ANL-AFCI-150

Summary of Generation-IV Transmutation Impacts

Nuclear Engineering Division
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by
T. A. Taiwo and R. N. Hill
Nuclear Engineering Division, Argonne National Laboratory

June 30, 2005

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9700 South Cass Avenue
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ABSTRACT

An assessment of the potential role of Generation IV nuclear systems in an advanced fuel cycle has been performed. The Generation IV systems considered are the thermal-spectrum VHTR and SCWR, and the fast-spectrum GFR, LFR, and SFR. This report addresses the impact of each system on advanced fuel cycle goals, particularly related to waste management and resource utilization. The transmutation impact of each system was also assessed, along with variant designs for transuranics (TRU) burning.

The base fuel cycle for the thermal reactor concepts (VHTR and SCWR) is a once-through fuel cycle using low-enriched uranium fuels. The higher burnup and thermal efficiency of the VHTR gives an advantage in terms of heavy-metal waste mass and volume, with lower decay heat and radiotoxicity of the spent fuel per electrical energy produced, compared to a PWR. Fuel utilization might, however, be worse compared to the PWR, because of the higher fuel enrichment essential to meeting the VHTR system design requirements. The SCWR concept also featured improved thermal efficiency; however, benefits are reduced by the lower fuel discharge burnup.

The base fuel cycle for the fast reactor concepts (SFR, GFR, and LFR) is a closed fuel cycle using recycled TRU and depleted uranium fuels. Waste management gains from complete recycle are substantial, with the final disposition heat load determined by processing losses. The base Generation-IV concepts allow consumption of U-238 significantly extending uranium resources (up to 100 times).

For both thermal and fast concepts, recent design studies have pursued the development of dedicated burner designs. Preliminary results suggest that a burnup of 50-60% is possible in a VHTR burner design using non-uranium (transuranics) fuel. However, practical limits related to higher actinide buildup and safety impact may limit the extent of TRU burning in thermal reactors. Fast burner designs have been developed for both conventional and high TRU content fuel forms. In general, the conversion ratio can be varied within a system by changing the uranium loading. Recent studies indicate a low conversion ratio (0.25) SFR retains the favorable passive characteristics of conventional designs, and the cost is similar.

1.0 INTRODUCTION

The viability of nuclear energy, in the long term, requires that the issues of waste management, non-proliferation, and uranium supply be adequately addressed. It has been recognized in the U.S. that new technologies have to be implemented both domestically and internationally to provide the necessary solutions. These efforts are being planned or undertaken under the Advanced Fuel Cycle (AFCI) Program of the USDOE. The four AFCI program objectives from the 2005 Report to Congress [1] are:

- Reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.
- Enhance overall fuel cycle proliferation resistance via improved technologies for spent fuel management.
- Enhance energy security by extracting energy recoverable in spent fuel and natural uranium, ensuring that uranium resources do not become limiting for nuclear power.
- Improve fuel cycle management, while continuing competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system.

The AFCI Program has proposed a logical succession of technologies to evolve the current nuclear fuel cycle enabling the U.S. to actively lead the development of advanced fuel cycle technologies and to influence worldwide technology choices that will ensure safe, proliferation-resistant nuclear power. [1] The elements of this strategy include the completion of the once-through fuel cycle by opening the federal geologic repository, the application of extended burnup fuels, the development and demonstration of proliferation-resistant fuel cycle technology (possibly incorporating limited recycle in existing reactors), and the development of advanced fast reactors and transition to continuous and sustained spent fuel recycle.

The USDOE also has a program to develop advanced, next-generation nuclear reactors; this program has analogous goals related to safety, sustainability, economics, and proliferation resistance. Six of these advanced designs, called the Generation-IV (Gen-IV) nuclear energy systems, have been proposed internationally. The systems include thermal-spectrum designs such as the Very High Temperature Reactor (VHTR), the Supercritical Water-Cooled Reactor (SCWR), and the Molten Salt Reactor (MSR). The Very High Temperature Reactor is the

leading candidate in the U.S. Gen-IV Program to develop an inherently safe, economic nuclear system that could produce both electricity and hydrogen with relatively less waste. [2] Three fast-spectrum systems are also being considered: Gas-cooled Fast Reactor (GFR), Lead-cooled Fast Reactor (LFR) and Sodium-cooled Fast Reactor (SFR).

The purpose of this report is to consider the potential role of these Gen-IV concepts in an advanced fuel cycle. The focus will be on the AFCI waste management and resource utilization goals which are directly impacted by the transmutation performance of the deployed reactors. In Section 2, the basic transmutation physics of the proposed Gen-IV systems is contrasted; key differences between thermal and fast spectrum systems are illustrated. In the following sections, the fuel cycle performance of each base Gen-IV system is evaluated, and associated transmuter options are explored. The application of the base VHTR in the current U.S. once-through fuel cycle is explored in Section 3.1 by comparing its waste generation to that of typical PWR systems. Concepts that have been proposed for configuring the VHTR-type systems into a transuranics burner are discussed in Section 3.2. In Section 4, the base Gen-IV SCWR and an alternative fast spectrum SCWR are described. The Gen-IV fast-spectrum systems (SFR, GFR, and LFR) are evaluated in Sections 5 to 7 respectively; for each concept, burner design options are also considered. The MSR concept was not considered in this study, although Ref. 3 contains some relevant fuel cycle information. An intercomparison of the fuel cycle roles and burner performance of the Gen-IV systems is presented in Section 8.

2.0 COMPARISON OF TRANSMUTATION PHYSICS

Similar to conventional LWRs, the Gen-IV VHTR and SCWR designs employ moderators (carbon and supercritical water respectively) to slow down the fission neutrons. Conversely, in the fast spectrum systems moderating materials are avoided. The resulting neutron energy spectra of the five Gen-IV systems are compared to a PWR in Fig. 1. The nuclear interactions in the PWR/VHTR/SCWR are dominated by the thermal peak around 0.1 eV. The fast spectrum systems have no low energy neutrons, and most neutron reactions occur around the flux peak at 100 keV. Because the transmutation physics of the dominant actinides differs greatly between these two energy ranges (as shown below), thermal/fast energy spectrum is a key distinction with regard to transmutation behavior.

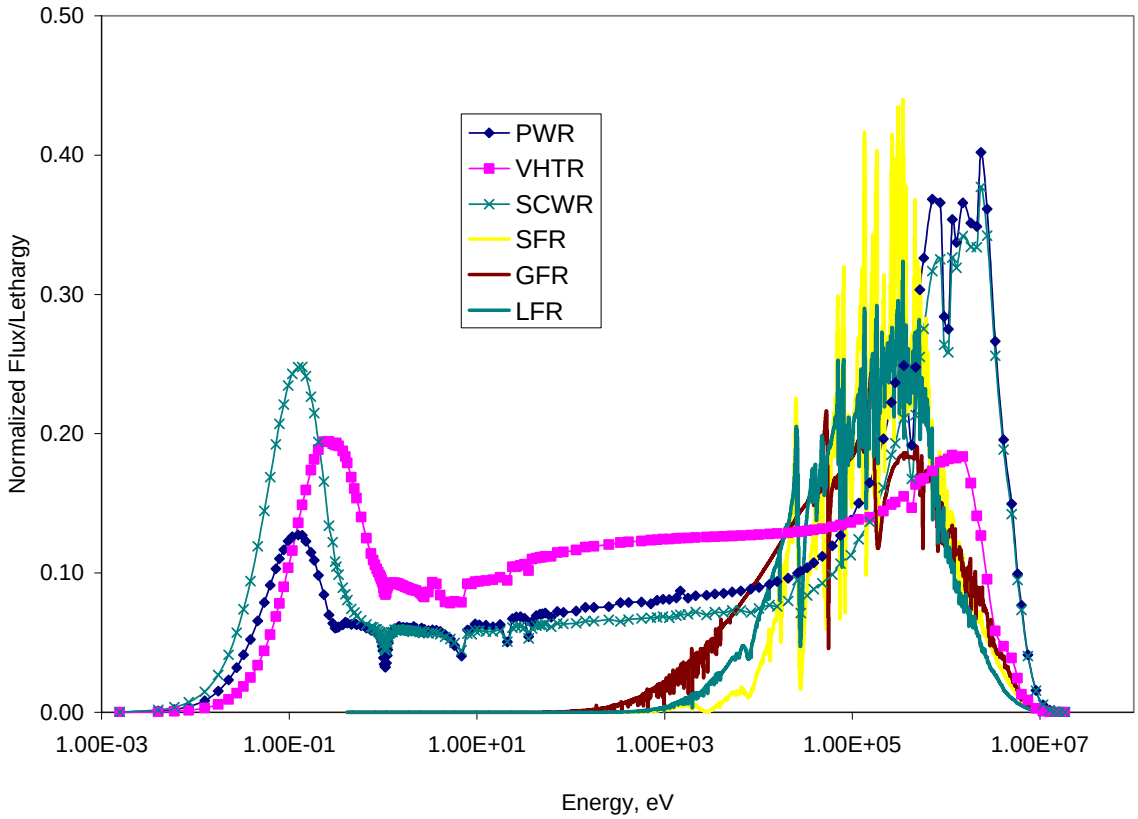

Figure 1. Comparison of Neutron Energy Spectra of Gen-IV Reactors

The variation of transmutation behavior with energy spectrum is illustrated in Fig. 2; the fission/absorption ratio is compared for dominant actinides in the PWR and SFR spectra. The fission/absorption ratios are consistently higher for the fast spectrum SFR. For fissile isotopes

(U-235, Pu-239, and Pu-241) over 80% of fast neutron absorptions result in fission, as compared to 60-80% in the PWR spectrum. In addition, the fast spectrum fission fraction can rise to 50% for fertile isotopes as observed for Pu-240 in Fig. 2, while remaining low (<5%) in a thermal spectrum. Thus, in a fast spectrum actinides are preferentially fissioned, not transmuted into higher actinides. This implies that fast systems are more "efficient" in destroying actinides because fewer neutrons are lost to capture reactions before eventual fission. Furthermore, the generation rate of higher actinides is significantly reduced at each transmutation step. For example, the Pu-241 content is seven times lower in a sustained fast reactor, compared to a typical LWR; and the higher actinides (americium and curium) continue to build-up with LWR recycle. These higher actinides tend to be more radioactive and can be problematic for fuel handling and fabrication in a closed fuel cycle.

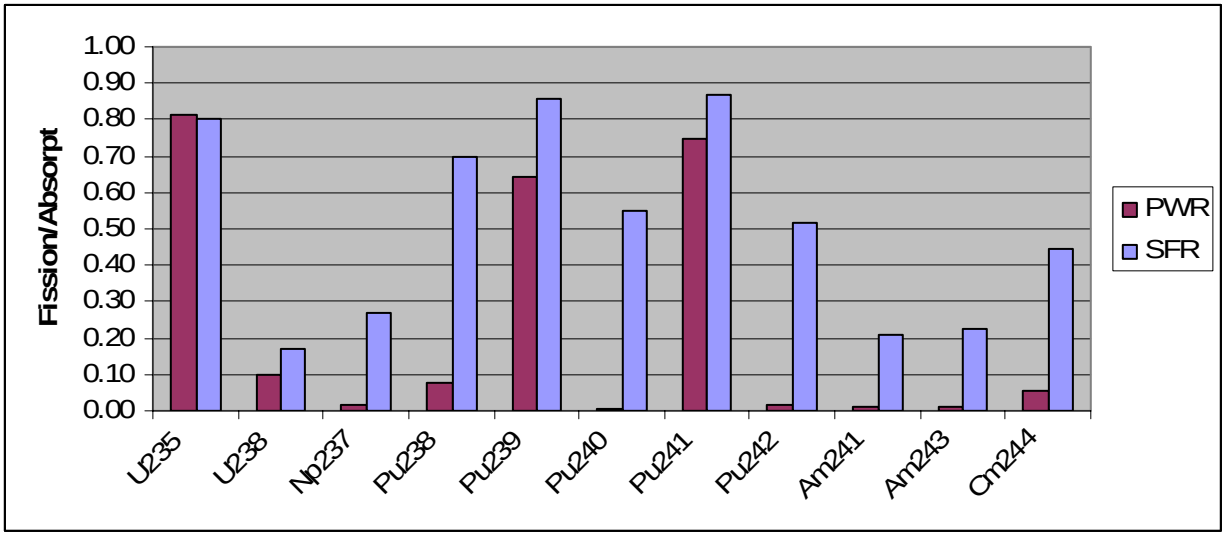

Figure 2. Comparison of Fission/Absorption Ratio for PWR and SFR

For resource utilization and sustainability goals, it is useful to compare the overall neutron "balance" of the transmutation process; a physics approach was developed in Ref. 4. The basic approach is to track the neutron balance for transmutation from the source isotope to extinction; fission and (n,2n) reactions produce neutrons while capture reactions consume neutrons. This approach was extended to compute equilibrium compositions for fixed feed compositions in Ref. 5. In that study, a variety of fast and thermal systems fuel cycles were compared. A previous analysis of the three Gen-IV fast reactor concepts was performed in the

ANL-AFCI-116 report [6] where the details of one-group cross section data generation and computational procedures are described.

In this report, the neutron balance (D-factors) are compared for the five Gen-IV systems; the results allow comparison of the feasibility of transmutation of the different isotopes in each reactor concept. The D-factors are given in Table 1 for the dominant actinides; note that a positive D-factor indicates net neutron consumption, while a negative D-factor indicates a neutron excess for transmutation of that isotope. As an example, in the case of the Am isotopes, the Am-241 transmutation is a neutron-consuming process in all the thermal concepts, while Am-243 is a slight producer. The fissile isotopes (U-235, Pu-239, and Pu-241) are net producers whatever the spectrum, with more excess neutrons in the fast concepts (e.g., average of about -1.6 for Pu-239).

Table 1. Comparison of D-Factors for the Generation-IV Concepts

Isotope	Thermal Concepts			Fast Concepts		
	PWR	VHTR	SCWR	SFR	LFR	GFR
U-235	-0.65	-0.53	-0.70	-1.04	-0.92	-0.84
U-238	-0.02	0.26	0.01	-0.89	-0.71	-0.62
Np-237	0.96	1.11	1.03	-0.88	-0.65	-0.51
Pu-238	0.01	0.12	0.07	-1.50	-1.36	-1.25
Pu-239	-0.83	-0.72	-0.80	-1.71	-1.59	-1.45
Pu-240	0.04	0.12	0.09	-1.28	-1.04	-0.94
Pu-241	-0.95	-0.88	-0.90	-1.42	-1.29	-1.27
Pu-242	0.72	0.79	0.86	-1.12	-0.72	-0.64
Am-241	0.80	0.90	0.88	-0.93	-0.67	-0.58
Am-243	-0.24	-0.20	-0.09	-1.14	-0.85	-0.83
Cm-244	-1.22	-1.19	-1.07	-1.70	-1.53	-1.53

Significant variations in the D-factor are observed between the concepts. In general, a harder neutron spectrum leads to a more favorable neutron balance. The metal-fueled SFR provides the most excess neutrons for each actinide isotope. D-factors in Table 2 indicate 0.1 to 0.5 fewer excess neutrons for the slightly softer LFR spectrum (see Fig. 1), where the nitride fuel yields an overall neutron spectrum similar in hardness to an oxide-fueled SFR. [6] As shown in Fig. 1, the GFR spectrum is even softer because of moderation by the carbide structural materials resulting in a further ~0.1 reduction in excess neutrons. However, all of the fast reactor systems exhibit a significantly more favorable neutron balance compared to the thermal systems.

Therefore, these results confirm the trends observed in Ref. 4 that fast neutron spectrum systems have a marked advantage in terms of neutron balance over thermal systems.

Of particular interest is the neutron balance of U-238 which comprises the bulk of natural uranium resources. The VHTR exhibits a significant neutron defect for U-238 (0.26 net consumption); this is because the fission peak is much less pronounced in the graphite-moderated systems (see Fig. 1) effectively suppressing the direct threshold fission of U-238. There is a slight neutron excess in the PWR, although the -0.02 will not overcome parasitic losses which will require roughly 0.38 neutrons. [5] Conversely, the fast reactors have a neutron excess ranging from 0.62 to 0.89. Thus, the fast systems have the potential to efficiently transmute the base U-238 resources, while the thermal systems will require an additional source of neutrons (fissile feed) to drive the transmutation of the U-238.

However, the neutron balance does not completely capture differences in transmutation behavior. For instance, none of the systems can go critical on U-238 alone despite any excess neutrons for complete transmutation. Furthermore, the interim product of thermal U-238 transmutation is reactor-grade plutonium, which can be used as a fissile material (neutron excess) in either type of system. Thus, another key distinction in transmutation behavior is the relative importance of U-238 capture compared to fission transmutation, particularly in the fissile isotopes. For the thermal and fast Gen-IV concepts, the one-group U-238 capture, U-235 fission, and Pu-239 fission cross sections are compared in Table 2. One observes that the fissile fission cross sections are much larger (>25 times) in the thermal systems; while the differences in U-238 capture are only ~5 times. However, the U-238 capture cross section is much larger in the VHTR (compared to the PWR and SCWR) because of its unique spectrum (see Fig. 1); the contributions of epithermal U-238 capture are much more important.

Table 2. Comparison of Key One-group Cross Sections for the Generation-IV Concepts.

Reaction	Thermal Concepts			Fast Concepts		
	PWR	VHTR	SCWR	SFR	LFR	GFR
U235f	37.23	44.18	52.80	1.53	1.69	2.09
U238c	0.91	**4.80**	0.95	0.20	0.26	0.32
Pu239f	89.17	164.54	138.78	1.65	1.69	1.90
U235f/U238c	40.77	9.20	55.75	7.54	6.60	6.59
P239f/U238c	**97.66**	34.26	146.55	**8.14**	**6.59**	**6.00**

Because U-238 capture is relatively more likely in a fast system this promotes their use as U-238 converters. However, this difference also implies that the fast systems will need a much higher enrichment to overcome the U-238 capture; compare the Pu239f/U-238c ratio of ~100 for a PWR, as compared to ~7 for the fast systems. Therefore, despite the inferior neutron balance for the individual isotopes (see Table 1), reactions in the fissile isotopes are favored in the thermal energy range. Therefore, the thermal spectrum system can operate on much lower enrichment because of the low relative probability of U-238 capture.

Based on these differences in transmutation physics, the Gen-IV thermal concepts have been configured for low enriched uranium (LEU) fuel utilization in a once-through fuel cycle. For this approach, waste management is improved by increased burnup (provide more energy from a given fuel mass) and improved thermal efficiency (more electrical power for a given energy production). In contrast, the base Gen-IV fast concepts are configured for U-238 utilization, the fissile material is recycled transuranics (TRU) and high in-core conversion rate is exploited to extend resources. For this approach, waste management is improved by increased burnup (more energy each recycle pass), improved thermal efficiency, and limited recycle losses.

Alternate Gen-IV designs (both thermal and fast) have been developed for inclusion in the fuel cycle as dedicated transuranic burners; and the transmutation performance of these burners is also compared in the subsequent sections. These burners are intended to address the transuranics (which dominate long-term heat, dose, and radiotoxicity) contained in the large stockpile of spent LWR fuel that has been generated by current U.S. nuclear plants. All of this spent fuel is currently located in prolonged storage at the reactor sites; and the total inventory will soon be equal to the original design capacity of the first geologic repository. The general approach in these burner designs is to maximize the fission of transuranics and minimize the production of new transuranics. This can be achieved by using recovered transuranics as the fissile material and removing uranium, to the extent possible. *Any fission-based system "burns" TRU at the same rate, ~1 MWt-day per gram fissioned.* However, the Gen-IV systems have a wide variety of transmutation characteristics based on spectral differences (Fig. 2) and design differences (e.g., power density, fuel enrichment, and fuel burnup limits). Burner design modifications can also lead to safety issues for both fast and thermal systems. Both thermal

(LWR) and fast (SFR) burner designs have been extensively studied in the AFCI, and this work is reviewed in Sections 3.2 and 5.2 respectively.

In the following sections, the base and burner designs for each Gen-IV concepts are described and key transmutation performance is compared. The VHTR, SCWR, SFR, GFR, and LFR concepts are addressed in Sections 3 through 7 respectively.

3.0 VHTR TRANSMUTATION PERFORMANCE

3.1 Base Gen-IV Design

The high coolant exit temperature of the VHTR makes the system quite attractive for the production of hydrogen. Consequently, the VHTR has become a leading concept for the Next Generation Nuclear Plant (NGNP) being proposed by the USDOE as a demonstration advanced system for electricity and hydrogen production. Two VHTR configurations using prismatic block and pebble fuel elements have been proposed. These are referred to as prismatic modular reactor (PMR) and pebble-bed reactor (PBR), respectively. [2] Some basic design data for the PMR design are summarized in Table 3. Both reactor types employ coated fuel particles (CFPs) dispersed in a graphite matrix. The fuel-graphite composite is contained either in a cylindrical fuel rod (fuel compact) in the prismatic block type or within a spherical graphite pebble in the pebble-bed type. In the PMR design, the fuel rods are contained in fuel holes in the hexagonal-prismatic fuel elements. Such fuel elements also have holes for coolant and control rods/materials passages, and fuel element handling. The fuel elements are stacked vertically to form a core column, with each core having multiple columns in an annular configuration. In the PBR design, the spherical graphite pebbles are contained in an annular core.

Table 3. Design Parameters for a VHTR Prismatic Block Concept.

Parameter	Value
Thermal power	600 MWt
Inlet /outlet coolant temperature	490 / 1000°C (Helium)
Average power density	6.6 MW/m^3
Fuel compound	$UC_{0.5}O_{1.5}$ or UO_2 TRISO particles
Fuel Enrichment	14 % U-235/HM
Target fuel burnup	> 100 GWd/t
Target core cycle length	~1.5 years
Thermal efficiency	47.7%
Fuel cycle	Once-through

In FY 2004, the USDOE funded a fuel cycle study to determine the feasibility of a once-through PMR to meet the cycle length and burnup goals within acceptable design ranges for the CFP kernel diameter, particle packing fraction in the matrix, and the fuel enrichment. [7] A PMR based on the GT-MHR annular core design, [8] was used in the study. A two-batch core design was found to meet the stringent requirements defined in the study: cycle length greater than 1.5 years, burnup greater than 100 GWd/t, fuel kernel diameter of less than 425 μm, fuel enrichment less than 20% (constrained to 15% in the study), and fuel packing fraction constrained to 30%. This 2-batch core design has been used to evaluate the fuel cycle performance of the VHTR; the spent fuel characteristics (composition, decay heat, and radiotoxicity) are compared to a conventional PWR in Section 3.1.1 and the uranium resource utilization is compared in Section 3.1.2.

3.1.1 Base VHTR Waste Characterization

The base VHTR operates in a once-through fuel cycle with direct disposal of the spent fuel. Thus, a detailed comparison of VHTR spent fuel to LWR spent fuel clarifies any changes in high level waste production. The spent nuclear fuel characteristics of a prismatic block-type VHTR design have been evaluated using the ORIGEN2 code and the results compared to those for medium- and standard-burnup PWRs. [9] The burnup-dependent cross sections for the ORIGEN2 calculations were generated using core analysis codes (REBUS-3 and WIMS8). These calculations gave the mass compositions, radiotoxicity, and decay heat values of the spent nuclear fuels and the results were then normalized to the corresponding values needed to produce a Giga-Watt day electric (GWe-d) of energy.

Spent Fuel Composition

The spent fuel masses of uranium, transuranics and fission product masses arising from VHTR and PWRs have been compared; results are summarized on Table 4. Although the VHTR is a thermal reactor, several features of the VHTR, which dominantly affect the spent nuclear fuel characteristics, are different from the typical PWRs. The discharge burnup for the VHTR (~100 GWd/t) is much higher than for typical PWRs (33 and 50 GWd/t in this study), and the enrichment (14%) is about 3 times higher. In addition, the VHTR core is operated under a high

specific power condition of ~100W/g, compared to about 36 W/g for PWRs. The high temperature operation of the VHTR results in a high thermal efficiency. The target thermal efficiency of the VHTR is 47.7%, compared to about 33% of PWRs. Finally, the spectrum of the VHTR is different from that of the PWRs, with the VHTR having a higher epithermal/thermal neutron component due to the use of graphite moderator (see Fig. 1).

Table 4. Comparison of Spent Fuel Properties of VHTR and PWRs.

Parameter	Reactor		
	VHTR	Medium Burnup PWR	Standard Burnup PWR
Burnup, GWd/t	~100	33	50
Thermal Efficiency (%)	47.7	33	33
Spent Fuel Mass, kg/MTU-initial (10-year cooling)			
Uranium	874.7	955.9	935.8
Transuranics	20.3	10.1	12.7
Plutonium	17.6	9.1	11.1
Normalized Spent Fuel Mass, kg/GWe-d (10-year cooling)			
Uranium	18	88	57
Transuranics	0.43	0.93	0.77
Plutonium	0.37	0.84	0.67
Pu-239	0.187	0.499	0.365
Pu-241	0.051	0.074	0.065
Am-241	0.033	0.048	0.043
Np-237	0.016	0.032	0.036
Fission Products (FP)	2.21	3.13	3.13

The high discharge burnup of the VHTR implies a relatively low normalized uranium mass in the spent nuclear fuel. Furthermore, the improved thermal efficiency reduces the initial heavy metal mass required to generate the same electricity. The net effect is that the energy-normalized uranium mass in the spent nuclear fuel is reduced by factors of 4.8 and 3.1 compared to the medium- and standard-burnup PWRs, respectively.

The high thermal efficiency of the VHTR also results in a reduction of the total fission product waste relative to the PWRs. The total mass of the fission products are directly proportional to the energy production in the core; the energy normalized fission product mass would be identical for a fixed thermal efficiency. However, the thermal efficiency of the VHTR is about 45% higher than that of the PWRs, and this results in a 30% reduction (1/1.45=0.70) in the VHTR normalized fission product waste.

Evaluations of the plutonium (Pu) and transuranics (TRU) masses reveal that the VHTR produces more Pu and TRU than the medium- and standard-burnup PWRs per initial heavy metal mass. This is because the TRU content of spent fuel increases with burnup in the range of interest. However, the Pu content saturates at high burnup because of in-cycle fission (see Fig. 3), slowing the TRU production rate. Other calculations showed that the TRU content per initial heavy metal of the VHTR and PWR at the same burnup level are comparable. When the Pu and TRU contents are normalized to the electrical energy production, however, the normalized values for the VHTR are significantly smaller (45%), because of the higher VHTR thermal efficiency and higher burnup (relative to medium and standard burnup PWRs), as shown in Table 4.

Table 5 summarizes the plutonium vector at discharge and 10 years after discharge for the VHTR and the PWRs. Figure 3 presents the plutonium isotopic masses during the irradiation period; all cases have similar depletion behavior. Following the capture of neutrons in U-238, the fissile nuclide Pu-239 is produced. A fraction of the Pu-239 produced is fissioned or converted to Pu-240 following neutron interaction. Thus, the instantaneous amount of Pu-239 depends on the balance between the production and destruction rates. This is the reason the Pu-239 mass first increases sharply and then gradually in the burnup range. By similar processes, other higher plutonium isotopes and transuranics are produced. Some transuranics decay to other nuclides (e.g., Am-241 from Pu-241, Pu-238 from Cm-242, etc.) In Figure 3, the Pu-239 masses of the PWR cases are stabilized around 6,000 g/IHMMT, while 9000 g/IHMMT for the VHTR case. This difference is attributed to increased importance of *epithermal* neutrons in the VHTR which leads to a higher U-238 capture rate (Pu-239 production), as shown in Table 2.

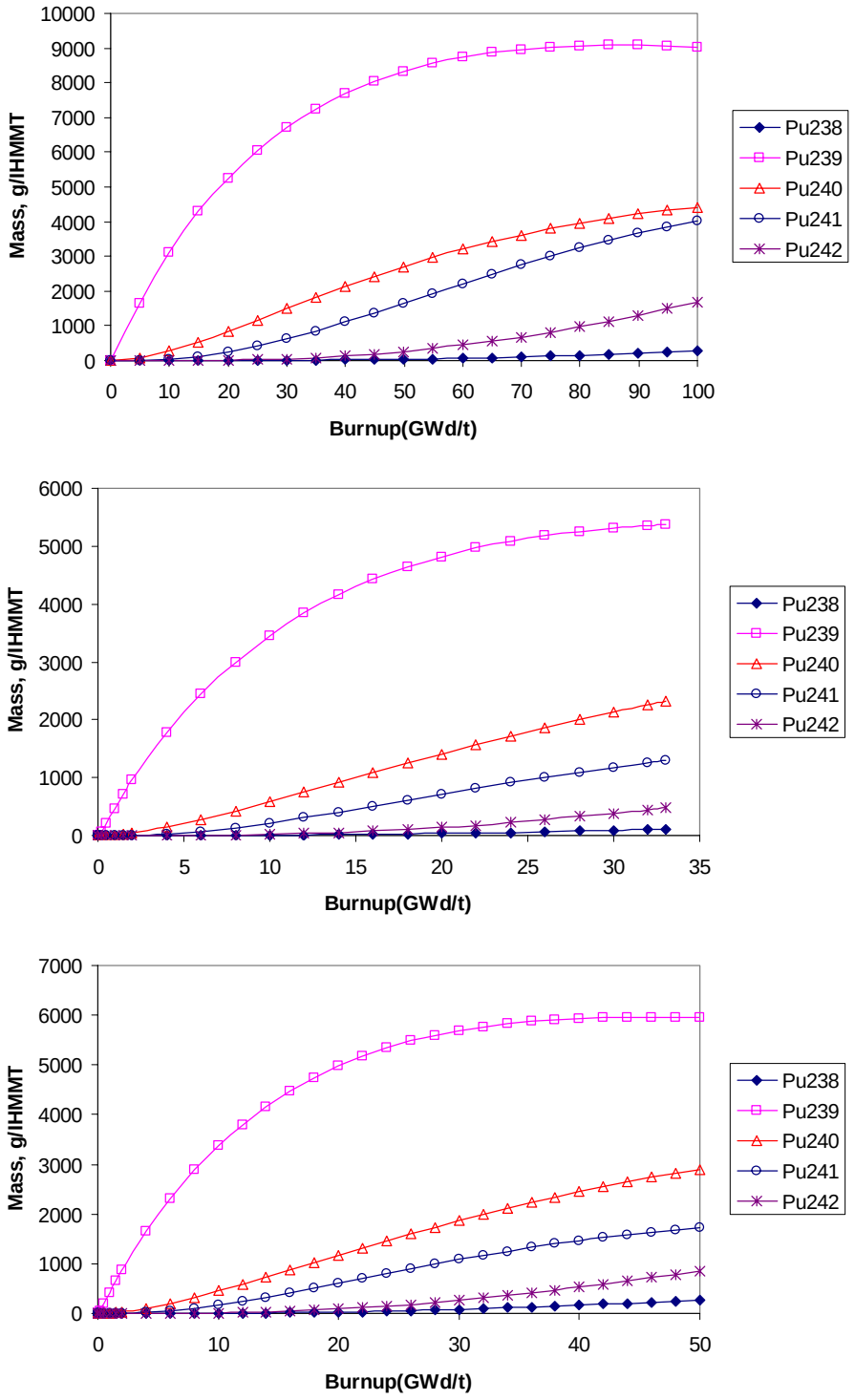

Figure 3. Plutonium Isotopic Mass over Irradiation Period in gram/IHMMT.
(Top: NGNP; Middle: Medium PWR - 33 GWd/t; Bottom: Standard PWR - 50GWd/t)

For both PWR and VHTR systems at very low burnup, Pu-239 is the predominant plutonium isotope (greater than 90% at 5.0 GWd/t for all cases). At such low burnups the plutonium material is quite attractive. The Pu-239 fraction however decreases with burnup to the 50 to 60% range as shown in Table 5. The differences in isotopic vectors between discharge and 10 years after discharge are due to the decay of the parent nuclides. For examples, the Pu-239 increases due to the decay of Np-239, and Pu-241 decays (mass decreases) into Am-241 due to its short half-life ($T_{1/2}$=14.35 years).

Table 5. Plutonium in PWR and VHTR Spent Fuel [a], %.

Isotope	VHTR 100 GWd/t Burnup		PWR 33 GWd/t Burnup		PWR 50 GWd/t Burnup	
	Discharge	10 years	Discharge	10 years	Discharge	10 years
Pu-238	1.6	1.9	1.2	1.3	2.3	2.5
Pu-239	46.2	50.7	56.1	59.5	50.7	54.0
Pu-240	22.5	24.2	24.0	25.1	24.6	26.1
Pu-241	20.9	13.8	13.7	8.8	14.8	9.6
Pu-242	8.8	9.4	5.1	5.3	7.4	7.8
Fissile	67.1	64.5	69.7	68.3	65.5	63.7

a) Pu vector is calculated as given Pu nuclide mass divided by total Pu mass.

Decay Heat

The decay heat arising from the spent fuel of the VHTR was also compared to those of the medium and standard-burnup PWRs in Ref. 9; results are summarized in Table 6. The decay heat decreases rapidly with increasing post-irradiation time. At discharge, short-lived fission products dominate the total decay heat. The short-lived fission products have half-lives (order of a day) that are much shorter than the irradiation period. Thus, they tend to saturate and their decay heat value is proportional to the fission rate. Therefore, the decay heat level per initial heavy metal mass is proportional to the specific power. The higher specific power of the VHTR results in a fission products decay heat per initial heavy mass that is a factor of 2.8 higher than that of the PWRs at discharge.

Table 6. Comparison of Decay Heat of VHTR and PWR Spent Fuel.

Parameter	Reactor		
	VHTR	Medium Burnup PWR	Standard Burnup PWR
Burnup, GWd/t	~100	33	50
Thermal Efficiency (%)	47.7	33	33
Normalized Decay Heat (DH), W/GWe-d			
Discharge	1.22E+5	1.95E+5	1.29E+5
1-year cooling	581	940	807
10-year cooling	69	101	109
30-year cooling	46	66	69
100-year cooling	18	27	26
1,500-year cooling	1.8	3.6	3.0
Integrated DH, KW-yr/GWe-d [a]	6.8	11.0	9.8
Estimated Repository Capacity	1.44	0.89	1.0

[a]Integration is over the period 100 to 1,500 years.

However, because of higher discharge burnup and higher thermal efficiency, the normalized decay heat for the VHTR is comparable to that of the standard-burnup PWR case as shown in Table 6. The fission products contribution to the total decay heat decreases with time because of their relatively short half-lives, compared to actinides in the spent fuel, and the actinides dominate after roughly 100 years.

Note that following irradiation (10 to 100 years), the short-lived *actinides* die away quickly and the decay heat of actinides per initial heavy metal mass increases with discharge burnup. This is because the longer irradiation time (higher discharge burnup) increases the content of the dominant long-term decay heat sources (Pu-238, Am-241, and Cm-244). Therefore, the decay heat per initial heavy metal mass for the VHTR is significantly higher than for the PWRs. However, due to higher discharge burnup and thermal efficiency, the energy-normalized decay heat of the VHTR is smaller than for the PWRs; the medium-burnup PWR has a 62% higher integrated energy-normalized decay heat value than the VHTR.

The nuclear waste loading in the repository is constrained by temperature, which is driven by the decay heat of the actinides. A previous analysis showed that the loading increase/decrease of directly-disposed spent fuel relative to that of the (reference) standard-burnup PWR could be reasonably estimated by taking the ratios of the energy normalized decay heat values integrated over the period from 100 to 1,500 years after discharge. [10] The energy normalized values and the deduced relative repository loading are presented in Table 6. If the fuel burnup is 33 GWd/t, which is roughly equivalent to the average burnup of existing stockpiles of spent nuclear fuel, the repository loading would be about 10% lower than the reference case to satisfy the same temperature constraints. For a VHTR that has a burnup of 100 GW/t, the loading can be increased by about 45%. A separate study indicated that relative to the standard-burnup PWR, an effective increase of about 25% in spent fuel loading is possible if the PWR has a burnup of 100 GWd/t. (No such PWR currently exists; issues of fuel/cladding performance, enrichment cost, and enrichment plant (availability) have to be addressed.) Therefore, increasing discharge burnup is one means of improving repository performance. The advantage gained by the VHTR system is due to the increased thermal conversion efficiency and burnup relative to the PWR.

Radiotoxicity

Radiotoxicity values of the VHTR and PWR spent nuclear fuels have been calculated using the cancer dose equivalent factors to convert the nuclide radioactivity to radiotoxicity. [9] The quantities derived provide an indication of the intrinsic radiotoxicity but do not account for the transport of the hazards to locations away from the repository (e.g., about 20 km used in Yucca Mountain assessments). Energy normalized radiotoxicity (cancer dose) values from discharge to one million years after discharge have been calculated. Results are summarized in Table 7.

Generally, the trend of the radiotoxicity is quite similar to that of the decay heat. The radiotoxicity of the fission products dominate at the discharge stage but quickly decreases due to their relatively short half-lives. Consequently, the radiotoxicity of the actinides dominates a few hundred years after discharge. The radiotoxicity per initial heavy metal mass of the VHTR is higher than that of the PWRs while the energy-normalized radiotoxicity of the VHTR is smaller

due to the higher discharge burnup and thermal efficiency. The results showed that the dominant contribution of fission products decays out within a hundred years and Am-241 becomes the dominant contributor in the time scale of 100 and 1,000 years. Following this time, Pu-239 and Pu-240 are dominant in the time scale of 1,000 and 50,000 years.

Table 7. Comparison of Spent Fuel Radiotoxicity.

Parameter	Reactor		
	VHTR	Medium Burnup PWR	Standard Burnup PWR
Burnup, GWd/t	~100	33	50
Thermal Efficiency (%)	47.7	33	33
Radiotoxicity (Cancer Dose), Sv/GWe-d			
Discharge	4.70E+7	4.12E+8	2.90E+8
10,000-year cooling	1.17E+5	2.89E+5	2.30E+5
100,000-year cooling	8.94E+3	1.89E+4	1.78E+4
1,000,000-year cooling	2.71E+3	5.93E+3	5.16E+3
Np-237 (1,000-year cooling)	195	302	286
Np-237 (100,000-year cooling)	229	350	329

Finally, it is noted that Np-237 is the leading nuclide of interest in the hundreds of thousand-year time scale in Yucca Mountain assessments. The results obtained in this work indicate that at 1,000 and 100,000 years, the normalized quantity of Np-237 for the VHTR is smaller than for the PWRs (difference is about 65%). It is noted that a large fraction of this Np-237 comes from the decay of Pu-241 and Am-241 and the sum of their masses is smaller for the VHTR (see Table 4). The lower value of Np-237 for the VHTR indicates that the source term will be less than for the PWRs. However, dose rate must include fuel degradation and geological modeling; this evaluation is underway for coated fuel particle forms in AFCI studies.

3.1.2 Uranium Ore Utilization

The AFCI effort has primarily focused on the back-end of the nuclear fuel cycle, with the aim of reducing the mass of spent UO_2 that must be disposed and/or reducing the masses of heat-producing transuranic nuclides in the nuclear waste. However, the AFCI energy security and Gen-IV sustainability goals also require an assessment of resource utilization; the limits and

availability of uranium resources is highly speculative, but constraints may be encountered if nuclear power continues. [11] Recently, the impact of advanced fuel cycles on uranium utilization was evaluated. The uranium requirements of the VHTR and several PWR systems are compared in Table 8. The enrichment requirements for the water-cooled systems were calculated assuming a three-batch fuel management scheme, compared to the two-batch VHTR as dictated by the NGNP study.

Table 8. Enriched UO_2 and Uranium Ore Requirements in PWR and VHTR Systems.

LWR and ALWR (*Thermal Efficiency=33%; Capacity factor=85%*)				
Discharge Burnup (GWd/MT)	**Enrichment**	**UO_2 Requirement (MTIHM/year/GWe)**	**Feed/Product[1]**	**Uranium Ore Requirement (MT/year/GWe)**
33	3.00%	29	5.48	156
45	3.85%	21	7.14	149
50	4.20%	19	7.83	147
100	8.50%	9.4	16.24	153
VHTR (*Thermal Efficiency=47%; Capacity factor=85%*)				
Discharge Burnup (GWd/MT)	**Enrichment**	**UO_2 Requirement (MTIHM/year/GWe)**	**Feed/Product[1]**	**Uranium Ore Requirement (MT/year/GWe)**
102	14.0%	6.5	27.02	175

[1]Feed/product ratio calculated assuming tails enrichment of 0.2 wt.% U-235.

While increasing the discharge burnup decreases the mass of spent fuel generated each year and the energy-normalized transuranic discharge mass, the normalized uranium ore requirements are marginally increased because of the higher enrichment of the VHTR compared to a similar burnup PWR (14% versus 8.5%). The higher uranium enrichment for the VHTR is due to its higher specific power density (98 versus 36 W/g) and lower fast fission fraction compared to the PWR, which results in a faster depletion of the U-235. [12] Because of its higher fuel enrichment, the uranium ore requirement/GWe for the VHTR is ~15% higher than for the PWR. It should be noted that fissile content of the *spent uranium* in the VHTR system is 5.6% (compared to 1.1% for the high-burnup LWR fuel), suggesting that the fuel design could

be optimized to reduce the fresh fuel enrichment; alternatively, the spent uranium could be recycled and re-enriched. Either approach would reduce the ore requirements.

3.2 VHTR Transmutation Concepts

3.2.1 General Thermal Recycle Considerations

Significant research on transmutation and multi-recycle of reactor-grade transuranics in thermal systems (e.g., conventional PWRs) has been conducted both in the AFCI program and internationally. These studies indicate that PWRs can be used for the transmutation and recycle of transuranics fabricated into MOX or non-uranium fuel. It should be possible to use VHTRs in a similar manner. However, several constraints for multi-recycle in thermal systems have been identified:

- An external fissile feed is required to "support" the transmutation.
- A technique to manage the higher actinides (Cm and above) must be employed.
- Key safety performance parameters (reactivity coefficients) may be adversely impacted.

As discussed in Section 2, the fissile nuclides are preferentially fissioned in the thermal spectrum; and the fertile species do not produce adequate neutrons for complete destruction. Thus, the continuous recycle of the transuranics (TRU) would be constrained by core reactivity balance. A variety of techniques to provide fissile support (e.g., using enriched uranium) have been conceived [13, 14] to facilitate extended recycle.

As shown in Section 2, fertile nuclides in the thermal-spectrum have a high neutron capture-to-fission ratio, resulting in the buildup of higher actinides with irradiation and recycle of the transuranics. Initial work (ANL-AAA-020: *Assessment of TRU Stabilization in PWRs* [15]) demonstrated that it is possible to recycle all TRU in the heterogeneous assembly PWR CORAIL concept with the 5% enrichment LEU pins and MOX pins at equilibrium TRU content of ~23% (significantly higher than Pu recycle). However, a build-up of higher actinides results from the relatively high capture-to-fission neutron interactions ratios in a thermal spectrum. Although initial recycles may be possible, eventually very high actinides (e.g., Cf-252) will build

in and make the fuel quite difficult to handle (e.g., neutron source four orders of magnitude larger than that observed for more conventional MOX forms).

Therefore, steps must be taken to suppress the generation of the higher actinides. One approach as taken in the recent ORNL study [16] if to interpose a long cooling time (40 years) to allow decay of the Cm-244 (18.1 year half-life with decay back to Pu-240) and the Pu-241 (14.4 year half-life with decay to Am-241, which is likely to bypass curium through Am-242 decay). However, this approach will only mitigate the generation of higher actinides; and at equilibrium significantly elevated decay heat and neutron sources levels (at least 100 times) are expected.

Another approach is to remove the higher actinide elements in the separations process; for example in Ref. 10 the Cm was removed and stored each recycle. This largely avoids the elevated neutron source, although increases in the decay heat and gamma source (of roughly an order of magnitude) are observed from the inclusion of Am. This strategy will incur the added expense of Cm separation particularly from Am and associated storage facility (unless the Cm stream can be directly disposed with the fission product wastes); however, it appears to be quite effective in preventing the generation of higher actinides.

Finally, the utilization of fissile TRU fuels in a thermal spectrum hardens the neutron spectrum; and the increase in the fertile nuclide fraction with burnup exacerbates the spectral hardening. These changes can result in adverse impacts to thermal reactor performance; for example at high Pu content, a positive void coefficient can occur in PWRs. This would limit the amount of TRU loading in the core. Previous studies have indicated that from a neutronics viewpoint some of these problems could be overcome by using external enriched uranium (~5%) and blending strategy to support the transmutation/recycle mission. [15]

Other thermal spectrum recycle concepts (e.g., VHTR and SCWR) will have to deal with these same issues. Similar approaches for fissile support and higher actinide management have been proposed. The VHTR may have some advantages with regard to the safety impact of TRU fuels. Because moderator voiding is not an issue, the void coefficient issue can be avoided making high enrichment fuels more usable. In addition, direct disposal of the TRISO fuel particles after limited recycle may provide a better waste form than LWR spent fuel.

3.2.2 Deep-Burn Concept

The deep-burn concepts aim at burning the transuranics recovered from spent nuclear fuel (SNF) to very high burnup levels, presumably 100%. The concept would therefore be very useful in a nuclear energy future where the intent is to *eliminate* the transuranics in spent fuel or to burn them down in order to stabilize the transuranic inventory. There are however two primary practical barriers to complete burn-down in a single reactor loading. First, since non-uranium or inert matrix fuel are used in the thermal reactor system to ensure that no net higher transuranics are produced, the core reactivity decreases as a function of burnup, particularly when using TRU compositions recovered from LWR spent fuel. Therefore, the designer is constrained by the requirement to keep the core critical, which might in turn limit the burnup level, and the requirement to minimize the burnup reactivity swing. Alternative designs in which other fuels (e.g., enriched uranium fuel) are used to support the irradiation of the TRU could be conceived. Even in this case, the second constraint, defined by fuel performance and fuel safety requirements, might become limiting. Issues requiring resolution in this area include (1) the ability of the fuel coating or fission-gas release barrier to withstand the enormous pressure from the fission gas buildup with irradiation, (2) significance of the fuel-to-coating interaction effect with increased burnup, and (3) the ability of the very high burnup fuel to withstand core transient involving insertion of positive reactivity (power ramps).

Deep-burn concepts have been proposed by different groups for the burnup of transuranics in nuclear systems. More recently, General Atomics (GA) has proposed this concept for the gas-turbine modular high temperature reactor. [17] In FY 2005, the U.S. Congress earmarked $3M for the evaluation of this concept to the University of Nevada, Las Vegas with support from GA. U. S. national laboratories are also contributing to this effort. In the following paragraphs, the intent and status of the concept are discussed.

The GA proposals for deep burning of TRU in the GT-MHR or VHTR have included one-pass, two-pass, and multi-pass options. A four-batch fuel loading scheme has been considered for the cases in general. In the one-pass scheme, TRU fuel extracted and fabricated from LWR spent nuclear fuel is burned in the VHTR for four core cycles. Fuel burnup as high as 60-70% has been postulated by GA. Following irradiation, the fuel is conditioned for long-term

burial in the repository. Work is currently being done in FY 2005 to determine if the TRISO fuel barrier (SiC) can contain the fuel particle/material for 1 to 2 million years in the repository environment. Separately, work is also being done to investigate the feasibility of reprocessing the VHTR fuel element. The reprocessing would allow the possibilities for the two-pass and multi-pass options.

Preliminary evaluations conducted by ANL have shown that a fuel consumption of about 60% is feasible in a single pass. The study showed that the fuel burnup is greatly dependent on the fuel particle dimension, with a value around 200 µm being optimum. The results from the study are summarized in Table 9.

Table 9. Consumptions in 800 MWt VHTR Equilibrium Core using 200 µm Fuel.

Nuclide	Charge mass (kg)	Discharge mass (kg)	Consumption (%)
Np237	13.75	5.87	-57.3
Pu238	3.99	19.29	383.1
Pu239	152.63	3.85	-97.5
Pu240	62.24	16.00	-74.3
Pu241	22.65	17.97	-20.7
Pu242	14.80	34.52	133.2
Am241	24.60	1.95	-92.1
Am242m	0.09	0.04	-57.5
Am243	4.52	9.42	108.1
Pu	265.32	91.63	-64.3
HM	299.29	119.44	-60.1

The results are preliminary in nature, because of the simplifying assumptions made in the core physics model. For example, burnable poison which would be required for controlling the core burnup reactivity has not been modeled in the study. Confirmation of core safety has also not yet been performed. The results, however, suggest that from a neutronic viewpoint, a high burnup can be achieved with the use of TRU in VHTR core.

An estimate of the repository loading benefit indicated that 60% burnup of non-uranium fuel in the VHTR would result in a repository benefit of about 2. However, further destruction may be needed to meet AFCI goals. A different version of the deep-burnup concept, the two-pass option, has been proposed to reduce the high heat and radiation generating nuclides (e.g., Pu-238,

Am-241) passing to the repository, and to increase the repository loading benefit. The approach uses both driver and transmutation fuels in the VHTR fuel element to burn the fuel further and thus reduce the amount of these nuclides. Practically, different particle dimensions may be used for the driver and transmutation fuel particle types in order to improve the transmutation performance. Burnups of the order of 70 to 75% have been postulated by GA for the two-pass scheme. An example of the fuel cycle of a two-pass scheme is shown in Figure 4.

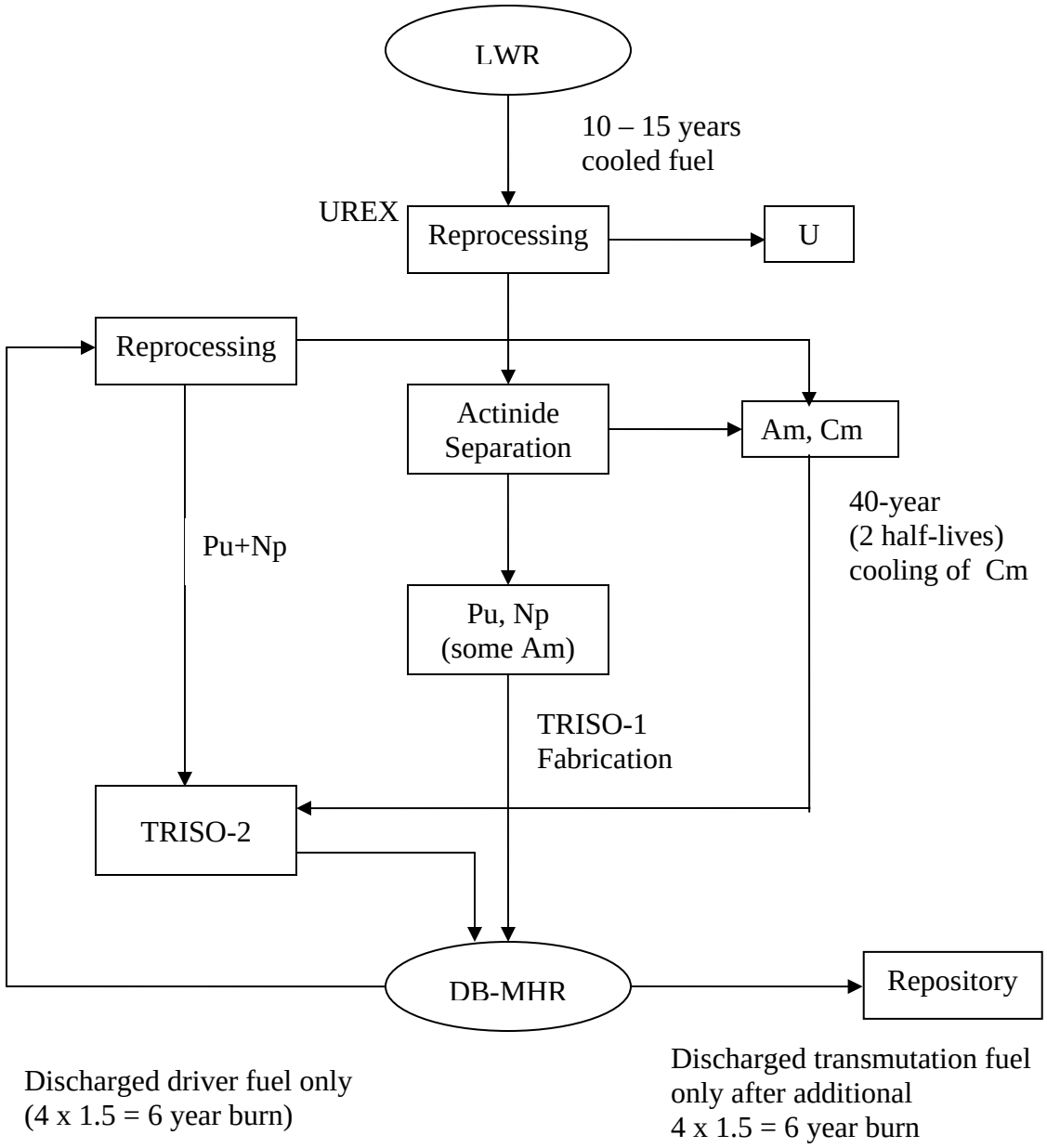

Figure 4. Schematics of the VHTR Deep Burn Two-Pass Transmutation Option. [18]

4.0 SUPERCRITICAL-WATER REACTOR DESIGNS FOR AFCI MISSIONS

4.1 Base SCWR Concepts

The supercritical water-cooled reactor (SCWR) has attracted renewed interest because of its potential for high thermal efficiency and considerable plant simplification. The SCWRs are high-temperature, high-pressure water-cooled reactors that operate above the thermodynamic critical point of water (374°C, 22.1 MPa). Core designs operating with thermal or fast-neutron spectrum, have been conceived; with the base design a thermal system.

Many thermal SCWR concepts have been considered, with the design focus being on economics. This is realized by achieving a high thermal efficiency and the relatively lower-coolant mass flow per unit thermal power resulting from the higher enthalpy content of the supercritical coolant, compared to LWRs. The lower coolant mass allows for significant reduction in the size of reactor coolant pumps, piping, and associated equipment, and a reduction in the pumping power. It also suggests smaller containment buildings. [19] Additionally, because of the physical state of the coolant, no boiling crisis (i.e., departure from nucleate boiling or dry out) exists, thus avoiding discontinuous heat transfer regimes within the core during normal operation. Furthermore, steam dryers, steam separators, recirculation pumps, and steam generators are not utilized.

In most of the thermal SCWR designs, including the Japanese supercritical light water reactor (SCLWR), the reactor vessel is similar in design to a PWR vessel. However, the containment and emergency core cooling systems are similar to those of BWRs. Generally, the direct energy-conversion cycle is actively pursued. Table 10 contains some design parameters for the U.S. reference SCWR concept. [20]

The thermal efficiency of the SCWR (45%) is similar to that of the VHTR and significantly higher than for LWRs. This suggests that for the same burnup as the PWR, the waste mass, decay heat and radiotoxicity per unit energy production would be proportionally lower in the SCWR. However, the significant differences observed between the VHTR and the PWRs (Section 3.1) would not be realized for the SCWRs because of its 45 GWd/MT burnup, comparable to current PWRs, but one-half the design target for the VHTR.

Table 10. Design Parameters for a Thermal-Spectrum SCWR Concept. [20]

Parameter	Value
Thermal Power	3575 MWt
Inlet /outlet coolant temperature	280/500°C
Average power density	69.4 MW/m^3
Fuel form	UO_2 pellets
Fuel enrichment	~5% U-235
Fuel burnup	45 GWd/t
Thermal efficiency	44.8
Fuel Cycle	Once-through

The uranium utilization of the SCWR fuel cycle should be about 25-30% better than that of the PWR; proportionately to the thermal efficiency (systems use similar enrichments).

By reducing significantly the moderation in the core, the SCWR can also be designed as a fast reactor. A direct-cycle, SCWR fast converter design (designated SCFR-D) was studied in Japan. [21] The system is designed for a large power rating (1,000 MWe), negative coolant void reactivity coefficient, and high discharge burnup. The coolant inlet and outlet temperatures are 310 and 431°C, respectively. The core burnup is 80 GWD/t and the conversion ratio is 0.95. Core average power density is 200 MW/m^3. The SCFR-D uses a mixed U-Pu oxide fuel; the plutonium is from spent LWR fuel. The plutonium content in the heavy-metal is about 15%; core fissile content is 6 tons. A stainless steel cladding was considered in the design. A zirconium-hydride layer between the core seed zone and the blankets is utilized to ensure a negative coolant void reactivity coefficient. With a conversion ratio of 0.95, the SCFR-D approaches a fissile sustainable design that conserves uranium resources. Some design parameters for the Japanese concept are presented in Table 11.

The fast-spectrum reactor design is attractive for full actinide recycle and consequently can reduce long-term waste toxicity source term, minimize the amount of weapons-useable material in repository, increase repository capacity by reducing long-term decay heat generation, and improve repository performance by reducing radiation damage on the final waste form. [19]

Table 11. Design Parameters for a Fast-Spectrum SCWR Concept. [21]

Parameter	Value
Thermal Power	3640 MWt
Inlet /outlet coolant temperature	310/431°C
Average power density	199 MW/m^3
Fuel compound	MOX pellets
Pu content in heavy-metal	~15%
Fuel burnup	77 GWd/t
Thermal efficiency	41
Fuel Cycle	Pu recycle

4.2 SCWR Transmuter

Because both fast and thermal spectrum SCWR configurations have been considered, the potential for a mixed spectrum single reactor design that accommodates both the economics and actinide management goals was the focus of a USDOE NERI project involving the University of Wisconsin-Madison and the Argonne National Laboratory. [22]. This design, called a mixed spectrum SCWR (MS^2), utilizes both an outer thermal-spectrum zone and an inner fast-spectrum zone, separated by a thermal blanket (shielding) zone. The water coolant first passes through the outer thermal-spectrum zone and then through the inner fast-spectrum zone. At the inlet of the fast-spectrum zone, the coolant is at the critical temperature. This design allows the MS^2 to be a boiler (outer zone) and an actinide burner (inner zone). A schematic of the MS^2 is shown in Figure 5.

Different assembly designs are used in the fast and thermal zones. The core design parameters for the MS^2 are summarized in Table 12. The pitch to diameter ratio (P/D) of the outer zone is similar to that of a PWR. This P/D value is reduced significantly for the inner zone in order to achieve a fast spectrum; the inner core utilizes more fuel pins. To ensure negative coolant void reactivity coefficient over the operating range, the design uses 25 void assemblies in the inner core zone, to enhance neutron leakage following coolant voiding. Chemical shim is not utilized in the core in order to allow the coolant to be used in the turbine of a direct cycle plant configuration. Enriched uranium-dioxide fuel in used in the outer thermal core zone. In the

inner core, a thorium-based mixed oxide fuel is used; composition is 15% Pu, 42.5% Th, and 42.5% U (6% enriched). The fissile fraction of heavy metal in the inner zone is ~11%. For the initial core, plutonium discharged from spent ALWR fuel is used. The MS^2 has a cycle length of 283 days at full power.

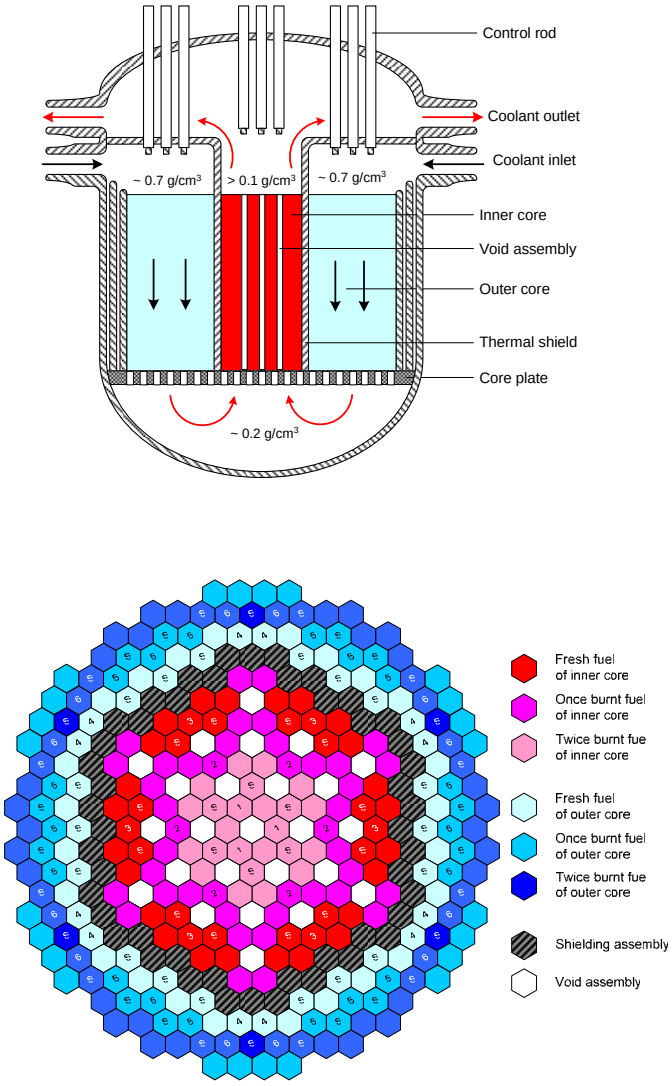

Figure 5. Axial and Planar Layouts of MS^2 Core. [22]

Table 12. Mixed-Spectrum SCWR Transmuter Design Parameters. [22]

	Parameter	Inner zone	Outer zone
General core properties	Thermal power, MW	3400	
	Thermal efficiency, %	44.1	
	System pressure, MPa	25	
	Number of fuel assemblies	108	68
	Number of shield assemblies	48	
	Number of void assemblies	25	0
	Active core height, cm	280	280
	Effective radius, cm	198	
	Power density, MW/m^3	157	85
	Average linear heat rate, W/cm	154	125
	Inlet temperature, ^oC	387	280
	Average outlet temperature, ^oC	553	387
Fuel and Assembly	Fuel form	TMOX*	UO_2
	Fissile enrichment, w/o	11	6.5
	Cladding material	Ni-Alloy	
	Fuel radius, cm	0.44	0.41
	Cladding thickness, cm	0.040	0.057
	Fuel pitch, cm	1.0	1.2
	P/D ratio of fuel pin	1.14	1.47
	Assembly shape	Hexagonal	
	Assembly pitch, cm	20.7	20.7

*TMOX means thorium-based mixed oxide fuel.

For the equilibrium cycle of the MS^2, the inner-core fuel is fabricated from external depleted uranium and thorium, and the Pu, Np, and Am (PNA) extracted from the discharged fuels of the outer and inner zones of a previous recycle stage. Thus, the PNA is continuously recycled in the inner fast zone. This limits the material passing to the repository to curium and the waste stream from the reprocessing stage (fission products, other actinides and PNA losses). As designed, the TRUs are created in the outer core and burned in the inner core, see Table 13. The fissile content in the PNA vector decreases from 64% to 58% over the first recycle stages. The uranium enrichment for the outer core increased from 6.0% to 6.5%. The results showed that the PNA is nearly stabilized over the core cycle. The net PNA mass balances is 47 kg in cycle 5, which is small compared to the total PNA mass of the fresh fuel in the inner zone (3,600 kg).

Table 13. Summary of Pu, Np and Am Multi-recycling in MS^2 Concept. [22]

Parameter		Cycle 1	Cycle 2	Cycle 3	Cycle 4	Cycle 5
U enrich. of outer zone,%		5.28	5.82	6.50	6.50	6.50
U enrich. of inner zone, %		6.0	6.5	7.0	6.5	6.5
PNA content in HM, %		15.0	14.9	14.9	15.1	15.0
Fissile content in PNA,%		64.2	61.1	60.3	59.1	57.7
Net mass balance, kg/Core (inner/outer)	Pu	-228.4 / 287.9	-200.9 / 243.2	-185.0 / 217.8	-179.7 / 200.9	-168.9 / 206.3
	Np+Am	42.2 / 9.2	27.5 / 10.2	15.3 /10.7	5.0 / 9.8	-2.6 / 10.1
	Others	3.5 / 0.0	5.4 / 0.2	6.6 / 0.3	8.6 / 0.3	9.6 / 0.3
	Net PNA	110.9	80.0	58.8	35.9	47.0
Charge vector of PNA	Am241	0.91	3.60	3.81	4.55	5.26
	Am242m	0.00	0.01	0.02	0.04	0.06
	Am243	0.00	0.15	0.30	0.43	0.55
	Np237	0.00	0.08	0.17	0.38	0.50
	Pu238	3.19	2.86	2.75	2.70	2.73
	Pu239	55.15	55.02	54.44	53.88	53.26
	Pu240	25.93	26.44	26.90	27.09	27.50
	Pu241	9.06	6.06	5.80	5.15	4.37
	Pu242	5.70	5.74	5.76	5.73	5.71
	U234	0.05	0.04	0.04	0.04	0.04
	U236	0.01	0.01	0.01	0.01	0.01

The feasibility study of the MS^2 concept is incomplete. Items requiring further evaluations include core stability, shielding requirements, the mechanical strength of the structure separating the fast and thermal zones in the upper plenum, impact of U-233 in the repository, and repository loading benefits.

5.0 SFR TRANSMUTATION SYSTEMS

Fast reactor systems are quite adaptable to the missions of the AFCI program. These systems can be configured as TRU sustainers or as TRU burners. The former mission is important in fuel cycles where fuel material utilization is a major consideration. In this mode, the fuel could be continuously recycled with minimal need for uranium enrichment, since either depleted or natural uranium would be used as make-up feed to the fuel cycle. TRU burners on the other hand are required in fuel cycles with a need to burn down waste materials; this could be either to complement once-through systems that are producing TRU or for the purpose of drawing down the accumulated legacy nuclear waste.

In this section, the transmutation impacts of the Generation-IV SFR nuclear system are explored; the other two fast-spectrum designs (GFR and LFR) are evaluated in Sections 6 and 7. A brief introduction of the base Gen-IV design is first given. For all three fast systems, the sustainable approach (first design option noted above) is employed. Subsequently, the transmutation capabilities of alternate burner designs are compared. Because more design work has been performed on the SFR burner concept, general considerations for fast spectrum burners are identified in Section 5.2. In addition, a limited evaluation of the safety and economic impacts of SFR burner design is included. An intercomparison of the fuel cycle roles and burner performance of the Gen-IV systems is presented in Section 8.0.

5.1 SFR Gen-IV Systems

Of the six reactor concepts favored under the Gen-IV program, the sodium-cooled fast reactor (SFR) is one that has been studied and developed extensively in countries such as the U.S., Russia, France, U. K., and Japan. A variety of SFRs have been operated, including critical facilities (e.g., ZPRs and ZPPR in the U.S., MASURCA in France, BFS in Russia, ZEBRA in the U.K, SNEAK in German, FCA in Japan), and experimental, prototype and power reactors (e.g., Fermi I , EBR-II and FFTF in the U.S., Phenix and Superphenix in France, PFR in the U.K, JOYO in Japan, BN-350 and BN-600 in Russia and FBR in India). Advanced sodium-cooled fast reactor concepts are also being pursued in several countries.

The possibility for resource extension with recycle in fast spectrum systems is undisputed. However, the primary mission recently identified for the SFRs under the Gen-IV program is the management of high-level wastes and, in particular, management of plutonium and other actinides (termed actinide management). [19] Other missions include electricity production and possibly nuclear hydrogen production. In the Gen-IV outlook, the latter missions are only possible if innovations to reduce capital cost are integrated in the design philosophy for these systems. Additionally, hydrogen production in such systems requires that more innovative and efficient approaches be developed at lower operating temperatures than being considered for the VHTR. This is because the outlet temperatures of existing and planned SFRs are typically less than 600°C.

Some of the design attributes that favor SFRs are the typically large margin to coolant boiling and operation near atmospheric pressure. In systems using steam for the power conversion, however, an intermediate loop is used to separate the radioactive sodium coolant in the primary loop from that of the steam/water in the Rankine-cycle side. This is a consequence of the strong chemical interaction between water/air with sodium.

Different primary coolant systems layouts have been considered for advanced SFR. These have included the loop type, favored in Japan, and the pool type. Reactor plant sizes ranging from 10 MWe to 1,700 MWe have also been considered. At the low end are the modular systems for remote site power sources (e.g., that being proposed for Galena, Alaska or military sites and the recent small modular fast reactor (SMFR) design being jointly pursued by ANL, CEA, and JNC).

For the SFR fuel cycle, two fuel forms have typically been considered: oxide fuel and metal fuel. Nitride fuel is also being considered in the SFR advanced fuels research and development scope. For all the advanced sodium-cooled reactor concepts, the fuel would be recycled using an advanced aqueous process or the pyrometallurgical process. While oxide fuels have generally been used and are mature for fast reactor applications, metallic fuels were developed in the U.S. fast reactor program (i.e., EBR-II) and shown to have safety advantages. Although the metallic fuel melting temperature is much lower than that of oxide fuel, it is more difficult to raise its fuel temperature because of a high thermal conductivity (10 times that of

oxide). As a result, operating margins can be greater for metal cores than for oxide cores. Additionally, because of the lower positive Doppler reactivity invested in the metallic fuel in generic anticipated transient without scram events, such as unprotected loss-of-flow (ULOF) and unprotected loss-of-heat-sink (ULOHS), the inherent safety characteristics of the metal fueled core are superior. [23]

SFR systems, configured for both high conversion ratio (greater than 1.0) and low conversion ratio (<0.5) have been extensively developed. However, the Gen-IV systems are targeting toward high conversion ratio, consistent with the sustainability (uranium resource) goals. For actinide transmutation missions, low conversion ratio systems have been analyzed primarily in the AFCI, as discussed in Section 5.2. The design features of the base Gen-IV SFR are summarized in Table 14.

Table 14. Basic Design Parameters for Current GIF SFR Systems. [20]

Parameter	**Value**
Plant Power, MWe	600 - 1,500 (Middle-Large)
Thermal Power, MWt	1,589 - 3,570 (Middle-Large)
Plant Efficiency, %	38 - 42
Outlet coolant temperature, ^{o}C	510 - 550
Inlet coolant temperature, ^{o}C	366 - 395
Average Power Density, MW/m^3	200 – 400 (typical)
Cycle length, months	18
Fuel Type	Metal(U-TRU-10%Zr Alloy), MOX(TRU bearing)
Fuel Enrichment, %TRU/HM	10 – 20 (typical)
Burn-up, GWd/t	70 - 200
Breeding ratio	1.0 - 1.2
Fuel Cycle	TRU and U recycle

As noted in Section 2, the enrichment levels (20%) for the SFR are significantly higher than a typical LWR; this is required to overcome enhanced U-238 capture in the fast spectrum. These high enrichment fuels are generally run to high burnups; the design target for advanced SFR fuels is 200 GWd/MT. Because of the excellent internal conversion in the fast spectrum, fuel life is not limited by reactivity constraints, but rather by structural irradiation damage. Thus, work on advanced cladding materials is also being pursued.

The SFR fuel cycle employs complete recycle of the TRU and uranium. This significantly extends fuel resources as only depleted uranium is required as a make-up feed. Furthermore, TRU are nearly eliminated from waste (~0.1% recycle loss each pass). The actinide-free waste has significantly reduced TRU mass, long-term heat, and radiotoxicity, compared to once-through fuel cycles. Thus, these features lead to exceptional performance with regard to the Generation-IV sustainability criteria related to resource utilization and waste management.

Another key feature of the SFR is the relative maturity of the technology. Several power reactors (e.g., BN-600, Superphenix) have deployed much of the basic technology. The estimated R&D costs for the Gen-IV SFR are lower than any of the other Gen-IV systems: $610M as compared to $670M for VHTR and $900-1000M for others.

5.2 SFR Burner Design Studies

The effectiveness of fast spectrum systems for actinide transmutation has been well documented. The key advantage of the fast spectrum resides in the high fission/absorption ratios, as shown in Section 2. For most fissile isotopes in a fast spectrum, nearly 90% of neutron absorptions result in fission, and fission fractions greater than 20% are observed for fertile isotopes. Thus, the production of higher actinides is inhibited in a fast spectrum system. These higher actinides tend to dominate the waste heat, dose, and radiotoxicity of spent fuel and thus, there are benefits to minimizing their amount in spent fuel.

The wide range of conversion characteristics achievable in a fast spectrum system provides for flexibility in TRU management strategy. As shown in Section 2, the neutron balance in a fast spectrum allows a net production of fissile material from U-238. The traditional fuel cycle approach exploits this behavior by employing excess fertile material (blankets) to breed additional TRUs. These fast breeder reactor configurations were envisioned for a rapidly expanding nuclear power economy where fissile material was scarce and expensive. Conversely, the fast reactor can readily be configured to achieve a net destruction (burning) of TRUs. Given the current status of the nuclear fuel cycle with stockpiles of excess weapons material, separated civil plutonium, and TRU-containing spent fuel, fast burner reactor configurations have been targeted in recent studies to reduce the global TRU inventory.

In a standalone mode, a fast burner reactor can reduce existing stockpiles of TRU material. Alternately, fast burner systems can be utilized to complement a sustained nuclear power capability using once-through (enriched uranium) systems that inherently produce TRUs. As shown in multiple strata studies, [24] a limited capacity of fast burner systems can be utilized to stabilize the total TRU inventory. This approach may be advantageous as it allows exploitation of current enriched uranium resources without continued buildup of once-through fuel cycle wastes. For either of these missions (stockpile reduction or waste transmutation), high net TRU consumption rates are advantageous; the number of systems required to complete the mission will be directly dictated by the consumption rate.

To achieve net consumption of TRU, their fission rate must exceed the production rate of new TRU. The conversion ratio is a useful measure of the relative rates, defined as

$$TRU\ conversion\ ratio\,(CR) = \frac{TRU\ \ production\ rate}{TRU\ destruction\,(\,fission)\,rate}$$

Thus, CR=1 denotes a TRU sustaining system, and low conversion rates are targeted for fast burner systems. It is important to note that the fission of actinides (regardless the reactor type) results in ~1 MWt-day of energy per gram destroyed. Assuming that TRUs are the dominant fission isotope, this implies that the net TRU destruction is (1-CR) grams per MWt-day of energy produced by the reactor system.

Fast burner reactor designs have been developed for a variety of missions. In Ref. 25, burner configurations were developed for weapons plutonium disposition. In particular, a moderate burner design (CR~0.5 using conventional fuel forms) and a pure burner design (CR=0, using nonuranium fuel form) were developed. In the French CAPRA study moderate burners were considered with enrichments as high as 45% Pu/HM. To summarize these original studies:

- Conversion ratio was constrained to ~0.5 by conventional fuel composition limits.
- Some evaluation of nonuranium fuels was conducted, and significant perturbations in the reactivity coefficients were observed.

In recent AFCI fast burner studies, the *entire range of TRU fuel compositions (not just conventional 0-30% TRU/HM) was investigated*. Eliminating the enrichment limit gives more

flexibility in the design of burner fast reactor systems. Furthermore, this allows the development of low conversion ratio systems (CR<0.5) without resorting to the extreme limit of nonuranium fuels (with associated safety implications). As the conversion ratio is reduced, the fast reactor reactivity feedbacks will vary. Thus, a critical evaluation of the safety performance was performed to refine fast burner design techniques and mitigate unfavorable impacts.

The SFR burner designs pursued for AFCI have continued to look at modular (840 MWt) ternary metal (U/Pu-10Zr) fueled SFR systems as defined in Ref. 25. Two options were considered to achieve core designs with low conversion ratio. First, geometric spoiling (increasing the effective radius of the core and/or decreasing core height) can increase neutron leakage. Second, directly reducing the fertile material content of the fuel can reduce the production of TRU elements. Parametric studies showed that in reducing the conversion ratio by modifying the core geometry, the effectively core radius can become unacceptably large; and a practical CR limit for this approach is about 0.5. [26] However, very low conversion ratios can be achieved by modest reductions in the fuel pin diameter while maintaining the assembly size; that is reducing the fuel volume fraction. Core designs with conversion ratios varying from 0.5 to 0.0 were considered, using transuranics extracted from spent LWR fuel. [27]

Reactor performance parameters and reactivity coefficients have been evaluated for compact core configurations having conversion ratios ranging from 0.0 to 0.5. Table 15 summarizes the reactor performance data for four such burners. [27] In the core designs, two enrichment (TRU content) zones were utilized to reduce radial power peaking. As would be expected, with smaller pins the fuel volume fraction decreases, leading to an increase in TRU enrichment to maintain criticality; this significantly reduces the fertile loading which in turn decreases the conversion ratio. As expected, the CR=0 burner achieve the maximum possible TRU consumption rate of 1 g/MWt-d since all TRU production (by U-238 capture) has been eliminated. The fuel burnup also increases significantly at low CR because the same energy is being produced at a reduced fuel volume fraction. The most important penalty for the improved TRU destruction rate is a significant increase in the burnup reactivity loss rate (results in Table 15 employ a constant cycle length). This will require additional reactivity control measures (e.g., more control rods, or more frequent batch refueling) to be employed at low conversion ratio.

Table 15. Compact Core Reactor Performance Data for Different CR Reactors. [27]

Targeted Conversion Ratio	0.5	0.35	0.25	0.00
Pin Diameter (cm)	0.67	0.62	0.59	0.45
Fuel Volume Fraction (%)	0.30	0.24	0.22	0.12
BOC Heavy Metal Loading (kg)	7485	5775	4571	2521
BOC TRU Loading (kg)	2254	2249	2250	2521
Low Enrichment Zone TRU Content (%)	27	35	44	87
High Enrichment Zone TRU Content (%)	33	44	56	93
TRU Conversion Ratio	*0.50*	*0.36*	*0.25*	*0.00*
External Feed (kg/cycle)	1129	886	715	424
Net TRU Consumption Rate (kg/yr)	122	161	193	259
Net TRU Consumption Rate (g/MWt-day)	0.47	0.62	0.74	0.99
Average Discharge Burnup (MWd/kg)	112	143	177	298
Burnup Reactivity Loss (%Δk)	2.6	3.5	4.3	5.9

Based on these results, the AFCI studies of SFR burners focused on the CR=0.25 design. As shown in Table 15, this configuration achieves 75% of the maximum possible TRU destruction rate, but utilizes a fuel form that remains ~50% uranium. This decision was made to limit the extrapolation in fuel form technology (conventional fast reactor technology utilized up to 30% Pu/HM in metal and 40% Pu/HM in oxide); no performance or safety limits were encountered that would exclude the CR=0 system.

A system point design studies was created for the compact core with a conversion ratio of 0.25. In the point design studies, both the recycle and startup cores were evaluated. The startup core uses only the TRU derived from LWR spent fuel with no recycled fuel from the SFR. The reactor performance parameters of the startup and recycle cores were found to be similar. The major difference is in the TRU enrichment, which is slightly higher in the recycle core because of a degraded TRU composition; this enrichment change also decreases the conversion ratio from 0.25 to 0.19, improving the TRU destruction rate.

Burner Safety Evaluation

The kinetics parameters and whole-core reactivity coefficients for the compact burner cores are summarized in Table 16. [27] The Doppler coefficient decreases significantly with the reduction in conversion ratio. It is important to note, however, that the 0.25 CR design maintains

a considerable fraction of the Doppler seen with the 0.50 CR designs; this was achieved by increasing the sodium (coolant) volume fraction leading to a softer spectrum for the 0.25 CR design. The delayed neutron fraction (beta) decreases because of the reduced U-238 fission at low CR. The decrease in delayed neutron fraction will make the core more sensitive to small reactivity changes and additional consideration will have to be given to normal operating transients. The prompt neutron lifetime increases at low CR because of the softer spectrum.

Table 16. Reactivity Worth Coefficients for the Compact Core. [27]

Calculated Conversion Ratio	**0.50**	**0.36**	**0.25**	**0.00**
Beta	3.13E-03	2.92E-03	2.76E-03	2.44E-03
Prompt Neutron Lifetime	3.60E-07	4.01E-07	4.35E-07	5.18E-07
Sodium Density Worth (cents/K)	0.13	0.12	0.10	-0.01
Sodium Void Worth ($)	5.6	4.9	3.8	-1.6
Radial Expansion Worth (cents/K)	-0.31	-0.35	-0.39	-0.47
Axial Expansion Coefficient				
Fuel & Clad (cents/K)	-0.23	-0.28	-0.32	-0.40
Fuel (cents/K)	-0.26	-0.31	-0.34	-0.43
Doppler (cents/K)	-0.070	-0.061	-0.047	-0.001
Control Rod Driveline Expansion ($/cm)	-0.54	-0.69	-0.81	-1.11

The expansion coefficients increase at low conversion ratio; these increases are caused by both the decrease in the delayed neutron fraction and the greater importance of leakage for the lower conversion ratio reactor designs. The sodium density and sodium void reactivity coefficients also decrease at low CR. This trend is due to increased leakage caused by the reduced fuel volume fraction. The control rod driveline expansion coefficient increases as the conversion ratio decreases, because of the high reactivity loss rate, reduced beta, and softer spectrum.

Because of the changes in key safety parameters (e.g., Doppler coefficient and delayed neutron fraction as shown in Table 16), a series of reactor dynamics calculations were performed to assess the behavior of the CR=0.25 burner. The intent of the safety study was to investigate the ability of the plant to undergo an extremely low probability accident sequence without permanent damage. The detailed analysis focused on the key transient of an unprotected loss-of flow (ULOF) accident. [28]. The computational results show that using best-estimate

assumptions, the low conversion ratio design responds to the double fault ULOF accident with a high level of self-protection, with large margins of safety.

Evaluations for other double-fault accident initiators (transient-overpower and loss-of-heat-sink accidents) were also performed for the low conversion ratio designs. [29] These evaluations were less rigorous than the ULOF analysis reported; they utilized whole-core reactivity parameters, instead of the SAS4A spatial model employed for the ULOF analysis. Results from these cases indicated that the transient response is relatively insensitive to the variation in the conversion ratio. Due to the large control rod worths (large burnup reactivity swing) of the designs evaluated in the study, the unprotected transient-overpower accident showed the largest sensitivity to the conversion ratio. In conclusion, the favorable passive safety behavior of the modular size metal-fueled SFR was retained at low CR; adequate safety margins to fuel and cladding melting and cooling boiling were maintained. [29]

The results of this safety study would need to be re-evaluated for alternate fast reactor systems (oxide-fueled SFR, GFR, and LFR) to make conclusive statements regarding their safety. However, these results offer promise that with proper design a fast-spectrum burner can achieve comparable safety to the base Gen-IV designs, with the primary concern being the additional reactivity control requirements.

Burner Cost Estimate

As discussed above, early fast burner studies employed geometric spoiling to reduce the conversion ratio using conventional enrichment fuels. For this approach, significant cost penalties were postulated to accommodate the large geometric changes. A recent AFCI study was conducted to evaluate the relative cost of both compact and spoiled geometry low conversion ratio systems to a reference fast reactor design. [30]

A marginal cost approach was developed to evaluate the cost differential to the reference PRISM Mod B cost estimate. A large number of design modifications and operation differences were considered in producing the cost estimate:

- Capacity Factor – Difference in cycle length, number of assemblies, and assembly design lead to different capacity factors and average annual energy generation.

- Core Size – Reactor vessel must be modified to accommodate geometric changes. Direct impact on capital cost.
- Fission Gas Plenum – A simple model was developed to evaluate the cladding stress resulting from the accumulation of fission gas. Indirect impact on core size.
- O&M – Differences in outage frequency impact the maintenance expenses.
- Fuel – A time dependent model was developed that includes variable processing rates of LWR and ALMR SNF.
- Decommissioning – A simple model was developed to adjust the expenses based on the capital costs.

The cost estimate results are summarized in Table 17. The study found that the compact burner design has the lowest estimated cost of electricity (40 mills/kWhr in 1994 dollars) and the high leakage burner has the highest (48 mills/kWhr). The compact core design shows an approximately 1% lower capital cost ($/kWe) than PRISM because of the smaller reactor vessel size; whereas, the larger diameter of the high leakage core will increase the capital cost by approximately 5%.

The shorter cycle length of the low conversion ratio cores leads to reduced capacity factors and average annual energy generation. This increases the levelized cost of electricity. If expenses were identical, the high leakage core would have a 3% higher cost of electricity and the compact core would have a 4% higher cost of electricity. The levelized expenses (not normalized to energy generation) are approximately 6% lower for the compact core and 14% higher for the high leakage core. The majority of the differences in estimated expenses are the fuel cycle costs. Note that the fuel cycle cost includes the capital cost to build all new fuel processing and fabrication facilities. The larger initial TRU inventory of the high leakage core requires larger upfront expenses to manufacture the fuel for the first core loading with the opposite being true for the compact core. For a given design configuration, the conversion ratio dependence is understated because of the assumption that the cycle length and/or number of control rods are independent of conversion ratio. If taken into account, the cost of electricity will become more conversion ratio dependent.

Table 17. Levelized Cost of Electricity for SFR Burners. [30]

Parameter	PRISM Mod B	High Leakage		Compact Core	
Conversion Ratio	0.80	0.47	0.00	0.50	0.00
Capacity Factor	0.86	0.83	0.83	0.82	0.82
Annual Generation (TWh/yr)	14.07	13.60	13.60	13.47	13.47
Total Capital Cost ($/kWe)	1,554	1,626	1,626	1,536	1,541
Total Levelized Cost (mills/kWhr)	40.4	47.7	47.7	39.6	39.7
Capital	20.0	21.7	21.7	20.7	20.7
O&M	7.0	7.3	7.3	7.5	7.5
Fuel	12.4	17.7	17.7	10.4	10.4
Decommission	1.0	1.1	1.1	1.0	1.0
Relative to PRISM					
Inverse Energy Generation		103%	103%	104%	104%
Total Capital Cost		105%	105%	99%	99%
Total Levelized Cost		118%	118%	98%	98%
Total Levelized Expenses		114%	114%	94%	94%
Capital		105%	105%	99%	99%
O&M		101%	101%	103%	103%
Fuel		138%	138%	80%	80%
Decommission		105%	105%	99%	99%

In conclusion, results indicate that with proper design an SFR burner (CR=0.25) can obtain comparable economics to the base Gen-IV SFR design. This result would need to be confirmed for alternate fast reactor systems (oxide-fuel SFR, GFR, and LFR) to make conclusive statements regarding their cost. However, a similar approach to burner design (compact, with reduced fuel volume fraction) can be employed in other fast spectrum concepts as demonstrated in Sections 6.2 and 7.2.

6.0 GFR TRANSMUTATION SYSTEMS

6.1 GFR Gen-IV Systems

Under an International Nuclear Energy Research Initiative Project, the USDOE and the French CEA are jointly developing a Gas-Cooled Fast Reactor (GFR) system that meets the Gen-IV goals of sustainability/non-proliferation, improved economics, increased safety, and waste minimization. [31] Unit reactor powers in the range of 300 MWe (modularity) to 1,500 MWe (economy of scale) are being considered.

The sustainability and non-proliferation goals are met with a core that has an internal conversion ratio of ~1.0, in order to minimize the need for uranium resources and to reduce the environmental burden. This system would use a closed fuel cycle with an integrated on-site fuel treatment facility. In order to facilitate the long term deployment of the GFRs, the initial plutonium inventory in each core has been limited to 15 tons per GWe. The improved economic goal is achieved by the use of a coolant with a high outlet temperature design (850°C) that allows high thermal efficiency and supplies the process heat for hydrogen production. Both direct and indirect cycle designs have been pursued. [31] Passive safety approaches are being considered in the designs to respond to both anticipated transients without scram and the decay heat removal accidents, particularly at low pressure conditions when external power is unavailable. Natural convection has been selected as the preferred passive decay heat removal mechanism. This option has necessitated a double containment/guard vessel design.

An initial core parameter that had to be selected for the GFR was the core power density. Though the earlier gas-cooled fast reactor design effort (about 25 years ago) considered power densities in the range 230-300 MW/m^3, the proposed Gen-IV safety criteria have resulted in lower power density designs. This reduction was necessitated by the requirement to passively remove decay heat during a depressurization initiator combined with total loss of electric power. The optimum power density is, however, also dependent on the requirements to reduce system cost and ensure safe decay heat removal. The core power density for the designs being evaluated is between 70 and 100 MW/m^3. [32] The design parameters for a sample GFR design are summarized on Table 18.

Table 18. Design Parameters for a GFR Concept. [32]

Parameter	Value
Power Plant	600 MWt
Inlet / outlet coolant temperature	480 / 850°C(Helium, direct cycle)
Core Volume	5.8 m^3 (H_{core} / D_{core} ~ 2 m / 2 m)
Average Volume Power	103 MW/m^3
Vol. Fraction Cercer / Gas / SiC	35 / 55 / 10 core vol.%
Reference fuel compound	(UPu)C / SiC (70 / 30 fuel vol.%)
Fuel Enrichment	15-20%TRU/HM
Maximum fuel temperature BOL	~ 1200 °C (normal operation)
In core heavy nuclei / Pu inventory	16 / 2.5 tons
Fuel burnup, % fima	~ 5% (target of 15%)
Fuel residence time	3 x 441 EFPD
Doppler effect (180°C-1200°C)	-1136×10^{-5}
Total He void effect	356×10^{-5}
Thermal Efficiency	~45% (typical)
Fuel Cycle	TRU and U recycle

A variety of candidate fuels and materials for the GFR are being studied. Both ceramic-ceramic (Cercer) and Ceramic-metallic (Cermet) fuel forms in block, plate and pebble geometries have been considered. [32] Pin geometry is an alternative that has also been studied. For all these geometries, carbide and nitride fuels have been evaluated because of their relatively higher actinide density than oxide fuel. Matrix materials such as SiC, TiN, and ZrC, are leading candidates. Reactor physics evaluations have shown that the nitrogen needs to be enriched in N-15 prior to fuel/material fabrication and must be recovered after used to minimize cost. For the pin geometry, the cladding materials have included SiC-SiC composites, ODS steel, Nb-1%Zr and other refractory metals. In general, however, the refractory metals are significant neutron absorbers.

As noted in Section 2, the enrichment levels (20%) for the GFR are significantly higher than for a typical LWR; this is required to overcome enhanced U-238 capture in the fast spectrum. The burnup for the base GFR is significantly lower (~1/2) compared to the base SFR

and VHTR designs because the initial GFR designs are fuel performance limited (development of new fuel types is original focus). However, the high thermal efficiency of 45% can be obtained (similar to the VHTR) because of the high operating temperatures.

The GFR fuel cycle employs complete recycle of the TRU and uranium. This significantly extends fuel resources as only depleted uranium is required as a make-up feed. Furthermore, TRU are nearly eliminated from waste (~0.1% recycle loss each pass). The actinide-free waste has significantly reduced TRU mass, long-term heat, and radiotoxicity, compared to once-through fuel cycles. Thus, these features lead to exceptional performance with regard to the Generation-IV sustainability criteria related to resource utilization and waste management.

6.2 GFR Burner Design Studies

The general approach for fast spectrum burner design was discussed in Section 5.2. For the GFR, initial burner studies have only evaluated the transmutation performance capabilities. Safety and costs studies for the GFR burner have not yet been performed; however, the SFR burner studies (Section 5.2) indicated minor penalties in safety and cost.

A recent study has investigated the neutronic feasibility and performance of gas-cooled fast burner (GFR) cores designed for low conversion ratio (0.86 to 0.00). [33] Two leading GFR (self-generations) concepts were evaluated for use as low conversion ratio TRU burner designs. They are 1) a block-type fuel assembly design using helium coolant, (U,TRU)C dispersion fuel, and SiC matrix and 2) a pin-type fuel assembly design using helium coolant, (U,TRU)C solid solution fuel, and Nb-1Zr alloy structure. These designs not only represent two leading concepts, but exhibit very different physics behavior. The block-fuel design contains a large amount of SiC, which has low absorption cross sections and effectively moderates the neutrons. The pin-fuel design uses primarily niobium for the structural material, which has a large absorption cross section and produces a much harder neutron spectrum. For both design concepts, the power level is 600 MWt and the power density is 100 W/cc. The feed materials are TRU recycled from LWR spent fuel and depleted uranium.

Similar to the strategy used for the sodium-cooled fast reactor system, the fertile content of the fuel (used for the self-generation design) was reduced by decreasing the fuel loading. Since the block-type fuel assemblies use dispersion fuel, this allowed for varying the relative quantity of fuel (dispersed particles) and the matrix without any dimensional changes to the fuel assembly. Because the pin-type fuel assemblies use solid solution fuel, geometric changes to the fuel assembly was required to adjust the fuel volume fraction. This was achieved primarily by varying the pin diameter.

Table 19 contains a summary of the reactor performance data for the block-fuel GFR for conversion ratios ranging from 0.79 (reference design) to 0.00. The fuel/matrix fraction is 18%/82% for the zero conversion ratio startup cores. The fuel volume fraction is reduced by 75% from the reference to zero conversion ratio case, while the SiC volume fraction is increased by almost 90%. Lower fuel fractions would not achieve the specified cycle length, but might be applicable for fuel cycles with shorter fuel residence times. With TRU recycle, the relatively high reactivity (high Pu-239) content LWR TRU feed will be degraded at low conversion ratios. This will require a higher TRU enrichment to operate for the given cycle length. This higher TRU enrichment is the primary reason for the reduced conversion ratio for a fixed fuel volume fraction. As expected, the CR=0 burner achieves the maximum possible TRU consumption rate of 1 g/MWt-d since all TRU production (by U-238 capture) has been eliminated. As observed for the SFR, the most important penalty for the improved TRU destruction rate is a significant increase in the burnup reactivity loss rate.

Similar results have been obtained for the pin-fuel GFR design using Nb-1Zr as structural material. The high absorption cross section of Nb leads to a much lower conversion ratio at a given fuel volume fraction because the parasitic absorption of the Nb requires a higher TRU enrichment. This GFR design has a much harder neutron spectrum relative to the block fuel using SiC as the matrix and structure. These differences in neutron spectrum and neutron economy lead to the large differences in performance and safety parameters between the pin fuel and block fuel designs. These generally lead to more favorable results for the block fuel design.

Reactivity parameters were also calculated as part of the study. The results of these calculations show several trends with potential safety implications. Very low conversion ratios

(CRs) result in large burnup reactivity losses. The delayed neutron fraction and prompt neutron lifetime decrease with decrease in conversion ratio. The magnitude of the Doppler reactivity coefficient is reduced at very low conversion ratios and is near zero for the pin fuel. For the block fuel, the Doppler reactivity coefficient has a large negative value for CR=0, which may be a significant safety advantage for this type of fuel. As the conversion ratio decreases, the helium void worth remains roughly constant for the block fuel, but increases significantly for the pin fuel. These differences were attributed to differences in the neutron spectra of the block and pin designs. [33]

The recycle of spent fuel uranium was found to have little effect on the performance of the GFR core. The use of recycled uranium results in a slight reduction in the conversion ratio, with core dimensions fixed. From a waste management point of view the recycle of this uranium is desirable as it minimizes the amount of heavy metal to be sent to the repository or waste burial sites. The impact of incomplete rare earth elements removal on core performance was also evaluated. From a proliferation perspective, a high content of rare earth elements in the recycled fuel may be beneficial as it provides a radiation barrier to material diversion. The difference in the core physics performance for the GFR cases with and without carry-over of rare earth fission products was found to be small.

Table 19. Block Fuel Reactor Performance with Decreasing Conversion Ratio. [33]

TRU Feed	Startup				Recycle			
Volume Fractions	Ref.				Ref.			
Fuel	0.240	0.172	0.103	0.062	0.240	0.172	0.103	0.075
Structure (Matrix & Structure)	0.204	0.273	0.342	0.383	0.204	0.273	0.342	0.370
Fuel / Matrix (%)	70/30	50/50	30/70	18/82	70/30	50/50	30/70	22/78
TRU Conversion Ratio	**0.79**	**0.52**	**0.24**	**0.00**	**0.78**	**0.51**	**0.18**	**0.00**
TRU Charge Enrichment (%)	23	33	57	100	23	37	69	100
Burnup Reactivity Swing (%Δk)	1.5	3.6	6.1	8.6	1.4	3.0	5.0	6.4
Discharge Burnup (%)	10	14	23	39	10	14	24	33
Net TRU consumption rate (kg/yr)	41	99	167	228	42	103	182	228
Net TRU consumption rate (g/MWt-d)	0.19	0.45	0.76	1.04	0.19	0.47	0.83	1.04
BOC Delayed Neutron Fraction						.0031		.0025
BOC Doppler Temp. Coeff. (cents/K)						-0.17		-0.12
BOC Helium Void Worth ($)						1.2		1.2

7.0 LFR TRANSMUTATION SYSTEMS

7.1 LFR Gen-IV Systems

Lead-cooled fast reactor (LFR) designs with power ratings between 50 and 1,200 MWe have been considered. [19] In the low power range (50-150 MWe) are the nuclear battery designs for long-life cores with long intervals between refueling (15 to 20 years). Other options include the modular systems (300–400 MWe) and the large monolithic plants at 1,200 MWe. The LFRs are designed, in general, to employ a closed fuel cycle.

The LFR battery options have been purposefully designed to meet market opportunities for electricity production on small grids, and for developing countries that may not wish to deploy an indigenous fuel cycle infrastructure to support their nuclear energy systems. Its small size, reduced cost, and full support fuel cycle services can be attractive for these markets. [19] The use of long-lived cassette cores makes access to fuel quite difficult and hence enhances the safeguard regime. In most concepts considering export to developing countries, it is expected that the spent fuel will be transported for reprocessing at a fuel cycle center or returned to the source country.

Under the Generation-IV program, a low-power LFR called the SSTAR (small secure transportable autonomous reactor) has been developed. Key features of the system include: proliferation resistance, molten lead coolant, nitride fuel, autonomous operation benefits of a self-regulating reactor based inherent physical processes, fissile self-sufficiency, supercritical carbon-dioxide (S-CO_2) gas turbine Brayton cycle power converter, natural circulation primary coolant heat transport, and factory fabrication. Core powers ranging from 10 to 20 MWe are being considered, with a thermal efficiency of 40%. [34] The design parameters and core performance of a 20 MWe (45 MWt) SSTAR are presented in Table 20.

Proliferation resistance is enhanced by the use of a single closed reactor cassette that has no moveable parts and that restricts access to fuel during the core lifetime. Long-core cycle lengths of about 20 years are being considered. The discharged transuranics are recovered and used in subsequent cycles. The use of lead coolant and nitride fuel allows operation at higher core temperatures than traditional liquid metal coolant. [34] These materials also enhance

passive safety. Lead is chemically inert, has a high boiling point (1740°C) allowing for a single phase coolant in both operational transient and accident scenarios. The nitride fuel has a high melting temperature (>2600°C for UN) and high temperature for decomposition (>1400°C). Its high thermal conductivity and the use of lead bond between the fuel and cladding allow for low temperature difference between the fuel and coolant. Generally, however, the nitrogen has to be enriched in N-15 to eliminate parasitic reactions in N-14 and the waste disposal problem associated with C-14 production. The nitride fuel has been found to be compatible with both the ferritic-martensitic stainless steel cladding and lead coolant, and nitrogen is insoluble in lead.

Table 20. SSTAR Core Conditions and Performance [34]

Parameter	Value
Power level, MWt	45
Thermal Efficiency	43 %
Fuel Form	U/PuN
Fuel Enrichment, % TRU/HM	15-20
Inlet/Outlet Temperatures, ^{o}C	405/561
Average/peak power density, MW/m^3	69/119
Specific power, KW/Kg	10
Average/peak discharge burnup, GWd/t	72/120
Peak fast fluence, n/cm^2	4 x 10^{23}
BOC to EOC burnup swing, %$\Delta\rho$	0.36
Fuel Cycle	TRU recycle

As noted in Section 2, the enrichment levels (20%) for the LFR are significantly higher than for a typical LWR; this is required to overcome enhanced U-238 capture in the fast spectrum. The burnup for the base LFR lies between the SFR (~100 GWd/t) and GFR (50 GWd/t) base designs. The thermal efficiency of 43% is slightly better than the SFR (38%) because of the higher temperature, but less than the GFR and VHTR 45+%.

The LFR fuel cycle employs complete recycle of the TRU and uranium. This significantly extends fuel resources as only depleted uranium is required as a make-up feed.

Furthermore, TRU are nearly eliminated from waste (~0.1% recycle loss each pass). The actinide-free waste has significantly reduced TRU mass, long-term heat, and radiotoxicity, compared to once-through fuel cycles. Thus, these features lead to exceptional performance with regard to the Generation-IV sustainability criteria related to resource utilization and waste management.

A modular LFR design, named STAR-LM, with a higher power rating of 400 MWt (176 MWe) has also been studied by ANL under a USDOE NERI project. [35] The features of the design include: sustainability and closed fuel cycle benefits of a fast spectrum converter core using transuranic nitride fuel; the passive safety attractions of the lead coolant; autonomous operation; cost advantages of modular construction and factory fabrication; natural circulation heat transport at 100% power; and a significantly improved thermal efficiency (44%). Consistent with the high operating temperature of the supercritical carbon-dioxide Brayton cycle, the lead coolant is used. This reduces corrosion unprotected steel and the coolant radioactivity compared to lead-bismuth.

7.2 LFR Burner Design Studies

The general approach for fast spectrum burner design was discussed in Section 5.2. For both the SFR and GFR burner designs, large increases in the burnup reactivity loss rate were observed. This behavior runs counter to the base LFR design feature of a long-lived cartridge core, because the reactivity loss rate would be even greater for single-batch fuel management. Therefore, multi-batch fuel management with annual refueling was employed for the LFR burner study. Initial studies have only evaluated the transmutation performance capabilities. Safety and costs studies for the LFR burner have not yet been performed; however, the SFR burner studies (Section 5.2) indicated minor penalties in safety and cost.

A low-conversion ratio (CR), lead-cooled fast reactor (LFR) design has also been designed. [36] For this study, metallic fuel (U/TRU-Zr) similar to the SFR burner was employed; this assumption is not expected to affect significantly the conclusion on the transmutation rates. The reactor was sized for 840 MWt and used two enrichment zones to flatten the power

distribution. The results for three core designs having different conversion ratios (0.5, 0.35, and 0.25) are summarized in Table 21.

Table 21. Attributes and Performance of Low Conversion Ratio LFR Designs. [36]

Parameters	CR=0.5	CR=0.35	CR=0.25
TRU Conversion Ratio	0.50	0.35	0.25
Cold Pin Diameter, cm	0.784	0.724	0.695
Core Volume, m^3	10.8	10.8	10.8
Power Density (W/cc)	77	77	77
Burnup Reactivity Loss, %Δk	2.8	3.9	4.7
Net TRU consumption rate, kg/yr	125	167	195
Net TRU consumption rate, g/MWt-d	0.48	0.64	0.75

This preliminary burner study demonstrates that a LFR design with a low conversion ratio is feasible. As expected, the CR=0.25 burner achieves 75% of the maximum possible TRU consumption rate of 1 g/MWt-d. As observed for the SFR and GFR, the most important penalty for the improved TRU destruction rate is a significant increase in the burnup reactivity loss rate. While a detailed optimization of the design has not been performed, it was found that limitations on the coolant velocity would lead to a modest power density in the range of 100 W/cc, as compared to 200-400 W/cc for the SFR. The low absorption cross section of the lead allows for an open lattice that is required to reduce the coolant velocity for the lead-cooled systems. The low power density has size (cost) implications that must be addressed for both the base Gen-IV and burner LFR applications.

8.0 ROLE OF GEN-IV SYSTEMS IN AN ADVANCED FUEL CYCLE

The Generation-IV systems are being developed to provide a long-term, sustainable fuel supply for the expanded use of nuclear energy. As stated in the Gen-IV implementation strategy, two research priorities are being pursued: [37]

- *"Mid Term Priority: Develop a Next Generation Nuclear Plant to achieve economically competitive energy products, including electricity and hydrogen."* This system is needed to meet growing energy demand and maintain the share of nuclear energy in the United States.
- *"Long-Term Priority: Develop a fast reactor to achieve significant advances in proliferation resistance and sustainability."* In this phase, the waste management and resource utilization gains are emphasized. Thus, the base Gen-IV systems are intended as a future replacement of existing plants for electricity production, and advanced technology for expanded nuclear energy production.

For the electricity production mission, the base Gen-IV systems offer a variety of fuel cycle options. Similar to advanced LWRs, the VHTR and SCWR concepts continue a once-through fuel cycle based on LEU fuels. As described in Sections 3 and 4, the Generation-IV systems provide improved thermal efficiency (by higher temperature operation), and higher fuel burnup; these features provide some improvement in the waste generation and resource utilization of the fuel cycle; for example, the waste generation/energy of the VHTR is 45% lower than the current LWR once-through fuel cycle. For low uranium prices and cheap direct disposal space, current reactors demonstrate that the once-through fuel cycle is economical.

The base Gen-IV fast systems (SFR, GFR, and LFR) operate in a closed fuel cycle using recovered transuranics (TRU) and uranium. As described in Sections 5, 6, and 7, these systems also provide improved thermal efficiency and higher fuel burnup. The application of continuous recycle provides large improvements in the waste generation and resource utilization of the fuel cycle; for example, the TRU waste/energy is reduced by a factor of 100, and depleted uranium can be utilized to extend resources. These benefits come at the cost of fuel reprocessing and complications in fuel re-fabrication associated with handling recycled TRU materials. Thus, the

decision to transition to this fuel cycle will depend on the future availability/cost of waste disposal space and uranium resources.

Another aspect of the Generation-IV approach is to expand nuclear power to alternative energy needs such as process heat, hydrogen production, and/or water desalination. Similar to improved thermal efficiency, high temperature operation is expected to improve application for these alternate energy missions. The VHTR and GFR concepts can achieve very high temperatures (>850C) which are targeted for chemical cycles for hydrogen production. Although the other concepts have higher outlet temperatures than current LWRs, their temperature range of (500-600C) would require lower temperature hydrogen production technology. Thus, evolution of hydrogen production and alternate energy technologies must also be considered in tandem with general fuel cycle and economic considerations.

Advanced reactor technologies are also being considered in the AFCI for a waste management mission. Continuation of the current once-through fuel cycle would produce several hundred thousand metric tons of spent fuel this century; thus, nuclear waste disposal solutions will be required for sustained nuclear power. As an alternative to vastly expanded permanent disposal space, the AFCI is investigating recycle technologies that avoid direct disposal of the spent fuel; in particular, the TRUs are removed from the spent fuel (reducing the long-term heat, dose, and radiotoxicity) and recycled in advanced reactors for consumption. As described in Sections 3.2-7.2, the Gen-IV systems can be modified to burner designs for this purpose.

In burner mode, the Gen-IV systems could be deployed to manage the TRU production of a sustained fleet of once-through LWRs; this strategy allows a gradual transition into advanced fuel cycle and reactor technology. At discharge, commercial LWR spent fuel (50 GWd/MT) contains 5.3 weight% fission products compared to a transuranic (TRU) content of 1.4%. Thus, for a sustained ***equilibrium*** with the existing LWRs a complementary pure burner system must comprise 1.4/5.3 ~ 25% the thermal power capacity of the LWR enterprise that is being supported. When fertile fuel is utilized, additional TRUs are created requiring additional fissions for their destruction; this results in higher capacity for the transmutation sector. For the Gen-IV burner concepts, the impacts of conversion ratio and thermal efficiency on the required power fraction are noted in Table 22.

Table 22. Comparison of Generation-IV Burner Requirements.

System	Thermal		Fast					
	PWR	VHTR	SFR			GFR		LFR
Conversion Ratio		0.00	0.50	0.25	0.00	0.24	0.00	0.25
TRU Production Rate, g/MWtd	0.25	-1.01	-0.47	-0.74	-0.99	-0.76	-1.04	-0.75
Thermal Power Ratio, % of PWR		**25**	53	34	**25**	33	**24**	33
Thermal Efficiency, %	33	47.7	38-42	38-42	38-42	45	45	43
Electrical Power Ratio, % of PWR		36	61-68	39-43	29-32	45	33	43

As noted above, all the pure burners (CR=0) require a thermal power of 25% the LWR capacity to completely consume the LWR TRU production. At higher conversion ratio, more burners are needed with the power fraction increasing to 53% (0.53 MWt of burner needed for every 1 MWt of LWR power) at CR=0.5. Because the Gen-IV systems have higher thermal efficiency than conventional PWRs, the electrical power ratio is slightly higher than the thermal power ratio; with the burner fraction ranging from 29-36% for pure burners; this would imply that the burners are providing ~25% of the total nuclear electricity. Thus, a significant infrastructure of burners will be required to manage TRUs that would be generated by sustained operation of once-through LWRs.

Because of its high thermal efficiency and non-uranium fuel form the VHTR has the highest electrical power ratio of the pure burners in Table 22. However, it may be difficult to completely destroy the TRU in this concept. As noted in Section 3.2, recycle in thermal systems may be constrained by neutron balance, higher actinide build-up, or safety impact of high TRU content fuels. Current studies show that 60% burnup can be obtained in a single pass. A single-pass application would reduce the power fraction to 22%, but nearly half the TRUs still remain in the waste, and the repository benefits (loading, dose, etc.) are limited to roughly a factor of 2.

Conversely, the Gen-IV fast reactor (FR) systems are designed to operate on TRU-based fuels with continuous recycle. As described in Section 5.2, the key limitation to achieving low conversion ratio is the desire to utilize conventional fuel enrichments: for perspective, CR=0.5 can be obtained at 30% enrichment (demonstrated fuels), CR=0.25 at 50% enrichment, and

CR=0 at 100% TRU enrichment. As shown in Table 22, the FR power fraction varies from 53% at CR=0.5 to 34% at CR=0.25 to 25% as a pure burner. The performance of the SFR, GFR, and LFR burner designs (as described in Section 5.2, 6.2, and 7.2 respectively) have been compared in detail; and key results are given in Table 23.

Table 23. Comparison of Gen-IV Fast Burner Performance (CR=0.25 Designs).

System	SFR	GFR	LFR
Net TRU Destruction, g/MWt-day	0.74	0.76	0.75
System Power, MWt	840	600	840
Outlet Temperature, ^{o}C	510	850	560
Thermal Efficiency, %	38	45	43
Power Density, W/cc	300	103	77
TRU Inventory, kg	2250	3420	4078
Fuel Volume Fraction, %	22	10	12
TRU Enrichment, % TRU/HM	44 - 56	57	46 - 59
Fuel Burnup, GWd/t	177	221	180

Because the transmutation physics behavior is similar for the fast burner concepts (see Section 2), a similar design approach was employed to achieve low conversion ratio; by reducing the fuel volume fraction, fertile material was removed. For a given conversion ratio, the TRU destruction rate and compositions are very similar. However, variations in other fuel cycle performance parameters are observed because of design differences. In particular, the power density of the SFR is much higher; the GFR requires low power density for safe decay heat removal, while the LFR requires low power density for slow coolant flow rates. The compact SFR approach has some economic benefits; although higher thermal efficiency and design simplifications (e.g., removal of secondary loop) are being pursued for GFR and LFR. Another impact of the low power density is a higher TRU inventory (by roughly a factor of 2 per MWt). Low power density implies a reduced fractional consumption rate (% of inventory/year) for the burner and significantly more LWR TRU would be needed to startup the lower power density options. The burners have a similar discharge burnup despite these inventory differences.

It has also been shown that fast burner performance is dependent on the isotopic vector of the recovered TRUs. This composition can vary with burnup and spectrum of the once-through feed system. For the SFR CR=0.25 burner, results are compared for a standard burnup

(50 GWd/t) LWR, ultra-high burnup LWR (100 GWd/t), and Gen-IV VHTR in Table 24. The primary difference in the fast burner design is that a higher enrichment is required at higher burnup, where the TRU vector is denatured. The enrichment increases by 5% and the TRU loading increases by ~10% between the 50 and 100 GWd/t LWR transuranic feed cases. The VHTR has a different trend because the spent uranium still has 5.4% U-235 enrichment. This reduces the startup TRU inventory by roughly 10% compared to the 100 GWd/t case. Because of the large variations in cross section and neutron balance in a thermal spectrum systems (see Table 1 and 2), one would expect thermal burner designs to be much more sensitive to changes in the isotopic feed vector.

Table 24. SFR Burner Performance with Varying PWR and VHTR TRU Feeds.

Feed	*50 GWd/t **LWR** TRU*		*100 GWd/t **LWR** TRU*		*103 GWd/t **VHTR** TRU*	
TRU Production Rate, g/MWt-day	0.25		0.19		0.20	
TRU Production Rate, g/MWe-day	0.77		0.58		0.43	
Burner Mode	Startup	Recycle	Startup	Recycle	Startup	Recycle
Fissile/HM	*29.0%*	*24.7%*	*26.5%*	*22.5%*	*29.7%*	*24.5%*
U-235/U	*0.81%*	*0.33%*	*0.71%*	*0.43%*	*5.41%*	*1.24%*
TRU/HM	*49.2%*	*62.7%*	*54.0%*	*66.4%*	*47.6%*	*65.7%*
BOC TRU Inventory (kg)	2,149	2,790	2,380	2,966	2,091	2,934
Reprocessed TRU Feed Rate (kg/year/unit)	0	804	0	858	0	850
Makeup TRU Feed Rate (kg/year/unit)	828	249	910	257	801	253
Conversion Ratio	0.26	0.19	0.23	0.17	0.27	0.18
TRU Destruction Rate, g/MWt-day	0.74	0.81	0.78	0.84	0.74	0.83
Thermal Power Fraction, % LWR	34%	31%		23%		24%
Electrical Power Fraction, % LWR	39%	36%		26%		**19%**

Results are also shown in Table 24 for the recycle case, where the LWR/VHTR feed is only used as make-up. A slight degradation in the TRU fissile content is observed for the burner core; this results in an enrichment increase to ~66% TRU/HM for the high burnup feeds. The

increased enrichment in turn reduces the uranium content of the fuel, resulting in a slightly lower conversion ratio.

It is also important to note that the higher burnup once-through systems produce less TRU/energy. Thus, the fast burner power fraction can be decreased significantly. At 100 GWd/t, the thermal power fraction is reduced from 31% to 23%. When the higher thermal efficiency of the VHTR is accounted, this implies that the fast burner will only be providing 19% of the VHTR power or 16% of total power, as shown in Table 24.

Another refinement investigated in AFCI is to utilize a *double-tier transmutation system* to further reduce the burner fraction. The primary goal of the transmutation mission remains – to transmute the hazardous components of spent nuclear fuel. Thus, the majority of the transuranic (TRU) material needs to be fissioned, producing ~1 MW-day of energy for every gram. However, other systems may be more efficient in consuming this material, particularly early in the process when the fissile content is still high. Thus, a variety of alternative fuel cycle strategies have been proposed where the transmutation campaign is divided into several phases, denoted tiers. For example, the TRU material could first be irradiated in a thermal reactor (to burn the plutonium) with subsequent irradiation in a fast spectrum system (to burn the minor actinides).

In Ref. 24 a variety of double and single tier transmutation systems, including thermal and fast spectrum systems using conventional and non-uranium fuel forms and their associated fuel processing, were evaluated in a consistent manner. Given clean fuel processing (0.1% losses), all scenarios were capable of reducing the TRU and plutonium losses to the waste to less than 0.6% of the once-through fuel cycle waste. This reduction in TRU content was observed to significantly impact the waste toxicity, reducing it to the level of natural ore in less than 1,000 years. It was demonstrated that the toxicity reduction is very sensitive to the processing loss fraction. Furthermore, the initial limited recycle in existing LWRs was shown to have little impact on the toxicity, confirming the need for a final tier fast spectrum system to complete the transmutation campaign.

If the initial irradiation is conducted in a thermal spectrum system, the fissile materials will be preferentially consumed. Furthermore, as discussed in Section 2, significant quantities of

higher actinides (americium and curium) may be generated, particularly with deep burnup. Thus, the feed material for the Tier 2 fast spectrum system is less reactive but highly radioactive; and the final destruction of the radiotoxic minor actinides is more difficult. For example, in Ref. 24 the makeup feed for a second tier burner is based on a combination of minor actinides (Np, Am, and Cm) which bypass the first tier and 50% burnup first tier discharge. This mixed feed is quite degraded from the initial ALWR TRU; the fissile plutonium fraction has decreased from 56% to 32% with buildup of higher actinides.

The significant change in feed isotopics impacts the performance of the fast burner system. The reduced fissile content of the heavy metal requires a significant increase in the TRU inventory; and the resulting change in core performance parameters is shown in Table 25 for a pure burner system. The TRU inventory increases by 35% compared to the direct fast system case; and this results in a corresponding 25% in the average discharge burnup. Because the systems exhibit similar discharge fluence levels, this *reduced burnup can not be recouped* by increasing the fuel residence time. On the other hand, the decreased fissile content (more fertile material) results in lower burnup reactivity loss rates. A significant reduction in burnup swing from 4.1 to 2.5 %Δk is observed.

Table 25. SFR Pure Burner Performance in a Multi-Tier Fuel Cycle.

Parameter		LWR Feed	Tier 1 Feed
BOEC Heavy metal inventory (kg)		2708	**3657**
Fuel enrichment (%TRU in matrix)	Inner Zone	59	61
	Outer Zone	70	72
Burnup reactivity loss (%Δk)		4.14	**2.52**
Discharge burnup (MWd/kg)		273	**209**
Peak fast fluence (10^{23} n/cm^2)		3.73	3.70

Overall, the AFCI multi-tier studies indicate that the utilization of a first tier thermal spectrum system can significantly reduce the power capacity requirements for complementary fast burners. For example, if 50% of the TRU (i.e., the majority of the plutonium) can be destroyed in a first tier burner; the power fractions in Table 22 would be cut in half. However, even in the double tier scenarios, the burner capacity requirements remain significant (maybe as

low as 10% the total nuclear generating capacity), indicating that a significant transition to Gen-IV burner reactors will be required to achieve the transmutation goals.

9.0 CONCLUSIONS

The potential role of Generation IV nuclear systems in an advanced fuel cycle has been evaluated in this study. The Gen-IV systems considered are the thermal-spectrum VHTR and SCWR, and the fast-spectrum GFR, LFR, and SFR. The study evaluated the impact of each system on advanced fuel cycle goals, particularly related to waste management and resource utilization. The transmutation impact of each system was also assessed, along with variant designs for TRU burning.

It was shown that the fast-spectrum systems have a consistently higher fission/absorption ratio, indicating that actinides are preferentially fissioned and not transmuted into higher actinides. This implies that fast systems are more "efficient" in destroying actinides because fewer neutrons are lost to capture reactions before eventual fission. Additionally, the generation of higher actinides is suppressed. The fast-spectrum systems were also shown to give a more favorable neutron balance that allows the efficient transmutation of U-238, allowing extension of uranium resources. Conversely, the thermal system would require an additional source of neutrons (fissile feed) to drive the transmutation of the U-238. These core physics characteristics have direct correlations to the base core designs and fuel cycle strategies of the thermal- and fast-spectrum Gen-IV systems.

The base fuel cycle for the thermal reactor concepts (VHTR and SCWR) is a once-through fuel cycle using low-enriched uranium fuels. The higher burnup and thermal efficiency of the VHTR gives an advantage in terms of heavy-metal waste mass and volume, with lower decay heat and radiotoxicity of the spent fuel per energy produced, compared to a PWR. Fuel utilization might, however, be worse compared to the PWR, because of the higher fuel enrichment essential to meeting the VHTR system design requirements. The SCWR concept also featured improved thermal efficiency; however, benefits are reduced by the lower fuel discharge burnup.

The base fuel cycle for the fast reactor concepts (SFR, GFR, and LFR) is a closed fuel cycle using recycled TRU and depleted uranium fuels. Waste management gains from complete recycle are substantial, with the final disposition heat load determined by processing losses. The

base Generation-IV concepts allow consumption of U-238 significantly extending uranium resources (up to 100 times). These benefits come at the cost of fuel reprocessing and fuel re-fabrication with recycled TRU. Thus, the decision to transition to this closed fuel cycle will depend on future availability of waste disposal space and uranium resources.

For both thermal and fast concepts, recent design studies have pursued the development of dedicated burner designs. An evaluation of these studies indicated that a burnup of 50-60% might be possible in a VHTR burner design using non-uranium (transuranics) fuel. However, practical limits related to higher actinide buildup and safety impact may limit the extent of TRU burning in thermal reactors.

Fast burner designs have been developed for both conventional and high TRU content fuel forms. In general, the conversion ratio can be varied within a system by changing the uranium loading. Similar transmutation performance has been shown for the SFR, GFR, and LFR concepts; however, some design features (e.g., long-lived core) need to be avoided to attain significant material consumption in a burner configuration. Furthermore, more detailed AFCI studies of a low conversion ratio (0.25) SFR show that the burner design retains the favorable passive characteristics of conventional designs, and that any cost penalty will be small with proper design (e.g., compact configuration).

REFERENCES

1. *Report to Congress Advanced Fuel Cycle Initiative: Objectives, Approach, and Technology Summary*, Prepared by U.S. Department of Energy Office of Nuclear Energy, Science, and Technology, March 2005.

2. P. E. MacDonald et al., "NGNP Preliminary Point Design – Results of Initial Neutronics and Thermal-Hydraulic Assessment," INEEL/EXT-03-00870 Rev. 1, Idaho National Engineering and Environmental Laboratory, September 2003.

3. C. W. Forsberg, "Reactors with Molten Salts: Options and Missions," *Proc. Federic Joliot & Otto Hahn Summer School on Nuclear Reactors: Physics, Fuel, and Systems*, Cadarache, France, August 25-September 3, 2004 (2004).

4. M. Salvatores, I. Slessarev, and M. Uematsu, "A Global Physics Approach to Transmutation of Radioactive Nuclei," *Nuclear Science and Engineering*, **116**, 1 (1994).

5. M. Salvatores, R. Hill, I. Slessarev, G. Youinou, "The Physics of TRU Transmutation – A Systematic Approach to the Intercomparison of Systems," *Proc. PHYSOR2004 – The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*, Chicago, Illinois, April 25-29, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004).

6. R. N. Hill, "Comparison of Fast Reactor Transmutation Potential," **ANL-AFCI-115**, Argonne National Laboratory, September 30, 2003.

7. T. K. Kim, W. S. Yang, T. A. Taiwo, H. S. Khalil, "Equilibrium Cycle Analysis in Support of Fuel Specification for Next Generation Nuclear Power Plant," *Proc. of ICAPP '05*, Seoul, Korea, May 15-19, 2005.

8. R. C. Potter et al., "Gas Turbine-Modular Helium Reactor (GT-MHR) Conceptual Design Description Report," GA Report 910720, Revision 1, General Atomics, July 1996.

9. T. K. Kim, T. A. Taiwo, R. N. Hill, J. A. Stillman, "Spent Nuclear Fuel Characterization for a VHTR," *Global 2005* Meeting, Tsukuba, Japan, October 9-13, 2005. Pending publications.

10. J. A. Stillman, "Homogeneous Recycling Strategies in LWRs for Plutonium, Neptunium, and Americium Management," **ANL-AFCI-124**, Argonne National Laboratory, August 31, 2004.

11. J. J. Steyn, "Preliminary Assessment of Global Uranium Resources," Energy Resources International Report for US-DOE, ERI-2103-0501, January 2005.

12. W. S. Yang, T. K. Kim and T. A. Taiwo, "Comparison of VHTR and ALWR Fuel Enrichments," *Trans. Am. Nucl. Soc.* **93** (2005). Pending publication.

13. T. A. Taiwo, T. K. Kim, J. A. Stillman, R. N. Hill, M. Salvatores, P. J. Finck, "Assessment of a Heterogeneous PWR Assembly for Plutonium and Minor Actinides Recycle," *GLOBAL*

2003: Advanced Nuclear Energy and Fuel Cycle Systems, New Orleans, November 16-20, 2003.

14. G. Youinou, F. Varaine, A. Vasile, "Plutonium and Americium Multirecycling in the European Pressurized Reactor (EPR) Using Slightly Over-Moderated U-235 Enriched MOX Fuel Assemblies," *GLOBAL 2003: Advanced Nuclear Energy and Fuel Cycle Systems*, New Orleans, November 16-20, 2003.

15. T. K. Kim, J. A. Stillman, and T. A. Taiwo, "Assessment of TRU Stabilization in PWRs," **ANL-AAA-020**, Argonne National Laboratory, October 18, 2002.

16. E. D. COLLINS and J. P. RENIER, "Evaluation of Actinide Partitioning and Transmutation in Light Water Reactors (LWRs)," presented at the 2004 Atalante Conference, Marcoule, France, June 21-24, 2004.

17. F. Venneri, et al., "Deep Burn Transmutation: A Practical Approach to the Destruction of Nuclear Waste in the Context of Nuclear Power Sustainability," **FDO-E00-N-TRT-X-000132**, General Atomics (Dec. 19, 2001).

18. F. Venneri, Private Communications, March 2005.

19. A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-002-00, USDOE, December 2002.

20. Generation IV Nuclear Energy Systems Ten-Year Program Plan," Volume II: Appendices, DOE Office of Nuclear Energy, Science and Technology, March 2005.

21. Y. Oka, S. Koshizuka, T. Jevremovic, Y. Okano, "Systems Design of Direct-Cycle Supercritical-Water-Cooled Fast Reactors," *Nucl. Tech*., **109**, 1, (1995).

22. T. K. Kim, P. H. Wilson, P. Hu, R. Jain, "Feasibility and Configuration of a Mixed Spectrum Supercritical Water Reactor," *Proc. PHYSOR2004 – The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*, Chicago, Illinois, April 25-29, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004).

23. "Small Modular Fast Reactor Design Description," Joint Effort by Argonne National Laboratory (ANL), Commissariat a l'Energie Atomique (CEA), and Japan Nuclear Cycle Development Institute (JNC), May 2005.

24. R. N. Hill, T. A. Taiwo, J. A. Stillman, D. J. Graziano, D. R. Bennett, H. Trellue, M. Todosow, W. G. Halsey, A. Baxter, "Multiple Tier Fuel Cycle Studies for Waste Transmutation," *Proceedings of ICONE-10; Tenth International Conference on Nuclear Engineering*, Arlington, VA, U.S.A, April 2002.

25. R. N. Hill, D. C. Wade, J. R. Liaw, and E. K. Fujita, "Physics Studies of Weapons Plutonium Disposition in the Integral Fast Reactor Closed Fuel Cycle," Nuclear Science and Engineering, **121**, 17 (1995).

26. E. E. Morris and M. A. Smith, “Development of Low Conversion Ratio Fast Reactors for Transmutation,” **ANL-AAA-057**, Argonne National Laboratory, October 18, 2002.

27. M. A. Smith and R. N. Hill, “Development and Analysis of a Compact Low-Conversion Ratio Fast Burner Reactor,” **ANL-AFCI-084**, Argonne National Laboratory, June 30, 2003.

28. J. E. Cahalan, M. A. Smith, R. N. Hill, and F. E. Dunn, “Physics and Safety Studies of a Low Conversion Ratio Sodium Cooled Fast Reactor,” *Proc. PHYSOR2004 – The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*, Chicago, Illinois, April 25-29, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004).

29. M. A. Smith, E. E. Morris, and R. N. Hill, “Physics and Safety Studies of Low Conversion Ratio Sodium Cooled Fast Reactors,” *GLOBAL 2003: Advanced Nuclear Energy and Fuel Cycle Systems*, New Orleans, November 16-20, 2003.

30. E. A. Hoffman, R. N. Hill, and T. A. Taiwo, “Estimated Cost for Low Conversion Ratio Burners,” *Trans. Am. Nucl. Soc.*, **91**, (2004).

31. Project Final Technical Report I-NERI Project #2001-002-F, “Development of Gen IV Advanced Gas-Cooled Reactors with Hardened/Fast Neutron Spectrum,” Document submitted to USDOE by T. Y. C. Wei, April 29, 2005.

32. J. C. Garnier et al., “Feasibility Study of an Advanced GFR, Design Trends and Safety Options, Status of France & US Studies,” *GLOBAL 2003: Advanced Nuclear Energy and Fuel Cycle Systems*, New Orleans, November 16-20, 2003.

33. E. A. Hoffman, T. A. Taiwo, and R. N. Hill, “Low-Conversion Ratio Gas-Cooled Fast Reactors,” *Proc. PHYSOR2004 – The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*, Chicago, Illinois, April 25-29, on CD-ROM, American Nuclear Society, Lagrange Park, IL. (2004).

34. J. J. Sienicki, et al., *ANL Contribution to the Interim Status Report on Preconceptual LFR Design Studies and Evaluations*, January 28, 2005.

35. J. J. Sienicki, et al., “The STAR-LM Lead-Cooled Closed Fuel Cycle Fast Reactor Coupled to a Supercritical Carbon Dioxide Brayton Cycle Advanced Power Converter,” GLOBAL 2003.

36. E. A. Hoffman and J. Maddox, “Development of a Low-Conversion Ratio Lead-Cooled Fast Reactor,” Intra-Lab Memo, Argonne National Laboratory, December 2004.

37. "The U.S. Generation IV Implementation Strategy," U.S. Department of Energy Report, September 2003.