Generation-IV Roadmap

Report of the Fuel Cycle Crosscut Group

March 18, 2001

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Generation-IV Roadmap
Report of the Fuel Cycle Crosscut Group (FCCG)

March 18, 2002

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The scenarios were generated by Anton Moisseysev, Texas AM& Grad Student in Nuclear Engineering during a summer assignment at ANL where he created the simulation code: DYMOND.

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Generation-IV Fuel Cycle Crosscut Group Report

Executive Summary

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FCCG
Executive Summary

Background

The Charter of the Generation IV Roadmap Fuel Cycle Crosscut Group (FCCG) is to (1) examine the fuel cycle implications for alternative nuclear power scenarios in terms of Generation IV goals and (2) identify key fuel cycle issues associated with Generation IV goals. This included examination of "fuel resource inputs and waste outputs for the range of potential Generation IV fuel cycles, consistent with projected energy demand scenarios." This report summarizes the results of the studies.

The membership of the FCCG comprised 8 US members and 7 members from Generation IV International Forum (GIF) countries including members from the OECD-NEA, the IAEA and the European Commission observer organizations to the GIF. Members of the FCCG were, in general, drawn from the Technical Working Groups (TWG's) and the Evaluation Methodology Group (EMG) of the Generation IV Roadmap organization. Five one-day working meetings were held between February 2001, and August 2001 – three of them in conjunction with Generation IV TWG quarterly meetings.

The FCCG reviewed energy projections and selected the authoritative IIASA/WEC projections of 1998 as the basis for performing a selected set of 100 year nuclear energy futures scenarios. We reviewed the uranium ore resource projections of the OECD-NEA, IAEA, and Uranium Institute, the thorium ore resource projections from multiple sources, and investigated independent models for prediction of new ore discoveries vs. cost of supply. A survey of Generation IV concept submittals was made to define the scope of proposed fuel cycles and fuel compositions. Fuel cycle infrastructure status was reviewed, and an extensive review was undertaken of fuel cycle R&D programs underway worldwide. A documentation of the status of institutional aspects providing the enabling legal basis and boundary conditions for worldwide fuel cycle deployment was made.

On the basis of these reviews, studies and evaluations, the FCCG has produced a set of principal findings and has generated a set of top level recommendations for Generation IV fuel cycle crosscutting R&D.

The FCCG's principal findings are based on two primary sources. First are the results from dynamic scenario simulations of various potential nuclear futures – driven by the 100 year world energy demand projections (and nuclear's share) provided by the 1998 IIASA/WEC. These nuclear futures scenarios were organized by generic fuel cycle type (once-through, partial recycle, full fissile recycle, and full transuranic recycle) and were constrained only by physically-achievable mass flows and lag times of potential Generation IV power plant and

fuel cycle concepts. They modeled idealized transitions from current and near term deployments to Generation IV fuel cycles and power plants and potential symbiosis of mass flow exchanges among Generation IV power plant concept types. These scenarios provide cornerstone indicators for the Roadmap of physically-achievable performance against Generation IV goals.

The second principal input to the FCCG's findings derived from an extensive and deep review of the technical status of fuel cycle technologies deployed and under development worldwide, and an evaluation of the underlying rationale for the choices of research focus that drive these development programs. While the technical approaches vary, it was found that the worldwide underlying motivations are closely aligned to the goals articulated for Generation IV in the areas of Sustainability, Safety and Reliability, and Economics. The fuel cycle plays a primary role in meeting the three elements of the Generation IV sustainability goals.

The principles of sustainability include meeting society's needs for energy services while using the earth's resources in an efficient and environmentally friendly way. Nuclear fission converts uranium and thorium resources to energy with fission products as the essential waste. The net production of long-lived transuranium isotopes is a characteristic of the specific reactor types and fuel recycling steps used. The goals of Generation IV include reduced waste generation and more efficient use of ore resources along with making the nuclear fuel cycle the least attractive route to proliferation of nuclear armaments.

Today the cost of uranium and thorium is not a major contributor to the cost of nuclear energy, and resources do not constrain the expansion of nuclear power. Within several decades the costs of fuel materials may become more significant as lower-grade resources are used. However, repository capacity is an increasingly expensive and politically divisive constraint on growth of nuclear power. The use of fuel cycles and reactors that minimize repository requirements is essential to increased use of nuclear energy.

Principal Findings

A. Reversing Waste Buildup in a Growing Nuclear Economy

1. Closed fuel cycles have already demonstrated a significant reduction of the volume and long-term radiotoxicity of nuclear high-level waste through the reprocessing of 20,000 tonnes of spent LWR fuel to recycle the plutonium and uranium. Closed fuel cycles provide the opportunity to partition classes of nuclear waste and to manage each class in a separate waste form according to its individual characteristics. Advanced waste management strategies include transmutation of selected nuclides, cost effective decay heat management, flexible interim storage, and customized waste forms

for specific geologic repository environments. These strategies hold the promise to significantly reduce the long-lived radiotoxicity of the waste destined for geological repositories by at least an order of magnitude via recovery of virtually all the heavy long-lived radioactive elements. Such reductions, and the ability to optimally condition the residual wastes and manage heat loads, will permit far more efficient use of limited repository capacity and further enhance overall safety of the final disposal of radioactive wastes. An equivalent reduction in secondary waste arisings is also possible.

2. Advanced once-through cycles also have a potential to provide useful improvements in repository performance, although smaller than closed cycles. These improvements may be achieved primarily through the increase of thermal efficiency and high levels of fuel burnup .

B. Sustainable Use of Resources

3. Virgin uranium ore supplies are assured for several decades but will fall short of demand for low cost uranium by the middle of the 21st century. Timely renewal of exploration campaigns may further extend this virgin uranium supply but probably at a higher cost. However, more efficient use of raw materials is sustainable while reducing the environmental burdens and worker radiological exposures from mining and milling activities. More efficient use of ore can be achieved in three ways: by further extraction of fuel from existing stocks of depleted uranium, by more fuel-efficient reactors, or by recycle of existing stocks of discharged fuel. (Additionally downblending of stockpiles of highly enriched uranium declared to be in excess of security needs and the use of stored uranium inventories will be temporary sources of fuel for the near-term.)

4. Nuclear energy has the important market advantage that its fuel cycle contributes only about 20% to the overall cost of energy. This advantage provides remarkable flexibility for decoupling the strategies for meeting Generation IV economics and safety and reliability goals from the strategies for meeting sustainability goals. Wide ranges of fuel cycle approaches can be undertaken with only weak influence on economics. Power plant design strategies to meet economic and safety and reliability goals, while intimately tied to fuel design, are only loosely influenced by fuel cycle choices. This flexibility is further enhanced by a symbiotic mix of reactor types.

C. Transition To Sustainability

5. Energy use is expected to grow substantially in this century; consequently nuclear energy deployment is projected by authoritative studies to increase by as much as a factor of five or six by 2050. Over this period the nuclear deployment will require an evolving mix of reactor types (thermal and fast spectrum) and fuel cycles (once-through and recycle) to serve different market sectors. Symbiotic mixes of reactor types and fuel cycles are essential for economically supplying the required energy - while minimizing waste generation and assuring efficient use of resources, including limited repository capacity.

6. New reactor types with favorable neutron economy and a variety of fuel compositions will be required to create a symbiotic mix. These reactors can either produce fuel by breeding or consume nuclear waste by burning – providing the flexibility needed to accommodate economic and social changes. Such reactors, e.g. those employing some combination of fast neutron spectra, altered fertile (U-238, Th-232) feed, reduced neutron losses and/or an external neutron source, rely on fuel recycle to fulfill their function.

   Near term evolutionary modifications of current reactors are needed to fill expanding energy needs while at the same time employing limited repository space with high efficiency. Within the symbiotic mix envisioned for the future, these reactors need not necessarily rely on recycle, but further increases in fuel burnup and sufficient flexibility to be adapted to a future closed fuel cycle are desirable.

7. Technologies for co-recycle of minor actinides with plutonium and/or for provisions of some residual contamination of recycle fuel from fission products are under current development. By focusing on creating clean waste streams (containing only the fission products) these technologies can significantly reduce the quantities of long-lived radionuclides consigned to waste and provide an intrinsic barrier to weapons proliferation. Recycle of these advanced fuels requires the handling of highly radioactive materials in all fabrication activities. Such highly radioactive and fission product contaminated fuels have traditionally been destined for use in the favorable neutron balance of fast reactors. However, in the symbiotic mix of reactor types envisioned for the future, such fuels may be destined for thermal reactors as well. Generation IV R&D must determine the optimal implementation of these options and develop such enabling technologies as remote fabrication for thermal reactor fuel as well as fast reactor fuel.

D. Fuel Cycle Safety

8. The radiological exposure of workers in the overall fuel cycle includes contributions from mining/milling and from recycling facilities. These worker exposures are limited and are comparable to those from reactor operation. Developments to further reduce radiological impacts, for example by use of already-mined resources and by cost-effective remote refabrication, will become increasingly important as the scale of nuclear power increases.

9. The glass waste form generated by today's fuel cycles has been shown experimentally to be very durable in repository conditions. The long-term behavior of waste forms and their ability to confine potentially mobile radionuclides are important and discriminating issues for comparing different fuel cycle strategies.

E. Thorium

10. Thorium-fueled thermal reactors and uranium fast spectrum reactors may become an attractive option in the longer term because of the depletion of uranium resources. In addition, it is believed that the joint use of thorium and uranium cycles could lead to significant reductions of the long-term radiotoxicity of the ultimate waste because of greatly reduced production of transuranium actinides. In addition, the joint use of both cycles might enhance proliferation resistance through increased U-232 and Pu-238 content in recycle fuel feedstock.

The FCCG's Recommendations for Fuel Cycle Crosscutting R&D for Generation IV Systems

Results of the scenarios studied by the FCCG indicate that the full fissile recycle and especially the full transuranic recycle generic fuel cycles may be needed to achieve Generation IV sustainability goals SU-1 and SU-2. Once-through power plant concepts, eventually symbiotically fueled with fissile material fed back from full recycle fuel cycles, will also be part of an overall sustainable nuclear energy park. Partial recycle fuel cycles can function as a bridge from the current situation to the long term full recycle (sustainable) cycles. These observations, combined with the FCCG's conclusion regarding the likely technical achievability of full recycle (which is based on the current state of knowledge and ongoing R&D) have influenced the selection of Generation IV fuel cycle crosscutting R&D recommended below.

R&D targeted to "breakthrough" enabling technologies have been favored by the FCCG over those directed toward refined understanding and/or incremental improvements in existing technology. While this latter type of "incremental" R&D is essential and important for the continual improvement against Generation

IV economic, reliability and safety goals for the global nuclear energy park, it is the "breakthrough" fuel cycle enabling technologies which are needed to attain the Generation IV waste management and resource utilization sustainability goals for the fuel cycle; these are the special domain of the fuel cycle crosscut R&D recommendations.

1. Improvement of fuels and development of advanced fuels is important *no matter what fuel cycle is used*. The fuel assemblies comprise the essential interface between the nuclear power plant and the fuel cycle. Fuel assemblies that achieve high discharge burnup, low reactivity loss with burnup, low fabrication cost and which can operate at high coolant temperature for improved station thermal efficiency will impact directly on economics and sustainability goals and indirectly on safety goals. If the fuel is intended to be used in a recycle-based fuel cycle, it must be designed with ease of recycle in mind; if it is intended for once-through, it must be designed for very high discharge burnup and for extreme robustness over geologic time scales in a repository environment. Finally, because ultimately the source of fissile will be fed back from full plutonium, uranium 233, or full transuranic recycle sources, the fuel must be remotely fabricable.

   Development of new fuel types and fuel compositions containing increased plutonium and minor actinide fractions is recognized to be a significant challenge. Fuels development campaigns typically require ten to fifteen years of significant development cost – requiring both in pile and out of pile testing (including safety tests) before the fuel is ready for deployment. While the cost is not to be understated, neither is the payoff; the fuel is the essential link between the fuel cycle (controlling resource and waste performance) and the reactor power plant (controlling cost, reliability, and safety) against Generation IV goals. Uranium oxide fuel is highly refined and dominates the current once-through commercial fuel cycle and is qualified for use in fast neutron spectrum reactors. Nonetheless new Generation IV reactor concepts and fuel cycles must be afforded opportunities to explore and exploit the potential of additional fuel types, and the R&D cost of doing so must be a central element of the Generation IV research program.

2. Cost effective advanced recycle technologies integrated with remote fuel refabrication technologies are the key enabling technologies for achieving the Generation IV sustainability goals. In the full recycle fuel cycles, the waste destined for the repository arises from losses in recycle/refabrication; these losses must therefore be small. Recycle technologies that achieve "clean waste" (i.e., fission

products only) and recycle "dirty fuel" (i.e., all transuranics and optionally some fission products) back to the nuclear power plants are favored; they extract the maximum energy from the ore resource, and they consign only the fission products to the waste. Such recycle technologies based on both aqueous and on dry processes are under active development worldwide. They are yet to be developed to the prototype and commercial scales, and their continued development for cost effectiveness and low losses must be one of the cornerstones of the Generation IV R&D program.

3. When "dirty fuel" provides the feedstock to the fuel fabrication link in the fuel cycle, remote fabrication technologies are required. Simple, few-step processes are favored and robust fuel form designs that minimize reject rates and rework are needed. Colocation of recycle and refabrication facilities is favored to minimize shipping and handling of radioactive materials in bulk form.

   "Dirty fuel" is readily useable in fast spectrum reactors. However, the future global nuclear energy park is envisioned to rely on symbiosis of both fast and thermal reactor power plants – with recycle fuel used in both thermal spectrum reactors and fast reactors. Therefore, the recycle, the fuel design, and the reactor core design must all be coordinated such that thermal reactor neutronics performance is not excessively spoiled by contaminants in the fed back fuel feedstock. R & D on this coordination, in concert with development of appropriate symbiotic mixes of reactor and fuel cycle types in the global energy park should be a central theme of Generation IV R&D.

4. While *the importance of near-term siting, licensing, and operating a geologic repository designed for the once-through fuel cycle cannot be overstated,* for the future, the development of recycle-based fuel cycles must go forward in close linkage with the development of repository designs that exploit new opportunities for customized waste forms and optimized decay heat management strategies to extend their capacities and reduce the level of stewardship required. Specifically, a singular importance in all fuel cycles rests on decay heat management for extending geologic repository capacity.

   Repository siting is and will always be difficult, so mechanisms for extending the capacity of given sites can significantly advance economic and social acceptance goals for nuclear energy. Capacity is controlled not by mass or volume, but by heat load, and heat load is dominated by Cs and Sr in the first hundred years and by minor actinides thereafter. Three alternative approaches are available for

preparation of the waste itself: interim storage of spent fuel prior to geologic emplacement; partitioning/conditioning with interim storage of tailored waste forms prior to geologic emplacement; and partitioning and interim storage of Cs/Sr with reactor recycle for consumption by fission of actinides. Design of repositories tailored for heat management and tightly integrated with the above approaches for waste preparation should be a priority development goal for Generation IV R&D.

5. Recycle and return to the reactor power plants of commixed plutonium and minor actinide bearing fuel creates an intrinsic radiation barrier to theft and diversion from the commercial fuel cycle; the presence of minor actinides and carryover fission products makes the recycled materials less attractive for weapons use as well as less accessible to theft or diversion. It also avoids consigning inventories of weapons-usable materials to interim storage and to geologic repositories, where over time their intrinsic protective radiation barrier decays away. On the other hand the requirement for extrinsic measures – materials control and accounting; physical protection to detect and prevent efforts at theft; and international safeguards to detect efforts at diversion – cannot be eliminated. R&D will be required to adapt existing regimes of extrinsic measures – which have been implemented at commercial scale for the once-through and partial recycle fuel cycles – to applicability to the full recycle fuel cycles. This R&D must be closely integrated with the development of the recycle/refabrication technologies and with the design of the facilities to execute these technologies.

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 1

Introduction: Conceptual Framework & Issues

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Chapter 1

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1.1 Conceptual Framework for the Generation-IV Fuel Cycle

Many of the central issues associated with nuclear power are tied primarily to the choice of fuel cycle. Resource limitations, non-proliferation, and waste management are primarily fuel cycle issues. The fuel cycle provides the mass flow infrastructure that connects the energy resources of uranium and thorium ore through the nuclear power plants to the eventual waste management of the nuclear energy enterprise. For the Generation-IV road-map effort, the cross cutting fuel cycle issues are central to defining a path forward. This report addresses fuel cycle issues in the context of the fuel cycle mass flow infrastructure common to all reactor types.

The overall fuel cycle involves resource mass flows, energy conversion, and resultant flow of waste mass. The FCCG is charged to evaluate the long term ore resource potential and the long term (one century) world energy supply demand projections and, given these, to consider the potential of Generation-IV technological deployments to meet needs within achievable mass flow constraints.

Generation-IV technologies are targeted for deployment by 2030; they will include assets with 60-year useful lifetimes; the Generation-IV energy supply approaches are meant to be sustainable. From these considerations, it is evident that a long planning perspective is required; in particular, it must extend over Generation-IV asset lifetimes of 60 years from time of introduction of Generation-IV technologies by 2030. The FCCG has used a one century planning horizon in its systems studies.

The evaluation over a one-century planning horizon of alternative Generation-IV fuel cycle approaches – taken one at a time or taken in symbiotic combinations involving mutually supportive interchange of mass flows – involves tracking all mass flows necessary for meeting energy demand. It also involves the evolution of fuel cycle enabling technologies – not all of which have currently achieved a state of commercial readiness. Because the evolution of world energy supply infrastructure will involve transitions from the current situation to the future Generation-IV deployment, and the Generation-IV technology readiness and market penetration will be gradual, dynamic tracking is required in the systems analysis with account taken for dynamic inertia elements in the mass flows, asset lifetimes, and construction lag times.

The purpose of this Section is to sketch out the conceptual framework, the relevant issues, and the approach being taken by the FCCG to accomplish its systems analysis responsibility.

1.1.1 Fuel Cycle Framework and Components

The Generation-IV goals of sustainability, safety and reliability, and economics require consideration of the entire fuel cycle – front to back. A particular fuel cycle may excel in one link of the fuel cycle chain, but be unacceptable because of deficiencies in other links.

A conceptual framework is required to facilitate comparisons between different fuel cycles and reactors. This section describes that framework. The fuel cycle (Fig. 1.1), from a global perspective, produces electricity by consuming (1) natural resources and (2) labor and capital. Various wastes are generated as a consequence of these actions. Natural resources include fuels (uranium and thorium), materials of construction, and renewable resources (such as water for cooling purposes). Wastes may include mill tailings, depleted uranium, spent nuclear fuel (SNF) and high level (radioactive) waste (HLW), other radioactive wastes, releases to the environment (air and water), and non-nuclear wastes.

Multiple technical facilities are deployed in the fuel cycle. In the simplified fuel-cycle schematic (Fig. 1.2), there are 6 major fuel cycle facilities.

- *Uranium and thorium mills*. Uranium and thorium ores are mined and then milled to extract uranium and thorium. Because most uranium and thorium ores are very dilute, this process generates the highest volume waste steam in the fuel cycle: mill tailings. There may be several purification and conversion facilities to produce purified nuclear materials.

- *Uranium enrichment*. In most fuel cycles, the uranium is isotopically separated into a stream enriched in the fissile isotope ^{235}U and a stream depleted in the fissile isotope ^{235}U. The enriched stream is used for fuel in most reactors. The depleted uranium may be stored for (1) further recovery of ^{235}U in the future or (2) use in some types of fast neutron spectrum fuel cycles. Alternatively, it may become a waste.

- *Fuel fabrication*. Reactor fuel is fabricated from some combination of natural uranium, enriched uranium, plutonium reclaimed from processing discharged fuel, or uranium-233, depleted uranium, and thorium.

- *Reactor*. The reactor produces energy and SNF.

- *Spent Nuclear Fuel Storage*. The SNF is stored for a time interval to allow its radioactivity and heat release rate to decrease. After storage, the SNF may be (1) considered a waste for disposal or (2) processed to recover fissile materials from the SNF for recycle back to the reactor as new fuel.

- *Processing*. If the SNF is recycled, the recovered fissile materials can be used to produce new fuel. Depending upon the design of the fuel cycle and the reactor, the quantity of fissile material recovered may be large or small. With low conversion ratio reactor designs, a continuing source of fissile from virgin ore is required. With fissile self sufficient reactor designs an outside source is still required if the deployment is growing. With breeder reactor fuel cycles, sufficient fuel may be recovered that fresh fissile material from uranium ore is not required even for a growing

deployment; the depleted uranium reserves would suffice for several centuries before mining would resume.

The preferred choice or choices of fuel cycles and reactors depends upon (1) the requirements for sustainability, safety, and economics, (2) the scale of global operations, and (3) the available technology. Four generic fuel cycles span the space of feasible conversion of ore resources to energy (Fig. 1.3).

- *Once through*. The fuel is fabricated from uranium and thorium, irradiated, and stored for a period to allow for reduction of deployed heat output, then directly disposed of as a waste. Light Water Reactors (LWRs) in the United States currently use this fuel cycle.
- *Partial recycle*. Some fraction of the SNF is processed, and some fraction of the actinide[1] material is recovered from recycle, and new fuel is fabricated. The fuel is returned to the reactor one time or several times (generally not exceeding three) to extract additional energy and the resulting SNF is then disposed as waste. An example is the French recycling system, in which (1) low-enriched-uranium SNF is recycled back to the reactors as MOX fuel and (2) the resultant MOX SNF may be directly disposed of. A second example is the proposed DUPIC fuel cycle which recycles LWR SNF into fresh CANDU fuel with direct disposal of the CANDU SNF. Depending upon the fuel cycle, High Level Waste (HLW) may or may not be produced.
- *Full fissile recycle*. All SNF is processed for recovery and recycle of plutonium and/or ^{233}U. The SNF is repeatedly processed and recycled to fully consume the fissile material through multiple burns in the reactor. Minor actinides and fission products are sent to the waste stream from the processing operation. An example of this is the traditional Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle.
- *Full-actinide recycle*. All SNF is processed, and all actinides are multiply recycled to fully consume the fissionable material. One or more fission products (^{99}Tc and ^{129}I) may be recycled as well. An example of such a fuel cycle is a system of LWRs, Liquid Metal Reactors (LMRs), and molten salt reactors. The LWRs produce power. The LMRs produce power and manufacture excess fissile fuel from fertile U238 to fuel the LWR's, and the molten salt reactor is used to destroy higher actinides that would otherwise be sent to the repository.

1.1.2 Fuel Cycle Evolution

Over time society can convert from one type of fuel cycle to another. The SNF does not disappear but remains in existence – holding a potential to be processed and recycled if desired. SNF remains a source of future fissile and fertile material. All repository programs worldwide are planning to use long-lived waste packages as part of

[1] Actinides are defined as actinium and all chemical elements lying above actinium in the periodic table of the elements. Uranium, Plutonium, Americium, Neptunium and Curium are actinides.

the repository disposal system, and most repository programs require that the waste will be retrievable for an extended period of time.

The fuel cycle infrastructure and the reactors deployed in the fuel cycle *is a dynamic system where market shares of reactor types; where fuel cycle sources of fissile; and where disposition choices for wastes may evolve with time in response to changing economic and social conditions*. Some of the economic and social issues which could drive such time evolution can be illustrated by examining the tradeoffs governing choices for source of fissile material (i.e., the energy resource) to be harvested for use as fuel as a function of time (Table 1.1).

Table 1.1 Trends vs. Time for Sources of Nuclear Fuel Over Time

Property	Source of Fissile Material		
	Natural Uranium	Depleted Uranium	SNF
Resource concentration (kg fissile/kg ore)	⇓⇓	⇓	⇔
Impact of technology (recovery efficiency)	⇑	⇑	⇑
Environmental impact (tailings and repository)	⇓	⇑	⇑
Economics	⇑⇓	⇑⇓	⇑

Referring to Table 1.1, if virgin natural uranium is taken for the source of fissile fuel, the uranium concentration in the ore will eventually decrease over time as the richest ores are mined out. In parallel, it would be expected that (1) uranium fuel costs would correspondingly increase and (2) the environmental impacts would increase because lower grades of ore imply more mining and discarded mill tailings per unit of harvested uranium. Additionally a once through fuel cycle implies growing quantities of SNF requiring disposal. However, technological improvements will take place over time and will tend to reduce costs and environmental impacts. These technological improvements include (1) reactors that produce more energy per unit of uranium or (2) better mining and milling methods for uranium recovery. The fissile resources in natural uranium are effectively unlimited (in theory) given the quantities of uranium in the earth's crust and seawater; it is the economic and environmental constraints that will ultimately limit practically recoverable reserves.

Enrichment tails are also a potential source of fissile material. The natural uranium must be isotopically enriched in ^{235}U (up to 2 to 5% ^{235}U content) in the uranium to produce feedstock for fuel production and existing enrichment processes leave about half of the fissile ^{235}U at depleted concentration in the uranium enrichment tails. As indicated in Table 1.1 this could become a cost effective source of fissile material given technological advances which lower the cost of reenrichment and increase in the cost of natural uranium. This resource is limited by existing inventories of depleted uranium.

SNF is also a potential fissile resource as indicated in Table 1.1. Currently it is placed into interim storage after discharge from the reactor. However, it contains fissile material at much higher concentrations than virgin ore and enrichment tails; over time it could become a cost effective fuel source. Depending upon the type of reactor through which the fuel has been burned, the fissile content remaining in the SNF may be less or greater than that needed to refuel the reactor. If more fissile material has been produced from fertile materials (uranium-238 and thorium-232) than was consumed, the externally supplied quantities of uranium or thorium required to maintain the fuel cycle at steady state are very small. Very little uranium or thorium is required for energy production if (1) the reactor can convert these fertile materials to fissile materials and (2) the fissile materials can be recycled into fresh fuel. Processing of SNF reduces the need for mined uranium and holds potential to reduce the costs and difficulties of disposing of radioactive SNF. It would be expected that the cost of fissile materials from this source would ultimately become less than that from low assay uranium ores or depleted uranium because of (1) the high concentrations of fissile materials in SNF that do not decrease with time and (2) technological progress.

While fissile resources (natural uranium) to supply the fuel cycle have historically been considered as the principal driving factor for evolution of the nuclear fuel cycle, it has become apparent that waste management may also incentivize the time evolution of fuel cycles. In the current situation, there is no shortage of uranium to fuel reactors, but major difficulties exist currently in the disposal of wastes. Some of theses difficulties are associated with technical issues; some are associated with institutional issues; and many are a combination of technical and institutional issues.

For evaluating the paths for evolution toward alternative nuclear futures and the accompanying evolutions of reactor types and choices for fissile supply and waste management, common *issues* arise that, in part, determine the feasibility and desirability of candidate fuel cycle evolutions.

- Availability of resources versus cost
- Future global energy demand
- Readiness of processing technologies
- Waste management strategies, and
- Dynamic response capability of the overall fuel cycle system

After briefly describing the historical evolution of the nuclear fuel cycle up to now, (Section 1.2), the main outside driver (energy demand) and constraint (ore availability) are discussed in Sections 1.4 and 1.3 respectively. Then these issues for the evolving fuel cycle choices are briefly discussed, and the strategy that the FCCG has taken to address them is outlined.

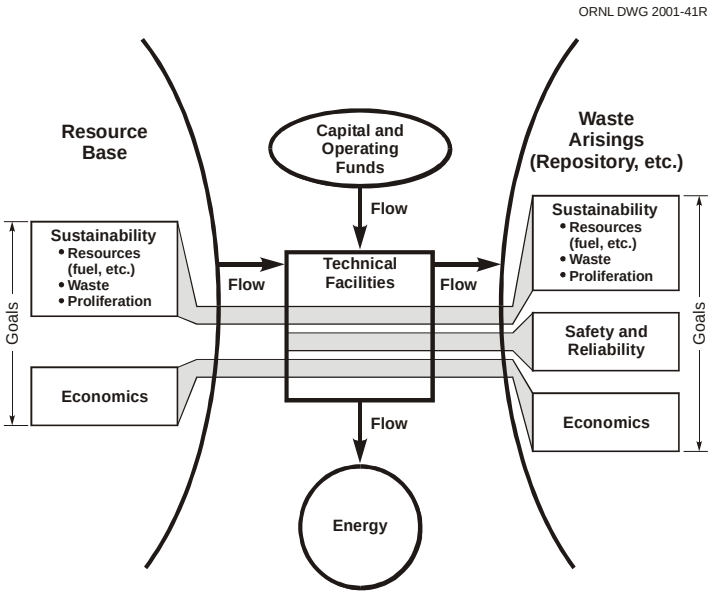

Fig. 1.1 The Fuel Cycle in the Abstract

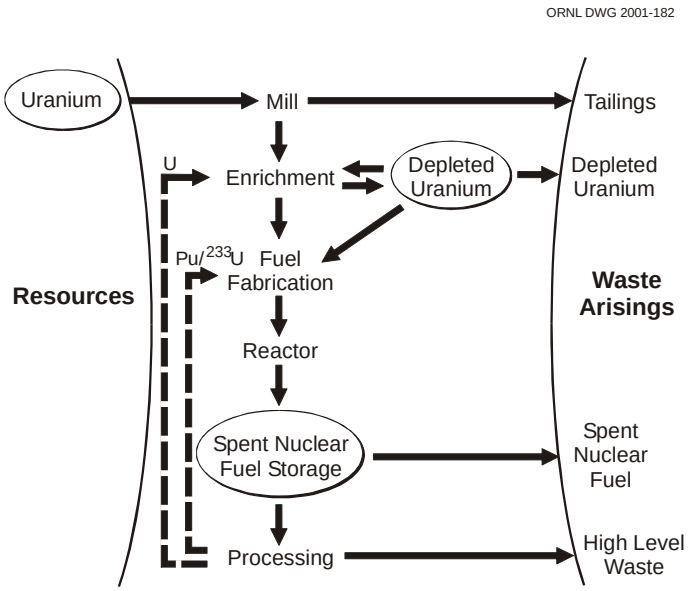

Figure 1.2 Steps in the Fuel Cycle

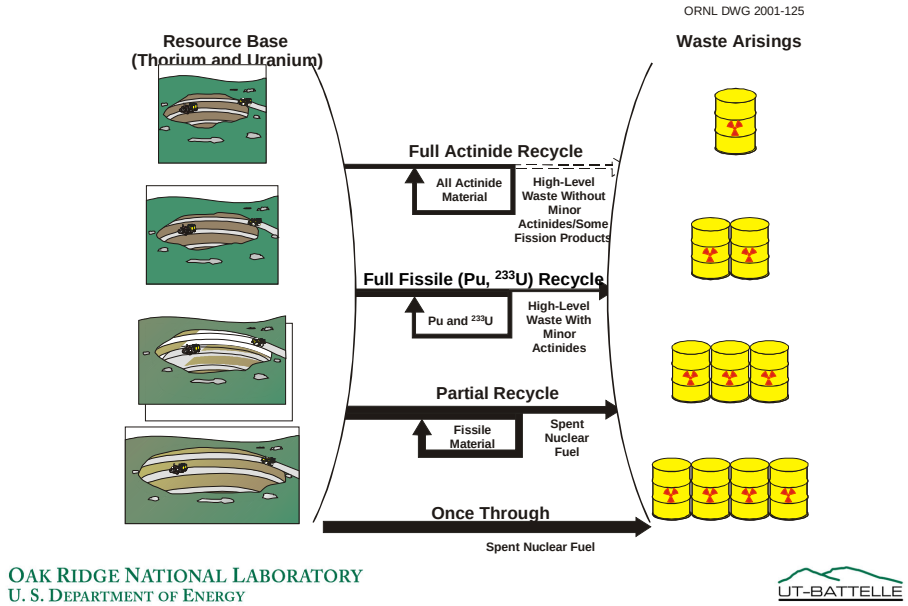

Figure 1.3 Alternative Fuel Cycles

1.2 History of The Nuclear Fuel Cycle

1.2.1 Technological History

Most of the technological steps in the uranium-plutonium nuclear fuel cycle were developed during the 1940s and 1950s. These technologies include mining, milling, conversion, enrichment, fuel fabrication and recycling of irradiated fuel for recovery of fissile. The dominant mission at that time was defense; materials production for nuclear weapons and for fueling naval propulsion reactors. In the years immediately following World War II attention shifted increasingly to commercial energy production via nuclear power. The Smyth report made much of the technology openly available for this purpose, and the Atoms for Peace program and its 1955, 1958, and 1964 International Conferences opened the technology to all nations.

Demonstration of electricity production from nuclear reactors was achieved in 1951 at the EBR-I test reactor in Idaho. The Shippingport reactor, which was ordered in 1953 and achieved criticality in 1957, was the first reactor to supply commercial electricity. It was built largely under government financing and navy leadership. The subsequent commercially developed design and construction of demonstration reactors during the 1950s and 1960s in the U.S., Canada, and the U.K. established the commercial feasibility of extracting useful amounts of heat and electricity from burning uranium fuels in light water, heavy water and gas cooled reactors, respectively. While development work on fast neutron spectrum reactors continued subsequent to the EBR-I demonstration, commercial deployment of nuclear power stations has been dominated by thermal neutron spectrum systems (LWR's, CANDU's, and MAGNOX).

With the rapid growth of nuclear power generation in the U.S., France, Japan, Germany, and the U.K. during the 1970s and 1980s, the back end of the fuel cycle, to include interim storage, spent fuel recycling, and final disposition received increasing attention and it still does.

Fuel cycle strategies in various countries have shifted over time. For example, in the late 1970s, national security of energy supply was of primary concern in many countries due to oil embargoes. Fuel recycling and fast breeder deployment were key elements of long-term energy security at that time in the U.S., United Kingdom, Germany, Belgium, France, Japan, and the Former Soviet Union. These strategies moderated in the mid 1980's, however, with the decline of demand growth rate, decreasing uranium fuel costs and technological and operational advances producing increased fuel burnup and average load factor in LWRs. The thermal reactor once through or MOX mono recycle have dominated the commercial deployment for the past 30 years.

Fuel cycle strategies vary from country to country depending on the availability of resources, indigenous industrial and technological capabilities, economic considerations, the desire for energy independence, and concerns over nuclear safety and nuclear weapon

proliferation. Currently two cycles are extensively deployed – the once-through UOX cycle (see Figure 1.4) and the MOX mono recycle (see Figure 1.5).

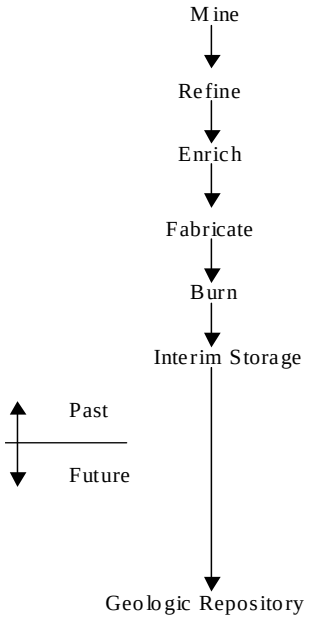

Figure 1.4 The Once-Through UOX Cycle

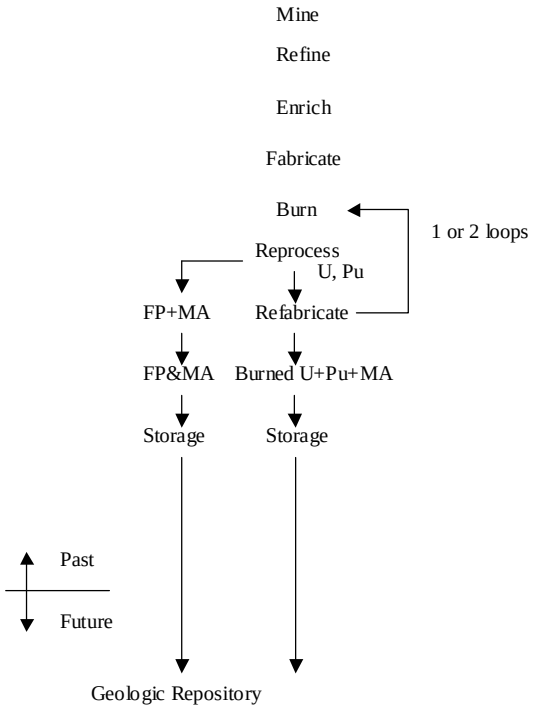

Figure 1.5 The LWR MOX Recycle Cycle

In the US, a UOX once-through cycle is employed. Policy leaders in the U.S. during the mid-1970s concluded "that the most severe risks from fuel reprocessing and recycle are the increased opportunities for the proliferation of national weapons capabilities and the terrorist danger associated with plutonium in the fuel cycle [1]." Acting on this finding, the Carter Administration decided in 1977 to defer indefinitely commercial reprocessing and recycling of separated plutonium. At that time the U.S. also initiated an extensive research program aimed at developing more proliferation-resistant reprocessing methods capable of recycling all actinides and of maintaining high radiation barriers at all stages of materials handling.

Other factors were also responsible for the abandonment of commercial fuel recycling in the U.S. in the late 1970s. Unreliable operation, excessive worker exposure, and escalating standards on radioactive releases also contributed to the shutting down of the West Valley (New York), Morris (Illinois) and Barnwell (South Carolina) reprocessing plants. Furthermore, the plentiful supply of low-cost uranium coming into the market at that time eroded the economic incentive for commercial fuel reprocessing in the U.S.[2]. In 1981 the nuclear waste policy act was passed, creating a once-through waste management approach based on a geologic repository.

Elsewhere, plutonium has been successfully recycled in the form of MOX fuel in thermal reactors starting more than thirty years ago. Today, about 35 thermal reactors use MOX-fuel in a partial core loading pattern. Table 1.2 lists the current status of MOX fuel utilization in thermal reactors world-wide. This commercial application of MOX fuel in LWRs was started in the mid 1980s when some modification or cancellation of fast reactor programs took place, and the already-developed technologies of recycling and fuel fabrication for fast reactors was applied for Pu-recycling in LWR-fuel sector for stabilization of separated plutonium inventories. MOX fuel achieved good irradiation performance, with average burnup between 36 and 44 GWd/tHM.

Currently, the use of MOX fuel has been established on an industrial scale in a number of countries. In Belgium, France, Germany, Japan and Switzerland a considerable number of the thermal power reactors (PWRs and BWRs) are either licensed (i.e. 40 reactors of which 33 have loaded MOX fuel in their reactor core) or have applied for a license (about 13) to use MOX fuel at levels of up to 30% of the reactor core (see Table 1.2).

Table 1.2 Status of Large Scale MOX Fuel Utilization in Thermal Reactors[a]

Status Year End 1998

	Number of Thermal Reactors[a]			
	Operating	**Licensed to use MOX FAs**	**Loaded with MOX FAs**	**Applied for MOX license**
Belgium	7	2	2	
France	57	20	17	8[b]
Germany	19	12	10	4
Japan	52	3	1	1
Switzerland	5	3	3	
Total	130	40	33	13

[a] There are a number of reactors, notably in Europe and India, not included in this Table, which are licensed to use MOX fuel on an experimental basis;

[b] Technically capable reactors planned to be licensed.

Whereas the U.S. and many other nations have adopted a once-through fuel cycle followed by interim storage and eventual direct disposal of spent fuel, other nations have developed and commercialized the recycling fuel cycle based on aqueous recycling. Today major plants for recycling are in operation and under construction in France, India, Japan, Russia, and the United Kingdom. The largest recycling programs for both domestic and foreign fuel can be found in France and the United Kingdom, with France having the most fully developed back-end fuel cycle. In addition to France, Japan has also planned on the increased burning of MOX fuel [3] while Belgium, Germany and Switzerland, embarked on recycling a part of their spent fuel and recycling the Pu in MOX-fuel.

Nuclear fuel cycle industries have achieved technological maturity and they continue to show improvements in economics and safety standards. Their deployment formerly was primarily in industrialized nations, but diffusion to developing countries is already in progress.

1.2.2 Future Prospects

Today, nuclear reactors and fuel cycle front and back end services can be considered a significant component of the global economy. Nuclear produces 17% of the world's electricity. Over 30 countries in most continents (excluding Australia and Antarctica) now own and operate nuclear power reactors. Nuclear fuel cycle services can no longer be considered exclusive enterprises of OECD countries. Twelve countries outside of OECD have constructed fuel cycle facilities, and each step of the total fuel cycle can be found under development or in operation in one or more of these countries (for example, uranium enrichment is done in India, Pakistan, and South Africa and spent fuel recycling is done in Argentina, Brazil and India). In contrast, ten of the thirty OECD countries provide no fuel cycle services.

Projected future energy growth, sustainability of the energy resource base beyond 2050 and global climate change issues have been gaining in importance for policy makers worldwide. In the US, energy independence is again emerging as a policy issue due to recent volatilities in oil and natural gas prices and supply.

Energy suppliers note that fuel costs for nuclear reactors are a much smaller fraction of electricity production costs compared with fossil energy generation, and any future increases in uranium costs would give rise to highly damped effect on cost of production, which is an important factor in the future sustainability of a secure, stable and affordable electric energy supply. Public opinion with regard to nuclear power may be shifting in the US. where nuclear power is being increasingly regarded as a well-established safe and commercially mature technology for electricity generation and one that can alleviate the risks of global climate change. These conditions have motivated the US Department of Energy to charter the Generation-IV Roadmap planning activity to specify how advanced nuclear power should be configured to safely and economically contribute to global energy sustainability.

The capacity for nuclear energy to contribute in meeting future energy needs is limited ultimately by the availability of the raw energy resource. These are estimated periodically by the OECD-NEA and the IAEA as a function of recovery cost. Known conventional resources are delineated as Reasonably Assured Resources (RAR) and Estimated Additional resources (EAR). Speculative Resources (SR) are those that are believed to exist but for which the evidence is indirect. In 1999 [4], known uranium resources recoverable at less than $130/kg U were estimated at 4 million tonnes U. Total known and undiscovered conventional resources were estimated at 15.4 million tonnes U. These resources and the technological and industrial capabilities available today make it unlikely that any reactors operating on a once through uranium oxide (UOX) cycle which are built in the next several decades would ever be shut down due to fuel shortages or high costs. However, by 2050 or earlier the projected life-cycle fuel costs for plants, as well as waste-disposal costs, could create economic incentives to use uranium more efficiently than in current once through or MOX mono recycle fuel cycles. Possible Generation-IV fuel cycle strategies for maximizing uranium resources utilization include [5]:

1. Specifying lower tails assay in enrichment plants
2. Achieving higher burnups in once-through cores with higher enrichment
3. Introducing advanced reactor concepts that have higher thermal efficiencies and thereby higher electrical output per unit of fuel burned.
4. Reprocessing and recycling uranium and plutonium (in the form of MOX fuel) in thermal reactors instead of final disposal.
5. Introducing thorium as a fertile feed material to extend fuel burnup through the production of fissile U-233, likely “denatured” by the inclusion of modest quantities of U-238.
6. Recycling uranium, plutonium, and minor actinides in fast reactors or in fast-thermal systems.

Another consideration for advanced fuel cycle development are the minimization of environmental impacts from fuel cycle operations. Among all fuel cycle operations, uranium mining, like any mining operation, is an important contributor to environment effects; efforts to minimize impacts through reduced mining and through improved mining technologies will be a consideration in the evolution of fuel cycles.

Spent fuel minimization and the reduction of the radioactive source term in high level wastes may also be important to the development and operation of final waste repositories. The full utilization of uranium resources will contribute simultaneously to waste minimization because burning of transuranic elements and minor actinides can substantially reduce the long term radiotoxicity source term of the wastes while at the same time producing energy. In addition, the transmutation of long-lived fission products is, in principle, feasible in symbiotic fuel cycle systems when enough excess neutrons are made available in critical or subcritical systems. In practice, the necessity of isotopic separations and difficulties in the preparation of targets present difficult obstacles for the fission product transmutation which currently reduce the number of candidate nuclides to only one or two, i.e., ^{99}Tc, and, possibly, ^{129}I. Various fuel cycles based on these schemes are under active study throughout the world.

The succeeding sections of this report will discuss in greater detail these and other fuel cycle developments in the context of the Generation-IV initiative.

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1. *Nuclear Power Issues and Choices,* report of the Nuclear Energy Policy Study Group, chaired by Spurgeon M. Keeny, Jr. (Cambridge, Mass.; Ballinger, 1977) p.333.
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6. National Academy of Sciences, *Management and Disposition of Excess Weapons Plutonium*. (National Academy Press, Washington, DC, 1994).
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1.3 Availability of Ore – Fueling Generation-IV

1.3.1 Expected Uranium Resources based on Geochemistry

The bulk amounts of uranium inherited by the earth during its accretion from primordial solar debris has been predicted through the use of various astrophysics models (see Deffeyes 1978). The cold accretion model predicts a uniform radial distribution of uranium at the conclusion of earth's initial accumulation.

Thermal models of the earth point to inevitable melting of the earth soon after its accretion due to gravitation energy and due to radioactive decay of the mass constituents. Because of its large ionic size and heating due to radioactive decay, uranium was transferred from its initial uniform distribution into low melting temperature fractions and out of the earth's core and mantle into the crust. These geochemical geophysical models predict that two thirds of the initial 63 Tt (63×10^{12} tonnes) of uranium present in the earth are now concentrated in the crust, which constitutes only 0.4 % of the earth's total mass. The low uranium and high iron concentrations predicted by such models for the earth's mantle and core have been confirmed by measured concentrations in iron meteorites and in mantle issuing from spreading zone (0.1 ppm U), compared with U concentrations in magma and crust in subduction zones (2 ppm U).

Uranium has a very complex range of geochemical behavior giving rise to a wide variety of economically recoverable deposits. Because it has two valence states in nature, with deficits of four and six electrons, and a variety of complex ions, uranium reacts geochemically in a variety of ways and exhibits a multimodal distribution of concentrations in various geological formations. Deffeyes and MacGregor (Deffeyes 1980) estimated the distribution of uranium in various types of ores, based on the multimodal geochemical behavior and an estimate of a log-normal distribution of concentration. Their results are summarized in Figure 1.6.

Note in Figure 1.6 that only the richest deposits are currently being mined. The slope at our present location on the concentration versus resource curve is 2.5 to 1. Thus one would expect a 300-fold increase in the available resource for every tenfold decrease in the ore grade.

Obviously, mining great quantities of shales and granites for their rather low concentrations of uranium is neither economically nor environmentally sustainable. However, recovery of uranium from previously mined ores via an in situ leach process or the co-recovery of uranium and phosphorus could be desirable, both environmentally and economically.

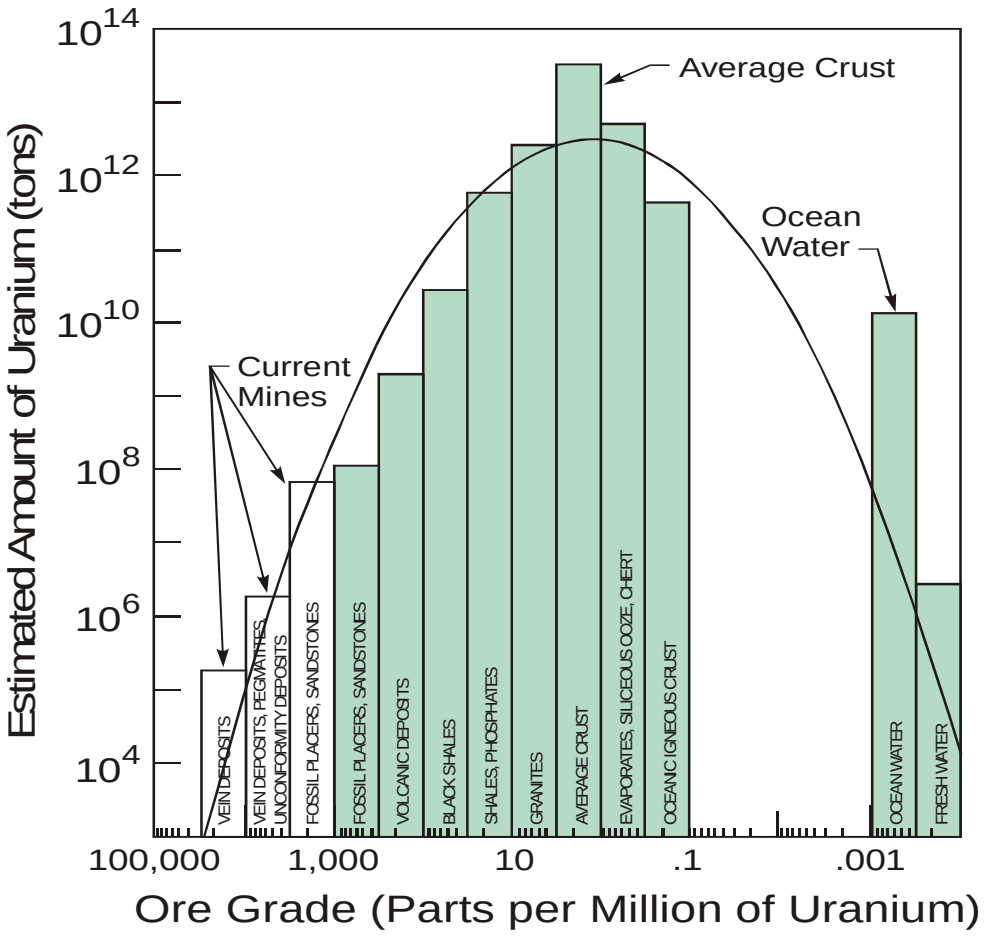

Figure 1.6 Distribution of Uranium in Earth's Crust (from Deffeyes 1980)

1.3.2 Uranium Resources

Commercially practical uranium resources can be categorized into conventional resources, which includes mining of uranium ores by mainly conventional mining technologies, and/or "mining" as a principal by-product of the mining and extraction of other commodities such as phosphates and gold. (An example of non-conventional mining used on a conventional resource is in-situ leaching, accounting for about 14% of current world uranium production.) Unconventional resources are defined to include low-grade resources for which uranium is not economically recoverable at present, e.g. sea water (estimated at 4200 million tones U). Known conventional resources are further sub-categorized as Reasonably Assured Resources (RAR) and Estimated Additional Resources (EAR). Speculative Resources (SR) are those that are believed to exist but for which the evidence is indirect.

International agencies such as the International Atomic Energy Agency (IAEA) and the OECD/NEA provide regular reporting of uranium conventional resources versus cost of recovery in the Redbook (NEA 1999). Their current evaluation is shown in Table 1.3:

Table 1.3 Overview of Uranium Resources, Production and Projected Capabilities

Resources (1 000 tU) reported in 1999						Number of years of present nuclear electricity production[2]
Uranium stocks			200			4
HEU and Pu			600			12
Known Conventional Resources		<40$/kgU	>1 254			
		<80$/kgU	3 002			
		<130 $/kgU	3 954			80
Undiscovered Conventional Resources		<80$/kgU	1 460			
		<130 $/kgU	5 338			
		Total	11 459			230
Uranium in phosphates			22 000			440
Uranium in seawater			4 200 000			80 000
Production (tU) in 1998						
World	34 986		OECD area		19 088	
Projected production capabilities based on Known Conventional Resources			**1999**	**2000**	**2005**	**2010**	**2015**
Existing and committed			45 800[3]	43 800	41 700	37 600[4]	33 000[5]
Planned and prospective					19 400		
Total			*46 000*	*45 750*	*61 200*	*64 800*	*55 000*

[2] The number of years of present nuclear electricity production (in 2000: 2540.5 TWhe [Nucleonics Week, February 8, 2001]), is calculated using a thermal efficiency of 35%, average load factor of 85%, and a ratio of natural to enriched uranium of 8 kgUnat/kgUenr(3.7%).

[3] With a projected plant capacity utilization of about 75%, Existing and Committed capability is about 74% of 1999 world uranium requirements.

[4] 50-16% of projected world uranium requirements in 2010.

[5] Based on the reported data, between 40% and 60% of the expected uranium requirements in 2015 could be satisfied with resources recoverable at $40kgU or less. Resources recoverable at higher costs and additional supplies would be necessary to fill the potential production shortfall indicated by some of the projections.

In considering the resources of any mineral, it is important to remember that those resources are a function of the recovery cost – as is shown in Table 1.3.

Deposits in Australia, Canada, Kazakhstan, and South Africa dominate the known uranium resources of the world, with about 28%, 15%, 20% and 10%, respectively, of the known resources recoverable at a cost of less than $80 per kilogram. The distribution of the resources and the 1999 uranium mine production, as estimated by the Uranium Institute (London), is shown in Table 1.4. The definition of resources used in this table is the OECD/NEA Reasonably Assured Resources category.

Table 1.4 Worldwide Uranium Production and Resources

Uranium Production and Resources		
Country	1999 Production tonnes U	Resources (tonnes U at < $80/kgU)
Australia	5,979	622,000
Canada	8,214	331,000
Kazakhstan	1,390	439,200
Namibia	2,689	156,120
Russia	2,000	145,000
South Africa	981	218,300
Ukraine	500	45,600
USA	1,807	110,000
Uzbekistan	2,130	66,210
others	5,382	113,000
Total	31,072	2,246,430

source: Uranium Institute, http://www.uilondon.org

Comparison to Fossil Reserves

It is interesting to compare the cited uranium resources with the estimated reserves of coal, oil and natural gas. The caveat in the paragraph above also applies to these fossil-fuel resource estimates, in that resources are often understated. However, it is noteworthy that the uranium resources are energetically equivalent to about 10% of natural gas plus oil resources (2.7% of all fossil reserves including oil, gas, and coal) even with their (inefficient) use in LWRs only (see Table 1.5). More efficient use by exploiting the fertile isotope ^{238}U in fast neutron spectrum systems shows a factor of 8 superiority of uranium to fossil energy potential.

Table 1.5 Comparison with Fossil Reserves

Comparison to World Fossil Resources			
Coal		1087	Billion tons (WEC,BP)
	=	2.43E+22	J thermal
	=	22,990	quad BTU
Oil		1033.2	Billion barrels (OGJ)
	=	6.32E+21	J thermal
	=	5,993	quad BTU
Natural Gas		5141.6	Trillion cubic feet (OGJ)
	=	5.59E+21	J thermal
	=	5,301	quad BTU
Total Fossil Energy		3.62E+22	J thermal
	=	34,283	quad BTU
Uranium Resources		2,246,430	tonnes U
	=	920	quad BTU (in LWRs)
	=	172,404	quad BTU (in breeder)

EIA: AER 99, pp. 277, 295

Uranium Ore Prospecting – Historical and Future

In the 1940's and 1950's uranium was believed to be a scarce resource. Since uranium-235 (^{235}U) is the only naturally occurring isotope that can readily sustain a fission chain reaction, in thermal neutron spectrum reactors, the resources of uranium (containing only 0.7% of ^{235}U) were believed to inherently limit the sustainability of nuclear energy based on fission. Concern that uranium would soon run out was one of the driving forces in the development of fast breeder reactors, which (through an intermediate step of converting ^{238}U to plutonium) consume the abundant isotope ^{238}U (>99%) of the ore. Similarly reactors capable of breeding thorium into the fissile (but not naturally occurring) isotope uranium-233 (^{233}U) were also developed at that time for the same reason. (Thorium, consisting almost entirely of the isotope ^{232}Th, is about three times more abundant as uranium).

Driven by those concerns, there have been two periods of extensive exploration for uranium, in the 1950s and in the 1970s, both followed by long periods of severe contraction in the market and the near-cessation of exploration activity. Over-expansion of the uranium supply infrastructure during the 1970's led to limited exploration and the closure of operating mines during the last 20 years. The situation has remained the same due to slower growth of commercial nuclear power than was originally anticipated. More recently a new disincentive to uranium exploration has emerged – disarmament. As much as one third of the natural uranium which has been mined since 1945 was used for the production of highly enriched uranium (HEU), primarily for weapons programs, and to a lesser degree for naval propulsion and for research reactors. The START weapons reduction treaties and heightened concerns for the proliferation of nuclear weapons have led in the past several years to programs for the down-blending of weapons-derivative high enriched uranium (HEU) in which HEU is mixed with natural or depleted uranium to produce low enriched uranium (LEU) to serve as fuel feedstock for commercial light water reactors (LWRs). The civilian use of that down-blended HEU effectively eliminates the need for as much as 600,000 tonnes of natural uranium, equivalent to ten years of current worldwide production. (Kidd 1998) Thus the market prices for natural uranium have recently been further depressed and the incentives for exploration have been very small.

The Reasonably Assured Resources cited in Table 1.3, would indicate that, at the current consumption rate of 64,000 tonnes of natural uranium per year, world resources would last only about 35 years. However, such a conclusion is likely to be incorrect for several reasons. The Redbook (IAEA 1999) assesses the current uranium resources base at 11 million tonnes if less well-proven, speculative and higher cost resources are included. Given the very low level of uranium exploration in the last two decades, the resource base is undoubtedly larger. Uranium, with a crustal abundance of 2.7 wppm, is twice as abundant in the earth's crust as molybdenum or tungsten, and 40 times more abundant than silver. Despite low prices and disincentive to exploration, additional resources of uranium continue to be discovered, particularly in northern Canada and in Australia. In Table 1.4 the Canadian resources are listed as 331,000 tonnes of uranium. However, the MacArthur River deposit alone contains 161,000 tonnes of uranium in ore that is 15% U_3O_8. The Jasper Lake deposit contains 123,000 tonnes of U in ore that is

13.6% U_3O_8. The two deposits cover less than 120 square miles and were first explored 20 years ago. Cigar Lake (137,000 t U) and Key Lake (70,400 t U) are also within 20 miles. Deposits in Australia have been of comparable concentration and size. The richness of the Canadian deposits and the relatively small area that has been explored to yield them suggest that additional demand and exploration will result in substantial further discoveries.

In addition to the discovery of new resources through increased exploration, improvements in mining technology are also lowering the cost of previously high-cost deposits. In particular, In Situ Leaching (ISL) is of growing significance and could be applied to existing gold and phosphates tailings piles. The resource base of 11 million tonnes U does not include the uranium in those tailings.

Uranium content of Enrichment Tails

The inventories of depleted uranium stored at enrichment plants constitutes a large fissile uranium resource. Because of low price of natural uranium in recent years, many plants have been operating with tails assays of up to 0.3 % ^{235}U. The 1.2 million tonnes of uranium currently stored at enrichment plants could supplant a few hundred thousand tonnes of natural uranium if demand required.

Uranium Content of Seawater

Uranium is present in seawater at 3 wppb and represents a well-quantifiable amount of 4 billion tonnes, >1500 times the Reasonably Assured (land-based) Resource shown in Table 1.3.

The recovery of uranium from seawater is highly speculative and may never prove to be economic and/or ecologically justifiable. One tonne of seawater contains 3 mg of natural uranium, which can deliver 244 MJ_{th} in a breeder or about 2.5 MJ_{th} in a present day LWR. Simple calculations show that the pumping energy needed in an extraction plant could easily consume all the energy available, particularly in the LWR case. (Thus seawater extraction conceptual designs relying on ion exchange have relied on natural currents to move the seawater past uranium-collecting surfaces.)

Uranium-recovery from seawater has been studied in Japan with an eye to the very long term or a very strong development of fission energy. On a laboratory scale, experiments have been performed where uranium is trapped on amidoxime adsorbent which were prepared from non-woven strands of polyethylene with the aid of radiation-induced co-grafting. These experiments produced 1.7 gU/kg-absorbent after 60 days of contact with seawater. At this stage of the study it is difficult to predict the practical application of uranium recovery from seawater. An economic assessment has been reported [4] indicating a possible cost for this uranium process in a 1,000-ton/yr commercial plant of approximately $600/kgU. Ultimately, with an improved absorbent, a goal of about $100/kgU was reported to be feasible in the longer term. Other studies have suggested much higher costs

As discussed later, the harvesting of remaining fissile material in enrichment tails or in discharged fuel provide more likely routes to resource extension than does seawater extraction. However, the magnitude of the seawater resource places an upper limit on the cost of uranium. Estimates of recovery costs have been in the neighborhood of $200/kg U, to 1000 $/kg U – though these are highly speculative.

1.3.3 Thorium Resources

Thorium, which averages 7.2 parts per million in the earth's crust, is the 39th most abundant of the 78 crustal elements. It is about three times more abundant than uranium in the earth's crust. Because ^{232}Th is the only isotope of natural thorium it provides no fissile material from nature. It is necessary to breed fissile ^{233}U from thorium through exposure to neutron fluxes – which requires some other source of fissile to initiate a fuel cycle based on thorium. When bred to the fissile ^{233}U, thorium releases about the same energy per unit mass (79 TJ_{th}/kg) as uranium when bred to ^{239}Pu (80.4 TJ_{th}/kg).

Thorium and its compounds have been produced primarily as a by-product of the recovery of titanium, zirconium, tin and rare earths from monazite. Only a small portion of the thorium produced is consumed. Limited demand for thorium, relative to the demand for rare earths, has continued to create a worldwide oversupply of thorium compounds and mining residues. Most major rare-earth processors have switched feed materials to thorium-free intermediate compounds to avoid the handling of radioactive thorium. Excess thorium not designated for commercial use is either disposed of as a radioactive waste or stored for potential use as a nuclear fuel or other applications. Increased costs to comply with environmental regulations and potential legal liabilities and costs to purchase storage and waste disposal space were the principal deterrents to its commercial use. Health concerns associated with thorium's natural radioactivity have not been a significant factor in switching to alternative non-radioactive materials. (USGS 1999)

In the short term, thorium is available for the cost of extraction from rare-earth processing wastes. In the longer term, large resource of thorium are available in known monazite deposits in India, Brazil, China Malaysia and Sri Lanka.

Quoted prices for thorium dioxide on Dec. 31, 1999, were $ 82.50 per kilogram for 99.9% purity and $107.25 for 99.99% purity. The relatively high prices for thorium are more attributable to the low level of consumption rather than to the inherent scarcity of the element.

Thorium is present in seawater at only about 0.050 wppb, due primarily to the insoluble nature of its only oxide, ThO_2. Thus the recovery of thorium from seawater is not a realistic option.

1.3.4 Sustainability Potential

The "known conventional resources" of uranium ore which are recoverable at less than 130 $/kg U are reported in the Red Book to be slightly under 4 million tonne of U. (See Table 1.3) When "undiscovered conventional resources" recoverable at less than 130 $/kg U ore added, the total rises to ~11.5 million tones U. (See Table 1.3). If used in current LWR's once through cycles to meet projected nuclear demand it will last for multiple decades but not for a century (see Chapter 3).

Figure 1.7 shows that overall fuel cycle costs are about 20% of cost of energy production in current LWR's operating on the once-through cycle and that the cost of the uranium itself currently comprises only 5% of energy production cost. This provides for a valuable absence of volatility to fuel costs for nuclear energy which is not shared by fossil sources. A detailed analysis is presented in Attachment 1 which quantifies the insensitivity of cost of LWR produced electricity to price of uranium ore. No reactor built in the next couple of decades and operating on the once-through cycle would ever be shut down due to uranium scarcity, regardless of its conversion ratio.

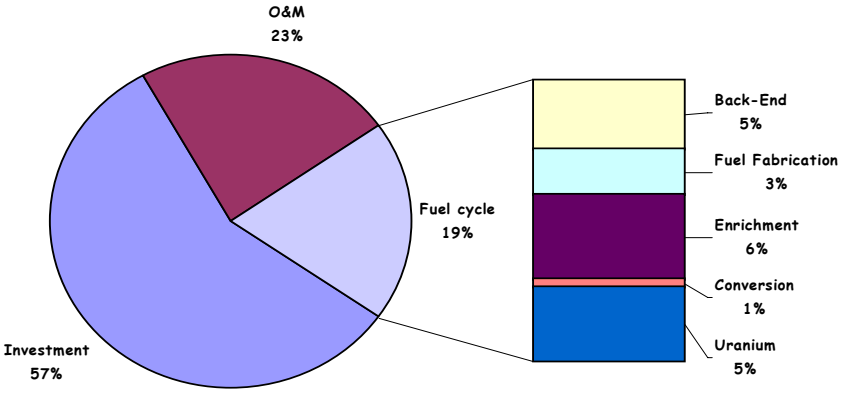

Figure 1.7 The Cost of Electricity from Nuclear Power

A requirement of the Generation-IV initiative is to produce *sustainable* technologies. They are to be deployable within thirty years; with asset lifetimes of 40 to 60 years for specific deployed plants. The Generation-IV planning horizon is therefore a full century into the future even for the first wave of deployments. Given a significant nuclear expansion forecast for the coming century by numerous international planning bodies, economic forces can be expected to emerge providing impetus to evolve the fuel cycle to include recycle of spent fuel, motivated both from rising uranium prices and from efforts to use repository space efficiently. For such recycle based fuel cycles, if exploited fully, at least a millennium of energy supply can be foreseen from the earth's endowment of economically recoverable uranium ore listed in Table 1.3. Likewise, the energy potential of the earth's endowment of thorium is greater still. *Thus the resource base is sufficient to support the Gen-IV sustainability goal, -- given that the fuel cycles and reactor types are deployed to fully exploit those resources.*

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1.4 Future Demand for Energy Services – Meeting Generation-IV Client Needs

1.4.1 World Energy Use Per Capita; Resultant Mass Flows

Providing energy to meet society's needs is an enterprise of gigantic scale. Figure 1.8 shows annual energy use per capita for various countries (1) expressed in units of tonnes of oil equivalent per capita. [One tonne of oil equivalent equals 42 GJ_{th} or of 40 million BTU]. In the industrialized nations, nearly 9 tonnes of oil equivalent (toe) per year are consumed to support the energy needs of each man, woman and child. These industrialized economies drive their economies with energy intensities in the range of 0.5 kgoe/$GDP (see Fig. 1.9). Of this energy usage about 1/3 is delivered in the form of electricity, 1/3 is in the form of liquid fuels for transportation and the remaining 1/3 in the form of heat for industry, farming, building heating, etc.

The mass flows to supply this energy to industrial societies (which comes dominantly from burning fossil fuels) is daunting – about 9 tonnes of drilling/mining, shipping, and burning of fossil fuels per person per year. Just as daunting are the flows of effluents and wastes produced when the energy is harvested – no less than 9 tonnes of ash and/or CO_2 emitted per person per year from combustion alone – not to mention secondary wastes. Since 20% of the world population of 6 billion currently consume energy at this level, mass flows in the range of $2*10^9$ tonnes of commerce and waste generation annually occur to support the energy need of the industrialized economies.

As regards to reduced mass flows of both energy resource materials and of effluents and wastes for supplying energy needs, *the million-fold greater energy density of nuclear vs chemical bond strength which is released as heat upon fission holds the potential to reduce the scale of mass flows and waste generation by upwards of a million fold were nuclear to supply a significantly larger fraction of the future world energy supply*. The complementary facets of nuclear energy's known multi-millenia resource base and its intrinsically low emission per unit energy benefit are the sustainability promise that Generation-IV seeks to exploit for meeting future world energy service needs.

1.4.2 Energy Demand Growth – The Generation-IV Clients

80% of the world population consumes energy at rates of as much as ten times lower than in industrial societies (Fig. 1.8) achieving a much lower GDP/capita not only because of smaller energy use but also because it is utilized much less effectively (Fig. 1.9). For a significant fraction of the world's population, their entire energy use is confined to burning straw or wood for cooking and heating. Were it possible by 2050 to develop most world economies to even half that of the Western world, the world's energy supply mass flows and waste generation rates would roughly triple their current levels.

By the time Generation-IV systems are significantly integrated (~2050 to 2100) into the world's energy supply market, the world's population will have nearly doubled (to 11 billion in 2050 and 12.5 billion by 2100) – leading to another factor of 2 for a total

factor of from three to six increase in the overall scale of world energy supply infrastructures over the next century.

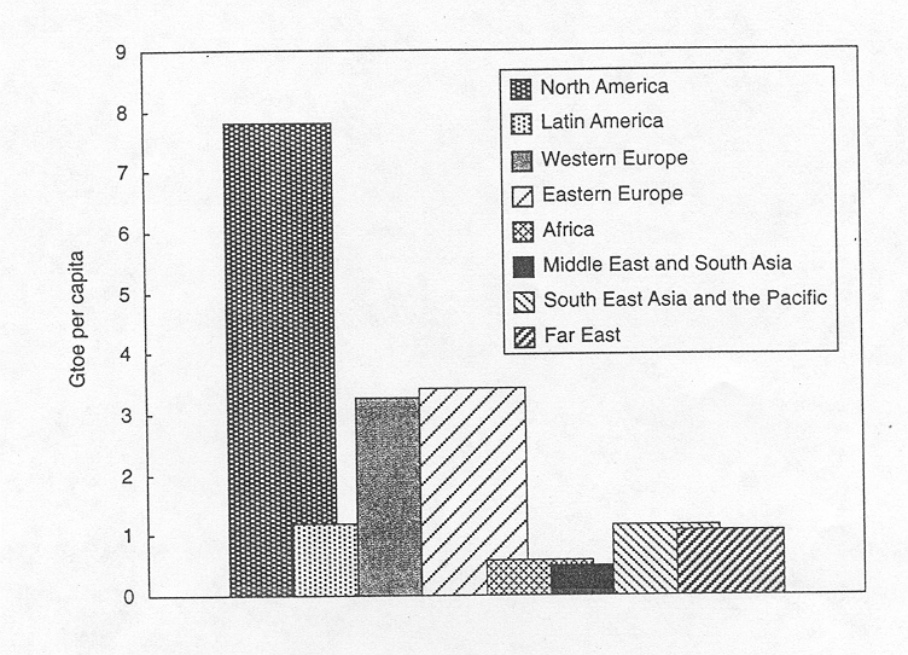

Fig. 1.8 Regional energy consumption (1995). [International Atomic Energy Agency, Energy, Electricity and Nuclear Power Estimates for the Period up to 2015, July 1996 Edition, IAEA, Vienna (1996)]

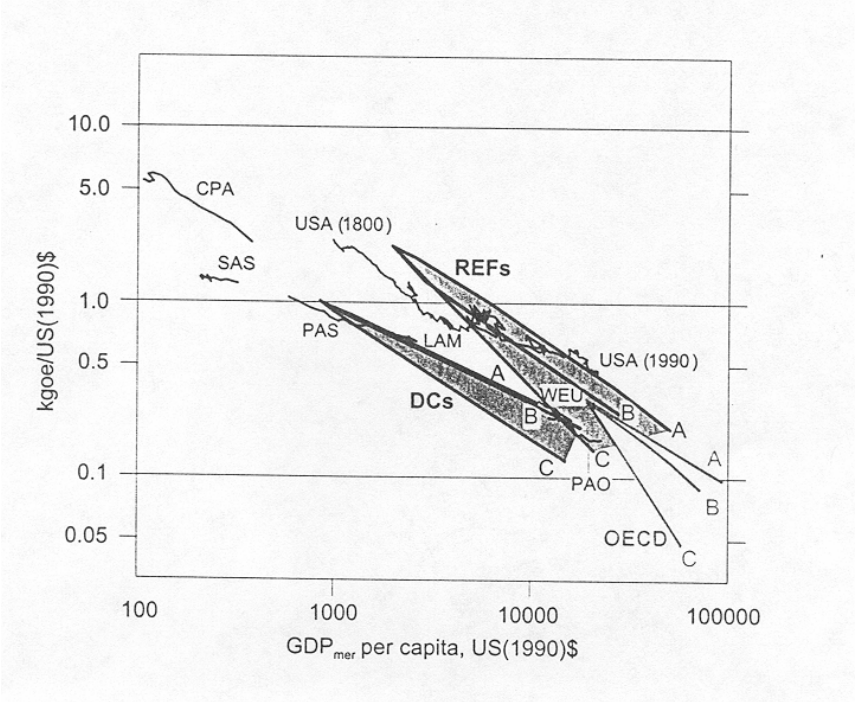

Fig. 1.9 Energy intensities, in kgoe per US (1990)$, as a function of degree of economic development, in GDP_{mer} per capita in US(1990)$. Historical data (black) and Cases A, B, and C (color).

The IIASA/WEC study (2) of the World's future region-wise energy needs, which was done with substantial rigor over an intensive multi-year campaign involving more than a hundred participants, produced predictions which are not inconsistent with the rough estimates produced above. Moreover, the WEC/IIASA study shows that energy demand growth rate will be highest in developing economies, and that the deployment of nuclear power will diffuse from the historical nearly exclusive application in the industrial countries; in the future it will support energy needs in the world as a whole. *The challenge for the Generation-IV initiative is to create and deploy nuclear energy supply options capable to contribute to applicability worldwide.*

1.4.3 Nuclear Energy's Potential for Broadened Energy Services

Nuclear energy currently supplies 17% of the world's electricity but only 6% of primary energy overall. As shown in Section 1.3, the earth's endowment of recoverable uranium and thorium ore is sufficient to support several millennia of the world's future energy needs – and not just electricity alone, but a more diverse fraction of primary energy needs. And numerous energy service needs beyond electricity alone can potentially be provided by Generation-IV concepts under consideration in the Roadmap.

Desalinization

Potable water can be produced via desalinization (Multi-Stage Flash or Multi-Effect Evaporation) bottoming cycles on electrical power production plants, -- or by use of nuclear generated electricity for driving Reverse Osmosis processes for off-site desalinization. The World Energy Council reports that even now one billion people lack adequate water supplies, and they predict the number to exceed 2.5 billion by 2025. This is a critical sustainability need for future society and one where Generation-4 can contribute.

Process Heat

Supply of emission-free high temperature process heat to the industrial sector was one of the driving forces for high temperature gas reactor development during the 1970's. Figure 1.10 displays some of the energy intensive industrial applications for process heat. Nuclear generated heat could find applications in industry at temperatures reachable by water reactors (~325°C), liquid metal reactors (~550°C) and gas cooled reactors (~950°C). This presents both an opportunity to diversify Generation-IV's contribution and a challenge to achieve safety levels compatible with siting needs for industrial applicability.

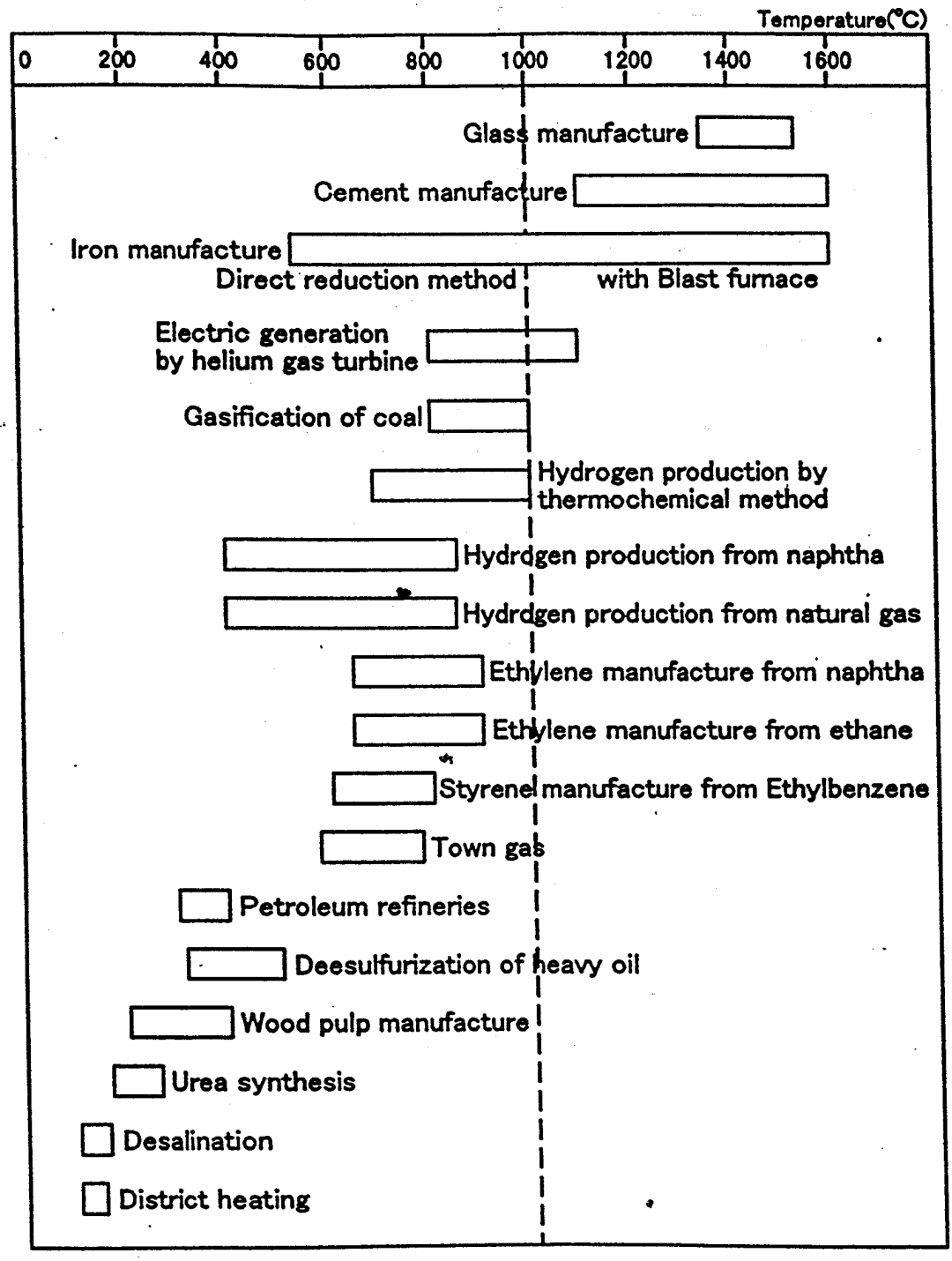

Figure 1.10 Temperature Region of Heat Used in Various Industries

Hydrogen Production

One third of primary energy use applies to the transportation sector. It is supplied exclusively by fossil (oil) resources. The commercial sector is currently expending hundreds of millions of dollars to exploit the potential emission reduction benefits of fuel cell powered vehicles. Onboard reformers to convert hydrocarbon-based liquid energy carriers (e.g., gasoline, methanol, etc.) to the hydrogen fuel used in the fuel cell are planned for the near term, but *emission reductions must ultimately rely on a carbon-emission-free source for hydrogen manufacture*. Nuclear production of hydrogen via electrolysis or via thermochemical water cracking cycles provides an avenue for Generation-IV concepts to vastly broaden nuclear energy's contribution to transportation's one third sector of the world's primary energy needs.

While transportation needs for hydrogen will be a growing market by 2030, extensive and growing markets for hydrogen already exist in the fertilizer manufacturing industry (ammonia) and in hydrocarbon refining. As the world's oil reserves are depleting, refiners are experiencing a growing need for hydrogen to sweeten (desulphurize) and lighten (increase the H to C ratio) of the degrading hydrocarbon feedstock.

As hydrogen gradually displaces hydrocarbon (solid, liquid and gas) energy carriers in the mid-century decades and beyond, the world's energy supply infrastructure will be based on electricity and hydrogen. In concert, they could service all primary energy needs. Generation-IV technology development has an opportunity for expansion of nuclear's role into a diversified and broader share of the primary energy service needs of society via the medium of hydrogen and electricity as complementary energy carriers.

References for Section 1.4

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1.5 Issues for Generation-IV Fuel Cycle 100 Year Systems Studies

The Fuel Cycle Crosscut Group's (FCCG) Charter provides for it to perform a systems analysis function in support of the Generation-IV Roadmap activity. Generation-IV technologies are targeted for deployment by 2030; they will likely include assets with 60 year useful lifetimes; the Generation-IV energy supply approaches are meant to be sustainable. From these considerations, it is evident that a century long planning perspective is required; in particular it must extend over Generation-IV asset lifetimes of 60 years from time of introduction in 2030.

As illustrated in Figure 1.2 the "system" transforms ore resources to energy products with the resultant production of waste. The fuel cycle infrastructure which executes this transformation includes resource mass inflows, energy conversion, and resultant waste mass outflows. The FCCG is charged to evaluate (1) the long term ore resource potential (discussed in Section 1.3) and (2) the long term (one century) world energy supply demand projections (discussed in Section 1.4) and, given these, to consider the potential of various Generation-IV fuel cycle deployments to meet energy needs within achievable mass flow constraints.

The evaluation over a one century planning horizon of alternative approaches – taken one at a time or taken in symbiotic combinations involving mutually supportive interchange of mass flows – involves tracking the mass flows necessary for meeting energy demand and keeping account of ore withdrawals and waste production. It must be a dynamic tracking because the evolution of world energy supply infrastructure will involve transitions from the current situation of technology availability and already-deployed assets to the future Generation-IV deployment, and the Generation-IV market penetration will be gradual and ultimately constrained by physically achievable time constants.

The purpose of this Section is to sketch out the relevant issues concerning the achievability and form of this evolution, and the approach being taken by the FCCG to accomplish its systems analysis responsibility.

1.5.1 Projected Resource vs Price

In the scenario evaluations discussed in Chapter 3 a fuel cycle cost index is used to provide a rough indication of which segments of the fuel cycle contribute in an important way to overall cost of fuel cycle services. We start with ore resources. To perform 100 year scenarios, a cost of uranium is needed versus cumulative ore withdrawals. The scenarios used a cost schedule based on the Redbook evaluations of resource vs cost of recovery up to ~15.1 million tonnes of U recoverable at $\leq$ 130 \$/kgU; it was assumed to increase to 200 \$/kg by 30 million tonnes cumulative withdrawals and to hold constant after that. This coarse representation for use in the scenarios was chosen in light of prior evaluations to determine a price elasticity of ore supply – which is presented here.

The EIA has estimated uranium prices through 2015 under two scenarios, shown in Table 1.6. The reference case assumes that the present fleet of LWRs will continue to operate through the projection period, but that no new reactors will be built. After 2010, the spot-market price is projected to rise above \$13.00 per pound U_3O_8. Even as uranium requirements decline after 2010, the spot-market price is projected to remain above \$13.00 per pound U_3O_8 as new mines will be required to offset the depletion of reserves. With their substantial high quality reserves, Canada and Australia are expected to be the principal uranium producers.

Table 1.6 Projected Uranium Prices through 2015

Projected Uranium Spot-Market Price 1999-2015
(1998 Dollars)

Year	Reference Case		Hign Inventory Case	
	per lb of U3O8	per kg U	per lb of U3O8	per kg U
1999	\$10.30	\$19.97	\$10.30	\$19.97
2000	\$11.10	\$21.53	\$10.70	\$20.75
2001	\$11.50	\$22.30	\$10.80	\$20.94
2002	\$11.50	\$22.30	\$10.70	\$20.75
2003	\$11.50	\$22.30	\$10.70	\$20.75
2004	\$11.90	\$23.08	\$11.10	\$21.53
2005	\$12.50	\$24.24	\$11.80	\$22.88
2006	\$13.00	\$25.21	\$12.50	\$24.24
2007	\$13.20	\$25.60	\$13.00	\$25.21
2008	\$13.20	\$25.60	\$13.20	\$25.60
2009	\$13.30	\$25.79	\$13.30	\$25.79
2010	\$13.40	\$25.99	\$13.40	\$25.99
2011	\$13.50	\$26.18	\$13.40	\$25.99
2012	\$13.40	\$25.99	\$13.30	\$25.79
2013	\$13.30	\$25.79	\$13.30	\$25.79
2014	\$13.40	\$25.99	\$13.30	\$25.79
2015	\$13.30	\$25.79	\$13.30	\$25.79

EIA projections, http://www.eia.doe.gov/cneaf/nuclear/special/uranproj.html

For the high inventory case, an additional 26 million pounds of U_3O_8, at increments of 1-4 million pounds per year, are assumed to have penetrated the market from the drawdown of inventories during 2000 through 2008. In the high inventory case, the three Western suppliers, Cameco, Cogema, and Nukem, are assumed to pursue more aggressive sales of HEU feed purchased from Russia. In response to additional drawdown of inventories, the uranium spot-market price is projected to remain below \$11.00 per pound U_3O_8 until 2004, four years later than projected for the reference case.

The EIA has also estimated the forward cost of uranium reserves using various mining methods. This estimate is summarized in Table 1.7.

Table 1.7 U. S. Uranium Reserves by Mining Method, 1999

U. S. Forward Cost Uranium Reserves by Mining Method, 1999								
	$30 per pound of U3O8				$50 per pound of U3O8			
Mining Method	Ore (million tons)	Grade (% U3O8)	U3O8 (million lbs)	Uranium tonnes U	Ore (million tons)	Grade (% U3O8)	U3O8 (million lbs)	Uranium tonnes U
Underground	25	0.271	138	55,293	143	0.163	464	185,912
Openpit	10	0.139	29	11,620	163	0.079	257	102,973
In Situ Leaching	40	0.132	106	42,471	121	0.076	183	73,323
Other	1	0.264	1	401	3	0.059	4	1,603
Total	76	0.179	274	109,785	429	0.106	908	363,811

EIA Uranium Industry Annual, 1999, p. 7

Using the admittedly sparse data, the FCCG determined the price elasticity of the various mining methods; shown in Table 1.8. Note that the elasticities are all greater than 1, e.g. a 10% increase in prices produces at an 11 % increase in supply. Furthermore, note that the open pit mining has the greatest elasticity, since idled mines would be re-opened if the market price rose above a mine-specific threshold. On the other hand, in situ leaching, which is often remediation or co-production operation, would tend to continue operating despite fluctuations in the price of uranium.

Table 1.8 Elasticity of Uranium Supply

Elasticity of Uranium Supply	
Mining Method	(change in supply/change in price)
Underground	2.4
Openpit	4.3
In Situ Leaching	1.1
Other	2.7
Total	2.35

Thus, for crudely estimating the uranium resource as a function of price, we might assume the same elasticity worldwide.

$$R = 77.4 \quad P^{2.35}$$

where R is the worldwide resource of uranium in tonnes U and P is the uranium price in dollars per kilogram U as UF_6. P is greater than $23, the current price, and less than $200, a lower bound assumed cost of extraction from seawater. Note that the predicted exponent based on the geochemical analyses by Deffeyes would be 2.48, as opposed to the value of 2.35 based on the EIA data.

Selected values of the global uranium supply are shown in Table 1.9, ranging between near-term prices and an assumed (lower bound) seawater extraction price. For this purpose, at the seawater extraction price of $200 per kg U, the price of uranium no longer increases with cumulative withdrawals.

This model has inspired the *form* of the virgin uranium cost model used for the scenarios in Chapter 3. That model does not precisely reproduce Table 1.9, it is described in Attachment 2.

Table 1.9 Resource - Supply Relationship

Price		Resource	
$/lb U3O8	$/kg U	tonnes U	
$15.00	$37.44	395E+3	
$20.00	$49.92	758E+3	
$25.00	$62.39	1.28E+6	
$30.00	$74.87	1.97E+6	
$32.06	$80.00	2.30E+6	
$35.00	$87.35	2.82E+6	
$40.00	$99.83	3.86E+6	
$50.00	$124.79	6.53E+6	
$60.00	$149.75	10.0E+6	
$80.14	$200.00	4.2E+9	Seawater assumed value

1.5.2 Projected Growth in Energy Demand

While the ore resource availability ultimately limits the energy that can be delivered, the driving force on the global nuclear fuel cycle is the growth of energy demand. One hundred year scenarios for energy demand are fraught with acknowledged uncertainty beyond even a few years. The FCCG elected to rely on the projections produced by the WEC/IIASA who have for many decades specialized in such long range economic, societal, and technical projections.

The world energy demand projections to 2100 and the nuclear share in them are selected from the set of aggregated regional scenario cases developed for IIASA/WEC study published [1] in 1998. The world projections result from aggregating studies of population growth, economic development, and resulting energy demand for the 11 world regions shown in Table 1.10 The study was conducted in two phases; from 1993 to 1995 six energy future scenarios were developed for each of the 11 regions in a ten person year effort. The second phase, from 1995 to 1998 was devoted to extensive outside reviews of the 11 regional forecasts by teams of regional experts and reviewers (totally over 100 individuals). Their findings were used to revise the 1995 results and produce the 1998 projections. This study is widely cited and highly regarded.

The regional forecasting process relied on a country by country population growth projection produced [2] by the World Bank in 1992. In this projection (the “center” projection among several), world population will be at 10.5 billion by 2060 and 11.7 billion by 2100. Virtually all the projected growth occurs in the currently non-industrialized countries. Given the population and economic development projections, the resulting regional energy demand projections were then developed on the basis of historical correlations relating energy consumption per capita vs GDP – accounting for historical trends of improvements in energy intensity (decreasing energy use per unit GDP) as the GDP increases (see Fig. 1.9).

Six patterns of how energy demand will grow and will be met were employed in the IIASA/WEC study to illustrate the range of potential energy futures. Given a pattern, that pattern was assumed to prevail in all world regions when producing the global aggregate. All patterns provide for substantial social and economic development with growth in quantity and quality of energy services provided and with improving energy efficiencies and environmental compatibility. While all six patterns of development are hopeful, the styles and drivers of development represented are different:

Pattern (Case) A - (with three scenarios) is one of high economic growth and assumed high degrees of technological ingenuity

- Scenario A1 high availability of oil and gas
- Scenario A2 return to coal; scarce oil and gas
- Scenario A3 nuclear and renewables; fossil phaseout

Pattern (Case) B - (single scenario) is the middle course – "muddling through" case with more modest energy demand, slower technological innovation, and less uniform rates of economic growth among developing countries.

Pattern (Case) C- (with two scenarios) is an ecologically driven pattern with assumed unprecedented progressive international co-operation focused on environmental protection and international equity and relying on North to South technology and institutional transfers

- Scenario C1 renewables grow dramatically; fossil reduction, and nuclear phaseout
- Scenario C2 nuclear ascendance, fossil reduction

Table 1.11 indicates the main characteristics of the three broad patterns in terms of world population, economy, and primary energy consumption out to the year 2100. In all cases shown in Table 1.11, the world is facing astounding and unprecedented absolute growth in energy demand – more than doubling by 2050 and quadrupling by 2100 (except for Case C). Notwithstanding any ecological consequences, the mobilization of financial, industrial, and institutional resources to meet demand increase of these absolute amplitudes will constitute a daunting societal challenge.

Given the total energy demand projections, the fractions supplied by fossil, renewables and nuclear were also projected by the WEC/IIASA, as a part of the pattern projections. Nuclear energy's role in meeting the demand for energy services was confined to the electricity sector only (which currently comprises about 1/3 of total energy needs and which was projected by the IIASA/WEC as growing to about ½ over the period of the study). No demand was placed on nuclear for providing additional energy services such as water desalinization, district heating, process heat, etc. Moreover, although technological innovation was presumed to drive the economic development in every case, the use of hydrogen as an energy carrier was modeled in the IIASA/WEC projections as occurring very late in the century; the use of nuclear heat or

electricity to manufacture hydrogen from water or hydrocarbons was not accounted for at all. In light of this and the envisioned role of Generation-IV to expand the scope of nuclear to non-electric applications, the projected nuclear share from the IIASA/WEC projections might turn out to be larger – if a broadened role for nuclear should be realized.

For the purpose of the Fuel Cycle Crosscut Group's scenario evaluations in Chapter 3, the IIASA/WEC Case B has been used as the base case and Case C2 has been used to provide a check on sensitivity of conclusions to demand growth rate. Case B is neither as technologically optimistic as the Case A scenarios nor as institutionally optimistic as the Case C scenarios, and was generated assuming a ragged dispersion of development rates in various developing countries. *In Case B the nuclear installed capacity grows from about 380 GW_e in 1990 to about 800 GW_e in 2020; to about 2000 GW_e in 2050 and to about 5500 GW_e in 2100.*

Alternately, the Case C2 scenario has nuclear growing to 1200 GWe by 2050 and to 1800 GWe by 2100 – thus providing a good spread from Case B for the purpose of assessing sensitivity of results for Generation-IV deployment strategies to assumed growth rate of demand.

Figure 1.11 displays the IIASA/WEC projected nuclear demand growth subdivided into industrialized and developing world regions – indicating that, as with energy in general, the developing world (in percentage growth rate) will dominate future demand. *During the time of Generation-IV introduction (2030-2050) the absolute sum of energy supply investments in developing countries will be as large as those in industrialized countries. After mid century the deployments in the currently industrialized countries will lie in the minority of overall growth*

In light of the difficult challenge presented by accurate economic forecasting, it is important to assure that the two nuclear energy demand projections taken from the IIASA/WEC study and used for Generation-IV scenario evaluations span the realm of consensus thinking of energy planners representing a range of viewpoints. The IIASA/WEC process itself sought to accomplish this through their use of seven separate scenarios and through their process of review by regional experts. As shown in Figure 1.12, the outcome is a significant variability in potential nuclear capacity growth over the next century. The case B base case for nuclear shows similar growth as the more aggressive case A patterns, but it shows significantly more growth for pattern B than the case C patterns.

The two nuclear growth rates drawn from the IIASA study for use in Generation-IV scenarios can be compared to the range being used by the United Nations in its studies of global climate change. Figure 1.13 shows the range of nuclear capacity reported in their Special Report on Emissions Scenarios (SRES). Their cases extend to very much

higher but not significantly lower[2] than the cases we have selected to use for Generation-IV scenario evaluations.

Projections of 50 to 100 years in advance are always to be viewed skeptically – too often they prove inaccurate within even a decade – yet they drive the market demand for nuclear energy services. *For this reason the Generation-IV Roadmap must be structured to produce a technology slate which is robust with respect to uncertainties in future energy demand – both absolute and rate of growth. On the one hand the technology should be able to penetrate the market incrementally without necessity of huge up-front expenditures; on the other hand the technology should not be innately constrained to some upper bound on scale of deployment.*

[2]The IIASA Case C1 is a nuclear phase out scenario. In light of Generation-4 sustainability goals, such scenarios are not relevant to the Roadmap planning. Note that the SRES nomenclature for their cases uses designations similar to the IIASA nomenclature – but with different meaning.

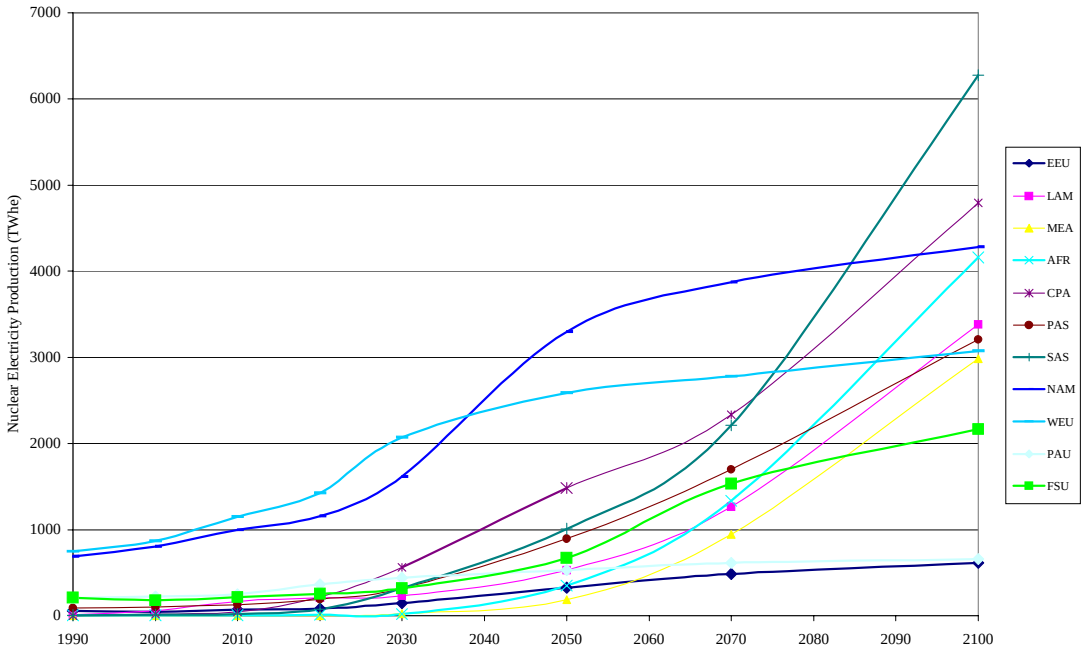


Figure 1.11(a) Geographical Distribution of Nuclear Electricity Demand (TWhe/yr) in the Scenario B Case[3]

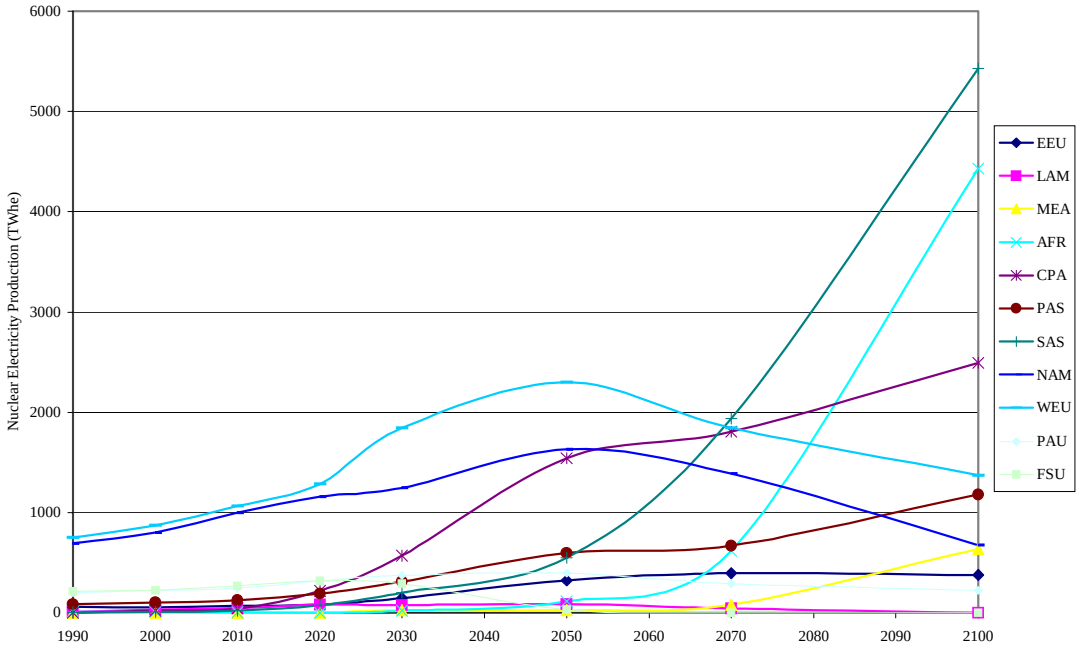


Figure 1.11(b) Geographical Distribution of Nuclear Electricity Demand (TWhe/yr) in the Scenario C2 Case

[3] EEU = Central & Eastern Europe; LAM = Latin America; MEA = Middle East & North Africa; AFR = Sub-Saharan & Southern Africa; CPA = Centrally Planned Asia & China; PAS = Pacific OECD (Japan, Australia, New-Zealand); SAS = South-East Asia; NAM = North America; WEU = Western Europe; PAU = Other Pacific Asia; FSU = Former Soviet Union.

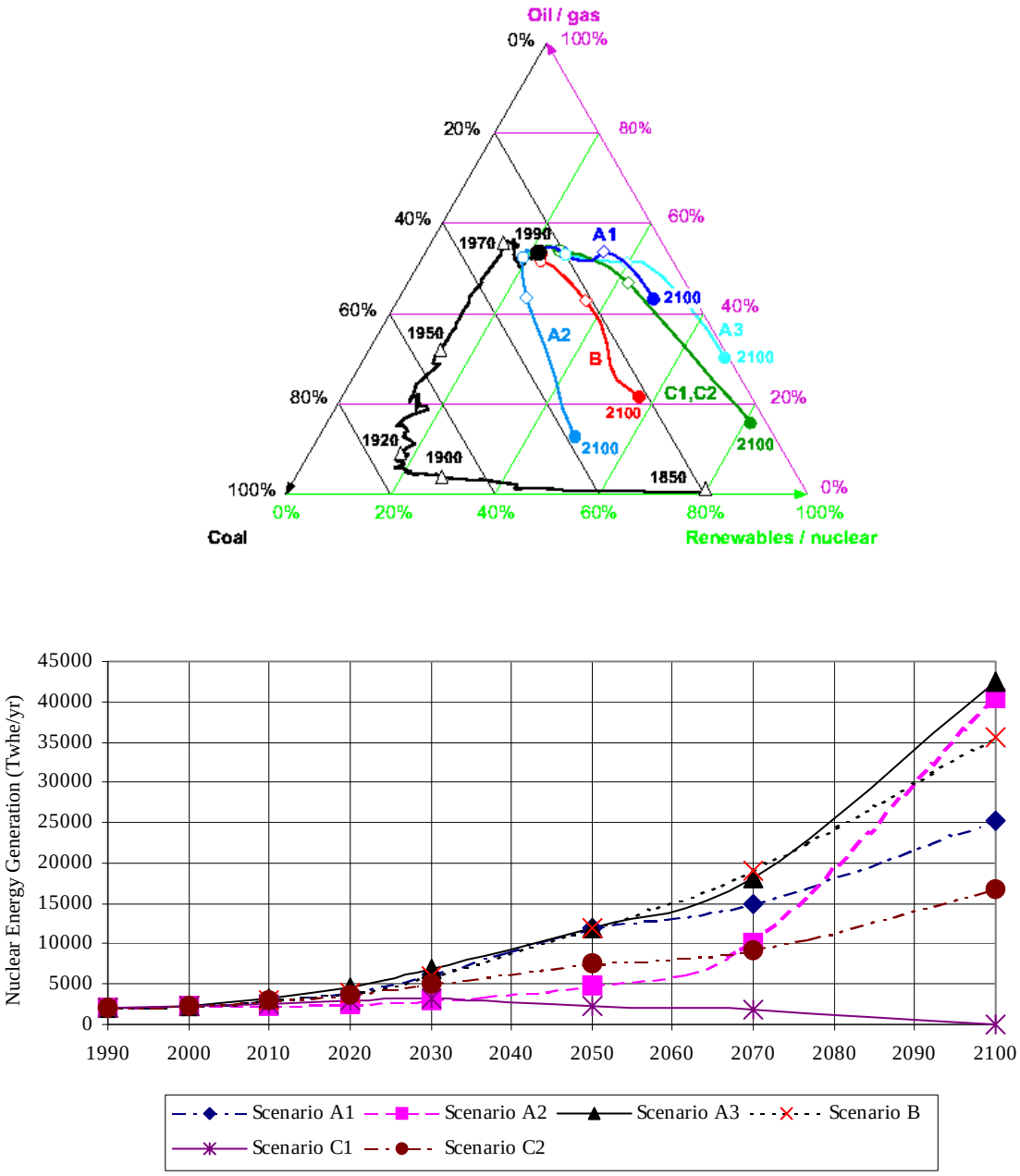


Figure 1.12 Six Cases of Nuclear Electricity Generation in the IIASA/WEC Study

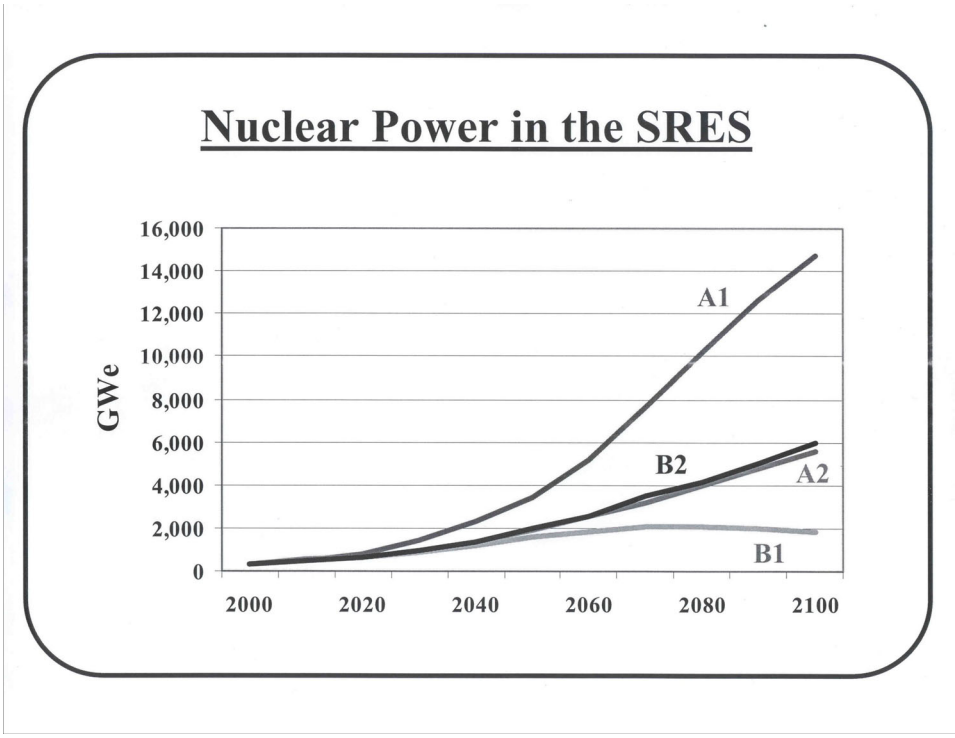


Figure 1.13 Nuclear Power in the Special Report on Emissions Scenarios

Table 1.10 World Regions Used by the IIASA/WEC to Produce Their Aggregate World Energy Demand Growth Scenarios Study

OECD		
	NAM	N. America
	WEU	W. Europe
	PAO	Pacific OECD (Japan, Australia, New Zealand)
REF		
	EEU	Central and Eastern Europe
	FSU	Newly Independent States of Soviet Union
DC		
	LAM	Latin America and Caribbean
	MEA	Middle East and North Africa
	AFR	Sub Saharan Africa
	CPA	Centrally Planned Asia and China (includes, Korea)
	SAS	South Asia
	PAS	Other Pacific Asia

Table 1.11 Summary of the Three Cases in 2050 and 2100 Compared with 1990

	Case		
	A High growth	B Middle course	C Ecologically driven
Number of scenarios	3	1	2
Population, billion			
1990	5.3	5.3	5.3
2050	10.1	10.1	10.1
2100	11.7	11.7	11.7
GWP, trillion US(1990)$			
1990	20	20	20
2050	100	75	75
2100	300	200	220
Global primary energy intensity improvement, percent per year	Medium	Low	High
1990 to 2050	-0.9	-0.8	-1.4
1990 to 2100	-1.0	-0.8	-1.4
Primary energy demand, Gtoe			
1990	9	9	9
2050	25	20	14
2100	45	35	21
Resource availability			
Fossil	High	Medium	Low
Non-fossil	High	Medium	High
Technology costs			
Fossil	Low	Medium	High
Non-fossil	Low	Medium	Low
Technology dynamics			
Fossil	High	Medium	Medium
Non-fossil	High	Medium	High
Environmental taxes	No	No	Yes
CO_2 emission constraint	No	No	Yes
Net carbon emissions, GtC			
1990	6	6	6
2050	9-15	10	5
2100	6-20	11	2

Abbreviations: GWP – gross world product; Gtoe – gigatons oil equivalent; CO_2 = carbon dioxide; GtC = gigatons of carbon.

1.5.3 The Role of Processing in Separating Resources and Wastes

Given the ore availability constraint and the energy demand driving function, the system which transforms ore to energy and waste is comprised to two main elements; nuclear power plants where mass is transmuted by nuclear processes to energy, fission products, and higher mass actinide elements and fuel cycle process plants which convert ore to fuel and convert discharged fuel to waste and recycle constituents by chemical processes.

The following paragraphs summarize the role of processing in the fuel cycle. For simplicity the discussion is limited to uranium based fuels. There are direct parallels with the processing of thorium and the transuranics but these will not be covered here other than by inference. (They are covered in depth in Chapter 4 and Attachment 4.)

Uranium is the heaviest naturally occurring element, and is a very reactive metal. The thermodynamic stability of its compounds places it in a chemical processing category similar to aluminum, titanium and magnesium.

Processing first refines the uranium ore so that it will be in a chemical, isotopic and physical form suitable to fuel a reactor, and so that following the release of energy the spent fuel can be discarded intact or can be processed so that the unused uranium and the newly formed transuranics can be recovered and fabricated into new fuel.

We divide the processes for uranium into two broad categories: 1) pre-irradiation processes and 2) post-irradiation processes, i.e., processing associated with fuel recycle and refabrication and processes associated with spent fuel and waste management.

Pre-irradiation Processing

The pre-irradiation processes encompass the many steps required to convert and refine uranium from the naturally occurring minerals in the earth's crust into a high-fired UO_2 fuel pellet, or other fuel form for use as fuel in a nuclear reactor. One process is isotope enrichment. Uranium has a natural isotopic distribution of ~99.3% ^{238}U and 0.7% ^{235}U, but for typical fuels, the uranium is moderately enriched up to 3 to 6 % ^{235}U to facilitate criticality and sufficient in-reactor residence time. Presently commercial isotopic enrichment is accomplished through isotopic separation of UF_6 by gaseous diffusion and by other techniques such as ultracentrifuge. The uranium is converted to uranium tetrafluoride UF_4 as an intermediate compound then to uranium hexafluoride UF_6 for isotope separation before being reconverted to uranium metal or uranium oxide. Isotopic separation of uranium metal vapor using lasers as in the AVLIS process was developed up to a point, but for various reasons its industrial development has been recently postponed or even abandoned in several countries.

Processes for isotopic separation and their efficiency will become important if the new Generation-IV reactor types require high enrichment for efficient operation. If the enrichment level required is <5% then new technology is not a pressing need. If

Generation-IV fuel cycles that require significant enrichment are chosen, then improved (lower cost) technology may be needed.

Post Irradiation Processing

In commenting on the role of recycle in the fuel cycle it is useful to refer to Figure 1.3 which illustrates alternative generic nuclear fuel cycles:

Once Through,
Partial Recycle,
Full Fissile (Pu, ^{233}U) Recycle, and
Full Actinide Recycle.

The post-irradiation processes are those technologies applied to discharged spent nuclear fuel at the "back end" of the nuclear fuel cycle. In the United States, the nuclear fuel cycle is as a "once-through" process. In this process the fuel cycle is open and there is a complete decoupling between fresh and discharged fuel; the discharged fuel is simply treated to put it into a suitable form for disposal. Current commercial nuclear fuel is deemed suitable for direct disposal, but new fuel types may require additional waste treatment.

The three generic closed fuel cycles (partial recycle, full fissile (Pu, ^{233}U**) r**ecycle and full actinide recycle) differ only in degree of recycle and the specifics of the material recycled. The intent in all cases is to recover some or all of the residual fuel, to first remove some or all of the fission products for disposal as waste, and then to refabricate and return some or all of the actinides (which had been formed by irradiating the fuel) back into the reactor power plant – to extract further energy and to reduce the mass of waste that finds its way to the repository.

There are six basic chemical separation technologies applicable for use in the recycle processes, solvent extraction, ion exchange, pyrochemical reduction, electrorefining, fluoride volatility, and plasma ion separation, and of course combinations of all five. The comments below are not intended to be exhaustive.

Solvent Extraction

Solvent extraction is the process most commonly used to close the fuel cycle. This process exploits differences in thermodynamic stability between complexes of the actinides. Using multiple extractions the concentration of a particular element can be reduced or increased by a factor of more than 10,000. Outside the United States, the PUREX method is mostly used to close the fuel cycle. In PUREX, spent fuel is dissolved, and usable fuel (typically >90% of the initial fuel) is extracted and recycled back into the pre-irradiation processing while fission products are extracted and processed into disposal waste forms.

Ion Exchange

The chemistry of ion exchange and solvent extraction are quite similar. Multiple exchanges between ions chelated to surface sites and those in solution exploit small differences in thermodynamic stabilities to concentrate specific elements on the surface of an ion exchange resin. With careful control of surface chemistry and solution pH the ion exchange can become highly specific, e.g., removing calcium from solution but not potassium or barium. Change the pH and the calcium can be easily stripped from the resin.

Ion exchange processes are not as useful as solvent extraction as in general they can not accommodate large volumes, and they operate with less efficiency and at a higher cost.

Pyrochemical Reduction

Pyrochemical reduction is used to reduce discharged oxide fuel, whether uranium or mixed metal, to a metal. The oxide is chemically reduced using an alkali or alkaline earth metal such as lithium (Li) or calcium (Ca) via the reactions shown below.

$$4Li + UO_2 \rightarrow 2Li_2O + U$$
$$2Ca + UO_2 \rightarrow 2CaO + U$$

This reaction occurs in a molten halide salt in which the resulting lithium or calcium oxide is soluble. Recycle of the lithium or calcium is desirable to reduce generation of secondary waste and it is achieved simply by electrolysis of the dissolved lithium oxide or calcium oxide. Electrolysis of the dissolved oxides results in deposition of lithium or calcium at a cathode and evolution of oxygen gas at an anode.

Electrorefining

The electrometallurgical process was adopted as a central part of Argonne's Integral Fast Reactor (IFR). Electrorefining was the core separation technology of that process. In an electrorefiner, chopped metal fuel is loaded into anode baskets and submerged in a molten salt along with a cathode. When current is passed between the anode baskets and a cathode, uranium and all components of the fuel that are less noble than uranium are oxidized and dissolve in the molten salt. At the cathode, uranium, which is the least noble species dissolved in the molten salt, is electrodeposited. Using a liquid metal cathode, plutonium and the minor actinides are electrodeposited along with the uranium. The active metal fission products accumulate in the molten salt and are eventually sorbed into zeolite and converted into a glass-bonded sodalite waste form. The noble metal fission products and the cladding that remain in the anode baskets are melted into an iron-zirconium alloy metal waste form.

Fluoride Volatility

Where electrorefining relies on differences in redox chemistry for achieving separation of uranium and transuranics from fission products, fluoride volatility relies on differences in the vapor pressures of fluorinated components of spent fuel. Fluorination of the spent fuel results in volatile actinide fluorides as well as a few volatile noble metal fluorides. The volatile species that are formed are sufficiently different in vapor pressure that the necessary separations can be made. The process has been successfully demonstrated on the pilot scale and is the subject of ongoing research. Fluoride volatility is compatible with the present fuel fabrication technology, which uses uranium hexafluoride (UF6) as the initial uranium feed material.

Plasma Ion Separation

Plasma-ion separation is a process recently proposed and under development for processing of Hanford high-level tank waste oxides using a coarse ion separator. Such a system might also be used to process uranium oxide spent fuel by vaporizing and ionizing all of the metal ions and separating the uranium and actinides from the lower mass fission products. This technology is not effective in separating uranium from other actinides, but the actinide/uranium product might then be blended into a MOX-like fuel. This process has the potential to achieve high separation efficiency for actinides and enable capture of all fission products including Kr-85 and tritium as waste.

Summary

Of these post irradiation processing recycle technologies for segregating wastes from resources, only PUREX solvent extraction is currently operated on a commercial scale. Generation-IV advanced fuel cycles based on the other identified technologies are proposed – motivated by Generation-IV goals for economics, sustainable waste management and proliferation resistance.

1.5.4 Waste Management from The Fuel Cycle

The nuclear fuel cycle which transforms ore into energy unavoidably produces residual and waste products as well. The objective of sustainable development with respect to these products is to (1) manage the residuals and to (2) dispose of the wastes while protecting humans and the environment. Depending on the fuel cycle strategy chosen, residual products may or may not be considered as final waste (for instance, SNF is considered a waste product in the once-through fuel cycle but not in the full actinide recycle fuel cycle). In every case, however a sustainable nuclear energy system will strive to minimize the amount of wastes produced in the system and strive to make optimal use of the residual products before they are to be disposed of as waste.

Both chemical and radioactive waste must be considered in a sustainable waste management policy. Some potential radioactive wastes may be chemical wastes as well – in which the chemical toxicity exceeds the radiotoxicity. (This is for instance the case for depleted uranium if treated as waste.)

There are essentially three fundamental waste management strategies:

- *Destruction.* Many chemical wastes are destroyed by incineration or other techniques. Proposals have been made to transmute long-lived radionuclides to stable or short-lived radionuclides.
- *Containment*. Radionulides can be isolated until they decay. The isolation period depends upon the half-life. Some medical wastes are stored until they are non-hazardous. In the United States, shallow-land burial is allowed for wastes that decay in limited periods of time (100 years for Class A and Class B wastes: 500 years for Class C waste). Geological disposal is required for long-lived radionuclides. Containment is also used for long-lived chemical wastes (heavy metals). There are several operating geological repositories for chemical wastes in Europe and one currently operating repository for long-lived radioactive wastes (WIPP) in the United States.
- *Dilution*. Hazardous materials can be diluted to safe levels. Whether one considers chemical releases (sulfur dioxide, carbon dioxide, etc.) or radioactive releases (tritium, krypton-85, etc.), there are limits to what the environment can safety accept. Therefore, after a critical energy generation scale is reached, releases per unit of energy produced must decrease as the scale of operations increases further.

The most radioactive or chemically hazardous wastes may not necessarily create the primary risks. With the current once-through fuel cycle, the major radiological exposures are in the mining and milling of uranium. Although the radioactivity of the mill tailings is low compared with that of SNF, geological disposal of the SNF and HLW is many orders of magnitude better at isolating radionuclides than shallow-land disposal of uranium mill tailings. The environmental and public health impacts of a fuel cycle in which almost all of the radioactivity is sent to a repository may be significantly less than

those of a fuel cycle with lower total levels of radioactivity but in which a larger fraction of the radioactive wastes (mill tailings, etc.) goes to shallow-land burial.

Beyond the technical considerations, complex political, social, and equity issues exist as well. Many human risks are immediate: a vehicle driver makes a mistake, an accident occurs, someone is hurt, someone is at fault, and actions can be taken to correct the problem. In other cases those who benefit from activities, say the use of electricity from coal, may not bear the full costs of the pollution, which affects populations remote from the plants. Many of the risks arising from management of chemical, elemental (heavy metals), and radioactive wastes also raise issues of intergenerational equity. Such wastes present long-term hazards as well as operational ones. It may take years or centuries for a waste form to degrade, the containers to corrode, and the radionuclides to be transported to man via groundwater. The individuals who benefit from the energy produced may be different from those who are at risk from waste management activities. Relatively simple packaging can minimize near-term risks and postpone risks into the future. Many waste management activities are more controversial because they involve political and philosophical questions that do not arise in many other industrial activities.

All the fuel cycles have the common front-end activities of mining and milling. Options to alter the waste management characteristics of front-end operations rely on separation and geological disposal of the natural longer-lived decay products in mill tailings, thus reducing accessibility of humans to this source of radiation.

Each of the four classes of fuel cycles provides different back-end waste management options.

- *Once through*. This option involves the minimum processing of the SNF for disposal.
- *Partial recycle*. Selected actinides are once or several times recycled and others are sent to waste in this option. Reduction of the fissile mass sent to the repository reduces safeguards and criticality issues within the repository. Partial recycle may increase or reduce the long-term radiotoxicity of the waste depending on how it is accomplished.
- *Full fissile recycle*. Recycling of the SNF provides multiple options to improve waste management. The fissile content of the waste is reduced by one to two orders of magnitude, thus significantly reducing the safeguards and criticality issues in waste management and at the repository. The radiotoxicity of the waste is also reduced.
    - Waste form. The chemical form and geometry of the waste form can be optimized to maximize performance. In some cases, the requirements for power production (high-surface-area SNF for heat transfer) are opposite those desired for a waste form (low surface area for low groundwater dissolution rate). There are very large differences in the expected behavior of different types of SNF in a repository environment; thus, there are large differences in the waste management benefits of processing

SNF. The SNF behavior also partly depends upon the particular repository geochemistry.

- Repository design. The wastes can be fractionated into separate streams to create alternative repository options. Current repository designs distribute waste packages over large areas to minimize the temperature rise from radioactive decay heat. The increased temperature from radioactive decay heat can (1) degrade the waste form, waste package, and local geology; (2) alter groundwater flow; and (3) increase uncertainties in the repository performance. If the major heat-producing fission product radionuclides (^{90}Sr and ^{137}Cs) are separated and managed individually, the repository size decreases. Repository performance may improve by reducing the temperatures in the repository that degrade waste forms and geology.

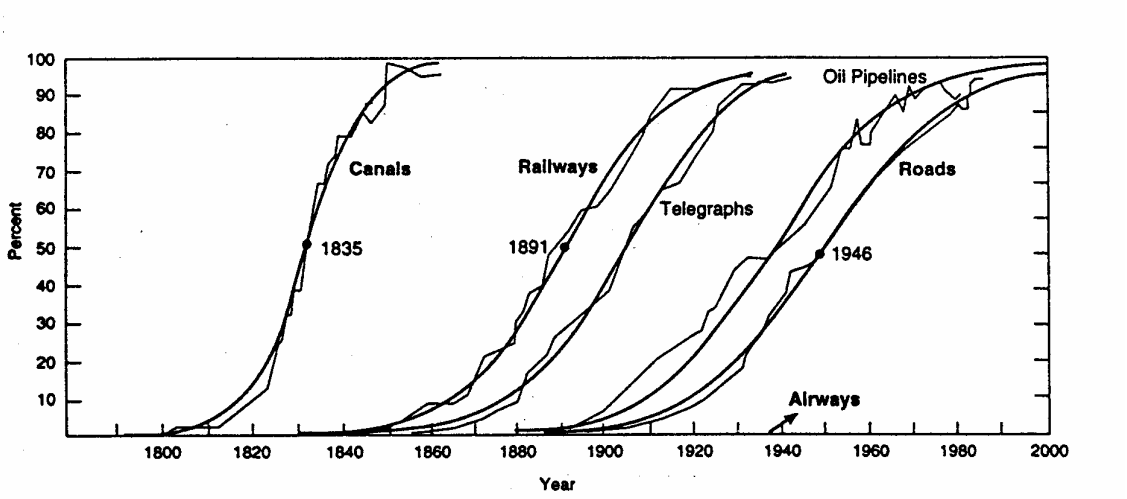
- *Full-actinide recycle.* Recycling of the SNF with recycle of all fissionable materials (plutonium, ^{233}U, and minor actinides) provides additional repository options beyond those associated with full fissile recycle. Safeguards and criticality issues are, for practical purposes, eliminated. Waste toxicity can be reduced dramatically. The options that were defined for disposal of HLW expand. Radically different repository design options become available. With removal of plutonium and the minor actinides, the only remaining heat generating radionuclides in the wastes are cesium and strontium. With their removal and separate management, the remaining radionuclides can be treated as low-heat wastes. This allows lower cost repository designs and potentially higher performance by avoiding the adverse impact of decay heat in a geological repository.

There are waste management trade-offs between fuel cycles. In most cases, additional processing increases near-term risks with the potential for reducing long-term risks. Waste management costs can vary as well. These issues must be addressed and balanced when considering Generation-IV fuel cycles.

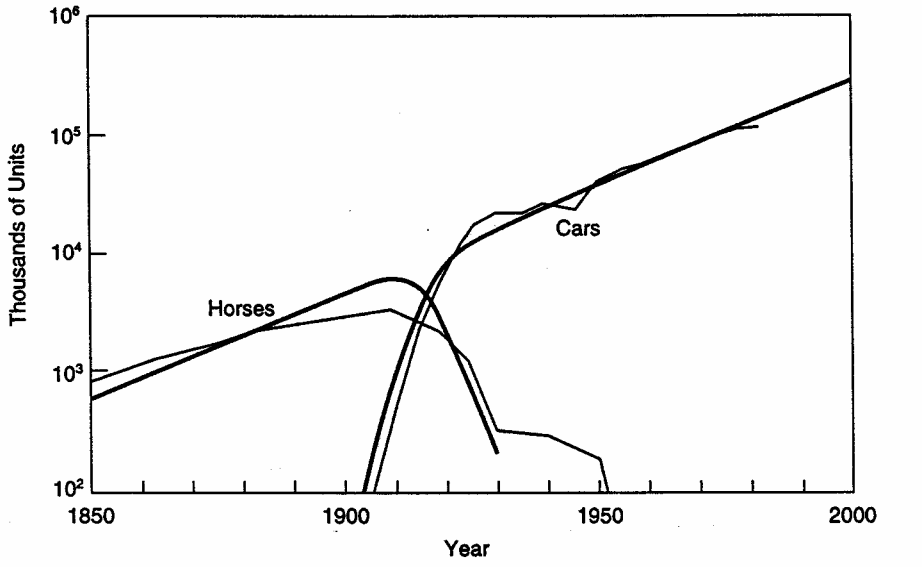
1.5.5. Dynamic Response Capability of The Nuclear Fuel Cycle To Follow Demand Changes

The nuclear energy enterprise is a dynamical system; it is driven by external demand for energy; it is constrained by availability of fuel and internal mass flows; it evolves from the current deployments and it incorporates inertial elements such as time delays to build assets and long asset lifetimes which carry the memory of previous conditions. The 30 year lead time to Generation-IV deployment recognizes the multi-decade time constant which is characteristic of innovation, development, and market penetration trajectories for major infrastructure deployments. Fig. 1.14 illustrates that this 3 or 4 decade time constant has displayed itself repeatedly – it is an innate characteristic of technological diffusion in society, and its underlying causes are comprised of physical, institutional, and societal inertia elements.[4]

(i)



Growth of infrastructures in the United States as a percentage of their maximum network size. SOURCE Grübler and Nakićenović (1991).



Number of nonfarm draft animals and automobiles. SOURCE: Nakićenović (1986).

Figure 1.14 Time Evolution of Major Infrastructure Development

The FCCG systems studies will develop and display several potential nuclear energy futures based on alternative Generation-IV fuel cycles. These projections will be based on physical inertia aspects of Generation-IV deployment i.e., they will be energy demand-driven, and actinide resource–constrained; the scenarios will not attempt to model the institutional and societal (including economic) inertial elements of actual societal behavior. *The scenarios developed therefore will illustrate what could be physically achievable energy supply capacities constrained only by the resource base and physical lag times under various assumed architectures for Generation-IV fuel cycles.*

For Generation-IV nuclear fuel cycles, the components of the system physical inertia derive from three general categories:

(i) lead time for construction of capital assets,
(ii) buildup and die away time constants for working inventories of in-process materials,
and
(iii) lifetimes of capital assets prior to their obsolescence (from either physical deterioration, from technological obsolescence, or from loss of cost competitiveness in the face of new economic and/or regulatory conditions).

Given the assumption that the technologies are ready for commercial deployment, and that vendors are positioned to fill orders, the construction lead times for fuel cycle capital assets, for reactors themselves, and for other deployment lag times are based on historical precedent and are displayed in Attachment 2. As to assets lifetime to obsolescence, the assumed power plant design life is targeted for 60 years. Fuel cycle facilities are assumed to have a 60 year life also.

The buildup and decay times of material in process working inventories are strongly influenced by the assumed market shares of Generation-IV concepts being modeled and their individual intrinsic properties. They depend on such intrinsic attributes as reactor specific inventories $\left(\frac{\text{kg fissile}}{\text{MW}_t} \text{ and } \frac{\text{kg HM}}{\text{MW}_t}\right)$; on discharge burnup $\left(\frac{\text{MW}_t \text{ days}}{\text{kg HM}}\right)$; neutron excess and resulting fissile conversion efficiency $\left(\frac{\text{kg fissile produced}}{\text{kg fissile fissioned}}\right)$ and on reactor refueling interval and refueling batch fraction (refueled disassemblies/total assemblies). They also depend on the out-of-reactor dwell times such as the lag times from fabrication to core loading; from unloading to recycling and from recycling to refabrication. They depend also on recycle losses as a fraction of throughput.

The resulting overall cycle working inventories when divided by cycle mass flows produce a time constant which characterizes the system dynamic response. When a

scenario is to represent an ensemble of several reactor types and fuel cycle types which together meet energy demand, and which (perhaps) symbiotically exchange mass flows, the dynamic response time constant depends on the energy fraction delivered by each system type (after internal mass balance has been satisfied, and considering the startup transient from consumption of pre-existing inventories such as LWR spent fuel).

A convenient overall figure of merit to characterize dynamic response of a fuel cycle enterprise is its system doubling time – how long it takes after a decision to do so for the candidate Generation-IV energy park to double in energy output capability – with account taken of all construction lag times and of the time required for building up the pipelines of internal working inventories. If the system is not depending on an outside source of fissile material, but is self generating its fissile, then the doubling time also depends on the overall neutron excess for the mix of plants which are deployed.

While steady-state or constant period demand growth scenarios for a single or for a specified mix of Generation-IV systems can be used to illustrate relative advantages and disadvantages of candidate Generation-IV systems, it will be essential to consider *transition scenarios* wherein Generation-IV systems penetrate the ongoing existing nuclear energy enterprise in the decades around 2030. In that case, transient shifts in the already-existing inventories of materials in process will take place. At the time of Generation-IV market penetration, three sources of actinide resource will be available from which to draw "fuel": uranium (and/or thorium ore) in the geosphere; uranium and transuranics present in fuel assemblies already discharged from today's nuclear enterprise which have been held in interim storage; and uranium enrichment tails held in interim storage at the world's enrichment plants. (In fact, a relatively small additional source derives from uranium and plutonium declared excess from nuclear weapons programs, but these weapons-program released stores are assumed to have been already consumed prior to Gen-IV introduction, and will be ignored in the scenario development.)

1.5.6 Summary

The Generation-IV fuel cycle systems studies will use dynamic modeling to evaluate candidate global nuclear energy parks which:

- Evolve from current deployments
- Meet projected global energy demand for nuclear energy
- By drawing on ore reserves
- With resulting waste generation
- And within physically achievable dynamic response times of the overall nuclear energy parks.

Opportunities for symbiotic mixes of Generation-IV concepts will be evaluated along with individual concept types so as to provide the Generation-IV Roadmap with insights regarding achievability of Generation-IV sustainability goals. The scenarios considered by the FCCG are presented and discussed in Chapter 3; they all consider the transitions from existing conditions to the eventual mid to late century Generation-IV nuclear energy park.

References for Section 1.5

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2. E. Bos, M.T. Vu, and R.A. Bulaloo, World Population Projections 1992-1993, Johns Hopkins University Press, (1992).
3. Intergovernmental Panel on Climate Change (IPCC), Special Report on Emission Scenarios (Figure 1.13 taken from Hans-Holger Rogner, Plenary Lecture, American Nuclear Society Winter Meeting, Reno, NV, Nov. 11-15, 2001).
4. A. Grübler, "Time For A Change: On the Patterns of Diffusion of Innovation," pp.14-32 in Technological Trajectories and The Human Environment, J. Ausubel and D. Langford, eds. National Academy Press, (1997).

1.6 Outline of the FCCG Report Contents

Chapter 2 of the report briefly summarizes the current scale of deployment of commercial fuel cycle facilities which service the world's existing nuclear power plants. These plants are currently providing ~350 GWe of installed capacity and are producing ~17% of the world electricity. In 1999, about 2,400 Terrawatt electric hours were delivered by nuclear plants worldwide. Chapter 2 also provides a synopsis of the scope of differing fuel types and needed fuel cycle services for the full range of candidate Generation-IV concepts.

Chapter 3 presents the results of numerical simulations of postulated scenarios for Generation-IV global market penetration. The scenarios are run as 100 year simulations with two alternative WEC/IIASA nuclear energy demand scenarios used as the forcing function and with internal mass flows and availability of ore as the constraints. Edits of ore cumulative drawdown and cumulative waste arisings are provided relevant to the Generation-IV sustainability goals. The scenarios are organized on the basis of the four generic fuel cycles delineated in Figure 1.3 – once through, partial recycle, full fissile recycle, and full actinide recycle.

Chapter 4 presents a summary review of the current status of fuel cycle technology and the directions of current fuel cycle R&D. A separate report carries a quite detailed discussion of the subjects which are summarized in Chapter 4.

Chapter 5 presents a synopsis of the status of the institutional regime which provides both enabling structure and boundary conditions for the globalization of nuclear energy in the future.

Chapter 6 collects and summarizes the lessons learned from reviewing status of R&D worldwide and from insights which were gained from the scenario outcomes – regarding efficacy of various generic fuel cycles in meeting Generation-IV goals. Fuel cycle crosscutting R&D is identified – drawing on the lessons learned in Chapters 3 and 4 – and prioritized recommendations are made for Generation-IV fuel cycle crosscutting R&D.

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 2

Current Fuel Cycles and Proposed Generation-IV Fuel Cycles

March 18, 2002

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Chapter 2

Table of Contents

2.1 Overview: Segments of The Fuel Cycle

The nuclear fuel cycle consists of a sequence of steps (see Figure 1) required to harvest and refine uranium ore, enrich it in the U235 isotope, convert it to fuel, convert fuel to heat and heat to electricity at the power plant, and then manage the discharged fuel. While there are some differences among cycles for LWR's, CANDU's and Magnox, only two generic cycle options dominate the commercial deployment of nuclear energy.[1] These fuel cycles diverge in character as to how the discharged fuel is managed. For the once-through cycle the discharged fuel is put into interim storage awaiting eventual disposal in a geologic repository – see Figure 2. The second option is the mixed plutonium uranium oxide (MOX) mono recycle option wherein the discharged fuel is recycled into three product streams: recovered irradiated uranium is set aside in interim storage, fission products and minor actinides are converted to a glass waste form for disposal and plutonium is recovered and used to fabricate MOX fuel for a single return pass through a reactor – after which the discharged MOX fuel is placed in interim storage awaiting further disposition options – see Figure 3. The uranium oxide (UOX) once-through cycle has been in growing commercial deployment for thirty years; the MOX mono-recycle option has gone into commercial deployment in Europe during the 1990's and is planned for Japan in the near future.

The scale of deployed fuel cycle support facilities is described in the next section. This deployment provides the technological foundation upon which future Gen-IV deployments will build.

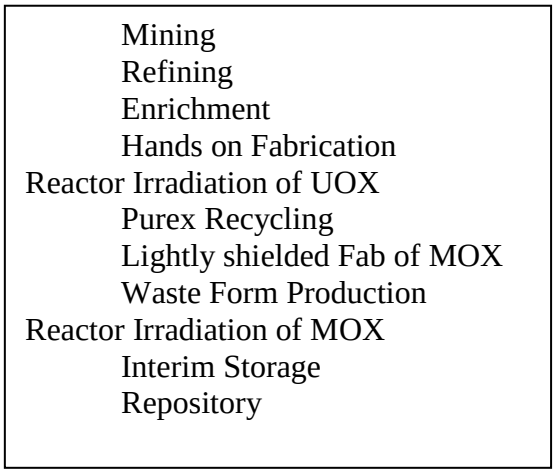

Fig. 2.1 Links in the Fuel Cycle Chain

[1] Limited deployment of Thorium-based fuel cycles have been initiated but have not achieved significant market share.

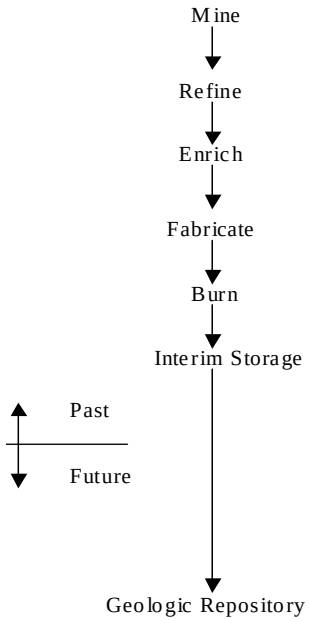

Fig. 2.2 The Once-Through UOX Cycle

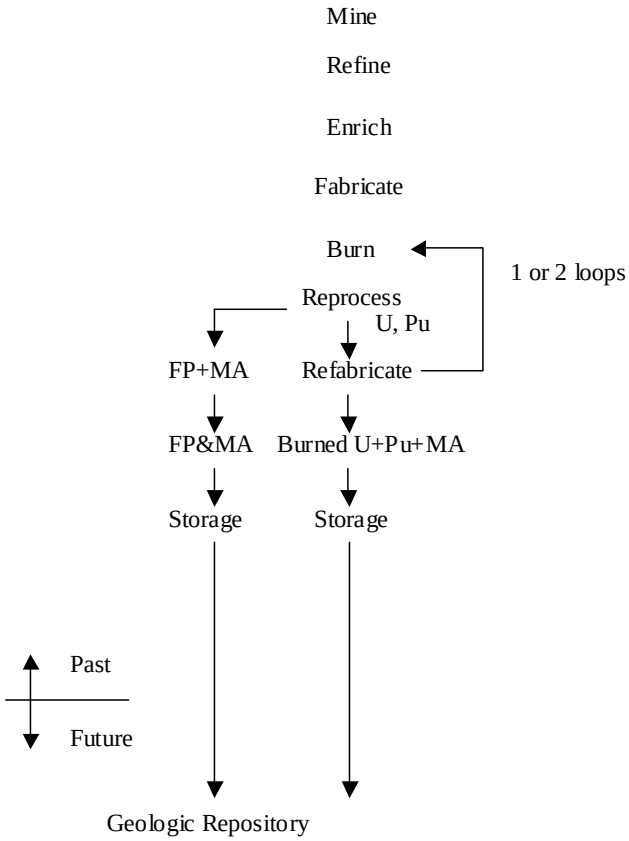

Fig. 2.3 The LWR MOX Mono Recycle Cycle

2.2 Current Worldwide Deployments of Fuel Cycle Infrastructure

The currently deployed nuclear power plants – supplying 17% of the world's electricity – are supported by an infrastructure of commercial fuel cycle facilities which are sited primarily but not exclusively in industrialized OECD nations and Russia.

Conversion

While the result of the first steps of the fuel cycle results in a "yellow cake", i.e. U_3O_8 concentrate, the second main step yields a high purity uranium product. The chemical and physical form of the converted product depends on the intended further use of the uranium, i.e. fuel type, and the technologies used for enrichment of the uranium. One can distinguish between conversion to metallic or oxide (non-enriched) uranium for use in Magnox and CANDU reactors and the conversion to uranium hexafluoride (UF_6) intended for enrichment with today's enrichment processes.

The conversion from yellow cake to UF_6 is achieved by dissolving the uranium concentrate in nitric acid, filtering and treating the solution with chemical solvents. The resulting uranyl nitrate (>99.95% pure) is then reconverted to uranium oxide and this, in turn, is converted to readily vaporisable uranium hexafluoride[2] used in the enrichment process. If enrichment is not required, for example for CANDU-fuel, then uranium dioxide is produced from the uranyl nitrate and shipped directly to a fuel fabrication plant.

Conversion plants are specialized chemical facilities and, because they handle very aggressive chemicals (F, HF), they are subject to rigorous operational conditions. They do not, however, produce significant amounts of radioactive effluents (principally containing natural uranium α-activity).

Re-conversion of depleted uranium hexafluoride to U_3O_8 currently is practiced in France and planned in the USA in order to cope with the growing stock of depleted uranium hexafluoride, taking the view that storage as stable uranium oxide is more appropriate for the very long term.

Table 1 gives an overview of the major commercial conversion plants and processes in use today or planned in the near future[3]. Some other conversion plants are mentioned, that are mainly annexes to the commercial plants and serving specific conversion needs.

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2. At ambient temperature, UF_6 is a colourless, high molecular weight (352) solid with a significant but less than atmospheric vapour pressure. It is readily transformed into a gas at atmospheric pressure by raising its temperature above 56.4°C and into a liquid by increasing the pressure above 1.5 atmospheres and the temperature above 64°C. All three phases, solid, liquid and gas, coexist at 64°C.

3. Argentina, Brazil, China, Korea, Pakistan and Japan have conversion plants as well.

Table 2.1 Conversion Plants and Processes

Country	Place	Operator	Capacity (tU/year)	Conversion process	First operation
Canada	Blind River	Cameco	18 000	$U_3O_8 \Rightarrow UO_3$, wet, UNH-process	1983
	Port Hope	Cameco	2 000		1985
	Port Hope	Cameco	10 500	$UF_4 \Rightarrow$ U-metal	1984
	Port Hope	Cameco	3 800	$UO_3 \Rightarrow UF_6$	1980
				$UO_3 \Rightarrow UO_2$	
France	Malvési	Comurhex	14 000	$U_3O_8 \Rightarrow UO_4$, wet, ADU-process	1959
	Pierrelatte	Comurhex	350		1976
	Pierrelatte	Comurhex	14 000	$UNH \Rightarrow UF_6$	1961
	TU 2	COGEMA	350	$UF_4 \Rightarrow UF_6$	1988
	TU 2 Recycling	COGEMA	1 200	Depleted $U_3O_8 \Rightarrow UO_2$ (MOX top-up)	1988
	TU 5 Recycling	COGEMA	2 000		1995
	W	COGEMA	20 000	$UNH \Rightarrow U_3O_8$	1984
	Defluorination			$UNH \Rightarrow U_3O_8$	
				$UF_6 \Rightarrow U_3O_8$	
Russian Federation	Angarsk	Minatom	20 000[4]	$U_3O_8 \Rightarrow UF_6$	1954
	Ekaterinburg	Minatom	4 000	$U_3O_8 \Rightarrow UF_6$	1949
UK	Springfields Line 4	BNFL	6 000	$U_3O_8 \Rightarrow UF_6$, wet, UNH-process	1974/94
	Springfields Line 3	BNFL	1 200	$U_3O_8 \Rightarrow UF_6$, wet, UNH-process	
USA	Metropolis	ConverDyn	12 700	$U_3O_8 \Rightarrow UF_6$, dry	1959

The Russian conversion plants are tightly integrated to Russian enrichment facilities, and do not export their product on the Western markets.

Although the UF_6 conversion capacities in the Western countries appear sufficient in the short-term to convert the primary uranium actually produced, there are uncertainties about adequacy of future conversion capacities when the BNFL plant will be closed (in 2006) and uranium primary production will increase to compensate for disappearance of secondary sources.

Future developments in enrichment technology (e.g. laser enrichment) could change the needed chemical and physical form of the converted product.

Enrichment

Uranium enrichment involves the partial separation of natural uranium into its two main isotopes ^{235}U and ^{238}U, yielding an enriched fraction containing more ^{235}U and a depleted fraction, called 'tails', containing less than the natural (0.711 wt%) concentration of ^{235}U.

The early history of uranium enrichment was associated with the production of nearly pure ^{235}U for nuclear weapons material, and many separation methods were considered. These included gaseous diffusion, centrifuges, aerodynamic processes, chemical and photochemical

4. The amount which can be supplied to customers in the Western World is currently limited by the US Suspension Agreement and by Euratom policy to about 7 000 tU per year.

methods in addition to a number of other techniques such as electromagnetic separators and distillation.

The main route for uranium enrichment immediately after World War II was gaseous diffusion. Centrifugation processes became commercial in the seventies. Today, gas diffusion and centrifuge technologies are deployed by the different enrichment companies world-wide (see Table 2).

Table 2.2 Commercial Enrichment Facilities[5]

Country	**Place**	**Operator**	**Capacity (1000 SWU/ year)**	**Enrichment process**	**First operation**
China	Lanzhou Shaanxi	CNNC	450 250 200	Gaseous diffusion Centrifuge	1980 1997 Being installed
France	Tricastin (George Besse)	Eurodif	10 800	Gaseous diffusion	1979
Germany	Gronau	Urenco	1 300	Centrifuge	1985
Japan	Rokkasho-Mura - 1 Rokkasho-Mura – 2	JNFL JNFL	600 450	Centrifuge Centrifuge	1992 1997
Netherlands	Almelo	Urenco	1 500	Centrifuge	1973
Russian Federation	Angarsk Ekaterinburg Krasnoyarsk Seversk	Minatom Minatom Minatom Minatom	1 400 10 000 2 900 5 700	Centrifuge Centrifuge Centrifuge Centrifuge (recycled U)	1954 1949 1964 1950
UK	Capenhurst	Urenco	2 000	Centrifuge	1976
US	Paducah Portsmouth	DOE/USEC DOE/USEC	11 300 7 400	Gaseous diffusion Gaseous diffusion	1954 1956[6]
		Total	*56 250*		

The market for enrichment services is expected to show little growth over the next two decades, as there is an over-capacity (approximately 15-20 million SWU). It is estimated that world demand will increase to about 37 million SWU by 2005, driven primarily by increased demand in Asia and will remain constant until 2010.

Nonetheless, Table 2 indicates that replacement of current facilities will become an issue in the coming decade. As of 2005, 90% of the enrichment industry's capacity will be in equipment which is more than 15 years old; about 70% will be more than 25 years old. This applies to the operating equipment, much of which is housed in plant facilities that are substantially older. The question of which technology to use for replacement of the gaseous diffusion plants -- high–performance gas centrifuges vs laser isotope separation technology --

5. Other centrifuge facilities are planned in Brazil and China.
6. Will cease operation in June 2001 (www.usec.com).

will have to be addressed soon. However, laser isotope separation development has been scaled–down drastically in some countries (France) and even stopped in the USA.

Storage of Enrichment Tails

The enrichment of uranium produces depleted uranium (DU) as a by–product. For a typical LWR, between 6 and 8 tons of DU are produced per ton of fresh fuel. Different countries have adopted different management strategies for this material. France is converting the DU to a stable oxide for long–term storage and possible eventual reuse as fuel in breeder reactors or for re-enrichment for extraction of additional fissile ^{235}U. No decisions have been made in the United States on long–term management of this by–product; however, the US Nuclear Regulatory Commission has stated that some type of deep disposal would be required -- if it were to be disposed.

UOX Fuel Fabrication

Table 3 shows current deployment of commercial UOX fabrication plants.

On a world–wide basis, low enriched (~4%) LWR UOX fuel manufacturing capacity appears to be nearly 50% above annual requirements. In the light of this situation, fuel fabricators continue to innovate and modify their manufacturing facilities in order to lower costs. Due to consolidations among fuel vendors, excess capacity is likely to be reduced in the future.

UOX Fuel Fabrication for CANDU

Natural uranium UOX fuel is used in D_2O moderated CANDU reactors. The short, simple CANDU fuel assemblies, called fuel bundles, for the CANDU (PHWR) reactors are produced by Canada, Korea, Argentina and Romania who all have independent PHWR fuel fabrication facilities sufficient to meet their indigenous demands, and China will build a fuel fabrication facility to meet the requirements of their CANDU reactors now under construction. PHWR fuel production costs are lower than those of other reactor designs, because natural uranium is relatively inexpensive, the uranium utilization (amount of energy produced from the mined uranium) is superior to that for enriched U once-through fuel cycles, and because of the simplicity of the PHWR fuel bundle. All manufacturing stages of natural UO_2 fuel can be accomplished without special criticality restrictions.

Several fuel bundle designs are in use in PHWRs; a typical fuel bundle design (the CANDU 6, 37-element bundle has thirty-seven fuel pins of natural uranium dioxide (UO_2) pellets sheathed in Zircaloy-4 (Zr-4) held together by Zr-4 end plates.

Table 2.3 Commercial Fuel Fabrication Plants

Country	Place	Operator	Fuel-type	Capacity (tHM/year)	First operation
LWR					
Belgium	Dessel	FBFC	LWR	500	1961
Brazil	Resende	INB	LWR	100	1982
China	Yibin	CNNC	LWR	100	1993
France	Romans	FBFC	LWR	800	1979
Germany	Lingen	Framatome ANP	LWR	650	1979
India	Hyderabad	DAE	UO_2 pellets	300	1998
India	Hyderabad	DAE	BWR	25	1974
Japan	Tokai-Mura	MNF	PWR	440	1972
	Kumatori-machi	NFI	PWR	284	1972
	Tokai-Mura	NFI	BWR	200	1980
	Kurihama	JNF	BWR	750	1970
Korea	Taejon	KNFC	PWR	400	1989
Russian Federation	Elektrostal	Mashino Stroitelny	UO_2 pellets	800	1996
	Elektrostal	Mashino Stroitelny	LWR (WWER)	620	
	Novosibirsk	TVEL	LWR (WWER)	1 000	1949
Spain	Juzbado	ENUSA	LWR	300	1985
Sweden	Västerås	ABB	LWR	600	1971
UK	Springfields	BNFL	LWR	330	1996
USA	Hematite	ABB-CE	LWR	450	1986
	Columbia	Westinghouse	PWR	1 150	1986
	Lynchburg	FC Fuels	PWR	400	1982
	Richland	Siemens	LWR	700	1970
	Wilmington	GE Nuclear Energy	BWR	1 200	1982
PHWR					
Argentina	Ezeiza	CNEA	PHWR	160	1982
Canada	Toronto	GE Canada Inc.	Pellets	1 300	1967
	Peterborough	GE Canada Inc.	PHWR	1 200	1956
	Port Hope	Zircatec Precision Industries Inc.	PHWR	1 500	1964
India	Hyderabad	DAE	PHWR	300	1974
	Trombay	DAE	PHWR	135	1982
Korea	Taejon	KNFC	PHWR	400	1987
Others (AGR)					
UK	Springfields	BNFL	Magnox (GCR)	1 300	1960
	Springfields	BNFL	UO_2 AGR	290	1996
MOX Fuel					
Belgium	Dessel	Belgonucléaire	LWR	40	1973
France	Cadarache	COGEMA	LWR, FBR	40	1961
	Marcoule – Melox	COGEMA	LWR	100	1995
UK	Sellafield SMP	BNFL	LWR	120	200?
	Sellafield MDF		LWR	8	1993
Japan	Tokai-Mura	JNC	ATR	10	1972
			FBR	5	1988
	Rokkasho	JNFL	LWR	130	200?
Russian Federation	Chelyabinsk	Minatom	FBR	60	
FBR					
France	Veurey-Voroise	CISN	FBR	150	1960
Russian Federation	Elektrostal	Mashino Stroitelny	FBR	50	

MOX Fuel Fabrication

Fueling of LWR plants with mixed uranium oxide/plutonium oxide (MOX) became a commercial reality during the 1970's. Up to now, more than 750 tHM MOX fuel (more than 2 000 fuel assemblies) have been loaded in 29 PWRs and in 2 BWRs in Europe, corresponding to the recycling of about 35 t of plutonium. Currently, MOX fuel fabrication capacities in OECD countries represent a flow of 190 tHM per year. This level corresponds to some 10 to 12 tPu used in MOX fuel per year. Table 3 shows current deployment of commercial MOX fabrication plants.

MOX fuel fabrication for LWR use is mainly based on the MIMAS process (Belgonucléaire, COGEMA) and on the Short Binderless Route (BNFL). In Japan, MOX fuels for core physics and irradiation testing have been manufactured in the JNC plant since 1966. A JNC MOX fabrication plant has been operating since 1972 (10 tMOX/yr) for the advanced thermal reactor ATR-'Fugen', with nearly 130 tons of fuel (about 750 fuel assemblies) having been produced by 1999, equivalent to about 1.8 tHM of plutonium. A microwave heating method for the co-conversion of plutonium-uranium nitrate to MOX powder was developed by JNC in order to obtain MOX powder directly from the product of the recycling process. This method avoids the existence of pure PuO_2 anywhere in the process.

In Russia, plutonium has been considered as a nuclear fuel since the second half of the 1950s. Two principal technologies are being developed to process the plutonium (mainly low burn-up) into mixed uranium-plutonium fuel: pelletising of U-Pu powders and vibrocompacting[7] which directly uses the acid solution from recycling and which could have economic merit compared to the classic pelletising route. These technologies are implemented at Mayak (Chelyabinsk) and at RIAR (Dimitrovgrad), respectively.

Today's MOX industry has become a mature business where parity between UOX and MOX fuel prices is requested by the utilities that employ a recycling strategy. Technology developments pursued focus on smoothing fuel cycle management by increasing burn-up of the MOX fuels, the possibility of load-following with MOX fuelled reactors (already demonstrated and licensed in France) and the possibility of increased MOX fraction in the core loading pattern (100% MOX cores) resulting in a net reduction of the Pu inventory (-15 kg/tHM per cycle in a 900 MWe LWR with a 100% MOX core).

Japan decided in November 2000 on a new LWR-MOX fuel fabrication plant with a capacity of 130 tHM/yr which will be constructed at Rokkasho-Mura and will be supplied with recycled fuel from the Rokkasho Recycling Plant.

MOX Fuel Fabrication for Fast Reactors

The use of MOX fuel in fast reactors allows its irradiation to achieve higher burnup and higher power density ratings when compared to LWR and offers the possibility to fission isotopes that would be neutron poisons to LWR fuels.

7. This production technique is applicable for future development of Am+Cm fuels or targets in advanced nuclear fuel cycles (see Chapter 4).

Liquid metal cooled fast reactor technology has been under development for as long as 50 years in many countries, including France, Germany, India, Japan, the Russian Federation, the United Kingdom and the USA. The liquid metal coolant is sodium for the most part, but in Russia lead and lead-bismuth are also used or under development. Other developments (US, France) have explored gas-cooled fast reactors. Development of fast reactor technology slowed during the last decade in the context of an overall slowing-down of nuclear power development.

Monju (Japan) and BN600 (Russia) are the only commercialized fast reactors currently deployed.

Commercial fast reactor fuel is MOX with higher Pu content compared to LWR-MOX fuel. Fuel pins clad in stainless steel are assembled in a hexagonal wrapper tube. The fabrication technology is well established, and fuels were supplied successfully in France, US, UK, Russia and Japan on a pre-industrial scale (see Table 3). In Japan, JNC has fabricated MOX fuel for the experimental Joyo and prototype Monju fast reactors since 1973. The Russian BN-600 operates on UOX fuel but has irradiated many MOX test assemblies. MOX fabrication capacity for fast reactor use is not expected to grow for several decades.

A long term development and demonstration program on metal alloy fuel for fast reactors has been carried at Argonne National Laboratory's EBR-II experimental fast reactor. Pre-industrial scale fabrication technology is well established. Similarly a long term technology development and demonstration program on vibrocompaction fabrication for MOX fuel pins for the BOR-60 test reactor at Dimitriovgrad, and pre-industrial scale fabrication technology is well established.

Recently, the interest in fast reactors and accelerator-driven fast spectrum systems as part of a partitioning and transmutation scheme has engendered R&D on new fuel matrices containing reduced or no fertile material. Next to known fuel fabrication techniques, such as pelletising and vibro-packed fuel, other developments suited to handling highly radioactive fuel compositions are being tested for deployment in the mid to long term. These are current R&D issues which are described in Chapter 4.

Thorium Fuel Fabrication

Over 50 tons of thorium fuel in ceramic form, clad in Zircaloy, was manufactured for the Shippingport LWBR core. No specific problems were encountered during fuel manufacture. US regulations governing maximum permissible concentration (MPC) limits on thorium in air have been dramatically tightened since the Shippingport fuel was produced; allowable limits have been reduced by a factor of 20. Thus, renewed manufacture of thorium fuel would increase in cost due to tight health physics control and monitoring.

In India, the use of thorium in reactors has been envisaged mainly in the form of thorium oxide where conventional dry powder metallurgy techniques for compacting and sintering were adopted.

Fabrication of thorium bearing fuel per se does not pose any serious radiological problems. However, one of the main issues for a closed thorium-^{233}U fuel cycle is the presence of hard γ-emitters (2.5 MeV) among the daughter products of ^{232}U that is always present with ^{233}U. This necessitates shielding and remote-handling facilities for manufacture of ^{233}U-based fuels – just as would be the case for TRU bearing fuels.

No commercial thorium fuel fabrication plants exist currently. Despite possible increases in the costs of manufacturing, the experience gained in thorium fuel manufacturing has established a pre-industrial status for thorium fuel fabrication processes.

Coated Particle Thorium Fuel Fabrication

In Germany and the USA, a fuel fabrication technology for thorium-based coated particle fuels has been developed up to a well proven, industrial-scale process under the High Temperature Reactor (HTR) program. Usually, coated particles were used, i.e. a spherical fuel kernel (oxide or carbide of U, Th or Pu) surrounded by layers of a coating material such as pyrolitic carbon or silicon carbide. Powder agglomeration processes or wet-chemical processes (sol-gel) for the gelation of droplets from a solution containing thorium and uranium could be used to produce the kernels. It was found from the development of prototype plants for the fabrication of mixed oxide fuels (Th/U or Th/Pu) during the 1960's in the USA that the sol-gel process could be a very suitable process although somewhat delicate to use and not suitable for remote manufacturing facilities. A process based on "sol-gel microsphere pelletisation" has also been developed in Germany for the fabrication of high-density oxide pellets for HTRs, at relatively low compaction pressure and low sintering temperature, avoiding dust generation.

Considerable experience was accumulated in the USA in coated particle thorium fuel fabrication. For the Fort St.Vrain high temperature gas reactor, 2,448 hexagonal graphite fuel elements using 26,000 kg of fissile and fertile material in TRISO-coated fuel particles were produced. This included almost 25,000 kg of thorium. It was irradiated at temperatures greater than 1,300ºC to a maximum burn-up in the fissile particles of 16% fissions of initial metal atoms (approximately 170,000 MWd/tHM) to a maximum fast neutron fluence of $4.5*10^{25}$ n/m^2 with no evidence of significant coating failure.

No commercial fabrication plants currently exist; the ESCOM project for PBMR's intends to construct a coated particle fuel fabrication plant for UO_2-based coated particle fuel.

PUREX Recycling of Discharged UOX Fuel

Table 4 lists the world's current deployment of PUREX recycling plants. Commercial recycling capacity today amounts to some 3 500 tHM/yr. The GR205 Magnox plant in the UK, with a capacity of 1 500 tHM/y, will be shut down sometime in the coming decade. World-wide commercial recycling capacity after 2015 is forecast to be some 4 300 tHM/y.

Table 2.4 Commercial Fuel Recycling Plants

Country	Company	Facility/location	Year of commissioning	Capacity (tHM/yr)	Fuel type
Plants in operation (01/01/2000)					
France	COGEMA	UP2-UP3/La Hague	1976 and 1989	1 700	LWR
India		Prefre-1, Tarapur	1974	100	PHWR
		Prefre-2, Kalpakkam	1998	100	PHWR
Japan	JNC	Tokai-Mura	1977	90	LWR, ATR
UK	BNFL	Thorp/Sellafield	1994	1 200	LWR, AGR
		B205 Magnox	1964	1 500	Magnox GCR
Russian Federation	Minatom	RT-1/ Tcheliabinsk-65 Mayak	1984	400	WWER
Plants under construction (01/01/2000)					
China	CNNC	Diwopu (Ganzu)	2002	25 - 50	LWR
Japan	JNFL	Rokkasho-Mura	2005	800	LWR

Geologic Repositories

No commercial geologic repositories exist. Countries currently developing geologic repositories for HLW disposal have generally adopted a stepwise approach that includes a period of intensive underground investigations and testing. Table 5 provides a list of the principal existing underground research facilities.

Table 2.5 Main Underground Research Facilities (information compiled by consultants at an IAEA sponsored meeting in 1999)*

COUNTRY	LOCATION	USUAL NAME/TYPE OF FACILITY	HOST ROCK/ FORMATION	TIME PERIOD
BELGIUM	MOL	HADES+URF PRACLAY	Plastic Clay	since 1980
CANADA	LAC DU BONNET Manitoba	URL	Granite	since 1984
FINLAND	OLKILUOTO (in VLJ repository)	Research Tunnel	Granite	since 1993
FRANCE	FANAY Augères/Tenelles	Galleries in U Mines	Granite	1980-1990
	AMELIE	Galleries in K Mine	Bedded salt	1986-1994
	TOURNEMIRE	Test Galleries	Shale	since 1990
GERMANY	ASSE	Test Galleries in K/salt mine	Dome salt	1977-1995
	GORLEBEN	URL	Dome salt	since 1997 (now halted)
	KONRAD	Test Galleries in Fe Mine	Shale	since 1980
JAPAN	TONO	Galleries in U Mine	Sandstone	since 1986
	KAMAISHI	Galleries in Fe-Cu Mine	Granite	1988-1998
SWEDEN	STRIPA	Galleries in Fe Mine	Granite	1976-1992
	ÅSPÖ	HRL	Granite	since 1990
SWITZERLAND	GRIMSEL	GTS at dam tunnel	Granite	since 1983
	MONT TERRI	Galleries at road tunnel	Shale	since 1995
USA	NEVADA Test Site	CLIMAX	Granite	1978-1983
	NEVADA Test Site	"G-Tunnel"	Tuffs	1979-1990
	CARLSBAD	WIPP	Bedded salt	since 1982
	YUCCA Mtn.	ESF	Tuffs	since 1993
	YUCCA Mtn.	Busted Butte	Tuffs	since 1997

*Existing facilities where tests were and/or are still undertaken. Table provided by IAEA from their Nuclear Technology Review, 2002.

2.3 Survey of Potential Gen-IV Fuel Cycles

As described in Chapter 1, the potential future Gen-IV fuel cycles can be conveniently "binned" into the four generic fuel cycles shown in Figure 4; these are organized based on the degree of consumption of the actinide feedstock – and concomitantly the fraction of the feedstock that ultimately is disposed as waste. Given any one of the four generic fuel cycles, the sequence of steps in the fuel cycle is as shown in Figure 5.

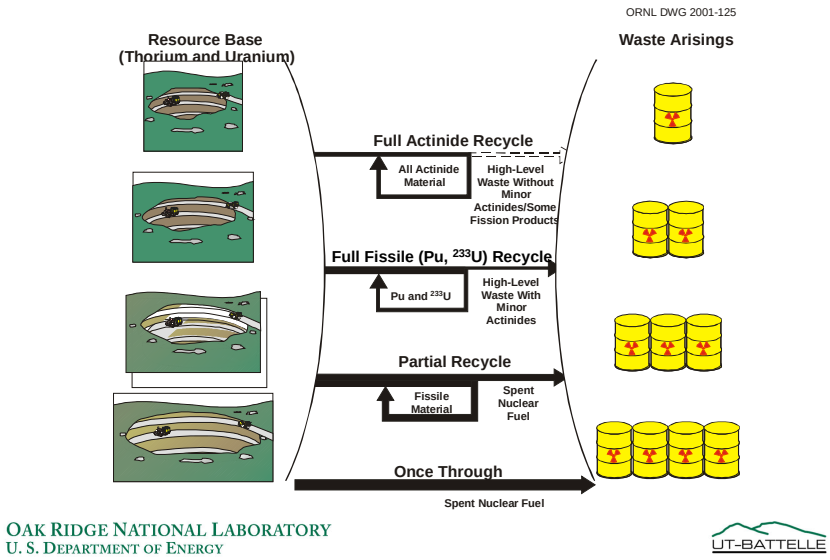

Fig. 2.4 Generic Gen-IV Fuel Cycles

In Figure 5, the "front end" links in the fuel cycle chain are shared by many of the Gen-IV concepts; similarly most concepts share similar "back end" waste disposition links in the fuel cycle chain. However, Gen-IV concepts employ very diverse fuel cycle technologies for the fuel cycle links in the middle of the chain – depending on choice of fuel composition; choice of neutron spectrum, choice of recycle vs no recycle, choice of recycle technology and choice of refabrication technology.

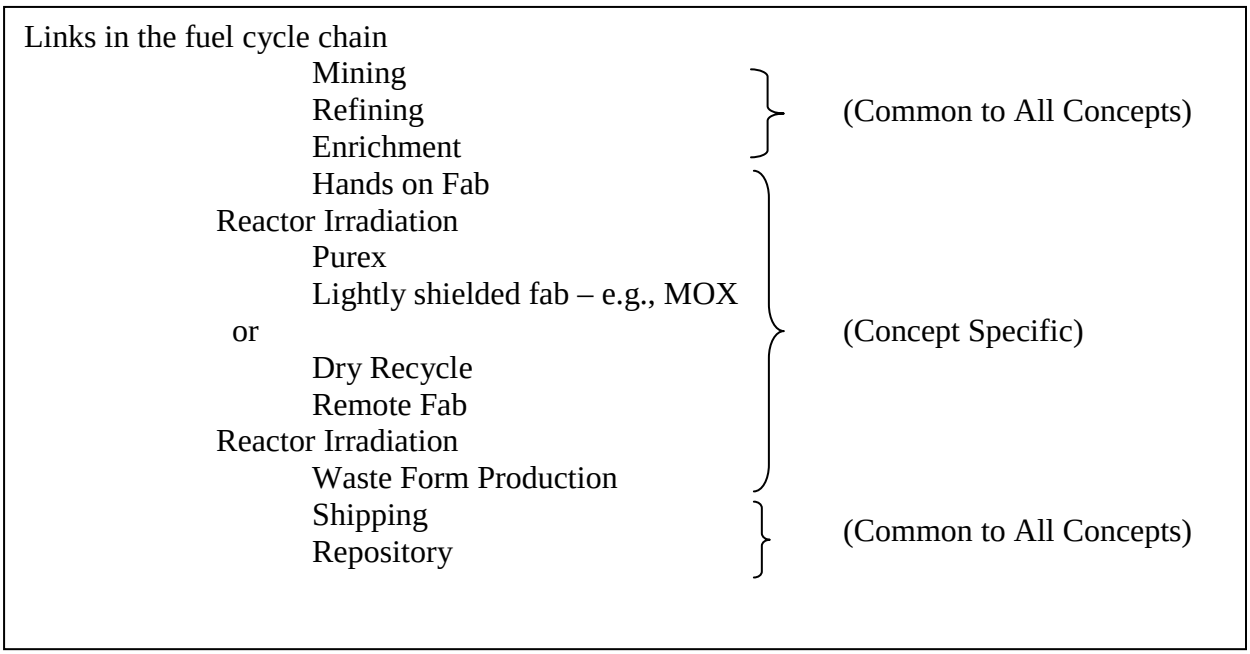

Fig. 2.5 Links in the Fuel Cycle Chain: Common Links and Diverse Links for Gen-IV Concepts

In light of this diversity, the FCCG conducted a survey early in the Gen-IV Roadmap campaign among the water, gas, liquid metal and non conventional Technical Working Groups (TWG's) to determine the breadth of fuel cycle technological approaches – as a first step toward identifying fuel cycle scenarios which should be evaluated and for identifying required cross cutting fuel cycle related R&D. Table 6 shows the results of the survey organized in the hierarchy:

For each Generic Fuel Cycle Type

First, Thermal Spectrum Concepts of the Type:

Water

Gas

Liquid Metal

Then , Fast Spectrum Concepts of the Type:

Water

Gas

Liquid Metal

While the survey results tabulated in Table 6 were developed early in the Roadmap process and could now be updated in light of additional information available to the TWG's, it is already evident that potential Gen-IV concepts exist for fuel compositions of oxide, nitride, metal, and carbon compounds and in forms of ceramics, metal alloys, coated particles, dispersions, liquids, and vapors. Both uranium and thorium are considered as fertile material, and inert matrix fuels containing no U^{238} or Th^{232} are considered as well. Recycle options include both aqueous

approaches and non-aqueous, and aim to recycle either plutonium alone or to recycle all transuranics. Refabrication technologies depend on this recycle distinction approach and are both contact handled (for high decontamination recycle products) and remote handled (for low decontamination recycle products).

The results of the survey (which are summarized in Table 6) helped the FCCG to determine the scope of its review of technological status for each link in the fuel cycle chain. This review is presented in Chapter 4 along with recommendations for crosscutting fuel cycle R&D in support of Gen 4. It also helped the FCCG to plan for the range of fuel cycle scenario cases which could span the space of potential nuclear futures. These scenarios are discussed in Chapter 3. They are organized in a sequence to help inform the Gen-IV Roadmap development.

Table 2.6 Summary of TWG Responses to Fuel Cycle Options Questionnaire

	Fuel Chemical Composition and Fuel Form							Recycle Choice		Refab Choice		
Fuel Cycle Scheme & *Specific Reactor Technologies*	Coolant	Oxide	Nitride	Carbide	Metal	Liquid	Other Compounds	Aqueous	Non Aqueous	Hands On	Remote Operations Contact Maintenance	All Remote
(1) Once-Through With disposal of FPs and Unused Actinides												
Thermal Spectrum												
LWR once-through	Water	UO_2 pellets						N/A	N/A	✓		
IRIS	Water	UO_2 pellets						N/A	N/A	✓		
Multi-application Small LWR	Water	UO_2 pellets						N/A	N/A	✓		
Homogenous Thoria-Urania	Water	ThO_2-UO_2 pellets, ~70% Thoria						N/A	N/A	✓		
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: enriched – uranium metallic dispersion fuel			N/A	N/A	✓		
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	UO_2 pellets						N/A	N/A	✓		
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	UO_2 pellets						N/A	N/A	✓		
Gas-cooled Reactor w/ LEU	He	Alt: TRISO-Coated UO_2					TRISO-Coated UCO					
Gas-cooled Reactor w/ LEU/Th	He	TRISO-Coated ThO_2 Alt: TRISO-Coated UO_2					TRISO-Coated UCO					
Gas-cooled Reactor w/ HEU	He	Alt: TRISO-Coated UO_2					TRISO-Coated UC					

	Fuel Chemical Composition and Fuel Form							Recycle Choice		Refab Choice		
Fuel Cycle Scheme & *Specific Reactor Technologies*	Coolant	Oxide	Nitride	Carbide	Metal	Liquid	Other Compounds	Aqueous	Non Aqueous	Hands On	Remote Operations Contact Maintenance	All Remote
(2) Limited Resource Recovery with disposal of FPs and some Actinides												
Thermal Spectrum												
IRIS	Water	PuO_2-UO_2 pellets						Yes	No		✓	
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: plutonium-bearing, metallic dispersion fuel			Yes	No	✓	✓	
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	PuO_2-UO_2 pellets						Yes	No		✓	
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	PuO_2-UO_2 pellets						Yes	No		✓	
HTGR with LWR Pu feed	He	TRISO-Coated PuO_2						PUREX				✓
Gas-cooled Reactor w/ LEU/Th	He	TRISO-Coated ThO_2 Alt: TRISO-Coated UO_2					TRISO-Coated UCO		Particle Separation			✓
Fast Spectrum												
Tight-lattice BWR Thoria Cores	Water	PuO_2-ThO_2 pellets						Yes	Maybe		✓	Maybe

	Fuel Chemical Composition and Fuel Form							Recycle Choice		Refab Choice		
Fuel Cycle Scheme & *Specific Reactor Technologies*	Coolant	Oxide	Nitride	Carbide	Metal	Liquid	Other Compounds	Aqueous	Non Aqueous	Hands On	Remote Operations Contact Maintenance	All Remote
(3) Multiple Recycle of Fissile Act's, with disposal of FPs and of non-fissile and MAs												
Thermal Spectrum												
Fast Spectrum												
EFR, DFBR	Na	MOX pellets						PUREX			✓	
S-PRISM	Na	MOX pellets						PUREX			✓	
(4) Multiple Recycle of all Act's with disposal of FPs												
Thermal Spectrum												
Gas-cooled Reactor w/ LWR Pu feed followed by Gas-cooled Reactor w/ GCR MA feed (not continuous recycle but multiple recycle)	He	TRISO-Coated PuO_2 & TRISO-Coated $(MA)O_2$ (mixed or separate)						✓	✓			✓
Fast Spectrum												
Supercritical Water-Cooled Fast Reactor	Super-critical Water		PuN-ThN or (TRU)N-ThN Pellets		Pu-Th-Zr or TRU/Th/Zr Dispersion Slugs			No	Yes, electrorefining or pyroprocessing			✓
Advanced Design GCFR using feed from LWR Pu with possible intermediate HTGR cycle	He	TRISO-Coated PuO_2 & TRISO-Coated $(MA)O_2$ (mixed or separate)						✓	✓			✓
Gas Cooled Fast Reactor	He	MOX pellets	Mixed Nitride pellets				CERMET, CERCER	✓	✓			✓

	Fuel Chemical Composition and Fuel Form							Recycle Choice		Refab Choice		
Fuel Cycle Scheme & *Specific Reactor Technologies*	Coolant	Oxide	Nitride	Carbide	Metal	Liquid	Other Compounds	Aqueous	Non Aqueous	Hands On	Remote Operations Contact Maintenance	All Remote
SPRISM	Na				U-Pu-Zr				Pyro			✓ Casting
BREST	Pb		Vibro or pellet						Pyro or Flouride			✓ Vibro or pelletizing
STAR-LM	Pb-Bi		Vibro or pellet						Pyro			✓ Vibro or pelletizing
STAR-LM	Pb		Vibro or pellet						Pyro			✓ Vibro or pelletizing
ENHS	Pb-Bi or Pb				U-Pu-Zr				Pyro			✓
IFR	Na				U-Pu-Zr				Pyro			✓

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 3

Illustrative Scenarios: Generation-IV Performance Against Goals

March 18, 2002

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Table of Contents

3.1 21st Century Gen-IV Global Nuclear Energy Scenarios

One hundred year global aggregate nuclear energy supply scenarios have been simulated for numerous potential Gen-IV concepts and combinations of concepts.

The scenarios use as initial conditions the year 2000 deployments of fuel cycle facilities and power plants which are specified in Chapter 2. Remaining productive lifetime of these assets are approximated with decommission spread over the next 40 years. The scenarios all use the uranium resources from the Redbook and thorium resources which were specified in Chapter 1; scenarios are not terminated upon exhaustion of these ore resources – rather, an edit is produced delineating the necessary future discoveries.

The scenarios are driven by the Case B world aggregate nuclear energy demand growth evaluations produced by the World Energy Council/IIASA and specified in Chapter 1 – growth of the world nuclear production of electricity rises from 350 GWe in year 2000 to 2000 GWe in 2050 to about 6000 GWe in year 2100. The deployment is thought of as the "global nuclear energy park" with no regional segmentation of energy demand or mass flows.[1]

So as to judge the robustness of the conclusions from the scenarios with respect to growth rate of demand, all scenarios were rerun with a slower growth rate – the C2 scenario from the WEC/IIASA which grows from 350 GWe to 1200 GWe in 2050 to 1800 GWe in year 2100.

In attempting to satisfy the pre-specified demand, the simulated deployment of new power plants is constrained only by internal mass flows which determine fissile availability – i.e., plants cannot start producing power unless they can be fueled – drawing on either virgin ore, or on fissile available in discharged fuel assemblies, or fissile remaining in enrichment tails and recovered irradiated uranium, or in new fissile bred in previously – deployed power plants (see Figure 3.1 which conceptually illustrates the several sources of fuel feedstock). The conversion ratios of each Gen-IV concept reactor are specified as input to the simulation and used to determine the composition of discharged fuel.

The sources of fissile to fuel new deployments are specified as input to the scenario. The physical inertial elements of the supply infrastructure are accounted for – including licensing and construction lag times for fuel cycle infrastructure elements and for power plants. The time lags for interim storage between links of the fuel cycle are also accounted for. When discharged fuel from once-through fuel cycles, (after a period of interim storage) is reprocessed to supply fuel for future power plant deployment, the buildup rate of the recycling capacity must be balanced between timely supply of sufficient fissile from recycle to meet new deployments vs. avoidance of buildup of interim stores of separated transuranics. This is done to recognize the financial disincentives which exist for deploying capital assets prior to timely need for their product. Additionally in the case of separated transuranics, their annual storage charge rate exceeds that of discharged fuel in the cost

[1] The demand is represented as a parabola fit through three data points: 353 GWe in year 2000; 2000 GWe in year 2050; and 6000 GWe in year 2100. The doubling time calculated for an exponential is 20-25 years.

model used here. Finally, depending on the form and content of the separated transuranics, it may require more or less costly safeguards stewardship than does discharged fuel.

Market economic penetration is not modeled; both the dates of commercial availability of various Gen-IV concepts and the fractional mix of Gen-IV concepts to be used to satisfy *new* demand *are pre-specified as input to the specific scenario case being evaluated*. In this sense, the scenarios serve to illustrate what could be *physically achievable* for specified Gen-IV strategies – but not what will actually occur when market and institutional dynamics and inertia effects are accounted along with physical inertia elements of mass flows and construction times.

The cases which have been selected for simulation are intended to serve the purpose of "cornerstones" which when taken together delineate the relative capabilities of physically achievable Gen-IV global energy parks in satisfying the Gen-IV sustainability goals pertaining to efficient use of resources and to ecological effects. To this purpose the edits from the simulations quantify the front and back end mass flows and inventories such as ore

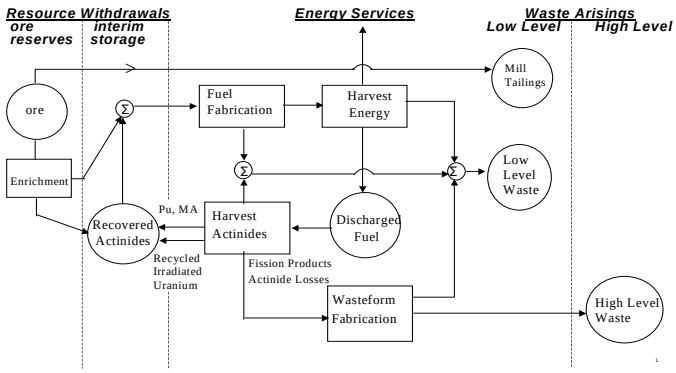


Fig. 3.1 The Fuel Cycle

withdrawals vs. time and their cumulative withdrawals and the high level waste arisings vs. time and their cumulative arisings. Additionally, inventories of spent fuel, plutonium, and minor actinides in interim storage or in the processing "pipelines", are shown. These edits are relevant to the third Gen-IV Sustainability Goal.

The edits quantify the scales of deployment of mining/milling, enrichment services, fuel fabrication plants, reprocessing capacities, and required capacities of interim storage facilities and final disposal repositories. These edits indicate the scale of the Gen-IV global infrastructure and are relevant to the Gen-IV Economic and Safety/Reliability goals.

Finally, the edits depict a year-by-year cost indicator for the *fuel cycle component of the cost of energy production* in the "global nuclear energy park" comprised of the specified mix of Gen-IV concepts. The edit is for fuel cycle services only; it does not include capital amortization and O&M costs of the power plants themselves. This indicator is quoted in year

2000 constant dollars per Terrawatt electric hr. The indicator covers the front end costs required to deliver fuel to the mix of deployed power plants (mining, milling, enrichment, fabrication) and the back end costs required to store and ultimately dispose the wastes. (When recycle is used to harvest materials for fuel assemblies destined for power production its cost is included in "front end" costs rather then back end costs). This fuel cycle services cost index serves as no more than an indicator for all of the following reasons:

- No discounting of the time value of money is used and moreover a cash flow accounting is used wherein expense is allocated to the index in the year a service is performed – even if the fuel won't be loaded until after a lag time.
- Unit process costs for future fuel cycle operations have been drawn up by reliance on expert judgment and therefore are highly uncertain.
- Capital amortization costs for future fuel cycle service plants are embedded in the unit process costs and also rely on expert judgment and are highly uncertain.
- The cost elements are sometimes borne by different entities (e.g., commercial ones and governmental ones) but they are aggregated for the purpose of the index.

The edit shows both the index itself and its fractional contributions from links in the fuel cycle. These components serve as useful indicators of sensitivities to the fuel cycle components of energy production cost and can suggest areas where R&D efforts could most usefully be directed. The unit costs and the methodology for tracking costs are described in Attachment 2.

The scenarios are run using the code, DYMOND[2] which was prepared specifically for the Gen-IV FCCG activities. At its current level of representation, DYMOND handles only a coarse representation of mass flows – Specifically, it does not distinguish plutonium of differing isotopic distribution; nor minor actinides of differing atomic and isotopic composition. At the current level of representation in the DYMOND code, differences among fuel composition i.e., oxide, nitride, etc. are not modeled and mass flows are all in terms of heavy metal.

The attributes of power plants are concept specific; the values of their attributes are expressible in terms of licensing and construction lag times, plant lifetime, working inventories, mass flows and spent fuel cooling lag times. The attributes of fuel cycle service plants such as throughput capacities, lag times for construction, plant lifetime, lag times for processing and post processing storage, etc. are also concept-specific.

The attribute values used up to now in the scenarios have not been provided by the Gen-IV TWG's – they are best judgment values culled from sources available to FCCG membership.

[2] Anton Moisseytsev, "DYMOND – A Dynamic Model of Nuclear Development," Internal Report, Argonne National Laboratory, (Aug 2001).

The initial conditions, fuel cycle unit process costs, uranium price vs. cumulative consumption, and power plant and fuel cycle facility attribute sets are all tabulated in Attachment 2.

3.2 The Scenario Set

The FCCG has organized its work using four generic fuel cycles which are depicted in Figure 3.2; -- the once-through, the partial recycle, the full fissile (plutonium) recycle, and the full actinide (transuranic plus uranium) recycle. The logic process for the selection of cases for scenario evaluation employed these four generic fuel cycles to generate "cornerstone" results which, when taken together, delineate the boundaries of physically achievable performance against the sustainability goals set for Gen-IV. For example, in principle there are four inventories of actinide-containing resource available to fuel the global nuclear energy park (see Figure 3.1):

Virgin ore
Enrichment tails
Spent Fuel
Separated irradiated uranium

However, as illustrated in Figure 3.3, the four generic cycles avail themselves differently of these resources and use different asset combinations to do so. The once-through generic cycles draw from the first resource only and potentially from the second. The partial recycle generic cycles draw fuel from the first three. These two generic cycles are the only ones which are currently deployed commercially – and they employ only thermal spectrum reactors. The fresh fuel is not radioactive and can be fabricated using hands-on, lightly-shielded technologies. Many of the candidate Gen-IV concepts rely on the once-through or partial recycle generic fuel cycles for fuel cycle services.

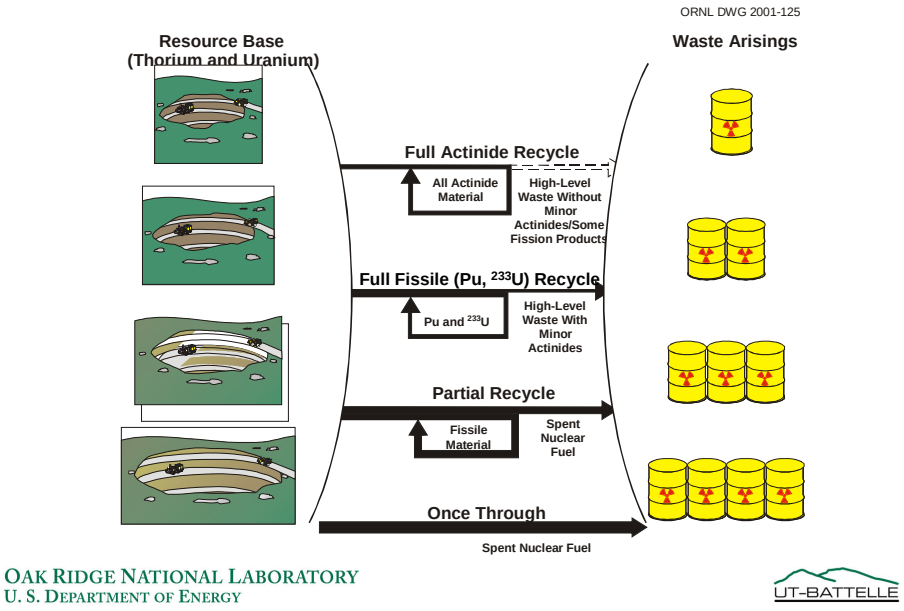


Fig. 3.2 The Four Generic Fuel Cycle Figures

The full fissile plutonium and full transuranic recycle generic cycles draw from the last three resource inventories shown in Fig. 3.3 and employ both thermal and fast spectrum reactors; they always employ *multiple* recycle. In the full Pu recycle case, minor actinides are consigned to waste; the recycled plutonium can be fabricated "hands-on" with only moderate shielding. In the full TRU recycle case, where minor actinides are recycled to the reactor, remote fabrication technologies are required. These latter two generic cycles are not currently[3] deployed commercially but are prominent among candidate Gen-IV concepts.

The first two generic cycles consign the enrichment tails, spent UOX fuel or spent MOX fuel, High Level Waste containing fission products and minor actinides , and separated irradiated uranium to waste management. The full plutonium recycle generic cycle consigns minor actinides and fission products to waste and enrichment tails and separated irradiated uranium to interim storage. The full transuranic recycle generic cycle consigns fission products to waste and enrichment tails and separated irradiated uranium to interim storage. Eventually over many decades even the enrichment tails and separated irradiated uranium are converted to plutonium and minor actinides and fissioned. Figure 3.3 illustrates what is fuel and what is waste for the four generic fuel cycles.

Given the roughly one hundred candidate Gen-IV concepts, and given these differences among their fuel cycle strategies, specific scenario cases have been selected as "cornerstones" to illustrate the broad implications of various choices in meeting Gen-IV goals.

Table 3.1 enumerates the scenario cases which have been analyzed. It is seen in the table entries that while not attempting to simulate economically – driven market penetration of alternate Gen-IV concepts, the pre-specified introduction dates and market penetration rates (specified as a fraction of new additions to capacity) which have been selected to drive the scenarios recognize the realities that:

(a) The Gen-IV market penetration will occur incrementally during an extended (multi-decade) transition from the Gen 3 and Gen 3+ concepts

(b) The introduction dates of different Gen-IV concepts and their supporting fuel cycles will likely extend from 2015 thru 2040 – owing to the fact that some concepts are nearer to commercial readiness than others and recognizing that economic conditions will evolve to alter the relative importance of capital vs operating costs, societal considerations, etc. which control market penetration.

(c) Once deployed, power plants and fuel cycle facilities have a long asset lifetime (~60 years), and will co-exist in the "global nuclear energy park" even as new entries of more advanced technologies gain market share of new construction.

(d) Different Gen-IV concepts will likely co-exist in the global nuclear energy park – filling different market niches -- and in some cases operating in symbiosis based on mutually beneficial exchange of mass flows.

Thus, all scenarios (except the 100% LWR-UOX once-through base case) involve a dynamically evolving mix of current generation and Gen-IV concepts.

[3] MONJU and BN-600 are exceptions; MONJU will run on MOX recycled from LWR's; BN-600 currently is fueled with UOX once through.

	Natural Resources		Fate of Fuel Resource/Waste				Technology Required			
	Virgin Ore	Enrichment Tails	Discharged Fuel				Recycle Technology Used	Remote Refab Used	Thermal Reactors Used	Fast Reactors Used
			Plutonium	MA	Irradiated Uranium	FP	No		Yes	No
Once Thru	Fuel	Interim Store	Waste	Waste	Waste	Waste		No		
Partial Recycle	Fuel	Interim Store	Fuel From UOX	Waste From UOX	Interim Store/From UOX	Waste From UOX	One Pass			
			Waste MOX	Waste MOX	Waste MOX	Waste MOX				
Full Pu Recycle	Fuel	Fuel	Fuel	Waste	Interim Store	Waste	Multiple Recycle			Yes
Full TRU Recycle	Fuel	Fuel	Fuel	Fuel	Fuel	Waste		Yes		

Fig. 3.3 Designation of Resource and Waste For The Four Generic Fuel Cycles

Table 3.1 Scenario Cases Examples to Illustrate Fuel Cycle Issues

- Use Case B of WEC/IIASA for World as a Whole; Use Case C2 to Assess Sensitivity to Demand Growth Rate
- Assume 90% of already deployed Gen 3 LWR's worldwide will receive 20y life extension
- Cases 1 and 7 are the extremes; setting physically achievable bounds; Cases 2 thru 6b and Case 8 allow for practicalities of markets

	Case B		Case C2	
Example Cases	Section	Figures	Section	Figures
A. Once-Through Thermal			All are	
1. New Additions = LWR UOX Once-Through	3.3.1	4-6	in	4-6
2. LWR UOX Once Thru; New additions (after 2010) HTGR Gas Turbine UOX Once-Through (50%) + LWR-UOX Once-Through (50%)	3.3.2 & 3.3.3	7-12	Attach. 3	7-12
3. Radkowsky Th Fueling in Existing PWR's	3.7.1	33-36		Not Run
B. Limited Thermal Recycle (after 2010)				
4. New Additions = LWR UOX → LWR MOX mono → Waste	3.4.1	13-15		13-15
5. DUPIC PWR/CANDU Symbiosis	3.4.2	16-18		16-18
C. Full TRU Recycle: Symbiotic Fast/Thermal (after 2025) for Waste Management				
6a. New additions = 80% LWR (UOX) + 20% Fast Burner (BR = ½) Waste Self Consumption Cycle	3.5.1	19-21		19-21
6b. New additions = Fast Burner (BR = 1) (use up LWR discharged fuel + future waste self consumption)	3.5.2	23-25		23-25
D. Full TRU Recycle: Breeding (after 2030) for Limiting Ore Drawdown	3.5.3	27-29		27-29
7. Start with Case 6b – followed by Switching in 2030 to New additions = Fast Breeders (BR = 1.7)				
E. Full TRU Recycle: Symbiotic Feedback of Fissile to Thermal Systems	3.5.4	30-32		30-32
8. New additions = fast breeders + LWR_MOX s(TRU)				
F. Transition to fissile self sufficient Molten Salt Closed Thorium Cycle	3.7.2	37-39		37-39

3.3 Scenario Results for the Once-Through Cycles

3.3.1 The LWR-UOX Once-Through Base Case Scenario

During the current century, the global nuclear energy park in the WEC/IIASA Case B) is assumed to grow from 350 GWe in year 2000 to 2000 GWe in year 2050 to 6000 GWe in year 2100. The first scenario considered for the once-through generic fuel cycle is the "business as usual" scenario where the LWR UOX once-through cycle continues to hold 100% market share in meeting the global nuclear demand growth all the way to the year 2100. This case is a base case against which the efficacy of alternate Gen-IV energy parks can be compared. The attribute set used for the LWR-UOX power plants is displayed in Attachment 2. (Enrichment of 4.2% and 50,000 MWd/tonne are used.)

Figure 3.4 displays the deployments of fuel cycle services and reactors needed to meet energy demand for the 100% LWR-UOX base case scenario. By 2050 a total of 2000 GWe of LWR-UOX once-through power plants are deployed; ~280 kilo tonne SWU/year of enrichment capacity is used (up by a factor of five from the 50 kt/y in year 2000); and ~60 kilo tonnes/year of UOX fab capacity are deployed (up by a factor of five from the 12 kt/y in year 2000).

Figure 3.5 displays the trends in fuel resource inventories and waste inventories. The 4.1 million tonnes of uranium in the Redbook known ore reserves (recoverable at $\leq$130 \$/kgU) is fully consumed by 2030; the additional uranium comprising the speculative reserves (recoverable at < 130 \$/kgU) are subsequently used up by 2050. After mid century the required rate of discovery and harvesting of new uranium ore reserves grows steadily from half a million tonnes per year to about one million tonnes per year and reaches a cumulative total (known plus speculative plus new discoveries) uranium consumption of 57 million tonnes by 2100. At year 2100, though the cost of recovery is assumed in the model to have risen by then to 200 \$/kgU (up from 20 \$/kgU in year 2000), the impact of fuel cycle services on the overall cost of energy production remains small at about 9 mills/kW_e hr (see Fig. 3.6)

Focusing back on the year 2050 – twenty years beyond the intended latest introduction date for Gen-IV concepts, in the business as usual scenario where LWR-UOX once-through holds 100% market share, not only would new ore discoveries be required, but the waste arisings would already be significant. The vast majority of the mined uranium ends up in one or another of the waste inventories either in interim storage or permanent disposal – mostly depleted uranium enrichment tails but also a quite significant spent fuel inventory.[4] The cumulative spent fuel inventory by 2050 is a million tonnes of heavy metal and is growing at 35,000 tonnes/year. These waste arisings can be compared to the 70,000 tonnes legislated capacity of Yucca mountain or the 4,000 tonne capacity of the Finnish repository at Eurajoki – more than ten times the legislated capacity of Yucca Mountain and growing at a rate which each two years requires an additional repository of similar capacity.

[4] In the middle plot of Fig. 3.5, the upper curve is spent fuel awaiting disposal (30 year lag time) and the lower one is spent fuel already in the repository; the bottom plot in Fig. 3.5 shows is the sum of the two.

The cost index for the fuel cycle component of the overall cost of energy production (see Fig. 3.6) is rising continuously due to the increasing cost of ore. The waste management burdens at the back end consist of the 1 mill/kwhr repository charge and costs of annual interim storage followed by a one time charge to package and emplace the fuel in a repository (see Attachment 2). The interim storage period is modeled here as 30 years. *This cost model places no cost penalty on the massive increase in repository deployment, so the disposal cost per terawatt electric hour remains small.*

This business as usual scenario serves as a base case against which to compare various Gen-IV concepts or symbiotic clusters of concepts.

3.3.2 The Mixed LWR-UOX and PBMR Pebble Bed Once-Through Case

The second scenario considered for the once-through generic fuel cycle assumes an aggressive PBMR market penetration. It is assumed that starting in 2010, all new deployments made to meet demand are shared 50/50 between LWR-UOX once-through and PBMR-UOX once-through power plants. The PBMR uses 8% enriched fuel and achieves 80,000 MWd/t discharge burnup. The attributes used for the PBMR are displayed in the Attachment 2. As an expedient measure in the absence of real data, the fuel cycle services unit process costs have been assumed to be the same for the PBMR as for LWR-UOX.

Figure 3.7 displays the time evolution of the market share of LWR's and PBMRs and the deployments of fuel cycle services needed to meet energy demand. By 2050 most of the LWR's which had been already deployed prior to 2010 have been decommissioned, and the 2000 GWe nuclear energy supply is shared 60% for LWR's with 40% for PBMR's.

As compared with the LWR 100% market share base case, the deployment of fabrication capacity is reduced owing to the higher discharge burnup of PBMR's (80,000 MWd/tonne vs 50,000 for the LWR's) and to the higher station efficiency of converting heat to electricity (46% vs 35%). This is easily understood by noting that annual mass flows in terms of deployed capacity, discharge burnup, and station efficiency is given by:

$$\frac{\text{kgHM}}{\text{y}} = \left(\frac{\text{MWe days}}{\text{y}} \right) \frac{1}{\eta} \frac{1}{\left(\frac{\text{MW days}}{\text{kg}} \right)_{\text{burnup}}}$$

Thus, the ratio of PBMR heavy metal mass flows to LWR mass flows for the same energy delivery is

$$\frac{\left(\frac{\text{kg HM}}{\text{y}}\right)_{\text{PBMR}}}{\left(\frac{\text{kg HM}}{\text{y}}\right)_{\text{LWR}}} = \frac{\eta_{\text{LWR}}}{\eta_{\text{PBMR}}} \frac{(\text{LWR Burnup})}{(\text{PBMR Burnup})}$$

$$\simeq \frac{35}{46} \frac{50}{80} \simeq 0.48$$

To the contrary, the needed enrichment services and the rate of ore withdrawals are approximately the same for the PBMR and the LWR because the required higher enrichment of PBMR fuel offsets the higher burnup and station efficiency.

Since

$$\frac{(\text{Mass/year})_{\text{ore}}}{(\text{Mass/year})_{\text{fuel}}} = \frac{\varepsilon_{\text{fuel}} - \varepsilon_{\text{tails}}}{\varepsilon_{\text{ore}} - \varepsilon_{\text{tails}}}$$

then the ratio of PBMR ore withdrawals to that of the LWR-UOX for the same energy delivery is

$$\frac{\text{Ore for PBMR}}{\text{Ore for LWR}} = \frac{8 - 0.3}{4.2 - 0.3} \simeq 2$$

and the gain made on burnup and station efficiency is offset by the higher enrichment.

Figure 3.8 shows the trends in fuel resource and waste inventories. The dates of exhaustion of known and speculative ore reserves are little changed from the 100% LWR UOX market share base case, and the inventories of enrichment tails likewise. The spent fuel inventories expressed on a mass scale are reduced a little by 2050 and much more by 2100.[5] The near term decay heat in fuel destined for the repository (dominated at early times by fission products) will be little changed – because the cumulative number of fissions to meet energy demand is always the same.

The time evolution of the energy production cost index and the fractional contributions of its components are displayed in Figure 3.9. Under the current modeling assumption that fabrication costs and disposal costs are the same for PBMR and LWR fuel, slight overall cost index reductions (a fraction of a mill/kwhr) accrue to reduced fabrication and reduced waste interim storage costs. As before, rising ore costs dominate the cost increases and altogether the cost index reduces by 1 mill/kwhr by year 2100.

The lower plots in Figure 3.9 shows the time evolution of several properties of this global energy park – *normalized year-by-year to the corresponding properties for the LWR*

[5] Detailed attribute data are needed to compare toxicity flows to the repository. However, as shown in Section 3.5.1.2 and Figure 3.22, increased buildup of minor actinide content resulting from higher discharge burnup is unfavorable to toxicity in the waste.

100% market share base case energy park. The plots are log-linear. Specifically, in the bottom plot on Figure 3.9, Curve 1 is cost of fuel cycle services in this scenario normalized year-by-year to the cost of the LWR-UOX once-through base case. Similarly Curve 2 is the normalized cumulative spent fuel arisings year by year (whether in interim storage or already in the repository). Curve 3 is the normalized cumulative plutonium mass in that spent fuel; Curve 4 is the same for minor actinides, and Curve 5 is the normalized cumulative ore withdrawals from reserves and new discoveries.

3.3.3 The Mixed LWR-UOX and Prismatic HTGR Once-Through Case

The third scenario considered for the once-through generic fuel cycle is identical to the previous one except that the Prismatic HTGR once-through concept replaces the PBMR. Starting in 2010 it is assumed that all new deployments to meet demand are shared 50/50 between LWR-UOX once-through and Prismatic HTGR once-through power plants. The attributes used for the HTGR are shown in Attachment 2; when compared to the PBMR the salient differences are in a higher enrichment (15.5% vs 8%) and a higher discharge burnup (121 GWd/tonne HM vs 80). The impact on the global fuel cycle of these differences is the focus of the comparison. In the absence of data, the fuel cycle unit process costs for the Prismatic HTGR are assumed to be the same as for the LWR.

The standard set of edits are displayed in Figures 3.10 thru 3.12. When compared to the PBMR case, the higher burnup attained in the Prismatic HTGR and the same high station efficiency somewhat reduces both the scale of fabrication plant deployment and the spent fuel waste arisings as expressed in mass (both per Terrawatt electric and cumulatively).

Again, the higher enrichment used in the Prismatic HTGR has more than offset the higher burnup and station efficiency of the LWR, so the drawdown of the ore reserves is unchanged (exhaustion of the Redbook known plus Speculative reserves again occurring by mid century); and the enrichment services are significantly increased. Under the current assumption that HTGR fuel fabrication and spent fuel handling costs are about the same as those for the LWR, overall costs of fuel cycle services, as before, don't increase above 9 mill/kwhr even though ore cost by late in the century is at the $200/kgU level.

Figure 3.12 shows the resource use, waste arisings and fuel cycle cost index versus time for this scenario *normalized to that of the LWR-UOX once-through base case*.

3.3.4 Observations Regarding Once-Through Generic Fuel Cycles

The once through concepts considered in these scenarios have the considerable merit of being nearer to commercial deployment than many of the other Gen-IV concepts and that is why the modeled scenarios introduce them as early as 2010. Additionally the gas cooled concepts are likely to facilitate nuclear energy expansion into new energy service markets which rely on high temperature – such as process heat and hydrogen manufacture.

Under the assumptions used for the scenarios, the once-through generic fuel cycles never fail to meet demand – because they are assumed to be able to draw from presumed new discoveries of ore subsequent to mid century. Alternately – though not specifically evaluated – the enrichment tailings could have been exploited to harvest some of the 0.3% U235 which remain in them.

The gas cooled reactor once-through generic fuel cycles which have been simulated do not improve on the LWR-UOX once-through base case in regard to the first Gen-IV sustainability goal – they exhaust the Redbook Known plus Speculative reserves by mid century and thereafter rely on a presumption that more than double the currently identified

ore reserves can be found to make it through the century. On most measures (mass, decay heat and toxicity) the waste arisings performance (second sustainability goal, SU-2) is not dramatically improved either – demanding a scale of geologic repository deployment which by 2050 is larger by an order of magnitude from current legislated capacities and larger by many factors of ten in 2100. However, the results illustrate that back end waste management scale can be *moderately* reduced by concepts which achieve high discharge burnup and/or high station efficiency in converting heat to electricity.

The performance of the once-through generic fuel cycle concepts against the Gen-IV cost goal will be dominated by factors other than the fuel cycle. This is because the base case fuel cycle costs are only a 20% contributor to cost of energy production, and they change by no more than a factor of two – even as ore becomes more scarce. Waste management costs modeled here remain fixed at 1 mill/kwhr even as the number of repositories deployed grows dramatically.

Concerning the fuel cycle contribution to the Gen-IV Safety Goal, the health effects to workers deriving from the overall fuel cycle are small compared to background. Moreover reference to life cycle comparisons of alternative energy supply approaches [6] (fossil, renewables, or nuclear), shows that the overall societal effects from the nuclear option are reduced from all other options in this respect. When the health effects to workers of the several links in the fuel cycle are compared, contributions are significantly influenced by mining and milling operations; recycle operations are much less, but occur on top of mining in any case (see Table 3.2). Both are related to harvesting fissile from one source or another to fuel the deployed reactor park. Reactor plant operations are a larger contributor compared to fuel cycle services on overall worker health effects of nuclear energy.[6]

Table 3.2 Human Health Effect to Workers Derived from Non-Reactor Links in Fuel Cycle

Fuel cycle stage	**Collective dose to workers (manSv/GWe)**	
	Once-through	recycle
Mining and milling	0.7	0.55 (1)
Conversion, enrichment	0.02	0.016
Fuel fabrication	0.00657 (2)	0.0941 (5)
Power generation	2.7 (3)	2.7 (3)
Reprocessing, vitrification and interim storage	0	0.012 (4)
TOTAL	**3.43**	**3.37**

Data from an ad hoc Expert Group on Spent Fuel Management Options under the OECD-NEA Committee on Radiation Protection and Public Health

Remarks: (1) Scaled for recycle option based on the need for U_{nat}; dose to workers taken from UNSCEAR88.
(2) Workers: Romans 6.57e-3, Melox 4.3e-1.
(3) Workers: average for French 900 MW(e) units.
(4) Workers: La Hague data.
(5) For recycle option weighted by UO2 and MOX fuel amounts 21.1 t & 55 t).

[6] In any case, the effects are quite negligible compared to background and compared to other energy supply options. W. Kroger, S. Hirschberg & K. Foskolos, "The Attributes of Sustainability of Energy Options" Proceedings of the International Symposium on the Role of Nuclear Energy in a Sustainable Environment, April 19, 2001, U Park Hotel at MIT, Cambridge, MA.

3.3.6 Sensitivity of Results to Energy Demand Growth Rate

All once-through scenarios were rerun for the slower nuclear growth rate of the WEC/IIASA Case C2 (deployed capacity for this case is 350 GWe in year 2000; 1200 GWe in 2050; and 1800 GWe in year 2100) as opposed to the Case B (2000 GWe by 2050 and 6000 GWe by 2100) used for the results displayed above. The plots of results are presented as Figures A3.4 through A3.12 in Attachment 3. (Figures in Attachment 3 are numbered to match the corresponding figures in this chapter.) The salient outcomes are as follows:

a. The date of exhausting the Redbook ore reserves is extended by a decade – from 2050 to 2060; however the required new finds of ore by the year 2100 is significantly reduced – from an additional 42 million tonnes of U down to 10 to 13 million tonnes of U (the smaller number for the 50/50 LWR/PBMR scenario).
b. The spent fuel waste arisings for the LWR-UOX once-through do not change much by year 2050 – down to 900,000 tonnes from 1,000,000 tonnes – in any case many multiples of the Yucca Mountain legislated capacity of 70,000 tonnes. By the year 2100, the arisings are significantly reduced – from 4,200,000 tonnes to 2,300,000 tonnes. The gas reactor improvements on waste arisings drop the year 2100 arisings further to around 1,800,000 tonnes.
c. Even by year 2100, as mentioned in (a), the cumulative ore withdrawals have not reached the 30,000,000 tonnes of U at which point the cost model used here caps ore costs at 200 $/kg. As a result the fuel cycle costs reach only 7 or 8 mills/kwhr – down from 9 for the case B growth rate.

3.3.6 Indicated Directions for Gen-IV Fuel Cycle R&D

The scenario results suggest that for once-through cycles, R&D attention should be given to innovations both in ore prospecting and in mining to facilitate new discoveries of rich deposits and for development of cost effective recovery methods and mill tailing management approaches. Additionally R&D could be allocated to cheaper enrichment technologies to harvest the fissile still remaining in enrichment tails as a way to reduce mining and its health effects.

Anticipating that recycle can improve performance against the first two sustainability goals, fuels development for once-through Gen-IV concepts which employ a fuel supply *fed back from recycle-based fuel cycles*, once they are deployed, may also be a fruitful avenue to meet Gen-IV goals – by *means of a symbiosis with other, more recycle-intensive Gen-IV concepts. Such a symbiosis scenario is evaluated in Section 3.5*.4.

Finally, since the discharged fuel is to go to waste, R&D could be devoted to developing fuels not only for high burnup but also for robust durability over geologic time scales in a repository.

3.4 Scenario Results for The Partial Recycle Fuel Cycles

The partial recycle fuel cycles introduce a new source of fuel-resource – i.e., that harvested from the accumulated inventory of spent fuel discharged from the LWR-UOX once-through power plants (see Figure 3.3). The net result is to extract additional energy from the ore – prior to disposition of the partially recycled materials to the waste.

3.4.1 The LWR-MOX Mono Recycle Case

In the scenario simulated here, it is assumed that starting in 2010, plutonium harvested from recycling LWR-UOX discharged fuel is used to fabricate MOX fuel assemblies for fueling 100%-MOX-fueled-LWR's newly added to the energy park to meet growing demand. However, if enough plutonium is not available for the initial working inventory of all the required MOX LWR's, then the deficit in capacity is made up by constructing enough new LWR-UOX plants in order to assure that demand is met. Once an LWR-UOX power plant is built (or is present as part of the initial condition) it is assumed to remain an LWR-UOX plant its whole life. Once a MOX power plant is built it remains a MOX plant its whole life. In summary:

Subsequent to 2010

Plants already in existence continue as before, and thereafter the available plutonium harvested by recycling LWR-UOX spent fuel goes to refuel already operating MOX plants first. Given that excess plutonium is available, the first new plants built are LWR-MOX plants fueled with 100% MOX core loadings. Enrichment tails are used as required for MOX fuel fabrication.

If sufficient plutonium is unavailable to supply working inventory for all the new plants required to meet demand, then the remaining new additions will be LWR UOX plants and they will operate in that mode for their whole life.

All LWR-UOX once-through fuel which is discharged adds to the already existing year 2000 legacy inventory of spent UOX fuel available for recycling. All MOX spent fuel goes to a waste inventory in interim storage for 30 years, then to repository emplacement. When UOX spent fuel is recycled, the recovered irradiated uranium goes to an interim storage inventory and the minor actinides and fission products go to a waste inventory in interim storage awaiting geologic disposal.

The attributes of the UOX power plant modeled here are the same as in the base case scenario, and those of the 100% MOX reactor are presented and compared with those of the UOX reactor in Attachment 2.

Given that a large initial inventory of UOX spent fuel already exists[7] as an initial condition, the rate of deployment of new MOX fueled power plants is constrained initially only by the recycling and MOX fabrication capacity which is deployed. A time can come when that initial inventory of spent UOX fuel plus all later additions to it has been drawn

[7] The current worldwide inventory of UOX spent fuel is modeled as 250,000 tonnes of heavy metal.

down to zero; thereafter the availability of plutonium is constrained by its production rate in the LWR-UOX power plants. *At that time the deployment ratio between LWR-UOX and LWR-MOX would settle out such that the plutonium supply to refuel the existing MOX reactors is provided by the UOX reactors in a symbiotic balance with no shortage and no excess.* Since the energy park is growing, the deployment ratio of MOX to UOX is lower then its steady state value by that amount which is needed to permit accumulation of the initial working inventories for the next additions of MOX reactors. Using the attributes of the UOX and MOX power plants shown in the Attachment 2 it can be found that a mix of UOX and MOX plants of 87%/13% is such as to just feed the MOX plants with reload Pu generated in the UOX plants at *steady state*. The doubling time of the global nuclear energy park is about 20 years (going from 350 to 2000 GWe over 50 years). When account is taken of the need to accumulate Pu for new MOX plant working inventories, the UOX to MOX plant ratio increases to about 90%/10% and so it is clear even without a scenario evaluation that the overall effect of the MOX mono recycle option is to extend the duration of ore reserves by about 10% -- or 5 years more before exhaustion of Redbook reserves at mid century. This negligible extension of ore reserves is illustrated in Figure 3.14 where the inventories of ore, enrichment tails and spent fuel are displayed for this scenario.

Figure 3.14 shows that the MOX spent fuel inventory and the recovered irradiated uranium inventories both grow subsequent to 2010, but that the growth of the UOX spent fuel inventory is capped in 2010 and subsequently drawn down by the mid century to only that which is in the pipeline awaiting recycling. As compared with the base case UOX once-through scenario, the spent fuel inventories (total of UOX and MOX) are *very substantially reduced* – amounting in 2050 to about 70,000 tons of HM in the spent fuel – down from 1,000,000 tonnes for the base case. The deficit appears in the form of ~900,000 tonnes of irradiated uranium recovered during the recycling of the UOX fuel and placed into interim storage. For the unit cost parameters assumed for these scenarios, this results in a cost saving owing to lesser cost of storing separated irradiated uranium as compared to storing discharged fuel for 30 years then packaging and shipping it to a repository. Disposal costs remain dominated by the one mill/kwhr repository charge which is unchanged from the base case.

Figure 3.13 displays the scale of fuel cycle services deployed in this scenario. Enrichment services are reduced slightly; overall fabrication rate is the same but is now divided 90% UOX and 10% MOX. By 2050 the recycling capacity has grown to ~38,000 tonnes/year – up by a factor of nine from the ~4,000 tonnes/year currently deployed.[8]

The cost index is shown in Figure 3.15. The overall increase of fuel cycle services is around a factor of two as MOX mono recycle penetrates the market (in 2010) at a scale needed to both draw down the 250,000 tonnes of legacy spent fuel and keep up with new additions. As time goes on, the cost index increase is still dominated by rising ore cost

[8] The PUREX reprocessing plant attributes are given in the Attachment 2. The unit process costs of fuel cycle services are listed there; reprocessing costs of 800 dollars/kgHM and refabrication costs of 1100 dollars/kg MOX HM are used.

because 90% of the park is still LWR-UOX and the waste management expenses benefit slightly from the cheaper storage of separated uranium as compared to UOX spent fuel.

An initial 25 year surge in the recycling component of the fuel cycle costs is seen between 2010 and 2035. This resulted from a scenario choice on how fast to draw down the initial legacy inventory of UOX spent fuel already in storage; the explanation is as follows. The simulated lag in construction of recycling plants is 5 year construction time plus 2 year licensing time. A decision to build was simulated when the projected need for plutonium seven years hence could not be met by the current-deployed recycling plants. When a new recycling plant comes on line, it initially has excess capacity – which is turned to the task of drawdown of the existing backlog inventory of UOX discharged fuel to create an inventory of separated plutonium available for meeting *future* refueling needs and providing initial working inventory for MOX power plants. A parameter study was run to evaluate the effects of a more or less aggressive buildup rate of recycling capacity – to more quickly drawndown the existing inventory of UOX spent fuel and more quickly build up the inventory of separated plutonium ready for fueling the initial working inventory of new MOX power plants. The parameter survey postulated additions of recycling capacity (worldwide) in increments of 1400, 2800, 4200, 5600 and 7000 tonnes/year plants – thereby more aggressively producing excess recycling capacity and more aggressively drawing down the UOX discharged fuel inventory. The survey showed the effect on UOX spent fuel inventory – elimination by 2030 for the fastest recycling buildup to a halving by 2100 for the slowest recycling buildup. Based on the survey we elected to add capacity in 18,000 tU/y increments (i.e., ~10 LaHague-sized plants added worldwide in year 2012) so as to draw the excess UOX spent fuel inventory to zero by mid century. However, Figure 3.13 shows that after a transient lasting 20 years, the LWR-UOX to LWR-MOX plant ratio settles into its quasi-equilibrium ratio of 90%/10% no matter what the rate of recycling. As displayed in the bottom plot of Figure 3.13, all that is accomplished by too fast a drawdown is to create an inventory of plutonium which can't immediately be placed in a reactor – so we exchange the cost of storage of spent fuel with a higher cost of storage of plutonium countered by a lower cost of storage of recovered irradiated uranium. Examination of the unit costs and assuming a 97%/3% U to Pu split shows that it is a wash as far as cost goes – but if the recycler is not the owner of the fuel, he has a disincentive to build excess recycling capacity. It is likely that recycling capacity buildup would be slower than modeled here, and the UOX spent fuel inventory would persist well beyond mid century.

Figure 3.15 shows the resource drawdown, waste arisings, and fuel cycle cost indicators for the MOX mono recycle/LWR-UOX symbiosis case *normalized year-by-year to the LWR UOX once-though base case*. Normalized cost does not increase once the 20 year surge in recycling is completed in 2030. By 2040 the benefits of waste reductions are seen – a factor of ten reduction in spent fuel mass already emplaced in – or destined for – the repository, and a factor of two reduction in the plutonium contained in that fuel.[9] Minor actinide content of HLW and spent fuel in or destined for the repository increases somewhat

[9] The behavior of the normalized waste arising curves between 2010 and 2030 is an artifact of the accounting algorithm – since discharged UOX fuel is already "declared" in year 2000 to be "destined: for reprocessing, it is not counted against the waste arisings "destined" for the repository. But there is no MOX spent fuel yet. So the numerator of the normalized waste arisings index is zero – which distorts the presentation on a log scale.

due to increased exposures of the original uranium to a thermal neutron flux (first as UOX then as MOX). There is little gain however in the drawdown of ore reserves – mid century still sees the exhaustion of Redbook Known plus Speculative reserves.

3.4.2 The LWR/CANDU DUPIC Partial Recycle Case

The DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors) scenario involves partial (mono) recycle of LWR spent fuel (SF) materials in CANDU reactors. The DUPIC recycle is a non aqueous process for removing volatile fission products from the LWR spent fuel. The LWR SF is mechanically separated into two major streams: (1) the UO2 with non volatile fission products and plutonium and minor actinides and (2) the Spent Fuel (SF) cladding. The UO2 with fission products and actinides is refabricated into CANDU fuel pins and assemblies. These fuel assemblies are used to fuel CANDU reactors and then disposed of as SF in a geological repository.

The scenario assumes that between the years 2010 and 2100, CANDU reactors are built at a sufficient rate (for example, for achieving the ratio of 2 PWRs to 1 CANDU reactor) so that almost all LWR SF including PWR and BWR spent fuel is mono recycled through CANDU reactors i.e., all LWR SF is mono recycled; none goes directly to waste.

CANDU (DUPIC) and LWR-UOX reactors will be built to meet new demand in a ratio such that when steady state is reached the SF from two LWRs will be used to fuel one DUPIC reactor. However, to reach a final balance between SF generated from LWRs and recycled to DUPICs, the initial LWR SF (250 kt) and SF continually generated by already-existing LWRs must first be used up. Thus an initial high buildup of DUPICs is assumed relative to LWRs – followed by gradual decrease in buildup rate until a steady rate of buildup of two LWRs to one DUPIC is achieved. The ratios used in this scenario are as follows. Between the years 2015 and 2020, DUPIC are deployed at rate which will satisfy 33% of the new and replacement electrical capacity while new LWRs satisfy the remaining 67%. The ratio is 30%, between 2020 and 2025. The share decreases to 25% between 2025 and 2037, and beyond 2037, a constant share of 22% is assumed. The 22% rate is based on an LWR SF discharge rate that is ½ of the DUPIC discharge rate. Based on Tables A2.5.5, 6, and 7 in Attachment 2 – which shows the attributes for DUPIC type reactors, the SF discharge rate is about 46000 kg/y. This is double the LWR discharge rate used in the scenario (about 23000 kg/y). Based on reactor powers of 1.3 GWe and 0.713 GWe, for LWRs and DUPICs, respectively, the pace of building two LWRs for each DUPIC will yield a ~80%/20% LWR/CANDU share of the energy deployment of satisfy new and replacement electrical capacity once the initial legacy consumption transient is over.

To simply the scenario simulation, a number of assumptions were made as follows:

- No special modeling for the CANDU reactors before the year 2020 is included in the DYMOND code. The actual existing CANDU capacity of about 5% (about 18GWe) of the world capacity, before the year 2020, is modeled as part of the existing LWR capacity. (Using the DUPIC fuel in the existing CANDUs will have small effect in depleting the existing SF (about 250 kt) and the newly generated SF from new and old LWRs.)

- After 2020, the DUPIC fuel fabricated from LWR SF is used only in new DUPIC type reactors that are built after the year 2020.
- No LWR SF (legacy or newly produced) will be sent to the repository since it will be needed for feeding DUPIC type reactors.
- To be consistent with other scenarios, SF storage charge is a one time charge of $50 per kg HM + $5 per kg-year of storage (this cost has been used for all types of reactors in other scenarios) instead of using $32 per kg-year specified by the CANDU/DUPIC developers and given in Table A2.5.7 of Attachment 2. (By the year 2100, the use of $32 per kg-year would have increased the total cost by about 10%.)

The results for this scenario are shown in Figures 3.16, 3.17, 3.18. Figure 3.16 shows the electricity production market share of the DUPIC reactors compared to the LWRs. By the year 2050 the quasi steady state is attained at about 18% of the total electricity generation capacity. Figure 3.17 shows the Known and Speculative U resources will be depleted by the mid century. The continuing buildup of LWRs will consume further resources; by the year 2100, about 33,000 kt of new U discoveries will be needed. This amount is about 20% less than the new resources needed for once-through LWR base case scenario.

The SF from this scenario includes SF from the LWRs (all in interim storage) and SF from the DUPIC reactors (in storage or sent to the repository). Figure 3.17 shows their time evolution. The UO_2 SF from the LWRs is the majority part of the total SF until it reaches a peak around the year 2040. Beyond 2040, the buildup of new DUPICs starts to consume the LWR SF at a larger rate than its production, and LWR SF starts to decrease. After 2052 the SF from the DUPIC reactors is the major inventory of SF. By the year 2085, all excess LWR SF will have been consumed, and LWR SF generation will be in balance with its recycling in DUPIC reactors. By the year 2100, the cumulative amount of SF from the energy park destined for the repository will be in the amount of about 2,500,000 tonnes. This is 40% less than that associated with the once-through LWR base case scenario.

The enrichment requirements for this scenario are lower than for the once-through LWR scenario as a result of the reduction on demand for enriched U in LWRs. By the year 2100, about 20% reduction in enrichment rate is expected. The fabrication rate, however, increases for this scenario compared to the LWRs-only scenario. CANDU reactors fueled by DUPIC fuel consume double the amount of fuel consumed by LWRs. About 40% increase in the total fabrication rate is expected by the year 2100.

Cost index estimates for this scenario are shown in Figure 3.18. The mining and enrichment costs, are the major factor but achieve a 20% reduction by the year 2100 compared to the LWR base scenario. The cost of fabrication increases compared to the LWR base case scenario, because of larger amounts of fuel fabricated for the DUPIC reactors (this cost will be more than double by the year 2100). Additional costs for this scenario over the base scenario are the recycling costs (mainly include the cost of shipping the LWR SF to the fabrication plants), and the cost of disposing of the HLW from the fabrication plants (0.5% loss fraction of fuel during fabrication). The increase in total cost due to those additional

costs almost offset the decrease in the enrichment and mining costs – leading to total cost estimates that are similar to the cost estimates for the base case LWR scenario.

The differences between the performance of this scenario and the base case LWR once through scenario are summarized in Figure 3.18, through the normalized indexes. The fuel cycle service cost of this scenario ends up about the same. The ore withdrawals index reaches about 0.82 by the end of simulation, reflecting an 18% savings in U resources through using DUPIC reactors to (in effect) extend the burnup of the LWR-UOX feedstock. About 40% reduction in the mass of SF by 2100, compared to the base scenario, is achieved as a result of the LWR SF mono recycling in the DUPIC reactors. This reduction leads to reductions in the amounts of Pu and minor actinide (MA) destined for the repository – reflected in the Pu and MA indexes. The Pu index is reduced to about 0.5 and the MA index is reduced to 0.64. This is a result of both the 40% reduction in SF, a lower Pu fraction compared to LWR SF, but a slight increase in MA fraction in DUPIC SF compared to LWR SF (~0.163% compared to ~0.149%, respectively).

3.4.3 Observations Regarding Partial Recycle Generic Fuel Cycles

Compared to the LWR-UOX once-through base case, the LWR-UOX/MOX mono recycle symbiosis adds little to the first sustainability goal, SU-1 – extending the date of Redbook ore exhaustion by less than a decade.

To the contrary, in terms of heavy metal of mass destined for the repository, a very significant gain is made by virtue of separating out the recovered irradiated uranium from the UOX spent fuel (~ 96% of its mass) and consigning it to interim storage. As always, short-term heat load destined for the repository depends only on cumulative fissions and is always the same so long as the energy demand is met. However as to long term toxicity destined for the repository, the minor actinide load (which dominate long term toxicity) is actually a little worse than the LWR-UOX once-through base case. All in all the gains against the Gen-IV waste management sustainability goal SU-2 which accrue to MOX mono recycle, are big enough to be important, but the gains against goal SU-1 on resource use are marginal.

Enrichment services and costs go down only slightly whereas significant recycling capacity is added, and 10% of fuel fabrication is the more expensive MOX fabrication. Even so, the cost index increases are small in any case, and after mid century savings accrue from reduced storage costs of UOX spent fuel – replaced by the lesser cost of interim storage of recovered irradiated uranium and of HLW disposal.

The CANDU DUPIC symbiosis with LWR-UOX does little (less than 5 years) to extend the date of exhausting Redbook ore resources. For DUPIC mono recycle, since the U is not removed from the UOX spent fuel and set aside (as it is for the MOX recycle case), two effects take place; the total new finds of uranium ore by the year 2100 are reduced relative not only to the LWR-UOX once-through base case but also relative to MOX mono recycle case (a 20% CANDU energy share vs a 10% LWR-MOX share). On the other hand, the mass of spent fuel destined for the repository from the overall energy park, while reduced somewhat from the LWR once-through base case, is not reduced by factors of 10 as it is for

the MOX mono recycle case. By year 2100 the park spent fuel masses are 4,200,000 tonnes; 380,000 tonnes and 2,500,000 tonnes for the base case, MOX, and DUPIC scenarios respectively.

Other differences also occur: DUPIC scenario enrichment capacity is reduced relative to both the base case and the MOX mono recycle case; fabrication capacity goes up relative to both. The cost index climbs to 9 mills/kwhr for both DUPIC and for MOX mono recycle approaches – and it is dominated by ore costs. The cost rises earlier in the century for MOX than for DUPIC, based on the PUREX and DUPIC recycle costs which were used to model these scenarios.

3.4.4 Sensitivity of Results to Energy Demand Growth Rate

The MOX mono recycle scenario was rerun for the slower nuclear growth rate of WEC/IIASA Case C2 and the plots are presented in Figures A3.13 through A3.15 in Attachment 3. As compared with the LWR-UOX once-through case for the same (C2) growth rate, the salient outcomes are as follows:

a. The date of exhausting the Redbook ore reserves is extended to 2065 i.e., again about 5 years longer than the once-through scenario for the same growth rate. New ore finds by year 2100 drop from 11 million tonnes to about 10 million tonnes.
b. As with the Case B growth rate, the big payoff in MOX partial recycle comes in the dramatic reduction in waste mass destined for the repository – achieved by separating and storing the recovered irradiated uranium discharged in spent fuel from the 90% LWR-UOX fraction of the park.
c. The fuel cycle cost index remains at or below 9 mill/kwhr

3.4.5 Indicated Directions for Gen-IV Fuel Cycle R&D

All-in-all, the gains against the Gen-IV waste management sustainability goal SU-2 which accrue to MOX mono recycle, are big enough to be important, but the gains against goal SU-1 ore resource use are marginal.

For DUPIC LWR/CANDU mono recycle, the Redbook ore exhaustion is not extended, but larger gains are made in reducing required new ore finds; waste mass destined for the repository is somewhat reduced but nowhere near the reductions from MOX mono recycle.

None-the-less, partial recycle (both MOX and DUPIC) has the significant virtue that it can be applied in the near term using power plants already deployed and (at 100% MOX loading or advanced CANDU) expected to be deployable within ten years. *Moreover, in the case of MOX, the gains in separating the 96% irradiated uranium component of spent fuel and the separate handling of the plutonium, minor actinides, and fission products may provide important opportunities to benefit Gen-IV Goals on efficient use of repository capacity and on costs of waste management. Similarly, the technology developed for remote fabrication of radioactive DUPIC fuel will benefit future needs when remote fabrication may*

be required for all fuel fed back from breeding after ore reserves are exhausted near mid century. This R&D needed now on integrated waste management and on remote fuel refabrication is discussed at length in Chapter 6 and should be a strong focus area for Gen-IV fuel cycle crosscutting R&D.

Lastly, partial recycle can serve as a technological and institutional transition to full recycle – where gains toward Gen-IV sustainability goals are significant indeed – as is illustrated next.

3.5 Scenario Results for The Full Fissile (Pu) and Full Actinide (Transuranic) Recycle Fuel Cycles

The full recycle fuel cycles introduce both multiple recycle and fast neutron spectrum reactors which as shown in Figure 3.3 further extends the sources of fuel and shrinks the sources of waste. These technologies also provide (as will be illustrated) the means for substantial flexibility in the nuclear energy park's capacity to respond to changing conditions of fuel resource availability and waste management goals. *Net fissile material production per unit of energy delivered by the energy park overall can be adjusted for keeping supply in balance with need and avoiding the buildup of fissile inventories in interim storage or in waste.* The flexibility can be exploited by changing core reload patterns *in fast reactor power plants already deployed*; this significantly shortens the time response of the nuclear energy park in responding to changing needs. The introduction of partial market share for full recycle fuel cycles and fast spectrum reactor types would occur over a multi-decade transition from the current and near term nuclear deployments of thermal spectrum power plants – which provides opportunities for beneficial symbiosis. Initial needs of full recycle and fast spectrum systems will be for waste management symbiosis; a second transition will occur later for fissile supply symbiosis.

Three scenarios are used to illustrate these features:

(1) In the first scenario, LWR UOX once-through plants are deployed paired with a fast *burner* reactor operating on a closed cycle to manage the back end of the fuel cycle – the goal is to reduce the waste burden destined for the repository from the nuclear energy park overall – while retaining a dominance of LWR-UOX once-through plants in the energy mix.

(2) The second scenario illustrates the use of fissile self-sufficient fast reactor closed fuel cycles in symbiosis with thermal once-through power plants to both manage the waste from the entire park and to reduce dependence on virgin ore reserves.

(3) The third scenario illustrates the efficacy of converting already deployed fast reactors from a burning (or fissile-self-sufficient) operation to breeding at the appropriate time as the way to cap virgin ore withdrawals when economic conditions favor the switchover – with no degradation of the continuing waste management function.

The reactor and fuel cycle attribute sets used for these scenario cases are documented in the Attachment 2. The fast burner has a transuranic conversion ratio of 0.5 (i.e., a net consumer of transuranics). The doubling times of the three different net transuranic producing "breeders" are ∞ for the fissile self sufficient core; ~45 years for the 1.25 breeding ratio design, and ~7 years for the 1.72 breeding ratio design -- (as compared to 20-25 year doubling time for the nuclear electrical component of the global energy demand growth being used from the WEC/IIASA Case B energy projection).

3.5.1 The Symbiotic LWR-Fast Burner Reactor Waste Management Case

In this scenario, it is assumed that starting in 2025, all new deployments to meet demand are filled by a symbiotic pair of power plants comprised of a LWR-UOX 1000 MWe plant and a 600 MWe fast burner reactor having a conversion ratio[10] of 0.5. The LWR's operate once-through, drawing their initial working inventory and refueling needs from virgin ore. Their discharged fuel is used to meet the refueling needs of its companion fast burner reactor which is operating as a net TRU burner on a full TRU recycle closed fuel cycle. Deployment of the fast burner reactors requires a source of TRU for initial working inventory as well as a source of TRU for annual refueling. The initial working inventories of the newly deployed fast burner reactors are derived from TRU obtained by recycling the legacy and still growing inventory of discharged fuel from LWR-UOX reactors which had been deployed previously to the 2025 start date of the new symbiotic deployment strategy. These previously deployed LWR's continue to operate once-through to the end of their 60 year lifetimes and to add to the inventory of spent fuel, whereas the TRU contained in the fuel discharged from all *new* LWR's is consumed by its companion fast burner reactor (and the recovered irradiated uranium is mostly set aside in interim storage except for that which is needed for fast burner fuel fabrication).

Figures 3.19 through 3.21 show the results from the scenario. The striking result of this symbiotic deployment is the dramatic reduction achieved in the mass and toxicity of material consigned to geologic disposal. The spent fuel from the newly deployed LWR's is not sent to waste but instead feeds the companion fast burner reactor's fueling needs. The LWR discharge fuel is recycled as follows: the majority of the recovered irradiated uranium is set aside in interim storage and the recovered TRU is sent to the fast burner reactor closed fuel cycle as incremental makeup to the closed fast reactor fuel cycle. This feedstock into the fast reactor cycle is eventually fully consumed through multiple recycle because the fast reactor is a net burner, not a breeder. Using this symbiotic pairing of thermal and fast reactors, *fission products and trace losses of TRU's from the recycle/refabrication steps are the only waste product going to the geologic repository from the nuclear energy park.* A factor of about 10^3 reduction in mass and about 10^2 in toxicity flow to the repository is achieved relative to the once-through cycle delivering the same energy services.

The legacy inventory of discharged fuel from LWR's deployed prior to 2025 and still operating on the once-through cycle subsequent to 2025 is not sent to waste either. This inventory is recycled to supply the initial working inventories for the new fast burner reactors as they are deployed. The initial legacy inventory (250,000 tonnes) of LWR spent fuel is eventually worked down to zero, and the scenario modeled here achieves a 80% LWR/20% fast burner reactor symbiotic ratio before mid century – where all initial spent fuel inventories are exhausted and all mass flows are in balance.

At zero growth rate, the symbiotic energy delivery mix of LWR's to fast burner reactors deployment is 63% LWR's to 37% fast reactors; each 1 GWe LWR is accompanied by a 600 MWe fast burner reactor in which the burner companion power rating and breeding

[10] Conversion ratio is 0.5 when for each two transuranic atoms fissioned, one new transuranic atom is created from neutron capture on U^{238}.

ratio are sized to exactly fully consume the TRU contained in the LWR discharged fuel so that no inventories of spent fuel or separated transuranics build up and no actinides from the power park are consigned to waste. But in an economy growing at the rate assumed here, once the initial legacy stockpile of spent UOX fuel from previous operations has been consumed, the ratio must increase to 80% LWR/20% fast reactor to account for the need for saving part of the new LWR spent fuel output to supply initial working inventories for new fast reactors.

The fact that part of the energy is supplied by further burnup of the ore mined originally for the LWR's, of course, reduces the drawdown rate on virgin ore as compared to the reference base case. Even so, as shown in Figure 3.20, it only negligibly extends the date of exhaustion of Redbook Known plus Speculative Reserves – by less than a decade. This is because it is already 2045 before the fast burner fleet has built up to a level of significance in the energy park and by then the Redbook Known plus Speculative reserves are nearly exhausted, – there are still 80% of LWR's in the energy park; and their forward fueling needs already exceed Redbook reserves. Further operation and further growth of this symbiotic energy park relies on an assumption of finding and exploiting new ore reserves.

In the scenario modeled here the fast reactor recycling/refabrication plants are represented as small-scale facilities sized for dedicated support of co-sited fast burner reactor plants; they handle 50 to 200 tonnes/year of throughput, depending on number of fast burners at the site. The plants for recycling the backlog of LWR spent fuel which provide initial working inventory for the fast reactor deployment are modeled as large central plants – as in the previous PUREX recycle partial MOX recycle scenario – i.e., ~1400 tonne/year plants. Unit process costs for recycle and refabrication in the UOX and fast reactor cycles are different as shown in the Attachment 2. Figure 3.21 shows the evolution of overall fuel cycle service costs for the symbiotic energy park.

The bottom plot in Figure 3.21 shows the performance values for this energy park scenario *ratioed year-by-year to those of the reference LWR once-through base case*. The symbiotic pairing of LWR's with fast burner companions achieves significant (three order of magnitude) reductions in waste mass from the nuclear energy park at a favorable cost benefit ratio in terms of the fuel cycle component of cost, but it does little to help the resource goal SU-1.

3.5.1.1 Effects of A Slower Demand Growth Rate

This scenario was rerun for the slower grow rate of the WEC/IIASA Case C2, and the results are displayed in Figures A3.19 through A3.21 in Attachment 3. As compared to the Case B growth rate, the results show:

a. The date of exhaustion of Redbook ore reserves is extended to the year 2070, (from 2060 for the LWR once-through base case for growth rate C2) and the required new finds by year 2100 is dramatically reduced – from 33 million tonnes U to about 7 million tonnes.

b. The slower growth rate of the park reduces the construction rate of new fast burner reactors and the demands for TRU to fuel their initial working inventories. Whereas, for Case B, the excess UOX spent fuel inventory was worked down to zero by 2046, it persists in case C2 to the year 2087; rising at its peak to 375,000 tonnes in interim storage (up from the 250,000 initial condition in year 2000.)

c. The ratio of fast burner to LWR-UOX plants in the park slowly rises above the 20%/80% value for Case B and by century's end is about 35%/65%. This is because it is not necessary to set aside as much LWR spent fuel for initial working inventory of fast reactors – driving the ratio toward the 37/63% split which applies at zero growth rate steady state.

d. In both cases, the fuel cycle cost index rises to and remains at about 9 mills/kwhr.

3.5.1.2 Full TRU Versus Full Pu Recycle – Minor Actinide Effects on Toxicity Sent to The Repository

The LWR/Fast Burner symbiosis scenario discussed above is based on a fuel cycle back-end management approach which was investigated (for the steady state) in much greater detail by an OECD-NEA Expert Group on the Comparative Study of Accelerator Driven Systems and Fast Reactors in Advanced Fuel Cycles. In that OECD-NEA study, the *full TRU recycle case* – which is the one represented above – was compared to a *full Pu recycle case* – where the deployment is essentially the same but where *the minor actinides are sent to waste along with fission products,* and only plutonium is multi recycled in the fast burner reactor. It was shown in the OECD-NEA study that for the full plutonium recycle case, reductions in *mass* consigned to geologic disposal are essentially identical to those obtained here, but *toxicity* reductions are very substantially less. *Toxicity reductions of several hundreds achieved with full TRU recycle reverted to reductions of factors less than ten when the minor actinides were sent to waste rather than sent to the fast burner reactor for multiple recycle to fission consumption* (see Fig 3.22).

In the OECD-NEA study, several alternative ways were evaluated to achieve the desired minor actinide recycle and consumption: one was the fast burner reactor using full TRU recycle technology modeled here; one was a fast Pu burner reactor of conversion ratio 0.5, using full Pu recycle technology – followed by a second strata accelerator-driven fast burner which consumed the minor actinides through multiple recycle. These two produced equally effective reductions in mass and toxicity flows to the geologic repository. Figure 3.22, taken from the OECD-NEA Expert Group study, indicates the importance to toxicity reduction of burning minor actinides as well as plutonium and illustrates that numerous technological alternatives are available to achieve that result.

This scenario displays the utility of multiple recycle through fast burner reactors for addressing the Gen-IV waste management goal, SU-2; the data from the OECD-NEA study shows the importance of minor actinide recycle to achieving desired levels of toxicity reduction. The scenario also illustrates that more efficient extraction of energy from the virgin ore by harvesting and burning the transuranics created in the LWR's *ultimately*

reduces the drawdown rate of ore reserves by the fraction of the energy output supplied by fast burner reactors.

But to effectively extend the time period over which uranium in the Redbook Known plus Speculative Reserves are available for thermal reactor plants already deployed or deployed in the future, two things are necessary: (1) this more efficient usage must be *initiated early in the growth of the energy park,* and (2) a switchover from burning to breeding must occur at a propitious time.

The next scenarios illustrate an architecture intended to simultaneously address waste management and further enhanced resource utilization goals.

3.5.2 Extending The Duration of Ore Reserves by Transition to Self Generation of Fissile

Starting in 2020 it is assumed that the first choice for capacity additions is fissile-self sufficient fast reactors which operate thereafter on a fissile-self-generated full TRU recycle fuel cycle. (This "breeding ratio" of 1.0 is achieved by the core loading pattern selected in a design which can achieve breeding ratios over a broad range simply by changing the core reload pattern.) The initial working inventories for the fast reactor additions are derived from TRU recovered when recycling the existing and growing inventory of LWR discharged fuel from the already-deployed LWR's. *If inadequate transuranics exist for new fast reactor deployments to meet demand, then new LWR plants operating on the once-through UOX cycle are constructed to meet demand.*

In the ensuing years after 2020, the fast reactor plants are fissile self-sufficient but produce no excess fissile. As recycling of the initial legacy LWR spent fuel inventory and its ongoing additions to supply initial inventories for new fast reactors consume the LWR spent fuel inventory, eventually the availability of legacy TRU to supply new fast reactor deployment nears depletion – thereafter the market shares of thermal and fast reactor deployment is controlled by balancing the source of fissile coming from LWRs in the park to supply initial inventory requirements needed to deploy new fissile self-sufficient fast reactors. This scenario illustrates those tradeoffs. Figures 3.23 through 3.25 show the results.

The bottom plot in Figure 3.23 shows that within a decade – before 2030 – fissile supplies from recycled LWR legacy discharge fuel would run short of requirements for fast reactors alone to keep up with demand i.e., the fast reactor deployment rate quickly becomes fissile limited because of the large fissile working inventory of fast spectrum reactors[11]. LWR deployments, fueled by virgin ore, would continue to supplement the shortfall in new fast reactor availability and their deployment would increase so as to meet demand. The growing energy park would settle into a symbiotic ratio of 80% LWR's/20% fissile self sufficient fast reactors – with LWR discharge fueling the initial inventories of new fast reactor starts; once deployed the fast reactors would be fissile self sufficient but would

[11] This case can be compared to the fissile self sufficient case for a thermal spectrum system having an exceedingly small fissile working inventory. That case is discussed in Section 3.7.2, and it is shown that the LWR spent fuel inventory is sufficient to support many decades of such reactor deployment.

produce no excess TRU. The availability of Redbook virgin ore reserves could be extended by only 5 years to 2055 before new discoveries become necessary. The cumulative needs (Redbook + new finds) of the LWR once-through dominated energy park to year 2100 would be slightly reduced to ~45 million tonnes, from ~55 million tonnes for the LWR base case because LWR's fueled on virgin ore still comprise 80% market share.

As in the earlier full TRU recycle scenario, the reductions in waste arisings from the energy complex are factors of a thousand lower in mass and factors of a hundred lower in toxicity as compared to the LWR once-through base case because nothing but fission products and trace losses of heavy metal in recycle operation go to waste.

Figure 3.25 displays the performance of the energy park for this scenario ratioed year-by-year to the reference base case scenario.

Comparing with the previous case, the deployment of fissile self sufficient fast reactor cycles (BR=1.0) in place of burner cycles (BR=0.5) can achieve the same waste management performance against goal SU-2, but it improves performance on efficient resource use Goal SU-1 only marginally. It is evident that the fissile self-sufficient fast reactor modeled in this scenario can never keep up with demand growth – no matter how early in the century it is deployed because, with a breeding ratio of only 1.0, its doubling time (the time for it to accumulate enough excess fissile material sufficient to start up a new reactor power plant including out of reactor fuel cycle inventory) is infinity – whereas the doubling time of the nuclear energy park is 20-25 years. *In order to achieve a transition to sustainability – where eventually no more ore drawdown takes place and the energy park is "fueled" by the ^{238}U inventories which are stored in the enrichment tails and recovered irradiated uranium inventories – it is necessary to deploy breeder fuel cycles having a doubling time less than 20 years.*

From results of several scenarios presented here and in the supplementary scenarios presented in Attachment 3, Table 3.3 tabulates salient results for introduction of fast reactors of increasing conversion and breeding ratio in the 2020 to 2030 time frame for the faster (Case B) and the slower (Case C2) nuclear park growth rates. Examination of Table 3.3 shows:

a. First, for the Case B energy demand growth rate (20-25y doubling time), and breeding introduction in the 2020 to 2030 period, the date of exhausting the Redbook ore reserves is extended only slightly with increasing breeding ratio – remaining near mid century except for the very high breeding ratio case. Similarly the required new finds of ore by year 2100 are not strongly reduced – remaining in the additional 25 to 35 million tonne range.
b. On the other hand, the effect of a reduction in energy demand growth rate is quite strong, and for the demand growth rate of Case C2, introduction of breeders of 1.25 breeding ratio in 2030 can extend Redbook ore reserves to the latter decades of the century. In the following century only slight extensions of Redbook ore reserves would be required.The high breeding ratio fast reactor can keep up with demand growth rate *within the Redbook ore*

reserves for either demand growth rate – reading 100% market share by year 2100 for Case B or by 2080 for Case C2. Thereafter, no new ore would be required for many centuries, and the growing energy park would be fueled by the ^{238}U contained in the already available inventories of enrichment tails and recovered irradiated uranium.

c. Well in advance of the 100% market share dates, these systems are producing *excess fissile* not needed for fast reactor market penetration. This excess fissile resource becomes available to fuel other reactor types and/or to extend the overall energy market share of nuclear.

The strong dependence of cumulative uranium ore utilization during the transition to sustainability as a function of fast reactor doubling time has been studied systematically by Koike, et al.[12] Figure 3.26, which is reproduced from their paper, shows how, given a fast reactor of breeding ratio equal 1.2, the uranium ore required in breeder introduction scenarios varies with (a) breeder introduction date, (2030, 2050, or 2070), breeder fuel cycle out of reactor dwell time (2 years up to 8 years), which is a surrogate for doubling time, and with growth rate of the global nuclear energy park (installed capacities in 2100 of 2,270 GWe (labeled FREE); 4380 GWe (labeled S1000); 4920 GWe (labeled S750); and 5920 GWe (labeled S550). (Note that their case labeled S550 corresponds to our scenarios for Case B of the WEC/IIASA 1998 projection; and their case labeled FREE is similar to our scenarios for WEC/IIASA Case C2.)

For our slower demand growth scenarios, C2, Koike's results show that breeding could be delayed until ~2070. The role of fast reactors and full TRU recycle would be for waste management until late in the century. Fast reactors and closed cycles would be introduced in the 2020 to 2030 time frame to cap and reverse the buildup of spent fuel waste inventories from once-through or partial recycle deployments.

For the faster energy park growth rate used here, Koike's results show that early fast reactor introduction (by year 2030) and short doubling time are necessities to transition smoothly to sustainability within the Redbook Known plus Speculative ore reserves. Otherwise, the limited TRU availability from LWR discharge plus self breeding retards the buildup achievable for fast reactor market share with the result that *new LWR's must be built in order to meet demand.* But and their lifetime forward fueling requirements exceeds the Redbook reserves. Figure 3.26 shows that the doubling time dependence is stronger than the introduction date dependence and that the doubling time dependence substantially increases in importance as energy demand growth rate increases.

d. As will be shown in the next scenario, early emplacement of full recycle and fast spectrum systems as a minority market share of the energy park for the purpose of a waste management function would provide the mechanism needed for control to hold the overall energy park mass balances in quasi-static equilibrium – avoiding waste buildup and avoiding fissile shortages as high assay ore becomes depleted

[12] R. Koike, Y. Shibata, T. Sanda, and T. Wajima, "A Semi-Empirical Long Term Scenario Analysis: Global Warming and Nuclear Energy", Proceeding of the Global 99 Conference, Jackson Hole, Wyoming, (Sept. 99).

Table 3.3 Transition To A Fissile Self- Sufficient Energy Park

Case / Growth Rate	LWR + FR(BR=0.5)		LWR + FR (BR=1.0)		LWR + FR(BR=1.25)		LWR + FR(BR=1.72)	
	B	C2	B	C2	B	C2	B	C2
Year of Introduction	2027	2027	2020	2020	2030	2030	2020	2020
Exhaust Redbook Ore (y)	2053	2070	2055	2082	2056	2100	∞	∞
Required New Ore by 2100 (tonnes)	$33 \cdot 10^6$	$6.5 \cdot 10^6$	$26 \cdot 10^6$	$2.4 \cdot 10^6$	$23 \cdot 10^6$	0	$1 \cdot 10^6$ excess	$4 \cdot 10^6$ excess
Spent UOX Inventory ⇒ 0 (y)	2045	2087	2066	2078	2042	2062	2070	2050
Max of UOX Excess (kt)	270	380	300	300	255	415	255	255
LWR/FR Ratio	80/20	65/35	65/35	50/50	50/50	50/50	50/50	50/50
year reached (y)	2045	2080	2100	2072	2100	2065	2044	2046
			& still growing	40/60 2100 & still growing	& still growing	10/90 2100	0/100 2100	0/100 2080
Max Pu Excess (kt)	3	0.9	1.2	1.2	4.5	3.1	28	27
start of escalation (y)					2092	2090	2060	2030
Cost Index mill/kwhr	10	10	12	12	14	16.5	25	26
Location of Full Set of Plots	Chapt. 3	Attach. 3	Chapt. 3	Attach. 3	Attach. 3	Attach. 3	Attach. 3	Attach. 3
Figures	19-21	19-21	23-25	23-25	33-35	36-38	39-41	42-44

3.5.3 Fuel Cycle Flexibility: Avoiding Waste Buildup; Avoiding Fissile Shortages via Adjustments in Market Shares of Symbiotic Systems

A persisting feature of all scenarios displayed thus far is the ponderous (many decade) pace of the transition to a Gen-IV energy park. The LWR-UOX market share never falls below 50% before mid century and in some cases the park has not settled into a quasi static distribution of market share even by century's end.

This occurs for several reasons. First, is the assumed 60 year lifetime of capital assets. Once deployed, the high capital cost assets are assumed not to be decommissioned due to obsolescence because fuel costs, even as they rise; remain the lesser component of the cost of energy production. Furthermore, *the plants, once deployed, have been assumed here to operate their whole lifetime on their original fuel cycle: in particular an LWR-UOX plant operates on the once-through cycle for 60 years, so cases occur where their forward fueling requirements may exceed the ore reserves on their first day of deployment.*

The second reason is a fissile availability constraint. New (Gen-IV) plants based on TRU fuel are not put on line until a working inventory of fissile has been accumulated – *not just enough to fuel one reload batch but enough to provide an entire fuel cycle working inventory both for the power plant and for the out of reactor fuel preparation and/or recycle plus fuel preparation pipeline. And in the scenarios, if a Gen-IV plant can't be deployed because of fissile shortage, then an LWR-UOX plant is put on line instead in order to meet demand.*

It is unlikely that plants once deployed would be decommissioned because of fuel cycle obsolescence, *but existing plants, both fast and thermal could be converted to different fuel loadings and fuel cycles.* Fast reactor plants, once deployed, provide a control variable for flexibility in management of transuranics; they need only to reconfigure their reload core patterns to alter their breeding ratio over a rather broad range – from a net burner of TRU with conversion ratio 0.5 to a net breeder of TRU with breeding ratio 1.25 to 1.3 or even higher.

Suppose that full TRU recycle fast burner reactors had been deployed initially to perform a waste management function for the energy park. Then, at the appropriate time, anticipating the exhaustion of affordable uranium ore fissile feedstock, the breeding ratio of all existing and newly deployed fast reactors could be increased to a net breeding value sufficient to manufacture excess fissile material at a rate to fuel the impending need. This switchover could be done simply by adding additional depleted uranium (or recovered recycled uranium) to blanket regions of the fast reactor core loading at their next reload outage – so the time constant for switchover is reduced from the lag time for building and licensing new plants to only the lag time to refuel and turn around the inventories of existing plants. The switchover of previously deployed fast burner or fast fissile self-sufficient power plants and supporting fuel cycle facilities provides a "kick start" to breeding which would be absent if no fast reactor plants had been previously deployed.

The physically achievable growth rate of the energy park on the basis of self generated new fissile is determined by the fast reactor versus thermal reactor energy share and the fast reactor breeding ratio, its working inventory and its cycle turnaround time – (which together determine its cycle doubling time). Given that the new loading pattern would produce a fast reactor fuel cycle with a doubling time shorter than that of the demand growth, the switchover lead time needed to achieve a smooth transition depends on the growth rate of demand vs the physically achievable growth rate of the *breeders in the* energy park and *what fraction of the park is already occupied by fast reactors*. (It depends specifically on the excess neutron balance of the energy park as a whole.) If the switchover occurs too late or the doubling time is too long, fissile shortages will result because the still deployed LWR's will continue to drawdown the ore reserves for decades until they reach their end of life. Alternatively if the switchover is too early or the doubling time is too short, inventories of self generated fissile will accumulate in excess of that needed to meet demand growth.

The next scenario illustrates the upper limits of physically achievable smooth switchover for the Case B demand growth rate. (To bound the range, an early switchover and a breeder design of extremely high fissile production are used.) The scenario starts like a previous case with fissile self sufficient (BR=1.0) fast reactor deployments starting in 2020. But starting in 2030, already deployed fast reactors and all future fast reactor deployments use a core loading pattern which achieves a breeding ratio of 1.72 and a doubling time of only 7 years. The results of this scenario are shown in Figures 3.27 through 3.29.

Starting from its "jump start" with 25% market share already held in 2030 by fast reactors (previously operating as fissile self sufficient) the breeders reach 50% market share by 2050 and although LWR's remain in operation until 2100, diminished LWR fueling needs can be satisfied within the Redbook Known plus Speculative ore reserves.

The legacy inventories of LWR-UOX spent fuel aren't worked down to near zero until about 2075; however, the interim storage inventories never exceed about 500,000 tonnes heavy metal (twice the worldwide current amount).

As in all full TRU recycle cases, the waste repository receives only fission products and trace losses of heavy metal from recycle operations.

By 2100, the park is delivering 6000 GWe of power, is keeping up with demand growth without reliance on discoveries of new ore beyond Redbook reserves, is being fueled on ^{238}U from the multi-century supply of enrichment tails, and is sending only fission products and trace losses of heavy metal to the repository – indicating that the sustainability goals, SU-1 and SU-2 can be met within physically achievable mass flows.

Figure 3.29 shows the performance of this nuclear energy park, normalized year-by-year to the performance of the LWR once-through base case. Gen-IV sustainability goals are achieved over a 70 year transition period and at a increase of about a factor of 4 in fuel cycle cost index over what it would have been for the LWR once-through base case.

Starting in the late 2060's an inventory of excess TRU starts to accumulate as the doubling time of the park's fissile production exceeds needs for keeping up with the doubling time of energy demand. That would be a propitious time to again change the core loading pattern of the breeders – to reduce breeding production so as to just meet requirements for new deployments. Alternately, nuclear park operators might use excess fissile production capacity to feedback fissile to already-deployed thermal reactors (which would change their loading pattern and source of fissile but cap their continued drawdown of virgin ore) – or the excess fissile supply could be used to expand overall energy market share, as fossil sources diminish in availability or desirability. Such a scenario is shown next.

3.5.4. Feedback of Excess Fissile to Thermal Reactors

The previous scenarios have illustrated symbiotic energy parks where thermal reactors draw their fuel from virgin ore and fast reactors manage the spent fuel from thermal reactors and eventually expand to 100% in market share so that withdrawals of virgin ore reserves cease and the fast reactor energy park is fueled from the extensive inventory of ^{238}U from the enrichment tails.

An alternative and *more likely symbiotic energy park is one of fast reactors and thermal reactors each filling appropriate market niches – but fueled exclusively by enrichment tails; with excess fissile from fast reactors fed back to fuel thermal reactors; and with fast reactors fueled by the spent fuel from the thermal reactors and from enrichment tails.* The energy park is shared such that the mass flows of transuranics just balance, no buildups or shortages of transuranics occur, and only fission products and trace recycle losses of actinides go the repository.

For example, consider a steady-state situation of LWR (100%) MOX once-through reactors (with a conversion ratio of 0.6) coupled to fast reactors of breeding ratio 1.4. The steady state transuranic balance is achieved when the LWR market share (energy share) denoted by, X, satisfies the equation:

$$X\ (0.6) + (1\text{-}X)\ (1.4) = 1$$

It is seen that x=0.5 and that the steady state park is comprised of LWR's and fast breeders in a 50/50 market share. When the park is growing, the right member of the equation has to exceed 1 by that amount of excess transuranics needed to replicate the park energy capacity within the energy demand doubling time. (From the equation it is easily seen that thermal reactors of high conversion ratio and low working inventory coupled to fast reactors of high breeding ratio and short out of reactor recycle time favor the maximization of thermal reactor market share.)

In the scenario illustrated next LWR-UOX once-through plants are deployed until 2035. Then, starting in 2035, fast reactors of breeding ratio 1.7 plus LWR-MOX mono recycle reactors fueled by excess transuranics from the fast reactor are deployed to meet demand, and new deployment of LWR-UOX is halted. However, the existing LWR-UOX plants continue to operate on virgin ore until the end of their 60 year life and their spent fuel

(along with the already existing inventory of UOX spent fuel) is recycled to provide initial working inventories for newly deployed fast reactors. An effort is made to favor LWR-MOX deployment over breeder deployment – but under a constraint that fast reactor market share must be sufficient to assure future needs for fissile.

The scenario results are shown in Figures 3.30, 3.31, and 3.32 for the Case B demand growth rate. As in the previous scenario it takes several decades before the fast reactors build up to a level where their excess transuranic production is sufficient not only to keep up with growing demand for new fast reactor deployment – but also to compensate for LWR-UOX plants going offline at the end of their 60 year lifetime. Starting in the 2040 time frame excess transuranics become available for a slow rate of LWR-MOX plant deployment but it isn't until 2080 that their growth rate matches the LWR-UOX rate of decline. By the end of the century the park is at an 80% FR/20% MOX/0% UOX market share and growths of market share is starting to favor MOX over FR's and approaching its asymptotic configuration (an unchanging roughly 50/50 market share of fast and thermal systems) early in the next century.

As in previous full TRU recycle cases, once closed fuel cycle/fast reactors deployment is initiated, the LWR-UOX spent fuel inventories are consumed and only fission products and trace losses of transuranics from recycle operations are destined to be emplaced in the repository. By initiating the transition in the late 2030 time frame, it is possible to convert to a sustainable growing energy park, while meeting demand without significantly exceeding the limits of the Redbook Known plus Speculative ore reserves. The cost index has risen to 20 mills/kwhr by year 2100 due to extensive mass flows through recycle and remote fabrication operations. The performance indexes, normalized year by year to the LWR-UOX once-through base case are shown in Figure 3.32. *The sustainability goals SU-1 and SU-2 are seen to be achievable in a symbiotic park of fast and thermal power plants at a doubling of the fuel cycle cost index from the 9 mill/kwhr predicted for the LWR-UOX base case in year 2100.*

3.5.4.1 Effects of Slower Demand Growth Rate

This scenario was rerun for the slower Case C2 energy demand growth rate and the results are displayed in Figures A3.30 through A3.32 of Attachment 3. For this slower growth rate, the LWR-MOX and FR market shares initially grow at the same rate – each reaching about 25% by 2065. At about that time the inventories of LWR-UOX spent fuel become exhausted and the pipeline of newly discharged UOX had to be preferentially directed to FR construction. By year 2100 the market shares were 30% MOX/70% FR/0% UOX and, as before, the buildup rates have reversed to favor MOX to approach the quasi-static symbiotic 50/50 market shares of a fissile self generating/waste self incinerating energy park. In this case the ore withdrawals remained within the Redbook reserves.

The scenarios was rerun yet again for the faster Case B energy growth rate with the fast reactor out-of-reactor dwell time reduced from 4 years to 2 years to reduce doubling time. The MOX fraction in the park grew to about 17% early (up from 5% for the 4 year out of reactor time) but the end result by year 2100 remained unchanged. Unless there is a large

supply of fissile (in excess of the production rate from the deployed LWR-UOX plants), most of the fissile must be devoted to FR construction leaving little for the MOX construction at mid century for growth rate Case B when the demand growth rate is relatively high.

A more practical scenario than the ones above is one wherein the already existing LWR's cease to rely on UOX, but rather are converted to use MOX fueling using fissile produced in the fast reactors – so as to facilitate a smoother transition to the asymptotic sharing of the market between LWR's and fast reactors. In that case the effective doubling time of the LWR-MOX/FR park overall will be slower than that of conversion to the fast reactors alone. However, the transition can be accelerated by immediately making use of existing LWR's as MOX burners rather than waiting until entire working inventories for new LWR's can be built up. At this time the DYMOND code has not been configured to model this case.

3.5.5 Observations Regarding The Full Recycle Generic Fuel Cycles

The full recycle generic fuel cycles provide the flexibility needed to manage the mass flows in a symbiotic global nuclear energy park. Fast spectrum reactors operating on closed fuel cycles can – on the basis of changeable core loading patterns – perform either as net consumers of transuranics or as net producers of transuranics. When combined symbiotically with thermal reactors in the energy park, such systems function to keep TRU production and destruction in balance and to avoid the buildup of fissile inventories in interim storage or in geologic repositories.

By deploying only 20% market share of fast spectrum closed fuel cycles the flow of all but trace losses of actinides to the geologic repository in a growing nuclear economy can be stopped – not only from future operations, but also from the legacy inventories of spent fuel from all prior nuclear power operations.

Thermal reactors dominate market share at least until mid century (and grow dramatically from current deployment levels no matter what full recycle case is considered). *Tying up in working inventory and burning down inventories of spent fuel from once-through and partial recycle thermal reactor deployments will be the first function for Gen-IV fast reactors/full recycle concepts.* Thermal reactor to fast reactor energy shares in the park needed to achieve Goal SU-2 lie in the range 80%/20% to 65%/35% -- with the necessary fast reactor share decreasing the greater the growth rate of the park – because of need to generate fast reactor initial working inventories in thermal reactors.

Reprocessing allows partitioning of the discharged fuel and handle each component according to its properties. Uranium goes to interim storage for later use in breeder blankets; TRU are used for initial working inventories of new starts or recycled as fuel, and fission products go to waste.

Fission product production depends exclusively on energy delivered from the nuclear power park; since repository capacity (packing density) is controlled by decay heat,

integrated waste management strategies which use multi-decade interim storage to allow decay heat to subside before geologic emplacement are clearly desirable.

Toxicity flow to the geologic repository is dominated initially for 100 years by fission products, but minor actinides dominate long term toxicity source term in the repository *Thus, full TRU recycle performs orders of magnitude better on this measure than does full Pu recycle – which consigns minor actinides to waste (see Fig. 3.22).* However, with full TRU recycle, the fabrication must be done remotely – requiring substantial R&D and increasing costs as represented in the cost index. Despite higher unit costs than for the LWR once-through base case, the cost index for full recycle scenarios changes by less than a factor of 4 from current levels (and factors of two from what it would become given growing ore costs in the once-through cycle). Since fuel cycle costs start out at only 20% of cost of nuclear power, this increase should be tolerable relative to the fuel cycle cost of energy production overall – (as compared with already experienced volatility of fossil fuel energy sources).

The scenarios have shown that to remain within the currently-identified high assay ore reserves while meeting significantly growing demand for electricity from the thermal-reactor dominated nuclear park, conversion from net burning to fissile self-generation must occur well before mid century for the WEC/IIASA Case B energy growth rate and around mid century for the Case C2 energy growth rate; thereafter the fast reactor energy share of the park grows above the 20% needed for waste management. Delay or less aggressive rates of deployment leads to reliance on re-enrichment of enrichment tails and/or prospecting/exploitation of new ore fields.

Whatever the introduction date and deployment rate, once closed cycle fast reactor systems achieve a significant market share, their role switches to a dual function – feeding back fissile fuel supplies to thermal spectrum systems – as well as continuing to manage the waste from the nuclear power park overall. *Results show that – were that to be required – it is physically achievable to deploy a nuclear energy park which is sustainable on self-generated fissile production within the remaining Redbook Known plus Speculative ore reserves.* The park sends only fission products and trace recycle losses of transuranics to the repository. It takes to the end of the century to achieve this transition.

The principal features regarding Gen-IV Sustainability Goals which are illustrated by the full recycle scenarios relies on exploitation of recycle and the flexibility in the neutron economy of fast spectrum systems. In the early decades of Gen-IV deployment these features are exploited to manage the spent fuel waste inventories from previous and ongoing once-through cycles by subsuming them into inventories of newly deployed fast spectrum power plants, -- then in latter decades to *transition to a fully sustainable symbiotic growing energy supply within the finite resource limitations of the reserves of virgin ore by converting from net consumption to net production of fissile.* By the time of full switchover to breeding, the uranium contained in the 15 million tonnes of Redbook ore reserves is residing almost entirely in already mined and milled inventories of enrichment tails and recovered irradiated uranium. Thereafter these already-mined uranium reserves are sufficient to provide for about a millennium of society's energy needs.

3.6 Cost Sensitivity Studies

3.6.1 Ore Costs

The fuel cycle services cost index for the reference base case scenario (100% LWR-UOX once-through) rises from a year 2000 value of 4 mill/kwhr by a little over factor of two by the year 2100 to around 9 mill/kwhr – in both the higher demand growth scenario, B, and in the lower demand growth scenario, C2. This escalation in cost index is dominated by rising ore costs which are assumed to rise from 20 $/kgU in year 2000 to 130$/kgU near mid century when the Redbook Known + Speculative reserves of ~15 million tonnes are exhausted and to be capped by century's end at 200 $/kgU the assumed upper bound price reached when twice the Redbook identified resource of U have been exhausted. This is seen in Figures A3.6 and in Figure A3.6 from Attachment 3 for growth scenarios Case B and Case C-2.

Not surprisingly, since ore cost dominates the change in cost index for the base case scenario, various assumptions on ore cost strongly affect the results. The effects of assumed 30% increase or decrease in cost of ore (at every level of cumulative withdrawal) computed for the base case for demand scenario B shows the cost index at century's end for fuel cycle services vary from 10.5 to 8 mills/kwhr around the base cost value of 9 mil/kw hr. A dilated cost schedule wherein the cumulative ore available is assumed to be doubled at each cost of recovery level shows only a moderate impact on cost index – it rises more slowly but reaches 9 mil/kwhr late in the century.

3.6.2 Repository Costs

Although the use of MOX partial recycle dramatically reduces the mass of spent fuel destined for the repository (380,000 tonnes HM by year 2100 vs 4,200,000 tonnes HM for the LWR-UOX once-through base case – compare Figures 14 and 5), *the costing assumptions of 1 mill/kwhr independent of number of repositories needed will yield no fuel cycle cost payoff for the waste management features of recycle.*

The same result is seen for the 80% LWR-UOX/20% Fast Burner Reactor full TRU recycle symbiotic park (compare Figures 3.20 and 3.5). Recycle costs peak early in the century while processing the legacy spent UOX fuel, then later fall to ~1/2 of the rising ore costs in Case B or nearly equal for Case C-2; *even though repository waste mass is reduced by three orders of magnitude, the 1 mill/kwhr repository costing schedule provides no cost index payoff for this reduction.*

No penalty was taken in any of the scenarios for the dramatic rise in required repository space needed for the once-through cycles. To assess what effect rising repository costs might have, a case was run raising the repository charge of 1 mill/kwhr by 0.1 mill/kwhr for each additional 70,000 metric tonnes of fuel to be disposed. For the LWR-UOX once through base case, by 2060, the repository charge is accelerating – though the fuel cycle cost index is still dominated by the ore cost. The cost index reaches 11 mill/kwhr by century's end – up from 9 mill/kwhr. It appears that the ore and repository components of

cost would become equal by 2130 at the assumed 0.1 mill/kwhr increase per 70,0000 tonnes – or by 2070 at an assumed 0.4 mill/kwhr increase per 70,000 tonnes.

3.6.3 MOX Recycle Costs

Returning to the base case set of cost assumptions, when LWR-MOX mono recycle is added at 10% market share, the initial addition to cost of recycle (of the 250,000 tonnes of legacy UOX spent fuel) adds dramatically to overall costs in the park – see Figure 3.15. However, after recycle of the legacy backlog is completed and the park settles into its asymptotic 90% UOX/10% MOX configuration, the ore cost again dominates (by factors of 3) and the overall fuel cycle costs remain at or below 10 mills/kwhr for both B and C2 scenarios.

3.6.4 Fast Reactor Advanced Recycle Costs

The fast reactor advanced (full TRU) recycle unit process costs are shown in Table A2.3.4 of Attachment 2 with their range of uncertainty. The nominal unit process costs for both advanced recycle and for remote refabrication are much higher than for the well established MOX recycle glove box fabrication values and moreover the uncertainty band is very large owing to the current state of development. All the scenarios used the nominal value from Table A2.3.4 of Attachment 2, and as seen in Figures 3.25 vs. Fig. 3.32, depending on the market fraction of fast recycle in the park, the recycle costs range from 4 to 15 mills/kwhr by year 2100. To assess the effect that a learning curve might have on the cost of fast reactor advanced recycle based cycles, the fast reactor 100% market penetration scenario – (Figure 3.27, 3.28, 3.29) was rerun using the lower end of the uncertainty band in the advanced recycle unit costs; i.e., recycle of the TRU fuel dropped to 1000 $/kgHM from the nominal value of 2000 and TRU fuel remote fabrication dropped to 1400 $/kgHM from the nominal value of 2600. The result (Note Figure 3.29) was to reduce the year 2100 fuel cycle cost index from 25 mills/kwhr to 16 – a dramatic reduction were it possible to reduce advanced recycle/refabrication costs by about 45% as experience is accumulated.

3.7 Scenario Results for Thorium Fuel Cycles

Scenarios have been evaluated for two Thorium-based Gen-IV concepts which lie at the two extremes of the range indicated in Figure 3.2. The Radkowsky once-through mixed uranium/thorium cycle is evaluated as a near term option for use in existing PWR power plants. Then a molten salt, liquid fuel, integrated Thorium recycle concept which starts up on enriched uranium, then consumes the transuranics contained in legacy and future LWR-UOX once-through fuel discharges, and finally achieves 100% market share on a closed Thorium/U233 fuel cycle is evaluated. Both scenarios were based on the WEC/IIASA Case B global nuclear energy demand growth rate.

3.7.1. The Radkowsky Once-Through Thorium/Uranium Fuel Cycle (RTF)

This scenario calls for starting to adopt the Radkowsky thorium fuel cycle (RTF) in the year 2020 and to fully adopt it by 2040. To implement the scenario, it was assumed that starting in 2020, 20% of the new plants (replacement and growth) will be RTF reactors, 40% share by 2025, 60% by 2030, 80% by 2035, and 100% by 2040. The mass flow data for the reactor are based on data in references. The RTF replacements for standard PWR assemblies consist of two sub-regions within the pin cluster; internal seed region and outer blanket region. The seed region contains U-Zr (~20% enriched U) metallic alloy pins and the blanket region contains ThO_2UO_2 (~10%UO_2 at ~20% enrichment) pins. The breeding ratio for the reactor is ~1, and the blanket part of the subassembly is removed from the reactor after ~10 years, while the seed part remains for 3 years. This leads to the higher blanket burnup shown in the reactor attributes table (Table A2.5-8 in Attachment 2) while the seed fuel reaches a burnup similar to LWR burnup. U233 is produced in the blanket fuel and the unburned part of it ends up in the repository as part of the spent fuel (SF).

Detailed cost data for the process steps of this cycle are not available, and it was assumed that the time lags and cost data for the RTF type reactor are the same as those for an LWR. For example, the reactor construction time, licensing time, SF storage time, and SF cooling time are the same as those for the LWRs. Also, cost parameters such as the fabrication cost, the SF storage cost, and the disposal cost (including the SF shipping cost) are assumed to be the same as the corresponding LWR costs. The virgin thorium cost is the only additional cost considered in this scenario. The cost is assumed to be about 95$/kg Th, based on average of $82.5/kg for 99.9% purity and $107.25 for 99.99% purity reported in Chapter 1, Section 1.3.2 The mining cost is assumed to be constant here, although the reference suggests that the mining cost can actually decrease if levels of consumption increase.

The DYMOND code tracked the seed and blanket fuel separately since its mass flows are independent from each other. The results for this scenario are shown in Figures 3.33 to 3.36. Figure 3.33 shows the market penetration of the RTF type reactors compared to the LWRs. By the year 2100, almost all power generation will be from RTF's after reaching a 50/50 market share in 2055. Figure 3.34 shows the depletion of Known and Speculative uranium resources by the year 2050. This is similar to the once-through LWR base case scenario because of the large LWRs capacity contribution up to that point in time (in the year

2050, about 60% of the total capacity is generated by LWRs). By the year 2100, the new U discovery needed and the generated enrichment tails are about the same as the corresponding once-through LWR scenario values. It might have been expected that a reduction in the required U resources would be achieved as the RTFs are introduced into the market (about 400 kgU is needed per year for a RTF compared to 23,000 kg needed for a LWR). However, the high enrichments associated with the U part of the fuel (~20% in RTF seed compared to ~4% in the LWR fuel) have offset the reduction in the U requirements for the reactors.

As shown in Figure 3.34, discharge fuel mass from the metallic seed fuel and the ThO_2UO_2 blanket fuel increase gradually with the introduction of the RTFs. However, by the year 2100, the contributions of those two types of fuel to the total are small compared to the LWR UO_2 spent fuel share. In general, the total spent fuel from the RTF scenario is smaller than from the once-through LWR. Not much reduction will be achieved by the year 2050, but about 30% reduction will be achieved by the year 2100. This is a result of a much smaller discharge from RTF (about 8,000 kg/y which includes the Th in the blanket resulting from the internal breeding and long residence time of the blanket pins) as compared to the LWRs annual discharge (about 23,000 kg).

Figure 3.33 shows the total enrichment and fuel fabrication capabilities associated with the scenario. The total enrichment rate is higher than the once-through LWR scenario which it exceeds by almost 30% by the year 2100. On the other hand, the fabrication rate by the year 2100 has decreased by about 50%.

The cost index estimates for this scenario are shown in Figure 3.35. The behavior of the individual cost parameters and the total fuel cycle cost per TW-h generated are similar to the once-through LWR scenario. At the end of simulation (year 2100) the total RTF scenario cost per terawatt hour is about the same as the base case scenario cost. The amounts of spent fuel generated by the RTF scenario reduce the storage and shipping costs; however the contributions of these costs to the total cost are very small.

Figure 3.35 shows the normalized indexes for this scenario. The cost and ore indexes do not change. The mass of fuel destined for the repository shows a 30% reduction by the year 2100. The Pu index is about 0.6 showing a substantial reduction in the content of Pu in the spent fuel going to the repository. However, as shown in Figure 3.36 substantial amounts of ^{233}U are generated and sent to the repository. Although those amounts of ^{233}U are substantial, the ^{233}U destined for the repository is denatured by the ^{238}U blanket uranium. Finally, Figure 3.36 shows the amount of mined thorium ore.

3.7.2 The Molten Salt Fissile Self Sufficient Closed Thorium Fuel Cycle

In this scenario, the Molten Salt Reactor (MSR) is introduced in 2030 first on enriched uranium, then on TRU from legacy and newly deployed LWR spent fuel -- until finally it is all consumed (in the 22nd century). Starting in 2030, 5% of the new plants (replacement and growth) are MSRs. In 2031, 10% of the new plants are MSRs. This linear market-fraction penetration extends until all reactors are MSRs. The MSR is assumed to have 44% energy conversion efficiency and a capacity factor of 90%. All MSRs starting up contain an initial inventory of 127,460 kg of thorium per 1 GW(e) capacity. The fissile material for startup of new MSRs from 2030 to 2050 is 19.9 wt % enriched uranium. The startup inventory is 3115 kg of ^{235}U in 15,653 kg of total U for 1 GW(e) capacity. After 2050, all new 1-GW(e)-capacity MSR plants at startup are loaded with 127,460 kg of thorium, plus 3115 kg of recycle LWR plutonium plus associated higher actinides (i.e., Pu and minor actinides or Pu+MA) plus 15,653 kg of depleted U. In a MSR, most of the plutonium is rapidly burnt out and replaced with bred-in ^{233}U. The depleted uranium is added with the startup plutonium to denature the ^{233}U as it grows in. There is no further enriched uranium requirements for MSRs once startup on plutonium is initiated in 2050.

Within a few years of startup, the fuel composition in the salt is essentially independent of what fissile material was used to start the reactor. For each GW(e)-year of electricity generated, the reactors require an added feed of 801 kg of thorium and 155 kg of depleted uranium, independent of what the startup fissile material is (enriched uranium or plutonium).

At the end of plant life, each retired MSR is replaced with a new MSR that uses the fuel load from the decommissioned MSR. MSRs used to replace retiring MSRs do not require added enriched uranium or recycle plutonium; the fuel salt is relocated into the replacement reactor to (1) continue use of the fuel, thorium, and the expensive ^{7}Li and (2) avoid disposal of the old salt (containing beryllium).

Subsequent to 2050 the fissile working inventory needed to build additional MSR's to meet growing demand comes from the TRU from the LWR discharged fuel inventory – until that inventory is consumed. (This doesn't happen until the twenty second century.) At that point a new external source of fissile would be required.) In performing the scenario simulation, a number of assumptions were made as follows.

- LWR fuel reprocessing is assumed to start 2045 and the reprocessing plants are built at a rate which will meet the demand of MSRs for Pu and higher actinides beyond the year 2050.
- The composition of Pu+MA startup fuel is assumed to contain proportions of Pu to MA that are found in the LWR spent fuel.
- APUREX process is used to reprocess the LWR SF.
- The composition of the spent fuel (SF) at the end of the MSR reactor lifetime is assumed to be the same as the composition used at the reactor startup except for the fissile material content. The fissile material used to startup the reactor (^{235}U or Pu+MA) is replaced with ^{233}U as the reactor reaches equilibrium.

- The processing of the molten salt during operations results in a processing trace loss of about 526 grams of Pu and MA, which will go to the high level waste (HLW) stream.
- The fission products (FP) are removed from the reactor during operations at the same rate that Th and depleted uranium (DU) are fed to the reactor. That is, the rate of FP removal is 801 kg + 155 kg – 0.526 kg = 955.4 kg/year.
- A cost of 95$/kg Th is assumed here. This cost is based on average of $82.5/kg for 99.9% purity and $107.25 for 99.99% purity reported in chapter 1. The Th conversion cost is assumed to be the same as the U conversion cost. Other costs, such as the storage cost, and HLW storage and shipping cost are assumed to be the same as the costs used in all scenarios.

The results for this scenario are shown in Figures 3.37, 3.38 and 3.39. Figure 3.37 shows the contribution of the MSR to the total electricity production capacity. The MSRs reach a 50-50 market share by the year 2067 and achieve about 95% market share by the year 2100.

Figure 3.38 shows the Known and Speculative uranium resources by the year 2050. This is similar to the once-through LWR scenario, since the LWRs remain dominant in power production until this point in time. However, by the year 2100 a large reduction in the required new finds of uranium resources has been achieved (about 10,000 kt are needed compared to about 42,000 kt of new finds for the once-through LWR scenario).

No LWR SF is sent to the repository; it remains in storage until it is sent to the reprocessing plants to generate initial working inventory for new MSR plants. The scenario shows a continuing increase in the stored LWR spent fuel, but at a much slower rate than that associated with the once-through LWR scenario. By the year 2065, the amount of SF in repository and storage for the base case once-through LWR scenario is about 1575 kt whereas LWR SF stored awaiting processing is 1145 kt for the MSR scenario. Beyond 2080, the amount of LWR SF associated with the MSR scenario starts to decrease as more SF is reprocessed to meet increased buildup of MSRs. By the year 2100, there are about 1000 kt of LWR SF remaining in storage compared to about 4000 kt of LWR SF in storage and repository in the case of once-through LWR scenario. A by-product of the reprocessing associated with the MSR scenario is an additional 1200 kt of burned uranium in interim storage.

Figure 37 shows a decrease in the rates of LWR fuel fabrication and of overall system uranium enrichment as the MSRs are introduced. The enrichment rate decreases since enriched uranium is used in the MSRs only between the years 2030 and 2050 to startup; it is not used as a feed for subsequent MSR reactor operations. Beyond 2050 no enriched uranium is needed to start the MSRs but is still needed for existing LWR refueling. By the year 2100 only about 25 kt SWU/yr enrichment rate is needed compared to a rate of about 750 kt SWU/yr in the base case once-through LWR scenario. The fabrication rate also decreases after MSR introduction, since fuel fabrication is unnecessary for the fluid fuel MSRs. Only fabrication of LWR fuel is needed, which decreases to a rate of about 5 kt HM/yr compared to about 130 kt HM/yr in the case of once-through LWR scenario. The figure also shows the reprocessing rates of the LWR SF. The reprocessing plants are built at

a rate that assures the presence of enough Pu supplies to meet the growing deployments of MSRs. The Pu that is separated and available for use in MSRs is shown in the figure; the reprocessing capacity constrained so as to keep the amounts of separated Pu in interim storage from being too excessive or being short of meeting the MSR needs.

Cost index estimates for the MSR scenario are shown in Figure 3.39. The mining and enrichment cost remains the major cost factor for this scenario as it was for the once-through LWR scenario. The amount of mined Th ore is much smaller than the amount of mined uranium ore. This is the case, even by the year 2100 where the majority of the electricity production is attributed to MSRs. This is a result of the low amounts of Th needed over the life of a fissile self-sufficient MSR fissioning all the thorium compared to the amounts of U needed over the life of a LWR which fissions less than 1% of the uranium – (about 175 tons of Th compared to about 1400 tons of 3% enriched U over a 60 years lifetime of a 1GWe MSR and 1.2 GWe LWR, respectively). Enrichment and conversion costs are usually minor compared to the mining costs. Fabrication and storage costs are associated with the operating LWRs, only since the MSR fuel simply circulates from reactor to fission product removal and back to reactor. Both costs decrease with time after the introduction of the MSRs.

Beyond 2050, after the introduction of the Pu based MSRs, the cost for Pu and MA extraction from LWR SF (based on APUREX process) becomes a major cost factor. This is especially after the year 2080, as it exceeds the mining costs. For this scenario, the disposal cost corresponds to the cost of temporary storing and transporting the HLW associated with the LWR SF reprocessing, and the repository cost. A cost of 1 mill/kWh is still charged for permanent storage of this HLW in the repository. The costs of temporary storage and transportation of the HLW are very small compared to the repository cost. Finally, a cost that is unique to the MSR is related to the salt processing in order to remove the FP and recycle the salt back into the reactor. It is considered a cost of the fuel to see its effect on the total fuel cycle cost. It is assumed that this cost is 1 mill/kWh. A recent study of the cost of the denatured MSR did not include the processing cost, since the reactor operations in this case do not require the FP removal from the salt. A 1970 estimate of this cost for the MSBR is about 0.3 mills/kWh, and adjusting it for the year 2000 cost, we assumed it to be 1 mills/kWh. In this case, as shown in Figure 3.39, the processing cost increases with time as more MSRs are introduced. Beyond the year 2090, this cost exceeds the costs attributed to the other parts of the fuel cycle. The total fuel cycle related costs for the scenario are shown in the figure. It is smaller than the cost associated with the base case once-through LWR scenario as discussed next in relation to the normalized indexes.

Figure 3.39 shows the normalized indexes, which compares the MSR scenario to the base case once-through LWR scenario. The cost of the MSR scenario is less than that of the once-through LWR, mainly as a result of decrease in the mining cost. The normalized cost index goes down gradually with the introduction of the MSRs until it reaches about 0.35 by the year 2100. Starting already in the year 2000, the cost index is slightly less than one since no LWR SF is shipped to the repository in this scenario, which eliminates the SF shipping cost. The SF will be used to fuel the MSRs, thus, the SF index is zero. The ore index also decreases with time after the MSR introduction because of the decreased demand for the U

ore. By the year 2100, the cumulative uranium ore withdrawals index will be reduced to about 0.4. The Pu and MA indexes increase above zero as the LWR SF processing and the MSRs operations start. This is a result of the losses during the LWR SF reprocessing operations and the losses during the recycling of the MSR salt. Finally, there are no noticeable changes in the indexes as the early built MSRs go off line (starting the year 2095) as a result of the assumption that core of a MSR at the end of life is used to fuel a new MSR. Ignoring the lag in time used to build the new reactor, this will be equivalent in its fuel cycle to a reactor that is running indefinitely.

3.7.3 Observations Regarding Thorium Fuel Cycles

The Radkowsky mixed uranium/thorium once-through cycle, like all the U once-through cycles discussed in Section 3.3 achieves performance on sustainability goals SU-1 and SU-2 which are little changed from the Base Case LWR-Once-Through cycle. The Redbook Known plus Speculative reserves are exhausted by mid century and the required new founds of U ore by year 2100 are not reduced – because the high enrichment required for the Radkowsky seed fuel pins offsets the benefit that some fissions are produced by ^{233}U bred from thorium. The spent fuel destined for the repository remains at the base case value of 1,000,000 tonnes at 2050 but is reduced from 4,200,000 to 2,900,000 tonnes by century's end.

The Molten Salt fissile self-sufficient closed thorium cycle displays all the benefits to sustainability goal SU-2 of full recycle which are achieved by closed uranium cycles – the mass flow from the entire energy park to the repository is reduced by a factor of a thousand.

As regards to sustainability goal SU-1, an interesting difference arises between the thermal-spectrum fissile self-sufficient MSR closed cycle and the fast spectrum fissile self-sufficient liquid-metal cooled closed cycle which was discussed in Section 3.5.2 (compare Figures 3.23, 3.24, 3.25 for LMR to Figures 3.37, 3.38, 3.39 for MSR). The total fissile mass contained in the LWR once-through spent fuel inventory (legacy plus ongoing additions) can supply the initial working inventories of very many more MSR's than it can for liquid-metal reactors – because the critical fissile mass of the MSR type of thermal spectrum system is much smaller than that of a fast spectrum system. (Compare the drawdown rates of LWR spent fuel inventories from Figures 3.24 and 3.38). Figure 3.24 shows fast reactor deployment exhausts the legacy inventory and becomes fissile constrained at mid century – with new starts relying on fresh discharged spent fuel from still-deployed LWR's. The MSR deployment based on TRU from LWR spent fuel on the other hand, (even granted that it starts 30 years later) is assured for many decades into the twenty second century – see Fig. 3.38. As a result, a conversion to breeding is not required in the MSR scenario throughout the entire 21st century, and new finds over and above the Redbook Known plus Speculative reserves are held to only 8,000,000 tonnes. This difference on the one hand delays a need for breeding (in excess of fissile self regeneration), but on the other hand it vastly increases the LWR interim storage requirements – both in mass and in longevity. More importantly, the MSR deployment delays but does not eliminate the longer term need for breeding – and it emplaces a thermal spectrum fuel cycle which (owing to small neutron excess inventory) cannot achieve the short doubling time which will be eventually necessary to manufacture

fissile fuel for the entire energy park in breeding systems of doubling time shorter than that of the energy demand growth rate of ~20 years. This suggests consideration of a symbiosis of fissile self sufficient MSR's and fast spectrum breeders starting some decades before century's end. This case has not yet been simulated.

3.8 Summary of Lessons Learned From the "Cornerstone" Scenarios

3.8.1 Lesson Learned; Achieving Sustainability Goals

The scenarios modeled here for the WEC/IIASA Case B growth scenario (2000 GWe by 2050 and 6000 GWe by 2100) are idealized in relying only on physical mass-flow constraints and in prescribing (as inputs) the Gen-IV concept introduction dates and target market shares of new deployments. They serve as mass balance constrained physically-achievable "cornerstones" to delineate the outer boundaries of achievable performance for *market driven* deployments.

The scenarios indicate that already deployed and new additions of once-through thermal systems will in all cases dominate market share for decades – until around mid century. However, their waste arisings become quite significant already by 2030 and the cumulative forward fuel requirements of once-through thermal spectrum systems deployed up to around 2030 already sum to the ore reserves in the Redbook Known plus Speculative reserves. Neither higher burnup achieved via higher enrichment in once-through designs nor MOX or DUPIC mono recycle is effective in producing significant delays in exhaustion of currently identified (Redbook) ore reserves – they last until about mid century. The slower growth rate of the WEI/IIASA case C2 (1200 GWe by 2050 and 1800 GWe by 2100) extends the Redbook ore reserves by ten years – to 2060.

MOX mono recycle can significantly reduce the waste arising *mass* destined for the repository – because the uranium which comprises 96 w/o of spent UOX fuel is set aside rather than sent to the repository. DUPIC mono recycle extends the energy derived from the fuel but does not reduce waste mass as much by separating and setting aside the uranium.

The benefits of waste mass reductions per se to the repository capacity are likely to be small in and of themselves because waste packing density in the repository is controlled by heat loading. Heat loading of waste is dominated for the first 100 years by fission products (principally Cs and Sr); thereafter heat loading is dominated by transuranics (primarily minor actinides). Repository capacity extension benefits of recycle therefore will rely on the development of customized waste forms and integrated heat management strategies for the fission product and TRU wastes destined for the repository.

The benefits of mass reductions per se to repository long term toxicity loading are also likely to be small in and of themselves. Long term toxicity is attributable to an almost insignificant mass fraction of the spent fuel – specifically the ~0.15 w/o of minor actinides; the ~1.5 w/o of plutonium; and a few specific long lived fission products. Even if plutonium is removed from the waste and fissioned, less than a factor of ten reduction in toxicity is achieved. It is essential to remove and fission the minor actinides as well – and if so, a factor of several hundred reduction is long term toxicity is attainable (see Fig. 3.22).

Full recycle and fast spectrum systems, if introduced to perform a waste management function in the 2020 to 2030 time frame, can stem the flow to the repository of all but fission products and trace losses of actinides from recycle steps – for the entire energy park

including the once-through thermal systems. The deployment of recycle/fast spectrum "burner" plants at ~20% of the energy park is sufficient to manage (hold in working inventory and burn) the spent fuel from the entire park such that only fission products and trace losses of actinides go to the waste. Moreover, they quickly draw down the inventories of legacy spent fuel from prior once-through operations as well – to use the material for initial working inventories of new deployments of the fast spectrum power plants.

With a symbiotic thermal/fast recycle energy park, the scenarios show that:

i. The waste arisings from the nuclear energy park as a whole can be limited to fission products only – (plus *trace losses* of actinides from recycle operations).
ii. The flexibility of fast spectrum systems using recycle fuel cycles is such that the fissile requirements versus fissile availability in the overall nuclear energy park can be kept in balance through timely adjustments in two degrees of freedom – the fractions of the park occupied by thermal and by fast systems, and the breeding ratio and doubling (or halving time) of the fast systems in the park – both of which can be adjusted. (Market forces by themselves may not lead to timely adjustments; governmental actions in the same vein as laws, central banks, tax incentives, etc. may be indicated if mass flows move out of balance.)
iii. When part of the park is already fast burner recycle systems, a time delay of only 1 or 2 years for switchover from full reliance on virgin ore to partial reliance on breeding can be made when economic or social conditions necessitate a change (i.e., no time delays to deploy altogether new reactor types or fuel cycle infrastructure are necessary – just a change in the annual reload core configuration of the fast systems.)
iv. Symbiotic energy parks which feedback excess fissile supply from full recycle fast breeders to thermal reactors operating on once-through cycles can achieve fully sustainable (Goals SU-1 and SU-2) global energy supplies while each filling specialized market niches – those suited to thermal reactors and those suited to fast reactors.
v. The switchover date from waste management to fissile production is delayed and/or the ratio of fast to thermal systems in the energy park is reduced when the thermal systems are of high conversion ratio and small fissile working inventory. (The extreme example is the Molten Salt Reactor closed thorium cycle discussed in Section 3.7.2.)
vi. When fissile manufacture from the fertile ^{238}U and ^{232}Th reserves (~4,000 years energy potential in the resource base) is required to sustain a growing nuclear energy park; the fast spectrum systems must be capable of short doubling times to support the entire energy park's (including thermal systems) fueling needs. This requires all three of the following features: fast spectrum, high power density, and short out of reactor recycle time lag and resulting fuel cycle pipeline inventory.
vii. The component of cost of energy supply which is due to fuel cycle services is small in the first place (~20%), and doesn't change dramatically anyway (a factor of about 2 by year 2100 for the base case LWR-UOX once-through cycle). Using current estimates of costs for TRU recycle and remote fabrication

produces a further factor of 1.5 to 2; *sustainable growing Gen-IV energy park fuel cycles would appear to have affordable fuel cycle service costs, and*

viii. The stored inventories of enrichment tails and recovered irradiated uranium are already more than sufficient for many centuries of energy demand growth in all energy sectors, not just electricity exclusively.

The physically constrained scenarios show that –were it necessary to remain within the Redbook Known plus Speculative ore reserves of ~15 million tonnes of U recoverable at <130 $/kgU, and were the demand growth in the range of the WEC/IIASA Case B growth (factor of 6 by 2050 and factor of 18 growth by 2100) – then new capacity additions subsequent to 2025 should favor fast spectrum systems deployed at the rate achievable based in fissile availability – until they reach the market share required to achieve a fissile self sufficient energy park. For the slower WEC/IIASA Case C2 growth rate (factor of 3 by 2050 and factor of 6 by 2100) a delay until near mid century would be acceptable, or the mix of thermal and fast deployments could include a larger share of thermal systems. Advanced fissile self sufficient closed thermal spectrum fuel cycles such as the Molten Salt Reactor operating on the liquid fuel thorium fuel cycle could sustain the Case B growth rate for many tens of decades fueled by the TRU contained in LWR once-through spent fuel.

3.8.2 Indicated Fuel Cycle R&D

R*eal deployment decisions will be market driven* and the planning horizon of the commercial decision making is shorter than the 25 to 40 year forward planning time frame for uranium ore resource management indicated in the scenarios. *What matters in market driven deployment decisions is the near term cost of energy production relative to the market price – not the 25-40 year hence future cost of fuel.* (Nor the longer term cost of waste management if, as in the US, a price cap has been emplaced by government). Thus, with respect to the Gen-IV sustainability goals controlled by the fuel cycle, these facts highlight the need for R&D in the following areas:

i. Cost effective recycle/refabrication technologies having low trace losses of actinides *which can become economically competitive early in the period of escalation of ore price*
ii. Recycle-based fuel forms (containing commixed TRU compositions) for thermal as well as fast reactor concepts and recycle-based waste forms and integrated decay heat management strategies for extending capacity of repositories and limiting the rate of future repository deployment even as nuclear capacity grows.
iii. *Cost effective fast spectrum systems* whose initial introduction into energy park growth will be to serve the function of back end waste management for the entire park – but whose prior deployment will, in the future, jump-start the transition to fissile self sustainability of the entire energy park.
iv. Cost effective high conversion thermal spectrum systems which benefit from small fissile working inventory but attain high energy output from ore feedstock as a result of high conversion ratio (perhaps requiring TRU recycle or perhaps in symbiosis with recycle fast spectrum systems.)

v. R&D on cost effective prospecting and mining technologies for finding and exploiting additional ore in an environmentally friendly fashion – to provide for robustness in the environmentally – responsible, energy security arena.

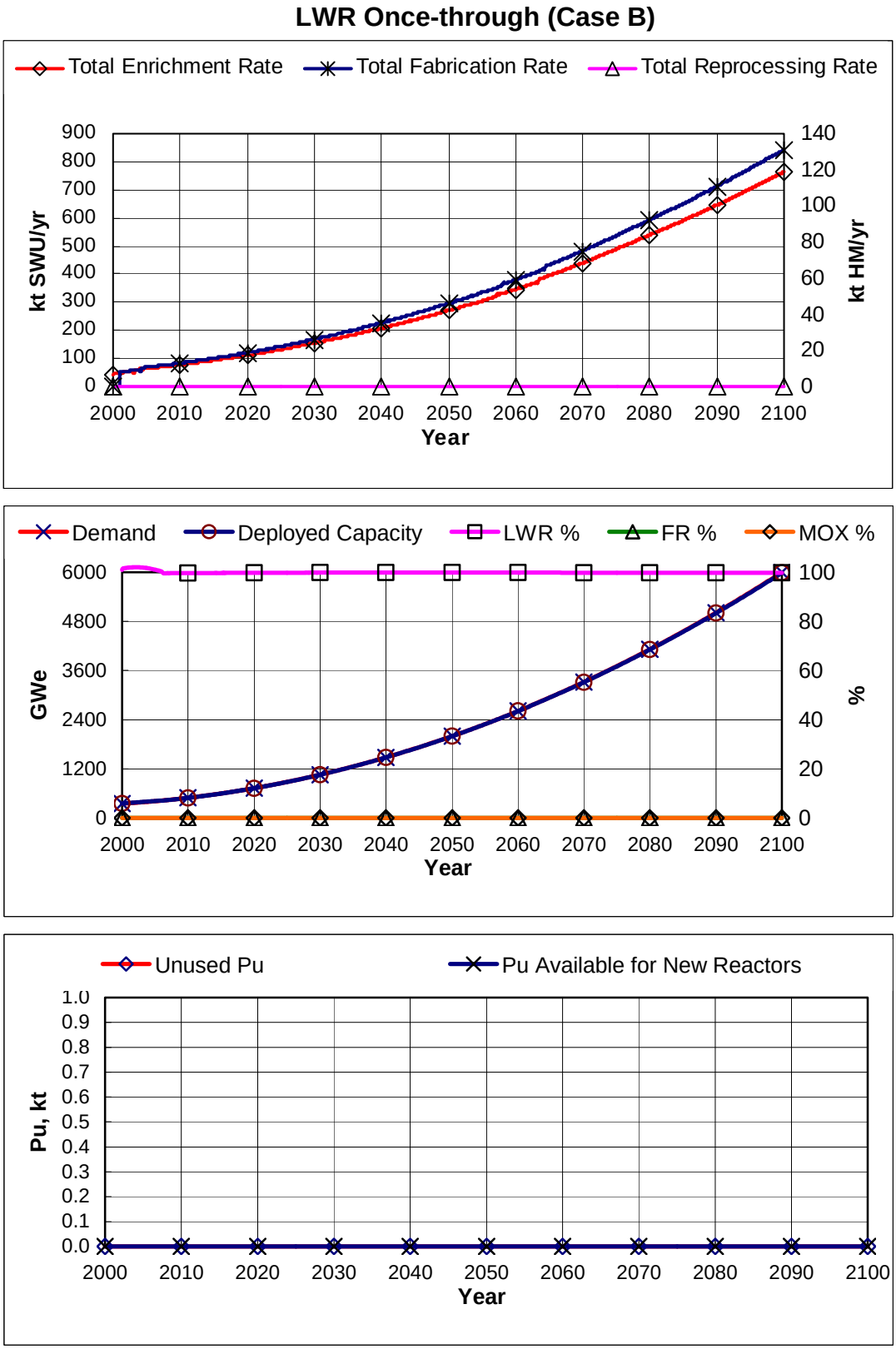

LWR Once-through (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.4

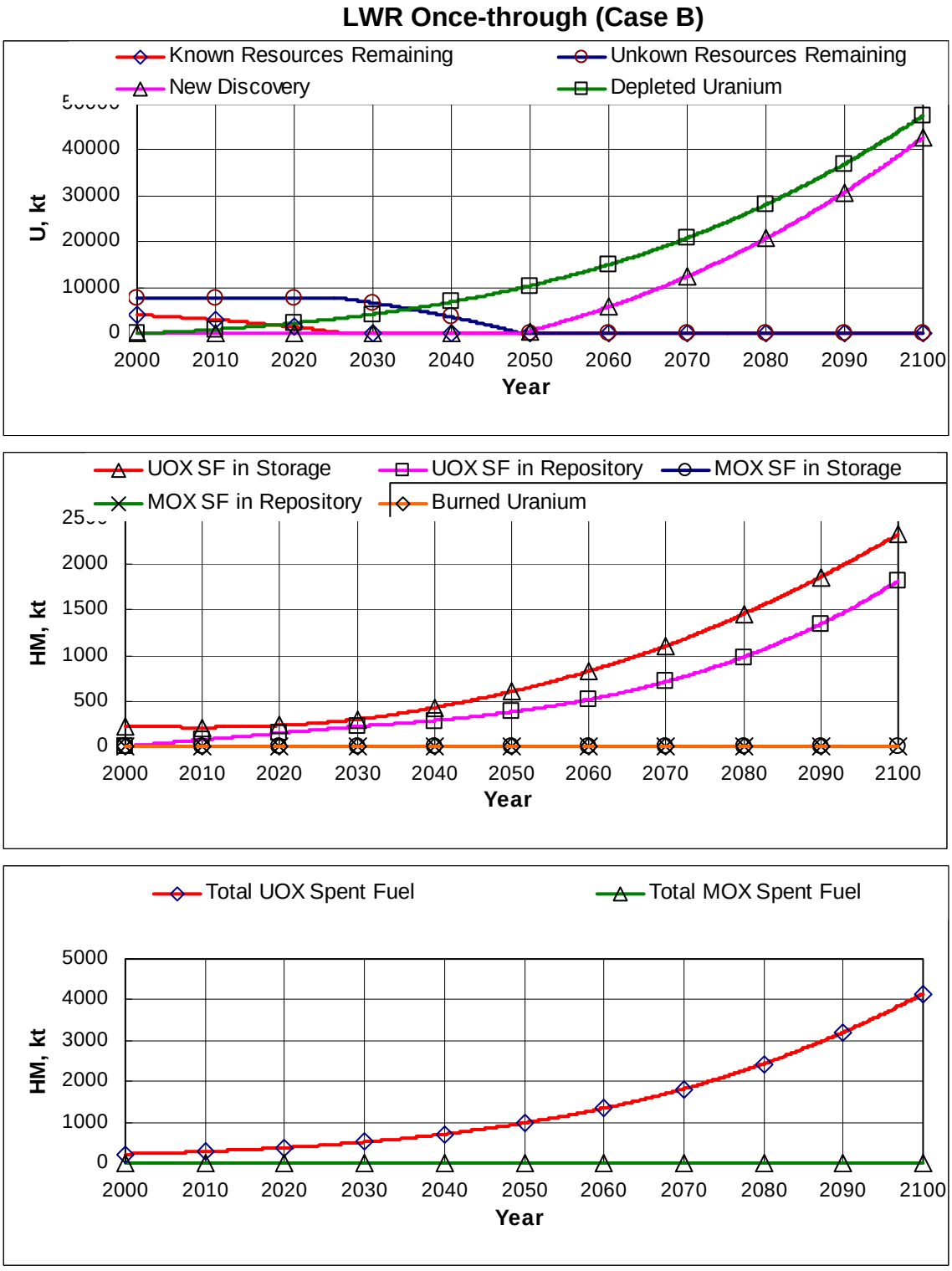

LWR Once-through (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
40000
30000
20000
10000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
2000
1500
1000
500
0
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
5000
4000
3000
2000
1000
0
Year

Fig. 3.5

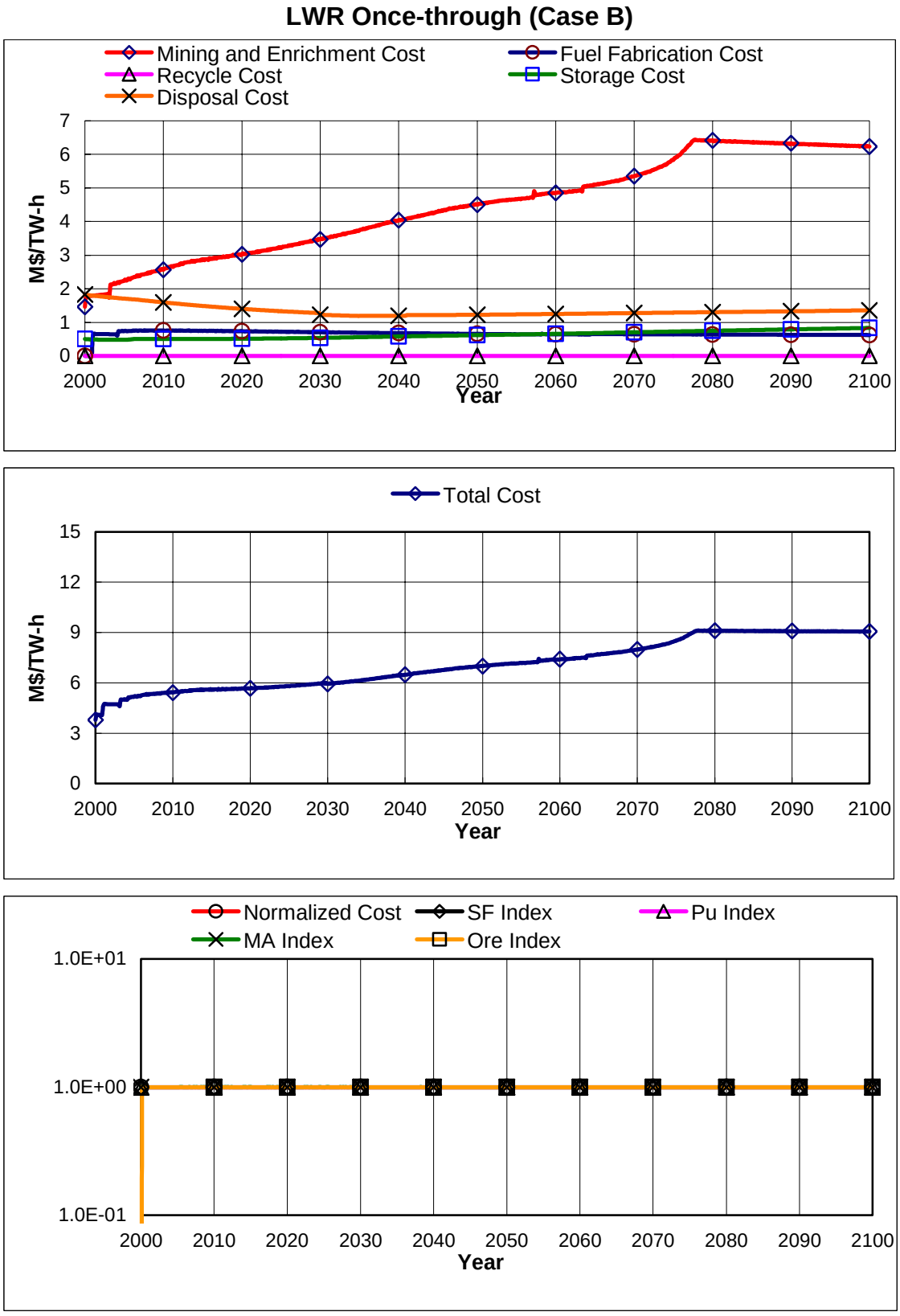

LWR Once-through (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
Year

Fig. 3.6

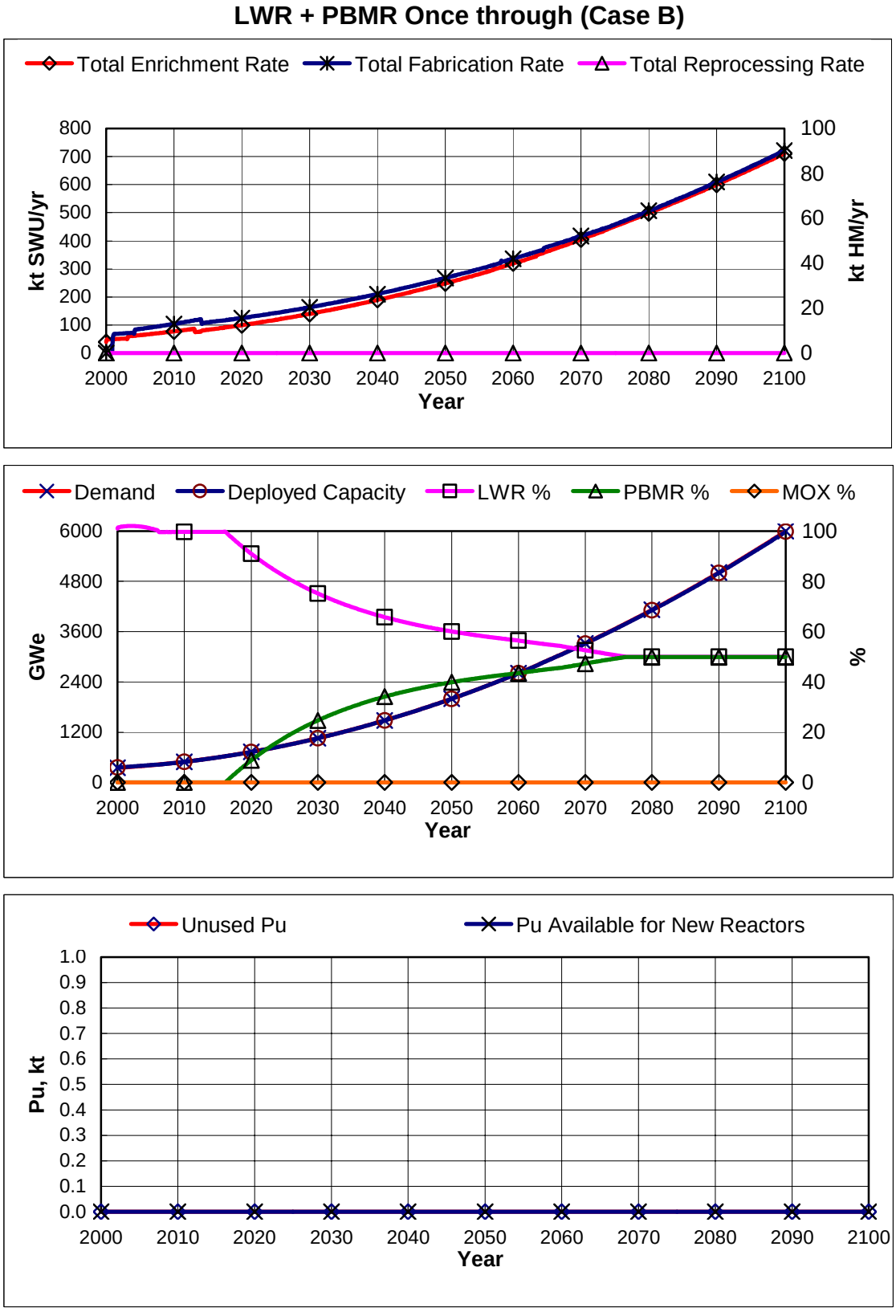

LWR + PBMR Once through (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
PBMR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.7

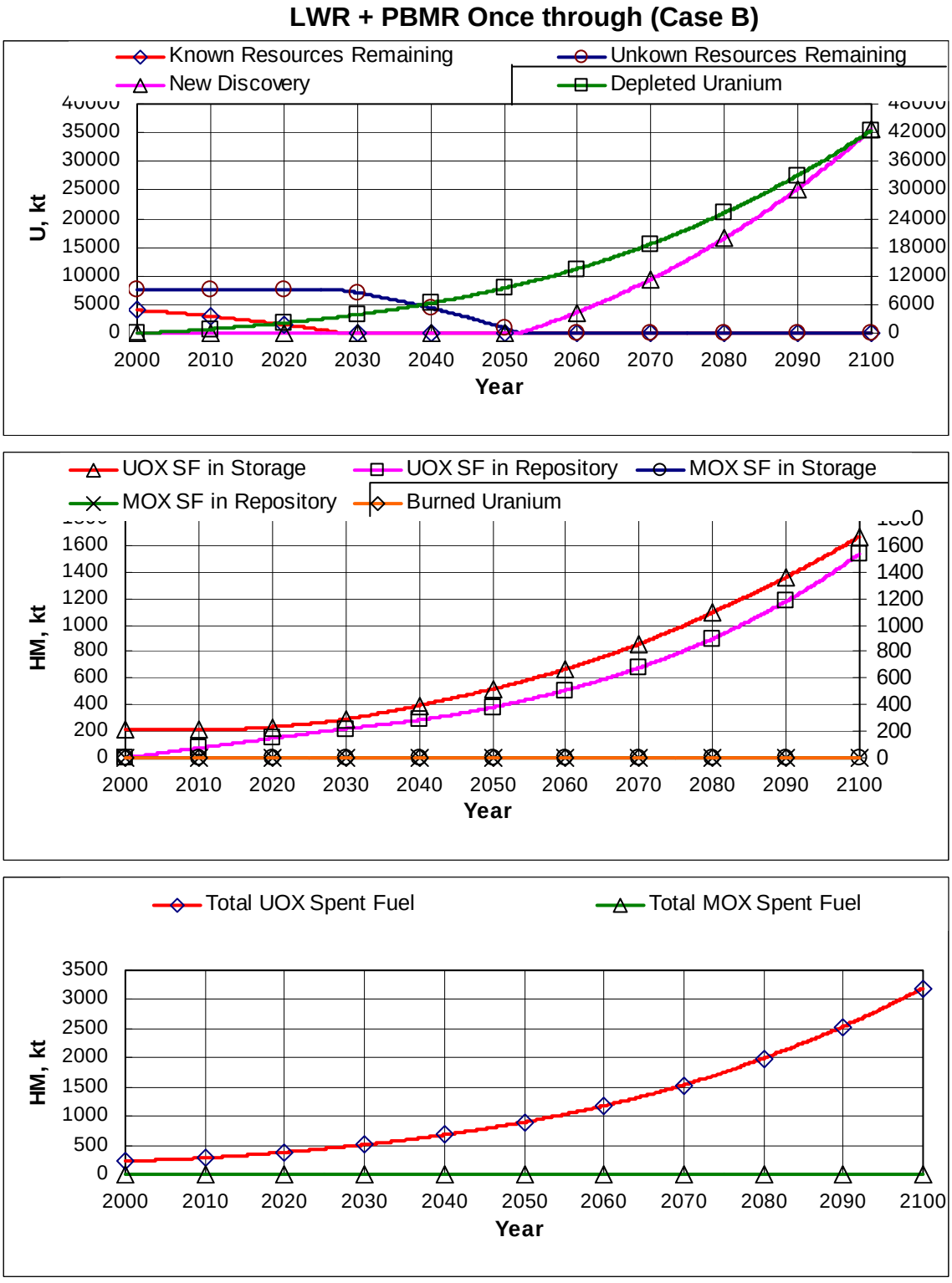

LWR + PBMR Once through (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.8

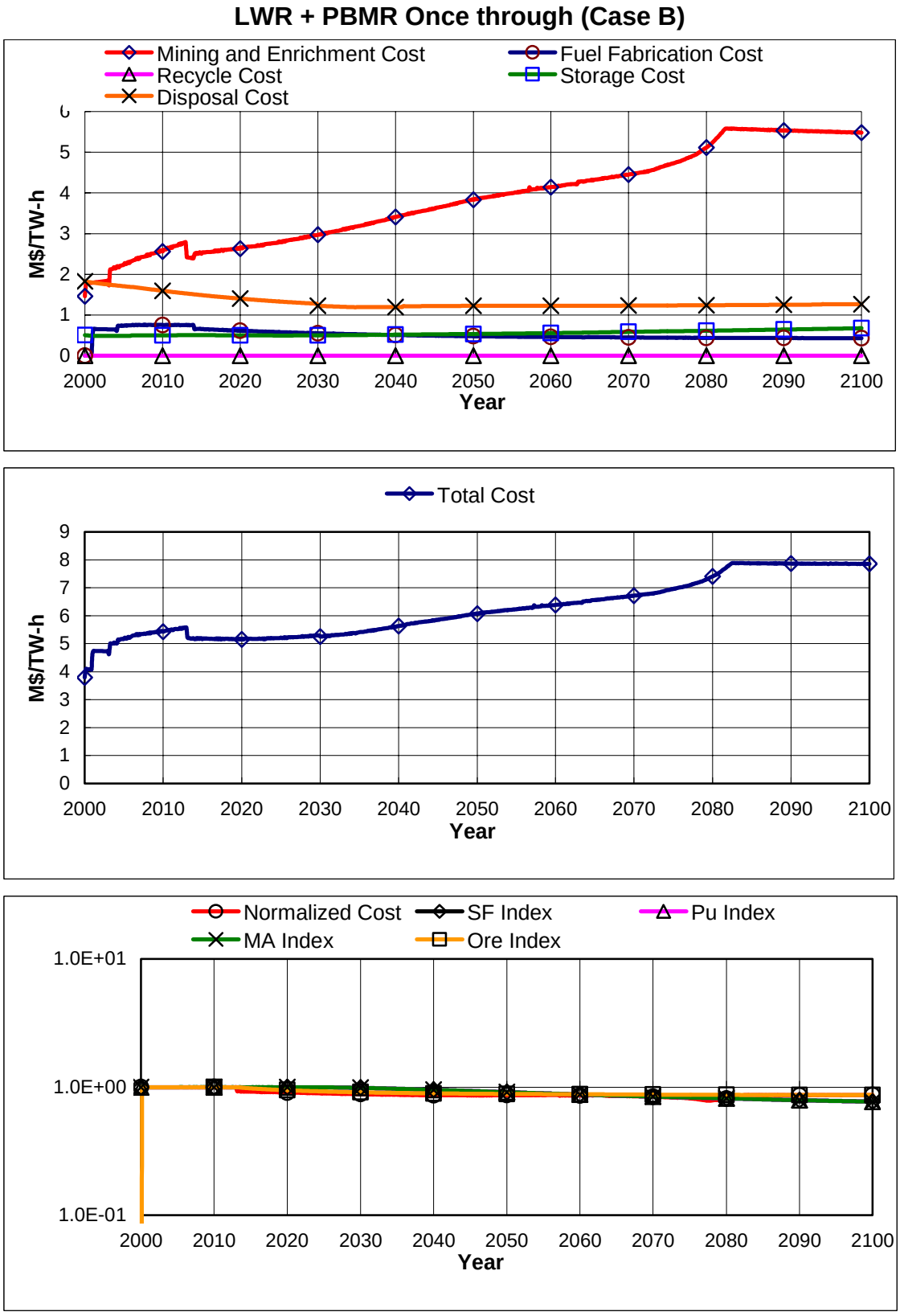

LWR + PBMR Once through (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
0
1
2
3
4
5
6
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total Cost
M$/TW-h
0
1
2
3
4
5
6
7
8
9
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Fig. 3.9

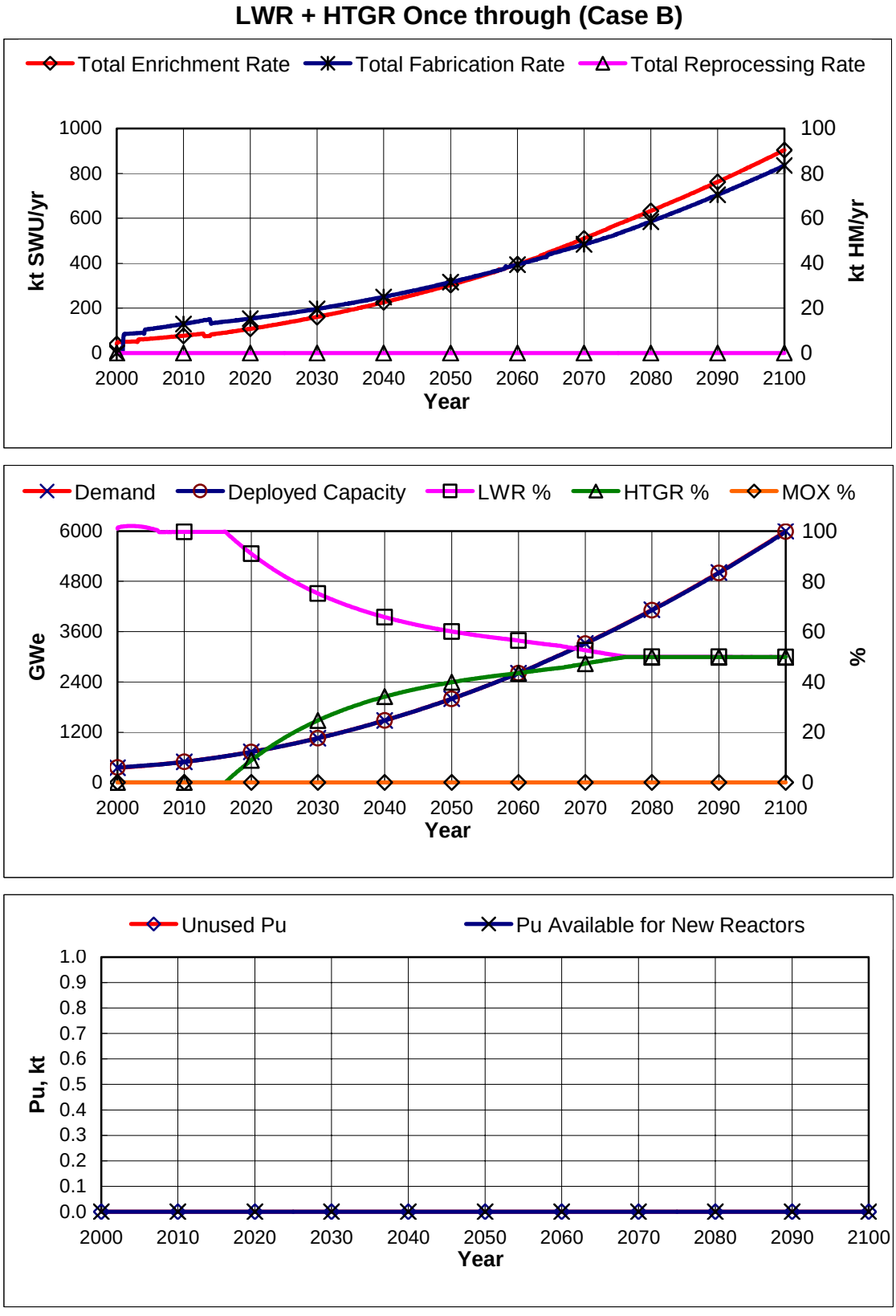

LWR + HTGR Once through (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
HTGR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.10

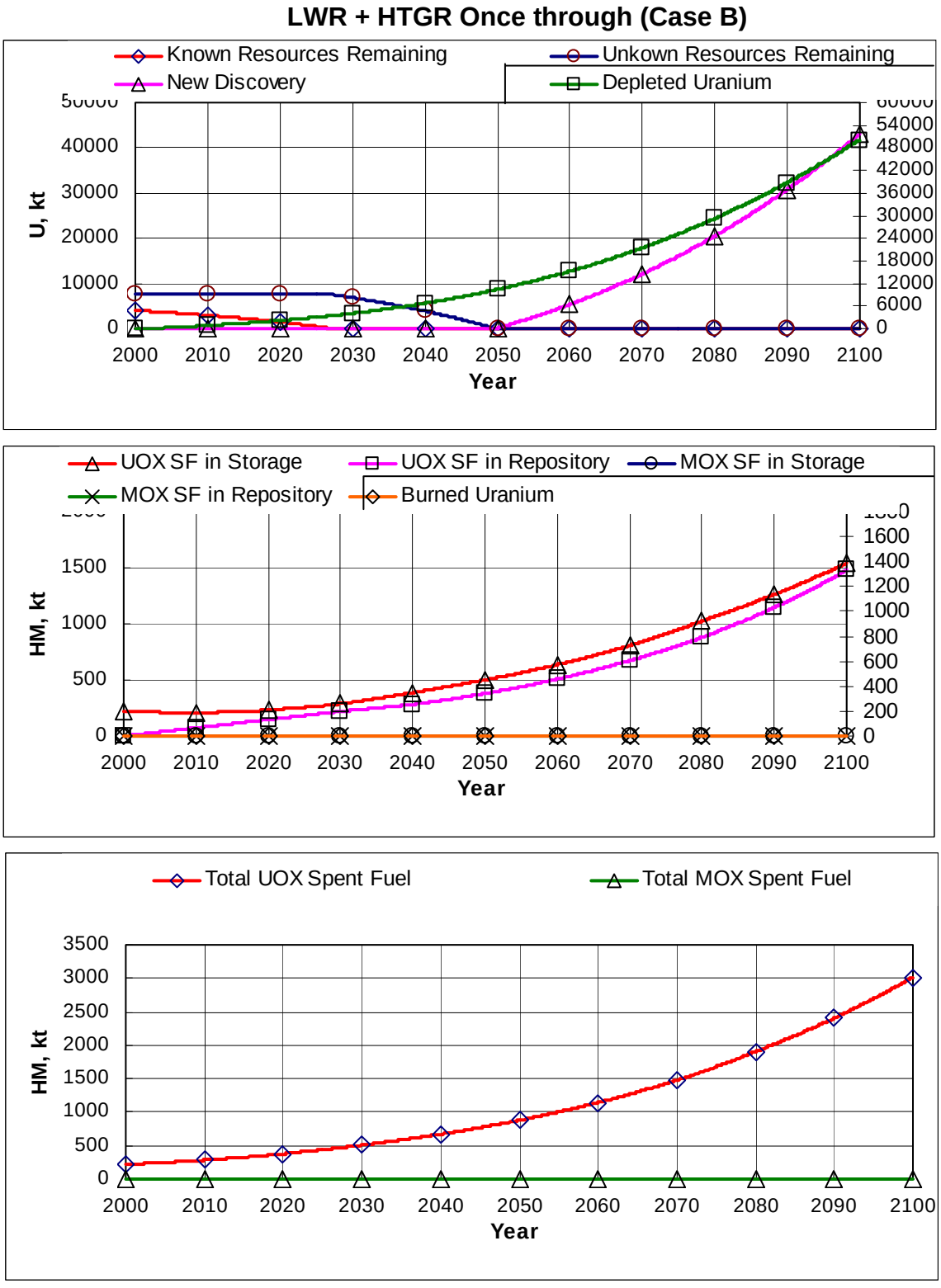

LWR + HTGR Once through (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.11

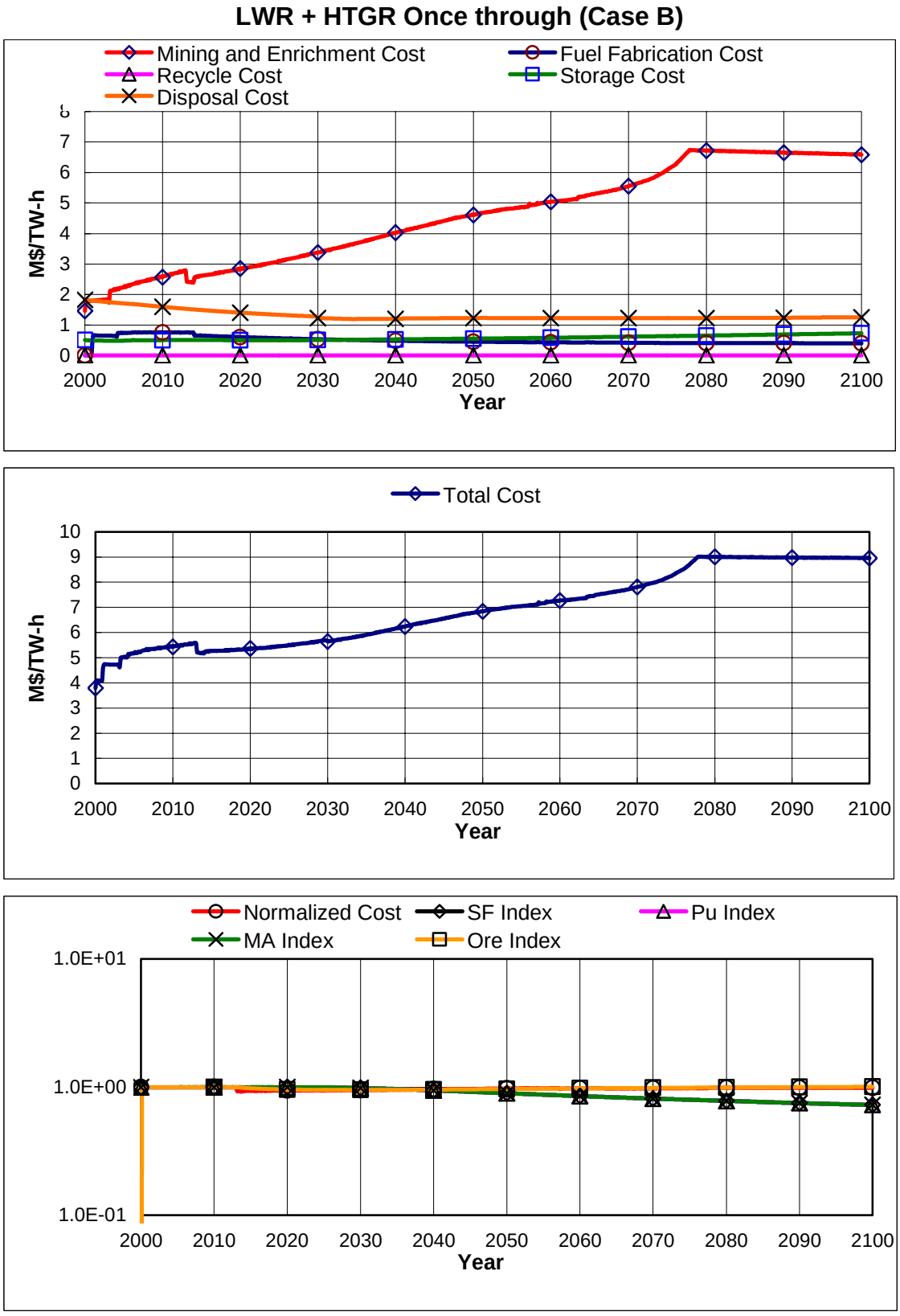

LWR + HTGR Once through (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
Year

Fig. 3.12

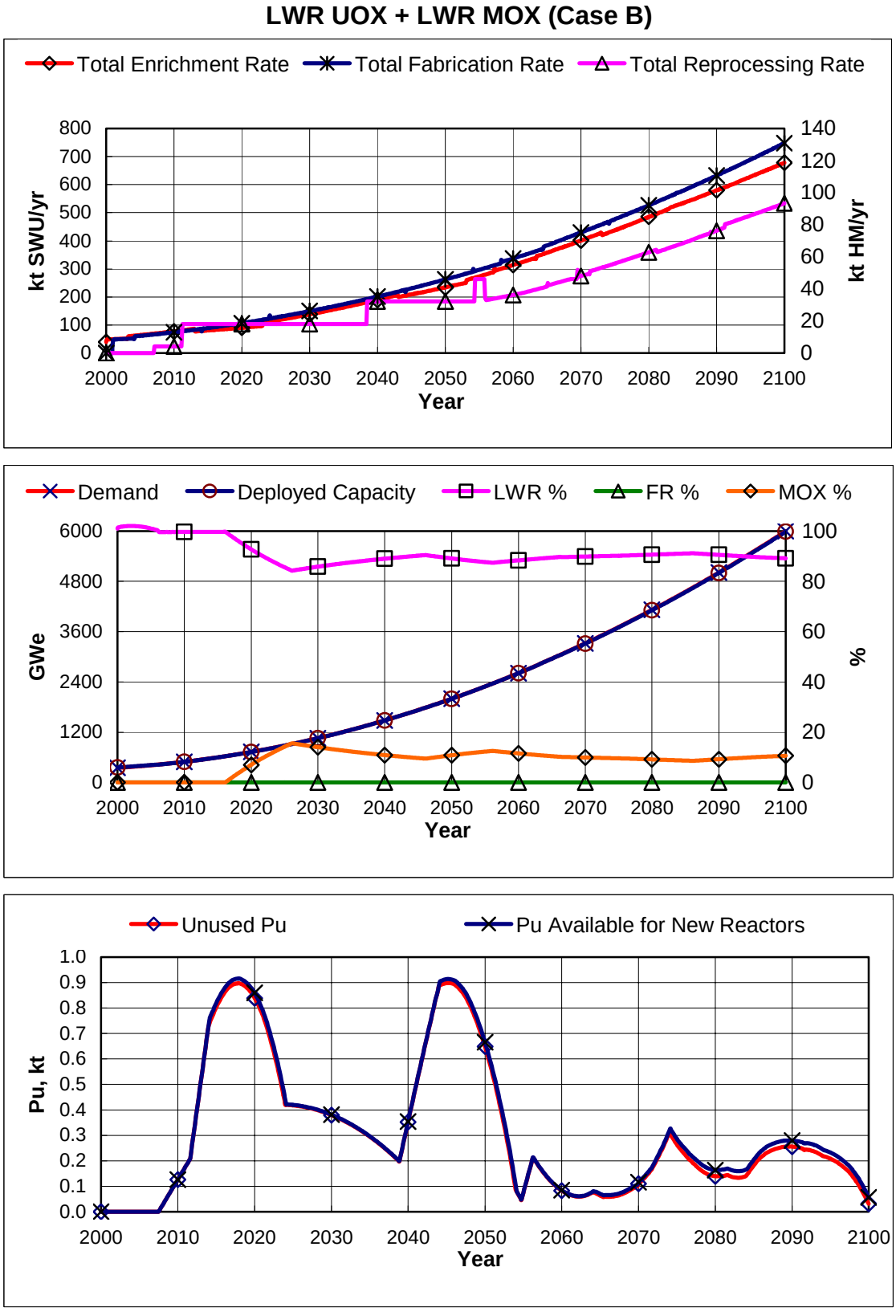

LWR UOX + LWR MOX (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.13

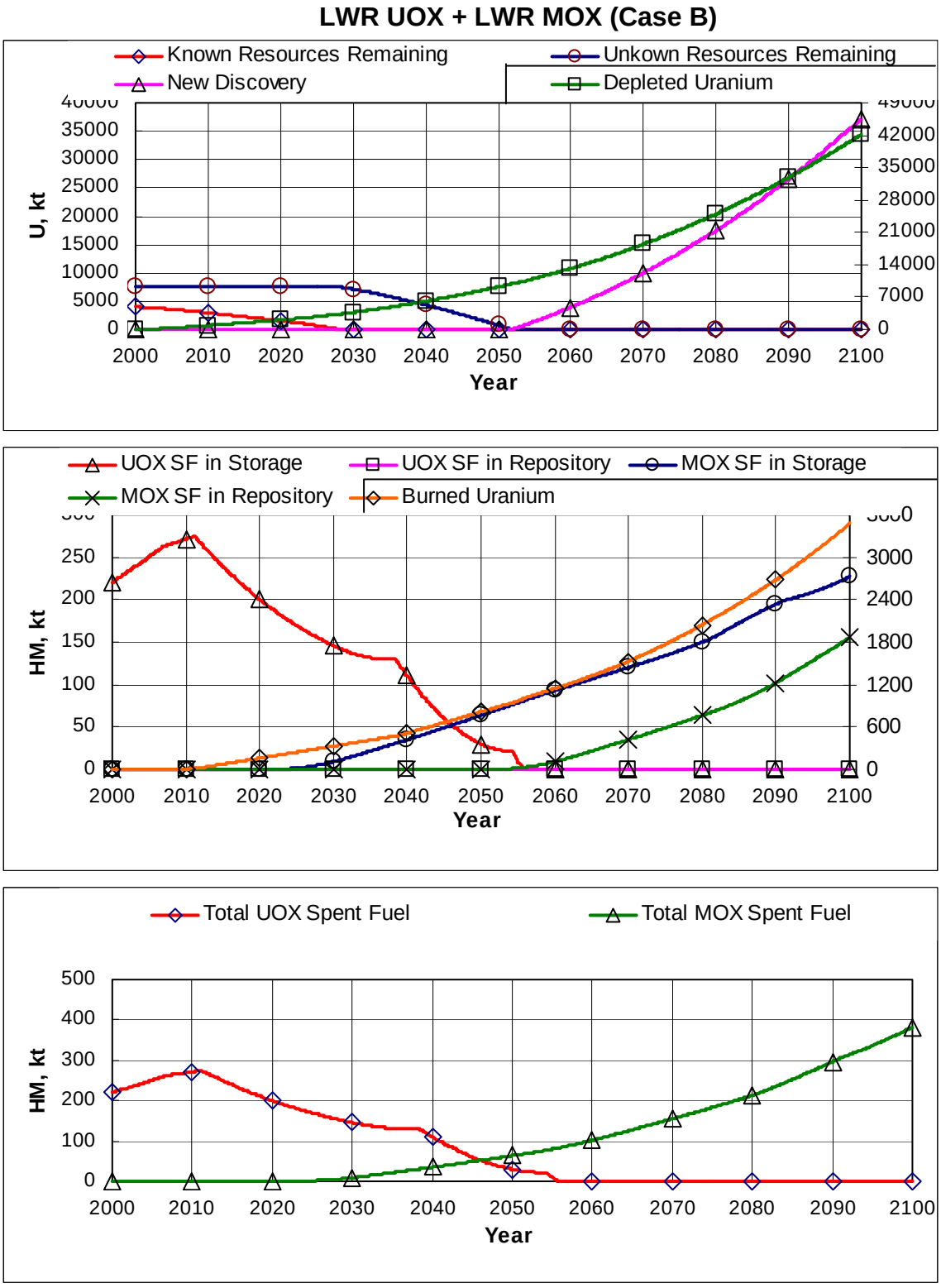

LWR UOX + LWR MOX (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.14

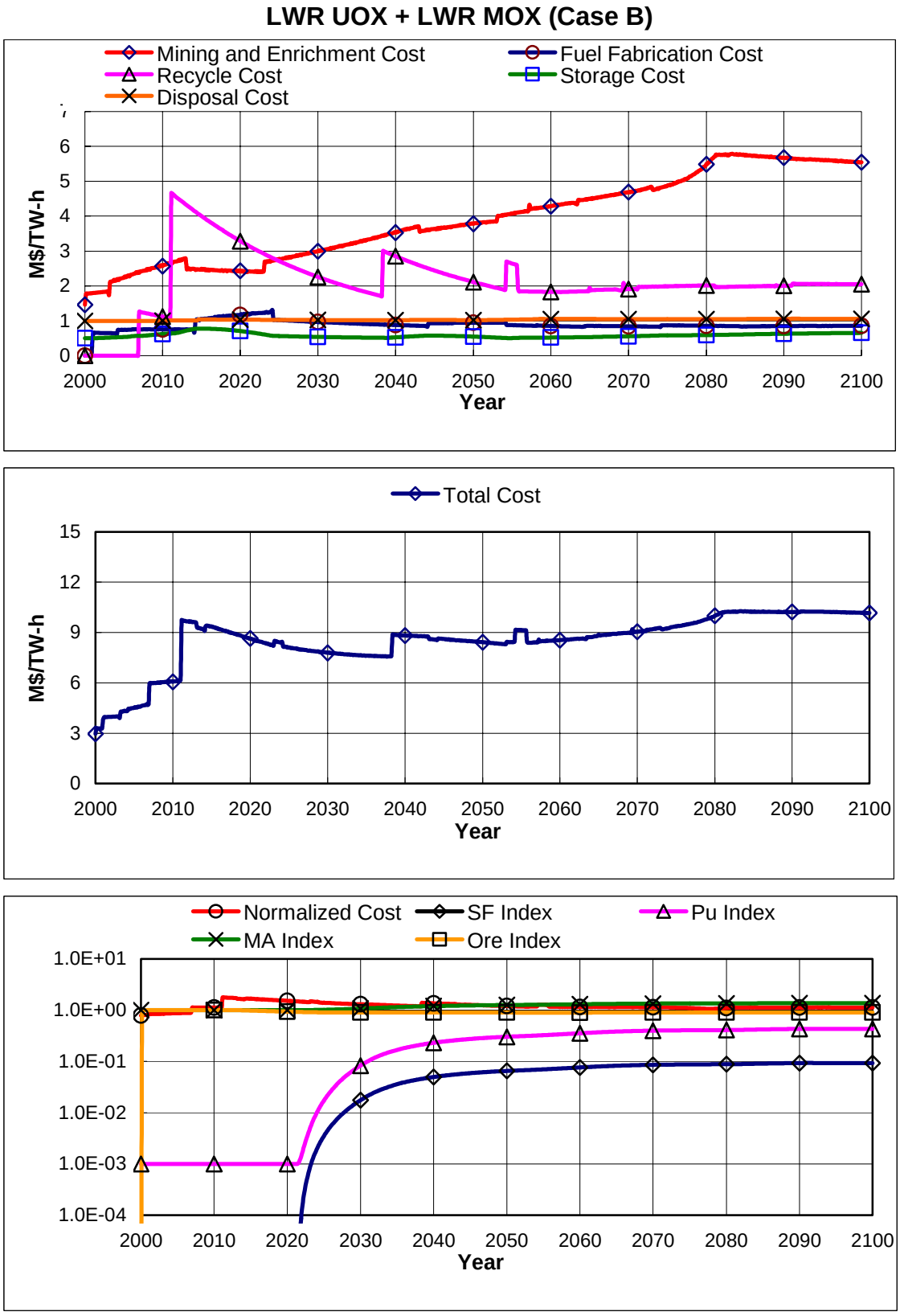
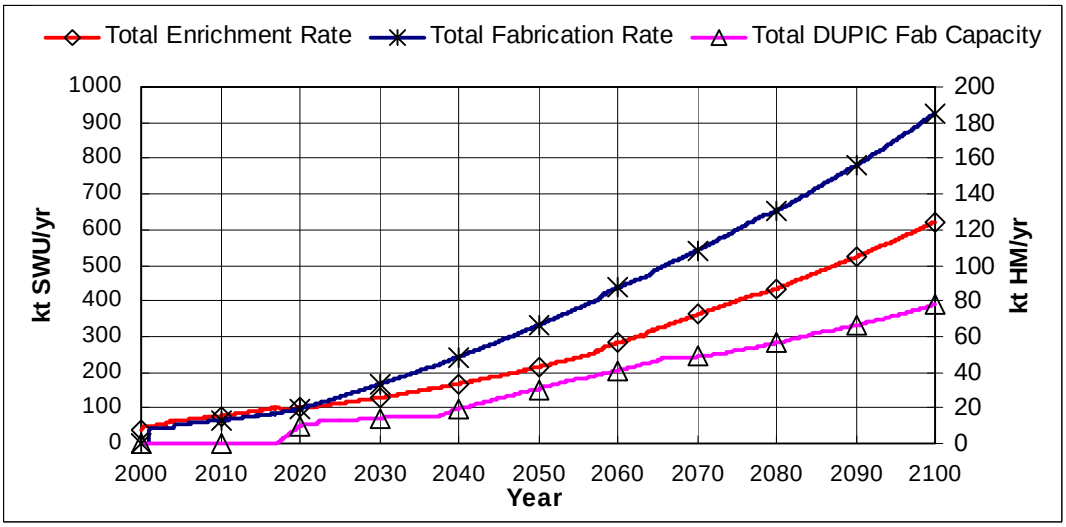
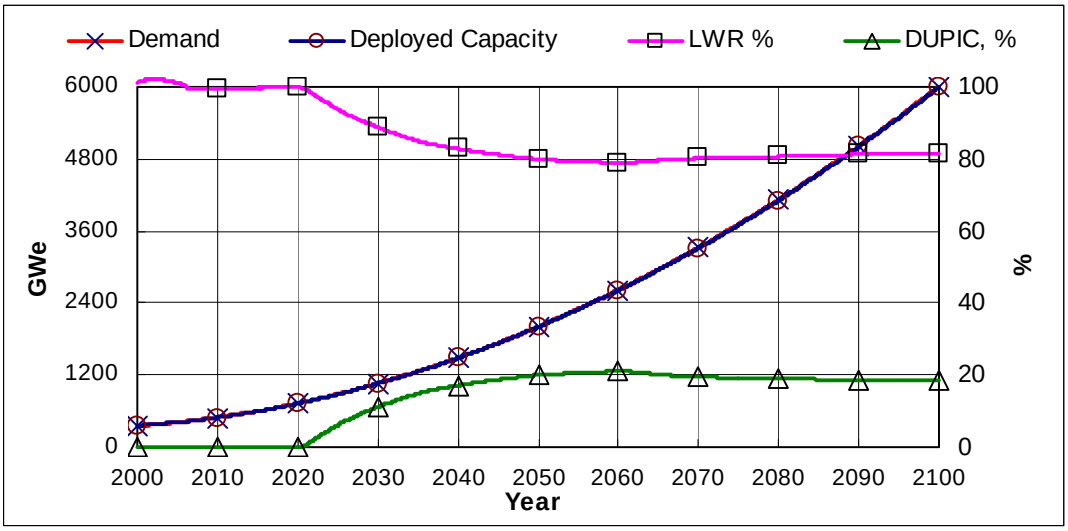

LWR UOX + LWR MOX (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
0
1
2
3
4
5
6
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total Cost
M$/TW-h
0
3
6
9
12
15
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
1.0E-02
1.0E-03
1.0E-04
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Fig. 3.15

DUPIC (Case B)





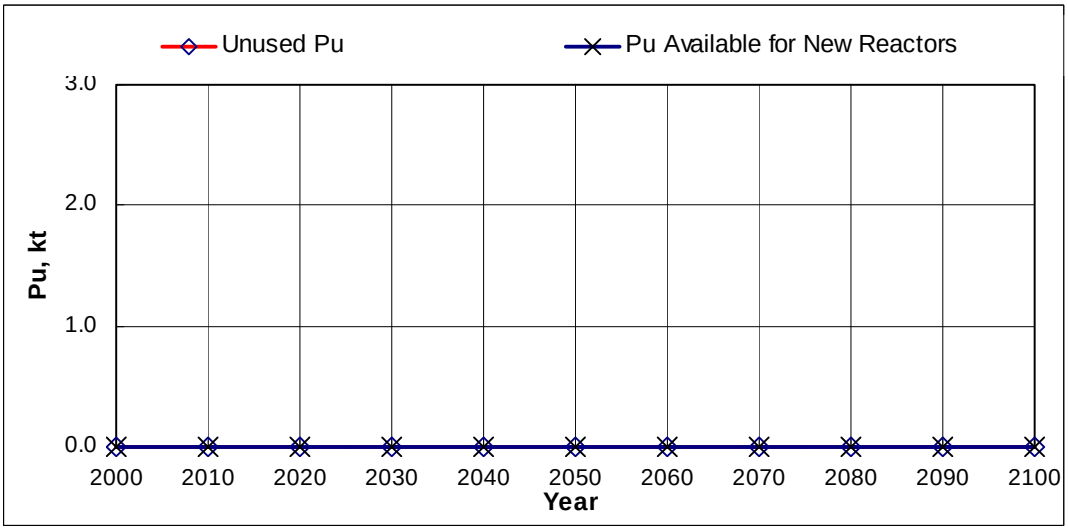


Fig. 3.16

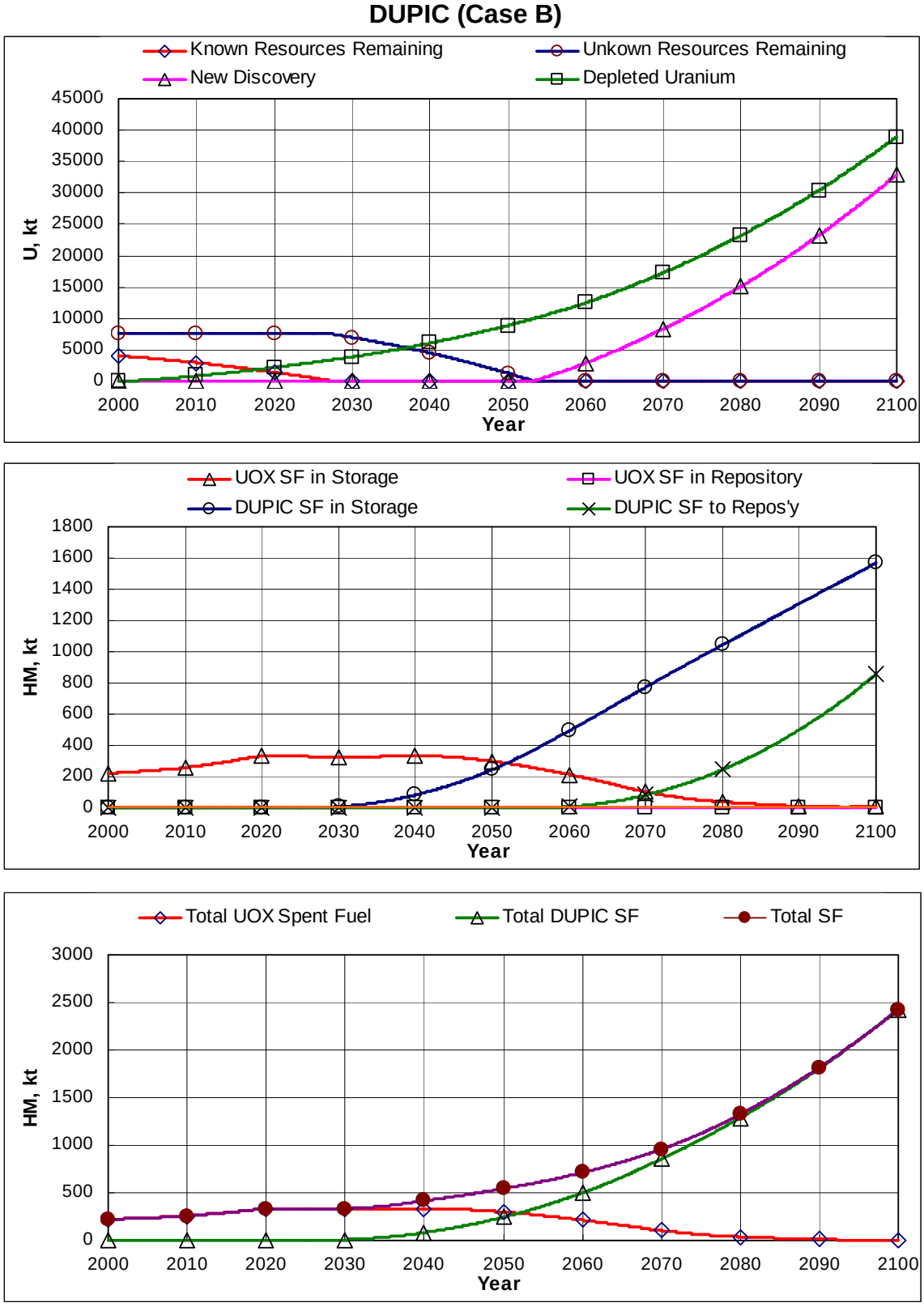
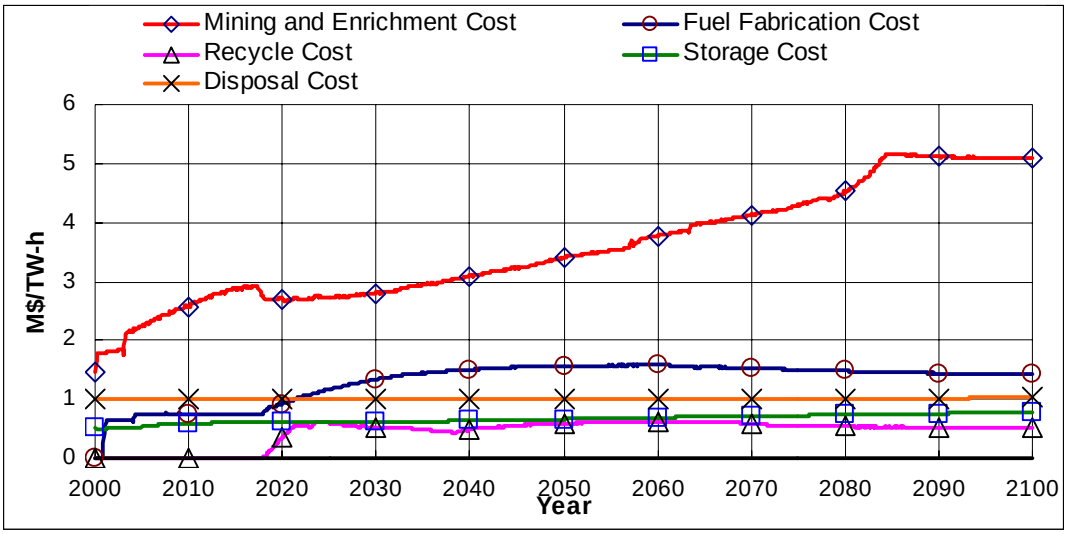
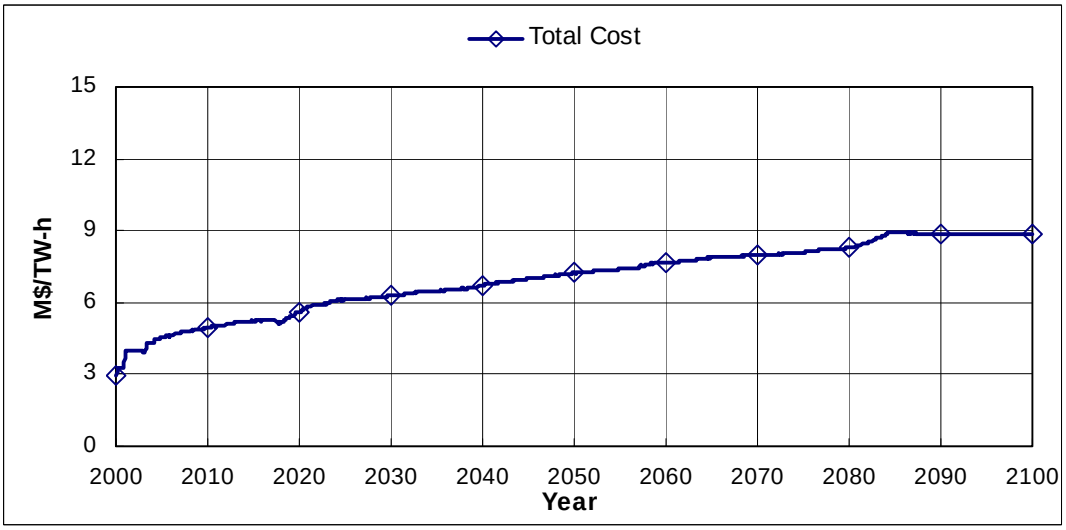

DUPIC (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
45000
40000
35000
30000
25000
20000
15000
10000
5000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
UOX SF in Storage
UOX SF in Repository
DUPIC SF in Storage
DUPIC SF to Repos'y
HM, kt
1800
1600
1400
1200
1000
800
600
400
200
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total UOX Spent Fuel
Total DUPIC SF
Total SF
HM, kt
3000
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Fig. 3.17

DUPIC (Case B)





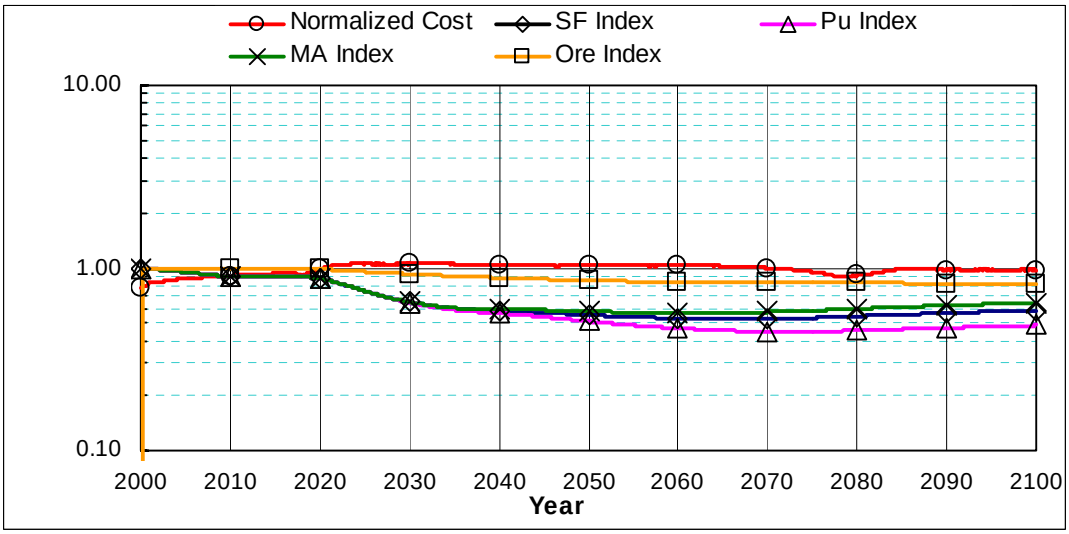


Fig. 3.18

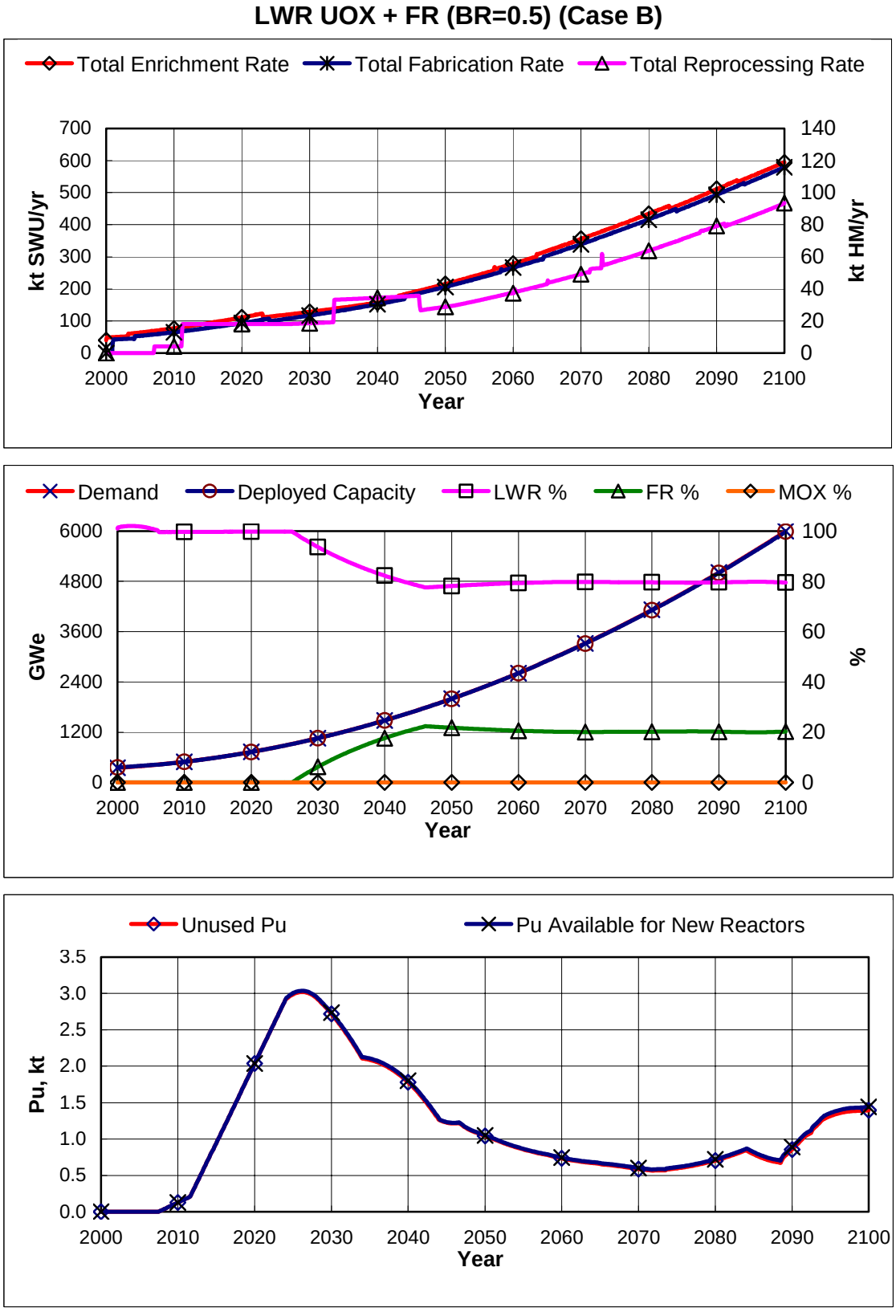

LWR UOX + FR (BR=0.5) (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.19

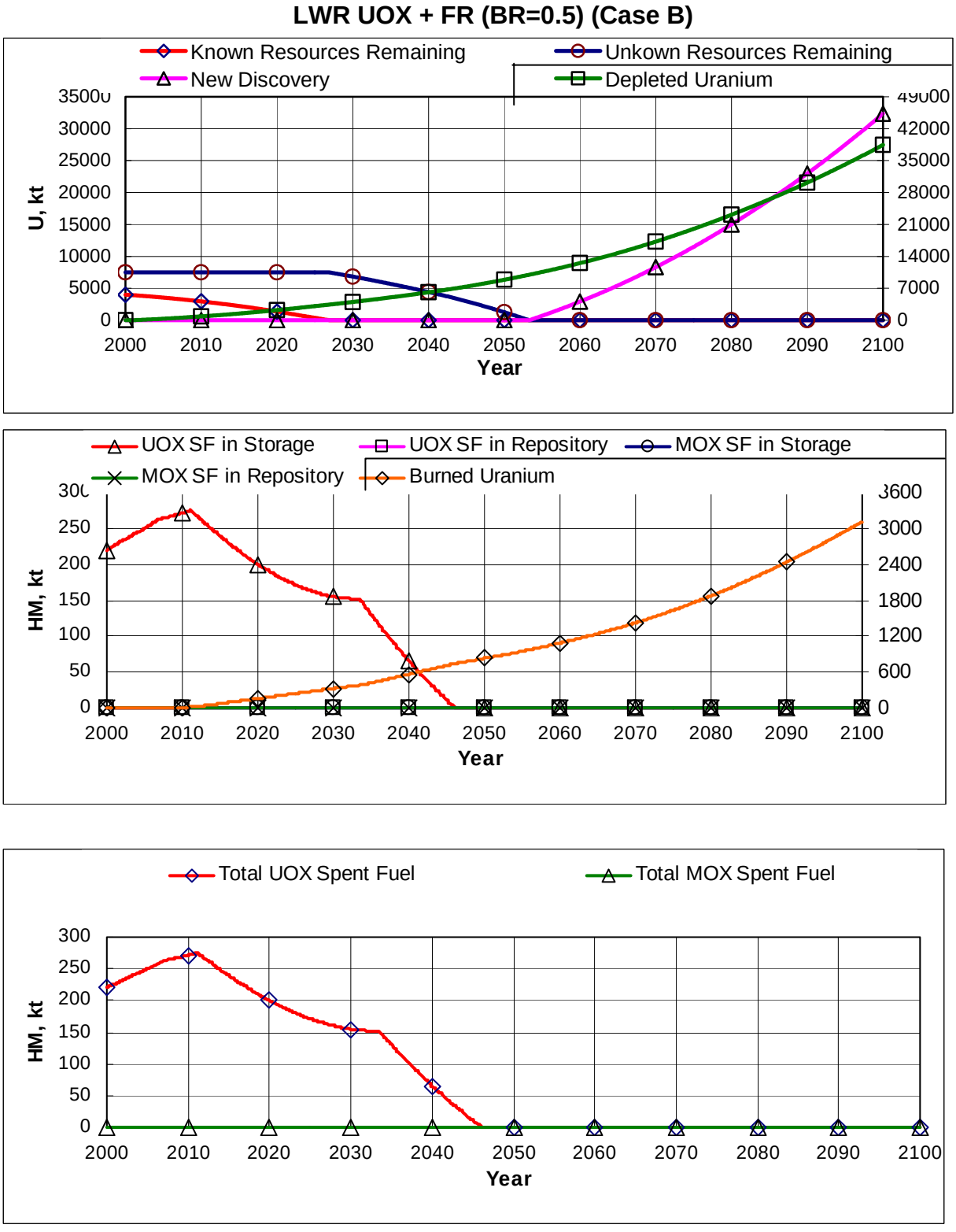

LWR UOX + FR (BR=0.5) (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.20

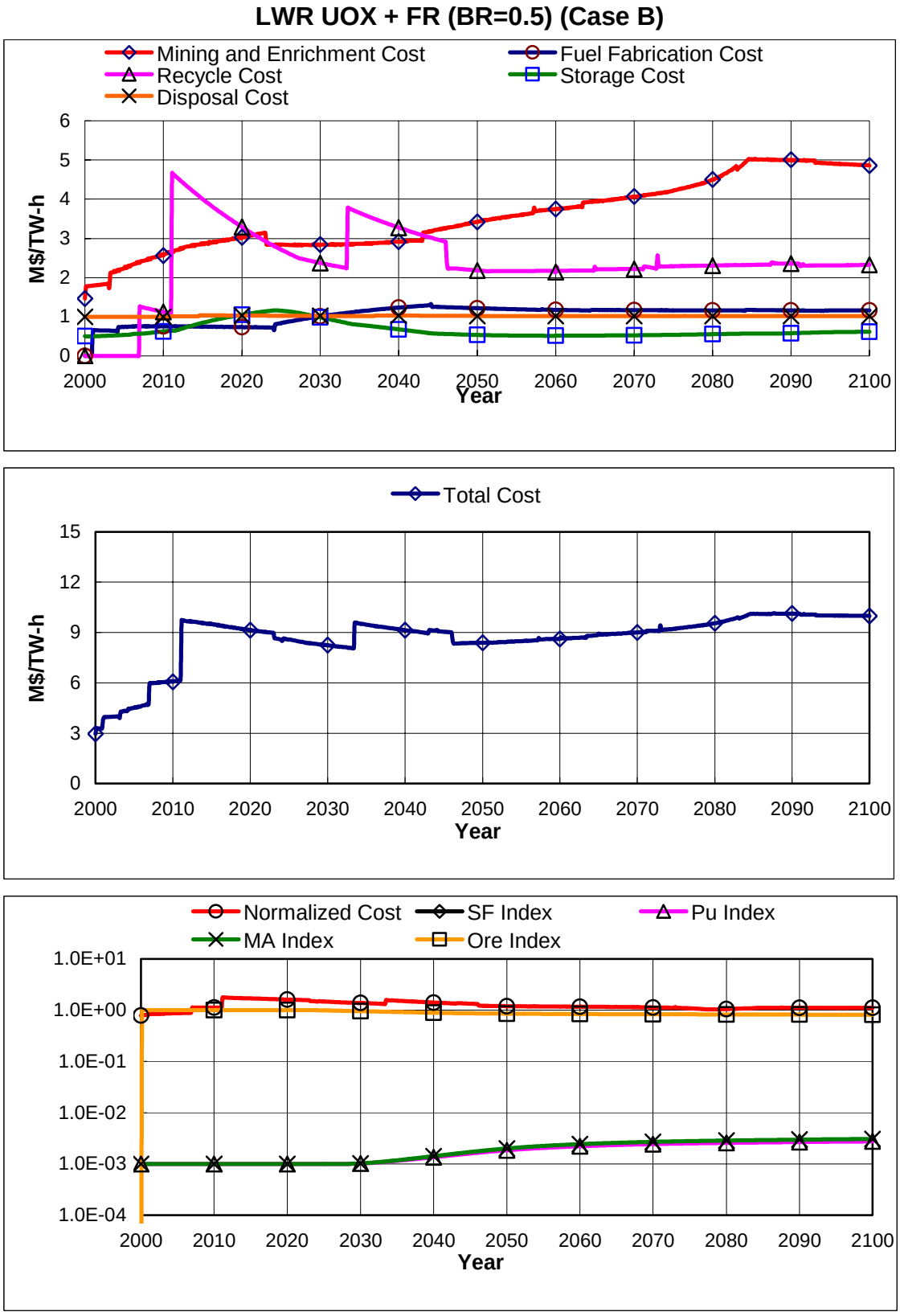
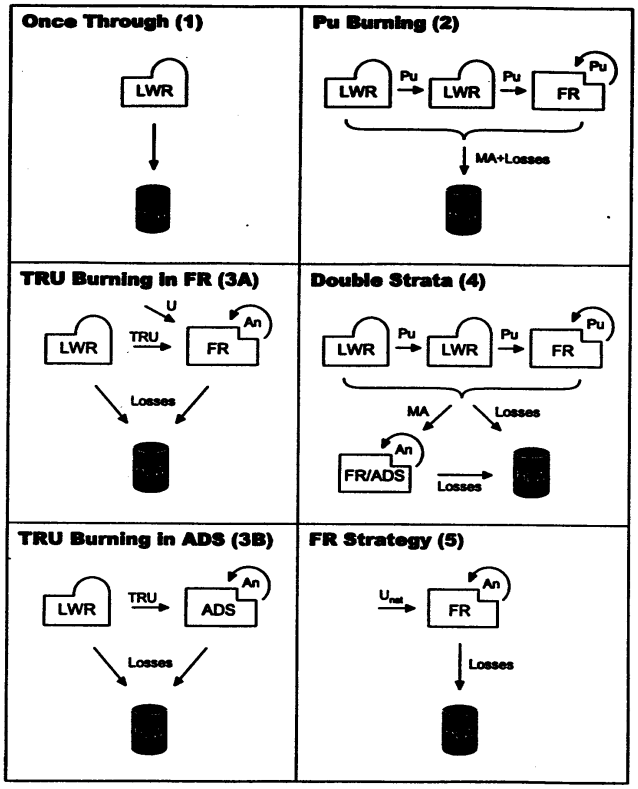

LWR UOX + FR (BR=0.5) (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. 3.21



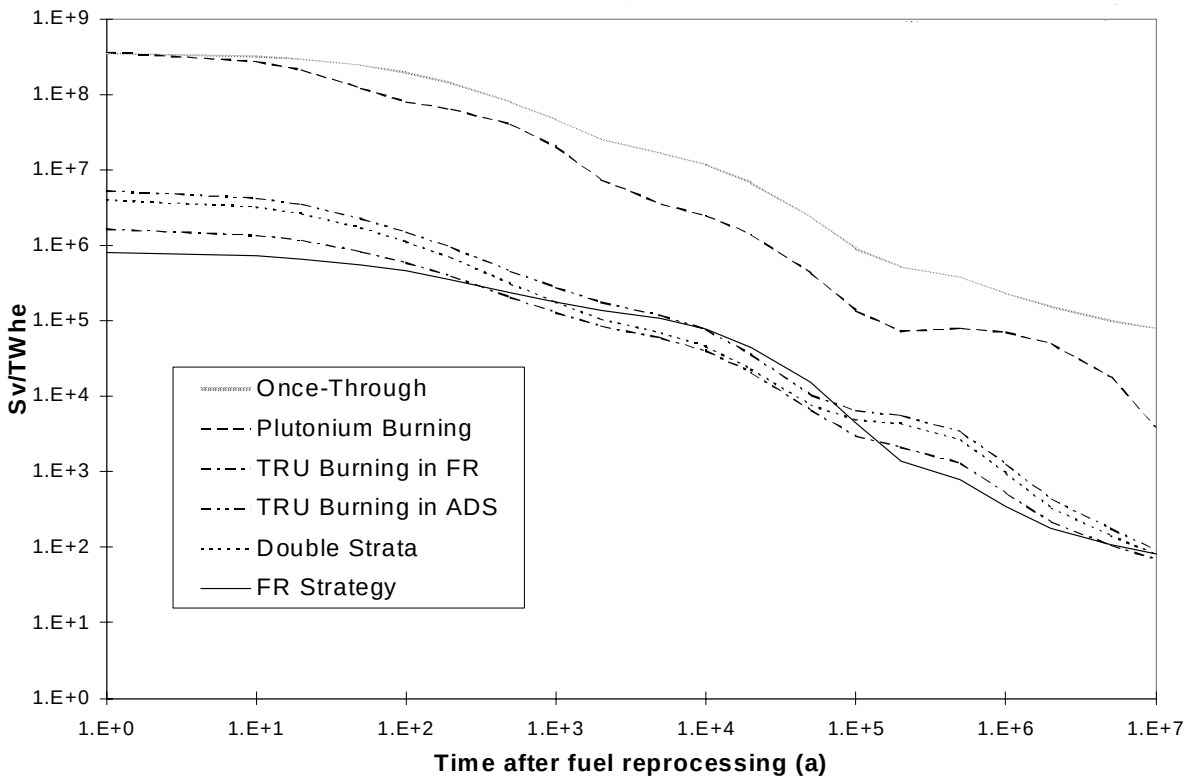


Fig. 3.22 Full Pu Recycle vs Full TU Recycle: MA Toxicity

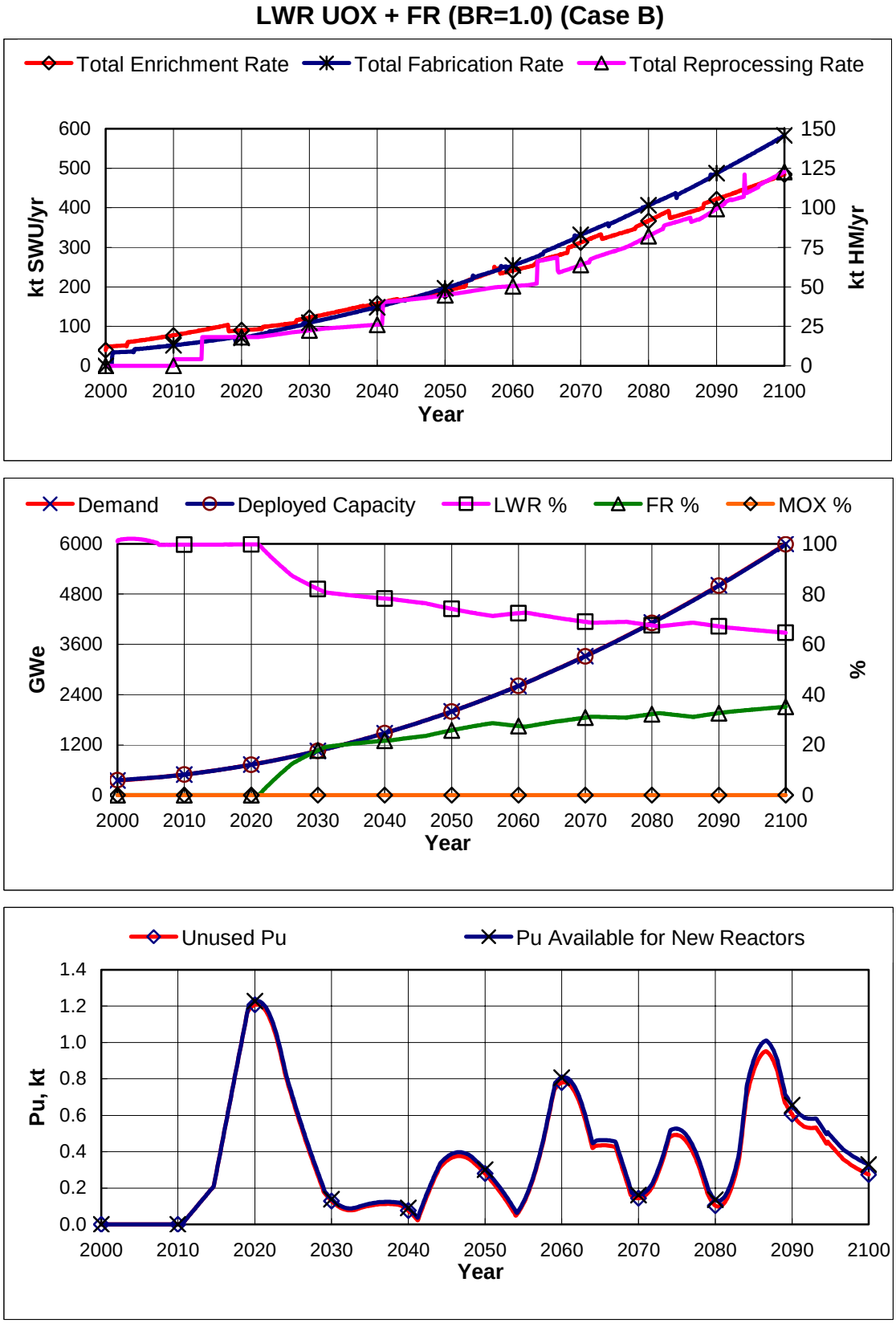

LWR UOX + FR (BR=1.0) (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.23

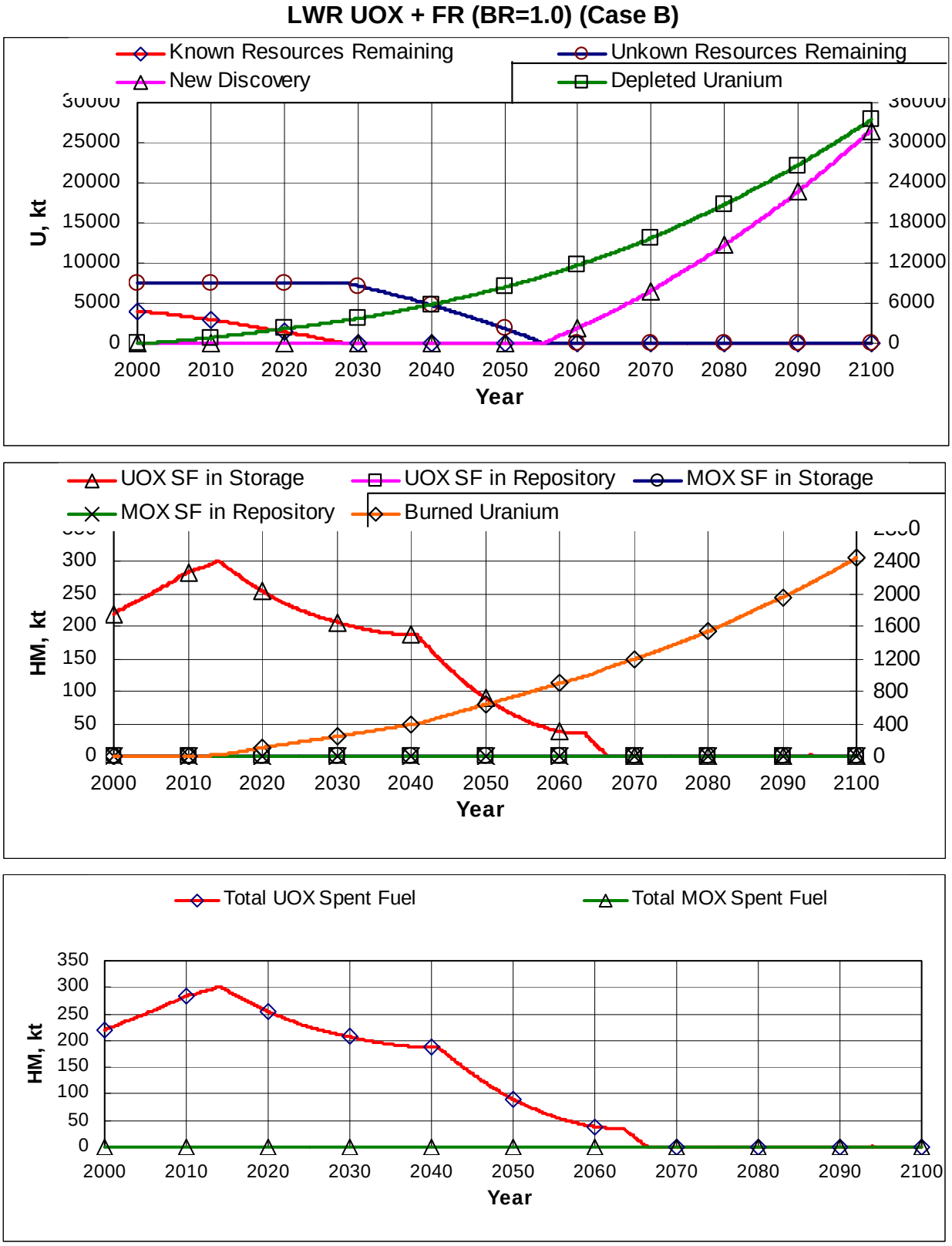

LWR UOX + FR (BR=1.0) (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.24

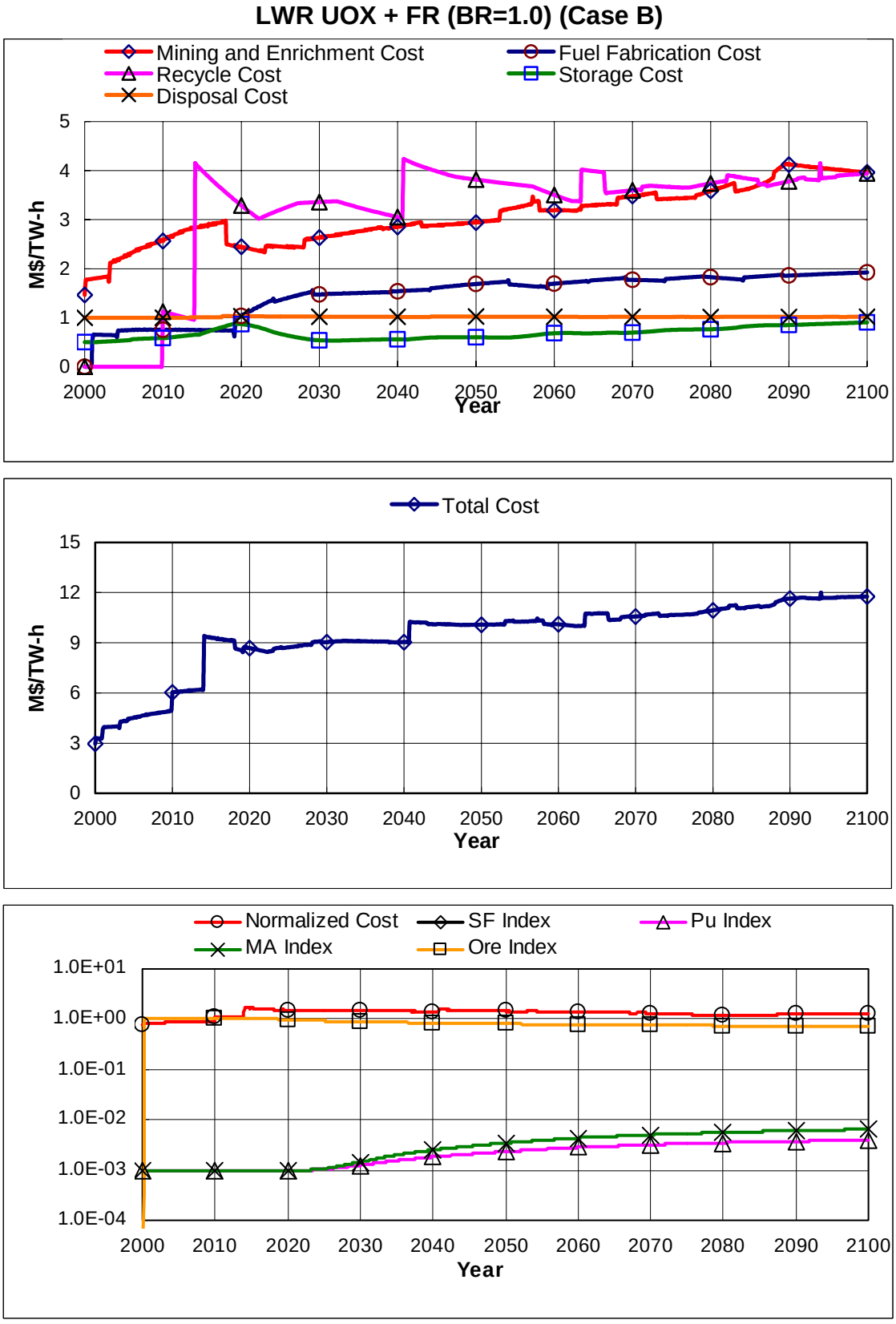

LWR UOX + FR (BR=1.0) (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. 3.25

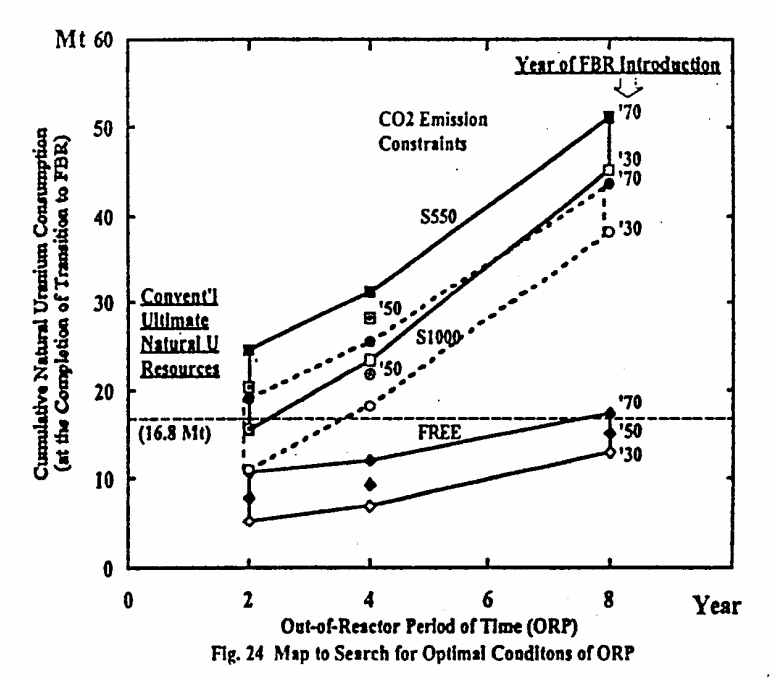


Fig. 24 Map to Search for Optimal Conditons of ORP

Koike et al Global 99
S550 ≃ WEC/IIASA Case B
350 → 2000 → 6000 GWe
S1000 350 → 1600 → 4200 GWe
Free 350 → 1100 → 2200 GWe

Fig. 3.26 Time Window of Opportunity for Transition to Resource Sustainability

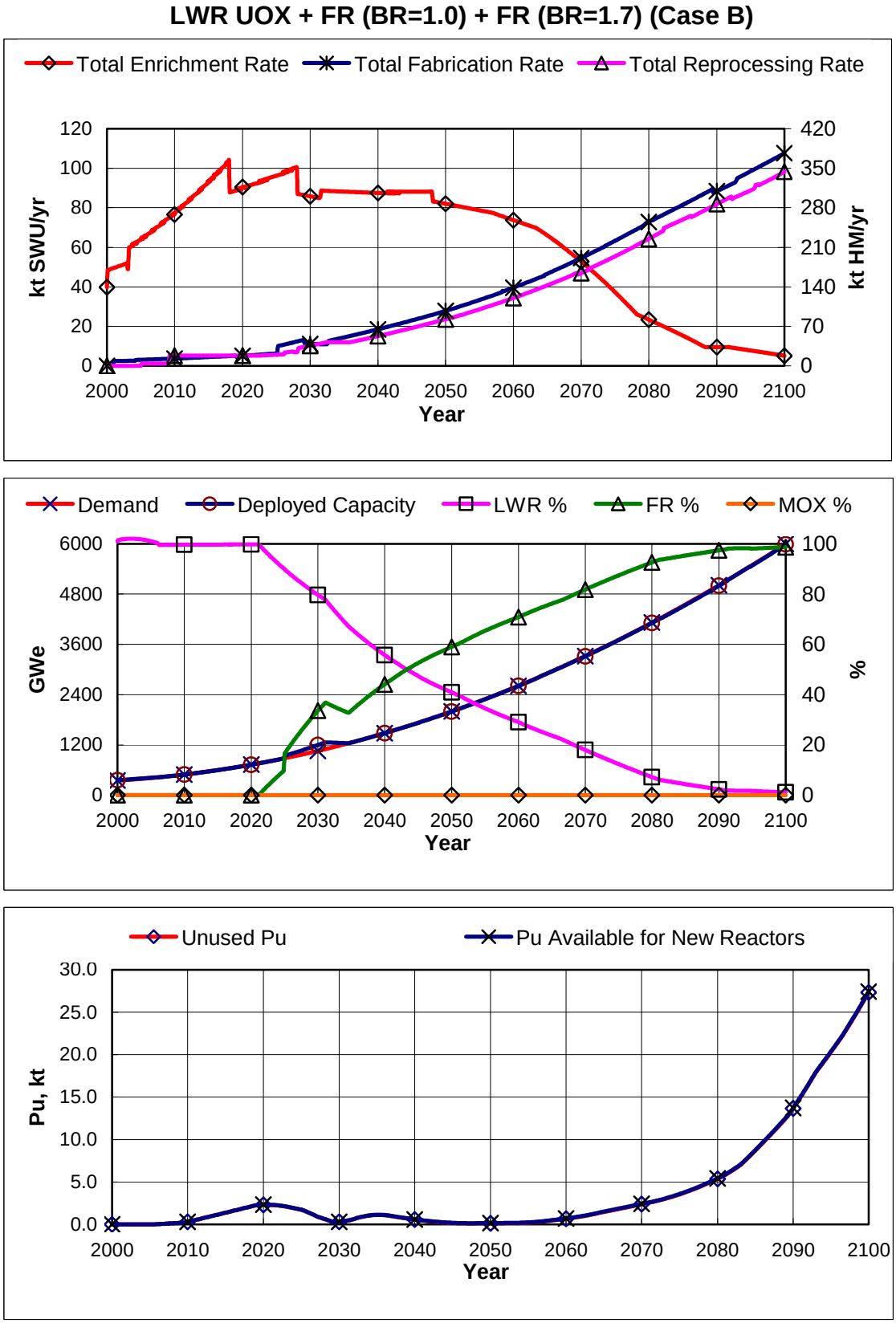

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.27

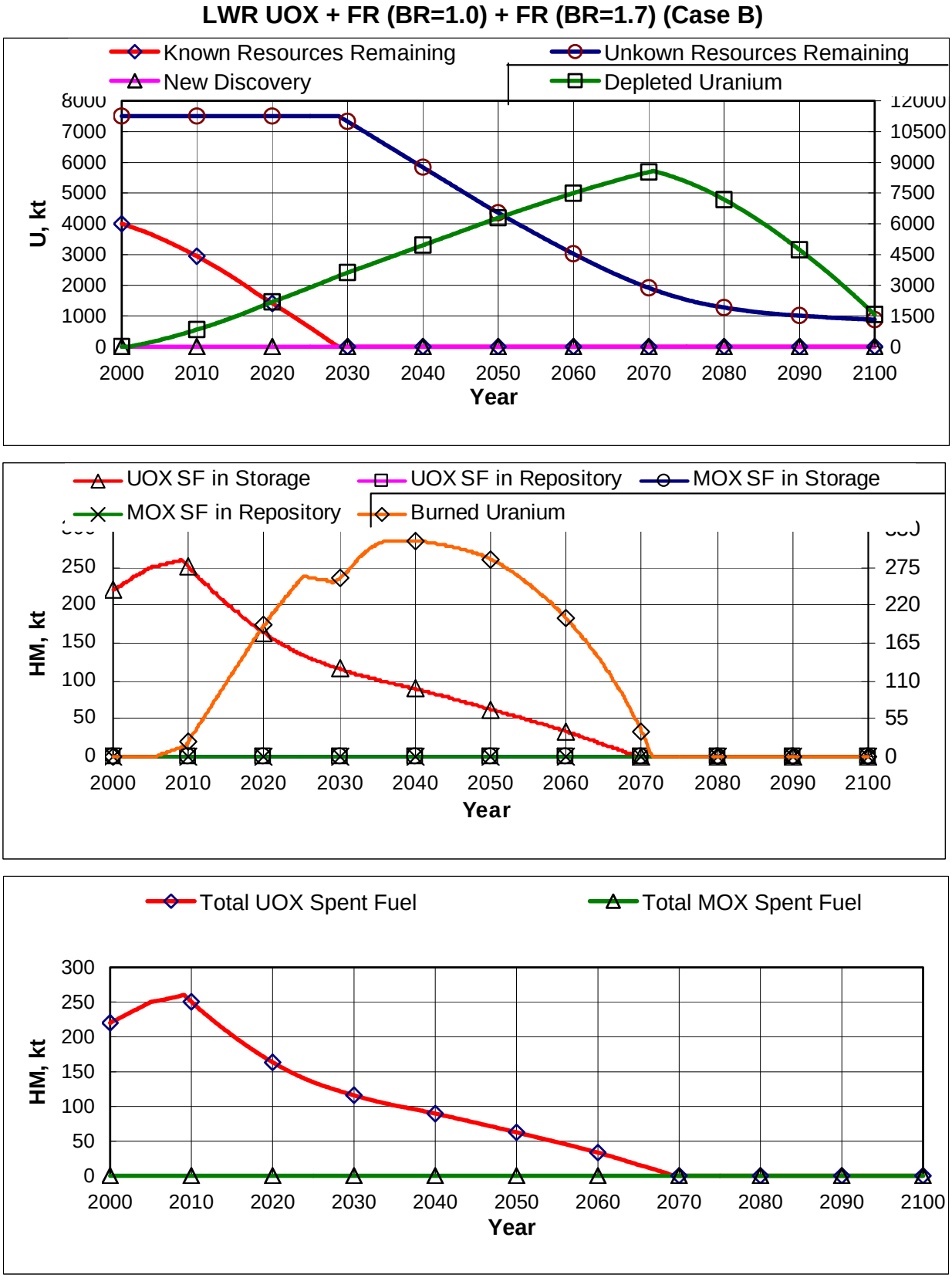

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. 3.28

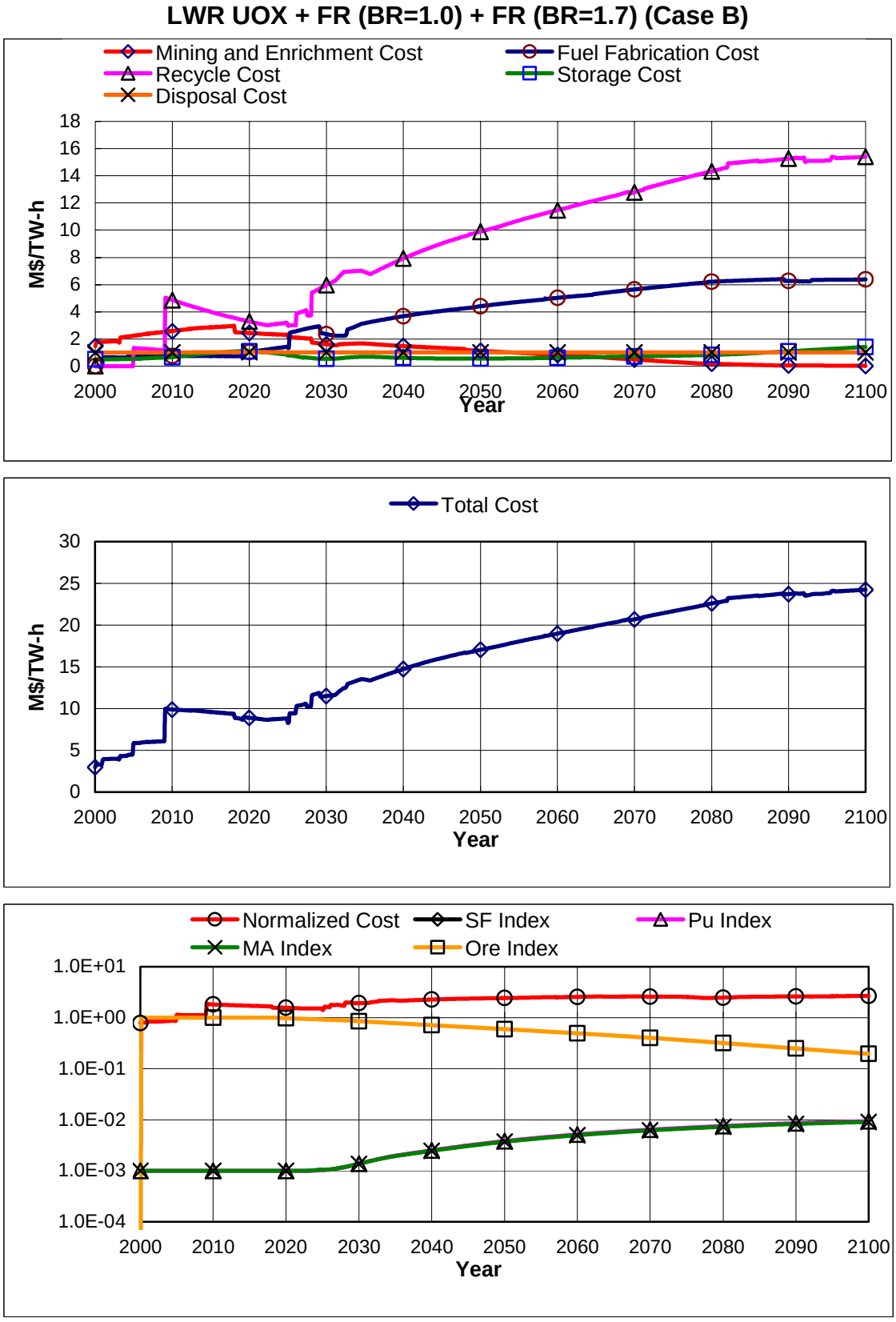

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index

Fig. 3.29

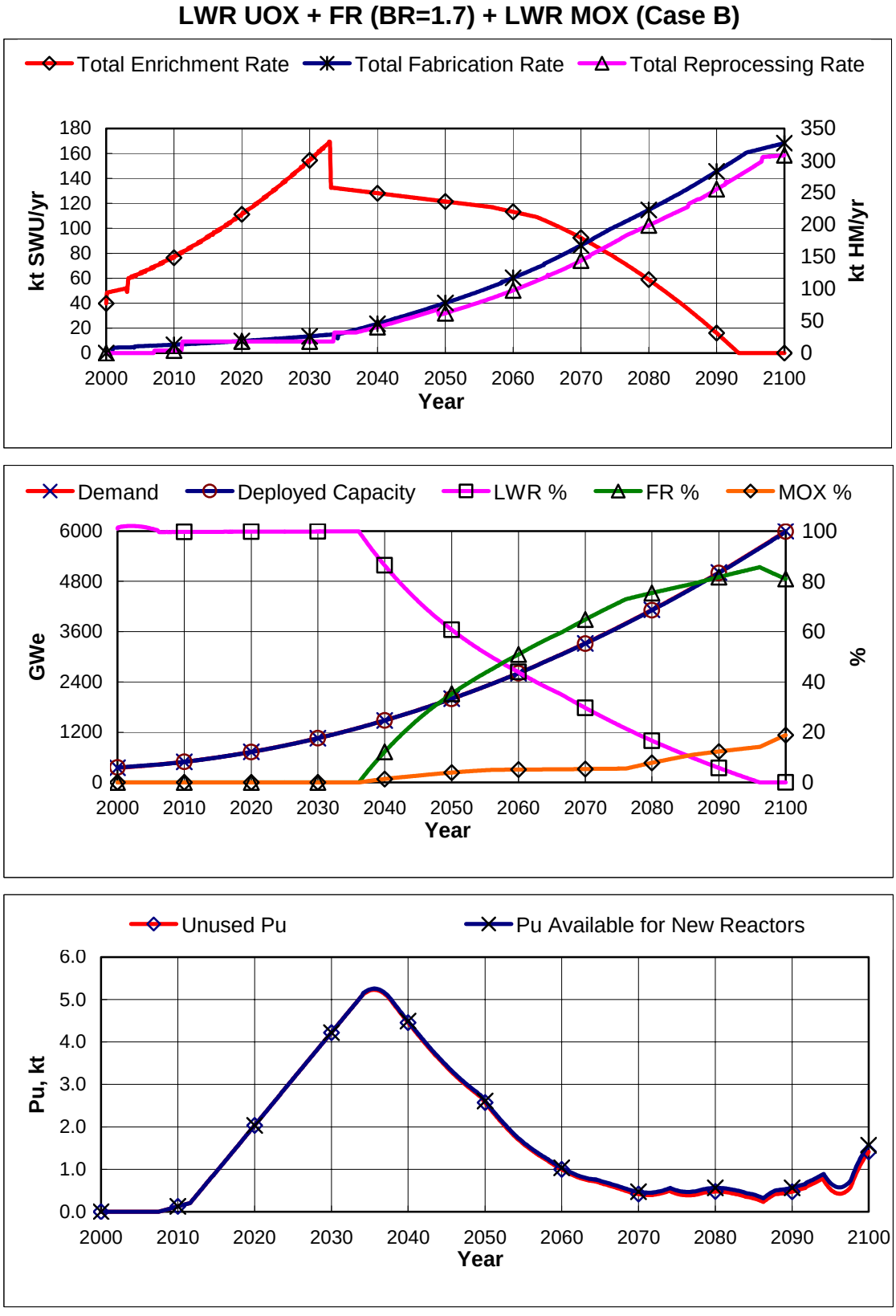
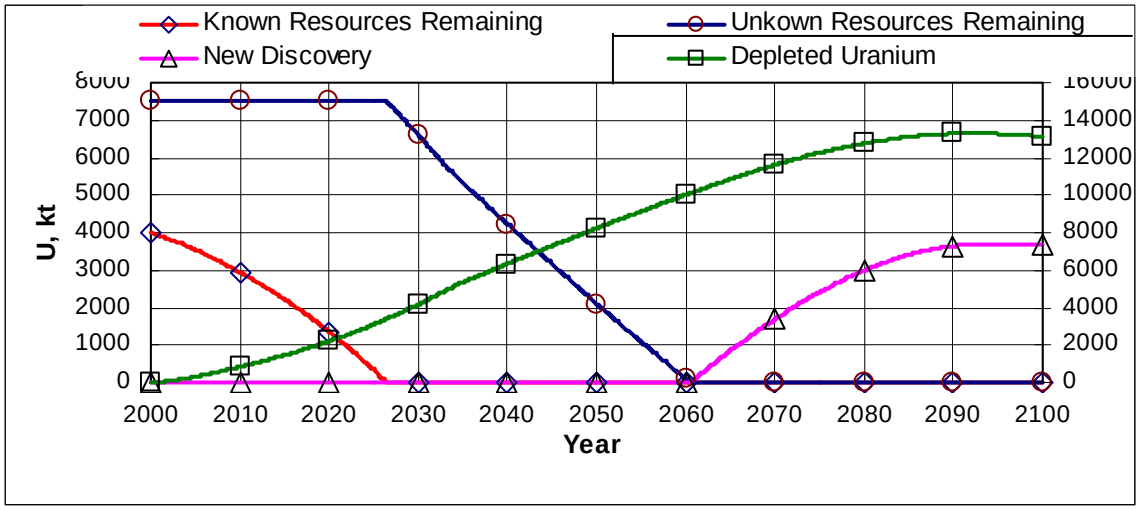
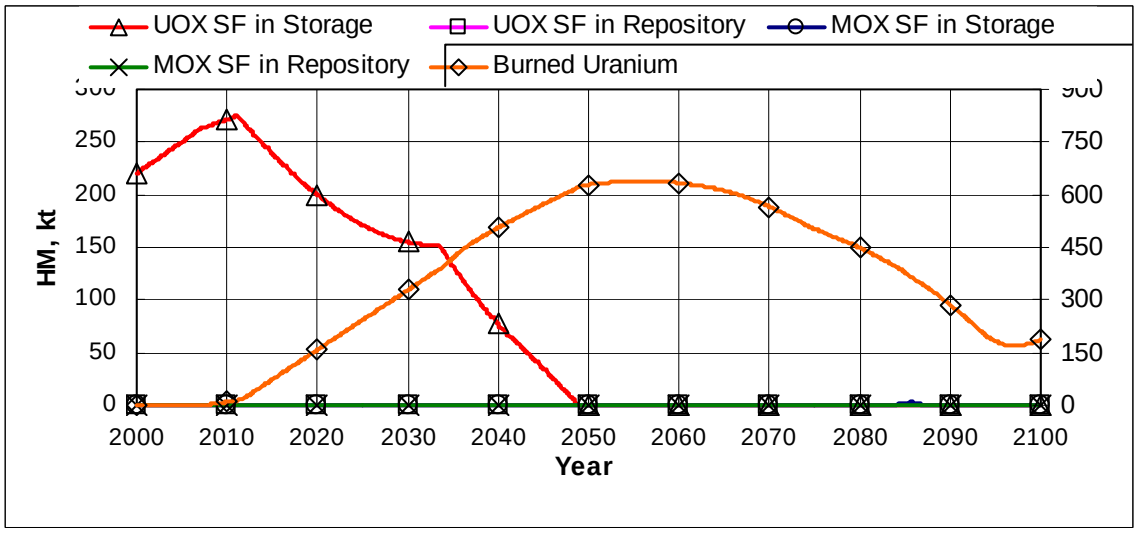

LWR UOX + FR (BR=1.7) + LWR MOX (Case B)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3.30

LWR UOX + FR (BR=1.7) + LWR MOX (Case B)





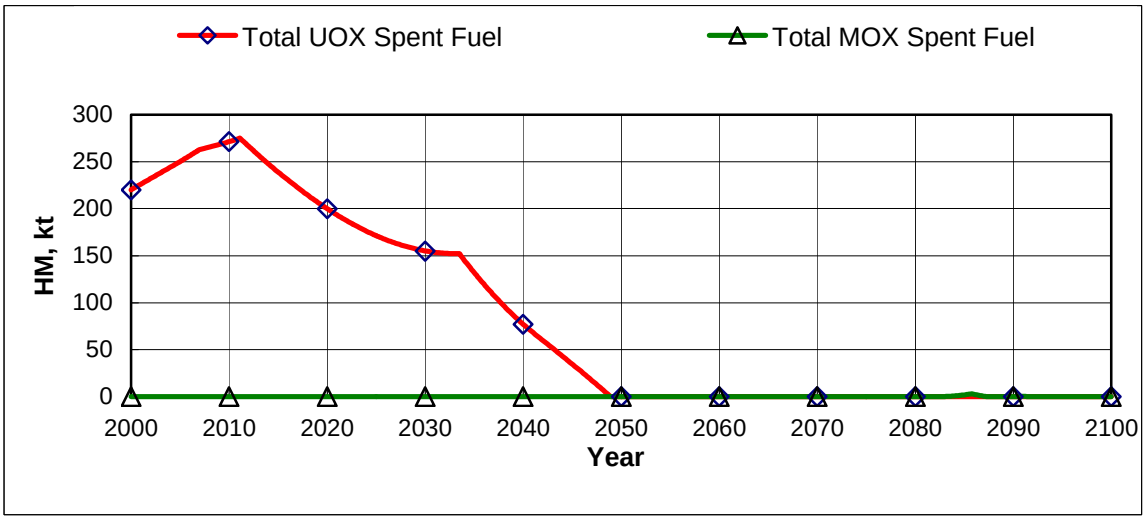


Fig. 3.31

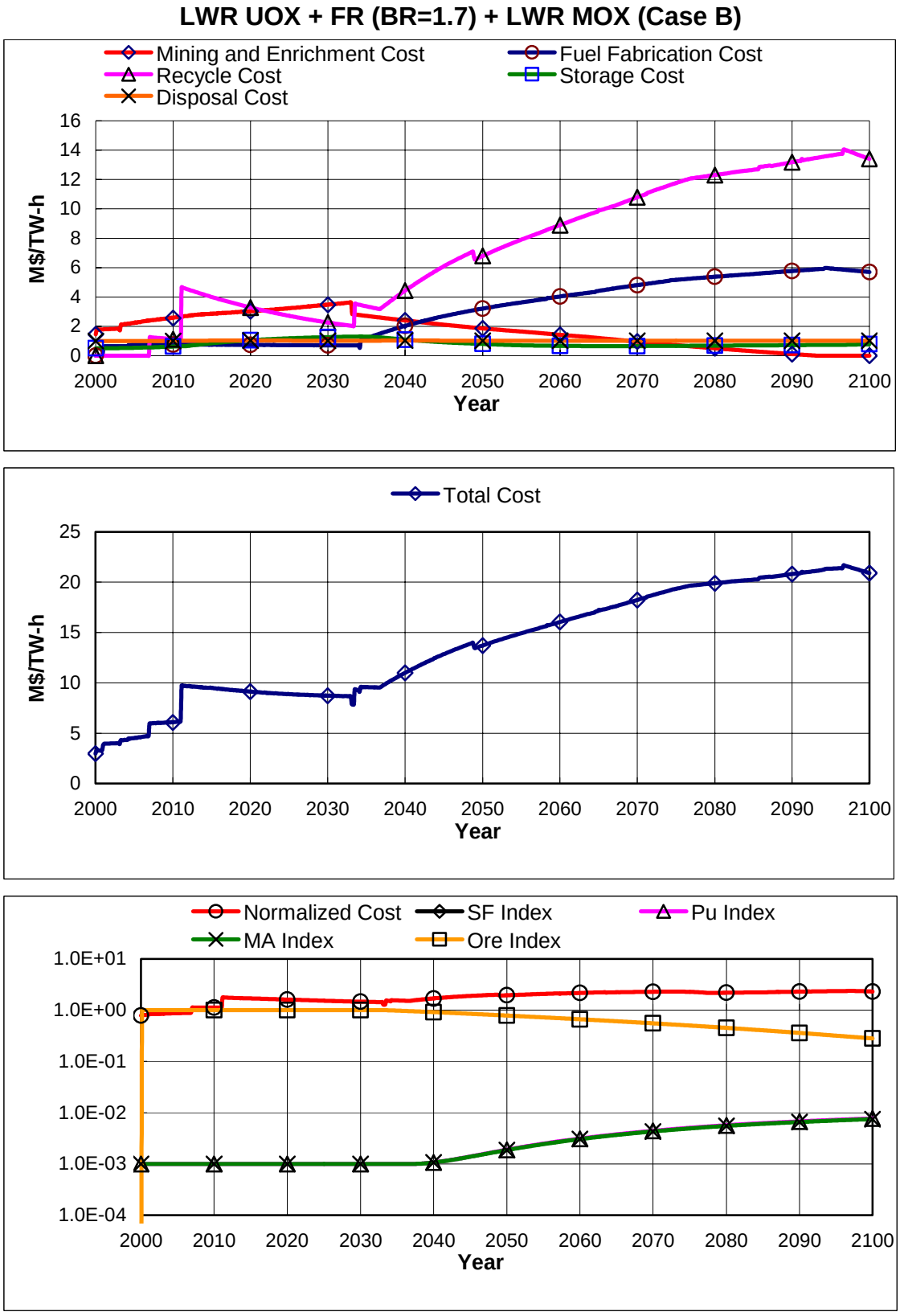
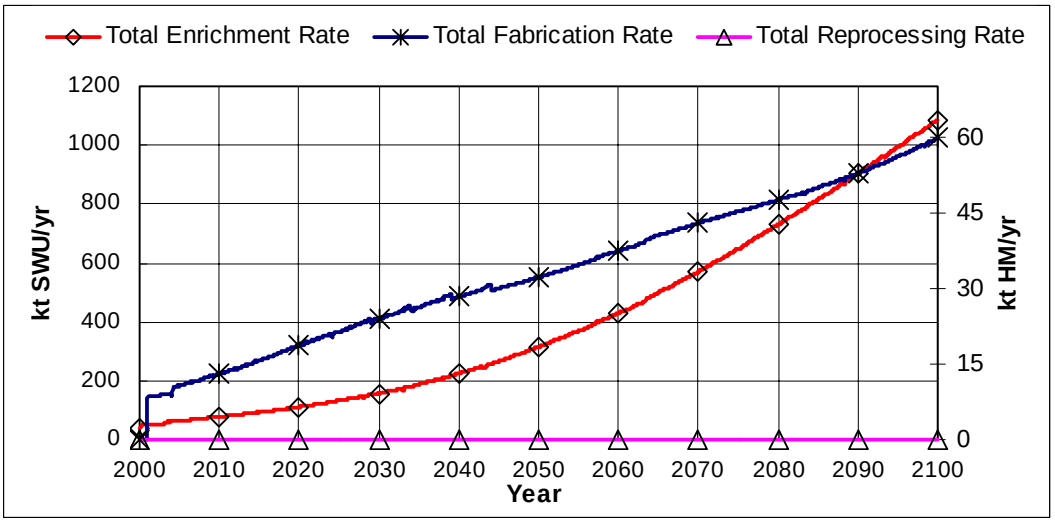
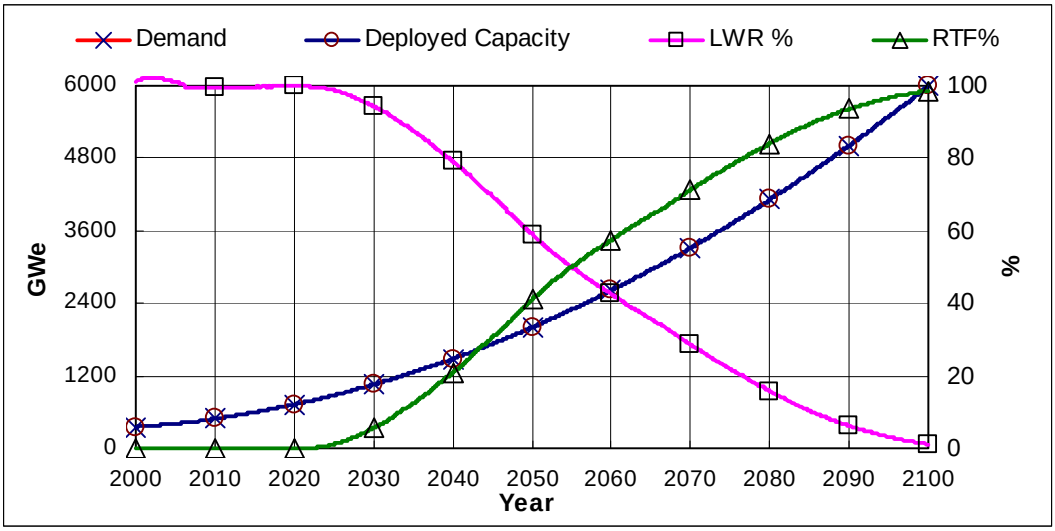

LWR UOX + FR (BR=1.7) + LWR MOX (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. 3.32

Radkowsky (Case B)





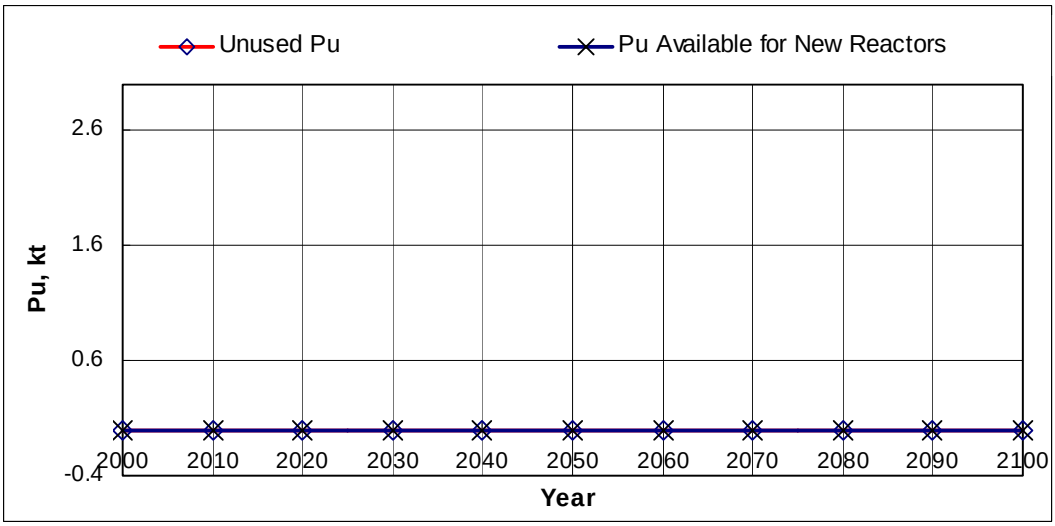


Fig. 3.33

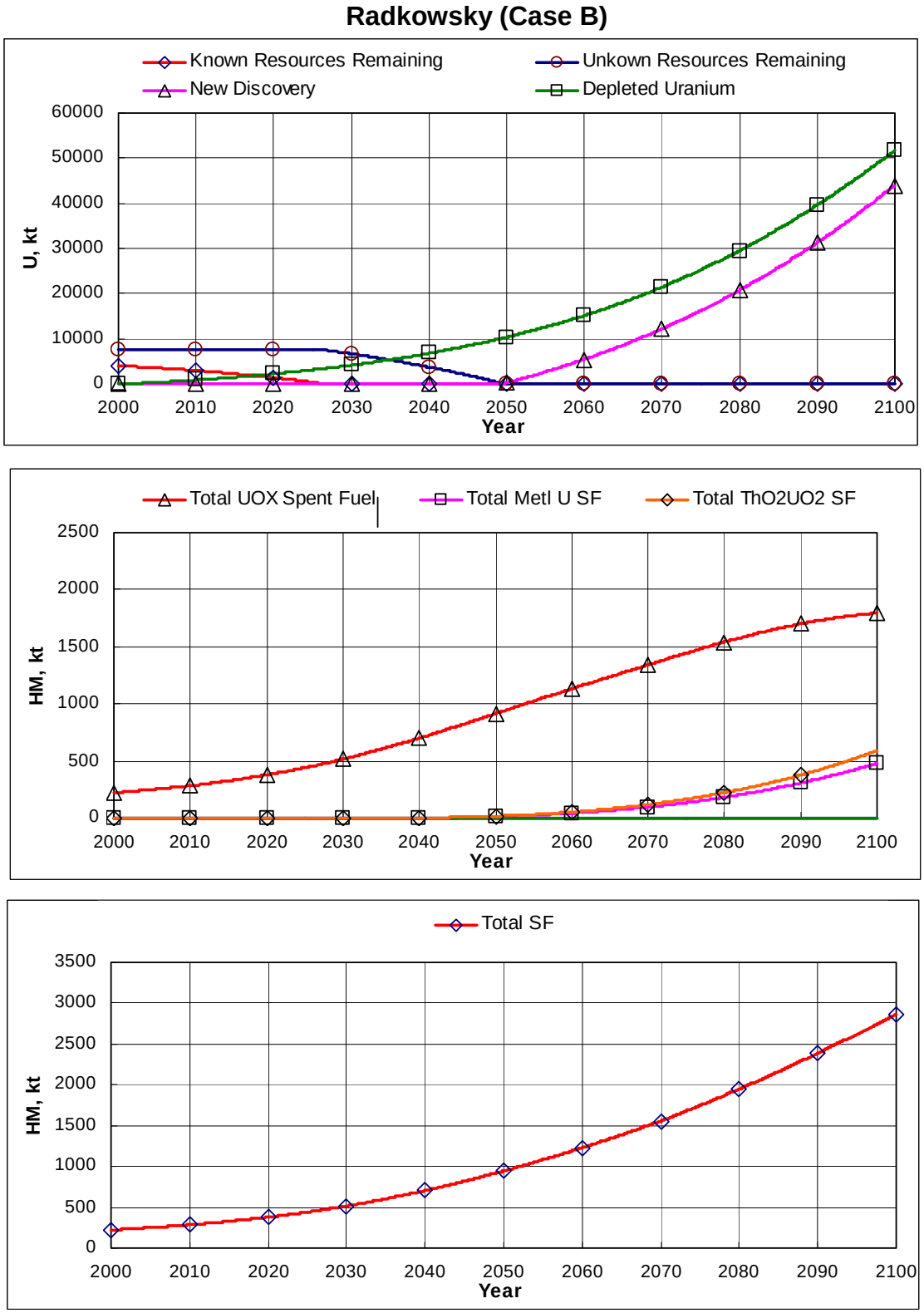

Radkowsky (Case B)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
60000
50000
40000
30000
20000
10000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total UOX Spent Fuel
Total Metl U SF
Total ThO2UO2 SF
HM, kt
2500
2000
1500
1000
500
0
Year
Total SF
HM, kt
3500
3000
2500
2000
1500
1000
500
0
Year

Fig. 3-34

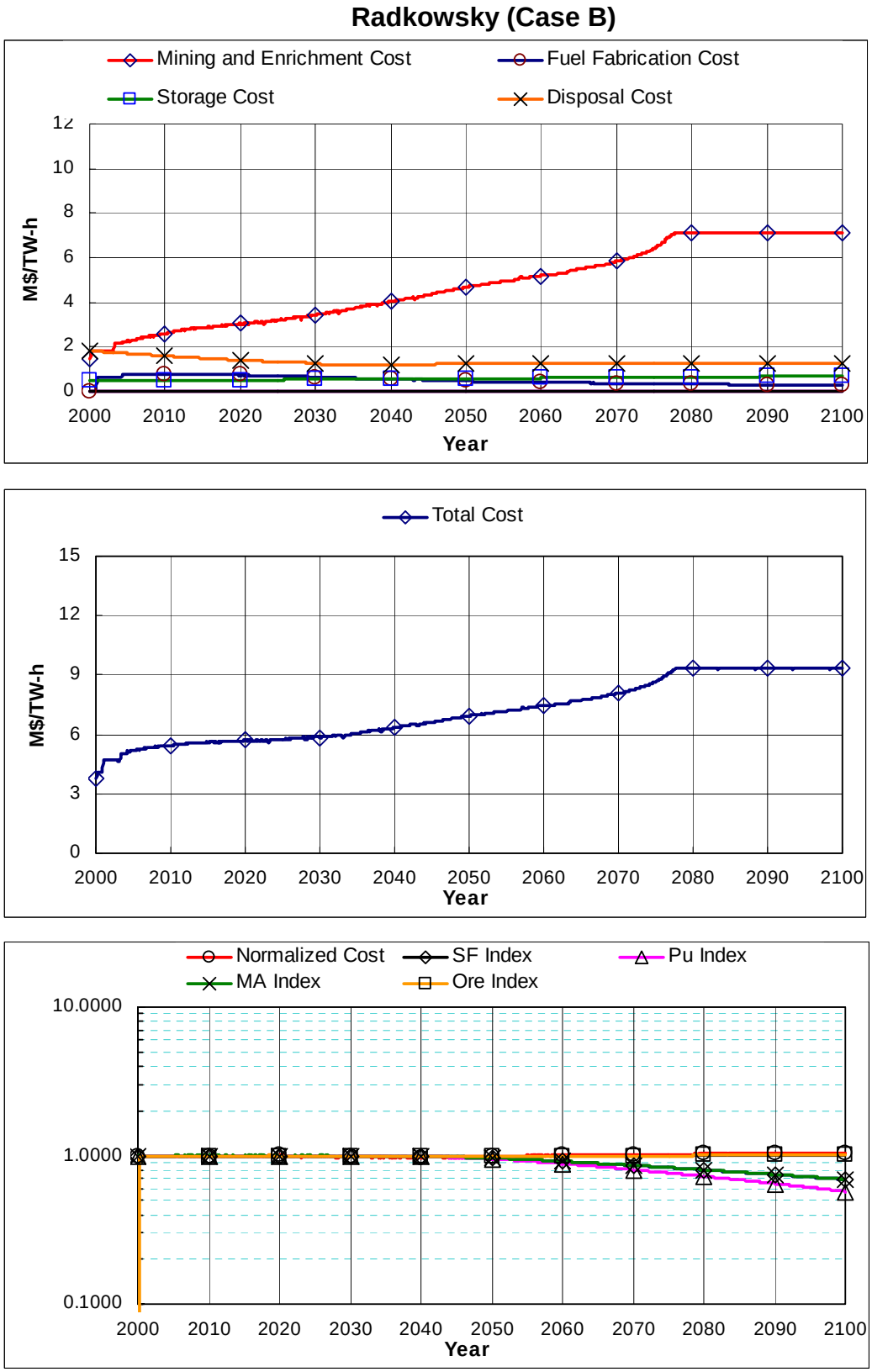

Radkowsky (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. 3-35

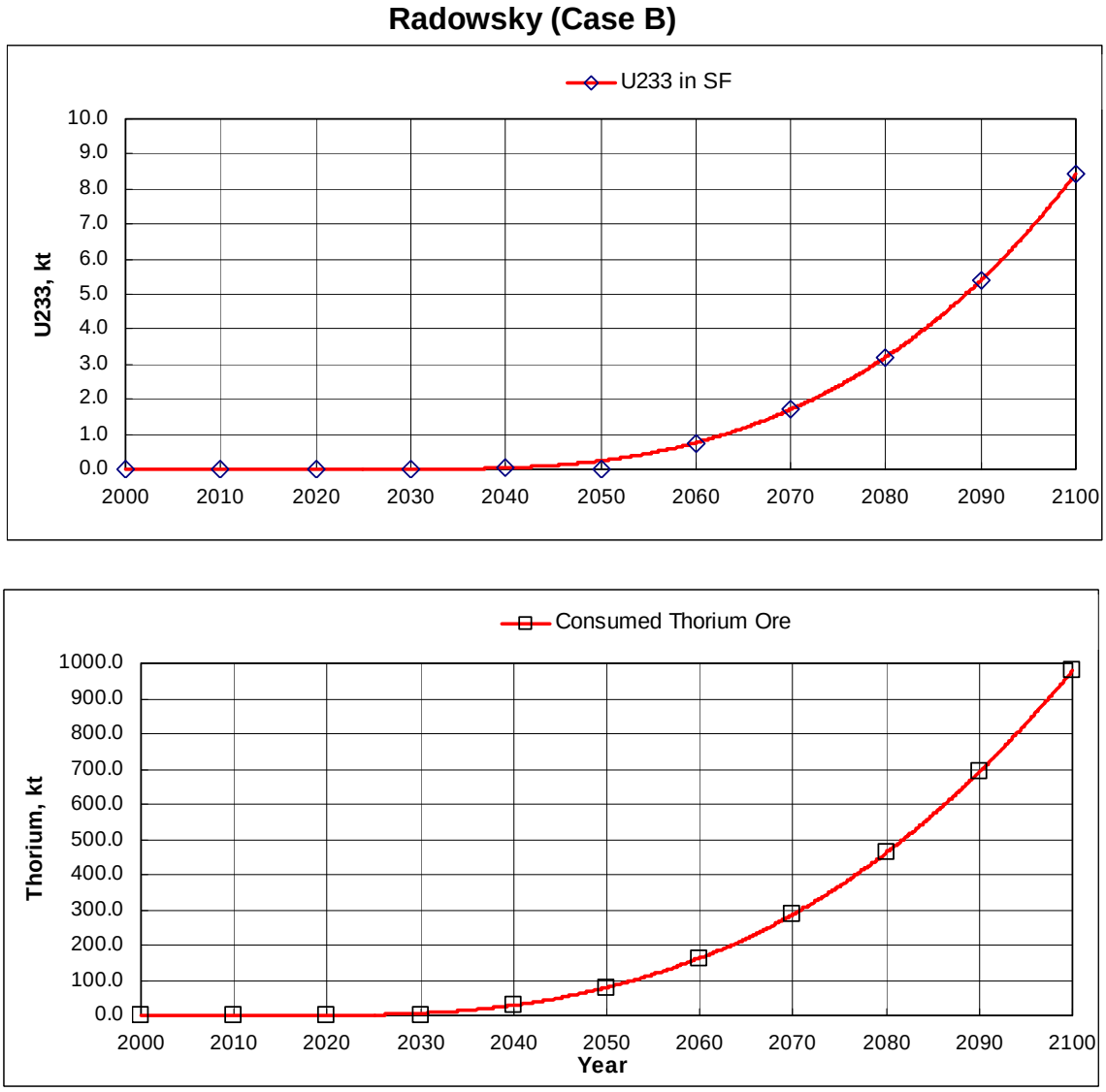
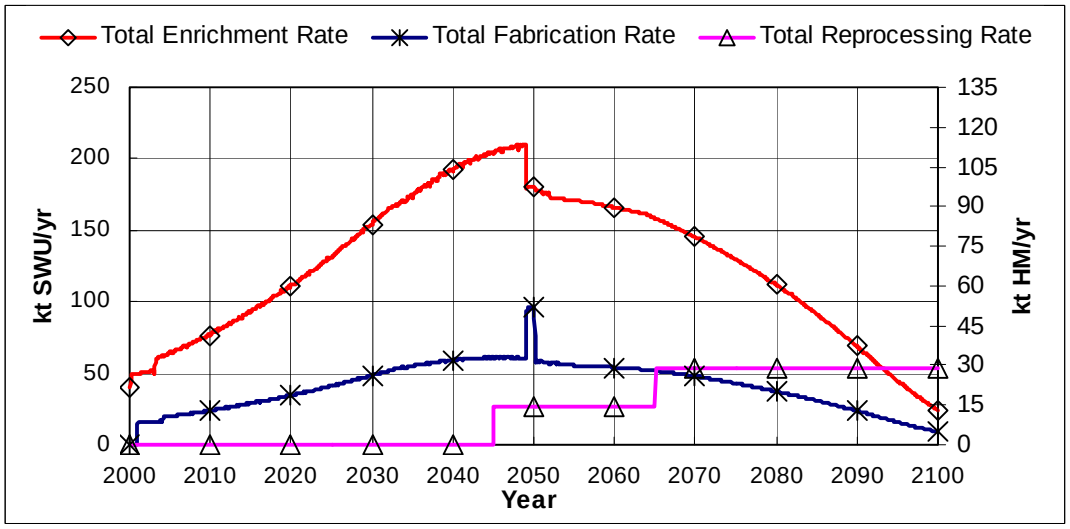
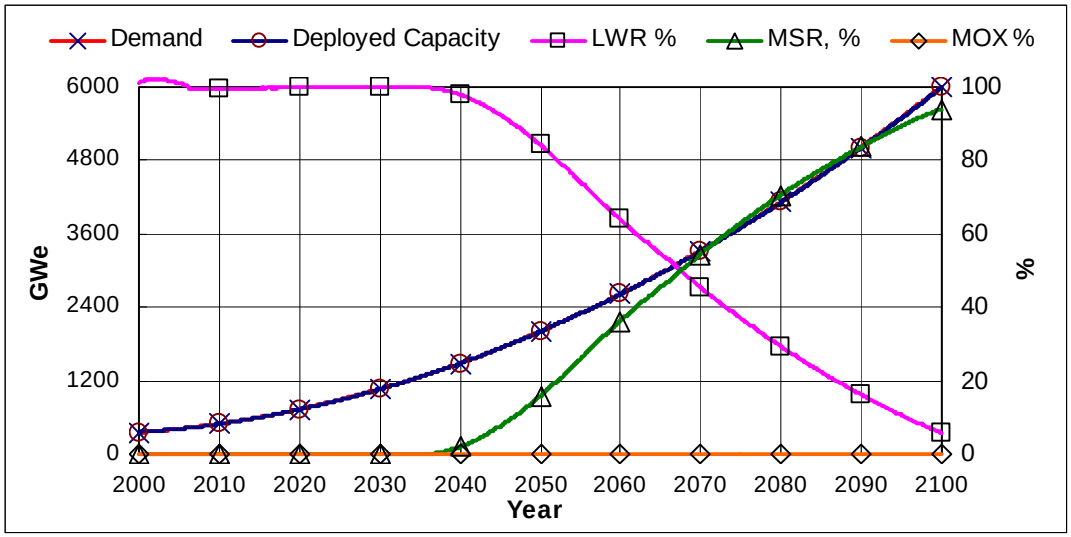

Radowsky (Case B)
U233 in SF
U233, kt
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Consumed Thorium Ore
Thorium, kt
1000.0
900.0
800.0
700.0
600.0
500.0
400.0
300.0
200.0
100.0
0.0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Fig. 3.36

MSR (Case B)





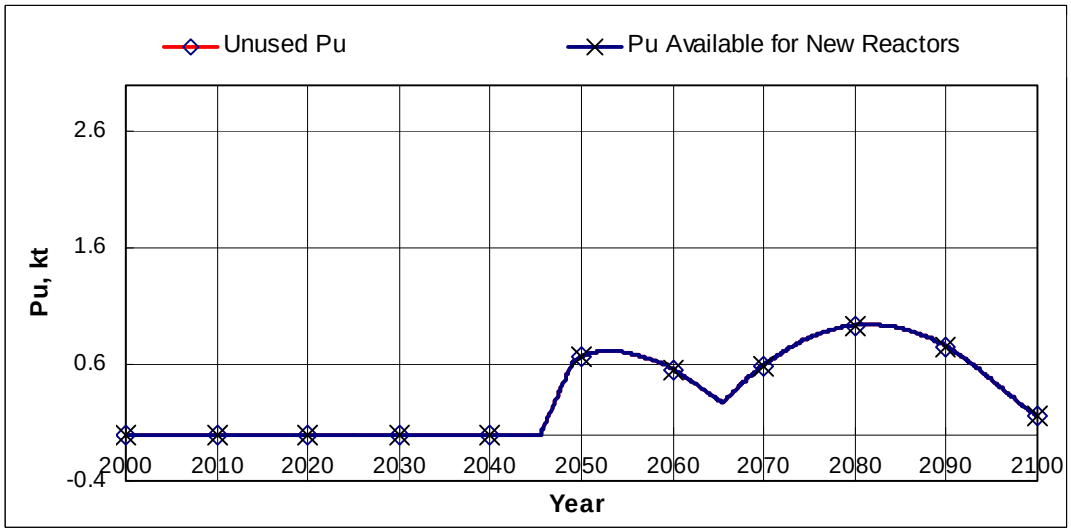
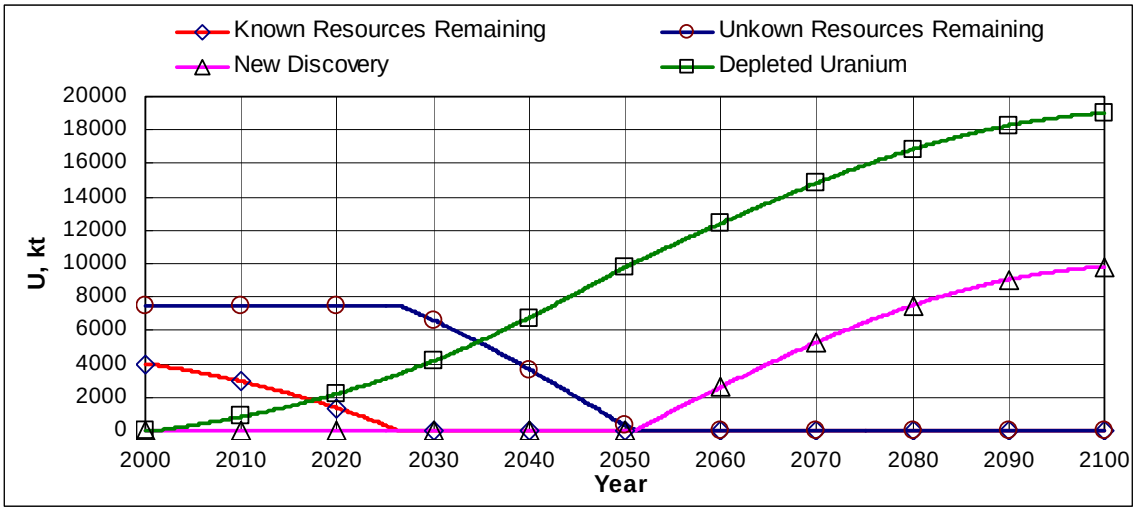
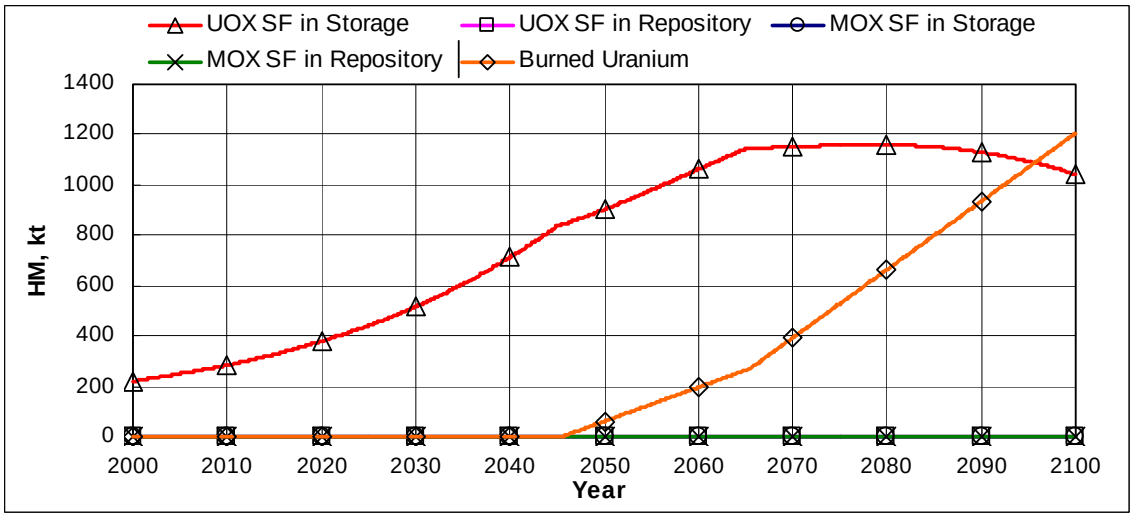


Fig. 3.37

MSR (Case B)





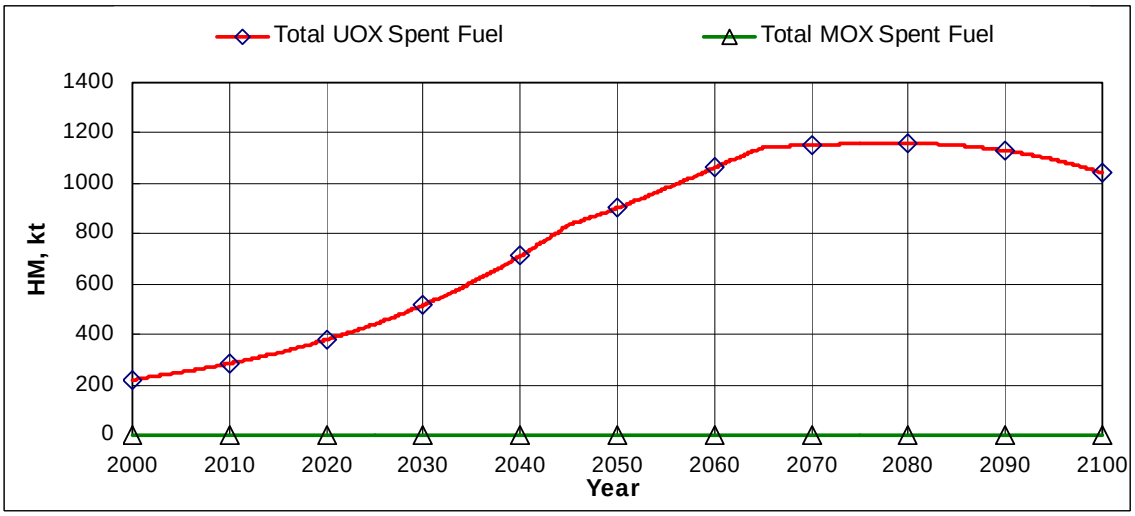


Fig. 3.38

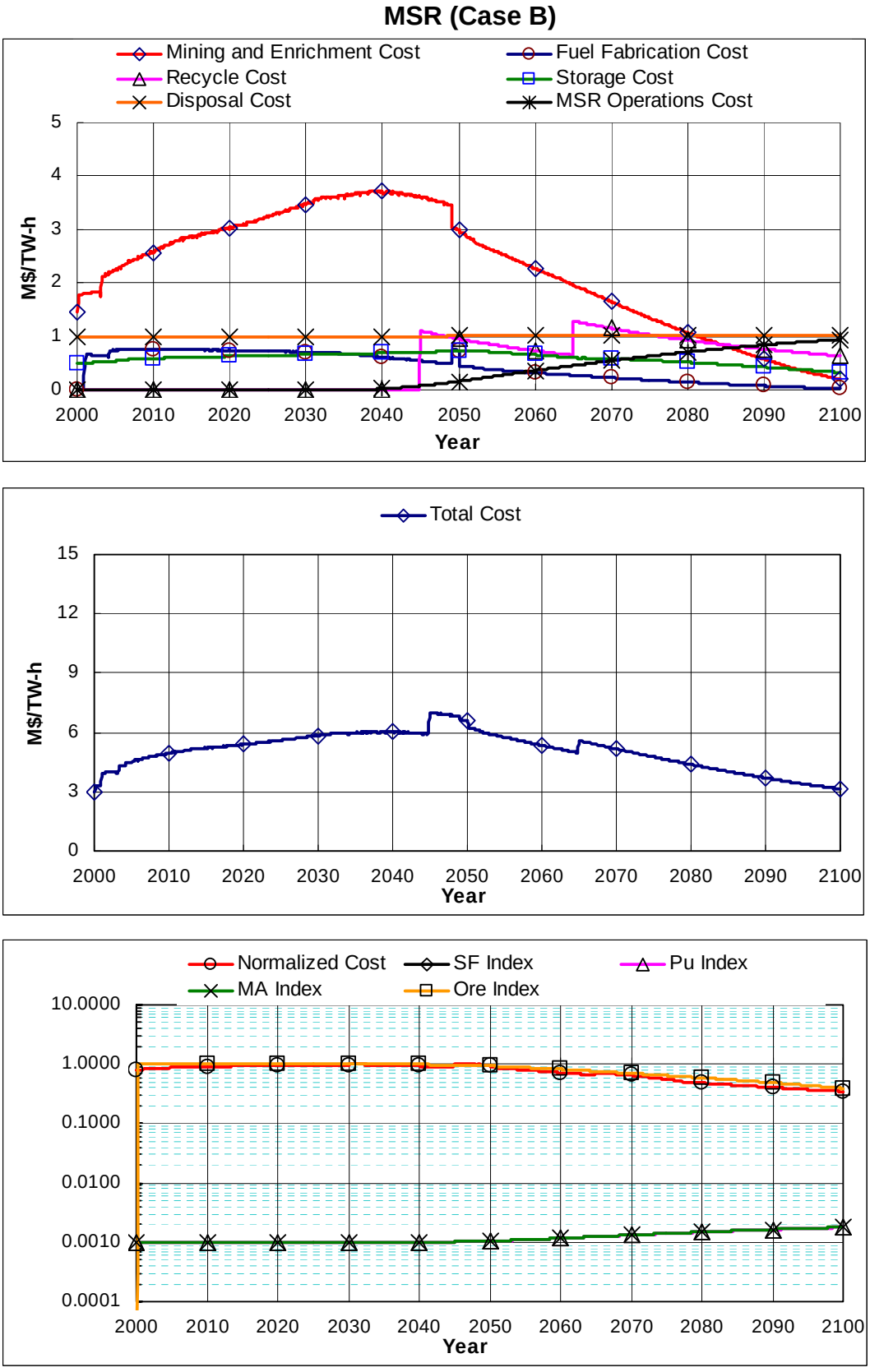

MSR (Case B)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
MSR Operations Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. 3.39

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 4

Fuel Cycle Technology Status and Needed R&D

March 18, 2002

"This document has not been patent-cleared and has not received DOE approval for external release. Do not distribute beyond the Generation IV Roadmap Project Participants"

Chapter 4

Table of Contents

Chapter 4
Fuel Cycle Technology Status and Needed R&D

4.1 Introduction

This chapter summarizes a survey of the status of existing technology for the several links in the fuel cycle chain and indicates directions of R&D ongoing and/or needed for the future fuel cycles contemplated as candidates for Generation IV. A detailed description of the technologies considered in the survey appears in a separate report. This chapter summarizes the information that appears in the report, with much of the text here being a simple condensation of the more detailed presentation presented there. Where possible, specific statements regarding R&D needs, drawn from the many submissions compiled in the report, are presented; however, the presentation of R&D needs in this chapter should not be considered exhaustive.

As shown in Figure 4.1 the links in the fuel cycle can be distinguished between two groups. First are those links at the front and back end which are shared by most existing and contemplated Generation IV fuel cycles (e.g., mining, milling, enrichment, conversion at the front end and storage, shipping, and disposal at the back end). Second are those links (fuel fabrication, fuel irradiation, reprocessing, fuel refabrication, and waste form production) which are concept specific.

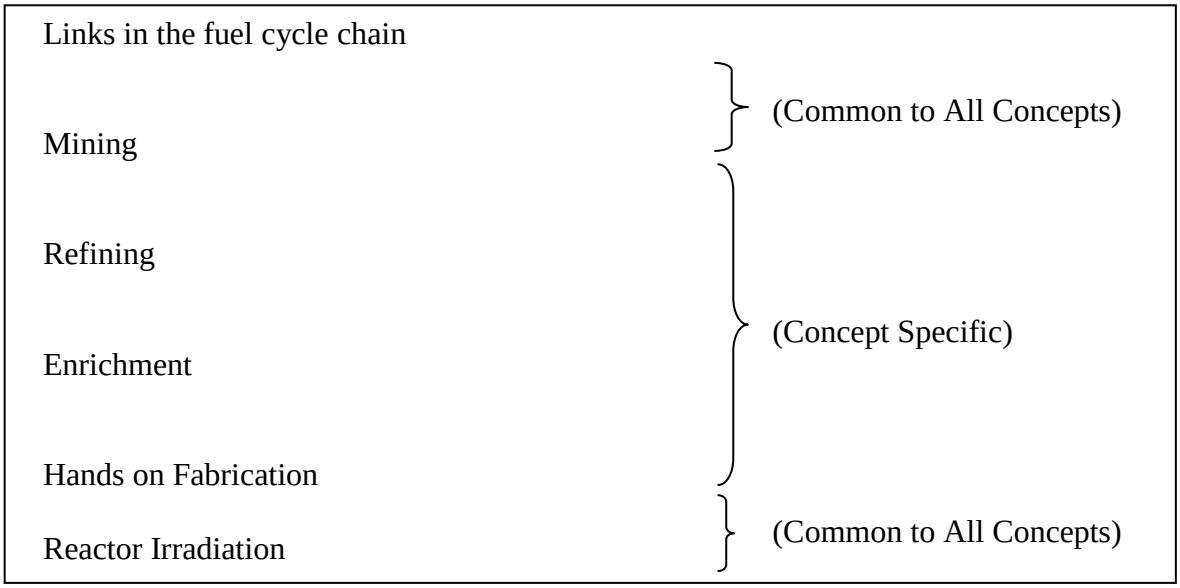


Fig. 4.1 Links in the Fuel Cycle Chain

The information which is summarized in this chapter, is organized by these links in the fuel cycle chain. The technological status and future R&D is discussed for the front end links in the fuel cycle chain which are common to most concepts. Then, concept-specific links are discussed in the order: Thermal Reactor Fuels Technology (Fabrication, Irradiation Performance); Fast Reactor Fuels Technology and Reprocessing Technology

(Separations, Waste Form Production, Waste Form Performance); and, as appropriate, Refabrication of Recycle Feedstock. The Waste Management Technologies are addressed in a non-specific manner. Finally, the thorium-specific fuel cycle technologies are addressed together, to allow consideration of those technologies in the context of the more-familiar uranium-plutonium fuel cycle technologies discussed in the previous sections of the chapter.

4.2 Concept-Independent Front End Links in the Fuel Cycle

As mentioned in the introduction, technologies for the front-end of the fuel cycle are developed and used independent of the specific nuclear energy technology. All Generation IV technologies that use uranium in the fuel cycle will require mining, milling, and extraction of uranium from minerals in which it is found in the earth. However, not all technologies will require uranium enrichment. Current technology development for these activities is directed toward more economic enrichment of ^{235}U. This is motivated by the larger economic investment in currently-deployed light water reactors and the once-through LWR fuel cycle.

Specific enrichment technologies include methods for chemical exchange separation of isotopes under development in Japan and France. The principle of chemical exchange is based on the oxidation/reduction reaction between UO_2^{2+} and U^{4+} and the use of oxidation-adsorption/reduction–desorption reactions on the surface of an ion exchange resin. During these reactions, ^{235}U has a tendency to be slightly oxidized (due to formation of U(VI) or UO_2^{2+}) and is adsorbed. Aqueous chemistry techniques can thus be applied to affect separation of isotopes. Laser isotope separation is another technology that offers promise of improved economics. However, work on laser isotope separation has virtually stopped in the U.S.

4.3 Concept-Specific Thermal Reactor Fuel Fabrication and Irradiation Performance Technology

Fuels used in light water reactors are typically uranium dioxide, UO_2, or mixed oxide, $(U,Pu)O_2$, pellets clad in zirconium alloys. Fabrication and irradiation performance technologies for each type of fuel are well established, to varying degrees of commercialization in varying countries. Technology development for LWR fuel, therefore, is typically driven by desires for incremental increases in economics or in performance. Fuel technology for gas-cooled reactors has been demonstrated, but has not yet been commercialized.

4.3.1 Uranium Dioxide Fuel in LWRs

Modern fuel designs and cladding materials appear to be capable of peak assembly burnup values of greater than 60 GWd/MTU. The degree to which those burnup values are actually attained depends, in part, on the economic factors that might motivate such utilization and the extent to which local regulators are satisfied that safety issues have been addressed. One set of issues that are currently being addressed with regulators are those associated with zirconium alloy-clad UO_2 behavior during reactivity-initiated accidents (RIAs); these issues are being addressed in international programs being performed by industry in collaboration with regulating agencies.

4.3.2 Mixed Oxide Fuel in LWRs

Since the early development of civil nuclear energy, the recovery of plutonium from spent fuels and its recycling in reactors in the form of plutonium fuel, and particularly mixed oxide (MOX) fuel, has been considered by several countries. Initially, MOX was studied and utilized in fast breeder reactors (FBRs), but because development of FBRs ceased in the 1990s, MOX fuel use in thermal reactors (mainly LWR's) was undertaken by some countries as an alternative back-end option. For these countries, plutonium recycle with LWR's has evolved to an industrial level, gaining high maturity through the incubation period during which FBR deployment was originally envisaged.

Within this framework, large R&D programs with plutonium fuel have already been conducted over the past decades. However, further improvement of overall MOX fuel performance is still necessary. A comprehensive examination of R&D needs must address not only topics linked to the interface with nuclear reactors, but also problems set by the management of the used fuel, which depend on specific back-end options, and which may introduce additional constraints to the fuel design. Because France is a leading country in this field, its ongoing R&D work is taken as a basis for an overview of R&D perspectives on plutonium fuel performance and technologies.

R&D Required for Generation IV

The major portion of R&D effort with plutonium bearing fuels in France is devoted to the improvement of MOX fuel performance, to achieve a higher burnup as well as an increase of reliability. This includes R&D toward a better understanding of fundamental phenomena that determine fuel behavior under irradiation and under transient or accident conditions. The program comprises experimental studies (irradiation and laboratory examination and tests) as well as theoretical development (modeling of fuel performance). Major topics addressed in these programs are:

- Fission gas release, which is governed by several phenomena and which particularly depends on the microstructure of the fuel matrix
- Evolution of mechanical properties under irradiation such as fuel rod growth (swelling) and creep of pellets
- Behavior under transient or accident conditions (primarily addressing issues associated with reactivity initiated accidents or loss of coolant accidents)

In addition, further R&D is necessary (and is ongoing in France) to obtain better computer codes able to run more accurate calculation of reactor cores containing plutonium fuels. To this end, nuclear data evaluations (i.e., determination of cross sections of plutonium and minor actinides) and development of calculation methods through international benchmarking are underway.

Another important subject is the burn-up credit (applicable to criticality control limits), which may have significant impact on economy and performance at all stages of irradiated fuel processing (e.g., storage, transportation, reprocessing, etc.). Continued, or even enhanced, R&D efforts must be devoted to achieving better knowledge in this domain.

4.3.3 Coated Particle Fuels for Gas Reactors

The development of the High-Temperature Reactor (HTR) has proceeded in two directions: a) the pebble bed concept pursued in the Federal Republic of Germany and Russia (now also in China and South Africa), and b) the prismatic core pursued in the United States, the United Kingdom, Japan and, recently with the GT-MHR, also Russia. Although fuel elements in the two HTR designs differ substantially, the basic fuel-containing unit, the coated particle is essentially the same, and coated particle fuel development has proceeded as an international effort quite independent of differences in reactor design.

The fuel elements for the pebble bed system consist of 60-mm diameter spheres made up of a fuel-free carbon outer zone and an inner fuelled region with coated particles uniformly dispersed in a graphitic matrix. The prismatic fuel element consists of a machined hexagonal graphite block ~750 mm long and 350 mm across flats. Alternate fuel and coolant holes are drilled in a hexagonal array. Fuel rods, consisting of coated

particles bonded in a close-packed array by a carbonaceous matrix, are stacked in the fuel holes in the prismatic blocks.

The two coated particle types most commonly used are:

- The BISO coating with a porous buffer and dense pyrocarbon (PyC) outer shell, and
- TRISO coating with its intermediate of SiC between two layers of high-density isotropic PyC.

Performance of a coated particle for HTRs relates to the ability of the fuel to retain fission products. Both BISO and TRISO particles are capable of nearly complete retention of gaseous fission products and iodine. Improvements in coated particle fuel technology are necessary to meet criteria for fabrication, irradiation performance, and safety-related behaviour.

R&D Required for Generation IV

The present coated particle design has not yet been optimised for the combination of high burnup and high temperature resistance under accident conditions that are associated with small inherently safe modular HTRs. The TRISO fuel design currently available resulted from fuel development for large HTRs with gas turbines before the advent of the modular concept. Thus, there is still a huge potential for improvement of coated particles, particularly if plutonium or actinide burning is also considered.

To develop the coated particle for the proposed modular applications, available results should be re-analyzed and re-investigated in the light of new requirements envisioned. Specific work to be performed includes the following:

- Re-evaluate ^{110m}Ag release data for normal operating conditions to derive source term data applicable for direct cycle applications.
- Determine the influence of burnup effects on irradiation performance at burnup values above 10% FIMA, in particular for potential reduction of 1600°C temperature capability for retaining fission products.
- Analyze accident condition performance at temperature above 1600°C for an improved coated particle modelling of accident response behavior.
- Investigate optimization of the coated particle design (e.g., ZrC could be a substitute for SiC, and a smaller kernel, a thicker buffer layer, and a thicker SiC layer could be utilized).
- Model the heating test results to aid the adaptation of the particle design to the needs of modular reactors.
- Investigate leach and corrosion behaviour of irradiated coated particles in geologic disposal conditions using high precision measurements.

4.3.4 Advanced Fuels and Inert Matrix Fuels

While well mastered and efficient recycling of plutonium in LWR with present MOX fuels is limited by inherent features that do not allow a higher degree of plutonium recycling. This is mainly due to the safety-related limit of 12.5% plutonium content in the fuel. Another limitation is the proportion of MOX fuel that can be accommodated in present PWR cores (for example, 30% in today's French 900 MWe PWRs). With these constraints (and because of some additional reasons) it is difficult to recycle the plutonium several times in large quantities in PWR's.

To address those issues, innovative fuels are being investigated. Such fuels include nitride, metal or inert matrix options that may be combined with alternative fuel assembly geometries. To accomplish a strategy to quickly reduce plutonium interim storage inventories, removal of uranium from the fuel form allows the maximum possible plutonium net consumption rate by eliminating conversion of fertile U-238 to new plutonium in situ. A typical example of research with such fuel may be found in the "APA" French concept (Advanced Plutonium fuel Assembly), which is a uranium-free plutonium fuel for which various assembly geometries are under study (including annular fuel pins). Preliminary results indicate that these new fuels present effective opportunity to increase plutonium consumption in LWRs.

In addition to fuel fabrication and irradiation performance considerations, the development of innovative fuels is intended to aid the multiple recycling of plutonium.

4.4 Concept-Specific Fast Reactor Fuels Technology

An extensive experience base exists for fast reactor fuels based on enriched uranium. Examples of such experience include oxide fuel in commercial service in BN600 and (until recently) in BN350 and metal alloy fuel in service in the experimental EBR-II power plant. Some experience in uranium nitride fuel for the SP-100 space reactor also exists. This experience base is directly applicable for the blanket fuel pins in breeder reactors and is the foundation upon which plutonium-bearing fast reactor fuels have been built.

Because the uranium-based breeder fuels are commercially established, and because fast reactors are considered for Generation IV due to the sustainability derived from their capability to either burn excess plutonium or to create a future fuel supply based on breeding, this discussion is confined to addressing mixed plutonium- and uranium-bearing fast reactor fuels.

4.4.1 Oxide Fuels

The development of mixed oxide fuel ($PuO_2 – UO_2$) was the cornerstone of Liquid Metal Reactor (LMR) Programs around the world for over twenty years. This development culminated with the demonstration of high-burnup mixed oxide cores in reactors in the United States, France, Japan, and United Kingdom. Mixed oxide was selected for this extensive development because of the excellent burnup potential of the fuel system, the relative ease of commercial fabrication, and the proven safety response by virtue of the Doppler effect.

Later, fuel development in Europe was carried out under a joint European program. This collaboration between France, Great Britain, and Germany brought together considerable experience covering a wide range of fuels. The economic incentive for lower fuel cycle costs produced a continual improvement in mixed oxide fuel, and burnup progressively increased. This was part of an international quest to constantly improve performance while ensuring safety and minimizing fuel cycle cost. Higher burnup performance was enabled by the development of advanced stainless steel alloys, which mitigated problems with irradiation-induced void swelling and irradiation-assisted creep.

Fabrication technologies include pelletizing using the MIMAS process or advanced fabrication processes such as those under development in Japan. Vibro-compaction (or vi-pac) technology is under development in Russia and recently in Japan; it offers the promise for simpler fabrication of oxide fuel.

R&D Required for Generation IV for Mixed Oxide Fuel Performance

Given the extensive experience in several countries with mixed oxide fuel over the past twenty years, there are few technical issues that impede its deployment in sodium-

cooled systems. On the other hand, there is an economic incentive for increased burnup, with an associated incentive for a better understanding of low-swelling alloys for duct and cladding materials. For example, oxide dispersion-strengthened stainless steels are being developed as cladding materials by the JNC in Japan, with the objective of enabling high burnup, 150 GWd/t, in fast reactors. And finally, if mixed oxide fuel is to be used with either lead or lead-bismuth coolants, there may be compatibility questions to be understood and resolved.

R&D Required for Generation IV for the Simplified Pelletizing Method

The following fundamental technologies are being investigated and developed to determine the technical prospects of the Simplified Pelletizing Method.

- Techniques for adjusting plutonium content during the mixing stage of uranium and plutonium nitrate solutions in a reprocessing plant,
- Techniques to enhance powder flow techniques, for example by controlling the temperature during calcination/reduction,
- Pellet-pressing equipment with die-wall lubrication, and
- Pneumatic powder transport systems, including the accountability system for nuclear materials.

In addition, in-cell application of the Simplified Pelletizing Method for MOX pellet fabrication requires development of the following technologies:

- Remote maintenance,
- Handling of low-decontaminated TRU fuel, including decay heat removal measures, and
- A turntable type denitration/calcinations/reduction system.

Eliminating the powder blending and granulation step that is part of conventional MOX pellet processes, will require that the powder preparation process of the Simplified Pelletizing Method be more reliable than the conventional process, which affects the process throughput rate.

R&D Required for Generation IV for Vi-pac Technology

At PSI in Switzerland, research on vibration-compaction with spherical particles derived using the internal gelation method (termed “sphere-pack” fuel fabrication) is being performed. In Japan, the JNC started research on gelation in 1990, and is presently proceeding to the collaborative study on fuel fabrication with PSI. At RIAR in Russia, irregular-shaped vipac fuel fabrication by a pyroelectrochemical method and its irradiation in BOR-60 is being continued.

With regard to the gelation method, the establishment of the optimum condition of gelation and treatment of waste solution remains to be solved, and more efficient granulation methods are to be developed. In the case of low-decontaminated and TRU fuel, it will be necessary to show the applicability of gelation to the multi-component systems. Regarding the Oxide Electrowinning method, the Pu enrichment distribution throughout the fabrication process should be monitored with quality control.

Development tasks that are common to fabrication using feedstock from both aqueous and non-aqueous reprocessing are optimization of vibration conditions in order to attain high-density fuel, a non-destructive inspection method for low decontaminated fuel, and irradiation experiments to confirm good irradiation performance.

4.4.2 Metal Fuels

Metal fuel was the first fuel used in fast reactors. The simple fabrication technologies for metal and metal alloys, the high thermal conductivity, and the relatively high fissile density all made metal fuel attractive to early reactor designers. The early metal fuel designs were not capable of achieving high burnup nor were they capable of performing at the high sodium-coolant outlet temperatures contemplated in the design of future fast reactors. Therefore, large scale development of metal fuels was discontinued in the late 1960's in favor of ceramic fuels. However, Experimental Breeder Reactor II (EBR-II) continued to further develop and operate with metal driver fuel; and this reactor was used as the test bed for all fast reactor fuels and materials irradiation testing through the 1970's and 1980's.

A simple design change was discovered that allowed metal fuel to attain high burnup. Subsequently, the high reactor outlet temperatures in the early designs were found to be too aggressive for the primary system structural materials. In addition, it was discovered that a high thermal conductivity and high density of metal-fuel comprised a set of enabling technologies to achieve inherent safety features that could not be duplicated with mixed oxide fuel. Additionally, metal fuel lends itself to inexpensive remote reprocessing and refabrication by the non-aqueous pyro-metallurgical process. As a result metal fuel again become a viable alternative to ceramic fuel by the mid 1980's.

The remote fabrication of metal alloy fuel proposed for Gen-IV concepts uses proven technology employed to fabricate cold fuel for EBR-II. In the 1960s this technology had already been employed for remote injection casting fabrication of over 34,500 fuel elements.

R&D Required for Generation IV

Knowledge of metal fuel is sufficiently mature that only limited and focused further R&D is required to support licensing of the fuel system. The performance of the fuel with increased minor actinide content, especially at high concentrations and with americium in particular, must be demonstrated with further testing. In addition, building

quality fuel in a hot cell environment must be demonstrated to prove the recycling concepts of metal fuel. Irradiation of reprocessed U-Pu-Zr fuel is therefore needed to demonstrate comparable performance to cold-line fabricated fuel. Irradiation of a full core of prototypic-length metal fuel would establish neutronics effects and inherent safety features of metal fuel.

Finally, the completion of an accurate, phenomenologically-based fuel performance code, benchmarked by the current database and further data gained from the remaining tests required, is needed to aid in licensing fuel for a fast reactor prototype.

4.4.3 Nitride Fuels

The state of development of nitride fuel is modest relative to that of mixed oxide or metal fuel. Nitride fuel is attractive for two reasons. First, it exhibits many of the desirable characteristics of metal fuel; (i.e., high heavy metal density, good thermal conductivity, and excellent compatibility with sodium and lead-based coolant), while at the same time having a high melting temperature characteristic of ceramic fuels. Second, testing of nitride fuel for space power applications did not reveal any undesirable characteristics. However, the amount of testing to date is small compared to that of either oxide or metal.

For fast reactor application, it is envisioned to enrich the nitrogen used for nitride fuel in N^{15}. There are two reasons – first to avoid the production of the extremely long lived radioactive activation product, C^{14}. The second reason is to mitigate the reactivity effects on neutron economy of the parasitic capture on N^{14}.

R&D Required for Generation IV

The R&D needs for nitride fuel are as follows:

- The fabrication experience with vibropacking is limited and needs further development,
- A low-cost technology to enrich the N^{15} component is needed to improve the economics of the fuel cycle,
- Irradiation testing is quite limited and often not well documented,
- Phenomenological characteristics affecting basic fuel design such as swelling, fission gas release, fuel-cladding chemical interaction, and thermal dissociation are not well known at higher burnups.

A considerable amount of research and development will be required to bring the status of nitride fuel up to that of either metal or mixed oxide fuel. But nitride fuel appears to have unique safety characteristics and its power margin to fuel melting is superior to that of either mixed oxide fuel or metal fuel. Thus for selected high temperature applications nitride fuel appears to hold unique potential, and irradiation testing results have been positive.

4.5 Concept-Specific Recycle Technologies and Waste Form Production

4.5.1 Recycle Overview

Two main types of processes can be applied to the separation of long-lived radionuclides: hydrometallurgical ("aqueous") and pyrochemical ("dry") processes. The industrial hydrometallurgycal reprocessing technique is used in the PUREX process to separate U, Pu and eventually Np (in a modified PUREX process) from dissolved spent fuel. For the extraction of minor actinides the process has to be extended; i.e. additional extraction steps follow the standard process. Extensive research is being carried out at present in this field mainly toward the extraction of americium and curium, including the separation of minor actinides from the lanthanides, which are generally co-extracted due to very similar chemical properties. An alternative to hydrochemical processes are pyrochemical processes in which refining is carried out in molten salt media, based on electrorefining or on distribution between non-miscible molten salt-metal phases.

The oxide fuels used world-wide in thermal reactor systems for energy production are easily reprocessed by aqueous techniques; these systems, primarily the PUREX process, are fully developed for full fissile (plutonium) recycle and are implemented commercially. Pyroprocess systems have largely been associated with fast reactors and metallic fuels and their development has reached only the pilot-scale stage, and the feasibility of transuranic recovery still needs to be demonstrated.

The major advantages of pyrochemical methods for reprocessing of advanced fuels, in comparison to hydrochemical techniques, are a greater compactness of equipment and less dependence on economy of scale – providing the possibility to form an integrated facility complex for irradiation, reprocessing, and (re)fabrication; thus considerably reducing the transport of nuclear materials. Pyrochemistry is to be preferred in particular for “advanced” oxide fuels (i.e., transuranic, inert matrix or composite) and metal fuels, as well as nitride fuels. Compared with aqueous methods, dry reprocessing of fuels results in “dirty” (less pure) and potentially more proliferation-resistant compositions of Pu, Np or Am. In addition, the radiation stability of the salt in the pyrochemical process compared to the organic solvent in the hydrochemical process offers an important advantage when dealing with highly active spent minor actinide (MA) fuel; this allows reprocessing after shorter cooling times, which reduces storage costs and out of reactor inventory (which is a favorable feature to reduce breeding doubling time.) Furthermore, due to the absence of water (neutron moderator) in the process the criticality hazard is lower.

Finally, it should be noted that reprocessing technology development is often directed toward meeting two important objectives consistent with Generation IV goals: increasing the proliferation resistance of the overall fuel cycle, and reducing the long-term radiotoxicity of wastes. This motivates use of technologies that produce “dirty fuel” and “clean waste”. Dirty fuel refers to fuel compositions with some amount of residual fission products or retained minor actinides, all of which are allowed into the fuel to make the

fissile material increasingly unattractive for diversion to weapons uses, and to provide for additional recycle for burnup of transuranic materials that would otherwise be disposed in waste streams. Clean wastes refers to those with reduced or miniscule amounts of long-lived radiotoxic constituents, such as minor actinides or certain fission products, which would be eliminated through means such as fission or transmutation, to the end of reducing long-term radiotoxicity.

4.5.2 Aqueous Processing Methods

The PUREX process is the most important hydrochemical technique to separate U and Pu from spent fuel containing natural, slightly or highly enriched uranium. It is employed at commercial scale in the spent fuel reprocessing industry and is the reference process for LWR/UOX and LWR/MOX reprocessing. Over the past decade substantial R&D has been directed to the recovery of minor actinides (MAs) in aqueous processing schemes. The European programs distinguish different options in the PUREX process which achieve the partitioning of MAs (Np, Pu, Am and Cm): they are the PUREX process adapted for Np recovery and the European Extended PUREX process for MA recovery with its three alternative approaches. In Japan, different institutions are studying several alternative techniques such as the JAERI Four Group Partitioning Process for recovery of MAs and selected fission products and the JNC Advanced Aqueous Process. These advanced processes all allow selected extraction of various MAs from solutions of dissolved spent fuel, but also represent attempts to advance the PUREX standard to improve economics of reprocessing and provide reductions in secondary waste streams.

R&D Required for Generation IV for Aqueous Processing

Table 4.1 gives a brief overview of the status of R&D on the various aqueous partitioning techniques, as was reported in the OECD-NEA first-phase partitioning and transmutation systems study "Status and Assessment Report of Actinide and Fission Product Partitioning and Transmutation". Three phases can be distinguished:

- Phase 1 corresponds to research on the principles of the process, which will establish the scientific feasibility of the process. In many cases, it overlaps the basic research conducted in the laboratory (for example, research on new extractant compounds).
- Phase 2 is the process development step. It includes all research designed to develop the complete flow chart, describe its application, and confirm its performance. The conclusion of this step corresponds to establishing the technical feasibility of the process.
- Phase 3 relates to the industrialization of the process, terminating with the establishment of industrial feasibility of the process and its potential application in an industrial installation. It is aimed at ensuring overall effective operations in industrial conditions.

For all the above-mentioned processes the development of new extractant molecules and the improvement of existing ones are carried out world-wide. Especially for the most challenging Ln/An separation, the work on diphosphines in Russia, the research on dithiophosphinic acid derivates in China and Germany, the improvement of TPTZ and BTP derivates in France, and the examination of sulfoxide type extractants in India have to be mentioned.

Table 4.1 Status of R&D on Aqueous Separation Techniques

	Phase 1	**Phase 2**	**Phase 3**	**Remarks**
U and Pu separation (PUREX)	X	X	x	Achieved industrially
Np separation (PUREX)			x	95% separation
(PUREX)		x		>95% separation
(DIDPA)		x		
(HDEHP)		x		
(TRUEX)		x		
Am + Cm separation:				
Based on An/Ln co-extraction				
(TALKSPEAK)			x	
(DIDPA)		x		
(TRUEX)		x		
(TRPO)		x		
(DIAMEX)		x		
Based on An selective extraction				
(TPTZ)	x			
(Picolinamides)	x			
(CYANEX 301)	X			SF=5 900
(BTP)				
based on precipitation				
(Ferricyanide)	X			
Am separation in the oxidized state (SESAME)		X		Am/Cm separation
Tc separation (PUREX)			x	soluble Tc
(PUREX)	X			insoluble Tc
Tc – PGM separation				
(Denitration precipitation)		x		
(Active carbon adsorption)		x		
I separation (PUREX)			x	95% separation
Zr separation (PUREX)		X		
Cs separation (Calixarenes)		X		
(Zeolite)		x		
Sr separation (Titanic acid)		X		
Cs and Sr separation (Dicarbollides)			x	
Pd (PGM), Se, Ru separation (Electrolytic extraction)	X			soluble Pd, Se, etc.

In view of process industrialisation, the economics and the radiation resistance of the organic molecules are important R&D issues. The aim is to develop sustainable, environmentally friendly processes. A direct selective extraction of An from the PUREX raffinate would reduce the number of process steps, and the pre-concentration of the raffinate could reduce by a factor of ~10 the volumes of liquid to be handled. Low-cost, efficient, robust and simple processes with either well-defined technologies (pulsed columns, mixer settlers and centrifugal contactors) or new technologies such as hollow fibre modules offer prospect for improvement. In all cases it is important to keep a good balance between fundamental chemical research, process development, and qualification using actual high activity wastes originated from real spent nuclear fuels.

R&D Required for Generation IV for the JNC Advanced Aqueous Process

The future R&D tasks necessary to develop advanced aqueous reprocessing technology that recycles TRUs in the closed fuel are the following:

- compacting head-end equipment,
- enhancing and optimizing equipment (corrosion resistance, extraction function, etc.), and
- enhancement of MA recovery process effectiveness to reduce waste

Furthermore, the safeguards optimization of blanket fuel management in the fast reactor must be addressed.

4.5.3 Pyrochemical Processing Methods

Since early in the development of the nuclear industry, a number of alternative pyrochemical processes to separate actinides from spent fuel have been investigated. The processes studied may be grouped into the following general categories: melt refining, zone melting, electrorefining, vacuum distillation, fractional crystallization, gas-solid reactions, and liquid-liquid extraction using either non-miscible molten metal phases or non-miscible molten salt-metal phases. Pyrochemical separations often rely on electrorefining or electrowinning techniques, in which fuel is preferentially separated or electrolytically dissolved from cladding into molten salts, and then recovered on some type of electrode.

Over the last couple of decades, pyrochemical recycle techniques have been studied, developed and tested (in some cases at a pilot scale) throughout the world; mainly in the US, Russia and Japan. Programs in the U.S. have sought to apply pyrochemical processing primarily to the recycle of fast reactor metal fuel and to the incorporation of spent LWR fuel as feedstock into the fast reactor fuel cycle. In contrast, programs in Russia have applied pyrochemical techniques to the preparation and recycle of oxide fuel

for fast reactor application. Programs in Japan are broader, addressing recycle of metal, nitride, and oxide fuels and the incorporation of spent LWR fuel as feedstock into those fuel cycles. The production of waste forms suitable for repository disposition has also been part of these programs, with the U.S. program perhaps having moved the furthest toward that end.

During the 1990s, new concepts for the transmutation of fission products and the associated fuel cycles provided renewed impetus for further investigation of these processes. New matrices are being designed for the elements to be transmuted, including heterogeneous recycling targets as well as dedicated fuels; longer-term perspectives also include new fuels for homogenous recycling concepts. Most of the scenarios considered require multiple recycling of targets or fuel to obtain high transmutation rates and thus a sufficient reduction in radiotoxicity. The far future options, such as dedicated fuel cycles and transmutation in molten salt reactors provide further motivation for the exploration of these processes. This is obvious in the case of molten salt reactors, for which pyrochemical techniques are the natural, and perhaps the only, possible reprocessing solution.

R&D Required for Generation IV for Pyroprocess Development.

The Spent Fuel Treatment Program at ANL has already demonstrated many parts of the pyroprocess recycle technology, but there are still key aspects that have yet to be demonstrated on a large scale with radioactive materials. The main outstanding issue is the recovery of transuranics. Large-scale equipment has been fabricated for transuranic recovery, but with the termination of the IFR program, the equipment and process was never tested beyond the laboratory scale.

Another challenge for a pyroprocessing system is selecting the appropriate materials of construction for the high temperature fuel cycle processes. Material improvements are needed in order to lessen the formation of dross streams to increase material recovery and throughput, and to minimize waste generation from process equipment degradation.

The quantity of high level waste generated from pyroprocessing requiring geological disposal appears to be comparable at present to modern commercial aqueous processes. Advancements are being pursued to further reduce the disposal volumes through zeolite ion exchange processes. This technology has not been demonstrated beyond the laboratory scale.

Most of the pyroprocessing work performed with radioactive materials to date has been for metal fuel. Laboratory work has been performed on the head-end operations for oxide reduction and on the nitride fuel cycle. Demonstrations of these technologies with actual spent fuel are still needed. Additionally for nitride fuels, demonstrating the recycle of nitrogen is critical as ^{15}N is specifically required for the fuel in order to eliminate the formation of radioactive ^{14}C.

R&D Required for Generation IV for Oxide Reduction Head-end Processing to Feed Pyroprocess Operations

Further work is required in a number of areas related to process design and equipment for reducing oxide fuel to a form suitable for pyroprocessing. (Such processes could find use in symbiotic systems where LWR-UOX fuel feeds into fast spectrum closed fuel cycles). Actual irradiated fuel should be used to verify the behavior of the TRUs and fission products. In addition, work remains to understand the fundamentals of the salt-recovery step to provide a basis for construction of more efficient cells and to understand the behavior of fission products in this step. Alternative, lower-cost oxygen-evolving electrodes must be developed for the salt-recovery step.

R&D Required for Generation IV for Oxide Fuel Pyroprocess Development

An alternative dry process retains oxide fuel in the oxide form rather than reducing it to a metal prior to processing. Research activities today (potentiometry, voltammetry, etc.) focus on the behavior of neptunium and americium for the fabrication of oxide fuel containing these elements. In the so-called DOVITA-process developed in Dimitrovgrad, the oxide fuel is converted into chlorides. UO_2 and PuO_2, as well as (U, Pu, Np)O_2, are separated by electrolysis in a melt of NaCl-KCl at 650°C. The transuranium elements are precipitated sequentially as the oxo-chlorides or oxides from the NaCl-KCl melt by gassing with Cl_2/O_2 and adding Na_2CO_3.

Because lanthanides and the transplutonium elements (Am, Cm) have similar behaviour, a fractionated precipitation of the oxychlorides is proposed in order to obtain an Am-Cm fraction with a sufficiently low lanthanide content. However, from the technological point of view, this is a cumbersome step; therefore, it would certainly be preferable to develop an electrorefining process also for Am and Cm.

4.5.4 Other Nonaqueous Processes

Other nonaqueous processes currently being considered include volatility and reductive extraction processes. The most common of the volatility processes applied in spent fuel processing are chloride volatility and fluoride volatility. Such processes can be extremely useful in the processing of complex fuel types, including inert-matrix fuels. Reductive extraction processes exploit certain well-behaved replacement reactions to separate certain fission products and actinides. Molten metal/molten salt systems are particularly useful for application of reductive extraction. An illustrative example is the treatment of chlorinated metallic fuel to separate the actinides from active metal fission products. Reductive extraction processes might find application in conjunction with other pyrochemical separation schemes that do not accomplish sufficient removal of lanthanides from recycled actinides.

4.5.5 Processes for Coated Particle Fuels

Development of HTR fuel reprocessing in Germany and the U.S. was based on reference fuel compositions containing HEU and thorium either as mixed oxides/carbides or with separate fuel and breeding particles embedded in the fuel matrix. In contrast to reprocessed LWR fuel, the extracted uranium will contain mainly U-233, with other U isotopes sufficiently radioactive to require remote fabrication of new fuel from this resource.

The main steps of HTR fuel reprocessing are:

- Head-end separation of graphite and fuel kernel
- Chemical separation of the different mass streams
- Back-end purification of products
- Reconversion of regenerated fissile materials

The development program in Germany included the operation of a development plant that processed more than 10,000 unirradiated fuel pebbles of different types (HEU-BISO, HEU-TRISO, LEU-TRISO, graphite balls) showing a very reliable operation and effective performance. Operations with irradiated materials were not started because the back-end strategy for the HTR was changed to direct disposal instead of reprocessing.

4.5.6 Secondary Wastes

Secondary Wastes arise at every link in the fuel cycle chain. Categorization of the secondary waste associated with fuel cycle operation is difficult due to the uncertainty inherent in any assessment of potential inventories. However, it is possible to suggest that secondary wastes arising from ancillary service materials and secondary liquid effluent treatment (e.g. secondary filter cartridges from steam generator blowdown systems that are contaminated with small quantities of fission products and activation products) would be considered low-level waste (LLW). Sludges and concentrates, contaminated with fission products and activation products with trace quantities of actinides would be categorized as intermediate-level waste (ILW), in countries where such a designation exists. Similarly, material contaminated with activation products, fission products, actinides and neutron-activated products would also be categorized as ILW.

It is generally acknowledged that in addition to the fission products, fuel cladding, etc., that constitute the primary fuel cycle waste from a PUREX recycle scheme, there would be secondary waste. This secondary waste comprises all insoluble active residues, degraded solvents or salts, ancillary materials, and analytical wastes, etc. that arise during additional fuel cycle operation. The majority of secondary wastes are generated during solvent/salt cleanup and recovery operations.

Because there is little or no reagent degradation, pyrochemical processing operations tend to produce little secondary waste. If the processes are properly designed and operated, high-level waste volumes can be minimized by recovery and recycle of salt and metal reagents. There is little published information available on wastes from pyrochemical process operations due to the relative technological immaturity and lack of industrial-scale experience. Therefore, the technology can be assessed only on the basis of extrapolations from laboratory-scale studies, and most of this experience has been with the electrorefining process used for separations of actinides and fission products in the metallic state.

4.5.7 Depleted and Recovered Irradiated Uranium

The management of depleted uranium and recovered irradiated uranium has received relatively little interest over the past years as their environmental impact is very low and, in today's fuel cycles, is overwhelmed by the potential radiological impact from the intermediate and especially high-level waste in the long term. The recycle schemes under consideration for the Gen-IV Roadmap may, however, reduce the amount of long-lived high-level waste by a significant factor; i.e. a factor of hundred or more. Therefore, the management of depleted uranium and recovered irradiated uranium may become a more apparent issue in the future if such recycle schemes are deployed.

Today, the strategy for the long-term management of depleted uranium is based on the consideration that depleted uranium is a valuable material, which may have various future applications, and is not considered a waste. The use of depleted uranium in fast reactor systems is one of the applications; alternately its re-enrichment is a second potentially valuable source of ^{235}U for LWRs, while the remaining ^{238}U may again be used in future fast reactor systems. In the absence of these, or other, large-scale applications, however, final disposition in some form of 'repository' would have to be considered.

Recovered irradiated uranium is currently not systematically recycled in UOX- or MOX-fuel. The reprocessed uranium distinguished itself from natural uranium by the occurrence of higher amounts of γ-emitting isotopes and neutron absorbing isotopes such as U-236. The nuclear industry has in place the facilities that are needed to recycle irradiated uranium (REPU) on a semi-industrial scale. This includes chemical conversion of REPU, enrichment, fuel fabrication and transport as well as reactor irradiation of REPU-based fuel.

As long as LWRs make up a significant market fraction of the nuclear power plant park in future fuel cycle schemes, increasing inventories of depleted and recovered irradiated uranium will build up. In particular, the latter would be of increasing importance as depleted uranium is used in fast reactor systems where the recycling of recovered irradiated uranium would not compensate for its production. Only the use of this recovered irradiated uranium in future fuel cycles can provide a steady decrease of the built-up inventory of depleted uranium and the reprocessed irradiated uranium. In the very long term (about 1 million years), as the natural decay chains reach secular

equilibrium this depleted and recovered irradiated uranium builds up in activity and radiotoxicity to a level comparable to natural uranium.

4.5.8 Conclusion

In view of the development of new reactor concepts the potential of hydrometallurgical and pyrochemical techniques has been assessed:

Aqueous processes have a high potential to reprocess spent fuels from commercial LWRs, including the separation of MAs as demonstrated by extensive research, especially in Europe. The major drawbacks of this technique are:

- The limited solubility of advanced fuel forms
- The limited stability of the organic extraction molecules in high radiation fields

Pyrochemical techniques, thus far developed only to a laboratory scale or pilot scale, offer potential for reprocessing of advanced fuels due to their good compatibility with most fuel forms and their high radiation resistance. In addition, the increased proliferation resistance and the compactness of the technique are important advantages, but the feasibility of the recovery of minor actinides has to be demonstrated. Electrorefining is generally considered to be the most promising pyrochemical method, and it is being investigated world-wide, especially in the U.S., Japan and Russia.

Aqueous and pyrochemical recycle techniques should be considered as complementary technologies. In a double-strata concept for instance, aqueous processes might be used to recycle the first LWR cycle (first stratum), including MAs. In the second stratum the FBR or dedicated burner reactor could rely on dry recycle techniques.

4.6 Concept-Independent Back End Links in the Fuel Cycle

Waste management, constrained by the number and capacity of available disposal sites, may be equal to, or a greater constraint than fissile material availability for Gen-IV fuel cycles. Major differences in waste quantities, characteristics, and costs arise for the four generic fuel cycle types. The major mass and volume of wastes are from the front-end of the fuel cycle (mining and milling). For the radioactive back-end wastes, repositories are required but the number, size, cost, and characteristics of those repositories are dependent upon the fuel cycle. While the waste management demands of once-through fuel cycles are reasonably well understood, as are those of the Pu recycle fuel cycle to a lesser degree, the implications of the other fuel cycles are yet to be determined through research and development.

The objective of sustainable waste management is to dispose of wastes while protecting humans and the environment to current standards for all time. Both chemical and radioactive waste must be considered. The world is in a transition from historical waste management practices to sustainable waste management practices. Different countries are at different points along this transition.

If nuclear power is used for a limited time on a limited scale, current waste management approaches are suitable. However, as with most energy technologies (fossil, with emissions of carbon dioxide, sulfur dioxide, etc. renewable sources, requiring energy and materials inputs for manufacturing, use of heavy metals in solar cells, significant land use, etc.; and nuclear), the scale of operation can alter the preferred waste management strategy. If nuclear power is used in the future on a global scale, changes may be required.

From the broad perspective, the most sustainable waste management policy is to not generate wastes; based on this principle, recycle/use is the next most sustainable waste management policy. If waste production is unavoidable, any wastes that are generated should be in a physical, chemical, and nuclear form that reduces waste management burdens.

Three approaches exist for waste management and are components of most waste management strategies. Different waste management strategies place more or less emphasis on one or more of these approaches.

- *Containment*. Radionulides can be isolated until they decay to non-hazardous concentrations. The isolation period depends upon the initial concentration and the half-life. Except for a small number of radionuclides, most fission products decay to low levels within a few centuries; the residual contents then have lower toxicity than the original uranium ore. A larger fraction of the transuranic elements created by neutron capture in reactors have long half lives; fuel design and recycle can be used to reduce or stop the accumulation of these elements. Three types

of containment strategies are considered: engineered disposal, engineered storage, and geologic isolation.

- *Dilution*. Hazardous materials can be diluted to safe levels. Whether one considers chemical releases (sulfur dioxide, carbon dioxide, heavy metals, etc.) or radioactive releases (tritium, krypton-85, etc.), there are limits to what the environment can safely accept. Therefore, for all energy sources and waste streams there exists a critical scale of operation, beyond which releases per unit of energy produced must decrease as the scale of operations increases.

- *Destruction*. Many chemical wastes are destroyed by incineration or other techniques. Analogously, proposals have been made to transmute long-lived radionuclides to stable or shorter-lived radionuclides. Examples of such proposals include the various partitioning and transmutation (P&T) schemes and the disposition of excess weapons plutonium in reactors.

R&D Required for Generation IV Waste Management

Advanced fuel cycles will generate different wastes with different characteristics. Some fuel cycles may reduce the toxicity of the waste and thus reduce the burden of waste management. However, the different characteristics also enable the use of advanced waste management techniques at repositories and other disposal facilities. In repositories, the waste characteristic that most strongly impacts engineering, cost, and expected performance is radioactive decay heat. Research on advanced repository designs should be in parallel with research on advanced fuel cycles to understand the costs and benefits of those fuel cycles.

Examples of two areas with potentially large benefits are (1) long-term storage of wastes to reduce decay heat before final disposal and (2) high-heat/low-heat repositories where the heat generating radionuclides are managed separately as wastes.

Decay heat creates the fundamental limitation on repository capacity. To avoid excessive temperatures that could damage repository performance, the waste is distributed over thousands of packages and many tens of kilometers of tunnels. For SNF the fission products—30-year half-life ^{137}Cs and 28-year ^{90}Sr— generate roughly half of the total repository heat load. Actinides— principally 458-year ^{241}Am and 86-year ^{238}Pu—provide the other half. The longer half lives of the principal actinide contributors make decay heat management impracticable, so actinide decay heat inevitably deposits in the geologic media unless one adopts a fuel cycle where these actinides are destroyed—a partitioning/transmutation fuel cycle.

Fission product heat can likely be managed by a variety of strategies. Longer interim storage, and separation and separate interim storage of cesium and strontium, have been proposed as potential heat management strategies. Even more interesting are methods that would permit active heat management following waste emplacement, since

these methods could reduce the burden placed on future generations to manage and emplace the waste following interim storage. For unsaturated media like Yucca Mountain, simple ventilation of the drift tunnels can recover some 50% of the repository thermal capacity every 30 years. For saturated repositories, where the waste must be emplaced in intimate thermal contact with the geologic media, active cooling systems that can be abandoned in place provide an option for controlling heat deposition in the geologic media and increasing the total repository capacity. The ability to continuously regenerate repository thermal capacity by active heat management would enable sustained use of nuclear energy while minimizing repository space requirements, and thus is a topic of interest for Gen IV research.

These advanced concepts do not change the need or siting of repositories. They may improve performance while reducing repository cost and size. Alternatively, the capacity of a fixed repository is significantly increased. All proposed repositories will be staged repositories; that is, they will be built in sections as wastes are disposed of. This generic characteristic of repositories implies that new technologies as they are developed can be implemented into existing repository systems.

4.7 Thorium Fuel Technologies

Because ^{235}U is the only naturally occurring fissile isotope, most of the world's nuclear reactors use this isotope to sustain the neutron chain reaction (the fuel utilizes either natural uranium or, in most cases, uranium enriched in ^{235}U). Thorium is a naturally occurring, fertile isotope (i.e. not thermally fissionable) that can also breed a fissile isotope, ^{233}U. A neutron chain reaction can only be sustained with thorium if fissile materials are available (^{235}U, ^{233}U, ^{239}Pu). By mixing such fissile isotopes with thorium it becomes possible to operate a nuclear reactor with a "thorium cycle" in which ^{233}U is produced. As with plutonium, ^{233}U is then partly consumed through in situ fission in reactors, and the remaining part contained in the discharged fuel may be recycled to generate fission power. Thorium then provides an alternative to the uranium cycle.

The main incentives for introducing thorium-based fuel cycles in the past have been

- the enhancement of fuel resources by breeding ^{233}U,
- the existence of domestic thorium in some countries and conversely, shortages of uranium,
- a desired reduction in overall ^{235}U requirements,
- good in-core neutronic and physical behavior of thorium fuel, and
- the lower excess reactivity requirements (higher thermal conversion ratio) of Th-based cores.

Set against these advantages, the principal reason that thorium has not been used more widely to date is that the ore contains no fissile isotope, as does uranium ore. Recently, however, renewed interest has arisen in thorium, not for its abundance as raw material, but because it may generate less long-lived minor actinides than the traditional uranium fuel cycle. Another incentive has appeared with the need for new fuel designs for burning plutonium in thermal spectrum reactors, where thorium may play an attractive role as a substitute for depleted uranium as the fertile material. These investigations include advanced reactor concepts based on thorium fuel cycles for future nuclear application such as LWRs, HTRs, MSRs, aqueous homogeneous suspension reactors, ADSs, and even fusion reactor blanket systems.

During the pioneering years of nuclear energy, 1950-1970, a large number of potential avenues for energy production with thorium were investigated. Since that time, quite significant experience on thorium-based fuel in experimental or power reactors has been gained worldwide. These experiments included post-irradiation examination of spent Th-based fuel from the reactors, and fabrication of Th-based fuel, on both pilot scale and semi-industrial scale. In addition, R & D was directed at recycling spent Th-based fuel on both laboratory scale and pilot scale and in laboratory re-fabrication tests. Several power reactors also used thorium, and some in India still use thorium fuel today.

With regard to recycling, the separation of ^{233}U and thorium is usually accomplished by wet liquid-liquid extraction using the THOREX process, for which a

pilot plant has been operated during many years at Oak Ridge. The recycling of U/Th is somewhat more complicated than with the traditional uranium cycle but should not pose too great problems to the processors today after the proper flow sheets have been selected. Head-end processing of pyrocarbon/silicon-carbide-coated carbide or oxide fuels for the HTR is challenging to address with the THOREX process, however.

Significant experience has been gained through past development on thorium-based fuel in both test reactors and power reactors. The feasibility of different types of existing reactors or prototype reactors based on Th-^{233}U has been successfully demonstrated and the fuel cycle technologies (mining, fuel fabrication, recycle and refabrication) are, in principle, available. However, the use of thorium at an industrial scale would still entail quite important R & D efforts and costs to master and optimize all the steps of the fuel cycle (including a better knowledge of thorium resources and extraction processes). However, modern technological breakthroughs, such as remote fuel fabrication techniques already applied to MOX fuels, should be able to overcome the previously envisioned technological hurdles to the implementation of thorium cycle.

Thorium-based fuel has attractive features for addressing long term radiotoxicity of wastes and plutonium stockpiles management, which may justify further studies to realistically assess its potential benefits under current conditions. However in the longer term, the main incentive for thorium utilization will remains the expansion of natural resources, which will be needed in the case of a sustained long-term application of nuclear energy.

4.8 Non-proliferation and Fuel Cycle Technology

Gen-IV sustainability goals include improved fuel cycle resistance to diversion or theft of weapons usable materials such that the nuclear energy supply provides the least attractive route to proliferation. Historically, there are many strategies and philosophies regarding proliferation resistance – (see the discussion in Chapter 5). Numerous groups, such as TOPS, have *recently* examined this issue; the results of their work will not be repeated here.

Much of the non-proliferation R&D required for Gen-IV concepts will be associated with specific Gen-IV concepts and is thus not discussed here. However, important *generic* fuel cycle non-proliferation R&D issues should also be addressed.

The proliferation resistance basis that impact on safeguards strategies for different fuel cycle options is as follows:

- *Once-through fuel cycle.* The basis for proliferation resistance in this strategy is to minimize processing of materials that contain weapons-usable materials. This minimizes the number of commercial facilities that could be modified to separate weapons useable materials from the SNF. The SNF is directly disposed of. Once-through fuel cycles result in a continuously increasing inventory of weapons-usable materials in SNF. Traditional methods to improve the proliferation resistance include: (1) fuel forms that make recovery of weapons-usable materials difficult, (2) higher burnup that produces smaller quantities of weapons-usable materials per unit of power, and (3) higher burnup that produces isotopic mixtures that are more difficult to use in the construction of weapons.

- *Closed fast-neutron breeder fuel cycle.* In this strategy traditional methods to improve proliferation resistance include (1) maintaining highly radioactive materials with the weapons-usable materials and minor actinide contamination during processing, (2) security in sensitive facilities and (2) destruction of weapons-usable fissile materials and thus minimizing the weapons-usable fissile materials that ultimately go to the repository. The total inventory of weapons-usable fissile materials is capped. The systems can be designed so the weapons-usable materials are limited to a subset of facilities.

- *Closed thermal-neutron breeder fuel cycle.* The basis for proliferation resistance in this strategy is to (1) minimize the total quantity of weapons-usable fissile materials in the fuel cycle, (2) design the system so the residual weapons-usable materials in the fuel cycle have very poor isotopics for manufacturing nuclear weapons, (3) eliminate off-site fuel-cycle facilities handling weapons-usable materials, and (4) limit process capability. All of these fuel cycles use mixtures of ^{233}U, ^{238}U, and thorium. The ^{233}U is isotopically mixed with ^{238}U to preclude its direct use as weapons material. Some plutonium is produced from the ^{238}U; but the reactors are designed to minimize the inventory of plutonium.

The Gen-IV Roadmap development will require that judgments be made about the proliferation risks of different fuel cycles. Reactor and fuel cycle designers must first know what improves or degrades proliferation resistance in order to develop and modify designs as appropriate. Since different fuel cycles have different strategic approaches to proliferation resistance, it is currently difficult to make cross comparisons. Continued development of a systematic basis to compare different systems is needed for improved capability to assess and to optimize their proliferation resistance.

Several examples illustrate the issues. The IAEA has defined weapons usable ^{235}U as uranium containing 20% or more ^{235}U in ^{238}U. Additionally, the quantity of plutonium necessary to build a nuclear weapon has been defined by the IAEA as 8 kg. These definitions provide a basis for the reactor and fuel cycle designers to design *traditional* systems that either do not use weapons-usable materials or minimize the inventories of such materials. However, advanced systems existing definitions are not sufficient because advanced fuel cycles employ new and nontraditional combinations of fissile materials. Several examples can illustrate this:

The usability of plutonium in nuclear weapons depends upon its isotopics. Plutonium isotopics depend upon the neutron spectrum and fluence employed in the fuel cycle. Table 4.2 shows isotopics for weapons-grade plutonium, PWR plutonium, and a proliferation resistant molten salt reactor. With some high-burnup fuels (high temperature gas cooled reactors and molten salt reactors), the dominant plutonium isotope is ^{242}Pu. Plutonium-242 has a critical mass about an order of magnitude larger than ^{239}Pu. It is clear that 8 kg of ^{242}Pu is a much smaller proliferation risk than 8 kg of ^{239}Pu. If 8 kg of ^{239}Pu is defined in safeguards regulating as that required to build a weapon, a currently undefined, quantity of plutonium should be defined for different isotopic mixtures.

Table 4.2 Plutonium Isotopics (%)

Isotope	Weapons grade	LWR[a] (PWR)	Denatured MSR
^{239}Pu	93	56.6	30
^{240}Pu	6.5	23.2	18
^{241}Pu	0.5	13.9	14
^{242}Pu	0.0	4.7	38

[a]The PWR SNF plutonium also contains 1.3% ^{238}Pu.

Many advanced fuel cycles propose to recycle and burn neptunium, americium, and curium. No definitions of significant weapons-usable quantities of these materials currently exist. Similarly, no definition of weapons-usable ^{233}U currently exists; although, technical analysis indicates that 12% ^{233}U in ^{238}U is equivalent to 20% ^{235}U in ^{238}U

(Forsberg,1998).[1] Many new fuel cycles propose fuels containing mixtures of ^{233}U, ^{235}U, and ^{238}U.

R&D Needed for Advanced Fuel Cycle Nonproliferation Strategies

Proliferation resistance of fuel cycles is achieved by combinations of intrinsic and extrinsic barriers at each link in the fuel cycle; the intrinsic and extrinsic barriers can be designed to complement each other and to address resistance to different threats at each link in the fuel cycle chain. But if fuel cycle and reactor designers are to improve "proliferation resistance", they will benefit from development of a generally agreed basis to judge proliferation risks so as to understand which materials present the greatest risks and what barriers present the more cost effective resistance against different threats. Current methodologies must be extended to enable such comparisons to be made. It is in this area that significant work is required. *The metrics to judge proliferation-resistance must be improved as a part of the Gen-IV R&D effort.*

[1] C. W. Forsberg, C. M. Hopper, J. L. Richter, and H.C. Vantine. *Definition of Weapons-usable Uranium-233*, ORNL/TM-13517, Oak Ridge National Laboratory, Oak Ridge, Tennessee (March 1998).

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 5

Institutional Boundary Conditions and Enabling Developments

March 18, 2002

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Chapter 5

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Chapter 5
Institutional Boundary Conditions and Enabling Developments

This chapter is intended to review the history and status of international institutional arrangements, which may form boundary conditions on achievable scales and conditions for Gen-IV fuel cycle and power plant deployment. While not yet exhaustive on all the possible institutional boundary conditions that might emerge in the future, it brings an overview of today's readiness for expanded global deployment.

5.1 The Institutional Challenges for Generation-IV

The scenarios for Generation-IV nuclear deployment in the 21st century are discussed in Chapter 3; they use WEC/IIASA estimates for two conceivable futures for growth in worldwide energy use and nuclear's role in that growth. The economic growth driven scenario (Case B) projection shows nuclear growing from 350 GWe today to 2000 GWe in 2050 to 6000 GWe in 2100. The ecology driven scenario (Case C2) shows growth to 1200 GWe in 2050 and 1800 GWe in 2100. In either potential future scenario, nuclear deployment is projected to grow dramatically (deployment rates of up to 35 GWe/year over 50 years) and in either potential future scenario, that growth includes major deployment in developing countries.

The Gen-IV sustainability goals, to be consistent with potential growth, contain three elements – long-term cost-effective supply of energy within available uranium ore resources, acceptable environmental impact, and acceptable level of proliferation risk associated with the Gen-IV fuel cycle. Meeting these goals will require more than just technological advances. The projected massive worldwide deployment will also be facilitated and guided through the use of government regulation and, specific to this Chapter, international regulations and treaties. Market forces will be tempered by both government regulation and international agreements.

The nuclear industry is already highly regulated on a country by country basis. The laws and regulations are complex and often overlapping, involving several government ministries, departments, and/or agencies. In many countries, individual states, provinces, and/or regional governments may also be involved in the regulatory process. The laws and regulations typically provide licensing of various aspects of the nuclear industry; government oversight; setting of standards (both technical and environmental); and protection of human health from radiological (and other) hazards. However, intra country institutional arrangements are not discussed here.

International institutional arrangements in the form of conventions or treaties have been introduced in domains where consequences of the use of nuclear energy cross national borders and become of an international nature and where a net benefit could be achieved in terms of sharing common norms. In some other transnational aspects the institutional arrangements have not resulted in such international conventions; for instance in the field of nuclear safety regulation, bilateral or multilateral information exchange has traditionally been considered as the more appropriate approach – without the need of a higher institutional organism to bring harmonization or agreement.

In the same sense that Chapter 4 has reviewed the current status of fuel cycle technology and R&D ongoing worldwide, this chapter provides an overview of the current status of international treaties and norms which have been put into place up to now in order to enable and to regulate international elements of nuclear energy. These existing international treaties and norms exist in five domains:

- Nonproliferation
- International nuclear damage liability and compensation
- Safety of Nuclear Installations
    * Standards and Conventions
    * Early Notification of Accidents
    * Radiological Emergency
    * Physical Protection of Nuclear Material (Shipping)
- Nuclear Waste Management and Disposal

and

- Radiological Protection

The status of their ratification and adoption which is reviewed below provides an indication of institutional readiness for an expanded nuclear deployment and identifies areas for further developments. For example, some issues are likely to arise upon substantially expanded deployment that were not relevant in the past; they will require new approaches. As an example, the scenarios displayed in Chapter 3 show that the fuel cycle for the global nuclear energy park as a whole provides a mass flow infrastructure and the technical mechanisms to regulate such flows in a way to achieve balance of fissile production vs. destruction in a park comprised of diverse power plants filling diverse energy needs. Such a symbiosis among power plant types achieved via the fuel cycle can in principle avoid:

- Shortages of fissile material
- Extensive inventories of fissile material in interim storage

and

- fissile material consigned to geologic repositories

However market forces alone may not be sufficient to achieve this desired balance in mass flows. (The current situation bears witness to this possibility; fissile material in discharged fuel builds up inexorably because market forces favor open (once through) fuel cycles over closed fuel cycles.) Thus, just as with other forms of commerce, governmental institutional measures may in the future be required to incentivize and regulate the mass flows in the symbiotic fuel cycle of a widespread nuclear deployment.

5.2 Review of the Status of International Institutional Aspects

5.2.1 Non-proliferation

From the earliest days of the nuclear era, the unrestrained spread of nuclear weapons has been widely recognized as a threat to international peace and stability. At the same time, however, it was also widely believed that the peaceful use of nuclear energy held great promise for the world's economic development. Reconciling these propositions has been one of the principal goals of the international nuclear nonproliferation regime developed over the last thirty years.

Both propositions are prominently reflected in the 1968 Treaty on the Non-Proliferation of Nuclear Weapons (NPT), the foundation of the current nonproliferation regime. Article II of the NPT, for example, provides that "Each non-nuclear weapon State[1] Party to the Treaty undertakes … not to manufacture or otherwise acquire nuclear weapons or other nuclear explosive devices and not to seek or receive any assistance in the manufacture of nuclear weapons or other nuclear explosive devices." However, NPT Article IV, paragraph 2, provides that "All Parties to the Treaty undertake to facilitate, and have the right to participate in, the fullest possible exchange of equipment, materials and scientific and technological information for the peaceful uses of nuclear energy." Article III of the NPT requires each non-nuclear weapon state party to accept International Atomic Energy Agency (IAEA) safeguards on all its peaceful nuclear activities, "for the exclusive purpose of verification of its obligations assumed under" the NPT.

One of the major boundary conditions in the development of Generation-IV systems is to support the NPT and the other elements of the nonproliferation regime, including IAEA safeguards. Sustainability Goal 3 (SU-3) declares that "Generation-IV nuclear energy systems including fuel cycles will increase the assurance that they are a very unattractive and least desirable route for diversion or theft of weapons-usable materials."[2] Several implications of this Goal should be noted.

First, SU-3 recognizes the practical reality that any civil nuclear facility containing weapons-usable material can be misused. The phrase "very unattractive and least desirable route" implies that proliferation resistance is a relative, not an absolute, value. Total proliferation resistance can never be attained, and for the foreseeable future there may be countries that cannot safely be given access to any civil nuclear fuel cycle facilities.

Second, under SU-3 the resistance of Generation-IV systems will be evaluated in relation to two types of proliferation risk—*diversion* of weapons-usable material and *theft* of weapons usable materials. As a general proposition, diversion involves the misuse of nuclear material by the government of the territory where the facility is located, and theft involves the misuse of nuclear material by others, such as a criminal group, political terrorists, or a foreign government.

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[1] Under NPT Article IX, paragraph 3, the "nuclear weapons States" are those which had manufactured and exploded a nuclear explosive device prior to January 1, 1967 (i.e., the United States, Soviet Union, United Kingdom, France and China). All other countries are "non- nuclear weapons States" for purposes of the Treaty. The Russian Federation has succeeded to the NPT rights and obligations of the Soviet Union.

[2] "Weapon-usable nuclear material" includes plutonium with any isotopic composition with the exception of "heat-source" plutonium containing 80% or more of the isotope ^{238}Pu, uranium enriched in the isotopes ^{233}U and/or ^{235}U to 20% or more, or other fissionable material having physical properties suitable for such purposes, including ^{237}Np.

IAEA safeguards are primarily intended to detect diversion of nuclear material. For safeguards in non-nuclear weapon states party to the NPT, the IAEA defines diversion as "the use of nuclear material for the manufacture of nuclear weapons or other nuclear explosive devices, or removal from peaceful uses for purposes unknown; or the withdrawal of nuclear material from safeguards except as provided for in the [safeguards] agreement"[3] between the IAEA and the state concerned.

The objective of NPT safeguards is timely detection of the diversion of a significant quantity of safeguarded nuclear material. A "significant quantity" is the approximate quantity of nuclear material needed to make a nuclear explosive device. For example, the IAEA assumes this quantity to be 8 kilograms for plutonium[4] and 22 kilograms for highly enriched uranium. Timely detection depends on time required to convert diverted nuclear material into the metallic components of a nuclear explosive device. For detection to be timely, for example, diversion of fresh fuel containing HEU, plutonium or MOX should be detected within a month of the diversion. Diversion of irradiated fuel containing HEU or plutonium should be detected within three months, while diversion of fresh fuel containing natural uranium, LEU or thorium should be detected within one year of diversion. Timeliness goals for other materials may be set out in safeguards agreements.

Theft of nuclear materials is usually discussed in the context of physical protection of nuclear materials and facilities. (Physical protection measures cover the prevention of sabotage in addition to theft.) Standards for the international transport of nuclear material are set forth in the 1980 Convention on Physical Protection of Nuclear Material. International standards for protection of nuclear material in use, storage or transport are found in the IAEA's Recommendations on the Physical Protection of Nuclear Material,[5] first issued in 1977 and periodically revised thereafter. Both instruments categorize material based on the time and effort required to convert it to nuclear explosive use, and require increasingly stringent security measures as the time and effort of conversion decreases.

Generation-IV systems might be made less attractive as the object of the diversion or theft of nuclear material for either intrinsic or extrinsic reasons. *Intrinsic barriers* to proliferation are defined by material qualities (isotopic composition, chemical separability, mass and bulk, fuel matrix radiation level, dilution and detectability characteristics), and by technical impediments that are inherent to a nuclear system, such as facility unattractiveness and accessibility, mechanical impediments to material and vital equipment access, and high fuel burn-up rates. *Extrinsic barriers* involve institutional controls, such as national materials protection, control and accounting (MPC&A) performed by the host nation to prevent theft and sabotage; and the detection of diversion and misuse performed by the IAEA or other international safeguards mechanisms. It is widely recognized that intrinsic proliferation barriers, by themselves, will never be sufficient.

It should also be noted that intrinsic barriers to proliferation may ease the application of extrinsic barriers by making the latter less costly and more efficient. For example, high burn-up of weapons usable nuclear materials in irradiated fuel will leave less such material to be safeguarded after removal from the reactor. Again, high radiation barriers to the handling

[3] IAEA INFCIRC/153, paragraph 14.
[4] Containing less that 80% Pu 238.
[5] IAEA INFCIRC/225.

of spent fuel may require protective measures that would make it easier for safeguards instrumentation to detect attempts to tamper with the fuel.

Finally, it should be noted that designers might be able to compensate for lower intrinsic proliferation barriers by making systems more friendly to extrinsic barriers. For example, a design might include built-in safeguards instrumentation that would be difficult to remove without detection.

It might be thought that application of international safeguards would only be of importance for fuel-cycle systems to be deployed in non-nuclear weapon states, since the NPT does not require IAEA safeguards on peaceful nuclear facilities in the five nuclear weapons states. For purposes of Generation-IV evaluation, however, the distinction between deployment in nuclear weapon states and non-nuclear weapon states is probably of little importance. To begin with, Generation-IV evaluators should not assume that a nuclear facility developed by a nuclear weapon state will only be employed there. A safeguards-friendly facility that can be exported to at least some non-nuclear weapons states will be economically much more attractive than one that has to remain in a nuclear weapon state. In addition, nuclear weapon states generally maintain a strict MPC&A system over material in their civil nuclear fuel cycle. Any feature that makes international safeguards more efficient is likely to be attractive from the standpoint of national MPC&A systems as well. Finally, it should be noted that all the NPT nuclear weapon states have entered into some form of voluntary safeguards agreement with the IAEA for some of their peaceful nuclear activities. Therefore, even a Generation-IV system deployed in a nuclear weapon state might sometimes be subjected to IAEA safeguards monitoring.

5.2.2 The International Nuclear Damage Liability and Compensation Regime

In the 1950's, the governments of many industrialized countries viewed the development of nuclear power generation as an important element in the development of their economies. However, the fear of financially devastating liability claims that could result from a nuclear accident inhibited both investment in nuclear power plants and the supply of goods and services to those plants, for under ordinary law of civil liability there is no limit to the liability of plant operators and nuclear suppliers for damages resulting from a nuclear accident. At the same time, governments wished to ensure that their citizens would be properly compensated for damage suffered if a nuclear accident actually did occur. The solution found to reconcile these conflicting interests was a legal regime based upon the strict and exclusive liability of nuclear operators, with limitations imposed upon both the amount of that liability and its duration in time.

The complexity of addressing nuclear accident liability mirrors the greater complexity of internalizing the environmental, economic, and public-health costs of all energy production activities. These complexities arise because many potential sources of specific pollutants exist, which creates uncertainty in ascribing specific pollutants to a specific source, and because multiple mechanisms exist that can create specific environmental, economic, or

[9] The Contracting Parties are: Belgium, Denmark, Finland, France, Germany, Greece, Italy, the Netherlands, Norway, Portugal, Spain, Sweden, Turkey and the United Kingdom.

public health damage, which creates uncertainty in ascribing specific damage to a specific pollutant.

The uncertainty in ascribing causation for damage to specific pollutants and sources has made it largely impossible to internalize the environmental and public health costs of fossil fuel use, particularly from those pollutants which act on regional and global scales. Also, to a lesser extent it has been difficult to internalize environmental and public health costs from the manufacture and construction of renewable energy infrastructure. Similar difficulties exist in ascribing causation for damages from emissions of radioactive materials from normal operation and accidents in nuclear energy facilities. The problem of ascribing causation has been dealt with in the current nuclear liability regime by the definition of the specific types of damage to be covered, as noted below.

5.2.2.1 The Basis of The Third Party Nuclear Liability Regime

Exclusive Liability of the Operator

To overcome the concerns expressed by nuclear suppliers, governments adopted the concept of "channeling" all liability for third party nuclear damage to the operator of the installation where the accident occurs, regardless of whose acts or omissions were the real cause of the accident. This saves suppliers the cost of defending complicated and expensive liability suits, and the high cost of obtaining insurance against such claims. For victims, channeling obviates the need to identify and pursue the person who actually caused the accident. In the case of transport of nuclear materials, liability for third party nuclear damage lies either with the operator sending the substances or with the operator receiving them, rather than with the carrier as is the normal rule. This principle is evidenced in the nuclear liability legislation of almost all nuclear-power generating countries.

Strict Liability of the Operator

From the beginning, it was clear that the concept of strict liability should apply to the nuclear power industry because of the unusual hazards involved, the difficulty of determining the real cause of an accident and the fact that the operator of a nuclear installation has complete control over all aspects of its operation. Strict liability means that the operator of a nuclear installation is liable for all third party nuclear damage resulting from its operation without the need to establish the operator's fault or negligence. Liability is established merely upon proof of a causal link between the damage suffered and the nuclear accident in question. This principle is also evidenced in the nuclear liability legislation of almost all nuclear-power generating countries.

Scope of the Liability

This special liability regime applies only to nuclear installations in which low probability, high consequence accidents could occur, such as nuclear power reactors, research reactors, factories for the manufacturing or processing of nuclear substances, for the separation of isotopes of nuclear fuel, and for the reprocessing of irradiated nuclear fuel, and facilities for the storage of nuclear substances (other than storage incidental to the carriage of such substances). Also covered are the transport and storage of nuclear substances and radioactive products and waste. Low risk activities, such as medical and industrial usage of radioisotopes and uranium mining and milling are not covered by the special regime.

Compulsory Financial Security

To ensure that funds are available to pay third party claims, financial security for an operator's liability is compulsory. Usually the security is provided by private insurance, but it could also be a bank guarantee, a State guarantee or indemnity or self-insurance.

Limits on Liability

Under ordinary civil liability law there is no limit on liability for damage caused by an accident. However, under the special nuclear liability regime, most national laws impose a limit on the operator's liability for nuclear damage suffered by third parties as a result of a nuclear accident. The amount varies from country to country. Usually States will provide compensation beyond the operator's liability limit, in recognition of the fact that the operator's insurance, and even its assets, may not be sufficient to cover all claims resulting from a major accident.

Time Limits

Insurance companies have limited their coverage to not more than ten years from the date of the accident. Neither insurance companies nor operators can maintain financial security for an extended period of time, and thus in many countries' national legislation, the time limit for submission of claims is the same ten years. In some countries the State will pay for damage in respect of which claims are brought beyond the limitation period.

5.2.2.2 The International Conventions

In the late 1950's, it was recognized by the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (OECD/NEA), the International Atomic Energy Agency (IAEA) and the European Atomic Energy Community (Euratom) that a nuclear accident might have trans-boundary consequences, and that States with nuclear power programmes needed to conclude an international agreement that would govern compensation for nuclear third party damage incurred both domestically and internationally. Such an agreement would operate to harmonize national laws, establish rules for cross-border claims by victims in one country against a liable nuclear operator in another country and govern liability issues arising from the transport of nuclear materials from one country to another and through a third country. For potential victims, such an agreement could serve to by-pass the complex and often inappropriate rules of private international law governing claims for compensation for cross-border nuclear damage. Within a few years, three international conventions in the field of nuclear liability were adopted:

1960 Paris Convention on Third Party Liability in the Field of Nuclear Energy (OECD)

The Paris Convention was the first to be adopted and it entered into force in 1968. It is open to all Member countries of the OECD by simple accession and to any other State by the unanimous consent of all Parties. At present, it is a regional agreement with 14 Contracting Parties, all from Western Europe[9]. The Convention generally does not apply to nuclear incidents occurring, or nuclear damage suffered, in the territory of non-contracting States. It does apply to damage suffered in a Contracting State as a result of a "nuclear

incident"[10] occurring in a "nuclear installation"[11] or involving substances coming from that installation.

The nuclear operator's liability is both exclusive and strict. No supplier or contractor may be held liable, even if it has been negligent or at fault, unless it has accepted liability by contract, in which case the operator has a right of recourse only. The "operator" of a nuclear installation is the person recognized or designated as such by the competent public authority. If the nuclear substances are in an installation at the time of an accident, the operator of that installation is liable to compensate the damage thereby caused. If the accident occurs during the course of carriage, the operator responsible is the sender unless the receiver has assumed responsibility therefore by contract or has taken charge of the substances[12]. The operator is not liable for damage caused by a nuclear incident directly due to an act of armed conflict, hostilities, civil war or insurrection. National law determines whether that exoneration extends to damage caused by a grave natural disaster of an exceptional character.

The operator is liable for damage to or loss of life of any person, and damage to or loss of any property other than property on the site of the accident. However, rights under public health insurance, social security, worker's compensation, or other occupational disease compensation systems under national law are not affected and if a victim is so compensated the body expending the funds may have a right of recourse against the operator.

The operator's maximum liability may not be greater than 15 million Special Drawing Rights (SDRs)[13], although national legislation may fix a higher amount if corresponding financial security is available[14]. A Contracting Party may set a lower amount, of not less than 5 million SDRs, for less dangerous installations, but must provide public funds to cover any excess damage up to the maximum limit. If more than one operator is liable, they are jointly and severally liable.

The time limit for instituting claims is ten years from the date of the incident, although States may extend that period as long as the operator's liability is covered by

[10] A "nuclear incident" is any occurrence or series of occurrences having the same origin which causes damage arising either from the radioactive properties or a combination of radioactive properties with toxic, explosive, or other hazardous properties of nuclear fuel or radioactive products or waste, or from ionising radiation emitted by any source of radiation inside a nuclear installation.

[11] "Nuclear installation" means reactors other than those comprised in any means of transport; factories for the manufacture or processing of nuclear substances; factories for the separation of isotopes of nuclear fuel; factories for the reprocessing of irradiated nuclear fuel; facilities for the storage of nuclear substances other than storage incidental to the carriage of such substances; and such other installations in which there are radioactive products or waste as determined by the Steering Committee for Nuclear Energy, the governing body of the OECD/Nuclear Energy Agency.

[12] Where nuclear substances are being sent to a person in a State not party to the Convention, the sending operator is liable until the substances are unloaded from the means of transport. Conversely, where substances are being sent from a person in a State not party to the Convention to an operator in a State party with its written consent, the latter will be liable from the time the substances are loaded onto the means of transport.

[13] The SDR is the unit of account used by the International Monetary Fund and is based upon a basket of weighted currencies. The amounts of compensation under the Paris Convention are to be converted into national currency in accordance with the SDR/national currency value established at the date of the incident.

[14] In most Contracting Parties, the operator's liability is far higher than 15 million SDRs and in one, it is unlimited. A 1990 Recommendation by the NEA Steering Committee calls for Contracting Parties to raise their liability limit to at least 150 million SDRs.

corresponding financial security. States may also specify a period after the victim knew of the damage and the liable operator within which he must bring his claim[15].

Operators are required to maintain insurance or other financial security, approved by the State in which their installation is located, in an amount corresponding to their liability under the Convention. Financial security may only be used to compensate claims for damages, and not for the payment of interest or costs.

The courts having jurisdiction to hear claims are those of the State where the nuclear incident occurs, unless the place of the incident cannot be determined or the incident occurs in a non-Contracting State, in which cases jurisdiction lies with the courts of the State in whose territory the installation of the liable operator is located. The courts will apply the Convention, and for matters not covered by the Convention their own law, without discrimination on the basis of nationality, domicile or residence. The nature, form, and extent of compensation, as well as its equitable distribution are determined by national law. Insurance premiums and monetary compensation must be freely transferable between Contracting Parties, and judgments must be enforceable in the territory of any Contracting Party.

1963 Brussels Convention Supplementary to the Paris Convention

The Paris Convention States were keenly aware that the limit of liability under that Convention would not go far in compensating damage resulting from a major nuclear accident. So shortly thereafter they adopted the Brussels Supplementary Convention in an effort to ensure that State funding would be available as supplementary compensation to that provided by the Paris Convention. The Brussels Supplementary Convention came into force in 1974 and its 11 Contracting Parties are the Paris Convention States except for Greece, Portugal and Turkey. It applies to damage caused by nuclear accidents, other than those occurring entirely in the territory of a non-contracting State. The incident must be one for which an operator would be liable under the Paris Convention and the courts of a Contracting Party must have jurisdiction.

The Convention establishes a three-tiered compensation system. The first tier is provided by the operator's financial security up to the maximum liability established by national law. The second tier is the difference between the first tier and 175 million SDRs and is provided by the State in which the nuclear installation of the liable operator is situated. The third tier, being the difference between 175 and 300 million SDRs is to be contributed jointly by all the Parties according to a formula derived from the gross national product (GNP) and the thermal nuclear power capacity of reactors situated in each State.

If the aggregate amount of liability resulting from a nuclear accident exceeds 300 million SDRs, any Party may establish equitable criteria for apportionment and the court having jurisdiction will decide upon the system of disbursements.

1963 Vienna Convention on Civil Liability for Nuclear Damage (IAEA)

[15] In the case of damage caused by a nuclear accident involving nuclear substances that have been lost, jettisoned or abandoned and not recovered, the time limit for making claims is 20 years from the date of the accident.

In May 1963, the members of the IAEA adopted another international nuclear liability convention, very similar to the Paris Convention, but with world-wide scope. The Vienna Convention on Civil Liability for Nuclear Damage came into force in 1977.

While this Convention embodies most of the basic principles found in the Paris Convention and in national legislation, it differs in some important respects. The Vienna Convention is more flexible than its Paris counterpart; for example, the Vienna Convention stipulates a *minimum* operator liability amount of US$5 million[16], with discretion given to Contracting Parties to set a maximum limit in their national legislation. The amount of financial security to be provided by the operator is also left to the discretion of the Contracting Party. In addition, the Vienna Convention contains a definition of "nuclear damage" which the Paris Convention does not, and the operator's liability is explicitly stated to be absolute. It also requires a State to guarantee the payment of compensation where the operators' financial security fails, which the Paris Convention does not.

Until the 1986 Chernobyl accident, there were only 10 Contracting Parties and only 2 of those had operating nuclear power reactors. In the 10 years following that accident, the number of Contracting Parties significantly increased[17] and a major revision of the Convention was negotiated, a revision that culminated in the adoption of an Amending Protocol in 1997.

1988 Joint Protocol Relating to the Application of the Paris Convention and the Vienna Convention (OECD and IAEA)

The inadequacy of the existing international nuclear liability regime became apparent when the Soviet Union refused to accept responsibility for damage suffered in other States as a result of the radioactive fallout from the Chernobyl accident. After that accident, many States saw value in revising the Vienna Convention so that it would attract most of the nuclear power generating States of Central and Eastern Europe.

However, it was also recognized that the Paris and Vienna Conventions existed in isolation from each other; victims in a Paris Convention State could not claim compensation, under either Convention, for damage arising from an accident for which the liable operator was situated in a Vienna Convention State. It appeared therefore that creating a link between the two Conventions, coupled with a revision of the Vienna Convention might result in the extension of benefits of the international civil liability regime to the entire European continent. The link, at least, was created by the adoption in September 1988 of the Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention.

The Joint Protocol deals with the civil liability of operators of nuclear installations governed by either the Paris or the Vienna Convention. It applies to nuclear damage caused by accidents occurring in land-based nuclear installations and during the transport of nuclear materials thereto and therefrom. The Joint Protocol permits victims in a Party to one Convention to obtain compensation for damages resulting from an accident occurring at an

[16] Defined by reference to its value in terms of gold on 29 April 1963; this is US$35 per one troy ounce of fine gold. That amount is worth approximately US$50 million today.

[17] The Contracting Parties are: Argentina, Armenia, Belarus, Bolivia, Bosnia-Herzegovina, Brazil, Bulgaria, Cameroon, Chile, Croatia, Cuba, Czech Republic, Egypt, Estonia, Hungary, Latvia, Lebanon, Lithuania, Mexico, Moldova, Morocco, Niger, Peru, Philippines, Poland, Romania, Slovak Republic, Slovenia, Macedonia, Trinidad, Tobago, Ukraine, Uruguay and Yugoslavia

installation located in the territory of a Party to the other Convention, and it prevents conflicts of jurisdiction by ensuring that only one Convention will apply to any one nuclear accident.

The Joint Protocol came into force in 1992 and the following are Parties to it: Bulgaria, Cameroon, Chile, Croatia, Czech Republic, Denmark, Egypt, Estonia, Finland, Germany, Hungary, Italy, Latvia, Lithuania, Netherlands, Norway, Poland, Romania, Slovakia, Slovenia, Sweden and Ukraine.

1997 Protocol to Amend the Vienna Convention

The revision of the Vienna Convention was carried out within the IAEA Standing Committee on Liability for Nuclear Damage, and the Protocol to Amend the Vienna Convention was adopted in September 1997. Three countries[18] have ratified the Protocol while five are required for it to come into force.

The 1963 Vienna Convention does not define its geographical coverage. The Vienna Amending Protocol provides that it applies to all nuclear damage wherever suffered, although the State in whose territory the installation causing the damage is situated may exclude damage suffered in another nuclear State which does not afford equivalent reciprocal benefits.

Under the Amending Protocol, nuclear installations and substances are defined as before, but the IAEA Board of Governors' powers to include new types of installations within the Convention's scope or to exclude them if the risk is sufficiently low, have been extended, making it easier to adapt the Convention to future needs. The Convention now clearly excludes installations used for non-peaceful purposes and there is no longer an exoneration from liability for damage caused by a nuclear incident resulting from a grave natural disaster of exceptional nature.

The Amending Protocol follows the trend of recent liability conventions by providing a detailed definition of nuclear damage. While retaining the categories of personal injury, loss of life and loss of or damage to property, the following new heads of damage are now included, although to what extent is left to the law of the competent court to decide:

- non-material damage (pure economic loss);
- cost of measures to reinstate significant environmental impairment;
- loss of income related to environmental damage;
- cost of preventive measures and any damage caused by such measures;
- any other economic loss (other than caused by impairment of the environment) permitted by national legislation on civil liability.

In addition, with respect to preventive measures, a nuclear incident now covers a situation which creates a grave and imminent threat of causing nuclear damage.

[18] Argentina, Morocco and Romania.

The US dollar has been replaced by the Special Drawing Right (SDR) as defined by the International Monetary Fund, following the example of many other international treaties. The new minimum limit of liability is 300 million SDRs. A State may set a lower limit of no less than 150 million SDRs, provided that the difference is made up from public funds[19]. In addition, for a transitional (phasing-in) period of up to 15 years from the entry into force of the Amending Protocol, a Contracting Party may fix a lower limit of 100 million SDRs. Financial security requirements correspond to the liability amount requirements. In addition, a Contracting Party may now contribute to the compensation of nuclear damage through public funds, thereby relieving the nuclear operator from a part of its obligation. The Amending Protocol authorizes States to set a lower amount, although not less than 5 million SDRs, in cases where there is a reduced level of risk and provided the State ensures payment of the difference up to the 300 million SDRs liability level.

The Amending Protocol now provides that actions for compensation must be brought within thirty years for personal injury and loss of life claims, and within ten years for other types of damage.

The rule designating the competent court as that of the Contracting Party in whose territory the nuclear accident took place remains intact. Now, however, that Party must designate which particular court will have jurisdiction to hear claims. Furthermore, in the event of an accident occurring in the exclusive economic zone of a Contracting Party during transport, the courts of the coastal State in question will have jurisdiction.

1997 Convention on Supplementary Compensation for Nuclear Damage

The need for additional funds to supplement the compensation payable by the nuclear operator's financial security, whether under national law or under an existing international convention, was recognized early on in the discussions within the IAEA Standing Committee on Liability for Nuclear Damage. That need was reflected in a totally new, “global” convention, the Convention on Supplementary Compensation for Nuclear Damage (CSC) which was adopted in September 1997. At present, three countries[20] have ratified this instrument. It will come into force when five States representing a minimum total of 400,000 units of installed nuclear capacity adhere to it.

The CSC is intended to generate funds to supplement the system of compensation available under the Vienna or Paris Conventions or under national legislation that reflects the principles of those Conventions. It is thus open to all States, an important fact when one considers that many States, both nuclear power-generating and otherwise, have not joined the basic Conventions[21]. Adherence to the CSC provides an opportunity for joining a global regime on liability and compensation for nuclear damage. The Convention applies to nuclear damage for which an operator of a nuclear installation used for peaceful purposes and situated in the territory of a Contracting Party is liable under either of the two basic Conventions or national legislation consistent therewith. A special “grandfather clause” is included in the Convention to address the particular situation of the United States whose national law on nuclear liability and compensation predates the basic liability conventions. That law is based on the concept of "economic channeling" which, in practical terms, leads to

19. To account for States which provide for the unlimited liability of their operators, the Amending Protocol provides that such States must require the operator’s financial security to be not less than 300 million SDRs.

20 Argentina, Morocco and Romania.

21 For example, the United States, the Russian Federation, Canada, Korea, Japan and China.

the same result as does legal channeling, but this difference prevents the United States from joining the basic Conventions.

Supplementary funds are provided, over and above a national compensation amount of at least 300 million SDRs the availability of which is ensured by the Installation State. The Convention does not prescribe the arrangements by which the national compensation amount is to be made available, but States in difficult economic circumstances may adhere by making use of a phasing-in mechanism that permits the fixing of a transitional amount of at least 150 million SDRs for a maximum of 10 years from the date of adoption of the CSC.

The supplementary fund will be available to compensate nuclear damage suffered (a) in the territory of States Party or (b) in or above maritime areas beyond the territorial sea by a national of a State Party, or on board or by a ship flying the flag of a State party, or on board or by an aircraft registered in the territory of a State Party, or on or by an artificial island, installation, or structure under the jurisdiction of a State Party; or (c) in or above the exclusive economic zone of a State Party or on its continental shelf in connection with the exploitation or the exploration of the natural resources. This is subject to the requirement that the courts of a State Party have jurisdiction pursuant to the Convention. Damage suffered in non-Contracting States is not covered, given that the fund comprises public funds.

The contribution of a State Party to the fund is calculated according to a formula based upon its installed nuclear capacity and its United Nations rate of assessment. A percentage limitation ("cap") is included to avoid a State Party having to provide an excessively large proportion of the fund but the cap is inapplicable to the State of the liable operator. Thus the amount of the supplementary fund will depend upon which States become Contracting Parties to the Convention.

The allocation of the supplementary fund aims to ensure that trans-boundary damage will be adequately compensated. While the national compensation amount is distributed on a non-discriminatory basis to victims both within and outside the installation State, the supplementary fund is distributed as follows: 50% to compensate nuclear damage within and outside the installation State, and 50% to compensate trans-boundary damage only. Where the State of the liable operator uses the phasing-in provision, the allocation of the supplementary fund is adjusted. On the other hand, if the national compensation amount is 600 million SDRs or greater, then the whole of the supplementary fund will be used to compensate nuclear damage both within and outside the installation State.

The CSC contains a jurisdiction clause comparable to that included in the Vienna Amending Protocol regarding the granting of jurisdiction to coastal states where a nuclear incident takes place within its exclusive economic zone.

Protocol to Amend the Paris Convention (yet to be adopted)

The Paris States decided to revise their Convention soon after the Vienna Amending Protocol had been adopted, as the existence of the 1988 Joint Protocol meant that any amendment to the Vienna Convention would have an impact on the Paris Convention. The Paris States also wanted to ensure that their revised Convention would not prevent them from adhering to the new Convention on Supplementary Compensation for Nuclear Damage. The negotiations are still ongoing and while agreement in principle has been reached on the vast majority of issues, the questions which remain unresolved are integral components of the whole Paris-Brussels scheme.

The first significant improvement in the revised Convention will be the expansion of its geographical scope. Under the existing Convention, a nuclear incident must occur in the territory of a Contracting Party and damage must be suffered there (unless otherwise provided by national legislation) before the Convention will apply. The amending Protocol will relax that rule by providing that the Convention will also apply to any nuclear damage suffered in a non-Contracting State as long as it is either:

- a party to the Vienna Convention/Joint Protocol, or
- it has no nuclear installations in its territory or in any of its maritime zones, or
- it has nuclear liability legislation affording equivalent reciprocal benefits and based on principles identical to those contained in the Convention.

In addition, the amount of the operator's liability under the revised Paris Convention is likely to be increased to 450-500 million SDRs. It is also likely that the unit of account will change from the SDR to the euro, which would mean that a 500 million SDR liability amount would be expressed as 700 million euros. This is a very considerable increase over the 15 million SDRs currently called for under the Convention. The revised Convention will allow for both limited and unlimited liability systems with the nuclear operator's liability amount being expressed as a *minimum* amount, rather than a maximum one. There will be a phasing-in provision in the Convention by which new adherents may limit their operators' liability to 225-250 million SDRs for a period of 5 years from the date of adoption of the amending Protocol. The provision will only be available to States who join the Convention after January 1, 1999.

The Paris Conversion States will continue to be able to fix reduced liability amounts for low-risk nuclear installations and for transport. The existing Convention prohibits such reduced amounts from being set at lower than 5 million SDRs; under the revised Convention, the minimum reduced amount for low-risk installations will be fixed at 50 million SDRs and for transport, at between 50-100 million SDRs.

Operators will still be required to provide financial security in the amount for which they are liable, and for States with unlimited liability regimes, operators will be required to provide financial security in an amount at least equal to the reference amount or the reduced liability amounts for low-risk installations and transport, whichever is applicable.

Following the amendments contained in the Vienna Amending Protocol, under the revised Paris Convention, prescription and extinction periods for nuclear damage claims will be extended to 30 years for actions respecting loss of life and personal injury, while remaining at 10 years for other types of damage.

As with the revised Vienna Convention and the Supplementary Compensation Convention, the revised Paris Convention will recognize the concerns of coastal states with maritime shipments of nuclear substances through their waters by including new jurisdiction provisions to ensure that where an incident occurs in the exclusive economic zone of a Contracting Party (or in an area not exceeding the limits of such a zone were one to be established) jurisdiction over claims for nuclear damage shall lie with the courts of that Contracting Party.

For the first time ever, the Convention will contain a definition of “nuclear damage” similar, but not identical, to that found in the Vienna Amending Protocol and the CSC; the most striking difference is that under the revised Paris Convention, the definition will not contain the reference to any other economic loss (other than caused by impairment of the environment) permitted by national legislation on civil liability.

The definition of nuclear installation will also be amended to include *all* installations for the disposal of nuclear substances as well as nuclear installations in the course of being decommissioned. Nevertheless, while deciding to include disposal facilities, the Paris States will rely upon the exclusion procedure provided for under Article 1(b) of the Convention to exclude any particular disposal facility in the post-closure phase from the application of the Convention where it no longer poses a significant risk and is therefore is no longer under active surveillance.

Under the revised Paris Convention, operators will no longer be exempt from liability for nuclear damage caused by a nuclear incident directly due to a grave natural disaster of an exceptional character. In addition, Paris Convention States will ensure the payment of damage claims where the operator’s insurance/financial security is unavailable/insufficient, but only up to the reference amount of liability.

5.2.2.3 Liability Boundaries to Generation-IV Development

Given the new geographical scope provision by the Paris Convention, the Vienna Convention and the Joint Protocol, no immediate new Conventions seem necessary if nuclear development would increase significantly.

5.2.3 Nuclear Safety Conventions

The IAEA has introduced in this domain conventions and safety guidelines that will be briefly addressed in what follows.

5.2.3.1 Convention on Nuclear Safety

The Convention on Nuclear Safety was adopted in Vienna on 17 June 1994. The Convention was drawn up during a series of expert level meetings from 1992 to 1994 and was the result of considerable work by Governments, national nuclear safety authorities and the Agency's Secretariat. Its aim is to legally commit participating States operating land-based nuclear power plants to maintain a high level of safety by setting international benchmarks to which States would subscribe. The obligations of the Parties are based to a large extent on the principles contained in the IAEA Safety Fundamentals document "The Safety of Nuclear Installations". These obligations cover for instance, siting, design, construction, operation, the availability of adequate financial and human resources, the assessment and verification of safety, quality assurance and emergency preparedness.

The Convention is an incentive instrument. It is not designed to ensure fulfilment of obligations by Parties through control and sanction but is based on their common interest to achieve higher levels of safety which will be developed and promoted through regular meetings of the Parties. The Convention obliges Parties to submit reports on the implementation of their obligations for "peer review" at meetings of the Parties to be held at the IAEA. This mechanism is the main innovative and dynamic element of the Convention.

The Convention came into force on 24 October 1996. As of January 31, 2000 there are 65 signatories and 53 Contracting Parties. The latter group comprises the following States: Argentina, Armenia, Australia, Austria, Bangladesh, Belarus, Belgium, Brazil, Bulgaria, Canada, Chile, China, Croatia, Cyprus, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Ireland, Italy, Japan, Korea, Latvia, Lebanon, Lithuania, Luxembourg, Mali, Mexico, Netherlands, Norway, Pakistan, Peru, Poland, Portugal, Republic of Moldova, Romania, Russian Federation, Singapore, Slovakia, Slovenia, South Africa, Spain, Sri Lanka, Sweden, Switzerland, Turkey, Ukraine, United-Kingdom, United States of America, EURATOM.

5.2.3.2 The Convention on Early Notification of a Nuclear Accident

This Convention was adopted and entered into force in 1986 following the Chernobyl nuclear plant accident. It establishes a notification system for nuclear accidents which have the potential for international transboundary releases that could be of radiological safety significance for another State. It requires States to report the accident's time, location, radiation releases, and other data essential for assessing the situation. Notification is to be made to affected States directly or through the IAEA, and to the IAEA itself. Reporting is mandatory for any nuclear accident involving facilities and activities listed in Article 1. Pursuant to Article 3, States may notify other accidents as well. The five nuclear-weapon States (China, France, Russia, the United Kingdom, and United States) have all declared their intent also to report accidents involving nuclear weapons and nuclear weapons tests.

As of October 9, 2000, there are 70 signatories and 86 Contracting Parties to the Convention, the latter group comprising the following States: Argentina, Armenia, Australia, Austria, Bangladesh, Belarus, Belgium, Bosnia and Herzegovina, Brazil, Bulgaria, Canada, China, Costa Rica, Croatia, Cuba, Cyprus, Czech Republic, Denmark, Egypt, Estonia, Finland, France, Germany, Greece, Guatemala, Hungary, Iceland, India, Indonesia, Iran, Islamic Republic of, Iraq, Ireland, Israel, Italy, Japan, Jordan, Korea, Republic of, Latvia, Lebanon, Liechtenstein, Lithuania, Luxembourg, Malaysia, Mauritius, Mexico, Monaco, Mongolia, Morocco, Myanmar, Netherlands, New Zealand, Nicaragua, Nigeria, Norway, Pakistan, Panama, Peru, Philippines, Poland, Portugal, Republic of Moldova, Romania, Russian Federation, Saudi Arabia, Singapore, Slovakia, Slovenia, South Africa, Spain, Sri Lanka, Sweden, Switzerland, The Frmr.Yug.Rep. of Macedonia, Tunisia, Turkey, Ukraine, United Arab Emirates, United Kingdom, United States of America, Uruguay, Vietnam, Yugoslavia, FAO, WHO, WMO.

5.2.3.3 Convention on Assistance in the Case of Nuclear Accident or Radiological Emergency

Adopted in 1986 following the Chernobyl nuclear plant accident, this Convention sets out an international framework for co-operation among States Parties and with the IAEA to facilitate prompt assistance and support in the event of nuclear accidents or radiological emergencies. It requires States to notify the IAEA of their available experts, equipment, and other materials for providing assistance. In case of a request, each State Party decides whether it can render the requested assistance as well as its scope and terms. Assistance may be offered without costs taking into account inter alia the needs of developing countries and the particular needs of countries without nuclear facilities. The IAEA serves as the focal point for such co-operation by channeling information, supporting efforts, and providing its available services.

As of October 9, 2000, there are 68 signatories and 82 Contracting Parties to this Convention, the latter group comprising the following States: Argentina, Armenia, Australia, Austria, Bangladesh, Belarus, Belgium, Bosnia and Herzegovina, Brazil, Bulgaria, China, Costa Rica, Croatia, Cuba, Cyprus, Czech Republic, Egypt, Estonia, Finland, France, Germany, Greece, Guatemala, Hungary, India, Indonesia, Iran Islamic Republic of, Iraq, Ireland, Israel, Italy, Japan, Jordan, Korea, Republic of, Latvia, Lebanon, Libyan Arab Jamahiriya, Liechtenstein, Lithuania, Luxembourg, Malaysia, Mauritius, Mexico, Monaco, Mongolia, Morocco, Netherlands, New Zealand, Nicaragua, Nigeria, Norway, Pakistan, Panama, Peru, Philippines, Poland, Republic of Moldova, Romania, Russian Federation, Saudi Arabia, Singapore, Slovakia, Slovenia, South Africa, Spain, Sri Lanka, Sudan, Sweden, Switzerland, Thailand, The Frmr.Yug.Rep. of Macedonia, Tunisia, Turkey, Ukraine, United Arab Emirates, United Kingdom, United States of America, Uruguay, Vietnam, Yugoslavia, FAO, WHO, WMO.

5.2.3.4 Convention on the Physical Protection of Nuclear Material (Shipping)

The Convention on the Physical Protection of Nuclear Material, signed at Vienna and at New York on 3 March 1980, obliges Contracting States to ensure during international nuclear transport the protection of nuclear material within their territory or on board their ships or aircraft.

Pursuant to Article 16 of the Convention, the first Review Conference was held in Vienna from 29 September to 1 October 1992 and attended by 35 States Parties. The Review Conference unanimously expressed its full support for the Convention and urged all States to take action to become party to the Convention. The Parties considered, in particular, that the Convention provides an appropriate framework for international co-operation in protection, recovery and return of stolen nuclear material and in the application of criminal sanctions against persons who commit criminal acts involving nuclear material.

As of April 25, 2001, there are 45 signatories and 69 Contracting Parties, the latter group comprising the following States: Antigua and Barbuda, Argentina, Armenia, Australia, Austria, Belarus, Belgium, Bosnia and Herzegovina, Botswana, Brazil, Bulgaria, Canada, Chile, China, Croatia, Cuba, Cyprus, Czech Republic, Denmark, Ecuador, Estonia, Finland, France, Germany, Greece, Guatemala, Hungary, Indonesia, Ireland, Italy, Japan, Korea, Republic of, Lebanon, Libyan Arab Jamahiriya, Liechtenstein, Lithuania, Luxembourg, Mexico, Monaco, Mongolia, Netherlands, Norway, Pakistan, Panama, Paraguay, Peru, Philippines, Poland, Portugal, Republic of Moldova, Romania, Russian Federation, Slovakia, Slovenia, Spain, Sudan, Sweden, Switzerland, Tajikistan, The Frmr.Yug.Rep. of Macedonia, Trinidad and Tobago, Tunisia, Turkey, Ukraine, United Kingdom, United States of America, Uzbekistan, Yugoslavia, EURATOM.

5.2.3.5 The IAEA Safety Standards

The IAEA Safety Standards comprise publications of a guidance nature. The Standards publications are:

- *Safety Fundamentals:* (F: blue lettering), states basic objectives, concepts and principles of safety and protection;

- *Safety Requirements:* (R: red lettering), establishes the requirements that must be fulfilled to ensure safety for particular activities or applications; and

- *Safety Guides*: (G: green lettering), recommends actions, conditions or procedures for complying with these safety requirements

The following safety guides have been published:

- *Building Competence in Radiation Protection and the Safe Use of Radiation Sources: Safety Guide*, Safety Standards Series No. RS-G-1.4 STI/PUB/1108

- *Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants: Safety Guide*: Safety Standards Series No. NS-G-2.2 STI/PUB/1100

- *Software for Computer Based Systems Important to Safety in Nuclear Power Plants: Safety Guide:* Safety Standards Series No. NS-G-1.1 STI/PUB/1095

- *Safety of Nuclear Power Plants: Operation: Safety Requirements:* Safety Standards Series No. NS-R-2 STI/PUB/1096

- *Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety: Safety Requirements*: Safety Standards Series No. GS-R-1
  STI/PUB/1093

- *Safety of Nuclear Power Plants: Design: Safety Requirements*: Safety Standards Series No. NS-R-1 STI/PUB/1099

- *Predisposal Management of Radioactive Waste, Including Decommissioning: Safety Requirements:* Safety Standards Series No. WS-R-2 STI/PUB/1089

- *Regulatory Control of Radioactive Discharges to the Environment: Safety Guide:* Safety Standards Series No. WS-G-2.3 STI/PUB/1088

- *Fire Safety in the Operation of Nuclear Power Plants:*
  Safety Guide: Safety Standards Series No. NS-G-2.1 STI/PUB/1091

- *Regulations for the Safe Transport of Radioactive Material –*
  1996 Edition (Revised): Safety Requirements: Safety Standards Series No. TS-R-1 (ST-1, Rev.) STI/PUB/1098

- *Decommissioning of Medical, Industrial and Research Facilities: Safety Guide*: Safety Standards Series No. WS-G-2.2 STI/PUB/1078

- *Decommissioning of Nuclear Power Plants and Research Reactors: Safety Guide*: Safety Standards Series No. WS-G-2.1 STI/PUB/1079

- *Assessment of Occupational Exposure Due to Intakes of Radionuclides: Safety Guide*: Safety Standards Series No. RS-G-1.2 STI/PUB/1077

- *Occupational Radiation Protection: Safety Guide*: Safety Standards Series No. RS-G-1.1 STI/PUB/1081

- *Assessment of Occupational Exposure Due to External Sources of Radiation: Safety Guide*: Safety Standards Series No. RS-G-1.3 STI/PUB/1076

- *Safety Assessment for Near Surface Disposal of Radioactive Waste: Safety Guide*: Safety Standards Series No. WS-G-1.1 STI/PUB/1075

- *Near Surface Disposal of Radioactive Waste: Safety Requirements*: Safety Standards Series No. WS-R-1 STI/PUB/1073

5.2.4 Waste Management and Disposal

Nuclear energy systems produce certain amounts of waste during their operation and also at the end of operation, i.e. decommissioning. An increase in deployment energy will result in an increase of radioactive waste to be handled. While the waste production per unit of societal benefit (i.e. sustainability criterion SU2-1), may be reduced, the overall effect of increased deployment will be an increase of the waste amount.

While Gen-IV nuclear energy systems are designed to minimise the production of high-level waste, a difficult technical challenge remains for low and intermediate level waste arising from fuel cycle and reactor operation, decommissioning activities and, in general, secondary waste arisings.

While technologically viable waste management and disposal solutions have been developed and implemented over the past decades, a significant increase in the volume of such LLW/ILW may become an increasingly important issue in the future. Pertinent questions may be posed in relation to the acceptable level of radioactivity contained in these waste and if clearance of certain very low-level radioactive waste types may be obtained (see Section 5.2.4.3).

The disposal of radioactive waste is guided by national regulatory bodies on the basis of national laws and regulations[22]. At the same time, international collaborative efforts have been beneficial to national waste-management programmes. These efforts provide world-wide exchange of technical information and expertise, and internationally recognised, non-binding safety standards and binding agreements among states.

The sharing of insights and resources in co-operative projects has proved valuable to both implementers and regulators. Dialogue and co-operative projects involving both types of organisation, at the international level, have the potential:

- To work towards common understanding of regulatory requirements across different types of waste materials and environmental risks;

- To (at least) rationalise differences between national regulatory guidelines.

22 . See http://www.radwaste.org/laws.htm for a list of national laws and regulations on waste management.

International for a allowing cross-party dialogue and co-operative projects are thus likely to continue to play an important role in the future for all those involved in waste management. Even the most encompassing form of international co-operation – sharing a common international repository in a volunteer host country – is a discussion topic in recent years.

5.2.4.1 Trends in National Laws and Regulations

There is a trend in national regulations towards the use of safety indicators in addition to calculational estimates of radiation dose or risk and the use of modelling approaches for assessing long-term impacts. This trend reflects international discussions, and the work of the International Commission for Radiation Protection (ICRP) and IAEA, regarding the difficulties of demonstrating compliance with radiological and other safety objectives far into the future.

As a result, safety indicators other than radiation dose and risk have been proposed, such as environmental concentrations and biosphere fluxes of radionuclides, as a way of broadening the safety case. In addition, it has been recognised that, although there is no scientific rationale for cutting off repository safety assessments at an arbitrary point in time, the nature of performance assessments over different timescales after closure of a repository cannot be the same. In particular, uncertainties increase with increasing time and so the results of assessments have to be regarded as indicators of safety, rather than as real predictions. These considerations have brought with them proposals to develop internationally agreed stylised approaches for assessing long-term impacts, e.g. the concept of reference critical groups and biospheres.

There is also tendency towards the embedding of national laws or regulations for waste disposal in a wider environmental regulatory framework that can consider the balance with the very-long-term management of non-radioactive hazardous wastes.

In many countries, an environmental impact assessment (EIA) must be carried out for facilities whose construction or operation might result in a significant impact upon the environment. The EIA is carried out by the operator or proponent of the facility and made available for public comment. Currently, however, only partial consistency exists in many countries for issues like parity in assessing the short and long term environmental impacts of alternative energy sources. Moreover, broad-based approaches at the legislative level to assure uniform environmental protection across waste sources which fall under authority of different regulative bodies or for uniformity between nuclear and non-nuclear hazardous waste forms are not yet in place generally.

5.2.4.2 The Joint Conventions on The Safety of Spent Fuel Management and on The Safety of Radioactive Waste Management

The disposal of radioactive waste is guided by national regulatory bodies on the basis of national laws and regulations, but at the same time, international collaborative efforts providing not only the world-wide exchange of technical information and expertise, but also internationally recognised, non-binding safety standards and binding agreements among states have benefited national waste management programmes.

Over that past decade, several binding agreements among sovereign states have been implemented with the support of the IAEA. These agreements are a component of a framework for fostering intergovernmental collaborative efforts in the area of nuclear safety. Specifically, regarding radioactive waste disposal, the "Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management" was adopted in 1997 and, when ratified, will be the first legally binding instrument in this area.

The Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management (the "Joint Convention") was adopted in September 1997 at a Diplomatic Conference convened by the IAEA. Its objectives are threefold: first, to achieve and maintain a high level of safety worldwide in spent fuel and radioactive waste management through the enhancement of national measures and international co-operation; secondly, to ensure that during all stages of spent fuel and radioactive waste management there are effective defenses against potential hazards, so that individuals, society and the environment are protected from the harmful effects of ionizing radiation, now and in the future; and thirdly, to prevent accidents with radiological consequences and to mitigate those consequences should they occur during any stage of spent fuel or radioactive waste management. This convention entered into force on 18 June 2001. As of August 2001, there are 42 signatories and 27 Contracting Parties, the latter group comprising the following States: Argentina, Austria, Bulgaria, Canada, Croatia, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Ireland, Latvia, Luxembourg, Morocco, Netherlands, Norway, Poland, Romania, Slovakia, Slovenia, Spain, Sweden, Switzerland, Ukraine and United Kingdom.

With certain restrictions, the Convention applies to the safety of spent fuel management which is defined as all activities that relate to the handling or storage of spent fuel, excluding off-site transportation, to the safety of radioactive waste management which is defined as all activities including decommissioning activities that relate to the handling, pretreatment, treatment, conditioning, storage or disposal of radioactive waste, excluding off-site transportation, to the safety of management of spent fuel or radioactive waste resulting from military or defense programmes[23], and to discharges which are defined as planned and controlled releases into the environment, as a legitimate practice, within limits authorized by the regulatory body, of liquid or gaseous radioactive materials that originate from regulated nuclear facilities during normal operation.

There are two basic types of obligations that must be undertaken by Contracting Parties to the Convention. The first type comprises general safety obligations in respect of both the safety of spent fuel management and the safety of radioactive waste management that are largely based upon the provisions of the Nuclear Safety Convention and on the principles contained in the IAEA Safety Series document No.111-F, entitled "The Principles of Radioactive Waste Management". These safety obligations extend to both existing facilities and past practices, and to new facilities. With respect to the latter these obligations are specific to the following facility stages: siting, design and construction, safety assessment, operation, disposal, and institutional measures after closure. Contracting Parties are also required to take appropriate legislative, regulatory and administrative measures to govern the safety of spent fuel and radioactive waste management and to ensure that individuals, society and the environment are adequately protected against radiological and other hazards. Such

[23] 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.

requirements extend to the implementation of quality assurance measures, emergency preparedness measures, and license holder and regulatory authority duties and responsibilities amongst other things.

Provisions are also contained in the Convention which address the transboundary movement of spent fuel and radioactive waste. These provisions are largely based upon the IAEA's Code of Practice on International Transboundary Movement of Radioactive Waste and they ensure, among other things, that transboundary movements of radioactive waste take place in accordance with internationally accepted safety standards and respective national laws and regulations. The sovereign right of every State to prohibit the import of radioactive waste into its territory is also recognised.

The second type of obligations imposed upon Contracting Parties are those flowing from the mandatory reporting and peer review mechanism. Under the Joint Convention, national reports are to be provided which address the measures taken by each Contracting Party to implement its obligations under the Convention, and which list all radioactive waste and spent fuel management facilities subject to the Convention, including their location, purpose, essential features and an inventory of spent fuel and radioactive waste that is in storage, that has been disposed or that has resulted from past practices. In addition, the inventory list which is to be submitted to review meetings of Contracting Parties must contain a list of nuclear facilities in the process of being decommissioned and the status of activities at those facilities.

Review meetings are to be held by the Contracting Parties for the purpose of reviewing the obligatory national reports. During these meetings, reasonable opportunity must be given to discuss the reports submitted and to seek clarification of such reports. Contracting Parties are required to attend all such meetings and subsequent to each meeting, the Contracting Parties are to adopt and make available to the public a document addressing the issues discussed and the conclusions reached during that meeting.

This new Joint Convention makes a contribution to what has become a new body of law, the international law of nuclear safety.

5.2.4.3 Exemption and Clearance Levels

The classification of radioactive waste is an important issue as it raises questions about how societies define and regulate all kinds of wastes. So far, national and international agreements have been elusive on two specific and distinct concepts, i.e. exclusion and/or exemption of radioactive wastes from regulatory requirements. In the context of this chapter, the discussion of exemption or clearance, i.e. exemption-a-posteriori, is important in the light of increasing amounts of LLW coming from operation and decommissioning of fuel cycle facilities and reactors.

In its Publication 64, the International Commission on Radiological Protection (ICRP) summarised the current criteria for exemption levels as follows: "In the case of normal exposure, most regulatory systems include provisions for granting exemptions from the regulatory system where it is clear that a practice is justified but regulatory provisions are unnecessary. The grounds for exemption are that the source gives rise to small individual doses (of the order of 10 μSv/yr, or about one hundredth of the average background dose) and the protection is optimised, i.e. regulatory provisions will produce little or no improvement in dose reduction."

The IAEA issued in 1996 as Safety Series No. 115 the "International Basic Safety Standards for Protection against Ionizing Radiation and for The Safety of Radiation Sources". These standards are co-sponsored by all the international organisations with interests in radiation safety. They set down requirements for protection against the risks associated with exposure to ionising radiation. These requirements are based on the estimate of the health effects attributable to radiation exposure, which are periodically submitted by UNSCEAR and ICRP.

In the early 1990s, the IAEA and OECD/NEA had set out the following principles for exemption: (I) individual risks must be sufficiently low as not to warrant regulatory concern; (ii) radiation protection, including cost for regulatory control, must be optimised; and, (iii) the practice should be inherently safe (IAEA Safety Series 89). These principles were later on developed. For instance, the IAEA and OECD/NEA suggested on cost-benefit grounds, that if the collective dose committed by one year of the unregulated practice were less than around 1 man-Sievert, the expected detriment would be low enough to permit exemption without more detailed consideration of other options.

The Safety Standard Series 115 also uses the concept of clearance, i,e., exemption a posteriori. Clearance is hereby defined as "Removal of Radioactive Materials or Radioactive Objects Within Authorised Practices From Any Further Control by the Regulatory Authority". Clearance levels have been, and continue to be, developed for a number of materials. Within the European Union, the Article 31 Group made recommendations on clearance levels for a number of important radionuclides in metals from the dismantling of nuclear installations. The IAEA has developed clearance levels for release of materials from medicine, industry and research and is also developing clearance levels for general applications to any solid material.

Work is needed to reach full international agreement on what constitutes "radioactive wastes" for regulatory purposes and this especially in relation to the exemption and clearance levels.

5.2.5 Radiation Protection

The International Commission for Radiological Protection (ICRP) formulated fundamental radiation principles and criteria for world-wide application. Since the beginning of the decade, the ICRP has established radiation protection principles for a number of practical situations where exposure is not foreseen but at the same time cannot be excluded, and has elaborated its general guidance on policy and ethical considerations relevant to radioactive waste management. The ICRP has also issued advise that is specifically relevant to the performance assessment of repositories for long-lived waste, e.g. the problems associated with a changing biosphere, the rationale for the exclusion of particular scenarios in a safety case presented as part of a licence application and the interpretation of the principle that future generations should enjoy the same protection as the present.

Based on the work of the ICRP, the IAEA has issued internationally recognised non-binding standards on radioactive waste safety, the Radioactive Waste Safety Standards (RADWASS). There are intended to establish an ordered structure for safety documents on waste management and to ensure comprehensive coverage of all relevant subject areas. The IAEA has, more recently, established a working group to explore and, where possible, develop consensus on issues relevant to deep geologic disposal and the findings, when

completed, together with guidance from the ICRP, are likely to be used in developing new safety standards in several nations.

5.2.5.1 Infrastructure Implications on Radiation Protection Institutions

Should a large number of new nuclear power reactors be commissioned in the mid-term, implications on public, worker and environmental radiation protection, most likely would necessitate evolution of the current system of radiation protection as recommended by the International Commission on Radiological Protection (ICRP). Very briefly, the areas most likely to be affected would be the following:

- *Regulatory criteria for the control of public exposures*: One of the pillars of the system of radiation protection is the concept of dose limitation. In the public context, a rationale has been developed to justify the selection of a numerical dose criteria that serve as a regulatory level above which members of the public should not be exposed as a result of the normal operation of radiation-producing activities. Part of the justification of the numerical criteria has been the assumption that it is unlikely that any single member of the public will be exposed to more than two sources of radiation at any one time. Should the number of nuclear power installations, and associated fuel cycle facilities, increase significantly, this numerical criteria should be revisited.
- *Radiological protection of the environment*: The international radiation protection community is currently reviewing its principles for the protection of the environment against harmful effects of ionising radiation. Similarly to the above-mentioned considerations for public exposure control, should the number of nuclear fuel cycle installations increase dramatically, their impact on the environment should be revisited, particularly in terms of any numerical criteria that might be established. (In this context, and with reference to public exposure above, ongoing studies of ecosystems may indicate that some radionuclides can accumulate in particular plant and animal species, thus having a proportionally larger effect on human exposures.)
- *Some governments are currently considering significant reductions in the gaseous and liquid emissions allowed from nuclear power plants*. Future designs could significantly limit fission and activation product build-up within plant systems, thus limiting the potential sources for release. Plant systems could also be designed to hold and "condition" liquid and gaseous effluents such that only solid radioactive waste is generated. The development of such plants would necessitate the review of policy and practice with respect to risk transfers; from the public to workers, and from current generations to future generations.
- *International co-ordination of accident management*: Should the number of nuclear power installations and associated fuel cycle facilities increase dramatically, it is likely that some could be located near national borders. Mechanisms for the co-ordination, across borders, of emergency response planning, preparedness and management activities currently exist; however significant increases in the number of facilities might require new mechanisms and international instruments (bilateral agreements, international conventions, etc.) to effectively and efficiently achieve the level of co-ordination necessary.
- *International co-ordination of the regulatory control of itinerant workers*: With increasing globalisation of the nuclear services market, and given increases in the number of nuclear power installations, it is likely that the employment of foreign-national workers will increase. In order to assure that workers respect

regulatory exposure criteria, institutional agreements and mechanisms may have to be developed to track and log itinerate worker individual exposures in real time.

- Currently, although most countries use the same numerical worker annual dose limit, this is not uniformally true. The United States, for example, has a worker limit of 50 mSv/yr, while most other countries use either 100 mSv/yr, or 20 mSv/yr. Should worker dose limits remain somewhat non-uniform, the managerial problems of *itinerant workers* will be additionally complicated, and international agreements might be necessary.
- In this same context, *institutional agreement would be necessary with regard to worker qualification and education requirements*, such that foreign workers could easily cross national borders and regulatory systems and be accepted as able to work in radiologically controlled areas.

5.3 Summary

The international institutional aspects of the civilian nuclear fuel cycle have evolved in parallel with technical aspects over a 50 year development period, and a number of conventions and norms which govern trans-national aspects of nuclear energy are in place. While these conventions and norms are not yet ratified by all nations which may in the future wish to deploy nuclear energy, the international organizations and the frameworks are in place for expansion of their coverage.

New issues are likely to emerge as the deployment of nuclear energy increases in size and expands to nations not currently generating nuclear power. Among these are trans-national nuclear waste shipments and disposition – which will likely become a desirable option to countries having a small deployment and not wishing to develop an indigenous fuel cycle infrastructure. Similarly, deployment of regional fuel cycle centers which support numerous client nations may become an option for international commerce. Mechanisms for maintenance of real time exposure records for an increasingly mobile workforce crossing national and regulatory boundaries may become desirable.

Finally governmental measures to incentivize the achievement of balance in the flows of fissile materials in a global nuclear energy park comprised of diverse types of power plants exchanging mass flows in symbiotic transactions may become required. Such governmental interventions on free market mass exchanges would be a natural extension of current interventions in monetary and fiscal policy, trade tariffs and agreements, tax incentives, etc. – interventions on the functioning of free markets which are the everyday activities of governments as they seek to maintain stability in flows of commerce and finance.

Generation-IV Fuel Cycle Crosscut Group Report

Chapter 6

Summary, Conclusions, and Recommendations

March 18, 2002

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Chapter 6

Table of Contents

Chapter 6: Summary, Conclusions, and Recommendations

Background

The Charter of the Gen-IV Roadmap Fuel Cycle Crosscut Group (FCCG) was to (1) examine the fuel cycle implications for alternative nuclear power scenarios in terms of Gen IV goals and (2) identify key fuel cycle issues associated with Gen IV goals. This was to include examination of "fuel resource inputs and waste outputs for the range of potential Gen-IV fuel cycles, consistent with projected energy demand scenarios."

The membership of the FCCG comprised 9 US members and 8 members from Generation 4 International Forum (GIF) countries including members from the OECD-NEA, the IAEA and the European Commission observer organizations to the GIF. Members of the FCCG were, in general, drawn from the Technical Working Groups (TWG's) and the Evaluation Methodology Group (EMG) of the Gen-IV Roadmap organization. Five one-day working meetings were held between February 2001, and August 2001 – three of them in conjunction with Gen-IV TWG quarterly meetings and a draft report was issued on November 1, 2001. The FCCG undertook supplemental work between November 1 and March, 2002 – which is incorporated in the final report issued in March 2002.

6.1 Summary of Report Contents

The Gen-IV goals, in addition to goals concerning safety/reliability and affordability include goals for three elements of sustainability: efficient use of resources; ecologically friendly waste management; and resistance to use of the Gen-IV fuel cycle for proliferation of weapons of mass destruction or nuclear terrorism. These sustainability goals depend strongly on choices made for Gen-IV fuel cycles, and attention to these goals has been central to all the FCCG's activities. In the following sections of this chapter, we describe the process by which the FCCG addressed its charter, the lessons learned, and the conclusions and recommendations for the Gen-IV path forward on issues controlled by the fuel cycle

Chapter 1 Synopsis

First, the conceptual framework for the FCCG work effort was laid out in Chapter 1. The FCCG reviewed energy projections and selected the authoritative IIASA/WEC projection of 1998 as the basis for performing a selected set of 100 year nuclear energy futures scenarios. We reviewed the uranium ore resource projections of the OECD-NEA, IAEA, and Uranium Institute, the thorium ore resource projections from multiple sources, and investigated independent models for prediction of new ore discoveries and their exploitation vs cost of supply. (See Fig. 6.1 which illustrates uranium supply vs assay.)

A challenge for Gen-IV is to meet a potential global nuclear energy demand, projected by IIASA/WEC, of ~2000 GWe (by 2050) and ~6000 GWe (by 2100) within affordable ore reserves, which currently are estimated at ~15 million tonnes of U

recoverable at <130 $/kg, and to do so in an ecologically friendly, safe, and affordable manner. The FCCG decided to organize its efforts to evaluate how best to deploy Gen-IV concepts to accomplish this by considering four generic fuel cycles (see Figure 6.2) – once-through, partial recycle, full fissile recycle, and full actinide recycle. These cycles progressively use more of the ore as fuel and send less of it to waste – simultaneously affecting sustainability goals SU-1 (resource) and SU-2 (waste) – (See Fig. 6.3).

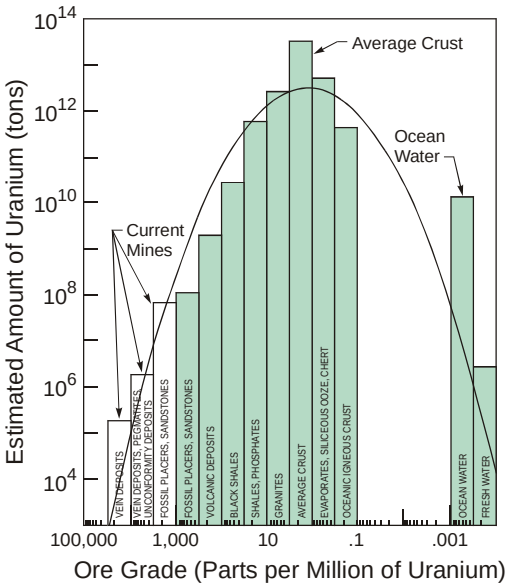


Fig. 6.1 Distribution of Uranium in the Earth's Crust

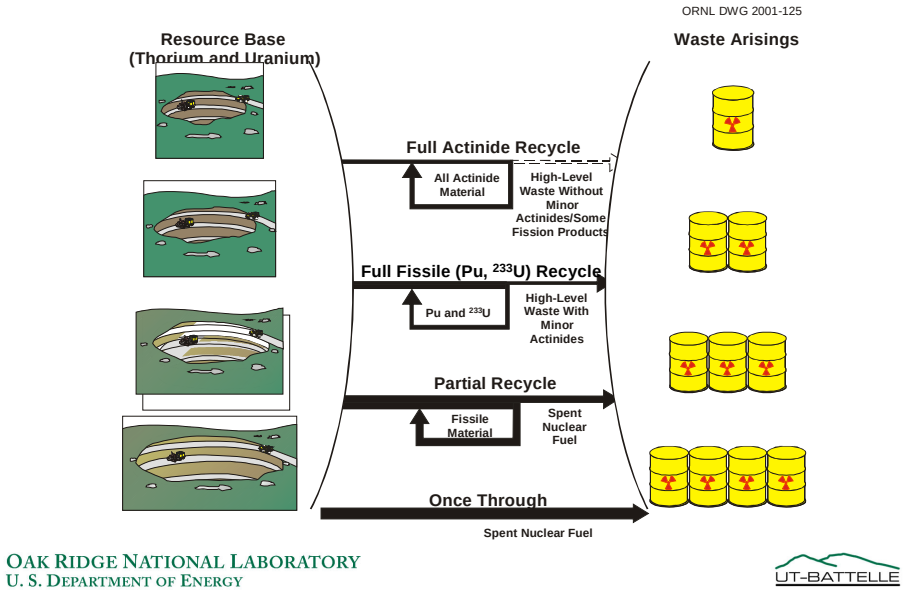
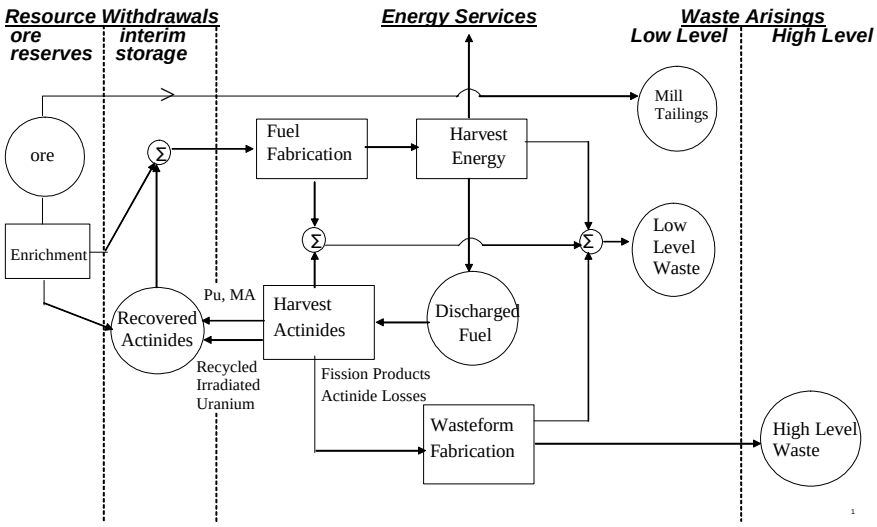


Fig. 6.2 Four Alternative Fuel Cycles Have Been Defined



The Fuel Cycle

	Natural Resources		Fate of Fuel Resource/Waste				Technology Required			
	Virgin Ore	Enrichment Tails	Discharged Fuel				Recycle Technology Used	Remote Refab Used	Thermal Reactors Used	Fast Reactors Used
			Plutonium	MA	Irradiated Uranium	FP	No		Yes	No
Once Thru	Fuel	Interim Store	Waste	Waste	Waste	Waste		No		
Partial Recycle	Fuel	Interim Store	Fuel From UOX	Waste From UOX	Interim Store/From UOX	Waste From UOX	One Pass			
			Waste MOX	Waste MOX	Waste MOX	Waste MOX				
Full Pu Recycle	Fuel	Fuel	Fuel	Waste	Interim Store	Waste	Multiple Recycle			Yes
Full TRU Recycle	Fuel	Fuel	Fuel	Fuel	Fuel	Waste		Yes		

Designation of Resource and Waste for the Four Generic Fuel Cycles

Fig. 6.3 Progression of Generic Fuel cycles: Transforming Wastes to Resources

Chapter 2 Synopsis

Next, the current scope of worldwide deployment of the nuclear fuel cycle practices and facilities was reviewed. Except for non-commercial ship propulsion, nuclear energy is used primarily for electricity generation. Nuclear energy currently supplies 17% of all the world's electricity (i.e., about 6% of all the world's primary energy use). Only two generic fuel cycles are deployed currently; the once-through fuel cycle using UOX fuel which has dominated the market for 30 years, and the MOX mono recycle (using PUREX reprocessing) cycle which has been commercialized in Europe during the past decade and will soon enter the energy mix in Japan.

The fleets of deployed thermal reactors (LWR's and CANDU's) are operating at levels of remarkable safety and reliability following a 15 year campaign of continuous improvement in management, operations, and regulatory practices. Most of the 100 GWe of deployed plants in the US are expected to file for and receive license extensions for twenty more years of continued operation. Consolidation of operator companies has resulted from deregulation, and existing US plants are viewed by the financial community as good investments. New nuclear capacity is being added primarily in regulated markets (in Japan, Korea); whereas the financial conditions currently favor combustion gas turbine plants in the US.

The world market in uranium mining and enrichment is experiencing a period of oversupply which is expected to last for 10-15 more years. There is currently little incentive for ore prospecting or for R&D on improved mining and enrichment technologies.

The fuel cycle currently contributes about 20% to the cost of nuclear energy production (see Fig. 6.4).

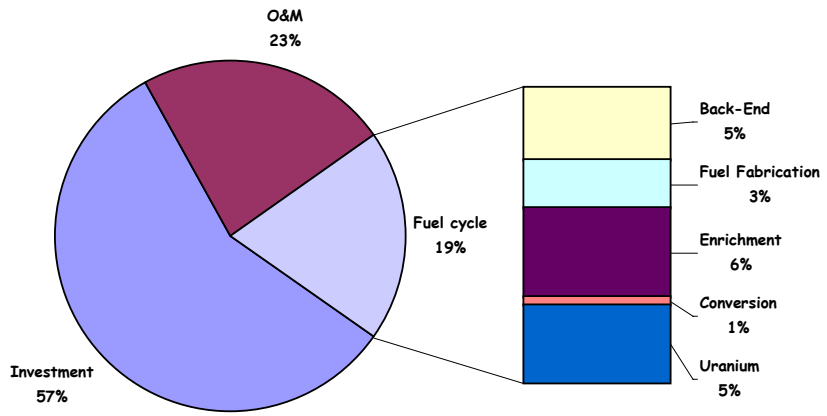


Fig. 6.4 Nuclear Electricity Generation Cost Are Not Currently Controlled By The Fuel Cycle

The issue for the nuclear fuel cycle which remains to be demonstrated is an operating regime of waste management. Discharged fuel remains for the most part in interim storage. One geologic repository (WIPP near Carlsbad, NM) has recently been licensed and is routinely receiving waste from the US nuclear weapons complex remediation program. Repositories for commercial nuclear waste are being researched and/or developed in numerous countries; Yucca Mountain in the US will likely be the first to enter into licensing review.

A survey of Gen-IV concept submittals was made to define the scope of proposed future fuel cycles and fuel compositions to be addressed in FCCG deliberations. Potential Gen-IV concepts exist for fuel compositions of oxide, nitride, metal, and carbon compounds and in forms of ceramics, metal alloys, coated particles, dispersions and liquids. Both uranium and thorium are considered as fertile material, and inert matrix fuels containing no ^{238}U or ^{232}Th are considered as well. Recycle options include both aqueous approaches and non-aqueous, and aim to recycle either plutonium alone or to recycle all transuranics. Refabrication technologies depend on this recycle distinction approach and are both contact handled in glove boxes (for high decontamination recycle products) and remotely in hot cells (for low decontamination recycle products).

The results of the survey helped the FCCG determine the scope of its review of technological status for each link in the fuel cycle chain.

Chapter 3 Synopsis

Results from one hundred year scenarios of potential nuclear futures were evaluated in Chapter 3. While representing only a subset of all of the possible fuel cycles, these scenarios provided numerous insights concerning achievability of the sustainability goals. The scenarios were driven by the IIASA/WEC projections of 21st century world energy demand and nuclear's share in it – from their Case B "just muddling through" scenario where nuclear, in the future, maintains it's current ~20% market share of electricity. Scenarios were considered for each of the four generic fuel cycle types; they were represented as transitions from the current situation, and were constrained only by physically achievable mass flows. Modeling was done at a coarse level; power plants were characterized by working inventories, heavy metal mass flows in and out of the plant and construction and refueling time intervals. Fuel cycle facilities were characterized in a similar manner and by their operational losses to waste. Ore withdrawals, waste arisings and inventories of in-process material were established for each scenario. The contribution of the fuel cycle services to the cost of energy from the global nuclear power park were estimated based on consensus unit process costs developed by the OECD-NEA. The scenario calculations were all repeated for a second, (lower) rate of demand growth to evaluate the sensitivity of the results to growth rate; the sensitivities of the fuel cycle cost index to ore price and to selected unit process costs and waste management costs were also determined. Lessons learned from the scenarios and sensitivity evaluations are described in Section 6.2, below.

Chapter 4 Synopsis

The FCCG surveyed the status of technology and the ongoing R&D for each link of the fuel cycle chain for each of the several generic fuel cycles. Table 6.1 indicates the current status of R&D activities worldwide. Currently there is little R&D focus on prospecting, mining, and enrichment technologies because the market is currently

Table 6.1 Worldwide Fuel Cycle Technology Status and R&D

Deployed
- Water Reactors/UOX Fuel: Dominates Commercial Power Plants
- Once-Through UOX Cycle: Dominates Commercial Fuel Cycles
- MOX Mono-Recycle (1/3 core loading) PUREX: commercialized in Europe and is starting in Japan

Nearly Ready to Deploy
- Multi (Several) Recycle LWR MOX: PUREX Pu Recycle
- Enriched U Coated Particle HTGR Once-Through
- Na-Cooled Fast Reactors with MOX Full Pu Recycle

Substantial Level of R&D Completed
- 100% MOX Core Loading
- Na-Cooled Fast Reactors: U/Pu/Zr Metal Alloy Pyro/Casting Full TRU Recycle

Active R&D
- Modified/Advanced PUREX $\Rightarrow$ Recover MA for recycle, Co-recover U and Pu
- Pyro and Other Dry Recycle $\Rightarrow$ Codeposit all TRU
- Simplified/Remote Fab for Radioactive Fuel: Simplified Pellet, Vibro, Casting
- Nitride fuel, Inert Matrix MA or Pu Fuel (for ADS missions)
- Particulate Fuels
- Thorium-based Fuels
- Tailored Waste Forms from Reprocessing: Glass, Ceramic, Metal Alloy
- Interim Storage Technologies
- Repository Site Characterization

Currently Dormant
- Prospecting, Mining/Milling Technologies
- Advanced Enrichment Technologies (some work in France, Japan and Australia)
- Fuels Designed Specifically to also be Waste Forms
- Integrated Waste Management Approaches: Repository/Interim Storage/Processing as a Coordinated System to manage decay heat and Extend Repository Capacity
- Integrated Intrinsic/Extrinsic Safeguards Regime for Future Fuel Cycles

oversupplied. Substantial activity on recycle/refabrication technologies is ongoing – in most cases in connection with the management of the back end of the fuel cycle. The recycle/refabrication technologies are integrated with fuel composition and form, and new forms – metal alloy, nitride, cermet, – are being developed in connection with full TRU recycle. In that same regard, long term interim storage of discharged fuel is under investigation in Europe. Activities on geologic repositories are strongly focused on site

characterization and developing a basic understanding of geochemical processes for retardation of waste specie migration in the very long term.

Chapter 5 Synopsis

The FCCG, with input from the legal staff of the OECD-NEA and the State Department, undertook a broad review of the current institutional basis of international commerce in the nuclear fuel cycle.

Treaties and norms are already in place in the areas of:

- Nonproliferation,
- Damage liability and compensation,
- Nuclear safety conventions and IAEA Safety Standards,
- Convention on early notification of a nuclear accident,
- Convention on assistance in nuclear accidents or radiological emergencies,
- The international regime of radiological protection (including shipping), and
- Conventions on spent fuel management and radioactive waste management (including high, intermediate, and low level waste).

Currently these treaties and norms are incompletely ratified by all relevant parties. Although much remains to be completed, the current status of international institutional fuel cycle related treaties and norms provides a well-founded basis to move forward in the further deployment of nuclear energy worldwide.

6.2 Lessons Learned from the Scenarios

One hundred year nuclear futures scenarios for each of the four generic fuel cycle types were evaluated – starting from current conditions and assuming initiation of market penetration by Gen-IV systems during the time interval 2010 to 2035 – depending on concept type.

The first case was for a 100% dominance of market share by LWR's operating on the UOX once-through cycle throughout the 21st century. This served as the reference base case against which the performance of alternative Gen-IV concepts in achieving Gen-IV goals could be compared. Figures 6.5 through 6.8 show the base case results.

Table 6.2 enumerates the other cases which were run. Selected illustrative results from these Gen-IV scenarios are compared to the base case in Figures 6.9 through 6.12. The overall nuclear energy park ore withdrawals, waste arisings and fuel cycle costs of the alternative Gen-IV scenarios are ratioed year-by-year to those obtained for the reference base case and are displayed in the figures along with market share of various concepts. These performance ratio comparisons are plotted on semi-log scales – with year on the horizontal (linear) axis and and ratio of performance to the base case on the vertical logarithmic axis. The principal lessons learned include the following:

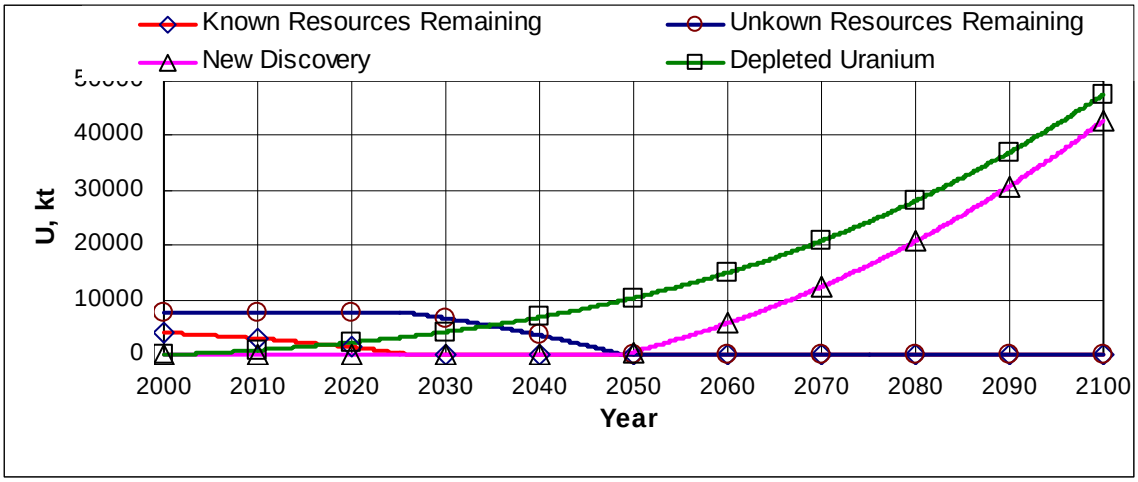


Fig. 6.5 Resource Use for LWR Once-Through Case

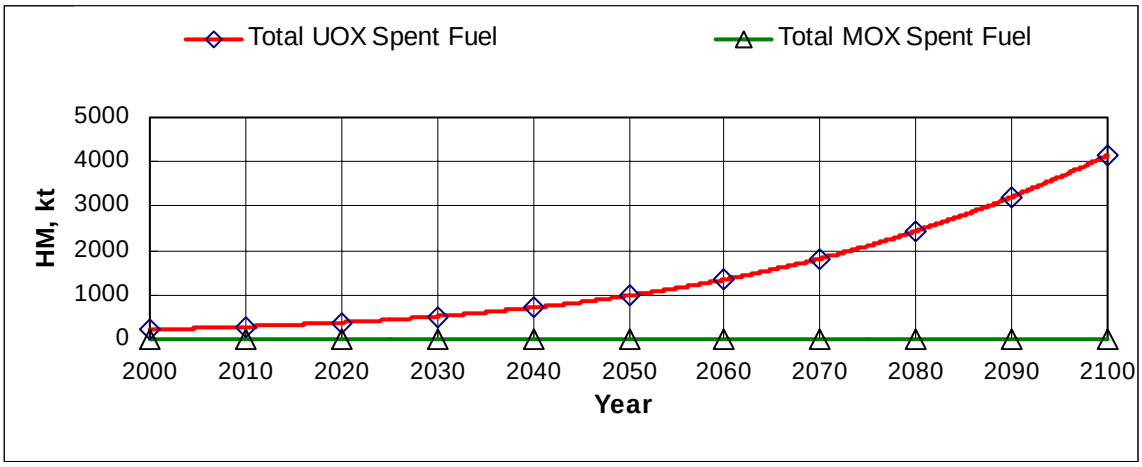


Fig. 6.6 Spent Fuel Accumulation for LWR Once-Through Case

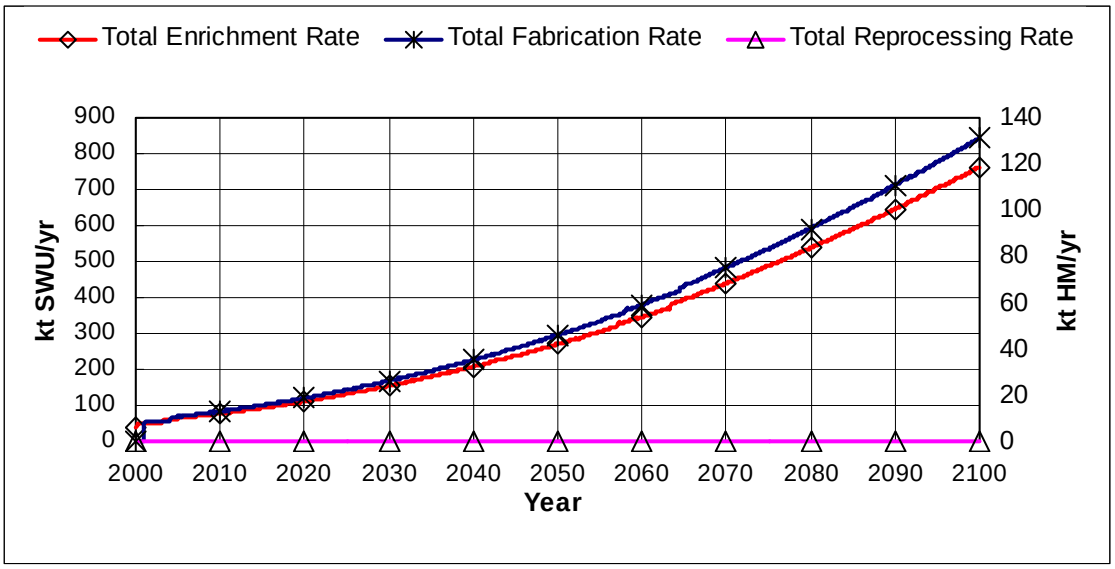


Fig. 6.7 Enrichment and Fabrication Rates for LWR Once-Through Case

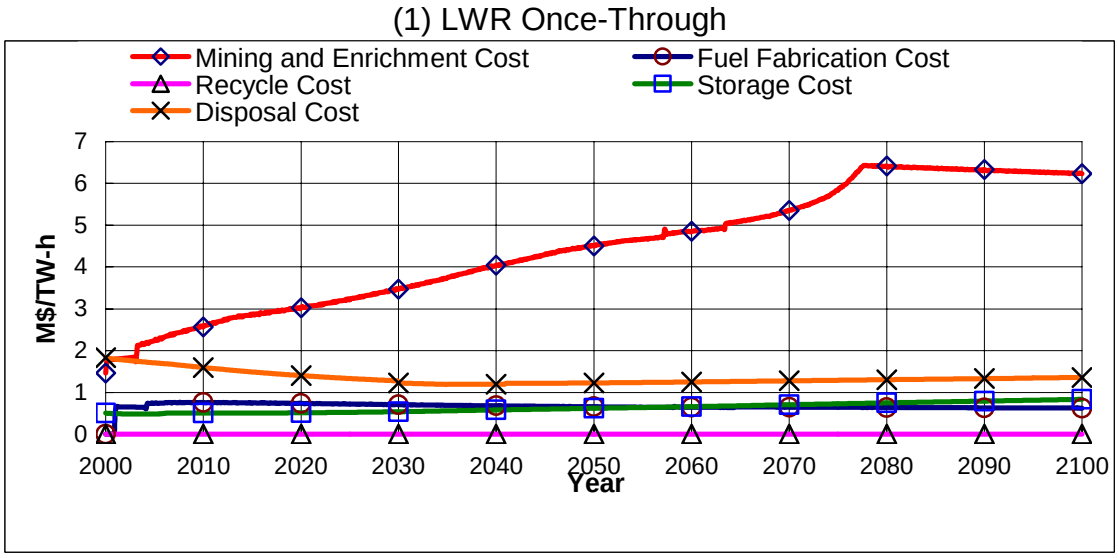
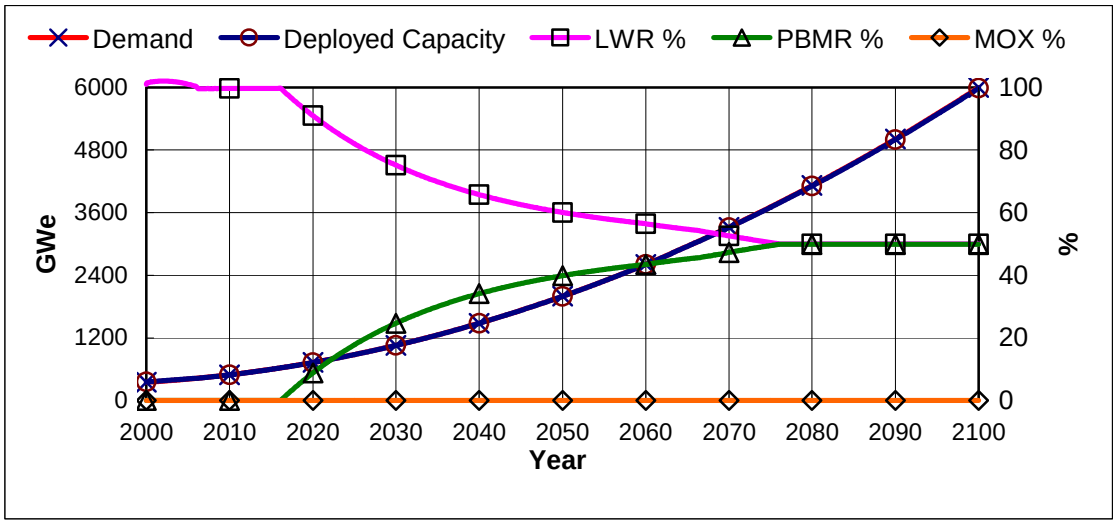


Fig. 6.8 Cost Indices for LWR Once-Through Case

Table 6.2 Scenario Cases Examples to Illustrate Fuel Cycle Issues

- Use Case B of WEC/IIASA for World as a Whole; Use Case C2 to Assess Sensitivity to Demand Growth Rate
- Assume 90% of already deployed Gen 3 LWR's worldwide will receive 20y life extension
- Cases 1 and 7 are the extremes; setting physically achievable bounds; Cases 2 thru 6b and Case 8 allow for practicalities of markets

	Case B		Case C2	
Example Cases	Section	Figures	Section	Figures
A. Once-Through Thermal			All are	
1. New Additions = LWR UOX Once-Through	3.2.1	4-6	in	4-6
2. LWR UOX Once-Through; New additions (after 2010) HTGR Gas Turbine UOX Once-Through (50%) + LWR-UOX Once-Through (50%)	3.3.2 & 3.3.3	7-12	Attach. 3	7-12
3. Radkowsky Th Fueling in Existing PWR's	3.7.1	33-36		Not Run
B. Limited Thermal Recycle (after 2010)				
4. New Additions = LWR UOX → LWR MOX mono → Waste	3.4.1	13-15		13-15
5. DUPIC PWR/CANDU Symbiosis	3.4.2	16-18		16-18
C. Full TRU Recycle: Symbiotic Fast/Thermal (after 2025) for Waste Management				
6a. New additions = 80% LWR (UOX) + 20% Fast Burner (BR = ½) Waste Self Consumption Cycle	3.5.1	19-21		19-21
6b. New additions = Fast Burner (BR = 1) (use up LWR discharged fuel + future waste self consumption)	3.5.2	23-25		23-25
D. Full TRU Recycle: Breeding (after 2030) for Limiting Ore Drawdown	3.5.3	27-29		27-29
7. Start with Case 6b – followed by Switching in 2030 to New additions = Fast Breeders (BR = 1.7)				
E. Full TRU Recycle: Symbiotic Feedback of Fissile to Thermal Systems	3.5.4	30-32		30-32
8. New additions = fast breeders + LWR_MOX s(TRU)				
F. Transition to fissile self sufficient Molten Salt Closed Thorium Cycle	3.7.2	37-39		37-39



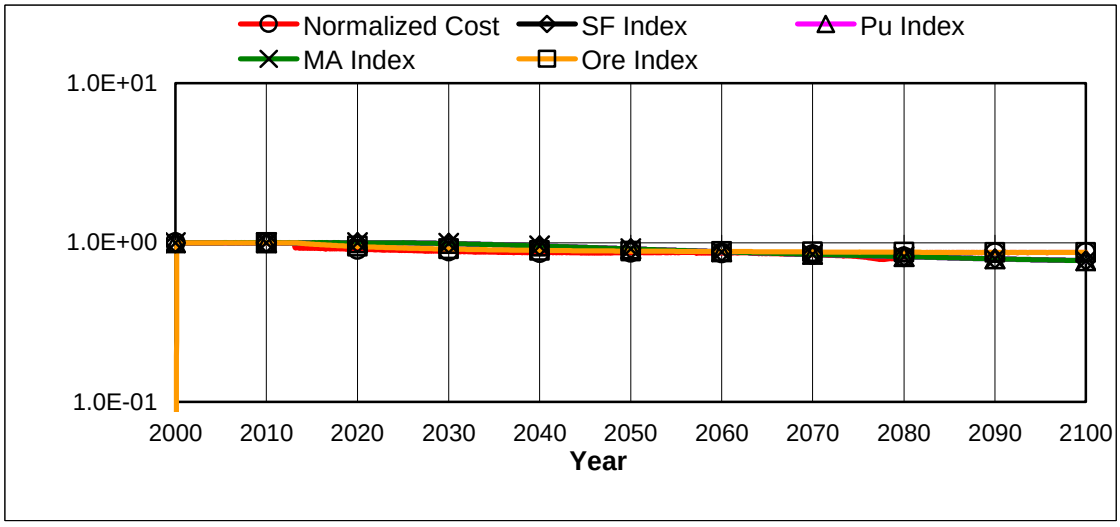
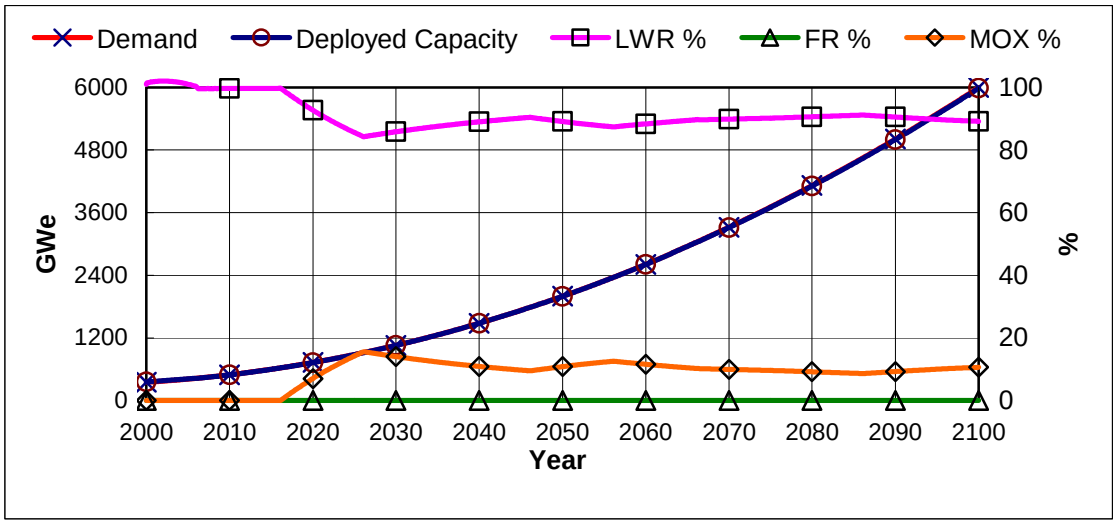


Fig. 6.9 Transition to 50/50 LWR-UOX and PBMR Market Share; Year by Year Ratio of Performance to the LWR-Once-Through Base Case Performance



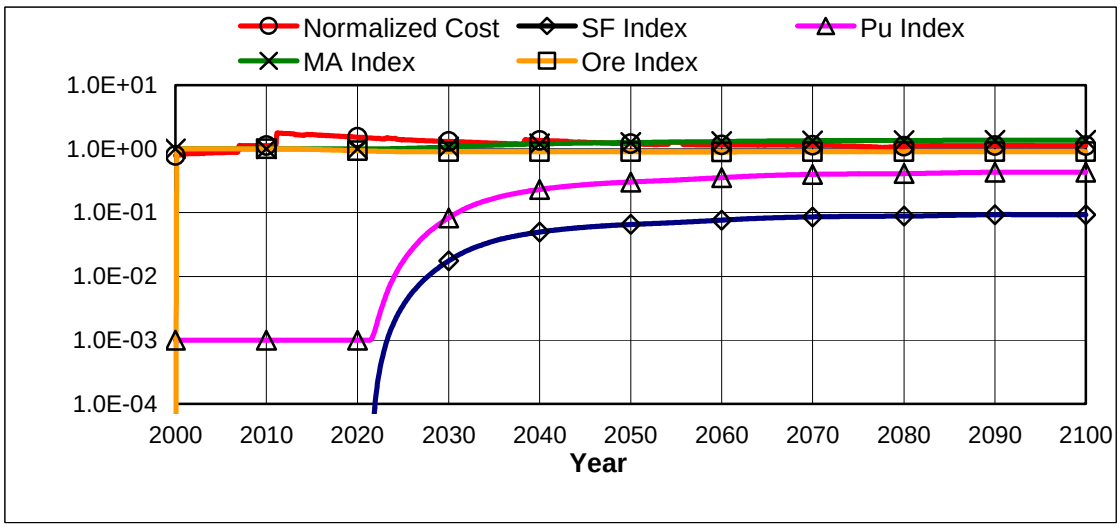


Fig. 6.10 Transition to 90% LWR-UOX/10%: LWR MOX Mono Recycle; Year by Year Ratio of Performance to the LWR Once-Through Base Case Performance

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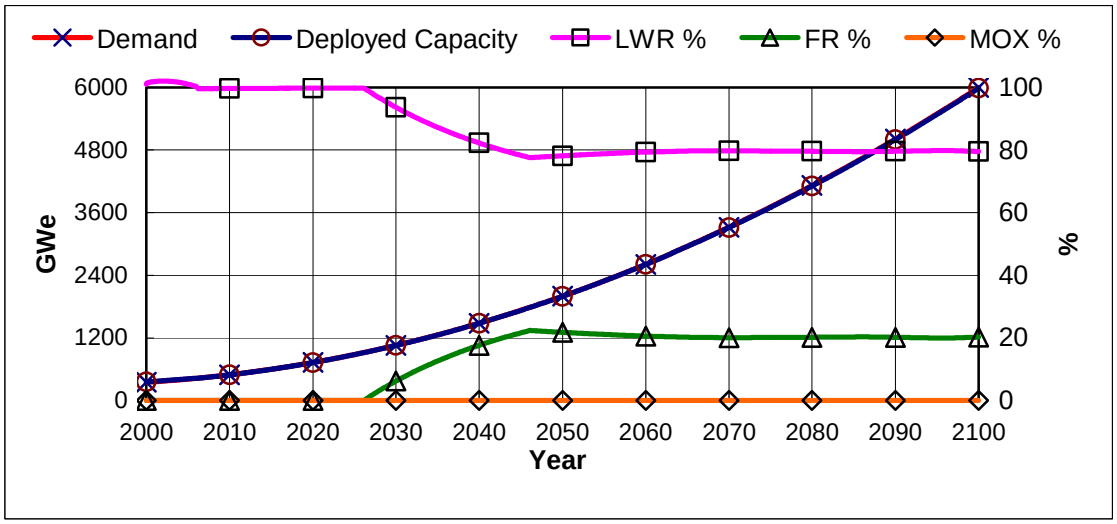

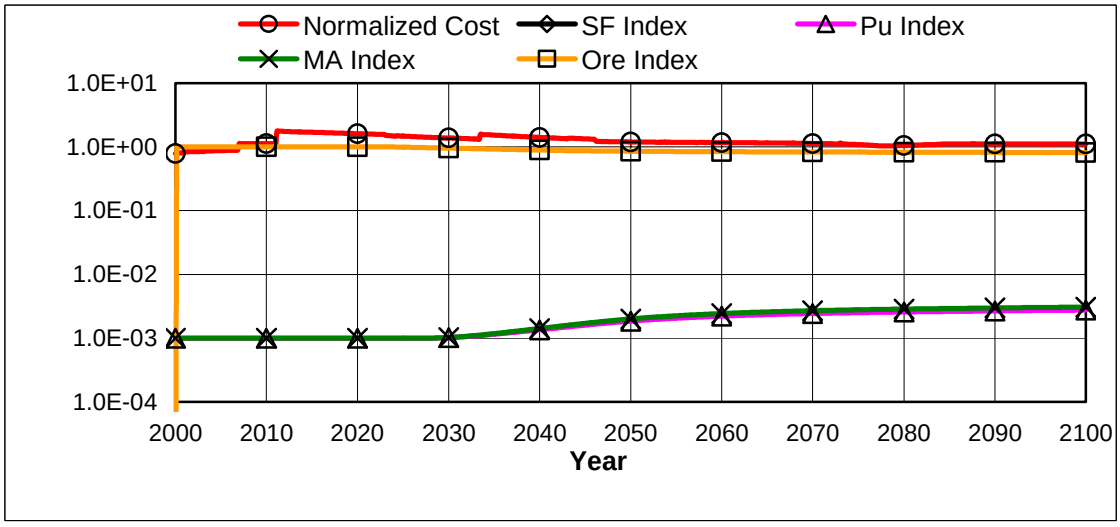

Fig. 6.11 A LWR-UOX/Fast Burner Reactor Symbiosis; Year by Year Ratio of Performance to the LWR Once-Through Base Case Performance

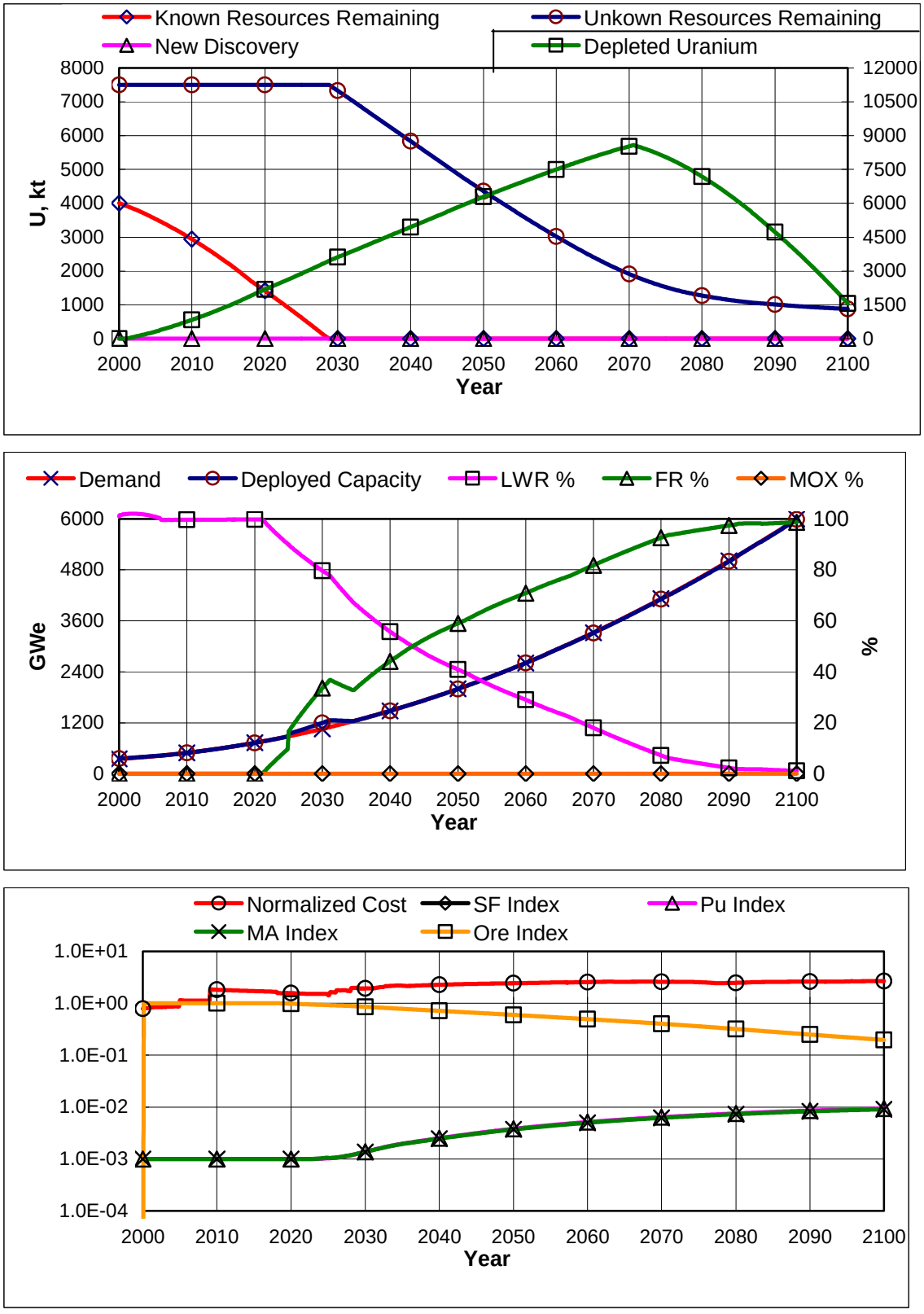

Fig. 6.12 Switch from Fissile Self Sufficient to Breeder in 2030

1. The “business as usual” base case will exhaust the Red Book Known plus Speculative uranium ore reserves by mid century. The worldwide spent fuel waste arisings by then will exceed the legislated capacity of Yucca Mountain (70,000 tonnes) by a factor of 10, and will be growing at a rate of one additional repository of that capacity every two years. Despite the increase in cost of ore with diminishing reserves, the fuel cycle contribution to the cost of energy remains smaller than the contribution from construction and operation of the power plants.
2. More efficient use of fuel via higher station thermodynamic efficiency and/or higher discharge burnup of fuel – illustrated through introduction of gas cooled thermal reactors – will moderately slow the rate of buildup of waste arisings but does not diminish the rate of ore withdrawals because of higher enrichment used for the HTGR or PBMR fuel.
3. Partial recycle of plutonium via MOX mono recycle in LWR’s in a 90% LWR-UOX/10% LWR-MOX mono recycle energy park (which balances mass flows)very significantly reduces the buildup rate of waste mass requiring geologic isolation – by recovering the 96% uranium content of the UOX spent fuel and setting it aside in interim storage. The discharged mono-recycled LWR MOX fuel remains as a waste arising as do the fission products and minor actinides separated from plutonium in the UOX reprocessing operation. Long term radiotoxicity of materials requiring geologic isolation remain relatively undiminished from the base case. Ore withdrawal rate is moderately reduced. The cost of reprocessing offsets the fuel cost increase of ore withdrawals and cost of waste management, but in any case the fuel cycle component of the cost of energy production remains a non majority contributor.
4. Waste arisings per unit of energy production is dramatically reduced when multi recycle through fast neutron spectrum reactors is introduced into the mix of power plant and fuel cycle types. The reductions come about through reprocessing LWR discharge fuel to supply the initial working inventories of new fast burner reactor deployments. Energy shares of 80% LWR’s/20% fast burner reactors bring the mass flows into balance. Ore drawdown is not significantly reduced (i.e., only 20%) but waste mass arisings are reduced by a factors of several hundred and toxicity sent to waste is reduced by nearly a thousand.
5. Various symbiotic couplings among fast and thermal power plants is illustrated by the scenarios. Fast burner reactors are deployed with thermal spectrum reactors to manage waste; the net result is a “clean waste” (fission products and trace losses of TRU) as the only waste emerging from the energy park and a reduced ore drawdown rate. All scenarios maintain more than 50% of the energy park as thermal reactors at least until mid century.
6. In one illustration, excess fissile generated in breeding blankets of fast reactors is harvested and fed back to fuel the deployment of thermal LWR-MOX mono recycle reactors or of gas cooled reactors to supplant a

continuing need for withdrawals of virgin ore reserves. An example for use of excess fissile beyond needs for electricity shows potential to penetrate into non-electric energy sectors such as hydrogen production.

7. Scenarios were run for a once-through thorium concept suitable for refueling current PWR's (the Radkowsky concept) and a Gen-IV closed thorium cycle concept (the molten salt, fluid fuel) concept. Because of use of high U235 enrichment in "seed" pins in the Radkowsky reactor, no benefits on ore utilization were achieved. Alternately, important reductions in ore drawdown were achieved with the closed thorium cycle. Specifically, the scenario for the molten salt reactor (MSR) involved drawing down the inventory of spent fuel discharged from LWR's to provide initial inventories for the fissile self sufficient MSR closed cycle, and because of the exceedingly small fissile working inventory of the MSR and its conversion ratio of unity, the inventory of LWR spent fuel was sufficient to support the growth of the MSR deployments well into the twenty-second century.
8. Taken together the scenarios illustrate the ability – within physically achievable mass flows – to transition to a ^{238}U resource economy when availability of high assay ore reserves diminish and to thereafter supply the fueling needs of the symbiotic mix of fast and thermal reactor power plants in the nuclear energy park by drawing from the inventories of enrichment tails and recovered irradiated uranium – inventories sufficient to fill many centuries of energy supply needs. The switchover can be made to occur seamlessly if a segment of the energy park had already been filled by fast reactors operating in a burner or fissile self sufficient mode – because only a change in core reloading pattern is needed – rather than the deployment of an entirely new fleet of previously undeployed reactor types and fuel cycle infrastructure.

The conclusions and recommendations for the R&D path forward are based in part on these scenario results, which show what elements in the fuel cycle must be emplaced in order to achieve Gen-IV sustainability goals. They are also based on an evaluation of the current status of worldwide fuel cycle R&D which shows what technologies are likely to be feasible with completion of further R&D. These R&D status and gaps are described in Chapter 4; they are summarized next.

6.3 Technology Gaps and Recommended Fuel Cycle Crosscutting R&D by Generic Fuel Cycle Type

The FCCG used the four generic fuel cycle types (once-through, partial recycle, full fissile and full actinide recycle) as an organizational mechanism for initial identification of Gen-IV fuel cycle crosscutting R&D. Additionally the notions of an input-output process (what material enters a system as fuel, how efficiently it is converted to useful energy services, and what material emerges as waste) were used to identify R&D that could enhance the effectiveness of each of these three functional links in the fuel cycle. Figure 6.3 illustrates the fuel sources and wastes for each of the four generic fuel cycles. "Cross cutting" R&D opportunities (i.e., R&D activities that are common to many or all technologies considered) are identified in this section for each fuel cycle type.

6.3.1 Cross Cutting R&D for Once-Through Fuel Cycle

For once-through fuel cycles, Table 6.3 lists the fuel cycle-related R&D areas that could improve the resource input, conversion, and waste output links in a manner to further enable achievement of Gen-IV goals. Because virgin ore and existing stocks of enrichment tails comprise the fuel resource for once-through cycles, development of cost effective enrichment technologies and development of improved prospecting and mining/milling technologies are stressed. For the conversion link, development of low-enriched/high-burnup fuel and/or high temperature fuel (to support higher station thermodynamic efficiency) could improve conversion efficiency. For the waste management link, the discharged fuel is itself the waste form, and development of fuels that not only meet the goals identified above, but in addition resist degradation over geologic time scales are needed. Moreover, integrated interim storage/geologic repository logistic strategies that manage decay heat loads should be developed to provide cost-effective ways to extend repository capacity. Near field designs to retard radioisotope diffusion during the post-containment-failure time period deserve continued development as does long-term remote monitoring for reduced stewardship costs of interim storage and of repositories.

Table 6.3 Once-Through Generic Cycles: Fuel Cycle Crosscutting R&D

Going In	Conversion	Coming Out
• Technologies for Cost Effective - Prospecting - Mining - Mill Tailings Management • Enrichment Technologies • Fuel Designs & Fuel Fabrication Methods for Fissile fed-back from recycle or • Low Enriched High Burnup Fuel - epithermal - burnable poison - partial substitution of thorium to replace U-238	• High Burnup Fuel • Fuel Suitable for High Thermodynamic Efficiency - High Temperature fuel - Fuel for Supercritical Water	• Fuel designed to be robust in repository environment for geologic times • Improved Heat Management Strategies for interim storage/repository • Improved near field retardants to diffusion of radioactive species • Improved remote monitoring

6.3.2 Cross Cutting R&D for Partial Recycle Fuel Cycles

The partial recycle fuel cycle types differ from the once-through cycles by the introduction of reprocessing and/or conditioning and refabrication of discharge fuel for mono recycle of plutonium or other fissile materials back into thermal neutron spectrum reactors. This adds a new fuel resource (plutonium and residual fissile constituents in spent fuel) to those already available in the once-through cycles. It also adds the opportunity for customized waste forms for part of the waste – (but not for the discharged mono-recycled fuel, which is itself either destined for a geologic repository or destined for interim storage awaiting deployment of full recycle fuel cycle types).

For the partial recycle generic fuel cycles, Table 6.4 lists the fuel cycle-related R&D areas that could improve the input, conversion, and output links of the partial recycle generic fuel cycles. First, because virgin ore and enrichment tails remain as primary fuel resources, the enrichment, prospecting, mining and milling R&D already identified for the once-through cycles remains a priority. The same commonality applies for development of high burnup and/or high temperature fuel – and for the integrated waste management of mono-recycle fuel.

Table 6.4 Partial Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences relative to once-through fuel cycles: harvesting Pu for return to thermal spectrum systems		
Going In	Conversion	Coming Out
• Same as Once-Through for harvesting virgin ore and enrichment tails		• Recycle Fuel which is robust in repository environment and
• For harvesting fissile from Spent Fuel: - Simplified Pu Recycle (S-PUREX) - Simplified Pu bearing fuel Refabrication - Fabrication using recovered irradiated U - Re-enrichment technologies for recovered irradiated uranium - Reduced Secondary Waste • Remote fuel fabrication for feedstock fed back from recycle • Front end reprocessing technology for high plutonium content fuel	• High temperature fuel • High Burnup fuel • Fuels (and core designs) for multi recycle in Thermal Spectrum Reactors	• Robust Customized Waste Forms MA FP High Heat FPs or Long-term storage Technologies (anticipating future adoption of a full recycle regime):

Fuel recycle offers the possibility for partitioning and production of customized waste forms for at least part of the waste – opportunities which have already been partially exploited in the development of highly robust glass waste forms for fission products and minor actinides. Further opportunities should be developed for integrated decay heat management. Finally, long-term interim storage technologies should be developed for discharged mono-recycled fuel – in anticipation that full-fissile or full-actinide recycle will ultimately be deployed and that the discharged mono-recycled fuel will then be reprocessed. Continued refinement of fabrication technologies for Pu-containing fuels is needed to further reduce worker dose, to reduce secondary wastes, and to lower costs.

For those Gen-IV concepts in which mono recycle is to be extended in the near term to several-time recycle in thermal spectrum systems, numerous developments are needed. These include improved front end steps for dissolving higher plutonium content MOX fuel used in multi recycle, semi-remote or remote fabrication of fuels derived from plutonium feedstock of degraded isotopic mix, and fuel assembly (and core layout) designs for use with multi-recycle fuel feedstock of degraded isotopics.

6.3.3 Cross Cutting R&D for Full Fissile Recycle and Full Actinide Recycle Fuel Cycles

Whereas the once-through and partial recycle fuel cycles are already commercialized, the full fissile recycle and full transuranic recycle fuel cycles are still in the development phase. Both cycles introduce two new elements: fast neutron spectrum reactors and multi recycle.

The higher fissile content of fast reactor fuels changes criticality control limits on all recycle and refabrication links of the fuel cycle chain. Additionally, in the case of full actinide (TRU) recycle, highly radioactive minor-actinide-containing fuel feedstock necessitates that remote refabrication technologies be deployed. Given that remote refabrication is required for full TRU recycle and that fast reactors are tolerant of fission product poisoning, simplifications in reprocessing technology can be considered – eliminating process steps and saving on cost while achieving only partial fission product elimination.

Tables 6.5 and 6.6 list fuel cycle-related R&D for the full fissile recycle and the full actinide recycle generic fuel cycles, respectively. Except for the criticality issues of high fissile content fast reactor fuel which are common to both, the majority of new R&D issues are seen to arise for the full actinide recycle case.

Full recycle, especially full actinide recycle, approaches can be characterized as cycles that produce "clean waste" and "dirty fuel." They are targeted to ease waste management challenges by recycling and self consuming in the reactors themselves many of the radioactive and radiotoxic wastes to avoid their consignment to geologic disposal – shortening the time span of required stewardship. The resulting requirement for remote fuel refabrication drives the need to develop simple, few-step techniques including simplified pelletizing or vibrocompaction for ceramic fuels and casting or powder metallurgical procedures for metal fuels. Dispersion fuel development is also considered. Significant challenges exist to develop fabrication technologies which retain volatile species (e.g., Am) in the fuel form during fabrication and generally to minimize or eliminate losses of the targeted isotopes during recycle and fabrication processes.

New fuel types consisting of compositions containing increased plutonium fraction, new compounds (such as nitrides, oxycarbides, spinels, apatites, etc.) and the potential presence of significant minor actinide fractions, fabricated using remote technologies, are recognized as challenging developments. Such fuels development campaigns typically require ten to fifteen years of significant development cost – requiring both in-pile and out-of-pile testing (including safety tests) before a new fuel is ready for deployment. The fuel is the essential link between the fuel cycle (controlling resource and waste performance against Generation IV goals) and the reactor power plant (controlling cost, reliability, and safety performance against Generation IV goals). Gen-IV reactor concepts and fuel cycles must explore and exploit the potential of a spectrum

of new fuel types which show promise, and the R&D cost of doing so must be a central element of the Generation IV research program.

The recycling of all transuranics including the minor actinides for consumption as fuel in a fast spectrum reactor offers the potential to achieve a dramatic reduction in the toxicity consigned to waste management – factors of several orders of magnitude may be possible. Additional benefits may accrue from substituting thorium as fertile material for some of the ^{238}U in the fuel cycle. The concomitant technical challenge is to develop recycle technologies capable of achieving the very small trace losses to waste during reprocessing and refabrication, at commercial scale, which are required to exploit this potential. Work started in the mid-1980s on this challenge, and it must remain a central theme in the Gen-IV research program.

Under a working hypothesis that the required low level of trace losses from recycle operations can be achieved, not only the number of repositories needed but also the capacities of future geologic repositories can be extended and their functional requirements can potentially be relaxed. For the future, the development of recycle-based fuel cycles should not go forward absent close linkage with the development of new repository designs to exploit opportunities for customized waste forms and optimized decay heat management strategies, which can extend their capacities and reduce the level of required stewardship. Therefore, R&D for new recycle and new repository design options should be closely-linked and evaluated as an integrated element of the Generation IV R&D activities.

Finally, R&D on recycle, refabrication, and fuel designs for recycle of materials derived from full fissile or actinide recycle to thermal spectrum reactors (as well as to fast reactors) should be a part of Gen-IV development. This applies as well to recycle of fissile from thorium closed cycles to fuel thermal spectrum reactors. For the base-line IIASA/WEC scenario, (and absent significant innovations in ore recovery technology) the feedback of fissile material bred in fast spectrum reactors to fuel thermal reactors, or advances in thermal reactor systems to employ thorium and improve neutron economy (e.g. continuous refueling as in pebble bed HTRs and molten salt reactors) will be needed to compensate for dwindling high assay virgin ore resources by no later than twenty years after the Gen-IV deployment date in the 2030's. A proper balance must be determined between the beneficial strategy of "dirty fuel" and "clean waste" (required to meet Gen-IV sustainability goals SU-2 and SU-3) versus the limited ability of current thermal spectrum reactors to accommodate minor actinides, degraded plutonium isotopics and lanthanide fission product carryover. (For example, molten salt reactors, if successfully developed, could accommodate very high minor actinide loading.) This is relevant especially to those simplified, full-TRU reprocessing technologies which coextract all transuranics and/or achieve only partial fission product decontamination – often at the expense of significant carryover of lanthanide fission products into the refabrication fuel feedstock. Optimization of these tradeoffs must be part of the Gen-IV research program.

Table 6.5 Full Fissile Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences from once-through and partial recycle fuel cycles: Multiple recycle in fast-spectrum reactors; exploitation of >99% of Energy Content of Ore		
Going In	Conversion	Coming Out
• Several Century Hiatus in Prospecting/Mining/Milling available for R&D on those issues • Recycle of fast reactor fuels (higher content of fissile) - Large fission product (FP) carryover is tolerable for fast reactors - Large FP carryover is not tolerable for thermal reactors • and/or advanced thermal reactor systems - Thorium-based fuels - improved neutron economy (e.g. molten salt reactors) • Fabrication of fast reactor fuel (high Pu content) • Remote Fabrication Thermal Reactor Fuel if poor FP separation • Simplified recycle Technologies (<10-4 decontamination from FP's)	• Same as once-through and partial recycle • And/or advanced thermal systems (e.g. molten salt reactors)	• Robust Waste Form for FP and TRU • Customized Waste Forms MA FP High Heat FPs or Long-term storage technologies (to wait for full recycle regime)

Table 6.6 Full TRU Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences from once-through, partial recycle, and full-Pu fuel cycles: Recycling of MA
→ requirement for remote fabrication
→ Issue: Can MA + Lanthanides bearing Fuel be used in Thermal Reactors

Going In	Conversion	Coming Out
• Remote Fabrication of very high radiation feedstock - Retaining Am during fabrication • Cost Effective, simplified recycle with secondary waste reduction and high TRU recovery • Integration/Co-siting for Recycle/Refabrication • Accountancy of very high radiation fuels & of processes having no input accountancy tank	• Same as for other fuel cycles	• Less stringent functional requirements on Repository Designs owing to changed source term • Issues - Cost/benefit of isotopic separations to get Cs for high heat separate from Cs for long life toxicity - Cost/benefit of trying to harvest the rare metal FP's

6.4 Principal Findings

On the basis of these reviews, studies and evaluations, the FCCG has produced a set of principal findings and has generated a set of top level recommendations for Generation IV fuel cycle crosscutting R&D.

The FCCG's principal findings are based on two primary sources. First are the results from dynamic scenario simulations of various potential nuclear futures – driven by the 100 year world energy demand projections (and nuclear's share) provided by the 1998 IIASA/WEC forecasts. These nuclear futures scenarios were organized by generic fuel cycle type (once-through, partial recycle, full fissile recycle, and full transuranic recycle) and were constrained only by physically-achievable mass flows and lag times of potential Generation IV power plant and fuel cycle concepts. They modeled idealized transitions from current and near term deployments to Generation IV fuel cycles and power plants and potential symbiosis of mass flow exchanges among Generation IV power plant concept types. These scenarios provide cornerstone indicators for the Roadmap of physically-achievable performance against Generation IV goals.

The second principal input to the FCCG's findings derived from an extensive and deep review of the technical status of fuel cycle technologies deployed and under development worldwide, and an evaluation of the underlying rationale for the choices of research focus that drive these development programs. While the technical approaches vary, it was found that the worldwide underlying motivations are closely aligned to the goals articulated for Generation IV in the areas of Sustainability, Safety and Reliability, and Economics. The fuel cycle plays a primary role in meeting the three elements of the Generation IV sustainability goals.

The principles of sustainability include meeting society's needs for energy services while using the earth's resources in an efficient and environmentally friendly way. Nuclear fission converts uranium and thorium resources to energy with fission products as the essential waste. The net production of long-lived transuranium isotopes is a characteristic of the specific reactor types and fuel recycling steps used. The goals of Generation IV include reduced waste generation and more efficient use of ore resources along with making the nuclear fuel cycle the least attractive route to proliferation of nuclear armaments.

Today the cost of uranium and thorium is not a major contributor to the cost of nuclear energy, and resources do not constrain the expansion of nuclear power. Within several decades the costs of fuel materials may become more significant as lower-grade resources are used. However, repository capacity is an increasingly expensive and politically divisive constraint on growth of nuclear power. The use of fuel cycles and reactors that minimize repository requirements is essential to increased use of nuclear energy.

A. Reversing Waste Buildup in a Growing Nuclear Economy

1. Closed fuel cycles have already demonstrated a significant reduction of the volume and long-term radiotoxicity of nuclear high-level waste through the reprocessing of 20,000 tonnes of spent LWR fuel to recycle the plutonium and uranium. Closed fuel cycles provide the opportunity to partition classes of nuclear waste and to manage each class in a separate waste form according to its individual characteristics. Advanced waste management strategies include transmutation of selected nuclides, cost effective decay heat management, flexible interim storage, and customized waste forms for specific geologic repository environments. These strategies hold the promise to significantly reduce the long-lived radiotoxicity of the waste destined for geological repositories by at least an order of magnitude via recovery of virtually all the heavy long-lived radioactive elements. Such reductions, and the ability to optimally condition the residual wastes and manage heat loads, will permit far more efficient use of limited repository capacity and further enhance overall safety of the final disposal of radioactive wastes. An equivalent reduction in secondary waste arisings is also possible.

2. Advanced once-through cycles also have a potential to provide useful improvements in repository performance, although smaller than closed cycles. These improvements may be achieved primarily through the increase of thermal efficiency and high levels of fuel burnup .

B. Sustainable Use of Resources

3. Virgin uranium ore supplies are assured for several decades but will fall short of demand for low cost uranium by the middle of the 21st century. Timely renewal of exploration campaigns may further extend this virgin uranium supply but probably at a higher cost. However, more efficient use of raw materials is sustainable while reducing the environmental burdens and worker radiological exposures from mining and milling activities. More efficient use of ore can be achieved in three ways: by further extraction of fuel from existing stocks of depleted uranium, by more fuel-efficient reactors, or by recycle of existing stocks of discharged fuel. (Additionally downblending of stockpiles of highly enriched uranium declared to be in excess of security needs and the use of stored uranium inventories will be temporary sources of fuel for the near-term.)

4. Nuclear energy has the important market advantage that its fuel cycle contributes only about 20% to the overall cost of energy. This advantage provides remarkable flexibility for decoupling the strategies for meeting Generation IV economics and safety and reliability goals from the strategies for meeting sustainability goals. Wide ranges of fuel cycle approaches can be undertaken with only weak influence on economics.

Power plant design strategies to meet economic and safety and reliability goals, while intimately tied to fuel design, are only loosely influenced by fuel cycle choices. This flexibility is further enhanced by a symbiotic mix of reactor types.

C. Transition To Sustainability

5. Energy use is expected to grow substantially in this century; consequently nuclear energy deployment is projected by authoritative studies to increase by as much as a factor of five or six by 2050. Over this period the nuclear deployment will require an evolving mix of reactor types (thermal and fast spectrum) and fuel cycles (once-through and recycle) to serve different market sectors. Symbiotic mixes of reactor types and fuel cycles are essential for economically supplying the required energy - while minimizing waste generation and assuring efficient use of resources, including limited repository capacity.

6. New reactor types with favorable neutron economy and a variety of fuel compositions will be required to create a symbiotic mix. These reactors can either produce fuel by breeding or consume nuclear waste by burning – providing the flexibility needed to accommodate economic and social changes. Such reactors, e.g. those employing some combination of fast neutron spectra, altered fertile (U-238, Th-232) feed, reduced neutron losses and/or an external neutron source, rely on fuel recycle to fulfill their function.

   Near term evolutionary modifications of current reactors are needed to fill expanding energy needs while at the same time employing limited repository space with high efficiency. Within the symbiotic mix envisioned for the future, these reactors need not necessarily rely on recycle, but further increases in fuel burnup and sufficient flexibility to be adapted to a future closed fuel cycle are desirable.

7. Technologies for co-recycle of minor actinides with plutonium and/or for provisions of some residual contamination of recycle fuel from fission products are under current development. By focusing on creating clean waste streams (containing only the fission products) these technologies can significantly reduce the quantities of long-lived radionuclides consigned to waste and provide an intrinsic barrier to weapons proliferation. Recycle of these advanced fuels requires the handling of highly radioactive materials in all fabrication activities. Such highly radioactive and fission product contaminated fuels have traditionally been destined for use in the favorable neutron balance of fast reactors. However, in the symbiotic mix of reactor types envisioned for the future, such fuels may be destined for thermal reactors as well. Generation IV R&D must determine the optimal implementation of these options and

develop such enabling technologies as remote fabrication for thermal reactor fuel as well as fast reactor fuel.

D. Fuel Cycle Safety

8. The radiological exposure of workers in the overall fuel cycle includes contributions from mining/milling and from recycling facilities. These worker exposures are limited and are comparable to those from reactor operation. Developments to further reduce radiological impacts, for example by use of already-mined resources and by cost-effective remote refabrication, will become increasingly important as the scale of nuclear power increases.

9. The glass waste form generated by today's fuel cycles has been shown experimentally to be very durable in repository conditions. The long-term behavior of waste forms and their ability to confine potentially mobile radionuclides are important and discriminating issues for comparing different fuel cycle strategies.

E. Thorium

10. Thorium-fueled thermal reactors and uranium fast spectrum reactors may become an attractive option in the longer term because of the depletion of uranium resources. In addition, it is believed that the joint use of thorium and uranium cycles could lead to significant reductions of the long-term radiotoxicity of the ultimate waste because of greatly reduced production of transuranium actinides. In addition, the joint use of both cycles might enhance proliferation resistance through increased U-232 and Pu-238 content in recycle fuel feedstock.

6.5 Recommended Fuel Cycle Cross Cutting R&D for the Gen-IV Path Forward

Given the initial identification of R&D needs for each generic fuel cycle, as developed above in Section 6.3, and given the insights gained from evaluating the scenario results (Chapter 3) and reviewing the current status of knowledge and ongoing R&D (Chapter 4), the FCCG has developed a prioritized set of recommendations for Gen-IV fuel cycle R&D. *This list reflects the FCCG view that priority should be given to those R&D areas which can potentially produce "breakthrough" changes in the effectiveness of the fuel cycle for achieving Gen-IV sustainability goals.*

Results of the scenarios studied by the FCCG indicate that full fissile recycle and especially the full transuranic recycle fuel cycles may be needed to achieve Generation IV sustainability goals SU-1 and SU-2. Once-through power plant concepts, eventually symbiotically fueled with fissile material fed back from full recycle generic fuel cycles, will also be part of an overall sustainable nuclear energy park. Partial recycle generic fuel cycles can function as a bridge from the current situation to the long term full recycle (sustainable) cycles. These observations, combined with the FCCG's conclusion regarding the likely technical achievability of full recycle which is based on the current state of knowledge and ongoing R&D have influenced the selection of Generation IV fuel cycle crosscutting R&D recommended below.

R&D targeted to "breakthrough" enabling technologies have been favored by the FCCG over those directed toward refined understanding and/or incremental improvements in existing technology. While this latter type of "incremental" R&D is essential and important for the continual improvement against Generation IV economic, reliability and safety goals for the global nuclear energy park, it is the "breakthrough" fuel cycle enabling technologies which are needed to attain the Generation IV waste management and resource utilization sustainability goals for the fuel cycle; these are the special domain of the fuel cycle crosscut R&D recommendations.

1. Improvement of fuels and development of advanced fuels is important *no matter what fuel cycle is used*. The fuel assemblies comprise the essential interface between the nuclear power plant and the fuel cycle. Fuel assemblies that achieve high discharge burnup, low reactivity loss with burnup, low fabrication cost and which can operate at high coolant temperature for improved station thermal efficiency will impact directly on economics and sustainability goals and indirectly on safety goals. If the fuel is intended to be used in a recycle-based fuel cycle, it must be designed with ease of recycle in mind; if it is intended for once-through, it must be designed for very high discharge burnup and for extreme robustness over geologic time scales in a repository environment. Finally, because ultimately the source of fissile will be fed back from full plutonium, uranium 233, or full transuranic recycle sources, the fuel must be remotely fabricable.

Development of new fuel types and fuel compositions containing increased plutonium and minor actinide fractions is recognized to be a significant challenge. Fuels development campaigns typically require ten to fifteen years of significant development cost – requiring both in pile and out of pile testing (including safety tests) before the fuel is ready for deployment. While the cost is not to be understated, neither is the payoff; the fuel is the essential link between the fuel cycle (controlling resource and waste performance) and the reactor power plant (controlling cost, reliability, and safety) against Generation IV goals. Uranium oxide fuel is highly refined and dominates the current once-through commercial fuel cycle and is qualified for use in fast neutron spectrum reactors. Nonetheless new Generation IV reactor concepts and fuel cycles must be afforded opportunities to explore and exploit the potential of additional fuel types, and the R&D cost of doing so must be a central element of the Generation IV research program.

2. Cost effective advanced recycle technologies integrated with remote fuel refabrication technologies are the key enabling technologies for achieving the Generation IV sustainability goals. In the full recycle fuel cycles, the waste destined for the repository arises from losses in recycle/refabrication; these losses must therefore be small. Recycle technologies that achieve "clean waste" (i.e., fission products only) and recycle "dirty fuel" (i.e., all transuranics and optionally some fission products) back to the nuclear power plants are favored; they extract the maximum energy from the ore resource, and they consign only the fission products to the waste. Such recycle technologies based on both aqueous and on dry processes are under active development worldwide. They are yet to be developed to the prototype and commercial scales, and their continued development for cost effectiveness and low losses must be one of the cornerstones of the Generation IV R&D program.

3. When "dirty fuel" provides the feedstock to the fuel fabrication link in the fuel cycle, remote fabrication technologies are required. Simple, few-step processes are favored and robust fuel form designs that minimize reject rates and rework are needed. Colocation of recycle and refabrication facilities is favored to minimize shipping and handling of radioactive materials in bulk form.

   "Dirty fuel" is readily useable in fast spectrum reactors. However, the future global nuclear energy park is envisioned to rely on symbiosis of both fast and thermal reactor power plants – with recycle fuel used in both thermal spectrum reactors and fast reactors. Therefore, the recycle, the fuel design, and the reactor core design must all be coordinated such that thermal reactor neutronics performance is not excessively spoiled by contaminants in the fed back fuel feedstock. R & D on this coordination, in concert with development of appropriate symbiotic mixes of reactor and

fuel cycle types in the global energy park should be a central theme of Generation IV R&D.

4. While *the importance of near-term siting, licensing, and operating a geologic repository designed for the once-through fuel cycle cannot be overstated,* for the future, the development of recycle based fuel cycles must go forward in close linkage with the development of repository designs that exploit new opportunities for customized waste forms and optimized decay heat management strategies to extend their capacities and reduce the level of stewardship required. Specifically, a singular importance in all fuel cycles rests on decay heat management for extending geologic repository capacity.

   Repository siting is and will always be difficult, so mechanisms for extending the capacity of given sites can significantly advance economic and social acceptance goals for nuclear energy. Capacity is controlled not by mass or volume, but by heat load, and heat load is dominated by Cs and Sr in the first hundred years and by minor actinides thereafter. Three alternative approaches are available for preparation of the waste itself: interim storage of spent fuel prior to geologic emplacement; partitioning/conditioning with interim storage of tailored waste forms prior to geologic emplacement; and partitioning and interim storage of Cs/Sr with reactor recycle for consumption by fission of actinides. Design of repositories tailored for heat management and tightly integrated with the above approaches for waste preparation should be a priority development goal for Generation IV R&D.

5. Recycle and return to the reactor power plants of commixed transuranic fuel creates and intrinsic radiation barrier to theft and diversion from the commercial fuel cycle; the presence of minor actinide and carryover fission products makes the recycled materials less attractive for weapons use as well as less accessible to theft and diversion. It also avoids consigning inventories of weapons-usable materials to interim storage and to geologic repositories, where over time their intrinsic protective radiation barrier decays away. On the other hand the requirement for extrinsic measures – materials control and accounting, and physical protection to detect and prevent efforts at theft; international safeguards to detect efforts at diversion – cannot be eliminated. R&D will be required to adapt existing regimes of extrinsic measures – which have been implemented at commercial scale for the once-through and partial recycle fuel cycles – to applicability to the full recycle fuel cycles. This R&D must be closely integrated with the development of the recycle/refabrication technologies and with the design of the facilities to execute these technologies.

Generation-IV Fuel Cycle Crosscut Group

Attachment 1

March 18, 2002

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Attachment 1

Sensitivity of the Cost of Electricity in LWRs to the Cost of Uranium

Figure 1.6 in Chapter 1 shows that geochemical theory indicates substantial uranium exists in the earth's crust, and the amount increases as the assay decreases. Recovery of uranium from sources of decreasing assay will both cost more, and yield increased ecological disruption. Focusing on the first issue as a start, this section addresses the sensitivity of the cost of energy production to the cost of virgin uranium recovery.

The cost of uranium in an LWR is only one component of the total cost of electricity. In order to determine whether increased uranium costs would have a significant impact on generation costs, this section documents calculations of fuel costs for six different scenarios, and shows a relative insensitivity to uranium prices.

Two irradiation cycles, to 45 MWth-d/kg of initial heavy metal (ihm) and to 60 MWth-d/kg ihm were chosen for comparison. In both cases the specific power is 37.9 kWth/kg ihm, typical of the average power in current pressurized water reactors. The two irradiation cases are then assumed to receive uranium at three prices: the current price of $30.17 /kg U, an optimistic price for recovery of uranium from unconventional sources at $200.00 per kg U and a pessimistic price at $1000.00 per kg U. (The average price paid by all U.S. utilities for uranium in 1999 was $11.63 per pound of U_3O_8, i.e. $30.17 per kg of uranium.)

In addition to the cost of the uranium, the calculations include the costs of conversion, separative work, fabrication and the interest on the fuel while it is in the reactor.

Conversion

Conversion is the chemical processing of the yellowcake (U_3O_8) into UF_6 for enrichment in a gaseous diffusion or gas centrifuge plant. The conversion charge of $5.00 per kg U is applied to the entire mass of natural uranium used in the fuel. Differences in the conversion costs are due only to the varying tails assay at the enrichment plant.

Enrichment

When uranium is cheap, the tails assay (the fraction of ^{235}U in the depleted uranium) is allowed to rise. Because of the currently low uranium prices the tails assay are 0.3 wt % ^{235}U. If uranium were to become so rare and expensive that recovery from poor assay sources is necessary, the tails assay would be decreased to reduce the natural uranium needed. These analyses have assumed that the tails assay would be 0.15 wt % at $200 per kg U and 0.05 wt % at $1000 per kg U. The different tails assays also lead to differences in conversion costs mentioned above.

The cost of separative work was assumed to be constant for all the cases, at the present price of $78 per kg-SWU. The number of separative work units, *E*, per kg of product, M_p was computed using the following defining formulae:

The mass balance is

$$M_f x_f = M_p x_p + M_w x_w$$

and the separative work is

$$\frac{E}{M_P} = \frac{x_P - x_F}{x_F - x_W}\phi_W + \phi_P - \frac{x_P - x_W}{x_F - x_W}\phi_F , \quad x_{P,F,W} = atomic\ fractions\ of\ feed, product\ and\ tails$$

and

$$\phi_w = (2x_w - 1)\ln\frac{x_W}{1 - x_W}, etc \qquad \frac{E}{M_P} = \phi_P - \phi_F - (x_P - x_F)\frac{\phi_F - \phi_W}{x_F - x_W}$$

Fabrication

This analysis assumes that the fabrication costs would not be significantly changed by the monetary value of the natural uranium. Therefore a constant fabrication cost of $300 per kg U was assumed. Fabrication includes the conversion from UF_6 to UO_2, the sintering and grinding of pellets and the cost of cladding and other fuel assembly hardware. While fabrication costs may be significantly different from this assumption, those costs would be the same regardless of the origin and cost of the natural uranium.

Table A1.1 Impact of Uranium costs on the Cost of Electricity

Impact of Uranium Costs on the Cost of Electricity							
	Current Uranium Prices		Optimistic: $200/kg		Pessimistic: $1000/kg		
	UO2 Fuel 45 MWd/kg	UO2 Fuel 60 MWd/kg	UO2 Fuel 45 MWd/kg	UO2 Fuel 60 MWd/kg	UO2 Fuel 45 MWd/kg	UO2 Fuel 60 MWd/kg	units
Specific Power	37.9	37.9	37.9	37.9	37.9	37.9	kWth/kg
Cycle Parameters							
Total cycle length	3.8	5.1	3.8	5.1	3.8	5.1	years
Capacity Factor	85%	85%	85%	85%	85%	85%	
Effective Full Power Days	1186	1582	1186	1582	1186	1582	efpd
Burnup	45.0	60.0	45.0	60.0	45.0	60.0	MWd/kg ihm
Feed U-235 content	0.72%	0.72%	0.72%	0.72%	0.72%	0.72%	atom %
Product U-235 enrichment	4.05%	5.40%	4.05%	5.40%	4.05%	5.40%	atom %
Tails U-235 content	0.30%	0.30%	0.15%	0.15%	0.05%	0.05%	atom %
Feed U-235 content	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	wt %
Product U-235 enrichment	4.00%	5.34%	4.00%	5.34%	4.00%	5.34%	wt %
Tails U-235 content	0.30%	0.30%	0.15%	0.15%	0.05%	0.05%	wt %
Separative work	5.277	7.852	7.554	11.053	11.663	16.829	kg-SWU/kg fuel
Natural uranium	9.009	12.257	6.845	9.216	5.973	7.990	kg/kg fuel
	0.200	0.204	0.152	0.154	0.133	0.133	kg/MWd
Rates							
Interest rate	8.0%	8.0%	8.0%	8.0%	8.0%	8.0%	per year
Natural uranium	**$ 30.17**	**$ 30.17**	**$ 200.00**	**$ 200.00**	**$ 1,000.00**	**$ 1,000.00**	/kg U
	$ 11.63	$ 11.63	$ 77.09	$ 77.09	$ 385.45	$ 385.45	/lb U3O8
Conversion U3O8 to UF6	$ 5.00	$ 5.00	$ 5.00	$ 5.00	$ 5.00	$ 5.00	/kg natural U
Separative work	$ 78.00	$ 78.00	$ 78.00	$ 78.00	$ 78.00	$ 78.00	/kg-SWU
Costs							
Natural uranium	$ 271.79	$ 369.80	$ 1,369.00	$ 1,843.19	$ 5,972.66	$ 7,989.80	/kg fuel
Separative work	$ 411.58	$ 612.44	$ 589.24	$ 862.17	$ 909.72	$ 1,312.66	/kg fuel
Conversion	$ 45.04	$ 61.29	$ 34.22	$ 46.08	$ 29.86	$ 39.95	/kg fuel
Fabrication	$ 300.00	$ 300.00	$ 300.00	$ 300.00	$ 300.00	$ 300.00	/kg fuel
Total cost	$ 1,028.41	$ 1,343.52	$ 2,292.46	$ 3,051.44	$ 7,212.25	$ 9,642.40	/kg fuel
Interest during use	$ 157.14	$ 273.81	$ 350.29	$ 621.88	$ 1,102.03	$ 1,965.12	/kg fuel
Total fuel cost	$ 1,185.55	$ 1,617.33	$ 2,642.75	$ 3,673.32	$ 8,314.28	$ 11,607.52	/kg fuel
	$ 26.35	$ 26.95	$ 58.74	$ 61.22	$ 184.80	$ 193.44	/MWth-day
	$ 0.322	$ 0.329	$ 0.717	$ 0.747	$ 2.257	$ 2.362	/million BTU
	$ 3.26	$ 3.33	$ 7.26	$ 7.57	$ 22.85	$ 23.92	/MWe-hr
COE increase due to uranium from seawater			$ 4.00	$ 4.24	$ 19.52	$ 20.58	/MWe-hr
			4.0	**4.2**	**19.5**	**20.6**	**mills per kWe-hr**

Interest

The reactor operator or, if the fuel is leased, the owner of the fuel, would presumably pay for raw materials, conversion, enrichment and fabrication at about the time the fuel is placed in the reactor. Assuming that the specific power is constant and that the operator is paid immediately for the electricity generated, the operator would pay a carrying charge on the fuel cost for half of total cycle length, here 3.8 or 5.1 years. The calculation has assumed a nominal interest rate of 8% per year.

Increment in Cost of Electricity

As can be seen in the last line of Table A1.1, the use of uranium recovered from low assay sources at $200 per kg U would raise the cost of electricity from LWRs on a once-through fuel cycle by 4.0 ($0.004) or 4.2 mills per kWe-hr.

Comparison with Natural Gas Price Violability

The years 2000 and 2001 have seen a wide variation in the spot and futures market prices for natural gas. As seen in Table A1.2 the price has varied from $2.33 to $11.00 per million BTU in the last three years. Thus a hypothetical increase in the cost of electricity due to recovery of uranium even at $1000 per kg U is no more than equal to the increase, from 1998 to 2000, in the actual cost of electricity generated by natural gas turbines.

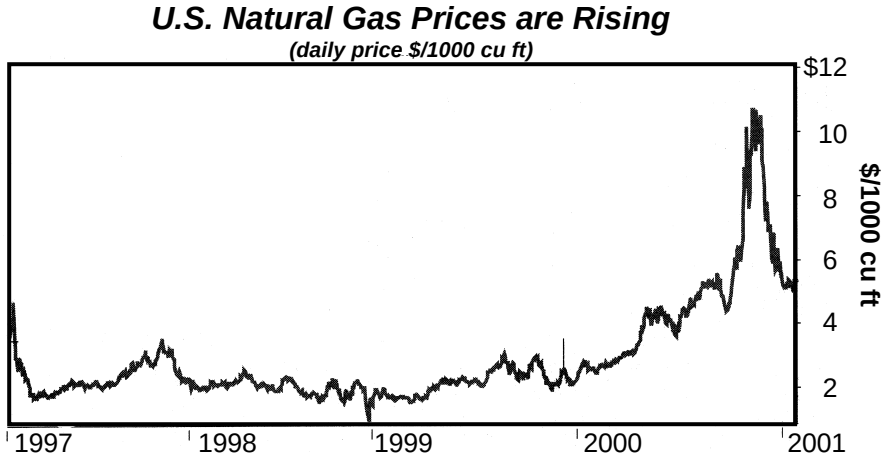


Figure A1.1 Natural Gas Price History 1997-2001

Table A1.2 Fluctuation in COE with Natural Gas

Fluctuations in Cost of Electricity due to Natural Gas Price Changes				
year	1998	Sept. 2000	Feb. 2001	
Natural Gas price	$ 2.33	$ 5.15	$ 11.34	per million BTU
	$ 2.26	$ 5.00	$ 11.00	per 1000 cu. Ft
Turbine/combined cycle efficiency	50%	50%	50%	
Fuel cost	$ 0.016	$ 0.035	$ 0.077	per kWe-hr
	15.9	35.1	77.4	mills per kWe-hr
Difference from 1998		**19.2**	**61.5**	mills per kWe-hr

Conclusions

The availability of uranium from conventional mines is of some interest in the development of new reactor types; it is seen in the scenario results in Chapter 3 that uranium resource limits are not foreseen for several decades and moreover need not by itself be a determining factor in the choice of a reactor type because the increase in the overall cost of electricity is relatively insensitive to fuel cost – unlike the situation for fossil based energy. While the cost of electricity is substantially variable region to region and country to country and also changes in time, it is clear that in all cases nuclear generated energy is extremely insensitive to the cost of the fuel resource and therefore

that a significant market share of nuclear would function as a damper on price violability in the energy marketplace. No reactor built in the next couple of decades would ever be shut down due to uranium scarcity, regardless of its conversion ratio.

Were it to develop that technical and environmental acceptable methods became available to do so, the additional cost of uranium recovered from low assay sources, would not add significantly to the cost of electricity. One would expect that, if conventional sources of uranium become limiting, a healthy competition in research and development would drive the recovery price down – and this is a FCCG research recommendation.

The sustainability goals for the Gen-IV nuclear fuel cycle are addressed to the effective use of the resource base and to the responsible management of the waste at the back-end of the fuel cycle. An additional motivation for technologies which make efficient use of the ore resource (recycling spent fuel) is that these technologies help address the waste management aspect of the Gen-IV development goals. Under significant nuclear expansion, economic forces can be expected to emerge for recycling spent fuel -- both from rising uranium prices and from efforts to use repository space efficiently.

Generation-IV Fuel Cycle Crosscut Group Report

Attachment 2

March 18, 2002

"This document has not been patent-cleared and has not received DOE approval for external release. Do not distribute beyond the Generation IV Roadmap Project Participants"

Attachment 2

Table of Contents

A2.1 The Dynamic Modeling of Nuclear Development (DYMOND) Code

The DYMOND code employs the ITHINK dynamic modeling platform to model 100-year dynamic evolution scenarios for postulated global nuclear energy parks.

The scenarios use as initial conditions the year 2000 worldwide deployments of fuel cycle facilities and power plants. Remaining productive lifetime of these assets are approximated with decommissioning spread over the next 40 years. The scenarios all use the world uranium resources which were specified in official publications such as the Redbook; scenarios are not terminated upon exhaustion of these ore resources – rather, an edit is produced delineating the necessary future discoveries.

The scenarios are driven by the Case B world aggregate nuclear energy demand growth produced in the World Energy Council/IIASA projections. The deployment is thought of as the "global nuclear energy park" with no regional segmentation of energy demand or mass flows.

In attempting to satisfy the pre-specified demand, the simulated deployment of new power plants is constrained only by internal mass flows which determine fissile availability - i.e., plants cannot start producing power unless they can be fueled - drawing on either virgin ore, or on fissile available in discharged fuel assemblies, or fissile remaining in enrichment tails and recovered irradiated uranium, or in new fissile bred in previously-deployed power plants. The sources to be exploited are specified as input to the scenario. The conversion ratios and other mass flow properties of reactor power plants are specified as input to the simulation and used to determine the composition of discharged fuel.

The DYMOND modeling of new capacity additions constrains plant startup in two ways. First, the construction and licensing lag times must have been completed. Second there must be fuel available sufficient to begin operations. The fuel requirement comprises not only the initial working inventory of the power plant itself, but also forward fueling for a specified number of reload batches. For the MOX-fueled LWR's (operating on an 18 month cycle and a 5 batch reload pattern, six years (4 batches) of forward reload fuel must be available at startup. For the liquid metal reactors (operating on a 12 month cycle and 3 batch or two batch loading pattern) two years (two batches) of forward reload fuel must be available at startup. This forward fueling constraint influences the value of the effective system doubling time and is included when quoting its value.

The physical inertial elements of the supply infrastructure are accounted for - including licensing and construction lag times for fuel cycle infrastructure elements and for power plants. The time lags for interim storage between links of the fuel cycle are also accounted for.

Market economic penetration is not modeled; both the dates of commercial availability of various power plant types and the fractional mix of plant types to be used to satisfy new demand are pre-specified as input to the specific scenario case being evaluated. In this sense, the scenarios serve to illustrate what could be *physically achievable* for

specified strategies – but not what will actually occur when market and institutional dynamics and inertia effects are accounted for along with physical inertia elements of mass flows and construction times.

The edits from simulations quantify the front and back end mass flows and inventories such as ore withdrawals vs. time and their cumulative withdrawals and the waste arising vs. time and their cumulative arising. Additionally, inventories of spent fuel, plutonium, and minor actinides in interim storage or in processing "pipelines", are shown.

The edits quantify the scales of deployment of mining/milling, enrichment services, fuel fabrication plants, reprocessing capacities, and required capacities of interim storage facilities and final disposal repositories. These edits indicate the scale of the global nuclear infrastructure.

Finally, the edits depict a year-by-year cost indicator for the fuel cycle component of the cost of energy production in the global nuclear energy park comprised of the specific mix of new plant types. The edit is for fuel cycle services only; *it does not include capital amortization and O&M costs of the power plants themselves.* This indicator is quoted in year 2000 constant dollars per Terawatt electric hour. The indicator covers the front end costs required to deliver fuel to the power plants (mining, milling, enrichment, fabrication) and the back end costs required to store and ultimately dispose the wastes. (When recycle is used to harvest materials for fuel assemblies destined for power production it's cost is included in "front end" costs rather then back end costs). This fuel cycle services cost edit serves as no more than an indicator for all of the following reasons:

- No discounting of the time value of money is used and moreover a cash flow accounting is used wherein expense is allocated to the index in the year the service is rendered -- even if the fuel won't be loaded until after a lag time.
- Unit process costs for future fuel cycle operations have been drawn up by reliance on expert judgment and therefore are highly uncertain.
- Capital amortization costs for future fuel cycle service plants are embedded in the unit process costs and also rely on expert judgment and are highly uncertain.
- The cost elements are sometimes borne by different entities (e.g., commercial ones and governmental ones) but they are aggregated for the purpose of the index.

The edit shows both the index itself and its fractional contributions from links in the fuel cycle. These fractions serve as useful indicators of sensitivities to the fuel cycle components of energy production cost and can suggest areas where R&D efforts could most usefully be directed.

The DYMOND code was developed by Mr. Anton Moisseytsev, a Texas A&M graduate student, as a summer project when employed in the Reactor Analysis and Engineering Division at Argonne National Laboratory during 2001. It has been extended for applicability to thorium cycles by Dr. Latif Yacout at Argonne National Laboratory.

References for Section A2.1

1. A. Moisseytsev, “DYMOND, A Dynamic Model of Nuclear Development,” Argonne National Laboratory Internal Report, Aug, 2001.

A2.2 Initial Conditions for The Scenarios

The scenarios all start in the year 2000.

Table A2.2.1 lists the initial conditions used in all scenarios for deployed facilities; these data are abstracted from the Chapter 2 detailed information.

Table A2.2.1 Scenario Initial Conditions: World Deployed Fuel Cycle Service Capacities in Year 2000

Fuel Cycle Link	Modeled Deployment in Year 2000	
Conversion[1,2]	113.1	t U/y
Enrichment	48850	t U SWU/y
LWR-UOX Fab	12099	t HM/y
CANDU-UOX Fab	4995	t HM/y
LWR-MOX Fab	513	t HM/y
FBR-MOX Fab	200	t HM/y
Purex UOX Reprocessing[3]	3590	t HM/y
Repositories	0	
Power Plants	353	GWe

[1] Not modeled explicitly
[2] 7000 t U/y limit on Russian supply to Western customers
[3] Reprocessing capacity ignores BNFL 1500 t/y Magnox to be decommissioned and 800 t/y Rokkasho-Mura not yet on line

The power plant deployment is taken as 350 Gwe. Their remaining lifetime is modeled as a linear distribution of remaining lifetime from zero years to 40 years.

The enrichment process is assumed to leave tailings of 0.3% U235.

Table A2.2.2 lists the estimates[1] of world inventories of nuclear materials in storage in the year 2000.

Table A2.2.2 Initial Values of World Material Inventories

Mill Tailings	TBD (Million tonnes U)
Enrichment Tails*	1.1882 (Million tonnes U)
Discharged UOX Fuel	250 (Kilo tonnes HM)
Separated Irradiated U**	14.239 (Kilo tonnes U)
Separated Pu	~200 (Tonnes)
MA in HLW	~22.65 (Tonnes MA)

*Assumed assay of enrichment tails: 0.3%
**Assumed assay of separated irradiated uranium: TBD

In running the scenarios, the initial inventory of enrichment tails was set to zero; these resources will not be drawn on until the twenty second century in all scenarios.

The 250 kilotonnes HM of discharged UOX fuel was assumed to be available for reprocessing in all scenarios except the once-through cases. In the once-through scenarios the 250 kilotonnes of initial spent fuel inventory is modeled as taken from interim storage and sent to the geologic repository in a linear withdrawal over a thirty year period from 2000 to 2030. In the scenarios where the UOX spent fuel is to be reprocessed the initial condition of 250 kiltonnes is modeled as 220 kilotonnes in interim storage ready for immediate reprocessing in year 2000 and 30 kilotonnes in pools which are emptied linearly over 5 years with shipping to interim storage.

The world initial inventories of Pu, MA, and recovered irradiated uranium were neglected in these scenarios. No uranium or plutonium declared excess of military needs was accounted for.

References for Section A2.2

1. Estimates produced from OECD-NEA sources. These are based on plausible calculations and indirect evidence and should be viewed as informed estimates.

A2.3 Fuel Cycle Unit Process Costs

Table A2.3.1 lists the unit process costs for links in the once-through and the MOX partial recycle generic fuel cycles; a range and a nominal value are shown. *The nominal value has been used to generate the cost index for the fuel cycle component of cost of energy production.* The values in Table A2.3.1 are based on consensus values developed by the OECD-NEA Expert Group on Trends in the Nuclear Fuel Cycle; [1] they are modified somewhat to account for recent deliberations of the OECD-NEA Expert Group on Comparison of Fast Reactors and Accelerator-Driven Systems in Advanced Fuel Cycles.[2]

Table A2.3.2 is a schedule for uranium ore cost versus cumulative drawdown on reserves. It rises from a current modeled value of 20 $/kgU to 130 $/kgU as a function of cumulative withdrawals per the Redbook; then is assumed to rise to $200/kgU at double the Redbook Known plus Speculative reserves; then stays flat at 200 $/kgU. (This is meant to represent the lower-bound number estimated for seawater recovery.) In the DYMOND code, this schedule is represented as a piece-wise linear fit which approximates but does not precisely reproduce this schedule (see Section A2.5, below).

Table A2.3.1 Unit Costs for Fuel Cycle Components Relating to LWR-Reactors[†]

Component	Description	Unit cost			Unit
		Lower bound	*Nominal Value*	*Upper bound*	
$Cost_{U}$	Uranium mining and milling	20	30	80	$/kgU
$Cost_{Uconv}$	Uranium conversion from U_3O_8 to UF_6	3	5	8	$/kgU
$Cost_{Uenr}$	Uranium enrichment	50	80	120	$/SWU
$Cost_{Ureconv}$	Uranium conversion from irradiated UO_2 to UF_6	15	24	30	$/kgU
$Cost_{UOXfab}$	UOX fuel fabrication	200	250	350	$/kgHM
$Cost_{Udepl}$	Depleted Uranium long-term storage	0.3	0.5	1	$/kgU/y
$Cost_{SFIntstore}$	Spent UOX-fuel interim storage*	5	5	5	$/kgHM.year
$Cost_{Pustore}$	Separated Pu storage	1000	1500	2000	$/kg$Pu_{tot}$.year
	Separated Irradiated U interim storage[*]		16		$/kgU/y
$Cost_{Pupurification}$	Separated Pu purification	10000	18000	28000	$/kg$Pu_{tot}$
$Cost_{Sftransport}$	Spent fuel transport	40	50	60	$/kgHM
$Cost_{UOXrepro}$	UOX reprocessing	500	800	1100	$/kgHM
$Cost_{MOXfab}$	MOX fuel fabrication	600	1100	1750	$/kgHM
$Cost_{MOXrepro}$	MOX fuel reprocessing	500	800	1100	$/kgHM
$Cost_{UOXgeo}$	UOX spent fuel conditioning and disposal	130	300	500	$/kgHM
$Cost_{HLWgeo}$	Vitrified HLW conditioning and disposal	80	200	310	$/kgHLW
*	One Time Charge for UOX Storage		50		$/kgHM

[†]All costs are expressed in 2000-dollars. In generating the table, unit costs available in the literature for other base-years were corrected using an escalation rate of 3%.

[*]Separated Irradiated Uranium Storage cost was estimated by a FCCG co-chair at 16 $/kgU/y

Table A2.3.2. Ore Cost Schedule from the Redbook*
Piece-Wise Linear Representation Between Data Points

Cost ($/kgU)	Cumulative Ore Withdrawn (kt U)
20	0
20	200
40	1,454
80	5,916
130	15,208
200	30,416
200	∞

See table A2.5.1 for the schedule used in the scenarios which approximates this schedule.

Table A2.3.3 shows the fuel cycle unit process costs for pebble bed and prismatic HTGR fuel cycles. In absence of data, these have been assumed to be identical to those for the UOX once-through fuel cycle.

Table A.3.3. Unit Process Costs For Particle Fuel Cycle Services

	PBMR Prismatic
Mining Milling Conversion Enrichment Conversion	Same as Table A2.3.1
Pebble Fab	
Prismatic Fab Compacts Fuel Assemblies	(Assume same as LWR-UOX) Same as Table A2.3.1
Pebble Interim Storage	
Pebble Conditioning and Disposal	
Prismatic Assembly Interim Storage	
Prismatic Conditioning and Disposal	

Table A2.3.4 shows the unit process cost estimates for advanced closed fuel cycles for fast neutron spectrum systems. These data are consensus estimates generated in deliberations of the OECD-NEA Expert Group on Comparison of Fast Reactors and Accelerator-Driven Systems in Advanced Fuel Cycles.[2] Estimates are provided for fast reactor MOX fuel, for fast reactor fertile-containing TRU fuel, for dedicated (U238-free) TRU fuel, and for dedicated (U238-free) Minor Actinide fuel. *The range between lower bound and upper bound estimates clearly indicates the degree of uncertainty attendant to the early state of development of these recycle and refabrication technologies.* The nominal value was used for these scenarios.

Table A2.3.4. Unit Costs for Advanced Fuel Cycle Components

Component	Description	Unit cost			Unit[4]
		Lower bound	*Nominal Value*	*Upper bound*	
FR MOX-fuels					
$Cost_{FR\text{-}MOX\ driver\_fab}$	FR-MOX driver fuel fabrication	650	1400	2500	$/kgHM
$Cost_{FR\text{-}MOXblanket\_fab}$	FR-MOX blanket fuel fabrication	350	500	700	$/kgHM
$Cost_{FR\text{-}MOXdriver\_repro}$	FR-MOX driver fuel reprocessing	1000	2000	2500	$/kgHM
$Cost_{FR\text{-}MOXblanket\_repro}$	FR-MOX blanket fuel reprocessing	900	1500	2500	$/kgHM
FR TRU Fuels					
$Cost_{FR\text{-}TRU\_fab}$	FR-TRU fuel fabrication	1400	2600	5000	$/kgHM
$Cost_{FR\text{-}TRU\_repro}$	FR-TRU fuel reprocessing	1000	2000	2500	$/kgHM
FR Blankets in Full TRU Recycle[5]					
$Cost_{FR\_blanket\text{-}fabr}$	FR blanket fuel fabrication	350	500	700	$/kgHM
$Cost_{FR\_blanket\_repro}$	FR blanket fuel reprocessing	1000	2000	2500	$/kgHM
Dedicated TRU Fuel					
$Cost_{ADS\text{-}TRU\_fab}$	ADS-TRU fuel fabrication	5000	11000	15000	$/kgHM
$Cost_{ADS\text{-}TRU\_repro}$	ADS-TRU fuel reprocessing	5000	12000	18000	$/kgHM
Dedicated MA Fuel					
$Cost_{ADS\text{-}MA\_fab}$	ADS-MA fuel fabrication	5000	11000	15000	$/kgHM
$Cost_{ADS\text{-}MA\_repro}$	ADS-MA fuel reprocessing	5000	12000	18000	$/kgHM

[4]All costs are expressed in 2000 dollars. Unit costs for other base-years were corrected using an inflation of 3%.
[5]Blanket and driver fuels are considered to be co-reprocessed and to recover MA as well as plutonium.

References for Section A2.3

1. OECD-NEA, “Trends in The Nuclear Fuel Cycle” -- An Expert Group Study, (to be published in Dec 2001).
2. OECD-NEA, “Comparison of Fast Reactors and Accelerator Driven Systems in Advanced Fuel Cycles – An Expert Group Study, (to be published in Feb 2002).

A2.4 Attribute Sets for Fuel Cycle Services Facilities

The fuel cycle service facilities include mining and milling, conversion and enrichment, fabrication, reprocessing and waste preparation, interim storage, and geologic repositories. Their attribute sets include construction and licensing lag times, throughput rates, fractions of throughput lost to waste, and throughput lag times.

At the level of development currently available in the DYMOND simulation code, these fuel cycle facility attributes are represented only very coarsely. For example, Table A2.4.1 indicates the modeling of construction and licensing lag time used for the various facilities. Only in the case of large PUREX reprocessing plants was a construction/licensing lag time actually modeled; for other facilities they were assumed to come on line at the required capacity as determined by the instantaneous mass flows required for meeting demand. (i.e., except for PUREX plants no prediction of future need and ordering construction in advance was modeled).

For PUREX plants, lag times were used and algorithms for forecasting future needs and ordering new capacity were developed and used in the scenarios. These prediction and build decision computational algorithms developed and employed in DYMOND for large PUREX reprocessing plants could easily be adapted to the other facilities if future refinement of the scenarios is required.

Some lag times which were neglected in the scenarios are significant indeed . As shown for the mining/milling and geologic repositories, multi decade lead times have been required in actual practice – between identified future need and actual deployment – 10 to 15 years from start of prospecting in a potential ore field to the start of production of a uranium mine (based on experience in Canada[1]) and 15 to 30 years for siting, licensing and constructing a geologic repository – (based on experience at WIPP). These lag times were neglected in the scenarios.

Processing lag times as modeled in DYMOND are shown in Table A2.4.2. UOX fabrication requires 2 years. Cooling of UOX assemblies prior to PUREX reprocessing is 5 years. The PUREX reprocessing time interval is ½ years with a 2 year lag for MOX refabrication and shipment. The out of reactor cycle time is 7 ½ years altogether.

For the metal fueled fast reactor full TRU recycle cases, based on pyrometallurigcal/casting recycle technologies, a spent fuel cooling time of 1 year is used. Reprocessing takes another 1 year; refabrication takes 1 year and storage takes a year. Total out of reactor dwell time for PYRO/Casting recycle of metal alloy fuel is modeled as 4 years altogether.

Discharged fuel assemblies destined for a geologic repository are modeled as being held in interim storage for 30 years, then placed in the geologic repository. Waste forms generated in PUREX reprocessing are stored for 5 years, then placed in a geologic repository. The 5 year lag times apply for waste forms produced by pyro full TRU recycle also. Table A2.4.3 lists these waste lag times.

These waste interim storage lag times affect the fuel cycle cost index; as shown in Table A2.3.1 and waste interim storage inventory charges are assessed annually during the interim storage period and then a one time charge is made when the waste is transferred and emplaced in the geologic repository.

Table A2.4.1. Construction and Licensing Lag Times for Fuel Cycle Facilities As Modeled in DYMOND

	Construction + Licensing Lag	
	Actual	Modeled
Mine/Mill Start of Prospecting to Production	10-15 y[(1)]	Available as needed
Conversion and Enrichment	TBD	Available as needed
Fabrication (UOX, MOX)	TBD	Available as needed
Reprocessing (PUREX) @1400 tonnes/y plant size	5 + 2	Need for new deployments is based on predictions of forward need
Reprocessing (Pyro or Adv. PUREX) @ 50 to 200 tonnes/y plant size	TBD	Available as needed
Interim Storage Facilities	TDB	Available as needed
Geologic Repositories	15-30 y	Available as needed

Table A2.4.2. Attribute Sets Fuel cycle Facilities

	Enrichment		UOX Fab	Recycle	MOX Fab	Recycle	TRU Fab
	Centrifuge Diff'n	Laser		Central PUREX		Colocated Pyro	Colocated Casting
Construction Lead Time*							
Licensing Lead Time*							
Plant Lifetime (years)				60			
Throughput Tonnes HN/y Rate							
Storage Prior to Processing (years)				5		1	
In Process Dwell Time	1		2	½	2	½	2
Total out of Reactor Time				-	7 ½	-	4
Loss to Waste per Recycle/Refab pass (%)	-	-	-	0.1		0.1	-

*See Table A2.4.1

Table A2.4.3 Interim Storage Lag Times Prior To Transfer To A Geologic Repository

Waste Product	Interim Storage Lag Time (years)
Spent Fuel	30
Glass HLW (PUREX)	5
Ceramic & Metallic Waste (Pyro)	5
Recovered Irradiated Uranium	∞
Enrichment Tails	∞

References for Section A2.4

1. D. H. Underhill, Analysis of Uranium Supply to 2050, IAEA-SM-362/2, (a paper based on the IAEA Study, Analysis of Uranium Supply to 2050, (to be published by the IAEA).

A2.5 Calculation of The Cost Index in DYMOND

The cost index was calculated year by year for the aggregate global nuclear energy park using the unit process cost values in Tables A2.3.1, .2, .3, and .4 and the lag times from Tables A2.4.1, A2.4.2, and A2.4.3. A cash accounting method was used – the processing costs were accumulated in the year the service was rendered. The time value of money was neglected; and constant year 2000 dollars were used.

No breakdown for each reactor or each fuel cycle type were edited.[1] Instead, the fuel cycle costs for the entire global nuclear park were aggregated for the edits. However the edits were broken down by functional link in the fuel cycle chain – as shown in Figures A2.5.1 and A.5.2:

Total Mining & Enrichment Cost
Total Fuel Fabrication Cost
Total Recycle Cost
Total Storage Cost
Total Disposal Cost

Each of these components was then divided by the total terrawatt hours of electrical energy produced by the entire global energy park that year and expressed as million $/terrawatt hours (equivalent to mils/kwe hr).

Storage costs are charged for enrichment tails; for materials in process (discharged fuel awaiting recycle and recycle products awaiting fabrication including recovered irradiated uranium, plutonium, and minor actinides); and for materials awaiting shipment to a repository (spent fuel in once-through and partial recycle cycles and high level waste in recycle cycles). Storage costs carry an annual inventory charge assessed each year over the time durations of storage as tabulated in Table A2.4.3, and in the case of spent UOX or MOX fuel awaiting recycle or destined for the repository, also include a one-time 50 $/kg HM charge for amortization of the interim storage facility.

Disposal (to a repository) costs include 1 mill/kwe hr assessed in the year the energy is delivered plus charges for preparation, packaging and shipping assessed at the end of the interim storage period, to move the material to the repository. For spent fuel from once-through or partial recycle cycles this is 300 $/kg HM preparation plus 50$/kg HM shipping. For High Level waste forms from recycle this is 200 $/(kg Fiss Prod + Minor Actinides) one time package and ship charge.

There are 250,000 tonnes heavy metal of legacy spent UOX fuel existing at the start of the scenarios. In those (once-through cycle) cases where it is destined for the repository, it is packaged and shipped to draw the inventory down to zero – done linearly over a 30 year period starting in the year 2000 and appropriate charges are made; however the 1 mill/kwhr

[1] Such a breakdown could be produced were it desirable in future evaluations. The DYMOND code has the capability to add in power plant production costs as well – but these cost components were always set to zero for the purpose of generating the fuel cycle service cost index.

is not assessed – since it was paid previous to year 2000. (In partial recycle and full TRU or full Pu recycle cases, this legacy material is viewed as a fuel resource and is saved for eventual recycle.)

Recycle and refabrication costs based on PUREX and LWR-MOX use the rather well established nominal unit process costs shown in Table A2.3.1. In the growing economies of the scenarios, separated plutonium is shown to remain in storage for short periods before refabrication and return to a reactor; no purification (from Am) operation is used.

For all cycles which use MOX multi recycle and full TRU recycle, advanced PUREX or PYRO recycle and remote refabrication in heavily shielded facilities are assumed, and the unit costs are modeled as the "nominal costs" for TRU fuel shown in Table A2.3.4. These operations are substantially more expensive than UOX PUREX and glove box Pu MOX fabrication due to added steps to recover minor actinides (in advanced PUREX) and due to remote refabrication in both technologies – due to the intense radiation fields caused by the minor actinides. Fast reactor blanket fabrication is assumed to use either depleted uranium or recovered irradiated uranium and to be conducted in lightly shielded facilities at the lower cost of 500 $/kg HM. When full TRU recycle fast reactors are used to manufacture fissile for return to fuel *thermal reactors*, in the latter part of the century, the same high cost of remote fabrication is used for the thermal reactor recycle fuel also (since that TRU bearing fuel will be highly radioactive). It may be seen in Table A2.3.4 that the range of possible costs for these unit process costs for MOX multi recycle and full TRU recycle are very uncertain at their current state of development. The nominal value was used, and no credit was taken for cost reductions as experience accumulates. Several sensitivity studies on these costs were run and are reported in Section 3.6 of the report.

Table A2.3.2 shows the cost of recovering uranium from ore versus cumulative ore harvested – this schedule is taken from the Redbook; it shows current estimated of cost of recovery, not price. Lacking other information, we have used it when calculating virgin U cost. However, the DYMOND code could not handle the particular piecewise linear segment representation as shown in Table A2.3.2; therefore for modeling the cost index calculation, a schedule (having uniform increments on the cumulative withdrawal axis) which nearly reproduces Table A2.3.2 was used. The schedule actually used in the DYMOND code is shown in Table A2.5.1.

Table A2.5.1 DYMOND Ore Cost Schedule

$/kgU	Cumulative Withdrawals ktU
15	0
40	1500
53	3000
67	4500
80	7500
103	9000
113	10500
120	12000

126	13500
130	15000
135	16500
138	18000
142	19500
146	21000
150	22500
156	24000
162	25500
170	27000
182	28500
200	30000

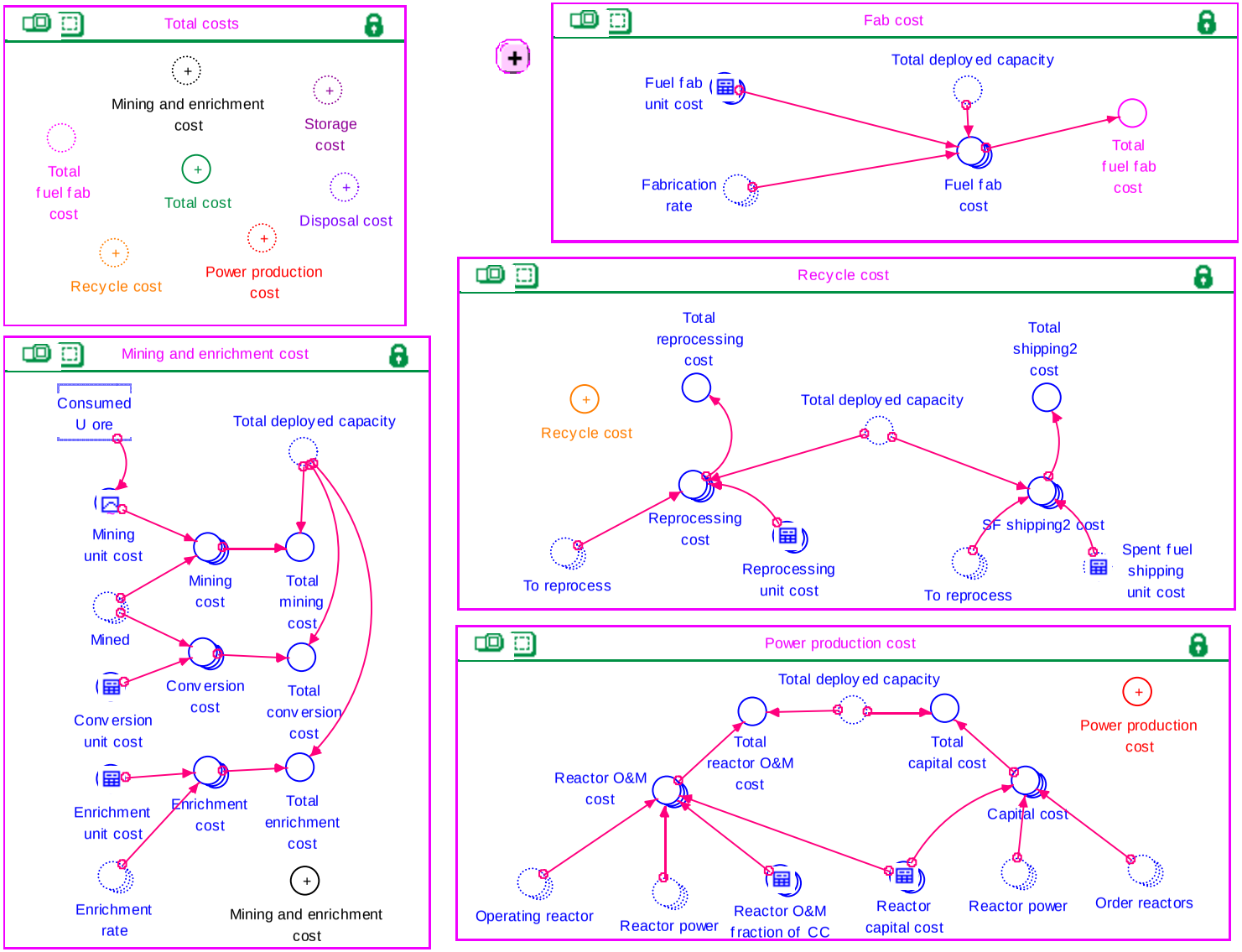

Total costs
Mining and enrichment cost
Storage cost
Total fuel fab cost
Total cost
Disposal cost
Recycle cost
Power production cost
Fab cost
Total deployed capacity
Fuel fab unit cost
Fabrication rate
Fuel fab cost
Total fuel fab cost
Recycle cost
Total reprocessing cost
Total shipping2 cost
Total deployed capacity
Reprocessing cost
To reprocess
Reprocessing unit cost
SF shipping2 cost
To reprocess
Spent fuel shipping unit cost
Mining and enrichment cost
Consumed U ore
Total deployed capacity
Mining unit cost
Mining cost
Total mining cost
Mined
Conversion cost
Conversion unit cost
Total conversion cost
Enrichment unit cost
Enrichment cost
Total enrichment cost
Enrichment rate
Mining and enrichment cost
Power production cost
Total deployed capacity
Total reactor O&M cost
Total capital cost
Power production cost
Reactor O&M cost
Capital cost
Operating reactor
Reactor power
Reactor O&M fraction of CC
Reactor capital cost
Reactor power
Order reactors

Figure A2.5.1

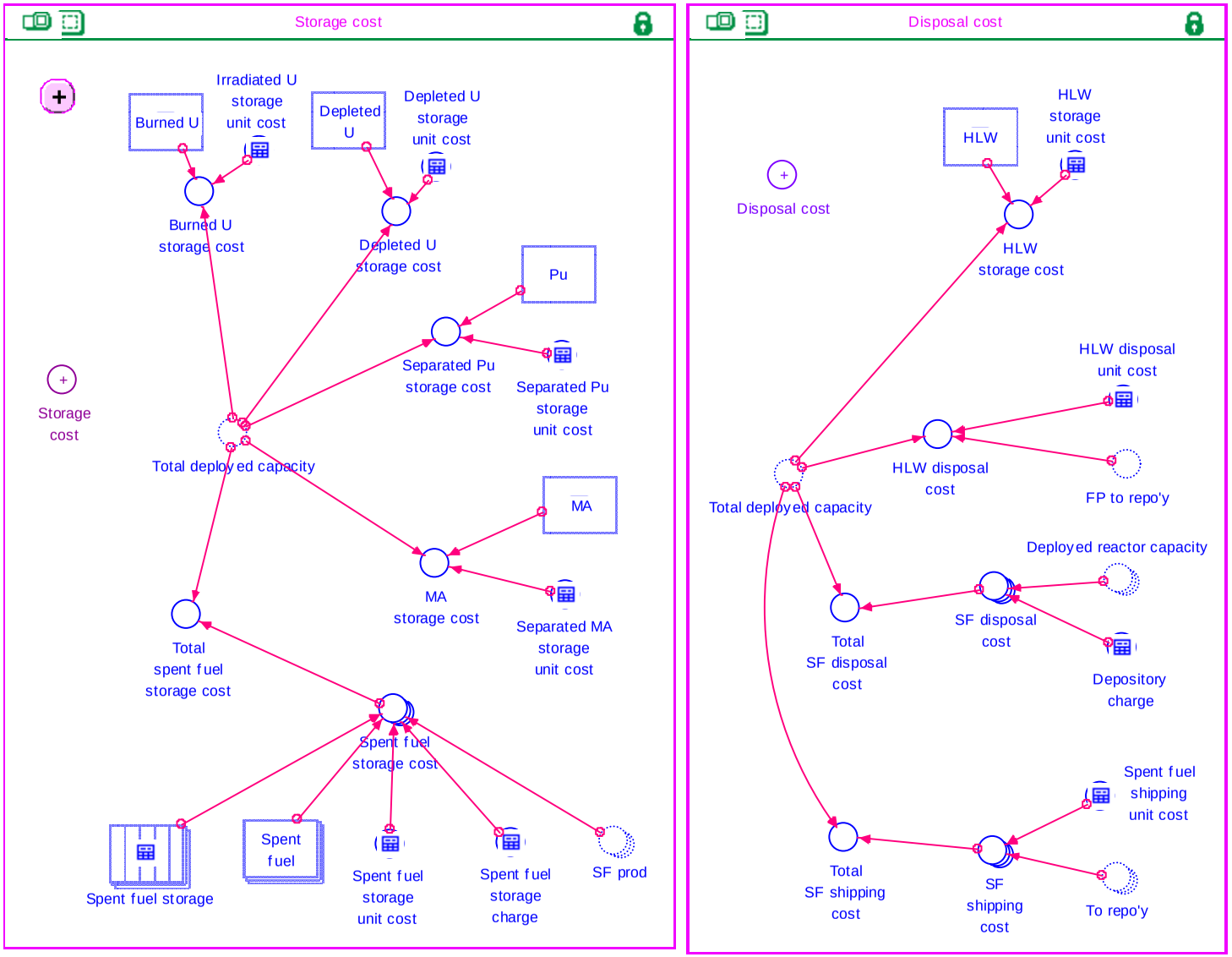

Storage cost
Burned U
Irradiated U storage unit cost
Depleted U
Depleted U storage unit cost
Burned U storage cost
Depleted U storage cost
Pu
Separated Pu storage cost
Separated Pu storage unit cost
Storage cost
Total deployed capacity
MA
MA storage cost
Separated MA storage unit cost
Total spent fuel storage cost
Spent fuel storage cost
Spent fuel storage
Spent fuel
Spent fuel storage unit cost
Spent fuel storage charge
SF prod
Disposal cost
HLW
HLW storage unit cost
Disposal cost
HLW storage cost
HLW disposal unit cost
HLW disposal cost
FP to repo'y
Total deployed capacity
Deployed reactor capacity
SF disposal cost
Total SF disposal cost
Depository charge
Spent fuel shipping unit cost
Total SF shipping cost
SF shipping cost
To repo'y

Figure A2.5.2

A2.6 Nuclear Power Plant Attribute Sets

Power plants are represented at a course level in the DYMOND scenario simulations – just sufficient to scope out mass inventories and mass flow rates and to represent construction and licensing lag times and plant design lifetime. The set of attributes needed for the simulations differ among three categories of power plant/fuel cycle

- Once Through Concepts
- Limited Recycle Concepts
- Multi Recycle Concepts

because of different categories of fuel assembly types they use. Table A2.5.1 lists the attribute set ideally required by DYMOND for each concept type.

At the level of detail currently employed in the DYMOND dynamic simulation code, heavy metal and its fissile and minor actinide fraction is all that is tracked; differences in *isotopic* distribution of uranium, plutonium and minor actinide elements cannot currently be represented.

The DYMOND modeling of new capacity additions constrains plant startup in two ways. First, the construction and licensing lag times must have been completed. Second there must be fuel available sufficient to begin operations. The fuel requirement comprises not only the initial working inventory of the power plant itself, but also forward fueling for a specified number of reload batches. For the MOX-fueled LWR's (operating on an 18 month cycle and a 5 batch reload pattern, six years (4 batches) of forward reload fuel must be available at startup. For the liquid metal reactors (operating on a 12 month cycle and 3 batch or two batch loading pattern) two years (two batches) of forward reload fuel must be available at startup. This forward fueling constraint influences the value of the effective system doubling time and is included when quoting its value.

Table A2.6.1 Power Plant Attribute Values Requested for Fuel Cycle Mass Flow Analyses (4/27/01)

Glossary:

Heavy Metal (HM)	are the actinides: U, Th, Pu, Am, etc.
Initial Heavy Metal (IHM)	heavy metal in the fuel as loaded
Transuranics (TRU)	are the elements heavier than U
Minor Actinides (MA)	are Np, Am, Cm
Fission Products (FP)	are fission products
Station Efficiency (η)	MW_e/MW_{th}

Enrichment (ε)

$\frac{\text{kg U235 or U233}}{\text{kgIHM}}$ for thermal UOX

or

$\frac{\text{kg Pu239 + Pu 241}}{\text{kgIHM}}$ for thermal of fast MOX

or

$\frac{\text{kg TRU}}{\text{kg IHM}}$ for fast systems

(A.) Attribute List for Once-Through Power Plant Concepts (e.g., LWR UOX; HTGR)

Power Plant:
- Rating (MWe); station efficiency (η)
- Construction lead time (y); lifetime (y)
- Capacity factor

Working Inventories:
- $\frac{\text{kg IHM}}{MW_{th}}$
- Average Enrichment
- Fertile material (U or Th)

Mass Flows:
- Refueling Interval y
- # of reload batches (i.e., 1/3 reload or ¼ reload, etc.)
- Ave discharge burnup $\frac{MW_{th}\ \text{days}}{\text{kgIHM}}$
- Mass fractions in Discharge Fuel $\frac{\text{kg Pu}}{\text{kgIHM}}, \frac{\text{kg TRU}}{\text{kgIHM}}, \frac{\text{kg U235}}{\text{kgIHM}}, \frac{\text{kg U233}}{\text{kgIHM}}$

(B.) Attribute List for Partial Recycle Power Plant Concepts (e.g., LWR MOX, DUPIC)

Power Plant: Same as (A)

Working Inventories: Mass loading fraction for each assembly type (e.g., UOX, MOX)

$$\frac{(\text{kg IHM})_{\text{assembly type i}}}{\text{kg All IHM in the core}}$$

then:

Same as (A) for each assembly type

Mass Flows:
- Refueling interval (y)
- # of reload batches

then

same as (A) for each assembly type
- ave discharge burnup
- mass fractions

Lag Times:
Lag from discharge to reprocessing (y)
Lag from reprocessing to refab (y)
Lag from refab to reload (y)

(C.) Multi-Recycle Power Plant Concepts (e.g, LMFBR)

Power Plant: Same as (A)

Working Inventories: Same as (B)

Where for fast reactors both fuel and blanket assemblies are treated

Where for thermal reactors it is necessary to Distinguish once, twice, thrice recycled assemblies – to account for evolution of MA content

Mass Flows: Same as (B)

Lag Times: Same as (B)

Values of Attribute Sets

The Gen-IV Roadmap schedule has FCCG and TWG activities occurring in parallel rather than in series – with the result that the Gen-IV power plant attribute sets used by the FCCG for running the scenarios comprise a "first cut approximation" *generated by the FCCG itself* – rather than provided by the TWG's. The attribute set values were none-the-less drawn from real designs reported in the literature. (See the References at the end of this section). While some details would change when refined attribute sets are used, the principal features of the scenario results will not be substantially affected *because the ore drawdown and waste arisings per unit of energy, and the doubling times of fuel cycles are fully determined by the coarse level of detail specified in Table A2.5.1*, and because the discharge burnups, enrichments and initial working inventories chosen by the FCCG all lie in the proper range for the various Gen-IV concepts sets.[2]

LWR's

Table A2.5.2 lists the attributes used in the scenarios for LWR UOX once-through reactors and LWR-MOX mono recycle reactors using 100% MOX assemblies[1]. The lag times for UOX reprocessing and MOX fab are given in Table A2.4.2.

References

1. Luc Van den Durpel, Personal Communication, August 2001.
2. OECD-NEA, "Trends in The Nuclear Fuel Cycle" – An Expert Group Study, (to be published in December 2001.

Gas Reactors

Table A2.5.3 lists the attributes used for the pebble bed[1] and prismatic[1] HTGR's fueled on enriched uranium and operating once-through.

The last column in Table A.5.3 is for a prismatic HTGR fueled on the transuranics derived from fast breeder reactor production of excess TRU. The design is an early version of a GA design for weapons plutonium disposition[2] – it operates once-through and achieves 87% transmutation of Pu239, and 62% burnup of transuranics overall at discharge. In the scenario using this once-through HTGR fueled with TRU from fast breeders, it is assumed to be driving a thermochemical water cracking plant for manufacture of hydrogen; it could equally well be producing electricity at the same conversion efficiency of 45%.

References

1. WG-2 Draft concept Summary Report
2. General Atomics, MHTGR Plutonium Consumption Study Phase II Extension, FY-94 Final Report, September 30, 1994.

[2] The initial working inventories were sometimes approximated by summing the equilibrium cycle batch loadings – this is perhaps the biggest error in the approximations.

Liquid Metal Fast Reactors

Four fast reactor attribute sets were used for the scenarios. All are sodium cooled. The first three were for three alternate core reload patterns of interchangeable core loadings for a 600 MWe, metal fueled, full TRU recycle concept which achieved breeding ratios of 0.5, 1.0, and 1.25 depending on interchangeable core reload pattern.[1] Mass Flows are represented separately for drivers, radial blankets (external plus internal) and axial blankets in DYMOND, and their fabrications are costed differently as shown in Table A2.3.3 (blanket fabrication using depleted uranium is less expensive than fabrication of TRU-bearing drivers). These mass flows are given in Table A2.5.4. Initial working inventories were *approximated* as sums over equilibrium batch loadings. For the breeding ratio 1.25 case its doubling time is rather long at 45 y, because of a spoiled geometry used to achieve a burning capability at breeding ratio of 0.5. Out of reactor lag times for reprocessing/refab are given in Table A2.4.3 and are 4 years in total.

The fourth fast reactor is a breeder design of very short doubling time (7 y's) developed [2] for the INFCE studies in 1979 to 1981. It is a 1000 MWe, radially – heterogeneous sodium cooled, metal fueled design with a breeding ratio of 1.72. It's mass flows are also given in Table A2.3.3 and are shown separately for drivers, radial blankets (internal and external), and axial blankets. At the time of the INFCE study, full TRU recycle had not yet been invented, and the published mass flows neglected minor actinides which at the time of INFCE were assumed to be lost to waste during PUREX recycle. For the scenario modeled here based on full TRU recycle using Pyro recycle methods, we assume that the minor actinide to plutonium ratio is similar to that of the other metal-fueled fast reactors, and the reported plutonium mass flows were partitioned into plutonium and minor actinide fractions according to that ratio. The out of reactor recycle/refab dwell times are modeled as in Table A2.4.3 and are 4 years in total.

The fast reactor designs modeled in the scenarios are metal fueled and pyro recycled with heavy metal losses assumed to be 0.1% per pass through recycle (see Table A2.4.2) and full TRU recycle back to the reactor.

Results are also reported from the literature for oxide-fueled, advanced PUREX recycled concepts with full TRU recycle back to the reactor and assumed 0.1% heavy metal loss to waste per recycle pass.

References

1. OECD-NEA, "Physics of Plutonium Recycling – Volume V: Plutonium Recycling in Fast Reactors, Appendix A: US Metal Fueled Benchmark Solution," (1996)
2. ANL-80-40, "Fast Breeder Reactor Studies," C. E. Till, Y. I. Chang, et al, July 1980.

DUPIC Fueled CANDU Reactors

Table A2.5.5 shows the attribute values for DUPIC-fueled CANDU reactors. Table A2.5.6 gives lag times for the DUPIC fuel cycle steps and Table A2.5.7 gives its unit process cost numbers.

References

1. Hedges and Yang, Personal Communication from TWG1, November 2001.

Radkowsky Thorium/Uranium Fueled PWR

Table A2.5.8 shows the attribute set used for the Radkowsky PWR concept.

References

1. A. Galperin, P. Reichert, and A. Radkowsky, "Thorium Fuel for Light Water Reactors – Reducing Proliferation Potential of Nuclear Power Fuel Cycle," Science & Global Security, 1997, Vol. 6, pp. 265-290, Princeton University.
2. A. G. Morozov, A. Galperin, and M. Todosow, "A thorium-based fuel cycle for VVERs&PWRs – a nonproliferative solution to renew nuclear power," Nuc Eng & Design, pp. 13-14, January 1999.
3. A. Radkowsky and A. Galperin, "The Nonproliferative light Water Thorium Reactor: A new Approach to Light Water Reactor Core Technology," Nucl. Tech., Vol. 124, Dec. 1998.
4. A. Radkowsky, "Using Thorium in a Commercial Nuclear Fuel Cycle: How to do it," Nuc Eng & Design, pp. 14-16, January 1999.

Molten Salt Closed Thorium Fuel Cycle

Table A2.5.9 shows the attributes used for the MSR concept. The market deployment sequence first uses enriched uranium; then uses TRU from LWR-UOX once-through discharge fuel and is described in detail in Section 3.7.2 of the main report.

References

1. Chapter 1, "Introduction: Conceptual Framework & Issues," Gen-IV Fuel Cycle Crosscut Group, November 1, 2001, section 1.3.2.
2. R. W. Moir, "Cost of Electricity from Molten Salt Reactors," to appear in Nucl. Tech., Vol. 138, April 2002.
3. A. M. Perry and H. F. Bauman, "Reactor Physics and Fuel-Cycle Analysis," Nucl. Appl. Tech., 8, 208 (1970).

Table A2.5.2 LWR-UOX and LWR-MOX Attribute Sets (Partial Recycle)

	LWR-UOX	LWR-MOX Mono Recycle 100% Core Loading of MOX
Rating	3714	3714
η	35	35
Const. Lead	4	4
Licensing Lead	2	2
CF	85	85
	UOX Assemblies	MOX Assemblies
$\frac{\text{kg IHM}}{\text{MW}_{th}}$		
$\varepsilon \left(\frac{U^{235}}{HM} \text{ or } \frac{Pu^{239} + Pu^{241}}{HM} \right)$	4.2	4.25
`Fertile	U238	U238 (0.3% tails)
Total Pu/IHM (%)	0	8.1
Refueling Interval (months)	18	18
# Batches	5	5
Ave. Burnup $\left(\frac{\text{MW}_{th}\text{d}}{\text{kg IHM}} \right)$	50	50
Mass Fractions in Discharge		
kg TRU/kg IHM kg Pu/kg IHM kg U^{235}/hg IHM kg U^{233}/kgIHM		
Initial Condition Deployment in Year 2000	350 GWe	0

Table A2.5.3 Attribute Sets of gas Reactors (Once Through)

		HTGR Pebble Bed	HTGR Modular	HTGR TRU Burner
Rating	MWt	250	600	600
η	(%)	46	47	45*
Const. Lead	(y)			
Licensing Lead	(y)			
CF	(%)			0.71
$\left(\frac{\text{kg IHM}}{\text{MW}_{th}}\right)$		10.1	7.5	1.06
ε	(%)	8	15.5	100
Fertile		U238	U238	--
Total Pu/IHM		0		1.0
Refueling Interval	(months)			1.2
# Batches				3
Ave. Burnup	$\left(\frac{\text{MW}_{th}\text{d}}{\text{kg IHM}}\right)$	80 to 100	121	618
Mass Fractions in Discharge				
Kg TRU/kgIHM		(TBD)	(TBD)	0.368
Kg Pu/kgIHM		(TBD)	(TBD)	0.353
Kg^{235}/hg IHM		(TBD)	(TBD)	--
Kg U^{233}/kgIHM		(TBD)	(TBD)	--
Initial Condition Deployment in Year 2000	GWe		0	0

Table A2.5.4 Fast Reactor Attribute Sets

	Convertible Fast Reactor[5]									Super Breeder(6)		
	BR=0.5 Net Burner			Breeding Ratio=1.0 Fissle Self Sufficient			Breeding Ratio-1.25			Breeding Ratio=1.72		
Rating MW_{th}	1,575			1,575			1,575			2,740		
η %	38			38			38			36.5		
Const'n Lead Time y												
Licensing Lead Time y												
CF %	85			85			85			75		
	Core	Ax Blk.	Rad. Blk.	Core	Ax Blk.	Rad. Blk.	Core	Ax Blk.	Rad. Blk.	Core	Ax Blk.	Rad. Blk.
Refueling Interval y	1	1	1	1	1	1	1	1	1	1	1	1
# Batches	3	3	4	3	3	4	3	3	4	2	2	3?
Cooling Time y		1			1			1			1	
Reprocessing/Refab y		1			1			1			1	
Kg IHM	17,525			17,526	19,694	12,460	17,526	39,388	12,460	21,712	24,838	111,542
Input kg/y												
HM	5,842			5,842	6,565	3,115	5,842	13,130	3,115	10,856	12,414	34,767
Pu	1,558			1,440	-	-	1,440	-	-	2,267	-	-
MA	220			193	-	-	193	-	-		-	-
Output kg/y												
HM	5,339			5,382	6,537	3,106	5,382	13,074	3,106	10,248	12,398	34,611
Pu	1,333			1,254	143	49	1,254	286	49	2033	126	661
MA	192			170	0.4	0.2	170	0.8	0.2	-	-	-

(5)

(6)

Table A2.5.5 Power Plant Attribute Values for the DUPIC Fuel Cycle

(B) Attribute List for DUPIC Fuel in CANDU Plant(CANDU DUPIC Fuel)

Power Plant (In case that the DUPIC fuels are used in existing CANDU reactor)

- Rating (MWe) : 713
- Station efficiency (η) : 33%
- Construction lead time (y); 4
- Lifetime (y) : 40
- Capacity factor : 90%

Working Inventories:

- $\frac{\text{kg IHM}}{\text{MW}_{\text{th}}}$: 86,640/2,159 = 40.13
- Mass loading fraction : DUPIC fuels are fully loaded in whole CANDU core.
- Average Enrichment : 0.54% for Pu239, 0.05% for Pu241, 0.98% for U235
  → total : 1.57% for DUPIC fuel.
  * DUPIC fuels are assumed to be directly fabricated with PWR spent fuels with 35,000 MWD/MTU of burnup.
- Fertile material (U) : U238

Mass Flows:

- Refueling Interval : on-line refueling
- # of reload batches : NA
- Ave discharge burnup $\frac{\text{MW}_{\text{th}}\ \text{days}}{\text{kgIHM}}$: 15,400
- Mass fractions in Discharge Fuel :

$$\frac{\text{kg Pu}}{\text{kgIHM}}, = 0.796\%, \qquad \frac{\text{kg TRU}}{\text{kgIHM}}, = 0.949\%$$

$$\frac{\text{kg U235}}{\text{kgIHM}}, = 0.21\%, \qquad \frac{\text{kg U233}}{\text{kgIHM}} = 0$$

Lag time;

- Lag from discharge to processing of spent PWR fuel : 10 years
- Lag from processing to refab. : 0.5 year
- Lag from refab. to reload : 0.5 year

Table A2.5.6 DUPIC Fuel Fabrication Facility Attribute Set

	DUPIC Fuel Fabrication facility
Construction Lead Time, y	5
Licensing Lead Time, y	2
Plant Lifetime, y	40
Throughput Tonnes HM/y Rate, y	400
Storage Prior to Processing, y	10
In Process Dwell Time, y	½
Total out of Reactor Time, y	1½
Loss to Waste per Recycle/Refab pass (%)	0.5

Table A2.5.7 Unit Costs for Fuel Cycle Components Relating to CANDU-DUPIC Fuel[†]

Component	**Description**	**Unit cost**			**Unit**
		Lower bound	*Nominal Value*	*Upper bound*	
$Cost_{DUPICfab}$	DUPIC fuel fabrication	448	616	784	$/kgHM
$Cost_{DUPICinstore}$	DUPIC spent fuel interim storage[1]	21	32	42	$/kgHM.year
$Cost_{DUPICsftransport}$	DUPIC spent fuel transport[2]	22	28	33	$/kgHM
$Cost_{DUPICgeo}$	DUPIC spent fuel conditioning and disposal[3]	73	167	279	$/kgHM

[†] All costs are expressed in 2000-dollars. All unit costs related to DUPIC fuel cycle are well described in the reference paper below. In the paper, unit costs of PWR fuel cycle are also described, but the values for transportation, interim storage, and disposal are a little different from the values described in the Table A2.3.1 (Unit Costs for Fuel Cycle Components Relating to LWR-Reactor) of Crosscut Group Report. In order to maintain the consistency of the DUPIC data with PWR data shown in the report, those data are converted by the ratios of DUPIC values to PWR values considering original Table A2.3.1's values.

[1] The unit cost of DUPIC spent fuel interim storage is estimated to be 109 $/kgHM.

[2] DUPIC spent fuel transport cost is estimated to be 28 $/kgHM, which is about 55% of LWR spent fuel transport cost in the paper.

[3] DUPIC spent fuel conditioning and disposal cost is estimated to be 167 $/kgHM, which is about 56% of LWR spent fuel conditioning and disposal cost in the paper.

Table A2.5.8 Radkowsky Reactor Attributes

Power Rating, MWth	3000			
η, %	33			
Load Factor, %	85			
	Seed		Blanket	
Refueling Interval, yr	1		10	
# Batches	3		1	
Burnup, GWd/t	54[a]		100	
kg, HM/ yr	Seed		Blanket	
	Input	Output	Input	Output
HM	3625[b]	3206	4450	4450
Total U/ yr	3625	2689	445	390
Thorium	0	0	4005	3819
U235/yr	725	128.4	89	5.46
Pu	0	36.6	0	11.8
U233	0	0	0	63.5
FP	0	475.6	0	222.5
MA[c]	0	4.809	0	6.675

a - Annual burnup
b- An average over cycles 2-9. Initial core loading is 6900 kg HM.
c- Assumed the same minor actinides fractions as LWR (~0.0015 HM)

Table A2.5.9 Power Plant Attribute Values for the Molten Salt Reactor (MSR) Fuel Cycle

Attribute List for MSR Fuel

Power Plant

- Rating (MWe) : 1000
- Station efficiency (η): 44%
- Construction lead time (y); 4
- Lifetime (y) : MSR is assumed to continue beyond the usual 60 years lifetime which is equivalent to reusing the molten salt with its heavy metal in a new reactor
- Capacity factor : 90%

Working Inventories:

- $\frac{\text{kg IHM}}{\text{MW}_{\text{th}}}$: 143,113/2273 = 63
- Mass loading fraction: MSR fuel is fully loaded in whole MSR core.
- Average Enrichment : 19.9% U235/(Total U) for startup with U235
  19.9% (Pu+MA)/(U238+Pu+MA) for startup with Pu + MA
  (i.e., 3115/143,113 = 2.2% enrichment, for both cases)
- Fertile material (U) : Th, U238

Mass Flows:

- Refueling Interval : on-line refueling
- # of reload batches : NA
- Ave discharge burnup: fuel remain in the core during the reactor lifetime
- HM feed during operations: 801 kg Th/year, 155 kg DU/year
- Processing waste: 526 gm Pu+MA (all goes to HLW and end up in repository with FP)
- FP removed from reactor: 801 + 155 = 956 kg/year
- Mass fractions in Discharge Fuel : Assumed to be the same as the startup fractions (the startup fractions used here are actually the equilibrium fractions and all generated actinides are consumed during operations and only a small fraction of the Pu and MA are lost during the processing of the molten salt)

Generation-IV Fuel Cycle Crosscut Group Report

Attachment 3:

Supplemental Scenario Results

March 18, 2002

"This document has not been patent-cleared and has not received DOE approval for external release. Do not distribute beyond the Generation IV Roadmap Project Participants"

Attachment 3: Supplemental Scenario Results

Attachment 3: Supplemental Scenario Results

A3.1 Introduction

All scenarios described in Chapter 3 for the WEC/IIASA Case B energy demand growth rate were rerun for the slower WEC/IIASA Case C2 energy demand growth rate. The results of those Case C2 scenarios are shown here with minimal elaboration and interpretation.

NOTE: The figure numbering scheme is coordinated with that in Chapter 3 – e.g., chapter 3, Figures 4, 5, and 6 for the LWR once-through base case for growth rate B are Figures 4, 5, 6 here for growth rate C2. Some figure numbers are missing here as a result.

This attachment also contains the results of several scenarios for the WEC/IIASA Case B energy demand growth rate which were developed as sensitivity studies during preparation of the "cornerstone" scenarios discussed in Chapter 3. They are documented here for supplemental information – with minimal elaboration and interpretation – and are referred to in Table 3 in Chapter 3.

Additionally this attachment contains the results of several scenarios run (for the Case B growth rate) to evaluate the sensitivity of the fuel cycle cost index to variations in unit process costs and are referred to in Section 3.6 of the main report.

Finally a case is shown here to evaluate the potential for nuclear to dramatically expand its scope into the 2/3 non-electric primary energy sector via manufacture of hydrogen as a synthetic chemical fuel to supplant hydrocarbons.

A3.2 Scenarios from Chapter 3 – Rerun for the (slower) WEC/IIASA Case C2 Growth Rate of Global Nuclear Energy

Scenarios described in Chapter 3 are for the WEC/IIASA Case B "just muddling through" global energy growth rate – one where fossil remains dominant and where for most countries economic growth takes precedent over ecological considerations. The nuclear capacity grows from 350 GWe in year 2000 to 2000 GWe by 2050 and 6000 GWe by 2100 in the Case B projections.

Among the other growth scenarios considered by the WEC/IIASA was scenario C2 – an ecologically driven scenario where conservation is stressed and renewables and nuclear grow in market share vs. fossil in a slower overall growth rate of overall energy demand. In the C2 scenario, nuclear grows from 350 GWe in year 2000 to 1200 GWe in 2050 and 1800 GW in 2100.

All scenarios shown in Chapter 3 for growth rate Base B have been rerun and displayed here for the Case C2 growth rate. The figure numbers here are in one-to-one correspondence with those in Chapter 3.

The C2growth rate scenario figure results (figures 4 through 32) are included here without comment; the interpretation of sensitivity to energy demand growth rate is discussed in Chapter 3 for each of the generic fuel cycle types.

A3.3 Sensitivities of Breeder Introduction Date; Breeder Doubling Time; and Fast Reactor Initial Market Share vs. Energy Demand Growth Rate

A3.3.1 Discussion of Controlling Features

The global nuclear energy park taken as a whole is a dynamical system; it grows in energy output to meet externally imposed demand; it draws resources from the finite uranium ore reserves; it is sluggish in its dynamic response owning to time lags, asset longevity and interim inventories; and it transforms the uranium to not only fission products (according to cumulative energy release) but also to bred fissile plutonium and minor actinides (according to cumulative neutron fluence on the U238).

Up to now, and in the early decades of the current century, the fissile plutonium and minor actinides which is bred into the fuel during exposure in the reactors has not been needed for use as fuel because virgin uranium reserves have remained sufficient and inexpensive. This bred fissile has simply accumulated in the inventories of spent fuel. However, during the middle decades of this century, the scenarios displayed in Chapter 3 show that the uranium ore resources recoverable at low cost will become scarce, -- and a transition to reliance on the bred fissile must be made in order to continue to meet growing demand for energy.

Managing a smooth transition from reliance on virgin ore to reliance on self bred fissile, is the role for the fuel cycle. Given the externally imposed growth rate for nuclear energy, the transition is controlled by the interplay of:

a. The date for introduction of breeding,
b. The energy production fractions of the thermal vs. the fast reactors
c. The conversion ratio and specific fissile inventory of the thermal reactors, and
d. The breeding ratio and doubling time of the fast reactors.

The scenarios show that the transition will have to be executed during the 2030 to 2060 time frame; clearly, it cannot be prescribed at this time because none of the relevant factors listed above can be foreseen accurately several decades ahead. None-the-less it is possible to *understand the systematics of the transition and the sensitivities to choices made for emplacing the technologies that will be required to execute the transition*. The several scenarios presented here and in Chapter 3, Sections 3.5.2, 3.5.3, and 3.5.4 serve that purpose.

The global nuclear energy park as a whole displays a neutron excess per unit energy release which depends on the ratio of fast to thermal reactors in the park. These

excess neutrons can be wasted to leakage and to parasitic capture in poisons or they can be used to generate new fissile atoms by capture on U238 to a greater or lesser extent. For an idealized example, assume a nongrowing energy park comprised of thermal reactors having conversion ratio 0.6 and market (fission energy) share, X, plus fast reactors of market share (1-X), having a breeding ratio of 1.4. The excess neutron population of the park overall will be used to exactly self replicate new fissile atoms to replace those destroyed by fission when X is such that the fissile balance satisfies the equation:

$$X\ (0.6) + (1\text{-}X)\ 1.4 = 1$$

A 50/50 market share (i.e. X=0.5) will give a fissile self sufficient energy park which requires no virgin ore and will be self supplied with fissile simply by drawing fertile U238 from the multi century inventory of enrichment tails and converting it to plutonium and minor actinides.

Suppose the park is growing at a constant annual percentage growth rate such that the reactor deployment must double in size every 20 years. In that case the right member of Eq. (1) must exceed 1 by an amount such that in 20 years enough excess fissile has been generated to double the size the energy park – i.e., double of the fissile working inventory. In fact, it must exceed 1 by even more – to account for the out-of-reactor fissile inventories in the recycle and fabrication pipelines. These out-of-reactor fissile inventories for the fuel cycle actually exceed the in-reactor inventories. For a non-growing (or a declining) park they are not important in Eq. (1); but for a growing park they must be accounted for – or else the park can not keep up with growing energy demand.

For a growing park, it is evident from Eq. 1 that the value of the thermal reactor market share, X, must decrease so that the breeder market share can increase sufficiently to generate the excess fissile needed for working inventory of incremental deployment – including the necessary out-of-reactor components. A still further decrease in X can be anticipated to account for the need to compensate the decommissioning of plants as they go off line at the end of their lifetime.

Consideration of Eq. 1, and the arguments above make it clear that to maximize the thermal reactor market share, X, in a fissile self sufficient growing Gen-IV nuclear energy park, the thermal and fast reactor types should have the following attributes:

Thermal Reactors Systems
- high conversion ratio
- small fissile specific inventory (kg fissile/kw)

Fast Reactors (Closed Fuel Cycle) Systems
- high breeding ratio
- high power density (to minimize fissile specific inventory (kg fissile/kw)
- short out of reactor turnaround time

(to minimize out of reactor fissile inventory)

Given that we understand what system features will favor a smooth transition to fissile self generation, the systematics of the interplay of date for breeding introduction vs. energy demand growth rate, virgin ore supply, and reactor plant attributes can be understood (at least in broad outline) by reference to Figure A3.3-1. A global energy park growing at a specified annual percentage growth (doubling time) is shown on a semi log scale (line A-B). With breeding fuel cycles introduced such that the overall park doubling time is shorter than that of the demand, the park will eventually become fissile self sufficient (line C-D). The area enclosed by (O-A-D-C-O) – if on a linear-linear plot – is proportional to the virgin ore requirements needed to support the park up until the time it becomes fissile self sufficient. Clearly those ore requirements are reduced by:

a. An earlier initiation of breeding (move point C to the left – closer to the origin)

or

b. A shorter park doubling time (steeper rise of line C-D). As discussed above this could be achieved by:

   i. Using thermal reactors of higher conversion ratio and/or smaller fissile specific inventory

and/or

   ii. Using fast reactors of shorter doubling time – i.e., higher breeding ratio, or higher power density, or shorter out of reactor fuel cycle turn around time.

Additionally, line C-E-F indicates that for a park of a given mix of reactor types (slope C-D specified) and a given date for breeding introduction (C), the virgin ore reserve withdrawals can be reduced by having anticipated and previously emplaced fast reactors and closed fuel cycles – but functioning prior to date (C) as burner or fissile self-sufficient reactors (for park waste management). Their market share can quickly become a breeder market share with only a 1 year time delay to change their core loading from a burner to a breeder. The virgin ore indicated by area C-E-F-D-C can be saved.

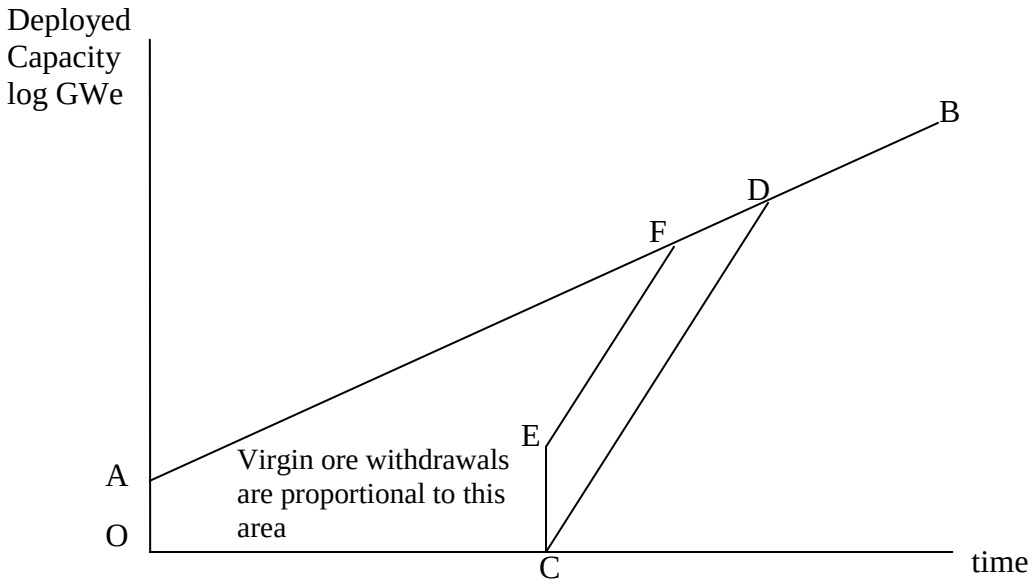


Fig. A3.3-1

A3.3.2 Illustrative Scenarios

The considerations discussed above are illustrated by scenario results which are tabulated in Table 3.3 of Chapter 3 which is reproduced here.

The scenario result plots indicated in the table are included here without commentary.

Table A.3-1 Transition To A Fissile Self-Sufficient Energy Park

Case / Growth Rate	LWR + FR(BR=0.5)		LWR + FR (BR=1.0)		LWR + FR(BR=1.25)		LWR + FR(BR=1.72)	
	B	C2	B	C2	B	C2	B	C2
Year of Introduction	2027	2027	2020	2020	2030	2030	2020	2020
Exhaust Redbook Ore (y)	2053	2070	2055	2082	2056	2100	∞	∞
Required New Ore by 2100 (tonnes)	$33 \cdot 10^6$	$6.5 \cdot 10^6$	$26 \cdot 10^6$	$2.4 \cdot 10^6$	$23 \cdot 10^6$	0	$1 \cdot 10^6$ excess	$4 \cdot 10^6$ excess
Spent UOX Inventory ⇒ 0 (y)	2045	2087	2066	2078	2042	2062	2070	2050
Max of UOX Excess (kt)	270	380	300	300	255	415	255	255
LWR/FR Ratio	80/20	65/35	65/35	50/50	50/50	50/50	50/50	50/50
year reached (y)	2045	2080	2100 & still growing	2072 40/60 2100 & still growing	2100 & still growing	2065 10/90 2100	2044 0/100 2100	2046 0/100 2080
Max Pu Excess (kt)	3	0.9	1.2	1.2	4.5	3.1	28	27
start of escalation (y)					2092	2090	2060	2030
Cost Index mill/kwhr	10	10	12	12	14	16.5	25	26
Location of Full Set of Plots Figures	Chapt. 3 19-21	Attach. 3 19-21	Chapt. 3 23-25	Attach. 3 23-25	Attach. 3 33-35	Attach. 3 36-38	Attach. 3 39-41	Attach. 3 42-44

A3.4 Sensitivity To Cost parameters

Chapter 3, Section 3.6 contains a discussion of the sensitivity of the fuel cycle cost index to variations in assumed unit process costs.

The scenario results which support that discussion of cost sensitivity are presented here without commentary in Figures A3.45 through A3.59.

A3.5 Feedback of Fissile from Breeding to Fuel Thermal Reactors

A3.5.1 Feedback of Fissile from Breeding to Fuel LWR-MOX Deployment

Chapter 3, Section 3.5.4 displays a case (for the Case B energy growth) where, after achieving sufficient market share, the breeders generate excess fissile – i.e., in excess of that required to provide for the working inventories of new fast reactor deployments -which is fed back to fuel LWR-MOX power plants so that the drawdown of virgin ore reserves can be reduced. Based on the discussions in Section 3.3 of this Attachment, it was expected that the energy park would settle into a quasi-steady-state 40/60 to 50/50 fast reactor/thermal reactor market share as the mass flows come into balance within several decades. The surprising result was (Figures 30, 31, 32 of Chapter 3) that the fast reactor share was required to first overshoot the asymptote market share in order to assure fissile self sufficiency within small overruns of the Redbook virgin ore reserves; and only late in the century was sufficient excess fissile available to increase the thermal reactor market share toward the asymptotic ratio. The asymptote would not be reached until the year ~2230 (a century after the transition to fissile self generated sustainability was initiated). This results from the extreme sluggishness of the energy park's dynamic response; the LWR's once-through plants have a 60 year life, are deployed up until ~2030, and continue on a UOX once cycle the whole time; the breeders have a short doubling time, but start out at zero and as illustrated in Fig. A3.1 take a while to catch up. While it takes a long time to get there, this illustrates the physical capability to transition to a symbiotic park of fast and thermal reactors fueled entirely by ^{238}U from enrichment tails and sending only fission products and trace losses of TRU from recycle to the geologic repository.

The case was re-run for the slower (Case C2) demand growth rate (Figures A3.30, A3.31, A3.32 of Attachment 3) with not much change in the time required to transition to sustainability. Finally, a shorter out of reactor fuel cycle turnaround time (2 years reduced from 4 years) was simulated. Again the results indicate a transition period of nearly a century and an initial overshoot of fast reactor market share required to avoid excessive virgin ore demand beyond Redbook reserves.

An overshoot in fast reactor market share with a later return to increased thermal reactor market share does not seem commercially realistic. The tradeoffs are clear from the prior discussions. On the one hand, fast reactors with short doubling time are needed to catch up to the growing energy park already filled with thermal reactors. But fast reactors have large specific fissile inventories (kg/kw) so to meet growing demands for energy (kw) takes a lot of fissile (kg).

On the other hand, when fissile is in short supply, one can produce more energy by putting the fissile into thermal reactor systems – which have a much smaller specific fissile inventory (kg/kw). The trouble is, that will reduce the fast reactor market share which will then reduce the ability of the park to self manufacture the new fissile which is required to keep up with demand growths.

As discussed in Section A.3.3.1, what is needed is a proper mix of thermal reactors of high conversion ratio and low specific fissile inventory working with fast reactors of short doubling time (high breeding ratio, high power density, and short out of reactor fuel cycle turnaround time). The scenarios studied so for have not found this proper mix.

Two additional scenarios are planned for the future to attempt a transition which maintains a higher thermal reactor energy share during the course of the transition. In the first case, the already deployed LWR-UOX reactors will be converted to 100% MOX during a 7½ year transition – refueling 1/5 of the core with MOX every 18 months. While new starts will be 100% MOX LWR's plus fast reactors, the priority use of bred fissile to fuel conversion of already deployed LWR-UOX reactors will reduce the demand for buildup of out of reactor MOX inventories – while still achieving growth in thermal reactors fueled by MOX.

In the second case still to be modeled, Molten Salt Thermal Reactors will be co-deployed with fast breeders and new LWR's will not be built. Because of the extremely small specific fissile working inventory and very high conversion ratio (i.e., CR=1) of MSR systems, it is expected that a smooth transition will take place – even allowing for a significant time delay prior to initiation of fast reactor breeding.

The DYMOND code is being modified to facilitate the simulation of these cases.

A3.5.2 Feedback of Fissile from Breeding to fuel HTGR's Producing Hydrogen

As shown in the cases above, by 2065 a source of excess fissile from the fast breeder systems becomes available to feed back to thermal spectrum systems – and this fissile is available to expand the nuclear energy output beyond what is demanded for the Case B electrical energy demand growth rate for nuclear. It could be used to expand nuclear market share-vis-à-vis fossil market share in the electricity sector. Alternately, in the future, a need may arise for a broadened role for nuclear energy – supplying energy services in sectors outside the electricity sector (which comprises only a third of overall energy use). The transportation sector, comprising one third of overall energy use and contributing significantly to both local and global waste emissions, is trending toward potential use of hydrogen as a synthetic chemical fuel. Fuel cell powered cars are expected to reach the market in the first quarter of the 21st century; when fueled with hydrogen, their emissions at point of use will be small. If market penetration of fuel cell powered vehicles succeeds, up to a third of the world's future energy delivery may rely on hydrogen as an energy carrier. Hydrogen currently is made by reforming methane, but in the future hydrogen production services which avoid emissions at point of production will be needed. In anticipation of this trend, Gen-IV concept submittals include those aimed at hydrogen production via closed thermochemical water cracking cycles driven by high temperature nuclear power plants, or by electrolysis using off peak electricity from the global nuclear energy park.

A scenario was run illustrate, within the bounds of physically – achievable self-generated fuel supply, the degree to which nuclear energy could expand beyond exclusive electricity production into supplying future energy sectors which require hydrogen as a synthetic chemical fuel.

Starting in 2025 it is assumed that new capacity to meet electricity needs will be met by adding fast breeder reactors operating on full TRU recycle closed fuel cycles. The initial working inventories for these breeders, as in the previous scenarios, will be generated by recycling the existing and still growing inventory of LWR spent fuel. With the goal to determine the upper limit of physically achievable market penetration, these fast breeder reactors operate at the upper extreme of achievable breeding – having a breeding ratio of 1.72 – with extensive mass flows for harvesting excess fissile from the breeding blankets and the out of reactor cycle dwell time is four years – yielding a doubling time of seven years.

After a six decade period of fast reactor increase in electricity market share, the bred transuranic supply eventually exceeds the needs for refueling and building additional breeder reactors devoted (with the remaining LWR's) to meeting demand for electricity production – and this excess fissile can then be used to supply initial working inventories and subsequent refueling of a growing deployment of Prismatic High Temperature Gas Cooled reactors assumed here to be *operating on a TRU-fueled, once-through deep-burn fuel cycle*. These HTGR's are assumed to drive thermochemical water cracking plants which convert heat and water to hydrogen and oxygen with an assumed efficiency of 45% (lower heating value of H_2 over supplied heat). Their attributes are listed in Table A2.5.3 of Attachment 2.

As the inventory of LWR discharged fuel is drawn down by withdrawals for fast reactor construction and as the supply of newly discharged spent fuel diminishes as LWR's go off line at their end of life, a time comes when the fuel bred in the fast reactors must supply *all* future needs – for initial inventories of new fast reactors to meet growing electric demand, for refueling of already deployed fast reactors, and for new construction and refueling of the HTGR's as they are deployed to supply an ever growing fraction of the hydrogen supply energy sector. At that time, ore withdrawals cease, the energy park is supplied by U238 "fuel" from the inventory of enrichment tails and the breeding capabilities of the fast breeder systems thereafter fully control the rate of nuclear energy market penetration.

Figures A3.60 thru A3.63 show the results of the scenario run for the Case B growth rate. The doubling time of the fast reactor fuel cycle (7 years) is shorter than the doubling time of electrical energy demand growth (20 years) – so it is evident that a penetration into the nonelectric energy sectors will be achieved eventually.

It takes until late in the century for the fast reactor cycle to assume the full burden of meeting electricity demand, and it isn't until 2060 that excess fissile becomes available to build HTGR's. Subsequently their market penetration is rapid – by 2100 the hydrogen production is the equivalent of 2000 GWe of additional energy delivery capacity

deployed. When this capacity is used instead for hydrogen production; nearly one Gigatonne oil equivalent of hydrogen is produced per annum.

Start of deployment in 2025 allows for transition to sustainability within the Redbook Known plus Speculative reserves – but delay to a 2035 introduction date for the breeder, means that the forward ore requirements of the LWR's already deployed by then will exceed the Redbook Known plus Speculative reserves; however, only a further 4 million tonnes of virgin ore is needed to fuel LWR's – the last of which would go off line in the 2190's. (Alternate sources of fissile could be considered as well – further extraction of U235 from the 0.3% enrichment tails *or fissile plutonium fed back from the breeder and conversion of the LWR-UOX loadings to 100% MOX mono recycle loadings.)*

As in all scenarios using full TRU recycle, the mass flows to the geologic repository are reduced by 3 orders of magnitude relative to the reference base case early in the scenario. However, by the late 2000's the waste flow to the repository is increasing compared to previous full recycle scenarios for two reasons: (1) the HTGR once-through concept modeled here achieves only ~60% a/o burnup of charged TRU, so 40% of the HTGR fuel supply ends up consigned to waste and (2) 30% more energy is being supplied by the nuclear park compared to other scenarios – not just electricity, but hydrogen as well.

The deployment of enrichment capacity diminishes as the LWR's go off line and demand for UOX goes to zero. The same is true for LWR spent fuel recycling plants – whereas the deployment of recycle/refab plants for the fast reactor cycle and fabrication plants for HTGR fuel grows as these reactors and fuel cycle concepts expand in market share. By 2100, about 375,000 tonnes of heavy metal are fabricated per year – 3 times the base case in 2100 and ~30 times the current day capacity. Recycling plants for fast reactor cycles are at a similar level of deployment ~375,000 tonnes HM/year – over one hundred times the capacity deployed currently.

Figure A3.63 shows the breakdown of mass flows and of recycling and refabrication costs which (as shown in Fig. A3.62) dominate the cost index. High flows of breeder blanket mass through recycling are needed to harvest the self generated fissile for the park. However, as compared with the LWR once-through cycle [(see Chapter 3, Figs. 3.4 to 3.6) whose year 2100 mass flows of fuel (~130,000 tonnes/year) or uranium enrichment (~750,000 tonnes SWu/y)] are very large, the mass flows in this scenario needed to harvest fissile from breeder blankets are in the same range and even less. The cost of full TRU recovery recycling is modeled as 2000 $/kgHM – (up from the 800 $/kgHM assumed for PUREX Pu recovery processing of LWR-UOX or MOX fuel) and fabrication of TRU-bearing fuel is modeled at 2600 $/kgHM – up from 250 $/kgHM for UOX fuel. Even so the absence of a need for speculative $200 kgU virgin ore serves to partially offset these increases in processing costs.

The performance indexes, normalized year by year to the LWR-once through base case are shown in Figure A3.62. To produce a fuel cycle cost index normalized per

terrawatt hour electrical, the expedient assumption was made that the HTGR's while actually making hydrogen at 45% efficiency, instead converts heat to electricity at 45% efficiency – i.e., at the same efficiency used in the model to convert heat to hydrogen – and that the fabrication cost for HTGR TRU fuel was identical to the cost for TRU-bearing fast reactor fuel, (2600 $/kg HM), in recognition that TRU fuels will all be fabricated remotely owing to their high levels of radioactivity. These assumptions in the absence of data are plausible and they serve to illustrate that *the cost index for fuel cycle services to support a fully sustainable Gen-IV nuclear energy future serving broad sectors of societal need are only four times current costs and only two times what the cost would have become for the base case.* In light of the fact that fuel cycle services comprise a 1/5 fraction of energy cost, this escalation may be tolerable.

The penetration of the nuclear sector into the primary energy market (limited in this modeling only by available fissile) occurs rapidly once excess TRU feedstock can be turned from growing the electrical sector. By 2100 nearly a Gigatonne of oil equivalent hydrogen is being produced; when expressed in units of GWe, this is an additional ~1800 GWe over and above the 6000 GWe of actual electricity. (Note that the worldwide primary energy need in year 2100 is 35 GTOE per annum as estimated by the IIASA/WEC – Case B projections for primary global energy usage.) The energy delivery capacity of nuclear is seen to hold significant potential for expanding into non-electric energy services by the end of the century while continuing to provide its targeted market share of electricity production.

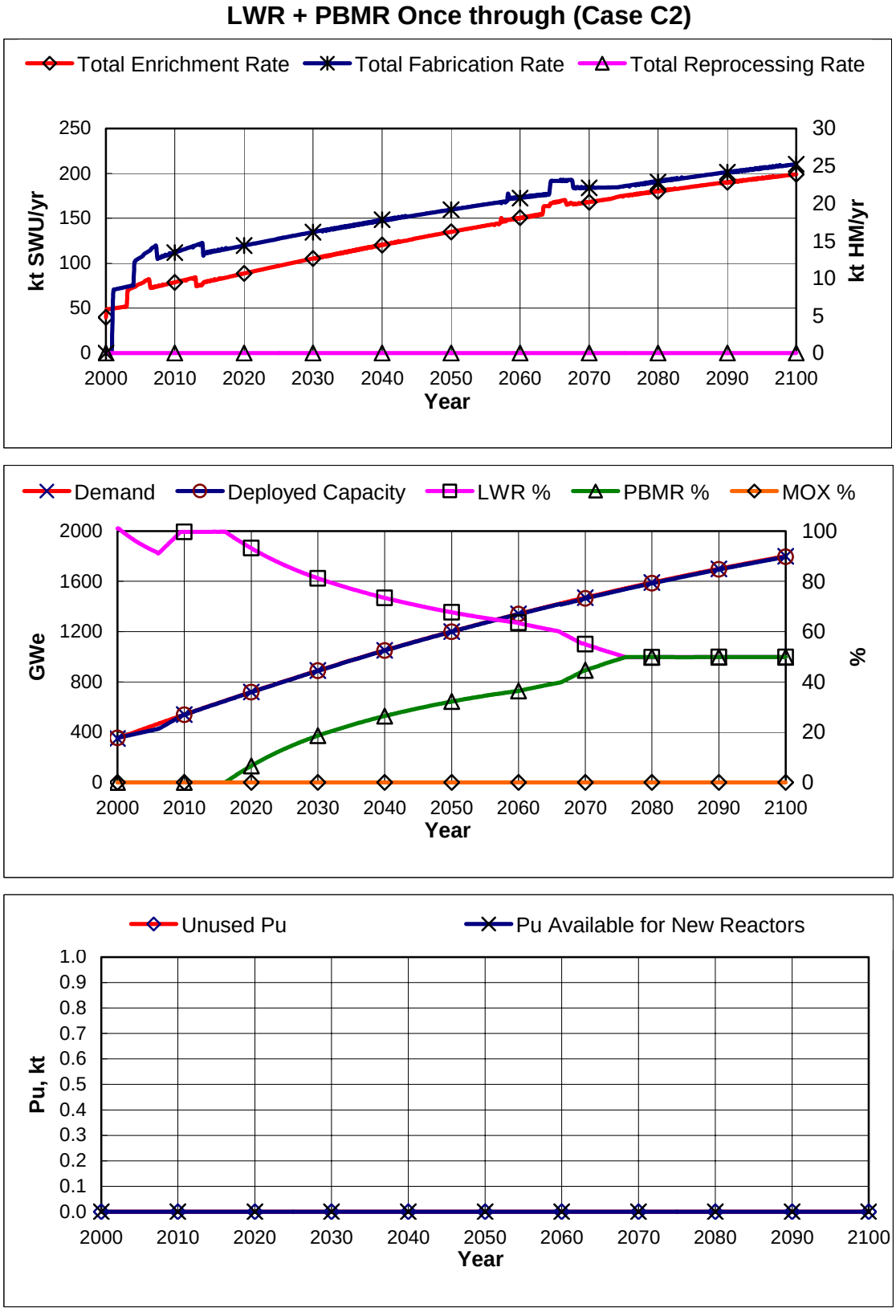

LWR + PBMR Once through (Case C2)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
PBMR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. A3-7

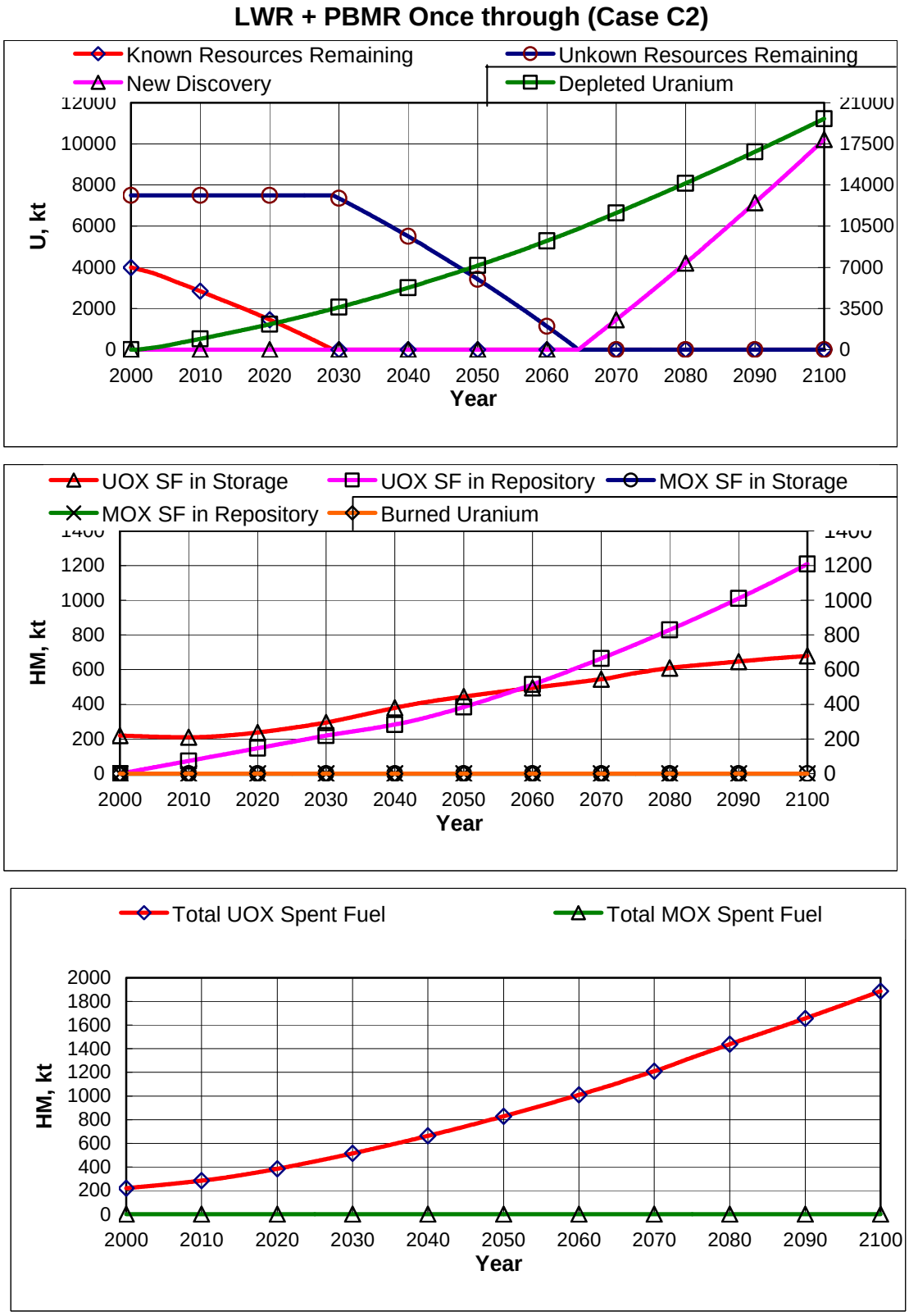

LWR + PBMR Once through (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-8

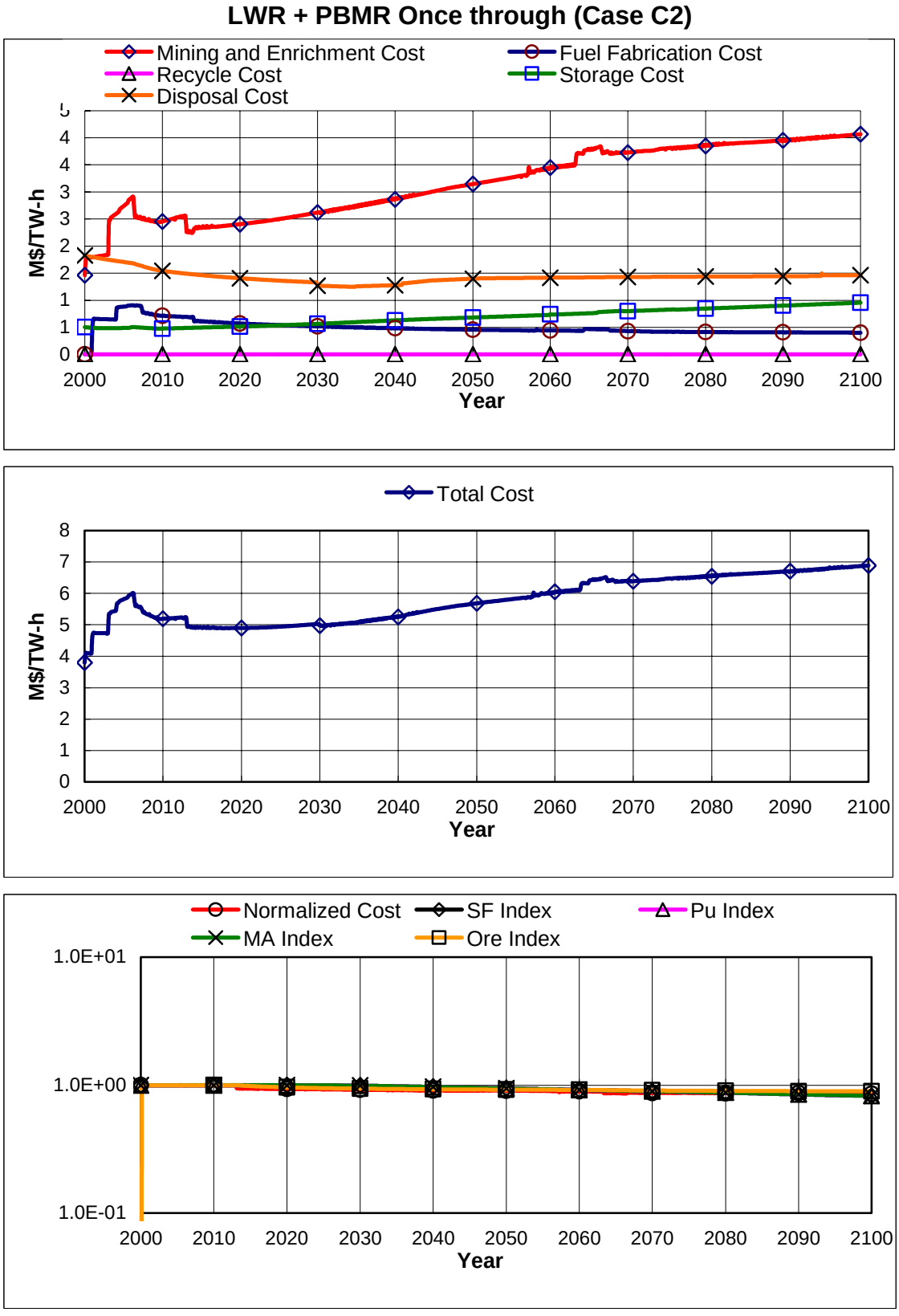

LWR + PBMR Once through (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
Year

Fig. A3-9

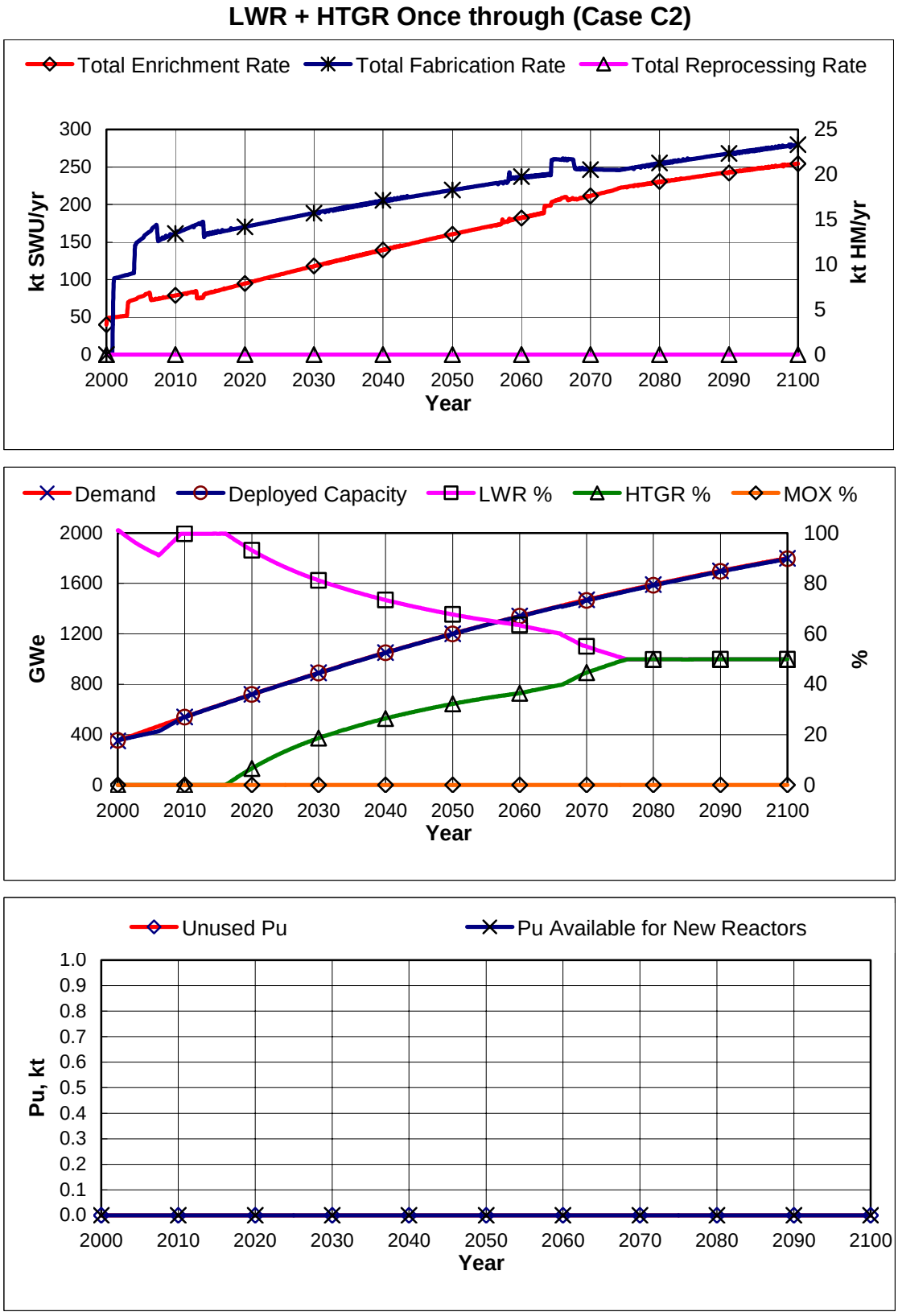

LWR + HTGR Once through (Case C2)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
HTGR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. 3A-10

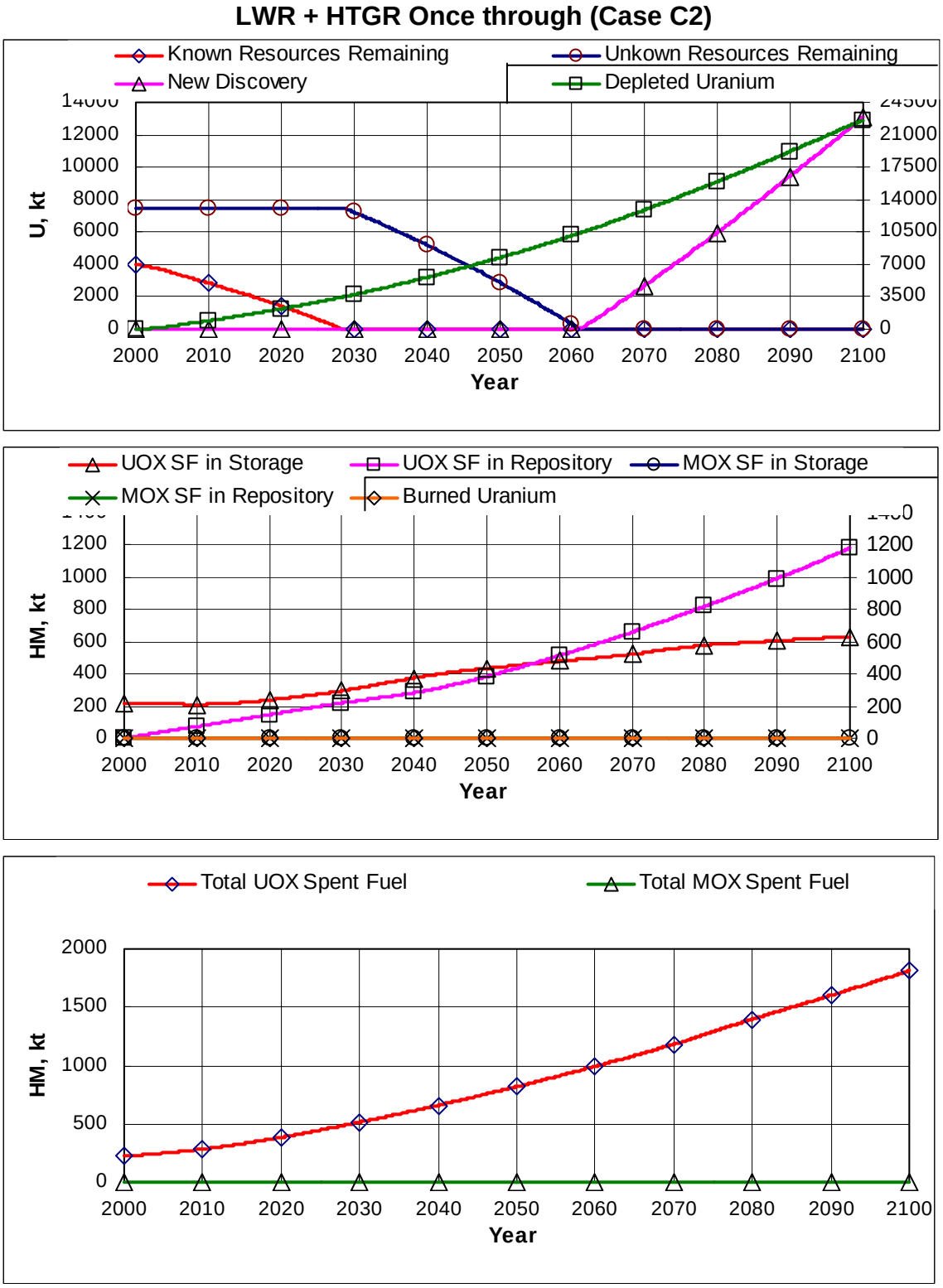

LWR + HTGR Once through (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-11

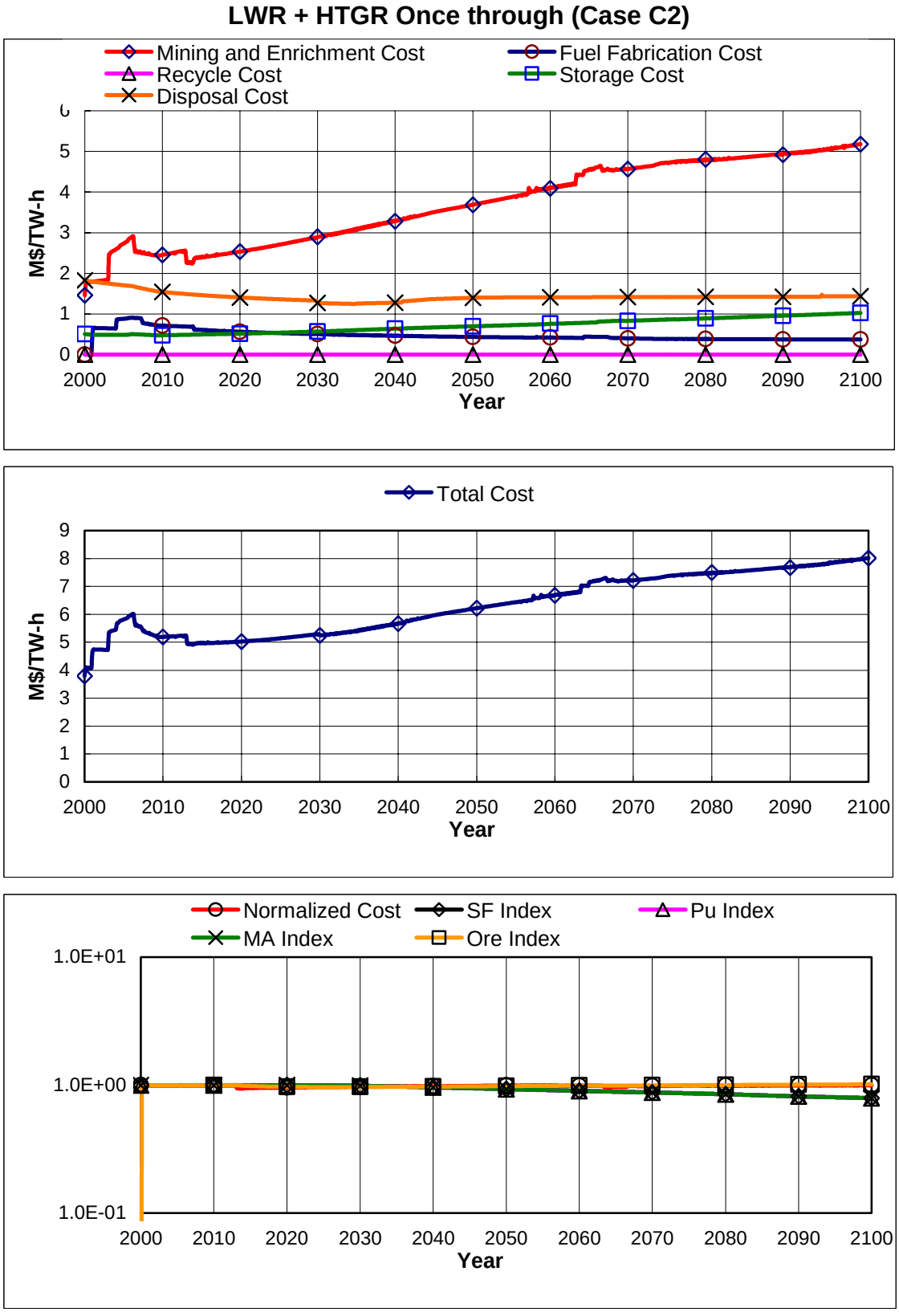
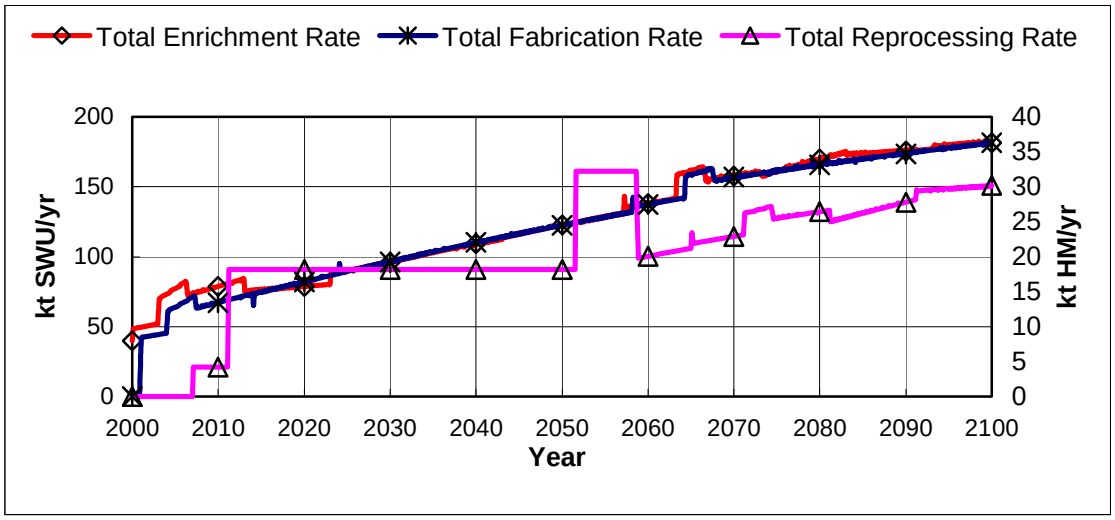
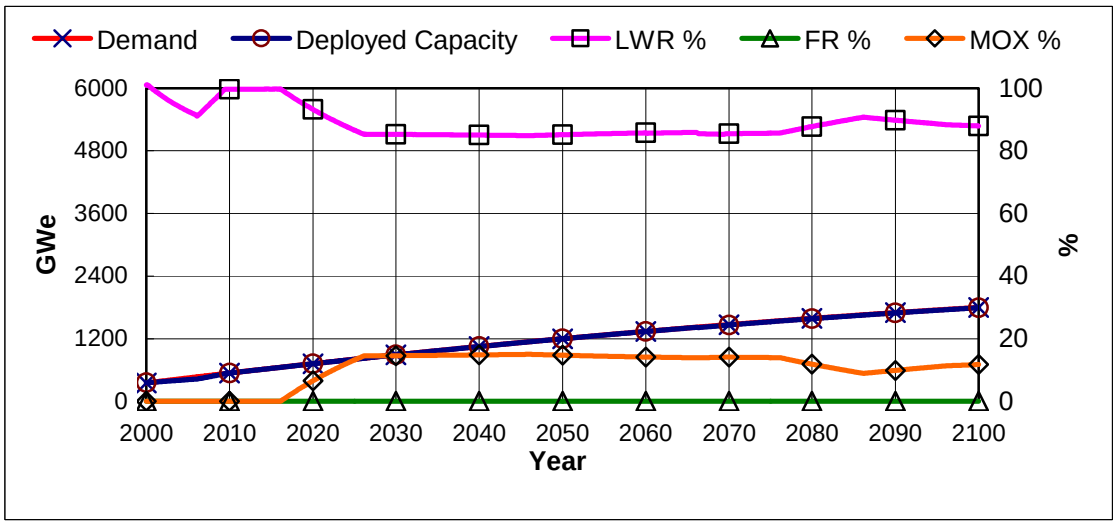

LWR + HTGR Once through (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
Year

Fig. A3-12

LWR UOX + LWR MOX (Case C2)





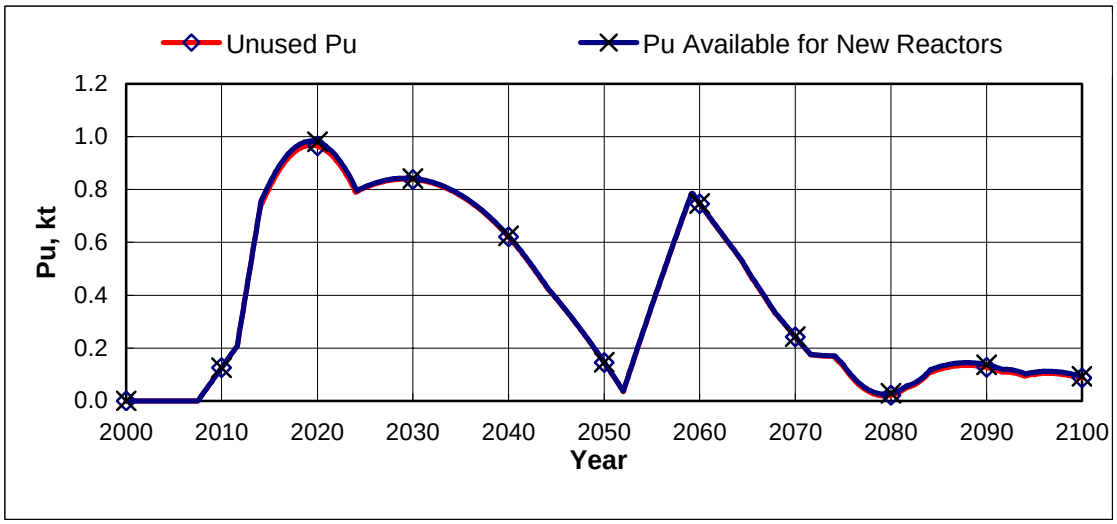


Fig. A3-13

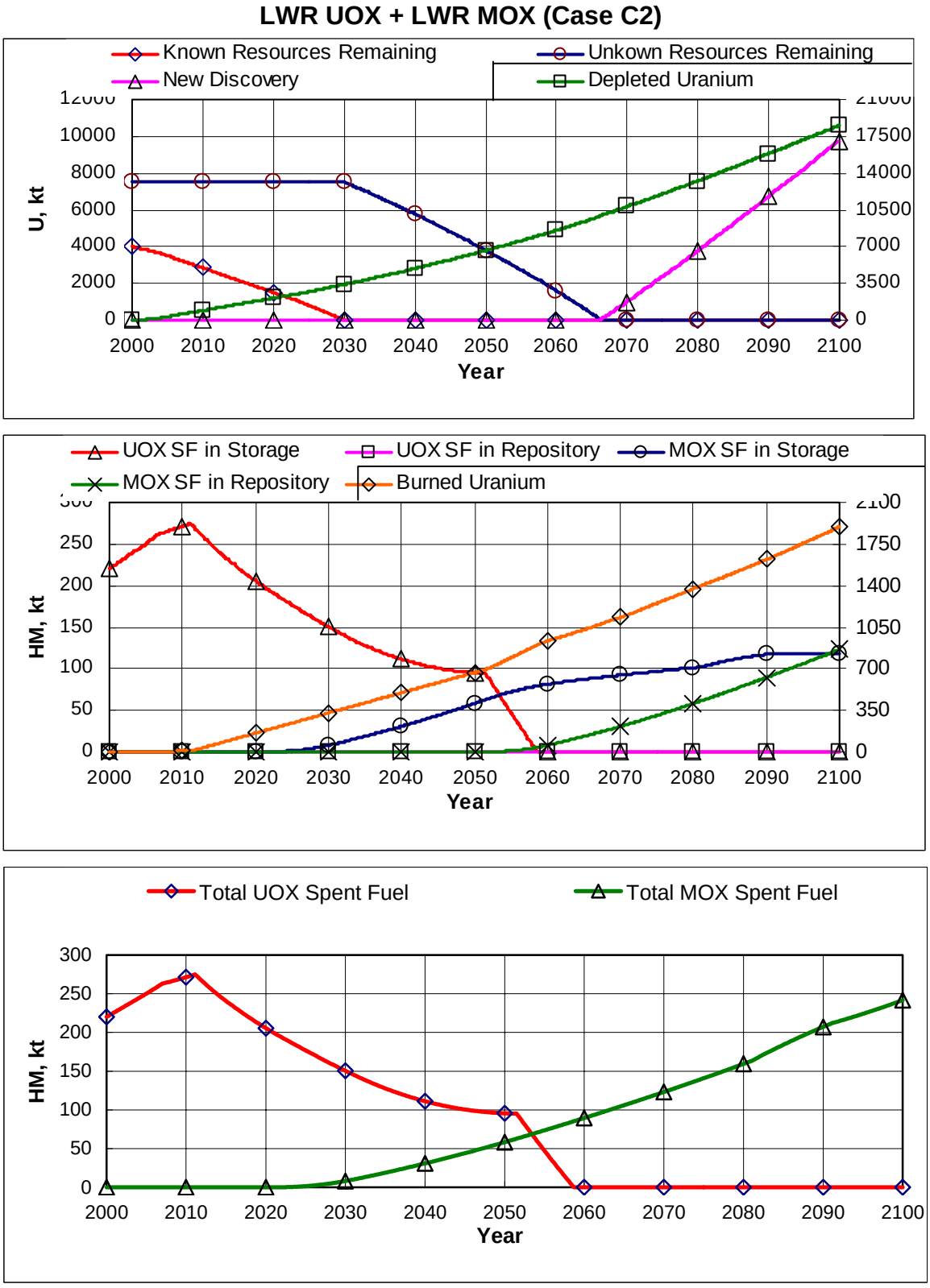

LWR UOX + LWR MOX (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-14

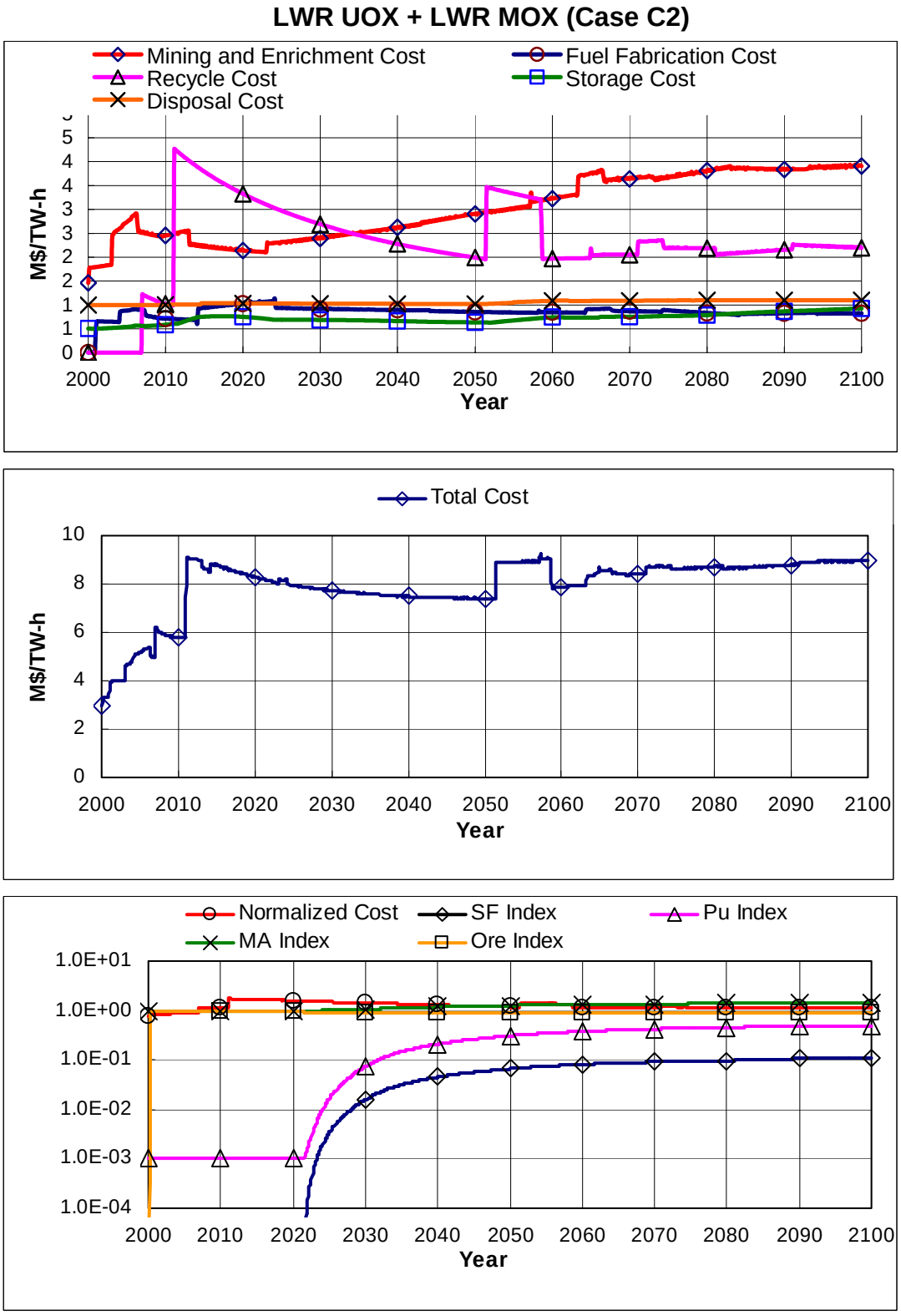
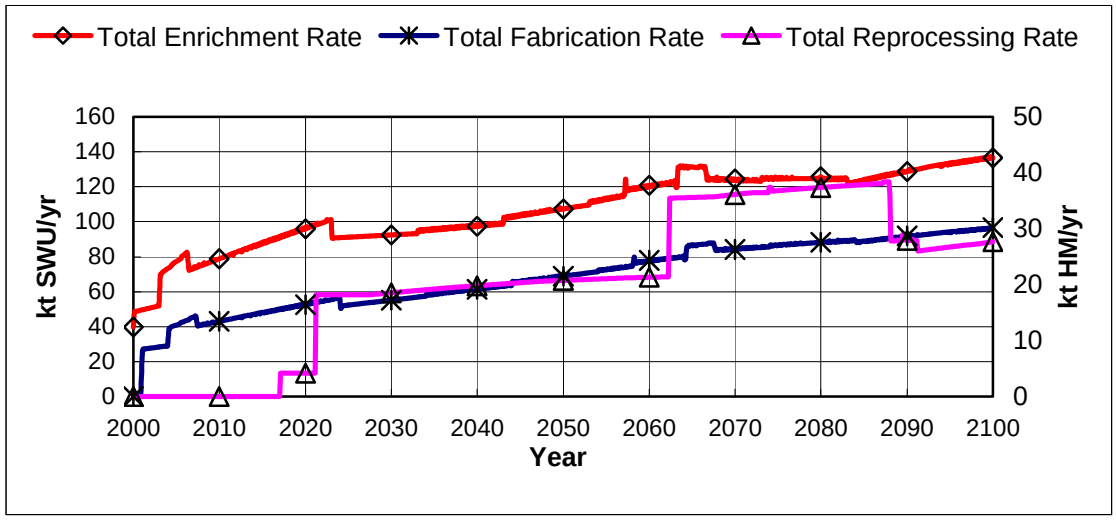
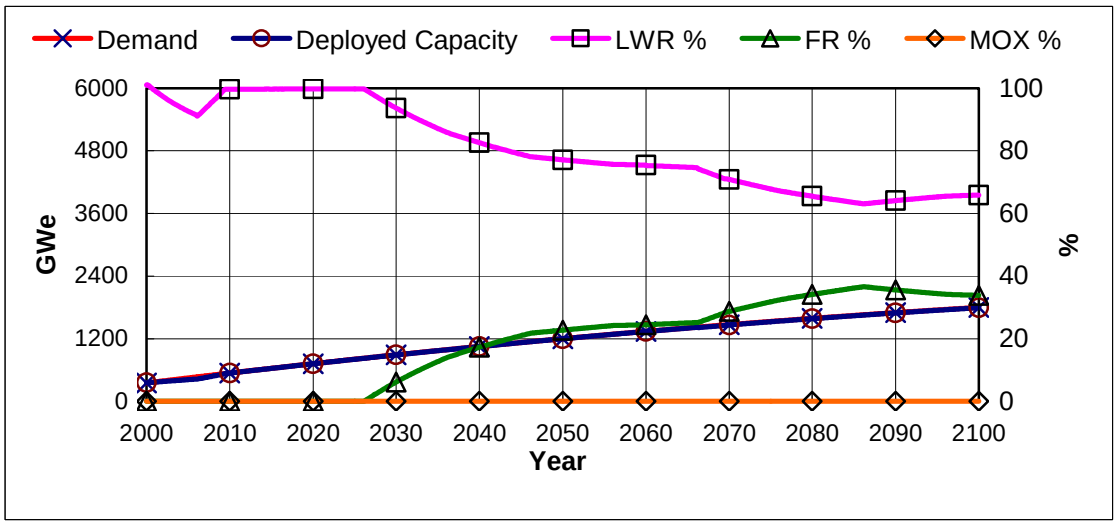

LWR UOX + LWR MOX (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
1.0E+01
1.0E+00
1.0E-01
1.0E-02
1.0E-03
1.0E-04
Year

Fig. 3A-15

LWR UOX + FR (BR=0.5) (Case C2)





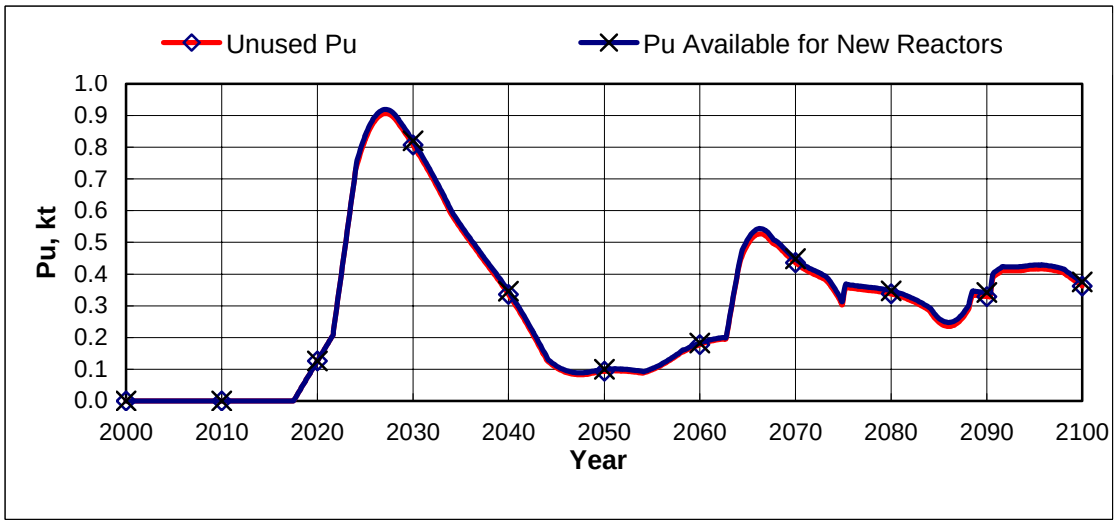


Fig. A3-19


LWR UOX + FR (BR=0.5) (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-20

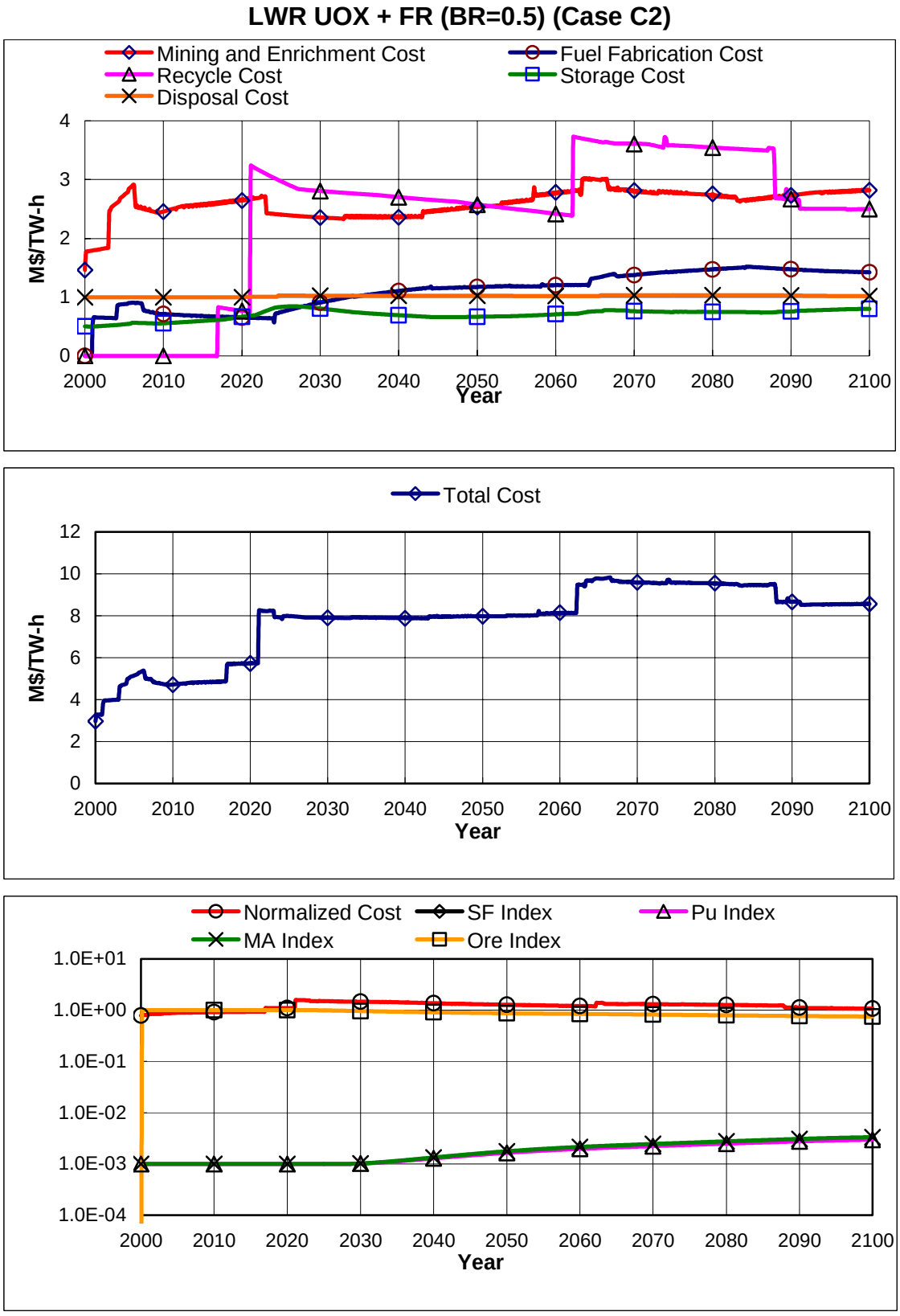

LWR UOX + FR (BR=0.5) (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. A3-21

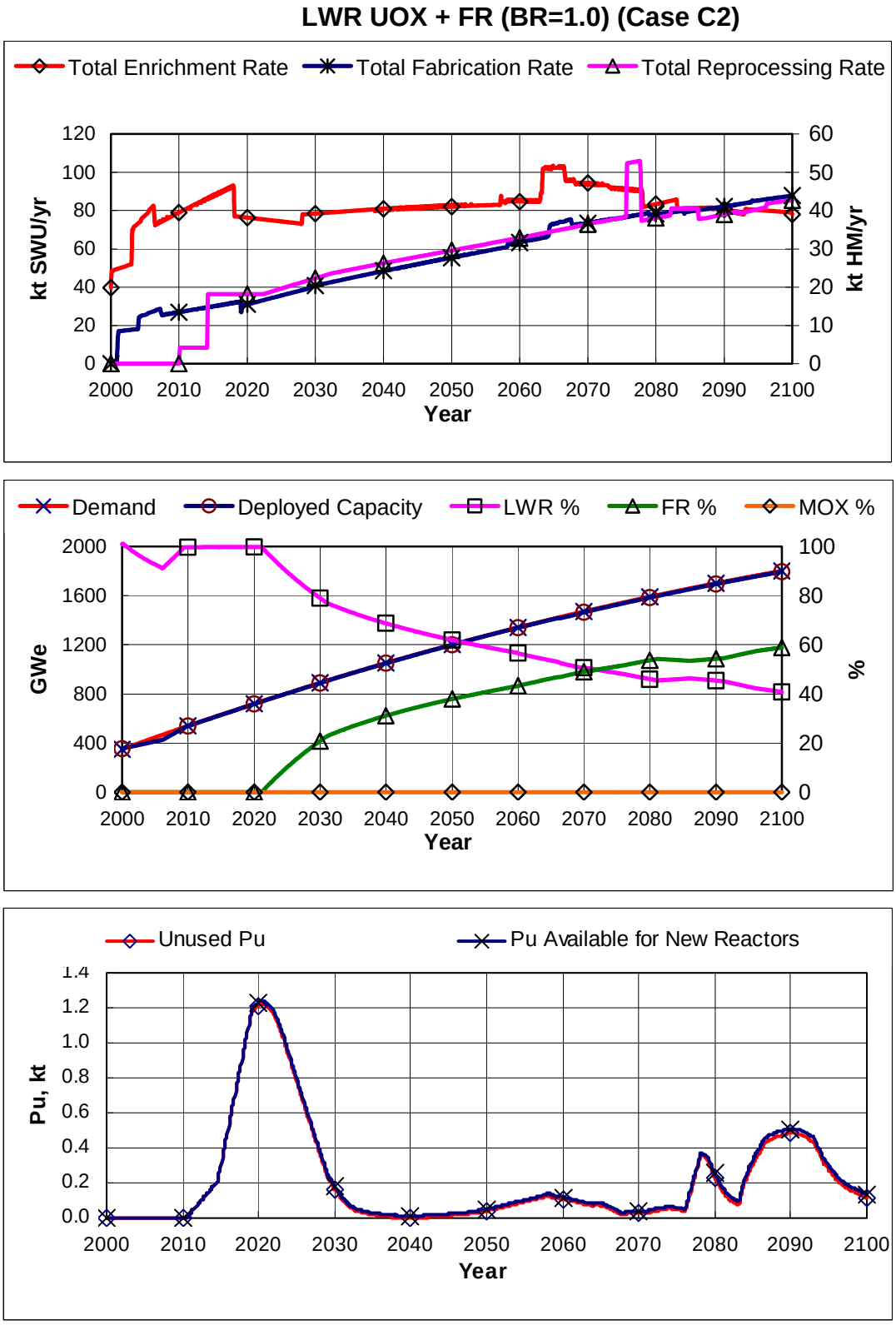

LWR UOX + FR (BR=1.0) (Case C2)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. A3-23

LWR UOX + FR (BR=1.0) (Case C2)

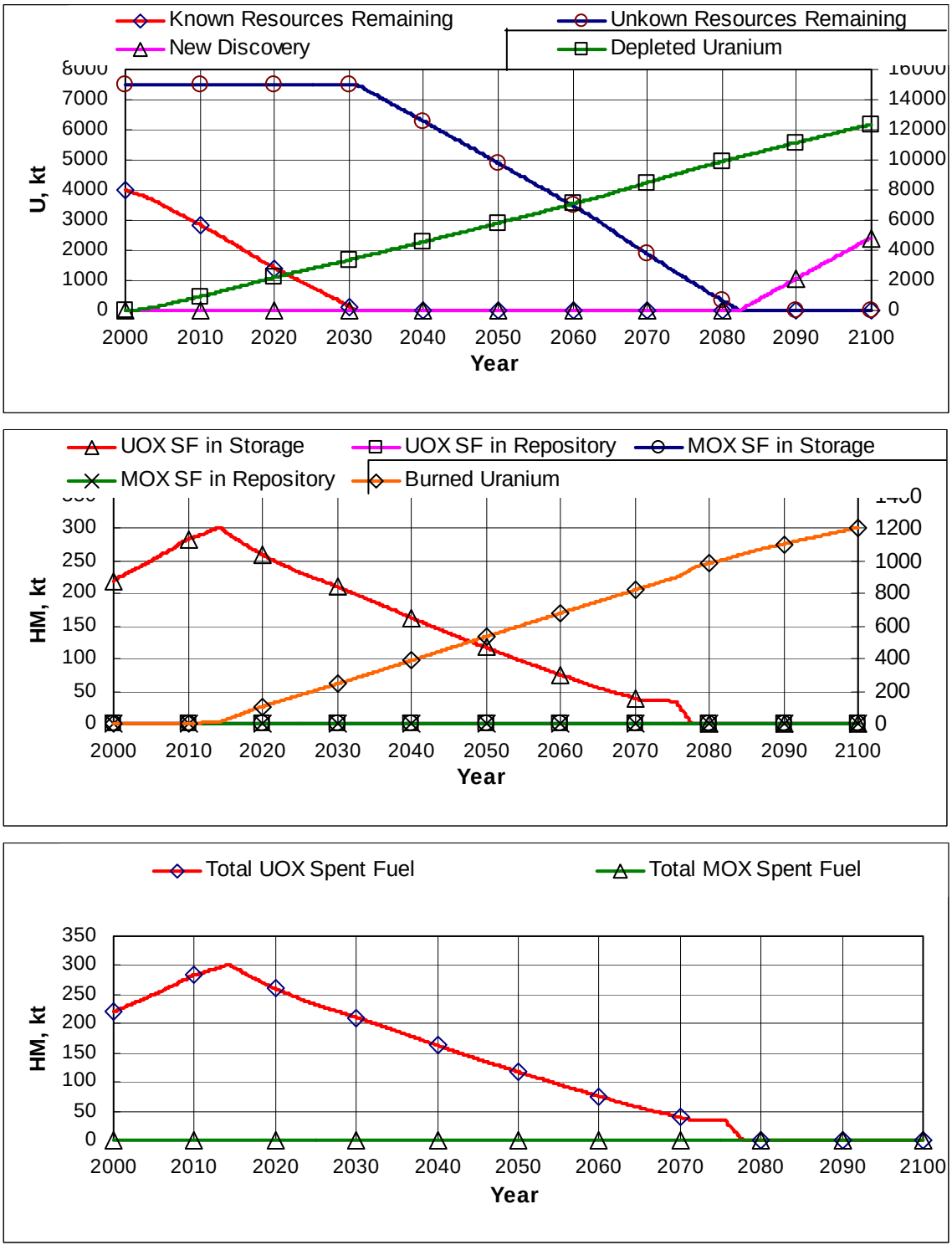


Fig. 3A-24

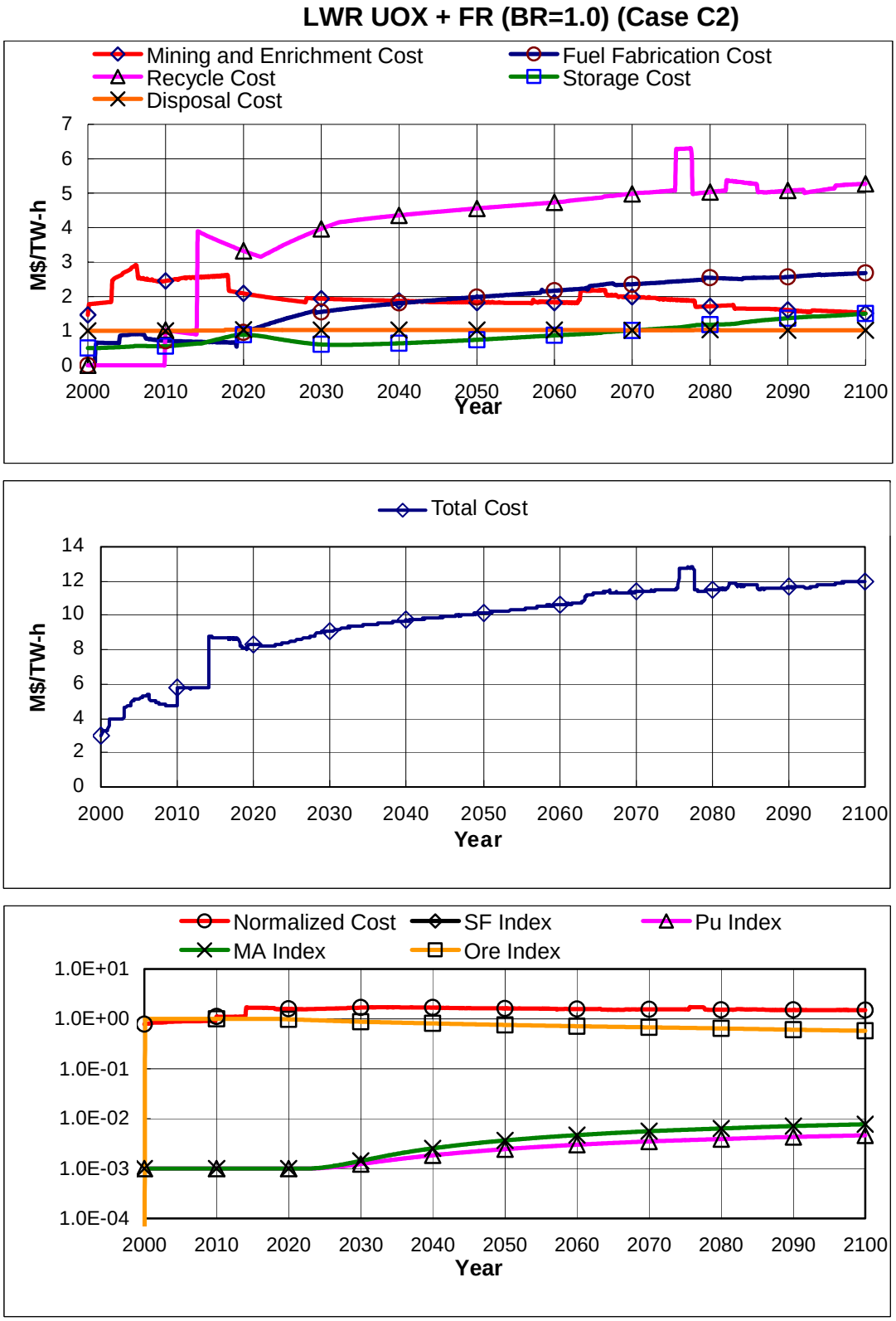

LWR UOX + FR (BR=1.0) (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. A3-25

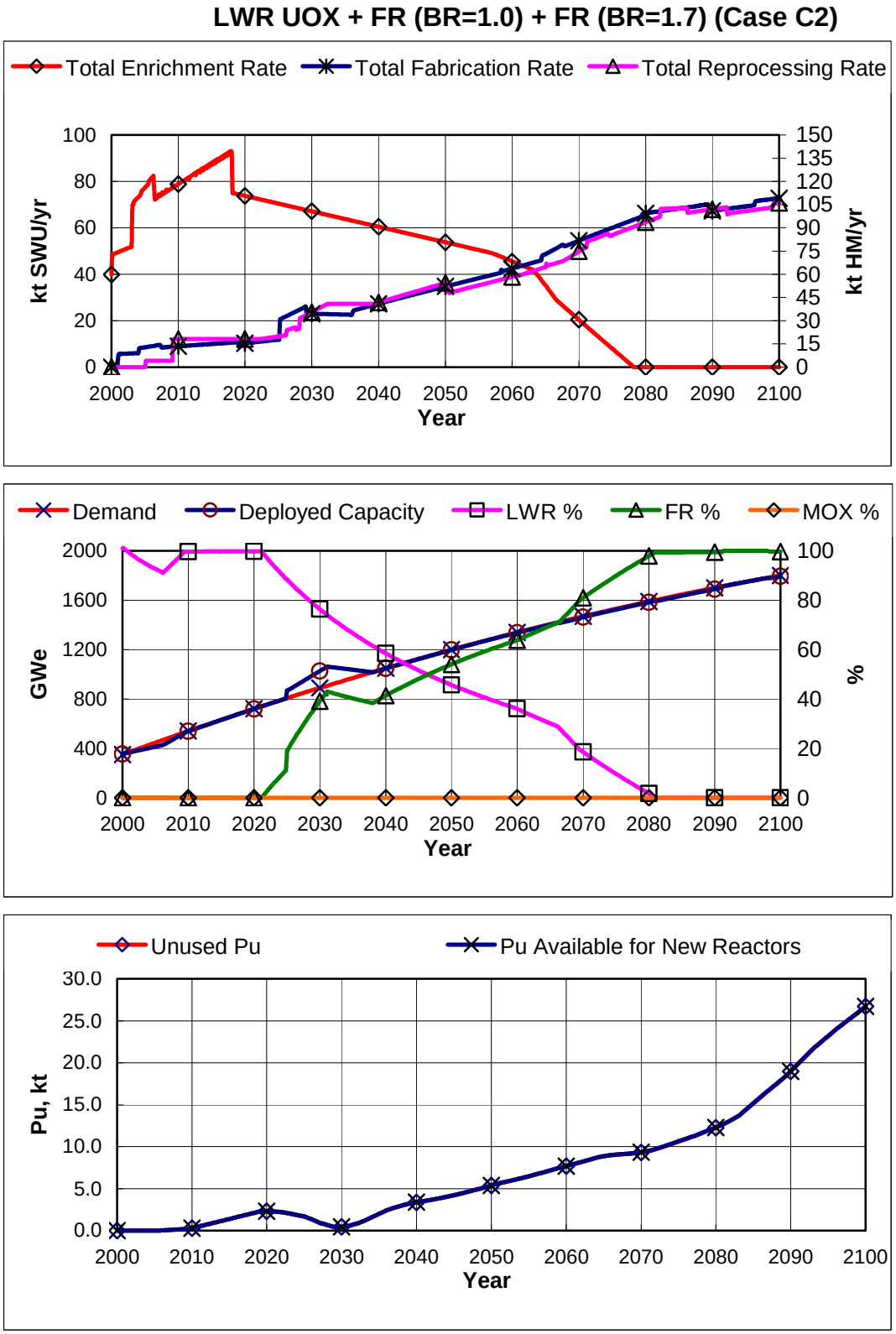

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case C2)
Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
Demand
Deployed Capacity
LWR %
FR %
MOX %
GWe
%
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
Year

Fig. A3-27

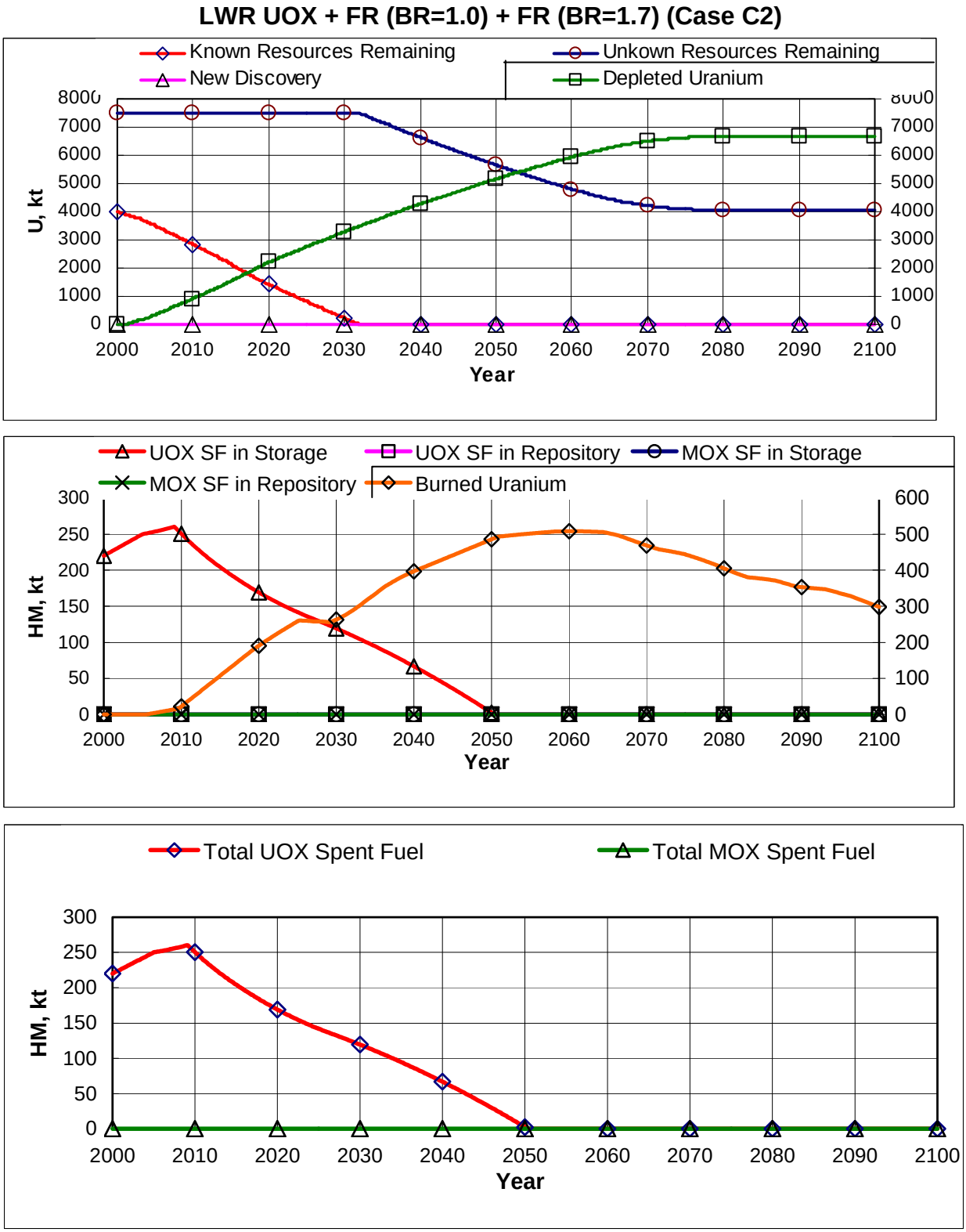

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-28

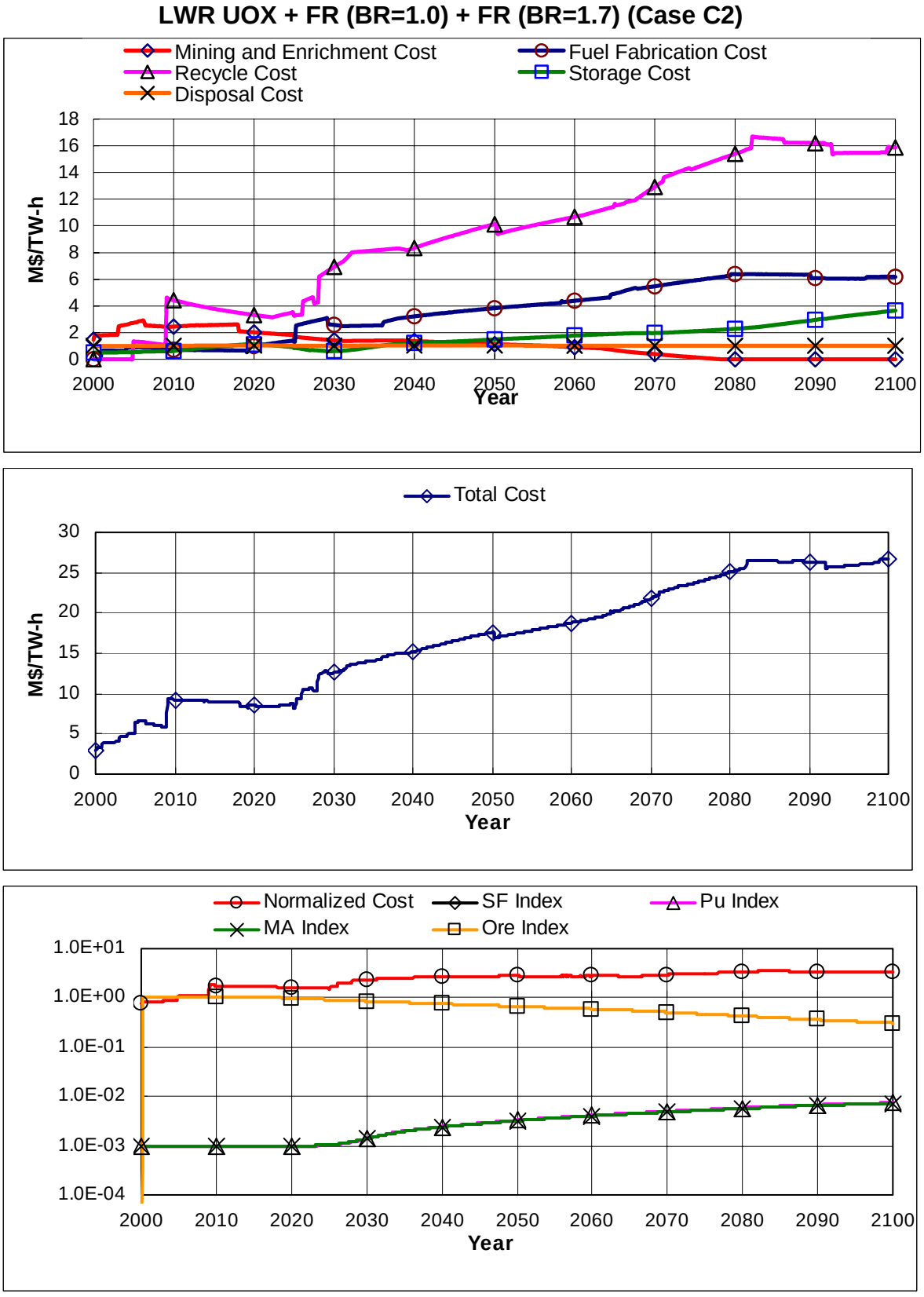
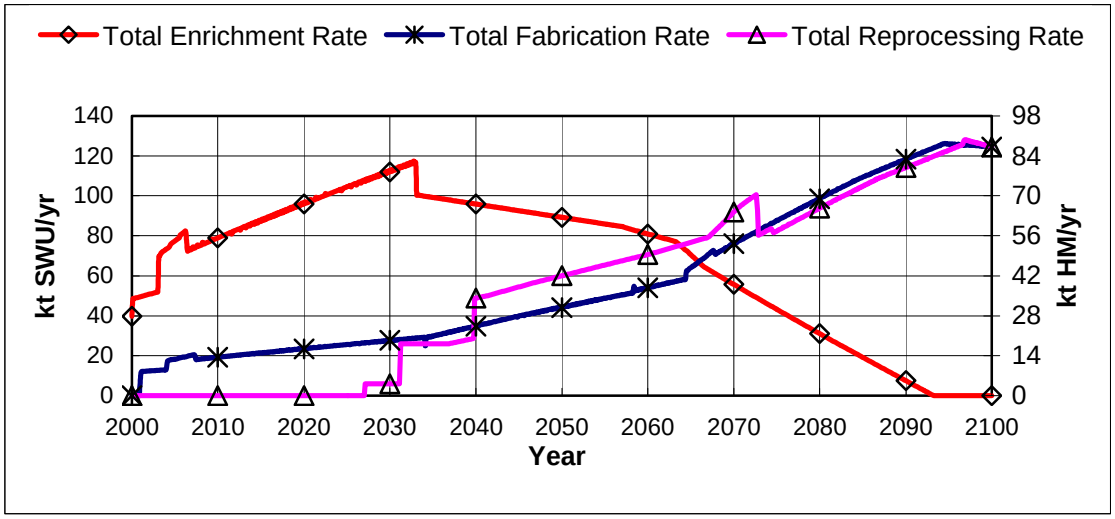
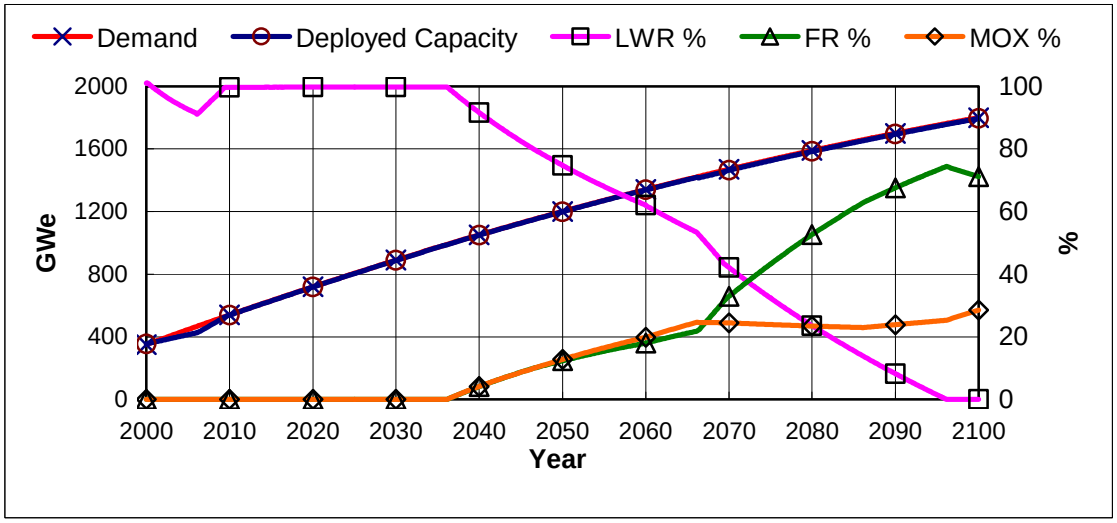

LWR UOX + FR (BR=1.0) + FR (BR=1.7) (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. A3-29

LWR UOX + FR (BR=1.7) + LWR MOX (Case C2)





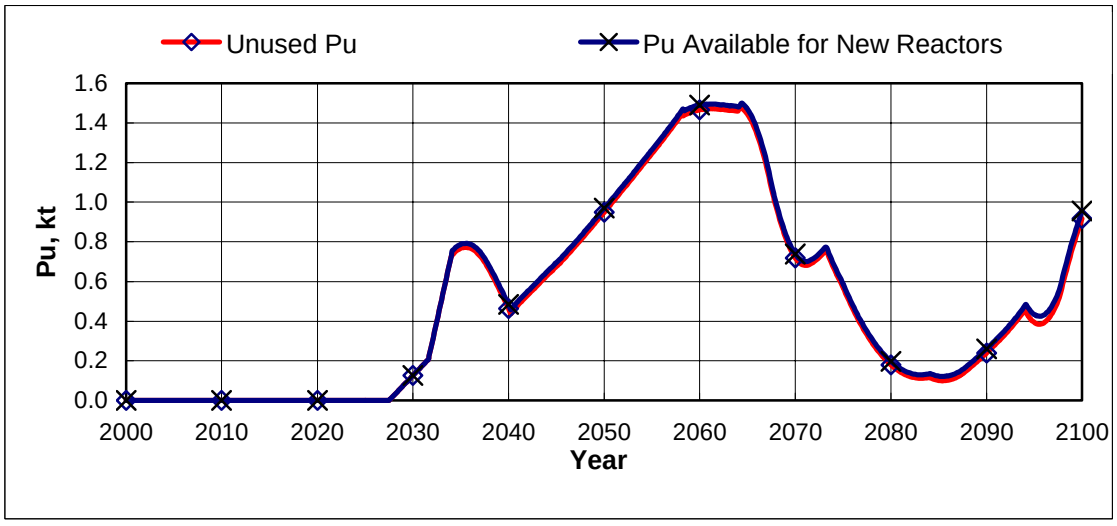


Fig. A3-30

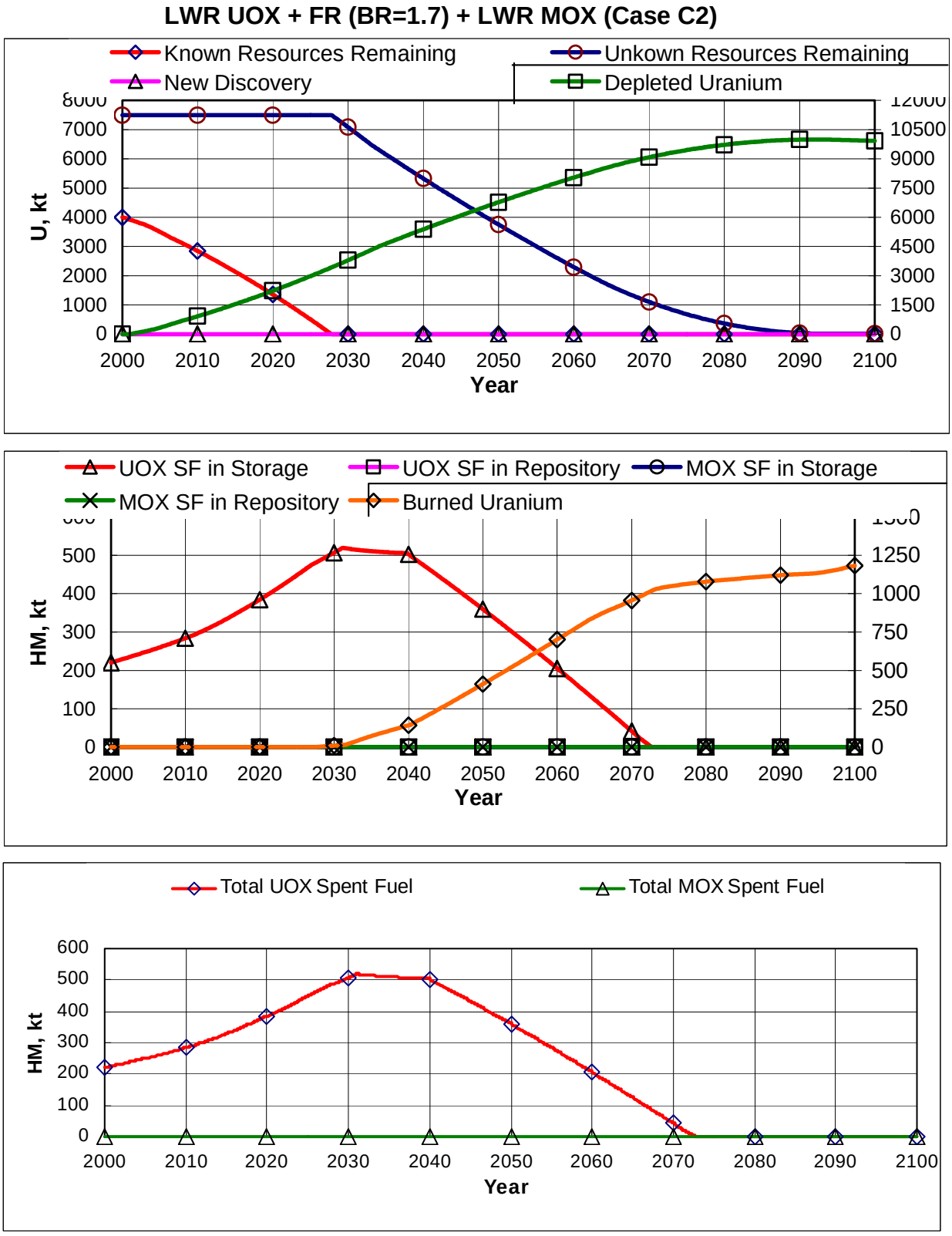

LWR UOX + FR (BR=1.7) + LWR MOX (Case C2)
Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Fig. A3-31

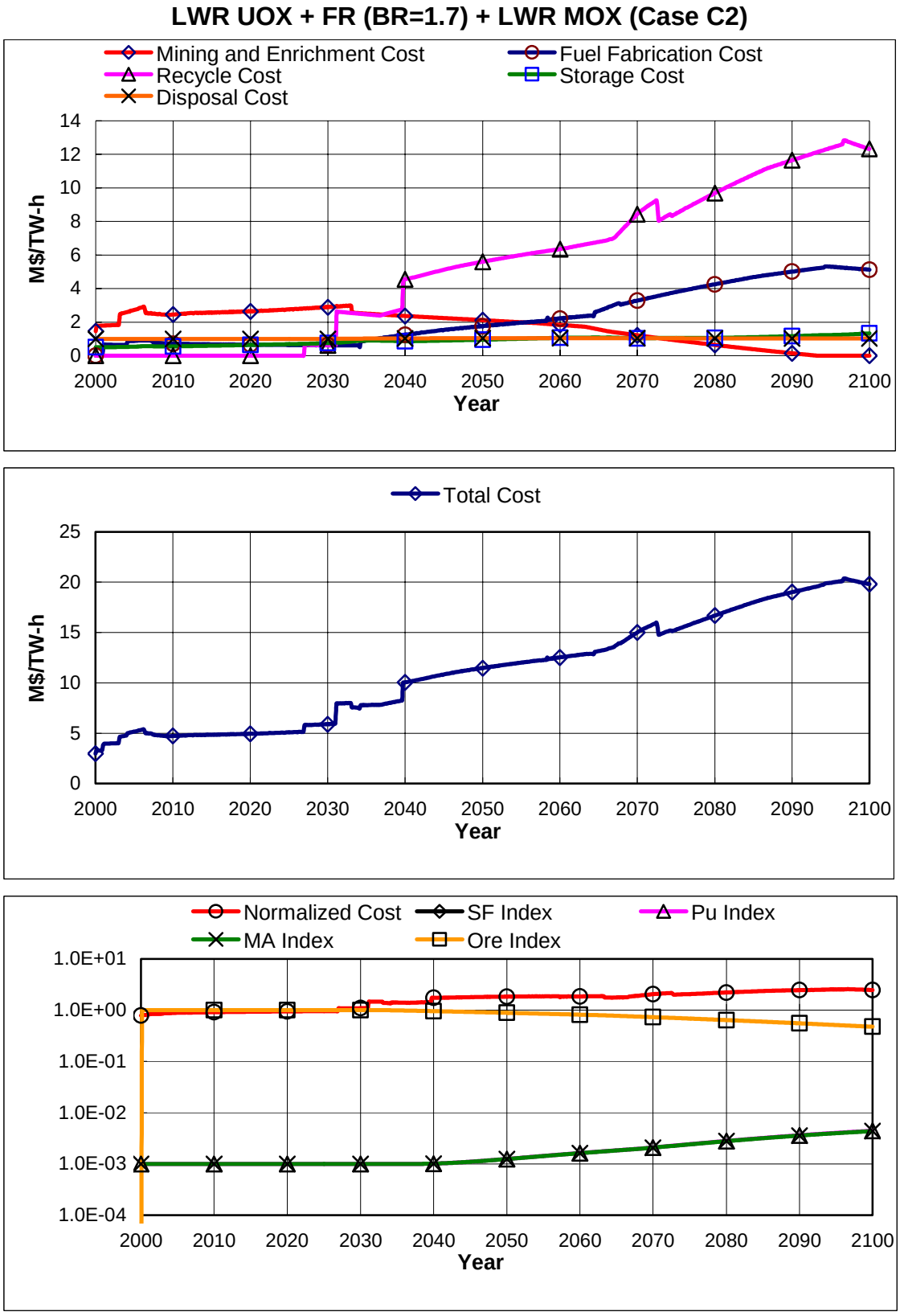

LWR UOX + FR (BR=1.7) + LWR MOX (Case C2)
Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
Year
Total Cost
M$/TW-h
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Fig. A3-32