Nuclear Fuel Cycles for Mid-Century Deployment

by

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Abstract

A comparative analysis of nuclear fuel cycles was carried out. Fuel cycles reviewed include: once-through fuel cycles in LWRs, PHWRs, HTGRs, and fast gas cooled breed and burn reactors; single-pass recycle schemes: plutonium recycle in LWRs and direct-use of spent PWR fuel in CANDU reactors (DUPIC); multi-pass recycle schemes: transmutation of transuranics in LWRs, fast reactors, double strata systems, and molten salt reactors. Mass flow calculations for the fuel cycles at equilibrium were carried out based on data available in the open literature, and results were used to compare the performance of the fuel cycles with respect to uranium utilization, waste management, proliferation resistance, and economics. Potential for mid-century deployment was assessed based on these results. Once-through fuel cycles based on solid fuel thermal reactors are found to be the best candidates for mid-century deployment because the substantial increase in electricity costs entailed by reprocessing schemes is unlikely to be justified by the afforded reductions in long-term proliferation and waste management risks. Furthermore, once-through cycles present lower proliferation and waste management risks in the short-term and their inefficient use of uranium is not likely to become an important issue before the middle of the century even under a high growth scenario.

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1 Introduction

The current nuclear industry is based predominantly on light water reactors (LWR), which account for almost 90% of installed capacity, with pressurized heavy water reactors (PHWR) also making a significant contribution with 5% of installed capacity. Most of these reactors operate on the once-through fuel cycle, where natural uranium is enriched to make uranium oxide (UOX) fuel, and, after irradiation, the spent fuel is encapsulated and disposed of directly in the repository. On the other hand, several countries choose to reprocess their spent fuel. Worldwide, a sizable fraction (>10%) of spent LWR fuel is reprocessed in order to extract plutonium[1], which is recycled and used to make mixed oxide (MOX) fuel. The mixed oxide fuel is irradiated in conventional PWRs and then sent to the repository[2]. This fuel cycle is usually referred to as single-pass plutonium recycling.

The reactors and fuel cycles identified above, which dominate today's nuclear electricity industry, represent only a small subset of possibilities. Researchers in the field of nuclear energy have devoted much time and effort to devise fuel cycles that improve on today's technology. Proposals for innovative reactors and fuel cycles abound in the open literature but none has emerged as a clear winner, as each one has its particular strengths and weaknesses.

This situation has spurred efforts to review, evaluate, and compare fuel cycles on a methodical basis. The present study is one such effort, where the emphasis is placed on identifying fuel cycles that constitute viable options for mid-century deployment, meaning that their characteristics must match the needs of the nuclear electricity industry in the coming decades and that a significant deployment is feasible on this time scale.

A literature review was conducted to identify fuel cycles that are currently generating interest, present attractive features for mid-century deployment, and together represent a broad spectrum of possibilities. The fuel cycles were evaluated on the basis of their performance in 4 areas: resource utilization, waste management, proliferation resitance, and economics. For each fuel cycle, mass flow calculations were performed based on data available in the literature. For simplicity, only equilibrium conditions were considered.

Conclusions on the outlook for mid-century deployment of the various fuel cycles are focused on issues that are likely to be most relevant in this period, and on the practical obstacles to deployment within this time frame.

[1] Uranium is also recovered, but, under current conditions, it is not economical to use it to make new fuel. Thus it is stored for possible future use.
[2] Future developments may allow one or two additional recycling passes before disposal.

2 Evaluation Metrics

Nuclear fuel cycles are evaluated and compared in chapter 3 on the basis of quantitative measures that fall into 4 categories: resource utilization, waste production, proliferation, and economics. In order to maintain a manageable level of complexity in the analysis, the set of indicators chosen is both simplified and limited. These quantitative indicators, or metrics, are detailed below.

2.1 Resource Utilization

Resource utilization is measured as the mass of natural uranium (or thorium) required per unit energy generated. Units chosen in this analysis are MTU/GWe·y, or metric tons of natural uranium per GigaWatt-electric-year. As shown in appendix A, the present uranium resource base is sufficient to support our current rate of consumption for about 250 years. In addition, although the availability of natural resources in the future cannot be predicted, there is good reason to believe that resources are adequate to support a large nuclear energy expansion based on the once-through fuel cycle until the middle of the century and beyond. Thus, resource utilization is perhaps the least important of the metrics considered in this analysis.

2.2 Waste

The impact of nuclear waste on the environment is difficult to evaluate. In-situ radiotoxicity, or the total radiotoxicity of all the isotopes present in nuclear waste, is an inadequate indicator of the risk posed by such wastes. Indeed, the long-term dose to the environment and the public from nuclear waste repositories is due to the radioactive isotopes that have the highest mobility in the geologic environment of the repository. As a general rule, the radiotoxicity of fission products is at least 2 orders of magnitude below that of the actinides after a few hundred years, but they are much more mobile. As a result, the dose to the public from a nuclear waste repository is dominated by fission products for the first million years or so after closure, except in the case of Yucca Mountain, an oxidizing environment, where actinides start to dominate after less than 100,000 years (see Figure 2.1 [24][1]).

The previous considerations have important practical implications: the vast majority of proposals for transmutation of nuclear waste have focused on actinides because of their large contribution to in-situ radiotoxicity. But such schemes would do little to reduce the dose from repositories. Furthermore, as indicated in appendix B, the prospects for

[1] In the case of Yucca Mountain, the curve is based on a 1998 DOE report (DOE/RW-0508), "Viability Assessment of a Repository at Yucca Mountain" and is consistent with more recent findings, reported in DOE/RW-0539-1 (2002) "Yucca Mountain Science and Engineering Report Rev. 1". The uncertainties associated with such calculations are significant, especially for 100,000 and 1,000,000 year time frames. For projections beyond 10,000 years, DOE/RW-0539-1 cautions that "these projections should not be interpreted as predictions of probable future performance. They are simply indicators of the possible range of performance." Note also that the curves for the different repositories on Figure 2.1 are based on separate and independent sources. For these reasons, Figure 2.1 should not be used as a basis for comparing the performance of various repositories.

transmutation of fission products are not good due to the long irradiation times required and the technical difficulties associated with the fabrication of targets. Therefore, the effectiveness of transmutation schemes in reducing the long-term dose from nuclear waste repositories can be questioned.

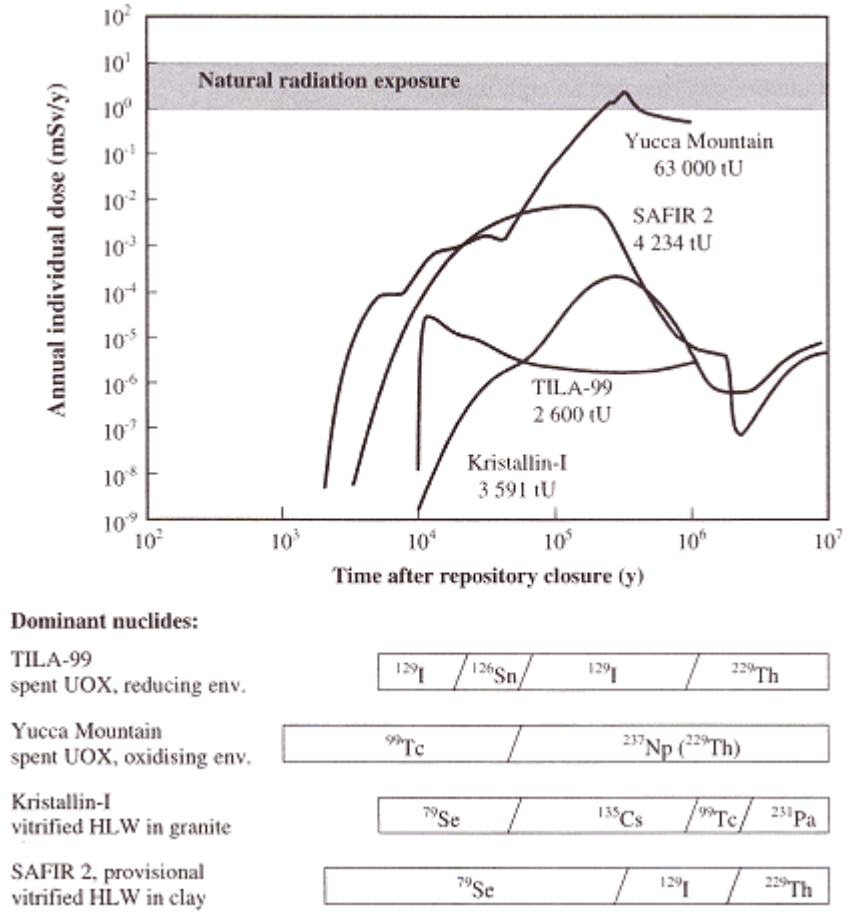

Figure 2.1. Annual dose from different repository concepts [24]

In this analysis, 2 metrics will be used to evaluate the waste production of fuel cycles: the mass of transuranics and the mass of fission products discharged per unit of energy generated, in kg/GWe·y or MT/GWe·y.

2.3 Proliferation

The proliferation risks posed by any given fuel cycle are difficult to quantify. Several activities related to the production of nuclear electricity raise proliferation concerns to some degree, either because they present opportunities for diversion of sensitive nuclear material or because they promote the dissemination of knowledge that could be used to manufacture a weapon. Thus numerous metrics are required to evaluate the proliferation dangers associated with a given fuel cycle, and it in many cases the risks are difficult to quantify. Brogli and Krakowski [27] have proposed a metric, called *proliferation*

attractiveness, which gives a single measure of the overall risk from all the activities in a fuel cycle. However, this method involves assigning attractiveness levels to each activity in the fuel cycle, which is somewhat subjective and can influence results significantly. Thus, the method's usefulness is undermined because results must always be carefully evaluated by taking into account the particular attractiveness levels used for the calculation. To avoid these difficulties, no such method is used in this analysis. Instead, a set of 5 metrics, describing the aspects of nuclear fuel cycles that present the highest proliferation risks, will be used:

- Diversion of nuclear material (short-term):
  The 2 metrics used to quantify this risk are the working inventories of separated plutonium and transuranics required per unit of installed capacity, in kg/GWe.

- Diversion of nuclear material (long-term):
  The 2 metrics used to quantify this risk are the mass flow rate of plutonium and transuranics going to the repository per unit of energy generated, in kg/GWe·y.

- Propagation of technologies related to nuclear weapons:
  The metric used to quantify this risk is the mass flow rate of nuclear fuel reprocessed per unit of energy generated, in kgHM/GWe·y.

Once-through fuel cycles will fare better in terms of diversion of nuclear material in the short-term and propagation of sensitive technologies, whereas reprocessing significantly reduces the quantities of plutonium and transuranics going to the repository.

2.4 Economics

Evaluating the economic performance of various nuclear fuel cycles is extremely difficult. The uncertainties involved in estimating reactor costs and costs associated with fuel cycle operations are significant. This is true of currently deployed fuel cycles, such as once-through UOX and plutonium recycling (MOX) in LWRs, and even more so for advanced fuel cycles, for which no reliable economic data exists. Consequently, the approach used in this analysis relies on an extremely simplified calculation of nuclear electricity costs, which is referred to as the cost of electricity index, or CEI, to reflect its purpose as an approximate indicator only. The CEI is calculated as the sum of the costs related to the reactor, CEI_R, and the costs related to fuel cycle operations, CEI_{FC}, as follows (note that financing charges for the fuel cycle operations are neglected):

$$CEI = CEI_R + CEI_{FC}$$

Eq. 1:
$$CEI = \frac{1}{8.76 \cdot CF}\left(\frac{I}{P_e}\right)\cdot\left(\phi + f_{OM}\right) + \frac{1}{8.76 \cdot 10^6}\sum_i M_i \cdot C_i$$

where: CF = capacity factor

I/P_e = overnight construction cost ($/kWe)

φ = carrying charge (/year)

f_{OM} = O&M costs as a fraction of I/P_e (/year)

M_i = mass flow rate for process i (kg/GWe·y)

C_i = unit cost for process i ($/kg)

For fuel cycles involving more than one type of reactor, the overnight construction cost is obtained as:

$$\left(\frac{I}{P_e}\right) = \sum_i f_i \cdot \left(\frac{I}{P_e}\right)_i$$

where f_i is the fraction of installed electric capacity for reactors of type i.

The assumptions used to arrive at the values of the various parameters in Eq. 1 are listed in Table 2.1. They are based on [27], except where noted otherwise.

Note that the separative work requirement per unit mass of enriched product is obtained as follows:

$$\frac{\text{kg SWU}}{\text{kg product}} = (2x_p - 1)\cdot \ln\left(\frac{x_p}{1-x_p}\right) + \frac{x_p - x_{nat}}{x_{nat} - x_t}\cdot(2x_t - 1)\cdot \ln\left(\frac{x_t}{1-x_t}\right) - \frac{x_p - x_t}{x_{nat} - x_t}\cdot(2x_{nat} - 1)\cdot \ln\left(\frac{x_{nat}}{1-x_{nat}}\right)$$

where x_p, x_t, and x_{nat} represent the mass fraction U-235 enrichment of the product, the tails, and the natural uranium feed, respectively.

Table 2.1. Assumptions used in the calculation of CEI

Parameter	Value
Reactor	
CF	0.9
Φ	0.10/y
f_{OM}	0.04/y
(I/P_e):	
Thermal reactors (except molten salt)	1,700 $/kWe
Fast reactors and molten salt reactors	2,100 $/kWe
Accelerator driven systems	3,600 $/kWe
Fuel Cycle	
Uranium ore ($/kgU)	30
Enrichment ($/kg SWU)	100
Fuel fabrication ($/kgHM)	
▪ LWR-UOX	275
▪ LWR-MOX	1,500
▪ PHWR[a]	65
▪ HTGR[b]	275
▪ WASB seed[c]	500
▪ WASB blanket[c]	300
▪ B+B	450
▪ DUPIC[a] (incl. OREOX process)	550
▪ LWR-FFF	11,000
▪ FR-TRU	2,300
▪ FR-MOX	1,500
▪ ADS-MA	11,000
▪ GT-MHR (deep burn)[d]	1,500
▪ MABR[e]	11,000
Reprocessing ($/kgHM)	
▪ LWR-UOX	1,000
▪ LWR-MOX	1,000
▪ LWR-FFF	7,000
▪ FR-TRU	2,000
▪ FR-MOX	2,000
▪ ADS-MA	7,000
▪ GT-MHR (deep burn)[f]	7,000
▪ MABR[g]	7,000
▪ MSR (on-line)[h]	1,000
Storage and disposal[i]	
▪ Spent fuel	1 mill/kWh
▪ HLW (0.1% Pu, 100% TRU)	300 $/kgHM$_{reprocessed}$
▪ HLW (0.1% Pu, 0.1% TRU)	200 $/kgHM$_{reprocessed}$

Notes for Table 2.1:

a. From [28].
b. Assumed equal to cost of UOX fabrication, as in [10].
c. From [16]
d. Assumed equal to cost of MOX fabrication, since this fuel is made with plutonium recovered from spent UOX.
e. Assumed equal to cost of ADS MA fuel since both are fertile free TRU fuels.
f. Assumed equal to cost of ADS MA fuel due to high TRU content and high burnup of spent GT-MHR deep burn fuel.
g. Assumed equal to cost of ADS MA fuel since both are fertile free TRU fuels.
h. Assumed modest cost of reprocessing due to limited shielding requirements (on-line processing of liquid fuel) and relatively low contamination of fuel due to frequent reprocessing.
i. Studies such as [18] and [24] assume low costs for HLW storage and disposal (less than 100 $/kgHM). Assumptions used here differ for the following reasons: while the volume of HLW is significantly reduced, it contains all the fission products present in the spent fuel, including Sr-90 and Cs-137, which are the most important contributors to decay heat in the first few centuries after discharge and therefore have a significant impact on storage and disposal requirements. When only Pu is recycled, the HLW also contains all the minor actinides from the spent fuel.

2.5 Table of Metrics

Table 2.2. Metrics used for evaluation of fuel cycles

Metric	Units
Resource utilization	
▪ Uranium consumption	MTU/GWe·y
Waste	
▪ Transuranic discharge	kg/GWe·y
▪ Fission product discharge	kg/GWe·y
Proliferation	
▪ Plutonium inventory	kg/GWe
▪ Transuranic inventory	kg/GWe
▪ Plutonium discharge	kg/GWe·y
▪ Transuranic discharge	kg/GWe·y
▪ Reprocessing rate	kg/GWe·y
Economics	
▪ Reactor cost index	mills/kWh
▪ Fuel cycle cost index	mills/kWh
▪ Cost of electricity index	mills/kWh

3 Analysis of Fuel Cycles

3.1 Once-Through

3.1.1 Light Water Reactor

The light water reactor (LWR) is the work horse of the nuclear energy industry. Out of a total of 437 reactors in operation as of April 2003, there are 213 pressurized water reactors (PWR), 90 boiling water reactors (BWR), 50 Russian type pressurized water reactors (VVER), and 2 advanced boiling water reactors (ABWR). Together these reactors account for 316,641 MWe, or 88% of the total worldwide installed nuclear capacity (358,461 MWe). 23 LWRs with a total capacity of 22,387 MWe are under construction, distributed as follows: 4 ABWRs (5,329 MWe), 1 BWR (1,067 MWe), 8 PWRs (7,681 MWe), and 10 VVERs (8,310 MWe). In addition, 17 light water cooled graphite moderated reactors (LWGR) account for 12,589 MWe, and one 925 MWe LWGR is under construction [1].

The world's LWRs operate on either of 2 fuel cycles: once-through or single-pass plutonium recycle. The latter option will be considered in more detail in section 3.2.1. the once-through fuel cycle consists of the following steps: mining and milling; conversion from U_3O_8 (solid) to UF_6 (gas); enrichment; conversion from UF_6 to UO_2 (solid); fuel fabrication; irradiation in the reactor; storage of irradiated fuel for cooling; direct disposal of spent fuel.

Several variations exist within the once-through option. For example, the fuel loaded into BWRs typically requires lower enrichment and achieves a lower burnup than the fuel loaded into PWRs. However, because such differences do not have a significant impact on the fuel cycle, one specific set of assumptions will be adopted as representative of the LWR once-through fuel cycle. Specifically, the analysis will be based on characteristics of PWRs, as this is the most widely deployed type of LWR.

Mass Flow Analysis

The mass of fuel that needs to be irradiated for the production of 1 GWe·y of electricity can be obtained from the burnup and the reactor thermal efficiency:

Eq. 2:
$$m = \frac{365}{B_d \cdot \eta_{th}}$$

where: m: mass of fuel (MTHM/ GWe·y)

B_d: discharge burnup (GWd/MTHM)

η_{th}: thermal efficiency

Assuming a burnup of 50 GWd/MTHM and a thermal efficiency of 33%, we find m = 22.12 MTHM. This figure is useful in evaluating the waste management requirements of this fuel cycle. For instance, comparing it with the nominal capacity of the proposed Yucca Mountain repository, 70,000 MTHM, it is found that roughly 3,167 GWe·y of electricity production will generate enough spent fuel to fill the repository to its currently licensed capacity. The current U.S. fleet, with an installed capacity of 98,230 MWe, would require a repository the size of Yucca Mountain every 32 years; the world's current LWR fleet, 316,641 MWe, every 10 years. In light of the significant difficulties encountered in proceeding with plans for a repository at Yucca Mountain, and considering that a solution for high level waste disposal has yet to be implemented anywhere in the world, this is significant. In fact, waste disposal is generally considered to be the most significant challenge for the LWR once-through option.

Another issue, which was once generally perceived as a major constraint on nuclear deployment based on the once-through fuel cycle, is the availability of uranium. A brief examination of today's reported uranium resources compared to the requirements of LWRs on the once-through cycle clarifies this question.

The feed-to-product mass ratio for the enrichment process is given as:

Eq. 3:
$$\frac{F}{P} = \frac{x_p - x_t}{x_n - x_t}$$

where: F: mass of feed (any unit)

P: mass of product (any unit)

x_p: product enrichment

x_t: tails assay

x_n: natural enrichment (0.711%)

The tails assay is determined by the enrichment plant operator and generally varies between 0.2% and 0.4%. Here, a value of 0.3% will be assumed. The product enrichment is determined by the needs of the reactor operator. As noted previously, different reactor types may require different fuel enrichments. In addition, for a given reactor type, higher burnup requires higher initial enrichment. The following correlation, valid for standard PWRs, can be used to estimate enrichment as a function of burnup [2]:

Eq. 4:
$$x_p = 0.41201 + 0.11508 \cdot \left(\frac{n+1}{2n} \cdot B_d \right) + 0.00023937 \cdot \left(\frac{n+1}{2n} \cdot B_d \right)^2$$

where: B_d: discharge burnup (GWd/MTHM)

n: number of batches

The number of batches is selected according to the fuel management scheme adopted by the reactor operator. In the U.S., the number is typically approximately 3. Using Eq. 4 with n = 3 and B_d = 50 GWd/MTHM, the resulting U-235 enrichment is x_p = 4.51%. Using (2), we find F/P=10.24. Recalling that the mass of fuel per GWe·y is 22.12 MTHM, the mass of uranium ore required is 226.5 MTU/GWe·y (MTU: metric tons of uranium).

Uranium resource estimates are reported in the OECD *redbook* [3]. The latest estimates show 16.2 million MTU of total conventional resources, or 71,523 GWe·y, according to the previous calculation. This is enough to supply the world's current LWR fleet of 316,641 MWe, running at 100% capacity, for 226 years. Therefore, the availability of uranium is not an obstacle to the growth of nuclear power based on the once-through fuel cycle in LWRs. Note, however, that uranium utilization in this fuel cycle is very poor. Indeed, for a burnup of 50 GWd/MTHM, approximately 5% of the uranium atoms initially present in the fuel, or 1.105 MTHM/GWe·y, undergo fission[1]. This is approximately 0.5% of the uranium initially mined; hence, fuel cycle technologies involving full actinide recycle could theoretically reduce uranium consumption by a factor of about 200.

LWR with High Burnup Fuel

There has been a clear trend over the last decades toward increasing burnup in light water reactors. For reactor operators, higher burnup means that a given batch of fuel stays in the reactor for a longer period before it needs to be discharged from the reactor. For a fuel management scheme based on a given number of batches, a higher burnup results in longer intervals between reactor reloads, which tends to increase plant availability and revenues from electricity production. In addition, burnup has a moderate influence on the fuel cycle cost, mainly due to its effect on enrichment and fuel fabrication requirements. On one hand, higher burnup fuel requires higher initial enrichment to sustain reactivity as fissile material is consumed and fission products build up. On the other hand, because high burnup fuel produces more energy per unit mass, fuel fabrication requirements are lessened. These two counterbalancing effects result in a minimum fuel cycle cost which, for U.S. conditions, is in the range of 60 to 65 GWd/MTHM. It should be noted however, that the effect of burnup on fuel cycle cost is minimal, with a variation of only about 5% (approximately 0.3 mills/kWh) over the 40-100 GWd/MTHM range (see Figure 3.1 [2]).

[1] Some of those fissions occur after neutron capture by uranium nuclei. For example, Pu-239, a fissile element created by neutron capture in U-238, accounts for a large fraction of the fissions.

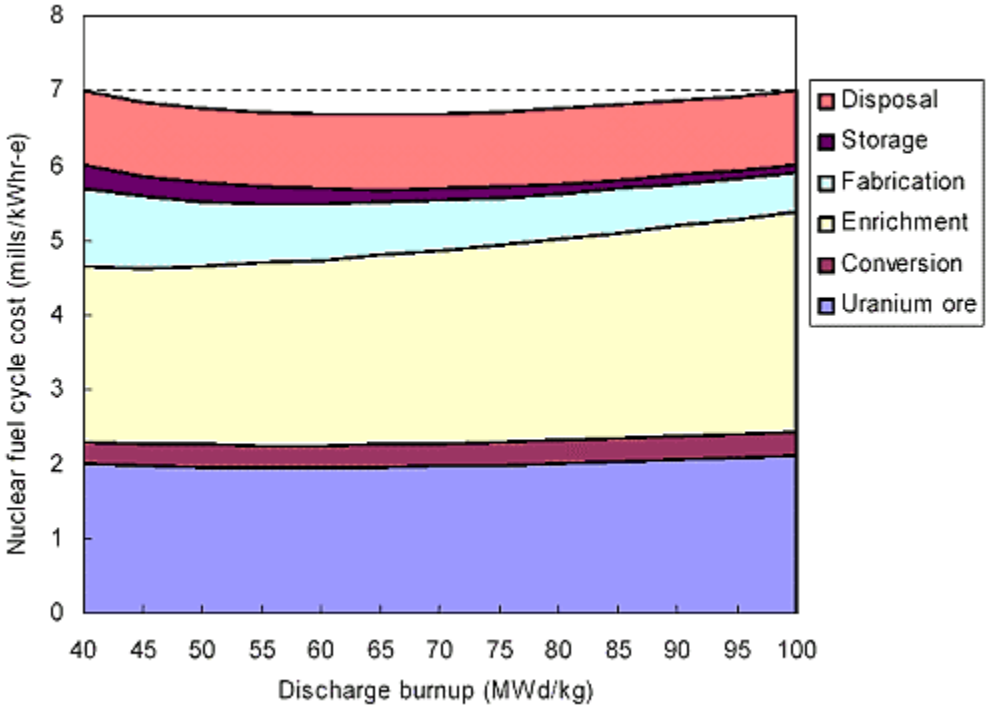

Figure 3.1 Fuel cycle cost as a function of burnup for conventional PWRs

Increased burnup places additional demands on fuel cladding. The accumulation of gaseous fission products inside the cladding results in significant internal pressure that the clad must withstand without bursting. In addition, the longer residence in the reactor requires a higher corrosion resistance. To illustrate the effect of high burnup on the fuel cycle, we will determine the characteristics of the LWR once-through cycle with a burnup of 100 GWd/MTHM. This value is an approximate upper bound on the burnup that can be achieved without radically redesigning LWR fuel. The effect of increasing burnup from the current level of 50 GWd/MTHM to 100 GWd/MTHM will now be examined.

Mass Flow Analysis for High Burnup Fuel

The relationship between fuel burnup and uranium ore requirements is determined by two counteracting effects. Higher burnup fuel generates more energy per unit mass so the mass of fuel loaded into reactors is lower. Specifically, if burnup is doubled from 50 to 100 GWd/MTHM, fuel requirement is halved and becomes 11.05 MTHM/GWe·y. However, the higher U-235 content of the fuel means that the ratio of feed to product in the enrichment process is higher. Eq. 4, using $n = 3$ and $B_d = 100$ GWd/MTHM, gives $x_p = 9.15\%$. Using (2), $F/P = 21.53$ and the uranium ore requirement is 237.9 MTU/GWe·y, or approximately 5% higher than for current burnup. If the number of batches is increased to 5, we find $x_p = 8.18\%$, $F/P = 19.17$, and the ore requirement is 211.8 MTU/GWe·y, or about 6% lower than for current burnup with 3 batches. Hence increasing burnup has only a modest impact on natural uranium requirements.

Table 3.1. Effect of burnup on uranium resource requirement

Enrichment (% U-235)	4.51%	9.15%	8.18%
Number of batches	3	3	5
Burnup (GWd/MTHM)	50	100	100
Uranium requirement (MTU/GWe·y)	226.5	237.9	211.8
Spent fuel discharge (MTHM/GWe·y)	22.1	11.05	11.05

High burnup results in a lower mass of spent fuel discharged per electricity generated, as this quantity in inversely proportional to burnup. This can significantly reduce spent fuel handling and transportation, but the impact on waste management is modest. Indeed, the mass of fission products produced in nuclear power plants is proportional to energy production and completely independent of burnup. If high burnup is used, the fission products are simply concentrated in a smaller mass of spent fuel. Therefore, the impact on storage requirements and repository footprint is minimal, as they are determined mainly by the decay heat from fission products. In addition, for repository environments where fission products dominate the risk to the public, fuel burnup will not influence the long-term performance of the repository. However, if transuranic actinides dominate repository risk, there is a modest benefit from high burnup because these nuclides occur in lower amounts, per unit energy generated, in high burnup fuel.

The spent fuel composition for the two burnup levels is shown in Table 3.2 [2, MCODE output]. The table shows that, for the same amount of energy generated, high burnup fuel produces 22% fewer transuranic elements and the same amount of fission products. For a repository where transuranics dominate long-term risk, there is some gain from going to high burnup, but it is not significant. For repositories where long-lived fission products dominate long-term risk, there is practically no benefit. Table 3.3 summarizes waste discharges on a per GWe·y basis. Note that the mass of plutonium discharged is significantly decreased by going to high burnup while the mass of minor actinides discharged is increased.

Table 3.2. Spent fuel composition at 50 and 100 GWd/MTHM

Component	composition @ 50 GWd/MTHM	Composition @ 100 GWd/MTHM	1 assembly @ 100 GWd/MTHM / 2 assemblies @ 50 GWd/MTHM
Uranium	93.40 %	87.42 %	0.47
Plutonium	1.33 %	1.97 %	0.74
Minor Actinides	0.12 %	0.30 %	1.22
Transuranics	1.45 %	2.27 %	0.78
Fission Products	5.15 %	10.31 %	1.00

Table 3.3. Spent fuel discharge at 50 and 100 GWd/MTHM

	50 GWd/MTHM	100 GWd/MTHM (3 batches)
Uranium consumption (MTU/GWe·y)	226.5	237.9
Spent fuel discharge (MTHM/GWe·y)	22.1	11.05
Transuranic discharge (kg/GWe·y)	320	251
Pu discharge (kg/Gwe·y)	294	218
Minor actinide discharge (kg/GWe·y)	26.5	33.2
Fission product discharge (MT/GWe·y)	1.14	1.14

The effect of burnup on decay heat load is shown in Table 3.4 [2]. As expected, decay heat characteristics for the two burnup levels are comparable, and therefore repository footprint cannot be greatly influenced by burnup.

Table 3.4. Effect of burnup on decay heat characteristics

		1 assembly @ 100 GWd/MTHM / 2 assemblies @ 50 GWd/MTHM
Storage time frame (10-100 years)	Decay Power (100 years)	1.06
	Integrated heat load	1.11
Disposal time frame (100-1000 years)	Decay Power (1000 years)	0.77
	Integrated heat load	0.89

Economics

The LWR once-through fuel cycle is a relatively attractive option economically. Light water reactors represent the most mature nuclear reactor technology (heavy water reactors are also well established). This is a significant advantage because the cost of bringing a new reactor technology to commercial deployment is enormous and because construction and O&M costs dominate the total electricity generation cost for any nuclear technology. Compared to the MOX option, which also relies on LWRs, the fuel cycle cost of the once-through option is lower, although estimates vary widely: European studies have found the two options to be comparable, while estimates based on U.S. conditions find the MOX option to be four to five times more expensive (see appendix C).

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.5. As expected, fuel burnup has a very modest impact on the cost of electricity.

Note that the unit cost for storage and disposal of spent fuel corresponds to the fee of 1 mill per kilowatt-hour of nuclear electricity generated paid to the DOE by each utility operating a nuclear power plant:

$$\frac{0.001\$}{kWh(e)} \cdot \frac{0.33 kWh(e)}{1 kWh} \cdot \frac{24h}{1d} \cdot \frac{10^6 kW}{1MW} \cdot \frac{50GWd}{1MTHM} \cdot \frac{1MT}{1000kg} \approx 400\$/kgHM$$

For a burnup of 100 GWd/MTHM, the cost of storage and disposal is doubled to $800/kgHM.

Table 3.5 CEI for LWR once-through (mills/kWh)

	50 GWd/MTHM	100 GWd/MTHM
CEI_R	30.18	30.18
CEI_{FC}	4.06	4.13
CEI	34.24	34.31

Proliferation

The LWR once-through fuel cycle has favorable characteristics for proliferation resistance. The front end of the fuel cycle requires enrichment to levels that are far below the threshold where proliferation concerns are raised (20%) and the direct disposal of spent fuel eliminates reprocessing operations where components of spent fuel can be separated. In particular, the separation of plutonium, as in the MOX option, or the separation of actinides, as in many fuel cycles currently proposed for actinide recycle, is worrying because streams of bomb usable material are produced. In the once-through cycle, illicit retrieval of plutonium or actinides is more difficult because the perpetrators would have to carry out separation operations. In addition, the presence of highly radioactive fission products in spent fuel requires adequate shielding for all handling operations, providing a barrier against proliferation in the first hundred or so years after discharge.

The plutonium in high burnup fuel is less attractive as a weapons material because it contains higher concentrations of Pu-238. Pu-238 is a powerful decay heat source and its presence significantly complicates plutonium handling and bomb assembly. Furthermore, it is a prolific neutron source, which causes fizzle detonation and severely degrades bomb yield. High burnup fuel is also better protected from illegitimate handling by its higher radioactivity level.

Table 3.2 shows that high burnup results in a 26% reduction in the quantity of plutonium produced per energy generated, but this needs to be put in perspective: the mass of spent fuel discharged from a single reactor operating with high burnup fuel is 11.05 MTHM/GWe·y. This amount of fuel alone contains 217.7 kg of plutonium (for current burnup, plutonium discharge is 293.9 kg/GWe·y). Considering that a bomb requires roughly 10 kg, this is still an enormous quantity. Furthermore, on a per fuel assembly basis, high burnup fuel contains more plutonium, so fewer assemblies would have to be removed from the repository to recover a given amount of plutonium. Indeed, assuming that a fuel assembly has a mass of 0.5 MTHM, a regular unit contains 6.65 kg of plutonium, as compared to 9.85 kg for a high burnup assembly.

Table 3.7 [2] shows the isotopic composition of the plutonium in spent fuel at various times after discharge. The plutonium in high burnup fuel contains more Pu-238 upon

discharge from the reactor, but because of the relatively short half-life of this isotope (87.75 years), its concentration is significantly reduced after 100 years. Thus, high burnup fuel does present a significant proliferation barrier due to its higher Pu-238 content, but only for the first few hundred years or so after discharge.

Table 3.6. Plutonium production in LWRs at 50 and 100 GWd/MTHM

	50 GWd/MTHM	100 GWd/MTHM
Pu discharge (kg/GWey)	293.9	217.7
Pu content of 1 assembly (kg)	6.65	9.85

Table 3.7. Plutonium isotopic composition at 50 and 100 GWd/MTHM

	Plutonium composition					
	10 years after discharge		100 years after discharge		1000 years after discharge	
Burnup(GWd/MTHM)	50	100	50	100	50	100
Pu-238	2.6%	6.6%	1.4%	3.7%	-	-
Pu-239	58.3%	53.7%	65.4%	61.6%	67.5%	65.2%
Pu-240	23.2%	22.1%	26.4%	26.5%	25.4%	26.1%
Pu-241	10.0%	10.6%	0.1%	0.2%	-	-
Pu-242	6.0%	7.0%	6.7%	8.1%	7.1%	8.7%

3.1.2 Pressurized Heavy Water Reactor

As of May 2003, 35 pressurized heavy water reactors (PHWR) are in operation around the world. Their net installed capacity, 17,180 MWe, represents about 5% of the total world installed capacity of 358,461 MWe. Heavy water reactor technology is alive and well, with 9 PHWRs (3,800 MWe) under construction. Canada's AECL has long been the sole developer of PHWR technology, but today India develops, builds, and operates these reactors independently. Of the 9 PHWRs under construction, 6 are in India [1].

Mass Flow Analysis

The neutron economy of PHWRs is superior to that of LWRs due to the very low parasitic capture of neutrons by the deuterium atoms in heavy water as compared to the hydrogen atoms in light water. As a result, natural uranium at 0.711% U-235 can be used as fuel for these reactors, which obviates expensive enrichment operations. However, due to its low fissile content, the fuel quickly loses reactivity and cannot sustain criticality beyond a burnup of about 7.5 GWd/MTHM. In addition, due to lower core outlet temperatures and pressures, the thermal efficiency of PHWRs is lower than that of LWRs. A value of 31% will be used for this analysis [5]. Using eq.1, the mass of spent fuel discharged per unit energy produced is 157.0 MTHM/GWe·y. Since there is no enrichment, the uranium consumption is 157.0 MTU/GWe·y.

A recent OECD document estimates that transuranic element discharges are 50 kg/TWhe for PHWRs [6]. This corresponds to 438 kgTRU/GWe·y, which implies that PHWR spent fuel contains 0.28% transuranics. The fission product content of spent PHWR fuel, assuming a fission energy yield of 1,000 GWd/MTHM$_{\text{fissioned}}$, is 0.75%, giving a fission product discharge of 1,178 kg/GWe·y.

AECL is currently developing a new fuel bundle, known as CANFLEX, which will allow currently deployed CANDU reactors (CANDU is an acronym for CANadian Deuterium Uranium and refers to PHWRs designed by AECL) to use slightly enriched uranium (SEU). The U-235 content of SEU can vary from 0.9% to 1.2%. For an initial enrichment of 1.2%, the average discharge burnup will be 21 GWd/MTHM [5]. The natural uranium requirement and spent fuel discharge for this case are shown in Table 3.8. The transuranic content of spent SEU fuel is approximately 0.62%[1], for a discharge of 347.8 kg/GWe·y. The fission product discharge, assuming a fission product content of 2.1% for spent fuel, is 1,178 kg/GWe·y.

[1] Personal communication with Gary Dyck of AECL (08/17/2003). Not an official AECL figure.

Table 3.8. PHWR uranium consumption and spent fuel discharges

	Natural uranium	SEU
Fuel enrichment	0.711%	1.2%
Burnup (GWd/MTHM)	7.5	21
Uranium consumption (MTU/GWe·y)	157.0	122.8
Spent fuel (MTHM/GWe·y)	157.0	56.1
Transuranic elements (kg/GWe·y)	438	347.8
Fission product discharge (kg/GWe·y)	1,178	1,178

The next generation of heavy water reactors from AECL, known as advanced CANDU reactors (ACR), use light water as a coolant and continue to rely on heavy water as a moderator only. The neutron economy of these reactors is inferior to that of previous CANDUs and, as a result, they will use 2% enriched uranium. However, the new ACR design affords several advantages, such as a smaller core, significantly reduced heavy water inventory, better safety characteristics (ACR is the first CANDU with a negative void reactivity coefficient), and slightly higher thermal efficiency due to a higher coolant pressure and core outlet temperature.

Some design data is available for the ACR-700, currently under review by the NRC [7]. This reactor's heavy water inventory is reduced by more than 75% compared to a CANDU-6 reactor of comparable power. Thermal power is 1982 MWth and gross electric output is 731 MWe, for a gross thermal efficiency of 36.9%. This can be compared to 2064 MWth, 728 MWe, and 35.3% for a CANDU-6, which suggests that net thermal efficiency of ACRs will be about 1.5% higher than that of CANDUs. An enrichment of 2% and a burnup of 20.5 GWd/MTHM are specified for the ACR-700. Hence, a thermal efficiency of 32.5%, an initial enrichment of 2%, and a burnup of 20.5 GWd/MTHM are assumed here, and the corresponding uranium requirement and spent fuel discharge are shown in Table 3.9. Note that uranium consumption is comparable to that of a typical PWR.

Table 3.9. ACR uranium consumption and spent fuel discharges

Fuel enrichment	2.0%
Burnup (GWd/MTHM)	20.5
Uranium consumption (MTU/GWe·y)	226.6
Spent fuel (MTHM/GWe·y)	54.8

Economics

CANDU reactors have enjoyed commercial success, with 5 units completed on time and on budget since 1990. For the ACR-700, AECL claims an overnight construction cost of $1,000/kWe and a levelized cost of electricity of 3¢/kWh [7]. In this analysis, however, we maintain our assumption of an overnight construction cost of $1,700/kWe for all thermal reactors. Hence, CEI_R will be the same as for LWRs for all three PHWRs options and the focus in this case is on CEI_{FC}.

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.10. Note that the unit cost for storage and disposal of spent fuel is 55 $/kgHM for a natural uranium PHWR, 155 for a SEU fueled PHWR, and 160 for the ACR-700 (corresponding to 1 mill per kilowatt-hour of nuclear electricity). It is seen that CEI_{FC} is highest for the ACR-700 due to its lower uranium utilization efficiency and higher enrichment costs. It must be noted that this difference (≤1 mill/kWh) is trivial compared to the cost reductions afforded by a significant reduction in the overnight construction cost of the reactor. If the overnight construction cost of the ACR could be lowered to $1,000/kWe as AECL claims, the CEI would be 21.0 mills/kWh. For a small improvement to $1,650/kWh, the CEI is 32.54 mills/kWh, on par with a PHWR using SEU.

Table 3.10 CEI (mills/kWh) for PHWR (natural U and 1.2% U-235) and ACR-700

	PHWR (natural U)	PHWR (1.2% U-235)	ACR-700
CEI_{FC}	2.69	2.15	3.25
CEI_R	30.18	30.18	30.18
CEI	32.87	32.33	33.42

Proliferation

Currently deployed PHWRs are viewed as having unfavorable proliferation characteristics as compared to LWRs. This is largely due to the fact that these reactors are refueled on-line, whereas LWRs need to be shut down for refueling. Illicit retrieval of irradiated fuel elements is therefore more difficult to prevent in a PHWR. Furthermore, PHWRs produce more plutonium per energy generated than LWRs. Indeed, Table 3.8 shows that PHWRs discharge 438 kgTRU/GWe·y. Plutonium accounts for most (>90%) of these transuranics and the plutonium discharge rate of PHWRs is approximately 394.2 kg/GWe·y; therefore, recalling that LWRs discharge 294 kgPu/GWe·y, the plutonium production in PHWRs is about 35% higher than in LWRs. For 1.2% SEU, the plutonium content of spent fuel is 0.59%, for a discharge rate of 331.0 kg/GWe·y, which is less than for natural uranium fuel but still more than for LWRs.

The plutonium production of ACRs will be lower than that of current PHWRs due to the higher fuel burnup. However, the ACR retains the on-power refueling feature of its predecessors.

3.1.3 High Temperature Gas Reactor

The concept of the high temperature gas reactor (HTGR), which has been evolving since the 1940's, has been revived in recent years as recent modular designs featuring TRISO fuel and a Brayton power cycle (direct or indirect) have shown promise for enhanced safety and attractive economics. Today, modular HTGRs are considered as leading candidates for the next generation of nuclear power plants. Two designs have emerged: the pebble bed type and the prismatic block type.

The essential difference between the two main HTGR designs lies in the design of the fuel elements. In both cases, the fuel is the form of small kernels known as TRISO particles, made up of fissile (and possibly also fertile) material at the core, surrounded by a low-density buffer zone (to accommodate fission products), an inner pyrocarbon coating, a silicon carbide coating, and an outer pyrocarbon coating. The coatings provide a corrosion resistant pressure vessel and a barrier for fission products. The diameter of TRISO particles varies from about 650 to 850 microns [8]. In the prismatic design, the TRISO particles are imbedded in a graphite matrix to form fuel compacts (typically cylindrical with dimensions of few centimeters), and then inserted in prismatic graphite fuel elements, which typically contain several thousand fuel compacts. In the pebble bed design, the particles are also imbedded in a graphite matrix, but in the form of spherical pebbles. The core contains hundreds of thousands of these pebbles, which move slowly through the core during reactor operation.

Reactors of both pebble bed and prismatic types have been built and operated, with the thorium high temperature reactor (THTR) representing the first category, and the Fort St. Vrain gas reactor representing the second[1]. These reactors have since been shut down[2], but other efforts have been undertaken. In Japan, the high temperature test reactor (HTTR), a 30 MWth prismatic reactor, went critical in 1998. In China, a 10 MWth pebble bed reactor known as HTR-10 went critical in late 2000. In addition, two major projects that aim to bring HTGR technology to commercial deployment are currently under way. The South African utility Eskom is leading the pebble bed modular reactor (PBMR) project, while the gas turbine modular helium reactor (GT-MHR) is being developed in a venture project led by General Atomics [9]. The PBMR and GT-MHR designs rely on a direct Brayton cycle[3].

High thermal efficiency and passive safety

One of the attractive features of HTGRs is their high thermal efficiency, due to the high core outlet temperature of the helium coolant and the use of a Brayton power cycle. Thermal efficiency for these reactors is generally around 45%. In addition, modular

[1] Several HTGRs were built and operated, but the THTR and Fort St-Vrain reactors are commercially the most important projects.
[2] The German THTR encountered licensing and funding problems while the Fort St. Vrain unit was plagued by helium circulator bearing problems.
[3] MIT is investigating a PBMR concept that relies on an indirect Brayton cycle.

HTGRs have particularly good safety characteristics due to their small core size and low power density, which allow passive removal of decay heat in the event of active heat removal system failure. The passive reactor cavity cooling system (RCCS) can carry heat away from the core by natural convection, but simulations have shown (Figure 3.2) that even if the RCCS fails, conduction and radiation are sufficient to prevent the peak core temperatures from reaching the limit of 1600°C limit beyond which TRISO particles may lose their integrity. Figure 3.3 shows the basis for the 1600°C temperature limit: beyond this point, particle failures start to occur in significant numbers and fission products can be released from the fuel, even though meltdown does not occur. This 1600°C fuel temperature limit can become an attractive basis for fuel performance once a fuel manufacturing and inspection program for continuous or batch production can be developed, qualified, and confirmed as sustainable.

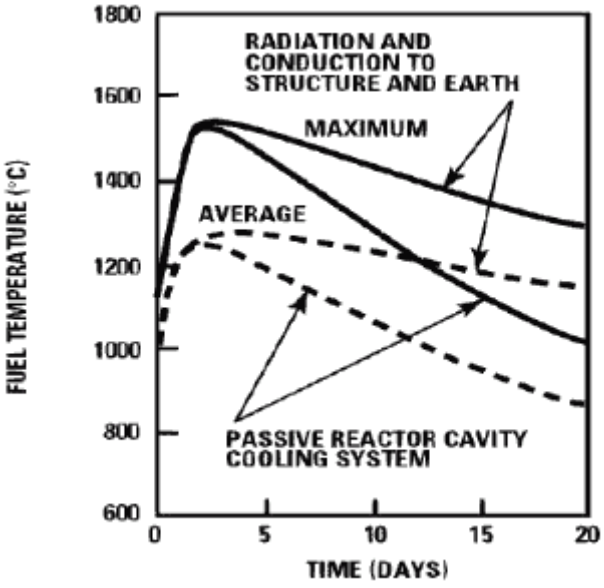

Figure 3.2. Temperature response to loss of coolant (GT-MHR) [8]

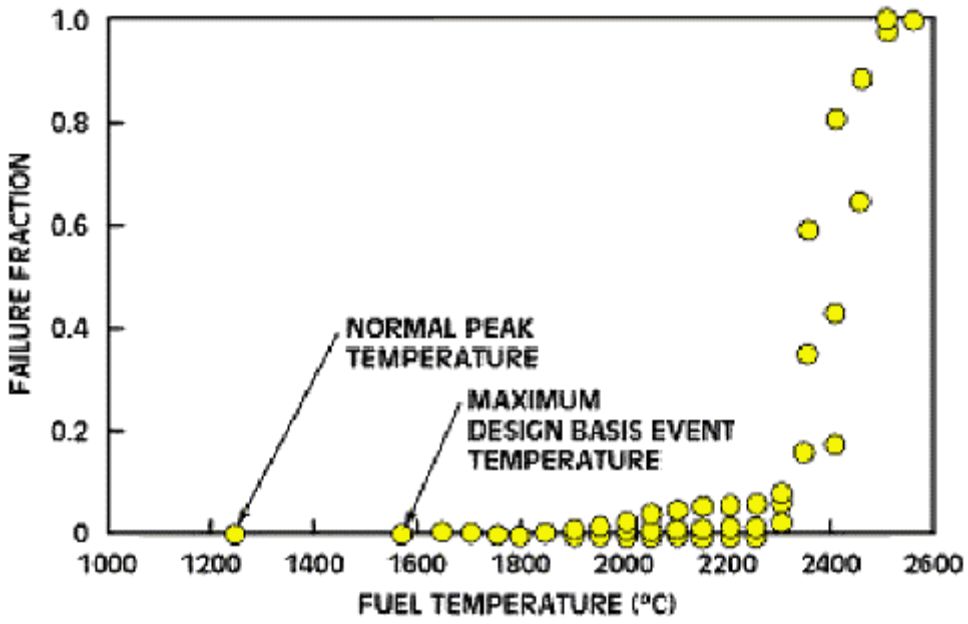

Figure 3.3. TRISO fuel performance as a function of temperature [8]

Mass Flow Analysis

To evaluate the uranium consumption and spent fuel discharge of the GT-MHR and PBMR, standard assumptions are used regarding thermal efficiency, burnup, and enrichment [10]. Results are shown in Table 3.11. The table also shows the volume of the spent fuel discharged, which is significantly higher for HTGRs than for LWRs because HTGR fuel is very dilute.

For the GT-MHR, the hexagonal fuel elements are 793 mm long and measure 360 mm across flats, giving a volume of 0.2670 m^3 [8]. The core contains 1,020 fuel assemblies and 4.57 MTHM [11]. Therefore, the specific volume of the fuel is 59.60 m^3/MTHM and the volume of spent fuel discharge is 381.4 m^3/GWe·y.

For the PBMR, each pebble has a diameter of 60 mm and a heavy metal content of 9 grams [12], for a specific volume of 12.57 m^3/MTHM. The volume of spent fuel discharge is therefore 124.4 m^3/GWe·y.

For a burnup of 50 GWd/MTHM, a typical LWR discharges 22.1 MTHM/GWe·y. Since a fuel assembly contains approximately 0.5 MTHM and has a volume of about 0.2 m^3, the volume of spent fuel discharged is by this type of reactor is 8.84 MTHM/GWe·y. The volume discharged by the GT-MHR is 43 times higher. For the PBMR, it is 14 times higher.

Table 3.11. Uranium consumption and spent fuel discharge for the GT-MHR and PBMR

	GT-MHR	PBMR
Thermal efficiency	47	46
Enrichment (% U-235)	15.5	8
Burnup (GWd/MTHM)	121	80
Uranium consumption (MTU/GWe·y)	237.4	185.8
Spent Fuel discharge (MTHM/GWe·y)	6.4	9.9
Spent fuel volume (m^3/GWe·y)	381.4	124.4
Fission product discharge (kg/GWe·y)	774.4	792

Because the thermal efficiency of HTGRs is roughly 1.5 times higher than for LWRs, the quantity of fission products generated per unit of electricity produced will be lower by a factor of 1.5 also. This means that storage and disposal requirements, which largely depend on fission product decay heat, will be lessened by about 50% for HTGRs as compared to LWRs. The high burnup achieved by HTGR fuel means that, compared to current LWRs, the transuranic production per unit of electricity produced will be diminished and the isotopic composition of the plutonium in the spent fuel will be degraded by a significant concentration of Pu-238. However, as noted earlier, a discharge burnup in the range of 100 GWd/MTHM is deemed possible in LWRs with major improvements over current technology.

Proliferation resistance

The average discharge burnup of PBMR and GT-MHR fuel is roughly twice as high as for current LWRs. Hence, the amount of plutonium produced per energy generated is expected to be lower and its isotopic composition will be of lower grade. The plutonium and transuranic discharge of the GT-MHR are shown in Table 3.12. The plutonium discharge is given in [13], and the transuranic discharge is obtained assuming that plutonium accounts for about 85% of transuranics in spent fuel, as in [14].

Table 3.12. Transuranic and plutonium discharge of the GT-MHR

Transuranic discharge (kg/GWe·y)	128
Plutonium discharge (kg/GWe·y)	109

Table 3.12 shows that the transuranic and plutonium discharge of the GT-MHR is significantly lower than that of the LWR, even with high burnup. Furthermore, no technology currently exists for extracting the heavy metals from the graphite fuel elements and TRISO particles. It should be noted, however, that there are no obvious obstacles to developing this technology (e.g. a process whereby fuel elements are ground to a powder and then exposed to a high temperature in order to burn off the carbon).

The fact that the PBMR uses online refueling raises proliferation concerns. More specifically, 4880 spheres are discharged from the core per effective full power day. The majority are recirculated to the top of the core, while 370 are removed and replaced by fresh pebbles [11]. The fact that fuel elements can be removed from the core without reactor shutdown facilitates their illicit retrieval, and the enormous number of pebbles that needs to be handled every day makes the removal of a small number of them difficult

to notice. However, the diluteness of the fuel implies that a large number of pebbles would have to be diverted in order to recover enough plutonium to make a nuclear weapon. Each pebble contains 9 grams of heavy metal. Assuming that this material contains 2% plutonium at discharge, almost 28,000 spent pebbles are needed to recover 5 kg of plutonium. Therefore, the diversion of a small number of pebbles poses no immediate danger. But a few pebbles could be sufficient to conduct research on a process for extracting plutonium from irradiated PBMR fuel.

Economics

Developers of the GT-MHR and PBMR often propound the very advantageous economics of these reactors. Very short construction times, on the order of 24 months, and low overnight construction costs, around $1000/kWe, are cited to support these statements [9]. While such claims remain to be proven, the modularity of current HTGR designs will allow factory assembly of major components and may indeed significantly reduce capital cost and construction time. Furthermore, the higher thermal efficiency of HTGRS would tend to make their construction cost lower on a per kWe basis. In this analysis, we consider two possibilities: first, we assume an overnight construction cost of $1,700/kWe; second, we assume that the overnight construction cost per kWe varies inversely with thermal efficiency: since the thermal efficiency of HTGRs is higher than that of LWRs by a factor of roughly 1.4, we assume a reduction in the overnight construction cost by a factor of 1.4, giving $1,200/kWe. In addition, the cost of HTGR fuel fabrication is assumed equal to that of conventional UOX fuel.

Note that the cost of storage and disposal, 1365 $/kgHM for the GT-MHR and 885 $/kgHM for the PBMR, is again obtained by equivalence with the current 1 mill/kWh fee paid to DOE. This fee does not take into account the fact that HTGRs have a significantly higher thermal efficiency than either LWRs or PHWRs and therefore generate fewer fission products per unit of electricity generated. In addition, HTGRs produce fewer transuranics per unit of electricity generated, which again is not rewarded by the 1 mill/kWh fee. Thus, if a fee structure rewarding higher thermal efficiency and burnup was implemented, it is reasonable to expect that the cost of spent fuel storage and disposal for HTGRs would be lower than the values that were assumed here. However, if increased spent fuel volume was penalized, this would be detrimental to the cost of HTGR spent fuel storage, transportation, and disposal.

Assumptions used for the calculation of the CEI for the GT-MHR and the PBMR are listed in Table 3.13.

Table 3.13 CEI for GT-MHR and PBMR (mills/kWh)

	GT-MHR	PBMR
CEI_{FC}	4.11	2.15
CEI_{R}	30.18 / 21.30	30.18 / 21.30
CEI	34.29 / 25.42	33.62 / 24.74

It is clear from the previous results that if the HTGR's high thermal efficiency translates into a lower construction cost per kWe (in other words, if the HTGR can be designed such that its construction cost per unit of thermal capacity is comparable to that of an LWR), then it has the potential to deliver electricity at a much lower cost than conventional LWRs.

3.1.4 Seed and Blanket Thorium Reactor

Thorium is attractive because the U-233 bred from Th-232 produces more neutrons per thermal neutron absorbed than all other fissile isotopes. The problem with thorium is that it must be supplemented with a fissile fuel, which sustains the nuclear reaction while investing neutrons in the fertile Th-232. The reactivity buildup in the thorium (as U-233 is bred) is quite slow and it is only when the burnup has reached 70-80 GWd/kg that the U-233 generates enough energy to compensate for all the neutrons absorbed. To reach such a high burnup, the required initial content of fissile material is so high that it "crowds out" the thorium. Indeed, for homogeneous Th/U fuel, the U-233 power share over a cycle does not exceed 22% (assuming that the uranium cannot be enriched to more than 20% U-235 for proliferation resistance) [15]. There are 2 solutions to this problem:

1. Closed cycle: stop at lower burnup and separate the U-233 from the fuel and use it as fissile material in fresh fuel;
2. Once-through cycle: divide the reactor core in 2 regions: the seed, which contains the fissile material, and the blanket, which contains the thorium. Different fuel management schemes can be used for the 2 regions. In this way, the blanket can remain in the reactor for a long period of time, allowing significant breeding and in-situ burning of U-233, while the seed can be replaced more frequently to maintain reactivity. This approach is known as "seed and blanket" or "heterogeneous".

Two prominent examples of seed and blanket approaches for using thorium fuel in conventional LWRs are the Radkowsky concept [17, 18, 19] (often designated as RTF, for Radkowsky Thorium Fuel, or RTR, for Radkowsky Thorium Reactor) and the Whole Assembly Seed and Blanket (WASB) concept [16]. In the RTR concept, all fuel assemblies contain both seed and blanket fuel pins (Figure 3.4 [16]). These heterogeneous assemblies are known as seed-blanket units (SBU). On the other hand, WASB calls for two types of homogenous assemblies: seed assemblies containing only seed fuel pins, and blanket assemblies containing only blanket fuel pins (Figure 3.4 [16]). Another difference between the two concepts is that RTR is geared toward the VVER, while WASB is based on a Westinghouse 1150 MWe PWR. A positive feature of these concepts is that they are fully compatible with existing power plants. Implementation will require only minor plant hardware modifications, all safety and operational parameters of existing power plants will be preserved, and the fuel design will be based mainly on existing fuel technology. In this section, we focus on the WASB concept, but the RTR and the WASB are based on the same principle and have similar characteristics.

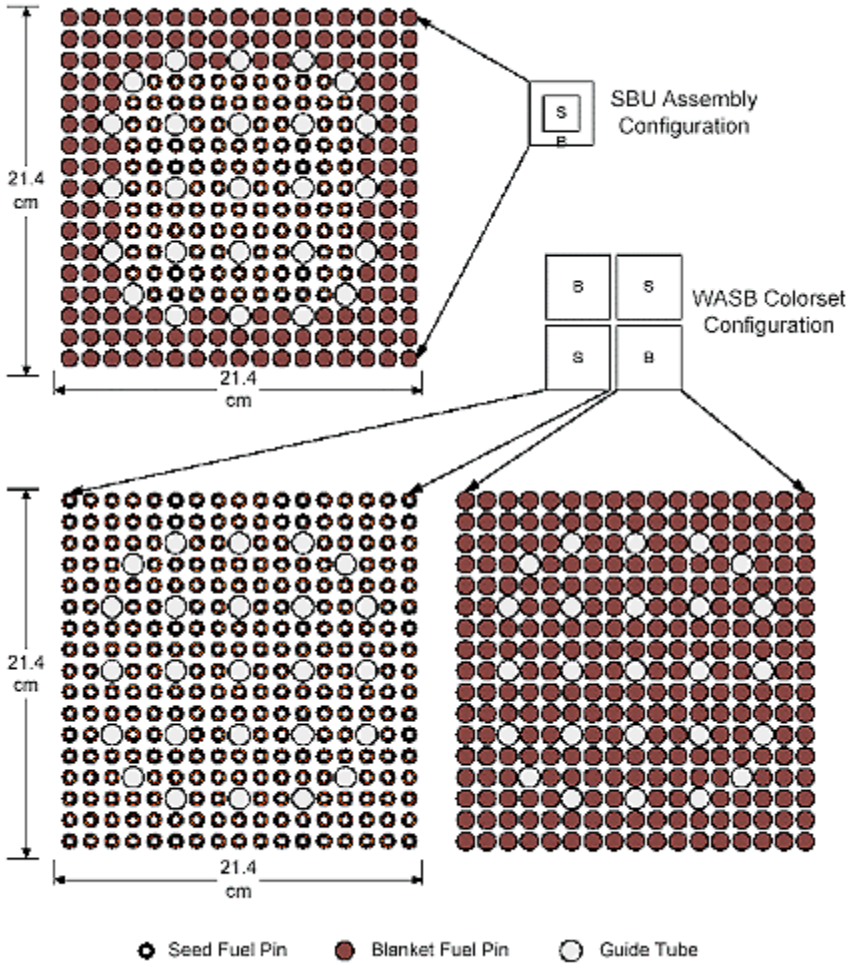

Figure 3.4. SBU and WASB assembly configuration

Mass Flow Analysis

In the WASB concept, the seed pins contain uranium oxide enriched to 20% U-235. The blanket pins contain 87% thorium oxide and 13% uranium oxide enriched to 10% U-235. The uranium in the blanket is added to provide power during U-233 buildup and to denature the U-233 for proliferation resistance. Under the proposed fuel management scheme for WASB, one third of the seed assemblies (7,703 kgHM) are discharged and replaced at every cycle, while the blanket (58,997 kgHM) is discharged and replaced only once every 9 cycles. The duration of a cycle is 18 months, over which the reactor generates 1,570 GWe·y (the assumed capacity factor is 91%). Thus the discharge rate is 5.01 MTHM/GWe·y for the seed and 4.26 MTHM/GWe·y for the blanket[1]. The uranium

[1] Note that the figures given for the spent fuel discharges assume a thermal efficiency of 33.7%. In order to establish a fair comparison with the other fuel cycles based on LWRs, a small correction has been made in order to take this into account: the mass of spent fuel was increased by the factor 33.7/33.

requirement, obtained using Eq. 3, is 240.2 MTU/GWe·y for the seed and 13.1 MTU/GWe·y for the blanket, for a total of 253.3 MTU/GWe·y.

The characteristics of the spent fuel for WASB are given in Table 3.14[1]. The discharge rate for plutonium, transuranics, and fission products, given in Table 3.15, is obtained by multiplying the spent fuel discharge rate by the spent fuel fractions shown in Table 3.14. Table 3.15 shows that the transuranic and plutonium discharge rates for WASB are about 3 times lower than for a conventional once-through fuel cycle.

Table 3.14. Spent fuel composition for WASB

	Seed	Blanket
Transuranics	1.97%	0.51%
Plutonium	1.56%	0.45%
Fission products	14.5%	8.8%

Table 3.15. Uranium consumption and spent fuel discharge for WASB

	Seed	Blanket	Total
Uranium consumption (MTU/GWe·y)	240.2	13.1	253.3
Spent fuel discharge (MTHM/GWe·y)	5.01	4.26	9.28
Transuranic discharge (kg/GWe·y)	98.8	22.1	120.8
Plutonium discharge (kg/GWe·y)	78.1	19.3	97.4
Fission product discharge (kg/GWe·y)	727	375	1102

Proliferation resistance

High proliferation resistance is the most important argument used in favor of seed and blanket proposals. Indeed, the use of thorium in the blanket and the high burnup reached by the fuel combine to produce spent fuel with a much lower plutonium content and significantly degraded isotopics.

The plutonium discharge for WASB is 97.4 kg/GWe·y., compared with 294 kgPu/GWe·y for a conventional PWR at 50 GWd/MTHM. This difference is not very significant when one considers that only ~10 kg are needed to make a bomb. However, the plutonium from spent seed and blanket fuel is less attractive than the plutonium from conventional spent fuel because of its higher heat generation rate and spontaneous fission source (SFS). The SFS and decay heat for plutonium of various grades is presented in Table 3.16 along with the critical mass.

[1] From [16]. Plutonium content determined from Table 6.3. Minor actinide content is approximated from Table 2.5.

Table 3.16. Characteristics of various Pu grades

	Weapon grade	PWR	WASB-seed	WASB-blanket
Critical mass (kg)	11.8	22.1	21.4	26.7
Decay heat (W/kg)	2.37	18.20	69.29	57.30
Critical mass decay heat (W)	27.97	402.76	1485.63	1530.18
SFS ($kg^{-1} \cdot s^{-1}$)	$6.05 \cdot 10^4$	$4.26 \cdot 10^5$	$5.52 \cdot 10^5$	$6.77 \cdot 10^5$
Critical mass SFS (s^{-1})	$7.14 \cdot 10^5$	$9.43 \cdot 10^6$	$1.18 \cdot 10^7$	$1.81 \cdot 10^7$

The above table shows that the WASB spent fuel from the seed is less proliferation resistant than the fuel from the blanket. Comparing WASB seed spent fuel to PWR spent fuel, it is seen that critical mass decay heat for WASB plutonium is 3 times higher. This could require that heat removal measures be incorporated in the bomb design to prevent metallurgical phase transition in the plutonium and/or degradation of the explosive. Expert advice would be required to evaluate how much this complicates bomb design. Table 3.16 also shows that the critical mass SFS for the WASB is 25% higher than for the PWR, a rather modest difference. Finally, it must be noted that the uranium in fresh WASB seed fuel is enriched to 20% U-235. Uranium having an enrichment level of 20% U-235 or more is considered to be weapons usable. Therefore, while the WASB concept alleviates problems associated with plutonium in spent fuel, it creates new concerns on the front-end of the fuel cycle.

Economics

Using Eq. 1 with the assumptions listed in Table 2.1, CEI = 35.52 mills/kWh, with CEI_R = 30.18 and CEI_{FC} = 5.34 mills/kWh. Thus the fuel cycle cost for the WASB is slightly higher than for the conventional uranium once-through cycle. However, the cost of waste disposal is assumed to be 1 mill/kWh and therefore does not reward WASB for its lower spent fuel mass and transuranic discharge rate. The WASB option could conceivably become attractive under a different waste disposal fee.

3.1.5 Breed and Burn in Fast Reactor

It is widely believed that nuclear fuel cycles that allow significantly improved uranium utilization over the current LWR once-through cycle must involve reprocessing. The breed and burn option, however, makes it possible to achieve high uranium utilization without any fuel reprocessing.

Breed and burn (B&B) can be defined as the use of fast reactors in which equilibrium reload fuel has lower enrichment than required to sustain criticality, but which breed sufficient plutonium in-situ to sustain criticality until high discharge burnup is achieved (on the order of 150 GWd/MTHM). The spent fuel discharged from the fast reactors is not reprocessed.

Investigation of B&B has been rather limited, with the few published papers on the topic focusing mostly on the physics aspects of the concept, leaving significant practical problems unresolved. G.I. Toshinsky [20] has investigated a lead-bismuth cooled fast reactor for B&B, while Ryu and Sekimoto [21] studied a pebble bed fast reactor cooled by helium. In addition, fast-mixed-spectrum reactor concepts, where the core contains both hard spectrum and moderated regions, have been considered [22]. A current effort involving MIT, INEEL, and ANL [23] will attempt to determine whether B&B is feasible in practical plant designs.

Breed and Burn concepts

Breed and Burn schemes currently receiving the most attention involve a gas coolant and carbide or metal fuel; in this way, moderation by the coolant and fuel material is minimized, allowing for an ultra-hard neutron spectrum and high breeding ratio; high power densities are required to accelerate fuel throughput and minimize the time required to reach equilibrium. Indeed, enriched uranium will be required for the initial core and for the first reload cores until sufficient plutonium has been bred in the reactor. The shorter the time to reach equilibrium, the sooner the uranium savings will come into effect. A consequence of requiring a high power density with a gas coolant is that post-LOCA decay heat removal would have to be handled by active systems. Hence, the emphasis on passive safety that prevails today is difficult to reconcile with the basic requirements of a good B&B reactor.

Another important feature of B&B is that the discharged spent fuel contains roughly 10% plutonium with a fairly clean isotopic composition. Because the fuel reaches a very high burnup and is extremely radioactive, the plutonium would probably be difficult to retrieve by illegitimate means. However, this issue can still raise proliferation concerns. To alleviate this problem without having recourse to aqueous or pyroprocessing, the B&B spent fuel could be recycled with the AIROX[1] process and burned in light water reactors (LWR). AIROX is a “dry” process whereby spent fuel is not dissolved but simply conditioned through subsequent phases of oxidation and reduction. It does not involve separation of any heavy elements (only volatile fission products are removed).

[1] Atomics International Reduction Oxidation.

Although it has been developed specifically for oxide fuels, there are no obvious reasons why it could not be extended to other fuels, but further research in this area is needed. The irradiation of the recycled fuel in LWRs would reduce the amount of plutonium, degrade its isotopic composition, and allow further gains in uranium utilization efficiency.

Uranium consumption and spent fuel discharge

Although B&B is at a very early stage of exploration, the natural uranium consumption and spent fuel discharges for such a fuel cycle can be estimated. If a burnup of 140 GWd/MTHM and a thermal efficiency of 40% are assumed, we find that the fuel requirement is 6.52 MTHM/GWe·y. The fuel contains approximately 10% plutonium and 0.5% minor actinides, so the discharge rate is 652 kg/GWe·y for plutonium and 33 kg/GWe·y for minor actinides, giving a transuranic discharge rate of 685 kg/GWe·y.

The natural uranium consumption depends on the enrichment required. In the best of cases, B&B cores will be able to run on natural uranium (0.711% U-235) once equilibrium is reached. Whether such performance can be achieved in practice is uncertain at this time. Therefore, it is useful to consider higher enrichments as well. Table 3.17 shows uranium requirements for enrichments of 2%, 3%, and 4%. It is clear from these results that the enrichment has a very important effect on uranium consumption: there is an order of magnitude difference between natural and 4% enriched uranium due to the amount of uranium ore that must be processed to produce enriched uranium.

If it is assumed that the B&B spent fuel is recycled with AIROX processing and irradiated in LWRs, uranium consumption is even lower. The B&B spent fuel contains ~10% Pu, whereas the typical content of MOX fuel is 8%. However, the AIROX-recycled fuel still contains an important quantity of fission products, which have a negative impact on reactivity. It will be assumed here that the recycled fuel is not diluted with any uranium, so that the AIROX-recycled fuel loaded in LWRs contains 10% Pu. In other words, it is assumed that the 2% excess plutonium compared to regular MOX fuel is needed to sustain reactivity to a discharge burnup of 50 GWd/MTHM. In a nuclear park with B&B fast reactors and LWRs running on AIROX-recycled fuel, the fraction of electricity production based on B&B versus AIROX-recycled fuel is determined as follows:

Mass of B&B fuel discharged = Mass of AIROX-recycled fuel loaded in LWRs

$$\left(\frac{P}{\eta_{th} \cdot B_d}\right)_{B\&B} = \left(\frac{P}{\eta_{th} \cdot B_d}\right)_{AIROX}$$

where: P = installed capacity (GWe)
η_{th} = thermal efficiency
B_d = discharge burnup (GWd/MTHM)

$$\frac{P_{B\&B}}{0.4 \cdot 140} = \frac{P_{AIROX}}{0.33 \cdot 50}$$

$$\frac{P_{B\&B}}{P_{AIROX}} = 3.39$$

For a total capacity of 1 GWe, 772.5 MWe is based on B&B and 227.5 MWe is based on AIROX recycled fuel. The uranium consumption and mass of spent fuel in this case are shown in Table 3.17 along with the case of B&B without recycling. As can be seen, for the AIROX recycled fuel a modest but not insignificant improvement in uranium utilization is achieved. The mass of spent fuel discharged is roughly 20% lower; however, the quantity of fission products is higher due to the lower thermal efficiency of the LWRs used to irradiate the AIROX-recycled fuel[1]. The real improvement on the waste side is the degraded isotopic composition and the reduced quantity of the plutonium in the spent fuel[2].

Table 3.17. Uranium consumption and spent fuel discharge for B&B, with and without AIROX recycling

	B&B	B&B with AIROX
Uranium consumption MTU/GWe·y (reduction factor[a]) ▪ no enrichment ▪ 2% enriched ▪ 3% enriched ▪ 4% enriched	 6.52 (34.8) 27.0 (8.4) 42.8 (5.3) 58.7 (3.9)	 5.03 (45.0) 20.8 (10.9) 33.1 (6.9) 45.3 (5.0)
Spent Fuel MTHM/GWe·y (reduction factor[a])	 6.52 (3.4)	 5.03 (4.4)

a. Compared to a once-through LWR (4.51% enrichment, 50 GWd/MTHM burnup, 33% thermal efficiency) consuming 226.5 MTU/GWe·y and discharging 22.1 MTHM/GWe·y of spent fuel.

Economics

The fuel cycle cost for B&B is expected to be very low because reprocessing is not necessary and because uranium consumption is low. Furthermore, enrichment costs are averted if equilibrium reloads can be done with natural uranium. Table 3.18 shows a detailed calculation of the fuel cycle cost at equilibrium for the case of natural uranium and 4% enriched reloads.

[1] The thermal power for B&B FRs having a capacity of 1 GWe is 1/0.4 = 2.5 GW. In the case where the spent fuel from the FRs is AIROX recycled for irradiation in LWRs, the thermal power for FRs and LWRs having a combined capacity of 1 GWe is 0.7725/0.4 + 0.2275/0.33 = 2.62 GW. Production of fission products is proportional to thermal power; therefore, a fleet that relies only on B&B fast reactors would generate about 5% less fission products than a fleet composed of B&B fast reactors along with some LWRs running on AIROX recycled fuel.

[2] No data is available at this time to quantify these benefits.

Table 3.18. Fuel cycle cost for B&B

Step	Mass flow	Unit cost	Lead time[a]	Direct charge	Carrying charge[b]
Natural uranium reloads					
Ore purchase	1 kg	30 $/kg	4	30	12
Enrichment[c]	0	100 $/kg SWU	3	0	0
Fabrication	1 kgHM	450 $/kgHM	2.5	450	112.5
Storage and disposal[d]	1 kgHM	1120 $/kgHM	-4	1120	-448
Fuel cycle cost: 1277 $/kgHM (0.95 mill/kWh)					
4% enriched reloads					
Ore purchase	9.0 kg	30 $/kg	4	270	108
Enrichment[c]	5.05 kg SWU	100 $/kg SWU	3	505	152
Fabrication	1 kgHM	450 $/kgHM	2.5	450	112.5
Storage and disposal[d]	1 kgHM	1120 $/kgHM	-4	1120	-448
Fuel cycle cost: 2269 $/kgHM (1.7 mill/kWh)					

a. Lead time is based on an irradiation time of 4 years. For fuel irradiated to 140 MWd/kgHM at a specific power of 100 kW/kgHM in a reactor running at 90% capacity factor, irradiation time is 4.26 years. We use 4 years for the purposes of this calculation.
b. The carrying charge factor is 0.10/yr.
c. SWU requirements are approximated as follows: kg SWU/kg of product = $2.07 \cdot x_p - 3.23$, where x_p is the enrichment of the product.
d. The cost of storage and disposal for 50MWd/kgHM UOX fuel discharged from LWRs is $400/kgHM. For B&B fuel, the cost would be higher due to the higher burnup (higher fission product content). Assuming that the storage and disposal cost is roughly proportional to burnup, we get $140/50 \cdot 400$ = $1120/kgHM.

Table 3.18 clearly shows very low fuel cycle costs for the B&B option. For comparison, the fuel cycle cost for the current once-through cycle is typically close to 5 mills/kWh. It is important to recall that the fuel cycle cost is only a fraction of the total cost of electricity generation. Because it relies on deployment of fast reactors, B&B is not likely to be more economic than the current LWR once-through option any time soon. However, compared with other fuel cycles that require fast reactors and reprocessing, B&B has a clear economic advantage.

Assuming natural uranium reloads, the CEI for this option is determined using the assumptions listed in Table 2.1. The CEI for Breed and Burn is 38.47 mills/kWh, with CEI_R = 37.28 mills/kWh and CEI_{FC} = 1.19 mills/kWh.

3.2 Closed Fuel Cycles

3.2.1 Plutonium Recycle in Light Water Reactor

The plutonium present in spent fuel can be recycled and used as fissile material in new nuclear fuel. Recycled plutonium is mixed with natural or depleted uranium to make MOX (Mixed OXide) fuel. Because of its degraded isotopic composition, which makes both recycling handling and reactor operation difficult, the plutonium in spent MOX is not recycled again[1]. Also, because of its detrimental effect on reactor control, the cores of most current reactors cannot be loaded entirely with MOX fuel assemblies. Typically, only about 1/3 of the fuel inside a core is MOX[2]. In this section, we will consider a fuel cycle where all spent UOX discharged from reactors is reprocessed for plutonium recycle but none of the spent MOX is reprocessed.

Mass Flow Analysis

The contents of freshly fabricated and irradiated MOX fuel are shown in Table 3.19 [24], for a burnup of 50 GWd/MTHM.

Table 3.19. Contents of fresh and irradiated MOX fuel (50 GWd/MTHM)

Component	Fresh MOX (%)	Spent MOX (%)
Uranium	91.9	88.79
Transuranic elements	-	6.06
Plutonium	8.1	5.52
Minor actinides	-	0.54
Fission products	-	5.15

Assuming that all spent UOX is reprocessed for plutonium recycle, the ratio of UOX based to MOX based capacity is determined by requiring that the amount of plutonium discharged in spent UOX be equal to the amount of plutonium required for MOX fabrication. Eq. 2 gives the mass of fuel irradiated per GWe·y. Multiplying this amount by the power and the capacity factor gives the mass of fuel per year:

Eq. 5:
$$M = \frac{P \cdot 365 \cdot CF}{B_d \cdot \eta_{th}}$$

where: m: mass of fuel (MTHM/y)

P: Power (GWe)

CF: capacity factor

[1] More recycles could eventually be carried out. However, no more than one or two additional passes could be done with the current technology.

[2] Advanced Light Water Reactors (ALWR) are designed to accommodate a full core MOX loading. The ABWR, AP-600, AP-1000, and System 80+ can all operate on a full MOX core.

Recalling that the plutonium content of spent UOX is 1.33% and that fresh MOX contains 8.1% plutonium, and allowing for 0.1% losses in plutonium recovery, we find:

$$\text{Pu in spent UOX} \cdot 0.999 = \text{Pu needed for fresh MOX}$$

$$M_{UOX} \cdot 1.33 \cdot 0.999 = M_{MOX} \cdot 8.1$$

$$\frac{P_{UOX} \cdot 365 \cdot CF}{B_d \cdot \eta_{th}} \cdot 1.33 \cdot 0.999 = \frac{P_{MOX} \cdot 365 \cdot CF}{B_d \cdot \eta_{th}} \cdot 8.1$$

$$P_{UOX} \cdot 1.33 \cdot 0.999 = P_{MOX} \cdot 8.1$$

$$\frac{P_{UOX}}{P_{MOX}} = \frac{8.1}{1.33 \cdot 0.999} = 6.10$$

Therefore, in the fuel cycle under consideration, the fraction of power derived from UOX is 85.91%, while 14.09% is derived from MOX.

The mass of fuel irradiated per GWe·y is obtained using Eq. 2. In a fuel cycle with more than one type of fuel, the mass of irradiated fuel of a given type can be obtained by multiplying Eq. 2 by the fraction of power supplied by that fuel:

Eq. 6:
$$m_x = \frac{365}{B_d \cdot \eta_{th}} \cdot f_x$$

where: m_x: mass of fuel x (MTHM/ GWe·y)

f_x: fraction of power supplied by fuel x

Using Eq. 6, we find m_{UOX} = 19 MTHM/GWe·y and m_{MOX} = 3.12 MTHM/GWe·y. Using Eq. 2, the mass of natural uranium required is 194.85 MTU/GWe·y. Since MOX is composed of 91.9% plutonium and 8.1% depleted uranium, MOX fabrication requires 2.86 MT/GWe·y of depleted uranium and 252 kg/GWe·y of plutonium.

The waste produced by the fuel cycle under consideration is composed of high-level waste (HLW) and separated irradiated uranium from reprocessing operations and spent MOX fuel. The reprocessing HLW is composed of all the fission products and minor actinides in the spent UOX. In addition, due to limitations in the PUREX (Plutonium URanium EXtraction) process used to reprocess the spent UOX, 0.1% of the uranium and plutonium cannot be recovered and ends up in the HLW. The contents of HLW are therefore obtained by multiplying the mass of spent UOX by the content fractions show in Table 3.20. The waste discharge in HLW is 979 kg/GWe·y of fission products, 22.8 kg/GWe·y of minor actinides, and 0.252 kg/GWe·y of plutonium.

Table 3.20. Content of HLW from PUREX

Component	Mass (MT per MTHM reprocessed)
Fission products	5.15·10-2
Minor actinides	$1.2 \cdot 10^{-3}$
Uranium	$9.34 \cdot 10^{-4}$
Plutonium	$1.33 \cdot 10^{-5}$

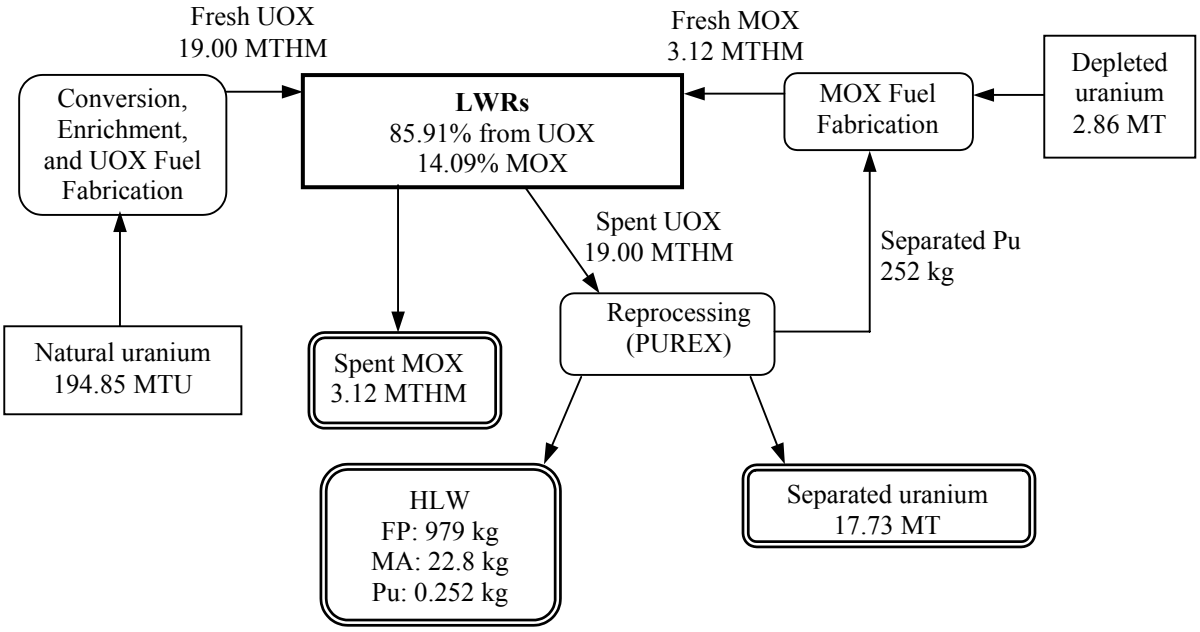

Figure 3.5. Plutonium single recycle in LWRs
(all quantities are per GWe·y)

Table 3.21 compares uranium consumption for plutonium recycle in LWRs and the once-through fuel cycle, and shows that savings from Pu recycle are a modest 16.5%.

Table 3.21. Uranium savings from plutonium recycle in LWRs
(all quantities are per GWe·y)

	Pu recycle	Once-through	Savings
Uranium consumption (MTU)	194.85	226.5	16.5%

Table 3.22 summarizes waste discharges for plutonium recycle in LWRs and shows the same data for the once-through cycle for comparison. Recycling plutonium is seen to reduce the amount of this element going to waste by about 40%. However, the production of minor actinides is higher because MOX fuel, which initially contains a substantial amount of various plutonium isotopes, leads to a higher production of minor actinides than UOX fuel for a given irradiation level.

Table 3.22. Waste discharge for plutonium recycle in LWRs
(all quantities are per GWe·y)

	Plutonium recycle			Once-through
	Spent MOX	HLW	Total	
Spent Fuel (MTHM)	3.12	-	3.12	22.1
Plutonium (kg)	172.2	0.252	172.5	294
Minor actinides (kg)	16.9	22.8	39.7	26
Transuranic elements (kg)	189.1	23.1	212.2	320

Proliferation

The PUREX process raises serious proliferation concerns because it produces separated plutonium. The conditions which currently prevail are particularly alarming because the rate at which separated plutonium is produced by reprocessing operations exceeds the rate at which it is consumed for MOX fabrication, leading to an accumulation of plutonium. About 22 tonnes of plutonium is separated by reprocessing plants in the OECD each year, while only 8 to 10 tonnes are used for MOX fabrication [25]. To date, about 200 tonnes of plutonium have been accumulated [26]. This illustrates a fundamental problem with balanced fuel cycles requiring several different types of fuels: a specific ratio between the various fuels is difficult to achieve in practice, and this may lead to some undesirable consequences – in the present case, the accumulation of separated plutonium. Moreover, even for the balanced fuel cycle shown in Figure 3.5, a minimum separated plutonium inventory is required. The inventory at any given stage in the nuclear fuel cycle is obtained as the product of the mass flow and the process time:

Eq. 7: $$I = M \cdot t$$

where: I = Inventory (MT or kg)

M = Mass flow (MT/yr or kg/yr)

t = Process time (yr)

On a per GWe basis:

Eq. 8: $$i = m \cdot t$$

where: i = inventory (MT/GWe or kg/GWe)

m = Mass flow (MT/GWe·y or kg/GWe·y)

t = Process time (yr)

Assuming a separated plutonium storage time of 6 months [27], 126 kg/GWe of plutonium will be held in storage at any given time, since the total plutonium mass flow is 252 kg/GWe·y.

The mass of plutonium and transuranics discharged in MOX fuel is obtained by multiplying the spent MOX discharge rate (3.12 MTHM/GWe·y) by the plutonium and transuranics content of spent MOX as shown in Table 3.19. The discharge rates of plutonium and transuranics are 172 and 189 kg/GWe·y respectively. The total transuranic discharge rate for the fuel cycle is 212 kg/GWe·y (note that the plutonium losses from reprocessing operations, 0.252 kg/GWe·y, are negligible compared to the plutonium discharged in spent MOX).

Economics

The economics of plutonium recycling in LWRs is the subject of much debate. The issue is difficult to resolve definitely because of the great uncertainties involved in fuel cycle cost calculations. Appendix C shows that this calculation is very sensitive to assumptions regarding the unit cost of various fuel cycle steps. With optimistic assumptions regarding the costs of the various operations related to reprocessing, plutonium recycle can be shown to be competitive with the once-through option. With more conservative assumptions, plutonium recycling is more expensive by a factor of about 4.5. Thus there seems to be no compelling economic case for plutonium recycling. However, proponents of this option can claim that it cannot be entirely dismissed on purely economic grounds for several reasons: 1) the cost increment associated with reprocessing and thermal recycle is small relative to the total cost of nuclear electricity generation; 2) assumptions for the unit cost of fuel cycle operations is highly uncertain and plutonium recycle can be shown to be competitive with the once-through option under certain assumptions; 3) in a balanced fuel cycle, MOX generates only about 15% of the electricity so its impact on the overall cost of electricity is mitigated.

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for plutonium recycle gives the results shown in Table 3.23. We note that the CEI is only about 5% higher than that of the LWR once-through option. This is because MOX accounts for a relatively small fraction of electricity production in a balanced fuel cycle and because fuel cycle costs are small relative to reactor costs. However, the fuel cycle costs associated with plutonium recycle and irradiation of MOX fuel are about 4.5 times higher than those of once-through UOX.

Table 3.23 CEI for plutonium recycle in LWR (mills/kWh)

	UOX	MOX	Overall
CEI_R	30.18	30.18	30.18
CEI_{FC}	4.06	18.66	6.12
CEI	34.24	48.84	36.29

3.2.2 DUPIC Fuel Cycle

The DUPIC (direct use of spent PWR fuel in CANDU) concept was introduced as a means of extracting more energy from PWR fuel without increasing initial enrichment, thus improving uranium utilization and minimizing spent fuel discharges. The DUPIC fuel cycle consists of recycling spent PWR fuel into CANDU fuel via a dry reprocessing process known as OREOX (oxidation and reduction of oxide fuel), which is essentially the same the AIROX process discussed in section 3.1.5. This is made possible by the excellent neutron economy of the CANDU reactors: spent PWR fuel, which typically has a fissile content of about 1.5%, can sustain criticality in a CANDU reactor up to a burnup of 10 to 20 GWd/MTHM. After irradiation in a CANDU reactor, the spent DUPIC fuel is sent to storage and disposal.

In studies on DUPIC carried out by AECL and KAERI, the reference PWR has a fuel burnup of 35 GWd/MTHM, a rather low value compared with today's average PWR burnup. It was estimated that DUPIC fuel could be irradiated to an additional 15 GWd/MTHM in a CANDU reactor, bringing the total burnup to 50 GWd/MTHM. This is now the average burnup in U.S. PWRs operating on the once-through cycle. A thorough analysis of the DUPIC fuel cycle under current conditions would be required to determine whether the benefits from this strategy outweigh the costs, namely those associated with OREOX reprocessing and DUPIC fuel fabrication.[1]

The OREOX process, which is at the heart of the DUPIC fuel cycle, will now be described[28]. During the dry OREOX process, uranium from spent PWR fuel is sequentially oxidized and reduced to a fine powder, which forms the starting material of DUPIC fuel pellets. The powder is pressed into pellets, sintered to a high density, ground to final size, and seal-welded within Zircaloy sheaths. All processing steps have to be carried out in hot cells due to the fuel's high radioactivity. Volatile fission products escape from the spent fuel and are trapped in ceramic based filters. Table 3.24 shows removal fractions expected for the OREOX process[29]. OREOX is at a fairly advanced stage of development; AECL (Atomic Energy of Canada Ltd.) and KAERI (Korea Atomic Energy Research Institute) have conducted experimental tests with this process, and have made test irradiations of the produced fuel

[1] The advanced CANDU reactor (ACR) has an inferior neutron economy, as compared to the original CANDU reactors, due to its light water coolant. Thus, the burnup that could be attained with OREOX recycled fuel in ACRs will be lower than in CANDUs and the DUPIC fuel cycle is less attractive under such conditions.

Table 3.24. OREOX fission product removal

Fission Product	% Removal
Cd	80
Cs	99
I	99
In	75
Kr	99
Mo	80
Ru	80
Se	99
Te	99
Xe	100
All others	0

Mass Flow Analysis

A study carried out at MIT has determined the burnup achievable for DUPIC fuel as a function of the initial enrichment and burnup of PWR fuel [29]. In particular, it was found that PWR fuel enriched to 4.5% and irradiated to 50 GWd/MTHM could be recycled into DUPIC fuel and further irradiated to 15 GWd/MTHM in a CANDU reactor, for a total burnup of 65 GWd/MTHM. To evaluate the mass of spent fuel discharged, the different thermal efficiencies of the PWR (33%) and CANDU (31%) reactors must be taken into account as follows:

Eq. 9:
$$m = \frac{365}{\left(B_d \cdot \eta_{th}\right)_{PWR} + \left(B_d \cdot \eta_{th}\right)_{CANDU}}$$

The spent fuel discharge for the DUPIC cycle is 17.3 MTHM/GWe·y. Using Eq. 3, the natural uranium requirement is 176.8 MTU/GWe·y.

The composition of DUPIC spent fuel made from recycled PWR fuel irradiated to 50 GWd/MTHM is not known because even the most recent studies on the DUPIC fuel cycle [30] are based on a reference PWR with a burnup of 35 GWd/MTHM. In this case, irradiation of DUPIC fuel leads to an 11% decrease in the mass of plutonium (due to fission of Pu-239 and Pu-241) and an 8% increase in the mass of minor actinides. Table 3.2 shows that the plutonium and minor actinide content of spent PWR fuel irradiated to 50 GWd/MTHM is 1.33% and 0.12%, respectively. Assuming that the mass of plutonium is reduced by 11% while the mass of minor actinides increases by 8% during irradiation of DUPIC fuel in the CANDU reactor, the plutonium and minor actinide content of spent DUPIC fuel can be approximated as 1.18% and 0.13%. Table 3.25 summarizes these results.

Table 3.25. Uranium consumption and spent fuel discharge for DUPIC

Uranium consumption (MTU/GWe·y)	176.8
Spent fuel discharge (MTHM/GWe·y)	17.3
Transuranic discharge (kg/GWe·y)	226.6
Plutonium discharge (kg/GWe·y)	204.1
Minor actinide discharge (kg/GWe·y)	22.5

In a balanced DUPIC fuel cycle, all the spent fuel discharged from PWRs is recycled into DUPIC fuel for CANDU reactors. The PWR to CANDU ratio in the fuel cycle can be determined using Eq. 5, as follows:

$$\text{Mass of spent PWR fuel} = \text{Mass of DUPIC fuel for CANDUs}$$

$$\frac{P_{PWR} \cdot 365 \cdot CF}{\left(B_d \cdot \eta_{th}\right)_{PWR}} = \frac{P_{CANDU} \cdot 365 \cdot CF}{\left(B_d \cdot \eta_{th}\right)_{CANDU}}$$

$$\frac{P_{PWR}}{P_{CANDU}} = \frac{\left(B_d \cdot \eta_{th}\right)_{PWR}}{\left(B_d \cdot \eta_{th}\right)_{CANDU}}$$

$$\frac{P_{PWR}}{P_{CANDU}} = \frac{50 \cdot 0.33}{15 \cdot 0.31} = 3.55$$

Thus, in a balanced DUPIC fuel cycle, PWRs account for 78.0% of the installed nuclear capacity, while CANDUs account for 22.0%. Note that a nuclear park consisting of 78% LWRs and 22% CANDUs would discharge 51.8 MTHM/GWe·y of spent fuel[1], so the DUPIC fuel cycle allows reduces the mass of spent fuel discharged by a factor of 3.

[1] 0.78·22.1 MHTM/GWe·y + 0.22·157 MTHM/GWe·y = 51.8 MTHM/GWe·y.

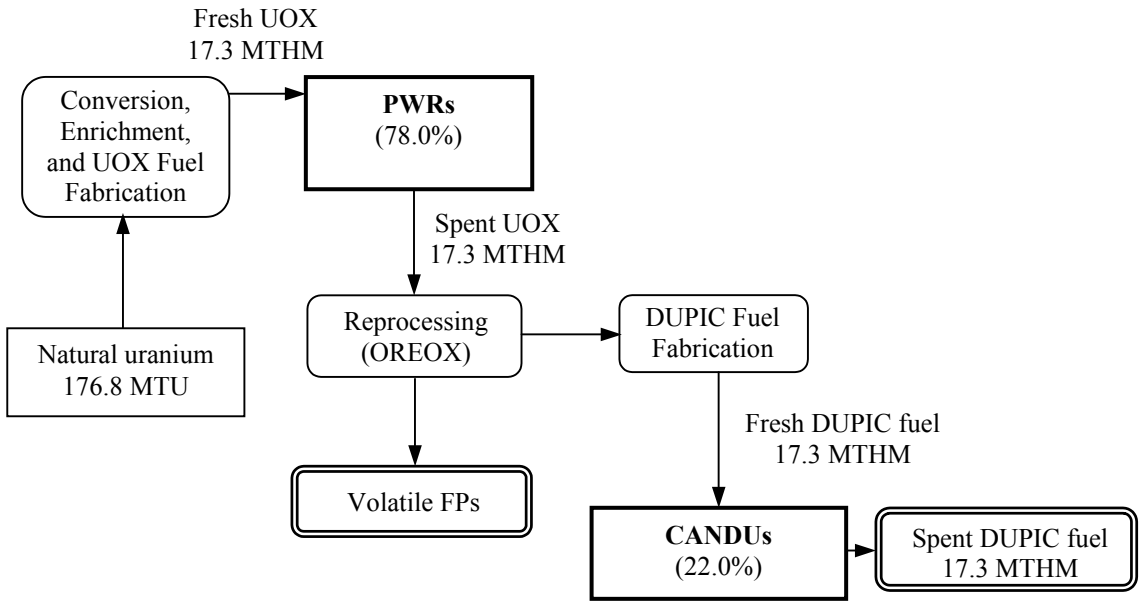

Figure 3.6. DUPIC fuel cycle
(all quantities are per GWe·y)

Proliferation

Although the DUPIC fuel cycle involves recycling spent fuel, the reprocessing method employed, OREOX, presents minimal proliferation risks because spent fuel is not dissolved and none of its components are extracted other than volatile fission products. For this reason, the reprocessing proliferation indicator will be taken as zero for the DUPIC fuel cycle.

In addition, the irradiation of DUPIC fuel allows for a reduction in the mass of plutonium discharged per unit energy generated as compared to the once-through fuel cycle in LWRs. The plutonium discharge is about 30% lower than for LWRs at 50 GWd/MTHM and roughly equivalent to LWRs at 100 GWd/MTHM.

Economics

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for the DUPIC fuel cycle gives the results shown in Table 3.26.

Table 3.26 CEI for DUPIC fuel cycle (mills/kWh)

	LWR-UOX	PHWR-DUPIC	Overall
CEI_R	30.18	30.18	30.18
CEI_{FC}	4.06	5.83	4.46
CEI	34.24	36.01	34.63

3.2.3 Actinide recycle in Light Water Reactor

Almost all fuel cycles involving transmutation of actinides rely on advanced reactors. However, the expenses associated with the development of such reactors are substantial. Furthermore, it is unlikely that they will be deployed on a significant scale in the coming decades without government involvement. As a result, the use of LWRs for waste transmutation has been proposed. While it is conceded that the combination of a thermal spectrum and solid fuel presents a serious challenge for fuel cycle operations, the fact that LWRs are likely to remain the most widely deployed reactors for many decades makes this an appealing option.

Several alternatives have been identified to incorporate transuranic elements in LWR fuel. One possibility is the use of mixed oxide fuel containing not just plutonium but also minor actinides. Another is the use of thorium fuel, which provides fertile content but generates fewer higher actinides, as compared to uranium, because of its lower atomic mass. Finally, the use of fertile free fuels (FFF) or inert matrix fuels (IMF), where no uranium or thorium is added to the transuranic elements, is also being studied [31]. While recent work has shown that thorium and mixed oxide fuels are also viable options [32,33], this section will focus on FFF.

Although TRU can be mixed homogeneously with the inert matrix, another possible design for FFF is to disperse TRU micro particles in the inert matrix. This approach allows more flexibility in the selection of materials, which must be chosen to provide good mechanical and chemical stability, radiation damage resistance, compatibility with cladding and coolant, good thermal properties, low parasitic neutron absorption, and advantageous properties with respect to the fuel cycle (e.g. chemical stability in the repository environment, simple reprocessing, etc.). Because FFF containing transuranic elements has undesirable features with respect to reactor control and safety, LWR cores can only be loaded with a small fraction of this fuel. One concept for using TRU bearing FFF in current reactors is to displace a limited number of UOX fuel pins in a conventional PWR assembly and replace them with FFF pins. This concept is known as the CONFU (combined non-fertile and uranium) fuel assembly. Figure 3.7 shows a CONFU configuration where 60 pins are replaced with FFF in a typical 17×17 PWR assembly.

The CONFU assembly reactivity coefficients and soluble boron worth differ only slightly from those of a conventional uranium fuel assembly, which indicates that CONFU is potentially compatible with conventional PWR systems. In addition, the generally low β_{eff} of transuranic elements does not pose a problem for the CONFU assembly. Even after 6 recycles, when higher actinide buildup is significant, the β_{eff} of the CONFU assembly at the beginning of irradiation is only moderately lower than for a conventional uranium assembly, and this difference diminishes with burnup.

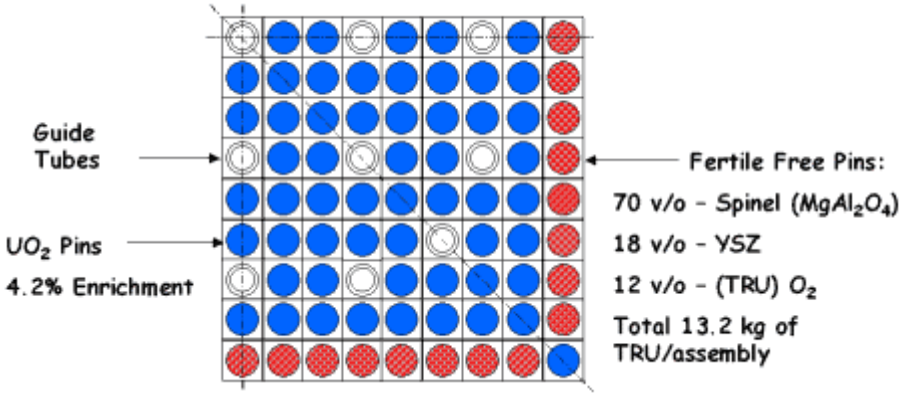

Figure 3.7. CONFU assembly concept.

All fuel pins discharged from a CONFU fuel assembly are reprocessed. The transuranic elements are recovered and incorporated in new FFF pins which will once again undergo irradiation in a CONFU assembly. Table 3.27 [31] shows the material flows, per GWe·y, that are obtained this fuel cycle. Although equilibrium values are not given, the table shows a convergence toward a net consumption of transuranics. This consumption rate is very small compared to the TRU generation rate of conventional PWRs; as a result, CONFU PWRs are not effective in reducing large existing stocks of transuranics even if all reactors are used for this purpose.

Table 3.27. PWR CONFU materials flow summary (per GWe·y)

	Recycle Stage					
	1	2	3	4	5	6
Total HM loaded, kg	16,149	18,698	19,330	19,538	19,538	19,582
Uranium loaded, kg	15,578	18,037	18,647	18,847	18,847	18,889
TRU loaded to FFF pins, kg	580	671	694	701	701	703
TRU discharged from UOX pins, kg	209	227	231	228	228	230
TRU discharged from FFF pins, kg	285	402	435	458	458	457
TRU discharged total, kg	495	629	665	685	685	687
Net TRU consumption, kg	85.1	42.4	28.7	15.8	15.8	16.0
Discharge burnup (assembly), GWd/MTHM	68.3	59.0	57.1	56.7	56.5	56.3

Mass Flow Analysis

For simplicity, the analysis presented here will consider a nuclear park composed entirely of LWRs loaded with CONFU assemblies with zero net TRU generation. It should be remembered that CONFU LWRs can in principle achieve a slight net TRU consumption and therefore, a LWR park with zero net TRU generation could accommodate a small fraction of conventional PWRs. In accordance with Table 3.27, a TRU content of 3.6% and a burnup of 56 GWd/MTHM is assumed for the CONFU assemblies (the burnup in the UOX pins is 45 GWd/MTHM; in the FFF pins, it is 350 GWd/MTHM). The enrichment of the uranium fuel pins is 4.2%. Reprocessing losses are assumed to be 0.1%.

Using these assumptions, the mass of fuel irradiated is found to be 19.75 MTHM/GWe·y using Eq. 2. Assuming a TRU content of 3.6%, the mass of FFF is 711 kg/GWe·y and the mass of UOX is 19.04 MTHM/GWe·y. Using Eq. 3, the mass of natural uranium required for the fabrication of the UOX fuel is 180.7 MTU/GWe·y. The content of the spent CONFU assemblies is determined as follows: assuming a fission energy yield of 1000 $GWd/MTHM_{fissioned}$, the mass of fission products generated is 1,106 kg/GWe·y. Since the net generation of transuranics is assumed to be zero, the spent fuel contains the same mass of transuranics as the initial fuel, 711 kg/GWe·y. The rest of the spent CONFU fuel assembly, or 17.93 MT/GWe·y, is uranium. The fuel cycle is represented in Figure 3.8.

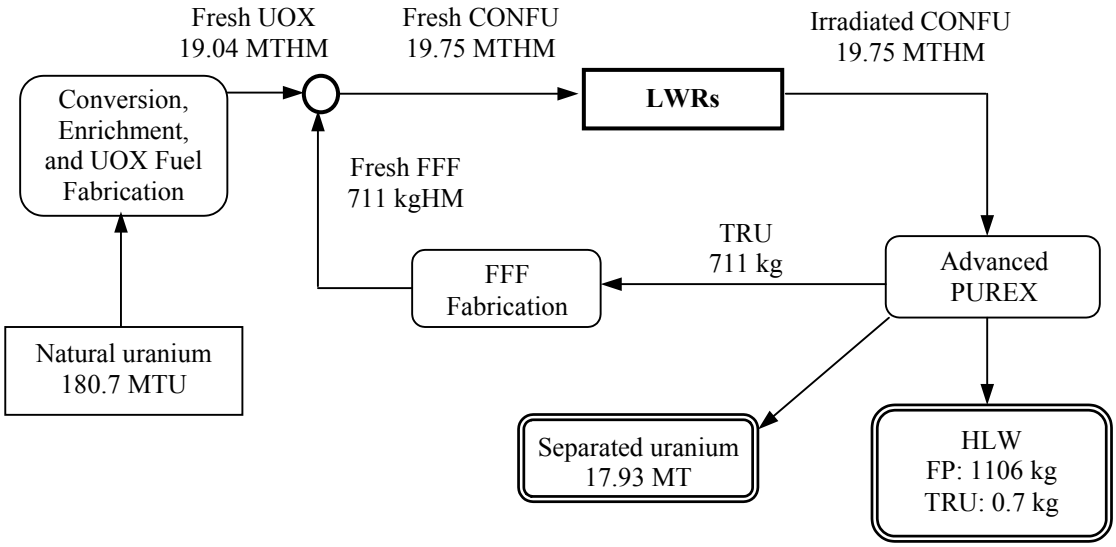

Figure 3.8. Actinide transmutation in LWRs (FFF in CONFU assemblies)
(all quantities are per GWe·y)

In practice, the equilibrium situation depicted in Figure 3.8 may be difficult to attain because LWR transmutation of actinides involves multiple recycle in a thermal spectrum and solid fuel, a combination that poses serious difficulties for fuel cycle operations. Indeed, certain higher actinides with undesirable characteristics accumulate with every recycle. In particular, Cm-244, which is a high decay heat source and a prolific emitter of neutrons, is generally viewed as the most problematic isotope. The buildup of Cf-252 also adds to the neutron source level and actually becomes the leading contributor after about 6 recycle stages. Indeed, although Cf-252 is present in smaller quantities than Cm-244, its specific neutron source (n/kg·s) is 215,000 times higher than that of Cm-244. Table 3.28 shows the radioactivity, thermal power, and neutron source for the spent CONFU assemblies 5 years after discharge and for the TRU recovered for FFF fabrication (7 years after discharge) [31]. Note that only the contribution from the actinides is considered.

Table 3.28. Characteristics of spent CONFU assemblies and FFF fuel

		Recycle stage				
		1	2	3	4	5
Spent CONFU assembly 5 years after discharge	HM mass flow, kg/GWe·y	15,018	17,560	18,197	18,320	18,436
	Radioactivity, Ci/kgHM	515	553	548	540	535
	Thermal power, W/kgHM	3.99	5.92	6.92	7.48	7.81
	Neutron source, n/kgHM·s	$8.72 \cdot 10^{6}$	$1.96 \cdot 10^{7}$	$4.46 \cdot 10^{7}$	$9.66 \cdot 10^{7}$	$1.76 \cdot 10^{8}$
FFF fabrication 7 years after discharge	TRU mass flow, kg/GWe·y	495	629	665	676	686
	Radioactivity, Ci/kgTRU	$1.43 \cdot 10^{4}$	$1.41 \cdot 10^{4}$	$1.37 \cdot 10^{4}$	$1.34 \cdot 10^{4}$	$1.32 \cdot 10^{4}$
	Thermal power, W/kgTRU	113	155	177	189	195
	Neutron source, n/kgTRU·s	$2.42 \cdot 10^{8}$	$4.64 \cdot 10^{8}$	$9.05 \cdot 10^{8}$	$1.78 \cdot 10^{9}$	$3.06 \cdot 10^{9}$

A disadvantage of the CONFU approach is that the transuranics are concentrated in the FFF pins, which constitutes only a small fraction of the fuel. This presents additional challenges for operations associated with fuel fabrication because the neutron source per unit mass of FFF is extremely high, as shown in Table 3.28.

Table 3.28 clearly shows that, whereas radioactivity and thermal power converge, the neutron source increases significantly with every recycle. This is shown graphically in Figure 3.9: whereas the decay heat level tends to approach an asymptote after about 5 recycles, the neutron source increases in an exponential way. This potentially limits the number of transuranic recycles that could be done in LWRs. For this reason, transmutation in LWRs is often seen as a delaying strategy for holding the total transuranic inventory constant until more advanced burners are deployed. If LWR transmutation of actinides is seen as a transitional technology limited to about 5 recycles, its potential for reducing the total actinide inventory is rather limited; indeed, whereas the TRU production at equilibrium would be about 500 times lower than for the once-through cycle (0.7 kgTRU/GWe·y versus 320 kgTRU/GWe·y), the production of actinides per energy generated over 5 recycles is only about 6 times lower than for the once-through cycle. This is because the quantity of transuranics generated by the irradiation of the initial UOX fuel is held more or less constant during the 5 subsequent recycle and irradiation stages. Thus, the amount of TRU discharged after the last irradiation is roughly the same as in the original spent UOX, but there has been a total of 6 irradiations.

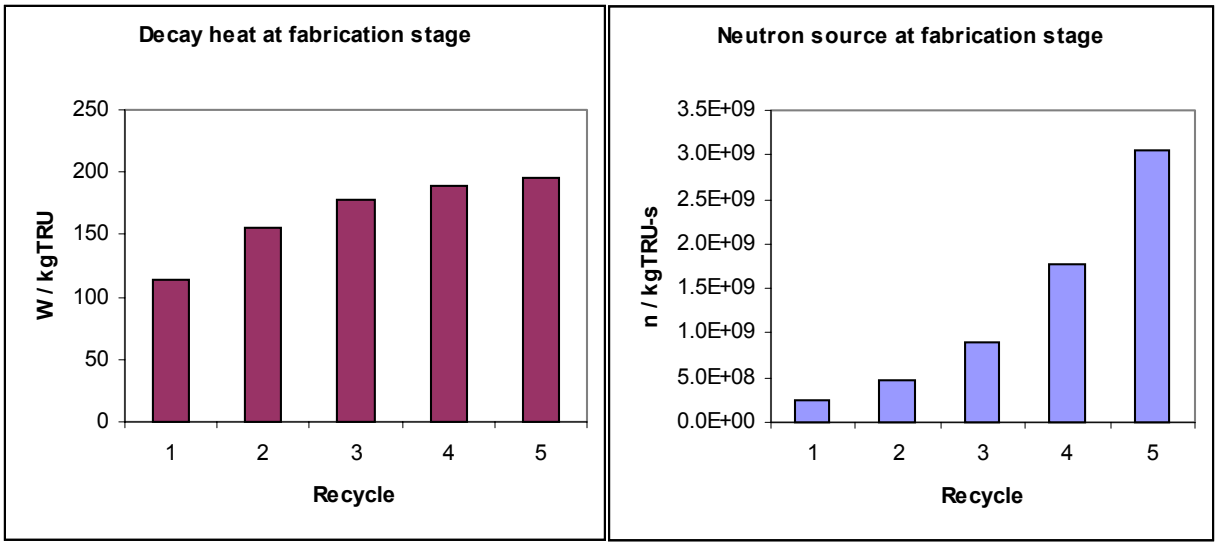

Figure 3.9. CONFU decay heat and neutron source at fabrication

A few strategies have been proposed in order to allow recycle of actinides beyond 5 or so recycles. One is to separate and store curium for an extended period to allow Cm-244 to decay into Pu-240. The half-life of Cm-244 is 18 years, so a storage period of about 60 years would allow quantities to decrease by a factor of 10, after which the stored material could be reintegrated into the fuel cycle. Obvious disadvantages of this approach are that it adds complexity and cost to the fuel cycle, especially considering that handling and storage of Cm-244 is likely to be particularly costly due to its high decay heat and neutron emission rate. Furthermore, the storage period needed (60 years is sufficient fo a factor of 10 reduction, but some authors suggest 360 years, or 20 half-lives, in order to reduce Cm-244 content by a factor of 10^6 [32]) is rather long, entailing large stored inventories and very long-term planning.

A second approach consists of simply storing spent fuel for longer periods between recycle stages. It has been shown that, with an elapsed time of 20 years between recycle stages (18 years for cooling, 2 years for fuel fabrication), the neutron source after 5 recycles is reduced by a factor of 8 [34]. The main benefit of this strategy is that no separation and storage of curium is required. However, the inventory of spent fuel in storage would be three times larger than for a delay time of 7 years between recycles. Furthermore, Table 3.29 (values for fuels other than CONFU FFF are from [24]) shows that the impact of this strategy is somewhat limited: even with 20 years between recycles, the neutron source after the 5^{th} recycle is still comparable to that of fast spectrum systems at equilibrium. In addition, the cooling time assumed for the fast spectrum systems is only 2 years, so the fuel in storage requirements are much smaller in this case.

To conclude, recycling actinides in LWRs presents difficulties for fuel cycle operations due to the accumulation of certain actinides having a high neutron emission rate, principally Cm-244 and Cf-252. For a cooling time of 5 years, the neutron source of spent fuel after about 5 recycles exceeds equilibrium values in fast spectrum systems;

therefore, further recycling of transuranics in LWRs will require more costly shielding measures, longer storage times, or added complexity (separation and storage of curium) as compared to fuel cycles based on fast spectrum systems.

Table 3.29. Neutron source at fuel fabrication for various fuel types

Fuel type	Cooling time	Neutron source (10^6 n / kgHM·s)
UOX	4	$2.1 \cdot 10^{-5}$
MOX	7	0.10
FFF in CONFU (5th recycle)	5	3,060
FFF in CONFU (5th recycle)	18	~383
TRU burner (FR)	2	92.1
TRU burner (ADS)	2	670
MA burner (ADS)	2	1,992

Proliferation

Transuranic recycling in LWRs would require very large scale reprocessing operations in order to extract transuranic elements from all the discharged spent fuel assemblies. The reprocessing technology used would probably be an advanced version of the PUREX process whereby minor actinides are extracted from spent fuel along with plutonium. Proliferation concerns are somewhat alleviated because no separated plutonium is produced, but a dirty bomb, having a low yield but still effective in dispersing radioactive material, could be made from a mix of transuranics. Assuming a storage time of 6 months, the transuranic working inventory in the fuel cycle under consideration is 355.5 kg/GWe. Considering that less than 10 kg would be sufficient to make a dirty bomb, this is a large quantity. Opportunities for diversion of this material could be reduced somewhat by collocation of the reprocessing and fuel fabrication plants, which would avoid transportation.

On the other hand, because all transuranics are recycled, the quantity of plutonium going to the repository is extremely small compared to a once-through cycle. The quantity of transuranics going to waste is 0.7 kg/ GWe·y, 85% of which is plutonium [34], so the plutonium discharge for this fuel cycle is 0.6 kg/GWe·y. This is roughly 500 times less than for the LWR once-through cycle.

Economics

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for actinide recycle in LWRs gives the results shown in Table 3.30. The fuel cycle cost for actinide recycle in FFF is extremely high because the cost of UOX reprocessing is assigned to the FFF part the fuel cycle, which accounts for only 3.6% of electricity generation. Thus the fuel cycle cost per electricity generated is exceptionally high. The overall fuel cycle cost is reasonable however, since FFF accounts for such a small fraction of electricity generation. Note also that the enrichment and burnup of the UOX fuel for CONFU assemblies differs slightly from the values assumed for conventional

assemblies, which results in a somewhat higher CEI_{FC} for the UOX component of the fuel cycle, as compared to the standard LWR UOX once-through cycle.

Table 3.30 CEI for transuranic recycle in CONFU (mills/kWh)

	LWR-UOX	LWR-FFF	Overall
CEI_R	30.18	30.18	30.18
CEI_{FC}	4.28	14.22	6.52
CEI	34.46	44.40	36.69

3.2.4 Actinide Recycle in Fast Reactor

Fast reactors (FR) have been developed on a limited scale in several countries, where they function as breeders on a Pu / U cycle. In this cycle, the reactor runs on plutonium fuel while simultaneously breeding plutonium from U-238 in a quantity that is at least sufficient to meet the reactor's own fuel requirements. Interest in this fuel cycle was sparked several decades ago by perceptions of uranium scarcity. Today, however, uranium resources appear sufficient to last well beyond the middle of the century, and while this cycle does indeed allow a very efficient use of natural uranium resources, it does not lead to significant reductions in the toxicity of nuclear waste. Furthermore, it involves extraction, storage, and transportation of large amounts of separated plutonium, which raises serious proliferation concerns. As a result, interest in fast reactors has recently shifted to their use as transuranic burners. Indeed, because of their superior neutron economy and their ability to fission all actinides, they are generally viewed as better suited to this role than thermal reactors.

Mass Flow Analysis

The assumptions used for fuel cycle calculations are shown in Table 3.31. They are based on a recent OECD study on advanced fuel cycles [24].

Table 3.31. Assumptions for the fast reactor transuranic burner

Thermal efficiency	40%	
Burnup (GWd/MTHM)	140	
Fuel composition	Charge	Discharge
Uranium	0.6696	0.5994
Transuranics	0.3313	0.2597
Fission Products	-	0.141

The fraction of installed capacity supplied by each type of reactor is determined by requiring that the mass of transuranics discharged in the spent fuel from LWRs and FRs be equal to the mass of transuranics required for the fabrication of new FR fuel. Assuming 99.9% recovery of TRU in reprocessing:

$$\text{TRU needed for FR fuel} = 0.999 \cdot \left[\begin{matrix} \text{TRU from} \\ \text{spent FR fuel} \end{matrix} + \begin{matrix} \text{TRU from} \\ \text{spent UOX} \end{matrix} \right]$$

Using Eq. 6, and recalling the TRU content of spent LWR fuel from Table 3.2 and of FR fuel from Table 3.31:

$$\frac{365}{140 \cdot 0.4} \cdot f_{FR} \cdot 0.3313 = 0.999 \cdot \left[\frac{365}{140 \cdot 0.4} \cdot f_{FR} \cdot 0.2597 + \frac{365}{50 \cdot 0.33} \cdot f_{LWR} \cdot 0.0145 \right]$$

$$\frac{f_{FR}}{f_{LWR}} = 0.684; \quad f_{LWR} = 0.594; \quad f_{FR} = 0.406$$

Thus, under the current assumptions, a nuclear park with a LWR :FR ratio of 1.5 :1 can maintain a constant inventory of transuranics. Using Eq. 2, the mass of fuel irradiated in LWRs is 13.1 MHTM/GWe·y. Using Eq. 3, the mass of natural uranium required for UOX fabrication is 134.6 MTU/GWe·y. The spent UOX fuel contains 12.27 MT of uranium, 677 kg of fission products, and 190 kg of transuranics. The spent UOX fuel is reprocessed and 99.9% of its transuranic content is recovered for FR fuel fabrication, while the uranium is separated and the fission products, as well the transuranic losses, are incorporated in HLW.

Using Eq. 2, the mass of fuel irradiated in FRs is 2.65 MTHM/GWe·y. Using the assumptions listed in Table 3.31, the initial FR fuel contains 1.77 MT of depleted uranium and 877 kg of transuranics, while the spent FR fuel contains 1.59 MT of uranium, 687 kg of transuranics, and 373 kg of fission products. The uranium and fission products end up in HLW, along with transuranic losses, while 99.9% of the transuranics are recovered for FR fuel fabrication. The fuel cycle is shown in Figure 3.10. Table 3.32 summarizes the calculation results.

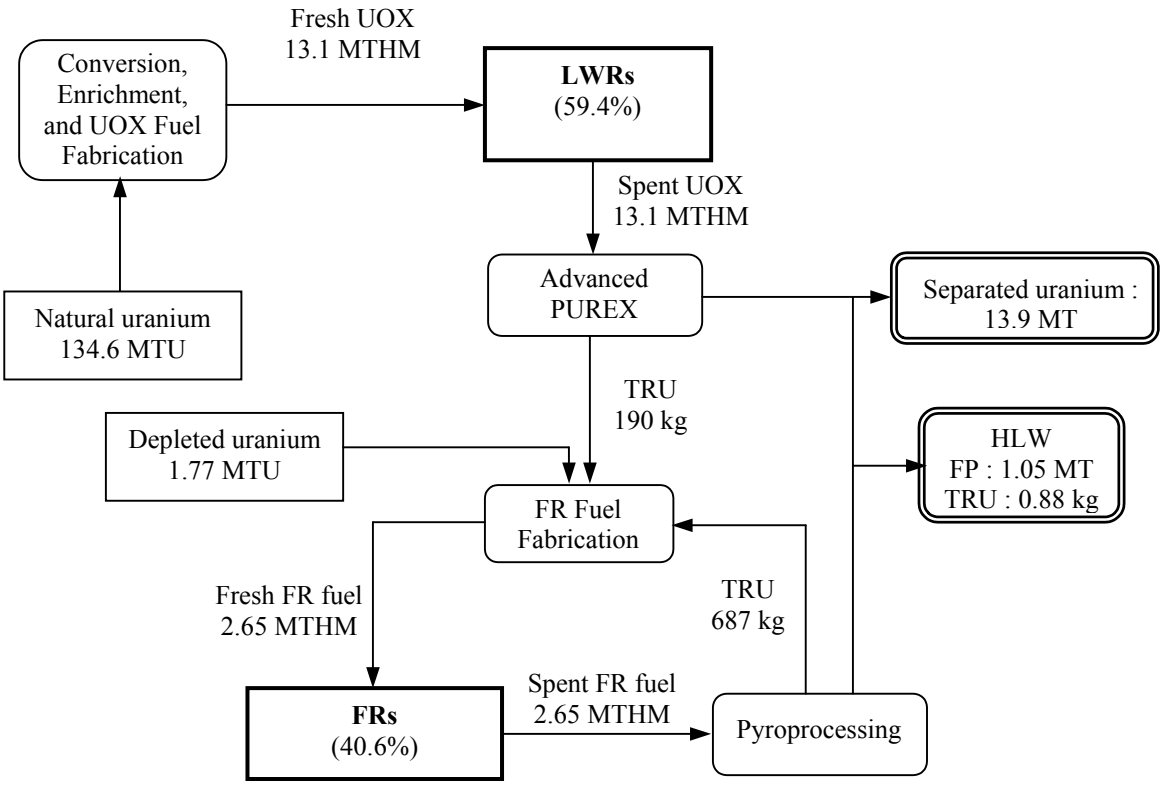

Figure 3.10. Balanced fuel cycle with LWRs and FR TRU burners
(all quantities are per GWe·y)

Table 3.32. Characteristics of the balanced LWR / TRU burner fuel cycle

Uranium consumption (MTU/GWe·y)	134.6
Fuel reprocessed (MTHM/GWe·y)	15.75
Transuranics discharged (kg/GWe·y)	0.88
Fission products discharged (MT/GWe·y)	1.05
Separated uranium (MT/GWe·y)	13.9

Proliferation

The total mass of spent fuel reprocessed in this fuel cycle is 15.75 MTHM. No separated plutonium is produced, but a considerable amount of transuranics must be held in inventory between separation and fuel fabrication. Assuming a storage time of 6 months, the working inventory is 438.5 kg. The quantity of transuranics going to waste is 0.88 kg/ GWe·y. Assuming that 85% of this amount is plutonium, the plutonium discharge for this fuel cycle is 0.75 kg/GWe·y.

Economics

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.33.

Table 3.33 CEI for transuranic recycle in fast reactor (mills/kWh)

	LWR-UOX	FR-TRU	Overall
CEI_R	30.18	37.28	33.06
CEI_{FC}	4.06	6.30	4.96
CEI	34.24	43.58	38.02

3.2.5 Double Strata Strategy

The purpose of the double strata (also known as 2-tier) strategy is to concentrate the burning of minor actinides into the smallest possible fraction of the nuclear park. This is accomplished by burning the plutonium discharged from LWRs in conventional systems while destroying the minor actinides in advanced burners.. The conventional systems that burn plutonium constitute the first stratum, while the second stratum consists of the advanced minor actinide burners. Because the bulk of transuranics in spent fuel is plutonium, the mass flow to the second stratum is relatively small. As a result, zero net production of transuranics is achieved with only a small fraction (about 5%) of the nuclear park composed of advanced burners.

The prevalent approach proposed for implementing the double strata strategy involves recycling plutonium in MOX fuel to be irradiated in light water reactors and sodium cooled fast reactors, while the minor actinides are fissioned in an accelerator driven system (ADS). Because of its subcriticality, it is believed that the ADS core can achieve a higher minor actinide transmutation rate than a critical reactor of comparable power. Recently, however, several authors have suggested that properly designed critical reactors may be able to fulfill the role of minor actinide burner in such fuel cycles [35, 36]. Accordingly, a 2-tier fuel cycle that relies on a high burnup GT-MHR for the first tier and a critical minor actinide burner in the second tier will also be examined in this section.

3.2.5.1 Subcritical Minor Actinide Burner

The ADS concept

An accelerator driven system (also called a hybrid system) combines a particle accelerator with a sub-critical core (see Figure 3.11 [24]). In most ADS designs, protons from the accelerator strike a liquid heavy metal target. Some of these protons strike neutrons or protons in the target nuclei and eject them in the forward direction with a lower energy than the incident particle. These neutrons, and sometimes protons, then strike other nuclides, and they may eject other forward moving but lower energy particles. This process, know as a spallation cascade, continues until the energy of the accelerator particles is spent. In any of these collisions, the struck nucleus is always excited to some degree and subsequently release this excess energy by emitting neutrons. In fact, neutrons produced in this way account for about 90% of the neutrons sent into the core. Altogether, a few tens of neutrons are produced per incident proton.

The energy conversion part of an accelerator driven system is similar to that of a normal power plant, but the energy required for the operation of the accelerator reduces the overall thermal efficiency of the system. In most ADS designs, which use molten salt fuel or a liquid metal coolant, heat can be extracted from the core and converted into electricity with an efficiency of about 40%, but the overall system efficiency is significantly lower, roughly 30%.

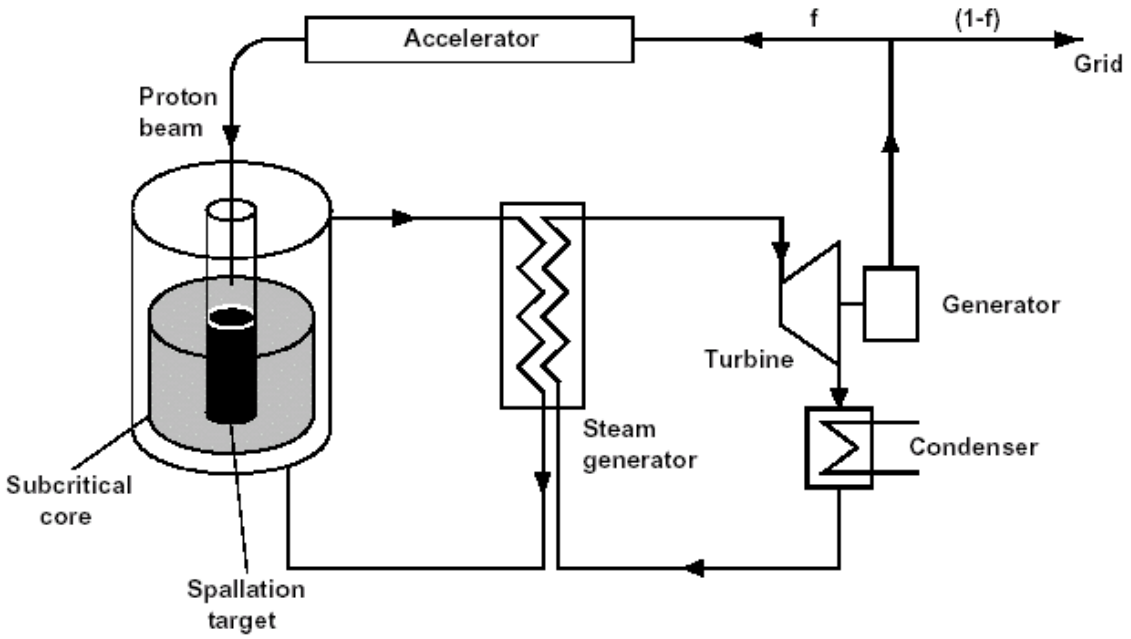

Figure 3.11. Concept of an accelerator driven system

The possibility to operate the core of an ADS at a neutron multiplication factor below 1 increases core design and fuel management flexibility. In particular, it allows transmuters to be designed as pure transuranic or minor actinide burners, which minimizes the fraction of specialized transmuters in the reactor park. Furthermore, the reactivity margin provided by subcritical operation potentially enables the safe operation of cores with a high loading of minor actinides, which typically have a very low delayed neutron fraction and unfavorable reactivity feedback coefficients.

It must be noted, however, that the potential advantages of the ADS concept must be weighed against several disadvantages. The presence of an accelerator increases the overall plant complexity and reduces thermal efficiency, which will have detrimental consequences on the economics of the plant. Important design and material problems arise from the installation of a spallation target in the center of a reactor: the interfacing of an accelerator with a reactor raises containment questions, and the target and surrounding structural materials are subjected to unusual stress, corrosion, and irradiation conditions. The reliability of current accelerators makes them unacceptable for use in an ADS because the frequency of beam trips, which would cause transients similar to a fast control rod insertion in a critical reactor, is orders of magnitude above current criteria for such transients. Finally, and perhaps most importantly, new types of reactivity and source transients have to be dealt with. A recent assessment of the transient response of ADS in comparison to critical reactors [35] has concluded that, for systems with unfavorable reactivity coefficients, subcriticality does not assure automatic safety and states that careful analysis and assessment is required for each different ADS core design to assure that its combination of reactivity coefficients and the selected operating level of

subcriticality does not lead to an unacceptable dynamic ADS plant behavior in case typical plant transient initiators such as loss of flow or loss of heat sink occur.

Mass Flow Analysis

Many ADS designs are possible: the fuel can be in liquid or solid form, the neutron spectrum can be fast or thermal, the system can be optimized for burning transuranics or only minor actinides. The ADS design considered here is based on the accelerator driven MA burner presented in Reference 24, a solid fuel, liquid metal cooled, fast spectrum system optimized for burning minor actinides. The main characteristics of this ADS are presented in Table 3.34.

Table 3.34. Characteristics of the ADS MA burner

Thermal efficiency	0.315	
Burnup (GWd/MTHM)	140	
Fuel composition	Initial	Final
▪ U	0.0462	0.04678
▪ Pu	0.4042	0.4092
▪ MA	0.5496	0.4039
▪ TRU	0.9538	0.8132
▪ FP	-	0.14

Table 3.35 summarizes the characteristics of the other reactors involved in the double strata fuel cycle, which are needed in order to determine the composition of the nuclear park and the mass flows between its various components.

Table 3.35. Characteristics of conventional reactors in the ADS double strata fuel cycle

	LWR-UOX		LWR-MOX		FR	
Thermal efficiency	0.33		0.33		0.4	
Burnup (GWd/MTHM)	50		50		185	
Fuel composition	Initial	Final	Initial	Final	Initial	Final
▪ U	1	0.934	0.919	0.8879	0.5557	0.4656
▪ Pu	-	0.0133	0.081	0.0552	0.4443	0.3222
▪ MA	-	0.0012	-	0.0054	-	0.0272
▪ TRU	-	0.0145	-	0.0606	-	0.3494
▪ FP	-	0.0515	-	0.0515	-	0.185

The ratio of UOX to MOX is determined by equating the plutonium discharge from the LWRs using UOX fuel to the plutonium consumption of the LWRs using MOX fuel. The losses in all reprocessing operations are assumed to be 0.1%.

Pu in spent UOX·0.999 = Pu needed for fresh MOX

$$P_{UOX} \cdot 1.33 \cdot 0.999 = P_{MOX} \cdot 8.1$$

Eq. 10:
$$\frac{P_{UOX}}{P_{MOX}} = \frac{8.1}{1.33 \cdot 0.999} = 6.096$$

The ratio of FRs to LWRs using MOX is obtained by equating the quantity of plutonium required for FR fuel fabrication to that which is recovered from spent MOX and spent FR fuel.

$$\text{Pu needed for FR fuel} = 0.999 \cdot \left[\begin{array}{c} \text{Pu from} \\ \text{spent FR fuel} \end{array} + \begin{array}{c} \text{Pu from} \\ \text{spent MOX} \end{array} \right]$$

$$\frac{f_{FR}}{185 \cdot 0.4} \cdot 0.4443 = 0.999 \cdot \left[\frac{f_{FR}}{185 \cdot 0.4} \cdot 0.3222 + \frac{f_{MOX}}{50 \cdot 0.33} \cdot 0.0552 \right]$$

Eq. 11:
$$\frac{f_{FR}}{f_{MOX}} = 2.020$$

The ADS fraction of installed capacity can be related to the MOX, UOX, and FR fractions by equating the mass of transuranics required for ADS fuel fabrication with the mass of transuranics recovered from spent ADS fuel, plus the total mass of minor actinides recovered from spent UOX, spent MOX, and spent FR fuel.

$$\begin{array}{c} \text{TRU needed} \\ \text{for ADS fuel} \end{array} = 0.999 \cdot \left[\begin{array}{c} \text{TRU from} \\ \text{spent ADS fuel} \end{array} + \begin{array}{c} \text{MA from} \\ \text{spent FR fuel} \end{array} + \begin{array}{c} \text{MA from} \\ \text{spent MOX} \end{array} + \begin{array}{c} \text{MA from} \\ \text{spent UOX} \end{array} \right]$$

Eq. 12:

$$\frac{f_{ADS} \cdot 0.9538}{140 \cdot 0.315} = 0.999 \cdot \left[\frac{f_{ADS} \cdot 0.8132}{140 \cdot 0.315} + \frac{f_{FR} \cdot 0.0272}{185 \cdot 0.4} + \frac{f_{MOX} \cdot 0.0054}{50 \cdot 0.33} + \frac{f_{UOX} \cdot 0.0012}{50 \cdot 0.33} \right]$$

Substituting Eq. 10 and Eq. 11 into Eq. 12, we get:

Eq. 13
$$\frac{f_{UOX}}{f_{ADS}} = 12.923$$

Using Eq. 10, Eq. 11, and Eq. 13, we find the composition of the double strata nuclear park to be as follows:

$$f_{UOX} = 0.636; \quad f_{MOX} = 0.104; \quad f_{FR} = 0.211; \quad f_{ADS} = 0.049$$

The mass of fuel irradiated in each type of reactor is determined using Eq. 6. Using fuel compositions shown in Table 3.34 and Table 3.35, the content of the various types of spent fuel in the double strata scheme can be determined. Results are summarized in Table 3.36.

Table 3.36. Spent fuel mass flows (double strata strategy – ADS)

Material (/GWe·y)	LWR UOX	LWR MOX	FR MOX	Total (first stratum)	ADS	Total
Spent Fuel (MTHM)	14.1	2.31	1.04		0.407	
U (MT)	13.14	2.05	0.48	15.67	$19.1 \cdot 10^{-3}$	15.67
FP (kg)	724.4	118.8	192.2	1035.4	57.0	1092
Pu (kg)	187.1	127.4	334.8	649.3	166.7	816
MA (kg)	16.9	12.5	28.3	57.6	164.5	222
TRU (kg)	204.0	139.8	363.0	706.8	331.2	1037

Assuming a recovery of 99.9%, the mass of plutonium recovered from spent UOX, spent LWR MOX, and spent FR MOX is 186.9, 127.2, and 334.4 kg/GWe·y, respectively. The mass of depleted uranium needed for MOX fabrication is the difference between the total fuel mass and the mass of plutonium. For LWR MOX, 2.12 MT of depleted uranium is required; for FR MOX, 577.4 kg.

Of the total mass of minor actinides generated in tier-1 systems, 99.9%, or 57.58 kg, is recovered in reprocessing and used for ADS fuel fabrication. In addition, 99.9% of the transuranics in spent ADS fuel, or 330.8 kg/GWe·y, is recycled for fresh ADS fuel fabrication. Since the total mass of ADS fuel is 407.2 kg, a small amount of recycled uranium (18.81 kg/GWe·y) is also needed for ADS fuel fabrication.

All the fission products generated in the fuel cycle, 1092 kg, are discharged in HLW. In addition, 0.1% of the total mass of transuranics going through reprocessing, or 1.04 kg, is lost and discharged as HLW.

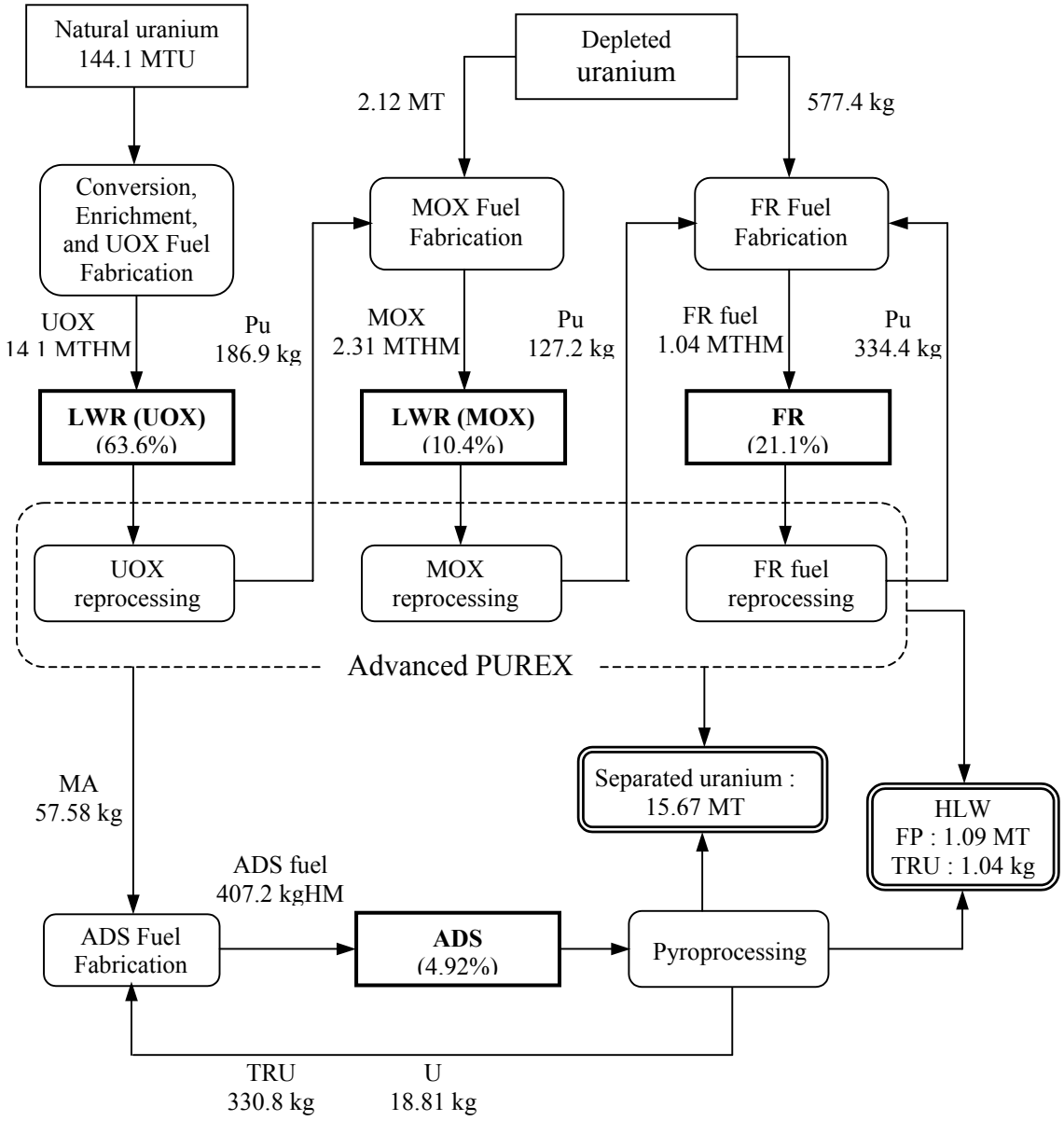

Figure 3.12. Double strata fuel cycle – ADS
(all quantities are per GWe·y)

Proliferation

The double strata fuel cycle requires separation of plutonium in the first stratum. The total mass flow of separated plutonium is 651.2 kg/GWe·y. Assuming a storage time of 6 months, the plutonium working inventory is 325.6 kg/GWe. In the second stratum, transuranic elements are extracted from the spent ADS fuel and recycled for further irradiation. The mass flow rate of transuranics is 330.8 kg/GWe·y, requiring a working inventory of 165.4 kg/GWe. The total mass flow rate of fuel to be reprocessed is 17.86

MTHM, of which 17.45 MTHM are reprocessed by advanced PUREX and 0.407 MTHM by pyroprocessing.

The mass of plutonium going to the repository is obtained by multiplying the mass of each type of fuel going to reprocessing by its plutonium content and the reprocessing loss factor (10^{-3}). The total mass of plutonium going to waste is 0.816 kg/GWe·y.

Fuel type	Losses (kg)
▪ UOX	$14.1 \cdot 103 \cdot 0.0133 \cdot 10^{-3} = 0.187$
▪ MOX	$2.31 \cdot 103 \cdot 0.0552 \cdot 10^{-3} = 0.128$
▪ FR	$1.04 \cdot 10^{3} \cdot 0.3222 \cdot 10^{-3} = 0.335$
▪ ADS	$407.2 \cdot 0.4092 \cdot 10^{-3} = 0.166$
TOTAL	0.816

Economics

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.37.

Table 3.37 CEI for double strata strategy – ADS (mills/kWh)

	LWR-UOX	Tier 1 and Tier 2	Overall
CEI_R	30.18	48.83	33.33
CEI_{FC}	4.06	9.02	5.87
CEI	34.24	47.86	39.20

3.2.5.2 Critical Minor Actinide Burner

While the ADS is widely viewed as the system of choice for actinide burning from the point of view of safety and control, its high capital cost makes it economically unattractive. For this reason, the possibility of using critical systems as minor actinide burners is receiving growing attention. A. Romano [36] has proposed a 2-tier system where, in tier 1, the plutonium discharged from LWRs is incinerated in deep burn GT-MHRs and, in tier 2, the minor actinides from the LWRs and the transuranics from the GT-MHRs are destroyed in a critical minor actinide burner reactor (MABR).

Mass Flow Analysis

Assumptions for the GT-MHR and MABR are shown in Table 3.38.

Table 3.38. Assumptions – GT-MHR (deep burn) and MABR

	GT-MHR (deep-burn)	MABR
Thermal efficiency	47%	40%
Burnup (GWd/MTHM)	700	200
Fuel Composition ▪ Fresh ▪ Spent	 100% Pu 70% FP, 15% MA, 15% Pu	 100% TRU 20% FP, 40% MA, 40% Pu

The ratio of LWRs to GT-MHRs is determined by equating the mass of plutonium recovered from the reprocessing of spent UOX from LWRs with the mass of plutonium needed for fuel fabrication for the GT-MHR. Using Eq. 6:

$$\frac{f_{GT-MHR}}{700 \cdot 0.47} = 0.999 \cdot \frac{f_{LWR}}{50 \cdot 0.33}$$

Eq. 14:
$$\frac{f_{LWR}}{f_{GT-MHR}} = 3.77$$

Equating the mass of TRU required for MABR fuel fabrication with the mass of TRU recovered from all reprocessing operations:

Eq. 15:
$$\frac{f_{MABR}}{200 \cdot 0.40} = 0.999 \cdot \left(\frac{f_{LWR}}{50 \cdot 0.33} \cdot 0.0012 + \frac{f_{GT-MHR}}{700 \cdot 0.47} \cdot 0.3 + \frac{f_{MABR}}{200 \cdot 0.40} \cdot 0.8 \right)$$

Substituting Eq. 14 in Eq. 15, the ratio of GT-MHRs to MABRs is obtained:

Eq. 16:
$$\frac{f_{GT-MHR}}{f_{MABR}} = 2.12$$

Using Eq. 14 and Eq. 16, we find the composition of the nuclear park for this fuel cycle:

$$f_{LWR} = 0.0.719; \quad f_{GT-MHR} = 0.191; \quad f_{MABR} = 0.090$$

The mass of fuel irradiated in each type of reactor is determined using Eq. 6. Using fuel compositions shown in Table 3.38, the content of the various types of spent fuel in the double strata scheme can be determined. Results are summarized in Table 3.39.

Table 3.39. Spent fuel mass flows (double strata strategy – MABR)

Material (/GWe·y)	LWR	GT-MHR	MABR	TOTAL
Spent Fuel (MTHM)	15.91	0.211	0.411	16.53
U (MT)	14.86	-	-	14.86
FP (kg)	819.6	148.0	82.1	1,049.7
Pu (kg)	211.7	31.7	164.2	407.6
MA (kg)	19.1	31.7	164.2	215.1
TRU (kg)	230.7	63.4	328.5	622.7

Assuming a recovery of 99.9%, the mass of plutonium recovered from spent UOX is 211.5 kg/GWe·y. The mass of minor actinides recovered is 19.1 kg/GWe·y. The mass of TRU recovered from pyroprocessing is equal to 99.9% of the total mass of TRU in spent GT-MHR and spent MABR fuels, or 391.5 kg/GWe·y.

All the fission products generated in the fuel cycle, 1,049 kg, are discharged in HLW. In addition, 0.1% of the total mass of transuranics going through reprocessing (622.7 kg/GWe·y), or 0.62 kg/GWe·y, is lost and discharged as HLW. The fuel cycle and its material flows are represented in Figure 3.13.

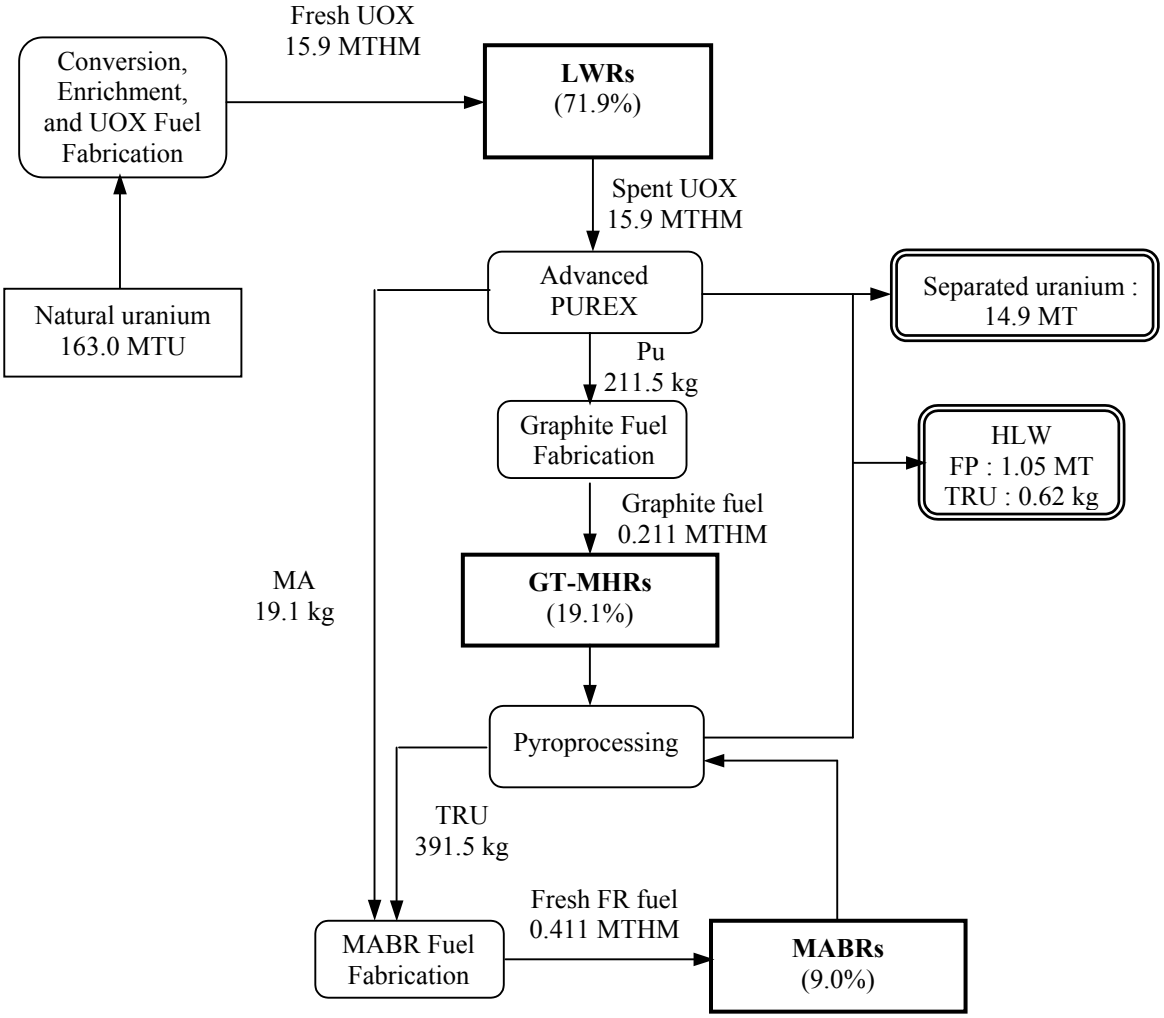

Figure 3.13. Double strata fuel cycle – MABR
(all quantities are per GWe·y)

Proliferation

The plutonium working inventory, assuming a storage time of 6 months, is 105.5 kg/GWe. This is about three times lower than for the double strata variant previously considered. The transuranics working inventory is 205.5 kg/GWe, or about 30% higher than in the previous case. The total reprocessing flow rate is 16.5 MTHM/GWe·y, only slightly lower than for the ADS double strata cycle. The amount of plutonium going to the repository, obtained by multiplying the reprocessing mass flow rate for each type of fuel by its plutonium content and the reprocessing loss factor (10^{-3}), is 0.41 kg/GWe·y. This is about 50% lower than for the ADS double strata fuel cycle, which was already minute.

Economics

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.40.

Table 3.40 CEI for double strata strategy – MABR (mills/kWh)

	LWR-UOX	Tier 1 and Tier 2	Overall
CEI_R	30.18	32.45	30.82
CEI_{FC}	4.06	8.95	5.43
CEI	34.24	41.40	36.25

3.2.6 Actinide Recycle in Molten Salt Reactor

The molten salt reactor (MSR) concept was extensively investigated at the beginning of the nuclear era, particularly with the molten salt reactor experiment (MSRE) and the molten salt breeder reactor (MSBR) at ORNL [37, 38, 39], when it was recognized as having a potential for very effective consumption or transmutation of nuclear fuel as well as several inherent safety features. Molten salts operate at high temperatures and low pressures and have favorable heat transfer properties, which result in a high thermal efficiency and elimination of the hazards associated with high pressures. The fluoride based salts, which are used in most MSR concepts, are chemically stable and non-flammable, averting fire hazards and energetic chemical reactions with water. In addition, they offer the potential for integrated fuel reprocessing, based on the high volatility of UF_6: by sparging the salt with fluorine, uranium can be removed as UF_6, and recycled into a fresh batch of fuel salt. The residual salt can be subjected to various processes to remove fission products, and the various carrier components of the salt can be recycled.

MSR development efforts have mostly focused on achieving effective use of nuclear resources through the introduction of thorium into the fuel cycle. Today, however, several investigation efforts are under way to explore the potential of molten salt reactors as transmuters of nuclear waste. Although the question of whether a thermal or fast neutron spectrum is preferable for incineration of transuranics is unresolved, the thermal spectrum has at least one clear advantage: because of the larger cross sections, the inventory of transuranics is much lower in thermal spectrum burners, which means that the time to reach equilibrium is greatly reduced. In addition, the inventory left behind after decommissioning of the burner represents a lesser burden. The molten salt reactor is an attractive option for a thermal spectrum transuranic burner because the use of liquid fuel and online reprocessing eliminates the complications associated with the accumulation of higher actinides.

AMSTER

Of the MSR transmuter concepts currently under investigation, AMSTER (actinide molten salt transmuter) is perhaps most closely related to the early ORNL designs. This critical thermal spectrum reactor uses the same salt composition as the MSBR and can be considered as revisiting the principles of the ORNL molten salt reactors, but with a focus on TRU incineration rather than breeding. To minimize the TRU losses associated with reprocessing operations, the molten salt inventory of AMSTER is reprocessed only once every 3,000 days (equivalently, only 1/3,000 of the molten salt is reprocessed per day). As a result, the fission product content of the molten salt is relatively high, leading to important parasitic neutron captures. Because AMSTER uses a thermal neutron spectrum, its neutron economy is relatively tight and it must rely on a U-235 support to maintain reactivity. This seriously hampers the transuranic incineration rate of the reactor because a significant fraction of the core power comes from fission of U-235 nuclei rather than transuranic nuclei. Indeed, the transmutation rate that can be obtained in a AMSTER reactor optimized for TRU incineration is 105 kg/GWe·y [40]. Recalling that the TRU

production rate of LWRs is 320 kg/GWe·y, the AMSTER:LWR ratio in a balanced fuel cycle is 3.05:1. In other words, 75% of the installed electric capacity in a nuclear park with no net generation of TRU would be based on AMSTER reactors, while LWRs would account for only 25%. Thus, the AMSTER concept cannot fulfill the role of transmuter in a nuclear park dominated by LWRs.

MSTB

A molten salt reactor concept that offers a very high transuranic incineration rate is the MSTB (molten salt thermal spectrum burner). The MSTB is an accelerator driven system that does not use any uranium or thorium support. The transmutation rate obtained is thus equal to the theoretical maximum because the core power is derived entirely from the fission of transuranic nuclei [40]. Thus the MSTB can be an effective TRU burner in a nuclear park dominated by LWRs. In comparison to a fast spectrum critical burner, however, the MSTB has an important drawback: to maintain reactivity in the absence of uranium or thorium support, the molten salt inventory must be processed very frequently to remove fission products which act as neutron poisons. Indeed, the MSTB fuel residence time is only 30 days, meaning that 1/30 of the core is recycled every day. This leads to reprocessing losses that are more than an order of magnitude greater than for fuel cycles that rely on fast spectrum burners, as will be shown below.

2 Tier System of Molten Salt Reactors

The AMSTER and MSTB concepts illustrate the tradeoffs required if a thermal spectrum system is used for transmutation of waste: if no uranium (or thorium) support is used, an external neutron source and frequent reprocessing are required, requiring more advanced technology and leading to increased transuranic losses. With a uranium feed, these problems are alleviated but the TRU destruction rate is limited, making such systems incapable of achieving zero net TRU production in a nuclear park dominated by LWRs. There are, however, a few authors who put forward thermal spectrum transmutation schemes that are, they claim, superior to systems based on a fast spectrum. One of the most notable proponents of thermal spectrum transmutation is Charles Bowman, who proposes a 2 tier transmutation system in accelerator driven molten salt reactors. This concept will be briefly reviewed.

Bowman's approach calls for a 2 tier system for transmutation of the transuranics discharged by LWRs (see Figure 3.14 [42]): Tier-1 consists of a once-through pass that eliminates a substantial fraction (>75%) of the transuranics, while Tier-2 involves full recycle of the transuranics and reduces the transuranics discharged from Tier-1 by a factor of several hundred. For both Tier-1 and Tier-2, the transmuters are 750 MWth (270 MWe) accelerator driven, graphite moderated molten salt reactors. The Tier-1 and Tier-2 systems differ mainly in the composition of the salt and the transuranic inventory: for Tier-1, a $NaF\text{-}ZrF_4$ salt is used and the actinide inventory is 193 kg; for Tier-2, the carrier salt is $LiF\text{:}BeF_2$ and the actinide inventory is 1126 kg. The Tier-1 system has been presented in considerable detail in the literature [43, 44] and, according to the most recent estimates [45], the mass of transuranics discharged from LWRs can be reduced by

75.5%[1] in a single-pass through Tier-1 units. Details on the Tier-2 system are more limited [42, 45], but the most recent information available states that the transuranics discharged as reprocessing losses from Tier-2 amount to only 0.062% of the transuranics in the spent fuel discharged from the LWRs. This is accomplished in a nuclear park with about 0.755 Tier-1 units and 0.245 Tier-2 units for every 1000 MWe LWR. Note that this is based on a transuranic discharge rate of 300 kg/y for the LWR and a destruction rate of 300 kg/y for a single 750 MWth transmuter[45][2]. If it is assumed there are transuranic losses of 0.1% during recovery of actinides from spent LWR fuel and again during recovery of actinides from spent Tier-1 salt, the total transuranic losses for this cycle are $300 \times 0.001 + 300 \times 0.245 \times 0.001 + 300 \times 0.00062 = 0.56$ kg per year. Since the total electric capacity is 1 GWe + 0.755×0.270 GWe + 0.245×0.270 GWe = 1.270 GWe, the transuranic discharge for the fuel cycle is 0.44 kg/GWe·y. This compares favorably to fast reactor transmutation and the double strata strategy, which are both based on fast spectrum transmuters. In addition, the thermal transmuters represent only 22% (0.27 GWe/1.27 GWe) of the nuclear park's total installed capacity. Hence Bowman's claim is that, with the 2-tier approach, thermal spectrum transmuters can match the performance of fast spectrum transmuters since they can achieve zero net transuranic production in a nuclear park dominated by LWRs AND minimize transuranic losses from reprocessing operations. Because thermal spectrum systems have the additional advantage of requiring a much smaller transuranic inventory (due to larger cross sections in a thermal spectrum), Bowman claims that his approach is preferable.

The preceding discussion shows that it is difficult to assess the effectiveness of thermal spectrum transmutation because different authors make diverging claims concerning the performance of their systems. For instance the MSBT and the 2 tier systems both use accelerator driven molten salt systems that burn transuranics with no uranium or thorium support, but the performance of the latter in terms of transuranic losses from reprocessing operations is vastly superior. A close examination of the calculations behind these results would be required to elucidate the differences and establish a consensus. At the present time, however, the majority of researchers involved in this debate favor fast spectrum systems and Bowman's claims do not seem to be widely accepted.

[1] More detailed simulations have found that a 93% reduction was possible [45]. However, these results have not been yet been validated.

[2] This destruction rate is unrealistically high: indeed, the maximum mass of transuranics that can be fissioned in a 750 MWth system, assuming an energy yield of 1000 MWd/kgHM$_{\text{fissioned}}$, is 273.8 kg/y. Even if we assume an energy yield of 0.95 MWd/kgHM$_{\text{fissioned}}$, the maximum transmutation rate is still only 288 kg/y.

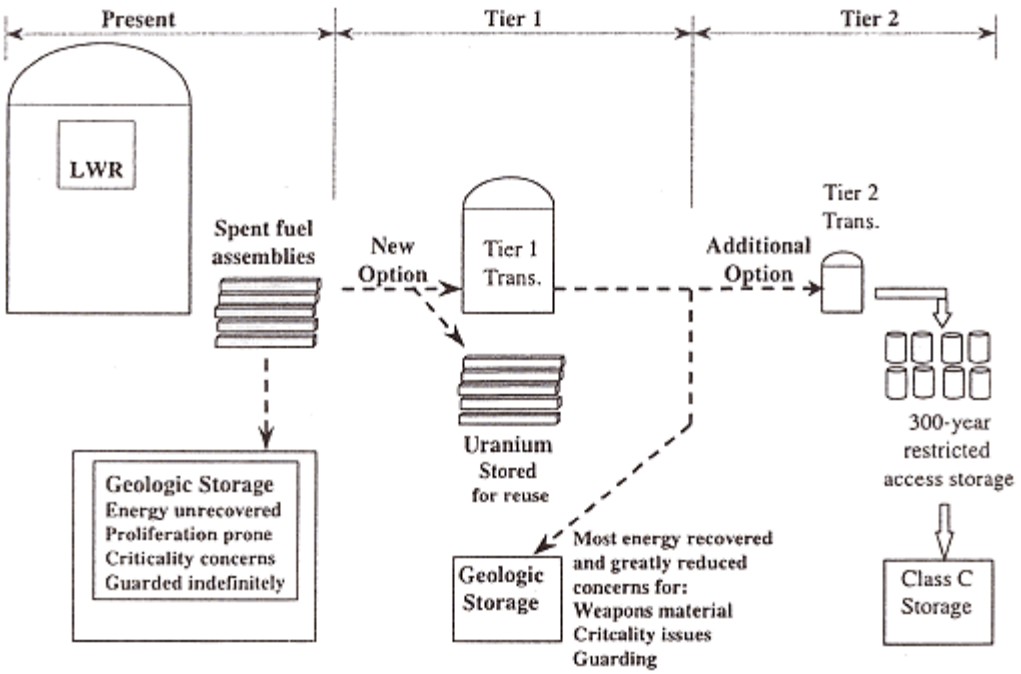

Figure 3.14. Charles Bowman's 2 tier thermal spectrum transmutation scheme

Fast Spectrum Molten Salt Reactor

Finally, it should be noted that molten salt reactors with a fast neutron spectrum are also being studied. JAERI is investigating a fast spectrum accelerator driven molten salt reactor which promises a high transuranic destruction rate without the drawbacks associated with the inferior neutron economy of a thermal spectrum[41]. Because of their neutron moderating ability, fluoride salts are not suitable for this concept and molten chlorides have been proposed (the mass number of Cl is about twice that of F). However, there are serious material challenges in chloride systems. Experimental data for various metals shows very poor corrosion resistance in a molten chloride environment (several orders of magnitude worse than fluorides). In addition, the chloride mixtures are less chemically stable than fluorides, presenting additional safety problems. Thus, for TRU burning in a fast spectrum, a molten salt ADS does not appear to be the most promising option and no further analysis will be presented here.

Mass Flow Analysis - MSTB

To establish the mass flows in a balanced LWR/MSTB fuel cycle, the ratio between the 2 reactor types must be established. This can be done by applying the mass conservation requirement on the MSTB. More specifically, the mass of transuranics supplied to the MSTB from reprocessed UOX fuel must be equal to the mass discharged from the MSTB as fission products and transuranic losses. The mass of spent fuel discharged from the

LWRs is given by Eq. 6, and recalling that spent UOX has a TRU content of 1.45%, we get:

$$\text{TRU supplied} = f_{LWR} \cdot \frac{365}{50 \cdot 0.33} \cdot 0.0145 \cdot 0.999$$

The mass of fission products discharged from the MSTB is equal to the transuranic incineration rate, which is 929.1 kg/GWe·y [40]. In a balanced fuel cycle, the mass of fission products discharged is:

Eq. 17: $$\text{FP discharge} = f_{MSTB} \cdot 0.9291$$

The MSTB transuranic inventory is 5.6 MT/GWe [40]. Given that this inventory goes through reprocessing every 30 days and that reprocessing losses are 0.1%, the transuranic losses from the MSTB are obtained as:

Eq. 18: $$\text{TRU losses} = f_{MSTB} \cdot 5.6 \cdot \frac{365}{30} \cdot 0.001$$

Applying mass conservation:

$$\text{TRU supplied} = \text{TRU losses} + \text{FP discharge}$$

$$f_{LWR} \cdot \frac{365}{50 \cdot 0.33} \cdot 0.0145 \cdot 0.999 = f_{MSTB} \cdot \left(5.6 \cdot \frac{365}{30} \cdot 0.001 + 0.9291 \right)$$

$$\frac{f_{LWR}}{f_{MSTB}} = 3.11$$

Thus in a balanced LWR/MSTB cycle, 75.7% of the capacity is based on LWRs while MSTBs account for 24.3%. Using Eq. 2 and Eq. 3, the mass of UOX fuel irradiated in LWRs is 16.74 MTHM/GWe·y, requiring 171.4 MTU of natural uranium. All the fission products, or 862 kg/GWe·y, and 0.1% of the transuranic elements, or 0.2 kg/GWe·y, present in spent UOX are discharged from the reprocessing plant as HLW. The transuranic and fission product discharge from the MSTB is obtained using Eq. 17 and Eq. 18 above with $f_{MSTB} = 0.243$. The transuranic loses are 16.8 kg/GWe·y, while the fission product discharge is 226 kg/GWe·y. A balanced LWR/MSTB fuel cycle is shown in Figure 3.15. Note that the mass of transuranics discharged in this case is about 20 times larger than in the case where fast reactors are employed as transuranic burners.

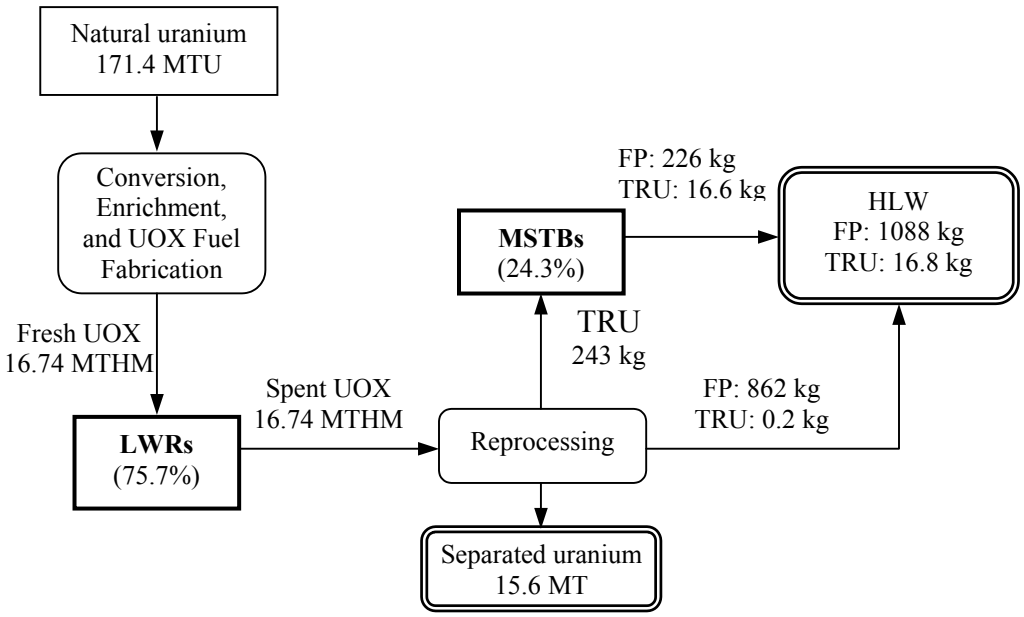

Figure 3.15. LWR/MSTB fuel cycle
(all quantities are per GWe·y)

Proliferation – MSTB

Because it uses liquid fuel online reprocessing, the molten salt reactor does not require separate fuel cycle facilities for reprocessing and fuel fabrication. In the fuel cycle considered here, the only transuranics that need to be separated are the ones generated by LWRs. The mass flow rate of transuranics is 243 kg/GWe·y, requiring a working inventory of 121.5 kg/GWe if a storage time of 6 months is assumed.

Because it relies on a thermal spectrum, the plutonium content of the transuranics sent to the repository is low (plutonium is displaced by higher actinides, mainly curium). Assuming a plutonium content of 15%[1], the plutonium discharge for this fuel cycle is 2.52 kg. Note that the heavy contamination of this plutonium with higher actinides provides an additional barrier against proliferation.

Economics - MSTB

Applying Eq. 1 with the applicable assumptions from Table 2.1 to determine the CEI for this fuel cycle gives the results shown in Table 3.41. The cost associated with online reprocessing of the molten salt is included in CEI_{FC} (the overnight construction cost of the molten salt reactor does not include the reprocessing plant). The high fuel cycle cost is due to the high reprocessing rate in the molten salt reactor.

[1] Bowman gives detailed isotopic composition data for his tier-2 molten-salt transmuters [45]. The molten salt at equilibrium contains 78.8% Cm, 13.3% Pu, 7.2% Am, and 0.5% Np.

Table 3.41. CEI assumptions for LWR/MSTB balanced fuel cycle (mills/kWh)

	LWR-UOX	MSTB	Overall
CEI_R	30.82	37.28	31.90
CEI_{FC}	4.06	15.63	6.87
CEI	34.24	52.91	38.77

4 Outlook for Mid-Century Deployment

4.1 Assessment of fuel cycles

In this section, the fuel cycles considered in this study will be evaluated and compared on the basis of the analysis presented in chapter 3. Table 4.1 (at the end of this chapter) summarizes the results of this analysis and presents the values of the metrics for each of the fuel cycles.

4.1.1 Uranium utilization

Light water reactors on the once-through UOX cycle utilize uranium resources poorly. Indeed, whereas they consume 226.5 MTU/GWe·y, advanced systems could fare much better in this regard. The energy yield of fission is 1000 GWd/MTHM, so the maximum energy yield of a system with a thermal efficiency of 0.33 is 333 GWe·d/MTHM. This corresponds to a consumption rate of roughly 1.1 MTHM/GWe·y. In other words, a fuel cycle that could extract all the energy from uranium with a thermal efficiency of 0.33 would require only 1.1 MTU/GWe·y (a higher thermal efficiency would make further gains possible) Thus, LWRs on the once-through cycle extract only about $1/200^{th}$ of the energy that could potentially be obtained from uranium. The GT-MHR offers no improvement in uranium utilization in spite of its higher thermal efficiency. PHWRs fare somewhat better, utilizing roughly $1/100^{th}$ of uranium's energy potential. The ACR-700, however, consumes uranium at a rate which is practically equivalent to that of LWRs. Note also the very limited uranium savings afforded by single-pass recycling in the MOX and DUPIC fuel cycles. Thus, currently deployed fuel cycles and candidates for near-term deployment do not use uranium resources efficiently.

In the past 2 decades or so, the purpose of reprocessing in advanced fuel cycles has shifted from uranium resource extension to nuclear waste management. This change was brought about by significant uranium ore discoveries, which have driven prices down and contributed to dissipating fears of uranium shortages, and by growing concerns about the long-term environmental impact of nuclear waste disposal. For this reason, fuel cycles that offer significant gains in uranium utilization but only limited improvements in waste management, such as plutonium recycle in fast breeder reactors, are no longer seen as the next step in the evolution of nuclear energy. In this study, the advanced fuel cycles considered all involve a large deployment of LWRs because these reactors (and other once-through systems with similar characteristics) will still account for the majority of nuclear power plants 50 years from now. The fraction of the nuclear park taken up by the advanced systems is minimized, and their purpose is to transmute the transuranics discharged from LWRs, which represent only a small fraction of LWR spent fuel. Uranium, which constitutes the bulk of the discharged fuel, is not recycled. As a result, these advanced fuel cycles do not offer significant gains in uranium utilization efficiency.

To conclude, currently deployed reactors and their most likely successors are inefficient in their use of uranium. Furthermore, advanced fuel cycles that aim to eliminate transuranics discharged by these reactors offer very modest gains in this regard.

Therefore, if uranium availability becomes a concern and significant improvements in uranium utilization are required, these reactors will have to be phased out and replaced by advanced designs (e.g. fast breeder reactors or molten salt reactors with thorium fuel). This is an unlikely prospect for the middle of the current century, since the next generation of reactors to be built will likely consist of LWRs, PHWRs, and possibly HTGRs. It should be noted, however, that uranium resources are sufficient to support even a large expansion of nuclear power based on such systems, as shown in appendix A.

4.1.2 Waste

Inspection of Table 4.1 reveals that fuel cycles fall into 2 categories with respect to the rate of discharge of transuranic elements: for once-through and single-pass recycle fuel cycles, the TRU discharge rate is the range of 100~300 kg/GWe·y; for fuel cycles involving full actinide recycle, the rate is on the order of 1 kg/GWe·y (with the exception of the MSTB, which will be discussed below). Since these advanced fuel cycles were modeled with LWRs discharging transuranics at a rate of 320 kg/GWe·y, it can be said that actinide recycle strategies can potentially reduce the mass of transuranics going to the repository by a factor of about 300.

Differences in the transuranic discharge rate between various transmutation strategies depend mainly on the burnup of the transmuter fuel and on the fuel's uranium content. A high burnup is desirable for transmuter fuel because it minimizes the number of reprocessing passes required per energy generated. The uranium content of transmuter fuel should be as low as possible because the transuranics created by neutron capture in uranium nuclei must also go through reprocessing and thus contribute to transuranic losses. The transuranic losses for the MSTB are high because the residence time of the molten salt is very short for this reactor[1]. Conversely, the double strata fuel cycle with the GT-MHR and MABR has low transuranic losses because transmutation is accomplished with uranium free fuels that reach very high burnups. It must be noted that the fractional reprocessing losses are also an important factor, but, in this analysis, they have been assumed equal to 0.1% in all reprocessing operations and therefore do not account for any differences between the fuel cycles.

The fission product discharge rate is affected only by the thermal efficiency of the power plants. Indeed, the energy released through fission is roughly equal to 1,000 GWd/MTHM for any type of heavy metal nucleus and any neutron spectrum. Thus the mass of fission products generated in a nuclear fuel cycle can be approximated as $365/\eta_{th}$, where η_{th} is the average thermal efficiency of the fleet. The fission product discharge rate for advanced fuel cycles involving fast reactors is slightly lower than that of LWRs because the fast reactors contribute to increasing the average thermal efficiency of the fleet. The HTGR allows the most significant reduction in the fission product discharge rate due to its very high operating temperature and correspondingly high thermal efficiency. But reductions in fission product discharge rates through increased thermal

[1] Low residence time for liquid fuel is equivalent to low burnup. For the MSTB, the TRU inventory is 5.6 MT/GWe and the residence time is 30 days, which means that the burnup is (1/0.4·30)/5.6 = 1.3 GWd/MTHM.

efficiency are limited because thermal efficiencies are not likely to go beyond 50%, which means that fission product discharge rates could at best be reduced by a factor of about 1.5 as compared to current LWRs. As section B.2 of appendix B demonstrates, there is no clear advantage between thermal and fast reactors for fission product transmutation. Therefore, on the basis of this analysis, no fuel cycle is clearly superior in minimizing fission product discharge rates.

4.1.3 Proliferation

The separation of plutonium from spent fuel is widely viewed as the most dangerous activity associated with commercial nuclear fuel cycles. Single-pass plutonium recycling requires a plutonium working inventory of 126 kg/GWe. To put this in perspective, a total installed nuclear capacity of 100 GWe, comparable to that of the U.S, would require a working inventory of 12.6 metric tons of plutonium. This quantity of plutonium is sufficient to make over a thousand nuclear weapons. The double strata strategy also requires separation of plutonium because the first stratum consists of plutonium burners. Plutonium inventories for the ADS based and MABR based double strata schemes are 326 kg/GWe and 106 kg/GWe, respectively. Thus, the proliferation issues associated with today's MOX fuel cycle will also cause problems for double strata schemes, should they ever be deployed. Furthermore, because of their higher complexity, it could prove even more difficult to balance plutonium production and consumption in such fuel cycles. As of 2002, imbalances between plutonium production and consumption in the MOX fuel cycle have led to the accumulation of over 200 metric tons of plutonium worldwide.

The plutonium discharge to the repository also raises proliferation concerns because the repository is effectively a plutonium mine. Once-through cycles discharge over 100 kg/GWe·y of plutonium. In particular, LWRs operating on the once-through fuel cycle send 294 kg/GWe·y of plutonium to the repository. Hence the repository for a fleet of 100 GWe operating for 50 years will contain about 1,500 metric tons of plutonium. This quantity is enormous compared to the 10kg or so needed to make a weapon. For this reason, fuel cycles such as single-pass plutonium recycling and even the Radkowsky thorium reactor, which do not reduce the plutonium discharge rate by even a single order of magnitude, cannot be said to offer a significant advantage in this regard. It must be noted, however, that the evaluation metrics used in this study do not take into account the isotopic composition of plutonium. Generally, the plutonium discharged from high burnup fuels is less suitable for weapons because it contains more even isotopes, in particular Pu-238, a prolific neutron emitter and a significant decay heat source. Spent MOX and spent WASB fuel contain high concentrations of Pu-238, but because the half-life of Pu-238 is 88 years, this barrier against proliferation lasts for only one or two hundred years after discharge.

Fuel cycles involving multi-pass recycling of transuranics send only very small amounts (~1kg/GWe·y) of plutonium to the repository. The plutonium is thus extremely dilute in the HLW and extraction of as significant quantity would require processing large amounts of radioactive waste, a difficult task for an illicit group. Furthermore, the isotopic composition of this plutonium is highly degraded due to the multiple high

burnup irradiations. Hence the proliferation risks associated with the repository is greatly reduced in such fuel cycles.

Most recent proposals for fuel cycles involving actinide transmutation call for reprocessing technologies that co-extract transuranic elements from spent fuel, so that no separated plutonium is ever produced. While a nuclear weapon could be made from a mixture of transuranics, it is conceded that it would be considerably more difficult and that a crude device would have a very low yield. The material could also be used in a radiological dispersion device (RDD), where conventional explosives are used to disperse the nuclear material. Actinide recycle in LWRs and in FRs are the two fuel cycles that require the largest transuranic working inventories, well above 300 kg/GWe·y. However, the double strata strategy fares better in this regard, but only because it involves separation of plutonium in the first stratum, thus reducing the transuranic inventory in the second stratum. Thus the reduced transuranic inventory should not be claimed as an advantage for these fuel cycles. The molten salt reactor presents lowest transuranic working inventory because the online processing in the reactor only involves extraction of fission products. The actinides remain in the liquid fuel at all times.

As noted previously, multi-pass actinide recycle allows a 300-fold reduction in the transuranic discharge rate as compared to once-through or single-pass recycle fuel cycles. As a result, the transuranics in the repository will be extremely dilute and difficult to extract. However, this must be weighed against the fact that these fuel cycles require actinide partitioning, thus producing proliferation risks in the short term.

Reprocessing rates differ little among the fuel cycles examined because, in all cases, the spent fuel discharged from LWRs must be reprocessed. Thus there will be practically no difference between the fuel cycles in terms of the spread of technologies that could be used to recover weapons material. Pyroprocessing, required for advanced fuel cycles due to the high burnup and short cooling times of transmutation fuels, presents fewer proliferation risks because it does not involve separation of plutonium. This advantageous feature is reflected in the *plutonium working inventory* metric, which is zero for fuel cycles that rely on pyroprocessing only. Reprocessing operations that involve no separation of actinides altogether present the smallest proliferation risk. In Table 4.1, numbers in parentheses designate processes where actinides are not extracted. The oxidation-reduction process involves no dilution of the spent fuel and no extraction of any of the heavy metals (only volatile fission products are removed), and thus does not contribute to the propagation of knowledge related to weapon production. As for actinide transmutation in a molten salt reactor, this fuel cycle requires actinide extraction from spent LWR fuel and thus the fact that the online reprocessing of the molten salt does not involve actinide extraction cannot be claimed as a significant advantage.

4.1.4 Economics

Nuclear electricity costs are difficult to determine. Even for currently deployed reactors, the construction and O&M costs are difficult to ascertain [46]. The same holds true for their associated fuel cycle facilities, as shown in appendix C. In this study, where advanced reactors and fuel cycles were considered, uncertainties associated with the analysis of nuclear electricity costs were even greater. For this reason, the cost of electricity index (CEI) is intended as an approximate indicator of costs only, and no conclusions should be drawn based on small CEI differences between the fuel cycles.

Inspection of Table 4.1 reveals that the cost of nuclear electricity is dominated by reactor construction costs. Fuel cycle costs are not negligible, but they are only relevant when comparing fuel cycles based on reactors with similar construction costs. For instance, the comparison between the fuel cycle cost of single-pass plutonium recycling and the once-through cycle is extremely significant because both fuel cycles rely on LWRs. This analysis reveals that single-pass plutonium recycling is not an economically attractive option (due to the high fuel cycle costs associated with reprocessing and MOX fabrication). In general, the fuel cycle cost of once-through schemes is lower than that of fuel cycles involving reprocessing of spent fuel.

For the above reasons, once-through fuel cycles that rely on conventional reactors present the lowest CEI values (The DUPIC fuel cycle is also relatively attractive economically because it relies on PWRs and PHWRs and the type of reprocessing involved is projected to be less costly than aqueous reprocessing or pyroprocessing). The CEI of these fuel cycles is very close because the assumed overnight construction cost is identical for all thermal spectrum reactors (except molten salt reactors). This assumption is necessary because the future construction cost for the various reactors cannot by established reliably. It is important to note, however, that some vendors estimate the overnight construction costs of next generation reactors at around $1,000/kWe, well below the value assumed in this study. If such claims can be realized in practice, the cost of nuclear electricity can be expected to drop dramatically. Furthermore, the differences in overnight construction cost that will emerge between various reactors could play a decisive role in their commercial success. It was noted, for instance, that a reduction in the overnight construction cost of HTGRs commensurate with their increased thermal efficiency would reduce the CEI by about 10 mills/kWh, making them significantly more attractive economically. Because of such uncertainties, the results of the current study should not be used to discriminate among the various once-through fuel cycles; indeed, the differences in CEIs are small compared to the uncertainties, namely those associated with reactor costs.

The CEI of the various fuel cycles involving reprocessing are falls between 40 and 44 mills/kWh, or roughly 3 to 7 mills/kWh more than the once-through cycles (except for the DUPIC fuel cycle, as noted previously). This rather modest difference is due to the fact that LWRs running on the once-through cycle account for the majority (60% or more) of installed capacity in these fuel cycles. This also explains why the variations in the CEI between the different closed fuel cycles are fairly small. Table 4.1 also shows the

CEI for the transmutation component of these fuel cycles, which is significantly higher. This indicates that there is no economic incentive to engage in reprocessing and transmutation. In other words, the value of the electricity produced in transmutation is insufficient to justify the costs associated with the required reprocessing, advanced fuel fabrication, and construction of advanced reactors. Therefore, without government intervention to create additional incentives, transmutation will not be pursued by the nuclear industry.

The double strata strategy using deep-burn GT-MHRs and MABRs appears to be the most cost effective transmutation strategy due to its relatively low reactor costs (only 9% of installed capacity is based on MABRs). LWR transmutation is also cost effective since it does not rely on fast reactors at all, but its fuel cycle costs are slightly higher. Transmutation in fast reactors is more expensive because of the large deployment of advanced reactors required. The double strata strategy with ADS does not appear cost effective due to the significant deployment of fast reactors (21% of installed capacity) and prohibitively high costs of the ADS, even if these systems represent only about 5% of installed capacity. Transmutation in molten salt reactors may well be the least cost effective option, since the CEI is high in spite of the favorable assumptions that were made regarding reactor and reprocessing costs. Indeed, the overnight construction cost of the reactor was assumed equal to that of fast reactors even though the latter represent a more mature technology. In addition, the cost of molten salt on-line reprocessing was assumed equal to the cost of conventional UOX reprocessing even though current reprocessing plants benefit from economies of scale due to their large size and employ a well-known process.

Finally, it should be noted that the reactor cost component for all fuel cycles based on advanced reactors may well be underestimated in the present calculations. Indeed, the assumption for the overnight construction cost of these reactors, $2,100/kWe, taken from [24], has been criticized as overly optimistic. Indeed, this figure is only 25% higher than the value used for LWRs, but it has been suggested that this margin should be increased to 50% [27]. In this case, the overnight construction cost of advanced reactors would rise to $2,550/kWe, resulting in an increase of roughly 10 mills/kWh in the cost of electricity generated by these reactors.

4.2 Outlook for mid-century deployment

Once-through

As of 2003, there are 390 LWRs and PHWRs under operation and another 32 under construction around the world. Thus it seems clear that these reactors will continue to provide a large fraction of nuclear electricity in the decades to come and will constitute an important part of the nuclear fleet at mid-century. The experience gathered over the past 50 years in building and operating these reactors in many ways compensates for their weaknesses compared to more advanced concepts. Operated on the once-through cycle, our analysis shows that these reactors are among the most economical. Their inefficient use of uranium does not harm their economic performance significantly and is not likely to become a concern before the end of the century. Furthermore, the once-through fuel cycle does not require any reprocessing and partitioning operations and therefore poses minimal proliferation risks. The greatest liability of the once-through cycle is that it sends large quantities of plutonium and minor actinides to the repository (of course, fission products are not partitioned either, but since their potential for transmutation appears very limited at this time, this does not weigh heavily against the once-through cycle). Thus the long-term risk associated with waste disposal is of great concern and is proving to be a significant obstacle to the continuing development of nuclear energy in many parts of the world. In spite of these concerns, many countries are currently developing disposal strategies for spent fuel, and current estimates of the associated long-term risks appear acceptable. It can also be argued that the repository presents a significant proliferation risk since it contains large amounts of plutonium, but fuel cycles where plutonium is transmuted inevitably involve separation of plutonium and/or transuranics, which presents a greater proliferation risk in the short-term.

The performance of HTGRs as measured by the metrics used in this study appears comparable to that of LWRs and PHWRs. However, HTGRs may be more attractive economically due in part to their high thermal efficiency. If this potential to deliver nuclear electricity at a lower cost can be fulfilled, HTGRs could account for a major fraction of installed nuclear capacity by the middle of the century. HTGRs also have good safety features, a factor not considered in this study, which may make them an attractive option.

The most important problem in the way of nuclear expansion is that, in many parts of the world, building a nuclear reactor is a financially risky proposal. Reactor vendors assert that evolutionary LWRs, PHWRs as well as HTGRs will have low overnight construction costs ($1,000-1,500/kWe) and short construction times, and will produce electricity at costs below 4¢/kWh, but investors fear that these claims will not materialize in practice. This wariness is based largely on past experience, which has shown that the cost of building and operating a nuclear plant can be prohibitive. Thus the main task for the nuclear industry is to dispel the fears of investors, which will require demonstration projects, preferably headed by joint ventures of private investors rather than by government. With America, many parts of Europe, and Japan all liberalizing their power industry [47], nuclear power must be shown to be a viable option for private investors in

an unregulated market. Otherwise, its continued existence will be limited to countries where the power industry is controlled by the state.

Other developments in the once-through fuel cycle

Our analysis indicates that the fuel cycle cost associated with a burnup level of 100 GWd/MTHM is higher than for a burnup of 50 GWd/MTHM. In addition, expenses will be required for the development of fuels capable of sustaining such high levels of irradiation. Under current conditions, the benefits of high burnup (lower spent fuel and plutonium discharge rates, degraded plutonium isotopics) are not rewarded. Hence there is no incentive for nuclear power plant operators to invest in high burnup fuels.

The use of thorium in LWRs faces similar obstacles: the lower spent fuel and plutonium discharge rates and degraded plutonium isotopics afforded by this concept are not rewarded under the current system of nuclear waste management. Thus there is no incentive for nuclear plant operators to incur the expenses associated with developing thorium fuels and refitting LWR cores to accommodate seed and blanket assemblies.

With the current fixed charge of 1 mill/kWh, the cost of waste disposal, as perceived by nuclear plant operators, is independent of the fuel cycle chosen. Thus developments that may be desirable from the point of view of waste management will not be pursued if they are not economically attractive. Therefore, unless a waste management system that recognizes differences between fuel cycles and rewards advantageous waste characteristics is implemented, high burnup and thorium fuels are not likely to be adopted. In any case, it should be noted that the benefits from these fuel cycles, although not insignificant, are insufficient to change the prospects for nuclear energy considerably.

The breed and burn concept is interesting in that it has the potential to improve uranium utilization considerably without reprocessing, but uranium availability is unlikely to become an important issue by the middle of the century. Thus there is no justification, at least in the coming decades, for incurring the large expenses required for the development of this technology to commercial maturity, and it is an unlikely candidate for mid-century deployment.

Single-pass recycle

The MOX cycle offers unimpressive uranium savings and no significant advantage for waste management while presenting significant proliferation risks and being more expensive than the once-through option. Plutonium recycling was introduced in the 70's in order to enable multi-pass plutonium recycling in fast breeder reactors, an option which has since lost its impetus due to problems with the development of fast reactors and large uranium ore discoveries. But because of large commitments made in the past, reprocessing plants are still being used to extract plutonium from spent fuel. Without breeder reactors, irradiation of MOX fuel in LWRs is the only alternative for using this plutonium. Today, a handful of countries remain committed to reprocessing and a large

reprocessing plant (800 MTHM/y) has recently been completed in Japan, so plutonium recycling in LWRs may still be in use at mid-century.

The DUPIC fuel cycle is a plausible mid-century candidate as it promises to deliver electricity at a cost which is only slightly above that of the once-through cycle and offers interesting gains in waste management (recall from section 3.2.2 that the mass of spent fuel discharged by a PWR/PHWR nuclear park is 3 times lower if DUPIC is used). The most significant hurdle for DUPIC is the development of the OREOX process. Work in this area is being carried out by AECL of Canada and KAERI of South Korea and is making good progress, with several successful tests carried out so far. It must noted that DUPIC deployment will be limited to countries that have both LWRs and PHWRs. Korea would most likely be the first country to proceed with DUPIC, but other countries where both types of reactors are deployed, notably China and India, could possibly follow.

Multi recycle

The present analysis shows that the costs associated with actinide transmutation are high. The CEI for the various transmutation options are roughly 10 to 20 mills/kWh higher than for the once-through cycle in conventional reactors despite a relatively optimistic assumption regarding the overnight construction cost of advanced reactors. Furthermore, the development of the reactors and fuel cycle technologies required to enable transmutation is substantial. In order to justify these costs, the benefits of transmutation have to be clearly established. In the current context, the gains from transmutation are difficult to assess, as several key issues related to nuclear waste disposal are not well understood. For instance, while it is clear that actinides dominate the long-term in-situ radiotoxicity of spent fuel, their contribution to the overall risk from the repository is not clear, and some studies suggest that long-lived fission products are in fact more worrisome. Another issue is the balance between the potential reduction in long-term risk from the repository and the increased short-term risk presented by reprocessing operations. Again, the lack of understanding regarding long-term disposal of nuclear waste makes this matter difficult to resolve. Furthermore, the risk from reprocessing operations has not been extensively studied, so the performance of future facilities is extremely difficult to estimate (present facilities have been responsible for substantial releases to the environment, but their record in this regard is improving). Finally, the benefits from transmutation must be compared to those that could be obtained by other means, such as deep-borehole disposal, which may reduce risk by several orders of magnitude. If transmutation is not the most cost-effective measure for dealing with nuclear waste, then it should not be pursued.

Due to their high level of complexity, fuel cycles involving transmutation are more likely to be deployed in countries where the nuclear industry is controlled by the government. Indeed, it is more practical to coordinate the construction of the transmuters and fuel cycle facilities needed for transmutation schemes under a system of central command. In a private sector nuclear electricity industry, it would be possible to bring about such an outcome by setting up a nuclear waste fee that creates an incentive for transmutation. Of course, a fee should not be set up purely to encourage transmutation: it should be based

on a thorough analysis and should reflect the true cost of nuclear waste disposal. If transmutation is indeed worthwhile, then it will reflected in the fee and nuclear electricity generators will seek to reduce their costs by engaging in transmutation rather than disposing of spent fuel directly. Otherwise, transmutation will not be pursued because of the significant cost increment associated with reprocessing operations and deployment of transmuters. As discussed above, there are many issues surrounding nuclear waste disposal that need to be resolved and much work would be needed in order to establish a strong analytical basis for such a nuclear waste fee. Note that the 10~20 mills/kWh increment for transmutation identified in this study is very large compared to the current 1 mill/kWh fee for nuclear waste disposal. Thus a fee that would create an incentive for transmutation would have to penalize direct spent fuel disposal very heavily as compared to HLW disposal. Note also that transmutation would make nuclear energy less competitive in the electricity market.

Actinide transmutation does not appear to be a plausible option for mid century deployment. The present analysis indicates that transmutation is significantly more expensive than once-through fuel cycles. Furthermore, the benefits of transmutation are not well established at present, and the deployment of complex fuel cycle schemes presents practical challenges, especially in countries where the nuclear electricity industry is not nationalized.

Table 4.1. Comparative Table of Fuel Cycles

		1	2	3	4	5	6	7
Resource utilization								
▪ Uranium consumption	MTU/GWe·y	226.5	237.9	157.0	122.8	237.4	253.3	6.52
Waste								
▪ Transuranic discharge	kg/GWe·y	320	251	438	NA	128	120.8	685
▪ Fission product discharge	kg/GWe·y	1140	1140	1178	1178	774	1102	978
Proliferation								
▪ Plutonium inventory	kg/GWe	-	-	-	-	-	-	-
▪ Transuranic inventory	kg/GWe	-	-	-	-	-	-	-
▪ Plutonium discharge	kg/GWe·y	294	218	394	331	109	97.4	652
▪ Transuranic discharge	kg/GWe·y	320	251	438	347.8	128	120.8	685
▪ Reprocessing rate	MTHM/GWe·y	-	-	-	-	-	-	-
Economics[a]								
▪ Reactor cost index	mills/kWh	30.18	30.18	30.18	30.18	30.18	30.18	37.28
▪ Fuel cycle cost index	mills/kWh	4.06	4.13	2.69	2.15	4.11	5.34	1.19
▪ Cost of electricity index	mills/kWh	34.24	34.31	32.87	32.33	34.29	35.52	38.47

1. next-generation LWR once-through at 50 GWd/MTH
2. next-generation LWR once-through at 100 GWd/MTHM
3. next-generation PHWR with natural uranium
4. next-generation PHWR with slightly enriched uranium (1.2% U-235)
5. GT-MHR
6. WASB (seed and blanket thorium in LWR)
7. Breed and Burn

a. The cost of electricity index is subject to large uncertainties because the underlying calculations are based on reactor and fuel cycle unit cost assumptions which are highly uncertain. For example, [27] gives a range of +/- 25% for the construction cost of advanced reactors. Since reactor costs account for a large fraction of electricity costs, this suggests an error of roughly +/- 25% on the cost of electricity index for fuel cycles that rely on fast reactors. As another example of the uncertainties involved, note that the fuel cycle cost for the balanced UOX/MOX fuel cycle (6.12 mills/kWh) is about 50% higher than for the once-through UOX fuel cycle (4.06 mills/kWh). But, as shown in appendix C, favorable assumptions can make the two options nearly equal. This suggests that fuel cycle costs are also subject to significant error. Thus, CEI values must be interpreted in the proper context, and small differences between the various options are not conclusive.

Table 4.1 (continued)

		8	9	10	11	12	13	14
Resource utilization								
▪ Uranium consumption	MTU/GWe·y	194.8	176.8	181	135	144	163	171
Waste								
▪ Transuranic discharge	kg/GWe·y	212	227	0.7	0.9	1.04	0.62	16.8
▪ Fission product discharge	kg/GWe·y	1140	1146	1106	1050	1090	1050	1088
Proliferation								
▪ Plutonium inventory	kg/GWe	126	-	-	-	326	106	-
▪ Transuranic inventory	kg/GWe	-	-	356	439	165	206	122
▪ Plutonium discharge	kg/GWe·y	172	204	0.6	0.75	0.8	0.41	2.5
▪ Transuranic discharge	kg/GWe·y	212	227	0.7	0.9	1.04	0.62	16.8
▪ Reprocessing rate	MTHM/GWe·y	19.0	(17.3)	19.8	15.8	17.9	16.5	16.6(16.7)
Economics[a]								
Overall:								
▪ Reactor cost index	mills/kWh	30.18	30.18	30.18	33.06	33.33	30.82	31.90
▪ Fuel cycle cost index	mills/kWh	6.12	4.46	6.64	4.96	5.87	5.43	6.87
▪ Cost of electricity index	mills/kWh	36.29	34.63	36.69	38.02	39.20	36.25	38.77
Transmutation:								
▪ Technology		8*	9*	10*	11*	12*	13*	14*
▪ Capacity share		14.1%	22.0%	22.5%	40.6%	36.4%	28.1%	24.3%
▪ Reactor cost index	mills/kWh	30.18	30.18	30.18	37.28	38.83	32.45	37.28
▪ Fuel cycle cost index	mills/kWh	18.66	5.83	14.22	6.30	9.02	8.95	15.63
▪ Cost of electricity index	mills/kWh	48.84	36.01	44.40	43.58	47.86	41.40	52.91

8. Plutonium recycle in next-generation LWRs
9. DUPIC fuel cycle
10. Actinide recycle in next-generation LWRs
11. Actinide recycle in fast reactors
12. Double strata strategy – ADS
13. Double strata strategy – MABR
14. Actinide recycle in MSTB

8*. MOX fuel in next-generation LWR
9*. DUPIC fuel in CANDU
10*. FFF in next-generation LWR
11*. TRU fuel in FR
12*. 1st (Pu recycle in LWR and FR) and 2nd (ADS) strata
13*. 1st (GT-MHR deep burn) and 2nd (MABR) strata
14*. TRU liquid fuel in MSTB

5 Future Work

The analysis of nuclear fuel cycles is a complex task and a full scale effort would require means not available for the present study. Future work in this area could improve on this study by conducting a more thorough review of fuel cycle options and establishing a better basis for comparison among them.

The limited set of fuel cycles to be evaluated must be selected from the multitude of proposals found in the open literature. The selection must be made so that options that present a unique set of features and open distinct new possibilities are included, subject to the availability of sufficient data to carry out a thorough analysis. This task requires discernment and a thorough knowledge of fuel cycle technologies. Some articles in the open literature are written by researchers who are in effect promoting their concepts, and these authors often gloss over the less advantageous features of their systems. Ideally, the selection of representative fuel cycles should be based on a consensus between unbiased experts and, as much as possible, calculations should be repeated or at least reviewed to ensure that the various options are compared on a fair basis. This would help to settle debates such as the question of whether fast or thermal spectrum systems are preferable for transmutation.

The set of metrics used for evaluating and comparing fuel cycles in this study was rather limited and the analytical background required to interpret their significance was lacking in some areas. Future efforts would benefit from carrying out the analysis required to improve our understanding of fuel cycle issues. In addition, the set of metrics should be expanded so as to take into account other aspects of fuel cycle performance.

In the area of uranium utilization, the position adopted in this study, presented in appendix A, is that uranium resources are not likely to be a concern before the middle of the century even under a high growth scenario. Thus uranium utilization was not an important factor in the assessment of fuel cycles and their potential for mid century deployment. These considerations also influenced the selection of fuel cycles; for instance, fast breeder reactors were not considered since it was a foregone conclusion that their efficient utilization of uranium did not justify the costs associated with the deployment of these reactors and the requisite fuel cycle facilities. However, future efforts should rely on a thorough assessment of worldwide uranium resources, so that the significance of uranium utilization can be put in a clear perspective.

The evaluation of the waste management aspects of fuel cycles is difficult at present because the impact of various components of nuclear waste on cost of disposal and long-term dose risk to the public is not well understood, both for mined repositories and other waste disposal methods such as deep boreholes. An important issue that must be resolved in order to properly assess the effectiveness of transmutation is the relative importance of actinides and fission products on heat load and long term dose risk from nuclear waste disposal. With regard to decay heat, the question of whether repository capacity is limited by the short term effect of the fission products or the long-term effect of the actinides must be resolved. With regard to dose risk, actinides dominate in-situ radiotoxicity but

fission products have a higher mobility and can therefore escape the repository more readily. The relative contribution of each to the dose risk must be clearly sorted out. Until such issues are better understood, it will be difficult to properly assess the effectiveness of transmutation schemes. Future fuel cycle studies would benefit enormously from further analysis in this area, as it would lead to a more focused evaluation and a clear interpretation of results.

Proliferation aspects of fuel cycles are difficult to quantify. There are several fuel cycle operations that present various degrees of proliferation risks, and the assessment of the overall proliferation risk of a fuel cycle must take all of them into account. This study focused on only a handful of fuel cycle characteristics, and no attempt was made to combine them into a single measure of proliferation. The task of developing a comprehensive measure of proliferation risk would require considerable judgment and would have to be undertaken by a group of experts, but such an effort would be extremely valuable in future fuel cycles studies.

The economic performance of fuel cycles is difficult to evaluate because of the large uncertainties involved. Reactor costs and unit costs of fuel cycle operations used should be based on a balanced and realistic consensus of experts rather than on the optimistic claims made by proponents of particular options, which are pervasive in the open literature. An evaluation of electricity cost based on such cost estimates would still be subject to considerable uncertainty, but the elimination of biases would allow a fair and consistent comparison of fuel cycles.

Safety was not considered in this study, but it is an important factor in determining the success of nuclear fuel cycles. Unfortunately, the safety performance of reactors and nuclear fuel cycle facilities is difficult to evaluate, especially for advanced systems for which no detailed designs are available.

Finally, future fuel cycle studies would have to address practical issues regarding deployment of advanced fuel cycles. For instance, many fuel cycles being proposed today are extremely complex (e.g. double strata fuel cycle, see Figure 3.12) and their deployment in unregulated electricity industries is very unlikely. Another practical issue in some advanced fuel cycles is the extremely long time scales involved; for example, curium storage for periods ranging from 30 to 360 years has been proposed for LWR transmutation. Such long planning periods involve considerable uncertainties and it simply cannot be assumed that developments will go as planned over such long periods. Current studies largely ignore these issues, but in fact they have a major impact on the viability of fuel cycles. Thus there is a need to inject pragmatism in fuel cycle studies, and options that present significant practicality issues should be penalized.

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Appendices

A Uranium Resources

Uranium resources and reserves

The most authoritative source for estimates of uranium resources is the OECD/IAEA Red Book [1]. Figures from the latest edition are shown in Table A.1.

Table A.1. Red Book Conventional Uranium Resources

(million metric tons, as of January 2001)

Known Conventional Resources			Reported Undiscovered Conventional Resources	
Cost Ranges			Cost Ranges	
<40$/kgU	40 – 80$/kgU	80-130$/kg	<130$/kgU	Cost Range Unassigned
2.1	1.0	0.8	6.8	5.5
Total Uranium Resources: 16.2				

The term "reserves" refers to the known conventional resources that can be extracted using current technology under current economic conditions at various recovery costs. For example, from Table A.1, reserves recoverable at costs ≤ $40/kgU amount to about 2 million metric tons of uranium (MTU), enough for about 30 years at the current consumption rate.[1] However, reserves are only a small fraction of the total uranium resource base, which also includes known deposits that are not economic to recover at present prices or are surmised to exist with varying degrees of uncertainty in the vicinity of well-mapped deposits or by similarity of one unexplored geologic structure to other mapped and productive ones. When uranium prices rise, presently uneconomic resources will become economic to recover and mining companies will also have an incentive to delineate presently unmapped resources. As a result, new reserves will be created that can be used to fuel a growing installed nuclear capacity.

A quantitative example of the increased reserves that would be created as a result of higher prices has been given by the Uranium Information Centre in Australia: a doubling of the uranium price – which has been declining steadily since the late 1970s; see Figure A.1 [2] – from present contract levels could be expected to create about a tenfold increase in measured resources [3]. The term "measured resources" in this context refers to reserves extractable at costs ≤ $80/kgU, which from Table A.1 amount to about 3 million MTU. Thus, a doubling of uranium prices from about $30/kgU to $60/kgU could be expected to increase these reserves to approximately 30 million MTU. This can be compared with the requirements of the following 1500 GWe mid century scenario: installed nuclear capacity grows linearly from the current 350 GWe to 1500 GWe over 50 years and, after this growth period, no new plants are built and existing ones are operated

[1] Current light water reactors consume approximately 226.5 MTU/GWe·y of electricity generated, hence the demand for today's fleet of 350 GWe is approximately 70,000 MTU per year, assuming a capacity factor of 90%.

for the rest of their lifetimes. The total production over the growth period is 41,625 GWe·y (assuming a capacity factor of 0.9), requiring 9.5 million MTU (assuming a uranium consumption of 226.5 MTU/GWe·y). Nuclear capacity then begins to decline: the newest plants still have 50 years of production ahead of them, but the units built at the beginning of the growth period must be decommissioned. Assuming an average remaining life of 25 years for the fleet, total electricity production over the decline period is 33,750 GWe·y, requiring 7.5 million MTU. The total uranium consumption for this scenario is therefore 17 million MTU. The 30 million MTU of reserves available if the uranium price doubled are more than sufficient to support this scenario.

The 2001 Redbook provides an analysis of uranium supplies to 2050 in which it considers various demand scenarios. The high demand scenario corresponds to an increase in uranium consumption by a factor of 4 by the year 2050 (similar to the scenario considered above). The analysis finds that, excluding speculative resources, conventional resources are adequate through 2029. The OECD does not conclude that such a scenario is unsustainable, but states that "to ensure a stable supply of relatively low-cost uranium needed to ensure the future of nuclear power, major exploration expenditures will have to be made within the next 5 to 10 years, which will only happen if near-term demand and market prices support such expenditures. To have the greatest impact on reducing the projected deficits, discoveries need to be made early enough that they can accommodate long environmental review and development lead times, and still contribute to fulfilling production requirements in a timely manner." Even under a high demand scenario, the OECD does not predict that uranium will run out before 2050, but states that exploration efforts will be required early on in order to avoid price shocks.

Increased reserves from high and low grade ores

The increase of reserves as a result of higher uranium prices could come from both high and low grade ores. The former are the "unconformity-related" deposits discovered starting in the late 1960s in Australia and Canada where typical ore concentrations exceed 10%. The world's largest, highest grade uranium mine at McArthur River in Saskatchewan, Canada is of this type. Estimates of reserves at McArthur River increased by more than 50% in 2001 [4], and further increases in reserves can be expected as a result of further exploration at this mine and other unconformity-related deposits. But such exploration followed by increased production is unlikely at today's uranium prices. Indeed, according to Bernard Michel, the former CEO of Cameco Corp., the McArthur River mine operator, uranium's current low price is "unsustainable"[5].

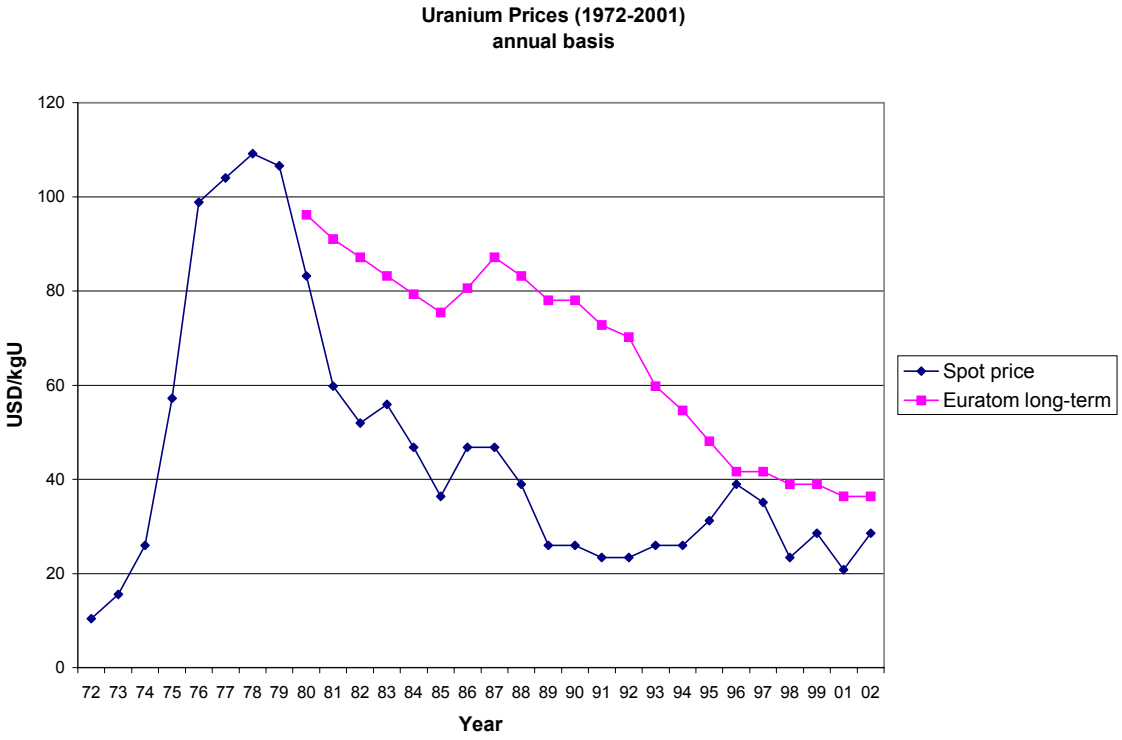

Figure A.1. Uranium prices, 1972-2001

Most of the terrestrial uranium resource consists of large quantities of low grade ore. For example, phosphate deposits, which typically carry 10 to 300 parts per million of uranium, are believed to hold 22 million tons of uranium. A 1980 Scientific American article [6] suggests that the distribution of uranium resources as a function of ore grade is such that, in the region of current commercial interest, a reduction in ore grade by a factor of 10 increases the amount of available uranium by a factor of 300. Equivalently, for a decrease in ore grade by a factor of 2, uranium resources expand by a factor of 5 (see Figure A.2).

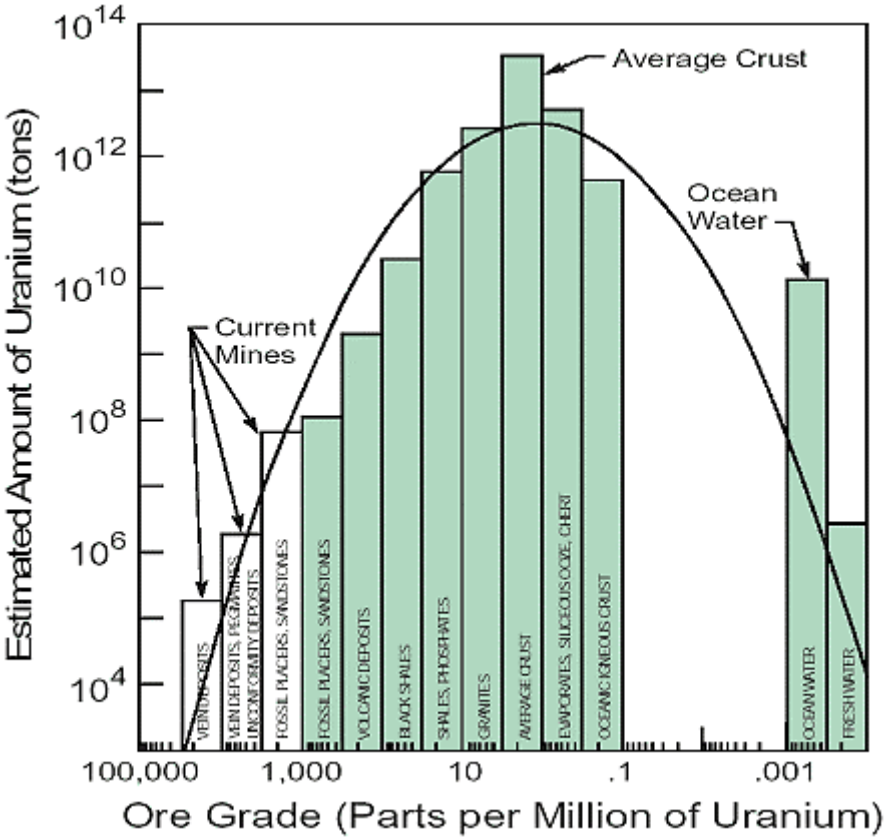

Figure A.2. Uranium Distribution as a Function of Ore Grade

Increased uranium prices and the competitiveness of nuclear electricity

Table A.2 shows that an increase in the price of uranium ore from 30$/kg to 60$/kg corresponds to an increase in ore price of about 1.10 mills/kWh. This corresponds to a modest increase of 2.2% in the cost of nuclear electricity.

Table A.2. Cost of Uranium Ore as a Fraction of Cost of Electricity

Ore price ($/kg)	Ore price (mills/kWh)			% busbar cost[c]
	Direct cost[a]	Carrying charge[b]	Total	
30	0.78	0.33	1.11	2.2%
50	1.29	0.55	1.84	3.7%
60	1.55	0.66	2.21	4.4%
100	2.59	1.10	3.68	7.4%
130	3.36	1.43	4.79	9.6%
200	5.17	2.20	7.37	14.7%

a. Assuming uranium consumption of 226.5 MTU/GWe·y for LWRs.
b. Assuming a lead time of 4.25 years and a carrying charge factor of 0.1
c. Assuming busbar cost of 50 mills/kWh, or 5 ¢/kWh.

Furthermore, even if uranium prices increase as the most attractive deposits are depleted, there is good reason to expect that prices will not soar to prohibitively high levels. Historical data shows that, over the past century, advances in exploration and extraction technologies have made it possible to recover lower grades and other less attractive resources at constant or even decreasing costs in constant dollars.

Economist Julian Simon notes that the historical evidence indicates a downward trend in the costs of natural resource, as measured by all reasonable concepts of costs [7]. For instance, Simon notes that the extractive share of U.S. GNP (the gross value of all extractive output – including minerals, coal, oil, and agriculture – divided by the gross national product) was 50% in 1890, but had fallen all the way to 3.7% by 1988. Hence, although the physical quantities of natural resources extracted have been rising, our expenditure on them as a fraction of total expenditure has been falling.

The U.S. Geological Survey [8] provides data showing that the U.S. mine production composite price index has decreased throughout the 20th century, even as consumption of minerals increased significantly (see Figure A.3 and Figure A.4). The USGS observes that advances in technology have been more than sufficient to overcome obstacles to supply. The USGS also provides striking data on the price and production levels of 4 selected commodities over the 20th century (see Table A.3).

Table A.3. 20th century world production and price for 4 selected commodities

Commodity	Period	Increase in production (percent)	Decrease in constant dollar price (percent)
Aluminum	1900-1998	3,250	89.7
Copper	1900-1998	2,465	75.0
Potash	1919-1998	3,770	93.9
Sulfur	1907-1998	6,000	89.4

Although uranium is different from other extractive resources because of its national security implications, this fact does not change the fundamental process by which higher prices not only lead to exploration efforts but also create an incentive to innovate, which leads to technological progress and tends to hold prices down.

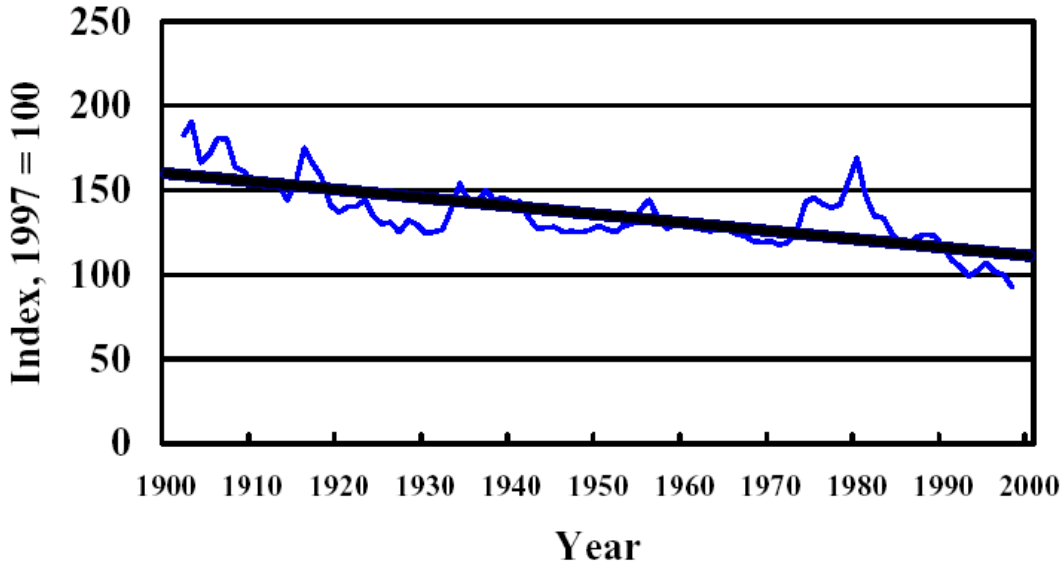

Figure A.3. Composite mineral price index for 12 selected minerals, 1900 to 1998, in constant 1997 dollars. Selected mineral commodities include 5 metals (copper, gold, iron ore, lead, and zinc) and seven industrial mineral commodities (cement, clay, crushed stone, lime, phosphate rock, salt, and sand and gravel).

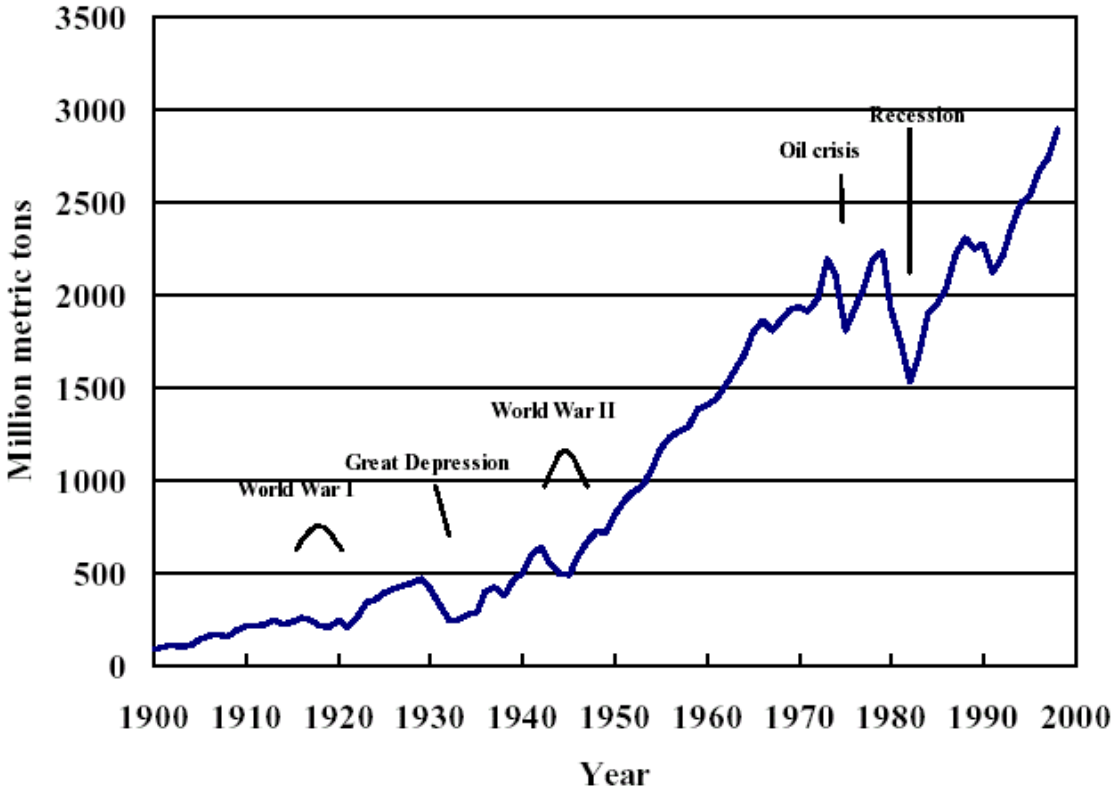

Figure A.4. U.S. apparent consumption of minerals, 1900 to 1998

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B Physics of Transmutation

The spent fuel discharged from nuclear reactors contains a host of elements which remain radiotoxic for extended periods of time. These elements fall into two major categories: actinides (i.e. uranium, plutonium, neptunium, americium, curium, and trace amounts of heavier elements) and fission products. Generally, actinides retain their harmful character for longer than fission products, but the latter have a higher mobility in repository environments. Depending on the specific characteristics of the repository, fission products or actinides can dominate the long-term risk to the environment. The problems associated with disposal of this material have prompted proposals involving transmutation by irradiation in a neutron flux. Several technical options have been proposed, but none has yet emerged as a clear winner. In particular, the debate on whether a fast or thermal spectrum should be used is still open. This appendix examines the basic physics of transmutation and elucidates the key differences between thermal and fast spectra.

B.1 Actinide Transmutation

Actinide transmutation refers to fission events whereby a heavy actinide nucleus is transmuted into a pair of fission product nuclei. We do not consider capture events, even though in some cases an actinide nucleus may be transmuted into another species with lower radiotoxicity or shorter decay time. The fission of one nucleus generates approximately 200 MeV of energy, which corresponds to roughly 1 GW_{th}·d per kg of actinides fissioned. Hence the transmutation rate of actinides depends only on the thermal power produced in reactors, independent of neutron spectrum. A simple expression for the actinide transmutation rate per unit of electricity produced is as follows:

$$RT = \frac{365}{\eta_{th}} \quad \left[\frac{\text{kg}}{\text{GWe} \cdot \text{y}} \right]$$

where RT is the transmutation rate, η_{th} is the thermal efficiency, and the energy release from fission is taken as 1 GW_{th}·d / $kg_{fissioned}$. For systems with a high thermal efficiency (40%), the maximum actinide transmutation rate is ~900 kg/GWe·y.

Neutron Economy

Although it has no bearing on the actinide transmutation rate, neutron spectrum determines the neutron economy of a system. Table B.1 shows the average capture and fission cross sections of the different actinide isotopes, as well as the ratio $\alpha = \sigma_c/\sigma_f$. This parameter is an indicator of neutron economy because capture is a neutron consuming process, whereas fission produces neutrons; hence, high α isotopes are detrimental to a system's neutron economy, whereas low α isotopes tend to create a neutron surplus. In this regard, Table B.1 shows a very significant advantage for the fast spectrum.

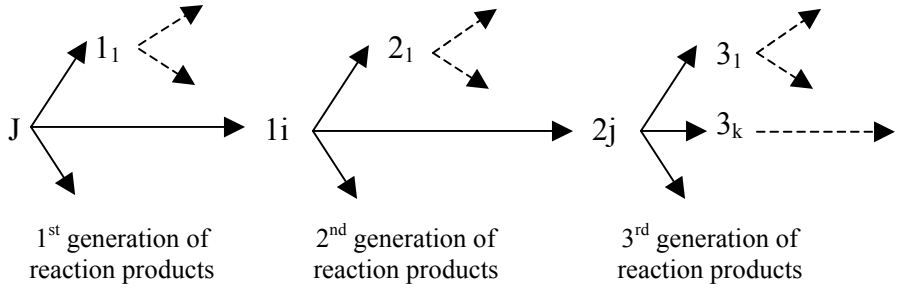
Figure B.1 and Figure B.2 show the fission cross sections of Am and Cm isotopes as a function of neutron energy. It can be seen that the even isotopes of Cm and the odd isotopes of Am have a threshold in the vicinity of 0.5 MeV. In addition, Table B.1 shows high α values for these isotopes. As a result, these isotopes can be expected to have a relatively low fission rate in a thermal spectrum.

Table B.1. Average fission and capture cross sections (barn) for various actinides [1]

Isotope	PWR spectrum			Fast neutron spectrum[(a)]		
	σ_f (b)	σ_c (b)	$\alpha = \sigma_c/\sigma_f$	σ_f (b)	σ_c (b)	$A = \sigma_c/\sigma_f$
Np-237	0.52	33	63	0.32	1.7	5.3
Np-238	134	13.6	0.1	3.6	0.2	0.05
Pu-238	2.4	27.7	12	1.1	0.58	0.53
Pu-239	102	58.7	0.58	1.86	0.56	0.3
Pu-240	0.53	210.2	396.6	0.36	0.57	1.6
Pu-241	102.2	40.9	0.40	2.49	0.47	0.19
Pu-242	0.44	28.8	65.5	0.24	0.44	1.8
Am-241	1.1	110	100	0.27	2.0	7.4
Am-242	159	301	1.9	3.2	0.6	0.19
Am-242m	595	137	0.23	3.3	0.6	0.18
Am-243	0.44	49	111	0.21	1.8	8.6
Cm-242	1.14	4.5	3.9	0.58	1.0	1.7
Cm-243	88	14	0.16	7.2	1.0	0.14
Cm-244	1.0	16	16	0.42	0.6	1.4
Cm-245	116	17	0.15	5.1	0.9	0.18
U-235	38.8	8.7	0.22	1.98	0.57	0.29
U-238	0.103	0.86	8.3	0.04	0.30	7.5

a. Sodium cooled, oxide fuel

For a more precise understanding of the transmutation potential of different neutron fields, the neutron consumption/fission, D, can be used. For a given isotope J, the neutron consumption/fission, D_J, is defined as "the number of neutrons needed to transform the nucleus and its reaction products into fission products". D_J must be evaluated numerically using the following algorithm:



$$D_J = \sum_i P_{J\to 1i} \times \left\{ R_{J\to 1i} + \sum_j P_{1i\to 2j} \times \left[R_{1i\to 2j} + \sum_k P_{2j\to 3k} \times \ldots \right] \right\}$$

where $P_{ni\to mj}$ is the probability of transmuting nuclide *i* into nuclide *j* in going from the n^{th} generation of reaction products to the m^{th} generation and $R_{ni\to mj}$ is the neutron loss in this process:

$$R = \begin{cases} 1 & \text{for neutron capture} \\ 0 & \text{for radioactive decay} \\ 1-\nu & \text{for fission} \\ -1 & \text{for (n,2n) reaction} \\ etc. & \end{cases}$$

Note that this algorithm does not consider the effect of the accumulation of fission products (fission products capture neutrons and therefore increase the neutron requirements for transmutation).

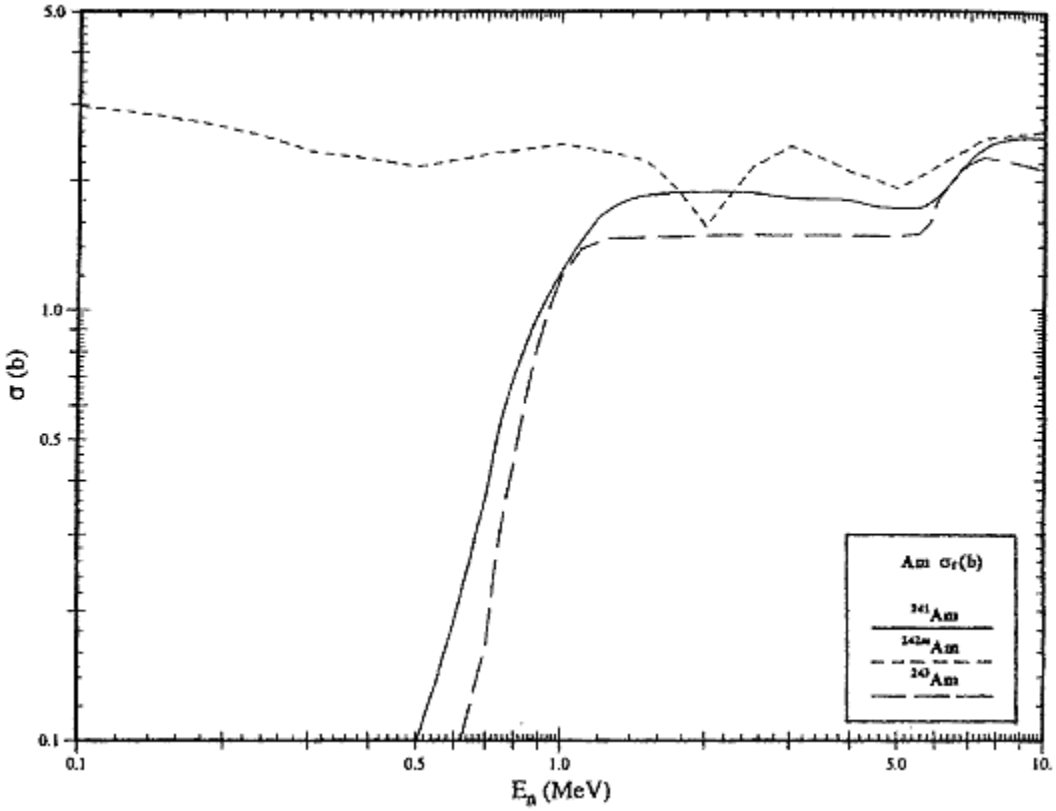


Figure B.1. Fission cross section of americium isotopes [Ref. 1]

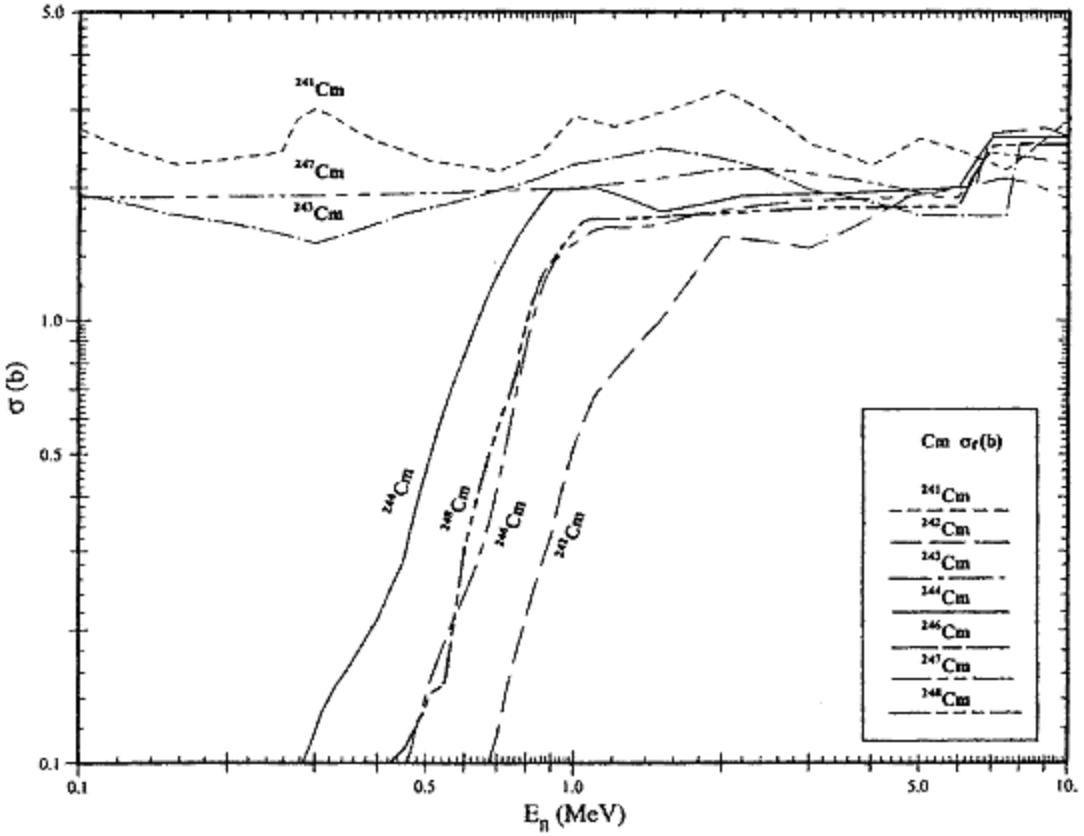


Figure B.2. Fission cross section of curium isotopes [Ref. 1]

Values of D for various isotopes are given in Table B.2 [2]. Note that negative values correspond to a neutron surplus, i.e. transmutation of the given isotope produces more neutrons that it consumes.

Table B.2 once again shows a clear advantage for fast spectrum reactors: for every single actinide isotope, transmutation produces more neutrons than it consumes; on the other hand, only a few of isotopes possess this desirable characteristic in a thermal spectrum. The table also shows that the neutron consumption/fission is lower for higher neutron fluxes. The neutron flux does not influence the reaction cross sections, but a higher flux increases the probability that a nucleus will interact with a neutron before it can undergo radioactive decay. In general, the α value of nuclei is lower before radioactive decay than after; therefore, the overall effect of a higher neutron flux is to reduce neutron consumption/fission. This effect once again favors fast reactors, as the neutron flux in these systems is about an order of magnitude higher than in LWRs.

Table B.2. Neutron consumption per fission (D) of various actinides [2]

Isotope	Fast reactor spectrum[(a)]		Standard LWR thermal spectrum		Superthermal spectrum[(b)]
	Flux, φ (n/cm^2·s)				
	10^{15}	10^{17}	10^{14}	10^{16}	10^{16}
U-235	-0.88	-0.90	-0.62	-0.71	-0.85
U-238	-0.62	-0.64	0.07	0.05	0.10
Pu-238	-1.36	-1.49	0.17	0.042	-0.13
Pu-239	-1.46	-1.51	-0.67	-0.79	-1.07
Pu-240	-0.96	-1.18	0.44	0.085	0.14
Pu-241	-1.24	-1.60	-0.56	-0.91	-0.86
Pu-242	-0.44	-0.75	1.76	1.10	1.12
Np-237	-0.59	-0.72	1.12	0.53	-0.46
Am-241	-0.62	-0.78	1.12	0.076	-0.54
Am-242m	-1.36	-1.54	0.15	-0.88	-1.53
Am-243	-0.60	-1.07	0.82	0.16	0.21
Cm-243	-2.13	-2.26	-1.9	-2.04	-1.63
Cm-244	-1.39	-1.92	-0.15	-0.53	-0.48
Cm-245	-2.51	-2.50	-1.48	-1.46	-1.37

Note: The standard flux level for fast reactors and LWRs is 10^{15} and 10^{14}, respectively.
a. Standard sodium cooled fast reactor with oxide fuel
b. Mean neutron energy of 0.025 eV as a theoretical spectrum for a performance estimation of reactor concepts based on the use of extremely good neutron moderators.

The practical implications of a superior neutron economy will now be outlined. In thermal reactors, only a few actinide isotopes are neutron producers. This can be seen in Table B.1, where only the odd isotopes of plutonium and the curium isotopes have a negative neutron consumption/fission; therefore, in order to have enough neutrons for transmutation, these nuclides must be present in a concentration that is sufficient to balance the neutron consumption of all the other actinides. Table B.3 shows the neutron consumption/fission for the mix of TRU discharged from a typical LWR with a burnup of 55 GWd_{th}/MTHM. When losses due to leakage and capture in non-fuel material are taken into account, the LWR spectrum gives a positive neutron consumption/fission; for that reason, the complete transmutation of TRU is impossible without an additional source of neutrons. In practice, this means that the fuel for these reactors must contain enriched uranium or plutonium (from a stockpile) to drive the nuclear reaction and provide neutrons for transmutation. As a result, fissile plutonium and uranium nuclei will account for a substantial fraction of the transmutations, crowding out non-fissile plutonium isotopes as well as the minor actinides neptunium, americium, and curium, and limiting the transmutation rate of these isotopes. Because of this lower transmutation rate, a large fraction of the nuclear park would have to commit to transmutation in order to achieve a balance between production and transmutation of transuranics in reactors. On the other hand, if a fast spectrum is used (or even accelerator driven thermal spectrum systems), transmutation can be done in a limited number of dedicated systems.

Table B.3. Neutron Consumption of TRU discharged from LWR[(a)] [4]

Isotope	Fraction	D		
		LWR	Fast	Superthermal
Np-237	0.0539	1.12	-0.59	-0.46
Pu-238	0.0364	0.17	-1.36	-0.13
Pu-239	0.451	-0.67	-1.46	-1.07
Pu-240	0.206	0.44	-0.96	0.14
Pu-241	0.121	-0.56	-1.24	-0.86
Pu-242	0.0813	1.76	-0.44	1.12
Am-241	0.0242	1.12	-0.62	-0.54
Am-242m	0.000088	0.15	-1.36	-1.53
Am-243	0.0179	0.82	-0.60	0.21
Cm-243	0.00011	-1.90	-2.13	-1.63
Cm-244	0.00765	-0.15	-1.39	-0.48
Cm-245	0.000638	-1.48	-2.51	-1.37
D_{TRU}[(b)]		-0.03	-1.16	-0.51
Less leakage and capture in non-fuel material (0.3n/fission)		0.27	-0.86	-0.21

a. Burnup is 55 $GWd_{th}/MTHM$

b. $D_{TRU} = \sum_J f_J \cdot D_J$, where f_J is the fraction of isotope J (we assume complete transmutation, i.e. all nuclei undergo fission)

Safety

Another important aspect of transmutation is the effect of a high minor actinide fuel loading on core safety. Introducing minor actinides in the fuel has an adverse effect on the effective delayed neutron fraction because the minor actinides generally have a lower β compared to the uranium isotopes (see

Table B.4). In addition, the displacement of U-238 in favor of minor actinides leads to a harder neutron spectrum, which in turn degrades Doppler coefficient, boron effectiveness, and coolant void reactivity effects [1]. Safety concerns arising from the previous considerations have sparked interest in the use of subcritical cores, which, it is believed, could accommodate a high actinide loading while maintaining acceptable safety characteristics. For critical reactors, however, minor actinide loading must be restrained. Here again, the fast spectrum has a clear advantage: in general, a prudent limit on the fraction of minor actinides in the fuel is approximately 5% for fast reactors, as compared to 1-2% in LWRs [1]. Consequently, fast reactors can be expected to achieve higher minor actinide transmutation rates.

Table B.4. Delayed neutron fraction for selected nuclei [1]

Isotope	β
U-235	0.0065
U-238	0.0172
Np-237	0.00388
Pu-238	0.00137
Pu-239	0.00214
Pu-240	0.00304
Pu-241	0.00535
Pu-242	0.00664
Am-241	0.00127
Am-243	0.00233
Cm-242	0.000377

Higher Actinide Buildup

Another disadvantage of using a thermal spectrum for actinide transmutation is that higher actinides occur in significant amounts at equilibrium. More specifically, Cm-244, Bk-249, and Cf-252, all prolific neutron emitters via spontaneous fission, reach concentrations high enough to make various fuel cycle operations, such as reprocessing and fuel fabrication, extremely difficult. The accumulation of these isotopes is due to the very high capture-to-fission ratio (α) of their predecessors. For example, Am-243 has σ_c = 49 b and α = 111. Upon neutron capture by this isotope, Am-244 is formed, which rapidly beta-decays ($T_{1/2}$ = 26 minutes) into Cm-244.

There are, however, various ways of dealing with this problem. For example, molten salt reactors[1] use liquid fuel and on-line reprocessing, which effectively obviates the need for expensive shielding measures in fuel cycle operations. Even for thermal reactors that use solid fuel, extended storage of irradiated fuel could reduce the neutron source to acceptable levels. Indeed, the problematic isotopes have relatively short half-lives (e.g. 18 years for Cm-244), so a few decades of storage may be sufficient to achieve this goal. Strategies for transmutation of waste in LWRs rely on such schemes to facilitate fuel cycle operations.

[1] Molten salt reactors can use thermal or fast spectrums. Most designs proposed today for actinide transmutation rely on a fast spectrum.

B.2 Fission Product Transmutation

Fission products are transmuted by neutron capture. The reaction rate for this process is given by the product of the capture cross section, $\sigma_{n,\gamma}$ and the neutron flux, φ. We define the transmutation half-life, T_{transm}, as the time required to transmute half of an initial amount of a given fission product subjected to a given flux:

$$T_{transm} = \frac{\ln 2}{\sigma_{n,\gamma} \cdot \phi \cdot 3.16 \cdot 10^{7}} \quad [\text{yr}] \qquad [4]$$

To determine whether a given fission product is significantly transmutable, we compare its decay half-life, $T_{1/2}$, with its transmutation half life, T_{transm}. A fission product is deemed transmutable if its transmutation half-life is much shorter than its decay half-life. This comparison is done in Table B.5, where the most problematic fission products are considered as potential candidates for transmutation. Table B.5 reveals that Sr-90 and Cs-137, the two major sources of decay heat in spent fuel, are not transmutable because their decay half-lives are much shorter than their transmutation half-lives. Sm-151 has $T_{transm}<T_{1/2}$, but $T_{1/2}$ is reasonably short. For these 3 isotopes, storage is the more sensible alternative. For Se-79, Nb-94, and Sn-126, $T_{transm}<<T_{1/2}$, but T_{transm} is excessively long. The irradiation time required is likely to make transmutation of these isotopes an unattractive option. Finally, Tc-99, I-129, Zr-93, Pd-107, and Cs-135 all have $T_{transm}<<T_{1/2}$ and reasonable transmutation half-lives. They are therefore considered transmutable on the basis of this preliminary evaluation. A very important conclusion to be drawn from Table 5 is that Sr-90 and Cs-137 are not transmutable; therefore, the thermal load from spent fuel cannot be reduced significantly by transmutation. Other problems with fission product transmutation are outlined later in this section.

Note that T_{transm} should not be used to evaluate the transmutation rate because it does not take into account the fission products that are created by the fissions that provide the neutrons for transmutation. For example, if 1,000 kg of Tc-99 is placed in a thermal flux, 13.6 kg will be transmuted after one year[1]. But this has to be balanced against the Tc-99 produced by fission in the reactor providing the neutron flux. For example, a 1000 MW_{th} reactor produces roughly 8 kg of Tc-99 over this period[2], so the net Tc-99 transmutation rate of this particular system is 5.6 kg/yr.

[1] Using $\sigma_{n,\gamma} = 4.3 \cdot 10^{-24}$ cm^2 and $\varphi = 10^{14}$ n/cm^{2}s, we find $\sigma_{n,\gamma} \cdot \varphi \cdot 3.16 \cdot 10^7 = 0.01359$ yr^{-1} , where $3.16 \cdot 10^7$ is the number of seconds in one year.

[2] Assuming an average mass of 0.240 kg/mol for the fissioning actinides, a fission energy yield of 1000 MWd/kgHM, and a fission yield of 5.5% for Tc-99 (see Table B.7).

Table B.5. Feasibility of fission product transmutation [4]

Isotope	$\sigma_{n,\gamma}$ (b)		$T_{1/2}$ (yr)	T_{transm} (yr)[a]		
	Fast spectrum[b]	Thermal spectrum[c]		Fast spectrum[b]	Thermal spectrum[c]	
Se-79	0.03	0.1	$6.5 \cdot 10^4$	$7.3 \cdot 10^2$	$2.2 \cdot 10^3$	Questionable
Sr-90	0.01	0.14	29	$2.2 \cdot 10^3$	$1.6 \cdot 10^3$	Not transmutable
Zr-93	0.03	0.28	$1.5 \cdot 10^6$	730	790	Transmutable
Nb-94	0.04	2.2	$2.0 \cdot 10^4$	$5.5 \cdot 10^2$	$1 \cdot 10^2$	Questionable
Tc-99	0.2	4.3	$2.1 \cdot 10^5$	110	51	Transmutable
Pd-107	0.5	0.3	$6.5 \cdot 10^6$	44	730	Transmutable
Sn-126	0.005	0.05	$1 \cdot 10^5$	$4.4 \cdot 10^3$	$4.4 \cdot 10^3$	Questionable
I-129	0.14	4.3	$1.6 \cdot 10^7$	160	51	Transmutable
Cs-135	0.07	1.3	$2.3 \cdot 10^6$	310	170	Transmutable
Cs-137	0.01	0.02	30	$2.2 \cdot 10^3$	$1.1 \cdot 10^4$	Not transmutable
Sm-151	0.7	700	89	31	0.3	Not transmutable

a. Standard flux levels are assumed: 10^{14} n/cm^{2}s for the thermal spectrum, and 10^{15} n/cm^{2}s for the fast spectrum.
b. Neutron energy 0.2 MeV, JEF-2.2
c. Neutron energy 1 eV, JEF-2.2

Table B.5 shows that T_{transm} for the fast and thermal spectra are comparable. In fact, fast reactors have the advantage of a higher neutron flux and thermal reactors are advantaged by higher cross-sections. These effects more or less balance each other. However, it would be possible to take advantage of both effects simultaneously by placing targets containing long-lived fission products (LLFP) in a moderated blanket region at the periphery of a fast reactor core. In this manner, the fission products to be transmuted would be subjected to a high flux of thermal neutrons. Table B.6 shows the transmutation half-lives that could be obtained using such a configuration.

Table B.6. LLFP transmutation in fast reactor moderated blanket

Isotope	T_{transm} (yr)
Se-79	220
Zr-93	79
Nb-94	10
Tc-99	5.1
Pd-107	73
Sn-126	440
I-129	5.1
Cs-135	17

The number of neutrons required for transmutation of a LLFP, D*, is shown in Table B.7 for each of the candidate isotopes. Note that D* is dependent on the interval between reprocessing steps, when all fission products except the transmutable isotopes (Zr, Tc, Pd, I, Cs) are assumed to be removed. The longer the reprocessing interval, the higher is D* because the reaction products of the transmutable isotopes keep capturing neutrons as long as they remain exposed to the flux. However, the dependence of D* on the reprocessing interval is weak because cross-sections are small and reaction rates are slow (for instance, increasing the reprocessing interval from 3 to 30 years increases D* for Tc-99 from 1.008 to 1.1 neutron/transmutation [4]). Table B.7 also shows the neutron consumption/fission, D, given by the product of D* and the yield per fission, Y. D is in fact the number of neutrons required per fission to keep LLFP production and transmutation in balance.

Table B.7. Neutron consumption for LLFP transmutation[(a)] [4]

Isotope	D* n/transmutation	Y[(b)] yeld/fission	D[(c)] n/fission
Zr-93	2.01	0.05	0.1005
Tc-99	1.01	0.055	0.05555
Pd-107	2.04	0.015	0.0306
I-129	1.008	0.009	0.009072
Cs-135	1.002	0.017	0.017034
Sn-126	2	0.0012	0.0024
Nb-94	0.985	6.30E-07	6.206E-07
Se-79	2	0.0004	0.0008
		Total:	0.22
all Zr	2.03	0.26	0.5278
All Tc	1.01	0.055	0.05555
All Pd	3.22	0.095	0.3059
All I	1.01	0.011	0.01111
All Cs	0.585	0.13	0.07605
		Total:	0.98

a. Time interval between reprocessing steps is 3 years
b. After 5 years of cooling
c. $D = Y \cdot D^*$

Table B.7 shows that transmutation of the 8 transmutable isotopes together requires about 0.22 neutron/fission. This assumes, however, that the LLFPs are subjected to isotope separation. This may become a practically feasible alternative with significant breakthroughs in isotopic separation, but otherwise it seems more realistic to consider neutron requirements when all isotopes of an element are transmuted together. In this case, the requirement is 0.98 neutron/fission, an excessively large figure. Setting aside Pd and Zr, which together require 0.83 neutron/fission, we find that all isotopes of Tc, I, and Cs could be transmuted with 0.15 neutron/fission. Note that Cs-137 would need to be handled as part of the Cs isotopes and that the presence of this high-heat isotope would

undoubtedly complicate target fabrication. If Cs is not transmuted, the requirement for transmutation of Tc and I alone is 0.07 neutron/fission.

The transmutation of fission products requires surplus neutrons and is therefore clearly facilitated by the use of fast reactors. While it may be possible to provide 0.15 neutron/fission (required for transmutation of Tc, I, and Cs) in an LWR with extra enrichment, a fast reactor can provide ~1 neutron/fission, thus promising much higher transmutation rates than LWRs. This means that LLFP transmutation could be done in a limited number of reactors: fast spectrum transmuters accounting for roughly 15% of installed nuclear capacity could eliminate the LLFPs from the entire nuclear park.

Thus, for fission product transmutation, fast spectrum systems are once again the preferred option due to their superior neutron economy. However, these considerations do not weigh too heavily against thermal spectrum options because fission product transmutation in any case does not seem very promising: first, Sr-90 and Cs-137 are not transmutable, meaning that there can be very little impact on decay heat management for geological storage. Second, if isotopic separation of LLFPs is ruled out, even a preliminary evaluation indicates that only 3 isotopes (Tc-99, I-129, Cs-137) are potential candidates for transmutation. Furthermore, the development of I-129 irradiation targets appears very problematic (but would not be needed if transmutation was done in a molten salt) and the transmutation of Cs-135 presents some serious practical difficulties because it must be transmuted along with Cs-137 [1]. Hence, Tc-99 may be the only good candidate for transmutation, which considerably limits the potential impact of LLFP transmutation and casts doubt on whether it is at all a worthwhile proposition. And even for Tc-99, one practical problem remains. The long transmutation times, e.g. 51 years for Tc-99 in a thermal spectrum (see Table B.5), mean that extensive repetitive recycle of LLFP targets will be required.

Conclusion

Basic physics considerations clearly favor fast spectrum options for radioactive waste transmutation; indeed, fast spectrum accelerator driven systems are currently receiving the most attention as potential transmuters due to their unsurpassed neutron economy and potentially superior safety characteristics. Nevertheless, actinide transmutation in a thermal spectrum is feasible. In particular, it is feasible in LWRs, which today represent the most well established nuclear reactor technology. In light of the significant difficulties associated with bringing new reactor designs to commercial deployment, this alone makes an extremely cogent argument in favor of LWR transmutation of waste. In the end, expert judgment will be needed to weigh the unmatched technological maturity of LWRs against the promise of better performance from systems based on a fast spectrum.

References:

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C Fuel Cycle Cost: Once-through vs. Pu Recycle

Spent UOX fuel typically contains a little over 1% Pu. Through reprocessing (PUREX process), it is possible to recover this plutonium and use it to make MOX fuel for use in LWRs. However, because of the high costs of reprocessing and of MOX fuel fabrication, the cost of repository disposal must be very high in order for the MOX option to become economically competitive with the once-through UOX cycle.

Fuel Cycle Cost Model

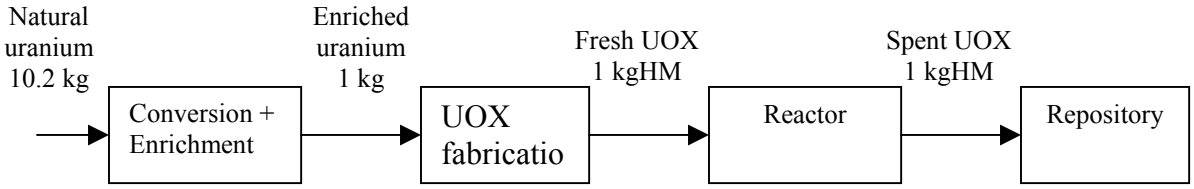
A simple expression for the fuel cycle cost is as follows:

$$FCC = \sum_i M_i \cdot C_i + \sum_i M_i \cdot C_i \cdot \phi \cdot \Delta T_i \quad [\$]$$

where: FCC = Fuel Cycle Cost [$]
M_i = mass processed at stage i [kg or kg SWU]
C_i = unit cost at stage i [$/kg or $/kg SWU]
ϕ= carrying charge factor (yr^{-1})
ΔT_i = delay between the investment for stage i and the midpoint of the irradiation of the fuel (years)[1]

UOX cycle

The once-through UOX cycle is represented below (for 1 kgHM of fuel):



Assumptions:

- U235 content of natural U: 0.711%
- Enrichment tails assay: 0.3%
- Fresh fuel enrichment: 4.5%
- Losses are neglected
- Burnup: 50 MWD/kgHM
- Capacity factor: 0.9
- Thermal efficiency: 0.33

[1] Note that ΔT_i can vary depending on the fuel management strategy

The Separative work per unit of enriched product can be obtained as:

$$\frac{\text{kg SWU}}{\text{kg product}} = (2x_p - 1)\cdot \ln\left(\frac{x_p}{1-x_p}\right) + \frac{x_p - x_{nat}}{x_{nat} - x_t}\cdot(2x_t - 1)\cdot \ln\left(\frac{x_t}{1-x_t}\right) - \frac{x_p - x_t}{x_{nat} - x_t}\cdot(2x_{nat} - 1)\cdot \ln\left(\frac{x_{nat}}{1-x_{nat}}\right)$$

where: x_p = product enrichment
x_{nat} = natural enrichment
x_t = tails assay

Using the values presented above for x_p, x_{nat}, and x_t, we get 6.23 kg SWU/kg product.[1] The fuel cycle cost can now be calculated (for 1 kgHM of fresh UOX fuel):

Table C.1. Once-through UOX Fuel Cycle Cost

	M_i	C_i	ΔT_i (yr)	Direct Cost $M_i \cdot C_i$ ($)	Carrying Charge $M_i \cdot C_i \cdot \phi \cdot \Delta T_i$ ($)
Ore purchase	10.2 kg	30 $/kg	4.25	307	130
Conversion	10.2 kg	8 $/kg	4.25	82	35
Enrichment	6.23 kg SWU	100 $/kg SWU	3.25	623	202
Fabrication	1 kgHM	275 $/kgHM	2.75	275	76
Storage and disposal	1 kgHM	400 $/kgHM	-2.25[a]	400	-90
			Total	1686	353
			Grand Total	2040	

a. The cost of waste storage and disposal is assumed to be paid at the end of irradiation, even though the unit cost of $400/kgHM is a proxy for the 1 mill/kWh paid by utilities during irradiation.

The calculations are based on the following assumptions:

- Fuel irradiation time : 4.5 years
- Lead times:
  - 2 years for ore purchase
  - 2 years for conversion
  - 1 year for enrichment
  - 0.5 year for fuel fabrication
- Carrying charge factor: $\phi = 0.1$

The cost is thus $2,040/kgHM. We can obtain the fuel cycle cost in ¢/kWh(e) as follows:

$$\frac{2040\$}{kgHM}\cdot\frac{kgHM}{50MWd}\cdot\frac{1MW}{1000kW}\cdot\frac{1d}{24h}\cdot\frac{1kW}{0.33kW(e)} = 5.15\cdot10^{-3}\,\frac{\$}{kWh(e)}$$

[1] Alternatively, a simple linear relationship can be used to approximate the SWU requirement. For a tails assay of 0.3%, the following holds:

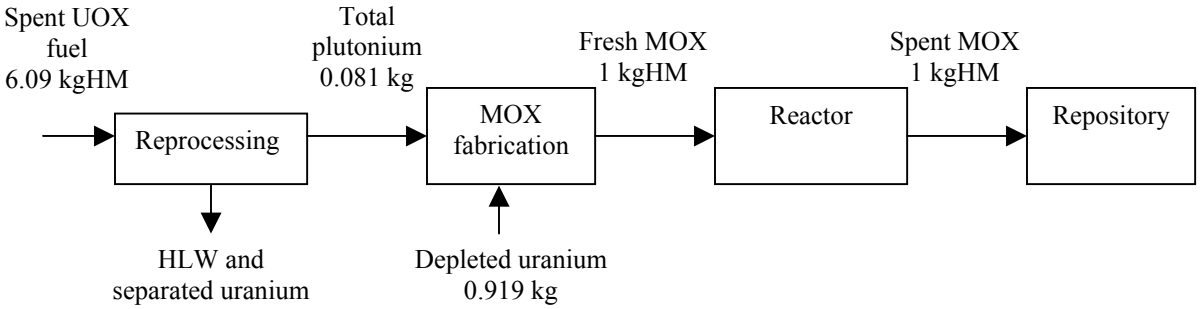
$$\frac{\text{kg SWU}}{\text{kg product}} = 2.07\cdot x_p - 3.23$$

Using the same values as above for x_p, x_{nat}, and x_t, we get 6.09 kg SWU/kg product.

The fuel cycle cost is therefore 0.515 ¢/kWh(e).

MOX cycle

The MOX cycle can be represented as follows (for 1 kgHM of fuel):



Assumptions:

- Pu content of spent UOX 1.33%
- Pu content of fresh MOX: 8.1%
- Losses are neglected
- Burnup: 50 MWD/kgHM
- Capacity factor: 0.9
- Thermal efficiency: 0.33

We now calculate the fuel cycle cost (per kgHM fresh MOX fuel):

Table C.2. Single Recycle MOX Fuel Cycle Cost

	M_i (kgHM)	C_i ($/kgHM)	ΔT_i (yr)	Direct Cost $M_i \cdot C_i$ ($)	Carrying Charge $M_i \cdot C_i \cdot \phi \cdot \Delta T_i$ ($)
Credit for UOX SF	6.09	-400	4.25	-2436	-1035
Reprocessing	6.09	1000	4.25	6090	2588
HLW storage and disposal	6.09	300	3.25	1827	594
MOX Fabrication	1	1500	3.25	1500	488
MOX Storage and disposal	1	400	-2.25	400	-90
			Total	7381	2544
			Grand Total	9926	

Assumptions :

- Fuel irradiation time : 4.5 years
- Lead times:
  - 2 years for acceptance of spent UOX fuel,
  - 2 years for reprocessing,
  - 1 year for storage of HLW from reprocessing;
  - 1 year for MOX fuel fabrication
- The cost of acquiring depleted uranium is neglected
- Both the cost of separated uranium storage and the potential value of separated uranium material are not included in the analysis. Under current conditions, separated uranium is not used for fuel fabrication because using natural uranium is less expensive. Separated uranium is simply stored for possible use in the future. Since cost of storing

separated uranium is very modest due to its low radioactivity, we ignore it in this analysis.

- The cost of HLW storage and disposal is assumed to be 25% lower than the cost of spent fuel storage and disposal. The HLW contains most of the fission products (including Sr-90 and Cs-137) and all the minor actinides present in the processed spent fuel, hence storage and disposal requirements are not expected to be much improved compared to spent fuel. However, because HLW has a lower volume and very small plutonium content, modest savings can be expected.
- The cost of storage and disposal for spent MOX fuel is assumed to be the same as for spent UOX fuel. Indeed, spent MOX is not reprocessed due to the degraded isotopic composition of its plutonium. We therefore consider it to be a liability comparable to spent UOX fuel.
- Carrying charge factor: $\phi = 0.1$

The fuel cycle cost is therefore \$9,926/kgHM, or 2.51 ¢/kWh(e). This is nearly 5 times higher than for the once-through UOX cycle under U.S. conditions.

The incremental MOX fuel cost compared to UOX fuel cost will contribute to an increase in the cost of electricity in proportion to the ratio of MOX to UOX fuel in the entire fleet. Accordingly the incremental electricity cost for the fleet will be:

$$0.515 \text{ cents/kWh} \cdot (6.09/7.09) + 2.51 \text{ cents/kWh} \cdot (1/7.09) = 0.796 \text{ cents/kWh}$$

or a blended increase in the cost of electricity of 0.28 cents/kWh in the MOX/UOX cycle compared to the once-through UOX cycle.

Conditions for competitiveness of the MOX option

It is important to determine under what conditions the MOX fuel cycle becomes cost competitive with the once through UOX cycle. Cost components to consider are: (1) cost of natural uranium, (2) cost of reprocessing, (3) cost of MOX fabrication, and (4) cost of waste storage and disposal. Table C.3 presents the value that would make the fuel cycle cost of both options equal (breakeven value) for each of these four cost parameters.

Table C.3. Breakeven Values

Cost component	Original value	Required value	Required/original
Natural uranium	\$30/kgU	\$570/kgU	19
Reprocessing	\$1000/kgHM	\$90/kgHM	0.09
MOX fabrication	\$1500/kgHM	Impossible	N/A
Waste storage and disposal	\$400/kgHM (SF) \$300/kgHM (HLW)	\$1120/kgHM \$100/kgHM	2.8 0.33

The cost of natural uranium is not likely to reach such high levels in the foreseeable future. The cost of reprocessing will probably never drop down to the required value of \$90/kgHM. As for waste storage and disposal, it is not reasonable to expect that the cost will be 11 times higher for UOX and MOX spent fuel than for HLW from reprocessing;

indeed, although the volume of the HLW is much smaller, it still contains most of the fission products and all the minor actinides from the spent fuel. Therefore, its heat load in the first few hundred years should be comparable to that of spent fuel. It can also be observed from Table C.2 that, even if we assume that HLW storage and disposal can be done at zero cost, the total cost of the MOX option is still $7505/kgHM (obtained by subtracting the cost of HLW disposal, $1827+$594, from the total cost, $9926). This is equivalent to 1.9 ¢/kWh(e), or more than 3 times the cost of the once-through option. It should be noted, however, that the original values selected for the costs of waste storage and disposal are not an absolute reference: important differences exist between countries because this cost depends on how difficult the nuclear waste problem is perceived to be. For some countries, the cost of waste disposal may very well be much higher than the reference values used here.

Finally, we consider the effect of changing our cost assumptions for ore purchase, reprocessing, MOX fabrication, and waste storage and disposal simultaneously. We find that the fuel cycle cost of the two options is equal under the following revised assumptions:

Table C.4. Breakeven Values (components adjusted simultaneously)

Cost Component	Unit	Original Value	Required Value
Ore purchase	$/kg	30	55
Reprocessing	$/kgHM	1000	600
MOX fabrication	$/kgHM	1500	975
Storage and disposal:			
▪ Spent Fuel	$/kgHM	400	600
▪ HLW	$/kgHM	300	100
Fuel cycle cost (both options)		6.5 mills/kWh	

Table C.4 shows that, by revising several cost assumptions in favor of plutonium recycling, we obtain equal fuel cycle costs for both options. Although the required ore purchase price is high and costs for reprocessing, MOX fabrication, and HLW disposal can be characterized as optimistic, they fall within the range of uncertainty defined by other fuel cycle cost studies (see Table C.6).

Comparison with other estimates

There have been a number of studies on the economics of reprocessing with significant differences in assumptions. The most comprehensive study has been carried out by the OECD/NEA [1]. This study thoroughly evaluated the cost of the once-through and plutonium recycling fuel cycles, and concluded that the cost of the once-through option is about 15% lower (based on the assumptions presented in Table C.5). Thus, the findings of the OECD differ significantly from the result presented earlier, where the cost of the once-through option was found to be about 5 times lower.

There are several differences between the methodology used in the OECD study and the simple fuel cycle cost model used in this appendix. The OECD model is more detailed and the methodology for dealing with carrying charges is more involved. In addition, it

sometimes uses different assumptions about the workings of the fuel cycles. For example, a credit is given for the irradiated uranium recovered in reprocessing, implying that it is used for fuel fabrication. In spite of such differences, assumptions regarding unit costs remain the dominant factor influencing fuel cycle cost estimates. The OECD study uses costs that are much more favorable to the reprocessing option. In fact, using the OECD assumptions in our model finds that the fuel cycle cost for plutonium recycling is about 11% higher than for the once-through option. This is shown in Table C.5.

Table C.5. Fuel Cycle Cost Using OECD estimates

Cost component	OECD estimate
Ore Purchase	50 $/kgHM
Conversion	8 $/kgHM
Enrichment	110 $/kg SWU
UOX fabrication	275 $/kgHM
SF storage and disposal	570 $/kgHM
Reprocessing	620 $/kgHM
HLW storage and disposal	60 $/kgHM
MOX fabrication	1100 $/kgHM
Fuel Cycle Cost	
Once-Through:	6.43 mills/kWh
Pu recycle:	7.11 mills/kWh

Table C.5 shows that OECD unit costs for the various back-end operations diverge significantly from the ones that were assumed in Table C.1 and Table C.2. Such differences can be expected, as fuel cycle cost studies generally show very large uncertainties on such estimates. Indeed, very little data on the cost of reprocessing and recycling operations is publicly available, and spent fuel or HLW disposal has not been implemented anywhere in the world, so the costs associated with these operations cannot be determined precisely. Furthermore, estimates are difficult to make for several reasons. First, engineering cost estimates for this type of activity are notoriously uncertain. Second, since fuel cycle facilities are high capital cost plants, the cost of capital assumption is very important[1]. Third, the cost estimates per unit product depend on assumption about both plant productivity and on allocation of fixed construction and development costs to unit output. Finally, the ultimate disposal cost for either spent fuel or HLW is not established. Certainly little confidence can be placed in any estimate on the difference in disposal costs for HLW and spent fuel.

Several other studies provide estimates of the unit costs for various fuel cycle operations. The OECD/NEA provides revised estimates in a recent study on advanced fuel cycles [2]. The Gen-IV Fuel Cycle Crosscut Group offers a range of estimates in its report [3]. Fetter, Bunn, and Holdren have offered an analysis of the economics of reprocessing versus direct disposal of spent nuclear fuel [4]. Finally, the National Research Council's study on Nuclear Waste [5] has an appendix on recycling economics. Table C.6 presents the unit cost assumptions used in these studies.

[1] For example, the National Research Council study estimates the levelized reprocessing cost for a 900 MTHM /year plant varies for different owner operators as follows: government $800/kgHM, utility $1300/kgHM, private venture $2000/kgHM.

Table C.6. Comparison of Cost for Once-through and Recycle Process Steps

Cost Component	Unit	Estimated Cost (lower bound – nominal – upper bound)		
		OECD/NEA (2002)	DOE GEN-IV	Fetter, Bunn, Holdren
Ore Purchase	$/kg	20-30-40	20-30-80	33
Conversion	$/kg	3-5-7	3-5-8	4-6-8
Enrichment	$/kg SWU	50-80-110	50-80-120	50-100-150
UOX fabrication	$/kgHM	200-250-300	200-250-350	150-250-350
SF storage and disposal	$/kgHM	410-530-650	210-410-640	0-150-300 more than HLW
UOX reprocessing	$/kgHM	700-800-900	500-800-1100	500-1000-1600
MOX reprocessing	$/kgHM	700-800-900	500-800-1100	-
HLW storage and disposal	$/kgHM	63-72-81	80-200-310	0-150-300 less than SF
MOX fabrication	$/kgHM	900-1100-1300	600-1100-1750	700-1500-2300

Conclusion

The simple fuel cycle cost model shows that the MOX option is roughly 4 times more expensive than once-through UOX, using estimated costs under U.S. conditions. Thermal recycle can be shown to be competitive with the once-through option only if the price of uranium is high and if optimistic assumptions are made regarding the cost of reprocessing, MOX fabrication, and HLW disposal.

It should be noted that the cost increment associated with reprocessing and thermal recycle is small relative to the total cost of nuclear electricity generation. In addition, the uncertainty in any estimate of fuel cycle costs is extremely large. Therefore, although there seems to be no case for reprocessing based on the analysis presented here, proponents of this option can claim that it cannot be entirely dismissed on purely economic grounds.

References:

1. OECD/NEA "The Economics of the nuclear fuel cycle," 1994.
2. OECD/NEA, "Accelerator-driven Systems and Fast Reactors in Advanced Nuclear Fuel Cycles", 2002
3. DOE, "Generation 4 Roadmap - Report of the Fuel Cycle Crosscut Group", 2001
4. Fetter, Bunn, Holdren, "The Economics of Reprocessing vs. Direct Disposal of Spent Nuclear Fuel", 1999
5. "Nuclear Waste – Technologies for separations and transmutation," Committee on Separation Technology and Transmutation systems, National Research Council, National Academy of Sciences, Appendix J, 1996