials, and subjected to multiple recycling for the most efficient utilization of natural uranium and plutonium resources.

Programme B of the IAEA, being implemented by the Nuclear Fuel Cycle and Materials Section (NFC&MS) of the Division of Nuclear Fuel Cycle and Waste Technology in the Department of Nuclear Energy, addresses all issues associated with the nuclear fuel cycle. The tasks and activities of Programme B carried out in NFC&MS are outlined in Fig. 2. The objective of this programme is to strengthen the capabilities of interested IAEA Member States for policy making, strategic planning, technology development and implementation of safe, reliable, economically efficient, proliferation resistant, environmentally sound and secure nuclear fuel cycle programmes. Figure 3 shows the integrated Nuclear Fuel Cycle Information Systems (iNFCIS) databases maintained by the IAEA. The databases available are: iNFCIS, the World Distribution of Uranium Deposits (UDEPO), the Minor Actinide Database (MADB), the Fuel Cycle Simulation System (VISTA) and the post-irradiation examination (PIE) facilities worldwide.

The present paper summarizes the status of uranium and thorium resources, conventional and advanced nuclear fuels, conventional and advanced methods of oxide and non-oxide fuel fabrication, and innovative nuclear fuel cycle initiatives worldwide, highlighting the fast reactor fuel cycle.

2. URANIUM AND THORIUM RESOURCES AND NUCLEAR FUELS

Table 2 indicates the status of global uranium resources [3], listing those countries with major uranium resources and those countries with major nuclear power programmes utilizing these resources. It may be noted that more than 50% of uranium reserves are in Australia, Kazakhstan, Mongolia, Namibia, Niger and Uzbekistan, where there are no nuclear power reactors. On the other hand, countries such as France, Germany, Japan and the Republic of Korea, where the contributions of nuclear power to generation of electricity are in the

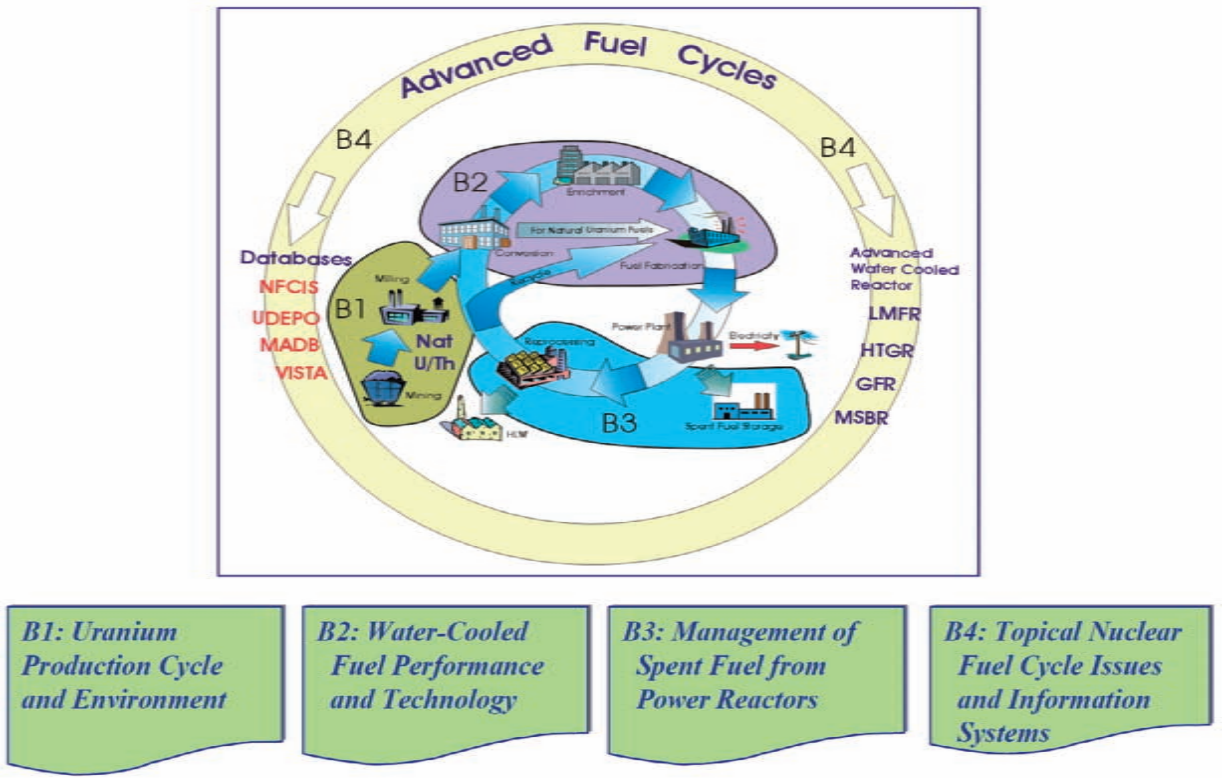

FIG. 2. Major activities on nuclear fuel cycle under IAEA Programme B.

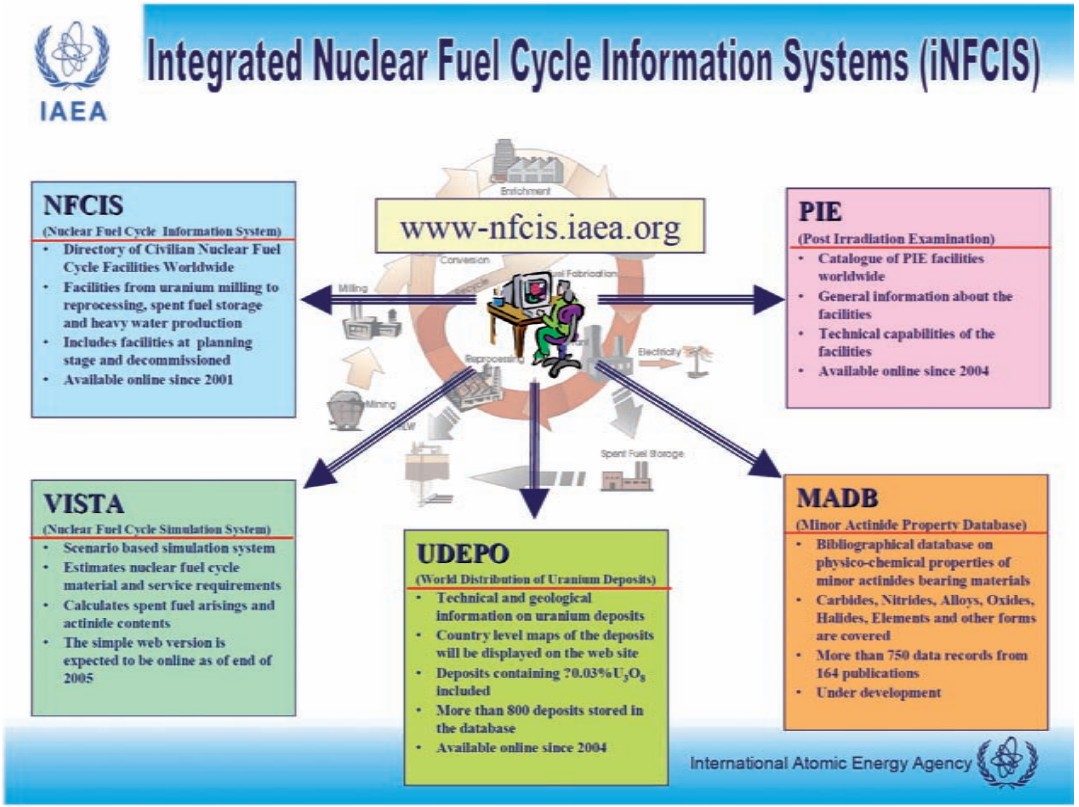

FIG. 3. The integrated Nuclear Fuel Cycle Informatioan System (iNFCIS) maintained by the IAEA.

range 30–78%, depend totally on uranium resources from overseas. The recent IAEA International Symposium on Uranium held in June 2005 concluded that there are adequate uranium resources worldwide to meet the fuel requirements of rapidly expanding nuclear power programmes up to 2050 and beyond, but that the gap between the 'uranium in the ground and yellow cake (uranium concentrate) in the can' has to be narrowed by augmenting the exploration, mining, milling and production activities of uranium.

The thorium reserves located so far, in Australia, Brazil, Canada, Egypt, Greenland, India, Norway, South Africa, Turkey and the USA, have been elaborated in a recent IAEA publication [4]. The exploration and mining activities for thorium have been limited because the present generation of nuclear power plants derive energy from the fission of ^{235}U. Fertile thorium resources are not likely to be used on a commercial scale globally in the coming decades, except in India, where a major R&D programme on the thorium fuel cycle is under way.

The present generation of nuclear power plants all derive energy mostly from the fission of ^{235}U in water cooled reactors. Light water reactors, consisting of pressurized water reactors (PWRs) of both Western and Russian (WWER) design, as well as boiling water reactors (BWRs), are most common (87%), followed by pressurized heavy water reactors (PHWRs), which account for 6%, and light water cooled graphite moderated reactors (LWGRs), which constitute 3%. Gas cooled reactors, namely Magnox reactors and AGRs, are in operation only in the UK and account for 3%. Liquid metal cooled fast reactors (LMFRs) have yet to be commercialized and constitute ≈1% of nuclear power. Light water reactors, LWGRs and AGRs use low enriched uranium (LEU), containing up to 5% of ^{235}U, as fuel in the form of high density uranium oxide pellets. PHWRs and Magnox reactors use natural uranium fuel in the form of high density oxide pellets and metallic uranium, respectively. The fissile material content in fast reactor fuel is in the range 15–25%. Water cooled power reactors use zirconium alloy as cladding, while LMFRs use stainless steel cladding of both austenitic and ferritic types. The trend in water cooled reactor fuel is towards improved burnup, particularly in LWRs.

The spent low enriched uranium (<5% ^{235}U) oxide fuel from light water cooled power reactors would contain on the average ≈95.6% uranium, ≈0.9% plutonium, ≈3% stable short lived fission products, ≈0.3% strontium and caesium, ≈0.1% long lived iodine and technetium isotopes, and ≈0.1% MAs (mainly Np, Am and Cm). The plutonium formed by the neutron capture of fertile ^{238}U consists of fissile ^{239}Pu and other isotopes of plutonium, namely ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{238}Pu. The MAs are also produced by the neutron capture reactions with ^{238}U and plutonium. The MAs are fissionable by fast neutrons and are considered as a source of fission heat energy. Plutonium-239 is the best

TABLE 2. COUNTRIES WITH MAJOR KNOWN CONVENTIONAL IDENTIFIED URANIUM RESOURCES AND COUNTRIES WITH MAJOR NUCLEAR POWER PROGRAMMES

(World uranium resources:

(a) Known conventional reasonably assured resources (RARs) at <130 US $/kg U: 3.297 million tons U

(b) Known conventional inferred resources at <130 US $/kg U: 1.446 million tons U

(c) Known conventional identified resources ((a) + (b)) at <130 US $/kg U: 4.743 million tons U

(d) Undiscovered conventional resources (prognosticated + speculative) at <130 US $/kg U: 7.07 million tons U

(e) Undiscovered speculative resources (cost range unassigned): 2.98 million tons U

(f) Unconventional resources in rock phosphates alone: 22 million tons U)

Country	Uranium resources (tons U)	Percentage of world resources (%)	Number of nuclear power reactors (% electricity)
Australia	1 143 000	24	Nil
Kazakhstan	816 099	17	Nil
Namibia	282 359	6	Nil
Niger	225 459	5	Nil
Uzbekistan	115 526	2.5	Nil
Mongolia	61 950	1.5	Nil
USA	342 000[a]	7	104 (20%)
Canada	443 800	9.4	20 (≈12%)
South Africa	340 596	7	2 (5.9%)
Russian Federation	172 402	3.6	30 (16%)
Brazil	278 700	6	2 (4%)
India	64 840 (>130 US $/kg)	1.4	15 (2.8%)
China (excl. Taiwan)	59 723	1.3	9 (2.2%)
France	100% from overseas sources		59 (78%)
Germany	100% from overseas sources		18 (32%)
Japan	100% from overseas sources		54 (30%)
Republic of Korea	100% from overseas sources		19 (38%)

[a] Only RARs; no inferred resources.

fissile material in the fast neutron spectrum for breeding from ^{238}U. Likewise, ^{233}U is the best fissile material in the thermal neutron spectrum.

At present, only France and the UK are reprocessing spent fuel from thermal reactors on an industrial scale, recovering the plutonium and manufacturing MOX fuel, containing up to 5% Pu, for use in LWRs. Semi-industrial/pilot scale reprocessing and MOX fuel fabrication plants are also in operation in India, Japan and the Russian Federation. Mixed oxide fuel is being used in more than 30 PWRs in Belgium, France, Germany and Switzerland. Nearly a third of the core of these PWRs is made up of MOX fuel. Japan and the Russian Federation are planning to use MOX fuel commercially in their LWRs in the near future. India has also recycled MOX fuel in a limited way in their two BWRs and in one PHWR.

At present, the trend in most countries is mono-recycling of plutonium in water cooled reactors and storing the spent MOX fuel for reprocessing at a later date when the fast reactor technology matures. Plutonium and MAs will be subjected to multiple recycling in fast reactors, as shown in Fig. 4, for burning MAs and for breeding and burning plutonium. Thus, the natural uranium resource will be efficiently utilized and the radiotoxicity in the waste will be significantly reduced. The ultimate objective of fast reactors is to breed

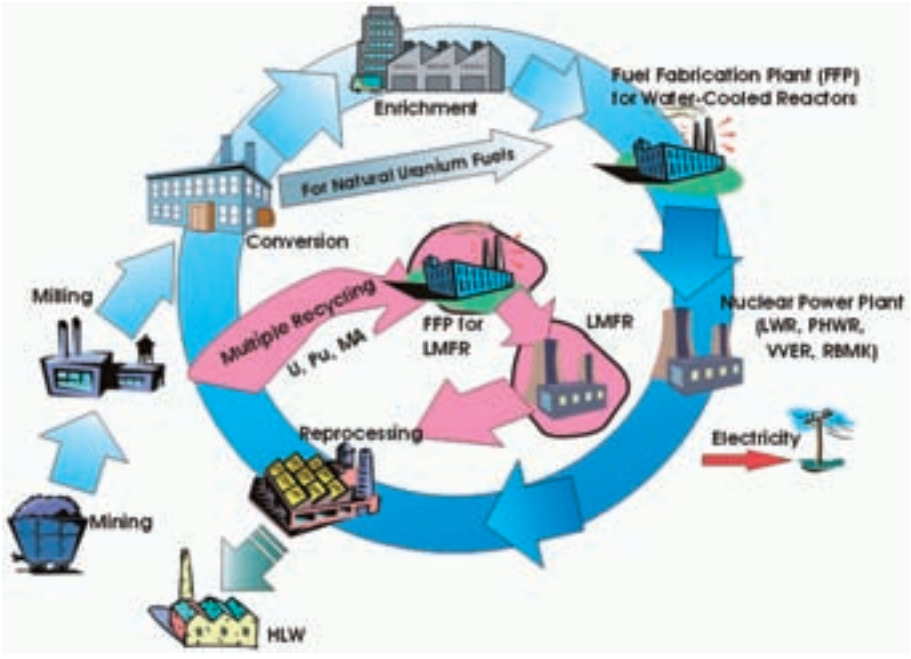

FIG. 4. Multiple recycling of plutonium and MAs in LMFRs with a closed fuel cycle.

fissile material efficiently and burn MAs effectively. Although the conventional fuel for LMFRs is MOX, mixed uranium plutonium monocarbide, mononitride and the metallic compound U–Pu–Zr are considered as advanced LMFR fuels because of their high breeding ratios, high thermal conductivities and excellent chemical compatibilities with sodium coolant.

High temperature gas cooled reactors (HTRs) are being developed for dual applications, namely generation of electricity and high temperature (500–950°C) process heat for generation of hydrogen by iodine sulphur (IS) or high temperature electrolysis. Hydrogen would progressively replace carbon based fossil fuels in the transportation sector. The conventional and advanced fuels and the cladding materials for the different power reactors are listed in Table 3.

3. INPRO AND GIF

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated by the IAEA in September 2000. The objectives of INPRO are to:

(a) Help to ensure that nuclear energy is available to contribute, in a sustainable manner, to energy needs in the twenty-first century;
(b) Bring together technology holders and users so that they can consider jointly the international and national actions required for achieving the desired innovations in nuclear reactors and fuel cycles.

At present, there are 24 IAEA Member States in INPRO and several others (e.g. Japan, the UK and the USA) participate as observers.

In July 2002, the USA initiated the Generation IV International Forum (GIF), with the aim of developing advanced nuclear energy systems for international deployment after the year 2030. At present, ten countries, namely Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, Switzerland, the UK and the USA, are participating in GIF. Six reactor concepts have been selected for international collaborative R&D and for each concept a lead country has been identified. The reactor concepts and lead countries are:

(1) Gas cooled fast reactors (USA);
(2) Lead cooled fast reactors (Switzerland);
(3) Sodium cooled fast reactors (Japan);
(4) Supercritical water cooled reactors (Canada);
(5) Very high temperature reactors (France);
(6) Molten salt reactors (lead country yet to be decided).

TABLE 3. CONVENTIONAL AND ADVANCED FUELS FOR LWRs, PHWRs, LMFRs AND HTRs

Reactors	Conventional fuels	Advanced/Alternative fuel
Light Water Reactor (LWR): BWR, PWR & WWER		
Fuel	LEU ($^{235}U \leq 5\%$) as UO_2	LEU (^{235}U 5–10%) Mixed Uranium Plutonium Oxide (≤10% PuO_2)
Cladding	Zircaloy 2 (BWR) Zircaloy 4 (PWR) Zr-1% Nb (WWER)	Zr-Sn-Nb-Fe & Zr-Nb-O alloys
Burnup	20 000–30 000 MWD/t	High: up to 60 000 MWD/t Ultra High: up to 90 000 MWD/t
Pressurized Heavy Water Reactor (PHWR)		
Fuel	Natural UO_2	REU, SEU in the form of UO_2, $(Th,Pu)O_2$ & $Th,^{233}U)O_2$, containing up to 2% fissile material
Cladding	Zircaloy 4	Zircaloy 4
Burnup	6700 MWD/t	15 000–20 000 MWD/t
Liquid Metal-cooled Fast Breeder Reactor (LMFBR)		
Fuel	HEU in the form of UO_2 & $(U,Pu)O_2$ (≤25% Pu)	(U,Pu)C, (U,Pu)N & U-Pu-Zr (≤25% Pu) with/without minor activities
Cladding	Stainless Steel D-9	S.S. (HT-9 or Oxide dispersed)
Burnup	100 000 MWD/t	Up to 200 000 MWD/t
Breeding Ratio	≤1.2	Up to 1.5
High Temperature Gas Cooled Reactors (HTR)	Multi-layer (pyrolytical carbon & SiC-coated) Uranium Oxide fuel particles (TRISO) embedded in graphite	Multi-layer (pyrolytical carbon & ZrC coated) Uranium Oxide, Mixed Uranium Plutonium Oxide, Mixed Uranium Thorium Dicarbide, etc., embedded in graphite

There is much commonality in the overall approach and objectives of INPRO and GIF. Both programmes are complementary to each other and aim to promote international cooperation to ensure the sustainable development and economic competitiveness of nuclear energy, as well as to meet high standards in the safety and reliability of nuclear power plants, environment, waste management and proliferation resistance.

Figure 5 shows the INPRO schedule. The first two steps, namely the INPRO Methodology Development (1A) and Methodology Validation (1B-first part), have been completed and documented [5, 6].

Several INPRO members have already started assessments of INSs on a national or international basis. The following assessments are being implemented:

(a) A joint assessment based on a closed fuel cycle with fast reactors (China, France, India, Republic of Korea and Russian Federation, with Japan as an observer);
(b) An assessment of INSs based on high temperature reactors (India);
(c) A study on the transition from LWRs to Generation IV fast neutron systems (France);
(d) An assessment of INS options for a country with a small grid (Armenia);
(e) A holistic assessment of the DUPIC fuel cycle with respect to proliferation resistance (Republic of Korea).

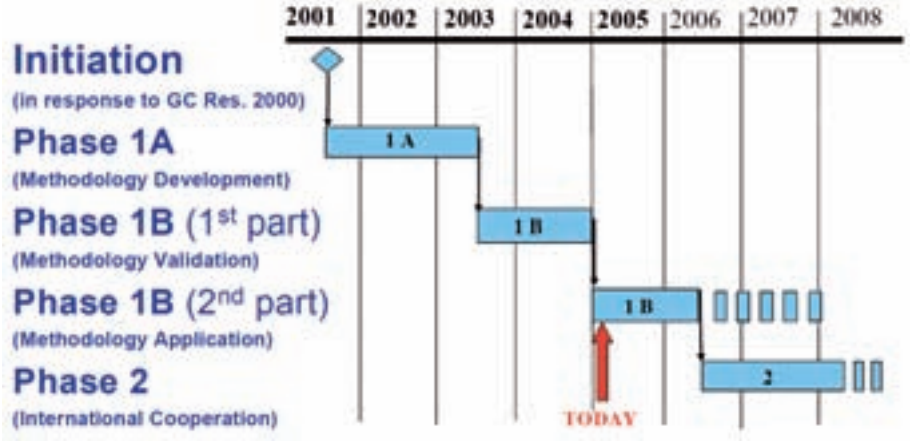

FIG. 5. The schedule for INPRO activities.

4. INNOVATIONS IN URANIUM EXPLORATION, MINING AND PRODUCTION

The world uranium demand in 2004 to fuel a nuclear power production of some 368 GW(e) was about 66 000 tU, but uranium production in the same year was some 40 250 tU. The gap between the annual demand for uranium and annual production has been met by secondary supply for more than a decade. This situation is likely to continue for at least a decade or two, after which the main stocks of secondary supply, namely the unused uranium inventory from past years (before 1990) and the highly enriched uranium (HEU) from nuclear warheads in the Russian Federation and the United States of America, will be likely to have been used up. Hence, the gap between the increasing annual demand for uranium, because of the expanding nuclear power programme worldwide, and annual uranium production has to be narrowed within the next two decades, when the secondary uranium supply will consist of meagre quantities of MOX, RepU and depleted uranium (DepU) tailings from enrichment plants.

Fortunately, the uranium industry is doing well again after a lull of nearly two decades. In the last three years, uranium exploration, mining and production activities have been significantly augmented in several countries. The uranium market is booming, with the price having nearly tripled in the last three years. However, innovations are needed in uranium exploration, mining, milling, purification, conversion, enrichment and fuel fabrication activities.

The first cycle of uranium exploration occurred in the 1960s and 1970s, and focused mainly on surface based prospecting techniques that have revealed only a little of the world's uranium potential. Since that time, nearly 80% of the uranium discovered has been in the Athabasca basin in Canada using deep exploration techniques. Innovative aerial and ground geophysical techniques need to be implemented in order to discover unconformity type deeply buried uranium deposits. With the use of new airborne electromagnetic (EM) equipment, capable of intercepting conductive materials at depths of up to 1 km, and improvements in the understanding of EM inversions, which promise to better integrate geophysical methods with geological models, and improved resolution from seismic methods, it should be possible to discover new uranium deposits in a cost effective way. A robust and successful international exploration effort is desirable. Likewise, the challenges of mining and milling can be met with the use of larger and better performing equipment, and radiometric ore scanning and sorting, which allows extraction of lower grade ores while reducing the barren waste rock. For increasing the productivity and ensuring low radiation exposure levels to mine personnel, direct reading

dosimeters (for gamma and alpha radiation), in combination with area radon progeny detectors, need to be introduced.

Unconventional uranium resources, particularly uranium bearing rock phosphates, were earlier proven to be commercially viable. Tapping uranium from this source will increase the resource base significantly. In milling, innovative high pressure filter technology has been efficient for solid–liquid separation. Full atmospheric leaching and underground in situ leaching (ISL) will enhance productivity and improve environmental protection. In recent years, smaller scale, but low cost, ISL mining has been gaining popularity, and is the sole method for extraction of uranium in Kazakhstan and Uzbekistan. In situ leaching is also being introduced in Australia and China, as well as accounting for nearly all of US uranium production in recent years. In the area of uranium enrichment, gaseous diffusion technology (GDT) is being phased out and replaced by ultra-centrifugation technology (UCT), because the latter is less energy intensive and hence more economic.

Argentina has recently developed a novel and low cost uranium enrichment technology, SIGMA (Separación Isotópica Gaseosa por Métodos Avanzados), based on gas diffusion, and the use of a much lower number of compressors has been developed with the objective of reducing capital cost and energy consumption [7]. Only 100 modules of serial SIGMA cascades could lead to 5% ^{235}U enrichment, which is adequate for nuclear power reactors. Thus, SIGMA has built-in proliferation resistant features because, with such a limited number of cascades, it is not possible to produce ^{235}U enrichment beyond 20%. In addition, SIGMA also has novel safeguard features and an on-line materials accounting system. The laser isotope separation technologies based on atomic vapour laser isotope separation (AVLIS) and molecular laser isotope separation (MLIS) have the inherent advantage of selectivity but have yet to be commercialized.

5. INNOVATIONS IN FUEL FABRICATION

Figure 6 summarizes the powder pellet and sol-gel processes for fabricating oxide, carbide and nitride fuels. UO_2, PuO_2 and ThO_2 are isostructural (FCC-NaCl type), completely solid soluble and have very similar thermodynamic and thermophysical properties. Hence, the manufacturing processes of uranium, thorium and plutonium oxide, as well as of mixed oxide fuel pellets, are similar. The same is true for carbide and nitride fuels. At present, the powder pellet route is universally followed for the manufacture of natural and LEU (<5% ^{235}U) uranium oxide and MOX fuel pellets. For nuclear power plants, the LEU oxide powder is mostly manufactured by the integrated

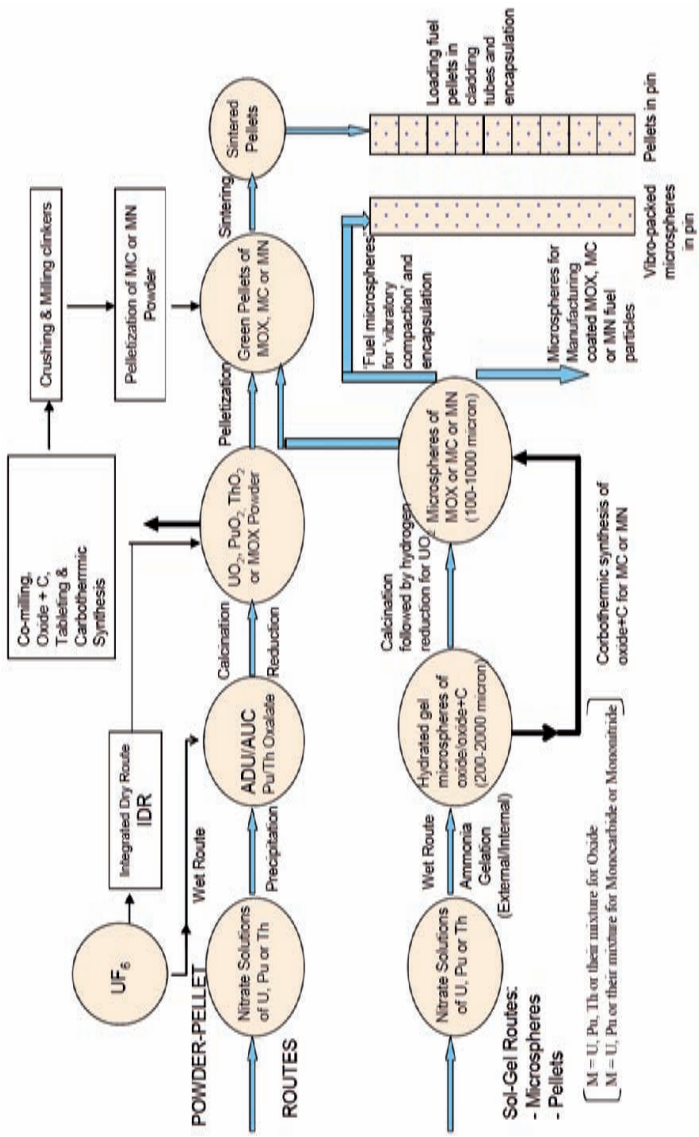

FIG. 6. Different methods for manufacturing MOX, MC and MN powder, microspheres and pellets.

dry route (IDR), although the wet route based on ammonium diuranate (ADU) and ammonium uranium carbonate (AUC) is also being used in some countries. Natural uranium oxide powder is mostly produced by the ADU process. The ThO_2 and PuO_2 powders are mostly produced by the wet chemical route, starting with the nitrate solutions, and involving oxalate precipitation and calcination. The oxide powders are subjected to co-milling, followed by cold pelletization and high temperature sintering in pure hydrogen or in an argon/nitrogen and hydrogen mixture to obtain pellets of controlled density, microstructure and oxygen to metal ratio. Dopants (silica, titania, niobia or alumina or their mixtures) may be added for obtaining pellets of large grain size and pore formers (methyl cellulose, polyvinyl alcohol or polyethylene glycol or U_3O_8) are added to obtain pellets of controlled porosity. The dopants and pore formers are admixed to the powder during the milling operation. Industrial scale manufacturing of MOX fuel is carried out in Belgium, France, Japan and the UK [8]. Pilot scale production facilities are in operation in India and the Russian Federation. In Belgium and France the micronized masterblend (MIMAS) process, and in the UK the short binderless (SBR) route are followed. In Japan, uranium and plutonium nitrate solutions are subjected to microwave denitration to obtain the mixed oxide powder.

The non-oxide ceramic fuels, namely, mixed uranium plutonium monocarbide and mononitride, are also manufactured by the powder pellet route, starting with UO_2 and PuO_2 powders. The oxide powders are co-milled with carbon, pelletized and subjected to carbothermic synthesis in vacuum and flowing nitrogen for obtaining MC and MN clinkers, respectively. The clinkers are crushed, ground, pelletized and sintered. Since MC and MN are highly susceptible to oxidation and hydrolysis and are pyrophoric in powder form, the entire manufacturing process is carried out under a high purity inert-cover gas in gloveboxes. In addition, close control of carbon and nitrogen stoichiometry is needed to produce nearly single phase MC and MN with minimum higher carbide and higher nitride second phase.

The powder pellet route involves generation and handling of fine powders and is associated with the problem of radioactive dust hazard. In most cases, the fine powders are not suitable for remote fabrication because they are not free-flowing and need to be granulated to obtain suitable ‘press-feed’ material. The alternative sol-gel process, based on the ammonia external/internal gelation process, produces dust-free and free-flowing hydrated gel microspheres of oxides or mixed oxides of uranium, plutonium and thorium, starting with nitrate solutions of the heavy metals. For preparation of mixed carbide and nitride microspheres, carbon black powder in predetermined amounts is added to the sol prior to gelation. The hydrated oxide plus carbon

microspheres are subjected to carbothermic synthesis in vacuum and flowing nitrogen for preparation of carbide and nitride microspheres, respectively.

Figure 7 shows a schematic diagram of the sol-gel and sol-gel microsphere pelletization (SGMP) processes. The SGMP route is dust-free and is suitable for remote and automated fabrication of highly radioactive Pu and ^{233}U bearing oxide, carbide and nitride fuel pellets. The SGMP process has been combined with low temperature oxidative sintering (LTS) for fabrication of high density UO_2 and (U, Pu)O_2 fuel pellets. The SGMP–LTS route has been successfully utilized for the manufacture and irradiation testing of UO_2 fuel bundles in PHWR-220 reactors in India.

6. INNOVATIONS IN THE BACK END OF THE FUEL CYCLE

National programmes on the back end of the nuclear fuel cycle were presented in the IAEA Technical Working Group Meeting on Nuclear Fuel Cycle Options and Spent Fuel Management (TWGNFCO), and the proceedings have been issued in the form of working materials. Some of these national programmes are summarized in this section.

6.1. United States of America

The Advanced Fuel Cycle Initiative (AFCI) programme in the USA is integrated with their GIF and Nuclear Hydrogen Initiative (NHI), in order to have a meaningful impact on the future of nuclear energy in the USA. The AFCI mission is "to develop and demonstrate technologies that enable the transition to a stable, long term, environmentally, economically and politically acceptable advanced fuel cycle." The AFCI strategy anticipates the transition over the next few decades from the current fuel cycle to one that is progressively more sustainable. The primary strategy for enhancing transuranic management during the once-through fuel cycle is the introduction of high burnup and ultrahigh burnup fuels that will reduce the transuranics produced per unit of generated energy.

Limited recycling will begin with the introduction of spent fuel treatment and fuel fabrication facilities (c. 2025). Limited recycling permits the recycling of transuranics through LWR and advanced LWR plants, and possibly through Generation IV very high temperature thermal reactors (VHTR), if deployed. The transitional recycling phase will begin with the introduction of the first Generation IV fast reactors (c. 2040). Transitional recycling will allow for the consumption of transuranics in a reactor fleet composed of thermal and fast spectrum reactors. Finally, sustained recycling is the end evolution point of the

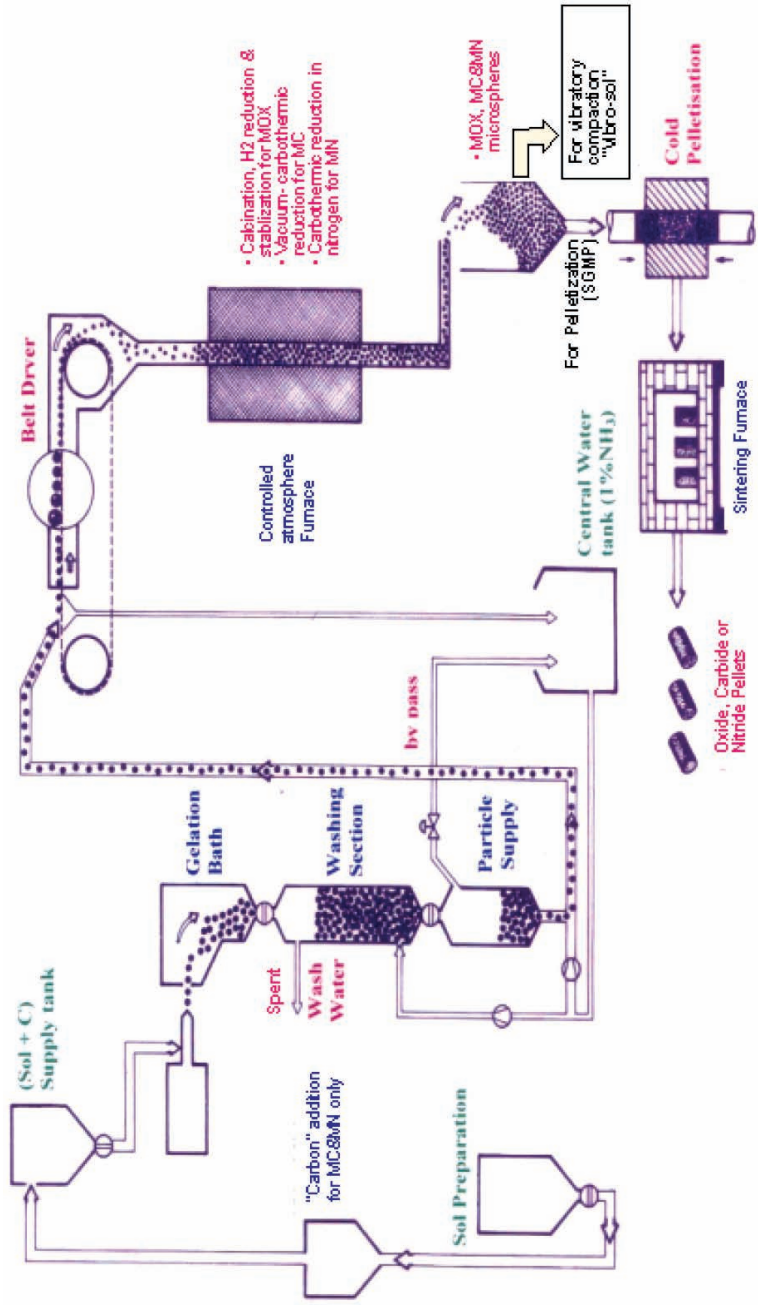

FIG. 7. Schematic diagram of the SGMP and vibro-sol processes (the ammonia external gelation process for preparation of hydrated microspheres).

fuel cycle, obtained when the reactor fleet consists of a high percentage of fast spectrum reactors. The sustained recycle option will allow not only for the consumption of transuranics during energy generation but also for the generation of new fuel through the transmutation of natural, depleted or recycled uranium.

Earlier, from the mid-1970s to the mid-1980s, extensive R&D was carried out in the USA on LMFR fuels, namely conventional mixed uranium plutonium oxide and advanced fuels, i.e. mixed uranium plutonium monocarbide and mononitride as well as U–Pu–Zr metallic fuels, in the Los Alamos and Argonne National Laboratories. Extensive irradiation testing has been successfully carried out in the Experimental Breeder Reactor-II (EBR-II) and the Fast Flux Test Facility (FFTF) up to high burnups (100 000–200 000 MW·d/t). An inherently safe integrated fast reactor (IFR) using U–Pu–Zr metallic fuel and a closed fuel cycle with co-located pyroelectrolytic reprocessing and refabrication facilities has been demonstrated. The metallic fuel pins were manufactured by vacuum induction melting followed by injection casting in high purity silica tubes. The results of LMFR fuel development in the USA and elsewhere have been very well documented [9–15].

6.2. France

In France, LWRs will continue to play a dominant role in the generation of electricity during most of the current century. The lives of the operating Generation II PWRs are being extended beyond 40 years (possibly up to 60 years) and will be progressively replaced by Generation III/Generation III+ PWRs, starting from 2015, which will operate over most of the twenty-first century. The transition from PWRs to Generation IV fast reactors is likely to start about 2040. Generation II PWRs were initially licensed to use LEU oxide fuel. Subsequently, they were slightly adapted to accept up to a 30% MOX fuel loading in the core.

A more efficient and economically acceptable plutonium management scheme is needed for PWRs until fast reactors are commercialized. Accordingly, European Pressurized Water Reactors (EPRs (Generation III)) have been designed to allow the loading of 100% MOX assemblies. The near term fuel programme aims at mono-recycling of plutonium and high burnup fuels (60 GW·d/t), keeping safety margins the same as for current UO_2 fuelled PWRs. The following three MOX based fuel concepts are under examination and are likely to be deployed commercially during the period 2015–2025:

(1) APA and Duplex assemblies consisting of a heterogeneous arrangement of PuO_2 in an inert matrix (e.g. CeO_2) surrounded by UO_2 rods.

(2) A cermet fuel has also been envisaged in which the PuO_2 particles are dispersed in a zircaloy metal matrix.
(3) CORAIL, using a heterogeneous arrangement of MOX rods (PuO_2 in a depleted UO_2 matrix) and UO_2 rods in a fuel assembly. The MOX rods could contain 0.25% ^{235}U and as high as 11% Pu.

France is pursuing R&D activities in the following three areas for future nuclear energy systems:

(1) Sodium cooled fast reactors (SFRs) and gas cooled fast reactors (GFRs) involving new processes for spent fuel treatment and recycling. Development activities on the group actinide extraction (GANEX) process, MA fuels, ferritic stainless steel cladding and wrapper materials, and multiple recycling of plutonium with MAs are under way. The objectives are the transition from plutonium mono-recycling in PWRs to integral and multiple plutonium and MA recycling in Generation IV fast neutron systems.
(2) Co-generation of electricity and nuclear hydrogen as well as supply of very high temperature process heat to industry by employment of VHTRs and water splitting processes.
(3) Innovations in LWRs.

France is collaborating with Japan and the Russian Federation in the utilization of the BOR-60, BN-600, Joyo and Monju fast reactors, and with the USA and EURATOM in fuel cycle activities.

6.3. Japan

In Japan, the main focus during the twenty-first century will be on energy security and a 'recycling society'. Hence, R&D efforts are under way to develop fast reactors with closed fuel cycles, aiming at breeding, reprocessing and multiple recycling of plutonium, and burning of MAs to minimize the radiotoxicity in the waste for disposal. All possible types of fast reactors, namely, SFRs, lead (Pb–Bi) cooled fast reactors (LFRs), GFRs (He cooled) and water cooled fast reactors, are being studied. Innovative fuel cycle technologies will be developed by 2015, after which the demonstration phase and economic studies will take another 15 years. Commercial fast reactors are likely to be deployed from 2030 onwards.

In the area of fuel fabrication, the powder pellet route is being developed for both oxide and non-oxide fuels with and without MAs. The advanced PUREX aqueous route is being developed, involving microwave denitration

and co-extraction of plutonium with MAs. The alternative 'sphere-pac' process is being considered in combination with the PUREX process for dust-free production of ceramic nuclear fuels, adapting the ammonia gelation process. In the area of non-aqueous reprocessing, the combined pyroelectrorefining, melting and injection casting process of ANL (USA) has been adapted for manufacturing U–Pu–Zr metallic fuels for SFRs and, in collaboration with RIAR (Russian Federation), oxide electrowinning followed by crushing and vibropacking of MOX fuel. An in-cell remote and automated fabrication facility has been set up for fabrication of pellet fuel. The power level of the experimental Joyo reactor is being progressively increased from 50 to 140 MW·t. Joyo and the demonstration type 280 MW(e) Monju reactors will be extensively utilized for fuel and structural materials development and international collaboration. The SFR reactors are also likely to be used for generation of hydrogen by thermochemical electrolysis, at around 500–550°C.

The HTR at Oarai, using multilayer coated TRISO fuel particles in prismatic blocks, has demonstrated a coolant temperature of 850–950°C. Thus, production of hydrogen by the alternative IS route by using the high temperature process heat would be feasible.

A feasibility study is under way to develop 'proliferation resistant fuel' under the Protected Plutonium Production (PPP) project. In the PPP scheme, it is proposed to admix ≈1% MA oxide with LEU oxide fuel in LWRs in order to have a significant quantity of ^{238}Pu in spent uranium fuel as a result of neutron capture of ^{237}Np and alpha decay of ^{242}Cm. Plutonium-238 has high neutron radiation and decay heat, thereby making the fuel inherently proliferation resistant. Likewise, it is proposed to add up to 5% of MAs in depleted uranium and thorium blankets in fast reactors, in order to make the irradiated blankets proliferation resistant.

6.4. Russian Federation

In the Russian Federation, it is planned to put new water cooled thermal reactors into operation at the rate of ≈1 GW(e) per year. The existing scheme for spent fuel management in the Russian Federation allows safe storage and a deferred nuclear fuel cycle plan for SNFs from WWER-1000 to RBMK-1000 and for reprocessing of SNF from BN-600 and WWER-440 as well as from research and transportation reactors. The Krasnoyarsk Mining and Chemical Combine (MCC) provides a service for the centralized intermediate storage of WWER-1000 SNF from Russian, Ukrainian and Bulgarian reactors. A dry storage facility for RBMK fuel is under construction at the site. The RT-1 complex of the Mayak Production Association has carried out radiochemical reprocessing of SNF from WWER-440 reactors operated in the Russian

Federation, Ukraine and Bulgaria, from fast BN-600 reactors with uranium cores, and from research and transportation reactors. All RMBK-1000 fuel is currently stored in water in on-site pools.

The first Russian plant for radiochemical reprocessing of SNF from civil NPPs — the RT-1 complex at the Mayak Production Association — was based on a military facility with an inadequate scheme of waste management. Three production lines of the RT-1 complex with the PUREX process allow reprocessing of different types of SNF. At present, the amount of fuel sent for reprocessing (250 t/a) at the RT-1 complex is considerably below the design output (400 t/a). Current reprocessing technology based on the PUREX process not only provides purification of Pu and RepU but also produces a spectrum of different wastes to be stored, discharged and disposed of. Any aqueous partitioning generates additional liquid wastes and fractions to be conditioned. Some new approaches may be developed for SNF treatment by the Compact Processing Scheme (CPS), aiming at waste minimization and simplification of the resulting nuclear materials management:

(a) Advanced dry technologies such as thermomechanical treatments provide an efficient separation of metal (zirconium and steel) components.
(b) The uranium matrix voloxidation process provides tritium and iodine removal from fuel.
(c) Uranium extraction with liquid CO_2 or a similar process should provide for removal of 90–95% RepU for re-enrichment, storage or permanent disposal.
(d) The resultant residue is a 'hot' composition of 5% U + Pu + MA + FP, and it should amount to only 10% of the initial fuel mass. It should be conditioned for storage in a safe and reliable way as long as is needed, for future use in commercial fast reactors or Generation IV reactor systems.
(e) The pyroprocess treatment with molten salts, described in detail in another paper at this meeting, may be easily applied in future to the resulting composition, with the aim of fresh U–Pu fuel production.

Implementation of the Compact Processing Scheme (CPS), shown in Fig. 8, at the MCC site near Krasnoyarsk, offers several advantages such as availability of underground space for long term disposal of metal wastes, near surface storage of RepU, a deep well injection site for tritium removal, and permanent isolation and provision for special dry storage for long term isolation of the hot plutonium and MA waste.

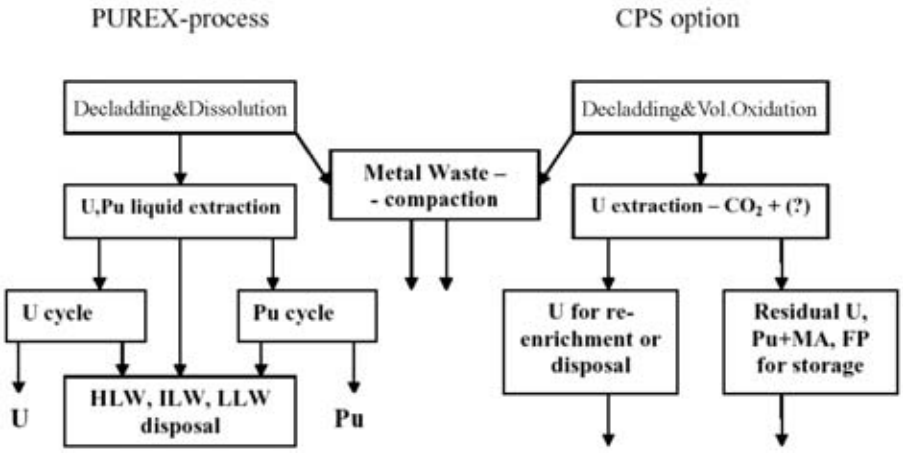

FIG. 8. Simplified comparison of the PUREX and CPS processes.

Commercial fast reactors are likely to be introduced in the Russian Federation from 2030 onwards. The performance of the largest operating fast reactor in the world, BN-600 MW(e), has been most satisfactory for more than two decades. The experimental fast reactors BOR-60 and BN-600 have been extensively utilized for R&D of fast reactor fuels. Construction activities for the BN-800 MW(e) fast reactor are expected to start soon.

Research and development activities are under way for both SFRs and LFRs and their fuel cycles, athough the focus is more on SFRs.

As part of LMFR fuel cycle activities, the Russian Federation has developed a novel dry pyroelectrolytic reprocessing of oxide fuel, using a chlorinator– electrolyser. The major process steps are:

(a) Dissolution of SNF in molten salt;
(b) Recovery of plutonium dioxide or mixed uranium plutonium oxide from the melt;
(c) Processing of the cathode deposit and production of granulated fuel;
(d) Vibropacking of fuel particles in cladding tube and encapsulation. Vibropacked MOX fuel manufactured by this route has been used in BOR-60 and successfully tested in BN-600. A semi-industrial vibropacked MOX fuel plant with an annual capacity of manufacturing some 50 BN-600 fuel assemblies will be in operation from 2005.

Simultaneously, the BREAST-300 LMFR design has been completed and the design of BREAST-1200 is under way. The essential features of BREAST reactors are the following:

(a) Use of lead as coolant and reflector;
(b) Use of high density and high thermal conductivity (U, Pu)N as driver fuel;
(c) No uranium blanket, and recycling of plutonium only as a mixture with uranium;
(d) Co-location of reactor, reprocessing and fuel refabrication facilities;
(e) Generation of lower amounts of MA and radwaste.

6.5. Fuel cycle initiatives in India

India has modest uranium resources but vast thorium ones and has no access to the international uranium market. India is pursuing an indigenous and self-reliant three stage nuclear power programme, as shown in Fig. 9, involving 'closed' uranium and thorium fuel cycles.

The first stage, consisting of natural uranium oxide fuelled PHWRs, is under way. The plutonium by-product from the PHWRs is being recovered by the PUREX process and will be subjected to multiple recycling in LMFRs in the second stage in the form of mixed uranium–plutonium ceramic or metallic fuel to breed ^{233}U from thorium blankets. In the third stage, a self-sustaining ^{232}Th–^{233}U fuel cycle is planned in thermal breeders. As a first step to the LMFR programme, India could miss out a step and use a hitherto untried

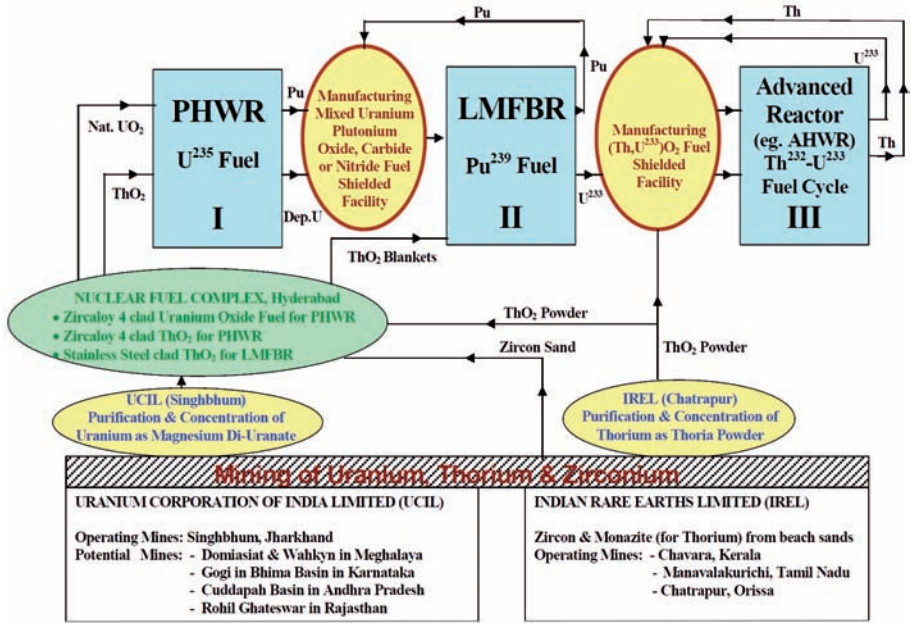

FIG. 9. Three stage nuclear power programme in India involving closed fuel cycles.

plutonium-rich mixed uranium–plutonium monocarbide driver fuel in the 40 MW(th) Fast Breeder Test Reactor (FBTR). The FBTR has been in operation for nearly two decades with this carbide fuel, which has reached a burnup of 140 000 MW·d/t without failure. The spent mixed carbide fuel has been successfully reprocessed on a laboratory scale. Mixed oxide fuel has been successfully utilized in the two BWRs and in one of the operating PHWR 220 MW(e) units. A prototype fast breeder reactor (PFBR-500) is under construction. The reference fuel for PFBR-500 is MOX, but metallic or nitride fuels are likely to be used in subsequent cores.

Significant progress has been made in the R&D on thorium based fuels and the associated fuel cycle. A process flowsheet based on the classical powder pellet route and advanced SGMP and impregnation techniques has been developed. Irradiated thorium pins from research reactors and PHWRs have been reprocessed by the Thorex process and the ^{233}U recovered has been used as fuel in the KAMINI research reactor. Design and development activity for an advanced heavy water reactor of 300 MW(e) (AHWR-300) is under way. The AHWR is a heavy water moderated, light boiling water cooled, vertical, pressure tube type reactor using ThO_2–UO_2(^{233}U) and ThO_2–PuO_2 as driver fuels.

6.6. Republic of Korea

The Republic of Korea (ROK) has 16 PWRs and four CANDUs currently in commercial operation with a total capacity of 17.7 GW(e). The ROK has an ambitious plan to expand the total installed nuclear capacity up to 26.1 GW(e) by the year 2015. The cumulative amount of spent fuel generated by December 2004 in the ROK reached 7286 tU. Projections indicate that approximately 20 000 tU of spent fuel will be accumulated by 2020. An AFR interim storage facility is scheduled for completion by the year 2016, and will be run at an initial capacity of about 2000 tU. The ROK is constructing an experimental fast reactor (KALIMER) as part of their plutonium utilization programme and continues to perform R&D activities in search of an optimal option for spent fuel management with the long term perspective of nuclear power utilization.

Extensive studies on innovative fuel cycle technologies are being carried out at the Korea Atomic Energy Research Institute (KAERI) in order to find an effective solution for the back end of the fuel cycle. Advanced fuel cycle development in the ROK is focused on the following three major areas:

(1) The DUPIC (direct use of spent PWR fuel in CANDU reactors) programme;

(2) The Advanced Spent Fuel Conditioning (ACP) programme to reduce the volume, heat load and toxicity of spent fuel;
(3) The Pyrometallurgical Partitioning and Transmutation (P&T) programme for P&T of long lived radionuclides, as shown in Fig. 10.

The DUPIC fuel cycle technology aims to directly fabricate CANDU fuel from spent PWR fuel through a dry thermal/mechanical process without any separation of stable fission products and transuranic elements. The existence of residual fission products in fresh DUPIC fuel is a distinctive feature of DUPIC fuel. Owing to the high radioactivity of the fuel material, all the manufacturing processes should be performed remotely in a highly shielded facility. Spent PWR fuel is first disassembled, and then the cladding is removed mechanically to retrieve the fuel material. The irradiated fuel material is treated by repeated cycles of oxidation and reduction, called the oxidation and reduction of oxide fuel (OREOX) process, to make the irradiated fuel material re-sinterable. Once the re-sinterable powder feedstock has been prepared, the remaining fabrication steps are similar to the conventional CANDU fuel fabrication process, i.e. powder treatment, compaction, sintering, end cap welding and bundle assembly. Since all the fabrication processes should be performed in a shielded facility throughout the whole process, a designated remote fabrication laboratory, called the DUPIC Fuel Development Facility (DFDF), was established in 2000 by refurbishment of an existing hot cell at KAERI.

A major DUPIC fuel fabrication campaign was started for fabrication of DUPIC fuel pellets and elements for the performance evaluation through irradiation tests at the HANARO research reactor. KAERI has successfully fabricated several DUPIC fuel elements in a remote manner at DFDF, and the performance evaluation through the irradiation tests at HANARO and the post-irradiation examination (PIE) at the Irradiated Material Examination Facility (IMEF) and at the Post-Irradiation Examination Facility (PIEF) is under way. The DUPIC technology is internationally acknowledged as a typical proliferation resistant fuel cycle technology.

6.7. China

The policies and strategies in China for the back end of the fuel cycle involve a closed fuel cycle, construction of a reprocessing pilot plant initially and construction of commercial reprocessing plants later in order to have plutonium for multiple recycling in fast breeder reactors (FBRs). At present, China has nine operating nuclear power plants (seven PWRs and two PHWRs)

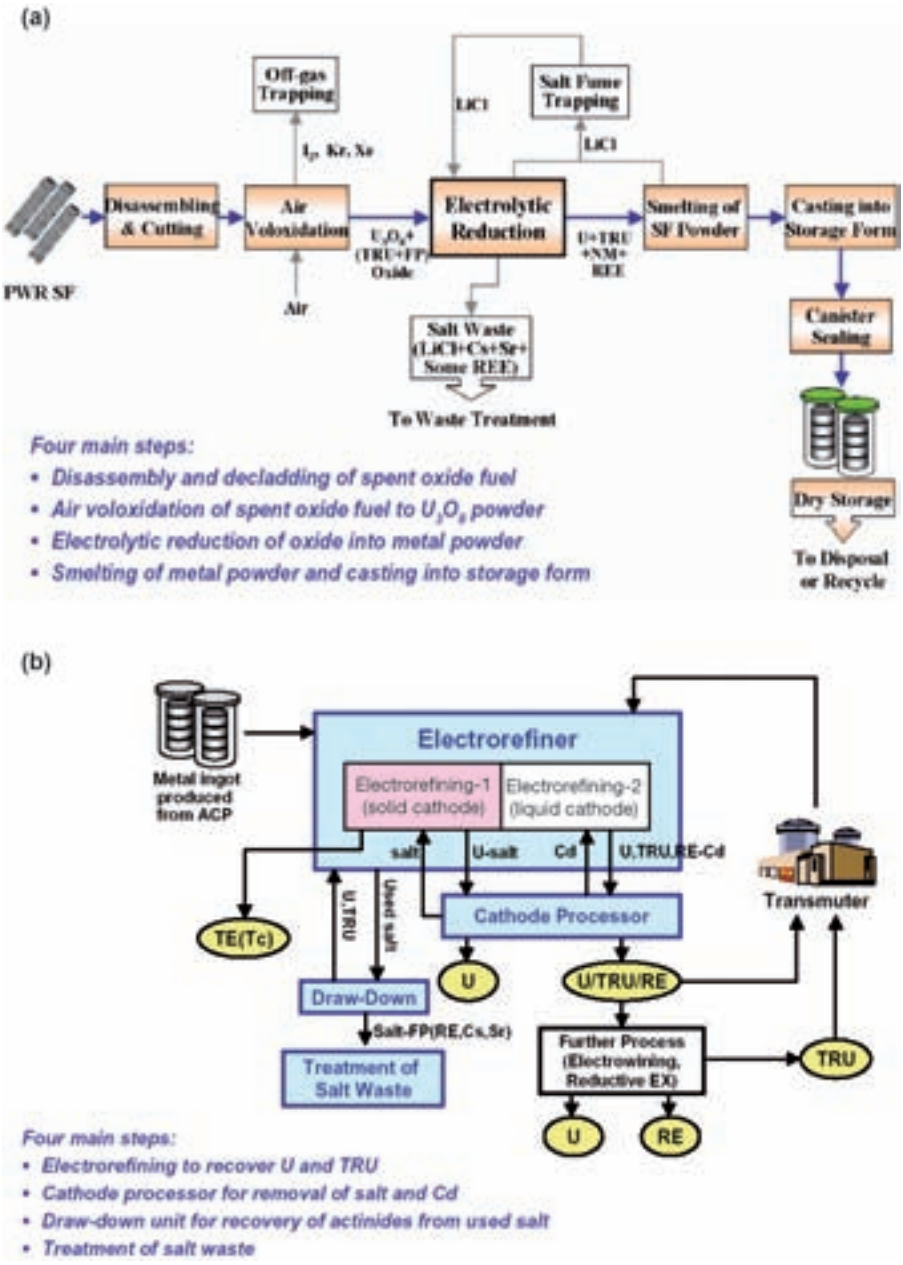

FIG. 10. Major steps in spent fuel reprocessing involving (a) pyropartitioning and (b) electrorefining.

with an installed capacity of ≈7000 MW(e), and there are plans to have ≈40 000 MW(e) of nuclear power by 2020, mainly from PWRs. The 25 MW(e) sodium cooled China Experimental Fast Reactor (CEFR), which is under construction, will be commissioned in 2008 and will use MOX fuel. At present, China has plans to construct only sodium cooled fast reactors, starting with a 600 MW(e) China Prototype Fast Reactor (CPFR) in 2020, followed by 300 MW(e) China Modular Fast Reactors (CMFRs) if needed, mainly for actinide burning, a 1000–1500 MW(e) China Demonstration Fast Reactor (CDFR) by 2025 and a series of 1000–1500 MW(e) China Commercial Fast Reactors (CCFRs), starting during 2030–2035. The CPFR would use MOX fuel initially, followed by metallic fuel. CDFR and CMFR will also use metallic fuel.

6.8. European Union activities on the fuel cycle

The European Technical Working Group (ETWG) on accelerator driven systems (ADSs) has played a coordinating role at the European level for P&T and ADS development as a route for waste management and final disposal (FD), and for closure of the back end of the fuel cycle. There is a need for first step demonstration of ADSs at the international level and for coordinated R&D at the European level with support from the European Commission for implementing the following strategies recommended by the ETWG:

(a) UO_2 in LWRs + ADSs + FD;
(b) UO_2 in LWRs + MOX in LWRs + ADSs + FD;
(c) UO_2 in LWRs + MOX in LWRs + (MOX + MAs) in FBRs + ADSs + FD.

Both critical and subcritical reactors are potential candidates for dedicated transmutation systems. Subcriticality is favourable and allows safe operation with a maximum load of MAs per unit.

The policy chosen is the closed fuel cycle. The strategy is to develop techniques in a reprocessing pilot plant, and to construct a commercial reprocessing plant based on the development of the pilot plant.

6.9. Multilateral approaches to the nuclear fuel cycle

Two primary factors dominate all assessments of multilateral nuclear approaches (MNAs), namely:

(1) Assurance of non-proliferation;
(2) Assurance of supply and services.

In 2004, the Director General of the IAEA appointed an expert group to:

(a) Identify and analyse issues and options;
(b) Provide an overview of the policy, legal, security, economic, institutional and technological incentives and disincentives for cooperation in multilateral arrangements for the front and back ends of the nuclear fuel cycle;
(c) Provide a brief review of the historical and current experiences, and analyses relating to multilateral fuel cycle arrangements.

The experts recommended the following five approaches [16]:

(1) Reinforcing existing commercial market mechanisms on a case by case basis through long term contracts and transparent arrangements with suppliers, with government backing. Examples would be fuel leasing and fuel take-back offers, commercial offers to store and dispose of spent fuel, as well as commercial fuel banks.
(2) Developing and implementing international supply guarantees with IAEA participation. Different models should be investigated, notably with the IAEA as a guarantor of service supplies, for example as administrator of a fuel bank.
(3) Promoting voluntary conversion of existing facilities to MNAs, and pursuing them as confidence building measures, with the participation of nuclear non-proliferation treaty (NPT) non-nuclear-weapon States, NPT nuclear weapon States and non-NPT States.
(4) Creating, through voluntary agreements and contracts, multinational, and in particular regional, MNAs for new facilities based on joint ownership, drawing rights or co-management for front and back end nuclear facilities, such as those for uranium enrichment, fuel reprocessing, and disposal and storage of spent fuel (and combinations thereof). Integrated nuclear power parks would also serve this objective.
(5) The scenario for a further expansion of nuclear energy around the world might call for the development of a nuclear fuel cycle with stronger multilateral arrangements — by region or by continent — and for broader cooperation, involving the IAEA and the international community.

7. CONCLUDING REMARKS

(a) The uranium conventional and unconventional (in phosphates) resources in the world are adequate to meet the fuel requirements for the different growth scenarios foreseen for nuclear power programmes up to 2050 and

beyond. However, innovative exploration, mining, milling and production techniques for uranium concentrates should be developed and introduced as soon as possible in order to close the gap between the 'uranium in the ground' and the 'yellow cake (uranium concentrate) in the can'.

(b) The diversity of nuclear reactor systems and fuel cycle options shows an extraordinary range. The scenarios foreseen are:

(i) Immediate reprocessing and plutonium recycling in LWRs and PHWRs as MOX fuel, to utilize the fissile worth of plutonium and to minimize the buildup of americium;

(ii) Immediate reprocessing and safe storage of plutonium for multiple recycling in FBRs along with MAs in future;

(iii) Temporary storage and postponement of reprocessing to the future, to recover plutonium for use in FBRs when they mature;

(iv) Direct disposal or long term storage followed by disposal;

(v) Development of ADSs for spent fuel and waste management.

Priority has been given to LMFRs with a closed fuel cycle in both INPRO and GIF. Sodium cooled fast reactors with metallic U–Pu–Zr fuel, with multiple recycling of plutonium along with MAs and involving pyro-electrolytic reprocessing of spent fuel are emerging as one of the most innovative fuel cycle options. In the area of high temperature gas cooled reactors, the focus is on utilizing high temperature nuclear process heat for production of hydrogen, either by the IS process or by high temperature electrolysis. Multilayered coated fuel particles with ZrC as one of the layers appear to be promising.

(c) The major studies that are likely to take place in the near future on the back end of the fuel cycle are assessment and intercomparison of advanced dry, aqueous and pyroelectrolytic reduction methods of reprocessing and how they fit into advanced methods of fuel fabrication. The advanced PUREX aqueous process (UREX, microwave denitration or GANEX) delivering nitrate solutions of uranium, plutonium and MAs as end products could be combined with ammonia external/internal gelation processes to have dust-free and free-flowing hydrated gel-microspheres, which could be subjected to remote and automated fabrication to obtain ceramic oxide, carbide and nitride fuel microspheres and then could be further processed to obtain pellet pins, vibropacked pins or multilayer coated fuel particles embedded in graphite. Likewise, remote pyroelectrolytic reprocessing, in shielded hot cells, could be tailored to obtain metallic or oxide deposits at the cathode, which could be further processed remotely for vacuum melting and injection casting into metallic fuel pins or crushed and vibropacked to produce oxide pins, respectively.

The assessment of P&T issues has to be carried out with respect to the benefits on their long term radiological impacts on the final disposal of the ultimate wastes and their environmental effects, proliferation resistance, additional costs and additional doses to workers.

(d) Thorium is three times more abundant than uranium in the earth's crust. The thorium fuel cycle offers the advantage of an additional fissile material resource, inherent proliferation resistance, lower MA formation, better in-core performance and better spent fuel long term storage options. However, the databases on thorium deposits, thorium fuels and fuel cycles are scanty. Exploration of thorium resources needs to be augmented. Research and development on thorium fuel fabrication, property evaluation and irradiation testing and reprocessing are needed before considering the thorium fuel cycle for commercial exploitation.

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THE DUPIC TECHNOLOGY CONTRIBUTION TO FISSILE MANAGEMENT

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Abstract

The paper addresses the contribution of DUPIC (direct use of spent pressurized water reactor (PWR) fuel in CANDU) technology to fissile management. For this, the fissile material inventory of the DUPIC fuel cycle is compared with those of other once-through fuel cycles such as the PWR and CANDU (Canadian deuterium uranium) cycles. A comparison is also made for the total toxicity generated from the DUPIC fuel cycle, for the effectiveness of waste management. From a fissile material analysis of fuel cycles, the total plutonium inventory, as well as the minor actinides, generated during 1 GW(e)·a is shown to be least in the DUPIC cycle. This means that the DUPIC option has some benefits for fissile material management. In addition, the DUPIC option has the lowest fissile plutonium content, which could be a measure of proliferation resistance. On the whole, the CANDU once-through cycle has the largest fissile, as well as gross, plutonium content on the basis of 1 GW(e)·a. From a radiotoxicity analysis of fuel cycles, the toxicity of the DUPIC option based on 1 GW(e)·a is much smaller than those of other fuel cycle options. It is shown that the value is just about half the order of magnitude of other fuel cycles until the radiotoxicity has decayed to a level below that of the initial ore. This means that the DUPIC option could have an indirect benefit on the environmental effects of long term spent fuel disposal.

1. INTRODUCTION

Recently, fissile material management has been more and more important in the area of the nuclear industry. There are three main reasons:

(1) The first reason is that the uranium resources of the earth are limited. Current demand for natural uranium amounts to 60 000 t/a. Stockpiles and known uranium resources represent some 70 years of consumption by present reactors. The actual conventional uranium resources are estimated to be about 15 million tonnes, representing some 250 years of present consumption [1].

(2) The second reason is concern for safe disposal, owing to the long half-lives of fissile materials. For more than four decades, nuclear experts have sought to develop practical methods for safe management and disposal of highly radioactive wastes containing long lived radionuclides. Proponents of reprocessing argue that burying plutonium-containing spent fuel creates an unacceptable long term hazard, since the half-life of the most important plutonium isotope, ^{239}Pu, is 24 000 years. Some of the solutions proposed have focused on separating the hazardous long lived radionuclide components of the waste and transmuting them by neutron bombardment to form nuclides that would be either stable or radioactive with a much shorter half-life. However, such systems would greatly increase the cost of nuclear power.
(3) The third reason is concern about nuclear proliferation, in which the fissile material can be diverted and then misused for nuclear weapons programmes.

In these respects, this study examines whether the DUPIC (direct use of spent pressurized water reactor (PWR) fuel in CANDU) fuel cycle will make fissile material management more effective compared with other once-through cycles such as the PWR and CANDU (Canadian deuterium uranium) cycles, as well as their mixing cycle.

To make a reasonable comparison, all amounts calculated in this study are expressed on the basis of one gigawatt (GW(e)) year of reactor operation, as well as tonnes of heavy metal (tHM) of spent fuels or high level wastes. For this, fuel cycle scenarios are first set up and reactor parameters and their fuel characteristics are assumed appropriately. Using these characteristics, fuel material flows are estimated based on 1 GW(e)·a. Then the various fuel properties are assessed by use of the ORIGEN computer code [2] and compared with each other.

Reference fuel cycle models and approaches for estimating the amount of fissile material are given in Section 2. The reference reactors and fuels and the material flow of each fuel cycle are described in Sections 3 and 4, respectively. The fissile material and toxicity index are described in Section 5.

2. NUCLEAR FUEL CYCLES

2.1. Concept of the DUPIC fuel cycle

The DUPIC fuel cycle concept [3] is to reuse the PWR spent fuel in the CANDU reactor without the need for the reprocessing operations typically

required for the recycling fuel cycle. Since 1991, the Korea Atomic Energy Research Institute (KAERI), Atomic Energy of Canada Limited (AECL) and the Los Alamos National Laboratory (LANL) in the United States of America, under the observation of the IAEA, have been engaged in DUPIC fuel cycle development as a practical exercise to develop a spent fuel recycling process that expands resource utilization and reduces waste accumulation with an enhanced proliferation resistance to those of the typical recycling options.

The residual fissile content of PWR spent fuel is ≈1.5 wt% for a discharge burnup of 35 000 MW·d/t. Because the CANDU reactor was originally designed to use natural uranium dioxide fuel with 0.71 wt% ^{235}U, it is possible to directly use the PWR spent fuel in the CANDU reactor without reprocessing, even though the spent fuel contains fission products and transuranic elements. The discharge burnup of DUPIC fuel is expected to be ≈15 000 MW·d/t, which is about twice that of natural uranium fuel [4, 5].

The DUPIC fuel fabrication process was developed on the basis of the oxidation and reduction of oxide fuel (OREOX) process to produce re-sinterable fuel stock material from PWR spent fuel. As shown in Fig. 1, after the PWR spent fuel assembly has been disassembled, the cladding is mechanically taken away in order to obtain the spent fuel materials from the PWR spent fuel rod. The key process in DUPIC fuel fabrication is conversion of the spent fuel material into the re-sinterable powder form through the OREOX process. Once the powder feedstock has been prepared, the DUPIC fuel pellet is fabricated following the conventional processes such as pre-compaction, granulation, final compaction and sintering. The fuel element welding seal is performed by a laser, which is installed in the hot cell through an optical fibre. Because there is no process to separate the fission products from the PWR spent fuel, all the fuel fabrication processes are remotely conducted in concrete hot cells to shield the high radiation level.

In order to experimentally verify the performance of DUPIC fuel, a series of irradiation tests have been performed in the HANARO research reactor of KAERI and the NRU research reactor of AECL. Recent results of post-irradiation examination showed that the performance of DUPIC fuel was similar to that of uranium dioxide fuel irradiated under the same conditions. The only difference was that the DUPIC fuel experienced more pellet centre micro-structural changes and a slightly higher fission gas release.

The DUPIC fuel cycle has a number of intrinsic features that enhance its proliferation resistance. For example, unirradiated directly usable weapons grade material is not produced and the fuel remains highly radioactive throughout the process. Another major factor that maintains the radioactive nature of the DUPIC fuel cycle is the OREOX process, which is not capable of producing separated plutonium and is not readily modified to do so. Owing to

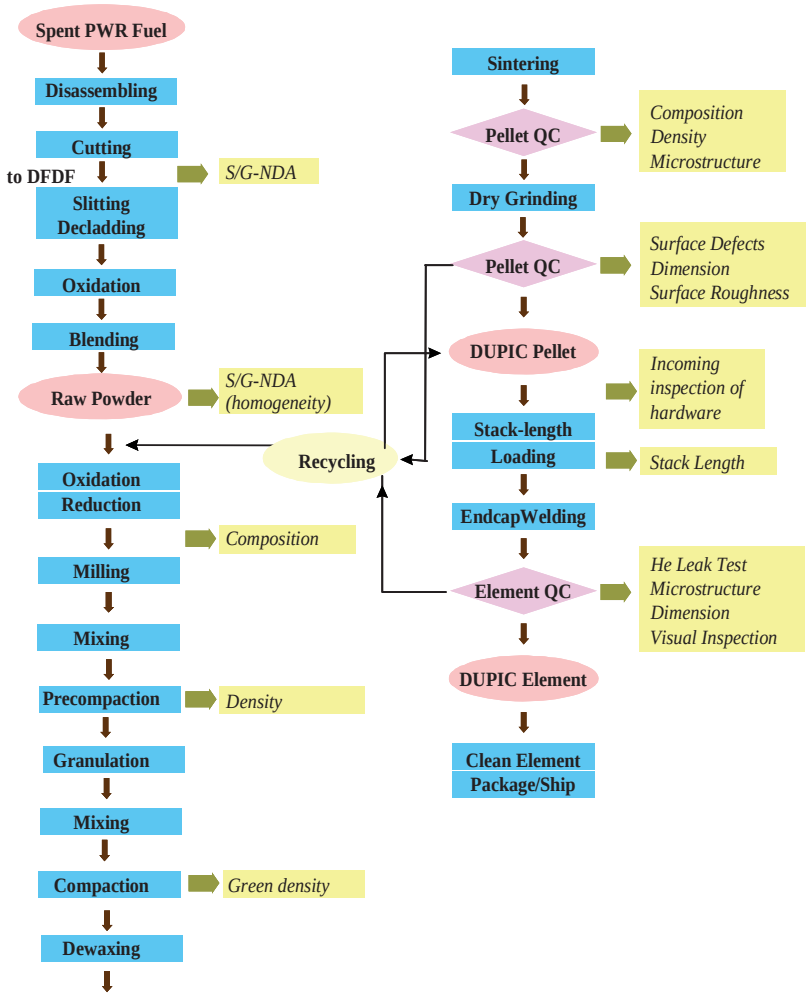


FIG. 1. Fabrication flow of DUPIC fuel.

the radioactive nature of the fuel material, all the steps in the fuel cycle should be undertaken remotely with substantial shielding, which provides another intrinsic feature that contributes to the proliferation resistance of the fuel cycle.

2.2. Alternative fuel cycles for the comparison

Figure 2 shows the fuel cycle options considered in this study and their steps or components consisting of fuel cycles. The first cycle is the DUPIC fuel

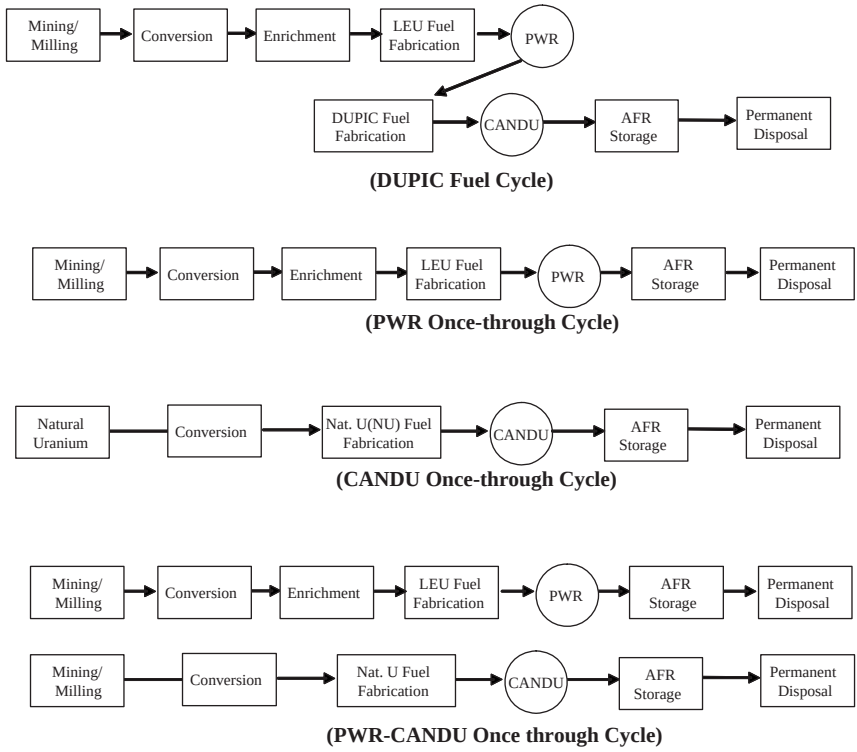


FIG. 2. The nuclear fuel cycle options considered in this study.

cycle, in which PWRs are linked to a CANDU reactor. The second cycle is low enriched uranium (LEU) in PWRs in once-through mode. The third cycle is natural uranium in CANDU in once-through mode. The fourth cycle is PWR fuel and CANDU fuel in once-through mode with a reactor grid equivalent to the DUPIC fuel cycle.

In the DUPIC fuel cycle, spent PWR fuel is directly refabricated into CANDU fuel to be burnt again in CANDU reactors before being disposed of permanently. On the other hand, for once-through fuel, all spent fuel generated from both PWR and CANDU reactors is disposed of. As shown in Fig. 2, the front end fuel cycle components for PWRs were established to be the same for both fuel cycles. For the DUPIC fuel cycle, however, several services such as DUPIC fuel fabrication are included but the front end fuel cycle components for CANDU are not needed.

3. REFERENCE REACTORS AND FUELS

For the material flow of each fuel cycle, reference PWR and CANDU reactors have to be chosen first, and their fuel characteristics (e.g., initial enrichment and discharge burnup) need to be defined reasonably. For practical purposes, a 950 MW(e) PWR and a 713 MW(e) CANDU reactor, which are now operating in the Republic of Korea, are taken as the reference reactor types. The characteristic parameters of the reference reactor systems that are used as input data for determining the fuel material balance are summarized in Table 1.

In this table, the amount of fuel loaded per reactor is estimated on the basis of reactor parameters such as:

$$\text{Fuel loading per core} = \frac{P \times 100}{\varepsilon \times SH} \tag{1}$$

where P, SH and ε are the electric power (MW(e)), the specific power (MW(th)/tHM) and efficiency (%), respectively.

Table 2 shows the reference fuels for each fuel cycle. It is assumed that LEU PWR fuels are burnt up to 35 000 MW·d/tU, although recent PWR fuels have had burnups mostly over 40 000 MW·d/tU fuel. The reason is that 35 000 MW·d/tU with initial enrichment of 3.5% ^{235}U was chosen as the reference PWR fuel in DUPIC fuel cycle development in the Republic of Korea [4, 5].

TABLE 1. CHARACTERISTICS OF REFERENCE REACTORS

Reactor parameter	PWR	CANDU
Electric power (MW(e))	950	713
Thermal efficiency (%)	34	33
Thermal power (MW(th))	2794	2161
Specific power (MW(th)/tU)	40.2	25.5
Load factor	0.8	0.9
Cycle length (full power day)	290	–
Number of fuel assemblies or bundles per core	157	4560
Number of batches for PWRs	3	–
Loading per core (tU)	69.5	84.7

In order to calculate how much plutonium is in PWR spent fuel burnt with 35 000 MW·d/tU, we have used the ORIGEN2 computer code. We have used the ORIGEN 2.1 PC version, which is able to use cross-sections that resulted from the processing of existing compilations such as ENDF/B-V.

In a CANDU reactor, the discharge burnup of natural CANDU fuel is assumed to be 7500 MW·d/tHM, and the discharge burnup of DUPIC fuel is assumed to be 15 400 MW·d/tHM, which is a reference fuel in DUPIC fuel development [4, 5].

The annual requirement of nuclear fuels is calculated on the basis of fuel burnup and other parameters such as

$$\text{Annual requirement} = \frac{P \times 365 \times C}{\varepsilon \times BU} \tag{2}$$

where C and BU are the capacity factor (%) and burnup (MW·d/tHM), respectively. The annual requirements per unit are translated into annual requirements based on 1 GW(e)·a, as shown in the last row of Table 2.

TABLE 2. CHARACTERISTICS OF REFERENCE REACTORS AND FUELS

Item	Characteristic parameters		
	PWRs with LEU fuel	CANDUs with natural uranium fuel	CANDUs with DUPIC fuel
Reactor:			
Loading per core (tU)	69.5	84.7	84.7
Annual fuel requirement (tU)	23.31	94.63	46.09
Fuel:			
Initial enrichment	3.5%	Natural U	PWR spent fuel
Number of fuel rods per assembly	264	37	43
Discharge burnup (MW·d/kgHM)	35	7.5	15.4
Normalization of fuel:			
Required amount of fuel for 1 GW(e)·a (tU or tHM)	24.54	132.73	64.64

4. MATERIAL FLOW FOR FUEL CYCLE OPTIONS

For the DUPIC fuel cycle, the equilibrium core ratio between PWRs and CANDU reactors has to be known so that all PWR spent fuels can be made into DUPIC fuel. It is possible to calculate with the annual requirement of PWRs and CANDUs with DUPIC fuel. The equilibrium core ratio between PWRs and a CANDU reactor can be calculated as follows:

$$\text{Equilibrium core ratio } (R_C) = \frac{M_{DUPIC}(1+L_{DUPIC})}{M_{PWR}} \tag{3}$$

where M_{DUPIC}, M_{PWR} and L_{DUPIC} are the annual requirement for DUPIC, the annual requirements of PWRs and the loss rate in DUPIC fabrication plants, respectively. In this study, the loss rate in the DUPIC fabrication plant is assumed to be 1%. Since M_{DUPIC} and M_{PWR} are 46.09 tHM and 23.31 tU, respectively, the equilibrium core ratio is 1.997.

The fraction of electricity generated by PWRs to that by PWRs and CANDUs for 1 GW(e)·a can be calculated as follows:

$$\text{Electricity generation fraction of PWRs} = \frac{P_{PWR}R_C}{P_{PWR}R_C + P_{CANDU}} \tag{4}$$

where P_{PWR} and P_{CANDU} are electric power generated by PWRs and CANDUs, respectively. Therfore, the percentages of PWR and CANDU generation will be 72.68 and 27.32%, respectively. These percentages of electricity generation will be applied to both the PWR-CANDU and DUPIC fuel cycles.

In this study, it is assumed that the loss factors are 0.5% for conversion, and 1% for all types of fuel fabrication and for reprocessing plants. The enrichment amount in separative work units (SWUs) is calculated as follows:

$$\text{SWU} = M_pV_p + M_tV_t - M_fV_f \tag{5}$$

where M_p is the mass of uranium to be charged into the fuel fabrication facility, M_f is the mass of uranium fed into the enrichment plant (and discharged from conversion plants), M_t is the mass of uranium discharged from the enrichment plant and

$$V_x = (2e_x - 1)\ln\frac{e_x}{1-e_x} \tag{6}$$

in which the subscript x is for f, p or t. Here

e_p = fraction of ^{235}U in the uranium feed (e.g. 3.5 wt%)

e_t = fraction of ^{235}U in the tails (e.g. 0.25 wt%)

and

e_f = fraction of ^{235}U of uranium to be charged into the enrichment plant (e.g. 0.711 wt%)

Then

$$M_f = M_p \frac{e_p - e_t}{e_f - e_t} \tag{7}$$

and

$$M_t = M_f - M_p \tag{8}$$

From the above equations, if M_p and three fractions of the ^{235}U in the enrichment plant are known, the number of SWUs as well as M_f and M_t (depleted uranium) can be calculated.

The requirement for natural uranium resources is converted to that for uranium (U_3O_8) by the following formulation:

$$M_n = M_R \frac{e_p - e_t}{e_f - e_t}(1+l_1)\frac{W_{U_3O_8}}{W_{U_3}}(1+l_2) \tag{9}$$

where M_n is the mass of uranium (U_3O_8) to feed, M_R is the mass of uranium charged to the reactor, and $W_{U_3O_8}/W_{U_3}$ is a conversion factor for the weight of uranium. That is, $W_{U_3O_8}$ is the atomic weight of U_3O_8 and W_{U_3} is the atomic weight of U_3. Here, l_1 and l_2 are the process loss rates of conversion and fuel fabrication, respectively.

The results of the material balance analyses, which were calculated from Eqns (1)–(9) with reference reactor parameters (given in Table 1) and their fuel characteristics (given in Table 2), are shown in Fig. 3. All values were expressed on the basis of 1 GW(e)·a for all fuel cycle options.

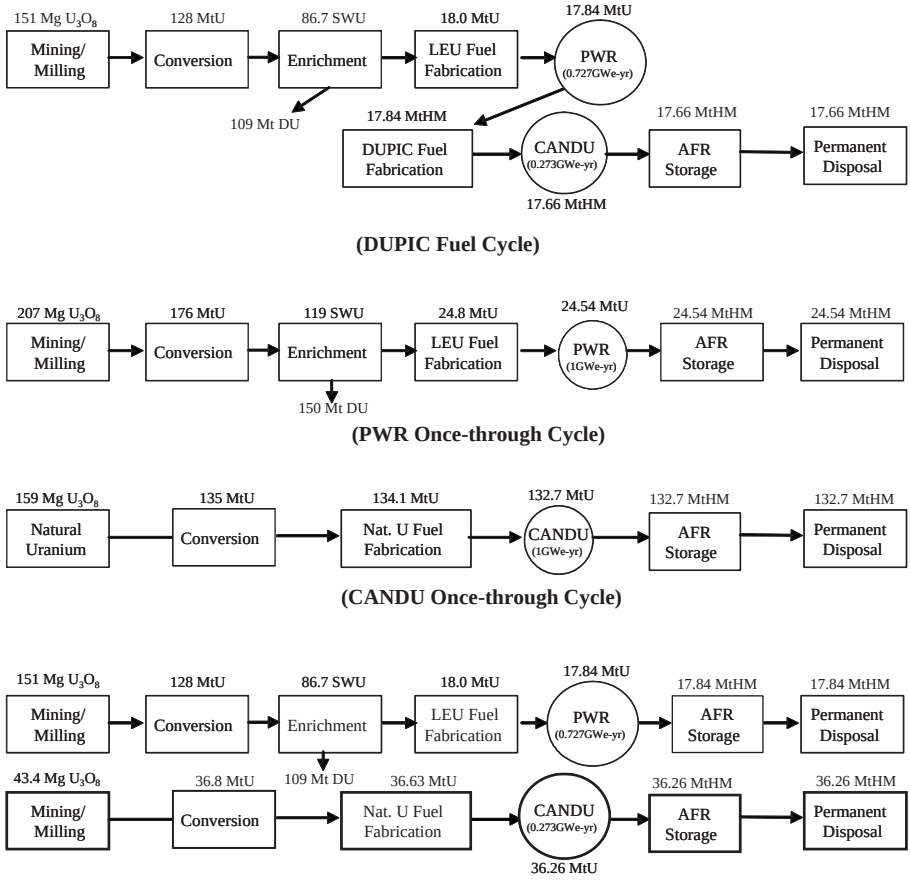
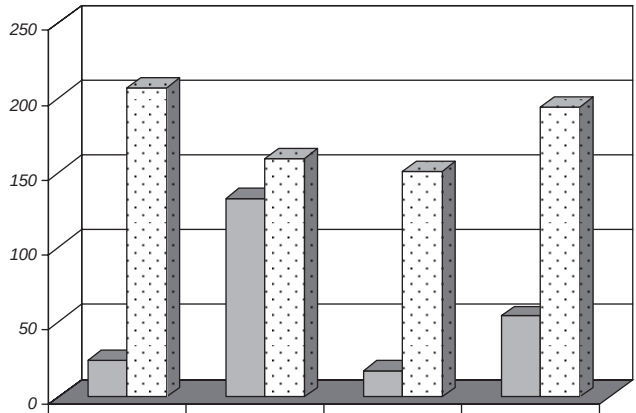


FIG. 3. Fuel cycle options and their material flows on the basis of 1 GW(e)·a. In this figure, Mt stands for tonne.

From the material flow, we can find interesting values of natural uranium resource requirements and spent fuel arisings for each fuel cycle, as shown in Fig. 4. These indicate that the DUPIC fuel cycle with PWR and CANDU reactor requires only 151 Mg U_3O_8 of natural uranium, which is just for PWR fuel with an enrichment of 3.5 wt% ^{235}U. On the other hand, 151 Mg U_3O_8 of natural uranium for PWR fuel and 43 Mg U_3O_8 of natural uranium for CANDU fuel are required for the PWR-CANDU once-through cycle. This means that the DUPIC option has a saving of ≈22% uranium resources based on weight (t), compared with the PWR-CANDU once-through cycle.



	PWR Cycle	CANDU Cycle	DUPIC Cycle	PWR-CANDU Cycle
Spent Fuel Arisings(MtHM)	24.5	132.7	17.7	54.1
Nat. Uranium (Mg U3O8)	207	159	151	194

FIG. 4. Natural uranium requirement and spent fuel arisings on the basis of one GW(e)·a. In this figure, Mt stands for tonne.

In addition, the amount of spent fuel discharged annually from the DUPIC fuel cycle generates only ≈18 tHM/GW(e)·a, while the once-through fuel cycle (PWR-CANDU) generates ≈54 tHM/GW(e)·a. The DUPIC fuel cycle generates ≈67% less spent fuel to be disposed of than the once-through fuel cycle (PWR-CANDU). Comparing the PWR and CANDU fuel cycles, it is indicated that the PWR fuel cycle requires the largest natural uranium resources (211 Mg U_3O_8/GW(e)·a) and the CANDU fuel cycle generates the largest amount of spent fuel (≈133 tHM/GW(e)·a).

5. FISSILE MATERIAL MANAGEMENT

5.1. Amount of fissile material

In this section, we will evaluate transuranium isotope compositions including the plutonium contained in spent fuels generated in the alternative fuel cycles. The compositions are assessed on a tonne basis, and then these values are translated into a 1 GW(e)·a basis. Plutonium and its content in spent fuels is especially important because it could be a measure of proliferation resistance.

In order to calculate how much plutonium there is in spent fuels, we have used the ORIGEN 2 burnup simulation code. Code users of ORIGEN 2 must

supply the input characteristics to the program. For reactor simulation of the DUPIC fuel in CANDU reactors, the isotope contents of PWR spent fuel, which are also calculated using the ORIGEN code, have been used. All actinides, transuranic isotopes and 140 fission products contained in PWR spent fuels were inputted into the code. Some fission products removed during the DUPIC fuel fabrication process have been excluded. It is assumed that volatile isotopes during the oxidation and reduction process are removed and that semivolatile isotopes such as caesium and ruthenium are removed during the sintering process working at 1700°C. The removed fission products referred to in a report by the Korea Atomic Research Institute (KAERI) [6] are described in Table 3.

Figure 5 shows a comparison of the major actinides contained in three different spent fuels after a cooling time of ten years. The weight per cent per heavy metal as well as the mass per initial uranium or heavy metal are shown in the figure. Spent PWR fuel still contains about 0.84 wt% ^{235}U and 0.88 wt% Pu, and about 68.4% of the plutonium is in the form of fissile isotopes (^{239}Pu and ^{241}Pu). The DUPIC and CANDU spent fuels contain only about 0.22 wt% ^{235}U, but DUPIC spent fuel contains about double the plutonium content, 0.84 wt%, of CANDU spent fuel. The minor actinide content of DUPIC spent fuel is a little more than that of PWR spent fuel.

Using the material flows of Fig. 3, the total plutonium embedded in spent fuels can be calculated on the basis of 1 GW(e)·a. As shown in Fig. 6, the total amount of plutonium generated during 1 GW(e)·a is the most (≈535 gPu/GW(e)·a) in the CANDU fuel cycle. The DUPIC option contains ≈141 gPu/GW(e)·a and also has the lowest fissile plutonium content, which could be another measure of proliferation resistance. On the whole, the CANDU fuel cycle has the largest amounts of fissile plutonium as well as of gross plutonium, which are negative points for resistance to nuclear proliferation.

TABLE 3. RELEASE RATE DURING THE DUPIC FUEL FABRICATION PROCESS

Isotope	Release rate (%)	Isotope	Release rate (%)
H	100	C	100
Kr	100	Ru	100
Cd	75	Te	75
Ir	75	I	100
Xe	100	Cs	100

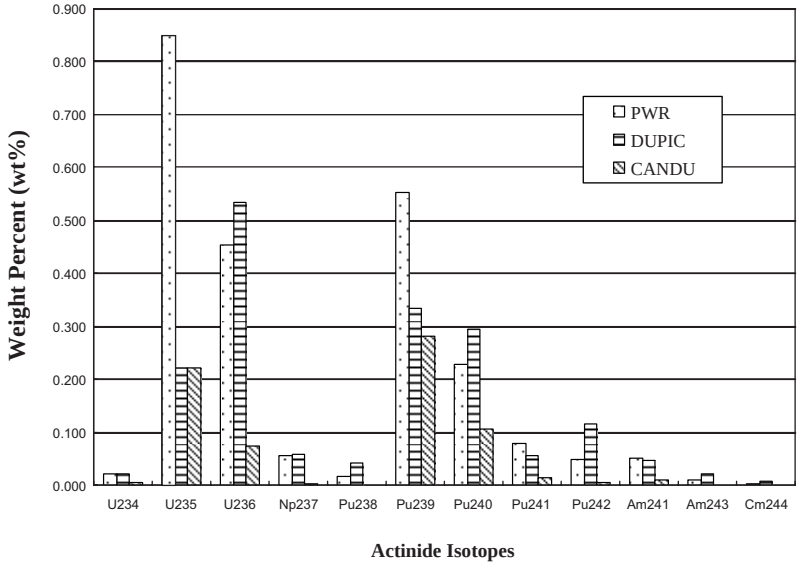


FIG. 5. Major actinide isotopes embedded in spent fuels for the PWR, DUPIC and CANDU cases.

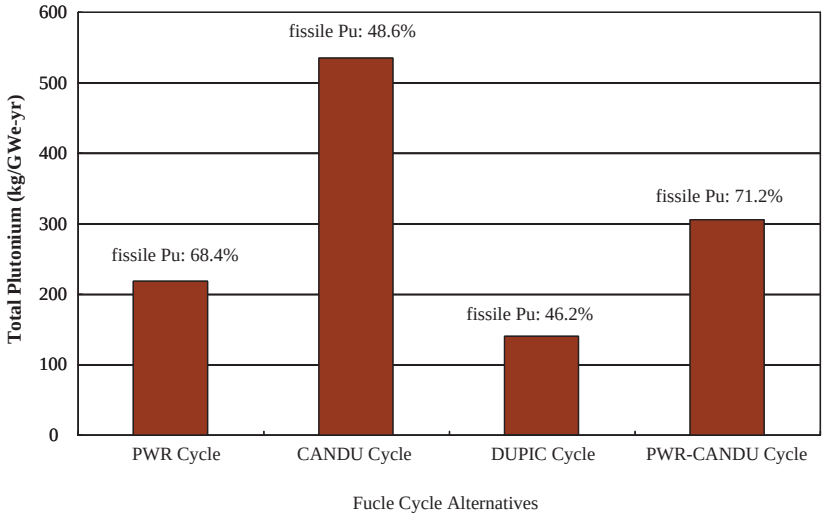


FIG. 6. Plutonium inventories of the various fuel cycles.

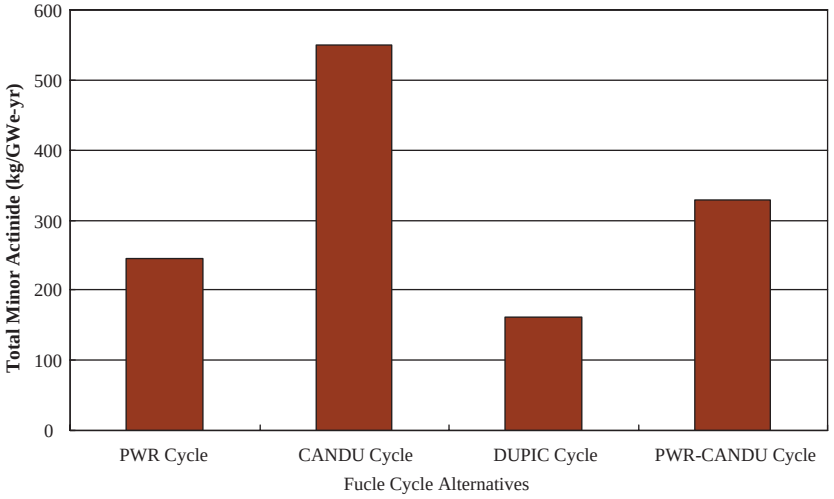


FIG. 7. Minor actinide inventories of the various fuel cycles.

The minor actinides (Np, Am and Cm) embedded in spent fuels were also calculated on the basis of 1 GW(e)·a. As shown in Fig. 7, the DUPIC cycle has the least amount of minor actinides, about half that of the PWR-CANDU once-through cycle.

5.2. Radioactive toxicity

The activity is only a crude measure of the importance of waste management in irradiated fuel and in radioactive wastes. A more meaningful measure of potential biological hazard must also include the sensitivity of humans to inhalation or ingestion of these radionuclides. For this purpose, a toxicity index has been used as follows [7]:

$$\text{Toxicity index} = \sum_i \frac{\lambda_i N_i}{C_{ik}} \tag{10}$$

where λ_i is the radioactive decay constant for nuclide i, N_i is the number of atoms of nuclide i and C_{ik} is the maximum permissible concentration limit for nuclide i in medium k (i.e. air or water).

The toxicity index is the volume of air or water with which the mixture of radionuclides must be diluted so that breathing the air or drinking the water will result in the accumulation of radiation dose at a rate no greater than the

dose limit. Because the actinide and most fission products are non-volatile and because the wastes are expected to be geologically isolated, ingestion toxicity is probably a more important measure than inhalation toxicity. Therefore, only ingestion toxicity indices are examined in this section.

The ingestion toxicity indices for fission products and actinides of PWR and DUPIC spent fuels are compared in Figs 8 and 9, respectively, as a function of storage time. It is apparent that the relatively high toxicity, low *C*, of bone seeking ^{90}Sr makes this nuclide more important than any other fission product in terms of potential ingestion toxicity during the first few hundred years after discharge from the reactor. Next, the long lived thyroid seeking ^{129}I is potentially the most important of the fission products. During the first 300 years, the total toxicity index is governed by the fission products, mainly ^{90}Sr. It is controlled by ^{241}Am and ^{243}Am from about 300 years to about 2000 years, followed successively by ^{239}Pu and ^{240}Pu from 2000 years to 80 000 years. Subsequently, the most important radionuclide is ^{226}Ra, which is formed from the decay of ^{234}U, ^{238}Pu, ^{242m}Am, ^{242}Cm and ^{238}U.

Comparing the ingestion toxicity of DUPIC spent fuel with PWR spent fuel, the DUPIC case is a little higher than the PWR case for actinide toxicity but is a little lower than the PWR case for fission product toxicity. Therefore, for the short term (about 200 years), when toxicity is governed by fission products, the DUPIC case is a little lower than the PWR case. On the other hand, for the long term, when toxicity is governed by actinides, the PWR case is a little lower than the DUPIC case.

Figure 10 compares the ingestion toxicities based on tonnes of heavy metal of spent fuel and high level wastes (HLWs). The ingestion index of DUPIC spent fuel is lower than that of PWR spent fuel during the period governed by fission products, but a little higher or similar during the period governed by actinides.

The ingestion toxicity index for five fuel cycle options is compared, on the basis of 1 GW(e)·a, in Fig. 11. Although the true hazards of radioactive wastes are not measured by these toxicity indices, some perspective can be obtained by comparing the total ingestion toxicity index with the similar toxicity index for the ore used to fuel the reactor to generate these wastes. The ore toxicity is due mainly to ^{226}Ra, which is in secular equilibrium. The ingestion index in Fig. 11 is also compared with the toxicity of uranium ore mined for 1 GW(e)·a of reactor operation. Since the ore toxicity is in secular equilibrium in the ^{238}U decay chain, we can calculate the ore toxicity. At secular equilibrium, the activities of ^{226}Ra and ^{238}U are the same. Therefore, the activity is:

$$\frac{(0.9927^{238}\,\mathrm{U/U})(10^{6}\,\mathrm{g/t})(6.0225\times10^{23}/\mathrm{g-atom})(0.693)}{(238\mathrm{gU/g-atom})(4.51\times10^{9}\,\mathrm{a})(3.154\times10^{7}\,\mathrm{s/a})[3.7\times10^{10}/(\mathrm{Ci.s})]} = 0.33\ \mathrm{Ci/tU}$$

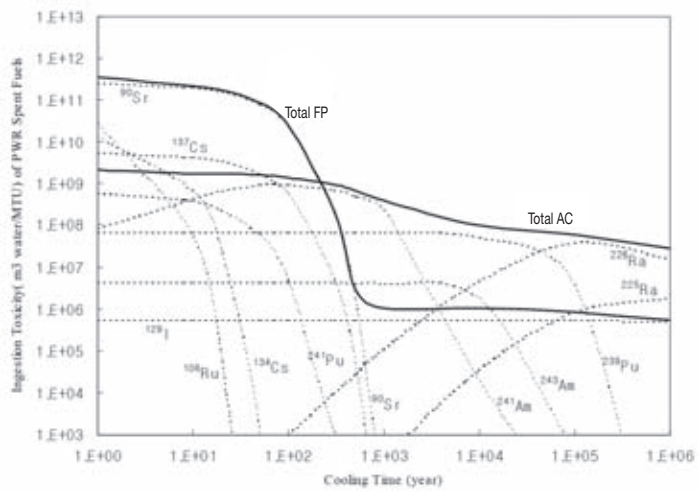


FIG. 8. Contributions of various isotopes to the long term ingestion hazard index of PWR spent fuels. In this figure, Mt stands for tonne.

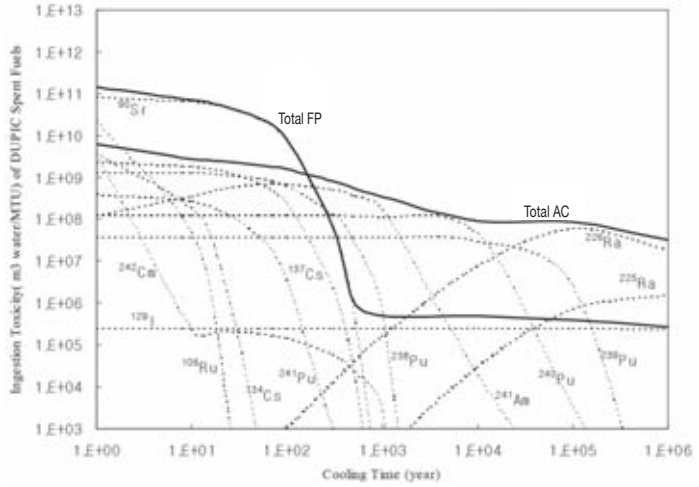


FIG. 9. Contributions of various isotopes to the long term ingestion hazard index of DUPIC spent fuels. In this figure, Mt stands for tonne.

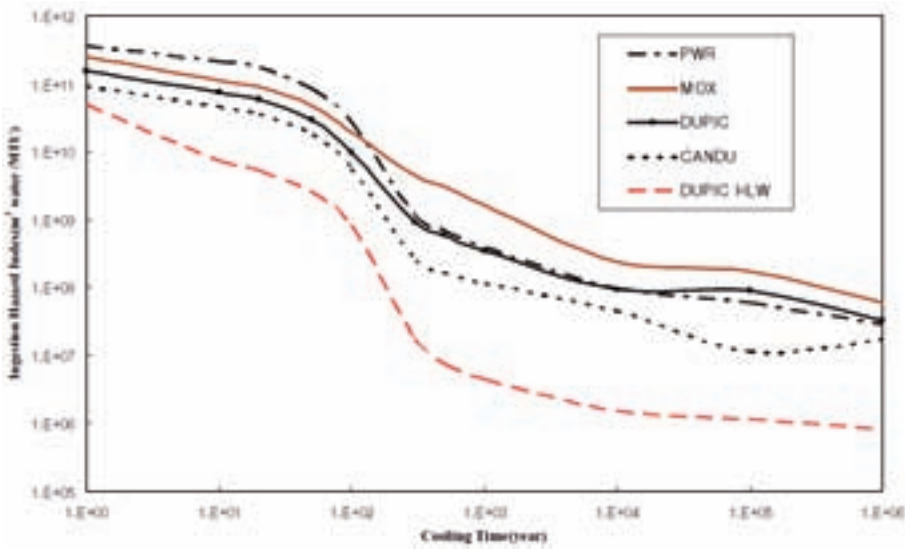


FIG. 10. Ingestion hazard indices for various spent fuels and HLWs. In this figure MT stands for tonne.

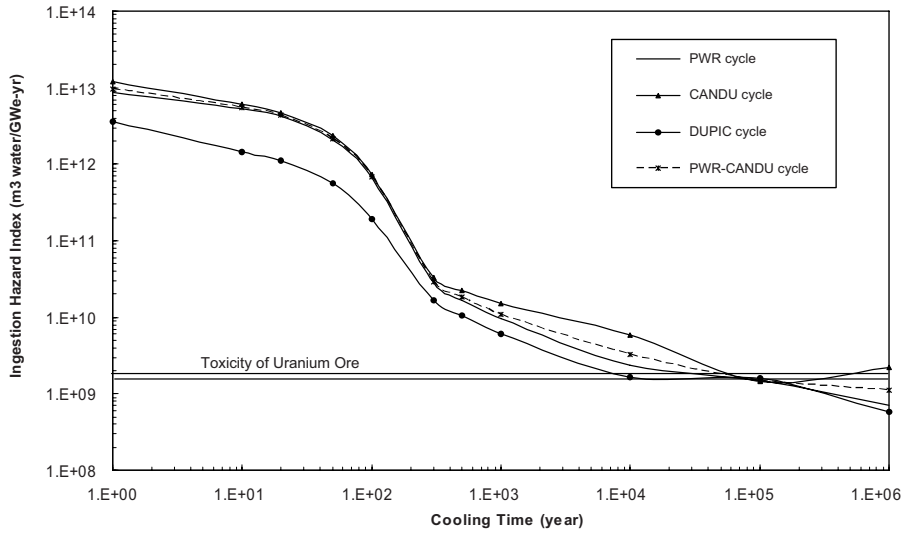


FIG. 11. Ingestion hazard index for various fuel cycle alternatives (based on 1 GW(e)·a).

The maximum permissible concentration limit (C_{ik}) for ^{226}Ra is 3×10^{-8} μCi/cm^3 from Korean regulations [8]. Therefore, the toxicity of uranium ore mined for 1 GW(e)·a of PWR reactor operation calculated from Eq. (10) becomes:

$$\frac{(176\text{tU})(0.33\times10^{6}\,\mu\text{Ci/tU})}{(3\times10^{-8}\,\mu\text{Ci/cm}^{3})(1\times10^{6}\,\text{cm}^{3}/\text{m}^{3})} = 1.93\times10^{10}\,\text{m}^{3}$$

The toxicities of uranium ore for all the fuel cycle options are compared in Table 4. From Fig. 11, the ingestion toxicity of each fuel cycle option decays to a level below that of the initial ore after a period of about 10 000–200 000 years. The toxicities ultimately decay to a fraction of a per cent of the toxicities of the original ore consumed to generate these wastes. From Fig. 11, it is indicated that the DUPIC, PWR-OT, PWR-CANDU-OT and CANDU-OT options decay to a level below that of the initial ore after about 10 000, 40 000, 80 000 and 200 000 years, respectively.

It is likely that the long term hazards from geologically isolated high level wastes will be less than those already experienced due to naturally occurring uranium minerals. The period of greatest importance in high level waste management is probably the earlier period, 100 000 years. In conclusion, up to that period, the toxicity of the DUPIC option is much smaller than that of any other fuel cycle option.

6. CONCLUSIONS

This study has examined the contribution of DUPIC technology to fissile management, for which the fissile material inventory of the DUPIC fuel cycle was compared with the inventories of once-through fuel cycles.

TABLE 4. TOXICITY OF URANIUM ORE FOR DIFFERENT FUEL CYCLE OPTIONS

Fuel cycle option	Ore toxicity for 1 GW(e)·a (m^3 water)
PWR-OT	1.93E+9
CANDU-OT	1.48E+9
DUPIC	1.41E+9
PWR-CANDU-OT	1.81E+9

It was found from the estimate of actinide content that the DUPIC option has the lowest plutonium arisings based on 1 GW(e)·a, and the fissile plutonium content generated in the DUPIC fuel is also shown to be the lowest among the fuel cycle options. On the whole, the CANDU cycle has the largest fissile plutonium content as well as gross plutonium content, which are negative features in regard to resistance to nuclear proliferation. It was indicated from the radiotoxicity analysis that the DUPIC option could have an indirect benefit on the safety of long term spent fuel disposal.

In conclusion, the implementation of the DUPIC fuel cycle will provide multiple benefits, such as:

(a) Increasing the effectiveness of fissile material management;
(b) Reducing the amount of accumulated PWR spent fuel by burning it in CANDU reactors;
(c) Reducing the CANDU spent fuel arisings owing to the higher burnup of DUPIC fuel;
(d) Making savings in the natural uranium resources required to produce CANDU fuel;
(e) Securing proliferation resistance.

In addition, it is known that the DUPIC fuel cycle option is economically competitive compared with the once-through fuel cycle, even though the cost difference between the DUPIC and once-through fuel cycles is very small [4].

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FUTURE FUEL CYCLE TECHNOLOGY OPTIONS

(Session 3)

Key Issue Paper of Working Group 3

SUSTAINABLE NUCLEAR ENERGY DEVELOPMENT BEYOND 2050: CROSS-CUTTING ISSUES

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Abstract

The paper describes three defined nuclear fuel management strategies — with an emphasis on expected developments. The main technological challenges and the economic implications, as outlined in the papers presented at this Conference, are described. The criteria are discussed that influence public acceptance of each of the nuclear energy generation options: non-proliferation and environmental friendliness. The sustainabilities of the three nuclear fuel management strategies and their variants are compared by cross-cutting through the nuclear energy generation options using all the criteria: economic and ecological implications, proliferation resistance, and technological challenges.

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1. INTRODUCTION

Expanded deployment of nuclear energy in the decades ahead will continue to require that public concerns about natural resources, economic viability, safety, waste management, proliferation and terrorism are adequately resolved. In this paper, we will examine the options for such strategies and consider their merits and disadvantages in terms of these cross-cutting issues.

Speculative uranium resources (recoverable at costs of up to US $130/kg) of 4 440 000 tonnes are expected to be adequate up to 2050 [1]. However, as these resources are consumed, a fissile material management strategy other than the once-through cycle may be required.

Over the timescale considered, many fundamental factors will no doubt change, for example:

(a) Additional rich reserves of uranium may be discovered, or the economics of uranium extraction from sea water may become attractive.
(b) The installed nuclear capacity may grow at a greater or lesser rate than that currently assumed.
(c) Nuclear energy systems using plutonium or ^{233}U may expand, thereby postponing the date when uranium is no longer available at attractive prices.

The arguments advanced in this paper lead to three options, under different circumstances, namely those with:

(1) No uranium or thorium resources;
(2) No ^{235}U enrichment technology capabilities;
(3) No reprocessing capabilities.

Under any of these circumstances, a nation may have to develop a strategy that does not rely on full access to all nuclear technologies, including enrichment and reprocessing. Sustainable fuel cycles may depend on the success of current efforts to limit the spread of enrichment and present PUREX reprocessing technologies, under which international commitments or even international nuclear production centres could make national self-sufficiency in nuclear fuel supply obsolete.

As stated before, currently known uranium resources are, under reasonable growth assumptions, sufficient to fuel nuclear power stations for about 50 years. However, beyond this date any new fuel management strategy will need development time and a test phase. This may result in a lead time of

30 years or more before widespread deployment of a new generation of nuclear power stations can start.

Decision makers in certain States will only support those new nuclear energy generation concepts that guarantee a long lasting energy supply and that meet public concerns about environmental friendliness, nuclear proliferation resistance, sustainability in resources and economic viability.

This paper only addresses related inherent properties of the fuel cycle concepts considered. It does not address institutional measures or engineered barriers, which may be intended to enhance proliferation resistance, reactor safety or environmental friendliness, or to prevent nuclear terrorism.

Section 2 outlines the basic fuel cycle options, touching on their economic implications and the technological challenges that lie ahead. Section 3 summarizes the social criteria which presently hamper the spread of nuclear energy generation in many countries. By cross-cutting the nuclear fuel management strategies with the following issues, we find the merits and acceptance problems of each option. We finally show that there is no general solution appropriate for the entire world.

2. FISSILE MATERIAL MANAGEMENT STRATEGIES FOR SUSTAINABLE DEVELOPMENT

To be sustainable, a fissile material management strategy must meet certain tests. Firstly, it must draw upon a resource of fissile materials adequate to allow for continued and expanded use for centuries. Secondly, it must embody technical measures and implementation arrangements that will gain and maintain public and political acceptance, including considerations of safety, non-proliferation, environmental protection and economic viability.

The extent to which the international community relies on nuclear power in the future will depend on a range of factors, including the following three cross-cutting issues:

(1) International security;
(2) The impact that global warming has on life on earth;
(3) Public acceptance — which is influenced most of all by safety, security and economic viability.

Taking all factors into consideration, the strategies for fissile material management for the future reduce to three options: the open ended uranium fuel cycle, a closed uranium–plutonium fuel cycle and a thorium fuel cycle.

Each option has benefits and drawbacks, and there are numerous variations on each. Five issues cut across all three options:

(1) Safety;
(2) The potential to in some way facilitate proliferation of nuclear weapons;
(3) The technological challenges that must be overcome for successful implementation in the context of sustainable development;
(4) Economic viability;
(5) The hazards to the environment associated with radiotoxicity.

Sustainable fuel management strategies can incorporate:

(a) Direct fission of ^{235}U, the only fissile nuclide occurring in nature in abundant supply;
(b) Fission of plutonium, which is produced as a natural by-product of fissioning ^{235}U (including thermal recycle and breeder operations);
(c) Fission of ^{233}U, which is produced by irradiating thorium, which occurs in nature in abundant amounts.

Future fissile management strategies can emphasize one of these elements or can combine two or all three.

Of the 440 nuclear reactors currently in operation, most are light water uranium fuelled thermal reactors. Less than 10% of LWRs recycle the plutonium from spent fuel. Reactors that do not use recycled fuel operate on what is referred to as the once-through or open fuel cycle. Heavy water reactors (such as CANDUs) and high temperature, gas cooled reactors (which are under development) also operate on once-through fuel cycles. This type of fuel cycle is the least efficient in terms of husbanding the uranium occurring in nature, and generates the greatest volume of spent fuel with the highest radiotoxicity. In order to be able to continue the use of the open fuel cycle strategy for a long time (e.g. several centuries), it will eventually become necessary to exploit unconventional uranium supply resources. While this is not an imminent problem, changes will become necessary in the future depending on the expansion of nuclear energy and the availability of commercially viable ore deposits. If uranium can be extracted from sea water at competitive prices, there will be a practically unlimited 'eternal' supply of uranium fuel.

Of all the factors driving towards recycle, management of spent fuel is the most important. Geological repositories are enormously expensive and politically controversial. Recycling the plutonium and setting aside the remaining uranium from spent fuel allows the waste volumes to be reduced dramatically. By separating strontium and caesium for interim storage and by

destroying the minor actinides (MAs) present in the spent fuel, the heat load and toxicity of the wastes will be reduced, thereby reducing the geological repository requirements.

So far, nuclear energy generation has been based on ^{235}U fission in thermal reactors[1]. The ^{238}U — also present in nuclear fuel — is transmuted to plutonium, some of which fissions in situ, contributing about a third to the total energy release.

The once-through, uranium fuelled, thermal fuel management strategy is mostly applied, although this strategy is the least efficient in resources and places the greatest demand on spent fuel disposal. The question of how to dispose of the spent fuel has not been answered. Repositories of the Yucca Mountain type are enormously expensive and politically controversial. Deep borehole options may prove to be an attractive alternative, and, in place of permanent geological disposal, an intermediate storage (for about 100 years) with retrievable storage thereafter may offer the advantage of treating spent fuel later. In the future, spent fuel management requirements may alone dictate future fuel management strategies, encouraging Pu or ^{233}U breeding and partitioning and transmutation (P&T) of MAs as a means to reduce repository requirements. This may sound too futuristic, but disposal in space — as mentioned below — might be the ultimate solution.

To continue in a resource sustainable open fuel cycle strategy, it may be necessary to exploit unconventional supply resources earlier than would otherwise be the case. If uranium can be extracted economically from sea water, there would be a practically unlimited 'eternal' supply. First studies are promising, but it is premature to build on this option. It has been recommended to pursue these studies despite the huge installations that would have to be installed in the sea. This is the main technological challenge to extending the sustainability of the open cycle with uranium as a fuel [2].

The present HWR fuel is natural uranium. As such, HWRs have a slight resource advantage over LWRs because HWRs need about 20% less natural uranium compared with present LWRs (which could be partially compensated by lowering the enrichment tails assay to 0.1%). With future rising uranium prices, this difference would widen the present HWR cost advantages in electricity generation. In contrast, fast reactors (FRs) use more than 90% of the potential energy in natural uranium. Hence, with FRs, low cost uranium

[1] 'Thermal reactors' include neutron moderator materials such that the kinetic energy of neutrons is moderated through elastic collisions from the energy at which they are released (≈2 MeV) to the thermal equilibrium energy (≈0.025 eV), at which the neutron cross-sections for fission are much greater.

resources could last much longer. Overall, therefore, the economic incentive to use the open fuel cycle will decrease with rising uranium prices; energy security will become increasingly susceptible to questions regarding the reliability of uranium supply. The geological repositories will have larger capacity requirements due to higher heat producing nuclear wastes. These are inherent to the open fuel cycle.

In a symbiotic manner, plutonium recovered from spent LWRs (possibly also from HWRs) will initiate the ^{238}U–Pu breeding cycle in FRs. It is likely that plutonium together with the MAs will be partitioned from spent fuel by advanced reprocessing, including pyro- or advanced aqueous recycle technologies, such as the envisioned GANEX (grouped actinides extraction) process in France (Fig. 1), and burned in future FRs, resulting in about a 10% additional savings in natural uranium [3].

Other advanced partitioning processes, mainly on a pyrochemical basis, have been developed and tested on a pilot scale [3]. The earlier aqueous processes, which include separation steps with purified plutonium, are of proliferation concern. Estimates of the additional cost to the fuel cycle are not prohibitive, and the capability of the FR in relation to transmutation has been demonstrated. The present obstacles to introducing FRs are the higher investment cost for the fuel cycle facilities, the processing of radiotoxic plutonium and the proliferation risks. With rising uranium prices, the cost threshold of LWR versus FR with MA transmutation will eventually be crossed.

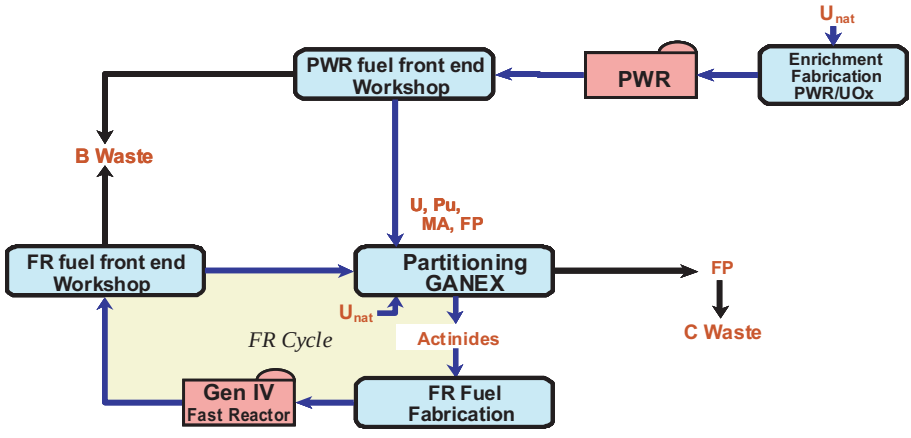

FIG. 1. Modular CEA design of a next generation spent fuel treatment (GANEX).

Significant development is required to produce fuel or targets containing MAs for transmutation for accelerator driven systems (Fig. 2) or by FRs. Compared with accelerator driven systems dedicated to MA transmutation, the development efforts necessary for achieving transmutation in FRs are minor [4].

The main technological challenges for FRs beyond the present very well known sodium cooled designs are [5]:

(a) Reactor designs allowing significant cost reductions in investment and generation: several options must be investigated, such as simplification of reactor circuits, utilization of alternative coolants allowing the use of higher temperatures, which increase plant efficiency and facilitate hydrogen production, and design features optimized to facilitate operational inspections of reactors and maintenance procedures.
(b) Fuel designs must be investigated that improve reactor safety through high fission product retention capabilities under normal and accident conditions: these future fuel designs must improve core behaviour under severe accident conditions on criticality and decay heat removal.
(c) The new fuels being designed will have an impact on reprocessing and increase the maximum MA loading to meet core safety requirements.

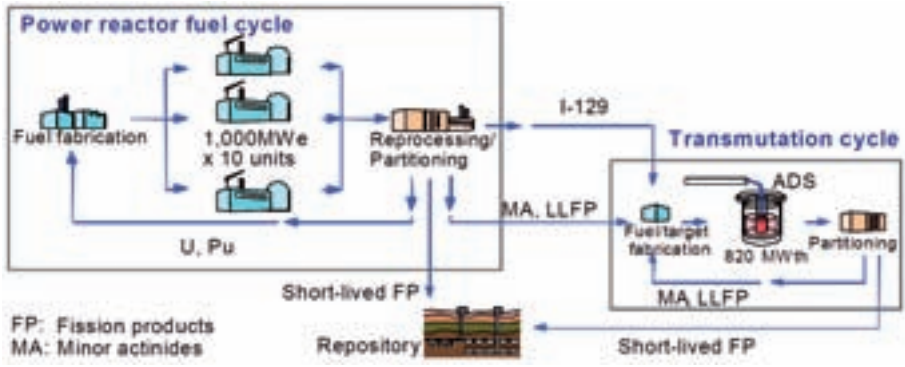

FIG. 2. JAERI double strata fuel cycle for transmutation of long lived radionuclides. A single unit of a burner system with a thermal output of ≈800 MW(th) can transmute MAs originating from ten 1000 MW(e) power reactor units. Accordingly, the heavy element mass in the transmutation cycle is significantly smaller than that in the power reactor fuel cycle. The support factor deteriorates with MOX fuelled power reactors, owing to increasing production of MAs.

A thorium fuel strategy could become a serious alternative to the FR cycle.[2] Given the great emphasis on uranium fuels, the thorium cycle with ^{233}U breeding is far less developed than the plutonium recycle nuclear fuel strategy. Several reactor concepts have been tested [6], and with the natural abundance of thorium, there is no resource reason why this option should not be broadly exploited. To start the thorium cycle, in the past, fuel enriched in ^{235}U has been used. Plutonium recovered from spent uranium fuels from thermal reactors is another choice. The use of a spallation neutron source driven by an accelerator has also been proposed to kindle a thorium reactor; about 10% of the electricity generated would be required to power the accelerator. However, so far no reliable accelerators with the required performance exist.

3. CRITERIA INFLUENCING PUBLIC ACCEPTANCE OF NUCLEAR ENERGY GENERATION

The public is concerned more about safety, waste management and economic issues than about proliferation issues. Governments are more concerned with security threats arising from:

(a) Proliferation of nuclear weapons or other nuclear explosive devices;
(b) Theft of fissile or hazardous radioactive materials for use in nuclear explosives or radiological dispersal devices (RDDs);
(c) Sabotage of nuclear facilities (especially reactors, spent fuel recycle facilities or waste conditioning/storage facilities, and sabotage of transport systems).

In a general sense, proliferation involves actions undertaken by a State to acquire nuclear weapons. Proliferation also involves actions by a State to provide assistance — intentionally or unknowingly — to other States or terrorist organizations to acquire or produce fissile material, or to supply equipment, components, specialized materials or technology suitable for use in a programme leading to the manufacture of nuclear weapons or other nuclear explosive devices.

[2] Thorium fuels were considered in the Generation IV programme, but no specific needs or benefits were identified, and no specific programmes are anticipated in that venue.

Any fuel management strategy that requires highly enriched uranium (HEU: uranium containing any mixture of the isotopes ^{235}U and ^{233}U such that $(\%\ ^{235}U + \frac{5}{3}(\%\ ^{233}U)) \geq 20\%\ U)$ or separated plutonium (especially containing high fractions of ^{239}Pu) in effect provides potential access to nuclear weapons. The use of enrichment technology (especially present gaseous centrifuge technology) and reprocessing technology (especially present PUREX reprocessing technology) makes such strategies vulnerable to misuse. This applies, to some extent, to the partitioning of neptunium and certain trans-plutonium nuclides as well.

Even with stringent nuclear material safeguards and with establishment of strict international physical protection, the danger remains that a State may decide to abrogate its non-proliferation undertakings or that weapons useable fissile material could be diverted or stolen by subnational groups, for example, terrorists.

Accordingly, fuel strategies that do not rely on ^{235}U enrichment or recycle technologies that lead to separated plutonium would be preferred by the public, as are those that make no use of HEU fuel. Enrichment technology is even more sensitive than plutonium recycle: the fissile uranium isotopes do not require sophisticated implosion technology — nuclear weapons made from ^{235}U or ^{233}U can use the simpler 'gun type' design. Plutonium with any isotopic composition (as occurs in the above fuel strategies) is regarded by the IAEA as potential weapons useable material — to some extent this depends, however, on the content of the plutonium isotopes ^{238}Pu, ^{240}Pu and ^{242}Pu, and on the ease of separating plutonium from the fresh fuel matrix — especially if mixed with other radiotoxic impurities and/or other spontaneously fissioning nuclides such as the curium isotopes ^{242}Cm and ^{244}Cm.

With these proliferation reservations, the once-through HWR and LWR fuel cycles are preferred over any fuel cycle containing plutonium in a purified form, for example, as obtained from the present PUREX process. Note that there is no stabilization of plutonium inventories under the once-through fuel cycle; the inventories with plutonium in spent fuels will continue to increase over time.

There are a number of ways in which the possibility of acquiring weapons useable fissile material from a peaceful nuclear energy programme can be impeded or even prevented. For example, an accelerator driven subcritical HWR with thorium fuel can be operated in an open fuel cycle. The spent fuel will be made 'proliferation resistant' because of the chemically inert fuel matrix and the presence of highly energetic gamma radiation from the ^{232}Th decay chain. (Uranium-233 created by irradiating ^{232}Th always contains ^{232}U, which soon builds up hard gamma emitting decay products.) In addition, by adding natural uranium to the thorium fuel, the ^{233}U can be kept below the 12%

isotopic concentration level. Similarly, the partitioning of transuranium (TU) elements by pyrochemical processes or the GANEX aqueous process (under development in France, which should coextract the TU elements) will result in highly radioactive FR fuels, which would deter terrorists from handling such material. Such mixtures of actinides and fission products are difficult and, under certain circumstances, impossible to use as nuclear explosives because of the high spontaneous fission neutron background leading to nuclear pre-ignition or degradation of chemical explosives.

In spent HWR fuels, the ^{239}Pu isotopic concentration is higher due to the lower burnup than that in spent LWR fuel. If these fuels are disposed of directly, as the fission product radioactivity decays, the plutonium remaining in the spent LWR and HWR fuels would become increasingly easy to extract (i.e. a 'Pu mine' for future generations). After 100 years of storage, for example, mining the plutonium may present a quite attractive means to acquire nuclear weapons. Apart from the expensive 'plutonium mining', it would remain necessary to master the implosion technique. High ^{239}Pu content plutonium is the material of choice for sophisticated and compact nuclear weapons; however, weapons or improvised nuclear explosives made using HEU are easier to produce.

Noting that all plutonium isotopes display fission properties that would make them useable in nuclear weapons, the lower the content of ^{239}Pu, the more difficult it becomes to manufacture any nuclear explosive. Weapons useable material with mixed-in hard gamma or neutron emitting nuclides can easily be detected by radiation monitors (which is not the case for HEU containing high fractions of ^{235}U).

Nuclear energy generation unavoidably produces radiotoxic by-products. This is common for any nuclear fuel strategy. The in-core radiotoxicity is directly proportional to the power of any plant. After Chernobyl, the public has become highly sensitive to accidental releases of radioactivity from power stations. Hence, reactor safety is an important acceptance requirement for a sustainable fuel management strategy. However, this paper does not discuss the different engineered safety concepts; there is recognition that passive safety features are likely to be essential for future public acceptance. Passive safety features would be designed in as inherent features of the nuclear energy generation concepts mentioned. They would include mechanisms to limit power density, and to prevent fuel melting and the reassembly of a critical mass in the case of core destruction. Passive safety would prevent accidents in the case of an interruption of cooling.

At present, the main radiation dose commitment to the public stems from former uranium mining and milling, when radioactive elements of the natural decay chains are released to the biosphere. In the future, in situ leaching of

uranium, returning the tailings to underground mines or to pervious surrounding systems, will protect groundwater [1]. Nevertheless, any fuel management strategy requiring less natural uranium (or thorium) would reduce the remaining risk. Note that in the extraction of uranium from the sea the natural decay products remain in the water.

In the waste discharged to geological repositories within the different strategies, the amount of long lived radiotoxic nuclides varies. They are most abundant in the open uranium fuel cycle, less in the thorium fuel cycle (because of the much lower buildup of transuranics, assuming that ^{231}Pa is recovered). In the FR cycle with the envisioned transmutation of long lived radiotoxic nuclides, we will observe the lowest discharged radiotoxicity to geological repositories.

If transmuted or not, there still remains the need to permanently dispose of the residual radioactive nuclides. Currently, disposal is envisioned in geological repositories — the number and capacity of repositories will be considerably less with the P&T strategy. The P&T strategy will minimize the potential radiotoxicity in geological waste disposal (Fig. 3).

The remaining volume of long lived radiotoxic fission products — which are very difficult to transmute — amounts only to less than 10 L/1000 MW(e) annually from a present-day LWR. Hence, in future with mature space

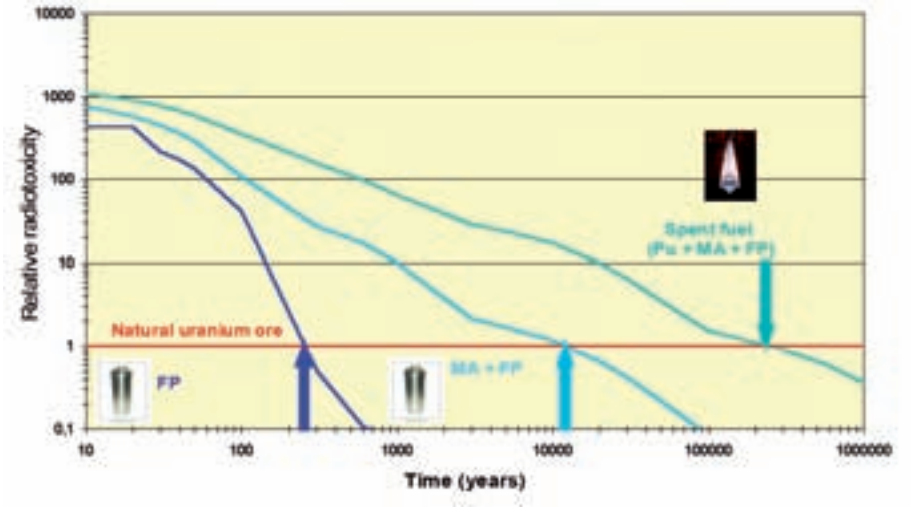

FIG. 3. Time dependence of radiotoxicity in a P&T scheme.

technology, a disposal in space seems the solution, and was once the subject of a study in the former USSR.

4. CROSS-CUTTING NUCLEAR FUEL MANAGEMENT SECTIONS BY ALL CRITERIA: ECONOMIC AND ECOLOGICAL IMPLICATIONS, PROLIFERATION RESISTANCE AND TECHNOLOGICAL CHALLENGES

Cross-cutting looks at individual sections of the three defined fuel cycle strategies, because it might well be that for various reasons the most economic, most ecological and most proliferation resistant strategy will consist of parts taken from different fuel cycle strategies. The characteristics of the different fuel cycle strategies are summarized in Table 1. To avoid repetition, only those characteristics of the fuel cycle sections are discussed that might favour or put into question a sustainable fuel management strategy.

All three fuel management strategies contain major technological challenges:

(a) The cost competitive mining of very low grade ores or the filtering of uranium in ppb concentrations from the sea;
(b) The Pu/MA group partitioning from spent FR fuel by advanced processing;
(c) The reprocessing in the thorium cycle — either in-line (as in the molten salt breeder reactor), or separately.

For the economic implications we should distinguish between the development and fuel cycle costs (including investment). Since the latter costs currently amount to only about 10% of the cost of electricity, they are less important for a management strategy that is sustainable in supply and public support.

In the open fuel cycle, the reactors are well developed and proven. In this case, the expenses for fuel from unconventional sources are significant. Of course, this strategy foresees direct disposal in geological repositories, which are expensive and politically controversial.

The P&T strategy for the uranium or thorium fuel cycle involves destroying MAs with high energy neutrons in an FR or in an accelerator system. Partitioning and transmutation substantially reduces the geological disposal capacity required. On the other hand, P&T requires the development and operation of the closed uranium and thorium fuel cycles with significant

TABLE 1. COMPARISON OF DIFFERENT FUTURE NUCLEAR FUEL CYCLE OPTIONS (*LESS FAVOURABLE*, NEUTRAL, **PREFERABLE**)

		Open fuel cycle		U–Pu (MA): Closed cycle			Th cycle: Open and closed		
		LWR	HWR	FBR (MOX)	FBR (MA)	FBR/ADR	Th/Pu LWR	Th/Pu HWR	Th–U
Proliferation aspect	Material	Enriched ^{235}U	**U_{nat}**	*U/Pu*	U/Pu/MA	*Pu + MA*	Th–Pu	Th–Pu (Dupic)	Th–^{233}U
	Technology	*^{235}U enrichment*	**No enrichment**	*PUREX*	Advanced processing	*Wet processing*	*PUREX*	*PUREX* ***(DUPIC)***	Pyro-processing
	Waste	*Pu mine*		No Pu	**Less Pu, MAs**		*Pu mine*		**No Pu mine**
Environmental impact	U mining	Mill tailings **but not U from sea**		**No mill tailings**			**No mill tailings**		
	Waste	*Pu, MA*		MA	**Less actinides, less repositories**		*Pu, MA*		**No actinides**
Technological challenges and economic implications	Fuel cycle technology	**Developed**		**Developed**	Under development	*Not developed*	Developed		*Not developed*
	Challenge	*(U from sea)*		*New cooling*	*~ + pyro.*	*~ + ADR*	*Th pyroprocessing*		

costs compared with the fabrication of fuel for the open fuel cycle, but fabrication is a small part of the total cost of nuclear power generation and this increase can be offset by the reductions in geological disposal requirements.

In the context of non-proliferation, a fissile material management strategy relying on ^{235}U enrichment or current PUREX reprocessing of spent fuel has a potential risk of proliferation, especially in States where such activities are not justified on the basis of an appropriate nuclear power programme. Here the once-through HWR or a P&T recycle scheme are the better choices.

Apart from radiotoxicity, environmental protection includes limitations on land consumption, which is reduced by fuel management strategies requiring less uranium mining (like any of the breeder concepts and like uranium extraction from sea water). Moreover, the discharge of mill tailings — at present the largest environmental burden — will also be reduced. As stated previously: the P&T concept reduces the size of a geological repository due to the elimination of long lived, heat producing nuclides and so reduces the radiotoxicity of long lived actinides. In this respect the thorium cycle with less of an MA buildup has a similar advantage.

5. SYNOPSIS AND CONCLUSIONS

5.1. OPEN FUEL CYCLE

The open uranium fuel cycle strategy is fully developed. Power stations using this strategy are operating at present at the lowest costs of any nuclear option. The existing fleet of nuclear reactors will in all likelihood continue according to this strategy. Even new reactors, such as the EPRs, are likely to carry this strategy forward over their entire lives. The anticipated life cycle of new plants is expected to be 60 years. Thermal recycle can help to reduce the uranium requirements in the future; however, with the uranium resources known to exist, this strategy can continue to serve as the option of choice for decades into the future.

The main drawbacks for long term reliance on the open uranium cycle fuel strategy are as follows:

(a) The utilization of uranium is less than 1% — long term resource sustainability of the open fuel cycle strategy will eventually require economically

viable methods for extracting uranium from low grade ores and from the sea.

(b) Continued reliance on enrichment for LWRs, unlike for HWRs, with increasing needs to serve expanding nuclear power programmes and the associated proliferation risks arising from the increased access to centrifuge technology (or other enrichment technologies).

(c) Heavy water reactors fuelled with natural uranium featuring on-load refuelling could facilitate the production of weapons grade plutonium, and the safeguards requirements to assure that such operations are not carried out require a substantially greater verification burden than for LEU fuelled LWRs.

(d) Rising mining and mill tailings constitute a growing environmental problem, more so than for any other fuel cycle strategy.

(e) The open fuel cycle is the least efficient strategy for geological repositories, requiring more capacity than any other fuel cycle strategy.

(f) Since the plutonium remains with the spent fuel, the repositories could later serve as 'Pu mines' for future generations.

5.2. URANIUM–PLUTONIUM CLOSED FUEL CYCLES

Variations of closed U–Pu fuel cycle strategies have been established, and advanced concepts are under consideration.

5.2.1. Thermal recycle

Current thermal recycling involves separation of plutonium using current PUREX reprocessing and production of MOX fuel assemblies for replacement of LEU fuel in LWRs. Currently, some 35 reactors are burning MOX, and the number is expected to increase. Thermal recycle provides for a reduction in the volume of waste requiring geological storage and can reduce the uranium resource requirements needed to sustain a fleet of reactors by about 15% or more.

There are several drawbacks associated with this example of a 'closed' fuel cycle:

(a) The present thermal recycle is limited to one pass; multiple passes are possible but are not currently performed.

(b) The costs of reprocessing and MOX fuel fabrication exceed the costs of LEU fresh fuel; hence this strategy does not offer a compelling economic advantage.

(c) The current PUREX reprocessing technology provides pure PuO_2 products, which can be readily converted to metal for use in nuclear explosives.
(d) PUREX reprocessing technology (and other plutonium reprocessing technologies as explored in the past) can be used to support clandestine heavy water or graphite production reactors as part of a proliferation activity.

5.2.2. Fast reactor recycling

Future fuel cycles anticipate multiple pass utilization, with efficiencies approaching 90% use of the latent energy in uranium. This will dramatically reduce uranium requirements, as well as reduce the mining and milling tails associated with open cycle operations. As such, this closed fuel cycle will meet expanding future energy needs for centuries.

In addition, future systems will include provisions for P&T that will reduce the volume and radiotoxicity of the waste discharged to geological repositories.

The advanced recycling technologies under development for this application (including pyroprocessing and the GANEX process) are intended to prevent access to pure plutonium, and thereby will increase the proliferation resistance of this technology over that of the present PUREX processes.

'Classic' plutonium recycling technology with sodium cooled FRs has been demonstrated at several places. As in any recycling, less uranium mining is needed than for the open fuel cycle strategy, and as a consequence mining and mill tailings are reduced. In the reprocessing waste, the remaining plutonium content is less than 0.2%, and the highly active waste, together with all the MAs, is suspended in less leachable glass with a lower volume in comparison with directly stored fuel. However, the present PUREX process is regarded as a proliferation sensitive technology, which limits its worldwide deployment.

To partition the heating and radiotoxic MAs from the waste stream and to transmute them in FR or dedicated MA burner reactors would improve public acceptance with regard to environmental friendliness. The P&T concept to partition plutonium together with MAs by advanced chemical processes is more favourable in regard to proliferation resistance than the present aqueous processes, which are, however, more advanced in development. The resulting 'dirty' fuel prevents its use as nuclear explosives. Research and development (technological challenge) is under way to test pyrochemical processes, to develop an advanced aqueous process (GANEX) and to demonstrate the feasibility of an accelerator driven transmutation reactor.

5.3. THORIUM FUELS

To initiate the Th–^{233}U cycle, use of the variant of Th/Pu fuel in HWRs is a sustainable strategy. The first step can be regarded as an open cycle, because the spent Th/Pu fuel, as well as the DUPIC fuel, will not be reprocessed and directly stored. The resulting 'Th/Pu mine', in comparison with directly stored LWR ^{235}U fuel, contains plutonium depleted in ^{239}Pu; this definitely constitutes an increase in proliferation resistance and also leachability. Nevertheless, the plutonium has to be separated for the Th/Pu fuel from spent LWR(HWR) fuel. If this turns out to be a problem, the proposal to refabricate spent LWR fuel for HWR use by the DUPIC process is a more proliferation resistant option. (The DUPIC fuel with separated thorium pins is irradiated in HWRs to breed ^{233}U.) The main challenge in the thorium cycle is the reprocessing step. Although the ^{233}U (^{232}U) produced by pyrochemical processes may deter diverters, an HWR fuel cannot contain the same amount of radioactive fission products as is tolerated by an FR fuel. Moreover, with ^{233}U, a 'gun type' nuclear explosive is possible.

We can conclude that the prospects for resources, fuel cycle cost and technological challenge allow for several nuclear energy options beyond 2050. However, the social criteria, non-proliferation and environmental friendliness, as perceived by a State, seem to be decisive in choosing the sustainable strategy.

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THE ETERNALLY OPEN URANIUM FUEL CYCLE*

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Abstract

The once-through uranium fuel cycle strategy, with no recycling of spent fuel, is currently the fuel cycle of choice. However, it is not without problems and complications, and its sustainability will increasingly deplete uranium sources and exacerbate spent fuel management demands. Moreover, the once-through uranium fuel cycle encourages expansion of enrichment services, and these could stimulate proliferation. In the paper, the fuel cycle material requirements are examined on the basis of the constraints of existing nuclear power plants and spent fuel storage and reprocessing facilities. Projected changes in population and energy use are considered as drivers for further expansion, and resource projections to support the continued use of the once-through fuel cycle are included. Possible new reactor technologies are briefly described, and proliferation concerns related to different fuel cycles are addressed.

1. INTRODUCTION

To be sustainable, a fissile material management strategy must meet certain tests. Firstly, it must draw upon a resource of fissile materials adequate to allow for continued and expanded use for an extended period, beyond the lifetime of the plants being considered and preferably for a long time into the future (e.g. centuries). Secondly, it must embody technical measures and implementation arrangements that will gain and maintain public and political acceptance, including considerations of safety, non-proliferation, environmental protection and economic viability.

The open uranium fuel cycle, a fuel cycle with no recycling of the spent fuel, is not a 'cycle' as such, but a sequence of one way operations from mining through use to disposal. While at present it is the fissile material management

* Work performed at the Pacific Northwest National Laboratory operated by Battelle for the United States Department of Energy.

strategy of choice, the once-through uranium fissile material management strategy is not without problems and complications:

(a) Long term resource sustainability of the once-through strategy will eventually deplete rich ore bodies; for this option to continue, economically viable methods for extracting uranium from low grade ores and from the sea will be needed.
(b) Continued reliance on enrichment, with increasing needs to supply expanding nuclear power programmes, may give rise to increased proliferation risks arising from the increased access to centrifuge technology (or other enrichment technologies).
(c) Heavy water reactors fuelled with natural uranium featuring on-load refuelling may facilitate the production of weapons grade plutonium, and the safeguards requirements to ensure that such operations are not carried out impose a substantially greater verification burden than for light water reactors (LWRs) fuelled with low enrichment uranium.
(d) Uranium mining and mill tailings constitute a growing environmental problem, and increasing these operations to sustain the once-through strategy will continue to make this problem worse.
(e) The once-through strategy, which has no reprocessing or partitioning and transmutation under its current implementation, is the least efficient strategy in terms of requirements for geological repositories.
(f) Since the plutonium remains with the spent fuel, geological repositories serving the once-through strategy will at some point in the future become 'Pu mines' that could provide fission energy at a later time, or worse, could serve as a future source of plutonium for use in nuclear weapons or other nuclear explosive devices.

The once-through strategy is mature and easily expanded. As plutonium recycle in LWRs is at present not economic, power stations using this strategy are operating at present at the lowest costs of any nuclear option. The existing fleet of nuclear reactors will in all likelihood continue according to this strategy. Even the new reactors now being introduced are likely to carry this strategy forward over their entire lives. The anticipated life cycle of new plants is expected to be 60 years, so every new plant coming on-line until Generation IV systems come on-line will follow the once-through strategy. Thermal recycle (i.e. recycling plutonium into current LWRs) can help to reduce the uranium requirements of the once-through strategy for the future; however, the net savings are limited.

The once-through strategy produces greater volumes of highly toxic waste than any other fuel cycle. From a practical point of view, disposal of spent

fuel remains an unsolved problem. Perhaps the chief motivation for selecting closed fuel cycles in the Generation IV programme was the realization that under anticipated growth rates, by the middle of this century, the demands for repository storage could require a new repository with the capacity of Yucca Mountain as often as every two years. Within ten years, the spent fuel being discharged in the United States of America (USA) will fill the planned capacity of Yucca Mountain. While noting the enormous complexity associated with licensing Yucca Mountain, while uranium resources appear to be adequate to sustain the once-through fissile material management strategy for decades (or centuries if the extraction of uranium from low grade ores and/or sea water becomes economic), waste management alone would appear to make this practice insupportable for the indefinite future.

Provided once-through fuel cycle applications do not encourage the spread of enrichment technology or reprocessing, the once-through cycle can be highly proliferation resistant. The issue from a proliferation perspective is what the reactor State does, recognizing that other States will carry out the enrichment needed and that eventually a global arrangement will evolve for spent fuel management. In the ideal case, the fuel cycle is what the former USSR used to employ, providing all fresh fuel and taking back all spent fuel. The solutions offered not only have to be reliable (assurance of supply) but they also have to undermine any economic rationale that could justify a State's efforts to be independent.

However, even if fuel supplies are assured, a State may be tempted at some future time to acquire indigenous enrichment capabilities, and having once acquired such capabilities, proliferation possibilities would open for misuse, replication or even spreading the technology to other States or subnational groups. Similarly, even if spent fuel waste removal services are provided, a State may decide at some point in the future that circumstances have changed and that it must establish a capability to reprocess its spent fuel, perhaps citing waste management pressures as the driving influence. In addition, the once-through strategy may provide possibilities for undeclared production of fissile material in some reactors (depending on their physical characteristics).

To summarize, for non-nuclear-weapons States, the once-through uranium fissile material management strategy can provide the benefits of nuclear power with manageable proliferation risks — provided the State enters into an arrangement for assured supply of fresh fuel over the life cycle of the reactor, and a corresponding arrangement is made for the assured removal and disposal of the spent fuel, preferably outside of its territory or control. From a proliferation risk perspective:

(a) Best would be an arrangement in which there is an assured supply of fuel from an existing supplier for the lifetime of the reactor;
(b) Next best would be an arrangement with negotiated suppliers, preferably with a guarantor (such as the IAEA);
(c) Next would be a multinational arrangement in which the State is a participant, under arrangements which inhibit its ability to master enrichment technology;
(d) Worst would be an arrangement that encourages or provides an excuse for the State to acquire an indigenous enrichment capability.

Public acceptance of the once-through strategy in large applications will thus depend on the magnitude of the application and the conditions for implementation.

2. THE RESOURCE DIMENSION

Let us consider the sustainability of the once-through strategies beyond 2050 using all possible uranium sources.[1] Assume that reprocessing stops and all nuclear power eventually moves to the once-through strategy. To consider this situation, we:

(a) Address the status of the nuclear fuel cycle, including:
  — Existing fuel production resources, facilities and capacities;
  — Existing nuclear power plants;
  — Current spent fuel storage strategy.
(b) Examine projected changes in power requirements by 2050, in particular, expected population growth and the resulting power requirements, expected percentages of nuclear contributions, resource requirements needed to support the expected growth in nuclear power, and possible variations in reactor operations or changes in the reactor technologies used to support the expected growth.

There are at present 441 nuclear power plants in operation with a total net installed capacity of 368 GW(e), and 24 nuclear power plants under construction [1]. Most of this electricity is generated via the once-through fuel cycle using enriched uranium in LWRs. As a starting point, we consider the

[1] Note the other papers at this conference on the subject of existing uranium reserves.

material and processing requirements of various elements of the nuclear fuel cycle needed to support this generation rate.

2.1. Fuel cycle material requirements

Figure 1 shows the annual material requirements to support one 1000 MW(e) reactor, amortizing the requirements of an eighteen month fuel cycle over one year.

For the purposes of this paper, we assume that the values in Fig. 1 are representative of the power plants in the worldwide fleet. Table 1 shows what the worldwide requirements are to support 367 GW of nuclear electricity. To simplify our analysis, if all existing power plants were rated at 1000 MW(e) per plant, there would at present be 367 power plants. Again, for simplicity, Table 1 amortizes the assumed 18 month reload cycle over a one year period.

The current worldwide uranium production capacity is 36 000 t of uranium; the remainder of the 57 000 t required is made up from military stockpiles left over from the cold war or blending of surplus weapons grade highly enriched uranium (HEU). Table 2 compares the required capacities for uranium mine production, refining and conversion, and enrichment needed to support current operations. It can be seen that the refining and conversion capacities are sufficient to meet current needs. It should be noted, however, that the average age of enrichment facilities is 31 years, and that 67% of worldwide enrichment capacity is over 40 years old [2]. Application of newer technology will be required to replace or upgrade these older facilities.

2.2. Existing nuclear power plants

Table 3 [2] gives the number of reactors by type and the amount of installed megawatts by type, compares the percentage each type contributes to the total number of reactors, and gives the total number of megawatts and the average age for each type, as of 2002.

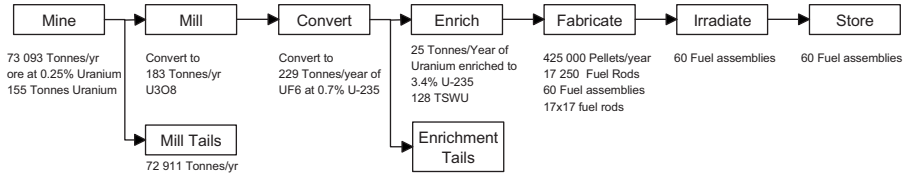

FIG. 1. Material requirements of a 1000 MW(e) LWR.

TABLE 1. MATERIAL REQUIREMENTS TO SUPPORT 367 1000 MW(e) REACTORS

Activity	Material
Mine	2.28E+07 Tonnes/a ore at 0.25% uranium
Uranium production	5.67E+04 Tonnes
After milling	6.71E+04 Tonnes/a of U_3O_8
Milling tails	2.27E+07 Tonnes/a
Convert ore to	8.42E+04 Tonnes/a of UF_6 at 0.7% U-235
After enriching	9.08E+03 Tonnes/a of UF_6 to 3.4% U-235
Enrichment tails	7.07E+04 Tonnes
Then fabricate	1.56E+08 Pellets/a
Fabricate	6.33E+06 Fuel rods
Fabricate	2.20E+04 Fuel assemblies with 17 × 17 fuel rods
Irradiate	2.20E+04 Fuel elements/a
Store	2.20E+04 Fuel elements/a
SWU required for enrichment	4.71E+04 TSWU

If the totals for PWRs and WWERs (which are Soviet designed PWRs) are combined, pressurized water reactors make up 58% of the total number of reactors and contribute 64% of the total number of megawatts produced. Boiling water reactors make up 21% of the total number of reactors and contribute 24% of the total number of megawatts produced.

Table 4 compares the number of reactors by moderator type. It can be seen that LWRs make up 79% of the total number of reactors and 87% of the

TABLE 2. COMPARISON OF EXISTING CAPACITY AND CURRENT NEEDS

Activity	2004 Capacity	Requirement	Per cent of capacity
Uranium mine production (tU/a)	3.58E+04	5.67E+04	158
Refining and conversion capacity (tU/a)	9.97E+04	6.21E+04	62
Enrichment capacity (TSWU/a)	4.57E+04	4.37E+04	96

TABLE 3. ELECTRIC POWER PRODUCTION BY REACTOR TYPE (2002)

Reactor type	Number of reactors	Installed megawatts	Per cent of total number of reactors	Per cent of total installed megawatts	Average age
PWR	200	1.8E+05	46.4	54.2	21
BWR	91	8.0E+04	21.1	23.7	23
WWER	48	3.1E+04	11.1	9.3	19
PHW	20	1.4E+04	4.6	4.1	17
LWGR	17	1.3E+04	3.9	3.7	23
AGR	14	8.4E+03	3.2	2.5	21
PHWR	14	3.6E+03	3.2	1.1	17
Magnox	18	2.9E+03	4.2	0.9	41
FBR	5	1.2E+03	1.2	0.4	25
Production (Russian PLWGR)	3	2.8E+02	0.7	0.1	44
LWCHWR	1	1.5E+02	0.2	0.0	25
Total	431	3.4E+05			

total number of installed megawatts. This is significant from a proliferation point of view because LWRs are not well suited for use as plutonium producers. Heavy water reactors, graphite moderated reactors and LMRs can all be more easily used as plutonium producers if the proper fuel is used and the fuel reload cycle is appropriately adjusted.

TABLE 4. ELECTRIC POWER PRODUCTION BY REACTOR MODERATOR (2002)

Reactor type	Number of reactors	Installed megawatts	Per cent of total number of reactors	Per cent of total installed megawatts
LWR	339	2.95E+05	79	87
HWR	34	1.73E+04	8	5
Graphite reactors	52	2.42E+04	12	7
LMR	5	1.19E+03	1	0

The average age of commercial LWRs is approximately 22 years. Since the original design life for many of these plants was 40 years, life extension activities are ongoing or being considered. However, to maintain the dominance of LWRs in the power production cycle, many will need to be replaced in the not too distant future.

2.3. Spent fuel storage and reprocessing facilities

While most spent fuel is currently stored on-site at each reactor, there are some central fuel storage facilities. Table 5 shows the existing worldwide storage capacity and compares it with the amount of heavy metal (uranium and plutonium) discharged on an annual basis. It can be seen that the existing facilities (even if they were currently empty) are completely inadequate to store more than a few years of total production. Since many on-site storage facilities are close to being filled to capacity, action must be taken soon to develop storage facilities.

TABLE 5. COMPARISON OF STORAGE FACILITIES AND ANNUAL HEAVY METAL DISCHARGES

	Worldwide total
Central storage (tHM)	49 000
Reprocessing (tHM)	5 400
Discharged yearly (tHM)	8 700

3. CHANGES PROJECTED BY 2050

3.1. Changes in population and energy use

Table 6 [3] shows projected growth patterns in population and energy use for 2000–2050. The table compares growth in the developed world, developing world and countries of the former USSR. This grouping approach was taken to account for relative differences in population growth and projected increases in

TABLE 6. CHANGES IN POPULATION AND NUCLEAR ENERGY PRODUCTION

	Total population (millions)		Country total electricity (billion kW·h)		Per capita consumption (kW·h)		Nuclear production (billion kW·h)						Nuclear capacity (GW(e))		
	2000	2050	2000	2050	2000	2050	2000	%	2050L[a]	%L	2050H[b]	%H	2000	2050L	2050H
Developed World	924	1010	8211	15 810	8888	15 659	1926	23	5197	33	8071	51	220	593	921
Developing World	4614	7395	4224	21 315	916	2882	91	2	2690	13	5347	25	10	307	610
Former Soviet Union	306	261	1202	1598	3925	6118	213	18	433	0.27	677	0.42	24	49	77
Totals	5844	8666	13 636	38 723	2333	4468	2230	16	8321	21	14 094	36	255	950	1609

[a] L: low projection.
[b] H: high projection.

electricity use in countries at different stages of development. Projections include low and high estimates of changes in nuclear production.

3.2. Estimated uranium resources

A review of the changes in nuclear capacity by 2050 shows expected increases ranging from four to six times that of 2000. The question of sufficiency of supply is addressed in Refs [4, 5]. In Ref. [4], a continued annual growth in nuclear energy is assumed that reaches 1120 GW(e) in 2050, a value slightly higher than the low projection of Table 6. In this case, the annual requirement for uranium would be 175 000 t of uranium, with a cumulative requirement (amount used) of 5.6 million tonnes by 2050.

Table 7 [5] shows the estimated uranium reserves of all types, including known conventional, undiscovered conventional, secondary and unconventional reserves.

TABLE 7. ESTIMATED URANIUM RESOURCES

Resource type	Estimate (1000 t)
Known conventional reserves	
Reasonably assured resources	2850
Estimated additional resources cat. I	1080
Undiscovered conventional reserves	
Estimated additional resources cat. II	2330
Speculative resources	9940
Secondary sources	
Commercial inventories	220
Surplus defence inventories	250
Re-enrichment	440
Subtotal	17 110
Unconventional resources	
In phosphates	22 000
In sea water	4 000 000
Total	4 039 110

Resources known to exist and that can be recovered using conventional mining techniques are classed as 'known conventional resources'. These resources are categorized into two subgroups: reasonably assured resources (RARs) and estimated additional resource category I (EAR-I) resources. Resources believed to exist and to be exploitable using conventional mining techniques, but not yet physically confirmed, are classed as 'undiscovered conventional resources'. These resources include estimated additional resource category II (EAR-II), which are uranium resources that are expected to be located in well defined geological trends of known ore deposits or mineralized areas with known deposits, and speculative resources (SRs), which are uranium resources that are thought to exist in geologically favourable, yet still unexplored, areas [5].

Secondary sources include:

(a) Inventories of previously mined uranium held by governmental and commercial organizations;
(b) Large inventories of previously mined uranium derived from military applications, and surplus plutonium which, converted to mixed oxide fuel, can reduce the need for fresh uranium;
(c) The remaining ^{235}U contained in large inventories of depleted uranium can be enriched in centrifuge plants and thereby reduce the need for fresh uranium.

Uranium has been recovered from phosphates in the past; however, high recovery costs limit their use in the current low price uranium market.

Studies [6] have indicated that as much as 45% of the uranium in sea water can be extracted, although the cost would currently be prohibitive under present market conditions. However, since the cost of uranium is only 5% of the overall cost [7] of reactor operation, it is not unrealistic to assume that extraction from phosphates or even sea water could become economic in the future.

3.3. Resource projections to support a once-through fuel cycle

Table 8 shows how long the different classes of resource could support a once-through fuel cycle. It should be noted that both conventional and phosphate reserves could provide sufficient resources to support nuclear power at its current rate (340 GW(e)) for 745 years, and at a projected rate in 2050 (1000 GW(e)) for 261 years.

Conservatively, it was estimated in this calculation that only 1% of the uranium in sea water could be recovered. Even so, the amount of uranium in

TABLE 8. RESOURCE USAGE

	Amount	
Total conventional and secondary sources	1.71E+07	Tons of uranium
Years at 2004 energy use (340 GW(e))	326	Years
Years at 2050 energy use (1000 GW(e))	114	Years
Unconventional resources — phosphate	2.20E+07	Tons of uranium
Years at 2004 energy use (340 GW(e))	419	Years
Years at 2050 energy use (1000 GW(e))	147	Years
Total of both sources	3.91E+07	Tons of uranium
Years at 2004 energy use (340 GW(e))	745	Years
Years at 2050 energy use (1000 GW(e))	261	Years
Unconventional resources — 1.0% sea water	4E+12	Tons of uranium
Years at 2004 energy use (340 GW(e))	72 983	Years
Years at 2050 energy use (1000 GW(e))	26 667	Years

sea water could support the existing once-through fuel cycle for 73 000 years and the projected 2050 power level for 27 000 years.

Accordingly, it can be concluded that sufficient global uranium resources are available to support a once-through fuel cycle if society makes the decision to continue to produce electricity in this manner.

3.4. Possible new reactor technologies

Research continues in a number of areas in reactor technology that could extend uranium resources, as described below.

3.4.1. Improving reactor burnup

An important aspect for existing reactors is to improve fuel burnup, i.e. the amount of energy produced per unit mass of fuel. The amount of burnup varies with reactor design and fuel management technique. In the USA, PWRs reach a burnup of approximately 50 GW · d/tHM (where GW · d is the annual thermal energy output and tHM is the amount of fuel loaded per year). Improving burnup decreases the mass of fuel loaded and discharged per unit of electric energy produced. A number of factors are involved in increasing burnup, including improved fuel cladding, better neutronics management and

increased enrichment. Alternative fuel concepts may allow for higher burnup while increasing reactor safety margins and enhancing proliferation resistance.

For example, Pacific Northwest National Laboratory has applied for a patent on an LWR concept based upon spherical fuel elements measuring a few millimetres in diameter, which are coated to prevent water incursion and to contain all fission products. The first concept under this scheme is for a 100 MW(e) reactor that would operate for 60 years on a single fuel charge, using low enrichment fuel and keeping all fuel within the pressure vessel for the full life of the plant. The reactor concept features cross-flow through constrained beds of these microfuel elements, with internal fuel transfers as required to sustain reactivity between a fresh fuel hopper, annular chambers within the reactor core and a spent fuel storage basin. The basic idea for this is shown in Fig. 2.

Such a concept offers ideal arrangements for proliferation resistance, for fuel efficiency and for safe and reliable operation. It is intended for use in developing countries and perhaps also in industrialized States.

3.4.2. Reuse of spent fuel

A new fuel cycle concept, DUPIC (Direct Use of Spent PWR Fuel in CANDU), is under development. The basis of this fuel cycle alternative is that spent fuel from LWR reactors still contains enough fissile material to be reused in CANDU reactors by direct refabrication, without separating the fissile material from the fission products.

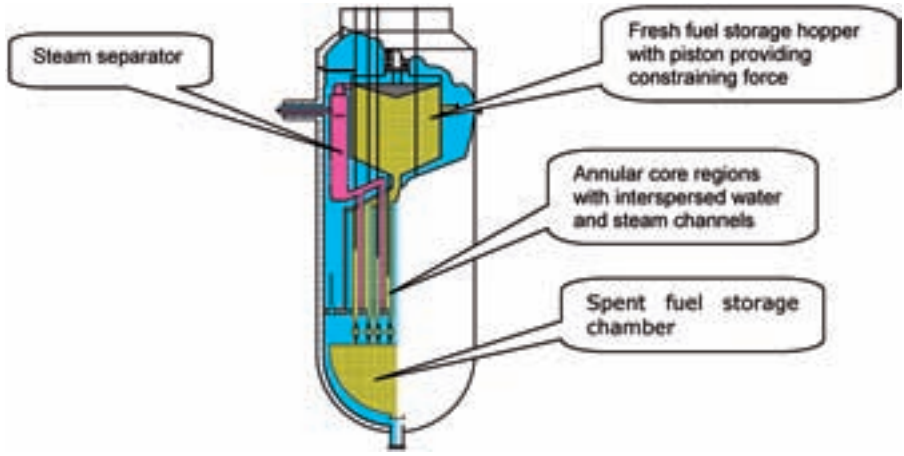

FIG. 2. The Atoms For Peace Reactor (AFPR).

The DUPIC fabrication process involves disassembling and decladding spent LWR fuel using thermal and mechanical processes. The material is then reformulated as powder, and the pellet and rod manufacturing processes are performed similarly to conventional fuel manufacturing. The manufacturing process must be performed remotely in hot cells, because the materials will still be highly radioactive [8].

3.4.3. Future reactor technologies

As shown above in Section 2.2, LWRs currently make up almost 80% of the reactors in operation, and they produce almost 90% of nuclear generated electricity. Evolutionary reactors are under development in a number of countries under national programmes, such as the IAEA INPRO programme or the Generation IV International Forum (GIF). For example, six reactor types are currently under consideration for Generation IV:

(1) Gas cooled fast reactor systems (GFRs);
(2) Lead cooled fast reactor systems (LFRs);
(3) Molten salt reactor systems (MSRs);
(4) Sodium cooled fast reactor systems (SFRs);
(5) Supercritical-water-cooled reactor systems (SCWRs);
(6) Very high temperature reactor systems (VHTRs).

The GFR system operates with a fast-neutron-spectrum helium cooled reactor and has a closed fuel cycle. The LFR system is a fast-spectrum lead or lead/bismuth eutectic liquid-metal-cooled reactor with a closed fuel cycle. The MSR system produces fission power in a circulating molten salt fuel mixture with an epithermal spectrum reactor and a full actinide recycle fuel cycle. The SFR system features a fast-spectrum sodium cooled reactor and a closed fuel cycle. The SCWR system is a high temperature, high pressure, water cooled reactor that operates above the thermodynamic critical point of water. The SCWR may have either a thermal or a fast spectrum, so it offers two fuel cycle options: an open cycle with a thermal spectrum reactor or a closed cycle with a fast spectrum reactor. The VHTR is a graphite moderated, helium cooled reactor with a once-through uranium fuel cycle. It will supply heat with core outlet temperatures of 1000°C for efficient hydrogen production and for other processes.

Of these types, two are advances on technologies currently in service; the SFR is an advance on LMRs, and the VHTR on HTGRs. Two of the proposed types are open cycle reactors with the same type of fuel requirements as the current LWR technology: the SCWR in a thermal spectrum mode operation, and the VHTR. All of these reactor types increase uranium utilization by

operating with greater thermodynamic efficiency, while the reactors with closed fuel cycles also increase uranium utilization by recycling the spent fuel.

4. PROLIFERATION CONCERNS

Recall that nuclear weapons are manufactured using HEU and/or plutonium. The once-through strategy allows possibilities for both.

Although the current commercial once-through nuclear fuel cycle is not considered an intrinsic proliferation risk, throughout the fuel cycle there are opportunities to divert material. Figure 3 shows the different elements of the fuel cycle and the different amounts of material that would have to be diverted at each step, as well as the subsequent operations that would have to be undertaken, to support a programme producing two weapons per year.

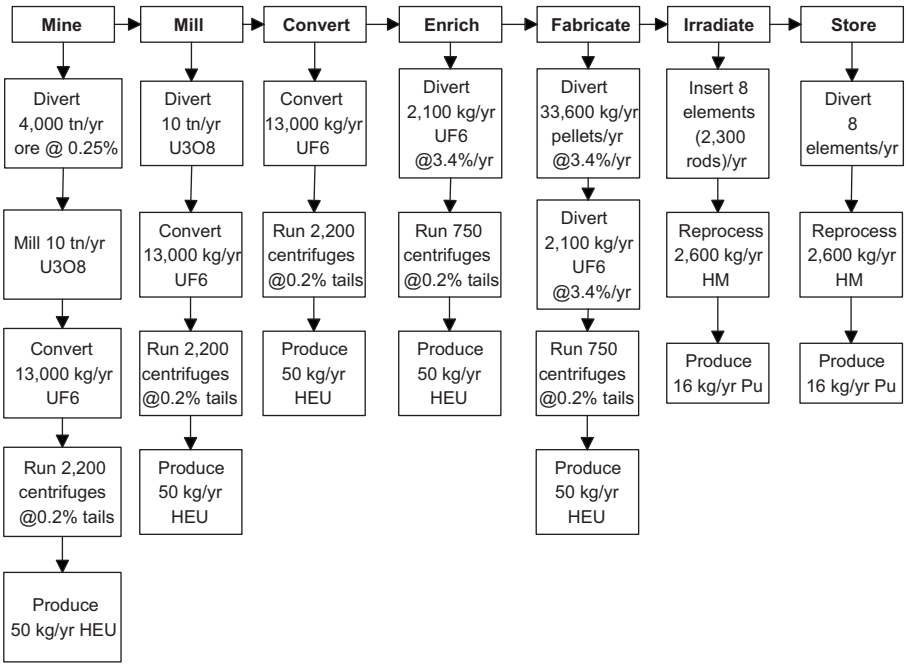

FIG. 3. Diversion facilities and throughput.

4.1. Uranium acquisition and enrichment

As described in Section 3.2, raw uranium (U_3O_8) is abundantly available, and large amounts of raw uranium can be obtained outside of the normal supplier arrangements.

To produce uranium weapons, a State would probably seek to acquire enrichment technology through whatever means are available to it, under the guise of its 'inalienable rights' as bestowed under Article IV of the Non-Proliferation Treaty (NPT). Having acquired enrichment technology, the State might choose to:

(a) Build a clandestine plant hidden from the world and thus not subject to IAEA safeguards;
(b) Use a declared and safeguarded plant to produce HEU by reflux (feeding the product back into the input until the enrichment level reaches the desired level);
(c) Have an HEU cascade concealed within a normal LEU plant;
(d) Produce excess LEU as a means of reducing the size of the clandestine plant.

All of these options raise possibilities for detection and, if detected, may bring intervention. That being said, a well concealed centrifuge plant could be very difficult to detect, and the most likely scenario involves acquiring centrifuge technology under the guise of peaceful nuclear programmes, mastering the technology, building a concealed plant for weapons use and proceeding to pass the threshold.

From the viewpoint of a State proliferation programme, any technology capable of producing HEU is suitable. A technology that is efficient and easily concealed would be preferable, but Iraq showed that even electromagnetic separation technologies (thought to be so inefficient as not to be a problem) might be pursued because technological information was available. Now, centrifuge technology has been made available. In the future, laser and plasma enrichment technologies may offer compelling commercial enrichment advantages but may also create additional means for proliferation.

Two primary means of enrichment have been successfully used to create material for weapons: gaseous diffusion and gaseous centrifugation. The use of other technologies has also been attempted; especially electromagnetic isotope separation, aerodynamic isotope separation and laser isotope separation. These technologies will be discussed briefly.

4.1.1. Gaseous diffusion enrichment

Originally, gaseous diffusion enrichment required extremely large, centralized, facilities, which were considered to be available only to major industrialized countries. Only six countries have successfully constructed gaseous diffusion facilities: the USA, the former USSR, the United Kingdom, China, France and Argentina. Gaseous diffusion plants are as efficient as centrifuge plants, and, because of their high power consumption, more readily detected.

Although most recent commercial enrichment facilities have been based on centrifuge technology, the large diffusion plants in France and the USA still make up 48% of total installed enrichment capacity.

4.1.2. Gaseous centrifuge enrichment

Centrifuge enrichment is currently the technology of choice for new enrichment facilities. Eight countries have established commercial enrichment facilities. Two new plants of commercial scale are planned in the USA: one will use indigenous technology and the other will be a Urenco partner.

The revelation of black market transactions involving centrifuge enrichment, UF_6 supplies and blueprints for certified uranium nuclear weapon designs, coupled with the inability to detect clandestine programmes that have been under way for years, has led to a reconsideration of how the benefits of nuclear energy can be made available without raising the risks of proliferation and nuclear terrorism. While there has been, and continues to be, an intense focus on plutonium weapons, at least for the present, uranium weapons acquired through centrifuge enrichment represent the greatest threat.

4.1.3. Electromagnetic isotope separation

Only the USA has successfully utilized electromagnetic isotope separation in a production enrichment process, and that was in a weapons programme. Its relatively high cost caused the USA to drop the programme when gaseous diffusion plants came on-line. The former USSR was never successful in using this technology in any significant way to produce enriched uranium.

4.1.4. Aerodynamic isotope separation

Three countries, Brazil, Germany and South Africa, have attempted to use aerodynamic isotope separation for enrichment. Only South Africa was successful, using the technology both for fuel enrichment and as a part of their

weapons programme. However, it was too expensive to compete with either gaseous diffusion or centrifugation, and South Africa halted its programme when it could import enriched uranium.

4.1.5. Laser isotope separation

A number of types of laser isotope separation technologies have been attempted. These include:

— Atomic vapour laser isotope separation (AVLIS/SILVA);
— Molecular laser isotope separation (MLIS);
— Chemical reaction isotope selective laser activation (CRISLA);
— Separation of isotopes by laser excitation (SILEX).

Active development has slowed on all technologies except SILEX. Although none of these technologies have fulfilled the expectations placed on commercial enrichment, it is possible that they could be used in a surreptitious fashion in a small (expensive) weapons programme.

4.2. Plutonium diversion

Plutonium diversion requires spent (irradiated) fuel and a reprocessing facility. All nuclear reactors produce irradiated fuel containing plutonium (other weapons usable materials may also be created, including ^{233}U, Np and Am, depending on the materials irradiated). The suitability of that plutonium depends upon the extent of the irradiation and the neutron energy spectrum of the reactor. Reactors capable of being refuelled while in operation (on-line reactors) are able to discharge spent fuel with low burnup values without the extensive and expensive refuelling disruptions of reactors that have to stop operation and be partially dismantled to remove the irradiated fuel.

Reprocessing involves releasing gaseous fission products and particulates, some of which are released from reprocessing plants. In addition, the wastes may be carried away from the plant in streams. Detection of these emissions is less challenging than enrichment.

While the once-through fuel cycle does not involve reprocessing, a State may acquire reprocessing technology and build a clandestine plant. It could then divert its spent fuel (either complete fuel assemblies or individual fuel rods removed from fuel assemblies; alternatively, it might be able to irradiate undeclared fertile material in its reactors for this purpose). Reprocessing technology is based on standard chemical operations. Literature on the chemistry and equipment required is widely available.

5. CONCLUSIONS

The following conclusions can be reached about the continuation of the once-through fuel cycle.

5.1. Sustainability of nuclear power production with existing and predicted uranium supplies

There are sufficient uranium resources available to support the once-through fuel cycle through at least the next 300 years. If other resources are used, i.e. the uranium in phosphates or sea water, there is essentially no limit on fuel availability. The costs for uranium recovery will increase as relatively rich ores are depleted. New technologies for extraction could make sea water recovery viable.

5.2. Technical and economic challenges of maintaining the nuclear power fuel cycle

Waste management would appear to be the strongest motivation for seeking to close the fuel cycle. Repositories are enormously expensive, and finding locations that are acceptable to governments (national and regional) — and to the public — provides a strong motivation to recycle and to reduce toxicity through transmutation.

5.3. Proliferation concerns

Superficially, the idea that a State might use 1000+ MW(e) LWRs and fuel cycle facilities scaled for such an enterprise to support a nuclear weapons programme seems questionable. However, every nuclear capability acquired for peaceful purposes also contributes to a latent proliferation capability.

Every nuclear energy system using enriched uranium creates a need for enrichment services. Once a State has acquired such a capability for legitimate peaceful use, it will also possess the ability to misuse its declared enrichment plants or to replicate the technology in clandestine plants. A State possessing this technology may become a source for other States — or terrorist organizations — to acquire enrichment.

One distinct disadvantage of closed cycles over the open cycle is that they include declared recycle operations, which currently produce separated and purified plutonium. Any nuclear power programme that leaves spent fuel issues unresolved leaves unanswered the question of what the State will do, as it eventually will have to do something.

Proliferation risk is relative. Every nuclear activity increases the knowledge of a State and enhances its technological capabilities. The acquisition of fissile material for a weapons programme requires many of the same processes, facilities and operations as needed for a peaceful nuclear programme. Accordingly, every peaceful nuclear activity carries with it a risk of contributing to a nuclear weapons programme through technology development or skills acquisition. States having indigenous enrichment or reprocessing capabilities already possess the means required to proliferate. States stockpiling separated plutonium and enriched uranium, especially HEU, have the materials available and could, therefore, produce nuclear weapons in a short period of time should they so choose.

Nuclear energy programmes should be carried out under arrangements that are demonstrably 'peaceful' in nature. A State can signal its peaceful intentions by pursuing nuclear energy under the following four principles:

(1) The State should have a sound programmatic basis for its nuclear power activities, including the justification for all the operations it intends to pursue. It should make public this justification.
(2) The State should not conduct 'peaceful nuclear operations' in secret.
(3) The State should engage in open cooperation with States with exemplary non-proliferation records, including the provision of facilities, equipment, components and materials.
(4) The State should have an Additional Protocol in force as a part of its IAEA Safeguards Agreement, and the IAEA should have reached a positive conclusion regarding safeguards implementation at the State level.

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PARTITIONING OF FISSILE AND RADIOTOXIC MATERIALS FROM SPENT FUEL: AN OVERVIEW

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Abstract

The term 'partitioning' means separation of one group of radioactive waste components from the others. These technological approaches are used mainly for extraction of long lived fission products (Tc and I) and minor actinides (MAs: Np, Am and Cm) from wastes after spent fuel reprocessing. Separated MAs must be transmuted by reactor or accelerator systems. The paper includes a brief review of information related to the main partitioning flowsheets developed in France, Japan, the Russian Federation and other countries. Recent approaches to partitioning are mainly directed towards reduction of the environmental hazards associated with radiotoxicity. In future, partitioning must be closely combined with reprocessing and other spent fuel treatment processes. Not only U, Pu and Th can be considered as recycled nuclear materials, but also Np, Am, Cm and Pa could be considered as fuels for some reactor systems. The development of new technologies must be directed in future towards the complete reprocessing and partitioning of fissile and radiotoxic materials from spent fuel. The technological optimization tasks can be formulated as follows: (a) reprocessing/partitioning for non-proliferation, (b) partitioning with minimum environmental impact, (c) partitioning by advanced and economic methods.

1. INTRODUCTION

The problem of the management of the various wastes arising as a result of human activity increased sharply in the second half of the last century. It has also related to atomic engineering in spite of the fact that the physical volume of its waste products is appreciably less than those of other power branches. The key problem of the management of accumulated radioactive waste (RW) and the study of radionuclide behaviour in the environment are described. The problem of reducing the radiotoxicity of RW has been actively studied, resulting in the partitioning and transmutation (P&T) approach.

The main goals of P&T are formulated as the following [1]:

(a) A reduction of the hazard associated with spent fuel over the medium and long terms (more than 300 years) by a significant reduction of the inventory of plutonium and minor actinides (MAs);
(b) A reduction of the time interval required to reach the reference level of the radiotoxicity inventory by recycling transuranic elements (TRUs);
(c) A decrease of the spent fuel volume by separation of uranium to enable more efficient storage or disposal. This should result in an increase in the effective capacities of final repositories. However, this approach might require special handling of strontium and caesium after partitioning.

Thus, the problem of reduction of the amount of RW and its radiotoxicity was separated into two subtasks. This paper is dedicated to one of these, namely: partitioning.

It is necessary to emphasize that the great volume of fundamental and applied research on the problem of management of RW is carried out in the following directions [2]:

(a) Creation of essentially new processes for RW treatment and spent nuclear fuel (SNF) reprocessing based on the use of effective extractants and selective sorbents, and of new methods for their realization;
(b) Partitioning of the current and accumulated high level liquid RW (HLLW) with the purpose of extraction of the actinide element fraction;
(c) Creation of new matrices and methods for immobilization of radioactive isotopes;
(d) Development of principles for the selection and creation of new types of natural and technical barriers to enable safe SNF storage and RW disposal.

The term 'partitioning' means, as a rule, the separation of one group of RW components from the others. The basic objective of this process in nuclear technology is to separate long lived components of high level waste (HLW) (fission products: Tc and I; minor actinides: Np, Am and Cm) from short lived ones. Being able to do this task will allow the problem of the long term radiotoxicity of waste products to be solved, as long lived components should be destroyed by the special method that we call 'transmutation'.

The following elements and isotopes are considered within the framework of the partitioning task [1]:

(a) All the isotopes of Np, Am and Cm **(**evidently Pa can also be considered for partitioning into Th cycle systems);
(b) Technetium-99, ^{129}I and ^{135}Cs (from fission products);

(c) Some other isotopes (^{79}Se and ^{93}Zr) — although their partitioning is not necessary;
(d) Activation products (^{14}C and ^{36}Cl).

Technologically, partitioning is the fine separation of components. From the point of view of reduction of RW radiotoxicity it should give good results. However, if PUREX type processes are used as the technological basis of the approach, they will lead to the formation of a great volume of low level liquid RW (LLLW), which makes partitioning economically unprofitable due to the increase in cost because of the additional treatment of wastes. On the other hand, partitioning will allow use of approaches typical of radiochemistry and to achieve a high recovery of the more dangerous components. However, it is necessary to take into consideration the fact that the methods of extraction and separation of the minor actinides and the various fission products were developed earlier and for other purposes [3].

The study of partitioning has generated a great volume of research in many countries during recent decades. The French programme has been particularly productive [4] (including development of the GANEX concept), with other European Union (EU) countries, the United States of America (USA), the Russian Federation, Japan, China and India also having national activities in the field of partitioning of RW.

Many publications are dedicated to the study of partitioning. Seven information exchange meetings have been organized by the OECD Nuclear Energy Agency (OECD/NEA) (Mito, Japan in 1990; Argonne National Laboratory (ANL), USA in 1992; Cadarache, France in 1994; Mito, Japan in 1996; Mol, Belgium in 1998; Madrid, Spain in 2000 and Jeju, Republic of Korea in 2002 [5]). The OECD/NEA issued a very comprehensive report on the P&T problem, which covered all the results obtained up to 1999 [6]. The IAEA has supported activity in the field of P&T and also issued some key TECDOCS [7] and reports [1, 8]. Namely, IAEA Technical Report No. 435 is one of the latest overviews of problems in partitioning.

The objective of this paper is not to carry out a universal review of completed studies. However, in the author's view it is necessary to analyse the general results of research and to develop proposals for future directions of research and development. The main aims of this paper are to consider the following questions:

(a) Which partitioning processes will be used in the future?
(b) What requirements should be put forward for new technologies for the realization of a reduction of HLW with a simultaneous reduction of new RW formation?

(c) Should the technological philosophy of partitioning be connected to tasks related to the steady supply of power with a high level of safety?

2. PARTITIONING AS A WAY OF REDUCING THE RADIOTOXICITY OF HLLW

Partitioning of wastes is closely connected with reprocessing of irradiated nuclear fuel. The method for reprocessing of spent fuel from thermal neutron reactors used in all countries is based on aqueous extraction flowsheets with production of purified uranium and plutonium. These processes have been used in military programmes and have retained their specific shortcomings.

The PUREX process is now very well developed and is used industrially in France, India, Japan, the Russian Federation and the UK. Each country has its own variants, but, as a rule, the processes used differ only in their details. The general flowsheet of products at a UP-3 reprocessing plant in France is shown in Fig. 1.

The reprocessing at the RT-1 facility (Mayak plant, Russian Federation) differs in that full extraction of the Np and Cs–Sr fractions is already realized [9] (Fig. 2). However, this plant generates an appreciably greater HLW volume in comparison with the La Hague plant.

Other methods for reprocessing irradiated fuel [10] have not been realized yet on an industrial scale, and consideration of them will be carried out below.

The MAs to be considered are Np, Am and Cm. Very high separation efficiencies are required to reduce the long term radiotoxicity of HLW by a significant factor. Since the MAs constitute the source of long term radiotoxicity, their removal from HLW before vitrification is a necessary step in a partitioning strategy.

Recent aqueous reprocessing of spent fuel has a separation efficiency in the range 99.8–99.9% for U and Pu, and this recovery rate can be improved to 99.9%. The goal of the MA partitioning step should be a separation efficiency of 99.9% (i.e. a decrease of the MA content of HLW by a factor of 1000).

2.1. Activity in the field of partitioning

Activity in the partitioning field related to recent reprocessing technology can take place along three directions:

(1) Modification of the PUREX process for direct extraction of long lived component fractions;

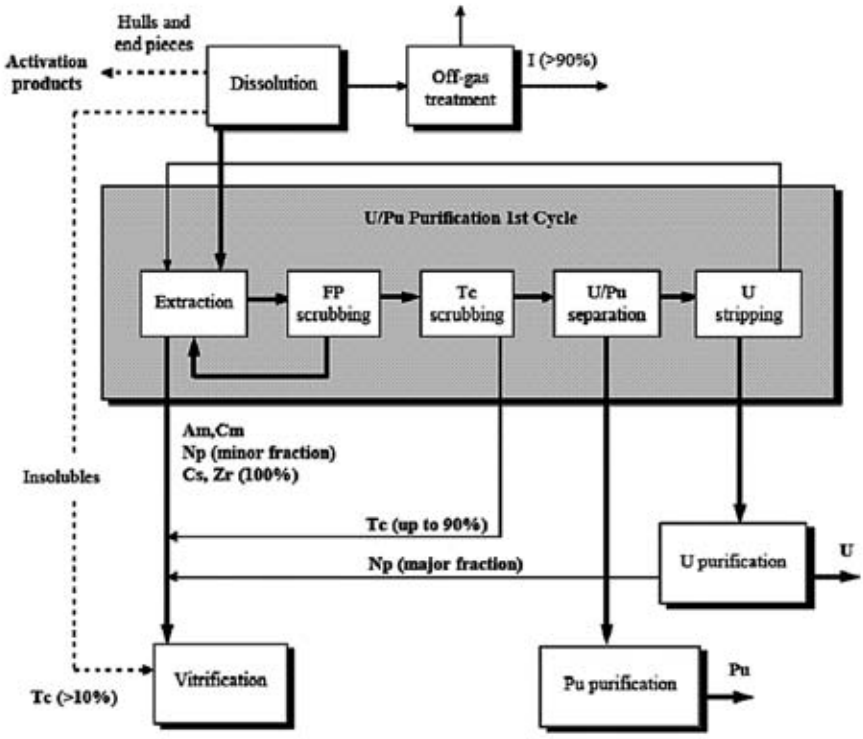

FIG. 1. Distribution of basic components, minor actinides and fission products in the technological flowsheet of the PUREX process realized at the UP-3 plant, France [6].

(2) Development of new extraction processes for partitioning of the necessary fractions from HLLW streams formed by the PUREX process without updating the basic technology.
(3) Development of separate technologies for treatment of HLLW after the PUREX process.

It will be shown in a number of examples that it is possible to realize all these variants.

2.1.1. Partitioning as a modification of the PUREX process

An example of activity in the first direction is the Russian work on development of an improved flowsheet for modernization of the RT-1 plant possible when the PUREX process is modified [11]. The task is achieved with a complex method involving partitioning and waste volume reduction (Fig. 3).

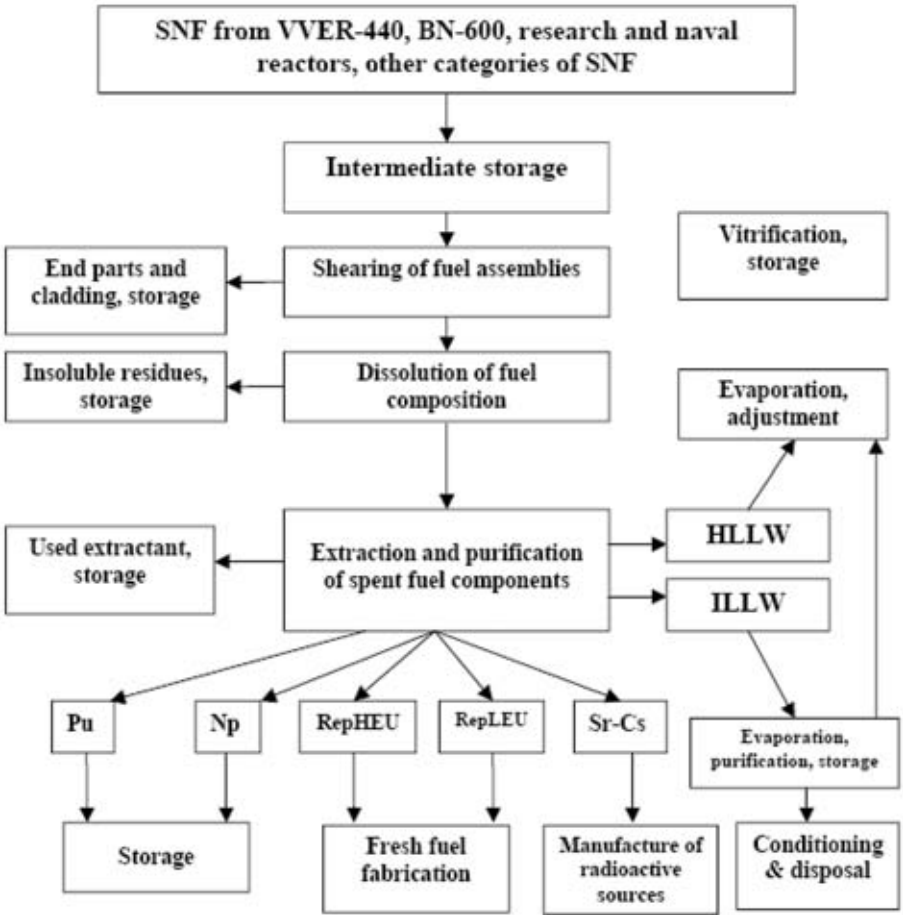

FIG. 2. Principal flowsheet of the RT-1 reprocessing facility at the Mayak plant, Russian Federation [9].

Among the waste processes allowing Cs, Sr and TPE extraction from highly saline acid HLW, three groups of processes are of the most interest for implementation of the RT-1 technological flowsheet:

(1) Processes based on application of macrocyclic compounds, primarily crown ethers;
(2) Processes using liquid cation exchangers, mainly chlorinated cobalt dicarbolite (CCD);

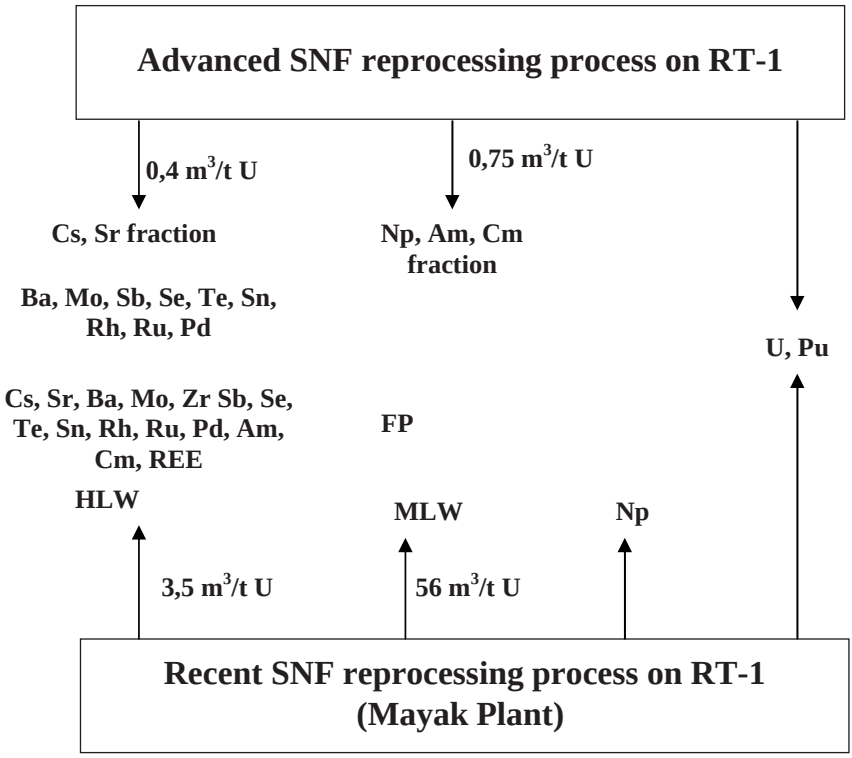

FIG. 3. Comparison of volumes and compositions of RW generated at different SNF reprocessing versions for existent and advanced flowsheets of the RT-1 plant.

(3) Processes using diphenyl-N, N-dibutyl-carbamoyl-methylene-phosphine-oxide solutions in a heavy nitroaromatic diluent-metanitrobenzotrifluoride.

The possibility of commercial application of HLW partitioning technologies at the Mayak plant was demonstrated at the end of the 1980s. Experimental–industrial checking of TPE and rare earth element (REE) extraction from the actual solution was carried out. Forty cubic metres of solution were processed; 10 m^3 of TPE re-extract and 10 m^3 of REE re-extract were received. Radionuclide extraction to the extract came to 97%. In the second stage of work, TPE re-extract was concentrated to 2.4 m^3; this concentrate contained 240 g of ^{241}Am and 20 g of Cm. This is the only system that passed large scale running-in on acute solutions in the Russian Federation [11].

Other examples of systems modified on the basis of PUREX and with new flowsheets are SUPERPUREX or the technology of united reprocessing and partitioning (PUREX–TRUEX) [12]. PUREX–TRUEX is a promising technology which provides reprocessing and partitioning within the framework of a single extraction cycle using dihexyl-diethyl-carbamoyl-phosphonate – $(HexO)_2/Et_2$ (Fig. 4).

2.1.2. Partitioning of MAs from aqueous reprocessing streams

The objectives and strategy of MA partitioning from HLLW are shown graphically in Fig. 4 (Institute for Transuranic Elements (ITU), Karlsruhe) [13]. The following paragraph is based mainly on material from Refs [1] and [6].

In an MA partitioning scheme, two main routes can be proposed (Fig. 5). The optimal strategy is, of course, a single partitioning process in which MAs are selectively extracted directly from the PUREX raffinate, HLLW. This requires an extractant capable of selective and efficient separation of the MAs at high acidities (>2M HNO_3) in a highly radioactive solution containing all the fission products, among them lanthanide elements in a mass excess of 20 times compared with MAs. To achieve the necessary specificity, however, of the extractant for MAs over lanthanides is extremely difficult to accomplish at such high acidities, and such a process is at present more at a conceptual level. Partitioning of MAs involving co-extraction of lanthanides and a subsequent separation of the two element groups is therefore the main option considered. (The shorter lived TRUs (Bk, Cf, etc.) are not considered here.)

2.1.2.1. Partitioning of neptunium

Recovery of ^{237}Np from the U–Pu product stream is technically possible in the PUREX process and has been realized at the Russian reprocessing plant

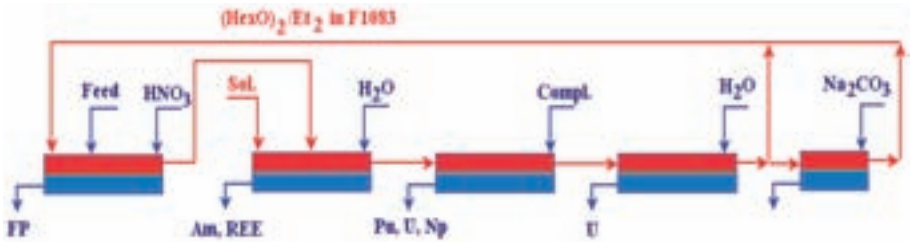

FIG. 4. Flowsheet for the PUREX–TRUEX combined process (ITU, Karlsruhe).

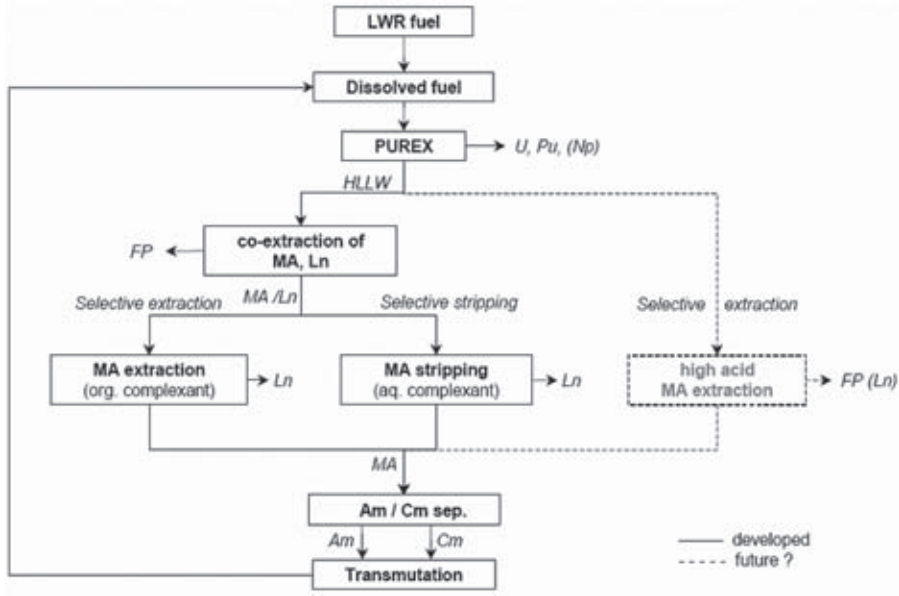

FIG. 5. Strategy for MA partitioning from the aqueous waste stream after the PUREX process [13].

RT-1. At the French reprocessing plant at La Hague, Np is partly discharged with the fission products into the HLLW and partly associated with the U, Pu and Np streams in TBP during current reprocessing operations. The purification of U and its quantitative separation from Np is achieved in the second extraction cycle of the PUREX process. It would be advantageous to modify the parameters in the first extraction cycle in order to co-extract the three actinides (U, Pu and Np) quantitatively and to recover the purified Np stream directly during reprocessing. Quantitative recovery of Np from dissolved spent fuel streams could be applied without expensive modifications of reprocessing equipment.

2.1.2.2. Partitioning of americium and curium from the HLLW resulting from spent light water reactor fuel

During conventional reprocessing operations, most of the MAs (Am + Cm are quantitatively more than 99.5%) are transferred to HLLW. The partitioning of Am (+Cm) from HLLW is the first priority from the radiotoxic point of view. The separation of ^{241}Am obviously also implies separation of the long lived ^{243}Am. Partitioning of all the MAs from HLLW is currently under

investigation in many laboratories around the world (in China, France, India, Japan, the Russian Federation and some other countries), having been first studied at US national laboratories (ANL and Oak Ridge National Laboratory), at Hanford and at Russian Institutes (the Khlopin Radium Institute, the Bochvar Institute, RIAR and the Mayak plant).

The Am and Cm fractions contain all the REEs, which are about 10–20 times more in weight than the actinides, depending on burnup. A number of processes have been studied and hot tested; among the most important are the TRUEX, DIDPA, TRPO, TALSPEAK and DIAMEX processes for actinide–lanthanide group separation, coupled to the Cyanex 301, SANEX, ALINA and BTP (bis-triazinyl-1,2,4-pyridines) processes, which allow actinide–lanthanide separation [1].

The most important criterion to be used in ranking the different methods is the overall DF obtained during extraction of HLLW and its comparison with the required DFs in order to reach the 100 nCi level of alpha active radionuclides in the HLW. The highest DFs should be reached for ^{241}Am separation, namely 3.2×10^4 if immediate separation is scheduled. However, this option cannot yet be realized in industrial facilities, in which DFs of 10^3 are the realistic limit. Another criterion is to minimize the feed conditioning in order to generate secondary waste during partitioning operations.

For transmutation in a fast neutron flux one would not need such high purification, since enhanced parasitic neutron capture of certain fission products does not exist in a fast neutron spectrum.

High level waste free of actinides, or at least HLW depleted of actinides, could be produced by vitrification plants and stored for cooling in surface facilities followed by geological disposal. There are no objective arguments to oppose geological disposal of such a waste stream, which decays with more than four orders of magnitude over 500 years.

2.1.2.3. Status of some partitioning processes

Two candidate processes, namely TRUEX and DIAMEX, are being tested at the Joint Research Centre–ITU at Karlsruhe. Both processes have the potential to obtain DFs ~ 10^3 for MA removal from acidic HLW raffinate [8]. The DIAMEX process represents the best compromise among the first series of methods. In laboratory conditions DFs ~ 10^3 have been obtained for the MAs from 3.5M acid concentrated HLW. The TRUEX process is also very effective for alpha decontamination of medium level and non-heating high active waste (HAW) streams. Except for the ex-military US facilities, in which kilogram scale separations have been performed, the present research facilities, in the EU, have strong limitations with regard to the quantities of MAs that can

be handled in shielded facilities; for example, the new MA laboratory of the Joint Research Centre–ITU is authorized for a maximum of 150 g of ^{241}Am and 5 g of ^{244}Cm [1].

The TRUEX process (Fig. 6), which was developed in the USA in the 1980s and is now being studied in India, Italy, Japan, the Russian Federation and the USA, is based on the use of the CMPO (octyl-phenyl-di-isoburyl-carbamoylmethyl-phosphine-oxide) extractant. The Russian version of the TRUEX process assumes use of a 0.06–0.12 mol/L solution of diphenyl-N, N-dibutylcarba-moylmethylene-phosphino-oxide in a heavy nitroaromatic diluent–metanitrobenzo-trifluoride [14].

The DIAMEX (Fig. 7) process was developed in France and is now under investigation in France, Germany, India, Italy, Japan and the USA; it is based on the use of a malonamide extractant. A process based on a new type of diamide, a diglycolamide, which is a terdendate ligand having better affinity for

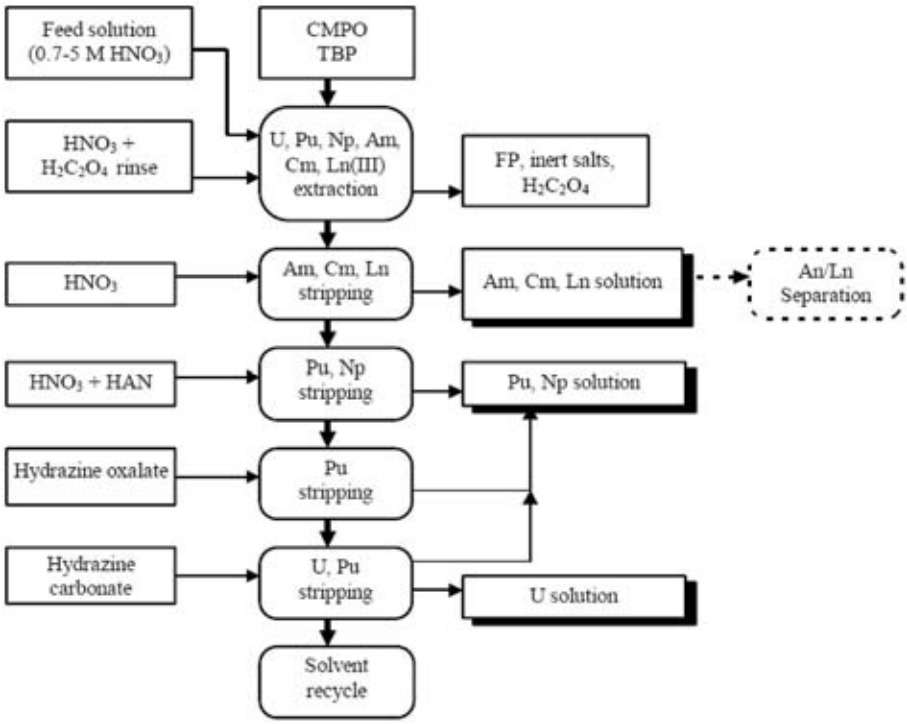

FIG. 6. The TRUEX process [6].

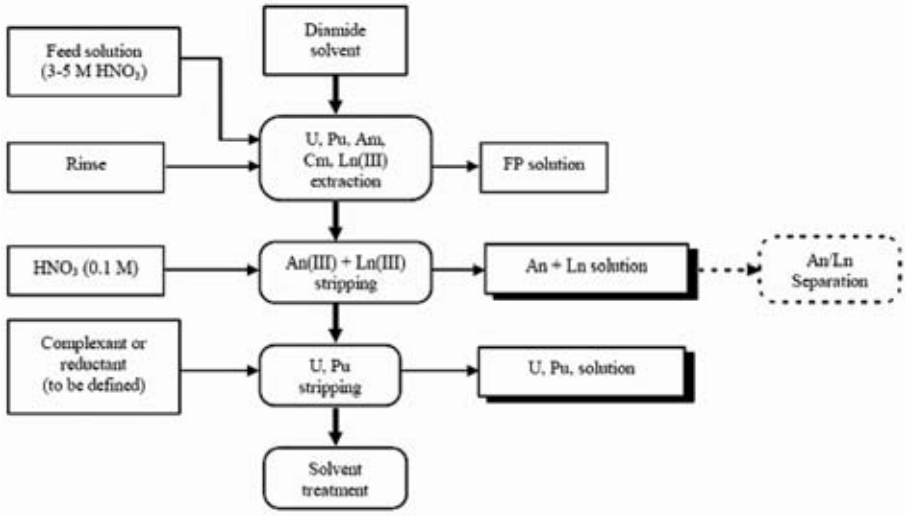

FIG. 7. The DIAMEX process [6].

trivalent actinides (An(III)) than the malonamide extractant, is under development in Japan [1].

An(III)–Ln(III) separation by TALSPEAK and CTH processes. The TALSPEAK process (Fig. 8), which was developed in the USA in the 1960s and then adapted (as the CTH process) in Sweden, can be considered as the reference process for An(III)–Ln(III) group separation. In the Russian Federation, this process is used in modified form at RIAR for purification of Cm and other actinides in the chain of Cf isotope production [15]. It is based on the use of HDEHP (di-2-ethyl-hexyl-phosphoric acid) as extractant and DTPA (diethylene-triamine-penta-acetic acid) as the selective An(III) complexing agent. The advantages of this process are the great experience that has been gained in it worldwide and its high efficiency.

Possible realization of An(III)–Ln(III) separation by SANEX process. The Cyanex 301 process is under investigation in China, Germany and India; the extractant consists of a dialkyl-dithiophosphinic acid (R2–PSSH, where R is an alkyl group). After the discovery in Germany of the astonishing properties of BTPs for An(III)–Ln(III) separation, successful hot tests have been achieved at both the Marcoule site of the French Commissariat à l'Énergie Atomique (CEA) and the ITU using n-propyl-BTP. A good efficiency for the BTP process was obtained.

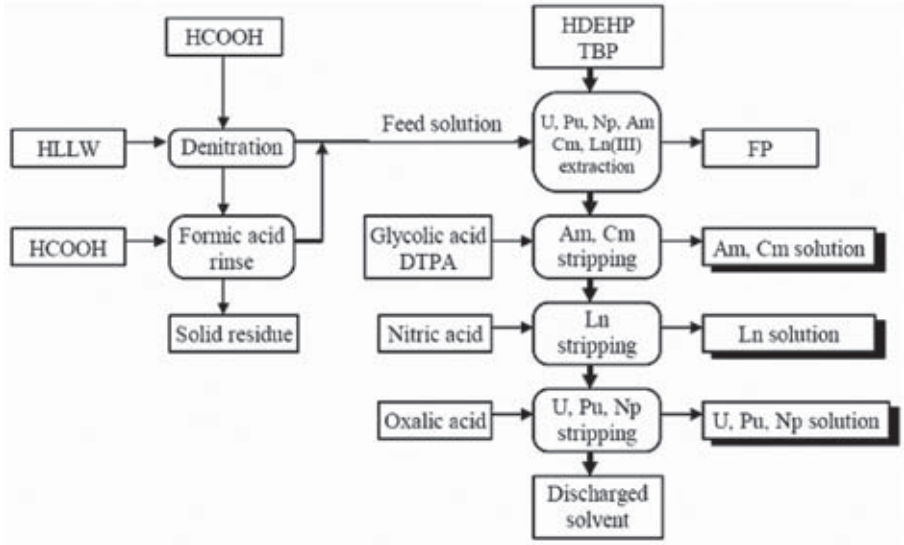

FIG. 8. The TALSPEAK process [6].

Nevertheless, it was shown at the CEA during a hot test that n-propyl-BTP is not sufficiently stable against hydrolysis and radiolysis to be proposed as an industrial extractant.

Great importance of americium and curium separation processes. For this step, processes based on selective oxidation of Am at the +VI or +V oxidation states have been developed, the Cm remaining unchanged as Cm(III), allowing simple Am and Cm separation processes to be defined:

(a) In the SESAME process, under strong oxidizing conditions, Am can be oxidized from Am(III) to Am(VI). This can be done, for example, by electrolysis in the presence of heteropolyanions acting as a catalyst. The Am(VI) generated can be separated from Cm(III) by extraction, for example, using TBP.

(b) In Am(V) precipitation, selective precipitation of the double carbonate of Am(V) and potassium is one of the oldest methods for Am and Cm or Am and lanthanide separation; it was developed at the end of the 1960s in the USA, used in the Russian Federation and is currently under development in Japan. This method requires the use of a 2M/L K_2CO_3 solution in which a mixture of Am(III) and Cm(III) is dissolved. After chemical or electrochemical oxidation of Am(III) to Am(V), Am(V) precipitates from the solution as solid crystalline $K_5AmO_2(CO_3)_3 \cdot nH_2O$,

while Cm(III) remains in solution. After filtration, Am is separated from Cm. This process is simple, selective for Am and has been used worldwide.

2.1.2.4. Comparative testing of advanced aqueous processes

Advanced aqueous processes (TRUEX, DIDPA, TRPO and DIAMEX) have been investigated and compared [1]. The common feature of the different methods consists of HLLW processing for extraction of the MAs. For all four extractants, continuous countercurrent extraction tests have been carried out in a centrifugal extractor battery. The most efficient stripping and the highest recovery rates are achieved with the DIAMEX and TRPO processes. The DIAMEX process therefore represents the best compromise of all four processes studied, and shows good extraction and excellent stripping properties. The separation efficiencies are shown in Table 1.

The separation efficiencies of liquid extraction processes are close to 99.9% for U and Pu. It is expected that similar efficiencies could be obtained in the MA separation schemes for Np, Am and Cm. However, the Am and Cm fraction is always accompanied by the bulk of the lanthanides. These have to be separated from Am and Cm, but the efficiencies are much lower than 99.9%;

TABLE 1. SEPARATION EFFICIENCIES (%) FOR VARIOUS ACTINIDES AND FISSION PRODUCTS IN DIFFERENT CHEMICAL PROCESSES [1]

Element	PUREX (industry)	Advanced aqueous reprocessing[a] (laboratory)	Pyrochemical reprocessing (laboratory)
Uranium	99.9	99.9	99.9 (prototype)
Plutonium	99.8	99.9	99.9 (prototype)
Neptunium	95	99.9	99.9
Americium	–	99.9	99.9
Curium	–	99.3	? (>95%)[b]
Lanthanides in MAs	–	<5	<10
Caesium-135, -137[c]	–	99.9	(>99.9)[b]
Technetium-99	–	≈80% (≈99%)[b]	–
Iodine-129	98	99.9	

[a] PUREX, DIAMEX and SANEX.
[b] Numbers in brackets are Russian data [16, 17].
[c] Calixarenes.

between 5–10% of lanthanides in separated actinides may be expected. Technetium in liquid form (TcO_4^-) contains 80% of the total Tc inventory; the residual quantity remains as insoluble residues. Iodine can be efficiently separated (>99%) from the dissolver off-gases.

Unfortunately, none of these processes allow separation of lanthanide fission products from MAs. For lanthanide separation from MAs, a two step partitioning process is required in which the aqueous lanthanide–MA fraction generated from the processes mentioned above is subjected to the SANEX process, in which the MAs are selectively extracted from the lanthanide–MA fraction. Excellent results have been obtained for the BTP process [1] using the n-propyl-bis-triazinylpyridine molecule. The experiment, carried out in a centrifugal continuous countercurrent set-up, achieved an MA–lanthanide separation with an efficient scrubbing of lanthanides and produced an MA fraction almost free of lanthanides. Minor actinide extraction and back-extraction were efficient, and a reasonably good recovery of Am (>99%) was achieved. Nevertheless, this process scheme has still to be improved to increase the recovery of Cm (at present 97.6%). By means of these tests it could be demonstrated that an efficient separation of MAs from genuine spent fuel is possible in a three step process.

The main results of the French studies [4] under the programme on partitioning are based on extraction, either by adapting the PUREX process used in the industry to reprocess fuel or in developing new complementary extraction processes downstream of PUREX (Fig. 9). The scientific feasibility of these processes was established in 2001 with values of separation as high as 99.9% for Am and Cm recovered from HLW [14]. Americium and Cm are partitioned by new extracting molecules specifically developed for that purpose with the French process that was selected at the end of the scientific feasibility stage; Cs is separated with calixarene crown molecules, whereas Np, I and Tc are extracted by technical adaptations of the existing PUREX process, which is already used at the La Hague plant.

As a 100 GW(e) nuclear power plant fleet produces annually about 1600 kg each of Np, Am and Cm [1], a major chemical engineering effort will be needed to scale the laboratory methods up to a pilot scale, and subsequently to an industrial prototype scale in future advanced reprocessing plants in order to include MA separation rigs from the design phase onwards. The first steps to implement the advanced fuel cycles are the installation of separation facilities for MAs from HLLW and the conditioning of these radionuclides for intermediate storage or as a potential target material for transmutation.

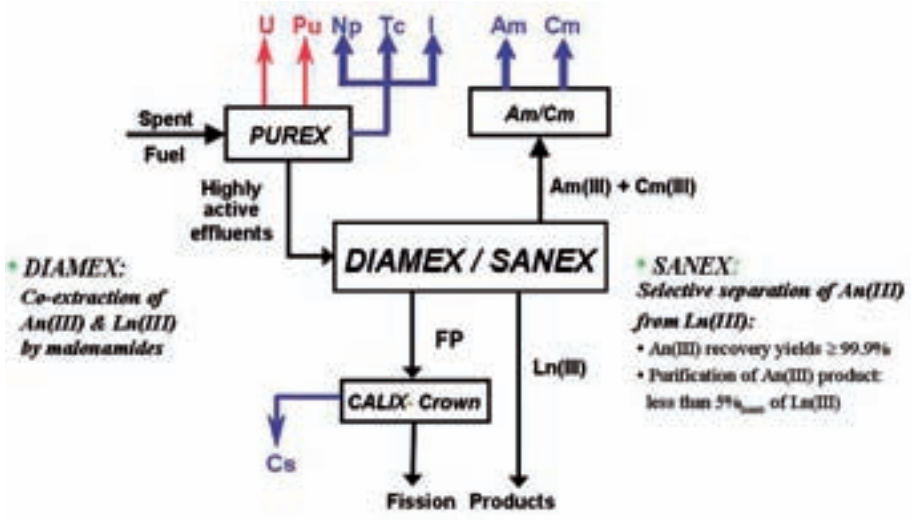

FIG. 9. The partitioning processes proposed for technological implementation of partitioning in France [4].

2.1.3. New technologies for partitioning of minor actinides from aqueous reprocessing streams

This class of technologies has a direct relation to waste treatment. There are two variants of the concepts investigated on the basis of pyroelectrochemical processing of HLW. These approaches can be considered as new processes within the framework of the management of existing HLLW. Two well known examples are considered in which HLLW is treated for the purpose of further converting it through new processes. Unlike the two previous approaches, such an approach allows the partitioning facilities to be located not only near SNF reprocessing plants (as for the aqueous extraction process) but also at other locations, for example, near fuel fabrication facilities for fast reactors.

2.1.3.1. Partitioning of HLLW by pyrochemical process

A well known concept has been developed by CRIEPI [10]. The main ideas here are conversion of HLLW after SNF reprocessing and reprocessing dried HLW by pyrometallurgical methods. Elements of this concept were taken from the US IFR concept, in which a similar process is used for reprocessing

and recycling of U–Pu–Zr fuel, and from reduction–oxidation processes investigated earlier for pyrochemical reprocessing. The flowsheet for this concept is shown in Fig. 10.

During the process, HLLW is dried and denitrified. The resulting solid powder includes oxide and metal particles, which are treated with chlorine for transformation to a molten chloride system. Some elements (mainly noble metals) are reduced to liquid Cd. As the next step, U metal is deposited (reduced) onto the solid cathode together with Zr. After recovery of uranium, the salt system is contacted by a liquid Bi cathode into which a mixture of transuranics and an amount of lanthanide fission products is recovered. This process will be tested soon on real HLLW. The experimental facility will be installed in the ITU hot cell laboratory in collaboration with CRIEPI. Reductive multistage extraction is another potential method to recover actinides or to separate actinides from lanthanides in a molten metal (Cd or Bi) system. The distribution coefficients between molten chloride salt and cadmium–bismuth have been measured. The recovery rates obtained in a laboratory facility by multistage reductive extraction are very encouraging: 99.7% for Pu, 99.8% for Np and 99.4% for Am.

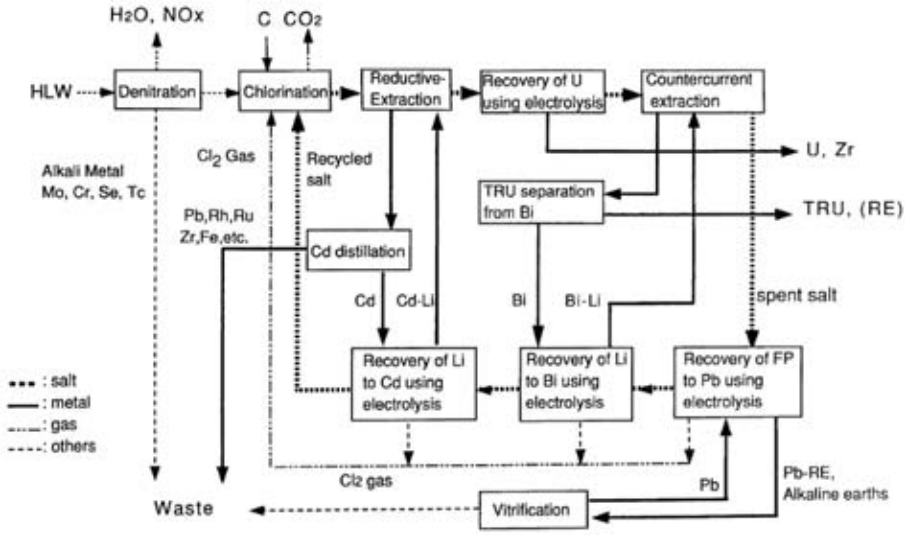

FIG. 10. Pyrochemical partitioning of HLW by the CRIEPI process [1].

2.1.3.2. Partitioning of HLLW in the UREX and PYRO processes

In the USA, LWR UO_2 spent fuel would be the material subjected to the conventional 'chop and leach' process in order to eliminate by the UREX process (aqueous extraction) (Fig. 11) the bulk of the U, Tc and I from the dissolved fuel mixture [18].

The raffinate of the extraction containing the fission products and the TRUs (≈6% of the initial mass) is calcined and transferred to the pyrochemical section of the plant. The calcined oxide mixture of fission products and TRUs then undergoes a lithium reduction step with Li–LiCl in a furnace at 650°C. The alkaline fission products Cs and Sr remain in the salt mixture and are treated as

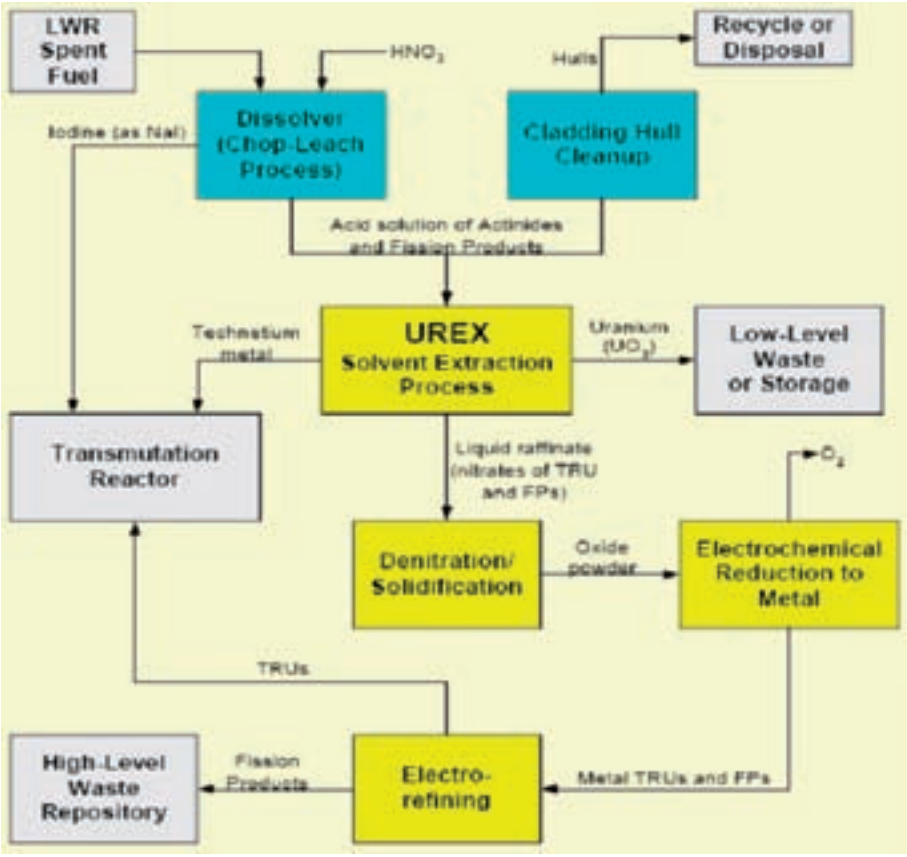

FIG. 11. A hybrid aqueous–pyrochemical process (UREX + PYRO-A) for partitioning of spent LWR oxide fuel [18].

waste. The TRUs, rare earths and residual metals (Zr, Mo, etc.) are treated in an electrorefining furnace at 500°C. The electric potential of the Cd electrode is selected to be a potential at which the rare earths are not yet deposited. The metal concentrate of the TRUs is mixed with Zr metal to produce a TRU–Zr metal alloy fuel by casting.

At the cathode of the electrolyser the TRUs and some rare earth metals are separated from the fission products. Complete separation from TRUs is theoretically impossible, because some free energies of formation (Pr, Ce, Nd and Y) overlap with those of Pu, Np, Am and Cm. This is not a fundamental drawback, since a certain amount of rare earths (between 1 and 10%) may be present in the TRU mixture, which is recycled in a fast neutron spectrum device. Recent studies have shown that a liquid Bi cathode is better suited to separate the TRUs from the bulk of the rare earths. Much R&D will, however, be necessary to transform the present IFR batch process into a countercurrent extraction system operating at high temperature.

Both examples have, in addition, proved that the variety of HLLW partitioning methods is high. However, these examples have also shown that partitioning processes could be semi-integrated into advanced aqueous extraction reprocessing flowsheets. Of course, some other systems are also under consideration, for example, fluoride salts, in which group separation MAs/REEs could give a higher separation factor [10]. However, no integral tests have been done in these systems.

2.2. Separation of long lived fission and activation products

A number of radiologically important fission and activation products play a potentially important role in the assessment of geological repositories and have been considered as a P&T option. This radiactive waste was mentioned in Section 1 [1]: the fission products ^{99}Tc, ^{129}I, ^{135}Cs, ^{79}Se, ^{93}Zr and ^{126}Sn, and the activation products ^{14}C and ^{36}Cl.

Caesium-137 and ^{90}Sr are the two main fission products that determine the radiological hazard and heat content of HLW during the initial 300 years. The radionuclide loading of the glass is determined by the concentration of these radionuclides. If they could be eliminated and conditioned in suitable matrices, the residual HLW could be disposed of much earlier in deep geological disposal facilities and the separated Cs–Sr radionuclides could be kept in engineered storage vaults.

Caesium radionuclides (^{137,135,134}Cs) can be effectively extracted from HLLW using several methods:

(a) Adsorption on inorganic exchangers;

(b) Liquid extraction with chlorinated cobalt dicarbolite (CCD);
(c) Use of calixarene crown ethers.

Pilot tests with CCD have been carried out in the USA, while representative industrial tests have taken place in the Russian Federation. A successful hot test with a calixarene crown ether extractant has been carried out recently in France by the CEA.

After separation, ^{135}Cs cannot be considered for transmutation because of the presence of other Cs radionuclides; separated ^{135}Cs will therefore be better directed towards specific conditioning into a stable crystalline matrix for disposal.

Separation of ^{90}Sr is in itself not essential within a P&T strategy, but since it generates a large amount of heat, its removal would contribute to a decrease of the heat load in the storage formation. Hot separation tests have been performed in the Russian Federation using CCD, and in India and the USA with crown ethers.

Technetium-99 is a fission product with a half-life of 213 000 years that occurs as Tc metal and TcO_2 in insoluble residues and as a soluble pertechnetate ion in HLLW solution. Its generation rate is high, with an overall specific concentration of ≈1.2 kg/t HM, depending on burnup. In order to effectively address the long term radiotoxicity problem, both soluble (80%) and insoluble (20%) fractions ought to be combined before any action is taken towards depletion by transmutation.

The extraction of soluble TcO_4^- is relatively easy. The similarity between Tc and the Pt metals in insoluble waste, as well as the nature of the separation methods, makes this partitioning operation very difficult, but separation from aqueous effluents is possible in an advanced PUREX scheme. The present recovery yield could approach 80% at best (DF = 5). A significant improvement in ^{99}Tc recovery from HLW is only possible if it is converted into a single chemical species, which is not easy to achieve. Pyrometallurgical processes are perhaps more adequate for carrying out a group separation with the Pt metals.

Iodine-129, which is included in most of the land based repository concepts for spent fuel, is the first radionuclide to emerge into the biosphere, owing to its very high mobility in aquifers. About 80% of this inventory is present as the very long lived (16 million years) isotope ^{129}I, and 20% as stable ^{127}I. During reprocessing by aqueous methods, it is removed from the dissolver solution with a yield approaching 95–98% (DF = 20–50). As its radiotoxicity is very high, and it is very soluble, it would be advisable to increase the separation yield from different waste streams to reduce the radiological impact. A target DF of ~10^3 could be proposed as a significant improvement. In order to

improve this separation yield, more complex chemical treatments are necessary. During high temperature pyrochemical processes, higher separation yields could in principle be expected. Adapted conditioning methods for separated iodine (AgI, $Pb(IO_3)_2$, Pb apatite, etc.) have been developed. The separated fraction can either be stored on a specific (zeolite) adsorbent or discharged into the sea. Since ^{129}I has a half-life of 16 million years, its worldwide dispersion in the geosphere or biosphere cannot be prevented. However, conditioning and confinement in, for example, a salt dome are the possible management options to reduce its radiological impact and final storage. This is an alternative management option that undoubtedly deserves much international attention.

Selenium-79 and ^{126}Sn are fission products with half-lives of 65 000 and 250 000 years, respectively, that occur in HLLW. Separation from liquid HLLW is not easy, owing to the very small chemical concentration in which it occurs in comparison with natural sulphur compounds. Zirconium-93 and ^{135}Cs are two long lived (1.5 and 2 million years half-life, respectively) radionuclides. Separation of these radionuclides from the other fission products for eventual transmutation is almost excluded, since they are accompanied by other radioisotopes. Their transmutation after a series of isotopic separation processes is at present considered to be an almost impossible endeavour from both the technical and economic points of view.

The activation product ^{14}C, with a half-life of 5730 years, is problematic because it can potentially enter the biosphere through its solubility in groundwater and play an important radiotoxicological role through its uptake into the biochemical life cycle. Its concentration in spent fuel is low, depending on the nitrogen contamination of the initial UO_2 fuel. Its role in long term radiotoxicity is dependent on the physicochemical conditions occurring in the deep underground aquifer or water unsaturated geosphere. The capture cross-section in a thermal neutron spectrum is negligible.

Chlorine-36, formed by transmutation of impurities of ^{35}Cl in zircaloy cladding, has a half-life of 300 000 years. This radionuclide cannot be considered in a recovery or transmutation scenario.

2.3. Conclusions about recent partitioning activities

This brief review of activity in the field of partitioning shows that the problem of partitioning of MAs and radiotoxic long lived fission products is solved as a whole. These solutions can be achieved by accessible technical means.

It is currently technically possible to treat the HLLW from a well operated reprocessing plant (e.g. La Hague) in a separate facility. It is planned

in France to put into operation such a pilot facility by 2012, and an industrial facility by 2025.

It is technically possible to carry out updating of the reprocessing technological flowsheet and to improve it, as is planned for the Mayak plant in the Russian Federation. It was originally planned to make these improvements stage by stage, including the improvements in the ecological and economic parameters of the plant that will be necessary in the next ten years.

It is technically possible to follow the route for allocation of long lived components from SNF that is assumed for the UREX process or to treat HLLW at a special separate facility.

From a general point of view, it is not necessary to connect partitioning and transmutation in time and space. Studies by the CEA have shown an opportunity for long term storage of MA concentrates as a mix with uranium oxide [19]. There are proposals in the Russian Federation to use, for this purpose, special glasses (alumino-fluoride-phosphates), which can be transferred over decades into the technological media, while MAs can be extracted from them for the manufacture of targets, or to enter them into another system for transmutation later [20].

The main conclusion about the recent status of partitioning studies is that a number of the methods developed will allow quantitative extraction not only of Pu and U from SNF but also of other actinides: Np, Am and Cm. In addition, there are no problems in extracting and concentrating certain fission products: Tc, I, Cs and Sr.

It is important that partitioning connects with other tasks: recovery of noble metals from SNF and HLLW, which are close to partitioning technically. The scientific basis for this type of recovery has been developed [21].

Unfortunately, questions related to Pa recovery in the Th cycle have not been considered here.

However, two key questions arise in connection with recent partitioning objectives and developments:

(1) Now the partitioning process becomes an additional procedure for reprocessing. Is it necessary in the future to design and construct similar systems where, except for extraction of the main SNF components, there are many problems with other SNF components? How will partitioning fit into the framework of advanced fuel cycles?
(2) The minor actinides are now considered as wastes. What is it expedient to do with MAs: destroy them or use them? In fact, they are heavy actinides that are fissile or fertile. Perhaps it is not necessary to transmute them at present. Perhaps it will be better to concentrate them and then store them as a resource.

3. THE FUTURE OF PARTITIONING

3.1. From reprocessing to partitioning and recycling

The conditions that now exist in the field of reprocessing irradiated fuel in combination with the main streams of SNF from LWR actually generate the problem of P&T. Recent approaches to partitioning are mainly directed towards reduction of the hazards associated with the radiotoxicity of HLLW after LWR SNF reprocessing, However, now mainly Pu is used from products of SNF reprocessing; other components are not used (utilization of reprocessed uranium (RepU) is now limited). This situation can be adopted at present, but for future systems with closed fuel cycles must be re-evaluated. In future, partitioning must be closely bonded with reprocessing and other processes of spent fuel treatment. Reprocessing/partitioning must be used for reasons of safety (including non-proliferation) and economy with closed fuel cycles.

The combined reprocessing/partitioning process for treatment of SNF must meet the following requirements:

(a) All extracted materials must be utilized. The main components of SNF — U, Pu and Th — must be reused in a nuclear fuel cycle (at present or in the future). This is necessary to exclude recovery of basic SNF components for stockpiles.
(b) New technology for reprocessing must be oriented only towards extraction of necessary components. Therefore, partitioning must be of SNF, not complete reprocessing with production of pure components.
(c) New technologies must be flexible and use the modular principle. In this case, changes of technological flowsheets will be easily possible for adaptation to new goals.

Therefore, the main goal for future technologies for SNF treatment is P&T instead of reprocessing.

New technological approaches will be necessary for advanced fuel cycles.

The above ideas can be illustrated by some concepts that involve complex systems avoiding traditional reprocessing approaches.

The DOVITA programme was one Russian concept related to a closed fuel cycle for fast reactors initially intended for transmutation of MAs [22]. This concept was based on a simple idea: dry technologies for MA oxide fuel reprocessing and preparation, and automated vibropacking technology for fuel pin production. Fuel recycling is carried out with the flowsheet shown in Fig. 12. The basic (driven) MOX fuel, after 15–20% burnup, goes through decladding, crushing, vacuum reprocessing and repeated vibropacking stages. After two

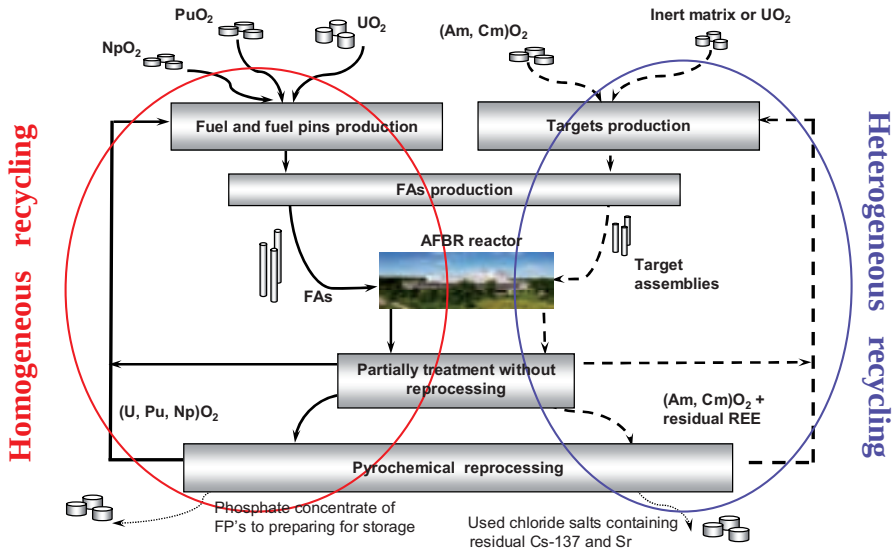

FIG. 12. The DOVITA fuel cycle [22].

cycles of irradiation, the basic fuel is reprocessed by pyroelectrochemical methods. Uranium, Pu and Np (and a portion of Am) return for irradiation. The materials of targets with Am, Cm and REE are irradiated in three or four cycles, with periodic decladding and treatment; they are then sent for pyrochemical processing. The pyrochemical process for separation of Am and Cm from REE will be carried out in a molten salt system with liquid metal electrodes; thus only the REE portion is discarded. The main part of the experimental studies have been completed over the last 13 years.

This system was, of course, initially developed particularly for fast reactor–burners, but the main part of the development could also be used for the closed fuel cycle of the BN-800 reactor, which is now under construction in the Russian Federation.

A prospective concept related to treatment of LWR SNF is under development in the Czech Republic [23]. Experimental and theoretical work in the area of the development of pyrotechnology for ADS is directed at the fields of fluoride volatility and material research into fluoride salts. This system includes a molten salt system for treatment of LWR SNF by fluorination for removal of uranium, but another component remains in the molten salt system for transmutation in molten salt reactors (MSRs) (Fig. 13).

As an example of advanced technologies, the UREX + PYRO process (Fig. 11), mentioned earlier, can also be considered to be a promising complex

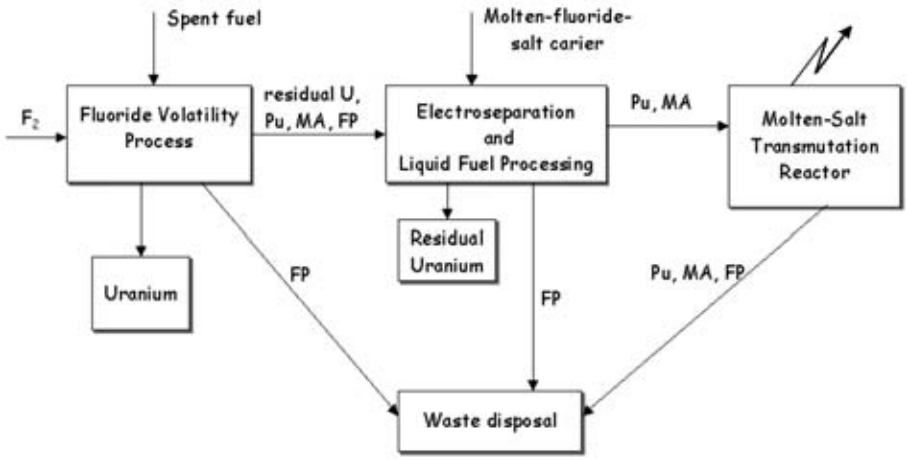

FIG. 13. Fuel cycle scheme for an MSR operated as a TRU burner in a once-through cycle, with a pyrometallurgical front end.

technology. Therefore, the path from reprocessing of SNF to partitioning might be realized in practice. More details about similar technologies have been demonstrated in the nuclear fuel cycle.

One of the first applications of dry processes, the melt refining of irradiated EBR-II fuel carried out in the 1960s, is a good example of partitioning SNF [24]. Spent nuclear fuel was divided into three components (Fig. 14): low decontaminated metallic uranium, deposits with solid FPs and volatile FPs.

3.2. Minor actinides: Waste or fuel?

The subjects related to partitioning can be considered from another point of view. What are MAs: wastes for destruction or useful materials? Neptunium, Am, Cm and Pa could also be considered as fuel for some reactor systems. This possibility would be better realized for fast reactor systems than for LWRs. If these elements are considered as wastes, additional expense must be incurred for their transmutation. If these elements are considered as fuel components, they could be recycled with the main actinides and co-extracted with them during reprocessing/partitioning processes. Special reactor systems also could be designed as systems for energy generation. For example, Np, Am and Cm could be considered as fuel components for fast reactors. Similar methods could be used for burning of parasitic uranium isotopes (^{232}U, ^{234}U and ^{236}U),

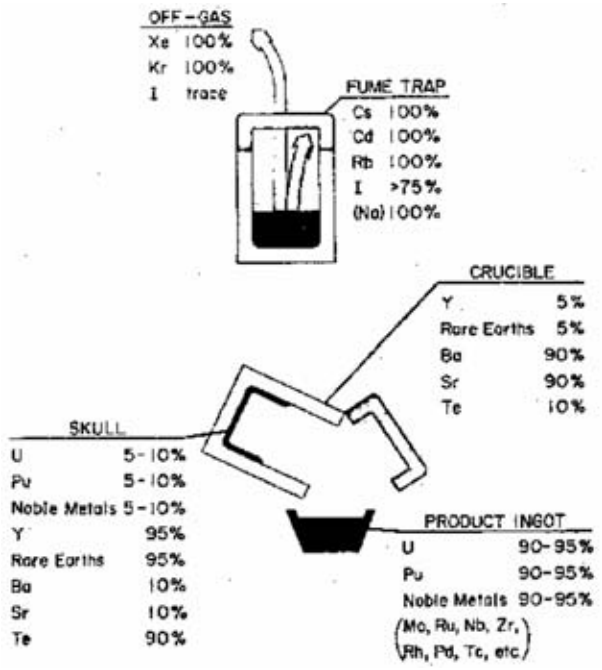

FIG. 14. Distribution of metallic fuel components after a melt refining procedure [24].

which must be produced in a concentrated form during the re-enrichment process.

There are other tasks connected with this problem: the development of reactor systems that can use MAs, in addition to Pu with a 'degraded' isotopic composition and U with a high content of parasitic isotopes as effective fuels or fertile materials. The problems associated with the development of such reactor systems could be solved with radioactive residues after re-enrichment of RepU, which contains a high content of ^{232}U.

It is necessary to develop new reactor systems that can usefully burn the above mentioned elements and isotopes, rather than just destroy them as wastes.

The criteria for partitioning/reprocessing and the other subjects for consideration in the future (beyond 2050) could be taken from INPRO methodology [25] or from other new conceptual studies.

4. CONCLUSIONS

A number of the methods developed will allow quantitative extraction not only of Pu and U from SNF but also of other actinides: Np, Am and Cm. In addition, there are no problems in extracting and concentrating certain fission products: Tc, I, Cs and Sr. Partitioning, as an additional step in standard reprocessing procedures, can provide for separation of long lived radionuclides and contribute in this way to reduction of radiotoxicity and the level of activity of the bulk volume of HLW for disposal. The relatively smaller amount of separated actinides could be stored until their transmutation or other utilization. For optimization of the fuel cycle, the development of new technologies must be shifted in future from complete reprocessing of spent fuel to partitioning of the fissile and radiotoxic materials from it. Technological optimization tasks could be directed to development of simple new technologies and development of new systems as a whole for effective utilization of all the components of SNF.

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TRANSMUTATION OF RADIONUCLIDES

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Presented by L. Koch

Abstract

The paper deals with the effects of transmuting radionuclides in spent fuel and the associated scientific and technical issues. Aspects of possible benefits of transmutation, which include toxicity reduction, an increase in effective repository capacity and ease of long term radioactive waste management, are discussed. Transmutation schemes for minor actinides (MAs) are largely classified into: (a) homogeneous recycling, (b) heterogeneous recycling, (c) recycling confined in a dedicated system. Long lived fission products (LLFPs) such as Tc-99 and I-129 would optionally be transmuted to achieve a further reduction of the long term dose level from the repository. The R&D on partitioning and transmutation have accumulated pertinent data for considering the technical feasibility of transmuting MAs and LLFPs in either an accelerator driven system or a fast reactor, but the technology for handling these highly radioactive materials in the fuel cycle is still in a nascent stage. There are also gaps in some of the fundamental data such as the nuclear data on americium and curium isotopes.

1. INTRODUCTION

For sustainable utilization of nuclear energy in any country, the waste problem has to be solved in a reasonable way. In particular, the accumulation of spent fuels on the surface has to be safely and economically controlled. In one alternative, spent fuels will be stored or disposed of in geological formations. Another alternative is to reprocess spent fuels to recover uranium and plutonium. In the latter case, plutonium will be recycled in power reactors; the reprocessed uranium, which contains about 0.9% of U-235, may also be used as feed for enrichment. Optionally, the remaining actinides (neptunium, americium and curium), which are often called minor actinides (MAs), and

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some of the long lived fission products (LLFPs), such as Tc-99 and I-129, may be transmuted into stable or short lived nuclides in a proper neutron field. This paper deals with the effects of, and the scientific and technical issues involved in, transmuting radionuclides in spent fuel.

Several aspects of the effects of transmuting radionuclides, which are partitioned from spent fuel, are discussed:

(a) Toxicity reduction — The total radiotoxicity that arises from the use of nuclear fuel can be reduced.
(b) Better use of repository capacity — A given repository can accommodate more wastes if they contain a smaller amount of heat emitting nuclides.

There are additional effects related to long term waste management:

(a) Elimination of recriticality concern — There may be a low probability of forming a recriticality condition by certain reconfigurations of wastes in a repository after a very long period of time.
(b) Enhancement of long term proliferation resistance. Permanent safeguard monitoring may be required for those waste materials that originated from safeguarded civilian facilities, unless they are converted into irrecoverable forms.

These benefits of transmutation will be discussed in this paper, along with the development status of transmutation technology.

2. POSSIBLE BENEFITS OF TRANSMUTING RADIONUCLIDES

2.1. Short term risks versus long term risks

Evaluation of any measure taken on wastes involves the very difficult task of balancing various risks, which are unevenly distributed in time and space. Here the length of the time frame to be considered exceeds everyday human understanding. It is certain only that both the short term and long term risks should be contained within the legally acceptable level if measures are taken according to well drafted regulation. The trade-offs may then be judged by the costs incurred in containing each risk within certain bounds, and by the social and economic benefits. Economic considerations tend to favour the once-through option rather than the recycling option; they also tend to favour plutonium recycling compared with the more thorough partitioning and trans-

mutation (P&T). However, there are factors that have yet to be taken into account in more or less sociological contexts.

2.2. Potential toxicity reduction

As explained below, the dose levels resulting from a repository are not directly connected to the amount of radiological toxicity of the wastes disposed therein. This is common knowledge among waste experts but not very obvious to the public. The public may not be persuaded on this one point because nothing is certain about the far future (say a hundred thousand years).

Figure 1 shows the annual limit of intake (ALI) ingestion hazard of the high level waste (HLW) from one tonne of fresh fuel. By transmuting more than 95% of MAs and the LLFPs (Tc-99 and I-129), the toxicity of HLW is reduced within ~1000 years to a level comparable to that of the natural uranium that has been used to produce the nuclear fuel[1]. If this reduction of toxicity is really achievable, it may help solve the problem of public acceptance of HLW disposal.

It has been argued by waste management experts that the total ingestion or inhalation hazard of wastes buried underground has little relevance to the health hazards of the population. Radionuclide migration from failed waste canisters in a repository has been modelled in various studies. Plutonium, americium and curium have low solubilities in groundwater, and their migration is so slow that they do not contribute to the health hazard. Neptunium contributes after 100 000 years, because this element has a relatively high solubility. However, there is no proportionality between the health hazard and the inventory of Np-237. Thus, the reduction of total actinide inventory does not significantly reduce the health hazard of the population as predicted by the migration model. Among LLFPs, I-129 and Tc-99 are considered to dominate the long term hazard, because they have high solubilities [2].

Ahn has recently proposed a view that may highlight the effect of P&T [3]. The geosphere has been traditionally treated as an important barrier to radionuclide migration. He pointed out the simple fact that the geosphere is also part of the environment. The impact of P&T can then be measured by the total radiotoxicity loading outside the repository (Table 1 and Fig. 2).

[1] The production of 3% enriched uranium with 0.15% tails concentration will require about 6 t of natural uranium; this figure increases with increasing enrichment and/or tail concentration. The current commercial figure for higher burnup fuel and 0.3% tails concentration is close to 9 t of natural uranium.

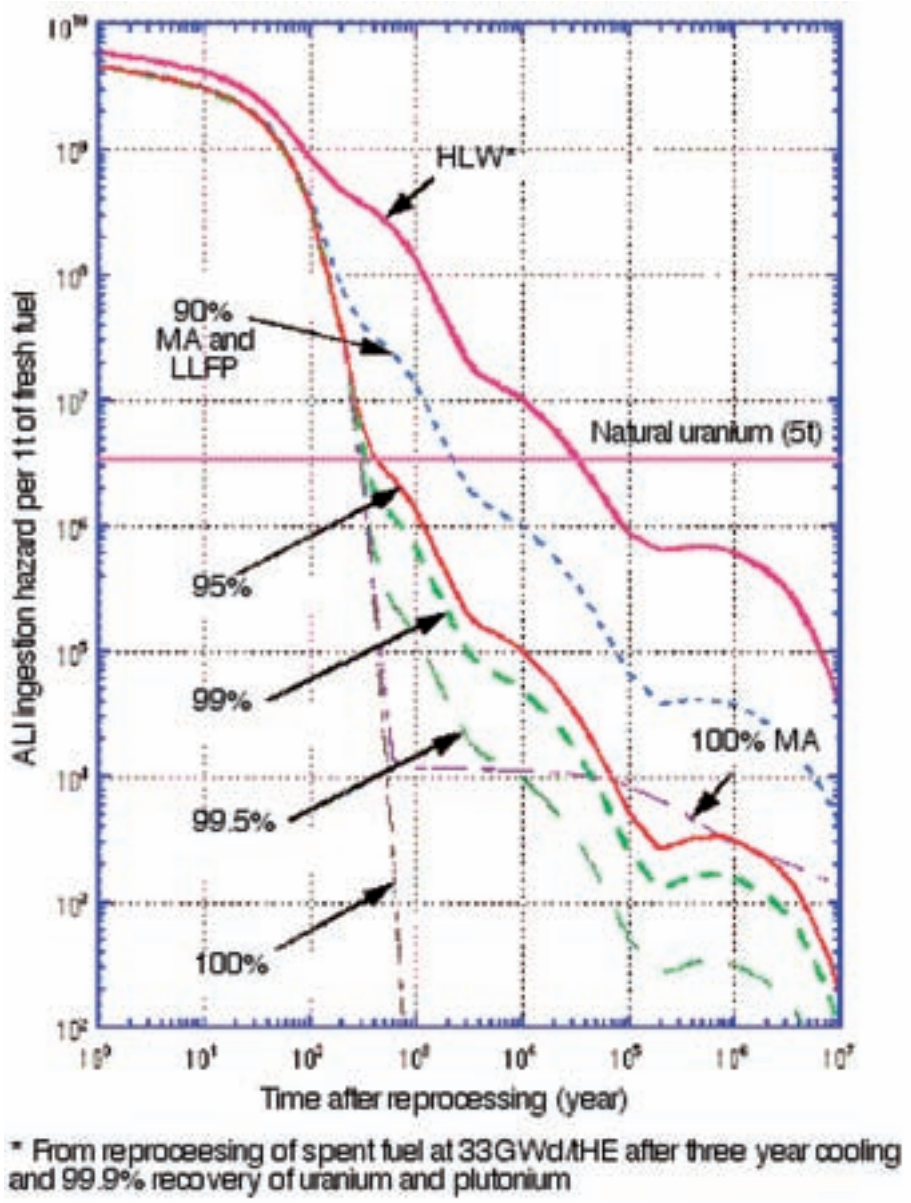

FIG. 1. Annual limit of intake ingestion hazard of HLW as a function of the transmutation fraction of MAs and LLFPs [1].

TABLE 1. ENVIRONMENTAL IMPACT MEASURES AND THE EFFECT OF TRANSMUTATION ACCORDING TO AHN [3]

Measure for repository performance	Significance of geosphere	Effect of reducing toxicity by transmutation
Exposure dose rate of a human living in a certain location relative to the repository	Natural barrier to migration of radionuclides	Low sensitivity
Environmental impact as the sum of toxicity indices existing in the far field	Part of the environment	Sensitive

2.3. Repository availability

Partitioning and transmutation would effectively remove or eliminate heat emitting nuclides from the wastes, resulting in better use of a given repository site. Figure 3 shows the result of an estimate of the effect of introducing P&T on an HLW repository area [4]. If strontium and caesium are also separated and a proper storage technology for these elements is developed, the required HLW repository area could be reduced by a factor of ten from the reference case where all liquid HLW is vitrified without P&T.

2.4. Long term proliferation consideration: Plutonium and neptunium

It is theoretically possible to recover plutonium from a spent fuel repository [5]. Spent fuels stored for more than 300 years, after which the radiation levels become tolerably low, may become an attractive source for reclaiming plutonium. The value of plutonium for nuclear weapons depends on

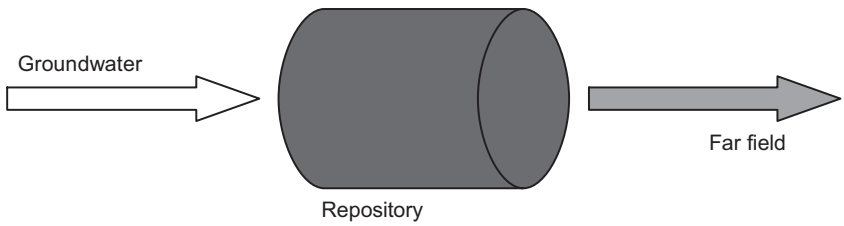

FIG. 2. Schematic view of repository and environment.

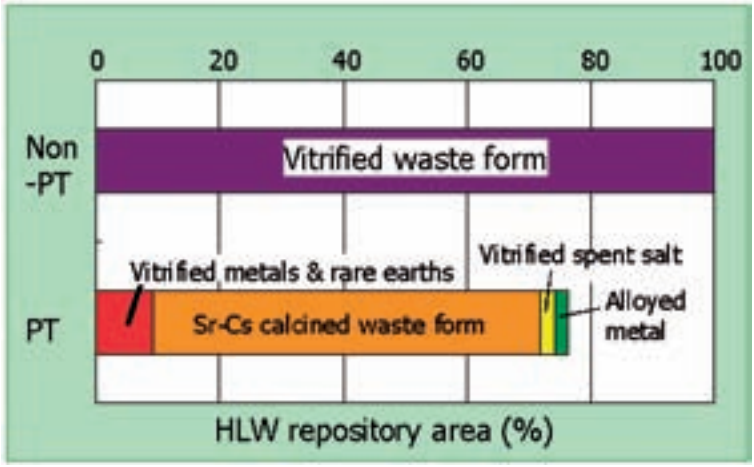

FIG. 3. Effect of P&T on an HLW repository area when liquid HLW underwent P&T by an accelerator driven system (ADS) combined with a pyroprocess.

the critical mass, the spontaneous neutron rate, the decay heat and the gamma radiation, which are functions of the isotopic composition. Among the deleterious factors for weapons applications, decay heat generation, mainly by Pu-238 ($T_{1/2}$ = 87.7 a), and gamma radiation are reduced significantly with plutonium stored for more than 300 years. In addition, the radiation from fission products in spent fuel decreases to a level that may no longer be lethal for humans without heavy shielding.

It has been argued that the HLWs from reprocessing may also pose some proliferation concerns. Neptunium in the wastes consists of only one isotope, Np-237. The bare sphere critical mass of Np-237 is 56 kg, which is comparable to that of U-235 [6]. In addition, the spontaneous neutron emission rate, which is a concern for designing an effective weapon, is low for Np-237. These features make neptunium usable for nuclear weapons. The concentration of Np-237 in the borosilicate glass for disposal is, however, a mere 0.25 wt% [7]; its recovery from the glass would be a formidable task, if not impossible.

2.5. Recriticality concern

Recriticality is an issue in spent fuel disposal during the post-closure phase. It is considered that most fissile nuclides are expected to decay to thermal fissile uranium isotopes (U-235 and U-233) before significant water

intrusion into the waste packages (WPs) occurs.[2] Uranium in WPs may be dissolved in water and concentrated in ore bodies by selective chemical processes in the underground environment. This may produce a situation similar to that of the natural nuclear reactor discovered at Oklo, Gabon. However, this may be effectively prevented by some engineered measures such as placement of depleted uranium (DU) in or around WPs: DU isotopically dilutes the fissile isotope [8].

The possibility of recriticality for the vitrified HLW repository has been examined and shown to be unlikely by Ahn [7].

3. TRANSMUTATION SCHEMES

Table 2 shows the amount of transuranium nuclides in PWR fuels [9]. With increasing burnup and initial plutonium content, the concentration of MAs increases. With regard to spent MOX, prolonged storage significantly increases the concentration of neptunium and americium due to the decay of Pu-241.

A remark is necessary about neptunium. Experience in the UP3 reprocessing plant indicated that about 2.3% of neptunium goes into recovered uranium and plutonium; the balance of 97.7% goes into high level liquid waste. About 85% of neptunium accompanies U + Pu in the first extraction cycle; the later extraction cycles remove neptunium from uranium and plutonium to place it into HLW streams [10]. Hence, the reprocessing process may be modified so as to recycle neptunium along with uranium and plutonium, if necessary.

There may be various scenarios for transmuting the long lived radionuclides, but basically there are three types regarding the treatment of plutonium and MAs.

3.1. Homogeneous recycling in power reactors

Minor actinides will be homogeneously mixed in power reactor fuel and recycled. Either thermal or fast reactors may be used. As for thermal reactors, use of light water reactors (PWRs and BWRs), high temperature gas cooled reactors (HTGRs) and molten salt reactors (MSRs) has been considered.

In the case of simultaneous recycling of plutonium and MAs in light water reactors, a type of MOX fuel enriched in U-235, which is called 'MIX', may be employed. Uranium is enriched in order to avoid significant reactor safety

[2] Careful design should eliminate the possibility of recriticality by water intrusion or reconfiguration at earlier stages.

TABLE 2. CONCENTRATION (g/tHM) OF TRANSURANIUM NUCLIDES IN PWR SPENT FUEL [9]

Nuclide	UO_2 45 GW · d/t 5 years	UO_2 60 GW · d/t 5 years	MOX 45 GW · d/t 5 years	MOX 45 GW · d/t 30 years
Np-237	6.329E+02	9.273E+02	1.708E+02	3.714E+02
Np total	6.329E+02	9.273E+02	1.708E+02	3.714E+02
Pu-238	2.677E+02	4.767E+02	1.112E+03	9.138E+02
Pu-239	6.028E+03	6.599E+03	1.797E+04	1.797E+04
Pu-240	2.678E+03	3.061E+03	1.379E+04	1.412E+04
Pu-241	1.322E+03	1.580E+03	6.492E+03	1.949E+03
Pu-242	7.707E+02	1.054E+03	4.584E+03	4.584E+03
Pu total	1.107E+04	1.277E+04	4.395E+04	3.953E+04
Am-241	4.113E+02	5.041E+02	2.455E+03	6.794E+03
Am-242m	7.940E–01	1.257E+00	1.579E+01	1.408E+01
Am-243	1.708E+02	2.746E+02	1.270E+03	1.267E+03
Am total	5.829E+02	7.800E+02	3.741E+03	8.075E+03
Cm-242	1.073E–02	1.610E–02	1.082E–01	3.407E–02
Cm-243	4.680E–01	8.515E-01	6.112E+00	3.327E+00
Cm-244	5.570E+01	1.137E+02	6.132E+02	2.355E+02
Cm-245	4.930E+00	1.255E+01	9.507E+01	9.488E+01
Cm-246	5.110E–01	1.518E+00	6.919E+00	6.894E+00
Cm total	6.162E+01	1.287E+02	7.216E+02	3.408E+02
TRU[a] total	1.234E+04	1.461E+04	4.858E+04	4.832E+04
MA total	1.277E+03	1.836E+03	4.633E+03	8.788E+03

[a] TRU: transuranium.

degradation[3], to balance the neutronic absorption of the actinides and to meet the fuel burnup requirement. Even with these techniques, the neutron emission from the spent MA-bearing fuels should become formidable due to buildup of

[3] The plutonium concentration in MOX should increase with multi-recycling in LWRs, ultimately leading to a positive coolant void reactivity. In MIX, enriched uranium is added to stabilize the plutonium content during multi-recycling.

curium and Cf-252. Therefore, use of fast reactors is considered a more reasonable option for burning MAs.

Recycling Np-237 with plutonium in light water reactors (LWRs), apart from americium and curium, is being investigated, but its chief purpose is to increase the proliferation resistance of spent MOX by introducing highly radioactive Pu-238 into plutonium by neutron capture of Np-237 [11]. Other concepts include reduced moderation BWRs, which may more readily accommodate MAs due to their harder neutron spectrum [12, 13].

3.2. Heterogeneous recycling in power reactors

Dedicated fuel elements, in which MAs are concentrated, may be inserted into the core of power reactors to achieve the required rating and burnup, which are different from those of normal fuel elements. Those dedicated fuels may be recycled in a separate materials flow. Mixed oxides with a high MA content may be used. The other approach in fuel design is to use inert matrix fuels, where MAs are dissolved or dispersed in a non-fertile 'inert' fuel matrix such as MgO. In the latter case, the inert matrix fuel would be irradiated to a very high burnup, which effectively makes further recycling of MAs unnecessary.

The advantage of heterogeneous recycling is to limit the technical difficulty and the economic penalty that are associated with handling MAs to a fraction of the fuel elements. Fabrication and reprocessing of most fuel elements are unaffected. Annexing separate fabrication and chemical treatment lines to the normal commercial lines is, however, a disadvantage.

3.3. Dedicated systems: The 'double strata' approach

This is the ultimate alternative to heterogeneous recycling: the idea is to annex the dedicated fuel cycle system (second stratum) to the commercial fuel cycle system (first stratum) [14]. Its advantage is to optimize the transmutation system independently from the power reactor fuel cycle system. Very innovative approaches such as those using accelerator driven systems (ADSs) may be adopted as illustrated in Fig. 4.

4. STATE OF THE ART TECHNOLOGIES

4.1. Properties of transuranium elements

Tables 3 and 4 show the nuclear properties and the basic chemical properties of transuranium elements, which are relevant in considering trans-

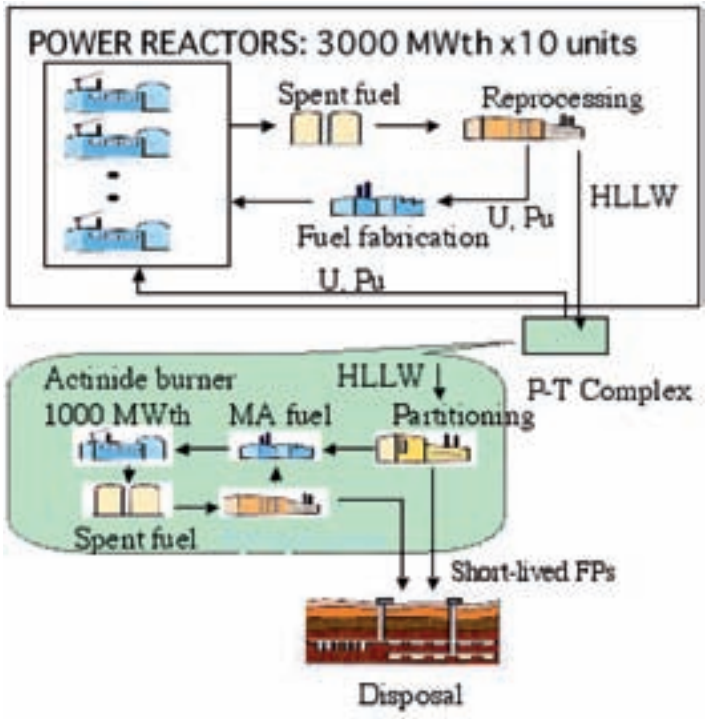

FIG. 4. Double strata fuel cycle scheme for P&T.

mutation of these nuclides. Plutonium-238 and transplutonium elements are characterized by significant neutron emission rates and decay heats.

There is a discontinuity in the physical and chemical properties on crossing from plutonium to americium. Therefore, simple extrapolation of past experience on fuels with uranium and plutonium cannot be made to those containing americium and curium.

4.2. Transmutation schemes and their impacts on fuel cycle processes

Table 5 compares the impact of recycling MAs on fuel fabrication in the various transmutation schemes [15]. Thermal loading in the fabrication facility is increased by a factor of 15 when 0.9% of MAs are introduced into the MIX fuel for PWRs and a factor of 6 when 1.21% of MAs are introduced into the MOX fuel for fast reactors. That of MA targets increases by a factor of 185 compared with MOX fuel for fast reactors. The increase in the thermal load is largely determined by Cm-244, with a decay heat of 2.8 W/g.

Forced cooling systems at all steps in the process, even including the transfer and storage of not only irradiated fuels but also fresh ones, have to be

TABLE 3. NEUTRON EMISSION AND DECAY HEAT FROM ACTINIDE NUCLIDES [17]

Nuclide	Half-life (a)	Fission cross-section[a] (b)	Total neutron emission[b] ($g^{-1} \cdot s^{-1}$)	Decay heat ($W \cdot g^{-1}$)
Pu-238	87.7	17.9	36 000	0.56
Pu-239	2.411×10^4	748	96	0.002
Pu-240	6.54×10^3	0.06	1300	0.007
Pu-241	14.4	1013	1.23	0.004
Pu-242	3.763×10^5	0.0026	2000	0.0001
Np-237	2.14×10^6	0.022	0.90	0.00002
Am-241	432.2	3.0	7000	0.11
Am-243	7.38×10^3	0.12	540	0.007
Cm-242	0.446	5.1	2.9×10^7	120
Cm-243	28.5	618	1.3×10^5	1.7
Cm-244	18.11	1.0	1.2×10^7	2.8
Cf-252	2.646	33	2.35×10^{12}	39

[a] For thermal neutrons with 2200 m/s.
[b] Assuming oxide.

TABLE 4. PROPERTIES OF ACTINIDES [18]

Element	M_p (K)	ΔH_f^o (kJ/mol)	I_1 (eV)	I_2 (eV)	I_3 (eV)	I_4 (eV)	r_m (nm)	$r_i(3+)$ (nm)	λ (W/(m·K))
U	1408	536	6.194	11.9	19.7	32.6	1.542	1.025	25
Np	913	465	6.266	11.7	20.7	33.6	1.503	1.01	6–8
Pu	913	342	6.062	11.7	21.6	34.6	1.523	1.00	5–8
Am	1449	284	5.993	12.0	22.1	36.2	1.730	0.98	–
Cm	1618	387	6.021	12.4	21.0	36.8	1.743	0.97	–

Note: M_p, melting point; ΔH_f^o, enthalpy of sublimation; I_n, ionization energy; r_m, metallic radius (CN = 12); $r_i(3+)$, trivalent ionic radius (CN = 6); λ, thermal conductivity (metal, room temperature).

TABLE 5. RADIOLOGICAL IMPACT OF MINOR ACTINIDE RECYCLING IN FABRICATION PLANTS [15]

	(12% Pu)-MOX for PWRs	(19.9% Pu)-MOX for fast reactors	(2.1% Pu)-MIX for PWRs	(20.2% Pu + 1.21% MA)-MOX for fast reactors	(2.8% Pu + 0.9% MA)-MIX for PWRs	(Am + Cm)-targets in a moderated fast reactor core
Activity (Bq/t_{HMi})	5.70×10^{16}	2.9×10^{16}	1.01×10^{16}	3.57×10^{16}	2.13×10^{16}	2.9×10^{17}
Power (W/t_{HMi})						
α	3.07×10^{3}	1.46×10^{3}	6.99×10^{2}	8.4×10^{3}	9.21×10^{3}	2.7×10^{5}
β	51.3	25.9	9.23	46.5	26.2	944.1
γ	1.25	0.8	0.26	9.3	4.3	552.4
Neutron source ($n/(s \cdot t_{HMi})$)	1.36×10^{8}	1.3×10^{8}	3.2×10^{7}	1.4×10^{11}	1.1×10^{12}	7.2×10^{11}

installed, but there is little industrial experience of such systems. One notable example of installation of a cooling system is the interbuilding fuel transfer cask in the demonstration of the fast reactor fuel cycle based on pyrometallurgical reprocessing of metal fuel in EBR-II [16]. The high decay heat would also cause problems in quality inspection and control such as dimensional measurements.

The effect of MA recycling on the neutron source strength is more pronounced for PWR fuel than for FR fuel, because the inventory of higher actinides, curium and californium, increases in a thermal flux. Remote fabrication and inspection in hot cells is therefore mandatory.

There are several alternatives in the combination of fuel types and reprocessing methods for transmutation of transuranium elements. Table 6 provides a brief summary obtained from the activity of the Working Party on Scientific Issues in Partitioning and Transmutation at the OECD Nuclear Energy Agency [19]. As for the reprocessing of fuels for heterogeneous recycling systems as well as for dedicated systems, pyrochemical methods (the pyroprocess) may be preferred in view of the high decay heat to be dealt with. Figures 5 and 6 illustrate the pyroprocess concept for metal fuels [20]. Figure 7 shows the concept of a nitride fuel cycle, whose molten salt reprocessing is based on the same principle as that shown in Fig. 6 [21]. However, the pyroprocesses are still under development and their technological maturity cannot be compared with that of the well established aqueous process, PUREX.

TABLE 6. RANKING OF REACTORS WITH RESPECT TO Tc-99 TRANSMUTATION CAPABILITY [23]

Reactor type	Configuration	Tc-99 inventory (kg)	Tc-99 transmutation rate		T^{transm} [a] (a)
			(kg/a)	(kg/(MW(e) · a))	
Fast reactor	Moderated S/A in inner core	2741	122	0.11	15
Fast reactor	Non-moderated S/A in inner core	2662	101	0.09	18
LWR	Pin in guide tube, UO_2 fuel	3633	64	0.07	39
LWR	Pin in guide tube, MOX fuel	1907	17	0.02	77

[a] Transmutation half-life $T^{transm} = \ln 2/\sigma\varphi$.

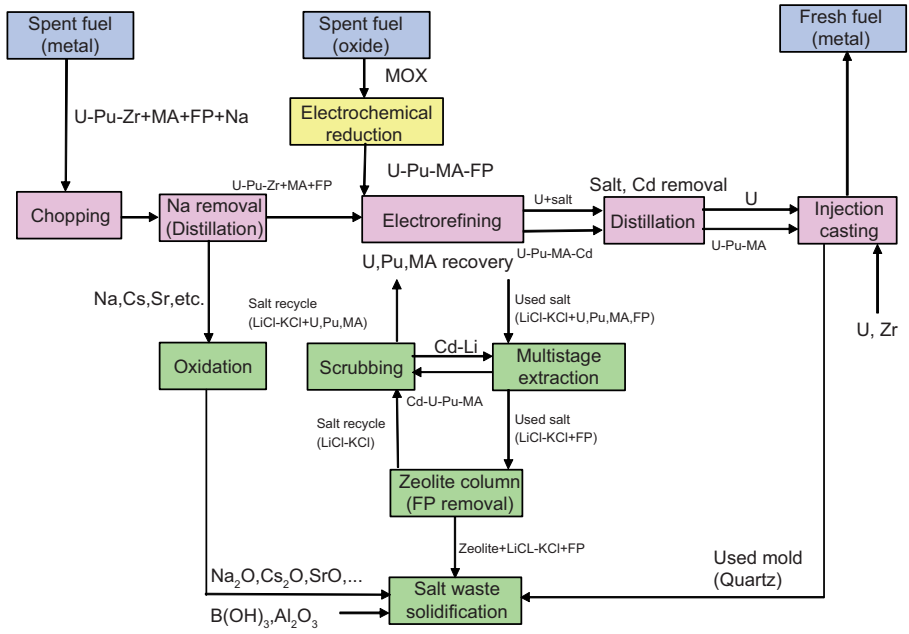

FIG. 5. Fuel cycle scheme for metal fuel fast breeder reactors [20].

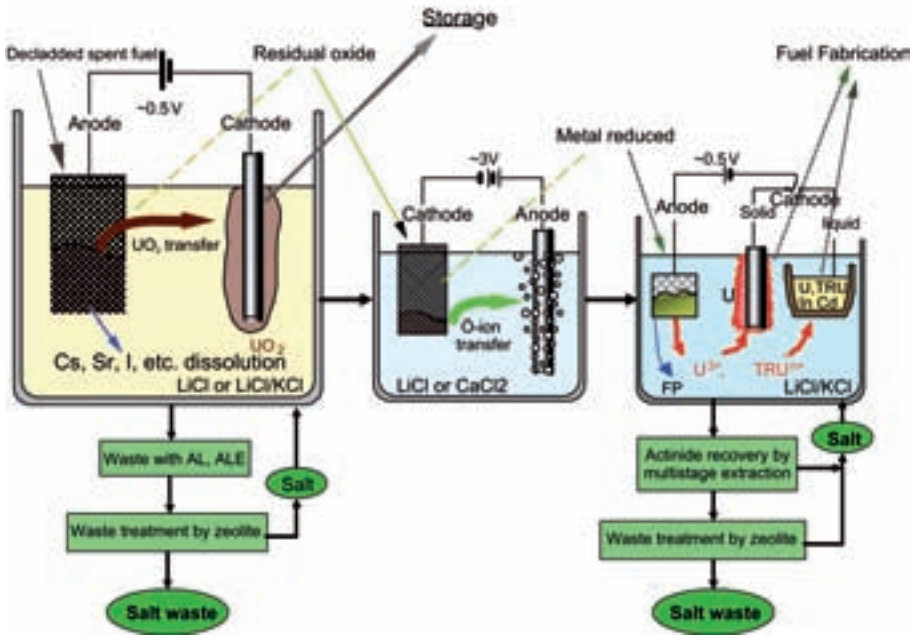

FIG. 6. Electrorefining process for the fuel cycle scheme shown in Fig. 7 [20].

4.3. Fission product transmutation

Transmutation of LLFPs is of little value from the viewpoint of radiotoxicity reduction. However, a selected number of LLFPs actually dominate the dose levels resulting from the repository. Among LLFPs, Tc-99 ($T_{1/2} = 2.1 \times 10^5$ a), I-129 ($T_{1/2} = 1.6 \times 10^7$ a), Zr-93 ($T_{1/2} = 1.5 \times 10^6$ a) and Cs-135 ($T_{1/2} = 2.3 \times 10^6$ a) are important nuclides that may have a potential impact on the long term dose levels around a repository.

Transmutation of Tc-99 and I-129 in current power reactors is not promising. Their irradiation half-lives are too long, and the inventories of fission products in reactors have to be huge to effectively realize a large transmutation rate. Either fast reactors or ADSs have to be utilized for transmutation of these nuclides.

The other significant LLFPs in long term dose evaluation of a repository are Se-79, Zr-93, Sn-126 and Cs-135, but to a lesser extent than Tc-99 and I-129. For instance, groundwater transport of Cs-135 would be retarded by a factor of 360 due to sorption on rocks [22]. The mass fractions of these isotopes in elemental fission yields are small: 20% for Zr-93 and 10% for Cs-135. The other

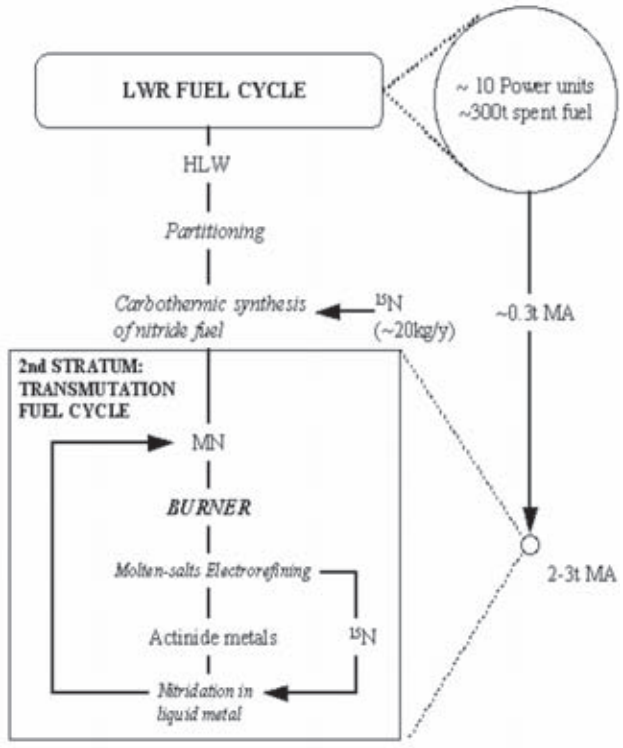

FIG. 7. Nitride fuel cycle for MA burning [18].

isotopes interfere in the transmutation of Zr-93 and Cs-135: isotope separation has to be performed. Until an economic means of isotope separation of these nuclides is established, transmutation of these nuclides does not seem feasible.

4.3.1. Transmutation of Tc-99

Transmutation of Tc-99 yields inactive Ru-100 with a reasonably large cross-section. Transmutation of Tc-99 in fast reactors may be achieved in either a moderated configuration or a non-moderated configuration. For moderation, the fuel subassembly (S/A) should incorporate metal hydrides, but the feasibility of using hydrides on an industrial scale has yet to be carefully assessed.

Table 6 shows the calculated transmutation rates and irradiation half-lives of Tc-99 in thermal and fast reactors.

Irradiation tests of Tc-99 metal rods have been made in the High Flux Reactor (HFR) at Petten [24]. Excellent in-pile behaviour has been demonstrated:

there were no observable changes in the microstructure from that of the unirradiated material.

4.3.2. Transmutation of I-129

Iodine is separated from dissolver off-gases in reprocessing plants. Transmutation of iodine to stable gaseous xenon has been studied in HFR [25]. For iodine, it is necessary to define the proper chemical and structural forms to be used as a target for transmutation. So far, pellets of MgI_2, CaI_2, CuI and NaI have been tested. Chemical interactions between the steel end-plugs and the CuI metal have been found. Since I-129 is transmuted to xenon, which would pressurize the pin containing I-129, the physical design of the iodine bearing target has yet to be investigated in detail.

Recently, a very innovative approach to transmutation of I-129 has been studied in several places. Ledingham et al. [26] have studied the use of a petawatt laser to achieve photo-transmutation of I-129. Its half-life of 15.7 million years may be shortened to 25 min by transmutation to I-128. The proper geometry, materials and energy balance have to be clarified to discuss the feasibility of these innovative approaches.

Because of the long transmutation half-life, recycling of the I-129 and Tc-99 targets seems inevitable. This means that all aspects of fabrication, irradiation and recycling of LLFP targets have to be considered. The processes of recycling and refabrication have yet to be studied.

4.4. Other technical issues

4.4.1. Fast reactors and accelerator driven systems

International efforts are now being directed again towards R&D of fast reactors within the framework of Generation IV reactors. Sodium cooled fast reactors are the most technically mature among the fast reactor concepts. Interest in gas cooled fast reactors (GFRs) is increasing, but the R&D effort in the past was very limited, while industrial experience in HTGRs may be fully utilized in developing GFRs, except for the design of the reactor core. All of these concepts have to be further improved from the point of view of construction and operational costs.

There is now intensive R&D being carried out on ADSs. Their advantages and disadvantages have been compared with those of fast reactors in a study performed by the OECD Nuclear Energy Agency [27]. Coupling of a high-energy high-current proton-beam accelerator with a subcritical fast reactor core is in its infancy. Stable operation of a high current accelerator has

yet to be demonstrated. Effects of beam interruption on the integrity of the reactor core are now being studied in international programmes.

The delayed neutron fraction, which is a key to safe operation of nuclear reactors, is significantly small for MA fuels free of fertile isotopes. This characteristic is a driver to pursuing the ADS option for realizing large transmutation rates. However, new types of reactivity and source transients have to be considered in view of this characteristic. A dedicated core of an ADS, which does not contain fertile materials, is characterized by insufficient prompt negative feedback. For instance, severe accident consequences are possible when the subcriticality level is eliminated by a phenomenon such as fuel compaction [28].

The performance and technical issues of burning MAs by homogeneous recycling in fast reactors and and the double strata approach with ADSs are summarized in Table 7.

4.4.2. Nuclear data

With any transmutation technology, the nuclear database on transuranium nuclides needs to be improved. Recent studies in JAERI showed significant discrepancies in the predicted reactivity swings with burnup of the ADS containing MA fuels, depending on the available nuclear databases employed in the calculation. The analyses of the MA samples irradiated at the Dounrey PFR have shown a significant difference between the experimental and predicted nuclear reactions (Fig. 8) [29]. Even the predictive accuracy of concentration of americium and curium in LWRs has to be improved (Fig. 9) [30].

4.4.3. Materials

Not only fuels and targets but also materials have to be developed for transmutation. With fast neutron transmutation systems, core structural materials are subjected to high radiation doses, up to around 200 dpa [31]. Transmutation systems using fast reactors may also operate at high temperatures. Fortunately, there is a substantial database on fast reactor materials. However, when the coolant for a fast neutron system is switched from sodium to either helium or lead–bismuth, the chemistry of the coolant under irradiation has to be studied further. Corrosion caused by lead and that in helium with low oxygen potentials have been experienced in the past in lead–bismuth cooled reactors in the Russian Federation and in HTGR reactors in several countries. However, the same problems in environments with more severe irradiation require further attention.

TABLE 7. COMPARISON OF ADSs AND FAST REACTORS FOR MA BURNING [33]

	Double strata with ADSs	Homogeneous recycling with FRs[a]
Advantages	Minor actinides are confined in the second stratum of the fuel cycle, which is much smaller than the first stratum of the fuel cycle of a power reactor. The technology choices for both strata are independent of each other.	Minor actinides spread through every stage of the FR power reactor fuel cycle. Power production, plutonium breeding and MA transmutation can be achieved by a single technology set.
Support factors	A single ADSs unit of 1 GW · t burns 300 kg/a of MAs, corresponding to the amount in the spent UO_2 fuels from 10 to 12 units of 3 GW(e) LWRs.	Fast reactor driver fuel would not accommodate MAs above 5%, owing to safety considerations. Therefore, FRs can burn only 50–60 kg/(GW·t/a) of MAs.
Safety	Accelerator driven systems can be shut down by turning off the proton beam. A core of ADSs without fertile materials is characterized by insufficient prompt negative feedback. Severe accident consequences are possible when the subcriticality level is eliminated by fuel compaction due to, for example, loss of cooling.	By adding 5% of MAs to FR driver fuel, the void coefficient increases by 10–20%. The Doppler coefficient decreases by 20–30%.
Other issues	The dedicated fuel cycle facility should be much smaller in scale compared with that for the power reactor fuel cycle, but the technology has to be developed to cope with fuels having high MA contents. Accelerator technology has to be advanced in reliability and efficiency. Extensive development effort is required for the window that couples the accelerator and the spallation target system placed in the subcritical core.	Massive deployment of FRs depends on economic considerations such as the availability of uranium and the cost of plutonium recovery. Minor actinide recycling may carry a significant penalty for the economy of the FR fuel cycle, if it is done in a homogeneous recycling mode.

[a] FRs: fast reactors.

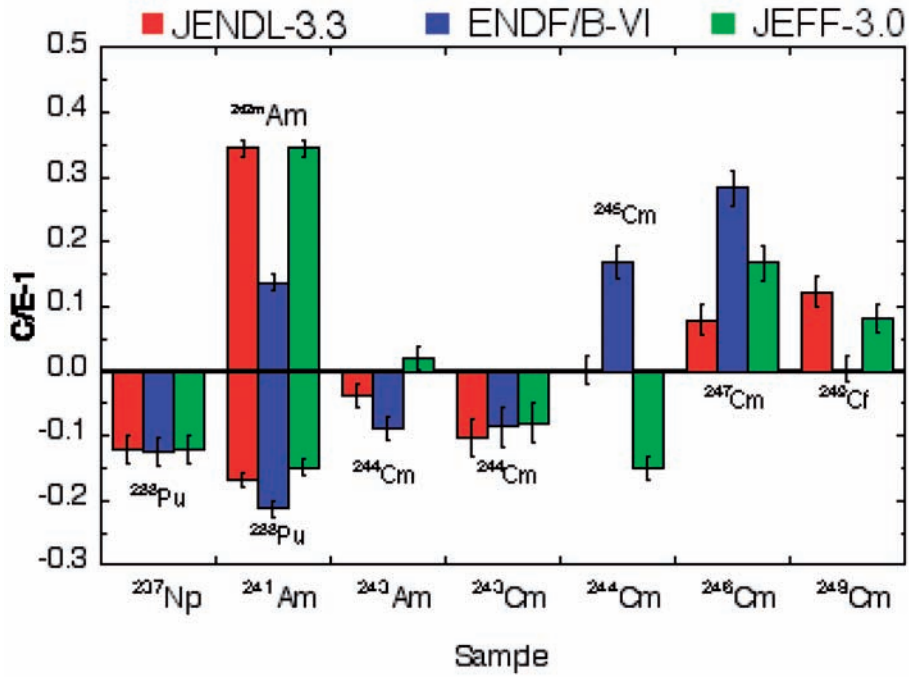

FIG. 8. Comparison of calculated (C) to experimental (E) amount in transuranium isotope samples irradiated in Dounray PFR. Significantly large errors associated with nuclear reactions of Am-241 are noted. Differences among the nuclear data are also noted, particularly for curium [29].

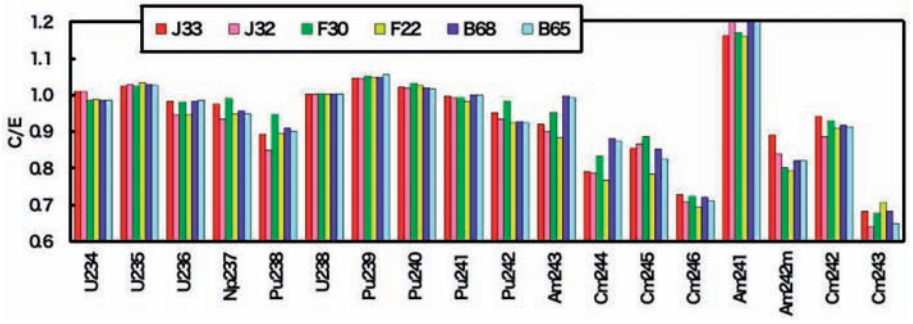

FIG. 9. Comparison of calculated (C) to experimental (E) amount of actinides in PWR fuel with 4.1 wt% U. The average was taken over ten samples with burnups ranging from 14.3 to 47.25 GW·d/t. A significant overestimate of Am-241 content and underestimates of curium isotopes are noted [30].

For accelerator driven systems, the high production of transmutation gases in structural metals is expected to cause serious problems, specifically void swelling and loss of low temperature toughness [31]. The helium production rate in a fast reactor is of the order of 0.1 appm He/dpa. Neutrons produced by the D–T fusion reaction will produce 10–15 appm He/dpa for iron based alloys. Gas production in the blanket region of ADSs will be intermediate between these two systems.

Allen et al. [31] have summarized the current status of the core materials for ADSs systems. In the temperature range 400–650°C, the ferritic steel HT9 has adequate swelling resistance, toughness, strength and ductility. In view of the increase of ductile–brittle transition temperature (DBTT) due to higher helium concentration, a minimum irradiation and minimum handling temperature for irradiated HT9 components should be set. Because of the large amounts of hydrogen and helium generated in ADSs, the synergistic effects of helium and hydrogen on mechanical performance and swelling should be studied.

4.4.4. Economics

In a European Union strategy study [32], the additional costs of recycling of 95% of americium and neptunium in power reactors increases the global cost of the overall fuel cycle by 10–50%. This increase is mainly caused by the

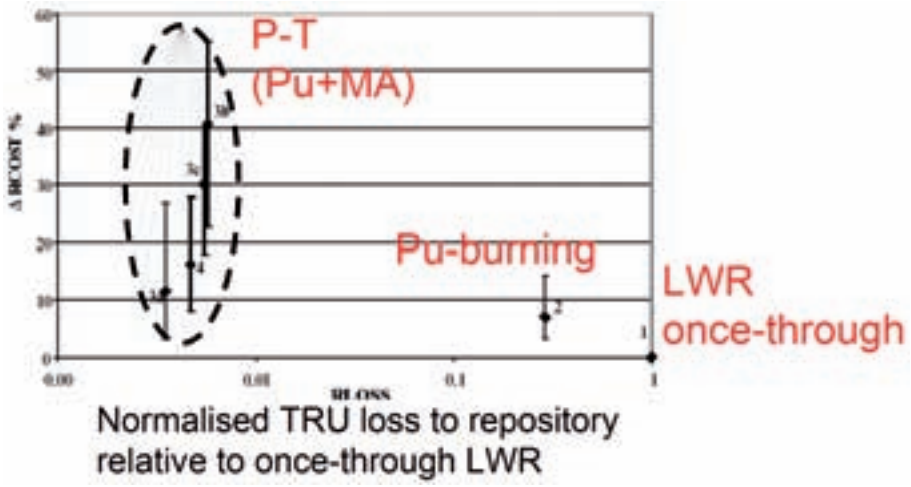

FIG. 10. Relative cost (%) as a function of relative transuranium loss to repository [27].

increase in reprocessing and MA fuel fabrication costs. A similar estimate, although quite approximate, by JAERI gave the net increment of electricity cost to be about 5% compared with plutonium recycling to LWRs. Also see Fig. 10.

These cost estimates are only bold extrapolations from the current technology at best. The economics of the fuel cycle technology remains the key issue in the deployment of any type of P&T system.

5. SUMMARY

Table 8 provides a summary of the status of developments in transmutation fuel and the major research issues in this field.

TABLE 8. SUMMARY OF THE STATUS OF TRANSMUTATION FUEL DEVELOPMENTS AND THE ASSOCIATED MAJOR RESEARCH ISSUES

MA recycling scheme	Homogeneous				Heterogeneous		Dedicated (double strata)
Reactor	PPWR	FR					ADS
Fuel type:	MA(<1%)-MIX	MA(1–2%)-MOX	(U,Pu,MA)N	Metal fuel: U-Pu-MA-RE-(<30wt%)Zr, Pu-MA-(40–60wt%)Zr	(Am, Cm) Oxide in MgO, (Am, Cm, Zr, Y) O_{2-X}[a]	(Am, Cm) Oxide in MgO	(Am, Cm)N in ZrN[b]
Fuel characteristics:	- With MA less than 1%, few effects are expected on the fuel behaviour. - The neutron emission by heavier actinides has serious impacts on the design of the fabrication plant and fuel transport.	Helium release into the plenum will be significantly increased from the ordinary MOX.	- Actinide nitrides are mutually soluble, retaining good thermal properties of (U,Pu)N for variety of isotope mix. - Sintering temperature has to be reduced to suppress americium vaporization loss. - Little database for the high-burnup behaviour even for (U,Pu)N. - Behaviour beyond cladding breach at high temperatures has to be studied.	- Casting process has to be modified to suppress americium vaporization loss. - Substantial high-burnup database exists for U-Pu-Zr. Design allows high fuel swelling and large gas release into plenum. - Redistribution of Am and its effects during normal and off-normal conditions have to be studied.	- (Am,Cm)Oxide microspheres (ϕ~100 μm) are dispersed in MgO matrix. MgO appears stable against fission-recoil damage. - Am and Cm are dissolved in yttrium-stabilized zirconia. - 90% of Am and Cm transmuted in once-through loading. - Hydride moderator has to be developed to reduce the fast-neutron damage on the cladding.	To be recycled, since the burnup level is limited by irradiation damage of the cladding.	Dissolving in ZrN stabilizes actinide nitrides. - Good thermal properties. - Little database on the irradiation performance. - Behaviour beyond cladding breach at high temperatures has to be studied.

TABLE 8. SUMMARY OF THE STATUS OF TRANSMUTATION FUEL DEVELOPMENTS AND THE ASSOCIATED MAJOR RESEARCH ISSUES (cont.)

MA recycling scheme		Homogeneous		Heterogeneous	Dedicated (double strata)
Reprocessing:	Aqueous	Dissolves in nitric acid (Exception: Metal fuel); Compatible with PUREX-based aqueous reprocessing with addition of appropriate partitioning process for recovering MA. Technical measures would be required to handle the increased decay heat.			
	Pyrolytic	- Either the RIAR process or the direct reduction in molten salts may be applicable, but the behaviour of americium and curium has yet to be clarified for efficient recovery.	- Molten salt electrorefining with dual cathodes may be applicable. - Sufficient throughput has to be demonstrated. - Effective separation of TRU from RE has yet to be studied.	(Application is not considered.)	- Oxidation potential of ZrN is a little too high, which may cause material problems. - Effective separation of TRU from RE has yet to be studied.

[a] Power to melt with ZrO_2 matrix would be too low for the ADS application.
[b] (Am,Cm)N in Mo-92 may be an alternative choice. Molybdenum is compatible with aqueous reprocessing.

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POTENTIAL CONTRIBUTION OF FAST REACTOR CYCLE TECHNOLOGIES TO TRU ELEMENT MANAGEMENT IN JAPAN

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Abstract

The paper describes a promising fast reactor (FR) cycle concept and the scenario for its introduction in Japan from the viewpoint of the management of transuranic (TRU) elements based on a feasibility study (FS) of commercialized FR cycle systems. An FS was started in July 1999 by a Japanese joint project team of the Japan Nuclear Cycle Development Institute and the Japan Atomic Power Company (as the representative of the nine electric utilities). From the current status of the phase II study (in Japanese fiscal years 2001–2005) of the FS, the combination of a sodium cooled FR with a mixed oxide fuel core, the advanced aqueous reprocessing process and the simplified pelletizing fuel fabrication process would be the most promising concept, because it exceeded other concepts in technical advances and compatibility with the development targets. In addition, a long term mass flow analysis of FR cycle deployment was performed to evaluate the potential contribution of FR cycle technologies to a possible scenario for TRU element management. The study of this scenario indicated that if FR cycle plants are commercially introduced around 2050, plutonium and all the minor actinides could be properly recycled in a closed TRU element cycle, and sustainable use of nuclear energy would be possible without a dependence on uranium resources from outside Japan. The FR cycle has potential benefits for a long term sustainable energy supply, prevention of global warming and a reduction of the long term waste toxicity source term sent to a geological repository. Therefore, the establishment of the FR cycle is expected as the ultimate goal of recycle technologies. To realize these benefits, Japan has developed FR cycle technologies, in which TRU elements are recycled, as the future energy system for replacing retired light water reactors.

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1. INTRODUCTION

Because Japan is poor in energy resources, most of these (96%) are imported from overseas. This is the highest figure among industrialized countries. To improve this situation, Japan has carried out development of nuclear power for the last fifty years based on the principle of peaceful use, and now 53 nuclear power plants are in commercial operation with a total installed capacity of 47 GW(e). Nuclear power is an extremely stable energy supply and generates 16% of primary energy resources in Japan. The dependence rate on energy resource imports is improved to 80% if nuclear power is added as a domestic energy resource. Nuclear power has become established as one of the major sources of energy production, supplying one-third of the electricity output. The use of nuclear power also reduces greenhouse gas emissions, and it is expected to provide a solution to the problem of global warming.

In addition, Japan has promoted development of the nuclear fuel cycle to enhance the efficient use of uranium resources and to reduce high level radioactive wastes (HLWs) as a national policy. Progress has been achieved in some fields, including uranium enrichment and nuclear waste management. A 1050 t separative work unit enrichment plant and a low level radioactive waste disposal facility are in operation. The Rokkasho reprocessing plant, with an annual throughput of 800 tHM, has started uranium testing, and its commercial operation is scheduled to begin in 2007. The construction of a mixed oxide (MOX) fuel manufacturing plant is also in progress at the Rokkasho site. Plutonium extracted from the reprocessing of spent fuel will be recycled into light water reactors (LWRs) in the form of MOX fuel. The legal framework for the disposal of HLWs was promulgated in 2000. Potential sites are now being surveyed in accordance with the law, and construction and operation of facilities are planned to commence by the late 2030s [1].

A nuclear fuel cycle using FRs can significantly increase the efficiency of uranium utilization and may reduce the long term radioactivity of HLWs, because it incinerates all long lived transuranium (TRU) elements in a closed cycle system. Improvement of the energy self-sufficiency ratio and diversification of the energy resources are significant issues, and thus steady effort in the development of fast reactor (FR) cycle technologies is necessary for the energy security of the next generation.

This paper describes a promising FR cycle concept and the scenario for its introduction in Japan from the viewpoint of management of transuranic (TRU) elements. From the preliminary evaluation of the feasibility study (FS), the most promising combination of FR cycle technologies is recommended as the main FR cycle concept for commercialization. With regard to the scenario for its introduction, the effects of FR cycle focusing on fuel utilization and waste

management are discussed on the basis of a mass flow calculation of uranium, plutonium and minor actinides (MAs).

2. FEASIBILITY STUDY OF COMMERCIALIZED FAST REACTOR CYCLE SYSTEMS

2.1. Development of fast reactor cycle systems

The Basic Law on Energy Policy was enacted in June 2002 to systematically promote long term comprehensive policies concerning Japan's energy supply and demand. As stipulated by the Basic Law, a Basic Energy Plan was prepared and submitted to the cabinet council in October 2003. The Basic Energy Plan calls for advancing the development of nuclear power generation and the related fuel cycle under the prerequisites of securing safety and non-proliferation.

The Atomic Energy Long Term Plan is revised about every five years, in recognition of the changes surrounding nuclear energy. In June 2004, the Atomic Energy Commission of Japan (AEC) decided to create a New Basic Plan and set up a nuclear energy programme planning committee to discuss various issues about the nuclear fuel cycle.

In the investigation of the importance of nuclear power and the related nuclear fuel cycle, the programme planning committee of the AEC discussed the reasons why a nuclear fuel cycle must be introduced. Plutonium generated in nuclear reactors, together with the remaining uranium in spent fuels, would allow effective use of resources. The usage efficiency of uranium in a nuclear fuel cycle with FRs could be improved dramatically by over 100 times compared with that in a direct disposal system in which spent fuel is disposed of as waste, or by an LWR cycle system. When viewed from the standpoint of making the most efficient use of energy resources, an FR cycle may therefore be considered as one of the candidates.

The FR cycle could contribute to the long term energy supply, to the countermeasures against global warming and to a reduction of potential toxicity of HLWs. Therefore, the Nuclear Energy Basic Plan calls for promoting the development of the FR cycle system towards its commercialization. In Japanese fiscal year (JFY) 2006, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) is scheduled to review the results of phase II of the FS. On the basis of the checking and review carried out in the FS, MEXT will make a recommendation of its main choice and the R&D policy for FR cycle system development.

2.2. Outline of the feasibility study

Japan has promoted the development of FR cycle technologies focusing on sodium cooled reactors, and accumulated engineering data through the design, construction and operating experience with JOYO and MONJU. To investigate a wide range of technical options for FRs and their related fuel cycles, a FS was started in July 1999 by a joint project team of the Japan Nuclear Cycle Development Institute (JNC) and the Japan Atomic Power Company (JAPC; as the representative of the nine Japanese electricity utilities) in cooperation with the Central Research Institute of the Electric Power Industry (CRIEPI) and the Japan Atomic Energy Research Institute (JAERI) [2–4].

The purpose of the FS is the adoption of prominent FR cycles, as indicated in Fig. 1, that will respond to the various needs of society in the future. The key specifications of the FR cycle system are high average burnup (> 150 GW · d/t), low decontamination reprocessing processes (DF < 100) and MA-containing fuel (<5 mass%). These issues contribute to basic objectives such as economic competitiveness, reduction of environmental burden and enhancement of proliferation resistance. Challenging technology goals for the FR cycle concepts were determined in five development targets as shown in Table 1, i.e. safety, economic competitiveness, reduction of environmental

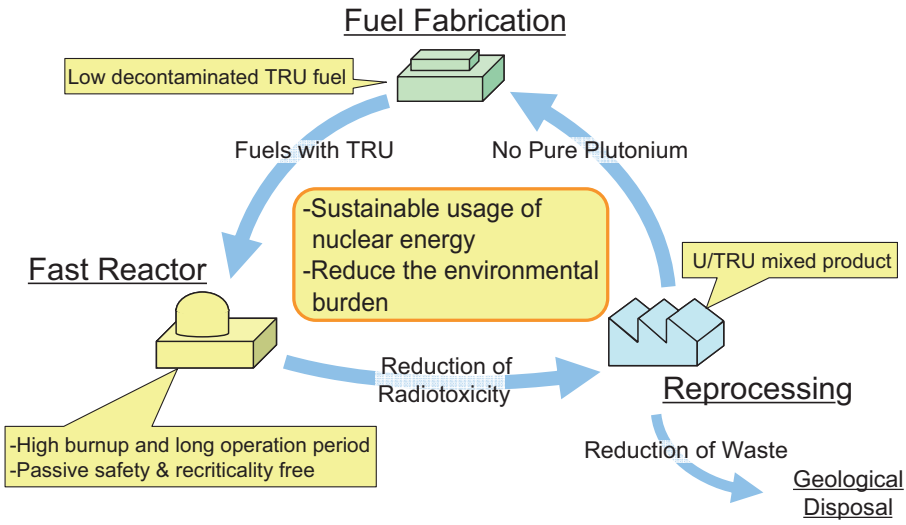

FIG. 1. Concept of the FR cycle system.

TABLE 1. DEVELOPMENT TARGETS FOR THE FAST REACTOR CYCLE CONCEPTS IN THE FEASIBILITY STUDY

Safety
Risks caused by introduction of the FR cycle should be small compared with the risks that already exist in society.
Economic Competitiveness
Achieve power generation cost comparable to that of future LWR cycle systems and other energy resources.
Ensure cost competitiveness in the global market.
Reduction of Environmental Burden
Reduce the amount of radioactive waste generated in the course of plant operation and maintenance as well as decommissioning.
Reduce the radiotoxicity of radioactive waste by means of burning or transmuting long lived radioactive nuclides.
Efficient Utilization of Nuclear Fuel Resources
Produce sustainable nuclear fuel.
Respond to diverse needs for future energy resources.
Enhancement of Nuclear Non-Proliferation
Reduce burden of nuclear PP and safeguards (no pure plutonium in any FR cycle process and increase radioactivity of fuel materials).
Effectively operate non-proliferation system (remote process and monitoring system).

burden, efficient utilization of nuclear fuel resources and enhancement of nuclear non-proliferation. In this section, the current status of the FS is briefly reviewed to indicate the prospects for FR cycle commercialization.

The time schedule of the FS is illustrated in Fig. 2. Phase I of the FS was carried out in the period from JFY 1999 to 2000. As a result of phase I, representative FR cycle concepts were screened out from FR systems using various coolants, i.e. sodium, lead bismuth eutectic (LBE), helium gas and water, with oxide, nitride and metal fuels. With regard to fuel cycle systems, fuel reprocessing methods (such as advanced aqueous methods, oxide electrolysis and metal electrorefining) and fuel fabrication methods (such as simplified pelletizing, sphere packing, vibropacking and injection casting) were selected.

In phase II (JFY 2001–2005), the design study of FR cycle concepts, the development of significant technologies necessary for feasibility evaluation and confirmation of key technical issues is being carried out to clarify the promising candidate concepts for commercialization. An interim report was published at the end of JFY 2003 to indicate the perspective of the characteristics and performance of the candidates. The results of phase II will be available by

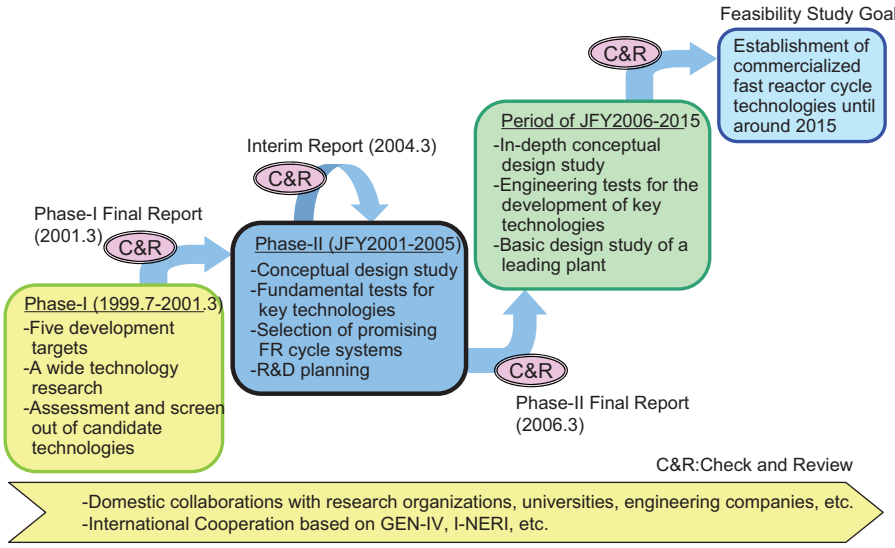

FIG. 2. Time schedule of the FS.

about March 2006, and then the prospects of promising concepts, the R&D plan until about 2015 and the key issues for commercialization will be summarized [5].

2.2.1. Preliminary evaluation of FR concepts

Design studies of four FR concepts have been conducted to elucidate potential performance by taking account of the characteristics of each coolant [6–17]. The core design should have a flexible breeding performance from the viewpoint of plutonium management on the basis of the principle of non-retention of surplus plutonium. A high breeding ratio (BR) of more than 1.2 is not necessary, because the increase in nuclear power generation capacity in Japan is assumed to be less than 20%. Thus, two types of core were designed in the FS:

(1) A resource saving core (a BR of about 1.1);
(2) A core with high burnup and a long operation cycle period, for an economical FR cycle (a BR of about 1.03).

The design requirements for the FR system are summarized in Table 2. Preliminary evaluation results of the FR concepts are as follows:

2.2.1.1. Main features of the sodium cooled FR concept

The main features of the sodium cooled FR concept are shown in Fig. 3. An advanced loop-type system with new materials (i.e. 12% Cr ferritic steel as structural material and oxide dispersion-strengthened steel as fuel cladding material) and innovative technologies (e.g. a compact reactor vessel, a short piping configuration, a reduced number of primary loops, an integral pump and intermediate heat exchanger, a high reliability double-wall-tube steam generator and an early detection system for defects of heat exchanger tubes)

TABLE 2. DESIGN REQUIREMENTS FOR THE FAST REACTOR SYSTEM

Development targets	Design requirements	
Safety	Core damage frequency	$\leq 10^{-6}$/reactor year
	Passive safety	
	Re-criticality free core	
Economic competitiveness	Electricity generation cost	4 JPY/kW·h (JPY: Japanese Yen)
	Burnup	Averaged over core: 150 GW·d/t Averaged over core + blanket: 60 GW·d/t
	Construction period	Large-scale plant: 42 months Medium-scale plant: 36 months
	Construction cost	200 000 JPY/kW(e)
	Operation cycle period	13 months–18 months
	Availability factor	93%
	Plant life	60 years
Reduction of environmental burden	TRU burnup	Acceptable for FR multi-recycling spent fuels and long-term storage of LWR spent fuels
	Compound system doubling time	≤30 years
Efficient utilization of resources	Breeding ratio	1.0–1.2
	FP transmutation	I, Tc
Nuclear non-proliferation	Fuel handling facilities	Physical protection and security system
	New fuels and spent fuels	Limitation of any unintended access

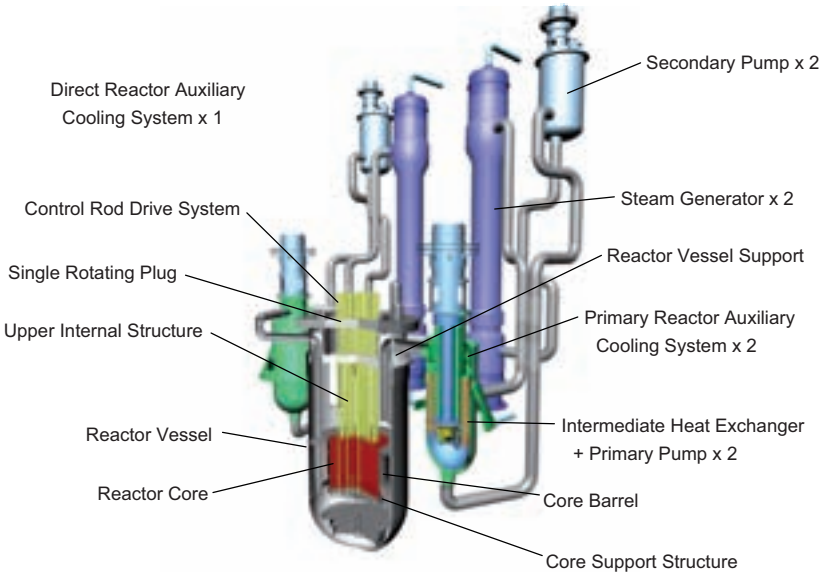

FIG. 3. Design of sodium cooled FR system.

was investigated to achieve a reduction of the plant materials and an improvement of plant reliability including in-service inspection and repair.

The fuel options are MOX and metal. Mixed oxide fuel is highly developed as a result of many years of work, and burnup in the range of 150–200 GW · d/t has been experimentally demonstrated. Metal fuel is selected because of its potential to achieve higher core performance in spite of further development issues. Sodium cooled FRs are the most technologically developed systems. There are extensive technology bases in the fields of design, construction and operating experience with JOYO and MONJU. Key technical issues for commercialization of sodium cooled FRs are cost reduction and in-service inspection and repair. These issues could be solved by R&D on innovative technologies. Sodium cooled FRs would fulfil development targets at a high level as shown in Table 3. The construction costs could be expected to attain around 90% of the target value (200 000 JPY/kW(e)).

2.2.1.2. Main features of the helium gas cooled FR concept

The main features of the helium gas cooled FR concept are shown in Fig. 4. Helium gas cooled FRs pursue a high core outlet temperature up to 850°C and a high thermal efficiency of 47% by employing direct cycle gas

TABLE 3. COMPATIBILITY OF FAST REACTOR SYSTEMS WITH DEVELOPMENT TARGETS

Development targets		Sodium cooled FR (1500MW(e)) MOX Fuel		Helium cooled FR (1500MW(e)) MN Fuel		LBE cooled FR (750MW(e)) MN Fuel		Water cooled FR (1356MW(e)) MOX Fuel
		Resource saving core	Economical core	Resource saving core	Economical core	Resource saving core	Economical core	
Safety		Out- and in-pile tests for passive safety feature and re-criticality-free core (in progress)		Possibility for re-criticality due to hollow fuel assembly and core catcher		Possibility for re-criticality due to prevention due to fuel floating		Possibility for neutron absorber
Efficient utilization of resources	Breeding ratio	1.10	1.03	1.11	1.03	1.10	1.04	1.05
	Compound system doubling time	65 years	—	99 years	—	76 years	—	≥200 years
	Fissile nuclide for initial core	5.7 t/GW(e)	5.8 t/GW(e)	7.0 t/GW(e)	7.0 t/GW(e)	5.9 t/GW(e)	5.9 t/GW(e)	≈11 t/GW(e)
Reduction of environmental burden	MA burnup	Acceptable for LWR spent fuel composition (MA content: 5 mass%) under low-decontaminated recycling condition (FP content: 0.2 vol.%)						Acceptable for low-decontaminated recycling condition (MA content: 2 mass%)
	FP transmutation	Possibility of LLFP (I-129, Tc-99) nuclear transmutation by loading fuel in radial blanket zone						Not yet evaluated

TABLE 3. COMPATIBILITY OF FAST REACTOR SYSTEMS WITH DEVELOPMENT TARGETS (cont.)

Development targets			Sodium cooled FR (1500MW(e)) MOX Fuel		Helium cooled FR (1500MW(e)) MN Fuel		LBE cooled FR (750MW(e)) MN Fuel		Water cooled FR (1356MW(e)) MOX Fuel
			Resource saving core	Economical core	Resource saving core	Economical core	Resource saving core	Economical core	
Economic competitiveness	Burn-up	Averaged over core	147 GW·d/t	150 GW·d/t	121 GW·d/t	123 GW·d/t	154 GW·d/t	155 GW·d/t	88 GW·d/t
		Averaged over core and blanket	90 GW·d/t	115 GW·d/t	69 GW·d/t	89 GW·d/t	105 GW·d/t	128 GW·d/t	45 GW·d/t
	Operation cycle length		26 months	26 months	18 months	18 months	18 months	18 months	18 months
	Availability factor		≈95%	≈95%	≈93%		≈93%		≈93%
	Primary coolant temp.		550°C		850°C		445°C		287°C
	Thermal efficiency		42%		47%		38%		35%
	Load factor		4%		3%		3%		3%
	Construction cost		90%		100%		100%		≈100%

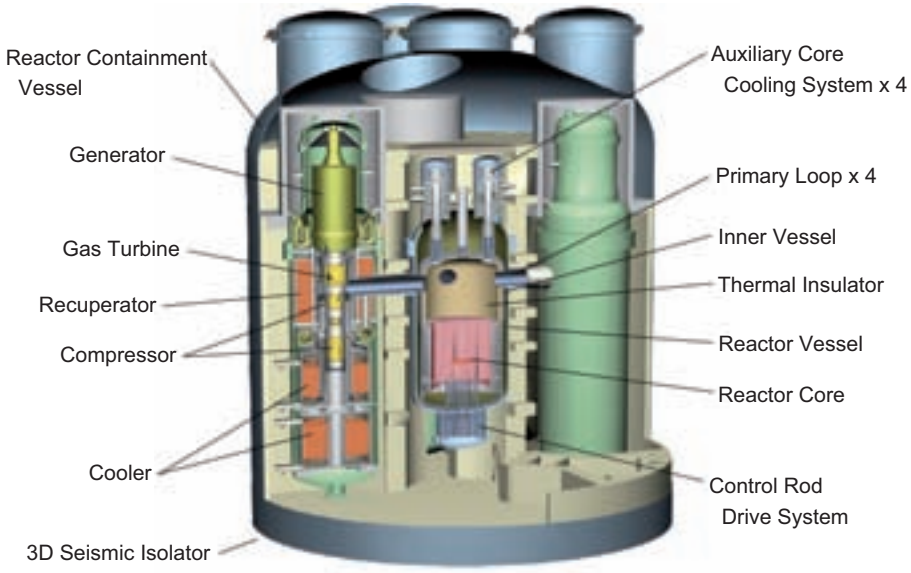

FIG. 4. Design of helium gas cooled FR system.

turbines. The development of a large scale gas turbine of the vertical type generating 300 MW(e) is needed to ensure a compact plant layout. Regarding the core design, a coated particle nitride fuel has been selected as the fuel form to meet the core performance and safety requirements. Development of the particle fuel is a particularly crucial issue for confirming the feasibility of the helium gas cooled FRs. There are many technical difficulties to be solved concerning the particle fuel, such as a fabrication method for reinforced coated particle fuel and fuel assembly, a reprocessing method, an ^{15}N enrichment method, and a fuel safety experiment to elucidate failure limits and molten fuel behaviour.

In spite of technical difficulties, there is strong interest in the helium gas cooled FR around the world due to its high outlet temperature, which enables multipurpose applications, such as an efficient hydrogen generation. To enhance the development of helium gas cooled reactors, international cooperation is preferable because international concern about gas cooled reactors is high. The construction cost was evaluated in the FS to be around 100% of the target value.

2.2.1.3. Main features of the LBE cooled fast reactor concept

The main features of the LBE cooled FR concept are shown in Fig. 5. In the design of LBE cooled FRs, earthquake resistance is one of the most significant issues. A large scale plant design is not acceptable due to the difficulty of satisfying the mandated standards to withstand earthquakes, so that plant size is limited to a medium scale modular FR of around 750 MW(e). In addition, the results of basic erosion tests indicated that the core outlet temperature is limited to 445°C, in order to control corrosion behaviour. The capital cost was evaluated to be about 100% of the target value. The LBE cooled FR would have the potential to achieve the development targets, but there are several fundamental issues to be solved before its technical feasibility can be clarified.

2.2.1.4. Main features of the water cooled FR concept

The main features of the water cooled FR concept are shown in Fig. 6. Much of the technology base for water cooled FRs can be found in existing

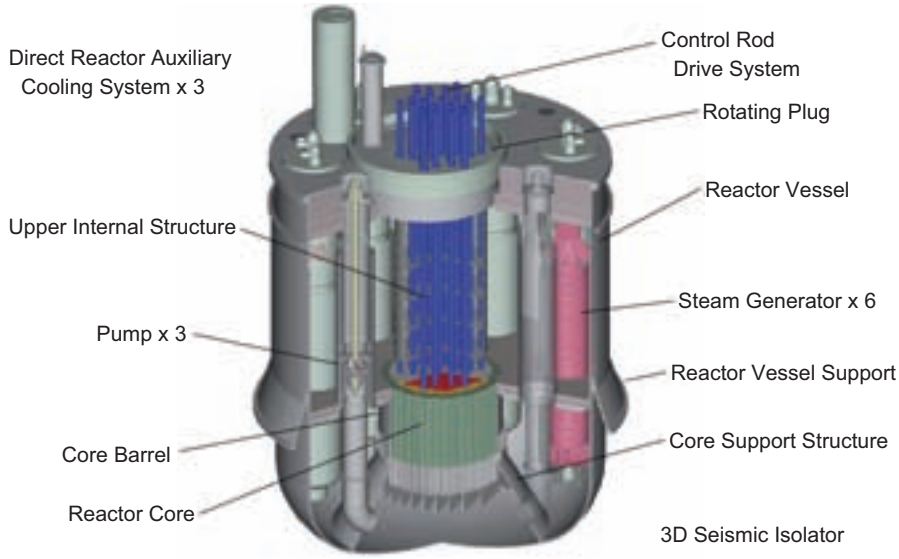

FIG. 5. Design of LBE cooled FR system.

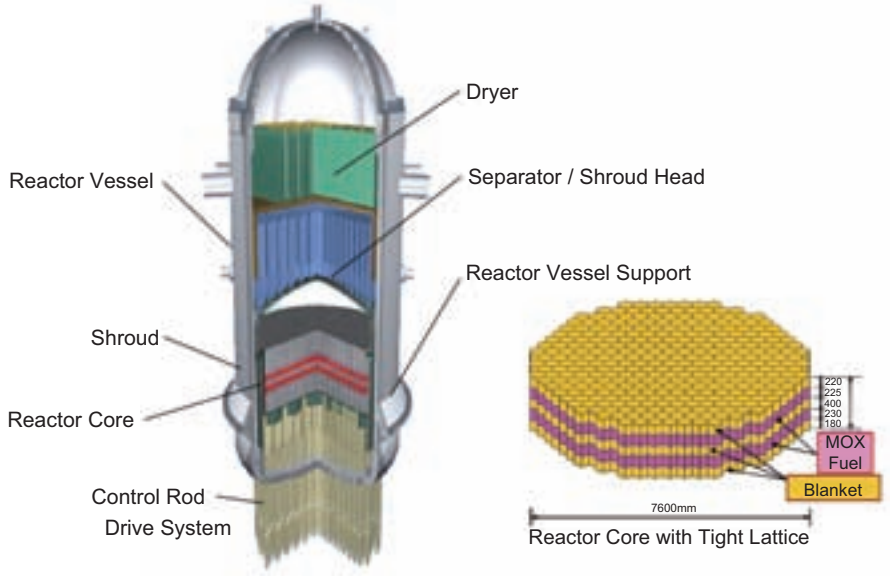

FIG. 6. Design of water cooled FR system (Courtesy of R&D of JAERI).

LWRs. Thus, the key issues are limited to the fuel cladding material and the safety related technology. The core performance is restricted, owing to the use of water as a coolant; the breeding ratio is slightly over 1.0, the average burnup including the blanket is 45 GW·d/t and the fissile plutonium inventory is 11 t/GW(e). The burnup is the lowest of the FR concepts in the FS, and the fissile plutonium inventory is two and a half times larger than that of sodium cooled FRs. The capital cost is considered to be around 100% of the target value, because it could be assumed to be the same as that of future LWR plants.

2.2.2. Preliminary evaluation of fuel cycle systems

In this paper, a fuel cycle concept is expressed in the paired form of 'reprocessing technology + fuel fabrication technology'. The four main fuel cycle concepts [17–28] examined in the FS are:

(1) Advanced aqueous processing + simplified pelletizing;
(2) Advanced aqueous processing + sphere packing;
(3) Oxide electrolysis + vibropacking;
(4) Metal electrorefining + injection casting.

Design studies of commercial plants were performed for two plant capacities (50 and 200 tHM/a), to check the dependence of the fuel cycle cost on plant capacity. The design requirements of the fuel cycle concepts are summarized in Table 4.

Preliminary evaluation results of the fuel cycle concepts are as follows:

(1) The main process flowsheet for advanced aqueous processing + simplified pelletizing is shown in Fig. 7. The advanced aqueous process consists of a simplified process with the addition of a uranium crystallization step, a single cycle co-extraction step of U, Pu and Np, and an MA recovery step.

TABLE 4. DESIGN REQUIREMENTS FOR THE FUEL CYCLE SYSTEM

Development targets	Design requirements	
Safety	Uncontrollable release out of the hot cell boundary	The occurrence probability of the extremely unlikely event that massive radioactive nuclides are uncontrollably released out of the hot cell boundary should be less than 10^{-6} /year/plant. If existing safety regulations and guidance of nuclear plants are not directly applicable, measures of safety assurance should be identified.
Economic competitiveness	Fuel cycle cost	Reprocessing and fuel fabrication cost <0.8 JPY/kW·h
Reduction of environmental burden	Radioactive waste volume	The radioactive waste volume per unit power generation should be less than those for the LWR systems; e.g., high level waste volume < 5×10^{-4} m^3 /GW·h.
	Leakage of the U and TRU	<0.1% (target)
Efficient utilization of resources	Leakage of each of U and TRU	>99%
Nuclear non-proliferation	Safeguard system and non-pure plutonium handling	Secure implementation of the safeguard system is facilitated by remote monitoring, remote operation, appropriate material accountancy and prevention of pure plutonium handling.

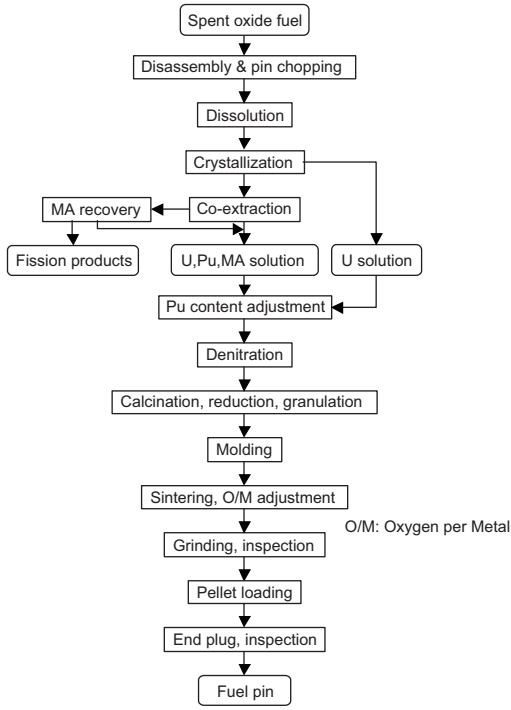

FIG. 7. Main process flowsheet for advanced aqueous processing + simplified pelletizing.

The crystallization step removes most of the bulk heavy metal and eliminates it from downstream processing. The purification step of U and Pu in the conventional process is eliminated, and they are co-extracted with Np. The simplified pelletizing process is rationalized by elimination of the powder blending step and the granulation step from the conventional MOX pellet process. The prospects for the technical feasibility at the commercial level of this concept would be relatively high as a result of many years of work by JNC at Tokai. The fuel cycle cost for a large scale plant (200 tHM/a) could satisfy the design requirement as shown in Table 5. Recovery of U/TRU was estimated to be greater than 99%. The key technical issue for commercialization of the advanced aqueous process is scaling up of the additional steps. Furthermore, it is important to demonstrate the production of MOX pellets containing MAs and trace amounts of fission products in a hot cell facility that is remotely operated and maintained.

TABLE 5. COMPATIBILITY OF FUEL CYCLE SYSTEMS WITH DEVELOPMENT TARGETS

Development Targets	Advanced aqueous process and simplified pelletizing	Advanced aqueous process and sphere packing	Oxide electrolysis and vibro-packing (vipac)	Metal electrorefining and injection casting
Safety	Existing regulations and guidance of nuclear plants are applicable	Existing regulations and guidance of nuclear plants are applicable.	Practical measures can be taken for criticality control, although detailed examination may be necessary.	Practical measures can be taken for criticality control, although detailed examination may be necessary
Economic competitiveness	200tHM/y plant: 0.28–0.48 JPY/kWh 50tHM/y plant: 0.61–1.17 JPY/kWh	200tHM/y plant: 0.34–0.56 JPY/kWh	50tHM/y plant: 0.56–0.89 JPY/kWh	50tHM/y plant: 0.44–0.83 JPY/kWh
Reduction of environmental burden	Bolosilicate glass: 2.8×10^{-4} m^3/GWh	Bolosilicate glass: 2.8×10^{-4} m^3/GWh	Phosphate glass: 2.1×10^{-4} m^3/GWh	Glass-bonded sodalite: 5.6×10^{-4} m^3/GWh
Efficient utilization of resources	99% recovery of each of U and TRU is possible	99% recovery of each of U and TRU is possible	MA recovery method should be explored	99% recovery of each of U and TRU is possible
Nuclear non-proliferation	Current material accounting system is applicable and Pu is always accompanied by U	Current material accounting system is applicable and Pu is always accompanied by U	Specific material accounting system should be developed and Pu is always accompanied by U	Specific material accounting system should be developed and Pu is always accompanied by U

(2) The main process flowsheet for advanced aqueous process + sphere packing is shown in Fig. 8. The advanced aqueous process is described above. In the sphere packing fuel fabrication process, spherical MOX fuel particles are made by use of the gelation method and then packed in the cladding tube. Two sizes of sphere are required to obtain a fuel smear density of more than 80%. The technical feasibility of the sphere packing process was confirmed from the recent test results by JNC at Tokai. However, the fuel cycle cost of advanced aqueous processing + sphere packing was about 20% higher than that of advanced aqueous processing + simplified pelletizing, because the increased number of fuel particle fabrication lines resulted in the process building having a larger volume (two fabrication lines are necessary for each type of fuel: inner driver, outer driver and blanket).

(3) The main process flowsheet for oxide electrolysis + vibropacking is shown in Fig. 9. This is a kind of non-aqueous fuel cycle concept for MOX fuel developed originally at the Research Institute of Atomic Reactors (RIAR) in the Russian Federation. A simultaneous electrolysis step, a

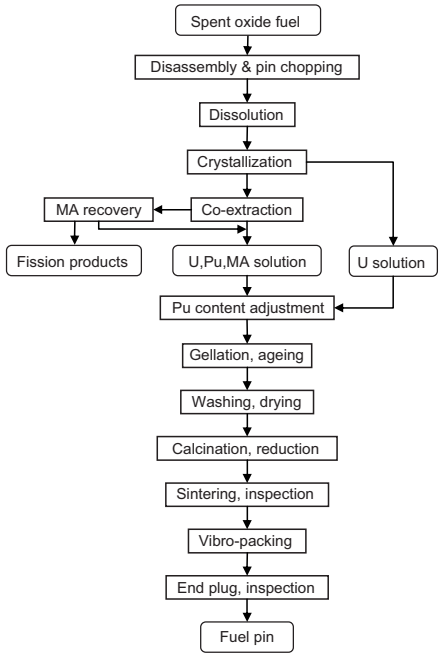

FIG. 8. Main process flowsheet for advanced aqueous processing + sphere packing.

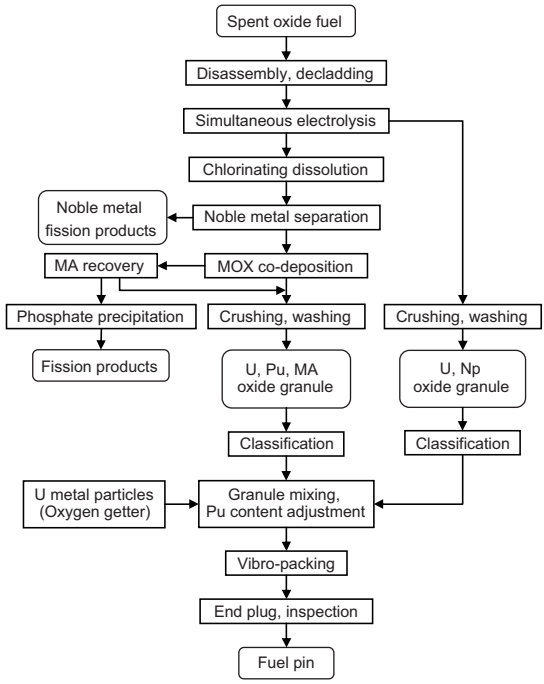

FIG. 9. Main process flowsheet for oxide electrolysis + vibropacking.

MOX co-deposition step and an MA recovery step are incorporated into the original scheme to satisfy the development targets of the FS. This concept has the potential for high economic competitiveness, and the fuel cycle cost for a small scale plant (50 tHM/a) could satisfy the target value; however, there are uncertainties about technical feasibility. Recent test data on the laboratory scale have indicated that the operating conditions for MOX co-deposition should be optimized to improve the efficiency currently achieved. In addition, the MA recovery process, which attains 99% recovery of MAs, should be explored. A long term R&D project, including construction of infrastructure, will be needed to realize commercialization of the oxide electrolysis + vibropacking concept.

(4) The main process flowsheet for metal electrorefining + injection casting is shown in Fig. 10. This pyroprocessing fuel cycle concept was originally developed for metal fuels at Argonne National Laboratory (ANL) in the United States of America, and some improvements have been made in the FS to enhance operationality and throughput. On the basis of extensive laboratory scale technical data, the fuel cycle concept could be compatible with the development targets; however, there are a few

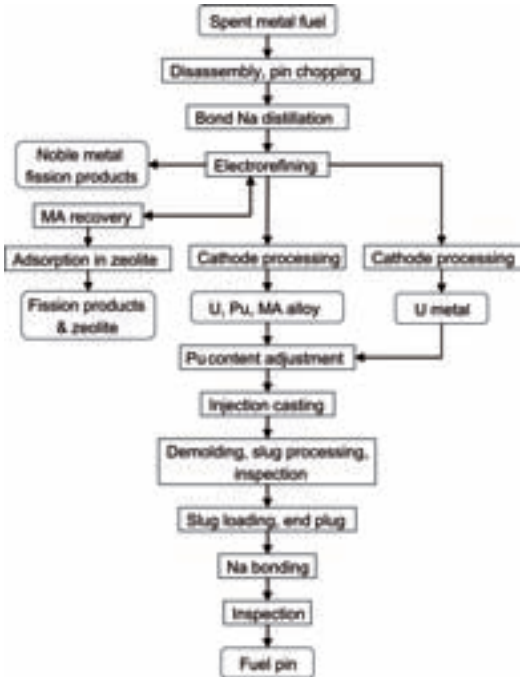

FIG. 10. Main process flowsheet for metal electrorefining + injection casting.

technical issues that need to be addressed to confirm its feasibility. Engineering scale tests of a high recovery process for TRU and a reduction of HLW volume are needed to realize the commercialization of the concept. International cooperation in R&D for the engineering tests will be necessary, because the existing worldwide infrastructure is insufficient in the area of the pyroprocessing fuel cycle.

2.3. Optimum coordination with reactor and fuel cycle concepts

On the basis of the technical summary of FR concepts and fuel cycle concepts in the preliminary evaluation of the FS, the sodium cooled FR would be more suitable for the design requirements than the other FRs, and the pairing of the advanced aqueous process + simplified pelletizing would be the most promising fuel cycle concept. From the viewpoint of the best performance FR cycle, the combination of 'the sodium cooled FR with MOX fuel core, the advanced aqueous reprocessing process and the simplified pelletizing fuel fabrication process' was expected to be the most preferable, because the concept is more technically advanced and conforms more with the

development targets than the other concepts. The electricity generation cost of the sodium cooled FR cycle was evaluated to be 2.2–3 JPY/kW · h. This could be lower than that of an advanced LWR cycle, and is the most cost competitive as the future baseload supply.

The combinations of 'the sodium cooled FR with metal fuel core, the metal electrorefining reprocessing process and the injection casting fuel fabrication process' and 'the helium gas cooled FR with nitride fuel core, the decladding advanced aqueous reprocessing process and the coated particle fuel fabrication process' were recommended as the options for the complementary concepts. The first concept of a metal fuel has the potential for higher core performance, and the second concept has attractive features, such as high thermal efficiency, hydrogen production and high temperature heat utilization. The helium gas cooled FR cycle has a good potential for reactor performance; however, there are several fundamental issues about its technical feasibility that need to be clarified: for example, the development of coated particle nitride fuel including fabrication, reprocessing and ^{15}N enrichment methods and materials with high heat resistance.

2.4. Development steps towards commercialization

The commercialization of the FR cycle system needs flexible and step by step R&D, taking the required budget and period into account with timely checks and reviews. In JFY 2006, MEXT is scheduled to review the results of the phase II study of the FS. The expected results of the FS are:

(a) Clarification of the most promising FR cycle concept as the main choice for commercialization and of the complementary concepts for continued development of the key technologies;
(b) The R&D plan until around 2015;
(c) Future problems to be solved for commercialization of the FR cycle system to be promoted.

On the basis of the check and review of the FS, MEXT is planned to recommend its main choice and the fundamental policy to be followed in the future plan.

In the R&D plan up to 2015, innovative technologies based on engineering scale R&D will be developed and the design study optimized to determine the feasibility of key technologies and the user requirements for commercial FR cycle plants. In this phase, technical data will be systematized and a commercialization project plan will be established as essential outputs of the FS.

Between 2015 and 2030, assuming successful completion of engineering scale R&D, licensing of construction and operation of a pilot (or demonstration) system will take place, to confirm the performance and reliability of the FR cycle system in partnership with industry and perhaps other countries. Research and development activities will undertake the acquisition of performance data and optimization of the system during this phase.

Between 2030 and 2050, assuming successful completion of demonstration R&D, the detailed design, licensing, construction and operation of commercial plants will take place, to confirm the economic competitiveness and operational availability. After 2050, a fully fledged commercial deployment of the FR cycle could start for replacement of retired LWRs.

On the basis of the above described development steps for commercialization, an example of transition scenarios from the LWR cycle to the FR cycle is shown in Fig. 11. In the evaluation of the transition scenario, it is important to indicate the timing of the technology demonstration, taking into account the lead time of R&D, design specifications, plant sizes, and technical continuity of the LWR cycle and FR cycle technologies. In addition, it is essential that the development of the FR and related fuel cycle is promoted under a unified strategy.

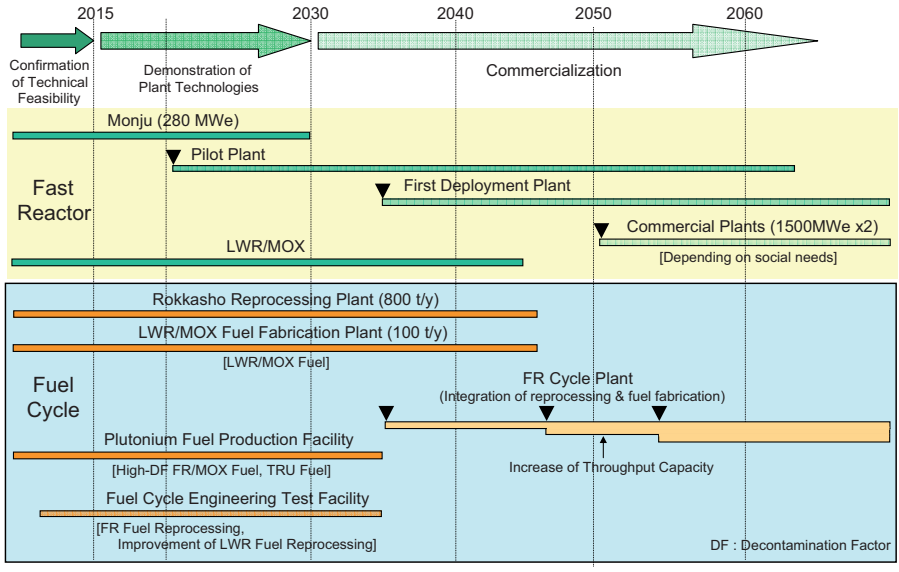

FIG. 11. Example of transition scenario from the LWR cycle to the FR cycle.

3. DEPLOYMENT SCENARIO FOR THE FAST REACTOR CYCLE

3.1. Prospects

On the basis of Japan's energy supply and demand perspectives, the electric energy requirements will grow slowly and then saturate at a level of 1.2–1.3 times the current value around 2020–2030, because of a balance between a steady increase of energy demand from the residential sector and a decrease in the population. The share that nuclear energy has of electricity generation is expected to increase steadily, owing to a reduction in the use of fossil energy sources in order to reduce greenhouse gas emissions. Therefore, the electricity generation using nuclear power plants is expected to supply at least 30–40% or more after 2030. Nuclear power generation is important as the baseload energy supply in Japan [29].

The LWRs at present in operation will come to the ends of their lives from around 2030, assuming their lives to be 60 years. The Rokkasho reprocessing plant will reach the end of its life in the late 2040s, so that renewal of this plant will then be needed. As for the FR cycle, R&D will be promoted step by step to realize commercial plants by around 2050.

3.2. Possible transition scenario to the FR cycle system

Continuous use of LWRs may lead to depletion of uranium resources and accumulation of HLWs. From the background of the sudden increases in the fossil fuel cost and the future energy demand, particularly from growth in China and other Asian nations, energy resources with a low environmental burden will be indispensable in the middle of the twenty-first century.

In this section, a possible transfer scenario from the LWR cycle to the FR cycle was studied to clarify the advantages of the FR cycle scenario by comparing it with the LWR once-through scenario. The deployment of commercial FR cycle plants is assumed to start from 2050. The nuclear power generation capacity was 46 GW(e) in 2000 and is expected to increase until 2030, to 58 GW(e), and thereafter remain at this capacity [30].

The main assumptions about characteristic data of reactor and fuel cycle systems are given in Table 6. The average burnups of LWR and FR fuels are 45–60 and 150 GW · d/t, respectively. As for FR cores, the economical type (with a breeding ratio of 1.03) and the resource saving type (with a breeding ratio of 1.1) are prepared to enable a switch-over according to the plutonium balance. At first, the resource saving type core is used to advance the replacement of retired LWRs, and then the economical type of core is used so as not to retain surplus plutonium for the replacement of FRs. The lifetime of

TABLE 6. MAIN ASSUMPTIONS ABOUT THE CHARACTERISTIC DATA OF REACTOR SYSTEMS AND FUEL CYCLE SYSTEMS

Item		Assumption
Reactor System	LWR	BWR, PWR: Burnup 40 GWd/t, for Reactor which will be deployed by 2019 Load factor 80% ABWR, APWR: Burnup 60 GWd/t, for Reactor which will be deployed after 2020 Load factor about 90%
	FR	Sodium cooled type reactor with mixed-oxide fuel Breeding ratio 1.1 (resource saving core), 1.03 (economical core) Load factor about 90%, MA content 5% (upper limit)
	Life time	60 years for both LWR and FR
Ex-core Time	LWR	4 years (cooling time 3 years, Reprocessing 0.5 year, Fabrication & transportation 0.5 year) (irradiation period about 4-6 years)
	FR	5 years (Cooling time 4 years, Reprocessing 0.5 year, Fabrication & transportation 0.5 year) (irradiation period about 8-11years)
Loss Factor	LWR	Conversion 0.5%, Fabrication 0.1%, Reprocessing U: 0.4%, Pu: 0.5%, MA: 0.1%
	FR	Fabrication 0.1%, Reprocessing 0.1%
Reprocessing Plant	LWRs	JNC's Tokai : 2001-2005; 40 ton/year, abolished in 2010 Rokkasyo : 2005-2010; Plan value, 2011-2046; 800 ton/year, abolished in 2047 2047- ; 800 tonHM/year (with MA recovery process)
	FRs	Primary plants introduce 50 ton/year, and are expanded at unit of 200 ton/year depending on FR deployment capacity appropriately
	Life time	40 years for both LWR and FBR
Other		The uranium recovered from spent fuel is re-concentrated

every type of reactor is assumed to be 60 years. The ex-core time periods are assumed to be six years for the LWR cycle and five years for the FR cycle. Losses from the fuel cycle are 1.1% for the LWR cycle and 0.2% for the FR cycle. The throughput of the LWR reprocessing plant is 800 tHM/a. Minor actinides are recovered from the HLW in the LWR reprocessing plant of the next generation after 2047. The recovered MAs are recycled into FRs in the form of MOX-TRU fuel. The FR reprocessing plants, which have a throughput capacity of 50 or 200 tHM/a, are installed in response to the number of installed FRs.

The calculated result for the FR cycle deployment scheme is shown in Fig. 12. Fast reactors will replace 1 GW(e) of LWR capacity every year after 2050, and the switch-over to FRs will be almost completed by the late 2120s. In addition, the maximum reprocessing capacity for LWR and FR spent fuels is estimated to be about 1400 tHM/a.

4. URANIUM AND TRU MANAGEMENT

The long term mass flow analyses of LWR once-through and FR cycle scenarios were performed focusing on management of uranium and TRU.

4.1. Cumulative uranium demand

A comparison of cumulative uranium demand between the continuous uranium use of the LWR once-through scenario and the FR cycle deployment scenario in Japan is shown in Fig. 13. The LWR once-through scenario will result in a proportional increase in the cumulative uranium demand, which will

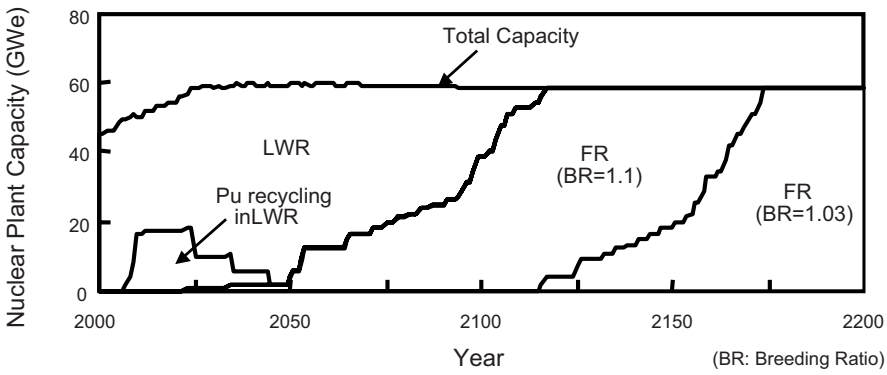

FIG. 12. Possible FR cycle deployment scheme.

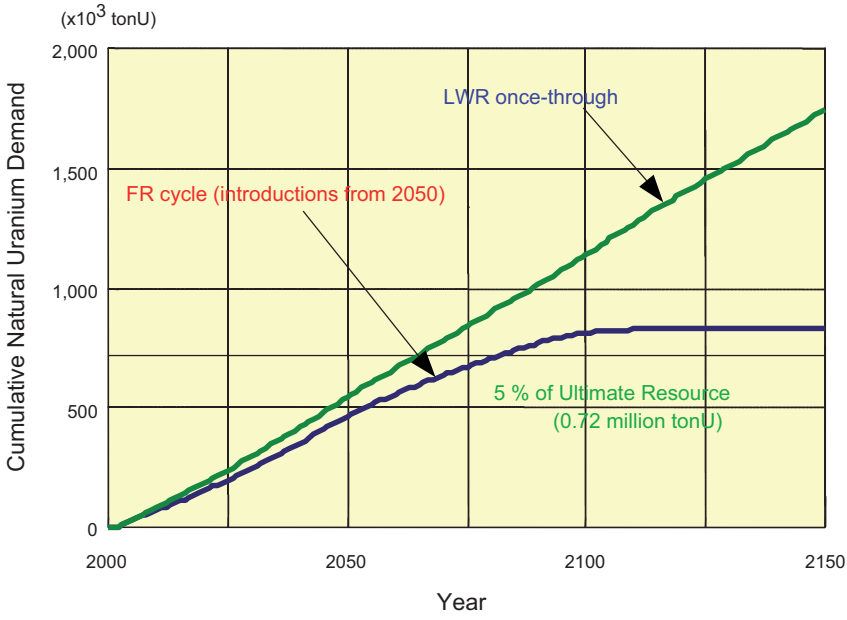

FIG. 13. Comparison of cumulative natural uranium demand for the FR cycle and LWR once-through cycle scenarios.

mean that more than 5% of ultimate resources will be consumed by about 2070. The 5% curve corresponds to the estimated ratio of Japan's nuclear power plant capacity to the world's capacity in the future.

In contrast, the FR cycle scenario will drastically suppress the cumulative uranium demand in the case of commercial introduction from 2050. The cumulative amount consumed will stop increasing at about 800 000 tons at the beginning of the twenty-second century, which corresponds to 6% of ultimate resources. Therefore, sustainable nuclear power generation may be attained, without any dependence on imported uranium resources, by the introduction of the FR cycle.

4.2. Plutonium and minor actinide storage

The accumulations of plutonium and MAs, which will be disposed of as HLW, are shown in Fig. 14. In the case of the LWR once-through scenario, the cumulative amount of plutonium will increase continuously, and will reach about 1800 tons at the end of the twenty-second century with direct disposal of spent fuel in a geological repository. In the case of the FR cycle scenario, the cumulative amount of plutonium will be less than ten tons. This means that the

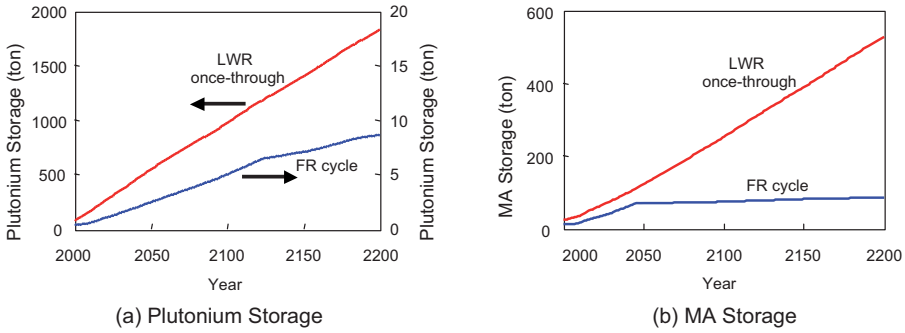

FIG. 14. Comparisons of storage of (a) plutonium and (b) MAs in HLWs for the FR and LWR once-through cycle scenarios.

amount of plutonium to be disposed of will be about three orders of magnitude lower in the FR cycle scenario than in the LWR once-through scenario, because plutonium is incinerated as fuel (the recovery rate in the reprocessing process is assumed to be 99.9%).

The cumulative amount of MAs in the LWR once-through scenario is calculated to rise to about 500 tons at the end of the twenty-second century. In contrast, the increased rate of MA accumulation will drastically decrease from 2050 in the FR cycle scenario, and the cumulative amount of MAs will be reduced to about 85 tons due to recovery of MAs. A great reduction in the contribution of TRU elements to HLWs is expected in the FR cycle scenario, and this may offer significant potential for reducing the volume and cost of a geological repository.

4.3. Potential radioactive toxicity of HLWs

The reduction of the long term radiotoxicity of HLWs from the FR cycle scenario in comparison with the LWR once-through scenario is shown in Fig. 15. The uranium and TRU element recovery rates of reprocessing are 99.9% for the FR cycle scenario. The radiotoxicity is normalized to that of one year after in the LWR once-through scenario. The dotted line indicates the radiotoxicity level of natural uranium necessary for a nuclear power plant of 1000 MW(e) · a.

The radiotoxicity level for the FR cycle scenario decays to approximately 1/240th of the value for the once-through scenario after a thousand years. From the point of view of the time taken to decrease down to the level of the radiotoxicity of natural uranium, the once-through scenario takes about a hundred

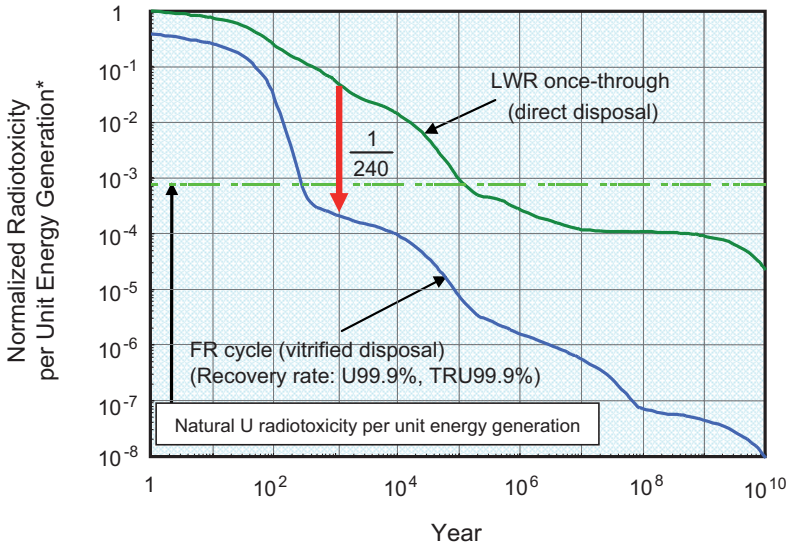

FIG. 15. Radiotoxicity reduction of HLWs.

thousand years. The FR cycle scenario could shorten this period to several hundred years, because all TRU elements will be recycled with a high recovery rate. Recycling and improvement of the recovery rate of TRU elements greatly reduce the potential hazard of HLWs. The reduction effect of the FR cycle on environmental burden is important in obtaining public approval for the FR cycle [31, 32].

4.4. Accumulation of HLWs

A comparison of the cumulative weight of all nuclides in HLWs for the LWR once-through scenario and the FR cycle scenario is shown in Fig. 16. The amount of all the nuclides to be disposed of as HLWs in the LWR once-through scenario will reach about 120 000 tons in the middle of the next century if the disposal of HLWs to a geological repository is started from the late 2030s.

The cumulative weight of all the nuclides in the FR cycle scenario could fall to about a 1/30th of that in the LWR once-through scenario because uranium and TRU elements are not disposed of. Separation of bulk uranium may permit more efficient use of uranium resources and repository capacity. Furthermore, the recovery of TRU elements may minimize emplacement in a repository of nuclear materials suitable for weapons use.

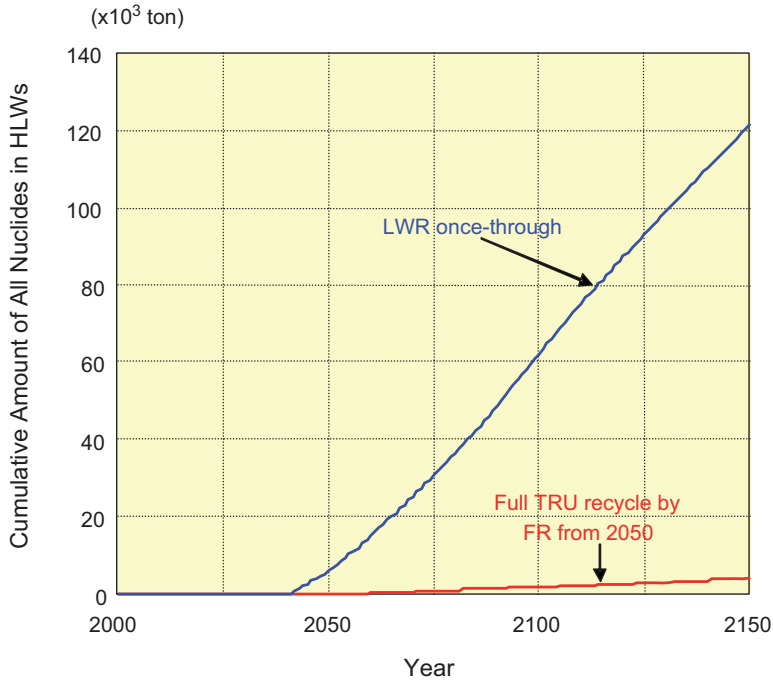

FIG. 16. Comparison of cumulative amount of all the actinides in HLWs for the FR and LWR once-through cycle scenarios.

5. CONCLUSIONS

The establishment of a nuclear fuel cycle is the basis of Japan's nuclear energy policy. Spent fuel from LWRs will be reprocessed by the Rokkasho reprocessing plant with an annual throughput of 800 tHM. Plutonium extracted from LWR spent fuels will be recycled in LWRs in the form of MOX fuel. Spent fuels exceeding the throughput of the reprocessing plant will be temporarily reserved in an interim storage facility.

The FR cycle has potential benefits for an energy supply sustainable in the long term, prevention of global warming and a reduction of the long term waste toxicity source term sent to a geological repository. To realize these benefits, Japan has been pursuing the development of FR cycle technologies, in which plutonium and MAs are recycled, as the future energy systems for replacing retired LWRs. To steadily promote commercialization of the FR cycle, a design study and the key technology R&D have been conducted in the FS. The phase II study of the FS is scheduled to be completed in March 2006,

and it will clarify the most promising FR cycle concept, the R&D plan until around 2015 and the future issues to be solved on the route to commercialization.

From the perspective of the preliminary evaluation in the FS to date, a combination of sodium cooled FRs with MOX fuel cores, the advanced aqueous reprocessing process and the simplified pelletizing fuel fabrication process was recommended as the main choice for the most promising concept. This concept exceeds the other concepts in technical advancement, and its conformity with the development targets was higher.

The study of this potential scenario indicated that if the commercial deployment of FR cycle plants is started in around 2050, sustainable use of nuclear power will be possible without depending on uranium resources from overseas in the future. In addition, the cumulative weight of all the nuclides in the FR cycle scenario could decrease to about 1/30th of that for the LWR once-through scenario because uranium and TRU elements are not disposed of. The separation of bulk uranium may permit more efficient use of uranium resources and repository capacity. Furthermore, the recovery of TRU elements may yield the minimization of emplacement of nuclear materials suitable for weapons use in a repository. The introduction of the FR cycle could lead to a drastic decrease of the uranium and TRU elements accumulating as HLWs.

Nuclear power generation contributes to economic efficiency, supply stability and environmental conservation of the Japanese baseload energy supply. Nuclear fuel cycle technologies have the potential to improve further on these attributes, and to permit an environmentally harmonized and sustainable nuclear power generation for a long time. Therefore, Japan has been promoting the technological development of spent fuel reprocessing and the recycling of recovered uranium, plutonium and MAs. Fast reactor cycle technologies could significantly increase the efficiency of uranium utilization, and could probably reduce the long term radioactivity of HLWs. Anticipating the diversity of future social needs, a steady effort in the development of FR cycle technologies is important in securing promising energy system options for the next generation.

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POTENTIAL CONTRIBUTIONS OF FAST REACTORS AND NEW REACTOR CONCEPTS TO FISSILE MATERIALS AND MA MANAGEMENT

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Abstract

The current management of spent uranium fuels in the LWR fleet includes direct disposal, temporary storage or processing, and recycling of plutonium in the form of mixed oxide fuel. This last option allows a reduction of the storage capacity required for spent fuels in the short term. In order to eliminate the main actinides (plutonium and the minor actinides) that represent the long lived radiotoxic component of current ultimate wastes (whether disposal is direct or not), a basic and physically optimal scenario (for the system consisting of reactors and fuel cycle facilities) is proposed, which foresees the optimal use of natural resources and partitioning of MAs in fourth generation fast neutron reactors, keeping the proliferation resistance level and remaining economically competitive. Following a physical analysis of the respective potentials of the fast neutron or thermal neutron spectra for transmutation and natural resources use, an analysis is presented of scenarios ranging from the current PWR fuel cycle to a full fourth generation system, including recycling stages for all of the actinides: uranium, plutonium and minor actinides. The paper will present a preliminary analysis of the various scenarios, taking into account the constraints and inventories in all

installations of the fuel cycle (fabrication and processing), including reactors and final disposal. Fast neutron systems allow global recycling of actinides or optimum use of natural resources by plutonium recycling on the basis of their intrinsic physical characteristics, minimizing the impacts on the fuel cycle facilities and improving the performance of the global fuel cycle by removing all front end facilities, this being strongly related to the availability and cost of uranium.

1. PHYSICS OF TRANSMUTATION

1.1. Physics of the transmutation of minor actinides

Transmutation can be achieved by fission or by capture reactions. In the case of fission, a heavy nuclide is transformed into fission products (FPs), most of which are short lived (less than 50 years). In the case of capture, a heavy nuclide is transformed into another element that does not necessarily yield a significant reduction in the medium or long term radiotoxicity. However, the isotopes produced can be transmuted by fission or capture. As much as transmutation by successive captures towards higher elements with higher activities has to be limited, transmutation by fission is to be favoured.

Thus, when analysing the physics of recycling, we evaluate two parameters:

(1) The ratio $\alpha = \sigma_c/\sigma_f$, which indicates the probabilities of decay through capture rather than through fission at the first neutronic interaction;
(2) The production of higher elements with curium (^{245}Cm), which is representative of the successive neutronic captures and the efficiency of transmutation.

Table 1 presents the mean effective cross-sections for capture and fission and their ratio, for the main isotopes of actinides and for three neutronic spectra: one PWR thermal spectrum with a UO_2 fuel, one PWR epithermal spectrum with mixed oxide (MOX) fuel and one fast spectrum for fast neutron systems.

The capture/fission ratio is reduced by a factor of 5–10 with the change from a PWR spectrum (thermal or epithermal) to a fast neutron system. Fast neutrons are therefore more efficient at transmuting minor actinides by direct fission.

For 'capturing' isotopes ($\alpha > 1$) such as ^{241}Am and ^{243}Am, the efficient capture cross-sections (Figs 1 and 2, respectively) remain higher than the fission cross-sections, up to a fission threshold at a few MeV. The main method of transmutation will therefore be through capture, with a larger probability when the α ratio is higher.

TABLE 1. NEUTRON CROSS-SECTIONS OF THE ACTINIDES AND THE σ_c/σ_f RATIO

Isotope	Thermal neutron spectrum (PWR)			Epithermal neutron spectrum (PWR-MOX)			Fast neutron spectrum (FNR)		
	σ_f	σ_c	$\alpha = \sigma_c/\sigma_f$	σ_f	σ_c	$\alpha = \sigma_c/\sigma_f$	σ_f	σ_c	$\alpha = \sigma_c/\sigma_f$
^{235}U	*38.8*	*8.7*	0.22	*12.6*	*4.2*	0.3	*1.98*	*0.57*	0.29
^{238}U	*0.103*	*0.86*	8.3	*0.124*	*0.8*	6.5	*0.04*	*0.30*	7.5
^{239}Pu	*102*	*58.7*	0.6	*21.7*	*12.2*	0.6	*1.86*	*0.56*	0.3
^{237}Np	*0.52*	*33*	**63**	*0.6*	*18*	**30**	*0.32*	*1.7*	**5.3**
^{241}Am	*1.1*	*110*	**100**	*0.8*	*35.6*	**44.5**	*0.27*	*2.0*	**7.4**
^{243}Am	*0.44*	*49*	**111**	*0.5*	*31.7*	**63.4**	*0.21*	*1.8*	**8.6**
^{242}Cm	*1.14*	*4.5*	3.9	*0.96*	*3.45*	3.6	*0.58*	*1.0*	1.7
^{244}Cm	*1.0*	*16*	**16**	*1*	*13.1*	**13.1**	*0.42*	*0.6*	**1.4**
^{245}Cm	*116*	*17*	0.15	*33.9*	*5.4*	0.2	*5.1*	*0.9*	0.18

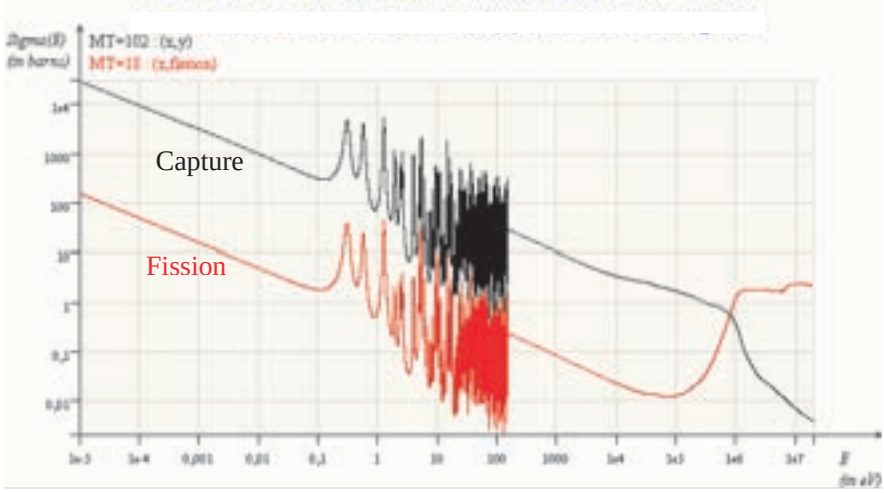

FIG. 1. Neutron cross-sections of ^{241}Am.

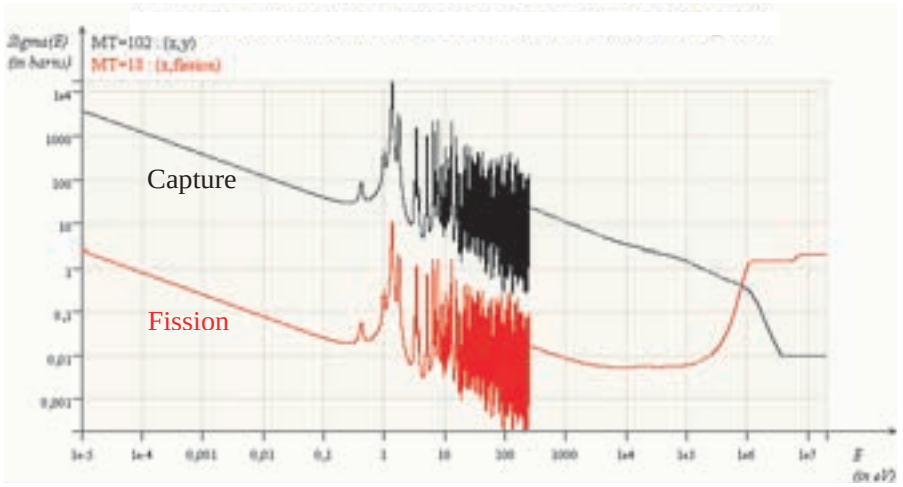

FIG. 2. Neutron cross-sections of ^{243}Am.

Conversely, 'fissile' isotopes (such as ^{239}Pu, Fig. 3) mostly transform into short lived FPs.

The main paths for transformation of neptunium and americium isotopes in PWRs are shown in Fig. 4.

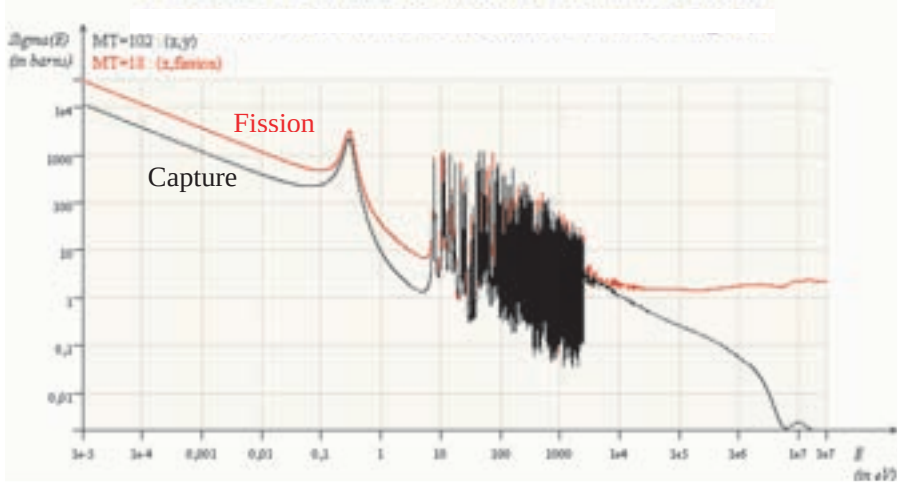

FIG. 3. Neutron cross-sections of ^{239}Pu.

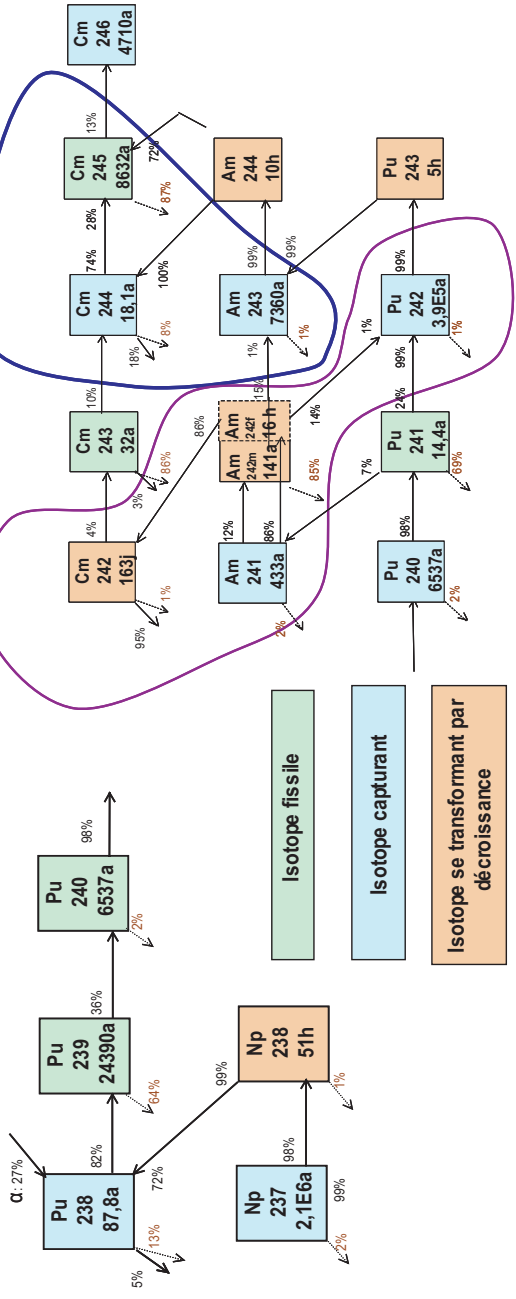

FIG. 4. Main transmutation chains in neutron fluxes for neptunium and americium.

We thus conclude that:

(a) The transmutation by capture of ^{237}Np leads mainly to the production of ^{238}Pu.
(b) The transmutation by capture of ^{241}Am leads mainly to the production of ^{242}Pu and ^{238}Pu via the decay of ^{242}Cm.
(c) The transmutation by capture of ^{243}Am leads mainly to the production of ^{244}Cm and some higher isotopes.

Beyond these main elements resulting from the transmutation, the production of higher elements derived from the successive captures of neutrons indicates significant differences between the PWR and fast neutron spectra: a factor of 10 or more, with the fast spectrum producing overall distinctly smaller amounts of ^{238}Pu, ^{242}Pu, ^{244}Cm and beyond, whether for the transmutation of ^{241}Am or ^{243}Am. These differences result from theoretical studies of the transmutation of ^{241}Am or ^{243}Am alone in both types of spectrum; without taking into account the method of transmutation and its feasibility in the reactor.

Another indicator in the comparison of the efficiency of transmutation depending on the spectrum, is the ratio of the masses of ^{246}Cm or higher ranking isotopes produced by transmutation of the ^{241}Am and ^{243}Am in PWR neutron and fast neutron spectra, which highlights an accumulation under the thermal or epithermal spectrum conditions of PWRs. The cumulative production of isotopes higher than ^{245}Cm (Fig. 5) from americium is thus very distinctly less in fast neutron systems than in PWRs (in the ratio of 5–30).

These physical characteristics allow us to propose a scenario for transmutation of the minor actinides in fast neutron systems in which the actinides are fully recycled without an accumulation of higher isotopes, since the trend is towards a low level concentration at equilibrium of the recycling. This trend cannot be transposed for transmutation in thermal or epithermal neutron spectra. In these spectra, it is essential to separate and not recycle curium, to avoid the constant accumulation of higher isotopes.

Figures 6 and 7 illustrate the evolution of the ^{252}Cf inventories in PWRs and fast neutron reactors (FNRs), respectively. The evolution of the ^{248}Cm inventories is similar. The cumulative inventories indicate a difference of more than three orders of magnitude between recycling in PWRs and recycling in FNRs (×2500 at ten recyclings).

The concentration of isotopes of higher rank than plutonium drives the radioactivity (α, β and γ radiation) and the neutronic emission of the fuel, and, consequently, the radioprotection arrangements or possibly the adaptation of the processes for partitioning of the spent fuels and re-manufacturing of the fuel for recycling.

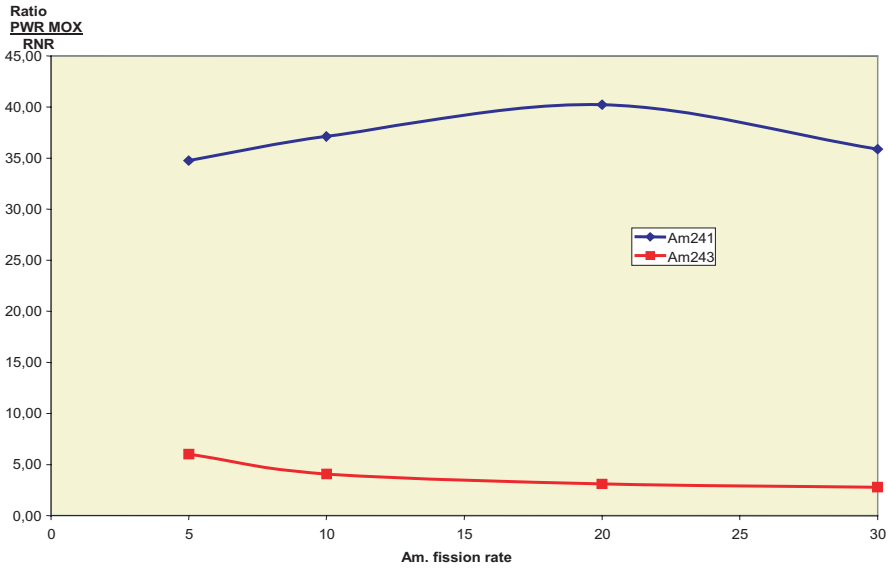

FIG. 5. Ratio of the masses (thermal/fast spectra) of the isotopes of higher rank than ^{245}Cm produced by transmutation of ^{241}Am or ^{243}Am.

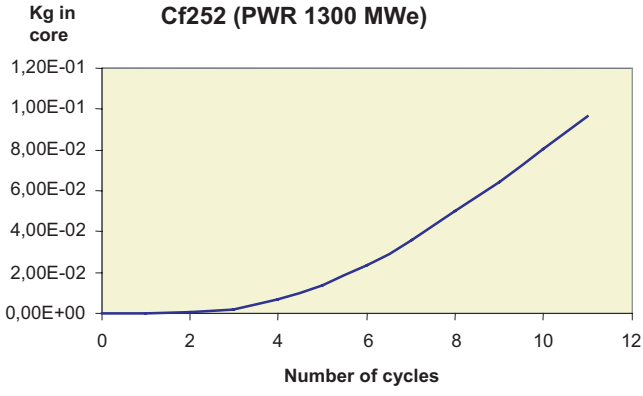

FIG. 6. Mass of ^{252}Cf in the reactor during multiple recycling of actinides in PWRs.

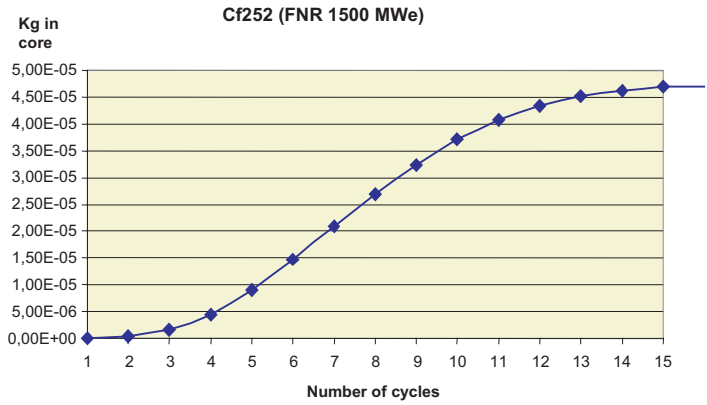

FIG. 7. Mass of ^{252}Cf in the reactor during multiple recycling of actinides (in homogeneous mode) in FNRs.

1.2. Physics of the core and fuel cycle: Constraints on the introduction of minor actinides

The main constraint on the introduction of minor actinides into a PWR or an FNR relates to their direct impact and the impact of their transmutation products on the reactivity coefficients and on the core kinetics parameters, as well as on the neutronic source, the power (α), the dose rate (γ radiation and neutrons) and the criticality to be managed in the fuel cycle. Indeed, among the main products of the transmutation of neptunium or americium:

(a) The ^{238}Pu fraction has an impact on the thermal output (α) during the stages of manufacture and reprocessing.
(b) The ^{242}Pu fraction has an impact on the coolant void effect in PWRs.
(c) The production of curium and higher elements will have an impact on the spent fuel processing and temporary storage stages.

A fuel containing minor actinides in PWRs or in a fast neutron system induces:

(a) *On the reactor physics parameters*: a reduction in the reactivity loss in the reactor, a degradation of the reactivity temperature coefficients and of the soluble boron efficiency (in PWRs), and an increase of the reactivity effect relating to the coolant voidage.

(b) *On the fuel cycle physics parameters*:
  (i) *In spent fuel processing*: A production of higher isotopes that display prejudicial characteristics in terms of the α source or neutronic sources.
  (ii) *In fuel manufacturing*: Consequences relating to the recycled elements:
    — Neptunium does not significantly modify the overall power outputs, activities or neutronic sources, but results in an increased dose rate.
    — Americium results in a strong increase in the γ dose rate.
    — Curium has major and sometimes unacceptable consequences in some respects: increases in the γ dose, the neutronic sources and the α source, as well as criticality in the case of its specific partitioning.

The principal parameters for analysis of the transmutation performance in reactors are:

(a) The specific power, which influences the masses of the inventories in the cycle;
(b) The minor actinides fraction, whose limits are given in the following sections;
(c) The effective cross-sections and levels of neutron flux, which define the transmutation rates by fission or by capture and which determine the inventories of a system at equilibrium.

1.2.1. Limitations on minor actinides loading into PWRs

The results of neutronic studies are used to evaluate the maximum minor actinide fraction allowed in PWRs to maintain acceptable values for the coefficients of reactivity (the temperature and density of the moderator, fuel temperature and the efficiency of soluble boron) as follows:

(a) Under 1% for moderation ratios of less than 2;
(b) Around 1% for moderation ratios of 2, and 2–3.5% for moderation ratios of 3–4, with a MOX fuel (Fig. 8).

In addition, the introduction of actinides requires an increase in ^{235}U enrichment or in the plutonium fraction to maintain the same core fuel management (burnup) as that of the reference UO_2 fuel. These values depend, secondarily, on the isotopic composition of the different actinides.

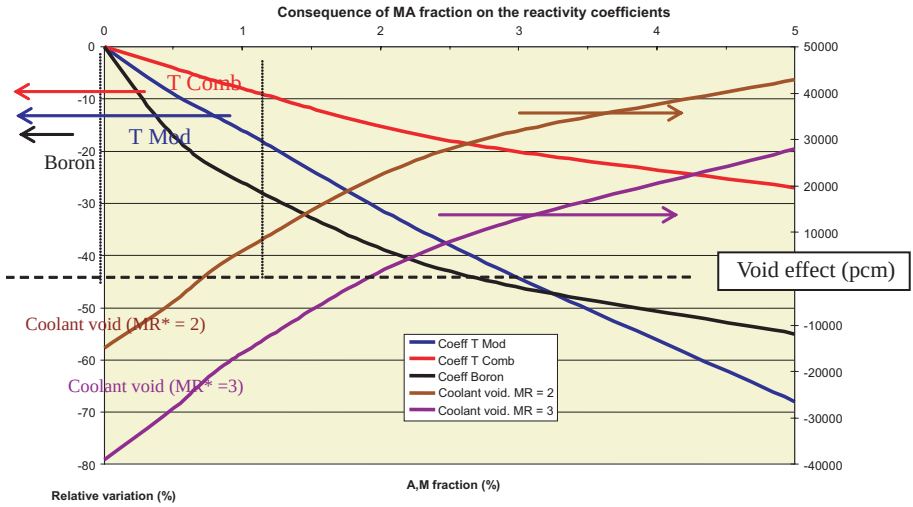

FIG. 8. Relative variation (in per cent) of the reactivity coefficients (moderator and fuel temperature, soluble boron) as a function of the minor actinide fraction (in per cent) in the fuel (MR: moderation ratio (Vm/Vf) EPR = 2 and HM-PWR = 3); T Comb: reactivity coefficient related to the fuel temperature (Doppler); TMod: reactivity coefficient related to the temperature and density of the moderator; Boron: reactivity coefficient related to the efficiency of soluble boron).*

A reduction of the temperature coefficients or boron efficiency indicates a lowering of the reactivity feedbacks. The reactivity coolant void effect must remain negative, imposing a minor actinide content of about 0.8% in a MOX fuel for a moderation ratio of 2, or 1.9% for a moderation ratio of 3.

In general, the main effect of introducing minor actinides into PWRs is to reduce the reactivity feedback effect relating to coolant void and the drop in boron efficiency (a means of control).

1.2.2. Limitations on minor actinides loading into FNRs

The results of the neutronic studies indicate that the maximum minor actinide fractions allowed in fast neutron systems with a sodium coolant in order to maintain acceptable values for the reactivity effects related to sodium voidage, the Doppler effect (temperature of the fuel) and the kinetic parameters (delayed neutron fractions) are as follows:

(a) About 2.5% for the large sized cores of the Super Phenix or European Fast Reactor (EFR) types (Fig. 9);
(b) About 5% for small sized cores of the PRISM or Phenix types.

With the impacts of neptunium and americium being comparable, and that of curium being lower, the reactivity effects indicated in Fig. 9 are also representative of a general recycling of all of the actinides.

For fast neutron systems, switching from a liquid metal coolant (sodium fast reactor (SFR)) to a gas coolant (gas fast reactor (GFR)) enables elimination of the constraint on the minor actinide fraction related to the coolant void effect, and taking into account the specific limits of the kinetic parameters and Doppler effect (Table 2). In this case, that fraction could be increased to 5%, in which case the limit relates principally to the fuel design and the release of helium induced by the α radioactivity into the fuel.

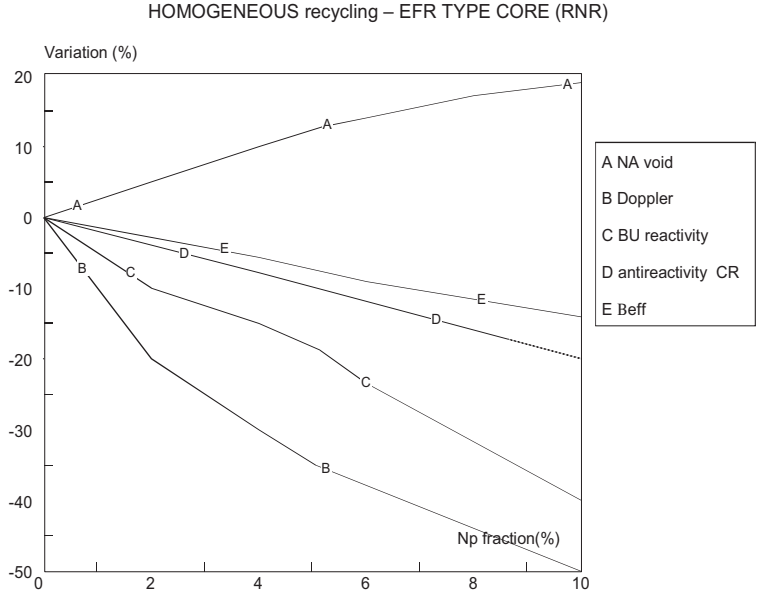

FIG. 9. Relative variation in the reactivity coefficients (fuel temperature, control rod (CR) antireactivity, efficiency, reactivity and coolant voidage) depending on the minor actinide fraction in the fuel.

TABLE 2. EFFECT OF COOLANT VOID (BEGINNING OF LIFE/END OF LIFE) AS PER CENT OF REACTIVITY

MA fraction in the fuel (homogeneous recycling)	Case of the SFR (EFR) with sodium coolant	Case of the GFR with helium coolant
0	1.8/2.7	0.2/0.24
2.5	2.2/2.9	0.24/0.26
5	2.4/3	0.27/0.28

1.2.3. Impact of the recycling method on the actinides

If the actinides are recycled in the homogeneous mode (distributed throughout the fuel), the initial loading in the core is limited by the following criteria regarding the kinetics and reactivity coefficients:

(a) In PWRs, mainly the efficiency of the soluble boron and the coolant void coefficient;
(b) In the fast neutron system, the Doppler effect and the coolant void coefficient.

These constraints limit the transmutation performances.

If the actinides are recycled in the heterogeneous mode, in the form of targets introduced into the core, the neutrons are supplied by the surrounding standard fuel. The transmutation performances are then limited by the number of targets that are acceptable in the reactor, resulting in limits that are fairly similar to the masses of minor actinides tolerable in the homogeneous mode, or to lower limits in the case of recycling in the form of moderated assemblies.

2. LIMITATIONS RELATED TO CURRENT OR ADVANCED TECHNOLOGIES

2.1. Current fuel cycle facilities: Limitations and constraints

The limitations and constraints relating to the introduction of plutonium and minor actinide recycling are described in this section. In the facilities of the MOX fuel manufacturing cycle or spent fuel processing, the limitations mainly relate to:

— Thermal outputs;
— Criticality risks;
— Dose rate and radioprotection;
— Qualification of the processes.

These constraints limit the mass flows managed in the workshops and installations of the cycle, taking into account the composition of the fuels and their characteristics, listed above.

The main limitations for a MOX fuel manufacturing plant are as follows:

(a) A 12% maximum content for plutonium.
(b) For criticality control purposes, the generic constraints are related to criticality limits. The systems have been designed (geometry, layout, etc.) on the basis of these constraints.
(c) The thermal outputs are limited. The main consequences of these limits for the manufacture of fuel containing high ^{238}Pu fractions or α emitters are production masses lower than the design or the maximum design potential production rate.
(d) A limitation both on the individual dose rates and on the collective dose rate. The manufacture of fuel containing neptunium or americium results in higher doses, thereby limiting annual production capacity.
(e) The industrial process of manufacturing from powder and sintering with americium requires different arrangements.

The main constraints and limitations to be taken into account for a spent fuel processing facility are as follows:

— Criticality;
— Thermal output and radiolysis effects.

2.2. Behaviour of minor actinides PWR fuel

The design of the MOX fuel has to be adapted, depending on the addition of minor actinides. In the case of an addition of approximately 1% of americium, a MOX fuel required:

(a) An enriched uranium (MOX EU) supply from the first recycling stage, typically with 8% plutonium and 1.13% americium;
(b) A fissile height reduced from 420 to 400 cm;

(c) An initial pressure in the rod reduced from 25 to 15 bar to accommodate the production and release of increased amounts of helium, mainly due to the α decay of ^{242}Cm.

For a more degraded plutonium isotopic composition, the design of the rod must be evaluated in greater detail (adaptation of the height of the fissile column and of the initial pressure).

The stability of americium oxide under irradiation and during manufacture (sintering) remains to be validated, as changes in stoichiometry can appear at temperatures of 1200°C and above. This may result in the need to develop a specific manufacturing process for MOX fuel with americium.

Additionally, the design of the EPR core results in a reduction of the linear power and an increase of fuel operating temperature (of approximately 200°C), compared with the previous PWRs. This effect facilitates obtaining high burn rates of the order of 60 GW/t or more. The increase of the operating temperature in the case of assemblies with a higher moderation ratio requires additional evaluation of the behaviour of the fission gases in MOX EU assemblies with or without americium, depending on the plutonium and americium fractions.

2.3. Temporary storage of minor actinides

The temporary storage facilities must meet the same safety requirements as do other nuclear facilities:

— Containment of radioactive nuclear materials;
— Limitation of external exposure;
— Control of subcriticality;
— Evacuation of the thermal output, with preference given to passive cooling.

The envisioned temporary storage could be that for material of a solid form (oxide pellets) consisting of actinides dispersed in a uranium matrix, cladded and placed in air cooled containers designed to control both heat and criticality, while allowing for the release of the helium related decay principally associated with the decay of curium.

In the case in which curium is managed separately by being stored temporarily to allow it to decay over a period of approximately a hundred years, two factors must be taken into account:

(a) After 100 years, sufficient ^{244}Cm will remain to ensure that it be recovered in a shielded chain (1 g of ^{244}Cm emits 40 rad in neutrons).
(b) The criticality risk will have to be evaluated very carefully, in particular during the recovery of these materials, whatever the end purpose is (irradiation, temporary storage or ultimate disposal).

3. THE GANEX PROCESS FOR FOURTH GENERATION SYSTEMS

The partitioning process envisaged at present to manage all actinides (GANEX) in a grouped manner is directly derived from studies carried out for the partitioning of americium and curium from FPs (DIAMEX-SANEX), and it is based on the same principles (Fig. 10).

The GANEX process includes two partitioning steps, so that some of the uranium can be extracted on the upstream side. This partial partitioning is imposed by the core design of FNRs, which requires that the core should be broken down into zones with different U/Pu ratios so as to have a better radial distribution of power. The remainder of the uranium and other actinides is then extracted from the FPs. Another reason for a preliminary partitioning of uranium is to reduce the material flow treated by the GANEX process. Finally, co-conversion of actinides can be incorporated into the process.

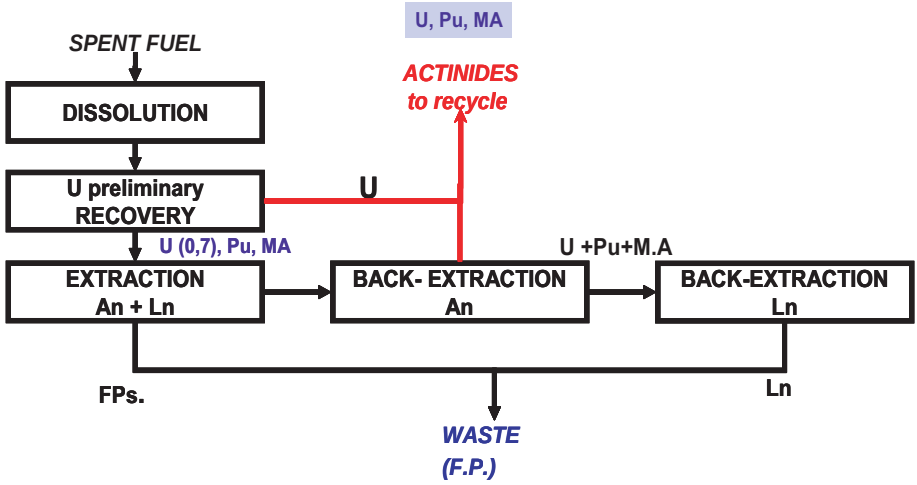

FIG. 10. Diagram showing the principles of partitioning processes for grouped partitioning of actinides (An: actinides, Ln: lanthanides).

4. SCENARIO STUDIES

Detailed scenario studies allow evaluation of the impact of fuel cycle strategies on reactors and facilities such as fabrication or processing plants, on long lived radionuclides storage and on the economy of electricity production by nuclear power plants. The term 'strategy' signifies the choice and implementation time of fuel cycle options such as a once-through uranium cycle, plutonium or overall actinide management and the thorium fuel cycle.

Before performing a detailed or transition scenario study using the COSI code, preliminary physical analysis of the systems and simplified steady state scenarios[1] are required to prepare the data, as well as to preselect some major characteristics that potentially fulfil the requirements.

This section presents various evaluations carried out in the context of the existing French fuel cycle. At present, in France, PWR spent fuel management involves the stages of processing and recycling plutonium in the form of MOX fuel, loaded into 20 PWRs of 900 MW(e) capacity. This strategy is advantageous in that it reduces the storage capacity needed for spent fuel and optimizes the packaging of the current long lived radionuclide waste consisting of FPs and minor actinides in highly resistant long lasting vitreous matrices. To some extent, these studies can also be applied to other countries.

In the context of the partitioning & transmutation (P&T) concept, solutions to extend and to improve current radioactive waste management were compared, using scenario studies to characterize the various steps of the fuel cycle (enrichment, fuel fabrication, reactors, processing, interim storage and final disposal). Thus, before presenting transition scenarios, the characteristics of different fuel cycles with or without minor actinide management are presented at steady state. On the basis of an analysis of their characteristics, a scenario is proposed that involves introduction of fourth generation fast neutron systems in order to develop a sustainable form of energy production, having an intrinsic transmutation potential to minimize production of long lived radioactive waste.

An analysis is presented of the transition from the current fuel cycle, from single plutonium recycling to the implementation of fourth generation systems, with the purpose of multiple recycling of all actinides: uranium and plutonium for energy production and minor actinides for transmutation and waste mass and activity minimization.

On the basis of the scenario outlined above, we have also examined the consequences of different launch dates for fourth generation systems, to

[1] Multiple and successive material recycling up to an asymptotic status.

determine the advantages of temporary scenarios for recycling plutonium in PWRs before achieving the full capability to recycle all of the actinides in fourth generation systems.

4.1. Preliminary scenario evaluations at steady state: Case of the partitioning & transmutation scenarios

A description follows of scenarios that implement the P&T concept. These scenarios are based upon four main reactor fleet families:

(1) Pressurized water reactors loaded with MIX or MOX-EU fuel (Pu on an enriched U support):
   (i) Plutonium-only recycling in PWRs;
   (ii) Global TRU element recycling (Pu, Np, Am and Cm), all TRU elements being diluted in all of the fuel assemblies, leading to a low MA content in the fuel but requiring higher uranium enrichment and lower plutonium content (the void effect limits the TRU element content).

(2) Fast reactor type reactor self-breeders:
   (i) Plutonium-only recycling;
   (ii) Global TRU element recycling (Pu, Np, Am and Cm);
   (iii) Neptunium and Pu recycling and (Am, Cm) burned in moderated target subassemblies (Am, Cm). The fuel targets in the FR cores are designed for a fission rate of 90% of the initial heavy nuclides, in 10–15 years time. The irradiated targets could be processed after a long decay time and disposed of in the glass.

(3) A PWR reactor fleet loaded with uranium oxide (UOX) fuel, Pu and MAs burned in accelerator driven systems (ADSs).

(4) A PWR reactor fleet loaded with UOX fuel, Pu recycling in FRs and MAs burned in ADSs.

The main fuel cycle characteristics are recorded in Table 3.

The analysis of Table 3 shows that a P&T strategy is efficient in terms of mass reduction in a geological repository. The gain in mass of the TRU elements to be stored is due to partitioning and recycling, and is not linked to the technology of the reactor.

Comparing the different P&T strategies, burning americium and curium in moderated subassemblies does allow a factor of 60 or more to be obtained on the TRU elements stored in the other P&T strategies. Accelerator driven systems and the scenario with once-through burning of americium and curium

TABLE 3. FUEL CYCLE CHARACTERISTICS FOR A STEADY STATE POWER PARK (400 TW(e) · h AND 60 GW(e))

Reactor type	PWR			FR			PWR + ADS	PWR + FR + ADS
Fuel cycle type	Once-through cycle	Pu recycling	Pu + MA recycling	Pu recycling	Pu + MA recycling		Pu + MA recyling	
Pu recycling	No	Yes	Yes	Yes	Yes	Yes	In PWRs	In PWRs and FRs
MA recycling	No	No	Yes	No	Together with Pu	In target subassem.	In ADSs	In ADSs
Front end facilities (annual requirements)								
Natural uranium (t/a)	8360	7320	7600	–	–	–	7580	5400
SWU (10^5 a^{-1})		5.7	6	–	–	–	5.7	
Fabrication capacities (t/a)	820	820	820	340	340	340 FR 2 target mod. subassem.	730 UOX 8.6 ADSs	540 UOX 90 PWR-MOX 60 FR 1 ADS
Reactors								
PWR-UOX (%)	100			–	–	–	89	66
PWR-MOX (%)	–	100	100	–	–	–	–	9.8
FR (%)	–	–	–	100	100	100	–	19
ADS (%)	–	–	–	–	–	–	21	5.2
Back end facilities (annual requirements): Processing and refabrication facilities								
Processing capacities (t/a)	0	820	820	340	340	340	740	690
Storage (annual masses)								
TRU stored (t/a)	118	3.3	0.03	2	0.06	0.2	0.07	0.05
Gain (related to once-through cycle)	1	4	400	6	200	60	170	240

have a lower mass of fuels containing minor actinides but higher impacts on the main fuel cycle parameters (Table 4), far from what can be envisaged at present.

In terms of utilization of natural uranium resources, the closed fuel cycle scenarios with FRs allows a large reduction of the annual requirements for natural uranium compared with PWR scenarios (with or without ADSs). In parallel, these scenarios also allow avoidance of any use of enrichment technologies or investment in enrichment facilities.

The impact on radiotoxicity or on the radioactivity of the spent fuel processing and fabrication steps has been evaluated for the steady state situation, knowing the detailed material isotopic composition at any step of the fuel cycle.

The relative impact on the fuel fabrication plant in terms of decay heat ($W \cdot g^{-1}$ iHM, where iHM stands for initial heavy metal) and neutron source ($n \cdot s^{-1}$) normalized (per kgHM) and compared with a standard 12% MOX PWR fuel is summarized in Table 4.

The impact of the partitioning & transmutation strategy is much lower for the homogeneous recycling in FRs, particularly for neutron sources. As fabrication steps, all P&T scenarios will require fuel fabrication in hot cells. An increase in neutron sources can be envisaged for homogeneous recycling in

TABLE 4. MULTIPLYING FACTOR WITH RESPECT TO REFERENCE CASE (STANDARD MOX FUEL) OF THE FUEL ACTIVITY AND NEUTRON EMISSION IN STEADY STATE SCENARIOS, AT THE FABRICATION STEP AND AT THE USED FUEL PROCESSING STEP

Fuel type	PWR			FR		ADS
Fuel cycle management	MOX as reference	Pu + MA recycling	Pu-only recycling	Pu + MA recycling	MA target subassembly	MA recyling
At used fuel processing step						
Decay heat ($W \cdot g^{-1}$ iHM)	1	×2	≈1	≈2	–	×70
Neutron sources ($n \cdot s^{-1} \cdot g^{-1}$ iHM)	1	×130	≈1	×2	–	×200
At fabrication step						
Decay heat ($W \cdot g^{-1}$ HM)	1	×3	0.5	×2.5	×80	×90
Neutron sources ($n \cdot s^{-1} \cdot g^{-1}$ HM)	1	×8000	1	×150	×5000/ ×10 000	×20 000

FRs. For the other scenarios, despite the reduced annual mass flow, research will have to prove the feasibility of fuels having neutron sources that are thousands of times higher compared with MOX fuel.

In addition, for a country like France, which has already deployed industrial facilities, the scenarios including ADSs would require additional and new dedicated facilities with new processes (pyrochemical or hydrometallurgical) compared with scenarios with PWRs or FRs, owing to the fact that the decay heat is far higher than the level tolerated by existing hydrometallurgical processes.

To assess the efficiency of the actinide incineration option, the radiotoxicity[2] is analysed for between 500 and 100 000 years, where the gain is obtained from Pu, Am and Cm recycling (Fig. 11).

With respect to the open cycle, plutonium-only recycling allows a reduction factor ranging from 3 to 10, according to the fuel cooling time, compared with direct disposal of UOX spent fuels. The reduction factor obtained with FRs is double that of MOX loaded PWRs, and reaches values close to the existing vitrified waste after a few thousand years.

Homogeneous minor actinide multi-recycling, with a loss rate of 0.1%, allows a reduction factor ranging between 200 and 400, irrespective of the reactor type.

Assuming minor actinide processing losses of 1% instead of 0.1% has a limited influence, since at best it has an impact on waste radiotoxicity by a factor of 4. As a final remark, if the minor actinide loss rate is 0.1%, plutonium losses (0.1%) in wastes again make a significant contribution.

The introduction of innovative P&T technologies to manage actinides in a fuel cycle has been evaluated by means of several studies of steady state scenarios with different systems and concepts, in order to choose some options and to perform detailed transition studies applied to the French nuclear power fleet.

Fast reactor system scenarios have more potential to achieve all the main goals together (economic competitiveness, flexibility, sustainability, non-proliferation and MA management), using existing or extended facilities. On the basis of this analysis, the introduction of fourth generation systems has been analysed using a transition scenario from the present to the long term situation.

In addition, detailed studies on the overall fuel cycle (fabrication, reactors, storage of actinides, spent fuel treatment, waste conditioning and disposal) have been made for the same transition scenario.

[2] Normalized to the equivalent mass of natural uranium required to produce the same amount of electricity in the open cycle.

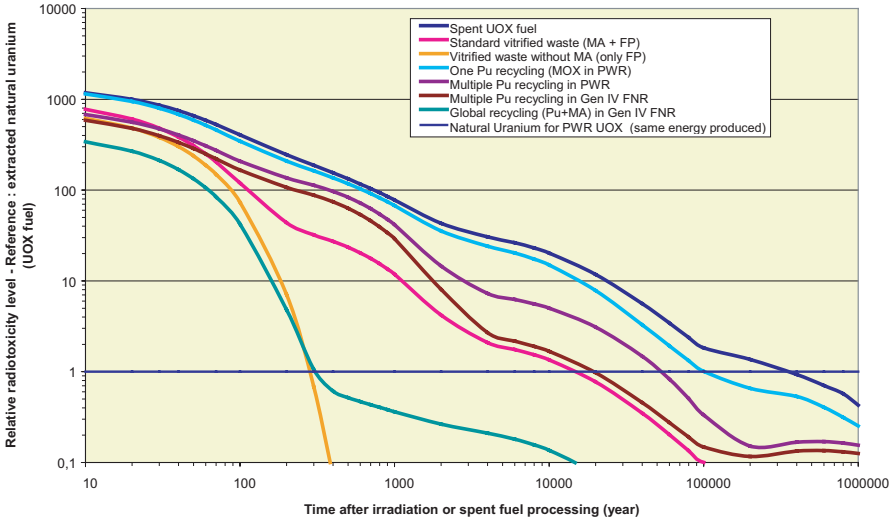

FIG. 11. Radiotoxicity level of TRU elements placed into storage. Key: Spent UOX fuel: direct disposal of irradiated fuel; Standard vitrified waste (MA + FP): glasses with MAs and FPs from UOX spent fuel processing (as produced currently at the La Hague facility); Vitrified waste without MA (only FPs): standard vitrified waste (see above) but without any MAs (only FPs from UOX spent fuel processing); Single Pu recycling in PWR: all TRU elements after single plutonium recycling in PWRs; Multiple Pu recycling in PWR: MAs and FPs from the UOX and MOX spent fuel processing in the case of a scenario with multiple plutonium recycling in PWRs; Multiple Pu recycling in Gen IV FNR: MAs and FPs from FR spent fuel processing in the case of a scenario with multiple plutonium recycling in FRs; Global recycling (Pu + MA) in Gen IV FNR: FPs from FR spent fuel processing in the case of a scenario with multiple plutonium and MA recycling in FRs.

4.2. Transition scenarios: Proposal of a reference for the future

4.2.1. Recall of objectives

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

(a) A reduction of the actinide fraction in vitrified waste to minimize the potential radiotoxicity and thermal load, which determine the size of the deep geological repository;

(b) Use of current facilities and installations to take best advantage of them up to 2030–2040, their planned replacement date, and to prepare for the deployment of future facilities (2040–2100), whether using current technologies or not;
(c) Preparation for the introduction of fourth generation FR (GFR or SFR) systems.

4.2.2. Key steps

To meet these objectives, the steps described in Sections 4.2.2.1–4.2.2.3 were identified as the most important (Fig. 12).

4.2.2.1. In the time frame 2020–2030

In this period, a start will be made to renewal with EPR reactors of 50% of the nuclear power fleet. This renewal relates to the end of the service life of the first PWR plants introduced in 1975–1985 and will be carried out, depending on the prospects at Électricité de France, at the rate of 2 GW(e)/a.

Alternative scenarios to the reference scenario (Section 4.2.3) are the following:

(a) Implementation of advanced partitioning and production of so-called 'light' glass matrices, independently of the scenario that is later deployed;

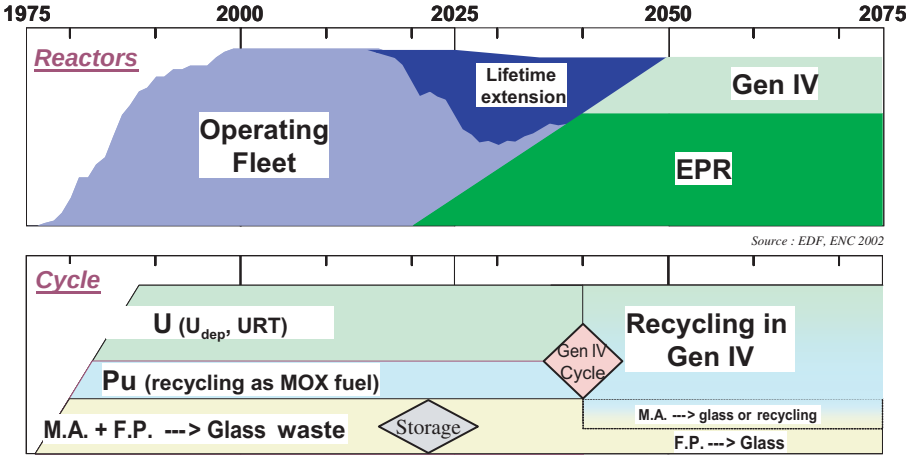

FIG. 12. Transition scheme for French nuclear reactors and fuel cycle facilities.

creation of a temporary storage solution for minor actinides (Am, Np and Cm in a mix or separately, depending on the scenario). This implementation can occur on an industrial scale by adding a workshop to the existing processing facility at La Hague after 2025 or 2040. The date for this study (2020) was chosen before the analysis of industrial optimization, which led to a date of 2025 at the earliest.

(b) Implementation of the advanced processing of spent MOX fuel to perform a second recycling of plutonium in PWRs, by temporarily storing the minor actinides for later recycling in fourth generation systems.

4.2.2.2. In the time frame 2035–2040

In this period, a start will be made to renewal of the remaining 50% of reactors of the previous generation by:

(a) Fourth generation fast neutron systems;
(b) Or EPRs if fourth generation systems are not industrially mature by that date.

Advanced processing will start of spent MOX fuel in order to recycle plutonium and minor actinides in the fourth generation fast neutron systems.

4.2.2.3. About 2080

A start will be made about 2080 to renewal of the EPRs that will first be introduced in 2020 by fourth generation FRs.

4.2.3. Brief description of reference scenario

The reference scenario is based on an introduction of fourth generation FNRs (GFRs or SFRs), at the rate of 2 GW(e)/a, up to 50% of the nuclear power fleet, followed about 2080 by replacment of EPRs (introduced between 2020 and 2035) by FNRs. In 2095, fourth generation (GFR or SFR) reactors will make up 100% of the fleet.

4.2.4. Alternative options in the case of postponed deployment of fourth generation systems

In these options, the fourth generation systems are introduced starting in 2080, at the rate of 2 GW(e)/a, until 100% of the fleet has been renewed. Until that period, the alternative options considered are:

(a) Alternative 1:

(i) A continuation of plutonium mono-recycling in PWRs;
(ii) Interim storage of separated minor actinides until they are recovered;
(iii) Storage of MOX spent fuel.

Finally, recycling will take place in fourth generation fast neutron (GFR or SFR) systems.

(b) Alternative 2:

(i) Multiple plutonium recycling in PWRs in the form of MOX EU fuel, to stabilize the plutonium inventory;
(ii) Temporary storage of the minor actinides from 2020–2040 until their recovery for recycling in fourth generation fast neutron (GFR or SFR) systems.

The minimum delays prior to spent fuel processing are:

(1) Five years for spent fuel from PWRs;
(2) Two years for spent fuel from fourth generation fast neutron systems, which are considered in this study to be self-breeders or slight breeders.

The reduction of the delay between two recycling processes in the reactor enables a reduction of the inventory of recycled actinides in the facilities of the cycle. That same reduction induces handling constraints (on transportation or processing) associated with the higher residual powers that will have to be taken into account in future facility design. Conversely, longer cooling periods prior to reprocessing increase the ^{241}Am inventories due to decay of ^{241}Pu (both outside and inside reactors).

The results obtained for these scenarios (the reference and alternative options) are summarized in Tables 5 and 6 on the basis of the hypothesis described before.

In the once-through cycle, production of ^{241}Am by decay of ^{241}Pu is not negligible, and is related to the long cooling time and no fuel processing or recycling in this case.

TABLE 5. INVENTORIES IN THE FUEL CYCLE FOR SCENARIOS WITH PWRs

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

Inventories (t)	Alternative 1: Single recycling of Pu (MOX)			Alternative 2: Multiple recycling of Pu (MOX-EU)			Alternative 3: Once-through cycle (UOX)			
	2035	2050	2070	2035	2050	2070	2035	2050	2070	2100
Natural U (annual values/ aggregates)[a]	7400/410 $\times 10^3$	7500/520 $\times 10^3$	7500/670 $\times 10^3$	7160/410 $\times 10^3$	7100/520 $\times 10^3$	7000/660 $\times 10^3$	8360/420 $\times 10^3$	8360/550 $\times 10^3$	8360/720 $\times 10^3$	8360/ 970 $\times 10^3$
UTS (annual, MSWU/a)	5.8	5.8	5.8	5.3	5.1	5.1	6.4	6.4	6.4	6.4
Pu (total)[b]	396	479	596	373	398	400	474	612	793	1062
Np	20	32	48	18	31	46	33	46	64	92
Am	51	81	121	52	84	130	63	88	123	174
Cm	4.7	8	9	6.3	10	15	3.4	4	4.3	4.7
MA (total)	76	120	178	76	125	191	99	138	191	271
TRU total	472	599	774	449	523	591	573	750	984	1333
Percentage fuel with TRU in fleet[c]	12%	10%	10%	23%	26%	33%	0%	0%	0%	0%
TRU in storage, including 26.9 t of vitrified waste before 2020[d]	389	529	703	128	175	240	573	750	984	1333

[a] Annual requirements in t/a: cumulative requirements for natural uranium since 1980, in tonnes. World reserves are conventionally accepted as four million tonnes, while world reserves are estimated as 11.4 Mt (<130 $/kg) (cf. Red Book of the OECD NEA/IAEA).

[b] Total comprises that in interim storage, reactors, factories and disposal. These values include the Pu from the decay of Cm in temporary storage. The Pu produced from the stored Cm later recovered is much higher than for alternatives 2 (20 t in 2100) and 3 (32 t in 2100).

[c] Reactor fraction in the fleet containing actinides (Pu or minor actinides).

[d] Inventory of TRU in tonnes (Pu, Np, Am and Cm) in ultimate storage (vitrified waste), in long term storage of non-recyclable irradiated fuels according to the scenario or specific materials (for instance, Cm in the case of alternative 3). For an open cycle, the values at 2035, 2050, 2070 and 2100 are, respectively, 573, 750, 984 and 1333 t. The loss rate during spent fuel processing is 0.1% for all actinides.

TABLE 6. INVENTORIES IN THE FUEL CYCLE FOR SCENARIOS WITH FAST REACTORS

(The inventory values in this table have been rounded up or down to the first significant figure, after summation, except for Cm, rounded up or down to the nearest decimal. They do not include the inventories already vitrified in 2020, except in the last row.)

	Alternative 1: Single recycling of MOX in PWRs and Pu recycling in fourth generation FR systems (SFRs), MA disposed in storage				Alternative 2: Single recycling of MOX in PWRs and global multiple recycling (Pu, Np, Am, Cm, ...) in fourth generation FR systems (SFRs)				Alternative 3: Single recycling of MOX in PWRs and global multiple recycling (Pu, Np, Am, Cm, ...) in fourth generation FR systems (GFRs)			
Inventories	2035	2050	2070	2100	2035	2050	2070	2100	2035	2050	2070	2100
Natural U (annual values/ aggregates[a])	7850/430 $\times 10^3$	4200/510 $\times 10^3$	4200/600 $\times 10^3$	0/660 $\times 10^3$	7850/430 $\times 10^3$	4200/515 $\times 10^3$	4200/600 $\times 10^3$	0/660 $\times 10^3$	7850/430 $\times 10^3$	4200/515 $\times 10^3$	4200/600 $\times 10^3$	0/660 $\times 10^3$
UTS (annual, MSWU/a)	5.9	3.3	3.2	0	6	3.2	3.2	0	6	3.2	3.2	0
Pu (total)[b]	450	566	672	802	455	576	685	848	455	577	698	815
Np	21	32	43	58	22	28	30	22	22	26	19	11
Am	45	69	99	139	50	61	65	47	50	56	45	39
Cm	4	5	6	7	4	7	10	17	4	7	12	14
MA (total)	70	106	149	205	76	96	105	86	76	89	76	64
TRU total	519	671	821	1007	531	672	790	934	531	666	774	879
Percentage fuel with TRU in fleet[c]	0%	50%	50%	100%	0%	50%	50%	100%	0%	50%	50%	100%
TRU in storage[d]	65	103	149	208	27	28	29	30	27	28	29	30

[a] Annual requirements in t/a (cf. Red Book of the OECD NEA/IAEA).

[b] Total comprises that in temporary storage, reactors, factories and disposal. These values include the Pu from the decay of Cm in temporary storage. The Pu produced from the stored Cm later recovered is much higher than for alternatives 2 (20 t in 2100) and 3 (32 t in 2100).

[c] Reactor fraction in the fleet containing actinides (Pu or minor actinides).

[d] Inventory of TRU in tonnes (Pu, Np, Am and Cm) in ultimate storage (vitrified waste), in long term storage of non-recyclable irradiated fuels according to the scenario or specific materials (for instance, Cm in the case of alternative 3). The loss rate during spent fuel processing is 0.1% for all actinides.

4.2.5. Analysis of the results of each scenario

4.2.5.1. Reference scenario: Single plutonium recycling in PWR-EPR followed by recycling in fourth generation fast neutron systems

As from 2095, 100% of the fleet will be made up of fast neutron systems (first introduced in 2035).

The plutonium inventory grows even in 2100 because of the mildly positive increase in breeding in the calculation. A reduction in the breeding gain should enable stabilization of the plutonium inventory, if necessary, at approximately 850 t.

The fuel for a homogeneous recycling situation at equilibrium contains approximately 1.2% of MA (Np + Am + Cm) in a fuel with 20% plutonium. However, absorption of the stock accumulated during the transitional period can be envisioned, with a maximum fraction of the order of 2.5–3% MA in a large sized SFR and up to 5% for a small one (such as Phenix). The introduction of GFR systems capable of accepting a 5% fraction limit would enable the consumption of minor actinides to be increased and therefore a reduction of the inventory in 2100 to a lower level than that of SFRs.

The inventory of minor actinides would be lower in 2100, at a level of about 86 tonnes (64 for GFRs):

(a) The neptunium inventory is down to 22 t (11 for GFRs);
(b) The americium inventory is also down, to 47 t (39 for GFRs);
(c) The curium inventory is approximately 17 t (14 for GFRs).

The ratio of plutonium inventory to minor actinide inventory starts dropping in 2050. The minor actinide inventories in 2100, after the 100% FR fleet has been in operation for five years, come very close to the inventories in 2035, when the fourth generation fast neutron reactors are first introduced, to replace 50% of the fleet over the period 2035–2080.

The natural uranium requirements are 30–40% less than those in the other scenarios. Considering that the French installed fleet of nuclear power plants amounts to 17.5% of the overall world fleet, and extrapolating that fraction to the use of the world's uranium reserves, the introduction of fast neutron systems means that the conventionally agreed level of reserves will not be reached. Without these FR systems, and if the energy requirements were to remain the same or increase, these reserves would be depleted by the end of the century.

The timetable for introduction of specific facilities for the cycle of fourth generation systems is as follows:

(a) In 2030, facilities for fuel manufacture;
(b) In 2040, facilities for reprocessing of fuel in a shielded plant.

A modular reprocessing facility with hydrometallurgical processes would enable, starting in 2040, processing of both spent UO_2 fuels from PWRs and the fuels from fourth generation systems, and would enable grouped management of the actinides. The current process would be transformed into a GANEX type process, after partial reduction of the flow of uranium materials. The resulting products would in this case be a set of uranium and transuranium elements for reuse in the manufacture of fuel assemblies to be recycled in fourth generation (FR) systems. This modular design is based on the GANEX process, which is the subject of a programme of research and experiment.

4.2.5.2. Alternative 1: Single plutonium recycling in PWRs

Mixed oxide manufacture remains fully compatible with the current MELOX factory, i.e. ≈130 t/a production capacity. The amounts of plutonium and MAs (771 t Pu + 264 t MAs in 2100) continue to grow continuously, due to the decay of the ^{241}Pu in ^{241}Am and to the production of minor actinides in the MOX fuel. The percentage of the reactor fleet affected by the recycling of plutonium is between 12 and 10%, depending on the availability of plutonium (assuming just-in-time management of separated plutonium).

In the case of recovery in 2070 of TRU elements from the spent fuels available for reprocessing and their introduction into fourth generation (GFRs or SFRs) systems in 2080, the average MA fraction in the GFR (or SFR) fuel is close to 3.4%, which remains below or compatible with the allowable content in FR cores (5% for GFRs, 2.5% for large SFRs and 5% for small SFRs).

4.2.5.3. Alternative 2: Multiple recycling of the plutonium in EPRs

The need for an enriched uranium support for MOX fuel (associated with the degradation of the isotopic vector and the limit of 12% for the amount of plutonium in the fuel) is effective at the third recycling (support with ≈1.8% ^{235}U), starting in 2045–2055. Prior to 2040, a support of U_{dep} or U_{nat} type is sufficient.

The manufacture of MOX-EU remains compatible with a MELOX type facility, with production increased to ≈230 t/a in 2040 (280 t/a after 2055).

The americium and neptunium inventories increase and differ little in the open cycle, single plutonium recycling and multiple plutonium recycling options, showing the importance of the ^{241}Pu decay for the production of ^{241}Am. The plutonium inventory is stabilized from 2040–2050 at a value of the

order of 400 t, and the fraction of the reactor fleet operating with MOX EU fuel is stabilized at around 33%.

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

The plutonium inventory, stabilized by 2050, will not allow introduction of 60 GW(e) capacity of FRs in 2110. This means either that EPR reactors will be in the fleet up to 2170 or that plutonium recycling has to be stopped in 2060 and UOX burnup reduced to 42 GW · d/tHM from 2060 to 2080, leading to an increased use of natural uranium resources compared with alternative 1.

4.2.6. Reference scenario versus alternatives

The P&T scenario to be implemented in fourth generation FRs in the period 2025–2040 also allows:

(a) To minimize the mass disposed in the final waste at the end of the century, by a factor of 40–50 or more compared with the once-through cycle and by a factor of close to 10 compared with plutonium recycling (in PWRs or FRs) without recycling of minor actinides (Tables 5 and 6);
(b) To minimize the thermal output of the final wastes, allowing a strong and rapid decrease of power with time (Fig. 13);
(c) To minimize the potential radiotoxicity inventory (and radioactivity) in the final disposal;
(d) To make a saving in natural uranium resources by 40%.

Regarding items (c) and (d), after a few hundred years (100–300 years), the activity of the waste is below that of the natural uranium extracted to produce the same amount of energy using a PWR once-through cycle, and the decay heat represents a few watts per gram of waste disposed.

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

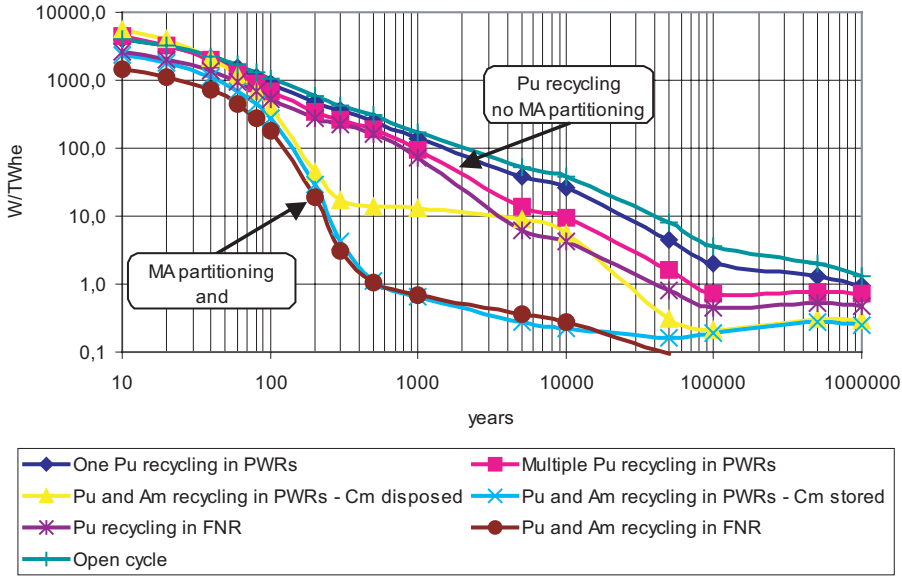

FIG. 13. Decay power of the final wastes (actinides + FPs).

5. CONCLUSIONS

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

The prospect of starting deployment of a first series of fourth generation systems in 2035 bolsters the objective of implementing towards 2020–2030 a system to manage the back end of the PWR cycle with partitioning (and temporary storage) of the minor actinides. Should the deployment of fourth generation systems be delayed, the preceding strategy would still be possible and would still offer all the same advantages, because of the capability of fast neutron systems eventually to recycle the TRU elements produced by PWRs through to the end of the twenty-first century (with, however, increasing restrictions relating to the accumulation of minor actinides because of the ageing of nuclear materials and possible multiple recycling processes in PWRs).

The increasing difficulty involved in recycling plutonium and efficiently burning up all the minor actinides in the PWRs under quite realistic economic and industrial conditions, should favour the deployment, around the middle of

the twenty-first century, of a first series of fast neutron systems to manage the actinides produced by the PWR fleet.

Fast reactors can also allow a saving of up to 40% of the natural uranium consumed during the twenty-first century in the case of the French context and do not require any use of uranium enrichment technologies at the end of the century.

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CORE SAFETY FEATURES OF ADVANCED FUEL CYCLES

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Abstract

Various fuel cycle options have been considered for Generation IV systems for future sustainable nuclear developments, including recycling of minor actinides (MAs). The purpose of the paper is to provide an analysis of the core safety implications related to these advanced fuel cycles, which are designed to optimize natural resources and minimize waste. Plutonium and MA recycling are considered for both thermal and fast reactor cores, although fast reactors are favoured since they are the only ones to produce the necessary extra neutrons for the general objectives of Generation IV systems. These parameters mainly concern temperature feedback coefficients, coolant void coefficients, control rod efficiency, delayed neutron fractions and the mean prompt neutron lifetime. The physical reasons for these changes are emphasized. Typical values of MA maximum loadings are situated in the range of 2–5% for fast reactors and 1–4% for PWRs.

1. INTRODUCTION

Advanced fuel cycles including transmutation of minor actinides (MAs) are part of the strategies for a long term deployment of nuclear power. Several systems (reactors and associated fuel cycles), including ones with transmutation capabilities, are under study worldwide.

Loading standard fuel with MAs for transmutation purposes has limited value since there are several different constraints related to:

— Fabrication;
— In-core fuel behaviour;
— Economics;
— Reactor safety.

The aim of this paper is to summarize the main limitations on both fast reactors and PWRs related to the last constraint.

The relative efficiencies of the two options are not considered here even if it is generally admitted that fast reactors have a clear advantage for this specific function of the nuclear fuel cycle.

2. PHYSICAL PROPERTIES

2.1. Neutron capture cross-sections

As shown in Fig. 1, MAs (Np-237, Am-241, Am-243 and Cm-244) have higher neutron capture cross-sections than U-238. Then, when MAs are loaded into standard mixed oxide (MOX) fuels, a hardening of the neutron spectrum occurs due to higher capture rates.

2.2. Delayed neutrons

The delayed neutron fraction, commonly called β_{eff}, is the main parameter for reactor control. Table 1 provides the corresponding values for

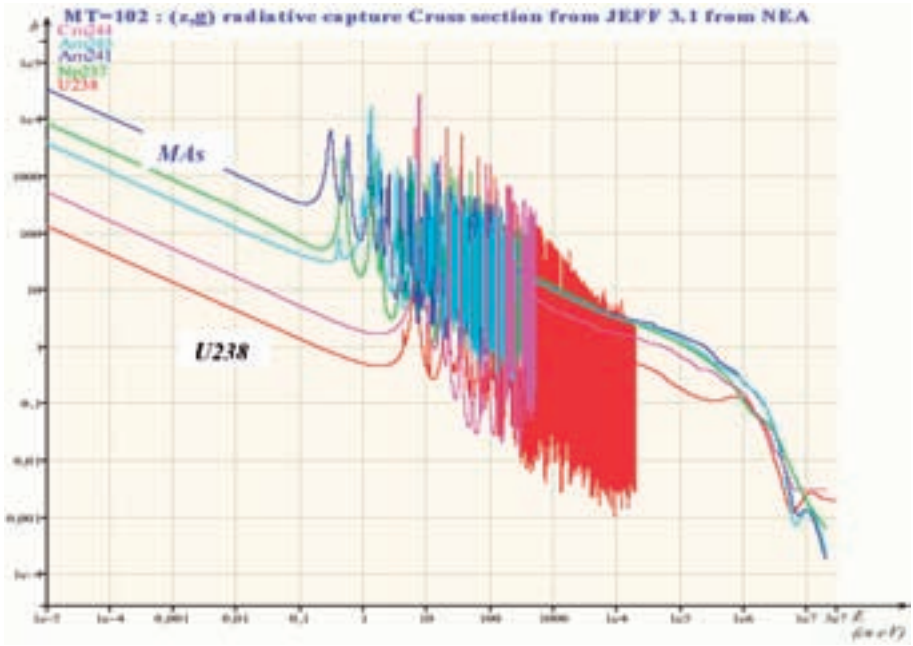

FIG. 1. Minor actinides and U-238 neutron capture cross-sections.

TABLE 1. DELAYED NEUTRON FRACTIONS FOR DIFFERENT ISOTOPES

Isotope	^{235}U	^{238}U	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{237}Np	^{241}Am	^{243}Am	^{244}Cm	^{245}Cm
β_{eff} (‰)	670	1680	220	270	490	640	440	110	250	100	130

different isotopes. The replacement of U-238 by MAs reduces the total β_{eff} of the core.

2.3. Impact of actinide loadings and safety aspects

The modifications to the void, Doppler and delayed neutron coefficients are the main limitations on the total amount of MAs that can be loaded in fuel for transmutation purposes.

2.4. Void coefficient

In the case of sodium cooled fast reactors, the void coefficient corresponds to the impact on the core reactivity of the sodium voiding due to the hardening of the neutron spectrum (sodium has a moderating effect). Two opposing effects appear:

(a) Increase of neutron leakage (reactivity decreases);
(b) Modifications to the cross-sections (reactivity increases).

This last effect is explained by the fact that the capture cross-sections of neptunium and americium are 6–7 times higher than those of U-238 (Fig. 1), so in the case of sodium voiding the corresponding reduction of neutron capture will be higher than that in the case of fuel without MA loadings.

These effects lead to a limit of about 2.5% if no specific sodium void reduction measures are adopted. Such measures include a reduction of the core height, axial heterogeneity, introduction of a moderator into the subassemblies (SAs) and use of neutron absorbing blankets. The combination of the last two modifications was identified as the most promising option, allowing an increase of the limiting MA core fraction of up to 5%.

For gas cooled fast reactors (GFRs), there is no problem with void coefficient modifications because neutron interactions with the gas are limited. The main results of the calculation for a 2400 MW(e) GFR are summarized in Table 2.

TABLE 2. MAIN CORE CHARACTERISTICS OF GFRs

Parameter	Units	Value			
Core diameter/height	(m)	4.44/1.55			
Core management	(fped)[a]	3 × 831 = 2493			
		First cycle		Equilibrium cycle	
Transuranium element fraction	(%)	15.2		18.5	
Pu + MA inventory	(tHM/GW(e))	7.7		10.1	
Discharged BU	(fima)	10.1		10.1	
Maximum damage	(dpa)	163		152	
		BOC[b]	EOC[c]	BOC	EOC
Breeding gain		–0.07	–0.04	0.05	–0.05
Doppler coefficient	(‰)	–1872	–1175	–1405	–968
Voiding coefficient	(‰)	121	253	253	257
Delayed neutron fraction	(‰)	388	344	347	332

[a] fped: Full power equivalent days.
[b] BOC: beginning of irradiation cycle.
[c] EOC: end of irradiation cycle.

One reason for the move to GFRs arises from the low coolant void reactivity coefficient, which allows the relaxation of safety related constraints in cores heavily loaded with MA fuels. Dedicated cores using gas coolant allow larger sized cores because of the low coolant void reactivity.

Even if there is an impact on the relative values of the reactivity coefficients, the absolute values remain favourable, in particular the low helium depressurization effect (Fig. 2). The equilibrium state of GFR cores leads to an amount of MAs in the core of about 1–2%. These MAs increase the breeding gain and degrade the reactivity coefficients in a limited way.

Dedicated cores based on the Existing Technology Gas Breeder Reactor design were studied. It has been demonstrated that a gas cooled reactor can be designed to obtain a power output of 3600 MW(th). A nitride fuel matrix comprising a solid solution with ZrN was found to be suitable from a reactor physics point of view. An acceptable Doppler coefficient can be obtained by incorporating zirconium hydride moderator pins into the fuel S/As. Safety transient studies have been made on a design with a gas cooled MA-burning fast reactor core.

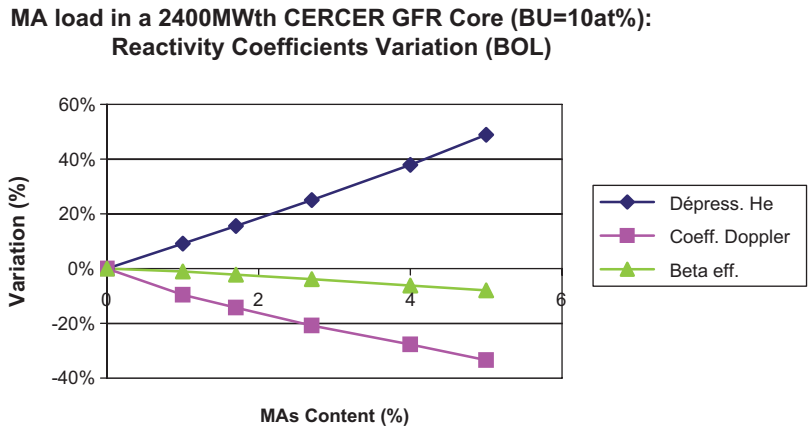

FIG. 2. Effect of MA content on reactivity coefficients in a 2400 MW(th) GFR.

The impact of MA loadings in the range from 0 to 5% on the voiding effect for both sodium and gas cooled reactors are summarized in Table 3.

The same limitations can be observed for thermal reactors, in which neutron captures by americium occur essentially in the energy range from 0.1 eV to a few electronvolts (Fig. 1). In the case of moderator voiding, there are no more neutrons in this range, giving a positive effect on the core reactivity.

Recent calculations on MOX assemblies with enriched uranium support (MOX EU) showed a 1% americium loading limit to cope with voiding issues (Fig. 3).

TABLE 3. RELATIVE REACTIVITY VARIATIONS IN THE CASES OF COOLANT VOIDING FOR SODIUM AND GAS COOLED FAST REACTORS (BOL/EOL)

Minor actinide content in the fuel (%)	Reactivity variation (%)	
	Sodium cooled fast reactor (EFR)	Gas (He) cooled fast reactor
0	1.8/2.7	0.2/0.24
2.5	2.2/2.9	0.24/0.26
5	2.4/3	0.27/0.28

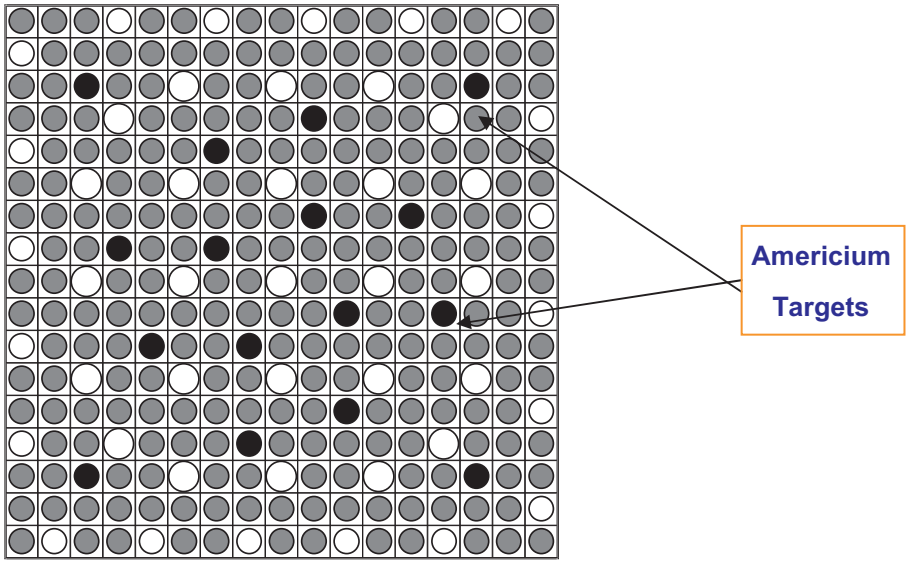

FIG. 3. Americium heterogeneous recycling in MOX EU PWR standard assemblies.

The maximum americium loadings allowed by safety considerations are summarized in Table 4, with two options being considered: homogeneous and heterogeneous recycling. In the latter option, americium is loaded into specific pins (targets).

The neutron spectrum hardening in thermal reactors can be balanced by modifying the core moderation ratio.

2.5. Doppler effect

The Doppler coefficient measures the increase of neutron capture in the resonance range and the corresponding decrease of reactivity, in the case of an increase of the fuel temperature.

TABLE 4. MAXIMUM AMERICIUM LOADINGS IN A MOX EU SUBASSEMBLY (8% Pu)

Homogeneous		Heterogeneous	
Maximum Am loading (%)	U-235 enrichment (%)	Maximum Am loading (%)	U-235 enrichment (%)
1.1	4	1.6	4.6

In the case of a loading with MAs, two main trends can be observed:

(1) The U-238 resonance range (10–10 000 eV) is much broader than those of neptunium and americium (1–100 eV), so the impact of MAs on the Doppler effect is much lower than that for U-238.
(2) The absorption cross-sections of neptunium and americium are lower than those of U-238.

As a consequence, the replacement of U-238 by neptunium and americium will reduce the neutron density and consequently the Doppler effect of U-238 in this energy range. Figure 4 shows the effect of moderation ratio on reactivity increase in the case of coolant voiding.

Pressurized water reactors may accommodate maximum MA loadings of about 1% for a moderation ratio (MR) of 2 (the standard value), and up to 2 and 3.5% for MR = 3 and 4, respectively.

2.6. Reactivity control worth

The neutron spectrum hardening induced by MAs in the fuel leads to a reduction of control rod worth in the case of fast and thermal reactors and of soluble boron worth in the case of PWRs. Even if this is not a limiting factor for maximum loading values, it must be considered in safety studies, and eventually the number and/or the design of the control rods are modified.

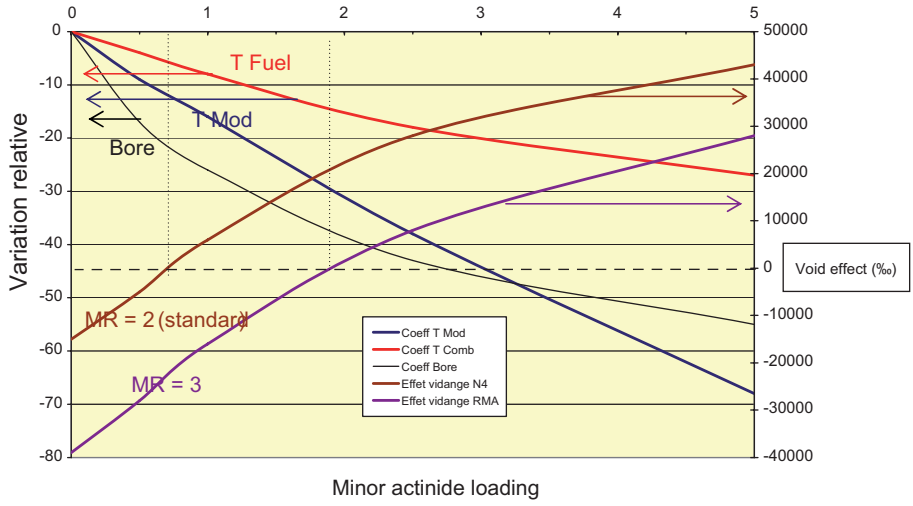

FIG. 4. The effect of MR and MA loadings on PWR core reactivity coefficients.

3. CONCLUSIONS

Recycling of MAs for transmutation purposes is one of the strategies being considered at present for future sustainable fuel cycles. The addition of MAs to the fuel hardens the neutron spectrum and leads to degraded reactivity coefficients and kinetics parameters. Core calculations on fast and thermal reactor cores of various sizes have been performed to identify the maximum core fraction of MAs allowed by safety constraints. For sodium cooled fast reactors, these indicate a limit of about 2.5% if no specific sodium void reduction measures are adopted. Such measures include a reduction of core height, axial heterogeneity, introduction of a moderator into the SAs and use of neutron absorbing blankets. The combination of the last two modifications was identified as the most promising option, allowing an increase of the limiting MA core fraction of up to 5%.

For small cores, the maximum loadings that could be attempted without specific measures are up to 5%. For thermal reactors, typical values are around 1% but could be increased up to 4% when using cores with increased moderation ratios.

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THE THORIUM FUEL CYCLE

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Abstract

There has been renewed interest in thorium fuels and fuel cycles because of their intrinsic proliferation resistance, the favourable properties of thorium oxide as a matrix for high burnup fuels, the possibility of self-sustaining ^{232}Th–^{233}U fuel cycles in several thermal reactor systems and the attractive features of thorium related to accelerator driven systems. The IAEA has published several technical reports on thorium fuels and the thorium fuel cycle during the last five years covering front and back end issues, proliferation resistance and economics. The paper summarizes the IAEA's programmes with reference to thorium utilization in the once-through mode and as closed fuel cycle concepts worldwide.

1. INTRODUCTION

Thorium is three times more abundant in nature than uranium and occurs mainly as the fertile ^{232}Th isotope. Various approaches to thorium based fuel cycles were explored worldwide from the mid-1950s to the mid-1970s, particularly in Germany and the United States of America (USA). Several experimental and power reactors utilized thorium based driver fuels: high temperature gas cooled reactors (HTGRs), light water reactors (LWRs) and molten salt breeder reactors (MSBRs). The initial enthusiasm was not sustained because of the ready availability of uranium and the slowing down of nuclear power programmes in developed countries. In recent times, there has been renewed and additional interest in thorium fuels and fuel cycles worldwide for the following reasons:

(a) Their intrinsic proliferation resistance;
(b) The favourable properties of ThO_2 as a matrix for high burnup fuels for once-through fuel cycles utilizing ^{239}Pu or ^{235}U fissile materials in commercial LWRs, pressurized heavy water reactors (PHWRs) and fast reactors;

(c) Lower transuranic (TRU) wastes;
(d) The possibility of self-sustaining ^{232}Th–^{233}U fuel cycles in MSBRs, HTGRs and heavy water moderated reactors;
(e) The attractive features of thorium related to accelerator driven systems (ADSs) and energy amplifiers (EAs).

The thorium fuel cycle satisfies the main objectives of two recent and ongoing international programmes, namely, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the Generation IV International Forum (GIF), which address issues related to long term sustainability and favourable economics of nuclear power, inherent and passive safety features of future reactors, safety and environmental protection for interim storage and final disposal of nuclear waste, actinide management, and proliferation resistance in nuclear fuel cycles.

Updated information on thorium and the thorium fuel cycle has been covered in IAEA-TECDOC-1155 [1] and IAEA-TECDOC-1319 [2], published in 2000 and 2002, respectively. In addition, IAEA-TECDOC-1450 on the potential benefits and challenges of the thorium fuel cycle [3] was published in 2005. This TECDOC summarizes the potential of the thorium fuel cycle with the emphasis on forthcoming thorium based reactors, the current information base, front and back end issues, including manufacturing and reprocessing of thorium fuels, proliferation resistance and economics.

2. BENEFITS OF THE THORIUM FUEL CYCLE

The main benefits of the thorium fuel cycle are as follows:

(a) Thorium is three to four times more abundant and widely distributed in nature than uranium, occurs mainly as the mineral monazite (thorium rare earth phosphate) in many countries, in beach or river sands along with heavy minerals (ilmenite, rutile, zircon, sillimanite and garnet), and involves easy and less expensive mining operations. Thorium resources have not been exploited commercially so far and therefore complement uranium reserves and ensure the long term sustainability of nuclear power.
(b) The thorium fuel cycle is an attractive way to produce nuclear energy in the long term with low amounts of radiotoxic waste. In addition, the transition to thorium could be achieved through the incineration of weapons grade plutonium (WPu) or civilian plutonium.
(c) The absorption cross-section for thermal neutrons of ^{232}Th (7.4 b) is nearly three times that of ^{238}U (2.7 b). Hence, a higher conversion (to

^{233}U) is possible with ^{232}Th than with ^{238}U (to ^{239}Pu). Thus, thorium is a better fertile material than ^{238}U in thermal reactors but thorium is inferior to depleted uranium as a fertile material in fast reactors.

(d) For the fissile ^{233}U nuclei, the number of neutrons liberated per neutron absorbed (represented as η) is greater than 2.0 over a wide range of the thermal neutron spectrum, unlike ^{235}U and ^{239}Pu. Thus, unlike the ^{238}U–^{239}Pu cycle in which breeding can be obtained only with fast neutron spectra, the ^{232}Th–^{233}U fuel cycle can operate with fast, epithermal or thermal spectra.

(e) Thorium dioxide is chemically more stable and has a higher radiation resistance than uranium dioxide. The fission product release rate for ThO_2 based fuels is one order of magnitude lower than that of UO_2. Thorium dioxide has favourable thermophysical properties because of the higher thermal conductivity and lower coefficient of thermal expansion compared with UO_2. Thus, ThO_2 based fuels are expected to have a better in-pile performance than that of UO_2 and UO_2 based mixed oxide.

(f) Thorium dioxide has high chemical and radiation stabilities, is relatively inert and does not oxidize, unlike UO_2 (which oxidizes easily to U_3O_8 and UO_3); hence, for ThO_2 based fuel, long term interim storage and permanent disposal in a repository are simpler without the problem of oxidation.

(g) Thorium based fuels and fuel cycles have intrinsic proliferation resistance owing to the formation of ^{232}U via (n, 2n) reactions with ^{232}Th, ^{233}Pa and ^{233}U. The half-life of ^{232}U is only 73.6 years, and the daughter products have very short half-lives and some, such as ^{212}Bi and ^{208}Tl, emit strong gamma radiations. From the same considerations, ^{232}U could be utilized as an attractive carrier of highly enriched uranium (HEU) and WPu to avoid their proliferation for non-peaceful purposes.

(h) For incineration of WPu or civilian plutonium in a once-through cycle, $(Th, Pu)O_2$ fuel is more attractive than $(U, Pu)O_2$, since plutonium is not bred in the former and the ^{232}U formed after the once-through cycle in the spent fuel ensures proliferation resistance.

(i) In the ^{232}Th–^{233}U fuel cycle, a much smaller quantity of plutonium and long lived minor actinides (MAs: Np, Am and Cm) are formed than with the ^{238}U–^{239}Pu fuel cycle, thereby minimizing the radiotoxicity associated with spent fuel. However, in the back end of the ^{232}Th–^{233}U fuel cycle, there are other radionuclides such as ^{231}Pa, ^{229}Th and ^{230}U, which may have a long term radiological impact.

(j) The past performance of ThO_2, $(Th, U)O_2$, ThC_2 and $(Th, U)C_2$ fuels in several HTGRs has been excellent.

3. CHALLENGES OF THORIUM FUEL CYCLE

The melting point of ThO_2 (3350°C) is much higher than that of UO_2 (2800°C). Hence, a much higher sintering temperature (>2000°C) is required to produce high density ThO_2 fuel and ThO_2 based mixed oxide fuel. Admixing of 'sintering aid' (CaO, MgO, Nb_2O_5, etc.) is required to achieve the desired pellet density at a relatively lower sintering temperature (≤1700°C):

(a) Thorium dioxide and ThO_2 based mixed oxide fuels are relatively inert and, unlike UO_2 and (U, Pu)O_2 fuels, do not dissolve easily in concentrated nitric acid. The addition of small quantities of HF in concentrated HNO_3 is essential, but causes corrosion of stainless steel (SS) equipment and piping in reprocessing plants. The corrosion problem is mitigated by the addition of aluminium nitrate. Boiling THOREX solution (13M HNO_3 + 0.05M HF + 0.1M $Al(NO_3)_3$) at ≈393 K and a long dissolution period are required for ThO_2 based fuels.

(b) The irradiated thorium or thorium based fuels contain a significant amount of ^{232}U, which has a half-life of only 73.6 years and is associated with strong gamma emitting daughter products, ^{212}Bi and ^{208}Tl, with very short half-lives. As a result, there is a significant buildup of radiation dose with storage of spent thorium based fuel or separated ^{233}U, necessitating remote and automated reprocessing and refabrication in heavily shielded hot cells and an increase in the cost of fuel cycle activities.

(c) In the conversion chain of ^{232}Th to ^{233}U, ^{233}Pa is formed as an intermediate product, which has a longer half-life (≈27 days) than that of ^{239}Np (2.35 days) in the uranium fuel cycle, thereby requiring a longer cooling time of at least one year for completion of the decay of ^{233}Pa to ^{233}U. Normally, protactinium is passed into the fission product waste in the THOREX process, which could have a long term radiological impact. It is essential to separate protactinium from the spent fuel solution prior to the solvent extraction process for separation of ^{233}U and thorium.

(d) The three stream process for separation of uranium, plutonium and thorium from spent (Th, Pu)O_2 fuel, although viable, has yet to be developed.

(e) The database and experience of thorium fuels and thorium fuel cycles are very limited, as compared with UO_2 and (U, Pu)O_2 fuels, and need to be augmented before large investments are made for commercial utilization of thorium fuels and fuel cycles.

4. REACTORS WITH THORIUM BASED FUELS

Table 1 summarizes the experimental reactors and power reactors where thorium based ceramic nuclear fuels have been used in the form of coated fuel particles (microspheres) in graphite matrices in HTGRs or as zircaloy/SS clad fuel pin assemblies containing high density fuel pellets or vibrocompacted fuel particles or microspheres. In the past, in the two helium cooled pebble bed HTGRs in Germany, namely AVR (15 MW(e)) and THTR (300 MW(e)), coated fuel particles of HEU, thorium, mixed oxide and mixed dicarbide, embedded in graphite matrices and consolidated in the form of spherical fuel elements of diameter ≈60 mm, were successfully utilized. Later, in the wake of international non-proliferation requirements, the HEU was replaced with low enriched uranium (LEU: <20% ^{235}U). Coated fuel particles of mixed uranium thorium oxide and dicarbide, embedded in graphite, were also employed in the form of prismatic blocks in helium cooled HTGRs in the USA, namely Peach Bottom (40 MW(e)) and Fort St. Vrain (330 MW(e)). The HTGR in the United Kingdom (UK), namely the Dragon reactor, has also used coated fuel particles of mixed thorium uranium oxide and dicarbide in a graphite matrix.

Thorium cycles are feasible in all existing thermal and fast reactors, for example, LWRs (including WWERs, especially WWER-T (thorium)), PHWRs, HTGRs, MSBRs and liquid-metal-cooled fast breeder reactors (LMFBRs), and in ADSs. In the short term, it should be possible to incorporate the thorium fuel cycle in some of these existing reactors without major modifications to their engineered systems, reactor controls or reactivity devices. However, for innovative reactors and fuel cycles, extensive reactor physics studies and other technological developments would be required before the thorium fuel cycle could be implemented.

5. RATIONALE FOR THORIUM BASED FUEL CYCLES

Thorium based fuel cycles are under consideration for the following reasons:

(a) The abundance of thorium in nature, as well as thorium mining operations being easy and inexpensive;
(b) The better nuclear characteristics of ^{232}Th and ^{233}U;
(c) The higher chemical and radiation stabilities of ThO_2;
(d) The excellent past performance of ThO_2, $(Th, U)O_2$, ThC_2 and $(Th, U)C_2$ fuels in HTGRs;

TABLE 1. THORIUM UTILIZATION IN DIFFERENT EXPERIMENTAL AND POWER REACTORS

Name and country	Type	Power	Fuel	Operational period
AVR, Germany	HTGR, Experimental (pebble bed reactor)	15 MW(e)	Th + ^{235}U driver fuel, coated fuel particles (oxide and dicarbides)	1967–1988
THTR, Germany	HTGR, Power (pebble type)	300 MW(e)	Th + ^{235}U, driver fuel, coated fuel particles (oxide and dicarbides)	1985–1989
Lingen, Germany	BWR Irradiation testing	60 MW(e)	Test fuel (Th, Pu)O_2 pellets	Terminated in 1973
Dragon, UK OECD–Euratom also Norway, Sweden and Switzerland	HTGR, Experimental (pin-in-block design)	20 MW(th)	Th + ^{235}U driver fuel, coated fuel particles (dicarbides)	1966–1973
Peach Bottom, USA	HTGR, Experimental (prismatic block)	40 MW(e)	Th + ^{235}U driver fuel, coated fuel particles (oxide and dicarbides)	1966–1972
Fort St. Vrain, USA	HTGR, Power (prismatic block)	330 MW(e)	Th + ^{235}U driver fuel, coated fuel particles (dicarbide)	1976–1989
MSRE ORNL, USA	MSBR	7.5 MW(th)	^{233}U molten fluorides	1964–1969
Borax IV and Elk River Reactors, USA	BWRs (pin assemblies)	2.4 MW(e), 24 MW(e)	Th + ^{235}U driver fuel oxide pellets	1963–1968
Shipping Port and Indian Point, USA	LWBR PWR (pin assemblies)	100 MW(e), 285 MW(e)	Th + ^{233}U driver fuel, oxide pellets	1977–1982, 1962–1980
SUSPOP/KSTR KEMA, Netherlands	Aqueous homogeneous suspension (pin assemblies)	1 MW(th)	Th + HEU oxide pellets	1974–1977

TABLE 1. THORIUM UTILIZATION IN DIFFERENT EXPERIMENTAL AND POWER REACTORS (cont.)

Name and country	Type	Power	Fuel	Operational period
NRU and NRX, Canada	MTR (pin assemblies)		Th + ^{235}U test fuel	Irradiation testing of a few fuel elements
KAMINI, CIRUS and DHRUVA, India	MTR Thermal	30 kW(th), 40 MW(th), 100 MW(th)	Al–^{233}U driver fuel, 'J' rod of Th and ThO_2, 'J' rod of ThO_2	All three research reactors in operation
KAPS 1 and 2, KGS 1 and 2, RAPS 2, 3 and 4, India	PHWR (pin assemblies)	220 MW(e)	ThO_2 pellets For neutron flux flattening of initial core after startup	Continuing in all new PHWRs
FBTR, India	LMFBR (pin assemblies)	40 MW(th)	ThO_2 blanket	In operation

(e) The intrinsic proliferation resistance of the closed ^{232}Th–^{233}U fuel cycle and the once-through fuel cycle in both thermal and fast reactors for burning civilian or WPu;
(f) The excellent possibilities of the once-through thorium cycle or the closed ^{232}Th–^{233}U fuel cycle in CANDU PHWRs, ACRs (Canada) and advanced heavy water reactors (AHWRs) (India);
(g) The compatibility of the thorium cycle with ADSs;
(h) The lower generation and handling problems of TRU waste.

6. THORIUM FUEL CYCLE

6.1. The open fuel cycle

The open fuel cycle avoids the engineering processes and other complications associated with reprocessing and refabrication of highly radiotoxic ^{233}U based fuels. An example of thorium utilization in the once-through mode is the Radkowsky concept [4] for LWRs, which is also applicable to the Russian WWER-T reactor concept [5, 6]. The essence of the core layout of such a concept is that each fuel assembly (FA) is made up of a central seed with fissile

material (a medium enriched with uranium and plutonium) and with a thorium blanket. The seed components are more frequently replaced as compared with FAs during refuelling. Separation of seed and blanket, optimization of moderator (water) to fuel ratio and the very long fuel campaign (900 and 2620 effective full power days for seed and blanket, respectively) offer the possibility of such a system up to ≈40% of power to be defined by fission of ^{233}U. Such an open fuel cycle concept for introduction of thorium into a nuclear power reactor is very attractive from the point of view of in situ utilization of ^{233}U and avoiding handling of 'dirty' ^{233}U outside the core.

Another incentive to using thorium in a once-through fuel cycle is the possibility of incineration of WPu in combination with thorium in LWRs of WWER-1000 type to burn ^{239}Pu and not breed it. For this, mixed thorium plutonium oxide, containing ≈5% PuO_2, could be used as the driver fuel. The exclusion of uranium from the composition of the fuel results in an essential increase in the rate of plutonium incineration compared with the use of standard mixed uranium plutonium oxide (MOX) fuel [7, 8]. The spent mixed thorium plutonium oxide on achieving the standard burnup (≈40 MW·d/kgHM (heavy metal)) of LWR fuel is not only degraded in terms of WPu content but also becomes 'proliferation resistant' with the formation of ^{232}U (by the (n, 2n) reaction with ^{232}Th), which has very strong gamma emitting daughter products.

Likewise, the stock of civilian plutonium could be significantly decreased by using the same in combination with thorium in WWER-1000 type reactors. A direct replacement of low enriched uranium oxide fuel is possible with mixed thorium plutonium oxide fuel without any major modifications of core or reactor operation. In such a reactor, there is no need to use a burnable absorber, in the form of gadolinium, integrated into the fuel. The ^{240}Pu isotope, present in significant quantities in civilian grade plutonium, is a good burnable absorber. One of the main advantages of such thorium plutonium mixed oxide fuel in WWERs is the reduction in neutron flux on the reactor vessel. Light water reactors using a mixture of plutonium and thorium oxides have, in fact, better safety characteristics than ones using enriched uranium oxide. Depending on the strategy of nuclear power development, the spent thorium plutonium mixed oxide fuel from WWERs can be disposed of or subjected to long term interim storage until the technology for reprocessing and separating ^{233}U becomes economically attractive.

6.2. The closed fuel cycle

Reprocessing of irradiated thorium based fuels and separation of converted ^{233}U are necessary steps of the closed fuel cycle. In this case, LWRs such as WWER-1000s using mixed thorium plutonium oxide fuel can be

considered as converters for ^{233}U. For recycling the ^{233}U thus formed in LWRs (such as WWER-1000s), an important factor is the ^{232}U content in ^{233}U. For a standard burnup of 40 MW·d/kgHM for a WWER-1000 fuel, the ^{232}U content would be in the range of 3000 ppm. The two recycling options [9] are as follows:

(1) Use of (^{232}Th–^{233}U)O_2 fuel;
(2) Use of (depleted U–^{233}U)O_2 or (reprocessed U from WWER–^{233}U)O_2.

In the first option, there will be a buildup of ^{232}U in ^{233}U in subsequent cycles, whereas with the use of reprocessed uranium, two 'dirty' uranium fuels would be utilized in the same technology. The option with depleted uranium enables a smooth changeover to the thorium fuel cycle with the minimum of modification in reactor design and technology of handling spent fuel. However, the use of depleted/reprocessed uranium in combination with ^{233}U is not part of a pure thorium cycle since ^{235}U is also being used along with ^{233}U and there is a buildup of ^{239}Pu from the conversion of ^{238}U. In addition, recycling of ^{233}U with ^{232}U does not utilize the main advantages of the thorium fuel cycle; for example, using the entire energy potential of thorium, excluding buildups of MAs and plutonium, and minimizing the radiotoxicity of the disposed wastes.

Replacement of ^{235}U by ^{233}U in WWER-1000 reactor fuel results in a shift in the water temperature coefficient of reactivity to the positive region. On the other hand, when ^{235}U is replaced by plutonium, the shift in temperature coefficient of reactivity is in the negative region. Hence, it is possible to judiciously combine plutonium and ^{233}U in the fuel composition such that the safety requirements with respect to the temperature coefficient of reactivity are met. Addition of plutonium compensates for the deficiency of the reduction of ^{233}U. Separate allocations of ^{233}U and plutonium appear to be preferable in comparison with a mixed allocation in terms of improved efficiency in reactor control, lower neutron flux on the reactor vessel and a simpler fresh fuel fabrication and reprocessing of spent fuel. The transition to a tight lattice in WWER-1000s raises the conversion ratio of ^{232}Th–^{233}U fuel but cannot convert these reactors into thermal breeder reactors such as the Shipping Port light water breeder reactor (LWBR) [10].

Calculations performed by Russian experts demonstrate the possibility of achieving self-sufficiency in the ^{232}Th–^{233}U fuel cycle with a breeding ratio ≥1.0 in a BN-800 type sodium cooled LMFBR [11]. Similar results have also been reported from France. In other types of reactors also, namely HTGRs or heavy water reactors, calculations show the possibility of a breeding ratio approaching 1.0 but not exceeding it.

In India, the vast thorium reserves there are being judiciously utilized by pursuing a three stage indigenous nuclear power programme, as shown in

Fig. 1, linking the 'closed' fuel cycles of PHWRs, LMFBRs and self-sustaining ^{232}Th–^{233}U based advanced thermal reactors [12]. In all three stages, ThO_2 is being introduced. In stage I, namely that with PHWRs, zircaloy clad ThO_2 pin assemblies are being used for neutron flux flattening of the initial core. The ^{233}U obtained by reprocessing the spent ThO_2 blankets from PHWRs is found to contain ≈500 ppm ^{232}U. In stage II, namely that with LMFBRs, SS 316 clad ThO_2 blanket assemblies are in operation in the Fast Breeder Test Reactor (FBTR).

7. FEASIBILITY STUDIES OF THORIUM UTILIZATION IN LIGHT WATER REACTORS

Several thorium based fuel design options investigated in recent years [13–19] have demonstrated the basic feasibility of thorium based fuel cycles for LWRs of current and next generation technologies. Activities have focused on examining the Th–^{233}U cycle as a replacement for conventional uranium based fuels in existing LWRs, as well as a way to manage the growth of plutonium stockpiles by burning plutonium, or achieving a 'net zero production' sustainable recycle scenario. The fuel has to be designed to withstand very high burnup (above 100 000 MW·d/kg). The fuel cycle costs are similar to those of conventional fuel. Two main implementation scenarios have been the focus of recent studies for pressurized water reactors (PWRs): homogeneous and heterogeneous designs. Homogeneous designs employ a mixture of ThO_2 and UO_2, within each fuel rod, with a uranium volume fraction and enrichment sufficient to obtain the required burnup and cycle length. Heterogeneous designs use a seed blanket approach, where uranium and thorium fuel parts are spatially separated either within a given assembly or between assemblies. Homogeneous studies have also considered 'microheterogeneous' schemes in which the uranium and thorium fuels are spatially separated within a given fuel rod.

The use of thorium based fuels in combination with plutonium has two advantages. Firstly, the production of plutonium and higher actinides is reduced, thus controlling the growth of plutonium. Secondly, the existing stockpiles of plutonium from spent nuclear fuels and dismantled nuclear weapons could be disposed of by burning them. The viability of thorium based fuels in PWRs for burning plutonium and TRU elements is being investigated in detail [20–23] in Europe and the USA. Destruction rates and residual amounts of plutonium and MAs in the fuel used for transmutation were examined. In general, the thorium based concepts consume approximately twice as much plutonium as conventional (U–Pu) mixed oxide fuel.

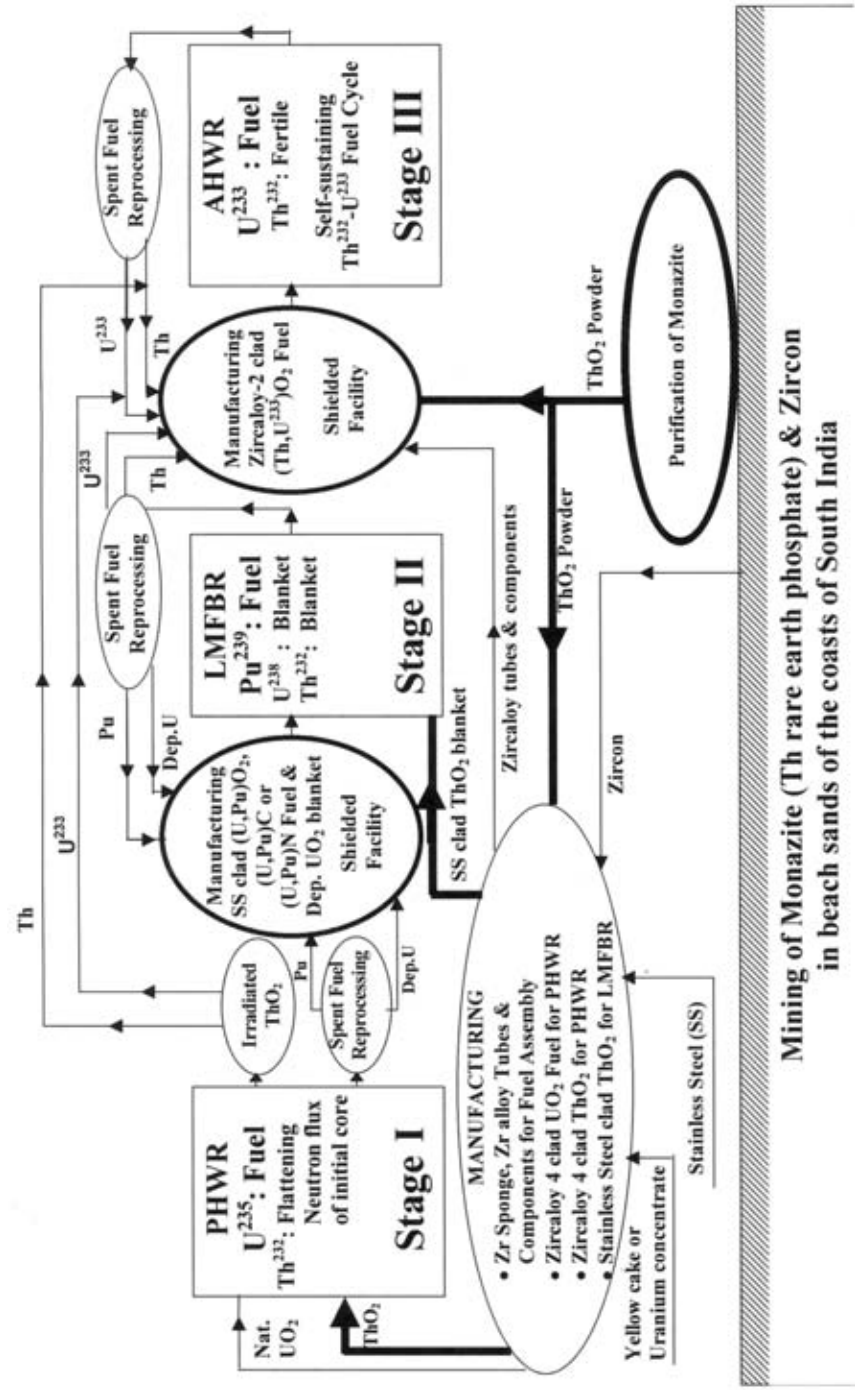

FIG. 1. Thorium utilization in PHWR (stage I), LMFBR (stage II) and AHWR (stage III) closed fuel cycles in India.

Destruction of up to 1000 kg of reactor grade plutonium can potentially be achieved by burning in thorium based fuel assemblies per gigawatt-year. The addition of MAs to the fuel mixture degrades the efficiency of burning. In general, evaluation of reactivity coefficients demonstrated the feasibility of designing a Th–Pu or Th–Pu–MA fuelled core with negative Doppler and moderator temperature coefficients. Introduction of TRU containing fuels into a PWR core inevitably leads to lower control material worths and smaller delayed neutron yields in comparison with conventional UO_2 cores. Therefore, a major challenge associated with the introduction of Th–TRU fuels into PWRs will be the design of the whole core and reactor control system to ensure safe reactor operation.

Research on the utilization of thorium based fuels in the intermediate neutron spectrum of a tight pitch BWR lattice has been performed at Purdue University and Brookhaven National Laboratory as part of a United States Department of Energy (USDOE) Nuclear Energy Research Initiative (NERI) project. The focus of the work at Purdue University was on the performance of thorium in tighter pitch BWR fuel lattices and on whether thorium based fuels possess advantages in the intermediate neutron spectra [24–29]. The results of these studies showed that thorium based fuels do have several attractive characteristics in tight pitch lattice designs, such as a more negative void coefficient, a higher fuel conversion ratio, improved non-proliferation characteristics and a reduced production of long lived radiotoxic wastes than the corresponding uranium based fuels. Most high conversion LWR concepts fuelled with plutonium in a tight pitch lattice have struggled with ensuring a negative void coefficient of reactivity and have had to introduce some mechanical measures to augment neutron leakage effects, such as void tubes within the fuel assemblies. One of the motivations for the work carried out at Purdue University was to investigate whether thorium fuels in an intermediate spectrum possessed inherent neutronics properties that would ensure negative void reactivity and thereby obviate the need for any mechanical measures to ensure safe reactor operation.

8. FEASIBILITY STUDIES OF THORIUM UTILIZATION IN HEAVY WATER REACTORS

The majority of heavy water power reactors in operation at present are of a pressure tube design, employing simple small fuel bundles and allowing on-power fuelling. The fuel bundle design greatly facilitates the production of exotic, potentially radioactive, fuels. On-power fuelling permits careful reactivity management without the need for excessive neutron absorption by

control devices or neutron poisons. The ability to fuel individual fuel channels also offers the possibility of independently adjusting the residence time of different fuel types in the same reactor core. All of these features could be of great benefit in the implementation of thorium fuel cycles.

Advanced heavy water designs are being pursued in Canada and India that use pressurized, boiling or supercritical light water as the coolant. These designs offer substantially reduced capital costs and allow improvements in other reactor operating characteristics. The thorium fuel cycle options in HWRs are as follows:

(a) The once-through thorium (OTT) cycles, where the rationale for the use of thorium does not rely on recycling the ^{233}U (but where recycling remains a future option);
(b) Direct self-recycling of irradiated thoria elements following the OTT cycle (no reprocessing);
(c) The self-sufficient equilibrium thorium (SSET) cycle, a subset of the recycling options, in which there is as much ^{233}U in the spent fuel as is required in fresh fuel;
(d) High burnup open cycles.

8.1. The OTT fuel cycle and mixed bundle option

The OTT cycle produces a mine of valuable ^{233}U in the spent fuel, at little or no extra cost, available for future recovery, depending on economic or resource considerations.

High neutron economy, on-power fuelling, channel design and simplicity of the fuel bundle provide a great deal of flexibility in approaches to the OTT cycle. In the original OTT concept, it was termed the 'mixed channel' approach, whereby channels would be fuelled either with ThO_2 bundles or with 'driver' fuel, typically slightly enriched uranium (SEU) [30]. The driver fuel would provide the neutrons required to convert ^{232}Th to ^{233}U in the thoria fuel. In such a system, the thoria would remain in the core much longer than the driver fuel.

At low burnups, the thorium represents a load on the uranium, and therefore the presence of thorium causes a reduction in the energy obtained from uranium. With increasing thorium burnup, the ^{233}U, which builds in, produces power, and the sum total of the energy extracted from the SEU and the thorium can become larger than that achievable with SEU alone. At still higher burnups, the accumulated fission product poisons cause the energy extracted to decrease once again. The total energy extracted will be the sum of the energy obtained from the thorium and the SEU. As the residence time of the thorium in the core increases, the energy obtained from a unit of mined

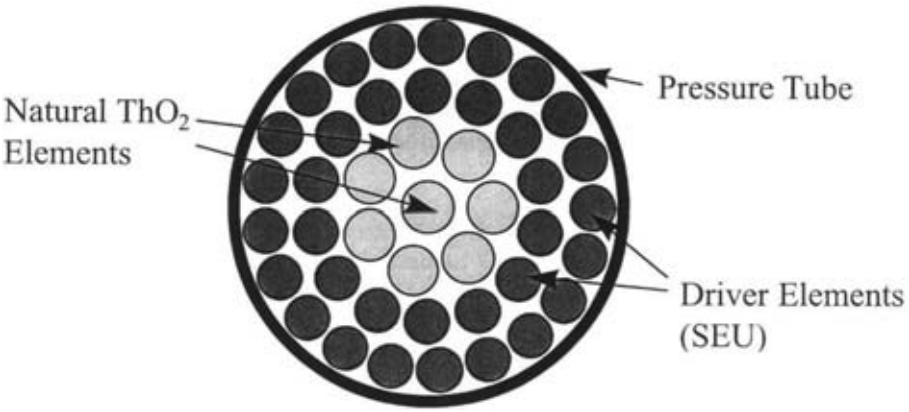

FIG. 2. A CANFLEX mixed bundle.

uranium will first decrease, then, after passing through a minimum, will start to increase, finally becoming higher than it would have been had no thorium at all been present.

In the optimal mixed channel approach to the OTT cycle, a combination of feed rates, burnups, uranium enrichment and neutron flux level would be chosen in order that the cycle be economic (in terms of either resource utilization or monetary cost) compared with either natural uranium or SEU, without taking any credit for the ^{233}U produced. Simple 'scoping' studies (using a lattice code) have shown that such OTT cycles do indeed exist, although their implementation would pose technical challenges to fuel management because of the disparity in reactivity and power output between driver channels and thorium channels [30]. Other driver fuels could also be considered, such as DUPIC fuel from recycled PWR fuel or MOX fuel [31].

An alternative approach has been developed in which the whole core would be fuelled with mixed fuel bundles, which contain both thorium and SEU fuel elements in the same bundle. The CANFLEX mixed bundle shown in Fig. 2 contains ThO_2 in the central eight elements and SEU in the two outer rings of elements. This mixed bundle approach is a practical means of utilizing thorium in existing HWRs, while keeping the fuel and the reactor operating within the current safety and operating envelopes established for the natural uranium fuel cycle. Compared with natural uranium fuel, this option has better uranium utilization, comparable fuel cycle costs are not as low as for SEU, or for an optimized OTT cycle using the mixed channel approach. This mixed bundle option is a practical means of utilizing thorium in operating HWRs, within the current safety and operating envelopes, and does not involve making any significant hardware changes.

Atomic Energy of Canada Limited (AECL) has examined two mixed bundle strategies for burning thoria in six existing CANDU reactors [32]. In option 1, only one type of fuel was used throughout the entire core, and the adjuster rods were removed. The reference fuel design for this study was a CANFLEX fuel bundle with 1.8% SEU in the outer 35 elements and natural ThO_2 fuel in the inner eight elements. The initial fissile content was chosen to give UO_2 burnups that would be readily achievable without requiring significant development.

The second option illustrates the flexibility of existing CANDU reactors in accommodating both thorium fuel and adjuster rods. In option 2, the reactor core is divided into three regions, each containing a different type of thoria fuel bundle. The fuel in the 196 outer region channels is the same as that used in option 1. The fuel in the 124 inner region channels is identical to that in the outer region channels, except that the central ThO_2 element contains 6.0 wt% of gadolinium to shape the axial flux distribution. The gadolinium doped bundles are only used in the inner, adjuster rod, region of the core. The 60 outermost channels contain thorium bundles designed to achieve burnups of over 50 MW·d/kgHE. These high burnup thorium bundles use natural ThO_2 in all 43 fuel elements. However, the initial fissile content in the outer 35 elements is increased from 0 to 1.7 wt% using 20 wt% enriched uranium. These high burn-up thorium bundles are located strategically at the edge of the core in order to utilize a large percentage of the leakage neutrons to produce power. This arrangement significantly increases the amount of thorium fuel in the core and improves the overall fuel efficiency of thorium burning reactors.

8.2. Direct self-recycling

Additional energy can be derived from thorium by recycling the irradiated thoria fuel elements (which contain ^{233}U) directly, without any processing, into the centre of a new mixed bundle [33]. Recycle of the central eight thoria elements results in an additional burnup of ≈20 MW·d/kgHE from the thoria elements, for each recycle. The reactivity of these thoria elements remains remarkably constant during irradiation for each recycle. This direct self-recycling results in a significant improvement in uranium utilization compared with OTT: after the first recycle, the uranium requirements are ≈35% lower than those of the natural uranium cycle, and more than 10% lower than those of the optimal SEU cycle, and remain fairly constant with further recycling. The cumulative uranium requirement averaged over a number of cycles is 30–40% lower than that of a natural uranium fuelled CANDU reactor.

8.3. Self-sufficient equilibrium thorium cycle

The ultimate uranium conserving fuel cycle would be the self-sufficient equilibrium thorium cycle, in which no fissile topping (and hence, no natural uranium) would be required in equilibrium, i.e. the ^{233}U concentration in the recycled fresh fuel matches the ^{233}U concentration in the spent fuel [34]. Further improvements in neutron economy would be required to achieve this: reducing the fuel rating to lower the flux and hence neutron capture in ^{233}Pa, increasing the moderator purity, removing the adjuster rods from the core, enriching the zirconium used in the pressure and calandria tubes to remove most of the high cross-section isotope, ^{91}Zr. However, the following studies do not give credit for such improvement.

The major shortcoming of the self-sufficient equilibrium thorium cycle is its low burnup, between 10 and 15 MW·d/kgHE, which will not be economic in a cycle that requires reprocessing and remote fabrication of the ^{233}U bearing fuel. To address this issue, a small amount of ^{235}U make-up could be added to each cycle, allowing the burnup to be increased as desired.

8.4. High burnup open cycle

The high burnup thorium open cycle avoids the issues related to closing the fuel cycle by reprocessing. In this cycle, the burnup is increased by making a trade-off with the conversion ratio. The thorium is enriched with ^{235}U to give whatever burnup the fuel can achieve. The spent fuel is not recycled (although this option would not be precluded). A high burnup is equally possible with SEU, but the advantage of thorium over SEU lies in the fact that for very high discharge burnups, the initial fissile content required is lower with thorium fuel. In the case of low enrichment, SEU gives a higher discharge burnup for a given ^{235}U enrichment, but with very high discharge burnups, the enrichment required for the thorium fuel is lower than that required for SEU. In theoretical assessments, pure ^{235}U has been added to the thoria.

The main advantage of this thorium cycle compared with an equivalent enriched uranium cycle stems from the fact that as ^{235}U is burnt, so ^{233}U is built up, and as ^{233}U is a superior fissile material than ^{235}U, the reactivity versus burnup curve falls off more gradually for thorium than it does for enriched uranium. This means that to attain the same discharge burnup, the initial ^{235}U content can be lower in the thorium cycle. To achieve a discharge burnup of around 66 MW·d/kgHM, SEU requires an enrichment of 4.5% in a CANDU reactor, whereas thorium needs only 3.5% (in total HM). Added to this is the fact that thermal neutron absorption in thorium is about three times that in ^{238}U, and that consequently the initial reactivity in the thorium core will be well

below that in the SEU core for the same discharge burnup. This leads to lower swings in reactivity, which is a definite operational advantage. This cycle is also an attractive method for plutonium annihilation, as it would have a very high efficiency for destruction of plutonium.

9. ADVANCED HEAVY WATER REACTOR DESIGNS

An AHWR of power 920 MW(th)/300 MW(e) has been designed as a forerunner of thorium based reactors in India to maximize the energy potential of the vast thorium resources there (≈518 000 t, in terms of thorium metal). Table 2 summarizes the major design parameters of the AHWR 300 design. Some of the salient features of this reactor are:

(a) Use of ThO_2 based driver fuel — zircaloy-2 clad (Th, Pu)O_2 and (Th, ^{233}U)O_2 fuel pin clusters;
(b) Heavy water moderator as heat sink;
(c) Boiling light water coolant;
(d) Vertical pressure tube;
(e) Heat removal through natural circulation;
(f) On-power fuelling.

The design logic and reactor physics objectives are now described. The AHWR core has 500 lattice locations of which 452 are for fuel channels, 36 for shut-off rods and 12 for control rods. The fuel assembly in each channel has a length of 10.5 m and is suspended from the top. The assembly consists of a single long fuel cluster of length 4.3 m and two shield subassemblies. These subassemblies are held to each other through a quick connecting/disconnecting joint to facilitate handling.

An AHWR fuel cluster, shown in Fig. 3, consists of 54 zircaloy-2 clad fuel pins of outer diameter 11.2 mm, wall thickness 0.6 mm and pellet stack length 3500 mm. The fuel pins in a cluster are arranged in three concentric rings having 12, 18 and 24 pins in the inner, middle and outer rings, respectively. The 24 fuel pins in the outer ring contain high density (Th, Pu)O_2 pellets with 3.25% plutonium. The fuel pins in the middle and inner rings contain high density (Th, ^{233}U)O_2 fuel pellets having 3.75 and 3% ^{233}U concentrations, respectively. The outer diameter of the fuel cluster is 118 mm. The fuel pins are assembled in a cluster by top and bottom tie-plates, with the central rod connecting the two tie-plates. There are six spacers along the length of the cluster. The hollow central rod contains ZrO_2–Dy_2O_3 and also functions as a water injunction tube for the emergency core cooling system (ECCS), as spacer capture rod and

TABLE 2. MAJOR DESIGN PARAMETERS OF ADVANCED HEAVY WATER REACTORS IN INDIA

Parameter/component	Value/number
Reactor power	920 MW(th)/300 MW(e)
Core configuration	Vertical pressure tube, 500 lattice locations (452 fuel channels + 36 shut-off rods + 12 control rods).
Number of fuel clusters in the core	452
Dimensions of fuel cluster	118 mm dia. × 4.3 m length, six spacers
Number of fuel pins in each cluster, their configuration and other components	54 fuel pins in three concentric rings Outer: 24 pins (Th–Pu)O_2 (3.25% Pu) Middle: 18 pins (Th–^{233}U)O_2 (3.75% ^{233}U) Inner: 12 pins (Th–^{233}U)O_2 (3% ^{233}U) Hollow displacer rod: ZrO_2–Dy_2O_3 Water tube: zircaloy-2, 36 mm OD[a] × 2 mm thickness
Fissile material per cluster	^{233}U: 2.3 kg; Pu: 1.75 kg
Active fuel length (pellet stack)	3500 mm
Cladding material	Zircaloy-2
Cladding dimensions	11.2 mm OD, 0.6 mm thick
Top and bottom tie plates	SS
Annual fuelling rate	102 fuel clusters
Average heat rating of fuel	10.56 kW/m
Fuel burnup	24 000 MW·d/t
Moderator	Heavy water
Reflector	Heavy water
Coolant	Boiling light water under natural circulation
Total core flow rate	2306 kg/s
Core inlet temperature	261.4°C (nominal)
Feedwater inlet temperature	130°C
Average steam quality	17.6%

TABLE 2. MAJOR DESIGN PARAMETERS OF ADVANCED HEAVY WATER REACTORS IN INDIA (cont.)

Parameter/component	Value/number
Steam produced	405 kg/s
Steam pressure/temperature	70 bar/285°C
Main heat transfer loop height	39 m
Lattice pitch	270 mm: square pitch
Pressure tube ID[b]	120 mm
Primary shutdown system	36 shut-off rods having B_4C
Secondary shutdown system	Lithium pentaborate solution injected into 32 poison tubes

[a] OD: outer diameter.
[b] ID: inner diameter.

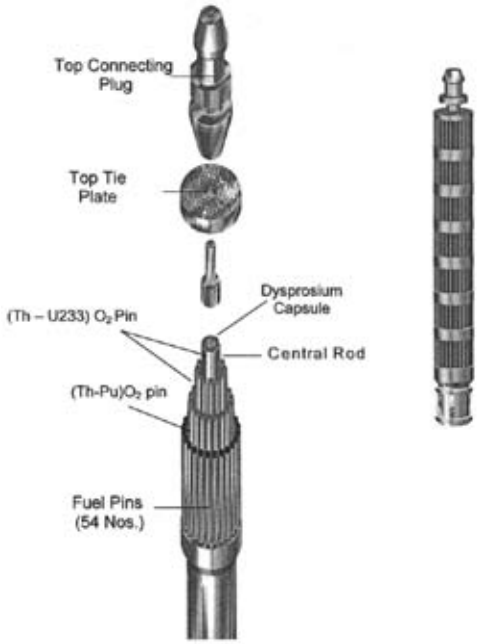

FIG. 3. An AHWR fuel cluster.

as tie-rod for the cluster. The dysprosium helps in achieving a negative void coefficient of reactivity.

10. THE ADVANCED CANDU REACTOR

Atomic Energy of Canada Limited has developed a reactor, the ACR™ (Advanced CANDU Reactor) that is an evolution from the well established CANDU 6 reactor (with over 90 reactor-years of operating experience plus current construction experience). The ACR is designed to be an economical reactor choice for current conditions with enhanced safety and reliability, while meeting expectations for sustainability. The potential for economical use of MOX fuel provides an economic means of ensuring adequate fuel supplies through recycling, thus enhancing sustainability. In the future, the ACR will be able to operate using a thorium fuel cycle.

The ACR reference design provides a design life of 40 years with an option for extension to 60 years. The design will be completed in time to have an operation start date of 2011. The ACR is a heavy water moderated, light water cooled reactor. Evolutionary changes from CANDU 6 allow for a compact reactor core design, with the core of the 700 MW class having 284 channels, with 12 CANFLEX fuel bundles per channel. A once-through fuel cycle is assumed in the reference design but the potential exists for alternative MOX, DUPIC or thorium fuel cycle options. Using light water as coolant requires some enrichment of the fuel, thus a fuel enrichment of 2% ^{235}U with 4.6% dysprosium in natural uranium in the central fuel element in the bundle is the proposed SEU reference fuel for the once-through fuel cycle. This reactor is designed for operation with low enriched fuels such as SEU at approximately 2% enrichment, for 21 MW·d/kgHM burnup, or at up to about 4% for future operation at up to 45 MW·d/kgHM.

In particular, ACR, as designed for the reference SEU fuel cycle, can operate using MOX fuel with no design or operational changes. This is an important benefit, since a programme of power operation can start with ACR using SEU fuel, and then, at the operator's choice of timing and degree, can move to MOX fuel with a seamless transition involving no unit downtime or power derating, or added capital cost.

As a thorium burner, the small simple fuel bundle, fuel channel design, on-power fuelling and good neutron economy all facilitate a variety of possible fuel cycles. The lattice design provides an inherently negative coolant void coefficient for many thorium bundle designs without the need to add neutron poisons to the middle of the bundle.

Looking further into the future, AECL is pursuing designs for a super-critical LWR. In these designs, the thermal efficiency of the reactor is increased by running the coolant at much higher temperatures and pressures. The reactor coolant is operated beyond the supercritical point of light water.

This design is the next evolutionary step beyond the ACR. Increasing the temperature and pressure of the coolant builds on the success of the ACR design. Preliminary physics calculations show that the reactor should perform well with thorium fuel cycles designed to meet a variety of user requirements.

11. FEASIBILITY OF THORIUM UTILIZATION IN HIGH TEMPERATURE GAS COOLED REACTORS

In the past, thorium based fuels have been successfully utilized in helium cooled HTGRs in Germany, Japan, the Russian Federation and the USA [35]. The fuels were in the form of coated particles of ThO_2, (Th, U)O_2, ThC_2 and (Th, U)C_2, popularly known as TRISO, with a fuel kernel of diameter 350–500 μm with multilayer carbon and silicon carbide coatings (a ≈100 μm buffer carbon layer on the fuel kernel followed by inner and outer pyrolitic carbon coatings of ≈40 μm with a 35 μm SiC layer in-between). In Germany, two pebble bed HTGRs, namely AVR (15 MW(e)) and THTR (300 MW(e)), successfully operated until the late 1980s after which they were terminated. In pebble bed reactors, coated fuel particles are embedded in a graphite matrix and shaped into spherical fuel elements of diameter ≈60 mm. Coated fuel particles of mixed uranium thorium oxide and dicarbide, embedded in graphite, were also employed in the form of prismatic blocks in the helium cooled HTGRs of the USA, namely Peach Bottom (40 MW(e)) and Fort St. Vrain (330 MW(e)). The HTGR in the UK, namely the Dragon reactor, has also used coated fuel particles of mixed thorium uranium oxide and dicarbide in a graphite matrix.

The US led GIF has identified very high temperature reactors (VHTRs) with a helium coolant outlet temperature of 1000°C as one of the candidate nuclear energy systems deployable by the year 2025. For this, the reference reactor concept has been a gas turbine modular helium reactor (GT-MHR) of 600 MW(th), helium cooled and with a prismatic block fuel, or the pebble bed modular reactor (PBMR) with a pebble fuel. Thorium based, ZrC coated fuel particles (termed 'TRISO') of oxide, mixed oxide, dicarbide or mixed dicarbides in graphite matrices are strong candidate fuels for these types of reactor.

High temperature gas cooled reactors have considerable adaptability for different fuel cycles without a change of active core design and main plant components and offer attractive opportunities for utilization of thorium in combination with enriched uranium and plutonium [36]. Studies of fuel loads on the basis of thorium with weapon quality ^{235}U and ^{233}U–Th fuel, as well as experience of operation of the Fort St. Vrain reactor [37], which is the

GT-MHR prototype, showed the high effectiveness of these fuel compositions from the point of view of minimization of consumption of fissile isotopes. Thus, the operational conditions (ratio of fuel reloading, time between fuel reloading and limitations on available operative reactivity margin) met the design requirements.

To use the HTGR neutron spectrum effectively in the ^{233}U–Th fuel 'open' cycle, a high metal component in a fuel compact is preferable, corresponding to a moderator (carbon) to high metal (N_C/N_{TM}) ratio of ≥200. In this case, compared with other fissile isotopes, the mean load of uranium in an active core, as well as the consumption of ^{233}U per unit of energy generated, will be least. The changeover to a self-sustained fuel cycle can further lower consumption of ^{233}U. The minimum consumption of uranium in a closed fuel cycle amounts to ≈0.27 g/(MW·d), and this corresponds to the maximum possible load of high metal component in a fuel compact (6.1 g/compact). In comparison, the consumption of uranium in the case of ^{235}U–Th fuel is ≈1 g/(MW·d). For various reasons (e.g. technological limitations and economic factors), the optimum possible load of high metal component is 3–4 g/compact at a minimum enrichment of ≈4.5%. In the case that thorium is injected into the reflectors, the uranium consumption in a self-sustained fuel cycle can be reduced to ≈30%; however, for economic reasons, this alternative would hardly be expedient. A self-sustaining ^{233}U reactor has not yet been achieved.

12. FEASIBILITY OF THORIUM UTILIZATION IN FAST REACTORS

In the fast neutron spectrum, ^{232}Th is less fissile than ^{238}U and has a higher fission threshold energy. In addition, the η value increases much less with energy for ^{233}U than for ^{239}Pu.

Extensive investigations have been carried out in France on the closed thorium fuel cycle in fast neutron reactors, and the following sequential approach has been proposed [38]:

(a) Design (Th, Pu)O_2 cores to burn plutonium and breed ^{233}U to initiate the Th–U cycle in LMFBRs;
(b) Design a (Th, ^{233}U) self-sustaining core with a breeding ratio slightly higher than 1.0 for multiple recycling of recovered thorium and uranium with or without other actinides.

Thorium utilization has been investigated with the reference European Fast Reactor (EFR: 3600 MW(th), 1450 MW(e)) and CAPRA cores. In an EFR-like (Th, Pu)O_2 core (100 cm height × 405 cm diameter), the percentage

volume fractions of PuO_2 in $ThO_2 + PuO_2$ were assumed to be 20.45, 24.95 and 29.98% in the three enrichment zones of the reactor, and the isotopic compositions of plutonium were assumed to be 52.54% ^{239}Pu, 25.49% ^{240}Pu, 9.8% ^{241}Pu, 7.84% ^{242}Pu and 1.96% ^{238}Pu, and the plutonium contained 2% ^{241}Am. An EFR-like core contains axial (lower: 25 cm; upper: 15 cm) and radial (one row of 78 assemblies) fertile ^{232}Th blankets. In a CAPRA-like core, the two enrichment zones had a PuO_2 vol.% in (Th, Pu)O_2 in the range from 43.2 to 45%, with no axial or radial fertile blankets. The isotopic composition of plutonium for the CAPRA-like core was 38.95% ^{239}Pu, 26.71% ^{240}Pu, 13.06% ^{241}Pu, 14.42% ^{242}Pu and 5.5% ^{238}Pu, with an ^{241}Am content of 1.31% in plutonium. The following conclusions were drawn from these studies:

(a) There is a decrease of ≈35% in the sodium void reactivity of (Th, Pu)O_2 compared with the reference (U, Pu)O_2 core of EFR and even more (≈65%) in a (Th, U) core. The Doppler constants are similar in (Th, Pu)O_2 cores and ≈50% greater in Th–U cores than in standard U–Pu cores.

(b) Large plutonium consumptions in (Th–Pu)O_2 fuel, both EFR-like and CAPRA-like, compared with the reference (U–Pu)O_2 fuel for EFRs. The plutonium consumptions were higher in CAPRA-like fuel than in EFR-like fuel, being 880 and 660 kg/(GW(e)·a), respectively.

(c) In EFR-like (Th–Pu)O_2 fuel with a ThO_2 fertile blanket, enough ^{233}U is produced to feed a similar Th–U reactor after ≈15 years of operation. Thus, with plutonium burning, it is possible to initiate a ^{232}Th–^{233}U fuel cycle sooner. A closed self-sustaining ^{232}Th–^{233}U cycle is possible with indefinite recycling, but with a very long linear doubling time of nearly 300 calendar years without taking the operating, ageing and cooling time of the fuel into consideration. Hence, the ^{232}Th–^{233}U fuel cycle is not so attractive for LMFBRs if rapid growth of a nuclear power programme is to be achieved.

(d) The ^{233}U and ^{232}U contents in EFR-like cores after 1700 effective full power day (EFPD) operation + 5 years were 92.93 and 0.23%, respectively. The ^{233}U contents in the blanket regions were much higher at 95, 98 and 96% in the lower axial, upper axial and radial blankets, respectively. The ^{233}U contents in the ThO_2 blankets of EFR-like (^{238}Th, ^{233}U)O_2 cores were higher than with (Th, Pu)O_2 cores, at 97.9, 99 and >98% in the lower axial, upper axial and radial blankets, respectively. These values were 90.28 and 0.22%, respectively, for the CAPRA-like core after 990 EFPDs + 5 years.

On the basis of recent experimental irradiation of thorium blankets in BN-350, it has been reported from the Russian Federation that the ^{232}U content in bred ^{233}U could be brought down to extremely low levels (≤11 ppm) by locating thorium blankets at a distance of 15–20 cm away from the border of the core [39].

Thorium based metallic fuel is of great interest for commercial LMFBRs with excellent safety features. The compositions under consideration are: Th–20%U, Th–20%Pu, Th–20%Pu–4%U and Th–20%Pu–4%U–8%Zr. The metallic fuel pins have been manufactured by Argonne National Laboratory (ANL, USA), as part of the Experimental Breeder (EBR-II) fuel development programme, by induction melting of Th, U, Pu and Zr metal buttons in graphite crucibles, followed by vacuum injection casting in high purity silica tubes. However, reprocessing of thorium based metallic fuel has not been attempted so far. Thermochemical modelling of the reprocessing of Th–U–Pu–Zr metallic fuel by a electrorefining process has indicated that thorium will remain in the anode compartment forming metallic waste together with zirconium and noble metals, while uranium–plutonium and MAs together with some fraction of rare earth elements (REEs) can be separated and co-deposited in the liquid cadmium cathode [40].

As part of the inert matrix fuel development programme for incineration of plutonium, Japan has developed a rock-like fuel consisting of a polyphase mixture of $(Th, Pu)O_2 + MgAl_2O_4 + AlO_3$, which was prepared and irradiated in the JPR3 reactor for a burnup of 21–28% [41]. In addition, a thorium based nitride fuel is also being developed in Japan for lead cooled fast breeder reactors for enhancement of safety, economics and breeding potential [42].

13. FEASIBILITY OF THORIUM UTILIZATION IN ACCELERATOR DRIVEN SYSTEMS

In recent years the study has started, with specific objectives and in various countries and at international level, of accelerator driven subcritical systems, popularly known as ADSs. Accelerator driven systems might be utilized for safe and efficient breeding of ^{233}U from abundant thorium resources.

The advantages of the use of thorium in ADS fast critical systems are:

(a) It avoids production of higher actinides.
(b) It limits the reactivity swing over the cycle.
(c) Thorium-232 is easily converted into ^{233}U.
(d) It is relevant if TRU waste is absent.

Fast energy amplifiers are suitable for the following core loadings containing thorium bearing fuels:

(1) Mixed plutonium thorium oxide or mononitride for burning weapons grade or civilian plutonium;
(2) Mixed thorium–^{233}U oxide or nitride for 'clean' energy production;
(3) Mixed high HEU–thorium oxide or nitride for burning HEU.

14. INNOVATIVE THORIUM/^{233}U BASED FUELS

The NERI project of the USDOE has developed an innovative metal matrix dispersion, or cermet, fuel consisting of (Th, U)O_2 microspheres (using LEU: <20% ^{235}U) of diameter ≈50 μm in a zirconium matrix that can achieve high burnup in a once-through cycle and be disposed of, without processing, as nuclear waste. The volume fraction ratio of fuel microspheres and zirconium matrix is 50:50. The use of mixed oxides prohibits direct chemical separation of pure ^{233}U or ^{239}Pu. Blending of the actinide oxides helps to improve the proliferation resistance of this innovative fuel. The high thermal conductivity of the zirconium matrix enhances heat removal and keeps the fuel central temperature significantly lower than that of the 'pellet-pin' design, thereby minimizing fission product migration and fuel swelling.

The metal matrix fuel was manufactured by the novel 'powder-in-tube-drawing' technique, which consists of dry mixing or wet vibratory milling of zirconium powder with (Th, U)O_2 microspheres, loading the powder mixture into the cladding tube and using vibratory packing to obtain a smear density in the range of 40–50% of the theoretical density. The tube containing the cermet powder mixture is then subjected to multiple drawing/heat treatment cycles for progressive densification of the cermet and reduction in the fuel pin diameter. Wet milling facilitates coating the fuel microspheres with zirconium metal, which enhances the fission heat transfer from the fuel through the metal matrix to the zirconium alloy cladding [43].

Japan is pursuing R&D activities on innovative thorium based hydride fuels for advanced MA and plutonium burners with high safety characteristics [44]. The U–Th–Zr–H fuel has a high thermal conductivity and consists of uranium metal, Th–Zr_2H_x and ZrH_x phases. The hydride fuel is manufactured by melting and casting a ternary alloy of U–Th–Zr, keeping the atomic ratios of U:Th:Zr = 1:4:10. Next, the alloy was hydrogenated and clad with SS. The SS clad alloy fuel U–Th–Zr–H was irradiated in the Japan Material Testing Reactor (JMTR) with a thermal neutron fluence of 1.2×10^{20} n/cm^2 at a maximum pellet temperature of 554 K and a linear heat rate of 140 W/cm.

Efforts are under way to develop thorium based hydride fuel containing plutonium as an effective fuel for burning weapons grade plutonium.

15. FRONT END ISSUES AND CHALLENGES OF THORIUM FUEL CYCLES

15.1. Mining and milling

The largest known reserves of thorium are contained in beach and inland placer deposits of monazite, a mixed phosphate mineral with the following chemical formula: [(Rare earth, Th, U)PO_4]. Monazite is a primary source of light REEs and thorium, in addition to being a secondary source of phosphate and uranium.

Thorium concentrate and nuclear grade ThO_2 are produced from monazite by involving the following process steps:

(a) Extraction and pre-concentration of beach sands;
(b) Conversion of ore (beach sand concentrates) to monazite;
(c) Conversion of monazite into thorium concentrate, uranium concentrate and rare earths;
(d) Storage of thorium concentrate in a suitable form or conversion of thorium concentrate to nuclear grade ThO_2 powder.

Mining and extraction of thorium from monazite is relatively easy and significantly different from that of uranium from its ores. Most commercially exploited sources of monazite are from beach or river sands along with heavy minerals. The overburden during mining is much smaller than in the case of uranium, and the total radioactive waste production in mining operations is about two orders of magnitude lower than that of uranium. The so-called radon impact is also much smaller than in the case of uranium due to the short half-life of thoron compared with that of radon, and needs, therefore, much simpler tailings management than in the case of uranium to prevent long term doses to the public. As far as occupational doses are concerned, there is no need to control ventilation with respect to ^{220}Rn inhalation because monazite extraction is carried out in open pits. However, inhalation and ingestion dose factors are high for thorium and thoron.

Monazite deposits are formed by weathering of parent rock, followed by gravity concentration of heavy minerals in sand-beds through the actions of wind and water in the coastal areas of tropical countries. The individual heavy minerals, namely ilmenite, rutile, monazite, zircon, sillimanite and garnet are

separated from each other by methods dependent upon their physical properties, i.e. specific gravity, magnetic susceptibility, electrical conductivity and surface properties. A representative procedure for separating monazite from heavy minerals in beach sands [45] is as follows:

(a) The electrically conductive ilmenite and rutile constituents are first separated using a high tension separator.
(b) Next, the non-conducting monazite, which is heavy and moderately magnetic, is isolated from non-magnetic sillimanite and zircon and from magnetic garnet by the use of high intensity magnetic separators and air or wet tables. The resulting concentrate contains 98% monazite.

The monazite is finely ground and in most countries dissolved in 50–70% sodium hydroxide at ≈1400°C and subjected to a series of chemical operations, including solvent extraction and ion exchange processes to obtain pure thorium nitrate, which is precipitated in the form of thorium oxalate and subjected to controlled calcinations to obtain ThO_2 powder.

15.2. Types of thorium based fuels and fuel elements

There is a great diversity of thorium based nuclear fuels and fuel elements, depending on the type of reactor, as shown in Table 3. Usually, solid fuels in the form of tiny ceramic fuel microspheres (100–1000 μm), ceramic fuel pellets or metallic alloy fuel rods are used.

15.3. Fabrication of thorium based fuels

The selection of the process flowsheet and the mode of fabrication to be employed for either the first cycle or recycled fuel would depend to a great extent on the radiotoxicity, quantity and form of the fuel material. Fuels containing naturally occurring fissile ^{235}U in combination with fertile ^{238}U or ^{232}Th, emitting only alpha particles of relatively low specific activity, can be manufactured by the so-called contact operations, in which the operator has direct contact with the fuel material. However, process operations that involve generation and handling of fine powders of ^{235}U, ^{238}U or ^{232}Th bearing fuels are carried out in ventilated enclosures for minimizing radioactive aerosols. The enclosures need not be hermetically sealed for handling ^{235}U, ^{238}U or ^{232}Th bearing materials, if they are not pyrophoric. Glovebox operations are those requiring hermetic sealing of equipment, and are essential for handling highly radiotoxic plutonium and ^{233}U bearing materials.

TABLE 3. TYPES AND GEOMETRY OF THORIUM BASED FUELS AND FUEL ELEMENTS

Reactor type	Composition	Fuel shape	Fuel elements
High temperature gas cooled reactors	ThO_2, (Th, U)O_2, ThC_2, (Th, U)C_2 (^{235}U or ^{233}U)	Microspheres 200–800 μm coated with multiple layers of buffer and pyrolytic carbon and SiC	Mixed with graphite and pressed into large spheres (≈60 mm) for pebble bed reactors or fuel rods for HTGRs with prismatic fuel elements
LWRs	ThO_2, (Th, U)O_2, (Th, Pu)O_2 (<5% Pu, ^{235}U or ^{233}U)	High density sintered pellets High density microspheres	Zircaloy clad pin clusters encapsulating the pellet stack Zircaloy clad 'vi-pac' pin clusters encapsulating fuel microspheres
Heavy water reactors:			
PHWRs	ThO_2 for neutron flux flattening of initial core		
AHWRs	(Th, U)O_2 (Th, Pu)O_2 (<5% Pu, ^{235}U or ^{233}U)	High density sintered pellets	Zircaloy clad pin clusters encapsulating the pellet stack
Fast reactors	ThO_2 blanket (Th, U)O_2 and (Th, Pu)O_2 (≈25% Pu, ^{235}U or ^{233}U) fuels	High density sintered pellets	Stainless steel clad pin clusters encapsulating the pellet stack
	Thorium metal blanket Th–U–Zr and Th–U–Pu–Zr fuels	Injection cast fuel rods	Stainless steel clad pin clusters encapsulating fuel rods
MSBRs	$^{7}LiF + BeF_2 + ThF_4 + UF_4$	Molten salt liquid form	Circulating molten salt acting as fuel and primary coolant

Often in such facilities, light beta–gamma and neutron shielding and semi-remote operations are necessary. Remote operations are those requiring heavy shielding and a high degree of automation and remotization. During the last four decades, several countries have manufactured thorium based oxide and non-oxide fuels in both particulate (microspheres) and pellet forms by employing contact, hooded, glovebox, semi-remote and remote operations. The reactors that use thorium fuel, the fuel forms and the viable routes for manufacturing these fuels are as follows:

(a) Fuels for LWRs and PHWRs: mixed oxide fuels based on ThO_2 in the form of high density microspheres or pellets of (Th, U)O_2 (mostly highly enriched ^{235}U and to a limited extent ^{233}U) and (Th, Pu)O_2 encapsulated in the following forms of zirconium alloy clad fuel pin assemblies:
   (i) Vibro-compacted fuel pins: high density fuel microspheres of one, two or three size fractions (1000, 100 or 10 μm) are vibropacked in cladding tubes followed by encapsulation. The fuel microspheres have been prepared by either 'dry' or 'wet' chemical routes. In the wet chemical route, the sol-gel process based on ammonia external/internal gelation has been followed for preparation of hydrated gel-microspheres, which, after controlled calcination and sintering, led to high density microspheres suitable for vibrocompaction.
   (ii) Pellet-pins: The pellets have been produced mostly by the classical 'powder pellet' route, involving co-milling of the oxide powders followed by granulation, pelletization and sintering. In a few cases, a sol-gel microsphere pelletization (SGMP) process has been adapted, where sol-gel derived microspheres have been directly compacted and sintered to high density mixed oxide pellets.

(b) Fuel for HTGRs: high density fuel microspheres (≈200–400 μm diameter) of thorium based oxide, mixed oxides, carbide or mixed carbides coated with multiple layers of pyrolitic carbon and silicon carbide (known as BISO and TRISO particles). These fuel particles are embedded in a graphite matrix and used in the form of spherical balls (known as pebbles) or prismatic bars.

15.3.1. Fuels for LWRs and PHWRs

The following techniques have so far been developed for the manufacture of ThO_2 and thoria based mixed oxide fuels for water cooled reactors:

(a) The powder pellet route: for preparation of high density fuel pellets, using ThO_2, UO_2 and PuO_2 powders as starting materials; the fuel pellet stacks

are encapsulated in cladding tubes. ThO_2, UO_2 and PuO_2 are isostructural (FCC, CaF_2 type), completely solid soluble, and have very similar thermodynamic and thermophysical properties. Hence, the manufacturing process of thoria based mixed oxide fuels is very similar to that of the well established processes for fabrication of UO_2 and $(U, Pu)O_2$ fuels.

(b) The 'vibrosol' and SGMP routes: for preparation of fuel microspheres using nitrate solutions of uranium, plutonium and thorium as starting materials and adapting the ammonia external gelation or ammonia internal gelation processes for obtaining hydrated gel microspheres. The microspheres are sintered and vibropacked in cladding tubes, followed by encapsulation. In the SGMP process, dust-free and free-flowing sol-gel derived oxide fuel microspheres are subjected to controlled calcination to obtain relatively soft, dust-free and free-flowing microspheres that can be directly pelletized and sintered.

(c) The impregnation technique: where (i) partially sintered ThO_2 pellets of relatively low density (≤75% theoretical density) or (ii) 'porous' ThO_2 microspheres are vacuum impregnated in uranyl nitrate (uranium as ^{233}U) or plutonium nitrate solution, followed by calcination and sintering to form high density ThO_2 based mixed oxide fuel pellets, which are encapsulated in cladding tubes.

15.3.2. Coated fuel particles for HTGRs

The higher melting points of thorium oxide and dicarbide compared with their uranium counterparts, make thorium-based ceramic coated fuel particles in a graphite matrix an ideal choice of fuel for HTGRs, where the objective is to have a high coolant outlet temperature (750–900°C) and, more particularly, a high fuel surface temperature (900–1100°C) and a compact core. Research and development and manufacturing of coated fuel particles for HTGRs have been under way for more than three decades in several countries. The core of an HTGR essentially consists of tiny, multilayered, coated fuel (TRISO) particles, dispersed in a graphite matrix and shaped into different forms depending on the design. Two major directions for the fuel element designs have emerged, namely:

(1) The German spherical fuel element design pursued in Germany (used in the AVR and THTR reactors), the Russian Federation (VGM reactors) and China (HTR 10);

(2) The block type US design that has been utilized in the USA (Peach Bottom Unit 1 and Fort St. Vrain (FSV)), the UK (Dragon) and Japan (HTTR).

In the German design, kernels of ≈500 μm of fissile and fertile materials, surrounded by layers of carbon buffer (≈95 μm), inner pyrocarbon (40 μm) and silicon carbide (35 μm), and finally outer pyrocarbon (≈40 μm), are homogeneously distributed in a graphite matrix and shaped in the form of fuel element balls of diameter 60 mm with a 5 mm fuel-free zone in the outer shell. In HTGRs, the coated layers confine fission products released from the 'fissile' or 'fertile' kernels. The pyrolytic carbon (PyC) layers essentially retain Kr and Xe, whereas the silicon carbide layer is effective in retaining solid fission products, e.g. Cs, Sr, Ba and Ag. Silicon carbide is an ideal coating material because of its low thermal neutron absorption cross-section, high ability to retain solid fission products, good irradiation stability, high thermal conductivity, high strength and no thermal creep up to 1900°C. The disadvantage of the high brittleness of SiC is overcome by embedding in dense PyC coatings, for which these particles are called TRISO. Earlier, the fissile and fertile kernels were coated with only PyC and were known as BISO particles. The reference fuel sphere contains approximately 11 000 TRISO coated fissile particles. The active core of the German THTR reactor consists of 360 000 such spherical fuel elements. The modular HTGRs of China and the Russian Federation are also pebble bed type reactors containing spherical fuel elements like those of German HTGRs.

The US fuel element is a hexagonal graphite block, 793 mm in length and 360 mm in width across the flat surface, and containing some 102 coolant channels of diameter 15.9 mm and with 210 fuel holes, which are filled with TRISO fuel compacts of diameter 12 mm and sealed. The active core of the 350 MW(e) HTGR steam cycle consists of 660 graphite fuel elements, while the core of the 600 MW(th) direct cycle GT-MHR consists of 1020 elements. The FSV initial core required about 20 000 kg of HEU, (Th, U)C_2 and ThC_2 TRISO coated particles assembled into some 1500 hexagonal prismatic fuel elements. For cores 1 and 2 of Peach Bottom Unit 1, some 3500 kg of BISO coated HEU (Th, U)C_2 particles were manufactured and assembled into more than 48 000 annual fuel compacts in cylindrical fuel elements.

The Japanese design consists of a block type fuel, similar to the pin-in-block design of the Dragon HTGR fuel in the UK. Each hexagonal block, 550 mm in length and 360 mm across the flat surface, has 31 or 33 fuel holes, each containing an annular fuel pin, which consists of 14 fuel compacts in a graphite sleeve. A fuel compact made of graphite matrix powder with the shape of an annular cylinder, 30 mm in height, 26 mm in outer diameter and 10 mm in inner diameter, contains 13 500 TRISO coated fissile particles. The HTTR active core is composed of some 70 000 fuel compacts.

The manufacture of HTGR fuel elements is carried out in three steps, namely:

(1) Preparation of fertile ThO_2 or ThC_2 and fissile (Th, U)O_2 or (Th, U)C_2 containing LEU or ^{233}U kernels;
(2) Giving the kernels a multilayer coating to form TRISO particles;
(3) Fabrication of fuel elements in the form of spherical balls or prismatic blocks.

The spherical fuel kernels are prepared by the ammonia external or internal gelation process, starting with a nitrate solution of thorium and uranium, followed by reduction at 900°C and sintering at 1500°C to form high density fuel microspheres. For preparation of the carbide fuel particles, carbon black is added to the sol prior to gelation and the hydrated sol containing a homogeneous mixture of oxide and carbon particles is subjected to reaction sintering at 1400–1500°C. The coating is carried out in fluidized bed reactors using a different hydrocarbon gas and methyl trichlorosilane (MTS: CH_3SiCl_3), as shown in Fig. 4. The buffer layer is coated in the temperature range from 1100–1400°C using a mixture of acetylene and argon (C_2H_2 + Ar). The inner PyC layer is coated at 1350–1450°C using a mixture of C_3H_6 and argon. The SiC

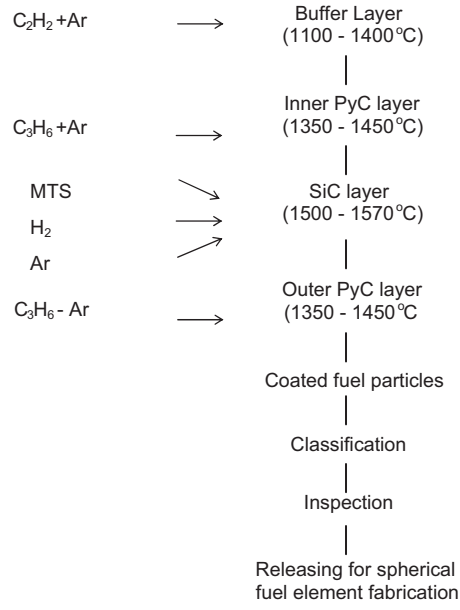

FIG. 4. Flowsheet for multilayer coating of fuel particles for HTGRs.

layer is coated at 1500–1570°C using methyl trichlorosilane (MTS: CH_3SiCl_3), a source of silicon and carbon. Argon and hydrogen are used as the carrier gas along with MTS for the SiC coating. In recent years, a ZrC coating has been preferred instead of SiC for higher burnup. The process parameter for the outer PyC layer is the same as that for the inner PyC layer.

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IMPACT OF THE TECHNOLOGY OF THE MSBR CONCEPT ON LONG LIVED RADIOTOXICITY AND PROLIFERATION RESISTANCE

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Abstract

The Molten Salt Breeder Reactor (MSBR) was an industrial project at the beginning of the 1970s at Oak Ridge National Laboratory in the United States of America and was based on thorium. Immediately before this, the Molten Salt Reactor Experiment (MSRE) worked very well for four years with molten fuel. The MSBR system, where maximum breeding was required, included a graphite moderated core with circulation of a 71.7%LiF–16%BeF_2–12%ThF_4–0.3%UF_4 salt and a pyrochemical reprocessing unit. To obtain the maximum breeding ratio, protactinium was extracted and stored, allowing decay out of the neutron flux. This required the entire volume of salt to be reprocessed in ten days, the gaseous fission products and minor actinides being extracted continuously by helium bubbling and pyrochemical methods. The doubling time was evaluated to be around 25 years. The project has since been re-evaluated, especially within the framework of the EURATOM Molten Salt Reactor Technology (MOST) review. To have an acceptable global reactivity feedback coefficient, studies have shown various possibilities based on core geometry, neutron moderation ratio and salt composition. When requiring only a breeding ratio of 1, it is possible to avoid continuous reprocessing and to strongly simplify it. These various options will be discussed. The detailed inventory will be given, showing clearly the interest in thorium molten salt reactors, where the production of americium and curium is a factor of 100 lower that for the U–Pu RNR. The amount of ^{232}U, which is always produced in the thorium cycle, will be calculated as well as its decay rate since its decay chain eventually results in a 2.6 MeV gamma ray, which may be used to detect and hence control ^{233}U fuel movements. As the ^{233}U has to be produced in other reactors (PWRs, RNRs or other molten salt reactors), special care has to be taken, and this will be discussed.

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1. INTRODUCTION

Until the present time, nuclear energy production has been based on ^{235}U, the only fissile nucleus still existing in nature. It represents only 0.7% of natural uranium, and resource problems are unavoidable if world nuclear energy production increases significantly. To massively extend the capabilities of nuclear fission, the solution is to use the only two fertile nuclei existing on the earth in reactors designed to be able to breed at least as much fissile material as they burn. After one neutron capture and two beta decays, the fertile nuclei, ^{238}U and ^{232}Th are transformed into fissile nuclei as follows:

$$n + {}^{238}U \longrightarrow {}^{239}U \xrightarrow{\beta} {}^{239}Np \xrightarrow{\beta} {}^{239}Pu$$

$$n + {}^{232}Th \longrightarrow {}^{233}Th \xrightarrow{\beta} {}^{233}Pa \xrightarrow{\beta} {}^{233}U$$

The information is summarized in the actinide chart shown in Fig. 1. The uranium cycle is already used in present reactors, where uranium only slightly enriched in ^{235}U and mainly composed of ^{238}U leads to the production of plutonium. That plutonium is partly burnt during the reactor operation, for example in PWRs; one third of the energy comes from plutonium fission. The unavoidable production of plutonium and other minor actinides in the uranium cycle is the main concern for the radiotoxicity produced in the spent fuel and also for the proliferation resistance, a question that is discussed in other contributions to this conference.

Natural thorium does not contain fissile material and must therefore be mixed with fissile material produced elsewhere. Until now it has been used only in experimental reactors fuelled with existing ^{235}U or Pu to extend the capabilities of the used fuel. For reasons that will soon become apparent, thorium, in comparison with uranium, has the potential for energy production as well as for nuclear waste minimization.

2. COMPARISONS BETWEEN THE THORIUM AND URANIUM CYCLES

The condition for breeding in a fission reactor is that the number of nuclei that fission is smaller than, or at most equal to, the number of fissile nuclei created by neutron capture on the fertile nuclei and subsequent decays during the same time span. If ν is the number of neutrons emitted by fission and α is the ratio of capture to fission cross-sections for a fissile nuclei as a function of energy,

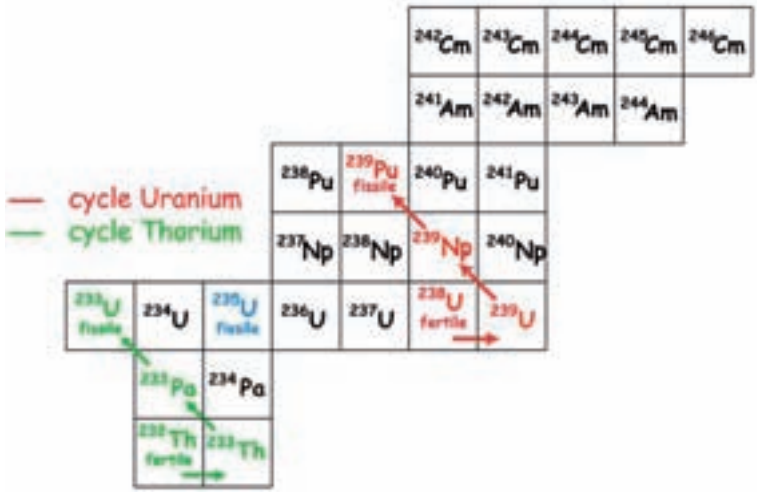

FIG. 1. Actinide chart, showing reaction chains for breeding fissile materials.

ν neutrons are produced
-1 neutron induces a new fission
$-\alpha$ extra neutrons are captured by fissile nuclei
$-(1+\alpha)$ neutrons are captured by fertile nuclei for fissile regeneration

$$\alpha = \frac{\sigma_{\text{capture fissile}}}{\sigma_{\text{fission fissile}}}$$

$= \nu - 2(1+\alpha) = N_b$ neutrons left for breeding

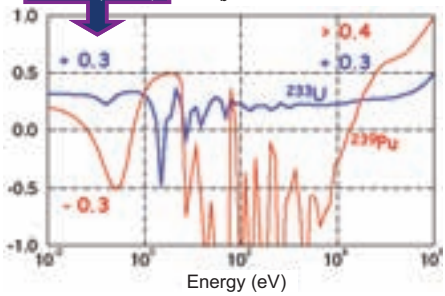

FIG. 2. Number of neutrons available for breeding.

the number of neutrons available for breeding is given by $N_b = \nu - 2(1 + \alpha)$ and is plotted in Fig. 2 for the two fertile elements as a function of neutron energy.

As long as the available neutron number is always slightly larger than 0, breeding is possible. This is the case for the whole neutron energy spectrum for ^{233}U, whereas it is only possible for neutron energies larger than a few tens of

keVs for ^{239}P. This explains why, if plutonium is produced and partly burnt in light water reactors, it is impossible to reach an interesting breeding ratio with a thermal neutron spectrum. The main advantage of thermal spectra is that the amount of fissile material required for starting a chain reaction is smaller (by a factor of up to 6, as will be seen later) than that for fast neutron reactors. Another interesting feature of the thorium cycle is the lower production of actinides, which are the main contributors to the radiotoxicity of spent fuel. Figure 1 shows that five successive neutron captures are necessary to reach neptunium, whereas the uranium cycle is already very close to the minor actinides. The radiotoxicity, a measure of the risk due to the spent fuel of the various fuel cycles, is shown as a function of time in Fig. 3, which clearly indicates the advantage of the thorium cycle, and the production of actinides is discussed further in Section 5. As the number of available neutrons is always small ($N_b \approx 0.3$), it is very important to minimize all the potential neutron losses. As some fission products capture neutrons easily, it is also important to remove them as soon as possible from the reactor core. This is one of the reasons why the thorium cycle has been linked to molten salt reactors (MSRs) from the beginning.

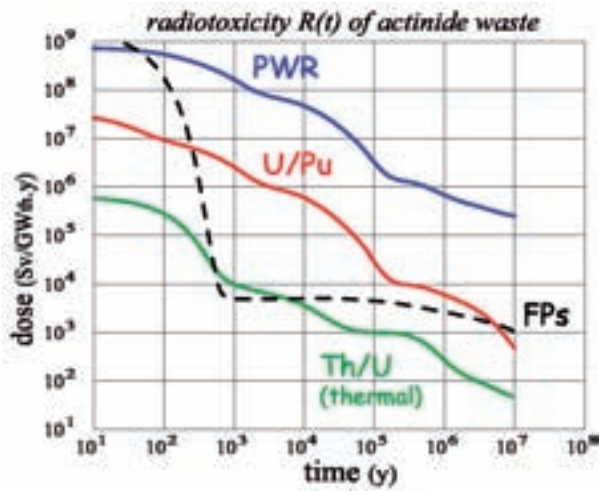

FIG. 3. The radiotoxicities of the various cycles.

3. THE MOLTEN SALT BREEDER REACTOR PROJECT: REVIEW AND DISCUSSION

Molten salt reactor concepts were first studied at the Oak Ridge National Laboratory (ORNL) in the United States of America with the Aircraft Reactor Experiment (ARE) [1], which was based on liquid uranium fluoride circulating in a BeO moderator and which ran for around a hundred hours. Studies were then oriented towards civilian applications such as electricity production. The Molten Salt Reactor Experiment (MSRE) [2] managed from 1964 to 1969 the operation of an 8 MW(th) graphite moderated MSR with a liquid fluoride fuel mixed with lithium and beryllium fluorides. Initially, the fuel was ^{235}U, then ^{233}U and finally Pu were also burnt. The main results were a very good operating performance for over four years, an improvement of the materials used against corrosion and a good understanding of the fuel behaviour. These studies led ORNL to the Molten Salt Breeder Reactor (MSBR) project [3] for a 1 GW(e) industrial reactor based on the thorium cycle and sketched in Fig. 4.

The $LiF–BeF_2–ThF_4$ salt ($\approx 45\ m^3$) circulates through a graphite moderated core, being pumped with a bubbling system and heat exchangers located near the core and partially through a pyrochemical reprocessing unit. The bubbling system is operating continuously and is assumed to extract continuously the gaseous and insoluble fission products that are the most neutron capturing. In the chemical unit, which reprocesses the entire salt volume within ten days, to obtain the largest breeding ratio, the protactinium is quickly extracted with full efficiency and temporarily stored to decay into ^{233}U, which is re-injected into the core with all the actinides, whereas the lanthanides are extracted with 20% efficiency and stored. The doubling time was calculated to be about 25 years for a ^{233}U initial loading slightly larger than one ton. In the end, the concept was not pursued and the ORNL studies were stopped in 1976. Some additional work has been done on the MSR concept in France, Japan and the Russian Federation, and since 2001 a review [4] has been carried out by EURATOM through the Molten Salt Reactor Technology (MOST) project. The recent studies on MSBR have come to two conclusions:

(1) The fast reprocessing scheme is somewhat impractical and may lead to too large thorium losses.
(2) The global temperature feedback coefficient is positive due to the positive effects of graphite, which means that the reactor is not intrinsically safe.

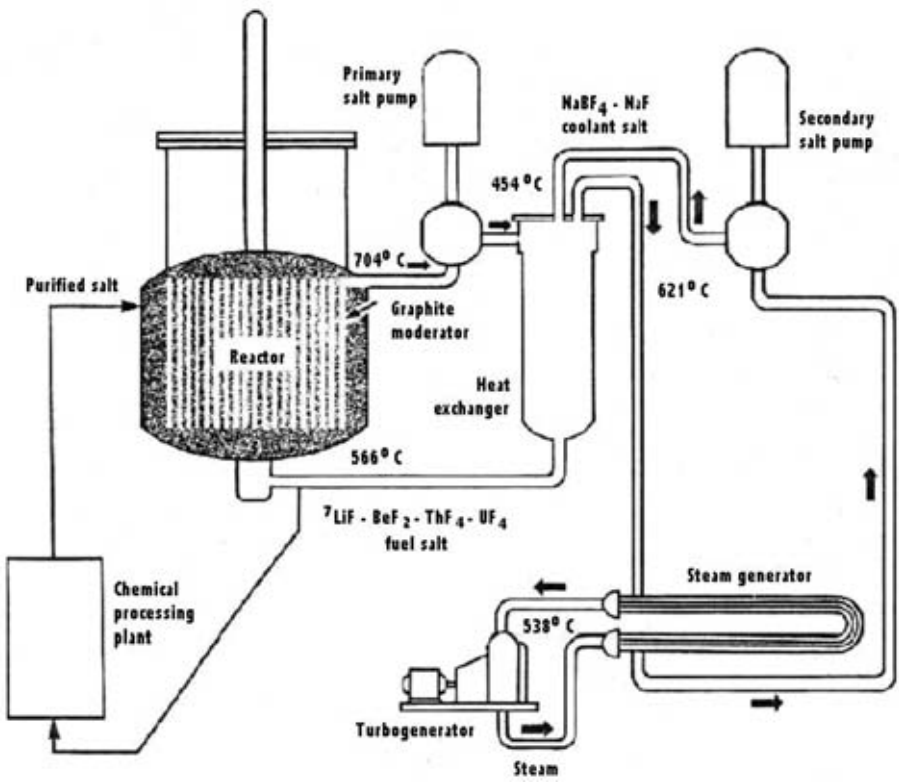

FIG. 4. The MSBR project.

In 1999, we decided, in collaboration with Électricité de France [5, 6], to revisit the MSBR concept from the points of view of the reprocessing constraints and the intrinsic safety [7, 8]. The results presented here have been obtained with a stochastic code based on the neutron transport code MCNP [9] used with the ENDF/B-VI, JENDL 3.2 and JEF 2.2 databases in that order to make feedback coefficient calculations and coupled with Bateman differential equations to make material evolution calculations that allow the exact composition of the core to be known as a function of time. The results of the evolution calculation are sketched in Fig. 5.

4. THE THORIUM MOLTEN SALT REACTOR CONCEPT

The reference concept chosen for our studies in the continuation of MSBR is now described. The core is shown in Fig. 6; it is a cylindrical assembly

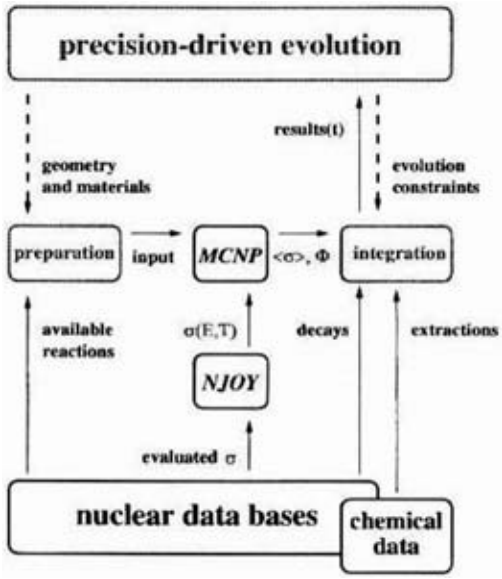

FIG. 5. Organization of evolution tools around the neutron transport code MCNP.

(1.6 m in radius and 3.2 m in height for the reference concept) of graphite hexagons (15 cm side), each pierced by a channel of variable diameter, which allows the moderation ratio to be varied during the studies.

The graphite in the core of a thorium molten salt reactor (TMSR) has a density of 1.86 g/cm^3 and the salt used is LiF (78%)–(HN)F_4 (22%) at 630°C near the eutectic point, where HN stands, at the beginning, for thorium and ^{233}U in a quantity suitable to make the reactor critical. The actinides produced and fission products are then taken into account in the neutron balance. The volume of the salt is equal to 20 m^3, whatever the concept, one third being outside the core. The core also has two plena above and under the graphite matrix and is surrounded by a radial reflector made with the same graphite hexagons hollowed out of cylinders with a radius of 10 cm, where a fertile salt with only thorium as heavy nuclei is included to increase the production of ^{233}U. This uranium is also extracted every six months.

Three concepts have been more extensively studied:

(1) The first, which is directly derived from the MSBR, is the reference TMSR in which the radius of the channels in the graphite is 8.5 cm (the epithermal concept).
(2) The second corresponds to the case where there is no graphite inside the core and the reflectors are made of non-moderating materials (the fast concept).

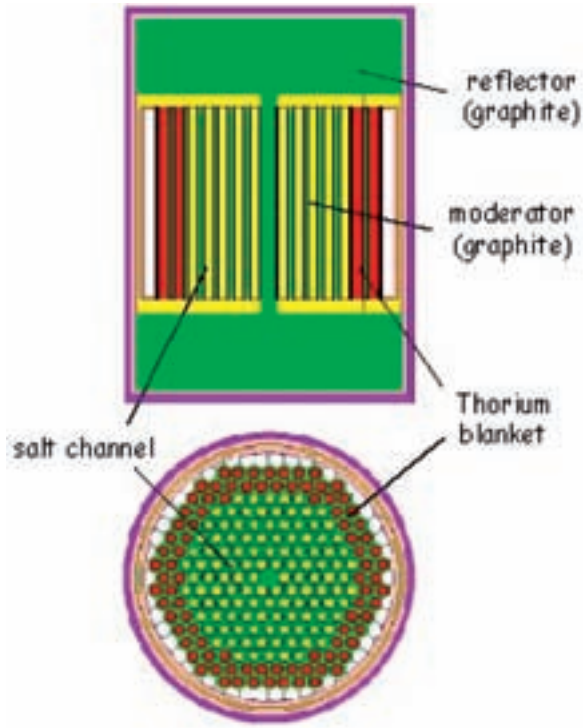

FIG. 6. Representation of the core of the thorium molten salt reactor (TMSR) concept.

(3) The third is the thermal concept in which, to obtain acceptable reactivity feedback coefficients, the hexagon sides have been reduced to 5 cm and the salt channel radii to 1.33 cm.

The geometrical characteristics are summarized in Table 1.

The second modification affects the reprocessing scheme; the new one is shown in Fig. 7. The only continuous action is the bubbling, which is assumed to extract the gaseous and non-soluble fission products present in the salt within 30 s. The properties of the salt are monitored on-line and, in addition, we consider a delayed and separated reprocessing of the whole salt over a six month period. Uranium (including ^{233}U) is extracted by fluorination, and directly and immediately reintroduced into the core. The other processes are aimed at extraction of the lanthanides, which may require the transuranium (TRU) elements and thorium to be extracted previously, and are carried out in a nearby but separate chemical unit.

The possibility of sending the protactinium and TRU elements back into the core or not leads to different inventories at equilibrium, which will be given

TABLE 1. DESCRIPTION OF THE VARIOUS MSR CONCEPTS

Dimension	Reactor type			
	MSBR	Thermal TMSR	Epithermal TMSR	Fast TMSR
Hexagon side (cm)	15.0	5.0	15.0	–
Channel radius (cm)	7.5	1.33	8.5	–
Core radius (m)	2.3	2.55	1.6	1.25
Core height (m)	4.6	5.3	3.2	2.6

later. For the three concepts we have checked that the global reactivity coefficients are negative (Table 2), which ensures the inherent safety of the reactor. The calculations have been made independently for the entire reactor, for the graphite and for the salt, taking into account the two main effects, i.e. the Doppler and density effects.

The sum of the reactivity coefficients for the salt alone is always negative, which is a stipulation for prompt safety. More detailed calculations, taking into account the salt circulation and the temperature differences, are under way to check the MSR safety more carefully. However, at that time, with these calculated negative reactivity coefficients, with the absence of pressure even at high temperature, with the absence of reactivity reserve and with the possibility

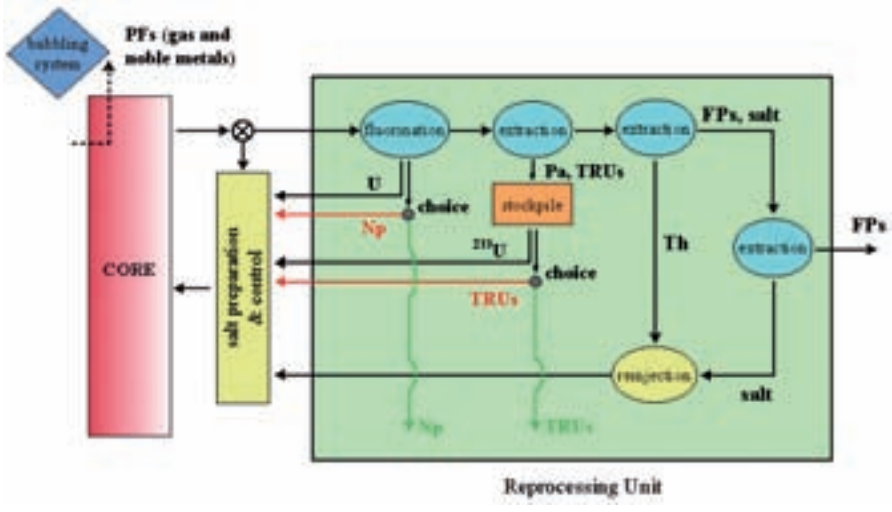

FIG. 7. The TMSR reprocessing scheme.

TABLE 2. REACTIVITY COEFFICIENTS FOR VARIOUS REACTOR TYPES

Effect	Reactor type			
	MSBR	Thermal TMSR	Epithermal TMSR	Fast TMSR
Doppler	–3.3	–3.2	–6.1	–3.2
Salt density	+2.4	+0.6	+3.2	–2.2
Graphite	+1.6	+1.8	+0.5	0
Global (‰)	+0.7	–0.8	–2.4	–5.4

of draining the salt quickly into safety tanks, thorium MSRs appeared to be very attractive from the safety point of view.

We will now give the results of calculations concerning the production of the various actinides during reactor operation and discuss these results from the point of view of reduction of waste production and of proliferation concerns.

5. ACTINIDE YIELDS

The thorium chain is shown in the upper part of Fig. 8, and it can be clearly seen that the transition to the uranium chain (in the lower part of Fig. 8) is made through the ^{235}U, a fissile nucleus, and that the ^{237}Np, the first minor actinide found in this diagram, is produced after five neutron captures and three beta decays.

The occurrence of these processes is mainly dependent on the neutron spectra and fluxes, so the results will be presented in various conditions up to reaching equilibrium, which, in some cases, would require the reactors to run for one hundred years or even longer. This is illustrated in Fig. 9, where the quantity of the various actinides and uranium isotopes present in the core of an MSBR-like reactor is given as a function of the operating time.

Even if equilibrium has not been clearly reached for curium and americium after one hundred years, we will take the values obtained at that time for comparisons in various situations. The values obtained for the MSBR and the TMSRs under various conditions are given in Table 3 under the same operating assumptions, which is that the actinides, if separated, are sent again into the core after their extraction, which explains the slow buildup of higher mass actinides.

The values obtained are very low, especially for the heaviest minor actinides and illustrate very well the interest in the thorium cycle for minimization of long life nuclear waste production. The amount of actinides sent to

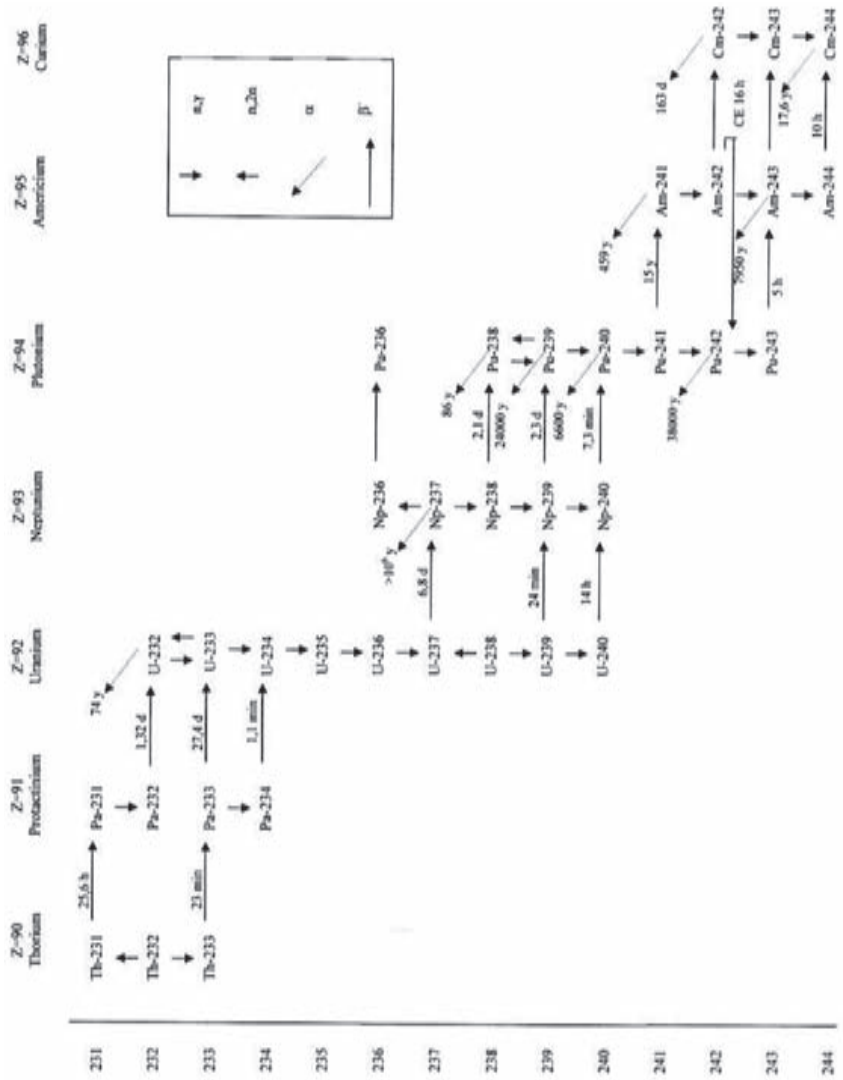

FIG. 8. The thorium and uranium chains.

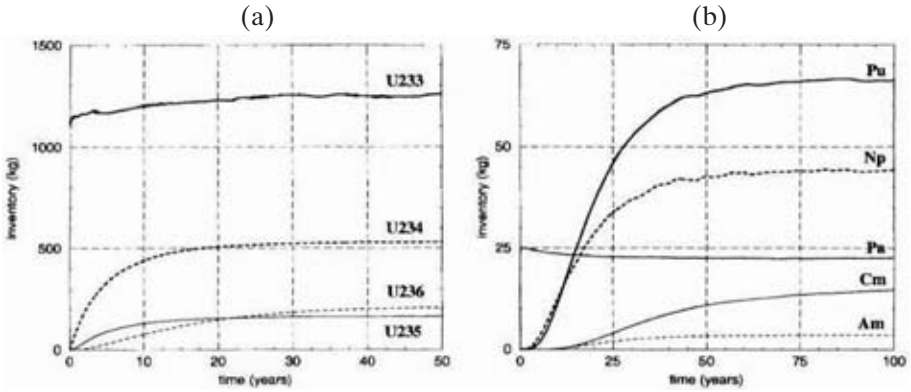

FIG. 9. Production of (a) uranium and (b) actinides in the MSBR.

the wastes will be proportional to the quantity of actinides present in the core and to the efficiency of the lanthanide extraction process, which is not chosen now but in any case will be very small. The production of minor actinides is compared in Table 4 with the values calculated for light water reactors and fast neutron reactors [10] in the case of 1 GW(e) (≈7 TW · h(e) annually). This table gives the inventories that have to be taken in charge when the reactor is stopped. The interest in MSRs for the thorium cycle appears clearly as the only comparable values are for the neptunium isotopes, the other actinide values

TABLE 3. ACTINIDE INVENTORY AT EQUILIBRIUM (kg) FOR CLOSED CYCLES

Element	Reactor type			
	MSBR	Thermal TMSR	Epithermal TMSR	Fast TMSR
Th	65 560	48 400	45 300	43 300
Pa	22.5	75	74	85
U	2 156	1 600	4 200	8 300
Np	44.0	29	110	150
Pu	66	38	260	270
Am	3.6	3.1	7.1	4.8
Cm	15.1	14.1	18.0	2.4
Bk	0.01	0.005	0.04	0.0002 (100 a)
Cf	0.081	0.06	0.11	0.0007 (100 a)

TABLE 4. COMPARISON OF INVENTORIES (kg) FOR VARIOUS NUCLEAR REACTOR CONCEPTS IN CLOSED FUEL CYCLES

Element	Reactor type				
	Thermal TMSR	Epithermal TMSR	Fast TMSR	FNR	PWR
Np	29	110	150	70	91.5
Pu	38	260	270	12 550	3850
Am	3.1	7.1	4.8	528	248
Cm	14.1	18	2.4	135	124

being significantly lower. This is the explanation of the radiotoxicity curves presented in Fig. 3.

The results clearly emphasize the interest in the thorium cycle for minimization of the heaviest actinides (plutonium and the heaviest elements), which make the major contributions to the radiotoxicity of nuclear wastes. Only the fission products (less than one ton each year) have to be managed and moved elsewhere.

Another way of operating might be to burn the extracted TRU elements in separate reactors that would be dedicated to burning actinides (mainly neptunium and plutonium). In this case, the minor actinide inventories are clearly lower and the calculation leads to the values given in Table 5, where the the quantities of actinides present in the core at equilibrium are compared again with those of other reactor cycles. The results are still very good; the values that have to be taken into account annually being very low and the quantities extracted to be burnt elsewhere annually being rather limited.

TABLE 5. COMPARISON OF INVENTORIES (kg) FOR VARIOUS NUCLEAR REACTOR CONCEPTS IN OPEN CYCLES

Element	Reactor type				
	Thermal TMSR	Epithermal TMSR	Fast TMSR	FNR	PWR
Np	7.0	15.0	9.7	23.0	102
Pu	1.9	2.8	0.6	12 250	1420
Am	4.0×10^{-4}	5.0×10^{-4}	7.01×10^{-7}	192	86.0
Cm	0.001	1.0×10^{-4}	2.0×10^{-8}	15.0	14.0

As a partial conclusion, we observe that the quantity of actinides in the core is very small even for neptunium and plutonium, whichever hypothesis is chosen, and that the quantity of radioactive materials that have to be sent out of the nuclear plant is also very small.

Production of tritium is often considered to be important in MSRs, so we have calculated the amounts produced for the concepts studied. The main source is ^{6}Li, through the following reaction:

$$n + {}^{6}Li \longrightarrow t + \alpha$$

Lithium-6 comes at the beginning from the non-separated part in the initial lithium which will disappear within the first five operating years and, in the case of the presence of beryllium in the salt, ^{6}Li is permanently produced through the following reaction:

$$n + {}^{9}Be \longrightarrow {}^{6}He + \alpha$$

^{6}He decaying quickly into ^{6}Li.

The second reaction producing tritium is directly due to ^{7}Li but the cross-section is smaller than that for ^{6}Li:

$$n + {}^{7}Li \longrightarrow t + \alpha + n$$

The quantity produced, in the case of the MSBR, amounts to 0.385 kg/a at the beginning and decreases to 0.150 kg/a at equilibrium after five years; in the case of the TMSR concepts, tritium production begins at a level of 0.185 kg/a and equilibrium is reached at 0.11 kg/a. These values are not negligible but, for TMSRs, are always smaller than the 0.280 kg/a obtained in the CANDU reactors in operation at present.

6. NON-PROLIFERATION CONSIDERATIONS

Our reactor is working with ^{233}U which, owing to its small critical mass, around 15 kg, and its half-life, 1.5×10^{5} years, is an interesting potential material with which to make nuclear weapons. We will therefore examine the uranium cycle during the whole reactor operation. The uranium fuel is diluted in salt and represents, by mass, a few per cent of the salt. Thus, to obtain a quantity of uranium sufficient to make a weapon requires a chemical unit able to process at least a few tons of salt. As the salt composition is continuously monitored, there is no reactivity reserve in the core. Moreover, as we have seen

in the preceding section, the various uranium isotopes are quickly produced and are mixed with the ^{233}U and are extracted together in chemical processes such as fluorination. The buildup of uranium isotopes is shown in Fig. 10, while the uranium isotope production values are given in Table 6 for the various concepts presented here. Their presence increases the critical mass and the required quantity of salt that has to be reprocessed to obtain it, which is equivalent to an isotope dilution. The only way to obtain pure ^{233}U is to use an efficient and fast protactinium separation and to let it decay out of the neutron flux. This was the case for the MSBR project, which could produce 38 kg annually; in the TMSR cases, there is no ^{233}U available because the regeneration is obtained from the uranium produced, not only in the core but also in the axial blanket from where it is extracted every six months.

Concerning proliferation resistance, the most interesting product is ^{232}U, which is mainly produced (Fig. 8) by an (n, 2n) reaction on thorium and, to a lesser extent, by an (n, 2n) reaction on ^{233}U. These reactions have a high energy threshold, but Fig. 10 shows that, whatever the concept, the production of ^{232}U will still occur. In Fig. 10(a) are, given as functions of neutron energy, the cross-sections for the (n, 2n) reactions and the neutron fluxes in the core and the blanket for the various reactor concepts. In Fig. 10(b), is presented the quantity of ^{232}U related to the whole uranium present in the salt as a function of the graphite channel radii, i.e. as a function of the moderation ratio for the neutron flux. Unlike the results given in Table 6, the ^{232}U production shown in Fig. 10(b) is calculated with the hypothesis that the ^{231}Pa was not sent into the core after separation and ^{233}Pa decay. This minimizes by a factor of 2 (thermal concept) to 7 (fast concept) the ^{232}U production, and we are sure even in this case that the quantity produced is sufficient to prevent ^{233}U diversion.

TABLE 6. URANIUM ISOTOPE INVENTORIES AT EQUILIBRIUM (kg)

Element	Reactor type			
	MSBR	Thermal TMSR	Epithermal TMSR	Fast TMSR
^{232}U	2.8	2.5	3.7	14.0
^{233}U	1250	790	2400	5200
^{234}U	530	470	1100	1900
^{235}U	160	100	410	560
^{236}U	210	250	380	580
^{238}U	2.8	4.8	1.5	0.7
Total U	2155.6	1616	4290	8254

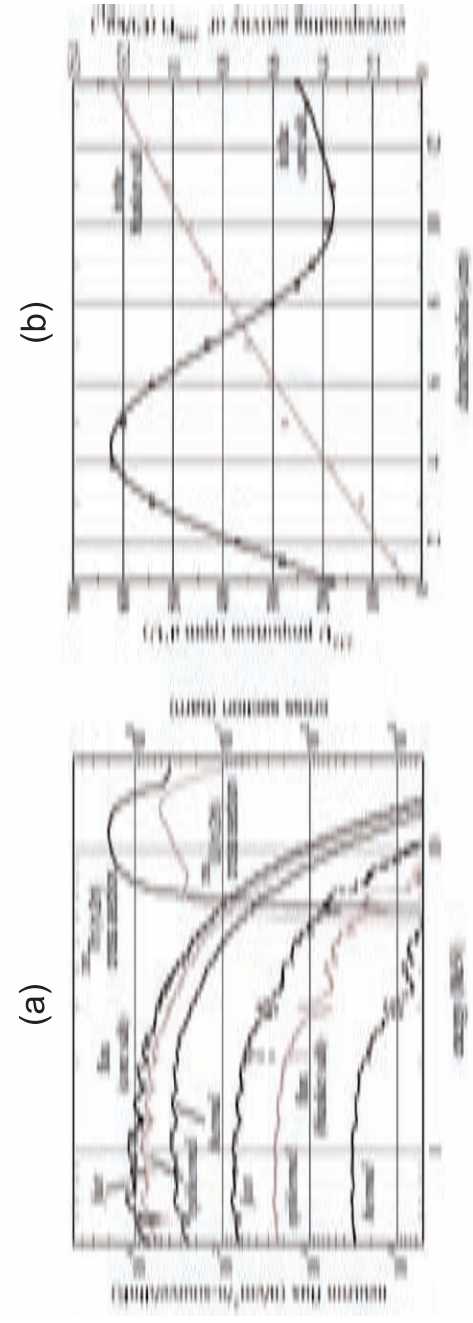

FIG. 10. (a) (n, 2n) Cross-sections for ^{232}U production and the high energy part of the neutron flux for the various concepts. (b) Proportion of ^{232}U in the uranium present in the core as a function of moderation ratio.

The proportion is also given for the thorium salt in the reflector and even in that case the ^{232}U production is not at all negligible. Another interesting result is the weak sensitivity to the neutron spectrum which implies that, whatever the initial load and operating condition of the reactor, the inventory will contain a noticeable amount of ^{232}U. The decay scheme of ^{232}U (half-life: 68.9 years) is given in Fig. 11.

The main feature related to proliferation resistance is the presence in 36% of ^{232}U decays of a very energetic (2.6 MeV) gamma ray, which prevents easy manipulation of the salt and above all of the extracted uranium and which may therefore help to detect the diversion of uranium even in small quantities. The slowest step in the decay chain is the ^{228}Th decay (1.91 a). The activity related only to that gamma ray, assuming equilibrium between its descendants, is 250 GBq for 1 kg of extracted uranium, with a $^{232}U/U$ ratio of 250 ppm. This value combined with the gamma energy explains why manipulation and transport of diverted uranium are virtually impossible without their detection and present a serious hazard to their transporters. An illustration of the growth of activity is shown in Fig. 12, which is extracted from the Ref. [11].

The weakest point of the MSBR project, where the uranium balance was favourable and where the protactinium was quickly extracted and efficiently separated to let the ^{233}Pa decay into ^{233}U, is the possibility of diverting some of that uranium at the most advantageous time to obtain rather pure ^{233}U. The TMSR project is calculated to run without the capability of producing extra uranium; the aim of the reprocessing unit (Fig. 7) is to extract the lanthanides but, before that extraction, it may be necessary to extract the TRU elements, palladium and thorium. If the extracted palladium, TRU elements and thorium are sent again directly into the core, there is no problem; in the other case, it would be necessary to leave at the first fluorination a sufficient percentage of uranium, which will be mixed with the uranium coming from the palladium decay to obtain uranium with enough ^{232}U.

Another interesting TMSR feature is that the fast neutron spectrum option may run for 20 years without salt reprocessing, i.e. without a reprocessing unit, but this option requires around 5.5 tons of ^{233}U to start the chain reaction. Some possibilities exist to reduce this quantity and have been presented in Ref. [12]. With the reprocessing unit in the vicinity of the reactor, the problems related to transport of fuel are greatly reduced, and this unit may be adapted at the beginning to receive the thorium irradiated in other reactors to produce the first ^{233}U load without any other manipulations. As the chemical reprocessing schemes are not firmly established at present, it is difficult to go further in drawing definite conclusions about the proliferation issues concerning the MSRs in the thorium cycle. However, it is clear that the unavoidable production of ^{232}U together with the production of ^{233}U is the

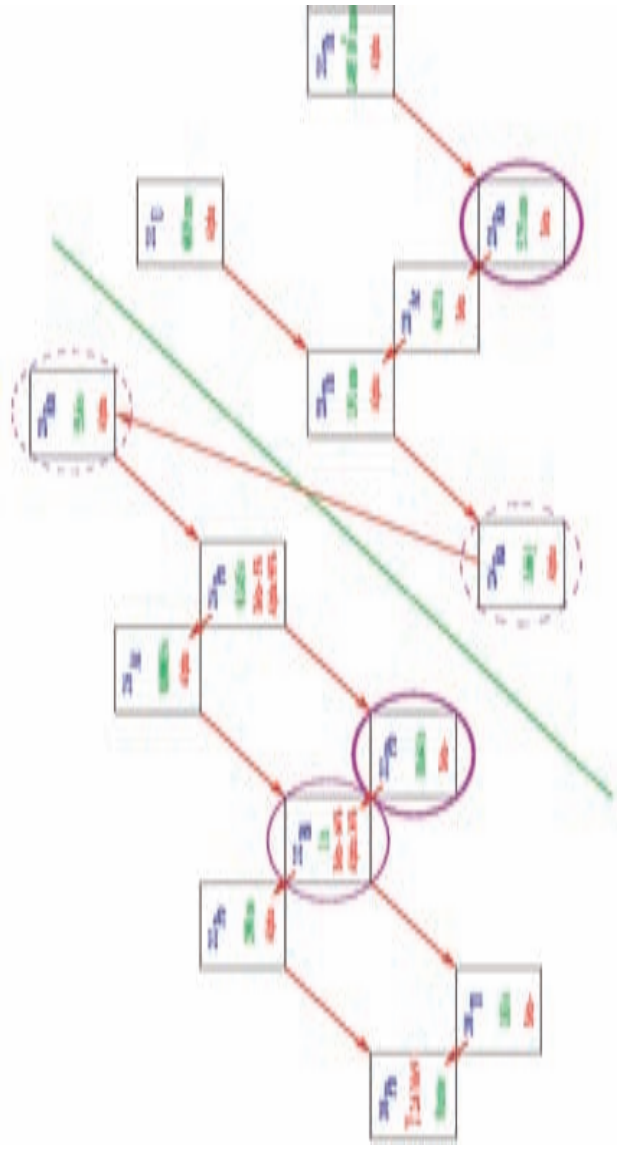

FIG. 11. Decay scheme of ^{232}U.

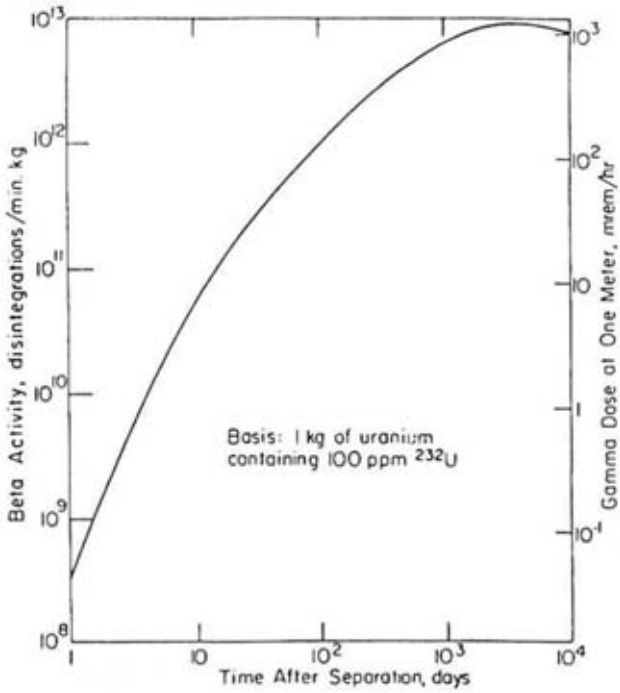

FIG. 12. Growth of the activity [11].

main obstacle, as its presence prevents easy manipulation and transport of fissile material.

From the point of view of proliferation, plutonium has been taken into account and does not matter because it is produced in very limited quantities and the larger part (more than 60%) of it is ^{238}Pu, which is characterized by a large heat release; so MSRs are not at all convenient for making nuclear weapons with the plutonium produced.

7. SUMMARY

The thorium cycle has very interesting characteristics for production of nuclear energy. It has been demonstrated that it is possible to obtain a breeding ratio at least equal to 1 with any kind of neutron spectrum in the reactor and to obtain good reactivity feedback coefficients, thus ensuring inherent safety. Therefore, these reactors are well suited to the objectives that have been defined for future nuclear reactors by the Generation IV international forum.

As shown in this paper, from the physical point of view, thorium MSRs allow the main criteria now required for sustainable nuclear energy production to be satisfied: saving resources, intrinsic safety, minimization of waste production, no reactivity reserves and good resistance to proliferation. Many possibilities have been presented, which require further studies according to the priorities and technologies that will be chosen. However, in any case, the production of minor actinides is very low, and this will greatly simplify the management of the wastes, which will mostly consist of fission products. In MSRs, the fissile material is disseminated in small quantities (1–3%) in the salt and requires reprocessing of a large amount of salt to obtain a sufficient quantity of fissile material. Moreover, the unavoidable production of ^{232}U accompanying ^{233}U production, which might be a major problem in the case of solid fuel preparation, places very strong constraints on the manipulation of uranium and helps prevent undesirable use and transportation.

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TECHNICAL BARRIERS FOR PROLIFERATION RESISTANCE IN MANAGEMENT OF NUCLEAR MATERIAL

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Abstract

Protected plutonium production (PPP) has been proposed to improve the proliferation resistance of plutonium by transmutation of minor actinides (MAs). The PPP studies focus on the transmutation of MAs such as ^{237}Np and ^{241}Am to increase the fraction of ^{238}Pu, which is a high spontaneous fission neutron source, to reduce the quality of nuclear explosives. In addition, the high decay heat of ^{238}Pu makes the processes of nuclear weapon manufacture and maintenance technologically difficult. A small amount of doping of uranium fuel with MAs is sufficient to increase the fraction of ^{238}Pu up to the level where it is practically unusable as nuclear explosives, even at the beginning of irradiation when the void and Doppler coefficients for LWRs are negative. Feasibility studies have also been performed for protected plutonium breeding in the blankets of FBRs by applying PPP technology. This technology has been extended to studies on the denaturing of the reactor grade plutonium from LWRs by doping with MAs of mixed oxide fuel, uranium-free fuels such as inert matrix fuel (IMF) and thorium fuel with small amounts of depleted or natural uranium, to avoid the production of pure ^{233}U from ^{232}Th. The PPP technology makes the fuel cycle more flexible with respect to fuel storage for future energy crises because of the enhanced proliferation resistance of plutonium. Instead of geological disposal or simply their burning through fission, MAs can be used effectively as burnable and fertile materials to improve the proliferation resistance of plutonium. The PPP technology will in future open up new markets for nuclear reactors with improved proliferation resistance.

1. INTRODUCTION

An approach to harmonization of nuclear energy systems with society and the global environment was proposed by Fujii-e in the form of the

following four universal principles of self-consistent nuclear energy systems (SCNESs), which should be implemented simultaneously [1]:

(1) Energy production;
(2) Fuel breeding;
(3) Transmutation of radioactive wastes;
(4) System safety.

Self-consistent nuclear energy systems are oriented to the ultimate goal of zero release of radioactive wastes from the nuclear energy system and full use of natural uranium resources.

Establishment of an SCNES that simultaneously meets the above four requirements strongly relies on the generation of excess neutrons. To improve the neutron economy and the neutron quality for transmutation, the studies have been extended to the multicomponent SCNES (MC-SCNES) [2] with external non-fission based neutron sources by spallation and fusion reactions, since for future sustainable growth the excess neutrons can be greatly upgraded to approach the ultimate goal of zero release of radioactive wastes with unlimited creation of fuel resources by breeding.

In the SCNES concept, minor actinides (MAs) are not treated as wastes but as part of the fuel. Protected plutonium production (PPP) has been proposed to increase the proliferation resistance of plutonium accumulated in the environment of LWRs [3, 4]. The studies focus on the transmutation of MAs such as ^{237}Np and ^{241}Am with large neutron capture cross-sections to increase the fraction of ^{238}Pu, as shown in Fig. 1 [5], which is a high neutron source from spontaneous fission (2.6×10^3 n · g^{-1} · s^{-1}), to reduce the quality of plutonium nuclear explosives. In addition, the high decay heat of ^{238}Pu (560 W/kg) makes the processes of nuclear weapon manufacture and maintenance technologically difficult. The decay heat and the neutron source of the spontaneous fission in plutonium vectors are compared in Fig. 2.

Protected plutonium production has been studied in thermal and fast breeder reactors [6–12]. The studies on PPP technology have been extended to denaturing of reactor grade plutonium by doping with MAs of mixed oxide (MOX) fuel, uranium-free fuels such as inert matrix fuel (IMF) and thorium (ThO_2) fuel with small amounts of depleted or natural uranium, to avoid the production of pure ^{233}U from ^{232}Th in LWRs [13–15].

The present paper summarizes these studies from the viewpoint of the technical barriers for the proliferation resistance of plutonium.

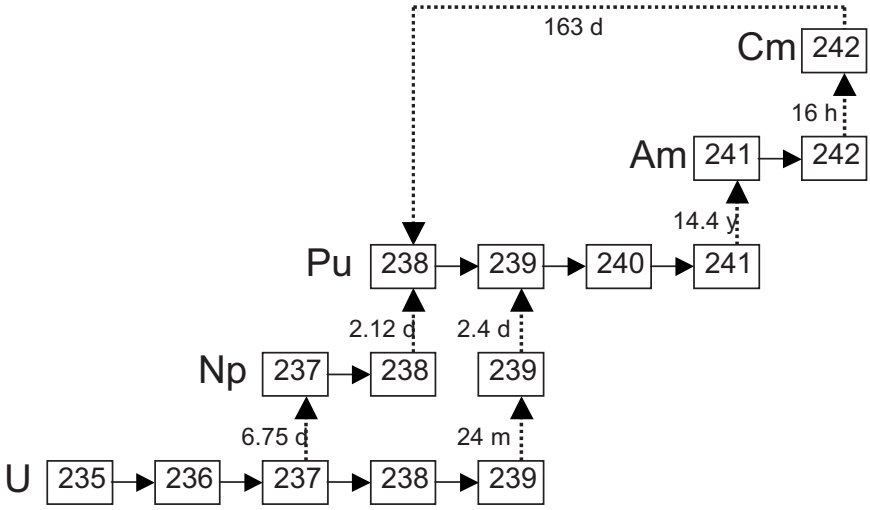


FIG. 1. Principal chains of nuclear reactions leading to accumulation of ^{238}Pu [5].

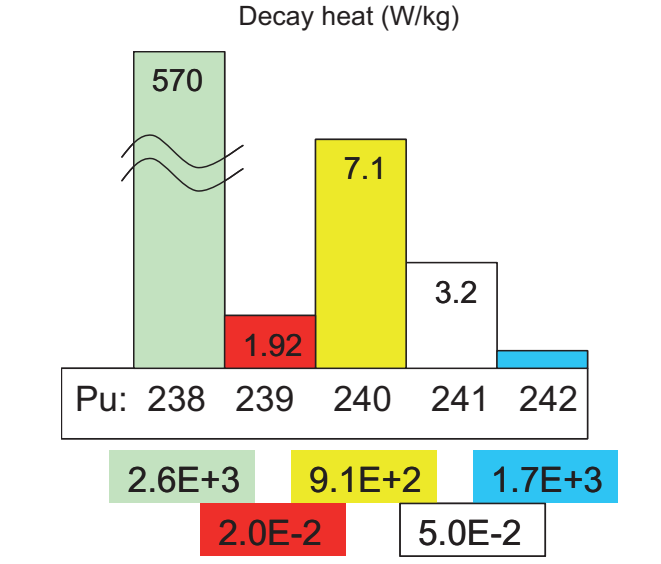


FIG. 2. Property of proliferation resistance of plutonium.

2. FUNDAMENTALS OF PROTECTED PLUTONIUM PRODUCTION

2.1. Protected plutonium production in LWRs

The infinite cell burnup calculations of advanced fuel with UO_2 by doping with ^{237}Np in the conventional LWR neutron spectrum are performed with the SCALE 4.4 code system. Figure 3 shows an example of the effect of 2% ^{237}Np doping on the burnup dependence of the criticality (k_{inf}) for a typical PWR configuration. As shown in Fig. 3, ^{237}Np doping shrinks the area of critical performance by reducing the initial criticality and the final burnup value (assumed at k_{inf} = 1). However, fuel burnup can be extended in medium enriched uranium fuel, for example, 12% enrichment of ^{235}U, because of the burnable poison effect of ^{237}Np.

The plutonium accumulations in the advanced fuel with ^{237}Np are compared in Fig. 4 with the results of conventional fuels with UO_2 (3.3%

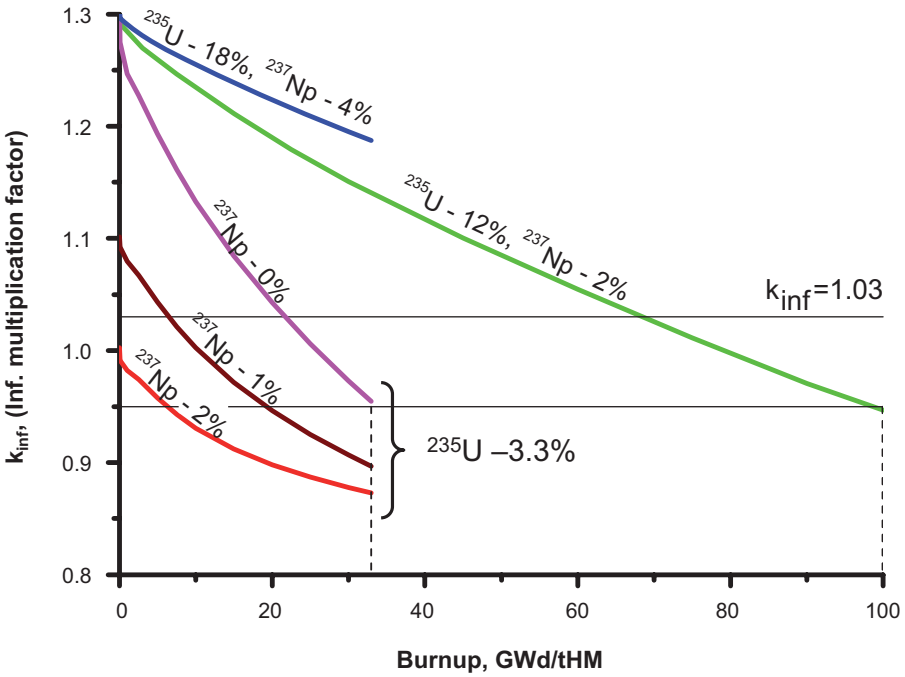


FIG. 3. Example of effect of ^{237}Np doping on burnup dependence of criticality (k_{inf}) for a typical PWR configuration.

(a) UO_2 (3.3% enrichment) (b) UO_2 (12% enrichment) + ^{237}Np (2%)

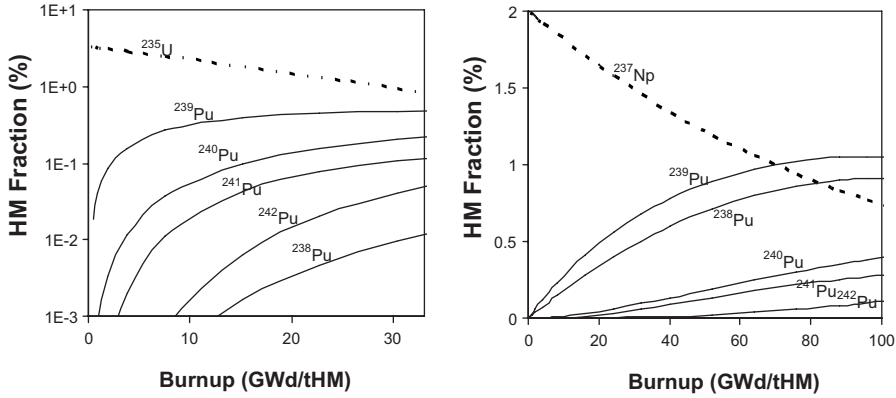
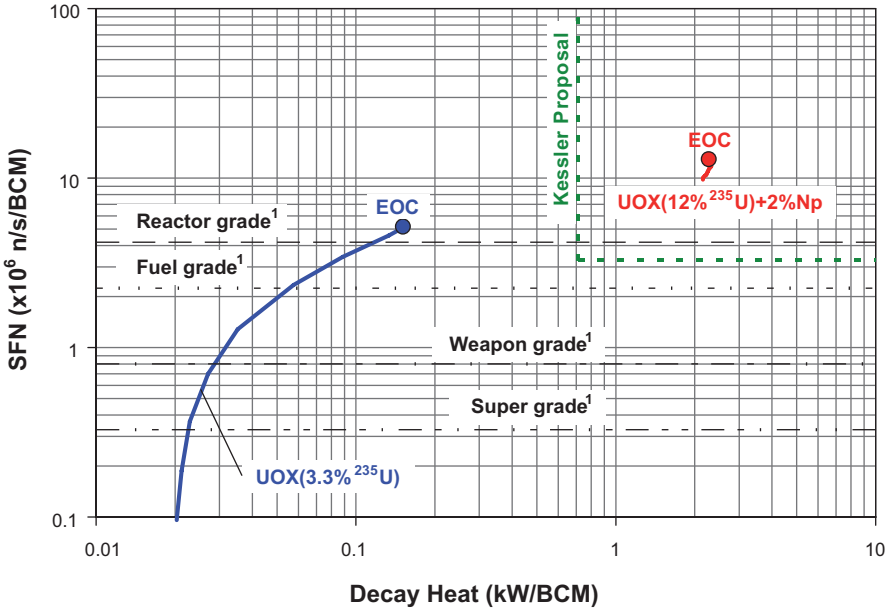


FIG. 4. Plutonium accumulations in (a) conventional fuel and (b) advanced fuel with ^{237}Np.

enrichment of ^{235}U). The proliferation resistance of plutonium is evaluated by the following indexes: the bare critical mass (BCM), decay heat (DH) and spontaneous fission neutron (SFN) rate. As shown in Fig. 4, with a decrease of ^{237}Np during burnup, ^{238}Pu is built up and increasing in the advanced fuel with ^{237}Np. Because of ^{238}Pu in the advanced fuel, the proliferation resistance based on the DH and SFN rate per BCM of plutonium in the advanced fuel with ^{237}Np is higher, even in the initial stage of burnup, than the resistances of conventional fuels, as shown in Fig. 5. This figure shows a proliferation resistance map based on decay heat versus spontaneous fission neutron rate, which is compared with the traditional suitability for explosives of a plutonium mixture based on ^{240}Pu [16] and Kessler's proposal based on ^{238}Pu [17].

Figure 6 shows the effect of transplutonium (Am–Cm) doping on the burnup dependence of k_{inf} for a typical PWR configuration. Mixtures of transplutonium isotopes are taken in the proportions that appear in PWR fuel with an initial enrichment of 5% burned up to 50 GW · d/tHM and cooled for one year after discharge. In general, ^{237}Np doping shrinks the area of critical performance by reducing the initial criticality and the final burnup value; however, for the same mass addition, transplutonium doping markedly reduces reactivity swings compared with the ^{237}Np effect. The combined effects of ^{237}Np with transplutonium elements (3%) results in more ^{238}Pu accumulating in the spent fuel, as shown in Fig. 7.

The larger amount of doping with MAs makes the operation of achieving criticality impossible. In the mixture of discharged MAs, the ^{237}Np content is



[1] Adapted from B. Pellaud, "Proliferation aspects of plutonium recycling", J. Nucl. Mater. Management **31** 1 (2002) (Reactor grade: 30% ^{240}Pu + 70% ^{239}Pu; Fuel grade: 18% ^{240}Pu + 82% ^{239}Pu; Weapons grade: 7% ^{240}Pu + 93% ^{239}Pu; Super-grade: 3% ^{240}Pu + 97% ^{239}Pu).

FIG. 5. Comparison of proliferation resistance of conventional fuel and advanced fuel with neptunium.

about 50% in MAs, so that the combined effects of MAs will bring the initial criticality for conventional fuel into the subcritical domain at an MA doping mixture of more than 1% for the standard LWR configuration (5% ^{235}U enrichment) with a pitch-to-diameter (P/D) ratio of 1.4. For the medium enriched uranium case (20% ^{235}U), the core becomes subcritical at an MA doping of more than 10%. It appears that an MA doping of 30% is close to the maximum that 20% enriched uranium could accept in subcritical operation. Doping with MAs oriented for the subcritical mode obviously gives an advantage in both the quality of protection and the mass produced; however, the extent to which the subcritical mode could retain this attribute depends upon the energy balance in an accelerator driven system. The doping effects of MAs on the proliferation resistance of plutonium in critical and subcritical operation modes are summarized in Fig. 8, compared with the traditional suitability as an explosive of a plutonium mixture based on ^{240}Pu content [16], Kessler's proposal based on ^{238}Pu [17] and IAEA guidelines that identify

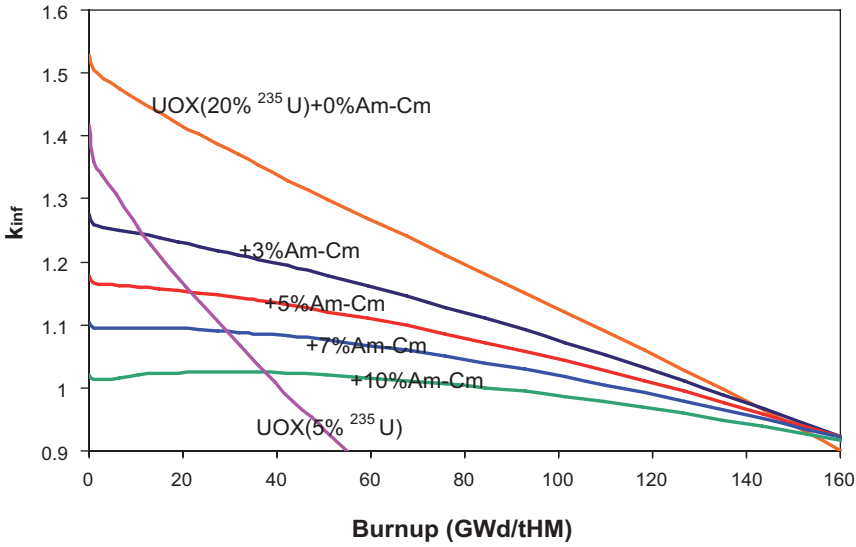


FIG. 6. Example of effect of transplutonium doping on burnup dependence of criticality (k_{inf}) for a typical PWR configuration.

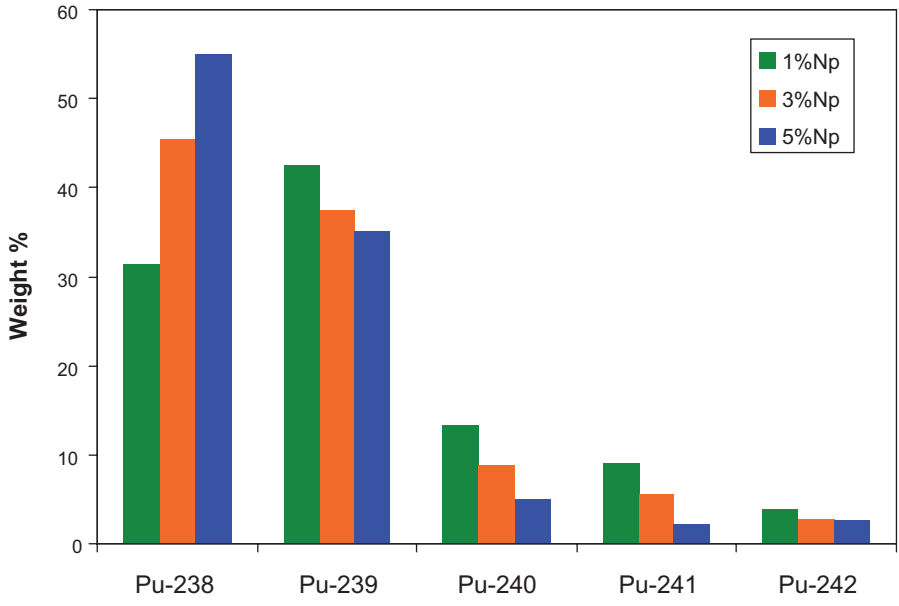
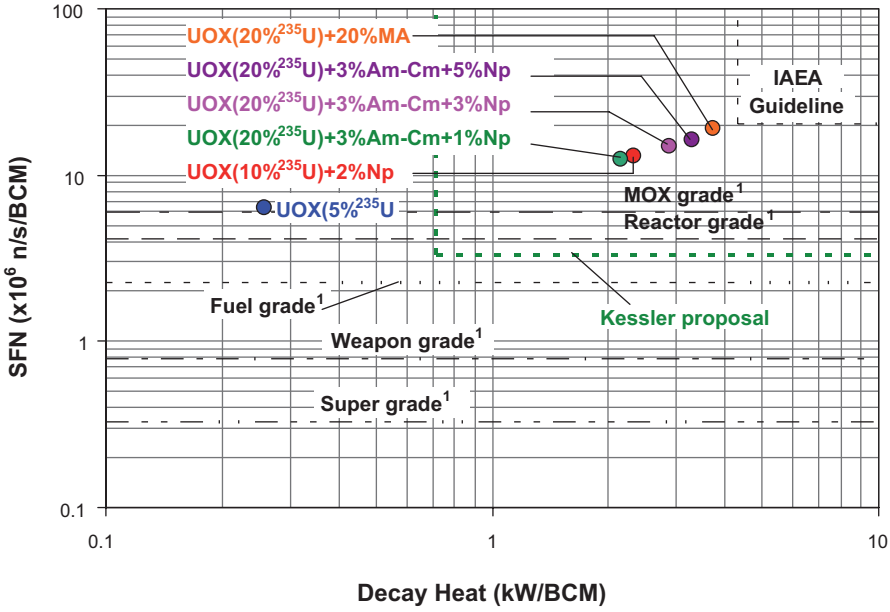


FIG. 7. Combined effects of ^{237}Np with transplutonium elements on ^{238}Pu accumulation.



[1] Adapted from B. Pellaud, "Proliferation aspects of plutonium recycling", J. Nucl. Mater. Management **31** 1 (2002) (Reactor grade: 30% ^{240}Pu + 70% ^{239}Pu; Fuel grade: 18% ^{240}Pu + 82% ^{239}Pu; Weapons grade: 7% ^{240}Pu + 93% ^{239}Pu; Super-grade: 3% ^{240}Pu + 97% ^{239}Pu).

FIG. 8. Summary of doping effects of MAs on proliferation resistance of plutonium in critical and subcritical operation modes.

plutonium with a ^{238}Pu fraction of more than 80% as being exempted from safeguards [18]. The IAEA guideline, however, is to be considered as the uppermost limit.

Figures 9 and 10 summarize the PPP mechanism and the contributions of PPP technology to advanced LWRs with uranium fuel, which will open up new markets for nuclear reactors with improved proliferation resistance in the future.

On the basis of fundamental theoretical studies on the PPP mechanism, the PPP-A project has been initiated with financial support from the Ministry of Education, Culture, Sports, Science and Technology in Japan [19]. The objectives of the PPP-A project are to develop innovative nuclear technology based on the transmutation of ^{237}Np, which is at present treated as high level waste (HLW), to reduce HLW, to establish high burnup cores and to produce protected plutonium with high proliferation resistance in a wide range of the

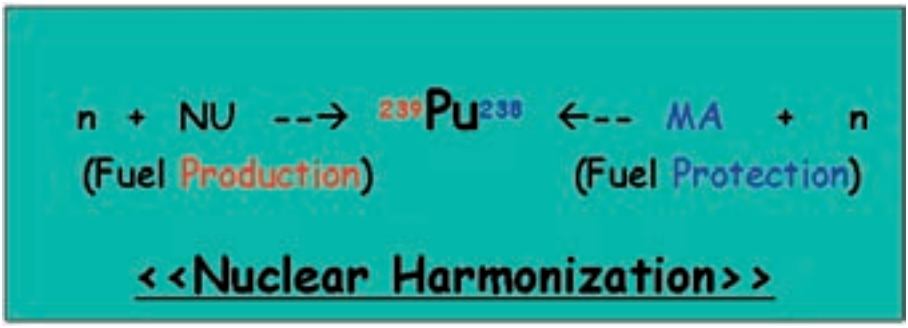


FIG. 9. Mechanism of protected plutonium production.

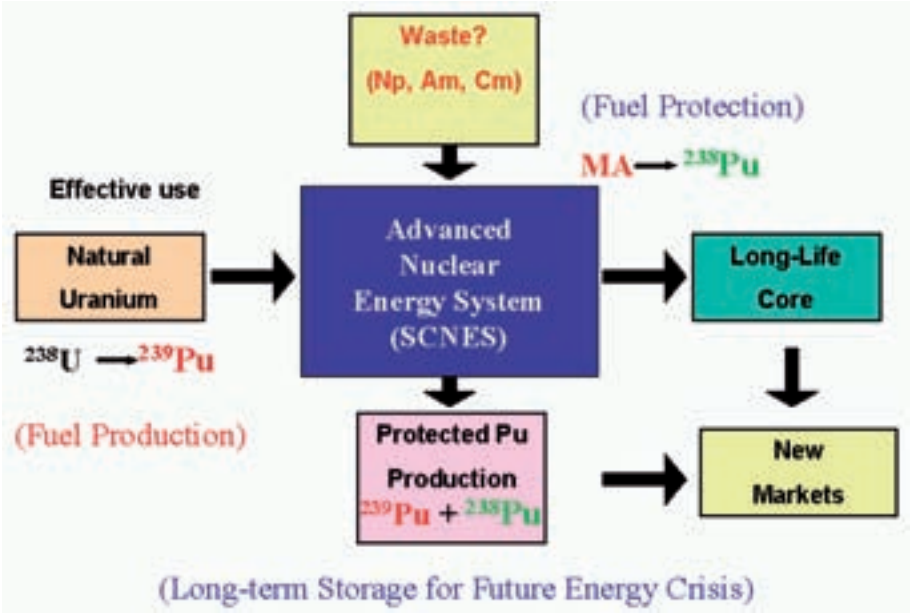


FIG. 10. Contributions of PPP technology to advanced reactors with uranium fuel.

neutron spectrum. To demonstrate the PPP mechanism, irradiation tests of samples of UO_2 with ^{237}Np will start in the Advanced Test Reactor (ATR) at the Idaho National Laboratory (INL) in the United States of America in September 2005.

2.2. Protection plutonium breeding in FBRs

Medium size FBRs (1600 MW(th)) were selected to study the feasibility of protected plutonium breeding [10–12]. The core layout and geometry are shown in Fig. 11 [12]. In the present study, MOX fuel in the active core and MAs extracted from 33 GW · d/tHM irradiated fuel with 3.3% ^{235}U in PWRs (with a cooling time of three years), together with computer codes such as SLAROM, JOINT and CITATION FBR, were used. The effective cross-section used in this calculation incorporated seventy effective group constants, which were provided from the Japanese standard constant set, JFS-3-J-3.2R, which is based on an evaluated nuclear data library, JENDL 3.2.

Examples of the effects of MAs and ZrH_2 (as the moderator) doped into the blanket on the growth of ^{238}Pu during burnup are shown in Figs 12 and 13, respectively. Without doping with MAs, super-weapon grade plutonium is produced in the FBR blankets. However, in a blanket with MA doping, ^{238}Pu builds up and increases even in the initial stage of burnup, as shown in Fig. 12 [12]. As shown in Fig. 13 [12], the fraction of ^{238}Pu is increased by increasing the content of ZrH_2.

The doping effects of MAs on the proliferation resistance of plutonium in FBR blankets are summarized in Fig. 14 [12], in comparison with the traditional suitability as an explosive of a plutonium mixture based on a ^{240}Pu content [16], with Kessler's proposal based on ^{238}Pu [17] and with IAEA guidelines that identify plutonium with a ^{238}Pu fraction of more than 80% as an

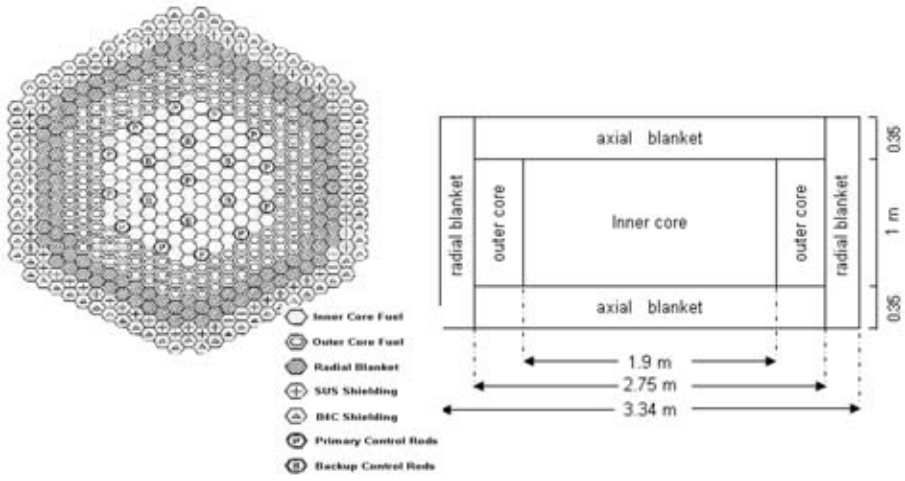


FIG. 11. Core layout and geometry of a medium size FBR [12].

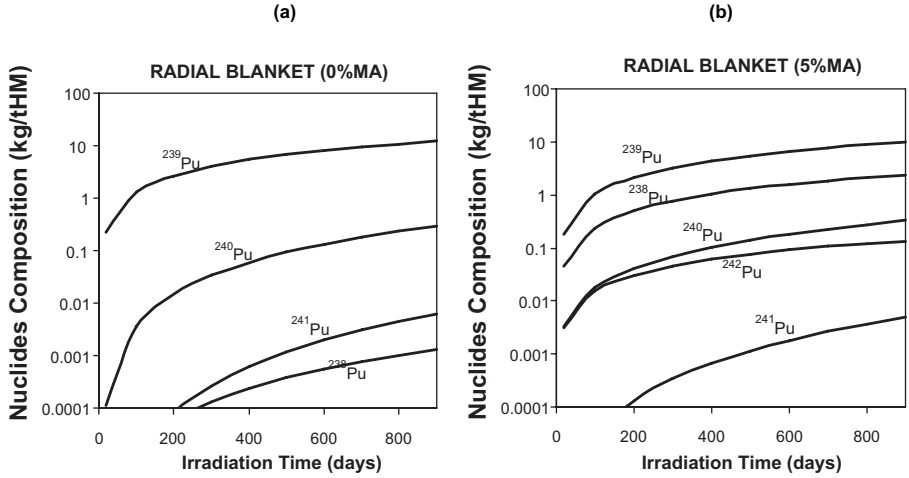


FIG. 12. Accumulation of plutonium (a) without MAs and (b) with MAs (5%) in radial blankets [12].

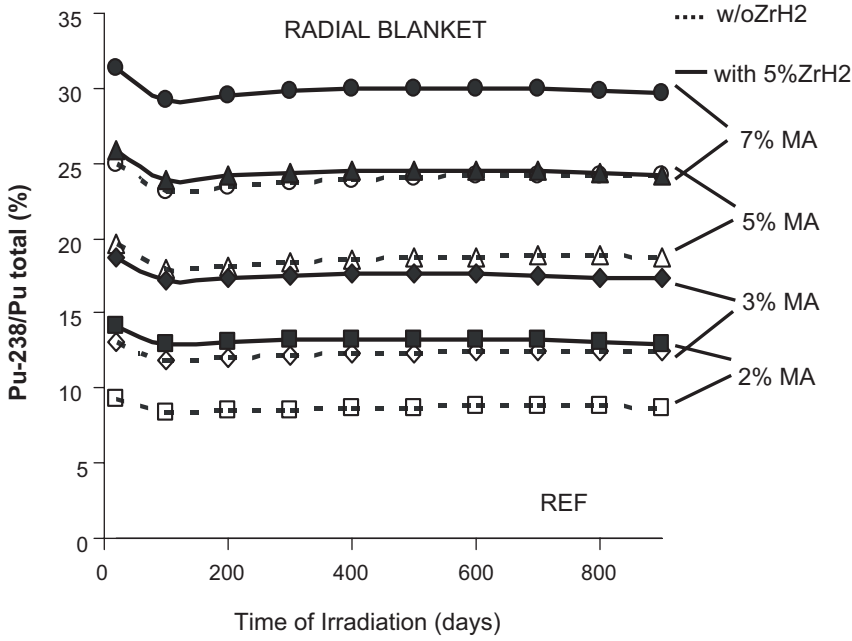
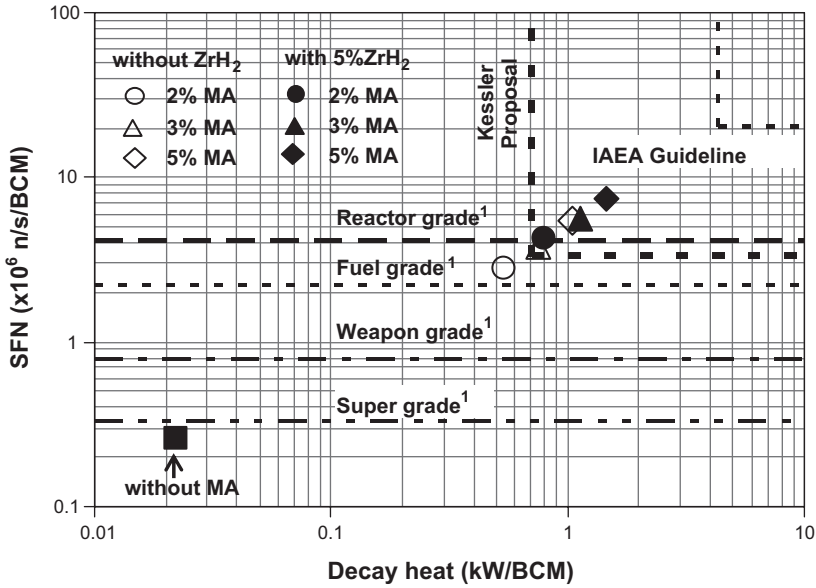


FIG. 13. Effect of ZrH_2 on ^{238}Pu accumulation in radial blankets [12].



[1] Adapted from B. Pellaud, "Proliferation aspects of plutonium recycling", J. Nucl. Mater. Management **31** 1 (2002) (Reactor grade: 30% ^{240}Pu + 70% ^{239}Pu; Fuel grade: 18% ^{240}Pu + 82% ^{239}Pu; Weapons grade: 7% ^{240}Pu + 93% ^{239}Pu; Super-grade: 3% ^{240}Pu + 97% ^{239}Pu).

FIG. 14. Summary of doping effects of MAs on proliferation resistance of plutonium in an FBR blanket [12].

exemption from safeguards [18]. In Fig. 15 [12], the PPP ratio is defined as the ratio of the protected plutonium (>12% ^{238}Pu) production in the blanket to the plutonium consumption in the active cores in FBRs. Figure 15 [12] shows that a doping of 2% MAs in the blanket is enough for protected plutonium breeding with ZrH_2.

3. DENATURING OF PLUTONIUM IN LWRS

Neptunium-237 and ^{241}Am, dominant nuclides of MAs in spent fuel from current LWRs, have rather large capture cross-sections in LWRs and are mainly transmuted to ^{238}Pu, which provides the essential protective effect of plutonium due to its large decay heat and neutron source from spontaneous fissions [13–15]. If plutonium is recycled together with MAs and incinerated within the environment of LWRs, there are possibilities to denature plutonium by ^{238}Pu

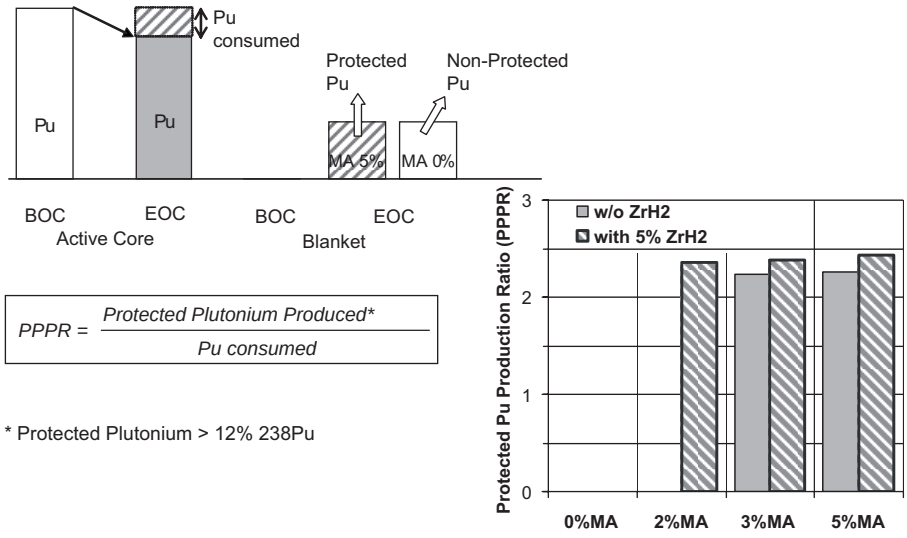


FIG. 15. Protected plutonium production ratio in FBRs [12].

through transmutation of MAs, and to enhance the intrinsic feature of proliferation resistance over that in the current MOX fuel cycle, as shown in Fig. 16.

In the present study, the incineration of three types of fuel have been compared, MOX fuel (U_{1-x-y}, Pu_x, MA_y, O_2), uranium-free IMF based on the zirconium matrix [20] (Pu_x, MA_y, Zr_{1-x-y}, $O_{2(x+y)}$) and thorium (ThO_2) fuel, with a small amount of depleted or natural uranium to avoid the production of pure ^{233}U from ^{232}Th. Uranium, plutonium and MAs are recovered from the conventional LWR spent fuel. The composition of plutonium and MAs in each fuel and the moderator to fuel ratio are set as parameters.

To estimate the enhancement of intrinsic features of plutonium proliferation resistance by incineration in LWRs, the fuel depletion and the criticality calculations were performed in a unit pin cell model with the help of the SRAC (VER.2002) code system. The PIJ, a collision probability method, of the SRAC (VER.2002), coupled with the 107-group cross-section library generated from JENDL3.3 were applied to the calculation of producing effective cross-sections. The cell burnup calculation of the SRAC (VER.2002) is used in 107 group depletion computation to determine the fuel composition for the next run. The reactor core safety parameters such as the Doppler coefficient (DC), moderator temperature coefficient (MTC) and void coefficient (VC) have also been investigated. Since all the evaluations were investigated with the infinite cell calculation, the neutron leakage from the reactors has not been

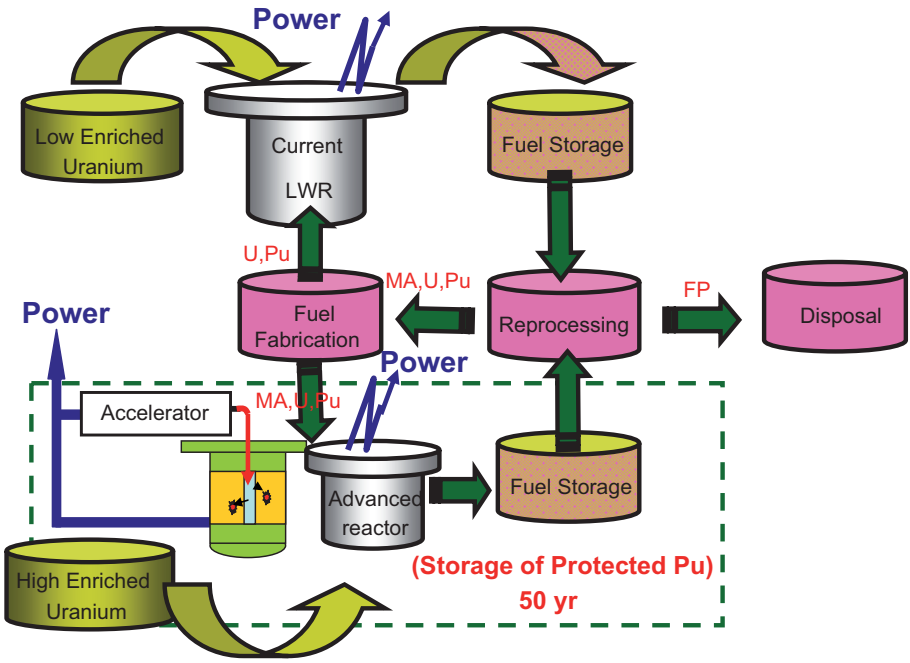


FIG. 16. Advanced concept of fuel cycle with PPP technology.

taken into account. Therefore, the VCs are always overestimated compared with the values in actual LWRs. The BCM of plutonium in the fuel at each step of burnup was obtained in a spherical geometry by the MCNP-4C(03) code with the nuclear data FSXLIBJ33 library based on JENDL3.3.

The results of a burnup calculation show that MAs have the attractive potential to reduce the initial excess reactivity and to be well transmuted to ^{238}Pu in any type of fuel. With the same initial load of plutonium and MAs in the MOX and IMF types of fuel, it is found that almost the same amounts of ^{238}Pu are accumulated, respectively. On the other hand, as shown in Fig. 17 [15], the amount of ^{239}Pu decreases more rapidly in uranium-free IMF fuel compared with MOX fuel due to the lack of ^{238}U, which is a fertile product of ^{239}Pu. Because of these characteristics, the decay heat and SFN rate from the BCM of plutonium at the end of incineration are increased. In addition, the intrinsic proliferation-resistance features of plutonium in IMF and ThO_2 fuels with MAs are much more enhanced than those of MOX with MAs, as shown in Fig. 18 [15], and in comparison with the traditional suitability for an explosive of a plutonium mixture based on a ^{240}Pu content [16], with Kessler's proposal based on ^{238}Pu [17] and with the IAEA guidelines, which identify plutonium with a ^{238}Pu fraction of more than 80% as being exempt from safeguards [17].

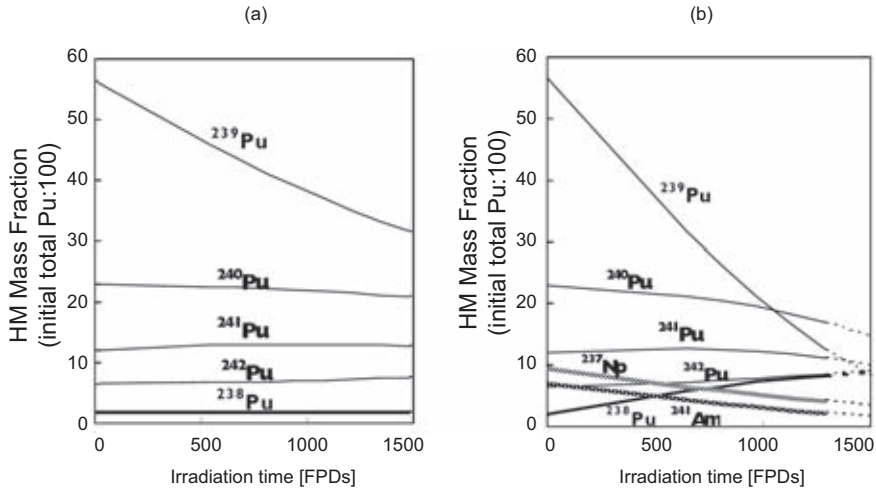
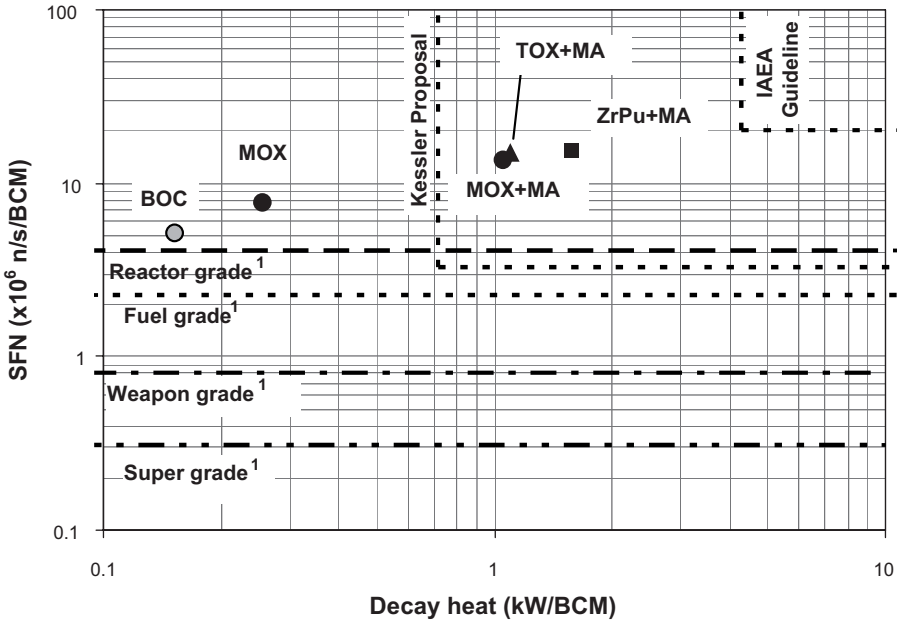


FIG. 17. Comparison of denaturing of plutonium between (a) MOX fuel (Pu 10%) and (b) uranium-free IMF fuel ($Zr(Pu + MA)O_2$) [15].

4. CONCLUSIONS

Protected plutonium production has been proposed to increase the proliferation resistance of plutonium by the transmutation of MAs. The PPP studies focus on the transmutation of MAs such as ^{237}Np and ^{241}Am with large neutron capture cross-sections in the thermal spectrum to increase the fraction of ^{238}Pu, which is a high spontaneous fission neutron source, to reduce the quality of the nuclear explosive. In addition, the high decay heat of ^{238}Pu makes the processes of nuclear weapon manufacture and maintenance technologically difficult. A small amount of doping uranium fuel with MAs is enough to increase the fraction of ^{238}Pu up to the practically unusable level for nuclear explosives purposes even at the beginning of the irradiation with negative void and Doppler coefficients in LWRs. The PPP technology has also been applied for protected plutonium breeding in FBRs. The PPP technology has been extended to the studies on denaturing of reactor grade plutonium in LWRs.

The PPP technology makes the fuel cycle more flexible with respect to fuel storage for future energy crises because of the enhanced proliferation resistance of plutonium. Instead of geological disposal or just their burning through fission, MAs can be used effectively as burnable and fertile materials to increase the proliferation resistance of plutonium. The PPP technology will



BOC: 3.5w/oEU, 40GWd/tHM, 5-year-cooling
MOX: $(U_{0.9}, Pu_{0.1})O_2$, V_m/V_f=1.84, 1340FPDs (40GWd/t)
MOX+MA: $(U_{0.88},Pu_{0.1},MA_{0.02})O_2$, V_m/V_f=4.0, 1613FPDs (55GWd/t)
TOX+MA: $(Th\text{-}U_{0.88},Pu_{0.1},MA_{0.02})O_2$, V_m/V_f =3.0, 1100FPDs (43.4GWd/t
Zr-Pu+MA : $Zr_{0.928}\text{-}(Pu_{0.06},MA_{0.012})O_2$, V_m/V_f =1.84, 1290FPDs

[1] Adapted from B. Pellaud, "Proliferation aspects of plutonium recycling", J. Nucl. Mater. Management **31** 1 (2002) (Reactor grade: 30% ^{240}Pu + 70% ^{239}Pu; Fuel grade: 18% ^{240}Pu + 82% ^{239}Pu; Weapons grade: 7% ^{240}Pu + 93% ^{239}Pu; Super-grade: 3% ^{240}Pu + 97% ^{239}Pu).

FIG. 18. Summary of doping effects of MAs on plutonium denaturing in LWRs [15].

open new markets worldwide for nuclear reactors with enhanced proliferation resistance in the future.

The demonstration of PPP mechanisms in the Advanced Test Reactor at INL, which starts in September of 2005, is an event of great historical significance in the second stage of the Atoms for Peace programme.

ACKNOWLEDGEMENTS

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PRESENT AND FUTURE SAFEGUARDS

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Abstract

The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) is the basis of world commitments to prevent the spread of nuclear weapons. States that are party to the NPT have committed themselves to preventing proliferation of nuclear weapons, pursuing nuclear disarmament and promoting peaceful uses of nuclear energy. The NPT recognizes five nuclear weapon States as parties to the NPT, and there are more than 180 non-nuclear-weapons States (NNWSs) party to the NPT, who have pledged not to develop or acquire nuclear weapons. The NPT authorizes the IAEA to verify compliance. The paper covers the features of the IAEA safeguards system, its history and major developments, the information flow on which it is based, and the methods used for verification. It then covers the process for evaluating information and for drawing conclusions regarding compliance or non-compliance with safeguards undertakings. The future of safeguards is discussed, covering: current and near future developments in implementation methods, particularly integrated safeguards; current worldwide forums for developing and evaluating future designs of nuclear energy systems and how safeguards are involved; and how future designs will include proliferation resistance and safeguardability concepts from the earliest stages. The paper includes the application of safeguards to special areas such as the minor actinides Np and Am, mixed oxide and breeder designs, and research areas such as accelerator driven systems and the ^{233}U–Th cycle. Safeguards in the weapon States and non-NPT States are also covered.

1. SAFEGUARDS ROLE — BASED ON THE NON-PROLIFERATION TREATY

The Non-Proliferation Treaty (NPT), the 1968 Treaty on the Non-Proliferation of Nuclear Weapons, is the basis of world commitments to prevent the spread of nuclear weapons. States party to the NPT and similar agreements have committed themselves to three objectives:

(1) Preventing the proliferation of nuclear weapons;
(2) Pursuing nuclear disarmament;
(3) Promoting the peaceful uses of nuclear energy.

The NPT recognizes five nuclear weapon States as party to the NPT, namely China, France, the Russian Federation, the United Kingdom and the United States of America. There are more than 180 non-nuclear weapon States (NNWSs) party to the NPT, and these have pledged not to develop or acquire nuclear weapons. With all except a few of the world's countries as parties to the treaty, the NPT is by far the most widely adhered-to legal agreement in the area, or in the history, of disarmament and non-proliferation.

The NPT authorizes the IAEA as the agency charged with verifying and reporting on the compliance of each State that is party to these commitments. Under the NPT, the NNWS parties are required to submit all nuclear material and nuclear activities to IAEA safeguards, and to conclude a safeguards agreement with the IAEA. The IAEA reports through the annual Safeguards Implementation Report (SIR). This is presented to the Board of Governors in the first half of each year, covering the previous year.

2. THE PURPOSE OF SAFEGUARDS

Safeguards is the set of activities the IAEA uses to verify whether a State is fulfilling its international undertakings not to use nuclear materials, activities or programmes for nuclear weapons purposes. Thus the IAEA's safeguards system consists of:

(a) Confidence building measures;
(b) An early warning mechanism;
(c) The trigger that sets in motion responses by the international community if needed.

IAEA safeguards provide credible assurances that:

(1) Nuclear material and activities are not 'diverted' or misused to develop or produce nuclear weapons;
(2) No material or activities remain undeclared that are required to be declared under the safeguards agreements.

The safeguards system is based on an assessment of a State's declarations to the IAEA of their nuclear material, facilities and activities: of the *correctness*, and for non-weapon States the *completeness*. To date, some 150 States have entered into such agreements with the IAEA, submitting nuclear materials and facilities to the scrutiny of the IAEA's safeguards inspectors.

Over the past decade, IAEA safeguards have been strengthened in key areas, an aspect that will be covered in more detail below. The measures aim to:

(a) Strengthen and build confidence that States are abiding by their international commitments;
(b) Increase the likelihood of detecting a clandestine nuclear weapons programme.

3. NUCLEAR MATERIAL AND ACTIVITIES SAFEGUARDED BY THE IAEA

In States under safeguards, the IAEA takes account of all nuclear material, especially that defined as 'source and special fissionable material', but the monitoring and verification activities are naturally concentrated on the nuclear materials that are the most crucial and relevant to nuclear weapons. These include the well-known fissile isotopes of uranium and plutonium, and any material containing these. Safeguards activities are applied routinely at over 900 facilities in more than 70 countries. Some 250 IAEA inspectors devote more than 21 000 days 'in the field' to verifying hundreds of tonnes of special fissionable material. Each year, the IAEA:

(a) Receives about 13 000 nuclear material accounting reports from States;
(b) Performs about 2400 inspections at around 650 nuclear facilities;
(c) Applies about 25 000 seals;
(d) Analyses about 700 material samples and around 500 environmental samples;
(e) Uses about 900 non-destructive analysis (NDA) systems;
(f) Draws conclusions for every State with a safeguards agreement in force.

4. SOURCES OF INFORMATION

There are four sources of information available to the IAEA:

(1) Information provided by a State, in formal declarations and in voluntary reporting;
(2) Information produced by the IAEA, from verification activities in the field, i.e. inspections and visits, and from its internal databases;
(3) Information from open sources, Internet searches, publications and news media, and commercial satellite imagery;

(4) Information from third parties: intelligence from other States and organizations.

These can be combined and cross-checked for consistency in various ways.

5. COMPREHENSIVE SAFEGUARDS AGREEMENTS

When safeguards agreements became a requirement under the NPT, the IAEA developed a standard for them, which became the comprehensive safeguards agreement (CSA) and is codified in INFCIRC/153 (Corr.) [1]. Each State brings into force its own CSA, but they are nearly all based on INFCIRC/153 text, paragraph numbers, etc. This agreement is suitable for application to both simple nuclear activities and to complex nuclear fuel cycles, and covers reactors as well as the conversion, enrichment, fabrication and reprocessing plants that produce and process reactor fuel. Some 140 States have such agreements at present.

Under INFCIRC/153, a State has to:

(a) Report all nuclear material shipments, receipts and holdings to the IAEA;
(b) Provide design information on facilities to the IAEA;
(c) Provide access for IAEA inspectors to verify the correctness of the reports and the design information.

Under INFCIRC/153, the conclusion that the IAEA can draw is limited to verifying the correctness of a State's declaration, and thus is limited to a narrow conclusion that there has been no misuse of a safeguarded facility to introduce, produce or process undeclared nuclear material, and, of course, no diversion of declared nuclear material.

6. STRENGTHENED SAFEGUARDS AND THE ADDITIONAL PROTOCOL

Under a safeguards system using INFCIRC/153 alone, the IAEA's capability to detect undeclared nuclear activities is limited. The IAEA inspections focus on declared nuclear material, and inspector access is normally limited to 'strategic points' in declared facilities.

Changes to safeguards were slow and gradual before the 1990s. However, events in the early 1990s dramatically changed attitudes towards IAEA safeguards and the NPT:

(a) The collapse of the former USSR and the end of the cold war;
(b) The discovery of Iraq's clandestine nuclear weapons programme;
(c) Difficulties in verifying the initial reports from the Democratic People's Republic of Korea, and its subsequent withdrawal from the NPT;
(d) The revealing of the weapons programme of South Africa.

The 1995 NPT review conference extended the NPT indefinitely, with a five year review cycle. It was clear that Member States wanted IAEA safeguards to be stronger and more cost effective. It was also clear that the limited information and access rights under INFCIRC/153 were seriously constraining the IAEA in detecting nuclear material and activities that a State did not declare, i.e. while the declaration of a State could be quite correct regarding its declared nuclear material and facilities, there was still the possibility that the declaration was not complete.

This prompted the IAEA to improve its ability to fully exercise its verification and detection capabilities. This was done in two ways –

(1) Some improvements were achieved by more fully exercising those powers granted to the IAEA under the existing safeguards agreements, which *strengthened* the implementation of INFCIRC/153.
(2) Further improvements required *new* legal authority, which was obtained through the well known INFCIRC/540 (Corr.) [2], the Additional Protocol to a State's safeguards agreement. This is the key to the strengthened safeguards system.

The combination of reinforced and new IAEA mechanisms for verification is known as strengthened safeguards, and has greatly improved the IAEA's ability to detect clandestine or undeclared nuclear activities. The IAEA now has the right to receive and use more information, and expanded rights of access. The total safeguards picture, including both the CSA and the Additional Protocol, now includes the following:

(a) States provide information on:
    (i) All aspects of its nuclear activities and material, including operating records;
    (ii) Capacities and usage of mines, mills, conversion plants, etc.;
    (iii) Future plans up to ten years ahead;

(iv) Design information about facilities and sites that handle safeguarded material;
(v) Plans for changes or for new facilities;
(vi) Descriptions of on-site buildings;
(vii) Facilities closed down/decommissioned before safeguards were in force;
(viii) Nuclear related research and development even when no nuclear material is used;
(ix) Manufacture and export of sensitive nuclear-related equipment or technologies even if it uses non-nuclear material;
(x) Locations of exempted materials and holdings of 'pre-safeguards' source material;
(xi) Responses to IAEA questions arising from a review of declarations.

(b) The IAEA can:
(i) Collect environmental samples in facilities and at locations where inspectors have access, and in other locations when deemed necessary;
(ii) Analyse the samples at the IAEA Clean Laboratory or another of the 12 certified laboratories worldwide;
(iii) Use unattended or remote monitoring of surveillance data from facilities, and transmit the data from unattended instruments.

(c) Inspection and access can now include:
(i) Inspector access to all buildings on a nuclear site;
(ii) Access to any location in a State if there is a 'question or inconsistency';
(iii) Access to locations of exempted and 'pre-safeguards' materials;
(iv) Access to verify this design information at any part of the facility life cycle, including decommissioning;
(v) Access rights to the manufacturing locations of sensitive nuclear related technologies;
(vi) Expanded use of unannounced or short notice inspections as well as scheduled routine inspections.

(d) Administrative procedures have been improved, including simplified access for IAEA inspectors through the issue of multiple entry visas for State entry and exit.

(e) Relevant information is obtained from various voluntary programmes:
(i) A voluntary reporting scheme (VRS), covering imports and exports of nuclear material; exports of sensitive equipment and non-nuclear material; and the alternative nuclear materials (ANMs) monitoring scheme, covering a State's production and holdings of separated neptunium and americium;

(ii) The Illicit Trafficking Database (ITDB), covering trafficking and smuggling incidents into and out of a State;
(iii) Additional Protocol 'declarations', covering source material production; and comprehensive reporting of source material exports and imports.

Much of the improvement in strengthened safeguards depends on the amount and quality of the information received from a State: i.e. from the State system of accounting and control (SSAC). These discharge the State's responsibility, making reports and providing the information needed for safeguards to be effective. The IAEA is in the process of setting up programmes to promote closer cooperation between itself and State or regional SSACs, since it is only to be expected that knowledge of the requirements and how to fulfil them is somewhat uneven. This includes provision of enhanced training for safeguards personnel in Member States, as well as for IAEA safeguards staff.

Now with the additional legal authority from INFCIRC/540, with wider access, broader information and better use of technology, the IAEA's capability to detect and deter undeclared nuclear material or activities is significantly improved. This thus enables the IAEA to provide assurance about both declared and possible undeclared material and activities: to verify for both correctness and completeness. The IAEA is now able to draw a broader conclusion about both the non-diversion of declared nuclear material and the absence of undeclared nuclear material and activities.

7. THE STATE EVALUATION PROCESS

In order to draw safeguards conclusions about States, the IAEA has established a formal State evaluation process. This is an ongoing process, documented via a State Evaluation Report (SER); for most States, the SER is produced annually. It ensures that all the information concerning a State's nuclear activities is taken into proper consideration in drawing the safeguards conclusion. The process is based on a State as a whole, not on the individual facilities. The IAEA:

(a) Reviews a substantial list of items, and produces the SER, to allow a clear and focused evaluation and discussion. The SER discusses a State's history, agreements with the IAEA, industrial infrastructure, any plausible acquisition paths for weapons or weapons-usable material, and weaponization capabilities.

(b) Draws a formal conclusion about each State, and publishes these conclusions in the Safeguards Implementation Report (SIR), annually in the first half of each year.

8. THE FUTURE OF SAFEGUARDS

8.1. Integrated safeguards

Once the broader conclusion has been drawn for a State, it becomes possible to change the way safeguards are implemented at nuclear facilities in a State. This 'integrated safeguards' (IS) is the optimum combination of all safeguards measures available to the IAEA under both comprehensive safeguards agreements and Additional Protocols. The approach takes into account a State evaluation, a State's nuclear fuel cycle, any interaction between its facilities, and other State-specific features such as the IAEA's ability to carry out unannounced inspections effectively, or the technical effectiveness of the SSAC. It is intended to:

(a) Achieve the maximum effectiveness and efficiency within the available resources in fulfilling the IAEA's rights and obligations;
(b) Make effective use of all the measures available under both the CSA and the Additional Protocol;
(c) Where appropriate, *make some reductions and find economies* in traditional safeguards at facilities;
(d) Maintain *cost neutrality.*

The stages are:

(1) The objectives specific to that State which support drawing of the broader conclusion are defined and specified in its State level IS approach (ISA).
(2) The safeguards measures to meet the objectives are specified in the ISA.
(3) The activities planned for each year form the annual implementation plan (AIP), which reflects the objectives and the measures.
(4) The IAEA and the State work out the detailed arrangements for the ISA and the AIP.
(5) The IAEA management approves the safeguards approach and implementation begins.

At present the IS approaches are in various stages of development. The model IS approaches for different facility types at various stages of development so far are:

— Light water reactors (including mixed oxide (MOX) fuel);
— On-load reactors;
— Spent fuel storage facilities;
— Conversion and fuel fabrication facilities;
— Transfers of spent fuel to dry storage;
— Storage facilities (other than spent fuel);
— Locations outside facilities;
— Enrichment facilities.

These model approaches are not binding, but are intended to be adapted to State-specific conditions.

Guidelines for specific areas of safeguards activity are also under preparation, including

(a) Use of unannounced and short notice inspections;
(b) A broader range of cooperation with State authorities and systems;
(c) Preparation of AIPs;
(d) Evaluation of IS results and any resulting review/revision of AIPs.

The IAEA must retain the ability to draw a conclusion, and thereby to continue to provide credible assurance of the absence of undeclared nuclear material and activities in a State. This requires continuous review and evaluation of information, continuing to take all actions necessary to resolve questions and inconsistencies, and conducting inspection visits as necessary.

One example of the methods in this more modern approach is the use of 'mailbox' systems, whereby operators report regularly to the IAEA, backed up by agreements on the use of short notice random inspections. This is applicable, for instance, to low enriched uranium bulk facilities such as fuel fabrication plants. The purpose is to verify 100% of flow components, including receipts and shipments, with a reasonable consumption of IAEA resources. Under the mailbox system, operators declare their transactions regularly, for example daily, to a web site at the IAEA. Secure communications and modern encryption technology make this possible, and the IAEA can then inspect and verify this information as they consider necessary.

8.2. Proliferation resistance in designs

Other papers in these proceedings cover the predictions for the growth of the energy supply and particularly of nuclear energy in the next few decades. We in the nuclear business, particularly the IAEA and safeguards, would be wise to be ready, because the demand is clearly predictable. Being ready involves improved designs to maximize the benefits and minimize the concerns associated with economic competitiveness, resources and waste management, safety, environmental impacts, proliferation resistance and physical protection. Plants must be safer, cheaper, quicker to build, more proliferation resistant and more efficient to run. All of these are possible.

The two major design and development initiatives in progress worldwide are the well known International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and the Generation IV International Forum (GIF). INPRO is an IAEA initiative, while GIF is an industry based initiative, and mainly, but not exclusively, of US origin.

The INPRO and GIF initiatives recognize that designs for new nuclear facilities must take safeguards into account at the earliest design stage, incorporating proliferation resistance features that reduce the system's attractiveness for diversion, or for misuse for weapons programmes. For instance:

(a) Spent fuel handling can be made completely inaccessible from the outside;
(b) The type of reactor and its fuel cycle and physics can be chosen so that there is no material that is 'attractive' for proliferation in the final product waste;
(c) Extralong refuelling intervals can be built in to prevent access to the fuel. Design characteristics or features, which either impede or support safeguards implementation, are aspects of a term called *safeguardability*, which is used in the optimization of proliferation resistance at the design stage of new systems.

Designing a nuclear energy system will start with identification of the major features:

(a) Major design areas for evaluation of safeguardability/proliferation resistance, such as source material and fuel production, use, transport, storage and disposal;
(b) Characteristics that can be used in proliferation resistance considerations: isotope content, chemical form, bulk and mass, radiation fields, heat

generation and spontaneous neutron generation rate (for detection purposes).

Safeguardability/proliferation resistance includes internal or system features such as:

(a) Design features that prevent/inhibit diversion:
  (i) Designs that limit access to nuclear material;
  (ii) Effectiveness of prevention of diversion or misuse;
  (iii) Time required to divert/misuse and convert to weapons material.
(b) Features that prevent/inhibit undeclared production of direct use material:
  (i) Complexity/time needed to:
    — Change civilian systems to weapons use;
    — Modify civilian systems to produce undeclared nuclear material;
  (ii) Skills, knowledge and expertise required to divert, produce and convert nuclear material to weapons use.
(c) Features that assist safeguards verification:
  (i) Ease of detecting diversion;
  (ii) Ease of measuring material flows and inventories.

Safeguardability/proliferation resistance also includes external features such as:

(a) A State's treaties, commitments, obligations and policies, such as:
  (i) Adherence with the NPT and and/or the nuclear-weapon-free zone (NWFZ) treaty;
  (ii) Adherence and compliance with the CSA and Additional Protocol.
(b) A State's agreements with other exporting and/or importing States:
  (i) That energy systems will only be used for civilian purposes;
  (ii) About export and re-export control policies;
  (iii) About supply and return of nuclear material.
(c) Commercial, legal and institutional controls on access to nuclear fuel cycle and energy systems:
  (i) Multinational ownership, management or control of nuclear fuel cycle or energy facilities;
  (ii) Agreements about supply and return of nuclear material.
(d) Application of IAEA safeguards to systems and transactions:
  (i) Declarations and reporting under the safeguards agreements and protocols;

(ii) Verification and inspection activities using all available technologies capable of detecting diversion, misuse or undeclared production;
(iii) Use of open source and third party information;
(iv) Cooperation with State/regional SSACs.

(e) Effective legal and institutional arrangements to address violations.

9. AREAS OF SPECIAL INTEREST IN PROLIFERATION RESISTANCE

These ideas, about proliferation resistance and safeguardability, apply to all aspects of the nuclear fuel cycle, not just to the power plants themselves. The same approach is used to focus on proliferation resistance in the design work for specialized problems, of which some examples are:

(1) The minor actinides — neptunium and americium;
(2) MOX fuel;
(3) ^{233}U — the Th cycle;
(4) Accelerator driven systems.

9.1. The minor actinides: Neptunium and americium

The voluntary reporting scheme for reporting on production and holdings of ANMs was mentioned earlier. The extent to which neptunium or americium are of real safeguards significance is still being debated both internally and with Member States. The characteristics and features of both are well known, and, if it becomes necessary, the IAEA would use these particular nuclear properties for detection and indication of these elements as appropriate.

9.2. Mixed oxide fuel and fast breeder systems

Fast breeder systems with MOX fuel involve plutonium recycling — there are no changes in the principal proliferation resistance and safeguards areas: the accounting of nuclear material is unchanged, as is containment and surveillance. The system saves enrichment work by replacing ^{235}U with plutonium, which is produced from the ^{238}U in the original fuel and recovered from the spent fuel by chemical reprocessing. The major differences are that:

(a) The need for safeguards extends to the reprocessing plant and MOX fuel fabrication.

(b) The safeguards requirements for these facilities are more stringent because of the safeguards sensitivity of separated plutonium.

Safeguards in fuel cycle plants, i.e. enrichment and reprocessing plants, are based on the same criteria, but involve bulk handing equipment and on-line analysis, measurement and read-outs, the same technology as in many kinds of processing plants. The plutonium is measured by both destructive and non-destructive analysis methods, and the individual elements are calculated for accounting purposes.

9.3. The ^{233}U–Th cycle

There are no power plants yet operational using this system, although the physics is well understood, and indeed the potential for future power production is great. There are no changes in the principal proliferation resistance areas: the design would be very similar to that of ^{235}U plants, and nuclear material accounting would be unchanged, as would containment and surveillance. The individual elements and isotopes are calculated for accounting purposes, but the objectives and methods of safeguards remain unchanged.

9.4. Accelerator driven systems

Many States are working in this area, investigating such fields as transmutation of actinides (actinide burning), to reduce or remove the HLW problem; and combining subcritical assemblies with accelerators to make intrinsically safe power production reactors. Both of these present new and different safeguards challenges — of great interest to IAEA Safeguards Concepts and Planning. The safeguards approaches have not yet been worked out, and are still at the research and experimental stages, but the approach can be seen from the earlier discussion of proliferation and safeguardability. The differences will be in the ways the concepts apply to the new designs, and the possibilities for effective and efficient safeguards application.

10. EXCEPTIONS TO ALL THE RULES: THE WEAPON STATES AND THE NON-NPT STATES

Despite the objective of safeguards being the prevention of proliferation of nuclear weapons to the non-weapon States, from the earliest days safeguards

has also been applied to varying extents both to the recognized nuclear weapon States and to States that are not parties to the NPT.

The States that are not party to the NPT have one kind of agreement, and the five recognized weapon States have another.

The first type of safeguards agreement was known as INFCIRC/66, and it is still in use by Israel, India and Pakistan, to cover nuclear materials and/or facilities supplied by other countries. INFCIRC/66 allows the IAEA to apply safeguards only to the specific nuclear materials, facilities or equipment agreed with the individual State. It thus only allows a conclusion about those specifically declared nuclear materials and facilities and nothing else, even if the IAEA is aware of other nuclear activities in that State.

Safeguards are also applied in the five NPT nuclear weapon States. All of these have in force voluntary offer agreements (VOAs); these allow IAEA safeguards to be conducted at specific sites and for particular purposes. The weapon States offer a chosen selection of their nuclear material or facilities, and from this the IAEA selects where it will apply safeguards. All of the weapon States have civilian nuclear programmes as well as military ones; and all of them have industries conducting import and export businesses in the nuclear field.

The voluntary offer safeguards agreements generally follow the format of INFCIRC/153 type agreements, but they vary in scope. None of the VOAs is a complete, State wide, agreement: for instance China has safeguards on one PWR, one centrifuge enrichment plant supplied by the Russian Federation, which required safeguards to be applied, and an experimental high temperature gas cooled reactor whose fuel is supplied by Japan.

Naturally, weapon States or any other States can and do voluntarily provide information, as already discussed.

In all of these areas, exceptional and specialized as they may be, the safeguards objectives remain unchanged, namely verification of a State's declaration; and the safeguards conclusion drawn will still depend on the information provided and the extent of the IAEA's verification authority and capability.

11. SUMMARY AND CONCLUSIONS

Ultimately, the strength of the IAEA safeguards system depends on three factors:

(1) How much we know: how much the IAEA is aware of the nature and locations of States' nuclear materials and activities;

(2) What we can find out: how much physical access the IAEA has, in order to provide independent verification of the peaceful intent of a State's nuclear programme;
(3) The will of the United Nations and the international community to take action against States that are not complying with their safeguards commitments to the IAEA.

By entrusting an impartial inspectorate with the task of verifying the peaceful use of nuclear applications, the international community has taken an important step in the direction of peace and international security. This is based upon the readiness of States to submit to inspection of their nuclear activities, to demonstrate their transparency and, thus, to mutually assure their peaceful nature. By bringing the Additional Protocol into force and using it, by dealing promptly and responsibly with cases of safeguards violations, and by providing the IAEA with the resources necessary to carry out its function, States demonstrate the political will to demand a peaceful world without the constant threat of nuclear weapons, and at the same time the right to the peaceful use of nuclear energy for power production, medicine, agriculture, drinking water and all its other potential benefits.

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The management of fissile material will play an increasingly important role in the nuclear fuel cycle. In view of this, the IAEA organized a technical meeting on fissile material management strategies for sustainable nuclear energy in order to provide Member States with essential information for their policy making and strategy planning requirements. The meeting was held in Vienna from 12 to 15 September 2005. The purposes of the meeting were to: (a) identify fissile material management strategies for different nuclear fuel cycle options; (b) clarify the issues and challenges existing in fissile material management; (c) seek possible solutions, focusing, in particular, on sustainability.

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