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SCOPING STUDIES OF ^{233}U BREEDING FUSION FISSION HYBRID*

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Neutronic calculations have been carried out in order to design a laser fusion driven hybrid blanket which maximizes ^{233}U production per unit of thermal energy (a 1 kg/ MW_{th} -year) with acceptable fusion energy multiplication ($M_{\text{FE}} \sim 4$). Two hybrid blankets, a thorium and a uranium-thorium blanket, are discussed in detail and their performance is evaluated by incorporating them into an existing hybrid design (the LLL/Bechtel design). The overall performance of the two laser fusion driven ^{233}U producers is discussed and estimates are given of 1) the number of equivalent thermal power fission reactors (LWR, HWR, SSCR and HTGR) that these fusion breeders can fuel, 2) their capital cost, and 3) the cost of electricity in the combined fusion breeder-converter reactor scenario.

INTRODUCTION

For several years we have been investigating the potential for producing fissile fuel and electricity with laser fusion driven hybrids. Our earlier studies^(1,2) primarily used neutronic methods of analysis to identify attractive reactor concepts and to provide an upper bound estimate on performance. Our more recent hybrid design studies^(3,4) have dealt with the engineering and economic issues as well as the neutronic aspects. Over the last two years, Lawrence Livermore Laboratory and Bechtel National, Inc. have collaborated in a joint effort to conceptually design a laser fusion driven hybrid. The hybrid concept chosen for this design study was a depleted-uranium fueled fast-fission blanket which produced electricity and plutonium for consumption in existing light water reactors (LWR). The results of this joint effort have been extensively reported in the literature.⁽⁵⁾

It was encouraging to note that, in general, the attractive features that became apparent from our neutronic studies were regained at the higher level of engineering detail. Several other important features of laser fusion driven

hybrids emerged from the process of performing the more detailed engineering design. First, our design was shown to produce enough plutonium to fuel six LWRs of equivalent thermal power. Second, we found that laser fusion driven hybrids will cost two to three times as much as LWRs and probably more than fission breeder reactors. Finally, and most importantly, the cost of electricity in a scenario with hybrids providing fissile fuel for LWRs was shown to be insensitive to the capital cost of the hybrid. Electricity costs increased by only 20 to 40% over present costs when hybrid costs ranged between two to three times more than those of an LWR. The results led us to conclude that fissile fuel production is the most attractive role for a fusion-fission hybrid. Electricity should be produced in the hybrid only to the extent that it makes the fuel product cheaper. In our design, revenues from the sale of hybrid electricity reduced the cost of fissile fuel by approximately 50%, thereby reducing the cost of electricity in the combined hybrid-LWR scenario by approximately 20%.

In this paper we consider the design and analysis of laser fusion driven hybrids which

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produce ^{233}U for consumption in modified LWRs and advanced converter reactors such as heavy water reactors (HWR), spectral shift control reactors (SSCR), or high temperature gas reactors (HTGR). A ^{233}U producing hybrid is attractive for two reasons: First, ^{233}U is a more efficient fuel for thermal fission reactors and second, the ^{233}U -thorium fuel cycle may prove to be a less proliferating nuclear fission option than the uranium-plutonium fuel cycle.

The current administration has been concerned about the possibility of commercial nuclear power increasing the proliferation of nuclear weapons. As a result, the present national energy program calls for an indefinite postponement of reprocessing and recycling of plutonium, together with an intensive search for alternative reactors and nuclear fuel cycles, which do not allow direct access to materials used in nuclear weapons.

The hybrid scheme we propose to prevent proliferation would work similarly to that proposed by Felveson and Taylor⁽⁶⁾ for a fission economy composed of fast breeders and thermal converters. In this approach thermal fission reactors could be supplied with ^{233}U that has been diluted with ^{238}U to a point where it is no longer a nuclear weapon material. A fission fuel producing complex consisting of a fusion breeder, a fuel reprocessing facility, and a fuel fabricating facility would be located in a safeguarded area. The complex would provide denatured uranium to thermal burner reactors located outside secure areas. However, with ^{238}U in their fuel, the thermal burner reactors will produce some plutonium; therefore, their spent fuel must be removed and shipped back to the safeguarded fuel complex where the plutonium can be removed and either burned in the fusion breeders or stored. The larger the number of thermal burner reactors that can be fueled from a ^{233}U producing hybrid, the smaller the fraction of energy generation that must be located in secure areas under Government control. For example, the fraction is 5% if a thorium fueled hybrid can produce enough denatured ^{233}U to fuel 19 thermal burner reactors of

equivalent power. This should be contrasted with a fast breeder reactor which could provide fissile fuel for at most three thermal burner reactors.

Neutronic scoping calculations have been carried out in order to find a hybrid blanket design which maximizes ^{233}U production per unit of thermal energy and provides acceptable energy multiplication. The results of the neutronic analysis are presented in the next section and the two most attractive ^{233}U producing blankets resulting from these calculations have been incorporated into the LLL/Bechtel hybrid design. By incorporating these blankets into our existing design we have been able to estimate and evaluate the overall performance of laser fusion driven ^{233}U producers. In evaluating the performance of the two fusion breeders, we have provided estimates of 1) the number of fission converter reactors (LWRs, SSCRs, HWRs, and HTGRs) they can fuel, 2) their capital cost, and 3) the cost of electricity in the combined fusion breeder-converter reactor scenario. We have also investigated the sensitivity of the cost of electricity in the combined scenarios to fusion breeder capital costs, and to whether or not the fusion breeder produces electricity.

NEUTRONIC SCOPING STUDIES

The primary objective of our neutronic scoping studies was to find a blanket configuration which maximized ^{233}U production per unit of thermal energy while maintaining an adequate tritium breeding ratio. We also wanted to maximize fusion energy multiplication but not at the expense of the primary objective. Specifically, we wanted a blanket design producing more than 1 kg of ^{233}U per MW-year of thermal energy while maintaining a tritium breeding ratio greater than one and multiplying the fusion energy by as much as a factor of four. With fusion energy multiplications in the neighborhood of four, a fusion breeder can operate with fusion energy gains that are as much as six times lower than required for pure fusion power production.

The exploratory calculations were performed in spherical geometry with the fission zone sandwiched in between a sacrificial first-wall and a tritium breeding region. The design and composition of the first wall and tritium breeding regions were taken from our latest hybrid design.⁽⁴⁾ The first wall is a thin sacrificial liner of graphite which is supported by a molybdenum structure. The tritium breeding region consists of a thick (50 cm) graphite region which is cooled on both sides with liquid lithium.

Two liquid-metal cooled fission blankets which produce ^{233}U have been considered in this study: a thorium (Th) blanket and a uranium-thorium (U-Th) blanket. A schematic representation of the U-Th blanket is shown in Fig. 1. As shown it consists

FIGURE 1. Schematic representation in spherical geometry of a ^{233}U breeding hybrid blanket.

of a uranium fast-fission zone followed by a moderator region and a Th blanket. The performance of both the Th and the U-Th blankets has been investigated as a function of parameters such as respective zone thickness, fission fuel type (metal or carbide), coolant type (sodium, lithium, enriched lithium), and moderator type (beryllium or graphite). The volume fractions of fuel, coolant, and structure in the fission blankets have been selected to be consistent with compositions used in the LLL/Bechtel hybrid. In all cases a ferritic steel (2 1/4 Cr-1 Mo) was used as the fuel cladding and blanket structural material.

All the neutronic calculations were carried out with ALICE,⁽⁷⁾ a Monte Carlo neutron trans-

port code which uses 175 energy group cross sections from the ENDL library and the probability table method to account for resonance self shielding. The 14-MeV neutron source in these calculations was assumed to be uniformly distributed in the compressed fuel region of a fusion target with a density-radius product $\rho R = 3 \text{ g/cm}^2$. The fuel region was surrounded by a 0.056-cm thick pusher region ($\rho R = 1.5 \text{ g/cm}^2$ 90% LiD, 10% Pb).

The blankets with the best performance in the spherical neutronic calculations were analyzed further by reconfiguring them to closely approximate the engineering features and cylindrical geometry of LLL/Bechtel hybrid. The results from the more realistic two dimensional neutronic calculation were used to evaluate the overall performance of a laser fusion driven ^{233}U producers.

These fusion breeders were then compared to the reference LLI/Bechtel hybrid design which is a plutonium producer.

THE THORIUM BLANKET

Neutronic calculations were carried out on several thorium metal blankets in order to determine their ^{233}U production, tritium breeding ratio and fusion energy multiplication. We looked at three coolants: sodium, lithium, and enriched lithium. Table 1 gives the results of the calculations for a 20-cm thick Th blanket and Fig. 2 gives the production of ^{233}U and tritium as a function of thorium zone thickness. All the calculations were carried out in spherical geometry with the thorium zone containing, by volume, 53% thorium metal, 12% steel and 35% coolant. Graphite and beryllium were compared as moderators in the lithium blanket located behind the thorium zone. From these results it can be seen that the highest ^{233}U production is obtained when the blanket is cooled with sodium. Unfortunately the production of tritium is too low; in fact, we were unable to find a sodium cooled Th blanket that provided a tritium breeding ratio ≥ 1.0 . With lithium as the coolant, the ^{233}U production decreases as the enrichment of ^6Li increases. The values listed in Table 1 show that a 20-cm thick

TABLE 1. Performance parameters of a 20-cm Th blanket (53% Th, 12% SS and 35% coolant) as a function of the type of coolant and moderator.

Coolant	Moderator	E_n	M_F	TBR	$^{232}\text{Th}(n,\gamma)$	^{233}U kg/ $\text{MW}_T\text{-year}$
Na	C	31.5	1.99	.620	.806	1.75
Na	Be	32.7	2.06	.679	.811	1.70
Li (natural)	Be	32.1	2.03	.831	.726	1.55
Li (20% ^6Li)	Be	32.4	2.05	.956	.606	1.28
Li (50% ^6Li)	Be	32.2	2.03	1.149	.443	0.95

FIGURE 2. Dependence of tritium breeding and ^{233}U production on blanket thickness and coolant selection for a thorium blanket.

Th blanket cooled with natural lithium produces 1.55 kg of ^{233}U per MW-year of thermal energy, while lithium enriched to 50% ^6Li reduces this value to 0.95 kg/MW_T-year. However the reduction on ^{233}U production is accompanied by a higher TBR, which increases from 0.831 with natural lithium to 1.15 with 50% enriched ^6Li . The choice of beryllium over graphite as a moderator increases the production of either tritium or ^{233}U by about 10%.

The blanket energy multiplication, M_F , defined as the ratio of blanket thermal energy to fusion

energy was found to be rather insensitive to the thickness of the thorium region, increasing from 1.9 and 2.2 when the thorium thickness ranges from 10 to 45 cm. M_F was also insensitive to coolant and moderator choices. The energy multiplication of these Th blankets is a factor of two lower than we would have liked. This led us to explore the performance of the U-Th blanket.

URANIUM-THORIUM BLANKET

The U-Th blanket that we have considered in these calculations is shown in Fig. 1. The uranium-fueled fast-fission blanket serves a dual purpose: It multiplies the fusion energy and increases the neutron population. Our previously reported results⁽⁸⁾ have shown that a depleted U blanket from 4 to 15 cm thick will multiply the fusion neutron energy by a factor of 2 to 7 and more significantly, the number of neutrons leaving the blanket will be 1.5 to 1.9 times larger than the incident fusion neutrons. The production of ^{233}U in the U-Th blanket was found to depend on several factors: The composition and thickness of the uranium zone, the composition and thickness of the moderator region between the uranium and the thorium regions, and the composition and thickness of the thorium region. Neutronic calculations in spherical geometry were carried out with the uranium and thorium zones containing, by volume, 53% fuel, 35% lithium coolant, and 12% steel. In these scoping calculations the production of ^{233}U was maximized for a uranium zone which consisted of either 6 cm of metal fuel or 8 cm of carbide fuel. Beryllium was found to be better than graphite in the region between the uranium and thorium zones with 10 cm being the

optimum thickness. The use of enriched lithium as the coolant in the U blanket was found to significantly improve the ^{233}U production performance of all the U-Th systems. This occurs because more tritium is produced in the uranium zone at the expense of reduced ^{239}Pu production, thereby allowing more ^{233}U to be produced in the thorium region while maintaining a tritium breeding ratio greater than one.

The ^{233}U production, blanket energy multiplication, and tritium breeding ratios from the more attractive U-Th blankets are plotted as a function of thorium thickness in Fig. 3. The uranium zone

FIGURE 3. Dependence of the ^{233}U production on the thorium thickness for a uranium-thorium blanket, where the uranium zone is 6 cm of uranium metal or 8 cm of uranium carbide.

thickness was fixed at 6 cm for metal fuel and 8 cm for carbide fuel. Lithium enriched to 50% ^6Li was used as the coolant in the uranium zone and natural lithium was used as the coolant in the thorium zone. The moderator was 10 cm of beryllium.

As shown, ^{233}U production increases from 0.85 to 1.1 kg/MW_t-year as the thorium zone thickness increases from 20 to 50 cm and is approximately the same for both types of uranium fuel. ^{239}Pu production remains relatively constant at approximately 0.7 kg/MW_t-year. Similarly, blanket energy multiplication is independent of thorium thickness, averaging 4.8 for metal fuel and 4.2 for carbide fuel. These energy multiplications are in line with the desired performance for the design. Tritium breeding decreases with increasing thorium thickness, however, it is adequate for thicknesses up to 50 cm. The results of a calculation with an 8-cm thick zone of uranium metal have also been plotted in Fig. 3 to illustrate the decrease in ^{233}U production as a result of the thicker uranium zone. Figure 3 also shows the performance of a U-Th blanket which uses spent fuel from a LWR in carbide form. This blanket exhibits excellent ^{233}U production and fusion energy multiplication. It is especially interesting since it provides some indication of the time iterated performance of a U-Th blanket.

TWO DIMENSIONAL CALCULATIONS

Two dimensional (2-D) calculations were carried out on the most attractive Th and U-Th blankets obtained from the spherical calculations. In these 2-D calculations, the ^{233}U producing blankets were reconfigured to closely approximate the engineering features and cylindrical geometry of the LLL/Bechtel hybrid. The fission blanket in these calculations is a 10-m high cylindrical shell with a radius of 5.93 m. The first wall is a sacrificial liner of graphite blocks which are attached to and supported by a lithium-cooled molybdenum structure. Lithium-cooled tritium breeding blankets are positioned in the top and bottom of the reactor and behind the radial fission blanket. The fission blankets selected for the 2-D calculations were a 35-cm thick thorium blanket cooled with natural lithium and a U-Th blanket with a 5-cm thick zone of uranium metal fuel which is cooled with enriched lithium (40 wt% ^6Li).

A side view of the U-Th radial fission blanket is shown in Fig. 4. The uranium fission zone is

FIGURE 4. Side view of fission blanket segment in laser fusion hybrid reactor.

composed of one row of hexagonally shaped steel process tube containing a 7-rod cluster of steel clad uranium-metal fuel pins. The moderator between the uranium and the thorium zones is a 10-cm thick region of beryllium. The thorium zone consists of three rows of steel process tubes containing a 19-rod cluster of steel clad thorium metal fuel pins. In the Th blanket system, the uranium and beryllium zones have been eliminated.

It was possible to use natural lithium as the coolant in the thorium system, even though the spherical calculations indicated an inadequate TBR, because the tritium breeding blankets positioned at the top and bottom of the reactor provided additional tritium. These tritium breeding blankets are described in Ref. 2. Combined, they cover 38% of the total surface area and they provide a TBR of 0.72. With the U-Th blanket, the

tritium production from the reactor is as high as 1.30 when the top and bottom blankets are included in the calculations. As a result, increased ^{233}U production was obtained by decreasing the uranium-zone thickness from 6 to 5 cm.

The composition of the fuel zones used in the 2-D calculations are slightly different than the ones used in spherical geometry. By volume they are as follows: for the Th blanket, 57.04% Th, 13.18% Fe, 25.93% Li and 3.75% Na; for the uranium-zone, 41.03% U, 15.12% Fe, 29.18% Li and 14.16% Na; for the thorium zone in the U-Th blanket, 60.14% Th, 13.18% Fe, 22.29% Li and 3.89% Na.

The performance parameters of the two blankets are listed in Table 2 and they are discussed in detail in the next section.

ANALYSIS OF THE FUSION BREEDERS

Laser and target parameters which appear feasible from the current understanding of the technology have been selected in order to evaluate and compare the operational performance of the Th and U-Th fusion breeders. The selected laser/target parameters and the resulting performance of the two ^{233}U producers are presented in Table 2. In both designs fusion targets producing 100 MJ of fusion energy are irradiated by a 1-MJ laser at the rate of 13 Hz. The 1300 MW_f of fusion power is multiplied by 1.77 in the Th blanket and by 2.53 in the U-Th blanket yielding 2300 and 3290 MW of thermal power, respectively. The higher blanket energy multiplication of the U-Th blanket leads to a fusion breeder which produces twice as much net electricity (671 MW_e vs 335 MW_e). This excess electricity can be sold to reduce the price of the ^{233}U produced. However, this enhanced energy multiplication is gained at the expense of a 40% reduction in ^{233}U production per unit of thermal energy.

The ^{233}U production capabilities of both fusion breeders can be put in perspective by looking at the number of thermal burner reactors which they can fuel. Figure 5 gives the number of equivalent thermal power LWRs, SSCRs, HWRs, and HTGRs that each of the fusion breeders can fuel. The ^{233}U

TABLE 2. Operational Performance of Two Laser Fusion Driven ^{233}U Producers

<u>Laser and Target Parameters</u>		
Laser Energy		1.0 MJ
Pulse Repetition Rate		13 Hz
Pellet Gain (Q)		100
Pellet Yield		100 MJ
Laser System Efficiency (η_L)		3%
Number of beams		4
<u>System Performance</u>		
	<u>Thorium</u>	<u>Uranium-Thorium</u>
Thermal Power	2300 MW _T	3290 MW _T
Thermonuclear Power	1300 MW _T	1300 MW _T
Plutonium Production	0	0.61 kg/MW _T -yr
^{233}U Production	1.9 kg/MW _T -yr	1.10 kg/MW _T -yr
Gross Electrical Power ($\eta_T = 0.36$)	828 MW _e	1184 MW _e
Laser Input Power	433 MW _e	433 MW _e
Auxiliary Power	60 MW _e	80 MW _e
Net Electrical Power	335 MW _e	671 MW _e
Recirculating Power Fraction	0.60	0.43
System Efficiency	14.6%	20.4%
Avg. Blanket Energy Multiplication	1.77	2.53
Avg. Tritium Breeding Ratio	1.05	1.15
Avg. First-Wall Neutron Flux	2.35 MW/m ²	2.35 MW/m ²

consumption rates for the LWR, SSCR, and HWR were obtained from Ref. 9. The consumption rate for the HTGR was based on a system with a conversion ratio of 0.85. It should also be noted that all the consumption rates are for denatured uranium (12% ^{233}U in U). In contrast to the fueling performance of the ^{233}U producing hybrids, the LLL/Bechtel hybrid produces 0.88 kg/MW_T-yr of ^{239}Pu . This is enough ^{239}Pu to fuel 6.25 LWRs, 7.73 SSRs, 10.1 HWRs and 6.6 HTGRs of equivalent thermal power. It is also interesting to compare the ^{233}U production rates of the Th and U-Th blankets to those of an LMFBR. The ^{233}U production rates for oxide, carbide and metal fueled LMFBRs have been estimated in Ref. 9. These fission breeders produce 0.077, 0.115 and 0.134 kg/MW_T-yr of ^{233}U , respectively.

COST ANALYSIS OF THE DESIGNS

With a reasonable estimate of the capital cost for Th and U-Th fusion breeders, it will be possible to estimate the cost of ^{233}U from these

facilities. This will allow us to estimate both the cost of ^{233}U from these facilities and the cost of electricity from fission burner reactors which use the fusion breeder ^{233}U . Since both of these blanket designs are quite similar to the LLL/Bechtel hybrid their cost has been estimated by extrapolating from the most recent costing of the reference blanket.⁽⁴⁾ The cost of the LLL/Bechtel hybrid⁽⁴⁾ without a laser and a target fabrication facility was shown to be approximately 1.75 times more expensive than a LWR on a $\$/\text{KW}_T$ basis. Major differences between LLL/Bechtel hybrid and the fusion breeders in this paper include:

1. Sodium has been replaced by lithium as a coolant in the fission zones of the ^{233}U producers.
2. Stainless steel has been replaced by ferretic steel as the structural and fuel cladding material in the ^{233}U producers.

FIGURE 5. Number of equivalent thermal-power fission reactors that the thorium and uranium-thorium fusion breeders can fuel.

3. The ^{233}U producers have lower blanket power densities for a given first wall flux.
4. The ^{233}U producers have less thermal power than the 4000 MW_t reference system. (4)

The first two points should make the ^{233}U producers less expensive than the reference system while points three and four should make them more expensive. The effects of less power output can be seen by noting that 600 MW_e -LWR costs approximately 45% more than a 1200 MW_e -LWR on a $\$/\text{KW}_t$ basis. From the comparisons made here we have estimated the following cost ranges for the Th and U-Th fusion breeders less lasers and target fabrication facilities.

Th system cost: $1.7\text{--}2.2 \times \text{LWR}$

U-Th system cost: $1.5\text{--}2.0 \times \text{LWR}$

The capital cost range of the thorium system was made slightly higher to account for its lower output power. The reference LWR being used here

costs $284\$/\text{KW}_t$ in 1978 dollars. It was costed on the basis of a 1200 MW_e plant which is ordered in 1978 for delivery in 1987. This reference LWR produces electricity at 35.2 mils/ KW-hr with a 16% rate of return on capital, 3.66 mils/ KW-hr operation and maintenance charge, and 43 $\$/\text{lb}$ uranium. With 100 $\$/\text{lb}$ uranium, it will produce electricity for 40.1 mils/ KW-hr .

Detailed cost studies have not been performed on the laser system or target fabrication facility. If one allows \$400M for the laser and \$100M for the target fabrication facility, the cost range of the thorium fusion breeder becomes 2.5 to 3.0 times the reference LWR and for the U-Th system the cost range becomes 2.0 to 2.5 times the reference LWR on a $\$/\text{KW}_t$ basis. The effect of these capital cost ranges on the cost of ^{233}U from the Th and U-Th fusion breeders is shown in Table 3. As shown, the cost of ^{233}U ranges from 65 to 80.9 $\$/\text{kg}$ for the Th system and 69.6 to 97.0 $\$/\text{kg}$ for the U-Th system. The last entry in Table 3 gives an indication of how much the cost of ^{233}U increases when the excess electricity is not sold.

The cost of electricity from fission burner reactors which have been supplied with ^{233}U from the Th and U-Th fusion breeders is shown in Fig. 6. In order to determine the cost of electricity from the HWR and SSCR relative to the LWR, the following estimates obtained from Ref. 10 were used: 1) the capital cost of the HWR was estimated to be 15% higher than the reference LWR with an additional cost of $\$160/\text{KW}_e$ for heavy water, and 2) the capital cost of the SSCR was estimated to be 5% higher than the reference LWR, with an additional cost of 40 $\$/\text{KW}_e$ for the heavy water reactor inventory. Finally, the ^{233}U -Th fuel cycle contribution to the cost of electricity for all the fission reactors has been estimated on the basis of the data provided in Table 4. The refabrication, spent fuel shipping, and reprocessing costs were compiled and averaged from Refs. 11-14. The high refabrication and reprocessing costs for ^{233}U -Th fuels results from the requirements to remotely handle radioactive fuel in shielded facilities.

The results shown in Fig. 6 are encouraging.

TABLE 3. Cost Summary for the ^{233}U Producing Fusion Breeders

Capital Cost Estimate	Thorium Hybrid		Uranium-Thorium Hybrid	
	2.5 x LWR	3.0 x LWR	2.0 x LWR	2.5 x LWR
Revenues required for capital cost at 16% rate of return, \$/KW-yr	114.2	137.0	92.5	115.2
Revenues required for fusion breeder fuel cycle,* \$/KW-yr	12.1	12.1	12.1	12.1
Revenues from sale of electricity at 35 mills/KW-hr and 75% capacity factor, \$/KW-yr.	-33.5	-33.5	-46.9	-46.9
Cost of ^{233}U at 75% capacity factor, \$/g	65.0	80.9	69.6	97.0
Cost of ^{233}U at 75% capacity factor with no revenues from electricity, \$/g	88.4	104.3	126.1	153.5

*For thorium fuel, fabrication = 30 \$/kg, reprocessing = 300 \$/kg, burnup = 10 MWD/kg.

For uranium fuel, fabrication = 30 \$/kg, reprocessing = 170 \$/kg, burnup = 6 MWD/kg.

TABLE 4. Fuel Cycle Cost Bases for Thermal Reactors Using ^{233}U -Th Fuels

Refabrication	\$300/kg HM	LWR, SSCR
	\$200/kg HM	HWR
Spent Fuel Shipping	\$20/kg HM	LWR, SSCR
	\$15/kg HM	HWR
Reprocessing	\$300/kg HM	LWR, SSCR, HWR
Plant Capacity Factor	75%	LWR, SSCR
	80%	HWR
Fixed Charge Rate on Capital	16%	
Present Worth Discount Rate	10%	
Basic Inflation Rate	6%	

They indicate that fusion breeder fueled thermal burner reactors can supply the world with reasonably priced electricity for hundreds of years. The cost of electricity from LWRs and SSCRs is quite similar in the cost range of fusion breeder produced ^{233}U . When these two thermal reactors are provided with ^{233}U costing from 65 to 100 \$/g, they produce electricity which costs from 38.5 to 44 mills/kw-hr. This range of values is only 10 to 25% higher than the cost of electricity from the reference LWR with 43 \$/lb uranium. Unfortunately the HWR with the best ^{233}U utilization (Fig. 5) produces electricity at a higher cost than the other two reactors: 46.5 to 51 mills/kw-hr when the price of ^{233}U ranges from 65 to 100 \$/g. We had expected the cost of electricity from HWRs to be more expensive at lower ^{233}U prices but less sensitive to ^{233}U price increases. Instead we found that the higher fuel cycle costs associated with the HWR's lower fuel burnup limits and higher ^{233}U inventory requirements outweighed the cost reduction from its better utilization.

FIGURE 6. The cost of electricity from fission burner reactors with fusion breeder produced ^{233}U .

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