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PRELIMINARY EVALUATION OF THE ^{233}U REFRESH CYCLE HYBRID POWER SYSTEM CONCEPT

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A preliminary evaluation was made of the technical feasibility of using a fusion reactor to breed ^{233}U from thorium in high temperature gas cooled fission reactor fuel and then using this fuel in a fission reactor without reprocessing. Estimates of the neutronic performance of thorium fusion reactor blankets indicate that adequate concentration of ^{233}U in thorium can be attained to allow operation of a high temperature gas cooled reactor using nonreprocessed fusion-bred ^{233}U fuel. Estimates of the fuel materials damage indicate that the breeding and subsequent burning of ^{233}U can be accomplished within the currently predicted materials limitations of high temperature gas-cooled reactor fuel. The system performance of this symbiotic fusion-fission power system, called the U-233 Refresh Cycle Hybrid Power System, was estimated. A refresh cycle fusion breeder reactor could support from two to three high temperature gas cooled fission burner reactors of equal thermal power without reprocessing. Greatly improved performance is possible if reprocessing is allowed. Thus preliminary evaluation shows that the concept is technically feasible and warrants more detailed study.

INTRODUCTION

The basic concept of producing fissile material in the blanket of a fusion reactor is promising because large quantities of high quality fissile fuel could be produced and because the fusion reactor needs no fissile inventory, thus freeing it from doubling time constraints.[1-3] The idea of using the fusion bred fuel without reprocessing is particularly attractive because it offers to alleviate concerns about nuclear proliferation. Only fertile material would be fed into a fusion hybrid breeder and the bred fissile material would have a high content of fission products that would never be removed from the fuel, as it would never be reprocessed. These fission products would protect the bred fuel from diversion just as fission products currently protect spent LWR fuel from diversion during shipping and storage.

To take full advantage of the no-reprocessing or "Refresh" cycle concept the thorium-uranium fuel cycle was selected. In this fuel cycle a neutron is absorbed by the thorium fertile material producing uranium-233 which is the best fissile material for use in a thermal spectrum fission reactor. High-Temperature Reactor (HTR) technology, which uses graphite moderator and helium coolant with the thorium-uranium fuel cycle, is attractive to the refresh cycle concept because it is capable of very high material burnup. The fuel cycle and reactor technology of the HTR are also developed and enjoy strong international interest through the High-Temperature Gas-Cooled Reactor (HTGR) program in the USA and the German High-Temperature Reactor (HTR) program.

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There are a number of technical considerations that have to be addressed to evaluate the feasibility of the Refresh Cycle concept.[4] In any fission or fusion reactor design the irradiation capabilities of the materials are of prime concern. The capabilities of the Refresh Cycle fuel materials are particularly critical as the fuel must survive irradiation in both the fusion and fission reactors and it may be desirable to recycle the fuel several times before the material limits are reached and it must be discarded. The nuclear performance of the fusion reactor breeding blanket must be analyzed. An adequate quantity of ^{233}U must be bred and the percentage of ^{233}U in thorium that can be reached must be sufficient to operate the burner reactor. The feed requirements of the burner reactor must be investigated to minimize the quantity of fuel that is required. Since the same fuel will be used in both fusion breeder and fission burner, it is important that the characteristics of the two reactors be matched together to obtain an optimum symbiotic fusion-fission power system.

MATERIALS CONSIDERATIONS

The fuel element for the HTGR reactor consists of a graphite block into which holes are drilled for the helium coolant and fuel rods. The fuel rod consists of a graphite matrix which acts as a binder to hold the fuel particles together. The pebble bed reactor fuel element consists of fuel particles dispersed in a graphite matrix which is surrounded by 5 mm of unfueled graphite. The fuel particles for both fuel elements are identical. The thorium BISO particle is composed of a spherical ThO_2 kernel in the center, surrounded by a buffer layer of porous carbon and a high density isotropic pyrocarbon shell which provides the primary containment for the fuel and fission products.

Because of its very high burn-up capability, the end of life for HTR fuel is determined by radiation damage to the fuel particles and to the graphite. Graphite materials exhibit rapid swelling in at least one direction when a certain fluence limit is reached. This fluence limit for end-of-life varies with the specific type of carbon material, its fabrication history, and its operating temperature. Fuel particle lifetime is limited by thermal migration of the fuel kernel, chemical attack of the coatings by fission products, buildup of fission gas pressures within the coatings, and radiation damage of the pyrocarbon coatings and the matrix material surrounding the particles. The critical limit on fuel particle lifetime will be that due to irradiation damage of the pyrocarbon coatings.

The radiation damage picture for graphite is shown in more detail in Fig. 1 for a typical near-isotropic extruded material — H-451 graphite. These curves show a limiting fluence of about 1.8×10^{22} at 800°C HTR conditions but suggest that the fuel life can be extended in a fission/fusion symbiotic system by operating the material at a lower temperature in the hybrid reactor. As an example, at 400°C, a lifetime of 4.5×10^{22} n/cm^2 would be projected. Experimental data on the cumulative effects of irradiation at changing temperatures are meager, however, and confirmation is needed. More isotropic, fine grained graphites have smaller neutron damage expansion rates, giving typically 50% more fluence lifetime.

The behavioral situation for fuel particle carbon coatings and the fuel rod graphite matrix

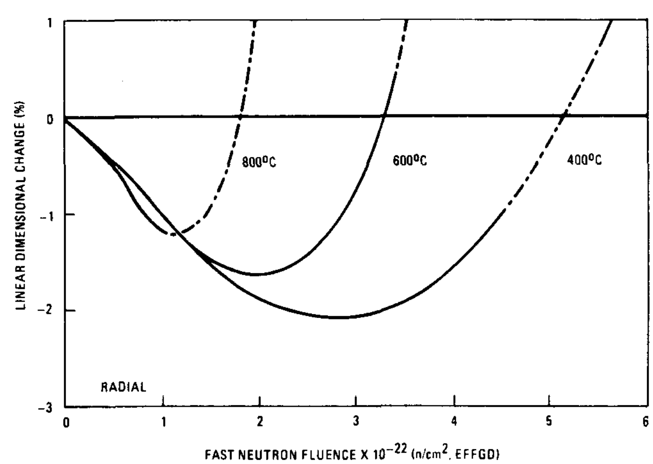

FIGURE 1. Dimensional changes in H-451 graphite as a function of fast neutron fluence, perpendicular to extrusion axis

material appears to be very similar to that for the fuel block graphite. This again suggests operation at lower temperatures to extend the lifetime. In this case however, irradiation behavior data do not exist below 1000°C and increased lifetime projections would need experimental confirmation.

In contrast to carbon materials, silicon carbide exhibits very stable irradiation behavior at fluences up to approximately 5×10^{22} n/cm^2 and at temperatures up to about 1000°C. Silicon carbide alloyed pyrocarbons could be used for particle coatings to increase fluence lifetime. It appears that 1.5×10^{22} could be achieved with SiC alloyed coatings under typical HTR conditions. Alternately, the coating thickness of the silicon carbide layer could be increased to the point where it becomes the structural unit and the outer carbon layer could be dispensed with, increasing the fluence limit to 1×10^{23} n/cm^2.

A point worth noting is that neutrons with energies between the fission spectrum and the 14 MeV fusion energy are about equally effective in displacement radiation damage to carbon. Thus the radiation damage rate per neutron in a fusion reactor should be comparable to that in a fission reactor for carbon and silicon carbide. No experimental data are available, however, to project the damage that might occur due to enhanced helium production from (n,α) reactions by the higher energy fusion neutron spectrum.

In summary, the present HTGR fuel design should provide a lifetime, including a 1.3 safety factor, of combined fusion-fission reactor fluence of about 1.4×10^{22} n/cm^2 Equivalent Fission Fluence for Graphite Damage (EFFGD) at 800°C. Modest changes through lowering operating temperatures, coating design, and graphite material selection should provide improvement to 2×10^{22} n/cm^2 EFFGD. Improvement by a factor of seven to a lifetime limit of 1×10^{23} n/cm^2 EFFGD appear possible through the extensive use of silicon carbide. These present limits and potential future limits are summarized in Table 1, and preliminary assessment of the particle designs required for survival of one, two and three fusion-fission cycles are given on Table 2 for the three HTR fuel types under consideration.

TABLE 1. HTGR fuel materials fluence limits

Component	Present Limit	Potential Limit
Graphite structure	1.4×10^{22} (800°C)	3.8×10^{22} by lower temperature 2.0×10^{22} by improved graphite 4.5×10^{22} by improved graphite and lower temperature
Coated particles	1.0×10^{22} (1000°C)	1.5×10^{22} by use of SiC alloyed pyrocarbon 4.5×10^{22} by lower temperature and possible use of SiC alloyed pyrocarbon 1.0×10^{23} or more by use of all SiC coatings
Matrix	$>1.0 \times 10^{22}$	4.5×10^{22} by lower temperature and possible use of improved materials

TABLE 2. Preliminary coated particle designs

No. of Cycles	Type of Fuel Element: Prismatic	Pebble Bed	Loose Particles
1	Ref BISO	Ref BISO	SiC
2	Alloyed SiC/PyC	Alloyed SiC/PyC	SiC
3	SiC	SiC	SiC

FUSION BLANKET ESTIMATES

In this study the full range of HTR fuel design possibilities was considered for the fusion blanket, including standard HTGR and pebble bed fuel which is characterized by a carbon to thorium atom ratio (C/Th) of 150, the heaviest thorium load the present HTGR and pebble bed fuel elements will carry (C/Th ≃ 80), and the very heavy thorium load that could be achieved by using coated particles of ThO_2 only with no graphite (C/Th = 12). Lithium-6 was added to suppress the thermal flux which would burn out the bred ^{233}U and to breed tritium. Previous studies of hybrid blankets have shown that the highest fuel production rate can be achieved in a blanket with a hard neutron spectrum.[5] Previous studies have also shown that a blanket of ^{238}U can achieve a higher fuel production rate than can one made of ^{232}Th, due to fast fission of the ^{238}U.[6] The potential blanket performance advantages of ^{238}U can be retained in a ^{233}U-producing thorium blanket by use of a thin "fission plate" of ^{238}U between the plasma and thorium breeding zone. The work of Lee at Lawrence Livermore Laboratory[7] indicates that a 5 to 10 cm thick fission plate of metallic uranium will double the number of neutrons incident upon the breeding zone, will multiply the incident fusion neutron energy by about a factor of five and will breed approximately 0.2 ^{239}Pu atoms per incident fusion neutron. While the use of a fission plate may enhance the fusion blanket performance significantly, the high power density may complicate blanket design and the fission product decay heat will require reliable emergency and shutdown heat removal systems.

The fusion blanket was assumed to consist of a 50 cm thick slab of the HTR fuels described above. The blanket was assumed to receive a neutron wall loading of 1.5 MW/m^2. The spatial details of the blanket were neglected and the results presented are average values for the blanket.

The calculational procedure used in this study was to do a detailed neutron spectrum calculation for each of the blanket designs using the General Atomic Company HTGR spectrum code MICROX to obtain accurate two-group microscopic cross sections for each material present: thorium, ^{233}U, lithium-6, oxygen and carbon. These cross sections were then used in two-group zero dimension static diffusion theory calculations to estimate the average nuclear performance of the blanket designs. The use of a fission plate was accounted for by use of an arbitrary source neutron multiplication factor of 2.0 and fusion energy multiplication factor of 5.8.[7] More accurate transport theory calculations are presently being done at Lawrence Livermore Laboratory and General Atomic Company to improve the accuracy of these blanket performance estimates.

To estimate the nuclear performance characteristics of the various blanket design options the following two-group, one region diffusion theory equations were used:

$$D_1 B_1^2 \phi_1 + \Sigma_{a1}\phi_1 + \Sigma_{12}\phi_1 = S + \nu_1 \Sigma_{f1}\phi_1 + \nu_2 \Sigma_{f2}\phi_2 \qquad (1)$$

$$D_2 B_2^2 \phi_2 + \Sigma_{a2}\phi_2 = \Sigma_{12}\phi_1 \quad , \qquad (2)$$

where D = diffusion coefficient,
B^2 = buckling,
ϕ = neutron flux,
Σ_a = absorption cross section,
Σ_{12} = cross section for scattering from group 1 to 2,
S = neutron source term,
Σ_f = fission cross section,
ν = neutrons released per fission,
Subscript 1 = fast neutrons,
Subscript 2 = thermal neutrons.

With a specified source, S, these equations were solved for the fast and thermal neutron fluxes, ϕ_1 and ϕ_2, which were then used to calculate various reaction rates to estimate the blanket performance. The results without fission plates are summarized in Figs. 2 through 4. The lithium-6 included in the fusion blanket designs serves two functions. It produces tritium to fuel the DT fusion reactor but its main function is to suppress the thermal flux to prevent the bred ^{233}U from being burned out. Figure 2 shows the uranium breeding ratio as a function of lithium load, as characterized by the Li/Th ratio, with and without uranium.

The lithium load should be minimized to achieve maximum uranium production. In order to achieve adequate ^{233}U concentration, however, a $^6Li/Th$ ratio of about 0.2 is required. The blankets with Li/Th > 0.2 all show an infinite medium multiplication factor, k_∞, substantially less than one, so criticality of the blanket is not a serious concern. At this 0.2 lithium-6 value, the tritium breeding ratio is only about 0.7. Regardless of the lithium-6 load, a tritium breeding ratio of 1.0 cannot be achieved by the HTR fuel blanket alone.

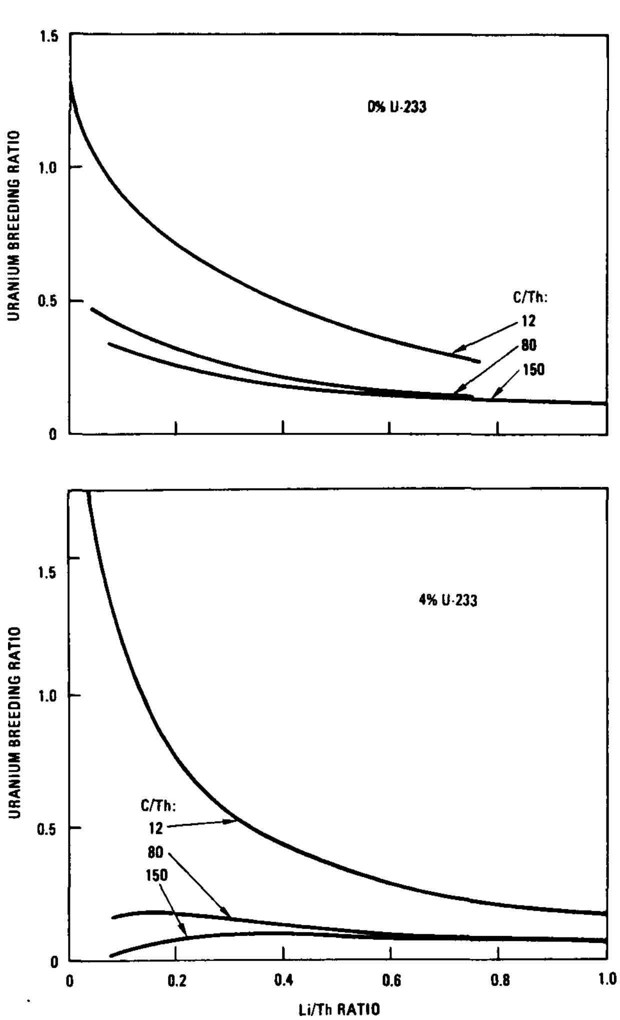

FIGURE 2. Uranium breeding ratio

The addition of a fission plate, however, would allow a ratio of 1 to be easily reached.

Selecting the ^{6}Li/Th = 0.2 point as a reference point, the nuclear performance of the blanket designs is plotted as a function of carbon/thorium ratio in Figs. 3 and 4. The uranium and tritium breeding ratios with 0 and 4% uranium concentration are shown in Fig. 3. The harder the neutron spectrum (lower C/Th), the more uranium is bred. The fast neutron fluence accrued during the time necessary to achieve 4% ^{233}U concentration, starting at 0%, and that needed to "refresh" the fuel from 3% to 4% are shown in Fig. 4. These values are used because they match the requirements of the HTR burner reactor. The softer spectrum designs, although

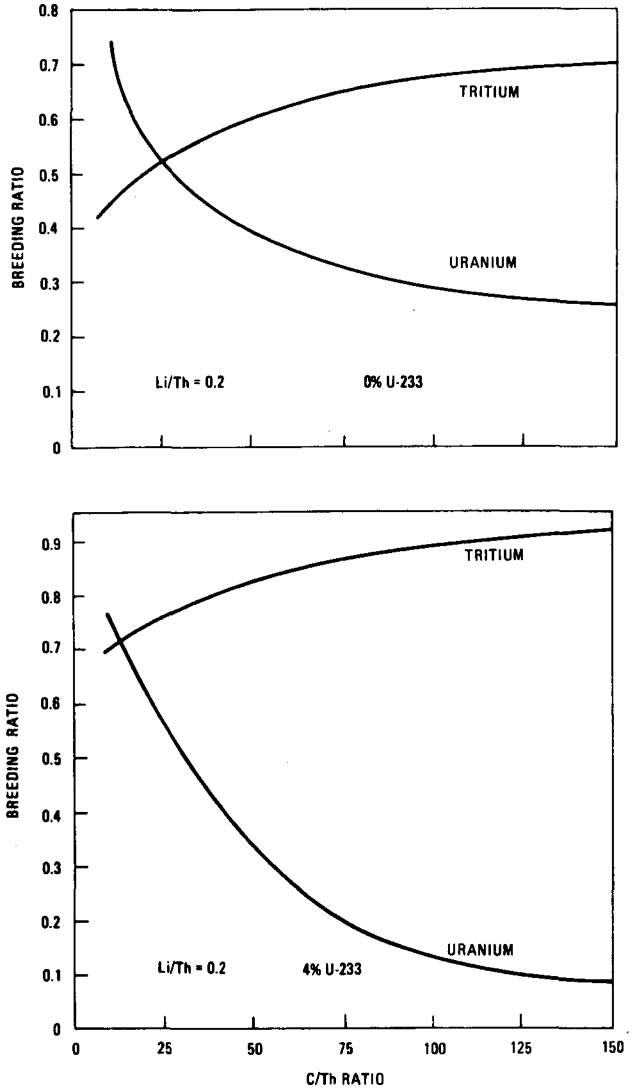

FIGURE 3. Uranium and tritium breeding ratios

producing less uranium, can achieve a 4% concentration sooner and with less fast neutron fluence to the fuel materials than can the low C/Th designs.

The blanket performance of four possible designs are shown on Table 3. Two breeding zone concepts, C/Th = 80 and C/Th = 12, are shown, each with and without a fission plate. The C/Th = 80 case corresponds to use of existing HTR fuel technology while the C/Th = 12 case may be thought of as the performance potentially available through use of the coated particles only.

Blankets are being developed for the doublet tokamak and tandem mirror fusion reactor concepts. Ball type fuel can be utilized directly in the

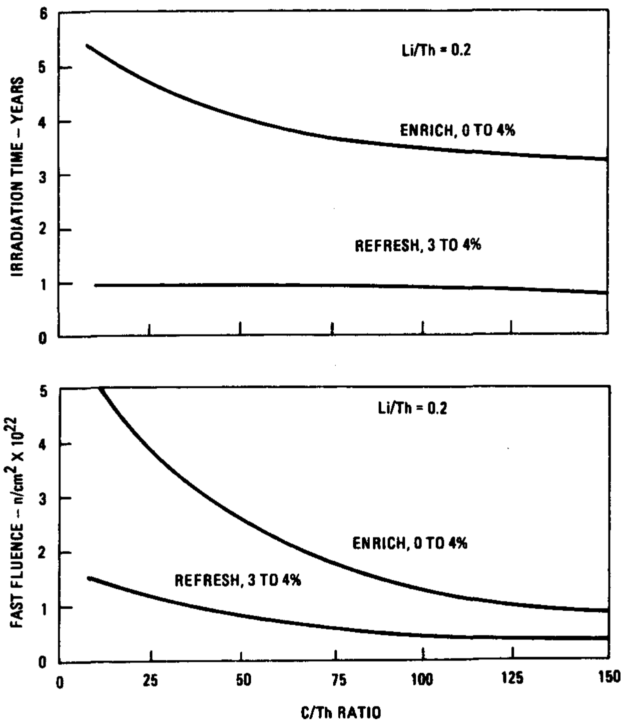

FIGURE 4. Irradiation time and fluence to enrich fuel

pebble bed HTR, and coated particle type fuel can be used in the HTGR reactor with fuel reconfiguration. Both ball and particle types of fuel are allowed to flow slowly from the top of the blanket to the bottom by gravity as shown in Fig. 5. The rate of material movement would be governed by the time necessary to irradiate the material to the required ^{233}U concentration. The blanket is sectioned parallel to the direction of flow to compensate for the non-uniform irradiation across the blanket by adjusting the fuel flow rate in different sections. For packed beds the minimum blanket section thickness required to prevent jamming of the moving balls or particles is about 5 diameters.

Although a fission plate may markedly improve blanket performance, it also poses a number of heat removal and structural problems which must be addressed. A neutronically optimum fission plate is relatively thin; in the present study the thickness was taken as 10 cm. A fission plate constructed of U_3Si and metal cladding has much higher material temperature limits ($\sim$ 900°C for the fuel and 700°C for the cladding) than the graphite of the blanket which is restricted to 400°C because of fluence concerns. Thermodynamically this is an advantage because the high grade heat of the coolant from the fission plate can be used to superheat steam for efficient power conversion. The achievement of a high coolant outlet temperature is difficult over a distance of only 10 cm. A design with the fuel contained in rods proved unsuccessful because the rods could not be packed sufficiently close to obtain a significant temperature increase of the coolant. A design with the fuel and cladding made into flat plates produced an outlet temperature of about 500°C,

TABLE 3. Blanket performance characteristics

	C/Th = 80, Li/Th = 0.20				C/Th = 12, Li/Th = 0.15			
	Without Fission Plate		With Fission Plate		Without Fission Plate		With Fission Plate	
Parameter	0% Uranium	4% Uranium	0% Uranium	4% Uranium	0% Uranium	4% Uranium	0% Uranium	4% Uranium
Breeding ratios (atoms/fusion)								
Uranium	0.32	0.18	0.64	0.36	0.78	0.94	1.56	1.88
Tritium	0.66	0.87	1.32	1.74	0.39	0.65	0.78	1.30
Blanket energy multiplication	1.1	4.7	7.0	14.2	2.2	9.0	9.2	22.8
Equilibrium ^{233}U concentration (%)	6.5		6.5		11.5		11.5	
Years to achieve 4% ^{233}U	3.7		1.8		4.5		2.25	
Fast fluence at 4% ^{233}U (n/cm^2)	1.4×10^{22}		1.4×10^{22}		4.8×10^{22}		4.8×10^{22}	
Fast fluence, 3% to 4% ^{233}U (n/cm^2)	0.7×10^{22}		0.7×10^{22}		1.4×10^{22}		1.4×10^{22}	

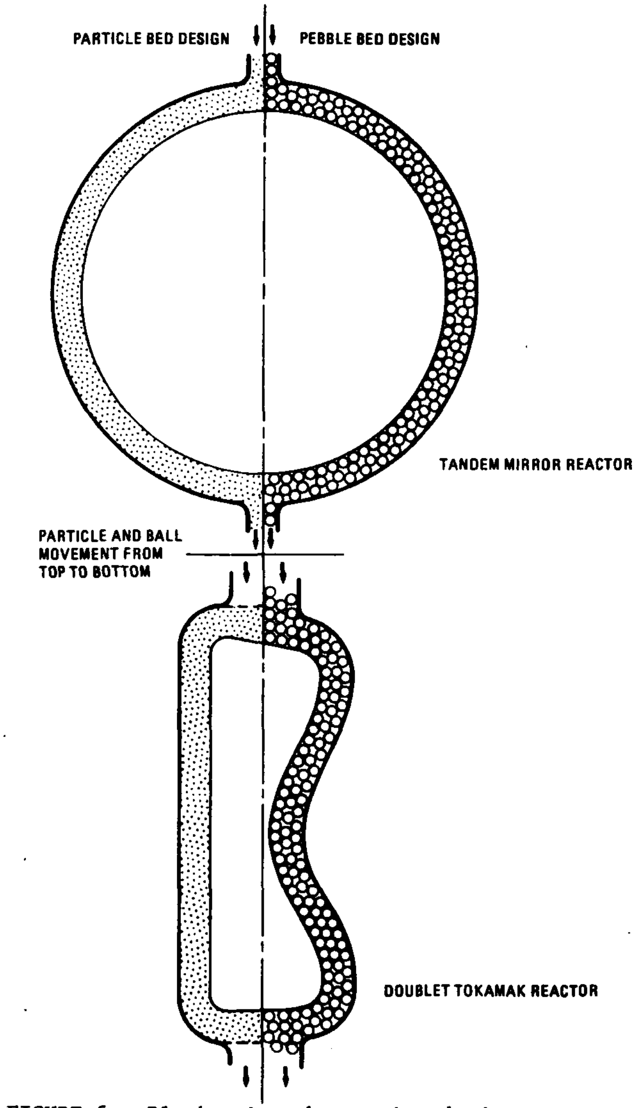

FIGURE 5. Blanket breeder region design concepts

comparable to that of gas-cooled fast reactors.(8) This high outlet temperature was achieved, within material temperature limitations, because the radial outward flow direction is optimum for the fission plate where the heat deposition is greater near the plasma (coolant inlet) than near the coolant outlet. Other flow orientations were considered but resulted in reduced thermodynamic performance.

A helium coolant pressure of about 60 atmospheres is required with gas cooling for adequate heat removal and minimal pumping losses. Because of the high coolant pressure it is necessary to pressure-equalize the plate-type fuel elements of the fission plate. Otherwise, the cladding will creep collapse onto the fuel. Another possibility is to provide pressure contact between the fuel and cladding at beginning of life. However, since the fission plate is designed for high burnup, there may be considerable swelling of the fuel which the cladding could not sustain. Thus, a pressure equalized design was chosen with adequate beginning-of-life clearance between fuel and cladding to accommodate the fuel swelling. Such a design is consistent with current gas-cooled fast reactor design practice.(8)

HTR FUEL CYCLE CALCULATIONS

The HTR is a very flexible reactor in terms of adaptability in its fuel use. As the main energy producer of a fusion/fission symbiosis, the HTR has to be optimized to achieve a high fuel utilization with the unreprocessed ^{233}U/Th feed material and operate on a low concentration of ^{233}U in the feed material.

In order to find some insight and understanding of the relationships between consumption, enrichment, inventory, etc., empirical formulas were developed from a large data base of similar HTR evaluations previously performed. The past experience is, however, not directly applicable since no data are available for use of unreprocessed ^{233}U feed material in HTRs. The empirical relationships derived from that past experience may be taken as indicative of trends, but the exact numerical values presented should be used with caution. Detailed HTR burnup calculations are now being done to confirm the results shown here.

Studies of high conversion ^{235}U-fed HTRs(9) showed that the conversion ratio can be well parameterized through the total thorium inventory and the irradiation period, the latter taking into account the fission product poisoning. The thorium inventory for a 1 GW(e) reactor is

$$\mathrm{Th} = 122{,}000\ (\mathrm{CR} + 0.02 \cdot \tau)^{3.94}\ \mathrm{kg/GW_e} \quad , \qquad (3)$$

where τ stands for the total effective irradiation period of the fuel (HTR and fusion blanket) expressed in years at 7 w/cc fission power density. Figure 6 shows this relation graphically and it

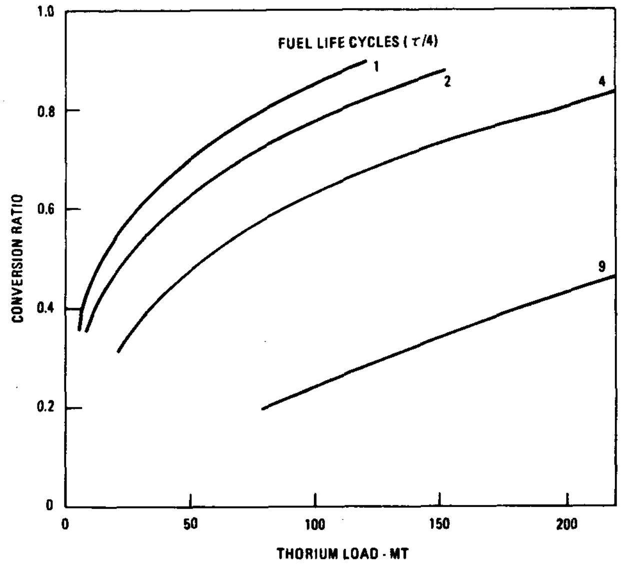

FIGURE 6. HTR conversion ratio - effect of number of enrichment - depletion cycles

points out that with an increased number of fuel life cycles (fusion re-enrichments) the conversion ratio drops due to fission product buildup.

The critical fissile uranium inventory of an HTR can be expressed as a function of the thorium load. The fissile uranium reactor inventory for a 1000 MW(e) HTR is about

$$U_{fissile} = 400 + Th/35 \text{ kg/GW}_e \quad , \tag{4}$$

where Th is the thorium inventory for a 1 GW(e) HTR defined above. Combining the expression for the uranium and thorium inventory allows one to calculate the fissile uranium inventory as a function of the conversion ratio and the number of fuel cycles.

The annual uranium requirement for a refresh cycle burner reactor consists of two parts. First, since more uranium is burned than produced with a conversion ratio smaller than unity, a net fissile uranium makeup of 866 (1 - CR) kg/GW$_e$ is required assuming ^{233}U fuel, 80% capacity factor and 40% thermal efficiency.[10] Second, fuel has to be added to compensate for the fuel being retired every year. In a graded fuel cycle where only a part of the fuel is discharged each reload, the end of cycle (EOC) discharge uranium load is reduced by half of that depleted each year so that

$$U_{discharge} = \frac{1}{R}\left[400 + Th/35 - \frac{866}{2}(1 - CR)\right] \text{ kg/GW}_e\text{-yr} \quad , \tag{5}$$

where R = fuel residence time in years. The fraction of the fissile inventory retired is inversely proportional to the number of life cycles M, times the HTR fuel residence time R. Therefore, the total annual ^{233}U requirements are

$$\begin{aligned} U_{consumed} &= 866(1 - CR) && \text{burned} \\ &+ \frac{1}{RM}\left[U_{fissile} - \frac{866}{2}(1 - CR)\right] && \text{retired} \end{aligned} \tag{6}$$

The uranium requirements are shown graphically in Fig. 7. The figure illustrates that minimum uranium requirements occur at conversion ratios of about 0.6. Higher conversion ratios reduce the fissile fuel burned but, due to the larger inventories, increase the amount of fissile fuel retired. Re-enriching fuel in the fusion reactor does reduce the uranium requirements somewhat. More than three fuel life cycles are, however, not desirable since the accumulated fission products reduce the conversion ratio and actually increase the uranium consumption.

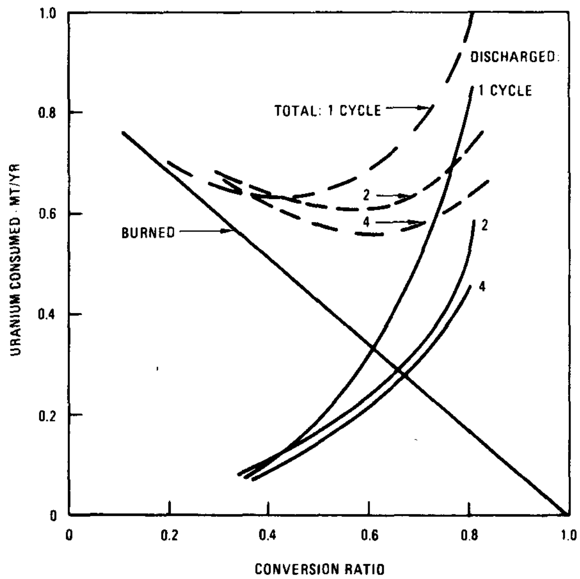

FIGURE 7. Uranium consumption

With the formulas so far established one can also calculate the necessary uranium concentration for the feed material for the HTR. Assuming a linear burnup and on-line refueling then the uranium concentration is about

$$U/Th = \left[866(1 - CR)/2 + U_{fissile}\right]/Th \quad . \qquad (7)$$

The Refresh Cycle HTR requires about 4% ^{233}U in thorium as feed material and discharges the spent fuel with about 3% ^{233}U concentration.

The empirical relationships defined above were developed using an extensive library of previous non-refresh HTGR studies as a basis. Preliminary fuel cycle computations for the Refresh Cycle HTR indicate that the ^{233}U concentration required can be reduced to about 3.5% and the amount of ^{233}U burned to about 150 to 200 kg/GW_e-yr. This is due to the low ^{234}U content in fusion-bred fuel and due to utilization of reduced power density and increased fuel residence time. This can allow the Refresh Cycle HTR ^{233}U requirement to be reduced below 400 kg/GW_e-yr.

SYMBIOTIC SYSTEM EVALUATION

Using the data developed above for the fusion breeder blankets and for the fission burner reactor, four conceptual symtiotic fusion power systems have been specified. These four power systems consist of two basic blanket concepts, each used with and without a ^{238}U fission plate. The fission plate significantly enhances the uranium breeding performance of the fusion blanket at the expense of added breeder reactor thermal power and added complexity to the blanket design. The two basic blanket concepts used represent existing technology and potential technology. The "existing technology" blanket is based upon C/Th = 80 HTR fuel and remains below presently expected graphite fluence limits by operating at modest temperature (400^{o}C) during irradiation in the fusion reactor. The "potential technology" blanket is based upon use of coated particles without any graphite in the fusion blanket which are incorporated into a removable graphite structure for subsequent depletion in the fission reactor.

Coupled with these fusion blanket design concepts is a fairly conventional HTR fuel cycle based on a C/Th = 175, which means that graphite will have to be added to the C/Th = 80 present technology fuel as well as to the C/Th = 12 potential technology fuel prior to irradiation in the fission reactor. If the C/Th = 80 fuel is in the form of pebble bed balls this can be accomplished easily by simply adding plain graphite balls to the fueled balls at a ratio of about 1 to 1. The symbiotic power systems formed in this exercise are not expected to be optimal but should give a reasonable estimate of the performance potential of the Refresh Cycle concept.

The system performance of the four conceptual symbiotic power systems, as characterized by the power produced by 1 GW of fusion plasma power, is summarized on Table 4.

TABLE 4. Symbiotic power system performance

Blanket	C/Th = 80 (Present Technology)		C/Th = 12 (Potential Technology)	
Fission plate?	No	Yes	No	Yes
Power, GW per 1 GW of fusion plasma power:				
Thermal power of fusion reactor	2.6	8.8	5.1	13.8
Thermal power of fission reactors	3.6	7.1	13.4	26.8
Total thermal power of symbiotic system	6.2	14.9	18.5	40.6
Total gross electric power of symbiotic system	2.2	5.5	7.4	16.2

Using the Refresh Cycle, a fusion hybrid reactor can support as many as 1.9 HTR burner reactors of equal thermal power. Factoring in the reduced ^{233}U feed requirements indicated by the preliminary Refresh Cycle HTR computations could increase this support ratio to 3.0.

IMPROVEMENT POTENTIAL WITH REPROCESSING

It should be noted that even with use of multiple refresh cycles, the fuel is discharged from the reactor and ultimately retired with a ^{233}U concentration of about 3%. This means that substantial amounts of valuable fissile material are retired. In addition, the no-reprocessing constraint pushes the optimum burner reactor conversion ratio for minimum net fuel consumption to fairly low values, typically in the range of 0.5 to 0.6. Since the HTR using ^{233}U fuel is easily capable of conversion ratios in excess of 0.8, this means that the burner reactors are not being used to their full potential. To take advantage of this potential, however, requires chemical reprocessing of the fissile fuel after discharge from the fission burner reactor. If this were to occur, the effective performance of the symbiotic fusion power system would improve markedly. The amount of uranium consumed then becomes simply the fuel burned as none would be retired. Assuming a conversion ratio of 0.85 in the HTR results in a ^{233}U consumption of 133 kg/yr per GW_e. This would allow a factor of 4.5 more fission burner reactors to be supported by each fusion breeder than were shown for the Refresh Cycle on Table 3.

CONCLUSIONS

This simple preliminary evaluation of producing ^{233}U in fusion reactors for use without reprocessing in fission reactors has shown that the no-reprocessing Refresh cycle concept appears to be technically feasible. This conclusion is important because it allows fusion power to offer a copious supply of fissile fuel for thermal burner fission reactors without violating any of the fuel cycle constraints that may be imposed due to concerns about nuclear proliferation. The thorium/uranium-233 fuel cycle may be used exclusively; virtually no plutonium need be produced. No initial fabrication or reprocessing of fissile material is necessary. At no time does fissile material exist in a weapons-grade form. That is, fissile material never exists without a substantial inventory of fission products. These features are important in that they allow the Refresh cycle symbiotic fusion power system to provide useful fissile fuel within possible nonproliferation constraints and guidelines.

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