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HTGR Technology Family Assessment for a Range of Fuel Cycle Missions

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The lead author thanks Brent Dixon for his review of the draft report and apologizes for not giving him more time.

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[a] VHTR is a subset of HTGR. The NGNP is a specific VHTR/HGTR reactor project.

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SUMMARY

This report examines how the HTGR technology family can provide options for the once through, modified open cycle (MOC), or full recycle fuel cycle strategies. The HTGR can serve all the fuel cycle missions that an LWR can; both are thermal reactors. Additional analyses are warranted to determine if HTGR "full recycle" service could provide improved consumption of transuranic (TRU) material than LWRs (as expected), to analyze the unique proliferation resistance issues associated with the "pebble bed" approach, and to further test and analyze methods to separate TRISO-coated fuel particles from graphite and/or to separate used HTGR fuel meat from its TRISO coating. The feasibility of these two separation issues is not in doubt, but further R&D could clarify and reduce the cost and enable options not adequately explored at present. The analyses here and the now-demonstrated higher fuel burnup tests (after the illustrative designs studied here) should enable future MOC and full recycle HTGR concepts to more rapidly consume TRU, thereby offering waste management advantages. Interest in "limited separation" or "minimum fuel treatment" separation approaches motivates study of impurity-tolerant fuel fabrication.

Several issues are outside the scope of this report, including the following: thorium fuel cycles, gas-cooled fast reactors, the reliability of TRISO-coated particles (billions in a reactor), and how soon any new reactor or fuel type could be licensed and then deployed and therefore impact fuel cycle performance measures.

HTGRs in the context of LWRs

As the HTGR and LWR are both thermal neutron spectrum reactors, the report frequently compares the two as suggested by authors of the Option Study.[Wigeland2009] There are four major LWR-HTGR differences with fuel cycle implications – solid moderator, higher operating temperatures, higher fuel burnup associated with the TRISO fuel coating, and the "pebble bed" design approach (as opposed to the "prismatic" design approach, which is more directly comparable to LWRs).

The solid moderator has several effects. First, it means that there is no accident sequence involving voiding of the core's moderator. Thus, there is no void coefficient problem in HTGRs. The void coefficient issue in LWRs constrains TRU loading in recycled fuel (modified open cycle, full recycle).[Youinou2009] The lack of the void coefficient problem in HTGRs suggests MOC or full recycle HTGR could consume transuranics faster than a LWR. Second, carbon is a less effective moderator than hydrogen (or deuterium), leading to neutron energy spectral changes that slightly decrease uranium utilization relative to an LWR. This reactor physics disadvantage is compensated by the higher operating temperatures hence higher thermal efficiency. Third, the solid moderator in an HTGR (graphite) represents more of a waste management issue than the liquid moderator in an LWR (water). Heavy metal is 57% (BWR) to 70% (PWR) of the mass of LWR UOX assemblies, i.e., the discharged mass to be managed from used fuel is ~1.5 times that of the heavy metal. But, in the illustrative HTGR designs considered here, which are not optimized, the discharged mass to be managed is ~19 times that of the heavy metal. (Fast reactor designs are intermediate between LWRs and HTGRs in this regard, with discharged fuel assemblies being 3 to 12 times that of heavy metal.)

The higher fuel burnup (~10% in the illustrative designs for UO_2 versus ~19% in recent tests for UO_2 fuel) decreases the mass of used fuel that must be disposed (as HLW) in either the once through or modified open cycle. Higher burnup slightly decreases radiotoxicity in the once through fuel cycle or modified open cycle. A single recycle in either LWR or HTGR reduces radiotoxicity very little in uranium-TRU fuels (MOX) and 2-4x in all-TRU fuels (inert matrix fuels, a.k.a. deepburn). Full recycle of all transuranics in any reactor reduces long-term radiotoxicity by ~2 orders of magnitude, assuming process loss rates below 1%; For full recycle, the reduction of radiotoxicity in waste depends not so much on the reactor, but the number of times material is recycled and what the processing loss rate would be. The speed (number of recycles) required to attain such order-of-magnitude reduction is improved as the TRU loading increases, which is constrained by void coefficient in LWRs but not in HTGRs.

The pebble bed design approach is outside common safeguard experience. LWRs, fast reactors, and prismatic HTGRs have fuel assembly masses in the range of 120 to 660 kg; HTGR pebble bed have fuel pebbles about 0.2 kg. (Each pebble has ~50,000 TRISO-coated fuel particles.) Thus, LWRs, fast reactors, and prismatic HTGRs have hundreds of fuel assemblies, which are easily counted as individual items. Pebble bed HTGRs have hundreds of thousands of fuel pebbles, e.g., the PBMR-400 design had 450,000 pebbles. These are thought to be counted as bulk material, not individual items. Moreover, the packing density of pebbles varies, making exact balance of the number of pebbles difficult; fortunately, several thousand pebbles are required to make up one "significant quantity" (SQ) of weapon-usable material.

Definitions and Terminology

This report uses the three DOE fuel cycle strategies.

- Any option using only uranium fuel is considered "**once through**" (OTC); used fuel is assumed directly disposed.
- "**Modified open cycle**" (MOC) refers to any option using material recovered from used fuel in which used fuel made from recovered material is eventually disposed after one or more recycles.
- "**Full recycle**" refers to options in which used fuel is never directly disposed, one or more TRU elements are always recovered and recycled.

Recycled fuel contains the full range of TRU elements (all-TRU) unless otherwise stated, in which case the fuel is denoted as Pu, NpPu, or NpPuAm. The only fast reactor case in this report that is not all-TRU is a uranium-fed fast reactor considered one of the once through options.

Recycled fuel falls into three categories. Mixtures of uranium and one or more TRU elements is called mixed oxide (MOX) or analogs thereof. Fuel without uranium is called inert matrix fuel (IMF) denoting the fact that the matrix holding the fuel together does not fission. Fuel with a heterogeneous mixture of UOX and IMF fuels is designated UOX&IMF.

A class of once-through fast reactor concepts is currently being called "breed and burn". These follow the once-through fuel strategy, fresh fuel is enriched uranium; the intent is to discard used fuel. Except for a reactor with 100% enriched U235 fresh fuel, all reactors breed some Pu239 (from U238) or U233 (from Th232) and burn some of it *in situ*. Thus, all reactors are, strictly speaking, breed and burn.

The phrase "deepburn" is used two ways in the literature – high burnup of uranium fuel or high burnup of uranium-free fuel. In this report, it is only used in the latter way. For similarity with LWR uranium-free fuel, in figures and tables, the common designator of IMF is used, e.g., LWR-IMF and HTGR-IMF.

By definition, the HTGR family is limited to high temperature gas coolant reactors and therefore has potential applicability to both electricity and process heat markets. Therefore, apart from fuel cycle considerations, the HTGR may be deployed for process heat with LWRs maintaining their electricity market, or the HTGR could also displace the LWR for electricity.

This report follows the definition of the Generation IV program: HTGR have a thermal neutron energy spectrum. This report does not address the related Gas Cooled Fast Reactor (GCFR), except to note where a few GCFR options border with HTGR options. The only potential fuel cycle mission that cannot be met by the HTGR (but can be by the GCFR) is fuel breeding so that all of the original uranium ore (or thorium in the case of those cycles) is eventually fissioned.

Option Space

This report is limited to the U-Pu set of fuel cycles. Per discussions at the beginning of the fiscal year, it does not address thorium fuel cycles.

The report discusses which HTGR characteristics are inherent. Fundamental considerations lead to the conclusion that carbon is the obvious moderator regardless of fuel cycle mission. However, the HTGR technology family is not limited to TRISO-coated fuel when considering potential application to the range of fuel cycle strategies. And, the HTGR technology family is also not limited to the common separation technology assumption of taking aqueous separation technologies developed for LWR oxide fuels and adding a step of mechanically breaking TRISO fuel coatings and applying LWR techniques to the fuel meat. Figure S1 illustrates the range of fuel and fuel separation options.

- If no separation, the logical fuel choice is TRISO-coated particles
- If separation after reactor irradiation (by dis-assembling the fuel), there are a range of options from crushing to powder to burning. Most of these would be followed by dissolution of the fuel exposed by the initial treatment. A range of fuel options exist: more robust options (such as TRISO) may be harder to dis-assemble.
- If (partial) separation occurs during operation, the most robust fuel options don’t apply.

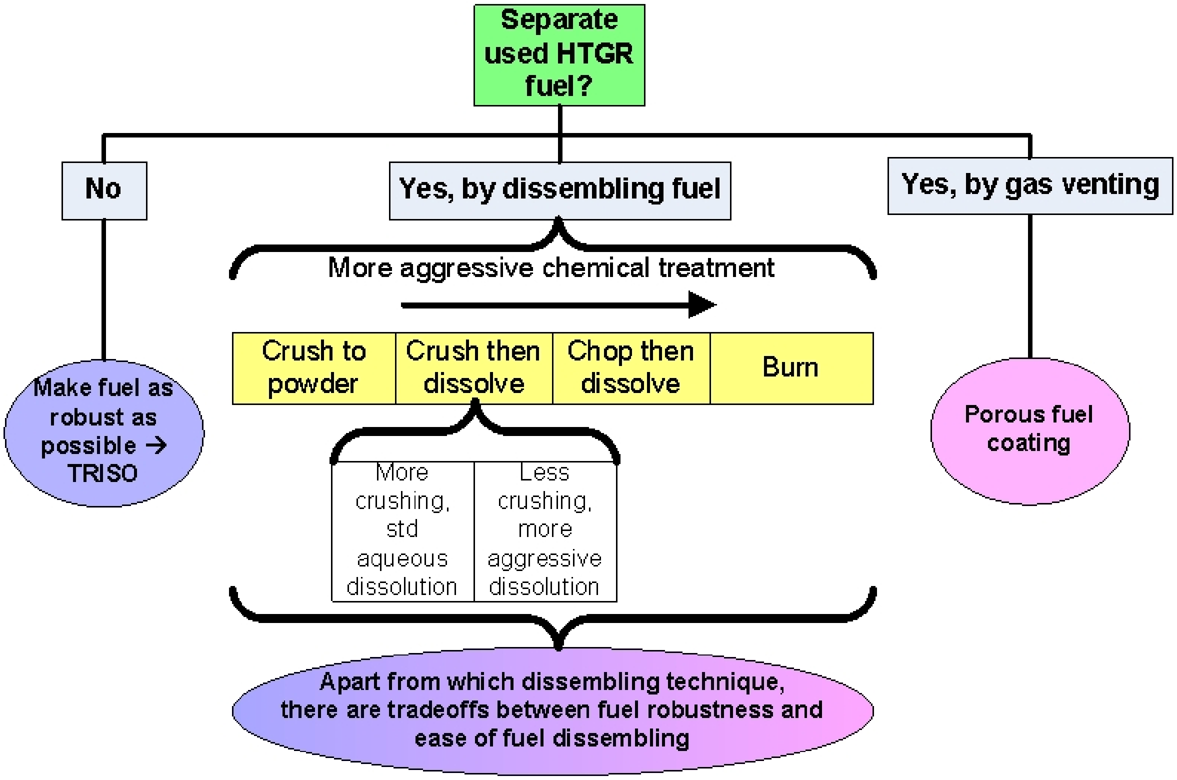

Figure S1. Range of HTGR Fuel and Separation Options

Additional Key Results and Considerations

The HTGR is considered for deployment independent of fuel cycle objectives; HTGR fuel cycle analyses are therefore appropriate regardless of perceived fuel cycle performance - to know how HTGR deployment would impact the overall fuel cycle prospects and options and what HTGR-relevant technologies should be researched. If substantial HTGR deployments occur, there is little reason at present to believe that that would worsen the prospects for advanced fuel cycles. Depending on the fuel cycle mission, those technologies could include new fuels with their fabrication techniques, new fuel separation approaches - either “full” separation or “minimum fuel treatment” - or new reactor designs.

To enhance reactor safety and reduce operational radiation fields around coolant equipment, the mainline HTGR approach is to develop a very robust fuel coating (TRISO). The HTGR potential for a very low radioactive coolant is unparalleled as helium (or CO_2) is non-radioactive and is less corrosive/erosive than water, sodium, lead, molten salts, etc.[Piet1982, Abdou1983, Smith1984] Therefore, the value of an extremely robust coating (nil “leakers”) is higher in an HTGR than for other reactor coolants and thus a long-standing objective of HTGR R&D is creation of very robust fuel coatings and assembly into structures that face the helium coolant.[b] However, for recycle strategies, a robust fuel coating poses a choice - develop technologies for undoing the robust coating or use a less robust coating. Options for undoing TRISO that have received little attention include jet-milling followed by mechanical re-forming, AIROX, or traditional chemical separation. Options for less robust coatings that have received little attention include ceramic tubes or porous coatings.

Options for separating particles from the pebbles or compacts include mechanical and passing electric currents, which can disintegrate the fuel form leaving matrix power and fuel particles.

The relatively poor coolant characteristics of gas, the single-phase nature of such coolants, and the desire for high thermal efficiency have led gas reactors to high temperatures, embedded in the names HTGR and VHTR. The high operating temperatures may offer a way to release fission products during operation; an idea being studied in GA’s EM^2 concept.[c] Although EM^2 is a once-through fast reactor concept, the approach has potential applicability to thermal reactors as well; indeed removal of neutron absorbing fission products would have more value in thermal reactors than fast reactors.

The FY2009 losses study report [Shropshire2009b] noted the linkage between impurities of waste in fuel products and impurities of fuel materials in waste products (e.g. TRU in waste). Higher tolerance of impurities in fuel may offer less TRU impurities in waste and conceivably even eliminate HLW altogether when defined by its characteristic components, studied in U.S. Japan, South Korea, and California.[Piet2009, Arie2009, Nam2009, Fowler2010]. (Of course, under current US law, all fission products would be HLW independent of their waste characteristics.) Meanwhile, to potentially reduce proliferation concerns and cost, DOE wants study of “minimal fuel treatment” options in addition to traditional full-separation techniques. This creates recycle fuels with high impurities. It is not clear how the potential of HTGRs to tolerate high-impurity recycle material differs from other technology families.

All commercial reactors in the US and 85% of those in the world are LWRs. About 10% of the reactors around the world are heavy water, 4% are gas cooled, and <0.5% are liquid metal cooled. Any non-LWR reactor’s contribution to advanced fuel cycle strategies is constrained by new technology deployment.

Figure S2 shows the radiotoxicity of waste for several LWR and HTGR cases. The three black lines are once through cases with similar burnups (100 to 110 MWth-day/kg-iHM); there is little difference among

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[b] Consider lead-bismuth coolant as an example; it is sufficiently radioactive and sufficiently corrosive that human access to the coolant system is precluded whether there are leakers or not. In water-cooled systems, significant radioactivity arises from both leakers and from corrosion products; until corrosion is adequately limited, preventing leakers has minor impact. In helium-cooled systems, leakers are the dominant pathway to coolant system radioactivity.

[c] The EM2 concept is a fast reactor in which fuel is vented during operation to allow volatile fission products to escape, thereby reducing gas pressure increase and neutron absorption by those fission products. Although fast reactors are not considered in this report; this fuel venting approach is.

them.[d] The uranium-free MOC cases for both HTGR and LWR are uranium-free IMF cases (called deepburn in HTGRs); they show significant radiotoxicity reduction between 100 and 100,000 years after reactor discharge primarily due to consumption of Pu239; there is an increase of radiotoxicity between 100,000 and 1,000,000 years due to production of some of the fertile isotopes. 1-recycle MOX fuels in LWR and HTGR (not shown) exhibit little or no radiotoxicity reduction versus once through. The body of the report shows that full recycle of LWR fuel does reduce long-term radiotoxicity by orders of magnitude versus once-through; this would be expected to occur with full recycle HTGR as well but there are no known analyses of that option.

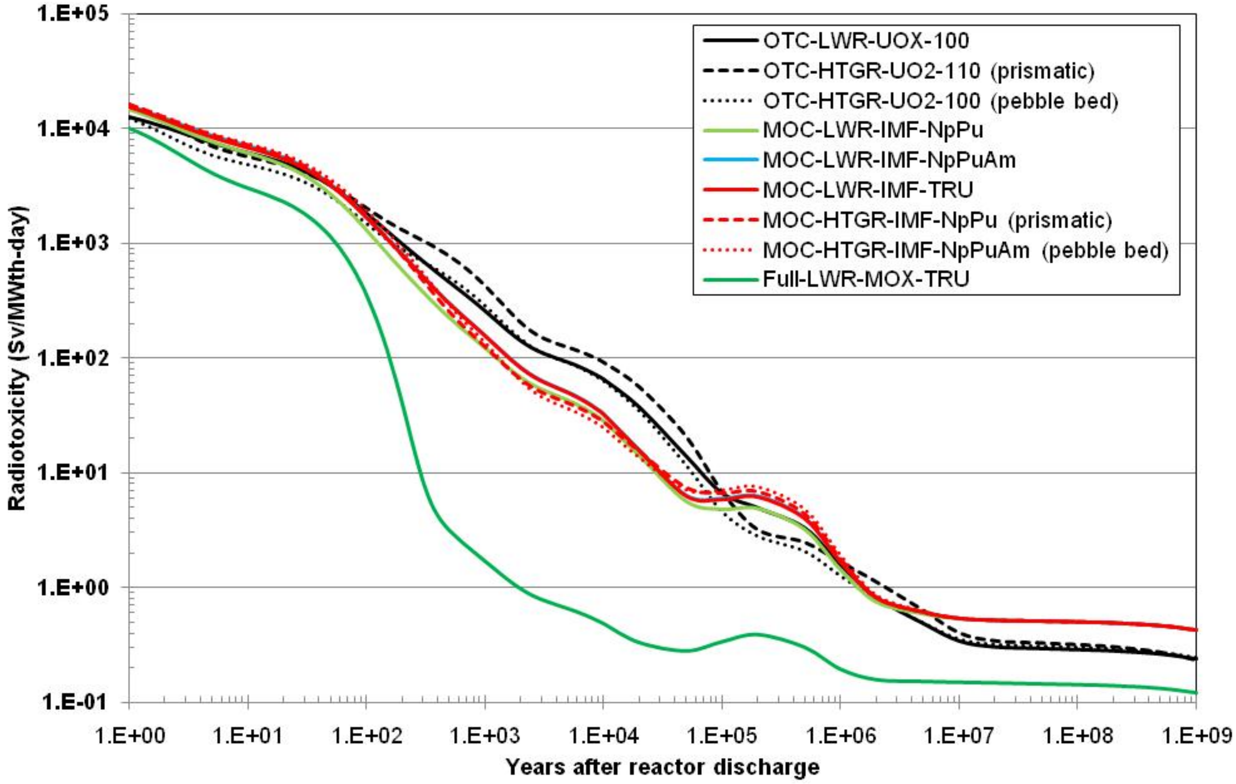

Figure S2. Radiotoxicity for LWR and HTGR options; solid lines are LWR; dashed lines are HTGR-prismatic; dotted lines are HTGR-pebble bed

Figure S3 shows uranium utilization for LWR and HTGR concepts. Uranium utilization in both LWR and HTGR is improved with recycling, but still remain below 1% uranium utilization. ~100% utilization can only be achieved by full recycle with fast breeder reactors. Recycling in thermal reactors improves uranium utilization, but only modestly. The two HTGR-UO2-prismatic cases have unusually low uranium utilization; as explained in chapter 2, this results from neutron spectral differences. The HTGR-UO2-pebble bed case does not appear to suffer as much, presumably because of the efficiencies associated with constant shuffling of fuel.

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[d] The prismatic UO2 case shows a slightly higher radiotoxicity than LWR-UOX; this may or may not be significant. The LWR analysis parameters are based on mature designs; the HTGR parameters are not.

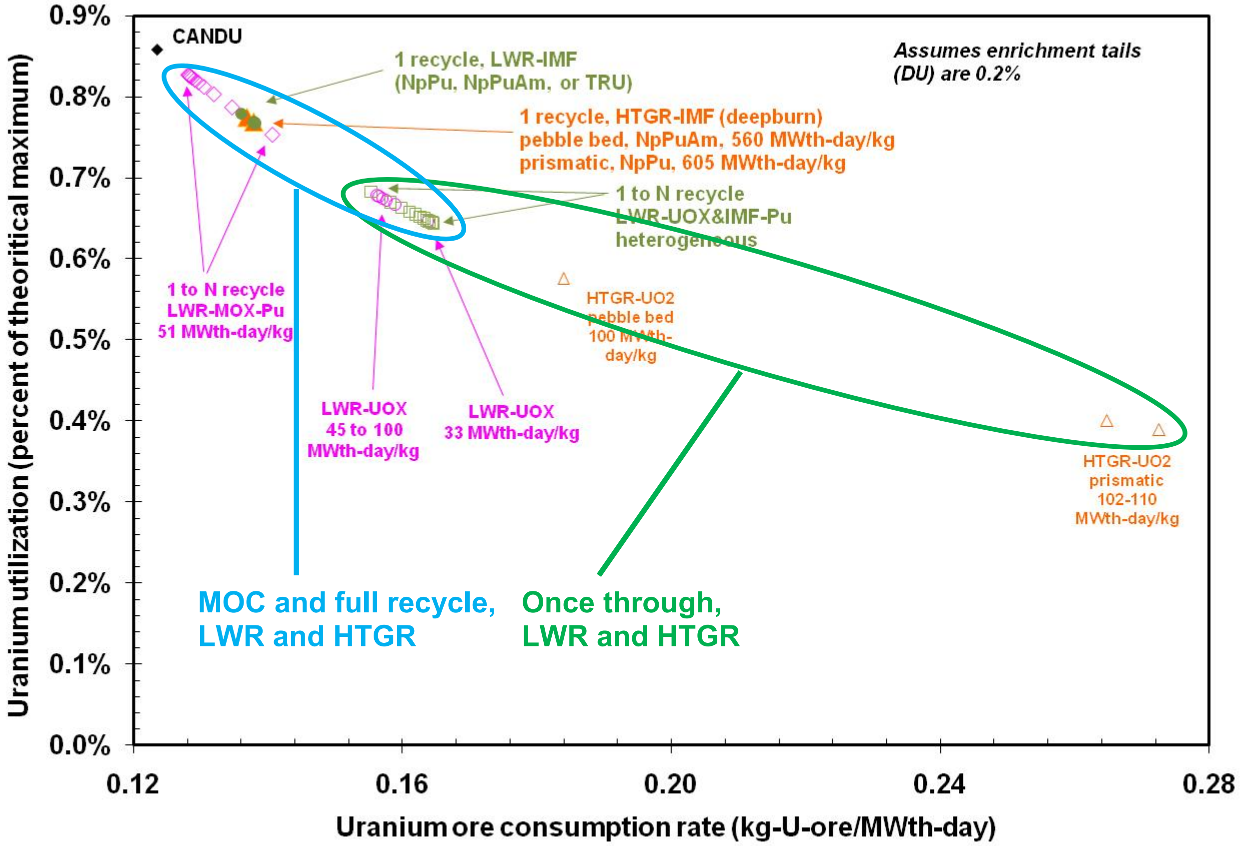

Figure S3. Uranium utilization and uranium ore consumption, without credit for thermal efficiency

There is a limitation in the preceding graph; it ignores the differing thermal efficiency among concepts. The GenIV roadmap says "The VHTR can also generate electricity with high efficiency, over 50% at 1000°C, compared with 47% at 850°C in the GTMHR or PBMR. Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes." This report is not limited to any specific HTGR concept or design and therefore it is appropriate to consider how the preceding figure changes with an HTGR thermal efficiency of 50% versus LWR efficiency of 33%.

Figure S4 is identical with figure S3 except that the normalization is changed from GW-thermal to GW-electric, i.e., the impact of thermal efficiency is included. In this graph, LWRs are assumed to have 33% thermal efficiency and HTGRs an upper-bound value of 50%. CANDUs are assumed to be 30%. Thus, figure S4 shows a best case HTGR/LWR comparison. The HTGR-UO2-prismatic case is still somewhat below LWR-UOX. The HTGR-UO2-pebble bed and the HTGR-IMF cases are significantly above LWR-UOX.

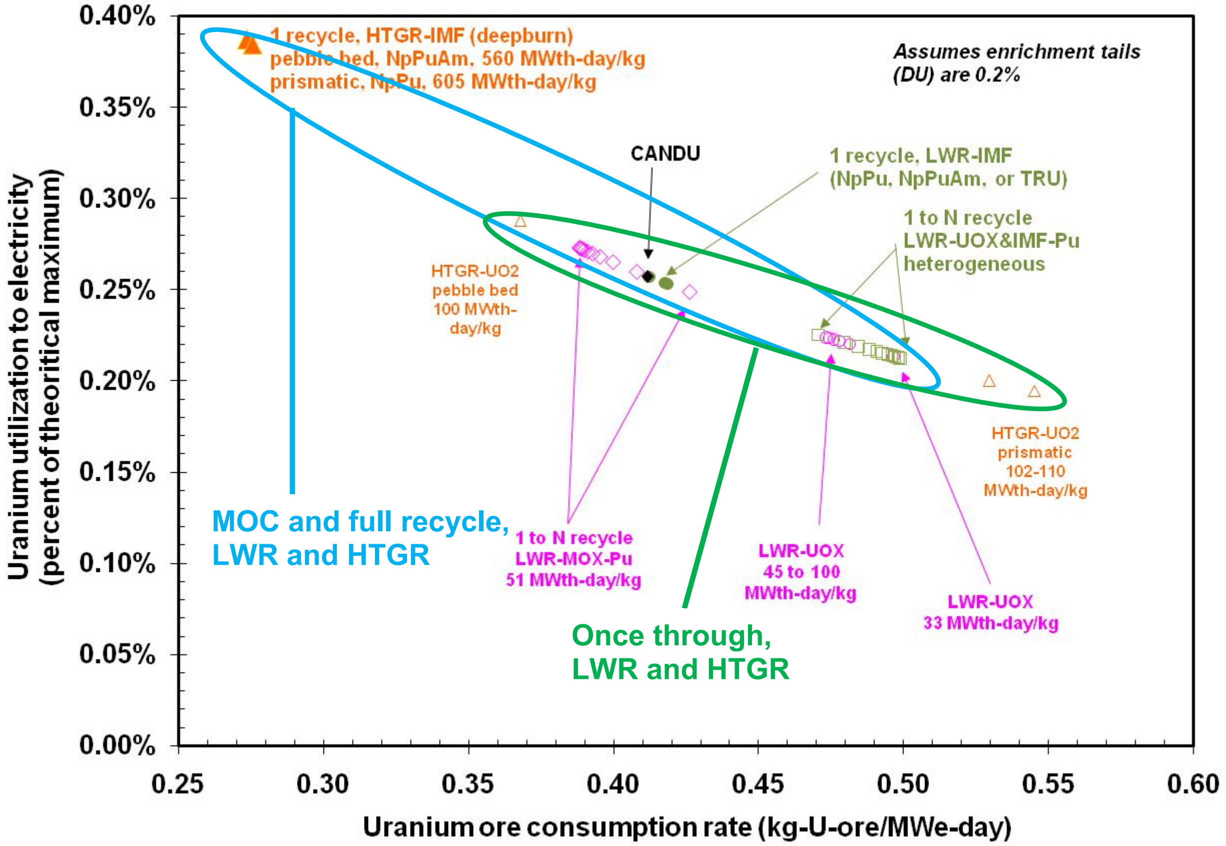

Figure S4. Uranium utilization vs. uranium ore for thermal reactor cases – with thermal efficiencies of 33% for LWR, 50% for HTGR, and 30% for CANDU.

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FIGURES

TABLES

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ACRONYMS

AAA	Advanced Accelerator Applications program
AFCI	Advanced Fuel Cycle Initiative program
AGR	Advanced Gas Reactor
AIROX	Atomics International Reduction Oxidation
aka	also known as
ATW	Accelerator Transmutation of Waste program
AVR	Arbeitsgemeinschaft Versuchsreaktor, the first pebble bed reactor (German), loosely translated as working group experimental reactor.
BISO	Bistructural-isotopic fuel coating, a predecessor coating to TRISO. The nuclear fuel kernel surrounded by a porous carbon layer (next to the fuel for gas expansion volume) and a dense C layer.
BWR	Boiling water reactor, a type of LWR
CANDU	CANadian Deuterium Uranium reactor
DB	Deep burn
DSARR	Dynamic Systems Analysis Report for Nuclear Fuel Cycle
DTC	Doppler temperature coefficient
DU	Depleted uranium
DUPIC	Direct Use of Pressurized water reactor spent fuel In CANDU
EM^2	Energy multiplier module, General Atomic concept for a gas-cooled breeder reactor
eV	electron-volt
FCR&D	Fuel Cycle R&D program, the successor to the Global Nuclear Energy Partnership (GNEP), which was the successor to the Advanced Fuel Cycle Initiative (AFCI), which was the successor of the Advanced Accelerator Applications (AAA) program, which followed the Accelerator Transmutation of Waste (ATW) program.
FFF	Fertile free fuel, same as IMF
FTC	Fuel temperature coefficient (=DTC)
FR	Fast reactor
GA	General Atomics
GNEP	Global Nuclear Energy Program
GT-MHR	Gas Turbine - Modular Helium Reactor, a variant of the HTGR
GW	Gigawatt
HEU	High enriched uranium (>20% U-235)
HLW	High-level waste
HM	Heavy metal

HTGR	High Temperature Gas Reactor, one of the Generation IV concepts
HTR	High Temperature Gas-Cooled Reactor (China)
HTTR	High Temperature Engineering Test Reactor (Japan)
HWR	Heavy water reactor
IAEA	International Atomic Energy Agency
iHM	Initial heavy metal
IMF	Inert Matrix Fuel (uranium-free)
IPyC	Inner pyrolytic carbon
LEU	Low enriched uranium (<20% U-235)
LFR	Lead-cooled fast reactor, one of the Generation-IV concepts
LLW	Low-level waste
LWR	Light water reactor (PWR, BWR)
MA	Minor actinide, any of the TRU elements other than Pu
Magnox	Magnesium non-oxidizing, a type of gas-cooled reactor and also the metal alloy cladding used in such reactors.
MCNP	Monte Carlo N-Particle, general purpose radiation transport code
MCNPX	Monte Carlo N-Particle eXtended
METROX	METal Recovery from Oxide Fuel
MOC	Modified open (fuel) cycle
MSR	Molten-salt cooled reactor, one of the Generation IV concepts
MT	Montana (MT is sometimes used for million tons and sometimes for metric tons; however, it is not a proper metric unit and is avoided here because of the ambiguity.)
MTC	Moderator temperature coefficient
MTS	Methyltrichlorosilane
MWe	Mega-watts of electrical energy
MWth	Mega-watts of thermal energy
NbC	Niobium carbide
NGNP	Next Generation Nuclear Plant
NpPu	Neptunium-plutonium based fuel
NpPuAm	Neptunium-plutonium-americium based fuel
OPyC	Outer pyrolytic carbon
OTC	once through (fuel) cycle
PMBR	Pebble Bed Modular Reactor (South Africa, canceled)
PBR	Pebble Bed Reactor, a variant of the HTGR
pcm	percent milli-rho

ppm	part per million
PWR	Pressurized water reactor, a type of LWR
PyC	Pyrolytic carbon
Q-value	Reaction energy, e.g., fission Q-value is the energy produced by the fission reaction
R&D	Research and development
RU	Recovered uranium
SCWR	Supercritical water-cooled reactor, one of the Generation IV concepts
SFR	Sodium-cooled fast reactor, one of the Generation-IV concepts
SiC	Silicon carbide
THTR	Thorium High Temperature Reactor (Germany)
THOREX	Thorium extraction, aqueous separation method for Th-based fuels
Tonne	1000 kg or 1,000,000 g = Mg
TRIGA	Training, Research, Isotopes, General Atomics - a uranium zirconium hydride fueled test reactor.
TRISO	Tristructural-isotropic, a type of micro fuel particle containing an oxide or oxycarbide kernel or fuel meat, surrounded by four layers of three isotropic materials - porous buffer layer of carbon, inner layer of dense pyrolytic carbon, ceramic layer of silicon carbide, and outer layer of dense pyrolytic carbon
TRL	Technology readiness level
TRU	Transuranic elements, neptunium (Np), plutonium (Pu), americium (Am), curium (Cm), berkelium (Bk), and californium (Cf).
TWR	Traveling wave reactor
UC_2	Uranium carbide
UCO	Uranium oxycarbide, a mixture of uranium dioxide (UO_2) and uranium carbide (UC) particles
UFF	Uranium free fuel, same as IMF.
UO_2 or UO2	Uranium dioxide, a ratio of 1 U atom to 2 O atoms
U_3O_8	Uranium oxide, a ratio of 3 U atoms to 8 O atoms
UOX	Uranium oxide, written without designating the specific uranium oxide because LWR UOX fuel has a U to O ratio that is close to but not exactly 1:2.
UOX&IMF	Heterogeneous IMF, namely a mix of UOX pins with IMF pins, in contrast with a homogeneous IMF core in which there are no UOX pins or fuel
UREX	Uranium extraction, aqueous separation method for U-based fuels
VHTR	Very High Temperature Reactor (the GenIV variant of HTGR, so-named to emphasize the very high temperature capability)
Zr	Zirconium, can also refer to zircaloy alloy (which is ~99% Zr)
ZrC	Zirconium carbide

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SYSTEM ANALYSIS CAMPAIGN HTGR TECHNOLOGY FAMILY ASSESSMENT

1. INTRODUCTION

This report fills in some of the knowledge gaps for the High Temperature Gas Reactor (HTGR) technology family with regard to the range of possible fuel cycle approaches. The objective is to examine this technology family in the sense of the Option Study.[Wigeland2009] This helps identify key issues, unknowns, and uncertainties. This report primarily addresses the breadth of possibilities and harvests past work. As the report analyzes past work with few original calculations, it cannot be a true independent validation of past claims and reports.

Several issues are outside the scope of this report, including the following: thorium fuel cycles, gas-cooled fast reactors, the reliability of TRISO-coated particles (billions in a reactor), and how soon any new reactor type could be licensed and then deployed and therefore impact fuel cycle performance measures.

The 2/2010 DOE budget request for FY2011 states that the Fuel Cycle R&D program's objective is to "research and develop nuclear fuel and waste management technologies that will enable a safe, secure, and economic fuel cycle." The fuel cycle R&D program is to "continue to develop advanced concepts for electrochemical processing and alternative waste forms, begin to develop innovative fuel systems that support advanced fuel cycles, and provide technical expertise to inform decision-making for storage, transportation, and disposal of used nuclear fuel and radioactive waste."

Accordingly, a comprehensive assessment of any fuel cycle option requires both

- A comprehensive set of technologies for the reactor, fuel, separation, and waste functions, and
- A comprehensive set of metrics encompassing waste management, proliferation resistance and physical security, natural resource utilization, economics, and safety.

Accordingly, for each possible fuel cycle approach, we have tried to identify candidate sets of reactor, fuel (TRISO or other), fuel separation, and waste management. Such a set is an "illustrative concept". There is no intent to down select or choose specific technologies, but to show how technologies can be combined to meet a particular fuel cycle approach with sufficient information to allow some assessment of that illustrative case.

1.1 HTGRs in the context of LWRs

As more readers are likely familiar with LWR technology than HTGR and more fuel cycle analyses have been done for LWR options, Table 1-1 provides translation between the two fields.

Table 1-1. Translation Table between LWR and HTGR

	LWR	HTGR
Coolant	Hot, pressurized light water	Hot, pressurized helium
Neutron energy	Thermal neutrons	
Neutron moderator	Light water - the coolant serves double duty	Carbon - generally in the form of graphite
Fuel form	Cylindrical pellets	Spherical kernels
Fuel + material surrounding the fuel	Fuel rods - a tube filled with fuel pellets	Coated particle = kernel + coating
		Fuel compacts = coated fuel particles mixed with carbon-based mastic - either rods or pebbles
Objects loaded and unloaded from a reactor	Fuel assemblies - a box of fuel rods	Fuel elements - contain fuel compacts
Material surrounding the fuel		
Roles of material surrounding the fuel	Provide heat transfer from fuel to coolant Keep fuel and fission products out of coolant Keep encapsulating material corrosion/erosion products out of coolant	
	Keep coolant out of fuel	
Typical material surrounding the fuel	Zircaloy cladding - an alloy of Zr made into tubes	TRISO coating - C/C/SiC/C
Alternative materials surrounding the fuel	Steel cladding - higher corrosion products and neutron absorption than Zr SiC cladding - lower corrosion products and neutron absorption than Zr	BISO coating - C/SiC Magnox cladding - Mg alloy
Fuel material as function of fuel cycle, with the following designations in this report		
Fuel used for once-through, i.e., enriched uranium	***LWR-UOX*** Uranium oxide (UOX)	***HTGR-UOX*** Uranium oxide (UOX) Uranium oxycarbide (UCO)
Fuel typically used for limited recycle, i.e., TRU-based fuel	***LWR-IMF-TRU*** Inert Matrix Fuel (IMF) Fertile Free Fuel (FFF) Uranium Free Fuel (UFF) *Pure TRU oxide has integrity and power/heat peaking issues; so, a fertile-free material is added to the fuel, e.g., TRU-Zr-oxide or TRU-Mg-Zr-oxide.*	***HTGR-IMF-TRU*** Deep burn (DB) *Extra carbon is added to the fuel to dilute the high power and heat peaking and provide volume for gas generation.*
Fuels typically used for sustained recycle, i.e., mixture of U and TRU in the fuel	***LWR-MOX-TRU*** Homogeneous mixture of uranium with one or more TRU elements	***HTGR-MOX-TRU*** Not yet studied, but logically would be analogous to LWR options.
	LWR-UOX/IMF-TRU Heterogeneous mixture of UOX pellets and IMF pellets	***HTGR-UOX/IMF-TRU*** Not yet studied, but logically would be analogous to LWR options.
To avoid excessive duplication, the once-through HTGR fuel option is designated HTGR-UOX; it is to be understood that UCO, a mixture of UO_2 and UC, is also an option. If the selection of TRU in the fuel is only a subset of the TRU elements (Np-Pu-Am-Cm-Bk-Cf) then the designations above (*X-TRU*) change to the elements in that subset, e.g., *X-Pu* or *X-NpPuAm*.		

In particular, in this report, analogs of LWR terminology are used to make the LWR-HTGR comparisons clearest where possible. For example, we rarely the phrase “deep burn”, which has often been used to mean TRU-based fuel in an HTGR, analogous to inert matrix fuel (IMF) in an LWR.[e] To make the analogies more apparent, these are called LWR-IMF and HTGR-IMF.

1.2 HTGR Fuel and Moderator Option Space

The HTGR technology family is defined to include all thermal neutron spectrum reactors using high temperature gas coolant with whatever fuels and separation technologies that may be appropriate for a given fuel cycle approach. Thus, the study is not limited to TRISO fuel, which was developed with the intent of high burnup and high radioactivity containment (during and after irradiation). The set of potential fuel cycle approaches considered is as follows:

1. Once through - one irradiation of uranium-based fuel in an HTGR, the used fuel is disposed.
2. Modified open cycle
    - Minimal fuel treatment - an irradiation of recycled uranium and transuranic (TRU) material (from LWR or HTGR) in an HTGR with minimal treatment of the LWR or HTGR used fuel.
    - Single recycle with HTGRs only - an irradiation of recycled TRU material (from LWR or HTGR) in an HTGR with full separation of the LWR or HTGR used fuel. This is analogous to the uranium-free Inert Matrix Fuel (IMF) option in LWRs but is typically called “deep burn” by advocates. The used fuel is disposed after a single pass in the HTGR.
3. Full recycle
    - Repeated recycle of the TRU and some of the uranium recovered (RU) from previous used fuel. This is analogous to repeated recycle of LWR-MOX or heterogeneous assemblies of LWR-IMF and LWR-UOX pins; either of which can be recycled indefinitely provided reactor safety limits such as void coefficient are met and fissile support is provided during subsequent recycles. Used fuel is never disposed. Only fission products are disposed. Depleted uranium (DU) and excess recovered uranium are put into indefinite storage.
    - Repeated recycle in mixture of HTGR and fast reactors: after TRU fuel is used once or twice in an HTGR, residual TRU material is used to fuel fast reactors. This is analogous to so-called 2-tier recycling in the former GNEP program. Used fuel is never disposed. Only fission products are disposed. Depleted uranium (DU) and excess recovered uranium are put into indefinite storage, but unlike sustained recycle in HTGRs only, the RU and DU could be used eventually if the fast reactors were in the breeder, rather than burner, mode.
    - Repeated recycle of the TRU and RU from previous used fuel. Only fission products are disposed. We find no evidence that this is possible in a thermal spectrum HTGR, but this fuel cycle is the intent of the related Gas Cooled Fast Reactor (GFR). The GFR is beyond the scope of this assessment.

Figures 1-1 and 1-2 (from Peterson 2006) illustrate the two main types of fuels based on TRISO-coated fuel particles. The prismatic reactor design incorporates fuel particles constructed of fuel particles (also called kernels or fuel meat) encapsulated inside four TRISO layers, combined into compacts, and placed inside graphite prism assemblies. Figure 1-2 shows the alternative design of placing many of the ~0.9-mm TRISO-coated fuel particles inside 60-mm diameter graphite pebbles.

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[e] “Deepburn” also sometimes refers to high burnup of enriched uranium fuel.

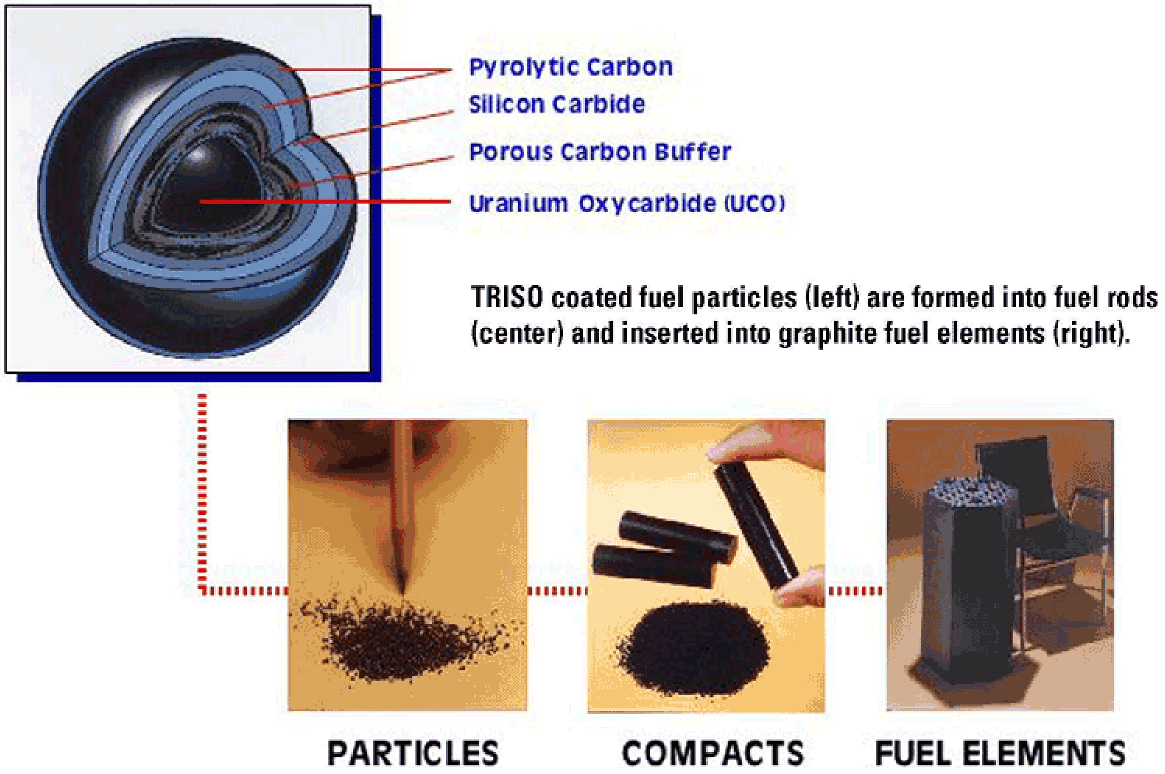

Figure 1-1. TRISO-coated particles assembled into compacts, placed inside prismatic graphite fuel elements; the fuel meat can be uranium oxycarbide or uranium oxide.[Peterson2006]

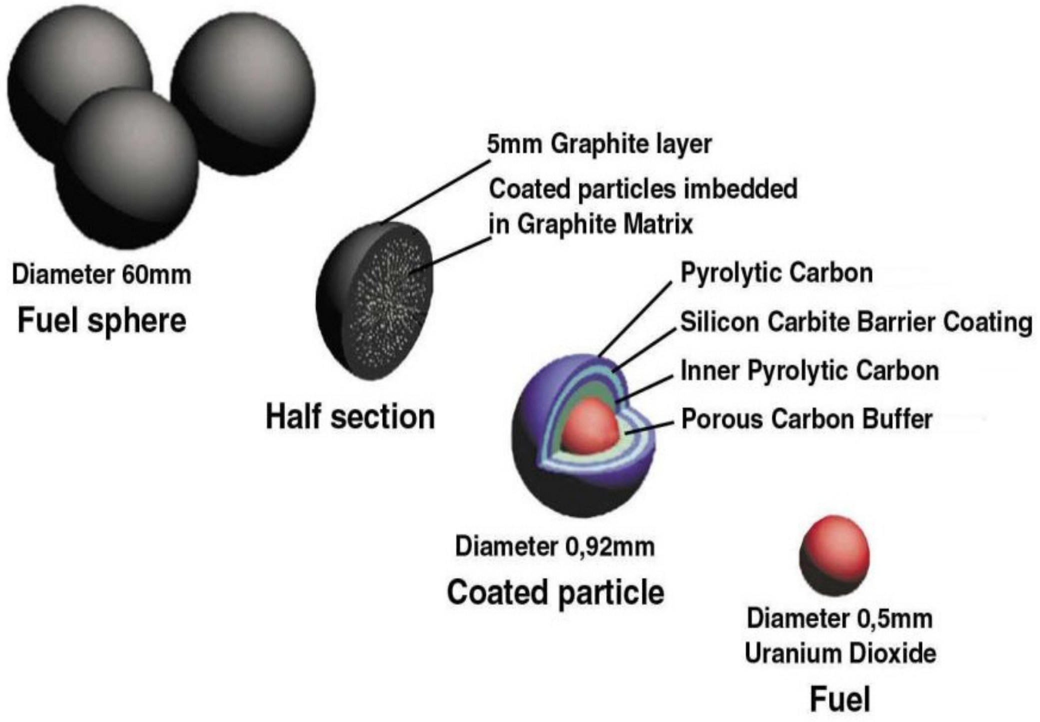

Figure 1-2. Pebbles constructed of TRISO-coated fuel particles, placed inside graphite pebbles; the fuel meat can be uranium oxycarbide or uranium oxide.[Peterson2006]

The TRISO coated U-based particles have achieved 19% burnup.[Grover2010a, Grover2010b] The TRU-based TRISO-particles also seem promising.[Petti2009] Even if HTGR-IMF-TRU ran into trouble because of the pure TRU fuel, one could always back off to HTGR-MOX-TRU. So, why would one possibly not want to use the TRISO-approach? Some reasons could include:

- If in-reactor fuel venting to possibly extend fuel life is desired,
- If an easier-to-separate fuel for fuel cycles involving multiple recycles is desired, or
- If the TRISO coating process is incapable of tolerating fuel kernels made with "minimal fuel treatment" such as mechanical re-forming or AIROX.

Of the characteristics that define what is in a reactor (coolant, moderator, fuel), perhaps the one more inherently linked to selection of fuel cycle strategy is the fuel itself. Thus, it may be unexpected that although coolant and moderator choices appear fixed in the HTGR family as fuel cycle strategies are varied, the selection of fuels might vary. Indeed, gas reactors have used easier-to-recycle Magnox [short for the "magnesium non-oxidizing" metal alloy coating on fuel particles.] fuel. While this United Kingdom (UK) design for HTGR fuel was temperature-limited due to the melting point of the Mg-Al-based alloy, the UK produced many of these reactors in the 1950s and 1960s and also recycled used Magnox fuel. Magnox fuel designs are now considered obsolete compared to the more recently developed BISO and TRISO fuels, but recycling Magnox fuel did not have to deal with the associated carbon and SiC in the particle (although graphite was still used as a moderator).

TRISO fuel is an appropriate illustrative option for those fuel cycle approaches in which the used fuel is disposed. However, when used HTGR fuel is intended to be recycled, it may be appropriate to consider other fuel options.

By definition, the HTGR family is limited to high temperature gas coolants and therefore has potential applicability to both electricity and process heat markets. Like the Generation IV program, we also define HTGR to be limited to a thermal neutron energy spectrum.

Fundamental considerations lead to the conclusion that carbon is the obvious moderator regardless of fuel cycle strategy, see Chapter 2 and Appendix 1.

1.3 Fuel and Fuel Separation Option Space

One cannot totally separate the description of "fuel options" from the description of "fuel separation options." The description of HTGR fuel cycle options starts with fuel separation technologies that are actually independent of the HTGR as they enable the reactor-separation-fabrication portion of the fuel cycle to the waste management portion of the fuel cycle. Furthermore, DOE has differentiated two classes of fuel separation, which we interpret as follows:

- Minimum fuel treatment - partial separation of fuel from impurities and/or partial separation of one fuel element from another.
    - Mechanical re-forming - taking used fuel pellets or powders and forming them into new pellets or spherical kernels without high temperature or oxidation/reduction conditions
    - AIROX - a class of dry options involving sintering (at high temperature) and/or oxidation/reduction cycles of the used fuel
    - Melt-refining - melting the fuel (typically metal and therefore would require reduction of oxide material to metal)

- Full fuel separation - high-purity separation of fuel from impurities; if appropriate there is separation of one fuel element from another.
    - Aqueous separation – dissolution of used fuel in an acid.
    - Electrochemical separation – dissolution of used fuel in a liquid salt

Chemistry of the process and/or species in used fuel has a role in determining the performance of each of these treatments. For example, mechanical re-forming is not a chemical process per se; but it is still the case that the inert gas elements would be expected to be released. Each of these treatments in Table 1-2 shows that if temperatures are increased (sintering) and/or oxidation/reduction cycles are used, more chemical elements are separated. Still, AIROX does not remove the majority of neutron-absorbing fission products (majority as weighted by neutron absorption), see [Christian1999]. It is not known how the relatively volatile actinides (notably Am) would behave. Indeed, the volatility of Am is sufficiently high that making and keeping Am during *fabrication* of a HTGR-IMF-TRU fuel is an issue (apparently a solved one). Melt refining does remove a majority of neutron-absorbing fission products and also removes 95% of the Am.[Hesson1963] The 95% removal of key elements in melt-refining was due to "formation of oxides at the wetted crucible surfaces, with oxygen contributed by the zirconia crucible. In general, the oxides remain in a reaction layer which is formed on the wetted crucible surfaces. The zirconia is left in an oxygen-deficient condition. Introduction of zirconium into the uranium melt was negligible."[Hesson1963]

So, we emphasize:

- No known candidate fuel separation/treatment method is completely without a chemical aspect. That is, the chemistry of the process and/or the species present impacts the performance.
- Technically, a useful definition of "minimum fuel treatment" has to specify which elements can be separated and by how much.
- All known candidate fuel separation/treatment methods - especially "minimum fuel treatment" - have significant unknowns with regard to how the method would (or would not) separate potentially important impurity elements.

Table 1-2. Estimated Percent of Fission Products Removed From Candidate Minimum Fuel Treatment Methods

Element		Percent removed from used fuel					Chemical class
		Melt refining [Hesson1963]	Dry recycle (AIROX) [Christian1999]			Mechanical re-forming	
			Both oxidative vaporization & sintering	Oxidative vaporization only	Sintering only		
35	Br	~100%					Halogen
36	Kr	~100%	~100%	~100%	~100%	~100%	Noble gas
37	Rb	~100%					Group 1
38	Sr	95%					Group 2
39	Y	95%					Transition metal that behaves like lanthanides in aqueous separation
40	Zr	Nil	Nil	Nil	Nil		Transition metal
41	Nb	Nil					
42	Mo	Nil	~100%	~100%	Nil		
43	Tc	Nil	~100%	~100%	Nil		
44	Ru	Nil	22%	22%	Nil		Noble metals
45	Rh	Nil	Nil	Nil	Nil		
46	Pd	Nil	Nil	Nil	Nil		
47	Ag	Nil	~100%	20%	~100%		
48	Cd	~100%	~100%	96%	~100%		Transition metal
49	In	Nil	Nil	~100%	~100%		Metalloids
50	Sn	Nil					
51	Sb	Nil					
52	Te	95%					Non-metal
53	I	~100%	~100%	~100%	~100%		Halogen
54	Xe	~100%	~100%	~100%	~100%	~100%	Noble gas
55	Cs	~100%	~100%	~100%	~100%		Group 1
56	Ba	95%					Group 2
57-71		95%	Nil	Nil	Nil		Lanthanides
90	Th	95%					Actinide
91	Pa						
92	U	Nil					
93	Np	Nil					
94	Pu	Nil					
95	Am	95%					
96	Cm						

Additional investigations of melt refining and AIROX variations are warranted to provide more representative and complete data. Many of the cells in this table are unpopulated because the references do not provide data for those elements. Some of the data in this table seems inconsistent, such as when up to 100% removal is shown for many elements for melt refining, and nil removal is shown for the same element for AIROX. Other inconsistencies relate to the Cs and Ru separations. R. Henry notes that the New Waste Calcining Facility, used at the INL for about a decade to calcine high level waste, operated at 550°C and highly oxidizing conditions, and was known to not significantly volatilize Cs (inconsistent with the ~100% values in this table), but volatilize Ru in larger proportions than consistent with the nil and 22% values in this table. Also, reported values of 100% separations are not likely, and may be due measurement limitations. For many elements, the difference between 100% and 99%, 99,9%, and 99.99% separations, and between 0% and “nil,” is significant, but these distinctions are not possible using the referenced data.

Regardless of reactor and fuel design, discharged used fuel that is coated or clad in metal or other material is rarely subjected to a chemical separation method without a mechanical pre-treatment to expose the fuel meat for the separations process. Table 1-3 lists options for mechanical preparation before chemical separation.

Table 1-3. Options for Mechanical Used Fuel Preparation before Chemical Separation

Preparation of used fuel	Possible to follow with mechanical re-forming?	Possible to follow with AIROX?	Possible to follow with aqueous separation?	Applicable to TRISO?	Applicable to other fuel coating?
Leave fuel coating intact (applicable to porous coatings that allow volatile gases to vent (to the coolant or to a separate recovery stream) during or after operation (somewhat analogous to molten salt reactors)	No	Yes, that is the intent of the EM^2 concept (but not for TRISO fuel)	Yes, discharged fuel could be separated via aqueous	No	Yes – such as BISO
Crush to powder, mechanically separate coating and fuel	Yes	Yes	Yes	Yes	Yes
Crush to crack the fuel's coating, dissolve fuel	No	No	Yes	Yes	Yes
Chop, then dissolve fuel (as done with LWR fuel)	No	No	Yes	No	Yes, e.g., Magnox
Burn	No	No	Yes	Yes	Yes
Prior to any of these, one may mechanically separate fuel particles from larger prismatic fuel elements or fuel pebbles.					

Note that we differentiate "crush to powder" versus "crush to crack" the fuel. The former is intended to completely expose the fuel particles for subsequent treatment, e.g., jet milling to produce particles ≤ 50 µm, suitable for then segregating carbon from fuel by density.[Hill2007] The latter is intended solely to allow dissolving agents (liquids or gases) to bypass the coating and attack the fuel kernels.[DelCul2002]

Although "crush to powder" (followed by density/gravity mechanical separation) has been proposed as a pre-treatment for subsequent aqueous separation, we see no obvious reason that the fuel particles couldn't alternatively be mechanically reformed into fuel (with their fission products) or given "minimum fuel treatment" such as AIROX. So, "crush to powder" could be a pre-treatment for mechanical re-forming, AIROX, or any of the traditional full separation techniques.

Indeed, the "crush to crack" approach does not enable subsequent segregation of carbon (and SiC) from fuel kernels except by chemical attack of the fuel. So, only full chemical separation techniques match this pre-treatment. There have been a range of options proposed after "crush to crack":[DelCul2002]

- Oxidize outer carbon, crush SiC shells, oxidize (burn) inner carbon, leach ash and fuel with nitric acid, proceed to standard nitric aqueous separation. This produces gaseous C-14 as CO or CO_2.[f]

- Oxidize outer carbon and SiC with a molten salt such as NaOH. This also produces gaseous C-14.

- Fluoridate outer carbon and SiC with fluorine. This produces gaseous C-14 as CF as well as SiF_4.

- Chlorinate outer carbon and SiC with chlorine. This produces gaseous C-14 as CCl_4 as well as $SiCl_4$.

- Oxidize SiC with molten $NaNO_3$ and Na_2CO_3. This produces NO and C-14 as CO_2.

- "Digest" SiC with phosphoric acid. This produces C-14 as CO and CO_2.

Yet another method to dissolve SiC is a mixture of nitric acid (HNO_3), hydrofluoric acid (HF) and fuming sulfuric acid (H_2SO_4).[Docekal1992]

The Rover fuel (coated UC_2) was recovered via burning followed by dissolution and aqueous separation.[Reilly1966]

The less crushing of the coated particles, the more that has to be done with chemical means (such as dissolving SiC) before one can start on the fuel kernels themselves. All pretreatment and separation options evolve gaseous species that contain at least some of the radioactive C14, H3, I129, and Kr85 (along with stable noble gas, halogen, and other gaseous species). The pretreatment and separations processes that oxidize the carbon-based coating and structure materials will result in contamination of the stable carbon with radioactive C14, which will increase the amount of total carbon that must be captured (by several orders of magnitude, compared to used fuel recycle processes that do not oxidize carbon coatings and structure materials) if the C14 must be captured and converted to a solid waste from to prevent air emissions of this radionuclide.

Dave Petti leads Very High Temperature Reactor (VHTR) technology development for the Next Generation Nuclear Program (NGNP). He notes that various high purity gases are used to make TRISO coatings, such as acetylene (HC_2H), propylene (C_3H_6), and methyl-trichlorosilane (CH_3SiCl_3). Experts have also wondered it is possible to integrate used TRISO fuel separation and new TRISO fuel fabrication by using the C and Si off gases as feedstock for new TRISO fuel fabrication.

The next option is "chop and dissolve." The Magnox fuel cladding is not suitable for next generation HTGR because of its low temperature capabilities. However, why not consider ceramic tubes such as ZrC or SiC, which is being considered for LWR fuel? The used fuel rods could be chopped and the fuel meat easily exposed, rather than crushing countless fuel particles.

Figure 1-1 assembles the options for separating used HTGR fuel by asking the questions, separate used HTGR fuel? And, if so, how? If used fuel is not to be separated, the fuel can and should be as robust as possible. But, if fuel is to be separated to any degree, other fuel options require consideration. In short, there are two ways to proceed:

- Use TRISO fuel coating (developed for the once-through fuel cycle) and develop ways to uncoat the fuel, preferably without generating troublesome volumes of radioactive gases.

- Start with mechanical/chemical separation in mind, then select fuel coating options that match.

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[f] Chapter 2 explains the sources of C-14 in HTGRs. Like H-3, C-14 draws attention because of its prevalence in the ecosystem. The major concern is nitrogen impurity in bulk graphite; the C-14 in the high purity C and SiC TRISO coatings should be less.

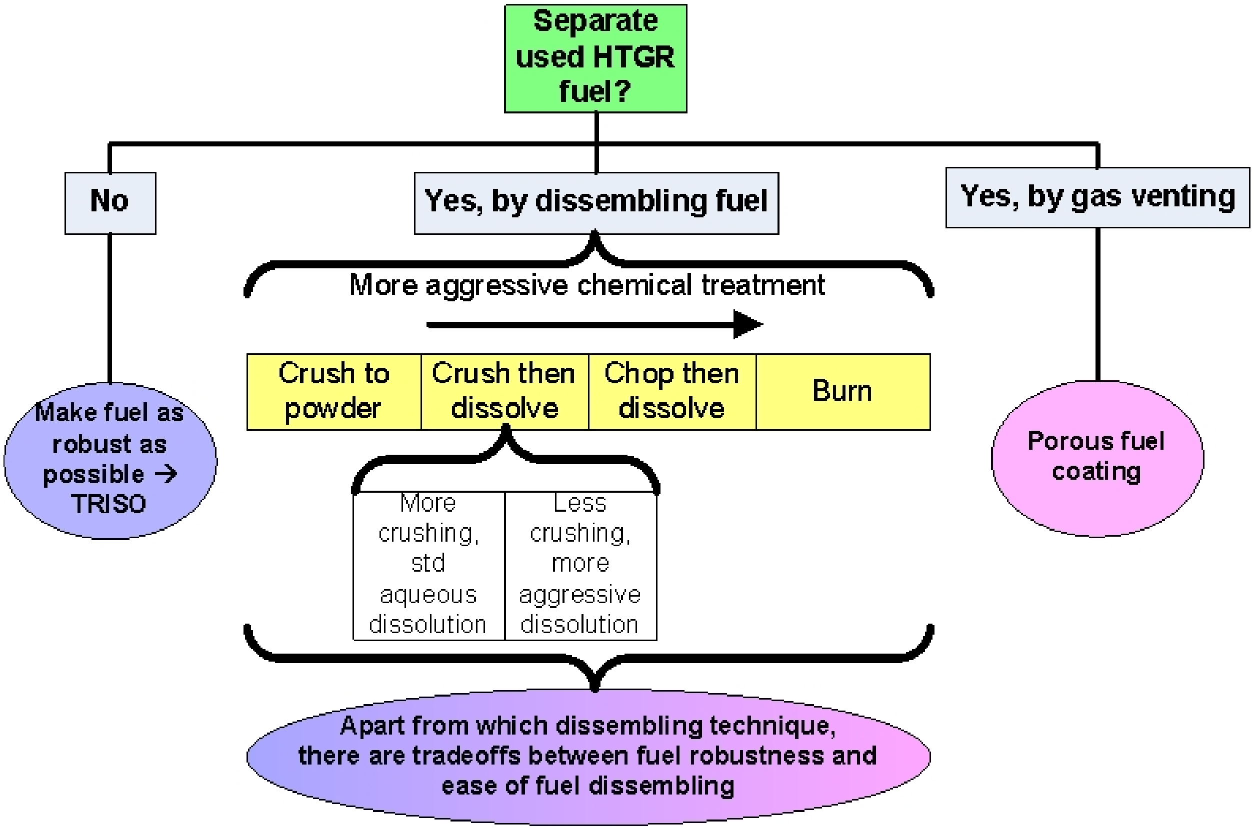

Figure 1-1. Separation Options for HTGR Technology Family.

Once the fuel meat is exposed, several chemical options can be considered, including

- Standard aqueous separation such as UREX+1
- Ammonium nitrate electrochemical [Tian2009a]

In summary regarding separation R&D issues:

- What is the practicality of "crush to powder"? If practical, are there advantages to "minimum fuel treatment" that warrant studies?
- What is the tolerance of TRISO to fuel kernels with high impurities, such as what would be present if used fuel were processed with AIROX or simple mechanical re-forming after "crush to powder"? If TRISO coating could not be used, then other approaches such as ceramic tubes might be applicable.
- Funded studies are ongoing that will determine the feasibility of using reactive gas chemistry to dissolve carbon and silicon carbide from TRISO form HTGR fuels. However, it may also be possible to recycle this material via physical treatment and separation methods and/or a combination of physical treatment and chemical methods that selectively separate/ dissolve the fuel matrix without reaction of the carbon or silicon carbide.
- Could porous fuel coating approaches such as GA has proposed for their EM^2 concept be applicable for HTGRs? Thus, another idea being considered for future HTGR fuel forms is to use other coating(s) on the fuel kernel with sufficient porosity to only allow the escape of helium,

tritium, and noble gases produced during irradiation. These gases would be separated from the helium coolant as they are produced, negating the requirement for robust pressure vessels (SiC) and the difficulty of assuring only perfect particles that are employed in the HTGR.

- Could high-temperature cladding materials, e.g. SiC, be considered as new gas-reactor tubular fuel cladding options?
- Can used TRISO off gases help feed new TRISO coating fabrication if separation and fabrication facilities were integrated?

1.4 Fuel Cycle Strategies for the HTGR Technology Family

Table 1-4 summarizes the potential fuel cycle strategies relative to the HTGR technology family and which reactor, fuel, separation, and waste technologies are considered in this report.

Table 1-4. Fuel Cycle Strategies Potentially Addressed by the HTGR Technology Family

Technical terminology	Reactor technologies	If TRISO coated fuel	If other fuel coating/cladding	Separation, used fuel treatment	Waste options
Once through	HTGR for electricity and/or process heat	HTGR-UOX-EU, developed for high burnup	Less robust fuel, vented to remove volatile fission products	Without venting, none needed. With venting, AIROX-like	Once through and single recycle options all include direct disposal of used HTGR fuel (high volume, high surface to volume ratio, low leach rate) and need to address recycle or disposal of graphite blocks (in the Prismatic reactor case) and pebble graphite (in the pebble bed case) used for core structure and as a moderator. Options with fuel venting and all single recycle options also include and disposal of separation products.
	Gas-cooled fast reactor (EM^2) in Adv once through study (outside scope of this study)	Not applicable, too much carbon	Less robust fuel, vented to remove volatile fission products		
Single recycle, minimal fuel treatment	HTGR for electricity and/or process heat	Impure-HTGR-IMF-TRU	Perhaps other options (SiC tubes??) if minimal fuel treatment doesn't produce adequately pure kernels	AIROX on used LWR UOX	
Single recycle of used LWR fuel		HTGR-IMF-TRU with TRU from used LWR fuel (aka deep burn)	If only a single recycle is planned and the feedstock is adequately pure, TRISO coating is the logical approach	Aqueous on used LWR UOX	
Single recycle of used HTGR fuel		HTGR-IMF-TRU with TRU from used HTGR fuel		• Same options as LWR-UOX or MOX with a pre-treatment step • Other options (e.g., AIROX, THOREX, METROX)	
Sustained recycle with thermal reactors only, only FP thrown away; excess DU and RU into storage		HTGR-MOX-TRU HTGR-UOX&IMF-TRU (mix of UOX fuel and IMF fuel)	Possibly use easier-to-separate fuels than TRISO		Dispose FP in waste forms same as for aqueous separations or in a proposed carbon waste form. Dispose or recycle graphite blocks.
Sustained recycle with mix of thermal and fast reactors, only FP thrown away; DU and RU into storage	HTGR followed by fast reactors (either burner or breeder mode)	HTGR-IMF-TRU HTGR-MOX-TRU			
Complete recycle of DU, RU, TRU	Gas-cooled fast reactor – (outside of the scope of this study)	Not applicable, too much carbon	Outside the scope of this study		

2. BACKGROUND AND PAST WORK

Over 99% of the commercial nuclear power plants in the world are light water cooled (85%), heavy water cooled (10%), or gas cooled (4%), see Table 2-1. All three have extensive commercial experience. All three are thermal reactors.

Table 2-1. Nuclear Energy Facilities in Operation as of June 1, 2009 [WNA2009]

Number	Name	Coolant	Moderator
264	Pressurized water reactors	Pressurized light water	Pressurized light water (same system as coolant)
92	Boiling water reactors	Boiling light water	Boiling light water (same system as coolant)
16	Light water graphite reactor	Pressurized light water	Carbon (graphite)
44	Heavy water reactors	Pressurized heavy water	Heavy water (separate system)
18	Gas cooled reactors	Pressurized CO_2	Carbon (graphite)
2	Sodium cooled reactors	Sodium liquid metal	None (fast reactors cannot have much moderating material)
436	Total		

2.1 Why Carbon as the Moderator?

Experts for other reactor types - most notably the Canadians for heavy water reactors - are re-examining their choice of moderator and coolant. This report does likewise. The Canadian developers of the heavy water reactor are pursuing a hybrid approach with light water coolant while maintaining heavy water as the moderator. This is an attempt to keep some of the advantages of both heavy and light water, such as higher neutron economy with heavy water but lower tritium content in the coolant (highest pressure and temperature, therefore, the more vulnerable place for tritium to reside). The penalty is that the neutron economy is not as high as in a pure heavy water system and therefore the uranium fuel must be enriched - lower than for LWR but unlike unenriched UOX for pure heavy water systems.

With the goal of examining the entire HTGR option space, and considering the Canadians' re-opening of moderator option space for heavy water reactors, it is appropriate to check the moderator option space for gas-cooled reactors. In gas-cooled reactors, the mass of coolant in the core is too low to provide significant neutron moderation. Therefore, thermal spectrum gas-cooled reactors require some neutron moderating material. Moderation is less effective the further one goes in the Periodic Table. So, our survey of candidate moderators in Table 2-2 extends only to element 10 (neon). Indeed, molten salt reactors based on fluorine (element 9) can be either thermal or fast spectrum, the amount of fluorine is not necessarily sufficient to force the neutron spectrum to be thermal, so there is no reason to look much farther.

Helium and neon can be dropped because, as gases, their density is too low to provide neutron moderation. Lithium and boron can be dropped because of high cross sections for wasting neutrons; we define "wasting neutrons" to be the sum of (n,γ), (n,α), and (n,p) reactions. Of course, in fusion reactors in which tritium production is desirable (instead of undesirable in fission reactors), lithium is an obvious candidate material for intercepting neutrons.

Table 2-2 shows that there are indeed alternatives to carbon as a moderator for HTGRs, but we see no obvious reasons to explore them further. When the carbon is only used for moderation, it is in the graphite form, a bulk form of carbon. Carbon is also used in thermal gas-cooled reactors as part of the fuel coating. Still, it is conceivable that as the HTGR technology family is considered against multiple fuel cycle strategies a non-carbon moderator approach could have value.

Table 2-2. Survey of Candidate Neutron Moderators

Element	Nuclear properties (lower is better)		Physical and chemical forms	Comment on candidate moderator
	Cross section for wasting neutrons (barns)	Number of neutron moderating collisions		
1. Hydrogen	0.028	15	Water, hydrides	Hydrogen is used in the hydride form, e.g., TRIGA reactors and some space reactor concepts. Hydrides are relatively unstable at higher temperatures and therefore relatively unsuitable for HTGRs.
1a. Deuterium	<0.000	20	Water, hydrides	
2. Helium	0.001	34	He gas	Helium has insufficient density to be an effective moderator.
3. Lithium	5.249	55	Li metal Li oxides Li fluoride	Lithium is relatively undesirable in a fission reactor because of its production of tritium, thereby wasting neutrons.
4. Beryllium	0.039	70	Be metal Be fluoride	Beryllium moderates, multiplies, and reflects neutrons. It is relatively expensive and chemically toxic, so its use is generally reserved in fission and fusion reactor concepts for neutron multiplication. The INL Advanced Test Reactor is partially beryllium moderated.
5. Boron	64.664	83	B_4C	Neutron waste is too high
6. Carbon	0.001	92	Graphite, C/SiC coatings, carbide/ oxycarbide fuels	Carbon is a decent moderator with very low neutron absorption. Stable chemical forms, with high purity products available.
7. Nitrogen	0.188	106	Nitride	Nitrogen is theoretically a candidate, but there are no obvious advantages versus carbon, at least for the traditional once-through HTGR fuel cycle. Nitrogen produces waste isotope C-14.
8. Oxygen	0.003	121	Oxides	Oxygen theoretically a candidate, but there are no obvious advantages versus carbon, at least for the traditional once-through HTGR fuel cycle.
9. Fluorine	0.007	143	Fluorides	Fluorine is relatively expensive and chemically active and relatively weak as a moderator.
10. Neon	0.009	151	Ne gas	Ne has insufficient density to be an effective moderator.

Cross section for "neutron waste" is the sum of neutron interactions that waste neutrons, (n,γ), (n,α), and (n,p). Illustrative thermal capture cross sections are for UOX in the ORIGEN 2.2 database with natural isotopic abundances except where noted.
More details on this table are found in Appendix 1.

Carbon comes with one particular waste management issue, C-14. Table 2-3 shows that carbon leads to C-14 in two ways, nitrogen impurity and activation of C13 (1.1% of natural carbon). At ~10 ppm-N, they are comparable.

Table 2-3. Sources of Carbon-14

Reaction producing C14	Relative mass of target element	Natural abundance of target isotope	Cross sections (barns)			Element x isotope x cross section
			σ(n,p)	σ(n,γ)	(n,α)	
N14(n,p)C14	1e-5 (N impurity)	99.63%	~ 1			~1e-5
C13(n,γ)C14	1	1.1%		~1e-3		~1e-5
O17(n,α)C14	1e-5 ?? (O impurity)	0.04%			0.24	~1e-9

Table suggested by Prof. M-L. Dunzik-Gougar.
Cross sections from 14th edition of nuclides and isotopes.

- (n,γ) has higher cross section at low (thermal) neutron energy.
- (n,p) and (n,α) have higher cross section at high (fast) neutron energy

Prof. Dunzik-Gougar and D. Petti independently note that N impurity is mostly an issue for graphite, rather than the extremely pure C/SiC used as fuel coatings.

2.2 Why TRISO as the Fuel Coating?

Table 2-4 summarizes past operating gas-cooled reactors to show just how prevalent TRISO fuel coating has been. The first fuel/cladding type was uranium metal with a Mg-based alloy cladding (Magnox); these remain the gas-cooled reactors most widely commercialized. These are being phased out due to relatively low temperature and performance and because the early approach of unenriched uranium fuel was replaced with enriched uranium fuel as uranium enrichment was widely commercialized.

Table 2-4. Past Operating Gas-Cooled Nuclear Energy Reactor Projects

Country	Project	Fuel Coating/ Cladding	Comments
UK	Magnox	Magnox clad uranium metal	Magnox (magnesium non-oxidizing) - metal alloy of magnesium, aluminum, etc. Late 1950s to present, unenriched U, no longer commercially available
UK	Dragon	BISO, TRISO	Dragon Reactor Experiment (materials test facility) 20 MWth, 1964-1975, enriched uranium
US	KIWI	Uncoated UOX UOX in graphite UOX in NbC UC_2/NbC	≤250 MWth nuclear rocket tests, 1959-1964
US	Phoebus	PyC coated UC_2 Some elements coated with NbC	≤3500 MWth nuclear rocket tests, 1965-1968
US	Pewee	NbC and/or ZrC coated PyC/UC_2	≤500 MWth nuclear reactor tests, 1968-1973
US	Peach Bottom	BISO	40 MWe, 1967-1974
US	Fort St. Vrain	TRISO	330 MWe, 1979-1989, uranium-thorium fuel
Germany	AVR	BISO, TRISO	Arbeitsgemeinschaft Versuchsreaktor (AVR) - working-group experimental reactor 10 MWth, 1966-1988, enriched-uranium
Germany	THTR-300	BISO	Thorium High Temperature Reactor 300 MWe/750 MWth, 1983-1989, enriched uranium and thorium
Japan	HTTR	TRISO	High Temperature Engineering Test Reactor 30 MW, 1988-present
China	HTR-10	TRISO	High Temperature Gas-Cooled Reactor 10 MWth, 2003-present
Assembled from Finseth1991, McEachern2007,1 Williams2009a, Williams2009b			

Although Magnox reactors are being phased out, they are included here to illustrate how the HTGR technology family can be adapted to a different fuel cycle - in this case no uranium enrichment and relatively easy separation of used fuel. Wikipedia describes Magnox:

> "Magnox reactors are pressurised, carbon dioxide cooled, graphite moderated reactors using natural uranium (i.e. unenriched) as fuel and magnox alloy as fuel cladding. Working pressure varies from 6.9 to 19.35 bar for the steel pressure vessels, and the two reinforced concrete designs operated at 24.8 and 27 bar. Magnox is also the name of an alloy—mainly of magnesium with small amounts of aluminum and other metals—used in cladding unenriched uranium metal fuel with a non-oxidising covering to contain fission products. Magnox is short for Magnesium non-oxidising. This material has the advantage of a low neutron capture cross-section, but has two major disadvantages: It limits the maximum temperature, and hence the thermal efficiency, of the plant. It reacts with water, preventing long-term storage of spent fuel under water."

> "Magnox fuel incorporated cooling fins to provide maximum heat transfer despite low operating temperatures, making it expensive to produce. While the use of uranium metal rather than oxide made recycling more straightforward and therefore cheaper, the need to recycle fuel a short time after

> removal from the reactor meant that the fission product hazard was severe. Expensive remote handling facilities were required to address this danger."

After Magnox, essentially all gas-cooled reactor efforts have been devoted to high temperature coolants and ceramic fuels. Gases are relatively poor heat transfer agents (compared to water/steam and liquid metals) and so high temperature is an important virtue. Inert gas coolants are relatively low in corrosion potential and erosion potential and therefore high temperature operation is relatively possible for gas-cooled reactors. So, high temperature and gas coolant go together.

The drive for high temperature tends to lead to ceramic fuels and ceramic fuel coatings. The fuels have been uranium oxide (UOX) more recently with the option of adding uranium carbide (UC), forming uranium oxycarbide (UCO).

The coatings have been combinations of carbon and SiC, maximizing moderation and stability at high temperature while minimizing neutron absorption. The first coating was BISO, a layer of porous carbon next to the fuel kernel, coated with SiC. To drive to very robust coatings to take advantage of potential high-temperature operation while keeping the coolant extremely clean (and thus the heat transfer systems low radiation zones), the BISO coating has given way to the more robust TRISO coating. Wikipedia describes TRISO as follows:

> "Tri-isotropic (TRISO) fuel is a type of micro fuel particle. It consists of a fuel kernel composed of UOX (sometimes UC or UCO) in the center, coated with four layers of three isotropic materials. The four layers are a porous buffer layer made of carbon, followed by a dense inner layer of pyrolytic carbon (PyC), followed by a ceramic layer of SiC to retain fission products at elevated temperatures and to give the TRISO particle more structural integrity, followed by a dense outer layer of PyC. TRISO fuel particles are designed not to crack due to the stresses from processes (such as differential thermal expansion or fission gas pressure) at temperatures beyond 1600°C, and therefore can contain the fuel in the worst of accident scenarios in a properly designed reactor. Two such reactor designs are the pebble bed reactor (PBR), in which thousands of TRISO fuel particles are dispersed into graphite pebbles, and the prismatic-block gas-cooled reactor (such as the GT-MHR), in which the TRISO fuel particles are fabricated into compacts and placed in a graphite block matrix. Both of these reactor designs are very high temperature reactors (VHTRs) [formally known as the high-temperature gas-cooled reactors (HTGRs)], one of the six classes of reactor designs in the Generation IV initiative."

2.2.1 TRISO for LWRs?

The potential advantages associated with TRISO-coated fuel has triggered consideration of TRISO-coated fuel in LWRs. This possibility is beyond the scope of this report.

2.3 How do LWR and HTGR Physics Differ?

First, basic differences among LWR, fast reactor, and HTGR physics explain one of the two major reasons there are fewer parameter variation studies for HTGR than for LWR. The other reason, of course, is that there have been fewer HTGR studies than LWRs.

LWR system analysis studies routinely use infinite lattice calculations based on a single unit cell. The mean free path of neutrons is relatively short and therefore core edge effects are ignored for studying the importance and effects of varying parameters.

Fast reactors have much longer mean free paths and therefore infinite lattice calculations are rarely used; instead full core calculations are done. More effect is required to study variation of parameters. But, since neutrons are not thermalized, many zones in the calculations can be homogenized.

HTGRs have lower mean free path than LWRs for neutrons because the interaction cross sections are lower. The energy loss is lower per collision and thus the number of collisions between neutron birth and thermalization is greater. Infinite lattice calculations are less usable than for LWRs. Zone homogenization is more restrictive than for fast reactor calculations. So, more work is required to study variation of parameters.

For the above reasons, we have less understanding of how selection of HTGR parameters impact physics-related results than for either LWR or FR.

2.3.1 Once through cycle - uranium fuels

Figure 2-1 provides representative neutron energy spectra for the PWR and five of the six Generation-IV concepts. The three thermal spectrum reactor concepts - PWR=Pressurized water reactor, VHTR= Very High Temperature Reactor, SCWR = Supercritical Water Reactor - are markedly different from the three fast reactor concepts - SFR = Sodium Fast Reactor, GFR = Gas Fast Reactor, LFR = Lead Fast Reactor. Below 10 eV, the VHTR has higher neutron energy; above 100 eV, the VHTR has lower neutron energy.

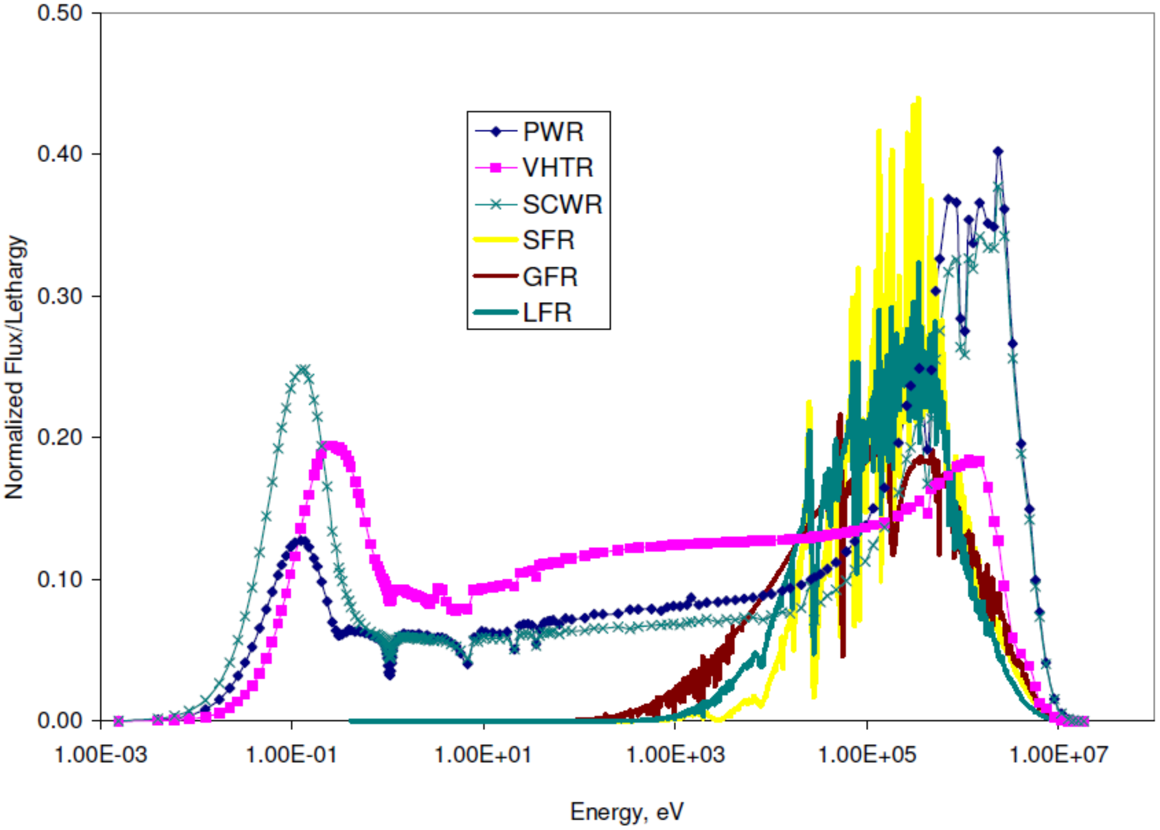

Figure 2-1. Comparison of Neutron Energy Spectra of Generation-IV Reactors [Taiwo2005]

Figure 2-2 shows the neutron energy spectrum for an HTGR-IMF concept.[Petti2009] It shows the same general behavior as the HTGR-UOX curve in figure 2-1. The thermal peak is shifted to slightly lower neutron energies (<0.1 eV versus > 0.1 eV). Presumably this would lead to somewhat higher capture by impurities and structural materials.

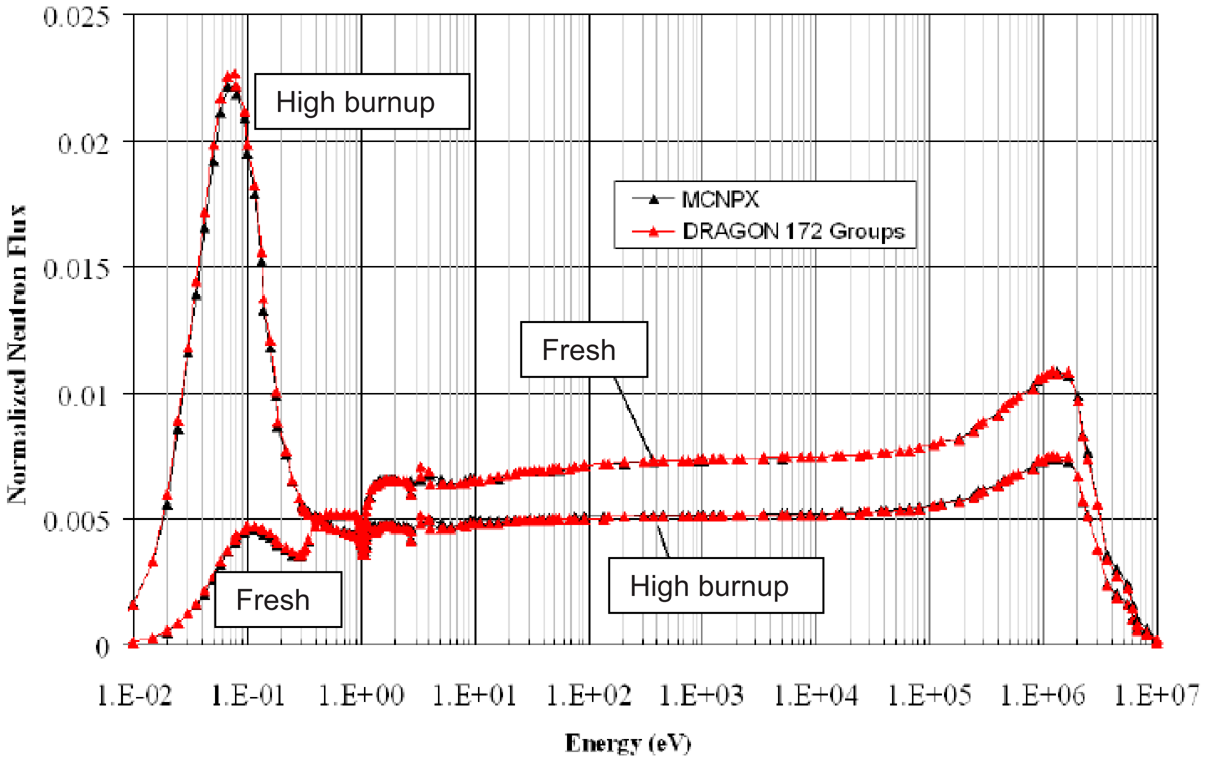

Figure 2-2. Neutron energy spectra for fresh and high burnup HTGR-IMF fuel calculated using DRAGON and MCNPX with diffuse boundaries [Petti2009, figure 2-125]

There are, of course, two key types of neutron reactions - fission and capture by structure and impurities. The cross sections for both are highest at the lowest neutron energy and therefore one would expect somewhat higher thermal neutron fission and structure/impurity capture in a PWR than VHTR. The cross sections for high-energy neutron fission (“fast fission”) are typically higher for the highest neutron energy; so again, one would expect somewhat higher fission in a PWR than VHTR.

Table 2-5 shows that the expectation is indeed correct - the PWR can obtain somewhat more neutrons than a VHTR from both fissile isotopes (fission at thermal neutron energies) and fertile isotopes (fission at high neutron energy and transmutation to fissile isotopes, which then fission at thermal neutron energies). D-factors are defined so that the more negative the number, the more neutrons are provided. Isotopes with D-factors below 0 are net neutron providers; isotopes with D-factors above 0 are net neutron sinks.

Table 2-5. Comparison of D-Factors for Dominant Actinides (more negative means more neutrons)

	D-Factors [Taiwo2005]		
Isotope	PWR-UOX	VHTR-UOX	Comments
U-235	-0.65	-0.53	Slightly more neutrons are provided by these fissile isotopes in the representative LWR.
Pu-239	-0.83	-0.72	
Pu-241	-0.95	-0.88	
U-238	-0.02	0.26	Slightly more neutrons are provided by these fertile isotopes in the representative LWR.
Np-237	0.96	1.11	
Pu-238	0.01	0.12	
Pu-240	0.04	0.12	
Pu-242	0.72	0.79	
Am-241	0.80	0.90	
Am-243	-0.24	-0.20	
Cm-244	-1.22	-1.19	

In addition to the comparison at the lowest and highest neutron energies, Figure 2-1 also shows the VHTR to have higher neutron population at the in-between or "epithermal" energies. This leads to more absorption by actinides - both useful (fission and transmutation of **fertile to fissile**[g]) and harmful (transmutation of **fissile to fertile** and transmutation of **fertile to fertile**[h]). Table 2-6 shows 1-group cross sections for the PWR and VHTR, incorporating contributions at thermal, epithermal, and fast neutron energies. The cross sections U-235 (fission), Pu-239 (fission), and U-238 (capture) are all higher in the VHTR.

Table 2-6. Comparison of 1-Group Cross Sections (thermal, epithermal, fast)

	1-group cross section [Taiwo2005]	
Isotope	PWR-UOX	VHTR-UOX
U-235 fission	37.23	44.18
Pu-239 fission	89.17	164.54
U-238 capture (leading to Pu-239)	0.91	4.80

Compared to a PWR, the older analysis of the VHTR at the same burnup (100 MWth/kg-iHM. 14%) requires higher U-235 enrichment because of the considerations noted above, leading to about 15% higher uranium ore required despite the higher thermal efficiency of the VHTR.[Taiwo2005] For example, consider the comparison in Table 2-7.

---

[g] The most important example of **fertile to fissile** transmutation occurs when the product of neutron capture is very short-lived, e.g., fertile U238 (n,γ) fissile U239, which immediately decays to fertile Np239, which quickly decays to fissile Pu239.

[h] **Fertile to fertile** transmutation occurs when the immediate fissile product is short-lived, e.g., fertile Np237 (n,γ) fissile Np238, which decays with 2.117 day halflife to fertile Pu238. So, neutron capture by Np238 has transformed one fertile to another.

Table 2-7. Previous Comparison of Representative VHTR versus PWR [Taiwo2005]

Parameter	PWR-UOX	VHTR-UOX	
Burnup, MWth-day/kg-iHM	100	102	
Uranium enrichment of fresh fuel	8.5%	14.0%	Presumably due to more absorption at epithermal neutron energies
Assumed thermal efficiency	33%	47%	
Assumed capacity factor	85%	85%	
UOX fuel requirement, tonne-iHM/yr/GWe	9.4	6.5	
Enrichment feed/product	16.24	27.02	Assuming 0.2% tails
Uranium ore requirement, tonne-ore/yr/GWe	153	175	VHTR requires 14% more ore at same burnup

We are unaware of a direct comparison of the wasting of neutrons by structure and impurity capture, but the VHTR/HTGR should be somewhat less subject to neutron absorption and capture by non-actinide isotopes. If true, the prospects for using "minimum fuel treatment" with VHTR/HTGR should be higher (from the physics perspective) than an LWR.

2.3.2 Modified open cycle - 1-recycle TRU fuels

When the fuel cycle objective is to consume the most TRU as quickly as possible, one considers the 1-recycle IMF class of fuels. By being uranium free, no new TRU isotopes are created and only TRU isotopes can be fissioned. The problem is that this leads to unacceptably high concentration of fission in the fuel so that neither LWR nor HTGR use a pure TRU oxide such as PuO_2.

In LWRs, the TRU is typically diluted by adding a third material to the oxide matrix such as ZrO_2. This material is fine unless one wants to have a relatively separable fuel to allow for subsequent recycling. Other options such as MgO (i.e. TRU-oxide/MgO) are inadequately stable during irradiation. This has lead to proposals for fuels such as TRU-oxide/MgO-ZrO_2 [Fridman2007] and TRU-oxide/Er_2O_3/ZrO_2.[Stanculescu1999] There is nil experience with such materials; they would have to be developed as fuels almost from scratch.

In HTGR, fundamentally new materials do not appear to be required. Carbon is already used in multiple ways (porous, pyrolytic, SiC, bulk graphite, and carbide or oxycarbide fuel in some cases) and absorbs few neutrons. Carbon is therefore an obvious option as a diluent of TRU-oxide or TRU-oxycarbide. Indeed, carbon diluted kernels of PuO_2 have been successfully irradiated in the U.K. Dragon experiment.[McEachern2007b]

Chemical getters can also be added to reduce oxygen pressure inside coated particles, such as Ce_20_3 and ZrC.[McEachern2007b]

Table 2-8 shows some key reactor safety considerations for LWR-IMF and HTGR-IMF. Neither LWR-IMF nor HTGR-IMF can be considered fully optimized. One set of possibilities is use of resonance absorber additives such as erbium.[Petti2009, Fridman2007, Stanculesu1999] A self-consistent analysis of such additives, their cost, impact on overall fuel performance, etc. is warranted; the HTGR-IMF analyses is closer to this goal (fully self-consistent analysis) than the LWR-IMF analyses.

Table 2-8. Reactor Safety Parameters for IMF Fuels

Consideration	MOC/1-recycle LWR-IMF [Hoffman2005] values used 1500 ppm-B for fresh fuel	MOC/1-recycle HTGR-IMF prismatic [Petti2009]	MOC/1-recycle HTGR-IMF pebble bed [Petti2009]
Fuel integrity as majority of HM fissions to fission products, most notably He, Kr, Xe	Not known to be studied - likely a problem	Studied in detail, e.g., stress analysis of fuel particles. Porous expansion volume (porous buffer) and gas getters can make performance acceptable.	
Power peaking	Add third component to TRU-oxide; e.g., ZrO_2	Dilute kernels with carbon and possibly ZrC or ZrO_2	
Void coefficient (all licensed US reactors have negative void coefficient)	Void coefficient for fresh IMF is negative up to ~80% coolant voiding [Hoffman2005]	Not a problem	
Moderator temperature coefficient (MTC) (negative values required)	-17.71 pcm/K [Hoffman2005, temperature range not given]	At high burnup (600 MWth-day/kg-iHM), positive up to around 700 °C (i.e. below operating temperature)	For burnup over 300 MWth-day/kg-iHM, MTC is positive for temperature < 500 °C (i.e. below operating temperature)
Fuel temperature coefficient (FTC) (negative values required)	-0.44 pcm/K (much less margin than LWR-UOX) [Hoffman2005, temperature range not given]	At high burnup (600 MWth-day/kg-iHM), remains smaller than -1 pcm/K up to ~700 °C	
Decay heat during reactor accidents, relative to once-through	Not known to be studied - may not be a problem as the relevant time scale is short	Is an issue starting a few hours after reactor shutdown because of Cm242; the passive safety objectives of the modern HTGR make decay heat particularly important. See chapter 4.	
Delayed neutron fraction	Requires study		
Prompt neutron lifetime	Requires study		
LWR-IMF void coefficient problem reduced if less than full core IMF, e.g., LWR-UOX&IMF. LWR-IMF FTC problem reduced if less than full core IMF, e.g., LWR-UOX&IMF.			

Table 2-9 shows calculated TRU consumption for LWR-IMF and HTGR-IMF. These numbers must be viewed with caution as different constraints, levels of detail, and amount of design optimization lie behind each. When considering the LWR-IMF results, recall that several safety considerations in Table 2-8 suggest that TRU consumption will have to be reduced from the existing results. For the HTGR-IMF, the MTC is positive and further tweaking or reduction in burnup would be required.

Table 2-9. TRU Consumption Comparison

Fuel Composition	MOC/1-recycle LWR-IMF	MOC/1-recycle HTGR-IMF prismatic	MOC/1-recycle HTGR-IMF pebble bed
LWR-IMF-NpPu	64% 633 MWth-day/kg-iHM [Stillman2004a]	64% 600 MWth-day/kg-iHM [Petti2009]	75% 702 MWth-day/kg-iHM [Petti2009]
LWR-IMF-NpPuAm	56% 553 MWth-day/kg-iHM [Hoffman2005]	50% 500 MWth-day/kg-iHM [Petti2009]	60% 560 MWth-day/kg-iHM [Petti2009]
LWR-IMF-TRU	56% 554 MWth-day/kg-iHM [Stillman2004a]		
Although numerically similar, it would be inappropriate to consider these as having the same credibility or basis without additional detailed analyses and design optimization. Consider the safety considerations in Table 2-8.			

One can conclude the following:

- Claims of 50-70% burnup in either modern commercial-scale LWRs and HTGRs with NRC licenses must be viewed with caution.
- IMF with NpPuAm or all-TRU will have slightly lower possible burnup than IMF-NpPu.
- HTGR-uranium-fuel burnup has been tested experimentally at higher burnup than LWR-uranium-fuel. HTGR-IMF test fuel has been taken to 67% of its initial Pu. It is hard to see how the LWR-IMF fuel pellet would maintain adequate integrity if taken to such burnups.
- The HTGR avoids the void coefficient problem but fuel and/or moderator temperature coefficient may be the dominant concern in either reactor type.
- The HTGR approach to IMF appears to be an easier R&D task (relative to U-based fuel) as the LWR-IMF contains a significantly different fuel matrix.
- The claims of such high burnup appear more credible in the HTGR than LWR.

2.3.3 Full Recycle MOX-like or UOX/IMF-like Fuels

Neither this team, D. Petti, nor F. Venneri have found studies of full recycle HTGR-MOX or HTGR-UOX&IMF.

It has been noted, “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.”[Taiwo2005]

Indeed, full recycle of LWR-MOX and LWR-UOX&IMF have been shown possible from the physics standpoint - whether the recycled TRU is Pu, NpPu, NpPuAm, or all-TRU.[Bays2009] Calculations were taken to 10 recycles and then the equilibrium composition was estimated. As with any burner reactor (thermal or fast neutron spectrum), fissile material must be added each recycle. (Only with a fast breeder reactor can the ultimate supporting material be fertile such as DU.)

On a "per kg-TRU-fuel" basis, the gamma and heat emission of full recycle LWR are less than an order of magnitude higher than for full recycle fast reactors On a per kg-HM-fuel" basis, the full recycle fast reactor has heat and gamma emission higher than a full recycle LWR; this arises because the full recycle fast reactor has higher TRU/heavy metal than a full recycle LWR.

The neutron emission penalty of LWR versus fast reactors is higher, approaching 2 orders of magnitude per kg-TRU-fuel or 1 order of magnitude (per kg-HM-fuel).

Table 2-10 shows the same safety considerations for full recycle MOX as for 1-recycle IMF. Many of the concerns appear lower than for 1-recycle IMF. The most notable exception is the void coefficient; a known critical constraint on the TRU loading in MOX.

Table 2-10. Reactor Safety Parameters for Full Recycle MOX Fuels

Consideration	Full Recycle LWR-MOX	Full Recycle HTGR-MOX
Fuel integrity as majority of HM fissions to fission products, most notably He, Kr, Xe	Appears manageable	No reason to believe a problem
Power peaking	Manageable	No reason to believe a problem
Void coefficient (all licensed US reactors have negative void coefficient)	Dominant constraint, limits the TRU content to ~10% if only Pu or ~8% otherwise, of the initial heavy metal [Youinou2009] Uranium enrichment increased from LWR-UOX to keep k > 1	Not a problem
Moderator temperature coefficient (MTC) (negative values required)	More negative than LWR-UOX because of higher resonance absorption [Youinou2009]	
Fuel temperature coefficient (FTC) (negative values required)	More negative than LWR-UOX because of higher resonance absorption [Youinou2009]	
Decay heat during reactor accidents, relative to once-through	Not known to have been studied	
Delayed neutron fraction	Acceptable [Youinou2009]	
Prompt neutron lifetime	An issue, appears acceptable [Youinou2009]	
LWR-IMF void coefficient problem reduced if less than full core IMF, e.g., LWR-UOX&IMF. LWR-IMF FTC problem reduced if less than full core IMF, e.g., LWR-UOX&IMF.		

The favorable results for LWR full recycle and the lack of void coefficient difficulties in an HTGR suggest it is very appropriate to look at full recycle in an HTGR.

2.4 Fuel Burnup

Fuel burnup is measured two ways:

- Fraction of initial heavy metal consumed (percent)

- Energy released per initial heavy metal (MWth-day/kg-iHM = GWth-day/tonne-iHM)

Comparisons among fuel cycles appear different depending on the measure used. For example, a LWR-IMF fuel that lasts as long (calendar time) as LWR-UOX fuel and provides the same power density (per fuel assembly) will nonetheless have these ***calculated*** burnups:[Fuel Cycle Transmutation Library]

- LWR-UOX - 51 MWth-day/kg-iHM, 5.3% of the initial uranium turned into fission products
- LWR-IMF - 554 MWth-day/kg-iHM, 56% of the initial TRU turned into fission products
- LWR-IMF-NpPu - 633 MWth-day/kg-iHM, 64% of the initial TRU turned into fission products

If fission products are retained in the fuel, the maximum theoretical fraction burnup of enriched uranium was estimated by S. Bays and S. Piet to be 80-85% in a thermal reactor; locally higher burnup is possible if other parts of the reactor are “driving” it, i.e., supplying extra neutrons. This could be pushed somewhat higher if neutron absorbing fission products were released from the fuel and removed from the reactor during operation. TRU fuel should have a similar result.

In 2009, TRISO-coated HTGR-UOX fuel achieved 19% burnup of its enriched uranium fuel. For 19.75% initial uranium enrichment, this is 178.5 MWth-day/kg-iHM.[12010a, Grover2010b] The highest LWR-UOX burnup we know of is 81 MWth-day/kg-iHM or ~8.4%.

The Dragon test reactor in the UK achieved “good success” with HTGR-IMF-Pu (coated PuO_2) to peak values of 750 MWth-day/kg-iHM and 80% burnup of the Pu and average values of 630 MWth-day/kg-iHM and 67% burnup of the Pu.[McEachern2007b] This is approaching the maximum theoretically possible. We do not know the corresponding experimental value for LWR-IMF.

2.5 Brief Summary of Current Reactor Projects

Table 2-11 summarizes current gas-cooled reactor projects. With the demise of the South African project, the number of continents involved in gas-cooled reactor projects has dropped from 4 to 3. Still, there is considerable work in progress and considerable locations in which new fuels and fuel coatings could be tested.

Table 2-11. Current Gas Cooled Reactor Projects [Venneri2007]

Country	Program
US	Next Generation Nuclear Plant, Texas MH3R, AGR Fuel
France	AREVA/CEA Technology Development and Design of Power Reactors
South Africa	Creating Infrastructure to construct and operate fleet of Pebble-Bed Modular Reactors. [*Defunct 2010*]
China	Operating experimental reactor (HTR-10), plan to build HTR power reactors
EU	Innovative reactor and fuel cycle designs, safety, RAPHAEL for GT-MHR technology; PUMA for DB-MHR technology
Japan	Operating experimental reactor (HTTR), developing technology for high-temperature and hydrogen production
Russia	GT-MHR design, developing technology for disposition of excess weapons plutonium
South Korea	Developing MHR technology for hydrogen production
MHR, AGR, GT-MHR, HTTR, HTR are roughly equivalent acronyms	

2.6 What Distinguishes the HTGR Technology Family?

There are four categories of characteristics that can cause differences between the HTGR technology family and alternatives:

- **Definition** of the technology family, i.e., high-temperature gas-cooled reactors.
- **Inherent** properties associated with defined characteristics.
- **Priorities**, some parts of option space have been developed over others because of past priorities, e.g., PUREX for separating U and Pu from used LWR-UOX fuel because of the need (at the time) to recover Pu for energy value for both civilian and military purposes. Techniques that might have recovered all the TRU elements were not developed because there was no perceived need at the time.
- **Technology availability,** some parts of option space have been developed and some have not, often because of the availability of relevant support technologies. For example, fundamental metal-related technologies (fabrication, separation) were developed long before the nuclear age. High quality ceramics for high-temperature service are a relatively new technological area.

The first two are not changeable; the latter two are. Table 2-12 summarizes the distinguishing characteristics of the HTGR technology family.

Table 2-12. Distinguishing Technological Characteristics of the HTGR Technology Family

Characteristic	Reasons	Fuel cycle ramifications
Thermal spectrum	Definition	HTGR technology family is most comparable to LWRs, HWRs, some molten salts, etc.
Gas coolant	Definition	• Coolant does not moderate neutrons • Coolant does not capture neutrons • Coolant has low radioactivity, increasing incentive to have robust barriers keeping fuel and fission products out of the coolant • Fuel does not have to be protected from coolant
High temperature (key HTGR virtue)	Definition	• High thermal efficiency • Fuels, fuel coatings, and other materials must survive high temperature operation, hence emphasis on ceramic materials (oxides, carbides, oxycarbides, graphites) more so than metals • Online fission product removal conceivable in vented pebble designs
Carbon as moderator	Inherent	• Positive void coefficient problem disappears (often constrains TRU loading in LWR fuels) • Carbon and nitrogen impurities in carbon produce C14 • Mass (waste) of graphite blocks in prismatic design; pebble graphite in pebble design; and graphite, SiC, pyrolitic carbon, and other possible kernel coatings
TRISO coating of small particles	Priorities (very robust containment)	• Long-lived and leach resistant TRISO particles in HTGR is a used fuel direct dispose option • TRISO-coated fuel can be dissolved in only very aggressive acids (but note that this is also true of Zr alloys in LWRs); recycling TRISO fuels therefore requires either mechanical preparation or aggressive acids or burning • Small particles cannot be simply chopped (as can Zr alloy LWR cladding) to expose the fuel; therefore more aggressive mechanical means are required
Pebble bed design (fuel in small, mobile form)	Tech availability Priorities	• On-line refueling possible (possible fuel cycle advantage, availability enhancement, proliferation issues) • Easier to divert or steal# (even if online refueling is not used)#
Prismatic design (fuel in large, stationary form)	Tech availability Priorities	• Harder to divert or steal# than pebbles • Added head-end separations step to remove fuel from graphite block; recycle graphite if practical; handle TRISO fuel

"Divert" refers to host country diversion of safeguarded material. "Steal" refers to subnational group theft of material. IAEA safeguards are focused on host country behavior, not subnational theft.

2.7 Technology Readiness

Having surveyed past work, we can usefully describe the technology readiness of the illustrative concept technologies. Table 2-13 shows the technology readiness level (TRL) definitions we considered for possible use in this report. At the present time, there seems inadequate reason to depart from the simplest TRL scheme.

Table 2-13. Description of Technology Readiness Levels

TRL	AFCI comparison reports and other reports of that era	Generic TRL definition from GNEP reviews (Appendix 0) from appendix B for "GNEP Deployment Studies" DE-PS01-07NE24448	TRL definition per GNEP technology plan
1	**Concept Development** – The concept is still at a basic level. Suitable options for various applications are defined based on first principles and fundamental knowledge, with the critical technical issues or "showstoppers" identified, a work-around for showstoppers defined, and a verification plan developed.	Basic principles observed and formulated	Lowest level of technology readiness. Scientific research begins to be translated into applied research and development
2		Technology concepts and/or applications formulated	Invention begins. Once basic principles are observed, practical applications can be invented. Applications are speculative and there may be no proof or detailed analysis to support the assumptions.
3		Analytical and experimental demonstration of critical function and/or proof of concept	Active research and development is initiated. This includes analytical studies and laboratory studies to physically validate analytical predictions of separate elements of the technology.
4	**Proof of Principle** – The concept has been shown to be technically feasible, but performance characteristics for operational plant performance are uncertain. Development is performed using laboratory scale experiments and analytic extrapolations to full-scale behavior.	Component and/or bench-scale validation in a laboratory environment	Invention begins. Once basic principles are observed, practical applications can be invented. Applications are speculative and there may be no proof or detailed analysis to support the assumptions.
5		Component and/or breadboard validation in a relevant environment	Active research and development is initiated. This includes analytical studies and laboratory studies to physically validate analytical predictions of separate elements of the technology.
6		System/subsystem model or prototype demonstration in relevant environment	Integration of basic technological components for testing in laboratory environment. Includes integration of "ad hoc" hardware in the laboratory.
7	**Proof of Performance** – The concept is known to be technically feasible and there is considerable performance data, but scale-up to commercial scale is uncertain. Large-scale demonstrations on portions of the processes are performed, yielding final performance specifications, including statistical assessments and initial indications of economic performance.	System prototype demonstration in prototypic environment	Demonstration of prototype system in an operational environment at the engineering scale.
8		Demonstration of integrated system with full-scale equipment completed successfully; short (~1 month) periods of sustained operation	End of system development. Technology proven to work in operational environment at the engineering to full scale.
9		Full-scale demonstration successfully completed; sustained operations for a minimum of three months.	Full scale application of technology in its final form at mission conditions.
10	**Commercial Experience** – The technology has analogous commercial experience somewhere in the world and there is good understanding of economic performance.	Not included	Not included

Our evaluation of the technology readiness of some of the more important HTGR options are given in the following Tables. Gas-cooled reactors have been in commercial experience in the UK (CO2 cooled), Germany (helium), and US (helium). Thus, Table 2-14 shows both the HTGR itself and TRISO-coated HTGR-UOX as entering commercial experience.

Table 2-14. Technology Readiness for Once Through (HTGR-UOX)

Technology Readiness	Reactor	Fuels	Separations	Waste
Concept Development				
Proof of Principle				Direct dispose - has not been done in any country
Proof of Performance	HTGR	TRISO coated HTGR-UOX		
Commercial Experience				

Table 2-15. Technology Readiness for 1-Recycle with Min Fuel Treatment (impure HTGR-IMF-TRU)

Technology Readiness	Reactor	Fuels	Separations	Waste
Concept Development		Some work on AIROX of used LWR UOX fuel, but not applied to subsequent use in an HTGR		
Proof of Principle				Used HTGR-IMF fuel would be disposed, no direct disposal has been done in any country
Proof of Performance	HTGR			
Commercial Experience				

Table 2-16. Technology Readiness for 1-Recycle of TRU Recovered from LWR (HTGR-IMF)

Technology Readiness	Reactor	Fuels	Separations	Waste
Concept Development		HTGR-IMF-TRU fuels are under development		
Proof of Principle		HTGR-IMF-Pu fuels have been made and tested to high burnup	Concepts applicable to recycling LWR-UOX have been tested	Used HTGR-IMF fuel would be disposed, no direct disposal has been done in any country
Proof of Performance	HTGR			
Commercial Experience				

Table 2-17. Technology Readiness for HTGR Full Recycle (HTGR-MOX)

Technology Readiness	Reactor	Fuels	Separations	Waste
Concept Development		No work has been known to have done on HTGR-MOX-TRU fuel		Graphite recycling is in development. Waste streams appropriate for HTGR-MOX separation would be partially based on aqueous separation waste forms
Proof of Principle			Concepts applicable to recycling HTGR-MOX would be similar to those for HTGR-UOX	
Proof of Performance	HTGR	Much work done on HTGR (U, Th)O_2 fuel		
Commercial Experience				

3. DESCRIPTION OF ILLUSTRATIVE CONCEPTS WITHIN HTGR TECHNOLOGY FAMILY

Chapter 1 described the option space for HTGRs. In that option space, resources and information availability have been adequate to quantitatively study four illustrative concepts with some comments and observations on three others. The four concepts are as follows:[Petti2009]

- Once through, HTGR, prismatic UO_2 fuel, 110 MWth-day/kg-iHM burnup - designated OTC-HTGR-UO2-110 (prismatic) in graphs
- Once through, HTGR, pebble bed UO_2 fuel, 100 MWth-day/kg-iHM burnup - designated OTC-HTGR-UO2-100 (pebble bed) in graphs
- Modified open cycle, HTGR, prismatic NpPu fuel, 605 MWth-day/kg-iHM burnup – designated MOC-HTGR-IMF-NpPu (prismatic) in graphs
- Modified open cycle HTGR, pebble bed NpPuAm fuel, 560 MWth-day/kg-iHM burnup – designated MOC-HTGR-IMF-NpPuAm (pebble bed) in graphs

Prior to the current study, there was one older once through HTGR case in the Transmutation Library and that is included in several of the graphs: 102 MWth-day/kg-iHM burnup – designated OTC-HTGR-UO2-102 in graphs.

The two MOC cases listed above are assumed to use fuel created with very pure NpPu or NpPuAm feedstreams from used LWR-UOX fuel, presumably via aqueous separation.

The biggest limitation is that we have been unable to find or create data for a full recycle case. Where possible, we provide comments and information on that type of fuel cycle strategy.

3.1 Key Parameters for Illustrative Concepts

Table 3-1 provides description of the illustrative concepts. Table 3-2 quantifies mass flow information.

Table 3-1. Preliminary Description of Illustrative HTGR Concepts

Parameter	Once through HTGR UOX (prismatic or pebble bed)	MOC (1-recycle) HTGR-IMF-NpPu (prismatic) HTGR-IMF-NpPuAm (pebble bed)	Full Recycle HTGR (prismatic or pebble bed)	
Fissile enrichment of fuel	≤19.75% achieved 15.5% prismatic 9.6% pebble bed	58%-66%, higher if UOX-Pu 59% prismatic, NpPu fuel 58% pebble bed, NpPuAm fuel	Not yet studied	
Burnup (MWth-day/kg-iHM)	190 achieved 110 prismatic 100 pebble bed	500-700, higher if Pu, lower if TRU 605 prismatic, NpPu fuel 560 pebble bed, NpPuAm fuel	Not yet studied	
Burnup (% of iHM)	19% achieved 11% prismatic 10% pebble bed	50-75%, higher if Pu, lower if TRU 60% prismatic, NpPu fuel 58% pebble bed, NpPuAm fuel	Not yet studied	
Fuel residence time	Prismatic: 1.5 years x 2 = 3 years Pebble bed: 183 days x 6 passes = 1100 days total	Prismatic: 300 days x 4 = 1200 days total Pebble bed: 250 days x 6 passes through core = 1500 days total	Not yet studied	
Refueling method	Prismatic - batch refueling Pebble bed - online refueling an option		Pebble bed (online)	Batch
Reactor capacity factor	Not known, 90% assumed in calculations in this report			
Reactor lifetime	Not known at this time			
Fuel physical form	Coated spherical particles	Coated spherical particles	Coated spherical particles	Pellets?
Matrix material of fuel	UO2 or UCO	TRU oxide or oxycarbide	U-TRU oxycarbide	U-TRU oxide
Fuel fabrication	TRISO	TRISO	TRISO	Other
Fuel cladding/ coating	TRISO	TRISO	TRISO	SiC tubes?
Separation - head-end process	N/A	Chop LWR fuel	Crush to powder	Chop
Separation - fuel/ fission product	N/A	UREX+1 on LWR fuel	Density based	TBD
Separation - Th/U/TRU if any	N/A		UREX+1	UREX+1
Nature of waste streams	Unprocessed used fuel	Unprocessed used HTGR fuel + wastes from UREX+1 separation of LWR fuel	Wastes from UREX+1 separation	Wastes from UREX+1 separation
Primary technical challenges	Implement NGNP	Show viability of high TRU loading	Estimate maximum TRU loading	Early TRL stages

Table 3-2. Main Mass Flows per GWe at 90% Capacity Factor and 50% Thermal Efficiency

Parameter	Once through HTGR-UOX prismatic	Once through HTGR-UOX pebble bed	MOC (1-recycle) HTGR-IMF-NpPu prismatic	MOC (1-recycle) HTGR-IMF-NpPu pebble bed	Full Recycle HTGR
Burnup MWth-day/kg-iHM	110	100	51 for LWR 605 for HTGR	51 for LWR 560 for HTGR	TBD
Uranium enrichment	15.5%	9.6%	4.3% for LWR-UOX None for HTGR-IMF		TBD
Fueling rate, tonnes-HM/yr	5.98	6.57	15.93-LWR 0.20-HTGR 16.13-total	16.09-LWR 0.21-HTGR 16.30-total	TBD
Ratio of NU to EU	30.0	18.4	8.02 for LWR-UOX None for HTGR-IMF		TBD
Ratio of DU to EU	29.0	17.4	7.02 for LWR-UOX None for HTGR-IMF		TBD
Natural uranium, tonnes-U/yr	179.1	120.9	127.8-LWR 0-HTGR	129.1-LWR 0-HTGR	TBD
Net TRU production, tonnes-TRU/yr	0.17	0.12	0.09	0.09	TBD
Unprocessed used fuel mass to disposal,					None
tonnes-U/yr	5.17	5.82	0.00	0.00	
tonnes-TRU/yr	0.17	0.12	0.08	0.09	
tonnes-FP/yr	0.64	0.64	0.12	0.12	
tonnes-total/yr	5.98	6.57	0.20	0.21	
Processed mass to disposal					
tonnes-RU/yr (LLW)	0	0	14.88	15.03	
tonnes-TRU/yr (HLW)	0	0	0.01 (AmCm)	0.00 (Cm)	(process loss only)
tonnes-FP/yr (TBD waste)	0	0	0.84	0.85	
tonnes-total/yr	0	0	15.73	15.88	
DU mass to storage, tonnes/yr	173.2	114.4	111.9-LWR 0-HTGR	113.0-LWR 0-HTGR	TBD
LWR assumed to have 33% thermal efficiency and 90% capacity factor. Note that the net flux of FP to disposal is higher for the 1-recycle (deepburn) cases because of the lower thermal efficiency of the LWR.					

3.2 Impurities

The baseline impurities in fuel are important for assessing activation products such as Cl36 as well as providing a comparison point for the potential high-impurity HTGR-IMF (deepburn) concept. TRISO-coated HTGR-UOX fuel kernels are exceptionally pure, as evidenced in Table 3-3. However, the elements covered in the current specifications do not include fission products. Some of the key fission products that cause difficulty during irradiation are Ag, Pd, and Cs.[Petti2009] The ratio of fission yield for some of the fission products of those elements between Pu239/U235 are as follows:

Pd105 (stable) - 6
Pd106 (stable) - 11

Pd107 (6.5e6 yr) - 23
Pd108 (stable) - 40
Ag109 (stable) - 47
Pd110 (stable) - 25
Cs133 (stable) - 1.1
Cs135 (2.3e6 yr) - 1.2
Cs137 (30 yr) - 1.1

Thus, the troublesome Pd and Ag are produced at about an order of magnitude faster rate in Pu239 fission than U235. This is being analyzed in the HTGR deep burn program. Looking back at the available separations information shown in chapter 2, we note that mechanical reforming does not remove any of these three elements. AIROX removes Ag and Cs, but not Pd. Melt refining removes Cs but not Ag or Cd. So, the presence of fission products, which may be incompletely separated from recycled fuel in MOC and full recycle HTGR options, but which have not yet been addressed in HTGR fuel purity specifications, presents a potential and as-yet-unresolved issue that could impact how separations of used fuel is done to minimize those impurities, and how recycled fuel is manufactured and used in those cases.

Table 3-3. TRISO-coated HTGR-UOX Fuel Impurities

Impurities	AGR-1 (laboratory scale)		AGR-2 (industrial scale)	
	Specification	Measured	Specification	Measured
Li	<15 ppm		<15 ppm	
Na	<50 ppm		<50 ppm	
Al	<30 ppm	2.1 ppm	<30 ppm	4.9 ppm
P	<30 ppm		<30 ppm	
S	608		387	
Cl	<25 ppm		<25 ppm	
Ca	<50 ppm	4 ppm	<71 ppm	6 ppm
V	<50 ppm	Ti+V = 6 ppm	<50 ppm	Ti+V = 3.8 ppm
Cr	<50 ppm	< 0.30 ppm	<50 ppm	< 0.18 ppm
Mn	< 8 ppm	< 0.30 ppm	< 8 ppm	< 0.18 ppm
Fe	<50 ppm	0.6 ppm	<50 ppm	0.7 ppm
Co	<5 ppm	< 0.30 ppm	<5 ppm	< 0.18 ppm
Ni	<50 ppm	< 0.30 ppm	<50 ppm	< 0.18 ppm
Cu	<30 ppm		<30 ppm	
Zn	<50 ppm		<50 ppm	
Sum (Cr, Mn, Co, Ni)		0.57 ppm		0.39 ppm

3.3 Used HTGR Fuel Separation and Waste Management

HTGR options that include separating HTGR fuel need to contend with three different types of materials that would arise from reprocessing operations:

- The fuel itself
- C/SiC coatings (or other coatings as appropriate)
- Graphite blocks

Reprocessing used prismatic reactor fuel may follow the flow diagram shown in Figure 3-1.[Petti2009] The fuel-bearing compacts would be core-drilled out of the graphite core structure. The graphite blocks would, if practical, be recycled new HTGR core material. The fuel compacts (also called sticks) would be pretreated through crushing and either physical separations or incineration steps to break open the TRISO coatings and (to the extent practical) separate those coatings from the fuel kernels. The fuel kernels would then be processed using a number of separations process choices to separate the fission product poisons from re-useable actinide isotopes. The re-useable actinides could then be processed into new fuel; the separated waste streams would be processed into waste forms designed based on the radiochemical and physical properties of the separated waste streams, to meet waste acceptance criteria for transportation, storage, and final disposition.

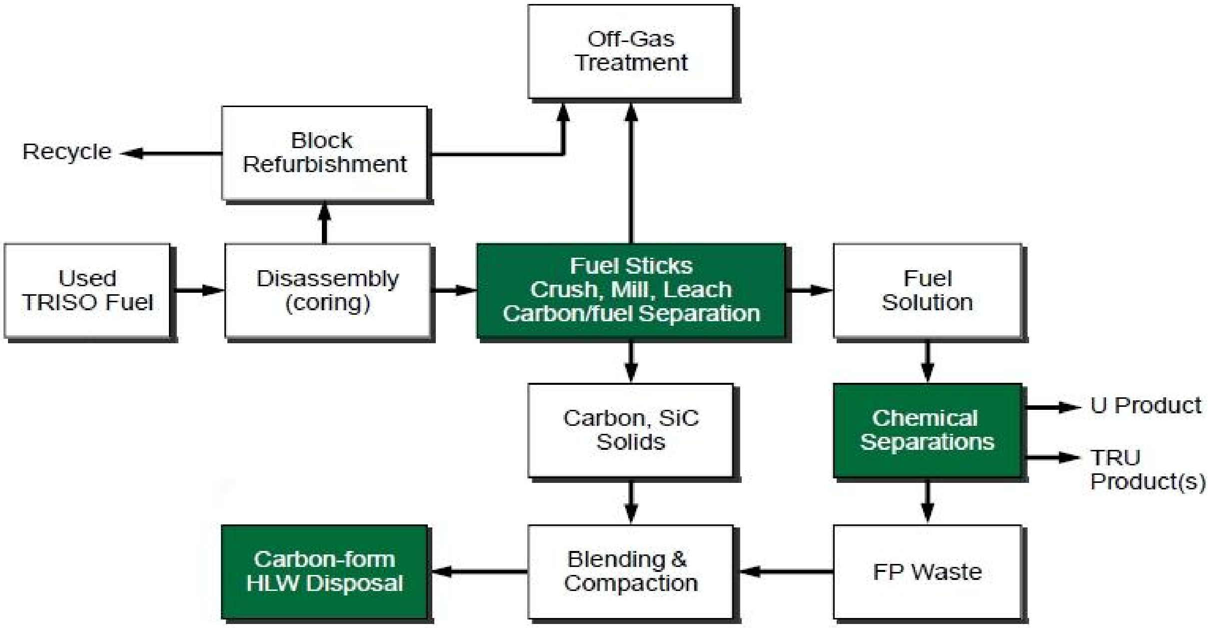

Figure 3-1. Overview of a potential TRISO-coated fuel recycle approach.[Petti2009, Figure 2-452]

Separations of crushed C/SiC coating particles from fuel kernels may be based on physical separations processes (such as a cyclonic or other process based on mass and/or size difference), or they may include thermal oxidation processes to convert the solid-phase C to CO_2. Thermal oxidation processes would also need to address how to separate Si carbide or oxide species that would not entirely volatilize from the fuel kernels. Either way, the separation of broken coating particles and the fuel particles cannot be expected to result in perfect separations; we can expect that some amount of heavy metals and fission products from the fuel kernels would contaminate the coating particle stream, and some amount of C/Si coating particles would contaminate the fuel kernel stream.

Both of these separated streams would require some additional processing to enable recycle of the heavy metals, separation of the fission products, and disposition of the C/SiC and fission product waste streams.

This figure, in which at least some separated fission products and C/SiC wastes are re-combined into a single C-SiC-FP HLW disposal waste form, pre-supposes that this waste form can be developed that would stabilize the fission products and meet waste acceptance criteria. If this waste form ultimately is not viable, then multiple waste streams and waste forms are likely, as shown in Table 4-1 and described below.

3.3.1 Separation and Processing of the Fuel Itself

Upon recycle, the fuel particles need to be separated from the coatings and graphite to enable separations processes and recycling, using methods mentioned above. Waste fission products can then be separated from recyclable components. Regardless of separations process, the most commonly considered radioactive gaseous fission products (H3, C14, I129, and Kr85) evolve in gas streams from the selected preparation and can be captured using the same gaseous fission product capture processes considered for LWR separations processes.

These gaseous FPs tend to start evolving into gas streams when the durable coatings are broken or; but essentially all of the gaseous FPs may quantitatively gasify by the time the carbon coatings are oxidized at high temperatures and the fuel meat is either dissolved into aqueous or molten salt solutions or separated at elevated temperatures. Species such as iodine that may, in part, dissolve into aqueous dissolver solutions can also eventually evolve into eventual treatment process off-gas streams.

- Gaseous species containing H3 are oxidized to tritiated water, which is captured from the off-gas systems using mole sieve sorbent. Periodically the mole sieve sorbent is cycled to remove the captured water, which is destined for a grout waste form in most recent separations process designs. Care is normally taken to minimize the amount of "tramp" non-radioactive hydrogen that could enter the process in water or other H-bearing chemicals, to minimize the final amount of the H3 waste form. Even-so, the amount of tramp water can be orders of magnitude larger than the amount of tritiated water in the process, but the relative amount of the H3 waste form (compared to amounts of other waste forms) is normally small enough that this increase in the H3 waste form is not considered significant (unless the tritium is not nearly all separated from fuel prior to entering an aqueous dissolver, at which time the large mass of dissolver water dilutes the relatively much smaller mass of tritiated water that would come from the tritium).

- The C14 could theoretically evolve in various forms depending on temperatures and stoichiometry, but is most commonly expected to evolve primarily in CO_2 during thermal oxidation, fuel dissolution, and fuel volatilization steps. If this C14 needs to be captured to comply with air emissions regulations (studies such as reported in Soelberg2009 show that C14 may not require any air emissions control) then capturing the C14 in the off-gas will be needed. The evolution of stable carbon from carbon coating oxidation will significantly dilute the C14 isotope in the off-gas species. This will not change the expected waste form which is typically a calcium carbonate grout containing CO_2 captured in an off-gas caustic scrubber, but it will certainly increase the total mass of that grout by orders of magnitude, unless much of the carbon coating material is physically separated prior to oxidation, volatilization, or dissolution steps, or unless some sort of isotopic carbon separation step becomes viable for HTGR fuel recycling.

- Prior studies have shown that, among the four most commonly evaluated gaseous fission products, emissions of I129 could require the highest levels of control (up to 99.99% in some cases) to comply with regulatory requirements (Soelberg2009). I129 and other halogens will evolve during pretreatment processes that break open the coatings, during separations processes that dissolve or heat/volatilize components of the fuel meat, and during high temperature fission product waste treatment processes such as vitrification. Current designs for capturing I129 include using inorganic zeolite or other sorbents that can then either be disposed as the waste form or can by cycled to recover the sorbed iodine, which is then placed into a separate waste form. Iodine waste loadings may be low, ranging from under 1 wt% up to 25 wt% for some innovative waste forms, and the actual loading of iodine will be reduced due to the presence of other competing halogens in the waste form. Even-so, the total mass of iodine waste forms is expected to be small compared to the expected masses of other fission product waste forms.

- Noble gases are captured from off-gas streams using cryogenic, zeolite-type sorbents, or sputtered-metal capture techniques. While cryogenic processes have been used in past used fuel reprocessing facilities such as the Idaho Chemical Processing Plant (ICPP), and sputtered metal techniques are being investigated in countries such as Japan, current U.S. research and designs are based mainly on zeolite capture. The zeolite sorbents are then cycled to desorb the noble gases, which, in most current thinking, would be compressed into gas cylinders. This approach is based on the fact that the noble gas that contributes the most to dose from air emissions, Kr85, has a sufficiently short half-live of 10.7 years that decay storage for a limited number of decades is a viable way to disposition the noble gases (presuming that waste policy and regulatory structure allows this approach).

The separation and disposition of other fission products depends in part on the separations process used to recycle HTGR fuel. Candidate processes include:

- Partial separations processes such as Melt Refining and AIROX, considered for fuel cycle options designed to enable used fuel reprocessing but avoid full separations processes to reduce proliferation and terrorist mis-use potential.
- Aqueous processes
- Electrochemical processes such as METROX (metal recovery from oxide fuel)

3.3.1.1 Partial Separations Processes

Little information is available at this time on the waste forms and disposition of waste streams that could evolve from partial separations processes. Several suppositions about partial separations processes, that rely on the volatility of different components of used fuel under varying temperature, reducing, and oxidative conditions, can be made:

- The fission products considered most volatile (H3, C14, I129, and noble gases) are expected to volatilize; the capture and disposition of these fission products can follow the designs described above.\
- Processes that rely on fluoride volatility can produce waste streams contaminated with fluorides that will impact the waste forms and waste loadings
- Separation of different fission products based on their volatility can open the door to separating some fission products into separate waste forms containing more volatile, less volatile, and mainly non-volatile fission products, as compared to separations made by aqueous and electrochemical separations that separate fission products based on their solubility in different chemical solutions.
- Partial separations processes might not separate recyclable actinides from fission products poisons as efficiently as full separations processes. This will result in potentially higher levels of fission product poisons and other contaminants in recycle streams, and potentially higher levels of TRU contamination in separated waste streams. Contaminants in the recycle materials will affect how recycled fuel materials are re-fabricated into new fuel, and how the new fuel is used in recycle reactors. TRU and other contaminants in the waste streams will affect the heat, radiotoxicity, dose, gamma, and neutron emissions from the waste streams, the utilization of fertile and fissile materials in the fuel cycle, and waste forms and waste loadings.

3.3.1.2 Aqueous Separations Processes

Aqueous processes can be expected to (after the pretreatment steps to deal with the fuel particle coatings) be similar to those researched and used for LWR oxide fuels. The waste forms for these separated waste streams can be similar in character and mass, to those from separating LWR oxide fuels. Dissolved carbonate levels in dissolver solutions may be higher for recycled HTGR carbide or oxycarbide fuels, unless those carbide and oxycarbide fuels are oxidized during pretreatment prior to dissolution. However, fission product waste stream processes such as vitrification would cause the evolution of those carbonates as mainly CO_2 in off-gas streams. To the extent that C14 has not already partitioned to pretreatment or dissolver off-gas streams, the CO_2 in fission product waste treatment off-gas streams may need to be scrubbed to control C14 emissions.

3.3.1.3 Electrochemical Separations/METROX

According to Petti2009, the variation of electrochemical separations called METROX has been used to recover uranium from simulated uranium TRISO fuel. Figure 3-2 summarizes a METROX process. In this process, a basket of UOX fuel kernels, after removal of the graphite and C/Si kernel coatings, is placed in a molten LiCl-KCl salt bath held in an electric furnace at a temperature of 500°C. (It is not known at this time if this process is also suitable for carbide or oxycarbide fuels). Uranium and all fuel components less noble than U (including U, TRU, all fission products, not including noble metals) dissolve into the molten salt bath. Residual carbon and SiC, and noble metal fission products, tend to not dissolve.

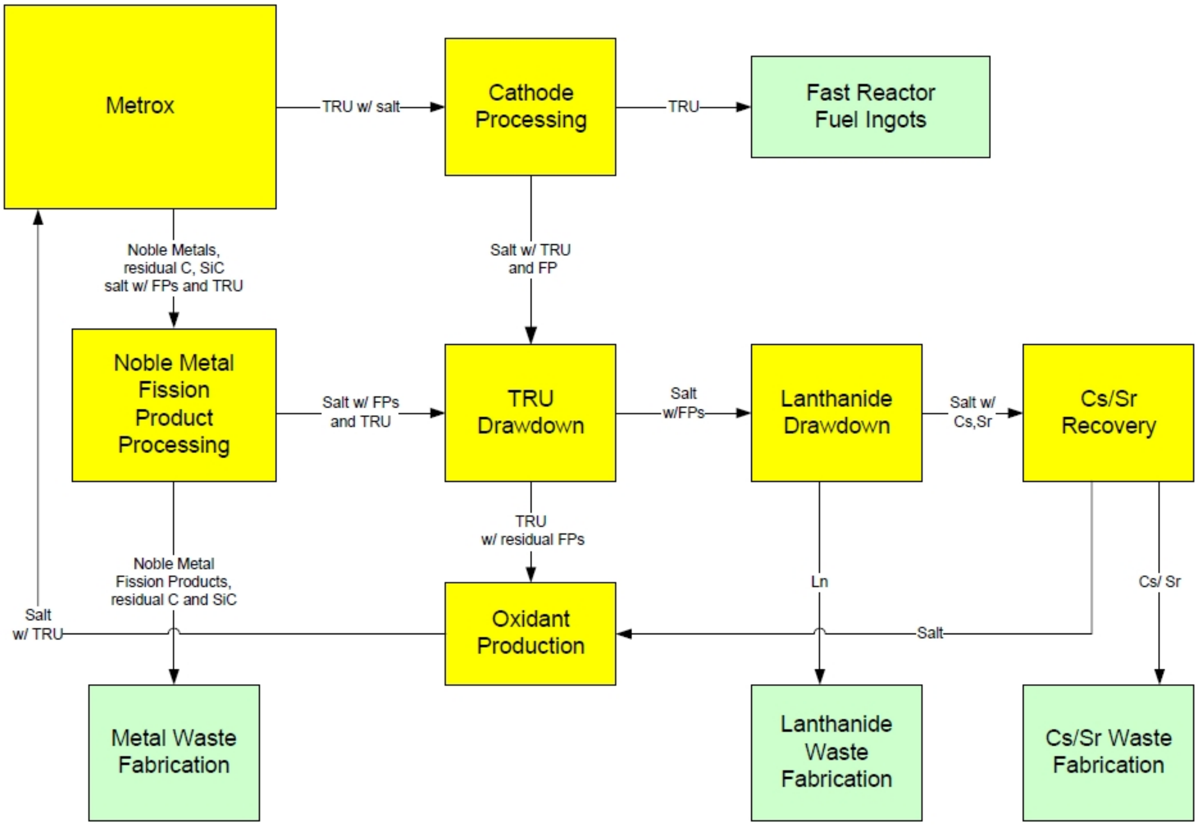

Figure 3-2. An overview of the METROX process.[Petti2009, figure 2-475]

Undissolved materials are removed and processed in a noble metal fission product process. Dissolved U and TRU can be electrodeposited on a cathode in the molten bath. With proper control of electrical conditions, dissolved fission products remain in the molten bath. The molten bath is also known to dissolve and retain halides including I129 that have not yet been volatilized from the used fuel in pretreatment processes.

This dissolution and electrodeposition process results in somewhat incomplete separations. Some of the fuel may not entirely dissolve and may partition to and contaminate the noble metal fission product waste stream. Salts of all dissolved constituents will also coat or be physically encapsulated in the mass of undissolved basket materials, and in the electrodeposited U/TRU material. This physical incorporation of FP-laden salts could cause the need to use salt distillation processes to separate and recover this salt contamination from those U/TRU recycle streams. And, some portion of U/TRU isotopes will remain in the salt solution and could contaminate the eventual fission product waste stream that evolves from the salt bath material. So all of these separated product and waste streams may require secondary processing to improve separation efficiencies and reduce losses and contamination levels.

3.3.1.4 Electrochemical Separations/Cathode Processing

An induction furnace is used to purify the U/TRU metals collected on the cathode. Upon heating in the furnace, salts incorporated in the mass of U/TRU metal volatilize, are condensed from the melter off-gas, and transferred to the TRU drawdown operation for further purification. The U/TRU metal melts and is removed from the furnace in the form of a cooled ingot, for potential re-use as fast reactor fuel.

3.3.1.5 Electrochemical Separations/Noble Metal Fission Product Processing

A furnace similar to the cathode processor is used to process the undissolved material remaining in the fuel baskets. The baskets contain noble metal fission products, carbon, SiC, residual unreacted TRU oxides, and molten salt. The undissolved solids are placed into a crucible in a furnace until the salts volatilize. The volatilized salts are condensed from the off-gas system and sent to the TRU drawdown operation. The remaining solids are transferred to metal waste form fabrication.

3.3.1.6 Electrochemical Separations/TRU Drawdown

The salts distilled from the cathode processing melter and the noble metal fission product melter still contain amounts of U/TRU that, if not separated more efficiently from these salts, will represent a TRU loss and will contaminate the salt waste stream. These distilled salts are melted into a molten salt bath, where electrolysis is used to recover most of the remaining transuranic elements dissolved in the salt.

The TRU metal product recovered from the TRU Drawdown melter is sent to the oxidant production operation. The TRU-free salt is transferred to the Lanthanide Drawdown operation (if used). A small amount of chlorine gas that is evolved in the operation is captured, converted to a stable chloride and disposed.

3.3.1.7 Electrochemical Separations/Ln Drawdown

The electrochemical separation process is theoretically capable of separating Ln from the rest of the fission products dissolved in the salts. If a separate Ln waste stream is desired, then the Ln can be separated from the salts collected from the TRU Drawdown process, and salts that remained in the electrorefining melter, by electrodeposition on the cathode. The recovered Ln product is transferred to Ln waste fabrication while the TRU/Ln-free chloride salt is sent to Cs/Sr recovery for the final purification operation. A small amount of chlorine gas that is evolved in the operation is converted to a stable chloride and disposed of.

3.3.1.8 Electrochemical Separations/Cs/Sr Recovery

The final theoretical salt purification operations can recover the Cs/Sr as well as the Se, Te, Br, and iodine from TRU-/Ln-free drawdown salt. The recovered products are transferred to Cs/Sr waste fabrication and the salt is sent to oxidant production. Several operations, such as ion exchange, are available for Cs/Sr recovery, but no single process is selected at this time.

3.3.1.9 Electrochemical Separations/Oxidant Production

Transuranic metal collected from the TRU drawdown operation is used to produce the oxidant (transuranic trichlorides) required for electrorefiner operation. The oxidant provides the transport ion needed for the electrochemical process. The oxidant is mixed with purified salt from the Cs/Sr recovery process and recycled into the bulk molten salt electrolyte.

3.3.1.10 Electrochemical Separations/Noble Metal Waste Fabrication

The noble metal fission products, carbon, SiC, and unreacted HM oxides result are typically an unconsolidated, primarily metal waste form that is compacted into a waste form for final disposal.

3.3.1.11 Electrochemical Separations/Ln Waste Fabrication

If a separate Ln waste stream is produced, it can be melted with waste loadings up to 50 wt% in a lanthanide borosilicate (LaBS) glass for final disposition.

3.3.1.12 Electrochemical Separations/Cs/Sr Waste Fabrication

If a separate Cs/Sr waste stream is produced, it may be stabilized in a glass-bonded ceramic waste form in a two-stage process. The waste stream is first ground into a powder, and combined in a mixer with ground zeolite material, and heated. The zeolite incorporates the Cs, Sr, and other isotopes into the crystalline structure and inside cages in the crystalline structure. Then, the mixture is melted into a borosilicate glass which bonds the powders into a glass monolith. Waste loadings up to 20 wt% may be theoretically possible, but operations to date have not achieved such high loadings.

A mass balance was reported in Petti2009 that includes assumptions and results for waste form mass flowrates for METROX recycling. Key assumptions and results that may apply to this and other HTGR fuel cycle scenarios include:

- Complete recovery of graphite block in coring separation
- Process breaks carbon and SiC layers to expose inner fuel meat
- Initial compact composition is 49 wt% filler carbon, 25 wt% kernel carbon, 14 wt% SiC, 12 wt% TRU-oxide fuel
- Complete recovery of compact particulate
- During crushing/milling: Noble gases released to off-gas system is estimated at 50 wt%; Tritium gas released to off-gas system is estimated at 60 wt%; Halogen gas released to off-gas system is estimated at 10 wt%
- Physically separated compact carbon and coating waste stream contains 75 wt% of the initial compact filler carbon, 10 wt% pellet carbon and SiC, and 1 wt% of fuel

- During particle and waste stream separation: Noble gases released to off-gas system is estimated at 25 wt% (75% of total noble gases now released); tritium gas released to off-gas system is estimated at 30 wt% (90% of total tritium now released); halogen gas released to off-gas system is estimated at 5 wt% (15% of total halogens)
- During METROX separations: 99 wt% HM oxide to metal product recovery, 1% of HM oxide left in UDS; complete conversion of active metal fission products to molten chloride salts; complete carbon, SiC, and noble metal retention in baskets; residual salt on products is estimated at 20 wt%; remainder of gaseous fission products (except for halogens) are released to off-gas system; process gas generated is in the form of carbon dioxide only
- During cathode processing: Complete salt distillation from metal; crucible surface remains inert and does not react with actinides; ingot free of fission products, carbon, SiC, and salt impurities (these assumptions should not be considered valid and future evaluations, if performed in more detail, should better estimate separations inefficiencies at this stage)
- During noble metal fission product, TRU drawdown, Ln drawdown, and Cs/Sr drawdown: Complete salt distillation from noble metal, carbon, and SiC; crucible surface is inert to noble metal product complete recovery of transuranic elements from salt; negligible amount of Ln and Cs/Sr in the HM product; complete recovery of lanthanides from salt; negligible amount of Cs/Sr in the waste salt product; Chlorine gas evolved is converted to stable chloride; all remaining impurities (Cs, Sr, Se, Te, Br, and I_2) are recovered from the salt (these assumptions should not be considered valid and future evaluations, if performed in more detail, should better estimate separations inefficiencies at this stage)

3.3.2 Separation and Processing of the C/SiC Coatings

The graphite, pyrolytic carbon, and SiC coatings on fuel kernels represent a mass that is much greater than the mass of the initial heavy metal. At this time, HTGR fuel recycle designs all include breaking open these coatings to expose the fuel meat to separations processes. Potential HTGR separation methods also include separating the fuel kernels from the coatings using either (a) physical separations processes based on mass, size, or density differences between broken coating fragments and fuel kernels, or (b) thermal oxidation processes that burn the coating components, converting the carbon to CO_2. Issues not addressed in the references reviewed at this time include how to separate Si species such as unreacted SiC and Si oxides, some of which would remain mingled with the fuel kernels.

Table 3-4 gives the sizes of individual fuel particles used in the illustrative concepts. Table 3-5 provides the particle chemical composition. Calculations were done and checked against information from M. Pope and [Petti2009].

Table 3-4. Sizes of Illustrative TRISO-Coated Particles

TRISO fuel particle layers and densities (from Mike Pope)					Calculated here							
	Enriched uranium		TRU (deepburn)		Enriched uranium				TRU (deepburn)			
	Thick -ness (μm)	Density (g/cm^3)	Thick - ness (μm)	Density (g/cm^3)	Inner radius (cm)	Outer radius (cm)	Volume (cc)	Mass (g)	Inner radius (cm)	Outer radius (cm)	Volume (cc)	Mass (g)
Fuel Kernel*	350	10.86	200	10.00	0	0.0175	2.24e-5	2.44e-4	0	0.0100	4.19e-6	4.19e-5
Buffer Layer	100	1.05	120	1.05	0.0175	0.0275	6.47e-5	6.79e-5	0.0100	0.0220	4.04e-5	4.24e-5
IPyC	35	1.90	40	1.90	0.0275	0.0310	3.77e-5	7.16e-5	0.0220	0.0260	2.90e-5	5.51e-5
SiC	35	3.18	35	3.18	0.0310	0.0345	4.72e-5	1.50e-4	0.0260	0.0295	3.39e-5	1.08e-4
OPyC	40	1.90	40	1.90	0.0345	0.0385	6.70e-5	1.27e-4	0.0295	0.0335	4.99e-5	9.49e-5
In the case of the fuel kernel, thickness value is actually diameter						Totals	2.39e-4	6.61e-4		Totals	1.57e-4	3.42e-4

Table 3-5. Chemical Composition of Illustrative TRISO-Coated Particles

	Enriched uranium		TRU (deepburn)	
	Mass (g)	Percent of particle	Mass (g)	Percent of particle
Fuel (HM only)	2.15E-04	32.52%	3.70E-05	10.80%
Fuel (O)	2.89E-05	4.37%	4.93E-06	1.44%
Coating (Si)	1.05E-04	15.92%	7.55E-05	22.08%
Coating (C)	3.12E-04	47.19%	2.25E-04	65.68%
Total	6.61E-04	100.00%	3.42E-04	100.00%

In the pebble bed illustrative concept there are 50,000 TRISO-coated particles in each 3-cm pebble. Their chemical composition is given in Table 3-6. These particles are in a carbon matrix (1.7 g/cc density) with 181 g-carbon/pebble. Note that the entire pebbles are 91-97% carbon. The loading of the TRU-pebble is lower because the specific heat generated by that fuel is much higher than from enriched uranium. Apart from the pebbles, graphite reflector blocks add another 340 tonnes-carbon to the core.

Table 3-6. Chemical Composition of Illustrative Pebbles

	Enriched uranium		TRU (deepburn)	
	Mass (g)	Percent of pebble mass	Mass (g)	Percent of pebble mass
HM in particles	10.75	5.0%	1.85	0.9%
O in particles	1.44	0.7%	0.25	0.1%
Si in particles	5.26	2.5%	3.78	1.9%
C in particles	15.59	91.8%	11.24	97.0%
C in pebble, outside of particles	181.00		181.00	
Total mass of pebble	214.04	100.0%	198.11	100.0%

In the prismatic design, the particles are first made into compacts, about 10,753 particles per compact, in a matrix of carbon. Table 3-7 gives the compacts' chemical composition. Then, 3126 compacts are made into a single fuel element. Table 3-8 provides the fuel element composition. And, again, the carbon content ranges from 91% to 97% of the material discharged from the reactor.

Table 3-7. Chemical Composition of Illustrative Fuel Compacts in Prismatic Concept

	Enriched uranium		TRU (deepburn)	
	Mass (g)	Percent of compact	Mass (g)	Percent of compact
HM in particles	2.31	16.3%	0.40	3.4%
O in particles	0.31	2.2%	0.05	0.4%
Si in particles	1.13	8.0%	0.78	6.6%
C in particles	3.35	73.6%	2.52	89.5%
C in compact, outside of particles	7.10		8.04	
Total mass of pebble	14.21	100.0%	11.79	100.0%

Table 3-8. Chemical Composition of Illustrative Fuel Elements in Prismatic Concept

	Enriched uranium		TRU (deepburn)	
	Mass (kg)	Percent of compact	Mass (kg)	Percent of compact
HM in compacts' particles	7.2	5.4%	1.2	1.0%
O in compacts' particles	1.0	0.7%	0.2	0.1%
Si in compacts' particles	3.5	2.6%	2.4	1.9%
C in compacts' particles	10.5	91.3%	7.9	97.0%
C in compacts,	22.2		25.1	
C in element, outside of compacts	90.0		90.0	
Total mass of pebble	134.4	100.0%	126.9	100.0%

3.3.3 Separation and Processing of the Graphite Blocks

Graphite is separated from the coated particles, and the C/SiC coatings are separated from the kernels, in all expected separating scenarios. Prismatic core graphite can be separated from the fuel compacts by coring the compacts out of the graphite blocks. Graphite in the pebbles must be separated when the pebbles and particles are crushed and/or burned.

The graphite core material gradually degrades in a reactor environment due to irradiation and oxidation.(Petti2009, Smith2009, Contescu2009, and Wichner2009). Irradiation can cause it to swell in the (c) direction and shrink in the (a) direction shown in Figure 3-3. Oxidation that can occur in an HTGR reactor core at high temperatures if oxidizing agents are present can reduce solid graphite mass as shown in Figure 3-4. Irradiation and oxidation can cause graphite to lose dimensional integrity, thermal conductivity, density, and strength. Properties of irradiated graphite removed from HTGR reactors worldwide are contained in a world-wide database summarized in Metcalfe2009.

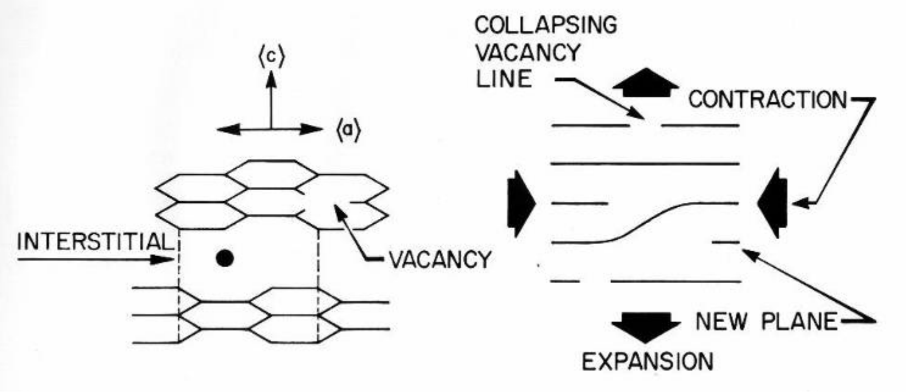

Figure 3-3. Radiation-induced damage to graphite (shrinking and swelling in different directions) that could occur in an HTGR reactor core.

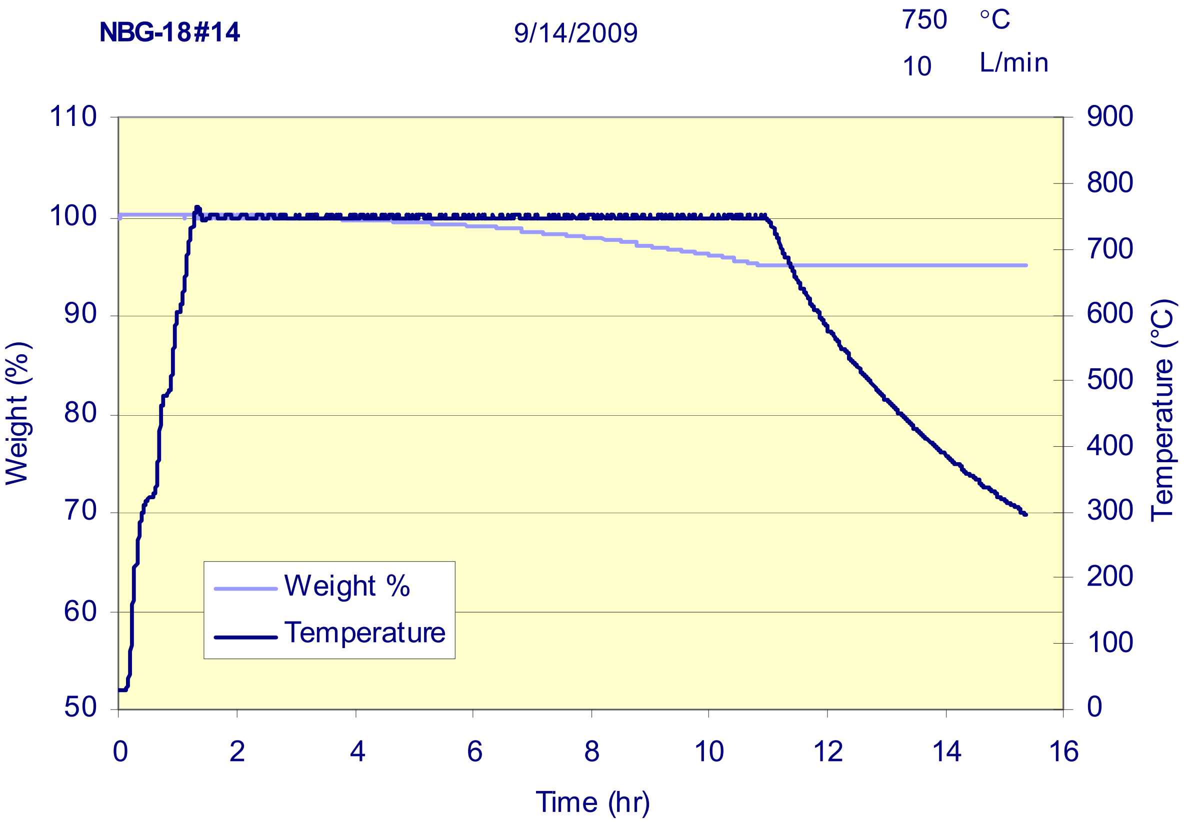

Figure 3-4. Mass loss of a surrogate graphite sample during oxidation tests [Smith2009]

Graphite in HTGR reactor cores must be replaced on a schedule than ensures reactor core integrity. It cannot be re-used in an HTGR reactor core again unless it can be re-fabricated to meet the necessary performance and dimensional specifications.

Most core graphite references cite the Prismatic graphite block configuration rather than the Pebble Bed configuration. Without more data specific to pebbles, we assume that issues and masses reported for graphite blocks can represent, in a first-order evaluation, graphite used in the pebbles of Pebble Bed reactor designs.

Two methods have been proposed or evaluated for re-using irradiated core graphite material removed from HTGRs. One method includes using the irradiated graphite to produce impermeable graphite that could be used to encapsulate spent HTGR fuel in a waste form that relies on the impermeability of the graphite as one of the barriers against radionuclide release in a repository environment (Fachinger2009).

Studies have also been performed to evaluate how to recycle the irradiated graphite into new HTGR core graphite material.[Petti2009, Pappano2009, and Albers2009] The recycling scheme proposed in these references involves crushing, grinding, chemical purification, pressing, and machining of irradiated graphite. Initial feasibility studies have been done to determine if irradiated graphite can be processed to overcome the radiation damage that weakens the graphite structure through annealing the crystalline structure, and if irradiated graphite recycling is practical on a large scale. These studies have indicated that recycling irradiated graphite is possible, the use of irradiated graphite does not significantly change the manufacturing process, and recycled graphite densities and other properties are comparable to those of virgin graphite. However, annealing to remove crystalline damage has thus far not been successful, perhaps because the annealing temperatures were too low.

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4. ASSESSMENT OF ILLUSTRATIVE CONCEPTS WITHIN HTGR TECHNOLOGY FAMILY

This chapter provides an assessment of the illustrative concepts described above. Consistent with the recent metrics review,[Dixon2009a] the assessment here examines the six areas of waste management, used fuel storage, resource sustainability, proliferation resistance and physical protection, economics, and safety. Where possible, the metrics from that report are used. In many cases, information is currently inadequate to calculate those metrics so instead qualitative and quantitative indicators are given. For example, it is not possible to provide credible estimates of levelized unit cost of energy or fuel cycle costs. Instead, observations are made that provide insights to potential economic performance.

The authors of the Options Study [Wigeland2009] have meanwhile asked for information on various parameters that somewhat overlap the metrics proposed by Dixon. Where quantitative information is lacking, they ask for whether the performance of the fuel cycle option is above or below the “nominal” value. For this report, we interpret “nominal” as the performance of the analogous LWR fuel cycle.

There is no consistency in depth or quality of information that is available.

4.1 Used HTGR Fuel Disposal, Separations, and Waste Management

The basic objective is to “reduce the long-term environmental burden of nuclear energy through more efficient disposal of waste materials.”[Dixon2009a] The recommended metrics are as follows:

- Quantity of actinides in high level waste for disposal, normalized per unit of energy produced
- Decay heat of waste at the time of waste form creation, normalized per unit of energy produced
- Radiotoxicity of disposed high level waste at 10,000 and 1,000,000 years, normalized per unit of energy produced
- Estimated dose for disposed high level waste, normalized per unit of energy produced
- Mass of initial heavy metal in waste disposed (all waste types), normalized per unit of energy produced
- Packaged volume of waste disposed (all waste types), normalized per unit of energy produced
- The packaged volume of material to be stored, normalized per unit of energy produced.” [Dixon2009a]

4.1.1 Used HTGR Fuel Direct Disposal

Used HTGR fuel may be direct-disposed in once-through or modified open fuel cycles. In direct disposal cases, the used fuel and TRISO or BISO coatings on the fuel become components of the multiple barriers in a disposal scenario that prevents or at least minimizes release of potentially hazardous fuel constituents into the biosphere. In this respect, the performance of the TRISO or BISO fuels in a repository environment should be compared to the performance of HLW forms and other forms of disposed spent fuels.

Perhaps used TRISO and BISO fuels should be eventually qualified as waste forms. Modeling analyses have been done to evaluate corrosion of TRISO fuel coatings.[Peterson2006] Figure 4-1 shows results of preliminary modeling for five different TRISO fuel kernel types, and indicate that one might expect

TRISO coatings to provide a barrier against release of particle contents to the environment out to about 10,000 years. More details are available in Peterson2006 and Peterson[date unknown].

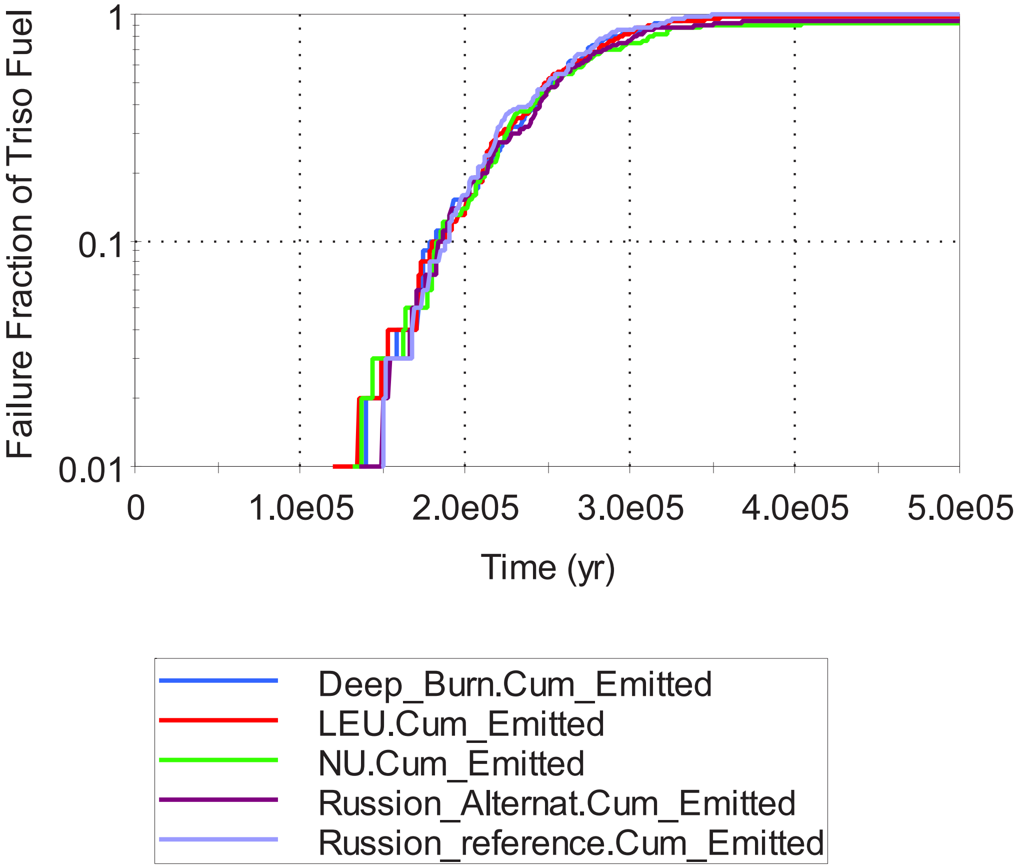


Figure 4-1. Results of corrosion modeling for five different TRISO fuel kernel types in a geologic repository environment.[Peterson2006]

An innovative alternative for direct disposing used HTGR fuel is to first encapsulate the fuel particles, compacts, or pebbles in a specially-designed low-porosity carbon that can be made using a portion of the used graphite core structure material, of which there is presently about 250,000 tonnes worldwide from HTGR reactors to date.[Fachinger2009] Figure 4-2 shows how specially designed low porosity graphite can increase the resistance of the graphite to corrosion. This could, if proven to be viable, provide an added barrier that may last in a repository environment for up to or over 10,000 years, to augment the roughly 10,000 year durability of the TRISO fuel coating.

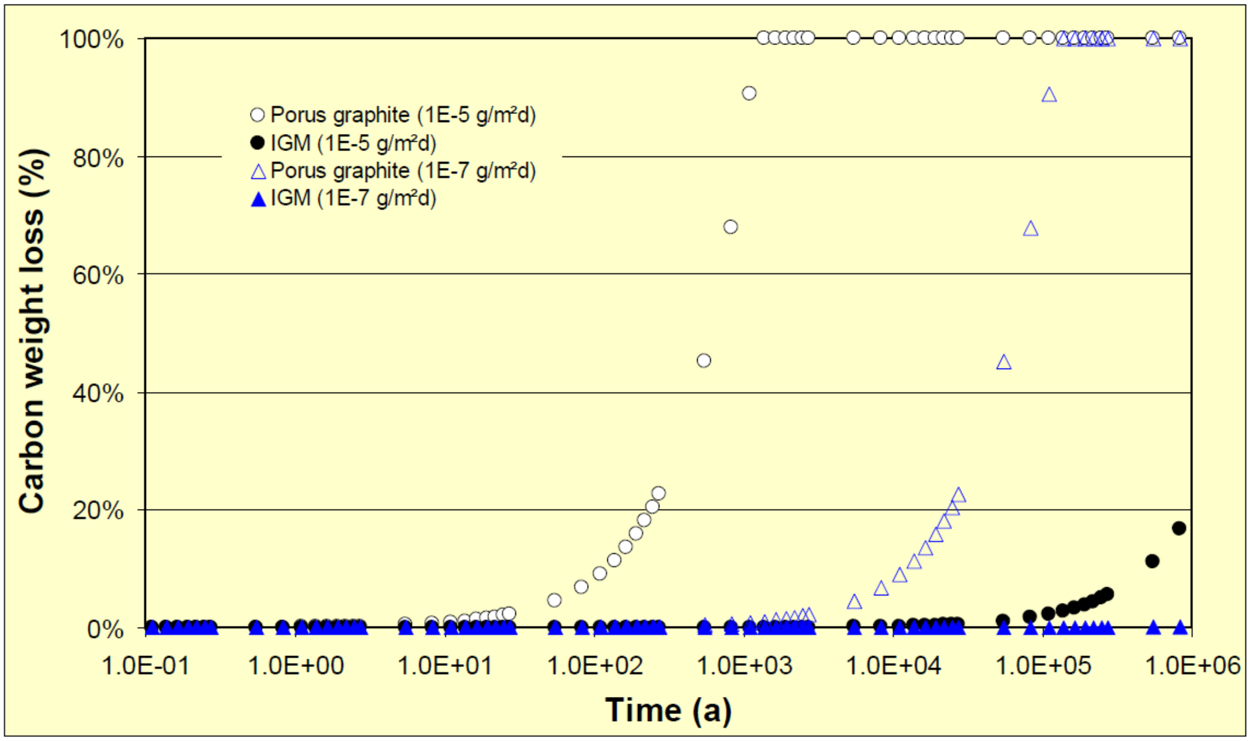


Figure 4-2. Increasing graphite corrosion resistance by reducing the graphite porosity.[Fachinger2009]

Conclusions from Peterson[date unknown] include:

> "Results of the analysis indicate corrosion rates and thicknesses of the outer pyrolytic carbon and silicon carbide layers, along with the time dependent temperature of the spent fuel in the repository environment, have a significant effect on the time to particle failure. The thicknesses of the kernel, buffer, and IPyC layers along with the strength of the SiC layer and the pressure in the TRISO particle did not significantly alter the results from the model. …a better understanding of the corrosion rates of the OPyC and SiC layers, along with increasing the quality control of the OPyC and SiC layer thicknesses, can significantly reduce uncertainty in estimates of the time to failure of spent TRISO fuel in a repository environment."

4.1.2 Radiotoxicity

All calculations for radiotoxicity, heat, gamma, neutrons, etc. of spent fuel and separated and discarded U and HLW are based on isotopic information in the Transmutation Library. The data for the four illustrative HTGR cases come from M. Pope and B. Boer, based on work done for the NGNP deepburn study.[Petti2009, Pope2010, Boer2010]

Figure 4-3 shows the radiotoxicity of the spent fuel from four once through LWR and HTGR cases with similar burnups (100 to 110 MWth-day/kg-iHM); there is little difference among them. The prismatic UO2 case shows a slightly higher radiotoxicity than LWR-UOX; this may or may not be significant. The LWR analysis parameters are based on mature designs; the HTGR parameters are not.

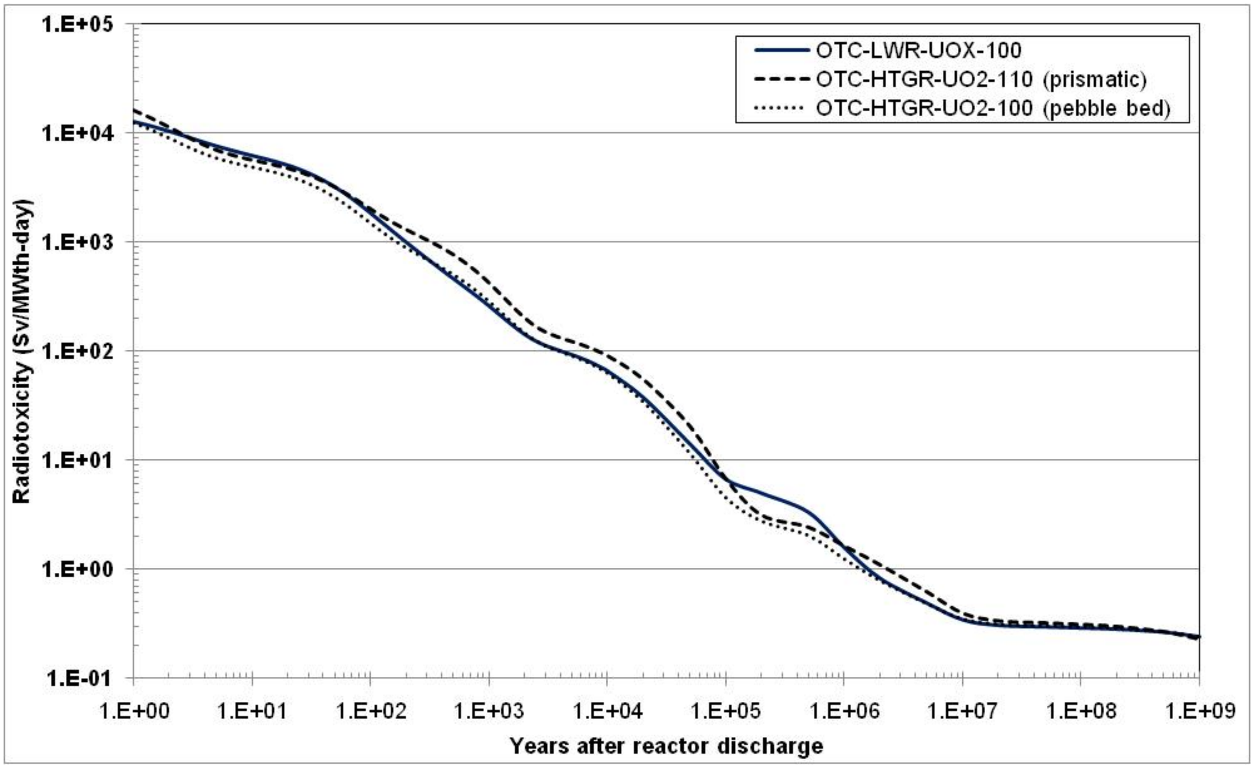


Figure 4-3. Radiotoxicity for three once through fuel cycles at 100-110 MWth-day/kg-iHM

Figure 4-4 shows the radiotoxicity of spent fuel for a wide range of once through cases ranging from a burnup of 7.5 up to 148 MWth-day/kg for the uranium-fed FR case at CR=0.50. There are four sets of cases to mention.

The bulk of the cases shown in this figure are for LWR UOX with burnup from 33 to 100 MWth-day/kg-iHM and these show a consistent pattern. On a "per MWth-day/kg-iHM" basis, there is more Pu239 for the lower-burnup cases and this increases the radiotoxicity in the 1e3 to 1e4 year time period. There is also more net uranium; this increases the radiotoxicity after 1e6 years. By 1e6 years, the radiotoxicity of fission products is mostly gone, most of the TRU have decayed into uranium isotopes, and the decay progeny of U235 and U238 have built to equilibrium levels. So, at such very long times, radiotoxicity is basically dominated by how much U+TRU mass each system starts with, which scales as 1/burnup.

The two CANDU cases mostly follow the LWR pattern; they are at such low burnups that they look to be separated from the LWR parameter scan.

As shown in the previous figure, the HTGR cases pretty well match that of LWR at 100 MWth-day/kg-iHM.

The final case is the uranium-fed fast reactor at CR=0.50. This is not one of the current "breed and burn" fast reactor concepts (for which isotopic data have not been received), but rather a past fast reactor at CR=0.50 [Hoffman2006] with an alternative feed material, namely 29% enriched U235.[Ferrer2007, Ferrer2008] Its burnup is 148 MWth-day/kg-iHM; it has significantly less radiotoxicity between 1e2 and 1e3 years, due to less Pu238 being generated by the chain of U235 (n,γ) U236 (n, γ) U237$\rightarrow$Np237 (n, γ) Pu237$\rightarrow$Pu238; these (n, γ) reactions are less likely in the fast neutron spectrum. The radiotoxicity is somewhat higher in the 1e3 to 1e4 period because there is relatively more Pu239, which is more preferentially consumed in a thermal spectrum.

Basically, among once through fuel cycles, the HTGR pebble bed results in spent fuel with radiotoxicity about as low or lower than any of the alternatives, other than the lower Pu238 case (OTC-U-fed-FR CR=0.50). The spent fuel radiotoxicity for the HTGR prismatic is slightly higher, but still relatively low compared to the lower-burnup LWR and CANDU cases.

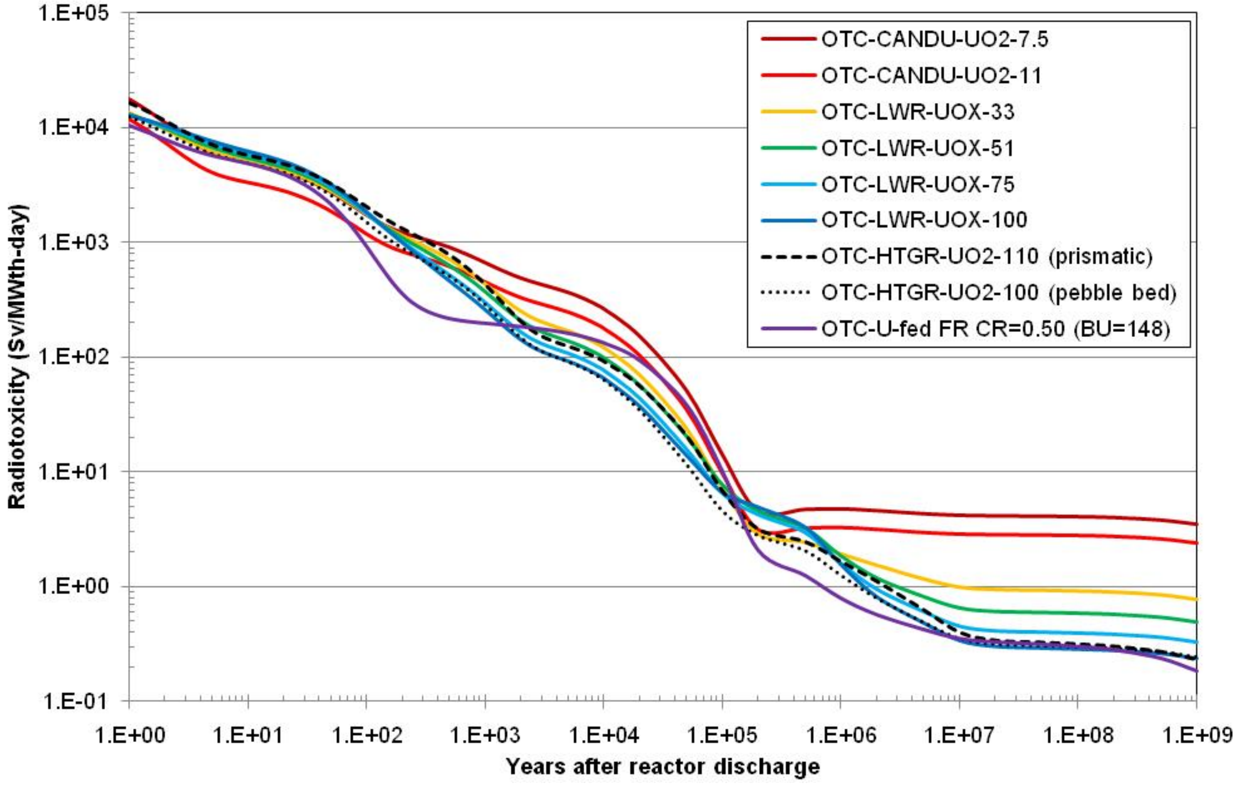


Figure 4-4. Radiotoxicity for a wide range of once through cases, 7.5 to 148 MWth-day/kg

Figure 4-5 shows the total radiotoxicity of the spent fuel and discarded U and HLW for LWR and HTGR cases for once through and for 1-recycle IMF-based modified open fuel cycle (MOC) fuels. The once-through cases are all near 100 MWth-day/kg-iHM and the MOC/IMF cases are all near 600 MWth-day/kg-iHM. The uranium-free MOC/IMF cases for both HTGR and LWR show significant radiotoxicity reduction compared to the once through cases between 100 and 100,000 years after reactor discharge, primarily due to consumption of Pu239; there is an increase of radiotoxicity between 100,000 and 1,000,000 years due to production of some of the fertile isotopes. 1-recycle MOX fuels in LWR and HTGR (not shown) exhibit little or no radiotoxicity reduction versus once through. As noted in chapter 2, the confidence in being able to burn TRU fuels to ~600 MWth-day/kg-iHM appears higher in the HTGR than the LWR; but, at the same burnup (if achieved), the resulting radiotoxicity is similar.

Some have compared HTGR-IMF performance with LWR-MOX and emphasized the lower radiotoxicity in HTGR-IMF. This is an apples-oranges comparison. The proper comparisons are HTGR-IMF to LWR-IMF and HTGR-MOX to LWR-MOX.

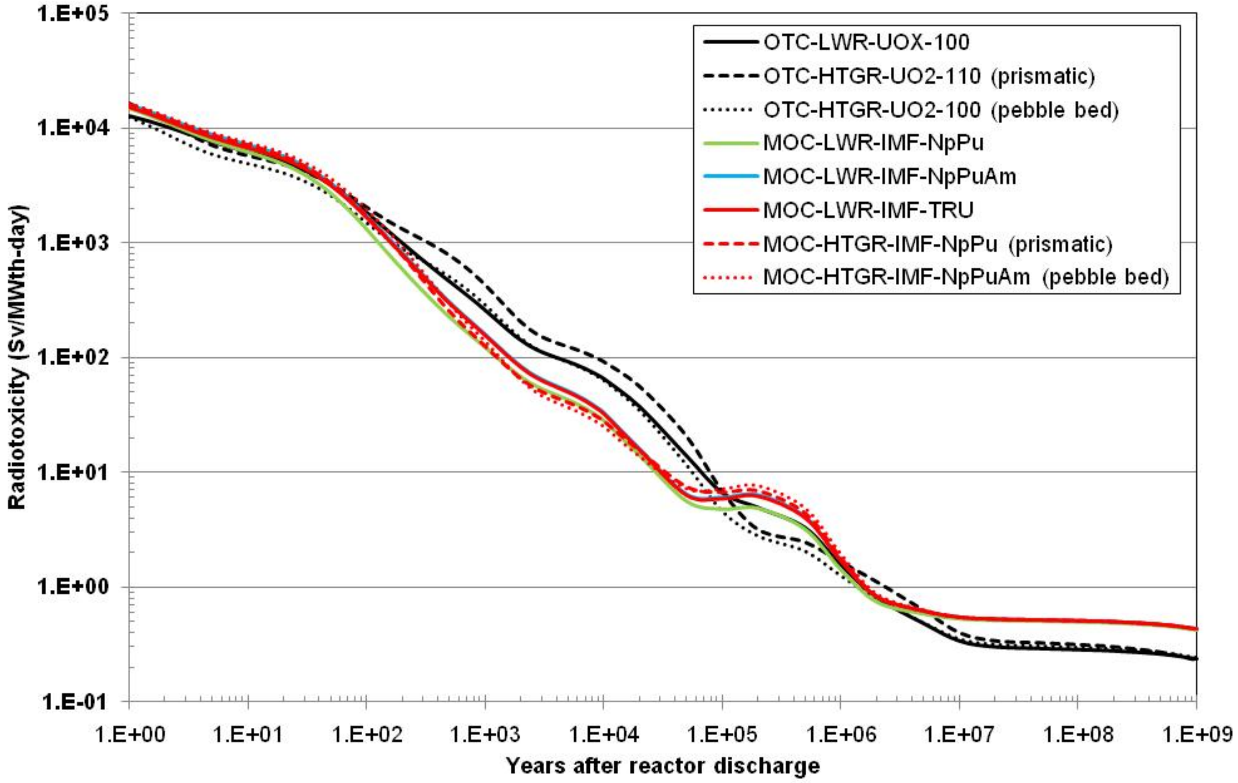


Figure 4-5. Radiotoxicity for a LWR and HTGR cases for UOX and 1-recycle IMF fuels

Since the previous figures show that at the same burnup LWR and HTGR spent fuels have similar radiotoxicity, it is useful to look at other LWR cases to see how their HTGR analogs might perform. Figure 4-6 includes the radiotoxicity of LWR-MOX taken to equilibrium (full recycle), with the required fissile support (enriched uranium) to keep successive recycles of unconsumed TRU critical.[Youinou2009] The processing losses were assumed to be 0.1% per recycle, consistent with the assumptions in the DSARR study.[Dixon2009] No used fuel is direct-disposed in these full-recycle cases, only HLW that contains the separated fission products, the separated and stored used U, and the 0.1% of TRU lost to the waste streams each recycle.

There are four such full recycle cases, recycling Pu, NpPu, NpPuAm, or all-TRU. When all-TRU are recycled (dark blue line), the radiotoxicity reduction is dramatic versus either once through or MOC, and even, to a lesser extent, versus the other three full-recycle cases. The HTGR analog of LWR-MOX-TRU should show similar or better behavior. For the other LWR-HTGR comparisons, the burnup and loadings of the fissile material were similar and the radiotoxicity results were similar. But, for full recycle MOX-TRU (or MOX-Pu, MOX-NpPu, MOX-NpPuAm), the TRU loading in the HTGR case could be higher. The LWR cases were limited to 10% Pu (for MOX-Pu) or 8% (for the other cases) because of the void coefficient issue.[Youinou2009] This constraint would not exist for the HTGR-MOX cases and so TRU consumption should be improved.

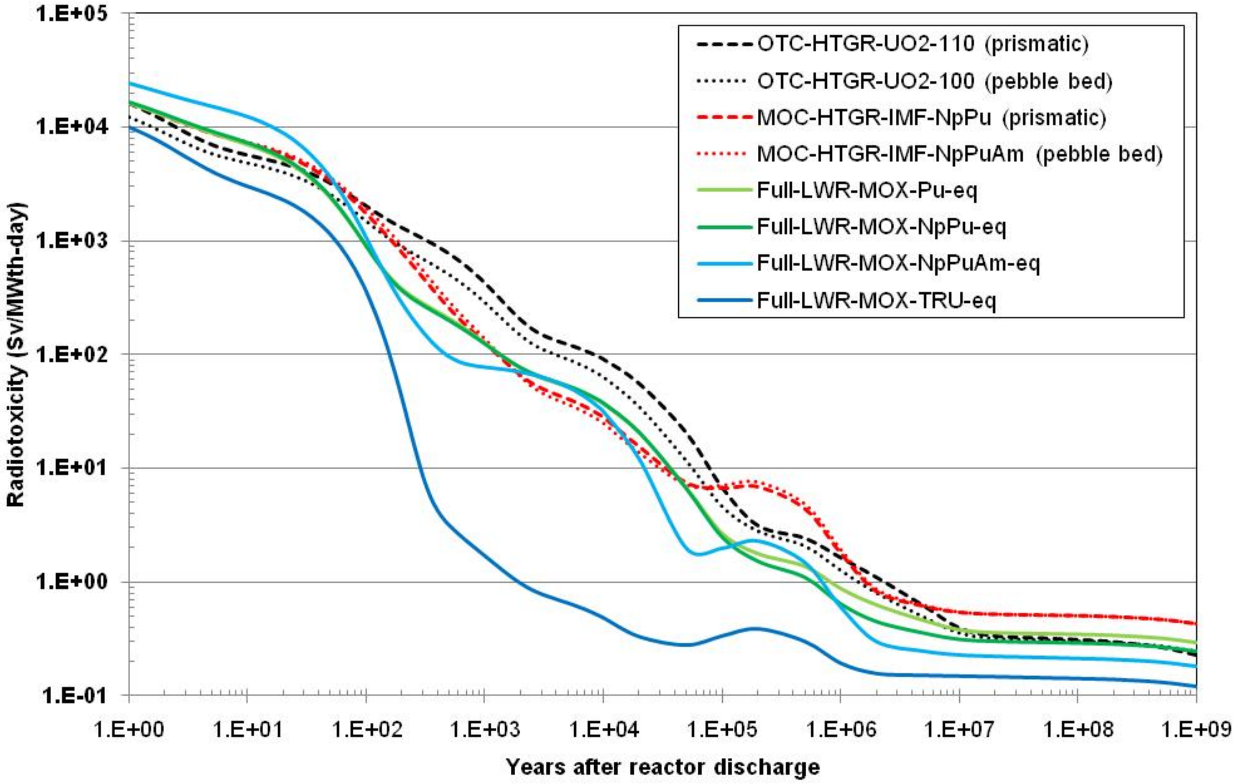


Figure 4-6. Radiotoxicity for a once-through HTGR cases, 1-recycle HTGR-IMF fuels, and full recycle LWR-MOX fuels.

4.1.3 Heat, Photon Energy, Neutron Emission

This subsection shows the degree of commonality among radiotoxicity, heat, photon energy, and neutron emissions of discarded used fuel, HLW, stored, depending on the fuel cycle. The discarded HLW is assumed to contain 0.1% of the TRU in that recycled fuel The same set of diverse cases is used in the next four figures, Figure 4-7 to Figure 4-10.

Figure 4-8 shows heat emissions from the waste for the same fuel cycle cases. The trends are very similar to radiotoxicity.

Figure 4-9 shows photon energy from the waste for the same fuel cycles. The sustained recycle fast burner reactor and LWR-MOX are now closer in this respect to the other cases with only the fast breeder reactor significantly different - and then only after 100,000 years. Basically, the isotopic mix among the actinides does not matter much for this parameter; gamma emitters are scattered throughout the actinide chain. Future work will translate these results into a more meaningful measure of gamma dose to a human at, say, a 1-meter distance.

Figure 4-10 shows neutron emissions from the waste for many of the same fuel cycles. One has to be careful, however. Many calculations do not include the impact of including the higher Cm, Bk, and Cf isotopes in the input feed to the equilibrium fuel cycle reactor physics calculations. Neutron emission is heavily dominated by several isotopes at the top of the actinide chain. So, calculations that are truncated at Cm246 (for example) are not usable for calculating neutron emission and are not included here. Figure 4-11 shows the advantage in some fuel cycles that minimize production of isotopes at the high end of the actinide chain - most notably the fast breeder reactor (CR=1.43) and some of the other fast reactors.

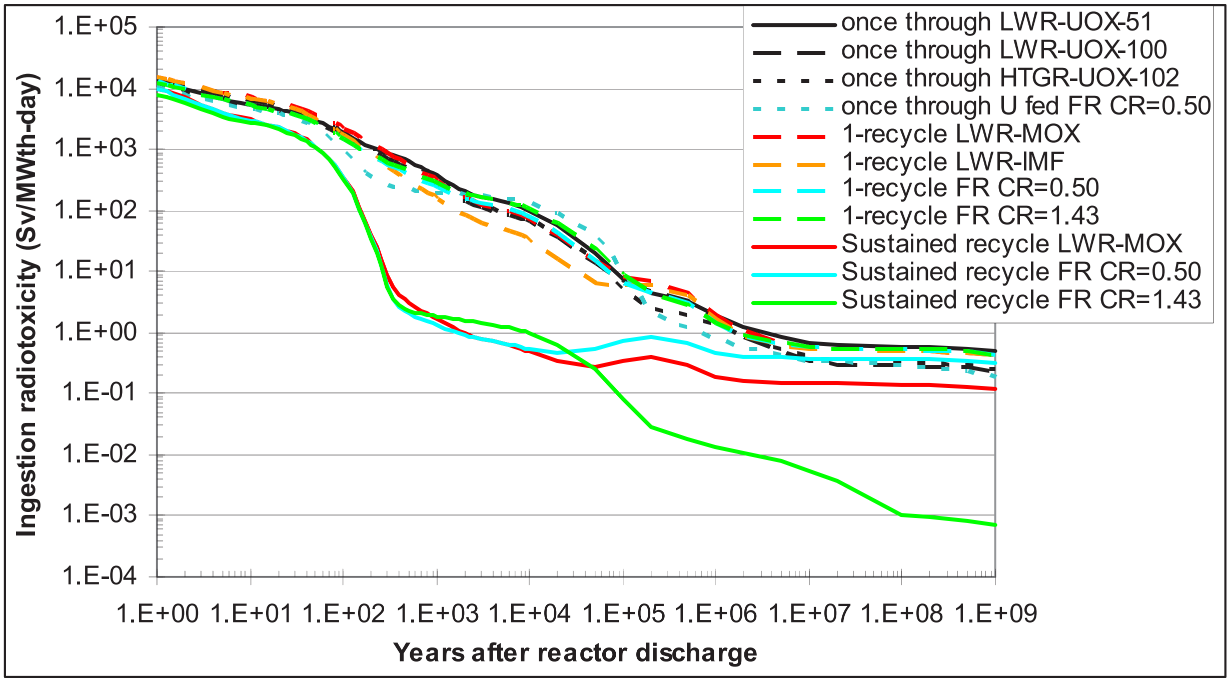


Figure 4-7. Radiotoxicity for a wide range of fuel cycles

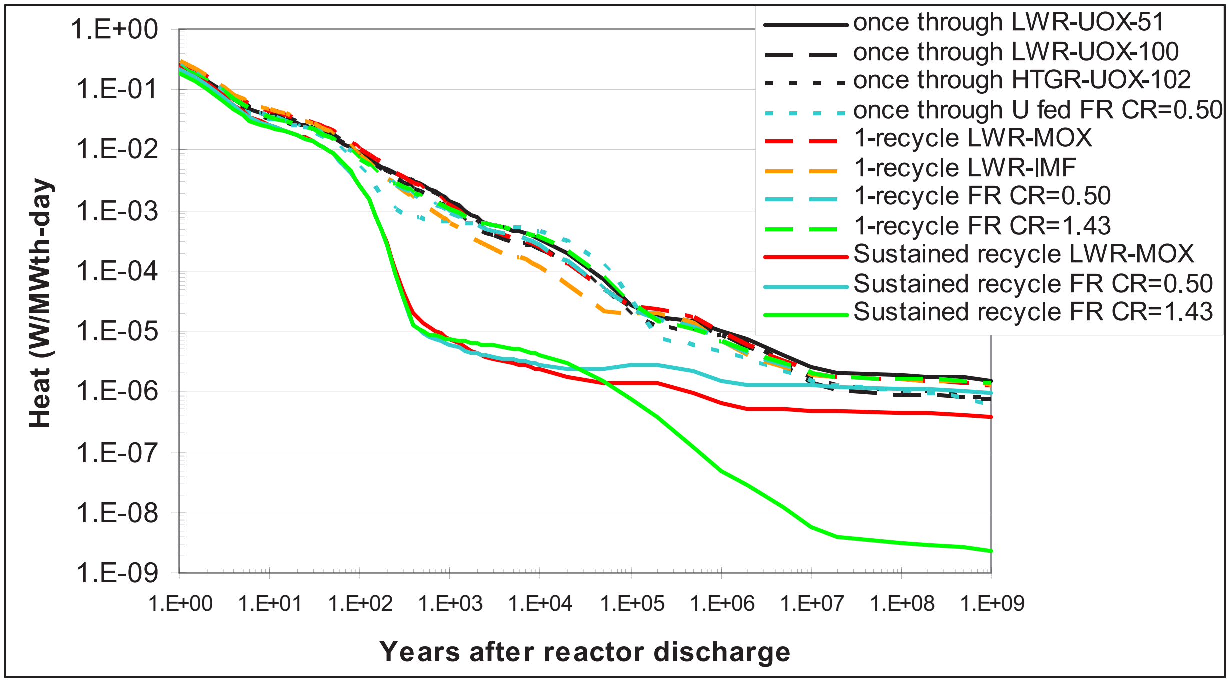


Figure 4-8. Heat for a wide range of fuel cycles

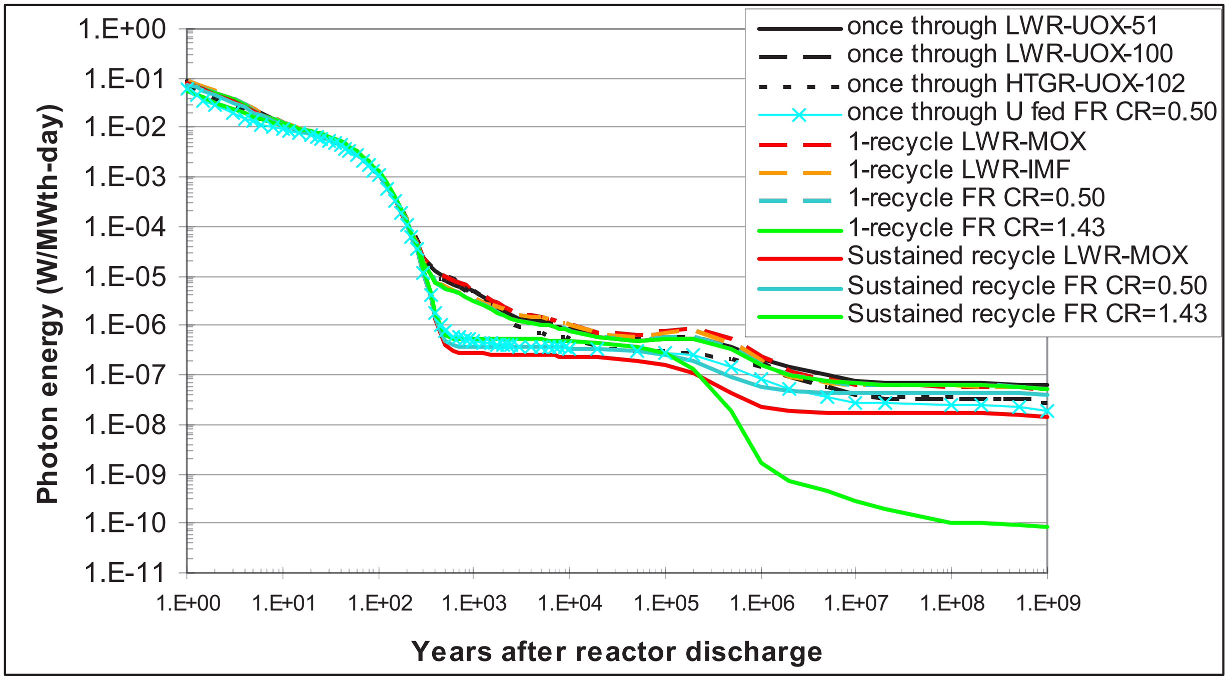


Figure 4-9. Photon energy for a wide range of fuel cycles

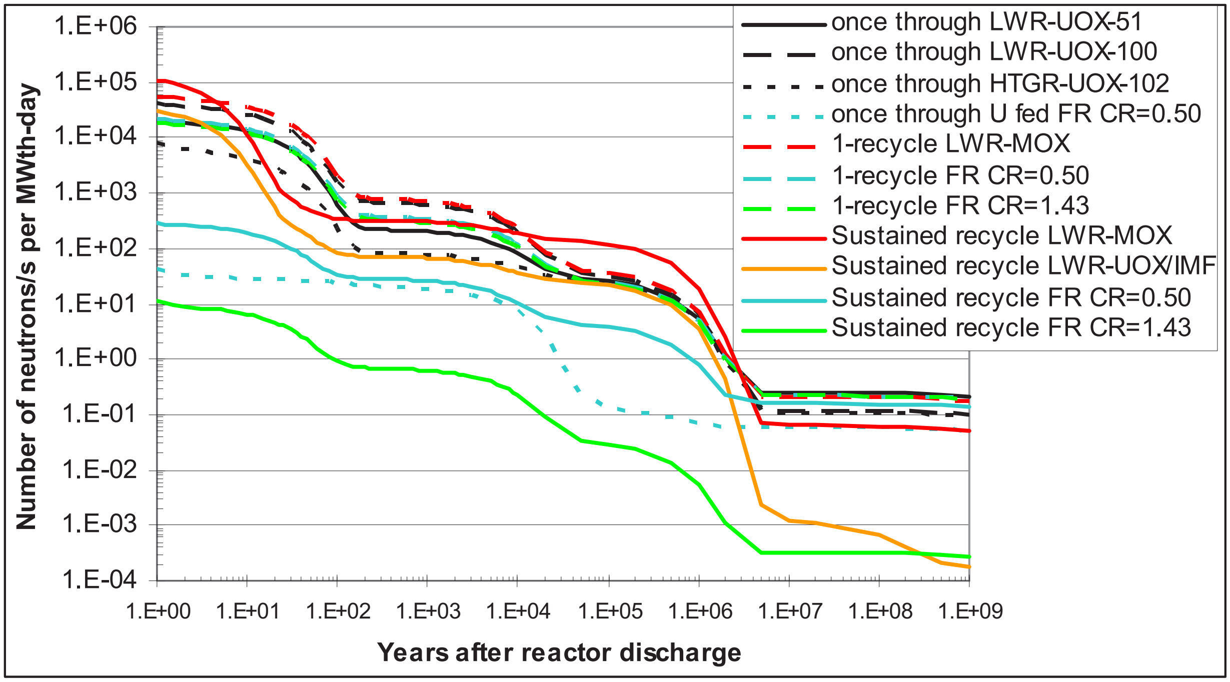


Figure 4-10. Neutron emission for a wide range of fuel cycles

4.1.4 Radiotoxicity at 1,000 Years and HLW Mass

Another waste management consideration is the mass of HLW. Of course, by itself, mass does not measure hazard or burden to future generations. It is a factor in handling, transportation, disposal site engineering, and public perception.

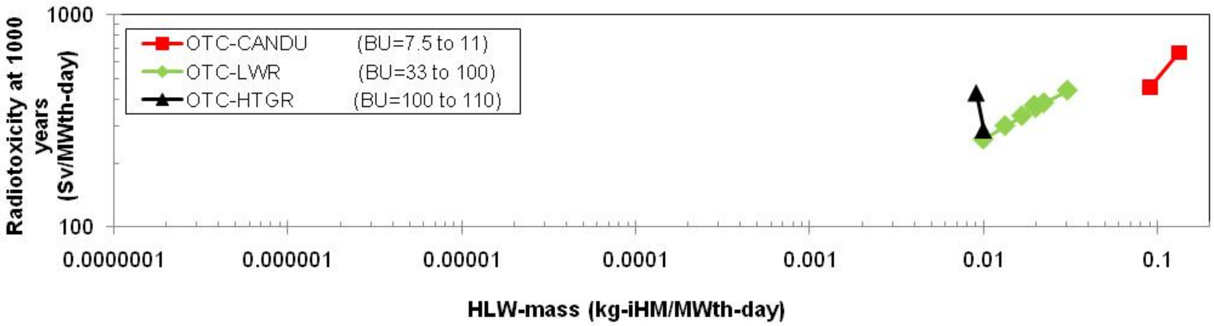
The units in measuring HLW mass are important to understand. In this report, we have used only the most basic units, the mass of initial-heavy-metal (iHM).[i] No mass of material added when the waste form is fabricated is included – the values discussed here are just the fission products and any TRU isotopes that are not recycled. The analysis of the mass of waste forms for separated HLW is a separate level of analysis, not included in this discussion.

The minimum amount of TRU isotopes that are not recycled is the amount of incompletely separated TRU isotopes (assumed to be 0.1% of the total TRU isotopes in the used fuel) that would partition with the fission products in a theoretical full separations process. In this unit, the mass of fission products alone is always about 0.001 kg-iHM/MWth-day, as the energy produced by complete fission of fissionable material is 940-980 MWth-day/kg, depending on the mix of isotopes fissioning.[j] A burnup of 51 MWth-day/kg-iHM therefore means that about 5.3% fractional consumption of the initial heavy metal has occurred, which matches the 5.3% mass percentage of fission products in used UOX-51 fuel.

For a half-century, an international possibility for fuel cycles has been the reduction in the radiotoxicity of the fission waste relative to natural uranium ore so that used fission waste is less hazardous than natural ore in the time period of human engineered structures (~1000 years). Therefore, the following graphs use the radiotoxicity at 1000 years. As shown later, the patterns are unchanged if this is changed to 10,000 years.

In the following graphs, points of the same type of option are linked by a line to better show patterns, e.g., in the first figure, all LWR-UOX points are connected by a green line, all HTGR-UO2 points are connected by a black line.

Figure 4-11 shows the radiotoxicity versus HLW-mass for a wide range of once-through cases. As burnup increases from 7.5 to 148 MWth-day/kg-iHM across a range of reactor types, HLW decreases (scaling as 1/BU) and radiotoxicity decreases slightly. Among HTGR cases, the pebble bed has somewhat lower radiotoxicity and lower BU (hence higher HLW mass), but this may not be significant among the HTGR cases. Higher burnup means lower HLW-mass and slightly lower radiotoxicity at 1000 years.



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[i] This is equivalent to the mass of fission products or other waste streams without any of the waste form or packaging material.

[j] The energy per fission is about 200 MeV, but this varies slightly among isotopes. When converting this to energy/mass fissioned, there is additionally the factor of the atomic mass of the isotope being fissioned.

Figure 4-11. Radiotoxicity and HLW mass (tonnes-iHM) for once through thermal reactor cases; higher burnup always moves points to the left

Figure 4-12 shows two sets of MOC cases. The cases with open symbols use the assumption that all fission products separated in the recycle of the used UOX-51 fuel are HLW. The cases with orange-filled symbols use the assumption that separated fission products are not HLW but that the 0.1%/recycle processing losses of TRU are HLW. The red line has 4 CANDU cases involving 1-recycle of material from used UOX-51: recycle of RU, RU+TRU (relatively high), RU+Pu, DUPIC (relatively low). The green and black cases are IMF from LWR and HTGR, these overlap.

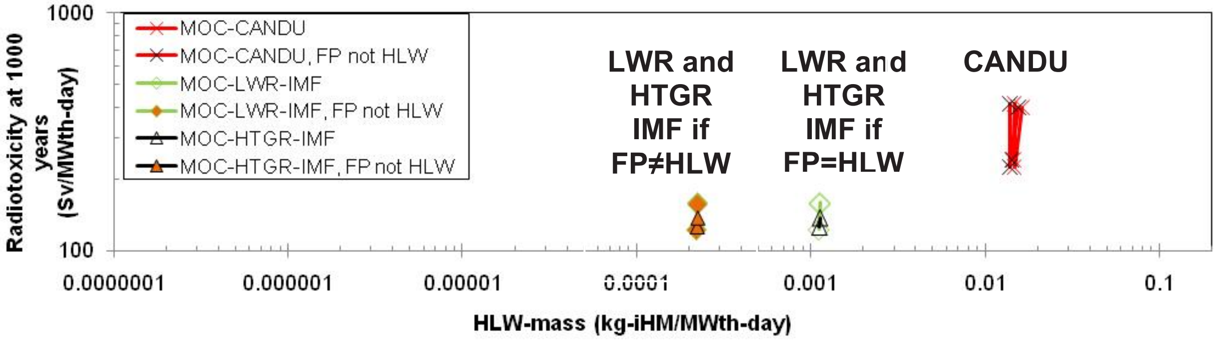

Figure 4-12. Radiotoxicity and HLW mass (tonnes-iHM) for MOC cases; MOC-LWR-IMF (diamonds) overlaps MOC-HTGR-IMF (triangles)

HLW-mass reduction in the units here (initial heavy metal) versus once through depends on four factors. The first three factors are burnup (as with once through), support ratio, and transuranic conversion ratio, which are interrelated. The support ratio is the mass of LWR-UOX divided by the mass of the MOC fuel (normalized by MWth-day). As the support ratio increases, the TRU is therefore concentrated in less mass. For example, MOX has TRU content of ~10%, a support ratio up to 8 relative to the 1.3% TRU in LWR-UOX. IMF has TRU content of ~100%, a support ratio up to 80 and therefore the HLW mass is lower. Similarly, in typical burnup units of MWth-day/kg-iHM, the BU of IMF is ~600 versus BU of MOX at ~50. These examples illustrate how high BU, high support ratio, and low CR correlate.

LWR and HTGR generally perform the same as the LWR-IMF and HTGR-IMF points overlap.

The radiotoxicity decreases slightly as support ratio increases and CR decreases.

The fourth factor is the FP waste mgt strategy. If all FP are HLW, the lowest HLW mass is ~0.001 tonnes-iHM/MWth-day, which corresponds to the mass of FP per fission. The diagram shows the opposite case in which no FP are HLW; HLW is the direct disposal of used fuel (after the one recycle) plus process losses. The leftward movement is greatest for the lowest CR points: the HLW for LWR and HTGR-IMF (CR=0) are reduced by a factor of 5. (open symbols → dark orange symbols). But, the CANDU points (CR >0) are not noticeably reduced. One of the reasons for the higher HLW reduction (for FP≠HLW) for lower CR is that there is less processing of TRU, i.e., more TRU is consumed each time through the reactor.

Figure 4-13 adds full recycle LWR-MOX cases, connected by green lines. Again, the open symbol points are for FP=HLW and the orange symbols are FP≠HLW. As shown in the previous radiotoxicity graphs, recycle of Pu, NpPu, or NpPuAm does not greatly reduce the radiotoxicity relative to MOC cases with IMF, these are the first three points in each of the green lines. Only recycling all TRU reduces the radiotoxicity by orders of magnitude, as shown by the lowest points on the green lines, which are in the region of radiotoxicity for all all-TRU, full-recycle cases. The last point is all-TRU.

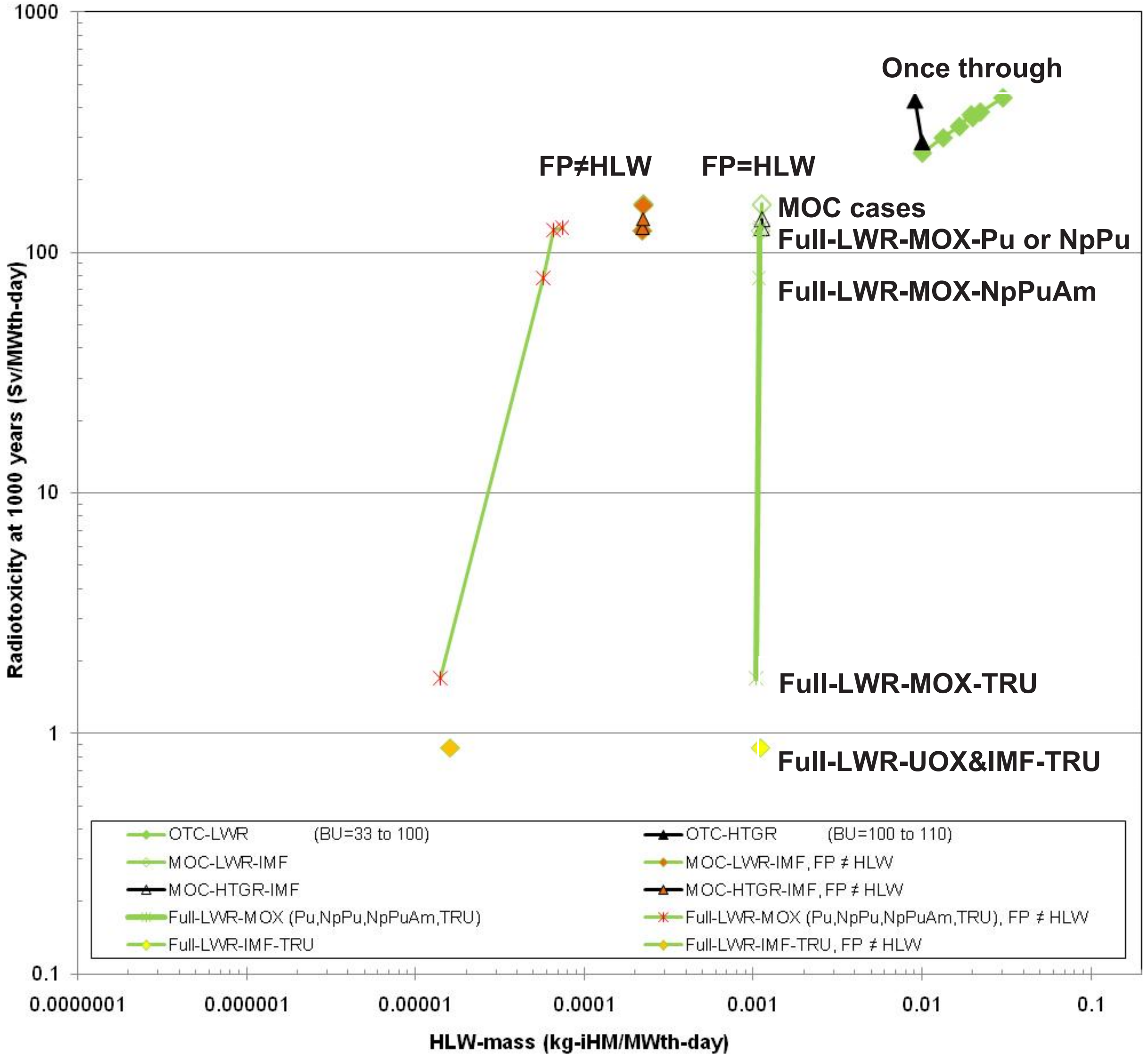

Figure 4-13. Radiotoxicity and HLW mass (tonnes-iHM) for thermal reactor cases in which FP are or are not HLW

The two unconnected points (one with FP=HLW and one with FP≠HLW) at the bottom of this figure are full recycle heterogeneous LWR-UOX&IMF-TRU, in which a heterogeneous mix of UOX pins and IMF pins is used instead of MOX. They exhibit somewhat lower radiotoxicity than their MOX analogs; however, they are somewhat academic because the recycling TRU is in balance without any new TRU coming from previous UOX-51. That is, the net feed is solely enriched uranium and no additional used UOX-51. Thus, although those cases can exist in steady state, it is not obvious how one builds toward it. Nonetheless, it shows the potential higher performance (in the sense of radiotoxicity reduction) of UOX&IMF versus MOX, presumably true for both LWR and HTGR.

4.1.5 Radiotoxicity at 10,000 Years and HLW Mass

Figure 4-14 is the same as previous Figure 4-13 except the radiotoxicity is at 10,000 years instead of 1,000 years. The points have all moved down, as the radiotoxicity decreases with aging from 1,000 to 10,000 years, but the patterns remain the same.

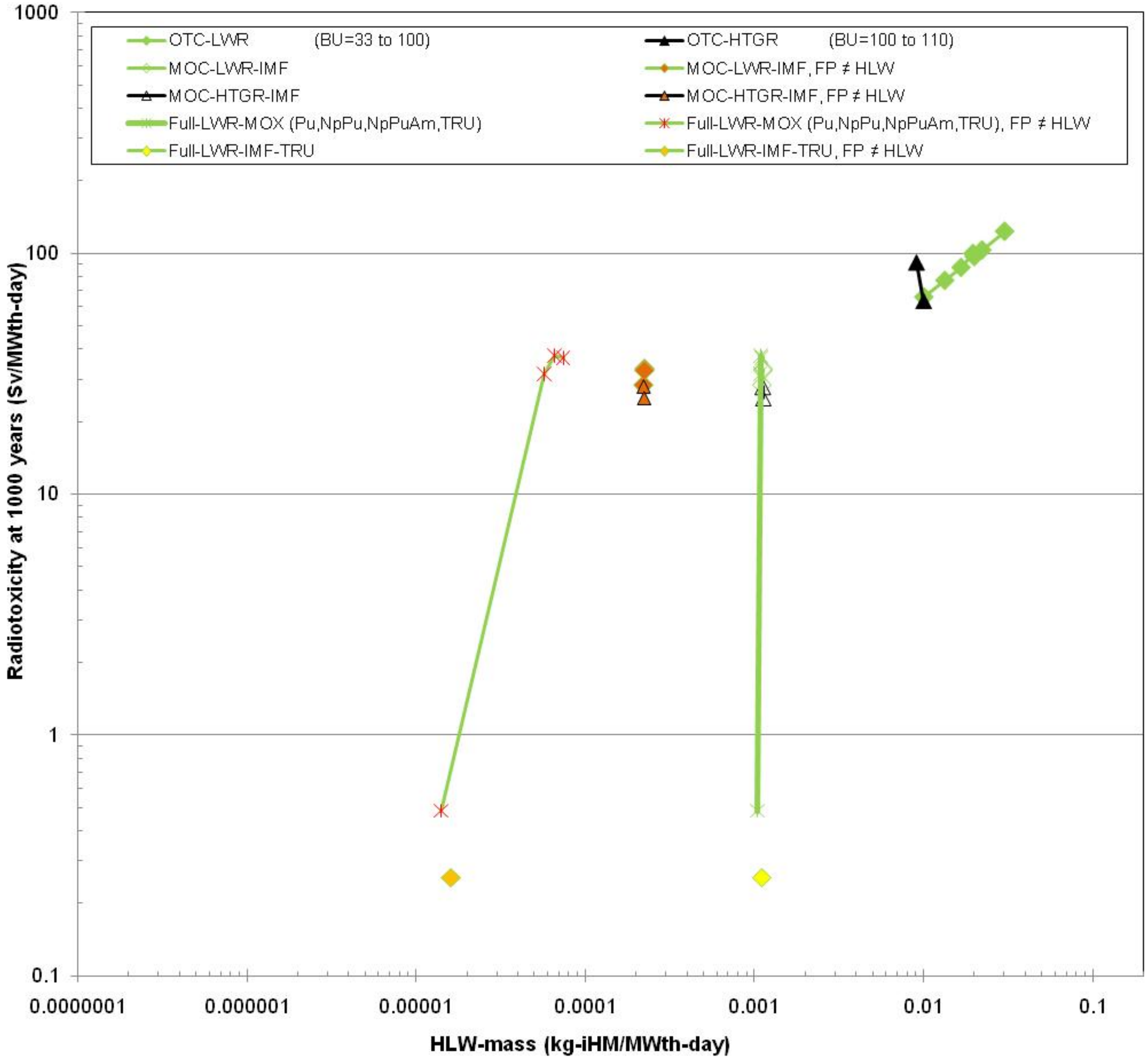

Figure 4-14. Radiotoxicity at 10,000 years and HLW mass; HLW mass cannot go below 0.001 tonnes-iHM/MWth-day unless some of the fission products are separated so that they are not HLW

4.1.6 Radiotoxicity at 100,000 Years and HLW Mass

Figure 4-15 is the same as previous Figures 4-13 and 4-14 except the radiotoxicity is at 100,000 years. The points have all moved down, as the radiotoxicity decreases with aging from 1,000 to 100,000 years, but the patterns remain the same – with the important difference that the vertical scale has shrunk from 4 orders of magnitude to 2, i.e., the rank orderings are about the same, but the absolute range among concepts have decreased considerably.

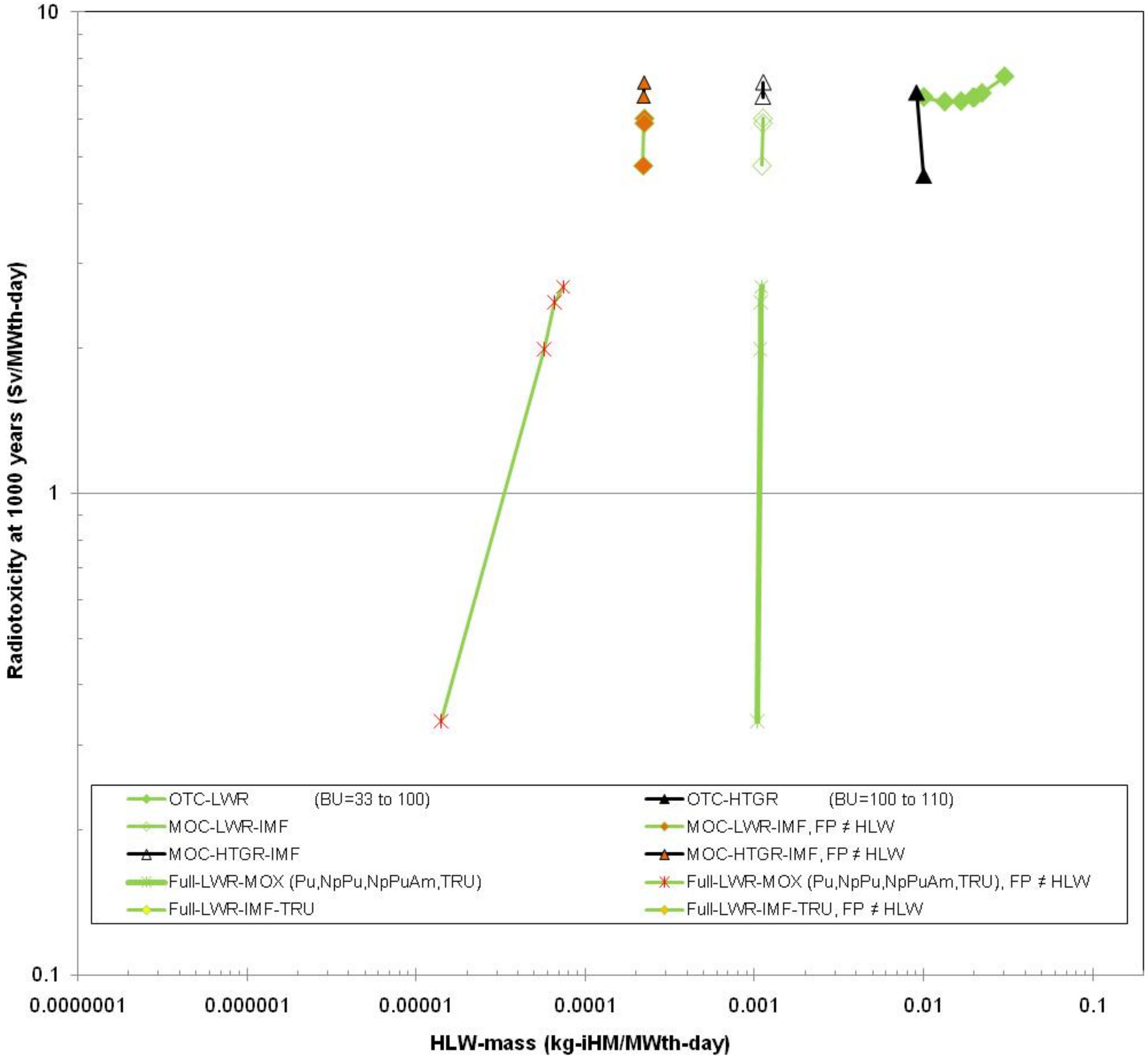

Figure 4-15. Radiotoxicity at 100,000 years and HLW mass; HLW mass cannot go below 0.001 tonnes-iHM/MWth-day unless some of the fission products are separated so that they are not HLW

4.1.7 Radiotoxicity at 1,000,000 Years and HLW Mass

Figure 4-16 is the same as previous Figures 4-13, 4-14, and 4-15 except the radiotoxicity is at 1,000,000 years. The patterns are about the same as for 100,000 years, except there is more difference between the MOX-Pu and MOX-NpPu. The MOX-NpPu points have moved toward the MOX-NpPuAm.

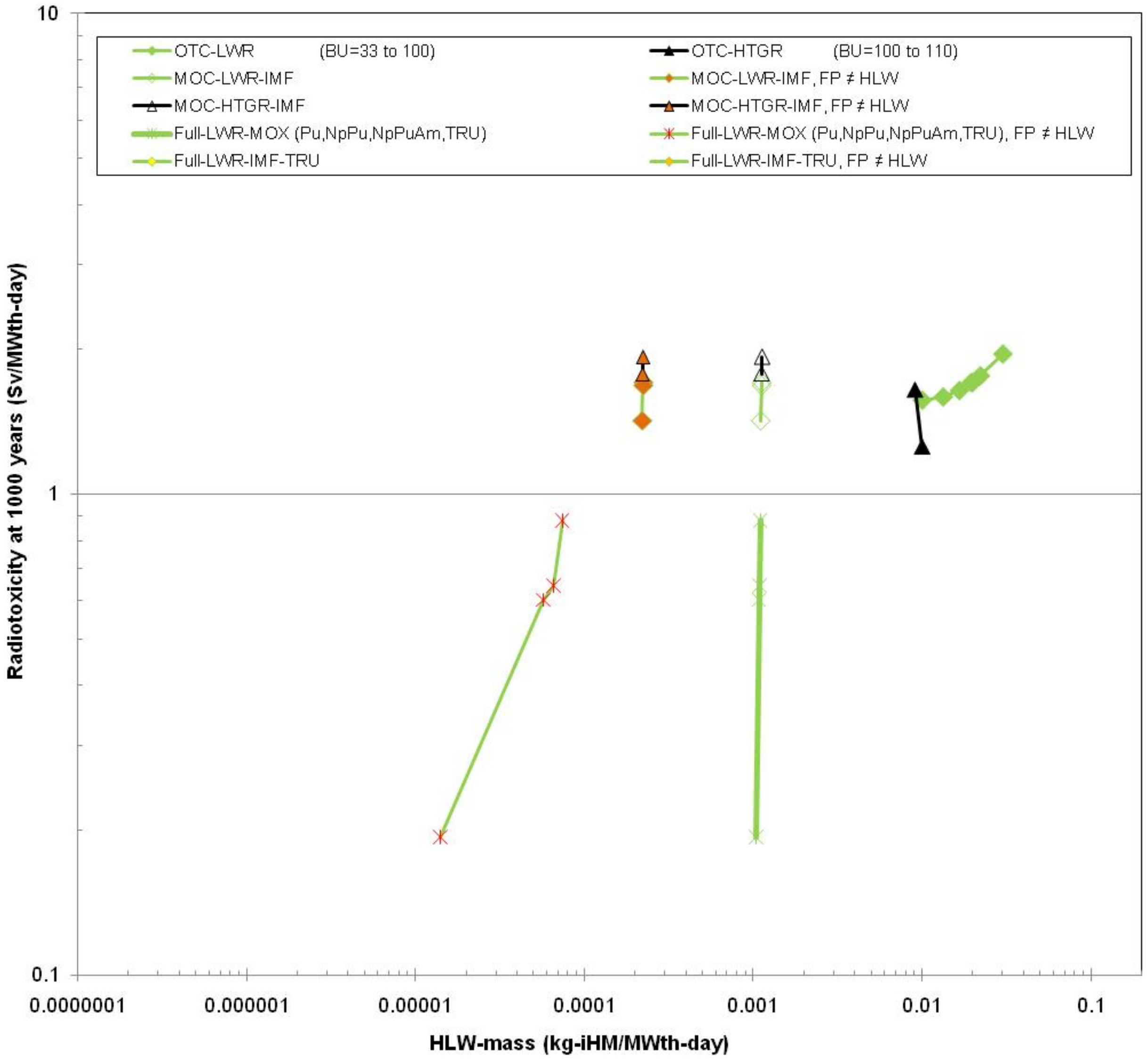

Figure 4-16. Radiotoxicity at 1,000,000 years and HLW mass; HLW mass cannot go below 0.001 tonnes-iHM/MWth-day unless some of the fission products are separated so that they are not HLW

4.1.8 Waste Management Summary

In terms of the amounts of TRU and fission products in the used fuel on an iHM basis, the waste streams for HTGR are basically the same as for analogous LWR cases with similar burnups. For once-through direct disposal of used fuel cases, the total mass of used fuel will be 1-2 orders of magnitude higher than analogous LWR cases because of the carbon that cannot be separated from the fuel meat without some form of separations; however, available information suggests it will be a more robust waste form.

Tables 4-1 lists potential waste streams from HTGR direct disposal or recycling. This list is generally consistent with current reference and optional waste forms recommended by the 2010 Waste Form Campaign.[Vienna2010]

Table 4-1. Waste Information for the HTGR Technology Family

Potential waste streams	Potential waste forms	Remarks
Used fuel (pebbles)	Direct disposal inside pebble coatings and with graphite not easily separated from the pebbles	Without pebble breakage, the pebble coatings can help isolate the spent fuel from the storage and disposal environments. The mass and volume of coatings and graphite will result in orders of magnitude greater mass and volume of the waste compared to the mass and volume of the spent fuel itself.
Tritium	Captured tritium contained in tritiated water molecules, grouted in a Ca-Al-Si cement	If capture of H3 is not necessary to meet air emission limits, then environmental release is an option. Another option is capture of H3-laden H_2O followed by decay storage and release to the environment.
C14	If captured, C14-laden CO_2 molecules are grouted in a Ca-Al-Si cement	If capture of C14 is not necessary to meet air emission limits then gas species containing C14 can be released to the environment and this waste form may be eliminated
Noble Gases	If captured, contain radioactive Kr85 (with or without co-captured stable Xe and Kr) in high pressure cylinder	If capture of H3 is not necessary to meet air emission limits, then environmental release is an option.
Halogens	Captured I129 and other co-captured halogens on silver mordenite, immobilized in grout or Si-glass	I129 capture with efficiencies up to 99.99% may be necessary to meet air emission limits.[Soelberg2009]
Group 1A/2A	Cs/Sr ceramic waste form or combine with other FP in a Si-glass or glass-ceramic WF	This waste stream is not separated from other fission product waste streams in some recycle options.
Lanthanides	Lanthanide glass or combine with other FP in a glass or Si-glass-ceramic WF	This waste stream is not separated from other fission product waste streams in some recycle options.
Transition metals	Metal alloy (when metal fission products are chemically reduced in a metal alloy with other metals) or combine with other FP in a Si-glass or glass-ceramic WF	This waste stream is not separated from other fission product waste streams in some recycle options. Even if this waste stream is separated, there may be little other metal to combine into a metal alloy waste form.
Undissolved solids (UDS)	Si-glass if combined with other FP; or metal alloy if combined with transition metals and other metals	This waste stream is not separated from other fission product waste streams in some recycle options.
Non-fuel materials - C/SiC coatings	Recycle or direct dispose amounts separated from used fuel prior to thermal or other separations processes.	Petti2009. Studies are in progress to determine how to recycle these materials.
Non-fuel materials - core internals	Recycle or direct dispose amounts separated from used fuel prior to thermal or other separations processes.	Petti2009. Studies are in progress to determine how to recycle these materials.

Table 4-2 provides a waste management assessment of the illustrative cases as requested by the authors of the Option Study. Details and references that support this table are provided in Section 3.

Table 4-2. Waste Assessment

<table>
<tr><th>Comparison Measures</th><th>Once through HTGR UOX prismatic</th><th>Once through HTGR UOX pebble bed</th><th>MOC (1-recycle) HTGR IMF (deepburn) prismatic</th><th>MOC (1-recycle) HTGR IMF (deepburn) pebble bed</th><th>Full recycle HTGR (prismatic or pebble bed)</th></tr>
<tr><td>Estimated Peak Dose Rate</td><td colspan="5">Should be improved due to robust TRISO coating</td></tr>
<tr><td>Radiotoxicity of high level waste and unprocessed used fuel</td><td>Similar to LWR-UOX-51</td><td>Nominal to LWR-UOX-51, improved vs. LWR-UOX-51</td><td colspan="2">Similar to LWR-IMF</td><td>Should be less than LWR</td></tr>
<tr><td>Unprocessed used fuel mass or HLW-mass (tonnes-iHM)</td><td colspan="2">Similar to LWR-UOX</td><td colspan="2">Similar to LWR-IMF</td><td>None, by definition</td></tr>
<tr><td>Unprocessed used fuel mass or HLW-mass (tonnes of total material)</td><td colspan="2">1-2 orders of magnitude higher because of the carbon that accompanies the heavy metal in unprocessed used fuel</td><td colspan="2">Nominal to 2 orders of magnitude higher depending on degree of separation</td><td>None, by definition</td></tr>
<tr><td>HLW-mass (tonnes of total material)</td><td colspan="2">None, by definition</td><td colspan="3" rowspan="2">The carbon must either be recycled or it significantly adds to the waste burden. If waste, how much of it would be HLW vs. LLW is not clear.</td></tr>
<tr><td>LLW Mass</td><td colspan="2">LLW during reactor operation will be significantly lower (improved) vs. LWR because the coolant is clean of radioactive corrosion products.
LLW generation during used fuel separations and waste disposal is unknown.</td></tr>
<tr><td>Heat load</td><td colspan="5">Similar to analogous LWR cases</td></tr>
</table>

4.2 Used Fuel Storage

The basic objective is to “efficiently manage the safe and secure storage of used nuclear fuel until disposition pathways are established.”[Dixon2009a] The recommended metric is “the packaged volume of material to be stored, normalized per unit of energy produced.” [Dixon2009a] The actual amount of material to be stored depends on when a repository is opened, when new reactor types and fuel cycle facilities are deployed, etc. Such factors are behind the scope of this report.

Instead, this report considers “used fuel storage” only on a per energy output basis, see Table 4-3. When the parameters of mass, heat, and so forth are expressed “per kg-iHM”, then HTGR and LWR are about the same. When expressed as “per kg-total” – meaning the carbon/graphite is included, then the HTGR is close to 2 orders of magnitude different than the LWR. For mass, the HTGR is higher. This means that the heat density is correspondingly lower. The lower heat density leads to the expectation that used

HTGR fuel would be stored dry, whereas LWR fuel must be stored wet when it is discharged from a reactor. HTGR storage would thus seem to be more stable chemically.

Table 4-3. Used Fuel Storage Assessment; "Nominal" Interpreted as the Analogous LWR Result

Comparison Measures	Once through HTGR UOX prismatic	Once through HTGR UOX pebble bed	MOC (1-recycle) HTGR IMF (deepburn) prismatic	MOC (1-recycle) HTGR IMF (deepburn) pebble bed	Full recycle HTGR (prismatic or pebble bed)
Interim storage effect on fuel stability	Waste mass (initial heavy metal) – about the same as LWRs Waste mass (total mass of fuel assemblies before disposition or recycling) – up to 2 orders of magnitude higher than LWRs Waste heat (W/kg-iHM) – about the same as LWRs Waste heat (W/kg-total) – up to 2 orders of magnitude lower than LWRs Storage environment – better (dry) than LWR (wet, corrosion an issue)				

4.3 Resource Sustainability

The basic objective is to "enhance energy security by extracting energy recoverable in used nuclear fuel, blankets and enrichment tails, ensuring that heavy metal resources do not become a limiting resource for nuclear energy."[Dixon2009a] The recommended metric is the "quantity of natural heavy metal per unit of energy produced – to be measured on a log scale." [Dixon2009a]

Figure 4-17 shows the uranium utilization versus the rate uranium must be extracted from the ground. This graph allows one to plot any reactor type. Uranium utilization (y axis) is calculated versus the theoretical maximum (corresponding to the complete fission of a mass of heavy metal, equivalently the Q-value of energy release of fission), which can be achieved by full recycle fast breeder reactors and by fission-fusion hybrids completing burning actinides (minus processing losses). The uranium ore extraction rate is the rate uranium ore must be mined. For a once-through fuel cycle, it is the inverse of the burnup (kg-fuel/MWth-day) times the ratio of uranium-ore to uranium-fuel (U-ore/U-fuel).

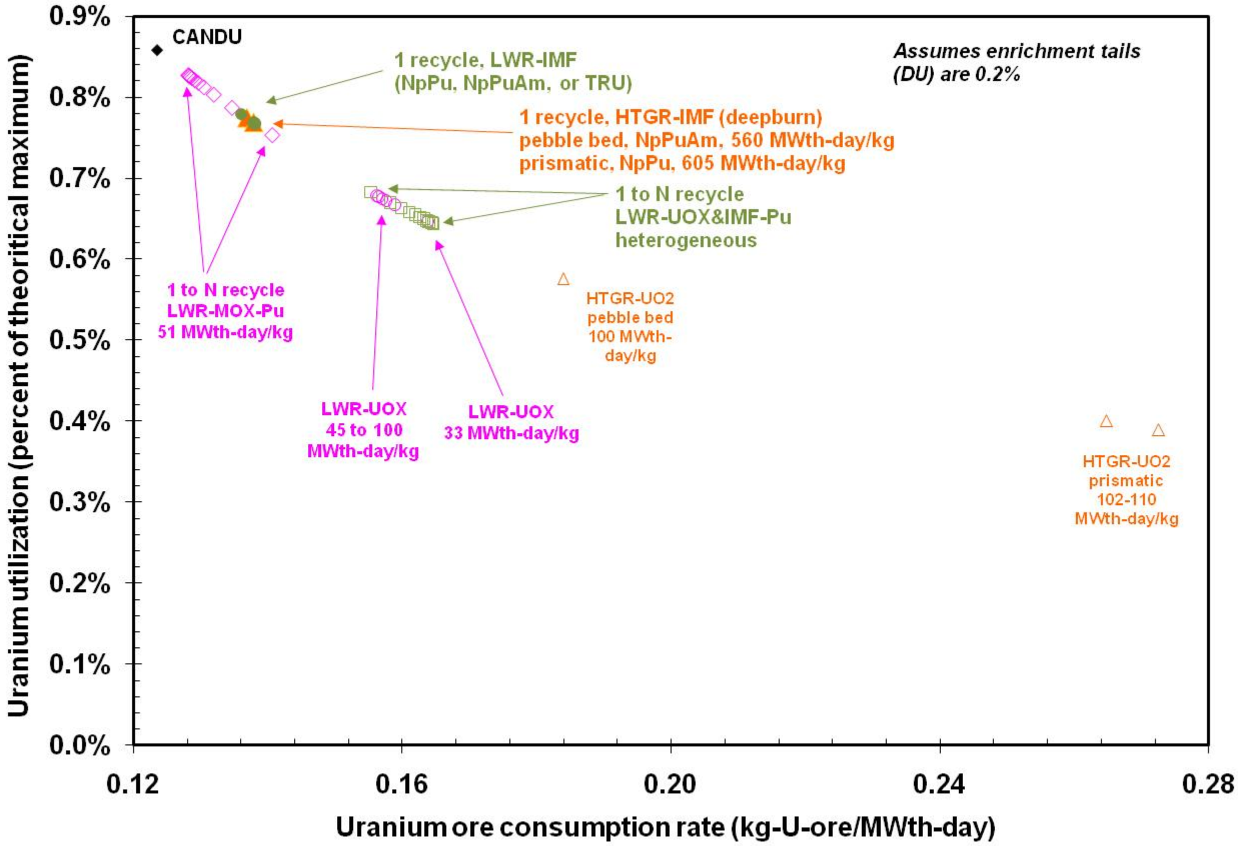

Figure 4-17. Uranium utilization vs. uranium ore - thermal reactor cases - ignoring thermal efficiency

There are various groups of reactors to consider:

- CANDU heavy water cooled and moderated reactors are the most uranium efficient of the thermal reactors, presumably due to the low neutron absorption of heavy hydrogen, which is 3 orders of magnitude lower than hydrogen-1 (appendix A).
- 1 to N recycle of LWR-MOX-Pu, shown are 10 recycles and the equilibrium recycle. By definition, these are MOC unless and until they are taken to equilibrium (infinite number of recycles), at which point they are full recycle.
- 1 to N recycle of LWR-UOX&IMF-Pu (that is, heterogeneous IMF which uses a mixture of UOX pins and IMF pins) , shown are 10 recycles and the equilibrium recycle. By definition, these are MOC unless and until they are taken to equilibrium (infinite number of recycles), at which point they are full recycle.
- 1-recycle LWR-IMF with three cases: NpPu, NpPuAm, all-TRU. These are the most analogous cases to the HTGR-IMF-NpPu/prismatic and HTGR-IMF-NpPuAm/pebble bed cases for which we have data.
- 1-recycle HTGR-IMF-NpPu/prismatic and HTGR-IMF-NpPuAm/pebble bed (deep burn), that is homogeneous IMF that does not have any UOX fuel.
- Once through LWR-UOX with 33 to 100 MWth-day/kg burnup

- Once through HTGR-UO_2

Note that all thermal reactor cases have uranium utilization under 1%. The CANDU case has the highest uranium utilization, consistent with the well-known property of the low neutron absorption of deuterium, see Appendix A. Recycling in thermal reactors improves uranium utilization, but only modestly. The two HTGR-UO2-prismatic cases have unusually low uranium utilization; as explained in chapter 2, this results from neutron spectral differences. The HTGR-UO2-pebble bed case does not appear to suffer as much, presumably because of the efficiencies associated with constant shuffling of fuel.

There is a limitation in the preceding graph; it ignores the differing thermal efficiency among concepts. The GenIV roadmap says "The VHTR can also generate electricity with high efficiency, over 50% at 1000°C, compared with 47% at 850°C in the GTMHR or PBMR. Co-generation of heat and power makes the VHTR an attractive heat source for large industrial complexes." This report is not limited to any specific HTGR concept or design and therefore it is appropriate to consider how the preceding figure changes with an HTGR thermal efficiency of 50% versus LWR efficiency of 33%.

Figure 4-18 is identical with figure 4-17 except that the normalization is changed from GW-thermal to GW-electric, i.e., the impact of thermal efficiency is included. In this graph, LWRs are assumed to have 33% thermal efficiency and HTGRs an upper-bound value of 50%. (CANDUs are 30%.) Thus, figure 4-18 shows a best case HTGR/LWR comparison. The HTGR-UO2-prismatic case is still somewhat below LWR-UOX. The HTGR-UO2-pebble bed and the HTGR-IMF cases are significantly above LWR-UOX.

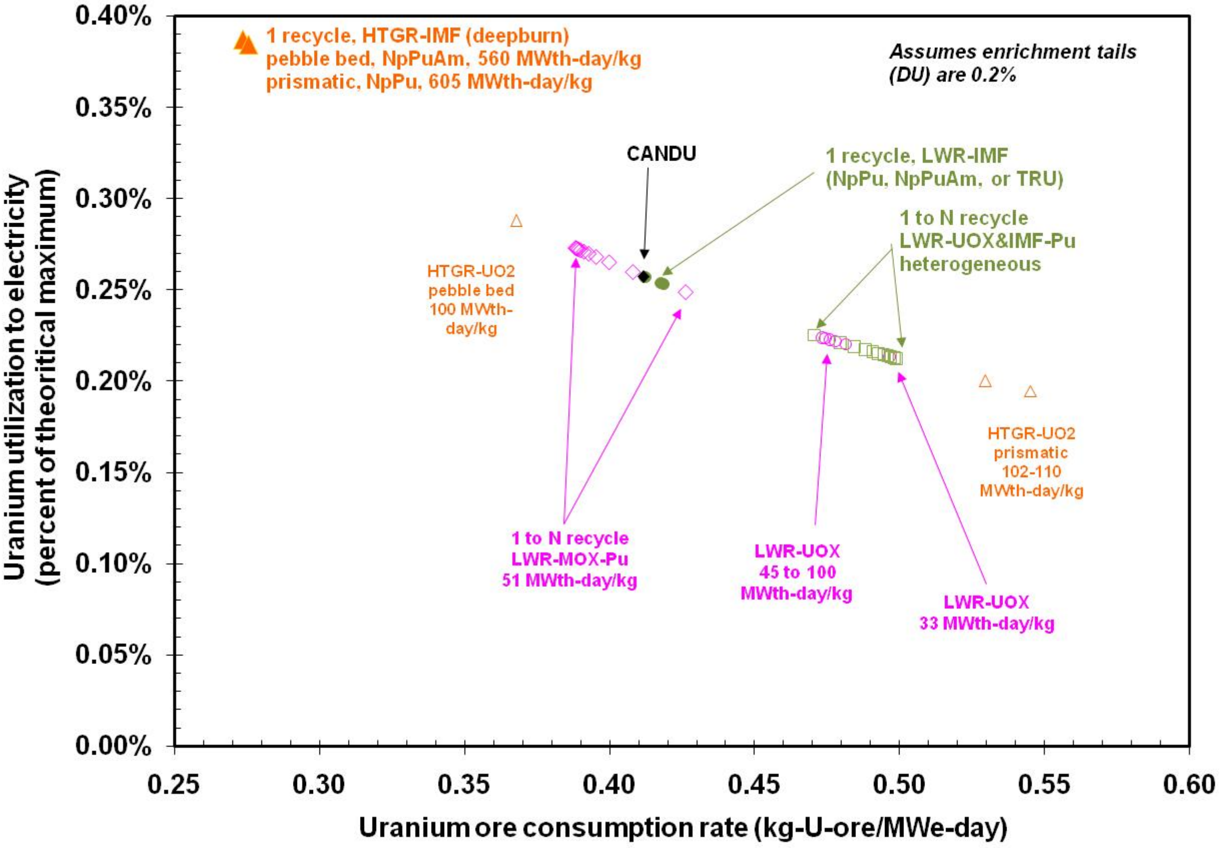

Figure 4-18. Uranium utilization vs. uranium ore for thermal reactor cases – with thermal efficiencies of 33% for LWR, 50% for HTGR, and 30% for CANDU.

Note that the range of uranium utilization for HTGR-UO2 is 0.39 to 0.58%, versus 0.68% for LWR-UOX. However, the thermal efficiencies of the reactor types differ, ~33% for LWR and ~50% for HTGR. If one includes this consideration, the comparison changes. The range of uranium utilization for HTGR-UO2 is 0.19% to 0.29% versus 0.22% for LWR-UOX.

The 1-recycle HTGR-IMF cases have the same thermal uranium utilization as the LWR-IMF cases, about 0.77%. Therefore, when one considers the higher thermal efficiency of the HTGR, the uranium utilization of HTGR-IMF is ~50% higher than for the LWR-IMF.

Table 4-4 summarizes uranium resource assessment.

Table 4-4. Resource Sustainability Assessment; "Nominal" Interpreted as the Analogous LWR Result

Comparison Measures	Once through HTGR UOX prismatic	Once through HTGR UOX pebble bed	MOC (1-recycle) HTGR IMF (deepburn) prismatic	MOC (1-recycle) HTGR IMF (deepburn) pebble bed	Full recycle HTGR (prismatic or pebble bed)
Fuel resources per GWe-day generated	13% worse to 30% better		~50% better		Likely higher than LWR-MOX because of higher thermal efficiency, higher TRU loadings possible

4.4 Proliferation Resistance and Physical Protection

The basic objective is to "enhance overall nuclear fuel cycle proliferation resistance and physical protection via improved technologies for used fuel disposition."[Dixon2009a] The recommended metrics are as follows:

- Net quantity of weapons-usable material produced, normalized per unit of energy produced – the IAEA definition of weapons-usable (HUE and Pu) is extended to include uranium enriched with U-233 above 12% and all minor actinides.
- Inventory of weapons-usable material in the fuel cycle, normalized per unit of energy produced – Attractiveness of materials could be added as a modifier once a tool for the calculation is created
- Quantity of weapons-usable material disposed as waste, normalized per unit of energy produced
- Quantity of enrichment required, normalized per unit of energy produced
- Quantity of separations required, normalized per unit of energy produced
- Qualitative measure of the intrinsic safeguardability of materials."[Dixon2009a]

The lead author of this report participated in mutually-beneficial sessions with proliferation resistance and physical protection experts Robert Bean, Richard Metcalf, Fernando Gouveia, and intern Amanda Rynes. This helped them in a report they are preparing on methodologies and helped the lead author better understand challenges for the range of HTGR options. This activity involved five half-day intense analysis and discussion sessions and review of unpublished documents.

After considering the HTGR and a few other concepts from the perspective of multiple assessment methodologies such as Proliferation Resistance and Physical Protection [PRPP2006], Technical Opportunities to Increase the Proliferation Resistance of Global Civilian Nuclear Power Systems

[TOPS2000], International Project on Innovative Nuclear Reactors and Fuel Cycles [INPRO2003], the following are noted as key differences and issues.

4.4.1 Differences in fuel fabrication and separation technology

Some of the assumptions

- Broken pebbles are detectable, i.e., if someone puts faulty pebbles into an HTGR their faults would cause abnormally high radioactivity levels in the helium coolant, which would normally be very clean.
- The host country would have significant industrial capability but little pre-existing nuclear engineering expertise.

Thus, a key issue is whether fuel fabrication technology would be provided to a host country in addition to the reactor for two reasons.

First, one mis-use scenario is making “fake” fuel with U-238 leading to higher quality Pu than routine uranium or recycled Pu or TRU fuel. This Pu would also be “off the books” and thus not safeguarded. The low budgets for IAEA inspectors mean they cannot watch everything. Detection of “fake” fuel would be difficult prior to insertion. However, if the host country does not obtain TRISO technology and nonetheless attempts to make fake fuel, it would seem highly likely to fail. Given the high purity and low radioactivity of HTGR coolants, such failures of fake fuel might be detectable.

Second, TRISO fuel fabrication technology involves a degree of separation chemistry to make high chemical purity U (or U-TRU or TRU) feedstock. So, having TRISO fuel fabrication technology would mean a country automatically has a certain chemical separation technology and experience in handling nuclear materials.

With regard to separation of used fuel, an existing and obvious way to recover U or TRU from used TRISO fuel is to burn the pebbles (if pebble bed) or fuel elements (if prismatic). This has two problems. First, it should be very easy to detect offsite. Second, without some crushing, one still hasn’t gotten to the fuel meat, which lies inside both carbon layers (which will burn) and a SiC layer.

Table 4-5 shows the judged technical difficulty of some proliferation pathways. Basically, the HTGR would have an advantage versus the LWR because of the increased difficulty of making fuel, especially if fuel fabrication technology is not obtained by the reactor host country.

Table 4-5. Unofficial Judgment of Proliferation Technical Difficulty

		What is in the reactor?		
		U238 Mis-use scenario	LEU fuel	TRU fuel
What do you get for your trouble if you succeed?	Fresh fuel		LEU	RG Pu or TRU
	Used fuel	Better than usual RG-Pu	RG-Pu	DB Pu or TRU
Scenarios for what technologies are provided to host country	Only with reactor	Very hard	Hard	Hard
	Reactor, fuel fabrication, and fuel separation technologies	Medium or low	Easy	Easy

4.4.2 Mass of Material to be Diverted or Stolen

The various methodologies consider the smallest unit of mass that can be diverted or stolen. The higher, the better. Table 4-6 provides the mass of heavy metal and total mass of fuel assemblies (or fuel elements or fuel pebbles). Except for the HTGR pebble bed, all the options have total assembly masses of 120-660 kg. It is unknown whether those are significant differences from a diversion or theft perspective. Even if they were, the values shown in the table were never selected from this perspective. That is, if significant advantage were given to concepts with say 600-kg assemblies versus 100-kg assemblies, it would seem likely that prismatic fuel elements could be increased in size and mass.

The pebble bed approach is an obvious outlier, with 0.2 kg/pebble, each containing only a small mass of heavy metal. This leads to the issue of counting, see next subsection.

Table 4-6. Mass of Fuel Assemblies in Various Options

	kg-iHM/assembly	kg-total/assembly
Once through PWR UO2	461.3	657.9
MOC or Full Recycle LWR MOX	461.3	657.9
MOC or Full Recycle LWR IMF-Pu	37.2	530.9
MOC or Full Recycle LWR IMF-NpPuAm	42.7	580.8
MOC or Full Recycle LWR IMF-TRU	42.5	578.8
Once through BWR UO2	183.3	319.9
	kg-iHM/assembly	kg-total/assembly
Full recycle FR metal CR=0.00	29.7	349.7
Full recycle FR metal CR=0.25	44.9	342.8
Full recycle FR metal CR=0.50	69.9	366.1
Full recycle FR metal CR=0.75	97.7	411.5
Full recycle FR metal CR=1.00	114.1	446.1
Full recycle FR oxide CR=0.00	37.7	264.9
Full recycle FR oxide CR=0.25	60.6	335.6
Full recycle FR oxide CR=0.50	91.9	426.1
Full recycle FR oxide CR=0.75	126.0	483.2
Full recycle FR oxide CR=1.00	148.6	540.0
	kg-iHM/element	kg-total/element
Once through HTGR-UO2 prismatic	7.2	134.4
MOC HTGR-IMF prismatic	1.2	126.9
	kg-iHM/pebble	kg-total/pebble
Once through HTGR-UO2 pebble bed	0.011	0.21
MOC HTGR-IMF pebble bed	0.002	0.20

4.4.3 Item versus Bulk Counting

Safeguards is all about counting. What do you have to count? How well?

Although not “safeguards” *per se*, similar issues exist for theft scenarios. How much material must you steal? As you transport it elsewhere, how much material must you move and how could that be detected.

The pebble bed design approach is outside common safeguard experience. LWRs, fast reactors, and prismatic HTGRs have fuel assembly masses in the range of 120 to 660 kg; HTGR pebble bed have fuel pebbles about 0.2 kg. (Each pebble has ~50,000 TRISO-coated fuel particles.) Thus, LWRs, fast reactors, and prismatic HTGRs have hundreds of fuel assemblies, which are easily counted as individual items. Pebble bed HTGRs have hundreds of thousands of fuel pebbles, e.g., the PBMR-400 design had 450,000 pebbles. These are thought to be counted as bulk material, not individual items. Moreover, the packing density of pebbles varies, making exact balance of the number of pebbles difficult; fortunately, several thousand pebbles are required to make up one "significant quantity" (SQ) of weapon-usable material.

4.4.4 Integral Measurements

LWR safeguard practice now includes integral measurements, monitoring the Cherenkov glow in used fuel ponds. This provides a very gross-level measurement to ensure that there are not a large number of fake fuel assemblies, attempted substitutes for real assemblies that have been diverted.

This approach is unique to water-based storage and is therefore not applicable to other fuels, including HTGR and fast reactor fuels. New techniques would have to be developed.

4.4.5 Summary

Table 4-7 summarizes some of the above observations in the format requested.

Table 4-7. Proliferation Resistance and Physical Security Assessment; "Nominal" Interpreted as the Analogous LWR Result

Comparison Measures	Once through HTGR UOX prismatic	Once through HTGR UOX pebble bed	MOC (1-recycle) HTGR IMF (deepburn) prismatic	MOC (1-recycle) HTGR IMF (deepburn) pebble bed	Full recycle HTGR (prismatic or pebble bed)
Special Nuclear Material Inventory	Nominal (similar to analogous LWR cases)				
Material Attractiveness	Nominal (similar to analogous LWR cases)				
Uranium Enrichment Requirement	Slightly higher than analogous LWR cases because of the lower conversion of U238 to Pu239, see chapter 2.				
Safeguardability	About same as LWR-UOX	Worse than LWR-UOX	About same as LWR-IMF	Worse than LWR-UOX	
Radioactive Material Inventory	Nominal (similar to analogous LWR cases)				
Physical security	About same as LWR-UOX		About same as LWR-IMF		

4.5 Economics

The basic objective is to "maintain an economical nuclear fuel cycle."[Dixon2009a] The recommended topic of technology readiness level is addressed in section 2.2, rather than here, to provide clearer description of the illustrative concepts within the HTGR technology family. Furthermore, it is particularly important to understand how the technology readiness varies as the HTGR technology family is used for various fuel cycle strategies.

There is insufficient information to calculate "fuel cycle costs, normalized per unit of energy produced" or "levelized unit cost of energy" for the illustrative concepts. Instead, the assessment will point to relative cost factors, starting with the six requests from the Option Study authors. To their list, we add in Table 4-6 two additional factors:

- What is the impact of changing gamma dose on fuel/waste form fabrication difficulty/cost? For comparison, the relative impact seems high for LWR oxide fuel and relatively low for metal fuel pins.
- As each impurity element changes relative to the "impurity limit", how does fuel performance and fabrication change?

Table 4-8 provides some qualitative observations on HTGR economics in the fuel cycle context.

Table 4-8. Economics Assessment; "Nominal" Interpreted as the Analogous LWR Result

Comparison Measures	Once through HTGR UOX prismatic	Once through HTGR UOX pebble bed	MOC (1-recycle) HTGR IMF (deepburn) prismatic	MOC (1-recycle) HTGR IMF (deepburn) pebble bed	Full recycle HTGR (prismatic or pebble bed)
Infrastructure Similarity	For electricity: similar types of facilities as LWR HTGRs do, however, have an entirely new market to explore				
Capital	Insufficient data				
Technical Maturity	Less experience				
Technical Risk	Higher risk				
Development Time	Relatively low		Modest		Higher
Life-Cycle Costs	Insufficient data				
Relative penalty to make fuel remotely	Not applicable	Not applicable	Higher		
Relative tolerance of impurities	Insufficient data. The C and SiC layers must be pure of course, but the relative tolerance of impurities in the fuel meat vs. LWR pellets is unclear.				

4.6 Safety

The basic objective is to "maintain excellent safety performance of nuclear fuel cycle facilities and operations."[Dixon2009a] The recommended metric in that study is a "qualitative measure of the relative difficulty of developing licensable facilities." There are three particular measures worth discussing in this study:

- Impact on decay heat in first several hours after reactor shutdown
- Impact on reactor accident source term
- Impact on hazard of separation facilities

4.6.1 Decay heat in first several hours after reactor shutdown

The first of these is unlikely to differentiate between LWR and HTGR, but the deepburn study [Petti2009] found that it does differentiate between HTGR-IMF and HTGR-UOX. The cause is the higher Cm242 inventory due to Am in the fresh fuel. Note that Cm242's halflife is sufficiently short, 163 days, so it does not accumulate so much with multiple recycles, but it is generated from Am242 and Am242m during irradiation. Figure 4-19 shows the decay heat (fraction of nominal operating power) as a function of time.[Petti2009] Cm242 is sufficiently long-lived, however, that it does not reach saturation during reactor operation.

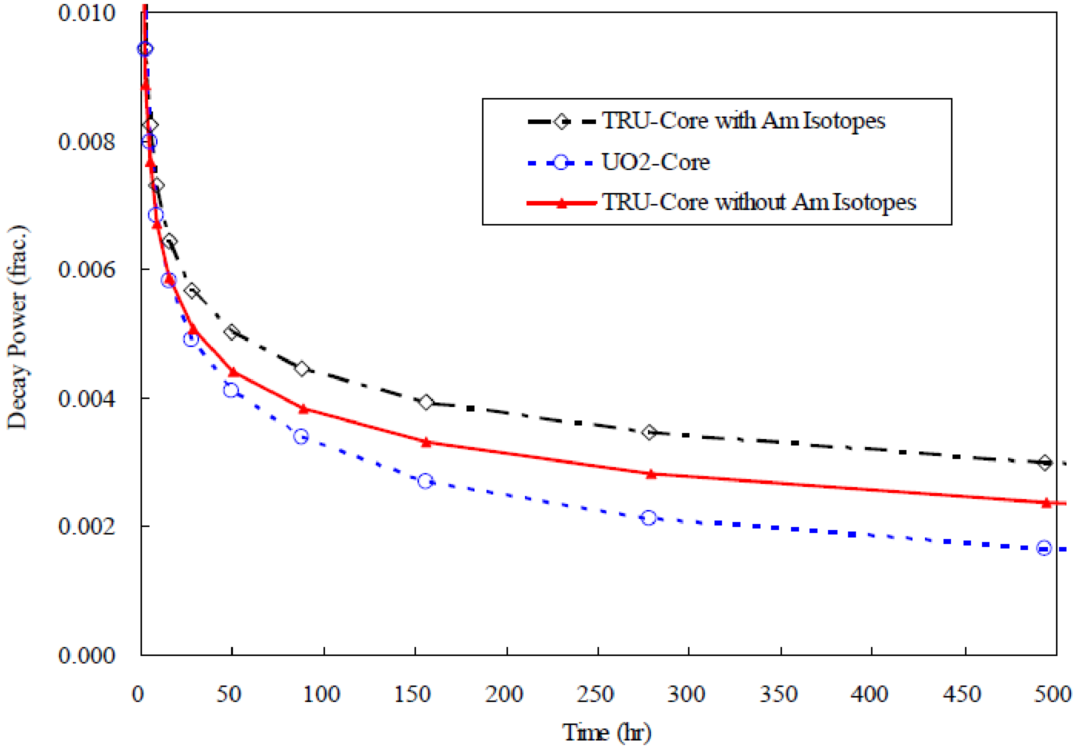

Figure 4-19. Decay power curve of TRU kernel fuel core with and without americium isotopes in fresh fuel [Petti2009, figure 2-90]

To shed more light on the situation, a quick ORIGEN calculation was done by B. Boer. Figure 4-20 shows the decay heat for four HTGR case:

- HTGR-UO_2 at 50 MWth-day/kg-iHM
- HTGR-UO_2 at 90 MWth-day/kg-iHM
- HTGR-IMF-TRU at 300 MWth-day/kg-iHM
- HTGR-IMF-TRU at 600 MWth-day/kg-iHM

The decay heat curve of HTGR-IMF-TRU is similar to that of `standard' UO2 fuel in general except for very high burnup levels. Note that during the first couple of hours the fission products determine the decay entirely. The HTGR-IMF-TRU fuel gives a different spectrum of fission products resulting in a lower decay heat level than UO2 fuel. In the following hours, the contribution of the minor actinides become more important, resulting in a higher decay level for the HTGR-IMF-TRU fuel, which increases with burnup. The figure shows a change in slope at about 2 hours after reactor shutdown. At that point the higher burnup cases for both HTGR-UO_2 and especially HTGR-IMF-TRU increase relative to HTGR-UO_2 at 50 MWth.

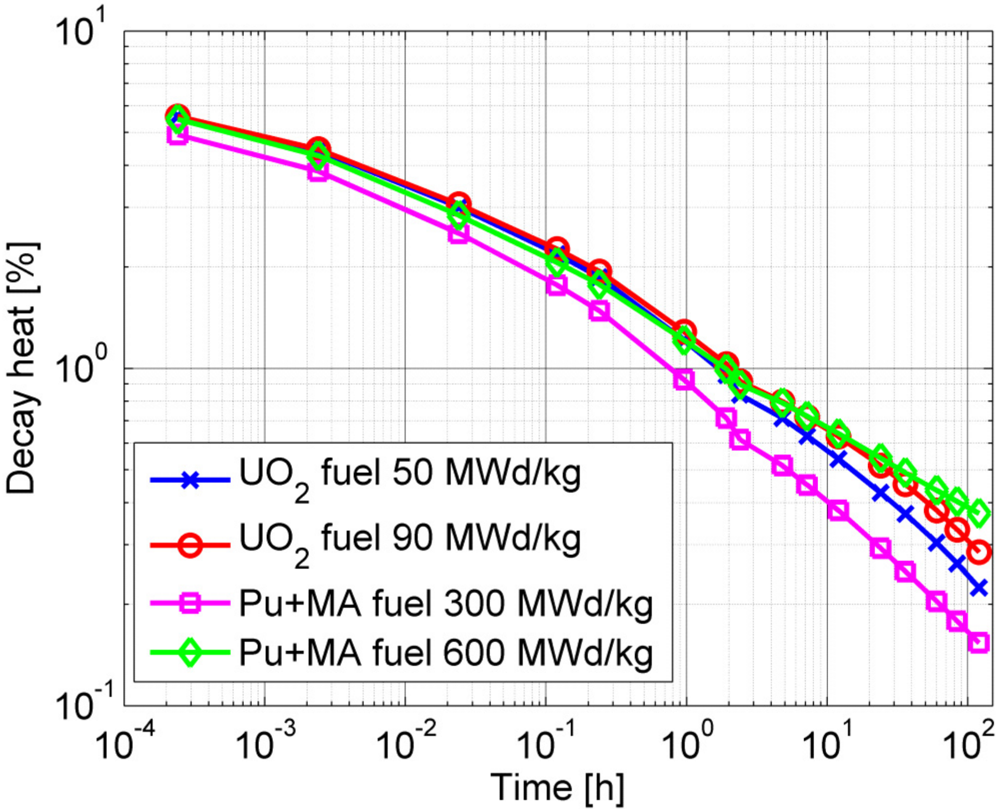

Figure 4-20. Decay power curve of uranium and TRU fuels [Brian Boer, personal communication, 3/2010]

4.6.2 Impact on reactor accident source term

For present purposes, there are two aspects to the reactor accident source term, the inventory of radioactive materials and the fuel coating.

The inventory of radioactive materials includes fission products and actinides. Per thermal energy released by fission, the inventory of fission products is about the same among reactors and fuels. There is one exception – venting of highly volatile fission products in the fuel venting concept (GA EM^2). It is, of course, precisely these isotopes that tend to dominate accident releases – iodine, Xe, Kr. So, fuel venting should lead to reduction in accident source terms – but with the downside of having another release path of material from the fuel to outside of the core.

The inventory of actinides rarely appears in reactor accidents as they have low volatility. In any case, the inventory of actinides is not significantly different between LWR and HTGR of the same fuel cycle strategy.

The TRISO fuel coating is proposed to be a more robust accident containment barrier than LWR fuel cladding. This will have to be shown to regulatory bodies before receiving “credit”.

4.6.3 Impact on hazard of separation facilities

Spills and leaks from existing aqueous fuel cycle facilities have been an issue.[Cadwallader2005] This is one publically-stated reason that General Electric has not included aqueous separation in its fuel cycle proposals. It happens that the three major alternatives to aqueous separation do not involve materials that are liquid at room temperature and therefore there is the potential for reduced hazard.

- AIROX involves no liquids at all.
- Melt refining has liquid metals, but these would freeze if leaked.
- Electrochemical has liquid salts, but these would freeze if leaked.

On the other hand, these three alternatives have higher operating temperatures than aqueous, which must be assessed for accident potential. In particular, melt refining involves liquid metals that can oxidize and must be considered therefore to have more chemical accident potential than aqueous, AIROX, or electrochemical.

4.6.4 Comparative safety assessment

Table 4-9 provides the current comparative safety assessment.

Table 4-9. Safety Assessment Relative to LWR

Comparison Measures	Once through HTGR UOX (prismatic or pebble bed)	MOC (1-recycle) HTGR IMF (deepburn) prismatic	MOC (1-recycle) HTGR IMF (deepburn) pebble bed	Full recycle HTGR (prismatic or pebble bed)
Licensing	Significantly better (passive safety)	Significantly better (passive safety)	Significantly better (passive safety)	Significantly better (passive safety)
Impact on decay heat in first several hours after reactor shutdown	Should be similar to LWR-UOX	Should be similar to LWR-IMF Appears higher than HTGR-UOX	Should be similar to LWR-IMF Appears higher than HTGR-UOX	Should be similar to LWR-MOX
Impact on reactor accident source term	Actinides – nil difference between LWR and HTGR. Fission products – nil difference between LWR and HTGR. Fuel venting options have lower inventories of volatile fission products but another pathway for radioactivity to escape the core into containment			
Impact on hazard of separation facilities	Not applicable	AIROX, melt refining, or electrochemical should have lower spill hazard than aqueous as they do not involve liquids at room temperature. The higher operating temperatures of AIROX, melt refining, and electrochemical must, however, be assessed. In particular, melt refining has liquid metals that can oxidize and therefore have higher chemical accident potential than aqueous, AIROX, or electrochemical.		

5. CONCLUSIONS AND FUTURE WORK

This report examines how the HTGR technology family can provide options ranging from open cycle to closed cycle. The HTGR can serve all the fuel cycle missions that an LWR can. The higher thermal efficiency of the HTGR and the recent fuel burnup test results make the HTGR slightly more uranium efficient than the LWR. The solid moderator in an HTGR (graphite) represents more of a waste management issue than the liquid moderator in an LWR (water). The lack of the void coefficient problem in HTGRs (a known limiting constraint in MOC and full recycle LWR-MOX) suggests MOC and full recycle HTGR-MOX would have higher performance.

HTGRs can consume transuranics faster than a LWR-MOX. The reduction of radiotoxicity in residual waste in either the LWR or HTGR depends not so much on the reactor, but on the number of times used fuel is recycled, whether all TRU or only some TRU isotopes in the used fuel is recycled, and the TRU processing loss rate. A single recycle in either LWR or HTGR does little to reduce long-term radiotoxicity. Full recycle of all the transuranics in any system does reduce long-term radiotoxicity by ~2 orders of magnitude, assuming process loss rates below 1%.

The HTGR is considered for deployment independent of fuel cycle objectives; HTGR fuel cycle analyses are therefore appropriate regardless of perceived fuel cycle performance - to know how HTGR deployment would impact the overall fuel cycle prospects and options.

Fundamental considerations lead to the conclusion that carbon is the obvious moderator for HTGRs regardless of fuel cycle mission. However, the HTGR technology family is not limited to TRISO fuel when considering potential application to the range of fuel cycle strategies. And, the HTGR technology family is also not limited to the common separation technology assumption of taking aqueous separation technologies developed for LWR oxide fuels and merely adding a step of mechanically breaking TRISO fuel coatings and applying LWR techniques to the fuel meat.

To enhance reactor safety and reduce operational radiation fields around coolant equipment, the mainline HTGR approach is to develop a very robust fuel coating (TRISO). The HTGR potential for a very low radioactive coolant is unparalleled as helium (or CO_2) is non-radioactive and less corrosive/erosive than water, sodium, lead, molten salts, etc.[Piet1982, Abdou1983, Smith1984] Therefore, the potential value of an extremely robust coating (nil "leakers") is higher in an HTGR than for other reactor coolants and thus a long-standing objective of HTGR R&D is creation of very robust fuel coatings. However, for recycle fuel strategies, a robust fuel coating poses a choice - develop technologies for undoing the robust coating or use a less robust coating. Options for undoing TRISO that have received little or no attention include jet-milling followed by mechanical re-forming, AIROX, or traditional chemical separation. Options for less robust coatings that have received little or no attention include ceramic tubes or porous coatings.

From the standpoint of flexibility and adaptability, the fundamental observation is that, in principle, the same reactor with the same general type of fuel (TRISO-coated particles) could be used for each of the fuel cycle missions, except breeding.

All commercial reactors in the US and 85% of those in the world are LWRs. (10% are heavy water, 4% are gas cooled, and <0.5% are liquid metal cooled.) Any non-LWR reactor's contribution to advanced fuel cycle strategies is therefore constrained by new technology deployment.

Top-level observations regarding waste management radiotoxicity and HLW-mass include:

- Any HTGR concept will have 1-2 orders of magnitude more waste mass unless the large amount of carbon is separated and recycled.

- MOC vs. once through – There is only a slight reduction in disposed waste radiotoxicity but there is an order of magnitude reduction in HLW mass. The relative reduction in radiotoxicity and HLW-mass is improved as TRU CR decreases, i.e., IMF vs. MOX or FR-burner vs. FR-breeder. If all FP are HLW (FP=HLW), the HLW-mass cannot drop below 0.001 kg-iHM/MWth-day, which is the mass of fission products per unit energy. If FP are not HLW (FP≠HLW), there is some additional reduction in HLW mass for MOC cases, but this is limited because used fuel is directly disposed.

- Full recycle vs. MOC or once through – There are orders of magnitude reduction in disposed waste long-term radiotoxicity but little HLW mass reduction versus MOC if all FP are HLW. The radiotoxicity reduction is ultimately limited by the processing loss rate for TRU (in this sense, deliberately not recycling minor actinides means the process loss rate is high). If FP are not HLW (FP≠HLW), the HLW mass can be further decreased (relative to MOC or once through) by orders of magnitude. Thus, full recycle options with low process loss rates and most/all FP not HLW can result in disposed waste that has radiotoxicity and HLW mass that are each orders of magnitude lower than for once through options.

Top-level observations regarding uranium utilization.

- In the grand scheme of all options, the LWR and HTGR have similar uranium utilization; lower than 1% for both.

- Between LWR and HTGR, the comparison depends on prismatic vs. pebble bed and the thermal efficiency of the HTGR. The prismatic once-through case appears to be somewhat lower uranium utilization than does LWR-UOX. The pebble bed and HTGR-IMF (deepburn) cases appear to have higher uranium utilization than the LWR.

Top-level observations regarding proliferation resistance and physical protection.

- The prismatic options appear very similar to their LWR counterparts.

- The pebble bed design approach is outside common safeguard experience. LWRs, fast reactors, and prismatic HTGRs have fuel assembly masses in the range of 120 to 660 kg; HTGR pebble bed have fuel pebbles that weigh about 0.2 kg. (Each pebble has ~50,000 TRISO-coated fuel particles.) Thus, LWRs, fast reactors, and prismatic HTGRs have hundreds of fuel assemblies, which are easily counted as individual items. Pebble bed HTGRs have hundreds of thousands of fuel pebbles, e.g., the PBMR-400 design had 450,000 pebbles. These are thought to be counted as bulk material, not individual items. Moreover, the packing density of pebbles varies, making exact balance of the number of pebbles difficult; fortunately, several thousand pebbles are required to make up one “significant quantity” (SQ) of weapon-usable material.

In summary regarding different fuel missions:

- *Once through*: higher fuel burnup has been shown in HTGRs versus LWRs (~20% vs. 8%), due in part to higher performance of HTGR TRISO fuel coating vs. LWR Zr-alloy fuel cladding. Advanced LWR fuel cladding, e.g., SiC, might reduce this differential.

- *Limited recycle*: Either HTGRs or LWRs can perform limited recycle via homogeneous core uranium-free fuel (called “deepburn” in HTGR and Inert Matrix Fuel in LWRs), homogeneous core mixtures of uranium and TRU (called MOX in LWRs), or heterogeneous core mixtures of uranium and TRU (mix of IMF and UOX fuel in LWRs). The first approach produces used fuel that is not easy to subsequently recycle; the latter two options can be subsequently recycled and are therefore best addressed as the first stages of continuous recycle. Homogeneous core uranium-free fuel (deepburn/IMF) can theoretically consume 50-70% of initial TRU, but this common result of both HTGRs and LWRs depends on addressing various constraints. To date, the analysis of these constraints across fuel cycle options (HTGR-IMF vs. LWR-IMF) varies in depth and therefore the fact that comparable consumption (50-60%) has been calculated for the two options does not mean equal probability of achieving such high consumption. The high TRU consumption results currently appear more credible for HTGR-IMF than LWR-IMF.

- *Full recycle*: Detailed studies of full recycle possibilities for HTGR are warranted. Although such studies have not been done, there are two HTGR/TRISO characteristics that suggest potentially higher performance than continuous recycle in LWRs. First, to date, TRISO fuels have higher burnup than Zr fuels. Second, HTGRs lack the LWR constraint of void coefficient (since the moderator is solid not liquid in HTGRs) and therefore higher sustained TRU loadings may be possible in HTGRs.

- *Recycle of non-fuel materials*: Waste management depends on more than fuel; all radioactive materials must be addressed, especially fuel coating/cladding and structural material holding the core internals in place. These are primarily Zr cladding and steel in LWRs and C/SiC TRISO coating and graphite in HTGRs. Inadequate work has been done on recycle possibilities of all of these materials. For example, although some graphite recycling work has been shown, the separation of fuel particles from structural graphite remains and issue and the recycle of the C/SiC coating poses different questions than recycle of bulk graphite.

In summary regarding separation R&D issues:

- No known candidate fuel separation/treatment method is completely without a chemical aspect. So, a technically useful definition of “minimum fuel treatment” has to specify which elements can be separated and by how much.

- All known candidate fuel separation/treatment methods have significant unknowns with regard to how the method would (or would not) separate important chemical elements in used HTGR fuels.

- What is the practicality of “crush to powder”? If practical, are there advantages to “minimum fuel treatment” that warrant studies?

- What is the tolerance of TRISO coatings to fuel kernels with high impurities (from both physics and performance perspectives), such as what would be present if used fuel were processed with AIROX or simple mechanical re-forming after “crush to powder”? If TRISO coatings could not be used, then other approaches such as ceramic tubes might be applicable.

- Interest in “limited separation” or “minimum fuel treatment” separation approaches motivates study of impurity-tolerant fuel fabrication.

- Funded studies are ongoing that will determine the feasibility of using reactive gas chemistry to dissolve carbon and silicon carbide from TRISO form HTGR fuels. However, it may also be possible to recycle this material via physical treatment and separation methods and/or a combination of physical treatment and chemical methods that selectively separate/ dissolve the fuel matrix without reaction of the carbon or silicon carbide.

- Could porous fuel coating approaches such as GA has proposed for their EM^2 concept be applicable for HTGRs? Thus, another idea being considered for future HTGR fuel forms is to use other coating(s) on the fuel kernel with sufficient porosity to only allow the escape of helium, tritium, and noble gases as produced during irradiation. These gases could be separated from the helium coolant as produced, negating the requirement for robust pressure vessels (SiC) and the difficulty of assuring only perfect particles are employed in the HTGR.

- Could high-temperature cladding materials, e.g. SiC, be considered as new gas-reactor tubular fuel cladding options?
- Can off-gases produced during used TRISO reprocessing help feed new TRISO coating fabrication if separation and fabrication facilities were integrated?
- The observation of higher decay heat (from IMF-type fuel) relative to UOX in the first few hours after shutdown needs further exploration.
- How much reduction in accident source term is possible from fuel venting and is it sufficient to compensate for the reduced fuel coat/clad robustness, i.e., there is another pathway for radioactivity to escape the core?

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Appendix A

Carbon is the Moderator

This appendix re-examines whether carbon is the best or only moderator for HTGRs, by re-examination of the first 10 elements (hydrogen to neon). The conclusion is that there are alternatives but none sufficiently attractive to warrant study of anything other than carbon.

A-1. Neutron Moderation

The first criterion to judge moderator candidates is indeed their moderating power. The following discussion is taken from J. R. Lamarsh, Introduction to Nuclear Reactor Theory, Addison-Wesley Publishing Company, Inc. 1966 and A. R. Foster and R. L. Wright, Jr. Basic Nuclear engineering, 2nd Edition, 1973.

Define, lethargy, as follows

$$u = \ln\left(\frac{E_0}{E}\right)$$

E_0 is an arbitrary energy, typically the "most energetic neutrons in whatever system is under considerations". In what follows, set E_0 as the initial energy of the neutron created by fission. Then, if A is the atomic mass, the average change in lethargy per collision is given by

$$\overline{\Delta u} = \xi = 1 - \frac{(A-1)^2}{2A}\ln\left(\frac{A+1}{A-1}\right)$$

This is also called the average change in the logarithm of neutron energy per collision. Consider that $\overline{u_n}$ is the average increase in lethargy at the nth collision. Then define

$$n = \frac{\overline{u_n}}{\xi} = \frac{1}{\xi}\ln\left(\frac{E_0}{E}\right)$$

"This formula must be interpreted with some care. Thus n is *not* the average number of collisions necessary to reduce the energy from E_0 to E, *nor* is it the number of collisions necessary to reduce the average energy to E. It is the number of collisions required to increase the average lethargy to the value $\overline{u_n}$ whose corresponding energy is E. All these numbers represent somewhat different concepts and they are slightly different numerically."

With E_0= 2 MeV and E = 1 eV, it is possible to the number of collisions, n, necessary to increase the average lethargy of 2-MeV neutrons to a value whose corresponding energy is 1 eV, assuming isotropic scattering in the center-of-mass system, as well as calculate the average energy loss per collision. See figures A-1 and A-2.

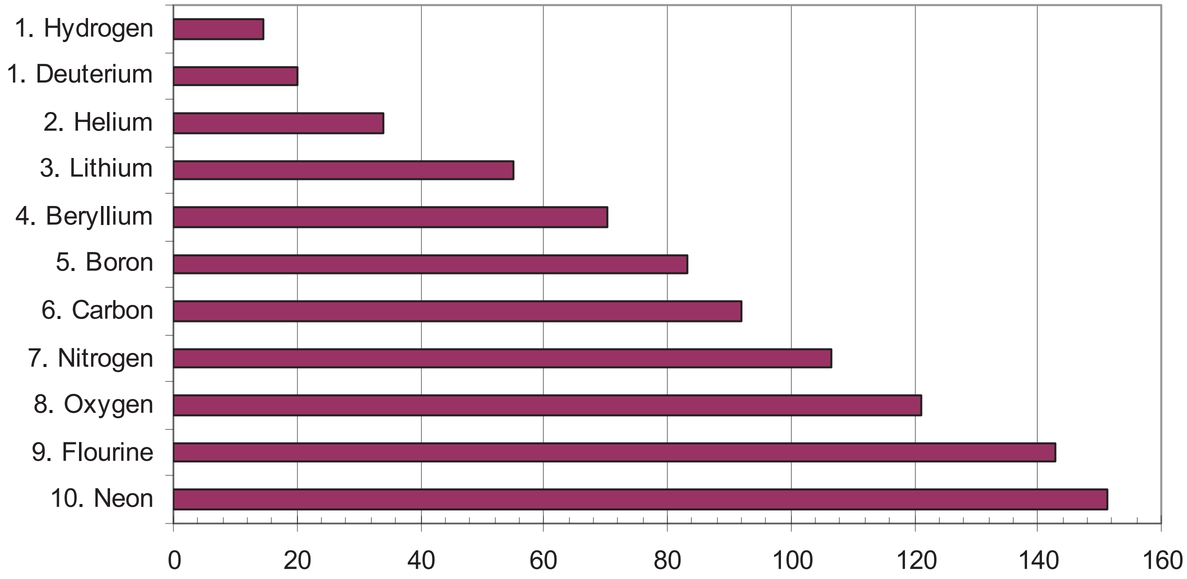

Figure A-1. Average number of collisions to increase the average lethargy of 2-MeV neutrons to a value whose corresponding energy is 1 eV.

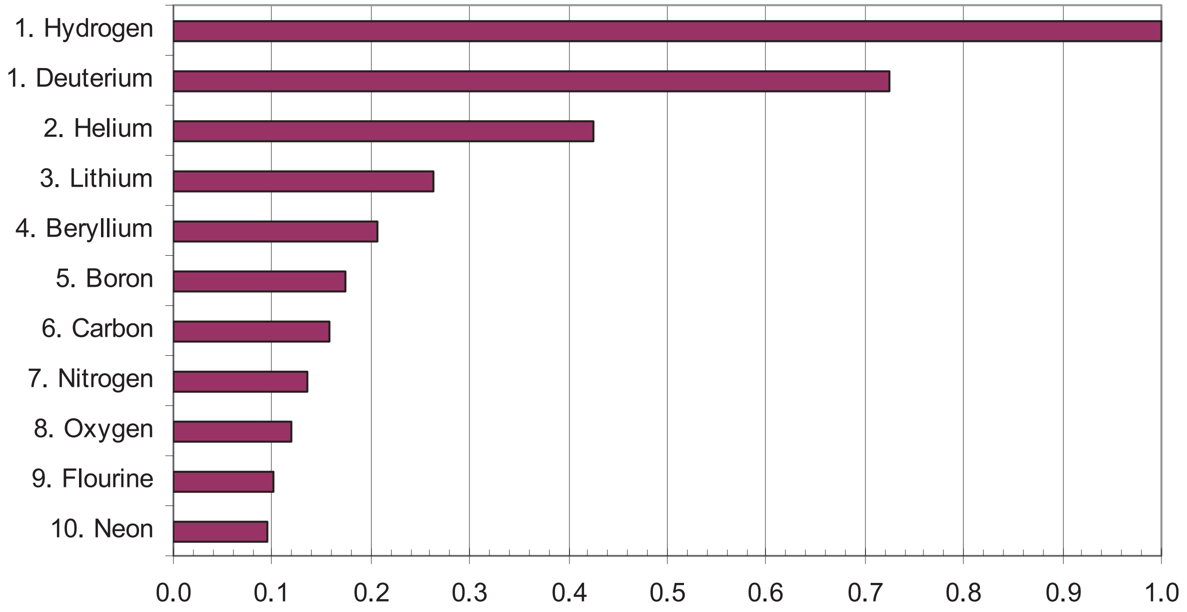

Figure A-2. Log of the average energy decrement.

Of course, helium and neon are inert gases and can only be present as gases. Their density is therefore 2-3 orders of magnitude lower than the other candidates. They are dropped from further consideration.

A-2. Neutron Absorption

The second criterion is neutron absorption, in a thermal spectrum. Calculations used ORIGEN-2.2 data for UOX-50. There are four relevant neutron reactions, as follows:

(n,γ) neutron capture

(n,p) proton emission

(n,α) alpha emission

(n,2n) neutron multiplication

Neutron multiplication does not waste or lose a neutron, in fact, it moderates it. So, we do not add the neutron multiplication cross sections to the others. The first three of the reactions, however, do lose neutrons, unless one wants to produce specific reaction products such as tritium from Li via the (n,α) on Li6 and Li7. One has to include all three neutron-wasting reactions to get a proper comparison, e.g., the dominant neutron wasting reaction for B-10, Li-6, Li-7 is (n,α) and for N-14 is it (n,p). When alpha emission is included, boron and lithium are seen as wasting more neutrons by 1-2 orders of magnitude than any of the other candidates. They were dropped from further consideration.

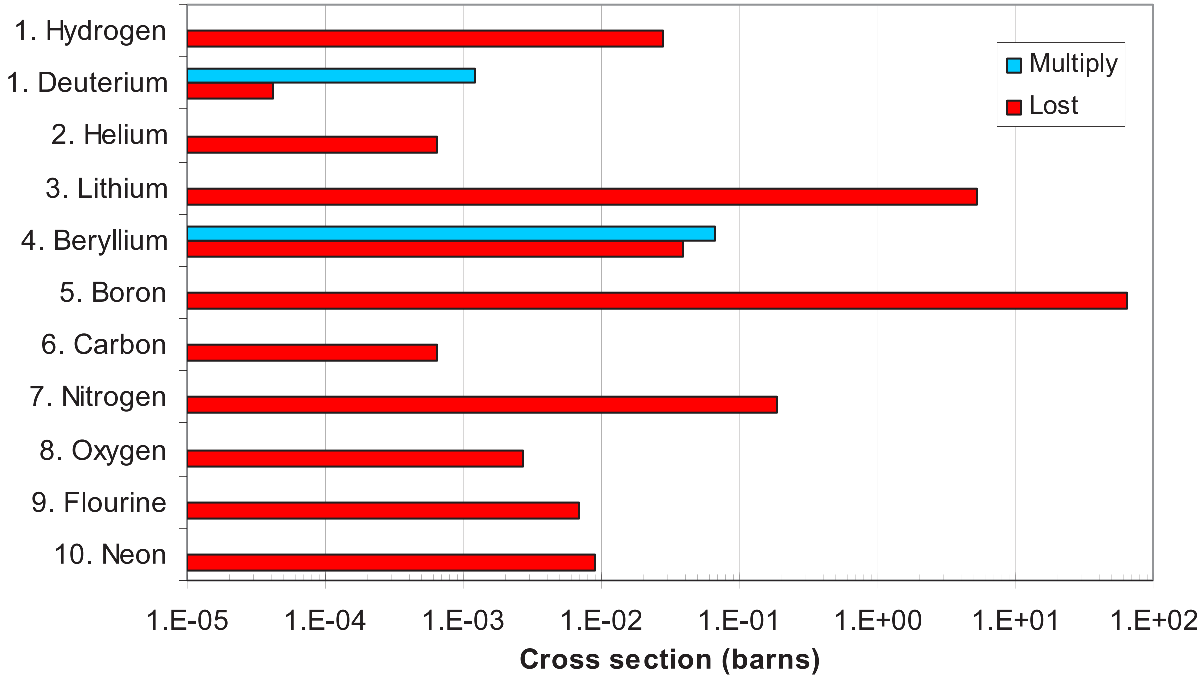

Figure A-3. Thermal cross sections for multiplying or losing (wasting) neutrons.

The lowest neutron consumers are deuterium, helium, and carbon. This is why reactors with mostly these materials can go critical with naturally enriched uranium - CANDUs and MAGNOX.

Figure A-4 and Table A-1 summarize moderator candidate information.

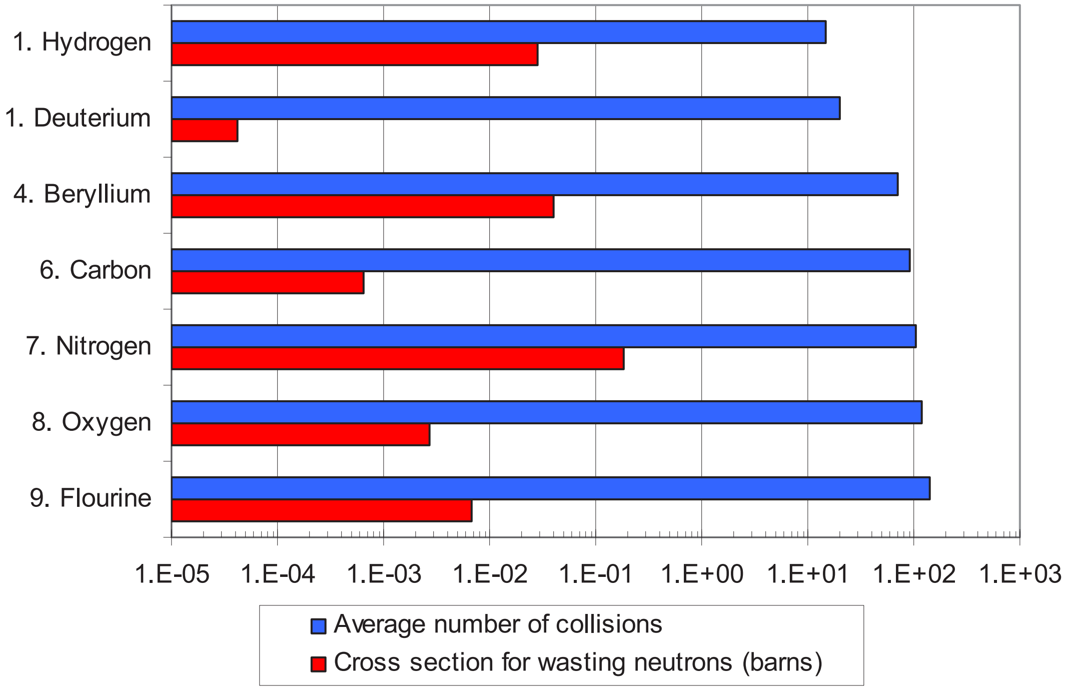

Figure A-4. Comparison of six moderator candidates from the standpoints of the average number of collisions (lower is better) and the cross section for wasting neutrons (lower is better).

Table A-1. Survey of Candidate Neutron Moderators

	Nuclear properties (lower is better)			
Element	Cross section for wasting neutrons (barns)	Number of neutron moderating collisions	Physical and chemical forms	Comment on candidate moderator
1. Hydrogen	0.028	15	Water, hydrides	Hydrogen is used in the hydride form, e.g., TRIGA reactors and some space reactor concepts.
1a. Deuterium	<0.000	20	Water, hydrides	
2. Helium	0.001	34	He gas	Helium has insufficient density to be an effective moderator.
3. Lithium	5.249	55	Li metal Li oxides Li fluoride	Lithium is relatively undesirable in a fission reactor because of its production of tritium, thereby wasting neutrons.
4. Beryllium	0.039	70	Be metal Be fluoride	Beryllium moderates, multiplies, and reflects neutrons. It is relatively expensive and chemically toxic, so its use is generally reserved in fission and fusion reactor concepts for neutron multiplication. The INL Advanced Test Reactor is partially beryllium moderated.
5. Boron	64.664	83	B_4C	Neutron waste is too high
6. Carbon	0.001	92	Graphite, C/SiC coatings, carbide/ oxycarbide fuels	Carbon is a decent moderator with low neutron absorption, stable chemical forms, with high purity products available.
7. Nitride	0.188	106	Nitride	Nitrogen is theoretically a candidate, but there are no obvious advantages versus carbon, at least for the traditional once-through HTGR fuel cycle. Nitrogen produces waste isotope C-14.
8. Oxygen	0.003	121	Oxides	Oxygen theoretically a candidate, but there are no obvious advantages versus carbon, at least for the traditional once-through HTGR fuel cycle.
9. Fluorine	0.007	143	Fluorides	Fluorine is relatively expensive and chemically active and relatively weak as a moderator. Indeed, fluoride-based salt-cooled reactors can be either thermal or fast spectrum. The fluorine moderator is not sufficient to force a thermal spectrum.
10. Neon	0.009	151	Ne gas	Ne has insufficient density to be an effective moderator.
Cross section for neutron waste is the sum of neutron interactions that waste neutrons, (n,γ), (n,α), and (n,p). Illustrative thermal capture cross sections are for UOX in the ORIGEN 2.2 database with natural isotopic abundances except where noted.				