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Fuel Cycle Scenario Definition, Evaluation, and Trade-Offs

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ABSTRACT

This report clarifies many technical issues being analyzed by the Advanced Fuel Cycle Initiative (AFCI) program, including Inert Matrix Fuel (IMF) versus Mixed Oxide (MOX) fuel, single-pass versus multi-pass recycling, thermal versus fast reactors, the need to recycle Np-Pu-Am to meet established AFCI objectives, the borderline case of Cm, the potential need for transmutation of technetium and iodine, and the value of separating cesium and strontium. This report represents the first attempt to calculate a full range of metrics, spanning all four AFCI program objectives [DOE2005a, DOE2005, DOE2006] - waste management, proliferation resistance, energy recovery, and systematic management/economics/safety - using a combination of "static" calculations and a system dynamic model, DYMOND.[Moisseytsev2001, Yacout2005a] (In late FY2006, DYMOND is being replaced with the VISION model.) In many cases, we examine the same issue both dynamically and statically to determine the robustness of the observations. All analyses are for the U.S. reactor fleet.

This is a technical report, not intended for a policy-level audience. A wide range of options are studied to provide the technical basis for identifying the most attractive options and potential improvements. No single fixed strategy guarantees optimal performance at all times in all possible futures. Instead, the objective in the next few decades should be to cost-effectively develop the tools to deal with the circumstances at that point in time. Technical maturity and readiness to deploy were outside the scope of this report.

Many dynamic simulations of option deployment are included. There are few "control knobs" for driving or piloting the fuel cycle system into the future, even though it is dark and raining (uncertain) and controls are sluggish with slow time response: what types of reactors are built, what types of fuels are used, and the capacity of separation and fabrication plants. Driving responsibilities are distributed among utilities, government, and regulators, compounding the challenge of making the entire system work and respond to changing circumstances. We identify approaches that would increase our ability to drive the fuel cycle system: (1) have a recycle strategy that could be implemented before the 2030-2050 approximate period when current reactors retire so that replacement reactors fit into the strategy, (2) establish an option such as multi-pass blended-core IMF as a downward plutonium control knob and accumulate waste management benefits early, (3) establish fast reactors with flexible conversion ratio as a future control knob that becomes available as fast reactors are added to the fleet, and (4) expand exploration of blended assemblies/cores and targets, which appear to have advantages and agility.

Results suggest multi-pass full-core MOX appears to be a less effective way than multi-pass blended core IMF to manage the fuel cycle system because it requires higher TRU throughput while more slowly accruing waste management benefits. Single-pass recycle approaches for LWRs do not meet AFCI program objectives and could be considered a "dead end." (We did not study the VHTR.) Fast reactors are effective but a significant number of fast reactors must be deployed before the benefits of that technology can be observed.

EXECUTIVE SUMMARY

This report clarifies many technical issues being analyzed by the Advanced Fuel Cycle Initiative (AFCI) program, including Inert Matrix Fuel (IMF) versus Mixed Oxide (MOX) fuel, single-pass versus multi-pass recycling, thermal versus fast reactors, the need to recycle Np-Pu-Am to meet established AFCI objectives, the borderline case of Cm, the potential need for transmutation of technetium and iodine, and the value of separating cesium and strontium. This report represents the first attempt to calculate a full range of metrics, spanning all four AFCI program objectives [DOE2005a, DOE2005, DOE2006] - waste management, proliferation resistance, energy recovery, and systematic management/economics/safety - using a combination of "static" calculations and a system dynamic model, DYMOND.[Moisseytsev2001, Yacout2005a] (In late FY2006, DYMOND is being replaced with the VISION model.) All analyses are for the U.S. reactor fleet.

The report's analyses were produced by INL, ANL, and SNL personnel under their Simulation, Evaluation, and Trade Study (SETS) work packages during FY2005 with followup work in FY2006.

This is a technical report, not intended for a policy-level audience. A wide range of options are studied to provide the technical basis for identifying the most attractive options and potential improvements. Indeed, we do not believe that any of the specific options presented here are the most optimum or economically feasible. Technical maturity and readiness to deploy were outside the scope of this report.

Options Considered in this Study

The range of fuel cycle options can be divided into two broad categories: (1) throw-away and (2) recycle. The throw-away, or once-through, approach does not meet the AFCI program objectives; indeed it leaves nuclear power vulnerable to nearer-term constraints of geologic repository capacity and longer-term uranium resource limits. Therefore, it is prudent and appropriate to determine the feasibility and attractiveness among recycle options.

The recycle options (strategy 2) can be usefully categorized by the type of reactors that **consume** recycled transuranic material:

- **2a**. Thermal reactors
- **2b**. Both thermal and fast reactors
- **2c**. Fast reactors in consumer (burner) mode
- **2d**. Fast reactors in breeder mode

This report includes several cases within each of these categories, see Table 1. Strategies 1, 2a, 2b, and 2c use thermal reactors; in this study, the light water reactor (LWR) is used as the thermal reactor. In this categorization scheme, the proposed technology for the Global Nuclear Energy Partnership (GNEP) is within strategy 2c; recycle is done in consumer fast reactors (CFR) but not thermal reactors. Strategies 2c and 2d both use fast reactors. Strategy 2c uses fast reactors in consumer or burner mode (conversion ratio less than 1) so that thermal reactors continue to be used as the source of transuranic fuel for the fast reactors. Strategy 2d uses fast reactors in breeder mode (conversion ratio greater than 1) so that thermal reactors are phased out. We do not claim that any of the specific cases is "optimum" within that category.

Table 1. Throw-away versus recycle strategies considered in this study

Strategy	Cases in this study
1. Throw away (once through)	LWR with UOX fuel (at burnups of 33, 51, and 100 MW-day/kg-HM)
2a. Recycle in thermal reactors	LWR with IMF-NpPuAm fuel (blended core, ¾ UOX, ¼ IMF) LWR with MOX-NpPuAm fuel (full core MOX)
2b. Sustainable recycle in thermal & fast reactors	LWR with IMF-NpPu & consumer FR (conversion ratio 0.25) LWR with MOX-NpPu & consumer FR (conversion ratio 0.25)
2c. Sustainable recycle in burner fast reactors	LWR with UOX & consumer FR (conversion ratio 0.25)
2d. Sustainable recycle in breeder fast reactors	Breeder FR

Limitations

There are four major limitations of this study. First, thermal reactors (TR) are always represented by Light Water Reactors (LWR) and both consumer fast reactors (CFR) and breeder fast reactors (BFR) are always represented by Sodium Fast Reactors (SFR). (We are aware many colleagues use the phrase "burner fast reactor;" however, this report uses "consumer" so the acronym (CFR) differs from BFR and minimizes the chance of misinterpretation of "burn" in the chemical reaction sense.) Processing of thermal reactor fuel is performed at centralized plants using UREX+ technology. Processing of fast reactor fuel is performed at power plants using pyrochemical technology. To first order, we do not believe the conclusions in this report would differ substantially for other thermal or fast reactor options based on the AFCI evaluation of Generation IV transmutation impacts.[Taiwo2005] We have not considered ultra-high burnup with the Very High Temperature Reactor (VHTR) concept.

Second, no attempt has been made to include economics *per se*. Instead, economic indicators are used such as separation and fuel fabrication throughputs and the relative amount of fuels that require remote handling (those including americium or curium), glovebox operation (those including plutonium), or current hands-on fabrication (uranium only).

The third assumption is that all options studied are technically feasible and available at the time indicated in various deployment scenarios (typically 2025), which implies the necessary underlying research, development, and demonstration have been completed. Thus, in this report, there is no analysis on the basis of technological maturity or readiness to deploy. Similarly, there is no analysis on the basis of R&D costs.

The fourth is that detailed fuel cycle data are only available for a finite subset of specific recycle cases. Although great care has been taken to assure the fuel cycle performance for each case has been analyzed in a consistent manner, not all promising options have been considered. In future work, the scenario evaluations will be utilized to define additional cases for detailed analyses; and new fuel cycle transmutation data on specific options will be incorporated into the dynamic model, as available.

Major Conclusions

Table 2 summarizes key results in this report. These results address the four AFCI objectives: [DOE2005a, DOE2005b, DOE2006]

1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.

2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for used fuel management.
3. Enhance energy security by extracting energy recoverable from used fuel and depleted uranium, ensuring that uranium resources do not become a limiting factor for nuclear power.
4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.

Table 2. Key Results that Clarify Selection among Recycle Options

Goal	2a. LWR with MOX	2a. LWR with IMF	2b. Consumer FR + IMF in LWR	2c. Consumer FR + UOX in LWR	2d. Breeder FR
Reduce long-term heat (LTH) to increase geologic repository capacity by 10-50x	~2x	~3x	>50x	>50x	>50x
Reduce long-term hypothetical dose (LTD) by 10-50x at peak dose (500,000 yrs)	~2x	~3x	>50x	>50x	>50x
Reduce long-term radiotoxicity (LTR) by 100x at 1,000 yrs so waste is less toxic than uranium ore	~2x	~3x	~100	~100	~100
Ensure recycling is sustainable	Number of recycles is limited by accumulation of higher TRU isotopes		Use of adequate number of FR ensures that recycling can be sustained indefinitely		
Minimize weapon-usable inventory	Intermediate values	Minimum of cases studied	Close to minimum	Intermediate values	Will eventually exceed once-through
Improve use of uranium ore by 50x in the long-term	~1.2x	~1.2x	~1.4x	~1.4x	>100x
Minimize TRU throughput, tonnes/yr per GWe to separation plant	0.94 at 5 cycles	0.34 at 5 cycles	0.85 at equilibrium	0.76 at equilibrium	1.45 at equilibrium
Minimize percent of fleet that must be fast reactors, which are yet unproven in the marketplace	None needed	None needed	19% at equilibrium	27% at equilibrium	100%
Maximize percent of fuel in the system that can be fabricated hands-on (UOX)	80%	75%	71%	73%	None
Minimize percent of fuel requiring remote fabrication	20% = MOX -NpPuAm	2% = Am targets (a)	19% = FR fuel with NpPuAmCm (b)	27% = FR fuel with NpPuAmCm	100% = FR fuel with NpPuAmCm
Minimize number of new fuels/reactor types needed (to minimize R&D risk)	1 / 0	1-2 / 0 "2" if use separate Am targets	2 / 1 Need both IMF and FR fuel	1 / 1	1 / 1

Color code	Pink	Yellow	Green
	Option does not meet goal	Option partially meets goal	Option meets goal

(a) Also, 23% of the fuel is IMF with NpPu, requiring glovebox fabrication
(b) Also, 11% of the fuel is IMF with NpPu, requiring glovebox fabrication.

Objective 1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.

We considered objective 1 using three primary metrics:

- **Long-term heat (LTH)** – the heat generated by waste in the time period from repository ventilation stoppage (minimum 50 yrs) to 1500 years. This has been calibrated versus detailed repository heat response calculations by Wigeland [Wigeland2003, 2004a, 2004b, 2006; Stillman2004c] so that it is a good indicator of heat-limited repository capacity.
- **Long-term dose (LTD)** – the hypothetical peak dose to the maximum exposed individual of the public scaled from 2004-vintage results for Yucca Mountain assuming linearity. In most cases, peak dose would occur at ~500,000 yrs after emplacement. The time that dose peaks can shift to other time periods if TRU are well recovered but U and Tc-I are not. LTD incorporates the relative mobility of species in that geochemical environment. If LTD is reduced by as much as LTH is, then there is no net increase in hypothetical dose from adoption of recycling.
- **Long-term radiotoxicity (LTR)** – the radiotoxicity of waste, independent of mobility and transport. LTR focuses on 1,000 years after discharge; a reduction of 100x relative to the throw-away fuel cycle means that the waste would be less radiotoxic than the equivalent initial uranium ore.

LTH and LTD are indicators of repository technical capacity. LTR has no regulatory or capacity value, but does help frame the hazard of the waste.

An AFCI program goal is to increase geologic repository utilization by factors of 10 to 50 so that the need for a second repository is deferred for a century or more.[DOE2005a, DOE2005b, DOE2006] The strategies with a fast reactor (2b, 2c, 2d) can accomplish this; the strategies without a fast reactor (1, 2a) cannot. Without fast reactors, transuranic material eventually accumulates and must be disposed. This limits the increase in repository utilization to ~2x to ~3x for strategy 2a, recycle in thermal reactors only. We denote that strategies 2b, 2c, and 2d are "sustainable" recycling. Strategy 2a can defer a second repository, but recycling is not sustainable.

Figure 1 compares heat-limit repository capacity improvements for multi-pass cases, either calculated here or in an earlier report.[Stillman2004c] The two MOX-NpPuAm cases match; the earlier report used actual thermal transient calculations; in this report we used the simpler LTH metric. CORAIL is a MOX-type approach with a different blending strategy than the MOX case used here and in Stillman2004c.

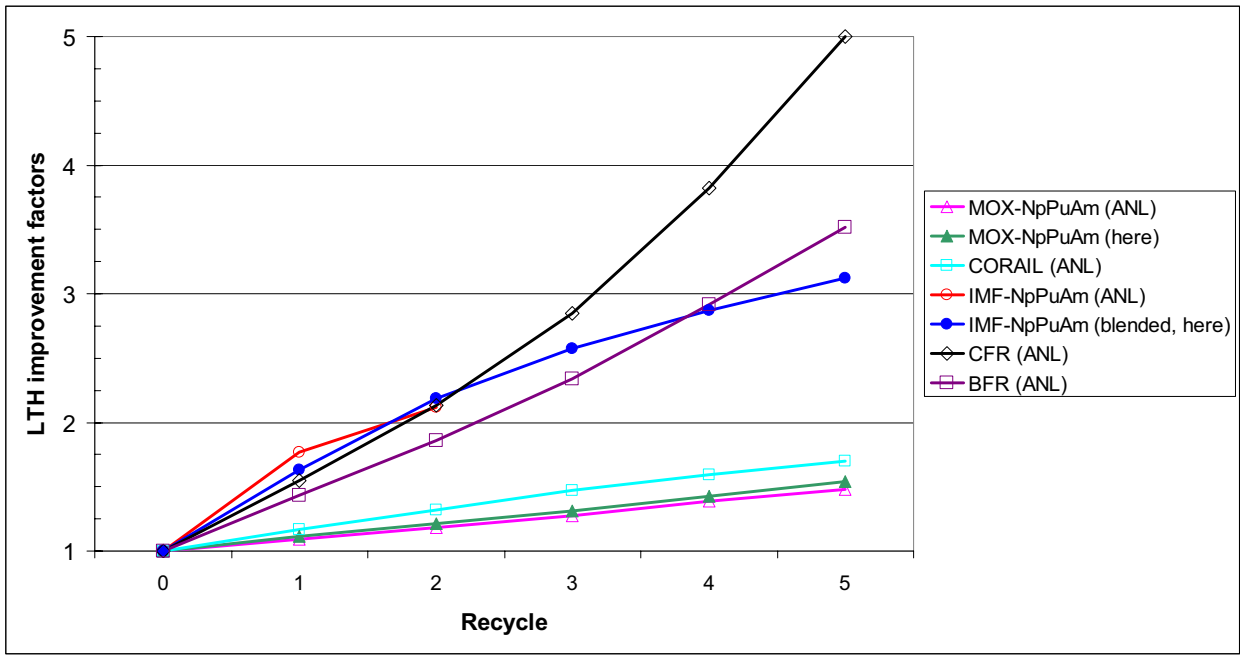

Figure 1. Comparison of multi-pass cases with 75-year ventilation in the repository. Cases marked "ANL" were calculated in [Stillman2004c].

Key observations from Figure 1 include:

- Even after 5 recycles, **none** of the cases achieve the AFCI program objective of 10-50x improvement in heat-limited repository capacity. Thus, sustained recycle is required.
- Full-core MOX does substantially worse than either blended core IMF or FR cases.
- 1-pass, full core IMF-NpPuAm is the highest after the first recycle, but at the cost of creating a residue that is very difficult to continue to use in an LWR. That curve stops at 2 cycles because then the accumulated TRU must be discarded. The blended core IMF case in this study more modestly burns TRU in the first recycle, but the approach can continue for additional recycles.

The same trends occur for long-term dose and long-term radiotoxicity. We find the following:

- Overall, sustained recycle of ~99.5% of Np-Pu-Am is required.
- In addition to 99.5% recovery of Np-Pu-Am, recovery of ~99% of Cs-Sr for independent heat management is required to meet LTH objectives.
- In addition to 99% recovery of Np-Pu-Am, LTD reduction can be constrained by Tc-I and uranium. Over 90% recovery of Tc-I, followed by tailored waste forms or transmutation appears needed. At least 90% recovery of uranium appears needed.
- In addition to >99% recovery of Np-Pu-Am, LTR reduction can be constrained by Cm as Cm and its daughters contribute ~1% to LTR at 1,000 years.

Regarding geologic repositories, we can rank order the strategies as follows:

- Throw-away [1] is least attractive as the cost of repositories increases.
- Recycle in thermal reactors [2a] defers the need for additional geologic repositories until the next century, but at some point accumulated TRU will lead to several repositories being needed.
- Recycle in fast reactors (with or without thermal reactors) [2b, 2c, 2d] can be sustained indefinitely and the need for additional repositories deferred for a considerable period of time.

So, the rank order is 1 < 2a < 2b/c/d (best).

Objective 2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for used fuel management.

We considered several indicators, such as the weapon-usable (WU) inventory measured in Pu239 equivalent masses (normalized by bare sphere critical mass). Figure 2 shows the WU inventory for a few of the cases studied.

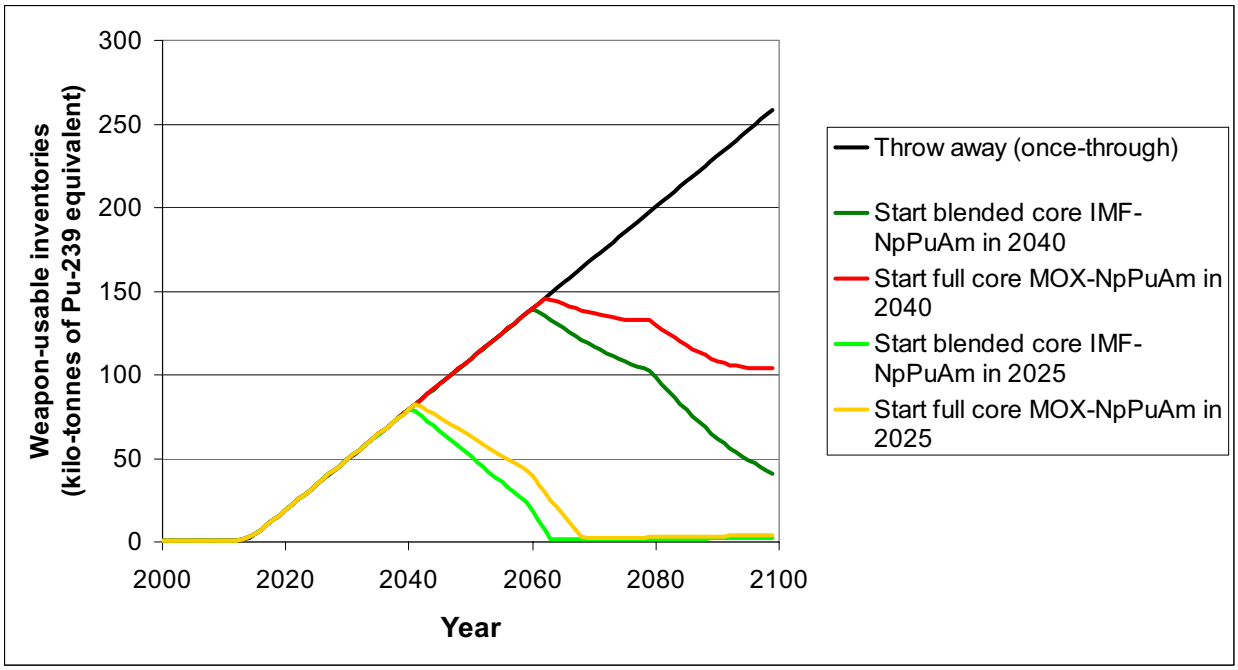

Figure 2. Comparison of weapon-usable system inventory for throw-away, blended core multi-pass IMF-NpPuAm, and full core, multi-pass MOX-NpPuAm.

The low conversion ratio systems – IMF and CFR – naturally minimize WU inventory. The blended core IMF case additionally has the lowest WU throughput as it consumes TRU the quickest (lowest conversion ratio), i.e., most consumption per unit mass flow.

An issue for the proliferation experts is how to view the intrinsic protective attributes of blended core approaches. For example, we considered an IMF-NpPuAm blended core approach with fuel assemblies with 75% UOX, 23% IMF-NpPu, and 1.5% Am targets. The transmutation effectiveness was similar (slightly better) than 75% UOX, 25% IMF-NpPuAm. UOX pellets/pins would be made hands-on; NpPu may qualify for glovebox fabrication; Am-containing fuels and targets would require remote fabrication. The concentration of Am into targets means that fewer fuel pellets and pins would have to be made remotely. The dose rate from the completed assembly with Am targets would be lower, because the targets would be inside; in contrast, the IMF-NpPuAm pins are on the outside of each assembly. Although these are economic benefits, are there proliferation risk penalties from separating Am from NpPu? For now, the rank order is unclear.

Objective 3. Enhance energy security by extracting energy recoverable from used fuel and depleted uranium, ensuring that uranium resources do not become a limiting factor for nuclear power.

Regarding uranium ore utilization, we can rank order the strategies as follows:

- Throw-away [1] is least attractive as uranium cost increases

- Recycle in thermal reactors or consumer fast reactors [2a, 2b, 2c] offers slight improvement, up to 40% relative to throw-away.
- Only becycle in breeder fast reactors [2d] offers major improvement, ~100x.

So, the rank order is 1 < 2a/b/c < 2d (best).

Objective 4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.

One safety indicator is throughput – the more material flowing through systems the more likely abnormal conditions arise. Throughput is also an economic indicator. Figure 3 shows the throughput for several multi-pass systems, compared to UOX-51. MOX-NpPuAm and IMF-NpPuAm are shown in the case where accumulated TRU is discarded after the 5th recycle; we do not know exactly when TRU accumulation will require TRU discard, but 5 cycles seems achievable. UOX/CFR, MOX/CFR, and IMF/CFR are equilibrium symbiotic systems; the CFR consumes TRU from the thermal systems. The BFR system equilibrium is 100% fast reactors. At CR=1.1, it has the highest recalculating inventory.

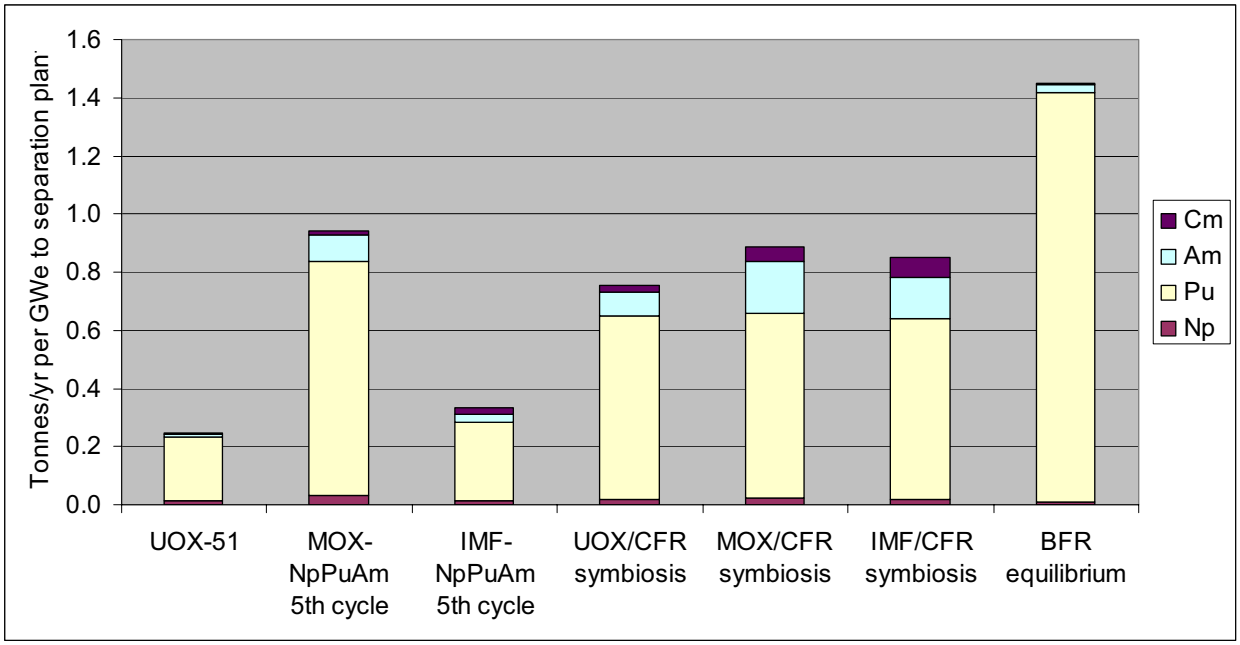

Figure 3. TRU throughput for selected systems, unlimited processing capacity

The strategies can also be rank ordered with regard to the relative importance of fast versus thermal reactor costs. In particular, as the cost of fast reactors relative to thermal reactors increases, one would expect recycle in thermal reactors [2a] to be superior to recycle in thermal+fast reactors [2b], in turn superior to recycle in fast reactors [2c]. Assuming fast reactors at CR=0.25, at equilibrium …

- Strategy 2c requires 27% fast reactors, 73% LWRs (all of the LWRs burn UOX)
- Strategy 2b requires 19% fast reactors, 81% LWRs (LWRs use a mix of UOX and IMF)
- Strategy 2a requires no fast reactors, 100% LWRs

So, the rank order if fast reactors are much more expensive than LWRs is 2d < 2c < 2b < 2a/1 (best).

Another Objective - Managing the Fuel Cycle System in spite of Uncertainties

Managing the fuel cycle system in a real-time fashion will not be easy. There is the potential to "out drive" our headlights. Consider that managing the fuel cycle is metaphorically like driving or flying a

plane. There are few "**control knobs"** available: what types of reactors are built, what types of fuels are used, and the capacity of separation and fabrication plants. All of the controls are very sluggish – with response times measured in decades. To compound the problem, there is no single driver; control is shared by utilities, other industry, government, and regulators. Worse, it is dark and raining (uncertain) and our headlights only illuminate a short distance into the future. Perturbations must be anticipated.

Table 3 lists potential control knobs that arose during this study. Future work is required to quantify each. Each potential control knob has costs; economic assessments are beyond the scope of this study. We can state based on hundreds of 2000-to-2100 fuel cycle simulations that at least some control knobs are required, but we do not see any evidence that all of these potential control knobs are needed.

Table 3. Potential Control Knobs of Varying Effectiveness.

Least flexible	Most flexible (most effective version of each control knob)
Miss the 2030-2050 window	Deployable before 2030
Single type of TR	FR+TR symbiosis (TR recycle serves as buffer for mismatch of FR building vs. UOX separation capacity.)
Fixed conversion ratio	Variable conversion ratio
UREX+1, PUREX, pyro	UREX+4
Homogenous core	Heterogeneous cores, e.g., IMF-UOX blends, targets, blankets
Don't recycle "legacy" used fuel	Use drawdown of "legacy" fuel as buffer

It would be beneficial to have a recycle strategy that could be implemented before the current reactor fleet retires in the 2030-2050 approximate time period so that replacement reactors fit into the strategy. The reactors built in that time period can determine much of the fuel cycle for the rest of this century, especially at low growth rates. As nuclear growth rates increase, the importance of the 2030-2050 time window decreases.

Multi-pass blended core IMF is a very effective downward Pu control knob. It can, for example, stabilize the Pu inventory even at 1.8% growth. And, for equivalent SNF throughputs, it can be implemented faster than MOX (*if the technology is available*) because of the low TRU throughputs; that is, for the same TRU separation capacity, IMF provides more ability to control inventories. IMF options can be tuned from conversion ratios near zero to at least 0.6. The capital investment of reactors would appear to exceed that of separation and fabrication facilities. If the IMF infrastructure is built and later not needed, thermal reactors can still be operated profitably. Blended core IMF appears a more effective and flexible control knob than MOX.

Establish fast reactors with ***flexible*** conversion ratio as a control knob. This "control knob" takes longer to become available because fast reactors must be tens of percent of the fleet before effects can be seen. The conversion ratio should be variable from ~0.25 to at least 1.3. Unlike the IMF control knob, this one can substantially reduce uranium ore needs if the conversion ratio is over one. IMF and MOX used in conjunction with CFR reduces the number of CFRs needed, and delays when they are needed.

Systems offering separation among TRU, e.g., UREX+4, offer more flexibility as the ratio among NpPu, Am, and Cm in fuels can be controlled. UREX+1 and pyroprocessing lead to group separation of NpPuAmCm, removing one potential control knob, but possibly having proliferation risk management benefits.

We should expand exploration of heterogeneous assemblies/cores and targets, which appear to have advantages and agility. The need for blended cores in fast reactors is well known (core + blanket).

Analyses suggest advantages for blended assemblies and targets in thermal reactors. In particular, the blended core multi-pass IMF approach in this study offers significant advantages as well as agility relative to full core MOX. Even better might be separating IMF-NpPu versus Am targets, so that little of the fuel would require remote fabrication. Preliminary analysis [Goldmann2005] indicates similar transmutation performance, but segregating the Am into targets minimizes the amount of fuel requiring remote handling.

Another Objective – Robustness and Agility

No single fixed strategy guarantees optimal performance at all times in all possible futures. Instead, the objective in the next few decades should be to cost-effectively develop the technologies to handle potential future circumstances. Thus, two criteria among options should be robustness and agility. Robustness measures how much preferences stay constant if postulated assumptions and future circumstances change. Agility measures the ease of adapting an option if new circumstances warrant.

As an example, we find that the multi-pass blended-core IMF approach would be more robust than the multi-pass full-core MOX approach in several ways. One is that the chemical composition of recycled material changes significantly cycle-by-cycle for MOX, but not for IMF. Separation and fabrication plants with fixed capabilities would thus be able to handle a wider range of IMF situations than MOX.

Figure 4 shows some of the key preferences described above. If the growth of nuclear power is low so that neither repository space nor uranium ore are constraints, then the throw-away fuel cycle [1] may be preferred. If nuclear growth is very high and/or the cost of uranium ore increases substantially, the breeder fast reactor [2d] tends to be preferred. If nuclear growth is intermediate, so that the constraints and costs of additional geologic repositories are serious issues, but the cost of uranium is less so, then recycle in consumer-mode reactors [2a, 2b, or 2c] should be preferred. The 2005 Report to Congress from the AFCI program [DOE2005a] indicates that this is the expected condition, i.e., that uranium costs are not expected to a serious problem for the foreseeable future but avoiding the cost of additional geologic repositories is a high priority.

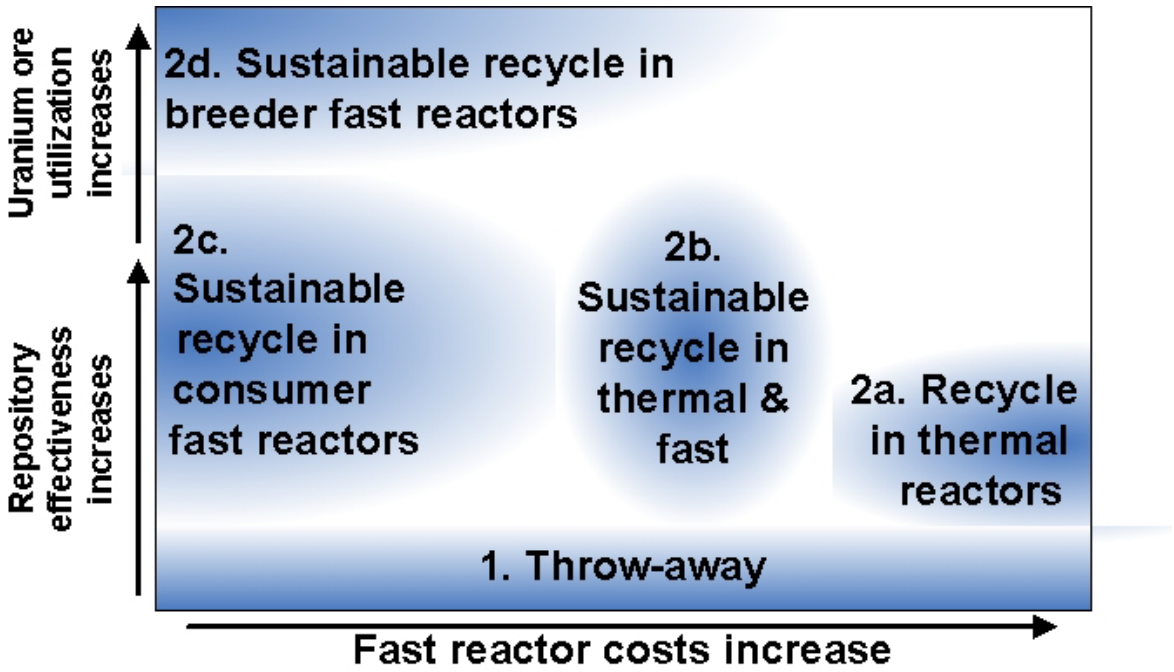

Figure 4. Hypothetical preferred fuel cycle strategies under different conditions

If fast reactors have similar costs as thermal reactors and a sizable fast reactor fleet can be built relatively quickly (compared to repository space concerns) then recycle in CFRs tends to be preferred over recycle in thermal reactors because recycling in the fast reactors is more sustainable than in thermal reactors.

Figure 4 also shows that demonstration of burner fast reactors is an important step to show the viability of much of the option space, especially since we already know we can operate fast reactors in a breeder configuration.

The most adaptable or flexible option would appear to be strategy 2b - sustainable recycle in a mix of thermal and fast reactors. It sits in the middle of option space in figure 4. If fast reactors prove too expensive, they could be deferred in favor of recycling in thermal reactors. If fast reactors prove very attractive, thermal reactors could be phased out. Such adaptability would come at a price – having to demonstrate both CFRs and either IMF or MOX in thermal reactors. For a given amount of desired plutonium or TRU consumption, calculations show that IMF is more effective than MOX and has lower mass throughput. Thus, unless IMF is infeasible or has abnormally high costs, it is preferred versus MOX. (Since the MOX throughput is 2-3 times higher than IMF throughout, the "per mass" unit costs of IMF could be 2-3 times higher than MOX and could be a "wash" economically.)

Strategy 2b, however, appears to have relatively high R&D costs because all technologies required for strategy 2c are also needed for strategy 2b, plus additional ones. Thus, a possibly better strategy is 2c – sustainable recycle in consumer fast reactors - which retains most of the flexibility of strategy 2b, but with lower R&D costs. This is indeed what GNEP proposes to do.

Figure 5 illustrates how strategy 2c, sustainable recycle in consumer fast reactors, is adaptable. If the cost of geologic repositories (either economic or political costs) are low, deployment of consumer fast reactors can be slowed relative to plan. If the cost of uranium ore increases substantially, the deployed fast reactors can be shifted from consumer to breeder mode. If the cost of fast reactors is too high, then recycling in thermal reactors should be considered to reduce the need for fast reactors.

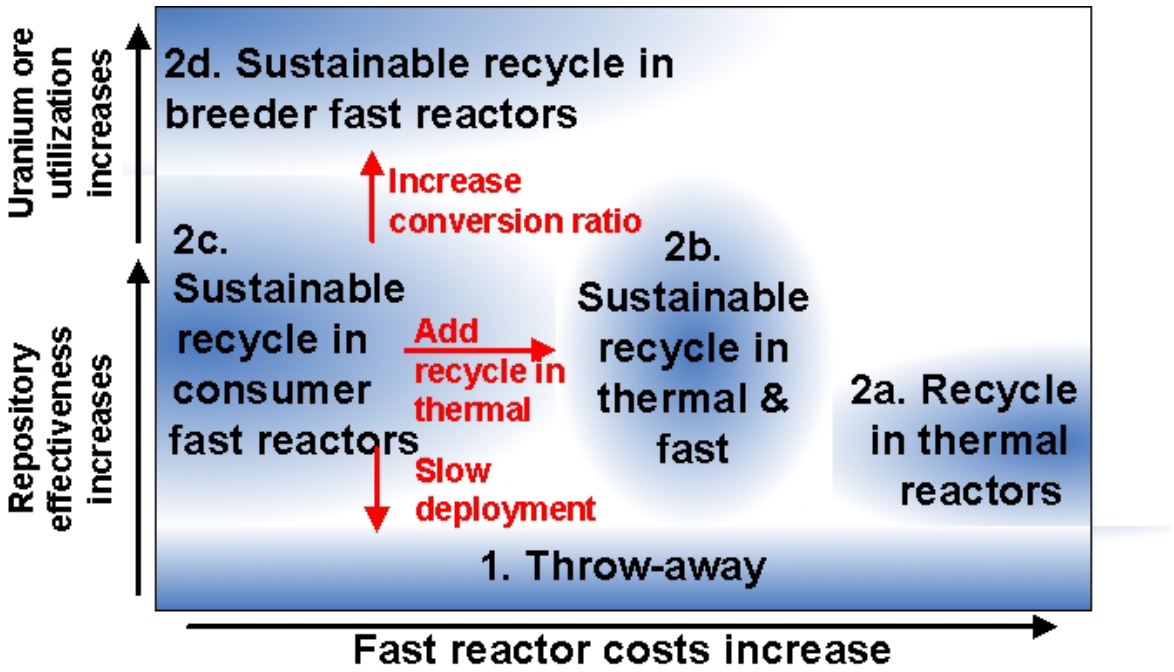

Figure 5. Adaptability of GNEP-like strategy (consumer fast reactors) if conditions change

CONTENTS

1. INTRODUCTION AND FRAMING

1.1. Purpose

This report clarifies many technical issues being analyzed by the Advanced Fuel Cycle Initiative (AFCI) program, including Inert Matrix Fuel (IMF) versus Mixed Oxide (MOX) fuel, single-pass versus multi-pass recycling, thermal versus fast reactors, the need to recycle Np-Pu-Am to meet established AFCI objectives, the borderline case of Cm, the potential need for transmutation of technetium and iodine, and the value of separating cesium and strontium. This report represents the first attempt to calculate a full range of metrics, spanning all four AFCI program objectives [DOE2005a, DOE2005, DOE2006] - waste management, proliferation resistance, energy recovery, and systematic management/economics/safety - using a combination of "static" calculations and a system dynamic model, DYMOND.[Moisseytsev2001, Yacout2005a] (In late FY2006, DYMOND is being replaced with the VISION model.) Thus, instead of only static results, such as reactor-1 can achieve 10% higher uranium ore utilization than reactor-2, the analyses compare various scenarios through the end of the 21st century in a dynamic fashion. These dynamic analyses are supplemented with static analyses where relevant, such as the observation that fuels containing americium (thermal or fast reactor) will require remote handling. In many cases, we examine the same issue both dynamically and statically to determine the robustness of the observations, e.g., IMF scenarios have lower throughputs of Np, Pu, and Am (mass/yr in separation and fuel fabrication plants) than either MOX or fast reactor scenarios. All analyses are for the U.S. reactor fleet. The results presented here were used in various FY2006 reports, such as the FY2006 Comparison Report to Congress.[DOE2006]

The report's analyses were produced by INL, ANL, and SNL personnel under their Simulation, Evaluation, and Trade Study (SETS) work packages during FY2005 with followup work in FY2006.

This is a technical report, not intended for a policy-level audience. A wide range of options are studied to provide the technical basis for identifying the most attractive options and potential improvements. Indeed, we do not believe that any of the specific options presented here are the most optimum or economically feasible. Technical maturity and readiness to deploy were outside the scope of this report.

Trade-offs can not be resolved without a "decision context," i.e., what decision are we trying to make, when, under what conditions.

For example, consider the issue of disposition of long-lived technetium and iodine. Any attempt to make a simple decision (transmute versus don't transmute) invariably is based on various assumptions (often implicit) as to what decisions would have been made prior to that decision and on how various uncertainties will be resolved. Different assumptions can lead to different conclusions; in this report we attempt to make the various assumptions explicit and indeed to show how a preference could change depending on assumptions, uncertainties, etc. Where possible, we then identify what is the most robust position *at present*. In the case of Tc and I, it would be to separate them, without commitment to whether they would be made into specialized waste forms (relatively likely) or transmutation targets (relatively unlikely) or taken to a different geologic disposal site. However, a decision to open the 1st repository (and therefore finalize its licensing basis) will have occurred before we need to make a final Tc-I decision.

For another example, consider the urgency of fast reactor deployment, which depends on (at minimum), (a) whether uranium resources are believed to be a constraint on a reactor-operation time frame, (b) whether fast reactors are needed to address the timing of AFCI waste mgt benefits (not just enough to

keep stuff out of repository but also must burn it up), (c) whether fast reactors success in the market place even in the absence of U resource and TRU-transmutation needs.

Throughout the report we use Light Water Reactors (LWR) as the reference thermal reactor (TR) and the Sodium Fast Reactor (SFR) as the reference fast reactor (FR). We do differentiate between a fast reactor used as a net consumer of TRU – breeding ratio and conversion ratio less than one (typically 0.25) – called the Consumer Fast Reactor (CFR) and a fast reactor used as net breeder – breeding ratio and conversion ratio about 1.07 at equilibrium – called the Breeder Fast Reactor (BFR). More analyses have been done with the LWR and SFR than other systems. Furthermore, available data [Taiwo2005] suggests that the differences among thermal reactor concepts, and the differences among fast reactor concepts, are minor compared to the differences among once-through, recycling in thermal reactors, and recycling in thermal reactors, which are the primary options considered in this report.

1.2. What is in this report?

We recommend Chapters 1 and 2 and 9 to everyone. Chapter 1 explains our approach and sets the context. Chapter 2 contains results from the standpoint of five key decisions that we identify:

- Open 1st geologic repository
- Determine credibility of recycling
- Build 1st separation plant (for UOX).
- Build 1st recycle-fuel fabrication plant.
- Build future separation and fuel fabrication plants.

For each decision, there are comparisons (qualitative, quantitative from DYMOND analyses, and quantitative from static analyses) of major choices available for each decision. In particular, we consider how robust decisions appear to be with regard to six key factors, which are introduced in section 1.5.

- Growth of nuclear energy?
- Cost and acceptance of additional repositories?
- Which reactors succeed in the market place?
- How much uranium is available?
- What proliferation policies exist?
- How much penalty is “hot” fuel separation and fabrication?

Chapter 9 contains the highest-level conclusions from the analyses. The other chapters contain the details underpinning chapters 2 and 9.

Chapter 3 describes AFCI objectives and the metrics used in this report to judge each. Of particular note are the waste management metrics – long-term heat (LTH), long-term dose (LTD), and long-term radiotoxicity (LTR).

Chapter 4 explains the AFCI options analyzed in this report. Of particular note is that the approach to multi-pass MOX involves using burned uranium (BU) for each recycle; to keep the fuel working, the plutonium content is increased each recycle. The approach to multi-pass IMF involves a blended core (3/4 of pins are UOX, 1/4 are IMF); the TRU in each generation of IMF pins comes from the residual TRU of the previous generation, both UOX and IMF. To keep the fuel working, the burnup is decreased slightly each cycle. Transmutation colleagues have taken multi-pass MOX and IMF to 5 cycles, which would be adequate through the end of this century.

Chapter 5 contains "static" analyses such as the support ratio of MOX-NpPuAm to UOX (ratio of 13.5) and metrics associated with each of the four AFCI objectives. The support ratios and the flux of recycling in the system (throughput) are quite important.

Chapter 6 contains scoping analyses for a broad range of development trees. Development trees denote selection among once-through, thermal recycling options, and fast reactor recycling options. In this analysis, we assess six development trees, each starting in 2025.

- Continue Once-Through– delaying recycling until at least 2040
- Start thermal recycling with Inert Matrix Fuel (IMF), specifically IMF-NpPuAm
- Start thermal reactor recycling with Mixed Oxide Fuel (MOX), specifically MOX-NpPu
- Start thermal reactor recycling with MOX-NpPuAm
- Start Consumer fast reactors (CFR)
- Start Breeder fast reactors (BFR)

There are decision points causing branching of each tree in 2040, 2060, and 2080. For example, one branch starts IMF-NpPuAm recycling in 2025, then phases out recycling in 2040 – providing an answer to the question, "what if you want to stop recycling once you start?" The dates 2025 and 2040 are consistent with the 2005 Report to Congress [DOE2005a], which set 2025 as the target date for beginning thermal recycling and 2040 as the target date for deployment of first fast reactors (hence fast reactor recycling).

Chapters 7 and 8 are DYMOND calculations. In Chapter 7, the comparisons are structured by "development tree." For example Development Tree 2 has IMF-NpPuAm starting in 2025, with several branches in 2040. In Chapter 8, the comparisons are by issue, e.g., single-pass vs. multi-pass, and multi-pass IMF vs. multi-pass MOX.

1.3. Making and framing decisions

Changing the status quo is especially difficult in the following circumstances:

- Multiple objectives are to be met.
- A wide range of options exist, each with their own advantages and disadvantages and with their own advocates and opponents.
- Consequences of decisions are high, e.g., high cost and long time scales. Fuel cycle facilities probably have price tags of 10's of billions of dollars, at least 0.5 decades between decision-to-build and facility-in-operation, and facility lifetimes of 3 to 6 decades.
- Uncertainties are large; the "right" thing to do is viewed as depending on resolving the uncertainties. See Section 1.6 for six key uncertain decision factors addressed in our analyses.
- Solutions are viewed as irreversible and inflexible, i.e., might regret the decision later. Decision makers at all levels tend to avoid decisions with the potential for "high regret." This is also posed as "what if you are wrong."
- The status quo is viewed as acceptable.
- There is much inertia to the status quo, e.g., large infrastructure and industry capacity associated with business-as-usual.

To illustrate – One line of argument [MIT2003, Wald2004] is that we do not know exactly what repository we need, how a repository would perform over millennia, nor if recycling makes sense, therefore the U.S. should neither open a repository nor commit to recycling. We should only study the problem; it is not stated when there would be enough "study" to justify action.

However, the U.S. fuel cycle status quo is not sustainable because there is no identified and implemented path for nuclear waste from current and future nuclear power plants. The U.S. status quo has the following characteristics.

- Unimplemented policy to open a first geologic repository.
- Policy to grow nuclear power (e.g. new energy bill), but unproven market implementation of this policy.
- No path identified beyond first geologic repository so that there is no identified way to handle nuclear growth, neither additional repositories nor recycling.

Changing the status quo requires the following:

- Clarify objectives as much as possible.
- Identifying solutions that are relatively robust in spite of uncertainties. A common example in the AFCI program is the robust conclusion that at least one geologic repository is required in all scenarios. An implicit component of this conclusion is that we know enough about the characteristics of the first repository to warrant proceeding whether once-through or recycling will be used throughout this century.
- Addressing the "what if you are wrong" question.
- Addressing agility, i.e., how one can adapt to changing circumstances.
- Addressing urgency, i.e. what happens if decision is delayed. In this report, urgency is addressed by consideration of delay in recycling from 2025 to 2040.

Figure 1-1 illustrates some of these concepts. Faced with opposition to making a decision, the classic thing to do is to push hard ("brute force") to overcome resistance. Indeed, sometimes this works, but sometimes it does not as the opposition can push just as hard in response. Instead, we can attempt to change the picture by (a) lowering the barrier by reducing the consequences of being wrong and (b) split a single decision into a network of staged decisions.[Piet2003] The "what if you're wrong" concept motivates such as questions as "how do you stop recycling" or "how can you exit recycling once you have started?". The answer should be

1. Take decisions one step at a time, while foreseeing future paths.
2. Take decisions when the wisdom of the action is relatively robust, i.e., will be seen to be appropriate under most foreseen futures. That is keep doors "open" until there is a good reason and good justification for closing them.
3. Always have a baseline that can be pushed forward if the need for a then-current decision arises.
4. Consider adaptability as one of the criteria for selecting among alternatives, i.e., have potential answers to "what if you are wrong" in hand before selecting an alternative.

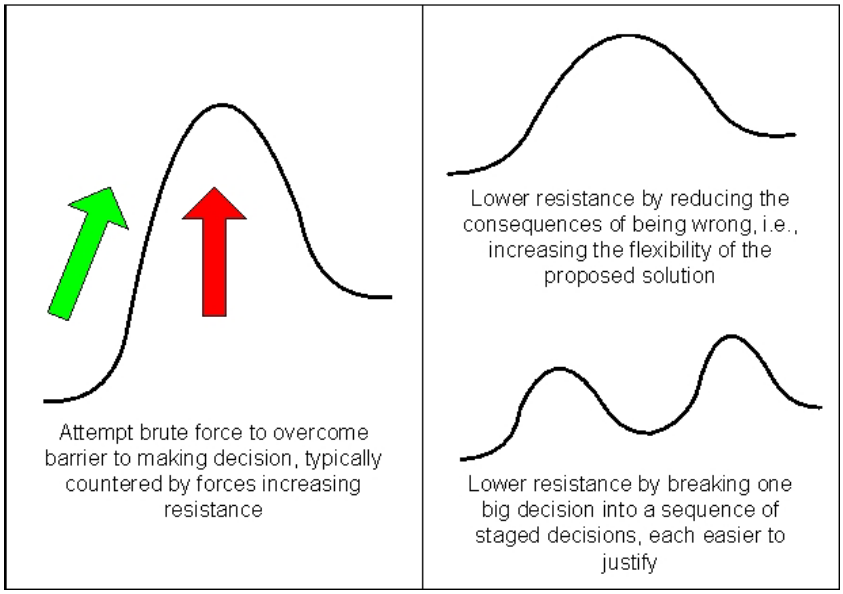

Figure 1-1. Overcoming resistance to making and keeping a decision

A related complication is that any future fuel cycle decisions are not made starting with a "blank slate" in which all options would be compared on a clear, consistent basis.

If we had a "blank slate," then we would have a systematic comparison of all possible nuclear energy technologies within the context of a comparison of all energy options. For example, we might, or might not, select Light Water Reactors (LWRs) if the nuclear energy enterprise were starting from scratch today.

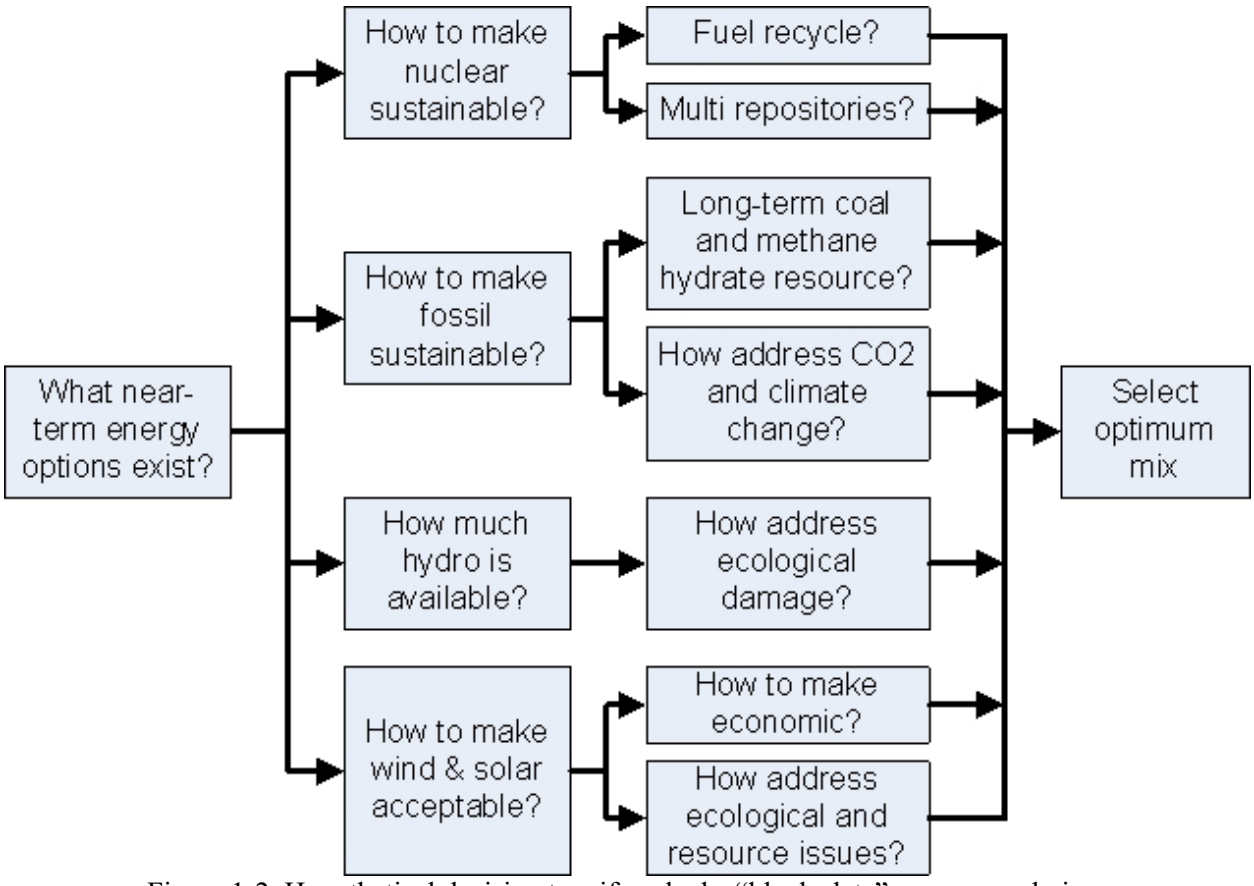

Figure 1-2. Hypothetical decision tree if we had a "blank slate" on energy choices

We do not have a blank slate. For example, our interpretation of a recent French report [Delpech2004] suggests the decisions (and their order) under consideration in France. Their "slate" is built on already deciding to continue nuclear and PUREX in all scenarios. The issue of fast reactors is posed as "when", not "if." The issue of Am is considered in this context, namely should Am be recycled in thermal reactors while waiting for fast reactors.

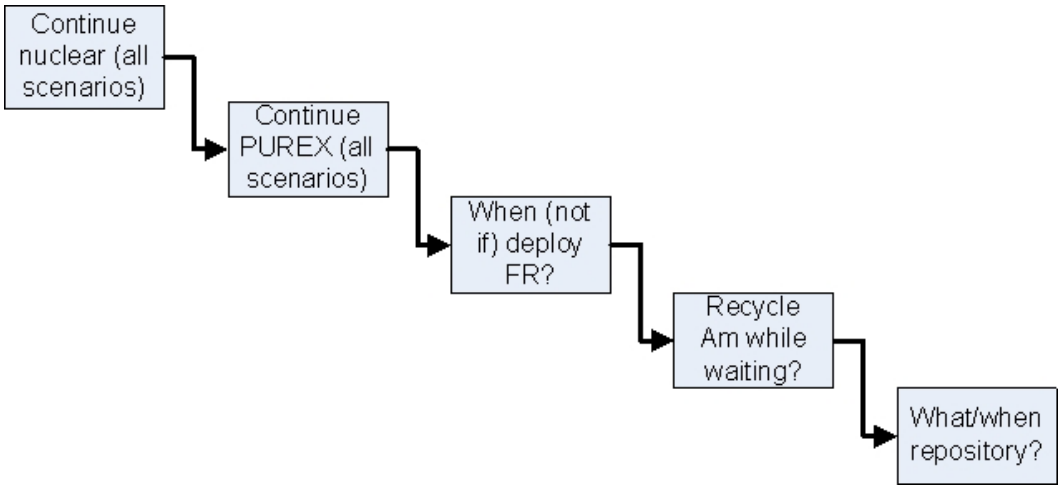

Figure 1-3. Interpretation of nuclear fuel cycle decisions in France

The U.S. situation is different. Indeed, Table 1-1 summarizes differences between the U.S. and France to illustrate that the context and status for decisions have significant differences.

Table 1-1. Comparison of Fuel Cycle Constraints and Status between U.S. and France

	U.S.	France
Policy and market commitment to nuclear power	Currently starting to rebound; the 2005 Energy Policy Act establishes the policy; we will now see how the market responds.	High, 80% of electricity produced in France is generated by nuclear power, France is a net exporter of electrical power, strong government/private industry collaboration
Policy and licensing commitment to 1st repository	Policy – yes Licensing – in progress The U.S. tends to use long-term heat and dose from a prototypical repository to judge geologic repository benefits (Long term heat addresses the physical design limitations of the repository and dose addresses the potential transport of isotopes assuming a specific engineered system and geologic location.)	Neither France tends to use radiotoxicity to judge geologic repository benefits. (Radiotoxicity addresses the maximum potential risk of the isotopes with no mitigation from engineered waste forms, containers, or geology. It is therefore independent of location and configuration.)
1st recycle plant	No- historically because of proliferation concerns in 1977, low price of uranium	PUREX plant in operation. Modified process may be useful with fast reactor oxide fuel.
Ability to dispose separated uranium in near-surface burial	Yes (if U separated at high purity to qualify under 10CFR61)	Probably not, because French regulations for near-surface burial appear more stringent in this respect.
Ability to dispose separated Cs and Sr in near-surface burial	Yes after 1-3 centuries cooling (if Cs-Sr separated at high purity to qualify under 10CFR61)	
Proliferation policy	No separate Pu is allowed	PUREX plant separates pure Pu

The difference in repository status is instructive. The U.S. is proceeding with the YMP, hence the U.S. AFCI program considers repository-specific long-term dose and long-term heat metrics, as opposed to only radiotoxicity. A radiotoxicity approach puts relatively more emphasis on the TRU, especially Pu. A dose approach puts relatively more emphasis on easily transportable elements such as uranium, Tc, I versus relatively slow transporting elements such as plutonium. The heat issue puts emphasis on plutonium and americium.

On the other hand, the French have a separation and recycle-fuel fabrication plant in operation, with all the experience (and inertia) associated with them. Consistent with this, they tend to emphasize fast reactor fuels, e.g., oxide in SFR, that could be handled by that plant or a modest modification thereof.

But, perhaps the key observation for the U.S. is that the continuation (let alone growth) of nuclear power is not assured nor mandated. Thus, some AFCI program critics argue against implementing recycling in part because of the lack of certainty that recycling will be needed ever because of the potential that nuclear power will be phased out or grow very slowly. Indeed, Figure 1-4 illustrates our understanding of key decisions facing the U.S., starting with opening the first geologic repository at Yucca Mountain

Project (YMP) observing/predicting nuclear growth, and then deciding either a multi-repository path or a recycling path.

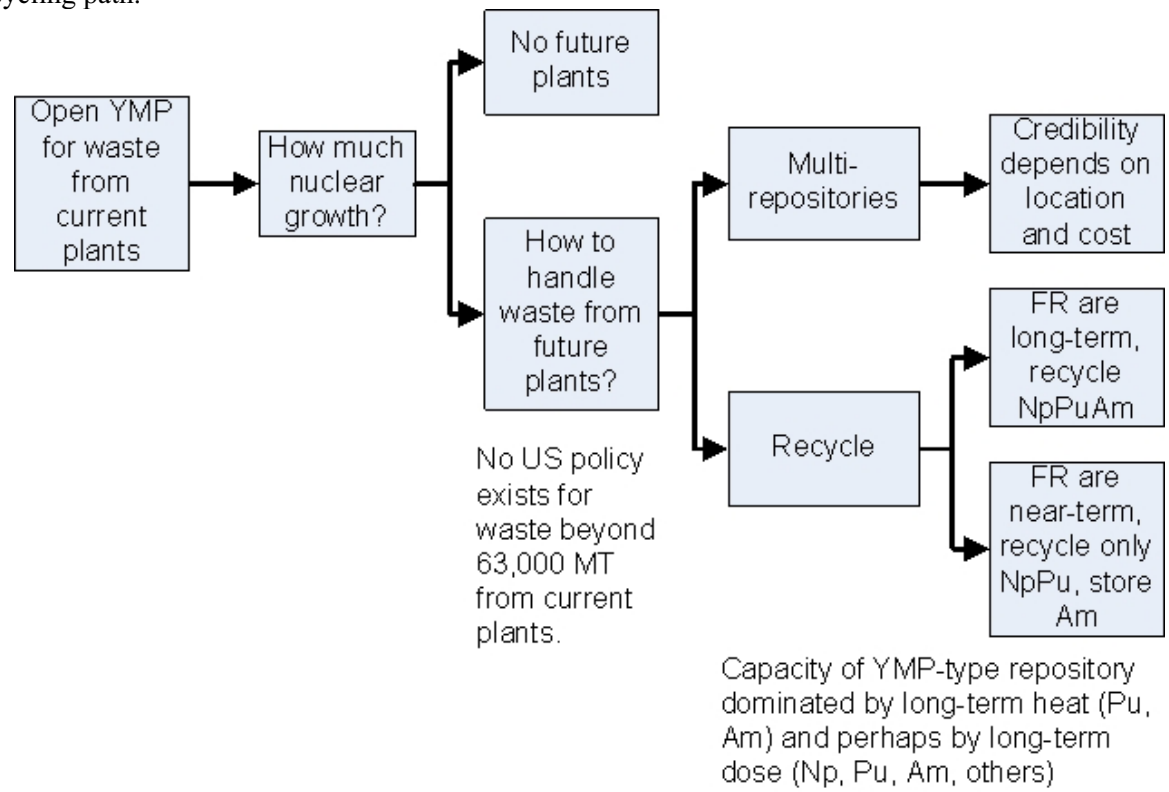

Figure 1-4. High-Level U.S. Nuclear Fuel Decision Tree

The decisions in Figure 1-4 form the columns (nuclear growth) and the rows (multi-repository vs. recycling without fast reactor vs. recycling with fast reactors) in the energy future table.[Dixon2004, Piet2004]

The version in Table 1-2 emphasizes the interplay among nuclear growth, repository capacity, fuel cycle approach, and required number of repositories. The "limited thermal recycle" rows correspond to a few recycles in thermal reactors. It is difficult to get beyond a repository capacity improvement factor of ~1.5 with multi-pass MOX but perhaps a repository capacity improvement factor of ~3 could be possible with multi-pass IMF using blended UOX/IMF assemblies, the basis for these number is presented in Chapter 5. The "repeated" recycle rows require fast reactors so that TRU never has to be discarded, the improvement factor is then controlled by the loss rates each recycle. Shaded (yellow) cells require more than one repository; green cells require only one repository. By definition, if the repository capacity is 70,000 tonne (63,000 of which is for commercial SNF), in only the "Legislative Limit" future is one repository adequate in the absence of recycling. The nation is already implementing "Existing License Completion" meaning that (a) the repository capacity must be increased to at least 120,000 tonne (plus allowance for defense wastes), (b) a second repository must be built, or (c) recycle of at least some SNF must be done. By 2100, without recycling of some type, there could be 22 repositories worth of SNF generated from the decay heat perspective at a growth rate of 3.2%/yr or 50 at a growth rate of 4.5%/year.[DOE2006]

Table 1-2. Illustration of how a Factor (Nuclear Growth in this case) influences Nuclear Fuel Cycle Decisions

Nuclear Futures		Legislative Limit	Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share Generation	Growing Market Share Generation
Corresponding Growth Rate		Nuclear phase out			0%/year	1.8%/year	3.2%/year
Cumulative discharged fuel in 2100 (tonne-iHM)		63,000	100,000	120,000	250,000	600,000	1,500,000
Fuel Management Approach	Repository Capacity (tonne)	**Number of Geologic Repositories Needed to Accommodate Fuel Discharged by 2100**					
No recycle	70,000	1.0	1.6	1.9	3.7	8.7	21.6
	119,000	0.6	0.9	1.1 (a)	2.2	5.2	12.7
Single-pass thermal recycle (capacity multiplication 1.7x)	70,000	0.7	1.0	1.2 (b)	2.3	5.2	12.8
	119,000	0.4	0.6	0.7	1.3 (c)	3.1	7.5
Thermal/fast symbiotic recycle (capacity multiplication of 50x)	70,000	0.2	0.2	0.2	0.2	0.3	0.6
	119,000	0.1	0.1	0.1	0.2	0.2	0.4

Repository capacity dominated by temperature limits, hence decay heat density.

a. Borderline case, can be reduced to 1 repository with further repository capacity increase from 119,000 to 127,000 tonne or a 10% increase in fuel burnup.
b. Borderline case, can be reduced to 1 repository with 20% increase in fuel burnup.
c. Borderline case, can be reduced to 1 repository with further repository capacity increase from 119,000 to 154,000 tonne or a 30% increase in fuel burnup.

As noted above, neither multi-repository nor recycling have been determined to be credible in the U.S. Recycling might be a net increase in cost, but not a prohibitively expensive increase. In the "recycle" decision branch, two main options denote whether one believes that fast reactors are relatively near-term, in which case it may be best to introduce the recycle materials directly into fast reactors or recycle only NpPu in thermal reactors (storing hard-to-handle elements like Am and Cm), or relatively long-term, in which case it maybe best to recycle NpPuAm (possibly Cm) in thermal reactors thereby reducing the inventory of Am241, which is so important to long-term heat in a repository.

However, Figure 1-4 is too simple because it ignores feedback loops and other factors (beyond AFCI control) that would influence the decisions denoted in Figure 1-4. For example, there are good reasons to suppose that the more sustainable nuclear power appears, the more nuclear growth. Thus, opening the first repository and establishing the credibility of either multi-recycle or multi-repository would seem to increase the potential for higher growth in nuclear energy. As an example, consider California state law, which prohibits construction of new nuclear power plants until a geologic repository opens. Thus, it is not as simple as waiting until we see if there is nuclear growth, and then plan to make nuclear power sustainable. Instead, there is a feedback between steps increasing confidence in nuclear sustainability and evidence that nuclear will indeed grow.

Therefore, we expand Figure 1-4 to include additional decisions, to explicitly mention six key decision factors (such as nuclear growth), and to show relationships among them. The six **factors** are described in Section 1.5; the five **decisions** are described and analyzed in Chapter 2.

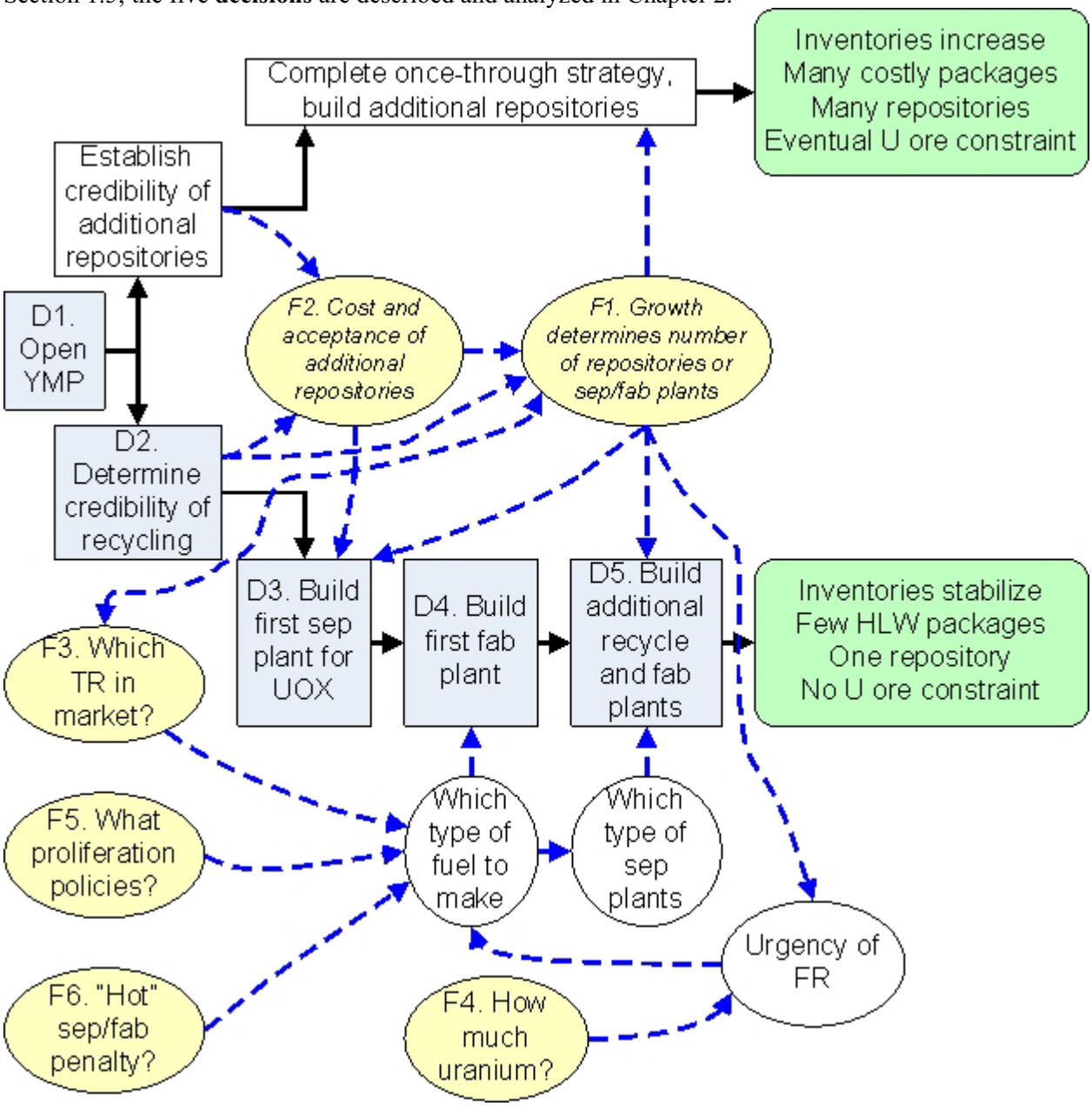

Figure 1-5. Suggested framework of decisions and key factors

Note that Figure 1-5 differentiates between "can recycle" (D2) and "should recycle" (D3). It is increasingly likely that we can get agreement on D2. A favorable D3 depends on D2 and several other factors, most notably nuclear growth (F1) and the cost and acceptance of other repositories (F2).

In 2004, the AFCI program spoke in terms of phases. The outcome of the decisions in Figure 1-4 map into those phases:

- D2=no ➔ continue once-through and start finding location for additional repositories.
- D4=no ➔ separation for waste management only
- D5=no ➔ limited recycle (only single-pass)
- D5=yes, F4=high ➔ low urgency for fast reactors ➔ continuous or "transitional" recycle using little or no fast reactors
- D5=yes, F4=low ➔ high urgency for fast reactors ➔ sustained recycle with fast reactors

As explained in Chapter 2, we define and order the decisions in decreasing order of robustness and hence decreasing order of readiness to make each decision. Essentially all pro-nuclear supporters support D1; in part because D1 makes sense regardless of how much growth one envisions and whether the 1st repository is considered to be followed by more repositories or by recycling. The support for D2 is increasing, because in the absence of establishment of the credibility of multiple repositories, there is a growing consensus of the need to determine the credibility of recycling.

A key feature of Figure 1-5 is the attempt to show feedback loops. For example, establishing the credibility of multi-repositories (F2) and/or recycling (D2), hence showing a path for waste from future nuclear plants, likely influences how much nuclear growth occurs (F1). F1, in turn, influences many of the other factors and most of the decisions (directly or indirectly).

1.4. Limitations and key assumptions

The first major limitation of this study is that thermal reactors (TR) are always represented by Light Water Reactors (LWR) and fast reactors (FR) are always represented by Sodium Fast Reactors (SFR). Processing of thermal reactor fuel is always assumed to be done at centralized plants using UREX+ technology. Processing of fast reactor fuel is always assumed to be done at power plants using pyroprocessing technology. To first order, we do not believe that the conclusions in this report would differ substantially for other thermal or fast reactor options, based on ANL transmutation analyses.[Taiwo2005] However, we emphasize that we have not looked at ultra-high burnup in any thermal reactors (LWR or VHTR) in this report.

The second major limitation is that there is no attempt to include economics *per se*. Instead, economic indicators are used. For example, although we do not attempt to provide relative or absolute cost values, we do examine metrics such as separation and fuel fabrication throughputs, and the relative amount of fuels that require remote handling (those including Am or Cm), glovebox operation (those including Pu), or current hands-on fabrication (uranium-only).

The third major limitation is that we assume that all options studied are technically feasible, in particular, that multi-pass Inert Matrix Fuel (IMF) is feasible. Similarly, all options are assumed available at the time indicated in various deployment scenarios, i.e., the necessary R&D&D has been done.

The fourth major limitation is that detailed fuel cycle data is only available for a finite subset of specific recycle approaches. Great care has been taken to assure that the fuel cycle performance for each case has been analyzed in a consistent manner. However, not all promising options have been considered. For example, detailed analyses of IMF recycle in a blended core have been included in this study; whereas, the current MOX recycle cases are for full-core loading. In future work, the scenario evaluations will be utilized to define additional cases for detailed analyses; and new fuel cycle data on specific options (e.g., BWR recycle, extended cooling options) will be incorporated into the dynamic model, as available.

The fifth is that isotopic decay is not accounted for at any of the long-term storage facilities. Isotopic change is accounted for in the reactor but as the fuel moves from the reactor to repository or reprocessing there is no isotopic change due to decay accounted for in the current model. In late FY2006, we are shifting to the VISION system dynamic model, which does account for isotope decay. So far, we do not observe changes that would change the conclusions in this report.

Only fuels containing only uranium can be fabricated hands-on, as is current industrial practice with uranium oxide (UOX) fuel. The addition of Pu requires glovebox fabrication. The further addition of Np does not appear to change this. Based on scoping analyses (Chapter 5), generally the addition of either Am or Cm requires remote fabrication of fuel pellets and pins, whether the fuel is to be used in thermal or fast reactors. This requires more analysis.

We emphasize that this does not necessarily mean that fuel assembly fabrication and fuel assembly handling at power plants must also be remote. The shielding by and within the assembly can be significant. For example, it may be that fuel assemblies with Am targets located near the center of the assembly would not cause power plant operators to require remote handling of the assembly; it is certain that concentrating Am toward the assembly center would have lower doses than distributing Am throughout the assembly (whether the assembly is for use in thermal or fast reactors). Heterogeneous assemblies (indeed blended cores) may therefore offer a way to obtain waste management and proliferation advantages of recycling Am with limited economic penalties.

As with other AFCI analyses, we assume that the first geologic repository has the characteristics of the Yucca Mountain Project (YMP), specifically the anticipated long-term heat and long-term dose limitations. Thus, we incorporate long-term heat (LTH) analyses by R. A. Wigeland [Wigeland2004a, Wigeland2004b, Wigeland2006] by calculating the heat released by each isotope emplaced in the repository in units of watts-years/gram of the isotope at time of emplacement, where the heat-interval is calculated from when repository ventilation stops to about 1500 years. We incorporate long-term dose (LTD) by scaling DOE-RW results provided by W. Halsey.[Halsey2005] Our calculated units are mrem/year at a particular future time from all isotopes that arise from a gram of each isotope at time of emplacement in the repository. A gram of Am241 at emplacement, for example, provides dose several thousand years later because of its decay eventually into Np237, itself having radioactive daughters. These LTD metrics therefore incorporate the various assumptions in the DOE-RW YMP analyses, especially its oxidizing conditions and the fact that Np solubility is limited.

We do not explicitly consider the potential of repositories with other heat or geochemical characteristics. The opposite heat case from YMP would be a “wet” repository, in which heat management would be a far easier problem and thus LTH metrics less important. (A wet repository would have, of course, other problems.) The opposite dose case from YMP would be reducing geochemistry. Although we do not analyze LTD metrics under reducing conditions, we do estimate long-term radiotoxicity (LTR) which is geochemistry-neutral. LTR incorporates decay and dose potential (once ingested) characteristics of the various isotopes; it does not incorporate transport from repository to potential receptors via the unsaturated (vadose zone) or saturated (aquifer) surrounding a repository as does the LTD metric.

Because of U.S. policy, we did not examine any cases where Pu and Np are separated from each other.

1.5. Key factors

Table 1-3 lists what we believe to be the six most important technical factors that will influence the five fuel cycle decisions; these were shown graphically in Figure 1-5.

Table 1-3a. Characteristics of Factor 1. Growth of Nuclear Energy

Case	Lower bound	Intermediate cases		Upper bound
Characteristic	No new reactors (nuclear phase out)	0% growth	1.8% growth	3.2% growth

Table 1-3b. Characteristics of Factor 2. Cost and Acceptance of Additional Repositories

Case	Most restrictive	Intermediate cases		Best case
Characteristic	Additional repositories precluded	Additional repositories acceptable, but costly	Only a 2nd repository is allowed for geographical balance	Low cost & high acceptance

Table 1-3c. Characteristics of Factor 3. Which Thermal Reactors Succeed in the Market Place?

Case	Status quo	Intermediate cases	Largest change
Characteristic	Only LWR succeeds	LWR dominates electricity market; VHTR dominates H2 market	VHTR displaces LWR

Table 1-3d. Characteristics of Factor 4. How much Uranium is Available? This has to be assessed both globally and domestically, i.e., how much of the world's uranium can be available to the US?

Case	Most restrictive	Intermediate cases	Best case
Characteristic	Conventional resources – pessimistic - 3.1 million tonnes U, [Herring2004, Steyn2003]	Conventional resources – optimistic - 16 million tonnes U [Herring2004, Steyn2003]	Unconventional resources (billions of tonnes of U) – [Herring2004]

Table 1-3e. Characteristics of Factor 5. What Proliferation Policies Exist?

Case	Status quo	Intermediate cases	Largest change
Characteristic	Keeping Np with Pu is adequate	Additional intrinsic protection required	Must meet 100 rem/hr criterion

Table 1-3f. Characteristics of Factor 6. How much Penalty is "Hot" Fuel Separation and Fabrication

Case	Most restrictive	Intermediate cases		Best case
Characteristic	Penalty from glovebox operation unacceptable (recycling thus precluded)	Penalty from glovebox operation ok, shielded operation unacceptable	Penalty from shielded operation acceptable, but cost an issue	Nil cost penalty from high gamma and neutron dose

It is possible to have a null-solution situation. For example, consider a combination of "additional repositories prohibited politically" (F2) with "recycle fuels must meet 100 rem/hr spent-fuel standard" (F5). None of the options currently receiving substantial attention in the AFCI program would be satisfactory; rather spiking of the recycle fuel with fission products would be required. Some combinations are technically incompatible under any circumstances, e.g., "recycle fuels must meet 100 rem/hr spent fuel standard" (F5) and "penalty from glovebox operation unacceptable" (F6). Thus, it behooves the government to try to increase option space by taking whatever actions are possible that push each factor in a way that increases options.

The primary way we developed these six factors was to analyze why various experts inside and outside the AFCI program had different opinions as to the "right thing" to do. Most have "the answer", but those

answers differ in large part because of differing opinions regarding one or more of these six decision factors.

For example, some believe that there will be little or no nuclear growth; others believe that there is a good chance for very high nuclear growth; this is factor F1. At present, each of these factors remains significantly uncertain. Some of those uncertainties can be reduced by AFCI actions (e.g. F6); some by DOE programs outside AFCI (e.g. DOE-NP2010 addresses F1, DOE-RW could address F2, and DOE-GenIV is addressing F3), but substantial components of all factors are outside DOE's control.

It is impractical to say – let's wait until we have perfect information before making any fuel cycle decisions. Instead, Chapter 2 assesses the robustness of the technical case for each decision at the present time.

In any case, reduction of uncertainties is desirable. It behooves the government to try to reduce uncertainties to the extent practical. So, Table 1-4 examines the six decision factors with regard to what AFCI, DOE in general, or other influences contribute to each decision factor.

Table 1-4. Characteristics of Key Decision Factors

	F1. Growth of nuclear energy?	F2. Cost and acceptance of additional repositories?	F3. Which thermal reactors succeed in the market place?	F4. How much uranium is available?	F5. What proliferation policies exist?	F6. How much penalty is "hot" fuel separation and fabrication?
Can AFCI actions influence each factor?	Yes – the more sustainable nuclear is, the most it is likely to grow	Yes – the more attractive recycling is, the less acceptable additional repositories may look	Yes – could influence LWR versus VHTR	No – AFCI cannot change how much uranium is there	Yes – AFCI can provide data on fuel content and form, provide options for managing inventory	Yes - AFCI can provide data on fuel reprocessing and fabrication content and form, provide cost estimates for options
Can DOE actions influence each factor?	Yes – the success of NP2010 would be evidence for moderate to high nuclear growth	Yes – DOE-RW could examine the issues associated with additional repositories	Yes – NP2010 and GenIV	Partly – DOE could reduce uncertainty by more analyses of uranium resource, could fund work on unconventional uranium resources.	Yes – increase dialog with other nations to explore options for reducing proliferation potential	Yes – advanced robotics could be a part of the maintenance strategy of GenIV reactors
Influences outside DOE	Economy, other energy sources	Growth of societal imperative to **reduce/reuse/recycle** will increase attractiveness of recycling and decrease attractiveness of additional repositories	H2 economy, other H2 sources	Other countries competing for U	The actions of groups and nations could influence our policies.	Robotics (the more robotic fabrication proceeds in other industries, the more likely the "hot" penalty is low for us)

2. ANALYSIS OF FIVE KEY FUEL CYCLE DECISIONS

This chapter both sets the context for the analyses in Chapters 5 through 8, but also uses their results. That is, the definition and order of decisions itself depends on what we know, what we don't know, and how uncertainties influence various potential decisions.

We have defined and ordered the decisions in a way that we believe represents decreasing readiness to make each decision, and therefore a logical chronological order. In particular, Table 2-1 shows which decision factors would have primary/secondary impact on each postulated decision, as well as which decisions would themselves inform decision factors.

Table 2-1. Influences between Decision Factors and Decisions

<table>
<tr><th>Factors

Decisions</th><th>F1. Growth of nuclear energy?</th><th>F2. Cost and acceptance of additional repositories?</th><th>F3. Which reactors succeed in the market place?</th><th>F4. How much uranium is available?</th><th>F5. What proliferation policies exist?</th><th>F6. How much penalty is "hot" fuel separation and fabrication?</th></tr>
<tr><td>D1. Open 1st geologic repository</td><td>Informed By decision</td><td>Informed By decision</td><td></td><td></td><td></td><td></td></tr>
<tr><td>D2. Determine credibility of recycling</td><td rowspan="3">Impacts both ways (the more attractive recycling, the more growth; and vice versa)</td><td rowspan="3">Impacts both ways (the more attractive repositories, the less need for recycling, and vice versa)</td><td rowspan="3">Impacts both ways for reactors that cannot recycle (if recycle is good and a reactor can't recycle, that reactor would lose support)</td><td></td><td>Informed By decision</td><td>Informed By decision</td></tr>
<tr><td>D3. Build 1st separation plant for UOX</td><td></td><td colspan="2" rowspan="2">Primary impact, could determine which elements are separated and hence fuel composition</td></tr>
<tr><td>D4. Build 1st recycle-fuel fabrication plant</td><td>Primary impact, couples with F4.</td><td>Primary impact, couples with F1</td></tr>
<tr><td>D5. Build future separation and fuel fabrication plants</td><td>Primary impact</td><td></td><td>Primary impact</td><td>Primary impact</td><td>Primary impact</td><td>Primary impact</td></tr>
</table>

The rest of this Chapter examines these five postulated decisions to determine uncertainties, which options appear the most robust, key unknowns, etc.

2.1. Open 1st geologic repository

All scenarios (even nuclear phase out) require at least one geologic repository. Therefore, the U.S. needs a geologic repository with these minimum characteristics.

- Minimum of 70,000 tonne of spent UOX. This is the current legal limit. In the absence of recycling, it would be inadequate. With multi-pass recycling, the current 10% allocation for DOE/DOE waste could be increased while also accommodating nuclear energy growth this century.
- Minimum of 50 years monitored retrievable operation per U.S. law.

This is settled US policy, but the decision has not been implemented. Once it is implemented, it will strengthen the foundation for AFCI waste management assessments. Table 2-2 is a decision table for opening the first geologic repository.

Table 2-2. Decision Table for "Open 1st Geologic Repository"

Decision table	Repository now, i.e., at YMP site	Repository later, site may not be YMP	No repository
Permanent solution to waste problem if once-thru	Yes	Yes, but delayed	NO
Permanent solution to waste problem if recycle	Yes, we see nothing in YMP design that precludes recycle option	Yes, but delayed	NO
Appropriate if major problem found with YMP	Only if the problem can be fixed	Yes	NO
Avoid need for other facilities to meet AFCI objectives, e.g., reduce at-reactor inventories	Yes	NO, centralized retrievable storage would be required	NO
Assess robustness against F1. Growth of Nuclear Power			
Appropriate for Nuclear phase out	Yes, but capacity would have to increase	Yes, but delayed	NO
Appropriate for 0% growth	Yes, but additional repositories required if most SNF is not recycled see Table 1-2	Yes, but the repositories would have to be large to avoid recycling, see Table 1-2	NO
Appropriate for 1.8% growth			NO
Appropriate for 3.2% growth			NO

Table 2-2 illustrates that "no repository" is never preferred. "Repository later" is only preferred if a major non-fixable problem is found with the YMP site. The table supports the proposition that the YMP site should go forward as expeditiously as possible.

From the AFCI perspective, the most important statement in the table is perhaps "We see nothing in the YMP design that precludes recycle option." Were this not true, then the possibility of recycling could be used to as an argument to delay proceeding with YMP. Hence, this statement requires more discussion here.

Implementing recycling could have four potential major impacts if we wish to avoid a second repository this century:

- If unrecycled used fuel fills up the repository then either it must be retrieved or filling must stop in time to reserve space from the HLW that would result from recycling. This is discussed below.
- The draft tunnel design (radius, spacing) may not provide optimum heat-management for recycle-HLW. This should be analyzed, but the mountain would be flexible for many years after licensing

and start of operations (assumed no earlier than 2012). There is no reason to believe that this is infeasible.

- The waste package and titanium drip shields may be over-designed if recycle-HLW waste forms are adequately superior to unrecycled UOX as a waste form. Since presumably the waste package, Ti drip shields (and drift tunnels) would be "build as you go," they could be redesigned if/when recycle-HLW waste forms became known. There is no reason to believe that this is infeasible.
- Long-term dose and radiotoxicity, see section 5.3.

Table 2-3 explores the retrievability issue. How much of the SNF must be recycled to "fit" the waste into the repository? How much of the repository must be used to store processed versus unprocessed fuel? Table 2-4 then looks at when filling of the repository must stop if waste is not to be retrieved. The repository section of Appendix B contains the assumed YMP fill rate, starting in 2012.

Table 2-3. How much of a geologic repository must hold processed waste, how much fuel must be processed?

Nuclear Futures		Legislative Limit	Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share Generation	Growing Market Share Generation
Nuclear growth rate		Nuclear phase out			0%/year	1.8%/year	3.2%/year
Cumulative discharged fuel in 2100 (tonne iHM)		63,000	100,000	120,000	250,000	600,000	1,500,000
Fuel Management Approach	YMP Capacity	**What % of repository must hold processed waste? What % of fuel generated by 2100 must be processed?**					
No recycling	70,000	Capacity sufficient					
	119,000		Capacity sufficient				
Single-pass thermal recycle (heat capacity improvement of 1.7	70,000		84% of YMP 90% by 2100				
	119,000		Processing not needed	10% of YMP 16% by 2100			
Thermal/fast symbiotic recycle (heat capacity factor improvement of 50x)	70,000		1% of YMP 38% by 2100	2% of YMP 48% by 2100	6% of YMP 76% by 2100	17% of YMP 91% by 2100	47% of YMP 98% by 2100
	119,000		Processing not needed	<1% of YMP 7% by 2100	3% of YMP 56% by 2100	9% of YMP 83% by 2100	25% of YMP 94% by 2100

Table 2-4. When stop filling YMP if non-retrievable, i.e., if we don't want to have to retrieve waste.

Nuclear Futures		Legislative Limit	Existing License Completion	Extended License Completion	Continuing Level Energy Generation	Continuing Market Share Generation	Growing Market Share Generation
Nuclear growth rate		Nuclear phase out			0%/year	1.8%/year	3.2%/year
Cumulative discharged fuel in 2100 (tonne iHM)		63,000	100,000	120,000	250,000	600,000	1,500,000
Fuel Management Approach	YMP Capacity	**When would filling of the repository have to stop if retrieval is to be avoided, assuming start in 2012?**					
No recycling	70,000	Capacity sufficient					
	119,000		Capacity sufficient				
Single-pass thermal recycle (heat capacity improvement of 1.7x)	70,000		Stop in 2019				
	119,000		Processing not needed	Stop in 2048			
Thermal/fast symbiotic recycle (heat capacity factor improvement of 50x)	70,000		Stop in 2035	Stop in 2035	Stop in 2034	Stop in 2032	Stop in 2027
	119,000		Processing not needed	Stop in 2052, i.e., just before repository filled	Stop in 2051	Stop in 2049	Stop in 2044

By definition, if discharged SNF is limited to 63,000 tonne, then a repository capacity of 70,000 (or 119,000) is adequate. In this case, 100% of YMP can hold unprocessed waste and 0% of the fuel must be recycled. This situation (shaded green in Table 2-3 and Table 2-4) is also true if the repository capacity is 119,000 and the accumulated fuel is 100,000.

These tables lead to several conclusions:

- The higher the heat capacity improvement factor, the less of YMP that needs to be reserved for HLW from recycling, therefore we can either fill longer or retrieve less.
- With an improvement factor of 1.7x (illustrative for single-pass recycling), additional repositories are likely to be required for any scenario involving continuation of nuclear power.
- With an improvement factor of 50, at least half of YMP could be filled with unprocessed waste – either commercial spent fuel or DOE/DOD waste.
- With high improvement factors, more of YMP could be allocated to DOE/DOD waste. That allocation is currently 10%, which is thought to be inadequate for the magnitude of cleanup wastes. If that allocation could be raised to 20 or 30%, the cleanup wastes could be dispositioned; this would also shift more of the cost of YMP from commercial utilities to DOE – a cost savings from recycling.
- The sooner we know the recycle target we will aim for, the clearer the way to use the YMP resource.

2.2. Determine credibility of recycling

US needs to establish the credibility of one or both ways to handle several times the waste legally allowable in the first repository, as was made clear in Table 1-2. There are two options: multi-repositories and multi-pass recycling.

Establishing the credibility of multi-repositories requires legal permission to look for additional sites, the political willingness to look for additional sites, and some rational for believing that additional repositories would be less expensive than the first. In most industries, indeed, there is a learning curve – subsequent products and facilities are less expensive. However, in non-nuclear waste management, e.g., landfills, this has not proven true. The lead author has worked with landfill experts on a previous project and they all report a negative learning curve with regard to siting new landfills – tougher and more expensive. The positive finding is that they are generally able to put more waste into existing landfills (by building up! – literally vertical) and to reduce the flux of waste to the landfill by encouraging recycling. These trends – difficult to site new repositories, reduce waste flux by recycling, and expanding the repository we have – are likely true for HLW repositories, but of course this is unproven.

The credibility of multi-pass recycling requires a legal basis for YMP capacity of residual HLW (i.e. not "initial heavy metal") and one or more technological options at advanced stages of technology readiness. So, which options make sense?

The short answer is that **multi-pass options should be pursued, single-pass options (that cannot transition to multi-pass) should not**. As shown in Chapters 5 and 8, the best single-pass option is IMF-NpPu implemented with minimum delay between reactor discharge and the single-pass recycle:

- LTH improvement of 2.0x, versus goals of 10-50x.
- LTD improvement of 2.1x, versus goals of 10-50x
- LTR improvement of 2.5x, versus goal of 100x
- Uranium utilization improvement of 1.15, meeting the near-term goal of 15% improvement.

Multi-pass options do exist to meet the objectives

- LTH improvement of ~100x, versus goals of 10-50x – provided that Cs and Sr are managed separately
- LTD improvement of ~100x, versus goals of 10-50x – provided that Tc and I are managed separately, in special waste forms, in special repositories, or transmuted
- LTR improvement of at least 100x, versus goal of 100x – limited by TRU loss rates
- Uranium utilization improvement of ~100x, versus goal of 50x.
- A combination of IMF and CFR can manage the inventory and quality of Pu for proliferation resistance purposes.

For current purposes, we assume that establishing the credibility of recycling would require a dedicated laboratory where separation and fabrication of multi-pass materials would be developed and demonstrated. Table 2-5 provides the decision tree for determining the credibility of recycling. The first option is to recycle immediately, presumably with the narrow range of options that exist today, specifically, MOX with Pu, using the PUREX process. We have not specifically analyzed multi-pass MOX-Pu, but the waste management results would be poor based on past and current (Chapter 5) analyses because only Pu would be burned. Since LTH, LTD, and LTR are all dominated by Np and Am, it is impossible to meet AFCI objectives with multi-pass MOX-Pu. Therefore, even if MOX-Pu were started "now", work would still be required on advanced recycling methods that would later be implemented to meet AFCI objectives.

Table 2-5. Decision Table for "D2. Determine Credibility of Recycling"

Decision table	Recycle now, i.e., commit to relatively narrow range of technical options (i.e. merge D2 and D3)	AFCL, decide on recycling later (i.e., keep D2 and D3 separate)	No AFCL, no recycle option
Assess robustness against F1. Growth of Nuclear Power			
Avoid second repository under nuclear phase out	Yes, but cost of AFCL would have been wasted, recycling wouldn't have been necessary	Yes	Yes
Avoid second repository under no growth	No, unless MOX-Pu is later changed to better recycle options	Yes, ideal	NO
Avoid second repository under low growth rates			
Avoid second repository under high growth rates		Yes, but SNF inventory continues to grow in the meantime	NO
Assess robustness against F2. Cost and acceptance of additional repositories			
We have maximum knowledge on how to improve recycle options in case multi-repositories are competitive	NO	Yes, ideal	NO

In the hypothetical dedicated facility, what separation targets would be warranted? Table 2-6 shows current program objectives and suggestions from this work.

Table 2-6. Suggested Separation Targets

	Recovery fraction (each recycle)		Product purity (each recycle)	
	Current goal [Vandergrift2004]	Differences identified in this study	Current goal [Vandergrift2004]	Differences identified in this study
Uranium – potential disposal	>90%	90% recovery is barely adequate to reduce LTD by 10x, would have to be 98% to reduce LTD by 50x.	Meet 10CFR61, e.g., <100 nCi-TRU/gram-U, requiring decontamination factor from Pu of > 10^5	
Uranium – potential recycle	Not specified	Not adequate for IMF options. Even 99% recovery would only make the U and Pu from UOX-51 comparable (0.17 and 0.22 tonnes/yr per GWe)	"If uranium is destined for recycle in reactor fuel, its purity requirements are greater and would be governed by ASTM C 877-98."	Not analyzed
NpPu	>99%	Possibly not adequate for MOX/CFR symbiosis, goal of 99.5% in DOE2005a appears adequate.	"The purity of this product stream is required to meet mixed-oxide (MOX) fuel specifications as described in ASTM C833-01."	
Am	>99.5% to provide 100x decrease in LTH	Appears adequate	"Based on fast reactor recycle of all TRU, the lanthanide content of the Am/Cm product must be <20 mg/g uranium plus TRU."	
Cm	>99.5% to provide 100x decrease in LTH			
Tc and I	>95% to provide 20x decrease in LTD	Recommend 98% to allow LTD reduction of 50x, comparable to LTH reduction.	"If transmutation of Tc is the chosen option, the Tc product must contain less than 16 μg of fissile actinides per g of Tc." 4 μg for I.	Not analyzed in this report, but we do not see the basis for this
Cs and Sr	"97% recovery required for Cs and Sr to make their recovery equal to that of all other fission products."	Recommend 99% to accommodate repeated recycling while maintaining high LTH benefits	Meet 10CFR61, e.g., <100 nCi-TRU/gram of Cs-Sr product	

2.3. Build 1st separation plant for UOX

Subject to (a) AFCL results and (b) nuclear growth, the US will need a 1st separation plant with these characteristics.

- If early nuclear phase out, would not commit to this facility because it would not be needed to stay within 1 repository. If evidence that multiple repositories are acceptable/cost effective, might not commit to this facility either.
- Purpose/role – process spent UOX. This is robust EVEN if LWRs appear to be displaced by VHTR or fast reactors. The separations plant would be useful for the large LWR spent fuel stockpile; and it may be possibly to develop a head-end process to allow introduction of alternate spent fuel forms. One would also want to be able to recycle a small amount of MOX and/or IMF, e.g. SFTF requirement was to be able to handle 2% MOX in the input stream.
- Capacity considerations: economics of scale, not committing to more than 30-yr or 60-yr facility could process relative to the amount of UOX expected. However, even conservative estimates suggest that the 1st plant could be as large as 5000 tonnes/yr with little risk of over-building relative to the amount of UOX expected.
    - Over ¾ of the fuel in all scenarios except BFR is UOX. Even at no-growth, by 2025 the amount of accumulated used UOX will be 96,000 tonnes. If no-growth continues to 2055 (hypothetical end of a 30-year first separation plant), 160,000 tonnes will have accumulated. The 1st plant with 30-yr lifetime would have to process 5300 tonnes/yr to eliminate this amount. If the 1st plant ultimately has 60-yr lifetime, it would have to process 3700 tonnes/yr to eliminate the 220,000 tonnes accumulated by 2085. These numbers increase with growth. If used, BFR would probably not decrease the UOX accumulation rate because of the finite time for BFR introduction against a backdrop of growth. (If there is no nuclear growth, it is difficult to imagine introducing BFR.)
    - DYMOND calculations in Chapter 7 all use 3,000 tonnes/yr for the 1st plant in 2025, followed by a 2nd plant of 3,000 tonnes/year in 2040. Only at 2040 does the rate of separation exceed the accumulation rate of used fuel at the nominal growth rate of 1.8%/year.
- Aqueous processing
- U separation – do not preclude future consumption of U, do not preclude 10CFR61-disposal.
- NpPu/Am/Cm should not be precluded at this time, as there are attractive scenarios that would require this separation approach. See Table 2-7.
    - Am recycle is required to meet LTD, LTH, and LTR program objectives; may be required for proliferation.
    - Cm recycle is not required to meet any of the program objectives. Therefore, whether to include Am with Cm requires a tradeoff on (a) cost of Am vs. Cm separation vs. (b) cost of including Cm in the recycled fuel (more complex chemistry, hotter handling)
    - Calculations show that IMF-NpPuAm and IMF-NpPu with Am targets have similar transmutation performance.[Goldmann2005] IMF-NpPuAm may have proliferation advantages, IMF-NpPu withAm targets would appear to have economic advantages (less remote handling). More work is therefore required to select.
- Recover Tc and I as separate products, which could be transmuted, put into special waste forms, or sent to a separate specialized repository later. At the present time, all three options should be retained. Analyses in Chapter 5 indicate that Tc and I removal may not be required to meet the 10x LTD reduction objective, but would be required to make LTD reduction match LTH reduction so that there would be no net increase in dose from the repository as a result of recycling. See Table 2-8.
- Recover Cs and Sr as a new class of Greater than Class C (GTCC) waste, could be stored at-grade or below-grade or in YMP. Waste form needs to accommodate any of these options. See Table 2-9.

Table 2-7. Decision Table for Transuranics for "Build 1st Separation Plant for UOX"

Decision table	NpPu/Am/Cm	NpPu/AmCm	NpPuAm/Cm	NpPuAmCm
Provides additional way to adjust k-eff of fuel	Yes, can vary Pu to Am ratio	Yes, can vary Pu to Am ratio	NO	NO
Assess robustness against F1. Nuclear Growth?				
Not relevant				
Assess robustness against F2. Cost and acceptance of additional repositories?				
Separation option would support sending Am to a hypothetical repository tailored to heat-generating isotopes	Yes	Only if hypothetical repository could also handle Cm	NO	NO
Assess robustness against F3. Which TR success in the market place?				
Supports possibility that non-LWR fuel can include NpPu but not Am (because of difficulties of incorporating high-vapor-pressure Am)	Yes	Yes	NO	NO
Supports possibility that non-LWR fuel could incorporate NpPuAm, but not Cm (because of difficulties of incorporating high-heat Cm	Yes	Yes, but Am would not be able to be burned without additional, later separation	Yes, ideal	NO
Assess robustness against F4. How much uranium?				
Not relevant				
Assess robustness against F5. What proliferation policies?				
Supports transmutation of Am with Pu (possible proliferation advantage)	OK, but possibly a wasteful separation; NpPuAm/Cm or NpPuAmCm might have been cheaper	Yes, but Cm would also be in the recycled fuel	Yes, ideal because no need to include Cm in the recycled fuel and because Am would always be with the Pu	Yes, but Cm would also be in the recycled fuel (highest cost but also highest intrinsic proliferation benefit)
Supports transmutation of Cm with Pu (possible proliferation advantage)		Yes	Yes, but the Am vs. Cm would have been unnecessary	
Assess robustness against F6. "Hot" fuel sep/fabrication penalty?				
Supports transmutation of Am in separate targets (possible economic advantage)	Yes, ideal	Yes, but Cm would be in the target	NO	NO

Table 2-8. Decision Table for Long-Lived Fission Products for "Build 1st Separation Plant for UOX"

Decision table	Put Tc and I in HLW	Put Tc and I in separate storage	Make Tc and I into targets
Supports possibility that transmutation of Tc and I needed to reduce YMP peak dose	NO	Yes	Yes
Supports possibility that transmutation of Tc and I not needed	Yes	OK, but the effort to separate Tc and I and put into their own waste/storage forms would have been wasted	NO
Supports possibility that Tc or I have some non-waste value	NO	Yes	OK, but the effort to have made the Tc/I into targets would be wasted and would have to be undone
Assess robustness against F1. Nuclear Growth?			
Not relevant			
Assess robustness against F2. Cost and acceptance of additional repositories?			
Supports possibility of a separate repository for Tc and/or I to take advantage of their geochemistry (if additional specialized repositories were acceptable)	NO	Yes	OK, but the effort to have made the Tc/I into targets would be wasted and would have to be undone if the target form was not suitable as a waste form
Assess robustness against F3. Which TR success in the market place?			
Not relevant			
Assess robustness against F4. How much uranium?			
Not relevant			
Assess robustness against F5. What proliferation policies?			
Not relevant			
Assess robustness against F6."Hot" sep/fabrication penalty?			
Not relevant			

Table 2-9. Decision Table for Short-Lived Fission Products for "Build 1st Separation Plant for UOX"

Decision Table	Put Cs and Sr into HLW	Put Cs and Sr into non-10CFR61 waste form	Put Cs and Sr into 10CFR61-type waste form	Put Cs and Sr into stable storage, pick waste form later
Supports possibility that Cs and Sr could have economic value	NO	Yes, but probably wasted the effort to make the waste form		Yes
Supports possibility to put Cs and Sr into special area of YMP	NO	Yes	Yes	Yes
Supports possibility to put Cs and Sr into at-grade storage that converts to 10CFR61 facility	NO	NO	Yes	Yes
Assess robustness against F1. Nuclear Growth?				
Not relevant				
Assess robustness against F2. Cost and acceptance of additional repositories?				
Supports possibility of a repository tailored for Cs and Sr	NO	Might be wrong waste form	Might be wrong waste form	YES
Assess robustness against F3. Which TR success in the market place?				
Not relevant				
Assess robustness against F4. How much uranium?				
Not relevant				
Assess robustness against F5. What proliferation policies?				
Supports possibility of having to spike recycle fuel with Cs and Sr	NO	Yes, but effort to make the waste form would have been wasted		Yes
Assess robustness against F6."Hot" sep/fabrication penalty?				
Not relevant because the baseline is to separate Cs-Sr from the recycled TRU.				

Note that at the time of separation, separated Cs and Sr would not quality for near-surface disposal per 10CFR61.

If the current legal definition of High Level Waste (HLW) is construed to apply to Cs and Sr, it would indeed be HLW, regardless of its characteristics. If so, the only legal disposal option would be a geologic repository. To preserve LTH benefits, the Cs and Sr would be placed in separate drift tunnels designed for that purpose, but without the need for millennia-long performance.

If Cs and Sr are not construed to be HLW by law, they become Greater than Class C (GTCC) waste. The Nuclear Regulatory Commission (NRC) has, for now, said that the only disposal option is geologic repository, but that other options could be considered. This provides the opportunity to make the argument that at-grade storage of Cs and Sr is an option; after 100-300 years, the Cs and Sr will have cooled sufficiently to then quality under 10CFR61 (if the regulations haven't changed by then).

There is one additional uncertainty regarding the status of Cs and Sr, namely long-lived Cs-135 (3e6 yr). It is not addressed in 10CFR61 so no definitive statement is possible. Two prior studies shed light on this.

First, the 10CFR61 methodology was applied to Cs-135 [Fetter1990] with the result that even pure Cs-135 would not be a problem.

Second, the original NRC technical report [NRC1981] did calculate a limit for Cs-135, 84 μCi/cc, in contrast to the limit on Cs-137, 460 μCi/cc. In the final 10CFR61 [NRC1982], there was a factor of 10 credit for Cs-137 to account for the waste being in activated waste form, raising the Cs-137 limit to 4600 μCi/cc. The study for the Spent Fuel Treatment Facility (SFTF) concluded that therefore the equivalent NRC limit for Cs-135 would be 800 μCi/cc, 10 times 84 μCi/cc. With the waste forms they were considering, they estimated the concentration from SFTF Cs waste to be 156 μCi/cc, which is lower than the hypothetical limit of 800 by a margin of 5. Although they were studying Cs from used UOX, the Cs flux from all of the fuels is similar, so the result should hold for other fuels.

Note that the same SFTF study estimated the Cs-137 concentration in waste to be 5.26e6 μCi/cc, or a factor of 1140 above the 10CFR61 limit. This requires 300 years of decay of Cs-137.

There is one other consideration in planning the 1st separation plant, the possibility that there may be other isotopes economically recoverable. There are at least 95 isotopes currently in use in various industries.[Waltar2004a] Table 2-10 lists six of those isotopes that are relevant to AFCI.

Table 2-10. AFCI-Relevant Isotopes Currently in Use (of 95 listed by Waltar2004a)

Isotope	Half life	Applications per Waltar	Potential to be a significant product?
Am-241	433 years	Smoke detectors; combines with beryllium to produce neutrons for material inspections (aircraft, [airport luggage screening] and others), oil well borehole analysis, soil density measurements, and coal ash measurements; used with targets (e.g. copper or silver) to generate pure fluorescent x-ray sources.	Would be contaminated with other Am isotopes, precluding many/most of current applications
Cf-252	2.65 years	Brain cancer therapy. Spontaneous fission neutrons for conducting materials radiography in aircraft and other crucial structures; soil density measurements.	Is there sufficient yield to matter? Would be contaminated with Cf251 (900 years).
Cs-137	30.1 years	Blood irradiation to make organ transplants successful. Determine ash content of coal entering a power plant (with Am-241); determine soil density; radiation monitoring calibrations. Determine soil erosion or sedimentation patterns in reservoirs or estuaries.	Would be contaminated with Cs-135, would probably not preclude current applications.
Kr-85	10.8 years	Thickness gauges; airport runway lights; methane gas tracer	Issue is cost effectiveness of capture of this noble gas, versus other noble gas isotopes.
Sr-90	28.8 years	With aluminum, produces x-rays to trace water or methane around petroleum fuels; thickness gauges; small power sources.	Other Sr isotopes decay away
Y-90	2.67 days	Cancer therapy and as labeled monoclonal antibodies for treating Hodgkins disease, non-Hodgkins lymphoma, breast, liver, and colon cancer; as microspheres for treating primary liver cancer, treating benign diseases (rheumatoid arthritis); treating restenosis of clogged arteries in heart disease.	Start with Sr90 and separate its daughter Y90

2.4. Build 1st recycle fuel fabrication plant

We noted above that none of the single-pass options come close to meeting AFCI objectives, we therefore limit the discussion on the hypothetical 1st recycle-fuel fabrication plant to those options that could move toward the objectives.

- IMF in a form that is recyclable, enabling one or more recycles in TR, possibly eventually transitioning to fast reactors.
- MOX in a form that is recyclable, enabling one or more recycles in thermal reactors, possibly eventually transitioning to fast reactor.
- CFR fuel, i.e., skip recycling in thermal reactors, go directly to CFR.
- BFR fuel, i.e., skip recycling in thermal reactors, go directly to BFR.

Much of the rest of this report compares those options. Here, we summarize the comparison and our conclusions in three ways. First, Table 2-11 provides a decision table. Second, Table 2-12 is a "regret analysis", what things might make us wish later that a given option had not been selected. Third, we suggest a decision tree.

Table 2-11. Decision Table for Multi-Pass Fuel Options for "Build 1st Recycle Fuel Fabrication Plant"

Decision table	Recyclable IMF	Recyclable MOX	CFR fuel	BFR fuel
Minimize technical risk	Loses because multi-pass IMF is very exploratory	Favored	Depends on the type of fast reactor and its fuel	Depends on the type of fast reactor and its fuel
Disposal of U becomes more of an issue (e.g. YMP U doses)	NO	Slightly favored, but only if we use burned U in the MOX instead of fresh U. Even then, ~90% of the burned uranium is still discarded.	Slightly favored, 70-80% of the uranium is still discarded.	YES – favored because all uranium is usable
Relatively high cost of separating and fabricating recycled material	Yes, lowest throughput of options studied	NO, option discouraged	NO, option discouraged	Yes, high throughput but relatively "clean" material
Assess robustness against F1. Nuclear Growth?				
Appropriate for nuclear phase out and relative low growth	Appropriate for nuclear phase out, relatively low growth	Appropriate for relatively high assurance of substantial, sustained nuclear growth because little short-term benefit (LT heat, LT dose, proliferation metrics) – consistent with French assumptions	Similar to IMF	NO
Appropriate for high growth	Adaptable to high growth by throttling back on IMF percentage. Full-core IMF to BFR allows about half the rate of BFR introduction as UOX to BFR.		Can adjust conversion ratio accordingly.	YES
Assess robustness against F2. Cost and acceptance of additional repositories?				
If repositories are readily accepted and cost effective	Were cost and acceptance of many repositories not an issue, the incentive for IMF, MOX, or CFR recycling would substantially decrease.			YES, still important because of uranium usage

Assess robustness against F3. Which TR success in the market place?				
Supports possibility that LWR continues to beat all competition	Yes	Yes	CFR component of fleet would be 19% if matched with IMF, higher if matched with UOX or MOX.	Not unless/until uranium forces BFR adoption
Supports possibility that VHTR enters the market either for hydrogen or for electricity.	Depends on finding analogs of these fuels for VHTR.			Not unless/until uranium forces BFR adoption
Assess robustness against F4. How much uranium?				
Supports possibility that U not a constraint for several centuries	Yes	Yes	Yes	NO
Supports possibility that U is a near-term constraint	Relatively poor, IMF is too successful in burning Pu	Yes	Yes, natural lead into BFR	Yes
Assess robustness against F5. What proliferation policies?				
Supports possibility that WU inventory must be reduced soon	Yes, Pu inventory can be "frozen" once multi-pass IMF adopted.	NO	Not quite as good as multi-pass IMF.	NO
Assess robustness against F6."Hot" sep/fabrication penalty?				
Supports possibility that remote fabrication uneconomic, i.e., only Pu or NpPu recycled	Possibly, using the blended core concept with separate Am targets	No repository benefits, little help with proliferation metrics, hence the only reason to adopt these options would be the modest savings in uranium ore required (17% for multi-pass MOX, 30-50% for CFR)		No repository benefits, but would address uranium ore limits

Some people prefer to look at these types of decisions from the standpoint of "regret" – why might I wish I had not made such a choice? Table 2-12 provides a "regret analysis" comparing the four options.

Table 2-12. Regret Analysis on Possible Recycle Fuel Options – What Could Go Wrong and What Could We Do About it?

Decision table	Recyclable IMF	Recyclable MOX	CFR fuel	BFR fuel
Regrets related to waste management (1)		MOX accrues benefits slowly compared to all other options. If this happens, the only fix would be to change to another fuel type.		
Regrets related to waste management (2)	Uranium disposal could become more of an issue, in which case all of these options suffer because most of the uranium is discarded. The only fix would be to shift to BFR, the only option that uses all types of "waste" uranium.			None
Regrets related to proliferation resistance		MOX does little to reduce Pu. The fix would be to shift to IMF.		Excess Pu accumulated, fix would be to shift to CFR or IMF.
Regrets related to uranium energy recovery	IMF is intended to destroy Pu at the maximum rate by excluding uranium and eliminating in-core Pu production. Therefore if taken too far it becomes more difficult to shift to BFR than had MOX been used.	MOX is not a breeder; the fix would be to shift to BFR.		None
Regrets related to economics and safety	IMF could be found to have flaws. Fixes could be to stop recycling, shift to MOX, or shift to FR.	MOX could be found to have flaws. Fixes could be to stop recycling, shift to IMF, or shift to FR.	FR not proven, could be viewed as a mistake. CFR option would require 27% CFR fraction in fleet (17% if IMF-CFR), meaning that significant fraction of the fleet could be at risk.	FR not proven, could be viewed as a mistake (indeed Superphenix has not led to a series of FR). BFR option requires most/all of the fleet to be BFR.

Figure 2-1 illustrates the concept of a decision tree. The example is used to assess the risk of a heart attack at hospitals. Easy to use. Simple in concept. Built on a huge number of extensive studies. Figure 2-2 is our first attempt at a decision tree for selecting among recycle fuels. Relatively simple. Built on many analyses – both in and out of this report.

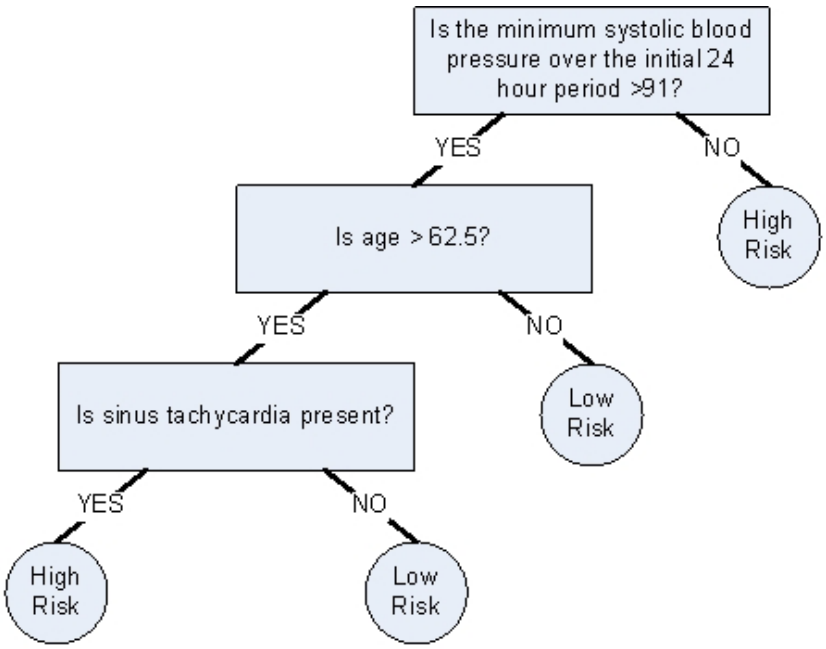

Figure 2-1. Sample decision tree for classifying incoming heat attack patients as high or low risk, easy to apply, but built on a large number of extensive studies.[Gigerenzer1999]

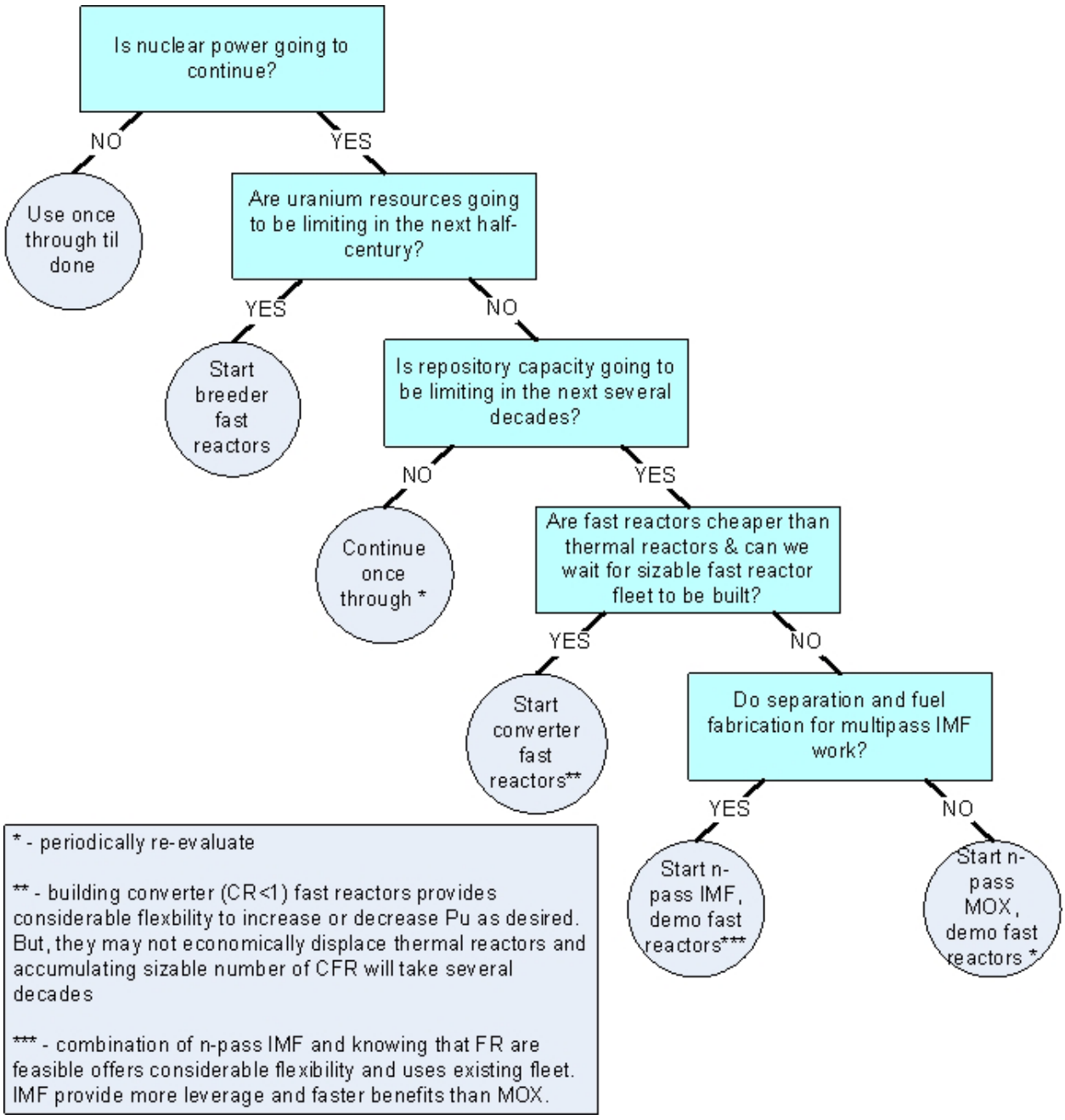

Figure 2-2. Decision tree for selecting among recycle-fuel strategies

The concept of this type of decision tree is to make a series of binary (yes/no) decisions that led to making a selection among options. As noted in Chapter 1, the first issue in this sense is whether or not nuclear is going to continue, if not, recycling does not make sense. If so, perhaps the most important question that would make the other questions moot is whether uranium resources are going to be limiting in perhaps a half century. If so, that dominates and BFR should be the preferred path. If not, there are a series of questions that lead to CFR, n-pass IMF, n-pass MOX, or keeping once-through.

Note that with either multi-pass IMF or multi-pass MOX, it is prudent to keep a FR component. As explained in later Chapters, by themselves, TR cannot meet AFCI objectives. It appears that the

strategies of multi-pass IMF and multi-pass MOX can indeed be continued indefinitely (until uranium resources become a problem). But, this would be accomplished by eventually throwing away unburned TRU that had accumulated. When that happens, the repository benefit will not meet AFCI objectives. This probably can be deferred until the next century, but it will happen eventually. Adding CFR to IMF or MOX means that accumulated TRU never have to be thrown away.

2.5. Build future separation and fuel fabrication plants

This decision must be faced only after building the first separation and fuel fabrication plants. The AFCI timeline calls for the first separation plant in 2025, with fuel fabrication shortly thereafter. The decision on subsequent plants would presumably be made so that FR are (or are not) available in 2040.

The basic options for subsequent plants would be as follows:
1. Drop nuclear
2. Keep nuclear, but drop recycling
3. Keep recycling, but change approach.

The ramifications of such choices are in Chapter 6 and 7. The main things that could go wrong, and what one could do about them, were already addressed in the regret analysis in Table 2-11.

The major point to make here is that the inertia and cost of changing reactor types, e.g. TR to/from FR exceeds the inertia and cost of changing fuels, e.g. UOX to/from IMF to/from MOX.

3. AFCI OBJECTIVES AND METRICS

The AFCI program objectives are now documented in a recent report to Congress.[DOE2005a] This Chapter starts with those objectives (Table 3-1) as the motivation for the metrics in this study. The most important metrics and targets are in Table 3-2; the full set of metrics analyzed are listed in Table 3-3. The rest of this Chapter motivates and explains individual metrics. In Table 1, "short-term" refers to the period through 2025, when the AFCI program recommends the need for a commercially-deployed spent fuel treatment facility. "Intermediate-term" refers to the period from 2025 until the commercial availability of Generation IV fast spectrum reactors, projected to be about 2040. "Long-term" refers to the time after several of these fast reactors have been built.

Table 3-1. AFCI Objectives from Report to Congress [DOE2005a]

Objective 1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.
In the short-term, develop and demonstrate fuel cycle technologies and facilities that remove more than 99.5 percent of transuranics from waste destined for geologic disposal and initiate their recycle in existing reactors.
In the short-term, improve management of the primary heat-producing fission products in spent fuel (cesium and strontium) to reduce geologic repository impacts.
In the intermediate- and long-terms, enable repeated recycling to reduce disposed transuranics by a factor of more than 100, delaying the need for additional geologic repositories for a century or more, even with growing energy production.
In the intermediate- and long-terms, reduce the long-lived radiation dose sources by a factor of 10 and radiotoxicity by a factor of 100, simplifying the design of a waste isolation system.
Objective 2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for spent fuel management.
In the short-term, develop fuel cycle technologies that enhance the use of intrinsic proliferation barriers.
In the short-term, demonstrate the capability to eliminate more than 99.5 percent of transuranic weapons-usable materials from waste streams destined for direct disposal by destroying these materials through recycling.
In the long-term, stabilize the inventory of weapons-usable material in storage by consuming it for sustained energy production.
Objective 3. Enhance energy security by extracting energy recoverable in spent fuel and recycled material, ensuring that uranium resources do not become a limiting resource for nuclear power.
In the short-term, develop the technologies needed to extend nuclear fuel supplies by up to 15 percent by recycling the fissile material in spent nuclear fuel.
In the long-term, extend nuclear fuel resources more than 50-fold by recycling uranium in spent fuel and depleted uranium, thereby converting current wastes into energy assets.
Objective 4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.
At all times, ensure that advanced fuel cycle technologies cause no significant decrease in the economic competitiveness of nuclear electricity.
At all times, maintain excellent safety performance of nuclear fuel cycle facilities and operations.
For the long-term, improve spent fuel management to reduce on-site storage at nuclear power plants.

Table 3-2. Most important metrics in this study

Metric	Targets (see Chapter 3)
Long-term heat (LTH) improvement	10x to 200x to achieve actual repository-heat improvements of 10-50x. As explained below, the LTH metric can overpredict the actual heat-based repository improvements calculated by Wigeland.
Long-term dose (LTD) improvement	10-50x
Long-term radiotoxicity (LTR) improvement	100x
Uranium ore use improvement	1.15 short term 50x long term
Pu239 equivalent tonnes/yr per GWe for fresh fuel	As low as possible
Pu239/Pu-total in fresh fuel	As low as possible (the value for discharged UOX-51 is 53%)
Avoid fully remote fuel fabrication	For as much fuel as possible
Minimize throughput of TRU (tonnes/yr per GWe)	As low as possible to minimize safety and economic issues
Percent fuel that are new	As low as possible to reduce safety and economic uncertainty
Percent of reactors that are new	As low as possible to reduce safety and economic uncertainty
Is option sustainable per repository limits?	Yes
Is option sustainable per uranium limits?	Yes

Table 3-3. Full Set of Metrics Used in this Study (Relative to Once-Through where Relevant)

AFCI Objective/Metric	Purpose	Weakness	Suggested Future Work
Objective 1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials			
Total mass in system Total mass in reactors Total mass in repository Total mass in separation & fuel fabrication	Basic parameters helping to understand how each case is functioning	None of these are directly tied to AFCI objectives and therefore should not be used to select among options.	N/A
Heavy metal (HM) mass in system	Common metric to understand waste management	HM mass is not directly an AFCI objective	N/A
TRU mass in repository TRU mass in system	Short-term objective to remove 99.5% of TRU from waste to repository	LTH, LTD, and LTR are more technically valid indicators of repository benefits	N/A
None	Short-term objective to improve management of Cs-Sr		TBD
Long-term heat (LTH) interval (watt-yr) from 50 to 1500 years after emplacement in a geologic repository	Indicator for repository capacity, hence the long-term objective to avoid need for 2nd repository for at least a century	See Section 3.1.1. LTH predicts repository heat-load capacity increases for cases constrained by mid-drift temperature, but not for cases constrained by drift wall temperature.	Improve metric to be a better predictor of heat-load capacity of a repository.
Long-term dose (LTD) (mrem/yr) at 1e4 to 1e6 years after emplacement in a repository. Key time is typically 5e5 years.	Long-term objective to reduce long-lived dose sources by 10x	See Section 3.1.2. We interpret this objective as requiring reduction of **peak** doses by 10x, **not** all doses at all times after emplacement.	Update on the basis of new numbers from the Yucca Mountain Project (YMP) when available.
Long-term radiotoxicity (LTR) (mrem/gram) at various times after decay starts. Key time is typically 1e3 years.	Long-term objective to reduce long-lived radiotoxicity sources by 100x	See Section 3.1.3. No direct regulatory value, but does suggest hazard relative to benchmarks such as uranium ore.	

AFCI Objective/Metric	Purpose	Weakness	Suggested Future Work
Objective 2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for spent fuel management			
Pu-239 in system Pu in system	Common simplified metrics for quantity of weapons usable material	Ignores all other weapon-usable isotopes, weights all Pu isotopes the same.	Should be discouraged, replace with Pu-239 equivalent metric. (see text)
Pu-239 fraction of total Pu in system	Indicator of quality of weapons-usable material, relevant to short-term goal of enhancing intrinsic proliferation barriers	Poor indicator of "quality", but simple to calculate.	Better "quality" metric needed. Dose calculations for representative fuels and geometries needed. Additional work needed for "key technologies" that facilitate export control (i.e. are not multi-use), technologies that become inoperable without international support.
Unshielded dose rate (duplicate of objective 4 metric)	Indicator of handling resistance, relevant to short-term goal of enhancing intrinsic proliferation barriers	Scaled from past calculation, not a new calculation. See section 3.3.	
Pu-239-equivalents in repository TRU mass in repository (duplicate of objective 1 metric)	Short-term objective to eliminate 99.5% of TRU weapons-usable material from repository	Pu-239 equivalent is the more technically valid measure of "weapons-usable" inventory, see section 3.3. "TRU mass" weights all TRU isotopes the same.	The metric "TRU mass" should be discouraged, replace with Pu-239 equivalent metric.
Pu-239-equivalents in system TRU mass in system (duplicate of objective 1 metric)	Long-term objective to stabilize weapons-usable inventory		
Objective 3. Enhance energy security by extracting energy recoverable in spent fuel and recycled material, ensuring that uranium resources do not become a limiting resource for nuclear power.			
Uranium ore needed	Compare versus 1.15x (short term) and 50x (long-term) goals	The metric is clear; the targets (1.15x, 50x) depend on beliefs regarding the size and cost of the uranium resource.	N/A
Burned U (BU) accumulated/used Depleted U (DU) accumulated/used	Helps understand how uranium resources are being used. Also, accumulation of BU and DU has economic and waste management implications.	Complete economic accounting would convert BU and DU from waste mgt ***liabilities*** to energy source ***assets*** once FR become available.	It therefore makes sense to account for the accumulation and use of BU and DU.

AFCI Objective/Metric	Purpose	Weakness	Suggested Future Work
Objective 4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system			
UOX operating reactors	Basic indicators of how each scenario functions		N/A
IMF/MOX operating reactors			
FR operating			
Unshielded dose rate	Indicator for economic and safety penalties from recycling TRU	Crude approximation of unshielded dose rate, which is itself a crude approximation of the underlying issue	Dose rates for representative fuels and geometrics.
Throughput = mass of discharge fuel per year by type (UOX, MOX, IMF, FR)	Indicator of economic (e.g. required facility capacity) and safety (e.g. transportation)		Examine how separation and fuel plants would change as the relative mix of U, Np, Pu, Am, Cm changes
Throughput = mass discharge fuel per year by element (U, Np, Pu, Am, Cm), by fuel type			
SNF sent to/retrieved from repository/year	Indicator of economic (emplacement/retrieval) and safety (transportation)		N/A
None at present	Economic value of other materials recovered from used fuel		TBD
Percent of fuel that is MOX or IMF in Fleet	Compare versus M. Todosow results.	Existing limits calculated for 1st pass	Need analogous limits for n>1 passes
Percent of reactors that are new types in Fleet	Desire as low as possible to reduce uncertainties		
Percent of fuel that is new in Fleet	Desire as low as possible to reduce uncertainties		
Percent of fuel that is Pu in Fleet	Compare versus French criterion of 10%	Existing limits calculated for LWR with 1st pass MOX	Need analogous limits for IMF and for n>1 passes
SNF at reactor in wet storage (< 5 yr)	Long-term objective to reduce at-reactor inventories	The 5-year threshold is an approximation that is ok for UOX but may not be for recycle fuels	A mini-model that determines when wet-to-dry can occur, and when aging pad at Yucca Mtn can be avoided.
SNF at reactor in dry storage (> 5 yr)			
Mass of Np, Am, and Cm in wine cellar storage (mass of Pu in wine cellar is zero)			Need more analysis of the "wine cellar" and how it can be managed.

We also note that the AFCI has two key strategic or schedule goals: [DOE2005a]

- "Develop and make available for industry the separations technology needed to deploy by 2025 a commercial-scale spent fuel treatment facility capable of separating transuranics in a proliferation-resistant manner for their recycle and destruction through transmutation."
- "Develop and make available the fuel cycle technology needed for commercial deployment by 2040 of fast spectrum reactors operating either exclusively as transuranics transmuters or as combined fuel breeders and transmuters. Actual decisions to deploy fast reactors will, of course, be made by industry in response to market needs."

The dates of 2025 and 2040 therefore help structure the development trees in Chapters 6 and 7. Specifically, baseline analyses use 2025 as the start date for recycling, generally in thermal reactors. Similarly, baseline analyses use 2040 as the start date for FR and FR recycling.

3.1. Waste Management

The three primary metrics are long-term heat (LTH), long-term dose (LTD), and long-term radiotoxicity (LTR). Calculations of associated coefficients in this study are described in Appendix D.

3.1.1 Long-Term Heat (LTH)

The AFCI program wishes to avoid the technical need for a second geologic repository this century. Figure 3-1 shows the number of repositories that would be required for the waste generated this century if the once-through strategy is kept. Consistent with Table 1-3, at a nominal growth rate of 1.8%, we would need 10 YMP-sized repositories, or therefore to improve the utilization of the first repository by 10x. At a growth rate of 3.2%, the highest considered in this year's report to Congress [DOE2005a], we would need an improvement of 22x. The 3.2% growth rate results in 370 GWe of installed capacity in 2050, similar to the MIT high-growth scenario of 477 GWe.[Dixon2004, MIT2003] The DOE laboratory directors published a report [Grunder2003] with targets that imply a growth rate of 4.5%/yr at least until 2050. Dixon and Piet estimated the growth rate by noting the objective of "50 percent of U.S. electricity and 25% of U.S. transportation fuels produced by nuclear energy by 2050," this results in 700 GWe installed capacity by 2050.[Dixon2004] If continued to 2100, this would require an improvement of 50x to stay within one repository. Therefore, we believe that avoidance of a second repository this century means we need to improve the utilization of the first repository by 10x to 50x.

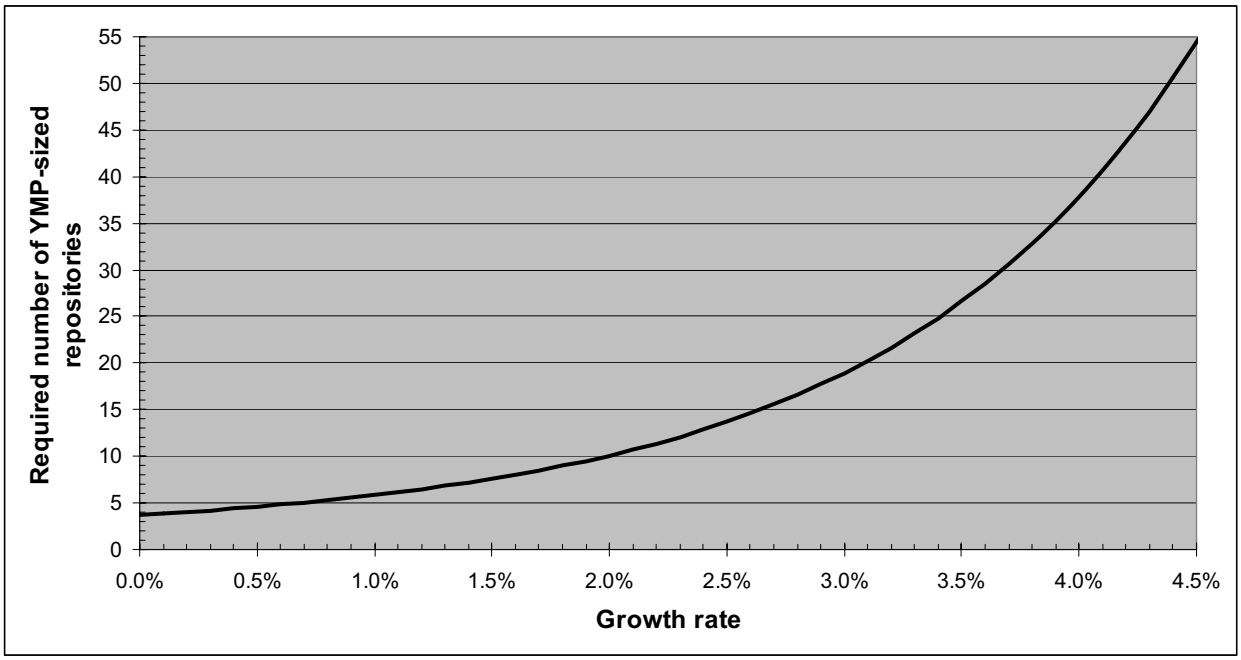

Figure 3-1. Number of YMP-sized repositories needed in the once-through strategy as a function of growth to 2100.

One of the key repository capacity factors is long-term heat (LTH). As analyzed and explained by Wigeland, [Wigeland2004a, Wigeland2004, Wigeland2005] a major factor determining the amount of waste that can be emplaced in a YMP-like repository is the heat generated from the waste from when ventilation of the repository stops to ~1500 years. The ventilation stops when the repository is closed (sealed). Current policy and regulations constrain the closure time from a minimum of 50 years to a maximum of 300 years. The end-period of the heating interval (~1500 years) is approximate; indeed, a single value is an approximation of a time-dependent heat transfer calculation. For present purposes, we use an LTH metric defined as the energy (watts-year) released per mass of isotopes emplaced in the repository. This requires us to account for the heat released during the time interval ventilation-stop to 1500 years from the isotopes and its decay products. Appendix D contains a list of these coefficients for three ventilation-stop cases, 50, 100, and 300 years after emplacement.

Wigeland's analyses show cases vary as to which temperature constraint dominates, as follows:

- Temperature below 96 °C between drifts, so that water can drain between drifts
- Drift wall temperature below 200 °C, at time that waste is emplaced
- Drift wall temperature below 200 °C, at time that waste is no longer ventilated, i.e., repository closure

Figure 3-2a shows the actual heat-load-limited improvement factors calculated by Wigeland.

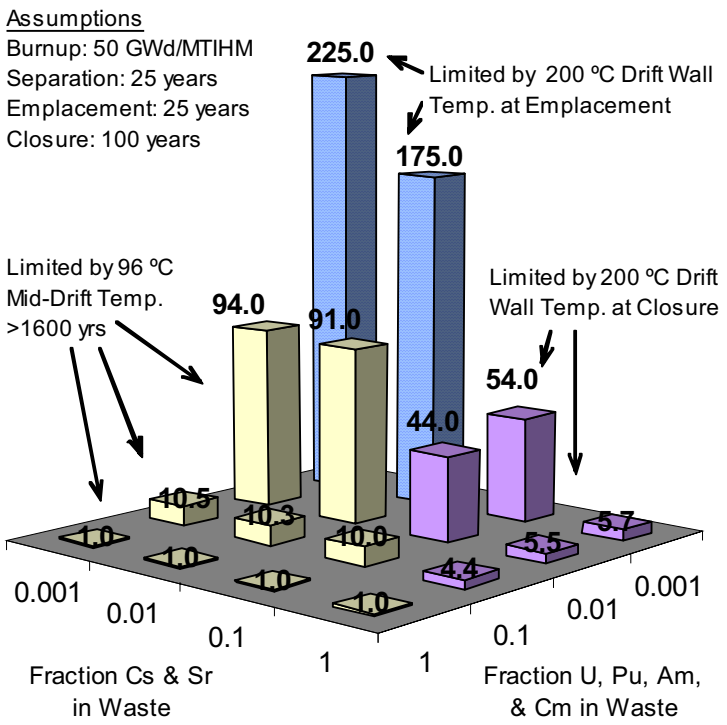

Figure 3-2a. Heat-limited repository capacity improvement factors [Wigeland2006]

The LTH used in this study is a merely a metric that allows us to approximate repository capacity improvement. Figure 3-2b shows the actual heat-limited repository capacity improvement [Wigeland2005, Wigeland2006] as a function of the LTH metric calculated by us.

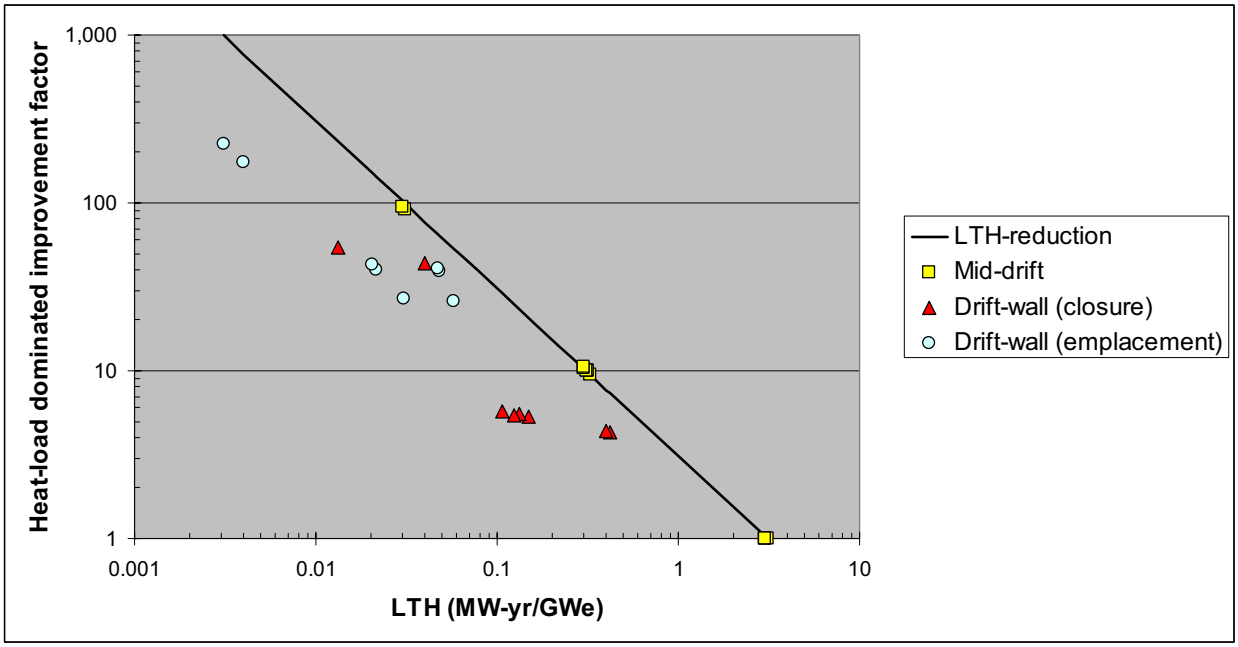

Figure 3-2b. Repository capacity improvement factors versus calculated LTH metric

The future contains several types of data points, as follows:

- Black line = repository capacity improvement if dictated solely by LTH improvement.
- Yellow squares = limited by limited by 96 °C mid-drift temperatures
- Red triangles = limited by 200 °C drift wall temperature at closure
- Blue circles = limited by 200 °C drift wall temperature at emplacement

We see that the LTH metric is an excellent predictor for cases dominated by mid-drift temperatures. It overpredicts for cases dominated by drift wall temperatures. Therefore, judging from Figure 3-2b, **if our goal is to reduce heat constraints on the repository by 10-50x, we should reduce LTH by 10-200x**.

Figure 3-3 plots the LTH values (W-yr/g-isotope) calculated for this study. Appendix D contains a table of the values. Note that the LTH values vary by 4 orders of magnitude. The highest isotopes are U232, Pu238, Pu240, Pu241, Am241, Am242m, Am243, Cm242, Cm243, Cm244, Sr-90, and Cs-137. The LTH values for Cs90 and Sr137 decrease significantly when the integration period changes from 50-1500 to 100-1500 to 300-1500 as their ~30 year halflives allow significant decay during the ventilation period. Shorter-lived TRU isotopes, e.g., Cm244 (18 yrs) do not show as rapid a decrease in LTH values because they are merely decaying into other radioactive isotopes, which themselves generate heat in the time period of interest.

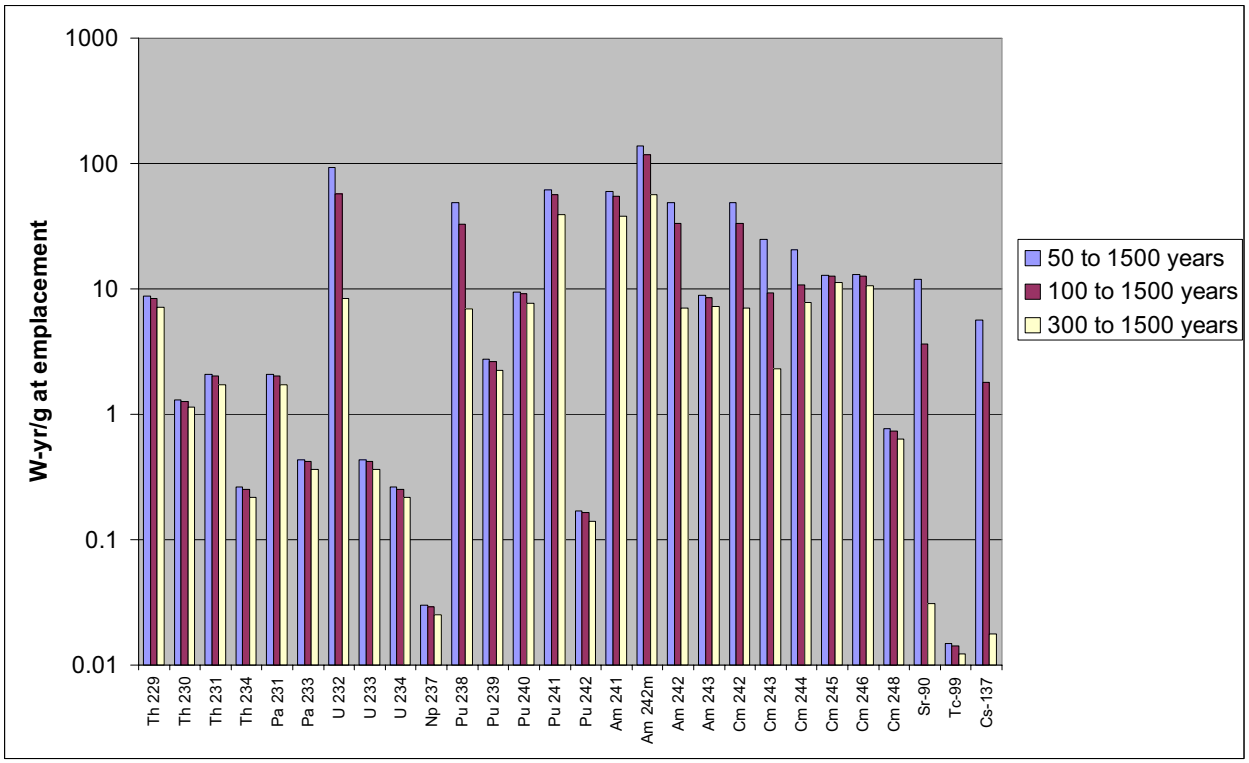

Figure 3-3. LTH values (watt-year/gram) for key isotopes

3.1.2 Long-Term Dose (LTD)

Another potential limitation on the amount and nature of waste emplaced in the repository is long-term dose. As of this writing, there are two dose standards for the maximally exposed individual (MEI) of the public living near the repository:

- 15 mrem/year at time periods less than 10,000 years; the peak dose in this time period is typically at 10,000 years
- 350 mrem/year at time periods between 10,000 and 1,000,000 years; the peak dose in this time period is typically ~500,000 years.

There are four possible AFCI targets:
1. Reduce all long-term dose sources (i.e. all long-term isotopes) by 10x.
2. Reduce the long-term dose sources so that the peak long-term dose is reduced by 10x.
3. Reduce the long-term dose sources so that the peak long-term dose is reduced by as much as the heat constraints are lowered (10-50x), i.e., so that as more reactor-years' worth of waste is emplaced, the net dose to the maximum exposed individual of the public does not increase relative to the throw-away fuel cycle.
4. Reduce the long-term dose sources so that emplaced waste meets the 15 and 350 mrem/year standards.

A literal reading of the current AFCI objectives [DOE2005a, DOE2005b, DOE2006] would lead to target version 1, but we reject this as outside the spirit of the AFCI objectives. It would automatically mean that all long-term isotopes would have to be reduced by 10x, regardless of the totals, regardless of how peak doses were impacted.

Target-2 is a minimum objective to show compliance with AFCI objectives.

Target-3 is more stringent than target-2 because the peak dose would have to be reduced by as much as 50x depending on the heat-reduction factor.

For once-through fuel, the current peak dose (at 500,000 years) per the most recent target-4 is 31 mrem/year,[Halsey2005] providing a margin of a factor of 11 versus the draft standard of 350 mrem/year. If we reduced the peak dose by 10x (target-2) and emplaced 50x more waste (achieve heat reduction of 50x), we would obtain 155 mrem/year. Thus, hypothetical target-4 is not controlling relative to target-3.

In summary, we need to reduce the peak doses by at least **10x** (target 2) and possibly as much as **50x** (target 3). Of course, these targets need to be reexamined if YMP-calculated hypothetical doses change significantly.

W. Halsey has provided (from DOE-RW), current estimates of maximum repository dose, shown in Figure 3-4. The peak dose is 31 mrem/yr at 500,000 years.

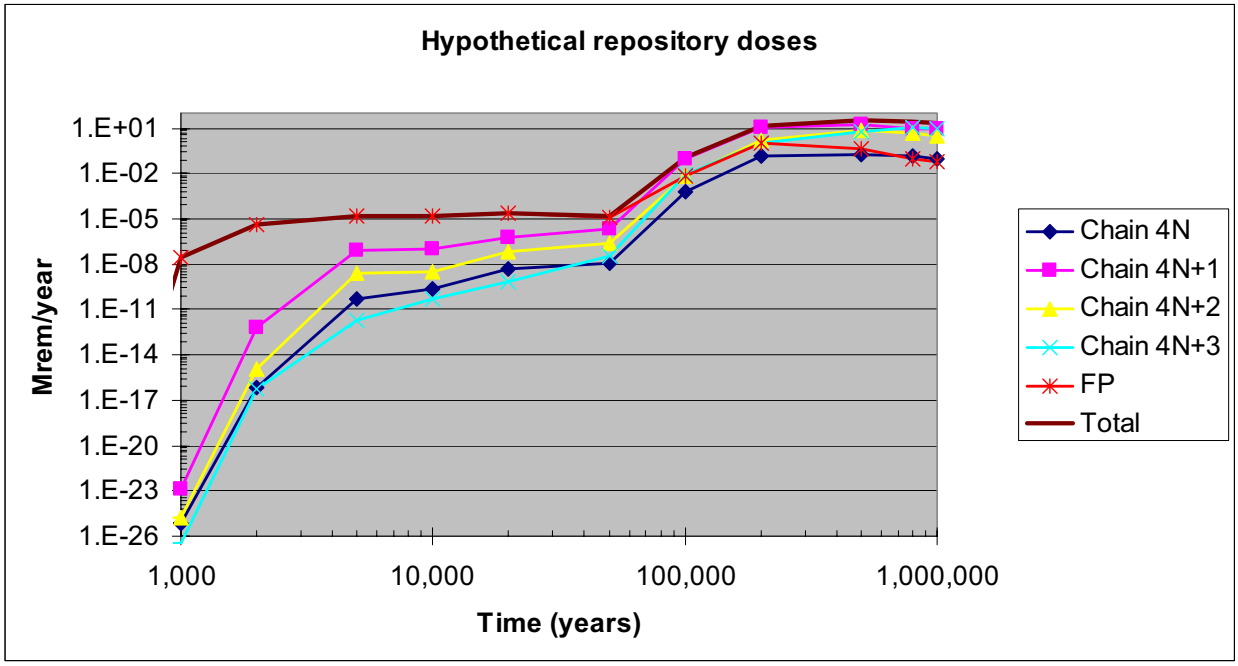

Figure 3-4. Hypothetical repository dose per DOE-RW via W. Halsey,[Halsey2005] dose from isotopes grouped by decay chain

For our assessments, we have estimated the dose at time T from all isotopes that result from a gram of each isotope emplaced into the repository, by scaling from the DOE-RW calculations. The units are therefore mrem/yr at time T/g-isotope emplaced in the repository. We calculate these coefficients at 10,000, 20,000, 50,000, 100,000, 200,000, 500,000, 800,000, and 1,000,000 years after emplacement as those are the time periods in the data set provided by DOE-RW. Figure 3-5 shows the results for 10,000 and 500,000 years. To improve readability, Figures 3-6 and 3-7 show only key isotopes at 10,000 and 500,000 respectively.

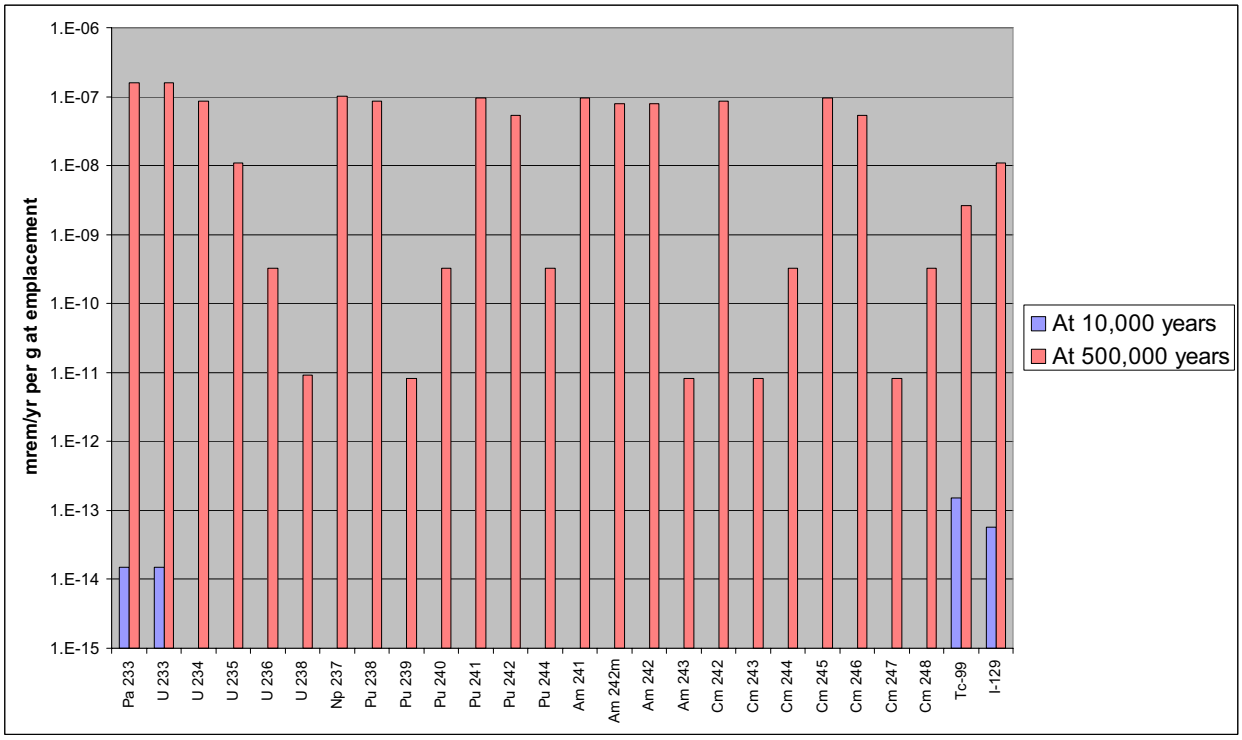

Figure 3-5. LTD values for several isotopes at 10,000 years and 500,000 years after emplacement

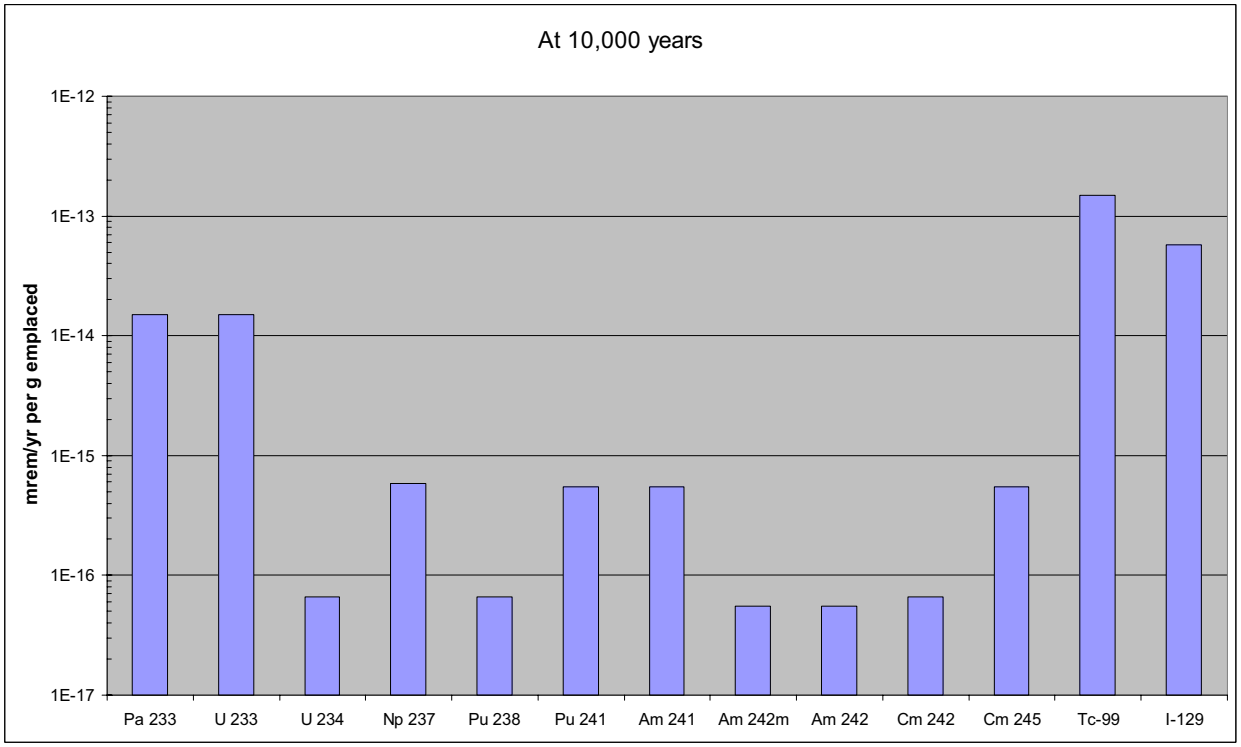

Figure 3-6. LTD values for key isotopes at 10,000 years after emplacement

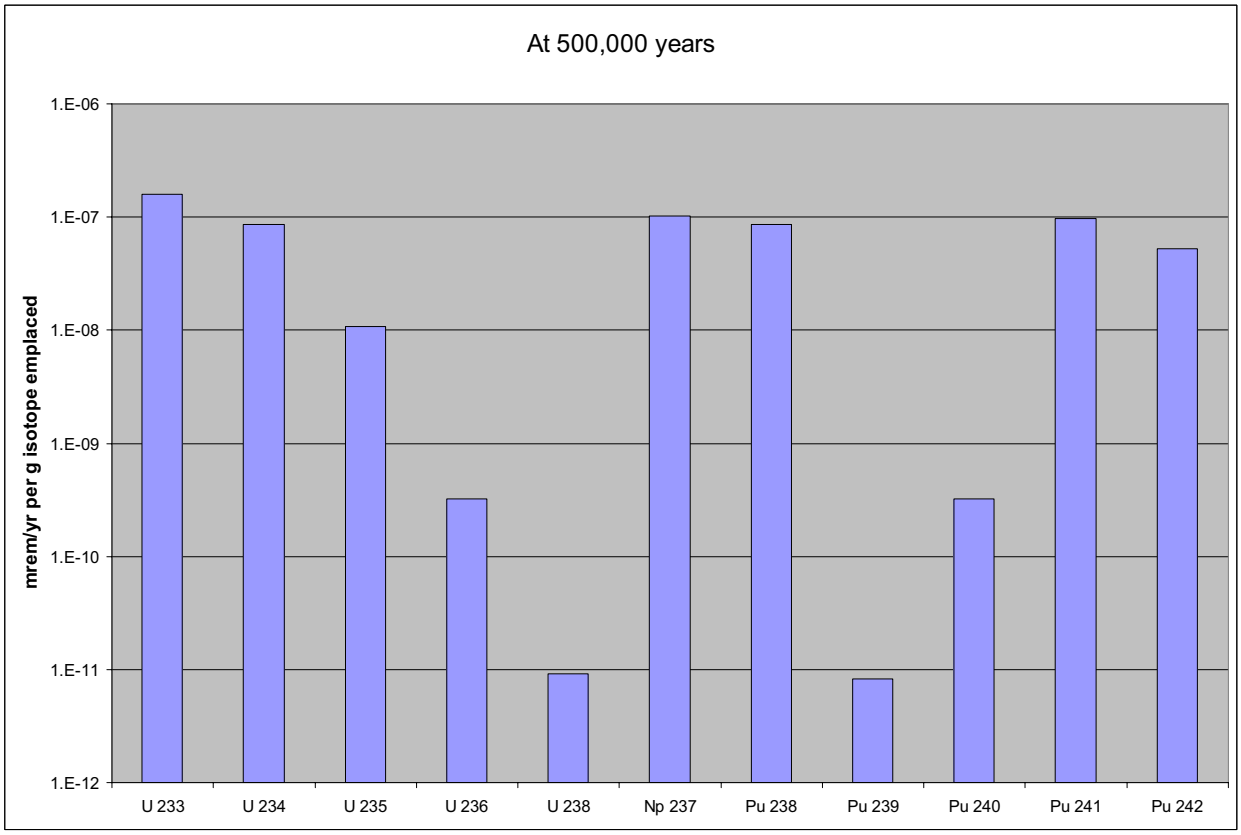

Figure 3-7. LTD values for key isotopes at 500,000 years after emplacement

In the above Figures, note that several groups of isotopes have the same value. For example, the values for Np237, Pu241, and Am241 are the same because on this time scale, any amount of Am241 or Pu241 will have decayed into Np237. And, Pu238 and U234 are the same; on the time scale of interest, Pu238 (88 years) decays into U234 (2.4e5 years).

Also note that Tc99 and I-129 dominate at 10,000 years, but that U and TRU isotopes dominate at 500,000 years.

Finally, note that the uranium values are high, especially for the relatively shorter-lived U233 and U234. We show in Chapter 5 that uranium doses can be significant. And, recycle approaches that generate U234/Pu238 are harmed in this metric.

3.1.3 Long-Term Radiotoxicity (LTR)

Long-term radiotoxicity differs from LTD because it ignores how much of isotopes emplaced in the repository can actually transport to human receptors. The advantages of LTR as a metric are that it is independent of repository location and design, independent of repository calculational uncertainties, and one can compare LTR directly to uranium ore. The first two advantages are why international assessments of waste management advantages tend to use LTR rather than either LTD or LTH as metrics. The last advantage warrants discussion here. Used UOX-51 tends to have LTR higher than uranium ore for ~400,000 years, coincidentally about the time period of peak LTD.

Figure 3-8 shows a classic decay of radiotoxicity relative to uranium ore.[DOE2005a] We note that a reduction of LTR by ~100x would mean that recycle waste would have lower radiotoxicity than uranium

ore within 1,000 years after emplacement. This brings the time scale for repository design hypothetically within engineering experience, whereas proving performance at 400,000 is problematical. This is the logic underlying the AFCI program objective of a factor of 100x reduction. So, the AFCI has a goal to reduce **LTR by a factor of 100**, at 1000 years after placement. To meet the underlying "no worse than uranium ore" objective, the LTR reduction can be less for times greater than 1000 years, e.g. a reduction of 30x at 10,000 years, 10x at 50,000 years, and 3x at 100,000 years would appear sufficient.

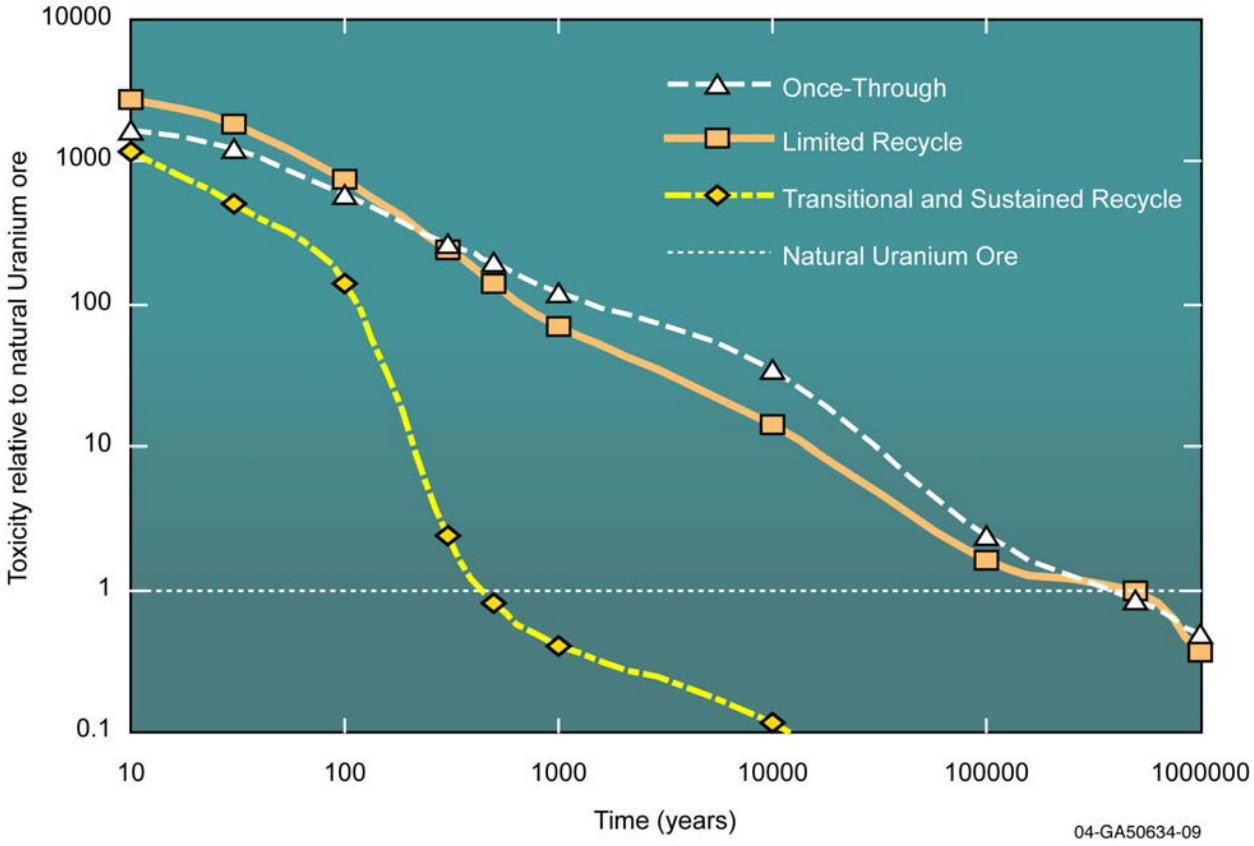

Figure 3-8. Radiotoxicity relative to natural uranium ore [DOE2005a]

To assess this objective, we calculated LTR coefficients, mrem/year per gram ingested (or drunk). Appendix D contains a list of these coefficients. The calculation of LTR per isotope is conceptually straightforward; the results calculated for this study are in Figure 3-9. Note that the coefficients include all daughters from the original isotope as they accumulate. So, for example, the coefficients for uranium isotopes in Figure 3-10 grow until all their respective daughters reach equilibrium and then decay slowly at their long halflives. U234 (2.4e5 yr) peaks at about 1e5 yr because Th230 (7.7e4 yr) is approaching equilibrium only at the time that U234 is significantly decaying. U235 (7.04e8 yr) reaches equilibrium at about 1e5 because Pa231 (3e4 yr) has grown in. U236 (2.3e7 yr) is at equilibrium early because it major daughter, Th232 (1.4e10 yr) is longer lived. U238 (4.5e9 yr) reaches equilibrium at about 1e6 yr because of U234 and Th230.

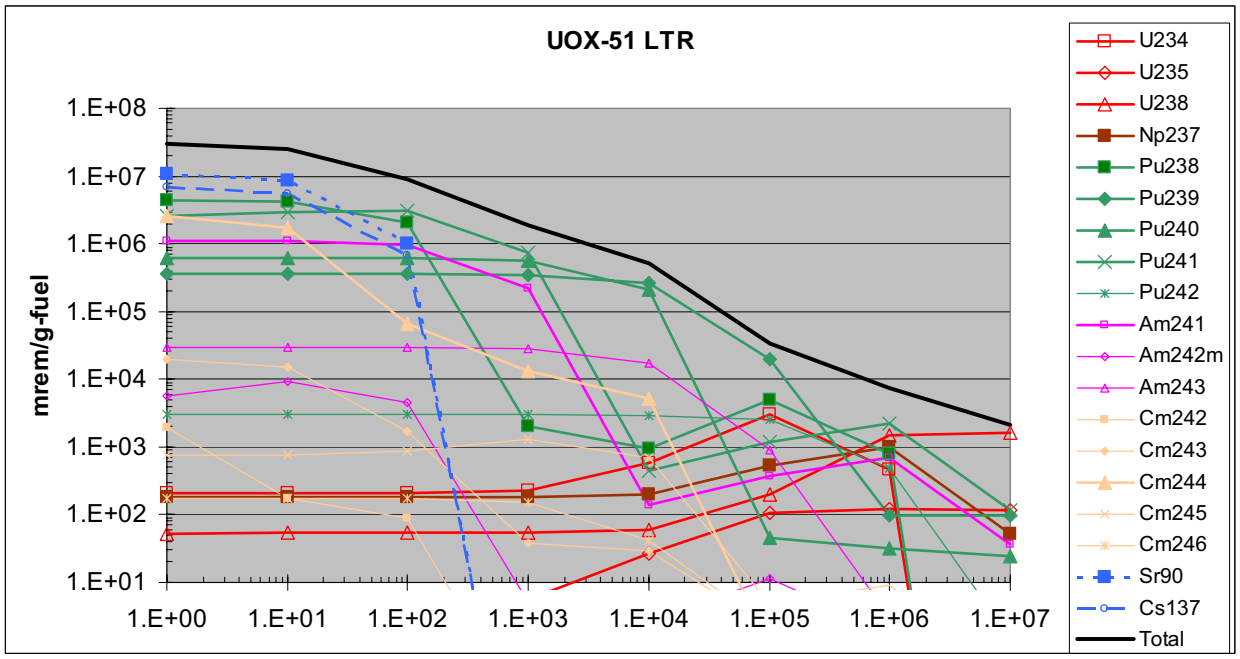

Figure 3-9. LTR values calculated for this study, selected isotopes only. The radiotoxicity of uranium ore equivalent to this much fuel is 1.3e4 mrem/g-fuel, or about 7.9 times higher than U238 in this figure to account for the conversion between ore and enriched uranium (7.9 g-ore/g-fuel)

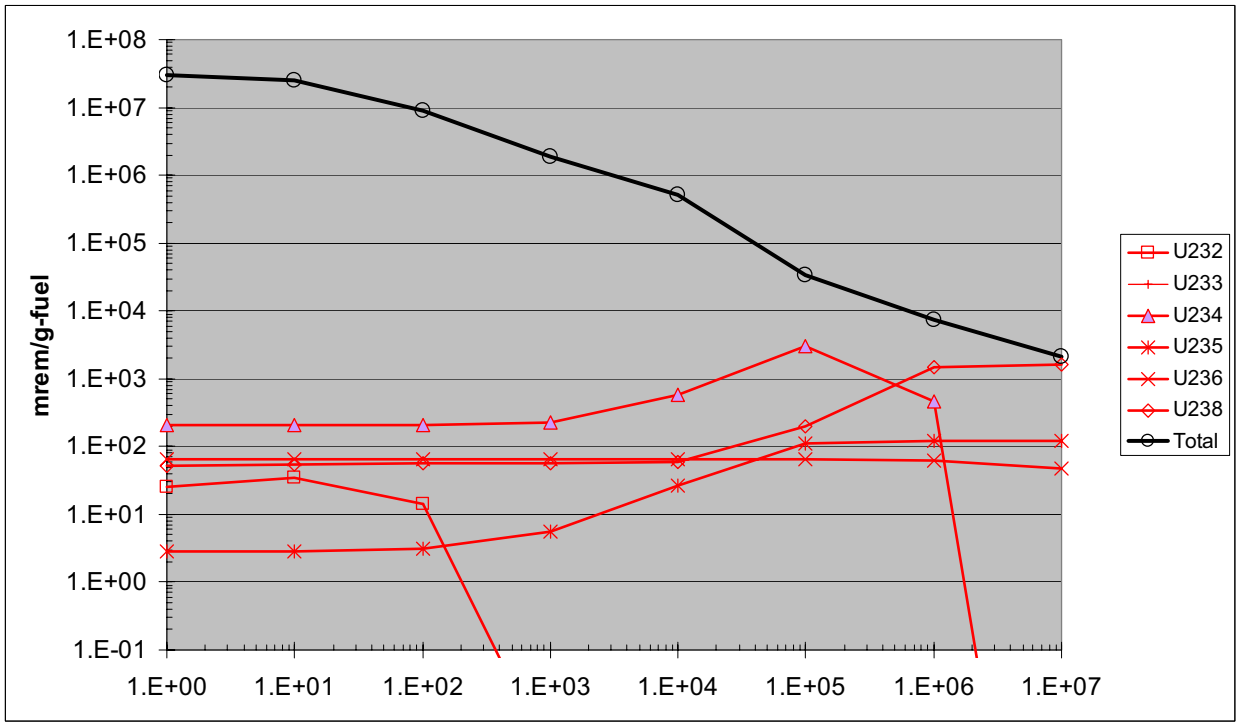

Figure 3-10. LTR values calculated for this study, uranium isotopes only. The radiotoxicity of uranium ore equivalent to this much fuel is 1.3e4 mrem/g-fuel, or about 7.9 times higher than U238 in this figure to account for the conversion between ore and enriched uranium.

With renormalization of our LTR results to uranium ore, it is possible to compare the current values with an entirely separate, independent calculation from the earlier Advanced Accelerator Application program.[AAA2001] as well as the 2005 Report to Congress [DOE2005a]. Figure 3-11 shows reasonable agreement, with the exception that the calculation in [DOE2005a] did not include the radiotoxicity of uranium because it was focused on relatively short time periods. Thus, at long times, the values in [DOE2005a] drop significantly below the current study or the 2001 study from the AAA program.

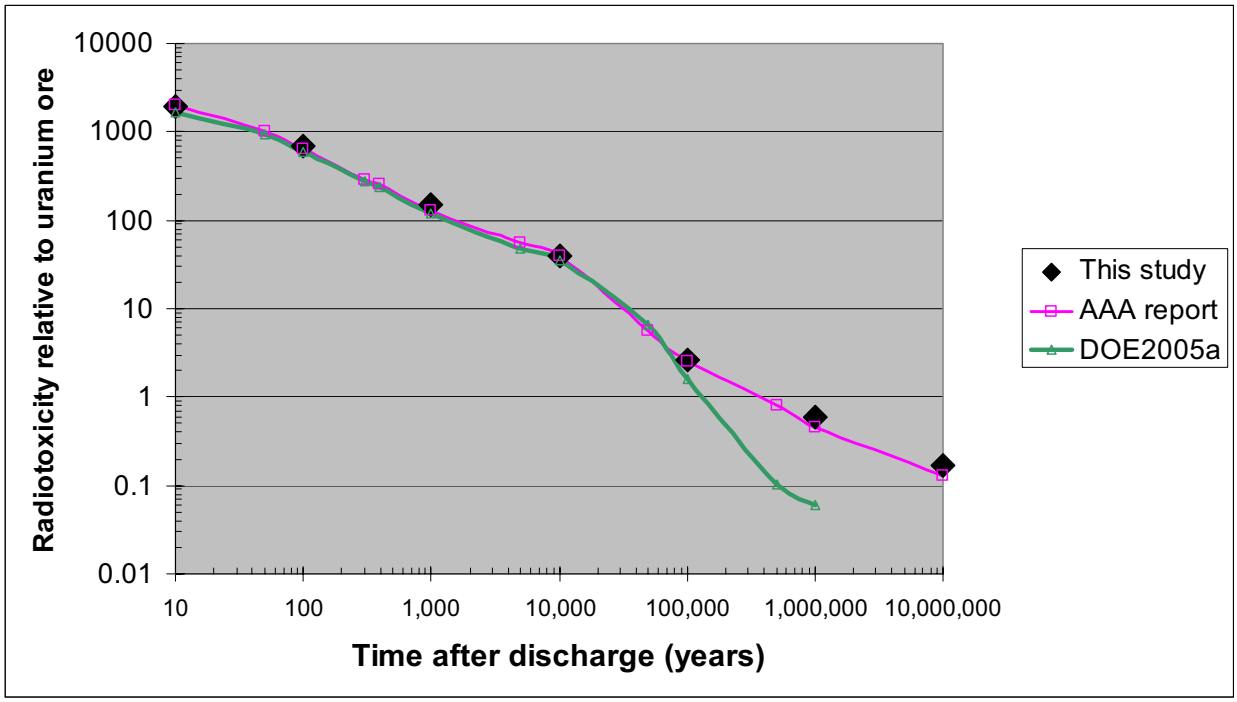

Figure 3-11. LTR values calculated for this study compared to previous reports.[AAA2001, DOE2005]

3.2. Proliferation Resistance

For present purposes, we consider proliferation resistance in three ways - quantity of weapons-usable material, quality of weapons-usable material, and ease of handling.

The IAEA defines "weapons-usable" to include any isotope with finite critical masses, excluding only those that generate so much heat that it is impossible/impractical to keep the material solid long enough to assemble and detonate a weapon. The first test allows U233, U235, and all TRU isotopes. The IAEA uses the second test to exclude Pu mixtures with more than 80% Pu238. This works out to an approximate threshold of 450 watts/gram. We tested this threshold against other isotopes and found one with sufficient heat to matter, Cm-244. A Cm mixture with more than 16% Cm-244 would appear therefore to not be weapons-usable. So, if Cm is separated from other TRU, it would not be weapons usable. However, Cm mixed with Am or other TRU falls below the threshold and would therefore be weapons usable. Furthermore, when the Cm cools for several decades it decays to Pu-240 which is once again considered weapons usable. Therefore, "weapons usable" includes [Piet2004]

- Uranium with >12% U233 or >20% U235
- Pu
- Np
- Am

- Cm only if mixed with other TRU, cooled for a long period

3.2.1 Quantity

The AFCI objectives refer to "TRU weapons-usable inventory." A literal interpretation of this phrase would mean we measure quantity by simply the TRU mass. We believe that that is inappropriate because it assigns the same value to all TRU isotopes, yet clearly they contribute differently to the ability to make a nuclear weapon. Instead, we primarily use "Pu-239 equivalent" masses. Each isotope is weighted by its bare sphere critical mass; see Appendix D for values used in this study (Figure 3-12). In particular, this weights Am isotopes much lower than Pu-239 and Pu-238.

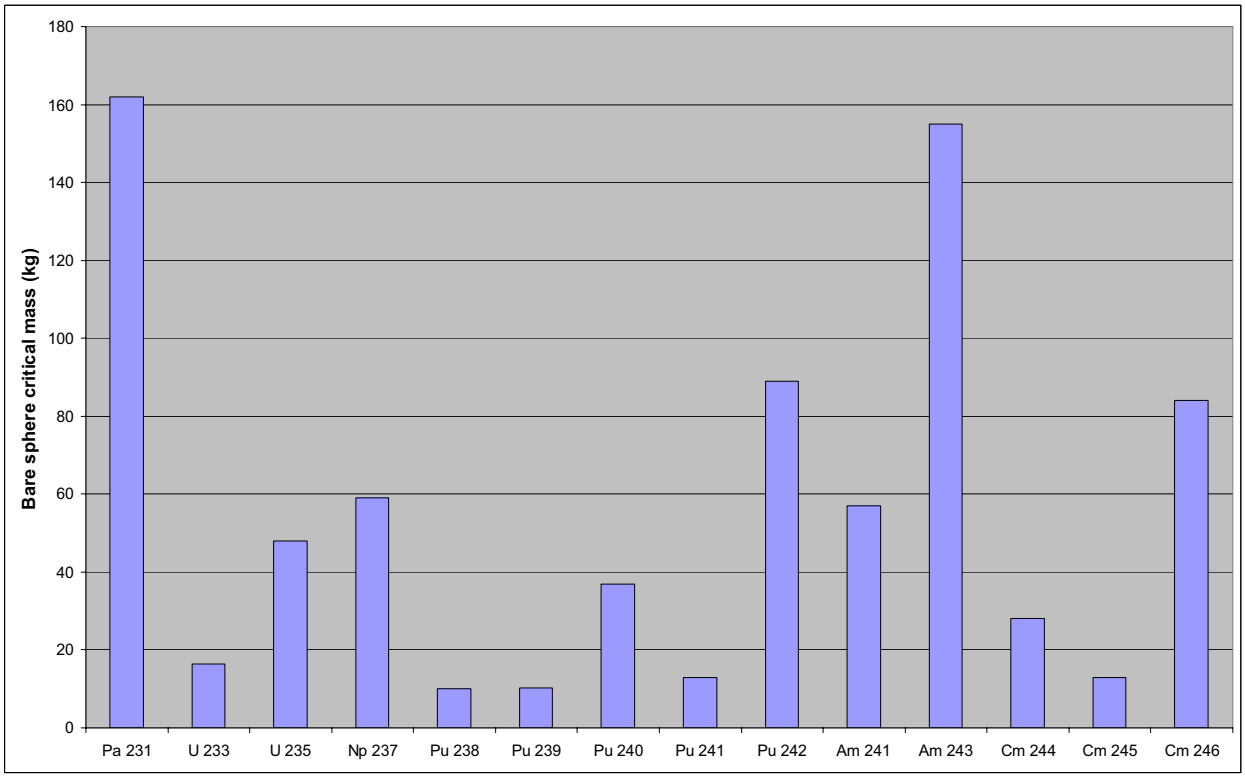

Figure 3-12. Bare Sphere Critical Mass, data from [NAS2000]

3.2.2 Quality

The AFCI objectives do not directly refer to the quality of weapons-usable material, but the potential importance of "quality" can be inferred by the reference to intrinsic proliferation resistance. In this study, we use the simplest "quality" metric, the ratio of Pu239/Pu-total.

3.2.3 Dose rates and the Spent Fuel Standard

There are wide differences of opinion as to the importance of dose rates for weapons-usable material in the fuel cycle, this is part of uncertainty F6. (There are also wide differences of opinion on the economic implications of dose rates, this is part of uncertainty F5; see section 3.4.)

The narrow interpretation of the "spent fuel standard" is simply 100 rem/hr contact dose rates. Stillman [Stillman2004d] has shown that this is not achieved with the first recycle of UOX unless the material is spiked with Cs. For metric purposes, we have generated a crude approximation of dose rates by examining Stillman's results. His calculations were for 4.3-kg of spent UOX in a prototypical canister geometry. It is important to recognize that self-shielding effects will be important for transuranic (TRU) mixtures, because most of the gamma production is low energy, which can be readily shielded. A truly unshielded configuration can allow high dose rates resulting from the TRU alone,[Hannum1995], but these gammas are shielded by even a thin wall container as demonstrated by Stillman. Only thin steel is needed to shield some of the actinide gammas, which is why the dose rates are relatively low (even for Am) compared to previous bare material evaluations. By analyzing Stillman's results, we obtain these "rules of thumb" for discharged fuels with fission products removed.

Gamma	Am241	0.100 rem/hr
	Pu-reactor grade (i.e. contains Pu238)	0.050 rem/hr
	Pu-weapons grade (i.e. Pu239	0.005 rem/hr
Neutron	Cm244	0.500 rem/hr
	Pu238	0.010 rem/hr
	Pu-weapons grade	0.001 rem/hr

These scaling analyses indicate that, consistent with Stillman's results for UOX, none of the recycle fuels meet the 100 rem/hr standard. (See Chapter 5.) So, the basic conclusion is that the only way to meet the narrow interpretation of the "spent fuel standard" is to spike the fuel with Cs or some other penetrating gamma or neutron emitter. Feasible, perhaps, but this is undesirable for several reasons.

But, perhaps this narrow definition (100 rem/hr) is a bit of a "red herring." A NAS panel led by John Holdren [NAS2000] used a broader definition of "spent fuel standard" to decide on the best course of action for disposition of excess weapons-grade Pu. In it, they considered other make-it-hard-to-mess-with parameters such as physical size, chemical form, etc. – quite similar to other methodologies. They did not insist on 100 rem/hr as a requirement.

Therefore, we do not consider 100 rem/hr as a requirement. Where there are other reasons to recycle material that increases dose (e.g. Am and waste management benefits), these should be considered. Even better is to search for ways to obtain the benefits of recycling Am without incurring the cost of fully remote fabrication and handling. If the only reason to recycle a material is to increase dose rates (e.g. Cs-spiking), that possibility should only be retained as a low priority alternative in case proliferation policies (F5) force adoption.

3.3. Energy Recovery

As describe in more detail elsewhere, [Piet2004] we identify three major possibilities for uranium resources

- Pessimistic, known recoverable resources – 3.1 million tonnes-U [Herring2004, Steyn2003]
- Realistic, estimated conventional resources – 16 million tonnes-U [Herring2004, Steyn2003]
- Optimistic, including unconventional resources, e.g., 4,200 million tonnes-U from sea water and phosphate.[Herring2004]

3.4. System Management

These sets of issues are typically underappreciated. The entire fuel cycle system must fit together, including:

- Balancing the types of facilities (different reactors, fuel enrichment, fuel fabrication, separation) – this becomes more complicated the more elements there are to the system.
- Readiness Levels – Is a proposed change to the status quo ready to proceed?
- Robustness as described in Chapter 1 – the property of a system that it is more likely to be found "correct" as future circumstances change.
- Agility (adaptability) as described in Chapter 1 – the property of a system that it is easier to adapt to new circumstances as they arise. [The combination of robustness and agility produce solutions that minimize the chance of "regret" later and are therefore are easier to sell now.]
- Economics – capital versus operating costs, balancing the short and long term costs and benefits, matching, working to ascribe costs to those receiving the benefit, short and long term energy security
- Safety – worker and community, short and long term risks and benefits, life cycle analysis compared to other energy resources

3.4.1 How does the system work together?

Once-through is simple and linear. We know how to make the pieces work – except that no country has put the final piece into the system, namely geologic disposal. Nonetheless, over time, we know how to make mining, fuel enrichment, fuel fabrication, reactors all balance. Currently, there may be a perturbation occurring (uranium price spike) in anticipation of higher uranium need. This will presumably lead to increasing uranium mining to re-balance the system. If the anticipated growth of reactors does not occur, an increase in mining could instead throw the system out of balance.

Pu recycle (TR and MOX-Pu) is relatively simple. There are several additional elements in the system – separation plant, MOX fabrication plant (in addition to the UOX fabrication plant), and additional types of waste. This requires balancing more elements in the system, each with their characteristic time frames. Several countries are making this work – except no country has geologic disposal and there is typically a stockpile of Pu accumulating.

The opposite extreme is a hypothetical mix of LWR (cheapest electricity production?) and VHTR (cheapest hydrogen production?) and consumer FR (garbage disposal units). Each of these reactors would have their own constraints and own fuel. Yet, the system would be expected to work together.

3.4.2 Readiness levels

We have not considered readiness levels in this report.. We do note that there are at least five types of "readiness" that should be considered in future work.

- Technical Readiness – dependability of R&D products.
- Infrastructure Readiness – ability, barriers to use as much of the existing infrastructure as practical
- Industrial Readiness – ability, willingness of industry to participate in a new fuel cycle
- Regulatory Readiness – ability to proceed from the regulatory standpoint
- Socio-Political Readiness – ability to proceed from the "public" perspective. Over long time periods, socio-political readiness will provide regulatory readiness.

3.4.3 Robustness and agility

These concepts are critical, but difficult to define.

We consider robustness of a decision to be high when preferences are unchanged as postulated "futures" change, i.e., people postulating different "futures" will come to the same decision, the same preference. We find, for example, that with two exceptions, the preference of multi-pass IMF over multi-pass MOX is unchanged as one considers differing importance among metrics depending on what one thinks the future looks like. The first exception is that if uranium resources are a major constraint, one would select neither and go straight to breeder fast reactors. The second, of course, is that the technical feasibility and relative cost of multi-pass IMF has not been established.

Another aspect of robustness, analyzed in Chapter 6, is the property of a system that reduces the need to change the system as circumstances change. Consider a scenario where uranium resources were found to be less of a constraint than previously thought, e.g., if new resources were found or the practicality of unconventional resources was established. Most of the systems would be robust in that this new information would not induce our successors to change how they were operating, indeed, one would not suddenly wish to change the system selected. The exception could be BFR; if adopted and later found that uranium was not a constraint, one might either lower the breeding ratio of the BFR or reduce their contribution to the reactor fleet.

Agility is a related but separate concept. Agility is the property of an option that allows us to change the option if new circumstances warrant. Fast reactors, for example, that can be altered from consumer to breeder are very agile.

3.4.4 Economics

A major late-FY2006 activity is merging the system dynamic model with the Economic Database activity led by D. Shropshire. The present report does include some factors that inform with regard to economics:

- Waste management metrics (covered in section 3.1)
- Uranium needed (covered in section 3.3)
- Percent of new reactors vs existing reactor types
- Percent of fuel that is hands-on (UOX), glovebox (PuNp), or remote (containing Am or Cm).
- Transportation mass flux
- Interim storage requirements

Future work would include

- Quantification of the economic impact of the above indicators
- R&D cost
- Possible alternative uses for separated products (rare earths, Cm/Bk/Cf, Tc as steel-strengthening agent)

In particular, our analyses have shown that the following need to be addressed in future economic analysis.

- The feed composition into a separation plant changes with time in most scenarios (see section 5.x); how much variation is acceptable with what cost impacts?
- The fuel composition into a reactor can change with time. We ignore isotope enrichment of any element other than uranium. So, the major "knobs" that can control net fuel composition are (a) how used fuel from different reactors are blended at the separation plant and/or fuel fabrication plant, (b) uranium enrichment, (c) changing the U to Pu ratio by fuel type or fuel composition, (d) changing the Am to Pu ratio. Using any of these knobs presumably has cost implications.
- Homogeneity vs. heterogeneity. As will be seen in Chapter 4, there are at least three ways to implement the IMF concept. What are the cost implications, e.g., economies of scale?

- "Full core" IMF – all fuel in a reactor is IMF (assuming this is feasible)
- Blended core IMF – some pellets (and pins) are IMF, some are UOX. All assemblies are the same.
- Blended core IMF with Am targets – some pellets/pins are IMF, some are UOX, and some are Am targets. All assemblies are the same

- Cost of lost resources for economic expansion due to consuming "excess" Pu.
- Cost of "mining" spent fuel from repository if once-through continues.

3.4.5 Safety

For present purposes, we divide safety issues into three categories.

- Reactors
- Fuel cycle facilities, primarily separation and fuel fabrication facilities
- Transportation

With regard to reactor safety, Table 3-4 lists safety constraints based on void coefficient and other considerations. It is important to note that these limits are based on the first recycle pass (MOX or IMF); little work has been done on subsequent passes. We therefore consider "% Pu" as a key metric to calculate.

Table 3-4. Thermal Reactor Safety Constraints on MOX/IMF [adapted from Todosow2004]

	single-pass MOX			single-pass IMF		
	% of reactors that can use	% of core	% of all fuel in fleet (reactors x core)	% of reactors that can use	% of core	% of all fuel in fleet (reactors x core)
Current PWRs and BWRs	50%	33%	16%	25%	25%	6%
Future PWRs and BWRs	100%	100%	100%	50%	100%	50%

There is also a French criterion: 10% Pu in core, based on MOX-Pu fuel [Salvatores2003]

Although we do not have corresponding limits on all the cases in this study, we do have these expectations.

- Systems with enriched U-235 reduces the problem with including TRU in the core [Salvatores 2003]
- As the Pu vector in MOX-cores degrades with subsequent recycles, the problems get worse [Salvatores2003]
- Including Np or Am or Cm – impact unclear.

The other safety concerns will generally scale with how much mass is being processed, fabricated, and transported. We therefore pay special attention to throughput, see section 5.1

3.4.6 At-reactor inventories

Per AFCI objectives, there is a simple metric we use – the at-reactor inventories of used fuel. In our calculations, we differentiate between TR fuel younger than 5 years, which is assumed not transportable

to centralized separation plants, and fuel older than 5 years, which is transportable. The short-hand terms are wet and dry storage.

An item for future work, however, is to assess when the wet/dry (non-transportable/transportable) threshold is crossed for recycle fuels. As fuels become "hotter," we would expect the waiting period to increase. However, much of the short-term heat driving this issue are relatively short-lived fission products that (a) we do not track in DYMOND because they are not relevant to fuel and waste management and (b) do not substantially change with recycle fuels. Per fission energy released, the short-lived fission products are similar. So, at relatively short times, one would expect the fission-product-dominated heat from MOX at 51 GWth-day/tonne-HM to be similar to UOX at the same burnup, as an example.

4. AFCI OPTIONS

This chapter describes options for reactors, separation, and fuels. Appendix B contains various input parameters for the specific cases in this study. Appendix C contains the input and output fuel composition recipes.

The highlights of this chapter pertain to the fuels (section 4.3) and how they would be used in recycle systems. The multi-pass MOX approach in this study uses once-burned uranium in the feed and varies the Pu/U ratio each cycle to sustain recycling. This leads to high recirculating TRU flows, discussed further in Chapter 5. The multi-pass IMF approach in this study uses UOX for about 3/4 of the pins in blended cores; the other 1/4 of the pins are IMF. The recirculating flows are much lower. There are three consumer fast reactor (CFR) cases analyzed – symbiotic with used UOX, with used MOX (which itself comes from UOX), and with used IMF (which itself comes from UOX). The equilibrium cycles for these CFR cases differ as the continuing makeup from the thermal reactors varies. There are two breeder fast reactor (BFR) cases analyzed – UOX transitioning to BFR and UOX-IMF transitioning to BFR. The equilibrium BFR cycle is the same between these two cases, but the path toward equilibrium varies.

4.1. Reactors

We limit our analyses to three reactor types:

- TR - Light Water Reactors (LWR) as representative of all TR. These advanced LWR are assumed able to burn UOX, IMF, or MOX. Appendix A lists acronyms.
- CFR - Consumer Fast Reactors, assumed to have characteristics as a Sodium Fast Reactor (SFR) with conversion ratio of 0.25. These are therefore not the typical FR, but rather ones modified to deliberately burn more Pu-239 (and other fissiles) than they create. As such, CFR would tend to serve the functions of thermal reactor with IMF.
- BFR – Breeder Fast Reactors, assumed to have characteristics as a Sodium Fast Reactor (SFR) with conversion ratio approximately 1.1.

So, how robust are our conclusions for other reactor types?

There are two other thermal reactors considered in the GenIV program: Supercritical Water Reactor (SCWR) and Very High Temperature Reactor (VHTR). As shown in [Taiwo2005] the current GenIV reference design for the SCWR has little difference in fuel performance compared to conventional LWRs. However, the VHTR is designed for significantly higher burnup (100 GW-day/tonne) and higher thermal efficiency (48%). It was shown in [Taiwo 2005] that for the GenIV base VHTR design the TRU production rate is 45% lower than conventional LWRs. However, the resource utilization may be slightly worse than LWRs because of the high enrichment required to meet the VHTR design goals.

There are two other fast reactors considered in the GenIV program: Lead Fast Reactor (LFR) and Gas Fast Reactor (GFR). The base fuel cycle for all three GenIV fast reactor concepts is a closed fuel cycle using recycle transuranics and depleted uranium fuels. The GenIV reference designs for each concept are BFR, not CFR configurations. Although some variability is observed between the fuel performances in the three concepts [Taiwo 2005]; the basic transmutation performance is expected to be similar.

There is one final system, the Molten Salt Reactor (MSR), where the salt is also the fuel. This is a fundamentally different system in many respects. And, it can be configured as either thermal or fast, with varying degrees of difficulty. The MSR remains to be analyzed.

4.2. Separation

We limit our analysis to two separation technologies:
- UREX+ as representative of wet (aqueous) separation.
- Pyroprocessing as representative of dry separation.

These are the only two technologies with active research in the AFCI program.

We match UREX+ with recycling of TR fuels and pyroprocessing with recycling of FR fuels.

From the standpoint of our analyses, there are four major considerations in selecting among UREX+ and pyroprocessing.
- What type of fuel can be sent to each plant?
- What TRU are separated from each other?
- What is the loss rate per recycle?
- Is it envisioned to be at-reactor or centralized plant?

The UREX+ technology is, of course, designed for processing oxide fuels and therefore is the logical match to UOX or MOX. We assume it is also the match for recycling IMF. Therefore, we make the simplifying assumption that the same technology works with any of the TR fuels in our analysis.

When VHTR fuels are considered in future analysis, this must be re-visited.

There are four UREX+ variations considered in the AFCI program
- UREX+1 provides group separation of all TRU elements (Np, Pu, Am, Cm), e.g., appropriate for remote fabrication of GenIV fast reactor fuel.
- UREX+2 provides NpPu and AmCm as two products, primarily intended for recycling NpPu in TR and keep AmCm for later FR.
- UREX+3 provides Np, Pu, Am, and Cm as four products.
- UREX+4 provides NpPu, Am, and Cm as three products, primarily intended for recycling NpPu and Am in TR, and then sending the small Cm stream to disposal or keeping it for FR.

We therefore consider that
- UREX+1 provides the product IMF-NpPuAmCm and for the first cycle of FR fuel when TRU are shifted from TR to FR,
- UREX+2 provides the product for IMF-NpPu and MOX-NpPu, and
- UREX+4 provides the product for all other cases.

The cost differences among these three cases requires analysis.

The pyroprocessing technology is designed for metal fuel. We therefore consider it to be used to process FR fuel, for recycle back to FR.

A critical parameter in our analysis is the loss rate per recycle. We assume 0.2%/recycle (half during separation and half during fabrication). We use the same value for all of the UREX+ variants and for pyroprocessing. Further experimental work is required to be able to differentiate among separation technologies on the basis of loss rates. In Chapter 5, we show that the required loss rate varies among options; the higher the content of heat-generating isotopes in recycling fuel, the lower the allowable loss rates. For example, it appears that for BFR fuels at equilibrium, the required loss rate is about 1/3 of that for UOX.

4.3. Fuels

Table 4-1 lists the fuels we wished to include in this study. We were unable to obtain input/output fuel compositions for a few cases that we desired, so that the comparison analyses sometimes miss some entries.

Table 4-1. Fuels Considered for this Study

Fuel type	Burnup (GWth-day/tonne-HM)	Notes
UOX	33	Performed static but not DYMOND analyses
UOX	51	Baseline once-through
UOX	100	Performed static but not DYMOND analyses
MOX-NpPu	51	Single-pass only
MOX-NpPuAm	51	This is the only multi-pass MOX case. It increases the Pu/U ratio each cycle to keep the fuel burning.
MOX-NpPuAmCm	Not available	
MOX (blended core)	Not available	
IMF-NpPu (full core)	633	Single-pass only. Calculated as if the core was 100% IMF; in reality, these IMF pins would probably be distributed among several reactors.
IMF-NpPuAm (full core)	Not available	
IMF-NpPuAmCm (full core)	554	
IMF-NpPuAm (blended core)	65-58	These are blended assemblies with 60 IMF pins and 204 UOX pins per assembly. The burnup decreases each cycle in the current approach.
IMF-NpPu with separate Am targets (blended core)	66-58	These are blended assemblies with 4 Am targets, 60 IMF pins, and 200 UOX pins per assembly. The burnup decreases each cycle in the current approach.
UOX/CFR symbiosis	177	No recycling in TR.
IMF/CFR symbiosis	176	Three fuels required – UOX, IMF-NpPu, CFR
MOX/CFR symbiosis	128	Three fuels required – UOX, MOX-NpPu, CFR
UOX-to-BFR	66	No recycling in TR.
IMF-to-BFR	66	Three fuels required – UOX, IMF-NpPu, BFR
MOX-to-BFR	Not available	
U ore to BFR	Not available	A BFR can be started with enriched U235 rather than depend on the availability of Pu from thermal reactors.

Our analysis has made it clear that in defining any fuel scenario, there are three important parameters.
1. Blending within fuel assemblies or cores.
2. Ratio of Pu to U
3. U235 enrichment

There are two types of blending implicit in our fuel scenarios.

First is the blending of TRU (and sometimes uranium) to make new fuel. For example, the 1st-pass MOX-NpPuAm fuel composition requires the NpPuAm from 13.5 used UOX fuel assemblies. (Chapter 5

has more on such support ratios.) The multi-pass MOX recipes also require burned uranium (BU), but there is an order of magnitude more used burned uranium from separation of UOX than is used in making MOX. That is why the UOX separation approach must in all cases separate U from TRU; otherwise there is no way to get the right U/TRU mixtures. The basic concept for this approach to multi-pass MOX is shown in Figure 4-1.

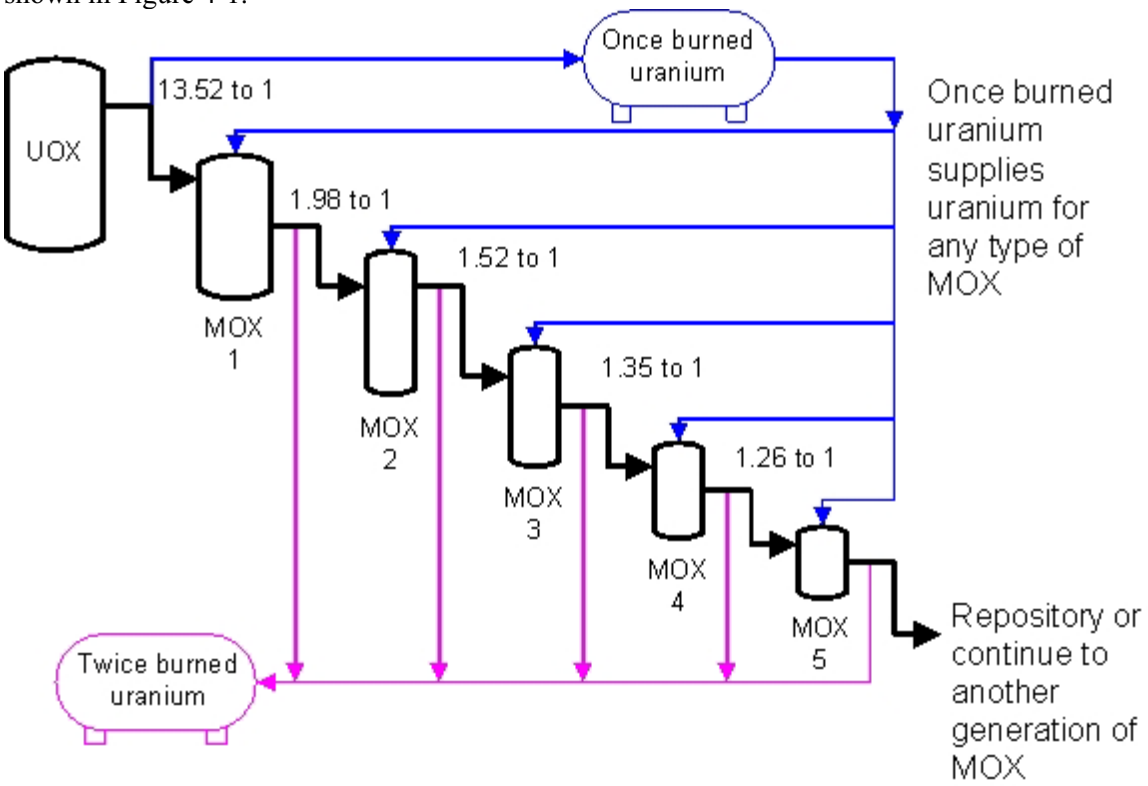

Figure 4-1. Multi-pass MOX approach used in this study

Figure 4-1 shows that once-burned uranium is an off-stream from separation of UOX; a few percent of this uranium is used as feed to the MOX cycles. Twice-burned uranium is an off-stream from separation of MOX; it is discarded. The support ratios each cycle vary; more details can be found elsewhere.[Wigeland2004a, Wigeland2004b]

The second is blending of different fuels in the blended core-IMF cases – fuel assemblies include both IMF pins and UOX pins. (In all other cases, the cores are considered homogeneous.) Table 4-2 provides the blending we used.[Goldmann2005] The intent was to take the remaining TRU from one generation to make the IMF for the next generation. That is, IMF in cycle N+1 is made from the TRU remaining in IMF cycle N plus the TRU in the preceding cycle's UOX. Figure 4-2 illustrates the concept. As the TRU from 264 UOX pins are used to make the IMF in 60 first cycle pins, we refer to the initial support ratio as 4.4 from the perspective of TRU mass. The support ratio is 1-1 from the perspective of reactor-to-reactor, i.e., 1 UOX reactor supports 1 reactor using the IMF-1 blend.

Table 4-2. IMF Blended Cores

	IMF-NpPuAm/UOX blends	IMF-NpPu/UOX/Am blends
Where is the Am?	In the IMF fuel	In separate Am targets
# UOX pins/assembly	204 (77.3%)	200 (75.8%)
# IMF pins/assembly	60 (22.7%)	60 (22.7%)
# Am pins/assembly	N/A	4 (1.5%)
U mass fraction in assembly (i.e. the heavy metal in UOX)	~98% (exact value depends on which IMF cycle)	~98%
TRU mass fraction in IMF pins	~2%	<2%
Am mass fraction in Am targets	N/A	~0.2%
Both cases also have 24 guide tubes and 1 instrument tube for a total of 289 pins (17 x 17)		

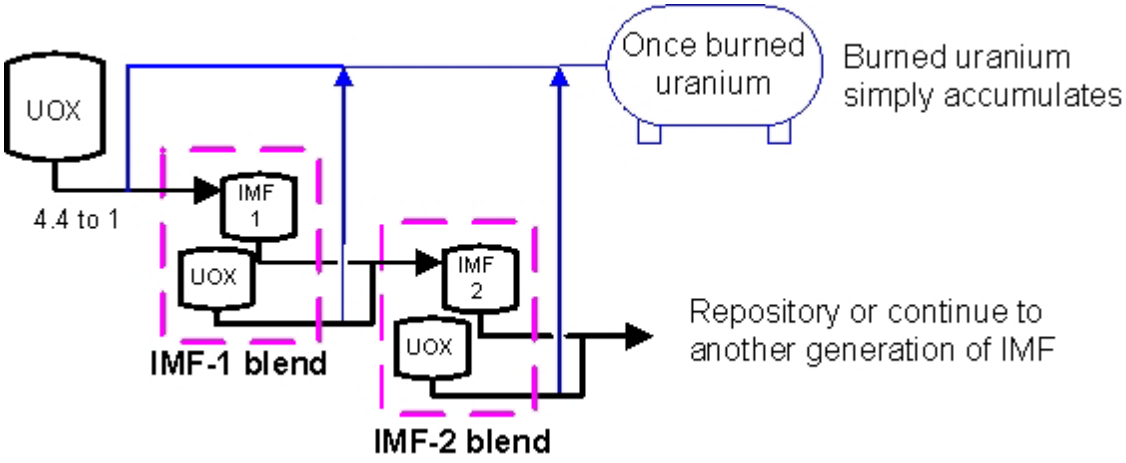

Figure 4-2. Multi-pass IMF approach used in this study

This approach to IMF is somewhat of a compromise between (a) IMF concepts that attempt to burn everything in a single pass (the single-pass cases in this study are good examples), leaving too much to dispose (to obtain high LTH, LTD, and LTR improvements) but too little fissile content to continue to burn and (b) the current MOX concept that has lower LTH, LTD, and LTR improvements.

In these blended cores, the asymmetry in mass between UOX and IMF is because uranium in UOX (or MOX) serves two functions – an "active" fuel ingredient and the matrix for the ceramic, but the matrix in IMF is some other "inactive" ingredient, such as $MgAl_2O_4$. That IMF matrix mass is not reflected in Table 4-2.

For any TR core, the fissile content must be sufficient (throughout fuel life) to keep k-eff>1. There are as many as 3 "knobs" as listed in Table 4-3 for a given target burnup.

Table 4-3. "Knobs" Available to Keep the Fuel Burning

	U235 enrichment	Ratio of Pu to U	Ratio of Am to Pu
UOX cores	Enrichment increased as desired burnup increases.	*Unavailable (no Pu in fresh fuel)*	*Unavailable (no Pu)*
MOX cores	Increase U235 enrichment to compensate for degrading Pu mixture. In current analysis, the U in MOX is always burned uranium (0.8% U235)	Adjust ratio of Pu to U in the MOX fuel composition.	In principle, could be used for cases where Am is recycled. In practice, this has not been done because in such cases the intent is to include as much Am as would be available.
IMF-full cores	*Unavailable (no U)*	*Unavailable (no U)*	
IMF blended cores	Increase U235 enrichment in UOX pins to compensate for degrading mixture in IMF pins. In current analysis, the UOX is always 4.3% enriched, the same as UOX in once-through (with 51 burnup)	Adjust ratio of IMF pins to UOX pins. This allows the composition of each fuel (UOX, IMF) to stay constant, only the number of pins changes.	

In UOX cores, of course, the U235 enrichment is set so that as the fuel burns there is always sufficient U235 (plus Pu239 bred *in-situ*) to keep the core going. One cannot independently adjust the Pu composition in UOX cores.

In MOX, there is an additional "knob", adjusting the ratio of Pu to U in the MOX. In this study, successive MOX cycles are kept fissile by adjusting Pu to U; the uranium in MOX is always burned uranium (0.8% U235). Thus, the n-pass MOX in this study does not use the U235 "knob".

In IMF blended cores, in principle there are also U235 and Pu/U ratio knobs, they are simply implemented differently than in MOX. E. Hoffman at ANL is exploring this option space.[Hoffman2005b] In this report, only the Pu/U "knob" is used; the uranium enrichment is always 4.3%, set the same as the UOX fuels in this study.

In IMF full cores, the only "knob" is how much fuel is put into the core. There is no uranium, hence no U235 nor Pu/U to adjust.

Figures 4-3 and 4-4 show how the composition of fuel changes by cycle for MOX and IMF. The MOX composition changes radically. Even the first cycle is 16% Pu, requiring 3 tonnes-Pu/yr per GWe. Since spent UOX has 0.22 tonnes-Pu/yr per GWe, the output of 13.5 UOX reactors is required for the first MOX recycle.

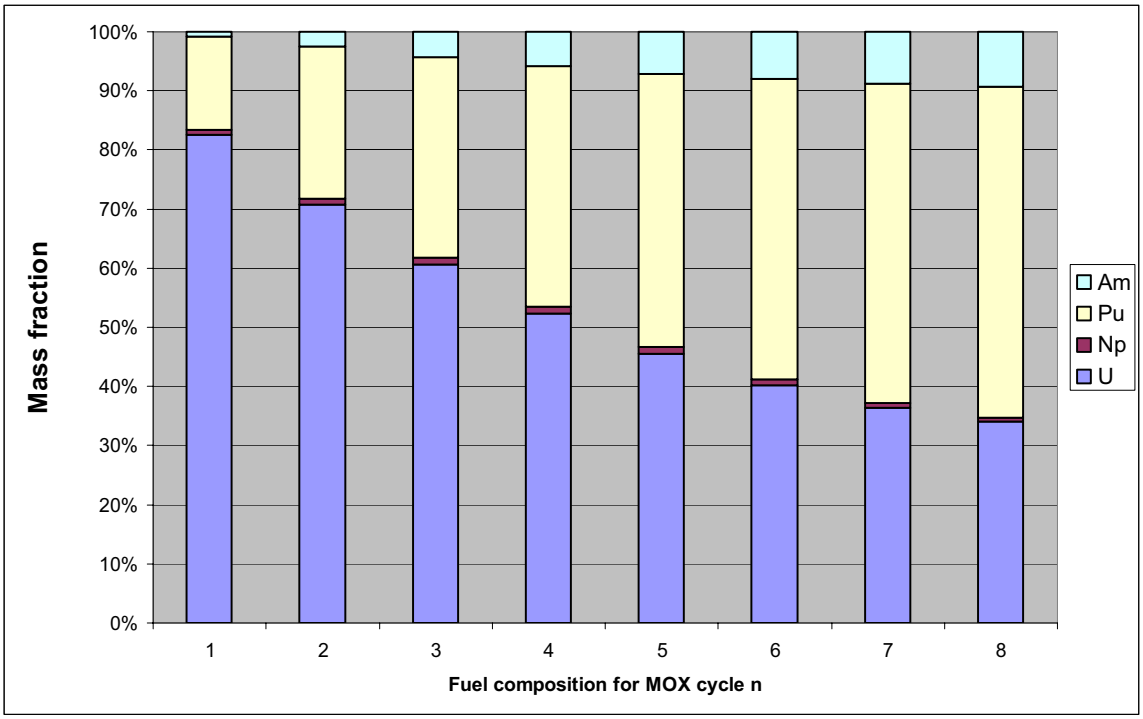

Figure 4-3. Elemental composition of full core MOX fuel by recycle generation. Generations 6-8 stem from extrapolations from earlier cycles; they are not results of transmutations calculations.

In contrast, Figure 4-4 shows that the composition of the blended IMF cores stays fairly constant.

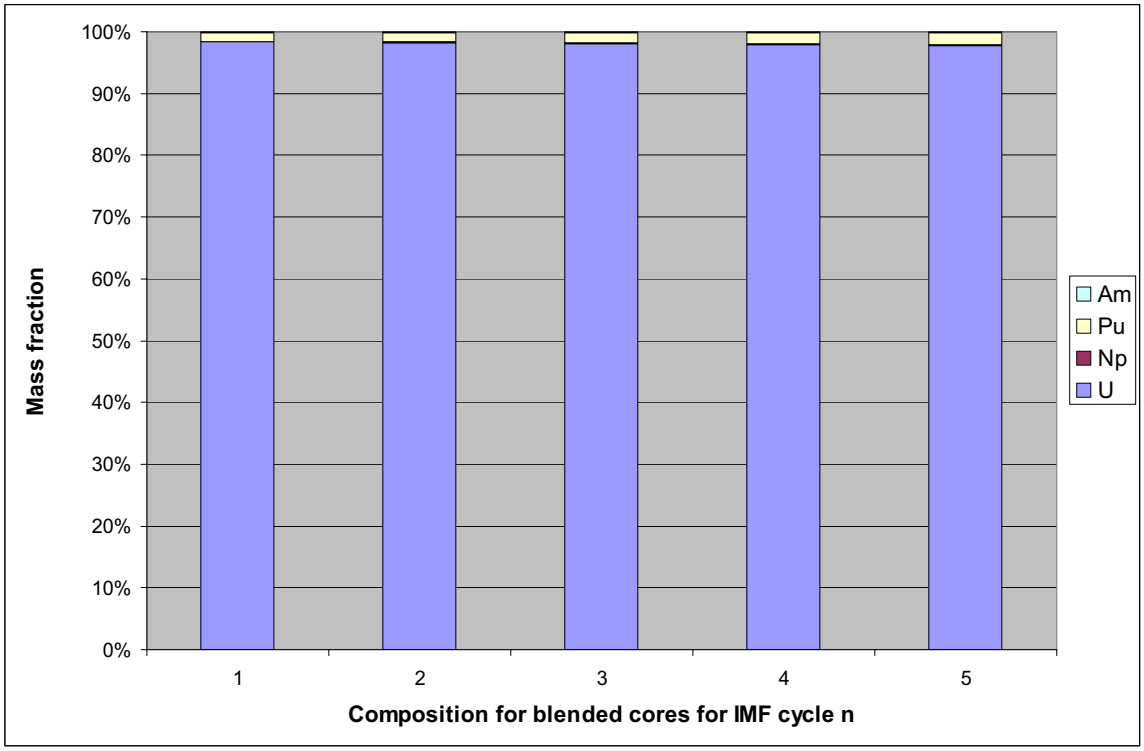

Figure 4-4. Elemental composition of blended core IMF fuel by recycle generation

The “knobs” in FR are different. It is still required to get each core started with fissiles. But, unlike TR, all U and TRU isotopes can contribute to having a favorable neutron balance. It is, for example, impossible to start a pure U238 or Pu238 core, even though both isotopes fission in a fast reactor.

We studied three types of consumer fast reactor systems, illustrated in Figures 4-5, 4-6, and 4-7. The fast reactor conversion ratios are approximately 0.25, thus considerable TRU must be supplied from thermal reactors each cycle. Unlike a pure-thermal system, however, the equilibrium composition in the FR will continue to burn. And, when recycling, the composition approaches equilibrium faster than in a thermal reactor. However, the equilibrium is not reached in just one or two cycles (see Chapter 5 for some discussion).

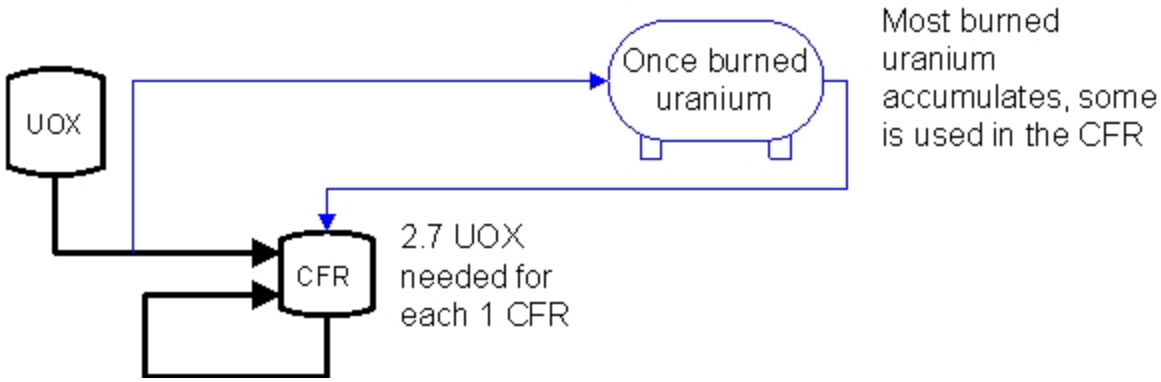

Figure 4-5. UOX- Consumer (Burner) Fast Reactor (CFR) symbiosis

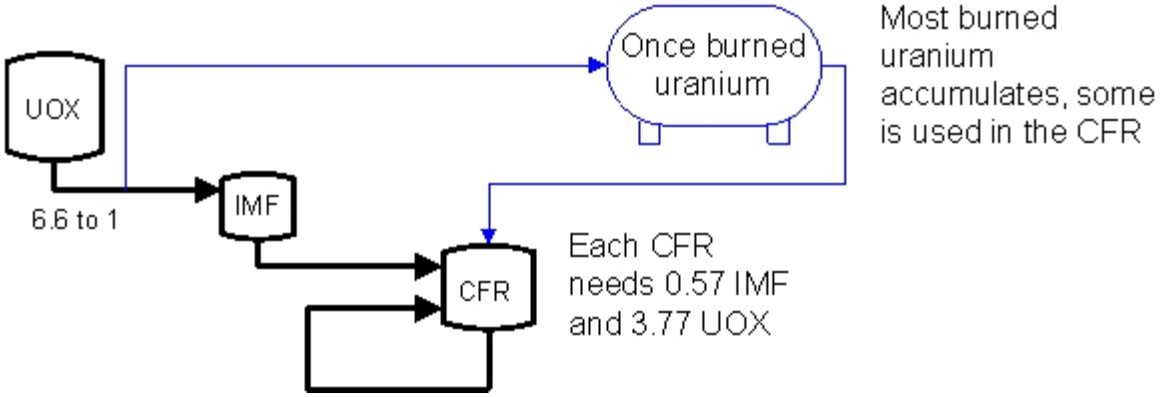

Figure 4-6. UOX-IMF- Consumer (Burner) Fast Reactor (CFR) symbiosis

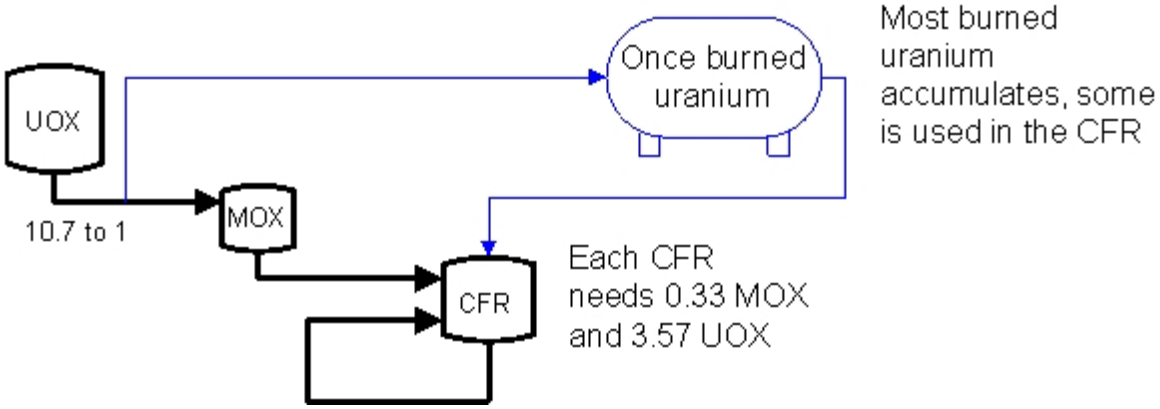

Figure 4-7. UOX-MOX-Consumer (Burner) Fast Reactor (CFR) symbiosis

All CFR cases, by definition, accumulate some burned uranium while using a minority of the BU from thermal reactors. In contrast, breeder fast reactors (BFR) use previously discarded uranium, either burned uranium or depleted uranium (DU). Figures 4-8 and 4-9 illustrate the two BFR cases in this study. There are two other BFR cases we were unable to study. One is the MOX analog to the IMF case, namely UOX-MOX-BFR. The other is starting BFR directly with enriched U235, bypassing the need for any thermal reactors. This would be the fallback position, for example, if thermal reactor recycling were too successful in drawing down Pu inventories so that a rapid BFR buildup was constrained by inadequate Pu stocks. This is explored in Chapter 8.

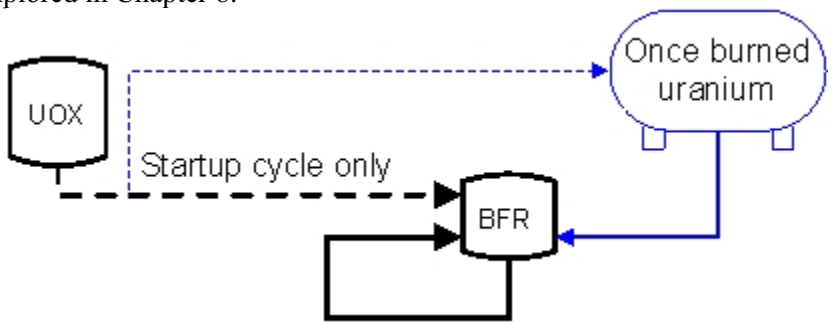

Figure 4-8. UOX-BFR case, UOX is only used for BFR startup, thereafter burned (shown) or depleted uranium (not shown) provides the input

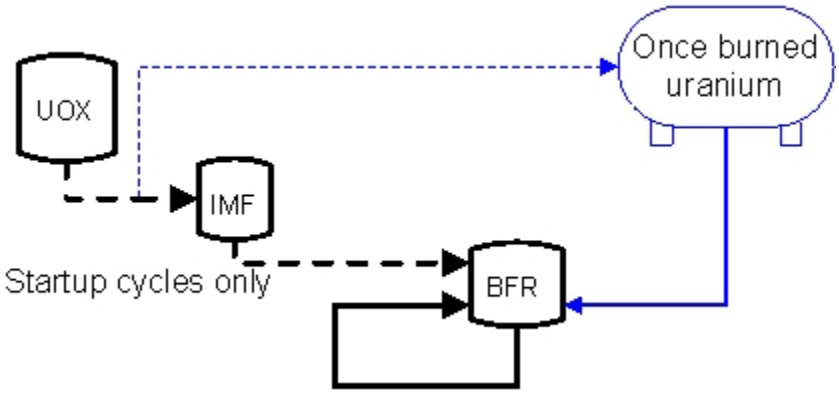

Figure 4-9. UOX-IMF-BFR case, UOX/IMF is only used for BFR startup, thereafter burned (shown) or depleted uranium (not shown) provides the input

5. "STATIC" ANALYSES

This Chapter contains "static" analyses using various TR and FR fuel compositions, which are described in Appendix C. These analyses have several purposes:

- Calculate parameters directly associated with individual fixed fuel compositions, e.g., the Pu239/Pu ratio for various fuels.
- Calculate idealized fuel throughputs and equilibrium mixes of reactor types, without constraints imposed by the need for building actual facilities (reactors, separation, fuel fabrication).
- Understand leverage points by comparing options.
- Inform the analysis of the dynamic DYMOND results.

The analyses are divided into five categories. The first pertain to mass throughputs and support ratios, which were introduced at the end of Chapter 4. The other four pertain to the four AFCI objectives.

Here are the highlights from this Chapter.

- Using Pu/U in MOX leads to varying compositions each cycle, with large recirculating Pu flows. This means that the separation plant used to process MOX will have to evolve as the cycles proceed.
- Fuel composition in IMF blends stays fairly constant. The recirculating Pu flows in multi-pass IMF are typically 1/2 to 1/3 of multi-pass MOX. The separation plant for IMF as the cycles proceed can stay constant, the elemental composition of the feed changes little. A similar approach for MOX blending may be able to improve the composition trends noted above.
- More than 80% of the fuel in the system at any given point in time (even at equilibrium) in multi-pass IMF or multi-pass MOX is still UOX. This means that the separation plants for thermal recycling primarily handle UOX, even with multi-pass MOX or IMF scenarios. When thermal recycling starts, the system is 100% UOX; this drops slowly to 80% as used UOX is processed and the TRU made into IMF or MOX. The 80% UOX value can only be temporarily reduced if separation and fuel processing is adequate to draw down the legacy UOX.
- More than 70% of the fuel in the system (at equilibrium) in thermal/fast-consumer symbiosis cases is UOX.
- To reduce uranium throughput, the only effective leverage is burnup. Recycling, per se, does not reduce uranium throughput.
- Can't reduce fission product throughput on a per GWe basis.
- To reduce plutonium throughput, use IMF. Avoid BFR. If one overburned Pu in 1-pass IMF, recycling cannot be sustained, the residual unburned TRU would eventually be discarded and waste management objectives not met. Backing off to blended multi-pass IMF avoids this problem for several decades, but not permanently.
- To reduce Am and Cm throughput, use BFR or IMF. From whatever the starting point following TR recycle, the higher actinide composition will slightly increase in the CFR.
- To reduce the required fraction of CFR in a thermal-CFR symbiotic system, use IMF. IMF-CFR requires 19% CFR. MOX-CFR requires 20% CFR. UOX-CFR requires 27% CFR. Starting recycling with IMF, therefore, reduces the need for CFR later.
- To reduce Pu inventory in the entire system, avoid once-through. UOX-51 makes Pu at 0.22 tonnes-Pu/yr per GWe. The next highest Pu production rate is the equilibrium BFR, 0.10 tonnes-Pu/yr per GWe. Thus, there is less Pu in a BFR system (with the design of a breeding ratio of 1.12) than in a once-through system! Of course, the Pu in the UOX-51 system is "self protecting" because it is unprocessed used fuel, whereas the Pu in the BFR system is recirculating with lower levels of self-protection.

- To reduce weapons-usable material, use single-pass full-core IMF. Avoid BFR and UOX-51. BFR cases have a slight increase in weapons-usable material; all pure-thermal-recycle and CFR cases have a net decrease in weapons-usable material.
- To reduce the amount of recirculating weapons-usable material, use IMF, avoid BFR. The flux of Pu239-equivalent/yr per GWe of fresh fuel is 0.26 for 5th pass IMF, 0.50 for 5th pass MOX, 0.36 for IMF-CFR, and 1.07 for BFR.
- To reduce Pu "quality" use IMF. Avoid BFR.
- If only looking at the Pu239 equivalent fraction in fresh fuel, avoid full-core IMF. The Pu239-equivalent fraction is 65-70%. Indeed, the fraction of Pu is over 90%. However, if one takes credit for the Pu "quality", the picture changes. The Pu239/Pu-total fraction for all the first pass fuels is the same as the UOX-51 output, 53%. For multi-pass IMF, multi-pass MOX, and CFR, the Pu239/Pu-total fraction steadily decreases. It increases for BFR cases, evolving toward an equilibrium value of 72%.
- Based on crude scaling analyses, we believe that MOX-NpPu and IMF-NpPu would require glovebox-fabrication, all other recycle fuels (which all contain Am) would require remote fabrication. The exception is part or all of the IMF blended cores. In multi-pass IMF-NpPuAm, the 204 UOX pins (of 264 total) would be hands-on; the 60 IMF-NpPuAm pins would be remote. The final assembly is probably remote, but more work is needed. In multi-pass IMF-NpPu with separate Am pins, the 200 UOX pins are again hands-on, the 60 IMF-NpPu pins would be glovebox (?), the 4 Am target pins would be remote, and more work would be required to know if the final assembly would be remote or glovebox.
- None of the single-pass-only systems come close to achieving program goals, see Table 5-1. They could be dropped from the program. (We did not study the VHTR.)
- Among multi-pass pure thermal systems, IMF appears superior to MOX, see Table 5-2. However, either only obtains ~17% improvement in uranium utilization and are therefore not sustainable from the uranium perspective in the long term. Although thermal recycling can be continued indefinitely, eventually sufficient TRU accumulates so that the TRU would be discarded with corresponding waste management penalties. However, this appears deferrable until the next century because 5 recycles appear practical. Building the infrastructure for multi-pass thermal recycling establishes some of the infrastructure for later fast reactors. And, using IMF would decrease the percent of CFR needed from 27% to 19%.
- CFR systems can be continued indefinitely; there is no need to discard TRU. Systems with 19% CFR, 11% IMF, and 71% UOX would be sustainable until uranium resources limit.
- BFR systems can be continued indefinitely and are the most sustainable. However, to preserve waste management and proliferation resistance benefits, one would not want to stop recycling without first transitioning back to CFR for several cycles. Even better would be rampdown with a combination of CFR and IMF.
- To increase the potential build rate of BFR, maximize its breeding ratio (esp. for early cycles), do not burn Pu239 in thermal reactors, and reprocess quickly to use fissile Pu241 in the BFR startup cycle. The ratio of output-fissile/input-fissile in this study are 1.06 (UOX-to-BFR startup cycle), 1.65 (UOX-to-IMF-to-BFR startup cycle), and 1.07 (BFR equilibrium cycle). The high ratio for the IMF-BFR startup cycle mitigates the burning of Pu239 in thermal reactors. (BFR equilibrium cycle is the same for UOX-to-BFR and UOX-to-IMF-to-BFR.)

Table 5-1. Key Results for Single-Pass Cases
(pink means option does not meet target, yellow means it partially meets target, green would have met it met target.)

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPu AmCm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Long-term heat (LTH) improvement	10x to 200x (to achieve actual repository improvements of 10-50x)	0.95	1.17	1.07	1.12	1.98	1.82	1.61	1.67
Long-term dose (LTD) improvement	10-50x	0.90	1.12	1.35	1.41	2.09	1.96	1.57	1.63
Long-term radiotoxicity (LTR) improvement	100x	0.89	1.38	1.12	1.18	2.46	2.39	1.79	1.85
Uranium ore use improvement	1.15 short term 50x long term	0.88	0.97	1.09	1.07	1.15	1.14	1.13	1.14
Is option sustainable per repository limits		NO							
Is option sustainable per uranium limits		NO							

Table 5-2. Key Results for Multi-Pass Cases. First number in each cell is the improvement factor this century (~5 cycles) if recycling stops. The second number is the improvement if recycling never stops (only feasible with fast reactors in the system).

	Improvement Targets (see Chapter 3)	Thermal recycling with MOX	Thermal recycling with IMF	Consumer fast reactor (CFR) with IMF thermal recycling	Breeder fast reactor (BFR)
Long-term heat (LTH) improvement	10x to 200x (to achieve actual repository improvements of 10-50x)	1.5x / Plateaus near this value	2.9x / Plateaus near this value	~4x / ~50x at 99.5% removal of TRU+Cs+Sr	~5x / ~70x at 99.5% removal of TRU+Cs+Sr
Long-term dose (LTD) improvement	10-50x reduction in peak dose, which is at 500,000 years after emplacement	1.9x / Plateaus near this value	3.0x / Plateaus near this value	~4x / ~60x at 99.5% removal of TRU+U+Tc+I	~7x / ~190x at 99.5% removal of TRU+U+Tc+I
Long-term radiotoxicity (LTR) improvement	100 reduction of radiotoxicity at 1000 years after discharge so that waste is less toxic than original uranium ore	1.9x / Plateaus near this value	3.2x / Plateaus near this value	~4x / ~100x at 99.5% removal of TRU+U+Tc+I	~7x / ~100x at 99.5% removal of TRU+U+Tc+I
Uranium ore use improvement	1.15 short term 50x long term	1.17x / ~1.2x	1.17x / ~1.2x	1.32x / 1.42x	2.0x / ~100x
Pu239 equivalent tonnes/yr per GWe for fresh fuel	As low as possible	0.50 / Slow increase	0.26 / Slow increase	Not estimated / 0.36	Not estimated / 1.07
Pu239/Pu-total in fresh fuel	As low as possible (value for discharged UOX-51 is 53%)	32% for 5th cycle fuel / Cannot drop much further	33% for 5th cycle fuel / Cannot drop much further	14% in CFR 53% in IMF / 14% in CFR 53% in IMF	72% / 72%
Avoid fully remote fuel fabrication	For as much fuel as possible	True for the 80% of the fuel that is UOX, untrue for MOX-NpPuAm itself	True for the ¾ UOX pins in blended assemblies, true for IMP-NpPu (with separate Am targets)	No	No
Minimize throughput of TRU (tonnes/yr per GWe)	As low as possible to minimize safety and economic issus	0.94 / Slowly increase	0.34 / Slowly increases	Not estimated / 0.85	Not estimated / 1.45
Percent fuel that is new	As low as possible	17%	17%	29%	100%
Percent of	As low as	Zero	Zero	19%	100%

reactors that are new	possible				
Is option sustainable per repository limits		NO, because unburned TRU must eventually be discarded, but probably after this century		Yes, unburned TRU does not ever have to be discarded, performance depends on loss rates	
Is option sustainable per uranium limits		NO			Yes
The LTH improvement factors for 5-cycles CFR and BFR come from Wigeland2004a as they use cycle-by-cycle compositions for those cases, which we did not use in this study. The 5-cycle value for MOX (1.5x) is the same in Wigeland2004a and here. The 5-cycle uranium improvement factors for BFR are probably understated, they depend on the breeding ratio (output-fissile/input-fissile) for the first few cycles, which was not optimized.					

5.1. Throughputs and Support Ratios

The results in later Chapters are difficult to understand without first examining throughputs and support ratios. Throughputs refer to the mass flux in and out of reactors, separation, or fuel fabrication plants. The support ratios refer to how many reactors (or pins) of one type of fuel are required to make the TRU required for the next generation.

We start with the support ratios of how much UOX is required (at equilibrium) to make IMF or MOX fuel with the compositions in this study. For example, Table 5-3 shows that 13.5 units of UOX are needed to make 1 unit of MOX-NpPuAm; this is caused by the UOX discharge rate of 0.22 tonnes-Pu/year versus the MOX-NpPuAm input requirement of 2.98 tonnes-Pu/year. (13.5=2.98/0.22) The support ratio for MOX-NpPuAm (13.5) is higher than for MOX-NpPu (10.7) because more Pu is needed in the former case to compensate for the Am in the fuel.

Table 5-3. Support Ratios for 1st pass in Thermal Reactors

Recipe used in this study	UOX to recycle fuel with NpPu	UOX to recycle fuel with NpPuAm	UOX to recycle fuel with NpPuAmCm
MOX-full core	10.7	13.5	Not calculated
IMF-full core	6.6	Not calculated	7.1
IMF-blended core	Not calculated	1.0 on basis of 1 reactor feeds 1 reactor. 4.4 from the perspective that 264 UOX pins make 60 IMF pins	Not calculated

For IMF-full core cases, the support ratios are lower than for analogous MOX-full core cases; there is no U238 in these cores; less Pu is needed. Again we see the trend that the support ratio for IMF-NpPuAmCm (7.1) is higher than for IMF-NpPu (6.6) because more Pu is needed in the former case to compensate for the Am and Cm in the fuel. For the IMF-blended core, the situation is fundamentally different. By design, the TRU from one UOX core is made into the IMF for the next core; hence the support ratio is 1.0. Indeed, the 1st pass of the IMF-blended core in this study still produces slightly more Pu239 than it consumes.

Table 5-4 shows more support ratios, this time for multi-pass cases. By design, the support ratio for the IMF-blended core case stays at 1.0. The support ratios for multi-pass MOX drop quickly from the UOX/MOX-1 support ratio (13.5) as MOX is itself recycled.

Table 5-4. Support Ratios for multi-pass in Thermal Reactors, support ratios for MOX 6-8 are extrapolated

	UOX to recycle 1	Recycle 1 to 2	Recycle 2 to 3	Recycle 3 to 4	Recycle 4 to 5	Recycle- 5 to 6	Recycle 6 to 7	Recycle 7 to 8
IMF (blended core)	1.0	1.0	1.0	1.0	1.0	Not calculated		
MOX (full core)	13.5	1.98	1.52	1.35	1.26	1.20	1.09	1.07

Figure 5-1 shows the equilibrium mix among fuels in a multi-pass MOX system. For a system of multi-passes, it is assumed that the Nth cycle is thrown away. For example, at equilibrium, in a MOX-1 system, 93.1% of the fuel is UOX and 6.9% is MOX-1. (7% = 1/(1+13.5)) The MOX-1 can either be thrown away, or made into MOX-2, in which case the equilibrium shifts to 90.0% UOX, 6.7% MOX-1, 3.4% MOX-2. The fraction of the preceding generations decreases slightly to "make room" for the newest generation, e.g., the MOX-1 fraction drops from 6.9% to 6.7%. Thus, the system is temporally stable as it proceeds to additional generations. Note at MOX-8, over 80% of the fuel in the system is still UOX.

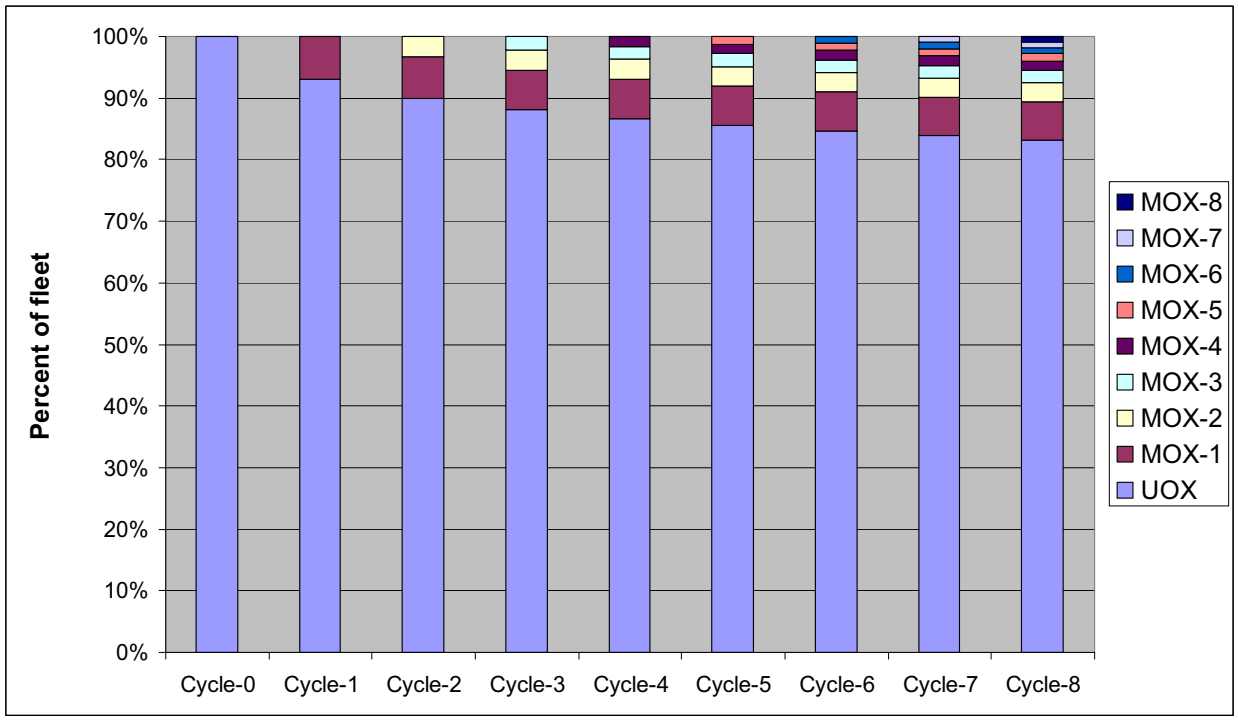

Figure 5-1. Equilibrium mix of fuels as a function of number of MOX recycle passes

Figure 5-2 is the same as Figure 5-1, except for IMF instead of MOX. Because the support ratios from each generation to the next is 1, the fleet is balanced (at equilibrium) among the blends available. For example, an equilibrium IMF-5 case would have 1/6 of the fleet each burning UOX, IMF-blend-1, etc. The numbers deviate slightly from the ideal because the burnup of successive IMF generations is reduced to keep the system going.

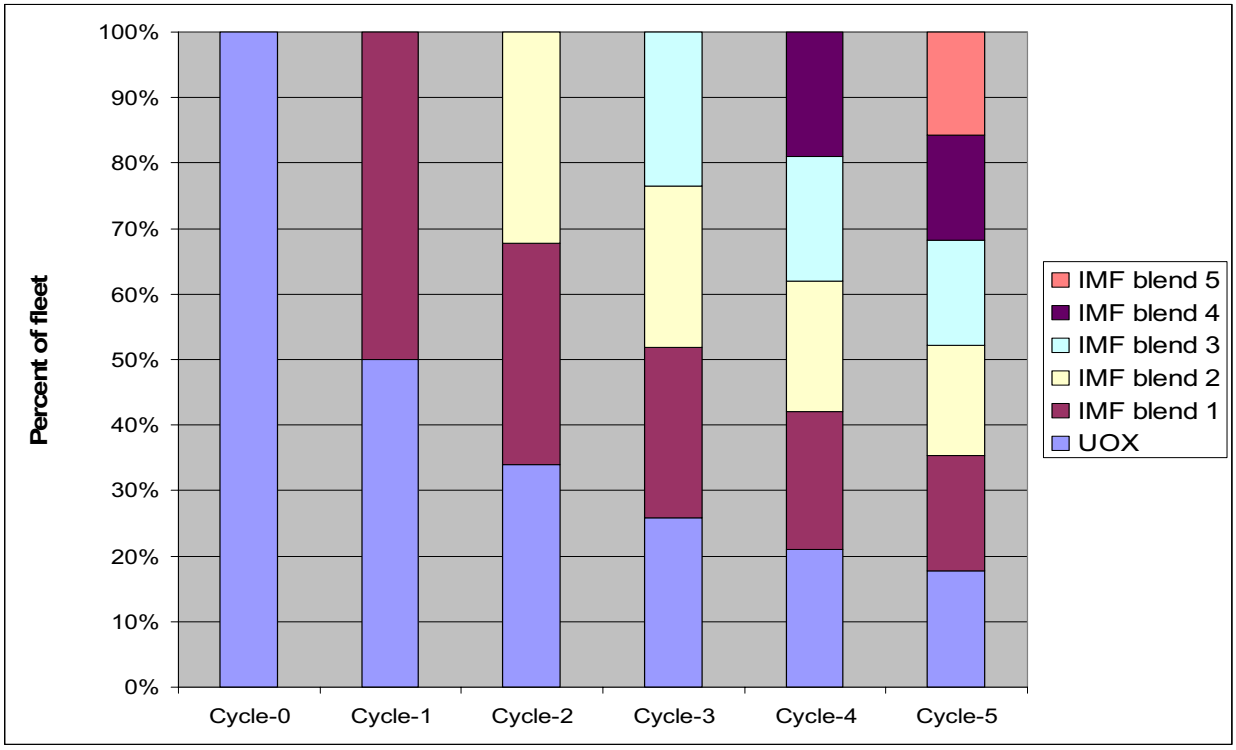

Figure 5-2. Equilibrium mix of fuel assemblies for different number of IMF recycle passes

There are two critical points to make about Figure 5-2.

First, the IMF system may not be temporally stable. If there is sufficient separation and fuel fabrication capacity, the entire fleet will progress through the cycles. Thus, at any given point in time, the entire fleet could be at cycle-4 if all reactors started recycling at the same time. In contrast, the MOX blending approach requires a mix of fuels at all times. Nonetheless, we still refer to Figure 5-2 as representing equilibria because over time, reactors would be at different generations. For example, in a 4-pass approach, overtime a "real" system would evolve so that roughly 1/5 of the reactors would be at the start (MOX), 1/5 each at IMF-blend-1 through 4. IMF-blend-4 would be discarded and those reactors would start over.

Second, Figure 5-2 would seem to imply that multi-pass IMF is dominated by IMF fuel, not UOX. However, this is not the case because each of the blends is itself dominated by UOX (204 pins out of 264). Figure 5-3 breaks down fuel usage accordingly. As in multi-pass MOX, we find that over 80% of the fuel in the system is UOX.

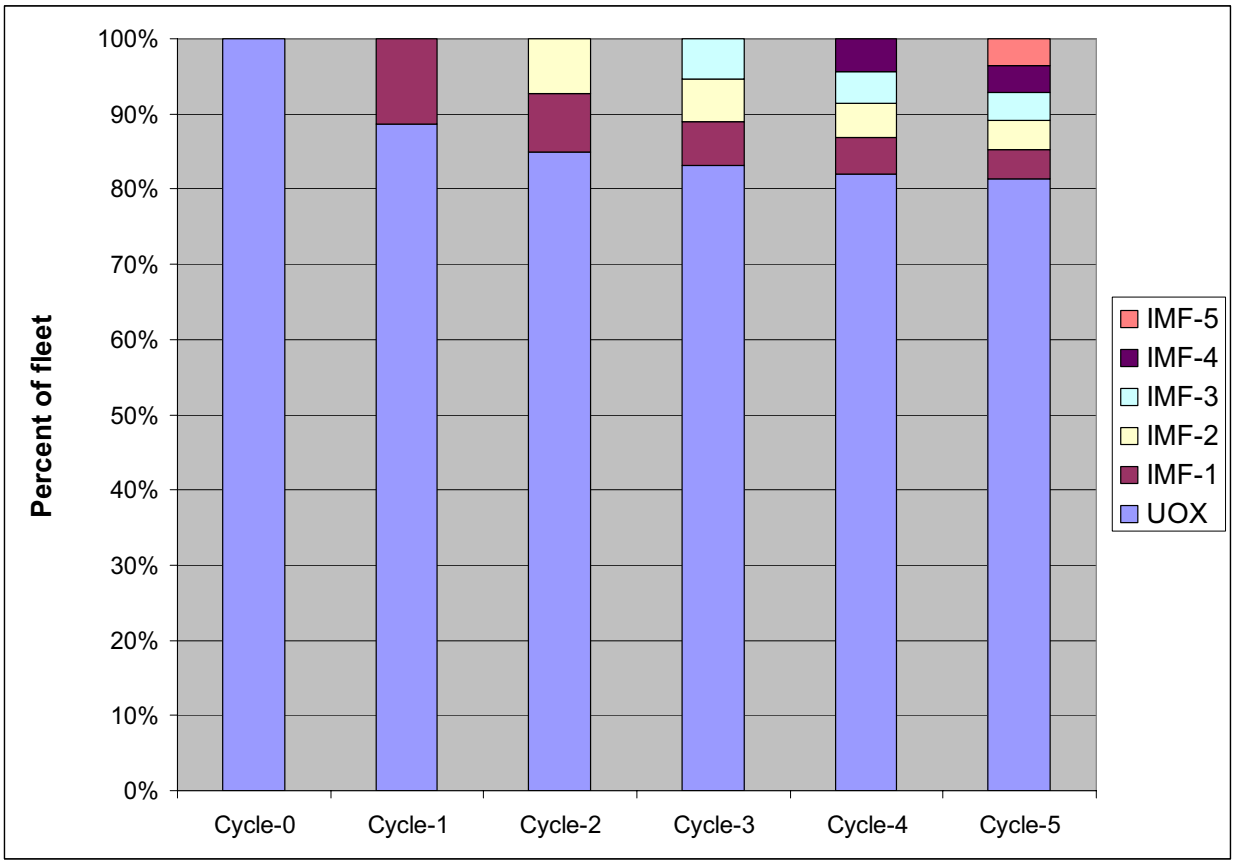

Figure 5-3. Equilibrium mix of fuel types for different number of IMF recycle passes

We now turn to fast reactor systems. The available transmutation analyses differ from those for thermal reactors. For thermal reactor systems, we have results generation by generation. For the fast reactor systems, we have only two results – "startup" and "equilibrium." The startup cycle takes only thermal reactor fuel as feed. The "equilibrium" cycle primarily uses recycled fast reactor fuel as feed, supplemented by thermal reactor discharged fuel only for CFR cases. The approximation is that the composition jumps immediately to the "equilibrium" composition after one startup cycle. This indeed is a good approximation for key isotopes such as Pu239. However, our analyses indicate that the approximation is poor for some isotopes that buildup slowly in fast reactors, such as Am242m. We therefore have to look at the generation-by-generation results for the fast reactor cases with caution.

Table 5-5 shows (on a per GWe basis) the Pu fluxes for the CFR and BFR cases, as well as the resulting support ratios. On a per-energy basis, to start up one CFR requires 8.9 UOX, 11.50 UOX (if first cycled through MOX-NpPu), and 42.5 UOX (if first cycled through IMF-NpPu). This is attributed to the factor that the IMF burns more Pu than MOX requiring additional spent fuel assemblies; in addition, the Pu vector recovered from the IMF is significantly denatured, requiring a higher fast reactor loading to achieve criticality.

Table 5-5. Support Ratios for Fast Reactor Cases

	UOX/CFR symbiosis	MOX/CFR symbiosis	IMF/CFR symbiosis	UOX to BFR	IMF to BFR
Required TRU for start-up cycle (tonnes-TRU/yr per GWe)	2.19	3.34	4.03	1.52	3.06
TRU feed from source for the start-up cycle (tonnes-TRU/yr per GWe)	0.246	2.23	0.63	0.246	0.63
Support ratio for start-up cycle	8.9x	1.5x from MOX, 16.1x from the UOX that went into the MOX	6.4x from IMF, 41.8x from the UOX that went into the IMF	6.17x	4.8x from the IMF, 31.8x from the UOX that went into the IMF
Required TRU for equilibrium cycle (tonnes-TRU/yr per GWe)	2.77	3.52	3.68	1.34	1.34
TRU feed from source for the equilibrium cycle (tonnes-TRU/yr per GWe)	2.11 from CFR 0.66 from UOX	2.77 from CFR 0.74 from MOX	3.32 from CFR 0.36 from IMF	1.34 + 0.11 excess	1.34 + 0.11 excess
Support ratio for equilibrium cycle	2.66x UOX	0.33 MOX 3.57 UOX	0.57 IMF 3.77 UOX	N/A	N/A

Figures 5-4, 5-5, 5-6 show the hypothetical evolution of a fleet for the CFR cases. The concept is the same as for the multi-pass IMF and MOX cases above, except here we make the approximation that the composition for the first CFR is the "startup" composition and for all later cycles is the "equilibrium" composition. We divide the total "equilibrium" portion of the fleet into the "last" cycle and the intermediate cycles. If a policy of N-cycles and then throw away were adopted, the "last" cycle would be the one thrown away. In each case, the systems are evolving toward an equilibrium including UOX, IMF or MOX, and CFR. For the major transuranic isotopes (e.g., Pu-239, Pu-240), the transition from fast reactor startup to equilibrium will be rapid; however, slow build-up has been observed for lower concentration isotopes that have equilibrium concentrations significantly higher than the initial LWR feed materials (e.g., Am-242m).

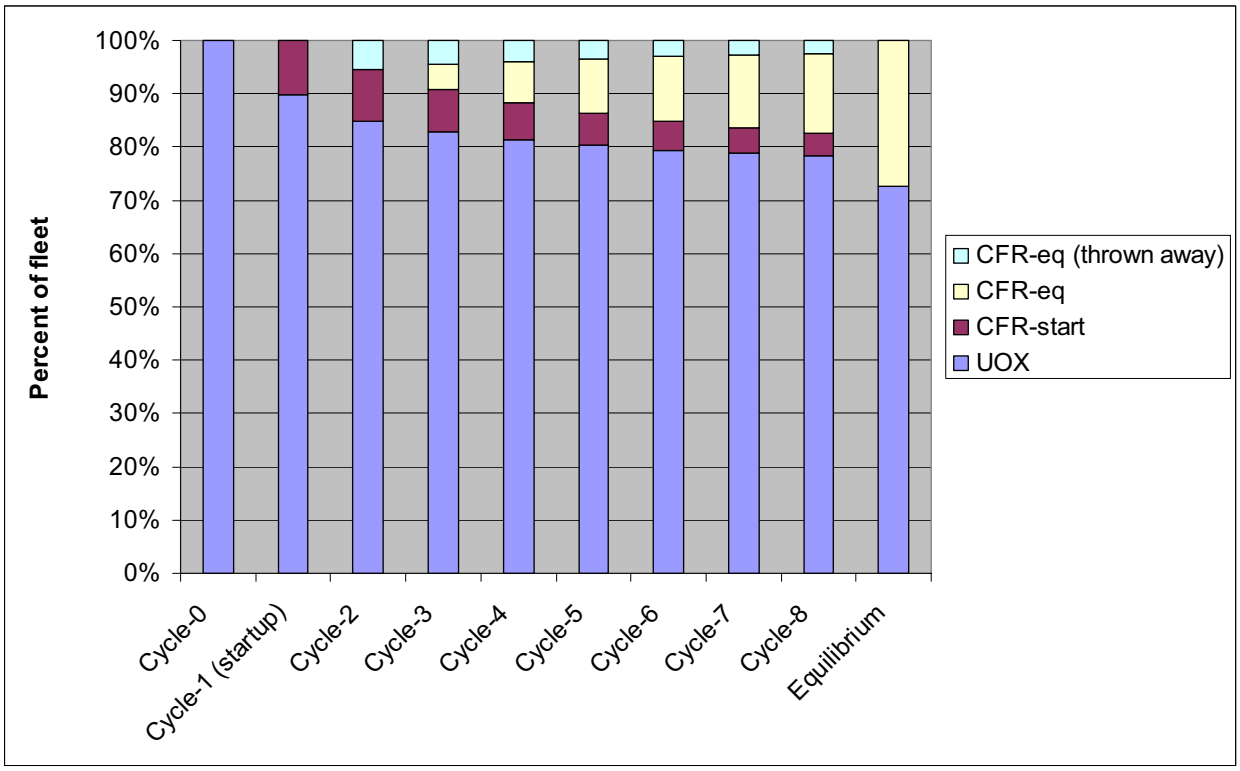

Figure 5-4. Hypothetical evolution of fleet for UOX-CFR symbiosis

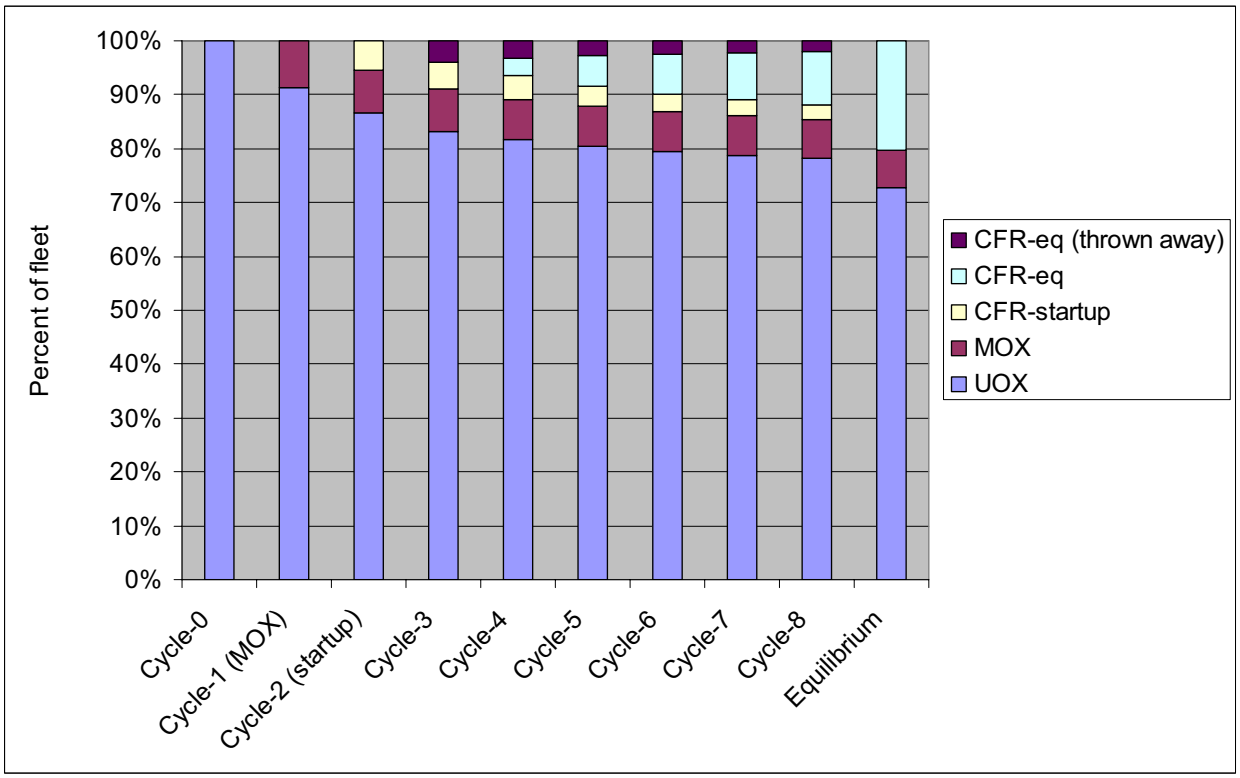

Figure 5-5. Hypothetical evolution of fleet for UOX-MOX-CFR symbiosis

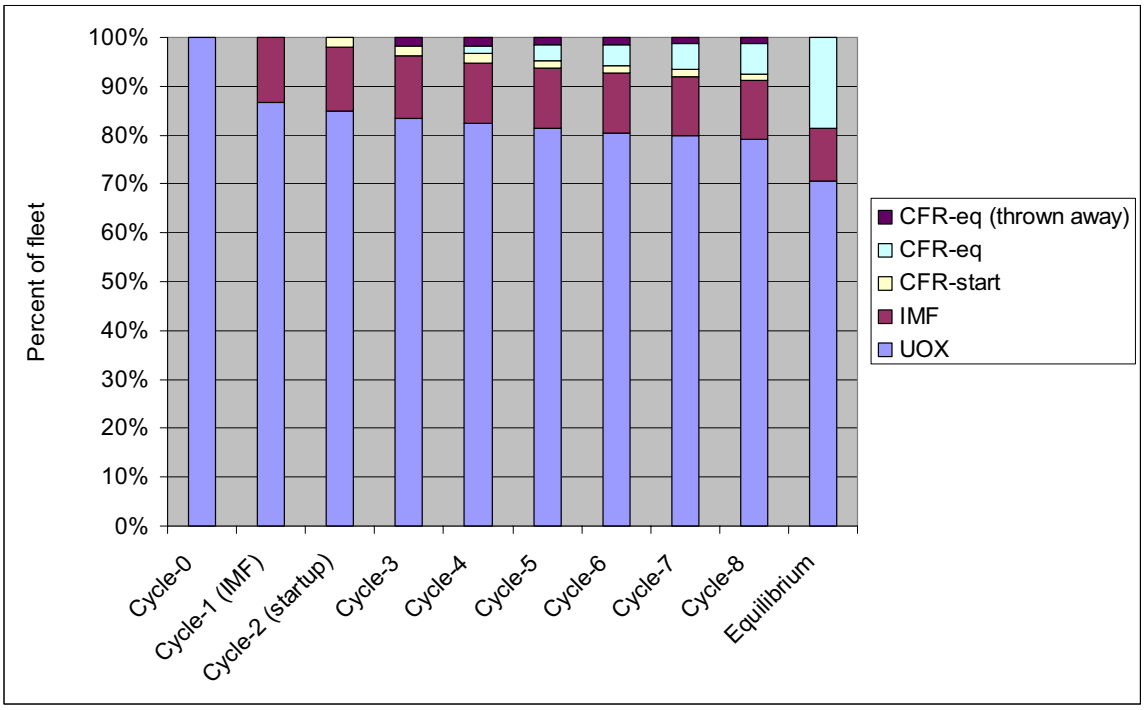

Figure 5-6. Hypothetical evolution of fleet for UOX-IMF-CFR symbiosis

Figure 5-7 compares the three equilibria. For both UOX-CFR and UOX-MOX-CFR, the equilibrium CFR portion of the fleet is 27%. Including MOX in the system basically only displaces some of the UOX. For UOX-IMF-CFR, however, the equilibrium CFR portion is only 17%. IMF displaces CFR, which is unsurprising as both are designed to primarily burn Pu. This option is therefore preferred if one is concerned with the possible high cost of CFR relative to thermal reactors. Again, the pure-thermal systems can continue indefinitely, but after some number of cycles, unburned TRU are discarded.

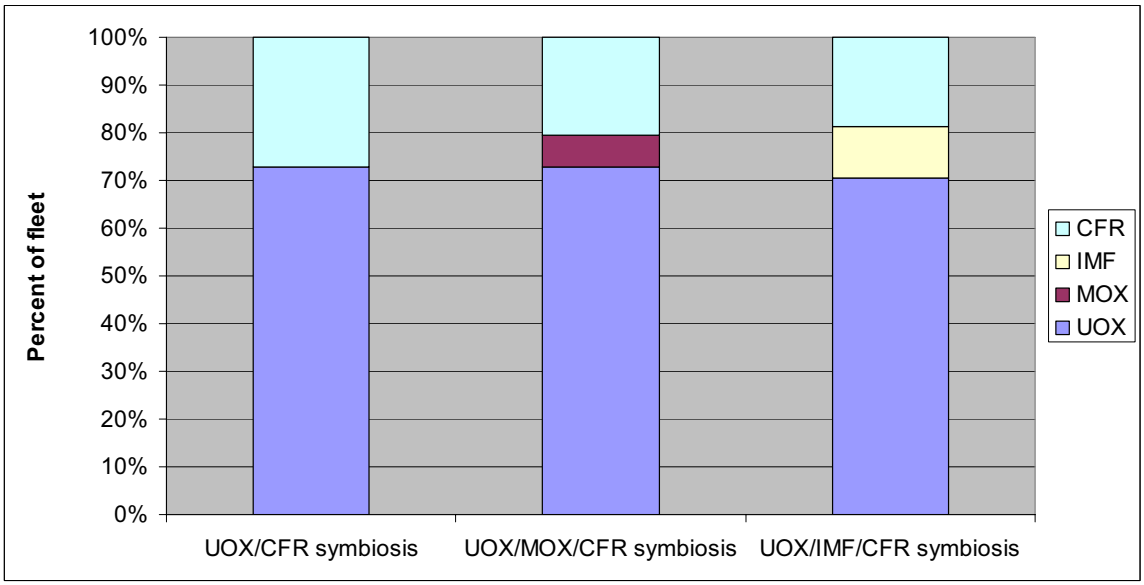

Figure 5-7. Equilibrium fleet mixes for CFR symbiotic cases

Of course, none of the pure-thermal cases nor thermal/CFR symbiotic cases avoid the need for continuing U235 consumption. Therefore, none are sustainable if uranium resources become a constraint. If so, the only options are BFRs.

Figures 5-8 and 5-9 show the evolution of BFR systems for UOX-BFR and UOX-IMF-BFR respectively. We again caution that these are suspect because of the "startup/equilibrium" approximation in the transmutation calculations. Both systems are evolving toward 100% BFR, but it will take awhile if these calculations are accurate. Of particular note is that the UOX-IMF-BFR case proceeds towards 100% BFR at about half the rate as UOX-BFR; of course, by design, the IMF is burning Pu. The UOX-IMF-BFR case is not a logical one for the long term, but it provides us some glimpse as to what happens if UOX-IMF runs for a few decades, reducing Pu stocks, and then BFR are introduced.

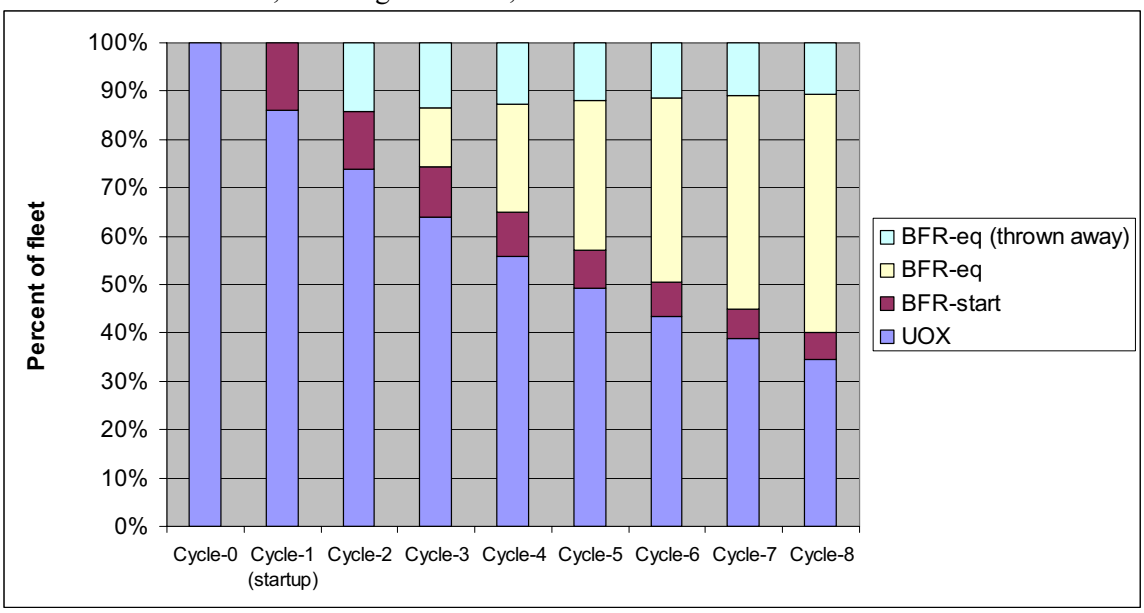

Figure 5-8. Hypothetical evolution of fleet for UOX-BFR

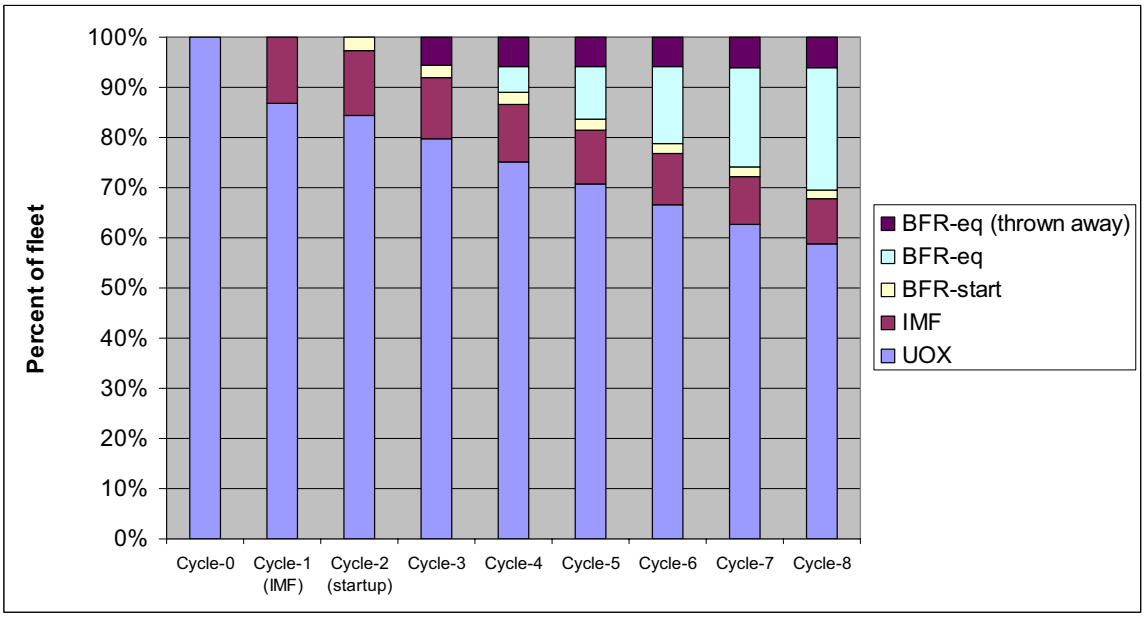

Figure 5-9. Hypothetical evolution of fleet for UOX-IMF-BFR

We recognize that conversion ratio can vary anywhere between 0.25 (CFR in this study) and 1.1 (BFR in this study. We therefore estimated how some key parameters would vary as a function of CR.

The **equilibrium** fraction of the reactor fleet that must be fast reactors is a strong function of the FR conversion ratio. With UOX feeding consumer fast reactors, about 27% of the fleet must be FR at conversion ratio of 0.25. Figure 5-10 shows that the required fraction of fast reactors increases sharply at the FR conversion ratio approaches 1.

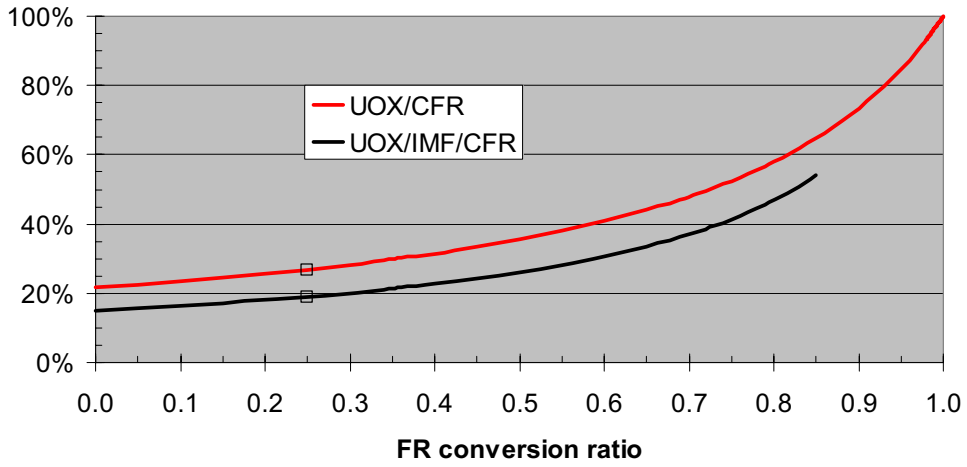

Figure 5-10. Required percent of fast reactors at equilibrium (FR/(FR+TR)) as a function of fast reactor (FR) conversion ratio. Red line shows UOX with consumer fast reactors. The black line shows UOX with 1 recycle of IMF in thermal reactors followed by consumer fast reactors (CFR).

If 1 pass recycle in thermal reactors with IMF is added to CFR, the required fraction of fast reactors is decreased, as shown in the curve above, because the IMF destroys some of the TRU that would otherwise have to be destroyed by the fast burner reactors. Additional recycles in thermal reactors would further lower the required fraction of fast reactors. However, the benefit of each additional recycle in a thermal reactor drops, e.g., the value of a second recycle is less than the first. And, although it is possible to indefinitely recycle in a fast reactor (the equilibrium concentration still works as fuel), it is not possible to indefinitely recycle in thermal reactors. In thermal reactors, eventually, the TRU mix becomes unusable and must be discarded. The unusable equilibrium is more unusable and is approached faster the more TRU isotopes are recycled. Thus, as a backup to the CFR case (only recycle in fast reactors), if we recycle Np-Pu-Am in thermal reactors, FY2005 analyses show that it is possible to get at least 5 cycles in thermal reactors. However, only about 2 cycles in thermal reactors if recycle Np-Pu-Am-Cm.

Figure 5-11 shows the uranium utilization improvement as a function of FR conversion ratio. As conversion ratio increases toward 1, the uranium utilization improvement (relative to once through) increases to about 160x, relative to once-through fuel cycle at 51 MW-day/kg-HM burnup.

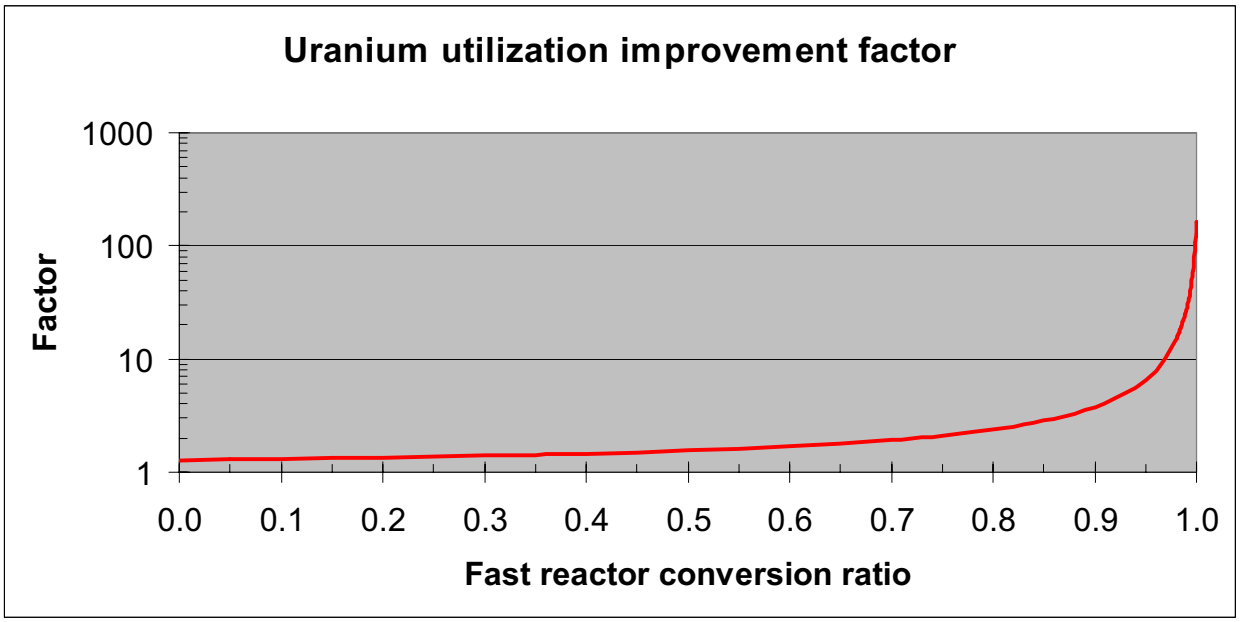

Figure 5-11. Uranium utilization improvement versus fast reactor conversion ratio

Figure 5-12 shows uranium flows as a function of FR conversion ratio. Here, we assume that discharged uranium is used as the source of uranium for the FR. A LWR at 51 MW-day/kg-HM burnup uses 19 MT-enriched uranium/year per GWe at 4.3% enrichment. This is 149 MT-uranium-ore/year per GWe, if the tailings are 0.2% U235. Consistent with figure 5-11, that number changes little until the FR conversion ratio approaches 1.0. At FR with CR=0.25, we find approximately …

109 MT/yr per GWe – Uranium ore needed
95 MT/yr per GWe - Enrichment tailings (unused)
13 MT/yr per GWe - Unused "burned" uranium discharged from the LWR portion of the fleet (73% of fleet)
0.9 MT/yr per GWe – Uranium consumed in LWR portion of fleet
0.05 MT/yr per GWe – Uranium consumed in FR portion of the fleet (27%)

Thus, little of the original uranium ore is used in the FR portion of the fleet. The uranium utilization factor is about 1.38, since 109 is a factor of 1.38 lower (better) than 149 MW-uranium-ore/year needed for once-through LWRs.

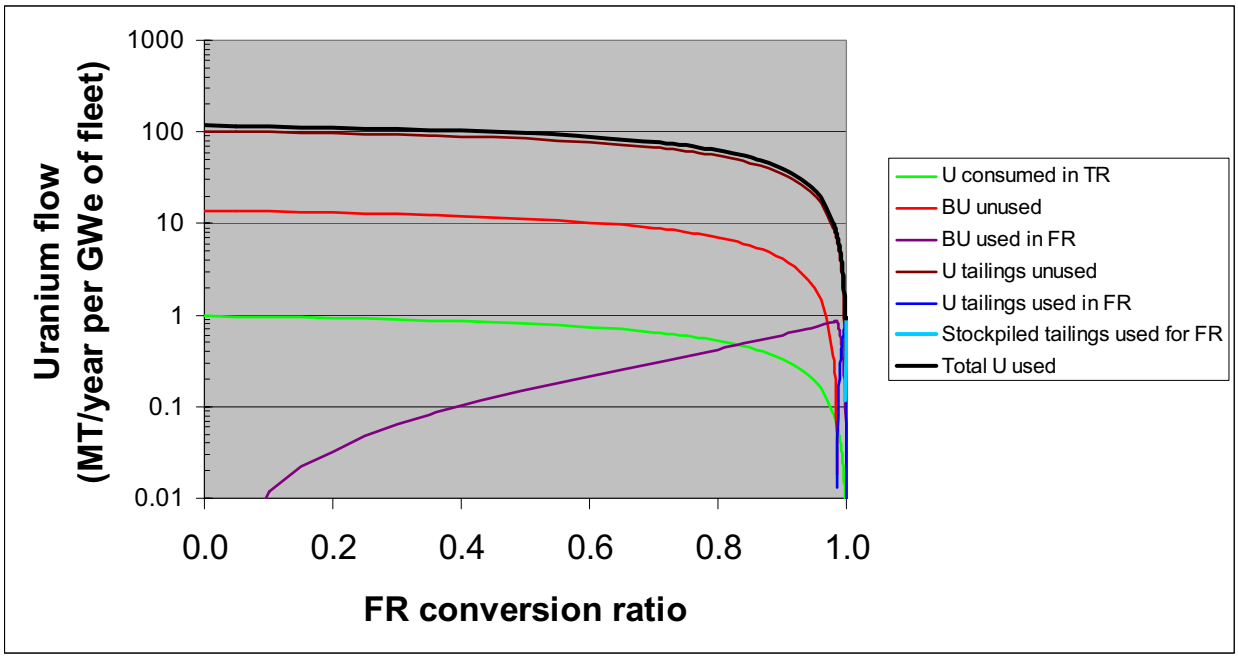

Figure 5-12. Uranium supply needed as function of FR conversion ratio. BU = burned uranium, the uranium discharged from thermal reactors (TR).

Figure 5-13 shows the uranium flows in the range of conversion ratio from 0.95 to 1.00 to show more detail as the conversion ratio approaches 1. At CR=0.986, all of the burned uranium discharged from LWRs is required to make fuel for the FR portion of the fleet. At this conversion ratio, 5% of the fleet is LWR, 95% is fast reactors. The total uranium ore required is only 7.2 MT/year per GWe, a factor of 20 improvement versus the once-through fuel cycle. Only above CR=0.986 are any of the uranium enrichment tailings needed.

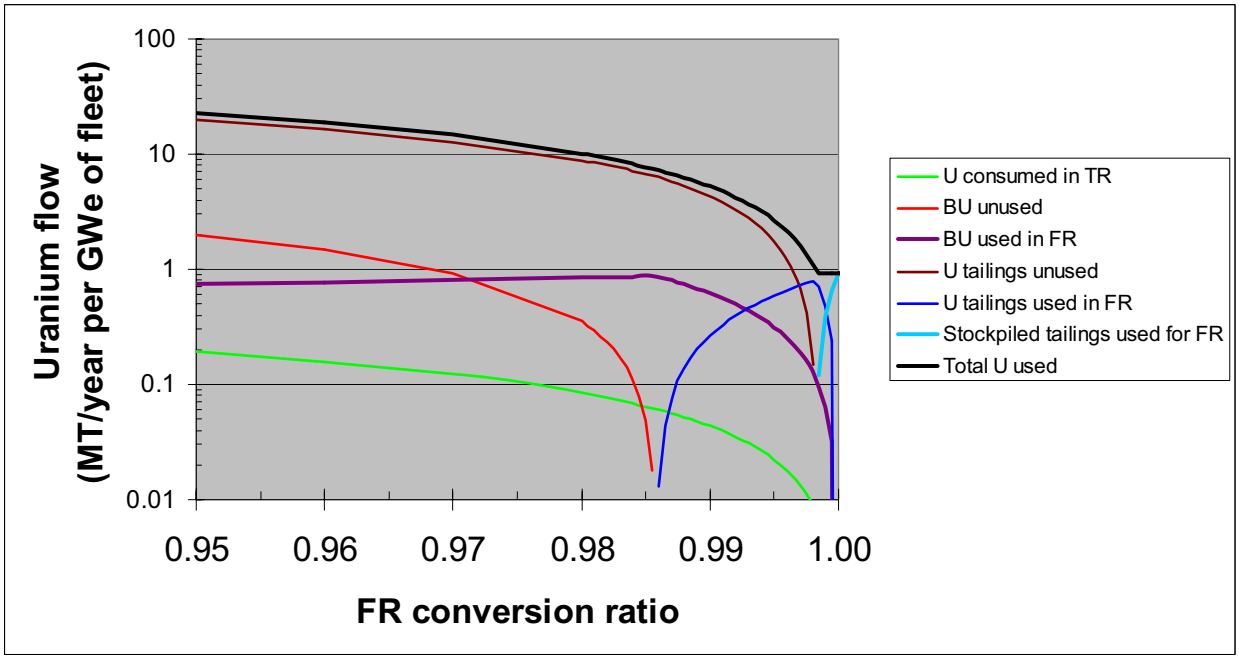

Figure 5-13. Uranium supply needed as function of FR conversion ratio. BU = burned uranium, the uranium discharged from thermal reactors (TR).

At CR=0.9985, only 0.5% of the fleet is LWR, 99.5% is fast reactors. At this conversion ratio, the uranium enrichment tailings (for the current LWRs) are no longer adequate to supply uranium for the fast reactors. Previously stockpiled uranium tailings (burned or depleted) would be required for part of the fast reactors. The uranium improvement factor at CR=0.9985 is about 90.

Having established the support ratios and the percent of the fleet burning each type of fuel, we turn to mass throughput.

Figure 5-14 shows the throughput (tonnes/year) of discharge fuel to separation plants for the various cases studied. The mass is dominated in all cases by uranium, consistent with the observations above that UOX remains the dominant fuel in the system, except in the BFR cases. The uranium in the BFR cases, of course, provides fertile U238.

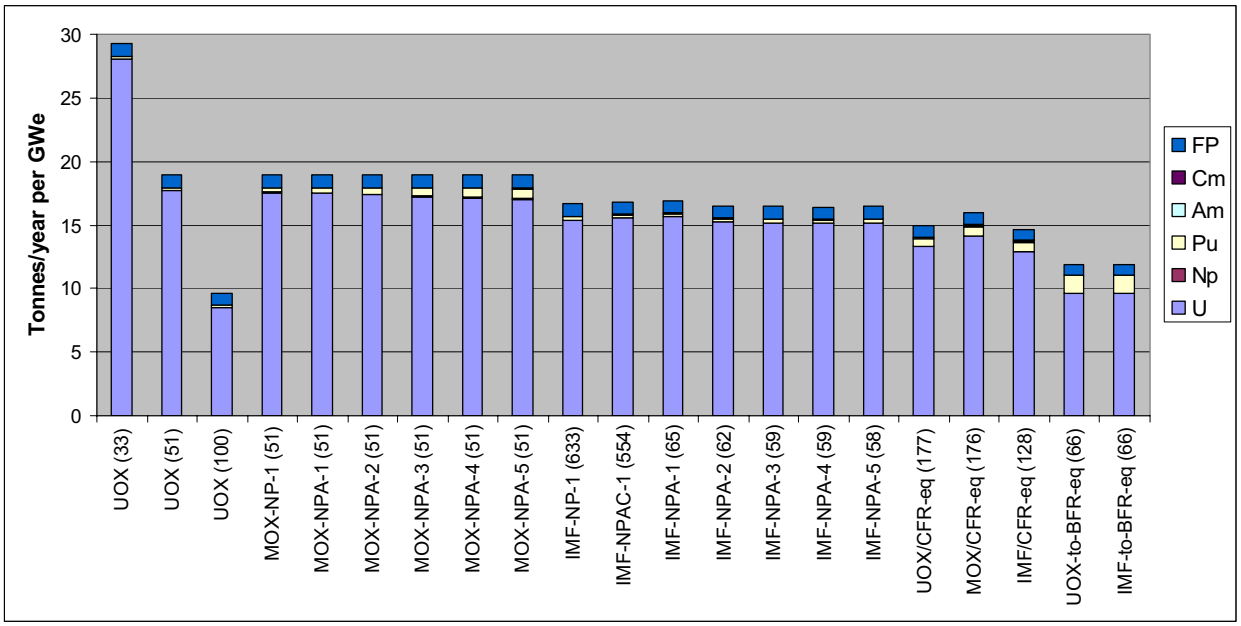

Figure 5-14. Equilibrium Mass Throughputs (numbers in the names of cases refer to burnup)

Recall that each case includes all the types of fuels involved. For example, IMF/CFR-equilibrium includes the UOX, the IMF, and the CFR output, normalized to 1 GWe.

The throughputs are in the range of 15 to 19 tonnes/year with only three exceptions: low UOX burnup (33), ultra high UOX burnup (100) and BFR. The various IMF cases show somewhat lower throughputs than the analogous MOX cases. With MOX, there is uranium coming from both used UOX and from used MOX; uranium continues to cycle around the system. With IMF, there is only uranium coming from used UOX; uranium does not continue to cycle around the system. The best way to reduce uranium throughput is ultra-high burnup.

The next several graphs divide the values in Figure 5-14 by element to better understand trends. Figure 5-15 removes uranium.

Figure 5-15 shows just the fission product throughputs. These are constant among the pure thermal cases. There is a slight reduction for the CFR and, even more, for BFR. This is presumably due to the increase in thermal efficiency for the fast reactor. Indeed, the fission product yield for U235 and Pu239 differ

(Figure 5-16). Recall that all discharge masses in this study reflect 5 years of decay after reactor discharge.

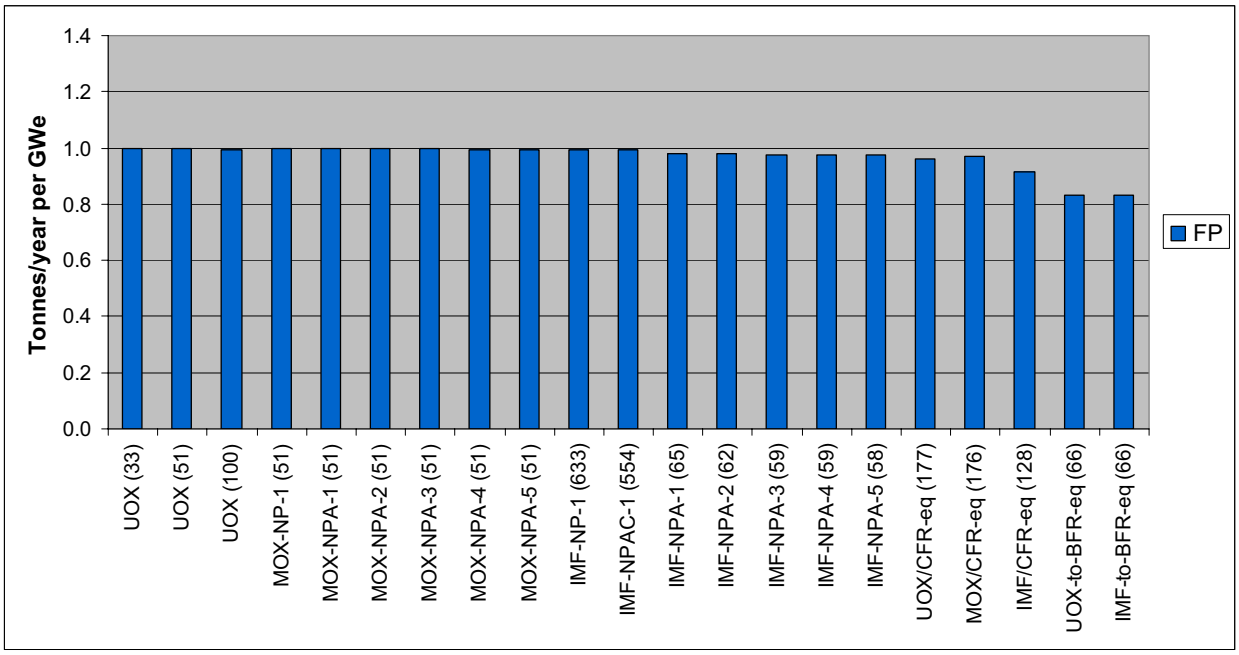

Figure 5-15. Equilibrium Mass Throughputs, Fission Products Only

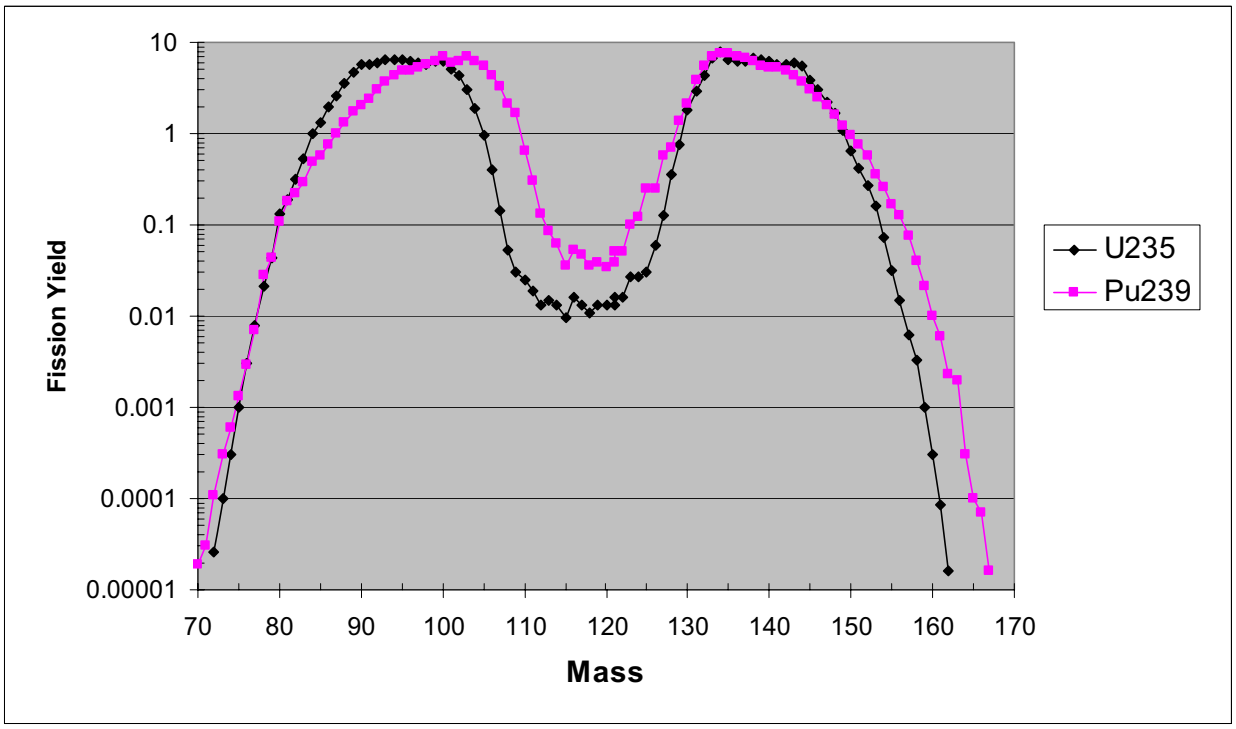

Figure 5-16. Fission yields for U235 and Pu239, data from [Walker1989]

Figure 5-17 shows just the TRU throughputs for the same cases. The lowest TRU throughputs are for UOX and IMF. The highest is for BFR. The MOX and CFR cases have significant recycling TRU flows

because none of them are as efficient in consuming TRU as does IMF. By continuing to recycle the TRU (multi-pass MOX, CFR symbiotic cases), the Am accumulates together with the Pu.

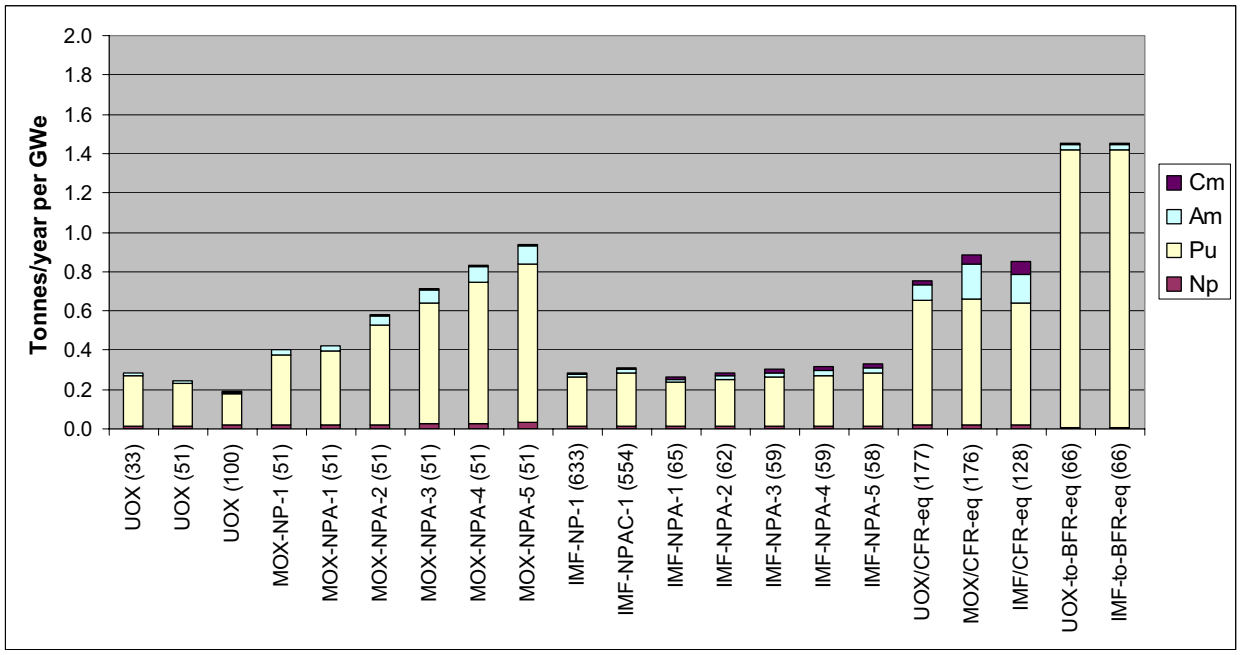

Figure 5-17. Equilibrium Mass Throughputs, TRU elements only

Finally Figure 5-18 focuses on the "hottest to handle" elements, Am and Cm. Consider multi-pass MOX versus IMF. Multi-pass MOX accumulates more Am each pass. The Am is not burning very efficiently. Am must be burning somewhat better in the IMF calculations, but there is a significant accumulation of Cm. Note that the CFR cases have the highest Am and Cm throughputs, because of the Am and Cm coming from the feed. The equilibrium BFR cases contain significantly less Am and Cm because the intermediate isotopes fission in the fast flux.

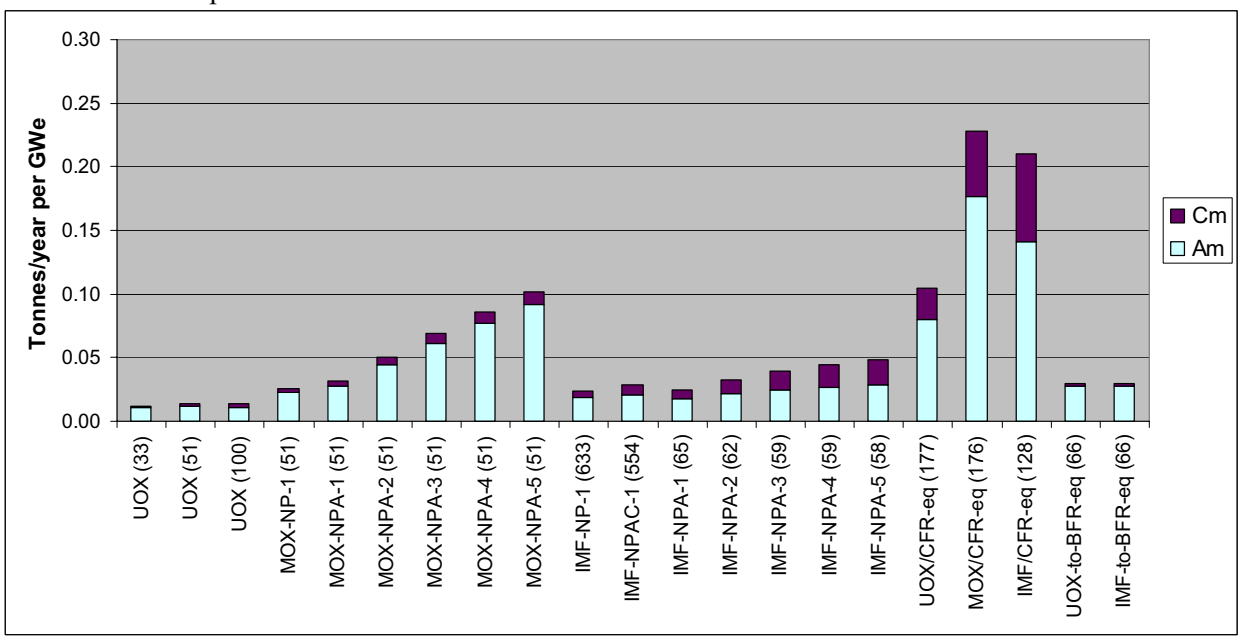

Figure 5-18. Equilibrium Mass Throughputs, Am and Cm only

The cost implications of these differing mass throughputs requires analysis.

5.2. Waste Management Calculations

This section examines long-term heat (LTH), long-term dose (LTD), and long-term radiotoxicity (LTR).

5.2.1 Long-Term Heat (LTH)

Unless otherwise stated, all analyses in this section are for the case of 50-year ventilation, the minimum required by law.

Table 5-6 shows the LTH improvement for the various single-pass cases. The best is IMF-NpPu, which is slightly better than IMF-NpPuAmCm. This was somewhat surprising. We believe it is caused by needing more Pu to make the NpPuAmCm case work; indeed, the IMF-NpPu case has 14% higher burnup. A single-pass strategy does not meet program objectives. Note that the IMF blended core case does not perform as well (for this first cycle) as the IMF full core cases, which go to higher burnups and leave little fissile material left. This of course makes it very difficult to continue to recycle those IMF cases. The IMF blended core does better than the 1st pass MOX cases. By design, the multi-pass IMF approach sacrifices some benefit the first pass in an attempt to keep going.

Table 5-6. LTH Results for Single-pass Cases, Normalized to UOX-51 as baseline

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPu AmCm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Burnup (GWth-day/tonne)		33	100	51	51	633.2	553.8	64.8	66.1
Long-term heat (LTH) improvement	10x to 200x (to achieve actual repository improvements of 10-50x)	0.95	1.17	1.07	1.12	1.98	1.82	1.61	1.67

Table 5-7 compares current results with those of Wigeland for multi-pass MOX. Recall in this study the LTH improvement is calculated solely by the metric of heat released from when ventilation stops (50 years) to 1500 years, whereas Wigeland's analyses determine the maximum loading to stay within the set of temperature limits. Nonetheless, we find good agreement.

Table 5-7. Current results compared to those of R. Wigeland

	Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5
50-yr ventilation (this study)	1.12	1.19	1.26	1.34	1.43
75-yr ventilation (Wigeland)	1.09	1.18	1.27	1.39	1.49
100-yr ventilation (this study)	1.09	1.19	1.28	1.38	1.48
300-yr ventilation (this study)	1.05	1.13	1.27	1.41	1.54

Figure 5-19 compares multi-pass cases, either calculated here or in an earlier report [Stillman2004c]. Note that the two MOX-NpPuAm cases agree; the earlier report used actual thermal transient calculations, as discussed in Chapter 3, in this report we used the simpler LTH metric. CORAIL is a MOX-type approach with a different blending strategy than the MOX case used here and in

Stillman2004c. The cycle-by-cycle isotope inventories in the CFR and BFR cases in [Stillman2004c] were estimated in that study.

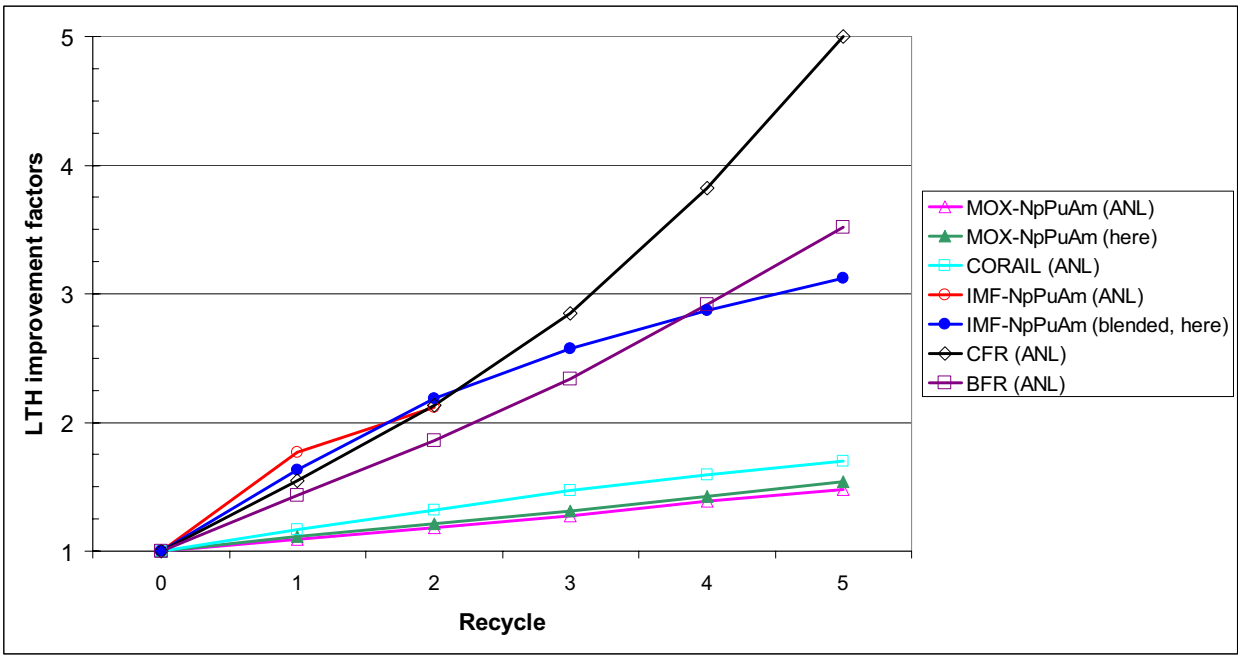

Figure 5-19. Comparison of multi-pass cases with 75-year ventilation in the repository. Cases marked "ANL" were calculated in [Stillman2004c].

Three observations of Figure 5-19 are as follows:

- Even after 5 recycles, **none** of the cases achieves the AFCI program objectives of 10-50.
- MOX does substantially worse than either IMF or the FR cases.
- 1-pass, full core IMF-NpPuAm is the highest after the first recycle, but at the cost of creating a residual that is very difficult to continue to use in an LWR. The blended core IMF case in this study more modestly burns TRU in the first recycle, but the approach can continue for additional recycles.

Figure 5-20 compares multi-pass MOX with multi-pass IMF. The IMF cases do better than MOX; however, none of these cases achieve the minimum 10x objective. As shown in Chapter 7, little mass in the system would get past 5 cycles this century. So, multi-pass IMF or MOX looks like a strategy that would work this century. The load on the repository would be controlled by losses. If recycling stopped after 5 cycles, there would be still be real benefit, ~3x for IMF.

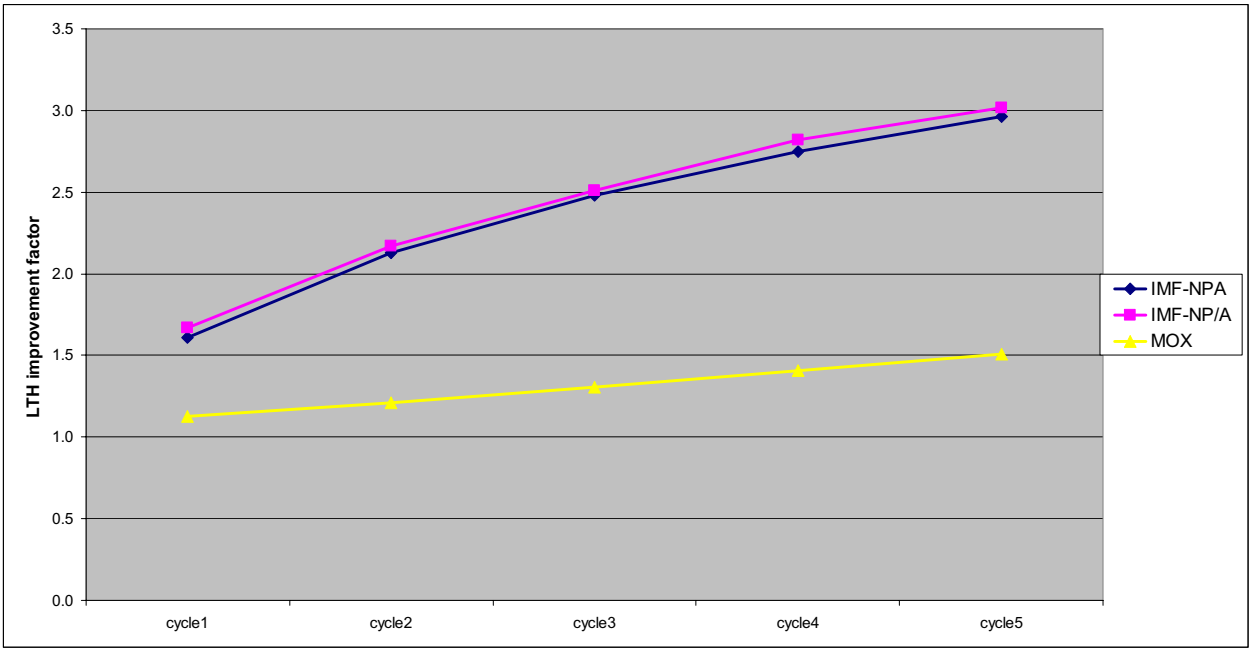

Figure 5-20. Comparison of multi-pass full-core MOX versus multi-pass blended core IMF (50-yr ventilation)

We now consider how "hot" each fuel becomes, all of the recycle cases have the effect of concentrating the heat-dominant isotopes in relatively less fuel. This is equivalent to saying that most of the mass of the system has been removed, i.e., the uranium. Figures 5-21a and 21b show the LTH in units of W-yr per gram of fuel for the various cases. These values are for fuels only, not how they would be combined in a given system. Figure 5-21a shows that the recycle fuels get significantly "hotter." Figure 5-21b shows that in all cases the LTH potential is dominated by NpPu at 5-yr after discharge. Note that the relative importance of fission products (Cs and Sr) declines relative to once-through fuels: fission products do not accumulate each recycle, TRU elements do.

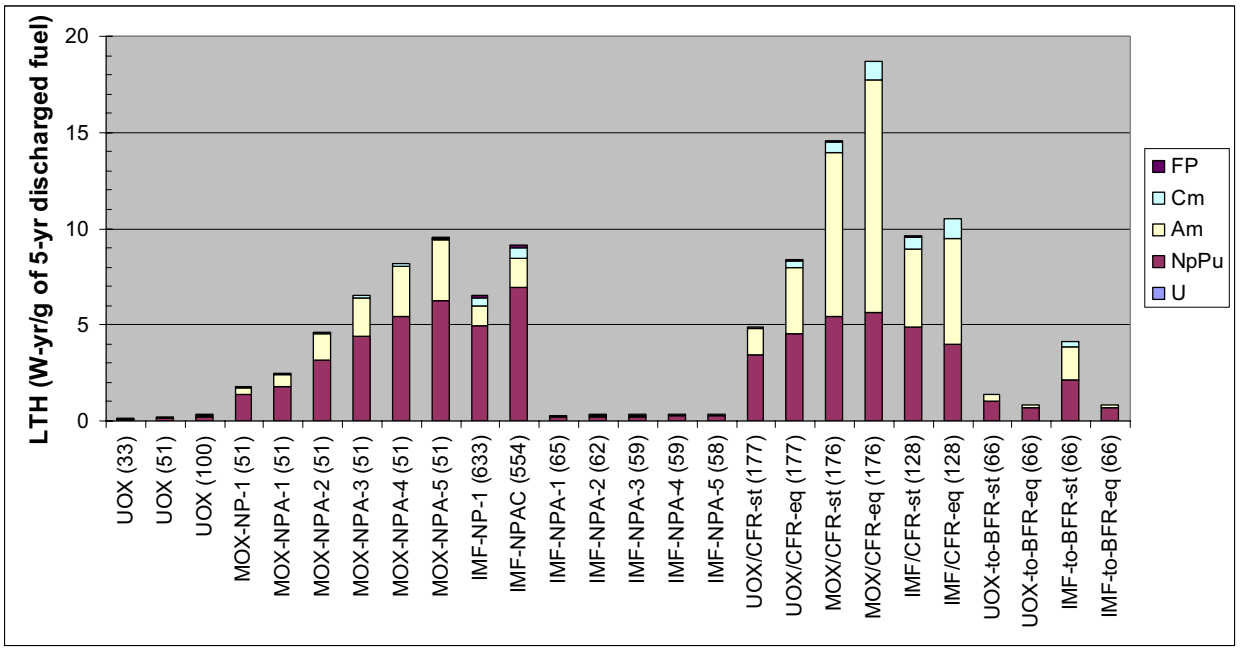

Figure 5-21a. LTH per gram of each fuel

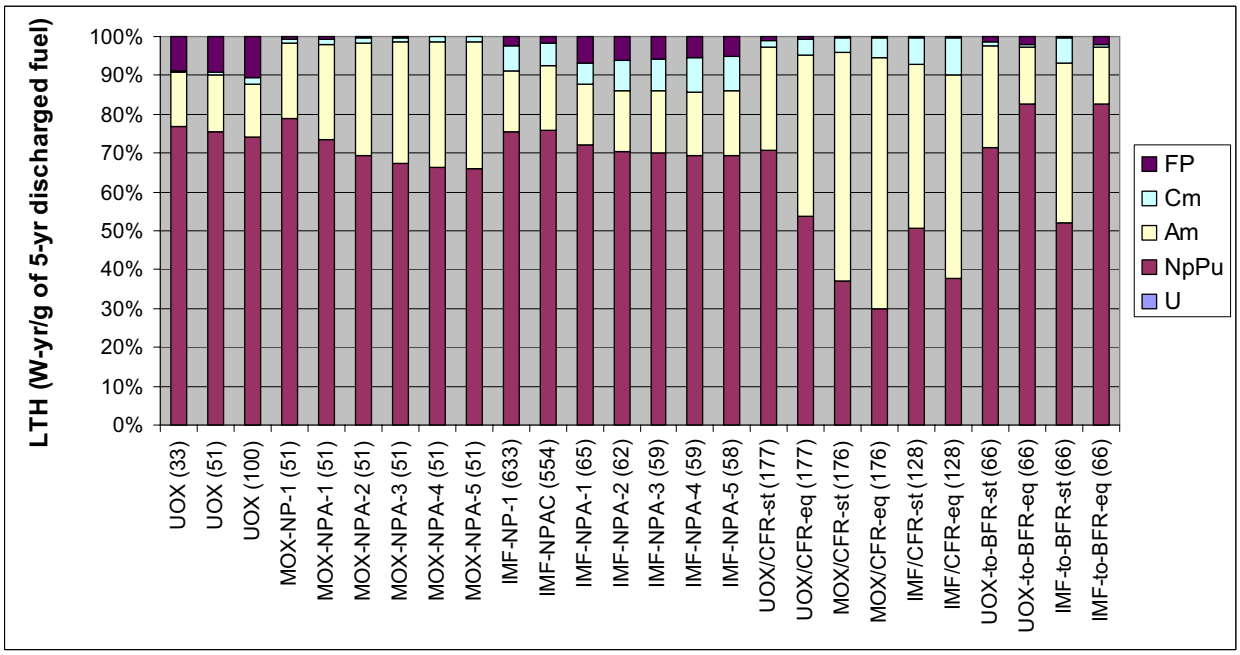

Figure 5-21b. LTH per gram of each fuel, normalized to 100% to show elemental contribution to total.

A more appropriate comparison is to normalize the above results on a per energy basis. Therefore, Figures 5-22a and 5-22b show LTH values, this time on a per GWe basis. As with Figure 5-16, these are not "fleet" calculations, they are simply the values for each type of fuel. The MOX cases are seen has attaining high LTH values, the dominant isotope becomes Pu238.

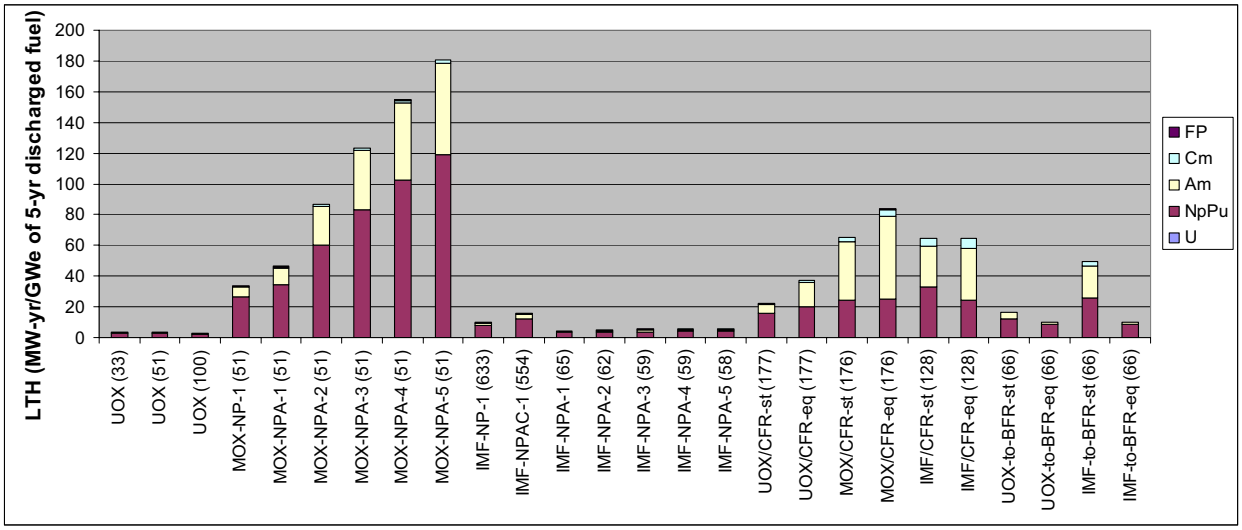

Figure 5-22a. LTH per GWe of each fuel

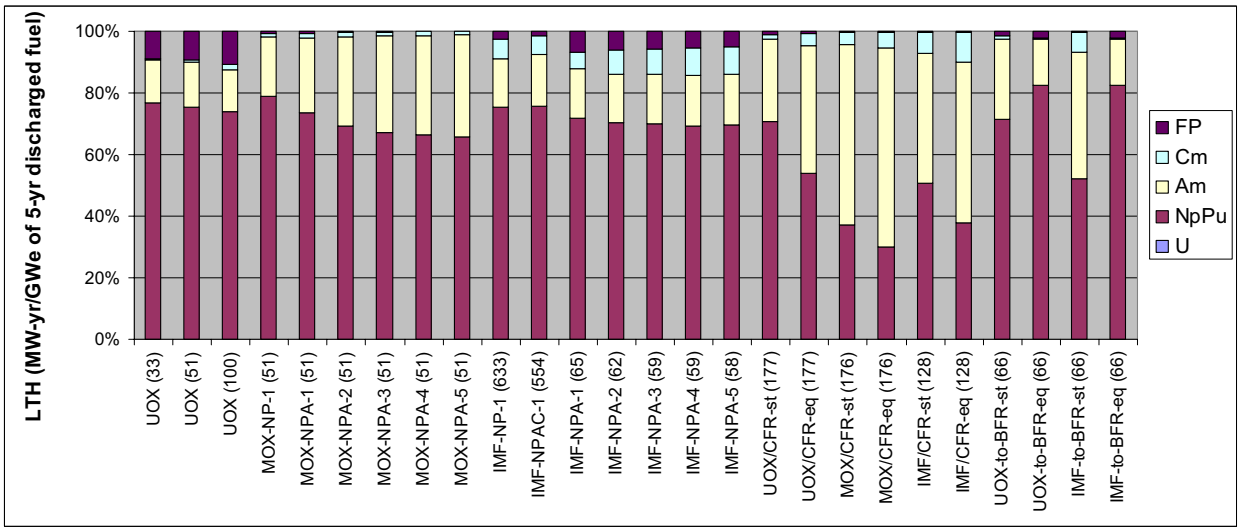

Figure 5-22b. LTH per GWe of each fuel, normalized to 100% to show elemental contribution to total.

We now turn to "equilibrium" calculations for the entire fleet:

5-pass MOX, then discard
5-pass IMF, then discard
UOX-CFR
UOX-MOX-CFR
UOX-IMF-CFR
UOX-BFR
UOX-IMF-CFR

The mix of fuels at equilibrium is shown in Figure 5-23.

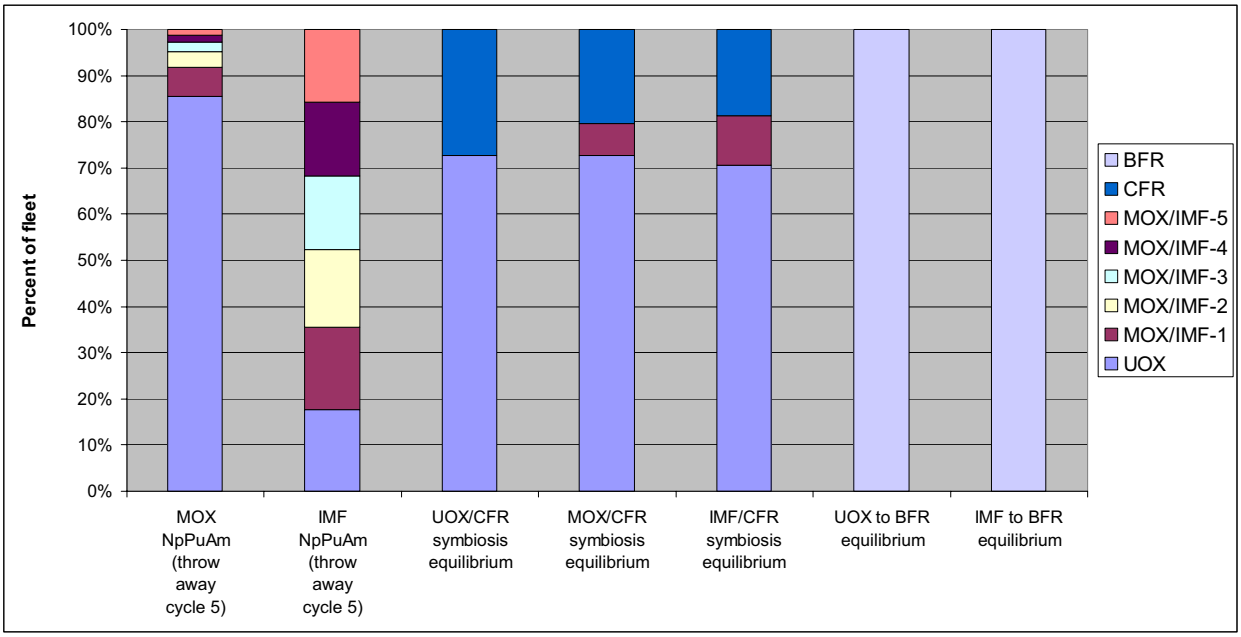

Figure 5-23. Equilibrium mix of fuels in seven systems

Figure 5-24 shows the resulting LTH improvement factors. For the pure-thermal cases, loss rates do not matter much because the 5th cycle is discarded. For the CFR and BFR cases, if Cs and Sr fission products are not separated, but 100% of TRU are recycled, it appears that an LTH improvement of 10x can be achieved, barely. Recall, however, that LTH improvement overpredicts actual repository heat improvement; see Chapter 3. Wigeland found [Wigeland2005] a maximum benefit of 6x (actual repository heat calculation, not simple metrics) without separation of fission products, but this was looking only at the composition of used UOX-51. Current results are not a basis for concluding that fission products can be kept with recycled FR fuel.

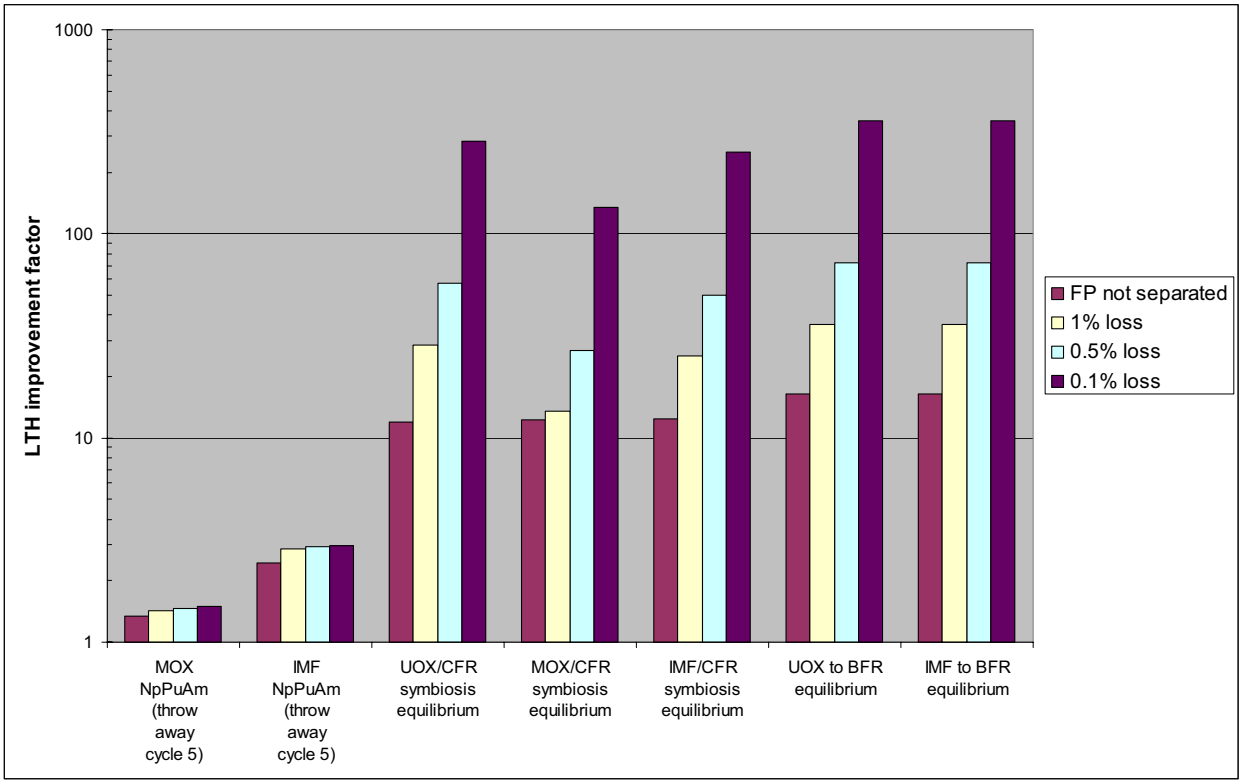

Figure 5-24. LTH improvement factors for seven systems

The program objective is 99.5% recovery of TRU.[DOE2005a] A tentative program objective for Cs and Sr during preparation of that report had been 99% recovery. In this case, the LTH improvement factors are 27 (MOX/CFR), 49 (IMF/CFR), 55 (UOX/CFR), and 70 (BFR), see Figure 5-25. Because the LTH improvement factors can overpredict actual repository performance, these results may not be adequate to avoid a second repository for high growth scenarios. It is envisioned that Cm is always recycled in fast reactor systems, nonetheless, the figure also shows a curious result with regard to separation of Cm. Because less Cm accumulates in the BFR systems, it is less important to recycle it. Even if fission products and Cm are thrown away each cycle, the BFR system keeps an LTH improvement of just over 10x. We caution that this is unlikely to actually achieve a repository improvement of 10x, however.

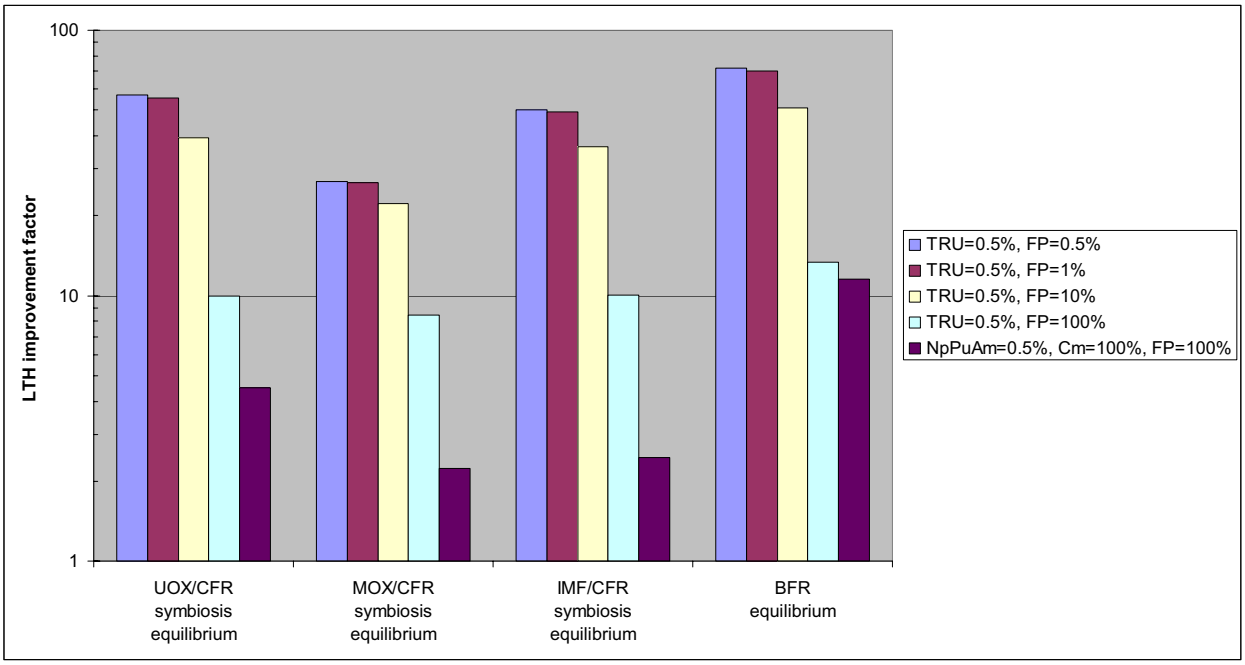

Figure 5-25. LTH improvement factors for FR systems

5.2.2 Long-Term Dose (LTD)

In many ways, the patterns for LTD and LTH are similar. The single most important set of TRU isotopes are Pu241/Am241/Np237 for LTD, Pu241/Am241 for LTH. There are five major differences, which we emphasize because the AFCI program has tended to be focused on LTH, but must also now examine LTD.

First, LTD must be assessed at multiple times after emplacement. The patterns, for example, at 10,000 versus 500,000 after emplacement are fundamentally different, as discussed in Chapter 3. In contrast, the patterns for LTH at different ventilation periods are basically the same. For UOX, peak doses occur at 500,000 years. As shown below, however, if some, but not all, components of waste are removed, the peak dose can shift to other time periods. **Our interpretation is the program goal is to reduce peak doses, whenever they occur.**

The second set of differences involve fission products. For LTH, the dominant fission products are Cs and Sr. They cannot be transmuted effectively.[Salvatores1998] The only options (Chapter 2) are to keep them with HLW or separate and dispose of them elsewhere, e.g., at-grade storage or in a special part of a repository. For LTD, the dominant fission products are Tc and I. Their relative importance depends on the time after emplacement. They can be transmuted, although slowly.[Yang2004, Salvatores1998] If separated from HLW, options include transmutation, putting in a more durable waste form than spent fuel (hence reducing their effective contribution to repository dose), or putting into a special repository tailored to their chemistry. **In the following analysis, we simplify these options to either keeping Tc and I with HLW, or separating them and doing something that would effectively eliminate their contribution to repository dose by some unspecified method.**

The third set of differences involve the importance of chemical form. The chemical form is irrelevant to LTH, but highly relevant to LTD. **In the following analysis, we have no way to apply credit for improved waste forms; all material is assumed to behave as if spent UOX.**

The fourth set of differences involve the benefit, or lack thereof, of delay. It is well known that for LTH there is a benefit of waiting for Cm244 (18.1 years) to decay. There are LTH tradeoffs for Pu241/Am241; delay allows Pu241 to decay to Am241, which is more difficult to transmute, but less likely to lead to some of the higher actinides. For LTD, delay is generally less important. The key long-lived isotopes for the 4 decay chains are as follows:

Chain 4N: U236 (2.3e7) and Th232 (1.4e10)
Chain 4N+1: Np237 (2.1e6) and U233 (1.6e5)
Chain 4N+2: U238 (4.5e9) and U234 (2.4e5)
Chain 4N+3: U235 (7.0e8) and Pa231 (3.0e4)

Delay only allows TRU to decay down to U236, Np237, U238, and U235. Indeed, on a "per gram" basis the LTD for many TRU isotopes are the same as one of those four isotopes. Delay does not eliminate a problem (as does the disappearance of Cm244 from LTH calculations once it decays); it is an advantage benefit or disadvantage only if the transmutation rate of its daughters are higher or lower than the original isotope.

The final set of differences involve uranium. For LTH, uranium is unimportant; whether uranium is separated, burned, put into near-surface disposal is irrelevant. For the LTD values in this study, based on recent unpublished YMP values (Chapter 3, Appendix D), uranium is important. Indeed, there is at least one long-lived isotope in each of the four decay chains. For LTH, decay into uranium isotopes (and Np237) is good because they are long-lived and therefore do not contribute much to heat in the 1500-year time period. For LTD, however, decay into uranium isotopes (and Np237) accomplishes nothing because these are the long-lived isotopes that can transport out of the repository. And, the underlying chemistry is that uranium is more soluble and transports faster than TRU.

Figures 5-26a and 5-26b show the hypothetical dose at 500,000 years. Note first that the uranium dose increases from MOX cycle 1 to 5, even though the fraction of uranium in the fuel is decreasing! Tracing back through the calculations reveals that the cause is slow accumulation of U234, a decay product of Pu238. Indeed, the small amount of uranium made in IMF-NpPu and IMF-NpPuAmCm is U234. Similarly, the uranium doses for the BFR-equilibrium fuels are low relative to the amount of uranium in the fuel because there is little U234 (less accumulation of Pu238) and relatively little U235. As shown in Chapter 3, the dose per gram from U234 is higher than U235, which is higher than U238.

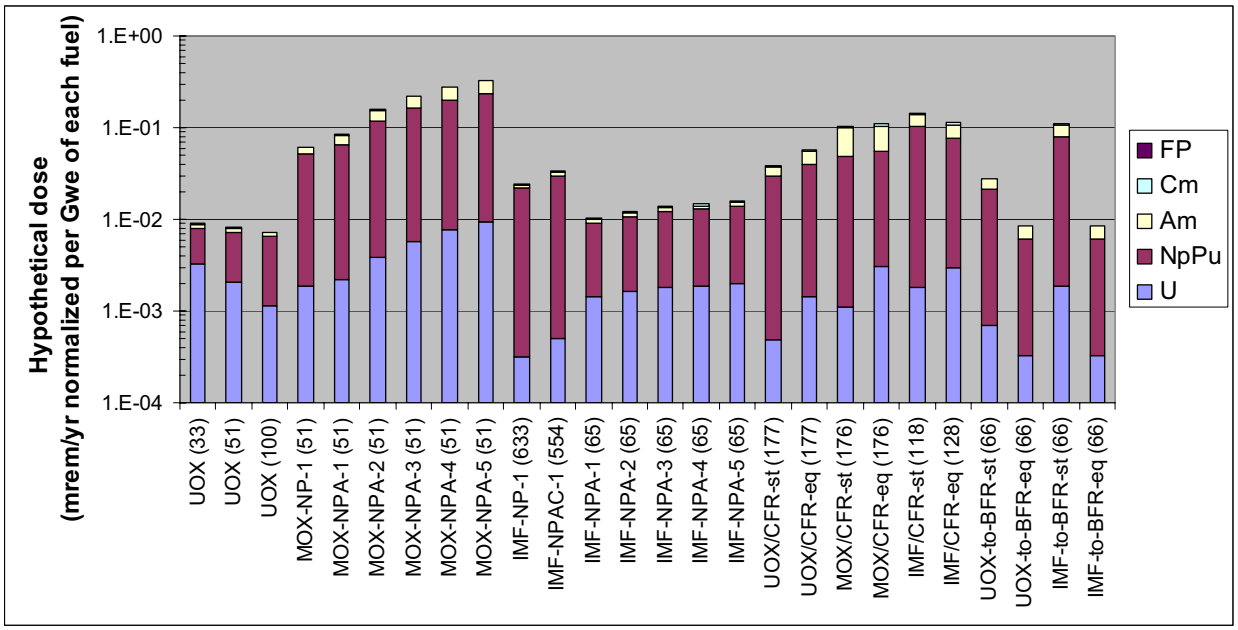

Figure 5-26a. Hypothetical dose per GWe at 500,000 years per GWe of fuel

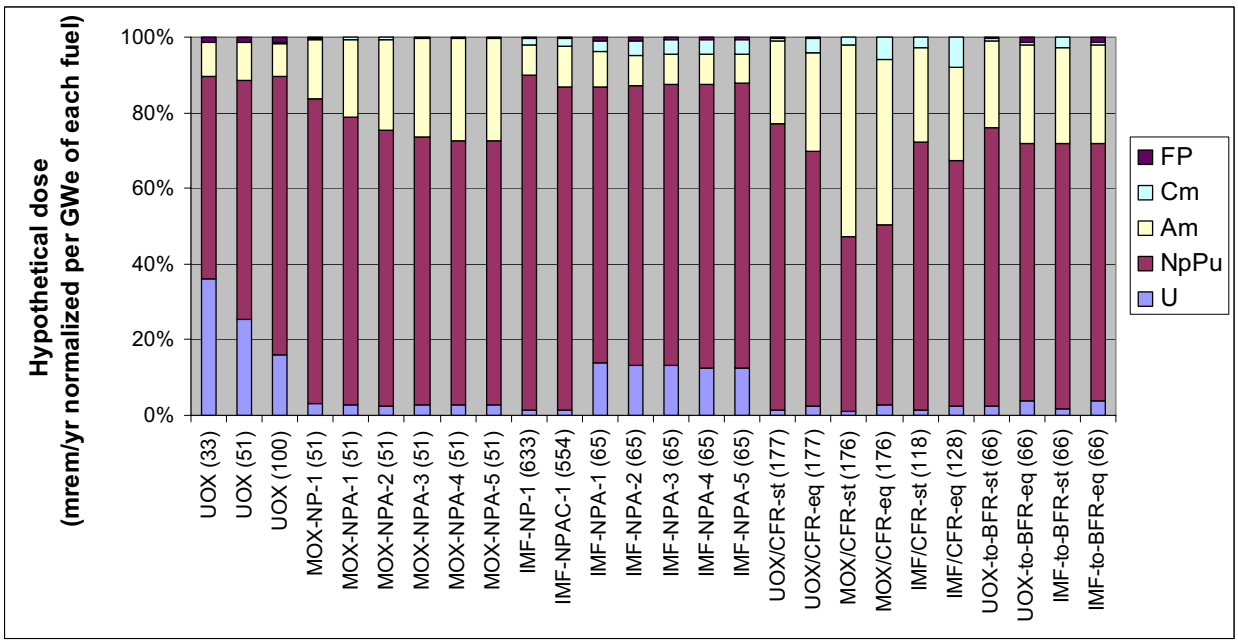

Figure 5-26b. Hypothetical dose at 500,000 years per GWe of fuel, showing the contribution among elements

As expected, in all cases the dose at 500,000 years is dominated by NpPu, especially Np237 and Pu241. The Am dose is dominated by Am241. At 500,000 years, the FP products are relatively unimportant, never over 1.4%.

Table 5-8 shows the LTD reduction for single-pass cases. The pattern is the same as for LTH. None of the systems come close to adequate performance.

Table 5-8. Key Results for Single-Pass Cases relative to UOX-51

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPu ACm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Long-term dose (LTD) improvement	10-50x	0.90	1.12	1.35	1.41	2.09	1.96	1.57	1.63

Figure 5-27 compares multi-pass MOX with multi-pass IMF. As with LTH, the IMF cases do better; however, none achieve the minimum 10x objective. Removing Tc and I helps, but not much.

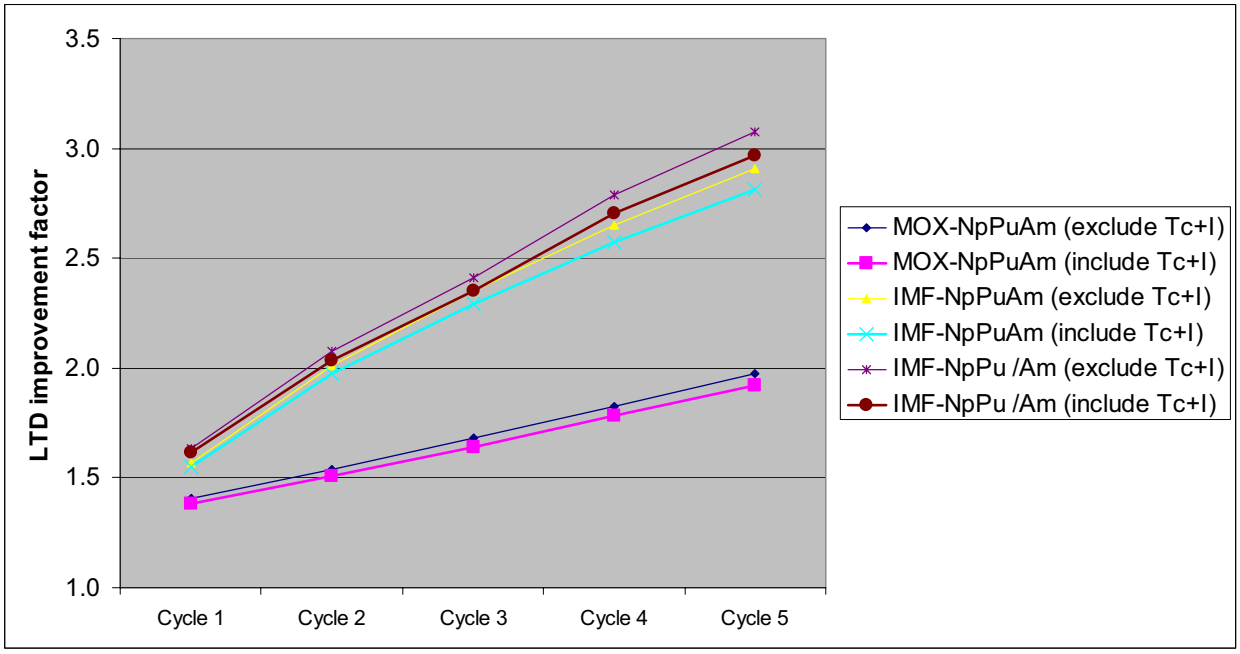

Figure 5-27. LTD improvement factor for multi-pass MOX and multi-pass IMF

So far, the improvement factors are so small, it does not matter that dose varies at different time periods, but as we look for higher LTD improvements, we have to consider that impact. Figure 5-28 shows the hypothetical dose for 70,000 tonnes of once-through UOX-51 with varying fractions of elements removed. Baseline (nothing removed) peaks at 31 mrem/yr at 500,000 years per the original numbers from DOE-RW. Removing 99% of NpPu, NpPuAm, or even NpPuAmCm does not meet the program objective of 10x improvement. Rather 90% removal of U is required in addition to removing 99% of NpPuAm; Cm is not an issue at this point. So, 99% NpPuAm and 90% U removal meets the minimum 10x objective, it actually achieves about 20x. Perfect removal of everything except FP achieves 26x.

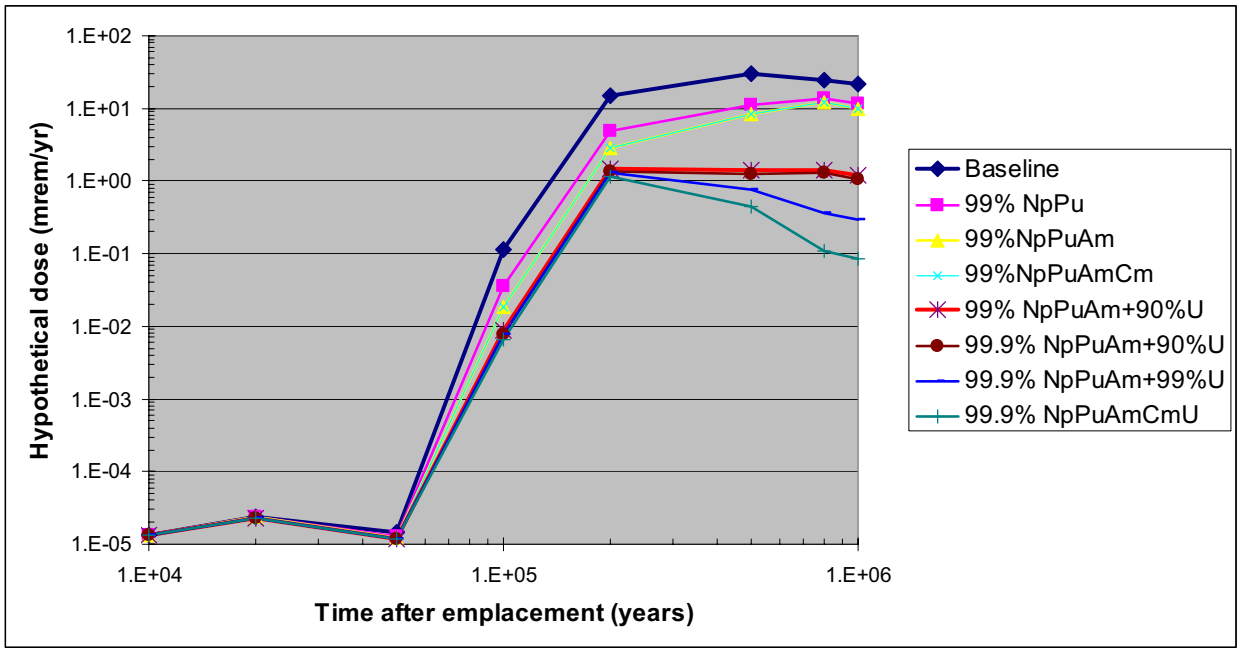

Figure 5-28. Hypothetical dose for 70,000 tonnes of UOX-51 with various elements removed; removal of 99% NpPuAm and 90% U achieves LTD reduction of 10x

We argued in Chapter 3 that it could be valuable to achieve better LTD performance, at least sufficient so that recycling is a net "wash" on dose – this would require the LTD improvement to be as high as the repository heat improvement, for example 50x.

Figure 5-29 shows that going to 99.9% removal of U, Np, Pu, Am, and Cm would not be sufficient to reduce peak dose by 50x. Once most of the U, Np, Am, and Am are removed, the problem becomes fission products Tc99 and I-129 at 100,000 years after emplacement. Figure 5-29 shows that 90% removal of technetium and iodine, in addition to 99% removal of NpPuAm and 90% removal of uranium, would indeed reduce peak doses by 50x.

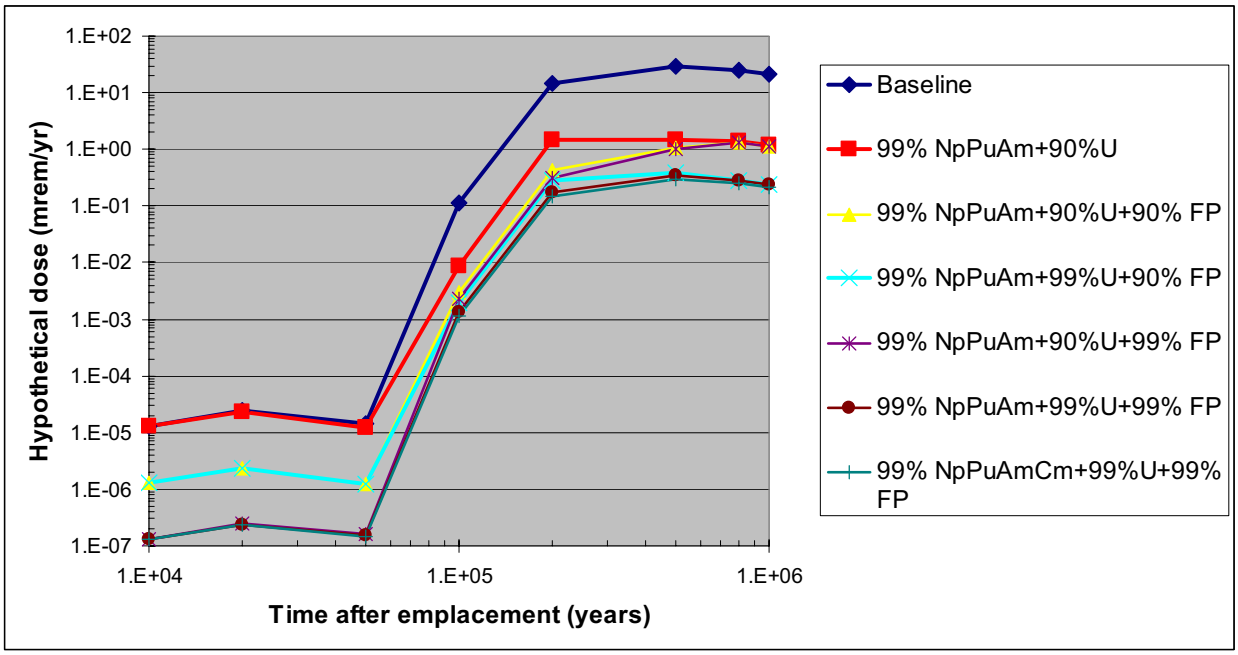

Figure 5-29. Hypothetical dose for 70,000 tonnes of UOX-51 with various elements removed; removal of 99% NpPuAm and 90% U and 90% of Tc and I achieves LTD reduction of 50x

Figure 5-30 shows LTD improvement factors for the same "equilibrium" cases. The first bar, "U+TRU=0%, FP=100%" denotes total recovery of heavy metals but all Tc and I going to the repository. This limits LTD improvement to about 20x. If we want to achieve LTD improvement as high as our target repository heat capacity improvements, something has to be done with the Tc and I.

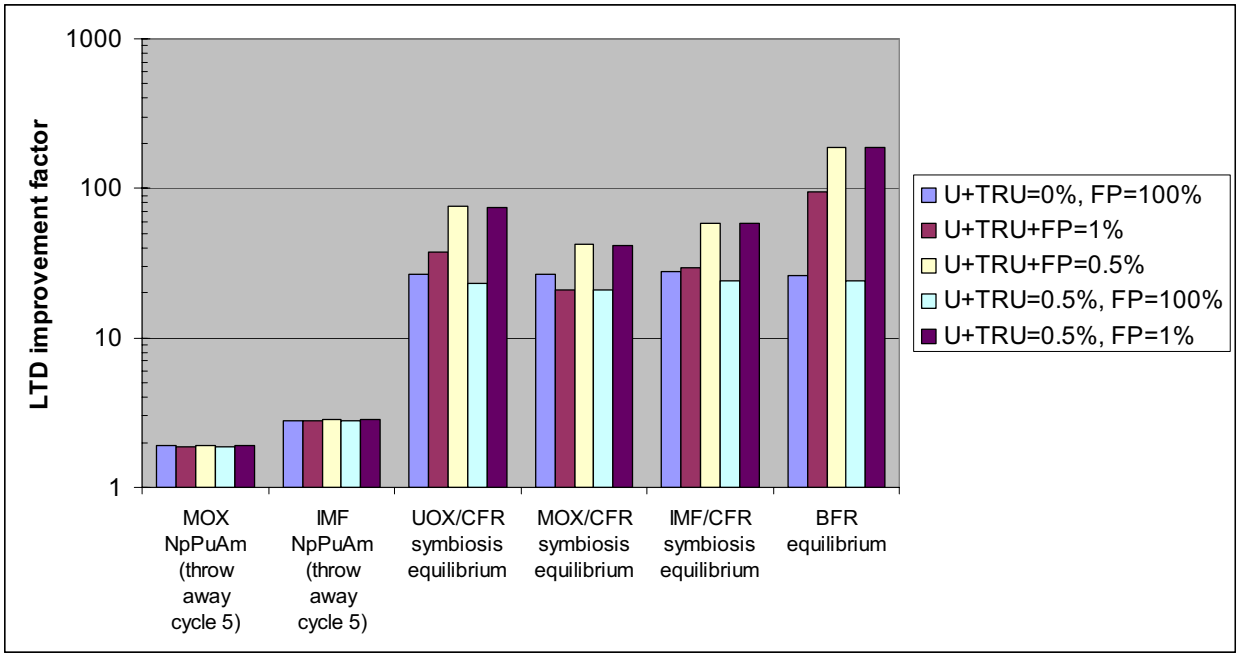

Figure 5-30. LTD improvement factors for sustainable cases

Therefore, the program should not exclude the possibility of recycling Tc and I. Fortunately, the UREX+ process (as embodied in the Spent Fuel Treatment Facility (SFTF) design study) extracts Tc and I so that they are separated for potential future transmutation, for dedicated waste forms (to reduce the effective dose relative to spent UOX) or potential geologic disposal; all options would be left open.

The above results are merely exploratory and future work is required to make more exact analyses.

5.2.3 Long-Term Radiotoxicity (LTR)

LTR is yet different than either LTD or LTH. Like LTH, the chemical form of waste is irrelevant. Like LTD, the value depends on the time after emplacement.

The conclusions regarding LTR are the following:

- Recycling of fission products such as Cs, Sr, Tc, and I are not needed to meet the LTR objective of 100x reduction at 1,000 years.
- Recycling of Pu and Am are required to meet the LTR objective. Recovery of 99.5% of the Pu during multi-pass recycling is probably not adequate because Pu itself must be reduced by almost a factor of 100x. (99.5% implies 200x reduction, but for only the first pass.)
- Strictly speaking, Np recycle is not required for the LTR objective.
- Recycling of Cm is borderline.
- Accounting for radioactive daughters is vital, including the daughters from uranium. For example, Pu238 decays into U234. The radiotoxicity of most of the uranium isotopes is dominated by their radioactive daughters, not themselves.

In contrast to LTD in the previous subsection, note that Np, Tc, and I are less important to LTR than to LTD. This arises because all three elements are relatively mobile in the YMP geology (hence a factor in LTD) but mobility is irrelevant to LTR calculations.

Figure 5-31 shows the radiotoxicity relative to uranium ore for the same case as in Figure 3-8[DOE2005a] except that this version shows the contribution by isotope. Np237 itself is not a major contributor. Note that **the LTR shown for each isotope in figure 5-31 is only that isotope**. When an isotope decays, its LTR drops. The LTR from its daughters is reflected in the daughters themselves, not the parent. In Figure 5-31, there are only two cases where LTR increases with time – Am241 because of in-growth from Pu241 and Np237 because of ingrowth from Am241 and Pu241. Figure 5-32 shows the same case, but the only dominant isotopes.

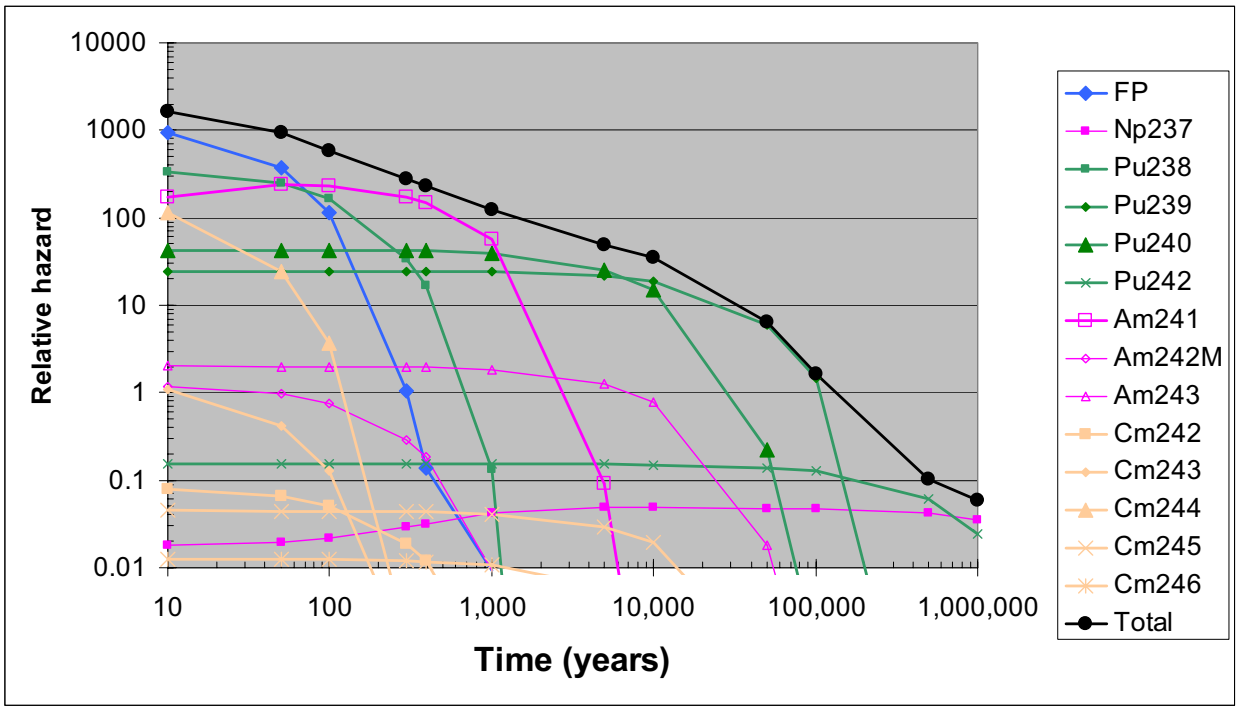

Figure 5-31. Radiotoxicity relative to natural uranium ore, by isotope (same case as Figure 3-8).

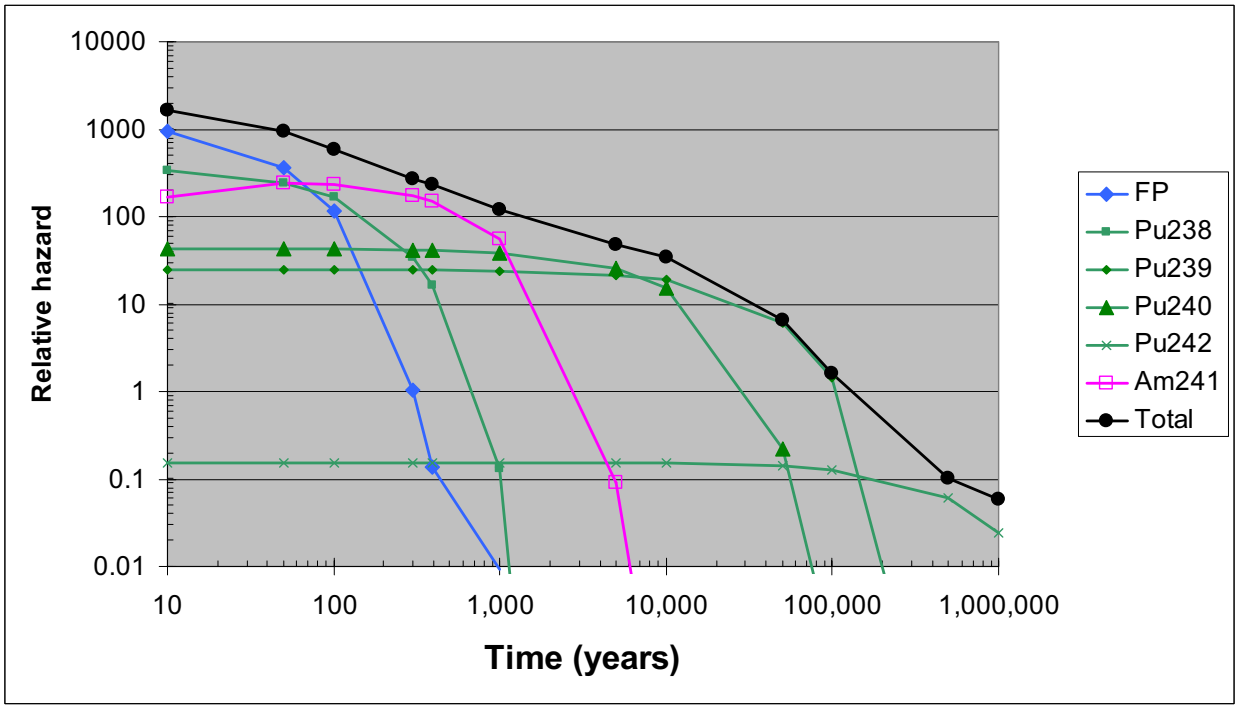

Figure 5-32. Radiotoxicity relative to natural uranium ore, by dominant isotope (same case as Figure 3-8).

Figures 5-31 and 5-32 show by inspection that neither fission products (FP) nor Np are important relative to the LTR reduction objective. These figures also imply that recycle of Cm is not required; however, as

shown below, this is an erroneous conclusion because the daughters of Cm244 are important. Recycle of Pu and Am are required.

In this study, we calculated LTR metrics such that **each isotope includes all of its associated daughters**; the results are in Figure 5-33 (most isotopes), Figure 5-34 (dominant isotopes), and Figure 5-35 (grouped by element). So, for example, Pu241 decays into Am241, that radiotoxicity is denoted as parent Pu241, rather than Am241. Thus, the Am241 curve in Figures 5-31/32 is essentially replaced by the Pu241 curve in Figures 5-33/34.

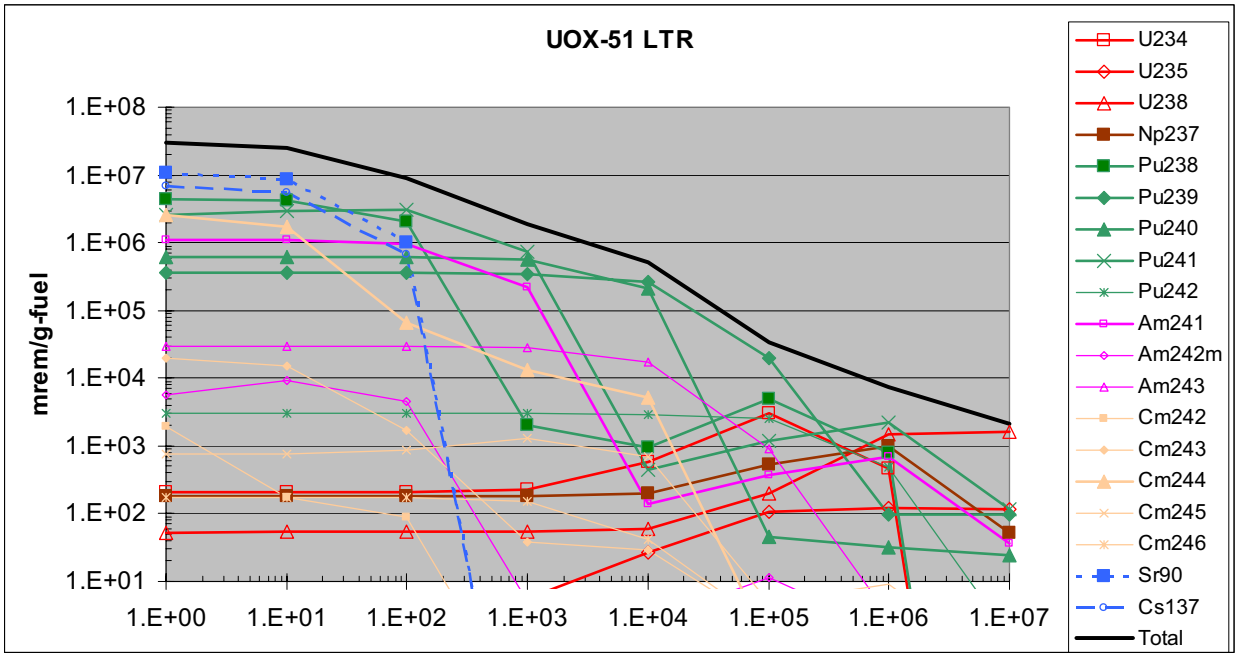

Figure 5-33. Radiotoxicity of discharged UOX-51 (same case as Figure 3-8), note the time scale from 1 to 10 million years

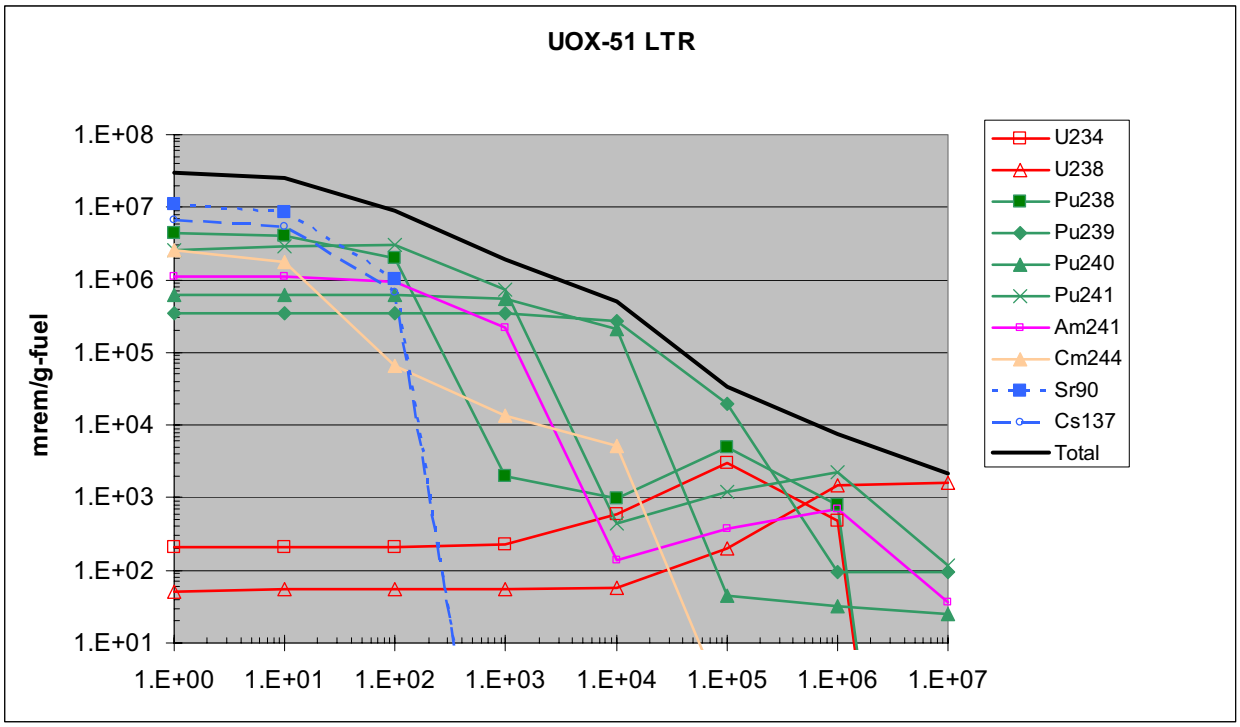

Figure 5-34. Radiotoxicity of discharged UOX-51, by dominant isotope, over the time period 10 to 1 million years for comparison with Figure 5-31/32.

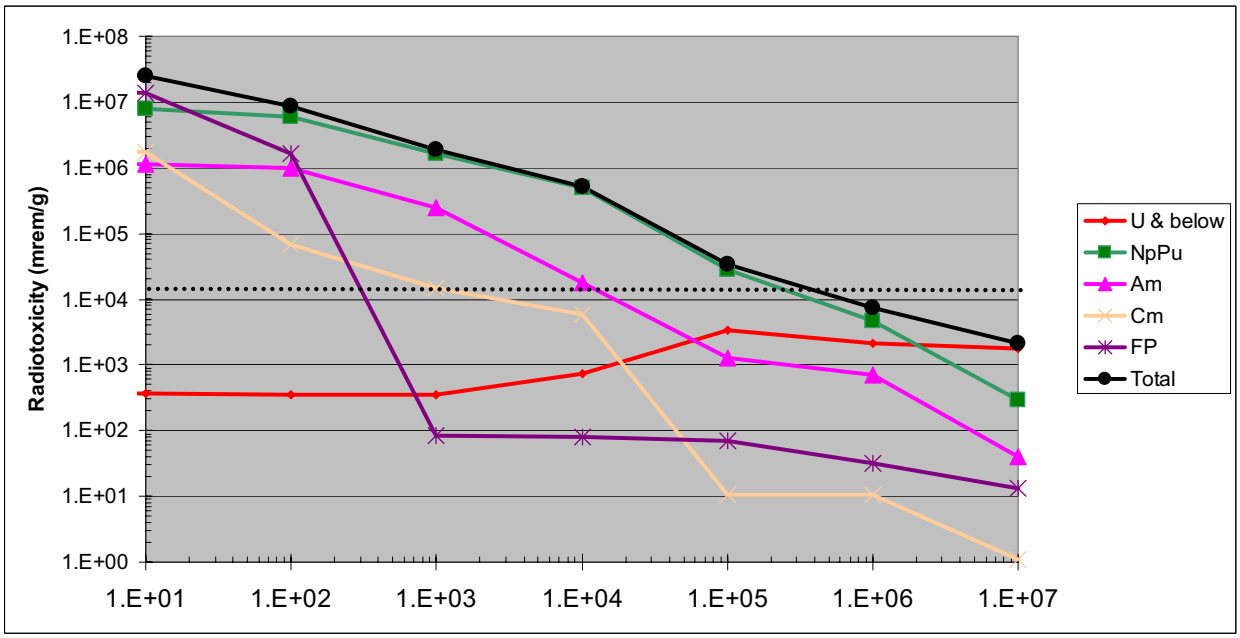

Figure 5-35. Radiotoxicity of UOX-51, grouped by element at time of discharge. The radiotoxicity of the uranium ore equivalent to fuel (dotted line in the figure) is 1.3e4 mrem/g-fuel.

In principle, either calculational approach – show each daughter explicitly in Figures 5-31/32 or incorporate daughters with their isotope of origin in Figures 5-33/34/35 will produce the same total

radioatoxicity. The latter approach has the advantage of more clearly identifying which elements have to be recycled. Isotopes have to be recycled if they – or their daughters – adversely impact total LTR.

Indeed, previous Figure 3-11 showed that the current calculation matches that of [DOE2005a] and [AAA2001], except that [DOE2005a] ignored uranium isotopes. The current calculation includes daughters with their parents; the two earlier calculations showed daughters separately.

Inspection of Figures 3-31/32 and Figures 33/34 show consistent trends, as follows:

- 10 to 100 years – both calculations dominated by short-lived fission products
- 100 to 1e3 years – [DOE2005a] calculation dominated by Am241; current calculation dominated by Pu241 and its daughters such as Am241
- 1e3 to 1e4 years – both calculations dominated by Pu240
- 1e4 to 1e5 years – both calculations dominated by Pu239
- 1e5 to 1e6 years – [DOE2005a] calculation dominated by Pu239/Pu242/Np237; current study's calculation dominated by Pu239/Pu242/Pu241 and their daughters such as Np237.

Figure 5-35 shows a dotted line representing the radiotoxicity of the uranium ore that gave rise to the equivalent amount of fuel. The radiotoxicity of uranium (see below) is 1.6e3 mrem/g-ore. About 7.9 g-ore are needed to make 1 g-fuel. Thus, the radiotoxicity of uranium ore equivalent to fuel is 1.3e4 mrem/g-fuel (1.6e3 x 7.9 g-ore/g-fuel).

Consistent with Figures 5-31 and 5-32, Figure 5-35 shows that fission products do not have to be recycled to meet the target of reducing LTR at 100x at 1,000 years after shutdown. Pu and Np are plotted together in Figure 5-35 because in the UREX+ process, they are never separated. Consistent with Figures 5-31 and 5-32, Pu and Am must be recycled and transmuted to achieve the 100x target at 1,000 years and therefore bring the radiotoxicity of residual waste below that of uranium ore.

The one key difference beween Figure 5-35 and Figures 5-31/32 is Cm. In contrast to Figures 5-31/32, Figure 5-35 shows that recycle of Cm is a borderline requirement. The difference is due to Cm244 decay to Pu240. In Figures 5-31/32, the LTR of Pu240 includes both the original Pu240 and the minor in-growth from Cm244 decay, thereby masking the contribution from Cm244 decay into Pu240. In contrast, Figure 5-33/34 includes the radiotoxicity of Pu240 (and other daughters) with the parent Cm244 because those daughters would be disposed on the basis of what happens to Cm, not Pu, in the separation process.

We now discuss uranium. Figure 5-36 shows only the radiotoxicity of uranium isotopes. The LTR of an isotope will increase as its radioactive daughters grow in and then eventually decay. Most of the uranium isotopes are important, as follows:

- U232 (72 yrs) – Its longest-lived daughter is Th228 (1.9 yrs) so its LTR builds quickly and decays simply per its 72-yr halflife. There is little U232 in UOX-51.
- U233 (1.6e5 yrs) – Its longest-lived daughter is Th229 (7.3e3 yrs) so that it takes until ~1e3 yrs for the LTR to grow, reflecting Th229 in-growth. However, there is little U233 in UOX-51 so it is not a major isotope.
- U234 (2.4e5 yrs) – Its longest-lived daughter is Th230 (7.7e4 yrs) so that its LTR increases as Th230 (and its daughters) grows in. It peaks near 1e5 yrs and then decays. There is substantial U234 in spent UOX-51 and its radiotoxicity should not be ignored.
- U235 (7.0e8 yrs) – Its key daughter is Pa231 (3e4 yrs) so that its LTR peaks about 1e5 yrs and then is constant.
- U236 (2.3e7 yrs) – Its key daughter is even longer-lived, Th232 (1.4e10 yrs) so that its LTR is not dominated by short-lived daughters.

- U238 (4.5e9 yrs) – Its key daughter is U235 (2.4e5 yrs) so that it takes until 1e6 yrs for the LTR to plateau. Thereafter, it is constant.

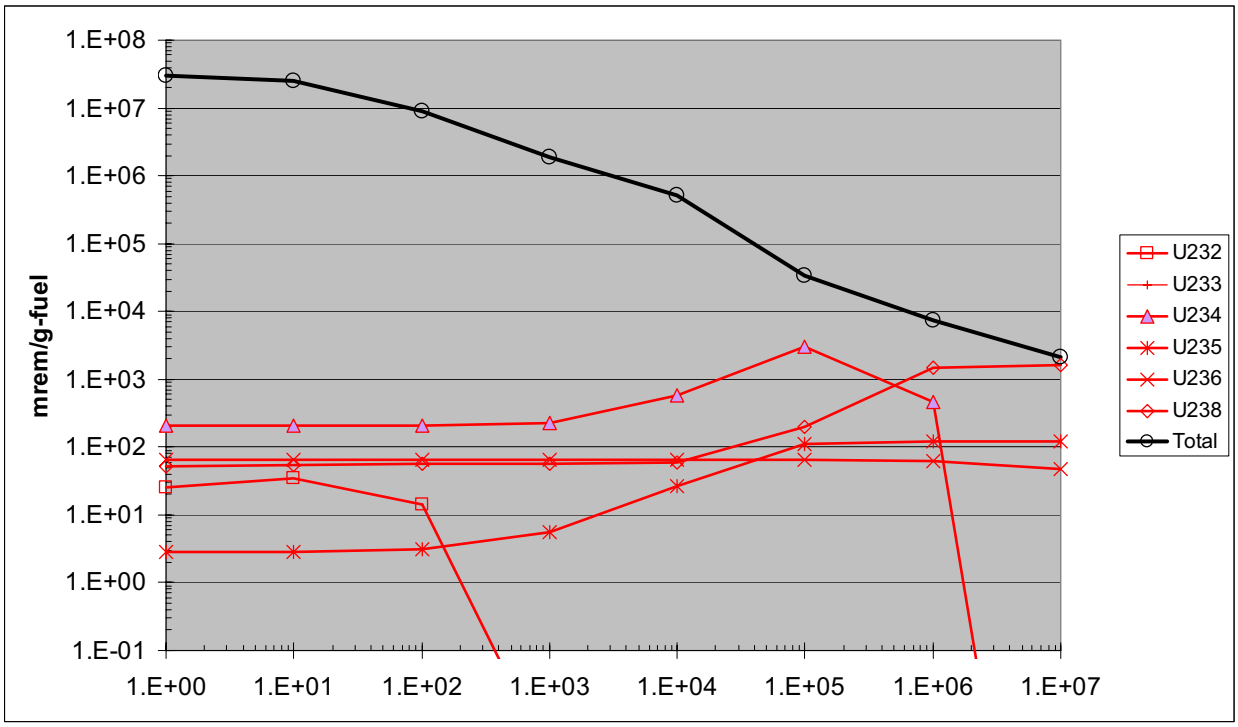

Figure 5-36. Radiotoxicity of UOX-51, uranium isotopes only.

Because natural uranium is 99.3% U238 and its radiotoxicity (once daughters build in) exceeds U235 and is close to U234, U238 defines the radiotoxicity of natural uranium ore (if one ignores natural thorium). Figure 5-36 shows that the radiotoxicity of uranium is 1.6e3 mrem/g-uranium. Indeed, by 1e7 years, the radiotoxicity of UOX-51 is also controlled by U238. On that time scale, spent fuel has returned to its starting point – uranium and its daughters. At times over 1e5 years, uranium isotopes must be considered in radiotoxicity calculations.

Figure 5-37 shows the radiotoxicity of used fuel for the three different UOX burnups considered in this study. Figure 5-38 shows the same thing, except normalized to uranium ore. Note that the normalization to uranium ore is itself a function of burnup because the uranium enrichment varies. At 33 MW-day/kg-HM, the conversion (0.2% tails) is 5.8 g-ore/g-fuel. At 51 MW-day/kg-HM, it is 7.9. At 100 MW-day/kg-HM, it is 15.9 g-ore/g-fuel. Figure 5-38 shows that the impact of burnup is very small, as we would expect – the accumulation of long-lived TRU isotopes (relative to the uranium ore required to make the UOX fuel) depends mostly on the amount of power generated (number of fissions), not whether the fissions are taking place in 3 batches of 33 MW-day/kg-HM fuel or 1 batch of 100 MW-day/kg-HM.

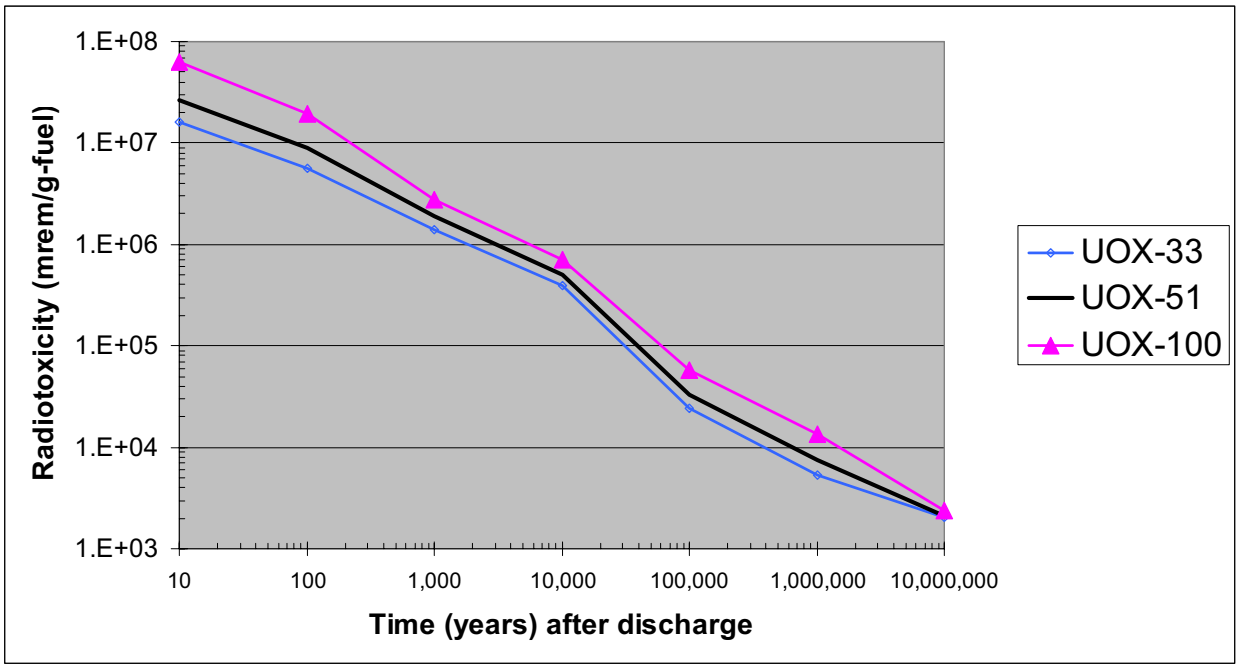

Figure 5-37. Radiotoxicity of UOX, showing the impact of burnup from 33 to 100 MW-day/kg-HM

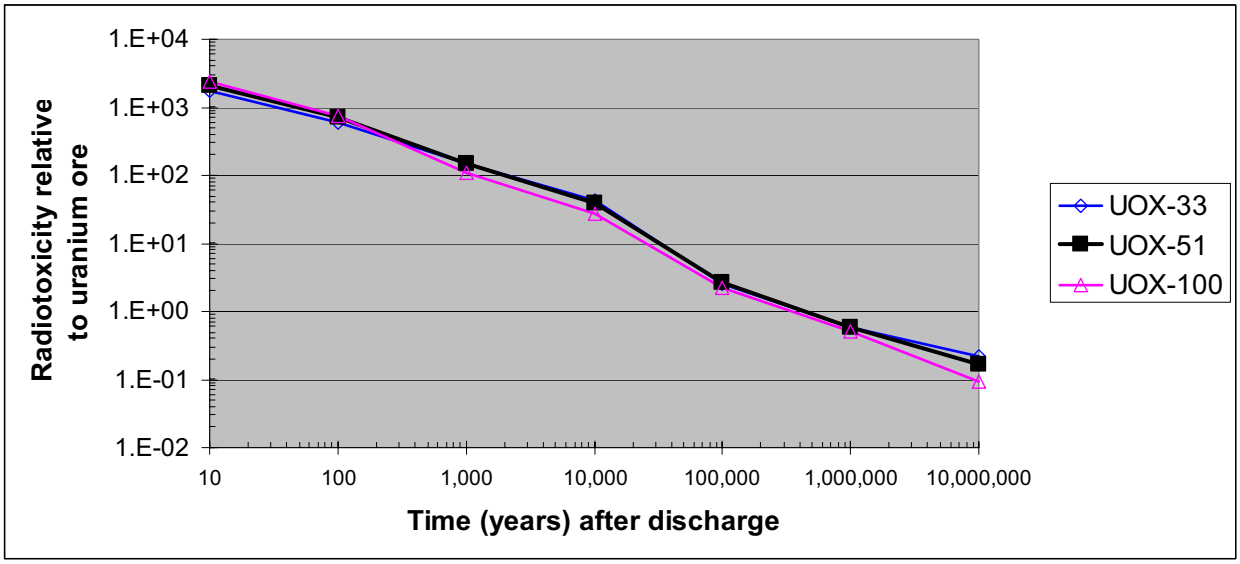

Figure 5-38. Radioxicity of used UOX, normalized to uranium ore, showing the impact of burnup from 33 to 100 MW-day/kg-HM.

Table 5-9 shows the radiotoxicity relative to UOX-51 for the single-pass cases in this study. Like LTH and LTD, the values are fairly close to 1, nowhere near the 100x target.

Table 5-9. Radiotoxicity results for single-pass cases, normalized to UOX-51

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPu ACm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Long-term radiotoxicity (LTR) improvement	100x at 1000 yr after discharge	0.89	1.38	1.12	1.18	2.46	2.39	1.79	1.85

Figure 5-39 shows the radiotoxicity improvement factor for mult-pass IMF and MOX; the trends are very similar to the analogous LTD and LTH improvement factors. This is unsurprising because the dominant elements – Pu, Np, and Am - are the same.

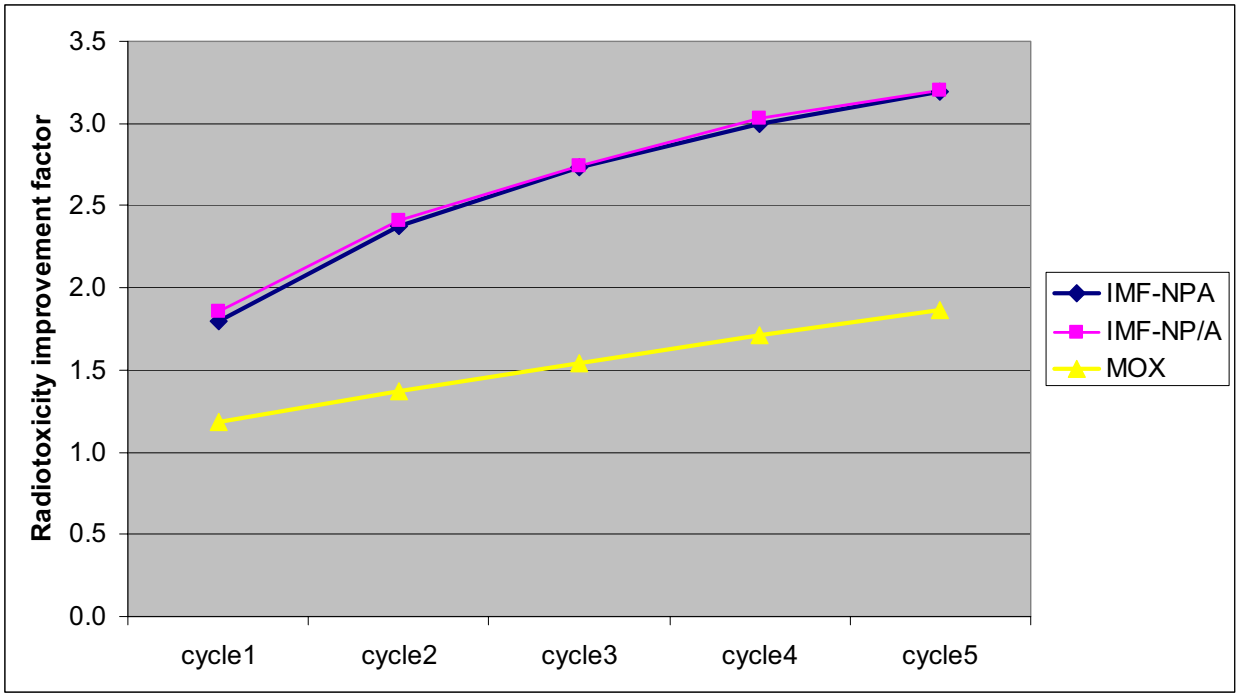

Figure 5-39. LTR improvement factors for multi-pass MOX and multi-pass IMF

Table 5-10 shows the results for multi-pass recycle cases in this study. Again, the results are quite similar to LTD.

Table 5-10. Multi-pass LTR results at 0.1% loss

	Targets (see Chapter 3)	MOX NpPuAm	IMF-NpPuAm	UOX/CFR symbiosis	MOX/CFR symbiosis	IMF/CFR symbiosis	UOX to BFR	IMF to BFR
		Throw away after 5 cycles		Equilibrium				
LTR improve-ment	100x at 1000 yr	1.86	3.19	>100	>100	>100	>100	>100

5.3 Proliferation Resistance

5.3.1 Quantity metrics

Figure 5-40 shows the Pu239 equivalent fraction of each fuel, which weights each isotope by their bare sphere critical mass relative to Pu239. There are five particular items to note, moving from left to right in the Figure. First, the Pu-239 equivalent fraction of the multi-pass MOX fuels steadily increases, this is because of Pu-238. Unfortunately, however, the Pu238 fraction is never high enough that Pu in the fuel ceases to be considered weapons-usable by IAEA [IAEA2001] definitions. Second, the Pu-239 equivalent fraction for single-pass IMF-NpPu and IMF-NpPuAm is high, by design! Third, the Pu239 equivalent fractions for multi-pass IMF are similar to once-through because they are mostly UOX. Less than 2% (by mass) of the multi-pass IMF blends is TRU. Fourth, the Pu239 equivalent fractions for the CFR cases are high. Five, the only cases where the Pu239 equivalent fraction is higher for the output fuel than the input fuel are the BFR cases, UOX-33, and UOX-51. In all others, it is literally true that recycling has the net effect of destroying weapons-usable material.

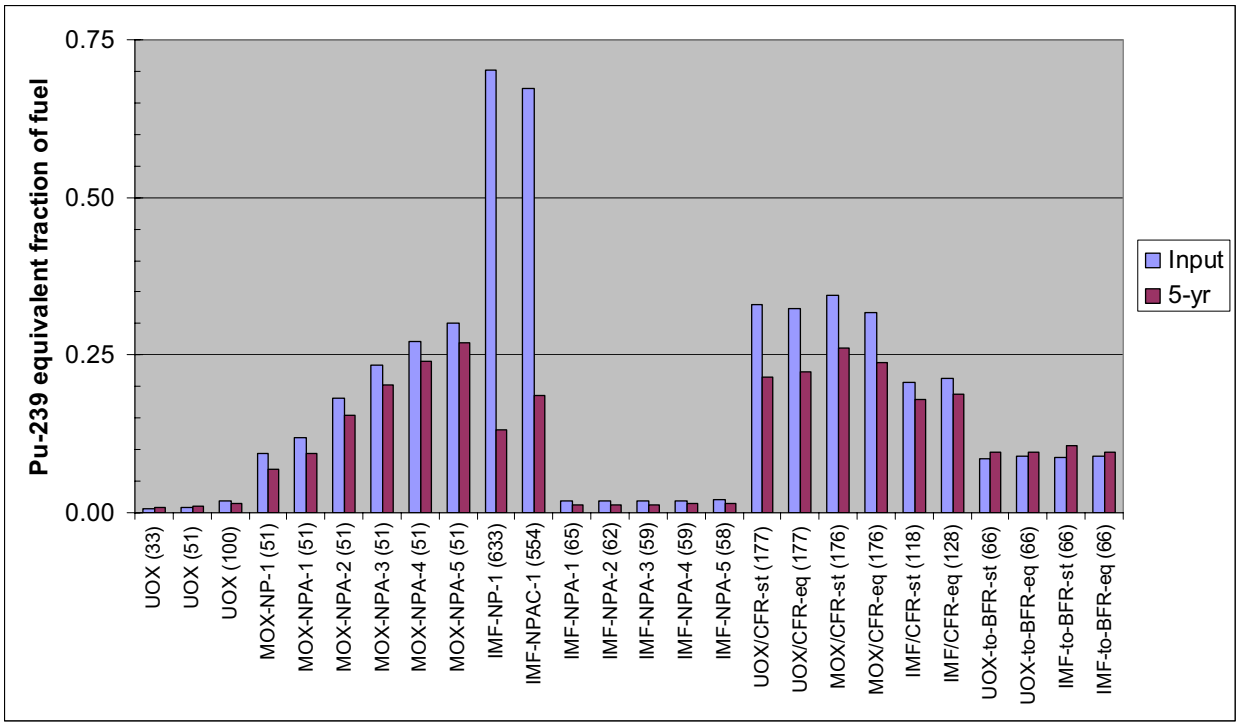

Figure 5-40. Pu239 equivalent fraction for fuels in this study

Figure 5-41 looks at the throughput of weapons-usable material in fresh fuel, again using Pu239-equivalent as the metric. As expected, the BFR system is the highest.

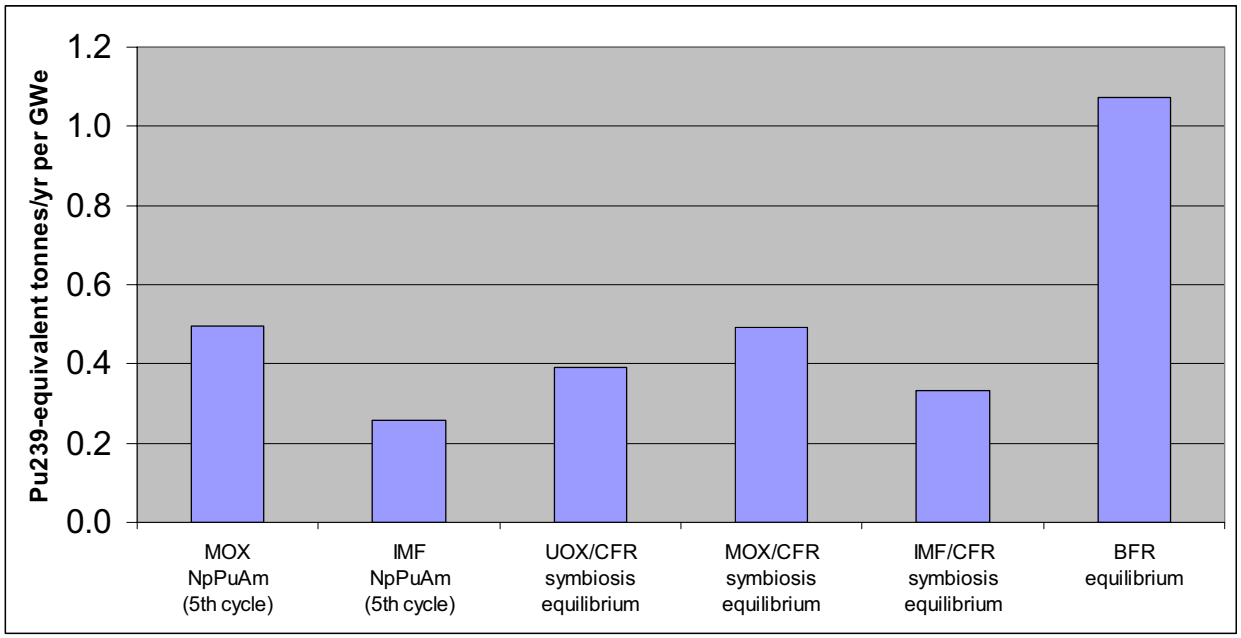

Figure 5-41. Throughput of Pu239-equivalent mass per year per GWe

Figure 5-42 shows the Pu fraction of each fuel. The trends are quite similar to Pu239 equivalent with the exception that the input value for once-through is zero here, but non-zero (because of U235) in Figure 5-30.

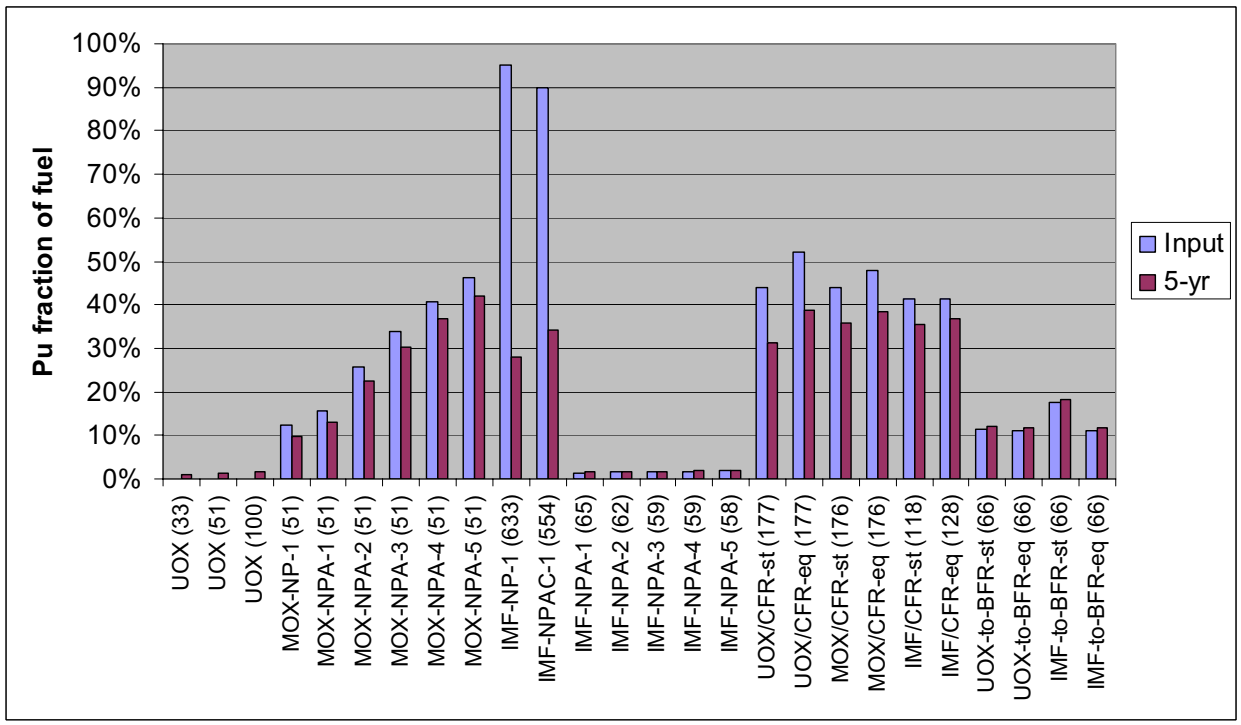

Figure 5-42. Pu fraction of fuels in this study

5.3.2 Quality metrics

Figure 5-43 shows the fraction of Pu239/Pu-total for the fuels in this study. The ratio is undefined for fresh once-through because there is no Pu. The input values for all fuels that directly use Pu from discharged UOX-51 is, of course, the same as the output value for UOX-51, 53%. The lowest Pu239/Pu fuel is IMF-CFR, but of course the IMF-CFR system requires full-core IMF in symbiosis with the CFR and the Pu239/Pu for fresh IMF is 53%.

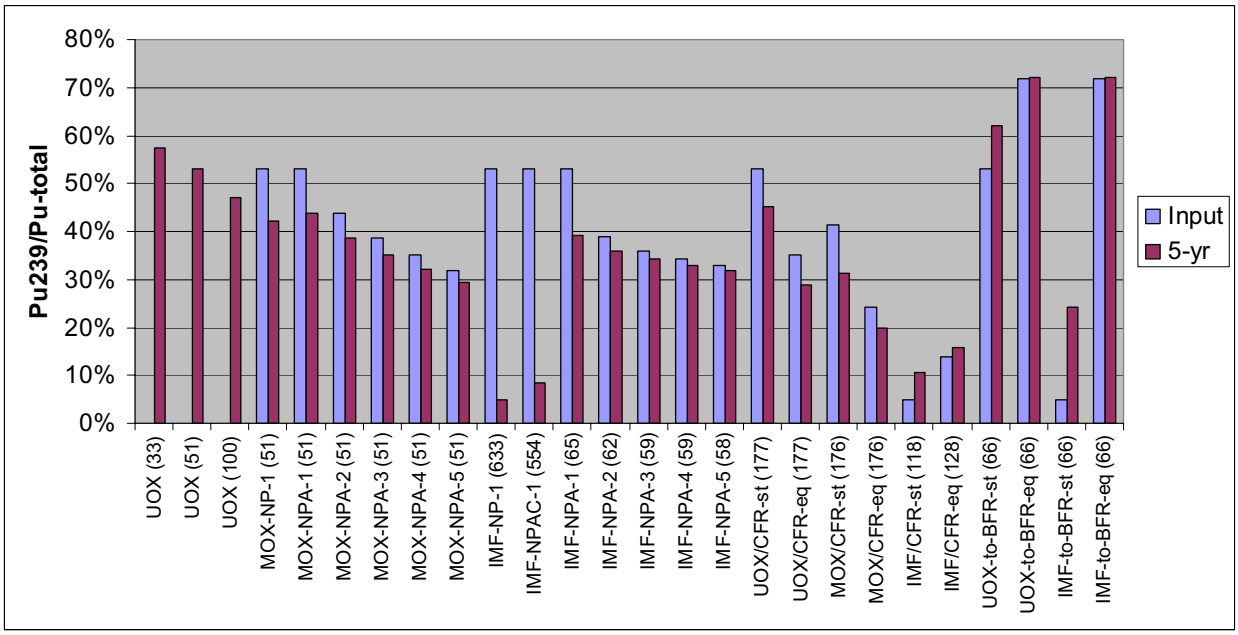

Figure 5-43. Pu239/Pu-total fractions for fuels in this study

5.3.3 Fuel Handling

Stillman [Stillman2004d] provided gamma and neutron results for 4.3-kg of spent UOX in a prototypical geometry. By analyzing his results, we obtain these "rules of thumb" for discharged fuels with fission products removed. Recall that a truly unshielded configuration, can allow high dose rates resulting from the TRU alone,[Hannum1995], but these gammas are shielded by even a thin wall container as demonstrated by Stillman. Only thin steel is needed to shield some of the actinide gammas, which is why the dose rates are relatively low (even for Am) compared to previous bare material evaluations.

Gamma	Am241	0.100 rem/hr
	Pu-RG	0.050 rem/hr
	Pu-WG	0.005 rem/hr
Neutron	Cm244	0.500 rem/hr
	Pu238	0.010 rem/hr
	Pu-WG	0.001 rem/hr

Figure 5-44 illustrates the expected gamma and neutron dose from different fuels (fresh or 5-year after discharge). There are several points to be made, remembering that this approach is only a crude estimate that neglects any changes in the important self-shielding effects (see Section 3.2.3).

In all cases, the true dose rate of discharged fuels is dominated by fission products. None of the TRU-based dose rates exceed 100 rem/hr; discharged fuels are much hotter. Indeed, it takes ~100 years before used once-through fuel decays to ~100 rem/hr. Since all fuels have roughly the same fission products, it will take all of them roughly the same time to drop to ~100 rem/hr. The discharge values in Figure 5-30 for the higher cases are dominated by Cm244 (18.1 year), so that the TRU component of dose rates will not stay near 100 rem/hr either. Therefore, the duration of "self-protection" (versus the 100 rem/hr value) does not appear likely to change significantly among fuels.

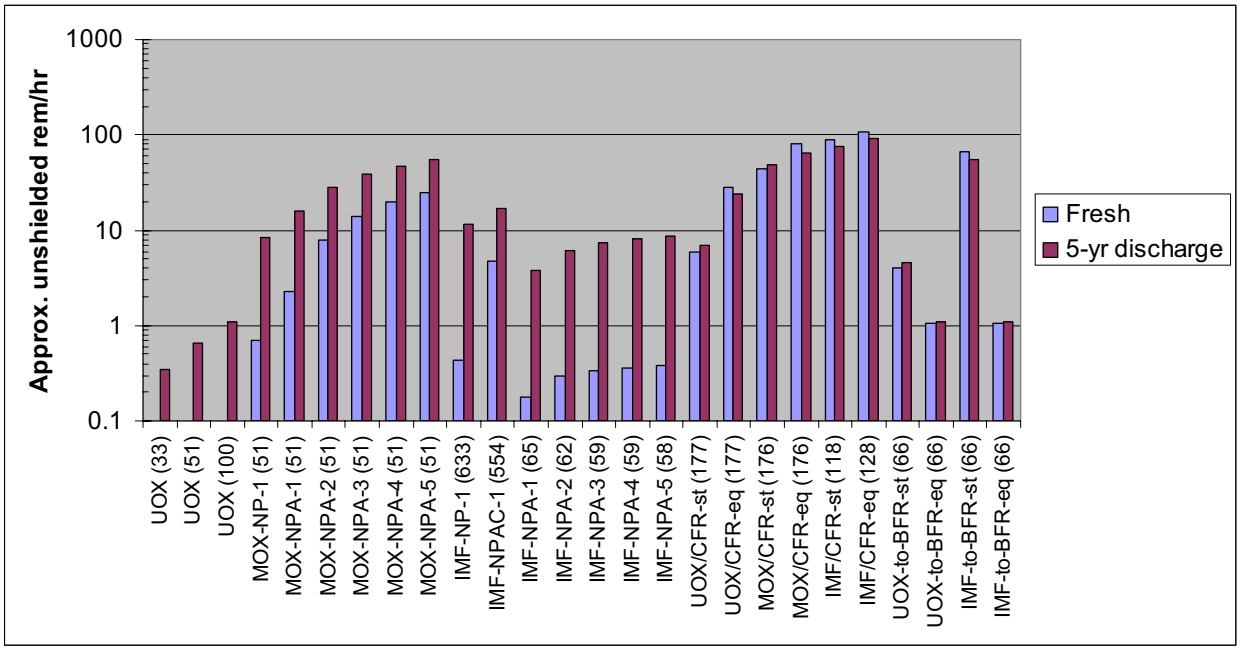

Figure 5-44. Crude approximation of unshielded dose rates for fuels in this study with fission products removed. We stress that these are crude, intended primarily to help compare options in a relative sense.

We now turn to fresh fuels. Within a factor of 3 up and down 1 rem/hr represents transition from hands-on/glovebox to fully remote fabrication. See also Appendix F. There are three types of fuels that may avoid fully remote fabrication.

1. First cycle fuels without Am, MOX-NpPu and IMF-NpPu. This is likely to be a **valid** conclusion as it matches international practice. Lacking recipes for multi-pass fuels without Am, we cannot validate whether this favorable result holds for subsequent recycles.
2. BFR fuels at equilibrium. The BFR spent fuel has an isotopic vector much more concentrated in Pu-239 than LWR spent fuel. Thus, Figure 5-34 shows that the dose rates for even the entire transuranic mix are quite low. However, the typical practice for fast reactor recycle is to process the fuel with very short cooling times (e.g., 2 years for pyroprocess in this study); this is done to reduce the total inventory contained in the fast reactor fuel cycle. These short cooling times would require remote fabrication, despite the favorable TRU dose characteristics.
3. Multi-cycle IMF-blended assemblies, with IMF-NpPuAm. The **validity of this finding is unclear**, but would seem unlikely. The crude estimates here indicate lower dose than for MOX-NpPu. The ¾ of each assembly that are UOX pins, of course, avoid fully remote fabrication. The IMF pins on the outer edge of each assembly contain all of the TRU. The mass of TRU in each assembly is small, less than 2%, in contrast to 100% for full-core IMF. That is what produces the result in Figure 5-30. These outer IMF pins will have relatively little shielding, so it would likely that the IMF pins and the final assembly will require remote fabrication.

4. Multi-cycle IMF-blended assemblies, with IMF-NpPu and separate Am targets. The difference from the preceding case is that the Am is not in the IMF-NpPu pins (which therefore are similar to the full-core IMP-NpPu pins), the Am is in 4 target pins (of 264 in the assembly), deeper within the assembly. It is quite possible that final assembly would not require fully remote fabrication. The UOX pins would be hands-on, the IMF-NpPu pins are possibly glovebox, and the Am targets would be remote fabrication.

The anticipated fabrication approach for the fuels in this study is therefore given in Table 5-11.

Table 5-11. Anticipated fabrication approach for the fuels in this study

Fuel type	Anticipated fabrication approach
UOX-31	Hands on
UOX-51	Hands on
UOX-100	Hands on
MOX-NpPu	Glovebox
MOX-NpPuAm	Remote
MOX-NpPuAmCm	Remote
IMF-NpPu (full core)	Glovebox
IMF-NpPuAm (full core)	Remote
IMF-NpPuAmCm (full core)	Remote
IMF-NpPuAm (blended core)	UOX pins – hands-on IMF pins – remote Final assembly –remote
IMF-NpPu with separate Am targets (blended core)	UOX pins – hands-on IMF pins – glovebox Am targets - remote Final assembly – remote
UOX/CFR symbiosis	Remote
IMF/CFR symbiosis	Remote
MOX/CFR symbiosis	Remote
UOX-to-BFR	Remote for short cooling time
IMF-to-BFR	Remote for short cooling time
MOX-to-BFR	Remote for short cooling time

5.4 Energy Recovery

Table 5-12 shows the uranium utilization improvement factor for single-pass cases; these are for the entire system, e.g., they account for the UOX needed to make MOX and IMF, neither of which use enriched uranium. Only IMF-NpPu meets the short-term objective of 1.15.

Table 5-12. Uranium Utilization Improvement factors (compared to UOX-51) for Single-Pass Cases

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPuAmCm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Uranium ore use improvement	1.15 short term 50x long term	0.88	0.97	1.09	1.07	1.15	1.14	1.13	1.14

Note that the uranium ore utilization for UOX-33 and UOX-100 are both lower than UOX-51. This agrees with past work. Hesketh and Robbins (BNFL, no date or conference ID available) concluded "A clear minimum in fuel cycle costs are seen at or below 55 GWd/MT" and that this conclusion is "robust against perturbations in the underlying assumptions" such as the price of uranium. Gregg and Worrall similarly concluded (Effect of Highly Enriched/Highly Burnt UO2 Fuels on Nuclear Design Parameters and Economics" ANFM 2003 Conference) that "there is no fuel cycle cost benefit in discharge burnups greater than 70 to 75 GWd/MT."

Figure 5-45 shows the uranium improvement factor for 5-pass MOX and IMF. The IMF-NpPuAm case has plateaued and peaked very slightly at cycle 4. The IMF-NpPu with Am target case does slightly better than the IMF-NpPuAm case.

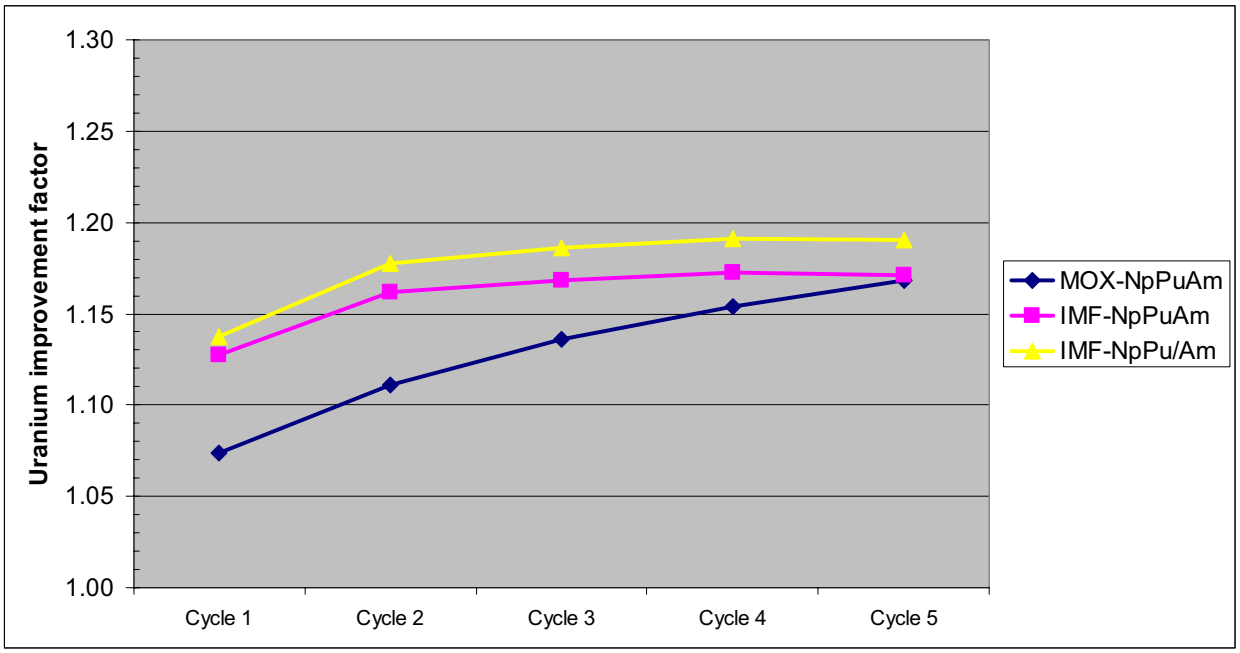

Figure 5-45. Uranium utilization factors for multi-pass IMF and MOX.

The equilibrium value for the CFR cases is determined simply by the fraction of UOX in the equilibrium fleets, with the results of 1.37 (UOX/CFR), 1.49 (MOX/CFR), and 1.38 (IMF/CFR).

The equilibrium value for the BFR cases is determined by the use of U238, not U235. At equilibrium, BFR has 10.57 tonnes-U/yr input (per GWe), 9.64 tonnes-U/yr output, requiring a makeup of 0.93 tonnes/yr. UOX-51 requires 18.96 tonnes-enriched U/yr (per GWe). At 4.3% enrichment and 0.2% tailings, 7.88 tonnes-U-ore needed to make 1 tonne of enriched U, so that UOX-51 requires 149 tonnes-U-ore/yr. Thus, the uranium utilization improvement factor is 149/0.93 = 160.

5.5 System Management (economics, safety, etc.)

From a system management perspective, the key is that the throughput of recirculating TRU varies so much among concepts, see section 5.1.

- The TRU throughput for the multi-pass full-core MOX cases in this study is always higher than for multi-pass IMF. At the 5th cycle, the values are 0.94 and 0.34 tonnes-TRU/yr per GWe.
- The TRU throughput for CFR cases is 0.75 (UOX-CFR), 1.04 (MOX-CFR), and 0.79 (IMF-CFR). These numbers, of course, include the UOX, MOX, and IMF required to keep the CFR going. However, for integrated FR-separation-fabrication plants, most of these mass flows would not have the leave the site.
- The TRU throughput for BFR is the highest, 1.45 tonnes-TRU/yr per GWe at equilibrium. However, for integrated FR-separation-fabrication facilities, as in the Integral Fast Reactor concept, only 7% of this would have to leave the site.

Worker dose and public risk throughput the fuel cycle will, to some degree, scale with the throughput of the "nastier" isotopes. In this regard, the rank order is roughly short-lived fission products, Cm, Am, Pu, Np, U. The flux of fission products is basically constant (per GWe). The flux of uranium does not change much either, except for ultra-high-burnup. The flux of the TRU elements changes significantly; see section 5.1 for details.

Finally, we note that the percent of new fuels and new reactors vary significantly. These can be seen directly from the blending Figures in section 5.1. Of course, the worst case are the BFR cases, with 100% new fuels and 100% new reactors. The pure thermal cases would use existing reactors, possibly with minor variations. The more than 80% of UOX fuel in these systems is of course unchanged, but the approximately 20% of IMF or MOX would be new. The UOX-CFR system at equilibrium is 27% new CFR fuel, 73% UOX. The MOX-CFR system is 20% new CFR fuel, 7% new MOX fuel, and 73% UOX. The IMF-CFR system is 19% new CFR fuel, 11% new IMF fuel, and 70% UOX.

6. SCREENING ANALYSES OF DEVELOPMENT TREES

Previous chapters contained key fuel cycle decisions (Chapter 2), AFCI objectives and metrics (Chapter 3), the alternatives (Chapter 4), and "static" analyses of the various fuel and reactor options and how they can be combined (Chapter 5). This Chapter is the first of three that address timing and dynamics. This Chapter approaches development "trees" using complex spreadsheets, carrying simulations to 2100. It concentrates on basic metrics such as mass to repository and, equally important, on semi-qualitative considerations such as what new technologies are needed or abandoned in each scenario. Chapters 7 and 8 analyze development trees using the DYMOND system dynamic model.

This Chapter has two parts. First is a broad screening analysis of ~160 scenarios. Six trees start in 2025: (1) continue once-through, (2) start IMF-NpPuAm, (3) start MOX-NpPu, (4) start MOX-NpPuAm, (5) deploy a few consumer fast reactors, and (6) deploy a few breeder fast reactors. Each tree has as many as 6 branches in 2040 denoting exiting technologies, shifting technologies, or adding technologies. There are additional branches in 2060 and 2080. Some branches are not studied, e.g., if a technology is exited in 2040, re-start in 2060 or 2080 is not analyzed. In all cases nuclear generation was level – that is, new types of reactors could be introduced only as existing LWRs were retired. Top-level conclusions are as follows:

1. The system has high inertia, it takes decades to affect major change. This may be overcome in a growth situation in which new reactors are needed to meet demand. A growth rate as low as 1% would introduce one new reactor each year.
2. In postulating development paths, consider the potential number of new technologies to be implemented at any given time, the fewer the better.
3. Even spacing decisions points 20 years apart, it is difficult to observe the impact of the n^{th} decision before it would be necessary to make the $n+1^{th}$ decision so that its implementation could begin 20 years after the implementation of the n^{th} decision. There is high potential to "out drive our headlights."
4. The two options with the most leverage (ie. the ability to change the amount of Pu in the system are IMF and fast reactors. MOX has less leverage.
5. There are more options to meet repository heat objectives than there are to meet uranium resources objectives.
    a. A screening analysis of 155 branches showed 5% had less than 10% of total energy produced from recycled fuel, 60% between 10 and 20% from recycled fuel, 31% between 20% and 50%, and 4% (six) over 50%. The six with over 50% start thermal or fast recycling in 2025 with fast recycling in 2040. The ones with less than 10% abandoned recycling.
    b. A similar screening analysis showed that (for 0% growth), 33% of the branches fill the repository less than 10%, 27% fill between 10 and 100% of the repository, and 40% exceed a 129,000 tonne repository. The first group uses fast reactors, particularly in the later years of the century. The last group rejects recycling sooner or later.

The second part is a more detailed screening of 36 scenarios, 6 branches for each of the 6 trees. These are the same 36 trees analyzed with DYMOND in Chapter 7. Top-level conclusions are as follows:

1. Don't use breeder FRs when the nuclear capacity is declining. SNF inventories increase with harmful results.
2. Be cautious in using IMF in high growth if you want some TRU to start up breeder FRs.
3. Starting up fast reactors commits their operation for a long time; loading fuel (IMF/MOX) lets you out in a very few years. This suggests that it will be difficult to make a decision to start a major rampup to fast reactors. Instead, a cautious introduction of fast reactors may be the only way to start. The only way to get full use of the TRU is to have fast reactors in the system. The only way to

minimize the number of repositories in the long term (if there is any growth) is to include fast reactors in the system.

4. So long as thermal reactors and burner FRs are used, uranium resources will eventually be limiting. Only by using breeder FRs can recycle fuel enable long-term sustainable growth.

6.1 Screening analysis of 160 scenarios

This section presents the results of developing a decision tree for which key decisions are made in 2025, 2040, 2060 and 2080 regarding the deployment of certain AFCI and Gen IV technologies. The data are provided at the end of section 6.1. The analysis, for the most part, assumes a "Level Generation" nuclear future – one in which the generation of nuclear energy remains constant throughout the 75-year period. The Level Generation future is one which significantly inhibits further development of nuclear power, as growth in generation generally enhances flexibility, agility and robustness.

There are nearly 160 different trajectories through the decision space, out of a potential 600 if all combinations were considered. At the end of section 6.1, Table 6-1 provides summary results. The first four columns describe the trajectory, identifying the decisions made in 2025, 2040, 2060 and 2080. The fifth column provides a "point score" for "Robustness," for which one point is awarded if reprocessing/recycling is enabled, another if fast reactors are enabled, and another if the Monitored Geologic Repository (MGR) is not full, and finally another if the repository is nearly empty. The sixth column provides an evaluation for "Agility" by indicating technologies available quickly, of which fast and thermal technologies are indicated separately. The seventh column shows the percent of energy generated from recycle fuel over the 75-year span of the decision tree. The eighth column shows the amount of SNF reprocessed in "system years," where a system year is the amount of SNF generated by the current 100 reactors in a year (approximately 2100 tonnes). The ninth column has the equivalent amount of SNF in the MGR (or other repository). The SNF-equivalent has been calculated using the Wigeland & Bauer [Wigeland2004a] loadings for recycle fuel and a 5% residue for fast reactor HLW. Fast reactors are presumed to consume 9 thermal reactor fuel loads of SNF. A numeric result gives the number of 129,000 tonnes-MGRs (in other words, a 1.3 means 1.3 x 129,000 or 167,700 tonnes) needed to handle the SNF. "OK" means the MGR is not full, and "E'er" means that it is less than 10% full. Note that the crude algorithms used to estimate the equivalent loading of the MGR are not applicable to situations in which there is an early introduction and then abandonment of the fast reactor technology.

Generally, the YMP is not filled only if fast reactors are used (one successful trajectory is all IMF). Generally, the more robust trajectories are those that involve fast reactors after 2060. The most agile trajectories include thermal technologies after 2060. This particular result requires some further comment regarding the value of agility. While a technology is being introduced, agility provides the ability to react quickly to "bad news." Thus, during the introduction of the fast reactor, it would be good to be able to switch back to thermal reactors. But as the fast reactors from the first generation are being retired, such agility is of very little value. Consequently, a low agility score in 2100 may be less important than such a score during the first four decades immediately after deciding to deploy fast reactors. The greatest energy production from recycle fuel occurs when the greatest use of fast reactors occurs. The most flexible trajectory, balancing these somewhat disparate results, would seem to argue for an early (2025) decision to introduce both a hermal recycle technology (either IMF or MOX, or both) and fast reactor technology fueled with recycled TRU. (The higher flexibility of this "dual-path" approach would likely come at the expense of higher R&D and deployment costs.) After a few decades with this "dual-path" approach, if nuclear growth is high, only fast reactors would be deployed. This approach capitalizes on the existing knowledge base for thermal reactors and maintains agility while FR technology is introduced and "proved," and finally comes to rely on a technology that promises to minimize the requirement for MGR space.

These results can be interpreted in term of "opening doors" and "closing doors," where an open door implies an opportunity and a closed door implies an opportunity lost. Within this metaphor, R&D is the "key" to opening doors. This key is time-sensitive, and after a decade or more without use, it loses its value because other technologies advance and the "old" R&D becomes increasingly anachronistic. The first door to be opened is recycling. Recycle implies reprocessing and a decision to maximize the capability of the MGR. This door is opened by the Secretarial Decision on a second repository in 2007-10 and the successful development of appropriate reprocessing, waste form and recycle fuel fabrication technologies. If that door is closed by a decision to continue to use the once-thru cycle and to initiate a second repository, there will be no justification for continued R&D and no development of "the key," so the door will remain closed, probably forever. If the door is opened, thermal recycle will increase the capacity of the MGR by 50-100%, based on results from Wigeland & Bauer. Perhaps more importantly, the demand for "permanent" space in the MGR will be dramatically reduced for a few decades until the final recycle batches must be emplaced, allowing time to work off the current SNF backlog. If IMF is the recycle fuel of choice, the final recycle batches will require emplacement around 2050 to 2070, whereas with MOX fuel that will not occur until after about 2090.

The second door to be opened is that representing the fast reactor. The "key" for that door is being jointly developed by the Gen IV and AFCI programs – Gen IV considering the reactor and ancillary systems and AFCI the fuel cycle. Within the door metaphor, this is a door that must be opened slowly because a new reactor technology is introduced and time will be required to gain operating experience and to implement "lessons learned" from the early years of operation. The promise of the fast reactor is that it can consume all the TRU materials present in SNF, so that the HLW consists primarily of relatively short-lived fission products. This has the potential to increase the capacity of the MGR by a factor of 20 to 40, or perhaps even more, especially when considering that the time required to fill the MGR will be comparable to or greater than the decay time of the HLW. The 21st century offers a potentially unique opportunity for opening this door: 1) R&D on fast reactors will be "fresh" so that it can be readily implemented in an operational setting; and 2) economic fuel resources for thermal reactors will still be plentiful, so that a ready supply of TRU materials for fast reactor fuels development and demonstration of reactor operation using these fuels will be facilitated. These factors would seem to argue for introduction of "Burner" fast reactors - fast reactors that "feed" on the TRU produced by recycling in thermal reactors. Keeping a mix of thermal and fast reactors operating for several decades also allows for agility in the event of a problem in the full deployment of fast reactors. If the fast reactor door is not opened, the main problem confronting nuclear power today – the lack of a suitable approach for disposal of spent fuel – will recur in a few decades. Assuming the MGR is expanded to about 129,000 tonnes-iHM, continued operation in the once-thru mode until around 2035 can be accommodated, operation in a recycle mode can be accommodated until somewhere between 2065 and 2090 (at which time the commitment represented by irradiation of the final recycle will overwhelm the MGR capacity), and operation with fast reactors is possible for the foreseeable future. Thus, closing the door on fast reactors simply delays the inevitable decline of nuclear power and provides only a stop-gap remedy to our dependence on foreign countries for energy resources.

6.1.1 Decision Guidelines

Selection of "No Recycle" precludes recycle options thereafter (implies failure of recycle concept). In 2025 this may occur because the 2007-2010 Secretarial decision was to build a second repository or because AFCI was unable to show economic recycling in compliance with US requirements. Even if recycling is beneficial, a decision to forego it will halt research and a 15 year hiatus will effectively close the door on that option.

Selection of a FR (fast reactor) option ("Breeder FR" or "Burner FR") followed by selection of TR (thermal reactor) option (IMF or MOX) precludes subsequent FR options (implies failure of FR option).

A decision to move to IMF after MOX implies a serious desire to destroy TRU. It involves "wasting" some of the "recycle energy" inherent in the MOX recycle (unless the MOX TRU can be re-fabricated into IMF). Therefore, only "No Recycle" and "IMF" are acceptable decisions after such a transition. In very general terms, IMF and MOX fuels are similar in that they consume TRU and can presumably be placed in existing reactors. They are also similar in that their use is limited by the amount of discharge TRU available to 15-20% of the fuel (or reactors, if special reactors were designed) and they do not completely destroy the TRU, as a fast reactor would. The IMF fuel produces the bulk of the "extra" energy and consumes the bulk of the TRU in a single cycle, and requires only two cycles to reach its end-point. The MOX fuel produces slightly more "extra" energy (17% compared to 15%), but requires five cycles to complete.

6.1.2 Reactor Introduction Constraints

There is a continuing demand for nuclear power – level generation or some growth.

Reactor life is 40-60 years; 60 for LWRs, 40 for first generation fast reactors. Only LWRs and FRs are considered.

All existing NPPs obtain 20-year extensions to their licenses and none shut down early. On that basis, existing plant retirements occur as follows.

Time Period	Capacity
Before 2035	6 GWe
2036 – 2040	31 GWe
2041 – 2045	14 GWe
2046 – 2050	36 GWe
2051 – 2055	10 GWe
2056 - 2060	2 GWe

New NPPs are added without premature shutdown of older plants. That means no new plants are introduced before about 2033 in the "Level Generation" nuclear future, and only about 37 GWe can be introduced in the 2025-2040 time period. After 2059, no plants are introduced until 2073, and probably not until 2093.

6.1.3 Other Analysis Assumptions

SNF accumulated by 2015 is 80,000 tonnes. The energy content/burnup is equivalent to 60,000 tonne at 50 GW-day/tonne. SNF is accumulated thereafter at the rate of 21 tonne/NPP per year with an assumed burnup of 50 GW-day/tonne. All fuel assemblies produce the same energy; therefore all fuels produce the same amount of "standard" SNF.

The compositions for MOX and IMF are based on analyses through May 2005. Therefore, the IMF cases are all "full core." The multi-pass IMF blended core case calculated later [Goldmann2005] was not analyzed.

Recycle options using IMF and MOX are as described by Wigeland & Bauer. Burner fast reactors (reactors fueled by the TRU in LWR SNF) require 10 assemblies of SNF to produce one new FR assembly. One tonne of FR SNF (equivalent to 50 GW-day) yields 0.05 tonne HLW (equivalent SNF).

LWRs are capable of substituting IMF/MOX assemblies up to about 20% of their core loading.

Note that IMF-NpPuAm produces about 115% the energy of the original uranium fuel, and does so in two cycles (23 years). MOX-NpPuAm produces about 117%, but it takes 60 years to capture it.

Although there is a significant cost difference between MOX-NpPu and MOX-NpPuAm, in this analysis, there seems to be little difference, so no sequence begins with MOX-NpPu.

The reprocessing plant is assumed to have a capacity of 5000 tonne/yr.

6.1.4 Definitions

Robustness: Allows for growth of nuclear power and provides for options to be exercised at a later date. Also accommodates some "upsets. Enabling reprocessing and recycle is critical. However, unless fast reactors are also enabled, the MGR will eventually fill. Fast reactors are also somewhat less sensitive to variations in fuel composition, so they can accommodate a greater variety of fuels. Timing is also important, and if the MGR is full (so a second is required), or filling, then additional pressure will be applied to find acceptable trajectories through "option space."

Robustness	0 to 4 points as follows:
	1 point if recycling enabled
	1 point if fast reactors enabled
	1 point if MGR is not full (at 129,000 tonnes)
	1 point if MGR is less than 10% full

If uranium resources are severe:

Uranium	0 to 2 points as follows:
	0 point if no FR
	1 point if Burner FR
	2 point if Breeder FR

If proliferation precludes recycling:

Proliferation	0 to 2 points as follows:
	0 point = shutdown reactors
	1 point = modify reactors
	2 point = no change required

Agile: Is capable of being implemented quickly, without major disruption to on-going operations. This requires that technologies be available. Fuel changes are easier to accommodate than reactor changes. However, fast reactors can probably handle fuel changes better than thermal reactors. Technologies that have failed or have not been used for 40 years are considered unavailable. Thus, in a sequence like Fast Reactor followed by IMF-NpPuAm, the fast reactor technology is considered unavailable. Similarly, a sequence such as IMF-NpPuAm followed by IMF NpPuAm makes the MOX-NpPuAm technology unavailable because of the 40 year hiatus in its use.

Agile	Technologies quickly available
	Th = thermal (IMF or MOX)
	F = fast (any fuel)

Recycle-Fuel Energy: % of energy from recycled fuel

SNF Reproprocessed: Reactor-years of SNF reprocessed/100

Equivalent SNF:

Equiv SNF	1 point if recycling enabled
	1 point if fast reactors enabled
	1 point if MGR is not full (at 129,000 tonnes)
	1 point if MGR is less than 10% full

6.1.5 Results

Table 6-1. Results of Screening Analysis of 160 Scenarios

2025	2040	2060	2080	Robust	2070 If U severe	2070 If prol precludes recycling	2070	Agile	Recyc Energy Percent	SNF Repro'ed Years	Equiv SNF 129,000 tonne-YMPs
Once-thru	No Recycle	No Recycle	No Recycle	0				None	0.00%	0	2.00
IMF-NPA	No Recycle	No Recycle	No Recycle	0	0	1.5		None	4.94%	36	1.68
	IMF-NPA	No Recycle	No Recycle	0	0	1.5		None	12.96%	83	1.28
		IMF-NPA	No Recycle	0	0	1.5		None	16.26%	103	1.09
		IMF-NPA	IMF-NPA	2	0	1.5		1 Th	19.60%	123	OK
		IMF-NPA	MOX-NPA	1	0	1.5		2 Th	17.93%	123	1.13
		IMF-NPA	Burner FR	3	0	1.5		Th + F	20.17%	123	OK
		IMF-NPA	Breeder FR	3	0	1.5		Th + F	20.70%	123	OK
		MOX-NPA	No Recycle	0	0	2		None	14.38%	103	1.34
		MOX-NPA	IMF-NPA	2	0	2		2 Th	19.96%	123	OK
		MOX-NPA	MOX-NPA	1	0	2		1 Th	17.71%	123	1.27
		MOX-NPA	Burner FR	2	0	2		Th + F	18.27%	123	1.18
		MOX-NPA	Breeder FR	2	0	2		Th + F	18.79%	123	1.10
		Burner FR	No Recycle	0	1	1		None	16.26%	103	1.09
		Burner FR	IMF-NPA	2	1	1		1 Th	19.60%	123	OK
		Burner FR	MOX-NPA	1	1	1		1 Th	17.93%	123	1.13
		Burner FR	Burner FR	3	1	1		Th + F	20.17%	123	OK
		Burner FR	Breeder FR	3	1	1		Th + F	20.70%	123	OK
		Breeder FR	No Recycle	0	2	0		None	16.26%	103	1.09
		Breeder FR	IMF-NPA	2	2	0		1 Th	19.60%	123	OK
		Breeder FR	MOX-NPA	1	2	0		1 Th	17.93%	123	1.13
		Breeder FR	Burner FR	3	2	0		Th + F	20.17%	123	OK
	V	Breeder FR	Breeder FR	3	2	0		F	20.70%	123	OK

2025	2040	2060	2080	Robust	2070 If U severe	2070 If prol precludes recycling	2070	Agile	Recyc Energy Percent	SNF Repro'ed Years	Equiv SNF 129,000 tonne-YMPs
|	MOX-NPA	No Recycle	No Recycle	0				None	10.62%	83	1.58
| |	|	IMF-NPA	No Recycle	0				None	14.75%	103	1.39
| |	|	IMF-NPA	IMF-NPA	2				1 Th	14.83%	123	OK
| |	|	MOX-NPA	No Recycle	0				None	14.11%	103	1.49
| |	|	MOX-NPA	IMF-NPA	1				2 Th	18.04%	123	1.32
| |	|	MOX-NPA	MOX-NPA	1				1 Th	18.18%	123	1.37
| |	|	MOX-NPA	Burner FR	3				Th + F	13.26%	123	OK
| |	|	MOX-NPA	Breeder FR	3				Th + F	13.92%	123	OK
| |	|	Burner FR	No Recycle	0				None	14.11%	103	1.49
| |	|	Burner FR	IMF-NPA	1				1 Th	18.04%	123	1.32
| |	|	Burner FR	MOX-NPA	1				1 Th	16.02%	123	1.49
| |	|	Burner FR	Burner FR	2				Th + F	18.74%	123	1.28
| |	|	Burner FR	Breeder FR	2				Th + F	19.25%	123	1.20
| |	|	Breeder FR	No Recycle	0				None	14.11%	103	1.49
| |	|	Breeder FR	IMF-NPA	1				1 Th	18.04%	123	1.32
| |	|	Breeder FR	MOX-NPA	1				1 Th	16.02%	123	1.49
| |	V	Breeder FR	Breeder FR	2				F	19.25%	123	1.20
| |	Burner FR	No Recycle	No Recycle	1				None	15.67%	83	OK
| |	|	IMF-NPA	No Recycle	1				None	19.72%	103	OK
| |	|	IMF-NPA	IMF-NPA	2				1 Th	23.09%	123	OK
| |	|	MOX-NPA	No Recycle	1				None	17.28%	103	OK
| |	|	MOX-NPA	IMF-NPA	2				2 Th	21.42%	123	OK
| |	|	MOX-NPA	MOX-NPA	2				1 Th	20.69%	123	OK
| |	|	Burner FR	No Recycle	1				None	22.29%	103	OK
| |	|	Burner FR	IMF-NPA	3				1 Th	26.32%	123	E'er
| |	|	Burner FR	MOX-NPA	2				1 Th	24.14%	123	OK
| |	|	Burner FR	Burner FR	4				Th + F	28.39%	123	E'er
| |	|	Burner FR	Breeder FR	4				Th + F	33.88%	123	E'er
| |	|	Breeder FR	No Recycle	1				None	22.29%	103	OK
| |	|	Breeder FR	IMF-NPA	3				1 Th	26.32%	123	E'er
| |	|	Breeder FR	MOX-NPA	2				1 Th	24.14%	123	OK
| |	V	Breeder FR	Breeder FR	4				F	33.88%	123	E'er
|	Breeder FR	No Recycle	No Recycle	2				None	21.41%	83	E'er
| |	|	IMF-NPA	No Recycle	2				None	25.30%	103	E'er
| |	|	IMF-NPA	IMF-NPA	3				1 Th	29.42%	123	E'er
| |	|	MOX-NPA	No Recycle	2				None	23.42%	103	E'er
| |	|	MOX-NPA	IMF-NPA	3				2 Th	27.75%	123	E'er
| |	|	MOX-NPA	MOX-NPA	3				1 Th	27.02%	123	E'er
| |	|	Breeder FR	No Recycle	2				None	39.51%	103	E'er
| |	|	Breeder FR	IMF-NPA	3				1 Th	44.06%	123	E'er
| |	|	Breeder FR	MOX-NPA	3				1 Th	41.89%	123	E'er
V	V	Breeder FR	Breeder FR	4				F	59.30%	123	E'er

2025	2040	2060	2080	Robust	2070 If U severe	2070 If prol precludes recycling	2070	Agile	Recyc Energy Percent	SNF Repro'ed Years	Equiv SNF 129,000 tonne-YMPs
MOX-NPA	No Recycle	No Recycle	No Recycle	0				None	2.85%	36	1.83
|	IMF-NPA	No Recycle	No Recycle	0				None	10.97%	83	1.52
| |	|	IMF-NPA	No Recycle	0				None	14.65%	103	1.29
| |	|	IMF-NPA	IMF-NPA	1				1 Th	18.17%	123	1.09
| |	|	IMF-NPA	Burner FR	2				Th + F	18.74%	123	1.00
| |	|	IMF-NPA	Breeder FR	3				Th + F	19.27%	123	OK
| |	|	Burner FR	No Recycle	0				None	14.65%	103	1.29
| |	|	Burner FR	IMF-NPA	1				1 Th	18.17%	123	1.09
| |	|	Burner FR	Burner FR	2				Th + F	18.74%	123	1.00
| |	|	Burner FR	Breeder FR	3				Th + F	19.27%	123	OK
| |	|	Breeder FR	No Recycle	0				None	14.65%	103	1.29
| |	|	Breeder FR	IMF-NPA	1				1 Th	18.17%	123	1.09
| |	V	Breeder FR	Breeder FR	3				F	19.27%	123	OK
| |	MOX-NPA	No Recycle	No Recycle	0				None	9.76%	83	1.67
| |	|	IMF-NPA	No Recycle	0				None	14.06%	103	1.46
| |	|	IMF-NPA	IMF-NPA	1				1 Th	17.65%	123	1.24
| |	|	MOX-NPA	No Recycle	0				None	14.11%	103	1.52
| |	|	MOX-NPA	IMF-NPA	1				2 Th	18.04%	123	1.34
| |	|	MOX-NPA	MOX-NPA	1				1 Th	18.31%	123	1.39
| |	|	MOX-NPA	Burner FR	2				Th + F	18.87%	123	1.30
| |	|	MOX-NPA	Breeder FR	2				Th + F	19.38%	123	1.22
| |	|	Burner FR	No Recycle	0				None	14.11%	103	1.52
| |	|	Burner FR	IMF-NPA	1				1 Th	18.04%	123	1.34
| |	|	Burner FR	MOX-NPA	1				1 Th	18.31%	123	1.39
| |	|	Burner FR	Burner FR	2				Th + F	18.87%	123	1.30
| |	|	Burner FR	Breeder FR	2				Th + F	19.38%	123	1.22
| |	|	Breeder FR	No Recycle	0				None	14.11%	103	1.52
| |	|	Breeder FR	IMF-NPA	1				1 Th	18.04%	123	1.34
| |	|	Breeder FR	MOX-NPA	1				1 Th	16.02%	123	1.52
| |	V	Breeder FR	Breeder FR	2				F	19.38%	123	1.22
| |	Burner FR	No Recycle	No Recycle	0				None	12.50%	83	1.19
| |	|	IMF-NPA	No Recycle	1				None	16.98%	103	OK
| |	|	IMF-NPA	IMF-NPA	2				1 Th	20.66%	123	OK
| |	|	MOX-NPA	No Recycle	0				None	15.18%	103	1.17
| |	|	MOX-NPA	IMF-NPA	2				2 Th	19.40%	123	OK
| |	|	MOX-NPA	MOX-NPA	1				1 Th	19.31%	123	1.05
| |	|	Burner FR	No Recycle	1				None	19.94%	103	OK
| |	|	Burner FR	IMF-NPA	2				1 Th	24.07%	123	OK
| |	|	Burner FR	MOX-NPA	2				1 Th	22.45%	123	OK
| |	|	Burner FR	Burner FR	4				Th + F	26.84%	123	E'er
| |	|	Burner FR	Breeder FR	4				Th + F	32.18%	123	E'er
| |	|	Breeder FR	No Recycle	1				None	19.94%	103	OK
| |	|	Breeder FR	IMF-NPA	2				1 Th	24.07%	123	OK
| |	|	Breeder FR	MOX-NPA	2				1 Th	22.45%	123	OK
| |	V	Breeder FR	Breeder FR	4				F	32.18%	123	E'er
| |	Breeder FR	No Recycle	No Recycle	2				None	25.71%	83	E'er
| |	|	IMF-NPA	No Recycle	2				None	31.00%	103	E'er
| |	|	IMF-NPA	IMF-NPA	3				1 Th	35.06%	123	E'er
| |	|	MOX-NPA	No Recycle	2				None	28.74%	103	E'er
| |	|	MOX-NPA	IMF-NPA	3				2 Th	33.39%	123	E'er
| |	|	MOX-NPA	MOX-NPA	3				1 Th	27.91%	123	E'er
| |	|	Breeder FR	No Recycle	2				None	51.29%	103	E'er
| |	|	Breeder FR	IMF-NPA	3				1 Th	56.36%	123	E'er
| |	|	Breeder FR	MOX-NPA	3				1 Th	54.48%	123	E'er
V	V	Breeder FR	Breeder FR	4				F	77.32%	123	E'er

2025	2040	2060	2080	Robust	2070 If U severe	2070 If prol precludes recycling	2070	Agile	Recyc Energy Percent	SNF Repro'ed Years	Equiv SNF 129,000 tonne-YMPs
Limited FR	No Recycle	No Recycle	No Recycle	0				None	1.09%	36	1.30
|	IMF-NPA	No Recycle	No Recycle	1				None	6.57%	83	OK
|	|	IMF-NPA	No Recycle	1				None	13.03%	103	OK
|	V	IMF-NPA	IMF-NPA	2				1 Th	16.85%	123	OK
|	MOX-NPA	No Recycle	No Recycle	0				None	5.90%	83	1.16
|	|	IMF-NPA	No Recycle	0				None	11.14%	103	1.02
|	|	IMF-NPA	IMF-NPA	2				1 Th	15.18%	123	OK
|	|	MOX-NPA	No Recycle	0				None	10.49%	103	1.13
|	|	MOX-NPA	IMF-NPA	2				2 Th	14.94%	123	OK
|	V	MOX-NPA	MOX-NPA	1				1 Th	15.09%	123	1.03
|	Burner FR	No Recycle	No Recycle	2				None	6.43%	83	E'er
|	|	IMF-NPA	No Recycle	2				None	12.58%	103	E'er
|	|	IMF-NPA	IMF-NPA	3				1 Th	17.05%	123	E'er
|	|	MOX-NPA	No Recycle	1				None	10.32%	103	OK
|	|	MOX-NPA	IMF-NPA	3				2 Th	15.37%	123	E'er
|	|	MOX-NPA	MOX-NPA	2				1 Th	14.79%	123	OK
|	|	Burner FR	No Recycle	2				None	11.43%	103	E'er
|	|	Burner FR	IMF-NPA	3				1 Th	16.91%	123	E'er
|	|	Burner FR	MOX-NPA	3				1 Th	14.89%	123	E'er
|	|	Burner FR	Burner FR	4				Th + F	14.44%	123	E'er
|	|	Burner FR	Breeder FR	4				Th + F	19.91%	123	E'er
|	|	Breeder FR	No Recycle	2				None	11.43%	103	E'er
|	|	Breeder FR	IMF-NPA	3				1 Th	16.91%	123	E'er
|	|	Breeder FR	MOX-NPA	3				1 Th	14.89%	123	E'er
|	V	Breeder FR	Breeder FR	4				F	19.91%	123	E'er
|	Breeder FR	No Recycle	No Recycle	2				None	15.00%	83	E'er
|	|	IMF-NPA	No Recycle	2				None	21.16%	103	E'er
|	|	IMF-NPA	IMF-NPA	3				1 Th	25.62%	123	E'er
|	|	MOX-NPA	No Recycle	2				None	18.89%	103	E'er
|	|	MOX-NPA	IMF-NPA	3				2 Th	23.94%	123	E'er
|	|	MOX-NPA	MOX-NPA	3				1 Th	23.37%	123	E'er
|	|	Breeder FR	No Recycle	2				None	38.23%	103	E'er
|	|	Breeder FR	IMF-NPA	3				1 Th	43.71%	123	E'er
|	|	Breeder FR	MOX-NPA	3				1 Th	41.69%	123	E'er
V	V	Breeder FR	Breeder FR	4				F	64.63%	123	E'er

6.2 Analysis of 6 development trees, 6 branches each

This section provides summary results from investigation of a "decision tree" containing 36 combinations of possible scenarios covering the time frame from 2015 to 2100, inclusive. Table 6-2, in section 6.2.3, contains the results. The first four columns set forth the case number, the reactor/fuel option selected for the period 2025 to 2039, the reactor/fuel option selected for the period 2040 to 2100, and the availability of reprocessing (none, start in 2025, or delayed to 2040). The remaining columns are described below.

Unless otherwise indicated (by "No Nuclear" in the third column), the analysis assumes level generation (i.e. 0% growth) throughout the time period. This means that new nuclear power plants (NPP) are added only when existing ones shut down. The analysis assumes that all existing NPP obtain 20-year extensions to their operating licenses. That means a modest amount of replacement capacity will come on-line between 2030 and 2040. Then the remaining existing NPP will shut down between 2040 and 2060. This will be followed by a period with no shutdowns and then, finally, the replacement capacity will shut down between 2070 and 2080. In the "No Nuclear" future, there are no replacements for the NPP shut down in

2040 and beyond. In the level generation future, they are replaced by new NPP. In the cases involving fast reactors, the CFRs and BFRs become the replacements for the shutdown NPPs Growth of the fast reactor NPP population is also limited by the amount of plutonium (or TRU) available for fuel.

LWRs are presumed to use either normal (no recycle) fuel, IMF or MOX fuel. CFRs are presumed to be used primarily to destroy TRU, and have a very low conversion ratio. This may be achieved through reactor design (a "skinny" reactor, for example), fuel type (an IMF-like fuel, for example) or both. BFRs are assumed to be oxide-fueled SFRs because that may simplify the transition from LWRs to BFRs, but they could be any kind of breeder reactor.

The analysis generally assumes that destruction (or use) of TRU is a priority and that all available SNF will be reprocessed to obtain the TRU. This is accomplished by making the initial reprocessing capacity larger than required for a sustainable fuel recycle. More details on this can be found in the assumptions. In this analysis, "delayed recycle" is interpreted as "delayed reprocessing." This minimizes the amount of *in situ* modification of the fuel material through radioactive decay.

When fuel technologies change (as in a change from MOX to IMF, for example) the fuel in the NPP is discharged at the next opportunity (not the next refueling), so freshly charged fuel would remain in the NPP for five years.

Note that the methodology is strained in the "Nuclear Phase-Out" (No Nuclear) cases. The core fraction devoted to recycle fuel grows as plants are shut down because the feed material remains relatively constant, since it reflects reactor loadings from a decade or more earlier when the nuclear capacity was larger.

6.2.1 Explanation of Columns in Table 6-2

Tech. Need. New technologies required for this case. The individual technologies considered are listed in the assumptions. For example, in case 2.2 one new technology is required for the duration of the scenario, and that is the Monitored Geologic Repository.

Tech. < >: New technologies that are required and then abandoned. For example, again in case 2.2, three technologies and needed, but then abandoned. They are IMF fuel fabrication, IMF reprocessing, and LWRs with mixed cores (UOX and IMF, in this case).

Tech. Gone: Technologies that are abandoned. For example, again in case 2.2, no technologies are abandoned. That means that the four in current use (LWRs, uranium acquisition, enrichment, and UOX fabrication) continue to be needed.

UOX SNF @ YMP: Thousands of tonnes iHM of uranium oxide SNF in storage at YMP by 2100. In general, this number will not exceed 83.8, the presumed capacity of YMP. The SNF is brought to YMP from the utilities on the schedule set forth by OCRWMP.

UOX SNF @ Util: Thousands of tonnes-iHM of uranium oxide SNF in storage at utilities by 2100. This is the excess over the 83,800 tonnes-iHM presumed capacity of YMP. In fact some or all could be stored at YMP, if that facility were to be expanded, or at another repository. This quantity will always include SNF discharged between 2096 and 2100, as that SNF will not have cooled sufficiently for reprocessing or storage.

Rcy SNF @ YMP: Thousands of tonnes-iHM of recycle SNF at YMP by 2100. Recycle SNF is sent to YMP if it is IMF-2 or MOX. Again, the last five years of discharges will be found at the utilities.

HLW @ YMP (Eq. MT=tonne): HLW sent to YMP, expressed as equivalent tonnes-iHM of UOX SNF. The analysis considers HLW to require 5% of the space needed for the SNF from which it was extracted. For example, case 2.3 (IMF/NPA for 75 years) indicates 11,800 $tonne_{eq}$ of HLW. This is the HLW from 236,000 tonnes-iHM (11,800/0.05) of SNF and it will occupy the same space as 11,800 tonnes-iHM of UOX SNF.

Eq. SNF @ YMP: The aggregate of HLW and SNF at YMP expressed as equivalent thousands of tonnes-iHM of UOX SNF. This is an indication of the extent to which space at YMP is required. For example, case 2.3 shows 93,300 equivalent tonnes-iHM. This would exceed the 83,800 tonnes-iHM capacity, but would be well within an expanded capacity of 129,000 tonnes-iHM.

Rcy Gen: The percent of nuclear electricity generation over the period 2015 thru 2100 that was generated from recycle fuel and fast reactors.

Nuc. Gen: Total nuclear electricity generation over the period 2015 thru 2100.

Max. Recy: The peak fraction of generation from recycle fuel in thermal reactors. The reprocessing plant is sized to process all legacy SNF and other SNF available for reprocessing during its presumed 30-year life. Thus, during those 30 years production of TRU for recycle fuel will exceed that which is sustainable under a level generation future. Consequently, the core fraction devoted to recycle fuel will increase rapidly to a figure determined by the reprocessing capacity, and will then drop, after about 30 years, to a level that is sustainable. Note that this does not include any contribution from fast reactors; it is a measure of the reactor core configuration devoted to fuel containing TRU.

HLW: Thousands of tonnes-iHM of SNF that were processed into HLW.

Eq. SNF: The grand total equivalent SNF generated by 2100. This includes SNF at YMP and at the utilities, HLW at YMP, and the SNF equivalent of the in-reactor inventory at the end of 2100.

6.2.2 Explanation of Table Columns

Technologies:

- Monitored Geologic Repository
- Reactor Systems – LWR
- Reactor Systems – LWR with mixed load (*e.g.*, UOX & IMF)
- Reactor Systems – CFR (assumes low conversion ratio attributable to design)
- Reactor Systems - BFR
- Uranium Mining, Milling, Conversion
- Enrichment
- Fuel Fabrication – UOX Fuels
- Fuel Fabrication – MOX, MOX/NP (Glovebox Operations)
- Fuel Fabrication – MOX-NPA (Hot Cell or Remote Operations)
- Fuel Fabrication – CFR Fuels (different design, criticality issues)
- Fuel Fabrication – BFR Fuels (different design, criticality issues)
- Reprocessing – Oxide SNF
- Reprocessing – IMF SNF
- Reprocessing – CFR SNF (presumably like IMF reprocessing, but different stream sizes)
- Reprocessing – BFR SNF (presumably like oxide reprocessing, but different stream sizes)

Normalizing the Starting SNF Inventory: As of 1/1/2000, approximately 2427 reactor years of operation were available to existing commercial nuclear power plants prior to expiration of their operating licenses. Since about 21 tonnes of SNF is discharged for each year of operation, this corresponds to about 48,900 tonnes of SNF to be discharged during all but the last year of operation. In the aggregate, core discharges after the final year of operation will amount to approximately 9,500 tonnes. Given that DOE has planned for about 83,800 tonnes of SNF[1], that means that about 25,500 tonnes had been generated at that time. Similarly, by 1/1/2015, an additional 31,500 tonnes will have been generated for a total, at that time, of 57,000 tonnes. It should be noted that these numbers are not entirely self-consistent. On this basis, during the 1573 reactor years operated, only 16.4 tonnes/yr of SNF was discharged.

YMP Capacity: The capacity of YMP is reported to be 63,000 tonnes-iHM. Yet, OCRWMP states that it can accept 83,800 tonnes-iHM of SNF. Some of this is accomplished by mixing "cool" SNF with "hotter" material. The current capacity is assumed to be 83,800 tonnes-iHM.

Reprocessing Capacity: The reprocessing capacity is based on reprocessing all legacy SNF plus all the 5 yr. old SNF generated during the life of the reprocessing facility. Because this is a "first of a kind" facility, a 30 year lifetime seems reasonable. Even though significant pre-operation R&D&D will occur and a significant "pilot plant" demonstration will be made, it seems reasonable to believe that technological and regulatory advancements during 30 years will render the plant obsolete, or nearly so. Total SNF to be processed during that time will be about 138,000 tonnes so the plant size should be 4,600 tonnes annually. It does not matter whether this is one 4600 tonnes/yr plant or ten 460 tonnes/yr plants. It is assumed that all fuel types envisioned can be handled by the plant capacity. This is probably true for oxide SNF, but would, as a minimum, require a different front end for IMF SNF.

IMF Fuel: IMF fuel data are taken from Wigeland & Bauer.[Wigeland2004a] The data are as given in the Table. The table indicates that 7.1 tonne of UOX SNF are required to make 1 tonne of IMF-1 fuel. Similarly, 19.9 tonne of IMF-1 SNF equivalent to 141 tonne of UOX SNF, is required to make 1 tonne of IMF-2 fuel. In the Wigeland & Bauer approach, the IMF-2 SNF and the HLW from 141 tonne of UOX and the 19.9 tonne of IMF-1 HLW would be emplaced together and would require 1/2.1 as much space as the equivalent SNF. This is interpreted to mean that the HLW and IMF-2 SNF together are equivalent to 77.1 tonne of UOX SNF. The last two rows in the table are an attempt to partition this between HLW and actual IMF SNF. This is done by assuming that all HLW is equivalent to 1/20 of the SNF from which it was produced and assigning the remainder of the equivalent SNF to the IMF SNF.

	IMF-1	IMF-2
UOX Required	7.1	141
IMF-1 Required	(1)	19.9
IMF-2		(1)
Total Fuel	8.1	161.9
Storage Factor	1.8	2.1
Equivalent SNF	4.5	77.1
HLW @ 0.05	0.36	8.05
SNF alone	4.15	69.05

MOX Fuel: MOX-NpPuAm fuel data are also taken from Wigeland & Bauer.[Wigeland2004a] The MOX-NpPuAm fuel are identified as MOX-1 through MOX-5 in the accompanying table. It is similar to the one above for IMF fuel. For example, it takes 1.98 tonnes of MOX-1 SNF (equivalent to 26.74 tonne

[1] *"Analysis of the Total System Life Cycle Cost of the Civilian Radioactive Waste Management Program", DOE/RW 0533, May, 2001, USDOE/OCRWM*

of UOX SNF) to produce 1 tonne of MOX-2 fuel. The partitioning of equivalent SNF is the same as the approach for IMF SNF. MOX-0 fuel is MOX/NP. The 4 tonnes-UOX per tonne-MOX-0 is assumed. MOX-0 SNF is assumed to be similar to MOX-1 SNF, both in its "SNF Value" and in its use to make MOX-2 fuel.

	MOX-0	MOX-1	MOX-2	MOX-3	MOX-4	MOX-5
Required	4	13.5	26.74	40.66	54.75	68.97
Req'd	(1)		1.98			
Req'd		(1)	1.98	3.01	4.06	5.11
Req'd			(1)	1.52	2.05	2.58
				(1)	1.35	1.70
Req'd					(1)	1.26
						(1)
Fuel		14.5	29.72	46.19	63.21	80.62
Factor		1.1	1.2	1.3	1.4	1.5
SNF		13.2	24.8	35.5	45.2	53.7
		0.68	1.44	2.26	3.11	3.98
SNF	12.51	12.51	23.33	33.27	42.04	49.77

Fast Reactors: Reactors consume about 1 tonne of fissile material each year. LWRs have conversion ratios between 0.25 and 0.5, so production of fissile material is about 0.25 to 0.5 tonnes/yr. Consumer fast reactors (CFRs) have conversion ratios around 0.25. Thus, since they produce only about 0.25 tonne fuel per year, they need about 0.75 tonne as makeup. This means that at least three LWRs are required to provide makeup for a single CFR. When other factors, such as neutron absorbing non-fissile diluent nuclides are taken into account, a ratio of 6 to 8 is reasonable. This study uses 8. Breeding fast reactors (BFRs) not only support themselves, they allow for growth because they produce more fissile material than is needed for their own refueling. Early generation BFRs will probably have conversion ratios between 1.1 and 1.2. They will therefore produce 0.1 to 0.2 tonnes of fissile materials that are excess to their needs. A BFR requires 2-2.5 tonnes of fissile material to get started, so a new one can start up for roughly every 15-20 BFR-years of operation. The fissile content of LWR SNF is about 1%, so a new BFR (or CFR, for that matter) will require 200-250 tonnes of LWR SNF or about 10-12 LWR years. Allowing for dilutions, as above, would increase that to around 24 (or about 500 tonnes of UOX SNF. Looking at it in a slightly different way, the fissile loading of a FR is about 3 three times the size of a reload. Thus, starting a new CFR or BFR will require about 24 reloads from LWR fuel. The "SNF values" were assumed on the basis that CFR-1 and -2 SNF would be similar to IMF-1 and -2 fuel, respectively in terms of their storage factor. The value for BFR fuel is approximately 5 times the CFR-1 value on the basis that it will have perhaps five times the TRU content.

	CFR-1	CFR-1	CFR-2	CFR-2	BFR
equired	8		80		8
	(1)		10		1
			(1)	10	
		1			7/8
Fuel	9		91		
	1.8		2.1		
NF	5		43.33	43.33	
	0.4		4.5		
	4.6	4.6	38.83	38.83	25

Thermal Reactor Transitions: In a transition from IMF to MOX fuel, it is assumed that 20 tonnes of IMF-1 SNF can be used to create 1 tonnes of MOX-2 fuel. In transitioning from MOX to IMF, it is assumed that 2 tonnes of MOX-1 SNF or 3 tonnes of MOX-2 SNF can be used to make 1 tonnes of IMF-1 fuel. Similarly, it is assumed that 1.35 tonnes of MOX-3 SNF or 1.7 tonnes of MOX-4 SNF can be used to make 1 tonnes of IMF-2 fuel.

Thermal-Fast Transitions: In using thermal reactor recycle fuel to make fast reactor fuels, ratios similar to those assumed for MOX-IMF transitions were used. The table to the right indicates the ratios used. The "SNF Values" are the same, regardless of the TRU origin, so they are not repeated here. It is assumed that IMF-2 and MOX-5 are still sent to YMP for emplacement, and are not used for generating fast reactor fuels.

	CFR-1	CFR-2	BFR
IMF-1 Req'd		20	20
MOX-1 Req'd	2		2
MOX-2 Req'd	3		3
MOX-3 Req'd		1.35	4
MOX-4 Req'd		1.75	5

Fast-Thermal Transitions: The assumptions for producing thermal reactor fuels from fast reactor SNF are given in the accompanying table. Again "SNF values" are independent of TRU origin and are not repeated here.

	IMF-1	IMF-2	MOX-1	MOX-2
CFR-1 Req'd	8		8	
CFR-2 Req'd		10		10
BFR Req'd	7/8		7/8	

6.2.3 Results

Table 6-2. Summary Results from Decision Tree Analysis

Case	2025-2039	2040-2100	Reprocess	Tech Need	Tech <>	Tech Gone	UOX SNF @ YMP Ktonnes-iHM	UOX SNF @ Util Ktonnes-iHM	Rcy SNF @ YMP Eq. Ktonne	HLW @ YMP Eq. Ktonne	Eq. SNF @ YMP Eq. Ktonne	Rcy Gen %	Nuc Gen GW-yr	Max Recy %	HLW Ktonne SNF	Eq. SNF Ktonne
1.1	Once-Thru	No Nuclear	None	1	0	4	83.8	56.8			83.8		4112			140.6
1.2	Once-Thru	Once-Thru	None	1	0	0	83.8	147.8			83.8		8521			241.1
1.3	Once-Thru	IMF-NPA	Delay to 2040	5	0	0	0.0	7.8	1.1	11.1	84.2	17.1%	8521	34.4%	221.2	117.2
1.4	Once-Thru	MOX-NPA	Delay to 2040	4	0	0	0.0	6.8	0.0	11.1	12.9	16.3%	8521	31.6%	222.4	160.7
1.5	Once-Thru	CFR	Delay to 2040	5	0	0	0.0	8.0	0.0	11.3	11.3	14.8%	8521		225.6	58.0
1.6	Once-Thru	BFR only	Delay to 2040	4	1	4	0.0	0.0	0.0	11.3	11.3	41.7%	8124		225.9	389.3
2.1	IMF-NPA	No Nuclear	Start 2025	1	3	4	0.0	6.8	1.4	7.3	69.4	23.8%	4112	93.8%	145.6	76.0
2.2	IMF-NPA	Once-Thru	2025-2039	1	3	0	31.5	126.3	7.1	3.7	64.7	5.0%	8521	34.4%	73.8	200.5
2.3	IMF-NPA	IMF-NPA	Start 2025	5	0	0	0.0	7.8	1.1	11.1	85.0	17.1%	8521	34.4%	221.2	110.7
2.4	IMF-NPA	MOX-NPA	Start 2025	4	2	0	0.0	7.5	0.6	10.7	39.5	16.2%	8521	34.4%	214.9	141.3
2.5	IMF-NPA	CFR	Start 2025	6	2	0	0.0	10.5	0.0	9.6	10.8	16.8%	8521	34.4%	192.9	50.7
2.6	IMF-NPA	BFR only	Start 2025	5	3	4	0.0	4.0	0.0	9.4	10.5	44.2%	8521	34.4%	187.9	352.6
3.1	MOX-NP	No Nuclear	Start 2025	1	2	4	21.6	3.7	6.9	5.9	141.8	46.1%	4112	89.5%	117.8	145.5
3.2	MOX-NP	Once-Thru	2025-2039	1	2	0	32.6	119.6	0.0	3.3	35.9	8.9%	8521	62.1%	66.7	164.9
3.3	MOX-NP	IMF-NPA	Start 2025	5	1	0	0.0	7.8	0.9	11.1	76.5	24.3%	8521	62.1%	221.5	104.0
3.4	MOX-NP	MOX-NPA	Start 2025	5	0	0	0.0	7.5	3.3	10.3	175.1	31.7%	8521	62.1%	205.5	285.5
3.5	MOX-NP	CFR	Start 2025	6	1	0	0.0	8.4	0.0	11.9	11.9	24.3%	8521	62.1%	237.3	41.1
3.6	MOX-NP	BFR only	Start 2025	4	3	4	11.9	1.1	0.0	11.9	202.6	24.3%	8521	62.1%	237.3	202.0
4.1	MOX-NPA	No Nuclear	Start 2025	1	2	4	0.0	4.5	2.8	7.8	108.6	24.1%	4112	91.4%	149.1	113.1
4.2	MOX-NPA	Once-Thru	2025-2039	1	2	0	33.7	127.7	3.5	3.8	82.0	2.7%	8521	18.7%	76.6	219.2
4.3	MOX-NPA	IMF-NPA	Start 2025	5	1	0	0.0	7.8	0.8	11.1	66.0	16.1%	8521	40.9%	221.7	93.5
4.4	MOX-NPA	MOX-NPA	Start 2025	4	0	0	0.0	7.5	1.1	10.8	65.0	17.3%	8521	28.1%	216.9	176.3
4.5	MOX-NPA	CFR	Start 2025	6	1	0	0.0	9.2	0.0	11.9	11.9	12.4%	8521	18.7%	237.3	37.6
4.6	MOX-NPA	BFR only	Start 2025	4	3	4	0.0	1.1	0.0	11.9	11.9	48.1%	8521	18.7%	237.3	202.0
5.1	CFR	No Nuclear	Start 2025	1	4	4	0.0	2.2	2.2	2.1	39.2	21.4%	4112		143.5	41.4
5.2	CFR	Once-Thru	2025-2071	1	4	0	0.0	69.4	6.1	8.6	92.6	12.4%	8521		171.8	171.5
5.3	CFR	IMF-NPA	Start 2025	5	3	0	0.0	9.0	0.5	4.2	43.2	17.9%	8521	19.4%	205.0	83.7
5.4	CFR	MOX-NPA	Start 2025	4	3	0	0.0	9.6	0.0	3.2	11.3	13.7%	8521	13.3%	206.1	63.8
5.5	CFR	CFR	Start 2025	5	0	0	0.0	8.4	0.0	2.7	11.3	15.1%	8521	20.1%	225.4	54.2
5.6	CFR	BFR only	Start 2025	4	4	4	0.0	1.7	0.0	21.0	11.2	48.1%	8521	8.9%	162.0	420.8
6.1	BFR	No Nuclear	Start 2025	1	4	4	0.0	0.8	16.3	20.8	414.7	30.0%	4112		133.0	415.5
6.2	BFR	Once-Thru	2025-2056	1	4	0	0.0	93.6	16.9	26.6	429.6	14.5%	8521		17.0	532.5
6.3	BFR	IMF-NPA	Start 2025	5	3	0	0.0	9.2	0.5	4.6	46.2	21.0%	8521	43.0%	204.7	93.0
6.4	BFR	MOX-NPA	Start 2025	4	3	0	0.0	9.7	0.0	3.0	10.8	13.6%	8521	11.8%	205.7	60.3
6.5	BFR	CFR	Start 2025	5	3	0	0.0	8.2	0.0	9.9	11.2	15.1%	8521		197.6	58.6
6.6	BFR	BFR only	Start 2025	4	1	4	0.0	0.0	0.0	11.1	11.1	60.4%	8521		222.5	389.1

6.2.4 Observations

1. **Agility:** Agility is here defined as the ability to change quickly. In general, fuel options have the ability to be changed much more rapidly than do reactor options. Consequently, fuel options are more agile. In most respects, IMF and MOX appear to have about the same effect on the LWR population. They both equilibrate somewhere around 15% of the core loading, for example. The one difference is that the equivalent SNF "value" of the in-reactor inventory of MOX fuel is much greater than that of IMF fuel, and on that basis the IMF fuel might be judged slightly more agile – at least less of a "downside" of a whole core discharge is required or if there is a sudden loss of reprocessing. In terms of agility, CFRs and BFRs seem roughly equivalent. This is due, in part, to the presumption that the CFR has a low conversion ratio because of its geometry, not because of its fuel. Thus, agility is limited by the ability to change out a reactor. Thus, arranged in order of agility, the options seem to be IMF, MOX, fast reactors.

2. **Robustness:** Robustness is here defined as the ability to perform under different conditions. There seem to be no major differences between LWRs with MOX or IMF fuels and CFRs. However, BFRs pose a different problem. The best solution in a "declining nuclear market" is to consume all (or as much as possible) of the TRU. However, the BFRs produce *more* TRU. Consequently, in a declining market, they worsen the SNF situation because they increase the amount of TRU that must be dealt with. The other three options destroy TRU and have less of a "downside" in a declining market. CFRs are "single purpose" in that they destroy TRU. In a framework that wishes to make the most use of TRU, they are less well suited. Arranged in order of declining robustness, the options are IMF/MOX, CFRs, BFRs.

3. On the other hand, BFRs appear to be able to sustain a growth rate of 2-3% per year, and are the only option that can do so. The other three options can equilibrate so that recycle is effectively destroying the TRU produced, but growth must be accommodated by starting new reactors on UOX fuel and allowing a cycle (12 years) to produce TRU.

4. **Effect of Assumptions:** The assumption of a 4600 tonnes/yr reprocessing plant causes the legacy SNF to be consumed by 2060 (2080 if reprocessing is delayed to 2040). Consequently, the production of TRU during those years is about 220% of "normal." As a result, the recycle content of LWR cores starts high and then declines to a more nearly equilibrium level. CFR start-ups must be constrained so that they do not run out of fuel during their later years. On the other hand, this gives BFRs a "boost" and allows them to more quickly enter the market. The results are "driven" by the assumptions on SNF "value" and on the utility of one fuel in making another. More studies like the Wigeland and Bauer study could do much to provide a stronger basis for studies such as this.

5. The equivalent SNF value of IMF-2 is much greater than that for any of the higher MOX SNF. IMF cases tend to produce more recycle SNF during this time period than the MOX cases – largely because there are 2 cycles to IMF SNF, but 5 cycles to MOX-SNF. Thus, the equivalent SNF at YMP is larger for the IMF-IMF (2.3) case than for the MOX-MOX (4.4) case. However, that means that the MOX fuel has a higher "holdup" in terms of reactor inventory at the utilities. The holdup for IMF fuel is around 30 tonne_{eq} whereas for MOX it is around 100 tonne_{eq}. The MOX holdup is strongly influenced by the long cycle time (~60 years for 5 cycles) – that means that the effects of the early high fuel production rate persist for a long while. For comparison, look at cases 5.4 and 6.4 to see where MOX fuel seems to equilibrate when its initial production is constrained.

6. The MOX/NP cases involve more recycle fuel than the other cases because of the high assumed conversion from UOX to MOX (1/4 to make MOX-NpPu and 1/13.5 to make MOX-NpPuAm). In

fact, in case 3.1 (MOX/NP-Nuclear Phaseout) the amount of recycle SNF overwhelmed the reduced demand for fuel and some of the lower (MOX-1 to -4) SNF had to be sent to YMP rather than converted to fuel. In terms of energy generated using recycle fuel, this increased amount of recycle fuel is good. However, in terms of SNF generation, this is not good. One would expect the equivalent SNF developed in Case 1.2 to be an index against which other cases could be compared, and that all other cases would have lower values. That seems not to be the situation for case 3.4 (MOX-NpPu-MOX-NpPuAm). The calculations have been checked and seem to be correct, but the result remains suspect. It is likely that the assumptions are internally inconsistent.

6.2.5 Case-by-Case Comments

Table 6-3. Comments on Individual Trees

Case	Comment
Case 1.1	30.8 GW of replacement LWRs added between 2033 and 2039. Remaining 68.4 GW of existing reactors retired at the end of their 60-yr extended licenses by 2059. Last replacement reactor shut down after 40 years in 2079.
Case 1.2	No Comment
Case 1.3	Because of the large initial output of Pu/TRU, the recycle content of LWR cores reaches about 34%. However, after equilibration, IMF fuel accounts for about 15% of the reactor loading.
Case 1.4	Because of the large initial output of Pu/TRU, the recycle content of LWR cores reaches about 31%. However, by 2100 MOX/NPA fuel accounts for about 23% of the reactor loading.
Case 1.5	23 GW of CFR capacity added by 2049 and maintained thereafter. Fuel inventory grows until 2081 (when legacy SNF is gone), then declines and is exhausted by 2100.
Case 1.6	BFR capacity grows to 45 GW by 2058 and holds until 2072 (no LWR retirements), then grows to 99 GW (100%) by 2086. BFR fuel inventory grows until 2076, drops to a low (essentially zero) in 2085 (when legacy SNF is gone and BFR capacity is growing), grows thereafter at a rate that should support about a 3% capacity growth rate.
Case 2.1	Recycle fuel production is stopped in 2073 and the last LWR shuts down in 2079. Recycle fraction builds quickly to 34% and then grows further to 94% as LWRs are shut down (94% at 35 GW) and then drops to zero as the in-reactor fuel is discharged.
Case 2.2	Recycle fraction grows to 34% by 2032. Recycle fuel production stopped in 2040 and fraction drops to zero by 2044.
Case 2.3	Recycle fraction grows to 34% by 2032, then drops slightly to 31% around 2052 and sort of equilibrates around 14% after legacy SNF is gone.
Case 2.4	Recycle fraction grows to 34% by 2032, holds, and drops to 16-17% as MOX replaces IMF. Only 1 year of IMF-2 fuel produced, 24 years of MOX-5.
Case 2.5	Recycle fraction as in prior IMF cases. 18 GW of CFRs replace retiring LWRs after 2040. CFR fuel inventory grows until 2056 (when legacy SNF is exhausted), then drops to essentially zero by 2100.
Case 2.6	Recycle fraction as in prior IMF cases. 41 GW BFR capacity added by 2050, starts growing again in 2057, reaching 87 GW in 2072 and then 89 GW by 2100. Fuel inventory peaks in 2075 (no capacity growth, so breeding increases fuel) then drops to essentially zero in 2086 as capacity grows; grows thereafter at a rate that could sustain a 1%-2% capacity growth.
Case 3.1	Recycle fraction grows to ~60% by 2032, holds until 2037; reaches 87% in 2042 as LWRs retire, then wobbles thru 2060 and ends at ~50%. Reprocessing plant shutdown in 2074.
Case 3.2	Recycle fraction grows to ~60% by 2032, holds until 2037; declines thereafter.
Case 3.3	Recycle fraction grows to ~60% by 2032, holds until 2037; reaches 50% around 2052, then drops to about 10% and equilibrates around 14%

Case 3.4	Recycle fraction grows to ~60% by 2032, holds until 2037; reaches 50% around 2052, then drops to about 44% and then further to 16% by 2100.
Case 3.5	21 GW of CFR added shortly after 2040. Fuel peaks at 2055 (when legacy SNF is gone) and is gone by 2100.
Case 3.6	61 GW of BFR added by 2057, then increases to 99 GW capacity between 2073 and 2079. Fuel inventory peaks in 2048, has a low in 2057, peaks again in 2075, another low in 2086, then grows enough to sustain 3% capacity growth.
Case 4.1	Recycle fraction grows rapidly, is 25-30% in years before 2040, then grows to 90 in 2054 as LWRs retire. Reprocessing plant shuts down in 2073, last LWR in 2083 with no recycle fuel.
Case 4.2	Recycle fraction grows to 18% by 2033 and then declines after 2040.
Case 4.3	Recycle fraction grows to 18% by 2033, then to 41% by 2044, equilibrates around 14% about 2070.
Case 4.4	Recycle fraction grows to 18% by 2033, then up to 28% after 2040, finally down to 17% around 2090.
Case 4.5	MOX fuel production starts in 2025. Replacement LWRs are brought on-line between 2033 and 2041. CFR fuel production starts in 2040 and the first CFR starts up in 2041, building quickly to 19.8 GW. CFR fuel inventory builds until about 2056, when the legacy SNF is exhausted. After that the inventory drops slowly and is exhausted by 2100. A mix of about 12-13% CFRs would be self-sustaining.
Case 4.6	MOX fuel production starts in 2025. Replacement LWRs are brought on-line between 2033 and 2041. BFR fuel production starts in 2040. BFR fuel inventory builds until 2047, when the FR capacity begins to climb steeply to about 58 GW. The inventory is essentially depleted from 2054 to 2059 (actually 3 replacement LWRs should have started up at this time, but the analysis assumed that they got 20-year license extensions). From 2060 to 2073, there are no retirements, so the fuel inventory builds. BFRs are added between 2073 and 2080 to fill out 100% of the generating capacity. The BFR fuel inventory grows until 2073 and then drops as the inventory is used to fuel the BFRs starting up from then on. In 2086 the fuel inventory is again depleted and after that it grows until the sequence ends in 2100. This growth in fuel inventory would likely sustain a growth in BFR population of somewhere near 3 percent per year after about 2089.
Case 5.1	CFR fuel production starts in 2025 when reprocessing begins. A significant inventory is produced before CFRs are added starting in 2033. 22 GW of CFRs are added as replacements for retiring LWRs between 2033 and 2038, and CFR operation continues thru 2079. The reprocessing plant is shut down at the end of 2074, so CFR fuel must be drawn from inventory starting in 2077. LWR retirements after 2040 reduce the LWR capacity to 8.8 GW in 2059, and the last LWR is shutdown in 2080.
Case 5.2	CFR fuel production starts in 2025 when reprocessing begins. A significant inventory is produced before CFRs are added starting in 2033. 26.4 GW of CFRs are added as replacements for retiring LWRs starting in 2033. Starting in 2040, no further CFRs are added but CFR operation continues until 2077. The reprocessing plant is shut down at the end of 2071, so CFR fuel must be drawn from inventory starting in 2074.
Case 5.3	CFR fuel production starts in 2025 when reprocessing begins. A significant inventory is produced before CFRs are added starting in 2033. 17.6 GW of CFRs are added as replacements for retiring LWRs starting in 2033. Based on the allocation of TRU from UOX SNF, this will exhaust the preproduced CFR fuel when the CFRs are themselves retired after 40 years of operation. Beginning in 2040 sufficient IMF-1 fuel is produced to achieve a 17% recycle fuel mix in LWRs. CFR fuel production is stopped in 2065. The amount of TRU from UOX SNF for recycle and CFR fuel drops significantly around 2055, when all of the legacy fuel has been reprocessed.

Case 5.4	CFR fuel production starts in 2025 when reprocessing begins. A significant inventory is produced before 13.2 GW of CFRs is added as replacement for retiring LWRs starting in 2033. Based on the allocation of TRU from UOX SNF, this will exhaust the preproduced CFR fuel when the CFRs are themselves retired after 40 years of operation. Beginning in 2040 sufficient MOX-1 fuel is produced to achieve an 11% recycle fuel mix in LWRs. CFR fuel production is stopped in 2065; however, in about 2057 (when the legacy SNF is gone) the consumption of UOX SNF for LWR MOX fuel requires the entire amount available, and only CFR-2 is produced during the last 8 years.
Case 5.5	19.8 GW of CFR capacity is added as LWRs are retired and by 2040 all CFRs are on-line. This number of CFRs is not sustainable; it exhausts the CFR fuel by 2100. The sustainable mix appears to be about 12% (12 GW out of 99)
Case 5.6	Only 9 GW of CFRs are added between 2033 and 2036. Beginning in 2040 BFRs are added as LWRs retire. By 2051 about 63 GW of BFR capacity is online. FR fuel production starts in 2027 and the inventory grows until about 2044 when the fuel needs of the growing BFR population overcome the production from legacy fuel and the inventory begins to decline. It reaches a low in 2058 after the legacy SNF is gone. No BFRs are added between 2051 and 2077. By then the FR fuel inventory has grown to support further additions of BFRs as well as replacing the retiring CFRs with BFRs. There is insufficient UOX SNF to completely replace all the retiring LWRs, however and by 2100, about 13% of the capacity is still LWRs. By the end of the century, the fuel inventory is growing fast enough to accommodate a 1-2% growth in capacity.
Case 6.1	About 31 GW of BFRs replace LWRs by 2039. The last LWR shuts down by 2060, last BFR in 2079. BFR fuel inventory peaks in 2058, when production of BFR fuel stops (reprocessing down in 2056). BFR fuel inventory essentially zero at time of last BFR shutdown. All BFR SNF produced after 2058 is “stranded.”
Case 6.2	About 31 GW of BFRs replace LWRs by 2039. Last BFR shuts down in 2079. BFR fuel inventory peaks in 2058, when production of BFR fuel stops (reprocessing down in 2056). BFR fuel inventory essentially zero at time of last BFR shutdown. Alternatively, could stop reprocessing in 2040. BFRs would run out of recycle fuel in 2062 and require UOX fuel (perhaps 10-15% enriched) thereafter. This would increase the equivalent SNF from 532,500 tonnes to 599,400 tonnes and generate 9,300 tonnes of uranium fast reactor SNF.
Case 6.3	BFR capacity limited to 22 GW. On this basis BFR fuel inventory peaks in 2056 and declines thereafter to reach zero with shutdown of last BFR about 2076. IMF production is limited to 8% of LWR capacity to provide TRU for BFRs. Recycle fraction climbs to 42% as BFR SNF is “worked off” and then declines to about 17%. BFR fuel production stops in 2067.
Case 6.4	BFR capacity limited to 17 GW. On this basis BFR fuel inventory peaks in 2056 and declines thereafter to reach zero with shutdown of last BFR about 2076. MOX production is limited to 8% of LWR capacity to provide TRU for BFRs. Recycle fraction climbs to 11% as BFR SNF is “worked off” and stays around there until the end. BFR fuel production stops in 2065.
Case 6.5	About 31 GW of BFRs replace LWRs by 2039. Last BFR shuts down in 2079. CFR additions amount to 22 GW. FR fuel peaks in 2053, declines to essentially zero in 2076 as CFRs consume the TRU, peaks again in 2089 and BFR cores are reprocessed and then declines through 2100, when a reasonable amount remains (enough for 3-4 reactor lifetimes).
Case 6.6	System is 100% BFR by 2079. BFR fuel peaks in 2075, drops slightly to a low in 2082 and then grows at a rate to sustain 3-4% capacity growth.

7. DYMOND ANALYSES OF SIX DEVELOPMENT TREES

The results presented herein for the scenarios were calculated using the nuclear energy systems dynamic analysis code DYMOND.[Moisseytsev2001, Yacout2005a] The code simulates the energy-demand driven nuclear energy system scenarios over time and allows the simulation of changing nuclear reactor parks and fuel cycle options. It includes different types of delays and feedbacks associated with the construction of nuclear facilities and the decisions to build such facilities. The mass flows of the different fuel cycle spent fuel streams are followed and the associated decay heat generations are calculated. More details about the models that are related to the global nuclear energy simulation are presented by Moisseytsev2001 and GIF2002. Other DYMOND modeling of the U.S. nuclear park within the AFCI context are provided by Yacout2004a and b, and Yacout2005a, b.

Table 7-1 summarizes the six development trees analyzed in this Chapter. Each is assumed to start in 2025. Each tree has six branches in 2040, the three that we considered the most interesting in addition to nuclear phase out, recycling phase out, and continuing what was started in 2025. We limited the analysis to 6 branches for tractability. One could, in principle, postulate an almost unlimited combination of scenarios. Unlike Chapter 6, we did not further complicate the analysis by postulating branches at 2060 or 2080. We calculated the various metrics in Table 3-2 for each of the 36 branches (6 trees x 6 branches/tree) using DYMOND, except 2 of the 36 could not be calculated because we lack fuel composition recipes for MOX-to-BFR.

For branches where nuclear is phased out (1.1, 2.1, 3.1, 4.1, 5.1, 6.1) the model is not currently capable of moving the remaining material to the repository. In these cases the final mass in the repository will be the total system mass for those elements that are to be placed in the repository (Note: reprocessing would eliminate the need to put uranium and fission products to the repository). For the discussion in this section, the system includes the repository, reactors, reprocessing , and all intermediate storage or processing steps.

For all cases, the initial reactor fleet was assumed to contain only thermal reactors. Some of the initial thermal reactors (35) were only capable of utilizing UOX fuel and their burn-up was set at 33. The remaining initial thermal reactors (68) were capable of utilizing UOX, MOX, or IMF fuel, and their burn-up for UOX was set at 51. This allowed the average burn-up of the existing reactor fleet to be met (45), while allowing some of the existing legacy reactors to utilize MOX or IMF in the future. In the model, the thermal reactors which only utilize UOX are retired before the thermal reactors with multiple fuel capability.

Note, when the Pu or TRU availability is inadequate for LWR reactors capable of running MOX or IMF, the model automatically substitutes UOX.

Table 7-1. Summary of Development Trees

Development Tree	Motivation for Analysis	Notes	Deployment constraints
1. Continue once-through until 2040, i.e., delay recycling	Explores continuation of once-through for an additional 15 years.	Branch 1.2 continues "once-through" until the end of the century.	N/A
2. Start IMF-NpPuAm in 2025 (using blended IMF/UOX cores)	Attempts fastest possible reduction in LTH, LTD, and LTR using thermal reactors and UREX+ separation technology, but an unproven fuel.	Assumes n-pass IMF fuels and their separation are practical. This IMF approach uses blended fuel assemblies, with ¾ UOX and ¼ IMF, with the TRU in used fuel UOX and IMF in one generation making the IMF in the next generation. Other n-pass IMF approaches require analysis, including increasing the IMF/UOX ratio to further accelerate benefits or require fewer reactors to use the blend.	3 kt/yr separation plant starts in 2025. All fuel that can be made from that separation plant is assumed to be used in the growing TR fleet.
3. Start MOX-NpPu in 2025	Closest to current international practice and current technology, while avoiding separation of Pu	Restricted to 1-recycling pass in current analyses.	
4. Start MOX-NpPuAm in 2025	Attempts modest repository benefits using thermal reactors, UREX+ technology, and fuels relatively similar to current UOX and MOX-Pu.	Assumes BU is the uranium component in MOX; the Pu/U ratio increases each cycle to keep the cores critical. Other n-pass MOX approaches require analysis, including keeping the core critical by increasing the uranium enrichment instead of the Pu/U ratio.	
5. Start consumer FR in 2025	Moves into FR, skipping recycling in TR. The early FR experience would set the stage for BFR when uranium resources warrant.	Balancing all the components of this type of system is not straightforward.	FR deployment is limited by the amount of Pu available for FR fuel, existing FR's have 1st priority on fuel over new FR's, if insufficient fuel is available for FR's to start, the missing capacity is met by starting thermal reactors
6. Start breeder FR in 2025	Moves into FR, skipping recycling in TR. Aims to accommodate a hypothetical combination of limited uranium resources and high nuclear growth.	Unique among the options in that BFR uses depleted uranium.	

The model is capable of running limited combinations of reactor types. The relative proportion of each type of reactor present in the system depends on the request, meaning the percent of power generation capacity coming online to be provided by specific reactor types.

The single-pass and multi-pass MOX recipes used in this chapter are full core recipes. The single-pass IMF recipe (IMF-NpPu) is also a full core recipe. The multi-pass IMF recipe (IMF-NpPuAm) is a blended core recipe. For multi-pass fuels, each pass is tracked in the model. Multi-pass MOX has 8 passes (pass 1-8 in the model), while multi-pass IMF has 5 passes (pass 1-5 in the model). All UOX fuel supplied to make up for missing MOX or IMF fuel is accounted for in a separate pass (pass 0 in the

model). When fast reactors are being requested for the reactor fleet all spent UOX (pass 0 in the model), spent pass 1 fuel (MOX or IMF depending on the case), spent pass 8 MOX fuel, spent pass 5 IMF fuel, and spent fast reactor fuel goes to the fast reactors. This means when fast reactors are being added to the system, no new MOX or IMF fuel is being added to the thermal reactors. In general this is a reasonable approximation of how the reactor fleet would be run given the current set of the reactor/fuel options in the model (currently VHTR reactors are not part of the model). Since IMF, MOX, and consumer fast reactors all perform somewhat similar functions of reducing transuranics, so directing fuel towards one type of fuel reactor system is reasonable. Breeder fast reactors utilize transuranics (especially plutonium) to increase the energy recovery from uranium, and work to some degree in opposition to thermal recycle and consumer fast reactors, directing fuel towards one type of fuel reactor system is again reasonable. In the future, the model will be modified to allow more flexibility in how spent fuel is allocated to existing reactors.

The support ratio (number of reactors in pass n required to provide fuel for reactors in pass n+1) varies among fuel types. For the multi-pass IMF (blended core) fuel used in this model, the support ratio is approximately one, this means the of IMF available for successive cycles of IMF remains fairly constant. This allows IMF fuel to move into a large proportion of the reactor system quickly. The multi-pass MOX (full core) and one-pass MOX (full core) fuels used in this model have a support ratio of 7-11, depending on the cycle of the fuel. This means that successive cycles of MOX move very slowly into the reactor system.

In the current version of the model, the amount of recycled fuel available for reactors is based on the elemental plutonium content of the spent fuel isotopes. Fuel control based on elemental plutonium is a reasonable approximation, especially when only type of recycled fuel is used. The isotopic composition of plutonium, other transuranics, and uranium in fuel are key to a reactors performance and the isotopic composition depends on the fuel and how it is burned. This means that the results from some of the fuel/reactor combinations presented in here may be refined in the future, especially once isotopic flow control is developed in VISION.

Note that DYMOND currently has a simple model for retirement of current reactors. As shown in Figure 7-1, the current estimates for reactor retirement have been pushed a bit further into the future. This is good from the standpoint of maximizing return on current investment and on the time available before selecting reactor types for the replacement reactors. The current model overpredicts how soon new reactor types would replace these retiring reactors but underpredicts the fraction of retirements that occur after 2040.

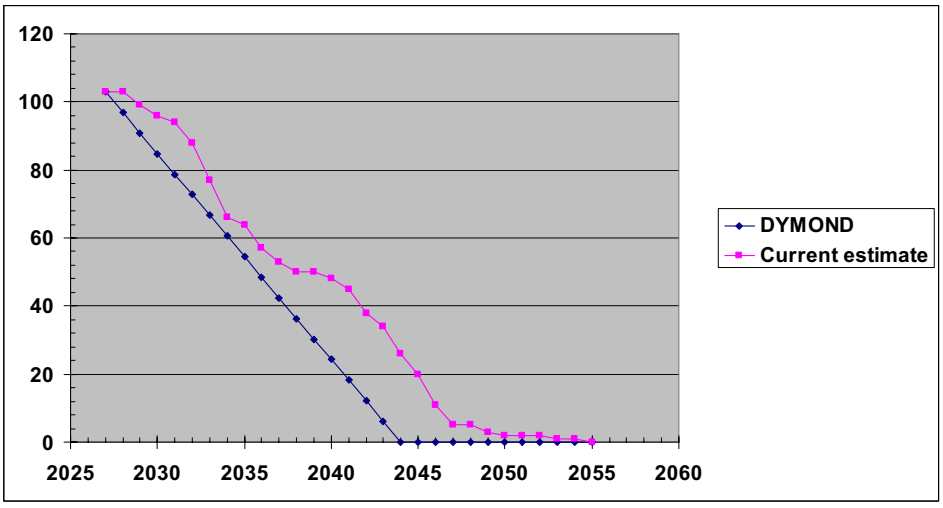

Figure 7-1. Estimated retirement of the current 103 reactors

7.1. Continue Once-Through until 2040 – delayed recycling

This development tree (Figure 7.1-1) explores continuation of the status quo (once-through) until at least 2040. In 2040, there are six branches. Branch T1.1 is phase out of nuclear power; no new reactors are built, existing reactors continue until the end of their lifetimes. Note that the current 103 reactors are projected to retire in the period 2027 until 2044 in the current DYMOND model. Thus, if nuclear phase out starts in 2040, most of the current reactors will have already been replaced. The last reactor would not stop until 2104 (60 years after 2044). Branch T1.2 is indefinite continuation of once-through. The other branches simply start recycling (trees T2, T4, T5, and T6) 15 years later than if they had started in 2025.

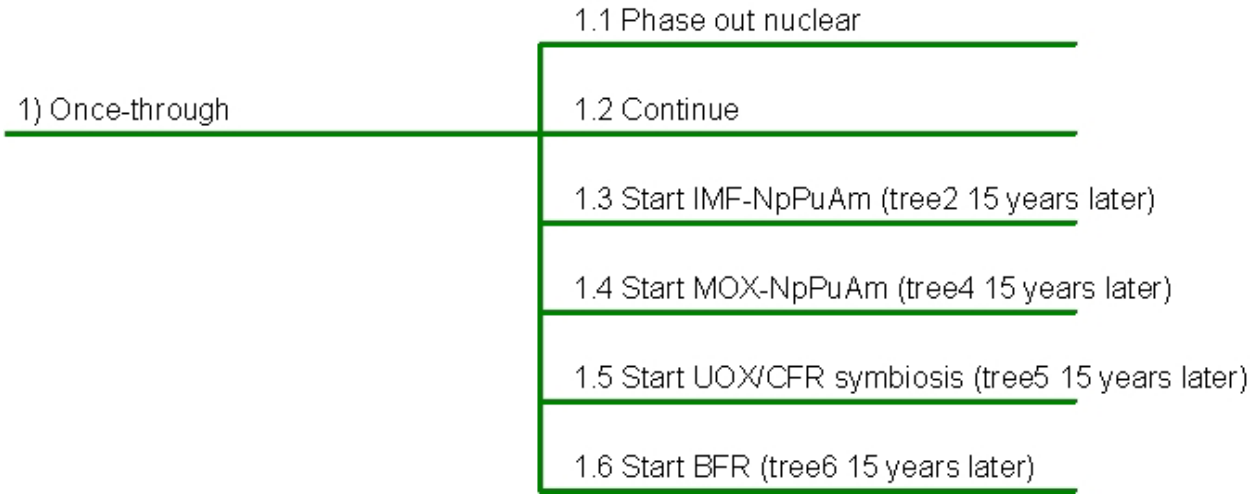

Figure 7.1-1. Development tree for continuing once-through until 2040.

The LWRu reactors in Figure 7.1-2 can only utilize UOX fuel; these reactors will be retired by 2032. In trees 1.5 and 1.6, the power provided by fast reactors is relatively small, even when the request (i.e. the percent of power generation capacity coming online to be provided by specific reactor types) for fast breeder reactors is set to 100% (tree 1.6). This is because the system is Pu limited for fast reactors given the current fast reactor recipes and breeding ratios.

Figure 7.1-3 shows the amount of mass in the repository as a function of time. The repository is assumed to open in 2012 with a maximum receiving capacity of 3 ktonne per year. Continuing with once through (T1.2) or phasing out nuclear, starting in 2040, results in greater than 250 ktonne of spent fuel in the repository by 2100. Compared to once through, switching to multi-pass IMF (T1.3) or breeder fast reactors (T1.6) in 2040 reduces the mass of isotopes in the repository by approximately 80% by 2100, while switching to consumer fast reactors in 2040 reduces the mass of isotopes in the repository by approximately 70%. Switching to multi-pass MOX (T1.4) in 2040 reduces the mass of isotopes in the repository by approximately 60% by 2100.

In order to achieve a 99.5% reduction in TRU content in the repository by 2100 multi-pass recycling of thermal or fast reactor fuels must start prior to 2040. Of the options explored, multi-pass IMF (T1.3) or breeder fast reactors provide the quickest reduction of TRU in the repository.

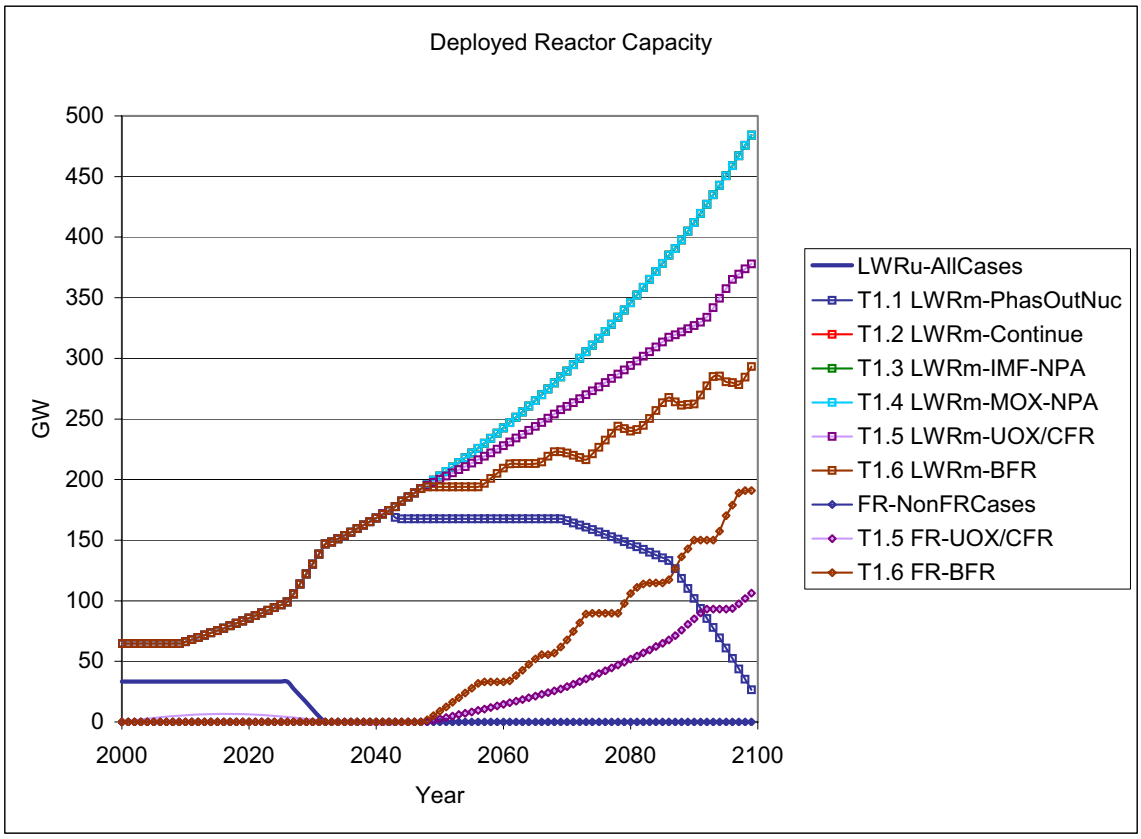

Figure 7.1-2. Power capacity of operating reactors by reactor type for each case in tree 1 (remain with once-through at 2025). LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR).

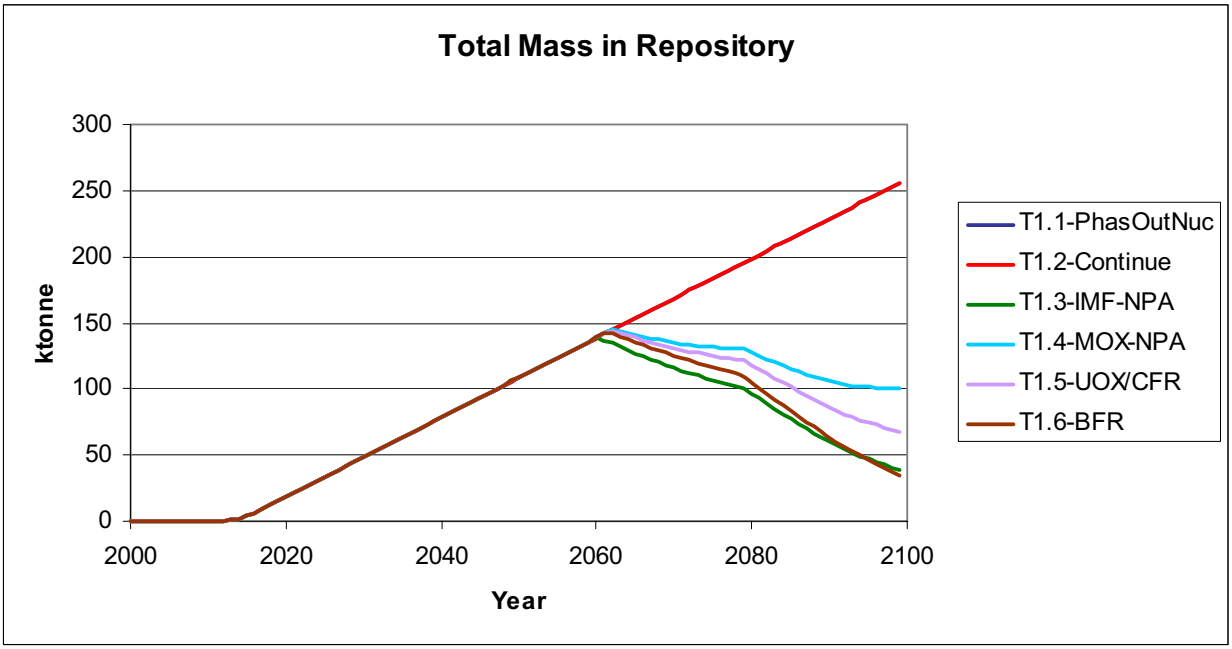

Figure 7.1-3. Total mass in repository for each case of tree 1 (remain with once-through at 2025). T1.1 and T1.2 overlay each other.

Figures 7.1-4 and 7.1-5 show the total and uranium mass (respectively) in the system for each case. Uranium accounts for the majority of the mass in the system and the relative order of mass for uranium and total system are the same for all six cases. As expected, phasing out nuclear (T1.1) results in the lowest mass in the system where continuing and once through (T1.2) result in highest mass in the system. Switching to multi-pass MOX or multi-pass IMF at 2040 reduces the total mass in the system a small amount, with IMF reducing the mass more than MOX. This is probably due to the relatively flat support ratio of multi-pass IMF (a blended core recipe) that allows it to move more quickly than multi-pass MOX (a full core recipe into the reactor fleet (see Figure 7.1-6). The consumer fast reactor (T1.5) sits between MOX and IMF; this is expected since the function of consumer fast reactor in the reactor system is similar to that of multi-pass MOX or IMF. Switching to breeder fast reactors decreases system mass more than consumer fast reactors, MOX or IMF by 2100.

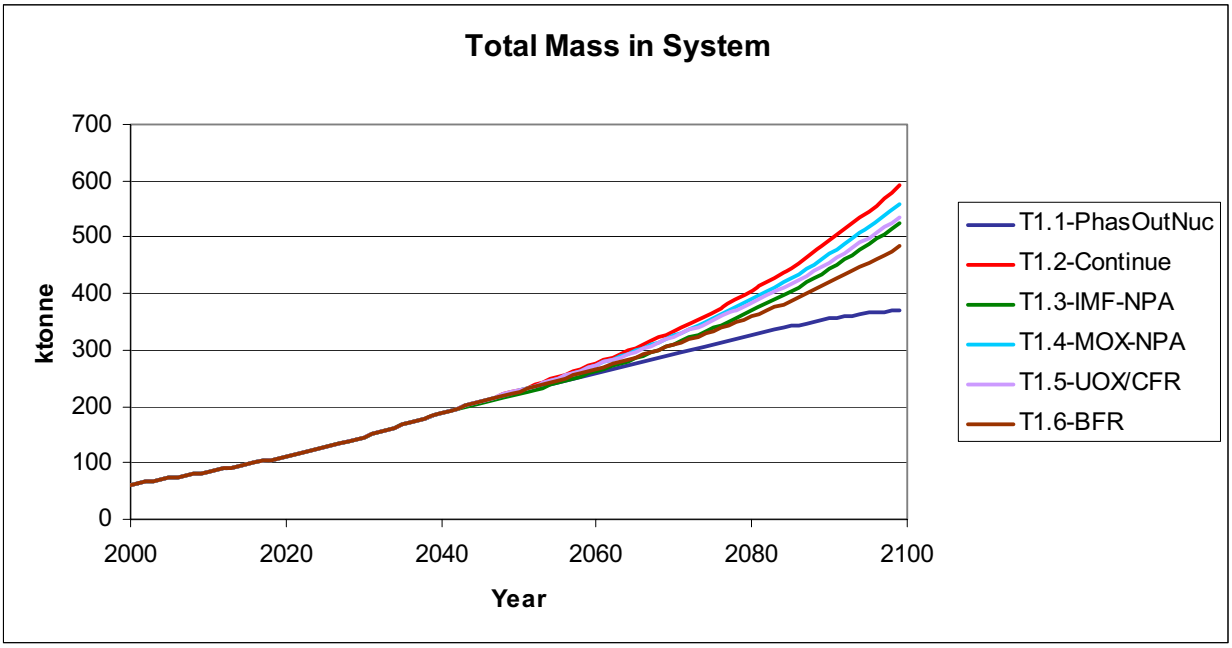

Figure 7.1-4. Total mass in system for each case of tree 1 (remain with once-through at 2025).

Figure 7.1-6 shows a clear difference between multi-pass IMF (T1.3) and multi-pass MOX (T1.4). The multi-pass IMF has a low support ratio and is blended core fuel. The multi-pass MOX has a higher support ratio and is a full core fuel. This means that the IMF moves much more quickly into the reactor fleet than the MOX. However, as discussed in Chapter 5. The amount of UOX in both systems is about the same since blended core IMF includes a significant amount of UOX in its recipe.

Plutonium (Figure 7.1-7) represents a small percent of the total mass in the system. Starting multi-pass IMF (T1.3) in 2040 significantly reduces the amount of plutonium in the system compared to all of the other cases, even compared to phasing out nuclear. This suggests that running multi-pass IMF may be desired even if nuclear power is phased out to reduce the amount of plutonium in the repository.

Once through UOX represents the worst case for uranium usage (see Figure 7.1-8). From a uranium conservation perspective, the faster recycling is started for thermal reactors or fast reactors the better. Of the continuing thermal reactor options, multi-pass IMF (T1.3) consumes the least uranium ore by 2100. This is probably because the support ratio (number of reactors in pass n required to provide fuel for reactors in pass n+1) is approximately one, so the system is not as limited with respect to Pu as for fast reactors and the amount of multi-pass IMF available for successive cycles of multi-pass IMF remains fairly constant. This allows multi-pass IMF fuel to move into a large proportion of the reactor system quickly (see Figure 7.1-6). The multi-pass MOX (T1.4) used in the model uses fresh uranium when

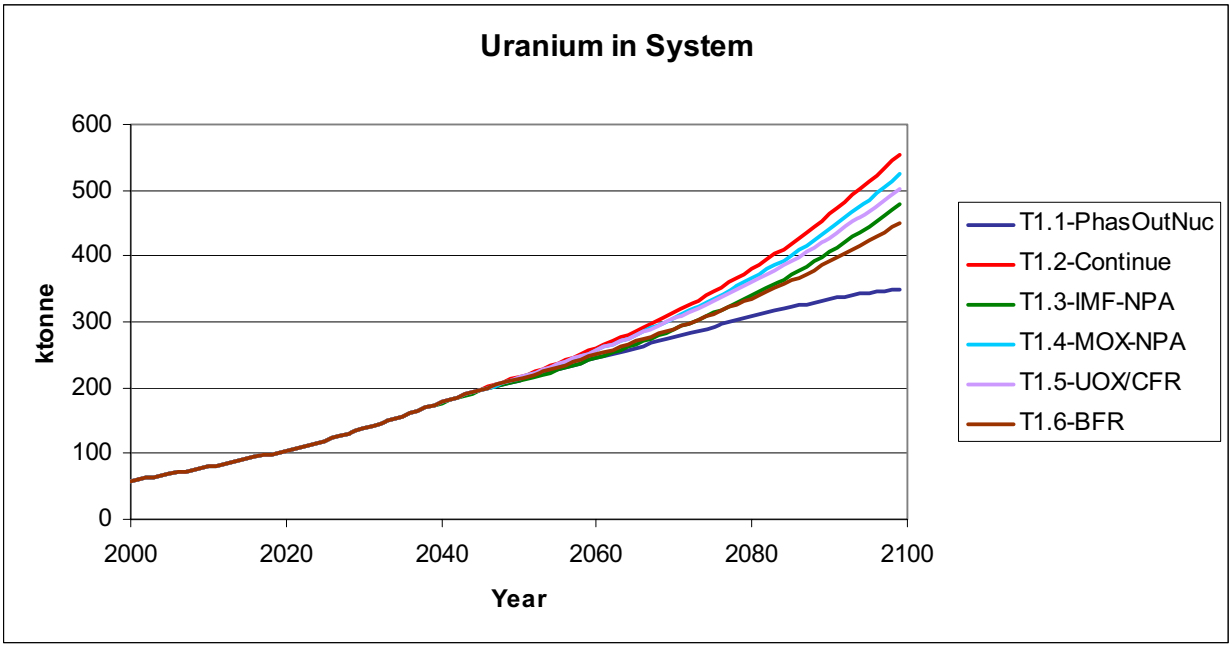

Figure 7.1-5. Mass of uranium in system for each case of tree 1 (remain with once-through at 2025).

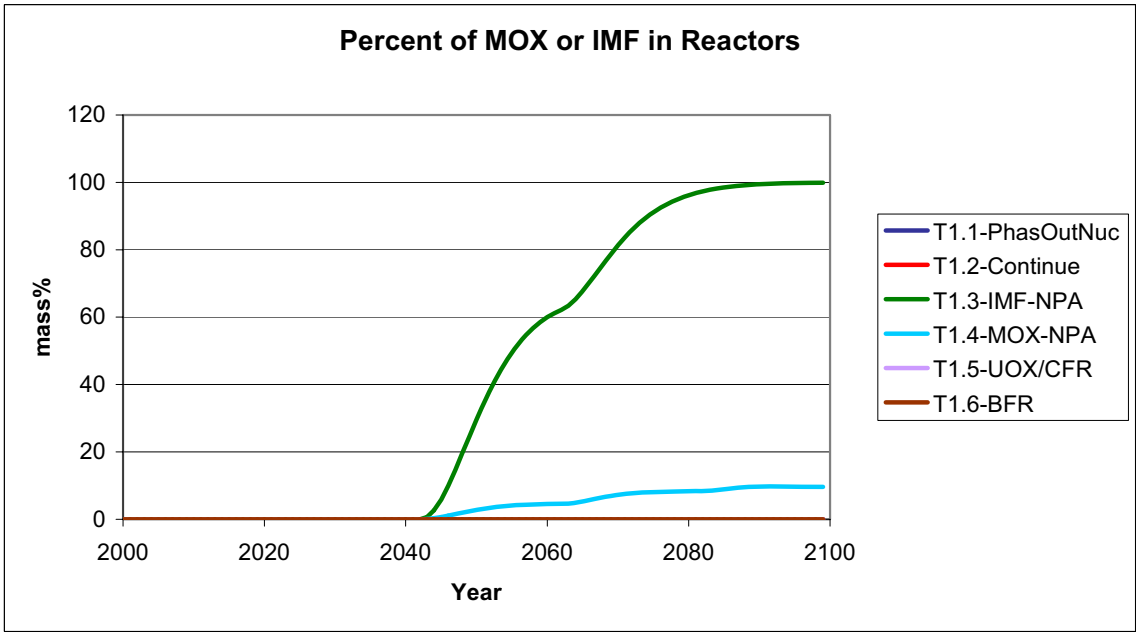

Figure7.1-6. Mass percent of multi-pass MOX or multi-pass IMF fuel in reactors for each case of tree 1 (remain with once-through at 2025). The multi-pass IMF for case T1.3 is a blended core rather than a full core; in T1.3 the entire fleet would use the 3/4-UOX/1/4-IMF blend. Case 1.4 is full MOX core; in T1.4 about 10% of the fleet would use full-core MOX. T1.1, T1.2, T1.5, and T1.6 do not contain MOX or IMF and overlay each other.

sufficient Pu is not available to make MOX and the support ratio for multi-pass MOX is much steeper than that for multi-pass IMF. This means that successive cycles of multi-pass MOX move very slowly

into the reactor system (see Figure 7.1-6), so a large portion of the fuel in the reactor fleet remains UOX (see Figure 7.1-6). The uranium usage for the consumer and breeder fast reactor cases (T1.5 and T1.6, respectively) is not surprising. As was shown in Figure 7.1-2, fast reactors represent only a small proportion of the reactor fleet by 2100, while the remaining fleet is operating with once through UOX (see the curve for T1.2), which uses the most uranium of all the cases.

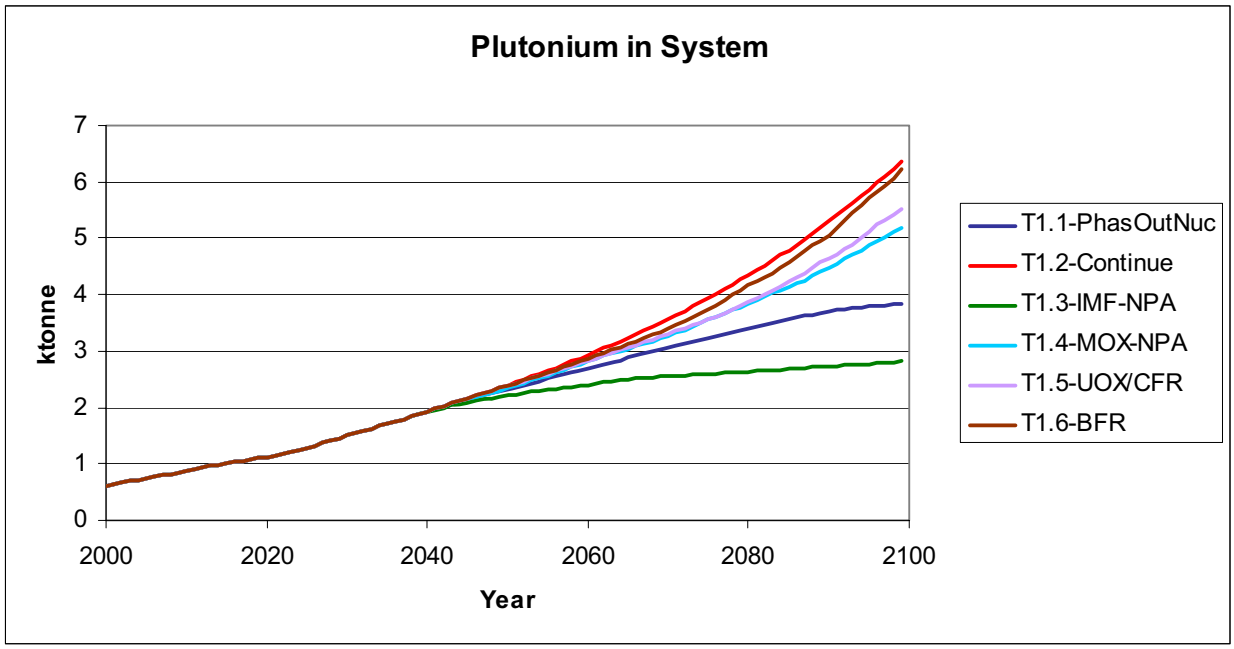

Figure 7.1-7. Mass of plutonium in system for each case of tree 1 (remain with once-through at 2025).

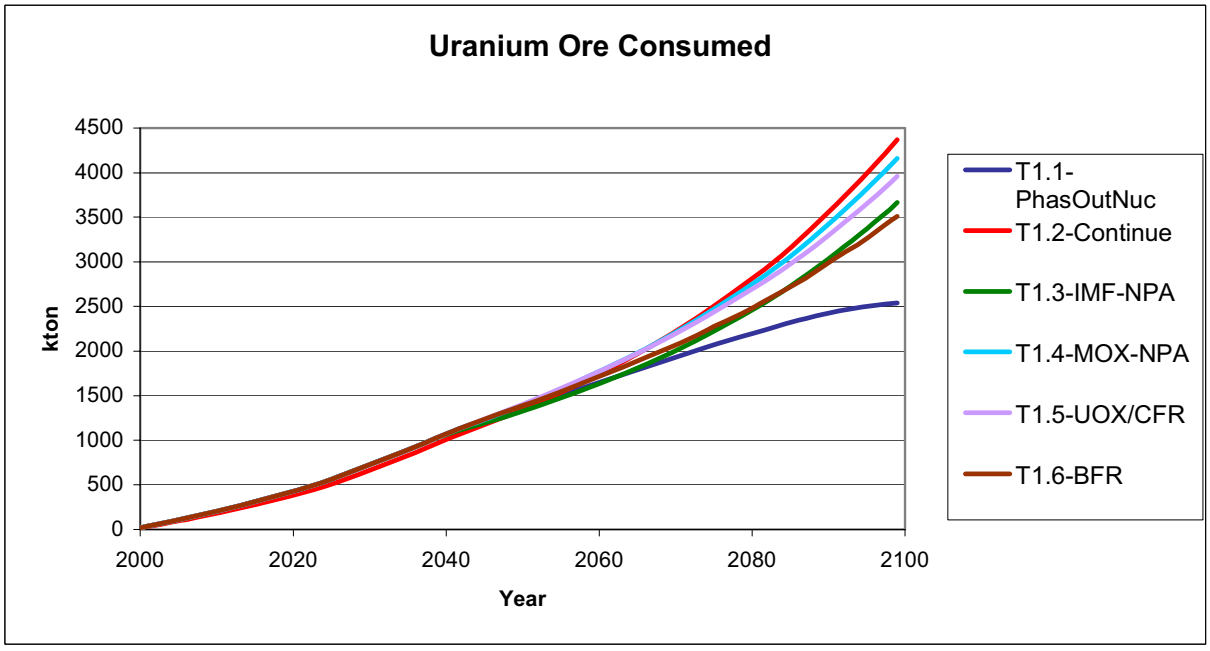

Figure 7.1-8. Uranium ore consumed for each case in tree 1 (remain with once-through at 2025).

7.2. Start IMF-NpPuAm in 2025

Development tree 2 is illustrated in Figure 7.2-1. This tree attempts the fastest possible reduction in LTH, LTD, and LTR using thermal reactors and UREX+ separation technology, but an unproven fuel: Chapter 8 shows that this was achieved. Basic physical principles assure us that IMF-NpPuAm is the fastest way to transmute these three elements, but the particular implementation of this approach is not necessarily optimum, see section 5.1.

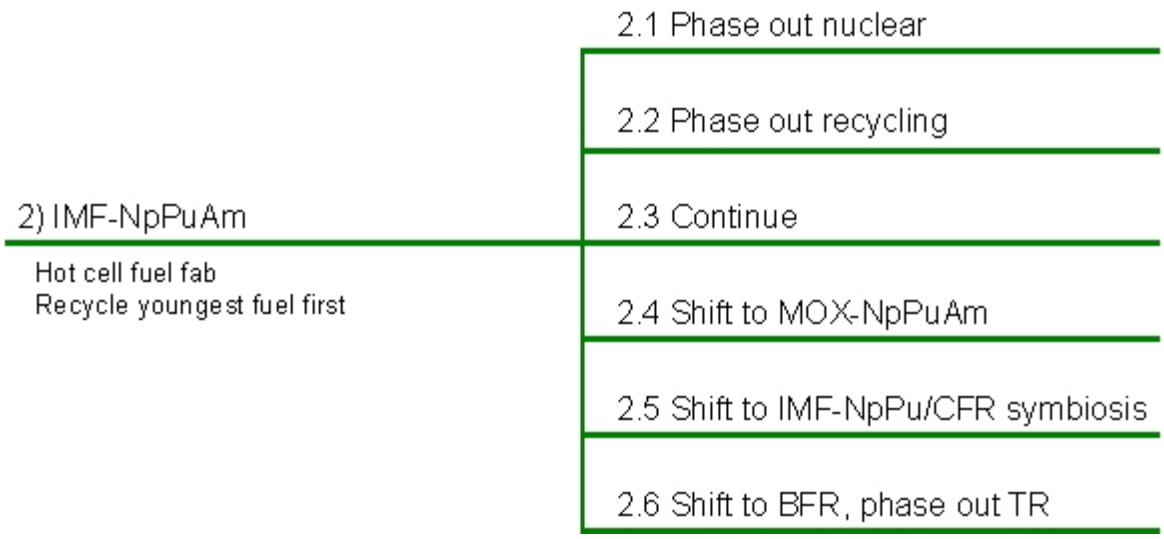

Figure 7.2-1. Development tree for starting IMF-NpPuAm in 2025.

Branches T2.1 and T2.2 reflect phase out, of nuclear power or thermal recycling. In branch 2.1, no new nuclear power plants are ordered after 2040; those in existence are allowed to continue to their 60-year lifetimes, as are the separation and fuel fabrication plants. In branch T2.2, nuclear power is continued, but thermal recycling is phased out quickly after 2040 in favor of the once-through fuel cycle. Branch T2.3 continues IMF-NpPuAm in 2040 thru the end of the century. Branch T2.4 is motivated by a hypothetical combination of potential problems encountered with IMF (i.e. retreat from IMF) and desire to continue recycling in thermal reactors (hence shift to MOX). Branch T2.5 reflects adoption of consumer FR's in 2040, leading to a symbiosis of TR (using UOX and IMF) and CFR. In this case, we assume that it makes more sense to transmute Am (and Cm) in the CFR than in the TR, thus, the IMF shifts from IMF-NpPuAm back to UOX once through since most available spent fuel is sent to FR reprocessing. Branch T2.6 reflects adoption of breeder FR in 2040, with eventual phase out of TR.

As shown in Figure 7.2-2, thermal reactors dominate the system through 2100. Neither consumer (T2.5) nor breeder (T2.6) fast reactors provide much power by 2100. The slow entry of fast reactors into the fleet occurs for two main reasons:

- the number of fast reactors requested is limited in the case of T2.5. Consumer fast reactors are requested (i.e. the percent of power generation capacity coming online to be provided by a specific reactor types) as 30% of the total reactors called but by 2040 most of the first wave of reactor replaced has already occurred and the growth rate is only 1.8% for non-replacement reactors.
- the number of fast reactors started is limited by the amount of plutonium present, MOX and IMF thermal reactors and consumer fast reactors are generally net consumers of plutonium, the recipe for

the breeder fast reactor used in the model has a low breeding ratio and therefore does not generate significant excess plutonium to start additional fast reactors.

These mechanisms restrict the number of consumer and breeder fast reactors present by 2100.

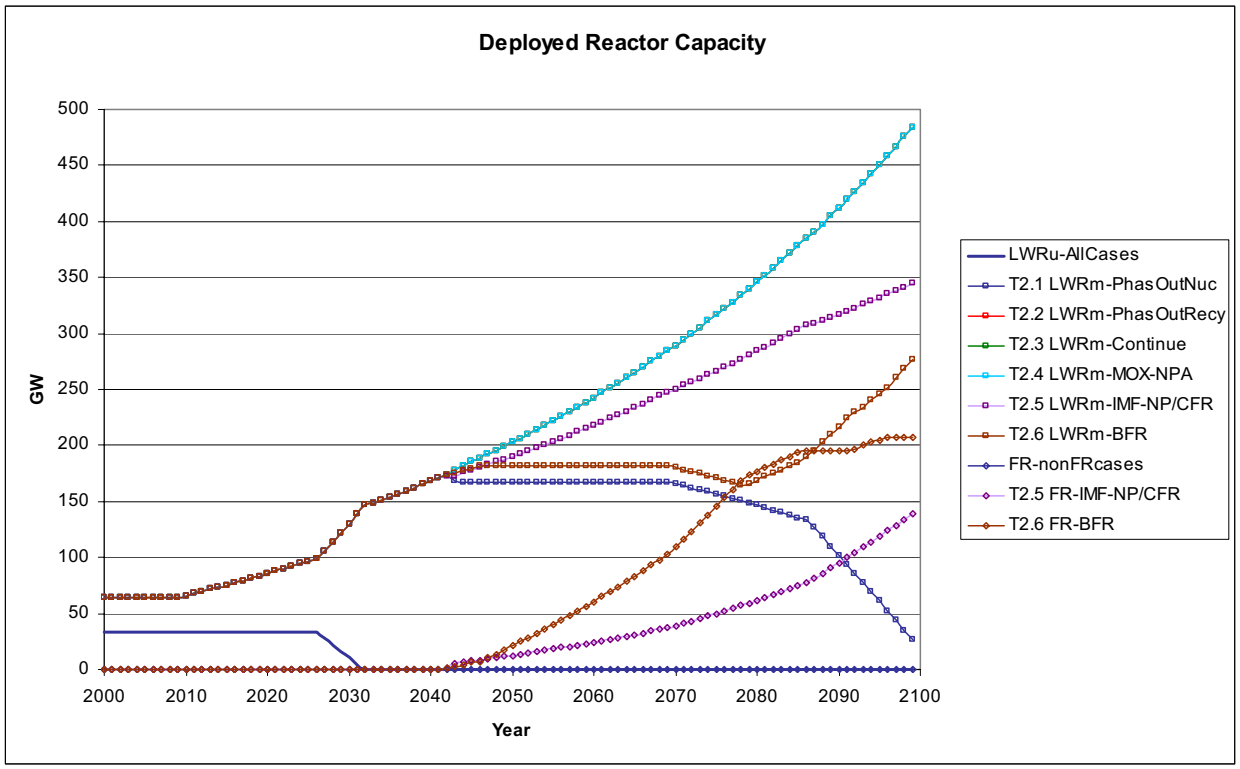

Figure 7.2-2. Power capacity of operating reactors by reactor type for each case in tree 2 (start IMF-multipass in 2025). LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR). T2.2, T2.3, and T2.4 overlay each other.

All of the cases which continue with recycling, IMF, MOX, consumer fast reactor, breeder fast reactor (T2.3 – T2.6) will achieve a 99.5% reduction in TRU content in the repository by 2100. Looking at the graph (Figure 7.2-3) it is clear that IMF could be started later than 2025 and still meet the repository reduction goal, although based on the results of tree 1 (Figure 7.1-3) IMF must start before 2040. Of the options explored, multi-pass IMF provides the quickest reduction of TRU in the repository. The plot for phase out of nuclear (T2.1) is misleading since the recycling of the fuel by IMF keeps fuel out of the repository until sufficient reactors are retired to result in excess IMF fuel; the mass in the repository would eventually equal the mass in the system for the phase out nuclear case (T2.1 in Figure 7.2-4).

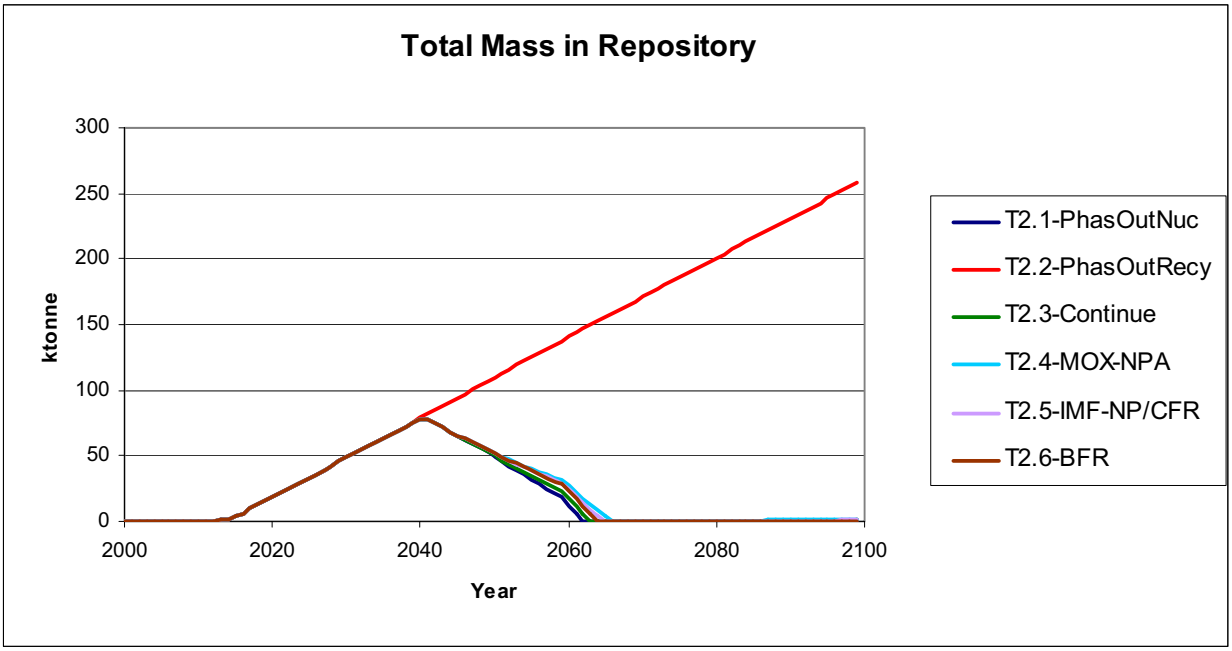

Figure 7.2-3. Total mass in repository for each case in tree 2 (start IMF-multipass in 2025). T2.5 and T2.6 are overlaid.

Figure 7.2-4 shows the total mass in the system as a function of time for each case. As with tree 1, the mass of uranium (Figure 7.2-5) and the total mass track together; this is expected since uranium accounts for the majority of the system mass. Phasing out nuclear power after 2040 (T2.1) results in the smallest total mass. The breeder FR case (T2.6) has the next least amount of mass in the system. This is because the breeder FR requires less uranium (T2.6, Figure 7.2-5) than the other reactor/fuel types and breeder FRs account for approximately 40% of the deployed reactor capacity by 2100 (T2.6 plots, Figure 7.2-2). Little differentiation is seen among the remaining cases; this is due to a combination of uranium usage and percent of deployed reactor capacity for the various fuel/reactor cases. Phase out of recycling moves the system back to UOX fuel which maintains a high uranium usage. Continuing with multi-pass IMF results in a high percentage of IMF in the system by 2100 (T2.3, Figure 7.2-6), but only reduces uranium usage slightly since this is a blended core fuel. Switching to multi-pass MOX reduces the uranium demand by the fuel, but percent of MOX fuel in the system by 2100 is very small (T2.4, Figure 7.2-6). Adding some consumer FRs to the system does not decrease the system or uranium mass significantly; this is because consumer FRs provide no more than 30% of the deployed reactor capacity by 2100 while the remaining LWRs use uranium and the consumer FR in this case has a conversion ratio of 0.25 which means that it has only slightly better uranium usage than thermal reactors (see Chapter 5).

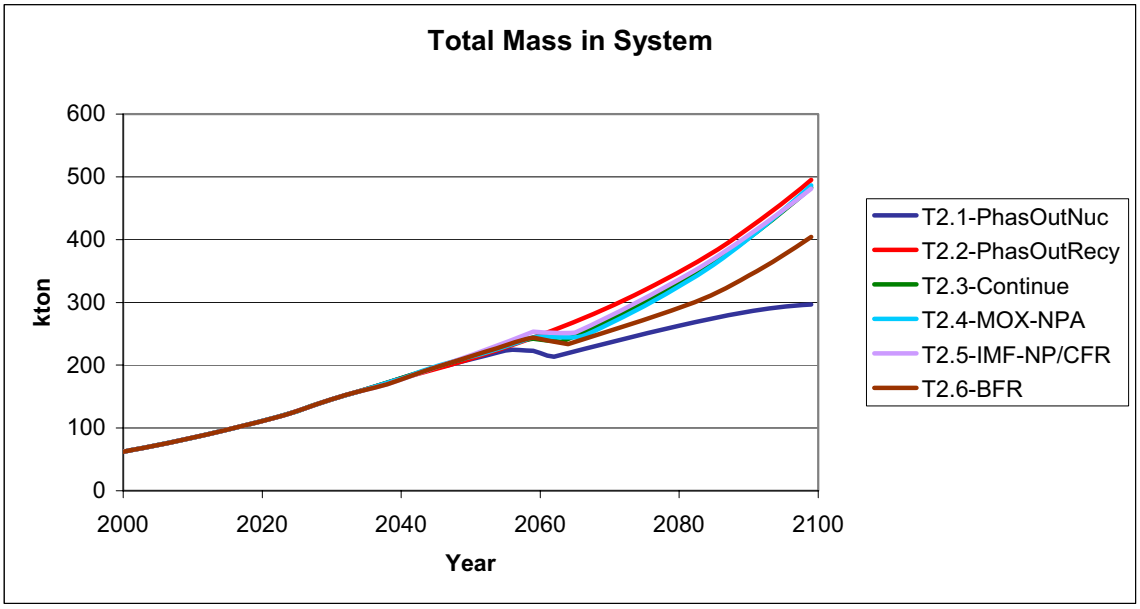

Figure 7.2-4. Total mass in the system for each case in tree 2 (start IMF-multipass in 2025). T2.3 is overlaid by T2.4 and T2.5.

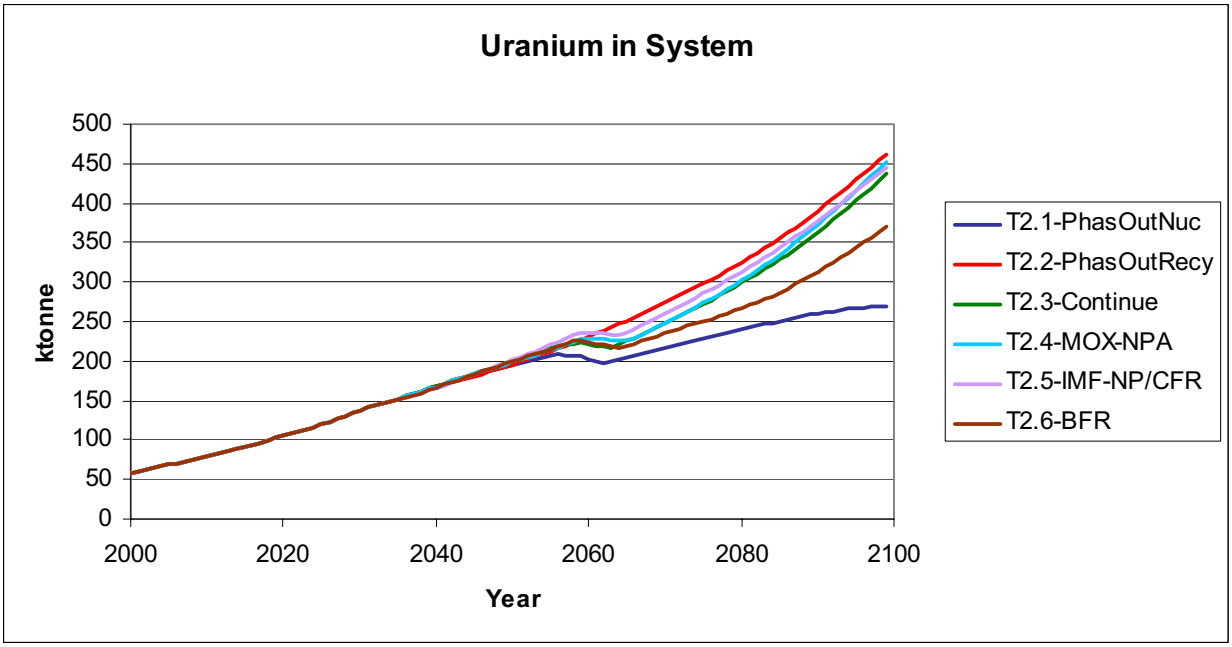

Figure 7.2-5. Mass of uranium in system for each case in tree 2 (start IMF-multipass in 2025).

Starting multi-pass IMF in 2025 results in a substantial movement of IMF into the reactor fleet by 2040. This is due to large number of legacy reactors retiring during this time and the reservoir of legacy spent fuel available for making IMF. When multi-pass IMF continues through the century, the system becomes IMF limited around 2085 (T2.3). This is likely due to two factors. First, in the current model, multi-pass IMF has a finite recycle life, after 5 cycles (each for 12 yrs) the fuel is sent to the repository or fast reactors (if available), IMF fuel started in 2025 is being retired in 2085. Second, there is no legacy fuel

remaining to provide extra IMF fuel for new reactors coming on-line, earlier in the century legacy spent fuel provided start-up fuel for new reactors, that otherwise would have been started with UOX.

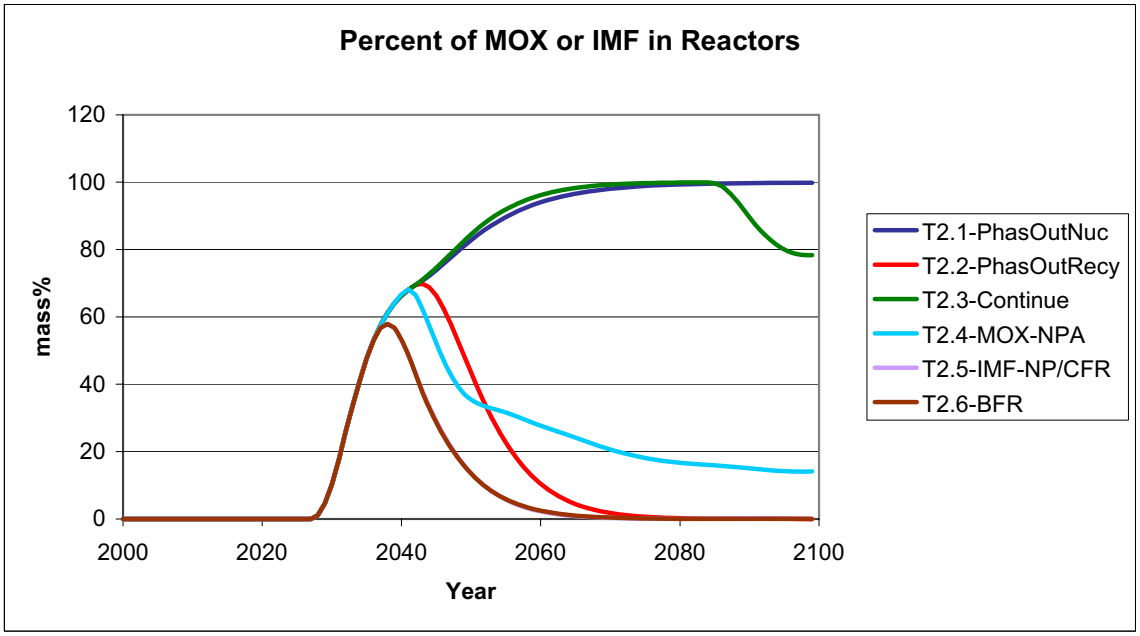

Figure 7.2-6. Percent of MOX or IMF fuel in reactors for each case in tree 2 (start IMF-multipass in 2025). T2.5 and T2.6 are overlaid.

Other than for phasing out nuclear, continuing with IMF results in the greatest reduction of plutonium (see Figure 7.2-7). Switching to MOX at 2040 (T2.4) would retain more plutonium in the system, while still keeping waste out of the repository. This option could be helpful if implementation of fast reactors is delayed.

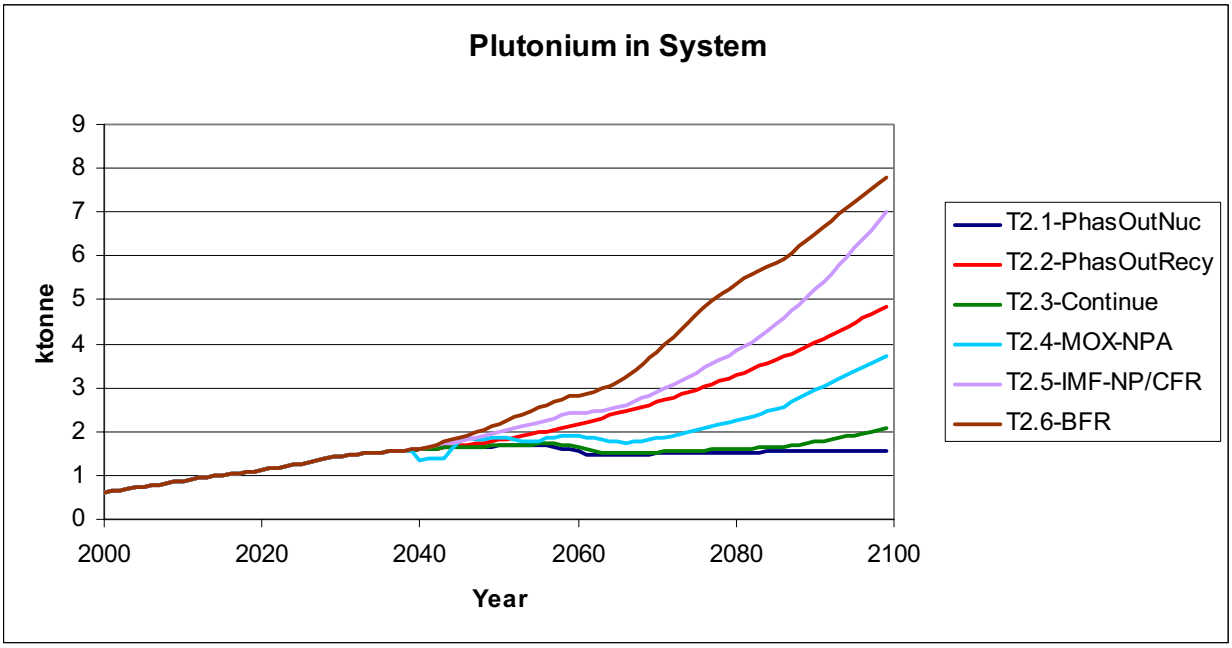

Figure 7.2-7. Mass of plutonium in system for each case in tree 2 (start IMF-multipass in 2025).

Figure 7.2-8 shows the uranium ore consumed for each case. Phasing out nuclear (T2.1) uses the least uranium, followed by breeder fast reactors (T2.6). The highest (by a small amount) uranium ore usage through 2100 is shown by switching to multi-pass MOX (T2.4) or adding consumer fast reactors (T2.5). This is not surprising since multi-pass IMF was started in 2025 and approximately 60% of the reactor fleet is using IMF by 2040. The support ratio for multi-pass MOX is higher than that for IMF so less reactors can initially be supported by MOX fuel than IMF fuel; therefore additional UOX is needed compared to remaining with IMF. Switching to consumer fast reactors essentially means converting most of the reactor system to UOX fuel, since the number of consumer fast reactors increases slowly (see Figure 7.2-2) and no new IMF is being made (recall the discussion in chapter 7 introduction about fuel reprocessing when fast reactors are being requested).

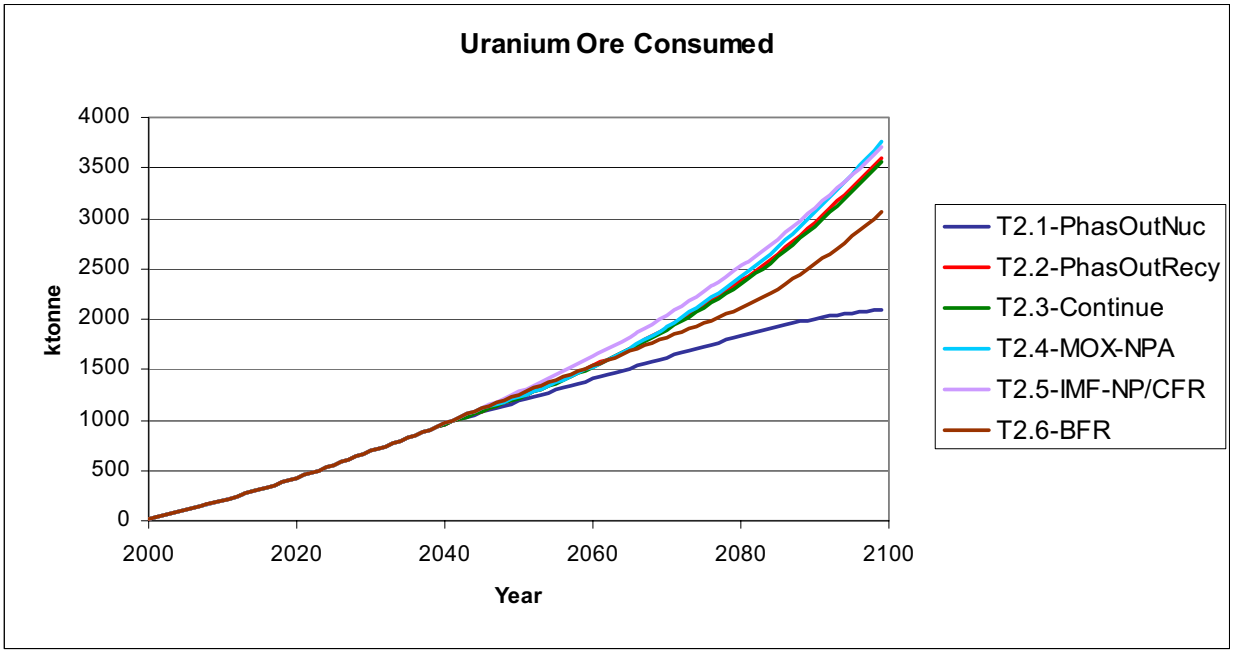

Figure7.2-8. Mass of uranium ore consumed for each case in tree 2 (start IMF-multipass in 2025).

7.3. Start MOX-NpPu in 2025

Development tree 3 is illustrated in Figure 7.3-1. This tree is primarily motivated as the closest to current international practice and closest to current technology, while avoiding separation of Pu. Instead, Np and Pu are always kept together. MOX-NpPu fuel would, like MOX-Pu, be fabricated with glovebox technology. For DYMOND calculations, we assume that the oldest fuel is separated and recycled first. However, this assumption makes little difference because (a) we are already assuming that glovebox fabrication is adequate and (b) without inclusion of Am in the recycle fuel, we get little reductions in LTH whether or not Pu241 has had time to decay.

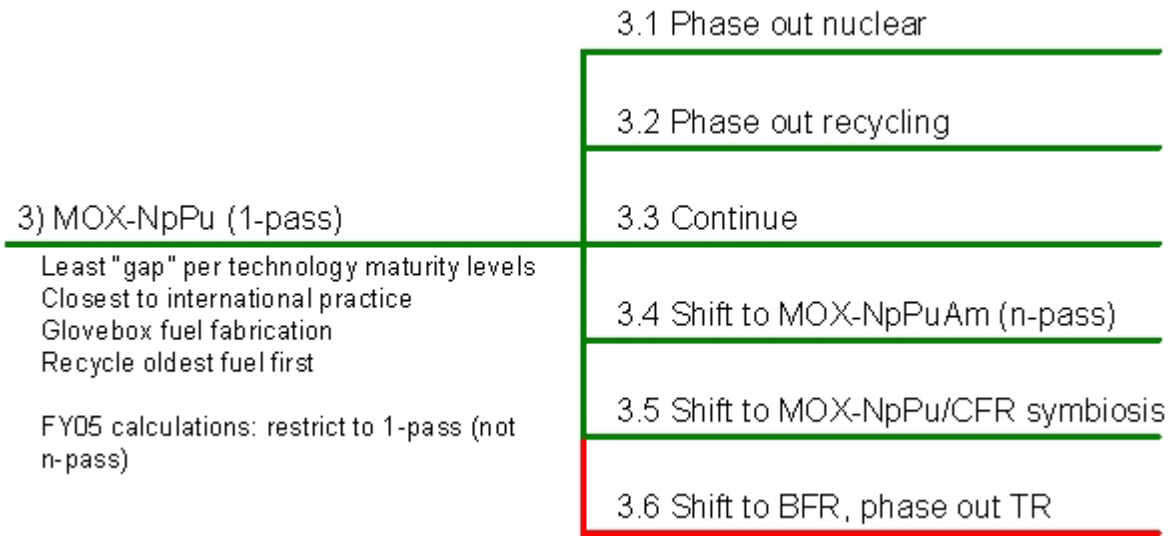

Figure 7.3-1. Development tree for starting MOX-NpPu in 2025. (Lacking fuel composition recipes, there are no DYMOND calculations for branch T3.6.)

In DYMOND calculations, we treat this case as being limited to 1-recycle pass of the MOX-NpPu. This is primarily because we lack fuel composition recipes for n-pass recycling.

As with tree 2, the first two branches at 2040 reflect phase-out of nuclear reactors and recycling and the third branch is simply continuing what was started in 2025. Branch T3.4 reflects a hypothetical shift to a long-term thermal recycling strategy; the penalty of going from glovebox to remote fuel fabrication would be accepted and reductions in LTH, LTD, LTR would accelerate because Am would be recycled. In contrast, Branch T3.5 assumes that consumer FR's (CFR) are introduced in 2040 and that they serve the function of reducing Am (rather than TR in Branch T3.4). Branch 3.6 shows introduction of breeder FR (BFR) and the phasing out of TR.

There are, of course, intermediate cases one could devise, such as a symbiosis of BFR and TR. Such a case would be appropriate if we needed high-temperature TR reactors (VHTR) for hydrogen production but wanted the uranium utilization and waste management benefits of BFR.

The distribution of thermal and fast deployed reactor capacity is shown in Figure 7.3-2.

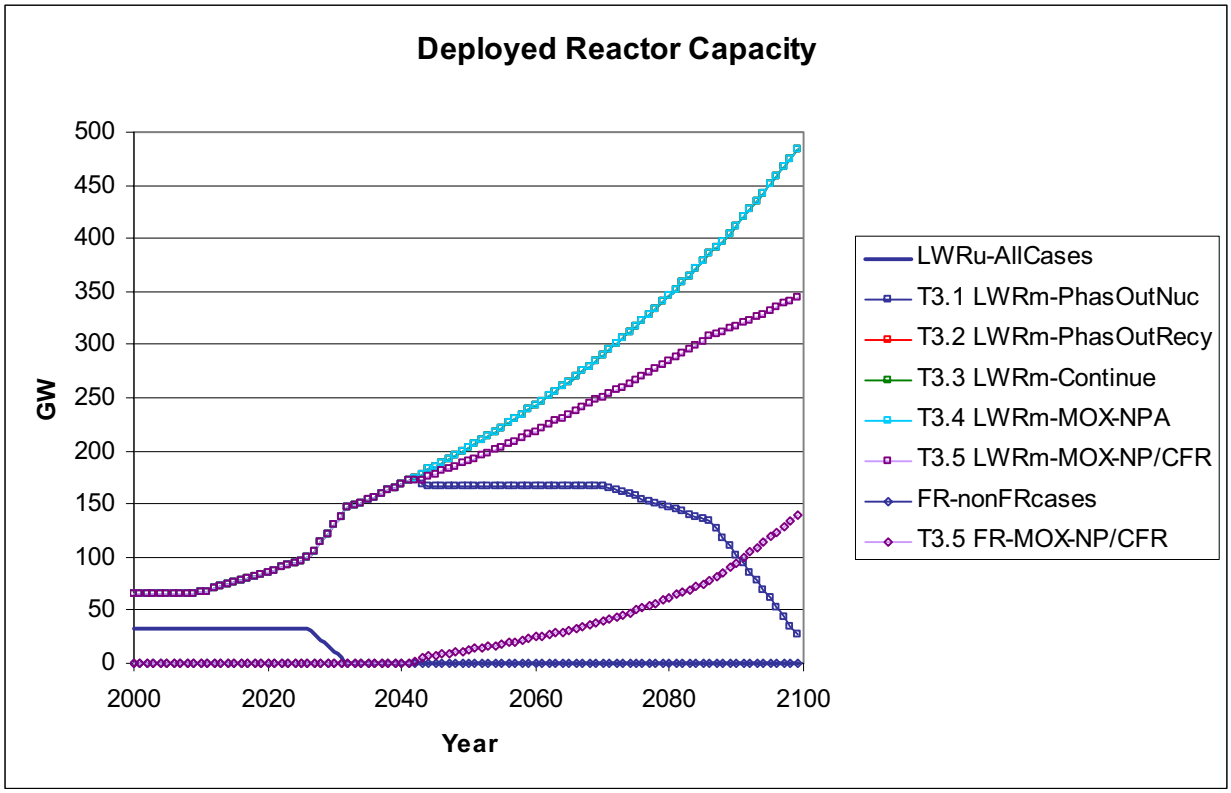

Figure 7.3-2. Power capacity of operating reactors by reactor type for each case in Tree 3 (start MOX-onepass in 2025). LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR). T3.2, T3.3, and T3.4 are overlaid.).

Only the two multi-pass recycling options, multi-pass MOX (T3.4) and consumer fast reactor (T3.6) can achieve a 99.5% reduction in TRU content in the repository by 2100 (see Figure 7.3-2). Single-pass

MOX (T3.3) is better than switching to once through (T3.2) in 2040, but only slows the rate of material entering the repository.

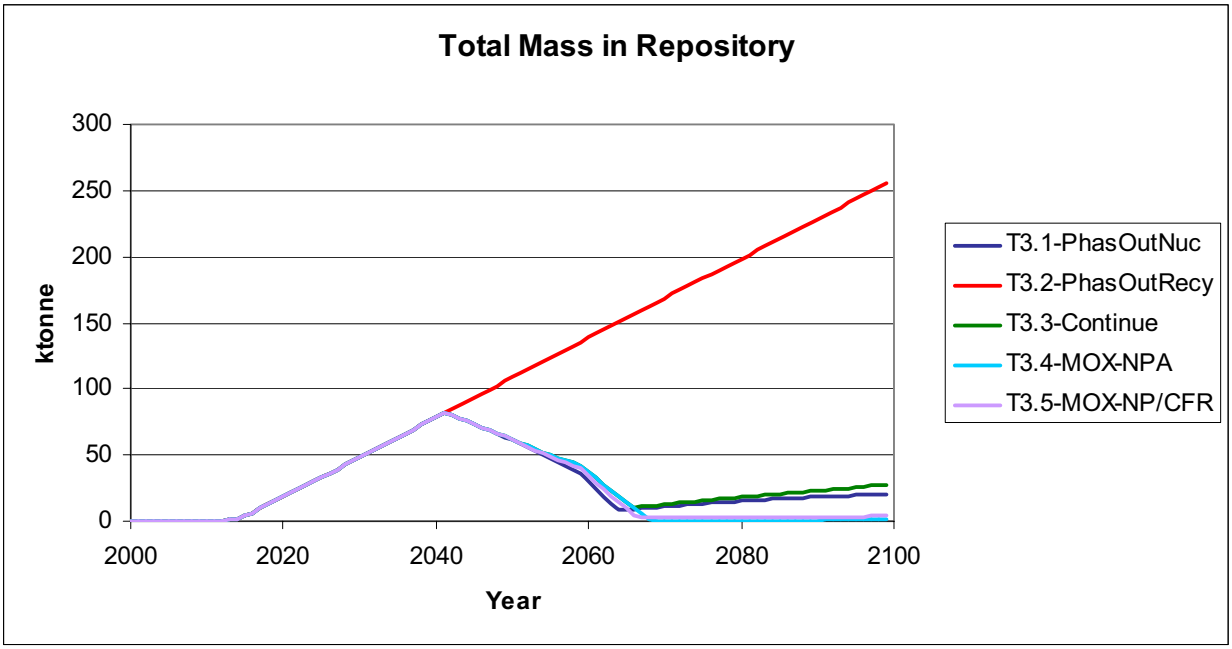

Figure 7.3-3. Total mass in repository for each case in tree 3 (start MOX-onepass in 2025).

Figure 7.3-4 shows the total mass in the system as a function of time for each case. Phasing out nuclear power (T3.1) after 2040 results in the smallest total mass in the system Of the thermal reactor cases, phasing out recycling in 2040 results in slightly more mass in the system than continuing with one pass MOX. Switching to multi-pass MOX or consumer fast reactors in 2040 reduces the total mass in the system. As with tree 1, the mass of uranium (Figure 7.3-5) and the total mass track together; this is expected since uranium accounts for the majority of the system mass.

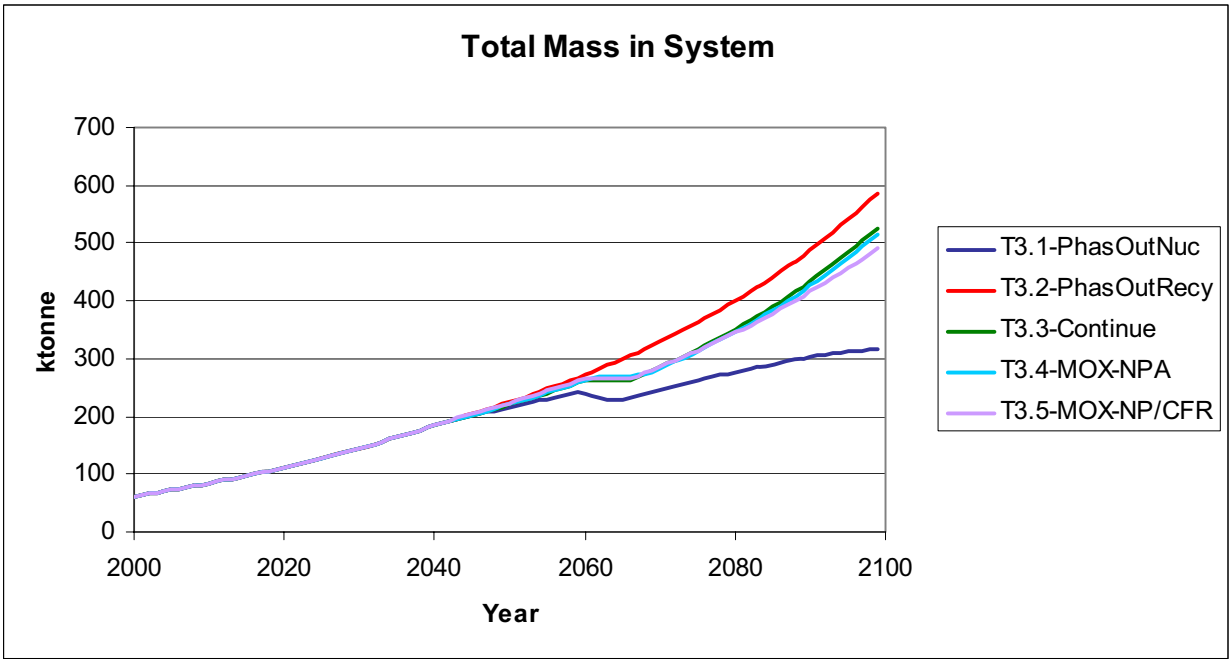

Figure 7.3-4. Total mass in the system for each case in tree 3 (start MOX-onepass in 2025).

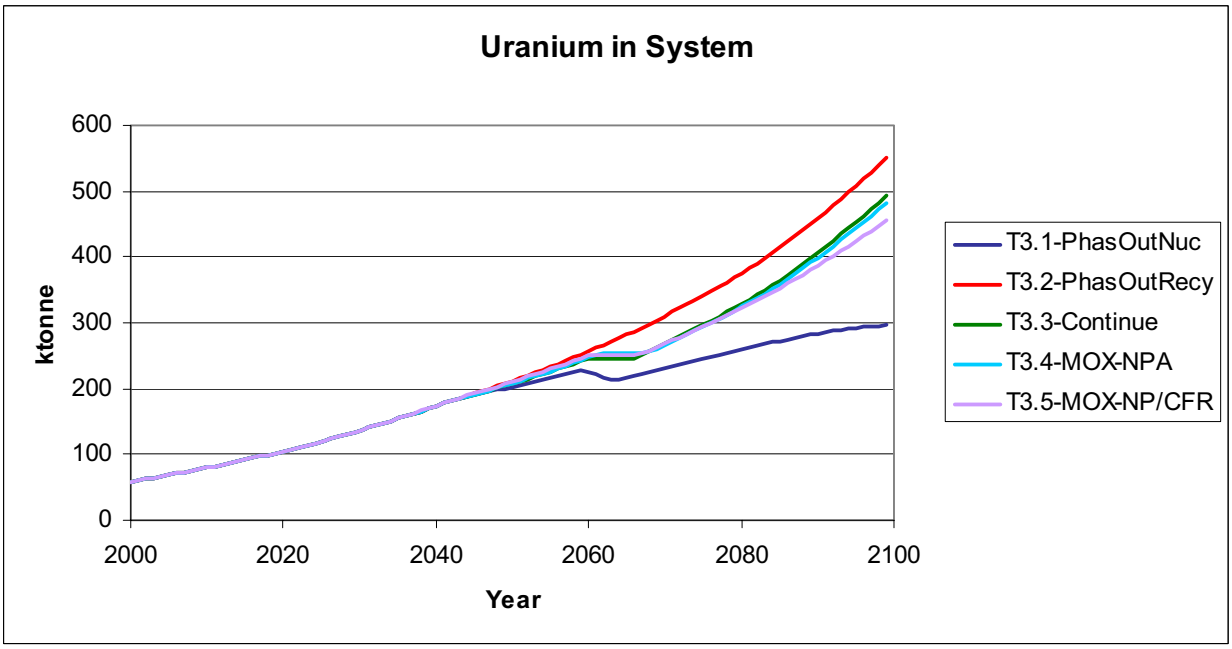

Figure 7.3-5. Mass of uranium in system for each case in tree 3 (start MOX-onepass in 2025).

By 2040, only approximately 6% of the reactor fuel is one-pass MOX (Figure 7.3-6); this is much lower percentage of the system fuel than was seen for multi-pass IMF in tree 2. Like multi-pass MOX, one-pass MOX has a steeper support ratio than multi-pass IMF; several reactors loads of spent fuel are needed for every reactor load of MOX. The doubling of the reprocessing rate in 2040 allows more legacy spent fuel to be reprocessed and temporarily increases the amount of MOX is available (see T3.1 and T3.3). If

nuclear is phased out (T3.1), the percent of MOX in reactors increases through 2060 because the number of reactors is steady while the amount of reprocessing capacity increases and legacy fuel is available. The second increase after 2090 is because the number of reactors is decreasing, while the reprocessing capacity remains constant. A switch to multi-pass MOX in 2040, results in a more than a doubling of the percent of MOX in reactors. The decrease in the percent of MOX in reactors after 2070 is due to a lack of additional fuel from legacy reactors, a constant reprocessing rate from 2060 – 2080 and the continued growth of the reactor fleet.

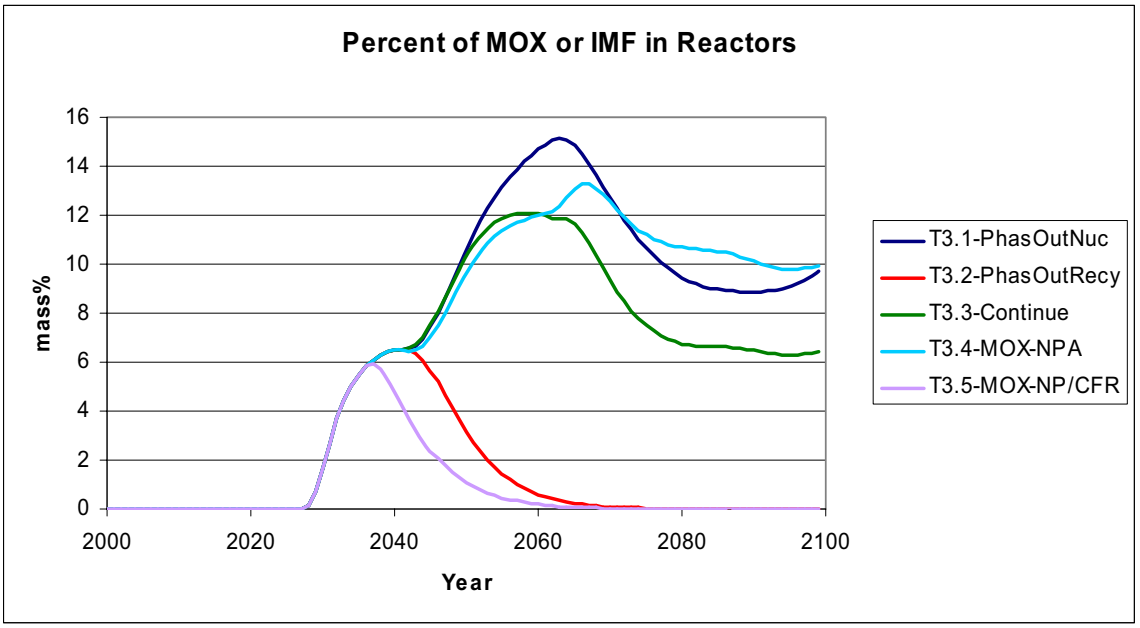

Figure 7.3-6. Mass percent of MOX fuel in reactors for each case of tree 3 (start MOX-onepass in 2025).

Other than for phasing out nuclear, moving to multi-pass MOX at 2040 results in the greatest reduction of plutonium (see Figure 7.3-7) in the system. Curves T3.3, 3.4, 3.5 peak around 2060, presumably because that is the time of maximum Pu availability as spent fuel has been withdrawn from the respository.

The plot of uranium ore usage (Figure 7.3-8) shows little differentiation among the cases (except for phasing out nuclear power), this is probably because MOX and consumer reactors account for only a small percentage of the reactor fleet; most of the fleet uses UOX through 2100.

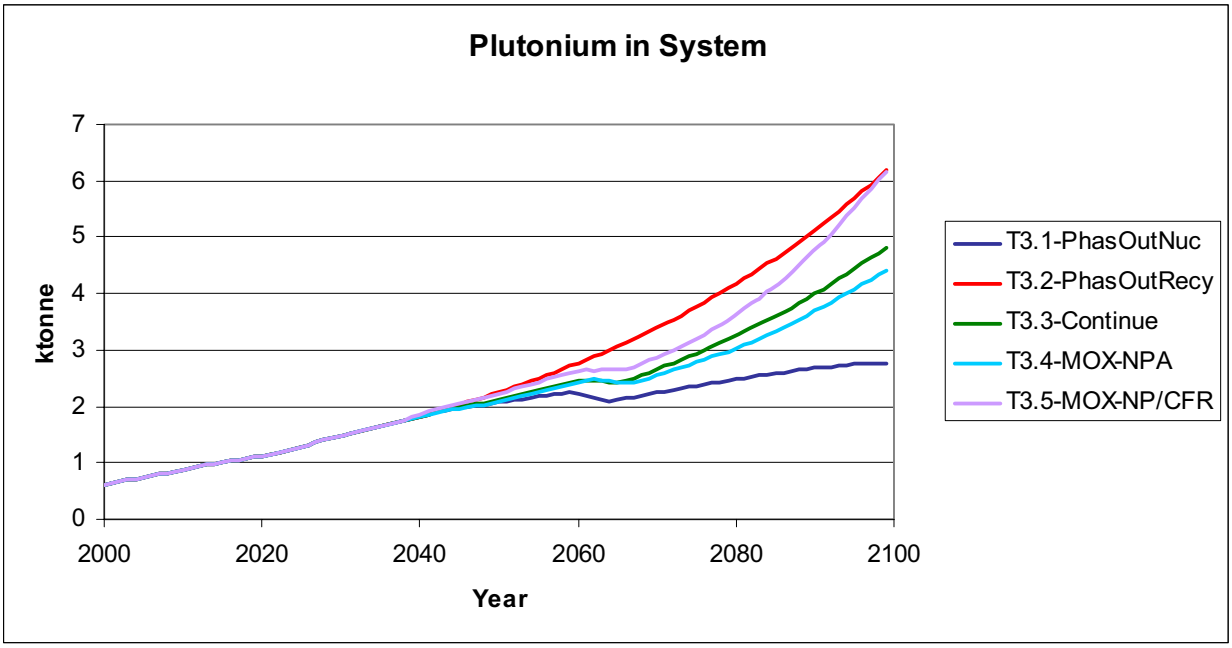

Figure 7.3-7. Mass of plutonium in system for each case of tree 3 (start MOX-onepass in 2025).

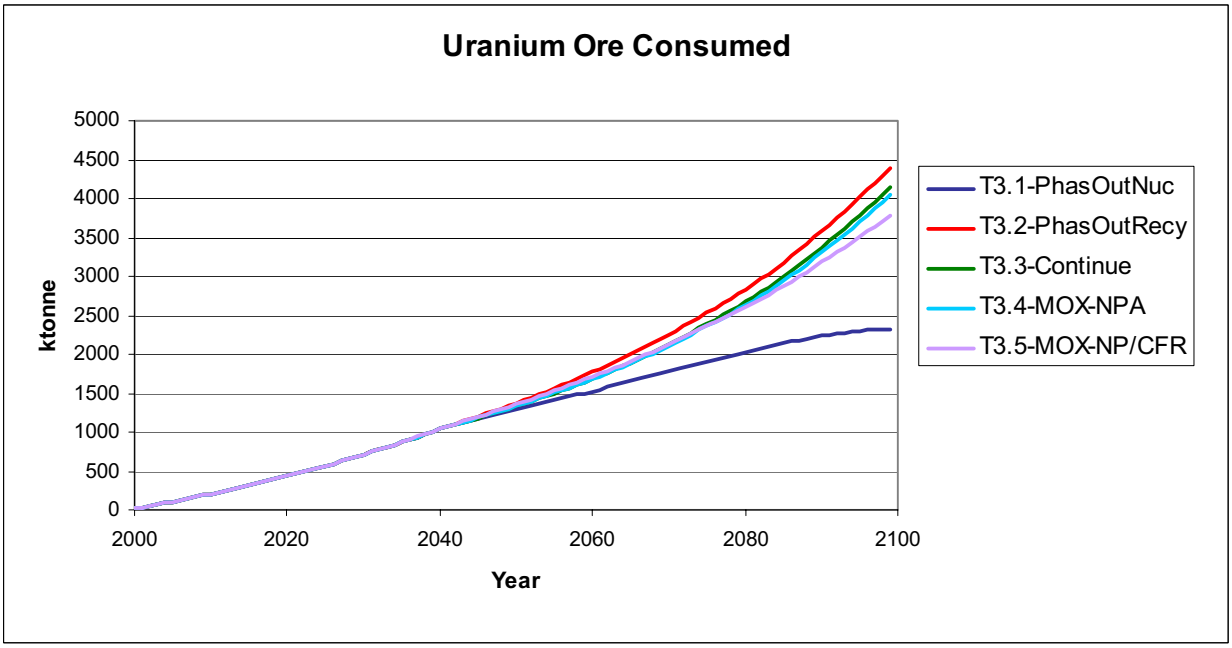

Figure 7.3-8. Uranium ore consumed for each case in tree 3 (start MOX-onepass in 2025).

7.4. Start MOX-NpPuAm in 2025

This tree attempts modest repository benefits using thermal reactors, UREX+ technology, and fuels relatively similar to current UOX and MOX-Pu. It can be said to be a U.S. analog to current international practice. The primary differences are inclusion of Np (and Am) to avoid separation of pure Pu and inclusion of Am and Np to start reduction in LTD, LTH, and LTR.

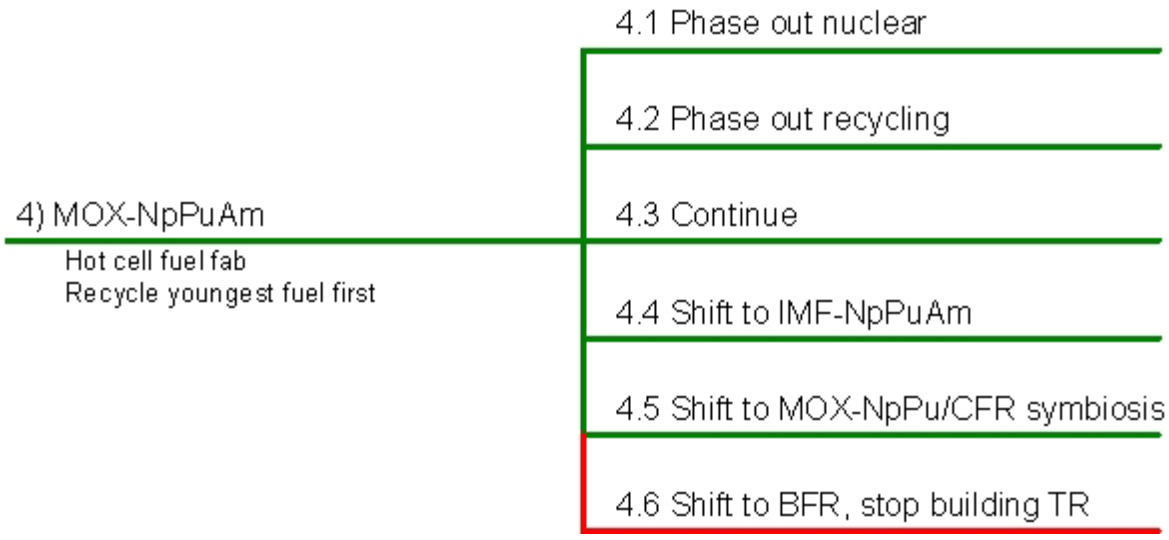

Figure 7.4-1. Development tree for starting MOX-NpPuAm in 2025. (Lacking fuel composition recipes, there are no DYMOND calculations for branch 4.6.)

As with trees T2 and T3, branches 1 and 2 reflect phase out scenarios. Branch T4.3 continues MOX-NpPuAm n-pass recycling in 2040, to the end of the century. Branch T4.4 could occur under two related conditions. First, IMF wasn't ready in 2025 (hence start thermal recycling with MOX) but is ready in 2040. Second, the reductions in LTH, LTD, LTR, and weapons-usable inventory obtainable by MOX are determined to be insufficient, hence motivating a shift from MOX to IMF.

Branches T4.5 and T4.6 reflect deployment of FR. As with tree T2 and T3, if CFR are built, it is assumed that they are superior for burning Am than TR, thus, the TR fuel shifts from MOX-NpPuAm to MOX-NpPu. As with branch T3.6, the lack of MOX-to-BFR fuel recipes prevents DYMOND calculations for that case.

Thermal reactors dominate the energy production through 2100 (see Figure 7.4-2). Consumer fast reactors come online very slowly because of the lack of plutonium and because they, like MOX and IMF, are net plutonium consumers.

Fuel for fast reactors comes from four places: 1. UOX fuel in the system, 2. MOX or IMF fuel after its first cycle, 3. MOX or IMF fuel after its last cycle, or 4. fuel in existing fast reactors.

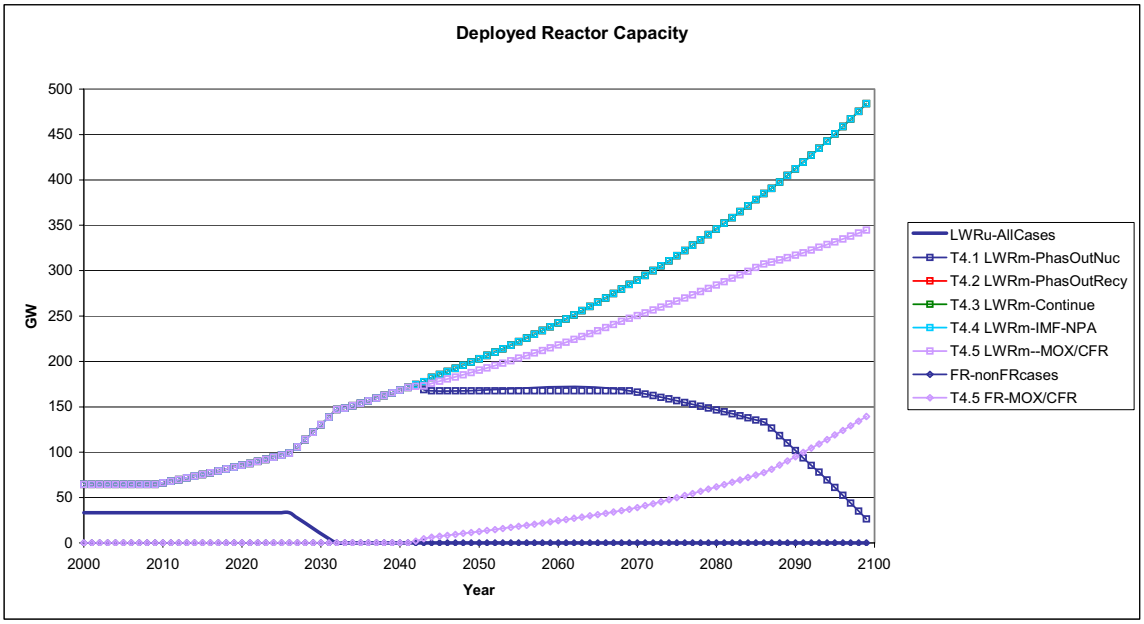

Figure 7.4-2. Power capacity of operating reactors by reactor type for each case in tree 4 (start multipass-MOX in 2025). LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR). T4.2, T4.3, and T4.4 are overlaid.

All of the cases except phase out recycling (T4.1) and phase out nuclear (T4.1) result in a 99.5% reduction in TRU content in the repository by 2100. Of the options explored, multi-pass IMF provides the quickest reduction of TRU from the repository. The plot of mass for phase out of nuclear (T4.1) would go back up and match the total system mass by 2100 (see T4.1 in Figure 7.4 -6).

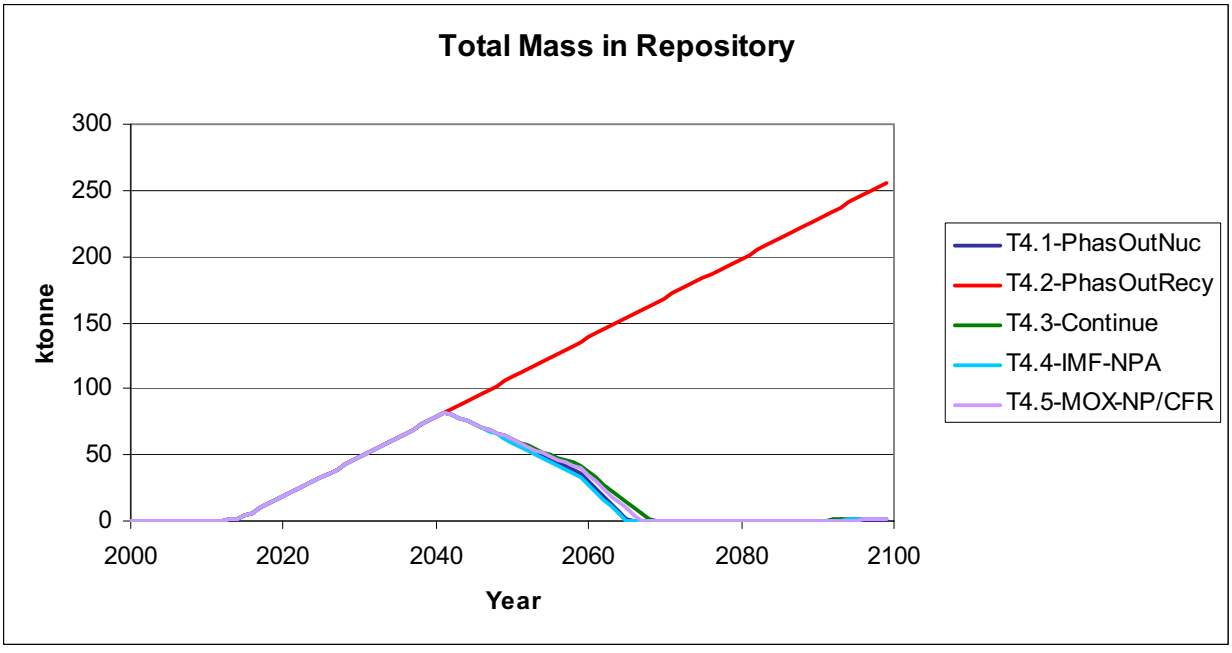

Figure 7.4-3 Total mass in repository for each case in tree 4 (start multipass- MOX in 2025). T4.1, T4.3, and T4.5 overlap.

As with the previous three trees, the total mass in the system (Figure 7.4-3) shows the same trends as the uranium in the system (Figure 7.4-4). Reverting to once through (T4.2) results in the highest mass in the system, while phasing out nuclear (T4.1) results in the smallest mass. The difference between staying with mult-pass MOX (T4.3), switching to multi-pass IMF at 2040 (T4.4) or adding consumer fast reactors at 2040 is very small, at least by 2100.

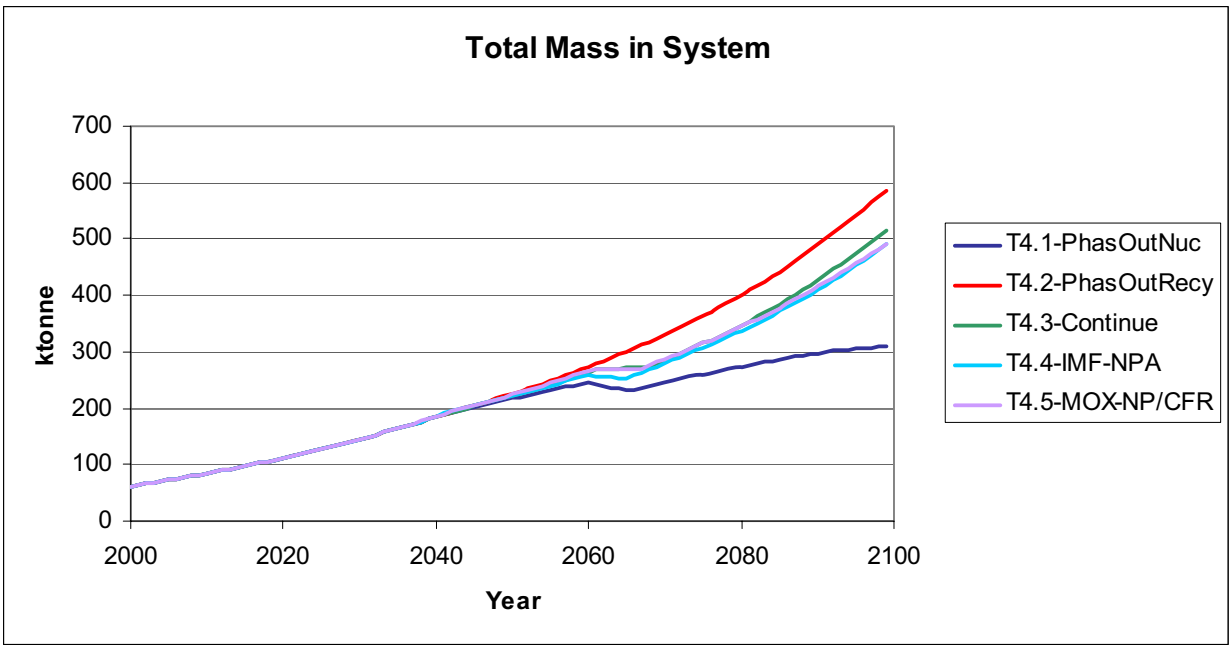

Figure 7.4-4. Total mass in system for each case of tree 4 (start multipass-MOX in 2025).

The mass of uranium in the system (Figure 7.4-5) for multi-pass MOX (T4.3) and multi-pass IMF (T4.4) and consumer reactors (T4.5) are very close. The amount of fresh uranium does not decrease much with MOX (T4.3) because of the steep support ratio between successive cycles of MOX means that only a small portion of the reactor fleet contains MOX (see Figure 7.4-6). In the multi-pass MOX and consumer reactor case presented here, only 30% of new reactors are consumers, so only a small number of consumer fast reactors are present in the system by 2100 (see Figure 7.4-2). This means that in both cases, most of the reactor fleet is running on UOX.

The mass of uranium in the system with IMF (T4.4, Figure 7.4-5) is slightly less than that for MOX or MOX and consumer fast reactor. IMF fuel moves into the reactor fleet more quickly than MOX fuel (see Figure 7.4-6). This is due to the recipe for multi-pass IMF which has close to a 1:1 support ratio. However, the IMF recipe used in this case is for a blended core, this means a large percentage of the core is UOX so the overall amount of uranium in the system does not decrease significantly.

Starting multi-pass IMF in 2040 results in a substantial movement of IMF into the reactor fleet by 2080 (T4.4, Figure 7.4-6). This is due to legacy reactors retiring during this time and the reservoir of legacy spent fuel available for making IMF. As multi-pass IMF continues through the century, the system becomes IMF limited around 2085 (T4.4). This is due to legacy fuel no longer being available (it was consumed prior to 2080) to provide extra IMF fuel for new reactors coming on-line, earlier in the century legacy spent fuel provided start-up fuel for new reactors, that otherwise would have been started with UOX. The slight rise in the percent of multi-pass MOX in the system around 2090 is due to reactors

leaving the system while reprocessing capacity remains constant which keeps the MOX supply constant or increasing slightly, so a higher percentage of the fuel in each reactor is MOX.

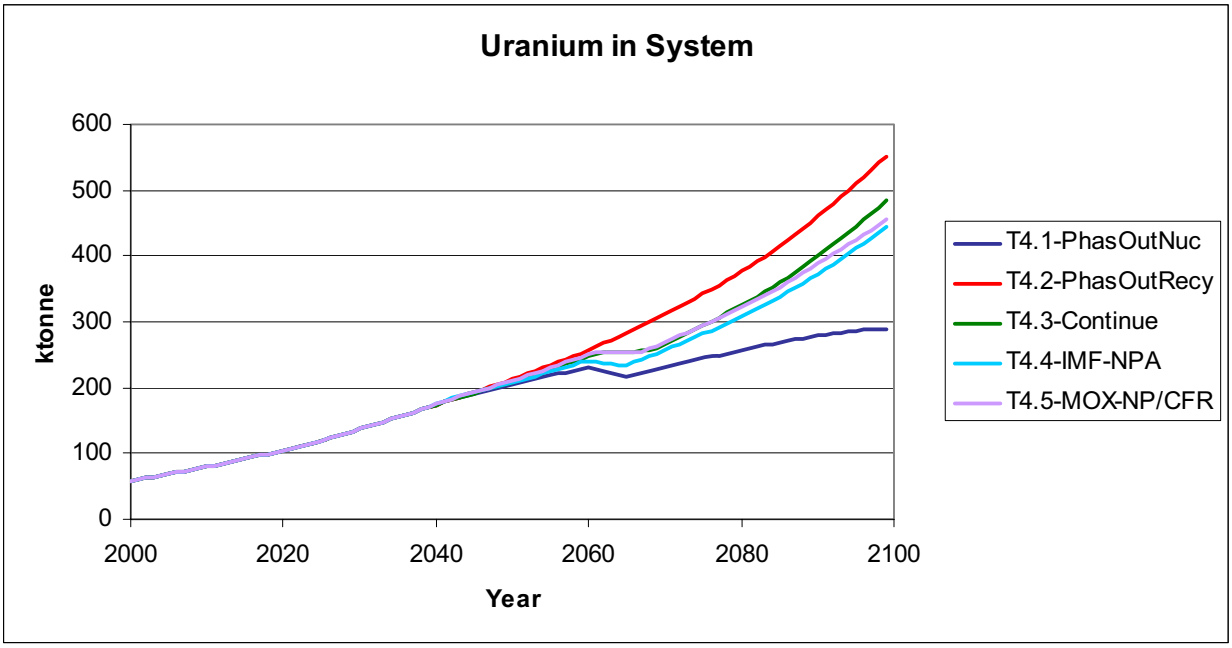

Figure 7.4-5. Mass of uranium in system for each case of tree 4 (start multipass-MOX in 2025).

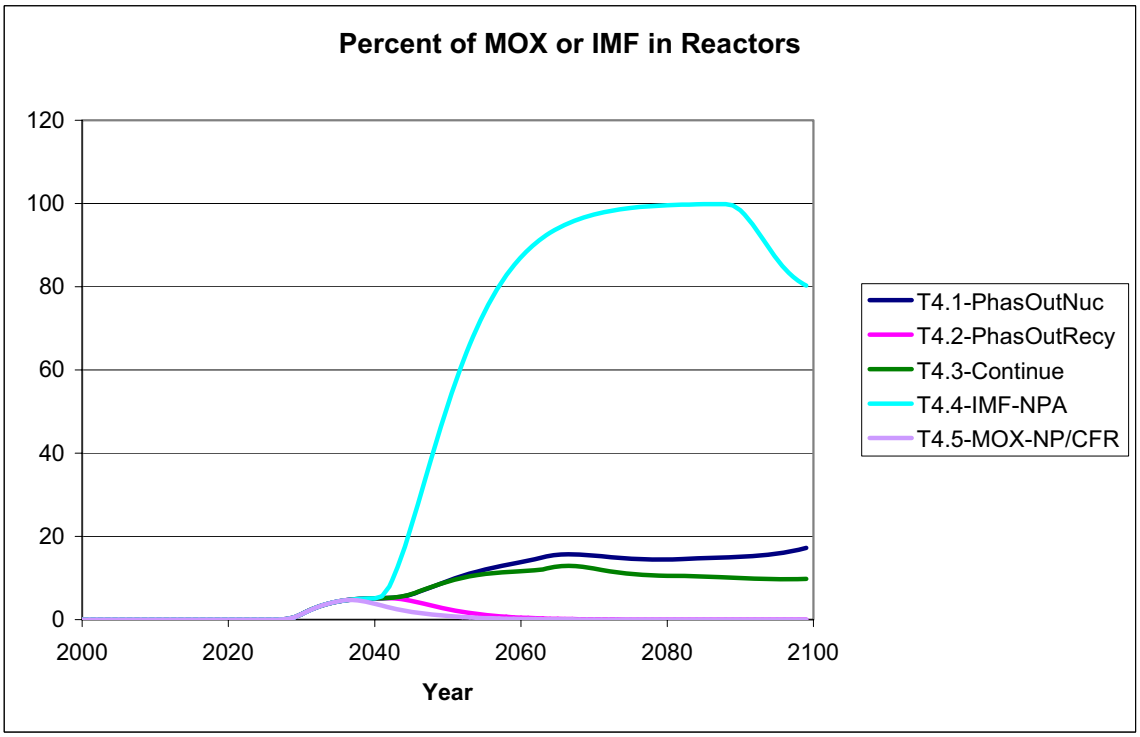

Figure 7.4-6. Mass percent of MOX or IMF fuel in reactors for each case of tree 4 (start multipass-MOX in 2025). The multi-pass IMF (T4.4) has a low support ratio and is blended core fuel. The multi-pass MOX (T4.3) has a higher support ratio and is a full core fuel.

IMF in this case (see T4.4 of Figure 7-4.7) reduces the amount of Pu in the system to a level comparable to the amount of Pu that would be in the system if we shut down nuclear power starting in 2040. IMF does a very good job of consuming Pu. Multi-pass MOX reduces the amount of Pu in the system more than the consumer FR case but only because of the limited number of consumer FRs that are on line by the end of the century.

Moving to multi-pass fuel systems (see Figure 7.4-8, MOX (T4.3), IMF (T4.4) or CFR (T4.5)) reduces the amount of uranium ore consumed compared to a once through system (T4.2). Moving to IMF fuel at 2040 results in the largest reductions in the amount of uranium ore consumed by 2100. All cases except, for terminating nuclear power (T4.1) consume a substantial amount of uranium ore.

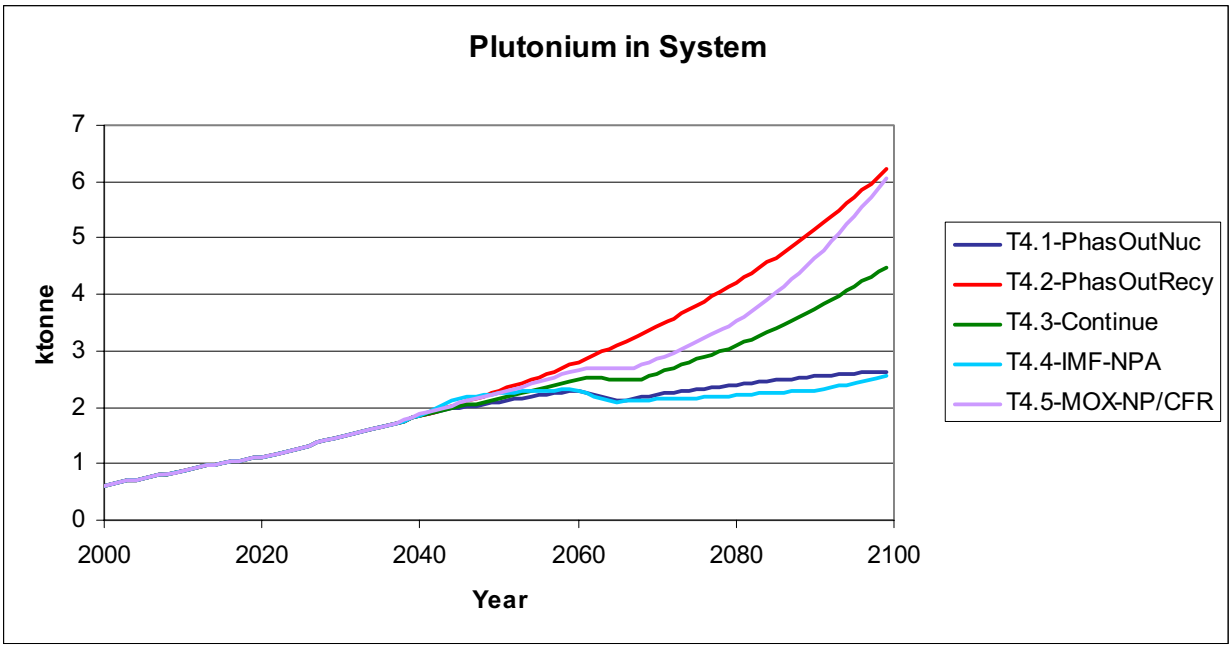

Figure 7.4-7. Mass of plutonium in system for each case of tree 4 (start multipass-MOX in 2025).

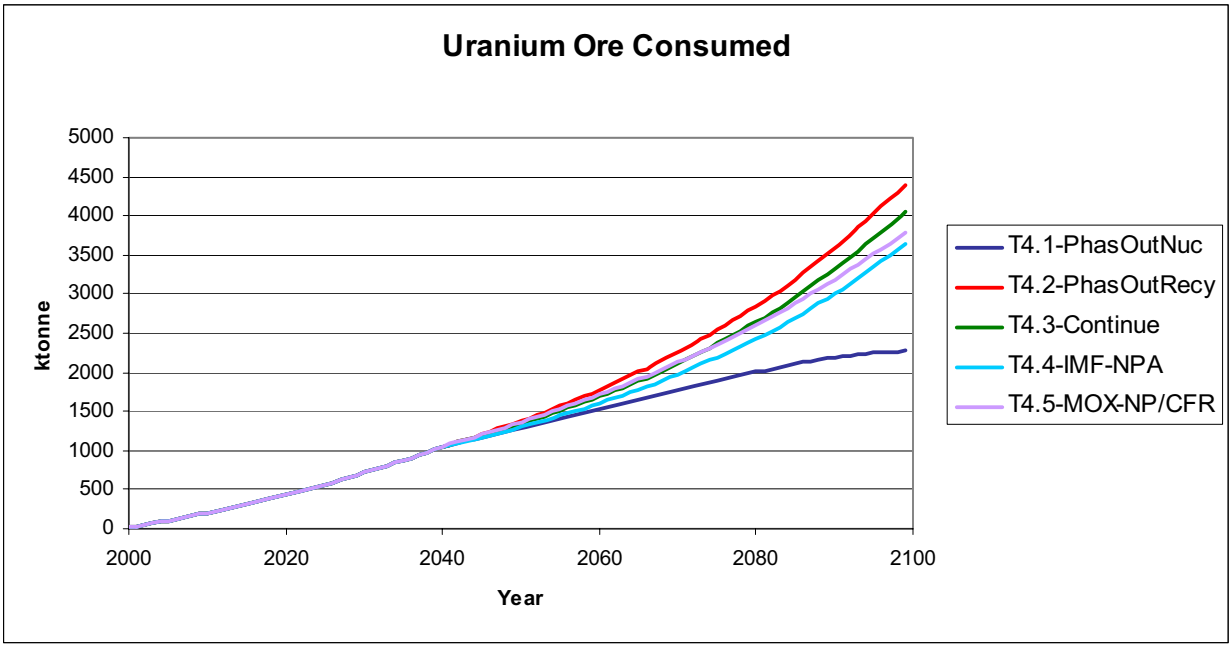

Figure 7.4-8. Uranium ore consumed for each case in tree 4 (start multipass-MOX in 2025).

7.5. Start Consumer fast reactors in 2025

This tree starts CFR, skipping recycling in TR. The early FR experience would set the stage for BFR when uranium resources warrant. As with the other trees, the first branch in 2040 is phase out of nuclear. Branches T5.2, T5.3, and T5.4 show phase out of the FR, e.g. if performance is inadequate. (One could argue that this is what the French did when they terminated SuperPhenix.) In branch T5.2, the FR is replaced with simply once-through. Thus, branch T5.2 is the same as once-through except for a few FR built between 2025 and 2040. In branches T5.3 and T5.4, the FR is replaced with thermal recycling via MOX and IMF respectively. Branch T5.5 continues the status quo and branch 6.6 accelerates the transition to FR.

Figure 7.5-1. Development tree for starting consumer FR in 2025.

The rate that consumer fast reactors are built depends on the request rate, the growth rate of the fleet, and the amount of plutonium available for starting the fast reactors. For the cases presented here, the request (i.e. the percent of power generation capacity coming online to be provided by specific reactor types) is 30% of new construction from 2025 to 2040 and then either continuing with that request (case 5.5), reducing that request to zero (cases 5.1, 5.2, 5.3, and 6.4), or increasing the request to 100% (case 5.6) and switching to breeder fast reactors. As for all the other cases discussed in chapter 7, the growth rate is 1.8% per year starting in 2010. The amount of plutonium available for starting the fast reactors depends on the recent history of plutonium generation and usage by the reactor fleet. When fast reactors are turned on, the fast reactors receive all UOX spent fuel, IMF or MOX pass 1 spent fuel, IMF pass 5 or MOX pass 8 spent fuel, and all fast reactor spent fuel. This means that once the fast reactor is turned on, the thermal reactor portion of the system will run MOX and IMF for the fuel that remains in the system, but the majority of the fuel in thermal reactors will be UOX.

Thermal reactors dominate the energy production through 2100 (see Figure 7.5-2). Consumer (T5.5 FR) and breeder fast reactors (T5.6 FR) come online very slowly because of the lack of plutonium.
For the case (T5.5) where consumer fast reactors are requested for 30% of the new reactor starting in 2025 and continue through 2100, the system becomes plutonium limited a little after 2090. This is not unexpected since the legacy spent fuel available earlier in the century would have provided “extra” plutonium compared to the current reactor fleet so that more consumer fast reactors could be built than

the fleet would normally support. Remember also that more plutonium is required to start a fast reactor than to continue its operation. The decrease in reactor capacity seen is from fast reactors built around 2030 retiring. A somewhat similar trend is seen when a switch is made to phasing out consumer fast reactors and continuing with once through (T5.2 FR), starting multi-pass MOX (T5.3 FR), or starting multi-pass IMF (T5.4 FR). In these cases, the plateau is due to discontinuing the building of fast reactors and the slope downwards is due to the retiring of fast reactors built around 2030. If a switch is made to 100% breeder fast reactors (T5.6 FR) in 2040, substantially more fast reactors enter the system. Early on, the difference between the breeder and convert fast reactor contributions is due mostly to the higher percent of breeder reactors requested. At the end of the century, the breeder fast reactor case (T5.6 FR) is plutonium limited, but not as badly as for the consumer fast reactor case (T5.5 FR), so a leveling rather than a decrease in fast reactors is seen. The plateau after 2090 is due to the rather weak breeding capacity of the recipe used for this case. Using a stronger breeder recipe would likely eliminate the plateau.

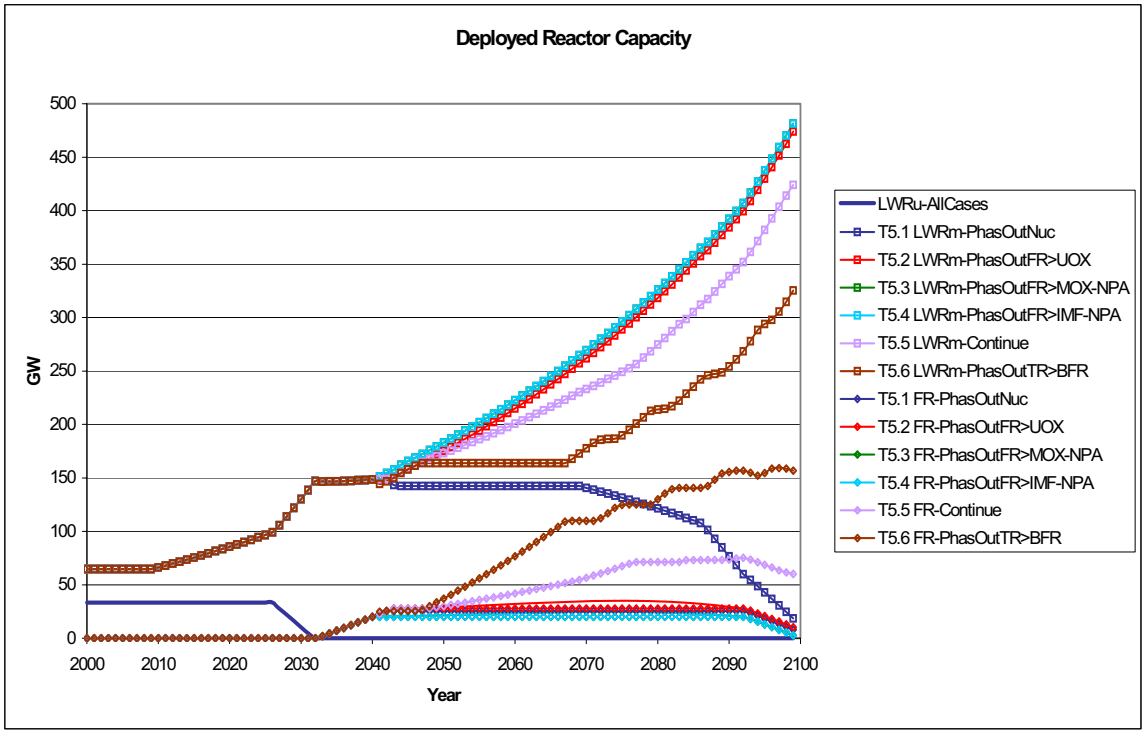

Figure 7.5-2. Power capacity of operating reactors by reactor type for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025). LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR).

All of the cases which continue with recycling, IMF, MOX, consumer fast reactor and breeder fast reactor (T5.3, T5.4, T5.5 and T5.6) will achieve a 99.5% reduction in TRU content in the repository by 2100. Looking at the graph (Figure 7.5-3) it is clear that shifting back from fast reactors to multi-pass MOX or IMF will still meet the repository reduction goal by 2010. The mass in the repository for phase out of nuclear (T5.1) would eventually equal the mass in the system for the phase out nuclear case (T5.1 in Figure 7.5-4). If consumer fast reactors are phased out at starting in 2040 and a once through thermal system is adopted at that time, then mass in the repository in 2100 will approach that of the continuous once through system shown by T1.2 in Figure 7.1-3.

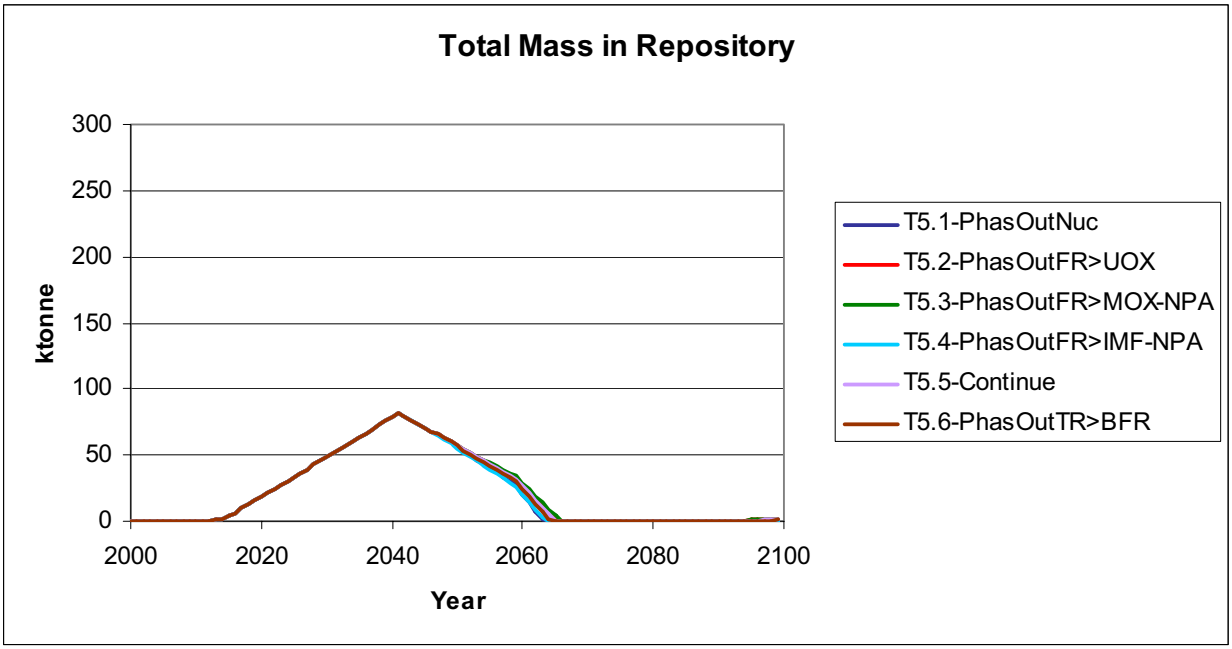

Figure 7.5-3. Total mass in repository for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025).

Figures 7.5-4 and 7.6-5 show the total and uranium mass (respectively) for each case. Uranium accounts for the majority of the mass in the system and the relative order of mass for uranium and total system are the same for all six cases. As expected, phasing out nuclear (T5.1) results in the lowest mass in the system while discontinuing consumer fast reactors and switching to once-through thermal reactors (T5.2) results in highest mass in the system. Phasing out consumer fast reactors and switching to MOX or IMF at 2040 reduces the total mass in the system compared to switching to once-through. This is probably due to the decrease in the uranium ore request (see Figure 7.5-8) that results from recycling fuel. Between MOX and IMF, IMF offers more reduction in total mass than MOX, this is more clearly seen in the plots of uranium in the system (see Figure 7.5-5) and plutonium in the system (see Figure 7.5-7). This is probably due to the relatively flat support ratio of IMF which allows it to move more quickly than MOX into the reactor fleet (see Figure 7.5-6). While the consumer reactor cases that continue with reprocessing (T5.3, T5.4, and T5.5) are close together, switching to a breeder fast reactor in 2040 significantly reduces the total mass and uranium in the system.

The mass of uranium in the system (Figure 7.5-5) for multi-pass MOX (T5.3) and multi-pass IMF (T5.4) is very close. The amount of fresh uranium does not decrease as much with MOX (T5.3) as for IMF (T5.4) because of the steep support ratio between successive cycles of MOX means that only a small portion of the reactor fleet contains MOX (see Figure 7.5-6). Whereas the recipe for multi-pass IMF which has close to a 1:1 support ratio which allows IMF to move into the system quickly.

Continuing with consumer fast reactors (T5.5) does not reduce the amount of uranium in the system any better than moving to MOX. When a switch is made to breeder fast reactors in 2040 (T5.6) the uranium in the system is further reduced from that achieved by switching to IMF.

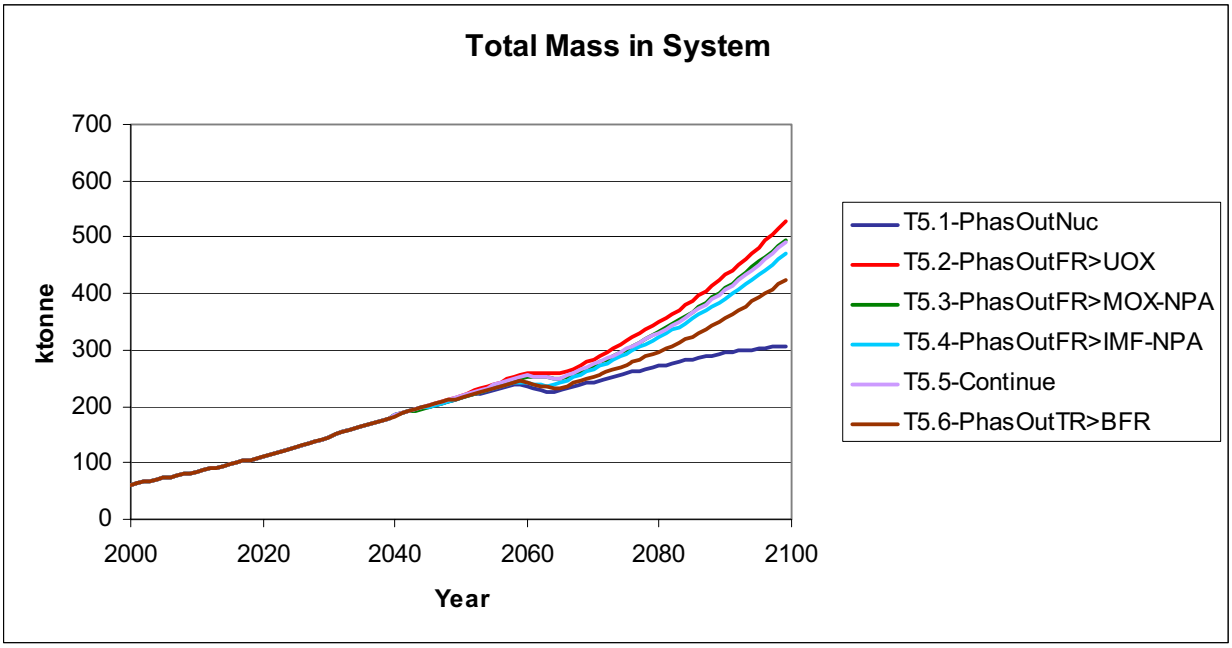

Figure 7.5-4. Total mass in the system for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025).

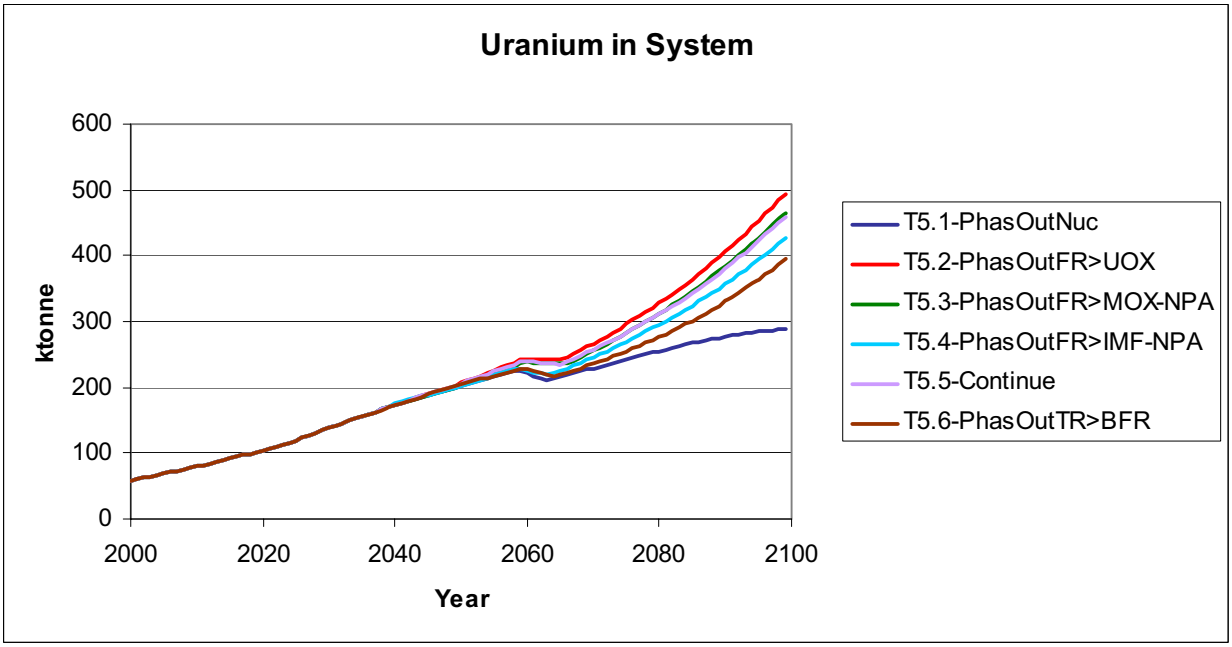

Figure 7.5-5. Mass of uranium in system for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025).

Starting multi-pass IMF in 2040 results in a substantial movement of IMF into the reactor fleet by 2080. This is due to legacy reactors retiring during this time and the reservoir of legacy spent fuel available for making IMF. As multi-pass IMF continues through the century, the system becomes IMF limited around 2085 (T5.4). This is due to legacy fuel no longer being available (it was consumed prior to 2080) to

provide extra IMF fuel for new reactors coming on-line, earlier in the century legacy spent fuel provided start-up fuel for new reactors that otherwise would have been started with UOX.

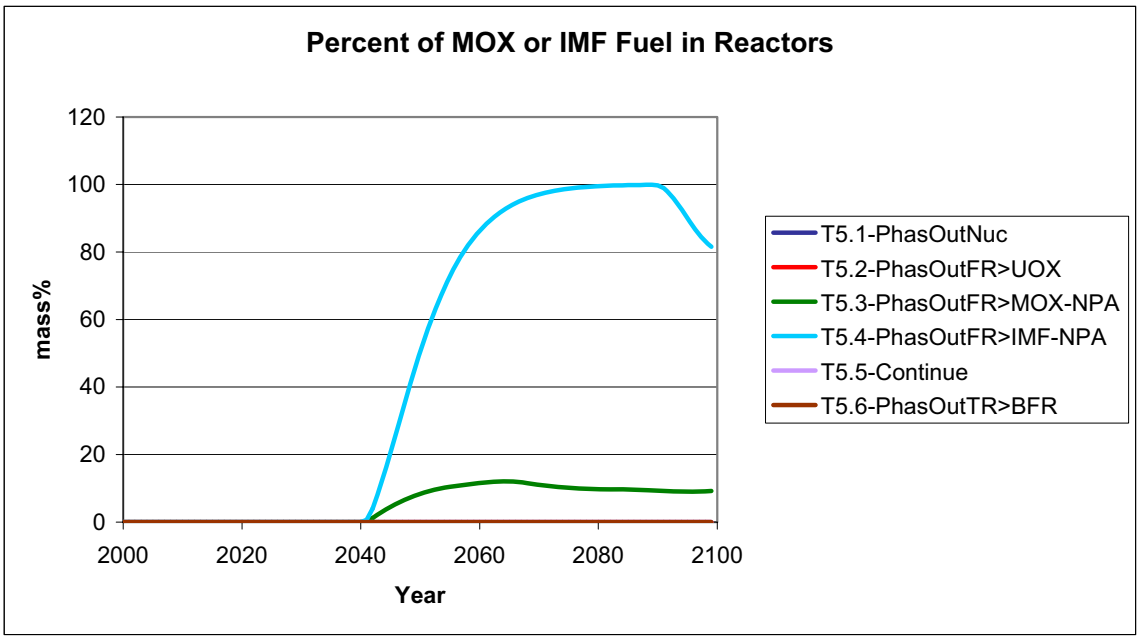

Figure 7.5-6. Mass percent of multi-pass MOX or multi-pass IMF fuel in reactors for each case of tree 5 (start once-through/consumer fast reactor symbiosis in 2025). The multi-pass IMF (T5.4) has a low support ratio and is blended core fuel. The multi-pass MOX (T5.3) has a higher support ratio and is a full core fuel.

IMF in this case (see T5.4 of Figure 7-5.7) reduces the amount of Pu in the system below that which would be in the system if nuclear power is phased out starting in 2040. IMF does a very good job of consuming Pu. As expected, shifting to breeder fast reactors (T5.6) increases the amount of plutonium in the system. Continuing with consumer fast reactors results in less plutonium in the system than for switching to once-through. This is not surprising since consumer fast reactors, IMF and MOX all reduce the amount of plutonium in the system. MOX is more effective than consumer fast reactors in consuming plutonium. The large difference in the amount of plutonium in the system between IMF and MOX is due both to IMF's better ability to consume plutonium and to the higher percentage of IMF in the system compared to MOX.

Looking at Figure 7.5-8, continuing with a consumer fast reactor – once through thermal reactor system (T5.5) uses the same uranium as moving toMOX thermal reactors (T5.3). As in previous trees, the once through thermal reactor case (T5.1) use the most uranium. Other than phasing out nuclear, switching to multi-pass IMF (T5.4) or breeder fast reactors in 2040 (T5.6) does the most to reduce uranium consumption.

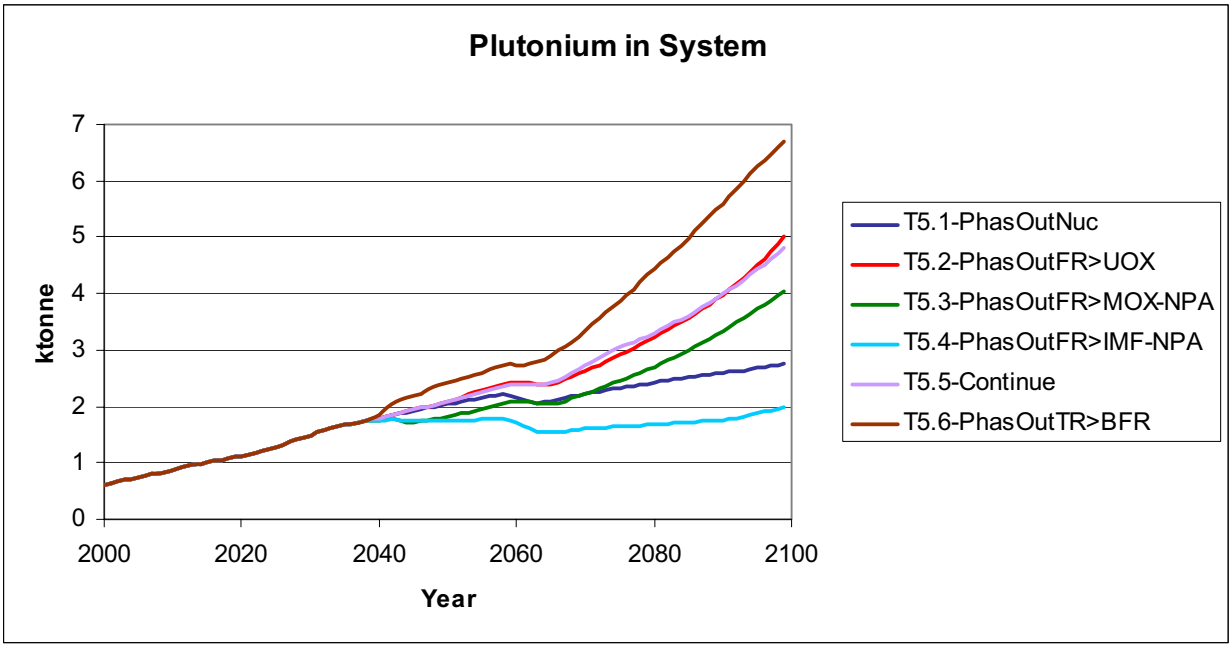

Figure 7.5-7. Mass of plutonium in system for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025).

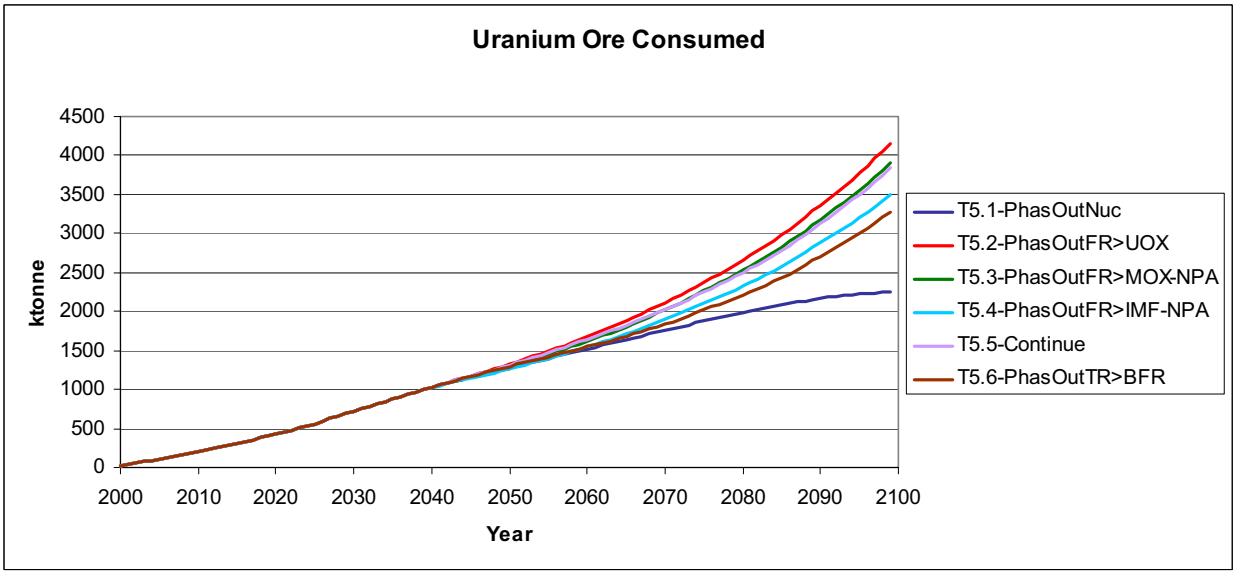

Figure 7.5-8. Mass of uranium ore consumed for each case in tree 5 (start once-through/consumer fast reactor symbiosis in 2025). Plots for T5.3 and T5.5 overlay each other.

7.6. Start Breeder fast reactors in 2025

This tree moves into FR, skipping recycling in TR. It aims to accommodate a hypothetical combination of limited uranium resources and high nuclear growth, without expending resources on recycling in TR. As with the other trees, the first branch in 2040 is phase out of nuclear. Branches 6.2-6.4 show phase out of the BFR, e.g. if performance is inadequate. (One could argue that this is what the French did when they terminated SuperPhenix.) In branch T6.2, the BFR is replaced with simply once-through. Thus, branch 6.2 is the same as once-through except for a few BFR built between 2025 and 2040. In branches 6.3 and 6.4, the BFR is replaced with thermal recycling via MOX and IMF respectively. Branch 6.5 continues the status quo and branch T6.6 accelerates it.

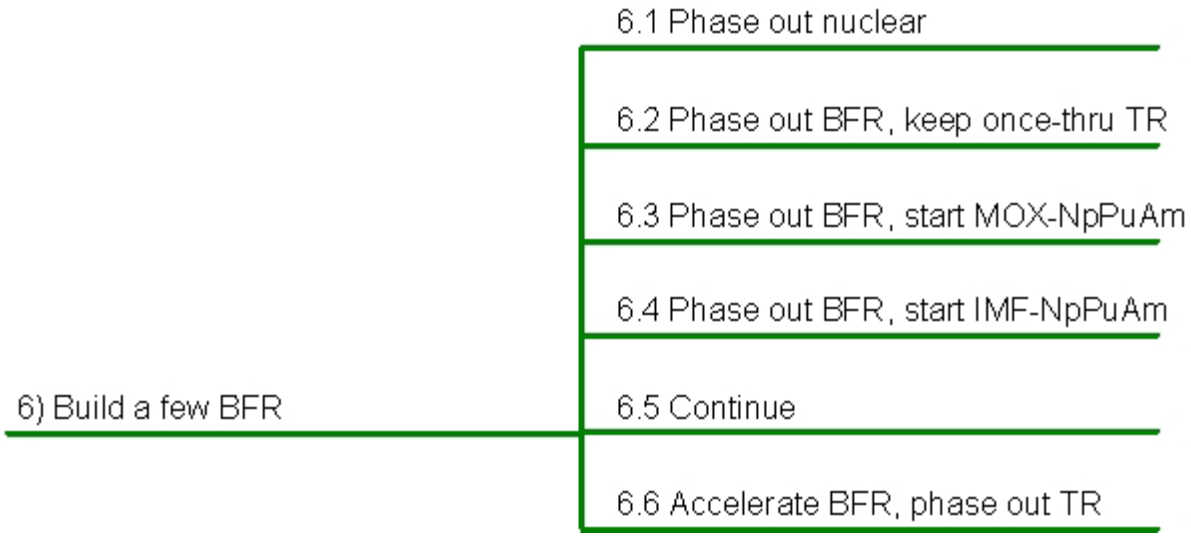

Figure 7.6-1. Development tree for starting breeder FR in 2025.

The rate that breeder fast reactors are built depends on the request rate (i.e. the percent of power generation capacity coming online to be provided by specific reactor types), the growth rate of the fleet, and the amount of plutonium available for starting the breeder reactors. For the cases presented here, the request rate is 30% of new construction from 2025 to 2040 and then either continuing with that request rate (case 6.5), reducing that request rate to zero (cases 6.2, 6.3, and 6.4), or increasing the request rate to 100% (case 6.6). As for all the other cases discussed in chapter 7, the growth rate is 1.8% per year starting in 2010. The amount of plutonium available for starting the breeder reactors depends on the recent history of plutonium generation and usage by the reactor fleet. When fast reactors are turned on, the fast reactors receive all UOX spent fuel, IMF or MOX pass 1 spent fuel, IMF pass 5 or MOX pass 8 spent fuel, and all fast reactor spent fuel. This means that once the fast reactor is turned on, the thermal reactor portion of the system will run MOX and IMF for the fuel that remains in the system, but the majority of the fuel in thermal reactors will be UOX.

Thermal reactors dominate the energy production through 2100 (see Figure 7.5-2). Consumer (T5.5 FR) and breeder fast reactors (T5.6 FR) come online very slowly because of the lack of plutonium. A plateau and decrease after 2090 is seen when a switch is made to phasing out nuclear (T6.1 FR), phasing out breeder fast reactors and continuing with once through (T6.2), starting multi-pass MOX (T6.3 FR), or starting multi-pass IMF (T6.4 FR). In these cases, the plateau is due to discontinuing the building of fast reactors and the slope is due to the retiring of fast reactors built around 2030. If 30% breeder fast reactors

continue to be requested past 2040 (T6.5 FR) or a switch is made to 100% breeder fast reactors (T6.6 FR) in 2040, substantially more fast reactors enter the system. Early on, the difference between the two breeder cases is due mostly to the percent of breeder reactors requested. At the end of the century, moving towards 100% breeder fast reactors results in periodic plutonium limited growth fairly quickly (by 2065), while continuing with 30% breeder reactors avoids plutonium limitations. This suggests that the optimum request rate for moving to breeder fast reactors (for these recipes) is a little above 30%. Using a stronger breeder recipe would reduce the amount of plutonium limitation.

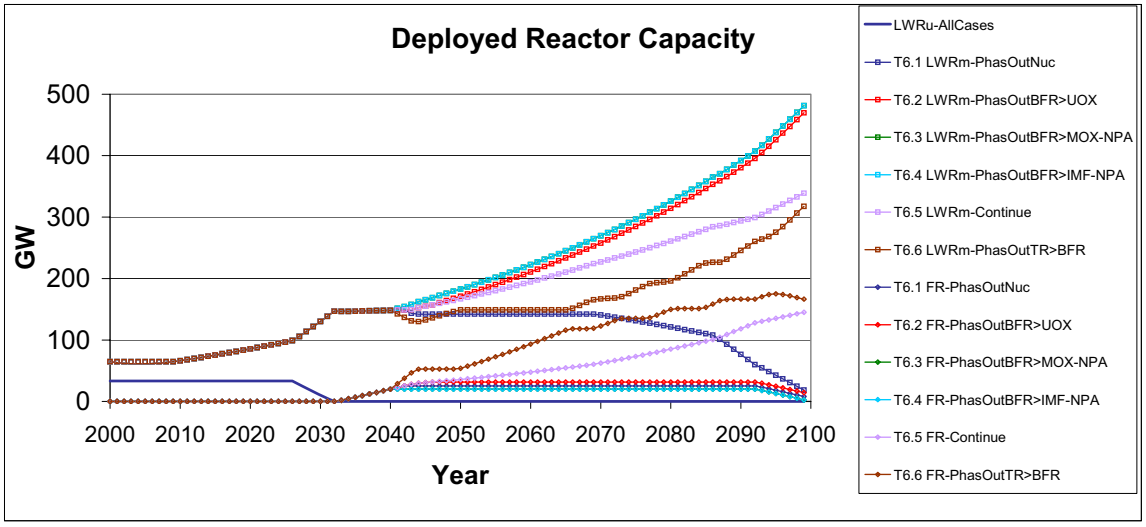

Figure 7.6-2. Power capacity of operating reactors by reactor type for each case of tree 6 (start breeder fast reactors in 2025). The lines for fast reactors for cases 6.1, 6.2, 6.3, and 6.4 overlay each other. LWRu is a thermal reactor that can only handle UOX. LWRm is a thermal reactor that can handle multiple fuels (e.g. UOX, MOX, IMF). FR is a fast reactor that may function as a consumer (CFR) or a breeder (BFR).

All of the cases that continue with recycling, IMF, MOX, and breeder fast reactor (T6.3, T6.4, T6.5 and T6.6) will achieve a 99.5% reduction in TRU content in the repository by 2100. Looking at the graph (Figure 7.6-3) it is clear that shifting back from fast reactors to multi-pass MOX or IMF will still meet the repository reduction goal in 2100. The mass in the repository would eventually equal the mass in the system for the phase out nuclear case (T6.1 in Figure 7.6-4). If breeder fast reactors are phased out at starting in 2040 and a once through thermal system is adopted at that time, then mass in the repository in 2100 will approach that of the continuous once through system shown by T1.2 in Figure 7.1-3.

Figures 7.6-4 and 7.6-5 show the total and uranium mass (respectively) for each case. Uranium accounts for the majority of the mass in the system and the relative order of mass for uranium and total system are the same for all six cases. As expected, phasing out nuclear (T6.1) results in the lowest mass in the system while discontinuing breeder fast reactors and switching to once-through thermal reactors (T6.2) results in highest mass in the system. Phasing out breeder fast reactors and switching to MOX or IMF at 2040 reduces the total mass in the system compared to switching to once-through. This is probably due to the decrease in the uranium ore request (see Figure 7.6-8) that results from recycling fuel. Between MOX and IMF, IMF offers more reduction in total mass than MOX, this is more clearly seen in the plots of uranium in the system (see Figure 7.6-5) and plutonium in the system (see Figure 7.6-7). This is probably due to the relatively flat support ratio of IMF that allows it to move more quickly than MOX into the reactor fleet (see Figure 7.6-6). While the cases that continue with reprocessing (T6.3, T6.4, T6.5, T6.6) are close together, switching to 100% breeder fast reactor in 2040 significantly reduces the total mass and uranium mass in the system.

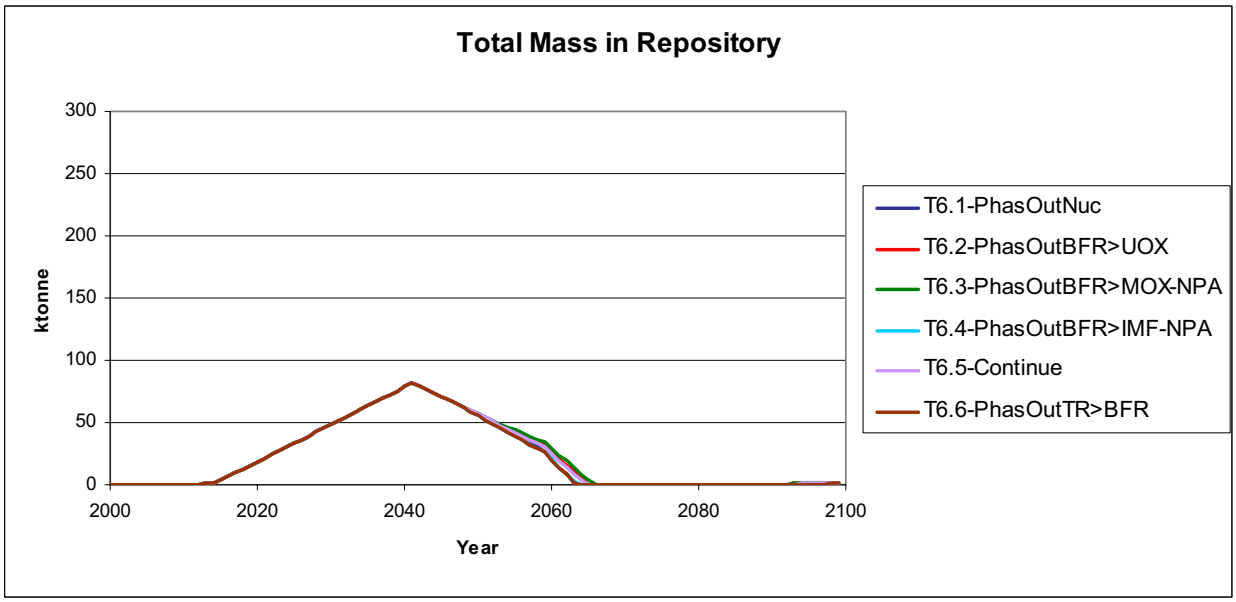

Figure 7.6-3. Total mass in repository for each case of tree 6 (start breeder fast reactors in 2025). Lines for cases T6.1, T6.2, T6.5, and T6.6 overlay each other.

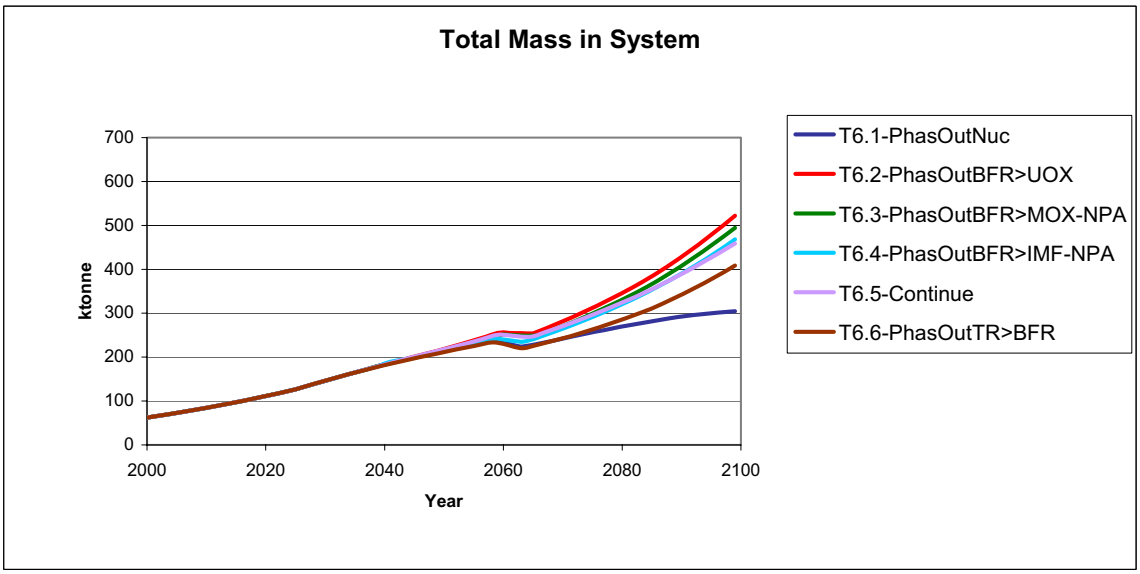

Figure 7.6-4. Total mass in system for each case of tree 6 (start breeder fast reactors in 2025).

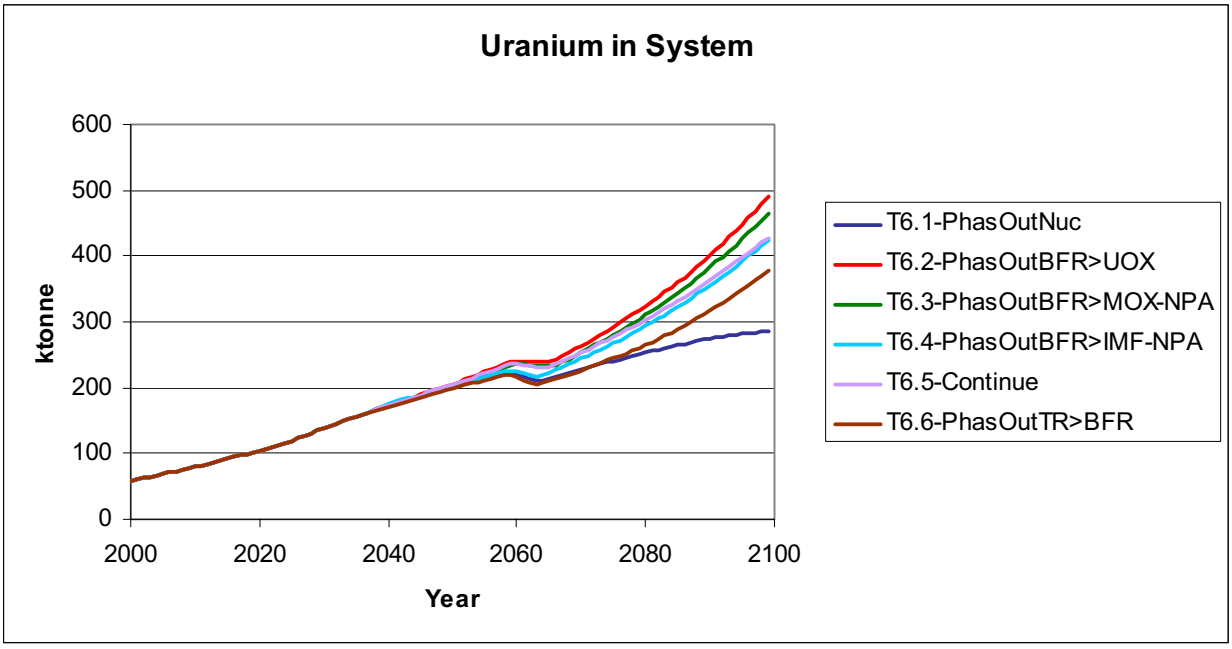

Figure 7.6-5. Mass of uranium in system for each case of tree 6 (start breeder fast reactors in 2025).

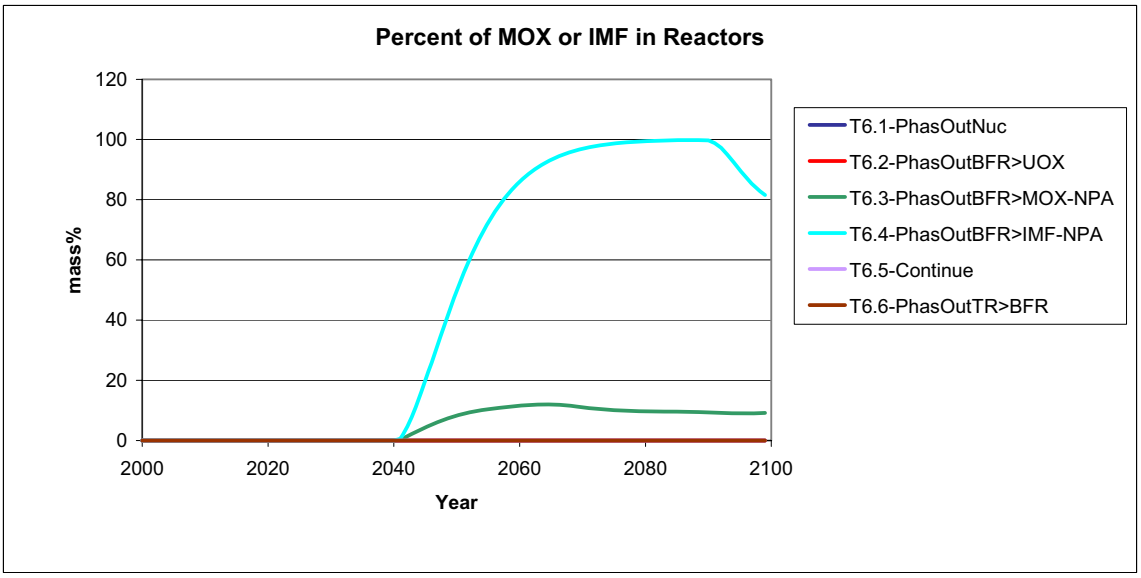

Figure 7.6-6. Mass percent of MOX or IMF fuel in reactors for each case of tree 6 (start breeder fast reactors in 2025).

IMF in this case (see T6.4 of Figure 7-6.7) reduces the amount of Pu in the system to a level comparable to the amount of Pu that would be in the system if we shut down nuclear power starting in 2040. IMF does a very good job of consuming Pu.

Switching to 100% breeder reactors (T6.6) for new construction in 2040 results in the highest mass of plutonium in the system at 2100. Phasing out breeder fast reactors and returning to once-through (T6.2) results in significantly more plutonium in the system than moving to multi-pass MOX (T6.3) or multi-

pass IMF (T6.4). Switching to IMF (T6.4)is the only case that reduces the level of plutonium in the system at 2100 to below that of phasing out nuclear (T6.1).

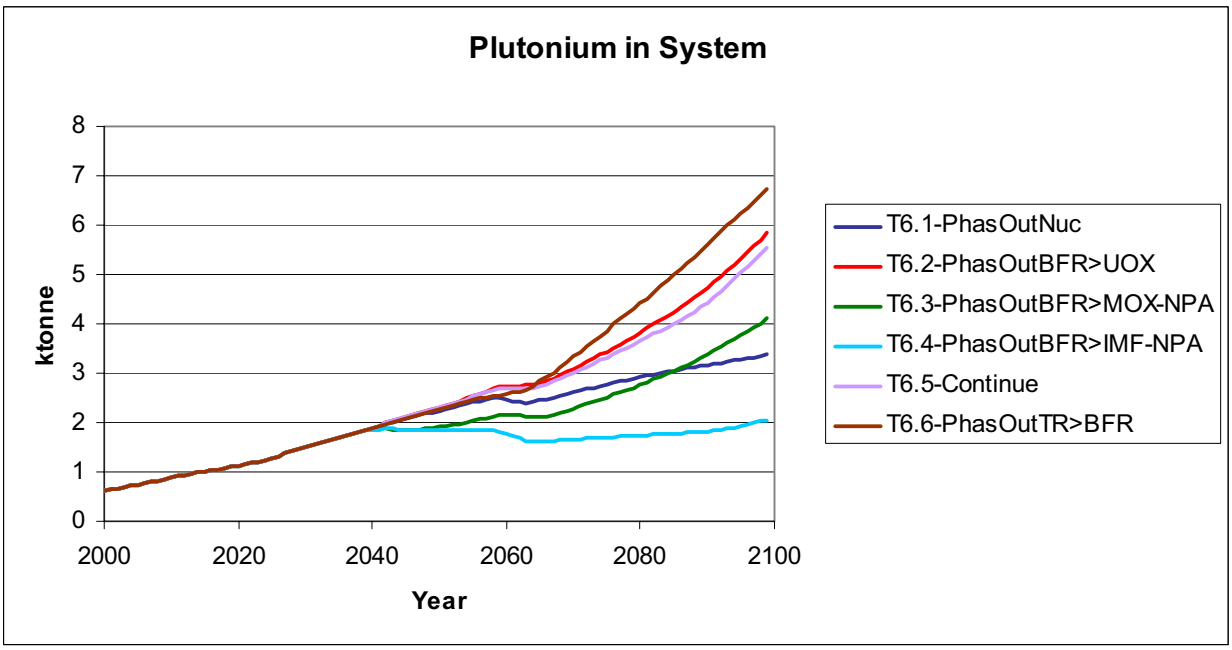

Figure 7.6-7. Mass of plutonium in system for each case of tree 6 (start breeder fast reactors in 2025).

Phasing out breeder fast reactors and returning to once-through UOX (T6.2, Figure 7.6-8) results in the highest uranium ore consumption by 2100. Switching to multi-pass MOX (T6.3) consumes only slightly less uranium than the once-through case; this is probably due to the low percent of MOX fuel (approximately 10%) in thermal reactors by 2100. Switching to multi-pass IMF (T6.4) consumes less uranium than the MOX case; this is probably due to the relatively high percent of IMF fuel (approximately 80%) in thermal reactors by 2100. Recycling fuel reduces uranium consumption compared to once-through. Switching to 100% of new reactors being breeder fast reactors (T6.6) reduces uranium consumption about twice as much as continuing with 30% of new reactors being breeder fast reactors (T6.5).

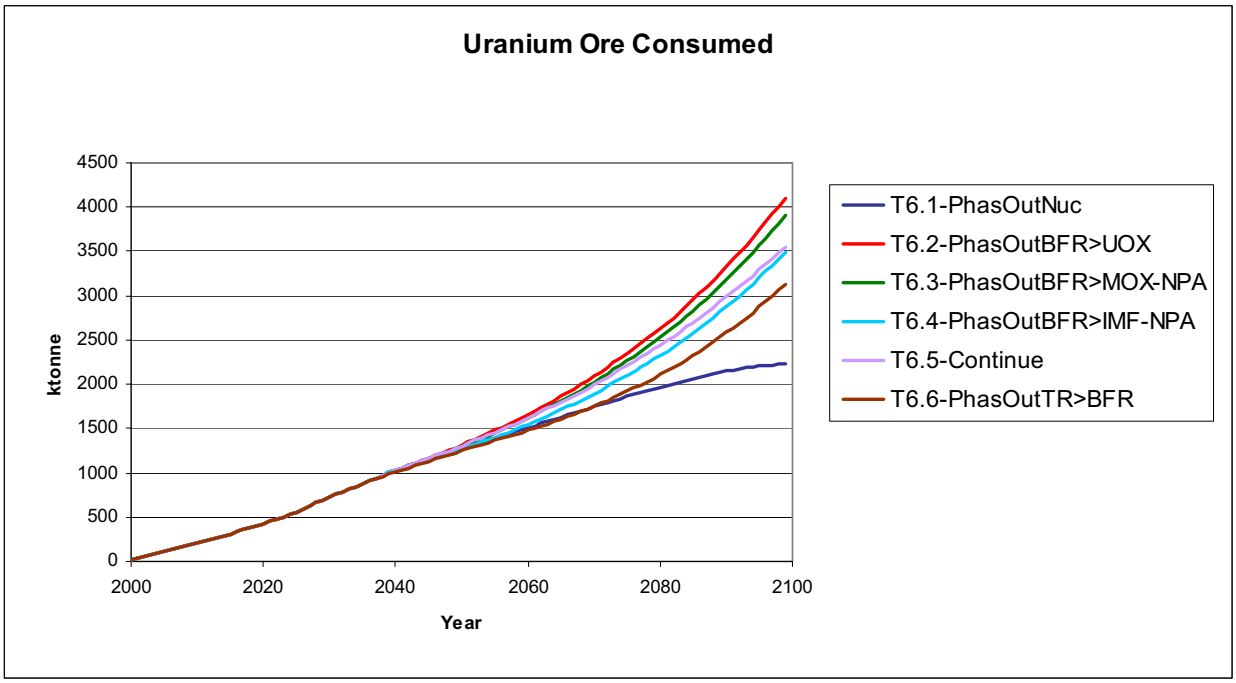

Figure 7.6-8. Uranium ore consumed for each case in tree 6 (start breeder fast reactors in 2025).

8. DYMOND ANALYSES OF SPECIFIC ISSUES

Previous chapters contained key fuel cycle decisions (Chapter 2), AFCI objectives and metrics (Chapter 3), the alternatives (Chapter 4), and "static" analyses of the various fuel and reactor options and how they can be combined (Chapter 5). This Chapter is the third of three that addresses timing and dynamics. Chapter 6 examined development trees using relatively simple analysis of more possibilities. Chapter 7 examined the same trees using DYMOND. This Chapter uses the results from Chapter 7 and other DYMOND calculations as needed to examine particular issues, including the following:

1. Comparison of single-pass options (uses trees 2, 3, 4, and variations thereof)
2. Comparison of multi-pass options (uses tree 2 and 4)
3. Comparison of single-pass versus multi-pass.
4. Transition to fast recycling (uses trees 5 and 6)

8.1. Comparison of single-pass options

This subsection shows DYMOND calculations for the following single-pass cases:

IMF-NpPu (full core)
IMF-NpPuAm (blended)
IMF-NpPuAmCm (full core)
MOX-NpPu (full core)
MOX-NpPuAm (full core)
UOX once through (for comparison)

None of these cases meet the AFCI objectives.

The following assumptions/settings were used for the calculations presented in this section:

- 1.8% annual growth rate in power demand starting in 2010
- repository opens in 2012
- reprocessing starts in 2025 with an annual reprocessing capacity for spent fuel from thermal reactors of 3 kton/yr in 2025, 6 kton/yr in 2040, 9 kton/yr in 2060, and 12 kton/yr in 2080
- all cases start with once through UOX in thermal reactors, all cases except OTC change to a one-pass MOX or IMF formulation in 2025
- the amount of MOX or IMF fuel that is available is calculated based on the amount of Pu available

Figure 8.1-1 shows the percentage of MOX or IMF fuel in the reactors (calculated as ktonne of MOX or IMF fuel / total ktonne of fuel in the reactors *100) as a function of time. As expected, the plot for OTC remains zero (no recycled fuel). Two of the IMF cases, IMF-NpPu and IMF-NpPuAmCm, and the two MOX cases, MOX-NpPu and MOX-NpPuAm, account for less than 15% of the total fuel in the reactors. The IMF-NpPuAm case shows a stiking different trend and accounts for up to 95% of the fuel in the reactors. This is due to the IMF-NpPuAm being a blended core recipe rather than a full core recipe. Full core recipes imply that all the pins in a given reactor are made of MOX or IMF; in the current model formation of MOX or IMF is based on the availability of Pu. If there is not enough Pu to make enough MOX or IMF for a full core, enough MOX or IMF is made for the available Pu and the remainder of the core is made of UOX. The total amount of MOX or IMF fuel in the reactors is then some fraction of the fuel capacity of the reactors. Blended core recipes assume that only part of the pins in a given core are IMF or MOX and the remaining pins are UOX. As in the full core cases, the current model formation of MOX or IMF is based on the availability of Pu. If there is not enough Pu to make enough MOX or IMF pins for a blended core, enough MOX or IMF is made for the available Pu and the remainder of the core is made of UOX. Since blended IMF or MOX fuel includes both tranuranic and uranium oxide pins, the amount of transuranic required to make a complete core of blended fuel is lower than that to make a

complete core of full core IMF or MOX. This means that for a given amount of available transuranics, blended core IMF or MOX fuel may account for a higher percentage of the total fuel in reactors than full core IMF or MOX fuel.

For these calculations, one-pass MOX or IMF fuel was used. For these calculations, one-pass means that fresh UOX fuel entering the reactor exits the reactor as spent UOX fuel and then goes to reprocessing to form fresh MOX or IMF fuel. Spent MOX or IMF fuel exiting the reactor goes to the repository; it is not reprocessed, hence the name one-pass, the MOX or IMF fuel makes one pass through the reactor.

A peak in the percent of reactor fuel provided by MOX or IMF fuels is seen for all recipes around 2060. This results from a combination of increasing reprocessing capacity and the availability of legacy fuel. Once all the legacy fuel is reprocessed in the 2060-2070 time period, the relative amount of transurantics decreases so percentage of MOX or IMF fuel decreases. The steeper increase observed for IMF-NPA (blended core) is due to the lower amount of tranuranic required to make a complete core of blended fuel and the extra transuranic available from the legacy fuel.

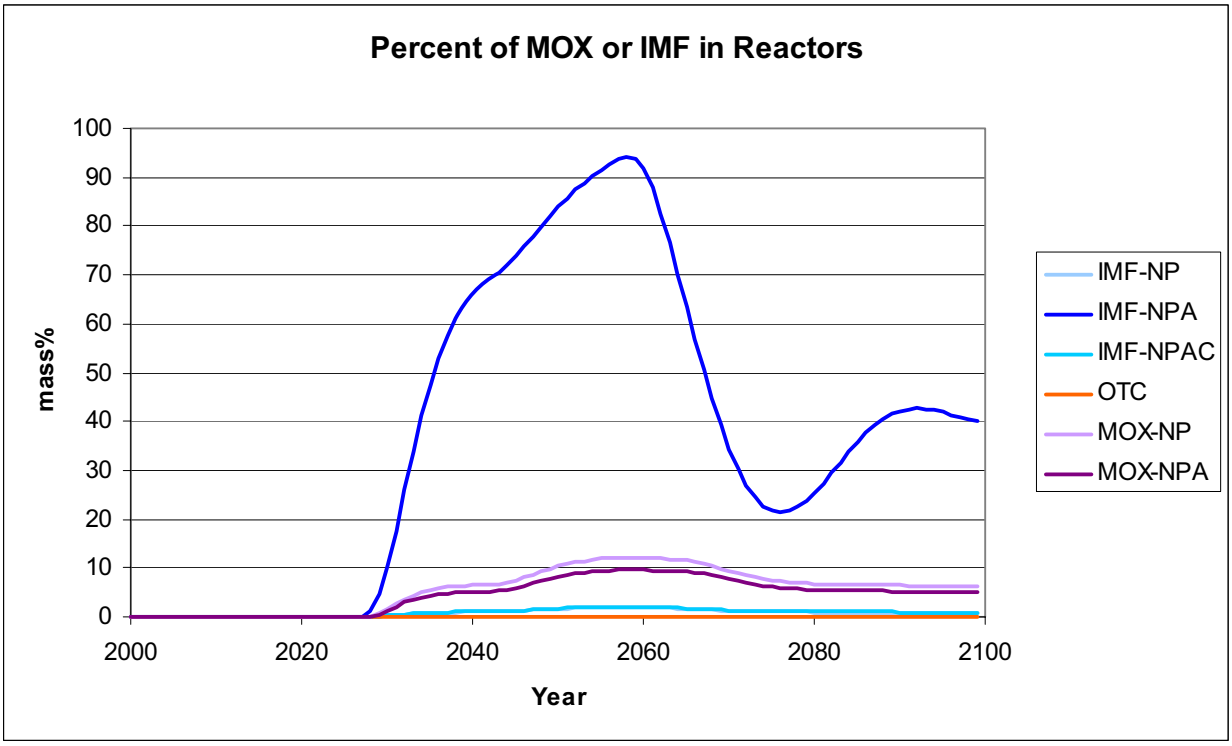

Figure 8.1-1. Percent of total fuel in reactors provided by MOX or IMF fuel recipes. The plots for IMF-NP and IMF-NPAC overlay each other.

The percent of the total mass in reactors that is Pu is presented in Figure 8.1-2. The results presented in the figure support the previous discussion. While the percent of reactor fuel provided by blended versus full core MOX or IMF fuel recipes is very different, the percent of Pu in the reactor cores is quite similar for all of the MOX or IMF recipes.

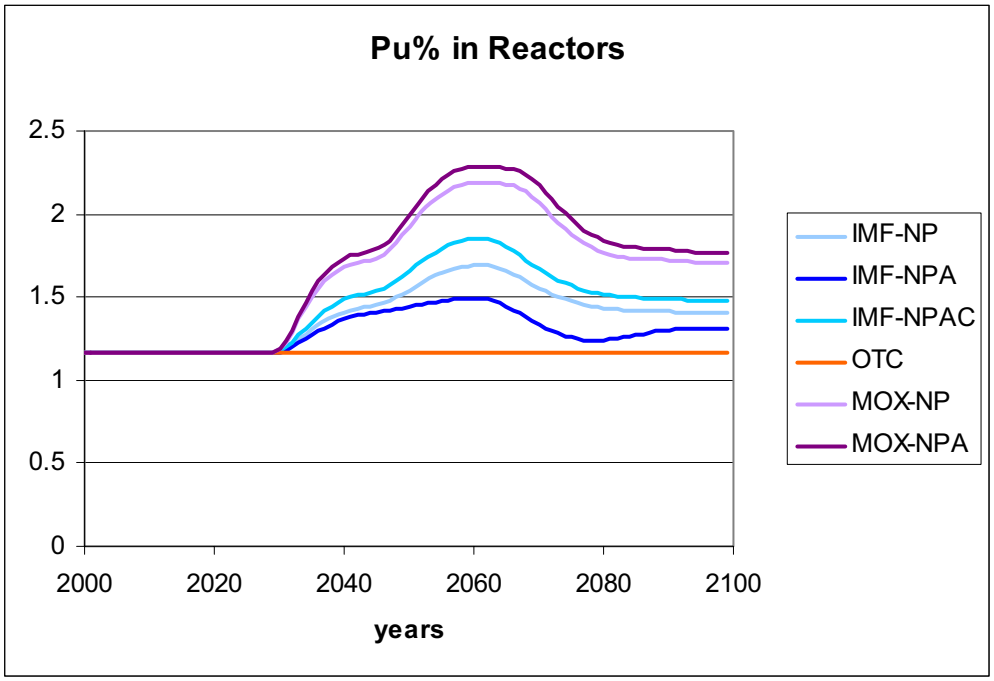

Figure 8.1-2. Percent of total mass in reactors that is Pu.

The total mass of spent fuel in the repository is shown in Figure 8.1-3. Continuing with once through results in the greatest amount of mass in the repository by 2100. Moving at 2025 to a one pass MOX or IMF full core fuel reduces the amount of spent fuel in the repository relative to once through. The large difference between the IMF-NPA blended fuel and the other MOX and IMF full core fuels reflects both the amount of each fuel type in the reactors and the way the fuel is handled in the current model. The IMF-NPA blended fuel accounts for a greater percentage of the fuel in reactors than the full core fuels do. Since any type of spent IMF or MOX fuel is sent to the repository, it is expected that more IMF-NPA fuel ends up in the repository since there is more in the reactors. However, from figure 8.1-2, it is clear that the percentage of transuranics in the reactors for all of the fuel types is fairly close. The large difference between the mass for the IMF-NPA and the other fuel types is how uranium is handled. In the current model, the fresh uranium used to make the IMF-NPA is counted as part of the mass of IMF or MOX fuel; the core is supposed to be heterogenous with respect to UOX and transuranics. When the IMF-NPA blended core spent fuel exits the reactor, all the fuel, including the spent UOX that is part of the normal fuel recipe goes to the repository. The core is not disassembled with the spent UOX going to reprocessing and the spent transuranics going to the repository. In the current model, when there is not enough full core MOX or IMF to complete a core, UOX is added. The added UOX is counted as part of the mass of UOX, not MOX or IMF. When the MOX or IMF full core spent fuel exits the reactor, only it goes to the repository. The UOX added to complete the load goes to reprocessing; this is the equivalent of disassembling the core. If the spent UOX for both types of recipes were treated the same way, then all five recipes would provide similar results. This difference in how spent cores and the spent UOX are treated siginificantly impacts the compositon of the material in the repository and the rate at which mass is placed in the repository.

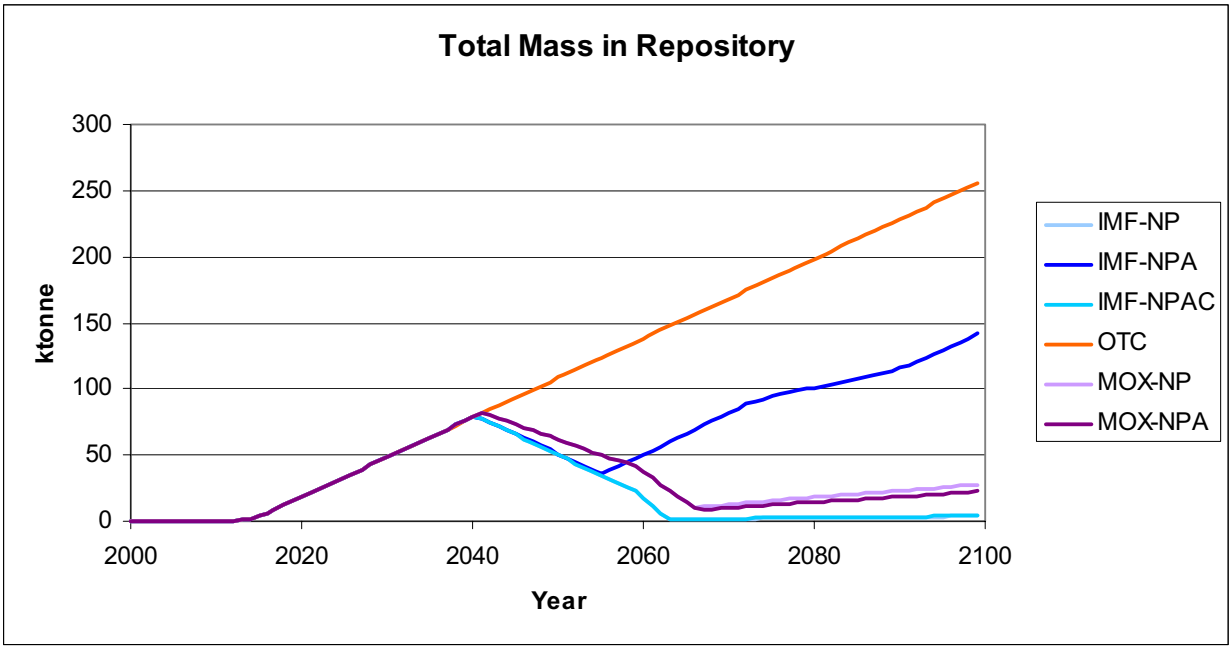

Figure 8.1-3. Total mass in repository. The plot for IMF-NP is overlain by the plot for IMF-NPAC. The IMF-NPA recipe is a blended core fuel while the other MOX and IMF recipes are for full core fuels.

Continuing with once-through UOX (Figure 8.1-4) results in the highest mass in the system. Compared to once-through, switching to one-pass MOX in 2025 reduces the total mass in the system by approximately 10%, while switching to one-pass IMF in 2025 reduces the total mass in the system by approximately 20%. No significant differences were observed amoung the different MOX and IMF recipes.

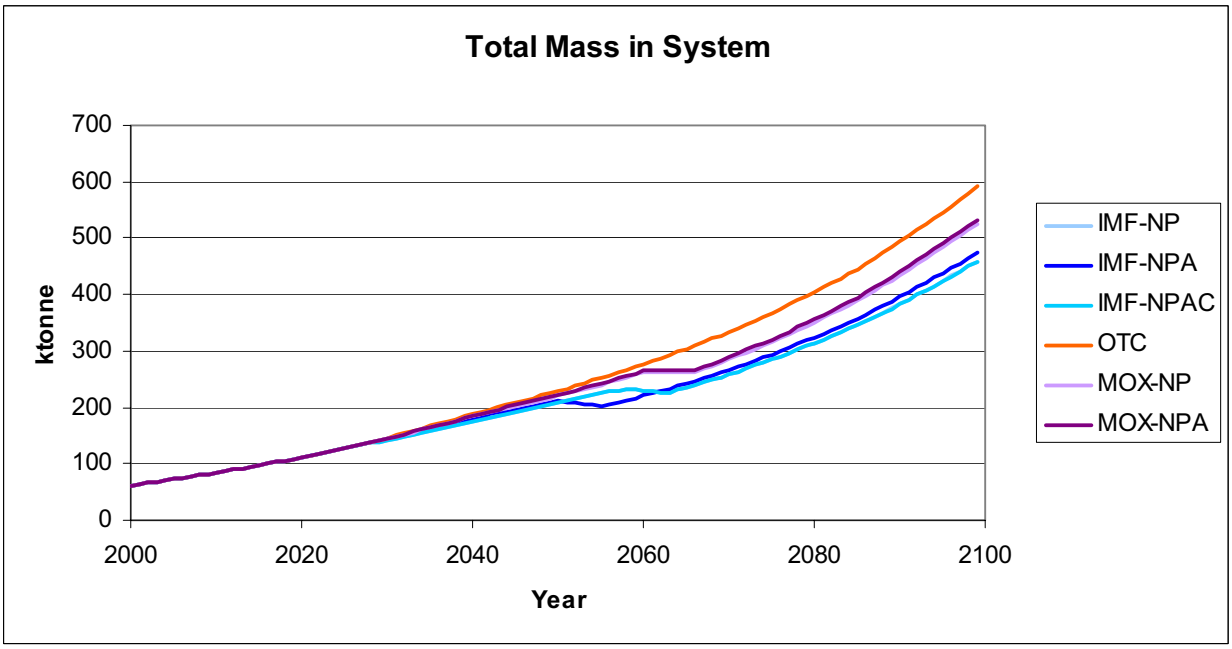

Figure 8.1-4. Total mass in the system. The plot for IMF-NP is overlain by the plot for IMF-NPAC. The IMF-NPA recipe is a blended core fuel while the other MOX and IMF recipes are for full core fuels.

The masses of plutonium (Figure 8.1-5) and uranium (Figure 8.1-6) in the system track with the total mass in the system. Continuing with once-through UOX results in the highest plutonium and uranium masses in the system. Compared to once-through, switching to one-pass MOX in 2025 reduces the plutonium in the system 19-24%, depending on the recipe. Compared to once-through, switching to one-pass IMF in 2025 reduces the plutonium in the system 48 – 62%, depending on the recipe.

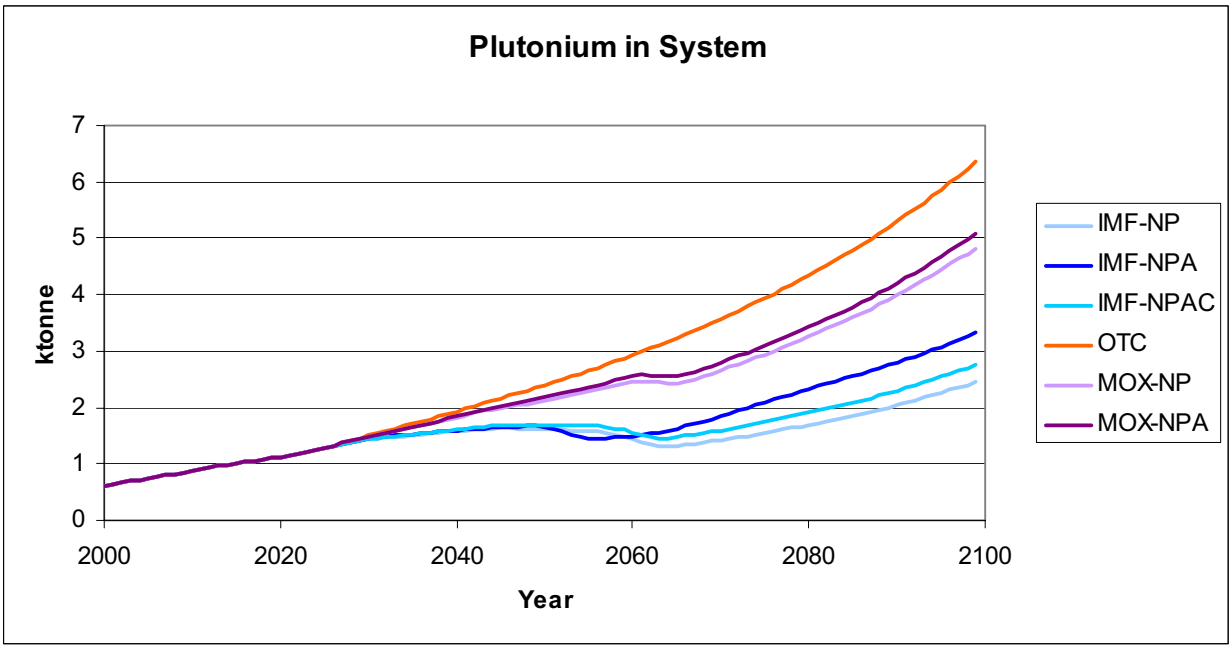

Figure 8.1-5. Mass of plutonium in system

Uranium is the largest single element in the mass of the system (see Figure 8.1-6) and "controls" the total mass in the system.

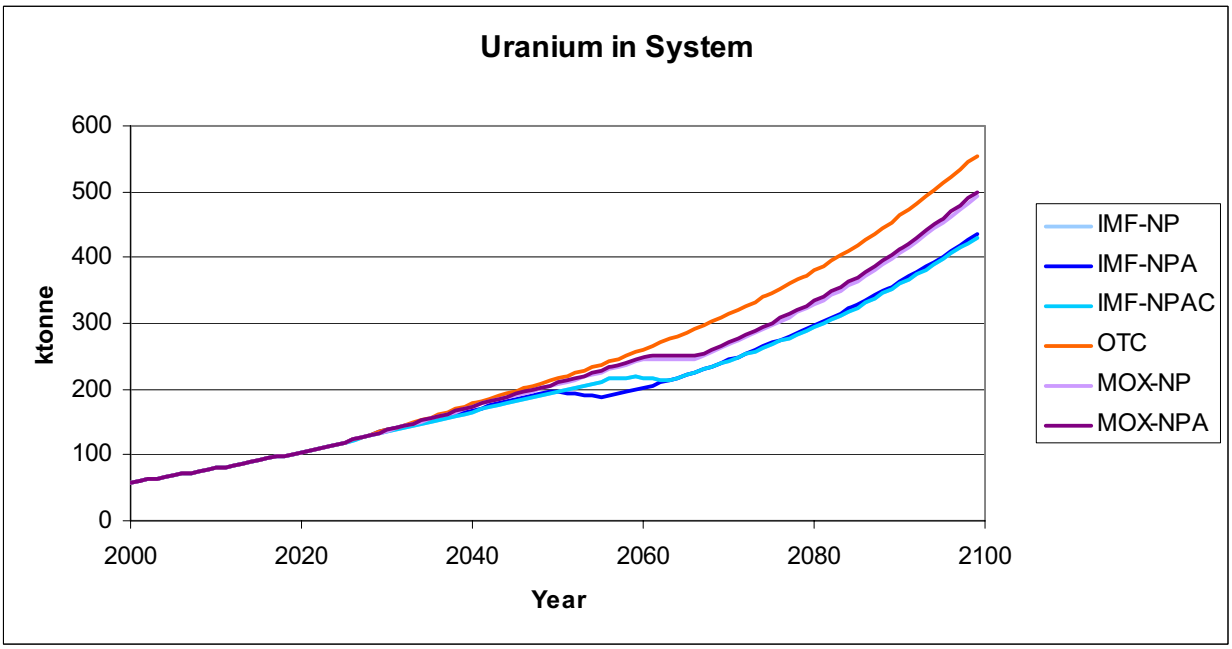

Figure 8.1-6. Mass of uranium in system.

The mass of plutonium-239 in the system (Figure 8.1-7) tracks closely with that of the total plutonium in the system (Figure 8.1-). This is expected since plutonium-239 is the dominant isotope of plutonium in the fuels. The mass fraction of plutonium-239 in fuel decreases between the inlet and outlet recipes for all of the MOX and IMF recipes presented; these recipes are net consumers of plutonium-239. As expected the mass fraction of plutonium-239 increases in the outlet recipes for once through UOX, making it a net producer of plutonium-239.

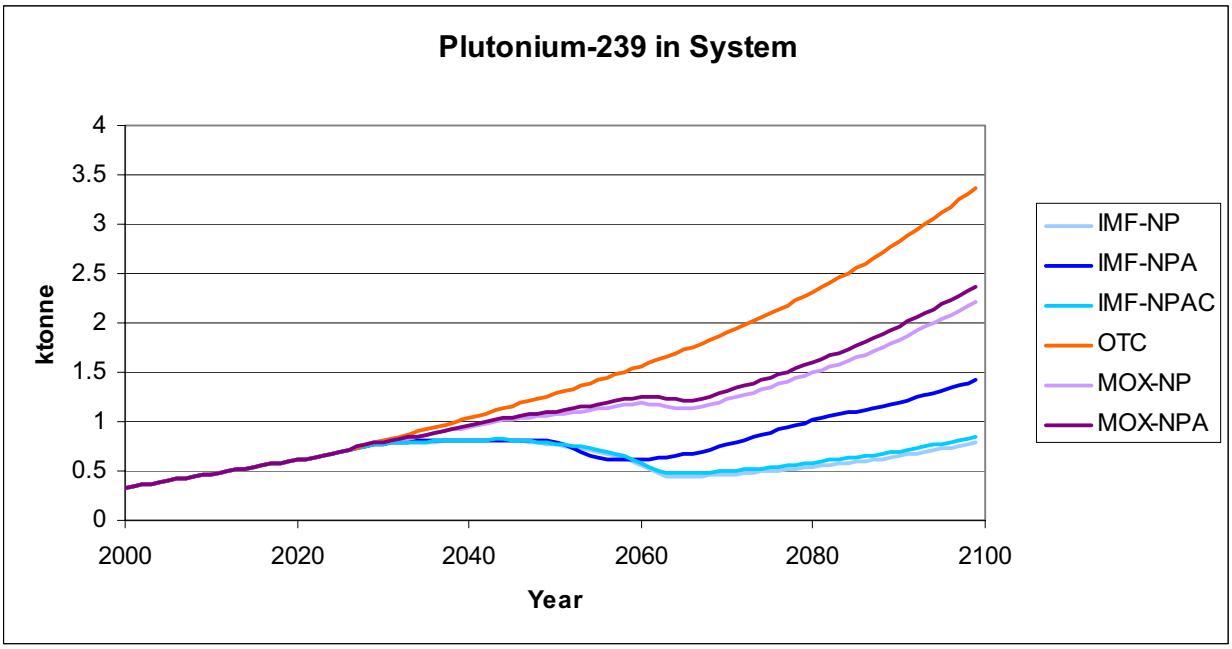

Figure 8.1-7. Mass of plutonium-239 in system.

The mass of plutonium-238 (Figure8.1-8) does not track with the mass of total plutonium in the system. Plutonium-238 accounts for a small fraction of the total plutonium in the fuels. In contrast to plutonium-239, the mass fraction of plutonium-238 in fuel increases between the inlet and outlet recipes for all of the MOX and IMF recipes presented; these recipes are net producers of plutonium-239. While for once through UOX, the trend is the same for plutonium-239, the mass fraction of plutonium-238 increases in the outlet recipes for once through UOX, making it a net producer of plutonium-238. The observed difference in order is due to the MOX and IMF fuels being net producers of plutonium-238 in contrast to them being net consumers of plutonium-239.

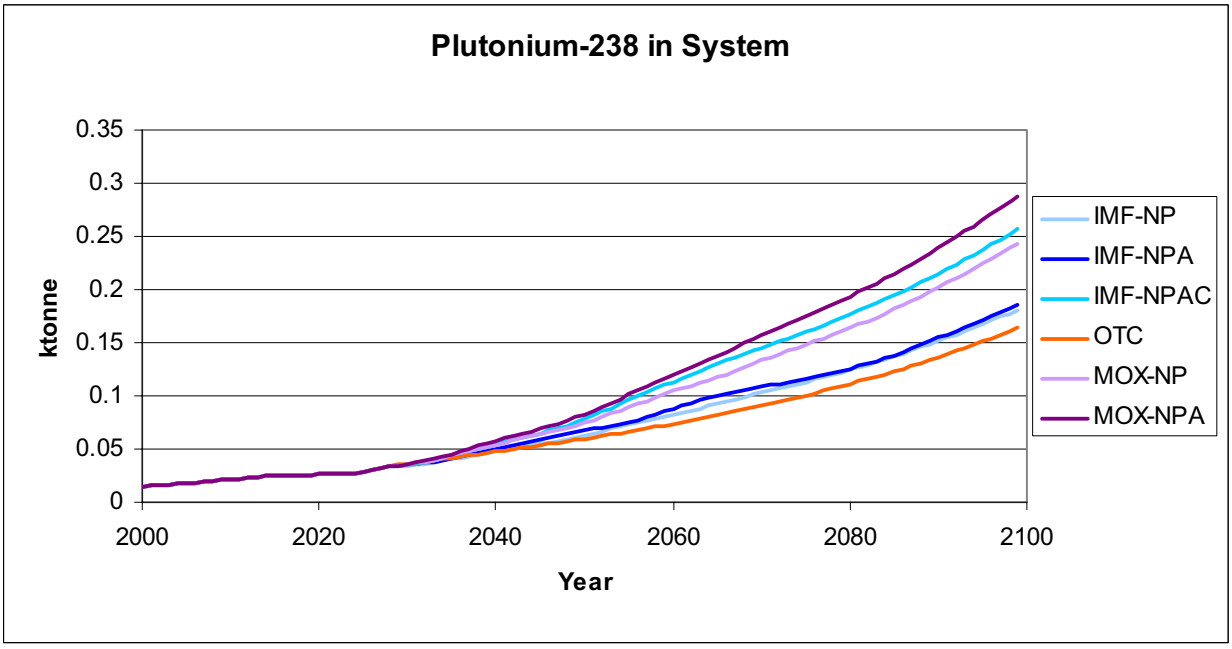

Figure 8.1-8. Mass of plutonium-238 in system.

The amount of weapons usable material in the system (see Figure 8.1-9) can be correlated to the potential for having sufficient material to create a critical mass. The current model uses the critical mass of plutonium-239 as a reference critical mass. The masses of all remaining isotopes, that are capable of forming a critical mass, are ratioed to mass required to form a critical mass of plutonium-239 and an equivalent mass of plutonium-239 is calculated. The equivalent masses are summed to estimate the mass of weapons usable material in the system. The relative magnitude of weapons usable material in the system is the same as the relative magnitude of plutonium-239 in the system, suggesting that for these fuels and operating modes, plutonium-239 is the major component of weapons usable material in the system. Switching to one-pass MOX in 2025 decreases the weapons usable material in the system at 2100 by 20 – 23%, depending on the recipe. Switching to one-pass IMF in 2025 decreases the weapons usable material in the system at 2100 by 45 – 57% depending on the recipe. For each of the recipes used, at any point in time, less than half of the total weapons usable material is in reprocessing, dry interim storage, or fuel fabrication (Figure 8.1-10).

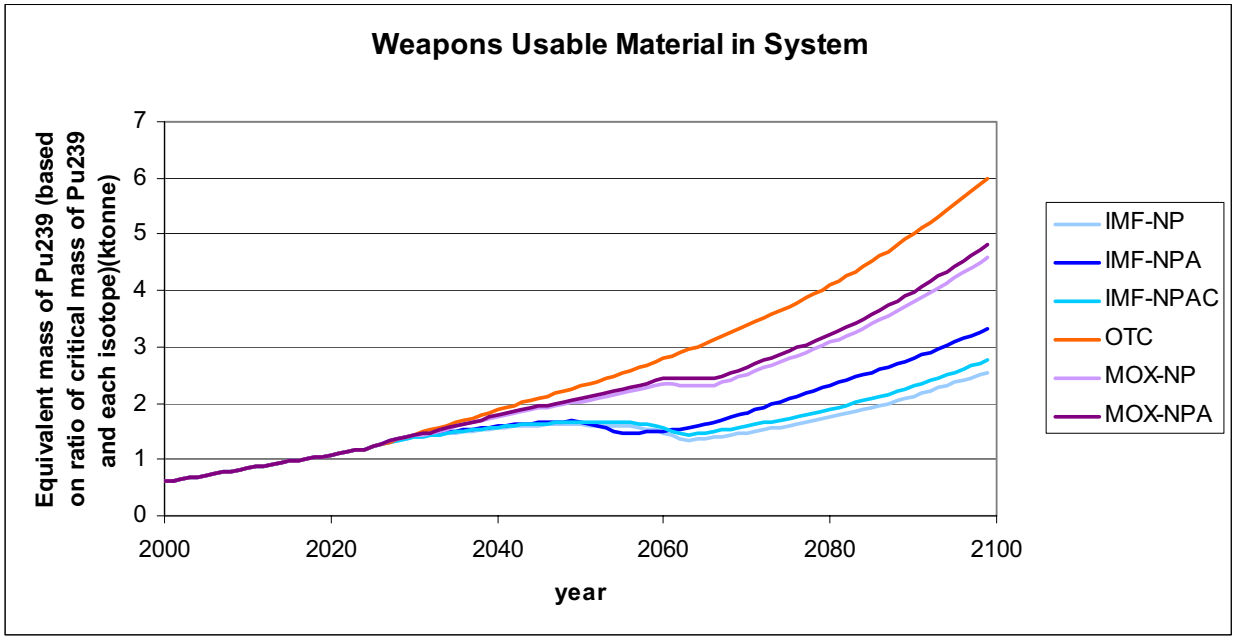

Figure 8.1-9. Weapons usable material in system.

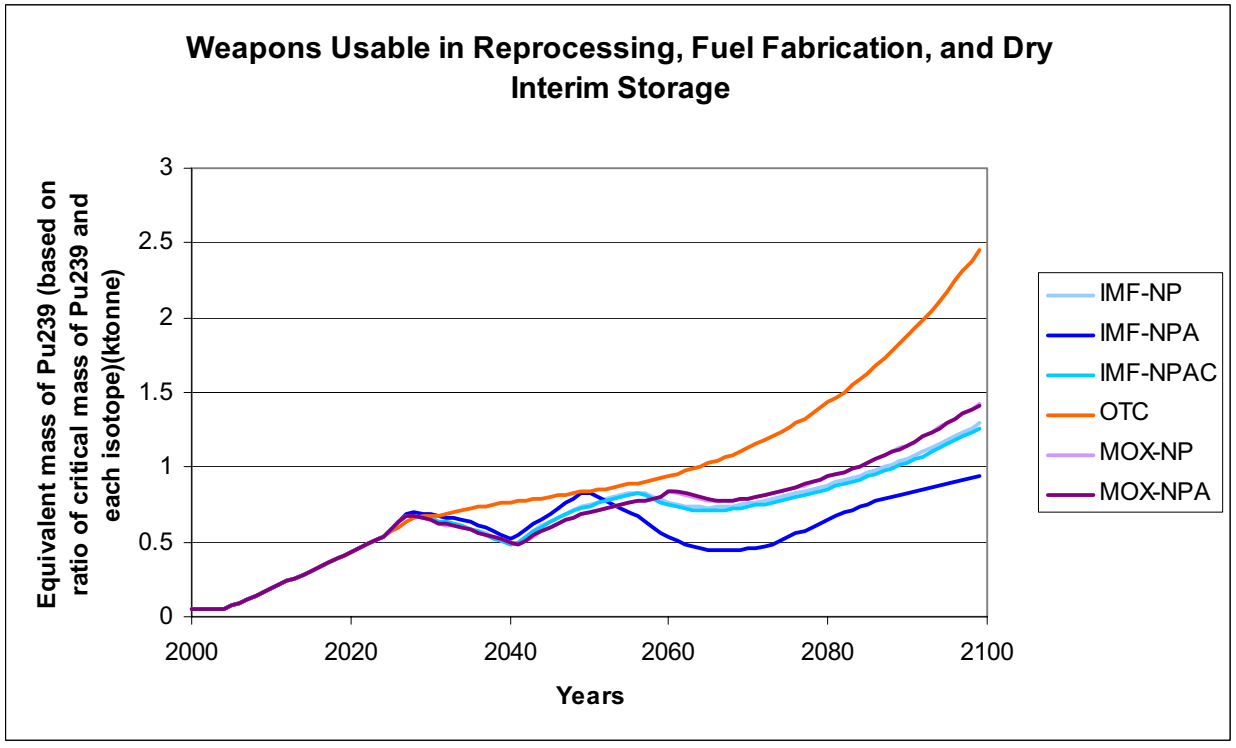

Figure 8.1-10. Weapons usable material in reprocessing, fuel fabrication, and dry interim storage

The long-term heat (50-1500 years) from the material placed in the repository is presented in Figure 8.1-11. The blended IMF-NPA fuel results in a significantly higher heat load than the other recipes. Looking

at the details of the model, most of the difference is due to the heat load contributions of Cm-245, Am-243, and other actinides.

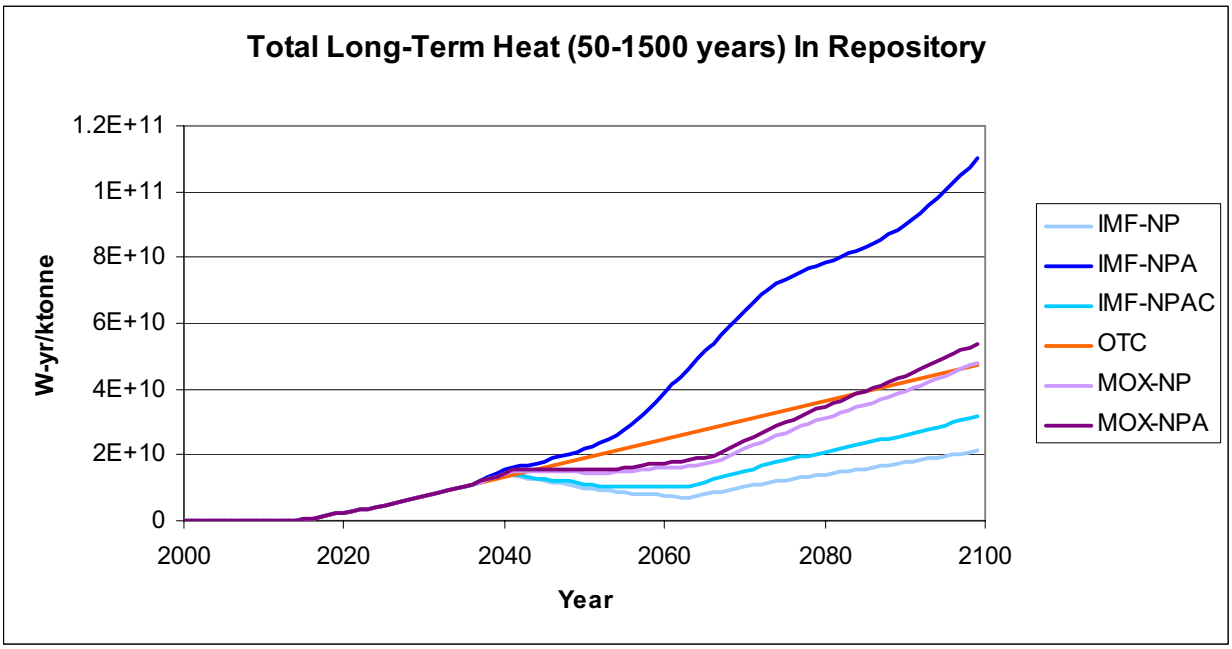

Figure 8.1-11. Total long-term heat (50-1500 years) in repository.

The dose at 500,000 years from material in the repository (see Figure 8.1-12) tracks with the total mass in the repository (see Figure 8.1-3).

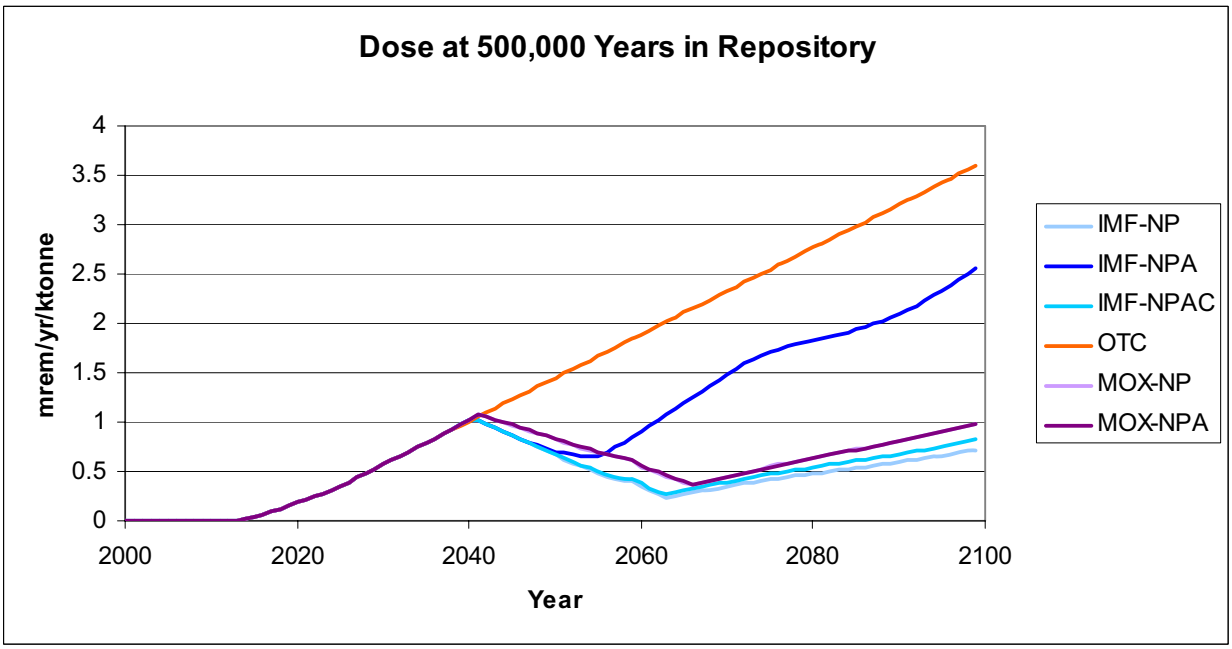

Figure 8.1-12. Dose at 500,000 years from material in the repository.

Continuing with once-through fuel consumes the most uranium ore (see Figure 8.1-13).

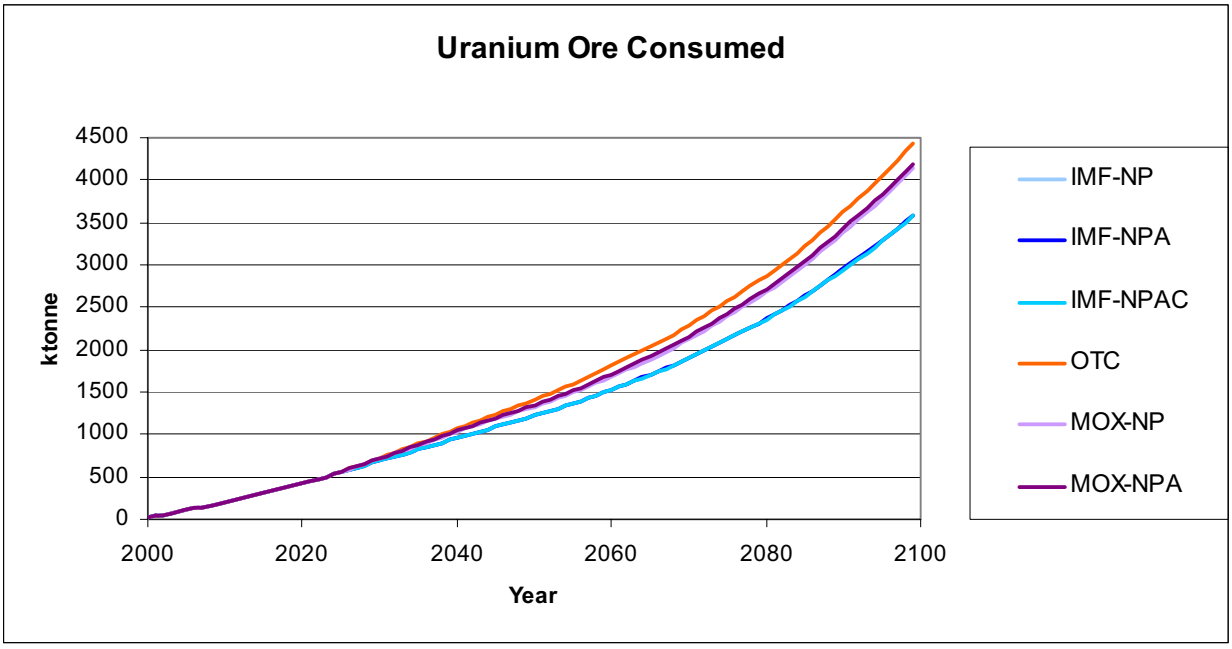

Figure 8.1-13. Uranium ore consumed.

The AFCI program has four major **objectives**,[DOE2005a] as follows:

1. Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials. Single pass full core IMF or MOX can significantly reduce the amount of material placed in the repository through 2100, however, neither MOX or IMF can fully consume the TRU so that MOX and IMF do not lead to a long reduction of required repository capacity. The long-term heat and dose in the repository are not significantly reduced compared to once-thru in the long term since one pass recycle does not significantly reduce the mass placed in the repository in the long-term.

2. Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for spent fuel management. MOX and IMF reduce the amount of total Pu and Pu-239 in the system compared to once-thru. IMF results in a greater reduction of total Pu and Pu-239 than MOX. MOX and IMF increase the amount of Pu-238 in the system compared to once-thru, but the weapons useable material in the system is decreased.

3. Enhance energy security by extracting energy recoverable in spent fuel and depleted uranium, ensuring that uranium resources do not become a limiting factor for nuclear power. MOX and IMF reduce the consumption of uranium ore with IMF reducing the uranium usage more than the MOX.

4. Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.

Overall, while single-pass strategies improve some aspects of the fuel cycle, they do not adequately meet all four AFCI objectives.

8.2. Comparison of multi-pass MOX versus IMF

In this section, we cross compare among some of the branches in Chapter 7 to examine multi-pass IMF versus MOX. Both are NpPuAm. The branches compared are as follows:

- T1.2 Once through
- T1.3 Multi-pass blended-core IMF starting in 2040
- T1.4 Multi-pass full-core MOX starting in 2040
- T2.3 Multi-pass blended-core IMF starting in 2025
- T4.3 Multi-pass full-core MOX starting in 2025

The graphs show that in almost all cases, IMF beats MOX. The primary counter example is the higher Cm in IMF after a few cycles; this was predicted from the analyses in Chapter 5. The cause seems to be that the IMF approach transmutes more of the Pu, Np, and Am with the byproduct of higher TRU isotopes such as Cm244.

The graphs also give a measure of the impact of delaying recycling from 2025 to 2040. In many cases, starting multi-pass IMF in 2040 catches up to benefits from multi-pass MOX started in 2025, an example of why we describe IMF has being a better "control knob" than MOX.

As we look deeper into some of the metrics here versus Chapter 7, we see more of the limitations of the existing model – most especially the lack of isotope decay while material is storage. (The first 5 years of decay after discharge is captured because we use 5-year-after discharge recipes.)

8.2.1 Basic parameters

We start with the mass in the system. Figure 8.2-1 shows the mass in the entire system. The two cases that start in 2025 diverge from once-through sooner than the 2040 cases, but there are time lags between when recycling starts and when the total mass in the system starts to change.

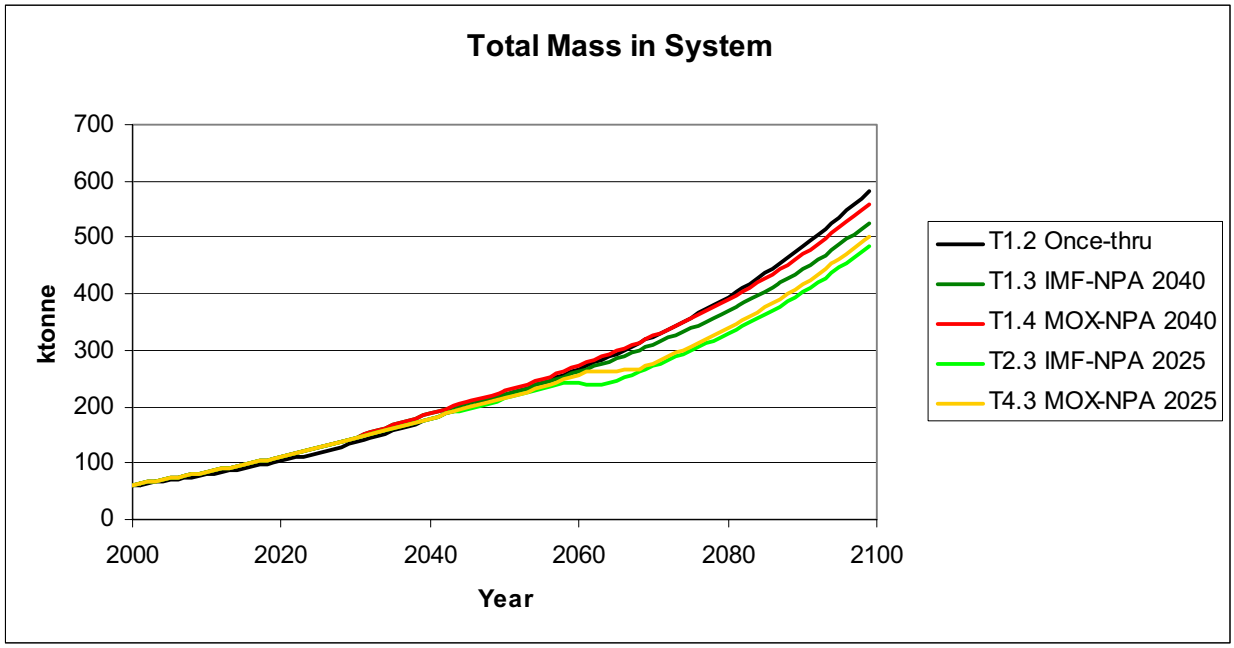

Figure 8.2-1. Total mass in the system for the cases being cross compared

Figure 8.2-2 is the first of several graphs that divide the total mass into key components, starting with uranium. Because uranium dominates the mass of the system, this figure is extremely similar to Figure 8.2-1.

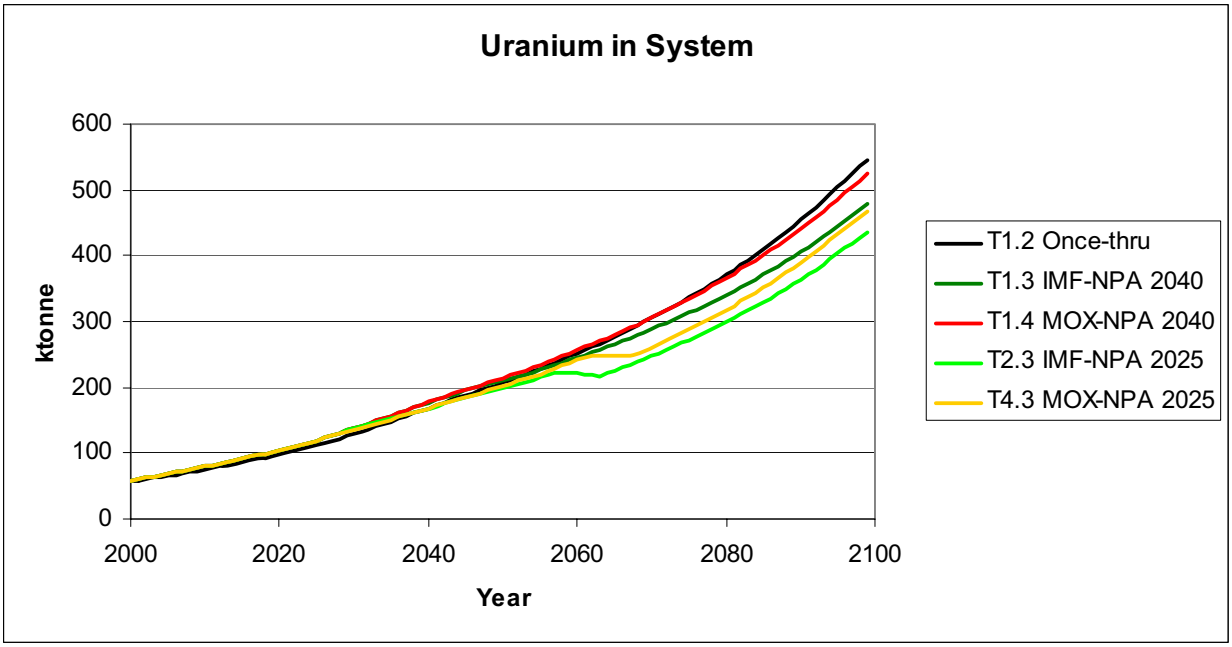

Figure 8.2-2. Total uranium in the system for the cases being cross compared

Figure 8.2-3 shows the total plutonium in the system. Multi-pass IMF can basically "freeze" the inventory of Pu even with 1.8%/year growth; MOX cannot. However, it appears that the later IMF is started, the more difficult it is to freeze the Pu inventory.

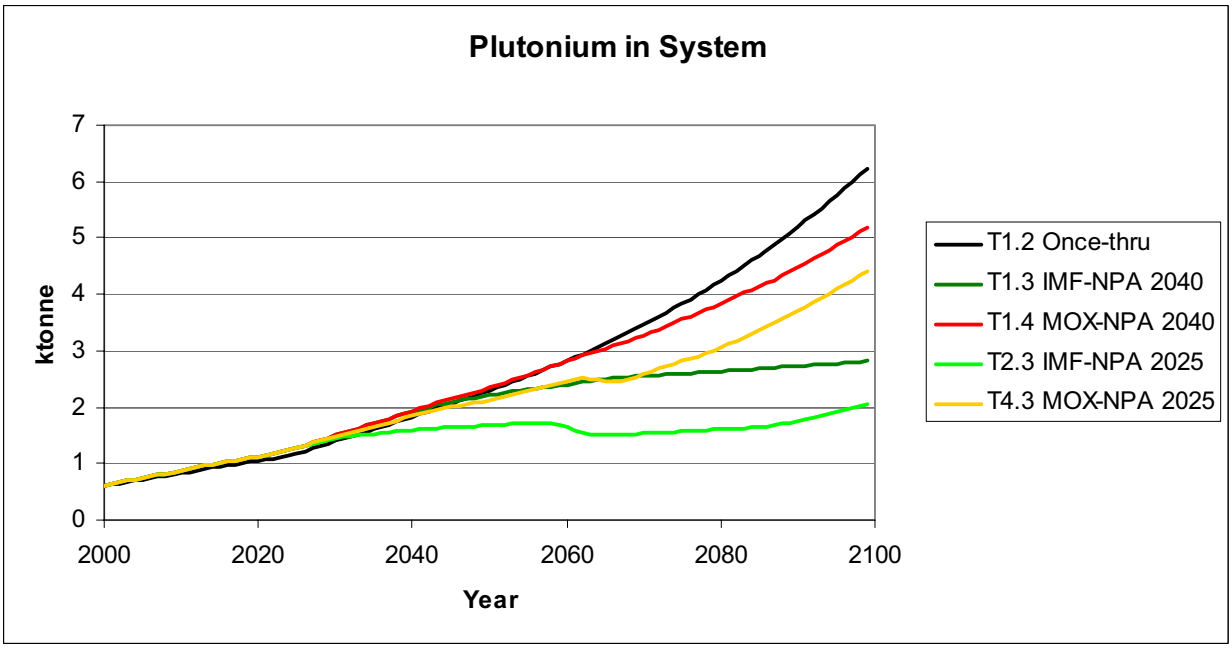

Figure 8.2-3. Total Pu in the system, note that there is more Pu in the MOX cases. Both MOX and IMF have lower Pu than once-through.

Figure 8.2-4 shows only Pu239; the trends are similar to total Pu. IMF's known ability to better burn Pu239 is quite evident. Figure 8.2-5 shows only Pu238; there is substantial increase in the Pu238 inventory in the system for the MOX cases. We do not fully understand (yet) the difference between MOX and IMF, but we do know that MOX has a larger recirculating inventory yet burns less of several TRU isotopes.

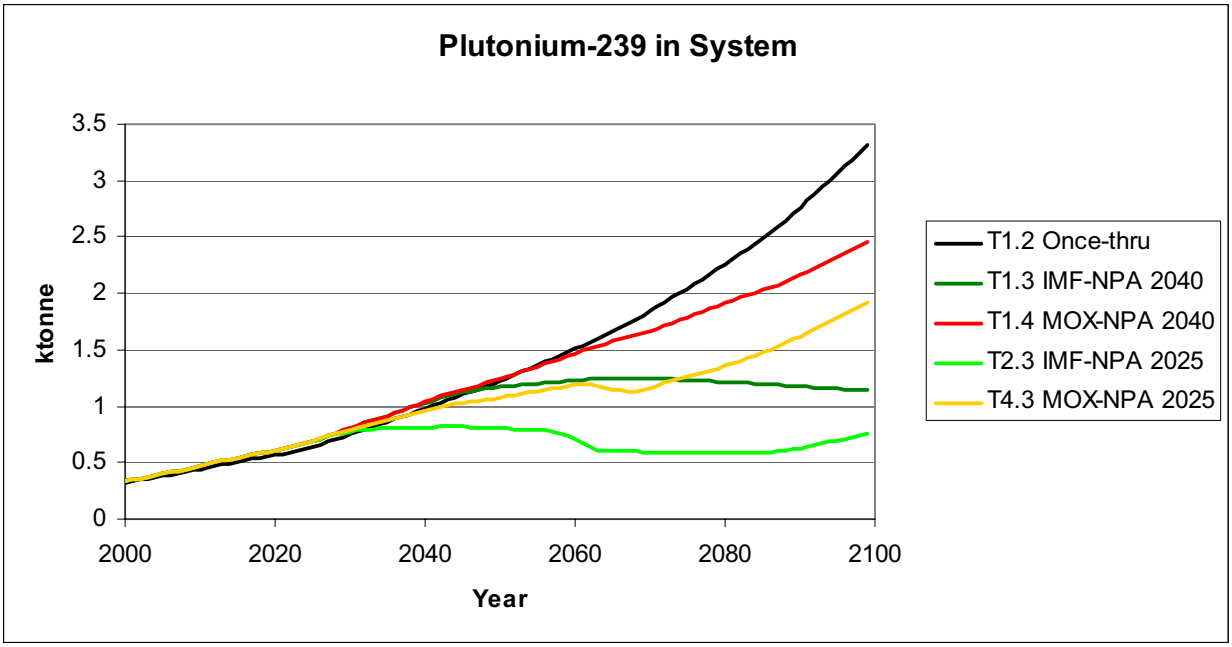

Figure 8.2-4. Pu239 in the system.

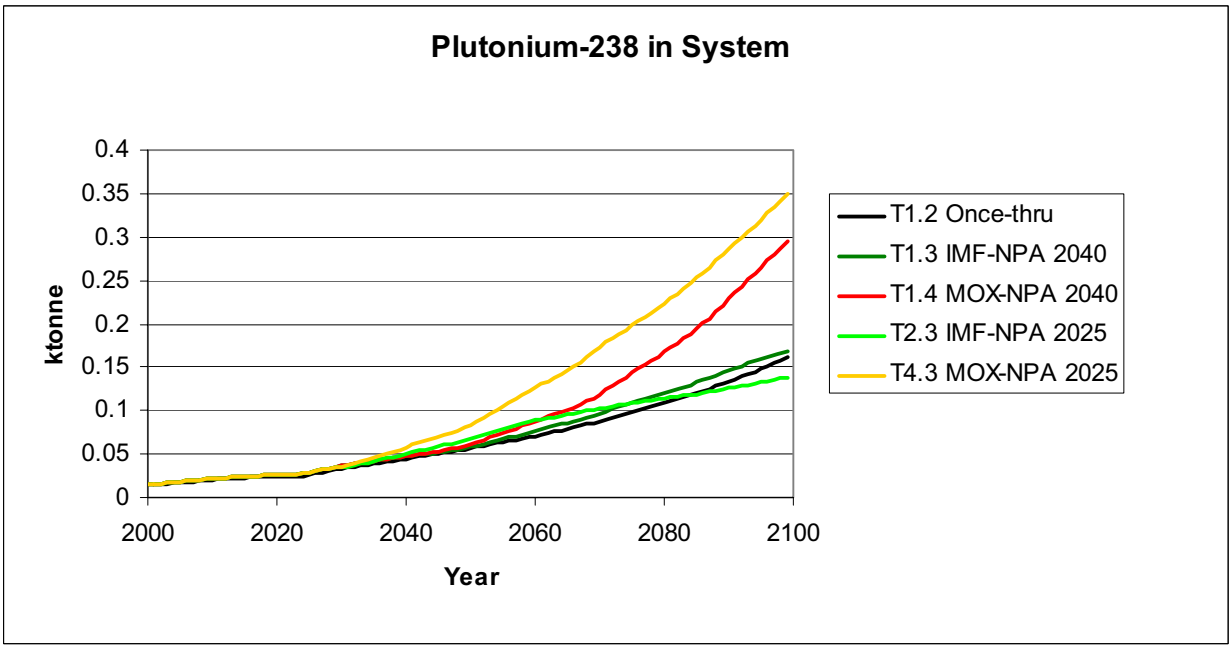

Figure 8.2-5. Pu238 in the system

Figure 8.2-6 shows total Np in the system, which is basically Np237. The IMF systems appear substantially more successful in ridding the system of Np237, which is a major contributor to long-term dose. Note that MOX/2025 has a 15 year head-start on IMF/2040; it takes until about 2085 before IMF/2040 passes MOX/2025. Starting earlier helps.

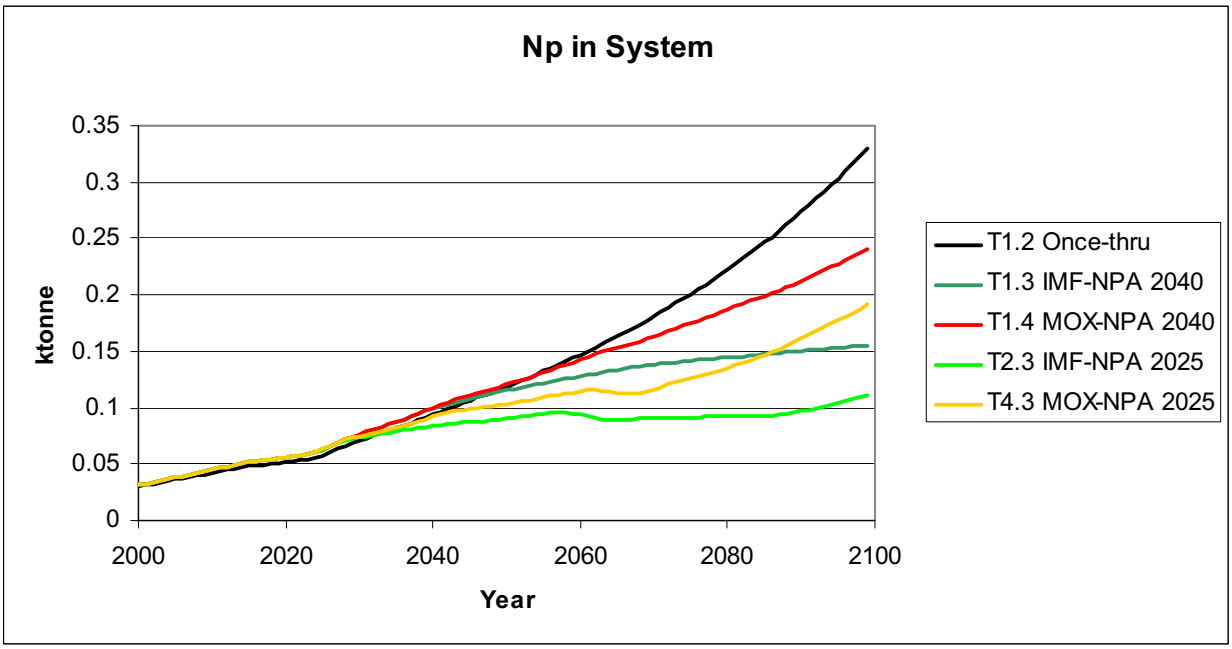

Figure 8.2-6. Total Np in the system, note that there is more Np in the MOX cases. Both MOX and IMF have lower Np than once-through.

Figure 8.2-7 shows total Am. We suspect that all of the Am curves have a systematic error, which is most pronounced for once-through. We speculate that if isotopic decay is included in future models we would see that the Am in once-through is higher than for IMF or MOX, but this is not for certain. Consider Figure 8.2-8, which shows Am241. Both Figure 8.2-7 and 8.2-8 show that MOX deviates up and IMF deviates down from once-through; more deviation the sooner recycling starts.

Recall that DYMOND does not explicitly account for isotopic decay while material is in storage. Isotopic decay is built into the 5-year-after-discharge recipes. Decay of Pu241 into Am241 is one of the major pathways to Am241. Thus, the increase of Am241 while SNF is in longer-term storage is not accounted for. Once fuel is discharged, the isotopics in DYMOND are accounted for using the 5-year-after discharge recipes. This overstates decay from zero to 5 years; but misses decay thereafter. So, most of the decay of Pu241 into Am241 for the once-through curves is missing.

For the recycle cases, however, most of the decay is accounted for; consider a 12-year loop (if separation capability is sufficient). Decay of Pu241 during the 4-5 years that fuel is in a reactor is accounted for; the decay of Pu241 during the first 5 years after discharge is accounted for. Decay during the 2 years of separation and fabrication is not accounted for. So, up to 83% (10/12) of the decay is accounted for, cycle by cycle.

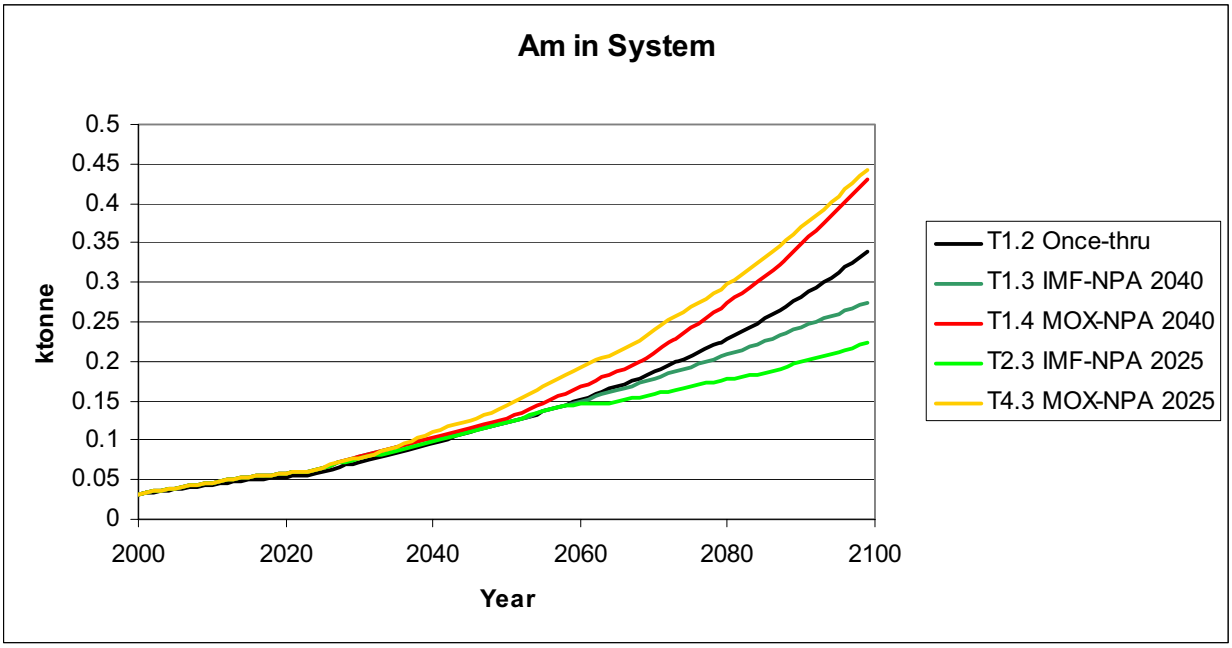

Figure 8.2-7. Total Am in the system, note that there is more Am in the MOX cases than in the IMF cases because some Pu is recycled, producing Am, but less Am is burned. However, believe that once-through actually as more Am than either IMF or MOX.

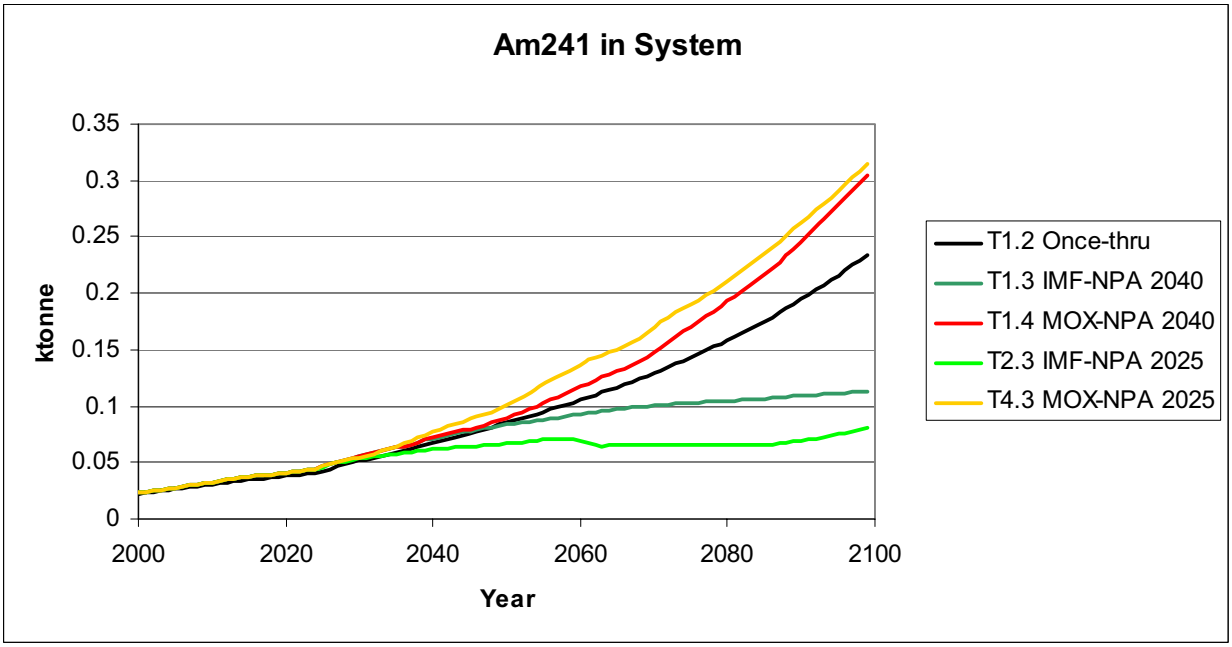

Figure 8.2-8. Total Am241 in the system. We suspect that the Am241 content in once-through is understated because it is missing the decay of Pu241 into Am241; the Am241 content in once-through is possibly higher than either MOX or IMF.

Figure 8.2-9 shows total Cm in the system. Consistent with observations in Chapter 5, there is more Cm accumulation in the IMF systems than in MOX. This is the major penalty for IMF's ability to more effectively and more quickly transmute the lower TRU. Figure 8.2-10 shows that indeed the majority of

the Cm mass is Cm244. Because of its 18.1-year halflife, the decay of Cm244 during the simulation is significant, but missed for once-through for reasons noted above. Thus, we believe the Cm and Cm244 mass for once-through is probably overstated.

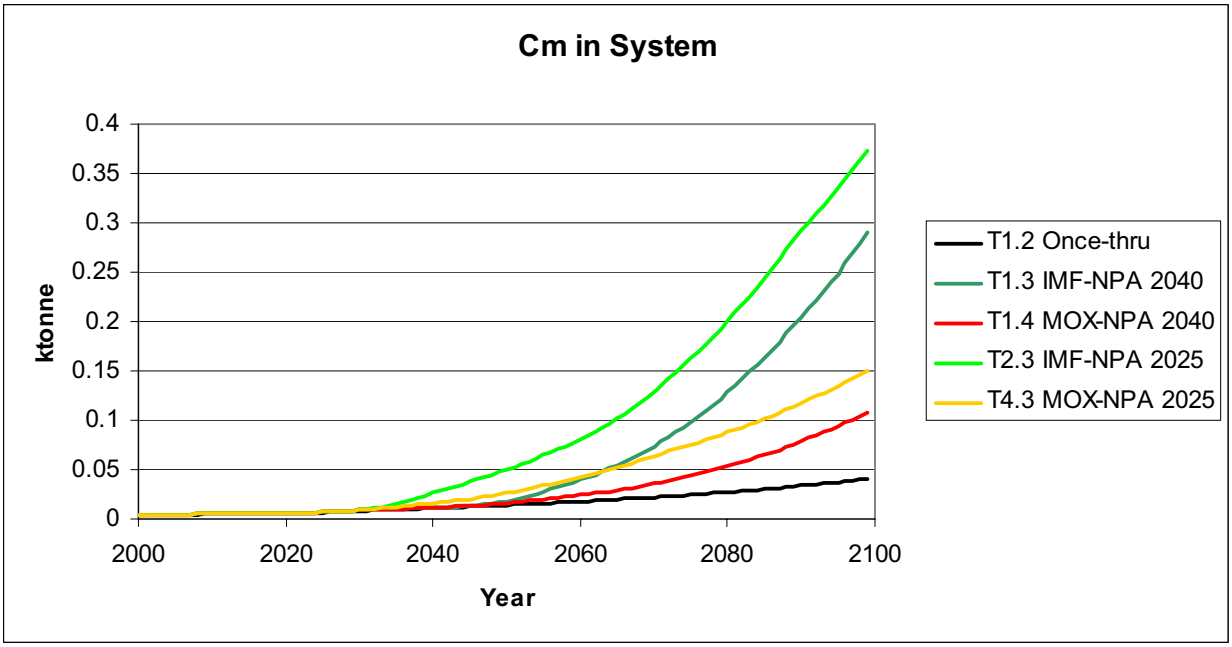

Figure 8.2-9. Total Cm in the system, note that there is more Cm in the IMF cases than in the MOX cases because the Am is more effectively transmuted with the side effect of generating Cm. Cm mass in once-through is probably overestimated.

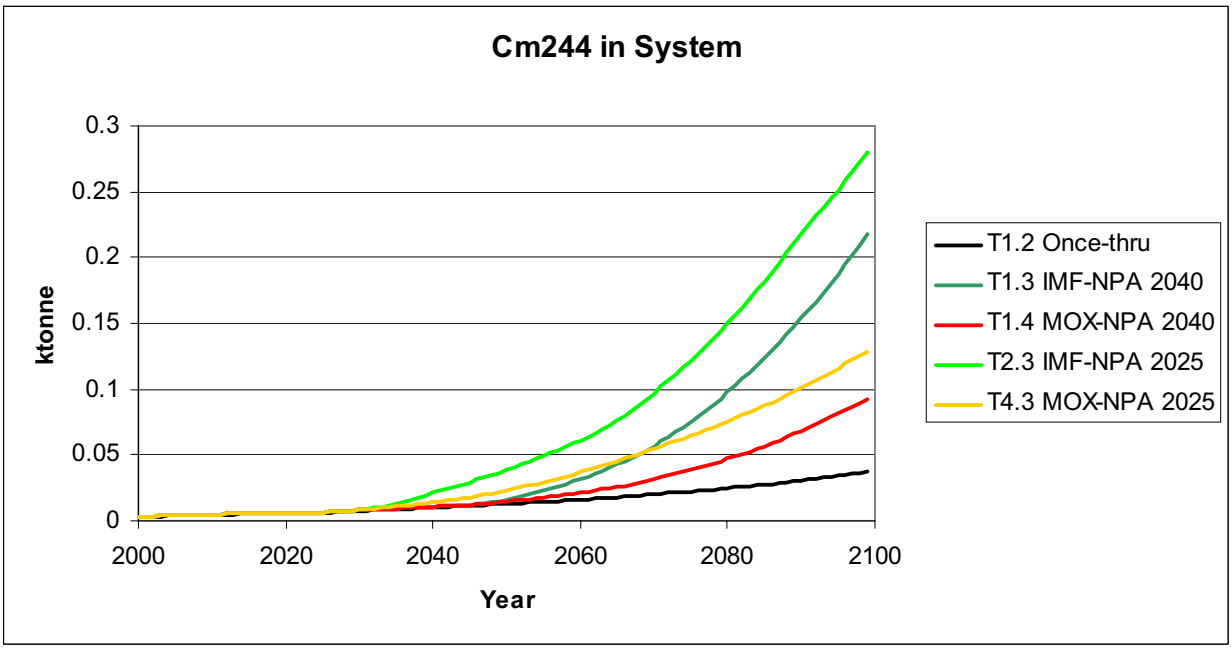

Figure 8.2-10. Cm244 in the system, note that there is more Cm in the IMF cases than in the MOX cases. Cm244 mass in once-through is probably overestimated.

8.2.1 Waste management

We now turn to waste management, starting with Figure 8.2-11, which shows the total mass in the repository. As in Chapter 7, the once-through curve simply increases at 3000 tonnes/year, the assumed receipt rate at the repository. For the two 2025-start cases, mass starts being withdrawn at 2040, when the second separation plant comes on line, doubling reprocessing capacity. Per GWe, IMF has slightly lower mass throughput, so that repository mass is withdrawn slightly faster. Between 2060-2070, all emplaced SNF has been withdrawn for either IMF or MOX; the only mass in the repository is HLW left over from recycling. The slope change in 2060 reflects the third separation plant coming on line. The 2040 cases start withdrawing mass from the repository about 20 years after recycling starts, after the second separation plant comes on line. The slope change in 2080 again reflects the third separation plant coming on line. But, the 15-year delay in recycling means that there is still emplaced SNF in the repository at the end of the simulation.

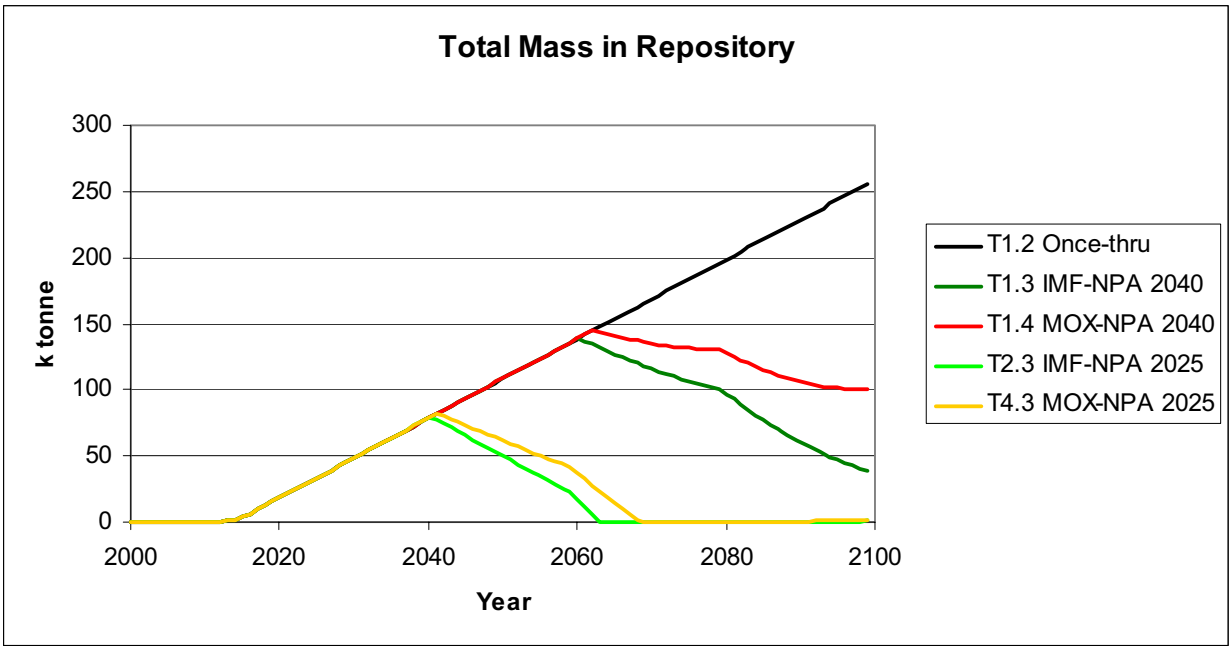

Figure 8.2-11. Total mass in the geologic repository

Figure 8.2-12 shows the LTH metric for the mass in the repository. Recall that LTH is defined in Chapter 3; it is a measure of time-integrated heat load to the repository. The metric here is for 50-years ventilation time. The trends mirror those of total mass, Figure 8.2-11.

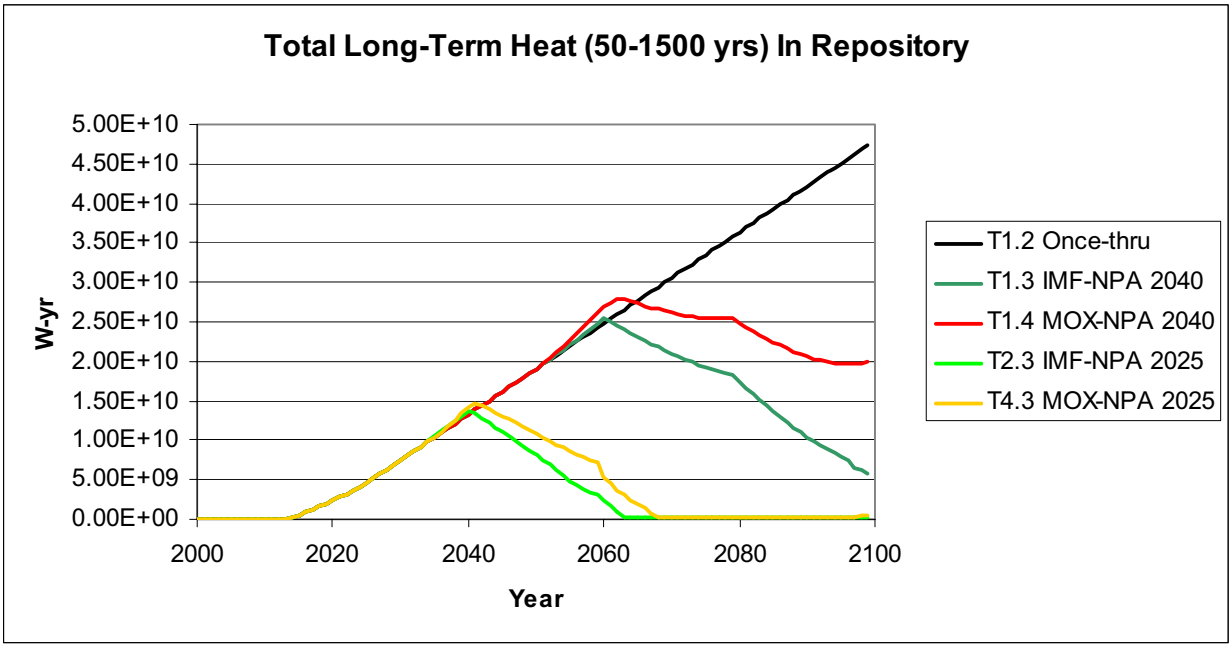

Figure 8.2-12. Long-Term Heat (LTH) integral for 50-year ventilation time for the mass in the repository.

Figure 8.2-13 is a subset of Figure 8.2-12, looking only at the lower part of the graph. The emplaced LTH is falling at 2060 as mass is withdrawn from the repository; with IMF faster than MOX. The LTH does not go back to zero, of course, because of the HLW from processing losses. More IMF goes through more cycles, which means that there is more accumulated HLW, so from 2070 to 2090, there is actually more IMF LTH in the repository than MOX LTH.

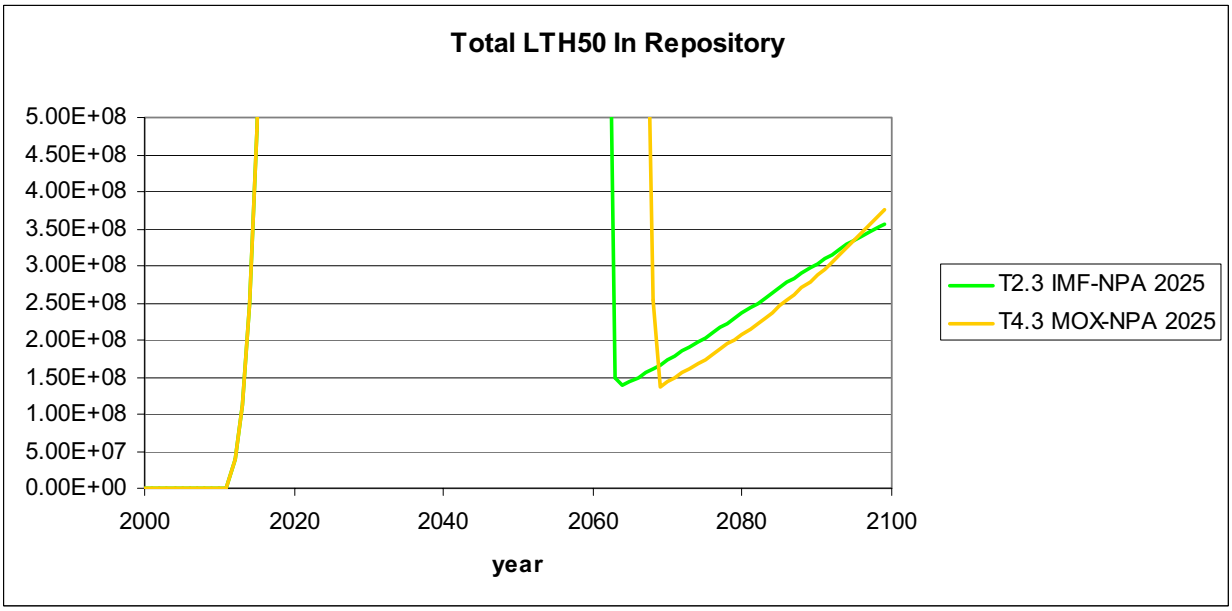

Figure 8.2-13. Long-Term Heat (LTH) integral for 50-year ventilation time for the mass in the repository, zooming on the lower part of the graph

Figure 8.2-14 shows the long-term dose (LTD) metric for mass in the repository. LTD is defined in Chapter 3. This plot shows LTD at 500,000 years after emplacement. The basic trends are the same as before, dominated by withdrawal of mass from the repository.

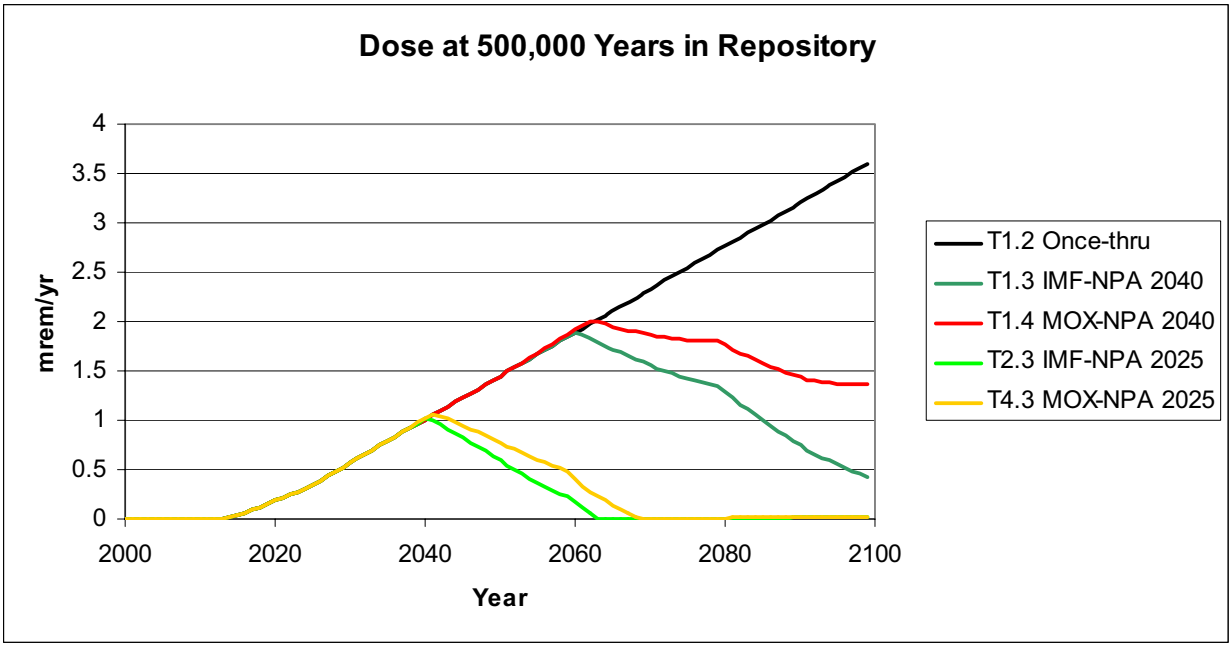

Figure 8.2-14. Hypothetical Long-Term Dose (LTD) 500,000 years into the future from mass emplaced in the repository

8.2.3 Proliferation resistance

Figure 8.2-15 shows how recycling degrades Pu, using the simple metric Pu239/Pu-total. IMF degrades the Pu faster and further than MOX. Remember, however, that here IMF is blended core, multi-pass IMF. It is therefore not designed to destroy the maximum amount of Pu as does 1-pass IMF; it is designed to burn as much Pu as possible while keeping recycling going. This moderates how far the Pu is degraded; more analysis would be required to better understand what, if anything, this degree of degradation accomplishes from the proliferation perspective.

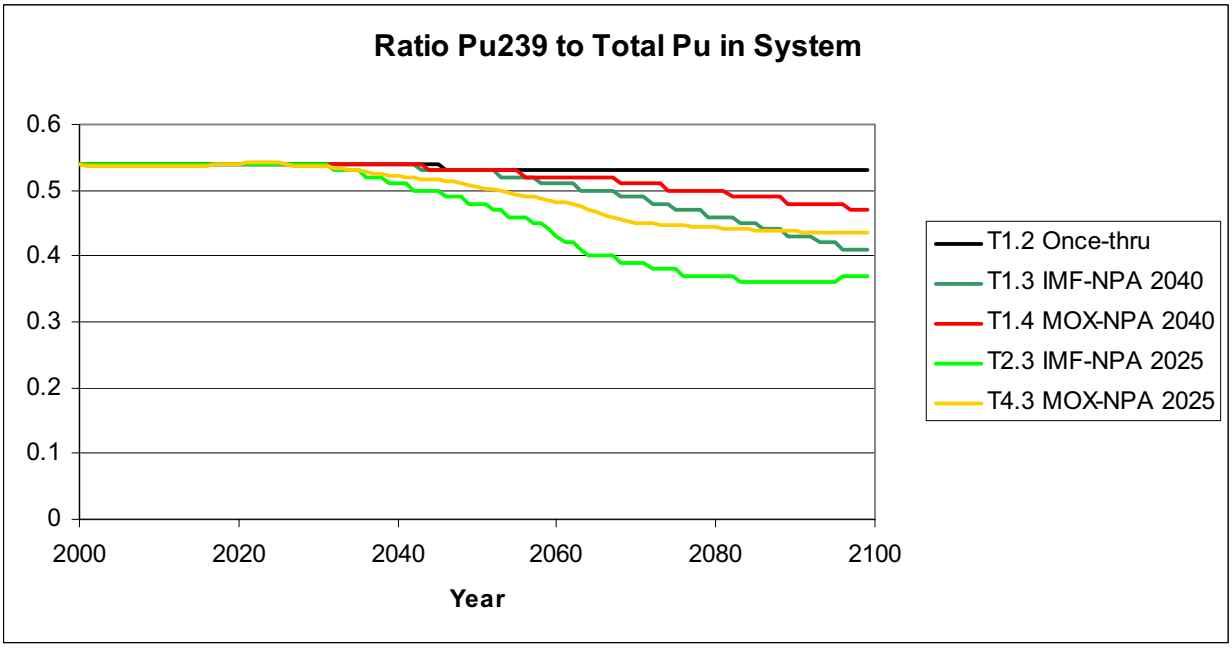

Figure 8.2-15. Ratio of Pu239/Pu-total throughout the system. IMF degrades the Pu vector more and faster than MOX.

Figure 8.2-16 shows the weapons-usable (WU) inventory. Per Chapter 3, "weapons-usable" is measured in Pu239-equivalent. For once-through, the slow increase is, of course, simply the mass in interim storage. Blended core multi-pass IMF-NpPuAm is superior to Full core multi-pass MOX-NpPuAm with regard to lowering the WU inventory.

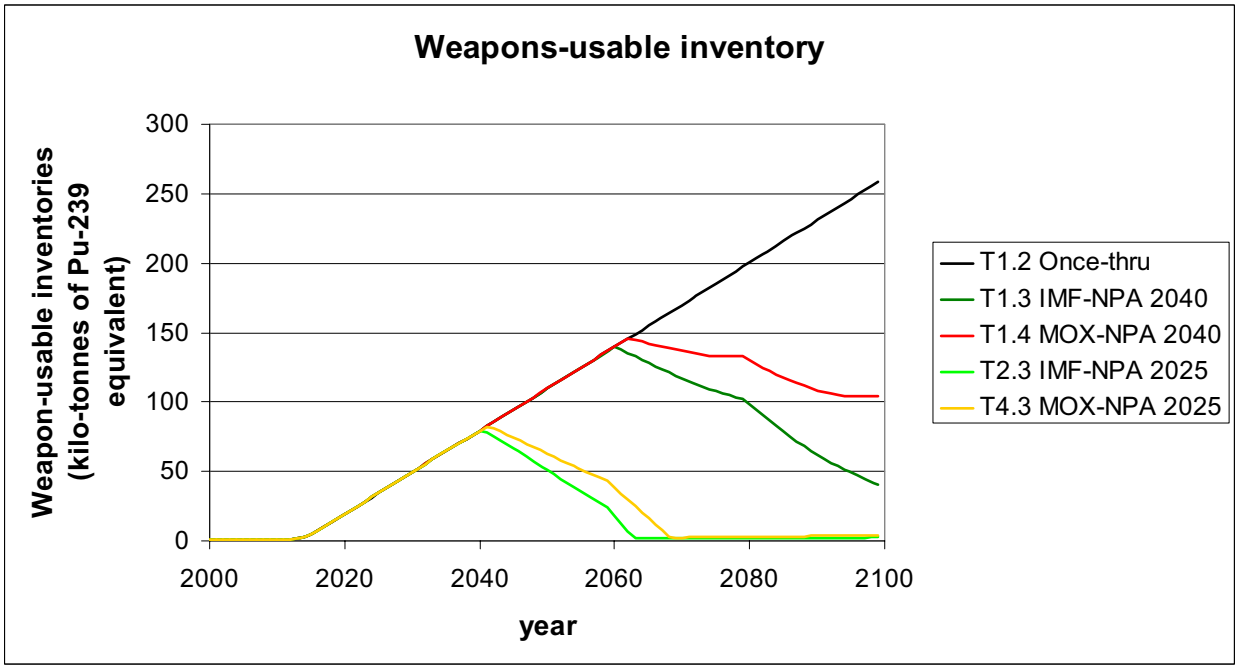

Figure 8.2-16. Weapons-usable inventory measured in "Pu239-equivalent" mass

8.2.4 Energy recovery

Figure 8.2-17 shows the uranium ore consumed. Consistent with Chapter 5, we observe that IMF needs less uranium ore, because it is more effectively burning plutonium. Of course, the uranium savings are higher the sooner recycling starts. By the end of the simulation, IMF/2025 has achieved almost 19% savings relative to once-through; this is slightly higher than the "equilibrium" value of 17% calculated in Chapter 5 because the system has used the legacy SNF. IMF/2040 lags slightly behind because it has not yet used all of the legacy SNF in the repository; it should eventually catch up to IMF/2025. At 2100, IMF/2040 is only at 16% savings. The MOX cases lag the IMF cases because (a) IMF burns plutonium better and (b) the lower TRU throughput allows IMF to proceed cycle-by-cycle further and faster than MOX.

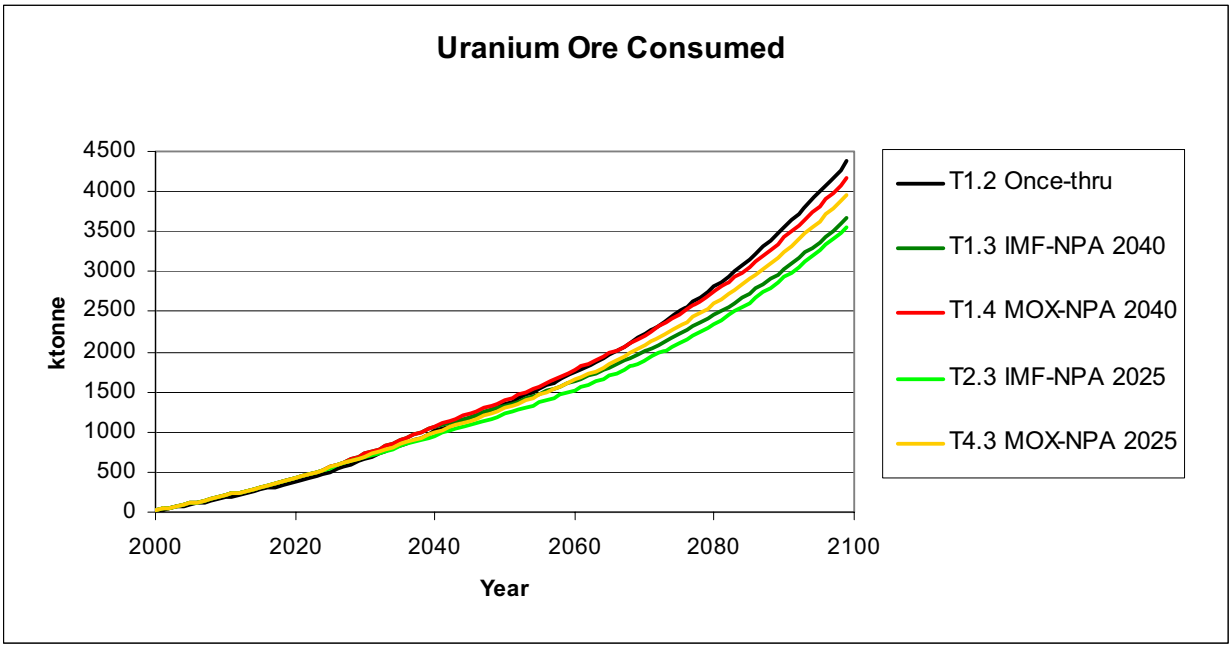

Figure 8.2-17. Uranium ore consumption

8.2.5 At-reactor storage inventories

Figure 8.2-18 shows the at-reactor wet storage inventory. In the current simulations, this is simply all SNF within 5 years of discharge. All the curves ramp up during the first 5 years of the simulation because we start the simulation with zero wet storage; it build to the correct value 5 years later as fuel is discharged from reactors starting in 2000. The MOX and once-through cases have almost the same SNF mass; they have the same burnup, 51 MW-day/tonne-HM. The IMF cases have slightly higher burnup, 66 to 58 MW-day/tonne-HM depending on which cycle; so for the same energy produced, there is slightly less SNF mass being discharged.

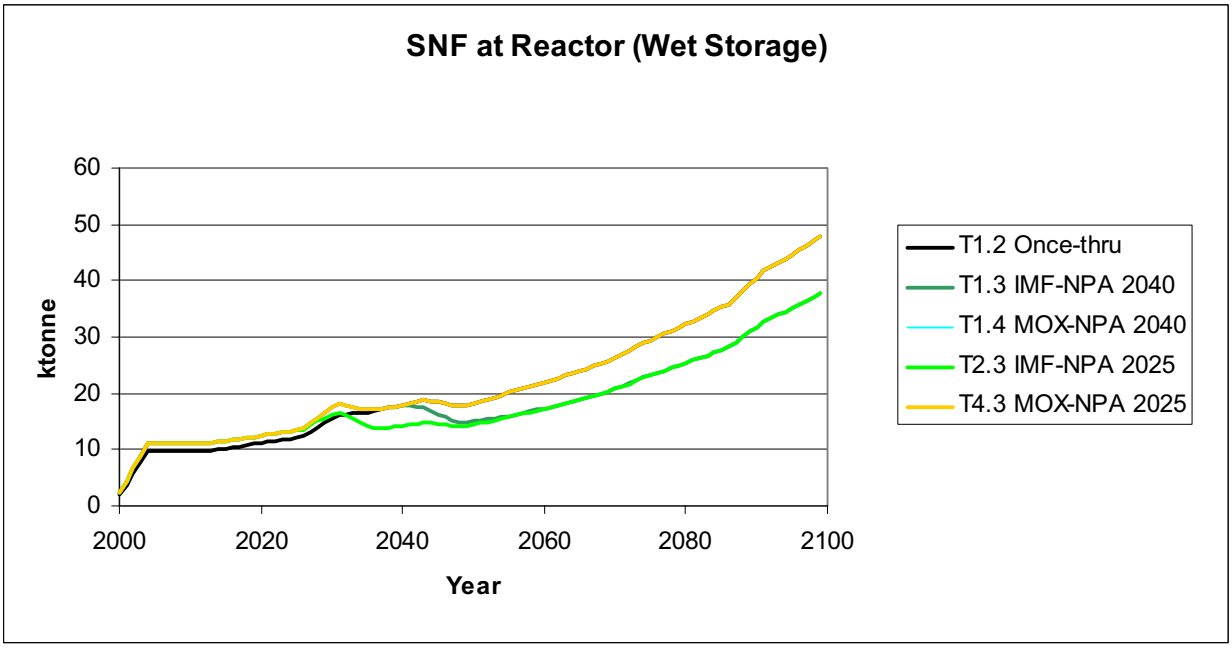

Figure 8.2-18. At-reactor wet storage inventories. The once-through and MOX cases overlay each other; the IMF cases overlay each other, except for at 15-yr mismatch.

Figure 8.2-19 shows dry storage at reactors. In the current model, this is all fuel that is more than 5 years old but has not yet been shipped to the geologic repository or separation plants. All curves are wrong for the first several years of the simulation because we start dry storage inventory at zero in 2000; all such inventory is instead considered "legacy SNF" and accounted for separately. The once-through curve is simply the difference between total SNF minus the 3,000 tonnes/year sent to the geologic repository. The 2025 curves start decreasing shortly after reprocessing starts. (Note that they do not decrease when the repository opens in 2012 because legacy SNF is sent to the repository first.) Once the separation plant opens in 2025, legacy SNF goes to the repository and dry-storage SNF goes to the separation plant. By 2040, the at-reactor dry storage inventory in the 2025-cases is gone; thereafter, the separation plants use a combination of fuel reaching 5-years age plus SNF withdrawn from the geologic repository. The pattern for the 2040-cases is the same, only delayed. The difference between the 2025-cases and the 2040-cases shows how much additional at-reactor storage is required because of the delay in recycling.

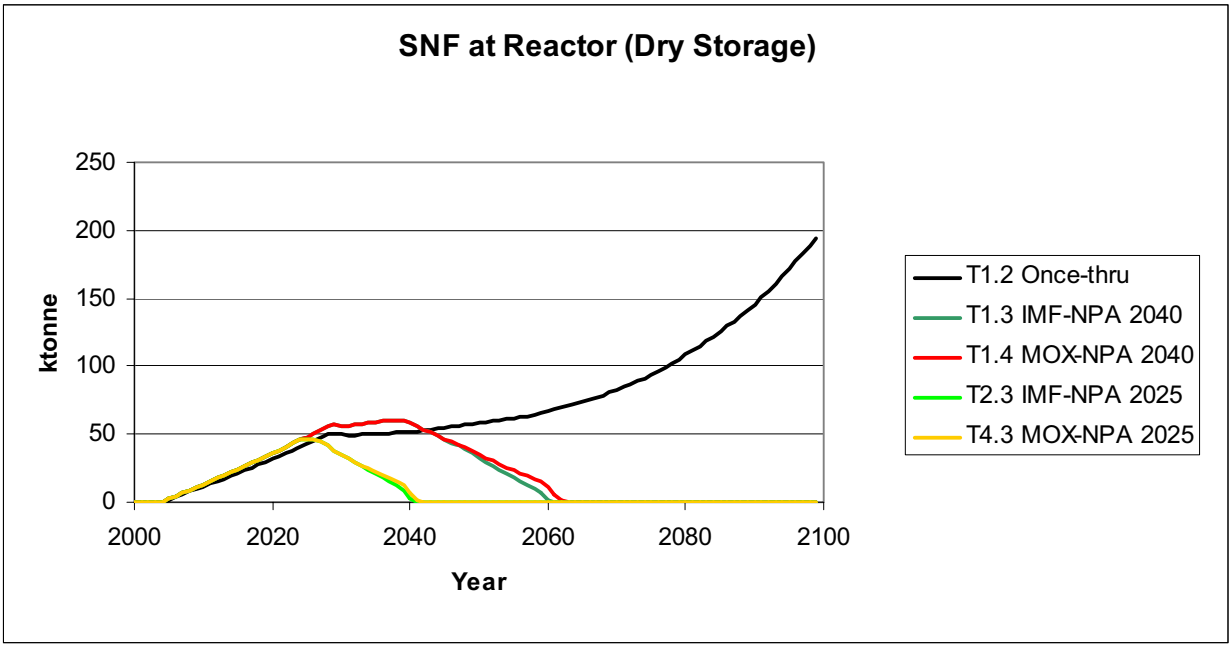

Figure 8.2-19. At-reactor dry storage inventories

8.3. Comparison of multi-pass versus single-pass

This section compares the results of single-pass fuels to multi-pass fuels for MOX and IMF. The current model has fuel recipes for five full core single-pass fuels, MOX-NpPu, MOX-NpPuAm, MOX-NpPuAmCm, IMF-NpPu, and IMF-NpPuAmCm and one full core multi-pass fuel, MOX-NpPuAm. The model also has a recipe for one blended core fuel, IMF-NpPuAm which can be run as a single or multi-pass fuel.

In the model, some legacy (reactors existing at time = 0 in the model) reactors are assumed to only be able to use UOX fuel, these are designated LWRu in Figure 8.3-1; the remaining legacy reactors are assumed to be able to use UOX, MOX and IMF fuels and are designated LWRm in Figure 8.3-1. All new (reactors built after time=0 in the model) light water thermal reactors (LWR) built are capable of handling UOX, MOX , and IMF fuels and are also designated LWRm in Figure 8.3-1.

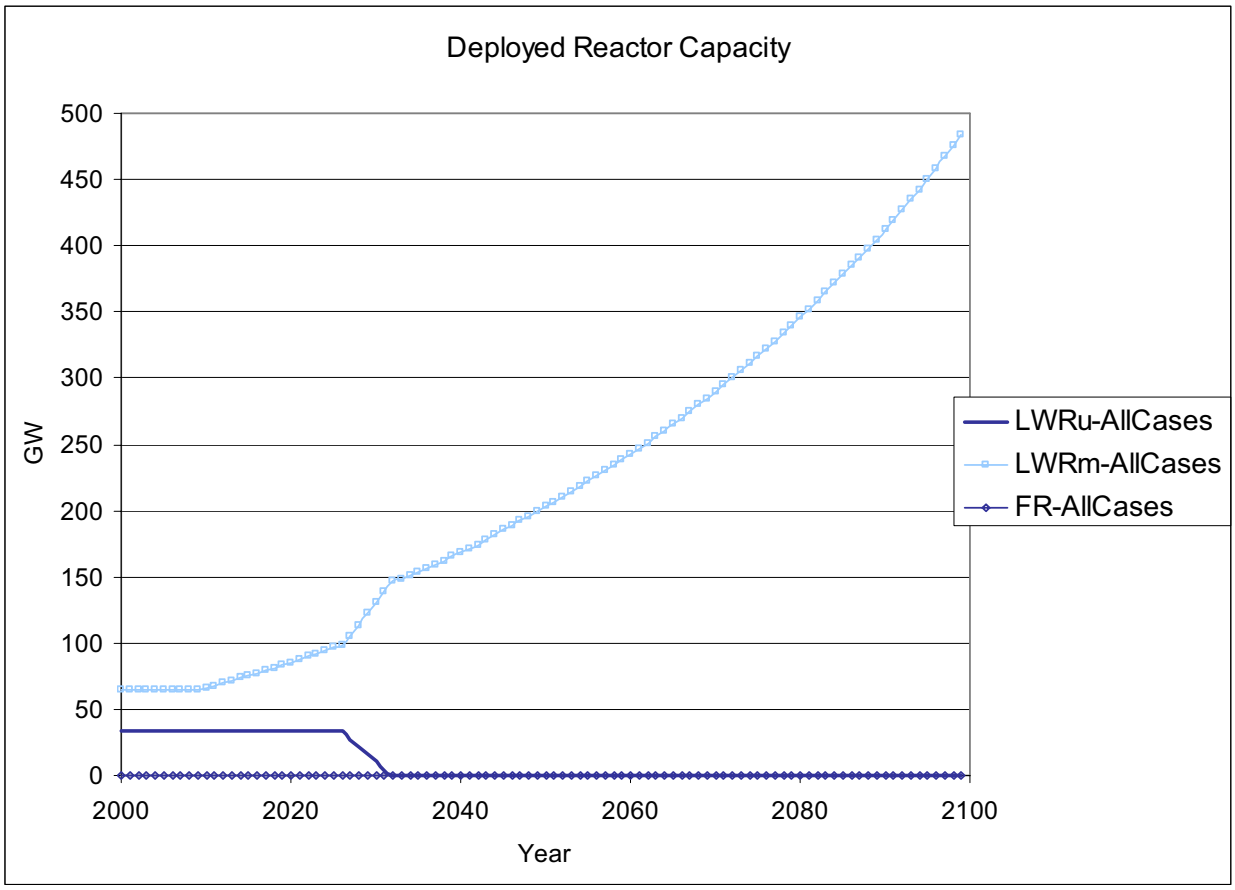

Figure 8.3-1. Deployed light water reactor (LWR) capacity. LWRu reactors can accept only UOX fuel. LWRm reactors can accept UOX, MOX, or IMF fuels.

The mass percent of plutonium in the reactors depends on the fuel recipe and whether the fuel is run in single or multi-pass mode. For IMF (see Figure 8.3-2), the single-pass blended core NpPuAm fuel results in the lowest Pu levels in the reactors by 2100; the same fuel run in a multi-pass mode results in more Pu in the reactors at 2100

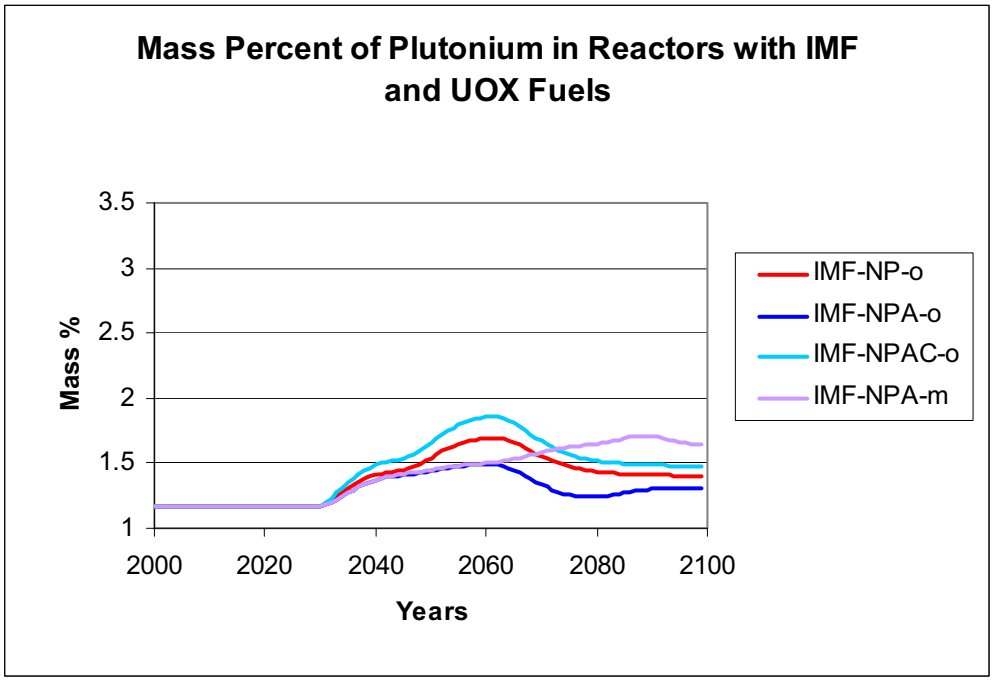

Figure 8.3-2. Mass percent of plutonium in reactors containing varying fractions of IMF fuels.

Combining the IMF fuel mass percent data presented in Figure 8.3-3 with the plutonium mass percent data presented in Figure 8.3.2 presents a slightly different perspective on the plutonium content in reactors as shown in Figure 8.3-4. The full core IMF recipes (IMF-NP and IMF-NPAC) account for less than 2% of the total fuel in the reactors by 2100, this means for these cases, most the fuel in reactors is UOX. The blended core IMF recipe (IMF-NPA) accounts for at least 40% and as much as 80% of the total fuel in the reactos at 2100. Even though there is substantially more IMF fuel in the reactors, the blended recipe has a lower effective plutonium content than the full core recipes. Multi-pass blended IMF results in a lower plutonium fraction than single-pass blended IMF.

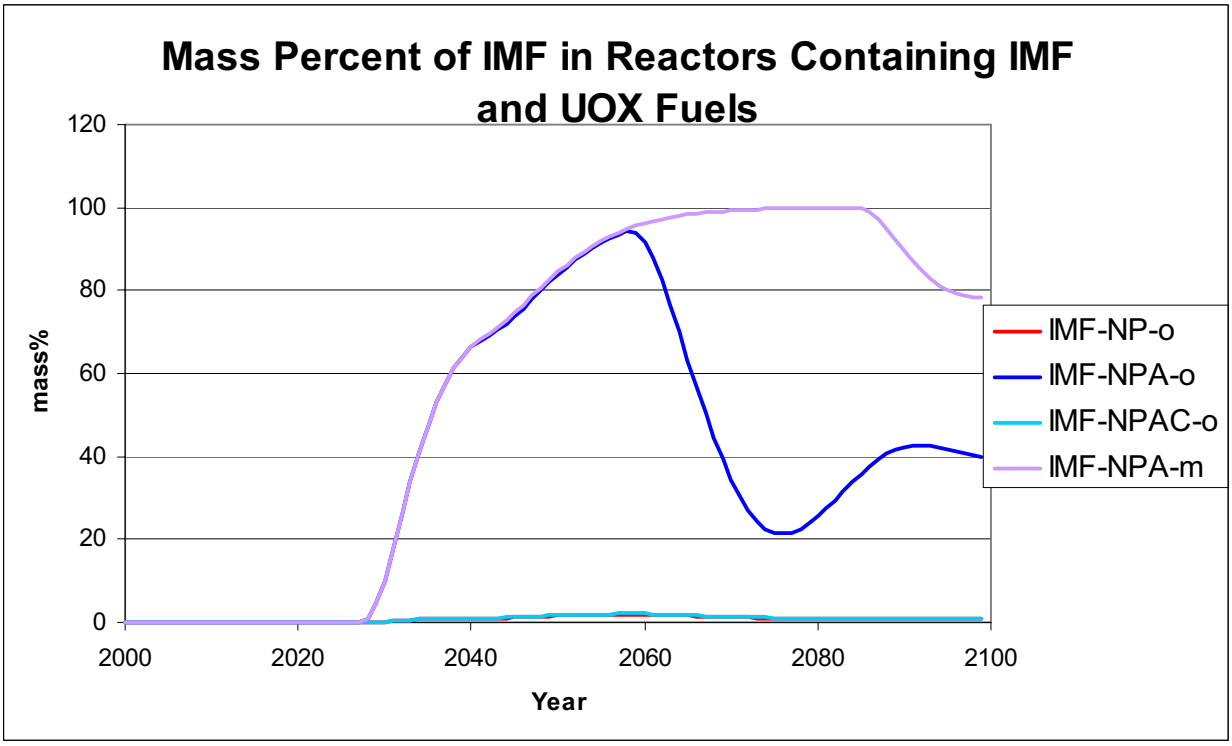

Figure 8.3-3. Mass percent of IMF in reactors containing varying fractions of IMF fuels.

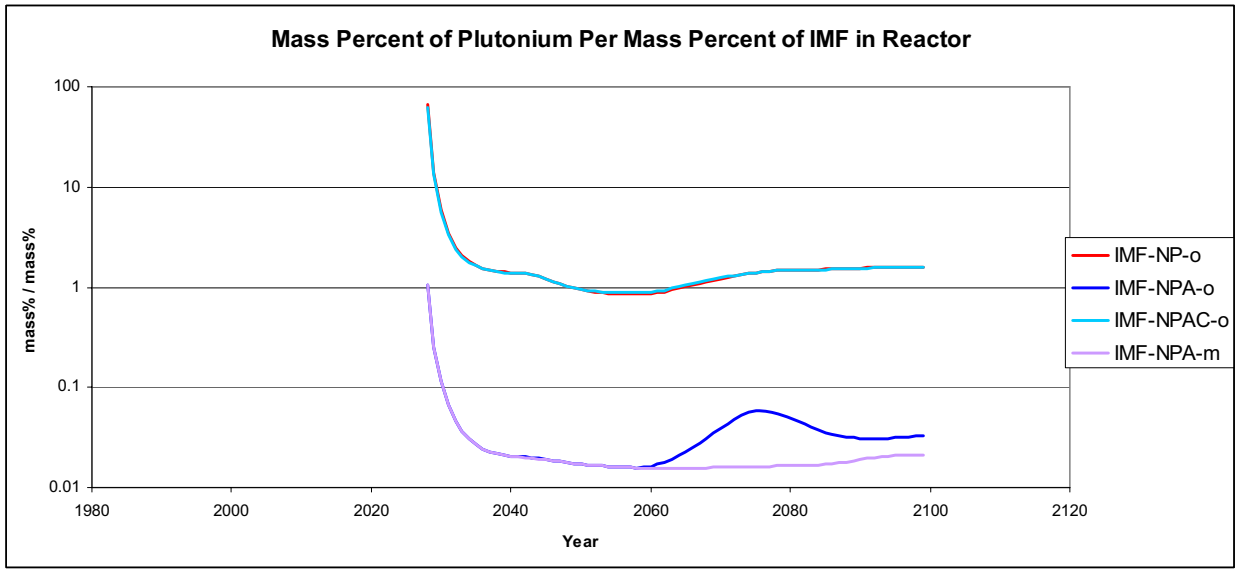

Figure 8.3-4. Mass percent of plutonium per mass percent of IMF fuel in reactors. Plutonium content normalized for IMF content in the reactors.

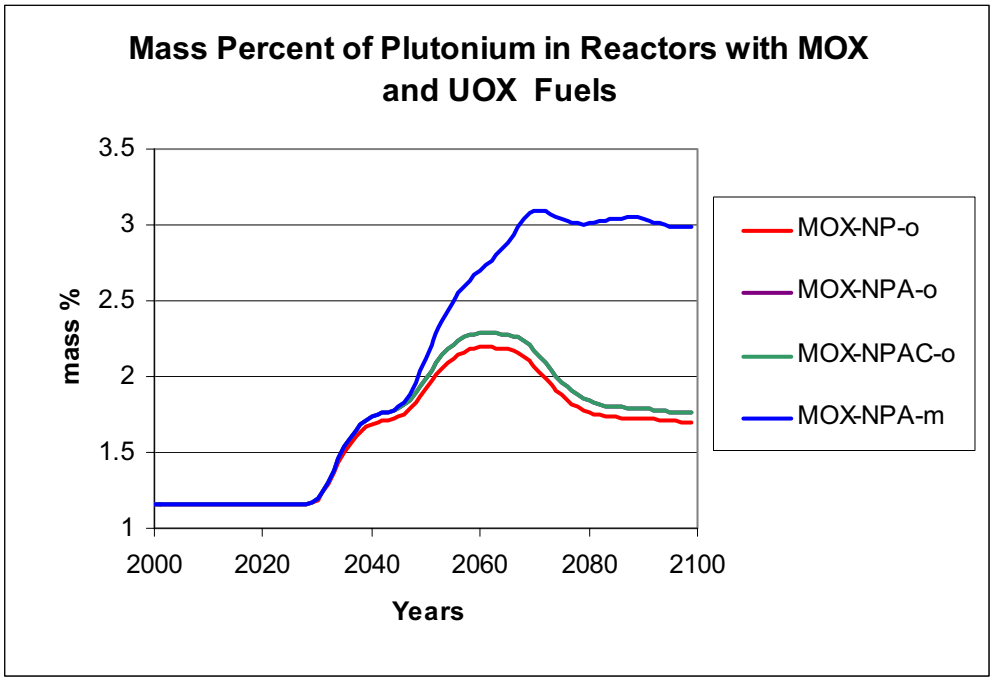

Figure 8.3-5. Mass percent of plutonium in reactors containing varying fractions of MOX and UOX fuels.

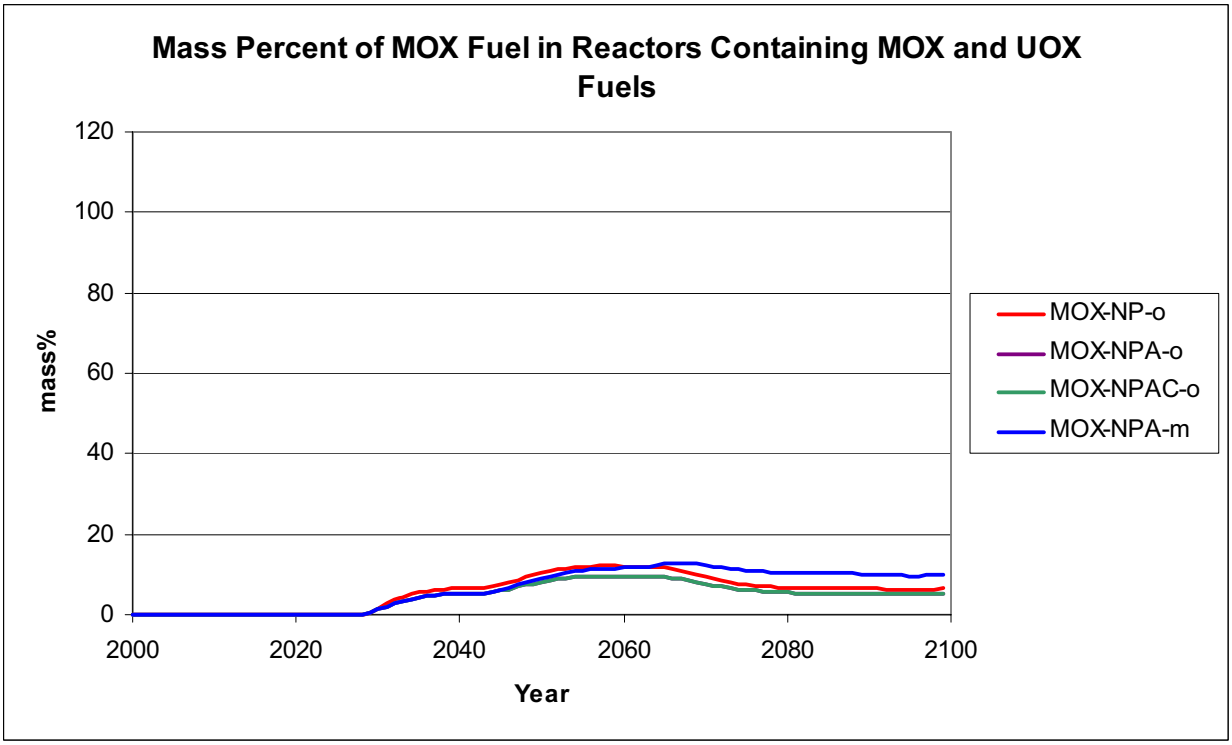

Figure 8.3-6. Mass percent of MOX in reactors containing varying fractions of MOX and UOX fuels.

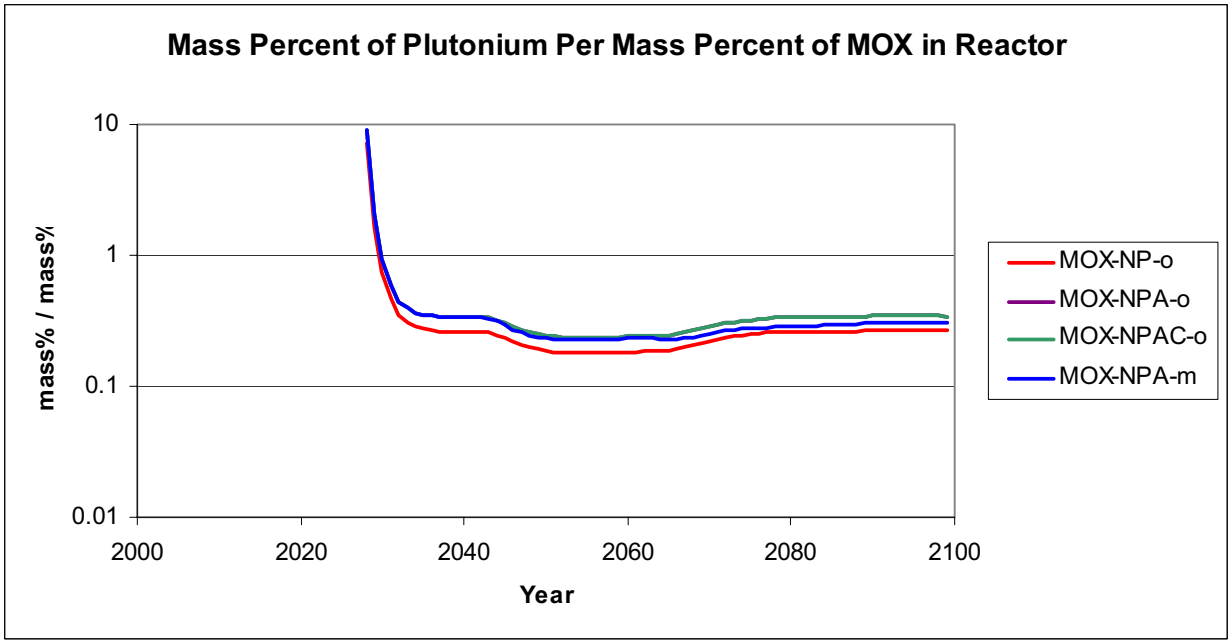

Figure 8.3-7. Mass percent of plutonium per mass percent of MOX fuel in reactors. Plutonium content normalized for MOX content in the reactors.

The current model treats full core and blended core single-pass fuels differently with respect to the repository. Single-pass fuels are not reprocessed before placement in the repository, so all MOX (Figure 8.3-8a) or IMF (Figure 8.3-8b) fuel goes to the repository. For fuel core recipes, all of the fuel categorized as MOX or IMF is burned uranium or non-uranium mixed oxides. While the blended core IMF includes fresh uranium in its recipe. In all cases, if there is not enough plutonium to make sufficient MOX or IMF, UOX is used to provide the remaining fuel required. In the current model, all UOX fuel is treated the same, so for single-pass MOX or IMF operation (all the cases with an "o" in figures 8.3-5 and 8.3-6) the spent UOX is split from the spent MOX or IMF and reprocessed rather than being sent to the repository with the spent MOX and IMF. The fresh UOX that is contained in the blended single-pass IMF (IMF-NPA-o in Figure 8.3-8) stays with the IMF so that spent UOX in the blended IMF stays with the spent IMF and goes into the repository. If blended IMF (IMF-NPA-m in Figure 8.3-8) is used in multi-pass mode, all of the spent fuel is recycled so only the HLW goes to the repository. This is why the mass in the repository for IMF-NPA-o is so much higher than the IMF-NPA-m and the other IMF recipes.

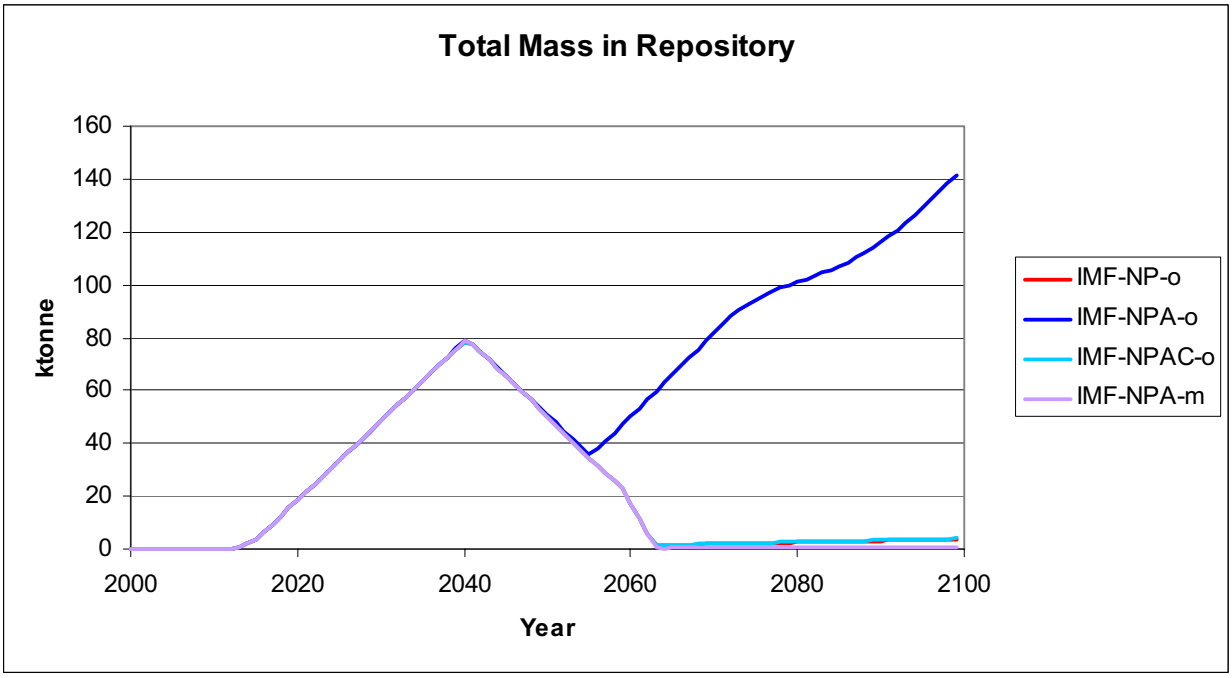

Figure 8.3-8a. Total mass in the repository for reactors containing IMF and UOX fuels. "o" indicates single-pass case and "m" indicates multi-pass case.

Looking at the full core single-pass cases, IMF (Figure 8.3-8) results in less mass in the repository by 2100 than MOX (Figure 8.3-9). This is not surprising since among the full core single-pass cases more MOX fuel than IMF fuel is in the reactors by 2100 (see Figures 8.3-3 and 8.3-5).

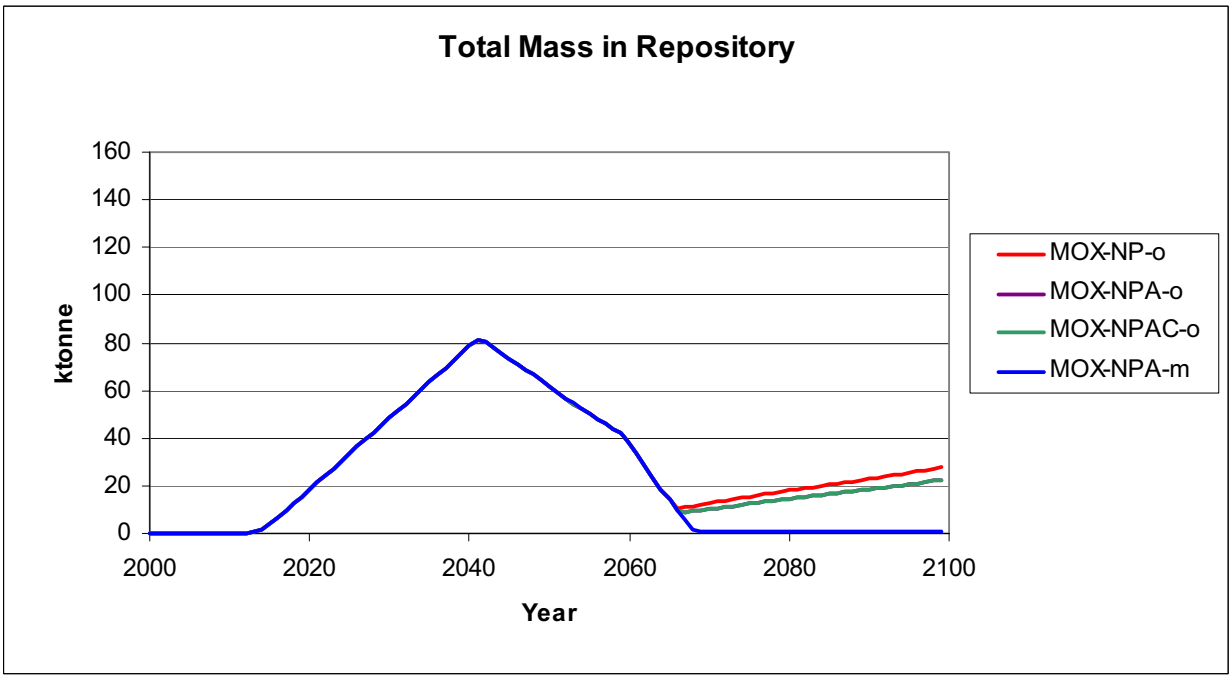

Figure 8.3-8b. Total mass in the repository for reactors containing MOX and UOX fuels. "o" indicates single-pass case and "m" indicates multi-pass case.

Moving to multi-pass fuels, either full core MOX or blended core IMF results in the highest amounts of high level waste in the repository (see Figures 8.3-9a and 8.3-9b), but these cases also significantly reduce the mass in the repository by 2100.

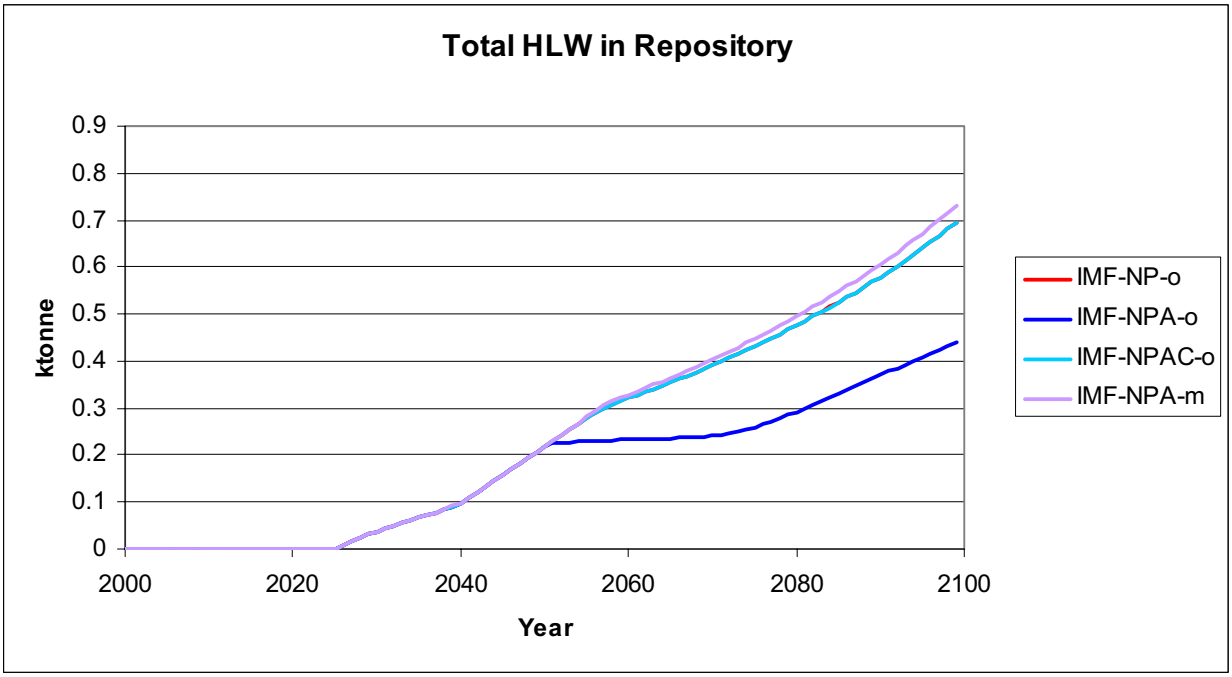

Figure 8.3-9a. High level waste mass from reprocessing and fuel fabrication in the repository for reactors containing IMF and UOX fuels. "o" indicates single-pass case and "m" indicates multi-pass case.

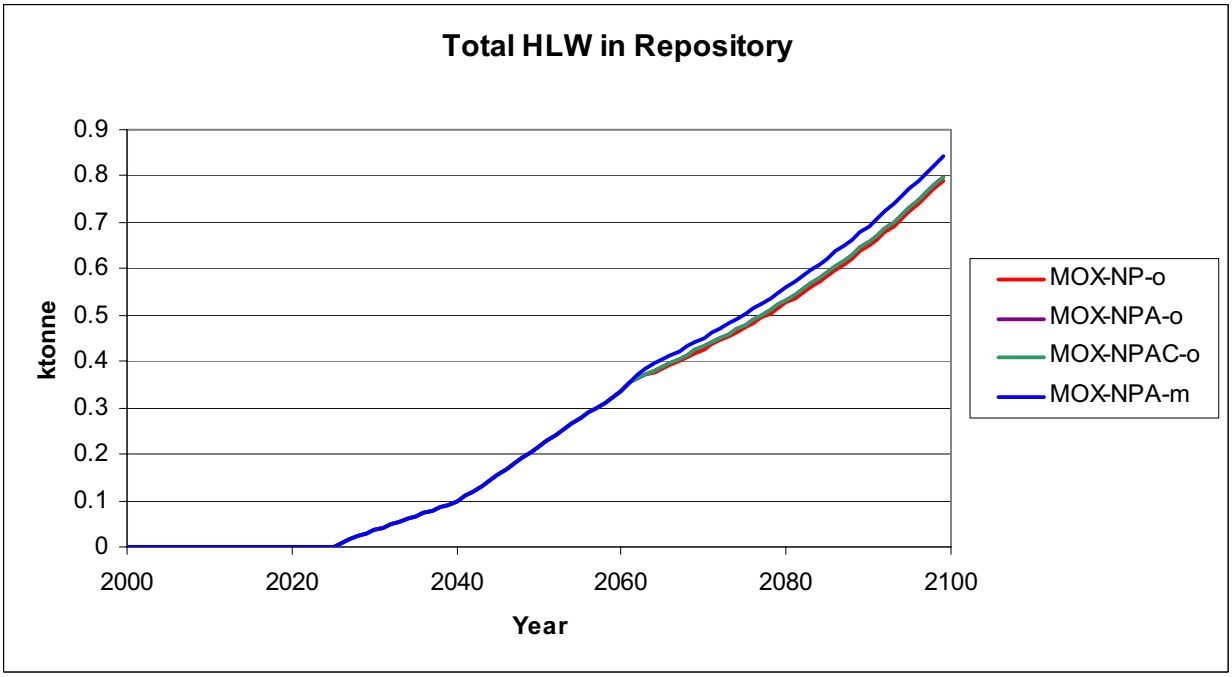

Figure 8.3-9b. High level waste mass from reprocessing and fuel fabrication in the repository for reactors containing MOX and UOX fuels. "o" indicates single-pass case and "m" indicates multi-pass case. The three single-pass cases ("o") fall on top of each other.

Uranium is the single largest element by mass in the system as illustrated by Figures 8.3- 10 and 11 for MOX or IMF fuels. Within a type of fuel, IMF or MOX, the specific recipe of fuel used and the number of passes for that fuel have little effect on the mass in the system. The IMF recipes result in a lower system mass than the MOX recipes by 2100.

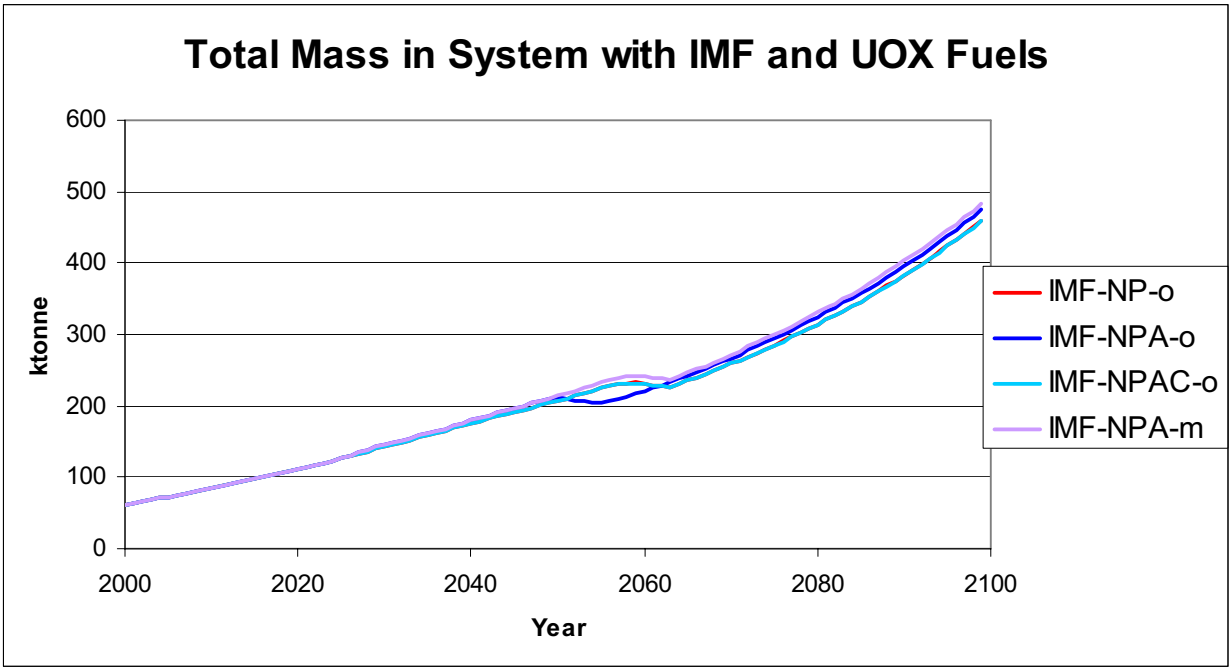

Figure 8.3-10a. Total mass in system with a combination of IMF and UOX fuels. The three single-pass lines ("o") fall on top of each other.

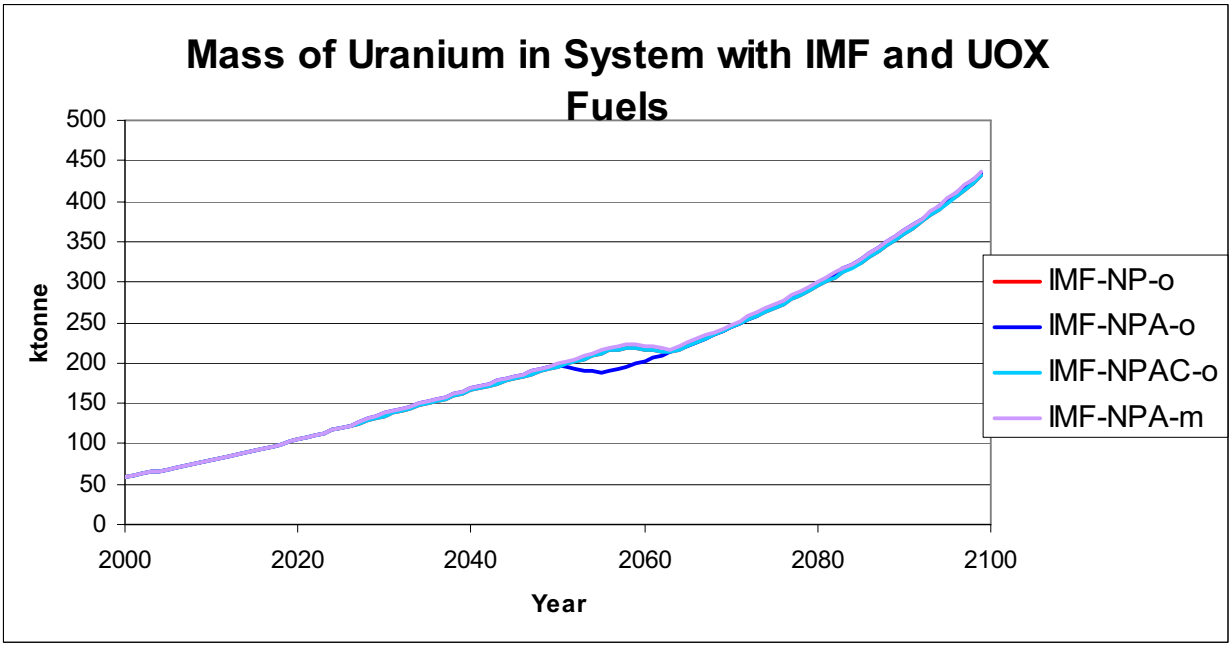

Figure 8.3-11a. Mass of uranium in system with a combination of IMF and UOX fuels. The three single-pass lines ("o") fall on top of each other.

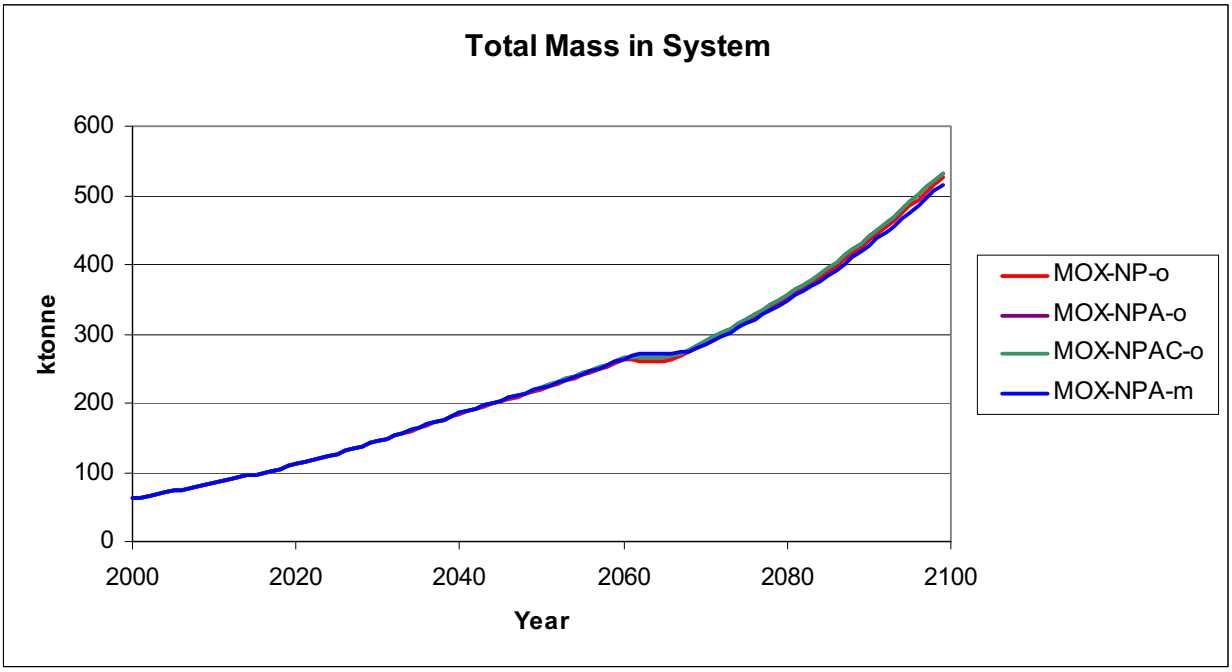

Figure 8.3-10b. Total mass in system with a combination of MOX and UOX fuels. The four lines basically fall on top of each other.

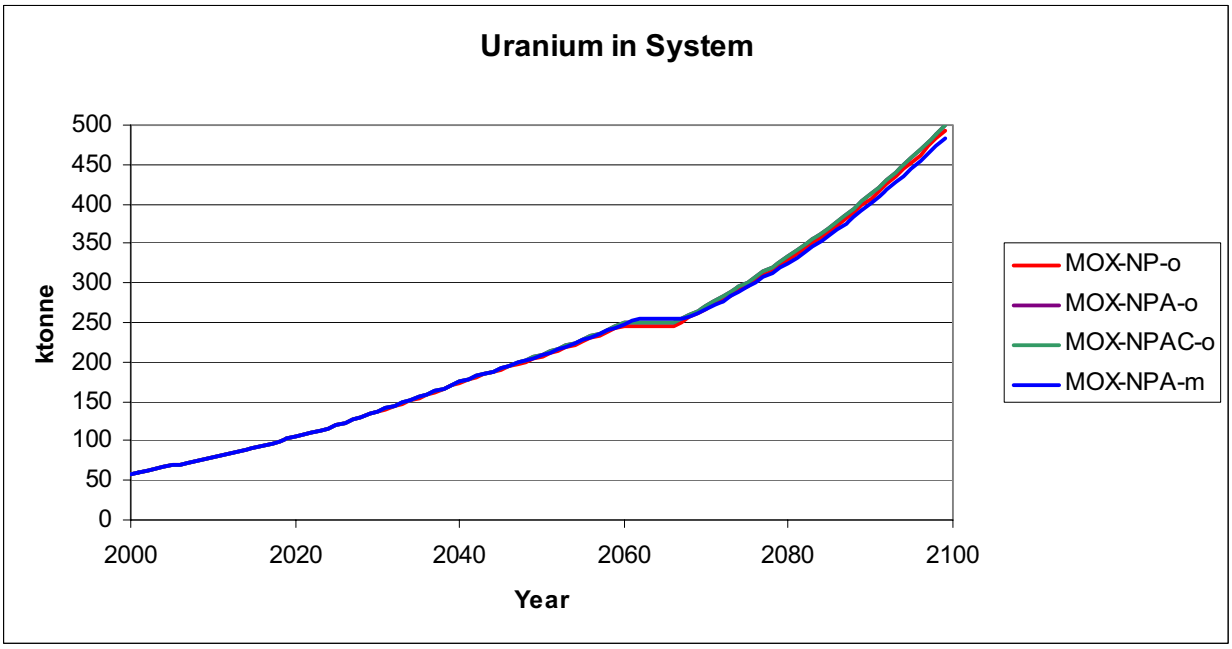

Figure 8.3-11b. Mass of uranium in system with a combination of MOX and UOX fuels. The four lines basically fall on top of each other.

The mass of plutonium in the system is dependent on the fuel recipe used and whether the fuel is single or multi-pass. For IMF (Figure 8.3-12), the blended fuel recipe brackets the full core recipes. Single-pass blended IMF results in the largest amount of plutonium in the system, while multi-pass blended IMF

results in the smalles amount of plutonium in the system. All of the MOX recipes are full core fuels. For MOX fuels, single-pass MOX-NPA or MOX-NPAC result in the highest amount of plutonium in the system, while multi-pass MOX-NPA results in the lowest plutonium in the system. Overall, IMF recipes resulted in less plutonium in the system than MOX recipes.

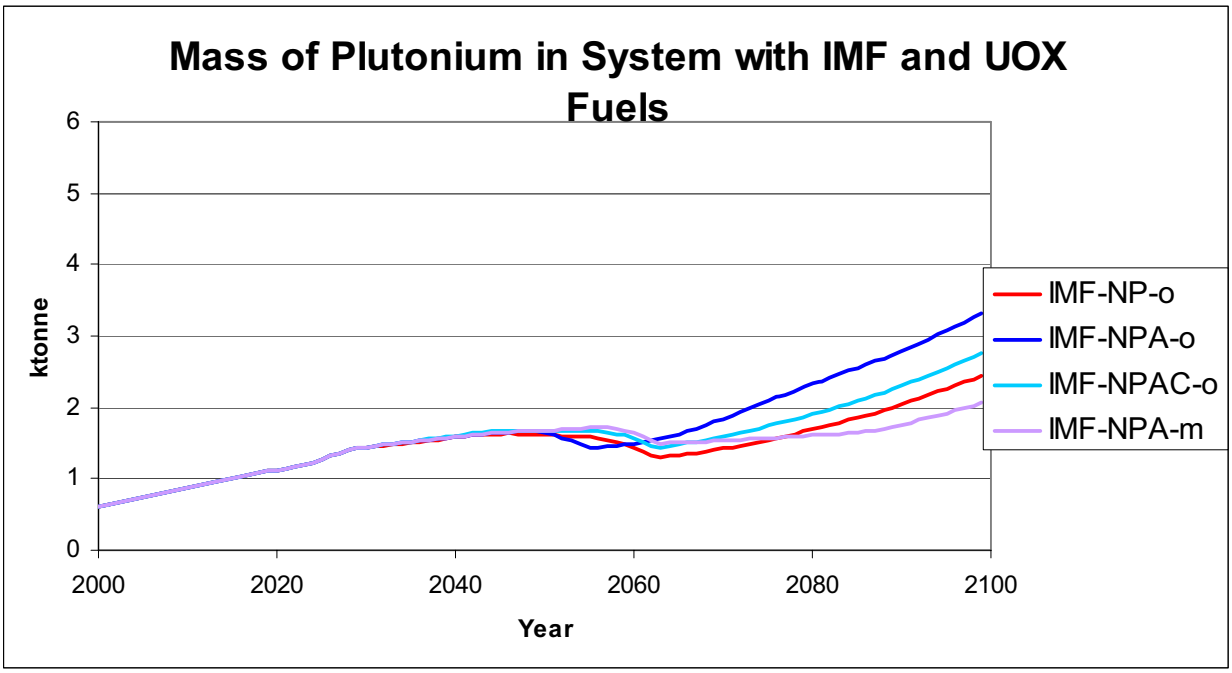

Figure 8.3-12. Mass of plutonium in system containing IMF and UOX fuels.

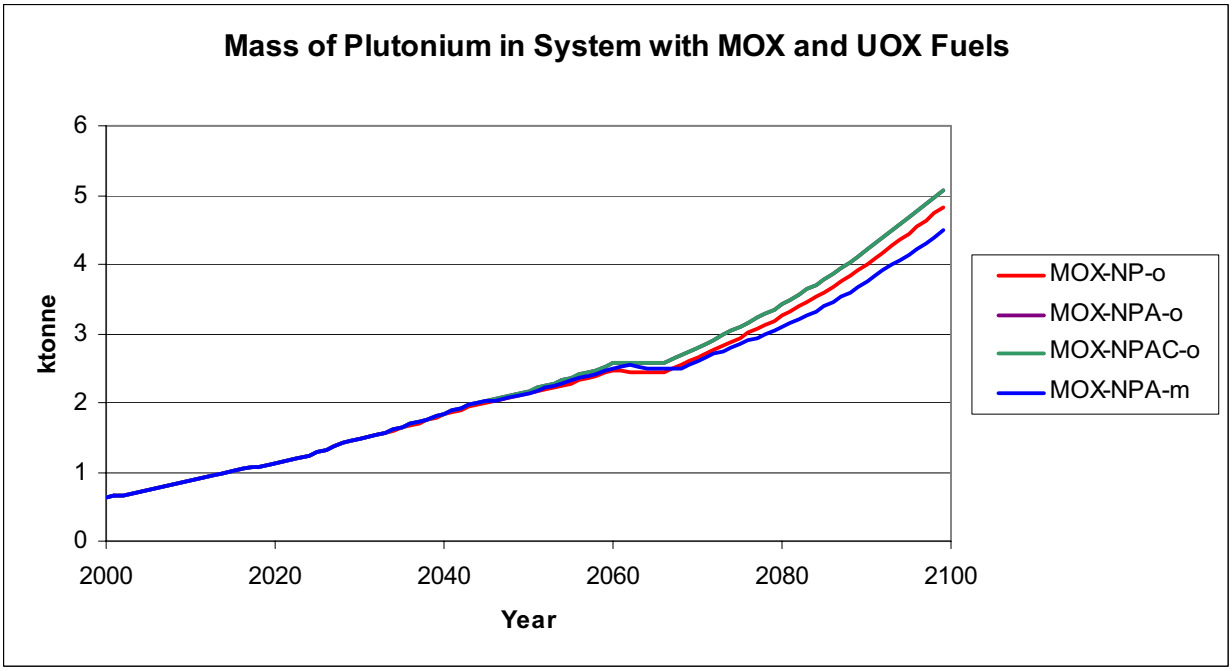

Figure 8.3-13. Mass of plutonium in system containing MOX and UOX fuels. The plot for MOX-NPAC-o overlays the plot for MOX-NPA-o.

Plutonium-239 (Figures 8.3-14 and 8.3-15) accounts for most of the plutonium in the system and shows the same trends and relative magnitudes as total plutonium.

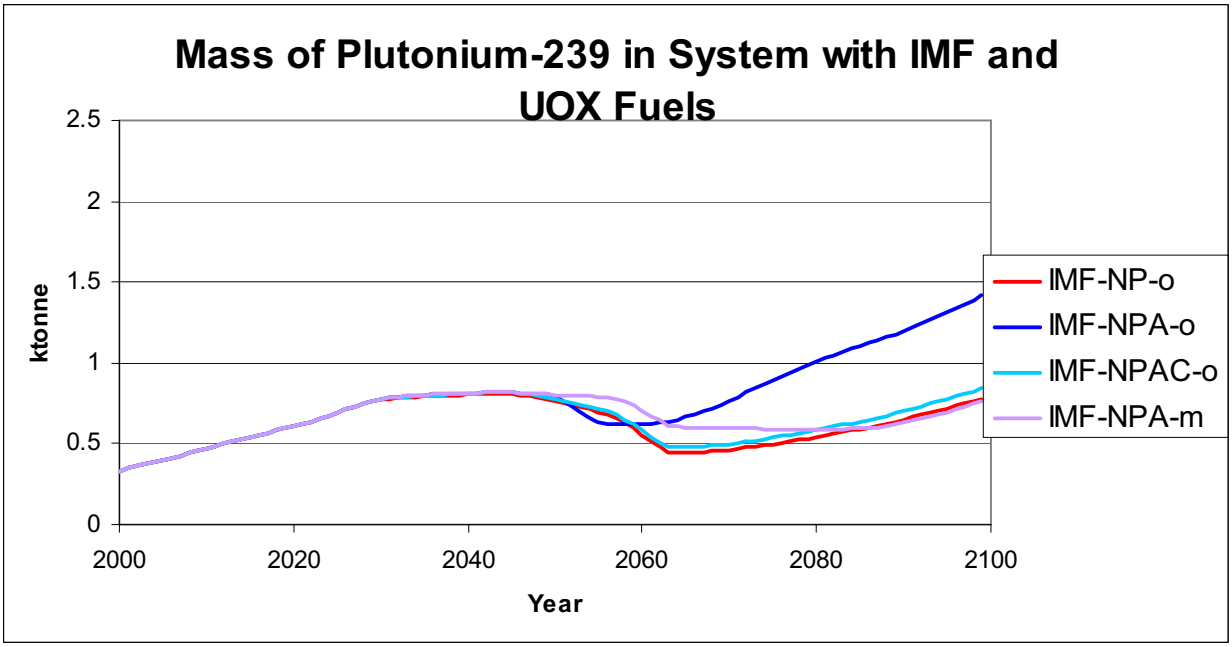

Figure 8.3-14. Mass of plutonium-239 in system containing IMF and UOX fuels.

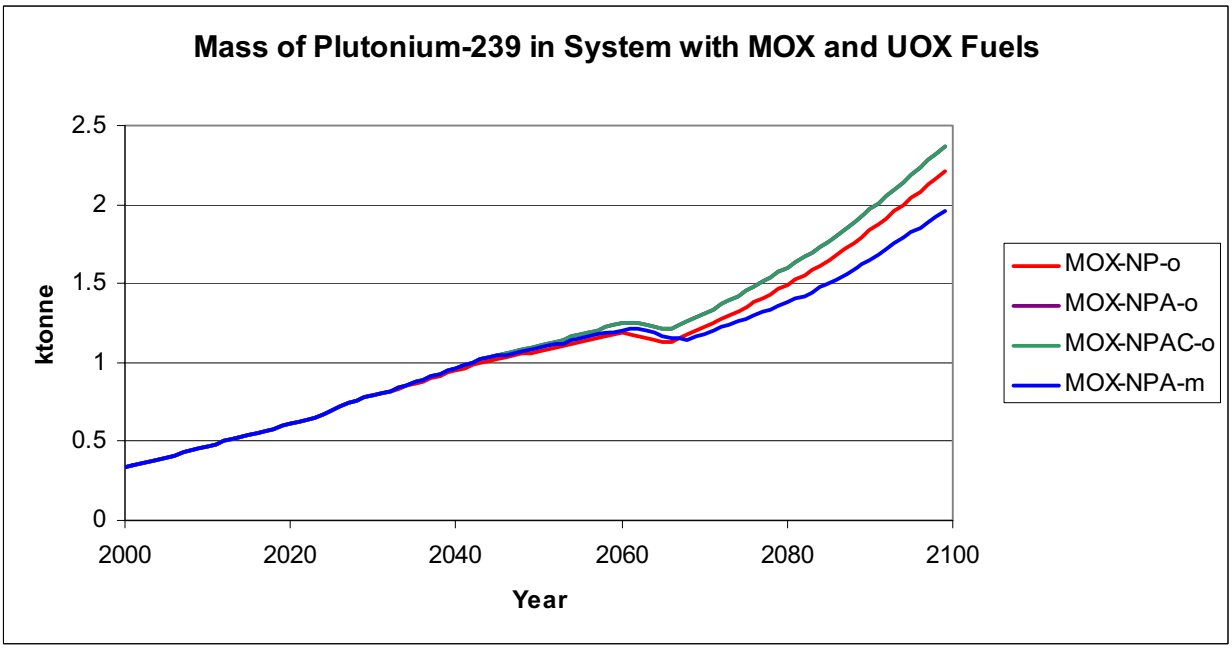

Figure 8.3-15. Mass of plutonium-239 in system containing MOX and UOX fuels. The plot for MOX-NPAC-o overlays the plot for MOX-NPA-o.

Plutonium-238 accounts for a significant but smaller fraction of the total plutonium in the system. For IMF, the single-pass IMF-NPAC fuel results in the largest amount of plutonium-238 by 2100 (Figure 8.3-

16). The multi-pass IMF-NPA fuel results in the smallest amount of plutonium-238, about half that produced by the single-pass IMF-NPAC recipe. For MOX, the multi-pass MOX-NPA fuel results in the highest amount of plutonium-238 in the system by 2100. The MOX-NP single-pass fuel results in the lowest amount of plutonium-238 in the system by 2100.

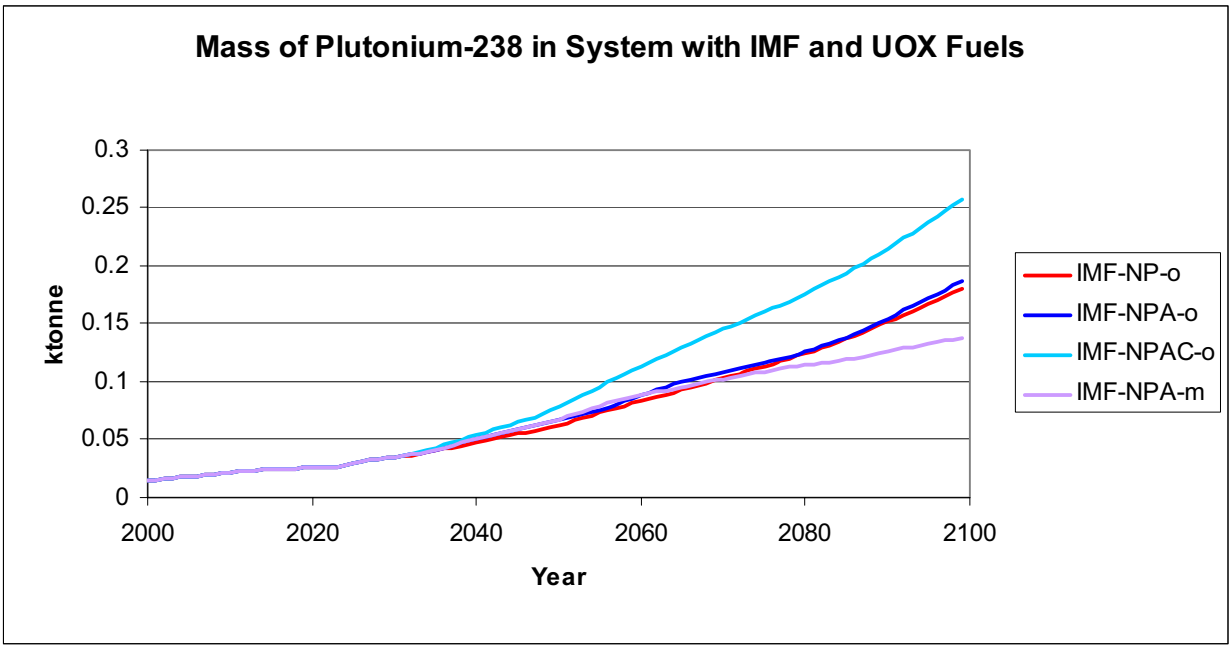

Figure 8.3-16. Mass of plutonium-238 in system containing IMF and UOX fuels

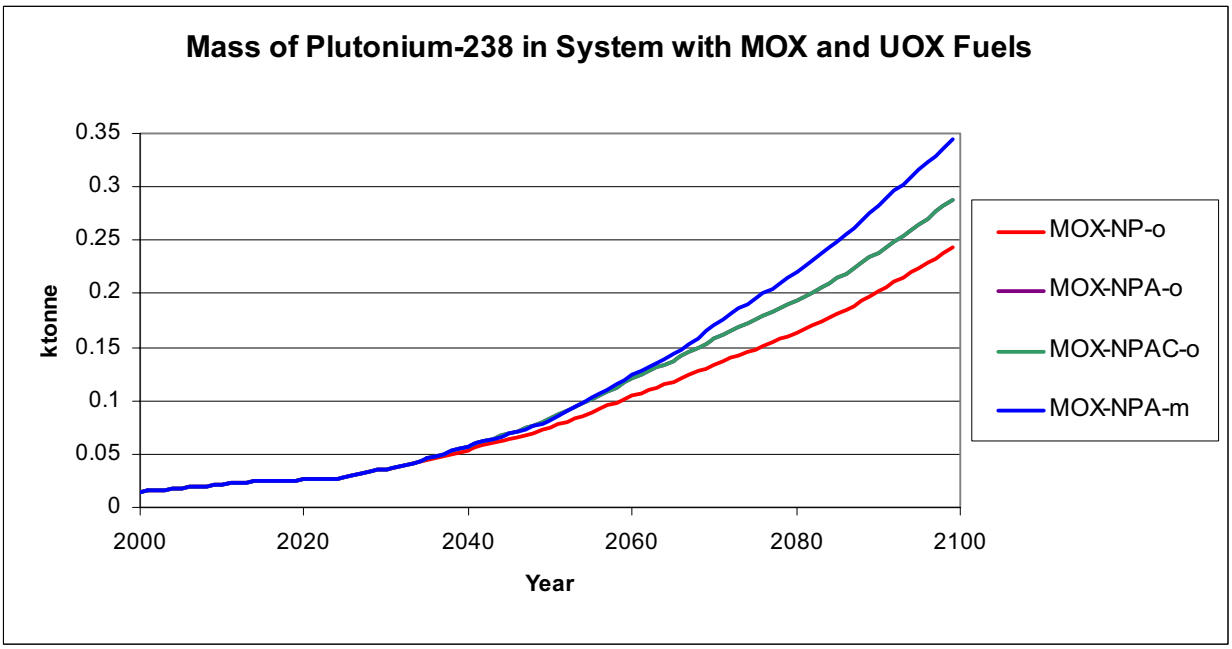

Figure 8.3-17. Mass of plutonium-238 in system containing MOX and UOX fuels. The plot for MOX-NPAC-o overlays the plot for MOX-NPA-o.

The plots of weapons usable material for the MOX (Figure 8.3-18) and IMF (Figure 8.3-19) fuels have the same shapes and relative positions and similar magnitudes to the plots of total plutonium for the same recipes. This suggests that plutonium drives the weapons usable content of material in the system. IMF recipes result in lower weapons usable quantities than MOX recipes and of the IMF recipes, multi-pass IMF-NPA results in the lowest weapons usable mass in the system.

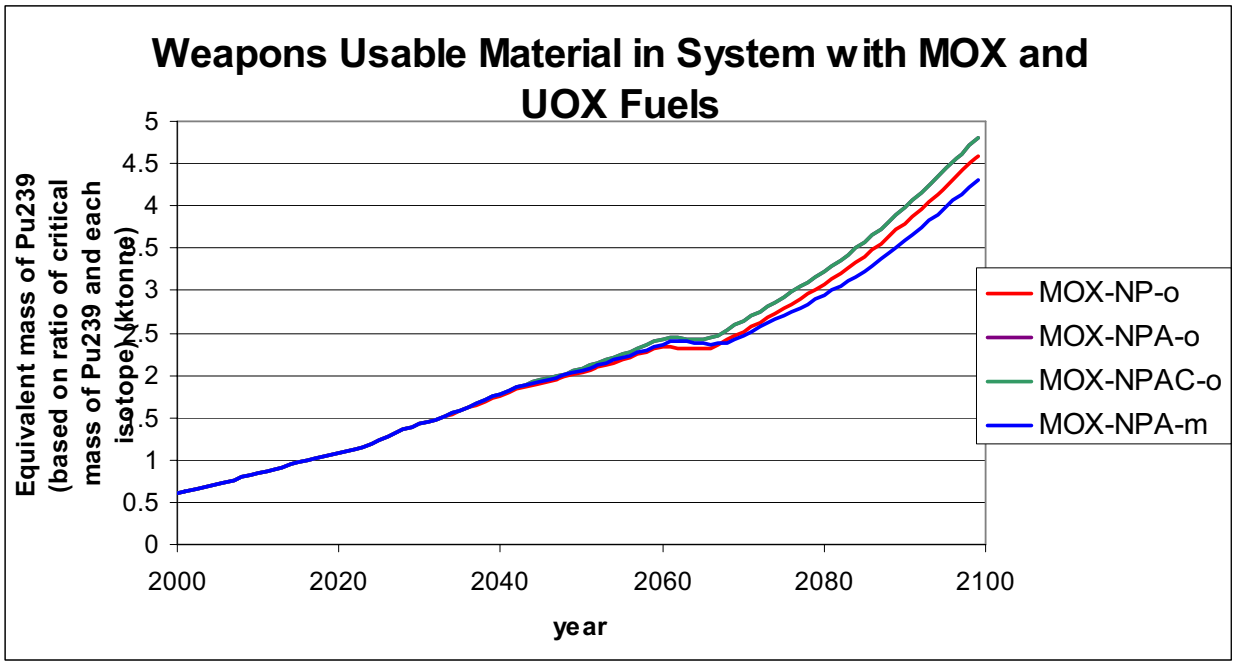

Figure 8.3-18. Weapons usable material in system containing MOX and UOX fuels. The plot for MOX-NPAC-o overlays the plot for MOX-NPA-o.

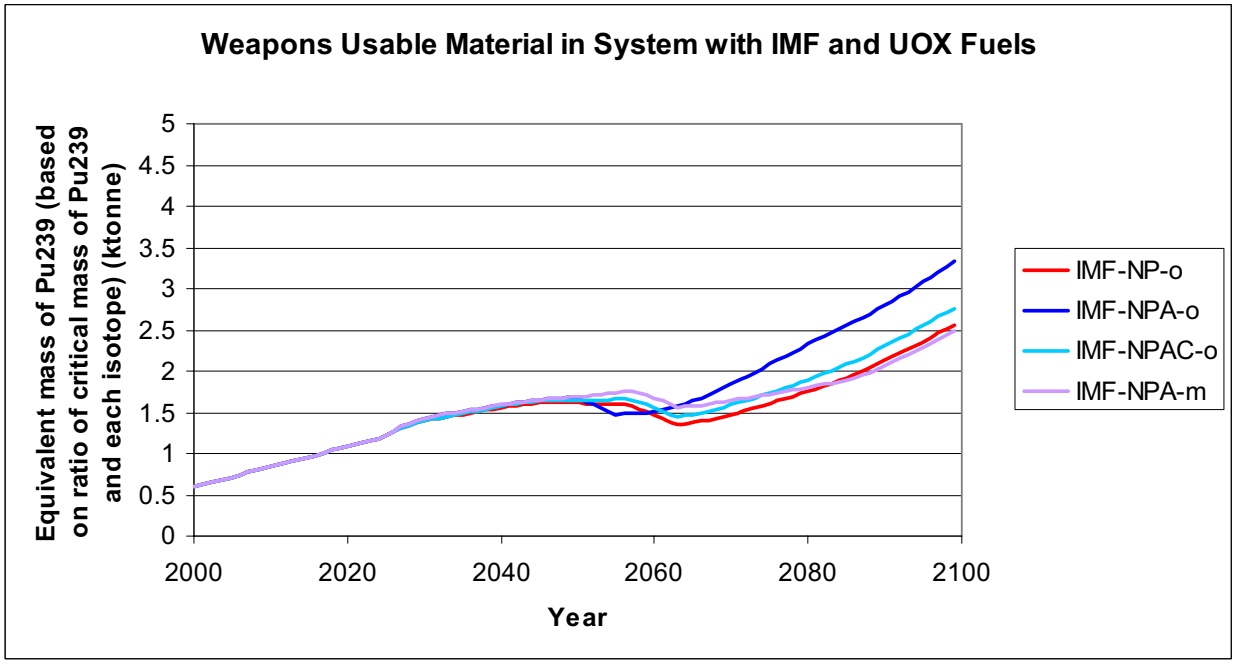

Figure 8.3-19. Weapons usable material in system containing IMF and UOX fuels.

Long-term heat in the repository generally tracks with the mass in the repository. For IMF (Figure 8.3.-20), the multi-pass IMF-NPA recipe results in the lowest long-term heat at 2100.

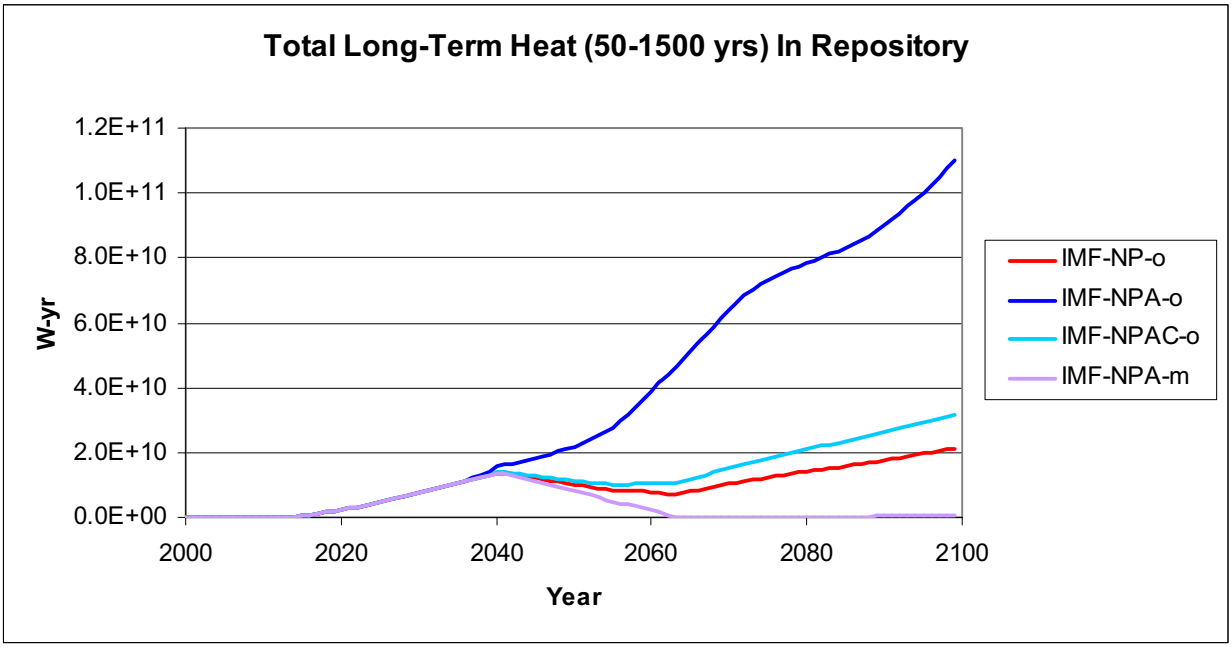

Figure 8.3-20. Total long-term heat (50-1500 years) in repository from a combination of IMF and UOX fuels in the system.

For MOX (Figure 8.3-21), the multi-pass MOX-NPA recipe results in the lowest long-term heat at 2100. For the single-pass MOX recipes, MOX-NP results in a lower long-term heat than MOX-NPA or MOX-NPAC eventhough the mass of MOX-NP is lower than that of the other two recipes. This suggests that americium and curium make a significant contribution to the long-term heat of the material. There is no significant difference in long-term heat between the multi-pass IMF and the multi-pass MOX.

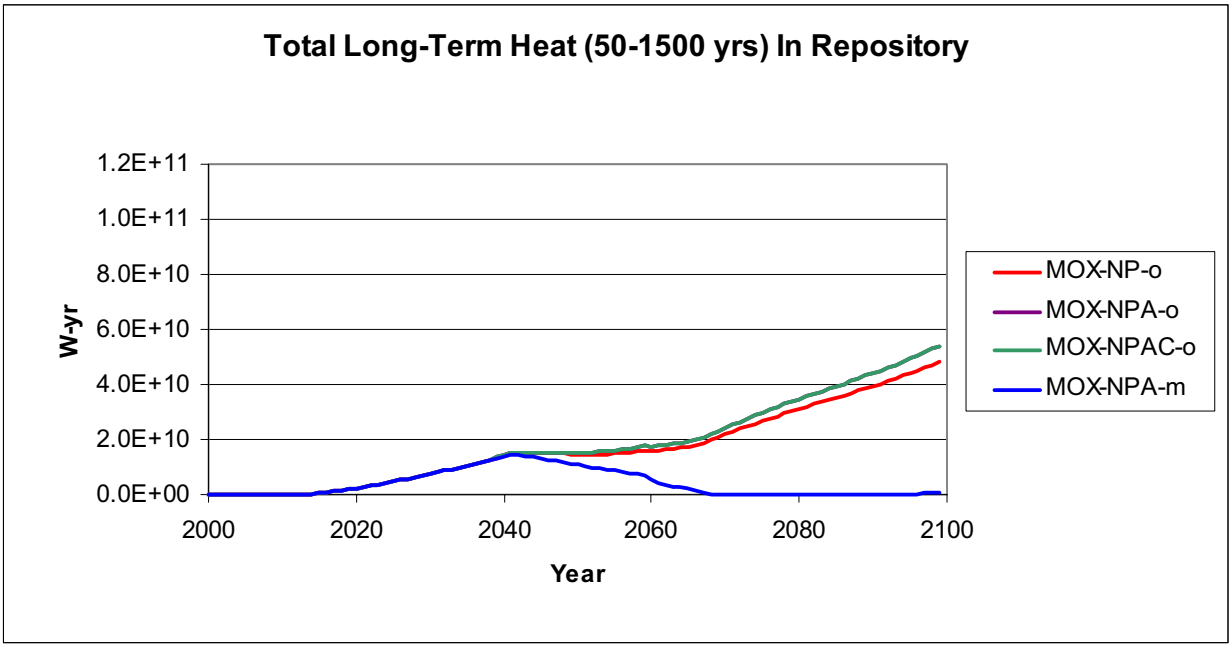

Figure 8.3-21. Total long-term heat (50-1500 years) in repository from a combination of MOX and UOX fuels in the system. The plot for MOX-NPAC-o overlays the plot for MOX-NPA-o.

The dose at 500,000 years from material in the repository generally tracks with the mass in the repository. For IMF (Figure 8.3-22), the multi-pass IMF-NPA recipe results in the lowest dose.

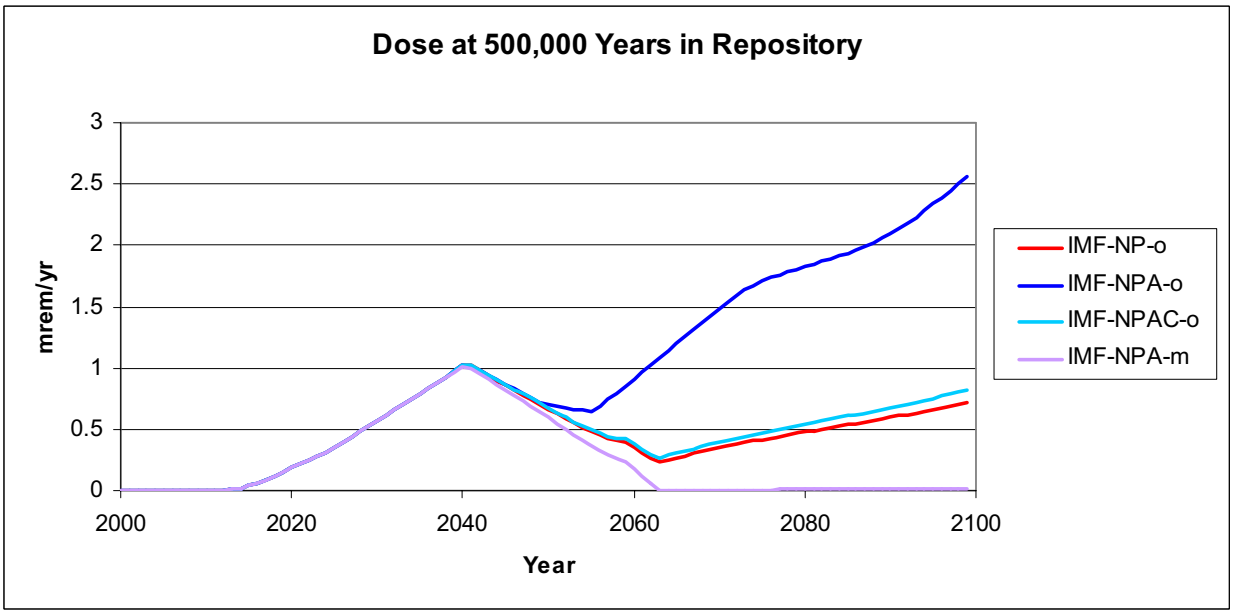

Figure 8.3-22. Dose at 500,000 years from material in repository for combinations of IMF and MOX fuels in the system.

For MOX (Figure 8.3-23), the multi-pass MOX-NPA recipe results in the lowest dose. For the single-pass MOX recipes, MOX-NP, MOX-NPA and MOX-NPAC result in the same dose, eventhough the

masses of MOX-NPAC and MOX-NPA are lower than that of MOX-NP. This suggests that americium and curium make a significant contribution to the dose from the material. There is no significant difference in dose between the multi-pass IMF and the multi-pass MOX.

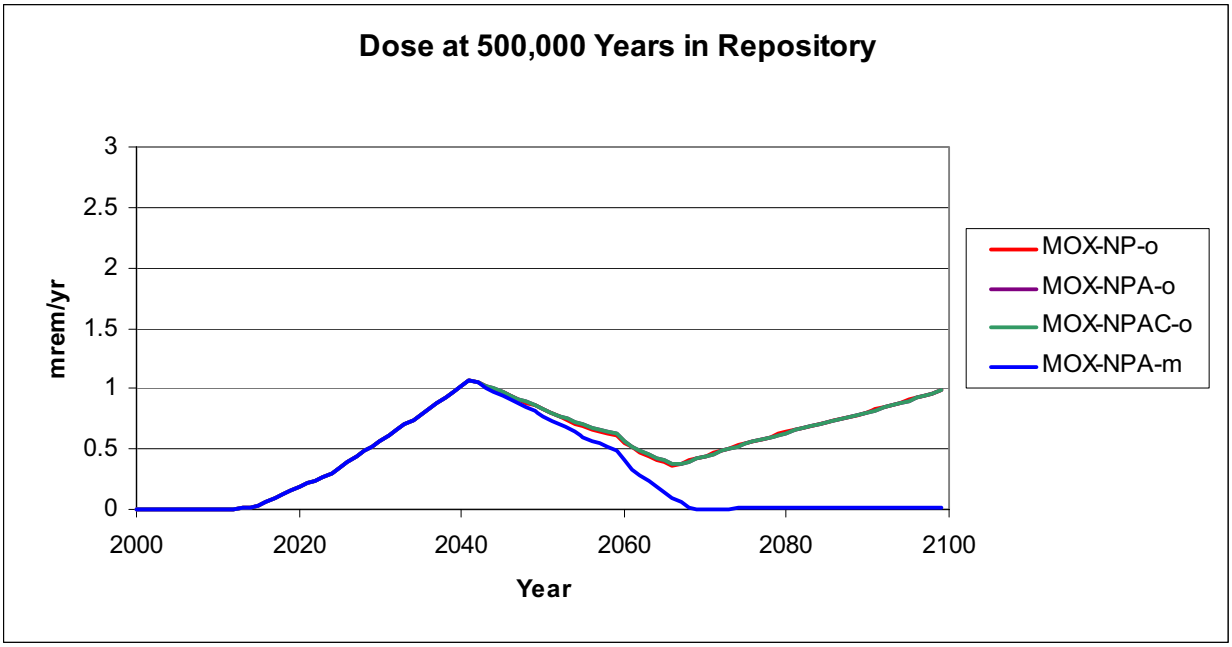

Figure 8.3-23. Dose at 500,000 years from material in repository for combinations of MOX and MOX fuels in the system. The plot for MOX-NPA-o and MOX-NP-o is overlain by the plot for MOX-NPAC-o.

8.4. Transitioning to, and management of, fast reactors

This section presents a set of practical scenarios that serve as examples of the complexity of the real-world scenarios focusing on transitioning to consumer fast reactor (CFR) systems. The current scenarios focus on improving permanent disposal utilization and/or reducing the temporary spent fuel storage inventory.

8.4.1 General assumptions and timelines

The basic assumptions of the scenarios, related to the existing U.S. reactors park and the future reactor systems, are the same as assumptions presented in previous sections except as follows. First, in addition to the baseline growth case (1.8% per year), there are also cases at 0% growth and 3.2% growth. Second, analyses assumed that ultra-high burnup fuels (e.g., 100 GW-day/tonne) replace existing fuel. Third, the deployment of reprocessing capacity is roughly half of the other analyses, corresponding to the ultra-high burnup. Fourth, all TRU from UOX SNF is transmuted by a consumer fast reactor (CFR). Fifth, deployment of CFRs is limited to about 1.6 GWe/yr (correspond to 5 CFRs of about 3.2 GWe each), beyond 2030.

The timeline for these scenarios is as follows:

- Starting 2010, demand growth at different rates (*0, 1.8, 3.2%).*
- Starting 2015, u*se ultra-high burnup, 100 GW-day/tonne fuel in all reactors*
- Starting 2025, SNF *reprocessing starts using a* first commercial plant (800 tonne/yr) starts in 2025 followed by an upgrade to 2,000 tonne/yr in 2035 and 3,000 tonne/yr total capacities in 2055.
- *FR deployment starts with a first of a kind plant (*FOAK) FR , followed by full deployment of FRs 5 years later, at a maximum rate of 1.6 GWe/yr (5 FR burners/yr)
- Starting 2028, *replace retiring LWRs with FRs to meet new energy demand if possible. If there is not enough TRU for FRs, build new ALWRs*

8.4.2 Scenario results

The base scenario is 1.8% growth rate with implementation of high burnup fuel starting in 2015. Figure 8.4-1 shows the deployment of both LWR and FR capacities according to this growth rate, where the CFR contribution to the total energy generation is as high as about 18%. The limited reprocessing capacity of LWR spent fuel, shown in Figure 8.4-2, does not limit the deployment of FR systems. The limitation here on the deployment of FRs is instead caused by the constraint of maximum deployment rate of 1.6 GWe/yr FR capacity per year, which is imposed to limit the number of FR burners to be deployed per year, as shown in Figure 8.4-3. The fast reactor percent of total capacity increases gradually to about 18%, and a significant decline starts 2090 because of the retirement of FRs built in 2030, while the TRU inventory is not large enough to make up for those reactors and also respond to increase in demand. However, this can be avoided by increasing the reprocessing capacity a few years earlier, or deployment of breeder reactors, to avoid the eventual shortages in transuranics.

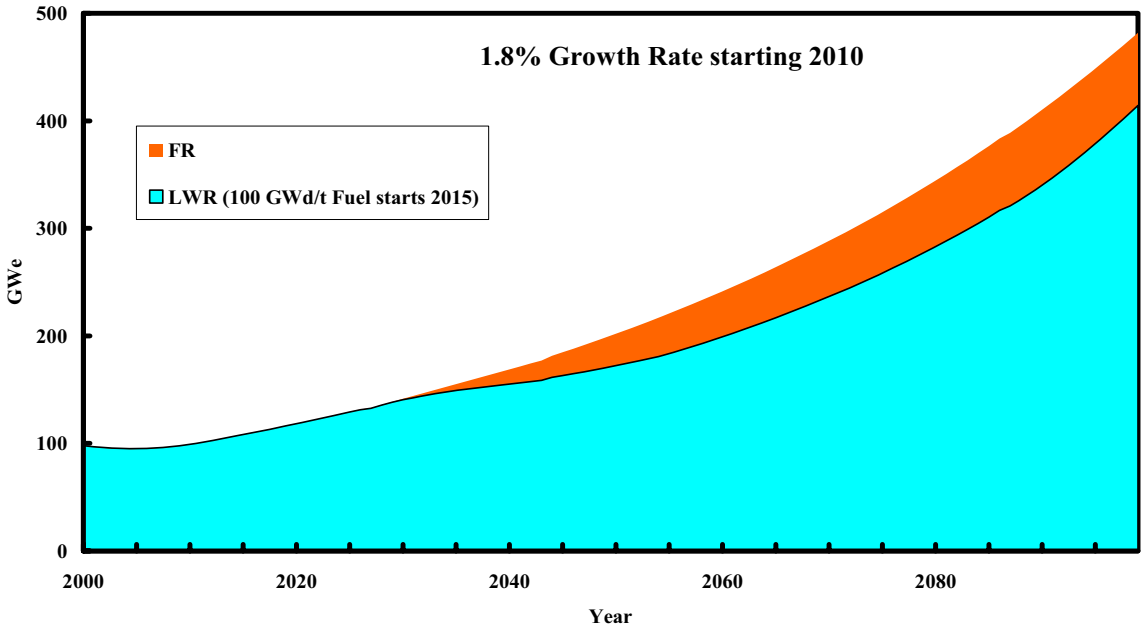

Figure 8.4-1. Thermal and fast reactor installed capacity for 1.8% growth

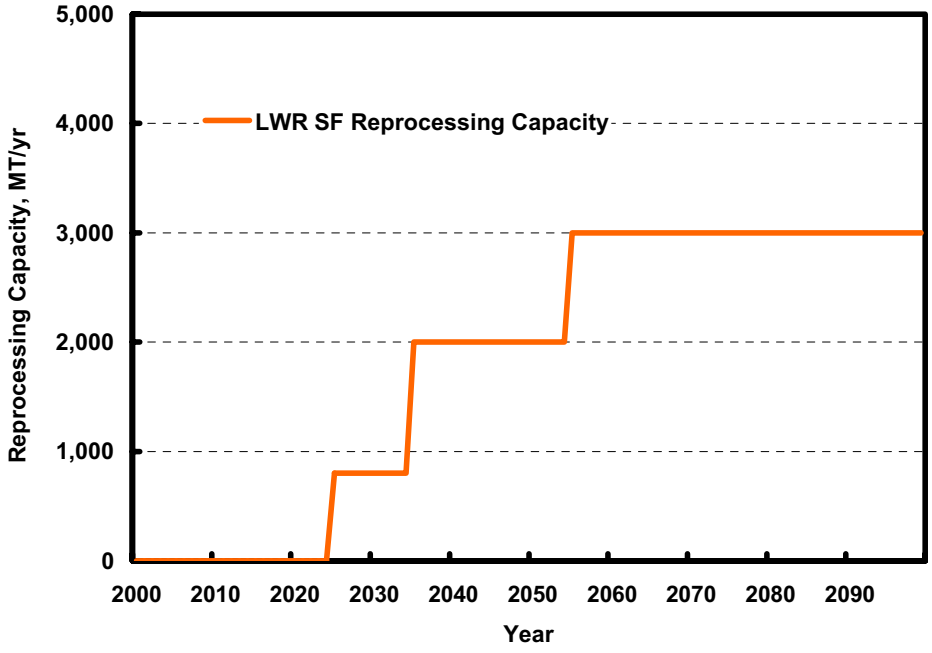

Figure 8.4-2. LWR SNF reprocessing capacity for 1.8% growth

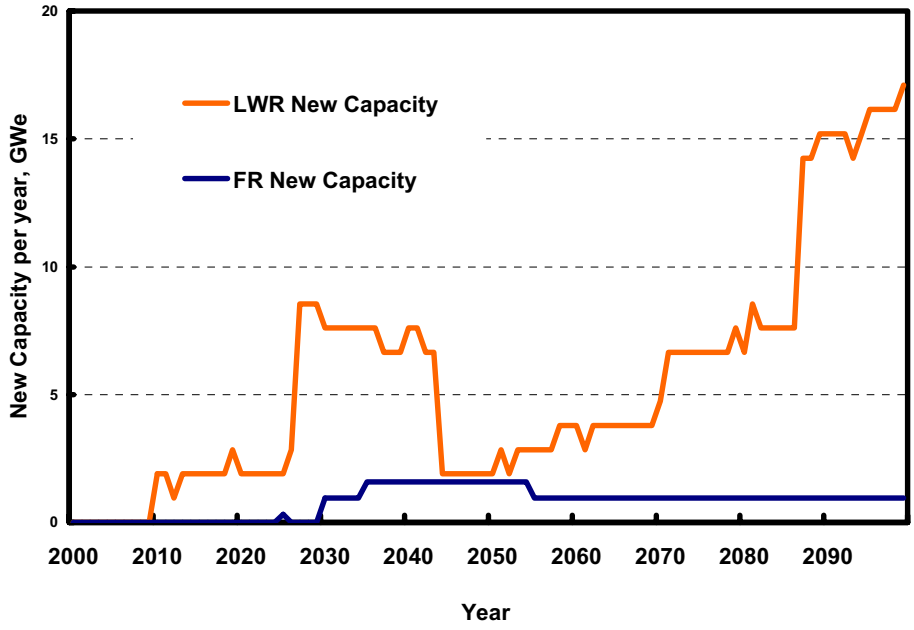

Figure 8.4-3. LWR and CFR capacity additions for 1.8% growth

The key scenario results are shown in Figures 8.4-4 and 8.4-5. As shown in Figure 8.4-4, SNF in temporary storage requirements are minimized. With reprocessing and transfer of SNF to repository, storage temporary requirement decline, and by about 2030, storage requirements are less than the storage requirements in 2000. Eventually storage requirements starts to increase after a 2043 minimum. Direct disposal of large amounts of SNF in repository is realized in this scenario. By 2028 all 2000 legacy SNF is transferred to repository, and by 2043, all SF production goes to reprocessing, and no more transfer of SNF to repository takes place until ~ 2088 when SF available exceeds the reprocessing needs. SNF in repository reach ~ 94,000 tonnes by 2043 (including military & DOE 7000 tonnes). The LWR spent fuel inventory in temporary storage, Figure 8.4-4, is gradually reduced to less than the year 2000 inventory by the end of the century. In addition, reprocessing capacity has been sized (Fig. 8.4-2) such that the available unused Pu inventory (from reprocessed spent fuel) as shown in Figure 8.4-5, is capped at less than the current worldwide inventory of 150 tonnes. Uranium utilization improves with this scenario where uranium consumption has decreased by about 14% by the year 2100. The CFR fraction of the total energy park reaches about 18% by the end of the century.

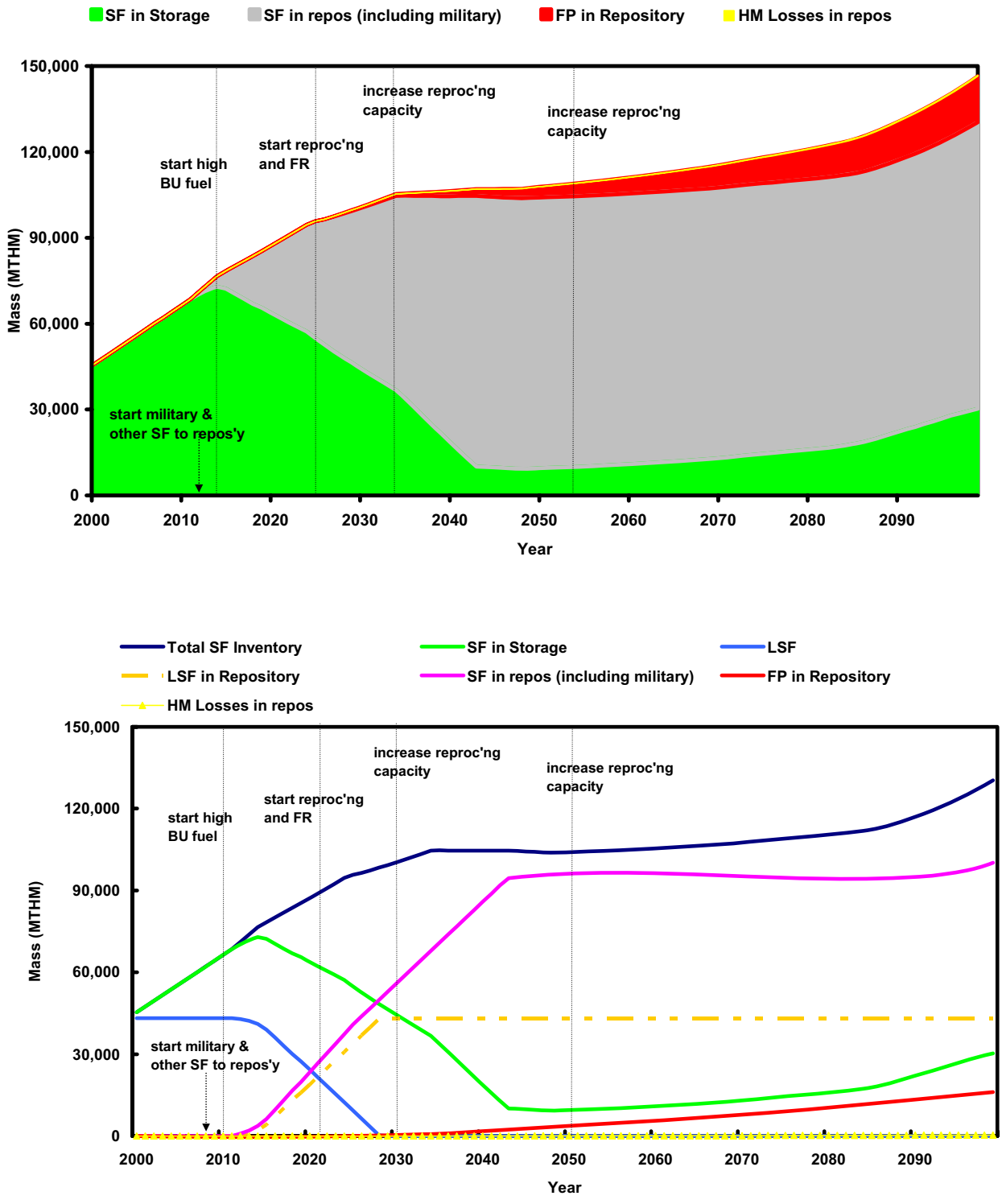

Figure 8.4-4. Dynamics of waste accumulation in both temporary and permanent storage for 1.8% growth

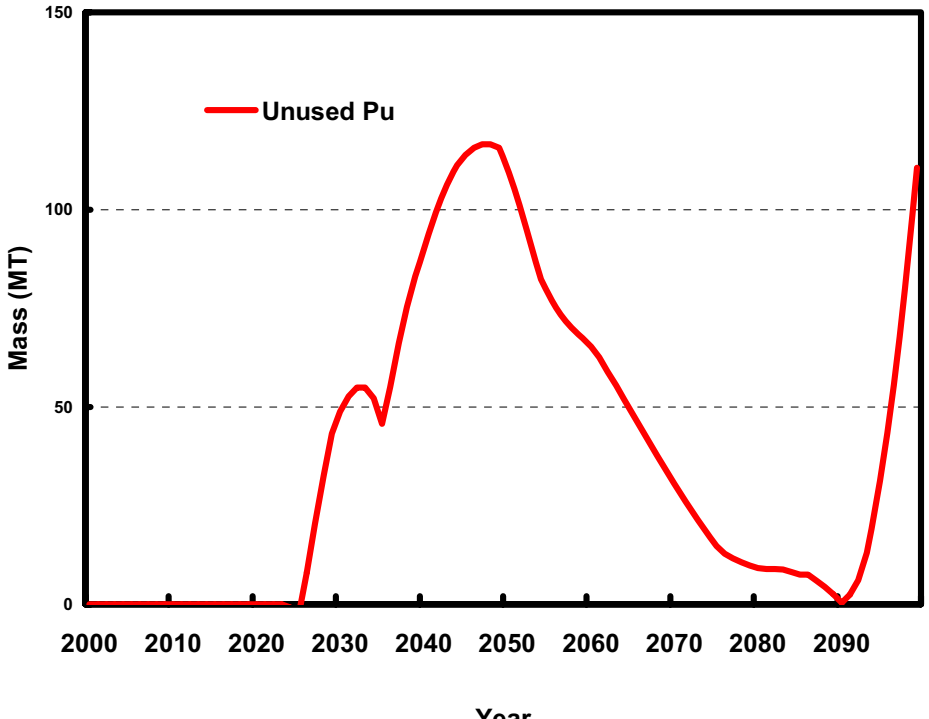

Figure 8.4-5. Pu from reprocessed SNF but not used (1.8% growth)

Figures 8.4-6 through 10 show scenarios results for the 0% growth rate. This limitation on growth of nuclear energy constrains the deployment of CFRs and limit it to the period between 2028 and 2043 to replace retiring reactors, until the next wave of retirement of LWRs in 2087, which are ALWRs that were built starting 2027. By 2043, the percent of fast reactors reaches about 22.5%, and remains constant until 2087, when ALWRs start to retire.

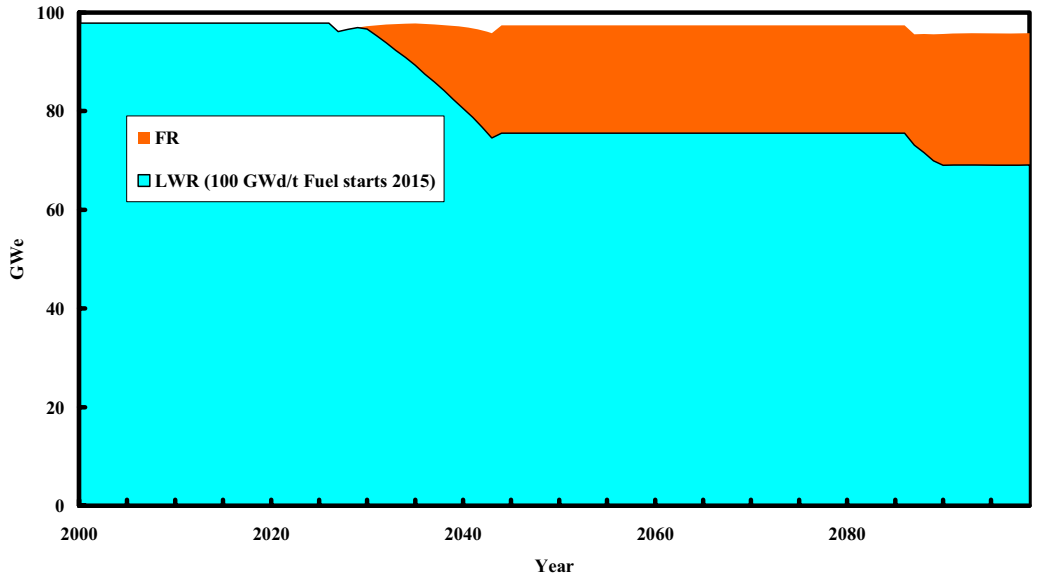

Figure 8.4-6. Thermal and fast reactor installed capacity for 0% growth

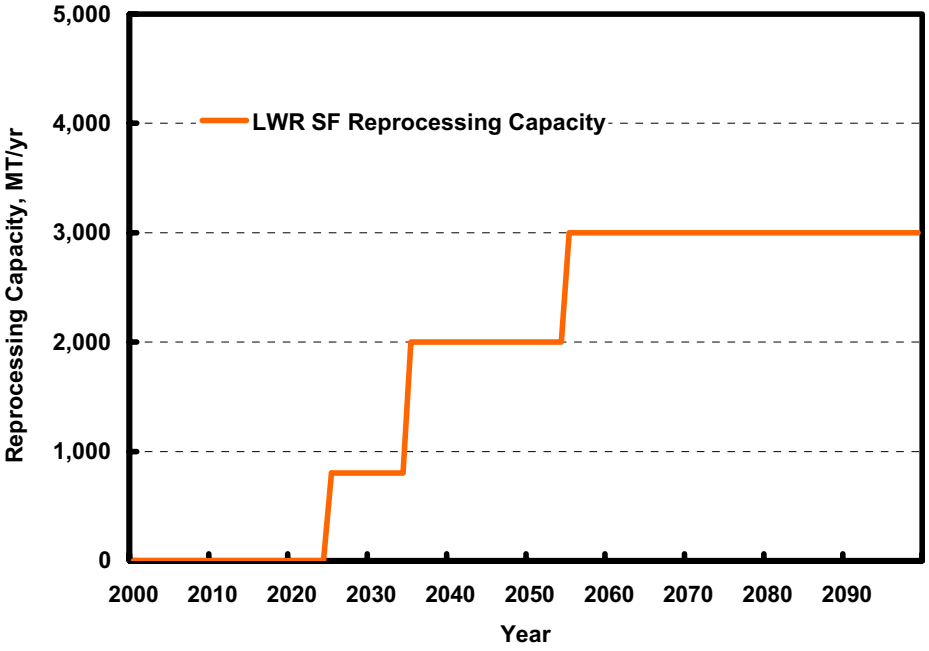

Figure 8.4-7. LWR SNF Reprocessing Capacity for 0% Growth Rate

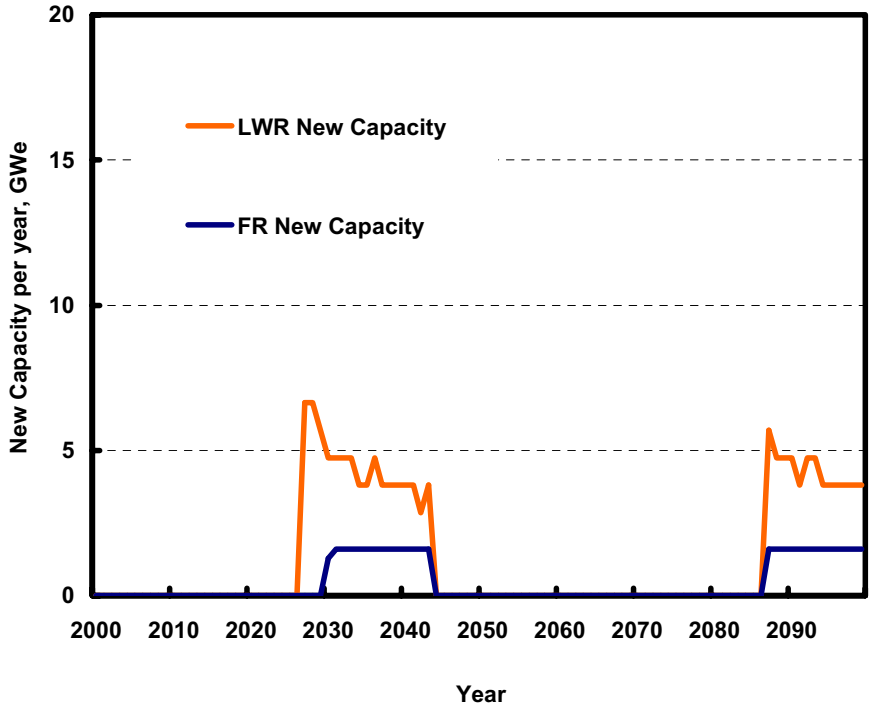

Figure 8.4-8. LWR and CFR capacity additions for 0% growth

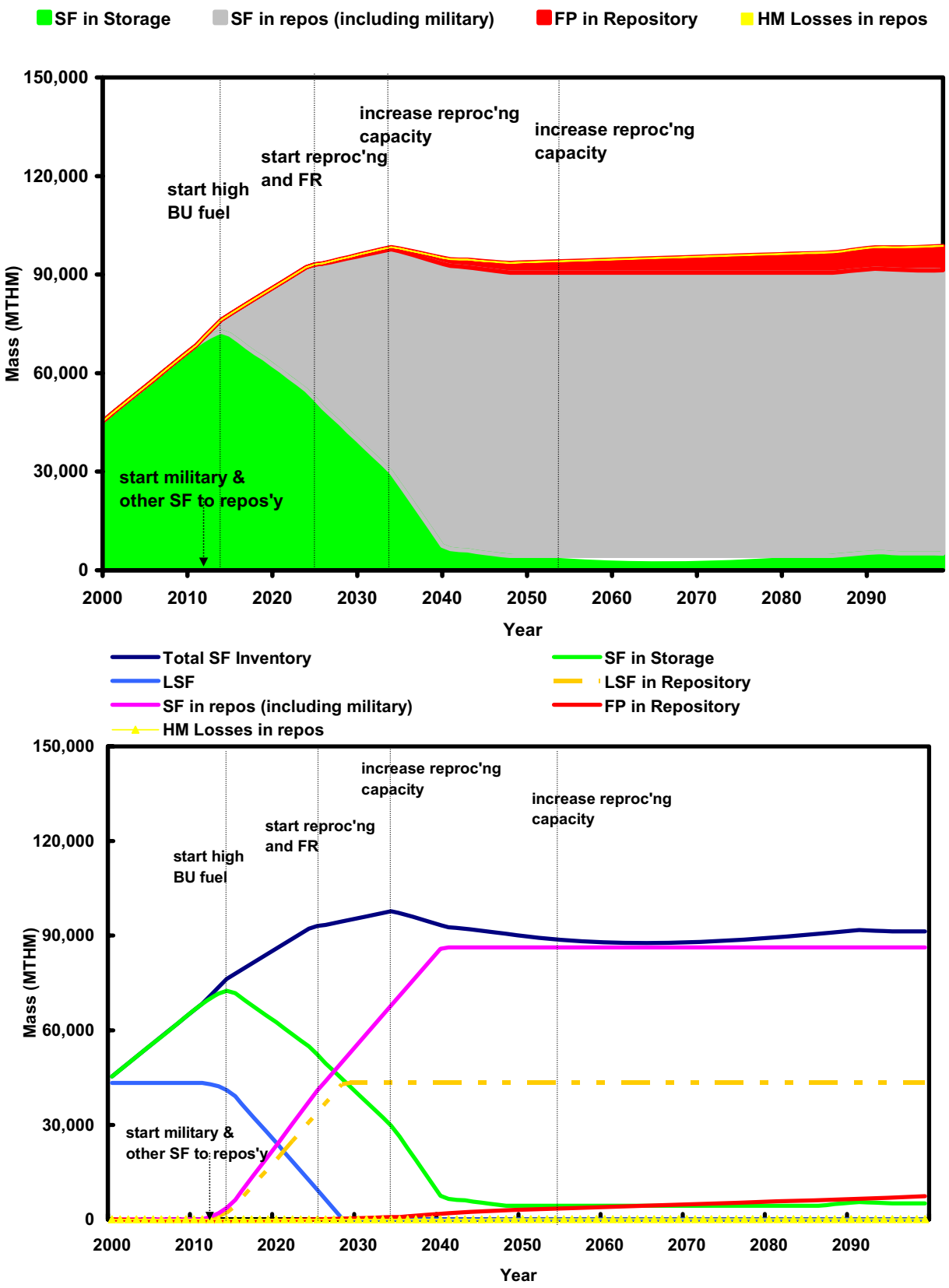

Figure 8.4-9. Dynamics of waste accumulation in both temporary and permanent storage for 0% growth

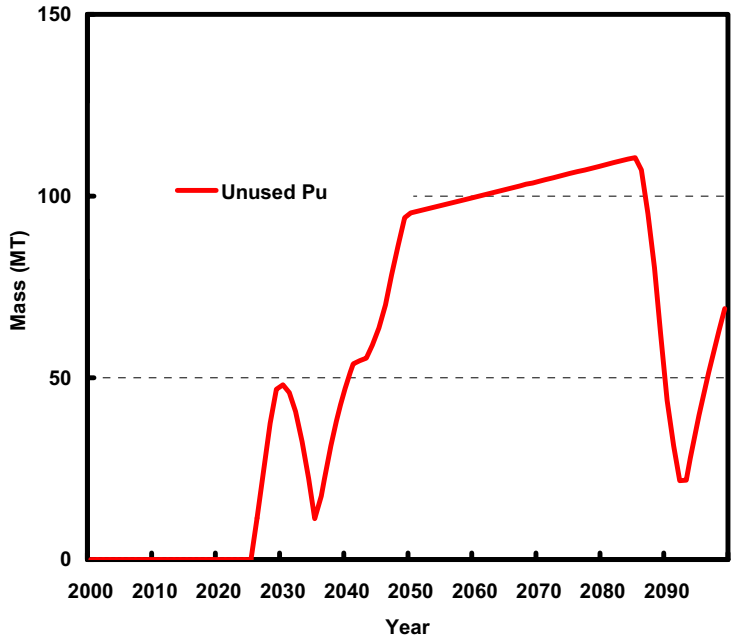

Figure 8.4-10. Pu from reprocessed SNF but not used (0% growth)

Those ALWRs retired in 2087 are replaced by CFRs, which increase the CFR% in capacity. Increase in CFRs starting 2087 leads to CFR% in capacity of about 28% by 2090. In this scenario, the SNF temporary storage requirements are also minimal. With reprocessing and transfer of SNF to repository storage requirements decline, and by about 2028, storage requirements are less than the storage requirements in 2000. Direct disposal of large amounts of SNF in repository is also realized here. By 2028 all the year 2000 legacy SNF is transferred to repository. By 2041, all SNF production goes to reprocessing, and no more SNF transfer to repository to the year 2100. SNF in repository reach ~ 86,000 tonnes by 2041 (including military & DOE 7000 tonnes). Again, inventory of Pu (from reprocessed SNF) at any point in time remain less than the current worldwide inventory of Pu of 150 tonnes.

The case of 3.2% growth is similar to the 1.8% growth rate case as shown in Figures 8.4-11 to 15. The reprocessing capacities, however, are different, where larger reprocessing capacities are needed to accommodate the increase in spent fuel production in this case. Until the year 2055, reprocessing capacity is assumed to be the same as the previous scenarios, and beyond 2055 it is increased rapidly to catch up with the high SNF production rate as shown in Figure 8.4-12, where 2000 tonnes-iHM/yr capacity is added every 4 years until the year 2087. Buildup of FRs/year is allowed to go up gradually from 1.5 GWe/year in 2055 to about 7.3 GWe by 2095, as shown in Figure 8.4-13. FR% reach about 14% (lower than the 1.8% growth rate because of the faster growth rate and the lack of enough TRU to build CFR fast enough to respond to increased demand). With reprocessing, and transfer of spent fuel to repository, temporary storage requirements decline, and by about 2035, storage requirements are less than the storage requirements in 2000. Eventually, storage requirements start to increase after a minimum in 2045. Direct disposal of large amounts of SNF in repository is also achieved. By 2028, all of the year-2000 legacy SNF is transferred to repository, and by 2062, all SNF production goes to reprocessing and no more transfer to repository to 2100. The SNF in repository reaches ~ 118,000 tonnes by 2062 (including military & DOE 7000 tonnes). Again, the Inventory of Pu (from reprocessed SNF) at any point in time remains below 150 tonnes.

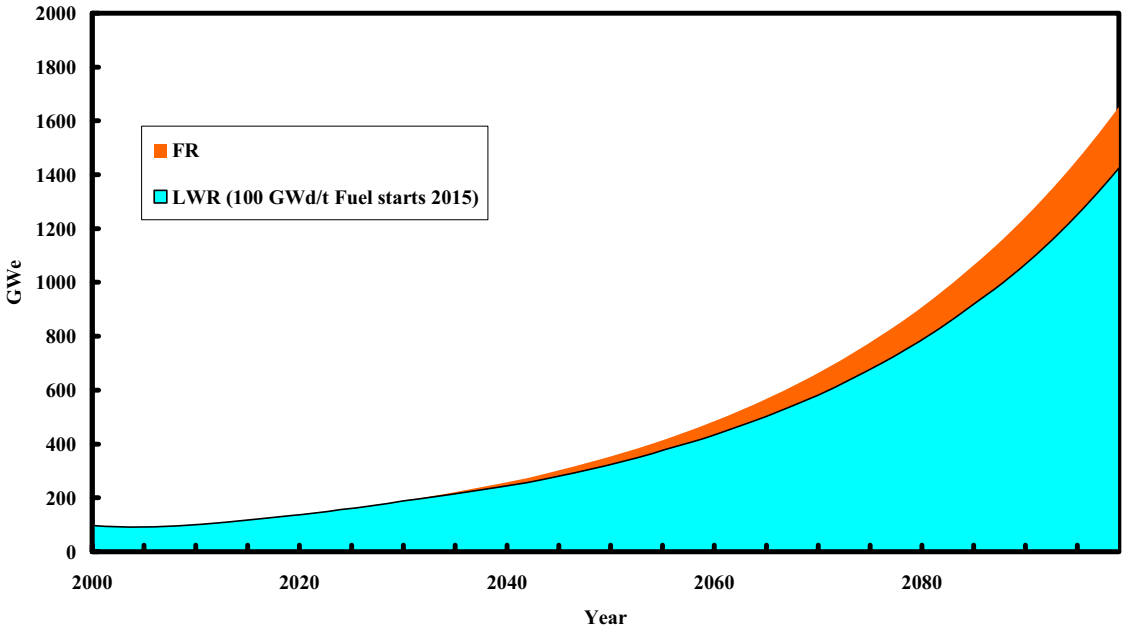

Figure 8.4-11. Thermal and fast reactor installed capacity for 3.2% growth

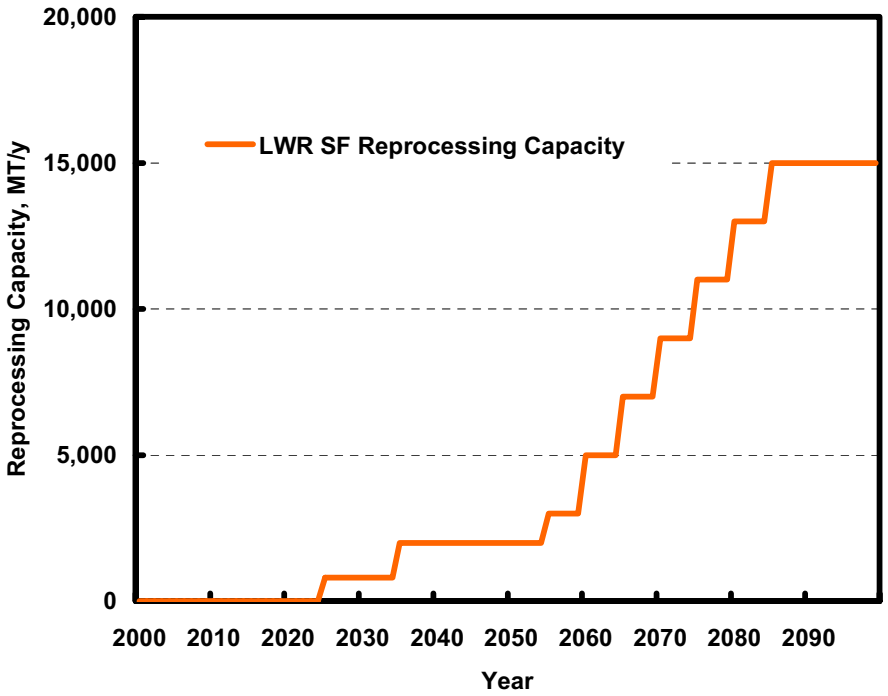

Figure 8.4-12. LWR SNF reprocessing capacity for 3.2% growth

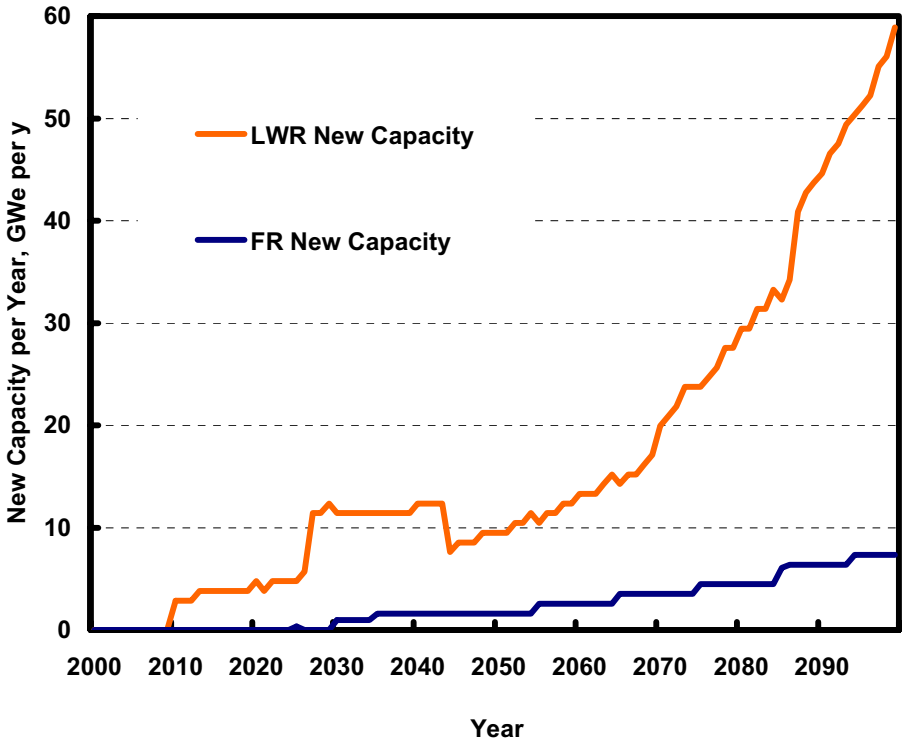

Figure 8.4-13. LWR and FR capacity additions for 3.2% growth

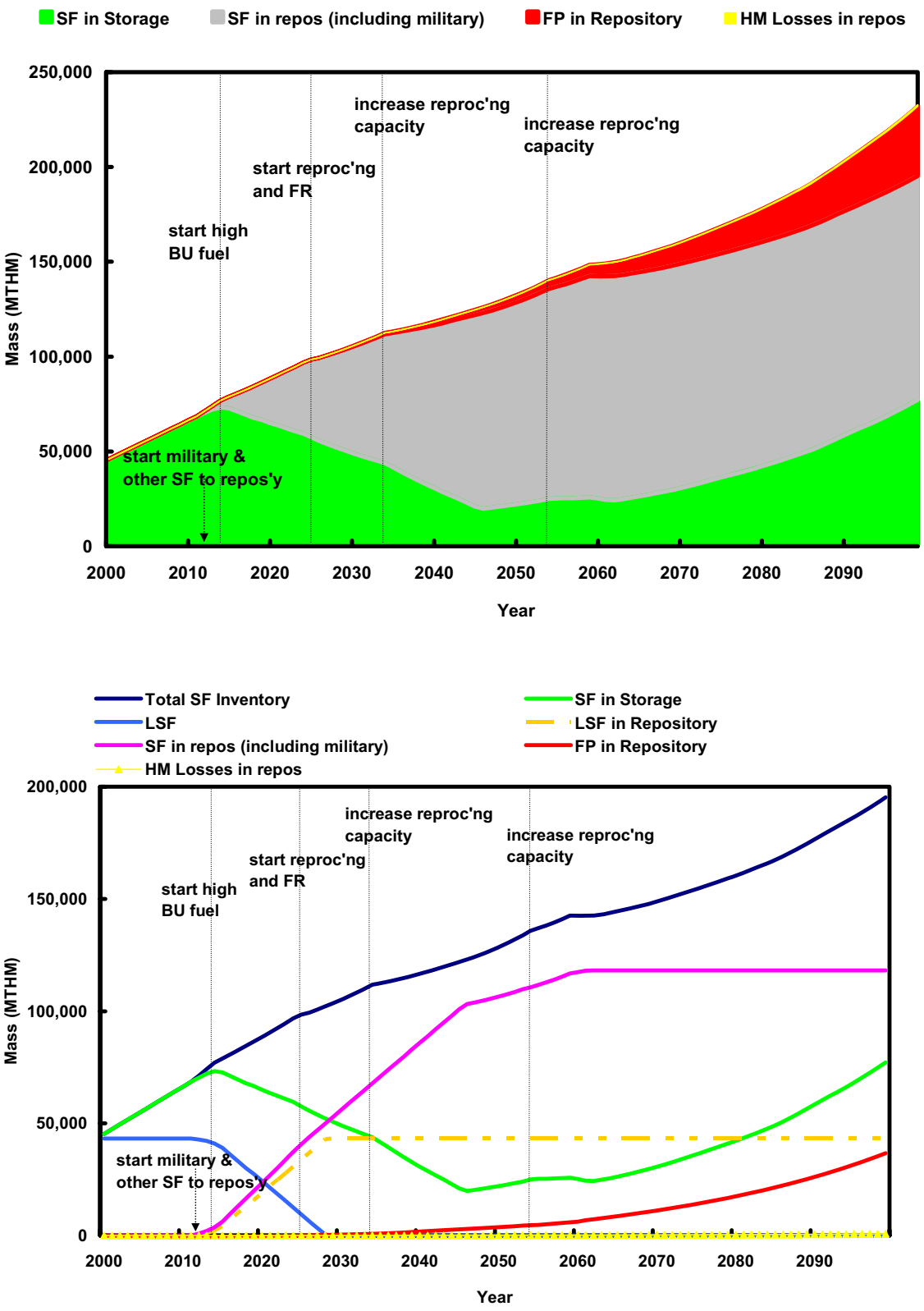

Figure 8.4-14. Dynamics of waste accumulation in both temporary and permanent storage for 3.2% growth

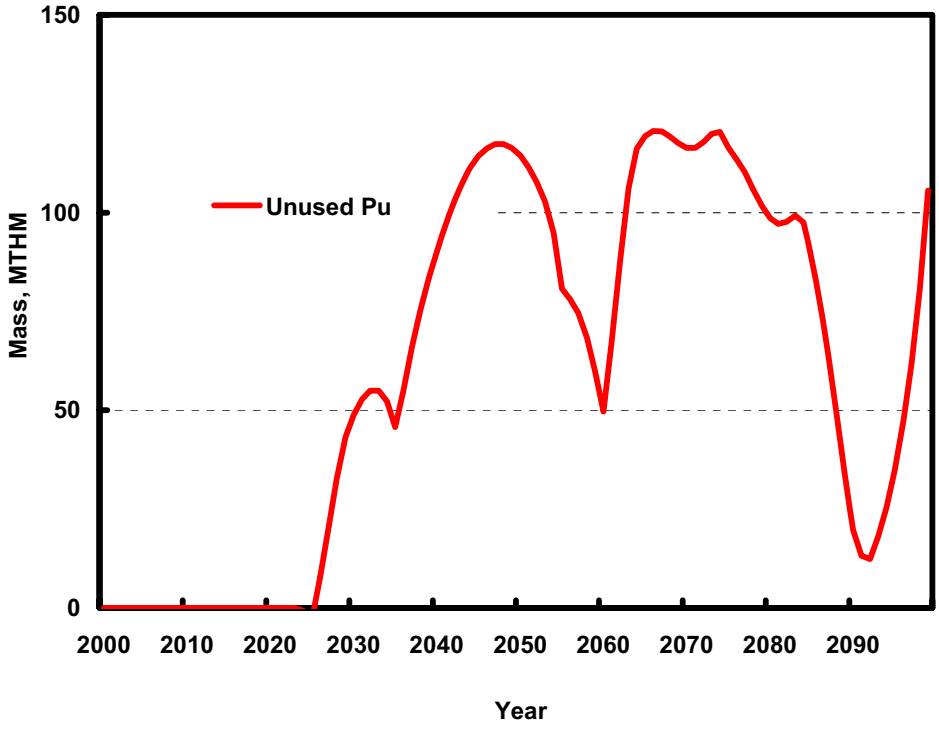

Figure 8.4-15. Pu from reprocessed SNF but not used (3.2% growth)

8.4.3 Conclusions of scenarios results

In conclusion, the continuation of the current once-thru fuel cycle practice should be re-evaluated as the demand for nuclear energy increases in the U.S. Potential consequences of the once-thru cycle include substantial increase in the number of geologic repository sites, continued accumulation of weapons-usable materials, and inefficient use of uranium resources. However, advanced fuel cycles as presented above, can limit spent fuel storage and direct disposal.

9. CONCLUSIONS AND PATH FORWARD

Reduce, Reuse, Recycle.

Reduce the number of repositories that cause so much controversy.
Reuse transuranics to maximize energy derived from uranium.
Recycle to minimize waste generation and manage weapon-usable inventories.

9.1. Top-level conclusions

Figure 9-1 summarizes our suggested high-level decision tree from a technical perspective. The branches of the first several decisions are relatively clear, the bottom half are the subject of most of this report.

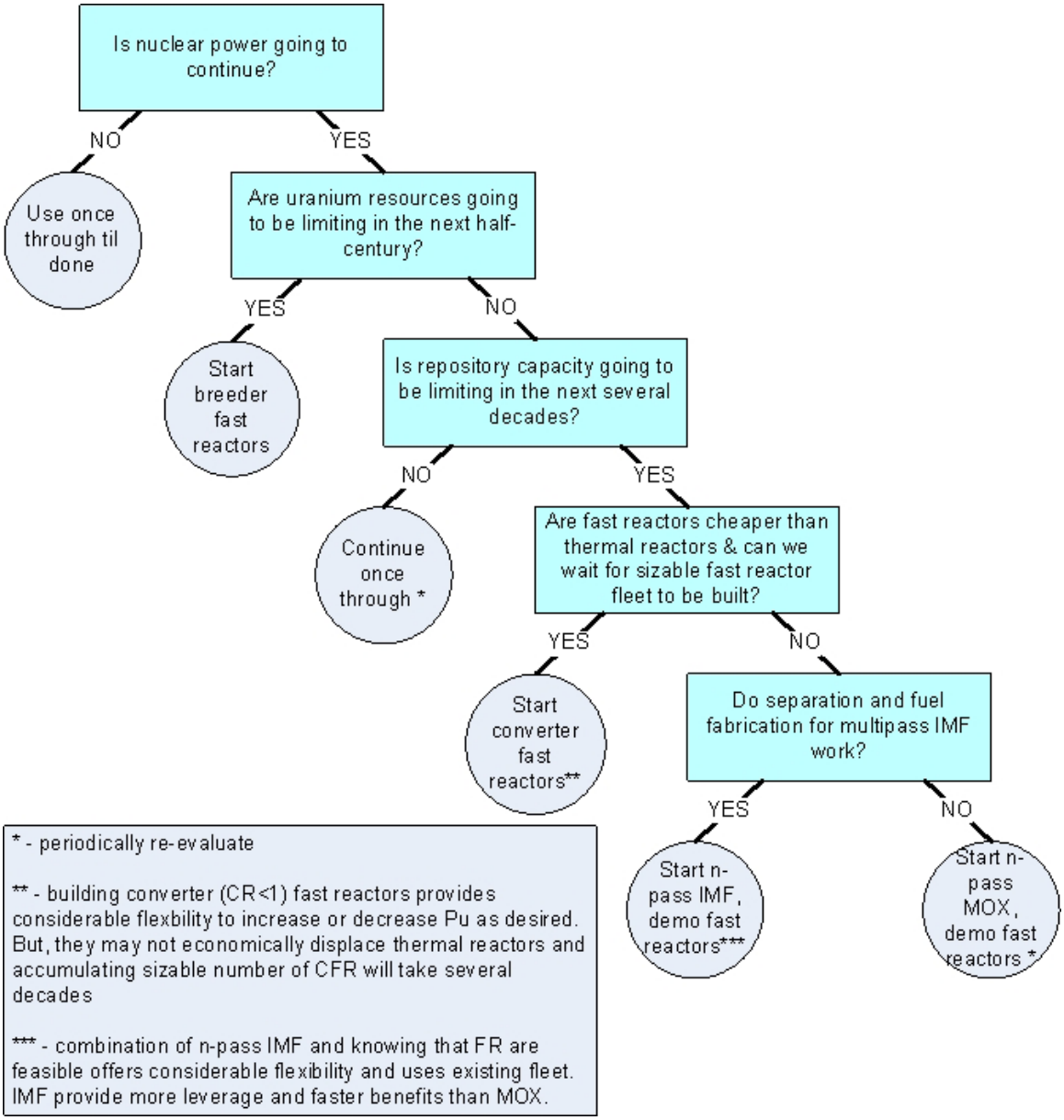

Figure 9-1 (Figure 2-2). Suggested decision tree for selecting among recycle options

Assuming that we plan on the continuation of nuclear power and that uranium resources are not an immediate crisis, Figure 9-2 illustrates that the nation must either establish the credibility of multiple repositories or establish the credibility of recycling. Repeated recycling can accomplish AFCI objectives.

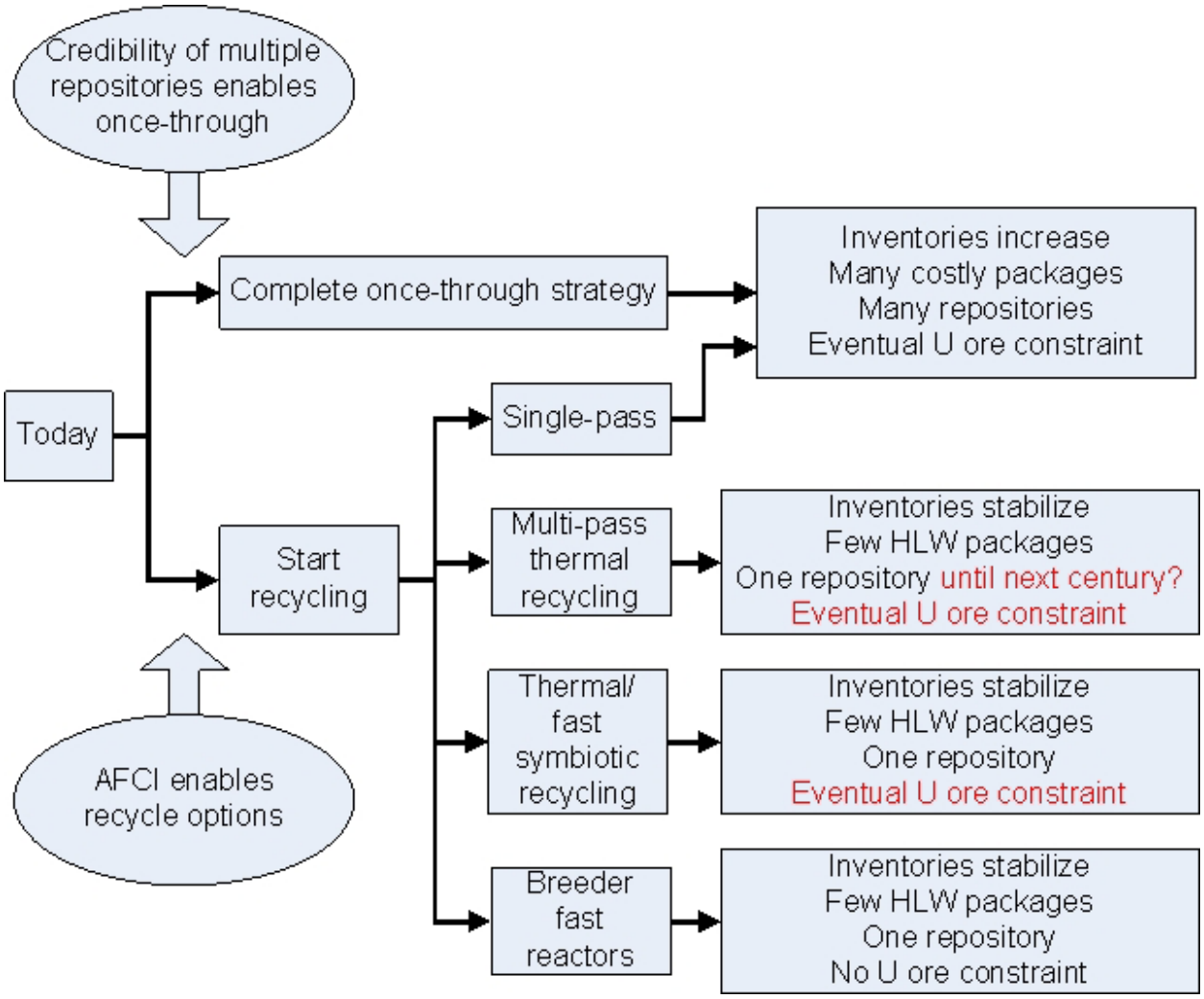

Figure 9-2. The nation needs either multi-repositories or multi-recycling

Table 9-1 summarizes single-pass results, when there are no constraints imposed by finite capacity of separation or fuel fabrication plants. None of these options are acceptable relative to AFCI program goals.[DOE2005a] Because of the insufficient benefits, we spend relatively little effort in this report on single-pass options. Some improvement could be expected with advanced once-through concepts that increase the fuel burnup and thermal efficiency (e.g., VHTR); however, these options are not explicitly considered in this report (see [Taiwo2005] for AFCI assessment of Gen IV options).

Table 9-1 (Table 5-1). Key Results for Single-Pass Cases
(pink means option does not meet target, yellow means it partially meets target.)

	Targets (see Chapter 3)	UOX-33	UOX-100	MOX-NpPu	MOX-NpPuAm	IMF-NpPu	IMF-NpPuAmCm	IMF-NpPuAm (blended core)	IMF-NpPu/Am (blended core)
Long-term heat (LTH) improvement	10x to 200x (to achieve actual repository improvements of 10-50x)[a]	0.95	1.17	1.07	1.12	1.98	1.82	1.61	1.67
Long-term dose (LTD) improvement	10-50x	0.90	1.12	1.35	1.41	2.09	1.96	1.57	1.63
Long-term radiotoxicity (LTR) improvement	100x	0.89	1.38	1.12	1.18	2.46	2.39	1.79	1.85
Uranium ore use improvement	1.15 short term 50x long term	0.88	0.97	1.09	1.07	1.15	1.14	1.13	1.14
Is option sustainable per repository limits		NO							
Is option sustainable per uranium limits		NO							

a. Program goal is to avoid a second repository for a century, implying the need for actual heat-limited repository capacity improvement of 10-20, even 50x to match the DOE laboratory directors goal. The LTH metric in this study can overpredict heat-capacity improvements, so the goal for the metric is 10-200.

Single-pass-only recycling in LWRs does not accomplish AFCI objectives, they can be removed from the program. We did not analyze VHTRs. The program should emphasize multi-pass options. There three types of multi-pass options, as follows:

- Recycling in thermal reactors only
- Recycling in a symbiotic mix of thermal reactors and consumer fast reactors (CFR)
- Recycling in breeder fast reactors (BFR)

With a few exceptions, these will all require remote fabrication. The possible exceptions are as follows:

- MOX-NpPu for one and possibly more cycles. Similarly for IMF-NpPu. However, this does not burn Am and therefore is only a "stop-gap" approach to recycling unless Am targets are added to the mix. Concentrating the Am in a few targets minimizes the amount of fuel that would require more expensive remote processing.
- The assemblies for the blended core IMF-NpPuAm have the lowest Am content of any of the Am-recycle fuels; they should be checked for dose rates.

Recycling in thermal reactors only – Although the strategy can be continued until uranium resources become a constraint, the benefits are limited because unburned TRU accumulates in the recycling fuel. Eventually, the unburned TRU would be discarded; however, we believe that this could be deferred until the next century as 5-cycles of either multi-pass IMF or multi-pass MOX appear feasible (on paper).

The multi-pass MOX-NpPuAm approach in this study varies the Pu/U ratio each cycle. It may give better performance. The multi-pass IMF-NpPuAm approach in this study uses blended cores – about 3/4 UOX pins and 1/4 IMF pins in each assembly, which results in ~98% of the heavy metal being in the UOX pins. (A variation puts the Am in 4 targets among 264 pins in each assembly.) In FY2006 we will validate these results and further examine the multi-pass IMF option space. The UOX fuel and UOX pins in the IMF blended core would be fabricated hands on. The MOX contains Np, Pu, and Am and there is little doubt it would require remote fabrication; see Chapter 5 and Appendix F. IMF with NpPuAm and Am targets would also require remote fabrication. IMF-NpPu pins would probably qualify for glovebox fabrication.

Both approaches meet the waste management objectives until unburned TRU is discarded; it appears that multi-pass IMF (which uses the blended core) accumulates waste management benefits almost twice as fast as multi-pass MOX. The IMF-pin component of fresh IMF assemblies is a relatively attractive proliferation target; however, like other IMF concepts, this one succeeds in burning Pu and degrading the Pu vector faster than MOX. Both meet the short-term uranium utilization objective (15% improvement) but only toward the end of the century when there has been time and sufficient separation/fabrication capacity to reach cycle-2 for multi-pass IMF and cycle-4 for multi-pass MOX. Safety and economics could prove dominated by the difference in TRU throughput (throughput of uranium and fission products varies little) – multi-pass IMF has typically 1/2 to 1/3 of the TRU throughput of multi-pass MOX. Building the infrastructure for thermal-only recycling (either MOX or IMF) provides much of the infrastructure for later CFR or BFR systems. However, if fast reactors are not readied for potential deployment, the pure-thermal strategy would require that much additional time to convert to one of the other strategies.

Other pure-thermal conclusions are as follows:

- Using Pu/U in full core MOX leads to varying composition each cycle, with large recirculating Pu flows. This means that the separation of MOX as the cycles evolve will require changes to the separation plant.
- Fuel composition in IMF blends stays fairly constant. The recirculating Pu flows in multi-pass IMF are typically 1/2 to 1/3 of multi-pass MOX. The separation plant for of IMF as the cycles proceed can stay constant, the elemental composition of the feed changes little.
- More than 80% of the fuel in the system at any given point in time (even at equilibrium) in multi-pass IMF or multi-pass MOX is still UOX. This means that the separation plants for thermal recycling primarily handle UOX, even with multi-pass MOX or IMF scenarios. When thermal recycling starts, the system is 100% UOX; this drops slowly to 80% as used UOX is processed and the TRU made into IMF or MOX. The 80% value can only be exceeded temporarily if separation and fuel processing is adequate to draw down the legacy UOX.
- To reduce uranium throughput, the only effective leverage is burnup. Recycling, per se, does not reduce uranium throughput.
- Can't reduce fission product throughput on a per GWe basis.
- To reduce plutonium throughput, use IMF. If one overburned Pu in 1-pass IMF, recycling cannot be sustained, the residual unburned TRU would eventually be discarded and waste management objectives not met. Backing off to blended multi-pass IMF avoids this problem.
- Based on crude scaling analyses, we believe that MOX-NpPu and IMF-NpPu would be glovebox-fabrication, all other recycle fuels (which all contain Am) would require remote fabrication. The exception is part or all of the IMF blended cores. In multi-pass IMF-NpPuAm, the 204 UOX pins (of 264 total) would be hands-on; the 60 IMF-NpPuAm pins would be remote. The final assembly is probably remote, but more work is needed. In multi-pass IMF-NpPu with separate Am pins, the 200 UOX pins are again hands-on, the 60 IMF-NpPu pins would be glovebox, the 4 Am target pins would

be remote, and more work would be required to know if the final assembly would be remote or glovebox.

- If only looking at the Pu239 equivalent fraction in fresh fuel, avoid full-core IMF. The Pu239-equivalent fraction is 65-70%. Indeed, the fraction of Pu is over 90%. However, if one takes credit for the Pu "quality", the picture changes. The Pu239/Pu-total fraction for all the first pass fuels is the same as the UOX-51 output, 53%. For multi-pass IMF, multi-pass MOX, and CFR, the Pu239/Pu-total fraction steadily decreases. It increases for BFR cases, evolving toward an equilibrium value of 72%.
- To reduce the amount of recirculating weapons-usable material, use IMF, avoid BFR. The flux of Pu239-equivalent/yr per GWe of fresh fuel is 0.26 for 5th pass IMF, 0.50 for 5th pass MOX, 0.36 for IMF-CFR, and 1.07 for BFR.
- Among multi-pass pure thermal systems, IMF appears superior to MOX, see Table 5-2. However, either only obtains ~17% improvement in uranium utilization and are therefore not sustainable from the uranium perspective in the long term. Although thermal recycling can be continued indefinitely, eventually sufficient TRU accumulates so that the TRU would be discarded with corresponding waste management penalties. However, this appears deferrable until the next century because 5 recycles appear practical. Building the infrastructure for multi-pass thermal recycling establishes some of the infrastructure for later fast reactors. And, using IMF would decrease the percent of CFR needed from 27% to 19%.

Recycling in a symbiotic mix of thermal reactors and CFR – On paper, this strategy can be continued until uranium resources become a constraint. Unburned TRU never has to be discarded. These options meet the waste management objectives provided the loss per recycle is acceptable and provided that one does not stop recycling. As the CFR fuels would contain Np, Pu, Am, and Cm there is little doubt that they would require remote fabrication.

We studied three cases: (1) using the TRU in discharged UOX in CFR, (2) using TRU from discharged MOX in CFR so that there are three types of fuels UOX, MOX, and CFR; and (3) using TRU from discharged IMF in CFR, again with three types of fuels. The three equilibria differ because of continuing makeup from the thermal reactors, which themselves differ.

The IMF-CFR combination generally provides the best performance. For example, the IMF-CFR symbiosis requires only 17% CFR in the fleet. The MOX-CFR system has the highest recirculating TRU throughput and the composition of the recirculating fuel has high fractions of undesirable isotopes. This could be good from the proliferation resistance perspective, but undesirable from other perspectives. Separation and fabrication loss goals derived from the first pass of used UOX are sometimes not adequate for CFR systems. Building the infrastructure for thermal/CFR symbiosis provides the experience and much of the infrastructure for later BFR systems. However, the thermal reactor component of this system would have to be phased out during transition to BFR; otherwise, the uranium utilization benefits are little better than pure thermal systems. Symbiotic systems have the most agility; if CFR performance is poor, they can be de-emphasized. If uranium begins to appear as a constraint, the breeding ratio of the fast reactors can be enhanced and eventually the thermal reactors phased out. If symbiotic systems had to be terminated, both the CFR and IMF can burn down remaining TRU leaving a relatively clean exit.

Other CFR symbiotic conclusions are as follows:

- More than 70% of the fuel in the system (at equilibrium) in thermal/fast-consumer symbiosis cases is UOX.
- Can't reduce fission product throughput on a per GWe basis.
- To reduce Am and Cm throughput, use BFR or IMF. Whatever the initial transuranic feed, the higher actinide content will increase slightly in the CFR.

- To reduce the required fraction of CFR in a thermal-CFR symbiotic system, use IMF. IMF-CFR requires 19% CFR; MOX-CFR requires 20% CFR; and UOX-CFR requires 27% CFR. Starting recycling with IMF, therefore, reduces the need for CFR later.

Recycling in BFR – This strategy can be continued indefinitely. Unburned TRU never has to be discarded. These options meet the waste management objectives provided loss per recycle is acceptable and provide that one does not stop recycling. As the fuels contain Np, Pu, Am, and Cm there is little doubt that they would require remote fabrication. (To reduce the fraction of fuel requiring remote fabrication, segregation of Am and Cm in "targets" is theoretically possible in fast reactors, as in thermal reactors; the separation technology would have to be capable of separating NpPu versus AmCm.)

We studied two cases: (1) using the TRU in discharged UOX to start BFR and (2) using the TRU in discharged IMF to start BFR. The equilibrium BFR is the same; thermal reactors would be phased out.

The recirculating TRU mass is relatively high and the Pu "quality" in that mass is also high, hence the known proliferation criticisms of this approach. (The actual total system Pu inventory is lower than once-through with the modest breeding ratio in this study, 1.07. The BFR has a net Pu flux of 0.10 tonnes-Pu/yr per GWe; UOX-51 creates Pu at 0.22 tonnes-Pu/yr per GWe.) The same characteristics mean that the recirculating mass appears easier to handle and slightly higher separation loss rates could be tolerated relative to CFR, once the isotopic mix evolved toward the BFR equilibrium values. If BFR systems had to be terminated, one would first want to convert the BFR into CFR to burn down as much TRU as possible.

Other BFR conclusions are as follows:

- To increase the potential build rate of BFR, maximize its breeding ratio (esp. for early cycles), do not burn Pu239 in thermal reactors, and reprocess quickly to use fissile Pu241 in the BFR startup cycle. The ratio of output-fissile/input-fissile in this study are 1.06 (UOX-to-BFR startup cycle), 1.65 (UOX-to-IMF-to-BFR startup cycle), and 1.07 (BFR equilibrium cycle). The high ratio for the IMF-BFR startup cycle mitigates the burning of Pu239 in thermal reactors. (BFR equilibrium cycle is the same for UOX-to-BFR and UOX-to-IMF-to-BFR.)
- To reduce the Pu inventory in the entire system, avoid once-through. UOX-51 makes Pu at 0.22 tonnes-Pu/yr per GWe. The next highest Pu production rate is the equilibrium BFR, 0.10 tonnes-Pu/yr per GWe. Thus, there is less Pu in a BFR system (with the design of a breeding ratio of 1.07) than in a once-through system! Of course, the Pu in the UOX-51 system is "self protecting" because it is unprocessed used fuel, whereas the Pu in the BFR system is recirculating with lower levels of self-protection.

If the processing capacity is unlimited, Figure 9-3 shows the mass flux to a separation plant for the multi-pass cases, compared to UOX-51. The fission product (FP) per GWe is unchanged, of course. The UOX and MOX cases have the highest uranium throughput. BFR has the highest Pu throughput. Figure 9-4 looks at the same cases, but only the TRU elements. The program needs a cost algorithm as a function of the throughput of individual elements. Note that for fixed waste management goals, as throughput increases, tolerable separation and fabrication loss rates decrease. Furthermore, safety and proliferation risk would appear to scale with the active inventory. The BFR case has the highest TRU recirculating inventory; it is mostly Pu, which makes handling and waste management goals easier than the other cases, but with higher proliferation issues. The MOX and MOX/CFR have the highest Am recirculating throughputs.

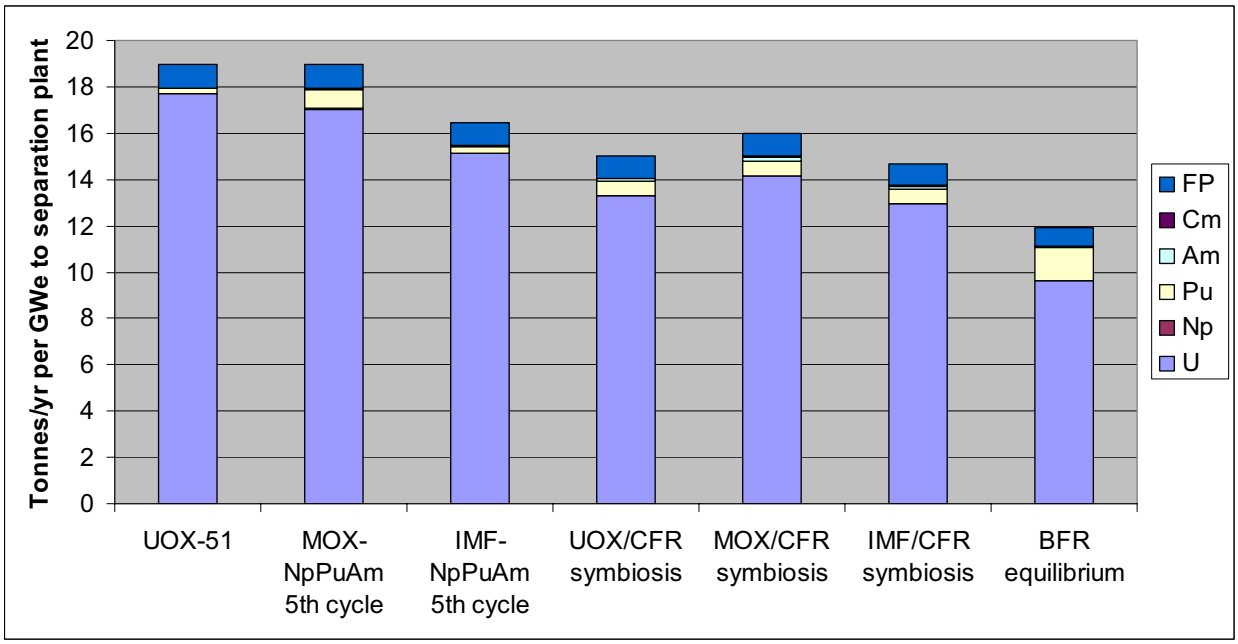

Figure 9-3. Throughput for selected cases, unlimited processing capacity (Figure 5-10 shows more cases)

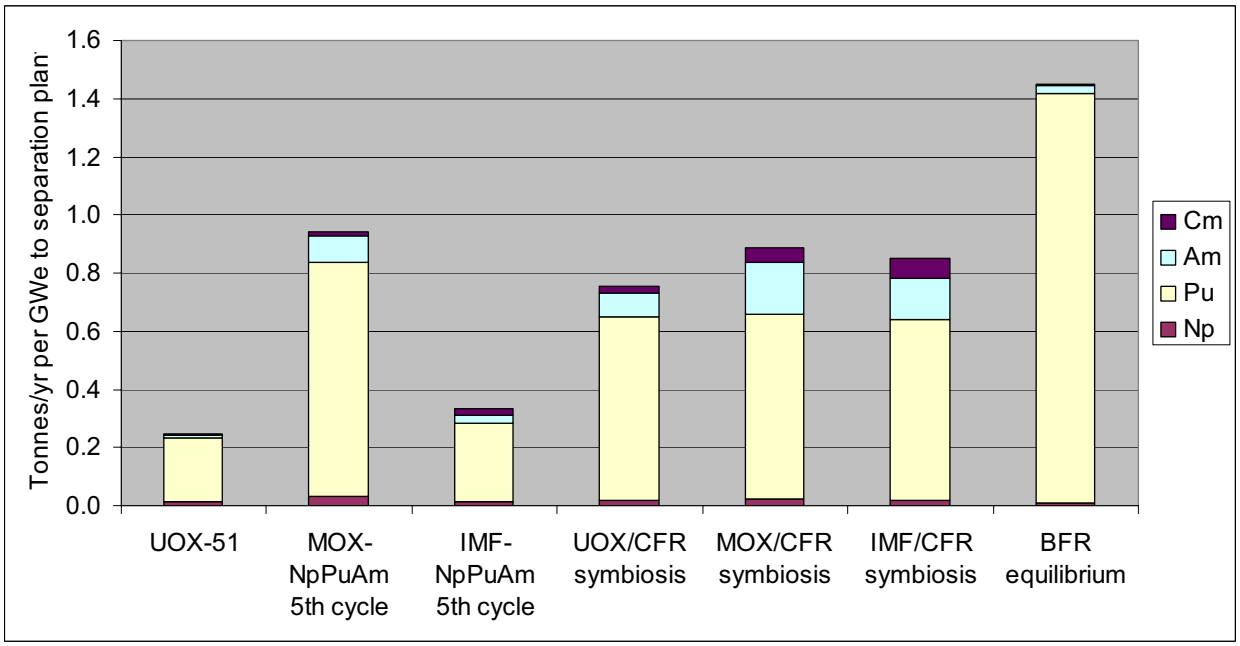

Figure 9-4. TRU throughput for selected cases, unlimited processing capacity (Figure 5-13 shows more cases)

Table 9-2 summarizes key results for multi-pass options when separation and fuel fabrication capacities are not limited. There are two numbers in most cells of the table. The first is the improvement factor this century (~5 recycles) if recycling then stops. The second is the improvement if recycling never stops, i.e., the system reaches a true equilibrium.

Table 9-2. (Table 5-2) Key Results for Multi-Pass Cases. First number in each cell is the improvement factor this century (~5 cycles) if recycling stops. The second number is the improvement if recycling never stops (only feasible with fast reactors in the system).

	Improvement Targets (see Chapter 3)	Thermal recycling with MOX	Thermal recycling with IMF	Consumer fast reactor (CFR) with IMF thermal recycling	Breeder fast reactor (BFR)
Long-term heat (LTH) improvement	10x to 200x (to achieve actual repository improvements of 10-50x)	1.5x / Plateaus near this value	2.9x / Plateaus near this value	~4x / ~50x at 99.5% removal of TRU+Cs+Sr	~5x / ~70x at 99.5% removal of TRU+Cs+Sr
Long-term dose (LTD) improvement	10-50x reduction in peak dose, which is at 500,000 years after emplacement	1.9x / Plateaus near this value	3.0x / Plateaus near this value	~4x / ~60x at 99.5% removal of TRU+U+Tc+I	~7x / ~190x at 99.5% removal of TRU+U+Tc+I
Long-term radiotoxicity (LTR) improvement	100 reduction of radiotoxicity at 1000 years after discharge so that waste is less toxic than original uranium ore	1.9x / Plateaus near this value	3.2x / Plateaus near this value	~4x / ~100x at 99.5% removal of TRU+U+Tc+I	~7x / ~100x at 99.5% removal of TRU+U+Tc+I
Uranium ore use improvement	1.15 short term 50x long term	1.17x / ~1.2x	1.17x / ~1.2x	1.32x / 1.42x	2.0x / ~100x
Pu239 equivalent tonnes/yr per GWe for fresh fuel	As low as possible	0.50 / Slow increase	0.26 / Slow increase	Not estimated / 0.36	Not estimated / 1.07
Pu239/Pu-total in fresh fuel	As low as possible (value for discharged UOX-51 is 53%)	32% for 5th cycle fuel / Cannot drop much further	33% for 5th cycle fuel / Cannot drop much further	14% in CFR 53% in IMF / 14% in CFR 53% in IMF	72% / 72%
Avoid fully remote fuel fabrication	For as much fuel as possible	True for the 80% of the fuel that is UOX, untrue for MOX-NpPuAm itself	True for the ¾ UOX pins in blended assemblies, true for IMP-NpPu (with separate Am targets)	No	No
Minimize throughput of TRU (tonnes/yr per GWe)	As low as possible to minimize safety and economic issus	0.94 / Slowly increase	0.34 / Slowly increases	Not estimated / 0.85	Not estimated / 1.45
Percent fuel that is new	As low as possible	17%	17%	29%	100%
Percent of	As low as	Zero	Zero	19%	100%

reactors that are new	possible				
Is option sustainable per repository limits		NO, because unburned TRU must eventually be discarded, but probably after this century		Yes, unburned TRU does not ever have to be discarded, performance depends on loss rates	
Is option sustainable per uranium limits		NO			Yes
The LTH improvement factors for 5-cycles CFR and BFR come from Wigeland2004a as they use cycle-by-cycle compositions for those cases, which we did not use in this study. The 5-cycle value for MOX (1.5x) is the same in Wigeland2004a and here. The 5-cycle uranium improvement factors for BFR are probably understated, they depend on the breeding ratio (output-fissile/input-fissile) for the first few cycles, which was not optimized.					

Fuel cycle time scales are long. Even after a decision is implemented, e.g., recycling starts, it generally takes decades before the impacts across the U.S. power plant fleet become significant.

Table 9-3 looks at anticipated decisions over the next several decades. As one goes further into the list, the uncertainties increase. Key uncertainties are discussed below.

Table 9-3. Status and Issues for Suggested Key Fuel Cycle Decisions in Decreasing Order of Readiness and Robustness

Key Decisions	Status and issues
D1. Open 1st geologic repository	Established US policy, implementation delayed. The basis for AFCI waste management calculations is YMP. We see no reason why YMP would not work well with a recycling strategy, but more work is warranted to confirm this.
D2. Determine credibility of recycling	There are only two sustainable high-level waste (HLW) approaches: multi-recycling and multi-repositories; neither is known to be credible today. This decision would not commit the U.S. to recycling, only determine "can recycle."
D3. Determine need for recycling and build 1st separation plant for UOX	If "should recycle" is established, the question is what separation plant should be built. All recycle scenarios include a UOX separation plant(s) for existing UOX, for the >80% UOX in IMF and MOX scenarios, and for the >67% UOX in CFR scenarios. Capacity should be 3,000 to 5,000 tonne/year to reduce at-reactor inventories without over-building capacity. At present, we suggest the UREX+ plant should be configured to provide NpPu/Am/Cm. (Alternative: NpPu/AmCm) Purity of separated Cs-Sr and U should meet 10CFR61 standards. Tc and I should be set aside for either specialized waste forms, specialized repositories, or transmutation targets.
D4. Build 1st recycle-fuel fabrication plant	Following closely behind a first separation plant would be a first recycle-fuel plant. The main categories of options are IMF, MOX, fuels for consumer fast reactors, fuels for breeder fast reactor. The selection among these options depends on too many factors to down-select today. We can say that non-recyclable fuels should be given low priority; multiple recycles are required to meet AFCI program objectives. We suggest analyses and tests that could improve each of the main options.
D5. Build future separation and fuel fabrication plants	Having made D4, the next decisions will involve follow-on separation and fuel fabrication plants. The dynamics of managing the fuel cycle are difficult. Assuming a 1-decade delay between decision and implementation, spacing major decisions by 2-decades (as we have in this study) means there is 1-decade of implementation and 1-decade observation between decisions. There

	is high potential to "outdrive our headlights."

Comparison of existing results versus the four AFCI programmatic objectives indicate that over 99.5% of NpPuAm must be recycled but that Cm can be disposed. Over 99% of Cs and Sr must be separated from repository wastes to accomplish the repository head-load objective. Preliminary long-term dose estimates suggest that 90% of uranium should be recycled. Similarly, it appears that the Tc and I dose must be reduced by 10x so that the long-term repository dose does not increase as more reactor-years worth of residual waste is emplaced; this is equivalent to saying that the dose reduction/mass-emplaced should be at least as high as heat reduction/mass emplaced (aka Wigeland Factor). The Tc and I reduction could in principle be accomplished by either transmutation (previous ANL work shows this is feasible but slow), specialized waste repositories, or specialized waste forms that receive associated regulatory credit for 10x slower dissolution/leaching than UOX.

Tc and I do not have to be recycled to meet the long-term radiotoxicity objective; neither do short-lived Cs and Sr. Pu and Am have to be recycled to meet the objective to reduce LTR by 100x; further analysis will clarify the separation targets. In the critical 1000 to 100,000 time period, preliminary calculations indicate that Pu dominates LTR. Thus, to reduce LTR by a factor of 100, Pu itself must be reduced by almost a factor of 100. Recovery of 99% of the Pu is therefore not sufficient for multi-pass recycling.

Table 9-4 provides current recycle targets for individual elements from the perspective of different possible goals. The current program targets are based on discharged UOX-51. There should now be a systematic re-evaluate the old goals to (a) reflect the 2005 AFCI program goals, (b) multi-pass recycling per this report, and (c) how pyroprocessing systems could be optimized. LTR analyses are only preliminary and have not been factored into this table.

Table 9-4 (Table 2-6). Suggested Separation Targets

	Recovery fraction (each recycle)		Product purity (each recycle)	
	Current goal [Vandergrift2004]	Differences identified in this study	Current goal [Vandergrift2004]	Differences identified in this study
Uranium – potential disposal	>90%	90% recovery is barely adequate to reduce LTD by 10x, would have to be 98% to reduce LTD by 50x.	Meet 10CFR61, e.g., <100 nCi-TRU/gram-U, requiring decontamination factor from Pu of > 105	
Uranium – potential recycle	Not specified	Not adequate for IMF options. Even 99% recovery would only make the U and Pu from UOX-51 comparable (0.17 and 0.22 tonnes/yr per GWe)	"If uranium is destined for recycle in reactor fuel, its purity requirements are greater and would be governed by ASTM C 877-98."	Not analyzed
NpPu	>99%	Possibly not adequate for MOX/CFR symbiosis, goal of 99.5% in DOE2005a appears adequate.	"The purity of this product stream is required to meet mixed-oxide (MOX) fuel specifications as described in ASTM C833-01."	
Am	>99.5% to provide 100x decrease in LTH	Appears adequate	"Based on fast reactor recycle of all TRU, the lanthanide content of the Am/Cm product must be <20 mg/g uranium plus TRU."	
Cm	>99.5% to provide 100x decrease in LTH			
Tc and I	>95% to provide 20x decrease in LTD	Recommend 98% to allow LTD reduction of 50x, comparable to LTH reduction.	"If transmutation of Tc is the chosen option, the Tc product must contain less than 16 μg of fissile actinides per g of Tc." 4 μg for I.	Not analyzed in this report, but we do not see the basis for this
Cs and Sr	"97% recovery required for Cs and Sr to make their recovery equal to that of all other fission products."	Recommend 99% to accommodate repeated recycling while maintaining high LTH benefits	Meet 10CFR61, e.g., <100 nCi-TRU/gram of Cs-Sr product	

Table 9-5 lists what we assess to be the six most important future unknowns, i.e., factors influencing major fuel cycle decisions. These are defined in section 1.5. Actions should be taken (a) decouple decisions from uncertain key decision factors by making options as robust as possible, (b) take actions that reduce uncertainties, and (c) prefer actions that cause favorable changes in factors, i.e., maximize option space.

Table 9-5. Key Factors Influencing U.S. Fuel Cycle Decisions

Decision Factors	How uncertainties could be reduced	How affected option space can be maximized
F1. Growth of nuclear energy?	Success of opening Yucca Mtn, NP2010, and other DOE programs will clarify the potential for growth.	Explore robustness and agility of options over range of nuclear growth scenarios.
F2. Cost and acceptance of additional repositories?	Explore the potential for (a) specialized repositories for Tc, I, Am, and (b) "standard" repositories for used nuclear fuel.	Consider three scenarios: no additional repositories, only "specialized" repositories, and many repositories. An Am-repository, for example, would decrease the need for CFR.
F3. Which thermal reactors succeed in the market place?	Ensure that VHTR fuels can be recycled so that either VHTR or LWR can fulfill required roles in sustainable fuel cycles.	Emphasize recyclable fuels, whether IMF or VHTR.
F4. How much uranium is available?	Update decades-old on-the-ground studies of uranium resources, both conventional and unconventional.	BFR should be kept within option space.
F5. What proliferation policies exist?	The relative importance of different proliferation objectives may be impossible to settle, therefore we need a suite of options that address each, e.g. IMF to burn and degrade Pu, co-located separation and fabrication, co-located reactors to minimize transportation.	Despite its undesirability to AFCI colleagues, estimate the cost of spiking fuels with Cs to meet the narrow definition of "spent fuel standard" to provide a cost/benefit comparison.
F6. How much penalty is "hot" fuel separation and fabrication?	We will not truly know without a detailed design of separation and fabrication facilities.	Explore blended fuel assemblies, e.g., Am targets, so that most of the assembly can be fabricated hands-on or in a glovebox.

Table 9-6 shows six development trees, which are analyzed in detail in Chapters 6-8..

Table 9-6. Summary of Development Trees

Development Tree	Motivation	Major findings
T1. Continue once-through until 2040, i.e., delay recycling	Explores continuation of once-through for an additional 15 years.	Requires a major increase in SNF storage capacity. Allows time to resolve R&D issues, but cost may exceed benefits. Delays "proof" of R&D to time when U resources may be more constraining., "fixes" more costly.
T2. Start multi-pass IMF-NpPuAm in 2025	Attempts fastest possible reduction in LTH, LTD, and LTR using thermal reactors and UREX+ separation technology, but an unproven fuel.	2-pass IMF tends to load YMP faster than 5-pass MOX (or 5-pass IMF). End point in 2100 not strongly influenced by 15-year "first recycle" approach. Fastest reduction in Pu inventory.
T3. Start single-pass MOX-NpPu in 2025	Closest to current international practice and current technology, while avoiding separation of Pu	Not significantly different from T2 and T4 so long as recycling continues; larger TRU inventory if it stops. Best "step-wise" approach to introduction of recycling technology. Most cost-effective if Pu Disposition program partners.
T4. Start multi-pass MOX-NpPuAm in 2025	Attempts modest repository benefits using thermal reactors, UREX+ technology, and fuels relatively similar to current UOX and MOX-Pu.	Not significantly different from T2 and T3 so long as recycling continues. Better energy recovery from SNF than IMF. Less complex transition than IMF, logical follow-on to MOX-NpPu
T5. Start CFR in 2025	Moves into fast reactors, skipping recycling in thermal reactors. The early fast reactor experience would set the stage for BFR when uranium resources warrant.	Tends to produce the least SNF & least demand on YMP. End point in 2100 not strongly influenced by 15-year "first recycle" approach. Runs risk of "out-driving headlights."
T6. Start BFR in 2025	Moves into fast reactor, skipping recycling in thermal reactor. Aims to accommodate a hypothetical combination of limited uranium resources and high nuclear growth.	Best approach to coping with demand growth, but SNF penalty high if recycling stopped. End point in 2100 not strongly influenced by 15-year "first recycle" approach. Runs risk of "out-driving headlights."

9.2 Managing the Fuel Cycle System in spite of Uncertainties

Managing the fuel cycle system in a real-time fashion will not be easy, with the potential to "out drive" our headlights. Consider that managing the fuel cycle is metaphorically like driving a car or flying a plane. There are few "**control knobs"** available: what types of reactors are built, what types of fuels are used, and the capacity of separation and fabrication plants. All of the controls are very sluggish – with response times measured in decades. To compound the problem, there is no single driver; control is shared by government, industry, and regulators. Worse, it is dark (uncertain) and our headlights only illuminate a short distance into the future.

Therefore, two criteria in selecting among options should be robustness and agility. Robustness measures how much preferences stay constant if assumptions and future circumstances change. Agility measures the ease of adapting an option later if new circumstances warrant.

As an example, we find that the blended multi-pass IMF approach in this study would be more robust than the full core multi-pass MOX approach in several ways. One is that the chemical composition of recycled material changes significantly cycle-by-cycle for MOX, but not for IMF. Separation and fabrication plants with fixed capabilities would therefore be able to handle a wider range of IMF situations than MOX.

One final way we attempt to summarize the wide range of static and dynamic analyses is to identify four approaches that would increase our ability to drive or pilot the fuel cycle system.

1. Have a recycle strategy that could be implemented before the current reactor fleet retires in 2027-2043 so that replacement reactors fit into the strategy. The reactors built in that time period will determine much of the fuel cycle for the rest of this century.
2. Establish multi-pass blended core IMF as a downward Pu control knob. It can, for example, stabilize the Pu inventory even at 1.8% growth. And, for equivalent SNF throughputs, it can be implemented faster than MOX (if the technology is available) because of the low TRU throughputs. IMF options can be tuned from breeding/conversion ratios near zero to at least 0.6. The capital investment of reactors would appear to far exceed that of separation and fabrication facilities. If the IMF infrastructure is built and later not needed, thermal reactors can still be operated profitably. IMF appears a more effective and flexible control knob than MOX.
3. Establish FR with flexible conversion ratio as a future control knob. This "control knob" takes longer to become available because fast reactors must first be a significant (several percent) of the fleet.. The breeding ratio and conversion ratio (conceptually similar but not numerically the same) should be variable from ~0.25 to at least 1.3. Unlike the IMF control knob, this one can substantially reduce uranium ore needs if breeding/conversion is over one. However, deployment of FR should proceed cautiously because once built there is high incentive to continue their operation. IMF, and possibly MOX, used in conjunction with CFR reduces the number of CFR needed and therefore is a logical way to move into fast reactors.
4. Expand exploration of heterogeneous assemblies and cores, which appear to have advantages and agility. The need for heterogeneous cores in fast reactors is well known. Analyses suggest advantages for blended (heterogeneous) assemblies in thermal reactors. In particular, the blended core multi-pass IMF approach in this study offers significant advantages as well as agility. Even better, perhaps, could be separating IMF-NpPu versus Am targets, so that little of the fuel would require remote fabrication. And, one could imagine turning down Pu consumption by reducing the fraction of IMP-NpPu pins while keeping the waste management benefits of Am targets.

9.3. Path Forward

We believe these analyses have addressed and clarified most. However, in support of future down-selection among options and the 2007-2010 Secretarial Recommendation on the need for a second geologic repository, additional work is needed along these lines. Within system analysis, we must convert the Stella-based DYMOND model to another software platform to resolve software-limitations we faced this summer. In doing so, the system dynamic model will be combined with the economic database. The combined model is tentatively called VISION, for Verifiable Fuel Cycle Simulation.

AFCI in general

- Closer cooperation and cross-reviews among program elements.
- Work to clarify the six factors noted above.
- Build consensus for D2 – determine the credibility of recycling.
- Improve the metrics for long-term dose, long-term heat, long-term radiotoxicity.
- Better integrate this work with proliferation resistance methodology and analyses.

- Identify and track all mass flows, including zirconium, carbon (TRISO), and nitrogen (nitride fuels).

Reactor and transmutation analyses

- Examine blended MOX strategies (such as the French MOX-UE concept) for potentially better performance than MOX-Pu/U in this study.
- Examine and validate multi-pass IMF
  - E. Hoffman will explore design space of U-235 enrichment in non-IMF pins & ratio of IMF pins to UOX-pins, e.g., going up to 60 pins in 264 assembly (17x17). Will concentrate on 3-batch, 1500-day burnup
- Examine VHTR options analogous to the LWR options in this study.
- Fill in missing elements in matrix
  - UOX-MOX-BFR
  - Single-pass IMF-NpPuAm (full core)
  - Single-pass MOX-NpPuAmCm
  - Multi-pass MOX-NpPu
  - Multi-pass MOX-NpPuAmCm (is it really worth the effort to separate Cm?)
- Perform scoping analysis for symbiotic thermal-BFR cases to explore how BFR could be slowly brought on line and how the symbiosis could maximize both waste management and uranium performance.
- Examine reactor safety limits for multi-pass MOX and IMF.
- VISION: Investigate wet-to-dry storage transition as function of heat and/or dose (replace fixed X-year)

Separation and system analyses

- Separation experts and system analysis colleagues should update separation and recovery targets, see Chapter 2 for our initial suggestions.
- Cost algorithm as function of throughputs of individual elements
- VISION: 1st order model of separation of individual elements with associated loss fractions
- VISION: Consider a "button" to build reprocessing if and only if committed SNF > 3000 x 30 years

Fuel fabrication and system analyses

- Fuel experts and system analysis colleagues should identify and start addressing issues associated with heterogeneous assemblies.
- Cost algorithm as function of hands-on/glovebox/remote fabrication for pellets/pins and for assemblies.
- VISION: 1st order model of fabrication of pellets/pins versus assembles.
- Calculate representative dose rates for prototypical fuel pins and assembly options.

Wine Cellar (how separation and fuel fabrication interact, where separated products are stored and blended into fuel fabrication)

- Identify algorithms for modifying both input/output fuel compositions with different strategies such as Pu/U, U235 enrichment, Am/Pu.
- VISION: 1st order model of a wine cellar is needed.
- VISION: Need algorithms for isotopic tracking of fuel

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APPENDIX A. ACRONYMS

A Americium, used for brevity, e.g., MOX-NPAC = MOX-NpPuAmCm denotes MOX with Np, Pu, Am, and Cm

Am Americium

AFCI Advanced Fuel Cycle Initiative

AMAD Activity Median Aerodynamic Diameter, a key parameter for aerosols and particulates

ANL Argonne National Laboratory

BFR Breeder Fast Reactor

BNL Brookhaven National Laboratory

BU Burned uranium, uranium that has been through a nuclear reactor at least once.

BWR Boiling Water Reactor

CFR Consumer Fast Reactor, breeding ratio and conversion ratio less than 1. Previously called Converter Fast Reactor.

C Curium, e.g., MOX-NPAC = MOX-NpPuAmCm denotes MOX with Np, Pu, Am, and Cm

Cm Curium

DOE Department of Energy

DU Depleted uranium, uranium containing less than 0.711% ^{235}U, currently discarded during uranium enrichment.

DYMOND Dynamic Model of Nuclear Development, the system dynamic model used for calculations in Chapters 7 and 8.

FP Fission products

FR Fast Reactors, either consumer (CFR) or breeder (BFR) depending on breeding ratio/conversion ratio.

FY Fiscal Year

GFR Gas-cooled Fast Reactor

GTCC Greater Than Class C waste, waste that does not qualify for near-surface disposal per 10CFR61, but is not considered HLW.

GWe Giga-watt (electric)

GWth	Giga-watt (thermal), GWth times thermal efficiency gives GWe
HM	Heavy Metal, in this report, actinides and TRU
HLW	High Level Waste
iHM	initial Heavy Metal, the amount of heavy metal in fuel it is irradiated
kt	kilotonne, i.e., a giga-gram
IAEA	International Atomic Energy Agency
ICRP	International Commission for Radiological Protection
IMF	Inert Matrix Fuel, fuel without uranium, only TRU
LANL	Los Alamos National Laboratory
LFR	Lead-cooled Fast Reactor
LTD	Long-Term Dose, see section 3.1.2
LTH	Long-Term Heat, see section 3.1.1.
LTR	Long-Term Radiotoxicity, see section 3.1.3
LLNL	Lawrence Livermore National Laboratory
LLW	Low Level Waste, which is actually not a regulatory term. In common language, we refer to LLW as waste qualifying for near-surface disposal per 10CFR61.
LWR	Light Water Reactor, either BWR or PWR
MGR	Monitored Geologic Repository
MOX	Mixed Oxide fuel
MSR	Molten Salt-cooled Reactor
N	Neptunium, used for brevity, e.g., MOX-NPAC = MOX-NpPuAmCm denotes MOX with Np, Pu, Am, and Cm
Np	Neptunium
NPP	Nuclear power plant
NRC	Nuclear Regulatory Commission
P	Plutonium, e.g., MOX-NPAC = MOX-NpPuAmCm denotes MOX with Np, Pu, Am, and Cm

Pu Plutonium

PUREX Plutonium-Uranium Extraction, an aqueous separation technology that separates pure Pu from dissolved used nuclear fuel.

PWR Pressurized Water Reactor

SCWR Super-Critical Water Reactor

SFR Sodium-cooled Fast Reactor

SNF Spent nuclear fuel

SNL Sandia National Laboratories

TR Thermal Reactors

TRU Transuranic elements

U Uranium

UREX Uranium Extraction, an aqueous separation technology that separates U and NpPu from dissolved used nuclear fuel. Pure Pu is not separated.

WU Weapons usable, per IAEA definition.

VHTR Very High Temperature Reactor

VISION Verifiable Fuel Cycle Simulation, planned successor for DYMOND during FY2006

YMP Yucca Mountain Project

APPENDIX B. INPUT PARAMETERS

Table B-1 lists input parameters. The combination of 97.2 GWe initial installed nuclear power, 51 GWth-day/tonne-HM burnup, 90% capacity factor, 34% thermal efficiency gives a current used fuel rate of 1840 tonnes/year, which is below the 2000 to 2100 tonnes/year in recent years. The mismatch is caused by the value of burnup (51), which is a bit of an improvement versus the current average, which must be about 45. Therefore, we model the current fleet has having average burnup of 45 GWth-day/tonne-HM.

We note that many of these parameters are unimportant to the results in this study, but they are required to make the model work. For example, the unit sizes of TR and FR only determine the number of each type of reactor, not the total installed capacity (determined by growth rate) nor mass throughputs (determined by capacity and fuel burnup).

Table B-1. Default Values of Input Parameters

Parameter	Default value	Alternatives considered in current studies	Future considerations, e.g., VISION
ENERGY			
Initial installed capacity	97.2 GWe		
Energy growth rate	1.8%/year	0%/year 3.2%/year	
Year energy growth starts	2010 (probably unrealistic)		
ENRICHMENT and MINING			
Uranium enrichment capacity	Unlimited, whatever required to fuel thermal reactors		Discrete plants (low priority)
Uranium enrichment	Whatever required to meet fuel recipe, typically 4.3% for 51 GWth-day/t burnup		
Uranium tails	0.2%		
Time spent in mining	1 year		
Time spent in enrichment processes	1 year		
REACTORS			
Initial number of U.S. reactors	103		
Growth rate of reactors	Starting in 2010, whatever required to meet energy growth plus reactor retirements		
Thermal reactor type	LWR		Study other thermal reactors such as VHTR
Fast reactor type	SFR		Study other fast reactors such as lead and gas cooled

Parameter	Default value	Alternatives considered in current studies	Future considerations, e.g., VISION
Reactor size (thermal)	0.95 GWe		Need to rethink how to match legacy reactors (average size < 1 GWe) but then have realistic sizes for new reactors
Reactor size (fast)	0.60 GWe	0.32 GWe for the fast reactor transition analyses in Chapter 8	
Reactor capacity factor (thermal)	90%		
Reactor capacity factor (fast)	82.2%	95% in the fast reactor transition analyses in Chapter 8	
Reactor thermal efficiency (thermal)	34%		
Reactor thermal efficiency (fast)	38%		
Reactor burnup (thermal), GWth-day/tonne-HM	Set by fuel recipe, e.g. 45 for current fleet		
Reactor burnup (fast), GWth-day/tonne-HM	Set by fuel recipe		
Cycle length (thermal)	1 year (33 or 51 burnup) 1.5 year (100 burnup)		
Cycle length (fast)	1 year		
Number of batches (thermal)	5 (33 or 51 burnup) 3 (100 burnup)		
Number of batches (fast)	4		
Minimum at-reactor storage time before processing (thermal)	5 years		Consider making a function of heat rate as function of time
Minimum at-reactor storage time before processing (fast)	2 years		
Retirement of existing reactors	103/17 or ~6/yr starting in 2027, last existing reactor retires in 2044.		
Construction time	5 years		
Licensing time	2 years		
Reactor lifetime (TR or FR)	60 years		

Parameter	Default value	Alternatives considered in current studies	Future considerations, e.g., VISION
SEPARATION PLANTS			
Separation plant capacity and date for thermal reactors	3 kt-SNF/yr in 2025 + 3 kt-SNF/yr in 2040 + 3 kt-SNF/yr in 2060 + 3 kt-SNF/yr in 2080	Varied in some parameter studies	For each plant, specify range of allowable input compositions, e.g., plant-1 would be tuned to processing UOX.
Separation plant capacity and date for fast reactors	Unlimited, i.e. whatever required to keep fast reactor going		
Lifetime (thermal)	Unlimited		Finite lifetime
Lifetime (fast)	Unlimited		Finite lifetime
Time mass spends in separation plant (thermal)	1 year		
Time mass spends in separation plant (fast)	1 year		
Loss rate for separation+fuel fabrication plant	0.2%/pass thru plant	Varied in some parameter studies	Differentiate separation versus fuel fabrication losses
Loss rate for separation+fuel fabrication plant	0.2%/pass thru plant		
FUEL FABRICATION PLANT			
Fuel fabrication plant capacity and date for thermal reactors	Production rate of MOX or IMF is constrained by Pu availability. (If available Np, Am, or Cm are inadequate to meet the fuel recipe, DU is substituted.) If Pu-limited supply of MOX/IMF is inadequate, UOX is substituted		Discrete plants
Fuel fabrication plant capacity and date for fast reactors	Unlimited, i.e., whatever required to keep fast reactors going		Allow either unlimited (i.e. at-reactor) or discrete centralized plants
Lifetime (thermal)	N/A		Specified lifetime for discrete plants
Lifetime (fast)	N/A		
Time mass spends in fabrication plant (thermal)	1 year		
Time mass spends in fabrication plant (fast)	1 year		
Loss rate for fuel fabrication plant	Included in a single value for separation+fabrication (see above)		Differentiate separation vs. fuel fabrication losses

Parameter	Default value	Alternatives considered in current studies	Future considerations, e.g., VISION
REPOSITORY			
Initial SNF inventory in 2000	29,624.00 tonnes at 33 GWth-day/t burnup 13,612.92 tonnes at 50 GWth-day/t burnup Total=43,236.92		
Receiving rate and date at geologic repository	400 tonnes in 2012 600 tonnes in 2013 1200 tonnes in 2014 2000 tonnes in 2015 3000 tonnes in 2016 and thereafter		Update when YMP status is better known.
Repository capacity	Unlimited		
Maximum retrieval rate	Unlimited		

APPENDIX C. TRANSMUTATION RECIPES

This appendix documents the fuel composition recipes (input and 5-year after discharge) for the calculations in this study. All units in the following tables are mass fractions; the total of heavy metal (actinides, U, TRU) and fission products equals 1. Some totals deviate slightly due to round-off errors.

With two exceptions, all calculations were performed by AFCI transmutation analysis colleagues at ANL. The multi-pass IMF calculations were done by AFCI fellow Andrew Goldmann, during his summer at the INL. Those calculations are documented in Goldmann2005. The UOX-100 case was calculated by M. Todosow at BNL.

C.1 UOX Isotopics

The system dynamics calculations utilize two types of LWR fuels with typical medium- and high-burnup PWR fuel as shown in Tables C-1 and C-2. The medium burnup fuel has an initial enrichment of 3.2% U-235 and a discharge burnup of 33,000 MW-day/tonne. The high burnup fuel has an initial enrichment of 4.2% U-235 and a discharge burnup of 50,000 MW-day/tonne. The depletion calculations were performed using the ORIGEN-2 [ORIGIN1980] computer code. ANL has performed ORIGEN2 calculations [Kim2003b] using the one-group cross sections that were provided with the code. No separate WIMS8 [WIMS8] cell calculations were performed to obtain new cross sections at those burnups. Another set of calculations was performed by BNL for the ultra-high burnup UOX fuel with 100 GWd/tonne, validated by calculations at ANL. In this case, ANL performed the WIMS8 cell calculations to estimate the one-group cross sections for ORIGIN2 calculations,[Kim2004] instead of using the cross sections provided with ORIGIN2 (which did not provide reasonable results). WIMS8 calculations used 172-group, JEF2.2-based cross section library which has been previously determined to provide accurate modeling of the important Pu-239, Pu-240, and Pu-241 resonances. ANL results for this ultra-high burnup UOX calculation were compared to BNL results listed here, and the two sets of results were found to be comparable.

Table C-1. Once-Through Recipes for Input Fresh Fuel

MASS FRACTIONS	Once-through		
	UOX-33	UOX-51	UOX-100
Burnup (GWth-day/tonne-HM)	33	51	100
U234	0.0003	0.0003	0.0000
U235	0.0320	0.0430	0.0850
U236			
U238	0.9678	0.9567	0.9150
[redacted]			
U	1.0000	1.0000	1.0000

Table C-2. Once-Through Recipes for 5-years after discharge

MASS FRACTIONS	Once-through		
	UOX-33	UOX-51	UOX-100
Burnup (GWth-day/tonne-HM)	33	51	100
Ra226	1.04E-13	2.68E-13	1.10E-14
Ra228	9.10E-20	1.81E-19	0.00E+00
Ac227	2.67E-14	1.17E-13	2.26E-15
Ac228	9.50E-24	1.89E-23	0.00E+00

MASS FRACTIONS	Once-through		
	UOX-33	UOX-51	UOX-100
Th228	3.09E-12	2.34E-11	2.13E-10
Th229	2.05E-13	2.78E-12	5.02E-14
Th230	3.19E-09	5.10E-09	5.96E-10
Th231	3.28E-14	3.11E-14	0.00E+00
Th232	7.19E-10	1.25E-09	1.89E-09
Th234	1.37E-11	1.34E-11	1.24E-11
Pa231	2.66E-10	9.65E-10	4.55E-11
Pa233	1.16E-11	2.11E-11	5.90E-11
U232	1.34E-10	9.90E-10	1.12E-08
U233	1.89E-09	3.26E-09	3.71E-09
U234	1.60E-04	1.84E-04	6.64E-05
U235	8.06E-03	7.65E-03	9.35E-03
U236	3.87E-03	5.71E-03	1.25E-02
U238	9.44E-01	9.21E-01	8.56E-01
Np237	3.41E-04	6.21E-04	1.74E-03
Pu238	1.16E-04	3.07E-04	1.19E-03
Pu239	5.13E-03	6.15E-03	7.73E-03
Pu240	2.26E-03	2.92E-03	3.97E-03
Pu241	9.62E-04	1.38E-03	1.96E-03
Pu242	4.73E-04	8.64E-04	1.52E-03
Pu244	1.25E-08	2.86E-08	7.01E-08
Am241	2.90E-04	4.38E-04	6.55E-04
Am242m	3.48E-07	8.34E-07	2.90E-06
Am242	4.16E-12	9.98E-12	3.74E-11
Am243	7.90E-05	1.98E-04	4.71E-04
Cm242	5.83E-09	1.32E-08	2.72E-08
Cm243	2.13E-07	6.83E-07	1.67E-06
Cm244	1.83E-05	7.08E-05	2.48E-04
Cm245	1.03E-06	5.72E-06	3.07E-05
Cm246	9.56E-08	7.29E-07	5.57E-06
Cm247	8.40E-10	9.97E-09	1.21E-07
Cm248	4.33E-11	7.70E-10	1.45E-08
[redacted]			
C14	2.63E-11	4.05E-11	7.92E-11
Sr90	4.77E-04	7.00E-04	1.41E-03
Zr93	7.19E-04	1.09E-03	2.18E-03
Tc99	7.86E-04	1.14E-03	1.99E-03
I129	1.83E-04	2.75E-04	4.58E-04
Cs135	3.06E-04	6.60E-04	1.37E-03
Cs137	1.07E-03	1.62E-03	3.04E-03
[redacted]			
Ra	1.20E-13	3.89E-13	1.11E-12
Ac	2.67E-14	1.17E-13	2.26E-15
Th	3.92E-09	6.39E-09	2.72E-09
Pa	2.77E-10	9.86E-10	1.05E-10
U	9.56E-01	9.34E-01	8.78E-01
Np	3.41E-04	6.21E-04	1.74E-03
Pu	8.93E-03	1.16E-02	1.64E-02
Am	3.70E-04	6.38E-04	1.13E-03
Cm	1.97E-05	7.80E-05	2.86E-04

Other Actinides	5.99E-07	1.91E-06	0.00E+00
Total actinides	0.9659	0.9474	0.8974
Sr	8.28E-04	1.23E-03	2.52E-03
Zr	3.49E-03	5.30E-03	1.11E-02
Tc	7.87E-04	1.14E-03	1.99E-03
I	2.39E-04	3.59E-04	5.94E-04
Cs	2.54E-03	3.95E-03	7.20E-03
Other FP	2.62E-02	4.07E-02	7.92E-02
Total FP	0.0341	0.0526	0.1026

C.2 Transmutation in LWR: MOX and IMF isotopics

In all cases, it was assumed that the recycled material originated from spent UO2 irradiated in a commercial PWR that produces the LWR-UOX spent fuel with 50 GWd/tonne in Table C-2. The plutonium, neptunium, and americium (in some cases curium) were assumed to be recycled in a similar system a finite number of times, using a MOX or IMF fuel assembly. Different isotopic vectors can be in the charged assembly, including Pu+Np (first tier of a double-tier MOX/IMF-FR system), Pu+Np+Am (single or multipass MOX), or Pu+Np+Am+Cm in single pass IMF. The isotopic vector of Pu+Np+Am in the charged assembly in recycle N was derived solely from discharged assemblies in recycle N-1. In the MOX and IMF cases, the mass of TRU charged in the fresh recycled assembly was adjusted to meet the same operational requirements (full-power days of irradiation) in each recycle. Consequently, the mass of spent fuel processed to produce a given assembly varied from one recycle to the next. Notice that the burnup achieved by the IMF fuel (about 550 GWd/tonne in the first pass) well exceeds that for the MOX fuel(50 GWd/tonne) As mentioned above for the ultra high burnup UOX calculations, for transuranic recycling in LWRs, the WIMS8 code was utilized for the mass flow analyses. The transuranic material initially irradiated in the MOX or IMF fuel was assumed harvested from UO2 which had been enriched to 4.2 wt.% U-235, irradiated to 50 GWd/tonne, and cooled for 5 years.

Unlike the other cases, the multi-pass IMF cases were generated using the suite of Monteburns, MCNP, and ORIGEN2.[Goldmann2005] These calculations are also unique in that there is a blend in each assembly – part UOX and part IMF pins, as given in Table C-3. And, the burnup is decreased each cycle to compensate for the degradation of the TRU mixture in the IMF pins, see Table C-4.

Table C-3. IMF Blended Cores

	IMF/UOX blends	IMF/UOX/Am blends
Where is the Am?	In the IMF fuel	In separate Am targets
# UOX pins/assembly	204 (77.3%)	200 (75.8%)
# IMF pins/assembly	60 (22.7%)	60 (22.7%)
# Am pins/assembly		4 (1.5%)
U mass fraction in assembly (i.e. the heavy metal in UOX)	~98% (exact value depends on which IMF cycle)	~98%
TRU mass fraction in IMF pins	~2%	<2%
Am mass fraction in Am targets		~0.2%
Both cases also have 24 guide tubes and 1 instrument tube for a total of 289 pins (17 x 17)		

Table C-4. IMF Blended Core Burnup Parameters

Cycle	1	2	3	4	5
Full power days for IMF-NpPuAm	1500	1440	1380	1380	1350
Full power days for IMF NpPu/Am targets					
GWth-day/tonne burnup	64.82	62.13	59.45	59.35	58.00
GWth-day/tonne burnup					

The isotopic vector for the recycled transuranics is provided in Tables C-5, 6, 7, 8, 9, 10 [Stillman2004a, Stillman2004b, Goldmann2005]

Table C-5. IMF Recipes for Input Fresh Fuel

Fuel type	IMF						
Composition	NpPu	NpPuAmCm	Np Pu Am (blended core)				
Cycle	1	1	1	2	3	4	5
Burnup (GWth-day/tonne-HM)	633.2	553.8	64.8	62.1	59.4	59.4	58.0
U234			0.0003	0.0003	0.0003	0.0003	0.0003
U235			0.0418	0.0417	0.0416	0.0416	0.0415
U236			0.0000	0.0000	0.0000	0.0000	0.0000
U238			0.9409	0.9395	0.9380	0.9365	0.9355
Np237	0.0503	0.0476	0.0008	0.0009	0.0009	0.0009	0.0009
Pu238	0.0250	0.0236	0.0004	0.0010	0.0012	0.0013	0.0014
Pu239	0.5041	0.4764	0.0077	0.0060	0.0060	0.0061	0.0062
Pu240	0.2385	0.2254	0.0035	0.0037	0.0038	0.0040	0.0041
Pu241	0.1122	0.1061	0.0018	0.0019	0.0019	0.0019	0.0020
Pu242	0.0699	0.0661	0.0011	0.0027	0.0038	0.0046	0.0052
Am241		0.0338	0.0006	0.0007	0.0007	0.0007	0.0007
Am242m		0.0001	0.0000	0.0000	0.0000	0.0000	0.0000
Am243		0.0151	0.0003	0.0009	0.0012	0.0014	0.0016
Cm242		0.0000					
Cm243		0.0001					
Cm244		0.0054					
Cm245		0.0004					
Cm246		0.0000					
U	0.0000	0.0000	0.9831	0.9815	0.9800	0.9784	0.9773
Np	0.0503	0.0476	0.0008	0.0009	0.0009	0.0009	0.0009
Pu	0.9497	0.8976	0.0145	0.0153	0.0166	0.0179	0.0188
Am	0.0000	0.0490	0.0009	0.0016	0.0019	0.0021	0.0023
Cm		0.0058					
Fission products							

Table C-6. IMF Recipes for 5-years after Discharge

Fuel type	IMF						
Composition	NpPu	NpPuAmCm	Np Pu Am (blended core)				
Cycle	1	1	1	2	3	4	5
Burnup (GWth-day/tonne-HM)	633.2	553.8	64.8	62.1	59.4	59.4	58.0
Ra226	7.86E-13	9.48E-13					
Ra228	5.55E-21	5.30E-21					
Ac227	1.90E-13	1.93E-13					
Ac228	5.79E-25	5.54E-25					
Th228	3.14E-11	2.93E-11					
Th229	3.94E-12	3.91E-12					
Th230	2.85E-08	3.67E-08					
Th231	1.26E-15	1.49E-15					
Th232	3.85E-11	3.73E-11					
Th234	2.92E-17	2.54E-17					
Pa231	1.62E-09	1.64E-09					
Pa233	4.61E-10	5.22E-10	3.09E-11	3.18E-11	3.15E-11	3.21E-11	3.17E-11
U232	1.35E-09	1.26E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U233	4.48E-08	5.47E-08	4.75E-09	5.31E-09	4.92E-09	5.27E-09	4.87E-09

Composition	NpPu	NpPuAmCm	Np Pu Am (blended core)				
Cycle	1	1	1	2	3	4	5
U234	2.40E-03	3.27E-03	2.01E-04	2.21E-04	2.31E-04	2.37E-04	2.45E-04
U235	3.10E-04	3.66E-04	6.49E-03	6.94E-03	7.67E-03	7.78E-03	8.30E-03
U236	2.01E-04	2.04E-04	5.60E-03	5.54E-03	5.46E-03	5.42E-03	5.40E-03
U238	2.01E-06	1.75E-06	9.03E-01	9.03E-01	9.05E-01	9.04E-01	9.03E-01
Np237	1.36E-02	1.54E-02	9.10E-04	9.35E-04	9.27E-04	9.47E-04	9.35E-04
Pu238	3.80E-02	5.46E-02	1.00E-03	1.21E-03	1.30E-03	1.37E-03	1.44E-03
Pu239	1.42E-02	2.88E-02	5.95E-03	5.98E-03	6.13E-03	6.22E-03	6.27E-03
Pu240	8.35E-02	1.10E-01	3.67E-03	3.81E-03	3.97E-03	4.09E-03	4.29E-03
Pu241	3.64E-02	5.05E-02	1.89E-03	1.87E-03	1.93E-03	1.99E-03	2.04E-03
Pu242	1.10E-01	9.94E-02	2.67E-03	3.75E-03	4.53E-03	5.12E-03	5.63E-03
Pu244	5.04E-06	4.02E-06	6.44E-07	1.38E-06	2.12E-06	2.87E-06	3.56E-06
Am241	1.30E-02	2.11E-02	6.64E-04	6.59E-04	6.89E-04	7.19E-04	7.45E-04
Am242m	4.14E-05	1.15E-04	3.31E-06	3.37E-06	3.68E-06	4.04E-06	4.46E-06
Am242	4.95E-10	1.38E-09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Am243	2.64E-02	2.70E-02	8.76E-04	1.21E-03	1.40E-03	1.56E-03	1.66E-03
Cm242	1.57E-06	2.53E-06	2.86E-08	2.93E-08	3.11E-08	3.30E-08	3.54E-08
Cm243	1.43E-04	2.66E-04	9.88E-06	1.04E-05	1.00E-05	1.04E-05	1.04E-05
Cm244	1.90E-02	2.36E-02	6.34E-04	1.00E-03	1.15E-03	1.30E-03	1.34E-03
Cm245	1.91E-03	3.47E-03	1.69E-04	2.69E-04	2.90E-04	3.15E-04	3.21E-04
Cm246	6.49E-04	1.07E-03	4.42E-05	7.31E-05	7.33E-05	7.73E-05	7.14E-05
Cm247	1.51E-05	3.16E-05	1.07E-06	1.85E-06	1.76E-06	1.84E-06	1.64E-06
Cm248	1.40E-06	3.91E-06	1.24E-07	2.09E-07	1.87E-07	1.96E-07	1.59E-07
C14	4.87E-10	4.26E-10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	3.76E-03	3.33E-03	7.69E-04	7.50E-04	7.28E-04	7.26E-04	7.11E-04
Zr93	8.58E-03	7.59E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc99	1.32E-02	1.17E-02	1.43E-03	1.39E-03	1.34E-03	1.34E-03	1.32E-03
I129	3.90E-03	3.49E-03	3.61E-04	3.46E-04	3.33E-04	3.33E-04	3.24E-04
Cs135	1.05E-02	9.31E-03	8.01E-04	7.60E-04	7.38E-04	7.45E-04	7.36E-04
Cs137	2.04E-02	1.78E-02	2.00E-03	1.93E-03	1.86E-03	1.86E-03	1.81E-03
Ra	9.48E-13	1.10E-12	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ac	1.90E-13	1.93E-13	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Th	2.86E-08	3.67E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pa	2.08E-09	2.16E-09	3.09E-11	3.18E-11	3.15E-11	3.21E-11	3.17E-11
U	2.91E-03	3.84E-03	9.16E-01	9.15E-01	9.18E-01	9.17E-01	9.17E-01
Np	1.36E-02	1.54E-02	9.10E-04	9.35E-04	9.27E-04	9.47E-04	9.35E-04
Pu	2.82E-01	3.44E-01	1.52E-02	1.66E-02	1.79E-02	1.88E-02	1.97E-02
Am	3.95E-02	4.82E-02	1.54E-03	1.88E-03	2.10E-03	2.29E-03	2.41E-03
Cm	2.18E-02	2.84E-02	8.59E-04	1.36E-03	1.52E-03	1.70E-03	1.74E-03
Other Actinides	4.61E-04	8.59E-04	1.34E-03	8.97E-04	0.00E+00	0.00E+00	6.93E-05
Total actinides	0.3598	0.4402	0.9354	0.9371	0.9406	0.9408	0.9418
Sr	6.60E-03	5.84E-03	7.70E-04	7.50E-04	7.28E-04	7.26E-04	7.11E-04
Zr	4.29E-02	3.79E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc	1.32E-02	1.17E-02	1.43E-03	1.39E-03	1.34E-03	1.34E-03	1.32E-03
I	5.12E-03	4.59E-03	3.61E-04	3.46E-04	3.33E-04	3.33E-04	3.24E-04
Cs	5.00E-02	4.41E-02	3.18E-03	3.04E-03	2.93E-03	2.94E-03	2.87E-03
Other FP	5.22E-01	4.56E-01	5.88E-02	5.74E-02	5.40E-02	5.48E-02	5.30E-02
Total FP	0.6401	0.5597	0.0646	0.0629	0.0594	0.0601	0.0582

Table C-7. IMF/Am Target Recipes for Input Fresh Fuel

Fuel type	IMF				
Composition	Np Pu with Am targets (blended core)				
Cycle	1	2	3	4	5
Burnup (GWth-day/tonne-HM)					
U234	0.0003	0.0003	0.0003	0.0003	0.0003
U235	0.0418	0.0417	0.0417	0.0416	0.0416
U236	0.0000	0.0000	0.0000	0.0000	0.0000
U238	0.9408	0.9398	0.9383	0.9371	0.9361
Np237	0.0008	0.0009	0.0009	0.0009	0.0009
Pu238	0.0004	0.0010	0.0011	0.0012	0.0012
Pu239	0.0078	0.0059	0.0058	0.0059	0.0059
Pu240	0.0035	0.0037	0.0036	0.0037	0.0039
Pu241	0.0019	0.0018	0.0018	0.0018	0.0018
Pu242	0.0011	0.0027	0.0037	0.0045	0.0051
Am241	0.0006	0.0006	0.0006	0.0006	0.0007
Am242m	0.0000	0.0000	0.0000	0.0000	0.0000
Am243	0.0003	0.0009	0.0014	0.0017	0.0019
Cm242	0.0003	0.0003	0.0003	0.0003	0.0003
Cm243	0.0418	0.0417	0.0417	0.0416	0.0416
Cm244	0.0000	0.0000	0.0000	0.0000	0.0000
Cm245	0.9408	0.9398	0.9383	0.9371	0.9361
Cm246	0.0008	0.0009	0.0009	0.0009	0.0009
U	0.9829	0.9818	0.9803	0.9790	0.9780
Np	0.0008	0.0009	0.0009	0.0009	0.0009
Pu	0.0147	0.0150	0.0161	0.0171	0.0179
Am	0.0009	0.0016	0.0020	0.0023	0.0026
Cm	0.9829	0.9818	0.9803	0.9790	0.9780
Fission products	0	0	0	0	0

Table C-8. IMF/Am Target Recipes for 5-years after Discharge

Fuel type	IMF				
Composition	Np Pu with Am targets (blended core)				
Cycle	1	2	3	4	5
Burnup (GWth-day/tonne-HM)	**66.1**	**63.4**	**60.7**	**60.6**	**59.2**
Ra226					
Ra228					
Ac227					
Ac228					
Th228					
Th229					
Th230					
Th231					
Th232					
Th234					
Pa231					
Pa233	3.06E-11	3.09E-11	3.05E-11	3.06E-11	3.05E-11
U232	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Composition	Np Pu with Am targets (blended core)				
Cycle	1	2	3	4	5
U233	4.58E-09	5.10E-09	5.41E-09	4.57E-09	4.92E-09
U234	1.99E-04	2.15E-04	2.23E-04	2.28E-04	2.29E-04
U235	6.19E-03	6.52E-03	7.21E-03	7.27E-03	7.38E-03
U236	5.63E-03	5.58E-03	5.52E-03	5.50E-03	5.50E-03
U238	9.03E-01	9.02E-01	9.04E-01	9.03E-01	9.02E-01
Np237	9.00E-04	9.11E-04	8.97E-04	9.00E-04	8.97E-04
Pu238	9.76E-04	1.13E-03	1.18E-03	1.21E-03	1.24E-03
Pu239	5.89E-03	5.86E-03	5.91E-03	5.94E-03	6.01E-03
Pu240	3.66E-03	3.66E-03	3.72E-03	3.89E-03	3.90E-03
Pu241	1.84E-03	1.78E-03	1.80E-03	1.83E-03	1.86E-03
Pu242	2.70E-03	3.73E-03	4.48E-03	5.06E-03	5.52E-03
Pu244	6.69E-07	1.41E-06	2.16E-06	2.91E-06	3.62E-06
Am241	6.40E-04	6.23E-04	6.34E-04	6.51E-04	6.67E-04
Am242m	3.01E-06	3.01E-06	3.15E-06	3.36E-06	3.55E-06
Am242	0	0		0	0
Am243	9.28E-04	1.35E-03	1.65E-03	1.90E-03	2.14E-03
Cm242	2.61E-08	2.65E-08	2.77E-08	2.87E-08	3.02E-08
Cm243	9.46E-06	1.01E-05	9.73E-06	9.79E-06	9.74E-06
Cm244	6.80E-04	1.06E-03	1.22E-03	1.36E-03	1.46E-03
Cm245	1.56E-04	2.21E-04	2.23E-04	2.37E-04	2.57E-04
Cm246	3.93E-05	5.56E-05	5.49E-05	5.55E-05	5.96E-05
Cm247	9.42E-07	1.37E-06	1.29E-06	1.27E-06	1.34E-06
Cm248	1.12E-07	1.52E-07	1.37E-07	1.37E-07	1.40E-07
C14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	7.79E-04	7.61E-04	7.40E-04	7.38E-04	7.35E-04
Zr93	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Tc99	1.45E-03	1.41E-03	1.36E-03	1.36E-03	1.36E-03
I129	3.67E-04	3.51E-04	3.37E-04	3.37E-04	3.37E-04
Cs135	7.99E-04	7.49E-04	7.26E-04	7.28E-04	7.34E-04
Cs137	2.04E-03	1.96E-03	1.89E-03	1.89E-03	1.89E-03
Ra					
Ac					
Th					
Pa	3.06E-11	3.09E-11	3.05E-11	3.06E-11	3.05E-11
U	9.15E-01	9.15E-01	9.17E-01	9.16E-01	9.15E-01
Np	9.00E-04	9.11E-04	8.97E-04	9.00E-04	8.97E-04
Pu	1.51E-02	1.62E-02	1.71E-02	1.79E-02	1.85E-02
Am	1.57E-03	1.98E-03	2.29E-03	2.55E-03	2.81E-03
Cm	8.86E-04	1.35E-03	1.51E-03	1.67E-03	1.79E-03
Other Actinides	1.12E-03	1.41E-03	0.00E+00	0.00E+00	0.00E+00
Total actinides	0.9345	0.9366	0.9392	0.9391	0.9393
Sr	7.79E-04	7.61E-04	7.40E-04	7.38E-04	7.36E-04
Zr					
Tc	1.45E-03	1.41E-03	1.36E-03	1.36E-03	1.36E-03
I	3.67E-04	3.51E-04	3.37E-04	3.37E-04	3.37E-04
Cs	3.21E-03	3.07E-03	2.96E-03	2.96E-03	2.96E-03
Other FP	5.97E-02	5.92E-02	5.54E-02	5.55E-02	5.54E-02

Total FP	0.0655	0.0648	0.0608	0.0609	0.0607

Table C-9. MOX Recipes for Input Fresh Fuel

Fuel type	MOX								
Composition	Np Pu	Np Pu Am							
Cycle	1	1	2	3	4	5	6	7	8
Burnup (GWth-day/ tonne-HM)	51	51	51	51	51	51	51	51	51
U234	0.0002	0.0002	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001	0.0001
U235	0.0071	0.0068	0.0058	0.0050	0.0043	0.0037	0.0033	0.0030	0.0028
U236	0.0053	0.0050	0.0043	0.0037	0.0032	0.0028	0.0025	0.0022	0.0021
U238	0.8563	0.8139	0.6966	0.5970	0.5152	0.4482	0.3959	0.3585	0.3358
Np237	0.0066	0.0083	0.0111	0.0122	0.0122	0.0117	0.0106	0.0090	0.0069
Pu238	0.0033	0.0041	0.0183	0.0351	0.0513	0.0657	0.0765	0.0826	0.0831
Pu239	0.0661	0.0835	0.1124	0.1315	0.1425	0.1480	0.1502	0.1510	0.1516
Pu240	0.0313	0.0395	0.0743	0.1060	0.1345	0.1598	0.1808	0.1967	0.2068
Pu241	0.0147	0.0186	0.0270	0.0286	0.0280	0.0272	0.0274	0.0294	0.0337
Pu242	0.0092	0.0116	0.0247	0.0380	0.0506	0.0623	0.0722	0.0799	0.0849
Am241		0.0059	0.0181	0.0309	0.0418	0.0504	0.0565	0.0601	0.0613
Am242m		0.0000	0.0003	0.0007	0.0012	0.0016	0.0018	0.0020	0.0020
Am243		0.0026	0.0070	0.0112	0.0151	0.0187	0.0221	0.0256	0.0289
U	0.8689	0.8259	0.7069	0.6058	0.5228	0.4548	0.4018	0.3638	0.3407
Np	0.0066	0.0083	0.0111	0.0122	0.0122	0.0117	0.0106	0.0090	0.0069
Pu	0.1245	0.1572	0.2566	0.3392	0.4070	0.4630	0.5071	0.5395	0.5602
Am	0.0000	0.0086	0.0254	0.0428	0.0581	0.0706	0.0805	0.0877	0.0922

Table C-10. MOX Recipes for 5-years after Discharge

Fuel type	MOX								
Composition	Np Pu	Np Pu Am							
Cycle	1	1	2	3	4	5	6	7	8
Burnup (GWth-day/ tonne-HM)	51	51	51	51	51	51	51	51	51
Ra226	2.81E-13	3.29E-13	7.63E-13	1.28E-12	1.80E-12	2.26E-12	0.00E+00	0.00E+00	0.00E+00
Ra228	3.07E-19	2.92E-19	2.51E-19	2.17E-19	1.88E-19	1.65E-19	0.00E+00	0.00E+00	0.00E+00
Ac227	9.02E-14	9.52E-14	1.87E-13	2.97E-13	4.06E-13	5.03E-13	0.00E+00	0.00E+00	0.00E+00
Ac228	3.20E-23	3.05E-23	2.62E-23	2.26E-23	1.97E-23	1.72E-23	0.00E+00	0.00E+00	0.00E+00
Th228	1.34E-11	1.36E-11	2.40E-11	3.65E-11	4.85E-11	5.91E-11	0.00E+00	0.00E+00	0.00E+00
Th229	1.74E-12	1.85E-12	3.09E-12	4.47E-12	5.75E-12	6.85E-12	0.00E+00	0.00E+00	0.00E+00
Th230	7.77E-09	9.79E-09	2.44E-08	4.16E-08	5.85E-08	7.36E-08	0.00E+00	0.00E+00	0.00E+00
Th231	1.77E-14	1.82E-14	1.76E-14	1.62E-14	1.47E-14	1.34E-14	0.00E+00	0.00E+00	0.00E+00
Th232	1.62E-09	1.54E-09	1.33E-09	1.15E-09	1.00E-09	8.79E-10	0.00E+00	0.00E+00	0.00E+00
Th234	1.20E-11	1.14E-11	9.81E-12	8.40E-12	7.25E-12	6.31E-12	0.00E+00	0.00E+00	0.00E+00
Pa231	7.30E-10	7.75E-10	1.54E-09	2.46E-09	3.36E-09	4.17E-09	0.00E+00	0.00E+00	0.00E+00
Pa233	1.48E-10	1.94E-10	2.81E-10	3.23E-10	3.35E-10	3.30E-10	3.06E-10	2.64E-10	2.04E-10
U232	5.67E-10	5.76E-10	1.02E-09	1.55E-09	2.06E-09	2.51E-09	0.00E+00	0.00E+00	0.00E+00
U233	1.77E-08	2.33E-08	3.98E-08	5.37E-08	6.48E-08	7.32E-08	6.82E-08	7.67E-08	8.69E-08
U234	5.05E-04	6.92E-04	1.77E-03	3.00E-03	4.19E-03	5.25E-03	3.40E-03	3.99E-03	4.68E-03
U235	4.35E-03	4.47E-03	4.33E-03	3.98E-03	3.63E-03	3.30E-03	3.15E-03	2.94E-03	2.81E-03
U236	5.12E-03	4.88E-03	4.23E-03	3.68E-03	3.23E-03	2.86E-03	2.60E-03	2.38E-03	2.25E-03
U238	8.28E-01	7.88E-01	6.75E-01	5.79E-01	4.99E-01	4.34E-01	4.02E-01	3.64E-01	3.40E-01
Np237	4.37E-03	5.72E-03	8.27E-03	9.50E-03	9.86E-03	9.71E-03	9.10E-03	7.84E-03	6.07E-03

Composition	Np Pu	Np Pu Am							
Cycle	1	1	2	3	4	5	6	7	8
Pu238	6.07E-03	9.23E-03	2.31E-02	3.81E-02	5.22E-02	6.44E-02	7.86E-02	8.45E-02	8.57E-02
Pu239	4.11E-02	5.68E-02	8.65E-02	1.06E-01	1.18E-01	1.24E-01	1.33E-01	1.33E-01	1.31E-01
Pu240	2.86E-02	3.75E-02	6.98E-02	9.99E-02	1.27E-01	1.51E-01	1.78E-01	1.94E-01	2.05E-01
Pu241	1.15E-02	1.36E-02	1.88E-02	2.08E-02	2.16E-02	2.20E-02	3.06E-02	3.30E-02	3.71E-02
Pu242	9.88E-03	1.24E-02	2.50E-02	3.76E-02	4.94E-02	6.04E-02	7.35E-02	8.16E-02	8.75E-02
Pu244	2.19E-07	2.25E-07	2.77E-07	3.12E-07	3.40E-07	3.65E-07	4.14E-07	4.58E-07	5.16E-07
Am241	5.24E-03	9.06E-03	2.01E-02	3.06E-02	3.94E-02	4.64E-02	4.81E-02	5.13E-02	5.25E-02
Am242m	4.92E-05	1.51E-04	4.68E-04	8.36E-04	1.19E-03	1.49E-03	1.86E-03	2.03E-03	2.11E-03
Am242	5.89E-10	1.81E-09	5.60E-09	1.00E-08	1.42E-08	1.79E-08	9.55E-06	1.07E-05	1.23E-05
Am243	2.08E-03	3.52E-03	7.34E-03	1.12E-02	1.48E-02	1.82E-02	2.25E-02	2.60E-02	2.95E-02
Cm242	2.37E-07	6.16E-07	1.58E-06	2.61E-06	3.54E-06	4.34E-06	1.85E-03	1.84E-03	1.75E-03
Cm243	1.12E-05	2.89E-05	5.10E-05	6.47E-05	7.32E-05	7.84E-05	9.53E-05	9.38E-05	8.87E-05
Cm244	8.41E-04	1.68E-03	2.68E-03	3.38E-03	3.96E-03	4.45E-03	6.17E-03	6.59E-03	6.89E-03
Cm245	1.27E-04	2.97E-04	4.18E-04	4.73E-04	5.10E-04	5.41E-04	5.99E-04	6.35E-04	6.78E-04
Cm246	6.14E-06	1.44E-05	1.44E-05	1.35E-05	1.30E-05	1.28E-05	1.40E-05	1.57E-05	1.88E-05
Cm247	1.21E-07	3.26E-07	3.19E-07	2.94E-07	2.79E-07	2.73E-07	2.98E-07	3.38E-07	4.08E-07
Cm248	9.16E-09	2.84E-08	2.70E-08	2.43E-08	2.29E-08	2.21E-08	2.42E-08	2.76E-08	3.38E-08
C14	3.96E-11	3.95E-11	3.95E-11	3.94E-11	3.94E-11	3.94E-11	0.00E+00	0.00E+00	0.00E+00
Sr90	3.62E-04	3.58E-04	3.49E-04	3.46E-04	3.44E-04	3.43E-04	4.04E-04	4.04E-04	4.03E-04
Zr93	7.71E-04	7.68E-04	7.60E-04	7.58E-04	7.58E-04	7.58E-04	0.00E+00	0.00E+00	0.00E+00
Tc99	1.12E-03	1.12E-03	1.13E-03	1.13E-03	1.14E-03	1.14E-03	1.19E-03	1.19E-03	1.19E-03
I129	3.52E-04	3.55E-04	3.56E-04	3.59E-04	3.63E-04	3.65E-04	3.82E-04	3.83E-04	3.82E-04
Cs135	9.36E-04	9.44E-04	9.61E-04	9.73E-04	9.81E-04	9.87E-04	1.04E-03	1.04E-03	1.04E-03
Cs137	1.63E-03	1.63E-03	1.63E-03	1.63E-03	1.64E-03	1.64E-03	1.93E-03	1.93E-03	1.93E-03
Ra	3.50E-13	3.99E-13	8.87E-13	1.47E-12	2.05E-12	2.57E-12	0.00E+00	0.00E+00	0.00E+00
Ac	9.02E-14	9.52E-14	1.87E-13	2.98E-13	4.06E-13	5.03E-13	0.00E+00	0.00E+00	0.00E+00
Th	9.42E-09	1.14E-08	2.58E-08	4.28E-08	5.95E-08	7.45E-08	0.00E+00	0.00E+00	0.00E+00
Pa	8.79E-10	9.69E-10	1.82E-09	2.78E-09	3.70E-09	4.50E-09	3.06E-10	2.64E-10	2.04E-10
U	8.38E-01	7.98E-01	6.86E-01	5.90E-01	5.11E-01	4.46E-01	4.12E-01	3.73E-01	3.50E-01
Np	4.37E-03	5.72E-03	8.27E-03	9.50E-03	9.86E-03	9.71E-03	9.10E-03	7.84E-03	6.07E-03
Pu	9.71E-02	1.30E-01	2.23E-01	3.02E-01	3.68E-01	4.22E-01	4.93E-01	5.26E-01	5.46E-01
Am	7.37E-03	1.27E-02	2.79E-02	4.27E-02	5.54E-02	6.61E-02	7.25E-02	7.94E-02	8.41E-02
Cm	9.86E-04	2.02E-03	3.17E-03	3.94E-03	4.56E-03	5.08E-03	8.73E-03	9.17E-03	9.43E-03
Other Actinides	3.76E-05	8.85E-05	1.71E-04	2.38E-04	2.90E-04	3.31E-04	0.00E+00	0.00E+00	0.00E+00
Total actinides	0.9482	0.9483	0.9483	0.9484	0.9484	0.9485	0.9951	0.9951	0.9951
Sr	6.39E-04	6.30E-04	6.14E-04	6.09E-04	6.05E-04	6.03E-04	4.04E-04	4.04E-04	4.03E-04
Zr	3.80E-03	3.78E-03	3.74E-03	3.73E-03	3.73E-03	3.73E-03	0.00E+00	0.00E+00	0.00E+00
Tc	1.12E-03	1.12E-03	1.13E-03	1.13E-03	1.14E-03	1.14E-03	1.19E-03	1.19E-03	1.19E-03
I	4.70E-04	4.73E-04	4.74E-04	4.78E-04	4.82E-04	4.85E-04	3.82E-04	3.83E-04	3.82E-04
Cs	4.22E-03	4.23E-03	4.26E-03	4.28E-03	4.30E-03	4.31E-03	2.97E-03	2.97E-03	2.97E-03
Other FP	4.15E-02	4.15E-02	4.15E-02	4.14E-02	4.14E-02	4.13E-02	0.00E+00	0.00E+00	0.00E+00
Total FP	0.0518	0.0517	0.0517	0.0516	0.0516	0.0516	0.0049	0.0049	0.0049

C.3 Transmutation in fast reactor systems

Transmutation in low conversion ratio fast reactor is based on a compact fast burner reactor design that can achieve low conversion ratios.[Smith2003] This design is the basis for all transmutation options that used TRU from UOX, MOX or IMF spent fuel into a burner fast reactor. The other type of fast reactors used in this study, that is the breeder fast reactor, has a different design from the consumer fast reactor.[Hill2002]

The ANL suite of fast reactor analysis codes was used to evaluate reactor operating parameters of either fast reactor designs. Specifically, the MC2-2, REBUS-3, and DIF3D codes were used. For each fuel composition, the MC2-2 code [MC2] is used to obtain regional group constants based on ENDF-V data by performing a critical buckling search (fundamental mode calculation). REBUS-3 is a fuel cycle analysis code [REBUS] for fast reactors which couples the DIF3D multigroup neutron flux code system [DIF3D] to a multigroup depletion code. In those designs the enrichment search option of the REBUS-3 code is used to compute equilibrium cycle compositions for each reactor design. The REBUS-3 code takes the user defined TRU feed (recycled transuranics from UOX, MOX, or IMF), the base feed (depleted uranium), the reactor operating cycle, and the fuel loading scheme and determines the necessary fuel enrichment and equilibrium discharge compositions (spent fuel composition) to assure criticality at the end of cycle (EOC). To get the detailed composition for key isotopes at discharge or a number of years after discharge, ORIGEN2 depletion calculations are performed using a one group cross-section set that is provided by the detailed REBUS-3/DI3D calculations. Thus, for each TRU isotopic vector from UOX, MOX, or IMF, the detailed MC2-2 and REBUS-3/DIF3D calculations, followed by the ORIGEN-2 depletion calculations are performed to provide the spent fuel vector for both startup and equilibrium cores of the fast reactors. Those vectors (recipes) are given in Tables C-11, 12, 13, 14.[Stillman2004, Hoffmann2004, Hoffmann2005a]

Table C-11. CFR Recipes for Input Fresh Fuel

Fuel type	Consumer Fast Reactor (CFR)					
System	UOX/CFR symbiosis		MOX/CFR symbiosis		IMF/CFR symbiosis	
Cycle	Startup	Equilibrium	Startup	Equilibrium	Startup	Equilibrium
Burnup (GWth-day/tonne-HM)	176.6	176.9	175.6	176.3	117.55	128.28
U234	0.0001	0.0030	0.0001	0.0066	0.0001	0.0045
U235	0.0042	0.0012	0.0021	0.0017	0.0032	0.0015
U236	0.0031	0.0025	0.0015	0.0023	0.0024	0.0026
U238	0.5030	0.3661	0.2512	0.1969	0.3879	0.3778
Np237	0.0233	0.0165	0.0222	0.0131	0.0209	0.0096
Pu238	0.0116	0.0267	0.0270	0.0588	0.0547	0.0393
Pu239	0.2333	0.1819	0.1824	0.1163	0.0207	0.0575
Pu240	0.1104	0.1952	0.1361	0.1828	0.1265	0.1363
Pu241	0.0520	0.0438	0.0513	0.0359	0.0525	0.0270
Pu242	0.0323	0.0723	0.0439	0.0862	0.1596	0.1531
Am241	0.0166	0.0249	0.1793	0.1008	0.0696	0.0361
Am242m	0.0000	0.0142	0.0005	0.0587	0.0002	0.0236
Am243	0.0074	0.0237	0.0800	0.0712	0.0608	0.0538
Cm242	0.0000	0.0001	0.0000	0.0003	0.0000	0.0001
Cm243	0.0000	0.0001	0.0002	0.0004	0.0003	0.0002
Cm244	0.0026	0.0154	0.0197	0.0442	0.0317	0.0447
Cm245	0.0002	0.0041	0.0026	0.0119	0.0032	0.0117
Cm246	0.0000	0.0022	0.0000	0.0059	0.0000	0.0059
U	0.5104	0.3728	0.2549	0.2075	0.3936	0.3864
Np	0.0233	0.0165	0.0222	0.0131	0.0209	0.0096
Pu	0.4396	0.5199	0.4407	0.4801	0.4139	0.4132
Am	0.0240	0.0629	0.2598	0.2307	0.1306	0.1136
Cm	0.0029	0.0219	0.0225	0.0627	0.0352	0.0626
Fission products	0.0000	0.0059	0.0000	0.0059	0.0059	0.0043

Table C-12. CFR Recipes for 5-years after Discharge

Fuel type	Consumer Fast Reactor (CFR)					
System	UOX/CFR symbiosis		MOX/CFR symbiosis		IMF/CFR symbiosis	
Cycle	Startup	Equilibrium	Startup	Equilibrium	Startup	Equilibrium
Burnup (GWth-day/ tonne-HM)	176.6	176.9	175.6	176.3	117.55	128.28
Ra226	2.65E-13	2.00E-12	5.65E-13	4.33E-12	7.30E-13	3.12E-12
Ra228	9.72E-20	7.40E-20	4.96E-20	7.08E-20	7.62E-20	8.36E-20
Ac227	1.34E-14	1.64E-13	1.56E-14	3.52E-13	2.49E-14	2.60E-13
Ac228	1.02E-23	7.73E-24	5.17E-24	7.39E-24	7.96E-24	8.73E-24
Th228	5.47E-09	5.06E-09	6.39E-09	5.09E-09	6.17E-09	3.85E-09
Th229	3.25E-11	4.13E-11	3.42E-11	4.28E-11	3.26E-11	3.28E-11
Th230	1.10E-08	6.30E-08	2.69E-08	1.37E-07	3.18E-08	9.77E-08
Th231	7.74E-15	3.68E-15	4.33E-15	6.11E-15	6.99E-15	5.06E-15
Th232	6.59E-10	5.00E-10	3.38E-10	4.80E-10	5.18E-10	5.64E-10
Th234	6.46E-12	4.69E-12	3.26E-12	2.52E-12	5.09E-12	5.19E-12
Pa231	1.22E-10	1.42E-09	1.43E-10	3.06E-09	2.27E-10	2.26E-09
Pa233	4.56E-10	3.31E-10	5.05E-10	2.91E-10	4.72E-10	2.24E-10
U232	2.90E-07	2.64E-07	3.41E-07	2.67E-07	3.29E-07	2.02E-07
U233	4.37E-08	1.17E-07	5.29E-08	2.35E-07	5.95E-08	1.70E-07

System	UOX/CFR symbiosis		MOX/CFR symbiosis		IMF/CFR symbiosis	
Cycle	Startup	Equilibrium	Startup	Equilibrium	Startup	Equilibrium
U234	9.67E-04	3.56E-03	2.71E-03	7.80E-03	2.87E-03	5.40E-03
U235	1.91E-03	9.11E-04	1.07E-03	1.51E-03	1.73E-03	1.25E-03
U236	2.99E-03	2.27E-03	1.56E-03	2.20E-03	2.36E-03	2.55E-03
U238	4.49E-01	3.26E-01	2.27E-01	1.75E-01	3.54E-01	3.61E-01
Np237	1.32E-02	9.59E-03	1.47E-02	8.43E-03	1.37E-02	6.51E-03
Pu238	1.55E-02	2.32E-02	5.08E-02	5.32E-02	4.68E-02	3.18E-02
Pu239	1.42E-01	1.12E-01	1.12E-01	7.59E-02	3.72E-02	5.75E-02
Pu240	1.03E-01	1.64E-01	1.24E-01	1.57E-01	1.09E-01	1.26E-01
Pu241	2.18E-02	2.47E-02	2.35E-02	2.15E-02	2.33E-02	1.70E-02
Pu242	3.10E-02	6.28E-02	4.76E-02	7.71E-02	1.38E-01	1.34E-01
Pu244	2.20E-07	4.63E-07	2.45E-07	5.22E-07	6.99E-07	9.48E-07
Am241	1.82E-02	2.34E-02	1.12E-01	6.36E-02	5.13E-02	2.79E-02
Am242m	1.14E-03	1.37E-02	9.63E-03	5.67E-02	3.68E-03	2.43E-02
Am242	1.37E-08	1.65E-07	1.16E-07	6.81E-07	4.42E-08	2.92E-07
Am243	7.81E-03	2.15E-02	5.69E-02	5.34E-02	5.33E-02	5.74E-02
Cm242	3.18E-06	3.37E-05	2.64E-05	1.39E-04	1.01E-05	5.94E-05
Cm243	5.54E-05	1.08E-04	3.93E-04	3.64E-04	2.14E-04	1.59E-04
Cm244	3.19E-03	1.31E-02	2.38E-02	3.56E-02	2.76E-02	3.76E-02
Cm245	6.24E-04	4.06E-03	4.76E-03	1.13E-02	5.86E-03	1.19E-02
Cm246	3.96E-05	2.18E-03	2.85E-04	5.89E-03	3.48E-04	6.27E-03
Cm247	9.95E-07	1.00E-04	6.24E-06	2.52E-04	7.29E-06	2.63E-04
Cm248	4.17E-08	7.34E-06	2.54E-07	1.83E-05	2.84E-07	1.95E-05
C14	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	1.22E-03	1.16E-03	1.13E-03	1.10E-03	7.18E-04	5.68E-04
Zr93	2.55E-03	2.47E-03	2.41E-03	2.35E-03	1.52E-03	1.19E-03
Tc99	4.44E-03	4.39E-03	4.36E-03	4.31E-03	2.82E-03	2.04E-03
I129	1.37E-03	1.28E-03	1.24E-03	1.16E-03	6.85E-04	5.81E-04
Cs135	7.78E-03	7.76E-03	7.76E-03	7.72E-03	5.07E-03	3.56E-03
Cs137	6.30E-03	6.31E-03	6.31E-03	6.33E-03	4.17E-03	2.88E-03
Ra	2.84E-11	2.80E-11	3.34E-11	3.05E-11	3.25E-11	2.29E-11
Ac	1.35E-14	1.64E-13	1.57E-14	3.53E-13	2.50E-14	2.60E-13
Th	1.72E-08	6.86E-08	3.36E-08	1.43E-07	3.85E-08	1.02E-07
Pa	5.79E-10	1.75E-09	6.48E-10	3.35E-09	6.99E-10	2.48E-09
U	4.55E-01	3.33E-01	2.32E-01	1.87E-01	3.61E-01	3.70E-01
Np	1.32E-02	9.59E-03	1.47E-02	8.43E-03	1.37E-02	6.51E-03
Pu	3.13E-01	3.86E-01	3.58E-01	3.84E-01	3.54E-01	3.67E-01
Am	2.72E-02	5.86E-02	1.79E-01	1.74E-01	1.08E-01	1.10E-01
Cm	3.91E-03	1.95E-02	2.93E-02	5.35E-02	3.40E-02	5.63E-02
Other Actinides	1.23E-04	2.34E-04	8.86E-04	7.23E-04	4.81E-04	4.38E-04
Total actinides	0.8128	0.8075	0.8132	0.8077	0.8712	0.9098
Sr	2.10E-03	2.00E-03	1.94E-03	1.88E-03	1.22E-03	9.77E-04
Zr	1.33E-02	1.29E-02	1.26E-02	1.24E-02	8.03E-03	6.16E-03
Tc	4.44E-03	4.39E-03	4.36E-03	4.31E-03	2.82E-03	2.04E-03
I	1.80E-03	1.69E-03	1.65E-03	1.53E-03	9.04E-04	7.67E-04
Cs	2.10E-02	2.10E-02	2.10E-02	2.09E-02	1.38E-02	9.63E-03
Other FP	1.44E-01	1.51E-01	1.45E-01	1.51E-01	1.02E-01	7.07E-02
Total FP	0.1871	0.1925	0.1867	0.1924	0.1288	0.0902

Table C-13. BFR Recipes for Input Fresh Fuel

Fuel type	Breeder Fast Reactor (BFR)			
System	UOX to BFR		UOX to IMF to BFR	
Cycle	Startup cycle	Equilibrium cycle	Startup cycle	Equilibrium cycle
Burnup (GWth-day/tonne-HM)	66.1	66.1	66.1	66.1
U234	0.0001	0.0002	0.0001	0.0002
U235	0.0039	0.0004	0.0029	0.0004
U236	0.0022	0.0003	0.0014	0.0003
U238	0.8667	0.8859	0.7404	0.8859
Np237	0.0061	0.0007	0.0089	0.0007
Pu238	0.0030	0.0010	0.0233	0.0010
Pu239	0.0606	0.0789	0.0088	0.0789
Pu240	0.0287	0.0258	0.0538	0.0258
Pu241	0.0135	0.0027	0.0223	0.0027
Pu242	0.0084	0.0014	0.0678	0.0014
Am241	0.0043	0.0013	0.0296	0.0013
Am242m	0.0000	0.0001	0.0001	0.0001
Am243	0.0019	0.0003	0.0259	0.0003
Cm242	0.0000	0.0000	0.0000	0.0000
Cm243	0.0000	0.0000	0.0001	0.0000
Cm244	0.0007	0.0002	0.0135	0.0002
Cm245	0.0001	0.0000	0.0014	0.0000
Cm246	0.0000	0.0000	0.0000	0.0000
U	0.8729	0.8868	0.7447	0.8868
Np	0.0061	0.0007	0.0089	0.0007
Pu	0.1142	0.1099	0.1760	0.1099
Am	0.0062	0.0017	0.0555	0.0017
Cm	0.0007	0.0002	0.0150	0.0002
Fission products	0.0000	0.0008	0.0000	0.0008

Table C-14. BFR Recipes for 5-years after Discharge

Fuel type	Breeder Fast Reactor (BFR)			
System	UOX to BFR		UOX to IMF to BFR	
Cycle	Startup cycle	Equilibrium cycle	Startup cycle	Equilibrium cycle
Burnup (GWth-day/tonne-HM)	66.1	66.1	65.7	66.1
Ra226	1.75E-13	1.94E-13	6.09E-13	1.94E-13
Ra228	1.03E-19	1.59E-20	6.95E-20	1.59E-20
Ac227	1.48E-14	1.68E-14	3.15E-14	1.68E-14
Ac228	1.07E-23	1.66E-24	7.25E-24	1.66E-24
Th228	1.85E-09	3.11E-10	3.35E-09	3.11E-10
Th229	2.02E-11	4.19E-12	3.43E-11	4.19E-12
Th230	5.37E-09	4.65E-09	2.12E-08	4.65E-09
Th231	7.95E-15	8.55E-16	6.51E-15	8.55E-16
Th232	5.95E-10	9.00E-11	4.05E-10	9.00E-11
Th234	1.14E-11	1.16E-11	1.01E-11	1.16E-11
Pa231	1.22E-10	1.31E-10	2.64E-10	1.31E-10
Pa233	1.42E-10	2.50E-11	2.18E-10	2.50E-11
U232	9.24E-08	1.54E-08	1.67E-07	1.54E-08
U233	1.86E-08	6.75E-09	3.81E-08	6.75E-09

System	UOX to BFR		UOX to IMF to BFR	
Cycle	Startup cycle	Equilibrium cycle	Startup cycle	Equilibrium cycle
U234	3.44E-04	2.02E-04	1.54E-03	2.02E-04
U235	1.97E-03	2.11E-04	1.61E-03	2.11E-04
U236	2.20E-03	3.28E-04	1.51E-03	3.28E-04
U238	7.95E-01	8.08E-01	7.00E-01	8.08E-01
Np237	4.11E-03	7.26E-04	6.32E-03	7.26E-04
Pu238	3.98E-03	1.02E-03	2.00E-02	1.02E-03
Pu239	7.55E-02	8.53E-02	4.41E-02	8.53E-02
Pu240	2.89E-02	2.80E-02	4.98E-02	2.80E-02
Pu241	5.34E-03	2.46E-03	9.39E-03	2.46E-03
Pu242	8.05E-03	1.57E-03	5.99E-02	1.57E-03
Pu244	2.86E-08	5.46E-09	1.79E-07	5.46E-09
Am241	5.39E-03	1.87E-03	2.30E-02	1.87E-03
Am242m	1.80E-04	7.78E-05	9.77E-04	7.78E-05
Am242	2.16E-09	9.34E-10	1.17E-08	9.34E-10
Am243	2.00E-03	3.60E-04	2.32E-02	3.60E-04
Cm242	5.07E-07	2.10E-07	2.70E-06	2.10E-07
Cm243	8.32E-06	2.91E-06	7.20E-05	2.91E-06
Cm244	7.26E-04	1.47E-04	1.11E-02	1.47E-04
Cm245	1.38E-04	3.93E-05	2.41E-03	3.93E-05
Cm246	8.12E-06	1.43E-05	1.41E-04	1.43E-05
Cm247	1.66E-07	5.32E-07	2.67E-06	5.32E-07
Cm248	6.90E-09	3.80E-08	1.06E-07	3.80E-08
C14	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr90	4.78E-04	4.90E-04	3.20E-04	4.90E-04
Zr93	9.70E-04	1.01E-03	6.47E-04	1.01E-03
Tc99	1.60E-03	1.68E-03	1.08E-03	1.68E-03
I129	4.83E-04	5.32E-04	2.86E-04	5.32E-04
Cs135	2.73E-03	2.86E-03	1.84E-03	2.86E-03
Cs137	2.15E-03	2.25E-03	1.46E-03	2.25E-03
Ra	9.69E-12	1.79E-12	1.78E-11	1.79E-12
Ac	1.48E-14	1.68E-14	3.16E-14	1.68E-14
Th	7.84E-09	5.07E-09	2.50E-08	5.07E-09
Pa	2.64E-10	1.56E-10	4.82E-10	1.56E-10
U	8.00E-01	8.09E-01	7.04E-01	8.09E-01
Np	4.11E-03	7.26E-04	6.32E-03	7.26E-04
Pu	1.22E-01	1.18E-01	1.83E-01	1.18E-01
Am	7.57E-03	2.31E-03	4.72E-02	2.31E-03
Cm	8.82E-04	2.04E-04	1.38E-02	2.04E-04
Other Actinides	3.45E-05	1.06E-05	2.34E-04	1.06E-05
Total actinides	0.9341	0.9304	0.9552	0.9304
Sr	8.27E-04	8.49E-04	5.50E-04	8.49E-04
Zr	5.04E-03	5.25E-03	3.38E-03	5.25E-03
Tc	1.60E-03	1.68E-03	1.08E-03	1.68E-03
I	6.32E-04	6.96E-04	3.73E-04	6.96E-04
Cs	7.33E-03	7.67E-03	4.97E-03	7.67E-03
Other FP	5.04E-02	5.34E-02	3.44E-02	5.34E-02
Total FP	0.0659	0.0696	0.0448	0.0696

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APPENDIX D. ISOTOPE COEFFICIENTS USED IN CALCULATING METRICS

This appendix documents the isotope coefficients and parameters used in this study for long-term heat (LTH), decay energy, long-term dose (LTD), long-term radiotoxicity (LTR), ingestion and inhalation dose conversion factors, neutrons per second, and bare-sphere critical masses.

D.1. Long-term heat and decay energy

The long-term heat (LTH) is defined as the heat released by waste in the period between when ventilation of the repository stops (i.e. repository closure) and 1500 years. The start of this interval is, by definition, when heat is no longer actively removed from the repository. The end of this interval is the time of approximate between-drift temperatures in Wigeland's relatively early calculations of repository thermal response. The LTH is simply the time integral of the inventory of each isotope, and its daughter isotopes, times the decay heat (watts/gram) of those isotopes. Accounting for heat from daughters is often critical, e.g., Pu241 decaying to Am241. Table D-1 provides the values calculated for this study, the units are watts-years per gram of isotope at time of emplacement. We calculated the most important coefficients two ways – by a system dynamic simulation of the isotope decay changes and by simple spreadsheet approximations. Also, the values for Pu238, Pu239, Pu240, Pu241, and Am241 matched those provided by ANL colleagues within 1%. The decay energy values used are shown in Table D-2; we took them from an official waste management assessment at Hanford. Note that for fission products, Cs-137/Ba-137m and Sr-90/Y-90, the Hanford data base includes the contribution of the short-lived daughter, so that we did not have to account separately for them.

Table D-1. Long-Term Heat (LTH) Interval Values

Unit	Long-Term Heat Intervals (Watts-years/g at time of emplacement)		
Interval	50 to 1500 years	100 to 1500 years	300 to 1500 years
Ra 226	70.29	67.26	55.04
Ra 228	0.28	0.00	0.00
Ac 227	0.23	0.05	0.00
Ac 228	0.00	0.00	0.00
Th 228	0.00	0.00	0.00
Th 229	8.77	8.44	7.17
Th 230	1.30	1.27	1.13
Th 231	2.09	2.02	1.73
Th 232	0.00	0.00	0.00
Th 234	0.26	0.25	0.22
Pa 231	2.09	2.02	1.73
Pa 233	0.44	0.42	0.36
U 232	92.83	57.35	8.36
U 233	0.44	0.42	0.36
U 234	0.26	0.25	0.22
U 235	0.00	0.00	0.00
U 236	0.00	0.00	0.00
U 238	0.00	0.00	0.00
Np 237	0.03	0.03	0.03
Pu 238	48.81	32.95	6.95
Pu 239	2.74	2.64	2.26
Pu 240	9.46	9.11	7.73
Pu 241	61.48	56.42	39.15
Pu 242	0.17	0.16	0.14
Pu 244	0.00	0.00	0.00
Am 241	59.65	54.57	37.84
Am 242m	137.70	117.77	56.04
Am 242	49.09	33.14	6.99
Am 243	8.89	8.57	7.29
Cm 242	49.05	33.12	6.98
Cm 243	24.95	9.23	2.32
Cm 244	20.52	10.76	7.75
Cm 245	12.89	12.58	11.17
Cm 246	13.09	12.59	10.63
Cm 247	0.00	0.00	0.00
Cm 248	0.76	0.74	0.63
[redacted]			
Sr-90	11.88	3.61	0.03
Tc-99	0.01	0.01	0.01
I-129	0.00	0.00	0.00
Cs-135	0.00	0.00	0.00
Cs-137	5.69	1.79	0.02

Table D-2. Decay Energy Parameters

			Table A-1 Conversion Factors for Generational Radiological Calculations, from the Hanford Site Solid Waste Acceptance Criteria, HNF-EP-0063 Rev 11
	Chain	Mass	Decay energy (W/g)
Ra 226	N+2	226	2.85E-02
Ra 228	N	228	3.79E-02
Ac 227	N+3	227	3.63E-02
Ac 228	N	228	not included
Th 228	N	228	2.68E+01
Th 229	N+1	229	6.50E-03
Th 230	N+2	230	5.82E-04
Th 231	N+3	231	not included
Th 232	N	232	2.66E-09
Th 234	N+2	234	9.88E+00
Pa 231	N+3	231	1.44E-03
Pa 233	N+1	233	not included
U 232	N	232	7.08E-01
U 233	N+1	233	2.81E-04
U 234	N+2	234	1.79E-04
U 235	N+3	235	5.99E-08
U 236	N	236	1.75E-06
U 238	N+2	238	8.51E-09
Np 237	N+1	237	2.07E-05
Pu 238	N+2	238	5.68E-01
Pu 239	N+3	239	1.93E-03
Pu 240	N	240	7.07E-03
Pu 241	N+1	241	3.27E-03
Pu 242	N+2	242	1.17E-04
Pu 244	N	244	5.33E-07
Am 241	N+1	241	1.15E-01
Am 242m	N+2	242	4.49E-03
Am 242	N+2	242	not included
Am 243	N+3	243	6.44E-03
Cm 242	N+2	242	1.22E+02
Cm 243	N+3	243	1.81E+00
Cm 244	N	244	2.83E+00
Cm 245	N+1	245	5.72E-03
Cm 246	N+2	246	1.01E-02
Cm 247	N+3	247	2.87E-06
Cm 248	N	248	5.27E-04
Sr-90		90	9.29E-01
Zr-93		93	2.84E-07
Tc-99		99	1.02E-05
I-129		129	8.19E-08
Cs-135		135	4.38E-07
Cs-137		137	4.17E-01

D.2. Long-term dose

R. Halsey (LLNL) has provided late-2004 DOE-RW dose numbers from the nominal Yucca Mountain scenario, listed in Table D-3. Compared to previous results, these numbers (which may continue to evolve) show lower long-term dose to the maximally exposed individual of the public, in part because the transportation of Pu and Np is lower than previously estimated. Figure D-1 graphs these results by grouping the uranium and TRU isotopes into the four decay chains.

Table D-3. YMP Dose Results provided by DOE-RW in late 2004

Time (yr)	1.00E+04	2.00E+04	5.00E+04	1.00E+05	2.00E+05	5.00E+05	8.00E+05	1.00E+06
Mean Annual Dose, mrem/yr	1.60E-05	2.50E-05	1.50E-05	1.20E-01	1.50E+01	3.10E+01	2.50E+01	2.20E+01
Max of RNs, mrem/yr	1.20E-05	2.10E-05	1.00E-05	8.30E-02	9.50E+00	1.30E+01	7.30E+00	7.50E+00
Am243	4.30E-16	4.60E-16	3.10E-16	6.60E-09	6.70E-09	3.00E-10	2.70E-10	3.10E-10
C14	2.50E-06	1.50E-06	2.20E-08	4.50E-08	2.70E-11	4.60E-19	1.50E-25	1.50E-25
I129	1.10E-06	2.30E-06	2.00E-06	9.90E-04	3.40E-01	2.10E-01	6.20E-02	5.20E-02
Np237	8.80E-08	5.60E-07	1.80E-06	8.30E-02	9.50E+00	1.30E+01	7.30E+00	7.50E+00
Pa231	2.80E-11	3.40E-10	1.20E-08	4.30E-03	5.00E-01	3.30E+00	5.90E+00	5.00E+00
Pu239	5.40E-13	1.20E-12	1.10E-08	9.40E-06	2.60E-01	1.30E-03	2.80E-04	2.90E-04
Pu240	1.30E-13	1.50E-13	7.70E-12	3.30E-09	6.00E-08	6.50E-11	7.60E-12	1.10E-11
Tc99	1.20E-05	2.10E-05	1.00E-05	5.50E-03	8.10E-01	2.10E-01	2.40E-02	1.10E-02
Th229	3.30E-09	2.70E-08	3.40E-07	3.70E-03	7.90E-01	3.10E+00	2.10E+00	1.30E+00
U233	4.70E-09	2.50E-08	2.00E-07	7.70E-03	7.10E-01	8.10E-01	6.40E-01	5.20E-01
U234	2.50E-09	5.80E-08	1.10E-07	5.00E-03	1.00E+00	6.50E-01	2.70E-01	1.90E-01
U236	2.20E-10	4.90E-09	1.10E-08	6.00E-04	1.40E-01	2.00E-01	1.50E-01	9.40E-02
U238	4.30E-10	7.30E-09	1.30E-08	9.40E-04	1.70E-01	1.60E-01	1.20E-01	1.40E-01
Pu242	1.30E-16	3.60E-16	2.00E-11	9.40E-08	7.30E-02	3.20E+00	1.90E+00	1.60E+00
Th230	2.20E-11	7.20E-10	1.40E-08	3.50E-05	3.20E-02	3.50E-01	2.40E-01	1.50E-01
Am241	5.30E-17	2.00E-17	1.60E-18	3.50E-19	4.30E-26	4.30E-26	4.30E-26	4.30E-26
Pu238	1.90E-25	1.80E-25	1.80E-25	1.80E-25	1.80E-25	1.80E-25	1.80E-25	1.80E-25
Ac227	2.40E-11	2.80E-10	9.90E-09	3.60E-03	3.70E-01	2.50E+00	4.60E+00	3.90E+00
Cs137	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26
Pb210	9.20E-11	2.80E-09	6.10E-08	1.50E-04	1.30E-01	1.40E+00	9.30E-01	5.40E-01
Ra226	8.60E-11	3.10E-09	5.80E-08	1.20E-04	1.40E-01	1.50E+00	1.10E+00	7.00E-01
Sr90	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26	5.30E-26
U232	7.30E-26	7.30E-26	7.30E-26	7.30E-26	7.30E-26	7.30E-26	7.30E-26	7.30E-26
Total	1.57E-05	2.55E-05	1.47E-05	1.16E-01	1.50E+01	3.06E+01	2.53E+01	2.17E+01

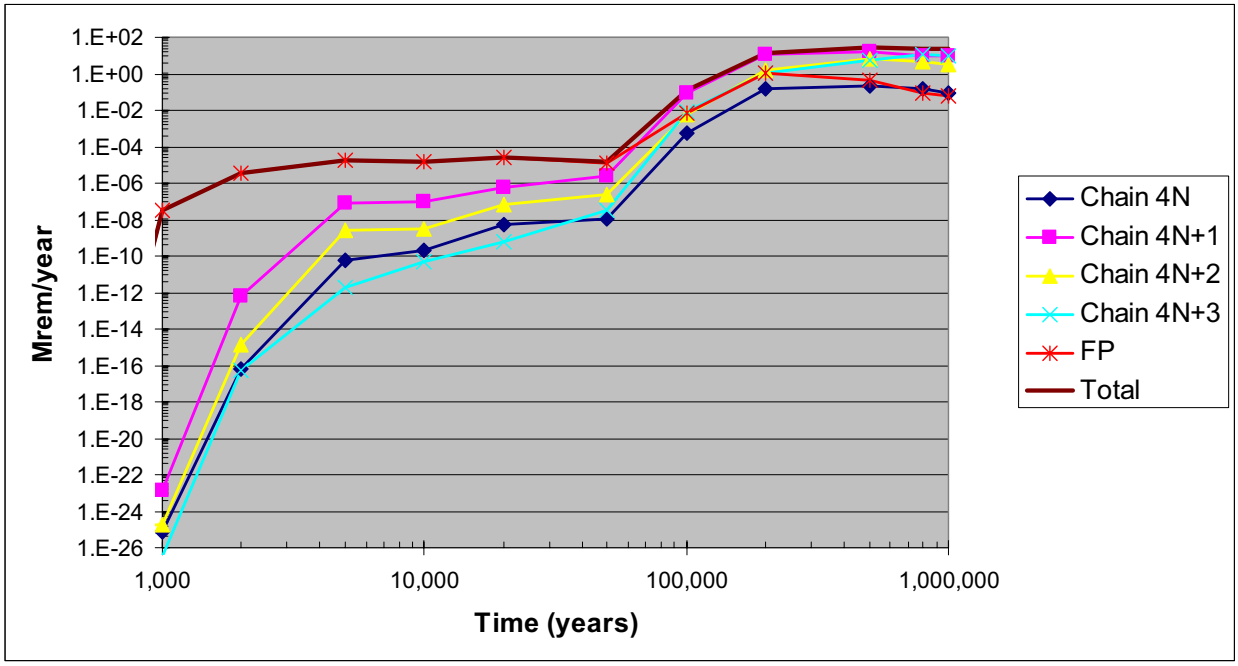

Figure D-1. Hypothetical repository dose, data from Halsey2004

To obtain metrics for this study, we analyzed these results and estimated how much of the mass of each isotope (at 10,000, 50,000, etc. years after emplacement) stems from the mass of each isotope at the time of emplacement. For example, much of the mass of Np-237 (hence the dose from Np-237 at X years after emplacement) stems from Pu-241 and Am-241 at the time of emplacement. By assuming linearity, we can therefore estimate dose/mass of isotopes prior to emplacement. Linearity is, we emphasize, suspect for Np-237 because of Np solubility limits in the YMP dose calculations. Nonetheless, some insights can be gained from this analysis. Indeed, we tend to find that the reduction in LTD is not too far from the reduction in LTH. Assuming that more GW-years worth of waste is emplaced in the repository corresponding to the LTH reduction, the actual amount of Np in the repository does not vary much. So, the Np solubility issue may not significantly impact these results.

Thus, the long-term dose (LTD) is defined as the maximum dose to a member of the public at various times in the future from transportation of isotopes that are released from waste in the YMP repository. Table D-4 provides values for this study, the units are mrem/yr at a future time per gram of isotope at time of emplacement.

We calculated the LTD coefficients two ways – by a system dynamic model of the four decay chains and by simple spreadsheet approximation. We also verified that we arrive back at the starting point when taking the composition for used nuclear fuel times the LTD parameters, i.e., we get back to the totals in Table D-3.

Table D-4. Long-Term Dose (LTD) Coefficients

	Hypothetical dose (mrem/yr) at some future time per g at time of emplacement							
	At 10,000 years	At 20,000 years	At 50,000 years	At 100,000 years	At 200,000 years	At 500,000 years	At 800,000 years	At 1,000,000 years
Pa 233	1.5E-14	4.9E-14	2.0E-13	2.2E-09	1.5E-07	1.6E-07	7.3E-08	4.0E-08
U 233	1.5E-14	4.9E-14	2.0E-13	2.2E-09	1.5E-07	1.6E-07	7.3E-08	4.0E-08
U 234	6.6E-17	1.6E-15	5.9E-15	1.3E-10	3.1E-08	8.6E-08	5.2E-08	3.1E-08
U 235	9.7E-20	1.2E-18	4.1E-17	1.5E-11	1.6E-09	1.1E-08	2.0E-08	1.7E-08
U 236	4.0E-19	8.2E-18	1.8E-17	9.6E-13	2.2E-10	3.2E-10	2.4E-10	1.5E-10
U 238	6.8E-21	1.2E-19	2.5E-19	1.7E-14	3.6E-12	9.1E-12	8.3E-12	7.0E-12
Np 237	5.8E-16	3.7E-15	1.4E-14	5.7E-10	6.7E-08	1.0E-07	6.1E-08	5.7E-08
Pu 238	6.6E-17	1.6E-15	5.9E-15	1.3E-10	3.1E-08	8.6E-08	5.2E-08	3.1E-08
Pu 239	1.2E-21	2.7E-21	2.5E-17	2.3E-14	5.8E-10	8.2E-12	1.6E-11	1.7E-11
Pu 240	2.6E-19	7.2E-18	1.8E-17	9.6E-13	2.2E-10	3.2E-10	2.4E-10	1.5E-10
Pu 241	5.5E-16	3.5E-15	1.3E-14	5.4E-10	6.3E-08	9.7E-08	5.8E-08	5.4E-08
Pu 242	1.3E-22	4.3E-21	3.5E-19	4.3E-15	1.2E-09	5.3E-08	3.1E-08	2.6E-08
Pu 244	2.6E-19	7.2E-18	1.8E-17	9.6E-13	2.2E-10	3.2E-10	2.4E-10	1.5E-10
Am 241	5.5E-16	3.5E-15	1.3E-14	5.4E-10	6.3E-08	9.7E-08	5.8E-08	5.4E-08
Am 242m	5.5E-17	1.3E-15	4.9E-15	1.1E-10	2.6E-08	8.0E-08	4.9E-08	3.0E-08
Am 242	5.5E-17	1.3E-15	4.9E-15	1.1E-10	2.6E-08	8.0E-08	4.9E-08	3.0E-08
Am 243	7.8E-22	2.3E-21	2.4E-17	2.3E-14	5.8E-10	8.2E-12	1.6E-11	1.7E-11
Cm 242	6.6E-17	1.6E-15	5.9E-15	1.3E-10	3.1E-08	8.6E-08	5.2E-08	3.1E-08
Cm 243	1.2E-21	2.7E-21	2.5E-17	2.3E-14	5.8E-10	8.2E-12	1.6E-11	1.7E-11
Cm 244	2.6E-19	7.2E-18	1.8E-17	9.6E-13	2.2E-10	3.2E-10	2.4E-10	1.5E-10
Cm 245	5.5E-16	3.5E-15	1.3E-14	5.4E-10	6.3E-08	9.7E-08	5.8E-08	5.4E-08
Cm 246	1.3E-22	4.3E-21	3.5E-19	4.3E-15	1.2E-09	5.3E-08	3.1E-08	2.6E-08
Cm 247	7.8E-22	2.3E-21	2.4E-17	2.3E-14	5.8E-10	8.2E-12	1.6E-11	1.7E-11
Cm 248	2.6E-19	7.2E-18	1.8E-17	9.6E-13	2.2E-10	3.2E-10	2.4E-10	1.5E-10
Tc-99	1.5E-13	2.6E-13	1.2E-13	6.9E-11	1.0E-08	2.6E-09	3.0E-10	1.4E-10
I-129	5.7E-14	1.2E-13	1.0E-13	5.1E-11	1.8E-08	1.1E-08	3.2E-09	2.7E-09

D.3. Long-term radiotoxicity and dose conversion factors

The LTR is related to the LTD, except that LTR is strictly a measure of hazard of the material in question and LTD incorporates the amount of material that transports to potential recipients. LTR is therefore simpler and independent of location, geochemistry, etc. However, LTR is not a measure of risk and has no regulatory value in the U.S. repository program. Its primary value is simplicity and comparison against benchmarks such as natural uranium ore.

Calculation of LTR metrics is conceptually straightforward, simply the inventory of all isotopes that arise from an initial isotope, times their respective dose conversion factors. Table D-5 lists the dose conversion factors we used; they come from the International Commission for Radiological Protection (ICRP). (It also lists accident release fractions, discussed below.) The ICRP database provides dose factors in units of Sv/Bq, these were converted to Sv/kg. The inhalation and ingestion dose factors are for the tissue-weighted "Effective Dose Equivalent" or EDE, for adult members of the public, integration time periods to 70 years age.

Table D-5. Dose and Safety Parameters

	The ICRP Database [ICRP]						Maximum Atmospheric Accidental Release Fraction to Containment per NUREG-1465 (LWR)
	Half-life		Inhalation (Sv/kg)				
	Halflife (seconds)	Halflife (years)	F	M	S	Ingestion (Sv/kg)	
Ra 226	5.05E+10	1.60E+03	N/A	N/A	N/A	1.02E+07	0
Ra 228	1.81E+08	5.75E+00	N/A	N/A	N/A	6.96E+09	0
Ac 227	6.87E+08	2.18E+01	N/A	N/A	N/A	2.94E+09	0
Ac 228	2.21E+04	6.99E-04	N/A	N/A	N/A	3.57E+10	0
Th 228	6.04E+07	1.91E+00	9.10E+11	9.70E+11	1.21E+12	2.18E+09	0
Th 229	2.32E+11	7.34E+03	1.89E+09	8.66E+08	5.59E+08	3.86E+06	0
Th 230	2.43E+12	7.70E+04	7.47E+07	3.21E+07	1.05E+07	1.57E+05	0
Th 231	9.19E+04	2.91E-03	1.53E+09	6.10E+09	6.49E+09	6.69E+09	0
Th 232	4.43E+17	1.41E+10	4.46E+02	1.83E+02	1.01E+02	9.33E-01	0
Th 234	2.08E+06	6.60E-02	2.14E+09	5.65E+09	6.60E+09	2.91E+09	0
Pa 231	1.03E+12	3.28E+04	not defined	2.45E+08	5.94E+07	1.24E+06	0
Pa 233	2.33E+06	7.39E-02	not defined	2.53E+09	3.00E+09	6.68E+08	0
U 232	2.27E+09	7.20E+01	3.17E+09	6.18E+09	2.93E+10	2.61E+08	0
U 233	5.00E+12	1.59E+05	2.08E+05	1.29E+06	3.44E+06	1.83E+04	0
U 234	7.72E+12	2.45E+05	1.29E+05	8.09E+05	2.17E+06	1.13E+04	0
U 235	2.22E+16	7.04E+08	4.16E+01	2.48E+02	6.80E+02	3.76E+00	0
U 236	7.39E+14	2.34E+07	1.27E+03	7.66E+03	2.08E+04	1.13E+02	0
U 238	1.41E+17	4.47E+09	6.22E+00	3.61E+01	9.95E+01	5.60E-01	0
Np 237	6.75E+13	2.14E+06	1.30E+06	6.00E+05	3.13E+05	2.87E+03	0.006
Pu 238	2.77E+09	8.77E+01	6.97E+10	2.91E+10	1.01E+10	1.46E+08	0.006
Pu 239	7.59E+11	2.41E+04	2.76E+08	1.15E+08	3.68E+07	5.75E+05	0.006
Pu 240	2.06E+11	6.54E+03	1.01E+09	4.22E+08	1.35E+08	2.11E+06	0.006
Pu 241	4.54E+08	1.44E+01	8.77E+09	3.43E+09	6.48E+08	1.83E+07	0.006
Pu 242	1.19E+13	3.76E+05	1.60E+07	6.97E+06	2.18E+06	3.49E+04	0.006
Pu 244	2.61E+15	8.26E+07	7.22E+04	3.08E+04	9.84E+03	1.58E+02	0.006
Am 241	1.36E+10	4.32E+02	1.22E+10	5.33E+09	2.03E+09	2.54E+07	0.005
Am 242m	4.80E+09	1.52E+02	3.31E+10	1.33E+10	3.96E+09	6.83E+07	0.005
Am 242	5.77E+04	1.83E-03	3.29E+11	5.08E+11	5.98E+11	8.97E+09	0.005
Am 243	2.33E+11	7.38E+03	7.08E+08	3.02E+08	1.11E+08	1.48E+06	0.005
Cm 242	1.41E+07	4.46E-01	4.05E+11	6.38E+11	7.23E+11	1.47E+09	0.005
Cm 243	8.99E+08	2.85E+01	1.32E+11	5.92E+10	2.86E+10	2.86E+08	0.005
Cm 244	5.72E+08	1.81E+01	1.71E+11	8.08E+10	3.89E+10	3.59E+08	0.005
Cm 245	2.68E+11	8.50E+03	6.29E+08	2.67E+08	1.02E+08	1.33E+06	0.005
Cm 246	1.49E+11	4.73E+03	1.11E+09	4.77E+08	1.82E+08	2.39E+06	0.005
Cm 247	4.92E+14	1.56E+07	3.09E+05	1.34E+05	4.81E+04	6.52E+02	0.005
Cm 248	1.07E+13	3.39E+05	5.66E+07	2.36E+07	7.55E+06	1.21E+05	0.005
C-14	1.81E+11	5.73E+03	3.30E+04	3.30E+05	9.56E+05	9.56E+04	Not defined
Sr-90	9.19E+08	2.91E+01	1.21E+08	1.89E+08	8.15E+08	1.55E+08	0.120
Zr-93	4.83E+13	1.53E+06	2.32E+03	9.30E+02	3.07E+02	1.02E+02	0.005
Tc-99	6.72E+12	2.13E+05	1.82E+02	2.51E+03	8.15E+03	4.01E+02	0.005
I-129	4.95E+14	1.57E+07	2.35E+02	9.80E+01	6.40E+01	7.18E+02	0.750
Cs-135	7.26E+13	2.30E+06	2.94E+01	1.32E+02	3.66E+02	8.52E+01	0.750
Cs-137	9.47E+08	3.00E+01	1.48E+07	3.12E+07	1.26E+08	4.18E+07	0.750

Table D-6 lists the resulting LTR values calculated for this study.

Table D-6. Long-Term Radiotoxicity (LTR) Coefficients (mrem/g)

	0	10	100	1,000	10,000	100,000	1,000,000	10,000,000
Ra 226	1.02E+09	1.77E+09	3.55E+09	2.48E+09	5.03E+07	2.39E-12	3.48E-203	
Ra 228	6.96E+11	2.51E+11	2.42E+06	1.50E-44				
Ac 227	2.94E+11	2.35E+11	1.28E+10	2.87E-03	9.65E-130			
Ac 228	3.57E+12	4.91E+09						
Th 228	2.18E+11	4.91E+09	1.27E-08	1.73E-184				
Th 229	3.86E+08	4.85E+08	4.81E+08	4.42E+08	1.89E+08	3.06E+04	4.72E-34	
Th 230	1.57E+07	1.58E+07	1.83E+07	4.26E+07	8.68E+07	3.91E+07	1.17E+04	6.36E-32
Th 231	6.69E+11	1.89E+08	3.50E+08	3.52E+08	2.86E+08	3.52E+07	2.91E-02	4.25E-93
Th 232	9.33E+01	3.35E+02	4.39E+02	4.39E+02	4.39E+02	4.39E+02	4.39E+02	4.39E+02
Th 234	2.91E+11	1.13E+06	1.14E+06	1.22E+06	3.20E+06	1.62E+07	2.57E+06	1.30E-05
Pa 231	1.24E+08	1.89E+08	3.50E+08	3.52E+08	2.86E+08	3.52E+07	2.91E-02	4.25E-93
Pa 233	6.68E+10	1.85E+06	2.04E+06	3.82E+06	1.50E+07	1.63E+07	3.29E+05	3.68E-12
U 232	2.61E+10	3.45E+10	1.45E+10	2.40E+06	3.71E-32			
U 233	1.83E+06	1.85E+06	2.04E+06	3.82E+06	1.50E+07	1.63E+07	3.29E+05	3.68E-12
U 234	1.13E+06	1.13E+06	1.14E+06	1.22E+06	3.20E+06	1.62E+07	2.57E+06	1.30E-05
U 235	3.76E+02	3.80E+02	4.07E+02	7.22E+02	3.54E+03	1.42E+04	1.57E+04	1.56E+04
U 236	1.13E+04	1.13E+04	1.13E+04	1.12E+04	1.12E+04	1.12E+04	1.09E+04	8.48E+03
U 238	5.60E+01	6.03E+01	6.03E+01	6.05E+01	6.34E+01	2.16E+02	1.61E+03	1.75E+03
Np 237	2.87E+05	2.89E+05	2.89E+05	2.90E+05	3.20E+05	8.64E+05	1.61E+06	8.42E+04
Pu 238	1.46E+10	1.35E+10	6.61E+09	6.56E+06	3.17E+06	1.62E+07	2.57E+06	1.30E-05
Pu 239	5.75E+07	5.75E+07	5.73E+07	5.59E+07	4.31E+07	3.20E+06	1.57E+04	1.56E+04
Pu 240	2.11E+08	2.11E+08	2.09E+08	1.90E+08	7.34E+07	1.54E+04	1.09E+04	8.48E+03
Pu 241	1.83E+09	2.10E+09	2.23E+09	5.28E+08	3.17E+05	8.61E+05	1.62E+06	8.42E+04
Pu 242	3.49E+06	3.49E+06	3.49E+06	3.48E+06	3.42E+06	2.90E+06	5.63E+05	1.75E+03
Pu 244	1.58E+04	1.58E+04	1.60E+04	1.75E+04	2.68E+04	3.26E+04	3.24E+04	3.08E+04
Am 241	2.54E+09	2.50E+09	2.16E+09	5.10E+08	3.17E+05	8.61E+05	1.62E+06	8.42E+04
Am 242m	6.83E+09	1.12E+10	5.51E+09	6.16E+06	3.21E+06	1.39E+07	2.23E+06	3.01E+02
Am 242	8.97E+11	1.13E+10	5.53E+09	6.08E+06	3.21E+06	1.39E+07	2.23E+06	3.03E+02
Am 243	1.48E+08	1.48E+08	1.47E+08	1.40E+08	8.77E+07	4.60E+06	1.57E+04	1.56E+04
Cm 242	1.47E+11	1.35E+10	6.66E+09	6.72E+06	3.17E+06	1.62E+07	2.57E+06	1.30E-05
Cm 243	2.86E+10	2.24E+10	2.49E+09	5.61E+07	4.33E+07	3.20E+06	1.57E+04	1.56E+04
Cm 244	3.59E+10	2.44E+10	9.30E+08	1.90E+08	7.36E+07	1.67E+04	1.09E+04	8.48E+03
Cm 245	1.33E+08	1.35E+08	1.51E+08	2.23E+08	1.21E+08	8.65E+05	1.62E+06	8.45E+04
Cm 246	2.39E+08	2.38E+08	2.35E+08	2.07E+08	5.81E+07	2.94E+06	5.70E+05	1.75E+03
Cm 247	6.52E+04	6.53E+04	6.59E+04	7.16E+04	1.16E+05	2.16E+05	2.15E+05	1.49E+05
Cm 248	1.21E+07	1.21E+07	1.21E+07	1.21E+07	1.19E+07	9.88E+06	1.59E+06	3.08E+04
Sr-90	1.55E+10	1.22E+10	1.43E+09	7.12E-01	6.52E-94			
Tc-99	4.01E+04	4.01E+04	4.01E+04	4.00E+04	3.89E+04	2.90E+04	1.55E+03	2.96E-10
I-129	7.18E+04	7.18E+04	7.18E+04	7.18E+04	7.18E+04	7.15E+04	6.87E+04	4.62E+04
Cs-135	8.52E+03	8.52E+03	8.52E+03	8.52E+03	8.49E+03	8.27E+03	6.30E+03	4.18E+02
Cs-137	4.18E+09	3.32E+09	4.15E+08	3.87E-01	1.90E-91			

Of the data in Table D-5, we only used the ingestion dose factors in this study. Ingestion would be the exposure pathway for waste in a repository. We considered also looking at inhalation, e.g., accidents during fuel cycle operations, during transportation, and during at-surface storage. Indeed, one criticism of recycling is that the reactor accident source terms would supposedly increase because of the buildup of TRU isotopes. So, we also assembled inhalation dose factors and maximum accident release fractions for reactor accidents. As described below, a preliminary analysis indicates that this is not a problem.

For the inhalation factors, two additional parameters must be specified to estimate the dose factor, the Activity Median Aerodynamic Diameter (AMAD) and the lung clearance class.

The former specifies the diameter of the particles entering the lung. "Most aerosols encountered in practical situations consist of a distribution (frequently log-normal) of particle sizes. The properties of the aerosol must therefore be characterised by some average of the properties for the individual particle sizes in the distribution. In recent ICRP reports the aerodynamic properties of an aerosol are specified in terms of the Activity Median Aerodynamic Diameter (AMAD). The AMAD is the median aerodynamic diameter of the distribution, thus 50% of the activity in the aerosol is associated with particles which have aerodynamic diameters in excess of the AMAD."[ICRP] The AMAD is typically used instead of the physical diameter because it is a better representation of how particle move in air flow. For the inhalation dose factors, the value of AMAD was set to 1 micron.

The latter pertains to how and how quickly the lung expels particles. "In the case of the 31 elements for which information on lung absorption is given in ICRP Publication 71, dose coefficients are given for the three absorption Types (F, M and S), together with a recommended default when no specific information is available on the chemical form of the radionuclide. The default Types are given in Table 2 of Publication 72. Inhalation dose coefficients for radionuclides of the additional 60 elements have been calculated on the basis that compounds assigned to lung inhalation Classes D [days], W [weeks] and Y [years] in ICRP Publication 30 (Parts 1-4) have been assigned to absorption Types F, M and S respectively, as in ICRP Publication 68. Information is given in the relevant original ICRP Publication (Table 2 of Publication 72) on the chemical forms appropriate to the different inhalation Classes/Types. This information is summarised here in a special table based on Annexe F of Publication 68."

As noted above, Table D-5 includes maximum atmospheric accident release fractions from light water reactors per the NRC. There are default values if no specific reactor and fuel values are available. We averaged the separate values for BWR and PWR, they differ slightly. Figure D-2 graphs the release fractions as a function of elemental atomic number. The TRU all have low release fraction, 0.005 and 0.006. In contrast, the volatile fission products, Xe, I, Cs have maximum release fractions well over 0.1. These, of course, dominate reactor accident consequences, not the TRU. As shown in Chapter 5 of the main report, the fission products changes little per fission energy released. Thus, it would not appear that accumulation of TRU isotopes would impact reactor accident source terms. It is, of course, an issue for the separation and fuel fabrication plants themselves, a topic for future work.

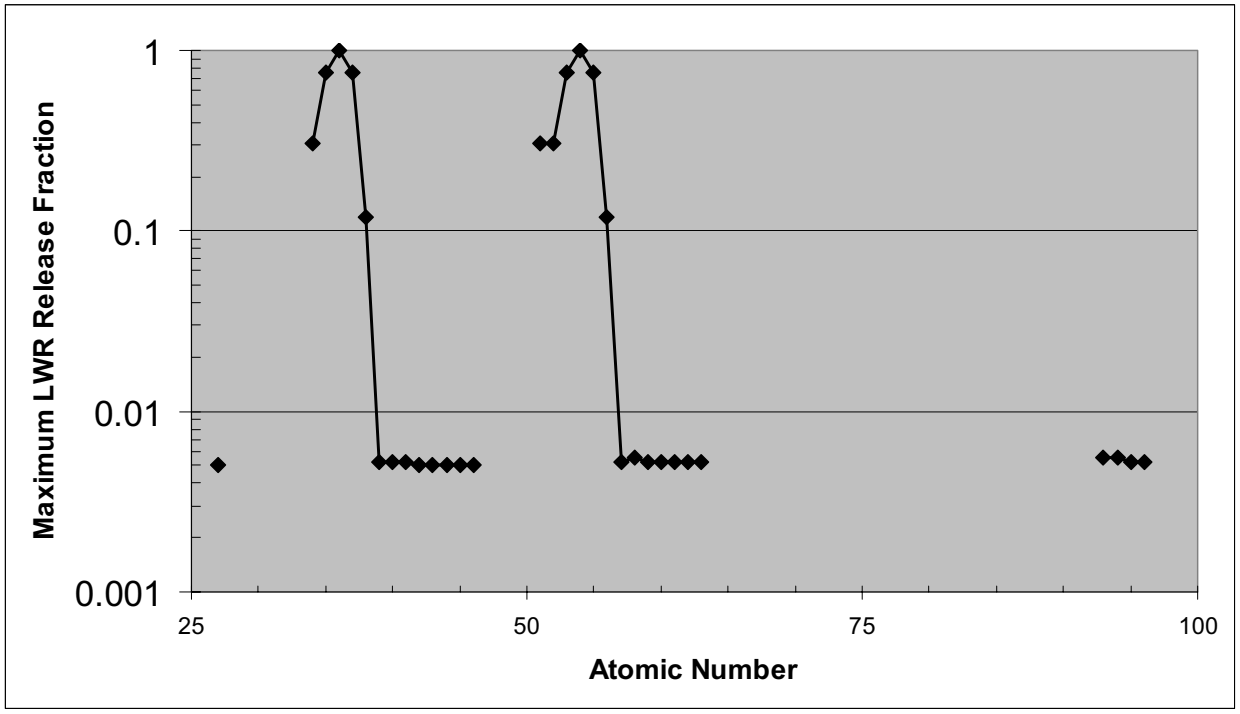

Figure D-2. Maximum LWR Release Fractions, data from [NRC1995]

D.4. Proliferation resistance parameters

The last set of isotope-specific coefficients used in this study are in Table D-7. These relate to proliferation issues, and so we used the data from the so-called TOPS report.[NERAC200, NERAC2001]. Isotopes below Th232 are not of concern.

Table D-7. Proliferation Resistance Isotope Parameters, data from [NERAC2000]

	Neutrons/ sec-kg	Bare Sphere Critical Mass (kg)
Th 232	nil	infinite
Th 234	not included	not included
Pa 231	nil	1.62E+02
Pa 233	not included	not included
U 232	not included	not included
U 233	1.23E+00	1.64E+01
U 234	not included	not included
U 235	3.64E-01	4.79E+01
U 236	not included	not included
U 238	1.10E-01	infinite
Np 237	1.39E-01	5.90E+01
Pu 238	2.67E+06	1.00E+01
Pu 239	2.18E+01	1.02E+01
Pu 240	1.03E+06	3.68E+01
Pu 241	4.93E+01	1.29E+01
Pu 242	1.73E+06	8.90E+01
Pu 244	not included	not included
Am 241	1.54E+03	5.70E+01
Am 242m	not included	not included
Am 242	not included	not included
Am 243	9.00E+02	1.55E+02
Cm 242	not included	not included
Cm 243	not included	not included
Cm 244	1.10E+10	2.80E+01
Cm 245	1.47E+05	1.30E+01
Cm 246	9.00E+09	8.40E+01
Cm 247	not included	not included
Cm 248	not included	not included

APPENDIX E. DYMOND STATUS

This appendix summarizes the status of DYMOND and some of its updated features that influence results in Chapters 7 and 8.

DYMOND now ensures mass balance of each transuranic element (Pu, Np, Am, Cm), controls the production of fuel and reactors by "elemental flow control" per Table E-1, and selects among available fuel for reprocessing according to the rules in Table E-2. Table E-3 lists available fuel options currently in DYMOND.

Table E-1. Elemental flow control approximations for thermal reactor recycling, i.e., what controls the availability of MOX or IMF fuel

	For MOX	For IMF
If TRU fuel not available	Use UOX fuel	Use UOX fuel
U	Use enriched U to meet recipe	Not needed
Pu	Require sufficient Pu to meet recipe; There is neither excess nor shortfall of Pu.	
Np	Replace shortfall with depleted U Excess accumulates	Makeup mass with IMF matrix
Am	Replace shortfall with depleted U Excess accumulates	Makeup mass with IMF matrix
Cm	None needed in current recipe, but would makeup shortfall with depleted U Excess accumulates	Makeup mass with IMF matrix

Table E-2. Priority in sending material to reprocessing or geologic disposal

	Priority in sending thermal reactor fuel to reprocessing and subsequently to thermal reactors	Priority in sending discharged fuel to geologic disposal	Priority in sending thermal reactor fuel to reprocessing and subsequently to fast reactors
Associated defaults	• SNF emplaced in geologic disposal may be retrieved. • Separation capacity is controlled by user input (histogram)	• Geologic disposal capacity is unlimited. • Rate of SNF acceptance at geologic disposal is limited (default value is 3000 MT/year); applies to the total of all unprocessed SNF. • Rate of HLW (residue after reprocessing) acceptance at geologic disposal is not limited	
Priority	1. SNF that has gone through the least number of passes (hence discharged UOX is processed before 1st pass MOX, which is processed before 2nd pass MOX, etc.) 2. SNF that is youngest 3. SNF that has been emplaced in geologic disposal	1. HLW 2. SNF that is oldest. 3. SNF that has gone through the most number of recycle passes	1. SNF that has gone through the most number of recycle passes
Alternative cases explicitly allowed in DYMOND	• Switch to disallow any emplaced SNF from being retrieved. • Switch to make separation capacity unlimited.	• Switch to disallow any SNF to be sent to repository (if sure to be following continuous recycling) • Switch to send X-pass fuel to repository where X is the terminal number of recycle passes, e.g., 1-pass IMF. (if sure to be following X-pass limited recycle)	

Table E-3. Available DYMOND cases

	Case	Neutronics data available?	Implemented in DYMOND?
Once-thru	Once-through at 33 MW-day burnup	Yes	Yes
	Once-through at 51 MW-day burnup	Yes	Yes
	Once-through at 100 MW-day burnup	Yes	Yes
IMF	1-pass IMF-NpPu	Yes	Yes
	1-pass IMF-NpPu & Am target	Yes	Yes
	1-pass IMF-NpPuAm	Yes	Yes
	1-pass IMF-NpPuAmCm	Yes	Yes
	N-pass IMF-NpPu	Not yet planned	
	N-pass IMF-NpPu & Am target	Yes	Yes
	N-pass IMF-NpPuAm	Yes	Yes
	N-pass IMF-NpPuAmCm	Not yet planned	
MOX	1-pass MOX-NpPu	Yes	Yes
	1-pass MOX-NpPuAm	Yes	Yes
	1-pass MOX-NpPuAmCm	Not yet planned	
	N-pass MOX-NpPu	Not yet planned	
	N-pass MOX-NpPuAm with constant U enrichment (but variable U/Pu ratio) (data thru N=5, extrapolation to N=8)	Yes	Yes
	N-pass MOX-NpPuAm with variable U enrichment	FY06 (from transmutation analysis)	FY06
	N-pass MOX-NpPuAmCm	Not yet planned	
Fast Reactor	UOX, then convertor fast reactor (CR =0.25) with continuing makeup from thermal reactor	Yes	Yes
	UOX, then breeder FR (CR=1.15), isolated from thermal reactors	Not yet planned	
	1-pass IMF-NpPu, then CFR	Yes	Yes
	1-pass IMF-NpPu, then BFR	Yes	Yes
	1-pass MOX-NpPu, then CFR	Yes	Yes
	1-pass MOX-NpPu, then BFR	Not yet planned	
	UOX, then CFR(CR=0.25), then BFR(CR=1.15)	Combination of other cases	Yes
VHTR	VHTR – once-thru, then TRU goes to LWR recycle	Yes	Yes
	VHTR-IMF analog	Fall 2005	Fall 2005
	VHTR-MOX analog	Fall 2005	Fall 2005

APPENDIX F. RADIOLOGICAL PROTECTION WHEN HANDLING MINOR ACTINIDES FOR NUCLEAR FUEL

F.1 Introduction

A question has been raised about the means to safely handle fuel materials for nuclear fuels incorporating minor actinides. The methods employed to protect personnel when handling radioactive materials are commensurate with the radioactive decay energy, intensity and type of radiation. The approaches may be classified as a) contact handling, b) ventilation control, c) sealed enclosures, d) semi-remote handling, and e) remote handling. The first three approaches allow human handling directly, usually wearing gloves and perhaps using simple tools (i.e., tongs, forceps). Very weak radiation sources of low specific activity, such as natural uranium, are handled by hand contact in well ventilated areas, either open areas of mill buildings, process facilities, or well-ventilated lab rooms. Small amounts of alpha particle emitters can be handled in fume hoods that provide ventilation control. As radiation levels increase, then sealed enclosures, such as gloveboxes, are used. These provide excellent alpha particle protection and some protection against low energy gamma and x-ray radiation. Shielded gloveboxes and leaded gloves provide higher protection against gamma and x-ray radiation. As the gamma and/or x-ray radiation intensity increases, these design features are insufficient radiological protection (Shuck, 1966). At that point, "semi-remote" equipment, such as in-glovebox tongs and manipulators can be used to reduce radiological exposure (NE, 1962), although these reduce dexterity, thereby increasing exposure time for completing the task. High energy radiation sources require thick shielding walls or a considerable distance from the source to the operator to reduce radiation levels to safe values. The high energy radiation sources are handled using electrically or mechanically controlled remote equipment, operated from remote stations. This increasing personnel protection comes with penalties – for example, using a factor of one to define the time to carry out an operation on an open lab bench with low-activity materials, then in-hood work is about 1.2, glovebox work is in the 1.5 to 3 range, and hot cell operations are 3 to an indefinite upper range (Stewart, 1981).

These pages will set practical limits on transitions between these five methods of personnel protection.

F.2 Contact handling

Cember (1996) states that handling radioactive materials in a laboratory room begins with a separated area of the lab room, on a benchtop. Cember states that if the material has a low radiation level and cannot release a gas, vapor or aerosol in a quantity exceeding one Annual Limit on Intake (ALI), then handling with open trays on benchtops allows adequate safety. Actual exposure to ALI inhalation values over a year result in a committed effective dose equivalent of 5 rem. The ALIs have been set by the International Commission on Radiological Protection (ICRP) and are also specified in the US Code of Federal Regulations (CFR, 2004) for a variety of radioactive materials; the ALI value for the given actinide radionuclides of interest here are generally in the 'small fraction of a microcurie' range. The actinides of concern are all classified as 'very high radiotoxicity' (IAEA, 1973). If the emitted radiation could result in a measurable external dose, then additional precautions of portable shielding and tongs or reach tools can be used to reduce occupational exposure. Cember implies that no special ventilation control beyond that for a typical laboratory space is needed at < 1 ALI for materials that do not release gas, vapor or aerosol. Considering that the US Department of Energy (DOE) has also set an Administrative Control Level (ACL) of 0.5 rem/year effective dose equivalent (DOE, 2004), then the ALI would be reduced from 1 ALI at 5 rem to 0.1 ALI at 0.5 rem. Other DOE regulations state that general

occupancies shall be designed for radiation exposure as low as reasonably achievable (ALARA) and maintained at radiation levels below 0.5 mrem/h for 2,000 hour work years (CFR, 2002), or 1 rem/year exposure.

F.3 Ventilation control

Cember (1996) stated that if the radioactive material can release gas, vapor or aerosol in the range of 1 to 10 ALI, the usual practice is to use a fume hood to provide for radiological protection. Considering the US DOE limitations, the lower limit would be less than 0.1 ALI. The IAEA (1981) more generously recommended a range for fume hood work with radium, plutonium, and americium of 10 microcuries to 1 millicurie. The IAEA did have a qualifier that if the dry box work was dusty, then the limit would be 0.1 microcurie to 0.01 millicurie (this would be on the order of ~ 10 to 1000 ALI for most actinide nuclidess). The fume hood sweeps the radioactive material release with the air drawn in to the hood, removing the inhalable material from the worker's breathing zone. The fume hood may filter its releases, but gas and vapor are not easily captured, and there is high air flow to treat. Also, fume hoods are susceptible to air flow issues, such as those created by the person standing in front of the sash, quick hand movements in and out of the hood, placement of room air ventilation ducts too close to the hood, and people walking past the hood passing too close to the front of the hood thus creating eddies that could allow the release of small amounts of airborne material out of the hood.

F.4 Sealed enclosures

If the radioactive material is over 10 ALI, or large amounts of air flow for a hood are not available or practical in the laboratory (Cember, 1996), or there is an environmental protection reason to confine/contain rather than dilute, disperse and vent the material, or an inert atmosphere is needed for material purity or safety, then the glovebox enclosure is the next engineering solution. The IAEA (1981) recommended moving to the glovebox at over 1 millicurie of Ra, Pu, Am, etc. The same qualifier from the fume hood discussion was given; if the work was dry and dusty then the transition limit would be 0.01 millicurie. This is on the order of 1,000 ALI for most actinides. The glovebox is a complete enclosure that allows workers to manipulate radioactive and hazardous materials with their hands, using flexible gloves – without high exposure to themselves or unfiltered release of material to the environment. The glovebox is capable of modest differential pressure and may use air, inert gas, or vacuum as an atmosphere. The glovebox may serve multiple functions – it can protect workers from exposure, protect the environment from material release, and it may protect the process material from air and humidity as well (DOE, 2003). Gloveboxes, gloves, and their windows work well to shield against alpha particles and low energy beta particles. Early in the exploration of nuclear energy, gloveboxes were a trademark of the nuclear industry and were considered to be good protection against alpha radiation (Ferguson, 1964). There is no longer an industry monopoly on gloveboxes; various types are now used in medical, life science, pharmaceutical, semiconductor manufacturing, and other industries.

An unshielded glovebox can be used for handling 'low exposure' plutonium (10 w/o Pu-240, 0.9 w/o Pu-241, balance Pu-239) up to the criticality mass limit of a dry process glovebox, which can be kilograms (Louwrier, 1976). Uranium and plutonium are low specific activity elements with low spontaneous fission rates. Their main radiological hazard is inhalation of these low solubility (long term lung retention) alpha particle emitters.

F.5 Semi-remote enclosures

If the radioactive material over 10 ALI emits penetrating radiation (e.g., gamma rays or energetic beta particles), then the glovebox steel walls, gloves, and windows may not provide adequate dose protection

to the glovebox worker. In the past, the US DOE design guidance was to consider use of remote handling equipment when exposures to the hands and forearms would approach yearly limits, or where contaminated puncture wounds could occur; appropriate shielding was required to minimize radiation to the skin and eyes (DOE, 1989). That guidance is no longer official, but it is still prudent from ALARA considerations.

Consider the example of an isolation glovebox used for removing actinides from irradiated plutonium samples (Schuman, 1957). The sample contained about 20 Curies of beta/gamma actinide activity and 1 Curie of alpha activity. The glovebox had 9-inch steel shielding on four sides but plexiglass walls in front and on top. Through-wall manipulators and tongs penetrated the plexiglass sides to conduct the separations work. In processing the sample (~1 day), operators each received 200 mrem. Such a dose would not be allowed any longer in the US DOE complex. The dose could have been reduced by additional precautions of front face shielding, but the implication is clear that 1E+01 Curie levels of actinides pose radiation exposure concerns in gloveboxes.

The federal annual limit for worker dose in the US DOE is 5 rem effective dose equivalent, with 15 rems for the lens of the eye, the sum of deep dose equivalent for external exposure plus committed dose equivalent to organs or tissues must be under 50 rems, and a shallow dose to the skin and extremities also has a limit of 50 rems (CFR, 2002). However, in a standard the US DOE also states that a limit of 2 rem/year is feasible and approval from a DOE Secretarial Officer is required to exceed that level. Furthermore, the US DOE has also set an Administrative Control Level (ACL) of 0.5 rem/year effective dose equivalent "As a challenging and achievable goal for personnel exposure in facilities" (DOE, 2004). Some level of ACL has been adopted at national laboratories and other DOE-operated facilities; for example, the INL uses 700 mrem/year as an ACL (INL, 2005). The INL does not have an ACL for extremities, the 50 rem/year is used for extremity dose. For a new facility, taking 500 mrem/year as the ACL and using a typical 2,000-hour work year, a maximum dose of 0.25 mrem/h is allowable. The radiation protection ALARA considerations call for whatever reductions are possible from that average dose.

If the material above 10 ALI emits gamma radiation that can penetrate the box walls, or beta radiation that can penetrate the windows and gloves, or neutrons that can penetrate any of the glovebox materials, then the glovebox can be fitted with radiation shielding to provide personnel radiological protection. The IAEA (1981) stated that shielded gloveboxes can accommodate radioactive materials in the millicurie to thousands of Curies range, but cautioned that this span was highly dependent on the energy of gamma rays emitted. Strong gammas produce too much exposure to allow hands-on work. As an initial assumption, the range was set as 10 to 1,000,000 ALI. For some actinides, 1E+06 ALI is still millicuries, while for some others it is multiple Curies. An example of a very heavily shielded glovebox with a ~1E+09 ALI (Nichols, 1963) housed up to 1 gram of Cm-244 (specific activity = 82 Ci/g). However, that appears to be a rather high value that was only accommodated by a specialized, highly shielded glovebox. Cm isotopes that emit neutrons would now be handled in hot cells.

The typical approach to shielding a glovebox is to build a general stainless steel-walled glovebox shell, then layer lead sheets onto the outer walls of the box for gamma shielding, either gluing or clamping the sheets, overlapping and perhaps melt bonding sheets together to reduce radiation streaming, then welding on another thin layer of stainless steel sheet to the glovebox as an outer cover over the lead. As an example, some gloveboxes in the TA-55 plutonium facility at Los Alamos National Laboratory used 0.25 inch-thick lead sheet on a 0.125 inch-thick stainless steel glovebox shell and then a 0.0625 inch-thick stainless steel cover over the lead. The lead sheet does not extend completely to the window frame or the glove port rings, so there are some areas of reduced shielding that allow the potential for radiation streaming in this approach. A thicker window – usually leaded glass rather than lexan - is framed in metal, the frame is gasketed and bolted to the steel shell.

The shielded glovebox design briefly described above is rather standard. However, if the radiation hazard is high, gloveboxes can be built with thicker shielding layers. For example, a glovebox for chemical analysis of radioactive liquids at INL uses ~1 inch thick shielding. For neutron shielding, a neutron thermalizer may be used (e.g., hydrogenous material such as polyethylene or water) and a neutron absorber such as boron carbide or boral (boron carbide mixed in aluminum) may be sandwiched with the lead used for gamma shielding. The glovebox windows, which may be constructed of lexan or other plexiglass for low radioactivity boxes, are usually lead oxide or lead silicate impregnated silica glass for radiation protection when penetrating radiation is confined in the glovebox. The gloves are 'leaded' by impregnating the glove material (neoprene, hypalon, etc.) with lead oxide powder to give some fraction of a mm of effective Pb shielding to the worker's hands, which are closest to the radioactive material. A typical glove is the North (see northsafety.com) lead-loaded hypalon dry box glove with 0.1 mm Pb equivalent for attenuation of soft gammas. The definition of 'soft gamma' varies, but can be assumed to be in the keV range, usually the upper bound of 'soft' is considered to be 200-300 keV, and certainly under 0.5 MeV. Glove work poses some concerns that are not easily solved: gloves reduce touch sensitivity and dexterity (so there is the chance of glove abrasion, puncture, or damage), gloves can become slippery when wet and operator perspiration in the anti-contamination hand gloves inside the glovebox gloves is very uncomfortable. The operator's sense of heat within the glovebox is greatly reduced, and the person may be under strain when operating the glovebox station (e.g., small hands in large gloves, arms in gloveports are poor ergonomic conditions) (Garden, 1962). Selection of leaded versus non-leaded gloves is not always obvious – radiological ALARA considerations must be balanced against glove durability and the extra "in-box" time required by reduced dexterity (Cournoyer, 2004). Leaded gloves tend to have shorter lifetimes than typical gloves (Dodoo-Amoo, 2003) and crack more easily than non-leaded gloves (Carmack, 2005). The designer defines what process or processes are to be carried out in the glovebox and designs the shielding to meet the yearly DOE ACL, plus a safety factor to avoid overexposures. For new facilities, the ACL would probably be 0.5 rem/year.

An important case history of actinide handling in shielded gloveboxes was given by Louwrier (1976). Handling 2.5 g increments of americium oxide and aluminum powder for cermet pellet batch preparation in a lead shielded glovebox, with leaded gloves and using tongs of 20-cm length to preclude glove contact with the material, resulted in the doses in Table F-1.

Table F-1. Doses from handling americium oxide in shielded gloveboxes (take from Louwrier, 1976)

	In-box gamma dose rates	Corresponding finger/hand doses
preparation and blending of powder	600 mrem/h	150-200 mrem
separation and weighing	800 mrem/h	400-600 mrem
loading powder in can	500 mrem/h	300-500 mrem
cold pressing pellets	200 mrem/h	30- 50 mrem
dimension control, transfer to furnace	200 mrem/h	100-150 mrem
discharging furnace, quality control checks	200 mrem/h	150-200 mrem

The finger/hand dose to prepare 132 pellets with a total of 27.5 grams of americium oxide was 19.6 rems. This is a significant fraction of the yearly 50 rem allowed for doses to the extremities, especially since only 132 pellets were fabricated. Typical fuel fabrication facilities with 4% enriched uranium oxide have had worker extremity doses less than 25% of the annual exposure limit, as measured by finger ring dosimeters (Sanders, 1975). Louwrier's exposure of 19.6 rems is 39% of the yearly limit. Louwrier (1976) also described some work with gram amounts of americium-curium; a spent isotopic power source, containing 3.3 g of Am-241 (10.7 Ci, γ dose=495 mrem/h [1E+09 ALI]), 18 mg of Cm-242 (61 Ci, 2.4 mrem/h dose is 9% γ and 91% neutron), and 380 mg of Pu-238 (6.6 Ci, γ dose=30 mrem/h),

was separated into constituent elements. The alpha radiation is not described since it was well-shielded. This Am-Cm work was judged to have high gamma radiation despite the low gamma energy, and was placed in a shielded cell with master-slave manipulators on the primary working side and leaded gloves on the opposite side. On the primary side, there was a lead sheet and a water-containing wall for radiation shielding, and a leaded glass viewing window. The work was performed solely with the manipulators rather than the gloves because of the high dose rate, and resulted in an exposure rate of < 1 mrem/h to the operators. The cell had an additional 2 mm thickness of lead shielding applied to the glove side assure future safe handling of Am-243. After that shielding augmentation, a test was performed using 5.7 micrograms of Cf-252 to simulate 1.2 grams of Cm-244. The neutron dose-rate outside the cell on the glove side (through the lead shielding) was 30 mrem/h and outside on the manipulator side (through the water and lead shielding wall) at the front of the cell was 0.4 mrem/h.

This operating experience shows that the hand doses with modest amounts of Am oxide were quite high even with use of 20-cm tongs. Also, as glovebox wall thickness increases with extra shielding, reach into the glovebox (normally ~26 inches is the design standard, based on the length of a person's arms) is reduced; hence the usability of the glovebox is reduced. Louwrier's experience shows that despite the higher allowable exposure to hands, leaded gloves cannot offer nearly the same level of hand protection as the shielded glovebox walls offer to the body. Although every effort is made to provide shielding protection, and the 50 rem/year extremities limit is not reduced to lower values by other DOE documentation, the leaded gloves at ~0.1 mm shield thickness do not offer the same level of radiation protection as the thicker lead sheeting clad on the glovebox walls. ALARA considerations for the extremities dose also affect the time operators work with the high Curie levels in shielded gloveboxes. At the Materials and Fuels Complex at INL, in-box work with materials reading over 100 mrem/h requires the operator to wear finger dosimeter rings to better record the dose to the hands. Working with such a high radiation level is rare at INL and steps are taken preserve ALARA. The staff endeavors to reduce the frequency that the evolutions occur and samples for chemical analysis are often diluted to reduce the radiation level. As an initial assignation, based on the practical operating experience described above, the transition point for moving from a shielded glovebox to a hot cell is on the order of 1 Ci, which for some of the actinides would be 1M ALI, and for higher hazard actinides would be 1E+09 ALI or greater.

F.6 Remote handling

As the materials to be handled increase in radioactivity, the glovebox cannot provide adequate radiological protection. The next step is the hot cell, also referred to as a shielded cell or "cave". Basically, a small hot cell is a robust walled glovebox with metal manipulator arms rather than gloves for handling the higher radioactive materials. The hot cell can be designed to accommodate the highest radiation level of material considered in a given process or operation. The radiation source term has been described in the past as some number of "MeV-Curies". The energy of the penetrating radiation is used with the Curie radioactivity measurement to define the shielding needed. Some early hot cells were designed for 100 Ci at 1 MeV gamma energy for analytical chemistry applications, 10,000 to 100,000 Ci at 1 MeV for post irradiation examination work, and up to 1,000,000 Ci at 1 MeV gamma energy for spent fuel handling work. The smallest hot cell facility noted during literature review was the INL Test Reactor Area cell 2, a lightly shielded cell for metallography and photography, was found to provide protection for a maximum of 10 Curies of Co-60 but some walls required temporary shielding for operator protection when operating above 2.5 Curies (Wagner, 1993). In general, the cell wall concrete thickness increases with increasing source term. The typical design practice has been to define the hot cell operator's allowable dose (such as 5 rem/year with 2,000 work hours/year), use a safety factor of 10 on that value, then determine the required concrete shield wall thickness to reduce the penetrating radiation source term to that dose level (Ferguson, 1964). The concrete walls would be several feet thick for hot cells, necessitating use of augmented reach tools. Hot cell windows are then designed to provide the same level of protection as the walls (Northrup, 1965). Other design steps were outlined by Long

(1978): the floor area and wall thickness would be established based on the process requirements, then the shielding window would be determined so that it offered similar radiological shielding as the concrete walls of the same thickness, followed by manipulator equipment selection based on process requirements, area to be covered, heaviest load to be lifted, versatility required, and if an airtight seal was needed at the manipulator wall penetration.

The hot cell viewing windows are multiple panes of lead silicate glass with mineral oil or zinc bromide to allow better optical transmission between panes. Tools are all remote manipulators, which have a reasonable reliability. As an example of manipulator reliability, a data point on metal tape driven units at the Materials and Fuels Complex (MFC) of the INL is given. The hot cell is roughly ~30 years old that has 21 left-right manipulator stations, and at any given time there may be three individual manipulators down for maintenance due to broken drive tapes or other problems. Tape slippage from a pulley and tape breakage are two of the most common problems of manual manipulator units (Smith, 1957). Considering work on a two-manipulator station basis, with three manipulators down at different stations, such faults give a 14% outage rate for the mature equipment, or 86% availability of the hot cell between maintenance outages to repair the manipulators. The design of the MFC hot cell is such that if the center of the cell is handling a 1,000 rem/h object, the inside wall is exposed to ~100 mrem/h, and the operators at manipulator stations are typically exposed to less than 0.1 mrem/h (Houser, 2005).

Likar (1988) pointed out that defense high level waste to be remotely handled at the Waste Isolation Pilot Plant would have canister surface dose rates of up to 15,000 rem/h. The hot cell purpose is to unpack canisters of radioactive material from shipping casks, inspect the canisters and overpack any damaged or leaking canisters, place canisters into facility casks for long term storage, and transfer the facility casks to an emplacement machine that places each cask in a position underground. The remote equipment has been designed to tolerate the high dose rate.

F.7. Conclusions

Table F-2 gives the suggested radioactivity ranges for moving through the approaches of successive levels of greater protection when handling actinide elements. The transition points given in the table are approximate, but for the multi-gram quantities of actinides to be mixed in to fuel, fully remote hot cell handling is warranted despite the costs of hot cells and the difficulty of handling 4.3-m fuel pins in cells.

Table F-2. Suggested progression of enclosures for handling radioactive materials

Work environment	Level of Radioactivity	Qualitative level
laboratory benchtop	< 0.1 ALI	sub-nano to nanocuries
laboratory fume hood	0.1 to 10 ALI	nano to microcuries
unshielded glovebox enclosure	10 to 1,000 ALI (a)	micro to millicuries
shielded glovebox enclosure	10 to 1,000,000 ALI	milli to unity curies
hot cell	> 1,000,000 ALI	unity to megacuries
(a) The glovebox may protect at levels greater than 1000 ALI if the emitted radiation is alpha or beta rather than gamma or neutron. If the radiation is penetrating, then a shielded glovebox may need to be used at a small ALI value.		

F.8. References for Appendix F

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APPENDIX G. SAFETY AND RELIABILITY ISSUES IN FABRICATION OF RECYCLE FUEL

Introduction. A question has been posed in the Advanced Fuel Cycle Initiative (AFCI) about using recycled uranium fuel with the addition of actinide wastes in reactors so that the excess neutrons in the reactor core will fission or transmute the actinide wastes in the recycled fuel. With the consumption of the long-lived actinides placed into recycle fuel, then there is less high level waste to store in an underground repository. This paper examines several safety and reliability issues of fabricating new fuel pellets and assemblies with the addition of the minor actinides in the uranium fuel.

Background. In the present uranium fuel cycle in the United States, the "fresh" fuel originating from uranium mines and enrichment for reactor use is composed mainly of U-238, ~5% of U-235, and ~0.04% U-234 (Rice, 1994). Therefore, with these long-lived radionuclides, the fuel is low specific activity. It is a low radiological hazard and has low dose rates. Table G-1 gives the radioactivity of unirradiated uranium dioxide fuel. The typical safety precautions in a fuel fabrication facility are mainly for laboratory-type cleanliness. Gloves are worn to protect hands from surface contamination and to keep skin oils, skin cells, etc., from the fuel. Hats or caps might be worn to prevent hairs from getting into the process, and lab coats are worn to protect against contamination of clothing and from clothing contamination of the fuel. General area ventilation is used in portions of the fabrication buildings where the uranium is in solid form. When handling uranium dioxide in powder form for individual fuel pellet manufacture, gloveboxes and respirators are used to prevent inhalation of fuel particles and control contamination by fuel dust. Uranium is a heavy metal, and most heavy metals have some level of chemical toxicity when inhaled (Klaassen, 2001; ATSDR, 2004; ATSDR, 1990; ATSDR, 1999). Heavy metals are generally not soluble in water and do not clear from the lungs easily when inhaled. While their radioactivity is low, these are alpha particle emitters. Depositing alpha particle energy in the lungs is very damaging.

Typically, meeting the 10CFR20.1201 occupational dose limit of 5 rem/year in the existing "once-through" fuel fabrication plants is not difficult if inhalation of dust is precluded. These plants use gloveboxes for dusty operations and ventilation control in other portions of the fuel fabrication line (Brodsky, 1985).

If uranium is recycled and some of the actinide elements are included in the recycled fuel, then concentrations of more highly radioactive elements will be present. Table G-2 lists the actinides from irradiated uranium fuel (Benedict, 1981). Note in Table G-2 that most of the principal decay modes are alpha particles, often in the 5 MeV range, accompanied by low energy gamma emission, with a few isotopes decaying by beta particles emission. Also note the annual limits on intake (ALIs) for airborne matter are also in the microCurie range for nearly all of these isotopes.

Table G-1a. Uranium radioactivity

Isotope	Half-life (years)	Principal decay
U234	2.46E+5	alpha at ~4.7 MeV
U235	7.04E+8	alpha at ~4.4 MeV
U238	4.47E+9	alpha at ~4.1 MeV
note: specific activity of natural uranium (0.72% by weight U-235) is ~7.1E-07 Ci/gram and for 5% enriched uranium the specific activity is ~2.6E-06 Ci/gram (from 10CFR71, Table A-4). These are low specific activity mixtures of uranium; protection must be provided to not inhale these alpha-particle emitters but otherwise they are not particularly dangerous to handle.		

Table G-1b. Uranium chemical toxicity as a carcinogenic heavy metal

Source	Value	Value
NIOSH	0.20 mg/m^3	NIOSH IDLH 10 mg/m^3 as U
OSHA	0.25 mg/m^3	
ACGIH	0.20 mg/m^3	
AIHA	Not listed	
These values are for U and insoluble compounds, the listed mass as U. The high density of uranium means that dust must be suppressed or confined to meet the ~0.2 mg level.		

The existing fuel fabrication processes would have to be enhanced with radiation protection if the "minor actinides' of americium, neptunium, and especially curium were included in recycled fuel. There is some experience in the world with mixed oxide (MOX) fuel fabrication, where the chemical and radioactive toxicity of plutonium required confinement. Glovebox lines are typically used in MOX fuel fabrication plants because of the chemical and radiotoxicity of plutonium. There has also been some work with reprocessed uranium. The experiences of this work have been surveyed to find any reliability or safety issues that should be addressed as the US considers the idea of fuel fabrication with reprocessed uranium.

There are several issues addressed here. These include the differences between typical fuel fabrication presently performed and the inclusion of actinide elements in the handling process. The main differences are radiation and chemical composition. Each of these is addressed below.

Table G-2. Actinide element radioactivity

Isotope	Half-life (years)	Principal decay mode	Decay product	Specific activity (Ci/g)	ALI (μCi)
Np237	2.14E+06	100% alpha at ~4.7 MeV	Pa-233	7E-04	5E-01
Np238	2.117 days	100% beta at 0.26 MeV	Pu-238	2.6E+05	2E+03
Pu236	2.87	~100% alpha at ~5.7 MeV SF=3.39E+04 n/s-g	U-232	527	2E-02
Pu238	87.7	~100% alpha at ~5.4 MeV SF=2.588E+03 n/s-g	U-234	17	7E-03
Pu239	2.41E+04	100% alpha at ~5.1 MeV	U-235	0.06	6E-03
Pu240	6.56E+03	~100% alpha at ~5.1 MeV SF=9.087E+02 n/s-g	U-236	0.23	6E-03
Pu241	14.4	99% beta at 20 keV	Am-241	110	3E-01
Pu242	3.75E+05	~100% alpha at ~4.8 MeV SF=1.718E+03 n/s-g	U-238	4E-03	2E-02
Am241	432.7	100% alpha at ~5.4 MeV	Np-237	3.2	6E-03
Am242	16 hours	82.7% beta at 0.18 MeV 17.3% electron capture	Cm-242 Pu-242	8.1E+05	8E+01
Am242m	152	~99.5% gamma at 0.04 MeV SF=1.247E+02 n/s-g	Am-242	9.7	6E-03
Am243	7.37E+03	100% alpha at 5.2 MeV	Np-239	0.19	6E-03
Cm242	162.8 days	~100% alpha at ~6 MeV SF=2.10E+07 n/s-g	Pu-238	3,400	3E-01
Cm-243	29.1	~99.74% alpha at 5.7 MeV 0.26% electron capture SF=1.22E+03 n/s-g	Pu-239 Am-243	52	9E-03
Cm244	18.1	~100% alpha at ~5.7 MeV SF=1.080E+07 n/s-g	Pu-240	82	1E-02
Cm245	8.5E+03	~100% alpha at ~5.3 MeV SF=3.875E+01 n/s-g	Pu-241	0.17	6E-03
Cm246	4.76E+03	~100% alpha at ~5.3 MeV SF=9.448E+06 n/s-g	Pu-242	0.31	6E-03
U234	2.46E+05	100% alpha at ~4.7 MeV	Th-230	6.2E-03	7E-01
U235	7.04E+08	100% alpha at ~4.4 MeV	Th-231	2.1E-06	8E-01
U236	2.342E+07	100% alpha at ~4.5 MeV	Th-232	6E-05	8E-01
U237	6.75 days	80% beta at 0.24 MeV 40% gamma at 64.5 keV	Np-237	8.2E+04	2E+03
U238	4.47E+09	~100% alpha at ~4.1 MeV SF=1.36E-02 n/s-g	Th-234	3.3E-07	8E-01

Table notes: specific activity of natural uranium (0.72% by weight U-235) is ~7.1E-07 Ci/gram and for 5% enriched uranium the specific activity is ~2.6E-06 Ci/gram (from 10CFR71, Table A-4). These are low specific activity mixtures of uranium; protection must be provided to not inhale these alpha-particle emitters but otherwise they are not dangerous to handle.

References: Lederer, 1978; Baum, 2002; spontaneous fission values over 1E+01 were cited, values were from report LA-UR-01-5572, September 2001

Annual Limits of Intake (ALI) are inhalation values taken from 10CFR20, Appendix B "Annual Limits on Intake (ALIs) and Derived Air Concentrations of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage", January 1, 2004. ALI values are in units of microCuries, and lung clearance class W was selected for consistency when multiple values were listed.

G.1 Issues of radiation

These alpha emitters, with some low energy gamma emission and a few fairly strong spontaneous fission isotopes (especially Cm-242 and Cm-244) present several handling problems (Louwrier, 1976) beyond typical uranium:

decay heat
radiolysis of solutions
radiation damage of solids
personnel exposure

Another issue is that the low energy gammas sometimes emitted by the alpha emitters and perhaps also the alpha particles themselves will interact with oxygen in hot cells. Even if a hot cell or glovebox uses an inert atmosphere (e.g., argon, nitrogen) there can be small amounts of inleakage oxygen since these enclosures operate at slightly negative pressure for particulate contamination control. The maximum permissible oxygen content is suggested to be in the 25 to 50 ppm range (ANS, 1988). The concern with oxygen is molecular dissociation. When oxygen molecules are dissociated, elemental or free radical oxygen atoms are created. These atoms generally recombine quickly with any nearby oxygen molecules to form ozone molecules. In a similar manner, nitrogen molecules can be dissociated, allowing nitrogen radicals to form nitrogen oxides (NO_x). Nitric acid (HNO_3) molecules can also be formed from ozone, oxygen radicals, and atmospheric humidity (Batchelor, 1982). Ozone is chemically very corrosive and is highly detrimental to plastics, such as electrical insulation, sleeves on manipulators, containers, etc., and also to rubber gloves used in gloveboxes. For that reason, hypalon (chloro-sulfonated polyethylene) gloves are favored for their resistance to ozone degradation (Louwrier, 1976). Ozone has also been known to attack metal. Hot cells or automated gloveboxes would need to be monitored for the ppm oxygen concentration. If the oxygen level increased, then there would be concerns about ozone creation and ozone degradation of materials.

Renard (1995) points out that incorporating actinides, especially curium, into a MOX fuel line may not be the best use of the line because of the contamination and retrofitting the line to accommodate the extra radioactivity and criticality concerns. Using americium and curium targets in a reactor core rather than blending these actinides into the fuel have some advantages.

Decay heat. The decay heat given off by actinides is a concern for shielded enclosures since these inert-atmosphere enclosures are not well ventilated and require special provisions to remove heat. A typical hot cell temperature should not exceed 30 C (86 F) and it is already heated by the intense lighting in the cell (the windows reduce the light intensity by more than half, so the lights must produce high luminosity) and other heat sources, such as motors, friction in machining parts, welders, and other equipment (Wahlquist, 1998). Inert gases may not cool as well as air when it is draw through an electrical motor casing. Electrical equipment is generally favored in hot cells due to the non-contaminating nature of electrical power (versus hydraulic or pneumatic power), but keeping electrical equipment cool is an important issue to guarantee reliable operation and long motor lifetimes. Wahlquist (1998) notes some types of motors may not be suitable for the very low humidity gas environment of a hot cell. High temperatures in the glovebox or cell tend to degrade electrical insulation in motors and wiring, reducing the useful life of the equipment, and high temperatures also place demands on lubricants for manipulators and on television cameras that might be in use. High temperatures can degrade gloves and window seals in gloveboxes. Some hot cells have cooling systems, refrigeration systems that allow the in-cell gas or metal parts of a machine to transfer heat to the refrigerant. These are rated to remove some level of kilowatts. If actinides increase the in-cell heating, changes would be needed to bring the temperature back down within normal limits.

Recalling that U-235 is ~1E-06 watts/gram, the specific heat values for several of the actinides of interest are listed below in watts/gram (Ferguson, 1963):

Pu-238	0.555
Pu-240	0.007
Pu-241	0.004
Pu-242	0.0001
Am-241	0.106
Am-242	1000.
Am-242m	0.003
Am-243	0.006
Am-244	88,000.
Cm-242	122.
Cm-243	1.47
Cm-244	2.91
Cm-245	0.006
Cm-246	0.0072

Obviously, the Cm-242 and Am-242 isotopes have high heat emission that must be dealt with in the design of the facility (Am-244 is very high; however, it is quite rare). Pillon (2003) stated that work with minor actinides would require continuous forced cooling throughout the fabrication facility. Forced cooling would be necessary for the fuel assemblies during pellet loading and pin mounting to keep the long, thin-walled pin structural material temperatures low enough to avoid damage during handling and storage. One possibility to consider was discussed in Weissert (1968); completed fuel pins of Th-232/U-233 were placed into fuel assemblies under water. The fuel element assembling machine operated in a canal for the pilot plant at Oak Ridge National Laboratory.

Radiolysis of solutions. Two materials are added to the powder before compaction. The first material is called "binder". It is usually a liquid adhesive, added to 'cement' the powder so that the fuel powder handles better for pressing into pellet form in a press. The second material is a lubricant, which is added to allow more positive and uniform compaction of the power in the press. Proper compaction results in a uniform density pellet, which is needed for consistent nuclear and thermal properties in the reactor core. There are some materials that can serve as both a binder and lubricant; since the liquid is expelled from the pellet during furnace heating at ~1700 C in the pellet sintering process, the liquid is not an impurity concern. Mobil #6 motor oil has been used as a combination binder and lubricant in the US (Carmack, 2005). Cochran (1999) stated that a binder sometimes used is polyvinyl alcohol. Alpha particles are short-range radiation, but they are directly adjacent to the binder and lubricant and are energetic enough to affect these materials. When hydrocarbons like alcohol and oil are bombarded with alpha particles, the molecular chains can be broken and they can release smaller hydrocarbons, namely hydrogen and methane. Such gases pose a concern in air filled gloveboxes or hot cells.

Pillon (2003) discussed that for an industrial scale of pellet production, pellet pressing required lubricants, but the minor actinide radioactivity causes the lubricant to break down and lose lubrication properties. The green pellets (the pressed pellets before heating in the sintering furnace) also can become mechanically unstable due to binder-lubricant degradation. The proposed solution was to use a separate binder and develop press dies that are automatically lubricated before each pellet rather than adding lubricant to the pellet material. This adds complexity to the machine, but the alternative is to allow fabrication of high numbers of substandard units that must be scrapped. This would mean that some material continues to traverse the fabrication line. Allowing material to "ride the circuit" means extra inventory is present and susceptible to accident events.

Radiation damage. Typically, alpha particles do not create damage in metals, and are too short range to be a concern for windows. Alpha particles may break down the lubricants in manipulators, especially the lube in the end effector joints close to the emitter. There are some concerns for gamma rays and beta radiation emitted by the actinides, although low energy gammas generally do not affect the metal components in hot cells (Wahlquist, 1998). Gamma radiation can have severe damaging effects on lubricants. Another example of a susceptible component in a hot cell is a television camera lens. Feraday (1981) stated that estimated dose rates to cameras in a remote fabrication hot cell would be < 2 rad/hour and this was acceptable for work with thorium-uranium fuels, where U-232 has some hard gamma ray emission in its decay chain. If a new facility to handle actinides in fuel is designed, then the radiation to cameras would be accounted for in the design. If an existing facility; that is, a MOX facility, began handling minor actinides, the additional radiation levels must be evaluated for the additional exposures and resulting damage from exposure. The spontaneous fission neutrons are another concern. Overall, robotic equipment in factories has exhibited high reliability in the 97% availability range (Nof, 1985), but the service lifetime of industrial robots is only 12 to 15 years (UN, 2002). The source documentation did not specify if the 12-15 year lifetime was based on obsolescence or wear out. The Materials and Fuels Complex at the INL has a large hot cell with 21 manipulator stations. The hot cell has operated for over 30 years (Houser, 2005). At any given time, there may be up to three manipulators out of service waiting for maintenance, or 18 operable stations out of a total of 21 stations gives 86% availability. Since some hot cell equipment has lifetimes of 30+ years, then perhaps factory retooling for new products and obsolescence are the dominant factors in robot lifetimes.

Remote manipulator "boots", that is, the sleeve-like covers over the in-cell portion of manipulator tools, are a source of continual concern for cell confinement integrity and contamination control. The boots are actually part of the seal for the manipulator penetration through the thick concrete wall, so they comprise a weak spot in the hot cell confinement boundary. Robinson (1969) stated that small particles of $^{244}Cm_2O_3$ at a curium hot cell facility were capable of burning pinholes through polyurethane boot material. Others noted that boot lifetimes in actinide processing hot cells were on the order of 10 months (Samsel, 1970), meaning annual replacement was standard procedure. Either more resilient materials are needed, or a changeout device is needed to make the boot replacement as quick as possible. If curium particles can damage the manipulator boots, then filters are also at risk. Some method of particle capture is needed ahead of the hot cell's gas filter banks.

Personnel exposure. Roepenack (1987) stated that the low energy gamma radiation of Pu and Am handled in MOX fabrication was negligible as long as the materials were held in bulky configurations. In bulk configurations there is high self-absorption and self-shielding. When the powders are spread out over widespread areas there is little self-shielding and the low energy gammas can contribute considerably to the glovebox operator's dose rate. The situation would be worse with actinides adding more gammas and also spontaneous fissions to challenge the shielding. Bemden (1981), Leblanc (1982), and Carmack (2005) also stated that dust is known to accumulate in the "powder" section of fuel fabrication lines. The powder section is where the uranium dioxide in a loose powder form is mixed and blended, then compacted into pellet form; this portion of fuel fabrication is always performed in some type of enclosure to control the spread of dust and protect personnel from dust inhalation and radiation exposure from dusts. When there are low energy gamma emitters accumulating with this fuel dust, the increasing radiation level versus a fixed shield wall thickness begins to pose an exposure concern for personnel working in and near the gloveboxes that house the powder section of the fuel fabrication line (Draulans, 1985). In fact, the powder tends to spread everywhere within the enclosure and in general is tedious to clean up (Carmack, 2005). Dust can also accumulate in the filters and ventilation systems of gloveboxes and hot cells. Some of the actinide dusts, especially Cm-244, would be difficult to detect through the shield walls because they have weak gamma emissions, but they pose a significant concern because they emit neutrons from spontaneous fission (Haggard, 1996). Typically, with low specific

activity uranium, the dust accumulation concern dwells on criticality rather than radiation exposure. The actinides have relatively high critical mass limits, as shown in Table G-3, the lowest being 13 grams of Am-242m in water. The fabrication processes should be dry, so criticality mass limits are in the kilogram range which should allow reasonable size batches of fuel. The criticality safety "double contingency" precautions of mass limits and geometry/moderator limits have served well in the fuel fabrication industry and would be needed with mixtures of non-fissile and fissile isotopes. Mixtures of oxide materials will need to be addressed for criticality safety issues, especially the spontaneous fission isotopes. The fuel dust would have to be regularly and thoroughly cleaned if gloveboxes are used to house an automated process line for fuel with actinides. Otherwise, any personnel intervention for surveillance or maintenance would expose the person to a high dose. Design provisions must be made to remotely decontaminate such gloveboxes.

Table G-3. Subcritical mass limits for non-fissile and fissile actinide nuclides

		Mass limit (kg) of nuclide	
Isotope	Chemical form	Water reflector	Steel reflector
Np237	Np	30	20
	NpO_2	140	90
Pu238	Pu	4	3
	PuO_2	11	7
Pu239 (fissile)	Pu	0.450	
Pu240	Pu	20	15
	PuO_2	70	45
Pu241 (fissile)	Pu	0.200	
Pu242	Pu	60	40
Am241	Am	24	16
	AmO_2	40	32
Am242m (fissile)	Am	0.013	
Am243	Am	35	25
	Am_2O_3	50	37
	AmO_2	60	45
Cm243 (fissile)	Cm	0.090	
Cm244	Cm	5	3
	Cm_2O_3	7	5
	CmO_2	7	5
Cm245 (fissile)	Cm	0.030	
Cm247 (fissile)	Cm	0.900	
Table taken from ANSI, 2005.			

Pillon (2003) discussed that for curium, powder metallurgy processes that generate dust must be avoided. Direct synthesis of dust-free spherical particles was proposed. Vibrocompaction to form pellets for sintering was suggested. Robinson (1969) discussed that low gas flow was important for reducing curium contamination, but with low gas flow, alternate cooling had to be installed in a curium hot cell. Pillon (2003) discussed the solution-gelation (sol-gel) approach to fuel fabrication. However, alpha particle radiation, especially from curium isotopes, degrades the solution in the sol-gel process. Pillon stated that the process must be performed rapidly to be successful. Based on that information, any unplanned process delays (power outages, equipment tripping off-line, instrument noise, etc.) mean retaining the material in the processing stream longer.

Renard (1997) described that 0.165 kg of americium oxide in a container gave an equivalent dose (0.0344 mSv/h) to 13.2 kg of plutonium oxide in a container. In a MOX facility, the storage cans and storage cavity rooms would need additional shielding if the americium supply increased. As shown in Table 2, Pu-241 decays to Am-241 with a 14-year half-life. Therefore, as more Pu-241 is bred, then more Am-241"grows in," increasing the americium quantity. Such increases need to be tracked and addressed, so actinide handling is more complicated than fresh uranium handling. Renard (1995) stated that neptunium recycling could occur without difficulty in existing MOX glovebox lines, that americium would require more shielding for safety, which was costly and cumbersome but feasible, and that curium recycling would give a factor of 100 increase in the neutron source. This increased neutron source would require such thick shielding that it precluded recycle fuel work in existing MOX plants; remote handling was suggested. Renard (1995a) suggested that the limitations in recycle were not in fuel fabrication since hot cells could provide adequate personnel protection, but in the core physics-neutronics issues of placing the actinide blended fuel in the reactor.

G.2. Issues of composition

Yoshimochi (2004) described fabrication of a MOX fuel with 3% americium oxide included. For that work they concluded that they required hot cell operation because handling the gamma-emitting americium in a glovebox was difficult. The fuel pellets were fabricated beginning with typical powder metallurgy - mixing, granulating, and pressing. The apparatus used in this fabrication was housed in stainless steel box enclosures (i.e., caissons) within a hot cell, presumably to confine dust contamination as well as reduce operator dose. The apparatus typically produced uranium oxide fuel and was remotely controlled by an operator at a control panel. The powder was weighed and an organic binder was blended into the mixture inside a ball mill that uses 10-mm diameter tungsten balls. The resulting pellet density of that initial batch was only 88% of theoretical density after sintering. The cause of the low density was traced to the fact that UO_2 and Am-PuO_2 powders have different morphologies and did not uniformly homogenize in the ball mill after the typical 4 hour milling time, despite attempts of enhance blending. Experimentation showed that the mixture required at least 10 hours of milling time to obtain a sintered pellet of 94.5% theoretical density. This is not a safety problem, but it does lengthen the time that powder form is being handled in the process, and may lead to more dust production. There may be other means besides ball milling available to mix the powders. Another issue occurred during the fabrication work. A large uranium spot was observed in one of the pellets; poor mixing was ruled out since the ball mill time had been more than doubled. The uranium spot was believed to have come from residue from a previous run of the powder feeder for UO_2 pellet production. Therefore, complete cleaning of the equipment prior to fabricating batches of actinide bearing fuel would be necessary to prevent residues from previous fuel batches from entering in to the new process.

Croixmarie (2003) also experienced the mixing issue when fabricating americium in magnesia targets. When the densities of the magnesia powder granules and the spheres composed of Am, Ce, Pu, Y, and Zr oxides were very different, mixing led to agglomeration and segregation. Increasing the magnesia granule size to 50-71 microns produced good mixing and produced pellets with greater than 95% theoretical density, no cracks, and the preferred random distribution of isolated spheres.

Krellmann (1993) described that MOX fuel pellets needed to be dried very thoroughly to remove any residual moisture prior to placing the pellets in a fuel pin. Moisture would bond well with sub-stoichiometric plutonium oxide, leaving hydrogen trapped in the fuel pin. Then, under reactor core heat and radiolysis conditions the hydrogen would react to form zirconium hydride on the fuel pin walls, leading to localized damage of the pin. Drying over 350 C in vacuum or in a dry inert gas was recommended.

G.3. Safety and reliability

Fullwood (1980, 1984) prepared a preliminary risk assessment for a fuel fabrication plant, both a reference MOX plant and a recycle plant with Am, Np, and Cm actinides incorporated into the MOX fuel. The MOX plant was postulated to have 1,078 Curies per kilogram of fuel powder (mainly from Pu-241 and Pu-238) and the recycle plant 1,408 Curies/kilogram (increases mainly from Cm-244, Cm-242, Am-241, and Np-239). The plant capacity was 600 metric tons of heavy metal per year. Preliminary hazards were identified as: powder leaks, filter failures, fire in the powder receiving area, powder overheating during processing, criticality, pellet press hydraulic fluid fire, solvent fire/explosion, pellet grinder failure, and improper fuel pin welding resulting in aerosol release. Scrap recovery in the plant also had several hazards, including criticality concerns, hydrogen explosions, solvent fires, red oil explosions, resin fire/explosion, and filter failures. External events included aircraft crash into the powder area, earthquake, and a tornado that reversed air flow in the facility. The fuel fabrication plant used thick concrete walls and remote handling. Fullwood's conclusion was that the hazard was well handled by the shielding and remote handling design, so the reference and recycle plant risks were identical at 6E-04 fatalities per GWe capacity-year from radiological releases, and industrial risks to workers were 1.1E-02 fatalities per GWe capacity-year. Occupational radiation doses were about equal between operators and maintainers, with most maintenance being performed remotely.

G.4. Conclusion

The fuel fabrication industry has enjoyed relatively easy fuel assembly since uranium is a low specific activity material. Moving to mixed oxide fuel increased the radiation hazard because of plutonium isotopes and their emissions, so MOX fuel fabrication required more shielding and enclosures of the process beyond the uranium processes. Moving to actinide inclusion in fuel poses another increasing level of difficulty. The operating experience thus far shows that handling the americium and curium has been performed in hot cells to provide for personnel protection and good confinement against environmental release. Renard (1995) stated that curium could probably not be handled in gloveboxes unless the boxes were specially shielded for that neutron emitter. Americium and curium appear to be the most challenging isotopes to handle during recycling. One recycle introduces a limited amount of radionuclides bred in the core, such as the plutonium isotopes. Additional recycle sessions would increase the concentrations of these radionuclides and their decay products. If a hot cell is used, it appears that multi-recycling fuel fabrication work can be performed safely. If an automated glovebox line is used, extensive glovebox shielding and remote reach tools would have to be provided to accommodate multiple recycling stages.

G.5. References for Appendix G

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