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Fusion-Fission Hybrid for Fissile Fuel Production without Processing

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Fusion-fission hybrid for fissile fuel production without processing

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Summary

Two scenarios are typically envisioned for thorium fuel cycles: "open" cycles based on irradiation of ^{232}Th and fission of ^{233}U in situ without reprocessing or "closed" cycles based on irradiation of ^{232}Th followed by reprocessing, and recycling of ^{233}U either in situ or in critical fission reactors. This study evaluates a third option based on the possibility of breeding fissile material in a fusion-fission hybrid reactor and burning the same fuel in a critical reactor without any reprocessing or reconditioning. This fuel cycle requires the hybrid and the critical reactor to use the same fuel form. TRISO particles embedded in carbon pebbles were selected as the preferred form of fuel and an inertial laser fusion system featuring a subcritical blanket was combined with critical pebble bed reactors, either gas-cooled or liquid-salt-cooled. The hybrid reactor was modeled based on the earlier, hybrid version of the LLNL Laser Inertial Fusion Energy (LIFE[1]) system, whereas the critical reactors were modeled according to the Pebble Bed Modular Reactor (PBMR) and the Pebble Bed Advanced High Temperature Reactor (PB-AHTR) design. An extensive neutronic analysis was carried out for both the hybrid and the fission reactors in order to track the fuel composition at each stage of the fuel cycle and ultimately determine the plant support ratio, which has been defined as the ratio between the thermal power generated in fission reactors and the fusion power required to breed the fissile fuel burnt in these fission reactors. It was found that the maximum attainable plant support ratio for a thorium fuel cycle that employs neither enrichment nor reprocessing is about 2. This requires tuning the neutron energy towards high energy for breeding and towards thermal energy for burning. A high fuel loading in the pebbles allows a faster spectrum in the hybrid blanket; mixing dummy carbon pebbles with fuel pebbles enables a softer spectrum in the critical reactors. This combination consumes about 20% of the thorium initially loaded in the hybrid reactor (~200 GWd/tHM), partially during hybrid operation, but mostly during operation in the critical reactor. The plant support ratio is low compared to the one attainable using continuous fuel chemical reprocessing, which can yield a plant support ratio of about 20, but the resulting fuel cycle offers better proliferation resistance as fissile material is never separated from the other fuel components.

[1] Note that LIFE is now a pure fusion design, so we denote the hybrid version as H-LIFE in this paper.

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Introduction

In the last few years there has been renewed interest by some in the fusion and fission communities to explore the synergistic coupling of neutron-rich fusion with energy-rich fission in fusion-fission hybrids and nuclear waste burners. Although a fusion-fission hybrid is not a new idea [1-4], several new approaches have been proposed. In the 1970s and 1980s, the emphasis was producing fissile fuel for light water reactors (LWRs) in the so-called fuel factory mode, where one hybrid could supply fuel for 10 or more LWRs and over 20 pebble bed reactors, based on either U/^{239}Pu or Th/^{233}U fuel cycles [5-6]. The motivation was that the cost of uranium would eventually increase as natural resources were depleted, making it economically attractive to breed fissile fuel in fusion reactor blankets if the fusion costs were low enough. As we know, the buildup of fission reactors nearly ceased, the high demand and high cost of uranium did not materialize, and interest in fusion-fission hybrids diminished, although some work continued on waste-burning hybrids [7].

In 2007, the Lawrence Livermore National Laboratory (LLNL) began a study of a laser inertial fusion-fission energy system [8]. Such systems can utilize a fission blanket to realize energy gain beyond that from the fusion targets. A once-through, deep burn-up fuel cycle option might be achievable. If successful, this would eliminate the need for enrichment and fuel reprocessing, and it could reduce the mass of long-lived nuclear waste per-unit energy produced that must be placed in a long term, deep geologic repositories. This is a power-mode hybrid; it seeks to more fully utilize the energy content of uranium or thorium and burn its own waste (or spent nuclear fuel), as opposed to providing fuel for fission reactors.

This work evaluates another hybrid application using the fusion neutron source to breed fissile fuel for critical reactors. In particular, it investigates the possibility to enrich thorium-based fuel for breeding ^{233}U and use such fuel directly in fission reactors, eliminating the need for reprocessing or reconditioning, as well as the need for an initial inventory of fissile material typically required to start a thorium fuel cycle. Multiple scenarios can be envisioned to accomplish this synergetic fuel cycle, but the constraint of no reprocessing requires both systems to operate with the same fuel form. For this study TRISO fuel particles carried in carbon pebbles were selected as they present significant advantages:

- High burn-up: in a thermal neutron spectrum TRISO particles can achieve up to ~20% FIMA (Fissions per Initial Metal Atoms) with no need for any reconditioning;
- Proliferation resistance: the fissile material is dispersed in millions of tiny particles, complex to reprocess;
- Flexibility: pebbles are well suited for the intricate geometries of hybrid systems' blankets and allow on-line refueling.

Consequently pebble-bed type fission reactors, gas-cooled or liquid-salt-cooled, were chosen. The companion hybrid system was based on the LLNL laser driven inertial fusion system [8].

This manuscript summarizes the analyses of a hybrid and critical reactor cycle starting from pure thorium fuel. In particular it describes: an extensive neutronic analysis that allowed determining the time dependent fuel composition as a function of multiple design parameters; a corresponding neutronic analysis of the fuel performance in the pebble bed reactors; results for attainable support ratio and possible ways for improvement.

Thorium fuel cycles

Thorium-based fuel cycles received considerable attention in the pioneering years of nuclear energy when uranium resources were considered inadequate to support a rapid growth of nuclear energy. As the shortage of uranium never became reality, thorium fuels went neglected. In more recent years the interest for thorium fuel was awakened by the possibility to develop fuel cycles that feature better proliferation resistance, higher burn-up, and reduced impact waste forms. Indeed, these and other possible benefits associated with thorium fuel include [9]:

- Thorium is more abundant than uranium and widely distributed; it would allow a long-term sustainable fuel cycle.
- Waste from thorium may have lower radiotoxicity.
- In the thermal range ^{232}Th capture cross section is about three times that of ^{238}U meaning a more efficient breeding (in contrast larger enrichments are required to maintain criticality and for this same reason heavy water systems are more suitable for thorium as the high absorption in thorium is compensated by the less absorption in the moderator).
- Thorium oxide is chemically more stable and has better radiation resistance than uranium oxide; fission product release rates are an order of magnitude lower.
- Hard gammas from ^{232}U daughters (73.6 y half-life), mainly ^{212}Bi (0.7-1.8 MeV) and ^{208}Tl (2.6 MeV) may make the burnt fuel self-protecting as ^{232}U is ~3% of all uranium and typically 2.4% is required for self-protection.
- Used thorium fuel contains less plutonium and minor actinides than uranium fuel, but contains ^{231}Pa, ^{229}Th, and ^{230}U that are isotopes of long-term radiological impact.

At the same time, thorium fuel presents new challenges [9]:

- Thorium oxide's high melting point (3350 °C vs. 2800 °C for uranium oxide) requires high sintering temperatures (>2000 °C).
- Post processing is more difficult as thorium oxide does not dissolve in nitric acid because it is a very stable dioxide. The THOREX process uses fluoric acid (HF) and long dissolution periods.
- High radiation doses from used fuel due to ^{232}U daughters may require remote and automated handling.
- ^{233}Pa has a long enough half-life (~27 days) that an off core decay time is often required to optimize ^{233}U breeding.
- No industrial scale demonstration of separation of U, Pu, and Th.
- Limited data and limited operational experience.

In order to implement a thorium fuel cycle, two options are often considered:

1. "Open" cycle based on irradiation of ^{232}Th and fission of ^{233}U in situ without reprocessing;
2. "Closed" cycle based on irradiation of ^{232}Th, reprocessing, and recycling of ^{233}U.

This study evaluates a third option based on irradiation of ^{232}Th bearing fuel in an inertial fusion source-driven system and utilization of the same fuel without any reprocessing in a critical reactor for fissioning of ^{233}U.

Models and methodology

The final scope of this analysis is to determine how many fission reactors can be supported using the fuel produced in a fusion-fission hybrid reactor. To this purpose it is necessary to determine how the fuel composition changes during the three phases of the envisioned fuel cycle: (1) breeding in the hybrid reactor; (2) cooling for decaying of ^{233}Pa to ^{233}U; (3) burning in critical reactors. The following paragraphs describe the models and the methodologies applied to track the fuel composition in each of these steps.

Fusion-fission hybrid

The fusion-fission hybrid was modeled according to the hybrid version of the LLNL Laser Inertial Fusion Energy system. The pseudo-spherical geometry was represented with a build-up of concentric spherical shells. The inner sphere—vacuum chamber, encloses the source at its center. The radial build-up (Figure 1) includes: a first wall; a main coolant inlet plenum; an optional neutron multiplier layer composed of metallic beryllium coated in oxide dispersion strengthen (ODS) steel and coolant; a fuel blanket filled with fuel pebbles; a reflector composed of graphite pebbles and coolant (Table 1.) Each section is separated by a 0.3 cm ODS steel (hereafter referred to simply as "ODS") wall. The first-wall is composed of tubes within which the coolant flows. The model renders the first wall using three concentric shells: ODS, lithium and ODS, of thickness 0.5 cm, 10 cm, and 0.5 cm respectively. In order to preserve masses, the ODS density in this two layers was increased by 37.4%, and the lithium density was reduced by 34.6%. A total of 48 beam ports penetrate through each layer. Besides the first-wall cooled by lithium, liquid salt flibe ($2LiF\text{-}BeF_2$) is the main blanket coolant. Flibe flows radially outward from the inlet plenum to outlet plenum beyond the reflector through perforated walls. The fusion source was assumed to operate at a low yield, 37.5 MJ at 13.3 Hz, and to produce 500 MWth total.

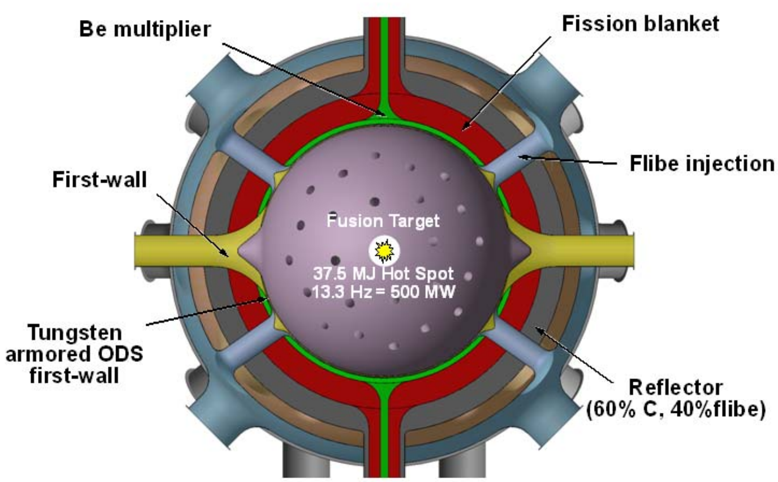

Figure 1. Center cross-section of the H-LIFE model.

The fuel pebble dimensions were chosen according to current assumptions for pebble bed reactors, that is 6 cm diameter with 0.5 cm carbon shell. TRISO particles dimensions are given in Table 2. The number of TRISO particles per pebble was set as a design parameter controlled by the packing factor—volume fraction of the inner pebble occupied by the TRISO, or by the ratio of carbon atoms to heavy metal atoms (C/HM) in the pebble.

Table 1. Dimensions of the radial components of the H-LIFE model.

Region	Thickness, cm	Material
Chamber (including first wall tubes)	500 (diameter)	Xe
First wall	11	Li/ODS
Gap	1	Xe
Coolant injection plenum	3	flibe
Multiplier	0-8	Be/flibe
Fuel blanket	30-100	fuel/flibe
Reflector	75	carbon/flibe
Walls in between each region	0.3	ODS

Table 2. Dimensions of TRISO particles.

Layer	Thickness, μm
Fuel kernel	450 (diameter)
Carbon buffer	100
Inner Pyrolitic Carbon	35
SiC	35
Outer Pyrolitic Carbon	35

Critical reactors

The gas-cooled critical reactor was modeled according to the PBMR (Pebble Bed Modular Reactor) design [10] and the liquid-salt-cooled according to the PB-AHTR (Pebble Bed-Advanced High Temperature Reactor) design [11]. Besides the coolant, the main difference between these systems (Table 3) is that the PB-AHTR operates at a higher power density of 10.2 MW/m^3, compared to the PBMR power density of 4.68 MW/m^3. For both fission systems, a cylindrical configuration was assumed and the same total thermal power (600 MW$_{th}$) was assumed.

Table 3. Gas-cooled and liquid salt-cooled pebble bed reactor features.

Property	PB-AHTR	PBMR
Coolant	Flibe	Helium
Thermal power, MW	600	600
Power density, MW/m^3	10.2	4.68
Core diameter, cm	430	558
Core height, cm	404	524
Leakage, %	6	6
Number of pebbles	~312,000	~680,000
Power per pebble, kW	1.923	0.882

Methodology

The time dependent fuel composition in the fusion-fission hybrid was determined using the LIFE Neutronics Code (LNC) that relies on MONTEBURNS (combining MCNP and ORIGEN2.2) for depletion while assuring that other system requirements (power level, tritium breeding ratio, etc.) are met [12-15]. A full 3D blanket model with accurate resolution of fuel geometry was employed. The fusion-fission hybrid was constrained to be tritium self-sufficient. The ratio of ^{6}Li to ^{7}Li in the coolants was varied to maintain a constant tritium-breeding ratio slightly above 1 to compensate for tritium losses, and the total thermal power level (fusion and fission) was allowed to vary accordingly. Before being transferred to a fission reactor, the fuel from the hybrid system was allowed to decay and its composition was tracked by ORIGEN2.2. It was found that a few 27 day half-lives of cooling time between pebbles unload from the hybrid blanket and load into a critical reactor is acceptable to reduce ^{233}Pa content in the fuel and maximize reactivity (Figure 2). One year cooling time was applied through this study.

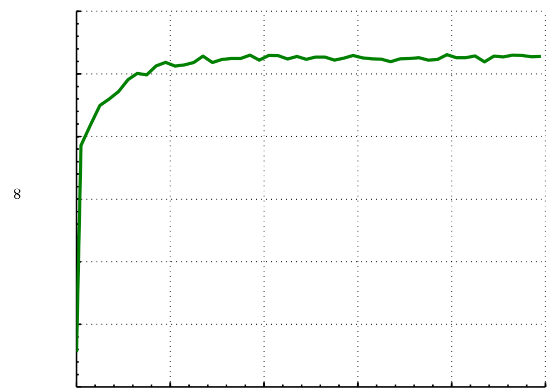

Figure 2. Beginning of life infinite multiplication factor in a liquid-salt-cooled pebble bed reactor as a function of fuel cooling time after discharge from the hybrid blanket.

The critical reactor models were represented as an infinite lattice of pebbles with surrounding coolant and depletion analysis was performed through MONTEBURNS assuming a constant neutron flux. The attainable burn-up was inferred from the depletion analysis of the infinite lattice model as in reference 16. The pebble bed reactors were assumed to operate in continuous refueling mode and to remain always critical; therefore, the infinite multiplication factor was set equal to the inverse of the neutron non-leakage probability, that for a 600 MW_{th} cylindrical core was calculated to be 94% for either coolant.

Ultimately the support ratio was determined combining the residence time in the hybrid and in the critical reactor. Traditionally, for fission-suppressed systems the support ratio is the defined as fission power generated per unit of fusion power. In this study, the hybrid system does not employ recycling and fissions are not suppressed effectively. Fission power generated in situ is about the same as fusion power; therefore, three different definition of support ratio are defined. The reference fuel cycle in this study features three power sources:

1. Total thermal power from fusion only (P_{fus})—that includes neutrons and alpha particles energy, total 17.6 MeV per fusion event;
2. Thermal power generated in the fission blanket of a hybrid system (P_{FF})—mainly from fission, but also from other nuclear reactions;
3. Total thermal power generated in the critical reactors (P_{fis}.)

The plant support ratio is defined as P_{fis}/P_{fus} and is directly proportional to the number of critical reactors that the hybrid system can support. The energy support ratio is defined as

$P_{fis}/(P_{fus}+P_{FF})$ and determines what fraction of the power is generated in the critical reactors rather than in the hybrid system. The source support ratio is defined as $(P_{fis}+P_{FF})/P_{fus}$ and determines the fission power produced per unit of fusion energy.

Results

Attainable enrichment

In order to optimize a thorium fuel factory for breeding ^{233}U to be used in a critical reactor with no reprocessing, three situations need to be accomplished:

1. Maximize neutron capture in ^{232}Th;
2. Minimize neutron absorption in ^{233}Pa;
3. Minimize neutron absorption in ^{233}U.

Figure 3 shows cross-sections for these three nuclides. The thorium capture cross-section is the smallest over the entire energy range. The optimal trade-off is expected to require neutrons with energy above 1 keV. Figure 4 shows the ratio between capture cross-section of ^{232}Th (^{233}U production) and absorption cross-section of ^{233}U (loss). This ratio is always below unity, except at resonance peaks, and reaches maximum about 0.2; therefore, to effectively breed ^{233}U, thorium concentration in the fuel must be at least 5 times that of ^{233}U. As absorption in ^{233}Pa also causes a loss of potential ^{233}U, more realistically the fuel must contain ~7 atoms of ^{232}Th for each atom of ^{233}U, assuming 1 atom of ^{233}Pa for every 5 atoms of ^{233}U. This means that the fraction of fissile in thorium fuel with no reprocessing cannot exceed about 15%.

From the consideration above, the breeding system is expected to optimize with a relative fast spectrum. The burner reactor features a very soft spectrum and this could potentially enable breeding as in the thermal energy range the number of neutrons produced per neutron absorbed in ^{233}U is mostly above 2 (Figure 5). One neutron is needed to maintain the chain reaction and another to breed fuel, leaving a small amount for leakage and parasitic losses.

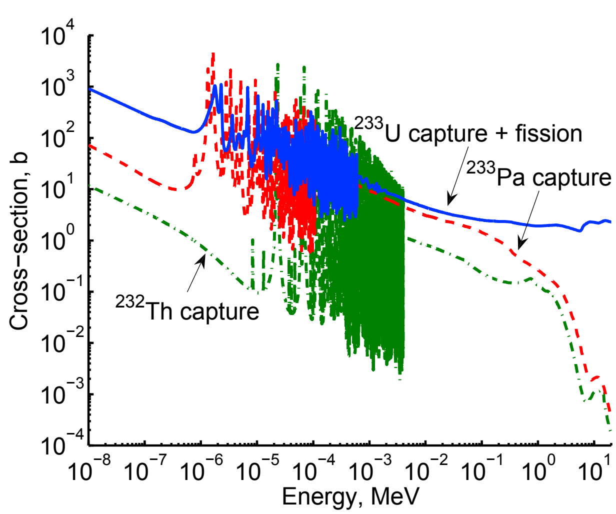

Figure 3. Comparison of ^{233}Th, ^{233}Pa, and ^{233}U cross-sections as a function of incident neutron energy.

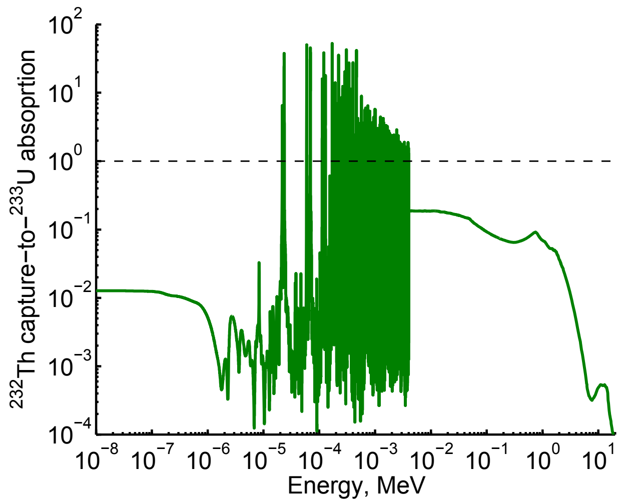

Figure 4. Ratio of ^{232}Th capture cross-section to ^{233}U absorption cross-section as a function of incident neutron energy.

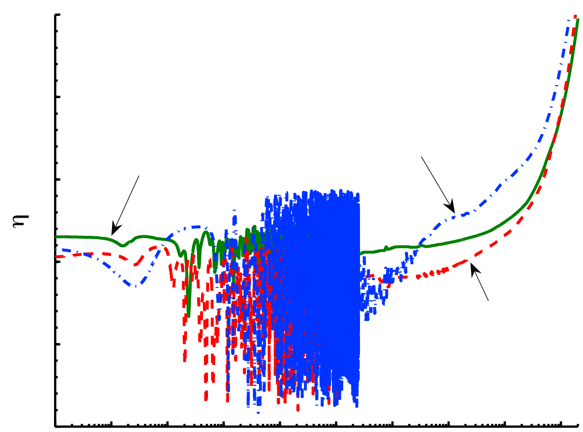

Figure 5. Comparison of neutrons produced per neutron absorbed as a function of incident neutron energy in ^{233}U, ^{235}U, and ^{239}Pu.

Figure 6 shows the enrichment level, which is defined as the ratio of ^{233}U and ^{233}Pa atoms to total heavy metal atoms in the fuel, as a function of breeding time and C/HM ratio for a hybrid reactor featuring a 50 cm blanket and no multiplier layer. The enrichment grows more rapidly for softer spectra but it stabilizes at a lower level. Faster spectra achieve higher enrichment regardless of burn-up (Figure 7), meaning fission products do not influence the attainable enrichment level.

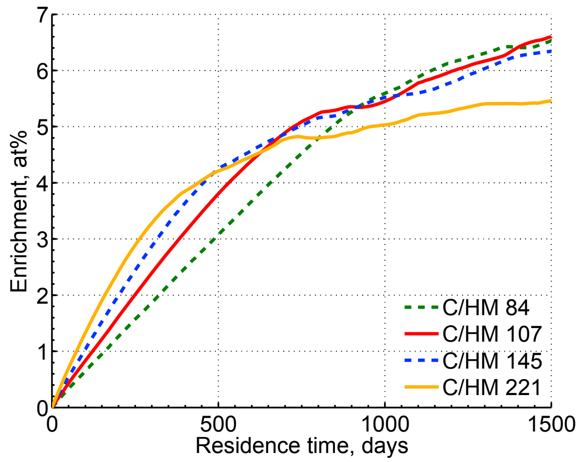

Figure 6. Fuel enrichment as a function of residence time and carbon-to-heavy metal ratio—50 cm thick blanket without multiplier.

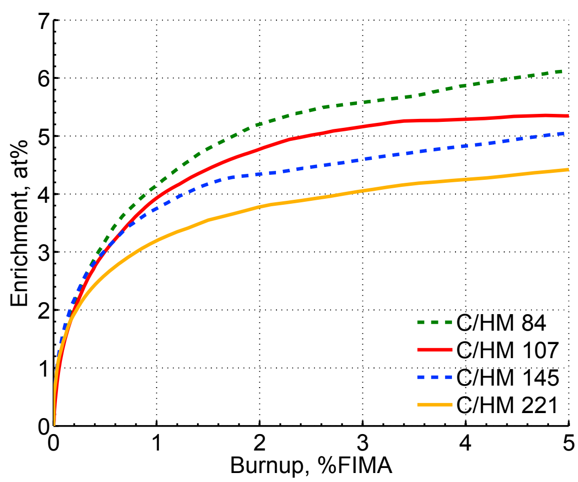

Figure 7. Fuel enrichment as a function of burn-up and carbon-to-heavy metal ratio—50 cm thick blanket without multiplier.

Loss of fissile inventory due to absorption in ^{233}Pa is about 8% and remains almost constant in the time range considered, as shown in Figure 8. Increasing absorption in ^{233}U effectively limits the fissile content in the fuel.

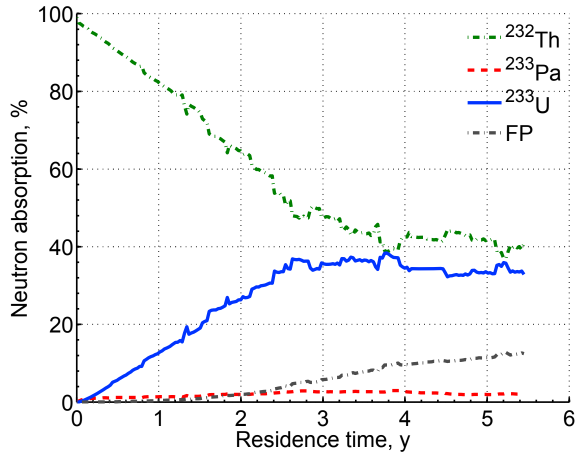

Figure 8. Fractional neutron absorption in ^{232}Th, ^{233}Pa, ^{233}U, and fission products as a function of breeding time for a 50 cm thick blanket, without multiplier, and 84 C/HM ratio.

At discharge from the hybrid system, the fuel enrichment is between 5% and 7%. This is significantly lower than the ~15% theoretical maximum estimated above for two main reasons: (1) only a fraction of the neutrons in the blanket have energy in the desirable range (Table 4); (2) larger enrichments would require longer breeding times, meaning larger accumulation of fission products that would be counterproductive once the fuel is transferred into the fission reactors.

Table 4. Fraction of neutrons with energy in the preferred range for breeding (>1 keV) as a function of C/HM ratio.

C/HM	Neutron fraction >1 keV
84	0.73
107	0.67
145	0.62
221	0.58

Parametric analysis

The maximum attainable support ratio was identified through a parametric analysis. Three main design parameters that directly influence the neutron spectrum in the hybrid blanket were varied:

1. Multiplier thickness;
2. Blanket thickness;
3. Carbon-to-heavy metal ratio or TRISO packing factor.

Table 5 shows that the plant and energy support ratio decreases when the multiplier thickness increases due to a softening of the spectrum. The source support ratio, instead, increases due to the increase in fission power produced in the hybrid. Similar behavior was observed for all combinations of critical system, blanket size, and C/HM. In the rest of this study the multiplier layer was eliminated from the hybrid blanket radial build-up.

Table 5. Support ratio as a function of multiplier thickness for liquid-salt-cooled pebble bed reactors—50 cm thick blanket and 145 C/HM.

Multiplier thickness, cm	Plant Support Ratio	Energy Support Ratio	Source Support Ratio
0	0.48	0.22	2.21
2	0.20	0.08	2.24
4	0.08	0.03	2.50
6	0.03	0.01	2.73
8	0.01	0.00	2.96

Table 6 and Table 7 show the support ratio as a function of C/HM ratio and blanket thickness in liquid-salt-cooled and gas-cooled pebble bed reactors, respectively. Gas-cooled reactors achieve a maximum plant support ratio of about 0.7 with a TRISO particles packing fraction of about 20% (C/HM 145). Liquid-salt-cooled reactors reach a plant support ratio of about 0.9 and require a higher TRISO packing of between 40% and 50% (84-107 C/HM), as the coolant provide part of the neutron moderation. Thin blankets (<50 cm) reduce the plant support ratio, whereas varying the thickness between 50 cm and 70 cm causes only small variations. The energy support ratio, by definition, follows the trends of the plant support ratio. The plant source support ratio increases with blanket size as the fission power generated in the hybrid increases.

Table 6. Maximum support ratio as a function of blanket thickness and C/HM for liquid-salt-cooled pebble bed reactors.

C/HM	Blanket thickness, cm	Plant Support Ratio	Energy Support Ratio	Source Support Ratio
84	30	0.52	0.23	1.79
	50	0.76	0.27	2.54
	70	0.89	0.24	3.60
107	30	0.53	0.23	1.82
	50	0.81	0.27	2.82
	70	0.93	0.20	4.66
145	30	0.48	0.23	1.63
	50	0.58	0.22	2.21
	70	0.57	0.19	2.51
221	30	0.30	0.16	1.23
	50	0.33	0.13	1.78
	70	0.27	0.10	1.95

Table 7. Maximum support ratio as a function of blanket thickness and C/HM for gas-cooled pebble bed reactors.

C/HM	Blanket thickness, cm	Plant Support Ratio	Energy Support Ratio	Source Support Ratio
84	30	- [a]	- [a]	- [a]
	50	- [a]	- [a]	- [a]
	70	0.01	0.00	2.81
107	30	0.18	0.08	1.47
	50	0.40	0.14	2.29
	70	0.45	0.15	2.44
145	30	0.51	0.26	1.51
	50	0.68	0.24	2.46
	70	0.62	0.22	2.48
221	30	0.48	0.26	1.29
	50	0.58	0.25	1.89
	70	0.57	0.24	1.98

[a] Fission reactor does not achieve criticality

Table 8 and Table 9 contain the main fuel cycle parameters that maximize the plant support ratio as a function of C/HM ratio and blanket thickness. The desirable fuel enrichment is typically 5%. A total burn-up between 110 and 130-GWd/tHM (about 30-GWd/tHM in the hybrid system and about 90-GWd/tHM in the fission reactor) can be achieved while maximizing the plant support ratio.

Table 8. Fuel cycle parameters that maximize plant support ratio as a function of blanket thickness and C/HM ratio for liquid-salt-cooled pebble bed reactors.

C/HM	Blanket thickness, cm	Breeding time, days	Enrichment, %	Burn-up in hybrid, GWd/tHM	Residence time in fission reactor, days	Burn-up in fission reactor, GWd/tHM	Total burn-up, GWd/tHM
84	30	680	5.67	30	930	88	118
	50	920	5.35	22	926	87	109
	70	1280	5.29	24	929	88	112
107	30	520	5.01	25	791	94	119
	50	750	5.09	26	859	102	128
	70	1070	5.08	29	843	100	129
145	30	520	5.25	36	729	115	151
	50	640	4.70	33	554	87	120
	70	790	4.26	30	415	65	95
221	30	390	4.68	30	387	91	121
	50	490	4.19	34	264	63	97
	70	580	3.75	32	163	38	70

Table 9. Fuel cycle parameters that maximize plant support ratio as a function of blanket thickness and C/HM ratio for gas-cooled pebble bed reactors.

C/HM	Blanket thickness, cm	Breeding time, days	Enrichment, %	Burn-up in hybrid, GWd/tHM	Residence time in fission reactor, days	Burn-up in fission reactor, GWd/tHM	Total burn-up, GWd/tHM
84	30	- [a]	- [a]	- [a]	- [a]	- [a]	- [a]
	50	- [a]	- [a]	- [a]	- [a]	- [a]	- [a]
	70	1360	5.47	30	28	1	31
107	30	520	5.01	25	578	31	56
	50	790	5.20	30	960	52	82
	70	980	4.81	23	836	45	68
145	30	510	5.20	35	1662	120	155
	50	510	4.29	17	1223	88	105
	70	730	4.11	24	930	68	92
221	30	360	4.53	25	1281	140	165
	50	410	3.89	23	923	100	123
	70	530	3.62	26	707	77	103

[a] Fission reactor does not achieve criticality

Segmented blanket

The parametric analysis showed that thicker blankets attain higher plant support ratio, but they also require longer breeding times. In order to more effectively use thicker blankets, a segmented model was considered, in which the blanket is divided into multiple zones and pebbles are segregated in the same zone. The ^{233}U enrichment in the inner zone increases rapidly. Once the desired level is achieved, the inner zone is discharged and moved to the critical system, the other zones are centrally promoted, and the outer most zone is loaded with fresh pebbles. A 100 cm thick blanket and five zone scheme was chosen for this analysis. Table 10 shows how for a set enrichment level the pebble yield increases using a segmented blanket compared to a fully mixed blanket. It was found that the plant support ratio could increase to 1.46 (energy support ratio 0.80, source support ratio 2.30) limiting enrichment to 6%. An optimization of breeding time, blanket thickness, and number of segments could further improve this value.

Table 10. Comparison of pebble yield (pebbles/day) as a function of enrichment for a fully mixed and a segmented blanket—84 C/HM ratio.

Enrichment, %	Fully mixed (50 cm)	Segmented (100 cm, 5 zones)
5	326	420
6	238	310
7	157	192

Spectrum tailoring

The optimal neutron spectrum for breeding is epithermal/fast spectrum, whereas for burning a thermal spectrum is usually preferred. The requirement of having the same fuel in the hybrid and in the fission reactor fixes the C/HM in the pebbles and therefore the spectrum. In order to optimize the spectrum in both systems the TRISO packing was maximized (50%) and when fuel pebbles were transferred in the critical system, they were mixed with inert carbon pebbles. It was found that introducing one carbon pebble per fuel pebble in the core increases the plant support ratio up to 1.38 (energy support ratio 0.38, source support ratio 4.06). If the spectrum tailoring were combined with a segmented blanket the attainable plant support ratio would reach 1.86 and the total burn-up would be as high as 199 GWd/tHM. Both gas-cooled and liquid-salt-cooled reactors can achieve similar support ratio. Cores cooled by liquid salt require less carbon pebbles since part of the moderation is provided by the coolant.

Table 11. Maximum support ratio as a function of number of carbon pebbles per fuel pebble for liquid-salt-cooled pebble bed reactors using a 70 cm blanket and 84 C/HM ratio.

Carbon pebbles per fuel pebble	C/HM	Plant Support Ratio	Energy Support Ratio	Source Support Ratio
0	84	0.89	0.24	3.60
1	176	1.38	0.38	4.06
2	267	1.11	0.294	3.89

Table 12. Maximum support ratio as a function of number of carbon pebbles per fuel pebble for gas-cooled pebble bed reactors using a 70 cm blanket and 84 C/HM ratio.

Carbon pebbles per fuel pebble	C/HM	Plant Support Ratio	Energy Support Ratio	Source Support Ratio
0	84	0.01	0.00	2.81
1	176	1.33	0.72	2.16
2	267	1.66	0.91	2.48

Conclusions

This study investigated the possibility to breed fissile starting from thorium only fuel in a fusion-fission hybrid engine and burn the same fuel in a critical reactor without any reprocessing and/or reconditioning. In order to implement this fuel cycle, the hybrid and the critical reactor are required to use the same fuel form. TRISO particles embedded in carbon pebbles were selected as the preferred form of fuel and an inertial laser fusion system featuring a subcritical blanket was combined with critical pebble bed reactors, either gas-cooled or liquid-salt-cooled. The hybrid reactor was modeled based on the LLNL laser inertial fusion-fission energy system (H-LIFE), whereas the critical reactors were modeled according to the Pebble Bed Modular Reactor (PBMR) and the Pebble Bed Advanced High Temperature Reactor (PB-AHTR) design. An extensive neutronic analysis was carried out for both the hybrid system and the fission reactors in order to track the fuel composition at each stage of the fuel cycle and ultimately determine the plant support ratio defined as the ratio between the installed fission power and the installed fusion power. The plant support ratio is enhanced optimizing the neutron spectrum at each stage: epithermal/fast spectrum for breeding in the hybrid; thermal spectrum for burning in the critical reactors. This was achieved using a high TRISO particles loading in the pebbles (50% of the active volume) and adding inert carbon pebbles when the fuel pebbles are transferred into critical reactors. The maximum plant support ratio is about 2 and is obtained employing a 100 cm blanket hybrid, operated with a five-batch scheme and no neutron multiplier, and adding one carbon pebble per fuel pebble in the liquid-salt-cooled core or two carbon pebbles in the gas-cooled cores. The corresponding attainable fuel burn-up is about 200 GWd/tHM.

The plant support ratio for this fuel cycle is significantly smaller than the one attainable using continuous fuel chemical reprocessing, but the resulting fuel cycle offers better proliferation resistance because fissile material is never separated form the other fuel components.

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