Laser Fusion Hybrids - Technical, Economic and Proliferation Considerations

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G.A. Moses, R.W. Conn, and S.I. Abdel-Khalik

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

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G. A. Moses
R. W. Conn
S. I. Abdel-Khalik

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Fusion Research Program
Nuclear Engineering Department
University of Wisconsin
Madison, Wisconsin 53706

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ABSTRACT

A possible role for the fusion-fission hybrid in the context of an immediate nuclear future that may not include fuel reprocessing or the LMFBR has been examined. In such a role, the hybrid is used to irradiate fertile fuel assemblies, thereby simultaneously enriching the fuel to the proper fissile concentration and rendering it proliferation resistant by making the fuel highly radioactive. Should reprocessing of spent LWR fuel be allowed, this hybrid concept can be incorporated into an internationally monitored, physically secure fuel production and reprocessing center that meets non-proliferation guidelines.

In the SOLASE-H study, a laser fusion hybrid is conceptually designed to meet the needs of the proliferation resistant fuel cycle. One hybrid operating at a fusion power of 1200 MW can fuel approximately 2.5 1000 MW_e LWRs requiring 4% enriched ^{233}U fuel. (With reprocessing this hybrid can fuel 10 LWRs.) The assemblies can be enriched to 4% fissile content in 1.9 years in an optimally designed case. The fuel burnup level in the hybrid itself is equivalent to 4300 MWD/MT. Flat enrichment profiles across the assembly can be achieved at the expense of the ^{233}U breeding ratio. A figure of merit that maximizes the ^{233}U breeding ratio subject to minimizing the peak to average enrichment (the hot spot factor) determines the optimum blanket design.

The substantial fusion power required to produce fissile fuel does not allow the laser fusion pellet gain (and hence laser energy) and pulse repetition frequency to be simultaneously relaxed. Laser efficiency can be

substantially relaxed due to the blanket energy multiplication. The hydrogen-fluoride laser, studied for the SOLASE-H system, appears to be scalable to an output energy of 2 MJ and the calculated net efficiency of 2.6% indicates HF is an attractive laser candidate for laser fusion hybrid applications.

1. INTRODUCTION

A fusion-fission hybrid reactor utilizes the 14.1 MeV DT fusion neutrons for breeding fissile material in the hybrid reactor blanket. This bred fuel can be removed periodically from the blanket and burned in conventional fission reactors or it can be burned "in situ" in the hybrid blanket itself. During the last five years there have been many studies of hybrids for a variety of fusion systems (i.e. tokamak, mirror, laser fusion, electron beam fusion).[1-7] Such fusion-fission hybrid reactors appear to be attractive because they produce two revenue sources, electric power and fuel for conventional fission reactors, at a fusion performance level that is less than that required for pure fusion reactors. The additional revenue source, fissile fuel, strengthens the economic perspective of the fusion system. The reduced fusion performance is allowable because the 14.1 MeV DT neutron energy is multiplied in the hybrid blanket by the fission process. For the hybrid operating as a fuel factory where fissions are minimized, the blanket energy multiplication is still typically 2-10. In the second option where the fuel is allowed to burn in the hybrid itself, the multiplication may be as high as 40-50 depending on how close the blanket approaches criticality. It is argued that this relaxation of the fusion energy requirement may allow hybrid reactors to make an impact on the world's energy production problem at an earlier date than pure fusion reactors. However, the hybrid reactor may also appear unattractive if it is considered to have both the disadvantages of complex fusion systems and the radioactive waste, criticality, and proliferation problems of fission reactors.

The SOLASE-H[8] laser fusion hybrid reactor study investigates the possibility of minimizing the perceived disadvantages of the hybrid by operating with a low k_{eff} in the blanket, and utilizing a proliferation resistant fuel cycle that allows direct enrichment of PWR fuel assemblies in the hybrid and transfer of the irradiated assemblies to the fission reactor, without intermediate reprocessing. The study established the potential role of the hybrid for a nuclear future that includes no immediate reprocessing or development of the LMFBR. This study is, in fact, a continuation of the SOLASE[9] conceptual laser fusion reactor design, reported by R.W. Conn at this meeting last year.

In the following sections, there first appears a generic discussion of the proliferation resistant fuel cycle. Any fusion system might be used to produce the fuel. This is followed by a description of the SOLASE-H laser fusion hybrid system. The final section summarizes the conclusions derived from the SOLASE-H study.

2. NON-PROLIFERATION POTENTIAL OF FUSION HYBRID REACTORS

Only 0.7% of natural uranium is the fissile ^{235}U isotope. The remaining 99.3% is ^{238}U. Other fissile isotopes can be manufactured by the absorption of a neutron in ^{232}Th and ^{238}U to produce ^{233}U and ^{239}Pu respectively. Once these artificial fissile materials have been produced they can be mixed with their corresponding fertile material at a 3-4% concentration and fabricated into fuel assemblies for use in fission reactors. The production of these artificial fissile isotopes is, of course, the purpose of the fusion hybrid reactor. However, the reprocessing of the material produced in the hybrid to remove fission products and the fabrication into cold, clean fuel assemblies exposes the hybrid fuel cycle to the same proliferation considerations as the fast breeder fuel cycle. Because this fuel is easily handled and the fissile material can be removed by chemical rather than physical processes, the fuel is most vulnerable to diversion for the purpose of nuclear weapons development. Feiverson and Taylor[10,11] have argued that spent or highly radioactive fuel is self-protecting. Such assemblies weigh nearly half a ton. They argue that stealing such irradiated assemblies would require heavy cranes, tons of shielding containers, and a large vehicle for transporting the stolen, shielded assemblies. Further, the fissile ^{233}U or ^{239}Pu must still be separated from the dangerously radioactive fuel.

The fusion-fission hybrid fuel cycle proposed here directly enriches the fertile fuel to 3-4% fissile concentration in the hybrid blanket. This process also makes the fuel highly radioactive so that it is rendered diversion resistant. The details of this fuel cycle are outlined in Fig. 1.

The cycle includes four steps:

1. Fertile fuel, ThO_2 or UO_2, is fabricated in a form that is directly usable in a LWR. (Other fission reactors could be included but the LWR is used here because it is the workhorse of the U.S. fission reactor industry.)

2. The cold, clean fuel assemblies, containing only fertile fuel, are placed in the hybrid blanket and carefully enriched to a nearly uniform concentration of 3-4% fissile fuel as required by the LWR.
3. The enriched, and now highly radioactive assemblies, are transferred as units directly to the LWRs for burning of the fuel.
4. The spent fuel from the LWR is stored until a decision is made on reprocessing or storing or both. If feasible, the spent fuel can be re-inserted into the hybrid to be re-enriched for further burning in the LWR. This possibility depends on both the importance of fission product buildup to LWR performance and the radiation damage to the fuel and cladding.

The attractive features of this cycle are the following:

1. The system is resistant to diversion because fissile material occurs only inside highly radioactive fuel assemblies. Only fresh fertile material is fed to the hybrid, and upon removal, the fuel pellets contain fission products that are highly radioactive and the pellets themselves are contained in rod assemblies with highly activated cladding. Access to the fissile material is thus very difficult making the entire cycle proliferation resistant according to the guidelines of Feiverson and Taylor.
2. The fissile fuel reserves are extended substantially. If the average LWR fuel enrichment is assumed to be 3%, the fissile fuel reserves are extended by 4.3 x (Thorium Resources+Uranium Resources). According to Staatz and Olsen,[12] the occurrence of thorium is widespread but the resources are not well known because present demand

is low. The demand in 1968 was for only about 125 tons of ThO_2. Estimates of the thorium content of the earth's crust range from 6 to 13 ppm. Identified world thorium resources recoverable primarily as a by-product or co-product are about 1.4 million tons, one-third of which occurs in a deposit near Elliot Lake, Canada. The general understanding is that large additional resources would be found with additional exploration. If we assume the thorium resources are no larger than the uranium resources, the fissile fuel supply is extended by a factor of 4 to 5 without reprocessing.

3. The extension of the fission fuel supply using the hybrid produces additional <u>time</u> that can be used to make deliberate decisions on issues such as internationally controlled, physically secure fuel production and fuel reprocessing centers.[10,11]
4. The manufacturing of fresh fertile fuel pellets can proceed without the handling problems inherent in the use of a radiation spiking material such as ^{60}Co. This avoids any legal or safety issues associated with the deliberate addition of dangerous materials.

The major disadvantage of this system is that it does not take full advantage of the fertile fuel reserves. To achieve a fuel supply measured in thousands of years, rather than just a few hundred, fuel reprocessing is essential. Without reprocessing, one hybrid reactor is only able to supply fissile fuel to about 2.5 LWRs of the same thermal power. This has the economic impact of increasing the effective fuel cost. With reprocessing of the spent LWR fuel, on the order of 10 LWRs can be fueled from one hybrid of equivalent power, depending on the conversion ratio of the LWR or other convertor reactor.

The proliferation resistant fuel cycle can be extended to include reprocessing of the spent LWR fuel if one follows the structure outlined by Feiverson and Taylor of internationally controlled, physically secure fuel production and reprocessing sites combined with many national convertor reactors "outside the fence". This process involves the four steps outlined in Fig. 2.

1. Fresh ThO_2 or UO_2 fuel is fabricated in assemblies that are directly usable in a LWR or other convertor reactor. This step will also involve the fabrication of enriched fuel assemblies at the secure site using fissile fuel from the reprocessing step. We propose that such fuel be only partially enriched (for example, to just 2% even though about 3-4% is required) and that the hybrid be used to produce the required additional enrichment.
2. The fuel assemblies are irradiated in the hybrid blanket to produce the required fissile enrichment.
3. The fuel is transferred directly to the fission reactor and burned.
4. The spent fuel assemblies are shipped back to the physically secure site for reprocessing. The reprocessing plant removes fission products and sends the fissile material to the fuel factory for fabrication into new fuel assemblies.

The advantages of this approach are the following:

1. The fuel supply is measured in terms of the fertile material abundance. All estimates show such fuel supplies will last for thousands of years.

2. Fuel shipped to and from the convertor reactors is always highly radioactive and would be resistant to diversion and reprocessing for the reasons described earlier.
3. The convertor reactor need not be restricted to a LWR although using these reactors will minimize the need to develop additional fission reactor technologies.

The potential success of these fuel cycles depends upon two key technical questions: (1) Can the hybrid reactor produce uniformly enriched fuel at an acceptable fusion performance level when the blanket design is constrained to accommodate LWR fuel assemblies? (2) Can a standard LWR burn the irradiated fuel? The first of these questions was the major emphasis of the SOLASE-H laser fusion hybrid study. The second question will be considered in future work.

3. THE SOLASE-H FUSION-FISSION HYBRID REACTOR STUDY

A. Introduction

The SOLASE-H study is a coupled set of investigations covering five separate topics:

(1) Overall proliferation resistant fuel cycles

(2) Blanket neutronics and mechanical design

(3) Laser fusion performance requiements for hybrids

(4) First wall protection using xenon cavity gas

(5) Hydrogen Fluoride laser design.

This study is distinguished from a true conceptual reactor design by the fact that not all systems are treated (e.g. pellet injection, tritium reprocessing) and there is not as much emphasis on a completely self-consistent set of parameters. However, the ranges of parameters studied in each area were chosen to overlap so that self-consistent sets of parameters can be derived from the study. It is convenient to consider a set of consistent parameters such as those displayed in Tables 1-3. Cut-away views of the reactor cavity and blanket are shown in Fig. 3. The reactor cavity and blanket have a cylindrical geometry to accommodate the fuel assemblies around the circumference. The cavity height allows 3 assemblies to be stacked on top of one another. The blanket structure is zircaloy to be compatible with the cladding of the fuel assemblies. The zircaloy first wall is protected from the X-ray and ion debris of the pellet microexplosion by 0.5-1 torr of xenon gas that is circulated through the cavity.

The fusion power in this system is 1200 MW. This is produced by irradiating pellets at the rate of 4 Hz with each explosion yielding 300 MJ of energy. The 14 MeV neutrons are assumed to contain 70% of this energy with the rest partitioned between ions and X-rays. The laser energy on target is 1.5 MJ thus implying a pellet gain of 200.

The blanket power multiplication varies between 2 and 5 during the fuel enrichment process, giving an average thermal power of 2650 MW. The neutron wall loading at the midplane is 2 MW/m^2. The coolant is sodium. It enters the blanket at 300°C and exits at 350°C. The tritium breeding ratio is 1. The upper and lower blankets, comprising 30% of the solid angle subtended at the target, are devoted only to breeding tritium.

Table 1

SOLASE-H PARAMETERS

Cavity Shape	Cylindrical
Cavity Radius	6 m
Cavity Height	12 m
Structure - Blanket	Zircaloy
- First Wall	2 mm Zircaloy
First Wall Protection	0.5 - 1.0 torr Xenon Gas
Fusion Power	1200 MW
Pellet Yield	300 MJ
Neutrons	210 MJ
X-rays and Ions	90 MJ
Pellet Gain	200
Pulse Repetition Freq.	4 s^{-1}
Laser Energy (on target)	1.5 MJ

Table 2

SOLASE-H PARAMETERS

Fusion Power	1200 MW
Average Thermal Power	2650 MWt
Thermal Power Range	2400-2900 MWt
Per Cent Variation	(19%)
Gross Elect. Output	925 MWe
Net Elect. Output	700 MWe
Recirc. Power Fraction	26%
Blanket Power Mult.	1.5 - 5
Neutron Wall Loading (Max)	2 MW/m^2
Coolant	Na
Coolant Temperatures	300-350°C
Tritium Breeding Ratio	1.0
Fertile Material	ThO_2
U^{233} Production Rate	0.65/Fusion Neutron
	2.5 Tonnes/yr
Fuel Form	(17x17) PWR Assemblies
Number of Assemblies	528
Time to 4% Enrichment	2.7 yr
Max/Ave. Enrichment	1.1
Neutron Multiplier	Pb

Table 3

SOLASE-H PARAMETERS

Laser Type	Hydrogen-Fluoride
Laser Energy	2 MJ
Net Efficiency	2.6%
Electrical Eff.	24%
Wavelength	2.7 - 3.5 μm
Maximum Power	300 TW
Pulse Length (Multiplexed)	3 ns
Number of Final Amplifiers	20
Last Mirror Position	22 m
Number of Laser Mirrors	56
Illumination	Non-uniform

The fertile material is ThO_2, clad in 17x17 PWR fuel assemblies. The blanket contains 528 assemblies and produces 0.65 $^{232}Th(n,\gamma)^{233}Th$ reactions per fusion neutron. This produces 2.5 tonnes of ^{233}U per year, enough to fuel about 2.5 1000 MW_e PWRs with no reprocessing. The time to reach 4% fertile enrichment is 2.7 years of exposure or 3.8 years of operation at a 70% plant factor. The maximum to average fuel enrichment in a fuel assembly is 1.1.

The SOLASE-H study includes a detailed conceptual hydrogen fluoride laser design. The laser energy is 2 MJ and the maximum power is 300 TW. The wavelength of the HF laser is actually a range of wavelengths, 2.7-3.5 μm, because the laser operates on many different lines. The net efficiency of the laser is 2.6%. This includes both the electrical efficiency of initiating the chemical reaction and the chemical efficiency of reconstituting the laser gas mixture back into its original constituents. The pulse length is 3 ns and there are 20 final amplifiers, hence there are 56 last mirrors. The last mirrors are located at a distance of 22 m from the cavity center and no attempt has been made to uniformly distribute them around the reactor.

B. Blanket Neutronics and Mechanical Design

The blanket design for SOLASE-H is shown in Fig. 3. The reactor cavity is cylindrical with fissile fuel only being bred in the circumferential blanket. The top and bottom blankets are devoted to breeding tritium. The radius of the cavity is 6 m and the height is 12 m. This allows 3 LWR fuel assemblies to be stacked in the blanket. The blanket structure is zircaloy, to be compatible with the cladding of the fuel assemblies. If stainless steel were used as the structure there is the possibility of carbon transport between it and the zircaloy cladding by the Na coolant. The first wall is 0.2 cm thick and is scalloped as shown in Fig. 3 to accommodate the Na coolant pressure in the blanket. Directly behind the first wall are pins of Pb, clad in zircaloy. This Pb serves as a neutron multiplier, thus enhancing the fissile production rate. When this zone is removed, the total number of breeding captures per fusion neutron is reduced from 1.63 to 1.46. If the neutron for breeding tritium is subtracted, then the reduction of neutrons available to produce fuel goes from 0.63 to 0.46, a 27% effect.

The zone containing LWR assemblies is surrounded in the front and rear with pins containing Li. These Li zones both breed tritium and filter thermal neutrons that might otherwise diffuse into the fuel assemblies and induce fission. By poisoning the thermal flux, they enhance the uniformity of enrichment across the LWR assembly. Behind the LWR fuel zone and its Li filter is a Pb and carbon reflector. The fuel zone is therefore surrounded by fast neutron reflecting material and thermal neutron filters. The assemblies behave as a fast neutron flux trap, thus maximizing the fissile fuel breeding rate. The reflector is followed by an outer Li zone to capture any leaking neutrons.

Numerous neutronics calculations using the ANISN neutron transport code were done to optimize this blanket (#13 on the figure) such that the uniformity across the fuel assemblies is that shown in Fig. 4. The maximum to average enrichment is 1.1 with a ^{233}U enrichment of 4.7% at the edge and 3.75% in the middle. The time required to reach this enrichment is 2.7 years of exposure.

The fuel assembly is rotated 180° at the end of 1.35 years to achieve the symmetric profile. The profile can be made flatter only at the expense of reducing the fissile breeding ratio. The optimized blanket is chosen to be the one having the smallest maximum-to-average Th(n,γ) reaction rate profile, denoted by R, while having a high value of the uranium breeding ratio, UBR. Thus,

$$FM = UBR/R. \qquad (1)$$

However, since the average Th(n,γ) reaction rate is proportional to UBR we find that

$$FM = (UBR)^2/Th(n,\gamma)_{max}. \qquad (2)$$

This quantity was chosen to represent the criteria for blanket optimization. A penalty is paid in the LWR for large values of maximum-to-average enrichment due to hot channel factors. However a penalty is paid in the hybrid for values close to one because these blanket designs have reduced values of UBR. Thus our figure of merit tends to minimize each of these penalities. For the base case blanket design the breeding ratio is 0.65 ^{233}U/fusion neutron. With this breeding ratio and 1200 MW of fusion power the hybrid produces ~2500 kg of ^{233}U per year, enough to fuel ~2.5 1000 MW_e LWRs without reprocessing.

Most of the blanket neutronics analyses were done using ANISN and assuming a one-dimensional, spherical blanket. A solid angle weighting of 70% is then applied to the results to account for the fact that the fissile fuel is only in the circumferential blanket in the cylindrical reactor. Once a near optimum detailed blanket configuration is determined, three-dimensional Monte Carlo calculations are performed on the entire blanket including the upper and lower tritium breeding regions. These calculations are done to determine the enrichment profile in the axial direction and to test the solid angle weighting approximation. This analysis shows that the upper and lower blankets can be strongly neutronically coupled to the circumferential blanket and hence the simple solid angle weighting technique must be cautiously applied. However, the total number of absorptions per fusion neutron is almost constant at 1.65. Therefore, a proper three-dimensional design can be established that will give the same results as the one-dimensional designs with solid angle weighting. Furthermore, alternate fuel assemblies can be replaced with scattering material and a thermal neutron filter so that the remaining assemblies are reduced in number and are surrounded by scattering material and thermal neutron filters. The three-dimensional analysis shows that this does not seriously reduce the total number of absorptions per fusion neutron but significantly reduces the fuel inventory. The fuel in the blanket is enriched more quickly, reducing the associated carrying charges.

The power generated by the hybrid averages 2650 MW_t. This swings between 2400 MW_t and 2900 MW_t due to the changing blanket multiplication during the fuel enrichment process. These values are for the equilibrium cycle where there are 4 different batches of fuel in the blanket. Therefore, the blanket contains fuel that is fresh, 1/4 enriched, 1/2 enriched, and 3/4 enriched. This fuel management scheme limits the power swing in the blanket to 19%. A thermal efficiency of 35% then gives a gross electrical output of 925 MW_e. The laser requires 225 MW_e and thus the net output is 700 MW_e.

C. Laser Fusion Performance

Two economic figures of merit serve as guidelines for the determination of acceptable laser fusion performance. These are (1) the recirculating power fraction and (2) the cost scaling of capital intensive components, such as power supplies to drive the laser. The first of these considerations is related to the return on investment in the thermomechanical equipment needed to produce the electricity. The recirculating power fraction can be related to the target gain, G, total laser efficiency, η_L, and the blanket energy multiplication, M, by the expression

$$f_R = [\eta_{th}\eta_L \; G(0.3 + 0.7M)]^{-1} \qquad (3)$$

where η_{th} is the thermal to electrical conversion efficiency and we assume 70% of the fusion energy is in neutrons. If M=1 and η_{th}=0.4, then a 25% recirculating power fraction implies η_LG=10. This is typically taken to be the performance constraint placed on pure laser fusion reactor systems. For instance, the laser efficiency in the SOLASE design was 6.7% and the target gain was 150. In a hybrid, where M>1, the product, η_LG, can be less than ten while the system still meets the condition of a 25% recirculating power fraction. Furthermore, the blanket multiplication increases the absolute power level, hence the fusion power necessary to produce a given amount of thermal power is reduced. Both of these effects lead to relaxed laser fusion performance requirements.

The second condition, the cost scaling of capital intensive equipment, determines the economy of scale associated with such systems. An analysis of power supply costs indicates that high efficiency lasers, which require modest power supply energy coupled with modest target gain are more likely to operate in an economically satisfactory way than low efficiency lasers and high gain targets. The lower bound on laser efficiency when power supply costs are limited to \$200/kW$_e$ of installed capacity is shown in Fig. 5 for different total electrical power and blanket multiplication factors. The economy of scale is clear because the minimum laser efficiency consistent with this power supply cost is 19% for a 100 MW_e plant, but only 4.4% for a 1000 MW_e plant when M=1. This maximum power supply cost of $200/$kW_e$ is chosen because this would be about 10% of the plant cost if the hybrid were to cost $2000/$kW_e$. We reason that it is unlikely that the power supplies, only one component of the plant, could be allowed to cost more than 10% of the total plant cost.

The SOLASE-H laser fusion parameters are compared to the SOLASE parameters in Table 4. A key relaxation of laser performance is the reduction of repetition rate from 20 to 4 Hz. State-of-the-art power supplies cannot meet the lifetime requirements of 10^8-10^9 shots and although, in principle, they can be derated in voltage to meet these demands, the prospect that such simple solutions will be successful is quite low. Therefore a relaxation of the repetition rate in the hybrid is important. This is also possible because the blanket multiplication allows a lower fusion power. The electrical efficiency of the HF laser in SOLASE-H is 24%, more than a factor of two greater than the electrical efficiency of the SOLASE laser. This allows the laser energy to be increased by a factor of two without increasing the power supply requirements. The larger laser energy is chosen to allow more conservative estimates of the laser beam transport efficiency and the target gain.

For the SOLASE-H study it was necessary to couple these considerations with the need to produce a copious supply of 14 MeV neutrons. The hybrid blanket is designed to produce a maximum amount of uniformly enriched fuel with a minimum fission rate. This results in a low blanket multiplication.

Table 4

Comparison of SOLASE and SOLASE-H Laser Fusion Performance Parameters

Parameter	SOLASE	SOLASE-H
Electric Power (net)	1000 MW	700 MW
Thermal Power	3300 MW	2650 MW
*Fusion Power	3000 MW	1200 MW
*Blanket Power Multiplication	1.1	1.5-5
Recirculating Power Fraction	25%	26%
*(Target gain)x(Laser efficiency)	10	5
Target Yield	150 MJ	300 MJ
% Yield in Neutrons	80%	70%
Target Gain	150	200
Laser Type	CO_2 Model	HF
Laser Wavelength	--	2.7-3.5 μm
Laser Energy (on target)	1.1 MJ (1.0)	2 MJ (1.5)
Max. Laser Power	1000 TW	300 TW
*Repetition Rate	20 Hz	4 Hz
*Net Laser Efficiency	6.7%	2.6%
*Laser Electrical Efficiency	10%	24%

A fusion power of 1200 MW produces enough fuel for about 2.5 LWRs. However, the reactor cavity size is determined by the energy that the first wall can accommodate in a single pellet microexplosion. This favors small explosions at a high repetition rate yet this is inconsistent with the anticipated scaling of target gain with laser energy. Such considerations lead to the range of parameters chosen for SOLASE-H.

D. First Wall Protection by Xenon Cavity Gas

As mentioned in the previous section, the cavity volume, and hence the blanket volume, is determined in laser fusion reactors by the size of a single microexplosion rather than by the average fusion power. It is therefore crucial to determine the maximum target yield that can be accommodated on a repetitive basis by the first wall. In most laser fusion reactor concepts thus far, the first wall has been shielded from the pellet blast by some protection scheme. The method of protection has in fact been the fundamental identifying characteristic of these reactor designs.[9,13-15] The first wall protection method proposed for the SOLASE-H study involves the introduction of a noble gas, such as xenon, in the cavity. The gas pressure is less than 1 torr to minimize the effects of gas breakdown by the laser beams. This gas absorbs the target ionic debris and X-rays from the pellet explosion and re-radiates the energy to the first wall over a time that is long enough to allow the energy to be conducted away. In Fig. 6 we show the heat flux experienced by the first wall as a function of time for 0.5 torr of xenon. In these calculations it is assumed that 90 MJ or 30% of the total 300 MJ yield is deposited in the gas. In Fig. 7 the transient temperature response of the wall is plotted as a function of time. This analysis indicates that 90 MJ is about the maximum amount of energy that can be withstood by a zircaloy first wall and a 6 m cavity. However, calculations also show that the heat flux at the first wall sensitively depends on the radiative properties of the hot (1-10 eV) xenon gas. These properties have not been accurately computed so that a final conclusion awaits further analysis.

E. Hydrogen Fluoride Laser Design

The SOLASE-H study includes the conceptual design of a 2 MJ hydrogen-fluoride laser. This laser is pumped by an electron beam initiated chemical reaction.

$$\begin{aligned} F_2 + H &\rightarrow F + HF \\ H_2 + F &\rightarrow H + HF \end{aligned} \tag{4}$$

The electron beam pulse is 20 nsec and the natural pulse width of the laser is 12 nsec. This is likely to be too long for target irradiation and therefore the 20 final amplifiers are multiplexed. Energy is extracted from 18 of the amplifiers in a series of three short pulses that are combined on the target in a manner that gives the desired pulse shape. From the remaining two amplifiers a single long pulse is extracted. The multiplexed pulse from one final amplifier is shown schematically in Fig. 8. The amplifiers have square optical apertures that are 102x102 cm and are 34 cm in length.

The net electrical and chemical efficiencies of the 3000/900/100 torr mixture of $F_2/O_2/H_2$ laser gas are 24% and 4% respectively. These are chosen to maximize the overall efficiency, including all recirculating power costs, to 2.6%. However, Fig. 9 shows that the electrical efficiency can be increased, but only at the expense of reducing the chemical efficiency and the overall efficiency. If this could be tolerated, then the power supply requirements could be even further reduced.

There is an increased interest in the HF laser for laser fusion applications because recent experiments have proven that beam quality is good and that amplified spontaneous emission in the very high gain amplifiers can be suppressed. This motivated our study of this 2.7-3.5 μm laser. The net efficiency is quite adequate for hybrid reactors and may be used for pure fusion reactors if target gains of 400-500 can be achieved with this relatively long wavelength. The amplifiers are compact, making the laser more easily scalable to 2 MJ. The HF laser may be repetition rate limited because the laser gas must be reprocessed to convert it from HF into H_2 and F_2. The chemical reprocessing facility may be the limiting capital cost item in the total laser cost. Just as with the power supplies, there is likely to be an economy of scale associated with the reprocessing plant, but this is not known at this time.

4. Conclusions of the SOLASE-H Study

The SOLASE-H study established the potential feasibility of hybrid reactors for fueling LWR fission reactors in a nuclear future that does not allow reprocessing, due to proliferation concerns. This involves direct irradiation of fertile fuel assemblies in the hybrid blanket. This simultaneously enriches the assembly of fuel to the proper fissile enrichment and renders it proliferation resistant by making the fuel highly radioactive. Such a hybrid reactor produces 0.6 - 0.7 ^{233}U atoms/fusion event while achieving a tritium breeding ratio of one. At this rate, one direct activation hybrid operated at a fusion power of 1200 MW can fuel ~2.5 LWR's requiring 4% enriched fuel. The hybrid can also be incorporated into a scenario where the spent LWR fuel is sent to an internationally monitored, physically secure fuel production and reprocessing center. This fuel is reprocessed, and then inserted back into the hybrid for re-enrichment. This would allow each hybrid to fuel 10 LWR's with conversion ratio of 0.75 for ^{233}U fuel. This scenario also meets the qualifications of the Feiverson and Taylor non-proliferation fuel cycle.

The economic feasibility of the nonreprocessing fuel cycle will be very sensitive to the cost of the hybrid because the support ratio of fission reactors to hybrid reactors is so low. This sensitivity is somewhat reduced for the case of the hybrid and reprocessing center. The first fuel cycle allows <u>time</u> to make deliberate decisions about reprocessing while still maintaining the LWR industry. Without reprocessing the fissile fuel reserves are extended by about a factor of ten over the ^{235}U resources. With reprocessing the fuel resources are measured in terms of the fertile fuel supply, extending the LWR fuel supply to thousands of years.

The potential success of these fuel cycles depends upon two key technical questions: (1) Can the hybrid reactor produce uniformly enriched fuel at an acceptable fusion performance level when the blanket design is constrained to accommodate LWR fuel assemblies? (2) Can a standard LWR burn the irradiated fuel? The first of these questions was studied in detail in the SOLASE-H study. The answer to the second question is currently being pursued.

Using careful blanket design, LWR fuel pins can be nearly uniformly enriched to 4% fissile concentration in approximately 3 years. The spectrum of neutrons incident on the assemblies must be carefully tailored to provide uniform enrichment. A hard spectrum is desired and this favors Pb, rather than Be, as a nonfissionable neutron multiplying material in the blanket. The Be has a large (n,2n) cross section but it also moderates the neutrons and this is not desired. There is also a serious question of resource availability for Be. The fuel is surrounded by thin zones of Li to filter the thermal neutrons that might otherwise diffuse into the fuel. This serves the dual purpose of breeding tritium and suppressing the fission rate in the fuel assemblies.

Nearly flat enrichment profiles across the assembly can be achieved, but only at the expense of the ^{233}U breeding ratio. Therefore, a figure of merit is developed that both takes account of minimizing the hot spot factor resulting from nonuniform enrichment and maximizing the ^{233}U breeding ratio. This shows that the optimum is not necessarily the blanket design that produces the flattest ^{233}U distribution. The fuel can be rotated at the half-way point in enrichment to provide a symmetric profile. Axial uniformity can be provided by a fuel management scheme in which the fuel spends 1/3 of its time in each of the three vertical locations relative to the point source. Three-dimensional neutronics calculations show that some of the fuel can be replaced by neutron scattering material and that the remaining fuel still has the same production rate, 0.6 ^{233}U/fusion neutron. This reduction of fuel inventory shortens the time to 4% enrichment from 3 years to 1.5 years of exposure. These 3-D calculations also show that the circumferential blanket and the upper and lower blankets can be strongly coupled, neutronically. This suggests that blanket design using simple 1-D calculations with solid angle weighting must be carefully evaluated for validity.

Burnup calculations show that approximately 13% of the total fuel generated is consumed before it is removed from the blanket. This burnup is equivalent to 4300 MWd/MT. The power swing, due to changes in the blanket multiplication during enrichment, is 19%. The minimum power is 2400 MW_t and the maximum is 2900 MW_t.

The damage rate to the zircaloy clad during exposure is about 7 dpa over the total 3 year period. This low value is the result of the neutron moderation caused by the Pb neutron multiplier zone and the Li filter zone in front of the fuel assemblies.

The direct enrichment fuel factory hybrid requires a substantial fusion power because the rate of fissile fuel production is proportional to the number of 14.1 MeV neutrons that are generated. Fissile or nonfissile neutron multipliers do not affect the rate of net fuel production by more than about ~25%. About 2.1 kg of ^{233}U is produced per megawatt-year of fusion energy. Hence, 1200 MW of fusion power is required to produce the fuel for 2.5 1000 MW_e LWR's. This large fusion power requirement does not allow the repetition rate and pellet gain to be simultaneously relaxed. In SOLASE-H the pellet gain remains rather high (~200) while the repetition rate is relaxed to 4 Hz. This low repetition rate allows more time to re-establish the cavity initial conditions before the next target micro-explosion. It also relaxes the gas handling capacity in the HF chemical laser. The laser efficiency can be substantially reduced in the hybrid while still maintaining an acceptible recirculating power fraction. The HF chemical laser has net efficiency of 2.6% and the pellet gain is 200. This leads to a recirculating power fraction of 26%. Such a reduction in required laser efficiency will admit more lasers to the "possible laser fusion driver" category than will pure laser fusion requirements.

The cavity and blanket volume are determined by the transient conditions following a single microexplosion rather than the time integrated fusion power. This leads to the desire for small explosions at a high repetition rate. But this is not likely to be possible because lasers and cavities may be repetition rate limited and a copious supply of fusion neutrons is needed. However, the repetition rate must be high enough to avoid thermal relaxation in the fuel assemblies between microexplosions. Repeated thermal transients in the fuel lead to thermal ratcheting which will destroy the fuel integrity. The minimum allowable repetion rate is about 1 Hz.

Gas protection of the reactor first wall from pellet debris and X-rays appears to be applicable to hybrid reactors. The major first wall response to the hot gas at densities of 0.75 - 3 x 10^{16} cm^{-3} comes from a thermal transient due to the gas reradiation to the first wall. The over pressure at the first wall due to blast wave effects is minimal.

The hydrogen-fluoride chemical laser appears to be scalable to at least 2 MJ. Its long pulse nature necessitates multiplexing of beams through the final power amplifiers if pulses shorter than 15 nsec are required. The electrical efficiency of this laser can be as high as 100% but optimum net laser efficiency is associated with an electrical efficiency of 24%. This can greatly relax the power supply requirements over those needed for low electrical efficiency lasers. The net efficiency of the HF laser including the chemical efficiency of reconstituting the H_2 and F_2 is 2.6%. As mentioned earlier, the HF gas handling and reprocessing limits the laser repetion rate.

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Figure Captions

Fig. 1 - Fusion-fission hybrid fuel cycle without reprocessing

Fig. 2 - Fusion-fission fuel cycle with reprocessing

Fig. 3 - Cutaway views of the SOLASE-H reactor cavity and blanket

Fig. 4 - Uniformity of enrichment across the PWR fuel assembly after 2.7 years of exposure.

Fig. 5 - Laser parameters for power supply costs that are limited to $200/kW$_e$ installed capacity.

Fig. 6 - Heat flux at the first wall as a function of time for 96 MJ of energy deposited into 0.5 torr of xenon.

Fig. 7 - Transient temperature response of 2 mm zircaloy first wall exposed to the heat fluxes in Fig. 6.

Fig. 8 - Multiplexed pulse from a single HF final amplifier.

Fig. 9 - Electrical and chemical efficiency of HF laser as a function of H_2 partial pressure.

THE HYBRID SYSTEM AS A FUEL FACTORY WITHOUT PROCESSING

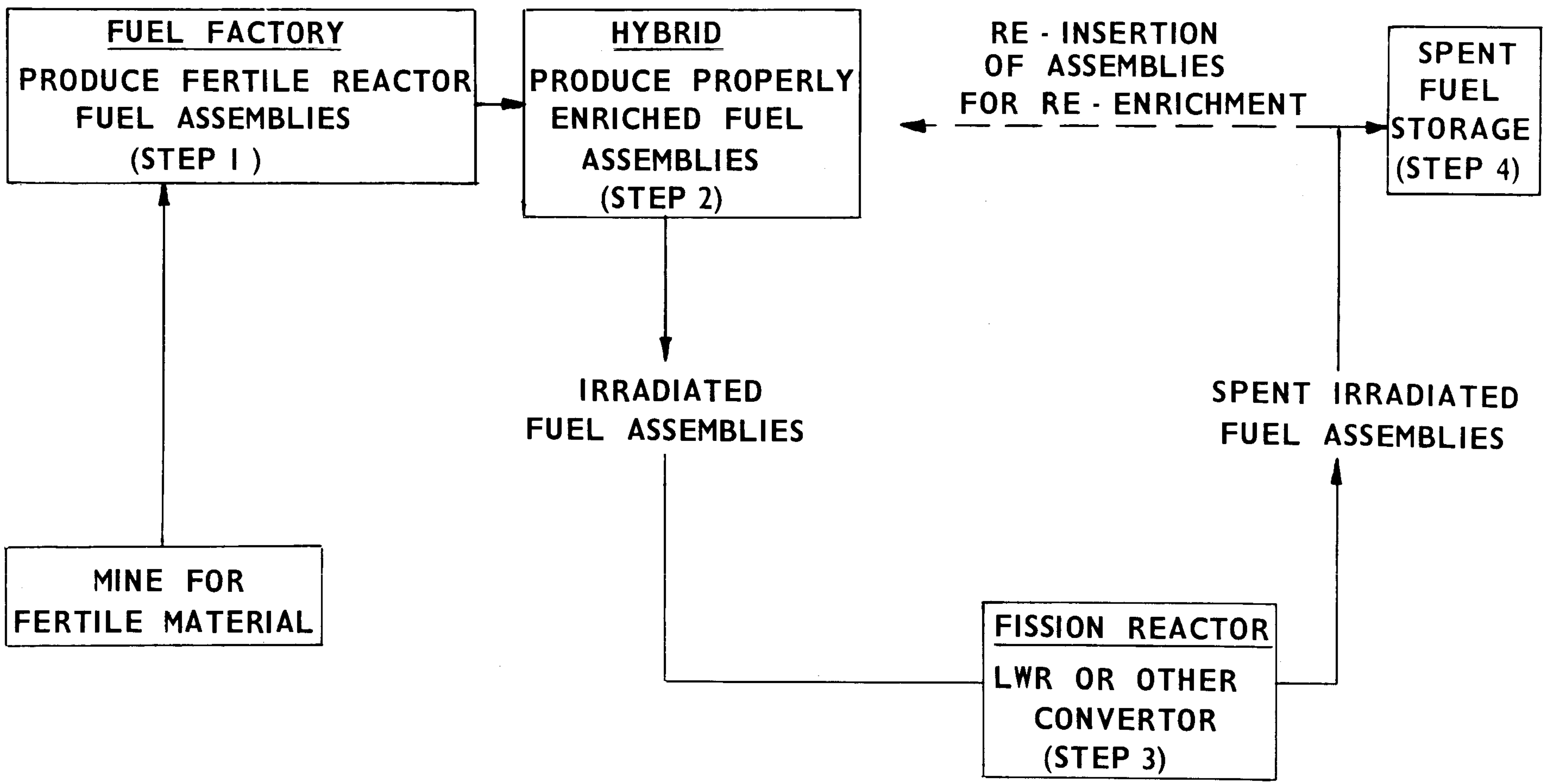


FIGURE 1

THE HYBRID SYSTEM AS A FUEL FACTORY WITH REPROCESSING

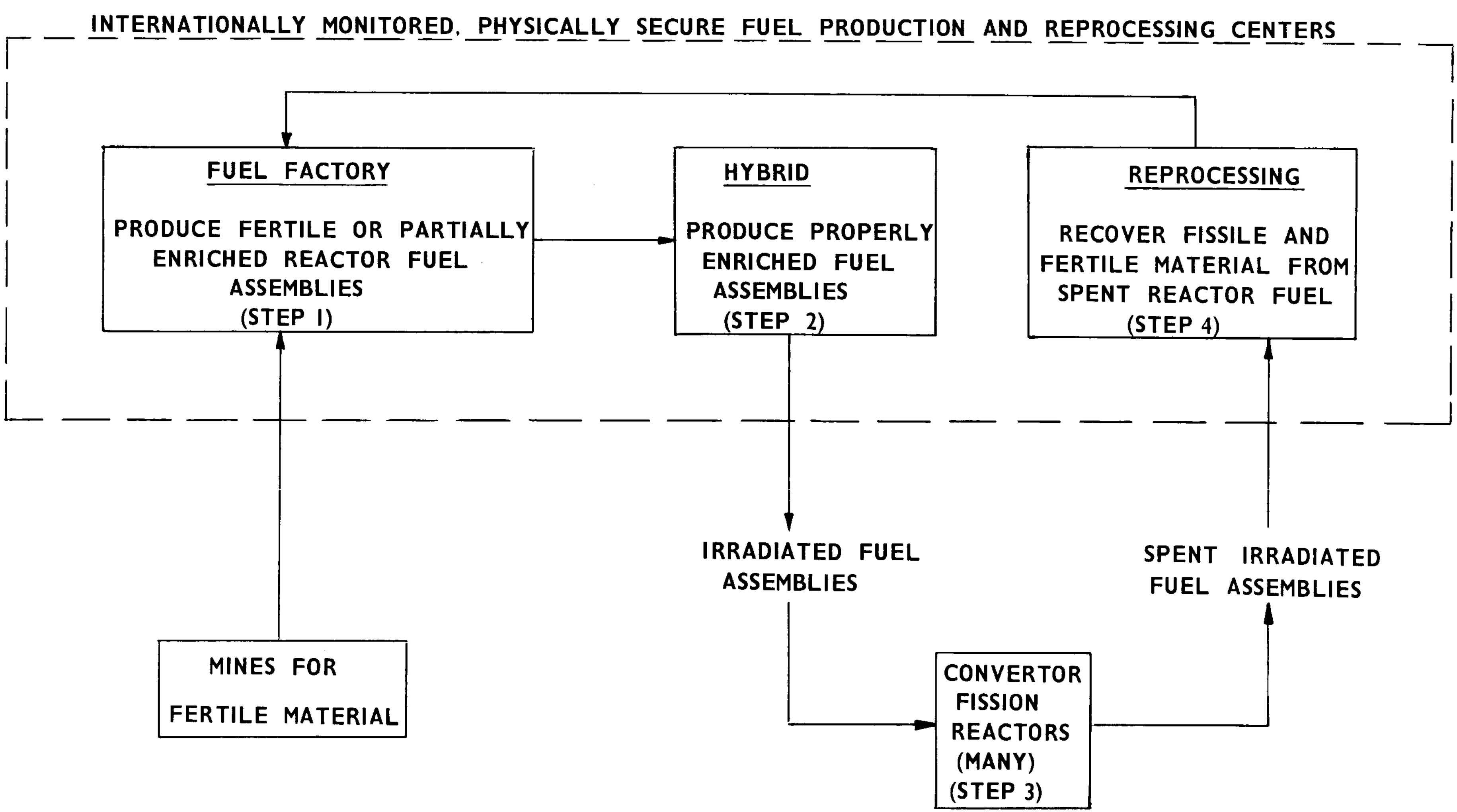


FIGURE 2

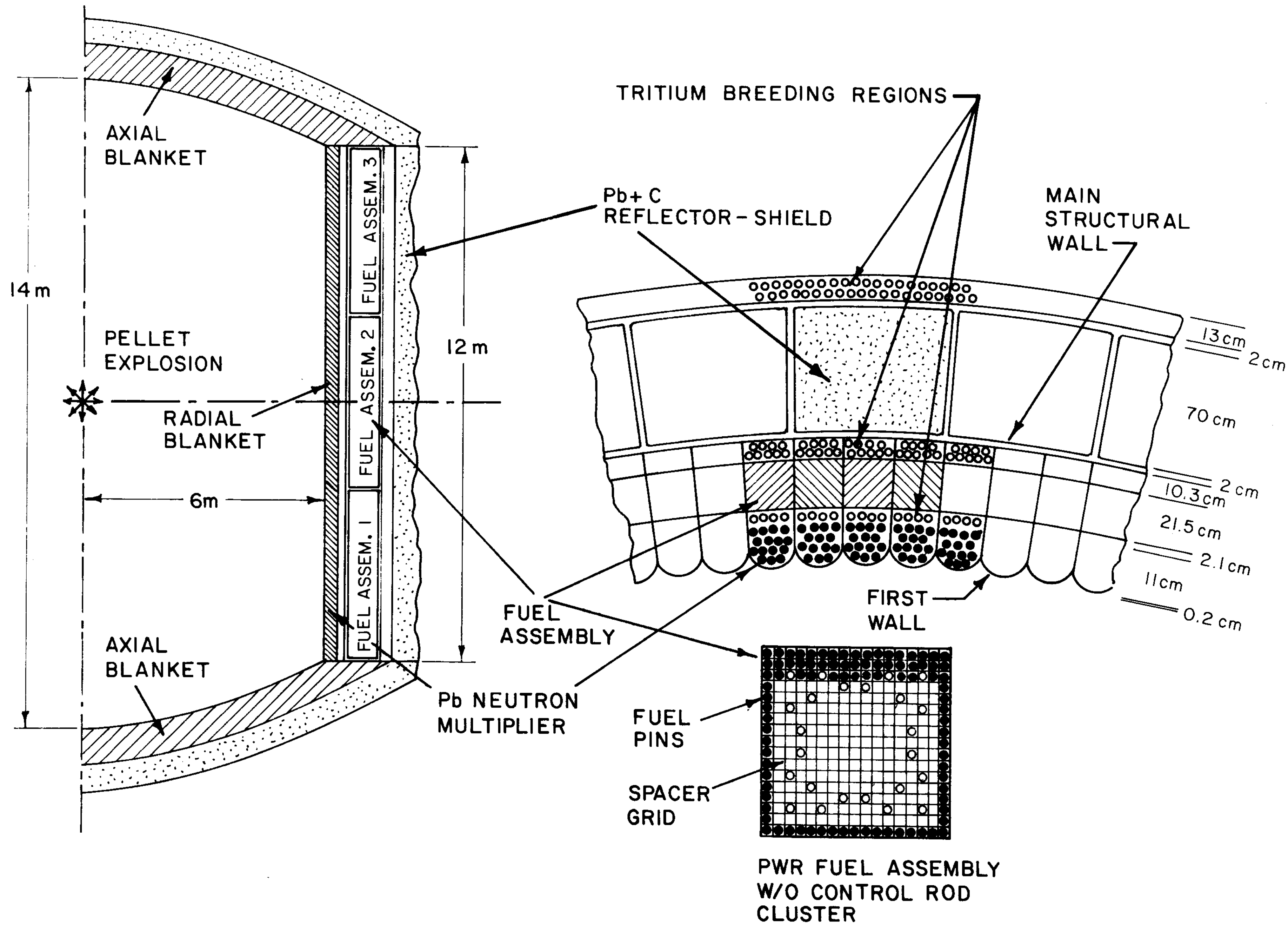
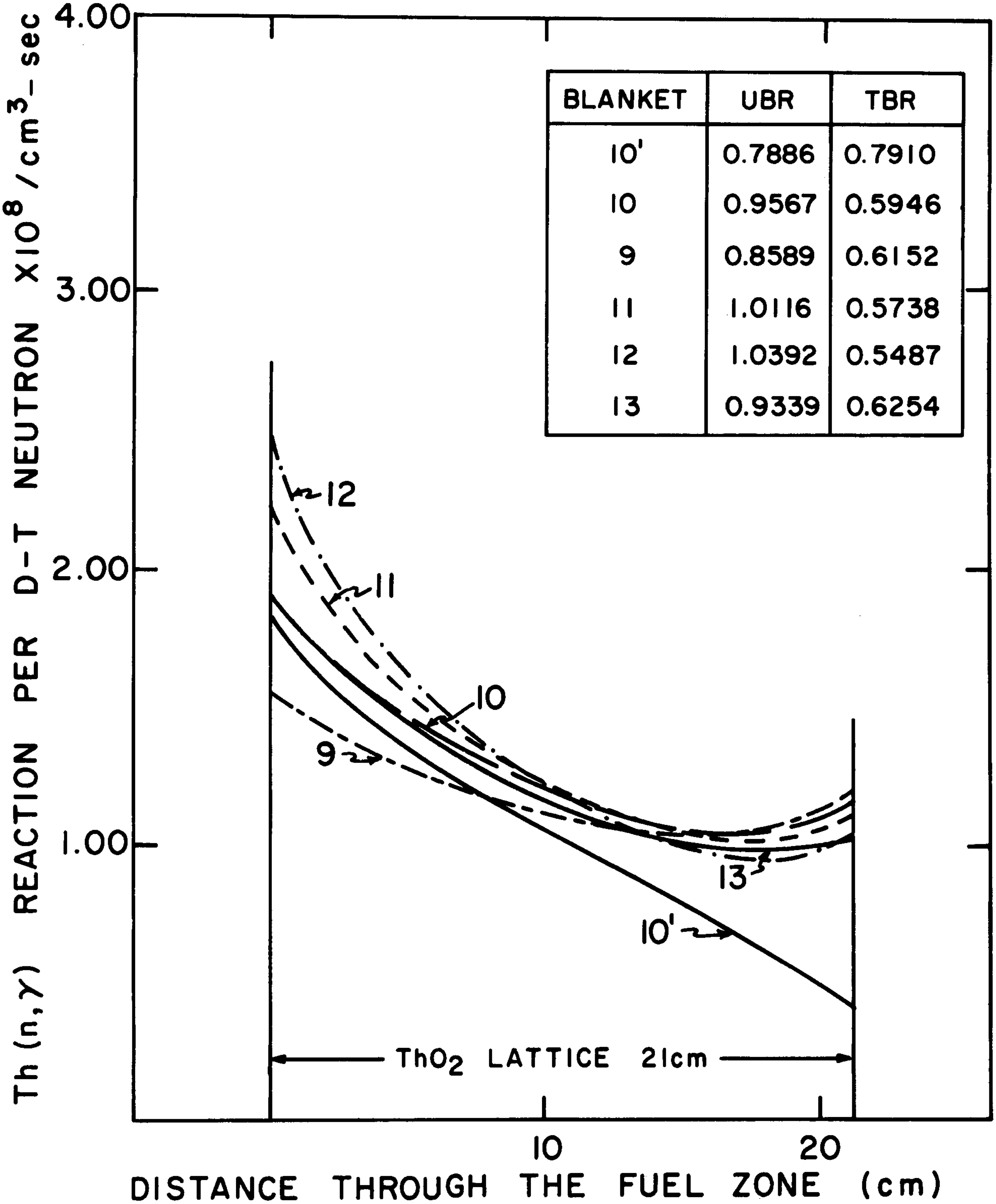

AXIAL
BLANKET
14 m
PELLET
EXPLOSION
RADIAL
BLANKET
6m
FUEL ASSEM. 3
FUEL ASSEM. 2
FUEL ASSEM. 1
12 m
AXIAL
BLANKET
Pb NEUTRON
MULTIPLIER
FUEL
ASSEMBLY
Pb+ C
REFLECTOR - SHIELD
TRITIUM BREEDING REGIONS
MAIN
STRUCTURAL
WALL
13 cm
2 cm
70 cm
2 cm
10.3 cm
21.5 cm
2.1 cm
11 cm
0.2 cm
FIRST
WALL
FUEL
PINS
SPACER
GRID
PWR FUEL ASSEMBLY
W/O CONTROL ROD
CLUSTER

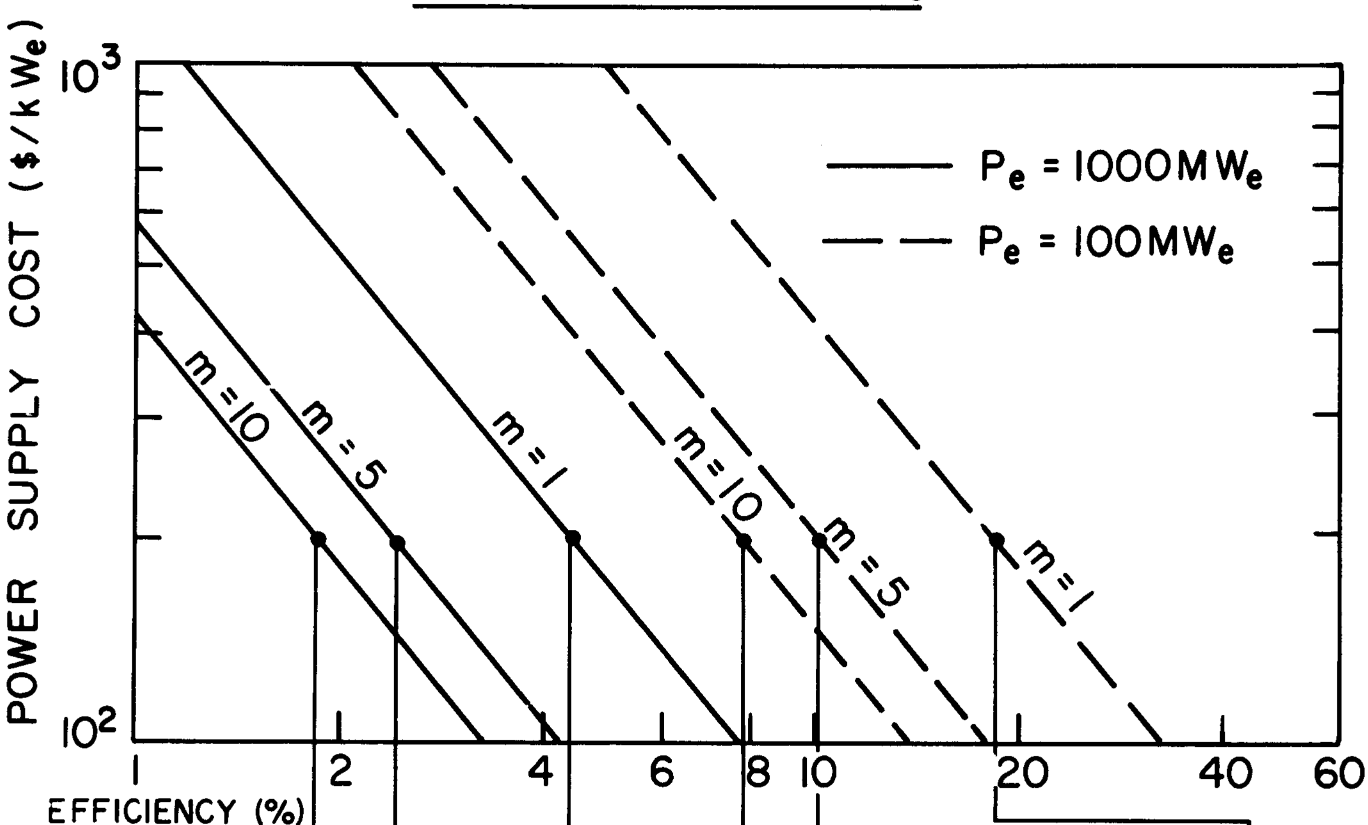
FIGURE 3



U-233 BREEDING RATE DISTRIBUTION IN FUEL ZONE PER D-T NEUTRON WITH Pb AS THE FRONT ZONE NEUTRON MULTIPLIER

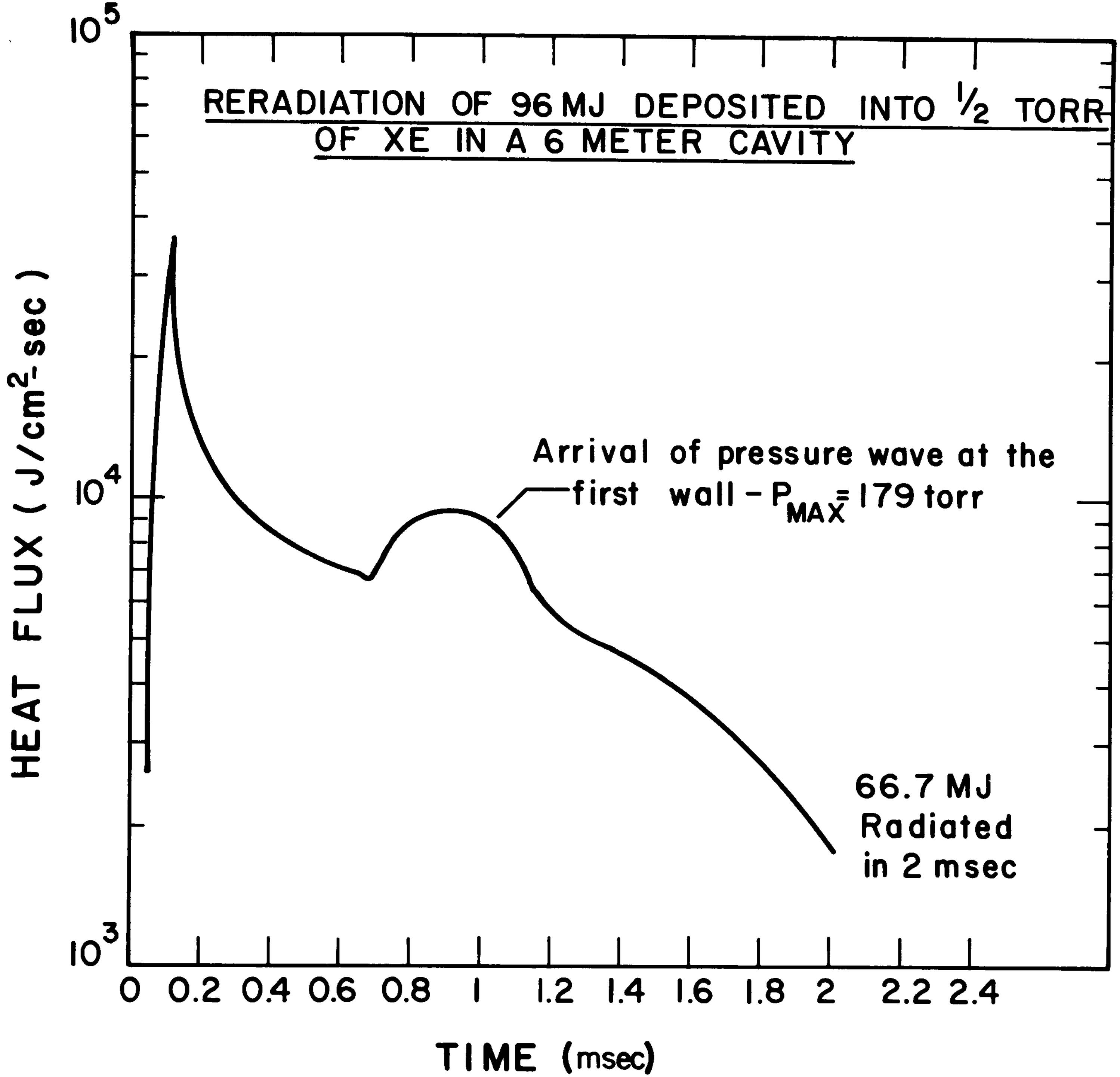
FIGURE 4

LASER PARAMETERS WHEN LIMITED TO POWER SUPPLY COST OF \$200/kW$_e$



P_e (MW_e)	1000	1000	1000	100	100	100
m	10	5	1	10	5	1
η_L	.019	.0245	.044	.081	.105	.19
G	52.6	81.6	227	12.3	19	52.6
E_L (MJ)	.685	.887	1.6	.292	.377	.685
y (MJ)	36	72	363	3.6	7.2	36
ω (s^{-1})	9.2	9.2	9.2	9.2	9.2	9.2

FIGURE 5



RERADIATION OF 96 MJ DEPOSITED INTO 1/2 TORR OF XE IN A 6 METER CAVITY
HEAT FLUX (J/cm²-sec)
10^5
10^4
10^3
Arrival of pressure wave at the first wall - P_{MAX} = 179 torr
66.7 MJ Radiated in 2 msec
0 0.2 0.4 0.6 0.8 1 1.2 1.4 1.6 1.8 2 2.2 2.4
TIME (msec)

FIGURE 6

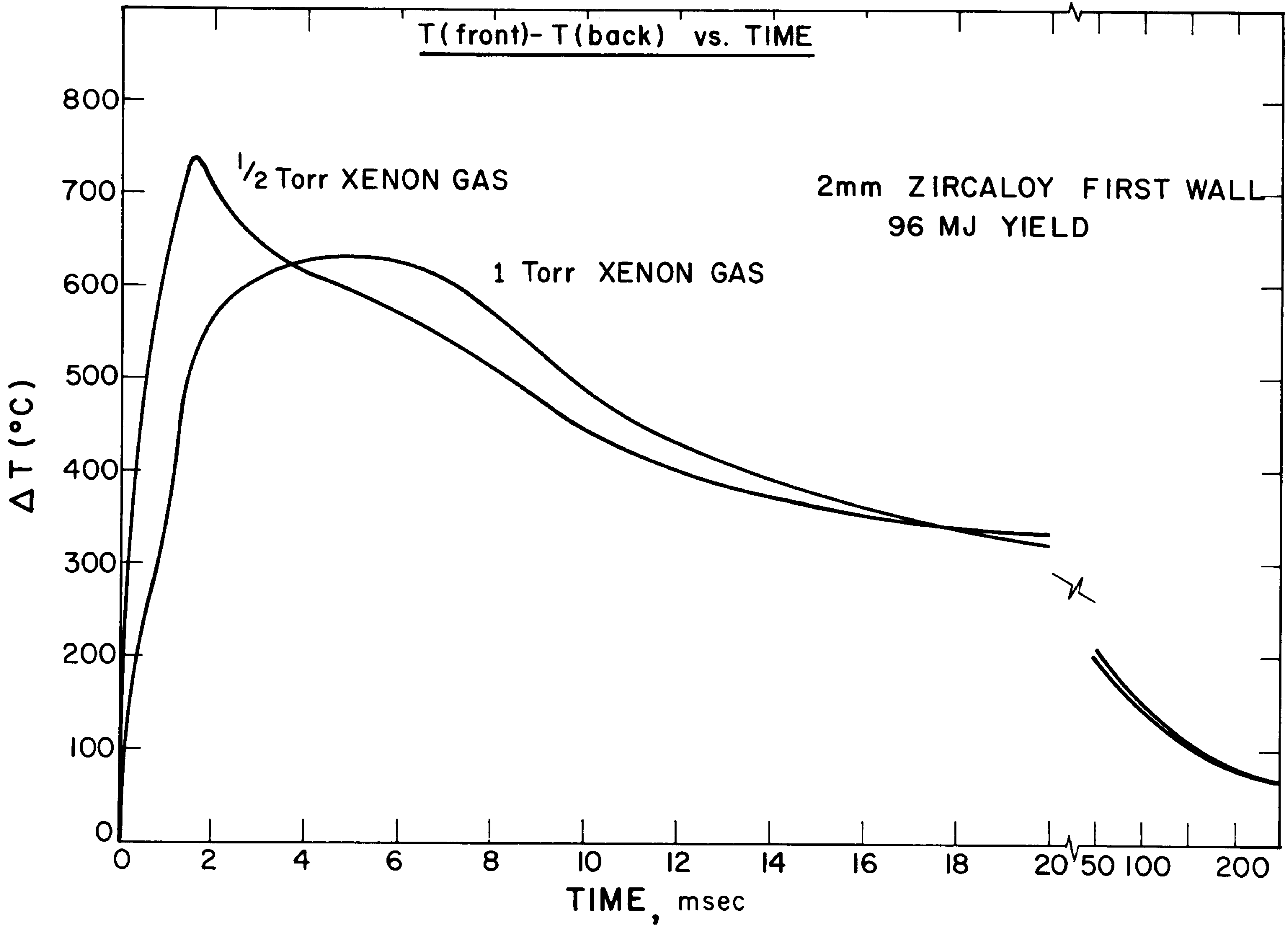
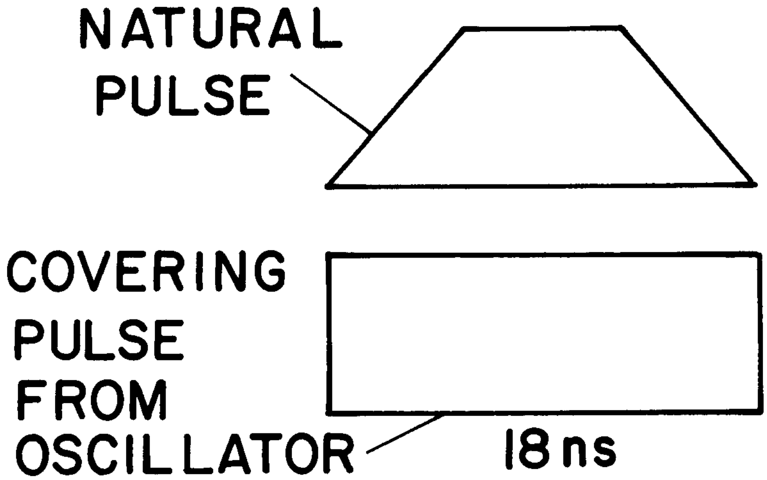
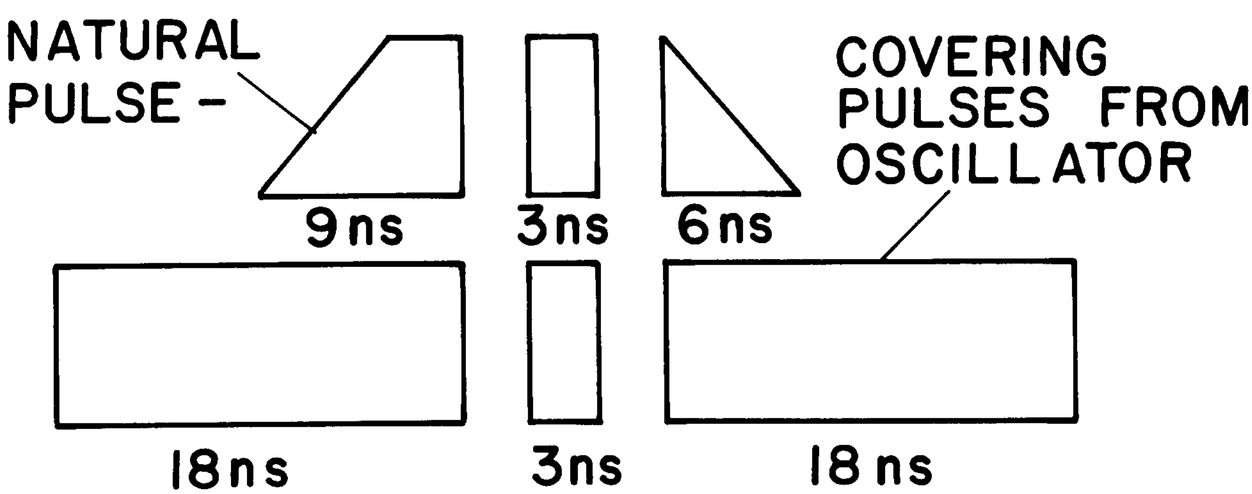

T(front)- T(back) vs. TIME
1/2 Torr XENON GAS
1 Torr XENON GAS
2mm ZIRCALOY FIRST WALL
96 MJ YIELD
ΔT (°C)
800
700
600
500
400
300
200
100
0
0
2
4
6
8
10
12
14
16
18
20
50
100
200
TIME, msec

FIGURE 7



(a) SINGLE PULSE EXTRACTION



(b) MULTIPASSED EXTRACTION

FIGURE 8

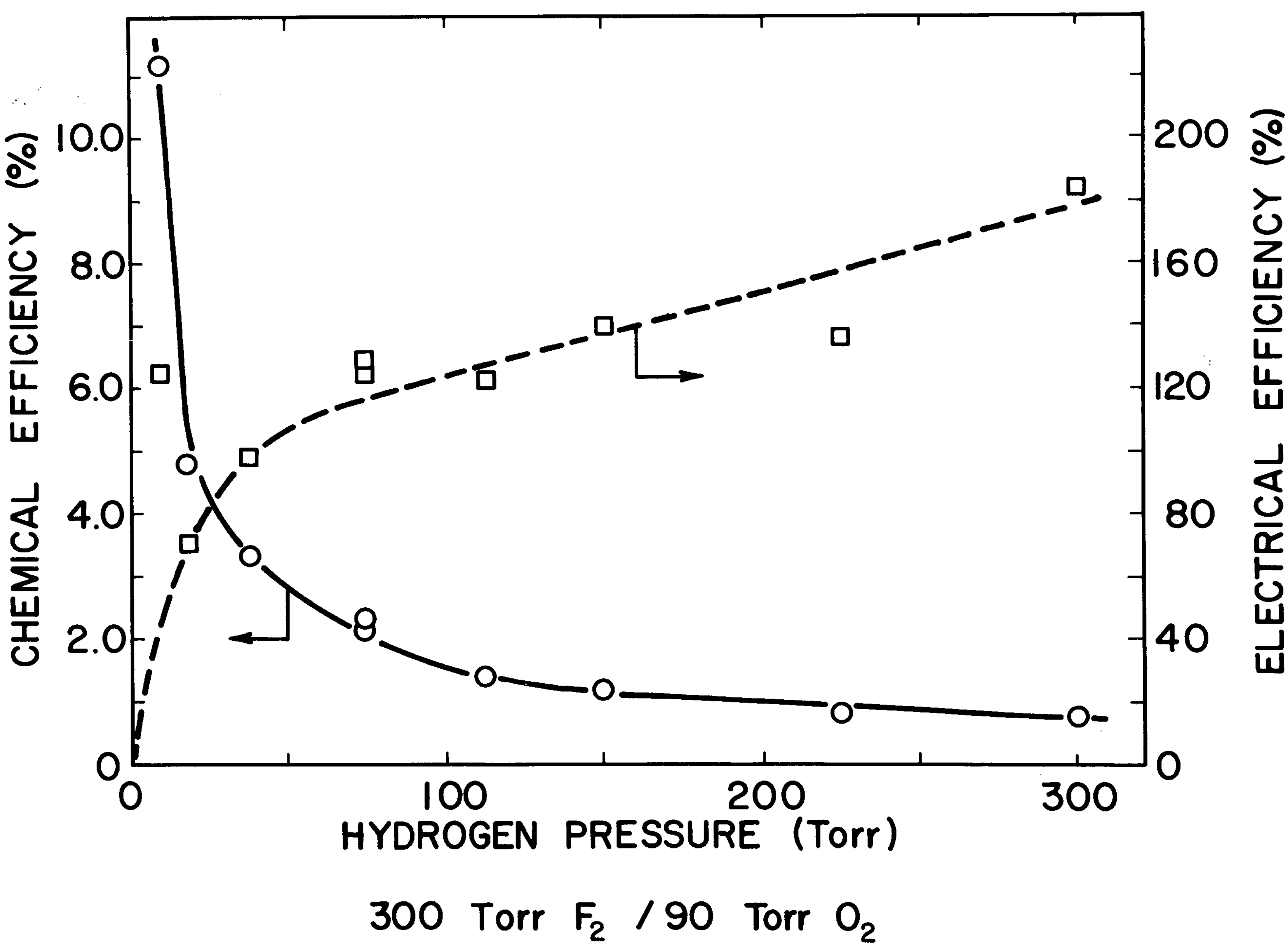

CHEMICAL EFFICIENCY (%)
ELECTRICAL EFFICIENCY (%)
HYDROGEN PRESSURE (Torr)
0
2.0
4.0
6.0
8.0
10.0
0
40
80
120
160
200
0
100
200
300
300 Torr F_2 / 90 Torr O_2

FIGURE 9