GENERATION 4 Roadmap

Report of the Fuel Cycle Crosscut Group

November 1, 2001

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Generation 4 Roadmap
Report of The Fuel Cycle Crosscut Group (FCCG)

November 1, 2001

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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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Executive Summary

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FCCG
Executive Summary

Background

The Charter of the Gen-4 Roadmap Fuel Cycle Crosscut Group (FCCG) is 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 included examination of "fuel resource inputs and waste outputs for the range of potential Gen-4 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 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-4 Roadmap organization. Five one-day working meetings were held between February 2001, and August 2001 – three of them in conjunction with Gen-4 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 Gen-4 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 status of institutional aspects providing the enabling legal basis and boundary conditions for worldwide fuel cycle deployment was contributed by the OECD-NEA.

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 Gen-4 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-thru, partial recycle, full plutonium recycle, and full transuranic recycle) and were constrained only by physically-achievable mass flows and lag times of potential Gen-4 power plant and fuel cycle

concepts. They modeled idealized transitions from current and near term deployments to Gen-4 fuel cycles and power plants and potential symbiosis of mass flow exchanges among Gen-4 power plant concept types. These scenarios provide cornerstone indicators for the Roadmap of what are physically-achievable performances against Gen-4 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 which 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 Gen-4 in the areas of Sustainability, Safety and Reliability, and Affordability. The fuel cycle plays a primary role in meeting the three elements of the Gen-4 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.

Today uranium and thorium are not major contributors to the cost of nuclear energy, and resources do not constrain the expansion of nuclear power. Within several decades the costs of nuclear 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

Reversing Waste Buildup in a Growing Nuclear Economy

1. Closed fuel cycles have already demonstrated a significant reduction of the volumes 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 transmuting 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 waste destined for geological repositories by at least an order of magnitude by recovering 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 arising 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 .

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 affordability safety and reliability goals from the strategies for meeting sustainability Generation IV Goals. Wide ranges of fuel cycle approaches can be undertaken with only weak influence on economics. Power plant design strategies to meet economic, 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.

Transition To Sustainability

5. Energy use is expected to grow substantially in the next century; consequently nuclear energy deployment is projected by

authoritative studies to increase by 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.

   Modified Generation III reactors are needed to fill expanding energy needs and to employ limited repository space with high efficiency. Within of the symbiotic mix, Generation III 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 some residual contamination of recycle fuel from fission products are under current development. By focusing on creating clean waste streams 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 poisoned 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 tradeoffs and develop technologies for remote fabrication of thermal reactor fuel as well as fast reactor fuel.

Fuel Cycle Safety

8. The radiological exposure to 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 using already-mined resources, and by cost-effective remote refabrication will become increasingly important as the scope of nuclear power increases.

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

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 both 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 would 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 Gen-4 Systems

The scenario results indicate that the full fissile recycle and especially the full transuranic recycle generic fuel cycles may be needed to achieve Gen-4 sustainability goals SU-1 and SU-2. Once-thru 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 Top-Ranking Gen-4 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 Gen-4 economic, reliability and safety goals for the global nuclear energy park, it is the "breakthrough" fuel cycle enabling technologies which are needed to handle the Gen-4 waste management and resource 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-thru, 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.

   New fuel types of compositions containing increased plutonium fraction, new compounds (such as nitrides, spinels, apatites, etc.) and the potential presence of significant minor actinide fractions – fabricated using remote technologies – is recognized to be a challenging and significant undertaking. 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 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 Gen-4 goals. Uranium oxide fuel is highly refined and dominates the current once-thru commercial fuel cycle and is qualified for use in fast neutron spectrum reactors. Nonetheless new Gen-4 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 Gen-4 research program.

2. Cost effective advanced recycle technologies integrated with remote fuel refabrication technologies are the key enabling technologies for achieving the Gen-4 sustainability goals. In the full recycle fuel cycles, the waste destined for the repository arises from losses in recycle/refabrication and from nowhere else; the trace losses during recycle must therefore be small. Recycle technologies which 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 for fissioning are favored; -- they extract the maximum energy from the ore resource, and they

consign only the fission products to the waste. Such recycle technologies are under active development worldwide – based on both aqueous and on dry processes. 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 Gen-4 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 which minimize reject rates and rework are needed. Colocation of recycle and refab facilities are favored to minimize shipping and handling of radioactive materials in bulk form.

   "Dirty fuel" is easily suitable for use 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 fed back to thermal spectrum reactors as well as to fast reactors. Therefore, the recycle, the fuel design, and reactor core design must all be coordinated such that thermal reactor neutron economy is not 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 Gen-4 R&D.

4. While *the importance of near-term siting, licensing, and deploying a geologic repository designed for the once-thru 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 which will 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 yield a significant benefit against cost 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 approaches: interim storage of spent fuel prior to geologic emplacement; partitioning/conditioning with interim storage of tailored waste forms prior to geologic emplacement; or partitioning and interim storage of Cs/Sr with

reactor recycle for consumption by fission of actinides are the approaches available for preparation of the waste itself. Design of repositories tailored for heat management and tightly integrated with the above approaches on the waste preparation should be a priority development goal for Gen-4.

5. Recycle and return to the reactor power plants of commixed transuranic fuel provides for an intrinsic barrier to weapons proliferation from the commercial fuel cycle; the presence of minor actinides and carryover fission products makes the materials less attractive for weapons use, less accessible to diversion, and it avoids consigning inventories of fissile materials to interim storage and to a geologic repository for time intervals long enough for their intrinsic protective radiation fields to decay away. On the other hand, extrinsic measures cannot be elimated, and R&D will be required to adapt existing regimes of extrinsic barriers – such as Materials Control and Accountancy, sampling, and physical inventories – which were originally developed for the once-thru and partial recycle fuel cycles – to applicability to the full recycle fuel cycles. This R&D must go hand-in-hand with the development of the recycle/refab technologies and the design of the facilities to execute these technologies.

Gen-4 Fuel Cycle Crosscut Group

Chapter 1

Introduction: Conceptual Framework & Issues

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FCCG
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1.1 Conceptual Framework of the Fuel Cycle

Many of the central issues associated with nuclear power are primarily associated with the choice of fuel cycle, not necessarily the choice of reactor. Resource limitations, non-proliferation, and waste management are primarily fuel cycle issues. For the Generation IV road-map effort, the cross cutting fuel cycle issues must be carefully evaluated before a path forward is defined. This report addresses fuel cycle issues; both in the context of specific reactors and in the context of general fuel cycle issues that are common to most reactors.

The Fuel Cycle Crosscut Group's (FCCG) Charter includes a systems analysis function in support of the Gen-4 Roadmap activity. Gen-4 technologies are targeted for deployment by 2030; they will likely include assets with 60-year useful lifetimes; the Gen-4 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 Gen-4 asset lifetimes of 60 years from time of introduction in 2030.

The overall fuel cycle involves resource mass flows, energy conversion, and the resultant waste mass flows. 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 Gen-4 technological deployments to meet needs within achievable mass flow constraints.

The evaluation over a one-century planning horizon of alternative Gen-4 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. Finally, because the evolution of world energy supply infrastructure will involve transitions from the current situation to the future Gen-4 deployment, and the Gen 4 market penetration will be gradual, dynamic tracking is required in the systems analysis.

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 Conceptual Framework of the Fuel Cycle

A conceptual framework is required to allow comparisons between different reactors and fuel cycles. 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. The Generation IV goals of sustainability, safety and reliability, and economics requires consideration of the entire fuel cycle. A particular fuel cycle may excel in one area but be unacceptable because of deficiencies in other areas.

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, SNF and HLW, other radioactive wastes, releases to the environment (air and water), and non-nuclear wastes.

When examined in greater detail there are multiple technical facilities within 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 breeder fuel cycles. Alternatively, it may become a waste.

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

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

- *Spent Nuclear Fuel Storage*. The SNF is stored for its radioactivity 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 breeder reactor fuel cycles, sufficient fuel is recovered that fresh fissile fuel from uranium ore is not required.

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. *It is a dynamic system where preferred reactors and fuel cycles may change with time*. Some of these effects can be understood by examining the sources of fissile material that can be used as fuel in a reactor as a function of time (Table 1).

Table 1.1 Impacts of different parameters on the sources nuclear fuels 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	[\	[\	[

If natural uranium is the source of fissile fuel, the uranium concentration in the ore will decrease over time as the richest ores are mined. In parallel, it would be expected that (1) uranium fuel costs would increase and (2) the environmental impacts would increase because lower grades of ore imply more mining and a once through fuel cycle implies large quantities of SNF requiring disposal. However, technological improvements over time will tend to reduce costs and environmental impacts. These technological improvements can be (1) reactors that produce more energy per unit of uranium or (2) better mining and milling methods for uranium recovery. The fissile resources are effectively unlimited given the quantities of uranium in the earth's crust and seawater. It is the economic and environmental constraints that ultimately limit practically recoverable reserves.

Current enrichment processes produce enriched uranium and depleted uranium. Existing enrichment processes leave about half of the fissile ^{235}U in the depleted uranium. This is a source of fissile material where (1) technological advances that lower enrichment costs and (2) the cost of uranium from the mine determine when the DU should be processed for added ^{235}U recovery. The resource is limited by existing inventories of depleted uranium

SNF is a fissile resource. Depending upon the type of reactor, the fissile content of the SNF may be less or greater than the needed fissile feed to the reactor. If more fissile material is produced from fertile materials (uranium-238 and thorium-232) than consumed, the quantities of uranium or thorium required to maintain the fuel cycle are very small. Less than 1% of natural uranium is fissile fuel. Over 99% of the natural uranium is fertile uranium-238. 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 or eliminates the need for mined uranium and can reduce the costs and difficulties of disposing of radioactive wastes. However, the technological and institutional requirements to use fissile fuels from SNF are significantly greater than for once through fuel cycles that use natural uranium as a source of fuel. It would be ultimately expected that the cost of fissile materials from this source would be less than from 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) has historically been considered a limiting factor for nuclear power that uses a once-through fuel cycle, it is important to recognize that waste management may also place limits on acceptable fuel cycles. In the current world environment, there is no shortage of uranium to fuel reactors but there are major difficulties in the disposal of wastes. Some of the difficulties associated with technical issues. Some of the difficulties are associated with institutional issues. Many of the difficulties are a combination of technical and institutional difficulties.

Four generic fuel cycles have been defined which 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 directly disposed of as a waste. LWRs in the United States currently use this fuel cycle.

- *Partial recycle*. Some fraction of the SNF is processed, the fissile material is recycled, and new fuel is fabricated. An example is the French recycling system, in which (1) low-enriched-uranium SNF is recycled and recycled back to the reactors and (2) the resultant MOX SNF may be directly disposed of. A second example is the proposed DUPIC fuel cycle of South Korea, which recycles LWR SNF into fresh CANDU fuel with direct disposal of the CANDU SNF. Depending upon the fuel cycle, HLW may or may not be produced.

- *Full recycle*. All SNF is recycled for recovery and recycle of plutonium and ^{233}U. An example of this is the traditional Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle. Minor actinides are sent to the waste stream.

- *All-actinide recycle*. All SNF is recycled, and all actinides are recycled. 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, LMFBRs, and the molten salt reactors. The LWRs produce power. The LMFBRs produce power and fissile fuel. The molten salt reactor is used to destroy higher actinides that would otherwise be sent to the repository.

Society can convert from one type of fuel cycle to another. For example, SNF that is disposed of remains a source of future fissile and fertile material. All repository programs worldwide are planning to use long-lived waste packages as part of their repository disposal systems. Most repository programs require that the waste be retrievable. The SNF does not disappear but remains to be recovered if desired.

For evaluating alternative nuclear futures, there are common issues (briefly discussed herein) that, in part, determine the feasibility of various fuel cycles:

- Availability of resources versus cost.
- Role of separations
- Waste management
- Future global energy demand
- Dynamic response capability of overall system

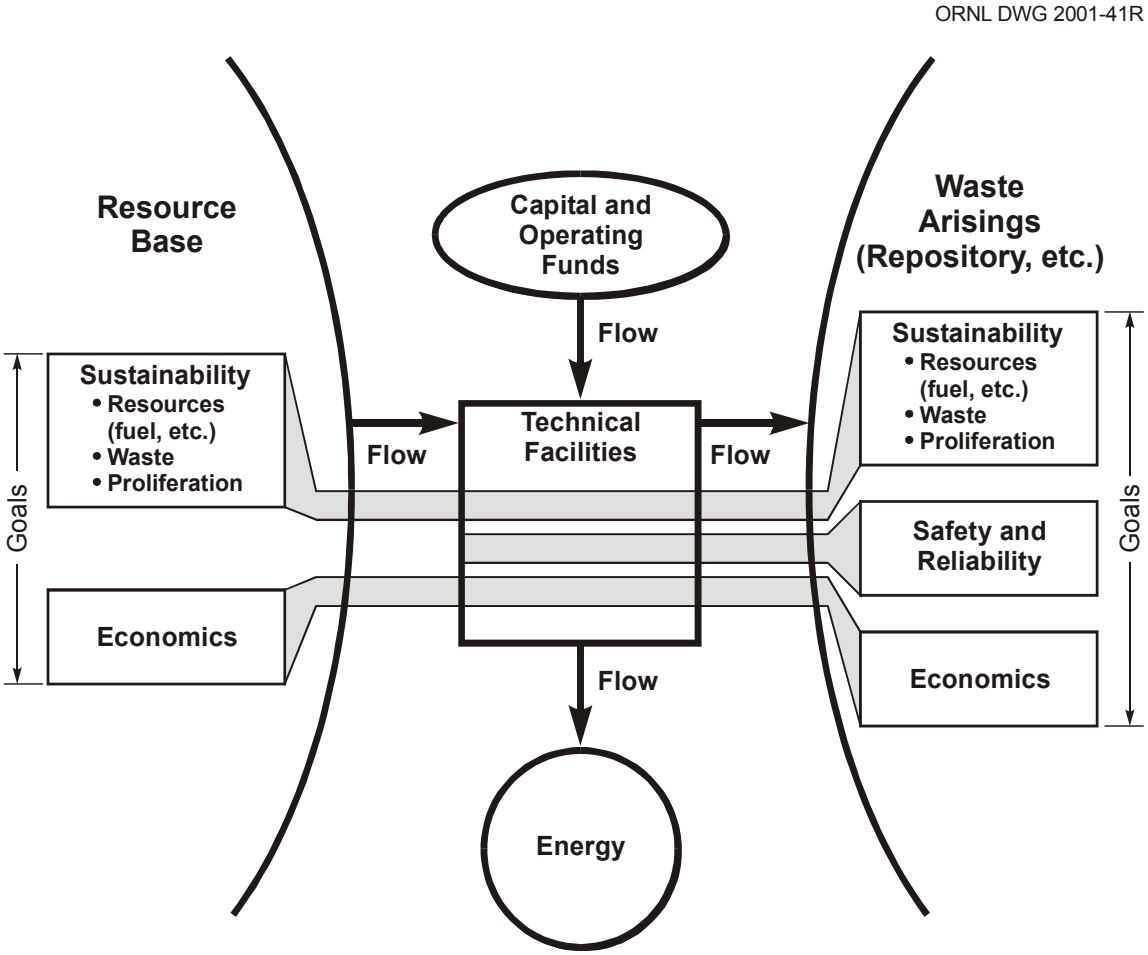

Fig. 1.1 The Fuel Cycle in the Abstract

ORNL DWG 2001-182

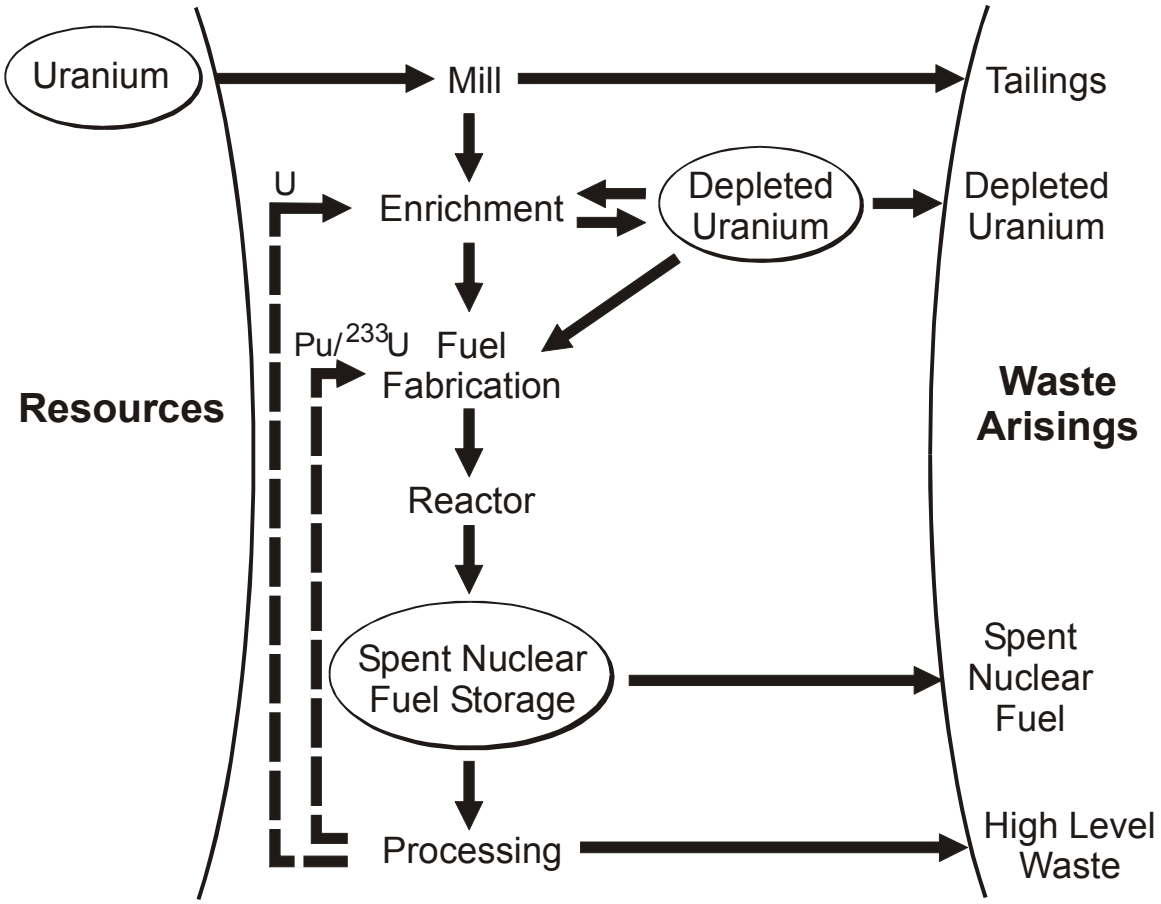

Figure 1.2 Steps in the Fuel Cycle

ORNL DWG 2001-40

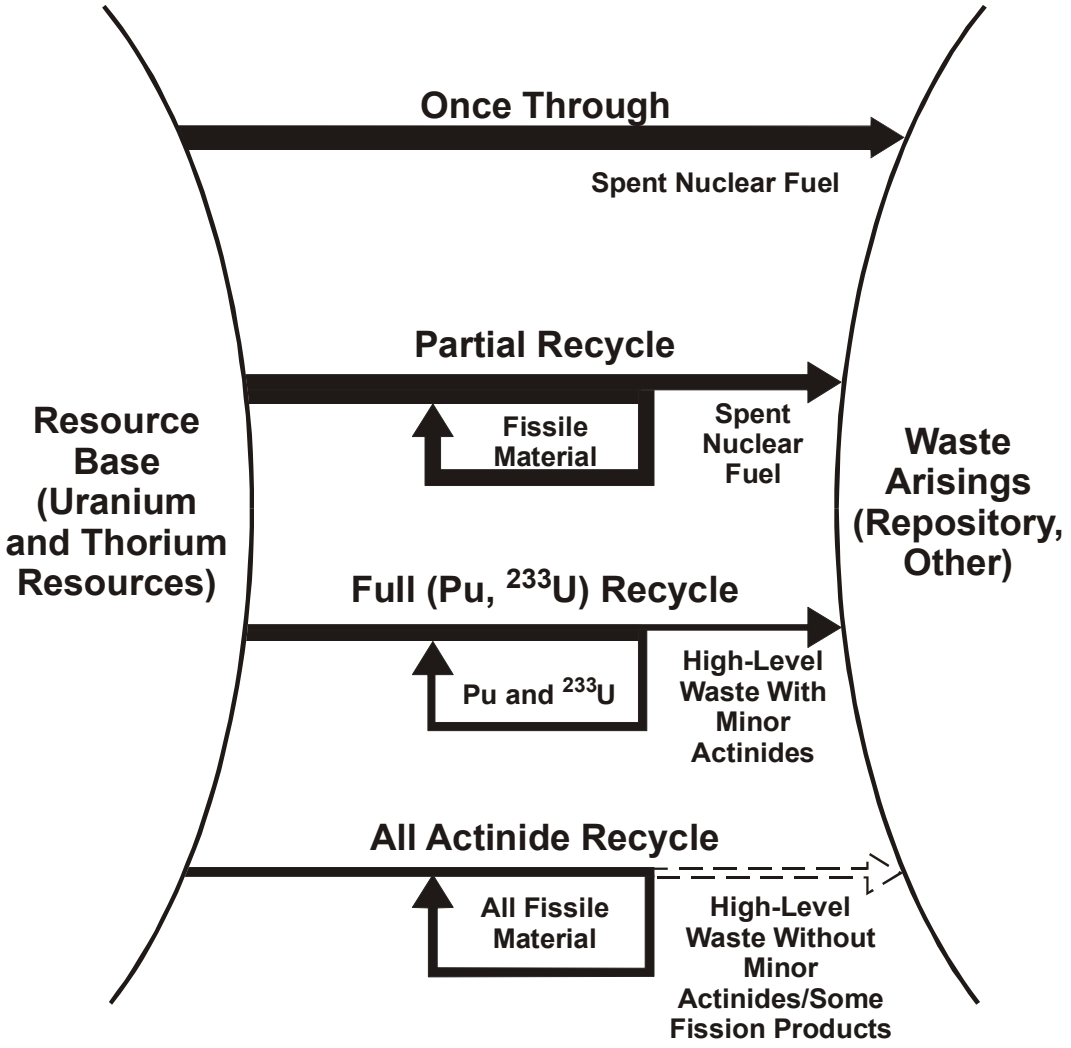

Figure 1.3 Alternative Fuel Cycles

1.2 History of 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 blossoming 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 commercially developed 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 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, of uranium fuel costs and increases in fuel burnup and average load factor in LWRs. The thermal reactor once thru 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 (see Figure 1.4); the once-through UOX cycle and the MOX mono recycle (see Figure 1.5).

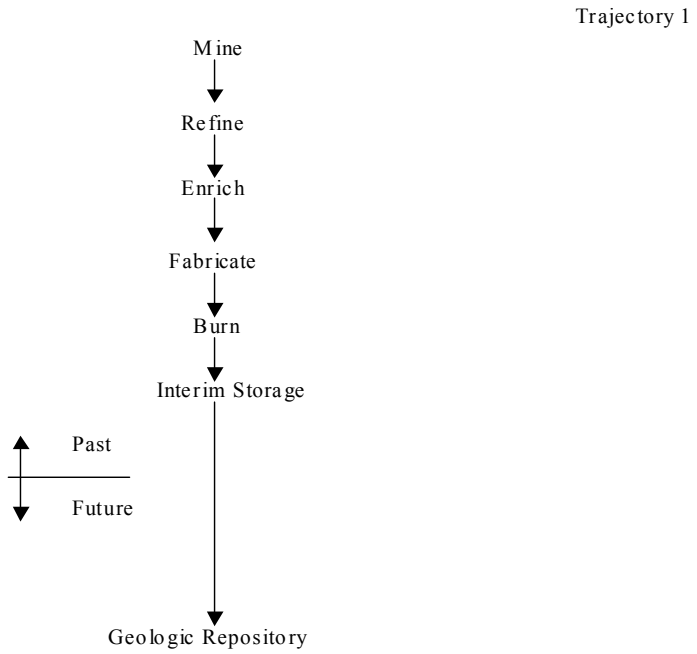

Figure 1.4 The Once-Thru UOX Cycle

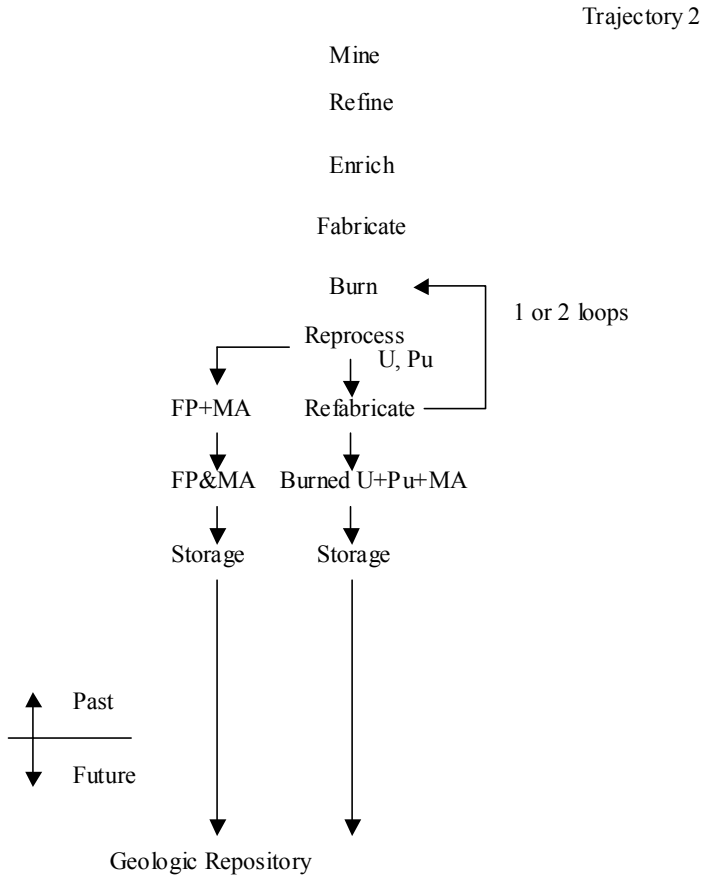

Figure 1.5 The LWR MOX Recycle Cycle

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 recycling 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 recycling and recycling of separated plutonium. At that time the U.S. also initiated an extensive research program aimed at developing more proliferation-resistant recycling 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) recycling plants. Furthermore, the plentiful supply of low-cost uranium coming into the market at that time eroded the economic incentive for commercial fuel recycling in the U.S.[2]. In 1981 the nuclear waste policy act was passed, creating a once-thru waste management approach based on a geologic repository.

Plutonium has been successfully recycled in the form of MOX fuel in thermal reactors for more than thirty years. Today, about 35 thermal reactors use MOX-fuel complying with 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 has been started in the mid 1980s when some modification or withdrawal of fast reactor programmes was enforced, and the developed technologies of recycling and fuel fabrication were applied for Pu-recycling in LWR-fuel sector, in the mean time focusing stabilisation of the separated plutonium inventory.

Currently, the use of MOX fuel has been established on an industrial scale in a number of reactors. In Belgium, France, Germany, Japan and Switzerland a considerable number of the thermal power reactors (PWRs and BWRs) are either licensed (i.e. 40, 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). MOX fuel achieved good irradiation performance, with average burnup between 36 and 44 GW·d/tHM.

TABLE 1.2 STATUS OF LARGE SCALE MOX FUEL UTILIZATION IN THERMAL REACTORS[a]

Status yearend 1998

	Number of Thermal Reactors			
	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 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 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 emerging as a policy issue again due to recent volatilities in oil and natural gas prices and supply.

Energy providers note that fuel costs for nuclear reactors are a much smaller fraction of bus bar electricity costs compared with fossil energy generation, and any future increases in uranium costs would be highly damped, which is an important factor in the future sustainability of a 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 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 Gen 4 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. In 1999 [4], known uranium resources (RAR) recoverable at less than $130/kg U were estimated at 4 million tones. Total known and undiscovered conventional resources (RAR and EAR) 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 thru UOX cycle which are built in the next several decades would ever be shut down due to fuel 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 thru or MOX mono recycle fuel cycles. Possible Gen-4 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. Recycling 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. Since uranium mining is a major contributor to the release of radioactive materials to the environment, efforts to minimize these impacts through improved mining technologies will be needed.

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 simultaneously contribute to waste minimization because burning of transuranic elements and minor actinides in fast reactors (FR), or fast sub-critical reactors with spallation neutron source or accelerator driven systems (ADS) can substantially reduce the long term radiotoxicity source term of the wastes while at the same time producing energy. Likewise, long-lived fission products can be transmuted by means of ADS and leakage neutrons in thermal and fast reactor blankets. 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 Generation IV.

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.
2. *Nuclear Energy, Principles, Practices, and Prospects,* David Bodansky (AIP Press: Woodbury, NY, 1996) p. 114.
3. *Ibid.* p. 116
4. *Nuclear Fuel Cycle and Reactor Strategies: Adjusting to Realities,* International Symposium held in Vienna, Austria, 3-6 June, 1997 (IAEA, Vienna, 1997) pp.81-83.
5. *Ibid.* p. 154.
6. National Academy of Sciences, *Management and Disposition of Excess Weapons Plutonium*. (National Academy Press, Washington, DC, 1994).
7. G. Linsley, A. Fattah, "The Interface Between Nuclear Safeguards and Radioactive Waste Disposal: Emerging issues" *IAEA Bulletin,* **36** (1994)

1.3 Availability of Ore – Fueling Gen-4

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 can be 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 constituents. Because of its large ionic size and heating due to radioactive decay, uranium is transferred 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 initial 63 Tt (63E12 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 for the earth's mantle and core have be supported by 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).

Geological Processes Determining Uranium Distribution

Uranium has a very complex range of geochemical behavior and a wide variety of economically recovery deposits. Because it has two valence states in natural, with deficits of four and six electrons, and a variety of complex ions, uranium reacts geochemically in a great 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.

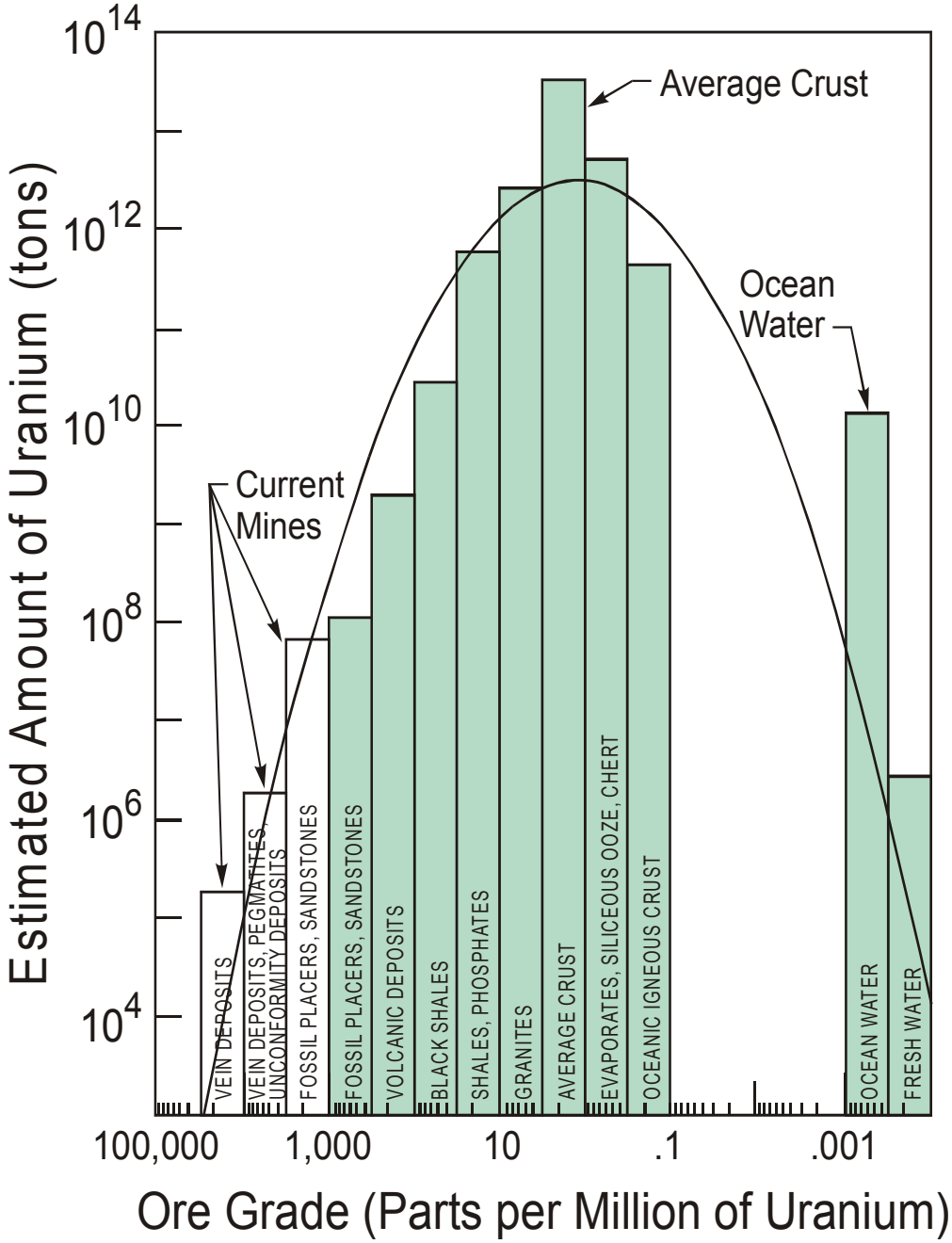

Figure 1.6 Distribution of Uranium in Earth's Crust (from Deffeyes 1980)

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 would be desirable, both environmentally and economically.

1.3.2 Known Uranium Resources

Future uranium resources can be delineated into conventional resources, including mining of uranium ores by mainly conventional mining technologies and as a principal by-product of the mining and extraction of other commodities, such as phosphates (estimated at 22 million tones U if fully exploited) 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 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 include 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 in the "Red Book" (NEA 1999) of uranium resources. Their current evaluation is shown in the following 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[1]
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[2]	43 800	41 700	37 600[3]	33 000[4]
Planned and prospective				19 400		
Total		*46 000*	*45 750*	*61 200*	*64 800*	*55 000*

1. 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%).
2. With a projected plant capacity utilisation of about 75%, Existing and Committed capability is about 74% of 1999 world uranium requirements.
3. 50-60% of projected world uranium requirements in 2010.
4. Based on the reported data, between 40% and 60% of the expected uranium requirements in 2015 could be satisfied with resources recoverable at $40/kgU 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.

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 in 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 20% of natural gas or oil resources, even with the use of LWRs only. (See Table 1.5.)

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

Uranium was early on seen as a scarce resource. Since uranium-235 (^{235}U) is the only naturally occurring isotope that can readily sustain a fission chain reaction, the resources of uranium (containing only 0.7% of U235) have been believed to inherently limit the sustainability of nuclear energy based on fission.

Concern that uranium would run out soon was one of the driving forces in the development of fast breeder reactors which consume the abundant isotope U238 (>99% of the ore) and of reactors capable of breeding thorium into the fissile, but not naturally occurring, isotope uranium-233 (^{233}U). (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. A second reason has been the much slower growth of commercial nuclear power than was originally anticipated. More recently a third disincentive to uranium exploration has emerged – this being disarmament. As much as one third of the natural uranium 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 to ongoing programs for the down-blending of weapons-derivative HEU. In the down-blending programs, HEU is mixed with natural or depleted uranium to produce low enriched uranium (LEU) fuel for commercial light water reactors (LWRs). The civilian use of down-blended that 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 been further depressed and incentives for further exploration have been very small.

Nevertheless, despite low prices and disincentive to exploration, additional resources of uranium have been discovered, particularly in northern Canada and in Australia. For that reason,

the estimates of resources while official must be used with some caution. Often the "reasonably assured resources" significantly under-estimate resources that will be found with further exploration. For example, although the present "to be mined" U.S. resources are listed as about 110,000 tonnes of uranium, 367,000 tonnes of uranium were produced from domestic mines from 1949 through 1999. U.S. uranium mines have been closed, not for lack of resources or of resources yet to be discovered, but because current prices of $23.05/kg U (Dec. 2000) no longer support operation.

The Reasonably Assured Resources, cited in Table 1.3, would indicate that, at the current consumption rate 64,000 tonnes of natural uranium per year, our resources would last only about 35 years. However, such a conclusion is incorrect for several reasons. The Red Book (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.

The Canadian reserves are a good example. 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.

Uranium, with a crustal abundance of 2.7 wppm, is twice as abundant as molybdenum or tungsten, and 40 times more abundant than silver. The richness of the Canadian deposits and their relatively small area that has been explored both 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 uranium in those tailings.

Uranium content of Enrichment Tails

The inventories of depleted uranium stored at enrichment plants is another 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 therefore 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. 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 currents to move the seawater past uranium-collecting surfaces.)

Uranium-recovery from seawater has been studied in Japan for the very long term or to face a very strong development of fission energy. On a laboratory scale, experiments performed by JAERI in Japan 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.

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.

1.3.2 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. 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 239Pu (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 $ 82.50per 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.

Because ^{232}Th is the only isotope of natural thorium, there are no enrichment plant tails from thorium nuclear fuel. Therefore the cost of thorium in a mixed thorium-uranium LWR fuel or in a pure thorium-^{233}U fuel cycle is relatively minor.

1.3.3 Sustainability Potential

The "known conventional resources" of uranium ore which are recoverable at less than 130 $/kg U is reported in the Red Book to be slightly under 4 million tonne of U. 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).

The energy content of the U235 in this 11.5 million tonnes of U, when used in current LWR once thru cycles, is equivalent to about 20% of known fossil (coal, oil, natural gas) resources (See Table 1.5). If used in current LWR's once thru cycles to meet projected nuclear demand (see Chapter 3) it will last for multiple decades but not for a century.

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-thru 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 Appendix 1 which quantifies the impact of uranium ore price on cost of LWR produced electricity.

Because exploration activities have been dormant for several decades, it is widely believed that additional ore deposits will be found when exploration resumes. Moreover, the production cost of energy is weakly dependant on cost of the raw ore. No reactor built in the next couple of decades and operating on the once-thru cycle would ever be shut down due to uranium scarcity, regardless of its conversion ratio.

A requirement of Generation 4 to produce sustainable technologies that are deployable within thirty years; with asset lifetimes of 40 to 60 years for specific deployed plants. The Gen-4 planning horizon is therefore a full century into the future. Given a significant nuclear expansion forecast by numerous international planning bodies, for the coming century, economic forces will emerge to recycle spent fuel, both from rising uranium prices and from efforts to use repository space efficiently. Recycle technologies will be a part of Gen IV research as a means to extract increasing fractions of the energy content of the already recovered ore. If exploited fully, at least a millennium of energy supply can be foreseen from the earth's endowment or uranium ore. Likewise, the energy potential of the earth's endowment of thorium is greater still. *Thus the resource base is sufficient to support the Gen 4 sustainability criteria, -- given that the fuel cycles and reactor types are deployed to exploit those resources.*

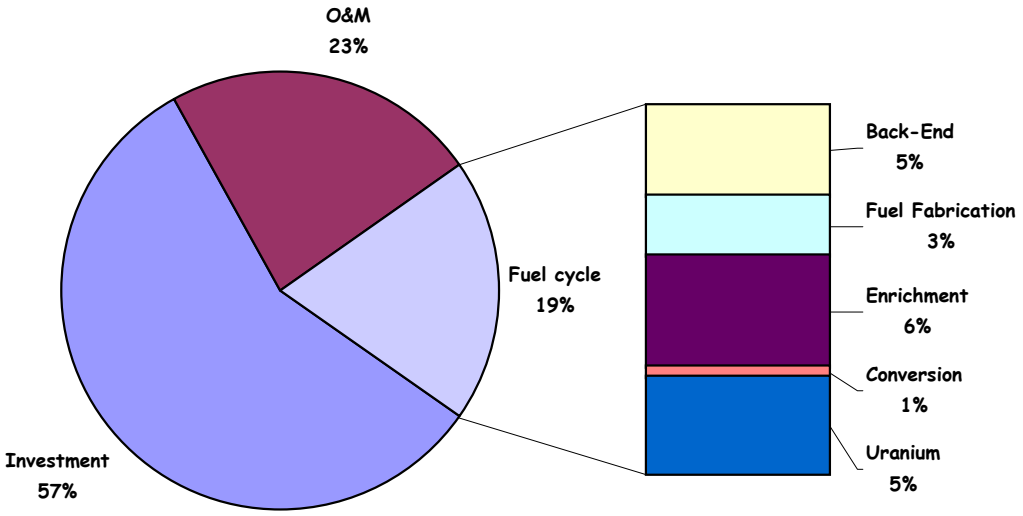

Figure 1.7 The Cost of Electricity from Nuclear Power

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1.4 Future Demand for Energy Services – Meeting Gen-4 Client Needs

1.4.1 World Current 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 or 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 (2) 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 (which comes dominantly from burning fossil fuels) is daunting – about 10 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 10 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 $0.2*10*10^9$ 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 Gen 4 seeks to exploit for meeting future world energy service needs.

1.4.2 Energy Demand Growth – The Gen-4 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 Gen 4 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

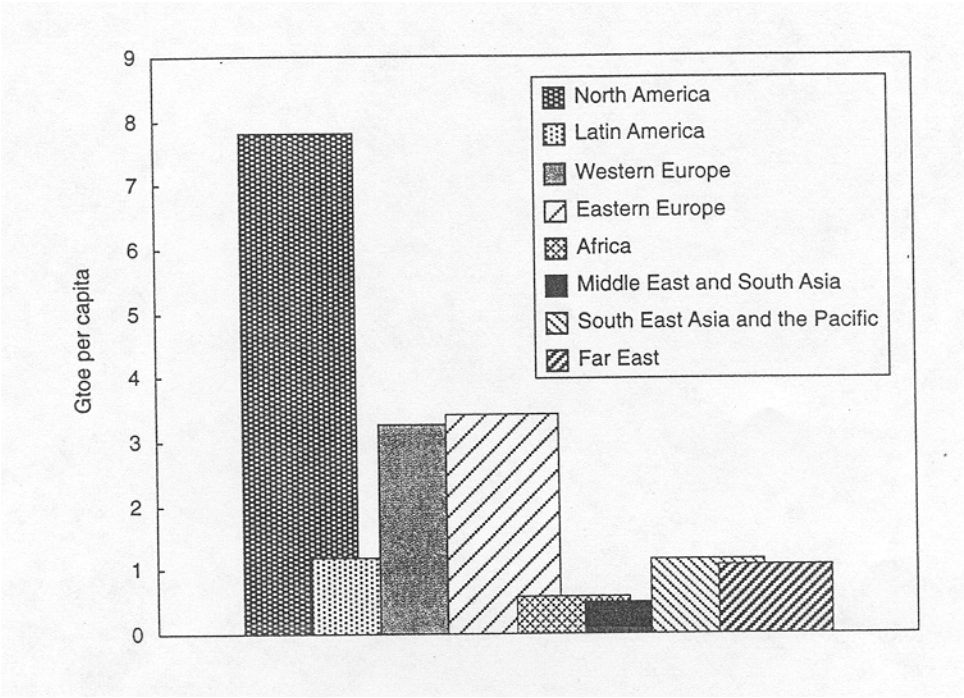


Fig. 1.8 Regional energy consumption (1995). [INTERNATIONAL ATOMIC ENERGY AGENCY, Energy, Electricity and Nuclear Power Estimates for the Period up to 2015, July 1996 Edition, IAEA, Vienaa (1996)]

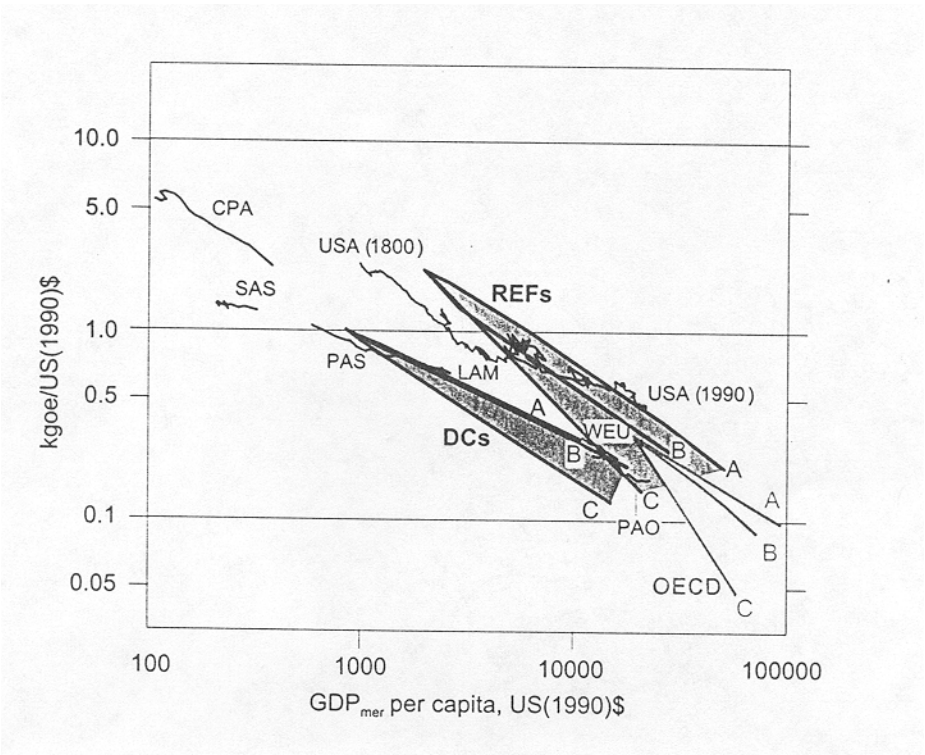


Fig. 1.9 Energy intensities, in kgoe per US (1990)$, as a function of degree of economic development, in GDP_{mer} per caita in US(1990)$. Historical data (black) and Cases A, B, and C (color).

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

The IIASA/WEC study (2) of the World's future energy needs, which was done region-wise 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 (see Section 1.5.4) – a way must be found during the next fifty years to increase energy supply by a factor about three while at the same time avoiding the concomitant buildup of waste production by the comparable factor.

The WEC/IIASA study shows that energy demand growth will be highest in developing economies, and that the deployment of nuclear power will diffuse from nearly exclusive application in the industrial countries to support energy needs in the world as a whole. *The challenge for the Gen-4 initiative is to create and deploy nuclear energy supply options capable to contribute to a worldwide applicability.*

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 discussed 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 electricity alone, but even a more diverse fraction of primary energy needs.

Numerous energy service needs beyond electricity alone can potentially be provided by Gen 4 concepts.

Desalinization

The first is potable water 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 Gen 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 Gen-4's contribution and a challenge to achieve safety levels compatible with siting needs for industrial applicability.

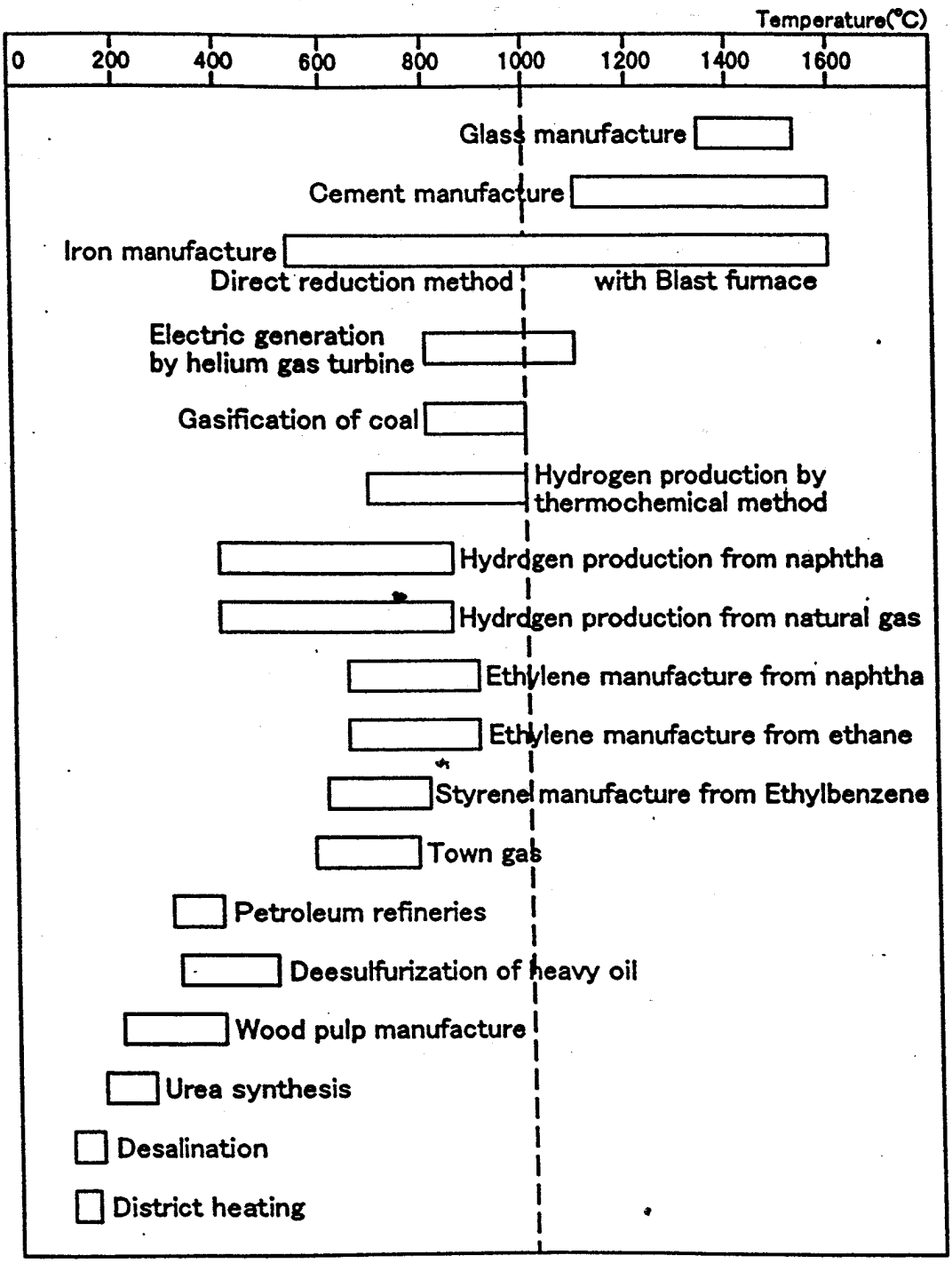


Figure 1.10 Temperature Region of Heat Used in Various Kinds of Industries

Hydrogen Production

One third of primary energy use applies to the transportation sector. 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 Gen 4 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 exist already 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. Gen 4 technology development should address this opportunity for expansion of nuclear's role into a diversified and broader share of the primary energy service needs of society.

References

1. IAEA, "Sustainable Development; Nuclear Power" Secretariate of the IAEA, (Nov. 1997).
2. IIASA/WEC, "Global Energy Perspective" Nakićenovivić, Grüber, and McDonald, eds. Cambridge, U. Press (1998).

1.5 Framework for Gen-4 Fuel Cycle Systems Studies

The Fuel Cycle Crosscut Group's (FCCG) Charter provides for it to perform a Systems Analysis function in support of the Gen-4 Roadmap activity. Gen-4 technologies are targeted for deployment by 2030; they will likely include assets with 60 year useful lifetimes; the Gen-4 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 Gen-4 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 which executes this transformation involves resource mass inflows, energy conversion, and resultant waste mass outflows. The FCCG is charged to evaluate (1) the long term ore resource potential and (2) the long term (one century) world energy supply demand projections and, given these, to consider the potential of various Gen-4 fuel cycle deployments to meet energy needs within achievable mass flow constraints.

The evaluation over a one century planning horizon of alternative Gen-4 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 to the future Gen-4 deployment, and the Gen 4 market penetration will be gradual and ultimately constrained by physically achievable time constants.

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.

1.5.1 Projected Resource vs Price

We start with ore resources. To perform 100 year scenarios, a cost of uranium is needed versus cumulative ore withdrawals. The FCCG has generated a price elasticity model based on two things: near term projections and a long term upper bound.

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. To induce additional production 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 in, we can determine of 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, the 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 estimated (lower bound) seawater extraction price. For this purpose, at the seawater extraction price of $200 per kg U, the oceanic resource of 4.2 billion tonnes of uranium are assumed to become available.

Table 1.9 Resource - supply relationship

Price		Resource	
$/lb U3O8	$/kg U	tonnes U	
$15.00	$37.44	385E+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

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 system 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 taken from the 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 projection. This study is widely cited and highly regarded.

The regional forecasting process relied on a country by country population growth projection produced 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 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 GDP) as the GDP increases (see Fig. 1.9).

Six patterns of how energy demand will be met were employed to illustrate the range of potential energy futures. Given a pattern, it 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"

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.

In all six scenarios, nuclear energy's role in meeting the demand for energy services was confined to the electricity sector (which currently comprises about 1/3 of total energy needs and which was modeled as growing to 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, ship propulsion, 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 the projected nuclear share from the IIASA/WEC projections might turn out to be even larger – should a broadened role for nuclear be realized.

For the purpose of the Fuel Cycle Crosscut Group's scenario evaluations, Case B has been taken as the reference case – neither as technologically optimistic as the Case A scenarios nor as institutionally optimistic as the Case C scenarios, and with a ragged dispersion of development rates in various developing countries. *With nuclear confined to electricity production, predicted global demand growth for nuclear is shown in Table 1.12 for the Case B scenarios. 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.*

Table 1.12 displays the projected nuclear demand growth not only for the world aggregate but also 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. Absolute capacity addition slightly exceeds a Gtoe of

cumulative additions made over the period from 1990 to 2050 in each the developing and the industrialized world regions. During the time of Gen 4 introduction 2030-2050 the energy supply investments in developing countries will significantly exceed those in industrialized countries.

In light of the difficult challenge presented by accurate economic forecasting, it is important to assure that the nuclear energy demand projections used for Gen-4 evaluations are in the realm of consensus thinking of energy planners representing a range of viewpoints. The IIASA/WEC process has sought to accomplish this through their process of review by regional experts. Beyond that, however, projections of other agencies, groups, and individuals have been compared to the IIASA/WEC Case B selected for Gen-4 studies. Table 1.13 lists several recent projections relevant to this comparison.

Projections of 50 to 100 years in advance are always to be viewed skeptically – too often they prove inaccurate within even a decade. For this reason the Gen-4 Roadmap must be structured to produce technology which is robust with respect to uncertainties in future energy demand. 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.

1. Global Energy Perspectives, Nakicenovic, Grübler and McDonald, eds. Cambridge University Press, (1998).
2. E. Bos, M.T. Vu, and R.A. Bulaloo, World Population Projections 1992-1993, Johns Hopkins University Press, (1992)
3. Bauquis, P.R. "Medium and Long Term World-Wide Energy Supplies (2020-2050): Constraints on Fossil Fuels and Potential Role of Renewables and Nuclear", Proceedings of the UNESCO International School for Science and Peace Forum: Can Advanced Energy Systems Cope with Energy Demands of Asia, OECD-Pacific and CIS? Como, Italy, July 5-8, 2000.
4. Sailor, et. al., "A Nuclear Solution To Climate Change?" Science, Vol. 288, pp. 1177-1178 (May 19, 2000).

Table 1.10 Regional Energy Demand Projections for IIASA/WEC Study

OECD

	NAM	N. America
	WEU	W. Europe
PAO	Pacific OECD (Japan, Australia, New Zealand)	

REF

	EEU	Central & Eastern Europe
	FSU	Newly Independent States of Soviet Union
DC		
	LAM	Latin America & 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.

Table1.12. Nuclear Electricity Demand

Region		1990	2020			2050			2100		
			A3	*B*	*C2*	*A3*	*B*	*C2*	*A3*	*B*	*C2*
World	(Gtoe)	0.45	1.03	0.90	0.85	2.82	2.74	1.77		8.3	
	GW_e		621	543	512		1652			5000	
OECD	(Gtoe)	0.36	0.72	0.66	0.62	1.58	1.45	0.98			
	GW_e			398			874				
North America	(Gtoe)	0.153	.268	.258	.258	.420	.737	.366			
	GW_e	92	162	156	156	253	444	221			
Developing	(Gtoe)	0.03	0.22	0.16	0.13	1.02	1.05	0.70			
	GW_e			96			633				
Reforming Economics	(Gtoe)	0.06	0.10	0.08	0.10	0.23	0.24	0.09			
	GW_e										

Table 1.13.

Comparison of Energy Demand Projections	
Study	World Nuclear Capacity Deployed
IIASA/WEC [1]	2.74 Gtoe (1652 GW_e) Case B in 2050
IPCC 2000	
OECD-IEA	
US DOE EIA	
US EPRI	
P.R. Bauquis Total/Fina/Elf [3]	4 Gtoe (2412 GW_e) in 2050
Sailor, et.al Center for International Security and Cooperation, Stanford University [4]	3300 GW_e in 2050

1.5.3 The Role of Separations

Given the ore availability 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 to energy and fuel cycle processes which transform ore to fuel and discharged fuel to waste and recycle constituents.

The nuclear fuel cycle is a series of pre and post processing technologies that support the operation of nuclear power plants designed to extract the energy released by fission and convert that energy into electricity. 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.

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. Although the processing of uranium has interesting similarities with these lighter reactive metals, there is a fundamental difference, i.e. the product is not uranium metal for structural purposes, but uranium for the energy it produces.

Processing first refines the uranium ore so that it will be well behaved in the 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 made 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 fabrication 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 unique to uranium 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 % U^{235} to facilitate criticality and sufficient in-reactor residence time. Presently commercial isotopic enrichment is accomplished through separation of UF_6 by gaseous diffusion and other techniques such as untracentrifuge. The uranium is converted to uranium tetra fluoride UF_4 as an intermediate compound then uranium hexa fluoride 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 is close to commercially viable.

Processes for isotopic separation and their efficiency will become important if the reactor requires high enrichment for efficient operation. If the enrichment level required is <6% then new technology is not a pressing need. If fuel cycles that require significant enrichment are chosen then improved 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 Alternative Fuel Cycles Diagram described in Section 1.1.The generic nuclear fuel cycles are

Once Through,
Partial Recycle,
Full (Pu, ^{233}U) Recycle and
All 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 mandated 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 (Pu, ^{233}U) Recycle** and **All 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 unburnt fuel, to first remove some or all of the fission products and then to refabricate and return some or all of the actinides formed by irradiating the fuel back into the power plant 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 encyclopedic.

Solvent Extraction

Solvent extraction is the process most commonly used to close the fuel cycle. The system relies on the exploitation of the differences in thermodynamic stability between complexes of the actinides. Using multiple extractions the concentration of a particular element can be reduced or concentrated by a factor of more than 10,000. Outside the United States, the PUREX recycling 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.

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, 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 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 invented as an integral part of Argonne's Integral Fast Reactor (IFR). Electrorefining was the core separation technology of that process and remains the core separation technology in the present EBR-II Spent Fuel Treatment Program. 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 an 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 wasteform. The noble metal fission products and the cladding that remain in the anode baskets are melted into an iron-zirconium alloy metal wasteform.

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 spent fuel. Fluorination the spent fuel produces volatile fluorides 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 still the subject of ongoing research. Fluoride volatility is extremely 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 being proposed and developed for processing of Hanford high-level tank waste oxides using a coarse ion separator such as that recently proposed for Hanford high-level waste. 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 recycle technologies PUREX solvent extraction is the only one operated on a commercial scale. However, Gen-4 advanced fuel cycles based on the other processes as well are proposed and they are motivated by Gen-4 goals for economics, sustainable waste management and proliferation resistance.

1.5.4 Waste Management

The system which transforms ore into energy unavoidably produces wastes as well. The objective of sustainable waste management in the nuclear fuel cycle is to dispose of wastes while protecting humans and the environment. Both chemical and radioactive waste must be considered. Some potential radioactive wastes, such as depleted uranium (DU), are also chemical wastes, in which the chemical toxicity exceeds the radiotoxicity. There are 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 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 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 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 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 reduced.
    - <u>Waste form</u>. The chemical form and geometry of the waste form can 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.
    - <u>Repository design</u>. 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.

- *All-actinide recycle.* Recycling of the SNF with recycle of all fissile materials (plutonium, ^{233}U, and minor actinides) provides additional repository options beyond those associated with full recycle. Safeguards and criticality issues are, for practical purposes, eliminated. Waste toxicity is reduced. 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 disruptive characteristics 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. All of these issues must be addressed when considering fuel cycles.

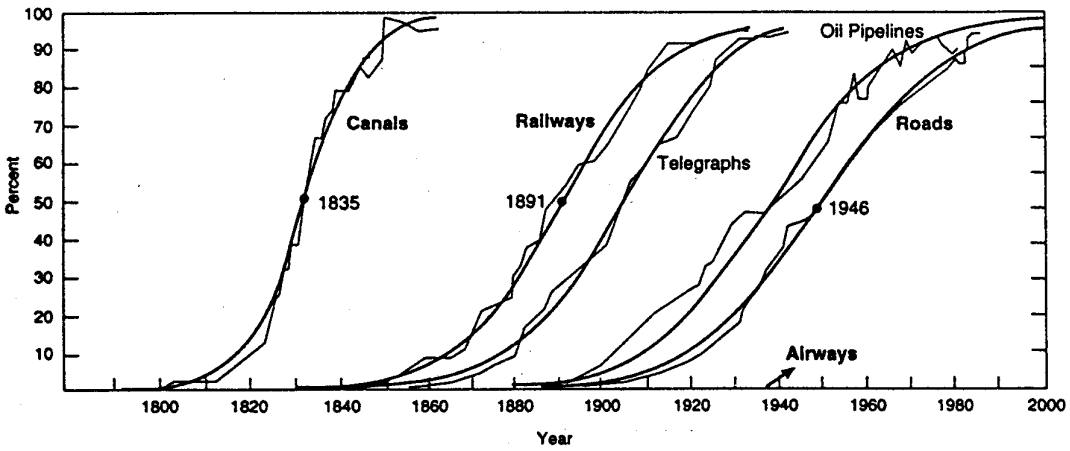
1.5.5 Dynamic Response Capability of The Nuclear Fuel Cycle

The nuclear energy enterprise is a dynamical system; it evolves from the current deployments and it incorporates time delays to build assets and long asset lifetimes which carry the memory of previous conditions. The 30 year lead time to Gen-4 deployment recognizes the multi-decade time constant which is characteristic of innovation, development, and market penetration trajectories for major infrastructure deployments. Fig. 1.10 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.[1]

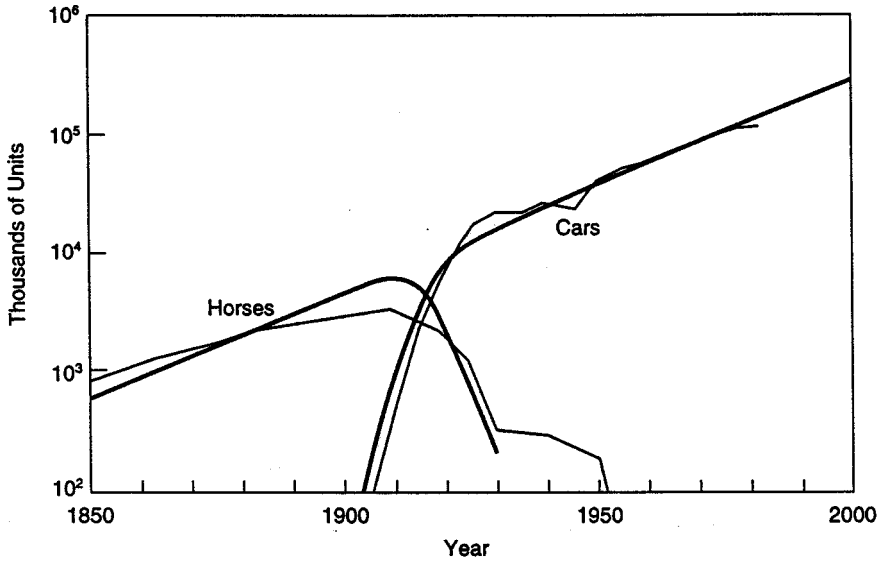
The FCCG systems studies will develop and display several potential nuclear energy futures based on alternative Gen-4 fuel cycles. These projections will be based on physical inertia aspects of Gen-4 deployment i.e., they will be energy demand-driven, and actinide resource – constrained; the scenarios 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 Gen-4 fuel cycles.*

For Gen-4 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,



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 1.10

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 capital assets, for reactors themselves, and for other deployment lag times are based on historical precedent and are displayed in Attachment 2.

As to assess 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 Gen-4 concepts being modeled. They depend on such intrinsic attributes as reactor specific inventories $\left(\frac{\text{kg fissile}}{\text{MW}_e} \text{and} \frac{\text{kg HM}}{\text{MW}_e}\right)$; on discharge burnup $\left(\frac{\text{MW}_e\ \text{days}}{\text{kg HM}}\right)$; core 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 mutually 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 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 Gen-4 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. A second useful figure of merit is the ratio of forecast years of resource availability remaining at the drawdown rate of the Gen-4 energy park – divided by the years remaining of capital asset physical lifetime. These two figures of merit can help characterize the achievable Gen-4 energy park's dynamic response within the context of either rising or diminishing energy demand vis-à-vis resource availability and asset lifetime constraints.

While steady-state or constant period demand growth scenarios for a single or for a specified mix of Gen 4 systems can well illuminate relative advantages and disadvantages of candidate Gen-4 systems, it will be essential to also consider *transition scenarios* wherein Gen-4 systems penetrate the ongoing existing nuclear energy enterprise in the decades around 2030. In that case, there will be transient shifts in the already-existing inventories of materials in process. At the time of Gen-4 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-4 introduction, and will be ignored in the scenario development.)

References

1. 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).

1.5.6 Summary

The Gen-4 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 Gen-4 concepts will be evaluated as well as individual concept types to provide the Gen-4 Roadmap insights regarding achievability of Gen-4 Sustainability Goals.

1.6 Outline of the FCCG Report

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 Gen-4 concepts.

Chapter 3 presents the results of numerical simulations of various postulated scenarios for Gen-4 global market penetration. The scenarios are run as 100 year simulations with the WEC/IIASA nuclear energy demand as the forcing function and internal mass flows and availability of ore as the constraints. Edits of ore cumulative drawdown and cumulative waste arisings are provided relevant to the Gen-4 sustainability goals. The scenarios are organized on the basis of the four generic fuel cycles – once thru, partial recycle, full Pu recycle, and full TRU recycle.

Chapter 4 presents a review of the current status of fuel cycle technology and the directions of current R&D.

Chapter 5 presents a synopsis of the status of the institutional regime which provides both enabling structure and boundary conditions of 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 the scenarios – regarding efficacy of various generic fuel cycles in meeting Gen-4 goals. Fuel cycle crosscutting R&D is identified – drawing on the lessons learned in Chapters 3 and 4 – and prioritized recommendations are made for Gen-4 fuel cycle crosscutting R&D.

Gen-4 Fuel Cycle Crosscut Group

Chapter 2: Current Fuel Cycles and Proposed Gen-4 Fuel Cycles

November 1, 2001

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Chapter 2

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2.1 Overview: Segments of The Fuel Cycle

2.2 Current Worldwide Deployments of Fuel Cycle Infrastructure

2.3 Survey of Potential Gen-4 Fuel Cycles

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 first – the once-thru cycle –. the discharged fuel is put into interim storage awaiting eventual disposal in a geologic repository – see Figure 2. The second option is the 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 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 UOX once-thru 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-4 deployments will build.

Mining
Refining
Enrichment
Hands on Fab
Reactor Irradiation of UOX
Purex Recycling
Fab of MOX
Waste Form Production
Reactor Irradiation of MOX
Interim Storage
Repository

Fig. 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.

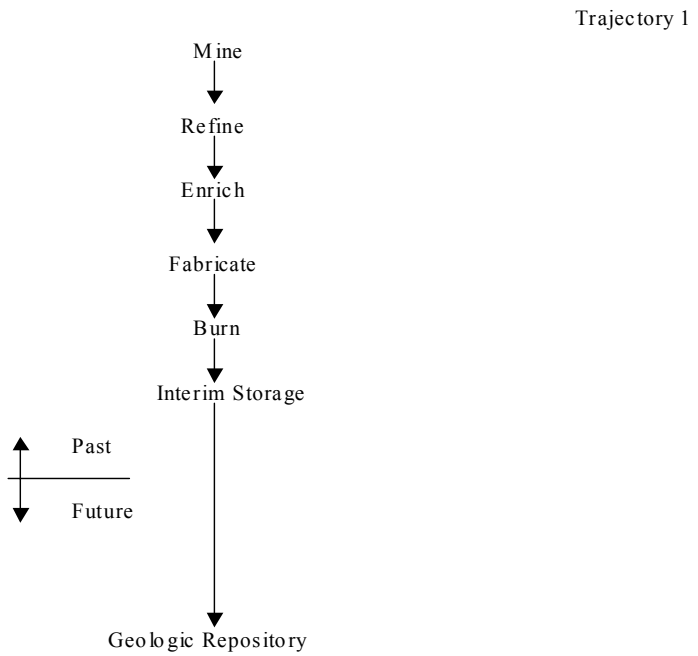

Fig. 2. The Once Thru UOX Cycle

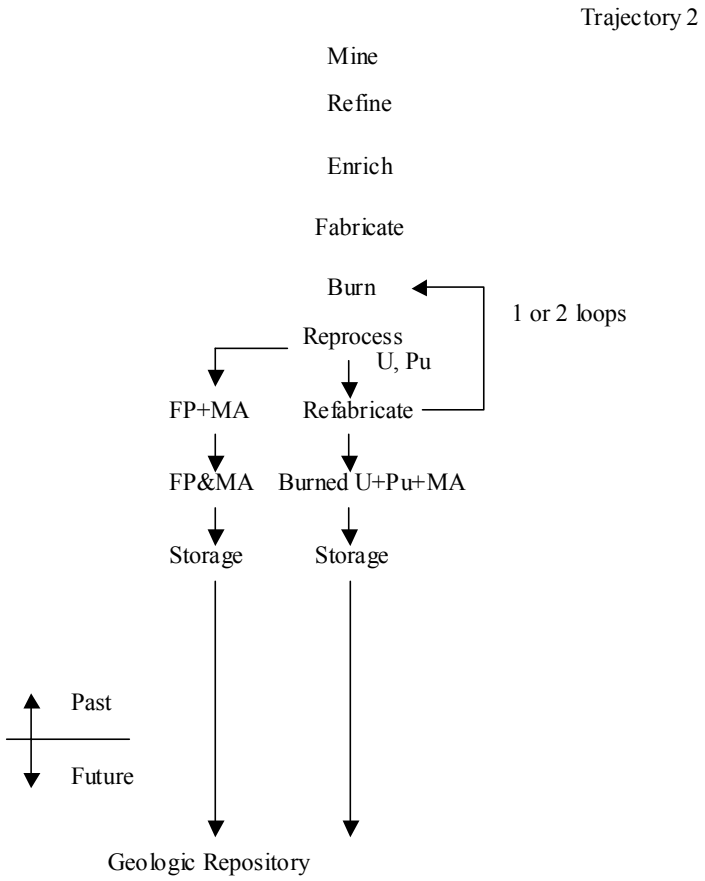

Fig. 3. The LWR MOX Recycle Cycle

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 could, in principle, 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] that is 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 highly specialised chemical facilities and, because they handle some 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 practised 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 1 Conversion plants and processes **[23]**

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

The Russian conversion plants appear to be tightly integrated to their 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 **Commercial enrichment facilities**[5] [23]

Land	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 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 replacement of the gaseous diffusion plants by high–performance gas centrifuges vs laser isotope separation technology will have to be addressed soon [24]. However, laser isotope

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

separation development has been scaled–down drastically and even stopped in the USA and France. New enrichment plants in the medium term in Japan, USA and probably France will, therefore, likely be based on centrifuge technology.

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 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 is to be disposed.

UOX FuelFabrication

Table 3 shows current deployment of commercial UOX fabrication plants.

On a world–wide basis, LWR 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 between fuel vendors, excess capacity is likely to be reduced in the future.

UOX Fuel Fabrication for CANDU

Natural uranium fuel is easy to process into fuel assemblies. The short, simple fuel assemblies, called fuel bundles, for the CANDU (PHWR) reactors are easily produced by countries that operate these reactors. Korea, Argentina and Romania all have independent PHWR fuel fabrication facilities sufficient to meet their demands, and China will build a fuel fabrication facility to meet the requirements of their CANDU reactors now under construction. Furthermore, PHWR fuel cycle costs are lower than those of other reactor designs, because natural uranium is relatively inexpensive, the uranium utilisation (amount of energy produced from the mined uranium) is good, and because of the simplicity of the PHWR fuel bundle.

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

Table 3 **Commercial fuel fabrication plants** [23]

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	Yusung	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	Kaeri	KAERI	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	

A thin layer of graphite (CANLUB) is applied to the inside surface of the sheathing to reduce the pellet/sheath interaction. The Zr-4 sheathing is closed at both ends by resistance welded end caps that serve three purposes: to provide a seal for the contents of the element, to provide effective element termination for attachment to the end plates, and to provide the structural component for interfacing with the fuel handling system.

The basic simplicity of PHWR fuel has led to extremely low fabrication costs, because only a small industrial factory is required and manpower needs are moderate. The use of natural uranium also eliminates dependence on foreign sources of enrichment (although fuel cycle costs are actually reduced with SEU in the PHWR). Thus, a country with PHWRs has the possibility of becoming independent from foreign nuclear fuel vendors within a fairly short time. In contrast to reactor systems requiring enriched fuel, all manufacturing stages of natural UO_2 fuel can be accomplished without special criticality restrictions.

MOX Fuel Fabrication

Fueling of LWR plants with 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. The 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 improves the non-proliferation character by avoiding 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 an 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 is requested by the utilities embarked in a recycling strategy. Developments currently pursued focus on smoothing fuel cycle management by increasing burn-up of the MOX fuels, the

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

possibility of load-following with MOX fuelled reactors (already demonstrated and licensed in France) and the possibility of increased MOX fraction (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).

JNFL 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 attractiveness of FBR-fuel when compared to LWR-fuel is the higher burn-up achievable, higher power ratings and the possibility to burn isotopes that would be neutron poisons to LWR fuels.

Liquid metal cooled fast reactor (LMFR) technology has been developed in many countries, including France, Germany, India, Japan, the Russian Federation, the United Kingdom and the USA [28]. The liquid metal is sodium but development is underway in Russia to use lead or lead-bismuth. Other developments (US, France) envisage gas-cooled fast reactors.

Monju (Japan) and BN600 (Russia) are the only commercialized fast reactors currently deployed. Development of fast reactor technology slowed in several countries during the last decade in the context of an overall slowing-down of nuclear power development.

Most FBR fuel is MOX with higher Pu content compared to LWR-MOX fuel. Fuel pins of stainless steel cladding tube are assembled in a hexagonal wrapper tube. The fabrication technology is 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 FR Joyo and prototype FBR Monju since 1973. MOX fabrication capacity for fast reactor use is not expected to grow for several decades.

Recently, increasing interest is going to fast reactors both for long term energy sustainability and, in the nearer term, as part of a partitioning and transmutation scheme; here new fuel matrices are needed. 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 industrial 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 subassemblies does not pose any serious radiological problems. However, one of the main drawbacks of 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.

No commercial thorium fuel fabrication plants currently exist. Despite possible increases in the costs of manufacturing, the experience gained in Th-fuel manufacturing has established a pre-industrial status for thorium fuel fabrication these processes. Even so, a Gen-4 shift to a Th-cycle would demand major investments in all fuel cycle steps.

Coated Particle Thorium Fuel Fabrication

Considerable experience was accumulated in the USA in *thorium coated particle 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. Performance of the fuel exceeded predictions; it was irradiated at temperatures greater than 1 300°C 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.

In Germany and the USA, a specific 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) programme. 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.

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 shutdown sometime in the coming decade. World-wide commercial recycling capacity after 2015 is forecast to be some 4 300 tHM/y.

Table 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

2.3 Survey of Potential Gen-4 Fuel Cycles

As described in Chapter 1, the potential future Gen-4 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 1.

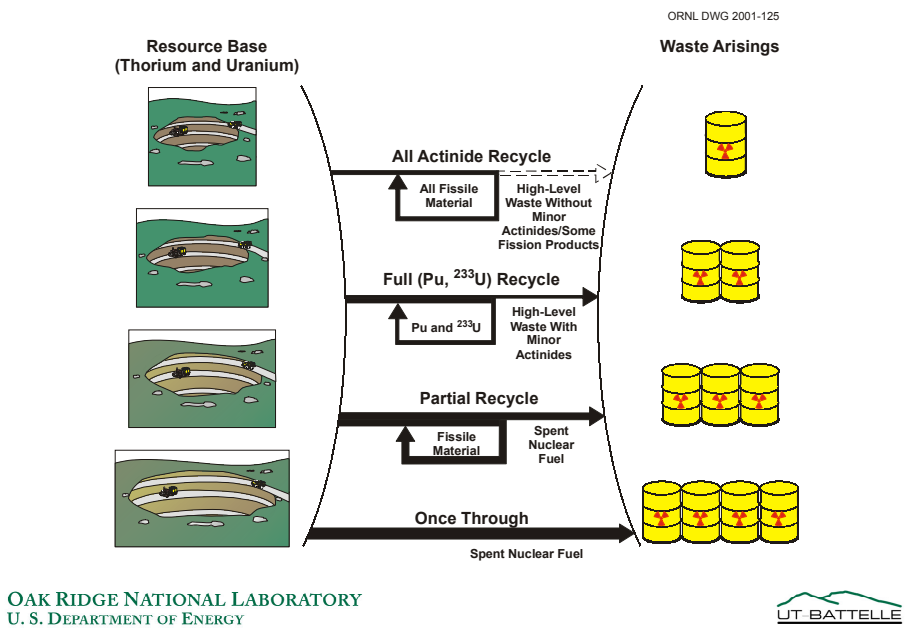

Fig. 4 Generic Gen-4 Fuel Cycles

In Figure 1, the "front end" links in the fuel cycle chain are shared by many of the Gen-4 concepts; similarly most concepts share similar "back end" waste disposition links in the fuel cycle chain. However, Gen-4 concepts are very diverse 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 and choice of recycle technology and refabrication technology. This is illustrated in Figure 5.

Links in the fuel cycle chain

Mining Refining Enrichment	(Common to All Concepts)
Hands on Fab Reactor Irradiation Purex Hands on fab – e.g., MOX or Dry Recycle Remote Fab Reactor Irradiation Waste Form Production	(Concept Specific)
Shipping Repository	(Common to All Concepts)

Fig. 5 Fuel Cycle Diversity of Gen-4 Concepts

In light of this diversity, the FCCG conducted a survey among the water, gas, liquid metal and non conventional TWG's to determine the breadth of fuel cycle technological needs as a first step toward identifying required cross cutting fuel cycle related R&D. Table 5 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 5 should now be updated in light of additional information available to the TWG's, it is already evident that potential Gen-4 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 U^{238} or Th^{232} are considered as well. Recycle options include both aquaeous 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 (for low decontamination recycle products).

The results of the survey helped the FCCG determine the scope of it's 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.

Table 5. 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 thru	Water	UO_2 pellets						N/A	N/A	T		
IRIS	Water	UO_2 pellets						N/A	N/A	T		
Multi-application Small LWR	Water	UO_2 pellets						N/A	N/A	T		
Homogenous Thoria-Urania	Water	ThO_2-UO_2 pellets, ~70% Thoria						N/A	N/A	T		
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: enriched – uranium metallic dispersion fuel			N/A	N/A	T		
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	UO_2 pellets						N/A	N/A	T		
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	UO_2 pellets						N/A	N/A	T		
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
Fast Spectrum												
(2) Limited Resource Recovery with disposal of FPs and some Actinides												
Thermal Spectrum												
IRIS	Water	PuO_2-UO_2 pellets						Yes	No		T	
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: plutonium-bearing, metallic dispersion fuel			Yes	No	T	T	
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	PuO_2-UO_2 pellets						Yes	No		T	
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	PuO_2-UO_2 pellets						Yes	No		T	
HTGR with LWR Pu feed	He	TRISO-Coated PuO_2						PUREX				T
Gas-cooled Reactor w/ LEU/Th	He	TRISO-Coated ThO_2 Alt: TRISO-Coated UO_2					TRISO-Coated UCO		Particle Separation			T
Fast Spectrum												
Tight-lattice BWR Thoria Cores	Water	PuO_2-ThO_2 pellets						Yes	Maybe		T	Maybe
(3) Multiple Recycle of												

	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
Fissile Act's, with disposal of FPs and of non-fissile and MAs												
Thermal Spectrum												
Fast Spectrum												
EFR, DFBR	Na	MOX pellets						PUREX			T	
S-PRISM	Na	MOX pellets						PUREX			T	
(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)						T	T			T
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			T
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)						T	T			T
Gas Cooled Fast Reactor	He	MOX pellets	Mixed Nitride pellets				CERMET, CERCER	T	T			T
SPRISM	Na				U-Pu-Zr				Pyro			T Casting

	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
BREST	Pb		Vibro or pellet						Pyro or Flouride			T Vibro or pelletizing
STAR-LM	Pb-Bi		Vibro or pellet						Pyro			T Vibro or pelletizing
STAR-LM	Pb		Vibro or pellet						Pyro			T Vibro or pelletizing
ENHS	Pb-Bi or Pb				U-Pu-Zr				Pyro			T
IFR	Na				U-Pu-Zr				Pyro			T

Gen-4 Fuel Cycle Crosscut Group

Chapter 3

Illustrative Scenarios: Gen-4 Performance Against Goals

November 1, 2001

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Independent check calculations of the scenario results are in progress at JNC to verify the DYMOND code. Additionally, the fuel cycle unit process costs and their application in the scenarios will be reviewed and coordinated with the work of the newly formed Gen-4 Economics Crosscut Group. Revisions will be made if required.

FCCG

Chapter 3 Scenarios

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3.1 21st Century Gen-4 Global Nuclear Energy Scenarios

One hundred year global aggregate nuclear energy supply scenarios have been simulated for numerous potential Gen-4 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 decommissiong spread over the next 40 years. The scenarios all use the world uranium 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 predictions produced in the World Energy Council/IIASA predictions 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 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]

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 1 which conceptually illustrates the several sources of fuel feedstock). The sources to be exploited 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. The conversion ratios of each Gen-4 concept reactor are specified as input to the simulation and used to determine the composition of discharged fuel.

Market economic penetration is not modeled; both the dates of commercial availability of various Gen-4 concepts and the fractional mix of Gen-4 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-4 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-4 global energy parks in satisfying the Gen-4 sustainability goals pertaining

[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.

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

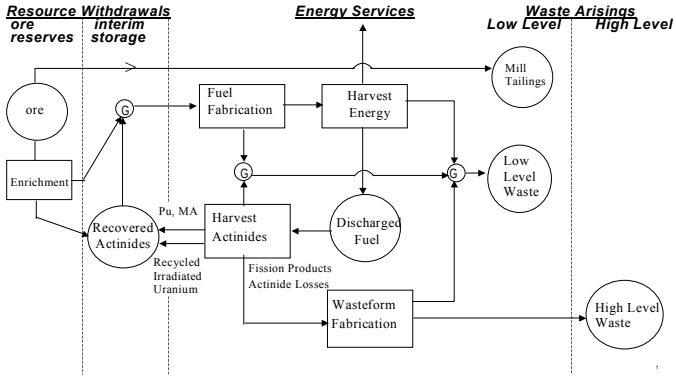


Fig. 1 The Fuel Cycle

withdrawals vs time and their cumulative withdrawals and the 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-4 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-4 global infrastructure and are relevant to the Gen-4 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-4 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 it's 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 it is spent – 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 scenarios are run using the code, DYMOND[2] which was prepared specifically for the Gen-4 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 thruput 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-4 TWG's – they are best judgment values culled from sources available to FCCG membership.

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 the attachment.

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

3.2 The Scenario Set

The FCCG has organized its work using four generic fuel cycles which are depicted in Figure 2; -- the once through, the partial recycle, the full plutonium recycle, and the full transuranic 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-4. For example, in principle there are four inventories of actinide-containing resource available to fuel the global nuclear energy park (see Figure 1):

Virgin ore
Enrichment tails
Spent Fuel
Separated irradiated uranium

However, as illustrated in Figure 3, the four generic cycles avail themselves differently of these resources and use different asset combinations to do so. The once-thru 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-4 concepts rely on the once thru or partial recycle generic fuel cycles for fuel cycle services.

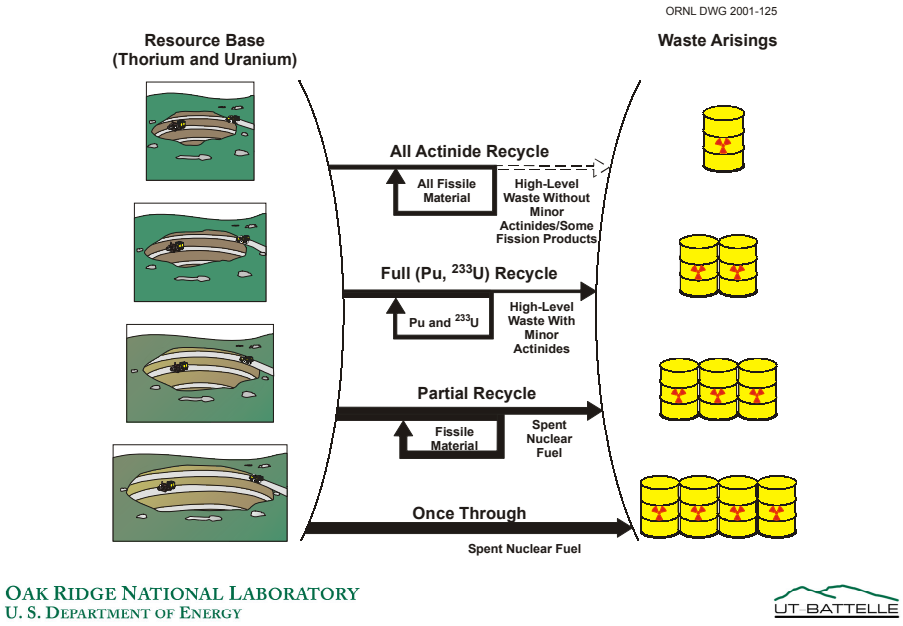


Fig. 2 The Four Generic Fuel Cycle Figures

The full plutonium and full transuranic recycle generic cycles draw from the last three resource inventories shown in Fig. 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-4 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. Figure 3 illustrates what is fuel and what is waste for the four generic fuel cycles.

Given the roughly one hundred candidate Gen-4 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-4 goals.

Table 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-4 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-4 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-4 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 weighting 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-4 concepts will 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-thru base case) involve a dynamically evolving mix of Gen 3 and Gen-4 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 thru.

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

Table 1. Scenario Cases Examples to Illustrate Fuel Cycle Issues

- Use Case B of WEC/IIASA for World as a Whole
- Assume 90% of already deployed Gen 3 LWR's worldwide will receive 20y life extension
- Cases 1 and 6 are the extremes; setting physically achievable bounds; Cases 2 thru 5a allow for practicalities of markets

Example Cases

A) Once Thru Thermal
   1. New Additions = LWR UOX Once Thru
   2. LWR UOX Once Thru; New additions (after 2010)
      HTGR Gas Turbine UOX Once Thru (50%)
      + LWR-UOX Once Thru (50%)

B. Limited Thermal Recycle (after 2010)
   3. New Additions = LWR UOX → LWR MOX mono
      → Waste

C. Full TRU Recycle: Symbiotic Fast/Thermal (after 2025) for Waste Management

   4a. New additions = 80% LWR (UOX) + 20% Fast Burner (BR = ½)
   4b. New additions = Fast Burner (BR = 1) (use up LWR discharged fuel)

D. Full TRU Recycle: Breeding (after 2030) for Limiting Ore Drawdown
   5. Case 4b – Switching in 2030 to New additions – Fast Breeders (BR = 1.7)

E. Full TRU Recycle: Symbiotic Feedback of Fissile to Thermal Systems
   6. New additions = fast breeders + LWR_MOX (TRU)

3.3 Scenario Results for the Once-Thru Cycles

3.3.1 The LWR-UOX Once Thru Base Case Scenario

During the current century, the global nuclear energy park is predicted 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 thru generic fuel cycle is the "business as usual" scenario where the LWR UOX once thru 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-4 energy parks can be compared. The attribute set used for the LWR-UOX power plants is displayed in the attachment. (Enrichment of 4.2% and 50,000 MWd/tonne are used.)

Figure 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 thru power plants are deployed; ~250 kilo tonne SWU/year of enrichment capacity is used (up by a factor of five from the 50 kt/y in year 2000); and ~40 kilo tonnes/year of UOX fab capacity are deployed (up by a factor of four from the 12 kt/y in year 2000).

Figure 5 displays the trends in fuel resource inventories and waste inventories. The 4.1 million tonnes of uranium in the known ore reserves (recoverable at #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 53 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 under 11 mills/kW_e hr (see Fig. 6)

Focusing back on the year 2050 – twenty years beyond the intended latest introduction date for Gen-4 concepts, in the business as usual scenario where LWR-UOX once thru 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 addition repository of similar capacity.

[4] In the middle plot of Fig. 5, Curve 2 is spent fuel awaiting disposal (30 year lag time) and Curve 4 is spent fuel already in the repository; in the bottom plot Curve 1 is the sum of the two.

The cost index for the fuel cycle component of the overall cost of energy production (see Fig. 6) is rising slowly by 2050 due in large measure to two factors: the increasing cost of ore and the growing waste management burdens at the back end where carrying costs for the accelerating spent fuel inventories are building up. The cost of spent fuel waste management consists of annual interim storage charges followed by a one time charge to package and emplace the fuel in a repository (see the attachment). The interim storage period is modeled here as 30 years and when the economy is accelerating, the interim storage cost per terawatt electric hour goes up.

This business as usual scenario serves as a base case against which to compare various Gen-4 concepts or clusters of concepts.

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

The second scenario considered for the once thru 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 thru and PBMR-UOX once thru 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. 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 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 significantly 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

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

$$\approx \frac{35}{46}\,\frac{50}{80} \approx 0.48$$

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

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

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

Figure 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 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 accrue to higher burnup and reduced waste management costs.

The lower plots in Figure 9 shows the time evolution of several properties of this global energy park – *normalized year-by-year to the corresponding properties for the LWR 100% market share base case energy park.* The plots are log-linear. Specifically, in the bottom plot on Figure 9, Curve 1 is cost of fuel cycle services in this scenario normalized year-by-year to the cost of the LWR-UOX once thru base case. Similarly plot 2 is the normalized cumulative spent fuel arisings year by year (whether in interim storage or already in the repository). Plot 3 is the normalized cumulative plutonium mass in that spent fuel; plot 4 is the same for minor actinides, and plot 5 is the normalized cumulative ore withdrawals from reserves and new discoveries.

[5] Detailed attribute data are needed to compare toxicity flows to the repository. However, as shown in Section 3.5.1.1, increased buildup of minor actinide content resulting from higher discharge burnup is unfavorable to toxicity in the waste.

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

The third scenario considered for the once-thru generic fuel cycle is identical to the previous one except that the Prismatic HTGR once thru 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 thru and Prismatic HTGR once thru power plants. The attributes used for the HTGR are shown in the attachment; when compared to the PBMR the salient differences are in a higher enrichment (15.5% vs 8%) and a higher discharge burnup (121 GW_t d/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 10 thru 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 actually greater during the second half of the century (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 significantly even though ore cost by late in the century is at the $200/kgU level.

Figure 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 thru base case*.

3.3.4 Observations Regarding Once-Thru Generic Fuel Cycles

Under the assumptions used for the scenarios, the once-thru 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 thru generic fuel cycles which have been simulated do not improve on the LWR-UOX once thru base case in regard to the first Gen-4 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 (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 thru generic fuel cycle concepts against the Gen-4 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 less than a factor of two – even as ore becomes more scarce and waste management costs increase.

Concerning the fuel cycle contribution to the Gen-4 Safety Goal, the health effects deriving from the overall fuel cycle are currently significantly influenced by mining and milling operations and by recycle operations (see Table 2). Both are related to harvesting fissile from one source or another to fuel the deployed reactor park. The reactor park itself is a slightly less but still a comparable contributor with fuel cycle services on overall health effects of nuclear energy.[6] The values in Table 2 suggest that for once thru 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 safer mining and mill tailings management methods. 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.

[6] In any case, the effects are quite negligible compared to background and compared to other energy supply options. [Ref. to Hirshberg]

Table 2. Human Health Effect Derived from Non-Reactor Links in Fuel Cycle
Man Sv/GWa

Fuel cycle stage	Once through	Recycling (MOX)	Source
Mining & Milling	1.0	0.8	^{230}Th, ^{226}Ru
Conversion, enrichment, and fabrication	0.0009	0.0009	
Power generation	0.6	0.6	^{14}C
Recycling/vitrification	N/A	1.2	^{14}C
Transport	Trivial	Trivial	
Disposal	0	0	
Total	1.6	2.6	

The once thru concepts considered in these scenarios have the considerable merit of being nearer to commercial deployment than many of the other Gen-4 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. Thus, anticipating that recycle can improve performance against the first two sustainability goals, fuels development for once thru Gen-4 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-4 goals – by *means of a symbiosis with other, more recycle-intensive Gen-4 concepts. Such a symbiosis scenario is evaluated in Section 3.5*.3.

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 thru power plants (see Figure 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 UOX and 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 thru fuel which is discharged adds to the already existing 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.

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 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

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

excess. However, if the energy park is growing (as it is), then the deployment ratio of MOX to UOX must be reduced from 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.

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

Using the attributes of the UOX and MOX plants shown in the appendix 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 14 where the inventories of ore, enrichment tails and spent fuel are displayed for this scenario.

Figure 14 shows that the MOX spent fuel inventory and the recovered irradiated uranium inventories both grow subsequent to 2010, but 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 thru 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 and eventually disposing discharged fuel in a repository.

Figure 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 ~30,000 tonnes/year – up by a factor of eight from the ~4,000 tonnes/year currently deployed.[8]

The cost index is shown in Figure 15. The overall increase of fuel cycle services is less than a factor of two as MOX mono recycle penetrates the market. The increase is still dominated by rising ore cost because 90% of the park is still LWR-UOX and the waste

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

management expenses benefit from the cheaper handling of separated uranium as compared to UOX spent fuel, -- in fact, in the long run costs decrease slightly.

An initial 20 year surge in the recycling component of the fuel cycle costs is seen between 2010 and 2030. This resulted from a scenario choice on how fast to draw down the initial 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 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 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 less than zero incentive 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 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 thru 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 due to increased

[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.

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 Observations Regarding Partial Recycle Generic Fuel Cycles

Compared to the LWR-UOX once-thru 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 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 worst than the LWR-UOX once thru base case.

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

All-in-all, the gains against the Gen-4 waste management sustainability goal SU-2 which accrue to mono recycle, are big enough to be important, but the gains against goal SU-1 on resource use are marginal.

Partial recycle has the significant virtue that it can be applied in the near term using LWR power plants already deployed and (at 100% MOX loading) expected to be deployable within ten years. *Moreover, 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-4 Goals on efficient use of repository capacity and on costs of waste management.*

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

3.5 Scenario Results for The Full Pu and Full TRU Recycle Fuel Cycles

The full recycle fuel cycles introduce both multiple recycle and fast neutron spectrum reactors which as shown in Figure 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 ability 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 full recycle fuel cycles and fast spectrum reactor types will occur by means of 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 will be for waste management symbiosis; a transition will occur later for fissile supply symbiosis.

Three scenarios are used to illustrate these features: in the first scenario, LWR UOX once-thru plants are deployed paired with fast *burner* reactor operating on a closed cycle to manage the back end of the fuel cycle – the goal is to reduce the waste management burden of the nuclear energy park overall. The second scenario illustrates the efficacy of converting already deployed fast reactors from burning to breeding at the appropriate time as the way to reduce virgin ore withdrawals when economic conditions favor the switchover – with no degradation of the waste management function. The third scenario illustrates the degree to which the nuclear energy park could expand beyond electricity production by late in the century – to enter other segments of the global energy market, specifically the production of hydrogen as a synthetic chemical energy carrier (while remaining within the limits of Redbook Known plus Speculative ore reserves). This expansion is accomplished with no degradation of performance pertaining to waste management.

The reactor and fuel cycle attribute sets used for these scenario cases are documented in the attachment. The doubling times of the three different "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 years for the nuclear electrical component of the global energy demand growth rate being used from the WEC/IIASA 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 one GWe plant and a 600 MWe fast burner reactor having a conversion ratio of 0.5. The LWR's operate once-thru, 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. The initial working inventories of the newly deployed fast burner reactors are derived from TRU obtained by recycling the existing 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-thru 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 16 through 18 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 as follows: the LWR discharge fuel is recycled; 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 recycling fast reactor fuel cycle. This feedstock into the fast reactor cycle is eventually fully consumed thru multiple recycle because the fast reactor is a net burner, not a breeder. Using this symbiotic paring of thermal and fast reactors, *fission products and trace losses of TRU's from the recycle/refab 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-thru cycle delivering the same energy services.

The inventory of discharged fuel from LWR's deployed prior to 2025 and still operating on the once-thru cycle is not sent to waste either. This inventory is recycled to supply the initial working inventories for the fast burner reactors as they are deployed. This initial inventory 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 ratio are sized to exactly fully consume the TRU contained in the LWR discharged fuel so that no actinides from the power park are consigned to waste. But in an economy growing at the rate assumed here, once the initial 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 of 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 17, 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 thruput, 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 refab in the UOX and fast reactor cycles are different as shown in the appendix. Figure 18 shows the evolution of overall fuel cycle service costs for the symbiotic energy park.

The bottom curve in Figure 18 shows the performance values for this energy park scenario *ratioed year-by-year to those of the reference LWR once-thru base case*. The symbiotic pairing of LWR's with fast burner companions achieves significant (three order of magnitude) reductions in waste from the nuclear energy park at a favorable cost benefit ratio in terms of the fuel cycle component of cost.

3.5.1.1 Full TRU versus Full Pu Recycle – MA Effects on Toxicity

This scenario 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 here – 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 19).

In that study, several alternative ways were evaluated to achieve the desirable 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 thru multiple recycle. These two produced equally effective reductions in mass and toxicity flows to the geologic repository. Figure 19, 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-4 waste management goal, SU-2; the data from the OECD-NEA study

shows the importance of minor actinide recycle to achieving significant toxicity reductions. 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* reduce the drawdown rate of ore reserves by 20%.

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 scenario illustrates 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 Breeding

Fast reactor plants, once deployed, provide 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. The next two scenarios illustrates the utility for including such a capability as part of the global energy park – providing the means to accommodate more rapidly to changes in financial, environmental, or social conditions.

Starting in 2020 it is assumed that capacity additions are made using fast reactors of breeding ratio 1.0 which operate thereafter on a fissile-self-generated full TRU recycle fuel cycle. (This breeding ratio is achieved by the core loading pattern selected in a reactor design which can reach breeding ratios anywhere over a broad range.) 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 and from the excess TRU produced in the already deployed breeders. *If inadequate transuranics exist for new fast reactor deployments to meet demand, the deficit is made up by constructing new LWR plants operating on the once-thru UOX cycle.*

In the ensuring years after 2020 the fast reactor plants are fissile self sufficient but produce no excess fissile. As recycling of the initial LWR spent fuel inventory and its ongoing additions to supply initial inventories for new fast reactors drawdown the LWR spent fuel inventory, the availability of TRU to supply new fast reactor deployment nears depletion. A time quickly comes after which the market shares of thermal and fast reactor deployment is controlled by balancing the source of fissile coming from LWRs in the park to initial inventory requirements needed to deploy new breeders. This scenario illustrates those tradeoffs. Figures 20 through 22 show the results.

The bottom curve in Figure 20 shows that within a decade – before 2030 – fissile supplies from recycled LWR discharge fuel would run short of requirements for fast reactors alone to keep up with demand. LWR deployments, fueled by virgin ore, would continue to 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. The availability of Redbook virgin ore reserves could be extended by only 5 years to 2055 before new discoveries become necessary. Even so, the cumulative needs of the LWR once-thru dominated energy park to year 2100 would be reduced to ~30 million tonnes.

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 business as usual LWR once-thru base case because nothing but fission products and trace losses of heavy metal in recycle operation goes to waste.

Figure 22 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 breeding cycles (BR=1.0) in place of burner cycles (BR=0.5) can achieve the same waste management performance against Goal SU-2. It improves performance on efficient resource use Goal SU-1 only marginally. It is evident that the fissile self sufficient breeder 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 and 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 U238 inventories which are stored in the enrichment tails and recovered irradiated uranium inventories – it is necessary to deploy breeder cycles having a doubling time less than 20 years.

Anticipating the exhaustion of 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 cycles 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[10] 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*. 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 strong dependence of cumulative uranium ore utilization during the transition attendant to physically achievable fast reactor doubling time has been studied systematically by Koike, et al.[11] Figure 23, 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

[10] i.e., within still available ore reserves.
[11] 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).

reactor dwell time (2 years up to 8 years), and 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 --- case B of the WEC/IIASA 1998 projection.)

For the 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 to sustainability within the Redbook Known plus Speculative ore reserves. Otherwise, the TRU availability from LWR discharge plus self breeding retards the buildup of the fast reactor market share, new LWR's must be built to meet demand, and their lifetime forward fueling requirements exceeds the Redbook reserves. The figure 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.

The next scenario illustrates a smooth switchover. It starts like the previous case with fissile self sufficient (BR=1.0) fast reactor deployments starting in 2020. But starting in 2030, all 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 7 years. The results of this scenario are shown in Figures 24 through 26.

Starting from its "jump start" with 25% market share already held by BR=1.0 fast reactors, in 2030, 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 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 finding new ore beyond Redbook reserves, is being fueled on U238 from the multi-century supply of enrichment tails, and is sending only fission products and trace losses of heavy metal to the repository.

Figure 27 shows the performance of this nuclear energy park, normalized year-by-year to the performance of the LWR once-thru base case. Sustainability is achieved in this scenario over a 70 year transition period at a increase of about a factor of 4 increase in cost of fuel cycle service index.

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 the slower 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 and just meet requirements for new

deployments. Alternately, nuclear park operators might use excess fissile production capacity to expand energy market share, as fossil sources diminish in availability or desirability. Such a scenario is shown next.

3.5.3. 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 in market share so that withdrawals of virgin or reserves cease and the fast reactor energy park is fueled from the extensive inventory of U238 from the enrichment tails. An alternative symbiotic energy park is one 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 these thermal reactors and from enrichment tails. The energy park is shared such that the mass flows of transuranics just balance and no buildups of transuranics occur.

For example, consider a steady-state situation of LWR-MOX once-thru 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:

$$X\,(0.6) + (1-X)\,(1.4) = 1$$

It is seen that the 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 example illustrated here LWR-UOX once-thru plants are deployed until 2030. Then, starting in 2030, 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.

The scenario results are shown in Figures 27, 28, and 29. 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 2055 time frame excess transuranics become available and LWR-MOX plants begin to be deployed. By the end of the century the park is rapidly approaching its asymptotic configuration comprising an unchanging roughly 50/50 market share of fast and thermal systems.

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. And as before, by initiating the transition in the 2030 time frame, it is

possible to convert to a sustainable growing energy park, while meeting demand without exceeding the limits of the Redbook Known plus Speculative ore reserves. The performance indexes, normalized year by year to the LWR-UOX once-thru base case are shown in Figure 29.

At this time the DYMOND code has not been configured to model the more practical scenario wherein the already existing LWR's cease to rely on UOX, but rather are refueled using MOX 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 reactor 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. The FCCG will evaluate that scenario (along with others recommended by the TWG's) during the second phase of its activities.

3.5.4 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 existing inventories of spent fuel from all prior nuclear power operations.

Thermal reactors dominate market share at least until mid century (and grow dramatically to 3000 GWe at their smallest) no matter what full recycle case is considered. *Tying up in working inventory and burning down inventories of spent fuel from once-thru and partial recycle themal reactor deployments will certainly be the first function for Gen-4 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.

Reprocessing allows to partition 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. 19). 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-thru base case, the cost index for full recycle scenarios changes by less than a factor of 4 and leads to tolerable changes in the 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 Redbook Known plus Speculative ore reserves while meeting growing demand for electricity from the thermal-reactor dominated nuclear park, conversion from fast reactor net burning to breeding must occur not

later than 2030; 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 breeders 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 – starting no later than 2030.

The principal features regarding Gen-4 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-4 deployment these features are exploited to manage the spent fuel waste inventories from previous and ongoing once-thru 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 Summary of Lessons Learned From the "Cornerstone" Scenarios

The scenarios modeled here are idealized in relying only on physical mass-flow constraints and in prescribing the Gen-4 concept introduction dates and target market shares as inputs. They serve as 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 thermal systems (rising to ~3000 GWe) will dominate market share for decades – until around mid century. Their waste arisings, however, become quite significant already by 2030. The cumulative forward fuel requirements of thermal spectrum systems deployed up to around 2030 already sum to the ore reserves in the Redbook Known plus Speculative reserves.

MOX mono recycle significantly reduces 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. On the other hand neither higher burnup once-thru designs nor MOX mono recycle is effective in producing significant increases in duration of the adequacy of currently identified (Redbook) ore reserves.

Full recycle and fast spectrum systems, if introduced to perform a waste management functions 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. They quickly draw down the inventories of spent fuel from prior operations as well – to use the material for initial working inventories of new deployments of the fast spectrum power plants. The deployment of recycle/fast spectrum plants at ~20% of the energy park is sufficient to handle the waste from the entire park.

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 by the breeding ratio of the fast systems in the park – both of which can be adjusted.
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 new reactor or fuel cycle types 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-thru cycles can achieve fully sustainable (Goals SU-1 and SU-2) global energy supplies while filling numerous market niches – those suited to thermal reactors and those suited to fast reactors.

v. The component of cost of energy supply which is due to fuel cycle services is small in the first place, and doesn't change dramatically anyway – so a sustainable Gen-4 fuel *cycle* would appear to be affordable, and
vi. 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, – then new capacity additions subsequent to 2025 should be 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. However*, real 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 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 for thermal and fast reactor concepts and recycle-based waste forms and integrated decay heat management strategies for extending repository capacity.
iii. *Cost effective fast spectrum systems* whose early introduction into energy park growth will be to serve the function of back end 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. R&D on prospecting and mining technologies for finding and exploiting additional ore in a cost effective and environmentally friendly fashion – to provide for robustness in the environmentally friendly global energy security arena.

Figures for The Scenario Results

Figures	Case
4-6	LWR Once-Thru Base Case
7-9	LWR/PBMR 50/50 Market Share
10-12	LWR/HTGR 50/50 Market Share
13-15	LWR/MOX-Mono Recycle 50/50 Market Share
16-18	LWR/FR (BR=0.5) 80/20 Market Share
19	Full Pu Recycle vs Full TRU Recycle: MA Toxicity
20-22	LWR/FR (BR=1.0) 80/20 Market Share
23	Time Window of Opportunity for Transition to Resource Sustainability
24-26	Burner Switch to Breeding in 2030 – Sustainability Within Redbook Limits on Ore
27-29	Feedback of Excess Fissile To Thermal Reactors

(1) LWR Once-Thru

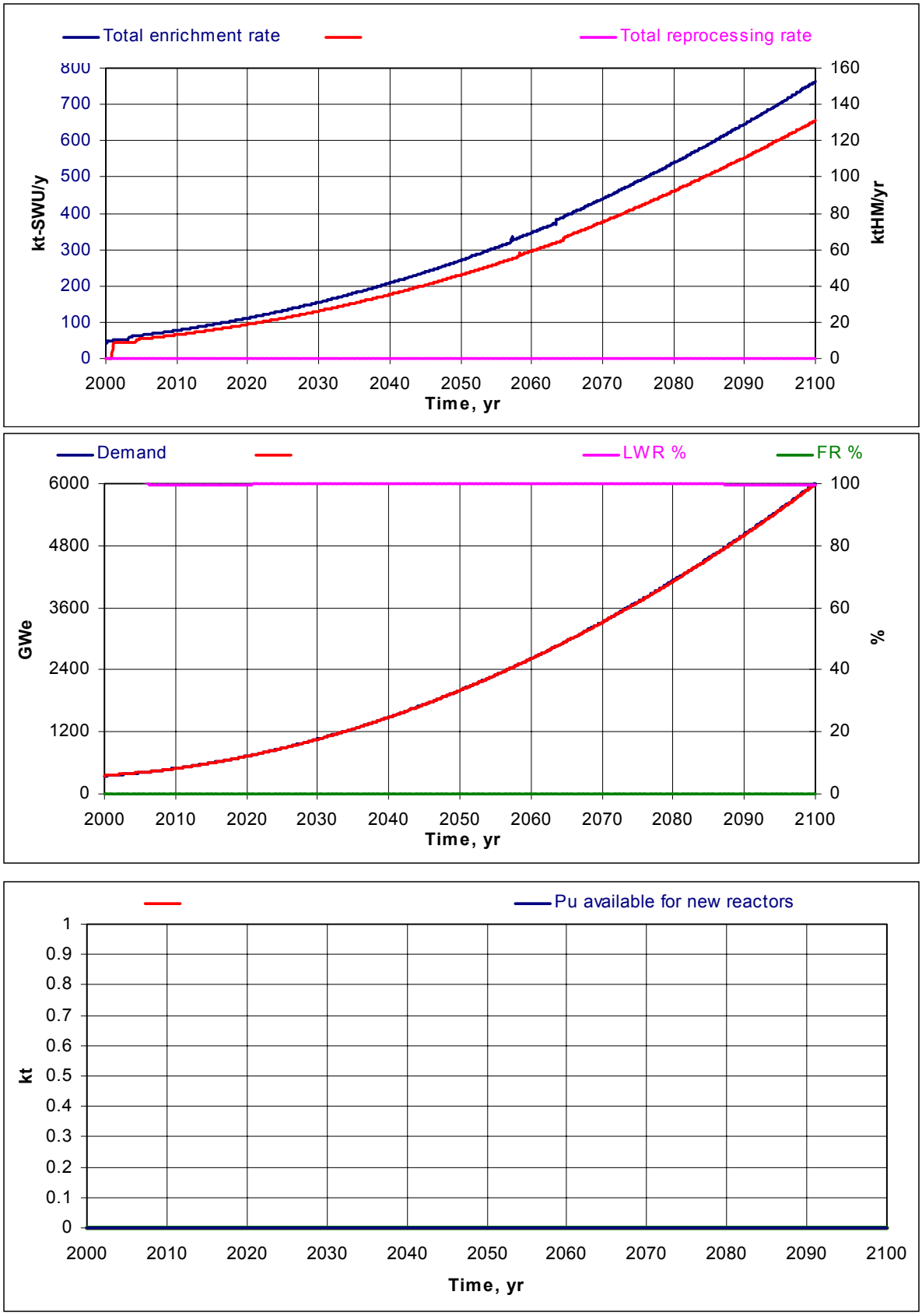
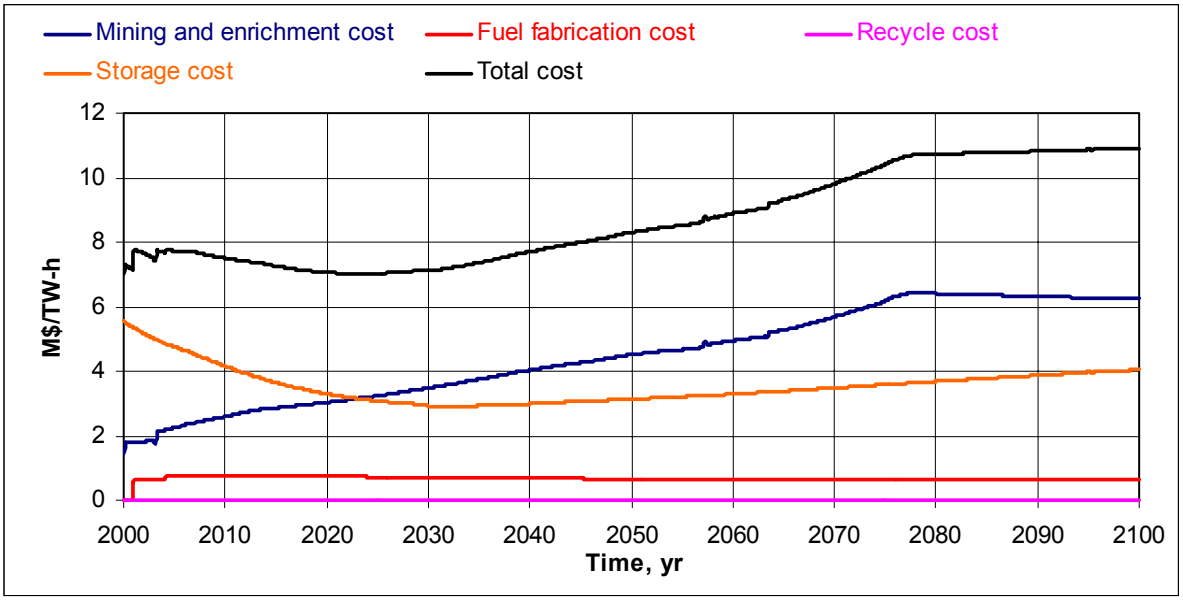


Fig. 4. LWR Once-Thru Base Case

(1) LWR Once-Thru

Fig. 5. LWR Once-Thru Base Case

(1) LWR Once-Thru



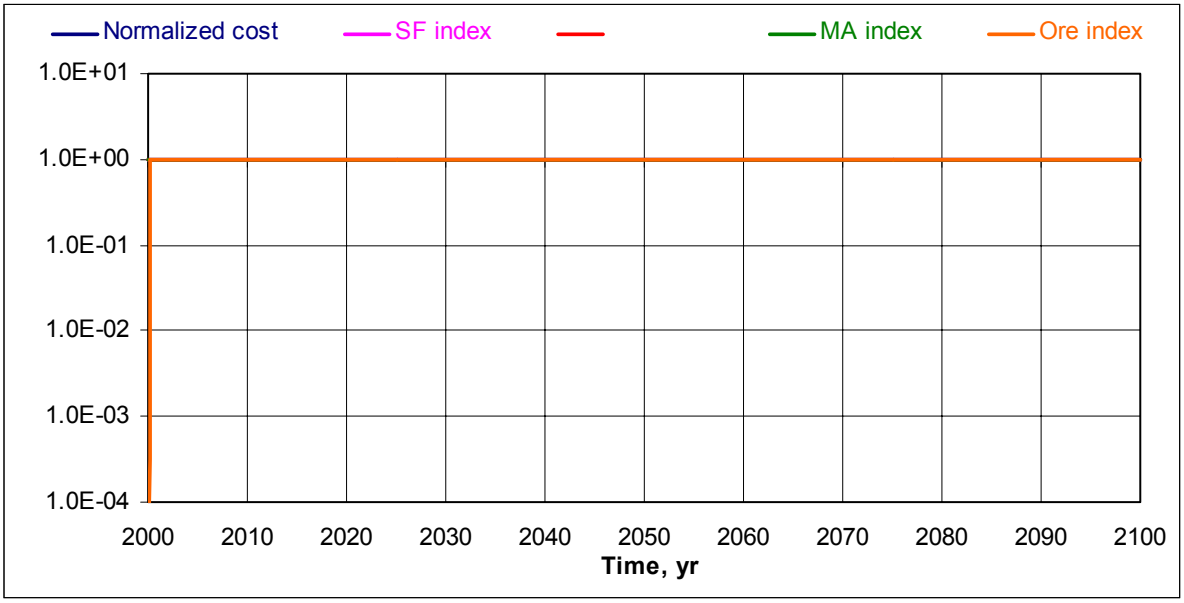


Fig. 6. LWR Once-Thru Base Case

(2) LWR Once-Thru + 50% PBMR

Fig. 7. LWR/PBMR 50/50 Market Share

(2) LWR Once-Thru + 50% PBMR

Fig. 8. LWR/PBMR 50/50 Market Share

(2) LWR Once-Thru + 50% PBMR

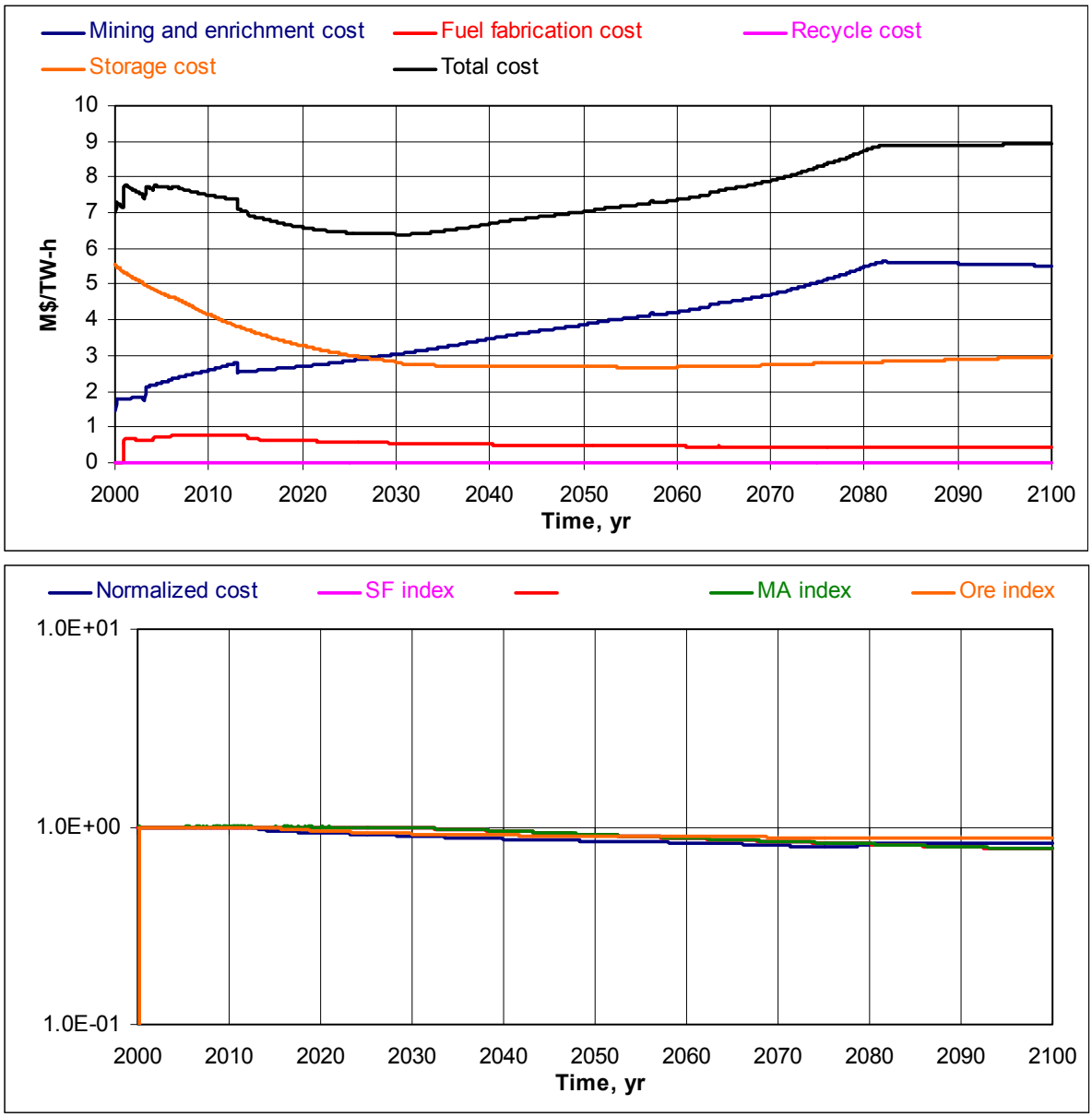
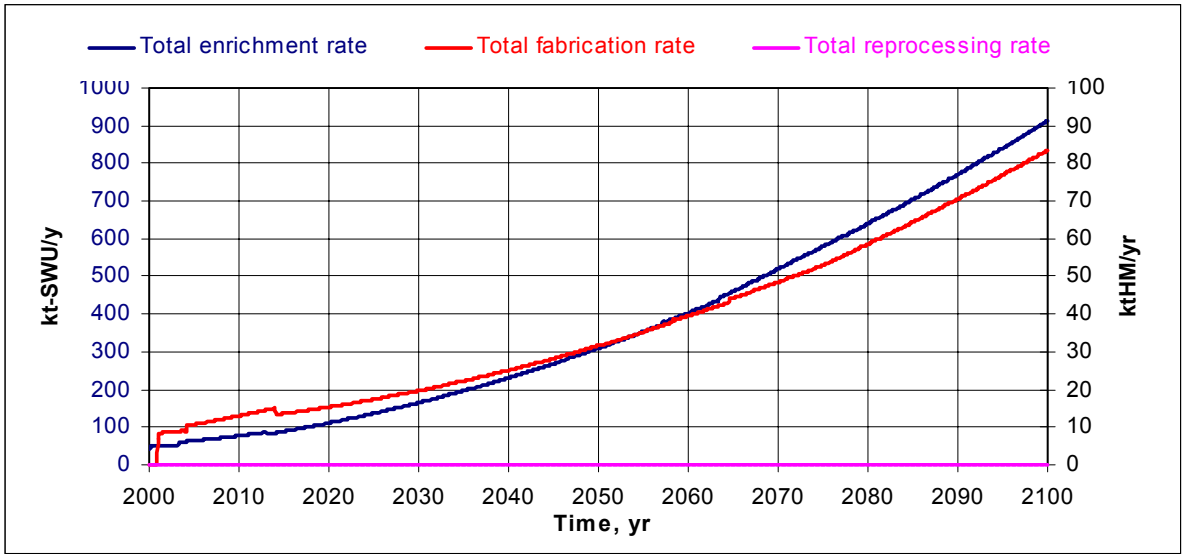
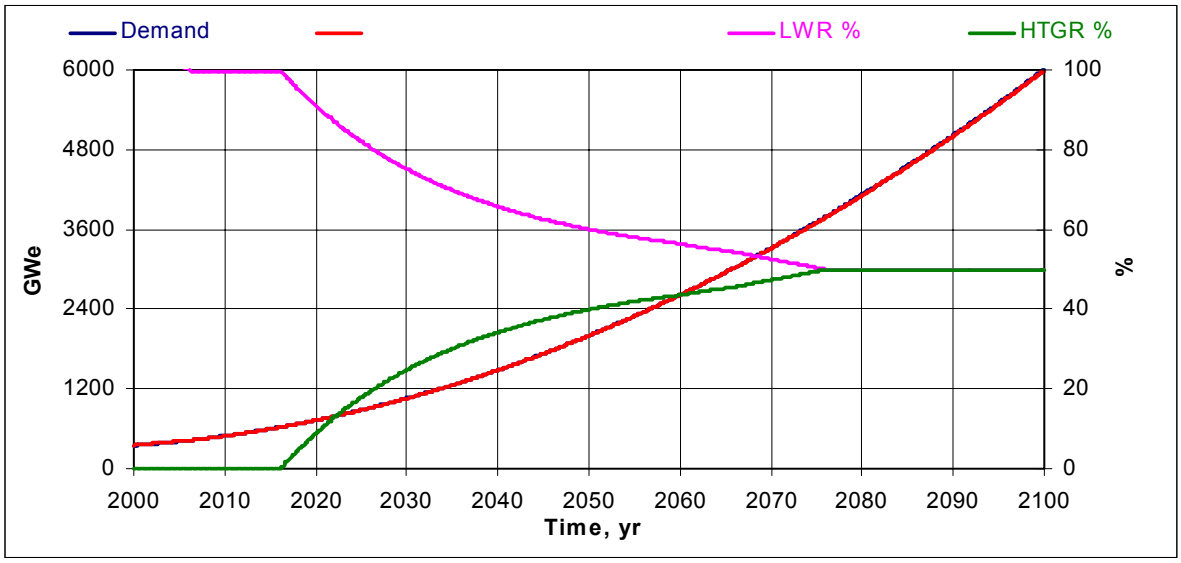


Fig. 9. LWR/PBMR 50/50 Market Share

(3) LWR Once-Thru + 50% HTGR





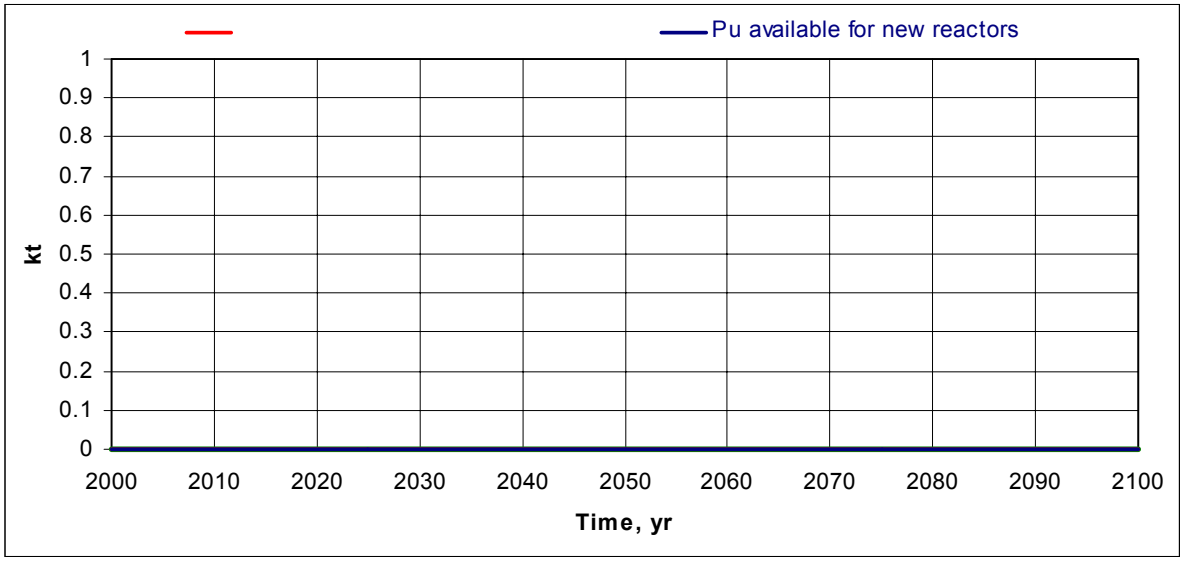
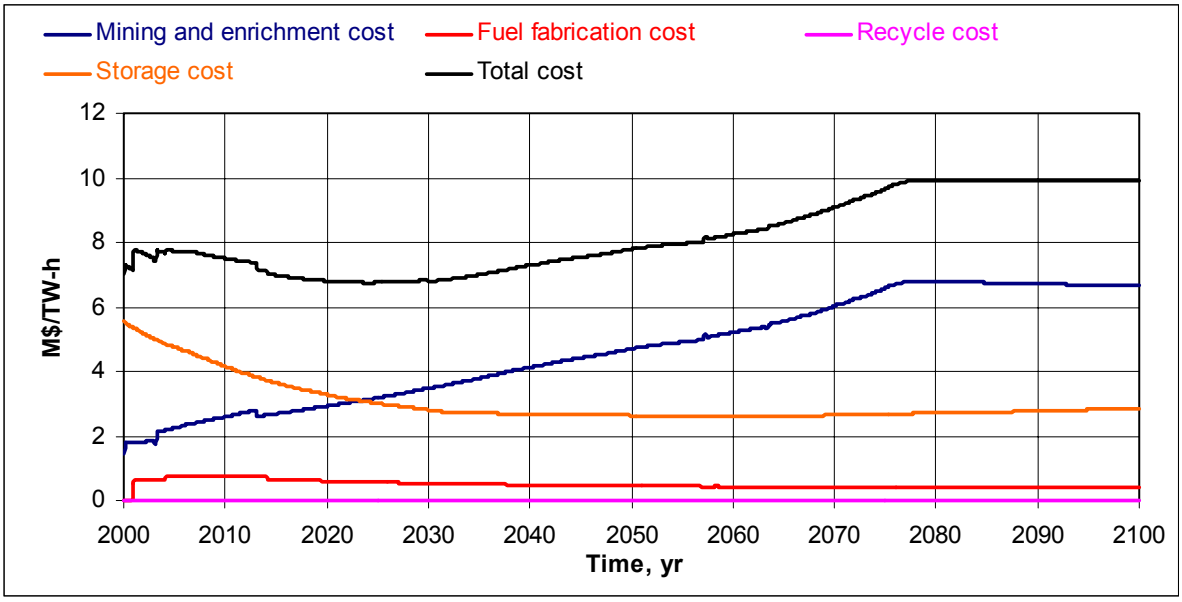


Fig. 10. LWR/HTGR 50/50 Market Share

(3) LWR Once-Thru + 50% HTGR

Fig. 11. LWR/HTGR 50/50 Market Share

(3) LWR Once-Thru + 50% HTGR



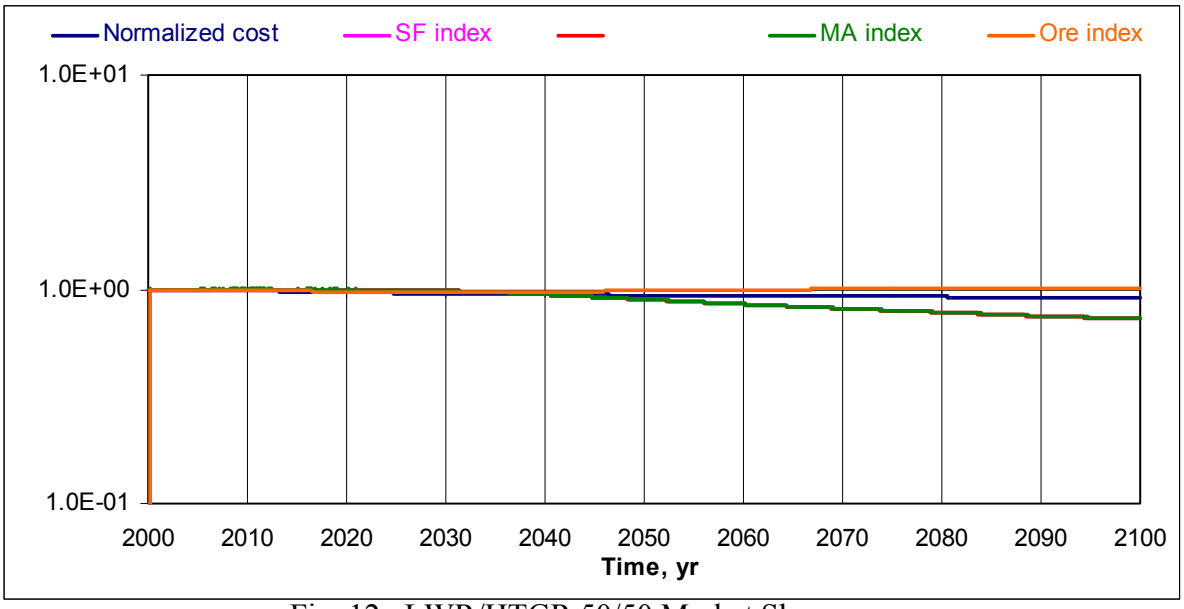
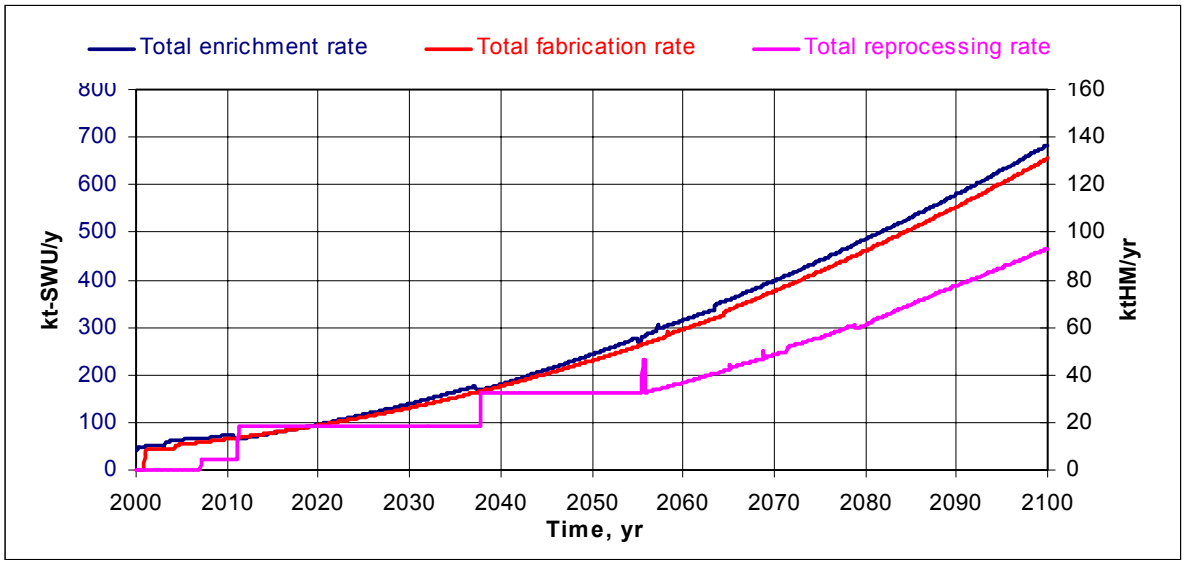
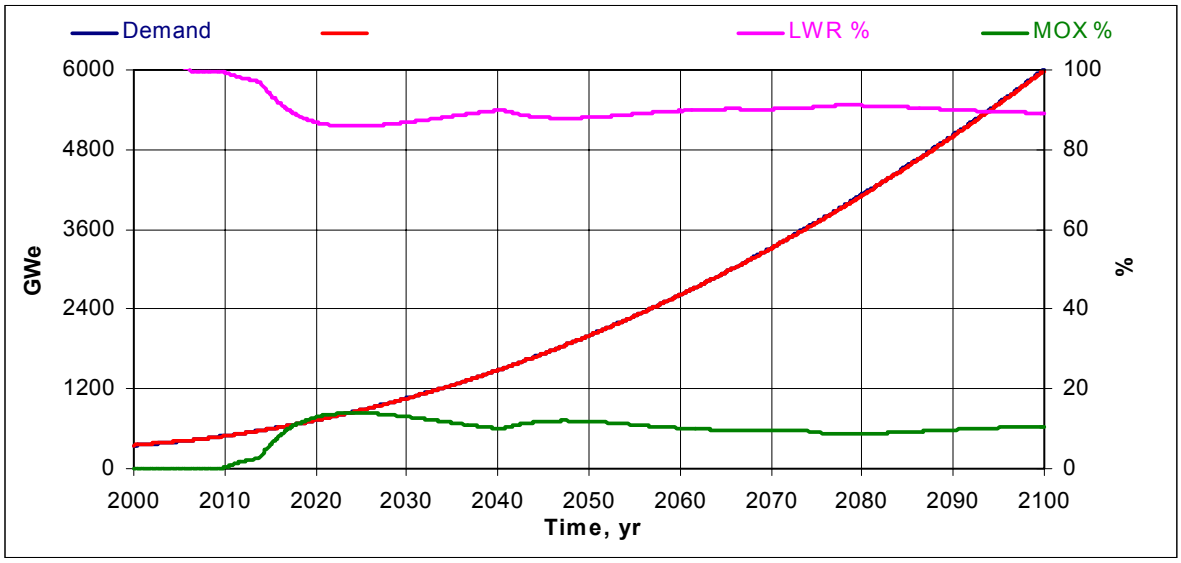


Fig. 12. LWR/HTGR 50/50 Market Share

(4) LWR Once-Thru + LWR MOX





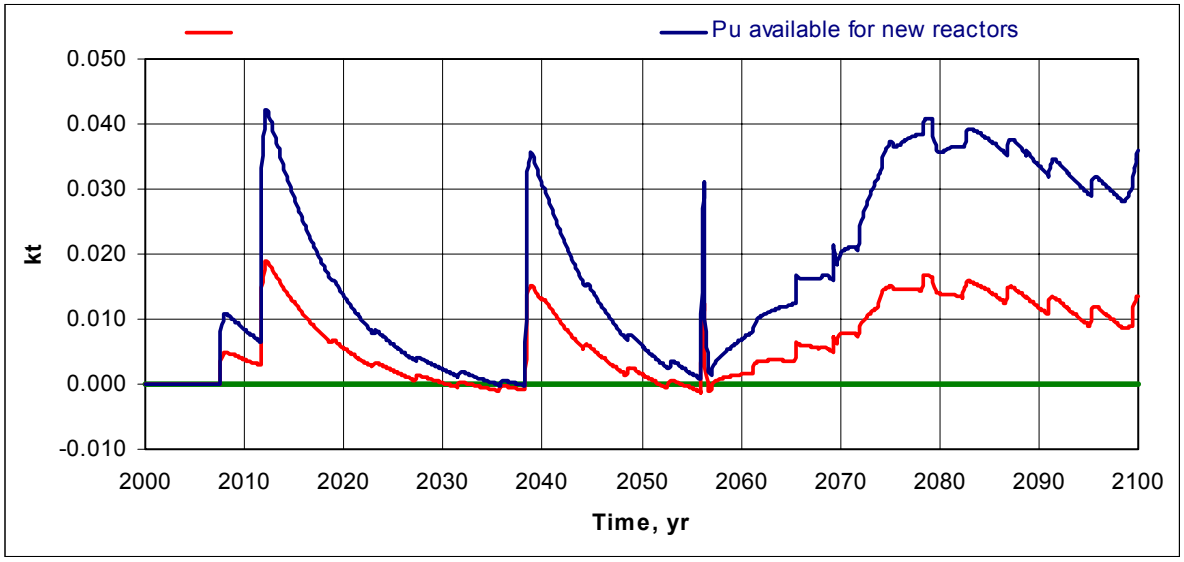
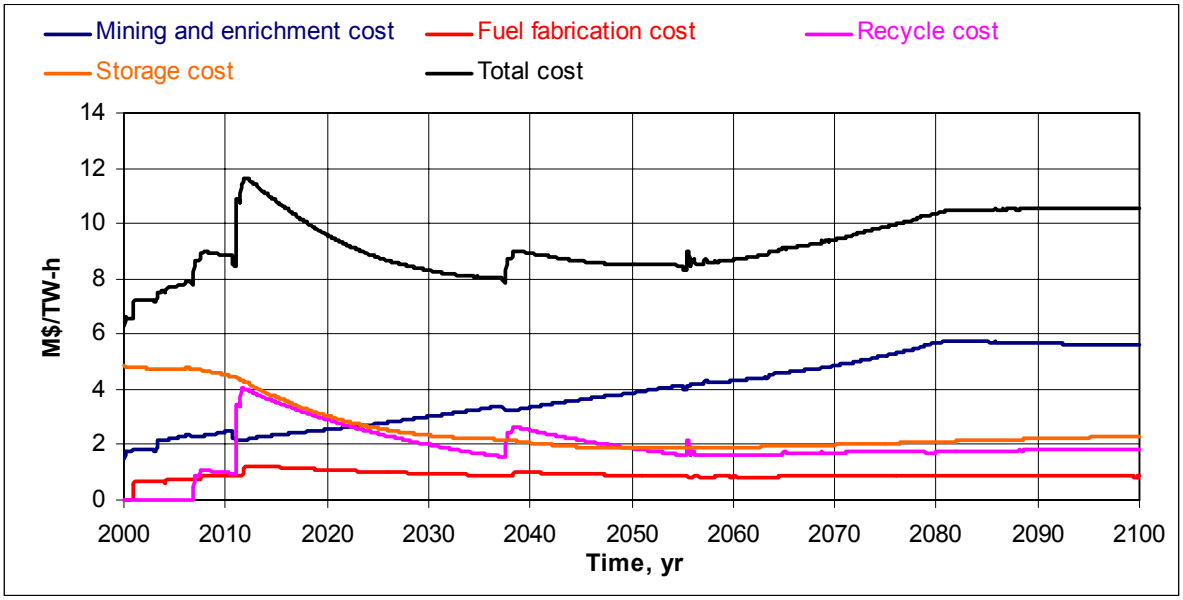


Fig. 13. LWR/MOX-Mon Recycle 50/50 Market Share

(4) LWR Once-Thru + LWR MOX

Fig. 14. LWR/MOX-Mono-Recycle 50/50 Market Share

(4) LWR Once-Thru + LWR MOX



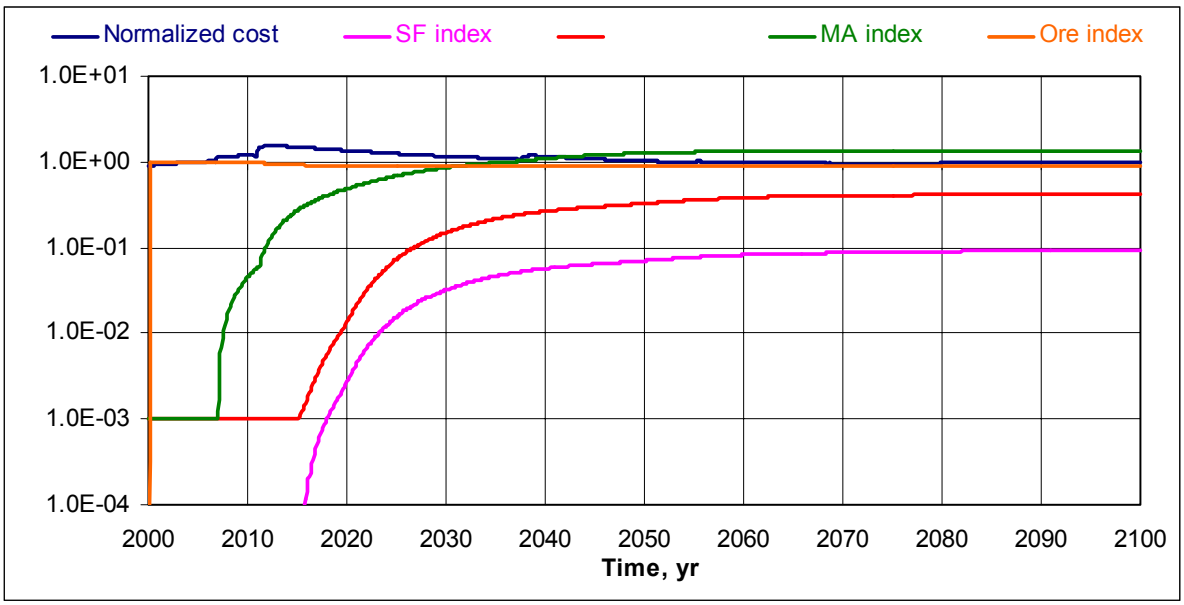


Fig. 15. LWR/MOX-Mono-Recycle 50/50 Market Share

Case #1 – LWR + FR (BR=0.5)

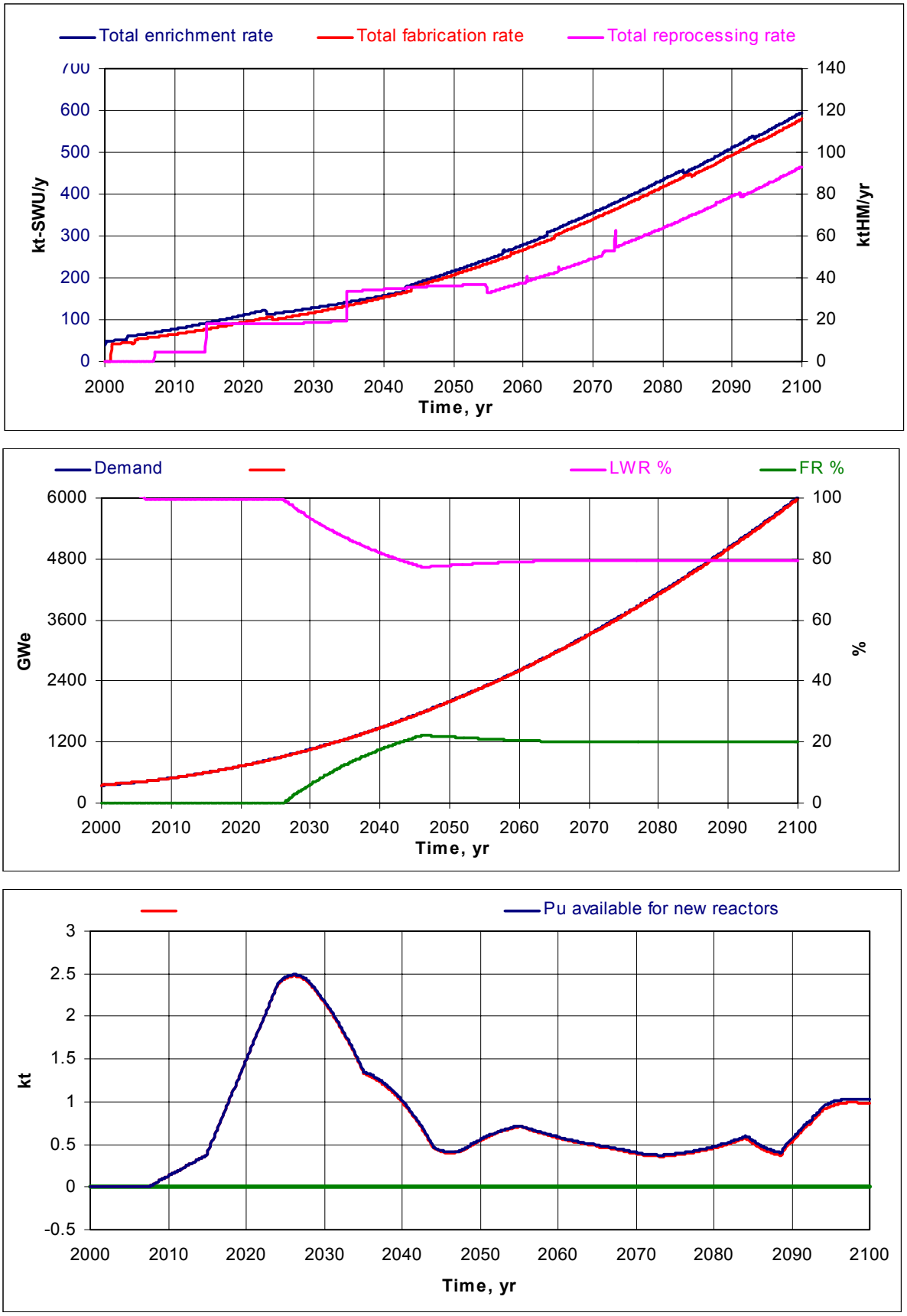


Fig. 16. LWR/FR (BR=0.5) 80/20 Market Share

Case #1 – LWR + FR (BR=0.5)

Fig. 17. LWR/FR (BR=0.5) 80/20 Market Share

Case #1 – LWR + FR (BR=0.5)

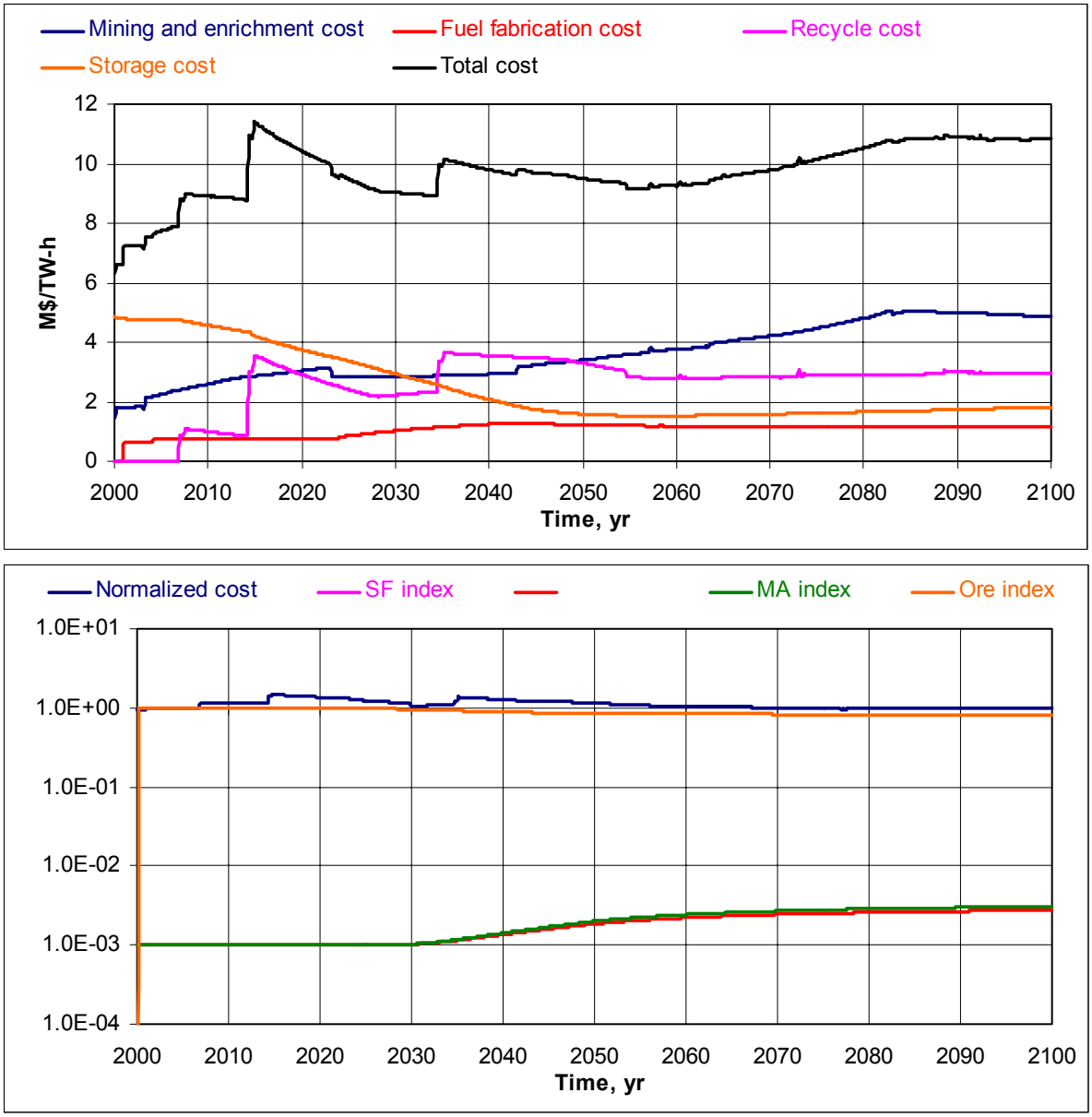
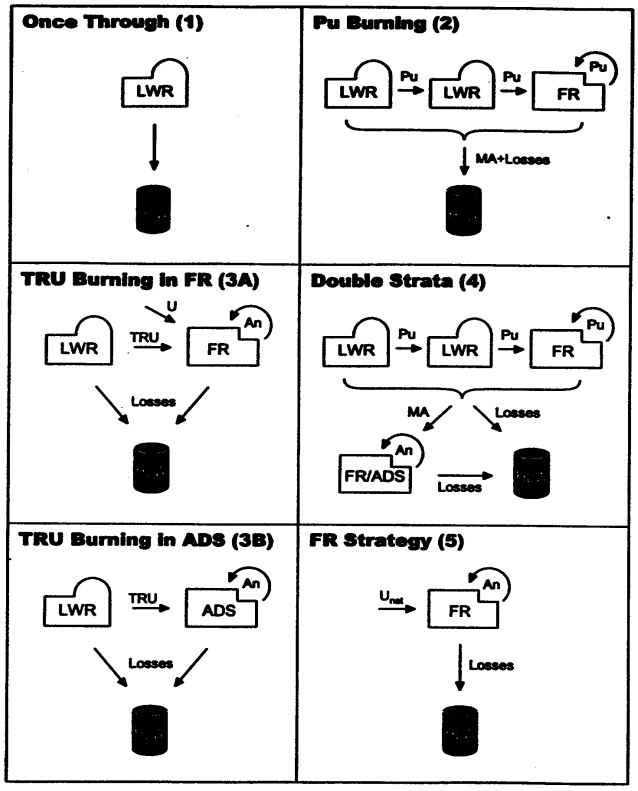


Fig. 18. LWR/FR (BR=0.5) 80/20 Market Share



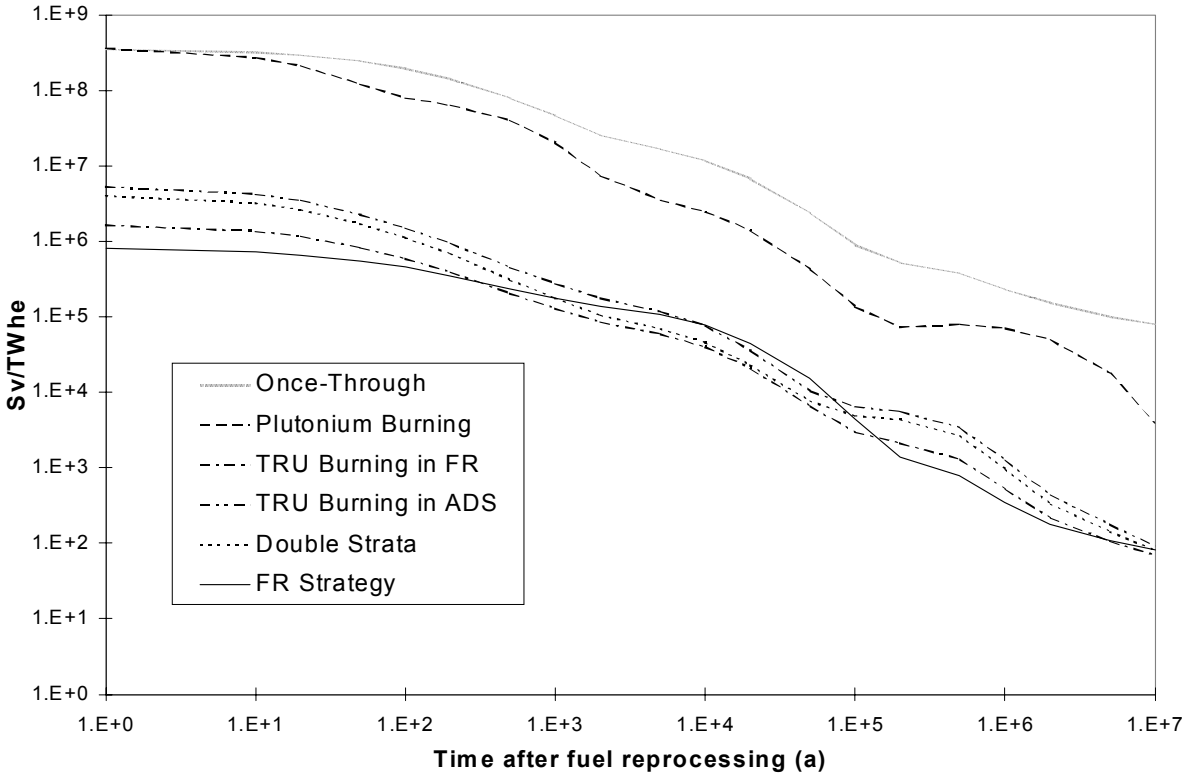
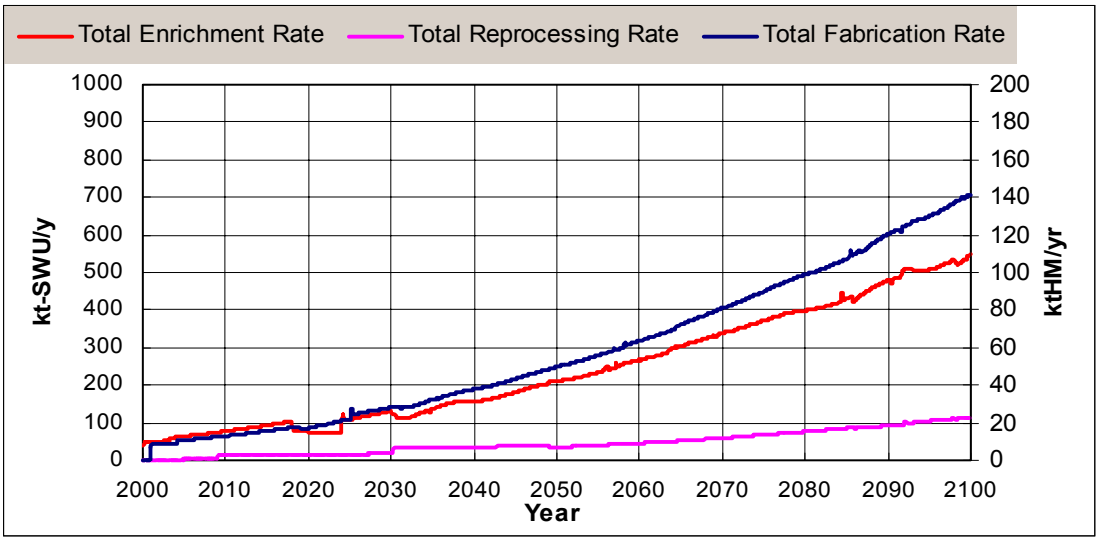
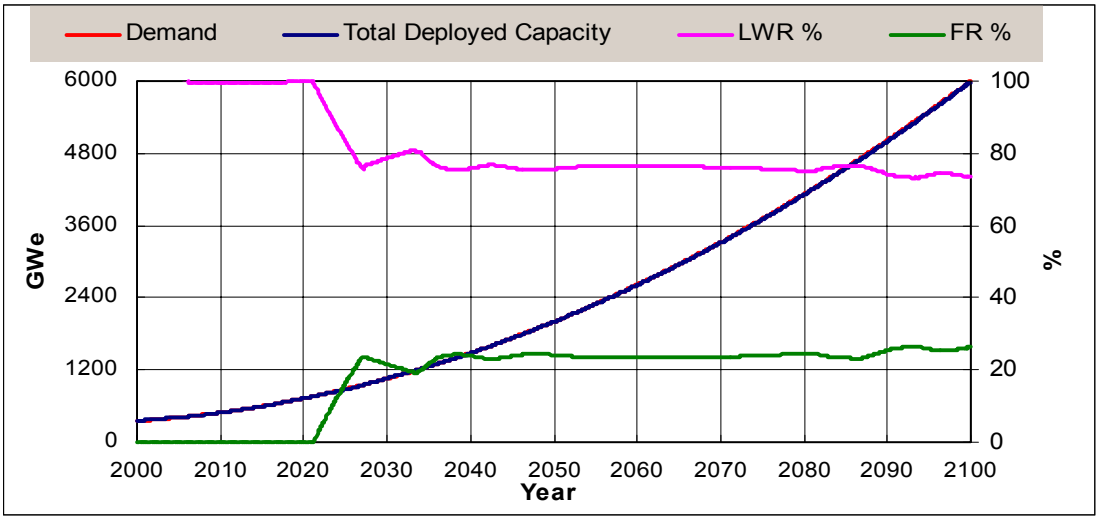


Fig. 19. Full Pu Recycle vs Full TU Recycle: MA Toxicity

LWR + FR(BR = 1)





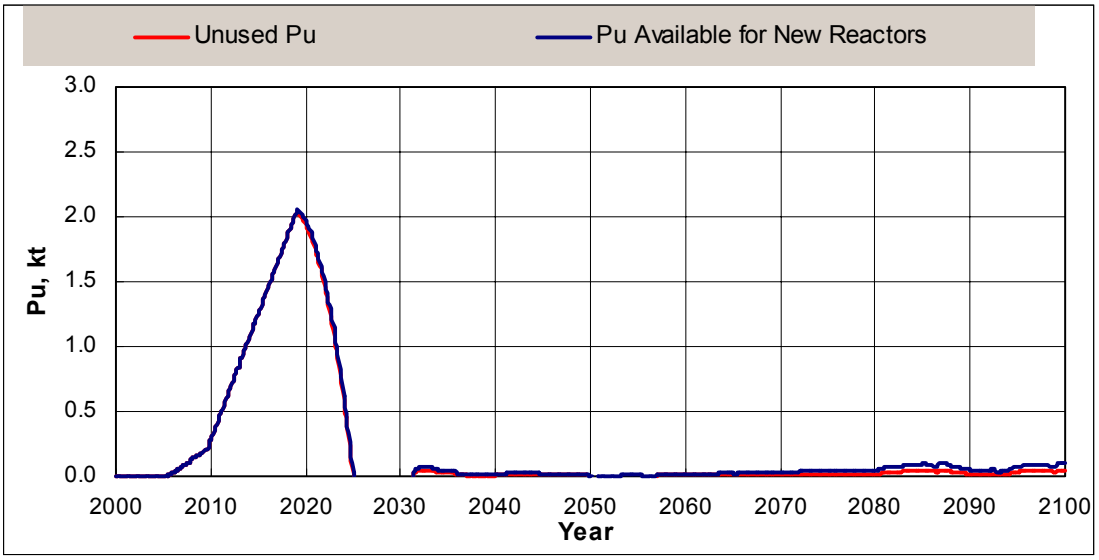
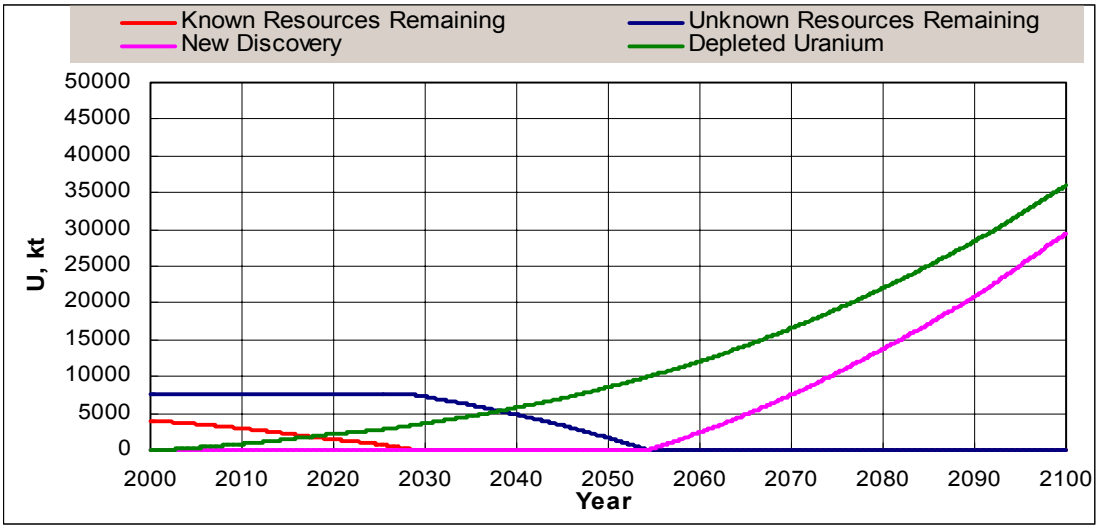
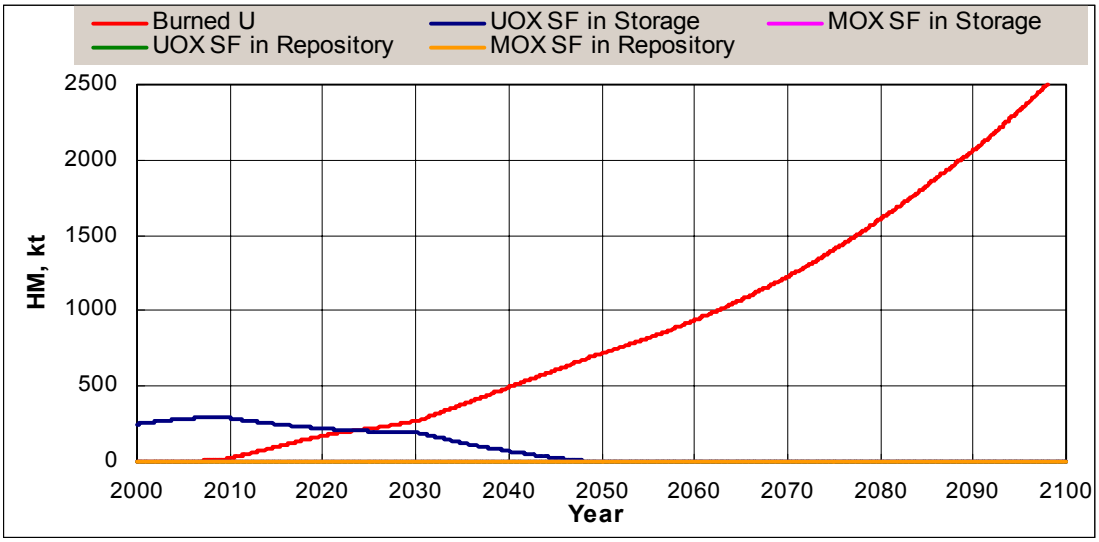


Fig. 20. LWR/FR (BR=1.0) 80/20 Market Share

LWR + FR(BR = 1)





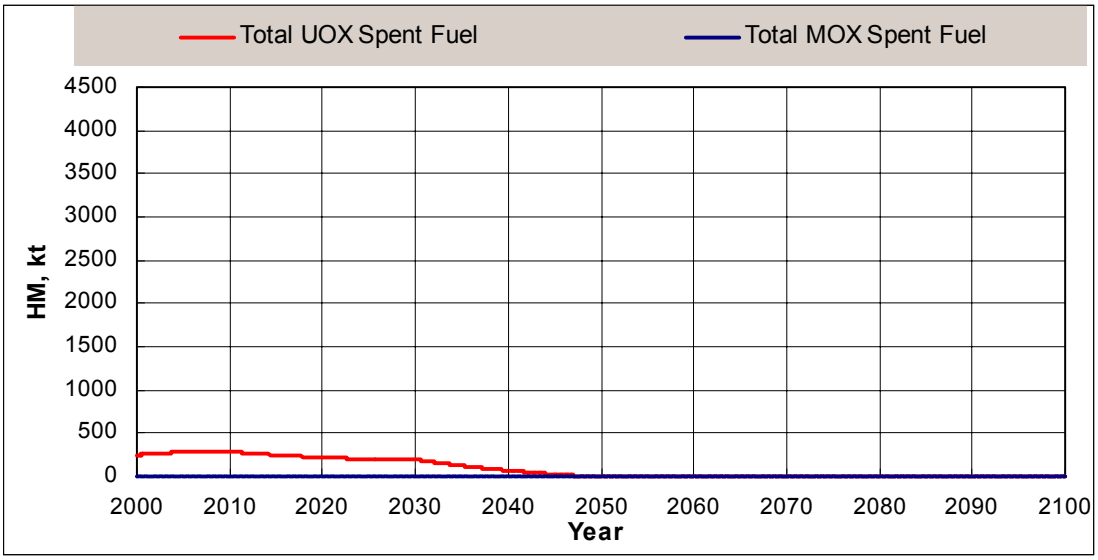
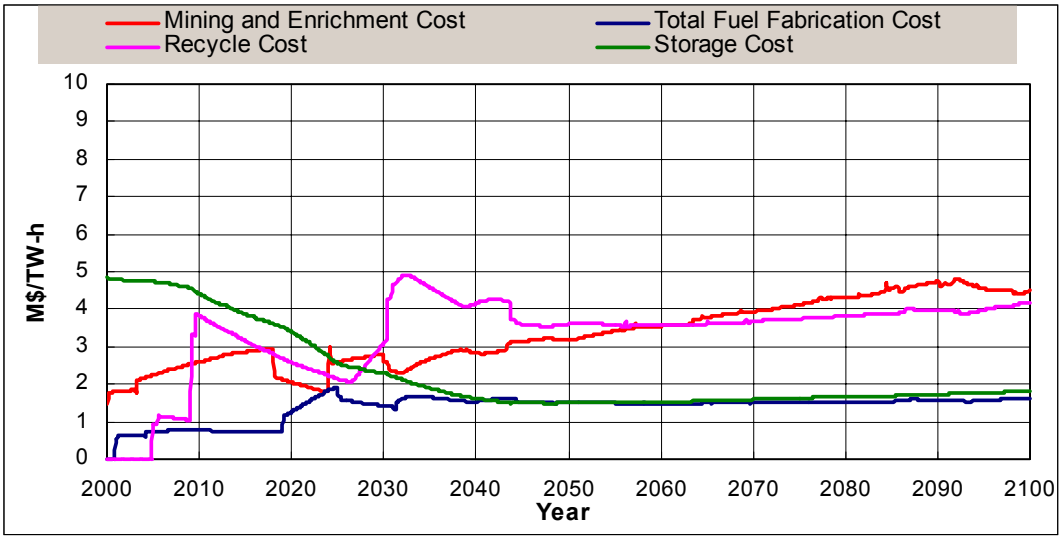
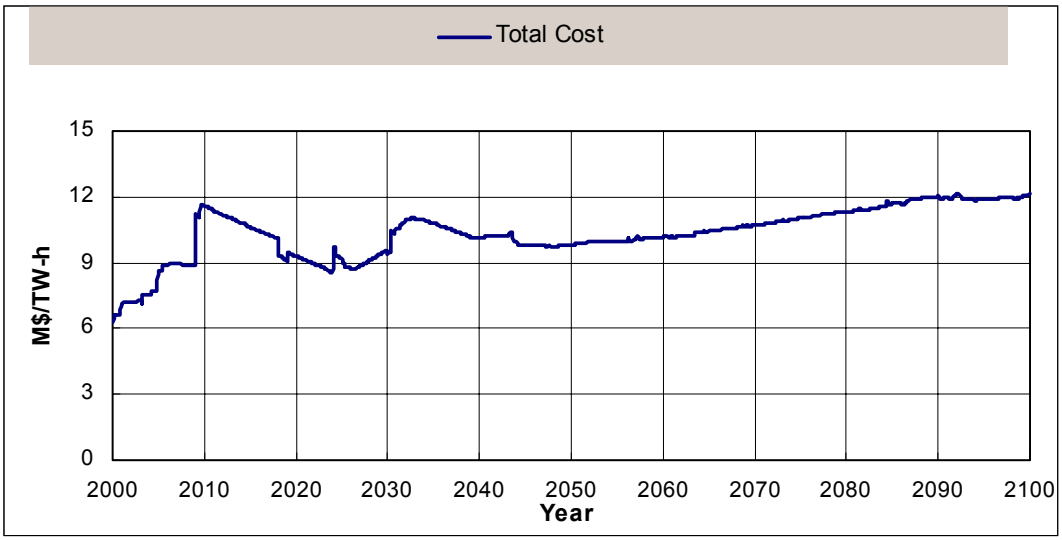


Fig. 21. LWR/FR (BR=1.0) 80/20 Market Share

LWR + FR(BR = 1)





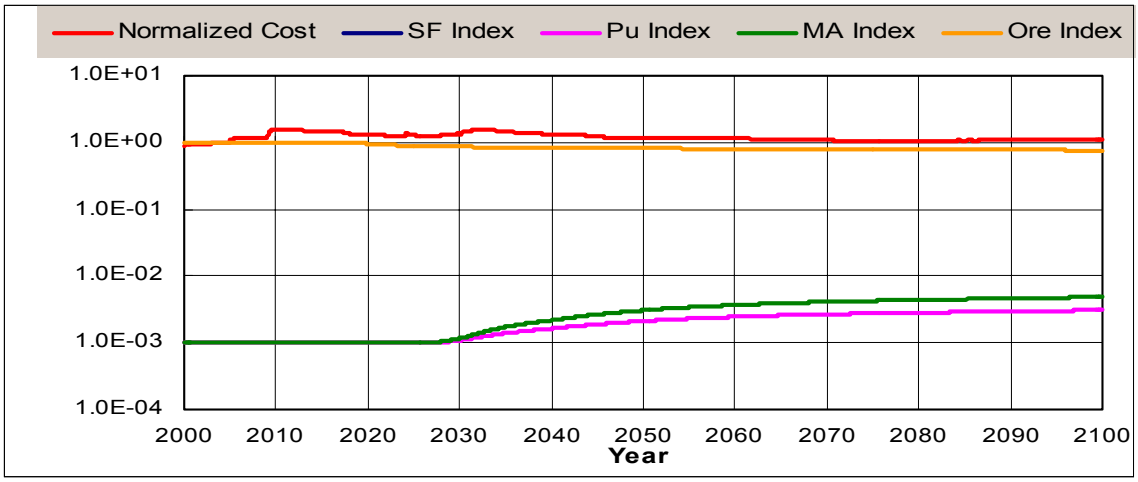


Fig. 22. LWR/FR (BR=1.0) 80/20 Market Share

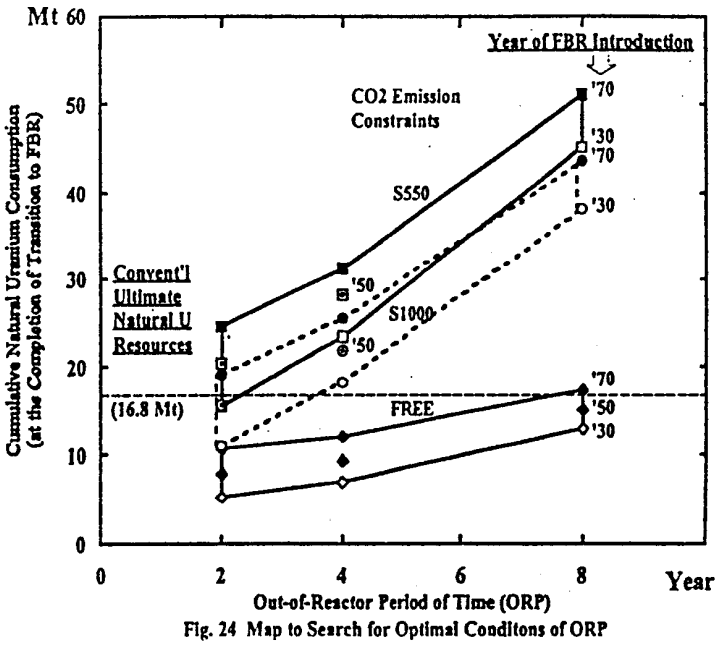
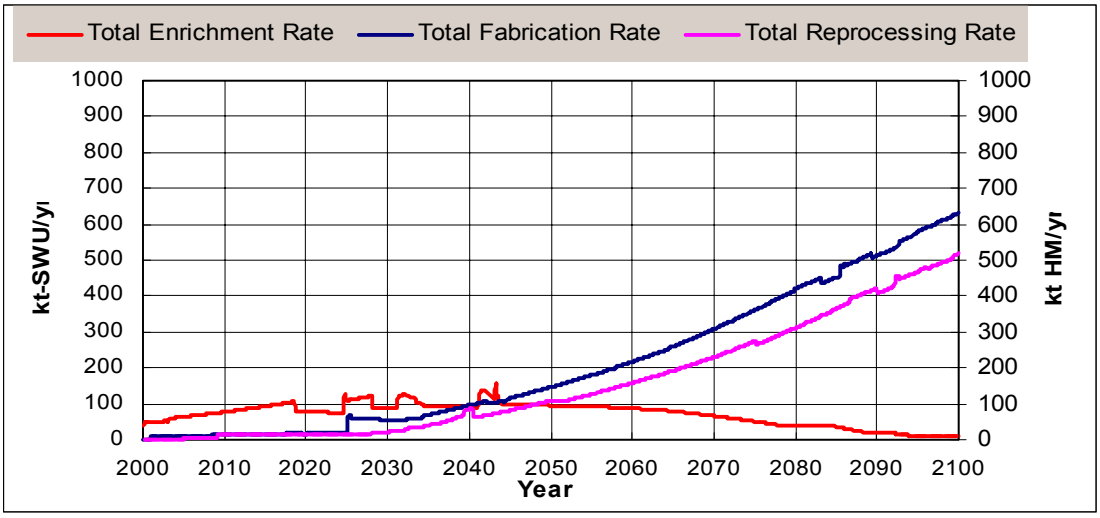


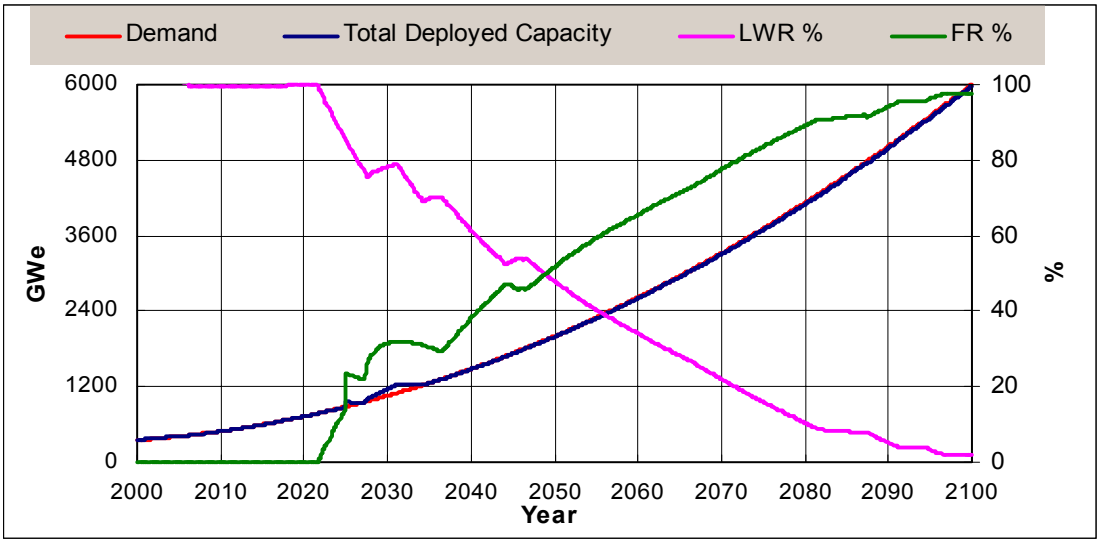
Fig. 24 Map to Search for Optimal Conditons of ORP

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

Fig. 23. Time Window of Opportunity for Transition to Resource Sustainability

LWR + FR(BR = 1, order 2015) + FR(BR = 1.7, order 2025)





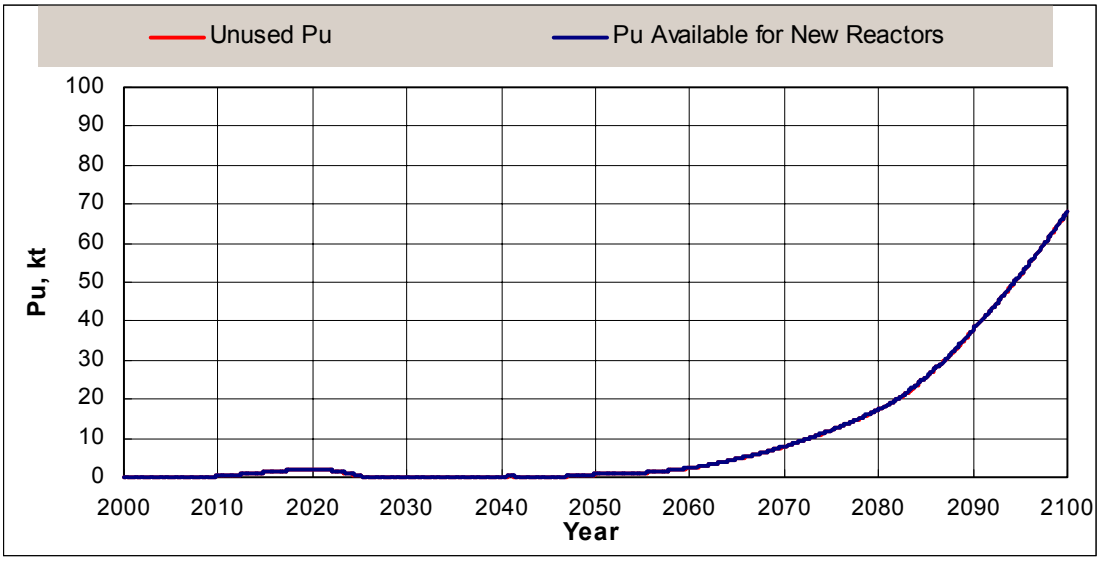
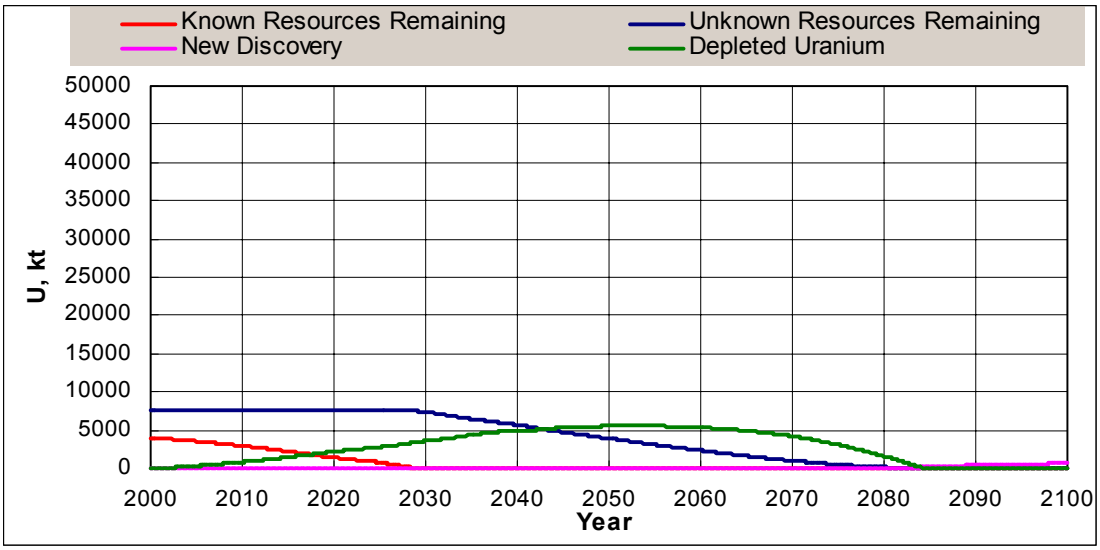
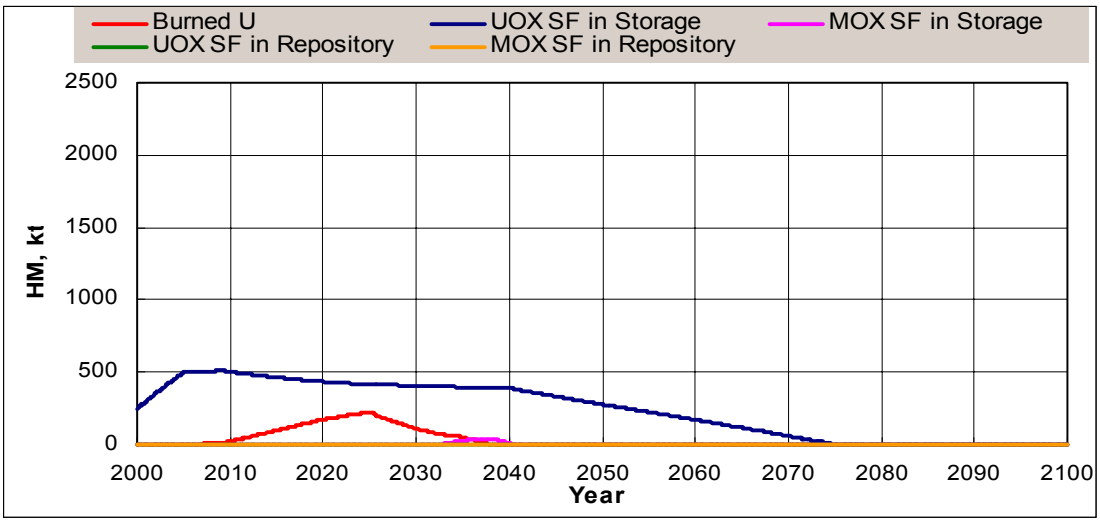


Fig. 24. Burner Switch to Breeding in 2030 – Sustainability Within Redbook Limits On Ore

LWR + FR(BR = 1, order 2015) + FR(BR = 1.7, order 2025)





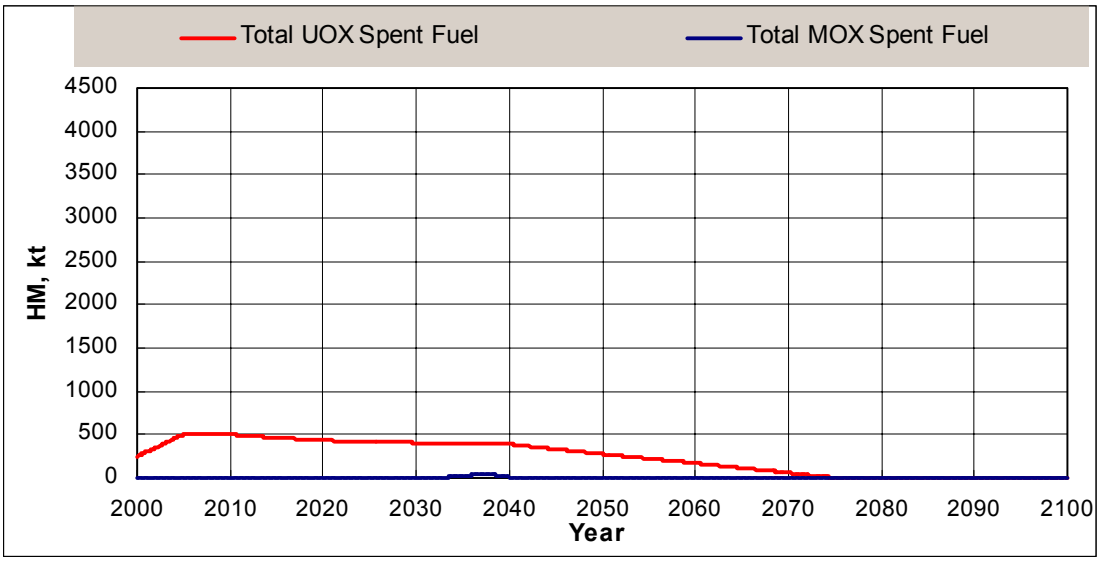
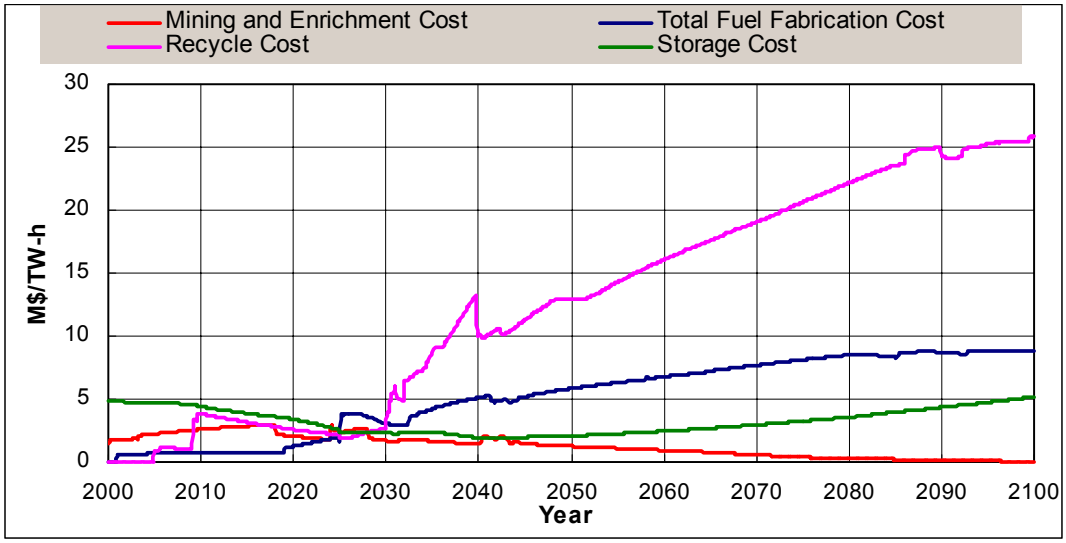
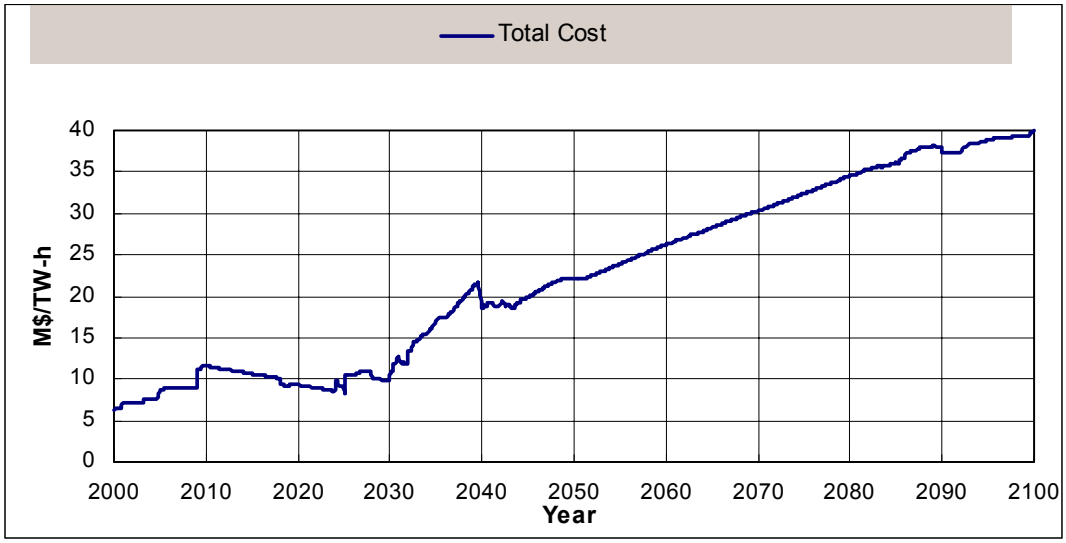


Fig. 25. Burner Switch to Breeding in 2030 – Sustainability Within Redbook Limits of Ore

LWR + FR(BR = 1, order 2015) + FR(BR = 1.7, order 2025)





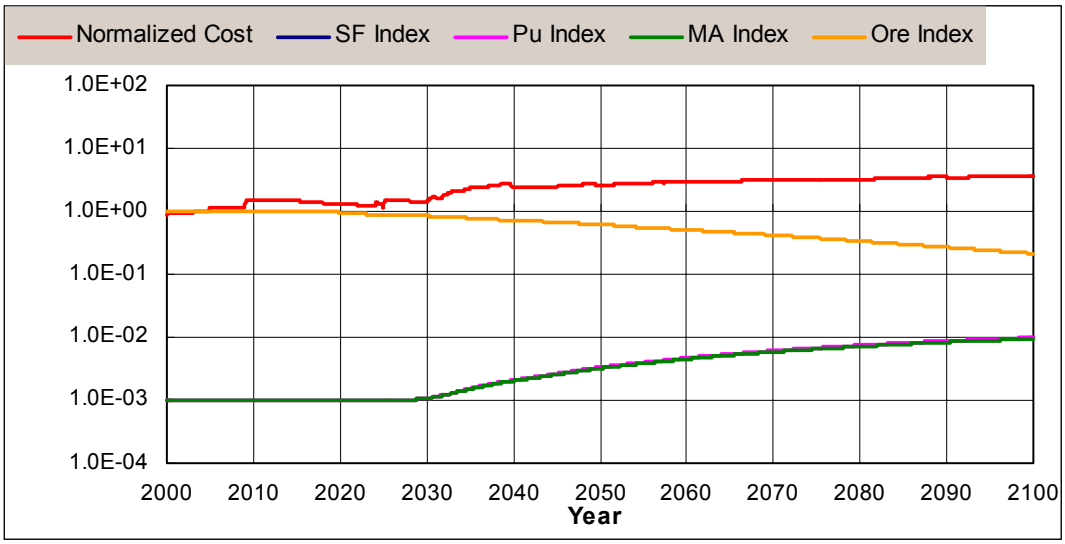


Fig. 26. Burner Switch to Breeding in 2030 – Sustainability Within Redbook Limits On Ore

Case #4 – LWR + FR (BR=1.7) + LWR-MOX

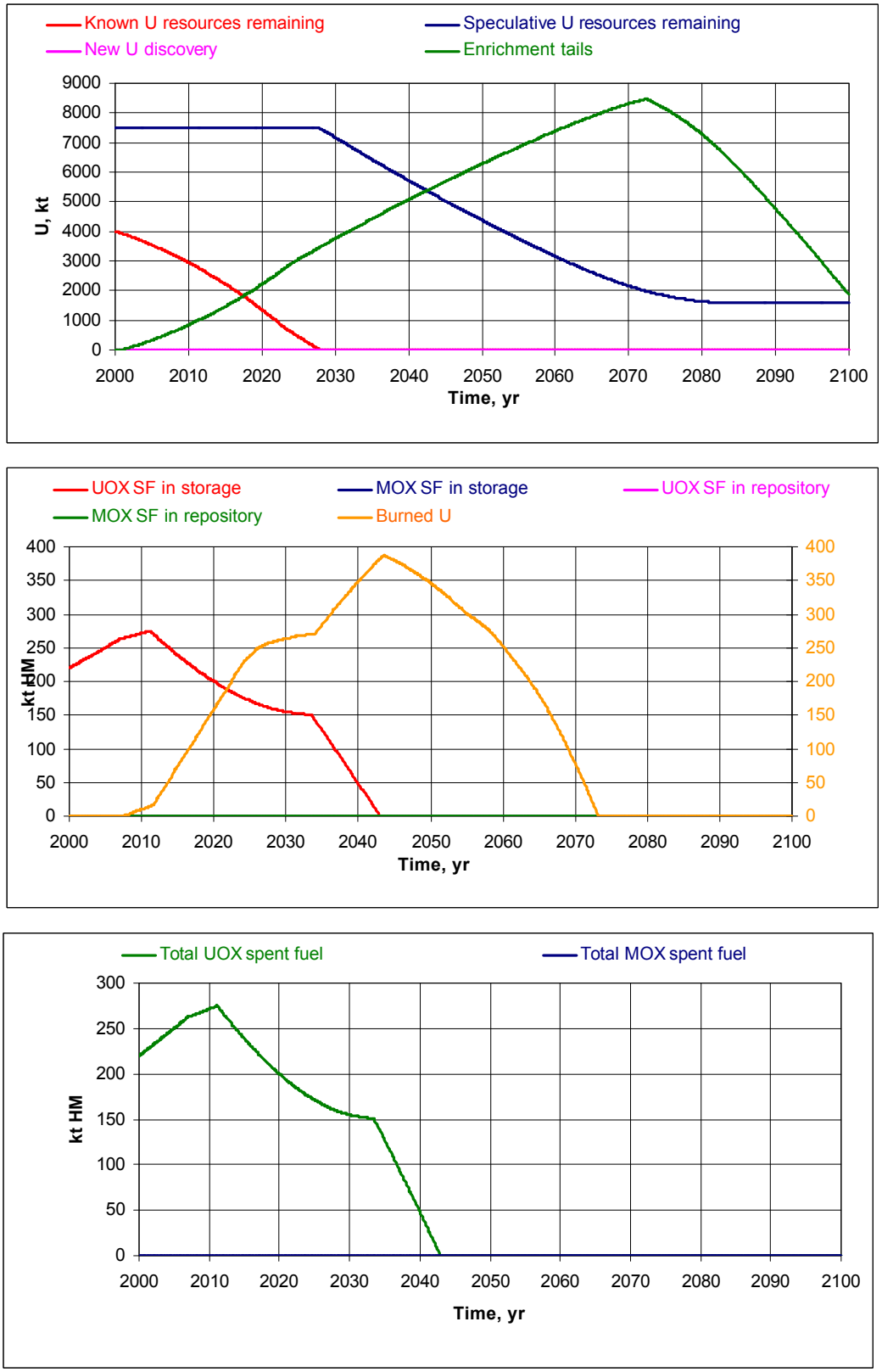


Fig. 27. Penetration Into Nonelectric Energy Sector

Case #4 – LWR + FR (BR=1.7) + LWR-MOX

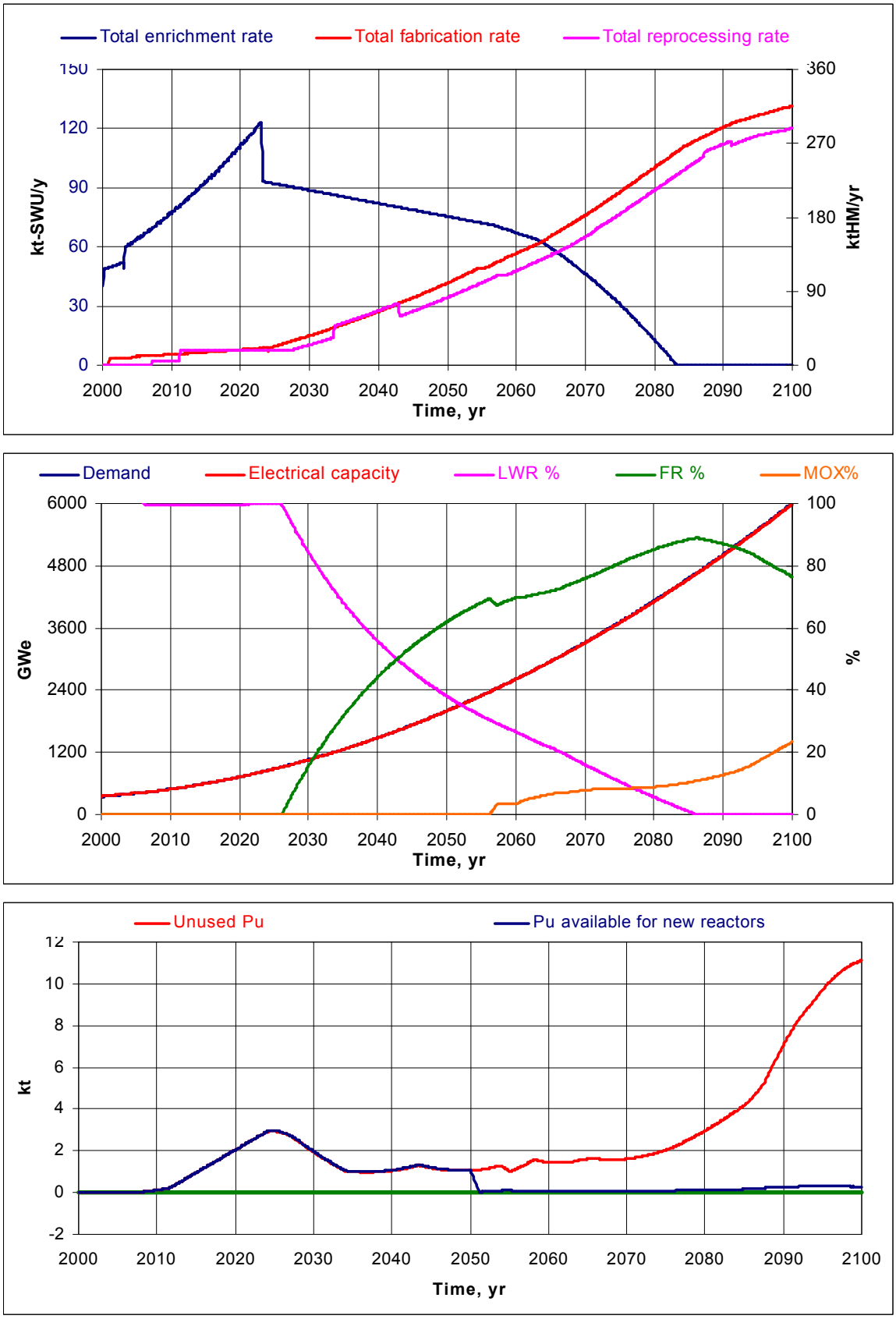


Fig. 28. Penetration Into Nonelectic Energy Sector
Case #4 – LWR + FR (BR=1.7) + LWR-MOX

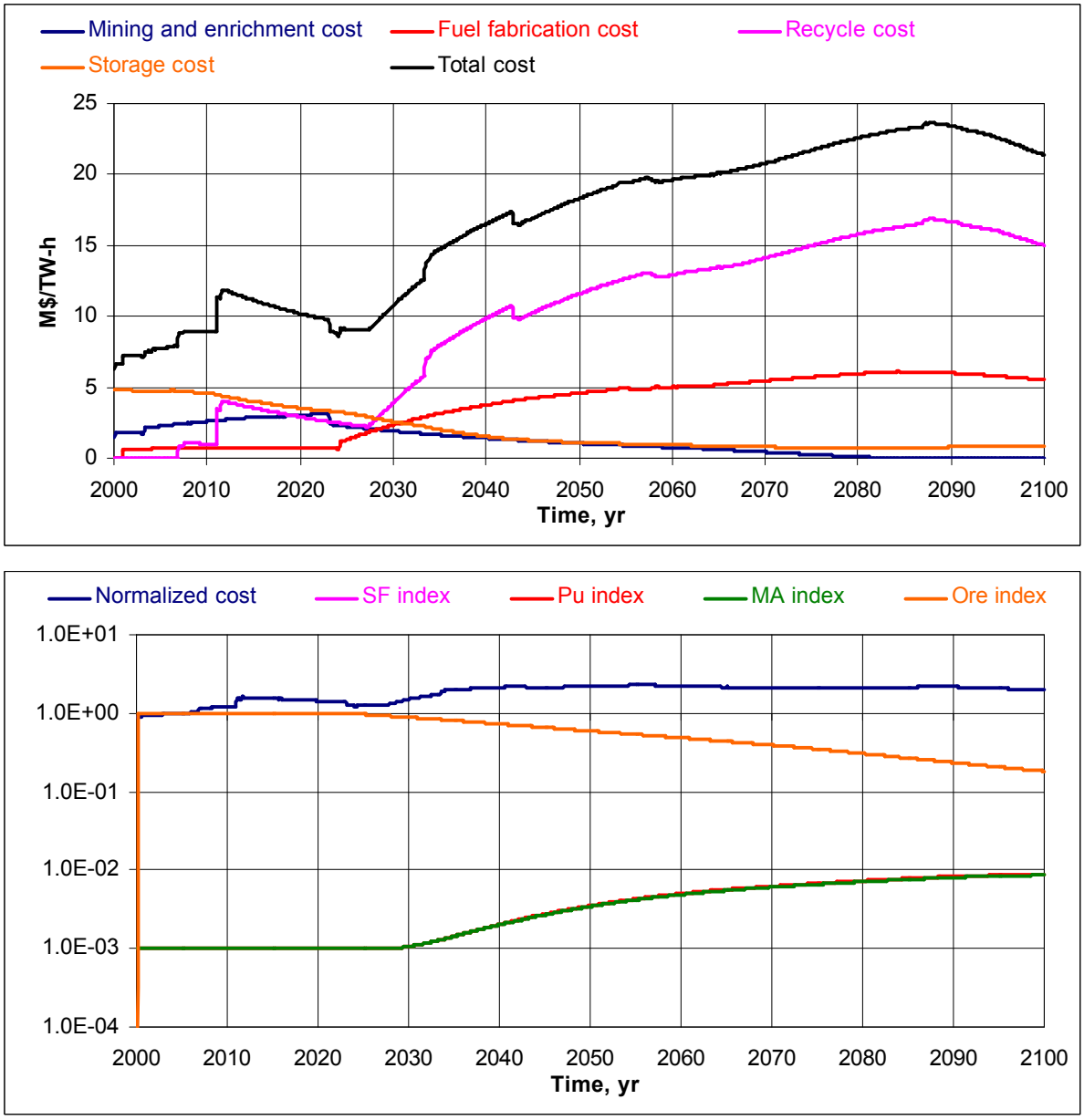


Fig. 29. Penetration Into Nonelectric Energy Sector

Gen IV – FCCG Report

Chapter 4

Fuel Cycle Technology Status and Needed R&D

November 1, 2001

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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 description of the technologies considered in the survey appears in Appendix 4. Although the survey is not complete (i.e., not all relevant technologies are considered), the technologies that are most pertinent to the proposed Generation IV system concepts are addressed. This section summarizes the information that appears in the appendix, with much of the text here being a simple condensation of the more detailed presentation in that appendix. Where possible, specific statements regarding statements of R&D needs drawn from the many submissions compiled in the appendix are presented in the chapter; however, the presentation of R&D needs in this chapter should not be considered exhaustive.

As shown in Figure 4.1-1 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.

The information presented in Appendix 4, and 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 then 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.

Links in the fuel cycle chain

Mining	(Common to All Concepts)
Refining	
Enrichment	
Hands on Fab	(Concept Specific)
Reactor Irradiation	
Purex	
Hands on fab – e.g., MOX	
or	
Dry Recycle	
Remote Fab	
Reactor Irradiation	
Waste Form Production	
Shipping	(Common to All Concepts)
Repository	

Figure 4.1-1. Links in the Fuel Cycle Chain

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 U235. 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, U-235 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 (MOX), $(U,Pu)O_2$, 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

For example, 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 depend, 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, which is a mixture of plutonium and uranium oxides, 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 policy 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 performances and technologies, as discussed in Appendix 4.

R&D Required for Generation IV

The major portion of R&D effort with plutonium 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 transients or accidental 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 particularly which 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 accidental 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, which may have significant impact on economy and performance at all stages of irradiated fuel processing (e.g., storage, transportation, reprocessing, etc.). Continuous, or even enhanced, R&D efforts must be devoted to achieving a 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 in the Federal Republic of Germany and Russia (now also in China and South Africa), and b) the prismatic core 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 fuel 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 fueled 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), and
- TRISO coating with its interlayer of SiC between two layers of high-density isotropic PyC.

Performance of a coated particle for HTRs is defined as the ability 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 behavior.

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 that is applicable for direct cycle applications
- Determine influence of burnup effects on irradiation performance at burnup values above 10% FIMA, in particular for potential reduction of 1600°C capability.
- Analyze accident condition performance at temperature above 1600°C for an improved coated particle model.
- 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 behavior of irradiated coated particles using high precision measurements.

4.3.4 Advanced Fuels and Inert Matrix Fuels

While efficient and well mastered, 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 accomodated in present PWR cores (for example, 30% in today French 900 MWe NPPs). With this constraint (and because of some additional reasons) it is difficult to recycle the plutonium several times in large quantities.

To address those issues, innovative fuels are being investigated. Such fuels include nitride, metal or inert matrix options that may be combined with alternative geometries. To accomplish a strategy to quickly reduce plutonium inventories, removal of uranium from the fuel form allows the maximum possible plutonium-consumption rate, by eliminating conversion of fertile U-238 to plutonium. 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 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 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 the fast reactor's 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 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,

that 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. Viro-compaction (or vi-pac) technology is under development in Russia and offers the promise for simpler fabrication of oxide fuel.

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

Given the extensive experience by 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 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 of 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, 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 operate with metal driver fuel, and this reactor was the test bed for all other fast reactor fuels and materials.

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. As a result metal fuel became a viable alternative to ceramic fuel. In addition, it was discovered that a metal-fueled fast reactor possessed inherent safety features that could not be duplicated with mixed oxide fuel. A key aspect of this is that metal fuel lends itself to proliferation-resistant remote reprocessing and refabrication by the non-aqueous pyro-metallurgical process.

R&D Required for Generation IV

Knowledge of metal fuel is sufficiently mature that relatively little remains 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 desirable 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 that gained from the few 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 same desirable characteristics as metal fuel; i.e., high heavy metal density, good thermal conductivity, and excellent compatibility with sodium. Second, testing of nitride fuel for space power applications did not reveal any undesirable characteristics. But, 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 N15 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 does appear to have unique safety characteristics that are superior to that of either mixed oxide fuel or metal fuel. And for selected high temperature applications such as space power, nitride fuel has unique advantages, 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 more easily reprocessed by aqueous techniques; therefore these systems, primarily the PUREX process, are more fully developed and implemented commercially. Pyroprocess systems have largely been associated with fast reactors and metallic fuels and their development has therefore reached only the pilot-scale stage, and the feasibility of minor actinide (MA) separation 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 the possibility to form an integrated system between irradiation reprocessing and (re)fabrication facility, thus reducing considerably transport of nuclear materials. Pyrochemistry is to be preferred in particular for advanced oxide fuels (mixed transuranium, inert matrix or composite) and metal fuels, but also nitride fuels. Compared with aqueous methods, dry reprocessing of fuels results in less pure and thus maybe more proliferation-resistant fractions 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 MA fuel; this allows reprocessing after shorter cooling times, which reduces storage costs. Furthermore, due to the absence of water (neutron moderator) in the process the criticality hazard is lower; i.e. the inherent safety is increased.

Finally, it should be noted that reprocessing technology development is often directed toward meeting two important objectives (which are 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 (in terms used by some of the international participants) “dirty fuel” and “clean waste”. Dirty fuel would be 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 additional

reduction of transuranic materials that would otherwise be disposed in waste streams. Clean wastes would be 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 reprocessing 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 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 improve on the PUREX standard to improve economics of reprocessing and provide better-manageable waste streams.

R&D Required for Generation IV for Aqueous Processing

Table 4.5-1 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 guarantee its performance. The conclusion of this step corresponds to establishing the technical feasibility of the process.
- Phase 3 relates to the industrialisation of the process, terminating with the establishing of industrial feasibility of the process and its potential application in an industrial installation. It is aimed to ensure overall active operations in industrial conditions. In practice, these studies are essentially conducted by engineering design.

Table 4.5-1 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 oxidised 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.

For all the above-mentioned processes the development of new extractant molecules and the improvement of existing ones is 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 in India the examination of sulfoxide type extractants have to be mentioned.

In view of process industrialisation, the economics and the radiation resistance of the organic molecules are important R&D issue. 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 preconcentration 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 genuine high active wastes originated from real spent nuclear fuels.

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

Advanced aqueous recycle is the technology that recycles TRUs in the closed system. In addition, if it is required to transmute fission products in the nuclear fuel cycle using the surplus neutron in fast reactor core, as does the SCNES concept, the separation and recovery function for FPs, (e.g., electrolyzing Tc from waste solution), will be developed and incorporated into the process.

The future R&D tasks necessary to develop advanced aqueous reprocessing technology are the following:

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

Furthermore, the safeguards optimization of blanket fuel management in the fast reactor must be attempted, due to the high Pu-239 quantities that are produced.

4.5.3 Pyrochemical Processing Methods

Since early in the development of the nuclear industry, a number of 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.

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 would certainly 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.

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, various 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 Russian 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.

R&D Required for Generation IV for Pyroprocess Development in the U.S.

The Spent Fuel Treatment Program at ANL demonstrated many parts of the pyroprocess fuel cycle, 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. The remote fabrication of IFR fuel was not part of the Spent Fuel Treatment Program, but the same technology was used to fabricate cold fuel for EBR-II, and a demonstration of another pyroprocess (melt refining) for recycling EBR-II in the 1960s employed remote injection casting fabrication for 34,500 fuel elements.

One challenge for a pyroprocessing system is selecting the appropriate materials of construction for the high temperature processes. Material improvements are needed in order to lessen the formation of dross streams and increase material recovery and throughput.

The quantity of waste generated from pyroprocessing that requires 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 Pyroprocess Development in Russia

Research activities today (potentiometry, voltammetry, etc.) focus on the behaviour 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.

R&D Required for Generation IV for Oxide Reduction Head-end Processing

Further work is required in a number of areas related to process design and equipment. Actual irradiated fuel should be used to verify the behavior of the TRUs and fission products. In addition, much 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.

4.5.4 Other Processes

Other pyrochemical processes that are 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 function. 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 categorised 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 unwanted fission products, fuel cladding, etc., that would constitute the primary fuel cycle waste from a PUREX recycle scheme, there would be generation of secondary waste. This secondary waste would comprise all insoluble active residues, degraded solvents or salts, ancillary materials, and analytical wastes, etc. that arise during additional fuel cycle operation. It is suggested that

the majority of secondary wastes, which arise during processing operations would be generated during solvent/salt cleanup and recovery operations.

Pyrochemical processing operations tend to produce little secondary waste because there is little or no reagent degradation. If the processes are properly designed and operated, high-level waste volumes can be minimised 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 in this study 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 this depleted uranium and recovered irradiated uranium may become a more apparent issue in the future if such recycle schemes would be deployed.

Today, the strategy for the long-term management of depleted uranium is based on the consideration that this depleted uranium is a valuable material, which may have various applications, and is not considered a waste. The use of this depleted uranium in fast reactor systems, as shown in this study, is one of the applications where the 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 form. The reprocessed uranium distinguished itself from natural uranium by the occurrence of higher amounts of γ-emitting isotopes. 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.

For as long as LWRs would make up a significant part of the nuclear power plant park in future fuel cycle schemes, an increasing amount of depleted and revovered irradiated uranium would emerge. In particular, the latter would be of increasing importance as depleted uranium would be 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 all FR scenarios can provide a steady decrease

of the build-up inventory of depleted uranium and the reprocessed irradiated uranium. In the very long term (about 1 million years), disposition of this depleted and recovered irradiated uranium results in a activity and radiotoxicity level comparable to natural uranium as the natural decay chains reach secular equilibrium.

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, seem to offer potential for reprocessing of advanced fuels due to their good compatibility with most fuel forms and their high radiation resistance. In addition, the 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 not be considered as competing but, rather, as complementary technologies. In a double-strata concept for instance, aqueous processes could assure the recycle of 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. There are major differences in waste quantities, characteristics, and costs for the four generic fuel cycle types. The major waste streams are from the front-end of the fuel cycle (mining and milling) and the back-end. For 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 characteristics 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 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 carbon dioxide, sulfur dioxide, etc. emissions; renewable sources, requiring energy and materials inputs for manufacturing, use of heavy metals in solar cells, required land use, etc.; and nuclear), the scale of operation can alter the preferred waste management strategy. If nuclear power is used on a truly 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.

There are three approaches to waste management that 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 occuring, thermally 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 fissile) 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 behaviour 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 modify 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 realistcally 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 development of nuclear energy.

4.8 Non-proliferation and Fuel Cycle Technology

Gen-4 sustainability goals include improved fuel cycle resistance to proliferation 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 herein.

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

There are three basic fuel cycle safeguards strategies:

- *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) better 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 in terms of use for manufacturing nuclear weapons, (3) no off-site fuel-cycle facilities [including the repository] beyond the reactor with weapons-usable materials, and (4) limited 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 convert the ^{233}U to non-weapons-usable ^{233}U. Some plutonium is produced from the ^{238}U; but the reactors are designed to minimize the inventory of plutonium.

The Gen-4 Roadmap development requires 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, this requires that the Gen-4 Roadmap process must develop an ability to compare very different systems. Because the philosophy and basis for proliferation resistance is different for each fuel cycle, it is difficult to make cross comparisons.

Several examples can illustrate the issues. The IAEA has carefully defined weapons usable ^{235}U as uranium containing 20% or more ^{235}U in ^{238}U. 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, these existing definitions are not sufficient for advanced systems. New fuel cycles imply different combinations of fissile materials. Several examples can clarify this:

Plutonium Isotopics

The usability of fissile materials in nuclear weapons depends upon the isotopics. Plutonium isotopics depend upon the fuel cycle. Table 4.8.1 shows isotopics for weapons-grade plutonium, PWR plutonium, and a proliferation resistant molten salt reactor. Fast reactor plutonium would have an isotopic composition between weapons-grade plutonium and PWR plutonium. The IAEA defines the quantity of plutonium necessary to manufacture one weapon as 8 kg. When this definition was developed, power reactors had very low burnup fuels and thus the plutonium was almost entirely ^{239}Pu. *Different nuclear isotopes have different nuclear properties.* If 8 kg of ^{239}Pu is required to build a weapon, a significantly larger, but currently undefined, quantity of plutonium would be required for different isotopic mixtures.

Table 4.8.1 Plutonium Isotopics (%)

Isotope	Weapons grade	LWR[a] (PWR)	Denatured MSR
^{239}Pu	38	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.

With some high-burnup fuels (high temperature gas cooled reactors and molten salt reactors), the dominate 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.

Higher Actinides

Many advanced fuel cycles propose to recycle and burn neptunium, americium, and curium. No official definitions of significant weapons-usable quantities of these materials exist.

Thorium Cycle

There is no legal definition of weapons-usable or non-weapons-usable ^{233}U; 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.

The ^{233}U-^{232}Th fuel cycles have a unique characteristic similar to that of minor actinide recycle – the recycle feedstock is highly radioactive and difficult to access. Unless very special production techniques are used, the production of ^{233}U results in secondary production of ^{232}U, which decays through several decay products to thallium-208 (^{208}Tl), which then emits a 2.6- MeV gamma ray. Consequently, the radiation field associated with ^{233}U increases with time. Within a decade the radiation levels from an 8-kg sphere of reactor grade ^{233}U will be many 10s of R/h at 1 m and hundreds of R/h at 1 ft. *Eight kilograms* is the IAEA equivalent definition of the quantity of ^{233}U needed to produce a weapon. For highly-irradiated ^{233}U with a high ^{232}U content, these radiation levels become significant barriers against use in nuclear weapons or purification of ^{233}U by isotopic separation from other uranium isotopes. Only isotopic separations can separate the ^{232}U from the ^{233}U and permanently eliminate the radiation hazard. This is a much more difficult separation than separating ^{235}U from natural uranium because of the small difference in mass and the high radioactivity. Note: this proliferation barrier is separate and distinct from isotopic dilution of ^{233}U with ^{238}U.

R&D Needed for Advanced Fuel Cycle Nonproliferation Strategies

Proliferation resistance of fuel cycles can be 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. But if fuel cycle and reactor designers are to improve "proliferation resistance", they must have a basis to judge proliferation risks and understand which materials present the greatest risks and what barriers present the more cost effective resistance. 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.*

[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).

Gen-4 Fuel Cycle Crosscut Group

Chapter 5

Institutional Boundary Conditions and Enabling Developments

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CHAPTER 5: INSTITUTIONAL BOUNDARY CONDITIONS AND ENABLING DEVELOPMENTS

This chapter is intended to review the history and status of 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 does bring an overview of today's situation and thoughts on these boundary conditions.

The nuclear industry is highly regulated. The laws and regulations are often complex and 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.

In the past, institutional arrangements in the form of conventions or treaties have been introduced in domains where consequences of the use of nuclear power became of an international nature and where a net benefit could be achieved in terms of sharing these consequences. Certain domains have never resulted in such international conventions, for instance in the field of nuclear regulation, as bilateral or multilateral information exchange was considered most appropriate without the need of a higher institutional organism to bring harmonization or agreement.

This chapter will address the existing conventions and treaties in the field on nuclear damage, nuclear safety, radioactive waste and non-proliferation. Insight will be given in those development deemed necessary in the future if nuclear energy would face a significant deployment.

5.1 The Institutional Challenges for Generation-IV

The Gen-IV sustainability goals contain, in essence, three elements – long-term cost-effective supply of energy, acceptable environmental impact, and acceptable level of risk that the Gen-IV fuel cycle cannot lead to political instabilities through avenues to proliferate weapons of mass destruction. Part of these goals will have to be achieved through the use of government regulation and, especially, international regulations and treaties. The market forces, as always, will therefore be tempered by such government regulation. These regulations cover, in essence, four domains:

- environmental impact and human health, i.e. radio-protection measures and regulation.
- reactor and fuel cycle installation safety, i.e. nuclear regulatory arrangements
- third party responsibilities in case of accidents, and
- non-proliferation

It is evident from the fifty-year evolution of civilian nuclear power that a combination of institutional and technological measures which operate in complementary and supplementary manner on each link of the fuel cycle chain will be required to achieve all three of the Gen-IV sustainability goals simultaneously. Innovations will be required in both areas to address the broad world-wide diffusion of civilian nuclear energy envisioned for Gen-IV technologies. Both facets are addressed by the Fuel Cycle Crosscut Group.

The following sections will bring an overview, including a short historical sketch, of the institutional issues that were recognised in the past and that may need to be developed in the future. Some of the issues may be limiting while others may become enabling for nuclear energy development. Different opinions on the need and development of new international treaties or conventions may exist and this document does not pretend to give a consensual view on these aspects. The FCCG aimed by collecting this information to give a picture of potential institutional boundaries where future additional work might be needed to find a more consensual view on these.

5.2 Institutional Aspects

5.2.1 Non-proliferation

The 1968 Treaty on the Non-Proliferation of Nuclear Weapons is the basis for the world's nuclear weapons non-proliferation regime. The purpose of the Treaty is to keep the number of countries with nuclear weapons to the five countries that possessed such weapons before 1967: the United States, Russia, the United Kingdom, France, and China. In addition to the five nuclear weapons States, 175 other countries are members of the Treaty. On May 12, 1995, the Review and Extension Conference of the Parties to the Treaty agreed by consensus to extend the Treaty for an indefinite period. The obligations for compliance with the Treaty on the Non-Proliferation of Nuclear Weapons apply to both nuclear weapons States and non-nuclear weapons States. While non-nuclear weapons States agree not to pursue development or acquisition of nuclear weapons or other nuclear explosive devices, the nuclear weapons States commit themselves to work toward the ultimate elimination of their nuclear arsenals. All States are thus bound to help reduce the global threat of nuclear weapons, but must do so without prejudice to a nation's ability to pursue the benefits of peaceful uses of nuclear energy.

This need to prevent the spread of nuclear weapons was evident from the first days of the nuclear era. On November 15, 1945, the United States, the United Kingdom, and Canada proposed the establishment of a U.N. Atomic Energy Commission for the purpose of "entirely eliminating the use of atomic energy for destructive purposes." The Baruch plan of 1946, offered by the United States, sought to forestall nuclear arms proliferation by placing all nuclear resources under international ownership and control.

But the early post-war efforts to achieve agreement on nuclear disarmament failed. The Soviet Union in 1949, the United Kingdom in 1952, France in 1960, and the Peoples Republic of China in 1964, became nuclear-weapon states. And increasingly it was becoming apparent that earlier assumptions about the scarcity of nuclear materials and the difficulty of mastering nuclear technology were inaccurate. South-Africa and Israel possess nuclear weapons as well where more recently, in the 1990s, India and Pakistan also demonstrated their nuclear-weapon capability. It should be remarked that none of these four countries are signatories to the Nuclear Non-Proliferation Treaty.

Other developments and prospects further underscored the threat of nuclear proliferation. In the early 1960s the search for peaceful applications of nuclear energy had brought advances in the technology of nuclear reactors for the generation of electric power. By 1966 such nuclear reactors were operating or under construction in ten countries. It was estimated that by 1985 more than 300 nuclear power reactors would be operating, under construction, or on order. Nuclear reactors produce not only power, but also plutonium, i.e. a fissionable material which can be chemically separated and used in the manufacture of nuclear weapons. By 1985 it was estimated that the quantity of plutonium being produced worldwide would make possible the construction of 15 to 20 nuclear bombs daily, depending upon the level of the technology employed.

If the diversion of nuclear materials from peaceful purposes was not prevented by an international nuclear non-proliferation regime, and if a growing number of nations came to possess nuclear weapon arsenals, it was believed that the risks of nuclear war as a result of accident, unauthorized use, or escalation of regional conflicts would greatly increase.

5.2.1.1 Brief historical overview

In the earliest days of nuclear power, the simple availability of plutonium was not considered to be a proliferation threat. The technology of weapons design was very sophisticated, was in the hands of only a few nations, and was held under the highest security. In those early days, it was U.S. policy that plutonium was a commercial material that should be available for private, uncontrolled ownership, like uranium. PUREX separation technology was unclassified and uncontrolled. This attitude changed when both the U.S.S.R. and France showed that they had acquired nuclear weapons technology without acknowledged U.S. help, and other countries were well on the way to gaining the capability. Private ownership of plutonium was eliminated in the U.S., and strict materials controls were edicted.

The early and ambitious goal of international ownership of all nuclear material, the Baruch Plan (Baruch, 1946), was never achieved. The lesser goals of the Non-Proliferation Treaty (NPT) – for weapons states to offer aid in developing peaceful nuclear applications in exchange for the verifiable commitment to refrain from nuclear weapons acquisition by a nation not already having these weapons – has been more succesful. International coverage was later broadened with the Treaty of Tlatelolco. Inspections by the International Atomic Energy Agency (IAEA) provide the evidence of compliance with these treaty commitments.

On the one hand, in the 1960s and early 1970s the projected growth in energy demand led the U.S. and others to plan for rapid expansion of plutonium stocks through the use of "breeder" reactors (AEC, 1962; 1967), and plutonium became viewed as a fuel resource.

On the other hand, as nuclear power and nuclear technology blossomed around the world in the 1970s and 1980s, the U.S. took the initiative to strengthen technical safeguards by unilaterally halting its civilian plutonium separation and purification, and by urging other nations to do the same (Carter, 1977). A major international study [the International Nuclear Fuel Cycle Evaluation, (INFCE)] reviewed proliferation risks in the light of "the urgent need… for nuclear power [to be] widely available" (INFCE, 1980). The study concluded that the sensitive points in the nuclear fuel cycle were: 1) enrichment facilities and their in-process product, and 2) separated (pure) plutonium and the facilities for such purification. However, the study also indicated that none of these sensitive points were not proliferation-resistant and that adequate measures could be taken to avoid any diversion of weapons-usable material. In response, in 1984 the U.S. undertook research to develop recycle technology which does not produce pure, separated plutonium.

Recently the Clinton Administration reiterated and continued implementation of the earlier Carter Administration policy for the US:

> *"The United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. The United States, however, will maintain its existing commitments regarding the use of plutonium in civil nuclear programs in Western Europe and Japan."*

Going further, R&D on recycle technology and advanced reactors relying on recycle was discontinued in the US in 1994.

A total of 20 countries, convinced of the wisdom of plutonium recycling both from an energy supply and from a waste management perspective, have declared their intent either to press ahead with plutonium recycling and mixed uranium-plutonium oxide (MOX) fuel in their LWRs, or at least to preserve the option of doing so. Russia has a long-standing policy to use plutonium in an expanding number of their fast reactors (Mikhailov et al., 1994). In their current situation, Russian leaders see this as a way to make more uranium available as an export commodity (Mourogov, 1994).

5.2.1.2 The Current Situation

The INFCE study resulted in achieving some degree of international consensus, but the level of proliferation risks attendant to certain fuel cycle steps have continued to be weighted differently in different national policies. But no matter which fuel cycle has been deployed, an inexorable buildup of bred transuranic material has been taking place.

The world inventory of plutonium produced in civilian power programs and for weapons use is huge, currently estimated at about 1000 tonnes (200 tonnes of these being from civil power programs) and growing at about 60 to 70 tonnes per year (Albright et. al., 1993). The vast majority of this material resides in spent nuclear fuel from existing power plants and is spread among at least 30 countries around the world. Additionally, in the 1990's the end of the Cold War has released substantial quantities of highly enriched uranium (HEU) and separated plutonium from military uses.

This release of materials due to disarmament, coupled with the degradation of physical security and accounting for civilian separated plutonium in the Former Soviet Union, motivated an extensive study in the United States [6] that developed a new measure of intrinsic proliferation resistance labeled the "*spent fuel standard*." Beyond defining intrinsic material characteristics that provide proliferation resistance, this work motivated more complete declassification of information on the utility of the isotopes of plutonium for use in nuclear explosives, showing that "reactor grade" plutonium is nearly equivalent to "weapons grade" for proliferation concern.[1] Also important in the

[1] A recent U.S. DOE Office of Arms Control and Non-proliferation report provides the most complete discussion available on reactor grade plutonium. "At the lowest level of sophistication, a potential proliferating state or sub-national group using designs and technologies no more sophisticated than those used in first-generation nuclear weapons could build a nuclear weapon from reactor-grade plutonium that would have an assured, reliable yield of one or a few kilotons (and a probable yield significantly higher than that). At the other end of the spectrum, advanced nuclear weapons states such as the United States and Russia, using modern designs, could produce weapons from reactor-grade plutonium having reliable explosive yields, weight, and other characteristics generally comparable to those of weapons made from weapons-grade plutonium. The greater radioactivity would mean increased radiation doses to workers fabricating such weapons, and military personnel spending long periods of time in close proximity to them, and the greater heat and radiation generated from reactor-grade plutonium might result in a need to replace

late 1990's was the declassification of the potential utility of the minor actinides for use in nuclear explosives, with a particular emphasis on neptunium-237 which is present in high-level waste in isotopically pure form and which has properties very close to HEU. During this period reviews by the International Atomic Energy Agency, which carries the responsibility for implementing international safeguards, also concluded that indefinite safeguards monitoring will be required for spent-fuel repositories, even following closure [7].

The US National Academy of Sciences (NAS) has said in 1995 that for the short- and medium-term (decades, or until the resource benefits of recycle become significant, or until advanced separations technologies become widely available), assuring physical protection, accounting, and control measures equivalent to those provided for nuclear weapons, and subsequently and expeditiously adulterating weapons plutonium to meet a "spent-fuel standard" are an essential response, from the US perspective, to the "clear and present danger to national and international security" presented by the release of separated nuclear materials from dismantled nuclear weapons (CISAC, 1994).

At the front end of the fuel cycle, non-proliferation concerns play a factor in uranium enrichment technologies.[2] During the Manhattan Project, gaseous diffusion, thermal diffusion (later abandoned), centrifuge separation, and electromagnetic separation were developed to enrich ^{235}U in natural uranium in the form of UF_6. Of these processes, gaseous diffusion is now the principal means of enriching uranium in the U.S. and centrifuge separation is the principal means[3] used in most facilities outside of the U.S. Laser enrichment through the atomic vapor laser isotope separation (AVLIS) process holds promise of a single-pass method of producing uranium sufficiently enriched for commercial fuel. This process has the lowest separative work and, hence, lowest energy consumed and the smallest thermal signature. In spite of the sophistication of the laser technology required, some harbor fears that laser enrichment, because of its lower capital costs and greater ease of concealment, will enable clandestine uranium enrichment for nuclear weapon production by proliferant countries.

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

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certain weapon components more frequently. Proliferating states using designs of intermediate sophistication could produce weapons with assured yields substantially higher than the kiloton-range possible with a simple, first-generation nuclear device." ("Final Non-proliferation and Arms Control Assessment of Weapons-Usable Fissile Material Storage and Plutonium Disposition Alternatives," U.S. Department of Energy, DOE/NN-0007, 1997.)

2 Electromagnetic separation was practiced in Iraq to secretly produce nuclear weapons materials until this attempt was discovered and terminated by U.N. inspectors.

3 Worldwide installed enrichment capacity is currently on the order of 55 million separative work units (SWU) annually, against a current demand on the order of 36 million SWU.

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 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 "channelling" 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, channelling 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 OECD/Nuclear Energy Agency (OECD), 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[4]. 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"[5] occurring in a "nuclear installation"[6] 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[7]. 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 SDRs[8], although national legislation may fix a higher amount if corresponding financial security is available[9]. A

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

[5] 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.

[6] "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.

[7] 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.

[8] 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.

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 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[10].

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 judgements 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 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.

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[9] 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.

[10] 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.

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)

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[11], 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[12] 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.

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[11] 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.

[12] 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

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 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[13] 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.

[13] Argentina, Morocco and Romania.

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.

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[14]. 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 Convention 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 recognised 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[15] 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[16]. 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

14. 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.

15 Argentina, Morocco and Romania.

16 For example, the United States, the Russian Federation, Canada, Korea, Japan and China.

on the concept of "economic channelling" which, in practical terms, leads to same result as does legal channelling, 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. It is hoped that the negotiations will be finalized within the next few months.

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 is will only be available to States who join the Convention after January 1, 1999.

The Paris 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, a category which most people, at least at the IAEA negotiations did not really understand.

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 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 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.

5.2.2.2 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.

5.2.4.1 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.

5.2.4.2 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 release 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.

5.2.4.3 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 channelling 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.

5.2.4.3 Convention on the Physical Protection of Nuclear Material

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.

5.2.4.5 The IAEA Safety Standards

The IAEA Safety Standards comprise publications of a regulatory 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 but also at the end of operation, i.e. decommissioning. An increase in nuclear energy production will therefore also result in an increase of radioactive waste to be handled. While the specific waste production, i.e. sustainability criterion SU2-1, may be reduced, the most probable overall effect would still be an increase of the waste amount. The latter would surely be applicable for the amount of waste from decommissioning activities of a growing amount of reactors and fuel cycle facilities.

While nuclear energy systems may be designed that minimise the production of high-level waste, this may not be the case for the low and intermediate level waste arising from fuel cycle and reactor operation, decommissioning activities and, in general, secondary waste arisings. Working methods and technologies have been improved in order to minimise also the amount of this kind of waste but, as a general rule, the volume of LLW and ILW remains very important.

While technological 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 verly low-level radioactive waste types may be obtained.

The disposal of radioactive waste is guided by national regulatory bodies on the basis of national laws and regulations[17]. At the same time, international collaborative efforts have benefited national waste-management programmes. These efforts provide not only a world-wide exchange of technical information and expertise, but also internationally recognised, non-binding safety standards and binding agreements among states.

Trends in the areas of national legislation and regulation include: the agreement that very long timescales (greater than 10 000 years) cannot be ignored for HLW, but require different treatments at increasing times; the advocacy of the use of safety indicators in addition to calculational estimates of radiation dose or risk; and the embedding of laws or regulations for waste disposal in a wider environmental regulatory framework. There is, however, currently little direct attention, at the legislative level, to the need for broad-based approaches that assure a uniform degree of environmental protection for society. For example, there is often little linkage between regulation of materials that are radioactive (whether or not they arise from the nuclear industry), and of non-nuclear hazardous or

17 . See http://www.radwaste.org/laws.htm for a list of national laws and regulations on waste management.

toxic waste streams. The discussion on exemption and clearance levels for waste also relates to this latter consideration.

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.

International fora 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 which has become more common in recent years.

5.2.4.1 Trends in national laws and regulations.

There is a trend in national regulations towards the advocacy of the use of safety indicators in addition to calculation estimates of radiation dose or risk and of stylised approaches for assessing long-term impacts. This trend reflects international discussions, and the work of the ICRP and IAEA, regarding the difficulties of demonstrating compliance with radiological and other safety objectives in the far 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 and its appeal to different audiences. 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 of impacts. 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. There is, however, little mention of issues like sustainability or of broad-based approaches at the legislative level to assure a uniform degree of environmental protection against radioactive waste sources and also between nuclear and non-nuclear hazardous waste forms and between the short and long term environmental impacts of alternative energy sources. Furthermore, although many countries claim coherence between legislation affecting radioactive waste disposal and other environmental legislation, there is currently little direct evidence that objective risk-based regulations are ensuring equitable treatment - and proof of treatment - of all potential hazards to humans. For example, in some countries, permits to allow injection of toxic materials into boreholes require in

principal that the applicant demonstrate that the toxins will 'never' endanger groundwater, but analysis projecting potential behaviour into the very fat future are not required to be submitted and reviewed in an exhaustive manner equivalent to that for radioactive wastes.

5.2.4.2 Progress in international legislation and regulation

The disposal of radioactive waste is guided by national regulatory bodies on the basis of national laws and regulations. At the same time, international collaborative efforts have benefited national waste management programmes, 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.

Over that past decade, several binding agreements among sovereign States have been implemented with the support of the IAEA. These agreements may be viewed as a component of a global 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 conventions on the safety of spent fuel management and on the safety of radioactive waste management

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[18], and to discharges which are defined as planned and controlled releases into the environment, as a legitimate practice,

[18] 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.

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 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, amongst 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 obligation 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 of or that has resulted from past practices. In addition, the inventory list 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.

In addition, review meetings are to be held by the Contracting Parties for the purpose of reviewing the national reports, which must be submitted. 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.

Undoubtedly this new Joint Convention makes a significant contribution to what has become a whole new body of law, the international law of nuclear safety.

5.2.4.3 Regulator and implementer consensus on key issues.

In early 1997, an NEA workshop took place in Cordoba, involving active regulators and implementers from a wide range of national programmes. The first novel feature of the meeting was that is was explicitly dedicated to encouraging dialogue between these groups. One aim was to establish where real consensus on key issues existed, in order to avoid confusing a wider audience

with debate on apparent differences which might be based more on semantics than substance. The participants examined waste disposal objectives and criteria, trends in performance assessments and the conduct of the regulatory processes. In the specific area of open regulatory issues, the key points agreed upon were that:

- A more sophisticated yardstick than only the 'high-level' criteria of dose or risk is needed.
- The waste management guiding principle that adequately protecting mankind also protects the broad environment was questionable.
- Long timescales (longer than 10 000 years) cannot be ignored, but require different treatments at increasing times.
- Communication between regulators and implementers, as well as with broader groups, needs improvement.
- Licensing requirements ('rules of the game') should be framed well ahead of when they are to be applied - and can justifiably include simplified or stylised approaches.
- The stepwise approach is valuable for implementers, regulators, decision makers and the public.

5.2.4.4 Exemption and clearance levels

The classification of radioactive waste is an important issue as it raises issues 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-postiriori, is important in the light of increasing amounts of LLW coming from operation and decommissioning of fuel cycle facilities and reactors.

In the ICRP Publication 64, the 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 estimated 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 form 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 continues to be 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, there would be some implications on public, worker and environmental radiation protection, most likely necessitating evolution and rethinking 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, rationale

have been developed to justify the selection of a numerical dose criteria that serves 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 not be exposed to more than two or so sources of radiation at any one time. Should the number of nuclear power installations, and associated fuel cycle facilities, increase significantly, this numerical criteria would have to be revisited.

- *Radiological protection of the environment*: The international radiation protection community is currently reviewing its principles for the protection of the environment against the 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 would need to be revisited, particularly in terms of any numerical criteria that might be established. In this same 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 that are 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 drastically, it is likely that many of these would 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 would 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 assuming dramatic increases in the number of nuclear power installations, it is likely that the use of foreign-national workers will increase. In order to assure that workers respect regulatory exposure criteria, institutional agreements and mechanisms will have to be developed to track and log worker individual exposures in real time.

    - Currently, although most countries use the same numerical worker annual dose limit, this is not completely 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.

Gen-4 Fuel Cycle Crosscut Group

Chapter 6: Summary, Conclusions, and Recommendations

November 1, 2001

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Chapter 6: Summary, Conclusions, and Recommendations

Background

The Charter of the Gen-4 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-4 fuel cycles, consistent with projected energy demand scenarios."

The membership of the FCCG comprised 8 US members and 7 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-4 Roadmap organization. Five one-day working meetings were held between February 2001, and August 2001 – three of them in conjunction with Gen-4 TWG quarterly meetings.

6.1 Summary

The Gen-4 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-4 fuel cycle for proliferation of weapons of mass destruction or nuclear terrorism. These sustainability goals depend strongly on choices made for Gen-4 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-4 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-4 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-4 concepts to accomplish this by considering four generic fuel cycles (see Figure 6.2) –

once-thru, partial recycle, full Pu recycle, and full TRU 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).

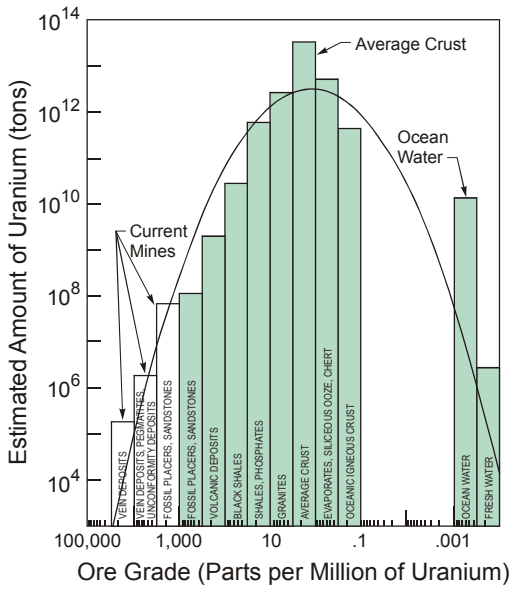


Fig. 6.1. Distribution of Uranium in the Earth's Crust

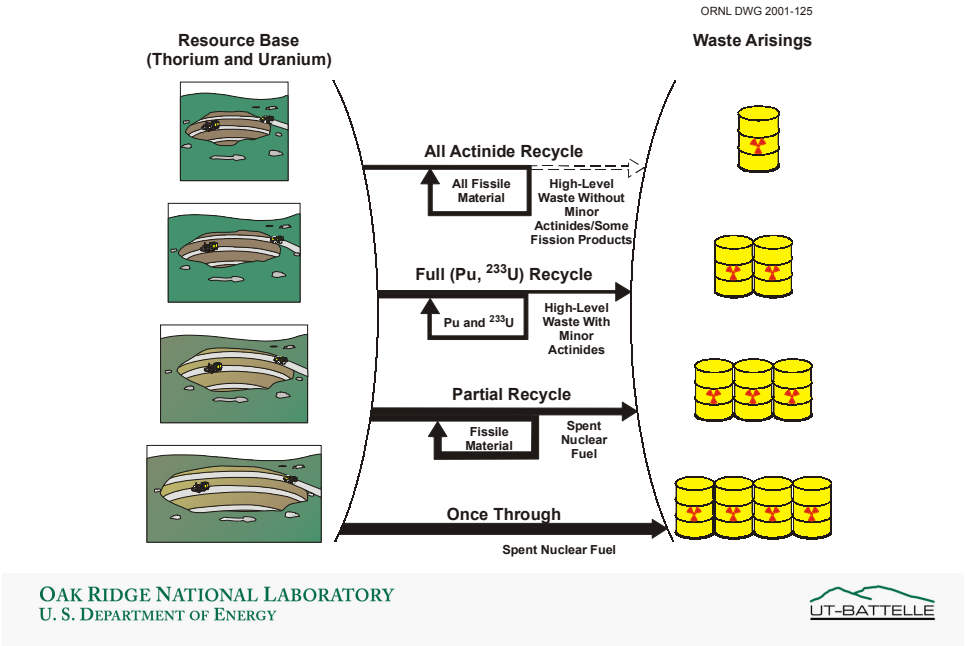
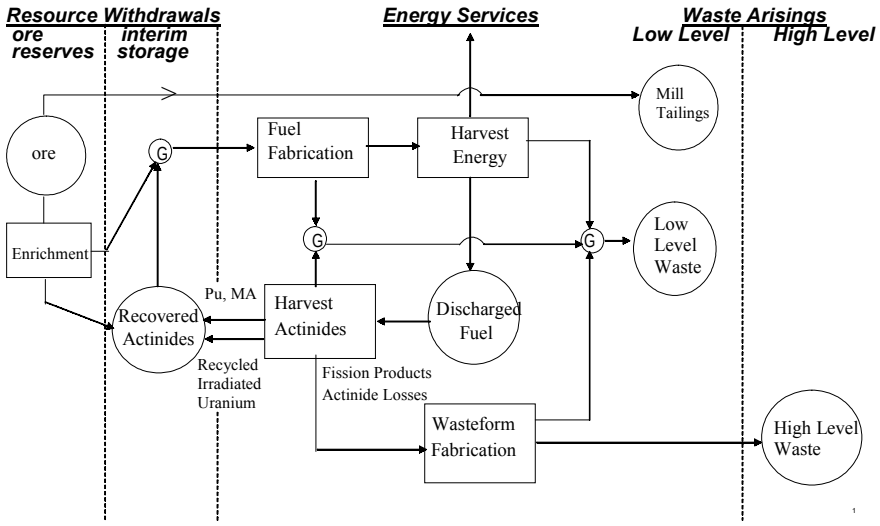


Fig. 6.2. Four Alternative Fuel cycles Have Been Defined



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				
Once Thru	Fuel	Interim Store	Waste	Waste	Waste	Waste	No	No	Yes	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 exclusively 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-thru 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 money makers and good investments. New nuclear capacity is being added primarily in regulated markets (in Japan, Korea); whereas the financial parameters 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 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.

The fuel cycle currently contributes about 20% to the cost of nuclear energy production (see Fig. 6.4).

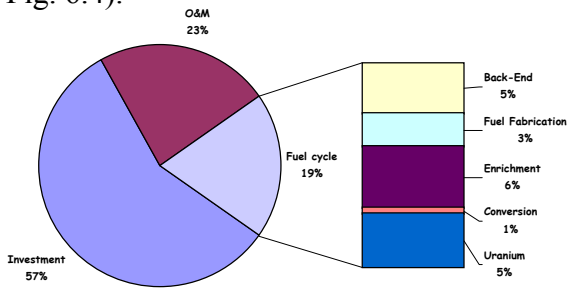


Fig. 6.4 Nuclear Electricity Generation Cost Are Not Currently Controlled By The Fuel Cycle

A survey of Gen-4 concept submittals was made to define the scope of proposed future fuel cycles and fuel compositions to be addressed in FCCG deliberations. (See Table 6.1). Table 6.1 shows the results of a survey of Gen-4 concepts 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

Potential Gen-4 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 U238 or Th232 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 remote 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 a subset of all of the possible fuel cycles (for example the use of thorium as fertile material has not yet been modeled), 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 its 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 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.[1] Lessons learned from the scenarios

[1] The cost data and their method of application in the scenarios require further peer review in the next phase of the study; the cost index and its components should be viewed as tentative pending this peer review.

are described in Section 6.2, below. Additional scenario cases have been identified for ongoing evaluations. These include the DUPIC cycle, various thorium cycles and a liquid fuel cycle.

Table 6.1. 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 thru	Water	UO_2 pellets						N/A	N/A	T		
IRIS	Water	UO_2 pellets						N/A	N/A	T		
Multi-application Small LWR	Water	UO_2 pellets						N/A	N/A	T		
Homogenous Thoria-Urania	Water	ThO_2-UO_2 pellets, ~70% Thoria						N/A	N/A	T		
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: enriched – uranium metallic dispersion fuel			N/A	N/A	T		
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	UO_2 pellets						N/A	N/A	T		
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	UO_2 pellets						N/A	N/A	T		
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					

	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
Gas-cooled Reactor w/ HEU	He	Alt: TRISO-Coated UO_2					TRISO-Coated UC					
Fast Spectrum												
(2) Limited Resource Recovery with disposal of FPs and some Actinides												
Thermal Spectrum												
IRIS	Water	PuO_2-UO_2 pellets						Yes	No		T	
Heterogeneous Thoria-Urania (Radkowsky Fuel)	Water	Blanket: ThO_2-UO_2 pellets, ~90% Thoria			Seed: plutonium-bearing, metallic dispersion fuel			Yes	No	T	T	
Supercritical Water-Cooled Thermal Reactor	Super-critical Water	PuO_2-UO_2 pellets						Yes	No		T	
Supercritical Water-Cooled CANDU Reactor	Super-critical H_2O, w/ D_2O moderator	PuO_2-UO_2 pellets						Yes	No		T	
HTGR with LWR Pu feed	He	TRISO-Coated PuO_2						PUREX				T
Gas-cooled Reactor w/ LEU/Th	He	TRISO-Coated ThO_2 Alt: TRISO-Coated UO_2					TRISO-Coated UCO		Particle Separation			T

	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
Fast Spectrum												
Tight-lattice BWR Thoria Cores	Water	PuO_2-ThO_2 pellets						Yes	Maybe		T	Maybe
(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			T	
S-PRISM	Na	MOX pellets						PUREX			T	
(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)						T	T			T
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			T
Advanced Design GCFR using feed from LWR Pu	He	TRISO-Coated PuO_2						T	T			T

	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
with possible intermediate HTGR cycle		& TRISO-Coated $(MA)O_2$ (mixed or separate)										
Gas Cooled Fast Reactor	He	MOX pellets	Mixed Nitride pellets				CERMET, CERCER	T	T			T
SPRISM	Na				U-Pu-Zr				Pyro			T Casting
BREST	Pb		Vibro or pellet						Pyro or Flouride			T Vibro or pelletizing
STAR-LM	Pb-Bi		Vibro or pellet						Pyro			T Vibro or pelletizing
STAR-LM	Pb		Vibro or pellet						Pyro			T Vibro or pelletizing
ENHS	Pb-Bi or Pb				U-Pu-Zr				Pyro			T
IFR	Na				U-Pu-Zr				Pyro			T

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 6.2 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 6.2 Worldwide Fuel Cycle Technology Status & R&D

Deployed
- Water Reactors/UOX Fuel: Dominates Commercial Power Plants
- Once-Thru 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-Thru
- 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 ⇒ Recover MA for recycle
- Pyro and Other Dry Recycle ⇒ 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 and Fuel Cycles
- 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 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 OECD-NEA legal staff undertook a broad review of the current institutional basis of international commerce in the nuclear fuel cycle.

Treaties and norms 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

and

- Conventions on spent fuel management and radioactive waste management (including high, intermediate, and low level waste)

are already in place – but are currently 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 growth 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-4 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-thru cycle throughout the 21st century. This served as the "no action" reference base case against which the performance of alternative Gen-4 concepts in achieving Gen-4 goals could be compared. Figures 6.5 through 6.8 show the base case results.

Table 6.3 enumerates the other cases which were run. Illustrative results from these Gen-4 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-4 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)

(1) LWR Once-Thru

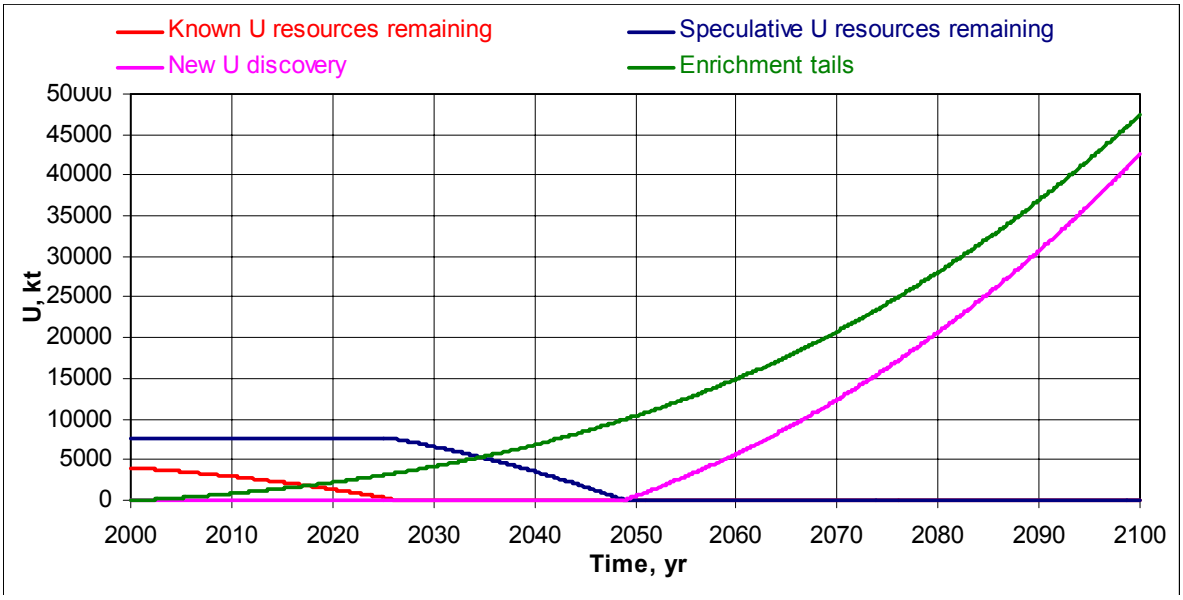


Fig. 6.5

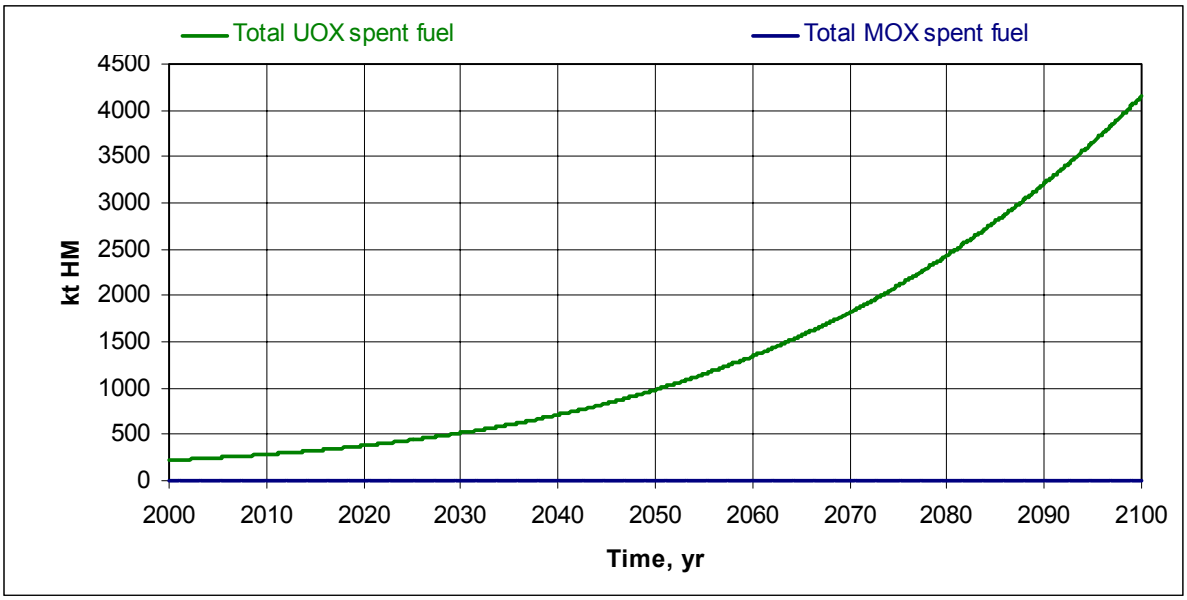


Fig. 6.6

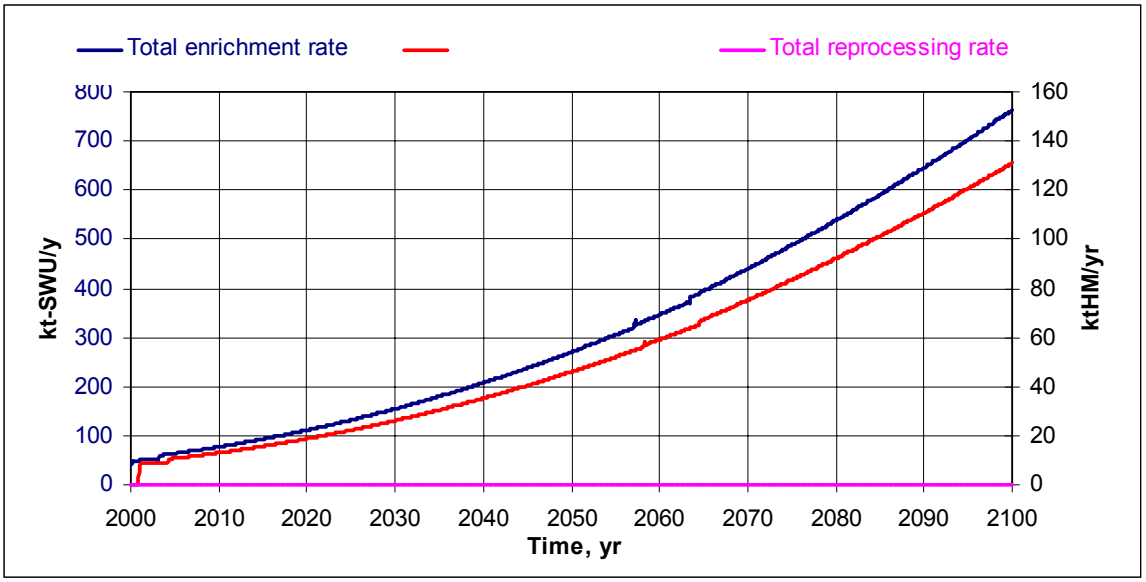


Fig. 6.7

(1) LWR Once-Thru

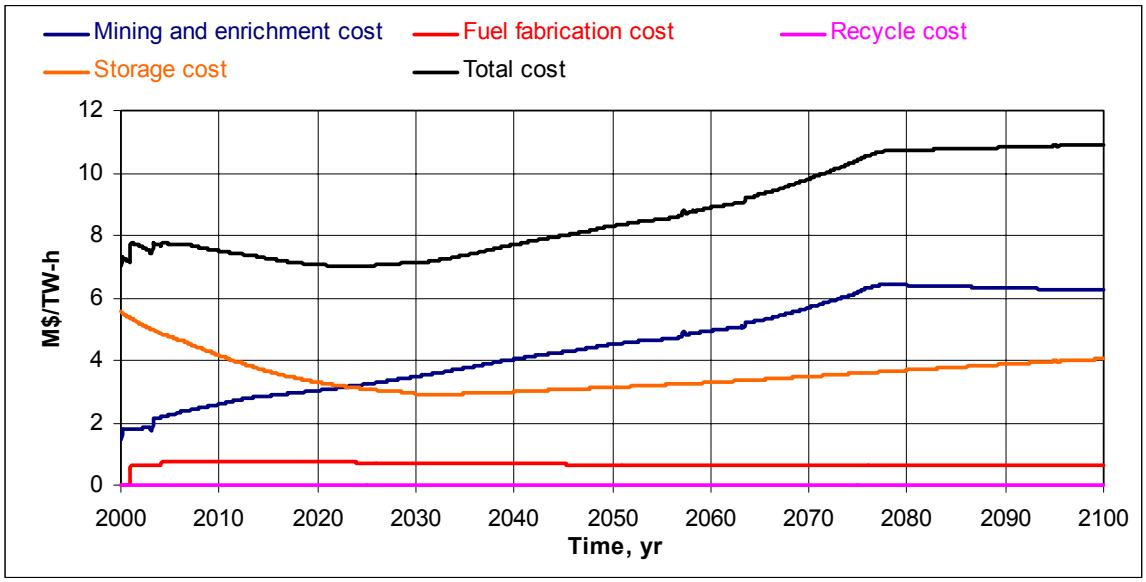


Fig. 6.8

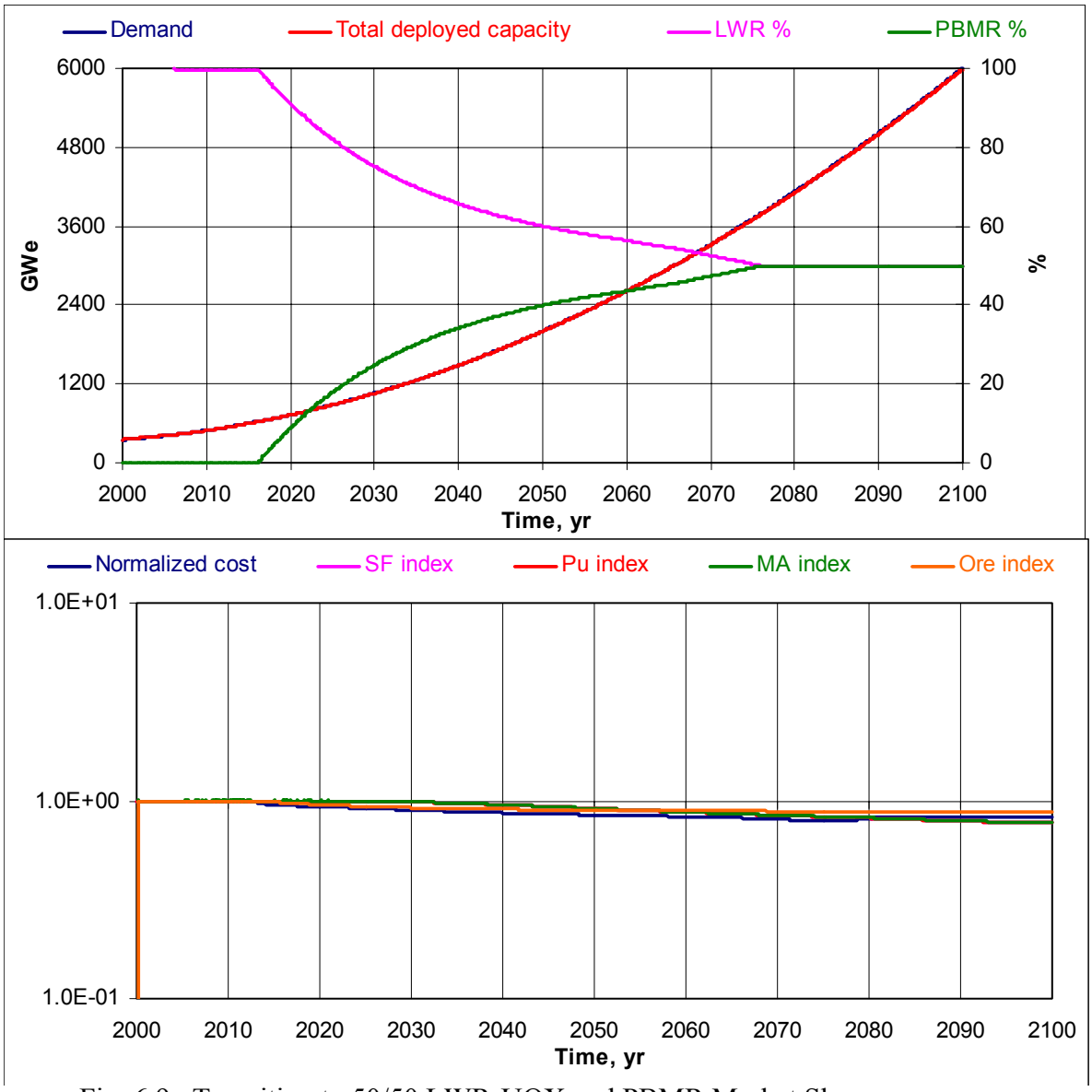


Fig. 6.9. Transition to 50/50 LWR-UOX and PBMR Market Share

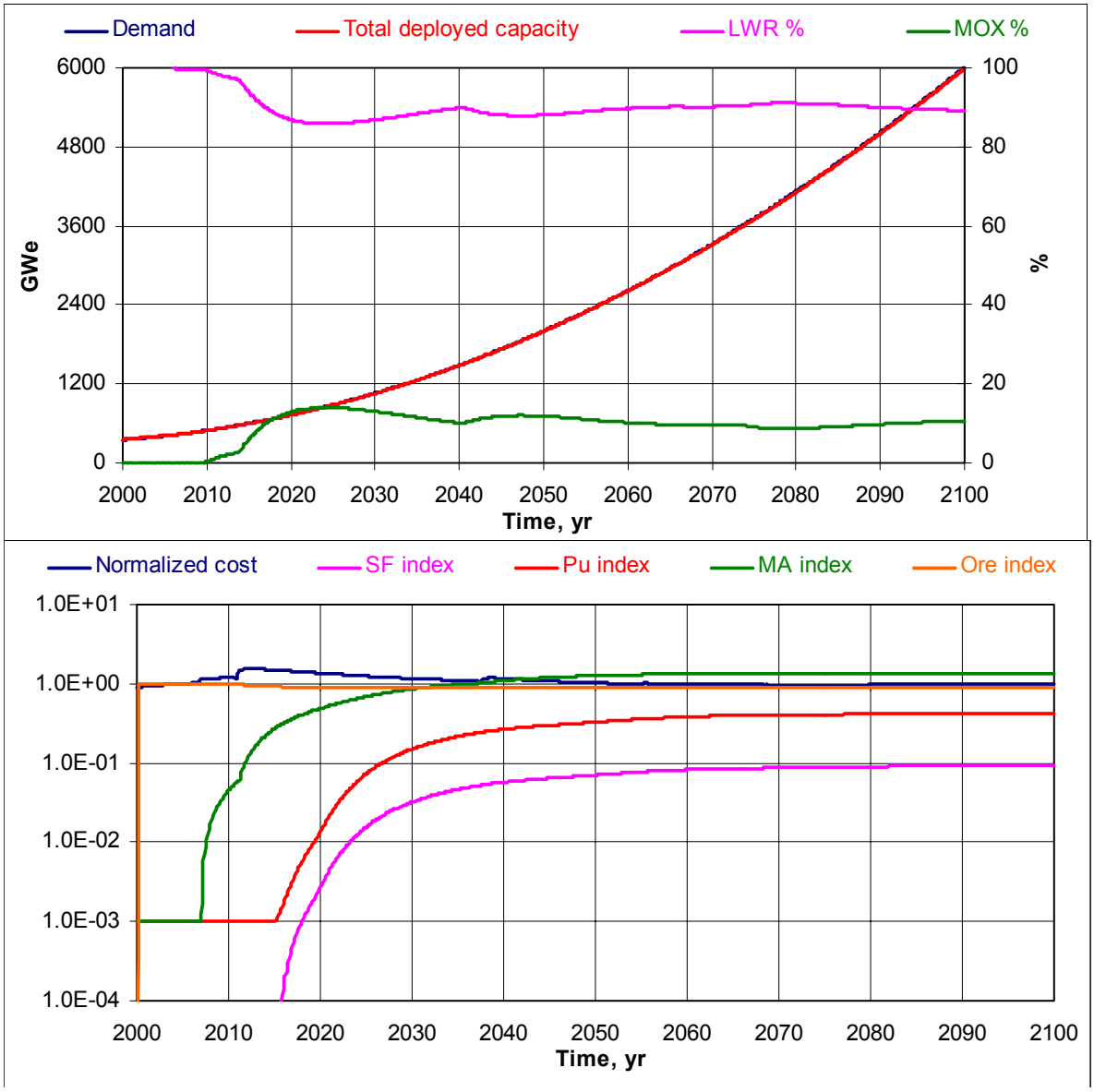


Fig. 6.10 Transition to 90% LWR-UOX/10% :LWR MOX Mono Recycle

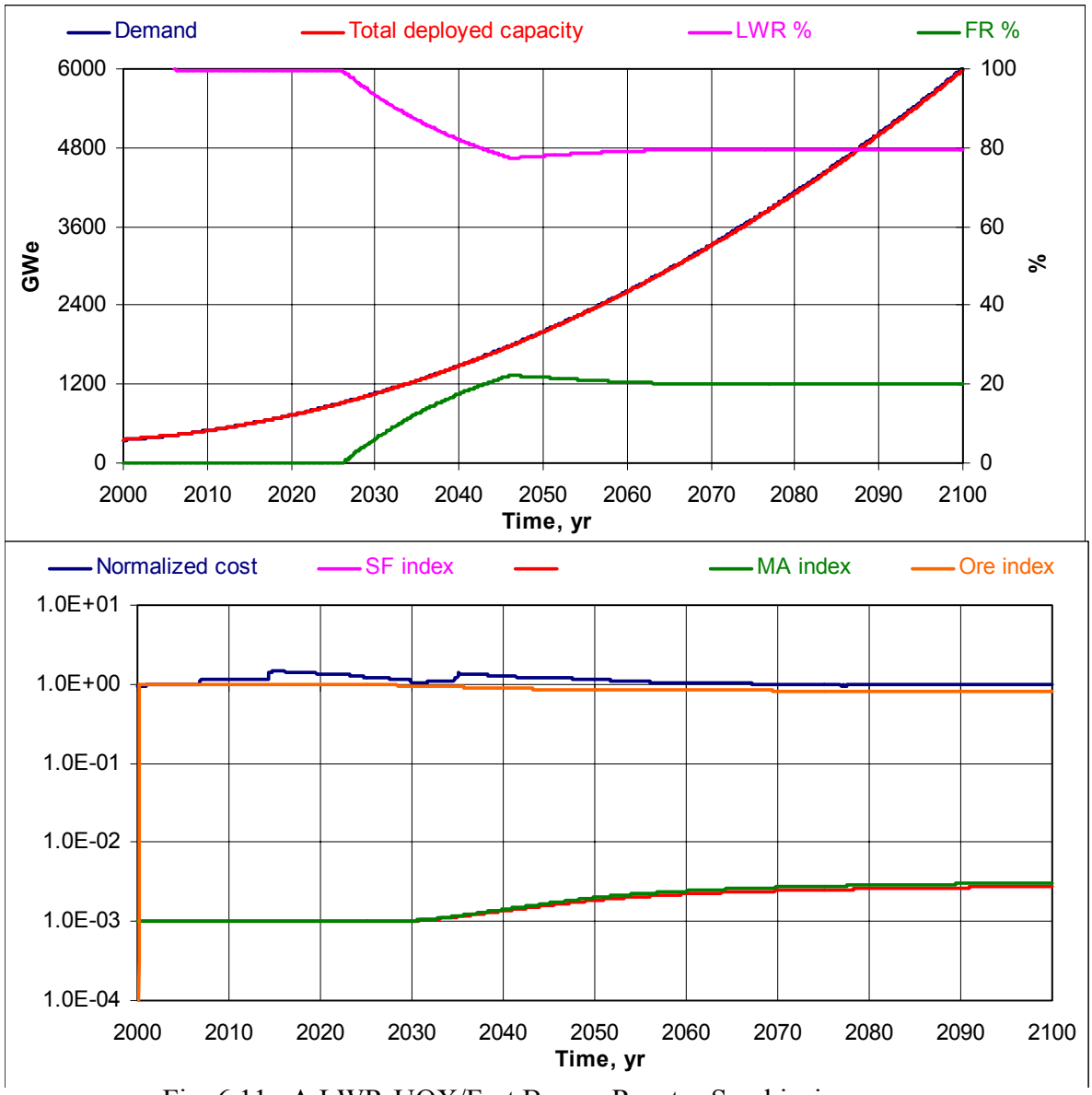


Fig. 6.11. A LWR-UOX/Fast Burner Reactor Symbiosis

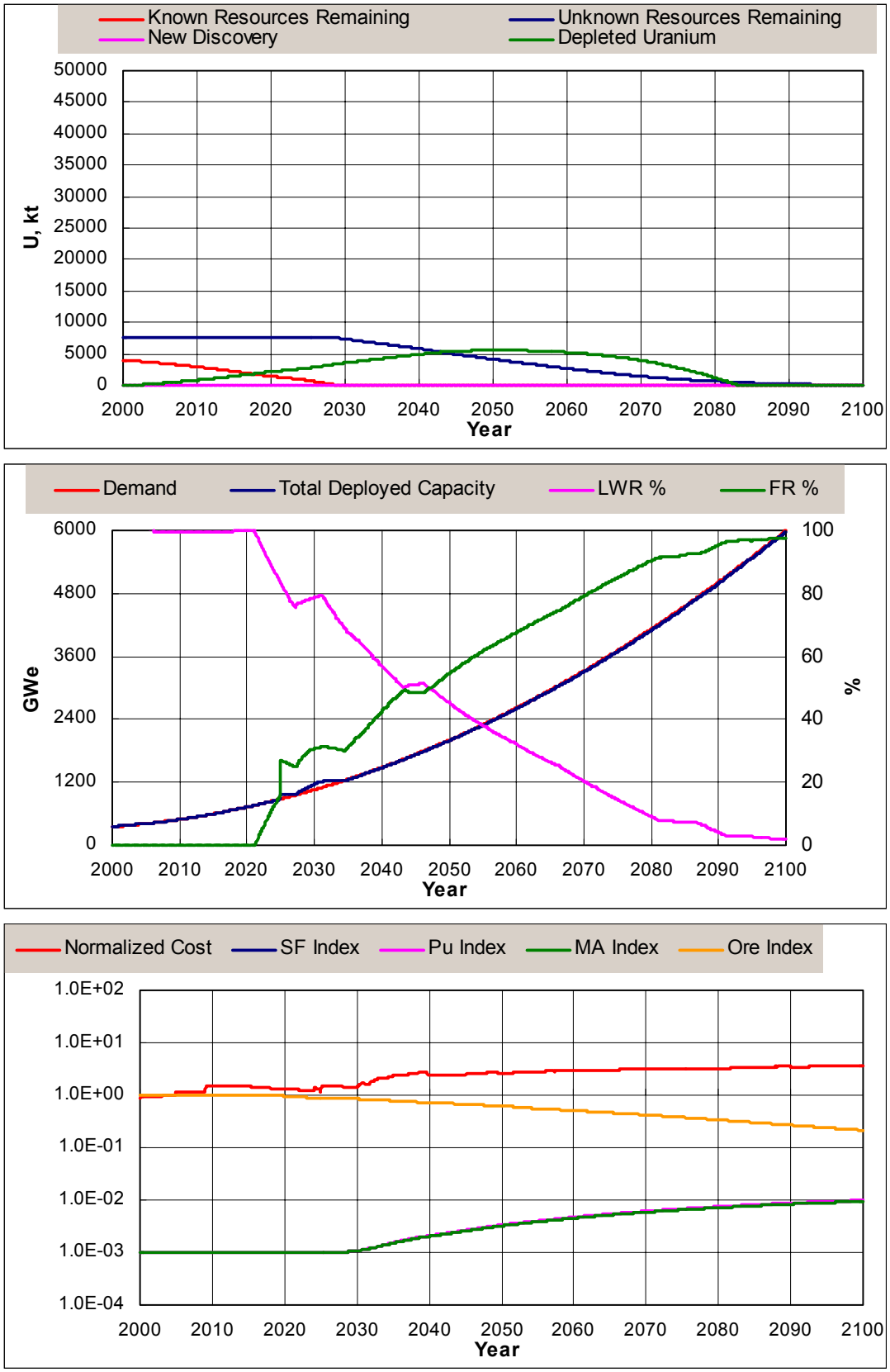


Fig. 6.12. Burner Switch to Breeder in 2030

Table 6.3. Scenario Cases Examples to Illustrate Fuel Cycle Issues

- Use Case B of WEC/IIASA for World as a Whole
- Assume 90% of already deployed Gen 3 LWR's worldwide will receive 20y life extension
- Cases 1 and 6 are the extremes; setting physically achievable bounds; Cases 2 thru 5a allow for practicalities of markets

Example Cases

A) Once Thru Thermal
   1. New Additions = LWR UOX Once Thru
   2. LWR UOX Once Thru; New additions (after 2010)
      HTGR Gas Turbine UOX Once Thru (50%)
      + LWR-UOX Once Thru (50%)

B. Limited Thermal Recycle (after 2010)
   3. New Additions = LWR UOX → LWR MOX mono
      → Waste

C. Full TRU Recycle: Symbiotic Fast/Thermal (after 2025) for Waste Management

   4a. New additions = 80% LWR (UOX) + 20% Fast Burner (BR = ½)
   4b. New additions = Fast Burner (BR = 1) (use up LWR discharged fuel)

D. Full TRU Recycle: Breeding (after 2030) for Limiting Ore Drawdown
   5. Case 4b – Switching in 2030 to New additions – Fast Breeders (BR = 1.7)

E. Full TRU Recycle: Symbiotic Feedback of Fissile to Thermal Systems
   6. New additions = fast breeders + HTGR's (TRU)

axis and and ratio of performance to the base case on the vertical logarithmic axis. The principal lessons learned include the following:

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 very significantly reduces the buildup rate of waste mass destined for the repository – 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 minor actinides separated from plutonium in the UOX reprocessing operation. Long term radiotoxicity of materials destined for the repository 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 in reprocessed to supply the initial working inventories of new deployments 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.
5. Symbiotic couplings among fast and thermal power plants is illustrated by the scenarios. In one case a fast burner companion reactor is deployed with each LWR to manage its waste; the net result is a clean waste (fission products and trace losses of TRU as the only waste from the energy park) and a reduced ore drawdown rate while maintaining more than 50% of the energy park as thermal reactors.
6. In a second 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 penetrate into non-electric energy sectors such as hydrogen production.
7. The addition of thorium as a fertile material in the fuel cycle, replacing some of the U-238 that would otherwise be used, has not yet been evaluated by the FCCG. However, it is known from the literature that the use of thorium in thermal reactors can reduce the generation of minor actinides, due to the higher fission to capture cross section of the U-233 bred from thorium. Measures to improve the neutron economy in thermal reactors, such as the continuous refueling in pebble-bed HTRs and molten salt reactors or the addition of an accelerator-driven neutron source, can also reduce minor actinide production. Thus the substitution of thorium for U-238 in the fuel cycle, and improved neutron economy in thermal reactors, have the potential to increase the ratio of thermal to fast reactors for fuel cycles achieving the same average conversion ratio and waste arisings.

8. Lastly the scenarios illustrate the ability to transition to a U-238 resource economy when availability of high assay ore reserves diminish and to thereafter supply the fueling needs of the fast and thermal symbiotic mix in the nuclear energy park off the inventories of enrichment tails and recovered irradiated uranium – inventories sufficient to fill many centuries of energy supply needs. The switchover can occur seamlessly if a segment of the energy park had already been filled using fast reactors operating in a burner or fissile self sufficient mode – because all that is needed is a change in core reloading pattern – rather than the deployment of a new fleet of previously undeployed reactor types.

The conclusions and recommendations for the R&D path forward are based on these scenario results which shows what will be essential to achieve sustainability goals. They also are based on an evaluation of the current status of fuel cycle R&D worldwide which shows what technologies are likely to be feasible with further R&D. These R&D gaps and status are described in Chapter 4 and Appendix 2 and summarized next.

6.3 Technology Gaps and Recommended Fuel Cycle Crosscutting R&D by Generic Fuel Cycle

The FCCG used the four generic fuel cycle types (once-through, partial recycle, full Pu and full TRU recycle) as an organizational mechanism for initial identification of Generation 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 major links in the fuel cycle for reaching Gen-4 goals. Figure 6.2 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 6.3-4 lists the fuel cycle-related R&D areas that could improve the resource input, conversion, and waste output links in a manner that would further enable Generation 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 are robust with respect to degradation mechanisms over geologic time scales are needed. Moreover, integrated interim storage/geologic repository logistic strategies that manage decay heat loads should be developed as a cost-effective way 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 6.3-4. Once-Through Generic Cycles: Fuel Cycle Crosscutting R&D

Going In	Conversion	Coming Out
• Technologies of Cost Effective - Prospecting - Mining - Mill Tailings Management • Enrichment Technologies		
• Fuel & Fuel Fab for Fissile fed-back from recycle or	• High Burnup Fuel • Fuel Suitable for High Thermodynamic Efficiency - High Temp. fuel - Fuel for Supercritical Water	• Fuel designed to be robust in repository environment for geologic times
• Low Enriched High Burnup Fuel - epithermal - burnable poison - partial substitution of thorium to replace U-238		• Improved Heat Mgt. 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 with the introduction of reprocessing and/or conditioning and refabrication of discharge fuel for mono recycle of plutonium back into thermal neutron spectrum reactors. This adds a new fuel resource (recycled plutonium) 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 6.3-5 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.

The feature of reprocessing 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-Pu or full-TRU 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 Generation 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 degraded isotopics plutonium feedstock, and fuel assembly (and core layout) designs for use with multi-recycle fuel feedstock of degraded isotope composition.

Table 6.3-5. Partial Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences relative to once-through fuel cycles: Reprocessing for harvesting Pu for return to thermal spectrum systems		
Going In	Conversion	Coming Out
• Same as Once-Thru for harvesting virgin ore and enrichment tails • For harvesting fissile from Spent Fuel: - Simplified Pu Recycle (S-PUREX) - Simplified Pu Refab - Fab using recovered irradiated U - Re-enrichment technologies for recovered irradiated uranium (U236) - Reduced Secondary Waste	• High temperature fuel • High Burnup fuel • Fuels (and core designs) for several recycle in Thermal Spectrum	• Recycle Fuel which is robust in repository environment and • Robust Customized Waste Forms MA FP High Heat FPs or Long-term storage technologies (to wait for full recycle regime):

Recycle • Remote fuel fabrication for feedstock fed back from recycle • Front end reprocessing technology for high plutonium content fuel		

6.3.3 Cross Cutting R&D for Full Pu Recycle and Full Transuranic Recycle Fuel Cycles

Whereas the once-through and partial recycle fuel cycles are already commercialized, the full plutonium recycle and full transuranic recycle generic 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 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 6.3-6 and 6.3-7 list fuel cycle-related R&D for the full plutonium recycle and the full TRU 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 TRU recycle case.

Full recycle, especially full TRU 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 many of the radioactive and radiotoxic wastes in the reactors themselves. The resulting requirement for remote refabrication drives the need to develop simple, few-step techniques including simplified pelletizing or vibrocompaction for ceramics fuel and casting or powder metallurgical procedures for metal fuels. Dispersion fuel development is also considered. Significant challenges are presented to develop fabrication technologies which retain volatile species (e.g., Am) in the fuel form during fabrication and to minimize or eliminate losses of the targeted isotopes during recycle and fabrication processes.

New fuel types of compositions containing increased plutonium fraction, new compounds (such as nitrides, spinels, apatites, etc.) and the potential presence of significant minor actinides fractions, fabricated using remote technologies, are recognized as challenging. 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. 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 against Generation IV goals) and

the reactor power plant (controlling cost, reliability, and safety performance against Generation IV goals). Uranium oxide fuel is highly refined and dominates the current once-through commercial fuel cycles and is qualified for fast reactor service as well. But new Generation IV reactor concepts and fuel cycles must be afforded opportunities to explore and exploit the potential of numerous other 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 U-238 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, which must remain a central theme in the Generation IV research program.

Under a working hypothesis that the required low level for trace losses 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. The importance of near-term siting, licensing, and deploying a geologic repository designed for the once-through fuel cycle cannot be overstated. For the future, however, the development of recycle-based fuel cycles should not go forward absent close linkage with the development of new repository designs to exploit new opportunities for customized waste forms and optimized decay heat management strategies, which can extend their capacities and reduce the level of required stewardship. Therefore, closely-linked R&D for new recycle and new repository design options should be considered 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 Pu or TRU recycle, and potentially thorium/U-233 recycle, generic fuel cycles to thermal spectrum reactors (as well as to fast reactors) will likely be valuable as a part of Generation IV development. For the base-line IIASA/WEC scenario, 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, accelerator-based neutron sources) will be needed to compensate for dwindling virgin ore resources by no later than twenty years after the Generation IV deployment date in the 2030's, absent significant changes in ore recovery technologies. A proper balance between the beneficial strategy of dirty fuel and clean waste (required to meet Generation IV sustainability goals SU-2 and SU-3) versus the limited ability of current thermal spectrum reactors to accommodate minor actinides (for example, molten salt reactors, if successfully developed, could accommodate very high minor actinide loading), degraded plutonium isotopics and lanthanide fission products must be determined. 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 Generation IV research program.

Table 6.3-6. Full Pu Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences from once-through and partial recycle fuel cycles: Multiple recycle in fast-spectrum reactors; >99% 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 OK for FR - Large FP carryover is not OK for thermal • and/or advanced thermal reactor systems - Thorium-based fuels - improved neutron economy (e.g. molten salt reactors) • Fab of fast reactor fuel (high Pu content) • Hands of Fab Thermal Reactor Fuel - But remote if poor FR Separation • Simplified recycle Technologies (<10-4 decontamination from FP's)	• Same as once-thru and partial recycle • And/or advanced thermal systems (e.g. molten salt reactors)	• Robust Waste Form for FP + TRU • Customized Waste Forms Ma FP High Heat FPs or Long-term storage technologies (to wait for full recycle regime)

Table 6.3-7. 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 Fuel Thermal Reactors		
Going In	Conversion	Coming Out
• Remote Fab of very high radiation	• Same as for other fuel	• Less stringent

feedstock - Retaining Am in the mix • Accountancy of very high radiation fuels & of processes having no input accountancy tank • Cost Effective, simplified recycle with secondary waste reduction and high TRU recovery • Integration/Co-siting for Recycle/Refab	cycles	functional requirements on Repository Designs owing to changed source term • Issues - Is it worth isotopic separations to get Cs for high heat separate from cs for long life - Is it worth trying to harvest the rare metal FP's:

6.4 Principal Findings

On the basis of the reviews, studies and evaluations, summarized above, the FCCG has produced a set of principal findings with a focus on Gen-4 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.

Today uranium and thorium are not major contributors to the cost of nuclear energy, and resources do not constrain the expansion of nuclear power. Within several decades the costs of nuclear 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

Reversing Waste Buildup in a Growing Nuclear Economy

1. Closed fuel cycles have already demonstrated a significant reduction of the volumes 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 transmuting 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 waste destined for geological repositories by at least an order of magnitude by recovering 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 arising 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 .

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 affordability safety and reliability goals from the strategies for meeting sustainability Generation IV Goals. Wide ranges of fuel cycle approaches can be undertaken with only weak influence on economics. Power plant design strategies to meet economic, 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.

Transition To Sustainability

4. Energy use is expected to grow substantially in the next century; consequently nuclear energy deployment is projected by authoritative studies to increase by 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.

   Modified Generation III reactors are needed to fill expanding energy needs and to employ limited repository space with high efficiency. Within of the symbiotic mix, Generation III 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 some residual contamination of recycle fuel from fission products are under current development. By focusing on creating clean waste streams 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 poisoned 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 tradeoffs and develop technologies for remote fabrication of thermal reactor fuel as well as fast reactor fuel.

Fuel Cycle Safety

8. The radiological exposure to 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 using already-mined resources, and by cost-effective remote refabrication will become increasingly important as the scope of nuclear power increases.

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

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 both 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 would 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-4 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 final prioritization ranking for Gen-4 fuel cycle R&D. This ranking is based on the principle 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 Generation IV sustainability goals.

The scenario results indicate that the full fissile recycle and especially the full transuranic recycle generic fuel cycles may be needed to achieve Gen-4 sustainability goals SU-1 and SU-2. Once-thru 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 Top-Ranking Gen-4 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 Gen-4 economic, reliability and safety goals for the global nuclear energy park, it is the "breakthrough" fuel cycle enabling technologies which are needed to handle the Gen-4 waste management and resource 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-thru, 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.

   New fuel types of compositions containing increased plutonium fraction, new compounds (such as nitrides, spinels, apatites, etc.) and the potential presence of significant minor actinide fractions – fabricated using remote technologies – is recognized to be a challenging and significant undertaking. 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 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 Gen-4 goals. Uranium oxide fuel is highly refined and dominates the current once-thru commercial fuel cycle and is qualified for use in fast neutron spectrum reactors. Nonetheless new Gen-4 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 Gen-4 research program.

2. Cost effective advanced recycle technologies integrated with remote fuel refabrication technologies are the key enabling technologies for achieving

the Gen-4 sustainability goals. In the full recycle fuel cycles, the waste destined for the repository arises from losses in recycle/refabrication (or in recycle for molten salt reactors with fluid fuel) and from nowhere else; the burn-up achieved between recycle steps must therefore be large, and/or the trace losses during recycle must therefore be small. Recycle technologies which 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 for fissioning are favored; -- they extract the maximum energy from the ore resource, and they consign only the fission products to the waste. Such recycle technologies are under active development worldwide – based on both aqueous and on dry processes. 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 Gen-4 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 which minimize reject rates and rework are needed. Colocation of recycle and refab facilities are favored to minimize shipping and handling of radioactive materials in bulk form.

   "Dirty fuel" is easily suitable for use in fast spectrum reactors or possibly advanced thermal systems such as molten salt reactors and accelerator-driven thermal transmuters. However, the future global nuclear energy park is envisioned to likely rely on symbiosis of both fast and thermal reactor power plants – with recycle fuel fed back to thermal and fast spectrum reactors as well as to fast reactors. Therefore, the recycle, the fuel design, and reactor core design must all be coordinated such that thermal reactor neutron economy is not spoiled by contaminants in the fed back fuel feedstock, or so that the thermal systems are designed to accommodate the effects of contaminants. 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 Gen-4 R&D.

4. While *the importance of near-term siting, licensing, and deploying a geologic repository designed for the once-thru 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 which will 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 yield a significant benefit against cost 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 approaches: interim storage of spent fuel prior to geologic emplacement; partitioning/conditioning with interim storage of tailored waste forms prior to geologic emplacement; or partitioning and interim storage of Cs/Sr with reactor recycle for consumption by fission of actinides are the approaches available for preparation of the waste itself. Design of repositories tailored for heat management and tightly integrated with the above approaches on the waste preparation should be a priority development goal for Gen-4.

Gen-4 Fuel Cycle Crosscut Group

Attachment 1

November 1, 2001

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Appendix 1

Impact of Cost of Uranium on the Cost of Electricity in LWRs

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, the following section documents calculations of fuel costs for six different scenarios. The calculations shown below document the 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 for recovery from seawater 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 seawater is necessary, the tails assay would be decreased to reduce the natural uranium needed. I 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

$$\frac{E}{M_P} = \frac{x_P - x_F}{x_F - x_W}\phi_W + \phi_P - \frac{x_P - x_W}{x_F - x_W}\phi_F\,, \quad x_{P,F,W} = \textit{atomic fractions of feed, product and tails}$$

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. Security concerns would be driven more by the handling of nuclear materials than by concerns for theft. Therefore a 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 1 Impact of Uranium costs on the Cost of Electricity

Impact of Uranium Costs on the Cost of Electricity							
	Current Uranium Prices		Uranium from Seawater				
			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. I have assumed a nominal interest rate of 8 % per year.

Increment in Cost of Electricity

As can be seen in the last line of Table 9 the use of uranium recovered from seawater 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. 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.

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 10 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.

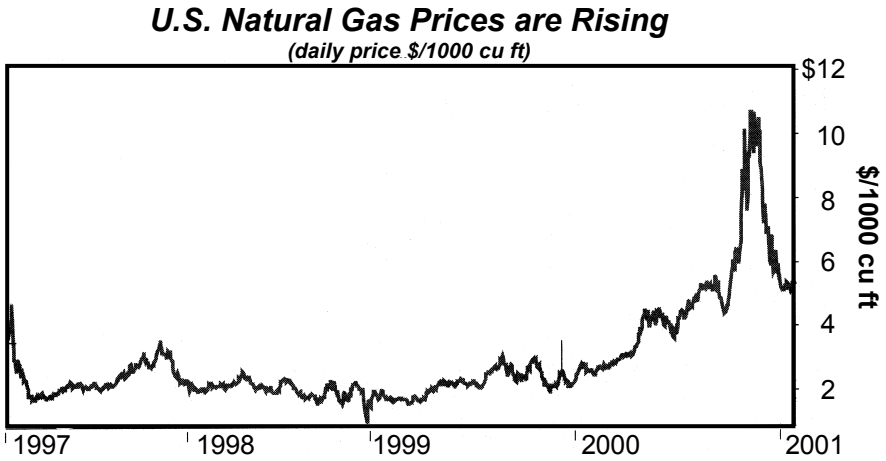

Figure 1 Natural Gas Price History 1997-2001

Table 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 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 would be less than what is routinely seen in fossil fuels, e.g. the price fluctuation of natural gas since 1998.

Were it to develop that technical and environmental acceptable methods became available to do so, the additional cost of uranium recovered from seawater, even under somewhat pessimistic costs of $1000 per kg U, would not add more than 30 % 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 price toward the more optimistic $200 per kg U.

No reactor built in the next couple of decades would ever be shut down due to uranium scarcity, regardless of its conversion ratio. None-the-less, 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. Recycle technologies must certainly be a part of Gen IV research.

Finally, the sustainability goals for the Gen-4 nuclear Fuel Cycle are addressed both the resource base and the responsible management the waste at the back-end of the fuel cycle. One motivation for technologies which make efficient use of the ore resource (recycling spent fuel) therefore are that these technologies help address the waste management aspect of the Gen-4 development goals.

Gen-4 Fuel Cycle Crosscut Group

Attachment 2

November 1, 2001

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Chapter 3 Appendix

A.1 The Dynamic Modeling of Nuclear Development (DYMOND) Code

A.2 Initial conditions for The Scenarios

A.3 Fuel Cycle Unit Process Costs

A.4 Fuel Cycle Facility Attribute Sets

A.5 Nuclear Power Plant Attribute Sets
- Attributes Used by DYMOND Code
- Attributes for Water Cooled (UOX and MOX) Reactors
- Attributes for Gas Cooled Reactors
- Attributes for Metal Cooled Reactors

A.1 The Dynamic Modeling of Nuclear Development (DYMOND) Code

The DYMOND code employs the ITHINK dynamic modeling platform to create 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 Red Book; 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 predictions produced in the World Energy Council/IIASA predictions. 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 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 it is spent - 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.

References:

1. A. Moisseytsev, “DYMOND, A Dynamic Model of Nuclear Development,” Argonne National Laboratory Internal Report, Aug, 2001.

A.2 Initial conditions for The Scenarios

The scenarios all start in the year 2000.

Table A.2-1 lists the initial conditions used in all scenarios for deployed facilities; these data are abstracted from the Chapter 2 detailed information.

Table A.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 A.2-2 lists the estimates[1] of world inventories of nuclear materials in storage in the year 2000.

Table A.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-thru cases. In the once-thru 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 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:

1. Estimates produced from OECD-NEA sources. These are based on plausible calculations and indirect evidence and should be viewed as informed estimates.

A.3 Fuel Cycle Unit Process Costs

Table A.3-1 lists the unit process costs for links in the once-thru 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 A.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 A.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.

Table A.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	40	60	80	$/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

†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 – lying between OECD-NEA consensus costs of spent fuel storage and depleted uranium storage.

Table A.3-2. Ore Cost Schedule
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	∞

Table A.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-thru fuel cycle.

Table A.3.3. Unit Process Costs For Particle Fuel Cycle Services

	PBMR	Prismatic
Mining Milling Conversion Enrichment Conversion	Same as Table A.3-1	
Pebble Fab		
Prismatic Fab Compacts Fuel Assemblies	(Assume same as LWR-UOX) Same as Table A.3-1	
Pebble Interim Storage		
Pebble Conditioning and Disposal		
Prismatic Assembly Interim Storage		
Prismatic Conditioning and Disposal		

Table A.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 A.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:

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).

A.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, thruput rates, fractions of thruput 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 A.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. (No prediction of future need and ordering construction in advance was modeled).

For PUREX plants, lag times were used and algorithms for forecasting furture 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 are required.

Some lag times neglected in the scenarios are indeed significant. 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 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).

Processing lag times as modeled in DYMOND are shown in Table A.4-2. UOX Fab 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.

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 Alloy is modeled as 4 years.

Discharged fuel assemblies destined for a geologic repository are held in interim storage for 30 years, then placed in the geologic repository. Waste forms generated in PUREX reprocessing are stored for 3 years, then placed in a geologic repository. The 3 year lag times apply for waste forms produced by pyro full TRU recycle. Table A.4-3 lists these waste lag times.

These waste interim storage lag times affect the fuel cycle cost index; as shown in Table A.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 A.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	5 + 2	Need for new deployments is based on predictions of forward need
Reprocessing (Pyro or Adv. PUREX) @ 50 to 200 tonnes/y	TBD	Available as needed
Interim Storage Facilities	TDB	Available as needed
Geologic Repositories	15-30 y	Available as needed

Table A.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			
Thruput 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 A.4-1

Table A.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)	3
Ceramic & Metallic Waste (Pyro)	3
Recovered Irradiated Uranium	∞
Enrichment Tails	∞

References

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).

A.5 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 Thru Concepts
- Limited Recycle Concepts
- Multi Recycle Concepts

because of different categories of fuel assembly types they use. Table A.5.1 lists the attribute set needed 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 A.5-1 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)

$$\frac{\text{kg U235 or U233}}{\text{kgIHM}} \quad \text{for thermal UOX}$$

or

$$\frac{\text{kg Pu239 + Pu 241}}{\text{kgIHM}} \quad \text{for thermal of fast MOX}$$

or

$$\frac{\text{kg TRU}}{\text{kg IHM}} \quad \text{for fast systems}$$

(A.) Attribute List for Once-Thru 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-4 Roadmap schedule has FCCG and TWG activities occurring in parallel rather than in series – with the result that the Gen-4 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 A.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-4 concepts sets.[1]

LWR's

Table A.5-2 lists the attributes used in the scenarios for LWR UOX once-thru 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 A.4-2.

Gas Reactors

Table A.5-3 lists the attributes used for the pebble bed[2] and prismatic[3] HTGR's fueled on enriched uranium and operating once-thru.

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[4] – it operates once-thru and achieves 87% transmutation of Pu239, and 62% burnup of transuranics overall at discharge.
In the scenario using this once-thru 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%.

Liquid Metal Fast Reactors

Four fast reactor attribute sets were used for the scenarios. All are sodium cooled. The first three were for interchangeable core loadings of 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.[5] 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 A.3-3 (blanket fabrication using depleted uranium is less expensive than fabrication of TRU-bearing drivers). These mass flows are given in table A.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

[1] The initial working inventories were sometimes approximated by summing the equilibrium cycle batch loadings – this is perhaps the biggest error in the approximations.

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 A.4-3 and are 4 years in total.

The fourth fast reactor was a breeder design of very short doubling time (7 y's) developed 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 A.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 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 A.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 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. Luc Van den Durpel, Personal Communication, Aug 2001.
2. WG—2 Draft Concept Summary Report for the MPBR Concept, C02.
3. OECD-NEA, "Trends in The Nuclear Fuel Cycle" – An Expert group Study, (to be published in Dec 2001)
4. General Atomics, MHTGR Plutonium Consumption Study Phase II Extension, FY-94 Final Report, Sept 30, 1994.
5. OECD-NEA, "Physics of Plutonium Recycling – Volume V: Plutonium Recycling in Fast Reactors, Appendix A: US Metal Fueled Benchmark Solution," (1996)
6. ANL-80-40, "Fast Breeder Reactor Studies," C. E. Till, Y. I. Chang, et al, July 1980.

Table A.5-2. LWR-UOX and LWR-MOX Attribute Sets (Partial Recycle)

	LWR-MOX Mono Recycle 30% Core Loading of MOX	LWR-MOX Mon 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
$\frac{\text{kg IHM}}{\text{MW}_{\text{th}}}$		
$\varepsilon \left(\frac{U^{235}}{HM} \text{ or } \frac{Pu^{239} + Pu^{241}}{HM} \right)$	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 $\left(\frac{\text{MW}_{\text{th}}\text{d}}{\text{kg IHM}} \right)$	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 A.5-3. Attribute Sets of gas Reactors (Once Thru)

		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
$\left(\frac{\text{kg IHM}}{\text{MW}_{\text{th}}}\right)$		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	$\left(\frac{\text{MW}_{\text{th}}\text{d}}{\text{kg IHM}}\right)$	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 A.5-3. 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)

Gen-4 Fuel Cycle Crosscut Group

Attachment 3: Supplemental Scenario Results

November 1, 2001

"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

3.1 Introduction

3.2 Fast Fissile Self Sufficient or Breeder With Doubling Time Longer Than The Growth Rate of Energy Demand

3.3 Penetration into Nonelectric Energy Sector: H_2 Production

3.5 Additional Scenarios Planned For Future Evaluation

Attachment 3: Supplemental Scenario Results

3.1 Introduction

This attachment contains the results of several scenarios which were developed during sensitivity studies during preparation of the "cornerstone" scenarios discussed in Chapter 3. They are documented here for supplemental information – with minimal elaboration and interpretation.

When they become available, the results of independent check calculations to help validate DYMOND will be documented here.

3.2 Fast Fissile Self Sufficient or Breeder with Doubling Time Longer Than The Growth Rate of Energy Demand

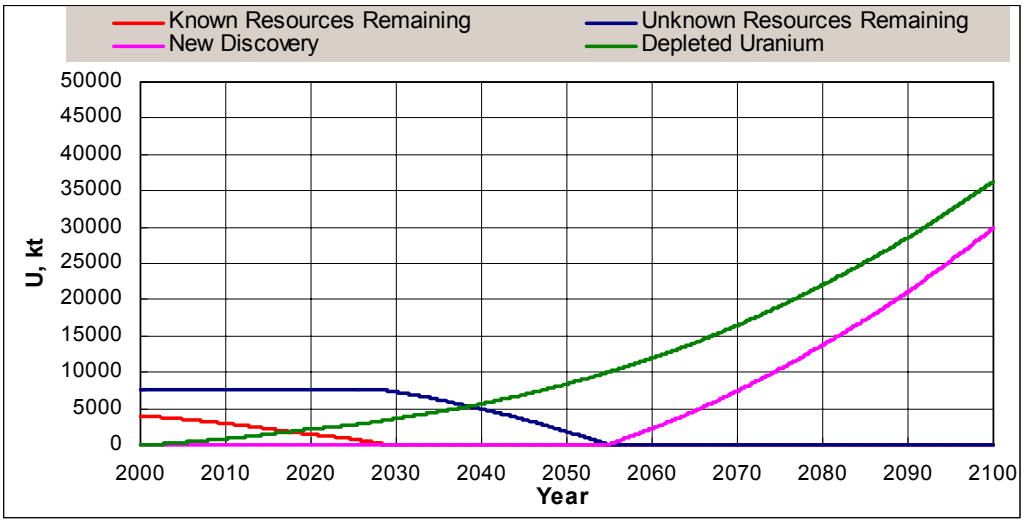
In developing the cornerstone cases discussed in Sections 3.5.1 and 3.5.2 fast reactors of BR=0.5, 1.0, and 1.25 operating in full TRU recycle were introduced in the 2020 to 2030 time frame at a rate determined by TRU availability from LWR spent fuel. LWR-UOX once-thru plants were built as required to fill the deficit in ability of fast reactors to meet energy demand.

Figures 16, 17 and 18 in Chapter 3 show the results for the BR=0.5 case. Figures 3.1, 3.2, and 3.3 below show the BR=1.0 case. Figures 3.4, 3.5, and 3.6 below show the BR=1.25 case.

These cases were similar to the LWR-MOX mono recycle case in that since the fast reactor doubling time is longer that the energy park doubling time, a time invariant energy ratio of LWR to FR plants establishes itself; ore reserves continue to supply the park, and waste to the repository reduces to fission products and trace losses of TRU from recycle.

In Chapter 3, Section 3.5.2 a high performance breeder (BR=1.72) is introduced by 2030 and since its doubling time is shorter than that of the energy park, it eventually achieves 100% market share, and thereafter ore reserve drawdown stops.

LWR + FR(BR = 1)


Known Resources Remaining
Unknown Resources Remaining
New Discovery
Depleted Uranium
U, kt
50000
45000
40000
35000
30000
25000
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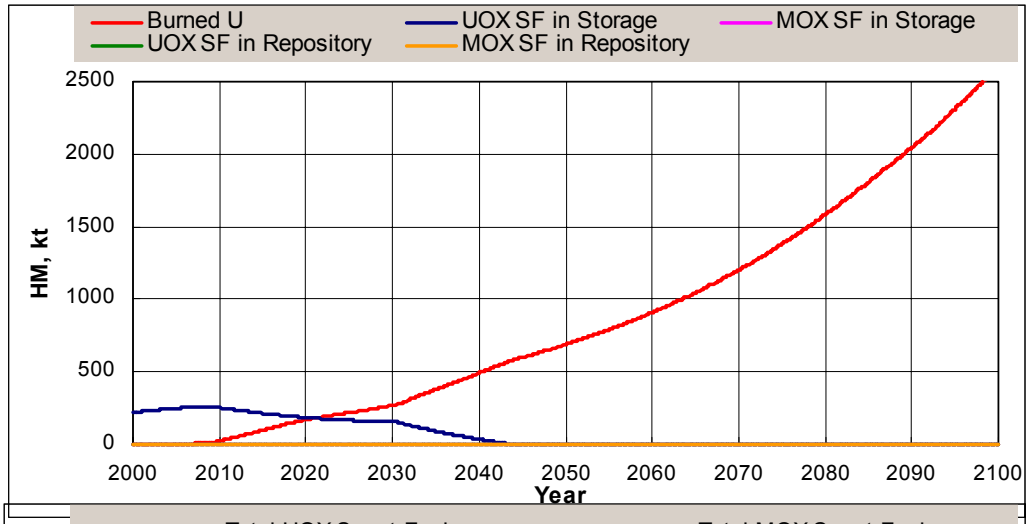
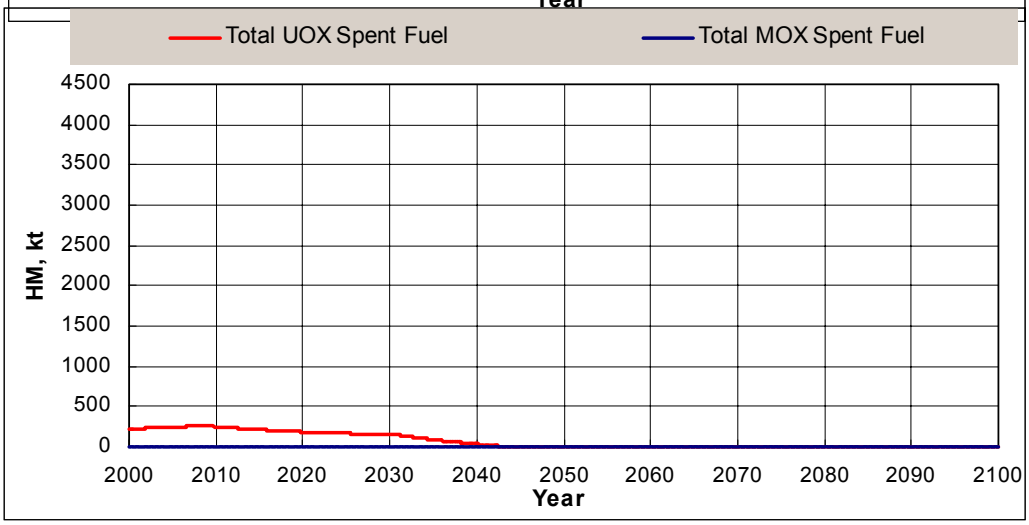
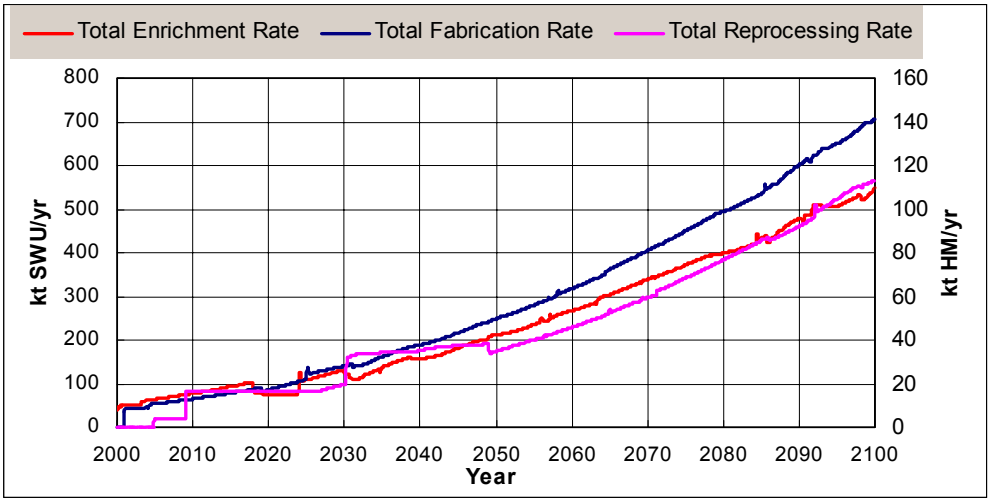
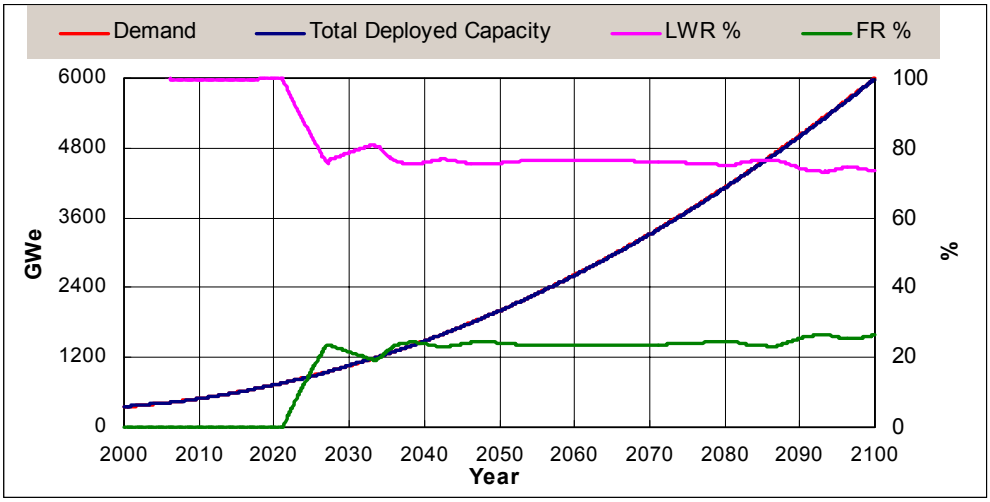
2100
Year

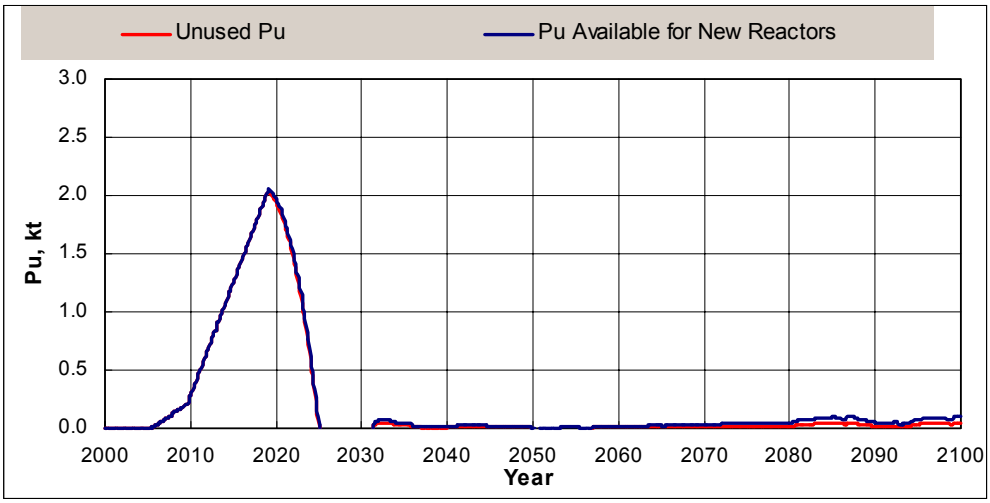

Burned U
UOX SF in Storage
MOX SF in Storage
UOX SF in Repository
MOX SF in Repository
HM, kt
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year


Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
4500
4000
3500
3000
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

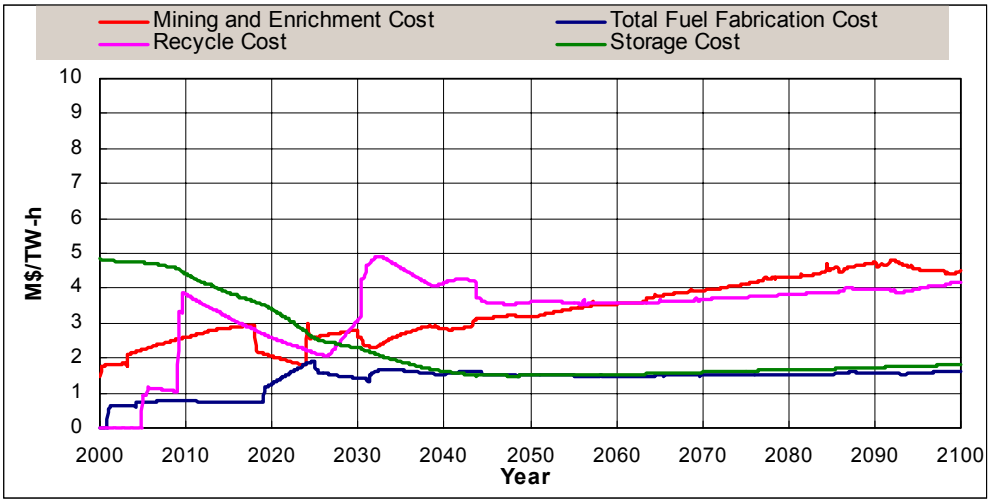
LWR + FR(BR = 1)

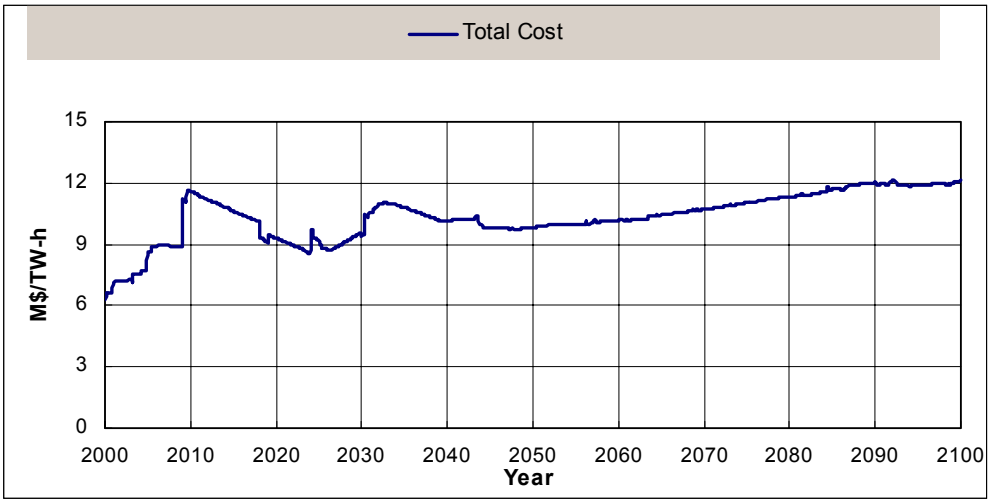

Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
0
100
200
300
400
500
600
700
800
0
20
40
60
80
100
120
140
160
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100

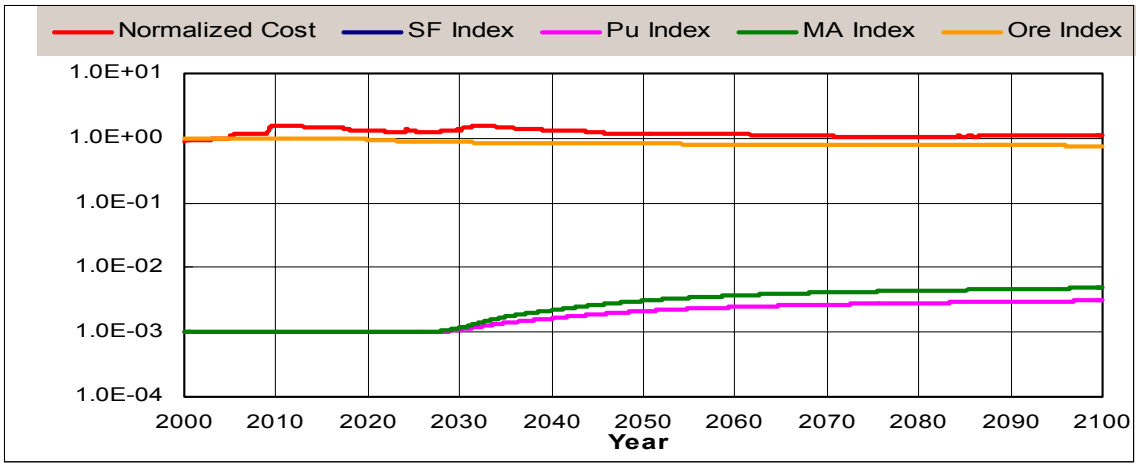

Demand
Total Deployed Capacity
LWR %
FR %
GWe
%
Year
0
1200
2400
3600
4800
6000
0
20
40
60
80
100
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100


Unused Pu
Pu Available for New Reactors
Pu, kt
Year
0.0
0.5
1.0
1.5
2.0
2.5
3.0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100

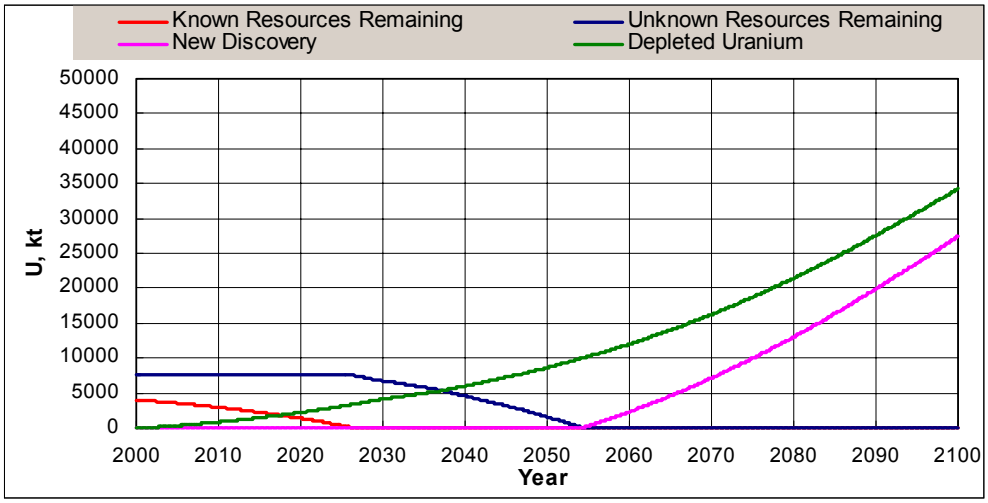
LWR + FR(BR = 1)

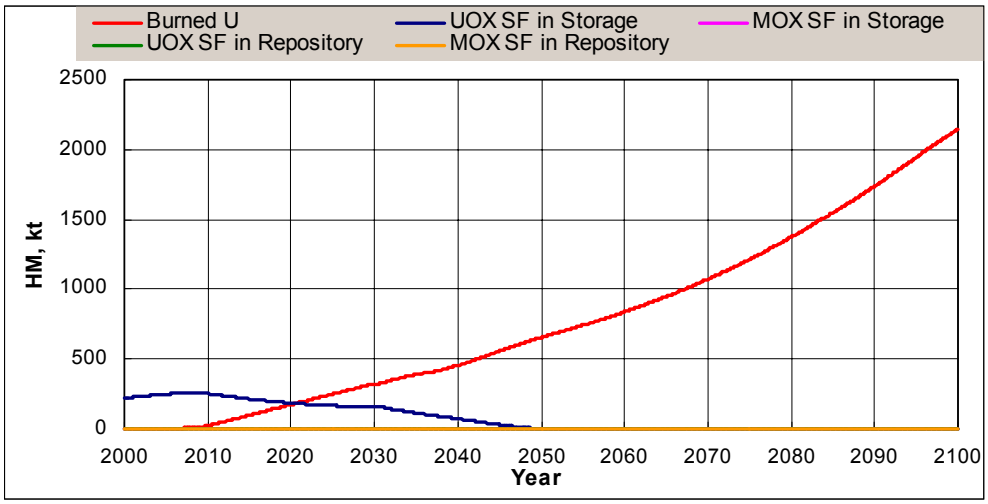

Mining and Enrichment Cost
Total Fuel Fabrication Cost
Recycle Cost
Storage Cost
M$/TW-h
0
1
2
3
4
5
6
7
8
9
10
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

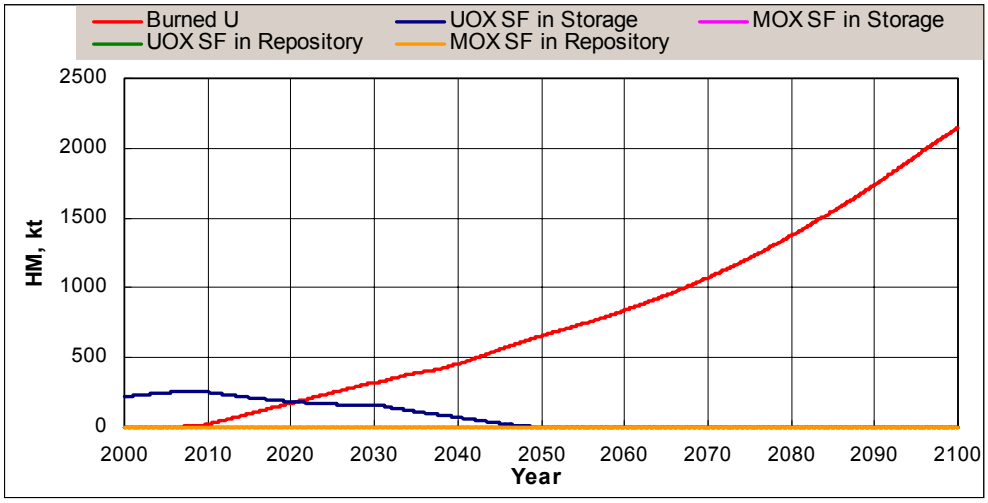

Total Cost
M$/TW-h
0
3
6
9
12
15
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year


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

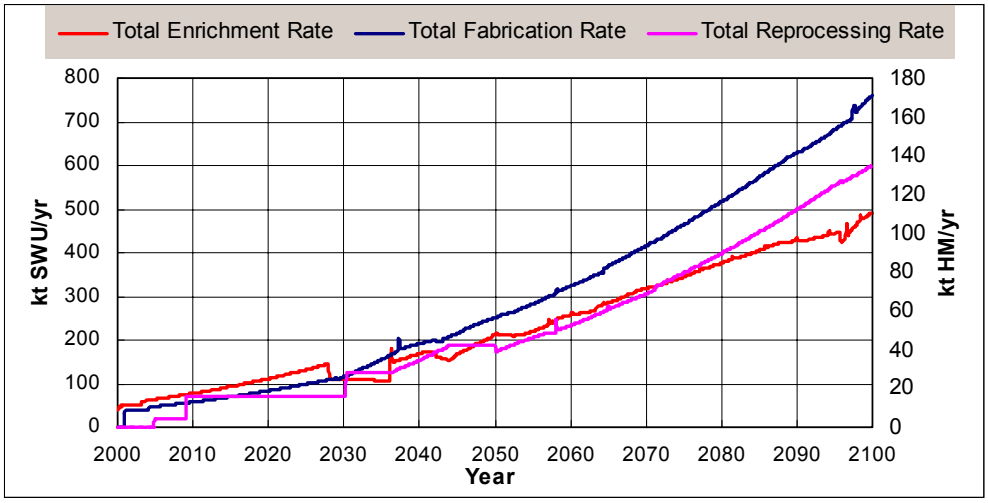
LWR + FR(BR = 1.25)

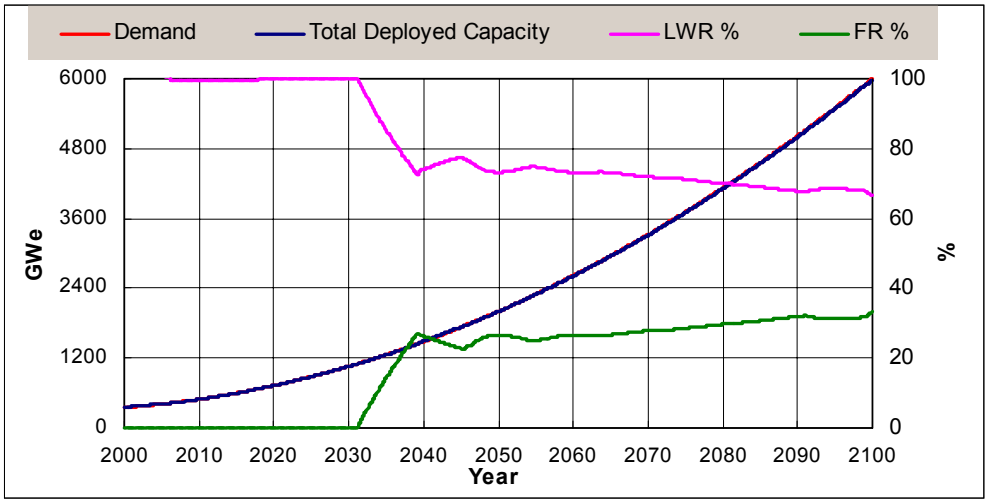

Known Resources Remaining
Unknown Resources Remaining
New Discovery
Depleted Uranium
U, kt
50000
45000
40000
35000
30000
25000
20000
15000
10000
5000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

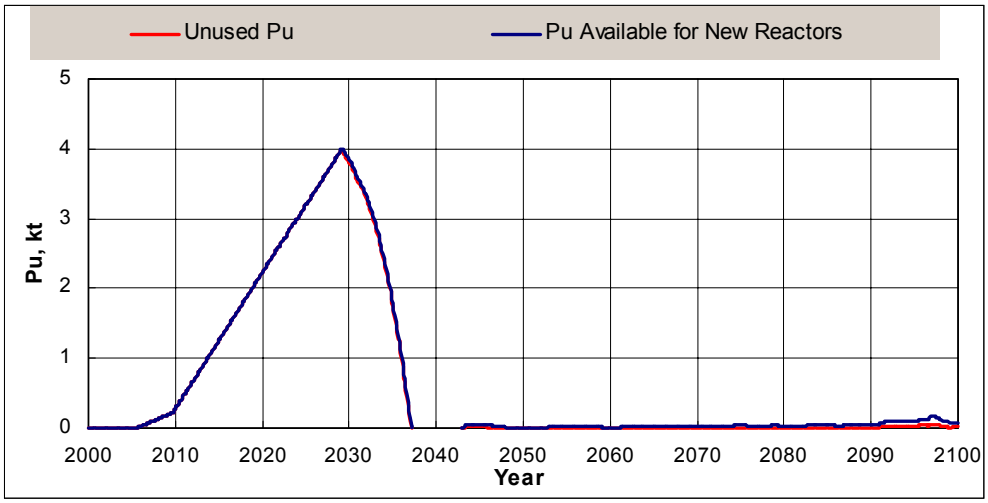

Burned U
UOX SF in Storage
MOX SF in Storage
UOX SF in Repository
MOX SF in Repository
HM, kt
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year


Burned U
UOX SF in Storage
MOX SF in Storage
UOX SF in Repository
MOX SF in Repository
HM, kt
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

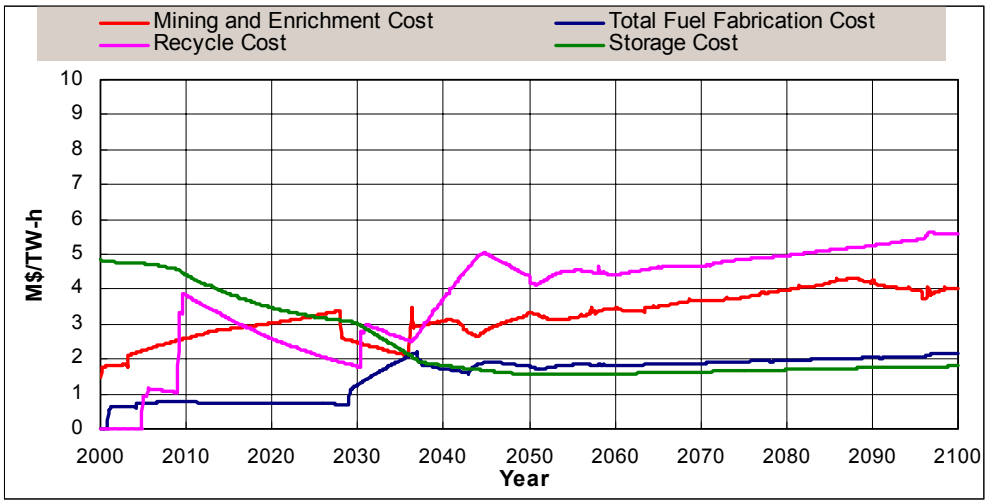
LWR + FR(BR = 1.25)

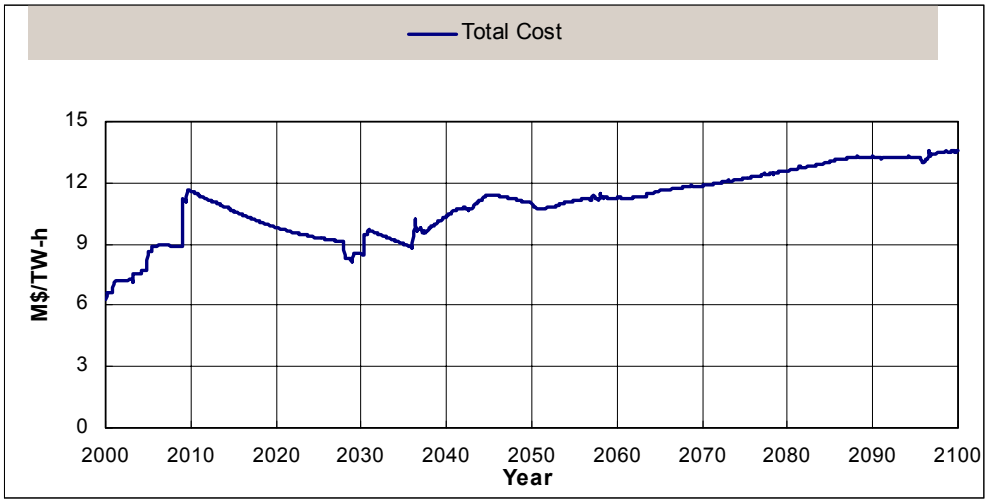

Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
Year
0
100
200
300
400
500
600
700
800
0
20
40
60
80
100
120
140
160
180
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100

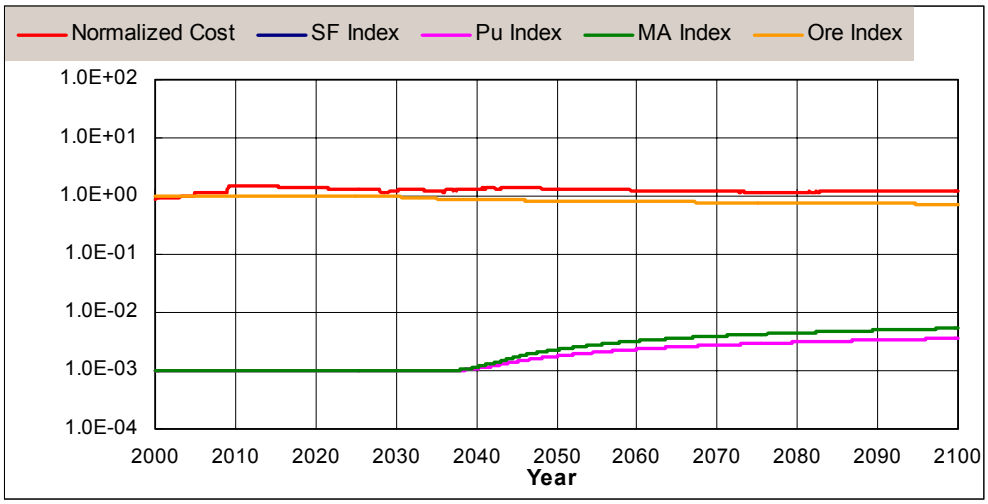

Demand
Total Deployed Capacity
LWR %
FR %
GWe
%
Year
0
1200
2400
3600
4800
6000
0
20
40
60
80
100
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100


Unused Pu
Pu Available for New Reactors
Pu, kt
Year
0
1
2
3
4
5
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100

LWR + FR(BR = 1.25)


Mining and Enrichment Cost
Total Fuel Fabrication Cost
Recycle Cost
Storage Cost
M$/TW-h
Year


Total Cost
M$/TW-h
Year


Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

3.3 Penetration Into Nonelectric Energy Sector: H_2 Production (Supplement To Chapter 3 Section 3.5.3)

In Chapter 3, Section 3.5.2 displays a case in which after achieving sufficient market share, a breeder generates excess fissile-which is fed back to fuel LWR-MOX power plants. This is a symbiotic park of fast and thermal reactors fueled entirely by U238 from enrichment tails and sending only fission products and trace losses of TRU from recycle to the geologic repository.

In the future, a need may arise for a broadened role for nuclear energy – supplying energy services in sectors beyond 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-4 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.

The next scenario serves to 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 *operating on a TRU-fueled, once-thru 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 the attachment.

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 3.7 thru 3.9 show the results of the scenario modeling. 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 equivalent of 2000 GWe of capacity is 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; and 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 the earlier 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 because (1) the HTGR once-thru 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 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 3.10 shows the breakdown of mass flows and of recycling and refabrication costs which (as shown in Fig. 3.9) 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-thru cycle [(see Chapter 3, Figs. 4, 5) 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 3.9. To produce a fuel cycle cost index normalized per terrawatt hour electrical, the expedient assumption was made that the HTGR's converted 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 are plausible and they serve to illustrate that *the cost index for fuel cycle services to support a fully sustainable Gen-4 nuclear energy future serving broad sectors of societal need are only four times current costs.* 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. (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.)

Case #1 – LWR + FR (BR=1.7) + HTGR=H2

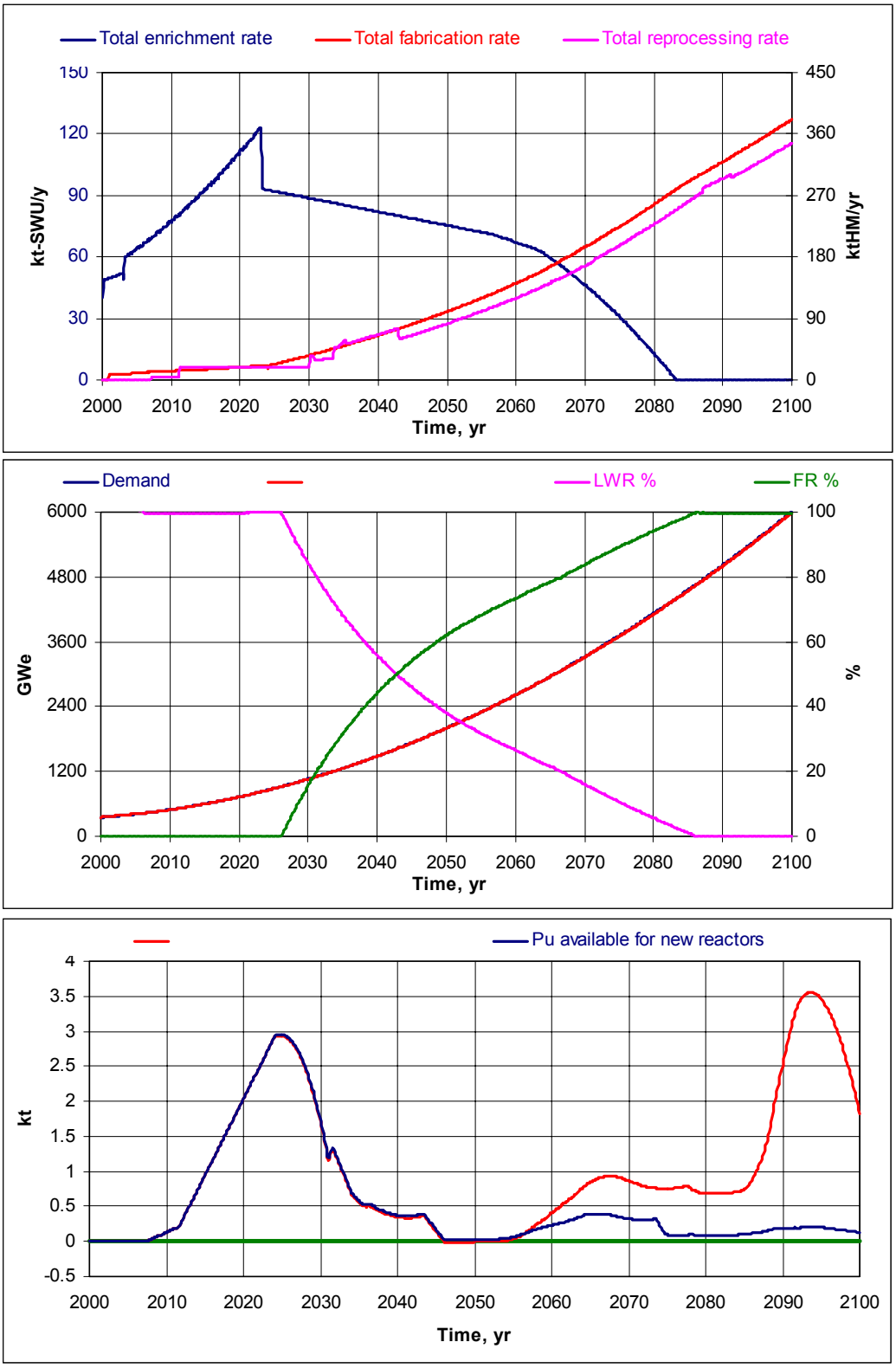


Fig. 3.7. Penetration Into Nonelectric Energy Sector

Case #1 – LWR + FR (R=1.7) + HTGR-H2

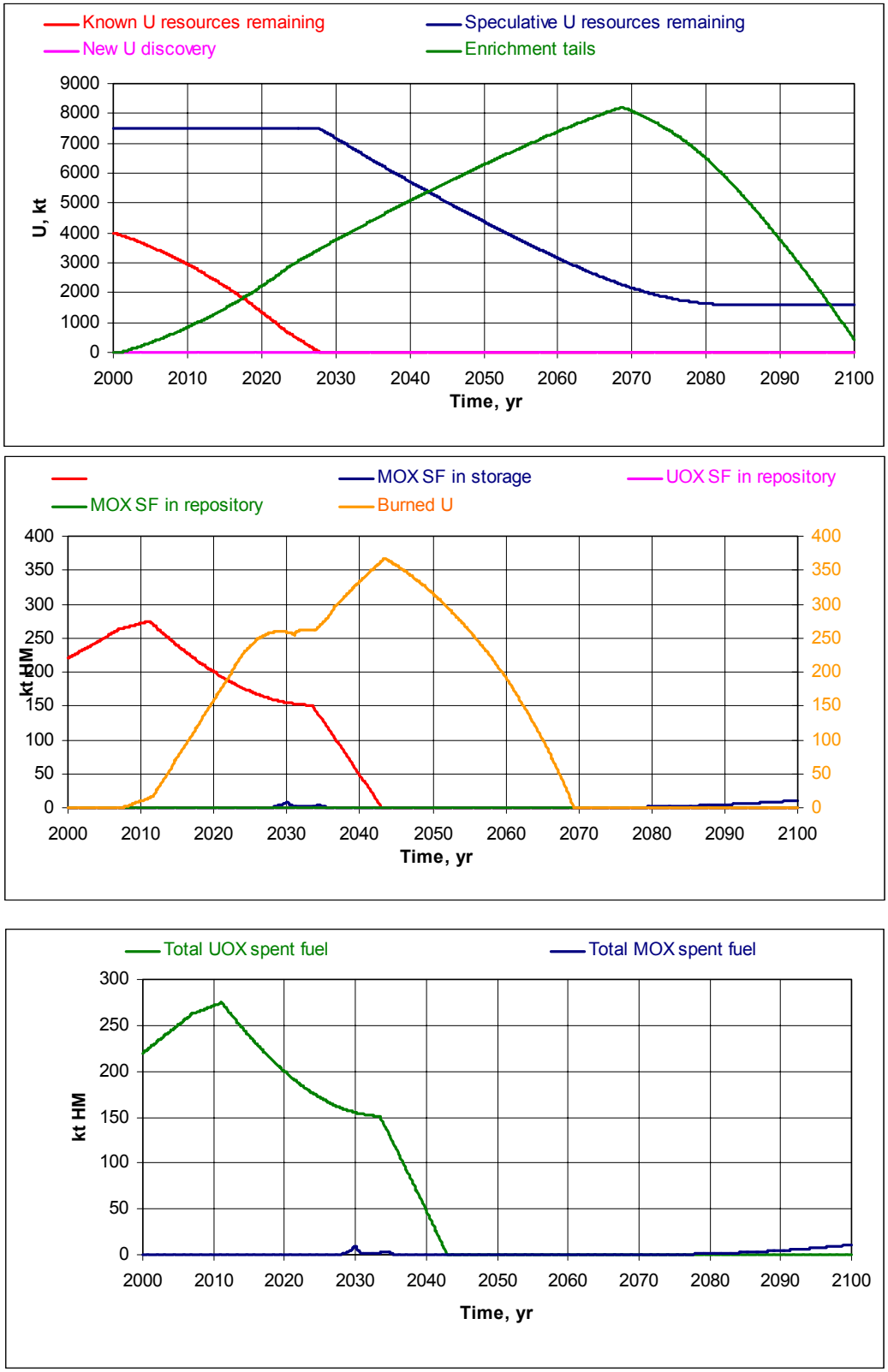


Fig. 3.8. Penetration Into Nonelectic Energy Sector

Case #1 – LWR + FR (BR=1.7) + HTGR-H2

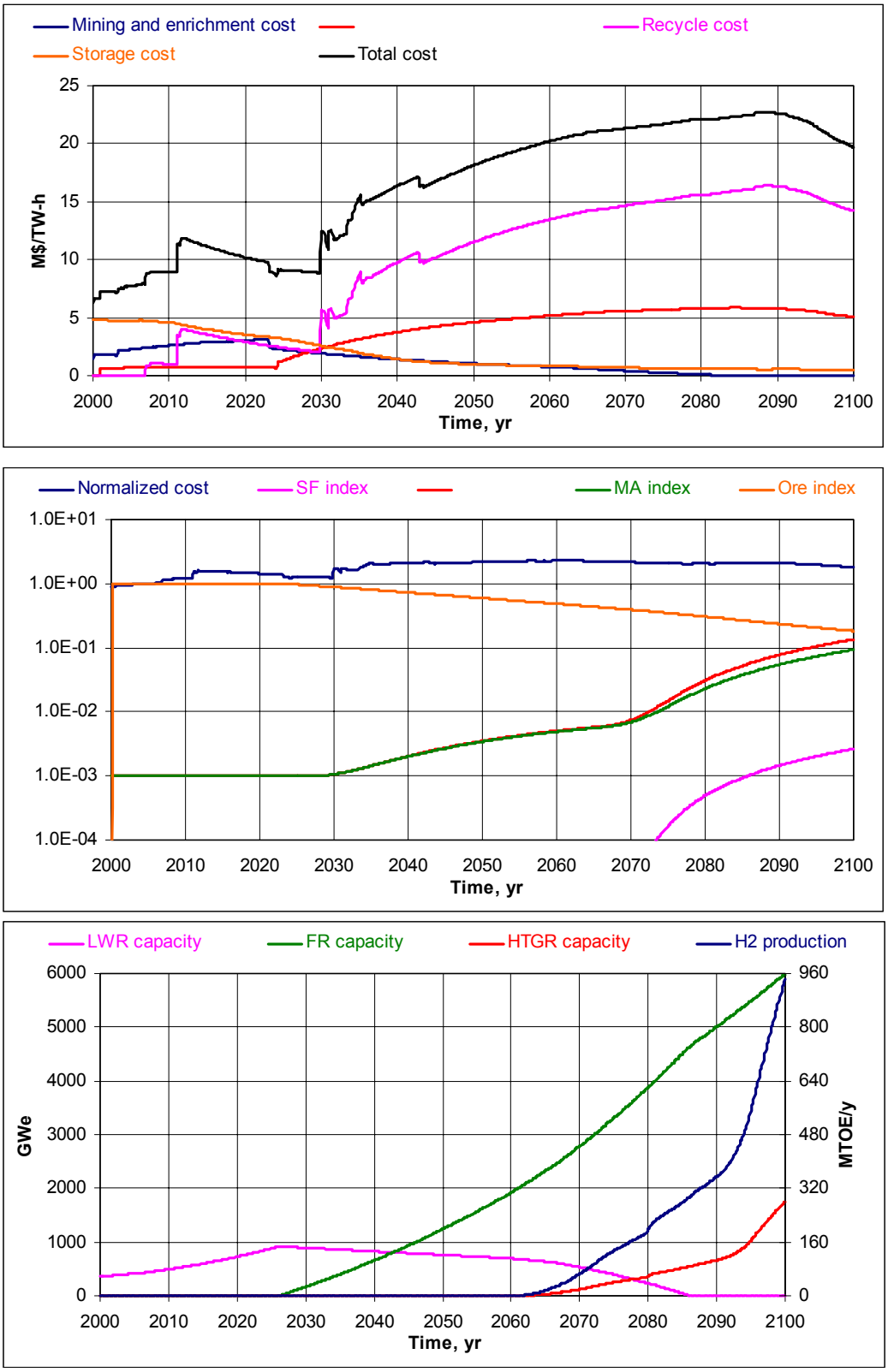


Fig. 3.9. Penetration Into Nonelectric Energy Sector

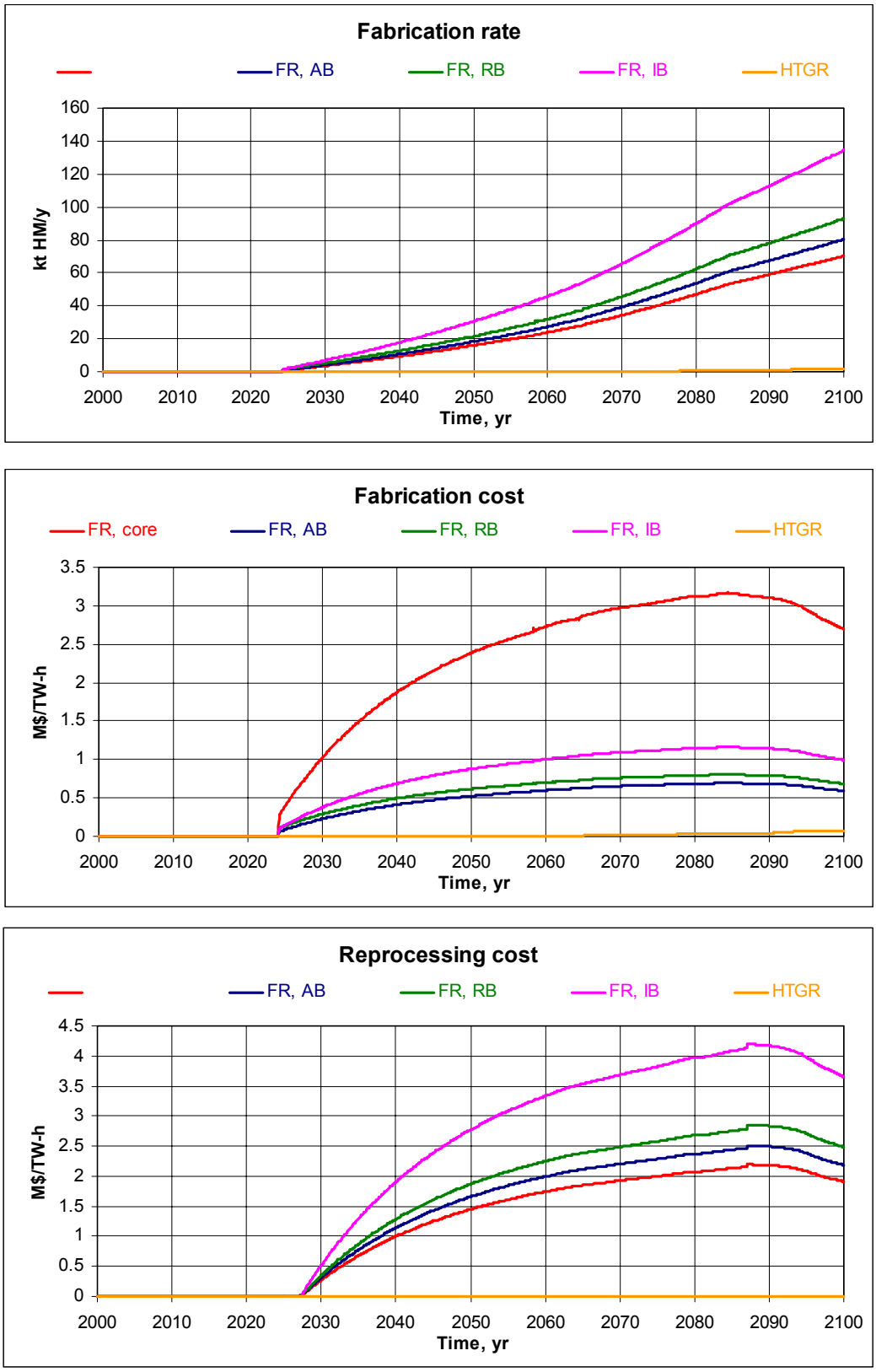


Fig. 3.10. Breeding Mass Flows and Costs

3.4 Additional Scenarios Planned for Future Evaluation

TWG Co-Chairs met with the FCCG on October 18, 19, 2001 to discuss coordination topics. Additional "cornerstone" scenario cases were identified; TWG's will provide attribute values to the FCCG; the FCCG will run the scenarios.

Short synopses are provided below to indicate the nature and rationale for these additional cornerstone cases.

Molten Salt Reactor with Full TRU Recycle

In this scenario, molten salt reactors (MSRs) with a conversion ratio of 1 replace light-water reactors. The MSR is introduced in 2030. MSRs fully replace LWRs by the year 2050. The startup of new MSRs is with enriched uranium from 2030 to 2050. After 2050, new MSRs are started up with recycle plutonium and minor actinides from LWR SNF. After startup of each MSR, each reactor is fueled with a mixture of thorium and depleted uranium. When a MSR is decommissioned, its fuel is used for the startup of the replacement MSR.

The proposed MSRs operate on a modified ^{233}U/thorium-fuel cycle with just enough ^{238}U to assure the ^{233}U are non-weapons usable by isotopic dilution. The MSR is designed as a thermal breeder reactor. Thermal neutron breeder reactors are only possible with a ^{233}U/thorium-fuel cycle. The breeding ratio of these machines under good conditions is only 1.06. If ^{238}U is added to convert the ^{233}U to non-weapons usable ^{233}U, the breeding ratio will only be slightly above one.

This scenario has several unusual characteristics. Like fast reactors, the fuel cycle that can fully use all fissile and fertile materials. However, the fissile inventory in the MSR is low compared to LWRs and fast reactors. The low inventory allows the use of (1) expensive uranium for startup of such reactors or (2) fissile materials in the SNF from one LWR over its lifetime to startup several MSRs.

There is no SNF from the MSRs. Fission products are removed from the salt and converted to a final waste form for the repository. The inventory of actinides in a MSR is small because the fuel cycle is primarily a ^{233}U/thorium-fuel cycle that generates only small quantities of higher actinides. The addition of the ^{238}U does generate some higher actinides. MSRs are minor actinide burners. The actinides remain in the salt until (1) fissioned or (2) converted to higher isotopes and then fissioned. The inventory of actinides is small and the thermal neutron flux results in very unusual actinide isotopic compositions. For example, ^{242}Pu are the primary plutonium isotope. Because of the technical difficulties and cost of removal of minor actinides from the fuel salt, MSR fuel cycles have always been assumed to burn minor actinides—although the minor actinide burning reduces the breeding ratio.

In the proposed scenario, MSRs are initially started on enriched uranium. After some experience is gained, latter MSRs are started on actinides from LWR SNF. All

LWR SNF is ultimately recycled for startup of MSRs; thus, no SNF goes to the repository. Since only two small experimental MSRs have been built (both in the 1960s), there are greater technical uncertainties with these fuel cycles than the more established fuel cycles associated with other scenarios.

(This scenario was chosen for inclusion in the FCCG report at the October18-19, 2001 Salt Lake meeting of the GEN-IV co-chairs. The scenario results are not included in this draft but will be included in the supplemental report.)

DUPIC Fuel Cycle For Partial Recycle of LWR SNF Into CANDU Reactors

This scenario involves the partial recycle of LWR SNF materials into CANDU reactors. The LWR SNF is mechanically separated into two major streams: (1) the uranium dioxide with fission products and actinides and (2) the SNF cladding. The LWR cladding is treated as waste. The uranium dioxide with fission products and actinides is fabricated into CANDU fuel assemblies. These fuel assemblies are used to fuel CANDU reactors and then disposed of as SNF in a geological repository. Because of the different neutronic characteristics of LWR and CANDU reactors, the fuel from LWR SNF can be used as fuel in CANDU reactors after refabrication to obtain the appropriate cladding and geometry.

The scenario assumes that starting in 2020, LWR SNF is recycled into CANDU reactors. By 2050, CANDU reactors are built at a sufficient rate that all LWR SNF is recycled 20 years after discharge from the reactor.

This fuel cycle option reduces the SNF and the quantities of fissile materials (^{235}U and plutonium) going to the repository. Unlike conventional recycle options, there are no chemical separation processes in the recycling system with high concentrations of weapons-usable materials. All weapons-usable materials are mixed with fission products. South Korea and Canada are currently investigating this process.

(This scenario was chosen for inclusion in the FCCG report at the October18-19, 2001 Salt Lake meeting of the GEN-IV co-chairs. The scenario results are not included in this draft but will be included in the supplemental report.)

Hydrogen Production With Once-Through LWR-UOX And Prismatic HTGR Fuel Cycles

This scenario is identical to the mixed LWR-UOX and Prismatic HTGR Once-Through case except additional HTGRs are built for the production of hydrogen for use as a fuel. Hydrogen production is assumed to start in 2020, equal 15% of the transportation fuel demand by 2050 and 20% of the transportation fuel demand by 2075 and thereafter.

The hydrogen production is assumed to be by thermochemical means with a 50% conversion efficiency of heat energy into hydrogen. All current thermochemical cycles require high temperatures; thus the HTGR is the chosen reactor type.

The choice of market penetration by hydrogen was based on how much hydrogen could be used in the production of traditional liquid fuels. Hydrogen penetration into the transport market would be considerably larger if hydrogen is directly used as a transport fuel. Hydrogen is currently used to convert crude oil streams into clean transport fuels (gasoline, diesel and jet fuel). The amount of hydrogen is variable and depends upon the transport fuel requirements. For carbon saver fuels, about 15% of the energy would come from the hydrogen. Carbon saver fuels [Forsberg 2001] are transport fuels where the hydrogen content of the fuel—within the constraints of conventional liquid fuels—is maximized to (1) minimize carbon dioxide releases from the fuel per unit of energy and (2) maximize the fuel produced per barrel of crude oil. If liquid transport fuels are made from coal, the hydrogen content is about 20% of the energy content of the liquid fuel. With conventional energy futures, coal will be used to make liquid fuels toward the end of the next century.

(This scenario was chosen for inclusion in the FCCG report at the October18-19, 2001 Salt Lake meeting of the GEN-IV co-chairs. The scenario results are not included in this draft but will be included in the supplemental report.)

C. W. Forsberg and K. L. Peddicord, "Hydrogen Production as a Major Nuclear Application" *Nuclear News*, **44** (10), pg 41-41, September 2001.

Once-through LWR Thorium Fuel Cycle

This scenario is a modified once-though LWR fuel cycle where a thorium fuel cycle begins to be adopted by the year 2020 and is fully adopted by 2040. In the modified fuel cycle, the traditional low-enriched uranium fresh fuel is replaced with a modified low-enriched uranium/thorium fuel. In effect, some of the ^{238}U in the LWR fuel is replaced with ^{232}Th. This reduces the production of plutonium and increases the production of ^{233}U. The ^{233}U is not weapons-usable because it is mixed with ^{238}U in sufficient quantities. For this scenario, the Radkowsky fuel cycle is chosen.

This fuel cycle has several potential advantages: (1) there is some reduction in uranium demand, (2) the quantity of plutonium in the SNF is reduced—a non-proliferation benefit and (3) thorium dioxide is expected to be a better waste form in the repository than the traditional LWR SNF.

(This scenario was chosen for inclusion in the FCCG report at the October18-19, 2001 Salt Lake meeting of the GEN-IV co-chairs. The scenario results are not included in this draft but will be included in the supplemental report.)

Gen IV – FCCG Report

Attachment 4

Review of Fuel Cycle Technology Status and Current R&D

November 1, 2001

"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"

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Chapter 4
Status of Technology and R&D Needs

4.1 Introduction

This chapter surveys 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.

As shown in Figure C.1-1 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 Gen 4 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, recycling, fuel refabrication, and waste form production) which are concept specific.

This chapter is organized by these links in the fuel cycle chain: in Sections 4.2, 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. In Sections 4.3 thru 4.5 the concept-specific links are discussed in the order: Thermal Reactor Fuels Technology (Fabrication, Irradiation Performance); Fast Reactor Fuels Technology then Recycling Technology (Separations, Waste Form Production, Waste Form Performance); and, as appropriate, Refabrication of Recycle Feedstock. The Waste Management Technologies are discussed in Section 4.6. The thorium-specific fuel cycle technologies are addressed together in Section 4.7, 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.

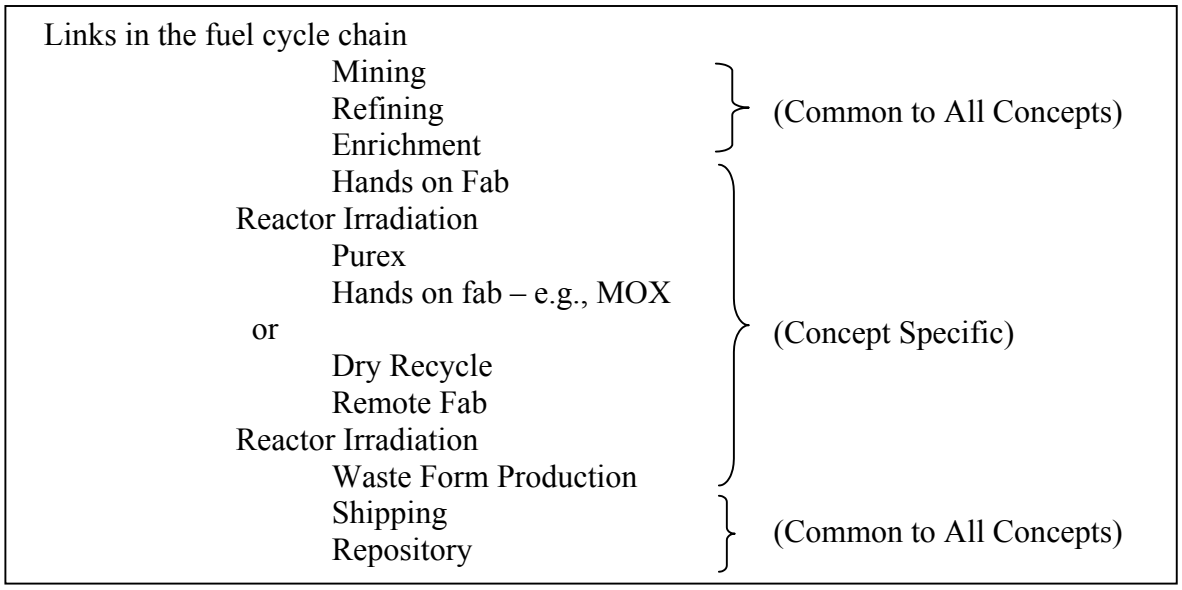

Figure 4.1-1. Links in the Fuel Cycle Chain

Sections 4.7 thru 4.11 discusses crosscut fuel cycle R&D needed to support each of the four generic fuel cycle types – once through; partial recycle; full Pu recycle; and full TRU recycle. In developing the recommended R&D, reference is made back to the results and insights from the scenarios presented in Chapter 3 as well as the current status as reviewed in this chapter.

4.2 Concept-Independent Front End Links In the Fuel Cycle

4.2.1 Mining/Milling

4.2.2 Enrichment

4.2.2.1 Chemical Exchange Separation of Uranium Isotopes

The chemical exchange method of isotope separation makes use of small variations in oxidation-reduction behavior of isotopes of uranium to provide a means of separation. In the United States, the Atomic Energy Commission determined in 1972 that the chemical exchange method was not economically competitive with the gaseous diffusion method [1,2], and U.S. development ceased. Since then, the developmental effort on the chemical exchange method has been carried out only in France and in Japan.

In France, the Chemex (Chemical Exchange) process based on solvent extraction using mixer-settler or pulsed column has been developed by the CEA, where the isotope exchange reaction $235U^{3+} + 238U^{4+} > 235U^{4+} + 238U^{3+}$ is the basic reaction. It was reported in the 1980s that the process would be in the commercial stage during the 1990s; however, there has been further report since that time.

On the other hand, Asahi Chemical Industry Co. Ltd. in Japan has been engaged in chemical exchange development, using anion exchange resin, since 1972 [3]. This section summarizes this process, which is termed the Asahi Chemical Enrichment Process (ACEP).

In the mid-1970s, ACEP was demonstrated basically using laboratory-scale apparatus, and by 1981 and 3% enriched U was recovered from natural U using continuous equipment. Based on these experimental results, a semi-commercial size column (100-cm dia. by 300-cm height) was built, and 17 kg of 3%-enriched U was produced. These results led to design of a commercial plant with 0.3 ~ 10 MSWU capacity, and it was assumed that a commercial plant had sufficient economic competitiveness.

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, U-235 has a tendency to be slightly oxidized (due to formation of U(VI) or UO_2^{2+}) and is adsorbed. In the ACEP process, uranyl nitrate solution is fed to a column filled with resin after oxidation solution feed. Uranus ion (U^{4+}) in the feeding solution is oxidized to uranyl ion (UO_2^{2+}), which is absorbed and a U absorption band forms in the column. Then the reduction solution is fed to reduce the uranyl ion (UO_2^{2+}) to uranus ion (U^{4+}), and the latter elutes from the resin downward through the column. During the phenomenon, the isotope exchange reaction occurs repeatedly. Finally, U-235 is collected at the upper part of the U absorption band. Crucial to the success of the technique was the invention of a new chemical process, which has more than 10^3 efficiency of selective absorption relative to

the conventional chemical process, and the development of new high temperature-resistant ion resin. [3,4,5]

Sufficient separation factor is expected with only one tower, which is the special characteristic of the technique; i.e., a cascade system is not necessary. Furthermore, as the system has no advanced mechanical equipment, operation and maintenance of the system is relatively simple. The fluctuation of concentration is expected to be small and the process control is expected to be simple. From the theoretical evaluation, and based on test results, it was shown that the capability of the process attains up to 500SWU/m^3. The estimated cost and amount of energy were about $41/SWU and 130~180kWh/SWU, respectively. Further details on the evaluation of the chemical exchange method are not available, because of their sensitivity

The main impediments to achieving an industrial-scale ACEP process were reduction of the required separation stage height, means of forming and retaining an effective U absorption band, and reduction in disturbance of the moving bed. In 2000, a report of the working group belonging to the Atomic Energy Commission of Japan to revise the "Long-Term Program for Research, Development and Utilization of Nuclear Energy determined that the chemical exchange method is probably sufficiently mature for commercialization.

4.2.2.2 Laser Isotope Separation

References for Section 4.2.2

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[2] Coates, J.H., et al.: Nucl. Eng. Int. vol.25, No.305, pp49-50 (1980).
[3] Miyake,T., et al.: "An Overview of the Asahi Chemical Enrichment Process (ACEP) and its Development", Bull. Res. Lab. Nucl. React., pp171-175 (1992).
[4] Seko, M., et al.: "Separation of uranium isotopes by the chemical exchange method", Sep. Sci. Tech.,vol.28 ,pp487-505 (1993).
[5] Seko,M., et al.,: "The chromatographic uranium enrichment process by Asahi Chemical", AIChE Symp. Ser., vol.78, No.221, pp41-48 (1982).

4.2.3 Conversion

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

4.2.1 Uranium Oxide Fuels for Water Reactors

4.2.2 Mixed Oxide Fuels for Water Reactors

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, which is a mixture of plutonium and uranium oxides, 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 policy 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 may be taken as a basis for an overview of R&D perspectives on plutonium fuel performances and technologies.

Fuel fabrication

The basic process implemented by COGEMA in France for MOX fuel fabrication is the well known MIMAS process (for MIcronised MASter blend), which is very similar to the OCOM (Optimized CO-Milling) process previously used at the HANAU plant in Germany. It is a well-proven and qualified process at industrial-scale. Although alternatives to the MIMAS process exist, those alternatives are less industrialized and would be subject of considerable R&D programs. Further improvement to the MIMAS process is sought through R&D in the following topics:

- Productivity and maintenance of the plants
- Reduction of scrap and waste
- Ability to use various sources of UO_2 powder (produced by wet or dry processes)
- Fabrication and irradiation tests to support increasing the plutonium content up to the maximum allowed value of 12.5% (for LWR's) in order to cope with higher burn-up requirements.

MOX design and behavior

The major portion of R&D effort with plutonium 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 transients or accidental 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 particularly which 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 accidental conditions

This last topic is an important part of the research, which primarily deals with RIA (Reactivity Injection Accident) and LOCA (Loss Of Coolant Accident) events for high burnup fuels.

Beyond this domain of fuel performance and technology, 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, which may have significant impact on economy and performance at all stages of irradiated fuel processing (e.g., storage, transportation, recycling, etc.). Continuous, or even enhanced, R&D efforts must be devoted to achieving a better knowledge in this domain.

Recycling of plutonium fuels

The recycling of MOX fuel has been already accomplished with success in industrial facilities, such as La Hague (about 10 tons). However, it has never been implemented at a large scale, so some further engineering development will be necessary to achieve an efficient and optimized industrial process (including waste management). These issues are discussed in Section 4.5.2.

4.2.3 Coated Particle Fuels for Gas Reactors

Introduction

The development of the High-Temperature Reactor (HTR) has proceeded in two directions: a) the pebble bed concept in the Federal Republic of Germany and Russia (now also in China and South Africa), and b) the prismatic core 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 fuel 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 fueled 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.

Coated particles are in themselves miniature fuel elements on the order of a millimetre in diameter. A commercial reactor core contains between 10^9 and 10^{10} individual fuel particles. The coatings provide the primary barrier to fission product release. The very small size of coated particles is an advantage in testing, since statistically significant numbers of "fuel elements" can be tested. Individual tests typically contain 10^3 to 10^5 coated particles. Fuel performance in-service can be predicted with high degree of confidence through properly designed fuel development and test programs.

The two coated particle types most commonly used are:

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

Both BISO and TRISO particles are capable of nearly complete retention of gaseous fission products and iodine. Intact TRISO particles also provide improved retention of metallic fission products at current peak HTR design temperatures (Figure 4.3.3-1).

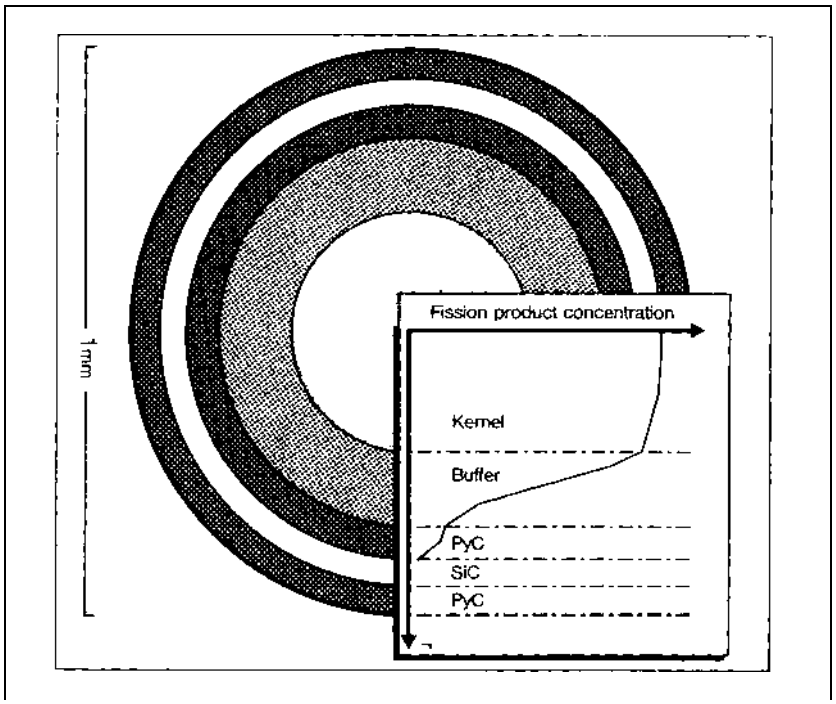

Figure 4.3.3-1. HTR coated fuel particle showing UO_2 kernel, buffer layer, inner pyrocarbon layer (PyC), silicon carbide layer and outer pyrocarbon layer.

The following three experimental reactors have been developed and successfully operated:

- AVR (Arbeitsgemeinschaft Versuchsreaktor): 1967-88, Pebble bed core, (Th,U)C2, $(Th,U)O_2$, BISO and (Th,U)O2 TRISO with high enriched uranium as well as low-enriched UO_2 TRISO coated particles; 46 MWth/15 MWe; power density: 2.6 MWth/m^3; Helium pressure: 1 MPa; He inlet/outlet temperature: 270/950°C.
- Peach Bottom 1 Reactor in US: 1967-74, Core with tube elements, (Th,U)C2, BISO; 115 MWth/40 MWe; power density: 8.3 MWth/m^3; He pressure: 2.5 MPa; He inlet/outlet temperature: 377/750°C.
- Dragon Reactor in the U.K.: 1968-75, Core with tube elements, various (UO2 driver fuel), TRISO; 20 MWth; power density: 14 MWth/m^3; He pressure: 1.0 MPa; He inlet/outlet temperature: 350/750°C.
- Fort St. Vrain in US: 1976-89, 842 MWth/330 MWe; power density: 6.3 MWth/m^3; Prismatic core; (Th,U)C2/ThC2, TRISO; He pressure: 4.5 MPa; He inlet/outlet temperature: 405/784°C.
- THTR: 1986-89, 750 MWth/300 MWe; power density: 6 MWth/m^3; Pebble bed core; (Th,U)O2, BISO; He pressure: 3.9 MPa; He inlet/outlet temperature: 270/750 °C.

All these HTRs have been operated using a conventional steam cycle. However, advanced designs with direct-cycle helium turbines and reformers for industrial process heat was under active development in Germany over two decades. Comparable HTR

research activities have been performed in the United States and Russia and for a process heat HTR in Japan.

Performance of Coated-Particles

Performance of a coated particle for HTRs is defined as the ability to retain fission products [1]. This ability can be affected by

- heavy metal contamination in the fuel element matrix introduced during;
- by particle defects during manufacture and particle failure during irradiation or during accidents;
- by fission product release from intact particles, e.g. by high temperature diffusion in kernel, coating and fuel element matrix material.

Modern fuel should satisfy criteria related to the following three topics:

- Manufacture
- Irradiation tests simulating normal operating conditions
- Heating tests simulating off-normal conditions

However, it might be considered beneficial to also consider fuel disposal options and characteristics when determining requirements for fuel. Therefore, disposal of coated particle fuels is also addressed in this report, although briefly.

Based on a combination of requirements for modern HTR designs, there is a consensus about the following main criteria for HTR fuel:

<u>Manfacturing criteria</u>

- burn-leach[1] fraction on particles $\leq 1x10^{-6}$ (necessary, but not sufficient)
- burn-leach fraction on fuel body $< 6x10^{-5}$ are required (GA $\leq 5x10^{-5}$ in final form).

Basically, coated particle manufacture by itself should be essentially perfect, and – within the limit of an acceptable quality control effort – this can be achieved. However, all the methods for making a fuel body (for example, cold isostatic sphere pressing with overcoated particles, warm pressing of annular compacts with overcoated particles, fuel rod hot injection) unavoidably break some particles contributing to a defect level in the range of $3x10^{-5}$ to $8x10^{-5}$ in modern fuel ($3x10^{-4}$ to $8x10^{-4}$ for fuel of the seventies).

<u>Irradiation performance criteria</u>

- in-reactor R/B $< 1x10^{-6}$ (necessary, but not sufficient), and

[1] The <u>burn-leach</u> consists of burning graphite and outer pyrocarbon at 800°C, leaching with nitric acid, and determination of uranium in the solution to detect defective and/or weak coatings.

- postirradiation examination results on fractional release of ^{137}Cs $F(^{137}Cs) < 2x10^{-5}$ and $F(^{110m}Ag) < 2x10^{-3}$ are required.

Irradiation conditions (i.e. reactor design) and particle properties shall be such that the fission product release contribution from the manufacturing defects shall not be significantly exceeded by irradiation effects. Therefore, postirradiation examination work is required to measure the release of solid fission products such as ^{137}Cs, ^{134}Cs, ^{106}Ru, ^{90}Sr, ^{154}Eu, and ^{110m}Ag. One unusual phenomenon is that ^{110m}Ag is released from intact particles by diffusion through coatings at irradiation temperatures > 1000°C.

Criteria for behavior during heating tests simulating accident conditions

- $F(^{85}Kr) < 2x10^{-6}$ (necessary, but not sufficient), and
- $F(^{137}Cs) < 2x10^{-4}$ are required.

The release of ^{131}I is the important consideration for accident consequences, but is difficult to measure due to its short half-life and easy deposition at cool surfaces prior to detection. A conservative indication for iodine release is krypton release. For HTR fuel investigated in accident condition heating tests, krypton can be directly released in the case of coated particle pressure vessel failure and/or after SiC deterioration first indicated by caesium release after diffusion through a still intact outer pyrocarbon layer.

Manufacture

Due to the different fuel element types, there are specific fabrication methods, as described in the following table:

Spherical Fuel Element	Annular Compact	Fuel Rod
cold isostatic pressing of overcoated particles	warm pressing of overcoated particles	hot injection, no overcoating
7-15% particle vol. Loading	33 % particle vol. loading	up to 60 % particle vol. loading
D, China, SA, Ru	J, UK	US, F, Ru

Particle manufacture can be made essentially perfect, but compact/sphere making still introduces defects, as shown in the German development [2] as well as in China [3] sphere production for HTR-10 and in Japan [4,5]. Burn-leach results are equally good with the manufacturing methods for spheres, compacts and rods, with relative defect fractions in the range of $3x10^{-5}$ to $8X10^{-5}$. The influence of overcoating of the particles is also essential for the quality of the fuel and has been improved by changing from manually controlled to automatic processes (see Figure 4.3.3-2).

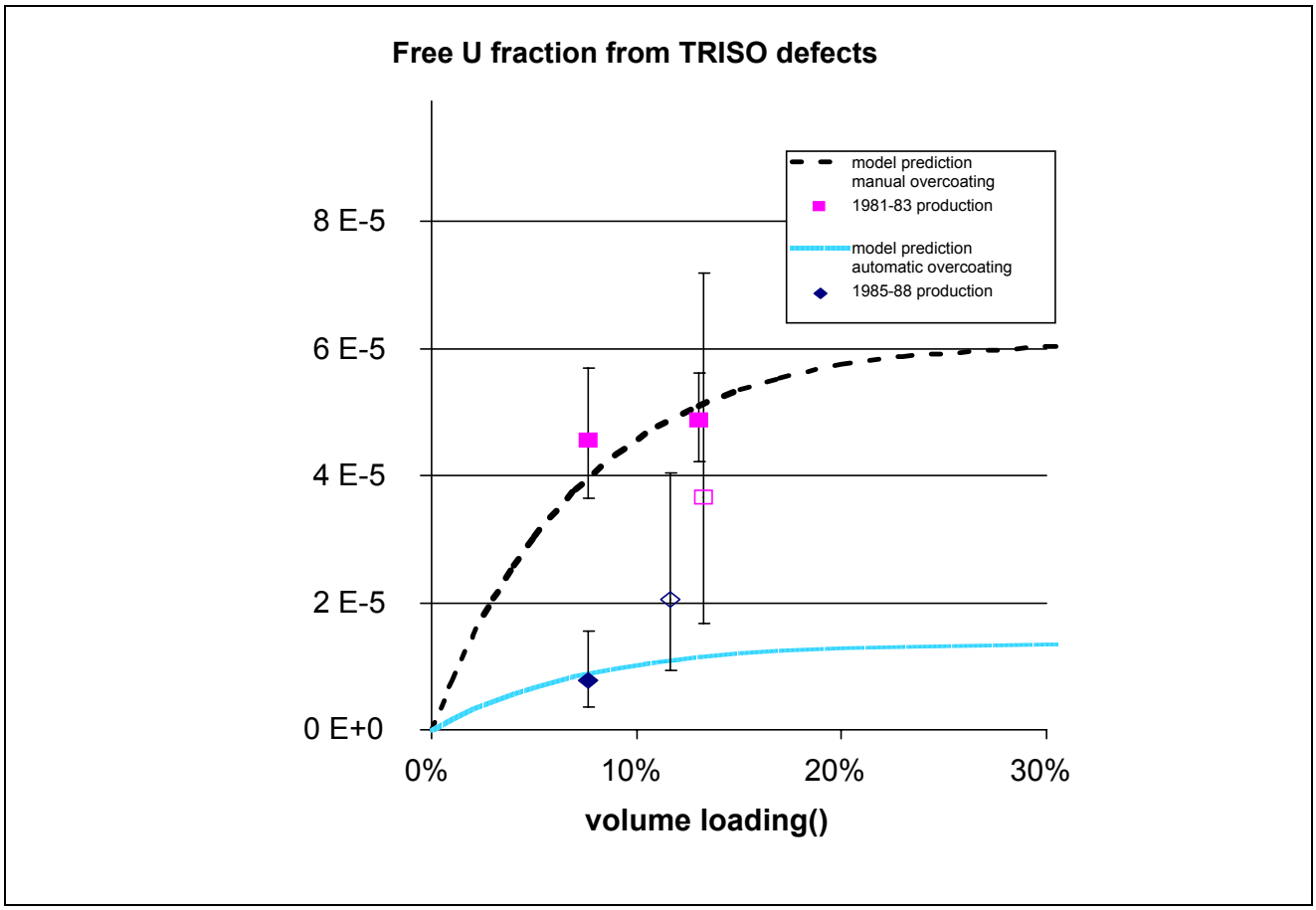

Figure 4.3.3-2. Fraction of LEU UO_2 TRISO defects during cold isostatic sphere pressing

Irradiation

To test in-reactor behaviour of particles, irradiation experiments have been performed in Material Test Reactors like the HFR Petten [6], Dido in Juelich, HFIR in Oak Ridge, ATR at INEEL, IVV-2M in Zarechny [7] and in AVR [8]. Not all HTR-specific parameters can be addressed in a single irradiation test, so there is a tendency to choose the following criteria for irradiation testing in approximate order of relevance:

- Temperature
- Burnup
- Fluence
- Power/ temperature gradients,
- Transients
- Real time

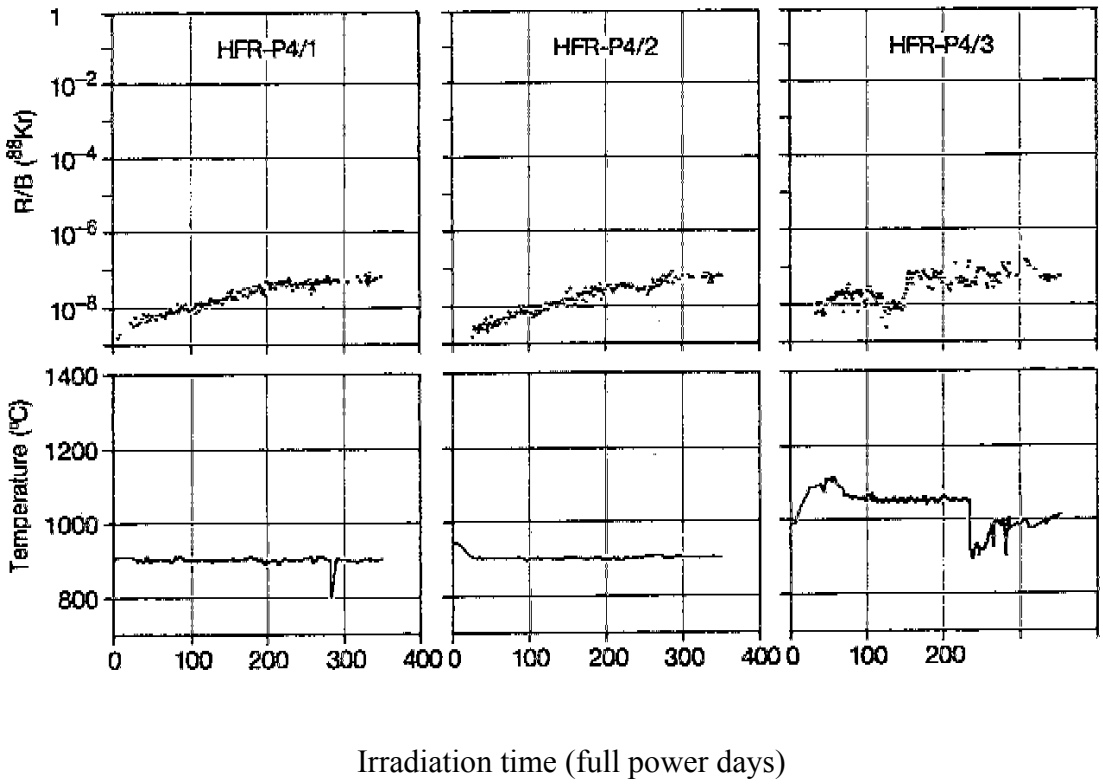

Figure 4.3.3-3. Irradiation test HFR-P4 with 19,560 particles in each leg of the Petten TRIO rig.

Figure 4.3.3-3 shows the release rate of short-lived fission gas ^{88}Kr during the irradiation of HTR fuel in the High Flux Reactor Petten and is typical of a large number of HTR fuel irradiation experiments conducted during in the 1980s and 1990s with high quality UO_2 TRISO particles [8,9,10]. In these sets of experiments, no manufacturing defects and no additional in-reactor failure could be observed, indicating that the requirement of $R/B<1x10^{-6}$ was easily satisfied. There is, however, the additional requirement that low levels of particle failure be demonstrated with a sufficiently large sample at sufficiently high burnup values. The statistical evaluation for the German test series on LEU UO_2 TRISO fuel is shown in Figure 4.3.3-4. When n failures are observed in a sample of size N, the one-sided upper confidence limit CONF is given by a failure fraction = BetaInv(CONF, n+1, N-n+1) and this is 1-exp[ln(1-CONF)/(N+1)] when n=0 as in all our irradiation tests. The figure also shows that there is still a great margin for enhancing the fuel burn-up beyond 80 GWd/t, which is still being taken as a design limit for LEU fuel.

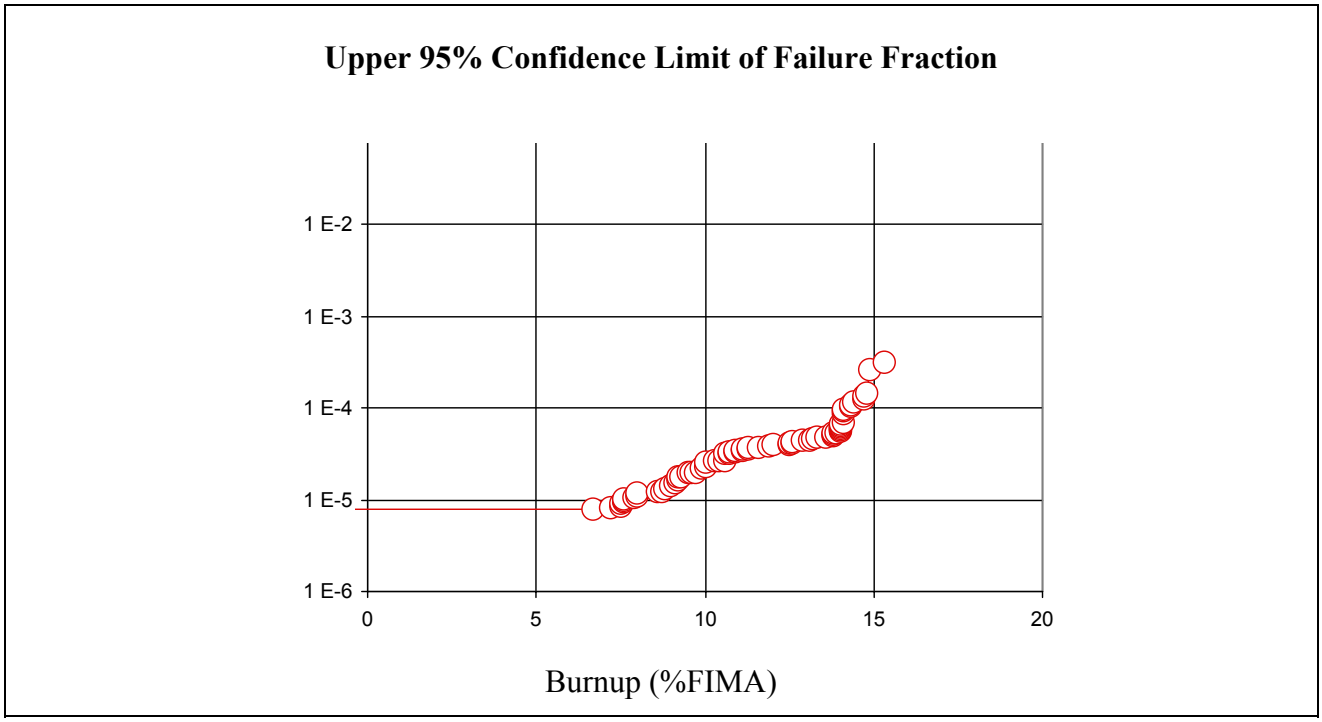

Figure 4.3.3-4. One-sided upper-95% confidence limit of in-reactor failure fraction when all 70 irradiation tests with UO_2 TRISO particles have shown zero in-reactor failure. It is planned to extend these limits significantly in the current EURATOM program with the irradiation of spherical fuel elements up to 20% FIMA [8, 13].

The burnup dependent failure curve in Figure 4.3.3-4 does not show a failure mechanism (after all, particles had remained intact in all cases). Rather, it displays the fact that there is less information available at higher burnups. Therefore, the confidence limits are larger at the high burnup range and this can only be improved by performing more tests at the high burnup range. This is, indeed, planned with the HFR-EU1 experiment in Petten within the European HTR-Technology Network (HTR-TN).

Ultimate particle coating quality is shown by the retention of solid fission products. Therefore, there is the need to measure the release of caesium, europium, strontium, zirconium and silver into components outside of the particle -- typically the graphite cups surrounding spherical fuel elements and the steel capsules of the irradiation rig. The increase of silver release with irradiation temperatures > 1000°C is noticeable. This is shown in Figure 4.3.3-5, which is a gross compilation of release data that does not take account of the variations in irradiation time and the space-time details of irradiation history.

The well-known silver release [11] from intact particles with perfect SiC coating has been of little concern for applications using high-enriched fuels and steam cycle systems. It is a potential concern with LEU fuel, direct-cycle applications and especially for

plutonium-based fuel due to the higher fission yield for silver. Therefore, the following is suggested:

- re-analysis of existing data (i.e., temperatures and silver releases);
- performance of dedicated irradiation tests;
- consideration of design/ material improvements.

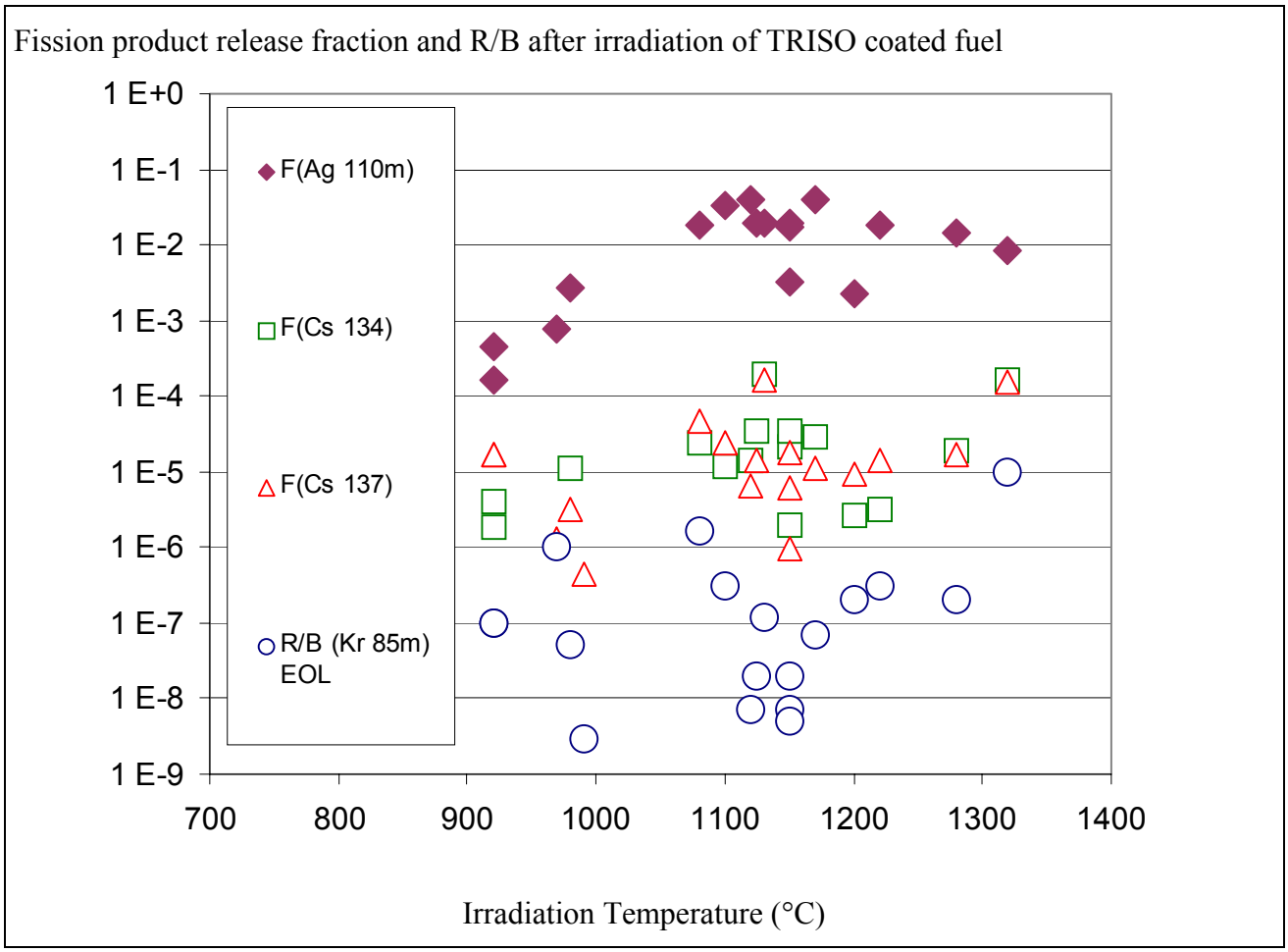

Figure 4.3.3-5. Diagram shows ^{110m}Ag and ^{137}Cs release fractions as a function of temperature in 230-530 day irradiation tests as measured in post-irradiation examination work.

Accident Condition Performance

Accident simulation tests have been performed since the mid-1970s in Germany, USA and Japan. The Forschungszentrum Juelich (FZJ) had concentrated on heating mainly complete spherical fuel elements. An early experimental program consisted of heatup ramp tests with $(Th,U)O_2$ BISO fuel at temperatures up to 2500°C. This program was followed by heating tests containing $(Th,U)O_2$ and UO_2 TRISO particles. Most of these were conducted at constant temperatures of 1600, 1700 or 1800°C, and some were fol-

lowing the MODUL accident scenario up to 1620°C. An overview of krypton releases is given in Figure 4.3.3-6, which indicates the fuel performance in terms of particle defects (manufacture) and accident-specific failure [12]. FZJ heating test results [13] have been used to demonstrate the safety inherent to the design of small modular reactors. More effort should now be spent in physical modelling of the heating test results to provide a higher degree of flexibility in reactor design and to aid the adaptation of the particle design to the needs of modular reactors.

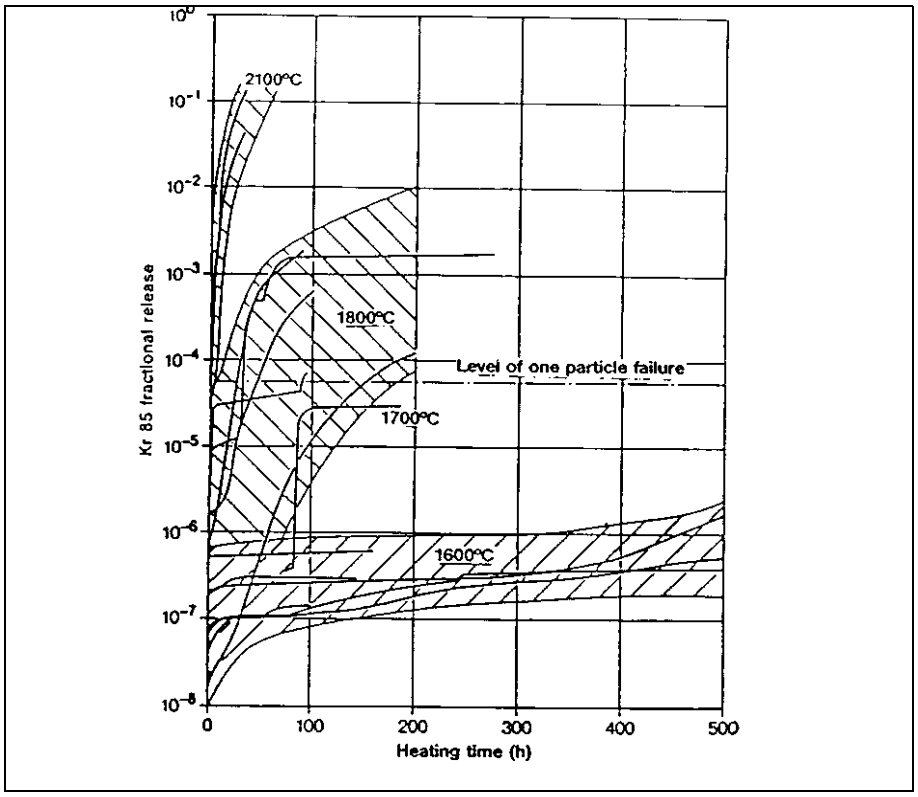

Figure 4.3.3-6. Caesium release from heated spheres as a function of heating times up to 500 hours.

Disposal of Spent HTR Fuel

Another important aspect of the HTR fuel elements is the capability for direct disposal of the spent fuel into a geologic repository. This is possible because the ceramic coating is rather resistant against leaching and corrosive attack. Furthermore, spent fuel encapsulated in coated particles appears to be rather proliferation resistant and can be compared with vitrified waste because the residual fissile material is similarily 'diluted' in the coating and in the fuel matrix. Disposal behavior of irradiated coated particles must be investigated with high precision measurements to allow reliable predictions. Disposal behavior characteristics will likely motivate further changes or enhancements to fuel design. Thus, the manufacturing of HTR fuel will have implications for final disposal as well as for irradiation performance.

R&D Requirement for Generation IV

Coated particles designed in the 1960s and 1970s were qualified in the 1980s and early 1990s primarily for steam cycle applications. These developments benefited from fuel improvements that were based on early direct cycle gas turbine concepts and which introduced a third layer into the particle coating sequence (TRISO). The features of this particle concept enabled the design of small inherently safe modular HTRs that retain fission products in the particle with self-limiting temperatures of < 1600°C even under core heat-up accidents.

In this context, it must be noted that the present coated particle design has not yet been optimised for the combination of high burnup and high temperature resistance under accident conditions. 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 the proposed modular applications, available results should be re-analyzed and re-investigated in the light of new requirements and envisioned. Specific work to be performed includes the following:

- Re-evaluate ^{110m}Ag release data for normal operating conditions to derive source term data that is applicable for direct cycle applications
- Determine influence of burnup effects on irradiation performance at burnup values above 10% FIMA, in particular for potential reduction of 1600°C capability.
- Analyze accident condition performance at temperature above 1600°C for an improved coated particle model.
- 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 behavior of irradiated coated particles using high precision measurements.

References for Section 4.3.3

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4.3.4 Advanced Fuels and Inert Matrix Fuels

While efficient and well mastered, 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 limit of 12.5% plutonium in the fuel. Another limitation is the proportion of MOX fuel that can be accomodated in present PWR cores (for example, 30% in today French 900 MWe NPPs). With this constraint (and because of some additional reasons) it is difficult to recycle the plutonium several times in large quantities.

To address those issues, innovative fuels are being investigated. Such fuels include nitride, metal or inert matrix options that may be combined with alternative geometries. To accomplish a strategy to quickly reduce plutonium inventories, removal of uranium from the fuel form allows the maximum possible plutonium-consumption rate, by eliminating conversion of fertile U-238 to plutonium. 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 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 intended to aid the multiple recycling of plutonium.

4.3.5 Molten Salt and Other Liquid Fuels

4.3.6 Summary

The large number of possible options makes the choice of advanced fuels for future nuclear energy systems not obvious at present. Oxides and metal are considered the most mature fuel materials, while nitrides and carbides appear to have promising potential. Oxide phases (including mixed transuranium oxide, inert matrix mixed oxide or thorium-based mixed oxide) have the advantage of high chemical stability and thus relative simple handling and fabrication. However, the relatively low thermal conductivity of oxide materials leads to a high operating temperature. Composites (ceramic-metal or ceramic-ceramic) may help to improve this. Nitride fuel, on the other hand, has much better thermal properties than oxide, providing a relatively lower operating temperature, and a fissile density nearly as high as that of metal fuel. Metal fuel has the advantages of high thermal conductivity, high fissile density and simpler fabrication, but has lower melting temperatures and often enhanced interdiffusivity with cladding materials.

State of the art and R&D needs

The state-of-the-art of the technology of the most relevant fuel forms for advanced reactor systems can be summarised as follows for the individual fuel forms:

- Oxide fuels have the general advantage that they are stable in air, which makes the safety requirements for the fabrication process less demanding. Three types of homogeneous oxide fuel have been discussed, mixed transuranium oxide, inert-matrix mixed oxide and thorium-based mixed oxide fuel. The MOX fabrication technology is developed at an industrial scale today (e.g. MELOX plant in France). However, the operating temperature of (mixed) oxide fuel is the expected to be high, caused by the relatively poor thermal conductivity, which may lead to limits in the linear rating. When thorium oxide is used as a matrix, this might be less significant.
- Nitride fuel has good thermal conductivity and the actinide nitrides show a mutual miscibility in a wide range of compositions. Nitride fuel is expected to have a good behaviour during normal operating conditions. The relatively high dissociation pressures of PuN and AmN at elevated temperature form, however, the major disadvantage of the mixed transuranium nitride fuel during off-normal operating conditions
- Metal fuel has interesting thermal and mechanical properties and fuel designs to deal with the extensive swelling of this fuel form exist. For uranium-based alloys, for

which the melting temperature is around 1350 ^{o}C, this gives a reasonable margin to melting. However, for Pu-rich alloys, the melting point and the thermal conductivity are significant lower. This problem can be overcome by addition of a non-fissile metal with a high(er) melting point and a high(er) thermal conductivity; a considerable amount of experience exists already with addition of Zr to U-Pu metallic fuels in the USA.

- Carbide fuels are, in spite of their good thermal and mechanical properties, considered of limited interest. The highly pyrophoric nature of these compounds requires strict safety measures for fabrication. The complex phase relations, especially between the sesqui- and monocarbides, seem to not facilitate the formation of a homogeneous fuel form and requires very specific fuel and core design in order to be able to safely operate nitride fuel.
- Composite fuels (ceramic-ceramic, ceramic-metal and metal-metal) have the advantage that the properties of the fuel can be tailored to a certain extent by a careful selection of materials. However, the number of relevant material combinations is limited, the most promising being MgO-$(Zr,An)O_{2-x}$ for ceramic-ceramic type, steel-$(Zr,An)O_{2-x}$ for ceramic-metal type and Zr-(An,Zr) for metal-metal type. It should be noted that the $(Zr,An)O_{2-x}$ oxide phase is favoured to the (An,Zr) metal phase. The fabrication of composite fuel is considerably more difficult, especially when specific the size and distribution of the dispersed actinide phase are required. Limited experience exists in this field. Also the irradiation behaviour forms a major uncertainty, especially the effect of locally high production of helium and fission gases on the mechanical behaviour.
- Coated particle fuel is essentially a special form of composite fuel. It is well known from the HTR development and offers the possibility of gas retention. However, this fuel type is not seriously proposed for fast-spectrum systems due to the moderating effect of graphite on the neutron spectrum and the unsatisfactory behaviour of graphite in a fast spectrum.

Considering the fuel development, mixed transuranium oxide fuel is the closest to current European MOX fabrication technology and for this reasons it is an important candidate fuel for future energy production. It has clear advantages with respect to fabrication and thermo-mechanical stability but less promising thermal properties may limit its application. Mixed nitride fuel forms a reasonable alternative for the (homogeneous) mixed oxide if the high temperature stability can be dealt with, for example by addition of ZrN. Composite fuel is considered the most innovative of these because the properties can be tailored to the application. However, the limitations of this fuel type need to be investigated thoroughly.

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 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 role fast reactors are considered for Generation IV due to the sustainability derived from the fast reactor's capability to either burn excess plutonium or to create a future fuel supply based on breeding, the discussion below is confined to addressing the status and R&D needs for mixed plutonium and uranium – bearing fast reactor fuels.

4.4.1 Mixed Oxide Fuels for Fast Reactors

Status of Oxide Technology

The development of mixed oxide fuel (PuO_2 – UO_2) was the cornerstone of Liquid Metal Reactor Programs around the world for over twenty years. This development culminated with the demonstration of high-burnup mixed oxide cores in the FFTF, PHENIX, MONJU, and PFR in the United States, France, Japan, and United Kingdom respectively. This was preceded by mixed oxide fuel testing in EBR-II, RAPSODIE, JOYO, and DFR. Mixed oxide was selected for this extensive testing 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. A summary of the European experience on fast reactor fuel development is shown in Table 4.4.1-1 [1].

Table 4.4.1-1. Summary of European Experience in Fast Reactor Fuel Development [1]

	Oxide	**Metal**	**Nitride**
Number of Pins	265,000	500	~ 30
Maximum Burnup (at %)	21.7	18.0	7.0
Burnup Achieved In	PFR	RAPSODIE	PHENIX
Maximum DPA	143	77	54
DPA Achieved In	PHENIX	DFR	PHENIX

The response of mixed oxide fuel to off-normal events has been extensively examined in TREAT in the United States and in CABRI and SCARABEE in France. These programs

included tests with fresh and irradiated fuel, as well as the development of analytical models to explain transient phenomenological behavior.

The economic incentive for a lower fuel cycle costs produced a continual improvement in mixed oxide fuel. Burnup progressively increased in reactors around the world as shown in Figure 4.4.1-1. This was part of an international quest to constantly improve performance while ensuring safety and minimizing fuel cycle cost.

In the United States, three cladding materials have been employed with mixed oxide fuel: 20% cold-worked 316 stainless steel, a modified stainless steel alloy D9 with improved swelling characteristics, and a very low swelling ferritic alloy HT9. Many fuel assemblies of each of these have been irradiated in FFTF. The effect of swelling as measured by duct elongation for each of these alloys is shown in Figure 4.4.1-2 (Leggett and Walters), which indicates the tremendous swelling resistance of HT9. Similar alloys have been developed in Europe. AIM1 and EM10 can be considered comparable to D9, and EM12 can be considered comparable to HT9. Even with these improvements, the maximum fluence still remains below the goals of some projects, such as that of the European Fast Reactor (EFR). In this case, the goal is 180 dpa or 3.6 10^{23} n / cm^2.

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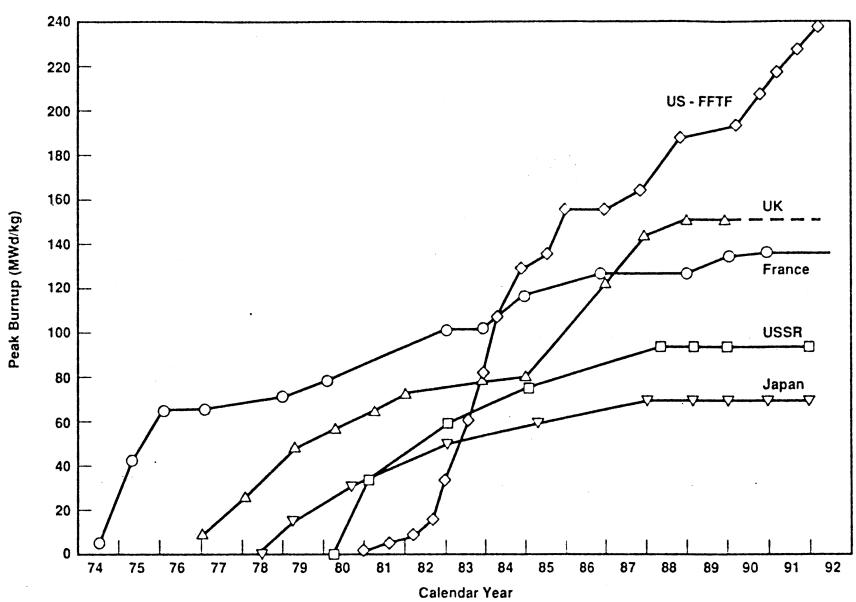

Figure 4.4.1-1. Burnup of mixed oxide fuel in Liquid Metal Reactors

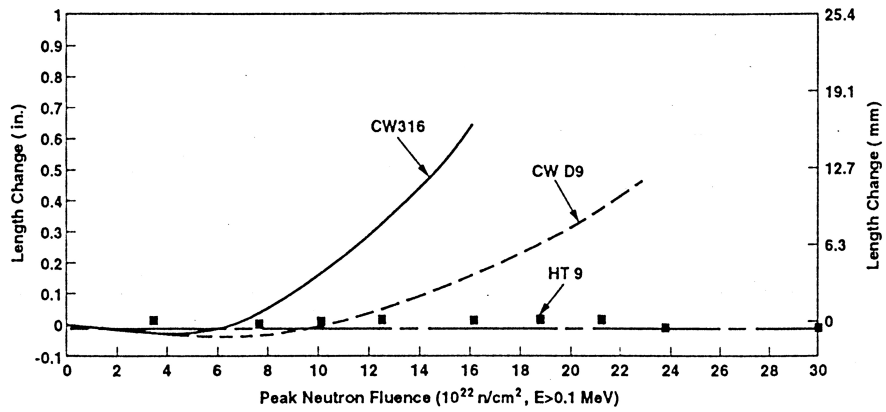

Figure 4.4.1-2. Assembly Elongation due to swelling in FFTF with different duct materials

Simplified Pelletizing Method for Fast Reactor-MOX Fuel Fabrication Developed by the JNC

1. Objective and progress of technology development [1]

The Japan Nuclear Cycle Development Institute (JNC) has been conducting research and development with MOX fuel fabrication for fast reactors (FRs). First, the Plutonium Fuel Production Facility (PFPF) with a capability of 5 ton-MOX/year was constructed in 1987 to demonstrate the mass production technology of MOX fuel, employing remote and automated operation technologies. Up to the present, the PFPF has fabricated more than 400 MOX fuel assemblies in total for the fast reactors Monju and Joyo, and the cumulative amount of MOX fuels has reached about 12 tons of MOX. MOX fuel pellet fabrication technology within glove boxes using high decontaminated materials from the current aqueous recycling plant is now well established through this fabrication experience.

However, simplification of the current MOX fuel fabrication process is essential to realizing a commercial FR fuel cycle with cost comparable to that of LWRs. Therefore, in 1999 the JNC initiated a program to determine, using laboratory-scale experiments, the fundamental process conditions for an advanced fuel fabrication process, termed the Simplified Pelletizing Method.

Further, JNC started a feasibility study on commercialized fast reactor cycle systems in cooperation with Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI) in July, 1999. The feasibility study is expected to indicate that the ability of FRs, to burn minor actinides and low-decontaminated fuel will reduce the recycling cost and to increase the proliferation-resistance by means of adoption of low decontamination factor in recycling. As a result, the nuclear materials are required to be handled in a hot-cell facility for production of fuel, fuel pins and subassemblies using completely automated equipment with remote-maintainability. The Simplified Pelletizing Method mentioned above is basically suited for in-cell fabrication because it would enable easier remote-maintenance for process equipment through drastic reduction in the number of process steps. Therefore, JNC adopted the Simplified Pelletizing Method as one of the promising candidates for MOX fuel fabrication technologies for a future FR cycle system, and initiated a conceptual design study with the Simplified Pelletizing Method for a commercial-scale fabrication plant.

2. Description of the process

The process flow diagram of Simplified Pelletizing Method is compared with the more familiar conventional process in figure 4.4.1-3. The subsequent pin loading and bundle assembly to be employed is the same as that used conventionally, without the pellet rework option, so the discussion here addresses only pellet fabrication.

In the advanced process, plutonium nitrate and uranium nitrate are mixed in a recycling plant to adjust the plutonium content to meet a fuel specification. The mixed solution is converted into MOX powder at a conversion facility by using the microwave direct denitration method (see Figure 4.4.1-4). The properties of the MOX powder are modified through appropriate calcinations and reduction to obtain certain flow characteristics that allow pelletization into green pellets without granulation. Nuclear materials generated from rejected pellets and pins are transferred to a wet-recovery process in the recycling plant.

After the realization of this advanced process, it will be possible to reduce the number of process steps compared to the conventional process, while achieving the same product quality.

3. Technical Features

By introducing the Simplified Pelletizing Method, the following benefits are expected.

- The fuel fabrication cost and the production of radioactive waste will be drastically reduced.
- By eliminating the powder blending and granulation steps from the conventional MOX pellet process, the mass of MOX powder falling and scattering in the process will be reduced. It results in increasing lifetime of process equipment because the dispersed MOX powder promotes the deterioration of cables, electronic equipments, etc. and decreasing personal radiation exposure during maintenance operation.

4. Current status of technical development [2]

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 recycling 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.

5. Technology to be developed

For the establishment of the Simplified Pelletizing Method, it is necessary to develop the fore-mentioned technologies. 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. Therefore the compact equipment concept of the conversion process includes microwave denitration and calcinations, which allows conversion from nitrate solution to MOX powder in the same container that is used to trasnfer the powder to the next step in the process. The equipment, illustrated in Figure 4.4.1-4, is based on a design philosophy that minimizes powder transfer in order to minimize dust generation and which incorporates simple mechanisms for remote-maintainability.

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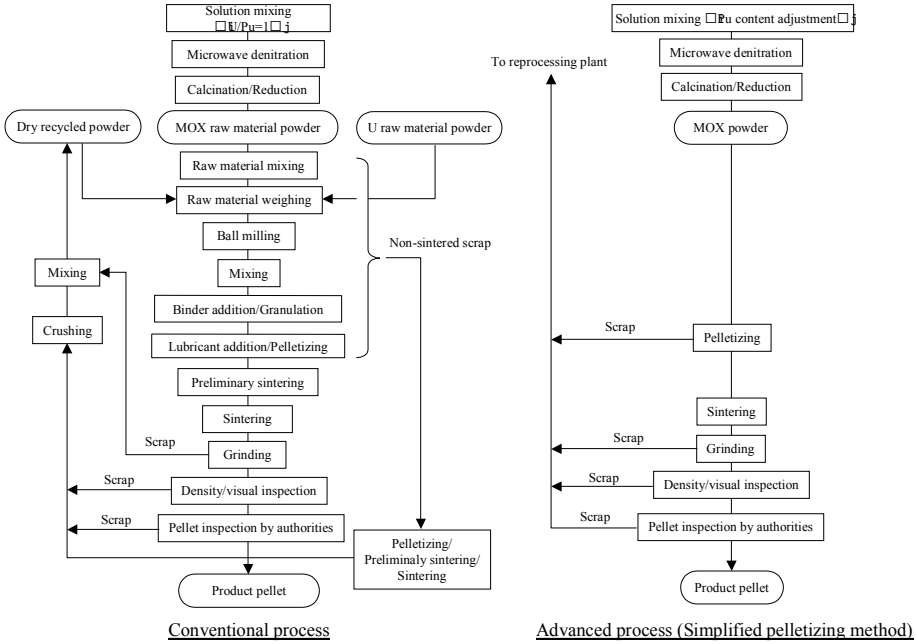

Fig.1 Comparison simplified pelletizing method with conventional one

Figure 4.4.1-3. Comparison of the Simplified Pelletizing Method with a conventional pellet fabrication process.

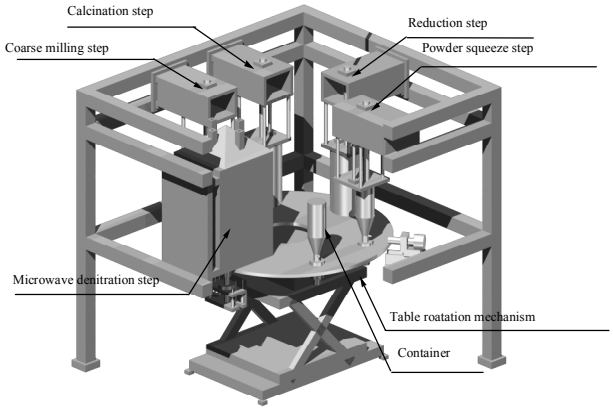

Fig.2 Turntable type denitration /calcination/reduction system

Figure 4.4.1-4. Turntable-type denitration/calcination/reduction system.

Vibration-Compaction Fuel Fabrication Technology

1. History of development

The vibration-compaction (or vibro-packing) technique of fuel fabrication was first attempted at Oak Ridge National Laboratory (ORNL) in the U.S.A in the late 1950s. At that time, the sol-gel process was developed for Thorium Oxide preparation. At that early stage of development, the particles obtained were irregular-shaped granules (similar to shards), which induced large stresses on the cladding tubes during compaction, which would degrade the cladding. For this reason, vibration-compaction fuel pin fabrication was thought to be difficult.

Later, in the early 1960s, the gel control technique that had been developed in the catalysis industries was applied to produce well-shaped, highly dense spherical particles, and vibro-packing of spherical particles with relatively low vibration energy was attempted. In Europe, fundamental research and development of the sol-gel method was activated in 1960s. In the 1970s, the gelation process was developed, by which it became possible to produce spherical UO_2 particles with diameter larger than 600 μm, which had been difficult using the sol-gel process. Since that time, many pilot plants have been constructed and operated, and also small scale irradiation experiments were performed in Italy, U.K., Germany and U.S.A., and elsewhere [1].

In the U.K,, irradiation experience reached about 700 pins in the Dounreay Fast Reactor (DFR) and about 3000 pins in the Prototype Fast Reactor (PFR) [2]. In the U.S.A., several pins were irradiated to burnup values as high as 12 at.% in EBR-II [3]. In Switzerland, carbide fuel was irradiated up to about 10 at.% burn-up [4]. In the late 1980s, as the development of the FBR was slowed, development activities of vibration compaction method also decreased.

In Japan, around the end of 1962, the predecessor of JNC initiated development of the sol-gel method. Development activity, however, did not reach full scale. Fundamental research was also conducted for ThO_2 fuel at the Japan Atomic Energy Research Institute (JAERI).

In Russia, development of vibration-compaction fuel fabrication combined with Oxide Electrowinning (non-aqueous recycling) started in 1970's at the Research Institute of Atomic Reactors (RIAR). In 1977, a pilot plant was constructed. Up to the present time, 426, 10 and 2 vibration-compaction MOX fuel assemblies were irradiated in BOR-60, BN-600 and BN-350, respectively, (about 18000 fuel pins in total). The maximum burnup attained in the BOR-60 with test pins of this type of fuel was about 32 at.% [5]. RIAR also has experience with production of granules sing vapor oxidation of fluorides [6].

2. Technology Concept

Gelation processes for aqueous recycling is categorized as either external gelation or internal gelation. In both cases, the process sequence is similar. Namely, a specific organic agent is added first to an aqueous solution containing heavy metals. Then, the solution is dropped as a droplet into another solution or bath. In the course of this process, the droplets become a spherical gel. In the Russian process applied to oxide-electrowinning, U and Pu dissolved in chloride salt solution are precipitated to an electrode as oxides, and such oxide granules are subsequently crushed and compacted into a fuel rod. Figure 4.4.1-5 provides a schematic of the two gelation processes.

The vibration technologies are categorized into following as either electrical vibration, pneumatic vibration. or electromagnetic vibration. The electromagnetic vibration method is applied to internal gelation at the Paul Scherer Institute (PSI) in Switzerland and to production of vibro-pack fuel in Russia using oxide electrowinning product. Figure 4.4.1-6 is a schematic illustration of the incorporation of vibration-compaction fuel fabrication into each of the various recycling schemes.

For the case of spherical, compacted fuel, the packing density is determined geometrically by the diametrical ratio of spherical particles and by the numbers of diameter sizes of particles. On the other hand, for the case of irregular-shaped granules, the edges of granules are ground by the vibration energy, which makes packing density higher.

The main development need for vibro-compacted fuel with spherical particles is the improvement of the smear density. For vibro-compacted fuel with irregular-shaped particles, it is important to find the vibration condition that minimizes the scars on the inner surface of cladding.

3. Characteristics of the technology

It is said that remote operation would be easily attained for sphere compacted fuel fabrication because the process does not include powder-handling steps, and as a result there will be no dispersion of fine powders.

Because the solution material of external gelation is thermally stable, the process of external gelation is considered to be simple and efficient. On the other hand, the amount of waste solution is produced largely and the sphere is likely to form a shell structure. Granules produced by the internal gelation method have a high degree of sphere-like shape and homogeneity; however, the solution material is thermally unstable such that appropriate cooling equipment is required. Further, the processing of used silicon-oil bath will be needed.

Vibration-compaction fuel fabrication by pyroelectrochemical processing is characterized by a small number of steps due to its combination with recycling, such that it would be suitable to a scale-up of the facility. On the other hand, there is a possibility that powder may be dispersed in the fabrication hot cell when granules are crushed, increasing

contamination. The use of chlorine for the dissolution of materials is taken as one of the inferior features of this method. In this process, oxygen getter made of metallic Uranium powders is used in order to control the oxygen potential in fuel rods.

4. Present status of technological development

Currently, technological development of vibration-compaction is underway in Switzerland, Russia and Japan etc. at the moment.

At PSI in Switzerland, research on vibration-compaction with spherical particles derived using the internal gelation method (termed "sphere-pack" fuel fabrication) is begin 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 pyroelectrochemical method and its irradiation in BOR-60 is being continued.

5. Items to be developed

With regard to the gelation method, the establishment of the optimum condition of gelation and treatment of waste solution remains to be solved. The means of solving such items and more efficient granulation method 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 both aqueous and non-aqueous recycling 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.

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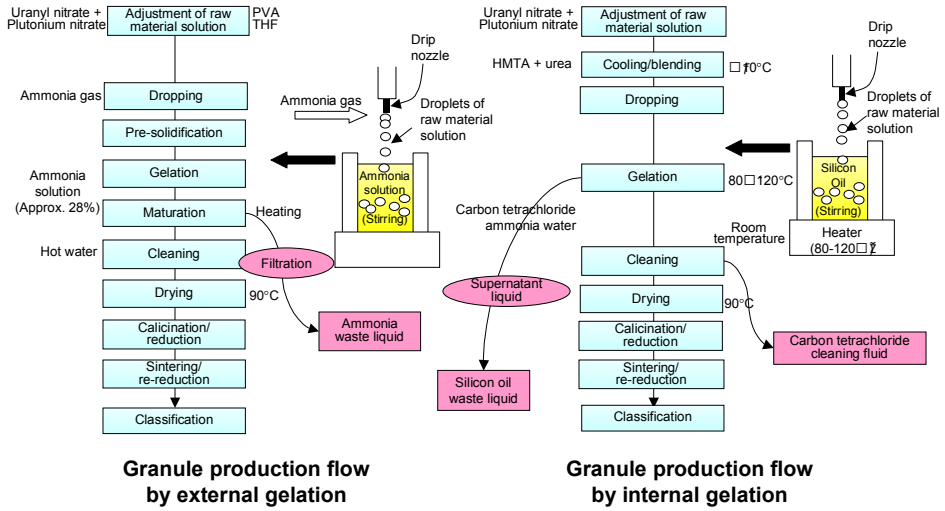
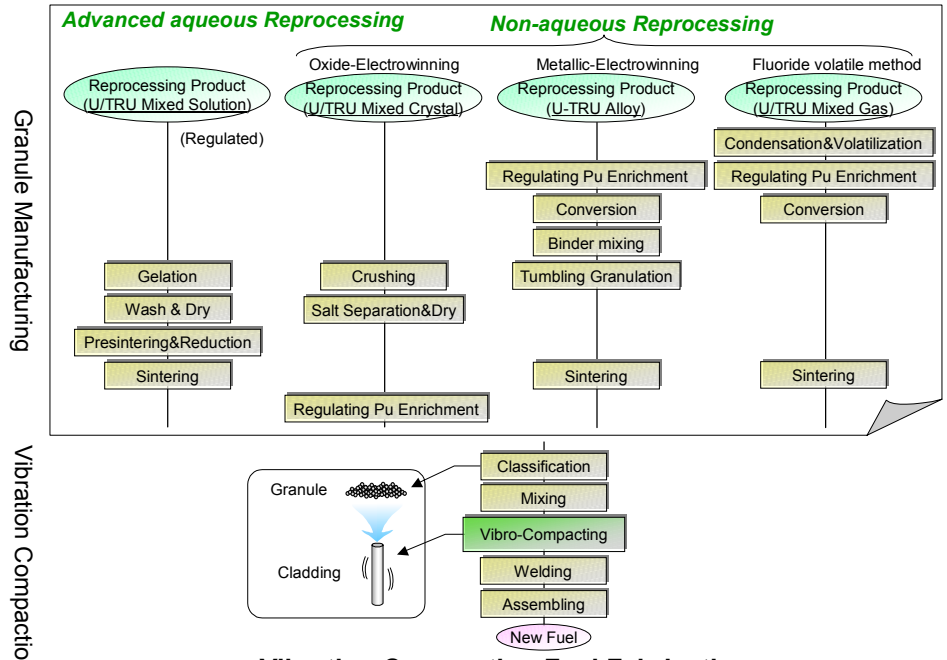


Particle Fabrication that Matches with Aqueous Reprocessing

Figure 4.4.1-5. Schematic illustration of external and internal gelation processes.



Vibration Compaction Fuel Fabrication

Figure 4.4.1-6. Schematic illustration of the incorporation of vibration-compaction fuel fabrication into various recycling methods.

R&D Related To Mixed Oxide Fuel Deployment

Given the extensive experience by 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.

4.4.2 Binary and Ternary Metal Alloy Fuels for Fast Reactors

Status of Metal Technology

Metal fuel was the first fuel used in fast reactors. The simple fabrication of metal and metal alloys, the high thermal conductivity, and the relatively high fissile density all made metal fuel attractive to early reactor designers. The Experimental Breeder Reactors I and II, the Dounreay Fast Reactor, and the Enrico Fermi Atomic Power Plant all used metal fuel. 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, 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 operate with metal driver fuel, and this reactor was the test bed for all other fast reactor fuels and materials; therefore, development of metal fuel continued at Argonne National Laboratory.

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. As a result metal fuel became a viable alternative to ceramic fuel. In addition, it was discovered that a metal-fueled fast reactor possessed inherent safety features that could not be duplicated with mixed oxide fuel.

When the Integral Fast Reactor (IFR) initiative was implemented at Argonne National Laboratory the development of metal fuel greatly accelerated. Later, General Electric adopted metal fuel for its PRISM reactor design. A key aspect of this is that metal fuel lends itself to proliferation-resistant remote recycling and refabrication by the non-aqueous pyro-metallurgical process.

Design Characteristics

A metal fuel rod consists of a solid fuel slug that has sodium in the gap between the fuel slug and the steel cladding. Early fuel designs had little or no gap between the fuel and

the cladding. When the fuel swelled due to solid and gaseous fission product accumulation, the fuel would contact and stress the cladding, leading to cladding failure. Alloying and heat treatment of the fuel to reduce swelling as well as the use of stronger cladding material resulted in little gain in the achievable burnup.

It was found that if the gap between the fuel and cladding was increased to allow the effective density of the fuel to be 75% or less, the fuel would swell, primarily from gaseous fission product accumulation, until the porosity produced by the gas interconnected. At a fuel density of 75% the interconnection would occur just as the fuel reached the cladding. The fission gas would then be released to an appropriately sized plenum, reducing the driving force for further swelling and thereby limiting the stress on the cladding. The interconnected porosity not only allowed the gaseous fission products to escape to the gas plenum, but the porosity also allowed the fuel to plastically flow into the porosity as the solid fission products accumulated, which also limited the stress applied to the cladding by the fuel.

Uranium-plutonium fast reactor fuel has an unacceptably low solidus temperature, so alloying elements are added to increase the solidus temperature. Of the many alloying elements that could be added to increase the solidus temperature, zirconium was chosen because the uranium-plutonium-zirconium alloys exhibited superior compatibility with the cladding materials, and because zirconium is compatible with the pyro-metallurgical process. The Zr content of the alloy is limited to 10 wt %; additional Zr added to the alloy would require casting temperatures that would challenge the use of simple and efficient casting equipment/materials, such as the quartz molds used in the casting of the fuel. Completing the development of a re-usable or integral mold remains a program goal. This would greatly reduce the waste produced in the fabrication.

Fabrication

The metal fuel slugs are fabricated by injection casting – a process which is well suited for the remote operations used for recycle of fuel after pyro-metallurgical processing. In the injection casting process, a fuel alloy is prepared by co-melting of fuel constituents in a crucible. The molten alloy is then injected into evacuated quartz molds where it is immediately quenched. As many as 100 slugs per batch of up to 18 inches in length have been cast routinely using this process. The slugs are removed from the molds and cut to length. No machining of the surface of the slugs is required. Any reject slugs and the material from the cropped ends are simply recycled into the next alloy melt. The slugs are loaded into cladding jackets with the sodium bond material where the cladding is heated to allow the slugs to settle in the sodium. An end-cap is welded onto the cladding to close the fuel rod. The completed fuel rod is at an elevated temperature to remove any defects in the sodium bond.

More than 200,000 metal fuel rods have been manufactured by this method. Of these, 35,000 have been manufactured remotely in a hot-cell. In addition to various Argonne fuel manufacturing campaigns, two commercial vendors have successfully employed this technology.

Irradiation Performance

Several metal fuel types that incorporate the current design have been irradiated. U-Fs, U-Mo, U-Zr, and U-Pu-Zr, over a range of Pu concentrations up to 28 wt %, have all been successfully irradiated to high burnup. All the alloys consistently exhibit interconnection of porosity and gas release. U-Pu-Zr fuel rods have been irradiated to as high as 20 at.% burnup without failure. For a period of its operation, the entire core of EBR-II was composed of either U-Zr or U-Pu-Zr fuel. Eight assemblies of metal fuel (five full, three partial) were irradiated to greater than 10 at.% burnup in the Fast Flux Test Facility.

Safety and Transient Performance

Metal fuel pins were purposely irradiated beyond the point of cladding breach in EBR-II. With extended irradiation (233 days) after breach, including start-ups and shut-downs, the fuel exhibited no further degradation, no fuel release to the sodium coolant, and no reaction with the sodium coolant.

Metal fuel was subjected to multiple overpower transients during the course of qualifying the EBR-II reactor for operational transient operation. It was found that the diameter of the fuel pins did not change during these events. Rather than stress the cladding the fuel tended to flow into the open porosity.

Irradiated metal fuel was also subjected to severe overpower transients in the TREAT reactor. Prior to cladding breach, the fuel slugs exhibited an increase in length, which would lead to a reactivity reduction during an overpower event in a metal-fueled fast reactor. Cladding breach was predictable based on the combination of stress due to the increase in plenum pressure and reduction in cladding wall thickness due to reaction of the cladding with molten fuel.

An extensive testing program was conducted to determine the temperature at which a liquid phase would form due to interdiffusion of fuel and cladding constituents. Limited liquid phase formation was detected for high-burnup fuel at a temperature of 650^0C. Typically, the peak cladding temperature during irradiation of metal fast reactor fuel is 590^0C. Operation of any fast reactor fuel above 590^0C is not recommended, irrespective of the liquid phase phenomenon, because the steel cladding becomes weak.

A broad data base exists for the irradiation of metal fuel over a wide range of fuels and materials variations and operating conditions. The evolution of the irradiated fuel microstructure and the interaction of fuel and cladding are well understood to burnups of 20%. In addition to the irradiation of fuel over a range of composition of the major alloying elements, U-Pu-Zr fuel with intentional additions of minor actinides has also been successfully irradiated.

The irradiation data base is complimented by a laboratory experiment program that includes phase diagram studies, diffusion experiments to understand fuel radial migration phenomena, and fuel-cladding compatibility studies. The laboratory test results and the irradiation data are included in the Life-Metal predictive modeling code.

R&D Related To Metal Fuels Deployment

Knowledge of metal fuel is sufficiently mature that relatively little remains to support licensing of the fuel system. Development of the binary metal alloy fuel was tied intimately with the development of the pyro recycle technology for recycling all transuranic elements in a commercial product stream. The performance of the fuel is not very sensitive to minor compositional variations and thus a fuel specification, which is based loosely on composition and more stringently on fissile/power density and reactivity, has been drafted that would allow remote fabricators the latitudes necessary for remote processing/fabrication. 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 recycled U-Pu-Zr fuel is therefore desirable 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 that gained from the few remaining tests required, is needed to aid in licensing fuel for a fast reactor prototype.

4.4.3 Mixed Nitride Fuels for Fast Reactors

Status of Nitride Technology

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 same desirable characteristics as metal fuel; i.e., high heavy metal density, good thermal conductivity, and excellent compatibility with sodium. Second, testing of nitride fuel for space power applications did not reveal any undesirable characteristics. But, the amount of testing to date is small compared to that of either oxide or metal [1]. Table 4.3-1 summarizing the status of nitride fuel development to date is shown below.

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}

Table 4.3-1

Nitride fuel development				
Items		**A. Status of technology**	**B. Status**	**C. R&D issues**
(1)	Fuel specification	[Fuel type (fabrication experience)] • Pellet/helium bond (over 1,000) • Pellet/liquid metal bond (tens) • Low "smeared density" fuel is understood as typical high burn-up fuel concept to have benign FCMI, but no firm specification.	• Vibropacking fuel, (TRU fuel with FP) • Vibropacking fuel, (TRU fuel) • (Pu enrichment: 8wt.%, TRU fuel)	[High burn-up fuel specification] • Same as for (3) below
(2)	Nitrogen isotopic composition	Requirement of N15 enrichment to exclude N14, the source of C14 by [N14(n,p)C14] • Conventional technologies of N15 enrichment - NITROX - Ion exchange, etc.	• It is desirable to use enriched N15 for nitride fuel. [M16]	[N15 enrichment technology] 4 Economical enrichment technology development - Ex:Zeolite PSA method
(3)	Burn-up	[Irradiation experience] • Sodium bonded fuel : ~160GWd/t (at peak) • Helium bonded fuel : ~70GWd/t (at peak) • High burn-up fuel concept is indispensable to realize economical advantage. • Irradiation experiments indicated significant FCMI at extended burn-up due to high fuel swelling rate and low creep rate.	4 Average discharge burn-up: □6at.% (UO_2 fuel core) 5 Average discharge burn-up: over 10at.% 6 Maximum burn-up : 105GWd/tHM (Metal fuel core) 7 Average discharge burn-up: 72MWd/kg, Maximum burn-up: 121MWd/kg Maximum burn-up: 15-20at.% (Metal fuel core) (This value is based on the irradiation achievements of metal fuel.) Average discharge burn-up: 100MWd/kgHM (Metal fuel core)	[High burn-up fuel specification] • High burn-up irradiation test of various type of fuel pin to identify the high burn-up fuel concept with benign FCMI Examples: Large gap with LM bonding Annular fuel with gas bonding etc. • [Core material development] • Core material development for high neutron dose to achieve high burn-up Ex.: High strength ferritic steels including ODS (oxide dispersion strengthened ferritic)
(4)	Core safety	Possible candidate approach 1) Utilization of inherent and/or passive shutdown mechanisms 2) CDA mitigation (Nitride fuel high temperature dissociation behavior is one of the critical	• This reactor has high passive safety characteristic, that limits the fuel temperature in accidents. (No description of thermal dissociation behavior) • This reactor has high passive safety characteristic, that limits the fuel temperature in accidents. Some concerns of nitride fuel high temperature	[Inherent/Passive safety] • (Precise investigation is indispensable to assure the feasibility.) [CDA mitigation] • In-pile or out-of-pile high temperature dissociation test of nitride fuel

Nitride fuel development				
Items		**A. Status of technology**	**B. Status**	**C. R&D issues**
		issues.)	dissociation behavior.	• Transient test to evaluate fuel pin failure limit

R&D Issues Affecting Nitride Deployment

The status of nitride fuel is well described in the table above. To summarize:

- The fabrication experience with vibropacking is limited and needs further development,
- A low-cost technology to enrich the N15 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 does appear to have unique safety characteristics that are superior to that of either mixed oxide fuel or metal fuel [2]. And for selected high temperature applications such as space power, nitride fuel has unique advantages, and irradiation testing results have been positive.

References for Section 4.4.3

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[2] A. Padilla, S. F. Kessler, K. D. Dobbin, and A. E. Waltar, "Fuel Aspects of Beyond Basis Event Analysis for the Fast Flux Facility," International Fast Reactor Safety Meeting, Snowbird, Utah (1990).

4.4.4 Fuels for Gas-Cooled Fast Reactors

4.4.5 Advanced Fuels and Inert Matrix Fuels

4.4.6 Molten Metal and Other Liquid Fuels for Fast Reactors

4.5 Concept-Specific Recycling Technologies and Waste Form Production

4.5.1 Recycling 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 recycling 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 more easily recycled by aqueous techniques; therefore these systems, primarily the PUREX process, are more fully developed and implemented commercially. Pyroprocess systems have largely been associated with fast reactors and metallic fuels and their development has therefore reached only the pilot-scale stage, and the feasibility of minor actinide (MA) separation still needs to be demonstrated.

The major advantages of pyrochemical methods for recycling of advanced fuels, in comparison to hydrochemical techniques, are a higher compactness of equipment and the possibility to form an integrated system between irradiation recycling and (re)fabrication facility, thus reducing considerably transport of nuclear materials. Pyrochemistry is to be preferred in particular for advanced oxide fuels (mixed transuranium, inert matrix or composite) and metal fuels, but also nitride fuels. Compared with aqueous methods, dry recycling of fuels results in less pure and thus maybe more proliferation-resistant fractions 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 MA fuel; this allows recycling after shorter cooling times, which reduces storage costs. Furthermore, due to the absence of water (neutron moderator) in the process the criticality hazard is lower; i.e. the inherent safety is increased.

4.5.2 Aqueous Processing Methods (PUREX and its Extensions)

4.5.2.1 Separations Steps

Overview of PUREX-type Processes

The PUREX process is the most important hydrochemical recycling 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 recycling industry and is the reference

process for LWR/UOX and LWR/MOX recycling. The process applies tri-n-butyl phosphate (TBP) in a kerosene-type aliphatic hydrocarbon diluent as the extractant and utilises the highly selective extractability of hexavalent U and tetravalent Pu nitrates to separate them from the bulk of fission products (FP) in a nitric acid (HNO_3) medium. Recently, alternatives such as crystallization of fuel solution, processing without plutonium purification, etc. have been studied to simplify the PUREX process. U and Pu are separated with an industrial yield close to 99.9 %. MAs and FPs are conditioned in a glass matrix for interim storage and final disposal. Am and Cm are not separated because their valences (III) are not extracted by TBP. They remain in the raffinate (HLLW) from the first cycle together with the bulk of the fission products, except the gaseous fission products like Kr-85 and I-129, which are separated in the off-gas of the spent fuel dissolution. Np is partially extracted by TBP, following the U stream and it is separated from the latter in the second U purification cycle. This Np stream is added to the HLLW.

Extended Aqueous Processing for Minor Actinide Recovery

Over the past decade substantial R&D has been directed to the recovery of MA in aqueous processing schemes. Three different options can be distinguished in the PUREX process which achieve the partitioning of TRUs (Np, Pu, Am and Cm). Each of these three is summarized in the following sections.

PUREX process adapted for Np recovery

An adaptation to the standard PUREX process allows improved separation of Np. Neptunium is present in dissolving solutions that are input to the first decontamination cycle as a mixture of Np(IV), Np(V) and Np(VI). If Np is to be co-extracted, a complete oxidation to the valence state VI is required. The Np is then extracted together with U and Pu in the first decontamination cycle and follows the U stream after the Pu partitioning. Finally it is recovered through a reducing scrub in the second U cycle. After separation, the Np nitrate, contaminated with β-γ emitters, may be purified by solvent extraction with TBP and finally transformed to oxide by calcination of the oxalate.

European Extended PUREX process for MA recovery

A further extension to the PUREX process allows the separation of minor actinides (Am and Cm) and some mainly long-lived fission products. Extensive R&D is ongoing to develop special extractants required for the separation of minor actinides (Am and Cm) from HLLW. Three alternative approaches are proposed as described below.

The first approach is based on co-extraction of minor actinides in valence state (III) and lanthanides in valence state (III). The developed or under development processes are:

- TALKSPEAK (USA) and DIDPA (Japan), which use acidic organophosphorus as extractants. The DIDPA process could also be used to recover Np from HLLW

- The TRUEX process (USA) and SETFICS (Japan) are based on the use of CMPO (n-octyl-phenyl-di-isobutyl-carbomoylmethyl-phosphine-oxide).
- The TRPO process (China) uses a trialkyl phosphine oxide. This process has also been developed at the Transuranic Institute (or ITU, in Karlsruhe).
- The DIAMEX (diamide extraction) process using malonamides as extractants has been developed at CEA (France).

The second approach is based on the separation of minor actinides in a single operation that leaves all the lanthanides in the HLLW. High-selectivity extractants like TPTZ (UK-France) (tripyridyltriazine) [85Mus] or CYANEX 301 (China) (trimethyl-pentyl dithiophosphinic acid) are used sometimes in synergistic combination with TBP. New extractants and methods are under research and development; i.e. the two projects PARTNEW and CALIXPART within the 5th Framework Program of the EC. The main extractants to be developed and tested (with some very significant experimental results already obtained with genuine fuels) are diamides (DIAMEX), nitrogen polydendate ligands, dithiophosphinic acids, and acidic S-, Se- or Te-bearing ligands. The last three groups of compounds are used in the SANEX (Selective Actinide Extraction) process to separate selectively the trivalent actinides (Am and Cm) from HLLW. The objective of the CALIXPART project is the synthesis of matrices able to extract and separate actinides from lanthanides in a single step. Several ligand chains including nitrogen or sulphur atoms properly organised on matrices such as calixarenes and others will be synthesised and tested for the oxidation of minor actinides and their selective extraction. This is based on the fact that, in contrast to lanthanides, Am can be oxidised in nitric acid solution. However the oxidation states IV and VI are unstable and must be stabilised with a strong oxidation reagent. The SESAME process, developed by CEA, uses an electrochemical method to oxidise the Am and then to separate it selectively from curium, lanthanides or other FPs. An overview of recovery yields by the above-mentioned PUREX processes is shown in Table 4.5.2-1.

Table 4.5.2-1. Recovery yields for PUREX hydrochemical recycling

Elements		**Standard PUREX**	**Improved PUREX**	**Extended PUREX**
U*		99.9 %	99.9 %	99.9 %
Pu*		99.9 %	99.9 %	99.9 %
Minor Actinides	**Np**		95-99.9 %	99.9 %
	Am			99.9 %
	Cm			99.9 %

* according to industrial process plant operators
** depending on the process parameters

The flow sheet shown in Figure 4.5.2-1 provides an overview on an advanced recycling of spent fuel based on hydrochemical processes.

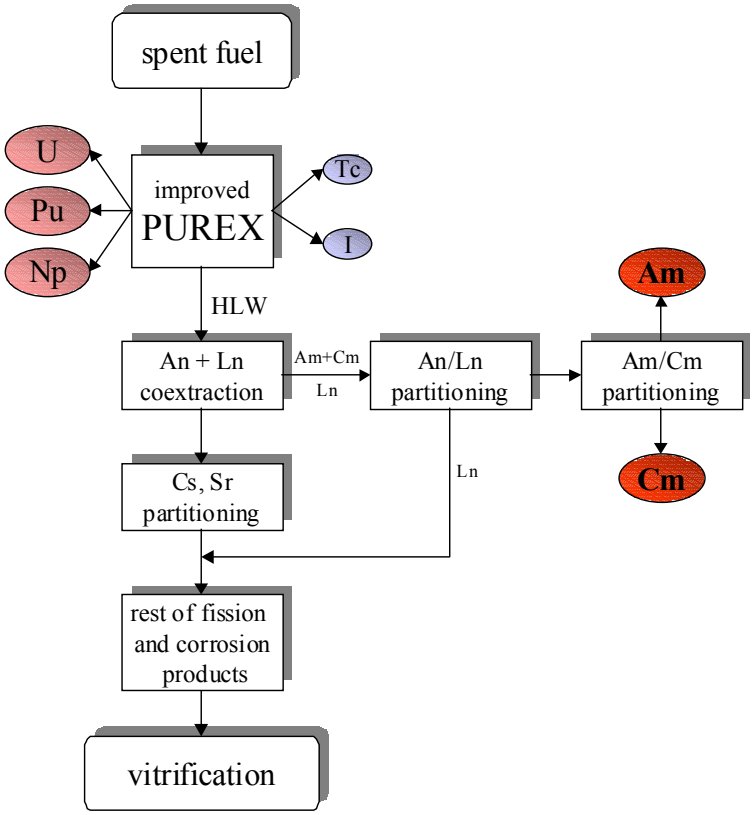

Fig. 4.5.2-1 Advanced recycling of spent fuel including MA recovery and separation of long-lived radionuclides.

Research Facilities

Dedicated facilities for research with hydrochemical recycling are available in France (CEA, Marcoule) and U.K. (BNFL, Sellafield) where extensive studies are being performed on spent fuel material. Facilities for smaller scale hot-tests are available in other countries and the JRC-ITU (Karlsruhe, Germany). In addition, European Universities are strongly involved in this research through the Framework Programmes, especially in the fields of synthesis of new extractant and molecular modelling.

The JAERI Four Group Separation Process for MA and Selected FP Recovery

The Four Group Partitioning process, developed by JAERI, separates the elements in HLLW into four groups: transuranic elements (TRU) using DIDPA, Tc-platinum group metals (PGM), Sr-Cs and others. The four group partitioning process has been constructed as shown in Figure Figure 4.5.2-2.

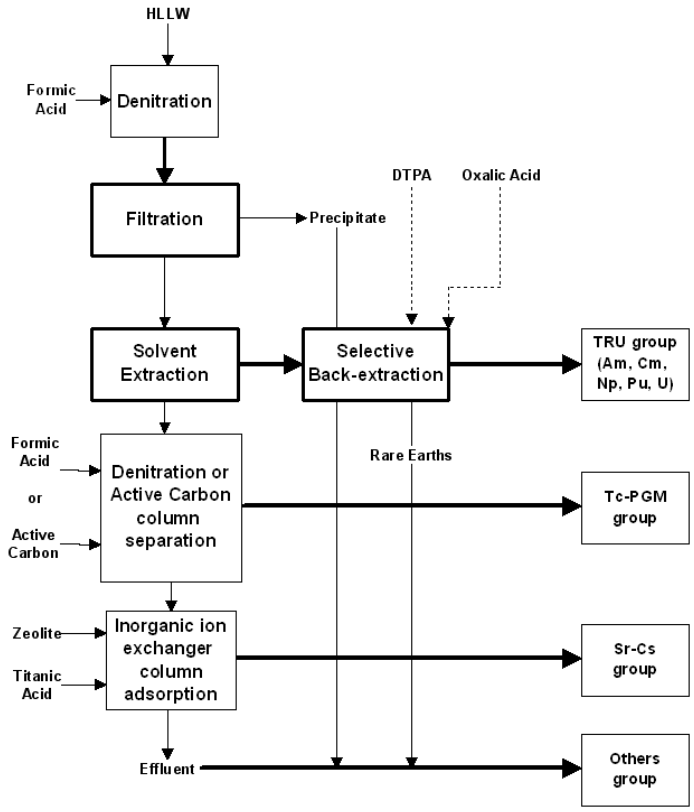

Fig. 4.5.2-2 Four Group partitioning process.

The main characteristics of the four group partitioning process are as follows:

- All TRU elements including pentavalent Np are extracted with DIDPA after the denitration of HLLW, reducing the nitric acid concentration from 2 M to 0.5 M.
- Tc and PGM are separated by precipitation through denitration or by adsorption with an active carbon.
- Sr and Cs are separated by adsorption with inorganic ion exchangers, titanic acid and zeolite, respectively.
- The TRU elements extracted in the DIDPA solvent are sequentially back-extracted with 4 M nitric acid for Am, Cm and lanthanides, 0.8 M oxalic acid solution for Np and Pu, and 1.5M sodium carbonate solution for U.
- Back-extracted Am, Cm and lanthanides are extracted again with DIDPA, and then Am and Cm are preferentially back-extracted with 0.05 M dietylentriaminepentaacetic acid (DTPA) solution, leaving lanthanides in the DIDPA solvent.

Effectiveness of the partitioning process

Separation efficiency and target recovery values, obtained experimentally with primarily simulated HLLW, and the estimated recovery through whole partitioning process are shown in Table 4.5.2-2. The separation efficiency of the element refers to the recovery in

the first chemical step for its separation. The target recovery of TRU elements is the value determined to make the potential radiotoxicity of a vitrified HLW after storage for 1000 years comparable to that for naturally occurring mineral containing 3wt% of U.

Table 4.5.2-2. Target recovery, experimentally obtained separation efficiency and estimated recovery of elements in the four group partitioning process

Elements		Target recovery (%)	Separation efficiency (%)	Estimated recovery (%)
T R U	Np	99.5	>99.95	99.85
	Pu	99.9	>99.99	99.85
	Am	99.99	>99.99	99.97
	Cm	99.9	>99.99	99.97
Tc		99	~98	~95
Sr, Cs		99	>99.9	>99.9

The separation efficiency and the estimated recovery of TRU elements are quite high and nearly satisfy the target recovery values. The recoveries of Tc and PGM are approximately 90 to 95%, which is lower than the target recovery. This lower recovery result is of lesser importance due to the relatively small contribution of Tc and PGM to the potential radiotoxicity of HLW.

Separation efficiencies obtained experimentally with simulated HLLW and the estimated recovery values for entire partitioning process attain or even surpass the target recoveries – indicating that the potential radiotoxicity of a vitrified HLW after storage for 1 000 years would be comparable to that for naturally occurring minerals containing 3 wt% of U. Regarding the separation of Am and Cm from Lanthanides, it is possible to reduce their lanthanide content to less than 25 wt% with 99.99% separation efficiency from actinides.

The radiation resistance of DIDPA is about 7 times higher than TBP and its degradation products have no harmful effect on the TRU extraction resulting in less secondary waste generation due to solvent retreatment. The four group partitioning process reduces the volume of waste by a factor of at least 3.3, compared with that of the vitrified HLW, and the volume of cemented waste is only about 20% of that generated in the PUREX process. Moreover, an α-activity concentration of the cemented waste becomes lower than 1 GBq/ton, and its shallow land disposal might be a possibility. Demonstration of the four group partitioning process with real HLLW is now in progress.

The JNC Advanced Aqueous Recycling Technology for Fast Reactor MOX

The JNC in Japan has proposed and is studying the "advanced aqueous process, which combines the "crystallization process" with the "simplified solvent extraction process" and with an added MA collection function. The main features compared with the conventional PUREX are summarized below.

- The purification process of U and Pu in the conventional PUREX process is eliminated, resulting in co-extraction of U/Pu/Np and the simplification of the system. Compact-sized centrifugal type equipment is adopted for the extraction process to allow a smaller facility size.

- The crystallization method is also adopted for the recovery of excess U before extraction of U/Pu/Np, which reduces the amount of liquid treated in the extraction process. The main stream is the salt-free process, which means the reduction of intermediate-level waste.

- An additional process is also considered for the recovery of AM and Cm. The combination of the SETFICS process, developed by the JNC, and the TRUEX process is applied for this stage.

- According to the evaluation of the mass balance, the recovery ratio of U/TRU has been estimated to be 99.7%, and the decontamination factor of the recycled product is higher than 10^2.

In addition, in order to improve the advanced aqueous process further, the ion exchange method, the amine extraction method, and the supercritical fluid extraction method are also investigated as an alternatives or supplementary aqueous technologies.

1. Overview [1],[2]

Aqueous recycling technology, which is used to recover elements from spent nuclear fuel encompasses several methods, such as solvent extraction based on solubility differences in solution and solvent, precipitation based on generation of a solid complex, and anion exchange. For all methods, the spent fuel constituents are dissolved in aqueous solution before treatment.

Because the solvent extraction method used in the PUREX requires an extensive and complicated system, the process has a disadvantage economically. However, the process has safety merit, because static chemical reaction at low temperature employed. Additional advantages to PUREX include decontamination performance, recovery function, criticality safety control and applicability to safeguards technology. In addition, the process is suitable for the mass production, because it has scale-up merit.

The fast reactor has the merit of being capable of accommodating recycle products that are not highly purified. Therefore, the JNC is developing advanced aqueous recycling technology, based on PUREX development, in which Pu is recovered with other elements to obtain a product without partitioning and purification. In this system, Pu is extracted with other elements, and partitioning and purification processes are removed from solvent extraction processes, which simplifies the process. Because isolated Pu does not exist in any step of the system, the new method enhances nuclear proliferation resistance. In addition, the equipment is compact, reducing the size if the required facility. Recovering surplus U using the crystallization method prior to extraction to reduce liquid waste.

Furthermore, TRU elements are recycled as a recycling product, which helps reduce the amount of long-lived MAs on the outside of the nuclear fuel cycle system (See Figure 4.5.2-3).

2. Background history [1],[3],[4]

PUREX recycling technology has already been commercialized in some nations for the recovery of pure Pu from the enriched U of spent LWR fuel. The system performance was enhanced and the equipment was improved through the operating experience.

Furthermore, the MOX spent fuel recycling technologies are also developed for the commercial LWR fuel cycle and the FR fuel cycle.

3. Current Status [2],[5],[6],[7]

Fast reactor spent fuel has higher quantities of Pu, MAs and FPs than than does the LWR spent fuel due to the higher burnup of the fuel. Therefore, improvements relative the LWR system are required in economic efficiency, nuclear proliferation resistance, environmental impact, etc.

Advanced aqueous recycling technology is developed for the purpose of meeting those objectives. In order to demonstrate technical feasibility of the technology, radioactive and non-radioactive experiments have been performed, and the extraction function, decontamination performance, etc. have been evaluated (See Figure 4.5.2-4). Equipment has been developed based on the operating experience of Tokai recycling plant.

4. Future R&D Tasks

The future R&D tasks necessary to develop advanced aqueous recycling technology are the following:

- compacting head-end equipment,
- enhancing and optimizing equipment (corrosion resistance, extraction function, etc.), and
- enhancement of effective MA recovery process to reduce waste (Figure 4.5.2-5)

Furthermore, the optimization of the blanket fuel management in the fast reactor must be attempted, from the viewpoint of safeguards, due to the high Pu-239 quantities that are produced.

Advanced aqueous recycling is the technology that recycles TRUs in the closed system. In addition, if it is required to transmute FPs in the nuclear fuel cycle using the surplus neutron in fast reactor core, as does the SCNES concept, the separation and recovery function for FPs, (e.g., electrolyzing Tc from waste solution), will be developed and incorporated into the process.

5. Research Facilities

The Chemical Process Facility at Tokai Works of JNC in Japan is available to develop the advanced aqueous recycling method.

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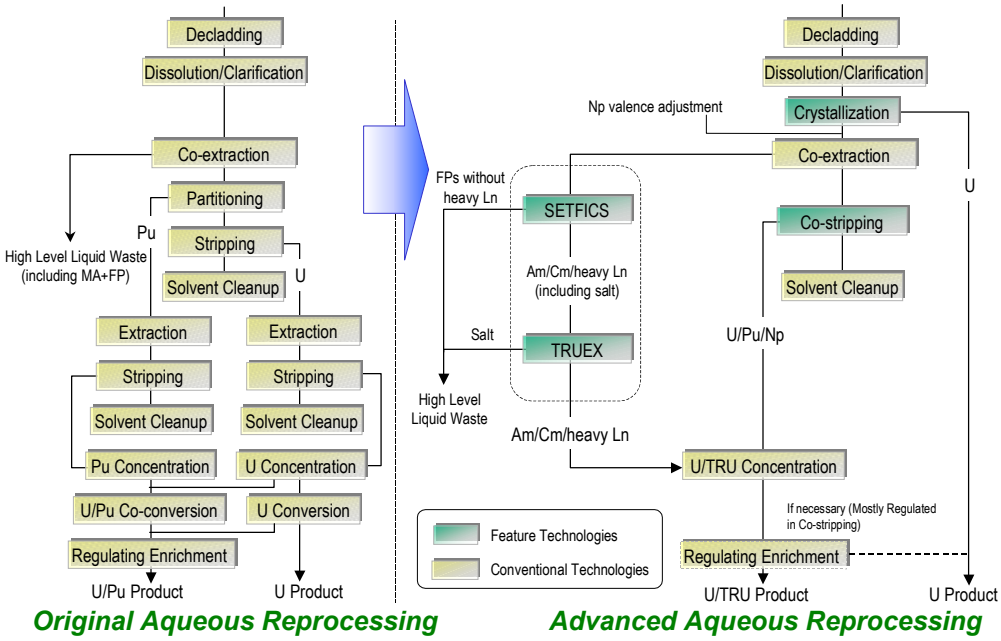

Figure1 Advanced Aqueous Reprocessing

Figure 4.5.2-3. Schematic flowsheet for JNC Advanced Aqueous Recycling technology for fast reactor MOX fuel.

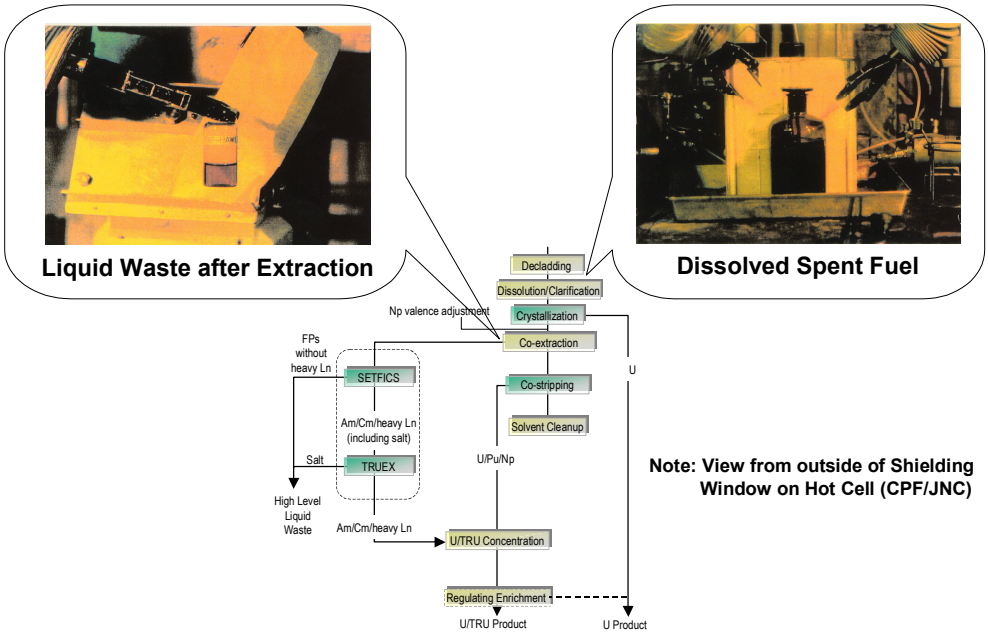

Figure 2 Experimental Confirmation of Feasibility

Figure 4.5.2-4. Illustration of experimental demonstration of JNC Advanced Aqueous Recycling technology.

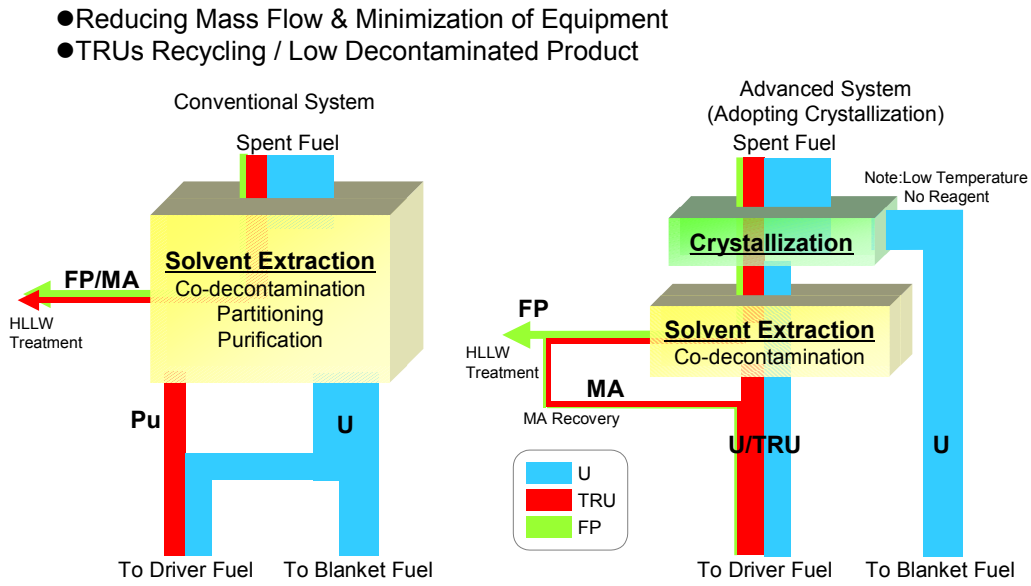

Figure 3 System Minimization and Optimization

Figure 4.5.2-5. Schematic illustration of flow path for minor actinides in conventional and of JNC Advanced Aqueous Recycling technology.

Summary Overview of Current R&D on Advanced Aqueous Recycling

A dual solvent extraction process has been proposed by the JNC and is based on U-Pu-Np separation from dissolver solution by the Advanced PUREX process and subsequently the additional recovery of Am and Cm from the high level waste by the Advanced TRUEX process. Actual R&D have been pursued in hot-lab conditions at the Chemical Processing Facility in Tokai-works using actual high-level liquid waste (HLLW). Actinides (An) and Lanthanides (Ln) were successfully separated from the highly active raffinates and application of some sort of 'soft' donor ligands, purified CYANEX 301 and ODP, provided excellent Am/Eu separation. Electrolytic extraction tests proved to be very promising for separating several rare metals including some LLFPs.

The NEWPART-programme, within the 4th European Framework program, focused on partitioning methods for Am and Cm within a PUREX raffinate (Np separation was considered within an improved PUREX process). Within this programme a two-step process has been targeted; the first step (DIAMEX process) aiming at the separation of trivalent actinides and Lanthanides from fission products (FPs) and, in a second step, (CYANEX process) the group separation of An(III)/Ln(III) by selective extraction of An(III), which is the most difficult goal to achieve. The SESAME process could complement the previous steps by separation of Am from Cm. Definite progress has

been made within the DIAMEX process in the understanding of the basic chemistry of malonamide extractants in their ability to react with trivalent actinide and lanthanide nitrates and active tests are foreseen in the near future. A real breakthrough has been reported by the independent confirmation by two institutes of the very high separation factors (6000) of Am and Eu using the CYANEX 301 as extractant. Another potential breakthrough has been achieved in An(III)/Ln(III) separation using BTP (pyridine-bis-1,24-triazines) resulting in separation factors of Am over Eu of 140 in high nitric acid concentrations. In conclusion, confidence has increased in that the processes based on the NEWPART programme can be developed to reach the goals required for modern efficient industrial MA partitioning processes.

Research activities in Russia are focused on the separation process of heat emitting nuclides (^{137}Cs and ^{90}Sr) and minor actinides by use of cobalt dicarbollide. The upscaling test of the process has shown to be very successful from the process- and safety-points of view. An industrial extraction facility using cobalt dicarbollide has been operated for several months at the Mayak plant and showed excellent results with real HLW. International discussions are underway to restart the process.

Table 4.5.2-3 gives a brief overview of the status of R&D on the various aqueous partitioning techniques, as was reported in the 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 guarantee its performance. The conclusion of this step corresponds to establishing the technical feasibility of the process.
- Phase 3 relates to the industrialisation of the process, terminating with the establishing of industrial feasibility of the process and its potential application in an industrial installation. It is aimed to ensure overall active operations in industrial conditions. In practice, these studies are essentially conducted by engineering design.

Table 4.5.2-3. 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) (DIDPA) (TRUEX) (TRPO) (DIAMEX) • based on An selective extraction (TPTZ) (Picolinamides) (CYANEX 301) (BTP) • based on precipitation (Ferricyanide)	 x x x x	 x x x x	x	SF=5 900
Am separation in the oxidised state (SESAME)		x		Am/Cm separation
Tc separation (PUREX) (PUREX) **Tc - PGM separation** (Denitration precipitation) (Active carbon adsorption)	 x	 x x	x	soluble Tc insoluble Tc
I separation (PUREX)			x	95% separation
Zr separation (PUREX)		x		
Cs separation (Calixarenes) (Zeolite)		x x		
Sr separation (Titanic acid)		x		
Cs and Sr separation (Dicarbollides)			x	
Pd (PGM), Se, Ru separation (Electrolytic extraction)	x			soluble Pd, Se, etc.

For all the above-mentioned processes the development of new extractant molecules and the improvement of existing ones is 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 in India the examination of sulfoxide type extractants have to be mentioned.

In view of process industrialisation, the economics and the radiation resistance of the organic molecules are an important R&D issue. 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 preconcentration 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 genuine high active wastes originated from real spent nuclear fuels.

4.5.2.2 Head End Steps

4.5.2.3 Back End Steps

4.5.3 Pyrochemical Processing Methods

Since early in the development of the nuclear industry, a number of pyrochemical processes to separate actinides from spent fuel have been investigated. Commencing in the early 1950s and extending through the 1960s, research and development work on pyrochemical and pyrometallurgical processes were conducted at various sites mainly in the US. 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.

During the 1990s, new concepts for the transmutation of fission products and the associated fuel cycles motivated 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 degrees of transmutation and thus a sufficient reduction in radiotoxicity. The far future options, such as dedicated fuel cycles and transmutation in molten salt reactors would certainly 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 recycling solution.

4.5.3.1 Separations Steps

For the recycling of nuclear fuel (or targets) the following basic steps must be considered:

- Separation of the fuel from the cladding material by mechanical techniques or
- Dissolution together with the fuel in molten salt at temperatures of the order of 500-800°C. Fluorides and chlorides are mainly proposed for this purpose, although other media (phosphates, borates, nitrates, etc.) could also be considered.

- Separation of the actinides by electrorefining, transfer to a non-miscible liquid metal phase or selective precipitation. Partitioning by extraction is based on the differences in the thermodynamic stability of compounds or complexes of these elements and on changes in their oxidation state. Extraction by a liquid metal involves stripping by means of a reducing agent. A multistage process improves, of course, the degree of refining and/or the recovery rate. In the case of electrowinning or selective precipitation, the elements are recovered by a physical method.

The main objectives of these operations are of course very similar to those of hydrometallurgical processes:

> Selective extraction of the long-lived fission products (LLFP) and/or the actinides for subsequent recycling; the extracted product must therefore meet specifications for purity with respect to the unextracted products.
>
> Minimisation of losses of the relevant elements in the various process steps.
>
> Generation of process waste (i.e., by-product waste rather than extracted fission products in salt or metallic form and technological waste) compatible with a suitable treatment process.

Because electrorefining and electrowinning are important techniques in pyrochemical processing, these techniques are described here.

Electrorefining

The electrorefining process can be characterized simply as a process in which an impure material is made as the anode in an electrochemical cell and, with the passage of current in the cell, pure material is deposited at the cathode. In spent fuel treatment applications, the anode contains the spent fuel, generally after chopping to expose the fuel material to the electrolyte salt. The fuel can be metal, oxide or nitride. It is not necessary to remove the fuel cladding or any other materials internal to the fuel rod. In the case of metal fuel electrorefining in which the fuel material is an alloy of uranium, transuranics and zirconium, the electrolyte salt of choice is a eutectic mixture of LiCl and KCl, operated at a temperature of 500°C. A concentration of about 2 mol% of actinide chlorides is maintained in the electrolyte salt to support electrotransport of the actinides to the cathode. With this concentration of actinide ions in the salt, electrotransport can be sustained at a high rate when a potential of 0.5-1.0 volt is imposed between the solid cathode and the anode. Reduction of the actinide chlorides occurs at the cathode and deposition of an actinide atom is accompanied by dissolution of a corresponding atom from the spent fuel. As anodic dissolution of the chopped fuel proceeds, fission products dissolve by reducing some of the actinide chlorides. This necessitates the occasional addition of a compensating amount of oxidant, usually $CdCl_2$ or UCl_3. If uranium is present in the spent fuel, and if the ratio of the concentration of UCl_3 to transuranic chlorides in the salt is greater than about 0.1, then only uranium will deposit at the solid

cathode. The alkali metal and alkaline earth fission products (e.g., Cs and Sr) will not deposit because their chlorides are much more stable than the actinide chlorides. Similarly, the chlorides of the transuranic elements and the rare earth fission products are more stable than uranium chloride. So, these elements will not deposit at the solid cathode as long as the uranium concentration in the electrolyte salt is sufficient. The transition metal fission products will not be oxidised and will remain in the anode basket as metallic elements.

The transuranic elements can be recovered in the case of the treatment of uranium-bearing fuels by the use of a different cathode - a liquid metal cathode. A common liquid metal cathode consists of a small amount of liquid cadmium contained in a ceramic crucible suspended in the electrolyte salt, with electrical contact made to the cadmium. The transuranic elements will electrodeposit in the liquid cadmium because their chemical activities are reduced by formation of intermetallic compounds such as $PuCd_6$. Uranium will also deposit in the cadmium cathode as an intermetallic compund. Even though the rare earth fission products will also form intermetallic compounds with cadmium, their chlorides are more stable than those of the actinides; thus, only small amounts of rare earths are deposited in the cadmium cathode. But it is important to recognize that in this system: (1) the uranium deposited at the solid cathode will be pure; (2) the transuranics deposited in the liquid cathode will be contaminated with 10-30 wt.% uranium and a few percent of the lanthanide elements; and (3) the properties of the transuranic elements are sufficiently similar that they cannot be separated in this process. The cladding materials of the spent fuel will not be dissolved, provided that the cell voltage is maintained below the decomposition potential of the constituents of the cladding.

In the case of non-fertile fuels (no uranium content), the transuranic elements can be electrodeposited directly on a solid cathode. Minor contamination of the transuranics with lanthanide fission products can be expected, and there will be no separation of the transuranic elements from one another.

If the fuel to be electrorefined is not metallic, one has the option of reducing the fuel to the metallic state, as in the case of oxide fuel using the Direct Oxide Reduction process (described in section 4.5.3.2) with calcium as the reductant. Alternatively, it is possible to use the Dimitrovgrad approach and capitalize on the increased electrical conductivity of oxides at elevated temperatures, directly electrodepositing the actinide oxides. Although this process has proven successful with MOX fuel, it requires further development or the use of additional process steps for complete recovery of all of the actinide oxides.

In the case of nitride fuel, actinides can be recovered much the same as with metallic fuel. Actinide elements can be electrotransported to a cathode, with nitrogen gas being liberated at the anode. An off-gas recovery system would be necessary in the case of nitride fuels highly enriched in the ^{15}N isotope to prevent build-up of ^{14}C. Noble metal fission products would remain in the anode basket, and other fission products would form stable chlorides and reside in the electrolyte salt, which could be a LiCl-KCl eutectic

mixture. In the processing of non-fertile fuel, transuranic elements could be collected on a solid cathode; with nitride fuels containing uranium, the collection of pure uranium at the solid cathode would not be possible, and the use of a liquid cathode for collection of transuranics may not be necessary. Experimentation is necessary to resolve this question.

Electrowinning

Molten salt electrowinning is simply an electrolysis process in which the material to be recovered is present as a halide compound in an electrolyte salt. For example, a metal or nitride fuel could be chlorinated in a chloride carrier salt to produce chlorides of the actinide elements and most of the non-gaseous fission products (depending on the chlorinating agent). Most fuel types could be fluorinated to produce fluorides in a fluoride carrier salt. The molten salt containing the dissolved spent fuel constituents could then be placed in an electrolytic cell, with a potential applied between the anode and cathode. At the appropriate voltage level (the decomposition potential), which depends on the species to be electrolysed and the cell temperature, actinides can be deposited at the cathode and the halogen gas will be liberated at the anode. Electrowinning can also be important in certain applications of volatility processes.

Concepts

Over the last couple of decades, various pyrochemical recycling techniques have been studied, developed and tested (in some cases at a pilot scale) throughout the world; mainly in the US, Russia and Japan.

Research in the U.S.

Through much of the 1980s and into the 1990s, the U.S. was engaged in the development of the Integral Fast Reactor (IFR) concept, which employs a pyroprocess for recycle of metal fuel utilized in a sodium-cooled fast reactor. In the later years of the program, an oxide reduction process was developed to remove actinides from spent LWR fuel for incorporation into the IFR fuel cycle. The process is illustrated schematically in Figure 4.5.3-1. The IFR program was terminated in 1994 before the technology for recycle of spent fuel from the Experimental Breeder Reactor II (EBR-II) could be demonstrated. Subsequently, the pyroprocess was modified for the treatment of sodium bonded metal alloy EBR-II fuel for eventual disposal. The key difference between the use of the technology for fuel treatment versus fuel recycle is that the transuranics are not recovered for fuel treatment. Instead, the transuranics are allowed to accumulate in the electrorefiner salt phase and are eventually disposed of in the resulting ceramic high-level waste.

The spent fuel treatment technology was successfully demonstrated with EBR-II fuel. During this demonstration by Argonne National Laboratory (ANL), which was conducted between June 1996 and August 1999, 100 EBR-II driver assemblies (with 400 kg highly-enriched uranium) and 13 EBR-II blanket assemblies (with 600 kg depleted uranium) were treated. This small, but representative quantity of fuel provided adequate fission

products and transuranics to produce waste forms for characterization. A subcommittee of the National Research Council (NRC), which had been established to review the progress and to evaluate the demonstration results, noted the following in their final report:

> Finding: The Committee finds that ANL has met all of the criteria developed for judging the success of its electrometallurgical demonstration project.
>
> Finding: The Committee finds no technical barriers to the use of electrometallurgical technology to process the remainder of the EBR-II fuel

Upon completion of the demonstration, review by the NRC, and a positive nonproliferation assessment, the Department of Energy (DOE) decided to use this technology to process the remaining EBR-II fuel (approximately 25 tonnes) and the small number of sodium-bonded metal assemblies from the Fast Flux Test Facility (FFTF). After completion of an environmental impact statement, these production operations started in September 2000.

The work performed to date on the treatment of nitride and oxide fuels has been less extensive than for metal; it has been on either the laboratory or engineering scale. The feasibility of the processes has been demonstrated, but large-scale tests have not been performed with irradiated spent fuel.

Technical Uncertainties

The Spent Fuel Treatment Program at ANL demonstrated many parts of the pyroprocess fuel cycle, 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. In addition, the remote fabrication of IFR fuel was not part of the Spent Fuel Treatment Program, but the same technology was used to fabricate cold fuel for EBR-II, and a demonstration of another pyroprocess (melt refining) for recycling EBR-II in the 1960s employed remote fabrication for 34,500 fuel elements.

One challenge for a pyroprocessing system is selecting the appropriate materials of construction for the high temperature processes. Material improvements are needed in order to lessen the formation of dross streams and increase material recovery and throughput.

The quantity of waste generated from pyroprocessing that requires geological disposal appears to be comparable 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.

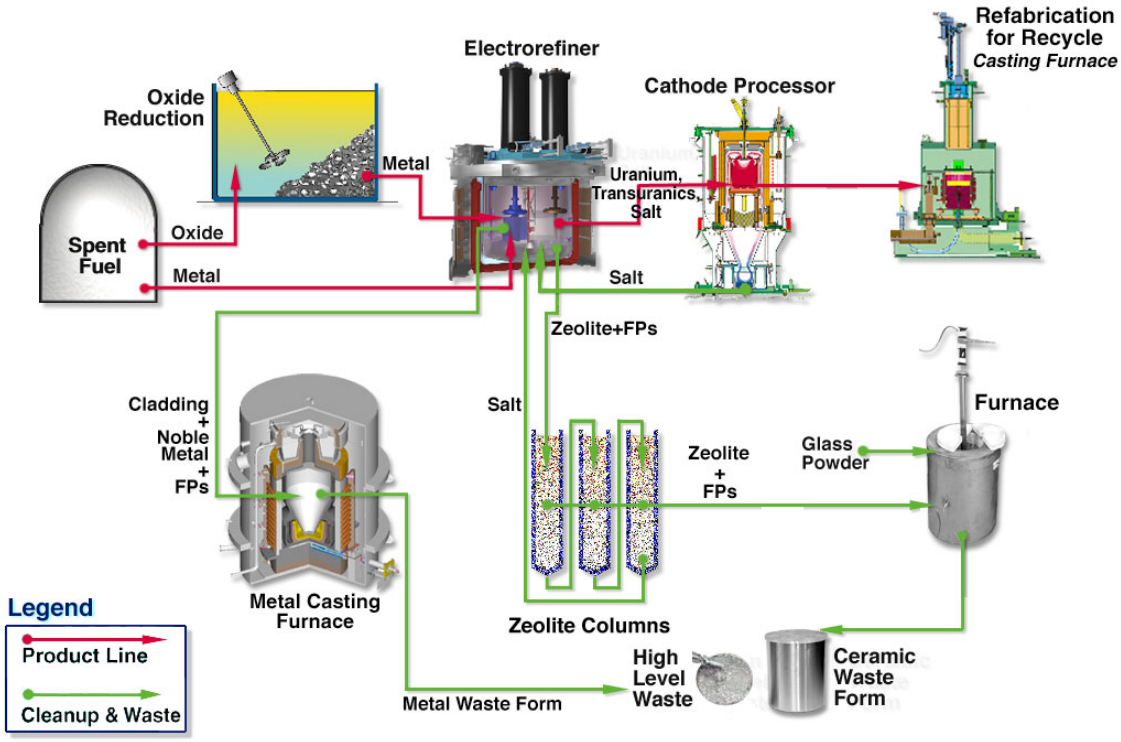

Figure 4.5.3-1. Schematic flowsheet for the IFR pyroprocess.

Research in Russia

In Russia, the Research Institute of Atomic Reactors (RIAR) in Dimitrovgrad has successfully demonstrated the technical feasibility of the pyro-electrochemical recycling of oxide fuel by direct electrorefining without reduction. Several kilograms of spent fuel from the BOR 60 fast breeder reactor have been recycled.

Russian experience with molten salt dates from the 1950´s with the creation of the High-Temperature Electrochemistry Institute at Ekaterinbourg. A wide range of laboratory studies (characterization of salt structures, electrochemistry, kinetics, etc.) was conducted there in molten chloride media on uranium, thorium and simulated fission products. Work began in 1964 on plutonium in the Electrochemical Processes Laboratory of the RIAR-Institute to specify a series of elementary operations for the fabrication and recycling of UO_2 and MOX fuels by the vibrocompaction method.

Research activities today (potentiometry, voltammetry, etc.) focus on the behaviour 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.

Since 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. The Arzamas and Kurchatov Institutes have also proposed a molten salt reactor concept for Pu and minor actinide management purposes.

Research in Japan

Japan's research and development activities on pyro-electrochemistry are performed primarily by:

- CRIEPI (Central Research Institute of Electric Power Industry) with the development of a metallic fuel process
- TEPCO (Tokyo Electric Power Company), which has embarked recently in a large collaboration with RIAR to study recycling of MOX fuel
- JAERI (Japan Atomic Energy Research Institute), which is investigating the potential of nitride fuels for minor actinide transmutation
- JNC (Japan Nuclear Fuel Cycle Development Institute). which has started an evaluation of different pyrochemical recycling technologies in view of future nuclear fuel cycles in Japan.

CRIEPI

CRIEPI is investigating the technology of recycling actinides from a metal-fueled FBR or from reduced LWR oxide fuels, and also their recovery from high-level liquid waste. The processes are essentially based on the metal fuel cycle as previously described using molten chlorides (LiCl-KCl) in contact with liquid metal (Cd or Bi). This recycling system should be operated flexibly by combining LWR and FBR fuel cycles. Its main features are the electrolytic deposition of uranium and plutonium to solid and liquid Cd cathodes and the recovery of transuranium elements from the salt by reductive extraction (Cd or Bi) or by electrorefining. Whereas the US and Russian concepts focus mainly on the recycling of U and Pu, CRIEPI has performed extensive experimental work on actinide/lanthanide separation in molten chloride and liquid metal (Cd, Bi) media by electrodeposition or by liquid-liquid extraction. Laboratory-scale experiments were performed with minor actinides in collaboration with several organisations, including the

University of Missouri for glove box tests and the Transuranium Institute (ITU) in Karlsruhe for hot-cell verification experiments on genuine metallic targets.

Results have shown that the electrorefining of minor actinide metals in molten LiCl-KCl at 450°C appears feasible, even for Am. The alternative approach is the combination of electrorefining with a reductive multistage extraction from the molten salt into molten Cd or Bi. In these tests conducted at trace levels, promising recovery rates of 99.9% for Am were achieved.

In the larger installation at ITU, a demonstration of the technical feasibility of the electrorefining, or reductive extraction, concept is to be made. Furthermore, conditions to improve the electrorefining of Am from lanthanides are being investigated. This installation will be the first to be operated at the gram-scale of minor actinides. (As mentioned above, in the conditioning of the EBR-II fuel at Argonne National Laboratory only uranium is separated by electrorefining, the transuranium elements remain together with the fission products, and the subsequent extraction of the TRU from the fission products planned for the IFR has not been conducted at a pilot scale.)

CRIEPI is also investigating a process to recover transuranium elements from high-level PUREX wastes. It consists of the following steps: denitration to oxides, chlorination, reductive extraction and electrorefining in a LiCl-KCl/Cd or LiCl-KCl/Bi system.

JAERI

Since the mid-1990s, JAERI has been working on a fuel cycle combining nitride fuel with pyrochemical recycling methods in molten chloride media at a laboratory scale. Although encouraging preliminary results have been obtained with uranium, major R&D work remains to be done on the transuranium nuclides. In the double strata concept foreseen in the OMEGA program, JAERI investigates the possibility of using nitride fuels and pyrochemical processing in the second stratum. In this case the recycling of ^{15}N to avoid the formation of ^{14}C during irradiation is much easier, because the reduction of fissile transuranics in the first stratum limits the amount of ^{15}N to be processed in the second stratum. A major theoretical and experimental research program undertaken by JAERI supports this proposal.

JNC

At the JNC, the examination of pyrochemical processes is performed on the laboratory scale and the engineering scale and is directed at an economic evaluation of the potential of this technology. Comparative assessments between dry processes and aqueous processes will be performed in the near future at JNC in collaboration with utilities. CRIEPI, and others.

A glovebox facility for plutonium experiments with a capacity to treat 200 g of fissile material (MOX) is installed in Tokai-mura. The behaviour of Pu is studied in each step of the pyrometallurgical process; i.e. fuel reduction, electrorefining, liquid-liquid

extraction, chloride distillation, metal distillation, oxidation. In this study, the performance and the safety aspects of the process are emphasized.

4.5.3.2 Head End Steps

It is evident that pyrochemical processes include not only a core process, in which the recyclable elements are extracted but also numerous ancillary operations at the front end or the tail end of the main process. Such operations include:

- Prior preparation of the feed stream to ensure compatibility with the core process; e.g. decladding of solid fuel, halogenation of oxide fuel and dissolution in molten salts,
- Purification of the output stream, with recycling of contaminants back into the core process and treatment of the process and conditioning of waste materials in accordance with their chemical and radiological properties (e.g. vitrification),
- Decontamination of technological waste (e.g. crucibles) with recycling of contaminants back into the core process and conditioning of the actual waste materials (e.g. melting of scrap metal),
- Decontamination of all process off-gases, with recycling of contaminants back into the core process.

The performance characteristics required for the separations process and for the ancillary operations depend directly on the targeted limits for losses of recyclable elements into the waste stream and on the desired degree of purity of the recycled elements. If a voloxidation of the spent oxide fuel is included in the pre-treatment, several fission products are distilled from the fuel material including ^{129}I, which could be collected with 99.9 % yield. During the electrorefining process, 95% of the ^{99}Tc is collected together with the noble metal fission products in the bottom Cd layer of the electrolysis cell. For the transmutation of the two fission products, special targets have to be fabricated.

<u>Pyrochemical Head-End Processes for Reduction of Oxide Spent Fuel</u>

Many pyrochemical process options have been developed over the past 60 years for recovery of transuranic (TRU) elements from irradiated nuclear fuel. In more recent years, as part of the U.S. Department of Energy's Actinide Recycle Program (ARP), the pyrochemical technology from past years was examined in detail to select the most promising method for recovery of TRU elements from light water reactor (LWR) spent fuel [1,2]. The purpose of the ARP was to develop a process for extraction of TRU elements from LWR spent fuel for utilization in the Integral Fast Reactor. A number of pyrochemical processes were extensively evaluated as part of the Actinide Recycle Program (ARP) during the time period 1990 through 1994. The most promising of the methods explored in the ARP included the Salt Transport Process, the Zinc-Magnesium Process, the Magnesium Extraction Process, the Lithium Reduction Process, and the Calcium Direct Oxide Reduction (DOR) Process. These candidates are described below, along with some of the major unresolved issues.

<u>Salt Transport Process</u>

This process effected reduction of the oxide feed using a Cu-Mg-Ca reductant alloy with a $CaCl_2$ molten salt flux at 850°C. During reduction, the uranium, TRU elements, rare-earths (REs), and other nobler metal (NM) fission products collect in the Cu-Mg-Ca alloy, while all other fission products collect in the molten salt. The salt is recycled by electrochemically decomposing the oxide, discarding the oxygen and recovering the calcium. The alloy is then separated from the reduction salt, and the TRU elements are extracted into molten $MgCl_2$, and deposited into a Zn-Mg alloy. This extraction and transport step is called "salt transport" and the purified TRU metals are recovered by retorting the Zn-Mg alloy for recycle in the process. The major issues with this process are that it is complex and requires many extraction steps.

Zinc-Magnesium Process

If the reduction step in the salt transport process described above is perfomred using a Zn-Mg-Ca alloy, the process can be greatly simplified. The reduction results in separation of the TRU elements, REs, and NMs from the other fission products, and these elements are recovered by retorting the Zn-Mg for recycle in the process. The TRU/RE/NM alloy could be used directly in fabricating fuel. The major issue with this process is that the reduction is effected by reaction of the oxides with Mg, rather than with Ca. MgO is not soluble in the salt, thus the Mg cannot be electrochemically recovered. It thus becomes a waste stream with volume of a magnitude approximately equal to that of the feed material.

Magnesium Extraction

In the Magnesium Extraction process, the oxide feed is reduced to metal using a Ca-Fe alloy in a molten $CaCl_2$ salt at 850°C. Reduced uranium forms a low-melting alloy with the iron, and the reduced TRU elements, REs, and NMs accumulate in this alloy. The other fission products collect in the molten salt. The salt is recycled by electrochemically decomposing the oxide, discarding the oxygen and recovering the calcium. The molten uranium-iron is then exposed to magnesium, which extracts the TRU and RE elements, leaving the NMs behind. Magnesium is subsequently recovered by retorting and reused in the process. The major problems with this process are that it requires the molten uranium-iron alloy, which is a corrosive, and it requires many magnesium extraction steps for each batch.

Lithium Reduction Process

The Lithium Reduction process effects reduction of the oxide feed using lithium metal in a molten LiCl salt at 650 ºC. The spent oxide fuel is loaded into porous metal baskets that are designed for maximum flow of reduction salt through the oxide feed material. The basket assembly is placed in the reduction vessel, and the salt is agitated to effect flow through the oxide. The lithium metal is contained in porous stainless steel sponge immersed in the salt. When the reduction step is completed, the basket, along with the reduced TRU metals, NMs, and RE-oxides, is withdrawn from the reduction vessel and

transferred to the electrorefiner for further processing. The spent reduction salt is transferred to an electrochemical vessel where the dissolved lithium oxide in the spent salt is electrochemically decomposed in a salt-recovery or electrowinning process step. In this process step, oxygen gas is released and lithium metal is deposited into the porous stainless steel sponge. At the end of this process step, the regenerated salt (low dissolved lithium oxide concentration) and lithium are recycled to the reduction vessel for reduction of the subsequent fuel batch.

Lithium is not as powerful a reducing agent as calcium, thus some more stable oxides, i.e., the RE-oxides are not reduced by lithium. Carryover of unreduced rare-earth oxides into the electrorefiner is a major concern in this process, as it will affect the electrorefiner chemistry. One major advantage of the lithium reduction process is that the waste salt stream that it generates is compatible with the ceramic waste form (CWF) that is already qualified as a high-level waste form.

Calcium Direct Oxide Reduction Process

The Direct Oxide Reduction (DOR) process was used for many years in production of plutonium metal from oxide. The DOR process was developed in the U.S. at the Rocky Flats Plant[3], Lawrence Livermore National Laboratory[4], and Los Alamos National Laboratory[5]. In this process, plutonium oxide is reduced with calcium metal to form plutonium metal and calcium oxide [5-7]. The reaction takes place in molten $CalCl_2$ or $CaCl_2$-CaF_2, which serves as a solvent for the calcium oxide and allows the plutonium metal to coalesce at the bottom of the crucible to form a button.

The reduction step for this process is similar to that described above for the lithium reduction step, except that calcium is used as the reductant rather than lithium and the reduction salt is $CaCl_2$ or $CaCl_2$-CaF_2 rather than LiCl. Because calcium is a much stronger reductant than lithium, the RE-oxides in the spent LWR oxide fuel will be converted to metals in this process, along with the TRU metals and the NM fission products. The spent LWR oxide is placed in a crucible with $CaCl_2$ and calcium metal at 850°C and agitated. The oxides are converted to metals that collect at the bottom of the crucible. The salt is recycled by electrochemically decomposing the oxide, discarding the oxygen and recovering the calcium. Because plutonium and neptunium melt below 850°C, they will likely alloy with the solid metals (both minor actinides and fission products). This solid material is then introduced as the anode in an electrorefiner, where the TRU metals are extracted and collected on a cathode. The DOR process has the advantage of minimizing the oxide carryover to the electrorefiner. However, the effect of $CaCl_2$ carryover on the electrorefiner chemistry and the interaction of $CaCl_2$ with the ceramic waste form are the major outstanding issues.

Recent Experimental Studies

Results of laboratory-scale experiments in the ARP program on the first four processes are reported in reference 8. Based on these results, the study initially concluded that the salt transport process (calcium reductant) was superior to the other treatment processes

but that the lithium reduction process also showed promise. However, the extent of TRU-oxide reductions in the lithium reduction process was a major uncertainty. As a result, the salt transport process was selected as the reference process for design of engineering experiments while the lab-scale investigation of the lithium reduction process was continued.

Subsequently, several laboratory-scale experiments were performed to study the lithium reduction process. The results of these experiments, reported in reference [9], demonstrated that the lithium reduction process was suitable for recovery of actinides from LWR fuel. High Li_2O concentration in the reduction salt was identified as the cause of incomplete TRU-oxide reductions in early experiments. Experiments confirmed that the TRU-oxides could be completely reduced if the Li_2O oxide concentration in the reduction salt was maintained below 3.3 wt% at 650ºC. At this point, the lithium reduction process was selected as the reference process primarily because of lower temperatures, less stringent materials requirements, and a waste salt composition that was compatible with the IFR waste salt [1,2]. Thus the engineering-scale development of the lithium reduction process was initiated.

The equipment for the engineering-scale (20-kg batch size) experiments was designed and fabricated and the operation of the process began in October 1993 [2]. An engineering-scale reduction experiment was performed in 1995 with simulated fuel consisting of a mixture of high-fired mixed oxide mock spent fuel pellets with a nominal composition of 5 wt% PuO_2, a high-fired mixed oxide mock spent fuel pellets with a nominal composition of 0.5 wt% PuO_2, and high-fired natural uranium oxide pellets [10]. Engineering-scale reduction experiments were also performed with fuel simulated to represent the core debris from the Three Mile Island-II reactor in 1996 [11]. Four engineering-scale experiments - reduction or electrowinning - were performed between 1995 and 1996. Although the results of these experiments showed that the oxide fuel could be reduced, they also revealed that the reduction times were much longer than those achieved in the lab-scale experiments. In addition, significant problems were encountered in measuring the extent of reduction. In summary, results of engineering-scale reduction experiments indicated that parameters affecting reduction kinetics such as fuel basket geometry, fuel morphology, and transport of lithium to the fuel (design of lithium source) had to be further studied in addition to developing a reliable means of monitoring the extent of reduction. Results of an engineering-scale electrowinning experiment indicated a need for improving the cell design.

A number of lab-scale reduction experiments were performed with UO_2 in 1996 and 1997 to study the many parameters affecting reduction kinetics and scale-up of the process. Lab-scale electrowinning experiments were also performed during this period to improve the cell design, including the design of the anode and the cathode, and parameters affecting process scale-up. Subsequently, engineering-scale reduction experiments successfully demonstrated UO_2 reduction using fuel basket designs that were compatible with the existing Mk-IV and Mk-V electrorefiners, which are being used for the Spent Fuel Treatment operations at ANL, at rates comparable to those achieved in the lab-scale experiments. Electorowinning was also successfully demonstrated at the engineering-

scale. The results of scale-up studies, between 1997 and 1999, are reported in references 12 and 13.

Further work is required in a number of areas related to process design and equipment. Actual irradiated fuel should be used to verify the behavior of the TRUs and fission products. In addition, much 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.

4.5.3.3 Back End Steps

In the case of the electrorefining process developed at the Argonne National Laboratory in the U.S., the high-level waste forms are identical regardless of whether the fuel type being treated is metal, oxide or nitride, metmet, cermet, or cercer. Different waste forms would probably be used for the case of coated-particle graphite reactor fuels or for a process utilizing a fluoride salt mixture, but the discussion will be limited here to the chloride-based processes for the sake of brevity.

In the reference ANL pyroprocess, the active metal fission products (Cs, Sr, etc.) will reside in the electrolyte salt together with trace amount of actinide elements. The transition metal fission products will remain in the anodic dissolution baskets together with the cladding hulls. So, there are two waste streams to be addressed, one salt and one metal. Because a chloride salt is not amenable to vitrification, development of a different waste form was necessary. A natural chloride-bearing mineral, sodalite ($NaAlSiO_4.NaCl$), exists in the geologic structures of the proposed Yucca Mountain repository in the United States. A waste in the form of this mineral can be synthesised by mixing the fission product-loaded electrolyte salt with a zeolite (Zeolite A, $Na_{12}Al_{12}Si_{12}O_{48}$) and heating to temperatures near 900°C. The synthesis is catalysed by the presence of borosilicate glass frit, which also serves to encapsulate the sodalite particles, providing an additional barrier to radionuclide release.

As is the case with vitrified high-level waste, the fission product loading of the waste form is constrained by the limiting centerline temperature of the waste form and the total heat flux that is acceptable in the high-level waste repository or interim storage site. The maximum temperature allowed in the composite glass-ceramic waste form developed by ANL is approximately 650°C - the temperature at which transformation of the sodalite to nepheline occurs. (Nepheline is a related mineral, but one that does not retain fission product cations well.) Because the thermal conductivity of the composite is similar to that of borosilicate glass, the maximum fission product loading is about the same as for glass, around 15 weight percent. The limits on actinide content are higher than for glass (which is limited by the solubility of actinides to around 1 weight percent), but the fission product decay heat limit prevails. Little is known about long-term degradation of the sodalite structure by α-particle damage.

A variety of leach tests conducted with differing leachant chemistries and environmental conditions have shown that the release rate of important radionuclides from the glass-ceramic waste form is comparable to the releases from the best vitrified waste forms and well over an order of magnitude less than the release from spent fuel under direct disposal conditions. These tests have been performed in evaluation of the potential behaviour of this waste form in the oxidising environment characteristic of the Yucca Mountain repository. Under reducing conditions, the behaviour is not known.

The metallic waste stream is generated by removing the anode baskets from the system and melting the baskets together with the metallic fission products and the cladding hulls to produce a corrosion-resistant metal alloy. The composition of this alloy is dependent on the nature of the cladding material, which dominates the mass of the waste form. A metal waste form based on stainless steel cladding hulls has shown a release rate for technetium that is several orders of magnitude less than than the release of technetium from spent LWR fuel.

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4.5.4 Other Processes – Volatility and Reductive Extraction

Volatility 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. A small-scale fluoride volatility processing system was operated jointly by NRI (Czechoslovakia) and RIAR (Russia) in the 1980s. This process was used for treatment of oxide fuel from the BOR-60 reactor. Powdered fuel was fluorinated, converting uranium to UF_6 and most of the plutonium to PuF_6, both very volatile products. Re-fluorination was necessary to convert the remaining plutonium to the PuF_6 species. Most of the fission products formed non-volatile fluorides, while the noble metal fission products formed semi-volatile fluorides that tended to accompany the uranium and plutonium. After extraction of niobium and ruthenium fluorides, uranium and plutonium fluorides were then condensed at –60 and –80°C. After collection in the condensers, the uranium and plutonium fluorides were

reheated, decomposing the PuF_6 to solid PuF_4, thereby allowing the volatile UF_6 to be separated from the plutonium. Column distillation was then used to purify the UF_6 by separating molybdenum, iodine and technetium fluorides.

Volatility processes have been proposed for application in the various separation schemes envisioned for the U.S. Accelerator Transmutation of Waste (ATW) program. A chloride volatility process forms the head-end of the treatment process for a fuel consisting of a metallic dispersion of TRU-Zr alloy in a zirconium matrix. Chlorination of the fuel in a LiCl carrier salt is followed by volatilisation of the zirconium as $ZrCl_4$. Digestion of the matrix material then provides physical access to the transuranics and fission products, which are subsequently separated by a reductive extraction process. Fluoride volatility processing has been proposed for ATW processing of TRISO (pyrolytic carbon coated) fuel. Without question, fluoride volatility processing is a powerful technology; on the other hand, it is a technological challenge to control the process so as to achieve very high efficiency, avoid ancillary recovery steps, and comply with environmental and safety regulations. Much more development is needed to realise the full benefits of this family of process methods.

Reductive Extraction Processes

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. The fuel can be chlorinated with $CdCl_2$ in a LiCl carrier salt, resulting in the formation of chlorides of the actinides, the Group I/II fission products, and the lanthanide fission products. The noble metal fission products will not form chlorides and will either sink to the bottom of the chlorination vessel or remain in the basket originally containing the fuel. The salt can then be contacted with a molten metal consisting of a dilute solution of lithium in cadmium. Lithium will reduce the actinide chlorides and, to some extent, the lanthanide chlorides. If the process is operated in countercurrent mode in a train of high-temperature centrifugal contactors, about eight contactor stages can effect extraction of over 99.99% of the actinide elements into the cadmium phase, with a lanthanide contamination of this product of less than 10% of the lanthanide elements present; i.e., excellent recovery of actinides contaminated with about 10 weight percent lanthanides. Greater purification requires an increased number of contactor stages. The process has been demonstrated with four contactor stages and provided excellent separation. A similar metal/salt reaction process has been reported for removal of noble metal fission products from oxide fuel that has been reduced with hydrogen and then fluorinated, with the noble metal fission product elements being digested in liquid antimony, tin, or zinc.

As described earlier in this chapter, several of the possible fuel cycle schemes can make use of pyrochemical processing methods. Because these methods tend to be non-selective with respect to the transuranic elements, they are applicable in a broad variety of systems. Process selection must be made with full awareness of the potential for

contamination of the separated TRU product with traces of lanthanide fission products. If the performance of the FR or ADS fuel is sensitive to the presence of lanthanides, then certain steps, such as the use of several reductive extraction stages, must be called upon to reduce the lanthanide contamination.

4.5.5 Processes for Coated Particle Fuels

Fuel economy, breeding and recycling were development goals in the early days of most nuclear developments in anticipation of large-scale deployment of nuclear energy and the limited resources of uranium. Today, such goals may again become important for sustainable development, but increasingly motivated by the need to reduce long-lived waste to be discharged for final disposal and by the increasing cost for waste transport, treatment and storage. Whereas the intended symbiosis of LWRs and FRs was primarily based on the uranium/plutonium fuel cycle with rather low fuel burnup and multiple recycling, the HTR development followed two specific objectives:

- Increase of fuel burn-up much beyond 100 GWd/t making recycling dispensable;
- Introduction of the thorium cycle to make use of an additional resource

In addition, the thorium cycle generates considerably less transuranics than does the uranium cycle (see Table 4.5.5-1) because the fission cross sections of the daughter products are always larger than the absorption cross sections. Table 4.5.5-1 illustrates that the fresh, three-component fuel (consisting of graphite, ^{233}U fuel, and ^{232}Th breeding material) is being converted to a multi-component system. However, the table indicates that the thorium cycle generates considerably less transuranics than does the uranium cycle (see Table 4.5.5-1); this occurs because the fission cross sections of the daughter products are always larger than the absorption cross sections. This characteristic also motivates consideration of the thorium cycle for LWRs, as it might help to minimize the generation of long-lived waste in existing reactors in a medium-term perspective.

Table 4.5.5-1 Composition of Fresh THTR Fuel and Spent THTR Fuel after Irradiation to 12% FIMA Burnup

Content	**Fresh fuel [g]**	**Spent fuel (12% FIMA) [g]**
Graphite	~190	~190
Th-232	10.2	9.25
U-233	0	0.237
U-235	1.0	0.028
U-236	0	0.179
U-238	0.075	0.056
Np-237	0	0.026
Pu-total	0	0.013
Fission products	0	1.2

Today, the development of HTR fuel cycles is devoted to different purposes:

- ultra-high burnup (~200 GWd/t) of Low-Enriched Uranium (LEU) fuel for direct disposal of spent fuel (no need for recycling);
- ultra-high burnup (~700 GWd/t) of weapon-grade plutonium for reduction of military plutonium stockpiles and safe enclosure with little danger for proliferation;
- ultra-high burnup (~700 GWd/t) of plutonium from recycled spent MOX fuel from LWR; direct disposal of the residual spent fuel (no recycling).

Given these objectives, there are no actual R&D programmes on recycling of HTR fuel underway. In addition to optimisation of the fuel particle design for the different targets, investigation of the behavior of HTR-fuel under various disposal conditions must become an additional focus of R&D. The coated particle represents a very effective encapsulation of the fission products and of the residual fissile materials. In addition to the advantages for retaining fission products, even during core heat-up accidents, the particle coating can also be considered as a preparation of the fuel for final disposal, incorporated during the initial fabrication step. In addition, the fissile content is reduced by ultra-high burnup and 'diluted' over a huge quantity of particles and thus comparable to vitrification with regard to proliferation resistance.

In the following, the state-of-the-art of recycling of spent HTR fuel is shortly described to show that this can also be a viable future option, if necessary, and that feasibility problems have already been overcome in former R&D programmes in Germany and the U.S. It should also be mentioned that the large stockpile of residual fresh fuel from THTR containing High Enriched Uranium (HEU) and thorium has been recycled at Dounrey in UK without major problems on a commercial base. The HEU/thorium cycle was mainly given up with regard to proliferation risk evaluations (refer to the International Nuclear Fuel Cycle Evaluation, or INFCE, of the early 1980s) although it shows significant advantages with regard to fuel economy and reduced generation of long-lived waste. The relative advantages and disadvantages of the HEU/thorium fuel cycle depend on the actual situations and could lead to other results if the waste aspects are dominating and if proliferation resistance can be improved by other means.

Requirements & Boundary Conditions

Development of HTR fuel recycling in Germany and the U.S. was based on the former reference fuel compositions containing HEU and thorium either as mixed oxides/carbides or with separate fuel and breeding particles embedded in the fuel matrix. The coatings were first applied as double pyrocarbon layers (BISO) and used in fuel for the demonstration reactors THTR and Fort St. Vrain. Later, TRISO coatings with an additional SiC layer were developed to improve fission product retention for application to direct cycle gas turbine plants.

The main requirements, in Germany for example, were:

- Separation of the fuel kernel from the graphite matrix and from the coatings (head end)
- Selective separation of fuel (HEU) and breeder (thorium) material

- Yield in fissile material recovery > 98-99%
- High separation / decontamination factors for the fission products ~ 10^6
- High reliability of the whole process; minimal number of process steps
- Value of recovered fissile material > cost for recycling

In contrast to recycled 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. On the other hand, the relatively hard γ radiation is an effective deterent against misuse of U-233, similar to the case of vitrified waste forms for plutonium disposition, in which relatively low-activity weapons-grade Pu is mixed and diluted with fission products.

State-of-the-Art in Germany

Whereas commercial recycling plants for the U/Pu cycle of LWR were already available, such a technology remained to be developed for HTR fuel. In Germany, a project entitled "Recycling of Thorium-based Fuel" was founded by KFA-Juelich and other partners from the chemical and nuclear industry. A first milestone was reached in 1970-71 by finalizing small-scale tests with "cold" and "hot" material on a laboratory scale. This allowed initiation of process engineering for commercial recycling plants and the defining of the process scheme for a demonstration plant called "Juelich Pilot Plant for Thorium Element Recycling (JUPITER)" to be installed in the KFA hot cells. These investigations were performed in conjunction with the AVR prototype providing the spent fuel and the test disposal site ASSE, a former salt mine, where the disposal behavior of HTR fuel and residue from recycling were tested.

The main steps of HTR fuel recycling 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

Figure 4.5.5-1 shows the JUPITER flow scheme, starting with destruction of the spherical fuel elements in a hammer mill to fragments of a size suitable for a fluidized-bed burner, where the fuel is burned with oxygen at 800-850°C. Fines escaping from the burner were recycled and the off-gas passed through packed-bed adsorbers to remove radioactive contaminants.

The remaining spent fuel, which is in the form of oxides, is dissolved in THOREX agent (13 M HNO_3, 0.01 M $Al(NO_3)_3$, 0.05 M HF). The dissolver product, a high-molarity nitric acid solution with a thorium concentration of about 1 mol/l, is first passed through filters for clarification and then transferred to the feed adjustment system where acid stripping is applied to produce a solution corresponding to solvent extraction conditions.

Mutual separation of uranium and thorium is achieved by tri-n-butyl-phosphate (TBP) extraction. The solvent extraction system of JUPITER was very flexible, permitting tests of various processes like Acid THOREX processes with 30 Vol% TBP/n-dodecane, interim processes with 5 Vol% TP/n-dodecane and combinations of both.

In addition to the main processing steps, auxiliary units like product handling, solvent regeneration, nitric acid recycling, clean-up of the dissolver, evaporator and vessel off-gas were also developed and tested.

It was foreseen to place spent fuel discharged from the reactor into an intermediate storage prior to recycling for two reasons:

- Allowance for decay of Pa-233 into the new fissile material, U-233;
- Limiting the radiolytic dissociation of process fluids, which is of essential importance for wet chemical processes.

The throughput was set at 2 kg of heavy metal per day as this seemed to be a good basis for extrapolation to commercial plant sizes. The plant got a first partial licence in 1975 as a pre-condition for construction. In 1978, tests began with surrogate material showing that the head end was working properly for BISO fuel.

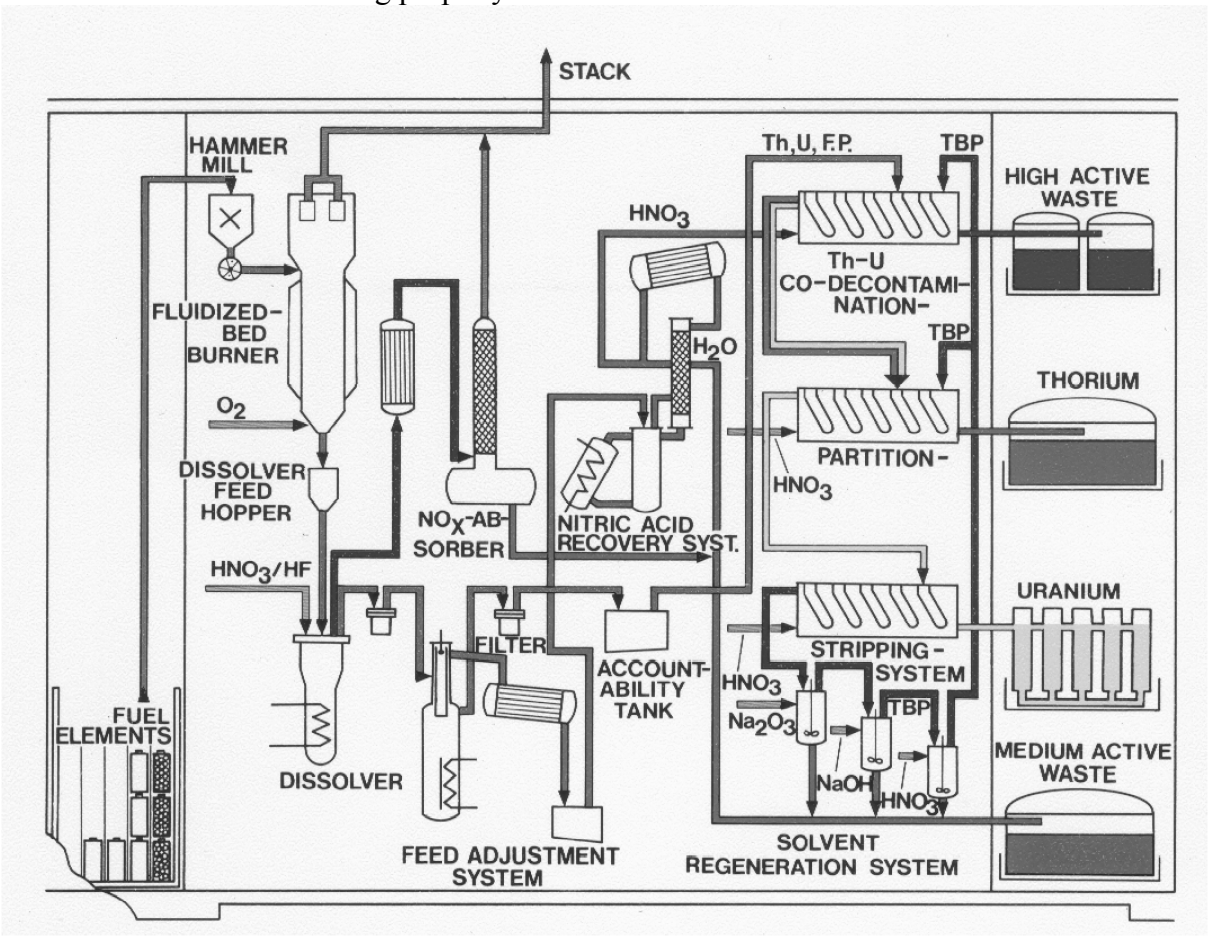

Figure 4.5.5-1. Flow scheme for Juelich Pilot Plant for Thorium Element Recycling, or (JUPITER)

In parallel to these developments, a change from HEU BISO towards LEU TRISO-coated particle fuel was initiated due to recommendations from INFCE and because of the improved fission product retention of TRISO fuel. This led to the need for considerable change in the head-end stage because the burning process ends at the SiC layer, whereas the BISO coating could be completely burned. Thus, the SiC layer must be destroyed mechanically before the residual carbon could be removed in the fluidized-bed. Different methods have been investigated in depth:

- Jet blast with bounce plate
- Diamant disk mill
- Particle jet mill
- Double-roll-crashers

The pneumatic particle jet mill method was slected as a reference because it allows a simple separation of the kernel material and the coating in the same stage and does not require moving mechanical parts like the double-roll-crasher, which also performed very well when tested with Fort St. Vrain fuel.

Until 1985, the head end of the JUPITER plant processed more than 10,000 fuel pebbles of different types (HEU-BISO, HEU-TRISO, LEU-TRISO, graphite balls) showing a very reliable operation and effective function. 'Hot' operation was not started because the back-end strategy for the HTR was changed to direct disposal instead of recycling. The results from the JUPITER plant have been documented in detail, allowing for future use of this know-how. Many alternative head-end processes have also been investigated for the disintegration of complete fuel elements, showing that it is possible to avoid the release of ^{14}C to the atmosphere, which is concern when burning the graphite for extraction of the kernel material.

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 categorised 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 categorised as ILW. In Tables 4.5.6-1 and 4.5.6-2, categories are assigned to the secondary wastes that would potentially arise during the operation of principal fuel cycle schemes.

It is notable that in the analysis of secondary wastes reflected in Tables 4.5.6-1 and 4.5.6-2, construction and decommissioning wastes have not been included. However, it is obvious that increased number of multistage processes and greater shielding requirements for handling product streams containing heavy concentrations of minor actinides would generate considerably more decommissioning wastes that would include secondary, active, wastes. Such wastes would arise from the sorption of material onto process surfaces and accumulation of active fines, etc.; but these cannot yet be realistically quantified.

Front End Secondary Wastes

The front-end of fuel cycles have common stages associated with the production of uranium oxide fuel for irradiation in conventional thermal reactors. These fuel cycles will generate wastes during the mining and milling stages, which are reported to represent the major health and environmental hazards of the nuclear industry [1]. Secondary wastes in the form of uranium residues in the tails from milling operations represent a definite anthropological hazard with respect to the in-growth of radioactive progeny. The generation of fugitive dust from tailings dams can have considerable impact on the local populations. Additional secondary wastes include sulphuric acid and ammonia that are used during the leaching and purification of uranium ore concentrate, used filter cloths, contaminated solvent extraction raffinate, dilute hydrofluoric and hydrochloric acids and magnesium or calcium fluoride slag. The use of process and ancillary service materials such as petrol and diesel will also generate additional secondary wastes. However, the stages that follow mining and milling generate considerably less waste and have a lower impact on the environment. Conversion and enrichment processes are essentially clean processes that generate very small quantities of secondary waste, mostly in the form of scrap metal and analytical wastes.

The power production, or reactor, stage of the nuclear fuel cycle generates relatively little secondary waste. The secondary waste that does arise will arise primarily from the treatment of coolant, and will include corrosion products, spent demineralization resins, filters, ancillary service material and general maintenance and analytical wastes. Secondary waste generation during fast reactor operation would be relatively small in comparison to that from recycling operations. Ancillary service wastes would be expected to be similar to those of the thermal reactor stage however, the secondary wastes associated with coolant could differ and would depend on the coolant media, e.g. sodium, employed in the fast reactor system. A coolant treatment system would be necessary that would operate on a continuous or periodic basis to remove potential corrosion and activation products.

Back End Secondary Wastes

It is assumed that the final disposal stage would entail long-term storage in a deep geological repository. The primary waste associated with this stage would be the waste rock inventory, which itself is radioactive, generated during construction of the

repository. The secondary wastes are difficult to identify; however, it seems feasible to assume that secondary wastes from the related ancillary services similar to those generated during mining operations would arise. In contrast, the irradiated fuel from the thermal reactor in the multiple recycle schemes would be subjected to processing following an appropriate post-reactor cooling period. The secondary wastes and the associated increase in dose in most multi recycle scenarios will arise from the separation and treatment of irradiated fuel components, and their subsequent fabrication into new fuel or transmutation target materials.

However, the stages downstream of thermal reactor operation in each of the fuel cycles differ considerably. Analysis of the back-end of the nuclear fuel cycle shows the increased secondary waste generation of multi-recycle schemes.

Table 4.5.6-1. Summary of secondary wastes arising from fuel cycle front end and reactor

Fuel Cycle Stage	Category*	Secondary Wastes
Mining & Milling	LLW	• Residual uranic wastes contained in drilling mud and displaced rock.
	LLW	• Sulphuric acid, ammonia, used filter cloths, activated solvent extraction raffinate, dilute hydrofluoric and hydrochloric acids, and magnesium or calcium fluoride slag.
	LLW	• Emissions from ancillary services.
Enrichment & Conversion	LLW	• Maintenance wastes: small quantities of coolant, inert gases, scrap metal, polythene, paper towels, etc.
	LLW/ILW	• Analytical wastes
	ILW	• Conversion: trace uranic quantities from scrub liquor and crud
	LLW	• Conversion: wastes from ancillary services such as polythene, paper towels, carbon and cloth filters, etc.
Power Production / Transmutation	ILW	• Cooling system maintenance – corrosion products, isotopic activation products
	LLW/ILW	• Spent demineralisation resins, and ancillary service material e.g. cooling tower blowdown, sewage, used oil, contaminated oil, wastewater treatment sludge, etc.
	ILW	• ADS: Spallation and activation products associated with coolant irradiation, e.g. formation of ^{210}Po from neutron capture in ^{209}Bi (in LBE coolant / target material), ^{202}Pb and ^{205}Pb, ^{208}Bi. Activation products such as Mo, Ni, Cr, Fe, etc., from irradiation of structural components such as the beam window and beam tube.
	ILW	• Corrosion products from mechanical degradation of structural materials
	LLW	• Activated air & offgas emissions from: target venting, coolant system, inert purge system, beam dumps
	ILW	• PPE, HEPA filters, batteries, scrap metal & glass, etc.

* Radioactive waste category: LLW – Low Level Waste; ILW – Intermediate Level Waste ; High Level Waste

Table 4.5.6-2. Summary of Secondary Wastes Arising from Fuel Cycle Back End

Fuel Cycle Stage	Category*	Secondary Wastes
PUREX	ILW	• Degraded solvent† and wash raffinate containing organic solvent degradation products such as DBP, MBP, alkanes, nitro-alkanes, carboxylic acids, carbon dioxide, and phosphoric acid.
	ILW/HLW	• Trace quantities of insoluble U, Pu, FP, and MA dissolver solids, and soluble traces in solvent wash raffinates, e.g. from processing of thermal oxide fuel, typically around 0.8 $g.l^{-1}$ U, 8 $mg.l^{-1}$ Pu and 0.3 $mg.l^{-1}$ Np.
DIAMEX (Fission Product and Transuranic separation)	ILW / HLW #	• 11% initial Ru in solvent wash raffinate, possibly removed in scrub liquor following oxidation to RuO_4,
	ILW	• Possible Fe (corrosion product) accumulation in solvent
SANEX (Trivalent Actinide and Lanthanide Separation)	ILW	• Scrub liquor, from off-gas scrubbing
	ILW	• Organic solvent degradation products – dependent upon extractants, e.g. TPTZ, or *di*-thiophosphoric acids in combination with TBP
	ILW	• Trace quantities of Am, Cm, from process raffinate
SESAME (Am / Cm separation)	ILW	• Contaminated condensate containing Am and Cm from product streams and associated scrub liquor
	ILW	• Waste solvent (solvent cleanup and recovery operations) possibly containing activated ^{110}Ag and waste heteropoly acid
	LLW	• Acidic wastes, phosphoric acid,
Pyroprocessing (Pu / MA recovery from spent fuel)	ILW	• Scrub liquor from off-gas treatment
	ILW	• Cladding hulls, noble metals @
	ILW	• Waste eutectic salt & trace inclusions (including the more electropositive FPs) **
	HLW / ILW	• Ion exchange zeolites, waste crucibles, scrap metallic waste.
Storage (Intermediate) & Final Disposal (repository)	LLW	• Ancillary service wastes

* Radioactive waste category: LLW – Low Level Waste; ILW – Intermediate Level Waste ; High Level Waste

† Solvent is stored for degradation

\# Dependent upon cooling times

@ Potentially HLW as the noble metals could include Ru

** Potentially HLW – could include ^{90}Sr etc.

Secondary Wastes from Aqueous Separations

It is generally acknowledged that in addition to the unwanted fission products, fuel cladding, etc., that would constitute the primary fuel cycle PUREX waste from a recycle scheme, there would be generation of secondary waste. This secondary waste would comprise all insoluble active residues, degraded solvents or salts, ancillary materials, and analytical wastes, etc. that arise during additional fuel cycle operation. It is suggested that the majority of secondary wastes, which arise during processing operations would be generated during solvent/salt cleanup and recovery operations [2].

PUREX is proposed for the processing of thermal uranium oxide fuel, and it is also proposed that advances in PUREX technology would allow the recovery of Np. It is suggested that a multistage processing operation involving the combination of DIAMEX-SANEX-SESAME technologies be implemented to allow the efficient separation of americium and curium from PUREX HAR and to strip the undesirable lanthanide fraction. The primary wastes produced by this multiplex processing system would comprise high-level waste (HLW) of lanthanides, fission products and trace quantities of actinides.

The secondary wastes from separations processes are included in Table 4.5.6-2. The solvent used in DIAMEX, DMDBTDMA in kerosene or TPH, conforms to the CHON principle, which implies that the spent solvent would be totally incinerable and as such would reduce the total solid secondary waste arising. However, a solvent waste stream containing up to 11% of the ruthenium in the feed to the DIAMEX process would be generated during solvent processing [3]. The ruthenium fraction could be oxidised to volatile RuO_4 prior to DIAMEX separation although this effectively generates a new RuO_4 secondary waste, which would require scrubbing from the solvent. Iron could accumulate in the solvent phase also, potentially resulting in the formation of a problematic third phase, which would require subsequent removal from the solvent during clean-up processing. Additional scrubbing cycles would be required to remove any residual ruthenium prior to incineration of the spent solvent. Radiolysis of the solvent would lead to the formation of carboxylic acids, amine, amide-acid and other degradation products, which decreases the solvent extracting properties [4].

SANEX would generate secondary waste from off-gas scrubbing and by radiolysis of the organic solvent. The presence of ^{242}Cm and ^{241}Am would increase the $\propto$-level activity and promote solvent degradation. The organic solvent degradation products would depend on which soft donor extractant is used, as several have been suggested. Similarly, SESAME would generate waste streams contaminated with trace quantities of Am and Cm from product finishing. The waste solvent could be polluted with activated ^{110}Ag, and quantities of waste complexing agents, heteropoly acids such as potassium phosphotungstate, would be generated. The advanced separation processes referred to above are currently in the development stage, and as such it is not possible to provide an accurate account of the potential secondary wastes associated with their operation. However, it seems feasible to suggest that implementing such complex multistage

processes would generate additional secondary wastes, which would further contribute to the operational dose of the fuel cycle and toxicity of the wastes.

The majority of secondary wastes that arise during processing operations would be generated during solvent cleanup and recovery operations [2]. It is feasible to assume that secondary wastes generated during PUREX processing for thermal uranium oxide compared to FR-MOX fuels would be similar, although probably different in magnitude. Solvent radiolysis or hydrolytic decomposition during PUREX operation would generate the major fraction of secondary wastes including organic degradation products such as DBP, MBP, alkanes, nitro-alkanes, carboxylic acids, carbon dioxide, and phosphoric acid. The α-level associated with recycle FR-MOX fuel would be approximately 30 times greater than the equivalent UO_2 fuel of the same irradiation. Consequently, the extraction of FR-MOX fuel that has been subjected to high irradiation would result in increased formation of DBP via solvent radiolysis, due to the increased Pu and MA inventories in the spent fuel (average 140 kg DBP/ TWhe for fast reactor). Trace quantities of insoluble crud would be found in solvent wash raffinates. For example, the solvent wash raffinate from the processing of thermal oxide fuel would typically contain around 0.8 g./l U, 8 mg/l Pu and 0.3 mg/l Np. The increased degradation of solvent and diluent expected during the processing of FR-MOX fuel might be accompanied by an increased inventory of insoluble actinide residues, the magnitude of which would depend on the complex interactions of many operating variables such as initial fuel composition, irradiation, dissolution conditions, etc. Distillation of the solvent wash raffinate reduces the volume but generates a higher-boiling residue containing traces of uranium, plutonium, and minor actinides. The recycle of Pu in the FR stage of the fuel cycle would generate quantities of secondary waste subject to accumulation with each cycle.

Secondary Wastes from Pyrometallurgical Separations

Pyrochemical processing operations tend to produce little secondary waste because there is little or no reagent degradation. If the processes are properly designed and operated, high-level waste volumes can be minimised 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.

The volume of waste emanating from pyrochemical processing operations is somewhat dependent on the nature of the fuel being processed and even more dependent on the complexity of the overall process. Although there is little extended large-scale experience with such operations, it is possible to estimate the resultant waste volumes on the basis of laboratory-scale testing.

The secondary wastes associated with the pyroprocessing of fast reactor TRU containing fuel would include zeolites, used for ion exchange purification of the process salt

medium, and waste eutectic salt and inclusions such as $CaCl_2$, FP, $MgCl_2$, LiCl, KCl, Li_2O, etc. It has been estimated that 0.3 m^3 of mineral wastes and 0.05m^3 of metallic wastes would be generated per tonne of recycled material [5]. Zeolite ion exchangers would be used for extracting fission products and rare earths, after which they would be collapsed and mixed with anhydrous zeolite to promote salt uptake into the crystalline matrix prior to conditioning for final disposal. It is suggested that such waste forms would typically contain around 4 wt.% FP. Recycling of the molten salt eutectic mixes could potentially reduce the generation of secondary wastes during pyroprocessing [6]. In addition there would be generation of process fines that would be subject to sorption onto the internal surfaces of equipment, cells and filters. However, the magnitude of secondary wastes generated during pyroprocessing is hypothesised as being relatively low owing to the reduced or negligible reagent degradation, based on extrapolations of current knowledge.

References for Section 4.5.6

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4.5.7 Depleted and Recycled Irradiated Uranium

The management of depleted uranium and recycled 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 Gen-4 may, however, reduce the amount of long-lived high-level waste by a significant factor; i.e. a factor hundred or more. Therefore, the management of this depleted uranium and recycled irradiated uranium may become a more apparent issue in the future if such recycle schemes would be deployed.

Today, the strategy for the long-term management of depleted uranium is based on the consideration that this depleted uranium is a valuable material, which may have various applications, and is not considered a waste. The use of this depleted uranium in fast reactor systems, as shown in this study, is one of the applications where the 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.

Depleted uranium, initially occurring as UF_6, can be stored safely for many decades in steel containers in the open air in storage yards. However, depleted uranium stored in the UF_6 form may represent a potential chemical hazard if not properly managed. Alternatives for the strategic management of depleted uranium therefore include the deconversion of UF_6 stocks to stable forms more suitable for long-term management. Due to their high chemical stability and low solubility, uranium oxides in general are favoured. Generally, though not exclusively, storage as U_3O_8, the most stable oxide, is considered for long-term storage where continued storage as UF_6 is not appropriate. Large-scale storage of compacted U_3O_8 has been undertaken in France since 1984. About 130,000 tU of UF_6 have already been converted into U_3O_8. The powder is held in about 3-m^3 painted mild steel DV 70-type containers, stacked three high in 'warehouses'.

The radiological characteristics of DU are a consequence of the properties of the three uranium isotopes: ^{238}U, ^{235}U and ^{234}U and their daughter products. DU is safe against criticality under all naturally occurring conditions. The initial activity of DU when it is newly produced is very low, around 23 Bq/g, and the radiotoxicity is also low, around 0.75 Sv/g. Whatever the ^{235}U assay of uranium, as time passes, decay products will appear - meaning that the activity and the radiotoxicity will increase. The activity and radiotoxicity levels for DU will become the same as for uranium ore (with the same original amount of uranium) after a time period of around 1 million years.

Recycled irradiated uranium is currently not systematically recycled in UOX- or MOX-fuel form. The recycled uranium distinguished itself from natural uranium by the occurrence of higher amounts of γ-emitting isotopes:

- ^{232}U, not present in natural uranium, has a higher specific activity than ^{235}U and some of the daughter-products are γ-emitting, i.e. ^{228}Th with a half-life of 1.9 years, as well as emitting hard γ-rays, i.e. ^{208}Ta. The γ-activity attains a secular equilibrium after about 10 years attaining significant higher values than for natural uranium.

- ^{234}U occurs as a natural $\alpha\gamma$-emitter and is accumulated in this irradiated uranium. In addition, Np and Pu are also occurring in recycled irradiated uranium and add to the α-activity.
- Trace amounts of ^{106}Ru occur which increase the α-activity.

The nuclear industry has in place the facilities that are needed to recycle recycled 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. In Japan, JNC carried out experiments on the REPU conversion technology at Ningyo-Toge conversion facility. The amount of REPU converted to UF_6 for re-enrichment reached 336 tU. Recycling of REPU from LWR fuel is now demonstrated, albeit limited to a fraction of the available material and to a few numbers of reactors. (Substantial quantities of REPU from MAGNOX fuel have been re-enriched in the UK to natural level for recycling, while the Doel-1 reactor in Belgium has been operated exclusively with re-enriched REPU for a number of years.) The enrichment of REPU is today based on centrifuge technology and is currently performed in Russia and in the Netherlands. Altogether, the recycling plants have delivered more than 12,000 tonnes of REPU, and this amount will further grow in the years to come in a pace comparable to the output of a large uranium mine. More recently, an alternative way of REPU reuse has emerged which relies on blending REPU with high-enriched uranium resulting in a reduced ^{236}U-content in the enriched REPU fuel, thus reducing neutron absorption and improving the economy of REPU recycling. If REPU is considered as a by-product from recycling, the economics realised in replacing natural uranium by REPU compensate for the extra expenditures related to storage, conversion, enrichment and fabrication, depending on the source material. The competitiveness of REPU is strongly related to the cost of recycling and of natural uranium, but also to its isotopic composition that varies from a batch to the other. The future of REPU is anticipated to be governed by economic and strategic, e.g. conservation, considerations. Considering the savings that can be realized by recycling REPU of relatively good specifications, despite an extremely depressed natural uranium market, it is possible that in the coming years, those utilities having selected recycling will also recover uranium through recycling.

In conclusion, as long as LWRs would make up a significant part of the nucler power plant park in future fuel cycle schemes, an increasing amount of depleted and recycled irradiated uranium would emerge. In particular, the latter would be of increasing importance as depleted uranium would be used in fast reactor systems where the recycling of recycled irradiated uranium would not compensate for its production. Only the use of this recycled irradiated uranium in future all fast reactors scenarios can provide a steady decrease of the build-up inventory of depleted uranium and the recycled irradiated uranium. In the very long term (about 1 million years), disposition of this depleted and recycled irradiated uranium results in a activity and radiotoxicity 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 recycle 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, seem to offer potential for recycling of advanced fuels due to their good compatibility with most fuel forms and their high radiation resistance. In addition, the 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 recycling should not be considered as competing but, rather, as complementary technologies. In a double-strata concept for ins,tance, aqueous processes could assure the recycling of the first LWR cycle (first stratum) including MA's. In the second stratum the FBR or dedicated burner reactor could rely on dry recycling techniques.

Regarding fuel development, mixed transuranium oxide fuel is the closest analogue to current European MOX fabrication technology, and for that reason it is an important candidate fuel for future energy production. It clearly has advantages with respect to fabrication and thermo-mechanical stability, but less promising thermal properties may limit its application. Mixed nitride fuel forms a reasonable alternative for the (homogeneous) mixed oxide if high temperature stability can be addressed, for example by addition of ZrN. Composite fuel is considered the most innovative of these because the properties can be tailored to the application. However, the limitations of this fuel type need to be investigated thoroughly.

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 reactors. There are major differences in waste quantities, characteristics, and costs for the four different fuel cycle types. The major waste streams are from the front-end of the fuel cycle (mining and milling) and the back-end. For 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 characteristics 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 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 clearly suitable. However, as with most energy technologies (fossil, with carbon dioxide, sulfur dioxide, etc. emissions; renewable sources, requiring energy and materials inputs for manufacturing, use of heavy metals in solar cells, required land use, etc.; and nuclear), the scale of operation can alter the preferred waste management strategy. If nuclear power is used on a truly 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.

4.6.1 Disposal Strategies: Overview of Approaches

This section describes, in a broad context, sustainable waste management principles and practices. The description provides the basis for understanding the recommended R&D for Gen. IV systems.

There are three approaches to waste management that 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.

- *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.

The relative emphasis on particular approaches to waste management depends upon societal goals, economics, and available technologies.

4.6.1.1 Containment Strategies

Radionuclides can be isolated until they decay to non-hazardous concentrations - with the isolation period dependent upon the initial concentration and the radionuclide half-life. This is the primary method for disposal of radioactive wastes. Three types of containment systems exist. The choice of system depends upon the half-life and radionuclide concentration.

Engineered disposal

Some wastes have sufficiently short half-lives that the wastes can be stored in an engineered facility until they are non-hazardous. Similarly, 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 and 500 years for Class C waste. (Note: the European waste classification system has similar classes of wastes but the definitions are fundamentally different). In each case the engineered disposal facility is designed to survive for sufficient time to allow decay of radionuclides.

Engineered storage

Wastes may be stored in engineered facilities to allow sufficient radioactive decay to reduce the cost and reduce the risks of ultimate disposal of the wastes. At the end of some period of time (1) the wastes are moved to a disposal facility, or (2) the engineered storage facility is converted into a disposal facility. Major studies on engineered storage are now underway in France (Section 4.6.2). Engineered storage allows for radioactive decay that provides several benefits:

- *Reduced radioactivity*. SNF and HLW have many short-lived radionuclides. Storage allows these radionuclides to decay away. This reduces handling, transport, and disposal risks.

- *Reduced heat load*. Decay heat can complicate radioactive waste transport, processing, and disposal. Decay heat can degrade disposal facilities by (1) accelerating degradation of waste forms, waste packages, and facilities, (2) degrading the capability of the disposal site to contain radionuclides, and (3) altering groundwater movement [see below]. Figure 4.6.1 shows the decay heat from SNF and HLW as a function of time.

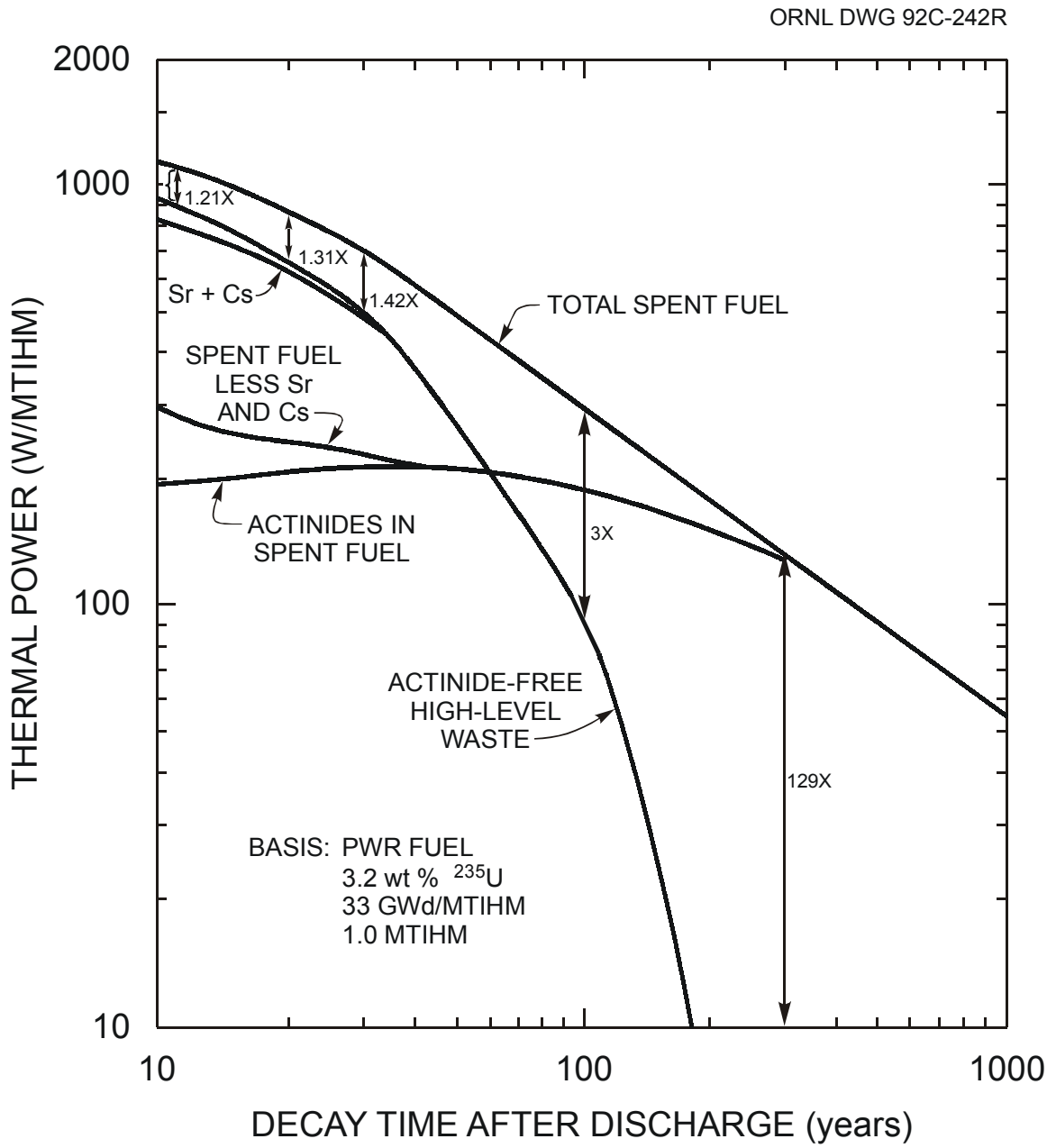

Figure 4.6.1-1. Decay heat from spent nuclear fuel as a function of time

Geological isolation

Geological isolation refers to disposal of radioactive wastes at substantial depths below the earth's surface, which is required for isolation of long-lived radionuclides on the earth. Wastes are isolated by engineered and natural geological barriers to provide time (over tens to hundreds of millennia) until radioactive decay occurs. Geological isolation may be accomplished under land or under the seabed. Methods of emplacement include construction of mined geological repositories and boreholes. Mined geological disposal is the primary disposal option being considered for technical, economic, and institutional reasons.

There are several operating geological repositories for chemical wastes in Europe and one operating repository for radioactive wastes in the United States. The oldest geological repository is Herfa-Neurode in Germany for hazardous chemical wastes. This repository was opened in 1975 and currently disposes of ~200,000 tons/year. It is located in a salt deposit at a depth of ~600m. The geological repositories currently used for hazardous chemical wastes are far larger than any proposed repository for radioactive wastes.

The Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico in the United States is the only currently operating geological repository for radioactive wastes. It surrently operates in a startup mode and is designed to accept alpha-emitting wastes with low heat generation. It is legally restricted to accepting defense-related transuranic wastes.

Multiple factors have led to the conclusion that geological disposal is required for isolation of long-lived radionuclides.

- *Geological history*. Geological history indicates that deep disposal should successfully isolate long-lived radionuclides from man until they are non-hazardous. Typical rock a few hundred meters underground has remained undisturbed for hundreds of millions of years. There are massive quantities of toxic elements (lead, mercury, etc.) and radioactive materials in the earth's crust. Most of these materials have had no impact on man because they are well isolated from the biosphere. Typically, repository performance will be a million or more times more effective than shallow-land disposal of radioactive wastes. Continents erode at an average rate of 600 m per billion years; thus, it takes a very long time to unearth a geological repository by erosion - the final failure mode.

- *Surface events and erosion*. The planet's surface is constantly changing. Erosion, glacial activity, earthquakes and many other natural phenomena change surface and near-surface facilities within decades to centuries. As discussed in the previous point, subsurface structures typically remain undisturbed for long periods of time.

- *Human behavior*. Several millennia of history show that humans driven by curiosity or gain break into, or disturb, surface and near-surface structures. Institutions have not successfully maintained unviolated facilities for durations longer than several centuries.

The performance of a geological disposal facility depends upon engineered and manmade barriers. These barriers are chosen to prevent or slow radionuclide releases by three failure modes.

- *Water migration*. The expected failure mode for a geological repository is dissolution of radionuclides in groundwater and migration of those radionuclides to the open environment. Low solubility waste forms, long-lived waste packages, slow groundwater movement, and sorption of radionuclides on rock provide delay times to allow for radioactive decay to occur prior to release of those raionuclides into the biosphere.

- *Intrusive events*. Some intrusive events, such as volcanoes, can expel wastes from a repository. Site selection is chosen to minimize these concerns.

- *Human intrusion*. Disposal-at-depth limits large-scale human intrusion to large programs that would necessarily be cognizant of, or able to deduce, the presence and function of a repository. Human intrusions by well drilling cannot be limited.

There are two basic types of geological repositories: low-heat repositories and high-heat repositories. If the concentration of heat-generating radionuclides is low, the total heat generated by the waste is low and no special features are required to address decay heat. Geological repositories designed for hazardous chemical wastes and the WIPP are low-heat repositories.

Repositories that contain SNF or HLW must be designed to handle significant decay heat from these wastes. Decay heat can increase temperatures that in turn degrade waste forms and waste packages and modify the geology. To reduce the maximum repository temperatures and minimize these problems, SNF and HLW will be packaged into small containers that are distributed over a sufficient area underground to dissipate heat. For example, in the proposed United States Yucca Mountain repository, the decay heat will necessitate packaging the SNF in over 10,000 waste packages that will be distributed over 100 kilometers of tunnel. If there was no decay heat, the same wastes could be placed in 10 silos with a few kilometers of access tunnels.

Decay heat greatly increases the cost of a repository. Equally important, the increased temperature creates uncertainties that (1) necessitate costly designs to assure the repository will perform as expected and (2) complicate licensing of such facilities. The effects of decay heat have strongly impacted repository programs worldwide and encouraged engineered storage of SNF and HLW prior to disposal. As shown in Figure 4.6.1-1, there are two sets of radionuclides that provide most of the decay heat: the fission products ^{90}Sr and ^{137}Cs and the actinides (Pu, Am and Cm). The economic and other

impacts of decay heat on repositories have led to investigations on alternative methods to manage decay heat from SNF.

- *Engineered Storage (see above)*. The decay heat can be reduced by an order of magnitude by interim storage for decades to centuries with multiple repository benefits.

- *Separations*. Two of the four fuel cycle schemes (full processing and partitioning and transmutation, or P/T) create options for alternative repository designs. In both cases, the intermediate-life heat-generating radionuclides ^{90}Sr and ^{137}Cs can be removed and separately stored or disposed of to minimize the heat impacts on the repository. Because of their limited half-lives (~30 y), there are many storage and disposal options. If a P/T fuel cycle is chosen (see below), most of the long-life heat-generating radionuclides (Pu, Am, and Cm) are destroyed. P/T fuel cycles with separate management of ^{90}Sr and ^{137}Cs result in very small repositories (small decay heat loads) with potential significant performance advantages from avoidance of decay heat in the repository.

Some characteristics of selected waste disposal programs are described below for illustration.

- *Sweden.* The Swedish waste program is operating a SNF storage facility to store SNF for several decades until the decay heat is reduced by time. This is required to provide high confidence in the performance of the repository. The Swedish program plans to use a copper waste package to help assure long-term isolation of SNF. Copper was chosen because native (metallic) copper has existed in stable form in Swedish granite for over a billion years. The SNF storage facility is to allow sufficient decay so the temperature of the repository is lower than the maximum temperature at which copper has been shown to have a very long lifetime under Swedish geological conditions.

- *United States*. The Yucca Mountain Project is currently investigating an alternative ventilated repository design where the repository would remain open for 50 to 300 years. The repository first operates as a storage facility and later is operated as a disposal facility. By ventilating the repository, the maximum rock and exterior waste package temperatures can be lowered to ~85°C. This cooling period substantially reduces the uncertainties in (1) the corrosion behavior of the waste package and (2) the long-term behavior of the geology.

- *France*. The French government has launched a large R&D program (Section 4.6.2) on long-term storage of SNF and HLW to (1) reduce repository uncertainties and (2) provide additional waste management options. HLW glass will be stored for several decades to allow reductions in decay heat rates from the waste.

The last important parameter in terms of repository performance is the waste form. The choice of fuel cycle has a major impact on the characteristics of the waste form.

- *SNF Disposal*. If a once-through fuel cycle is chosen, the fuel must meet certain minimum standards for disposal in a repository. Not all SNFs meet these requirements. For example, the Magnox SNF in Great Britain (uranium metal in magnesium-alloy cladding) is stable and suitable for reactor operations in gaseous carbon dioxide but does not have long-term stability in other environments. Therefore, the fuel is processed to (1) recover plutonium and uranium and (2) produce an acceptable waste form for storage and disposal. On the other hand, zirconium alloy-clad uranium dioxide, as is discharged from LWRs, and TRISO-coated particle fuel are considered to be suitable forms for geologic disposal.

- *HLW Disposal*. If the fuel is processed, the option exists to produce a superior waste form. The performance of such waste forms can be significantly better than SNF by proper choice of (1) physical form and (2) chemical form. The differing design goals for a nuclear fuel and a nuclear waste form imply that waste forms can be designed that are much superior to SNF in terms of repository performance. For example, nuclear fuel is designed with large surface areas to allow efficient transfer of heat from the fuel to the reactor coolant. The same high surface areas can accelerate the leaching of radionuclides from the SNF in the repository. Therefore, waste forms are can be designed to provide low surface area.

The geological disposal of SNF can be reversed, thus, a decision to adopt a once-through fuel cycle can be reversed by future generations with recovery and processing of the SNF. Most nations impose retrievability as a repository requirement when disposing of SNF or HLW, to allow access to the materials in the event that fuel cycle decisions are changed from geological disposal of certain wastes. All current repository designs use long-lived waste packages that will allow relatively easy recovery for periods of time that range from centuries to many tens-of-thousands of years. Design criteria for safe disposal of SNF, therefore, allows future societies to change their choices of fuel cycles.

Retrievability is a complex technical and policy issue. For some types of disposal systems, the impact is relatively small. For some other concepts, there are clear tradeoffs between retrievability and long-term performance.

4.6.1.2 Dilution Strategies

The earth contains very-large quantities of heavy metals and radioactive (uranium, thorium, etc.) materials. The impacts on man are limited because the concentrations of these hazardous components are low. Uranium, found in concentrations averaging 2.7 parts per million in crustal rock and soil (~4 gm/m^3), produces radon that provides approximately half of the average background dose of 300 mrem per year. Dilution can reduce the hazards of the waste to levels substantially lower than hazards posed by background radiation. The technical and policy issue is to determine under what conditions dilution is an appropriate strategy. There are three important dilution issues.

Deminimus

When nuclear facilities are decommissioned, a definition of clean must be developed. The definition can strongly impact the volumes of waste. Typical deminimus limits radiation exposures to 15 to 25 mrem per year. In comparison, the natural radiation background is a few-hundred millirem per year. As the deminimus level becomes a smaller fraction of the background radiation levels from natural occurring radioactive materials, it becomes harder to measure whether the goals have been met.

The choice of deminimus is a technical, political, and economic issue and not strictly a nuclear energy issue. For example, many other human activities (phosphate mining for the production of fertilizer, oil extraction and refinery operations) inadvertently concentrate natural radioactive materials and can create radiation hazards. From the perspective of public health and safety, the source of radiation dose is unimportant. All such sources should be controlled to the same standards of public protection.

Shallow land disposal

Some countries use shallow land disposal for radioactive wastes containing radionuclides with limited lifetimes (as described in the previous section on engineered disposal as a means of containment). The basic requirement is that the radionuclides decay to low levels before the engineered disposal site could potentially fail. The integrity of such sites cannot be assured for hundreds to thousands of years.

Shallow land disposal requires a definition of the allowable concentration of long-lived radionuclides in the waste. The importance of this definition can be seen by example. The Nuclear Energy Agency recently completed a detailed study of the health impacts of all fuel cycle operations. 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. A few curies of naturally-occurring long-lived radionuclides in shallow land disposal can have more impact than millions of curies in a geological repository.

In the United States, the generally accepted maximum concentration limit in a package for long-lived radionuclides in shallow land burial facilities is ~100 nCi/g. In France, the limit is 0.1 Ci/m^3. These are roughly equivalent limits. At these limits, the long-term hazard from a shallow land disposal site would be similar to many uranium ore deposits. As a second basis of comparison, depleted uranium has an alpha activity of several-hundred nCi/g. These are maximum limits for the waste packages with the highest concentrations of long-lived radionuclides. It would be expected that average concentrations in a shallow-land burial grounds would be several orders of magnitude below the maximum regulatory limit.

In some countries (England, Germany, Sweden, Switzerland, Finland), conventional shallow land disposal has been replaced by intermediate-depth or geological disposal. In

some cases, there were economic incentives. In other cases, national decisions have been made to avoid any disposal of hazardous wastes in or near the biosphere. Many of these national decisions address multiple hazardous wastes including chemical wastes, heavy metal (lead, arsenic, cadmium, etc.) wastes, and radioactive wastes. This is an area where different advanced countries have fundamentally different policies and no international consensus exists.

Dilution for Disposal

There are certain chemical and radiological wastes where dilution may be the preferred disposal option. The classical chemical example is salt. If salt is disposed of on land, it will eventually leach from its disposal site and may convert fresh groundwater into undrinkable salt water. This is a specific waste with a very specific chemical characteristic where ocean disposal is clearly the preferred disposal option.

The radiological equivalent may be ^{131}I. This is a very long-lived radionuclide ($T_{1/2}$ = $1.7x10^7$ y). The primary radiological risk is that iodine is concentrated in the thyroid. This is only an issue in the center of continents where iodine is not found in the soil in significant quantities because it has been washed to the sea. Many individuals in the center of continents, therefore, have iodine deficiency; thus, the body concentrates the iodine. In contrast, the oceans contain very large quantities of iodine, and individuals living near the ocean have excess iodine. For continental repositories, it is important that iodine leakage be minimized because of bio-concentration. Near infinite dilution may provide higher safety margins compared to other methods of disposal because of unique biochemistry and geochemistry of this element.

4.6.1.3 Destruction Strategies: Partitioning and Transmutation

What does not exist, cannot cause harm. Destruction of wastes provides an ultimate solution. However, there are limits to what can be accomplished in this regard. The technology does not exist to destroy some radionuclides having small nuclear cross sections. Chemical separations have finite efficiency and utlimately discharge some material to waste streams. There are significant costs. Last, there are risks associated with destroying particular wastes. Waste destruction does not necessarily have the lowest impacts on human health and the environment.

Current U.S. plans will use destruction as a waste management tool only for the eliminating excess weapons-usable plutonium. Recycle of reactor plutonium is primarily for resource utilization; but also has some waste management benefits. The choice of radionuclides to destroy depends upon the goals, as addressed below.

- *Reduced toxicity from nuclides most likely to impact human health under expected repository conditions*. Reducing the inventory of radionuclides reduces the potential future releases of radionuclides from the repository. However, not all radionuclides are equivalent. Shorter-lived radionuclides will decay before they can be transported by groundwater to the open environment. Many radionuclides are not easily

transported by groundwater; thus, their destruction would be expected to have little impact on repository performance. Last, of the nuclides that could potentially escape the repository, only a few would tend to concentrate in the body and thus impact public health. These considerations suggest that only a few radionuclides (^{99}Tc, ^{129}I, and ^{237}Np) are potential candidates for destruction based on the goal of destroying the radionuclides most likely escape a repository and cause a radiation dose to the public.

- *Reduced toxicity from intrusive events*. The risks from an intrusive event that disrupt a repository are considered to be low compared to the risks from movement of radionuclides via groundwater. However, the potential consequences of an event could be larger. If the goal is to reduce the consequences of low-risk, higher consequence events, a somewhat different set of radionuclides than discussed above must be removed. In most of these scenarios, one or more actinides control radiation exposure.

- *Reduced fissile inventory*. Reducing the fissile inventory of the repository reduces (1) required safeguards and the potential for recovery of weapons-usable materials and (2) the potential of human intrusion to recover valuable materials and in the process degrade repository behavior.

- *Reduced heat load.* Repository design is controlled by decay heat. Over half of the heat that governs repository loading comes from the decay of the fission products cesium-137 and its short-lived daughter barium-137 (with 30-yr half life, 0.0602 atoms/fission yield, and 0.80 MeV/decay decay energy) and strontium-90 and its short-lived daughter yttrium-90 (with 28.1-yr half life, 0.0443 atoms/fission yield, and 1.13 MeV/decay decay energy). For typical current nuclear power plants with thermal efficiencies of around 0.34, and spent fuel aged 50 years, the fission-product decay heat at 50 years, per gigawatt-year of reactor operation, is 10,400 W/GW-yr, Fission product decay heat will be generated by both spent fuel and recycling waste. For spent fuel, additional decay heat comes from the actinides and their shorter-lived precursors, predominately from Am-241 (4500 W/GW yr at 50 years), Pu-238 (2,400 W/GW-yr), Pu-240 (51 W/GW-yr); Cm-244 (370 W/GW-yr), and Pu-239 (350 W/GW-yr). For spent fuel, the total decay heat is then 80 percent greater than that for recycling waste, or 18,600 W/GW-yr. But the actinide decay heat persists longer, due to the longer half-lives of the primary contributors. Removal of heat-generating radionuclides may significantly improve repository performance and lower repository costs. The cross sections of ^{90}Sr and ^{137}Cs, combined with their relatively short half-lives, render the practicality of destroying these radionuclides. However their half-lives of 30 years are relatively short, which provides the opportunity for management strategies that allow the decay heat from these shorter lived radionuclides to drop substantially. Such strategies are not practical for the primary heat-generating actinides; however their cross sections are high enough to motivate their destruction.

The considerations above usually suggest that all of the actinides and a limited number of long-lived fission products are the primary candidates for waste destruction - if practical methods of waste destruction are developed.

Partitioning and transmutation is technically challenging. The higher actinides (americium, curium, etc.) have short half-lives compared to plutonium. Many radionuclides that are P/T candidates have many orders of magnitude higher radioactivity for corresponding heat loads than the major fissile fuels (^{235}U and ^{239}Pu) found in traditional light-water reactor and fast-reactor systems. Only several very-specialized facilities in the world have processed or irradiated these types of materials. Handling these radionuclides create unique fuel cycle issues. If a decision to implement P/T was undertaken, it may have major impacts on the choice of power reactors and fuel cycles. Some options include:

- *Thorium fuels*. If long-lived actinides are not to be sent to the repository, one strategy is to minimize creation of those radionuclides. The quantities of actinides produced in a thorium-^{233}U fuel cycle are between one and three orders of magnitude less than in uranium-plutonium fuel cycles. A thorium breeder fuel cycle produces ^{233}U. In any system, most of the ^{233}U fissions. The ^{233}U that does not fission ultimately captures two neutrons and becomes ^{235}U. It takes many neutrons to produce still higher actinides. If the amount of transuranics to be sent to repositories is to be minimized, avoiding their production becomes an important strategic option.

- *Special reactors*. In terms of technology and engineering, one of the major unresolved P/T questions is whether traditional reactor concepts (water-cooled reactors and metal-cooled fast reactors) can be practically used to transmute higher actinides. The decay heat loads of the higher actinides may require dilution of the higher actinides in fuel to meet reactor safety limits but this has major economic impacts and increases secondary loses of actinides to waste. The very high alpha activity compared to traditional fuels may have significant implications on reactor choice.

While some groups believe that existing reactor lines are suitable for P/T, other groups have concluded that P/T requires special reactors. For example, a recent study by the Kurchatov Institute suggested that an integrated system would contain light water reactors (power), fast reactors (power and fuel production), and molten salt reactors (higher actinide burning). In this example, a specialized molten salt reactor, producing about 8% of the total nuclear energy system power, is used for burning actinides. A specialized molten salt reactor was chosen for transmutation due to (1) very-low fissile (higher actinide) inventories (accident source term) per unit power compared to other reactors, (2) online removal of fission products with no need for fuel fabrication, and (3) potential for lower transuranic loses to waste streams. Groups in several other countries, including France and the United States, are examining such special reactors for P/T.

4.6.1.4 Combination Strategies

In any future waste management system, containment and dilution will be components of the waste management system. In many scenarios, destruction of selected radionuclides will also be a component of the overall strategy. Waste destruction or dilution may

reduce the inventory of radionuclides to the repository, reduce the potential for repository failure, and reduce costs. However, no credible case has been made that such technologies are capable of totally replacing repositories. Waste destruction and dilution should thus be considered as potential methods to enhance the performance of waste management systems and greatly increase the capacity of geologic repositories.

4.6.2 Waste Technology Status and R&D Needs for Gen IV

Waste management is a major issue for nuclear power. For Generation IV systems, there are two strategic questions:

- What are the waste management implications for nuclear power of the four types of fuel cycles?

- Do the waste management choices, beyond strict economics, have a major impact on the future of nuclear power?

Developments in the last decade have begun to reveal the implications of the different fuel cycles on waste management and the different options that become available with different fuel cycles. Some of the implications are only poorly understood. Contrasting scenarios can clarify the differences.

- *Once through LWR fuel cycle*. This fuel cycle generates volumes of mill tailings and requires large repositories for the SNF. The radiotoxicity and weapons-usable fissile inventories in the repository are large and grow with time. The operational risks (accidents, environmental releases, potential for loss of fissile materials) from the fuel cycle are minimized because operations are minimized. The technological and institutional requirements to support the fuel cycle are minimized. This fuel cycle is reasonably well understood.

- *Full Partitioning and Transmutation fuel cycle*. If the fuel cycle generates as much or more fuel than it consumes, the volumes of mill tailings are minimized. The repository size is very limited (limited inventory of heat-generating radionuclides in the repository from destruction of the long-lived heat-generating actinides) and no significant weapons-usable fissile repository inventory exists in terms of safeguards. Large-scale processing of SNF is required. Long-term risks are minimized, but near-term risks are higher due to the nature of the attendant processing operations. Advanced technologies and supporting institutions are required. The full implications are not well understood.

The cross cutting R and D identified herein is to create waste management options and an understanding of those options to assure the requirements of a Generation IV initiative can be met. There is no necessarily preferred path forward. The state of the technology and societal goals imply that different waste management systems may be preferred for different societies and that the preferred options may change with time.

4.6.2.1 R&D on Containment Strategies

4.6.2.1.1 Repository Implications for the Four Major Fuel Cycle Types

Recent work indicates that there may be significant technical, performance, size, economic, and public acceptance differences in the repository depending upon which of the four fuel cycles is adopted for Generation IV. This is in contrast to the historical perspective that the reactor choice and fuel cycle had relatively little impact on waste management and the repository. While the differences have begun to be recognized, the implications are not well understood and indicate a need for better understanding to aid decisions on which futures to pursue.

4.6.2.1.2 The Role of Interim Storage

In France, nuclear glasses and non-processed spent fuels are currently stored in industrial storage facilities, like pools for the spent fuel and dry storage for the glass packages, which are usually designed and authorised for about 50 years. Although studies on deep geological disposal are ongoing, no final decision has yet been taken regarding the ultimate disposition of the waste streams. France is also looking at whether and under which conditions a long-term interim storage may provide the fuel cycle back end with an effective gain in flexibility. The long-term interim storage can become a useful nuclear installation if it is able to:

- fill the gap between the industrial, decade-long storage options currently available and final disposal,
- provide the long-lasting waiting period with safe conditions for stored packages, and
- guarantee their protection and retrieval at conditions affordable to future generations.

Owing to the previously quoted uncertainties, the operational lifetime of long-term interim storage is envisaged to be typically of secular length, up to 300 years, making the design of such facilities a challenge.

The durability analyses have led French planners to consider four general principles underlying the approach:

- Different life phases can be defined for long-term interim storage: active phases, where a substantial activity occurs (loading, unloading, handling of packages…); passive phases, or waiting periods, where the activity is limited to maintenance, and no package handling is made. Those two operating modes are fundamentally different, and the specificity of the long-term interim storage lies in the waiting periods.

- The burden to future generations has to be minimised with respect to operating and maintenance costs. Thus, special attention has to be awarded to component reliability, the aging mechanisms have to be understood and addressed, the specifications must rely on maximum robustness and passiveness and not on

preferred social or environmental conditions, which are unpredictable on this time scale. In any case, an evaluation of this burden has to be performed and communicated to the decision makers.

- The back handling of the packages must be safely possible at any time, technically and economically. Thus, the design must be such that the society can keep track of all the relevant information on the waste and the packages, maintain competency and expertise in the relevant technical fields. The safety methodology used must also be known in subsequent years to facilitate decisions to be taken later on.

- The safety analysis is the same as the current one, based upon a defense-in-depth approach, but must be completed with specific dispositions related to the long term. Among those are a new evaluation of the risks usually considered for the analysis (although it is not possible to anticipate quantitatively the regulatory evolutions), and identification of relevant off-normal scenarios and evaluation of their impact, in each life phase of the facility. Regardless of the provisions made at the design stage to ensure acceptable safety conditions that will last for the facility lifetime, a periodic safety review has to be considered and the process of safety re-evaluation defined.

These general principles have to be translated into technical specifications

4.6.2.1.3 Management of Cesium and Strontium if SNF is Processed

Because ^{90}Sr and ^{137}Cs contribute the largest portion of decay heat of SNF and HLW, separate management of those isotopes may offer an opportunity to reduce repository costs. The cesium and strontium can be separated from HLW, stored until they decay to low levels, or separately disposed of. For unsaturated repositories active ventilation systems can potentially be used to manage these heat loads, while for saturated repositories separation and separate management are likely required. The costs and benefits of these alternative approaches to waste management that reduce repository impacts from decay heat should be determined.

4.6.2.2 R&D on Dilution

4.6.2.2.1 The Allowable Releases to the Environment

Nuclear fuel cycles that involve processing SNF release radioactive gases such as ^{85}Kr. These gases have traditionally been released to the atmosphere. With the current size of the nuclear enterprise, this is fully acceptable. However, with a very large enterprise, the concentrations of radioactive gases would build up in the atmosphere. Releases to the atmosphere could be limited by collecting and storing these gases until they decay to harmless levels.

The allowable releases to the atmospheres must be determined. This issue is the nuclear power equivalent to the release of carbon dioxide from the combustion of fossil fuels. In each case, the issue is trivial for a small enterprise but a significant issue on a sufficiently

large scale. This is both a technical and institutional issue. Ultimately, some type of international agreement will be required.

4.6.2.3 R&D on Destruction Strategies

4.6.2.3.1 The Need for Waste Partitioning and Transmutation

One of the four fuel cycle categories (partitioning and transmutation, or P/T) includes the processing of SNF, separation of selected radionuclides and their destruction. The motivation for implementing this category of fuel cycles must be assessed. There have been multiple P/T studies since the 1970s. Historically, these studies have focused on reduction of toxicity of specific radionuclides. There are other impacts of P/T on the repository, some of which may be more important in terms of economics and repository performance than the reduction in toxicity. The other implications of P/T on waste management must be understood to evaluate this category of fuel cycles. Such analysis may provide guidance on the direction forward for Gen. IV. Two examples can clarify the issues.

- *Heat load*. The experiences in repository design in multiple countries in multiple geologies indicate that heat management in a repository has major impacts on costs and potentially major impacts on repository performance. The longer-lived heat generating radionuclides in SNF are the actinides. It remains to be determined whether actinide destruction to minimize long-term heat loads in the repository should be a major goal of P/T.

- *Safeguards.* With the end of the cold war, major efforts have been initiated to reduce inventories of weapons-usable fissile materials. It remains to be determined whether this should be a goal for P/T. The appropriate criteria for safeguards termination must be determined. The question of whether the requirement for destruction of weapons-usable materials to levels is sufficient for qualification for safeguards termination shold be addressed.

4.6.2.3.2 The Appropriate Waste Partitioning and Transmutation System

Historically, most P/T studies have assumed that traditional reactor concepts would be used to destroy selected radionuclides - including the higher actinides. The alpha activity and heat loads of many candidate P/T radionuclides are orders of magnitude higher than the major fissile fuels (uranium and plutonium) found in traditional light water reactor systems and fast reactor systems. The characteristics of these materials raises the question of what reactors are suitable for these materials and whether mainline reactor concepts can be practically be used for such missions. If P/T is to be investigated, a central question is whether traditional reactors are suitable for the task or whether special reactors, accelerators, or other devices are required.

4.6.2.4 R&D on Combination Systems

4.6.2.5 Uranium and Thorium Mill Tailings

Whether the controlling radionuclides can be removed from mill tailings to create a non-radioactive waste stream and a concentrated radioactive waste stream for geological disposal must be assessed.

Mill tailings are the largest volume radioactive stream in the fuel cycle. Multiple studies show that the largest radiological impact of the once-through nuclear fuel cycle to the public is from mill tailings. The radiological impact is from the natural longer-lived decay products of uranium and thorium. There is a potential alternative disposal strategy: replace engineered (shallow-land) disposal of these tailings by separation of the tailings into (1) a concentrated stream containing the long-lived radionuclides and disposing of these radionuclides by geological isolation and (2) a very-low activity stream that meets the deminimus requirements for disposal as a non-radioactive waste. There are non-nuclear waste streams with similar characteristics. This alternative strategy separates the tailings into two streams to be managed separately. This strategy should be assessed to determine if it is a viable method to reduce the impact of this waste stream.

In this context, there is an important underlying technical consideration. The potential radiation exposure is primarily from radionuclides that have been solubilized during the milling process. To prevent radionuclide releases, some mills currently treat tailings water to precipitate these radionuclides and return them to the mill tailings disposal sites. This is a very small volume compared to the very large volumes of mill tailings. The regulatory, technical, and economic impacts of alternative methods of tailings management should be evaluated.

4.6.2.6 Low Level Waste

4.6.2.7 Interim Storage

4.6.2.8 Geologic Repositories

4.6.2.9 Waste Partitioning & Transmutation

4.7 Thorium Fuel Technologies [1,2,3]

Because ^{235}U is the only naturally occuring, thermally fissile isotope, most of the world's nuclear reactors use this isotope to sustain the neutron chain reaction (the utilizes either natural uranium or, in most cases, uranium enriched in ^{235}U). In this fuel cycle, neutron captures by the fertile material ^{238}U produce plutonium, and in particular its two thermally fissile isotopes ^{239}Pu and ^{241}Pu, which are burnt partly in-situ (the remaining part contained in the discharged fuel may be recycled into reactors). Thorium is a naturally occurring, fertile isotope (i.e. not thermally fissile). It can also breed a fissile isotope, ^{233}U. In a thermal spectrum, this isotope yields the highest average number of fission neutrons produced per thermal neutron absorption (2.27 for ^{233}U in a typical PWR, compared to 2.06 for ^{235}U and 1.84 for ^{239}Pu.).

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 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 uranium cycle.

Historical Context

During the pioneering years of nuclear energy, 1950-1970, a large number of potential avenues for energy production with thorium were investigated with great enthusiasm and with little regard for the costs, not only in the USA and USSR, but also in Europe and, to some extent, in Asia.

Thorium, as a source of energy that could supplement the limited uranium reserves, was widely investigated in those years, including studies in countries like Australia, Italy, Norway, and the Netherlands where nuclear energy has been virtually put to a rest today. For example, it is remarkable that the thorium-based Elk River (1963) and Peach Bottom (1967) reactors were started only a few years after the "founding fathers" of the two main reactor families of today, based on uranium fuel, PWR Shippingport (1957) and BWR Dresden (1960).

Since that time, quite significant experience on thorium-based fuel in experimental or power reactors has been gained. In particular, experiments have been conducted in:

- The AVR (Germany)
  127,000 hours of operation at about 95 per cent rated power with Th-based fuel at 15 MW(e), was achieved. Approximately 300 thousand fuel balls containing a total of 1360 kg of Th were used. Maximum burn ups of 150,000 MWd/t were achieved.

- The Dragon reactor (United Kingdom)
  Thorium fuel elements with a 10:1 Th:U (HEU) ratio were irradiated in this 20-MW(th) reactor (950 °C He-outlet temperature) for 741 full power days.

- KUCA facility (Japan)
  Many critical experiments have been conducted here with metallic Th and enriched U.

- The Peach Bottom HTGR (USA)
  The Peach Bottom reactor was a high-temperature, graphite-moderated, helium-cooled reactor experiment operating at 110 MW(th) with HEU/Th coated-particle fuel (with High Enriched Uranium enrichment of 93 %)

- CIRUS reactor (India)
  Irradiation of Th-based test fuel rods and Th/Pu-based test fuel rods (with up to 18,000 MW d/t burnup) has been conducted.

- MSRE (Molten Salt Reactor Experiment) in Oak Ridge (USA)
  This small prototype was operated for some years until 1976 and used the thorium/U233 cycle, with a continuous removal of the gaseous fission products.

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 (the latter in the case of the AVR). In addition, in most of these countries, R & D was directed at recycling spent Th-based fuel on both laboratory scale and pilot scale and at laboratory re-fabrication tests. For example, a prototype recycling plant and remote fuel fabrication plant (PCUT) operated briefly in Italy.

Several power reactors also used thorium, and some in India still use thorium fuel today. They are:

- THTR (Germany)
  A 300 MW(e) reactor was operated with 675,000 pebbles containing Th/HEU fuel, but was shut down after 3 years for non-technical reasons. Fuel fabrication was performed on an industrial scale.

- Fort St Vrain (USA)
  The Fort St Vrain (FSV) reactor was the most recent thorium-fueled (Th/HEU) nuclear power plant operating in the United States. FSV was a high-temperature, graphite-moderated, helium-cooled reactor with thorium and high-enriched uranium (HEU) fuel designed to operate at 842 MW(th). The fuel was contained in microspheres of thorium carbide and Th/^{235}U carbide (4Th:1U) coated with silicon oxide and pyrolytic carbon to retain fission products. Almost 25 metric tons of thorium fuel was manufactured for the FSV reactor.

- Shippingport/PWRs (USA)
  Thorium-based fuel for PWRs was investigated using both ^{235}U and plutonium as the initial fissile material, and it was concluded that use of Th-based fuel would not significantly affect operating strategies or core margins in PWRs. In this regard, a

nuclear energy centre (NEC) was proposed to utilize recycled plutonium. A thorium-based, plutonium-consuming, LWR fuel cycle was analyzed for the NEC to burn the plutonium and convert ^{232}Th to ^{233}U. The ^{233}U would then be denatured as fissile feed for reactors outside the NEC. The light water breeder reactor (LWBR) concept was also tested in the Shippingport reactor using thorium and ^{233}U fuel in Zircaloy-4 cladding. The core was operated from 1977 to 1982 and successfully demonstrated breeding of ^{233}U using the "seed/blanket" concept, allowing a "self-sustained" Th/^{233}U cycle.

- BWR, Lingen (Germany)
  This 60 MW(e) reactor utilized Th/Pu-based fuel test elements.

- ANGRA 1 (Brazil)
  Tests of Th/U and Th/Pu-based fuel test elements in the 900 MW(e) reactor were prepared (including licensing documentation) within a program of German-Brazilian cooperation, but were not carried out.

- KAPS units 1 and 2, KAIGA units 1 and 2, and RAPS 3 and 4 (India)
  Each KAPS unit is loaded with 500 kg Th-fuel. In June 1995, Unit 1 had achieved about 300 days of full power operation, Unit 2 about 100. The power plants for KAIGA and RAPS are under construction, but the use of Th-based fuel is planned for power flattening. Work on post-irradiation examinations, recycling (laboratory scale and pilot-scale) and refabrication (laboratory scale) continues and has been enlarged based on these experiments.

In conclusion, the feasibility of different types reactors based on U233/Th has been successfully demonstrated and extensive experience has been accumulated so far, theoretical as well as practical and engineering-wise.

Present Status Of Thorium Cycles

Thorium continues to generate interest in specialised circles, but more at an academic level for long-term prospects, especially in Japan where the HTTR could well be used with thorium in the future (as well as the HTR-10 in China). Only in India is thorium considered an industrial fuel for a not-too-distant future, because India has large thorium deposits compared to its limited uranium resources.

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 (see reference 2, pp. 3-5).

This subsection presents the findings of these various studies and viewpoints on thorium uses, along with a balanced view of the advantages and weaknesses of thorium as a fertile material for nuclear energy.

Early interest in thorium was motivated primarily by concerns about the limitations of uranium resources, within the framework of a large utilization of nuclear energy, which was anticipated at that time. Since that period, this picture has changed, and uranium resources are no longer considered as a major concern in the near term (except in India, which has large domestic resources of Thorium). New issues linked with nuclear energy development have now emerged, including proliferation resistance, accumulation of separated civilian plutonium, and radioactive waste management, for which the thorium fuel cycle may have promising features. Use of a thorium fuel cycle is reviewed with respect to each of these issues.

Extension of natural resources for nuclear energy

Thorium abundance and reserves

Thorium is essentially 100 % ^{232}Th, and is reputedly more abundant than uranium, although its limited uses so far had not led to very extensive prospecting (typically, thorium has been obtained as a by-product of uranium or rare earth mining). The world's reasonably assured reserves (RAR) are known to be at least as large as those of uranium, and quite probably higher. As a matter of fact, thorium is three times more abundant in the earth's crust than uranium, reflecting the differences in the half-lives of ^{232}Th (1.4×10^{10} years) and ^{238}U (4.5×10^{9} years).

Extensive thorium deposits exist in Brazil, Canada, India, and the USA. Most of India's thorium reserves (nearly 300,000 tons) are in the beach sands of Kerala and Orissa. The ore is very high grade, with a thorium content of about 10 percent, and because it is in beach sands the mining is inexpensive. Similar extensive deposits are found in Brazil and Canada. The USA is believed to have thorium in major deposits in California, Colorado, Idaho, Montana, New Mexico, and Wyoming; but there is no current interest in exploration or mining due to lack of demand.

It must be emphasized that the preparation of thorium, similar to that of the rare earths, entails its separation from many other (valuable) compounds; hence the chemical separation is not straightforward and necessitates many manipulations and chemical steps (with the possibility of corrosion by highly aggressive chemicals). The high melting points of thorium (1750 °C) and thorium oxide (3300 °C) give them remarkably resilient properties (stability, refractory and radiation damage resistance) which are balanced by added difficulties for preparation compared with uranium.

Thorium energetic potential

As said before, fission energy is not directly produced by thorium but primarily by the bred ^{233}U (except for a small fraction of direct fissions by fast neutrons). In a similar way, the 99.3 % of ^{238}U contained in natural uranium may be used primarily through plutonium produced by neutron capture in ^{238}U.

Using breeder-type reactors, the known RAR uranium reserves can provide an energy source roughly equivalent to those of coal RAR reserves. Thus it can be assured that the RAR thorium reserves can also double the potential available from uranium.

However, if thorium is used in classical thermal reactors, conversion ratios (the amount of fissile isotopes produced in the reactor divided by the amount of fissile isotopes consumed in the reactor) is generally less than one. In the CANDU-type heavy-water reactor, a self-sufficient equilibrium thorium cycle (SSET) appears possible (conversion ratio equal or slightly greater than one). Such breeding would be also possible in molten salt reactors and higher than for U-Pu cycles due to the better fission properties of ^{233}U. For example, with thorium the conversion ratio is 0.7 in a LWR, instead of 0.6 for U-Pu fuels, and may reach easily 0.8 or even 0.9 in some other types of reactors such as HWRs or HTRs.

Many studies have investigated the use of thorium in thermal reactors, because many combinations of fuel cycles are possible with a mix of various types of reactors, operating in the frame of symbiotic systems. To summarize simply, the results of these studies show that thorium use in non-breeder thermal reactors allows a global uranium savings from a few tens of percent to a maximum of roughly 80 %, when equilibrium of the reactor fleet is reached.

In summary, if thorium were to be used in non-breeder thermal reactors, the world's fissile resources would be increased by a factor of about 2 or 3. If breeder reactors were to be intensively used, the energetic potential of natural resources of uranium would be multiplied by a factor of 50 by the breeding process, and in that case the thorium breeding would multiply this already huge energetic potential by an additional factor of 2 to 4.

Proliferation resistance

The International Fuel Cycle Evaluation (INFCE) study (1978-1980) summarized thorium fuel activities world-wide and considered, in particular, issues related to the technical barriers to proliferation. It was shown that the technical characteristics that would inhibit proliferation for thorium cycles with up to 20 % of fissile material were similar to that of U-Pu cycles.

Indeed, about 5 kg of ^{233}U can be used as a weapon material (similar to plutonium). As with ^{235}U, a coarse bomb with ^{233}U is simpler to fabricate than with plutonium, because there are few spontaneous neutrons emitted. (The neutrons by the higher isotopes plutonium isotopes, ^{240}Pu and ^{242}Pu, which are always present in some quantities with

^{239}Pu, require that weapons made using plutonium use the more complex implosion approach).

Nevertheless, a specific technical hurdle to diversion of mateirals to weapons use does exist in the case of ^{233}U. This is due to the small quantities of ^{232}U that are always mixed with ^{233}U, coming from (n, 2n) nuclear reactions that occur in the reactor where ^{233}U was produced. (It must be noted, however that ^{232}U content is much less (10^{-6} to 10^{-7} only) if ^{233}U is produced in fast neutron reactors - see reference 2, p. 18). This ^{232}U has strong gamma emitters in its decay chain, especially ^{208}Tl (2.6 MeV), which creates a substantial problem for handling purified ^{233}U during fuel fabrication. Conversely, it also provides a radioactive "tag" attached to the ^{233}U that can help prevent proliferation.

Another deterrent to the use of ^{233}U for making nuclear weapons may be obtained by a dilution of ^{233}U with ^{238}U in the form of depleted uranium. (This isotopic dilution approach is not possible with ^{239}Pu because, unlike ^{238}U, all of the plutonium isotopes have sufficiently small bare-sphere critical masses to potentially permit use in nuclear explosives.)

To conclude this discussion, it must be emphasized that, to our knowledge, no nuclear weapon has ever been fabricated with U233.

Plutonium consumption

If thorium is used with Medium Enriched Uranium (< 20 %), called a MEU cycle, the reactor will still produce plutonium but in lesser amounts than the standard U-Pu cycle. If it is used with High Enriched Uranium (~93 %), called a HEU cycle, the production of plutonium will be very low. If only pure ^{233}U is used as a fissile material in combination with thorium, essentially no plutonium is produced. Thus, we can say that an extensive use of the thorium cycle would provide a significant reduction in the rate of plutonium accumulation.

Another potentially interesting use of thorium comes from the ability to burn existing stock piles of plutonium and in particular what is called "military" plutonium (that is weapons-grade plutonium now being declared excess-to-military-needs in the United States and Russia). This possibility is particularly attractive in high-temperature reactors where plutonium can be burned in a very efficient manner. This solution is now under study in Russia, in cooperation with other countries.

Waste management

A comprehensive study on thorium as a "waste management option" has been recently published in the framework of an EC contract [3]. The main finding of this study is that the waste radiotoxicity associated with Th-U fuels is much less than that of U-Pu fuels up to 10,000 years after disposal. After that, radiotoxicity due to ^{233}U daughter products is greater than that of ^{239}Pu. However, for the case of major actinides recycling only, the long-term radiotoxicity of the Th-U cycle becomes less than that of the U-Pu cycle at any

time scale. In both cases, a full recycling of all heavy nuclides gives impressively low radiotoxicity results for the wastes over a long period of disposal, starting after the bulk of the fission products has decayed. Another result of this study is that the long-lived radiotoxicity of thorium mining waste is expected to be relatively small compared to uranium mining wastes.

It must be emphasized that the often quoted benefit of the thorium cycle compared to U-Pu cycle with regard to long term radiotoxicity is not fully recognised by all specialists. (See for example the communication of Global 1995 of Peter Wydler and Enzo Curti of PSI in Switzerland "Closing the fuel cycle: consequences for the long term risk"). Therefore, this issue must be clarified, but it is clear that the production of long-lived minor actinides will be noticeably lower for thorium cycles than for uranium cycles due to the lower position of thorium on the Mendeleiev's table. Furthermore, if uranium and protactinium are removed by recycling, the wastes from thorium cycle will have a long-term radiotoxicity considerably decreased compared to those of a U-Pu cycle.

Thorium fuel properties in reactor cores

The high fission efficiency of ^{233}U (i.e., the relatively high number of fission neutrons per fission) has the result that variations of fissile content and reactivity of thorium fuels over the fuel lifetimes are much smaller than in uranium cycles. Thus, over the lifetime of a core, variation in reactivity and power distribution (power peaking) are less than in uranium core, which makes thorium reactor cores more manageable. This property gives a significantly greater flexibility to programs based on thorium use from reactor operation standpoint. Other positive attributes of thorium in reactor cores include the following:

- Thorium-^{233}U cycles increase the conversion ratio in current thermal reactors compared to U-Pu cycles (the increase is usually between 20 to 30 %).

- The nuclear properties of ^{233}U allow a much more flexible use in thermal reactors than the use of plutonium (because 3 isotopes of plutonium have great resonances at a very low energy, which complicate the neutronics behaviour of plutonium fuels).

- Globally, one can say that Th-^{235}U is more flexible in many dimensions for use in various types of reactors than the U-Pu cycle.

- A significant weaker power and temperature dependence of the ^{233}U nuclear parameters is favourable for reactor safety and operation, especially when changing LWR's from "cold" to "hot" conditions.

Other specific features of thorium fuel are as follow :

- The yield of fission products affecting reactor poisoning during operation (such as Xe, Sm, etc.) is significantly lower for ^{233}U than for ^{235}U and plutonium. This results in decreasing average cross section values of neutron absorption by ^{233}U fission

products (by 25-30 %) and hence, lowering reactivity loss and increasing the lifetime of the core.

- Thorium and its oxide (ThO_2) have better behaviour under irradiation than uranium and its oxide (UO_2), allowing higher burnups.

- The higher melting point of metallic thorium compared to that of uranium (1750 °C instead of 1130 °C) allows higher margins for the design and for operation of reactors using metallic fuels. This is the same thing for reactors using oxide fuels since the ThO_2 melting point (3300 °C) is higher than the one of UO_2 (2800 °C).
- The thermal conductivity of thorium is better than that of uranium.

- Metallic thorium interaction with water and steam is less intense than metallic uranium.

Conversely, one of the main disadvantages of the use of the thorium fuel in reactor cores is the rather high concentration of ^{233}Pa. The ^{233}Pa content arises because ^{233}Pa has a relatively long decay period of 27 days, compared to that of ^{239}Np (the decay precursor that forms ^{239}Pu), which is only 2.3 days. This isotope is created by ^{232}Th (itself formed by neutron capture on ^{232}Th) and decays to form ^{233}U. This entails a "delayed reactivity" increase after reactor shut-down, which must be carefully investigated.

Another drawback revealed by German studies on thorium use in HTR's was that this fuel exhibited higher decay heat production than U-Pu fuels.

Thorium-^{233}U fuel fabrication and recycling

It is clear that full advantage of the thorium fuel cycle is derived by retrieving the fissile ^{233}U through recycling, and regenerating it, as much as possible, in a "self-sustained" type of reactor in which thorium will breed at least as much fissile ^{233}U material as will be consumed. This means that recycling and refabrication of ^{233}U fuels are necessary steps.

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. (See the discussion of recycling in Section 4.5.5). The dissolution of thorium metal and thorium oxide is not as straightforward as with uranium and requires some addition of hydrofluoric acid, which is corrosive with chemical equipment and necessitates appropriate buffering agents. The recycling of High Temperature Reactor carbide fuels adds extra difficulties due to the impervious coatings on the carbide or oxide fuel. Although the operations have been demonstrated at pilot plant stage, these are not straightforward and present a real challenge, but this is also true for uranium-only coated particle fuels.

In this context, one of the challenges of a closed thorium cycle is the "contamination" of ^{233}U by ^{232}U, which generates strong high-activity daughter products. This considerably complicates ^{233}U fuel fabrication, which must be performed remotely behind shielding,

requiring costly and sophisticated remote handling technologies and heavy gamma protection devices. Furthermore, the ^{228}Th and ^{234}Th in recycled thorium prevent direct handling for many years. In this regard however, all full transuranic fuel cycle are also faced with a necessity for remote fabrication.

In summary, recycling of U/Th is somewhat more complicated than with the traditional uranium cycle but should not pose too great problems to the recycleors today after the proper flow sheets have been selected. The head-end processing of pyrocarbon/silicon carbide-coated carbide or oxide fuels for the HTR is challenging to address with the THOREX process, however.

Conclusion

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 behaviour 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.

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, recycling 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 modify the previously envisioned technological hurdles to the implementation of thorium cycle.

Today, emerging concerns associated to nuclear energy development, have stimulated renewed interest of thorium based fuels. Among them, two primary reasons are to be cited:

- long term radiotoxicity of wastes
- plutonium stockpiles management

On both topics, thorium-based fuel features quite attractive possibilities which may justify further studies to realistcally 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 development of nuclear energy.

References for Section 4.7

[1] "A present review of the thorium nuclear fuel cycle," UE Report : EUR 17771 – 1997

[2] "Thorium based fuel options for generation of electricity : development in the 1990's," AIEA – TECDOC – 1155 – May 2000

[3] "Thorium as a waste management option," UE Report : EUR 19142 - 2000

4.8 Cross Cutting R&D

The FCCG used the four generic fuel cycle types (once-through, partial recycle, full Pu and full TRU recycle) as an organizational mechanism for initial identification of Generation 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 major links in the fuel cycle for reaching Gen-4 goals. Figure 4.7-1 illustrates the fuel sources and wastes for each of the four generic fuel cycles. "Cross cutting" R&D (i.e., R&D activities that are common to many or all technologies considered) opportunities are identified in this section for each fuel cycle type.

4.8.1.1 Cross Cutting R&D for Once-Through Fuel Cycle

For once-through fuel cycles, Table 4.7-1 lists the fuel cycle-related R&D areas that could improve the resource input, conversion, and waste output links in manner that would further enable Generation 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 are robust with respect to degradation mechanisms over geologic times is needed. Moreover, integrated interim storage/geologic repository logistic strategies that manage decay heat loads should be developed as a cost-effective way 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 4.8-1. Once-Through Generic Cycles: Fuel Cycle Crosscutting R&D

Going In	Conversion	Coming Out
• Technologies of Cost Effective - Prospecting - Mining - Mill Tailings Management • Enrichment Technologies • Fuel & Fuel Fab for Fissile fed-back from recycle or	• High Burnup Fuel • Fuel Suitable for High Thermodynamic Efficiency - High Temp. fuel	• Fuel designed to be robust in repository environment for geologic times

• Low Enriched High Burnup Fuel - epithermal - burnable poison	- Fuel for Supercritical Water	• Improved Heat Mgt. Strategies for interim storage/repository • Improved near field retardants to diffusion of radioactive species • Improved remote monitoring

4.8.2 Cross Cutting R&D for Partial Recycle Fuel Cycle

The partial recycle fuel cycle types differ from the once-through cycles with the introduction of recycling and/or conditioning and refabrication of discharge fuel for mono recycle of plutonium back into thermal neutron spectrum reactors. This adds a new fuel resource (recycled plutonium) 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 4.8-2 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.

The feature of recycling 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 with 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-Pu or full-TRU recycle will ultimately be deployed and that the discharged mono-recycled fuel will then be recycled. 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 Generation 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

degraded isotopics plutonium feedstock, and fuel assembly (and core layout) designs for use with multi-recycle fuel feedstock of degraded isotope composition.

Table 4.8-2. Partial Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences relative to once-through fuel cycles: Recycling for harvesting Pu for return to thermal tpectrum systems

Going In	Conversion	Coming Out
• Same as Once-Thru for harvesting virgin ore and enrichment tails • For harvesting fissile from Spent Fuel: - Simplified Pu Recycle (S-PUREX) - Simplified Pu Refab - Fab using recovered irradiated U - Re-enrichment technologies for recovered irradiated uranium (U236) - Reduced Secondary Waste Recycle • Remote fuel fabrication for feedstock fed back from recycle • Front end recycling technology for high plutonium content fuel	• High temperature fuel • High Burnup fuel • Fuels (and core designs) for several recycle in Thermal Spectrum	• Recycle Fuel which is robust in repository environment and • Robust Customized Waste Forms MA FP High Heat FPs or Long-term storage technologies (to wait for full recycle regime):

4.8.3 Cross Cutting R&D for Full Pu Recycle and Full Transuranic Recycle Fuel Cycles

Whereas the once-through and partial recycle fuel cycles are already commercialized, the full plutonium recycle and full transuranic recycle generic fuel cycles are still in the development phase. Both cycles introduce two new elements: fast neutron spectrum reactors and infinite (multi) recycle, which have attendant technical issues. 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 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 recycling technology can be considered – eliminating process steps and saving on cost while achieving only partial fission product elimination.

Tables 4.8-3 and 4.8-4 list fuel cycle-related R&D for the full plutonium recycle and the full TRU recycle generic fuel cycles, respectively. Independent of 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 TRU recycle case.

Full recycle, especially full TRU 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 many of the radioactive and radiotoxic wastes in the reactors themselves. The resulting requirement for remote refabrication drives the need to develop simple, few-step techniques including simplified pelletizing or vibrocompaction for ceramics fuel and casting or powder metallurgical procedures for metal fuels. Dispersion fuel development is also considered. Significant challenges are presented to develop fabrication technologies which retain volatile species (e.g., Am) in the fuel form during fabrication and to minimize or eliminate losses of the targeted isotopes during recycle and fabrication processes.

New fuel types of compositions containing increased plutonium fraction, new compounds (such as nitrides, spinels, apatites, etc.) and the potential presence of significant minor actinides fractions, fabricated using remote technologies, are recognized as challenging. 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. 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 against Generation IV goals) and the reactor power plant (controlling cost, reliability, and safety performance against Generation IV goals). Uranium oxide fuel is highly refined and dominates the current once-through commercial fuel cycles and is qualified for fast reactor service as well. But new Generation IV reactor concepts and fuel cycles must be afforded opportunities to explore and exploit the potential of numerous other 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 offers the potential to achieve a dramatic reduction in the toxicity consigned to waste management – factors of several orders of magnitude may be possible. The concomitant technical challenge is to develop recycle technologies capable of achieving the very small trace losses to waste during recycling and refabrication, at commercial scale, which are required to exploit this potential. Work started in the mid-1980s on this challenge, which must remain a central theme in the Generation IV research program.

Under a working hypothesis that the required low level for trace losses can be achieved, capacities of future geologic repositories can be extended and their functional requirements can be relaxed. The importance of near-term siting, licensing, and deploying a geologic repository designed for the once-through fuel cycle cannot be

overstated. For the future, however, the development of recycle-based fuel cycles should not go forward absent close linkage with the development of new repository designs to exploit new opportunities for customized waste forms and optimized decay heat management strategies, which can extend their capacities and reduce the level of required stewardship. Therefore, closely-linked R&D for new recycle and new repository design options should be considered and evaluated as an integrated element of the Generation IV R&D activities.

Finally, R&D on recycling, refabrication, and fuel designs for recycle of materials derived from full Pu or TRU recycle generic fuel cycles to thermal spectrum reactors (as well as to fast reactors) must be a part of Generation IV development. As shown in the scenarios, the feedback of fissile material bred in fast spectrum reactors to fuel thermal reactors will be needed no later than twenty years after the Generation IV deployment date in the 2030's to compensate for dwindling virgin ore resources. A proper balance between the beneficial strategy of dirty fuel and clean waste (required to meet Generation IV sustainability goals SU-2 and SU-3) versus the limited ability of thermal spectrum reactors to accommodate minor actinides, degraded plutonium isotopics and lanthanide fission products must be determined. This is relevant especially to those simplified, full-TRU recycling 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 Generation IV research program.

Table 4.8-3. Full Pu Recycle Generic Cycles: Fuel Cycle Crosscutting R&D

Differences from once-through and partial recycle fuel cycles: Multiple recycle in fast-spectrum reactors; >99% Energy Content of Ore

Going In	Conversion	Coming Out
• Several Century Hiatus in Prospecting Mining/Milling available for R&D on those issues • Recycling of fast reactor fuels (higher content of fissile) - Large fission product (FP) carryover is OK for FR - Large FP carryover is not OK for thermal • Fab of fast reactor fuel (high Pu content) • Hands of Fab Thermal Reactor	• Same as once-thru and partial recycle	• Robust Waste Form for FP + TRU • Customized Waste Forms Ma FP High Heat FPs or Long-term storage technologies (to wait for full recycle regime)

Fuel - But remote if poor FR Separation • Simplified recycle Technologies (<10-4 decontamination from FP's)		

Table 4.8.4 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 Fuel Thermal Reactors

Going In	Conversion	Coming Out
• Remote Fab of very high radiation feedstock - Retaining Am in the mix • Accountancy of very high radiation fuels & of processes having no input accountancy tank • Cost Effective, simplified recycle with secondary waste reduction and high TRU recovery • Integration/Co-siting for Recycle/Refab	• Same as for other fuel cycles	• Less stringent functional requirements on Repository Designs owing to changed source term • Issues - Is it worth isotopic separations to get Cs for high heat separate from cs for long life - Is it worth trying to harvest the rare metal FP's: