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**PROCEEDINGS OF THE SYMPOSIUM ON THERMONUCLEAR  
FUSION REACTOR DESIGN, JUNE 2-5, 1970**

August 1, 1970

Plasma Laboratory  
Texas Tech University  
Lubbock, Texas

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PROCEEDINGS OF THE SYMPOSIUM ON  
THERMONUCLEAR FUSION REACTOR DESIGN, JUNE 2-5, 1970

Edited by  
M. Kristiansen  
M. O. Hagler

August 1, 1970

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Texas Tech University  
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## PREFACE

The Symposium on Thermonuclear Fusion Reactor Design was co-sponsored by the USAEC and Texas Tech University on June 2-5, 1970. The program consisted primarily of invited papers and was aimed at researchers with backgrounds in physics, electrical engineering, nuclear engineering, etc., but not necessarily at those with prior research experience in fusion reactor design. Some of the participants were already involved in various aspects of this research area and a small number of contributed papers were therefore also included to take full benefit of their experience.

Since the symposium was to a large extent tutorial in nature some of the papers were based on results which had been or were going to be presented at other conferences or in professional journals. Only the abstracts of these papers are included here in order not to preclude other publication. Several of the papers were prepared especially for this symposium as evidenced by the length of the proceedings.

The participants came from 24 states and 2 foreign countries. The distribution of participants between the various science and engineering disciplines was very broad and the participants contributed to the success of the symposium by their active comments and questions.

It has been found worthwhile to include in these proceedings a selected list of publications and reports pertaining to fusion reactor design studies. This list was compiled with the kind cooperation of W.C. Gough, USAEC. In the interest of saving time and enabling an early printing of the proceedings, only very minor editing and retyping of the various manuscripts have been done. Certain errors

and a non-uniform format will be evident but none of these were found important enough to justify delaying the printing of the proceedings.

The program covered a broad range of the various aspects and concepts of fusion reactor design. A status report on fast breeder reactors was specifically included in order to gain the proper perspective for evaluating the fusion studies. Some of the contributed papers were devoted to possible space application of fusion and this was also the topic of a special evening discussion session. The symposium appeared to succeed in its goal of presenting a broad review of the most important and recent fusion reactor studies in the U. S. to an audience with widely varied background and interests.

SYMPOSIUM ON FUSION REACTOR DESIGN  
Texas Tech University, June 2-5, 1970.

	Tuesday 6/2/70	Wednesday 6/3/70	Thursday 6/4/70	Friday 6/5/70
8:30-10:00	Welcome (University Pres.) Introduction and Organization (M. Kristiansen)	Current & Energy Consideration for Fueling Open-ended Fusion Reactors (H.K. Forsen, UW)	Blanket Neutronics and Radiological Aspects of Fusion Reactors (J.D. Lee, LRL)	Status Report on Fast Breeder Reactors (Why Breeders?) (G. Melese-d'Hospital, GGA)
10:00-10:30	Coffee	Coffee	Coffee	Coffee
10:30-12:00	Nuclear Fusion Reactors General Considerations (F.L. Ribe, LASL)	Fundamental Considerations for Toroidal Reactors (R.G. Mills, PPPL).	A Thermal Energy Conversion System for Fusion Reactors (G.A. Carlson, LRL)	Why Fusion? (W.C. Gough, AEC)
12:00-1:30	Lunch	Lunch	Lunch	Lunch
1:30-3:00	High- $\beta$ Pulsed Fusion Experiments and Reactors (F.L. Ribe, LASL)	Ignition and Control of Toroidal Reactors (R.G. Mills, PPPL)	Direct Recovery as it Relates to Mirror Machines. The Recovery of Tritium in a Fusion Reactor Using Heat Pipes (R.W. Werner, LRL)	<u>15 Minute Papers</u> D.G. Samaras (AFOSR) J.R. Roth (NASA/Lewis) T.C. Powell, (U. of Kentucky)
3:00-3:30	Coffee	Coffee	Coffee	Coffee
3:30-5:00	High- $\beta$ Pulsed Fusion Experiments and Reactors (F.L. Ribe, LASL)	Turbulent Heating of Mirror Machines (I. Alexeff, ORNL)	Fuel Burnup and Direct Conversion of Energy (T.A. Oliphant, LASL)	Summary and Conclusion (M. Kristiansen and M.O. Hagler, TTU)
7:00-9:00	Round Table Discussion of Space Applications of Fusion Reactors		<u>15 Minute Papers</u> A.S. Roberts (Old Dom. Univ.) F. Winterberg (Univ. of Nevada) C. R. Finfgeld (Roanoke) C. S. Shapiro (San Fran. State College) Z. Sabri (Univ. of Wisconsin) G. H. Miley (Cornell University)	

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# TECHNOLOGICAL ASPECTS OF STEADY-STATE AND PULSED REACTORS\*

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## Abstract

A comparison is made between the steady-state (S.S.) reactor, as described by Rose and by Carruthers, Davenport and Mitchell, and the pulsed, theta-pinch-like reactor described by Ribe, Oliphant, Quinn and Borkenhagen. In the case of the S.S. reactor an engineering demand on the thermonuclear energy flux through the vacuum wall ( $1300 \text{ W/cm}^2$ ) sets the size and plasma-physical properties of a reactor. The plasma radius  $R_p$  is 1.25 meters, and the inner diameter of the superconducting coil, which furnishes the steady magnetic field surrounding the vacuum chamber and lithium blanket is 6 meters. Both charged-particle-heated and injection-heated reactors are considered, with plasma  $\beta$  (ratio of plasma pressure to magnetic confining pressure) in the 5-10% region and necessary plasma containment times between 0.6 and 3 seconds. Such a reactor is estimated to produce its useful electrical output at a cost of about 2.5 mils/kwh with less than 10 percent circulating power.

In the pulsed reactor the plasma is admitted and burned during magnetic pulses of tens of milliseconds duration, separated by off periods 20 to 30 times as long. The plasma is much smaller ( $R_p \approx 10 \text{ cm}$ ) and denser, and the magnetic energy per MW(th) output is a factor of  $\sim 50$  less. Here a demand on the  $\beta = 1$  plasma D-T fusion-energy production determines the thermonuclear flux through the vacuum wall and coil, which are now inside the

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\* This work performed under the auspices of the U. S. Atomic Energy Commission

tritium-breeding lithium blanket. That portion (20%) of the flux which actually heats the coil can be removed either by fast He-gas flow or liquid-Li flow at a heat transfer coefficient  $h \approx 3 \times 10^3$  Btu/ft<sup>2</sup>hr°F. Either technique corresponds to present fast-reactor practice. The Cu coil and its Mo backing (for supporting the magnetic stress) allow an ample tritium breeding ratio,  $T/n = 1.28$ .

The circulating power fraction of the pulsed reactor is at least ~25%, set primarily by joule-heating of the internal coil. The cost/kw(E) output of the S.S. and pulsed reactors must be the same (~ \$200/kw(E)) on economic grounds, and the costs of their "conventional plants" for thermal-to-electrical energy conversion will also be about the same per kw(E). The extra conventional plant necessary to furnish the larger circulating power in the pulsed case must be made up by a smaller cost of the "core", consisting of magnet, blanket and vacuum chamber. Since the magnetic energy is ~50 times smaller in the pulsed case, the necessary factor-of-two saving of core cost can probably be made, even when an external pulsed cryogenic magnetic energy store is used. This may allow the cost per kw(E) of the pulsed reactor to be equal to or less than that of the steady-state reactor.

INJECTOR REQUIREMENTS FOR OPEN ENDED  
FUSION REACTORS

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May 1970

## I - INTRODUCTION

Over the past twelve years considerable effort has gone into the study of plasmas and plasma containment towards the goal of power from controlled fusion. Throughout most of these studies the containment volumes have been of such size that plasma sources of conventional design have been available to do a reasonable but usually inadequate job. If we are to continue making progress on the plasma containment problem then it will soon be clear if not already, that we need to expend much more effort on plasma sources in order to fill our experimental containment volumes with interesting plasma. This means densities near  $10^{14} \text{ cm}^{-3}$  and ion temperatures of kilovolts.

In this paper we will address ourselves to the problems in open ended low  $\beta$  reactors where the plasma can not be heated or created in situ but which must be generated outside the system. This does not mean that it will be impossible to generate or heat plasmas in such systems but we hope to outline the problem from the point of view of an external source.

In trying to determine injector requirements for open ended fusion reactors there are many factors to be considered and in this paper we will only deal with some of them. For instance, we will show that the Lawson criterion<sup>1</sup> will have a direct effect on the injected current per unit volume that is required to sustain a reactor or build up density in an experiment. If we are given an  $n\tau$  confinement criterion then one finds that the current required for fixed energy is only a function of the power output of the reactor or the square of the plasma density. Assuming the injector is to also provide the heat energy of the

system, then the current must be provided at some high energy. Should it eventually be possible to thermalize the charged reaction products such that they provide an internal source of heating, then the injector energy requirements can be reduced somewhat but this will have no effect on the current requirements. Thus, for the studies considered here we will neglect charged particle heating and therefore put an increased burden on the injector but it should provide an upper limit as to what must be done.

With the injector providing all the heating we then turn to problems of making a neutral atomic beam at the desired energy and injecting it into the reactor. Such problems as beam intensity, beam brightness, the conversion of ion beams to neutral beams and reactor wall area necessary for use by the injector are considered.

## II - CURRENT REQUIREMENTS

In order to determine the current required to fuel a reactor one needs only realize that in order to keep the plasma density about constant you need to replace the plasma particles which undergo fusion or are lost from the system. That is if we write for ions in a steady state D-T reactor with  $n_D = n_T = n/2$

$$\frac{dn_i}{dt} = S_i - \frac{n^2}{2} \langle \sigma v \rangle - \frac{n}{\tau_c} = 0 \quad (1)$$

where  $S_i$  is the source of ions per unit volume and  $\tau_c$  is the containment time due to geometrical losses. If we look at the

two loss terms we recognize that their ratio is just the fractional burn up ( $f_b \approx \frac{1}{2} n \tau_c \langle \sigma v \rangle$ ) for the case where  $\tau_c / \tau_f \ll 1$  and  $\tau_f$  is the fusion reaction time.

In the limit of low fractional burn up we can approximate that  $S_i = n / \tau_c$  or if we want a current per unit volume we have

$$\frac{I}{V} = \frac{ne}{\tau_c} . \quad (2)$$

Multiplying the numerator and denominator by  $n$  we see that the current (per unit volume) is  $n^2 e / n \tau_c$  and the Lawson<sup>1</sup> or  $n\tau$  criterion becomes all important. If we further write the fusion power for the system per unit volume we have

$$P_F = n_1 n_2 \langle \sigma v \rangle Q_{12} \quad (3)$$

where  $n_1$  and  $n_2$  are the particle densities for the two reacting nuclei which are usually taken such that  $n_1 = n_2 = n/2$  and  $Q_{12}$  is the energy released per fusion. We solve Eq. (3) for a D-T reactor and solve for the density to obtain

$$n^2 = \frac{4P_F}{\langle \sigma v \rangle Q} . \quad (4)$$

Substitute this into Eq. (2) and solve to find

$$I = \frac{4P_f V e}{\langle \sigma v \rangle Q n \tau} = \frac{2P_F V}{f_b Q_{ev}} \quad (5)$$

where  $Q_{ev}$  is in eV. This does not leave much to the imagination because if we are to run a 1500 MW(th) plant at some fractional

burn up, we find the current is determined except by uncertainties in the parameters. Table I lists values of the current from Eq. (5) for various reactions. However, in the table we use only the energy released per fusion and assume no contributions from the blanket or secondary reactions, i.e.  $Q_{DD} = 3.65$  MeV,  $Q_{DT} = 17.6$  MeV and  $Q_{D3} = 18.3$  MeV. If blanket reactions and secondary reactions are to be considered, the current requirements will not change appreciably but the fusion power will increase by as much as  $68.4/3.65$  or  $18.7$  for the D-D cycle according to calculations by Post.<sup>2</sup> One should be careful however because we are assuming a fractional burn up and this will change as a function of  $n\tau$  over which we may have little control.

There is a second almost independent way of arriving at the required current density. This involves dealing with Eq. (2) directly and trying to establish an economic size requirement. For instance, if we plot the volume required for a 1500 MW(th) plant as a function of density for various reactions using Eq. (3), we obtain Fig. 1. The two lines represent extremes for  $\langle\sigma v\rangle$  determined by the energies given.

Using the criterion established by Carruthers et al.<sup>3,4</sup> and based on limitations of first wall cooling and cost estimates, they suggest that an open ended cylindrical system must have dimensions of about 1.25 m plasma radius and 10 m long. This gives a plasma volume of about  $50 \text{ m}^3$  and from Fig. 1 we find that a density between 2 and  $30 \cdot 10^{14} \text{ cm}^{-3}$  is required



depending on the cycle and energy. Because open ended systems are limited in their confinement time to something like the ion-ion collision time and this varies as  $T^{3/2}/n$ , we choose the higher temperatures and thus the lower density.

To complete the requirements for solving Eq. (2) we need to again use the Lawson<sup>1</sup> criterion for the reaction under consideration. Several authors have recalculated this condition for different assumptions and we use that given by Carruthers et al.<sup>3</sup> for an injection heated system but do our own calculation for the D-H<sub>e</sub><sup>3</sup> cycle. Since calculations of this type generally assume  $T_e = T_i$  we run into difficulties at high energy because of the large bremsstrahlung contribution. Most injection heated systems will run with  $T_e \ll T_i$  unless electron heating is externally provided.<sup>5</sup> However, we approximate the  $nt$  criterion as the minimum value consistent with injection heating and high overall thermal conversion and injector efficiencies. These assumptions tend to put an optimistic value on the containment problem but result in a pessimistic value of the current required. Thus, for this analysis we establish Table II for the current density and total current required for the various reactions.

If we compare the results of Tables I and II we observe that the numbers compare quite well for the two cases even though the minimum value of  $nt$  was used whether it corresponded to the energy at which  $\langle\sigma v\rangle$  was taken or not. Also, the  $nt$  calculations for D-D normally take into account the energy gain

by neutron absorption in sodium and this was omitted here. With this understanding of the uncertainty in the numbers, it is still clear that these are formidable currents and we will see that they require a substantial scaling in size and/or numbers from existing injectors.

### III - ENERGY CONSIDERATIONS

For the three reactions considered we have roughly determined the current requirements and to do so required an estimate of the temperature. Now however we need to establish a requirement on the injection energy, independent of current or density considerations, and we do this to maximize reactor parameters or possibly containment parameters.

Whatever criteria we use to determine the injection energy, there are going to be many secondary things which must be considered. For instance if we choose the plasma temperature to maximize  $\sigma$ ,  $\langle\sigma v\rangle$  or  $\langle\sigma v\rangle/T^2$  we need to realize that electron cooling of plasma ions is important and we will have to inject at somewhat higher energies than the desired ion temperature. This is over and above the fact that  $kT_i = \frac{2}{3} U_0$  where  $U_0$  is the injection energy. To give an example of the numbers, Rose<sup>6</sup> calculates for a D-T mirror system where charged reaction products are not completely thermalized, that at 5% burn up and 120 keV injection energies,  $T_i \approx 90$  keV and  $T_e \approx 37$  keV. This is why the Lawson<sup>1</sup> criterion used in the last section is not completely valid because  $T_e \neq T_i$  as we pointed out.

If we simply use the cross section for the three reactions as a criterion for optimum injection, we see from Fig. 2 that they vary over a wide range. That is, for the two branches of the D-D reaction the peak occurs at about 2 MeV, for the D-T reaction it comes at about 120 keV and for the D-H<sub>e</sub><sup>3</sup> at around 400 keV.<sup>7</sup> Later we will see that for any of these energies, the stress imposed on the source at the required currents appears extremely difficult.

In order to integrate the cross section over the velocity distribution to obtain  $\langle\sigma v\rangle$ , we need to know the details of the velocity distribution. For mirror systems this will depend on the mirror ratio and therefore our results should contain these details. Post<sup>2</sup> and others have made these calculations and to show how they compare to a Maxwellian we use the calculations of Kuo-Petravic et al.<sup>8</sup> They find the peak in  $\langle\sigma v\rangle$  for D-T to be  $1.1 \cdot 10^{-15}$  cm<sup>3</sup>/sec at 60 keV in the cm system or 90 keV in the lab. The same cross section integrated over a Maxwellian is given by Glasstone and Lovberg<sup>9</sup> and Rose<sup>6</sup> who obtain  $9.6 \cdot 10^{-16}$  at 80 keV. The point of this is that the reaction rate parameter  $\langle\sigma v\rangle$  is greater for a loss cone distribution than a Maxwellian at the same temperature and increases as the anisotropy of the distribution produced by a larger mirror ratio.

For the present study we are not interested in the magnitude of  $\langle\sigma v\rangle$  but are trying to determine the temperature at which it is a maximum. Therefore, we will assume a

Maxwellian and use the values given by Glasstone and Lovberg<sup>9</sup> for the D-D and D-T reactions. That is, for a D-D reaction the peak in  $\langle\sigma v\rangle$  occurs above 1 MeV and for a D-T reaction it peaks at about 60 keV. For the D-He<sup>3</sup> reaction we use the value given by Post<sup>10</sup> which occurs with a broad peak at 200 keV.

Mills<sup>11</sup> and Rose<sup>6</sup> point out the advantages of using  $\langle\sigma v\rangle/T^2$  as a criterion for the operating temperature because the pressure the magnetic field can support is limited. Therefore for a fixed system or a system in which  $\beta$  and  $B$  are limited, we write the reaction rate as  $R = n^2\langle\sigma v\rangle/4$  and since  $n = \beta B^2/4\mu kT$  for  $T_e = T_i$ , we obtain

$$R = \frac{\beta^2 B^4}{64\mu^2} \frac{\langle\sigma v\rangle}{(kT)^2} . \quad (6)$$

Figure 3 presents a plot of Eq. (6) from Rose<sup>6</sup> and the dashed portion of the curves represent the region where for  $T_e = T_i$  the bremsstrahlung losses exceed the power produced by fusion.

In Table III we summarize the temperature one would hope to maintain in a reactor as a function of the three different criteria. From this and Fig. 3 we see that we should establish a higher operating temperature than the  $\langle\sigma v\rangle/T^2$  criterion would give because this is below the minimum ignition point for all but the D-T reaction.

Criteria such as that suggested by Table III does not take into account the classical confinement limit in open ended systems. That limit is the time for particles to diffuse into the loss cone and leave the system due to collisions. Since

it depends on the temperature as  $T^{3/2}$ , if one operates a mirror system at higher temperatures the confinement time increases. Kuo-Petravic et al.<sup>8</sup> survey calculations of this type and find the confinement time for a D-T plasma goes as

$$n\tau = 4.6 \cdot 10^{11} U_0^{3/2} \log R / \ln D \quad \text{cm}^{-3} \text{ sec} \quad (7)$$

where  $R$  is the mirror ratio,  $\ln D$  is the Coulomb logarithm and  $U_0$  is in keV. If we were to set this equal to the Lawson<sup>1</sup> criterion and solve for the injection energy  $U_0$ , we would find  $U_0 = 570$  keV for  $R = 3$ . This may be a bit unrealistic but it gives some idea of where the crossover occurs. One should be careful however because the Lawson<sup>1</sup> criterion is a function of energy and assumes steady state conditions during the time  $\tau$ . Equation (7) obtains the  $n\tau_c$  criterion by taking an instantaneous value of  $n^2 / \frac{dn}{dt}$  in a decaying plasma where loss is only by Coulomb encounters.

Most advocates of mirror reactors take the criterion of Eq. (7), couple it with other calculations and choose to operate somewhere between the  $\langle \sigma v \rangle$  and  $\langle \sigma v \rangle / T^2$  criterion. Similarly, if it is possible to convert the energy of particles lost through the mirrors to electricity at high efficiency as suggested by Post<sup>10</sup>, then one would like to increase  $U_0$  to minimize the percentage energy loss as the particles are collected. It might therefore be well to tabulate the energy suggested by various investigators rather than use any of the criteria of

Table III or Eq. (7) and this is done in Table IV. Here we see for most reactions that injection energies above 100 keV are desirable and this presents several problems which we will point out in the next section.

#### IV - POSSIBLE INJECTION SOURCES

Thus far we have determined the approximate current requirements and at least a range of energies for injection. Let us now look at the problems that come about in trying to satisfy these requirements and where we need to spend more of our time in research and development. For much of what is said in this and in the following section we describe effects with hydrogen gas but in real reactors the gas would be tritium, deuterium, or helium and the remarks apply equally well on a quantitative basis for these gases.

Since this analysis is confined to open ended systems, one has two limits on the injection direction, i.e. injection can be perpendicular to the field lines, parallel to them and presumably at any angle in between. Let us look at the two limits and see what kind of systems are possible. In order to inject across a magnetic confining field and have the plasma stop somewhere within a desired region, one can go about it in two ways. One way is to inject beams of charged or neutral atoms and have them dissociate or ionize within the region by either collisions with other ionized species or by Lorentz forces.<sup>13</sup> The other way is to shoot bursts or possibly a

steady state stream of plasma towards the region, have the stream polarize and cross the magnetic induction and then depolarize and stop before it gets out. A problem with this method appears to be that if the region already contains a plasma that has established magnetic field lines as equipotentials, it may not be possible to add plasma without losing an equal amount along field lines. Attempts to overcome problems of this kind have led investigators to shoot these plasma streams through gas cells in an attempt to change ions of the plasma stream into neutral atoms such that they could then be trapped in the containment region by ionizing collisions as with beams. Because of the large divergence angle of plasma from sources of this type, they have not been attractive for injectors and yet they may prove important because of their acceleration mechanism. We will take a further look at these sources later.

Let us now return to steady state neutral injection where atomic beams are created from ion beams using some form of source such as that shown in Fig. 4.<sup>14</sup> In these sources  $H_2$  gas, for instance, flows through an arc where it is ionized into fractions of  $H^+$ ,  $H_2^+$ ,  $H_3^+$  and  $H^-$  with the percentages dictated by the cross sections but depending strongly on the arc current and gas pressure. These ions in the form of a plasma pass through the anode of the arc and are then accelerated by a strong electric field. In order to separate the desired charge state from all of those extracted and accelerated, a magnetic lens is frequently used. The lens also serves to focus the beam

onto the aperture of the gas cell for charge exchange. Because initially the extracted beam is only an ion beam it becomes necessary to add charges of the opposite polarity to prevent space charge blow up. This is normally accomplished by simply ionizing the background gas with the energetic ions and allowing the beam to self neutralize. For low energy beams however, it is not possible to provide sufficient ionization without operating the background gas pressure undesirably high. If it becomes necessary to accelerate only one species from the source, say  $H_2^+$ , then it is possible to extract the ions at say 5 keV, magnetically select the desired ion and then post accelerate this specie to the necessary higher energy. In this way the energy spent on the undesired ions is minimized. These and other considerations make the generation and handling of intense beams extremely difficult and thus far highly collimated, energetic beams have been limited to less than about 1 ampere. However, a beneficial feature of such beam systems is the fact that one can make beams of almost arbitrary energy just by adjusting the extraction voltage.

Once we have a space charged neutralized ion beam of the desired energy we now must change it back into the atomic state if we are to inject neutral atoms. This is done by running the beam through a charge exchange cell where electrons from gas atoms are captured onto the fast ions. Because a fast ion or atom can ionize a gas atom, it follows that a gas atom, ion or electron can ionize the energetic atoms of the beam. Figure 5



gives the electron capture and ionization cross section for  $H^+$  in  $H_2$  as a function of the fast particle energy. Of course such cross sections depend on the gas used used for the cell but above a certain energy the cell gas which has the best conversion efficiency is frequently the same gas as the beam ions.

Looking at Fig. 5 one sees that for beam energies above 50 keV the ionization cross section  $\sigma_{01}$  is much greater than the electron capture cross section  $\sigma_{10}$ . This means that even for thin cells the charged beam is likely to emerge from the gas cell still charged and this is highly undesirable. Since the dissociation cross section does not decrease with energy as rapidly as the charge exchange cross section,  $H_3^+$  has the best neutral conversion efficiency at very high energy. D'yachkov<sup>15</sup> has studied this problem for a lithium vapor jet crossing beams of  $H^-$ ,  $H_2^+$  and  $H_3^+$  and finds the probability of producing a single  $H^0$  is given as

$$\begin{aligned}\eta(H^-) &= 0.65 \\ \eta(H_2^+) &= 255 U_0^{-0.4} \\ \eta(H_3^+) &= 500 U_0^{-0.37}\end{aligned}\tag{8}$$

where  $U_0$  is the initial ion energy in keV. To clarify the use of these expressions, if one desired to inject with 300 keV atoms you could start with 300 keV  $H^-$  and would end up with 650 ma of  $H^0$  for each amp injected into the gas cell. If you started with 600 keV  $H_2^+$  or 300 keV per particle, you would get 200 ma of  $H^0$  per amp of original ion and with 900 keV  $H_3^+$  you get 400 ma per amp.

A second problem that must be considered is the efficiency of the source in producing the various ion species. Morgan<sup>16</sup> and Stewart<sup>17</sup> have found that the source shown in Fig. 4 is capable of producing 65-80%  $H_2^+$  ions with a gas efficiency of 60-70%. Similarly Morgan et al.<sup>14</sup> have used slightly different versions of the duoplasmatron to produce even higher percentages of  $H^+$  with even higher gas efficiency. Unfortunately however, no one has yet produced a source of  $H_3^+$  with either a large fraction ion specie or high gas efficiency and this probably is because of the manner in which one gets  $H_3^+$  and the magnitude of the cross sections involved. It would therefore seem desirable from a point of view of gas efficiency to run  $H^+$  or  $H_2^+$  beams at sufficiently low energy that large fractions of  $H^0$  can be obtained from gas cells. We must either do this or perfect a mechanism whereby ions can be recycled through a cell a number of times until they become neutralized.

Assuming that we have neutral atoms let us now return to the problem of trapping them in the containment region by injecting across lines of magnetic induction. Rather than present any calculations let us just remark that if we are trying to refuel an operating reactor, then injecting anything into a high energy density plasma is likely to have a very short mean free path for ionization! Rose and others have worried considerably about this problem and conclude that since it is desirable to spread the new fuel across the diameter of the reactor

rather than just its outer skin, one will have to go to larger collections of molecules and form them into pea size pellets for injection. In the meantime, for present plasma physics experiments we do not have high energy density targets and the simple trapping of beams is considerably less than 100%. Dandl et al.<sup>18</sup> have shown that a modest energy density target plasma can be orders-of-magnitude more efficient in trapping than the  $10^{-4}$  obtained from Lorentz forces. This may be very important in trying to raise the temperature of plasmas in toroidal geometries to the ignition point. For these closed line experiments it is more important to provide high energy than high currents because the target plasma may already be sufficiently dense.

The Lorentz force or Lorentz trapping<sup>13</sup> that has been mentioned several times is caused by the  $\vec{v} \times \vec{B}$  electric field in the particle frame which can be intense enough to strip weakly bound electrons from their ions. Since most gas cells, which are used to convert the ion beam to an atomic beam, provide equilibrium distributions of excited atoms and these go as  $n^{-3}$  where  $n$  is the principal quantum number, the highly excited atoms are ionized and trapped and then the plasma builds up in this manner. Unfortunately the Lorentz trapping efficiency is extremely small. At high energies however, an excited neutral beam may be undesirable for reactor fueling because of Lorentz ionization in the beam filling ports which must pass through the magnetic field in the blanket region.

Unless these large fields are somehow diverted at the ports, plasma may build up in an undesirable area.

Moving now to injection along magnetic field lines as is done in the 2X experiment,<sup>19</sup> we find that we must consider whether we are dealing with a steady state or a pulsed reactor. In a steady state device there may be little opportunity to change the magnetic field in order to adiabatically trap particles and for that reason we must count on collisions. That is, if we inject a steady state beam of particles which are either in the atomic (neutral) or ionized state, or a burst of plasma from a pulsed source along magnetic field lines, the particles will leave as fast as they enter unless a momentum transfer collision is made. If such a collision is made, because of the presence of a dense already trapped plasma, then particles which were originally in the velocity space loss cone may be scattered out of such a region and could be trapped. This is an interesting possibility for refueling, providing of course such collisional trapping does not produce (equivalent) losses of their own, especially because of the plasma access at the ends of the device.

If the magnetic field of the system is to be pulsed, then pulsed plasma sources could be used and as the plasma moves into the evacuated containment region to provide a new fuel charge, the mirror fields can be programmed to adiabatically trap charged particles. This kind of trapping can be very efficient because the plasma can be guided along field lines

into the desired region and no inefficient conversion of ions to neutrals is required. A problem with this system is that all of the plasma fuel must be injected in a time short compared to the bounce time of the plasma from the far magnetic mirror. If we are talking about kilovolt plasmas and containment lengths of 10 m, then we have something like 40  $\mu$ sec to inject all of the particles suggested by Tables I and II. For a D-T reactor for instance this would be equivalent to  $N = nV = 10^{22}$  particles or  $4 \cdot 10^7$  amperes. While this current seems high it is only a factor of  $10^2$  from several existing gun sources; however, the requirements on a neutral beam source are at least a factor of  $10^3$  away from what is presently available.

From these arguments one might conclude that pulsed gun sources are a better approach than steady state beam sources and as far as the ability to produce large currents goes, this is correct. The problem with gun sources is that their energy and gas efficiency are not as high as beam sources and economics is going to be an important consideration in the injector for any future reactors. Never the less, if we are working with a pulsed reactor it may be possible to provide substantial heating of a dense cold gun plasma by magnetic compression and thereby get to the necessary ignition temperatures.

#### V - SOURCE CONSIDERATION

Thus far we have attempted to outline the source requirements and in the last section we have alluded to solutions with

pulsed guns and neutral beams--describing some problems as we see them for both systems. By considering a few additional details of the source problem we should be able to see directions in which to proceed. As fundamental considerations indicate limitations, we should also look for other ways to go. For a start, let us look at pulsed sources.

In pulsed sources gas is injected via a fast acting valve into the inter-electrode space of some kind of axisymmetric coaxial system. Since the coaxial or Marshall<sup>20</sup> gun is the simplest and most widely used, let us use it as our example. This system, shown in Fig. 6, normally has the gas valve located in the center electrode and is pulsed to dump a plenum containing  $10^{17}$ - $10^{20}$  atoms into the interelectrode region. After the plenum is emptied, 50 to 300  $\mu$ sec depending on the type, a voltage is applied across the gun electrodes by connecting a bank of capacitors charged to the desired voltage. Avalanche breakdown occurs and the radial current interacts with the azimuthal magnetic induction behind the current sheet to accelerate the plasma by  $\vec{J} \times \vec{B}$  forces. This simple picture leaves out many sophisticated effects but it clearly indicates that the force is a body force which does not act on each particle individually and that the amount of the force is determined by such external parameters as the capacitance and inductance of the system. The driving piston acts throughout the axial distance of the gun but frequently there is more than one current path and the coupling is less than anticipated. This is especially true when running with large particle densities or high filling pressures as would be required for fueling injectors.

Previously we have mentioned the lower gas and energy efficiencies of these sources. This comes about because of the inefficient coupling of the capacitor energy to the particles as well as the surface and atomic effects which take place in the source. The gas efficiency probably could be increased with an increase in size of the systems but solving the energy efficiency problem is not as evident. Larson et al.<sup>21</sup> have reported energy balance measurements of a rather typical system and they find only 45% of the capacitor bank energy goes into the particles as kinetic energy. However, if systems of converting high explosive energy to magnetic energy can really be developed as by Marshall and coworkers,<sup>22</sup> one may be willing to live with a lower energy efficiency for an inexpensive source of sufficient intensity. One cannot help but feel that obtaining scaling laws for these sources is just as important as obtaining those for containment vessels because unless in situ plasma generation turns out to be possible, such sources will be a limitation in any thermonuclear future. If pulsed sources could be developed and some means found to focus them through a gas cell, they could also play an important role in transverse neutral injection for fueling steady state devices. However, because of the intense beam flux, different gas cells for neutralization would also have to be found.

Moving on to steady state neutral beam sources we encounter a different kind of problem. Here the energy efficiency is capable of being high as is the gas efficiency; however, if one goes to the energies suggested in Table IV then neither of

these efficiencies may be very high. The problem was partially outlined earlier in this section where it was observed that gas cells are not efficient at high energy and thus we might be tempted to go to high energy  $H_3^+$  acceleration because of the larger dissociation cross section. Unfortunately efficient sources of these ions are not available and as a result the source gas and energy efficiency is also low.

Most ion sources used today have an energy scaling that in some fashion is governed by the Child-Langmuir law for space charge acceleration. This law, for plane parallel electrodes, governs the space charge limited current density which may be extracted at a potential  $U_0$  when the separation of the electrodes is  $z$  and is given by

$$J = \frac{4\epsilon_0}{9} \left(\frac{2e}{M}\right)^{1/2} \frac{U_0^{3/2}}{z^2} \text{ amp/m}^2 . \quad (9)$$

Since it is not possible to arbitrarily reduce  $z$  because of the need to be able to hold off the accelerating potential, we are basically limited in the extractable current. While many present day ion sources can approach the limit of Eq. (9), it does not appear possible to exceed this value within present acceleration methods.

If we use a modest energy like 50 kV where from Fig. 5 it would appear that the gas cell problem may not be too bad, and assume we can stably hold this voltage over long periods at a spacing of 0.4 cm, then Eq. (9) gives us  $7.8 \cdot 10^4$  amps/m<sup>2</sup> for



protons. Using the example of Carruthers et al.<sup>4</sup> again, we have a plasma to vacuum wall radius ratio of 0.7 or a wall radius of 1.75 m. This gives for the axisymmetric cylindrical reactor a surface area of 110 m<sup>2</sup>. For the current requirement of Table I for a D-T reactor at 5% burn up we then only need  $2.2 \cdot 10^{-2}$  m<sup>2</sup> or considerably less than 0.1% of the reactor surface area.

This sounds satisfactory until you realize that the source cannot be at the reactor vacuum surface but must be located some distance beyond the blanket or as much as 6 m from where it is needed. In order to transport the beam this distance and retain a reasonable fraction of the current, the beam brightness becomes an important consideration. Kelley<sup>23</sup> defines the effective brightness of a beam as

$$B = \frac{900 IM}{UA_1 A_2/L^2} \quad (10)$$

where M is the mass in amu, U is the energy in MeV, I the beam current in ma, A<sub>1</sub> the initial aperture size in cm<sup>2</sup> and A<sub>2</sub> the final aperture size in cm<sup>2</sup> when A<sub>1</sub> and A<sub>2</sub> are separated by L cm. Thus if our original area requirements were small we would need bright beams to keep the final aperture or area small. Morgan<sup>24</sup> reviews Kelley's paper and summarizes the effective brightness of beams by several investigators in Table V. It is interesting to note that the brightest beams are the lower energy ones even though one can write Eq. (10) in terms of Eq. (9) to show that  $B \propto U^{3/2}/U = U^{1/2}$ .

In order to generate bright beams one needs to work with reasonably small aspect ratios for the extraction aperture. The aspect ratio is defined for circular beams as the beam diameter ( $d$ ) divided by the extraction electrode separation ( $z$ ). For aspect ratios larger than unity the beam quality is down and for our study we choose  $d/z = 1$ . This means that the diameter of the beam of our previous example is only 0.4 cm or an area of  $0.13 \text{ cm}^2$ . Since we need  $2.2 \cdot 10^{-2} \text{ m}^2$  of beam, this requires approximately 1700 sources which may be an economic impossibility. Fortunately, however, we may be able to combine many sources into one by using the arc of a single device to illuminate a large surface. If the large surface can be a grid or a highly transparent electrode, then the extraction surface can also be a grid and spaced near the plasma grid. In this manner a single unit acts as though it were composed of many low aspect ratio sources all operating in parallel.

The problem with these grids or collimated extraction surfaces is that they get extremely hot when running high energy beams. That is, the periphery of each hole intercepts some of the energetic plasma and if the surface cannot conduct the heat away, it melts. Several groups are working on this problem and the conclusion seems to be that if the sources are required to operate at energies above a few keV steady state, the current limit is not determined by Eq. (9) but by materials properties and heat conduction. Thus far, a rule of thumb appears to be that a current density of  $0.1 \text{ A/cm}^2$  is probably close to the limit. Using this as a criteria we find that at 5% burn up in

a 1500 MW(th) D-T reactor we will need a minimum of  $1.7 \cdot 10^4 \text{ cm}^2$  wall area and this starts to look like a few percent of that available.

## VI - CONCLUSIONS

In this paper an attempt has been made to survey the filling problem of the present and next generation plasma physics experiments of the mirror type. These projections should also be relevant for the start up requirements of possible reactors although no attempt has been made to take the time history of build up into account.

There are two general conclusions that this study sought to bring out. The first is the need to recognize that the injection problem is far from solved and that there are many areas in pulsed and steady state beam sources that need investigation. The second tries to underline the difficulty of producing the required neutral atom currents at energies above 20-50 keV.

## REFERENCES

1. J.D. Lawson, Proc. Phys. Soc., B70 6 (1957)
2. R.F. Post, Nuclear Fusion, Supp. I, 99 (1962)
3. R. Carruthers, P.A. Davenport and J.T.D. Mitchell, Culham Lab. Report, CLM-R85 (1967)
4. R. Carruthers, BNES Nuclear Fusion Reactor Conf., Culham, (1969) (to be published)
5. N.H. Lazar and G.R. Haste, Oak Ridge Nat. Lab. Report, ORNL-TM-2945 (1970)
6. D.J. Rose, Oak Ridge Nat. Lab. Report, ORNL-TM-2204 (1968) and Nuclear Fusion, 9 183 (1969)
7. C.F. Barnett, J.A. Ray and J.C. Thompson, Oak Ridge Nat. Lab. Report, ORNL-3113 Rev. (1964)
8. L.G. Kuo-Petravic, M. Petravic and C.J.H. Watson, BNES Nuclear Fusion Reactor Conf., Culham, (1969) (to be published)
9. S. Glasstone and R.H. Lovberg, Controlled Thermonuclear Reactions, D. Van Nostrand Co. Inc. (Princeton, 1960)
10. R.F. Post, BNES Nuclear Fusion Reactor Conf., Culham, (1969) (to be published)
11. R.G. Mills, Nuclear Fusion, 7 223 (1967)
12. P.C. Thonemanr, G. Francis, J.D. Jukes, D.W. Mason, B. McNamara, D.R. Sweetman, J.B. Taylor, C.J.H. Watson and F.A. Julian, Culham Lab. Report, CLM-R94 (1969)
13. J.R. Hiskes, Nuclear Fusion, 2 38 (1962)
14. O.B. Morgan, G.G. Kelley and R.C. Davis, Rev. Sci. Inst., 38 467 (1967)
15. R.A. D'yachkov, Soviet Phys.-Tech. Phys., 13 1036 (1969)
16. O.B. Morgan - private communication
17. L.D. Stewart, Ph.D. Thesis, Univ. of Wisconsin (1970)
18. R.A. Dandl, G.E. Guest and N.H. Lazar, Oak Ridge Nat. Lab. Report, ORNL-4080 (1967)

19. F.H. Coensgen, W.F. Cummins, R.E. Ellis and W.E. Nexsen Jr.,  
Plasma Phys. and Controlled Nuclear Fusion Res.,  
(IAEA 1968)
20. J. Marshall, Phys. Fluids, 3 134 (1960)
21. A.V. Larson, T.J. Gooding, B.R. Hayworth and D.E.T.F. Ashby,  
AIAA J., 3 977 (1965)
22. I. Henins, R.A. Jeffries, D.M. Kerr Jr. and J. Marshall,  
Bull. Am. Phys. Soc., 14 1061 (1969)
23. G.G. Kelley, IEEE Trans. Nucl, Sci., 14 (1967)
24. O.B. Morgan, Ph.D. Thesis, Univ. of Wisconsin (1970)
25. P.M. Morozov and L.N. Pilgunov, Sov. Phys. Tech. Phys.,  
8 347 (1963)
26. A.L. Bezbatchesenko, V.V. Kuznetsov, N.P. Mamakhov and  
N.N. Semashko, Plasma Phys., 6 301 (1964)
27. L.I. Artemenkov, N.I. Klochkov, V.V. Kuznetsov, V.M. Kulygin,  
N.P. Mamakhov, P.A. Mukhin, D.A. Panov, V.S. Svishchev  
and N.N. Semashko, An Injector of Fast Hydrogen Atoms,  
Proc. VII Int. Conf. on Ionization Phenomena in Gases,  
Belgrade (1965)
28. F.J. Gordon and C.C. Damm, Rev. Sci. Instr., 34 963 (1963)
29. D.P. Hammond and D.R. Sweetman, The Phoenix II Injector and  
Burial Line Commissioning, Proc. 4th Symposium on  
Engineering Problems in Thermonuclear Research, Frascati  
(1966)
30. G.G. Kelley, O.B. Morgan and R.C. Davis, Source Development  
for Thermonuclear Research at Oak Ridge, to be published  
in the Proc. of the Int. Conf. on Ion Sources, Scalay,  
France, June 18-20, 1969.

TABLE I

Amperes required to maintain a steady state 1500 MW(th) reactor.  
Forsen:

Fractional Burnup	D-D	D-T	D-H <sub>e</sub> <sup>3</sup>
0.01	4.1.10 <sup>4</sup>	8.5.10 <sup>3</sup>	8.2.10 <sup>3</sup>
0.05	8.2.10 <sup>3</sup>	1.7.10 <sup>3</sup>	1.6.10 <sup>3</sup>
0.10	4.1.10 <sup>3</sup>	8.5.10 <sup>2</sup>	8.2.10 <sup>2</sup>

TABLE II

Amperes per unit volume and total amperes required for a steady state 1500 MW(th) plant for various reactions using a 50 m<sup>3</sup> volume and the lowest density from Figure 1.

	D-D	D-T	D-H <sub>e</sub> <sup>3</sup>
amps/volume	2.4.10 <sup>2</sup>	32	11
amperes	1.2.10 <sup>4</sup>	1.6.10 <sup>3</sup>	5.4.10 <sup>2</sup>

TABLE III

Energy in KeV at which  $\sigma$ ,  $\langle\sigma v\rangle$  and  $\langle\sigma v\rangle/T^2$  peak for various reactions assuming a Maxwellian distribution.

Parameter	D-D	D-T	D-H <sub>e</sub> <sup>3</sup>
$\sigma$ (7)	2000	120	400
$\langle\sigma v\rangle$ (9)	>1000	60	>1000
$\langle\sigma v\rangle/T^2$ (6)	15	14	70

TABLE V. EFFECTIVE BRIGHTNESS OF SOME ION BEAMS

<u>Device</u>	<u>Ion Source</u>	<u>Particle Energy</u> (keV)	<u>Particle Current</u> (mA)	<u>Effective Brightness</u> (mA/MV cm <sup>2</sup> rad <sup>2</sup> )	<u>Neutral or Ion Species</u>
ORGA-1 <sup>25,26,27</sup>	Calutron	160	155	4.9 x 10 <sup>6</sup>	H <sup>0</sup>
ALICE <sup>28</sup>	Calutron	20	50	3.3 x 10 <sup>9</sup>	H <sup>0</sup>
PHOENIX-II <sup>29</sup>	Duo-Plasmatron	40	50	5.7 x 10 <sup>7</sup>	H <sub>2</sub> <sup>+</sup>
DCX-1.5 <sup>14,30</sup>	Duo-Plasmatron	40	66	4.8 x 10 <sup>8</sup>	H <sub>2</sub> <sup>+</sup>
		20	92	1.35 x 10 <sup>9</sup>	H <sup>0</sup>
DCX-2 <sup>14,30</sup>	Duo-Plasmatron	600	100	5.17 x 10 <sup>7</sup>	H <sub>2</sub> <sup>+</sup>

TABLE IV

Injection energy (KeV) for open mirror systems  
by various investigators

	D-D	D-T	D-H <sub>e</sub> <sup>3</sup>
Post <sup>(2)</sup>	150-400	60-140	
Thonemann et.al. <sup>(12)</sup>		200	
Lawson <sup>(1)</sup>	100	50	
Post <sup>(11)</sup>	400	300	400
Lazar and Haste <sup>(5)</sup>		130	
Kuo-Petravic et.al. <sup>(8)</sup>		<100	



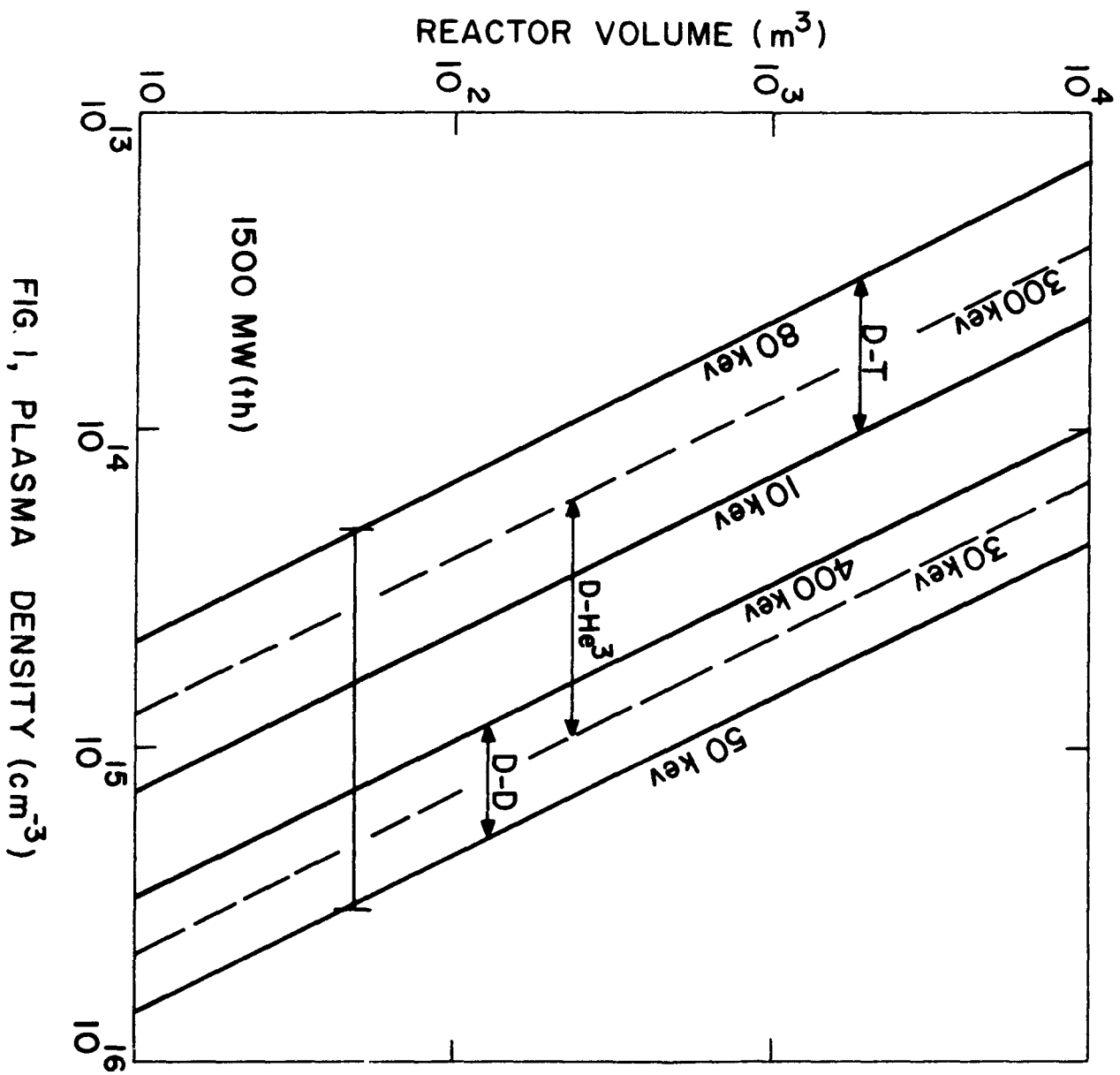


FIG. 1, PLASMA DENSITY ( $cm^{-3}$ )

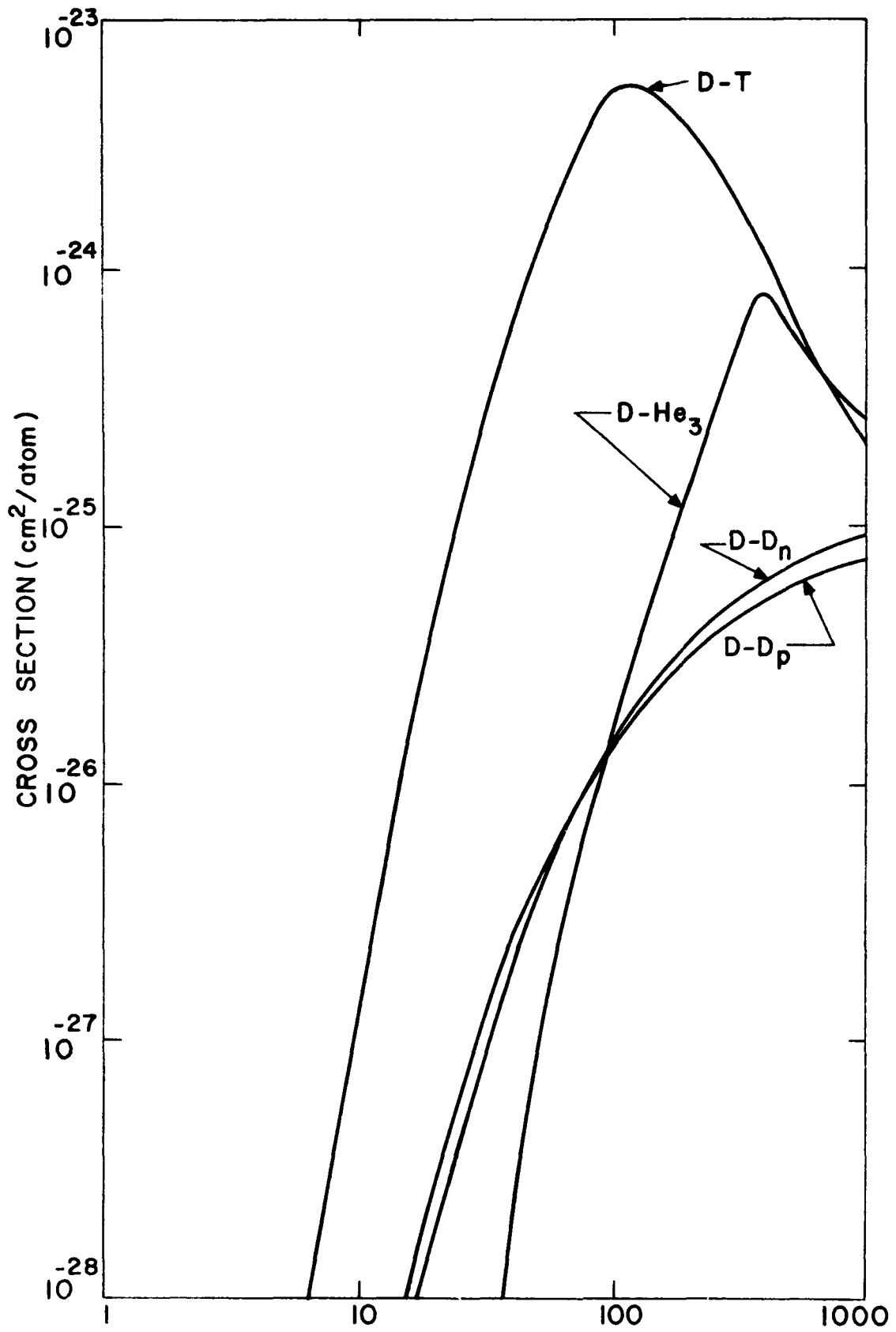


FIG. 2, ENERGY (keV)

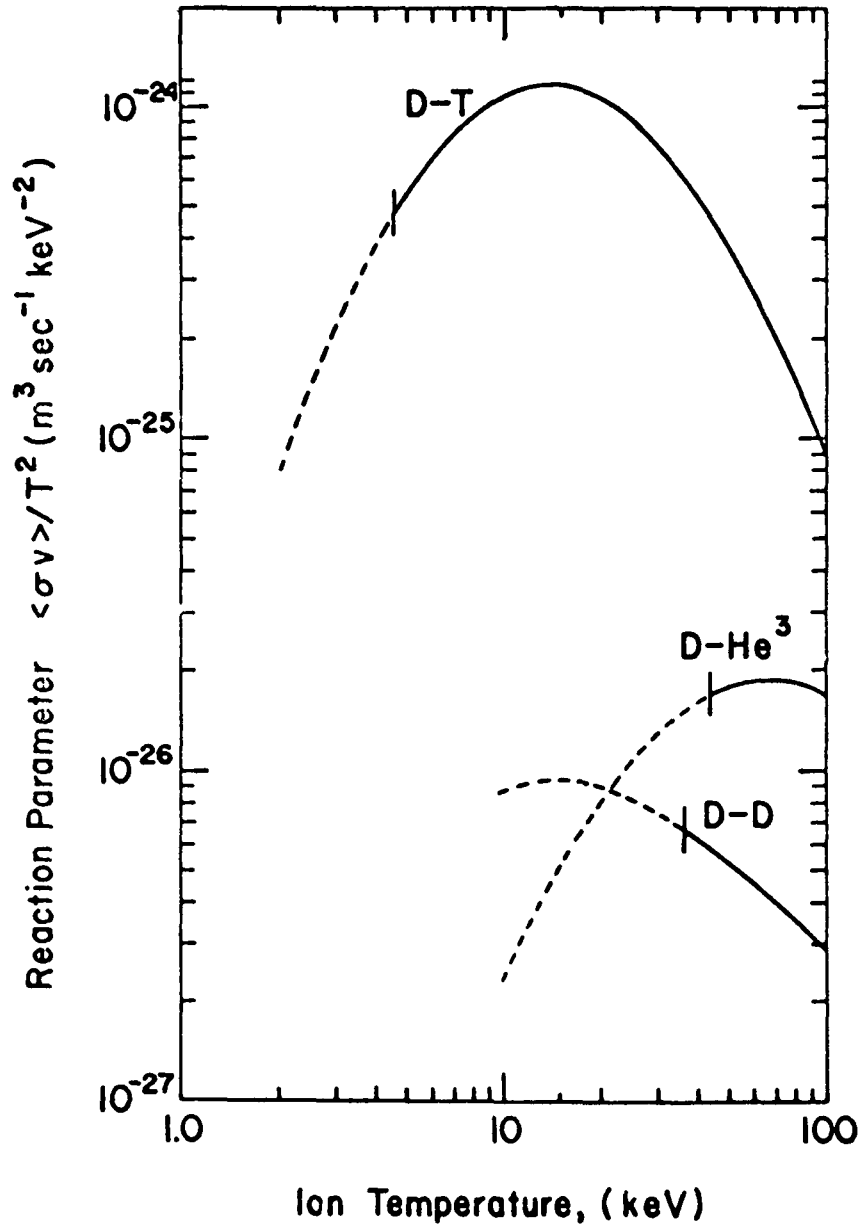


Figure 3

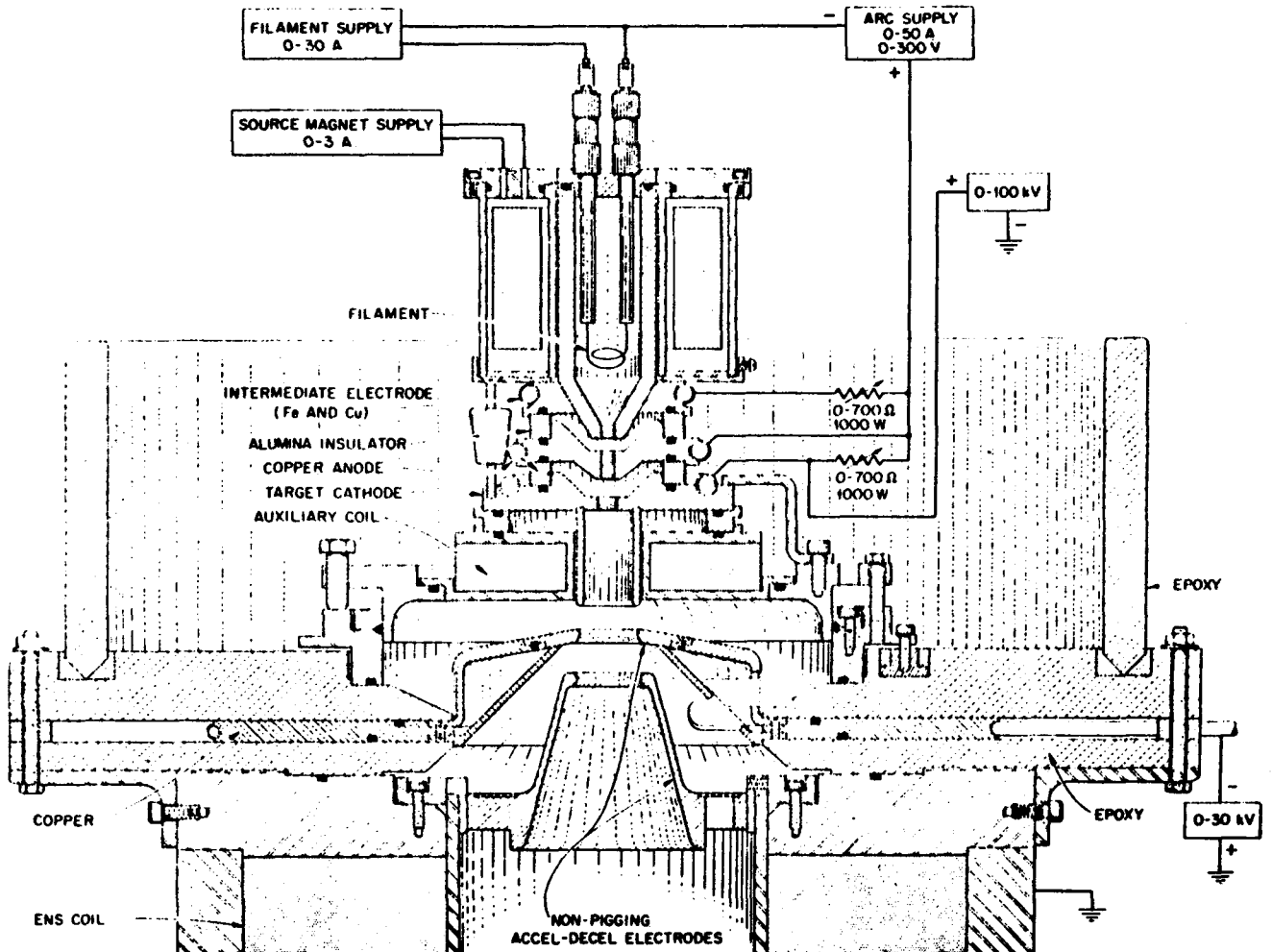
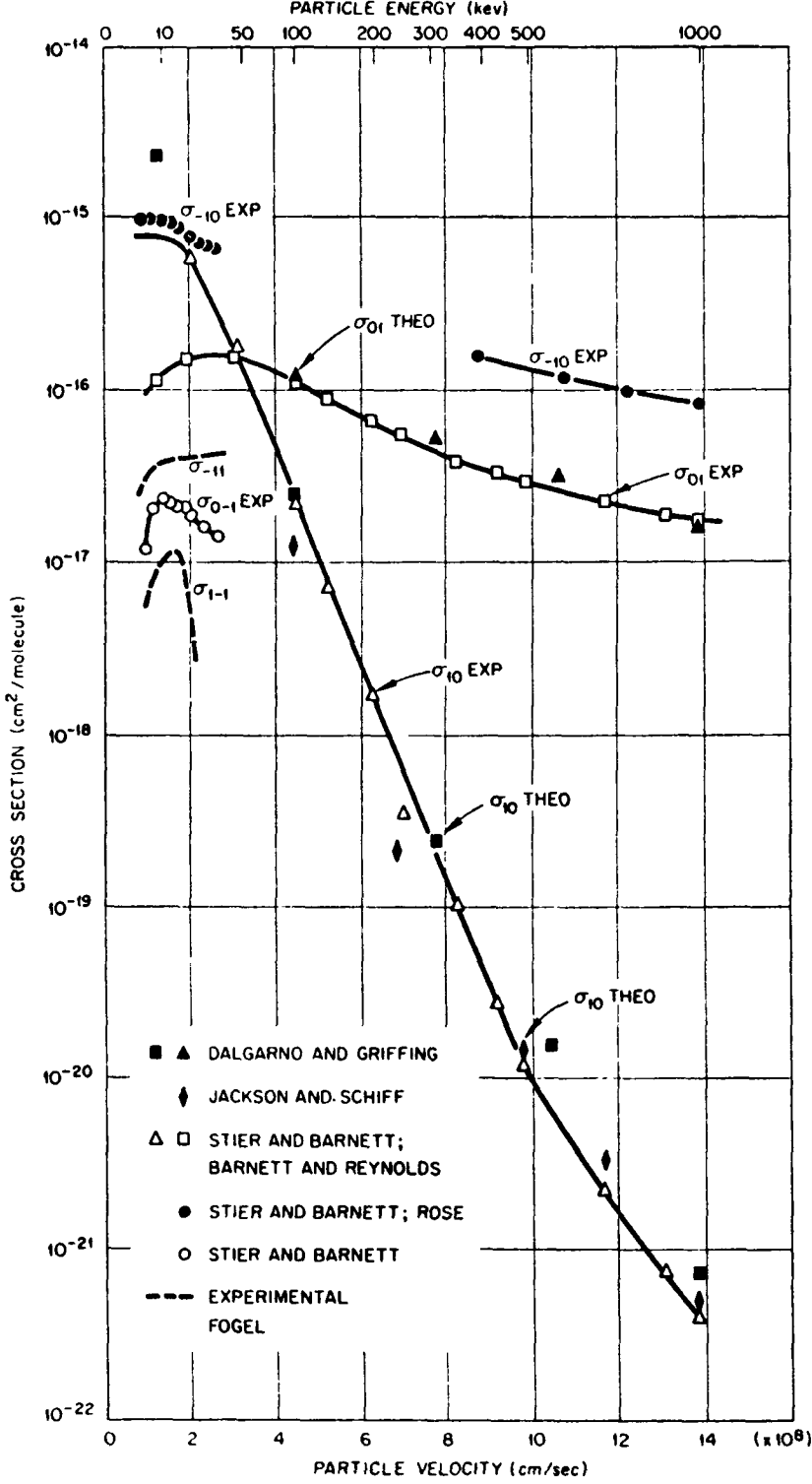


Figure 4. Four electrode ORNL source for producing intense beams of  $H_2^+$ .



Charge Transfer Cross Sections of Hydrogen Atoms and Ions in Hydrogen Gas.

Figure 5

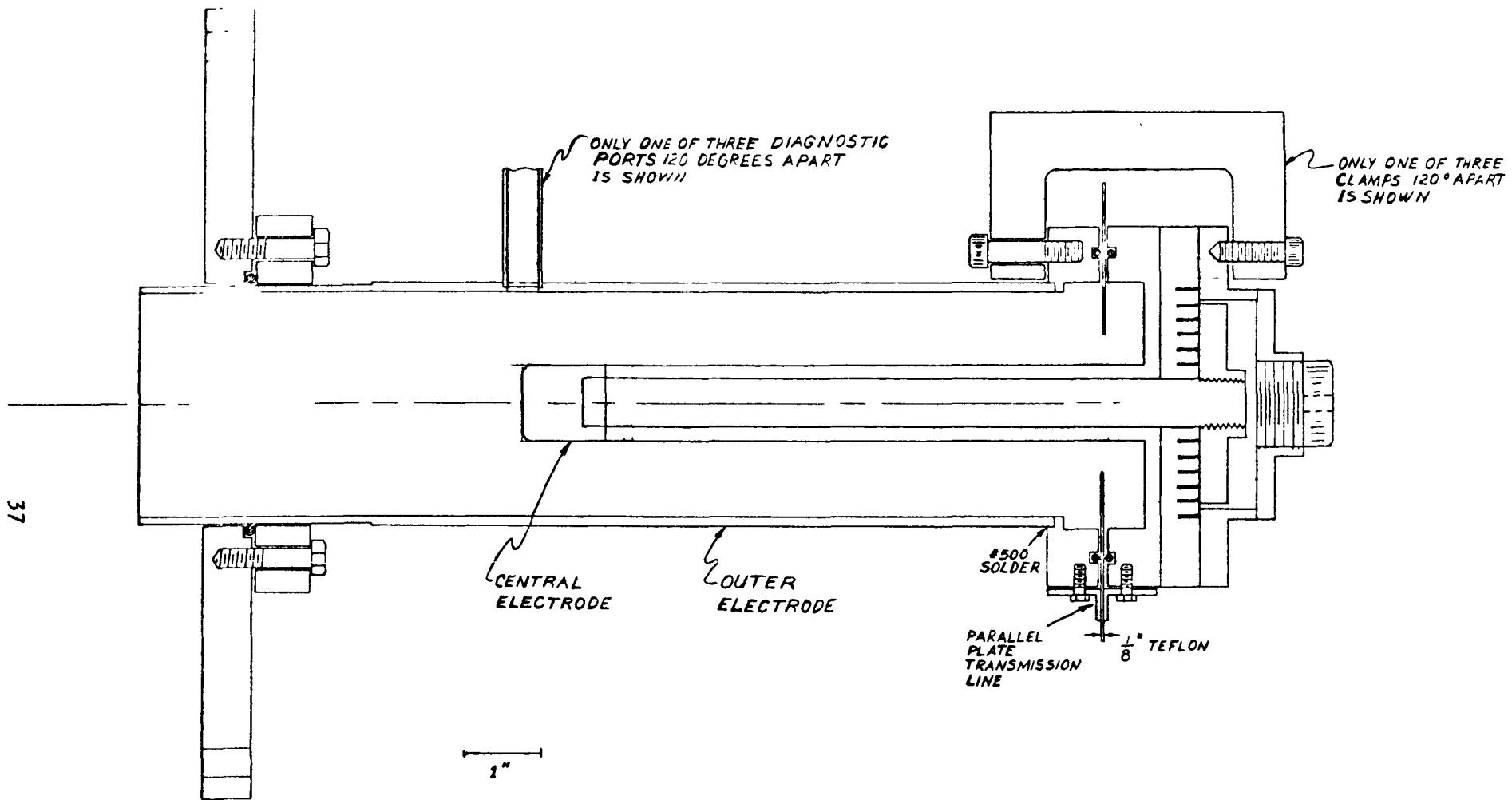


FIGURE 6

LECTURE NOTES

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- I      Fundamental Considerations for Toroidal Fusion Reactors
  
- II     Control and Ignition of Toroidal Fusion Reactors
  
- III    Bibliography

June 1, 1970

# I FUNDAMENTAL CONSIDERATIONS FOR TOROIDAL FUSION REACTORS

## A. Nuclear Physics

### 1. Reactions

Table I. Nuclear Fusion Reactions

		E	E*
1)	$D + T \rightarrow He^4 (3.52 \text{ MeV}) + n (14.06 \text{ MeV})$	17.58	3.52
	$D + D \rightarrow He^3 (0.82 \text{ MeV}) + n (2.45 \text{ MeV})$		
2)		3.6	2.4
	$D + D \rightarrow T (1.01 \text{ MeV}) + p (3.03 \text{ MeV})$		
3)	$D + He^3 \rightarrow He^4 (3.67 \text{ MeV}) + p (14.67 \text{ MeV})$	18.34	18.34
4)	$T + T \rightarrow He^4 + n + n$	11.32	

### 2. Cross Sections, Reactivity, and Relative Output

The first two figures are compiled from data presented in Glasstone and Lovberg.<sup>1</sup>

Figure 3 is a correction of Fig. 2 taking into consideration the difference in temperatures of the electrons and ions and the partial pressure of the product alphas.

### 3. The Blanket

Except for the nuclear reactions described under item 1, all the nuclear physics of a thermonuclear reactor is involved with the blanket. These questions are not in the scope of this lecture.



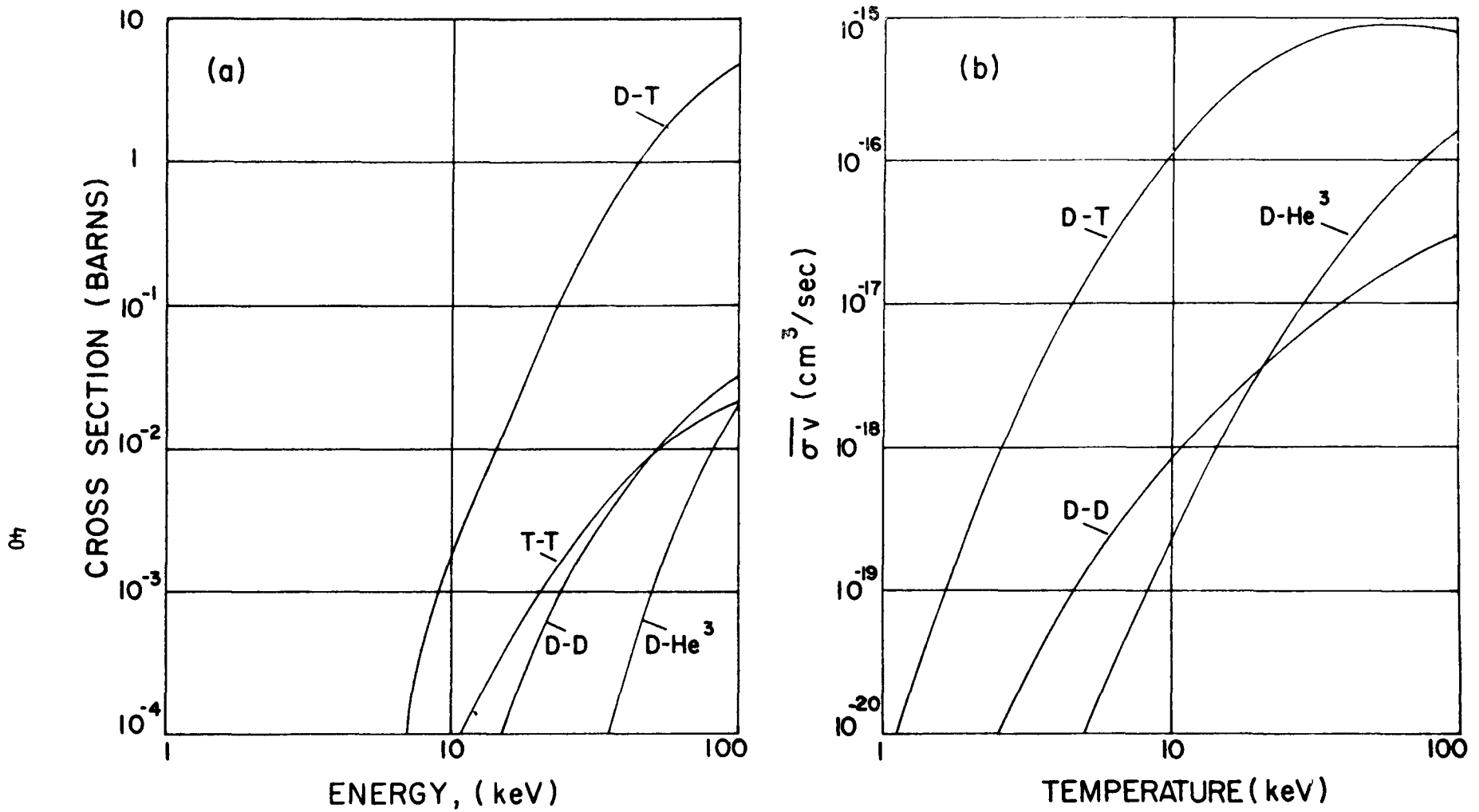


Fig. 1. Energy Dependence of Cross Section and Reactivity.

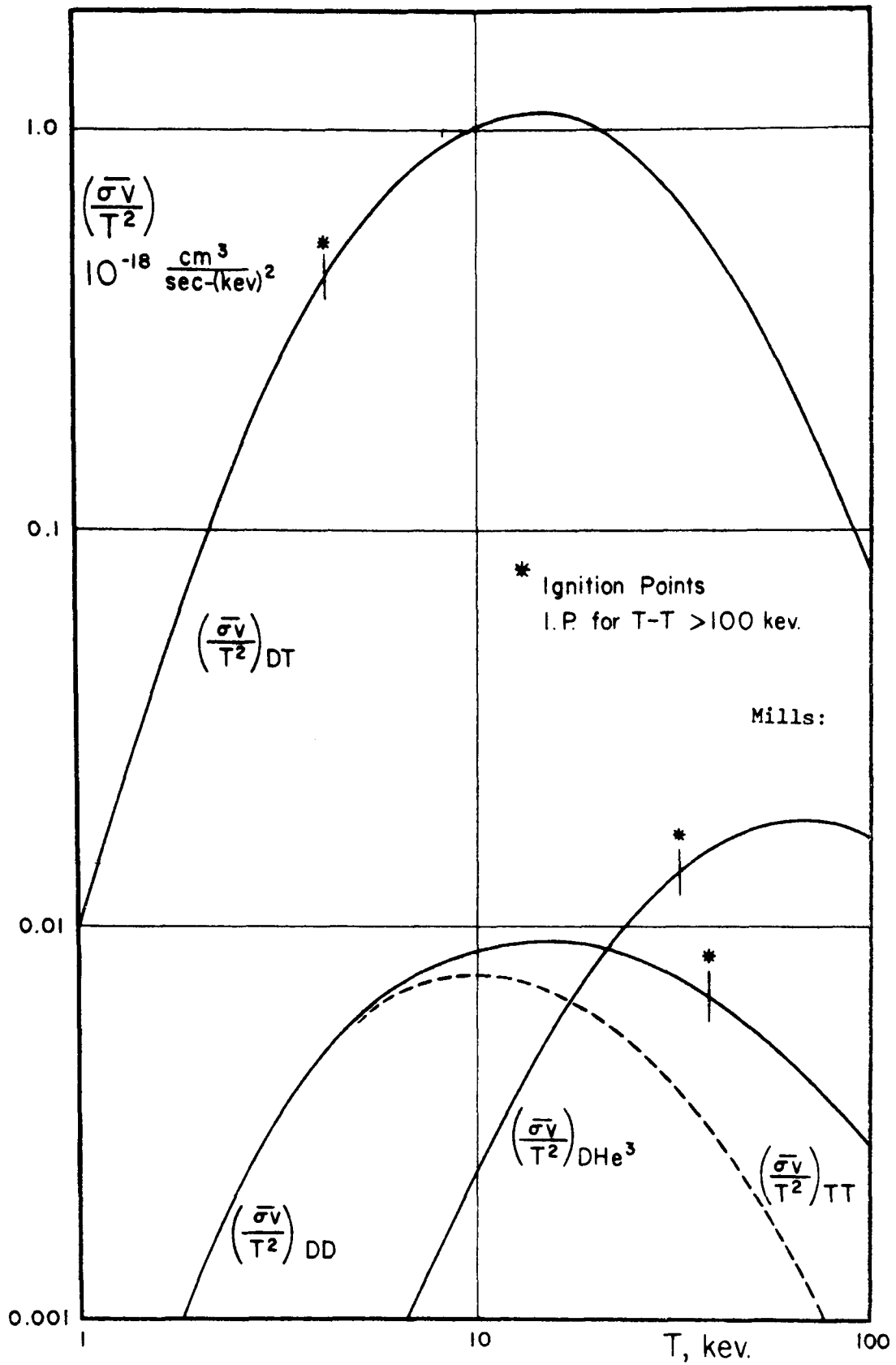


Fig. 2. Reaction Rate as a Function of Temperature.

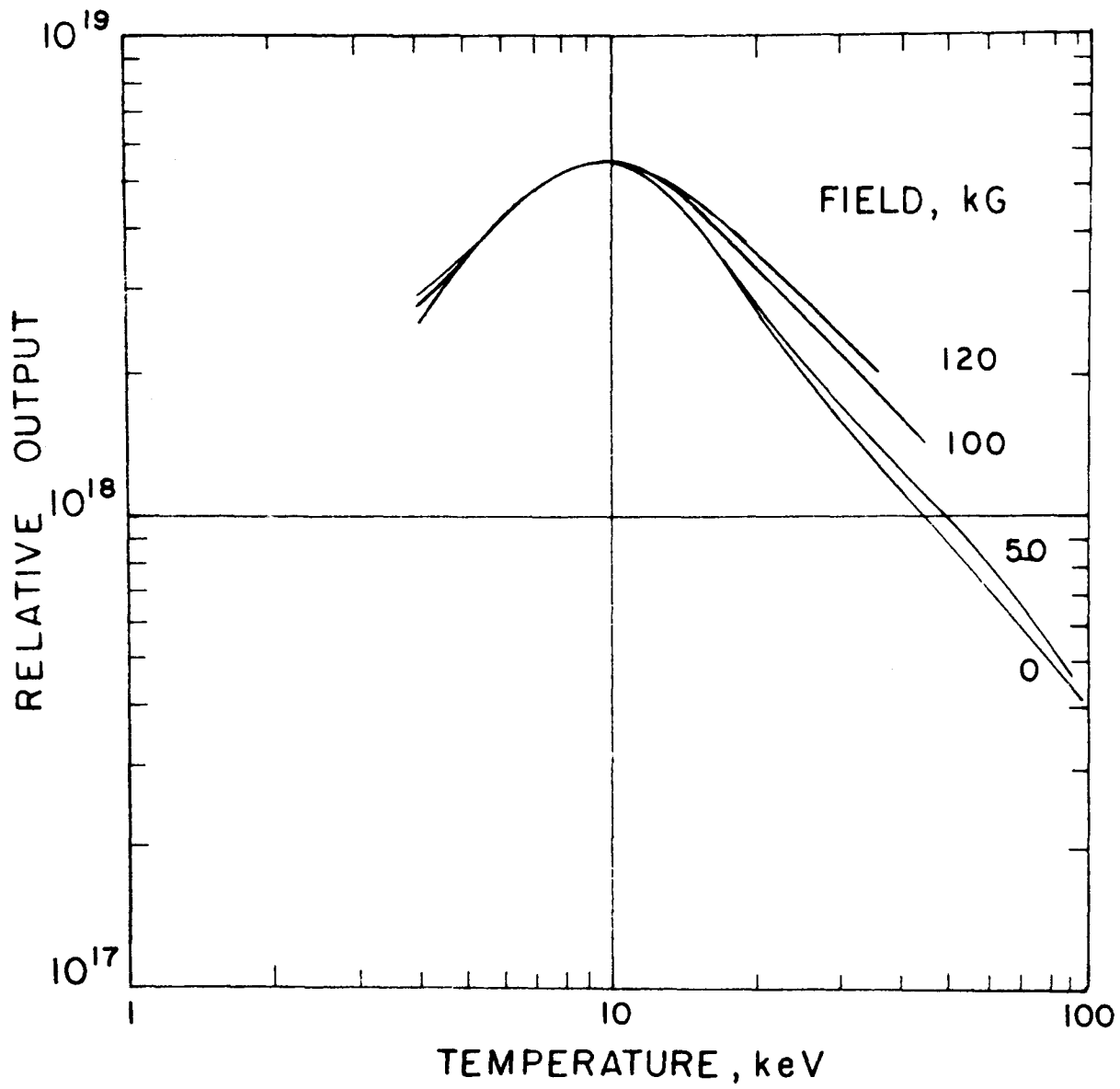


Fig. 3. Relative Power Output (in arbitrary units) of  $\beta$ -Limited Reactor.

## B. Plasma Physics

### 1. Resistivity

a) Spitzer<sup>2</sup> has calculated the resistivity of an ionized hydrogenic gas to be (the electron temperature,  $T_e$ , measured in keV)

$$\rho = (1.653 \times 10^{-7}) \ln \Lambda / T_e^{3/2} \quad (\text{ohm-cm}) \quad (1)$$

In this expression  $\Lambda$  is the ratio of the Debye shielding distance to the impact parameter of a close collision producing a  $90^\circ$  deflection. See Ref. 2.

Numerically,

$$\Lambda = 9.308 \times 10^{13} T_e / Z Z_1 n_e^{1/2} \quad (2)$$

with the electron density,  $n_e$ , expressed in  $\text{cm}^{-3}$ , and  $Z$  and  $Z_1$  are the charge numbers of the two interacting species. Figure 4 presents this relationship.

b) The presence of a strong electron current produces turbulence and anomalously high resistivity. Figure 5 shows the correlation of anomalous resistivity with the ratio of the electron drift velocity to the electron thermal velocity.

### 2. Equilibration Times

a) The self collision time is given by Spitzer<sup>2</sup> as

$$\tau_{\text{eq}} = (0.45 \times 10^{12}) A^{1/2} T_e^{3/2} / n Z^4 \ln \Lambda \quad (3)$$

where  $A$  is the mass number.

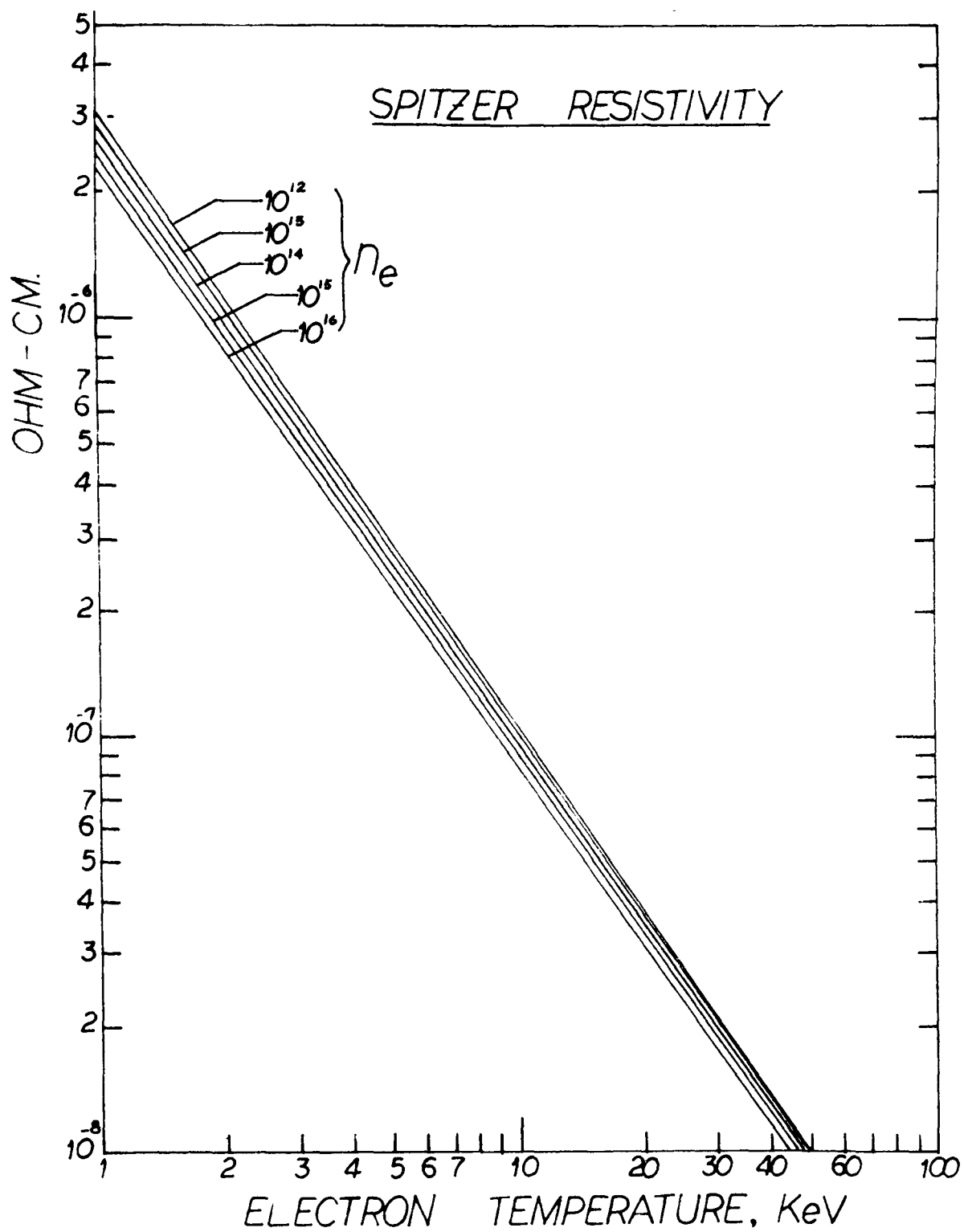


Fig. 4. Temperature Dependence of Plasma Resistivity.

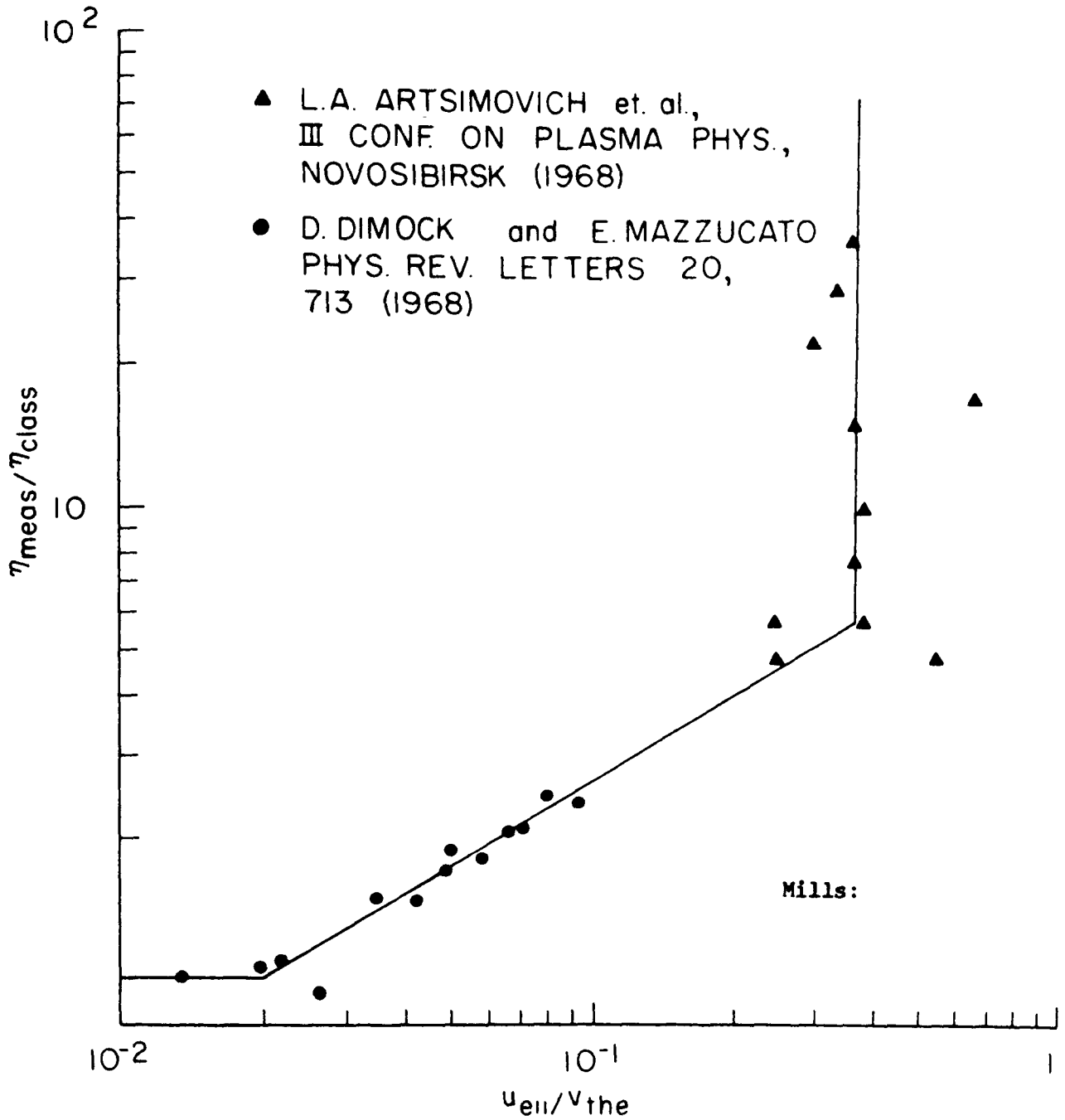


Fig. 5. Comparison of Anomalous Resistivity Experimental Results  
 Obtained in Tokamak-Confined and C Stellarator-Confined Hydrogen Plasmas.

Under plasma conditions typical of those assumed for a thermonuclear reactor, the self collision time for electrons will be of the order of a microsecond, and for the ions, of the order of 50 microseconds. Therefore, these populations may be considered to have a well defined temperature.

b) Electron-Ion Equilibration Time

Spitzer<sup>2</sup> gives this as

$$\tau_{ei} = (0.9899 \times 10^{13}) A_i T_e^{3/2} / n_i Z_i^2 \ln \Lambda_{ei} \quad . \quad (4)$$

For a 50-50 D-T plasma the electrons will be exchanging energy with a gas of average mass under 2.5. In this special case

$$\tau_{ei} = (2.472 \times 10^{13}) T_e^{3/2} / n_i \ln \Lambda \quad . \quad (5)$$

This is plotted in Fig. 6.

c) Alpha Heating of D-T Plasmas

With  $A_i = 4$  and  $Z_i = 2$  and a  $\Lambda$  appropriate for alpha-electron collisions, the alpha-electron cooling time may be calculated from (4) above. The alphas will cool according to this decay time towards the electron sea temperature until their velocity is low enough that they exchange more rapidly with the ions. See Butler and Buckingham.<sup>3</sup> The energy vs time curve for an energetic alpha will therefore be similar to Fig. 7.

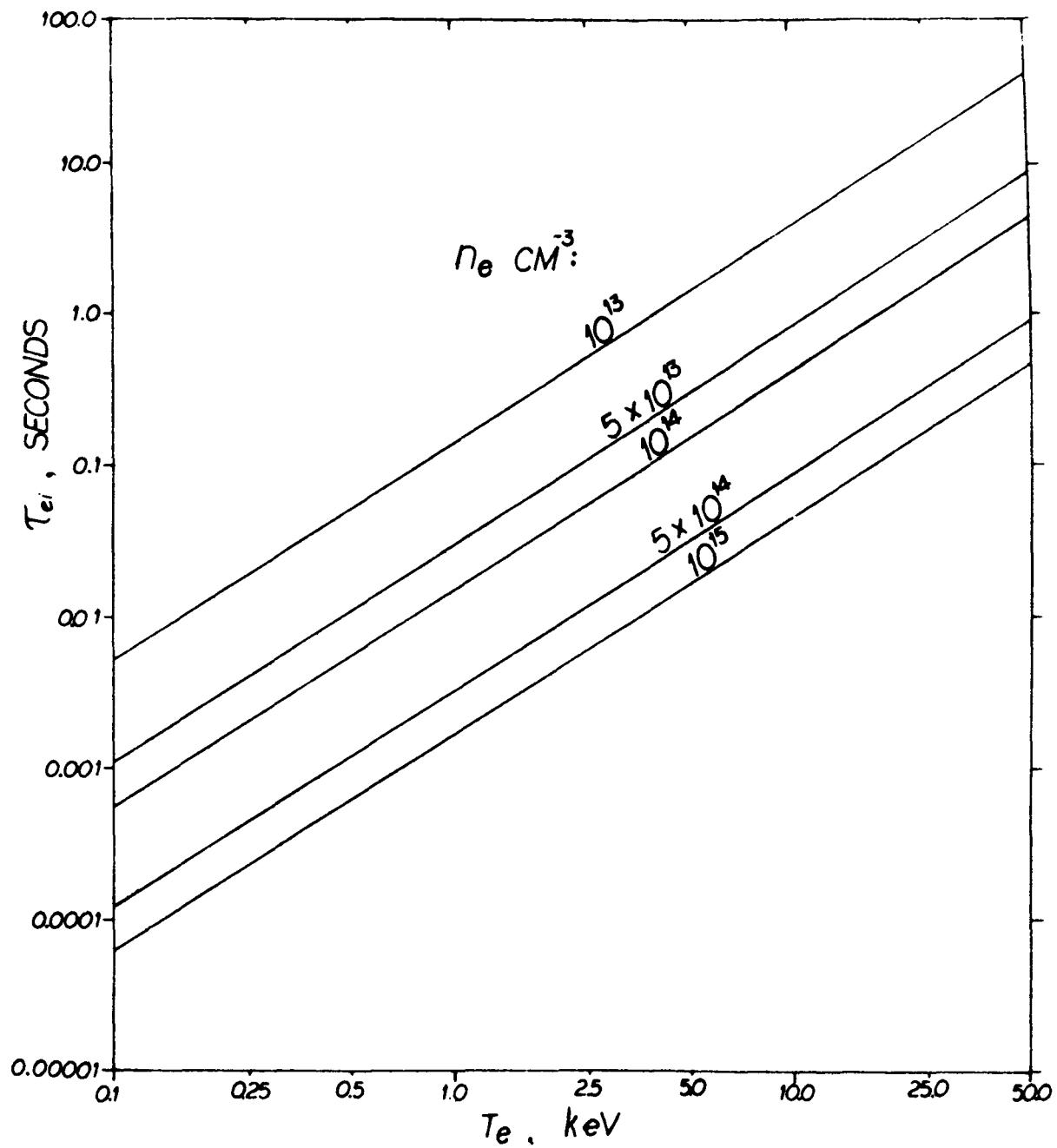


Fig. 6. Electron-Ion Equilibration Time.



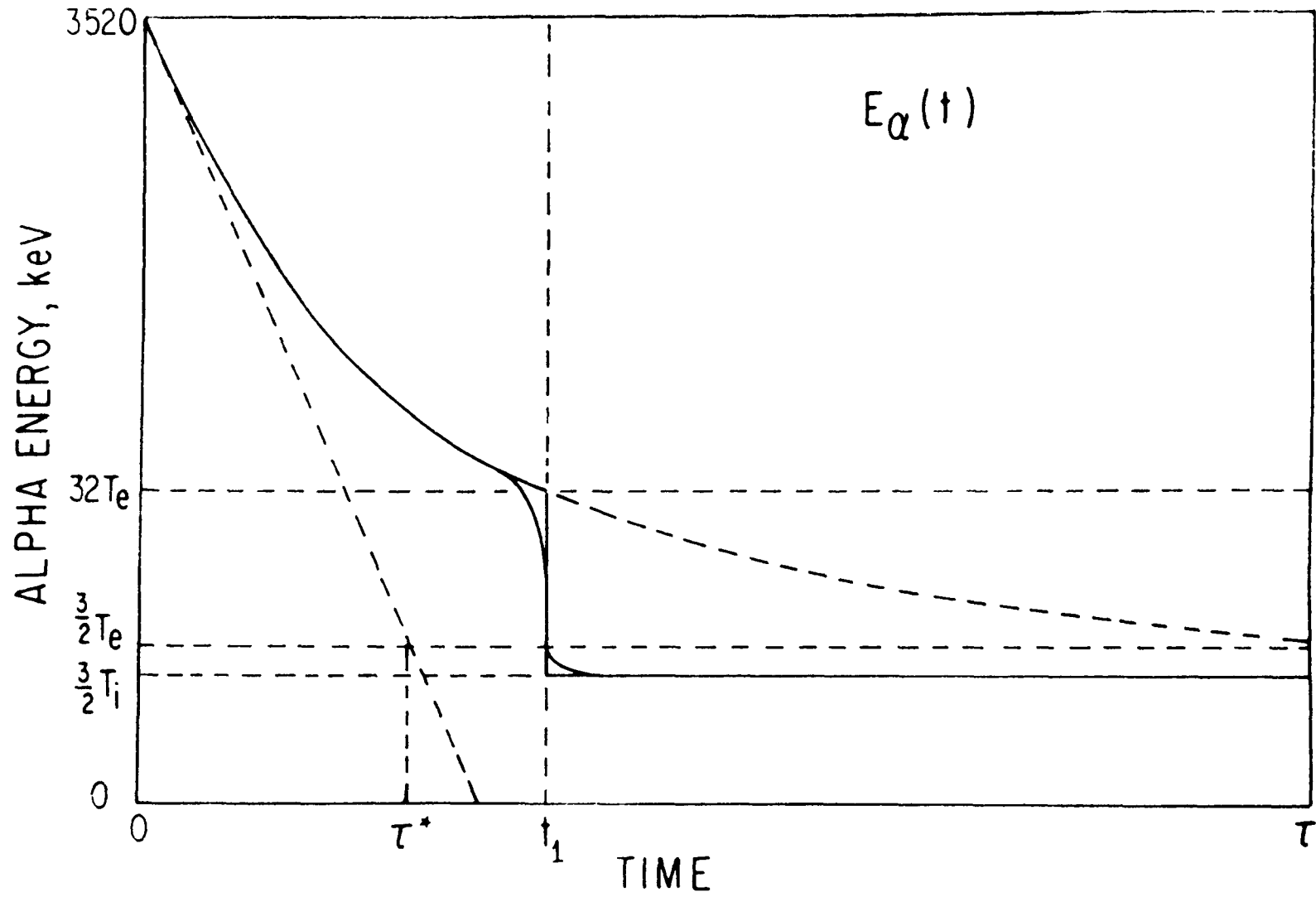


Fig. 7. Alpha Energy as a Function of Time.

### 3. Bremsstrahlung

Bremsstrahlung radiation from a hydrogenic plasma has been calculated by Spitzer.<sup>2</sup> The result is

$$P_B = (0.4854 \times 10^{-23}) n_e^2 T_e^{1/2} \left( 1 + \sum_i Z_i^2 \frac{n_i}{n_e} \right) \frac{\text{ergs}}{\text{cm}^3 \text{-sec}} . \quad (6)$$

### 4. Synchrotron Radiation

The point of departure for the study of synchrotron radiation is the work of Trubnikov and Kudryavtsev.<sup>4</sup> When metallic reflection is considered quantitatively, the amount of synchrotron radiation to be expected from a device may be calculated.<sup>5</sup>

Typical results are illustrated in Fig. 8.

### 5. Confinement and Stability

a) Magnetic confinement will depend on the ratio of plasma pressure to magnetic pressure,  $\beta$ , defined as

$$\beta = \frac{8 \pi n k T}{B_o^2} . \quad (7)$$

b) Classical analysis yields the following diffusion coefficient:

$$D = 130 \beta / T^{3/2} \quad \text{cm}^2/\text{sec} . \quad (8)$$

c) Bohm suggested the following semi-empirical relation,

$$D = 5.8 \times 10^9 T/B \quad \text{cm}^2/\text{sec} . \quad (9)$$

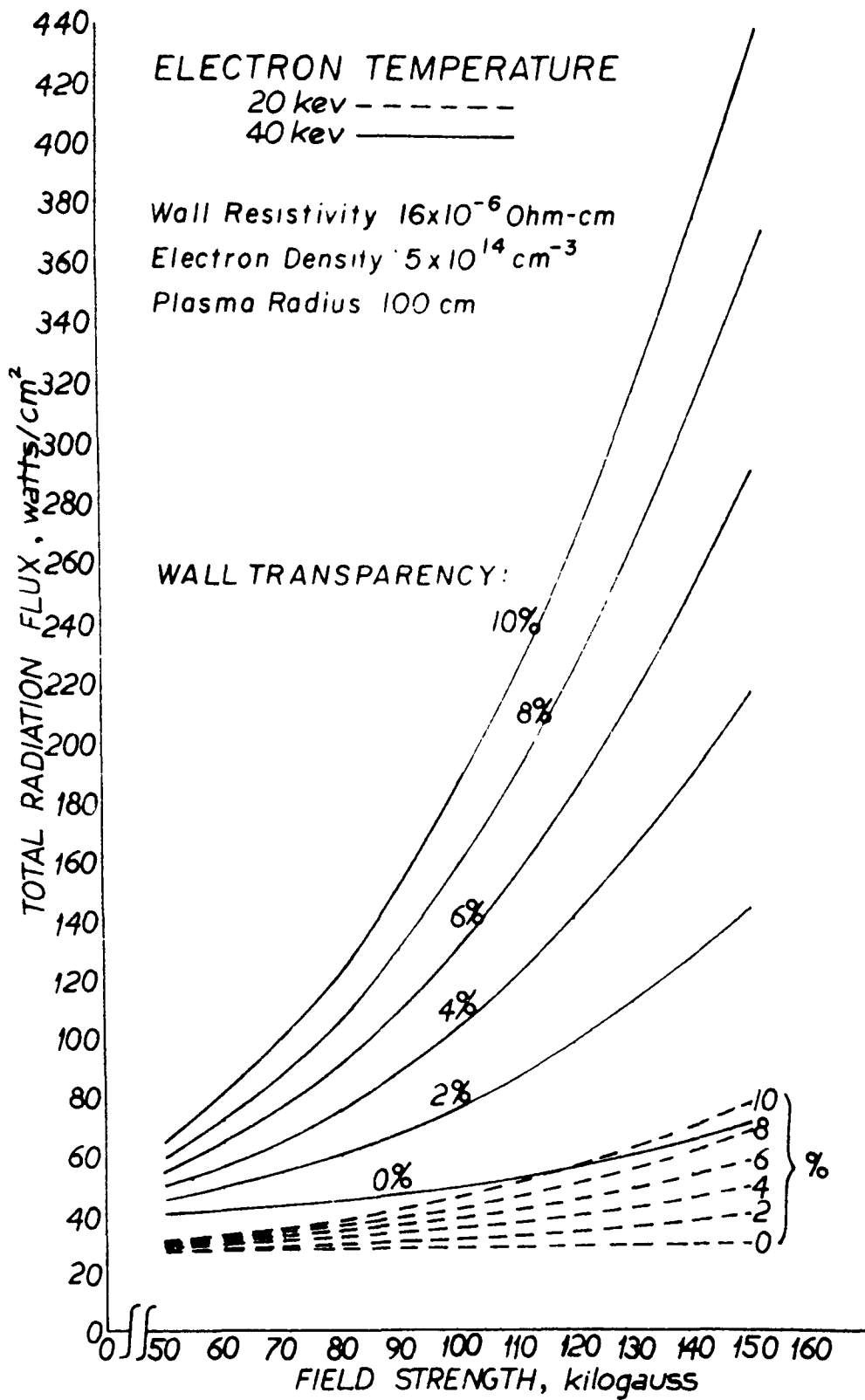


Fig. 8. Radiation Loss from a Cylindrical Plasma as a Function of Field Strength, Electron Temperature, and Wall Openings.

d) Experiment

For many years toroidal confinement seemed to be consistent with Bohm. In recent years Bohm has been exceeded by a factor of at least one order of magnitude and perhaps two in Tokamaks.

e) Confinement time,  $\tau$ ,

$$\tau \approx a^2/4D$$

where  $a$  is the radius of a cylindrical plasma. Assuming conditions of  $B = 10^5$  gauss,  $T = 10$  keV,  $\beta = 0.1$ , and  $a = 100$  cm,

$$\tau_{\text{classical}} = 10 \text{ minutes}$$

$$\tau_{\text{Bohm}} = 17 \text{ milliseconds}$$

One second is required.

C. Reactor Considerations

1. Lawson Criterion<sup>6</sup>

$$\left[ n_i^2 p(1-p)(\overline{\sigma v})U\tau + 3n_i kT + P_B \tau \right] \eta = 3n_i kT + P_B \tau \quad (10)$$

In this expression  $p$  is the fraction of  $n_i$  that is tritium,  $U$  is the total energy released in a single fusion event ( $\sim 22$  MeV),  $\tau$  is the plasma confinement time, and  $P_B$  is bremsstrahlung power, which from (6) above can be written as

$$P_B = b n_i^2 T^{1/2} \quad (11)$$

By dividing (10) by  $n_i$  we find that this gives  $n_i \tau$  as a function of

T:

$$\left[ p(1-p)(\overline{\sigma v})U(n_i \tau) + 3kT + bT^{1/2}(n_i \tau) \right] \eta = 3kT + bT^{1/2}(n_i \tau)$$

or

$$n_i \tau = \frac{3kT(1/\eta - 1)}{(\overline{\sigma v})p(1-p)U - (1/\eta - 1)(bT^{1/2})} \quad (12)$$

## 2. Equilibrium Condition

$$n_i^2 p(1-p)(\overline{\sigma v})cE^* = \frac{3}{2}mk(T_i + T_e) \quad (13)$$

In this expression  $E^*$  is the energy released to charged particles in the plasma (alpha energy), and  $c$  is the fraction of this that remains after bremsstrahlung and synchrotron radiation. The feed rate ( $m$  ions (or electrons) per second per cubic centimeter) determines the variation of density in time by

$$\frac{dn_i}{dt} = m - \frac{n_i}{\tau} \quad (14)$$

and in equilibrium

$$n_i^2 p(1-p)(\overline{\sigma v})cE^* = \frac{3}{2}k\frac{n_i}{\tau}(T_i + T_e)$$

or

$$n_i \tau = \frac{3k(T_i + T_e)}{2p(1-p)(\overline{\sigma v})cE^*} \quad (15)$$

a result of the greatest importance. Figure 9 compares these criteria.

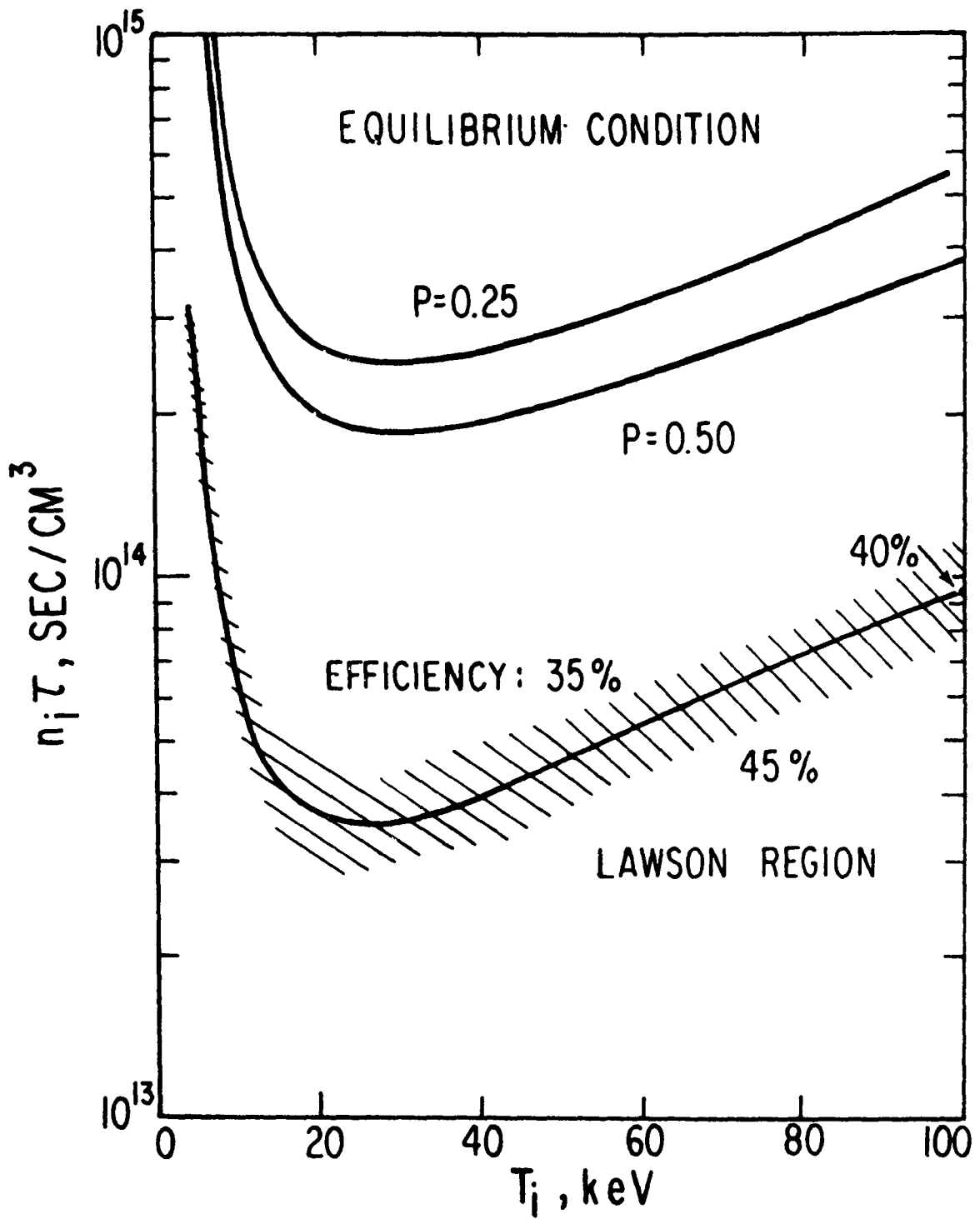


Fig. 9. The Lawson Criterion and the Equilibrium Condition as a Function of Ion Temperature.

### 3. Scaling Laws

From the definition of  $\beta$  and the fact that fusion results from binary collisions, it follows that the output of a reactor will scale as follows:

$$\text{Power} \sim r^3 \beta^2 B^4 \quad . \quad (16)$$

### 4. Wall Limitations

The power output available from a given size of machine will be limited by the physical properties of materials, no doubt the vacuum wall. The operating level may be characterized by the parameter introduced by Rose and his students,<sup>7, 8, 9</sup> the neutron energy transport per unit area of the wall. A typical characteristic is shown in Fig. 10.

#### D. Hypothetical Machines

##### 1. Closed vs Open

Open machines (mirrors) cannot provide the confinement time sufficient to meet the equilibrium condition. Energetic injection is therefore necessary to maintain the reactor in steady state.

##### 2. Closed Configurations

This category includes stellarators and Tokamaks, but also floating ring machines, toroidal  $\theta$ -pinches, and astronons. Closed machines imply unfavorable field curvature.

##### a) Stellarator<sup>10</sup>

This type of machine requires stabilizing windings.

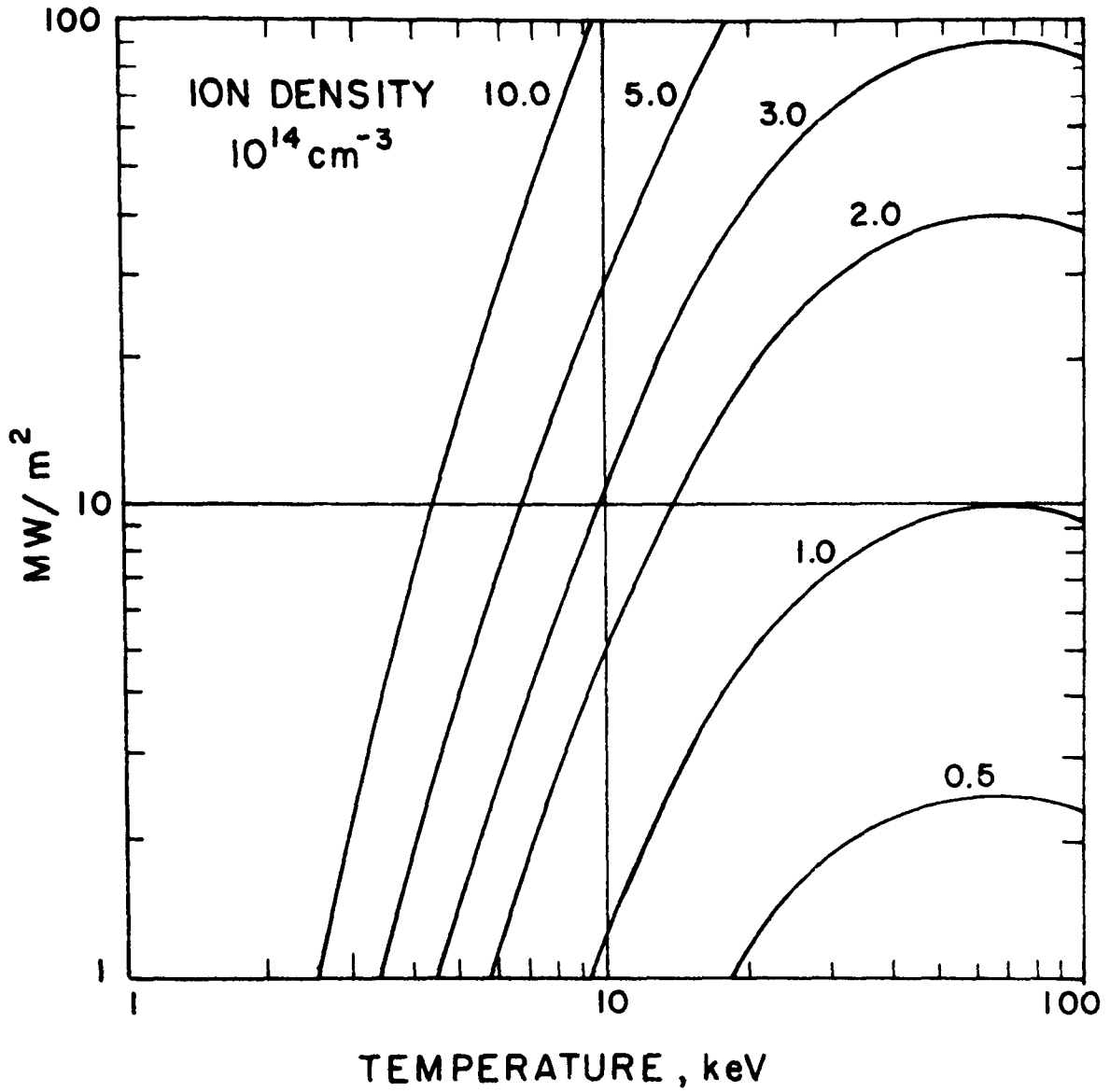


Fig. 10. Neutron Energy Transport Density as a Function of Ion Temperature and Density. Plasma Radius = 100 cm.



Theory is not sufficiently advanced to predict the  $\beta$  limit. It may be in the 10-20% range.

b) Tokamak<sup>11, 12</sup>

This type of machine uses an induced plasma current to provide the rotational transform. The following definitions are needed:

Aspect ratio,  $A \equiv \text{Major Radius/Minor Radius}$

$q = B_z / AB_\theta$ , where  $B_z$  is the externally applied solenoidal confinement field, and  $B_\theta$  is the poloidal field generated by the plasma current. The reciprocal of  $q$  is the rotational transform in units of  $2\pi$ .  $q$  is sometimes called the stability margin.

$$\beta_\theta = 8\pi nkT / B_\theta^2, \text{ therefore } \beta_\theta = \beta (1 + A^2 q^2) .$$

$\beta_\theta$  must be less than  $A$ , possibly as small as 1. Therefore  $\beta$  must be limited to the 1-4% range.

c) Common Features

Both machines are governed by the equilibrium condition (15) above, and the presence of the ohmic heating current is significant only at low temperatures. Therefore, characteristics such as relative output (Fig. 3), the difference in temperature between the electron and ion seas (Fig. 11), and the relative tritium burnup per pass (Fig. 12) are all similar.

d) Differences

Helical windings vs induced current. Possible beta ad-

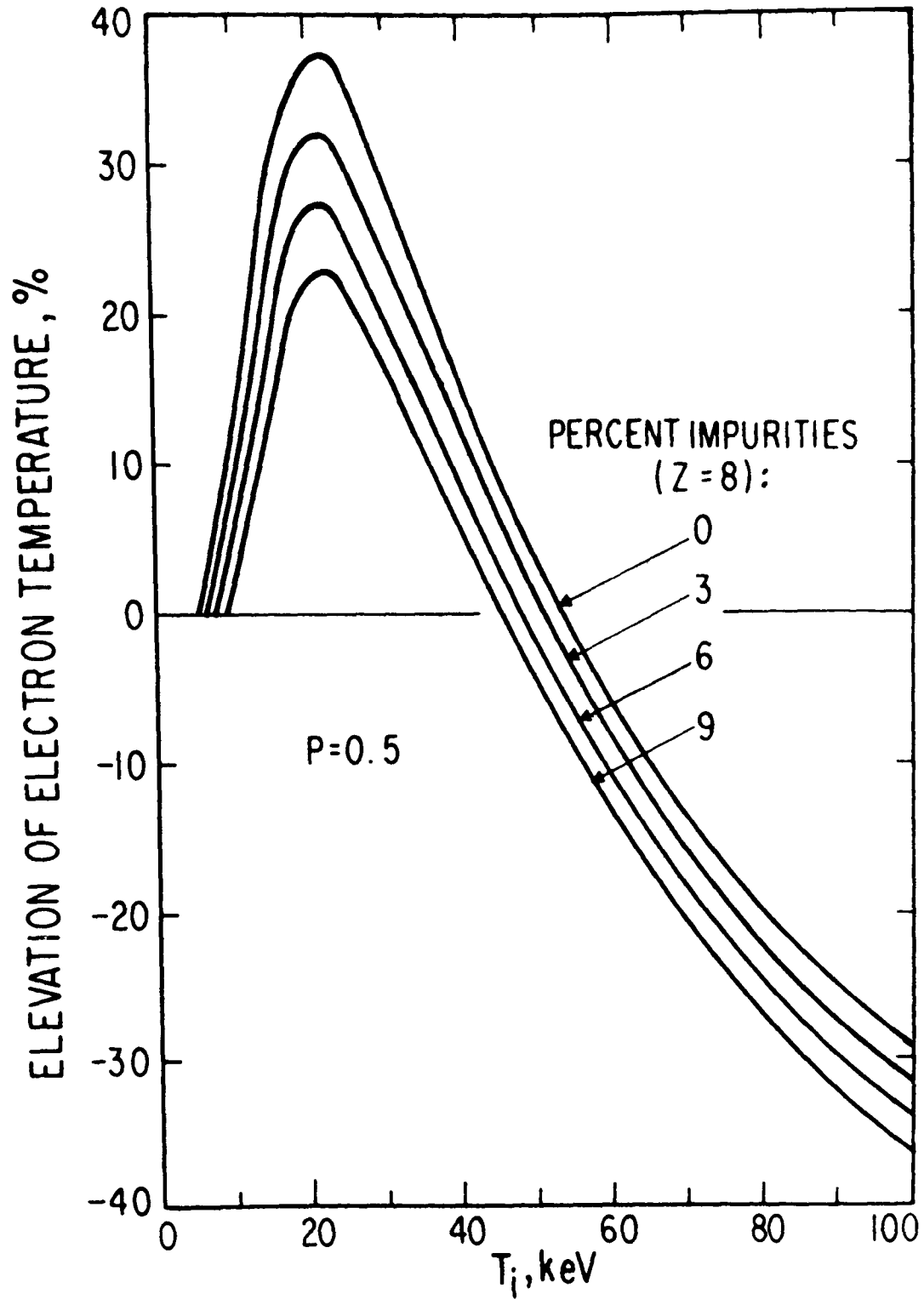


Fig. 11. Typical Difference in Temperature between the Ion and Electron Seas. The curves are field and density dependent.

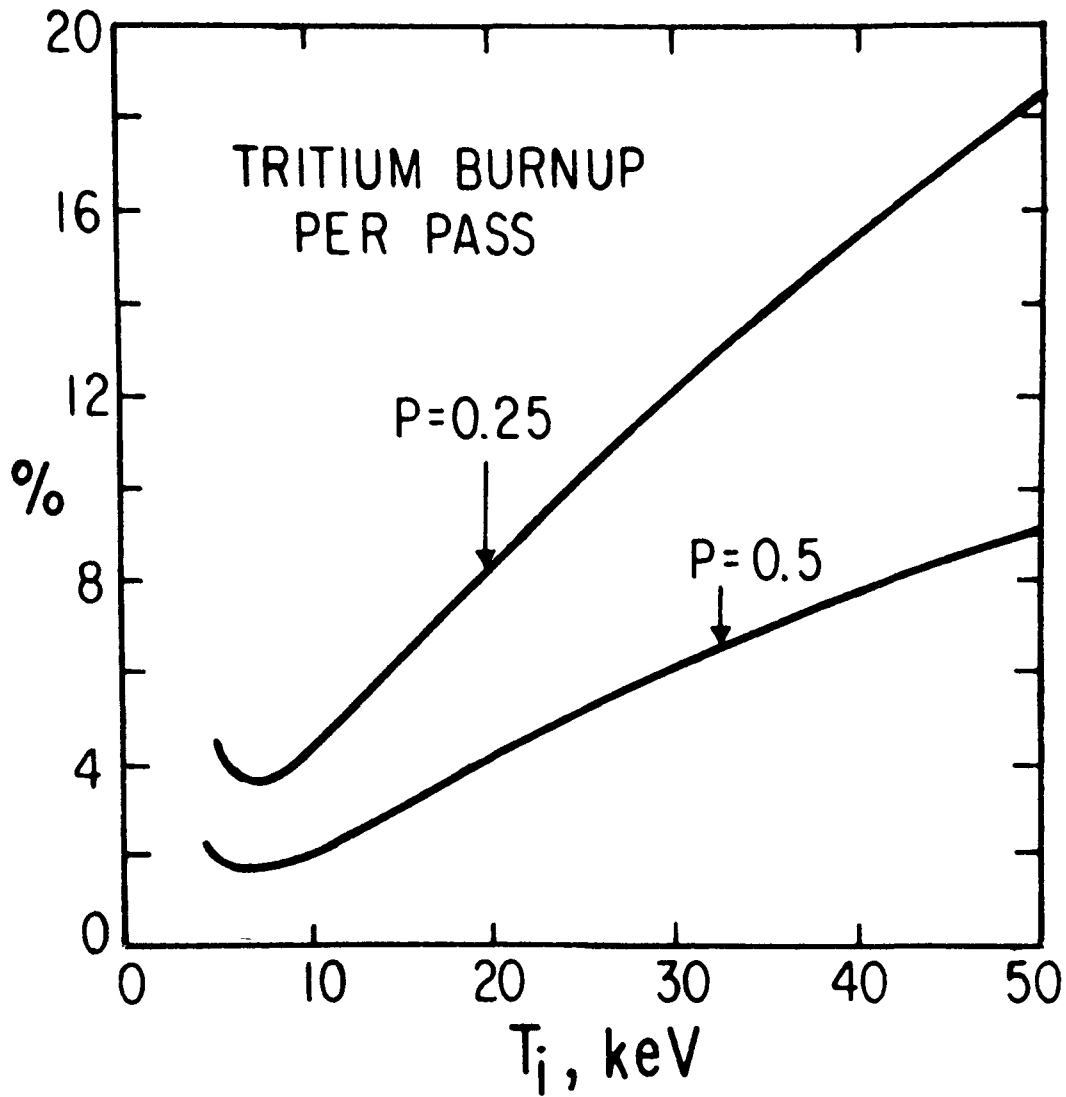


Fig. 12. Typical Relative **Fuel** Burnup.

vantage for stellarators. The cyclic operation of Tokamak is not a severe disadvantage, since the operating period is long ( minutes to hours ).

#### E. Sizes and Costs

##### 1. Estimates

Preliminary cost estimates have been made ( see, for example, Carruthers<sup>13</sup>, Mills<sup>10</sup>, Rose<sup>14</sup> ). These machines, if successful, would appear to be economically competitive.

##### 2. Fuel

Note from the following table that it is the fuel cost that dominates the cost difference.

Table II. Comparative Power Costs

	Oyster Creek Nuclear Station	Oyster Creek Coal Station	Fusion Power Station
Capital Charge	1.55	1.52	1.75
Operations and Maintenance	0.50	0.42	0.53
Fuel Cost	<u>1.62</u>	<u>2.37</u>	<u>0.02</u>
Power Cost, mills/kWh	3.67	4.31	2.30

## II CONTROL AND IGNITION OF TOROIDAL FUSION REACTORS

### A. Stability

#### 1. Variation of the Equilibrium Condition

The equilibrium condition requires the constancy of a function,

$$\psi = m \tau^2 p(1-p) c(\overline{\sigma v}) / (T_i + T_e) \quad . \quad (17)$$

Defining a function,  $S = c(\overline{\sigma v}) / T_i + T_e$ , we have

$$\psi = m \tau^2 p(1-p) S \quad , \quad (18)$$

and

$$\frac{\delta \psi}{\psi} = \frac{\delta m}{m} + 2 \frac{\delta \tau}{\tau} + \left( \frac{1-2p}{1-p} \right) \frac{\delta p}{p} + \frac{\delta S}{S} = 0 \quad (19)$$

for stability.

Thus the equilibrium tends to be unstable against fluctuation in the fuel feed rate, the confinement time, and the fuel mixture (unless  $p = 0.5$ ). The behavior with respect to  $S$  (essentially the temperature dependence) can be understood from Fig. 13.

#### 2. Need for Control

Feedback control is necessary if one desires to operate at more advantageous temperatures below the maximum of the stability function. Figure 14 illustrates the problem.

### B. Various Forms of Feedback Control

Stabilization can result from "Bohm-type" confinement. Most convenient, perhaps, is mixture control. Figures 15, 16, and 17 demonstrate these effects.

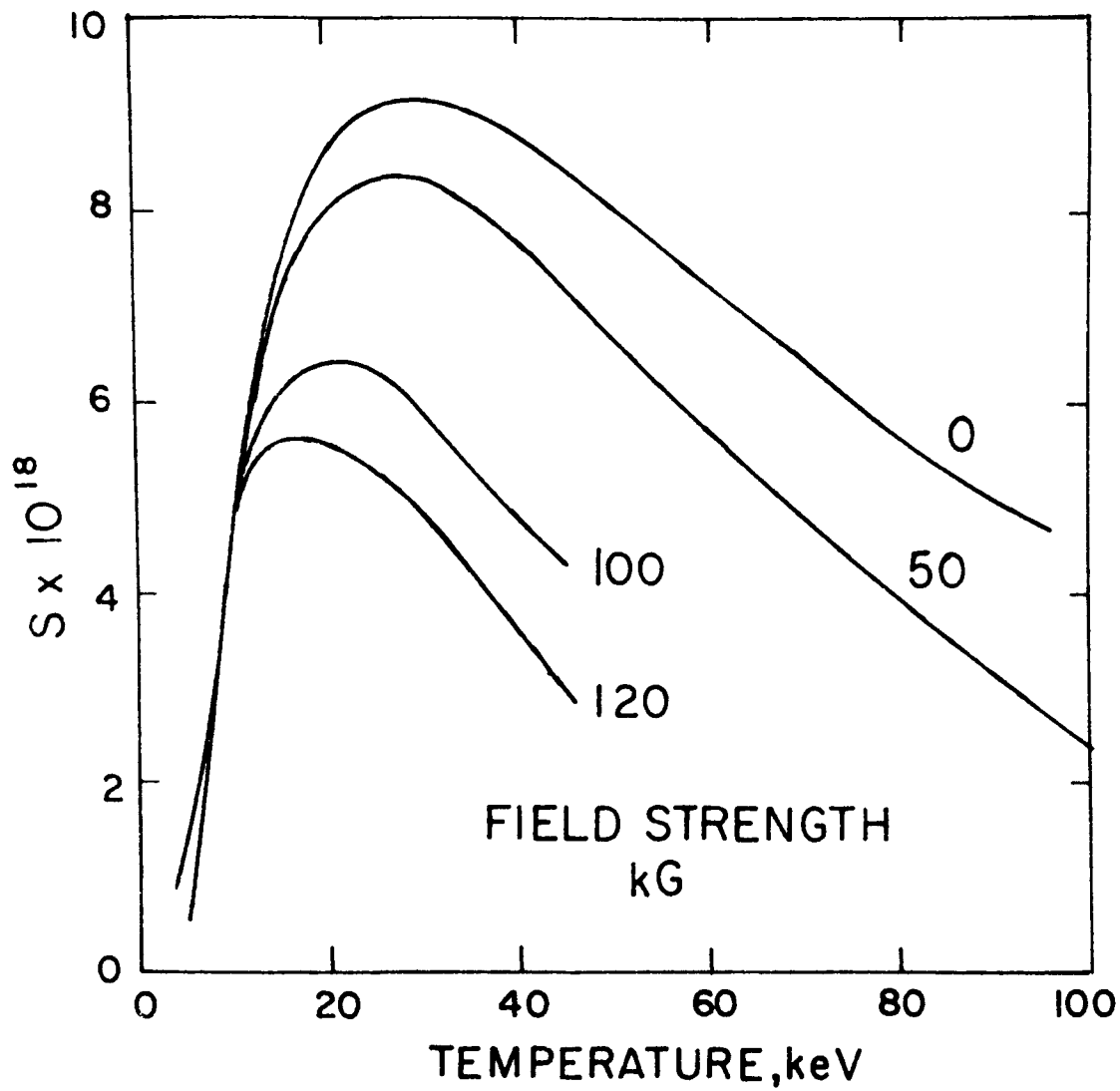


Fig. 13. The Stability Function.

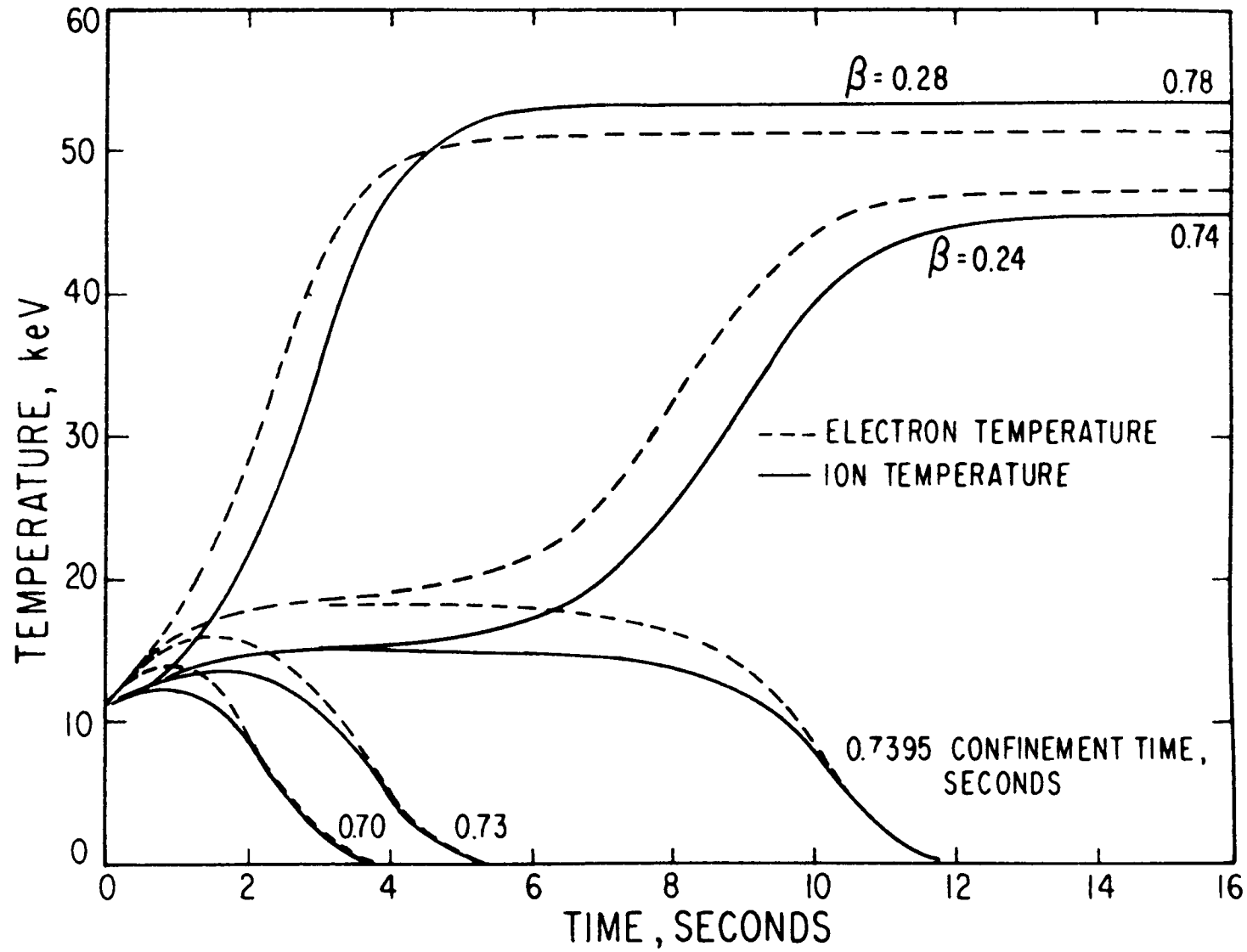


Fig. 14. Temperature Instability with Respect to the Fluctuation of a Fixed Confinement Time.



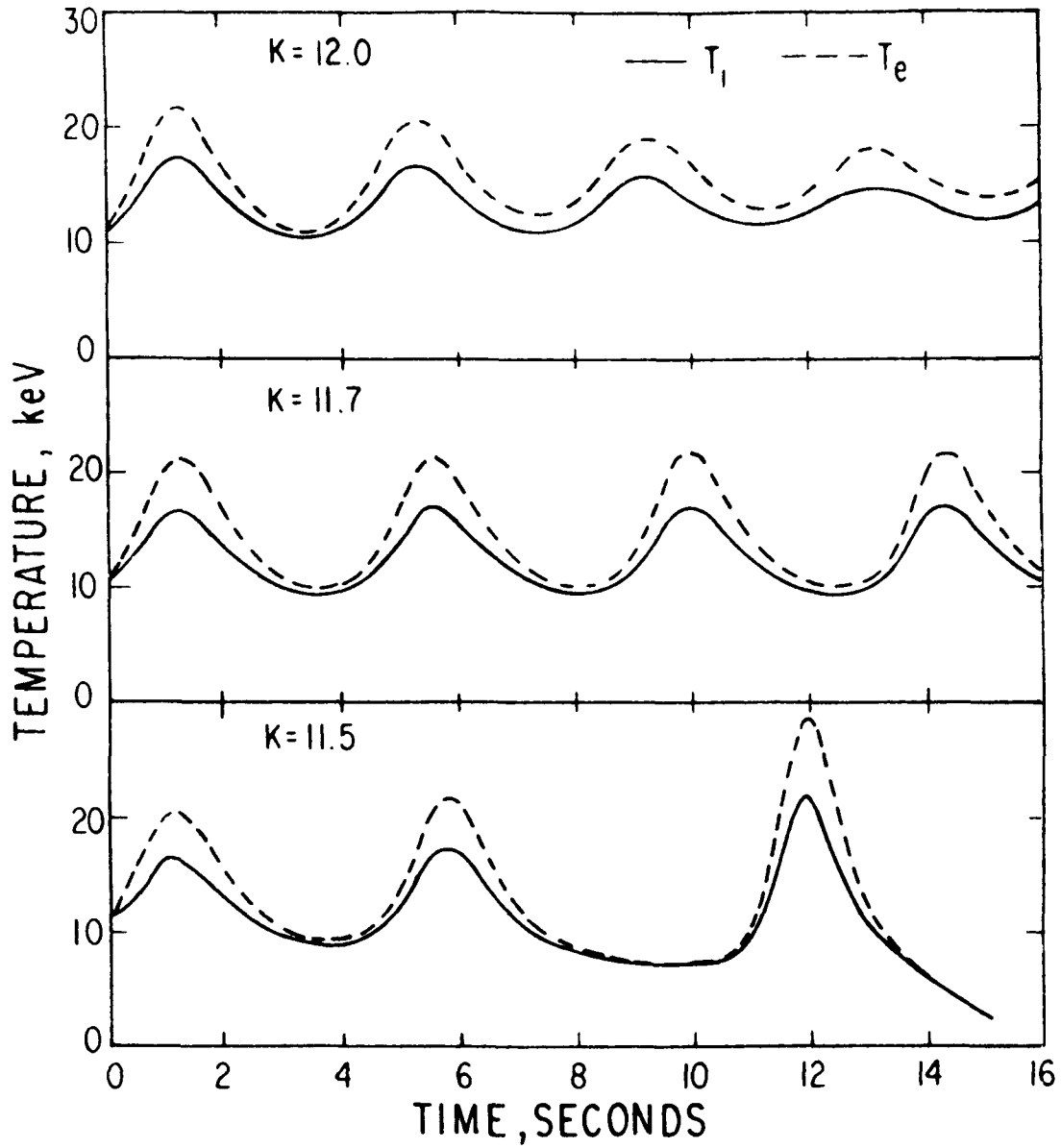


Fig. 15. Effect of "Bohm" Confinement Time,  $\tau = K/T_e$ .

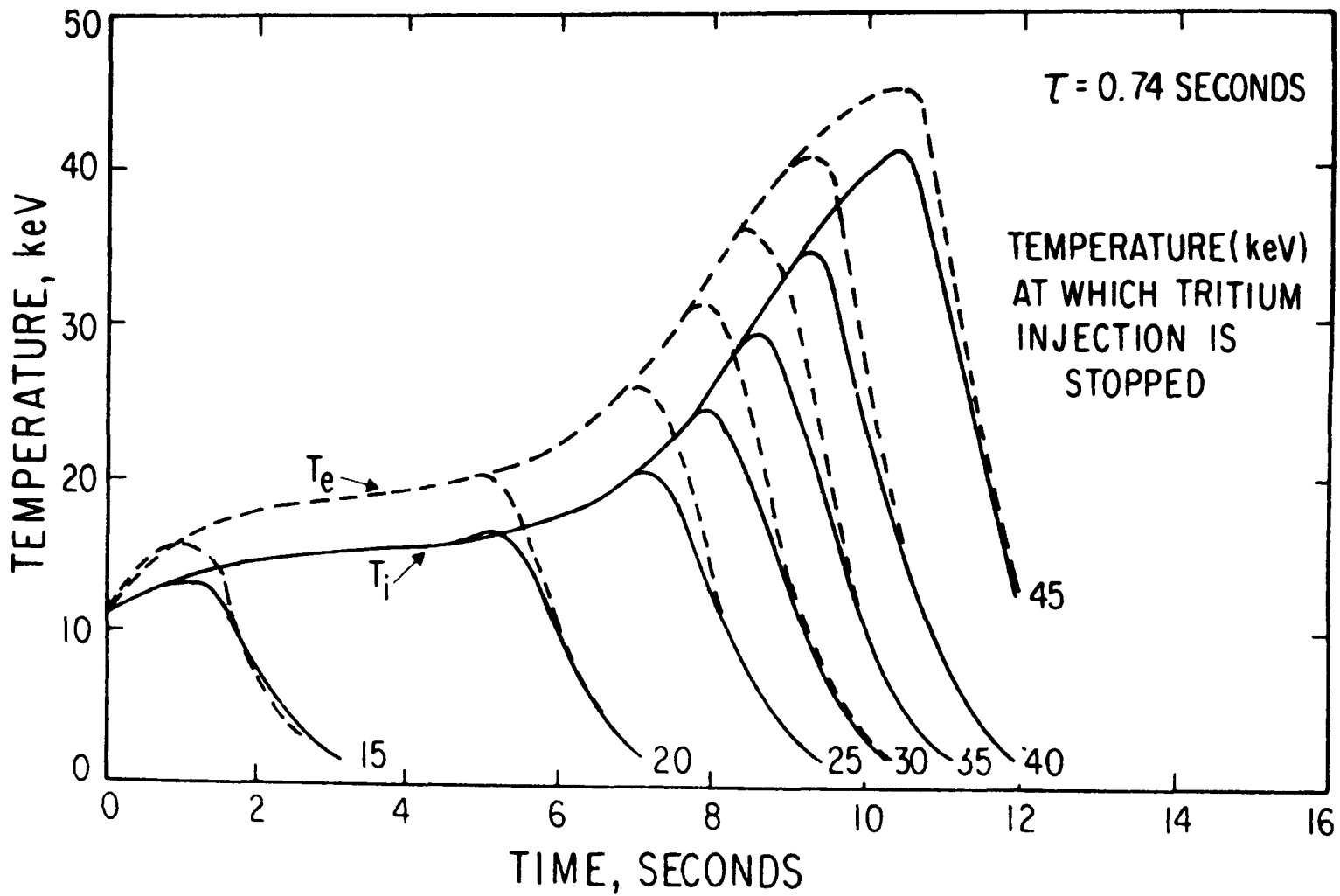


Fig. 16. Effect of Cutting Off Tritium Injection.

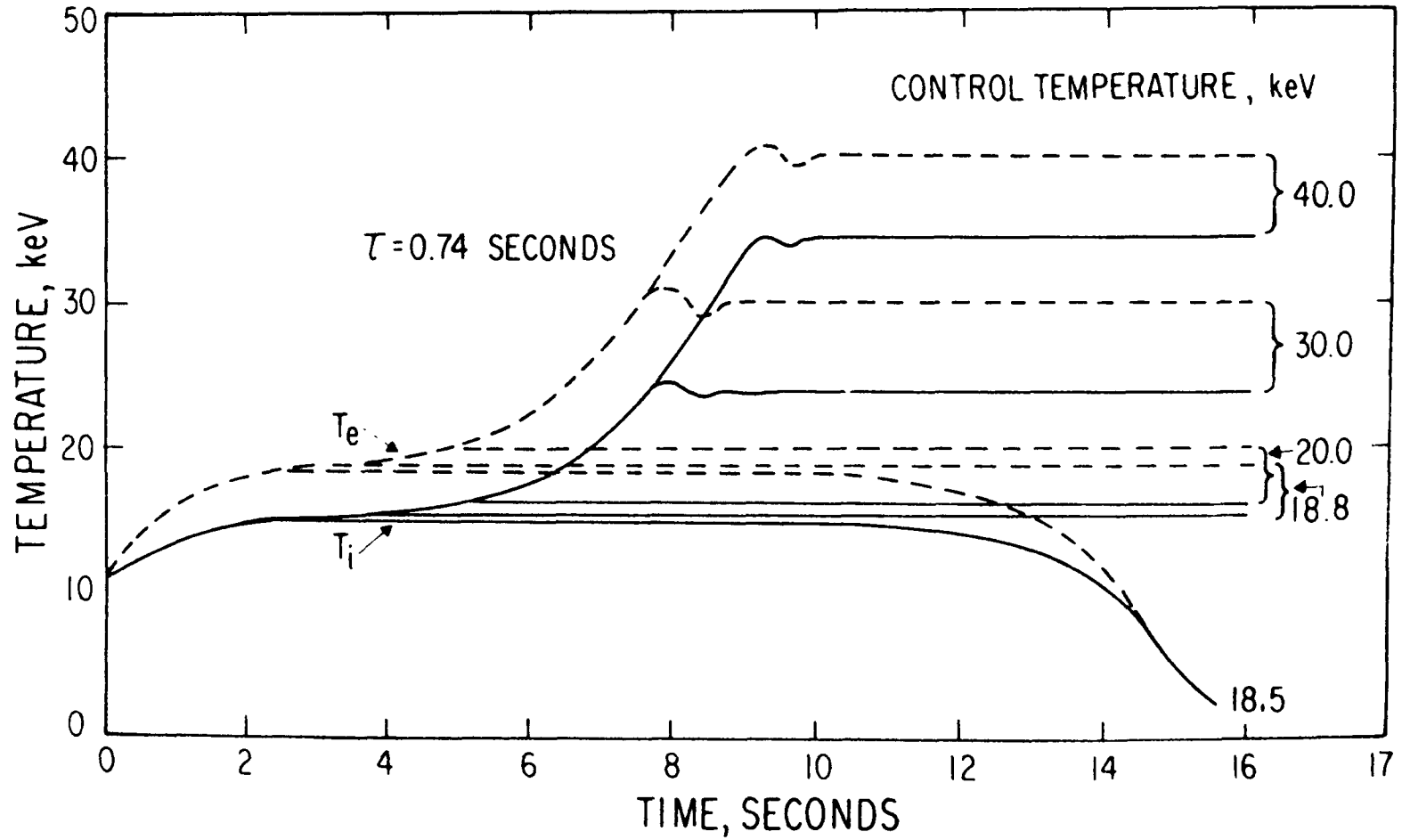


Fig. 17. Mixture Control.

### C. High Field Effects

Synchrotron radiation reduces the stable temperature point.

The effect on the stable operating point is shown in Fig. 18.

### D. Ignition Problems

#### 1. Ohmic Heating

As the temperature rises, bremsstrahlung increases, but the plasma resistivity falls. Consequently, with constant ohmic heating current density, the electron temperature saturates. The time,  $t$ , to reach a temperature,  $T$ , is given by:

$$t = \frac{AD}{2B} \left[ 2 \arctan\left(\frac{T^{1/2}}{D}\right) + \ln\left(\frac{D+T^{1/2}}{D-T^{1/2}}\right) \right] - \frac{2AT^{1/2}}{B} , \quad (20)$$

a solution of

$$A \frac{dT}{dt} + BT^{1/2} - CT^{-3/2} = 0 , \quad (21)$$

with  $D \equiv (C/B)^{1/4}$ .

The temperature approaches an asymptotic value,  $T_m$ , of  $D^2$ .

The time requirement to reach half the limit,  $t_{1/2}$ , is given by

$$t_{1/2} = 0.79AD/B .$$

$$T_m = (0.22 \times 10^{16}) j/n \quad \text{ev} .$$

$$t_{1/2} = (0.58 \times 10^{21}) j^{1/2} n^{-3/2} \quad \text{seconds} .$$

Figure 19 shows typical behavior.

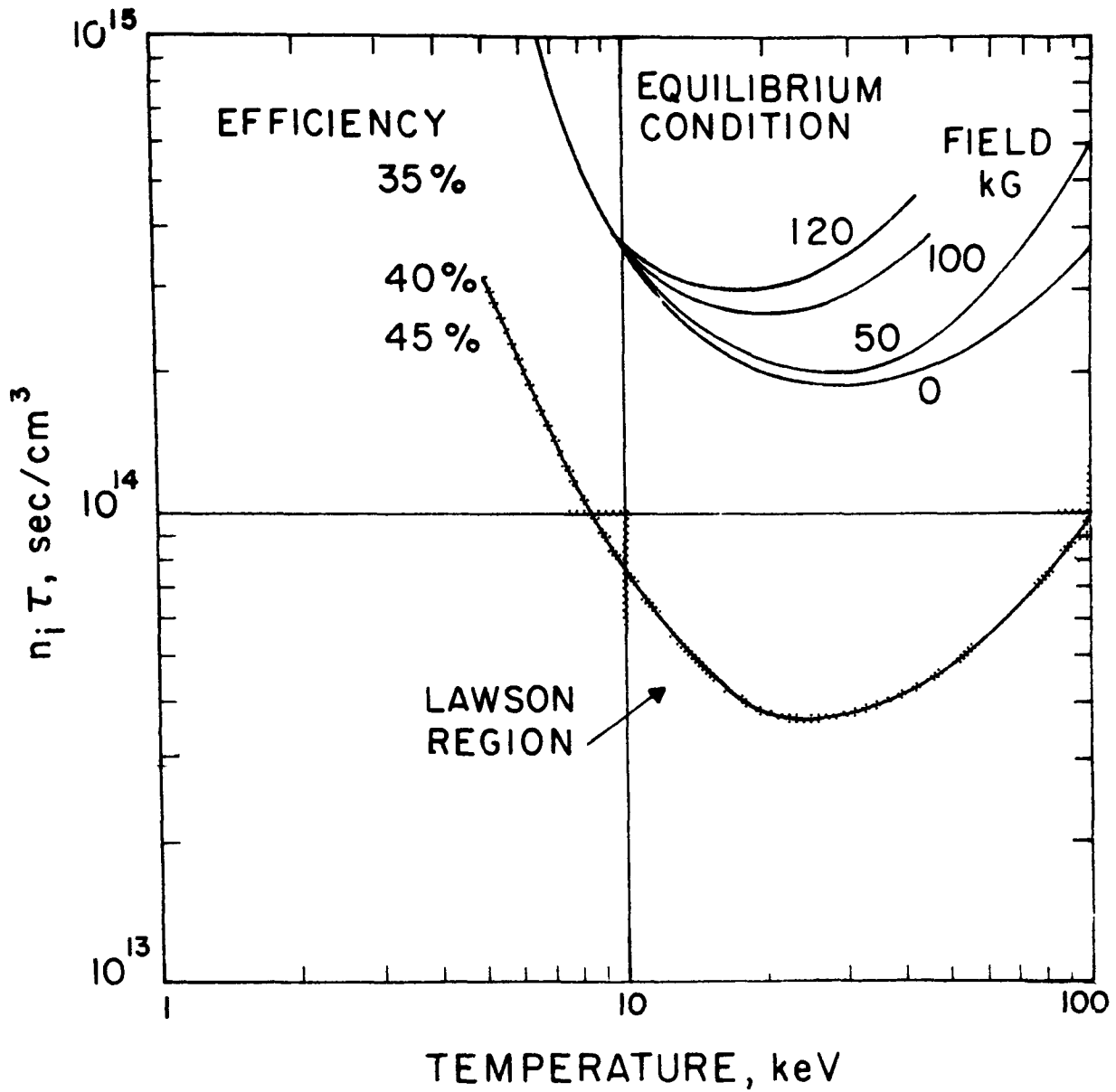


Fig. 18. Typical Effect of High Field on the Equilibrium Condition. Wall Resistivity =  $16 \times 10^{-4}$  ohm-cm. Plasma Radius = 100 cm. Impurity Level =  $10^{-5}$ . Radiation Leak = 0.05. Tritium Fraction = 0.5.

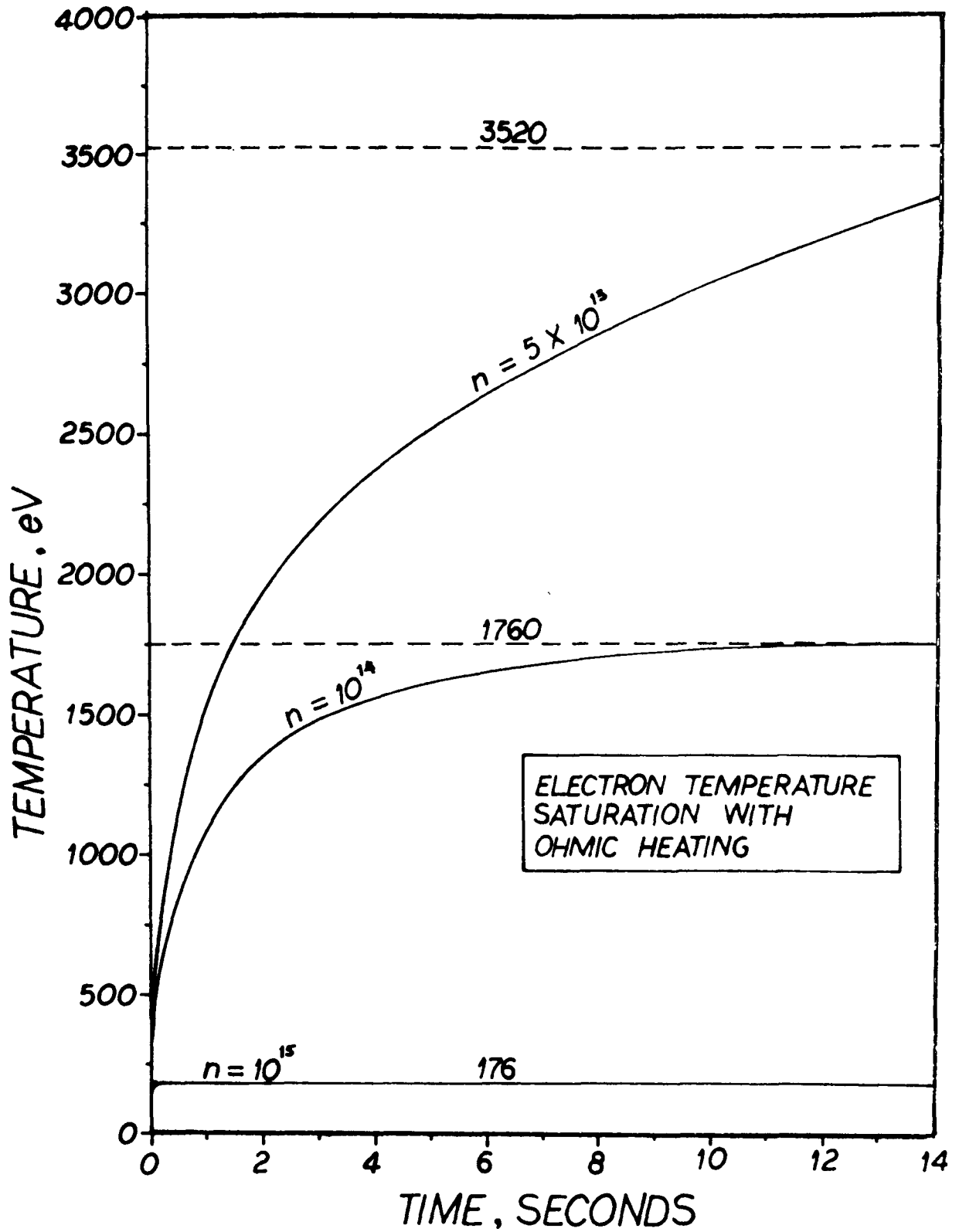


Fig. 19. Ohmic Heating.

## 2. Low Temperature Equilibrium

Figure 20 shows the equilibrium among 1) ohmic heating, 2) bremsstrahlung, and 3) alpha heating. The lower branch is stable. The upper branch is unstable and represents an ignition limit.

## 3. Intrinsic Slowness of Ignition by Ohmic Heating Alone

Figure 21 presents the number of seconds to reach 6 keV from various initial points by Tokamak ohmic heating ignition alone for a plasma of density  $1.0 \times 10^{14}$ .

### E. Compression

An analysis of Tokamak compression will be found in the literature.<sup>16</sup> Figure 22 shows its effectiveness in speeding ignition.

### F. Establishment of Equilibrium Regime

Figure 23 shows two examples of a complete ignition and injection sequence to establish an equilibrium burn. Compression takes place between 1 and 2 seconds, and injection of cold fuel begins at 4 seconds. Various injection rates lead to various equilibrium densities.

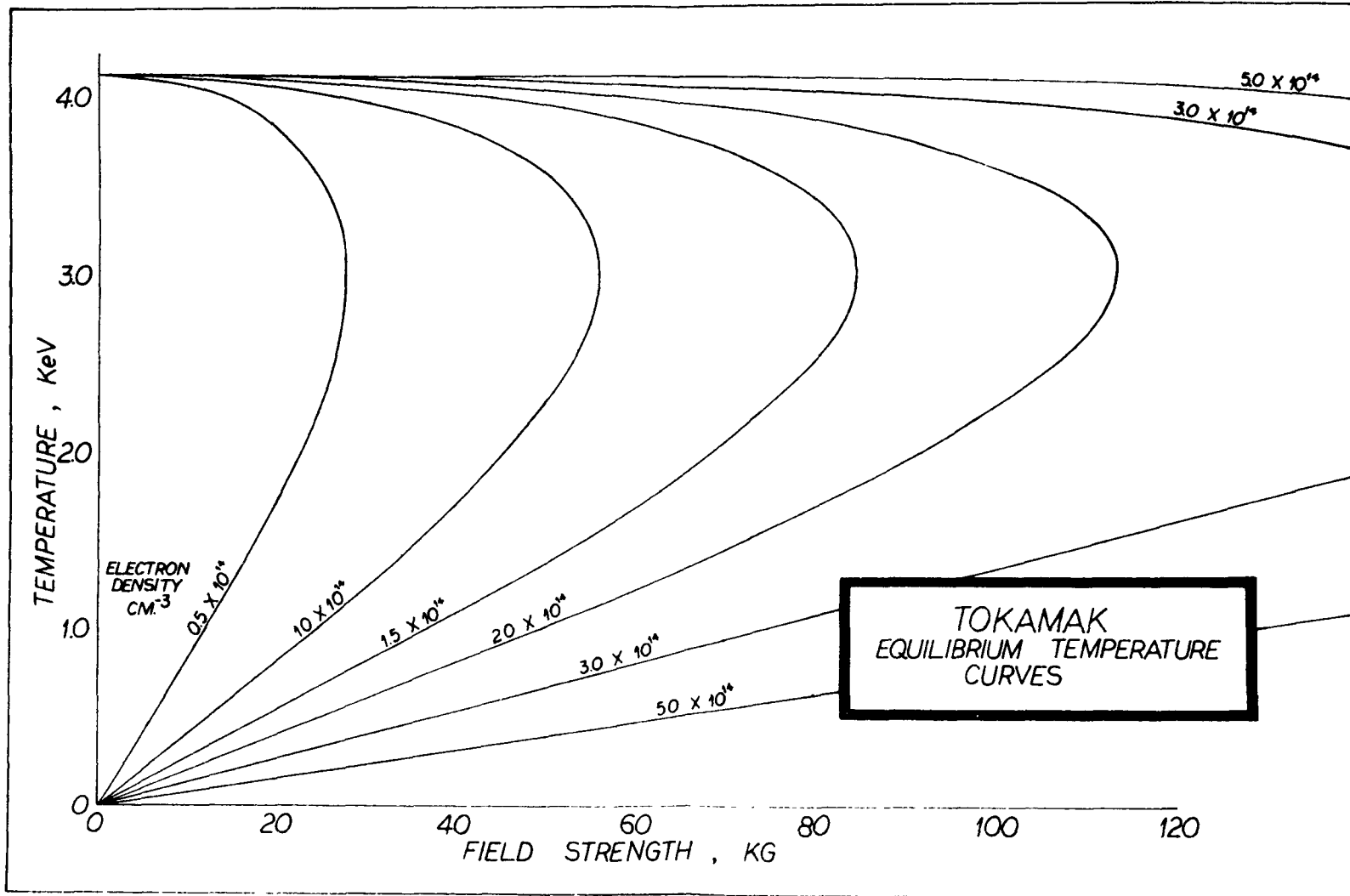


Fig. 20. Low Temperature Equilibrium of a Tokamak.  
A = 3, q = 3, Plasma Radius = 100 cm.



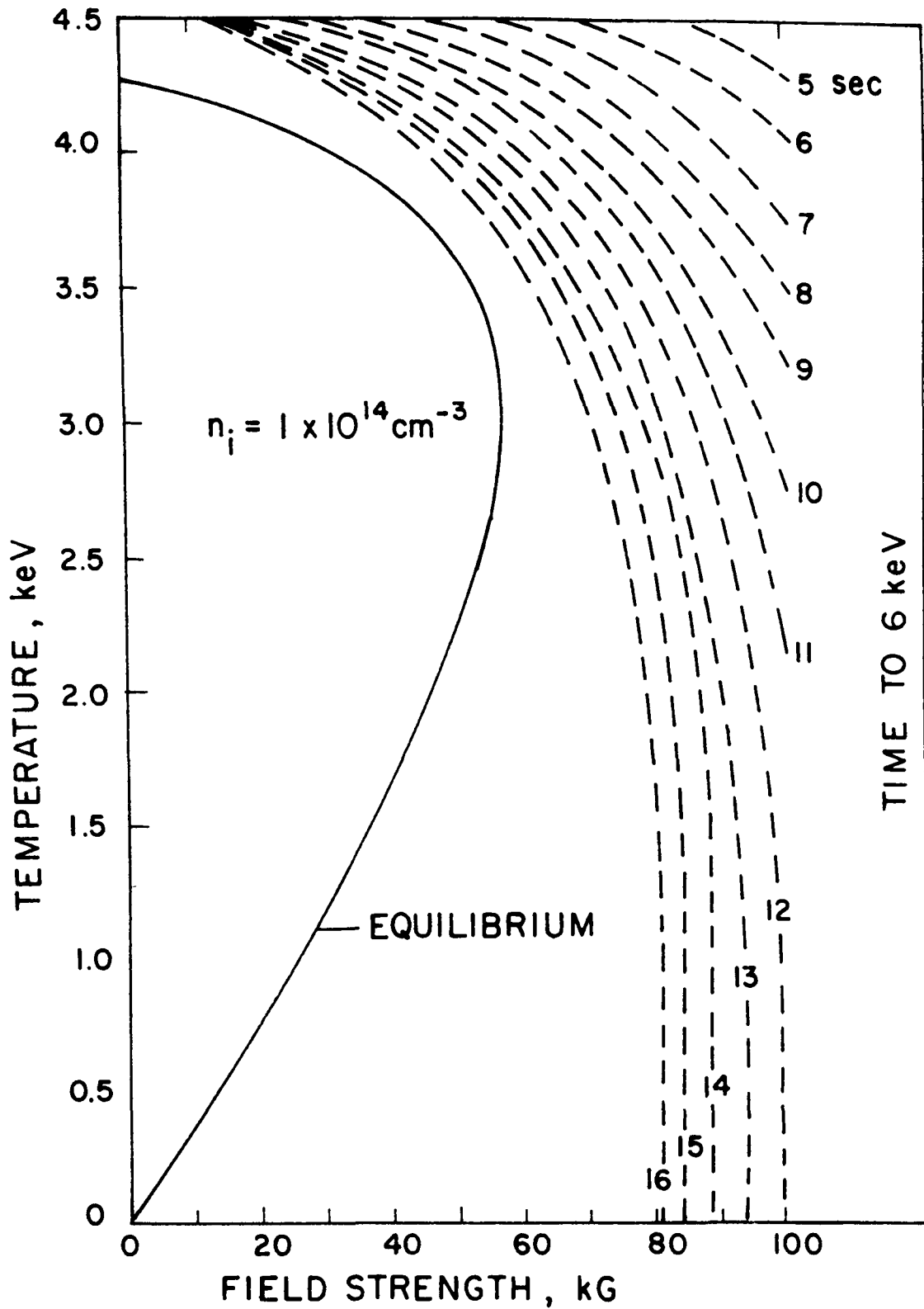


Fig. 21. Heating Time Required to Reach 6 keV.  $A = 3$ ,  $q = 3$ , Plasma Radius = 100 cm.

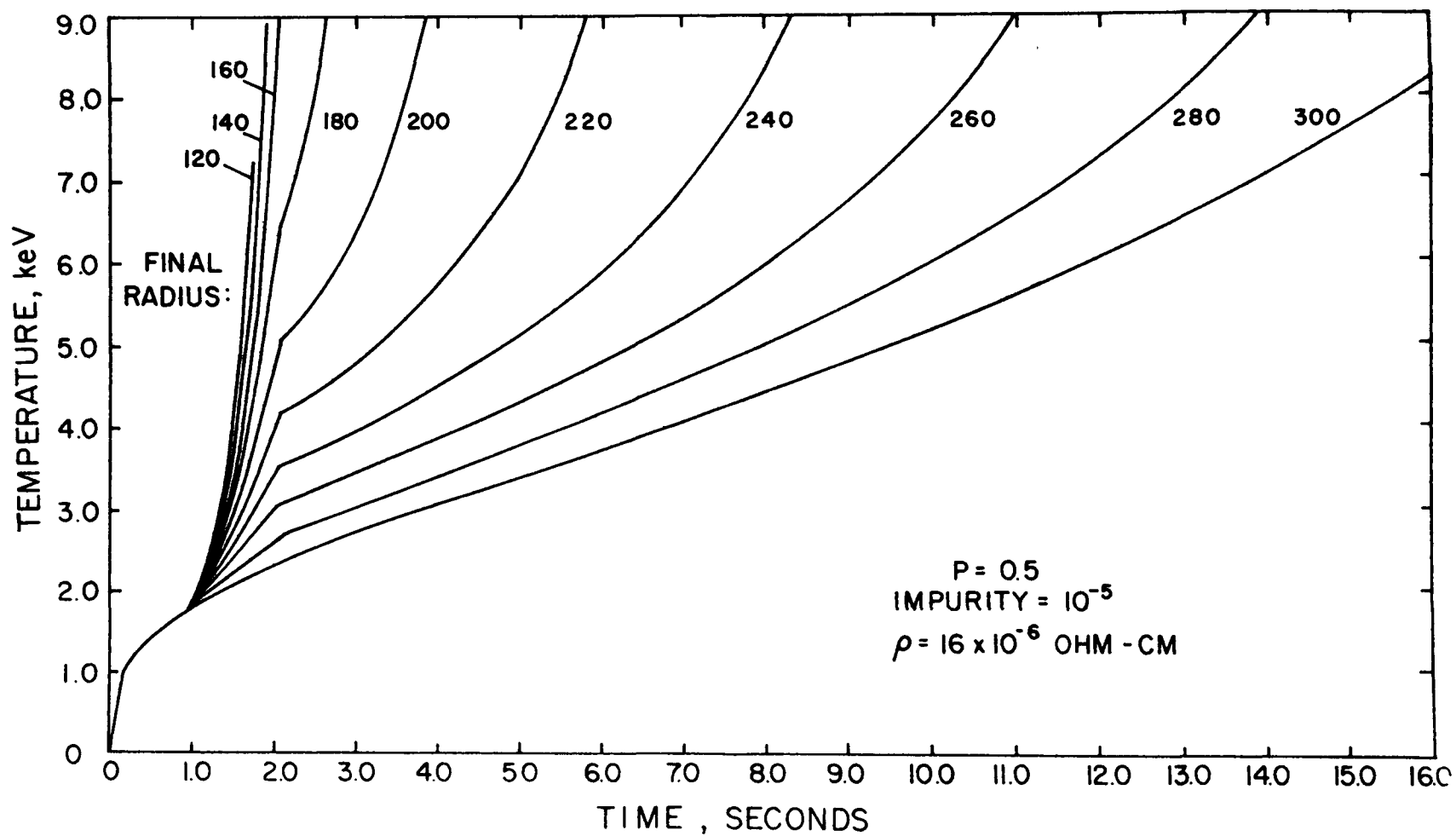


Fig. 22. Effect of Adiabatic Compression from an Initial Major Radius of 300 cm.

Density =  $1.0 \times 10^{14} \text{ cm}^{-3}$ . Radiation Leak = 0.05.

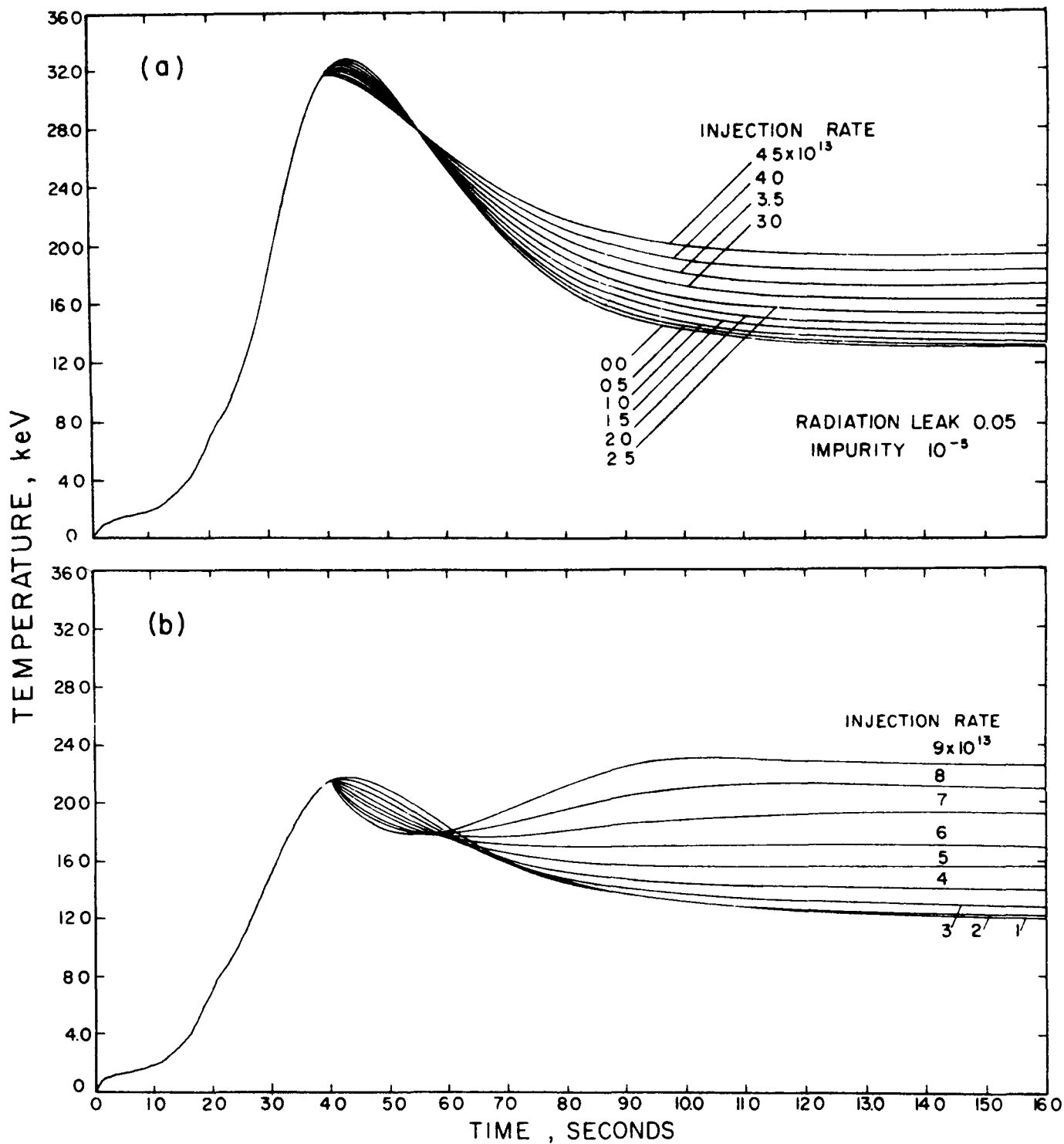


Fig. 23. Two Examples of Igniting a Reactor. Case (a) Is for a Confinement Time of 10 Seconds. Case (b), 5 Seconds. The Plasma Is Compressed from 300 cm Major Radius to 170 cm Major Radius during a One-Second Period between  $t = 1.0$  and  $t = 2.0$  seconds. Injection Begins at  $t = 4$  Seconds.

### III Bibliography

- <sup>1</sup>S. Glasstone and R.H. Lovberg, Controlled Thermonuclear Reactions, D. Van Nostrand Co., Princeton, N.J. (1960).
- <sup>2</sup>L. Spitzer, Jr., Physics of Fully Ionized Gases, Interscience Publishers, Inc., New York (1956).
- <sup>3</sup>S. T. Butler and M. J. Buckingham, Phys. Rev. 126, 1 (1962).
- <sup>4</sup>B. A. Trubnikov and V. S. Kudryavtsev, Proceedings Second International Conference on Peaceful Uses of Atomic Energy, Geneva (1958) vol. 31, p. 93.
- <sup>5</sup>R. G. Mills, Princeton Plasma Physics Laboratory MATT-658 (1969).
- <sup>6</sup>J. D. Lawson, Proc. Phys. Soc. (London) B 70, 6 (1957).
- <sup>7</sup>D. J. Rose, Nucl. Fusion 9, 183 (1969).
- <sup>8</sup>A. J. Impink, Jr., Technical Report No. 434, Massachusetts Institute of Technology (1965).
- <sup>9</sup>W. E. Homeyer, Technical Report No. 435, Massachusetts Institute of Technology (1965).
- <sup>10</sup>R. G. Mills, Nucl. Fusion 7, 223 (1967).
- <sup>11</sup>I. N. Golovin, Yu. N. Dnestrovsky, and D. P. Kostomarov, IAE-1903, Proceedings of the Nuclear Fusion Reactor Conference, Culham Laboratory, September 1969, to be published.

- <sup>12</sup>R. G. Mills, Princeton Plasma Physics Laboratory  
MATT-659 (1968).
- <sup>13</sup>R. Carruthers, P. A. Davenport, and J. T. D. Mitchell,  
Culham Laboratory, CLM-R85 (1967).
- <sup>14</sup>D. J. Rose, Oak Ridge National Laboratory, ORNL-TM-2204  
(1968).
- <sup>15</sup>R. G. Mills, Proceedings of the Symposium on Engineering  
Problems of Fusion Research, Los Alamos Scientific Laboratory,  
LA-4250 (1969) p. B1.
- <sup>16</sup>R. G. Mills, Proceedings of the 5th Intersociety Energy  
Conversion Engineering Conference, Las Vegas, Nevada (1970)  
to be published.
- <sup>17</sup>B. Coppi and E. Mazzucato, Princeton Plasma Physics  
Laboratory MATT-720 (1969).
- <sup>18</sup>R. G. Mills, Princeton Plasma Physics Laboratory  
MATT-728 (1970).
- <sup>19</sup>D. J. Rose and M. Clark, Jr., Plasma and Controlled  
Fusion, the M. I. T. Press and John Wiley and Sons, Inc., New  
York (1961).
- <sup>20</sup>L. Spitzer et al., NYO-6047 (PM-S-14), Atomic Energy  
Commission, Washington, D. C. (1954).

<sup>21</sup>R. F. Post, University of California, UCRL-6077 (1960).

<sup>22</sup>F. L. Ribe, T. A. Oliphant, Jr., and W. E. Quinn, Los Alamos Scientific Laboratory, LA-3294-MS (1965).

Turbulent Heating in Mirror Machines at

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ABSTRACT

By "turbulent heating" we mean that the appropriate direct current power applied to the plasma is converted by the plasma to radiofrequency oscillations that result in intense heating of the plasma. At Oak Ridge National Laboratory, we have been producing plasma in steady-state experiments in mirror machines either with hot electrons or with hot ions. In the hot-electron case, electron temperatures of  $10^5$  eV, and  $10^{11}$  electrons/cm<sup>3</sup> have been produced; in the ion case, ion temperatures of  $10^3$  eV and  $10^{13}$  ions/cm<sup>3</sup> are now being studied. In both cases, the heating efficiency (dc power in, related to hot-confined plasma leaking out) ranges from a few up to 10%.

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Recent studies have concentrated experimentally and theoretically on the mechanisms of the heating processes. We find two classes of instabilities present and competing for the applied power: plasma electron oscillations, driven by currents along the magnetic field, and resulting in hot electrons; and plasma ion oscillations, driven by currents across the magnetic field, and resulting in hot ions. (The electrons in the latter case are "frozen" to magnetic field lines and participate in the oscillations in only a "subdued" fashion.)

Recent theoretical and experimental work has produced the following results. First, in both the electron and the ion heating cases, the character of the "turbulence-excited" radiofrequencies is not such as to produce rapid, gross losses of plasma; second, density and temperature limitations on plasma heating observed in the past are now understood and possibly can be overcome; third, the details of the heating mechanisms are becoming sufficiently well understood that we think it is possible to apply these mechanisms to larger, different, and possibly more interesting magnetic field configurations, including systems of "closed" geometry.



We first survey the experimental results of past experiments in turbulent heating in mirror machines at the Oak Ridge National Laboratory. We restrict ourselves to discussing the Oak Ridge experiments and calculations because they lead directly to an understanding of the heating mechanisms. Since our theory has grown smoothly from our experiments, we have not made an exhaustive survey of the literature concerning turbulence calculations. Thus, we may have overlooked previous pertinent work by others.

The first set of experiments involved injecting an electron beam through a hollow, gas-filled anode, into a magnetic mirror, as shown in Fig. 1. Experimentally, we found that the plasma electrons were heated to a temperature of  $10^5$  eV, a density of  $10^{11}$   $\text{cm}^{-3}$ , had a lifetime on the order of  $10^{-1}$  sec, and were heated with an efficiency of a few percent.<sup>1,2</sup>

Since the plasma was hot and reasonably dense, we tried to use this hot-electron plasma as a blanket that would intercept and ionize gas atoms flowing into the hot ion plasma confined in a mirror machine. Experimentally, the blanket was formed by placing the beam-plasma interaction region off-axis in a large mirror machine. Electrons heated by the interaction were trapped between the mirrors, and precessed azimuthally in the radial magnetic field gradient, forming a shell or "blanket". This blanket<sup>3</sup> is shown in Fig. 2. Unfortunately, the empirical studies were not capable of forming plasma blankets of sufficient density ( $n_e > 10^{14}$   $\text{cm}^{-3}$ ) to accomplish shielding against gas influx, and the project was terminated.

A study of the radiofrequency emission from the hot-electron plasmas, using special high-frequency probes and correlators,<sup>4</sup> has led to an understanding of the heating mechanism<sup>5</sup> and of the density limitation. The

correlators suggest that noise, present in the plasma gun, excites oscillations that grow at  $\omega_{pe}$  and are convected along the electron beam. The parameters of the operating system were empirically so chosen that: (1) the amplitude of the rf is now computed to reach a maximum in the midplane of the mirror machine; and (2)  $\omega_{pe} \approx \omega_{ce}$ , so fringing-field effects couple the rf plasma oscillations to cyclotron oscillations. Thus the electrons are heated to high temperatures by electron-cyclotron resonance.

The density limitation on the heating of electrons comes from the fact that the e-folding distance  $x_0$  for the rf to grow is given approximately by  $x_0 = v_0/\omega_{pe}$ , where  $v_0$  is the velocity of the incident electron beam, and  $\omega_{pe}$  is the electron plasma frequency. Since  $\omega_{pe}$  is proportional to the square root of the electron density, a large increase in the electron density results in the growth distance becoming so short that the maximum rf amplitude is generated not in the mirror machine, but back in the electron gun.

To maintain electron heating at higher electron densities in systems of finite size, we must increase the growth e-folding distance. One possibility of accomplishing this is by using relativistic electron beams. The growth distance for plasma oscillations increases as  $\gamma^{3/2}$  for longitudinal oscillations,<sup>6</sup> and  $\gamma^{1/2}$  for transverse oscillations,<sup>7</sup> where

$$\gamma = \left(1 - \frac{v^2}{c^2}\right)^{-1/2},$$

v is the velocity of the electron beam, and c is the

velocity of light. Another technique might involve the placing of an electron gun closer to the desired plasma-heating region. In any case, since we now understand the limitation in heating plasma electrons at high density, we can begin to consider ways of overcoming this limitation.

One interesting observation is that the high-frequency rf fields in the plasma probably do not produce radial plasma losses. Our probe studies demonstrate that the plasma itself confines the rf to the central region around the beam. Thus the rf does not extend to the plasma surface to cause plasma losses there.

In the case of ion heating, our results have been rather encouraging. By placing a hollow, gas-filled anode between the coils of a magnetic-mirror machine,<sup>8</sup> we have obtained ion temperatures of a few keV at a density of about  $10^{13}$  ions/cm<sup>3</sup>.<sup>9,10</sup> The heating efficiency is high — on the order of 5 to 10%. The temperature and density are sufficient for a weak thermonuclear reaction to be observed. Unfortunately, the hot-ion lifetime is limited by charge exchange on neutral gas atoms to about 100  $\mu$ sec. The apparatus is shown schematically in Fig. 3.

In order to apply this turbulent heating process to higher plasma densities, volumes, and systems of better confinement, it is necessary to understand the heating mechanism. Also, we wish to understand some strange properties of this ion-heating device, such as why we observe hot ions in the device ( $T_i \approx 1$  keV) when at the same time the electrons are cold.

As a result of an integrated diagnostic study, coupled with a theoretical study, we now think that we understand the heating process for the ions. We suspect that in the ion-heating case, the heating occurs by means of the electron flow across the magnetic field, in contrast to the electron heating case where the heating occurs by the electron flow along the magnetic field.

Auto- and cross-correlation studies show that the plasma has a filamentary turbulence pattern, with potential "rods" extending along the mag-

netic field fluctuating randomly with respect to each other at high amplitudes with frequencies near  $\omega_{pi}$ . Since this leads to electric fields which are perpendicular to the magnetic field, high electron temperatures are not required to maintain high electric fields. Moreover, since the fluctuating fields are perpendicular to the magnetic field,<sup>9</sup> they are ideal for heating ions without accelerating them into the loss cone of the magnetic mirrors. In fact, due to the rapid growth rate of the heating instability (see Table 1), ions scattered into the loss cone are probably accelerated back out of the loss cone before being lost, leading to longer trapping.

The next step in this research was to determine how this instability near  $\omega_{pi}$  is excited. The critical step occurred when one of the authors (I.A.) attended the Fusion Torch Study Group in Washington, D. C.,<sup>11</sup> where a paper presented by H. K. Forsen described the "centrifugal splitting" that can occur when particles precess azimuthally in a radial electric field and an axial magnetic field. In cylindrical coordinates, particles of different mass precess with different azimuthal velocities due to the fact that centrifugal force is mass-dependent. In particular, in a cylindrical system, electrons and ions precess with sufficiently different velocities that a variety of "two-stream" azimuthal heating modes, described below, can occur. These instabilities all require a large azimuthal precessional velocity, i.e., a high radial dc electric field. We have attempted to confirm the presence of such fields and have obtained three pieces of experimental data that suggest they are, indeed, present:

1. X-Ray Data. We have used commercial xenon-filled proportional counters to examine the soft x-ray photon spectrum<sup>12</sup> emitted the plasma in the range of 2-10 keV (from the counter's sensitivity threshold to the energy corresponding to the applied potential). We find that in the plasma, the electrons from the initial electron beam are both spread in energy (which is expected from beam-plasma interactions) and degraded in energy to about half the value acquired in falling through the applied potential. This degradation in energy, with no electrons present having energies near that corresponding to the applied potential, is most easily understood if the core of the plasma is 5 kV negative with respect to the wall of the apparatus. Such a negative core would produce a radial  $\vec{E}$ -field of about 10 kV/cm in our finite geometry plasma.
2. Spectral Data. Using a recently discovered technique<sup>13</sup> of using normally forbidden satellite spectral lines from helium that is added to the plasma, we have measured electric field strength, electric field polarization, and electric field frequency in the plasma.<sup>14</sup> We find that the electric field has a peak value of about 8 kV/cm, is perpendicular to the magnetic field, and has a frequency  $\omega$  that lies in the range  $0 \leq \omega \leq \omega_{pi}$  or, in other words,  $\omega \ll \omega_{pe}, \omega_{ce}$ .
3. Precessional Velocity Measurements. By placing two probes at different azimuthal spacings, and by filtering out the intense rf near  $\omega_{pi}$ , we have followed the azimuthal propagation of small density fluctuations in the plasma. We find that they move in

the direction corresponding to the  $\vec{E} \times \vec{B}$  drift, and have a velocity corresponding to a radial  $\vec{E}$  field on the order of 10 kV/cm.

Table 1 summarizes our present theoretical knowledge of the various two-stream azimuthal instabilities which can result when a strong dc radial  $\vec{E}$  field such as described above exists in the plasma. Note that in all cases the computed maximum e-folding growth time is short compared to the observed ion charge-exchange lifetime of 100  $\mu$ sec. For all cases it has been assumed that  $\omega_{pe} < \omega_{ce}$ .

All the instabilities heat ions preferentially, except #2 which can heat electrons as well as ions under the proper conditions of wavelength parallel to the magnetic field relative to wavelength perpendicular to the magnetic field. All the instabilities in these high E-field modes tend to occur at short wavelengths, which tends primarily to heat the ions rather than to move them across the magnetic field. Number 5 does not require a relative drift of ions and electrons.

The instability that we suspect most strongly as causing the observed heating in our device is #1, since it has both the most rapid growth rate, and a predicted rf emission spectrum that agrees with experiment.<sup>17</sup> The physical nature of this instability is shown schematically in Fig. 4. We assume that although the ions precess azimuthally on the average, on the time scale discussed here they are essentially unmagnetized ( $\omega \sim \omega_{pi} \gg \omega_{ci}$ ). The electrons, on the other hand, are magnetized ( $\omega \ll \omega_{pe} < \omega_{ce}$ ) and precess with a velocity  $\vec{v} = \frac{c \vec{E} \times \vec{B}}{B^2}$ . The electrons are precessing more rapidly than the ions. In addition, we assume that the plasma density decreases rapidly with increasing radius (experimentally,  $\Delta n/n \sim 1$  over the radius of the plasma).

Table 1  
Theoretically Predicted Instabilities

Instability Requires	Growth Time (max)	Azimuthal Wavelength for max growth <sup>†</sup>
1. Radial density gradient. <sup>15-18</sup>	$10^{-9}$ sec	$k_r \approx \frac{\omega \pi}{\Delta u}$
2. Electrons move axially, <sup>16,19</sup> ions azimuthally.	$10^{-7}$ sec <sup>*</sup>	**
3. Radial density gradient caused by applied $\vec{E}$ field.	$10^{-7}$ sec	$k_r \approx \frac{\omega \pi}{\Delta u}$
4. Collisions between electrons <sup>20</sup> and ions.	$10^{-7}$ sec <sup>‡</sup>	$k_r \approx \frac{\omega \pi}{\Delta u}$
5. Shear in rotating ion cloud <sup>21</sup>	$10^{-7}$ sec <sup>‡</sup>	$k_r \approx \frac{\omega \pi}{\mu_i}$

<sup>†</sup>  $\Delta u$  is relative drift of ions and electrons;  
 $\mu_i$  is the ion drift velocity.

<sup>\*</sup> Computer solution for one case.

<sup>\*\*</sup> Not studied.

<sup>‡</sup> These are the growth times predicted by linear theory<sup>20,21</sup>;  
quasi-linear theory (not yet published) predicts growth times  
 $\sim 100$  times shorter.

Assume that an azimuthal potential fluctuation is produced from random noise, as is shown. We now demonstrate that this azimuthal potential fluctuation will grow. Ions, being unmagnetized, approach a positive potential fluctuation and slow down, thereby increasing the local positive charge. Electrons being magnetized, precess from the outer, more tenuous electron cloud to the region of the local positive charge and replace the initial more dense electron cloud, further enhancing the local positive region. Similarly, it can be shown that a negative potential fluctuation will grow so that both electron and ion perturbations provide positive feedback that causes rapid growth of radial plasma "spokes".

The strong charge fluctuations result in rapid ion heating. Strong phase mixing is present and is easily observed in a computer simulation of the response of the plasma column to a suddenly-applied potential step.

On the basis of the turbulence model above, we can compute the steady-state, or "saturated" turbulent electric field present in the plasma column.<sup>18</sup> We find that the calculated turbulent electric field yields a calculated flow of radial electron current that is in excellent agreement with the value observed experimentally in our plasma device.<sup>22</sup>

If we wish to ignite a large volume of plasma by our turbulence process, some modifications are necessary. The present fundamental limitation of the heating process is that it works best in columns of small radial extent, both because a small radial distance allows us to obtain high radial electric fields with modest applied potentials, and a small radius allows us to obtain large centrifugal effects for moderate azimuthal velocities. Therefore, to fill mirror machines having a large radial dimension by this turbulence process requires us to use many small columns in parallel, as shown



in Fig. 5. We can either have injected many separate plasma columns to utilize the density-gradient-dependent turbulence process (No. 1), or we can use one or more of the more slowly growing turbulence processes that do not require the gradient. In the latter cases, since a uniform plasma density suffices, we can simply fill the machine with cold gas or plasma and ignite it with an appropriate voltage pulse.

Applications of this system to heating plasma in toroidal configurations seem to be obvious. Rings placed around the plasma column, as used by Kawabe and Eubank at Princeton,<sup>23</sup> place a high potential drop along the surface of the plasma column, as is shown schematically in Fig. 6. As the Princeton workers have clearly demonstrated, turbulence near  $\omega_{pe}$  greatly increases the surface resistivity of the plasma, and the current, in consequence, diffuses inward to fill the plasma column. Since the current must flow across the magnetic field in the vicinity of the rings, our ion-heating turbulence process can occur there. Variations of various electromagnetic coupling processes may also help to couple in energy with the appropriate currents perpendicular to the magnetic field.

In conclusion, we now feel that we understand both the hot-electron and hot-ion turbulent heating processes in mirror machines sufficiently well to be able to deduce meaningful scaling laws and to make extrapolations to systems of larger volume, density, and temperature. In addition, we can discuss plasma heating in systems of closed geometry. The turbulent heating is simple, efficient, and need not lead directly to severe plasma losses.

## REFERENCES

1. I. Alexeff, R. V. Neidigh, W. F. Peed, E. D. Shipley, and E. G. Harris, Phys. Rev. Letters 10, 273 (1963).
2. I. Alexeff, R. V. Neidigh, and W. F. Peed, Phys. Rev. 136, A689 (1964).
3. W. L. Stirling, R. V. Neidigh, I. Alexeff, and V. J. Meece, J. Nucl. Energy, Part C 8, 157 (1966).
4. I. Alexeff, R. V. Neidigh, and W. R. Wing., Int. J. Engng. Sci. 7, 531 (1969).
5. I. Alexeff, G. E. Guest, D. Montgomery, R. V. Neidigh, and D. J. Rose, Phys. Rev. Letters 21, 344 (1968).
6. S. J. Buchsbaum and L. C. Hebel, also O. Eldridge, private communication.
7. M. Rosenbluth and F. Winterberg, private communication.
8. R. V. Neidigh, I. Alexeff, G. E. Guest, W. D. Jones, D. C. Montgomery, D. J. Rose, and W. L. Stirling, Plasma Physics and Controlled Nuclear Fusion Research (International Atomic Energy Agency, Vienna, 1969), Vol. 2, p. 693.
9. R. V. Neidigh, I. Alexeff, G. E. Guest, F. R. Scott, and J. R. McNally, Jr Bull. Am. Phys. Soc. 14, 1040 (1969).
10. I. Alexeff, W. D. Jones, and R. V. Neidigh, Phys. Rev. Letters 18, 1109 (1967).
11. Fusion Torch Meeting, Jan. 26, 1970, U. S. Atomic Energy Commission, Washington, D. C.
12. W. L. Stirling, I. Alexeff, R. V. Neidigh, and K. G. Estabrook, Bull. Am. Phys. Soc. 14, 1040 (1969).
13. M. Baranger and B. Mozer, Phys. Rev. 123, 25 (1961). See also, W. S. Cooper and H. Ringler, Phys. Rev. 179, 226 (1969); H. J. Kunze and

- H. R. Griems, Phys. Rev. Letters 21, 1048 (1968).
14. F. R. Scott, J. R. McNally, Jr., and R. V. Neidigh, Bull. Am. Phys. Soc. 14, 1040 (1969).
  15. I. Alexeff, "Advances in Understanding Turbulent Heating--V", Plasma Studies Technical Memorandum No. 61, March 10, 1970, ORNL.
  16. A. B. Mikhailovskii and V. S. Tsypin, [ZhETF Pis'ma 3, 247 (1966)], JETP Letters 3, 158 (1966).
  17. M. M. Widner, "Evaluation of the Dispersion Relation Given by Mikhailovskii and Tsypin", Plasma Studies Technical Memorandum No. 69, April 24, 1970. See also, M. M. Widner, "Turbulent Ion Heating in Burnout V", ORNL-TM-3036, June 1970, ORNL.
  18. A. Hirose, "Current Driven Instability in a High Temperature ( $T_i \gg T_e$ ) Plasma", Plasma Studies Technical Memorandum No. 70, May 7, 1970, ORNL.
  19. W. L. Stirling, "A High Frequency Electrostatic Instability in a Magnetically Guided Plasma Stream (D.E.T.P. Ashby and A. Paton) and Its Application to Burnout V", Plasma Studies Technical Memorandum No. 62, April 20, 1970, ORNL.
  20. A. Hirose and I. Alexeff, "High Frequency Dissipative Instability", Plasma Studies Technical Memorandum No. 72, May 14, 1970, ORNL.
  21. A. Hirose and I. Alexeff, "Velocity Shear Driven Instability", Plasma Studies Technical Memorandum No. 75, May 19, 1970, ORNL.
  22. A. Hirose, "Current Flow Across Magnetic Field in Burnout V", Plasma Studies Technical Memorandum No. 71, May 7, 1970, ORNL.
  23. Private communication.

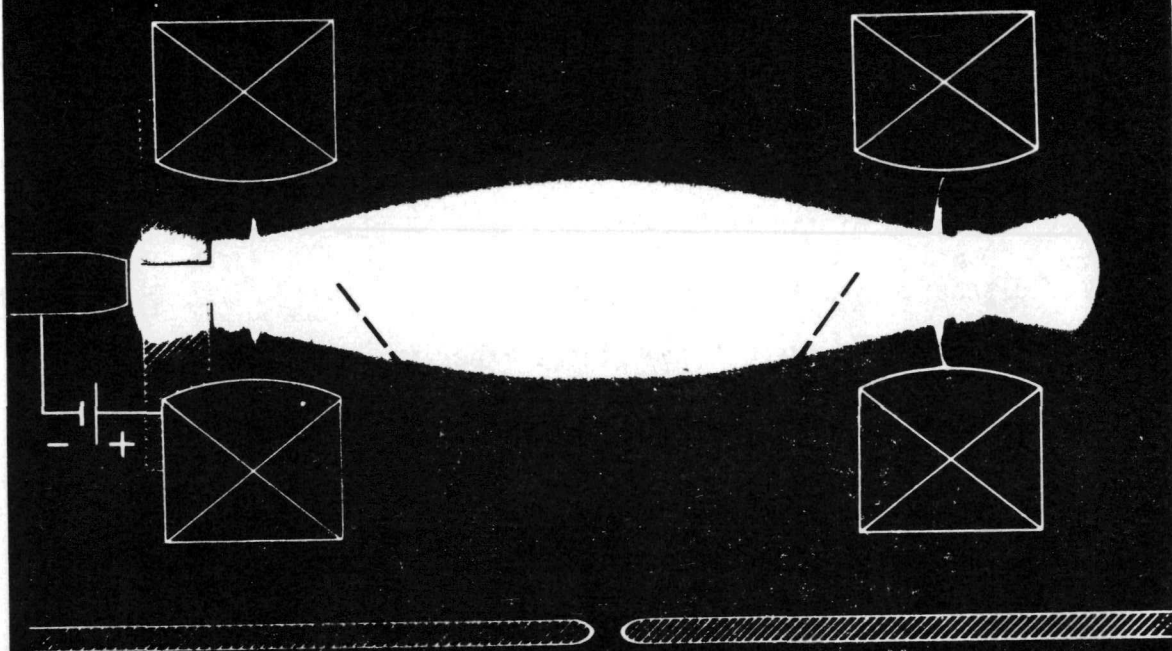
Availability of Plasma Studies Technical Memoranda:

These memoranda were given internal distribution only. A copy can be obtained by writing: Dr. Igor Alexeff, Oak Ridge National Laboratory, P.O. Box Y, Oak Ridge, Tenn. 37830.

## FIGURE CAPTIONS

- Fig. 1. Photos of hot-electron plasma taken with visible light (top) and with bremsstrahlung x-rays (bottom), with explanatory overlay showing schematic of apparatus. The intense x-ray spots at the ends of the x-ray image are due to hot electrons that escape from the plasma and emit intense x-rays during bombardment of the metal surfaces of the sides of the magnet coils.
- Fig. 2. Schematic of the hot electron blanket experiment. The reflex discharge was about three meters long.
- Fig. 3. Schematic of the hot ion plasma experiment, Burnout V.
- Fig. 4. Schematic of the proposed ion-heating instability in Burnout V. An azimuthal charge perturbation takes electrons from the dense core, superimposing these electrons on the negative density perturbation, thus reinforcing it. Similarly, ion and electron motion are such as to reinforce positive density fluctuations.
- Fig. 5. Concept of using a multi-electrode structure to turbulently heat the ions in a large-volume plasma device by means of many cylindrical, turbulent plasma cells.
- Fig. 6. Schematic of the turbulent-heating mechanism now used at the Princeton Plasma Laboratory to couple electrical energy radially into a plasma column (which could, of course, be toroidal).

**VISIBLE PLASMA**

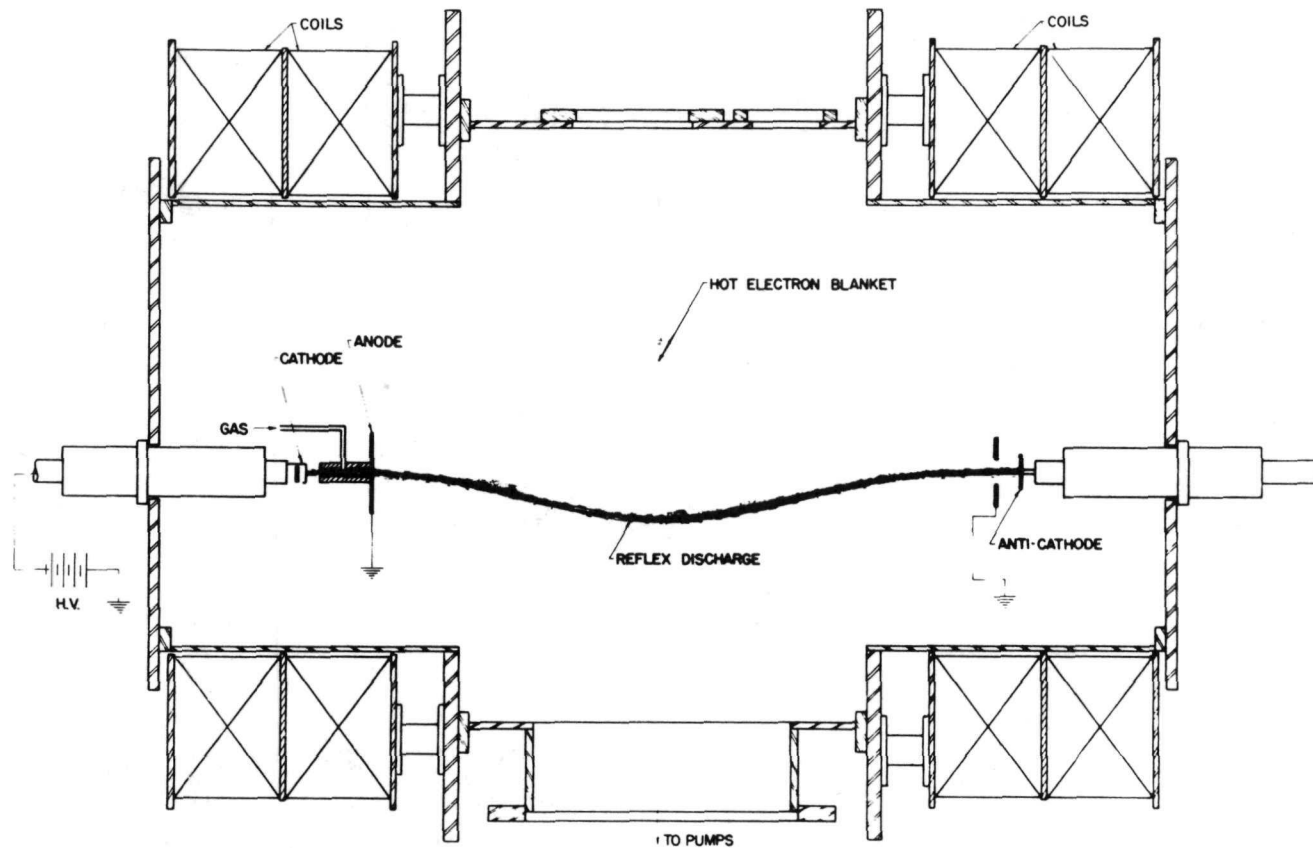


**LEAD**



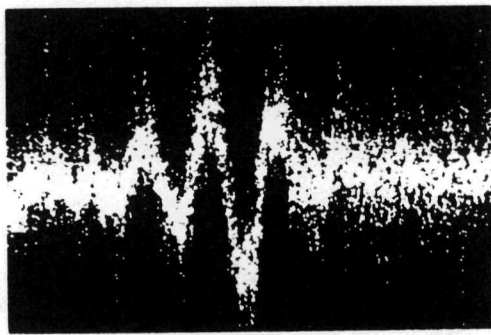
**X-RAY IMAGE**

UNCLASSIFIED  
ORNL DWG. 65-5528



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FIG. 2 BEAM PLASMA FACILITY  
AXIAL CROSS SECTION

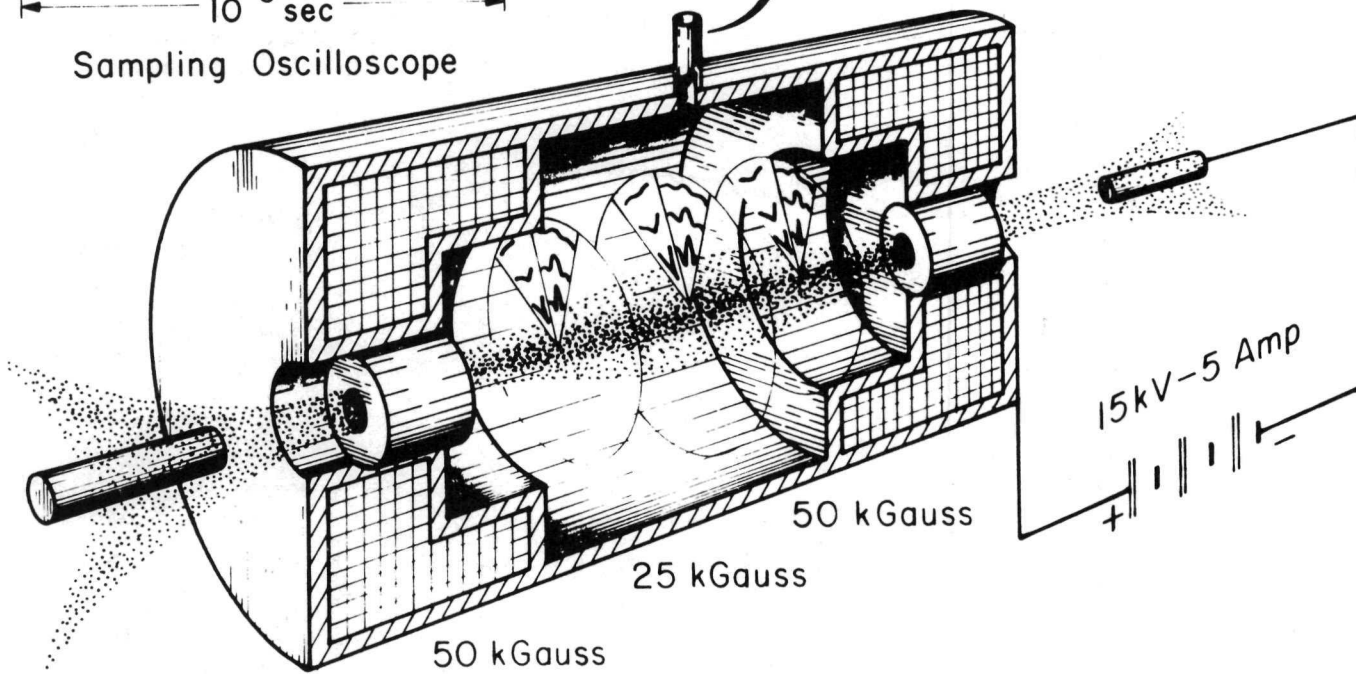


PROBE SIGNALS

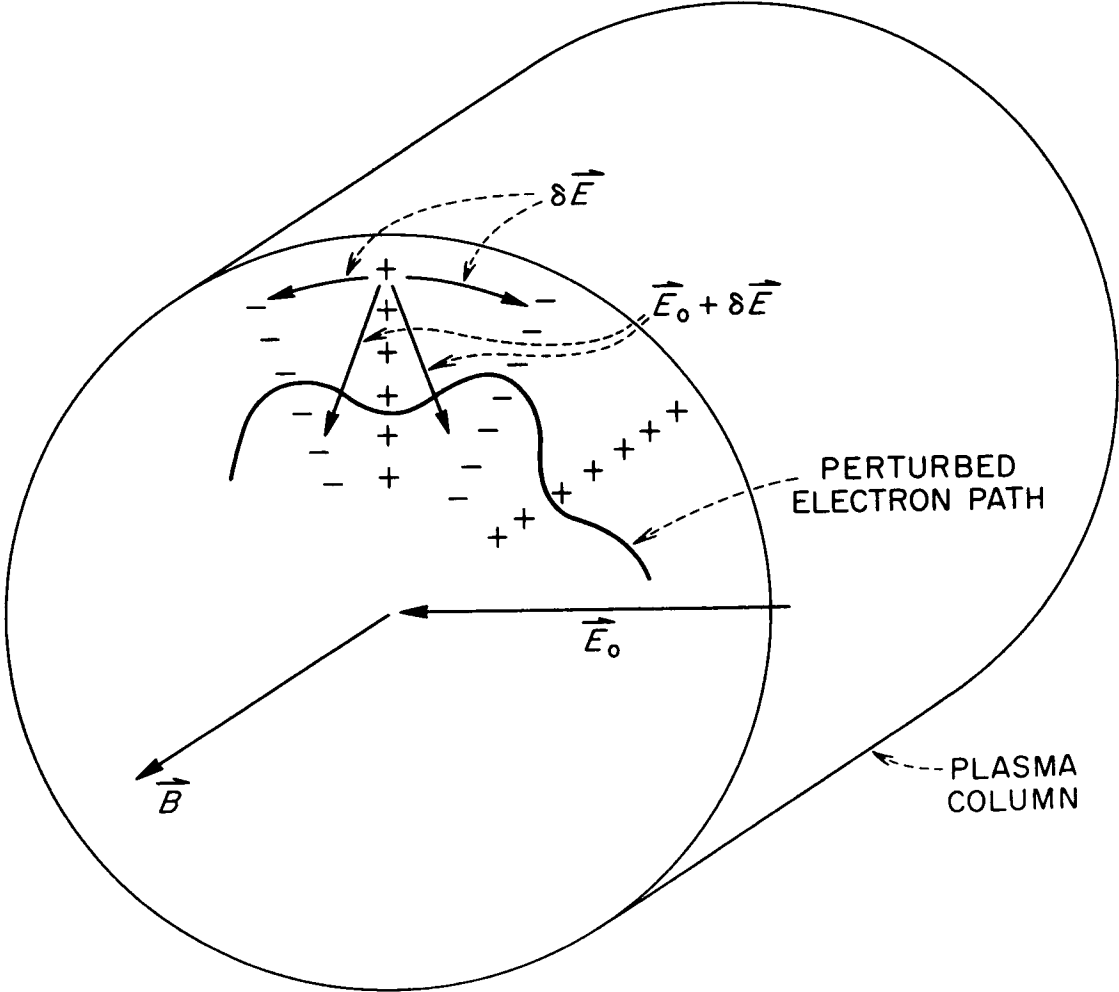


0 1000 2000 mega-hertz  
Spectrum Analyser

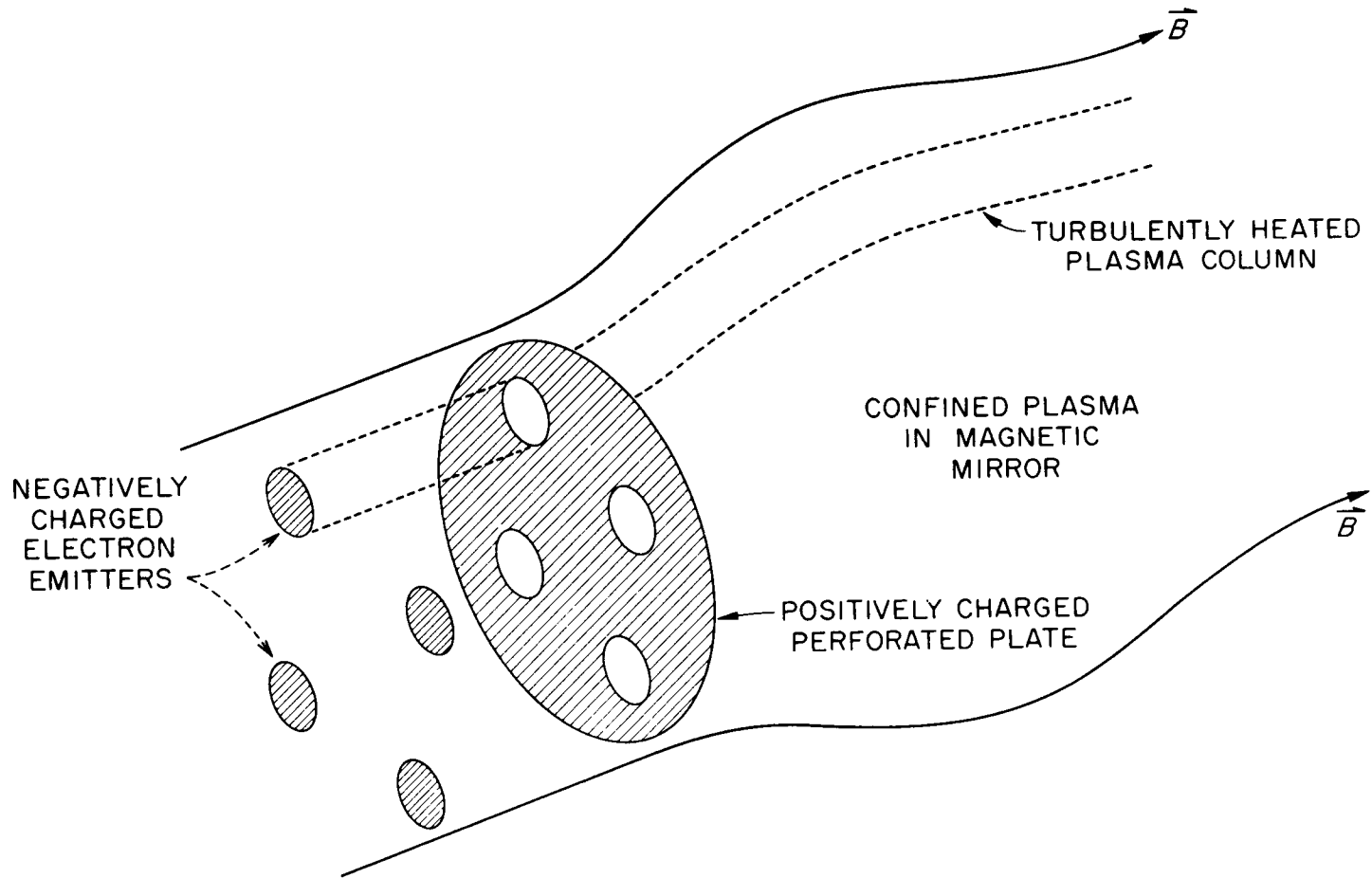
Deuterium input



Burnout  $\nabla$

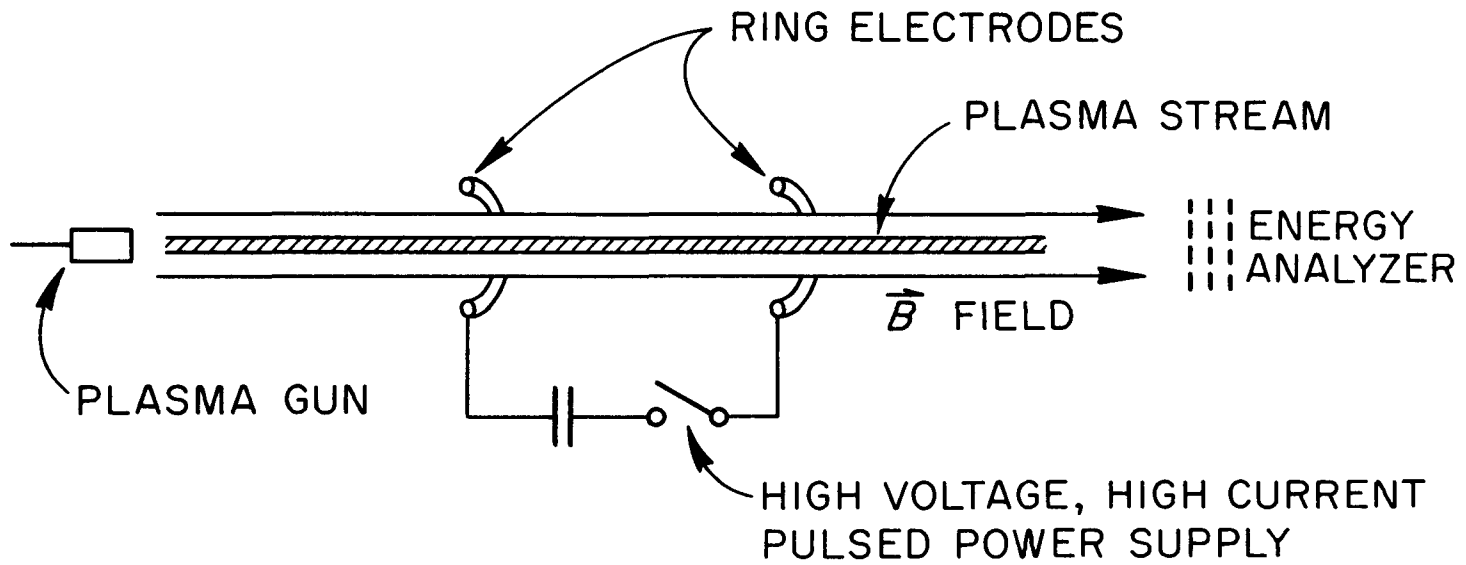






ORNL-DWG 70-7560

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SOME NEUTRONIC ASPECTS OF A DT FUSION REACTOR\*

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ABSTRACT

Fusion reactors based on the Deuterium-Tritium (DT) fusion reaction must produce tritium to be a viable power source. A survey of papers on the subject shows that attractive tritium production looks achievable. Some other neutronic considerations of DT fusion power are also discussed.

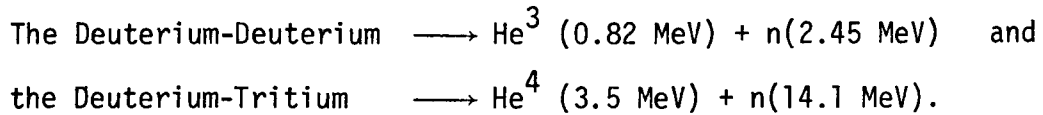
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\* This work was performed under the auspices of the United States Atomic Energy Commission

This paper was prepared for presentation at the Symposium on Fusion Reactor Design, June 2-5, 1970, at Texas Tech University, Lubbock, Texas

## INTRODUCTION

Two of the four fusion reactions being considered for fusion reactors produce energetic neutrons. These reactions are:



Hereafter, these reactions will be referred to as the DD and DT reactions. By virtue of its high cross section at low temperature as well as its high energy release, the DT fusion reaction is considered most likely to be used in early fusion reactors. Based on this, the DT reaction and its resulting 14.1 MeV neutron is the starting point for this discussion.

To make use of the DT fusion neutron, which carries 80% of the fusion reaction energy, the fusing DT plasma must be surrounded by a blanket. This blanket has two functions. First, it must intercept the 14.1 MeV neutrons and convert as much of their kinetic and potential energy to thermal energy as practical. Second, it must breed tritium to fuel the DT fusion reactions.

Unlike deuterium (D) which is a stable hydrogen isotope with a 0.015% abundance, tritium is an unstable, 12.6 year half life hydrogen isotope not found in nature. Tritium breeding is therefore the most crucial requirement the blanket must meet for if sufficient tritium cannot breed a fusion reactor based on the DT reaction would not be feasible.

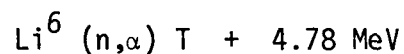
Fortunately, blanket calculations with lithium as the fertile material have shown tritium breeding in excess of that required is possible. The bulk of this presentation will deal with tritium breeding and energy generation in DT reactor blankets. Some of the radiological aspects such as tritium inventory, direct neutron and gamma radiation and contamination of a DT fusion reactor will also be discussed.

## TRITIUM BREEDING AND ENERGY GENERATION

As explained in the introduction, fusion reactors utilizing the attractive DT fusion reaction must have the means to breed tritium. The only conceivable means of producing tritium in the amounts needed is by neutron transmutation. The neutron source being the DT fusion reaction. They also must have the means to convert to heat the kinetic and as much of the potential energy of the 14.1 MeV DT neutron as practical. To accomplish this, the DT fusion source must be surrounded by a neutron intercepting blanket.

For a DT fusion system to be viable, the blanket must be able to breed at least enough tritium to make up for tritium consumed by fusion plus losses such as tritium's 12.6y half life  $\beta$  decay to helium 3. This means for each 14.1 MeV DT fusion neutron produced, one plus tritons must be bred in the blanket.

Lithium appears to be the only fertile element with any likelihood of producing required tritium from interactions with neutron. Natural lithium is composed of 2 stable isotopes,  $\text{Li}^6$  (7.42%) and  $\text{Li}^7$  (92.58%). Both isotopes undergo neutron reaction resulting in tritium. Lithium-6 undergoes an exothermic reaction of the form:



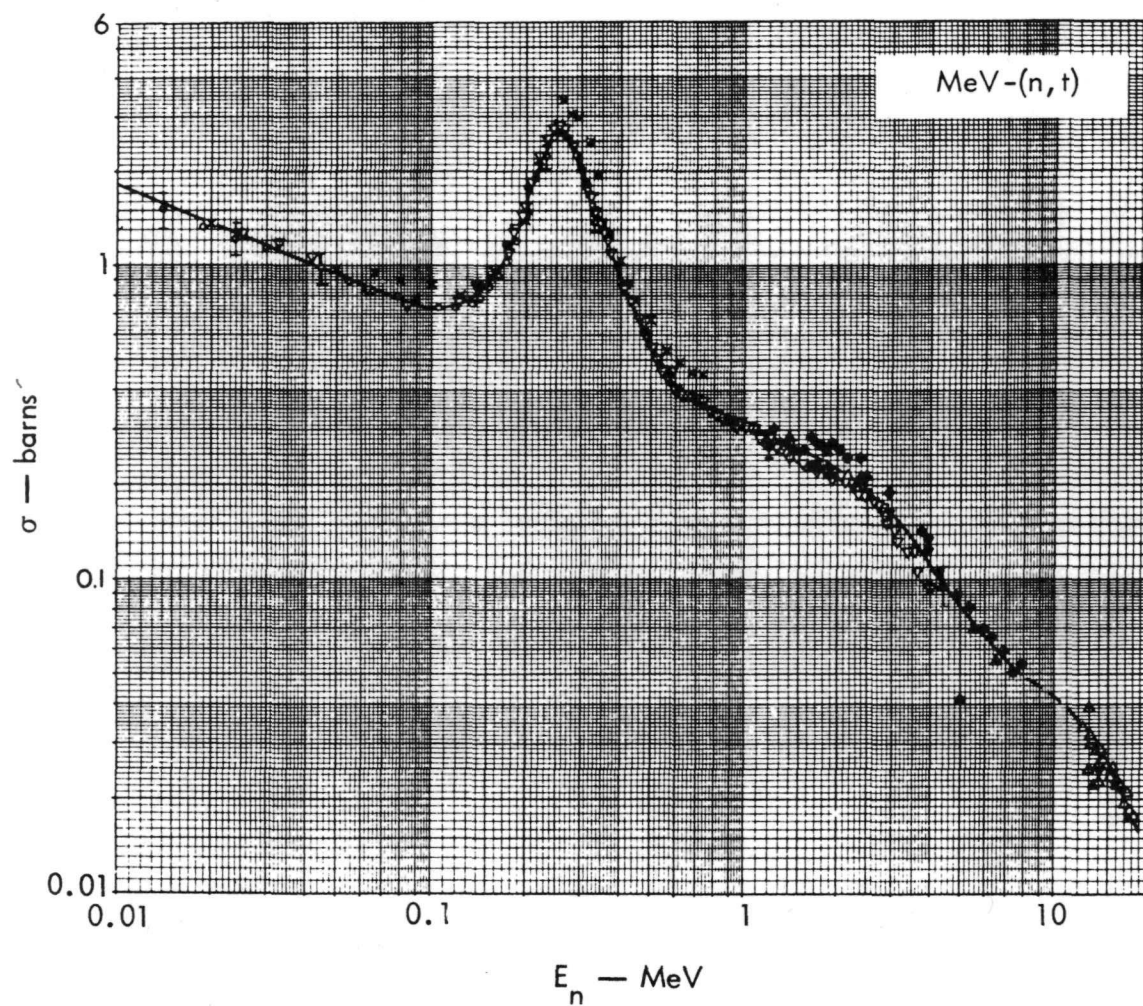


Fig. 1.  $\text{Li}^6(n,T)\text{He}^4$  cross section vs. energy.

Figure 1 shows the  $\text{Li}^6(n,\alpha)\text{T}$  cross section<sup>1</sup> for neutrons between .01 MeV to 11 MeV. The Lithium-6 thermal cross section is about 950 barns.

Lithium-7 undergoes an endothermic reaction of the form:

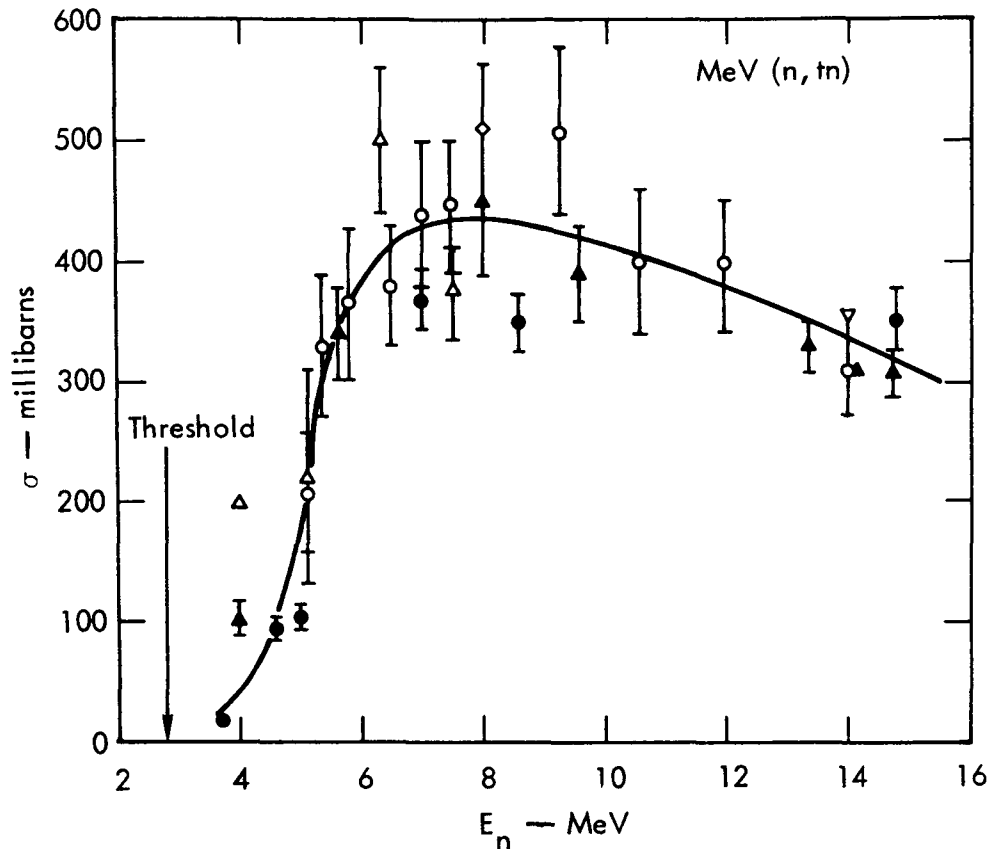


Fig. 2.  $\text{Li}^7 (n, tn') \text{He}^4$  cross section vs. energy.

Figure 2 shows the  $\text{Li}^7 (n, n'\alpha) \text{T}$  cross section for neutrons up to 14 MeV.

Early references to the tritium breeding problem such as in the books by Bishop,<sup>2</sup> Glasstone and Lovberg,<sup>3</sup> and Rose and Clark<sup>4</sup> mentioned the  $\text{Li}^6 (n\alpha) \text{T}$  reaction as a means to breed tritium.

In 1962 Myers<sup>5</sup> et al at Lawrence Radiation Laboratory performed some one dimensional diffusion theory calculations on a number of different blankets.

TABLE 1 - ZOOM 1 DIMENSION DIFFUSION CODE RESULTS BY MYERS

K - TOTAL TRITIUM ATOMS PRODUCED PER INCIDENT NEUTRON

K<sub>7</sub> - TRITIUM ATOMS PRODUCED BY  $Li^7(n,Tn)He^4$  PER INCIDENT NEUTRON

MATERIAL	TOTAL RADIAL THICKNESS OF BLANKET-CM					
	9	21	39	60	84	96
EUTECTIC FLIBE WITH WATER LAYER						
K	.295	.649	1.066	1.332	1.433	1.448
K <sub>7</sub>	.073	.135	.175	.187	.190	.190
EUTECTIC FLIBE						
K	.114	.496	1.133	1.532	1.641	1.662
K <sub>7</sub>	.092	.174	.225	.241	.245	.246
Li <sup>6</sup> EUTECTIC FLIBE						
K	.129	.604	1.116	1.361	1.440	1.451
30 MOLE PERCENT LiF FLIBE						
K	.066		1.036	1.465	1.645	1.679
K <sub>7</sub>	.051		.132	.144	.148	
15 MOLE PERCENT LiF FLIBE						
K	.033	.303	.946	1.418	1.638	1.682
K <sub>7</sub>	.024	.046	.063	.070	.072	.072
LITHIUM METAL						
K	.264	.578	.934	1.232	1.453	1.528
K <sub>7</sub>	.254	.503	.668	.722	.736	.737
LITHIUM-6 METAL						
K	.065	.298	.632	.853	.955	.976
LiNO <sub>3</sub>						
K	.091	.222	.652	1.013	1.238	1.298
K <sub>7</sub>						
LiNO <sub>2</sub>						
K	.114	.326	.742	1.104	1.313	1.365
K <sub>7</sub>	.109	.239		.405	.424	.428



Table 1 gives results of these calculations. The geometry used was cylindrical with an inner radius at 77 cm. Flibe is a mixture of lithium fluoride (LiF) and beryllium fluoride ( $\text{BeF}_2$ ). Eutectic flibe composition is 48 mole % LiF. These results indicate attractive tritium breeding ratios are attainable. In fact, only the all lithium-6 blanket failed to provide required tritium breeding. Also note the  $\text{Li}^7(n,n'\alpha)\text{T}$  reaction can be an important part of the total tritium breeding. In the case of the natural lithium blanket, material #6, lithium-7 reactions account for nearly half the tritium. One major drawback of Myer's calculations is that the fluoride and nitrogen cross sections were unavailable so oxygen cross sections were used in their places.

In a 1965 thesis Impink<sup>6</sup> studied the effect on tritium breeding for a number of blanket variables. Impink's standard blanket was made up of a 2 cm molybdenum first wall, a 6.25 cm coolant containing flibe, then a 56 cm primary attenuator containing 79 v/o flibe plus 21 v/o graphite. The flibe is composed of 66 mole % lithium fluoride and 34 mole % beryllium fluoride. A sample of Impink's results are listed in Table 2.

TABLE 2

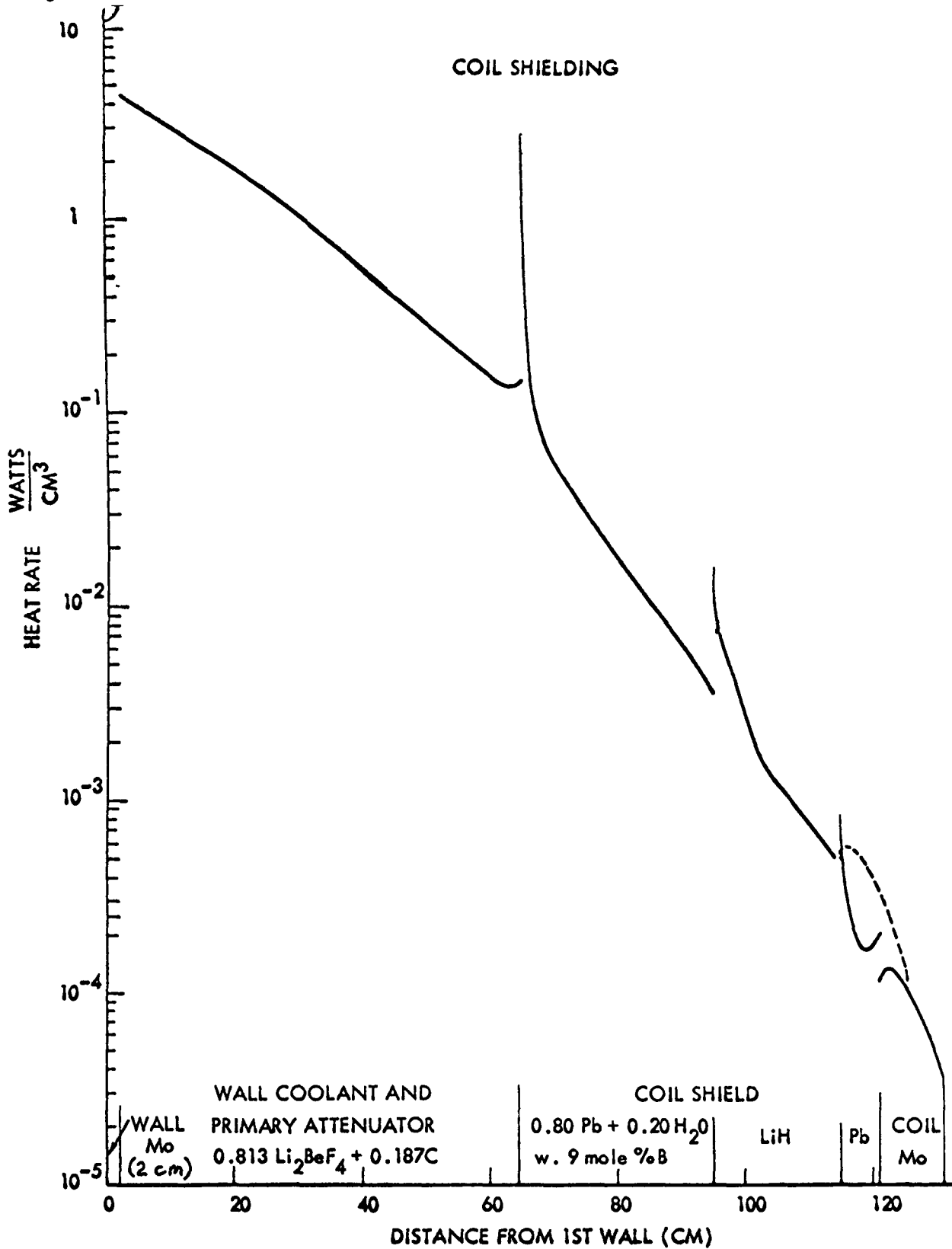
## TRITIUM BREEDING CALCULATIONS BY IMPINK

FIRST WALL	COOLANT CHANNEL	PRIMARY ATTENUATOR	TRITIUM PER NEUTRON
(VARIATION IN FIRST WALL)			
NONE	STD.	STD.	1.0717
Ni (1 cm)	"	"	1.0042
Mo (1 cm)	"	"	1.1414
Mo (2 cm) (STD)	"	"	1.1679
(VARIATION IN $Li^6$ ISOTOPIC ACTION)			
Mo (2 cm)	NAT	NAT	1.1499
"	0.20	NAT	1.1806
"	0.50	0.50	1.2163
(ADDITIONAL BE IN PRIMARY ATTENUATOR)			
Mo (2 cm)	STD.	NONE	1.1499
"	"	+ 5 cm BE	1.5526
"	"	+ 9 cm BE 0	1.4942
Ni (1 cm)	"	+ 10 cm BE	1.5328

Impink found tritium breeding in flibe to be less than Myers estimated. This lower breeding is attributed in part to Impink's better treatment of the fluorine cross sections.

In a companion thesis Homeyer<sup>7</sup> dealt with thermal and chemical problems in blankets. Figure 3 displays sample results of nuclear heating.

Figure 3 - Nuclear heating rates in the blanket and magnet coil by Homeyer



The heat rate in watts/cm<sup>3</sup> is based on a 14 MeV neutron energy flux of 1 MW/m<sup>2</sup>. Recoverable energy was 17.4 MeV per incident 14.1 MeV neutron.

TRITON PRODUCTION PER 14 MEV NEUTRON INCIDENT ON VARIOUS BLANKETS BY BELL

PROBLEM NUMBER	R <sub>1</sub> (CM)	M <sub>1</sub>	GEOMETRY R <sub>2</sub> (CM)	M <sub>2</sub>	R <sub>3</sub>	TRITON PRODUCTION				
						T <sub>6</sub>	T <sub>7</sub>	L	T	T <sup>+</sup>
TU0	0	Li	∞	INFINITE MEDIUM		1.05	0.83	0	1.88	1.88
TU01	0	FLIBE	∞	INFINITE MEDIUM		1.02	0.10	0	1.12	1.12
TU3	0	BE + 1.001 Li <sup>6</sup>	∞	INFINITE MEDIUM		2.74	0	0	2.74	2.74
TU3A	0	BE <sub>2</sub> C + 0.001 Li <sup>6</sup>	∞	INFINITE MEDIUM		2.03	0	0	2.03	2.03
TU1A	77	Li	112.0			0.16	0.59	0.88	0.75	1.63
TU1B	77	Li	137.0			0.40	0.73	0.64	1.13	1.77
TU1C	77	Li	177.01			0.72	0.80	0.33	1.52	1.85
TU2A	77	FLIBE	112.0			0.73	0.10	0.29	0.83	1.12
TU2B	77	FLIBE	137.0			0.97	0.10	0.05	1.07	1.12
TU2C	77	FLIBE	177.0			1.02	0.10	0	1.12	1.12
TU4	10	Cu	13.3		113.3	0.81	0.39	0.36	1.20	1.56
TU4A	10	Mo	13.3		113.3	0.86	0.41	0.39	1.27	1.66
TU5	10	VOID	13.3	0.5 BE + 0.5 LI	73.3	2.09	0.22	0.15	2.31	2.46
TU5A	10	VOID	13.3	0.75 BE + 0.25 LI	73.3	2.47	0.09	0.06	2.56	2.62
TU6	10	Cu	13.3	0.5 BE + 0.5 LI	73.3	1.60	0.11	0.08	1.71	1.79
TU6'	10	Mo	13.3	0.5 BE + 0.5 LI	73.3	1.701	0.12	0.09	1.82	1.91
TU6A	10	Cu	13.3	0.5 BE <sub>2</sub> C + 0.5 LI	73.3	1.35	0.11	0.10	1.46	1.57
TU6B	10	Cu	13.3	0.75 BE + 0.25 LI	73.3	1.75	0.05	0.04	1.80	1.84
TU7	76	Mo	77.0	0.5 BE + 0.5 FLIBE	117.0	1.46	0.04	0.13	1.50	1.63
TU7A	76	Mo	77.0	0.75 BE + 0.25 "	117.0	1.76	0.02	0.12	1.78	1.90
TU7B	76	Mo	77.0	FLIBE	117.0	0.87	0.08	0.19	0.95	1.14

Table 3

In another 1965 paper<sup>8</sup> Bell of LASL describes tritium breeding calculations in a number of different blankets. Table 3 lists Bell's blanket descriptions and their resulting tritium breeding. Calculations were made using the DTK neutron transport code with 25 energy groups and the  $S_4$  approximation. Blanket geometry in each case was an infinitely long cylindrical annulus. The 14 MeV neutron source is inside  $R_1$ , material  $M_2$  is between  $R_2$  and  $R_3$ , where the R's refer to radii in cm.  $T_6$  and  $T_7$  are the tritons produced per incident 14 MeV neutron by the  $Li^6(n,\alpha)T$  and  $Li^7(n,n'\alpha)T$  reactions, respectively. Total tritium produced ( $T$ ) is the sum of  $T_6$  and  $T_7$ .  $L$  is the leakage and  $T^+$  is the sum of  $T$  and  $L$ . Bell's infinite medium results indicate that lithium, beryllium plus a little lithium-6, and beryllium carbide plus a little lithium-6 are attractive tritium breeding materials, while flibe is marginal. Flibe's major problem appears to be inelastic scattering by fluorine which can degrade the fast neutron's energy below the  $Li^7(n,n'\alpha)T$  and  $Be(n,2n)$  thresholds. The fluorine cross sections are the most uncertain of the lot and should be suspect.

Bell's results also show that pure lithium blankets can give attractive breeding but need to be thicker than blankets containing beryllium, while problem numbers TU & and TU 7A show that flibe plus beryllium can give attractive breeding.

Up to this point, I have dealt with idealized blanket studies. These are very useful in that they indicate attractive tritium breeding looks possible and show what directions one might go to come up with real blanket designs.

I would now like to discuss in some detail neutron studies that consider to some extent structural, and heat transfer requirements, and also the tradeoff between tritium breeding and energy generation. The bases for these discussions are the papers presented at the International

Nuclear Fusion Reactor Conference held in England at the Culham Laboratory last September.

### Lithium Blankets

In a paper entitled "Tritium Breeding and Energy Generation in Liquid Lithium Blankets,"<sup>9</sup> I examine some of the neutronic aspects of a lithium blanket.

Liquid lithium appears uniquely qualified to meet all blanket requirements except structural.

1. It is the fertile material from which tritium must be bred.
2. It is a reasonably good fast neutron moderator.
3. It is an excellent heat transfer fluid, assuming MHD losses caused by the presence of strong magnetic fields can be made acceptably low.
4. It is not subject to radiation damage.

In its simple form, the liquid lithium blanket would consist of an annulus (cylindrical or spherical) through which liquid lithium flows. The D+T plasma would be maintained inside the inner surface of the annulus.

To calculate tritium breeding and energy generation, two neutron transport codes were used. One being a three dimensional Monte Carlo code called SORS-N;<sup>10</sup> the other, a one-dimensional discrete ordinates code called ANISN.<sup>11</sup> The SORS-N code employed 66 neutron energy groups. The ANISN problems were run with a 19-energy group,  $P_0$  cross section set and an SN order of 4. In both cases, neutron cross sections were derived from the Lawrence Radiation Laboratory, Livermore, (LRL), Howerton Evaluated Neutron Cross Section Library. All the SORS-N problems were run with five batches of 500, 14.1 MeV source neutrons each.

The SORS-N runs calculated tritium breeding and energy generation per source neutron. The ANISN runs calculated only tritium breeding per source neutron.

To examine the effects of structural material present in a liquid lithium blanket, a number of SORS-N calculations were run on blankets with various volume fractions of niobium.

Blanket geometry for these problems was a spherical annulus made up of three zones (zones 2-4). Outer radii ( $R_0$ ) of these zones (including the inner zone simulating the plasma region) were 100, 101, 202 and 302, respectively.

With only liquid lithium in the three blanket zones, tritium breeding and energy generation per source fusion neutron are 2.10 T/n and 17.14 MeV/n, respectively. Table 4 lists the complete results by zone

TABLE 4  
Tritium Breeding and Energy Generation with  
100 v/o Lithium in Zones 2, 3, and 4.

	Zones			Leakage	Totals
	2(1 cm)	3(100cm)	4 (100 cm)		
	(Input)				
Atoms/cm-barn	0.03679	0.03679	0.03679	—	—
Iso. fractions, $\text{Li}^6$	0.0742	0.0742	0.0742	—	—
$\text{Li}^7$	0.9258	0.9258	0.9258	—	—
	(Results per Source Neutron)				
Triton generation	0.02263	1.6894	0.3928	—	2.1048
$\text{Li}^6(n,\alpha)\text{T}$ reactions	0.00524	0.69084	0.3065	—	1.0025
$\text{Li}^7(n,n'\alpha)\text{T}$ reactions	0.01739	0.99857	0.08637	—	1.1023
(n,2n) reactions	0.00229	0.09284	0.00471	—	0.0998
Neutron energy deposition (MeV)	0.2196	14.3098	2.4989	0.116	17.144
Neutron energy deposition (%)	1.28	83.5	14.6	0.665	100.0
Standard deviation (%)	2.1	1.0	1.3	8.4	

for this idealized blanket. The tritium breeding ratio of 2.1 is quite attractive, about 10% higher than Bell got in his infinite medium Li case. But note that this is in an idealized structureless blanket presented for comparison with more realistic blankets containing structure.

To account for required blanket structure, zone 2 (1 cm) was made all niobium and the liquid lithium in zones 3 and 4 was varied for 100% lithium to a homogeneous mixture of lithium diluted with 20 v/o niobium. Figure 4 graphically displays the results of SORS-N analysis of these blankets.

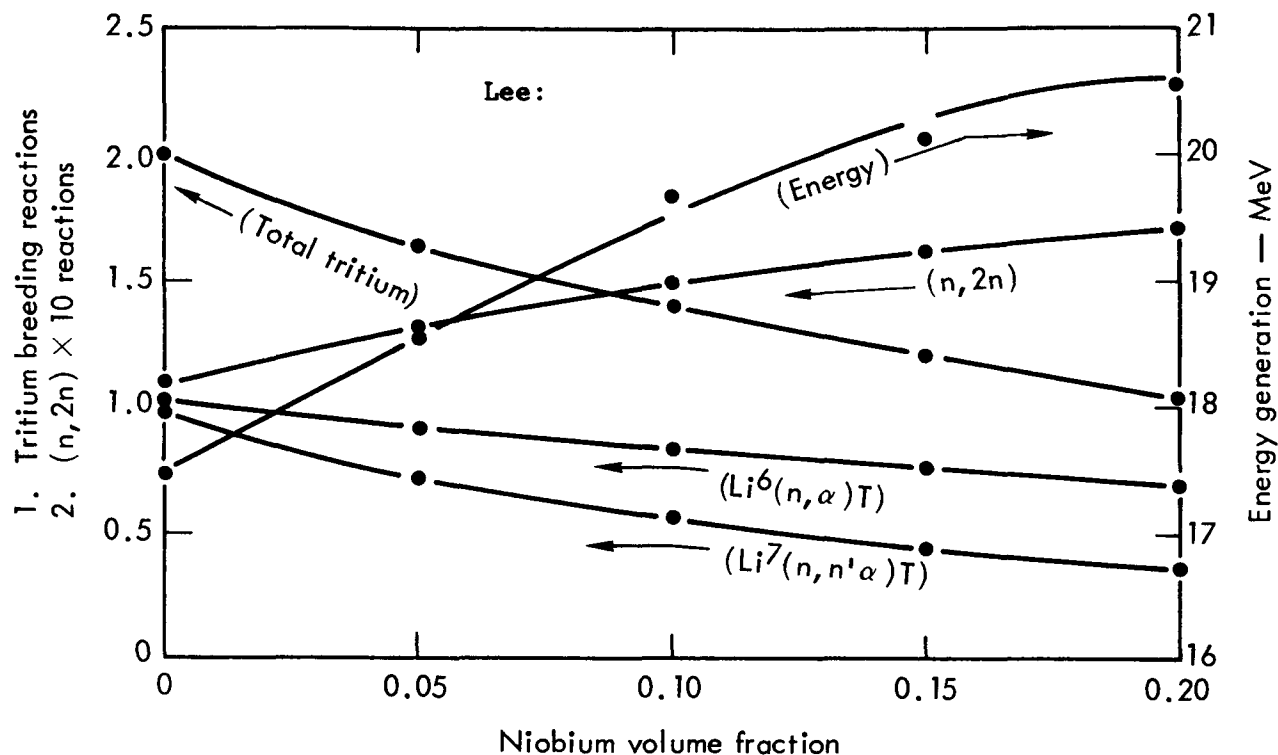


Fig. 4 Tritium & energy generation per 14.1-MeV source neutron vs. niobium volume fraction (first wall 1.0-cm niobium).

The increase in energy generation and the decrease in tritium breeding with increasing niobium is attributed to the increased capture in the niobium which has an energy release of 7.17 MeV vs. 4.78 MeV for the lithium-6 capture reactions.



The tritium breeding ratio drops to 1.0 at a niobium fraction of about 20%. This fraction is higher than that considered necessary for an achievable blanket structure. It appears blankets containing only lithium and required structure can meet tritium breeding requirements.

A second series of problems was run with the inner zone of niobium reduced in thickness from 1.0 to 0.1 cm. Figure 5 displays these results. The only noticeable effect caused by the change is the slight increase in  $\text{Li}^7$  tritons and a reduction in energy generation at low niobium fractions.

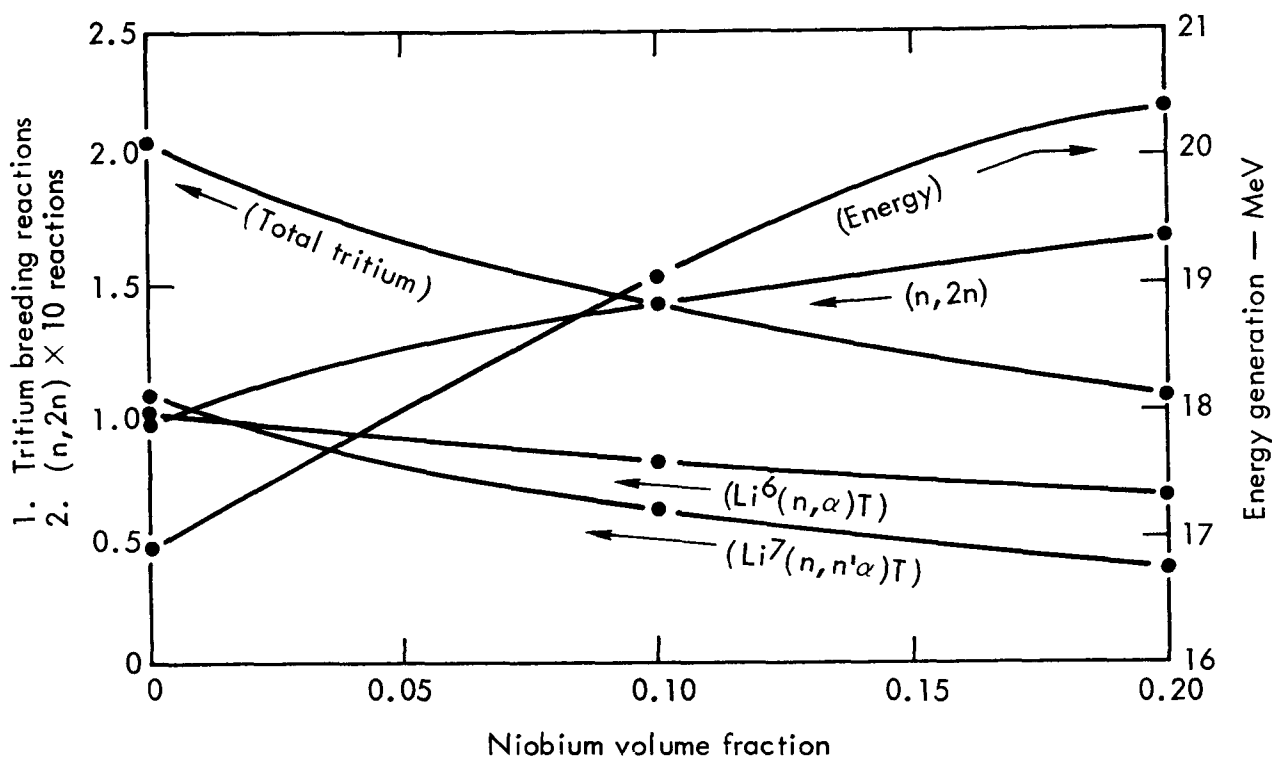


Fig. 5 Tritium & energy generation per 14.1-MeV source neutron vs. niobium volume fraction (first wall 0.1-cm niobium).

A major reason for a thin wall is heat transfer. Cooling the first wall (or vacuum wall) is a very demanding requirement. Bremsstrahlung and cyclotron radiation heat its inner surface, while neutrons and secondary gamma heat it volumetrically.

The neutron-caused heating can be estimated by using the results SORS-N calculated for the blanket with the 0.1 cm Nb first wall. Energy

deposition in this wall per 14.1 MeV source neutron is 0.17 MeV, therefore wall heating is estimated to be 12 watt/cm<sup>3</sup> for 1 MW/m<sup>2</sup> of neutrons incident on the first wall.

To get a better feeling for what goes on inside a lithium blanket containing structure, SORS-N was used to calculate the tritium breeding and energy-generation profiles in a blanket containing a homogeneous mixture of 95% liquid lithium plus 5% Nb by volume. The blanket was made up of seven cylindrical annular zones with the inner radius of the first blanket zone (zone 2) at 220 cm. The fusion neutron source was simulated by a 115 cm disc source of isotropic 14.1 MeV neutrons at the center of and normal to the blanket axis.

TABLE 5  
Tritium Breeding and Energy Generation  
per Zone per Source Neutron.

Zone	R <sub>o</sub> (cm)	T <sup>6</sup>	T <sup>7</sup>	T	Energy	SD(%)	E(%)	Ė(%/cc)
1	<b>220</b>	—	(Plasma Zone)		—	—	—	—
2	222	0.0202	0.0357	0.0559	0.698	1.4	3.75	1.35 × 10 <sup>-3</sup>
3	226	0.0410	0.0703	0.1113	1.369	1.6	7.35	1.305 × 10 <sup>-3</sup>
4	234	0.0824	0.1356	0.2180	2.598	0.9	13.95	1.205 × 10 <sup>-3</sup>
5	250	0.1635	0.2211	0.3846	4.370	1.2	23.45	9.64 × 10 <sup>-4</sup>
6	282	0.2718	0.2536	0.5254	5.451	2.5	29.20	5.45 × 10 <sup>-4</sup>
7	346	0.2711	0.1212	0.3923	3.508	3.0	18.82	1.49 × 10 <sup>-4</sup>
8	410	0.0685	0.0130	0.0815	0.634	5.5	3.40	2.24 × 10 <sup>-5</sup>
TOTALS		0.919	0.850	1.769	18.627	2.0		

Results are listed in Table 5. Note that 95% of the tritium breeding (1.69 triton per neutron) and 96.6% of the energy generation (18.0 MeV per neutron) occurs in the first 126 cm of the blanket. Also note that energy density, shown in the last column, varies by more than a factor of 10 over the blanket radius.

It is desirable to have a fast-running neutronic code for parametric blanket studies. The ANISN code (a one-dimensional discrete ordinates transport code) has the potential for short running time. It also can be recast in subroutine form for use in an overall blanket design code.

To compare the results of ANISN and SORS-N, the first set of blanket problems was rerun with ANISN.

The ANISN problems used the same zone geometry and material compositions as the SORS problems. Zone 1 [plasma-vacuum region ( $R_0 = 100$  cm)] was broken up into three equal-thickness intervals with a distributed 14.1 MeV neutron source in the first interval. Zone 2 [niobium vacuum wall (1 cm)] had three equal intervals. Zones 3 and 4 [2,100-cm-thick blanket regions with liquid lithium diluted with from 0 to 20% by volume with niobium] had ten equal intervals.

A quadrature (SN) order of four was used with the following set of direction cosines and weights:

<u>Cosine (<math>\mu</math>)</u>	<u>Weight</u>
-1.00000	0
-0.881917	0.166667
-0.333333	0.333333
+0.333333	0.333333
+0.881917	0.166667

And a nineteen-energy group set of isotropic neutron cross sections with six groups of downscatter was used.

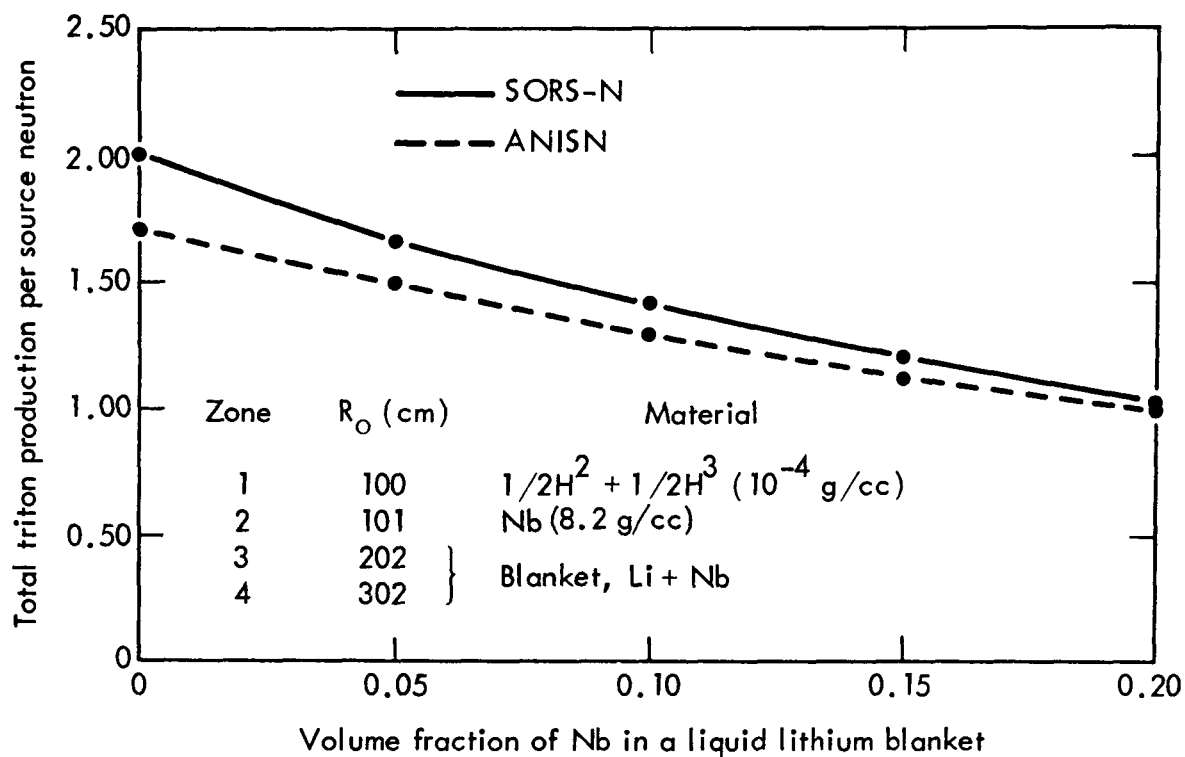


Fig. 6 Comparison between SORS-N and ANISN calculated blanket tritium production.

Figure 6 compares ANISN to SORS-N total tritium production per source neutron. ANISN underestimates the SORS-N results by  $\approx 15\%$  for the 100 v/o Li blanket to  $\approx 5\%$  for the 80 v/o Li, 20 v/o Nb blanket.

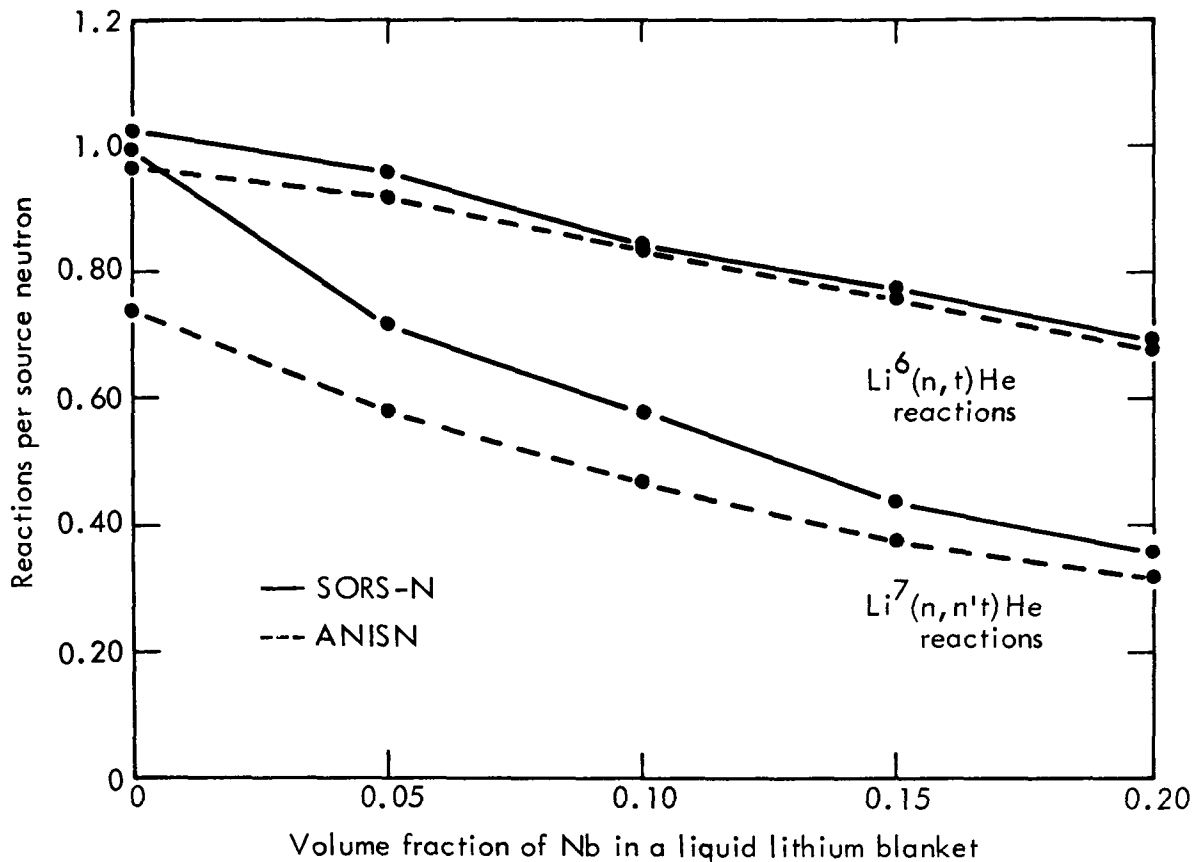


Fig. 7 Comparison of SORS-N and ANISN calculated blanket tritium-producing reactions.

Figure 7 compares individual triton reactions and shows that the fast neutron  $Li^7(n_f, n'\alpha)T$  reaction is the major contributor to ANISN's under-estimation of triton production.

Running time on the CDC 6600 averaged  $\approx 0.6$  min. per ANISN run compared to  $\approx 1.6$  min. per SORS run.

In light of the relative crudeness of the ANISN problems [specifically, the small number of energy groups (19), isotropic scattering, SN order of four and cross sections generated using a flat flux spectrum] the results of the ANISN runs compared to SORS are considered quite good. ANISN is considered a useful tool to run blanket parameter studies. It

also can be rewritten as a blanket neutronic subprogram for use in a thermo-nuclear reactor systems code while SORS cannot be. The more exact but more costly SORS code will be used to check and normalize ANISN results and for three-dimensional analysis when required.

To determine what effect lithium-6 concentration has on tritium breeding in a liquid lithium blanket, a series of blanket calculations were run using the ANISN code.

The basic blanket model consisted of a spherical annulus ( $R_i = 100$  cm,  $R_o = 330$  cm) with a homogeneous mixture of 95% lithium and 5% niobium (structure). The sphere enclosed by the blanket was a D-T neutron (14.1 MeV) source. The lithium-6 to lithium atom fraction was varied from 1 to 10% in five consecutive ANISN runs.

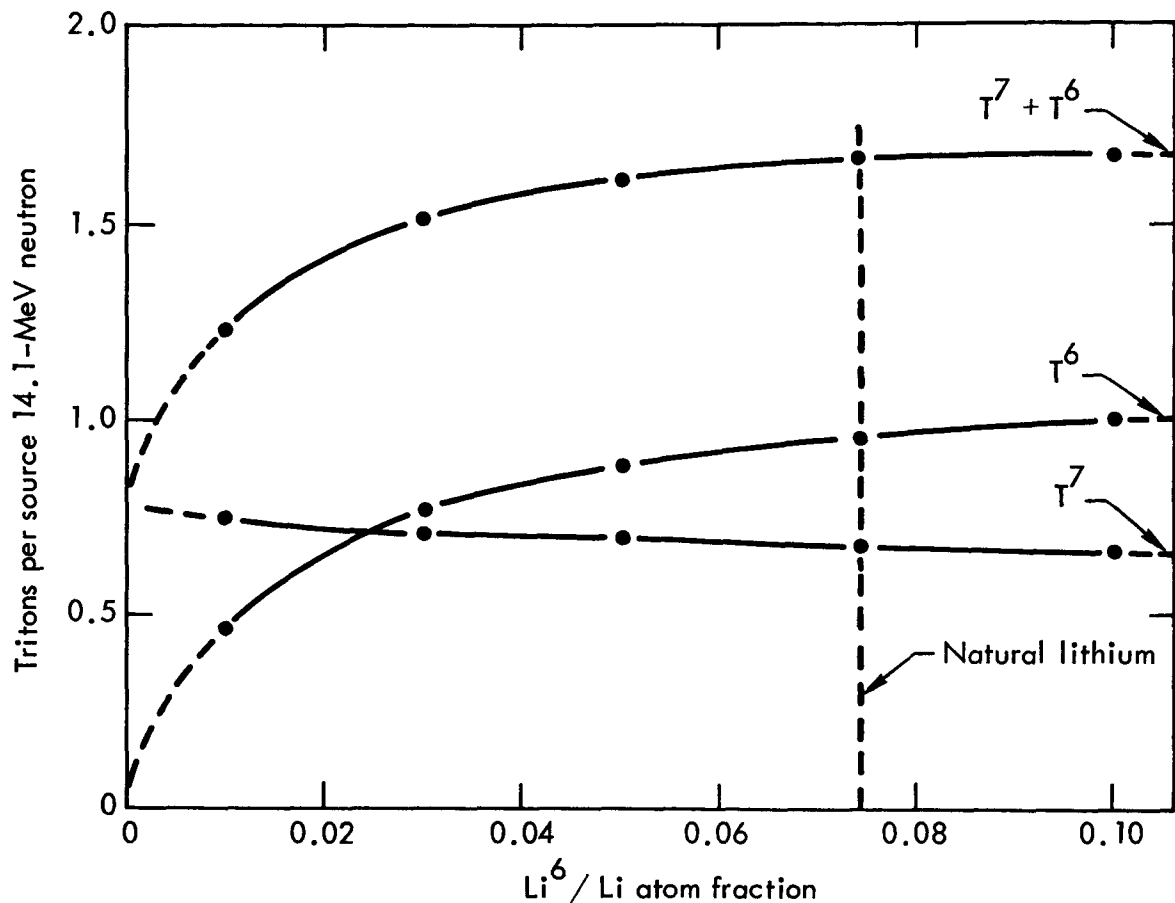


Fig. 8 Tritium breeding vs.  $\text{Li}^6/\text{Li}$  atom fraction in 95 v/o + 5 v/o Nb homogeneous blanket. [ $T^6$  and  $T^7$  refer to  $\text{Li}^6(n,\alpha)\text{T}$  and  $\text{Li}^7(n,n',\alpha)\text{T}$  reactions, respectively.]

Figure 8 shows the results of the five runs, where  $T^6$  and  $T^7$  represent the number of  $Li^6(n,\alpha)T$  and  $Li^7(n,n'\alpha)T$  reactions, respectively, per source neutron.

From these results it is apparent that lithium-6 enrichment of natural lithium is not worthwhile. In fact, the lithium can be depleted in lithium-6 to 1% while still giving the respectable tritium breeding ratio of  $\approx 1.2$ .

The raw material for fueling a deuterium-tritium fusion plant will be water and lithium, and possibly beryllium. Deuterium makes up 0.015% of natural hydrogen, and tritium (as shown) can be bred from lithium.

But what is the tritium and energy generation potential of our lithium resources? Given a gram of natural lithium (composed of 7.42 a/o  $Li^6$  + 92.58 a/o  $Li^7$ ) how much energy could be generated?

The answer, of course, depends on what kind of blanket is used. There are two basic choices:

1. Maximize energy generation per fusion reaction by converting only Lithium-6 to tritium [by the  $Li^6(n_{th},\alpha)T + 4.97$  MeV reaction] at a breeding ratio of 1.0 and using the fusion neutron's kinetic energy to multiply more neutrons for generating more energy.
2. Maximize energy generation for the lithium used by converting as much Lithium-7 as possible by the  $Li^7(n_f,n'\alpha)T - 2.47$  MeV reaction.

Four blanket schemes are presented as examples of these two basic choices.

#### Scheme 1

A blanket consisting of  $Li^6$  + Na(30 v/o), Be(60 v/o), and Nb (10 v/o) is an example of a blanket compatible with choice 1.

Such a blanket is estimated (by SORS) to give a total generation of approximately 30 MeV per D-T fusion at a breeding ratio of 1 triton per fusion. Such a blanket could potentially convert all the lithium-6 in natural lithium, producing  $6.37 \times 10^{21}$  tritons per gram of natural lithium or in terms of energy,  $8.54 \times 10^3$  kWh per gram of natural lithium supplied.

An example of a blanket compatible with choice 2 basically would be all lithium (95 v/o) plus required structure (5 v/o Nb). Figure 8 showed breeding ratio vs.  $\text{Li}^6/\text{Li}$  atom fraction for this blanket as estimated by the ANISN transport code. Total energy generation per fusion with this blanket was estimated by SORS to be  $\approx 22$  MeV. This blanket might be used in the following three ways: (schemes 2 through 4).

#### Scheme 2

The entire blanket loop (including required makeup) could be loaded with natural lithium and run until all the excess tritons produced when the breeding ratio was greater than 1.0 are consumed. If no tritium decay is assumed (by shipping the tritium between plants), the total triton atoms generated would be  $\approx 2.74 \times 10^{22}$  per initial gram of natural lithium. Therefore, at 22 MeV per fusion,  $\approx 2.60 \times 10^4$  kWh of energy could be generated per gram of lithium. Figure 9 graphically displays approximate integrated blanket reactions and tritium excess vs. integrated power for scheme 2.



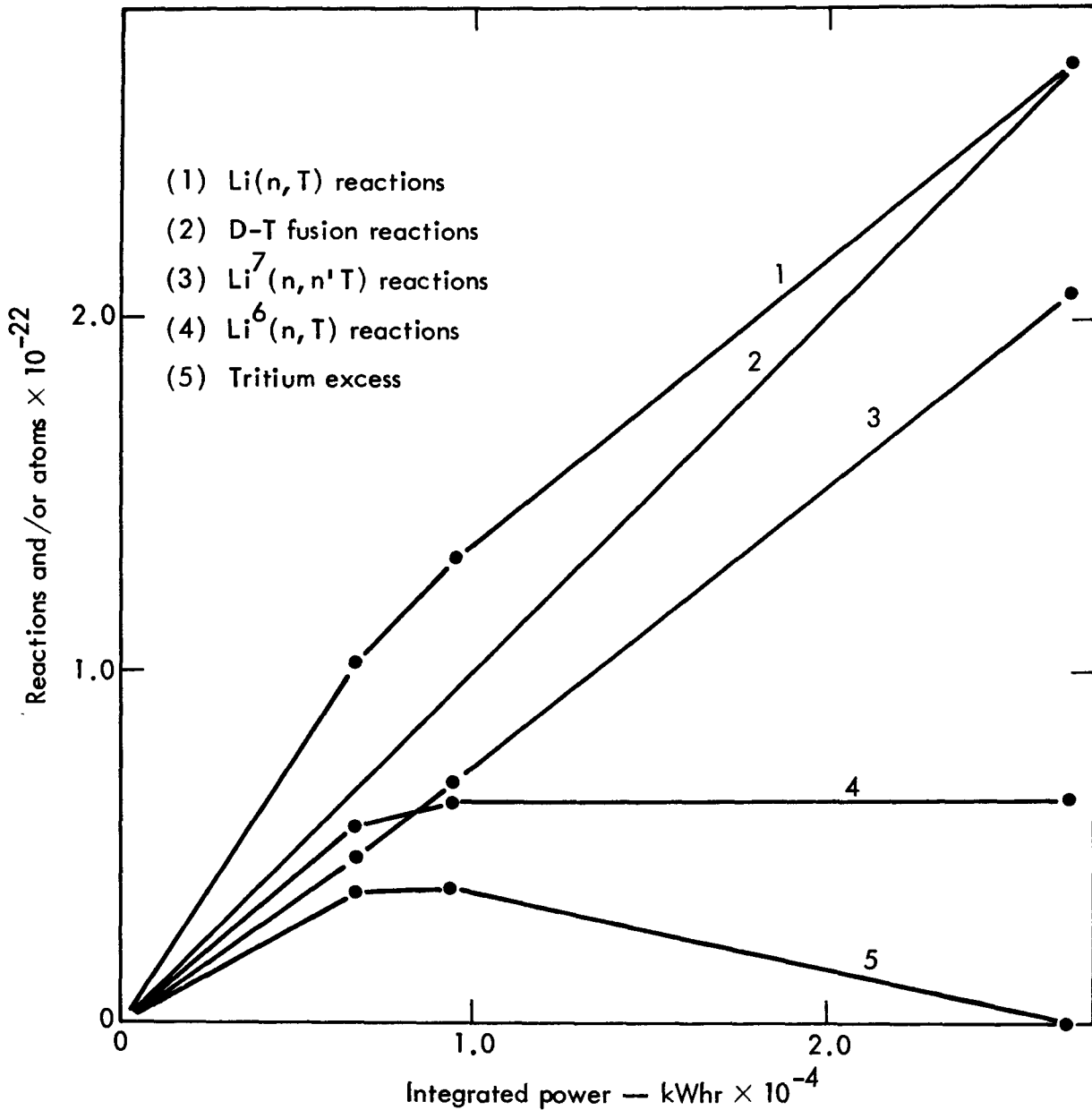


Fig. 9 Integrated lithium and tritium reactions and tritium excess vs. integrated power per initial gram of natural lithium (without tritium decay).

Scheme 3

Lithium utilization scheme 3 is the same as scheme 2, except that the excess tritium is stored, then used by the same reactor that produced it. Specifying an average power density of 40 watts/initial gram of lithium supplied, tritium decay of  $\text{He}^3$  reduced the energy generation by D-T fusion (at 22 MeV per fusion) to  $\approx 1.49 \times 10^4$  kWh per initial gram of lithium; or  $1.7 \times 10^4$  kWh, if the  $\text{He}^3$  produced is used for D- $\text{He}^3$  fusion.

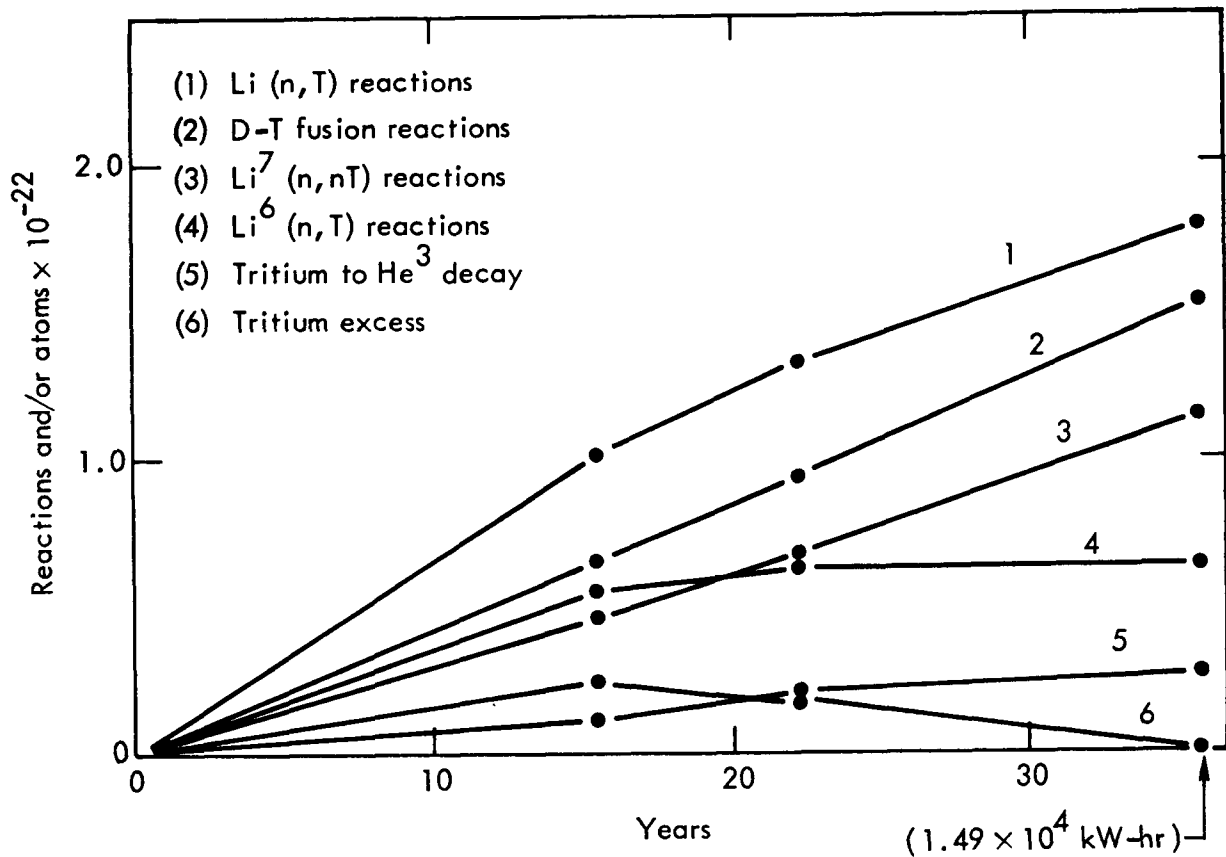


Fig. 10 Integrated lithium and tritium reactions and tritium excess per initial gram of natural lithium vs. time (average power density = 40 W/initial gram of lithium).

Figure 10 displays approximate integrated lithium and tritium reactions (including decay to  $\text{He}^3$ ) and tritium excess as a function of blanket age, for scheme 3.

#### Scheme 4

The last scheme considered also employs the basic lithium blanket but at a constant  $\text{Li}^6/\text{Li}$  atom fractions required to give an effective tritium breeding ratio of 1.0. From Figure 8 we estimate this atom fraction to be  $\approx 0.003$ . This fraction would be maintained by replacing depleted blanket lithium with natural lithium at the required rate. This scheme would produce and consume  $\approx 2.58 \times 10^{22}$  tritons per gram of natural lithium used while generating  $\approx 2.53 \times 10^4$  kWh of energy.

Table 6 compares the tritium and energy generation potential of a gram of natural lithium of the four schemes.

TABLE 6

Comparison of the Four Lithium Utilization Schemes.

	Schemes			
	1	2	3	4
Tritium potential, atoms per gram of lithium	$6.37 \times 10^{21}$	$2.74 \times 10^{22}$	$1.78 \times 10^{22}$	$2.58 \times 10^{22}$
Energy potential, kWh per gram of lithium	$8.54 \times 10^3$	$2.69 \times 10^4$	$1.49 \times 10^4$ ( $1.71 \times 10^4$ if the He <sup>3</sup> is used.)	$2.53 \times 10^4$

From a lithium utilization standpoint, scheme 4 appears to be the best choice. Its energy generation potential of  $\approx 2.53 \times 10^4$  kWh ( $8.64 \times 10^7$  BTU's) per gram of natural lithium used is only 5% below the highest achieved (scheme 2's) and requires tritium shipping and storage only for start up.

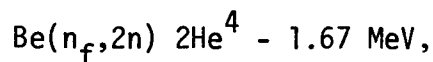
At initial start up, scheme 4's blanket loop would be filled with natural lithium. Excess tritium would be bred until the isotopic concentration of Lithium-6 is depleted to the point where the tritium breeding ratio is 1. Figure 9 shows that during this period,  $\approx 3.8 \times 10^{21}$  excess tritons are produced per gram of lithium which could be used to start up new D-T reactors. These excess tritons could also be allowed to decay into He<sup>3</sup> and "burned" in D-He<sup>3</sup> reactors.

Referring again to Figure 8, it is obvious that any tritium breeding ratio up to  $\approx 1.65$  could be maintained by tailoring the Li<sup>6</sup>/Li atom fraction if more excess tritiums were desired. But the higher the breeding ratio, the lower the energy generation potential of the lithium resources.

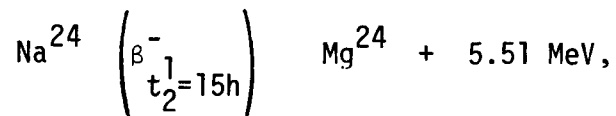
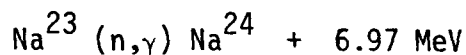
For example, if a breeding ratio of  $\approx 1.5$  were maintained by keeping the  $\text{Li}^6/\text{Li}$  atom fraction at  $\approx 0.03$ , the energy potential of a gram of natural lithium would be  $\approx 6.65 \times 10^3$  kWh, assuming all the tritium is "burned" in a D-T reactor giving  $\approx 22$  MeV per fusion. This is only 26% of what was obtained from scheme 4 at a breeding ratio of 1.0.

Economics will dictate what lithium utilization scheme is employed. If lithium is expensive, the high lithium utilization scheme 4 would be advantageous. On the other hand, if the capital cost of the fusion plant dominates and the cost of lithium-6, beryllium, and sodium is low, then a higher energy yield per fusion scheme like number 1 ( $\approx 30$  MeV vs  $\approx 22$  MeV) would be better.

In another Culham paper, entitled "Preliminary Design Considerations for an Astron Reactor System,"<sup>12</sup> I examine the tradeoff between tritium breeding and energy generation in a blanket containing beryllium for neutron multiplication by the reaction,



lithium for tritium breeding, sodium for energy generation by the respective reactions,



and niobium for blanket structure.

Neutronic analysis of the blanket employed the SORS<sup>10</sup> Monte Carlo neutron-transport code. The code uses 66-group neutron cross sections derived from the Howerton-evaluated LRL (Lawrence Radiation Laboratory, Livermore) neutron cross-section library. Five batches of 500 neutrons each were used for each problem.

Blanket geometry was taken to be an infinite length cylindrical annulus with an inner and outer radius of 280 and 380 cm, respectively. To reduce the power density near the inner radius and increase tritium breeding from lithium-7, the blanket was divided into two zones. The inner zone contained only natural lithium coolant and niobium structure with volume fractions of 75 and 5, respectively. Zone 1 thickness was 10 cm. The neutronic analysis treated zone 1 as two 5-cm-thick regions, regions 2 and 3. Region 1 simulated the D-T plasma and contained a 14.1 MeV isotropic neutron line source at the center of and normal to the blanket axis.

The outer blanket zone was 90 cm thick and subdivided into five regions (4 through 8) of increasing thicknesses. This zone contained 35% beryllium, 40% coolant, and 5% structure by volume. Both blanket zones 1 and 2 contain 20 v/o heat pipe void. Neutron analysis of a blanket in which zone 2 contained 50 v/o beryllium, 25 v/o coolant and 5 v/o structure, is also presented for comparison.

The materials in the two blanket zones are homogeneous mixtures of the required elements. Values of weight and atom densities of these are based on an average material temperature of 1075°K. The beryllium density used is 90% of theoretical to provide for a 10 v/o interconnected void for escape of helium generated by the Be(n,2n) 2He reactions. (See Table 7).

TABLE 7

Element Densities

Element	Density $\rho$ (g/cc)	$N(\text{atoms/cm}^3) \times 10^{-24}$
Be	1.592	0.1065
Li	0.447	0.0388
Na	0.742	0.01945
Nb	8.25	0.0535

With the volume fractions of the beryllium, sodium or lithium coolant, and the niobium structure fixed by heat-transfer and structural requirements at 35, 40 and 5 v/o, respectively, the ratio of lithium-6, lithium-7, and Na in the coolant was varied to examine the trade-off between tritium production and energy generation per 14.1 MeV fusion neutron. Table 8 lists the results of four such variations of the coolant mixture

TABLE 8

Tritium Breeding and Energy Generation  
vs. Coolant Mixture With Total Coolant = 40 v/o

Case	Coolant fluid (v/o)			Tritium production (T/n)	Energy generation, E (MeV/n)
	Li <sup>6</sup>	Li(nat)	Na		
1	0.1	0	39.9	0.936	24.21
2	0.2	0	39.8	0.998	23.94
3	40	0	0	1.84	22.46
4	0	40	0	1.639	23.08

For comparison, a blanket with 50 v/o beryllium, 25 v/o coolant, and 5 v/o niobium structure in region 2 gives the results listed in Table 9.

TABLE 9

Tritium Breeding and Energy Generation vs.  
Coolant Mixture With Total Coolant = 25 v/o

Case	Coolant fluid (v/o)			Tritium production (T/n)	Energy generation, E (MeV/n)
	Li <sup>6</sup>	Li(nat)	Na		
11	0.1	0	24.9	1.24	26.05
12	25	0	0	2.07	24.61
13	0	25	0	1.69	24.73

Figure 11 graphically displays these results.

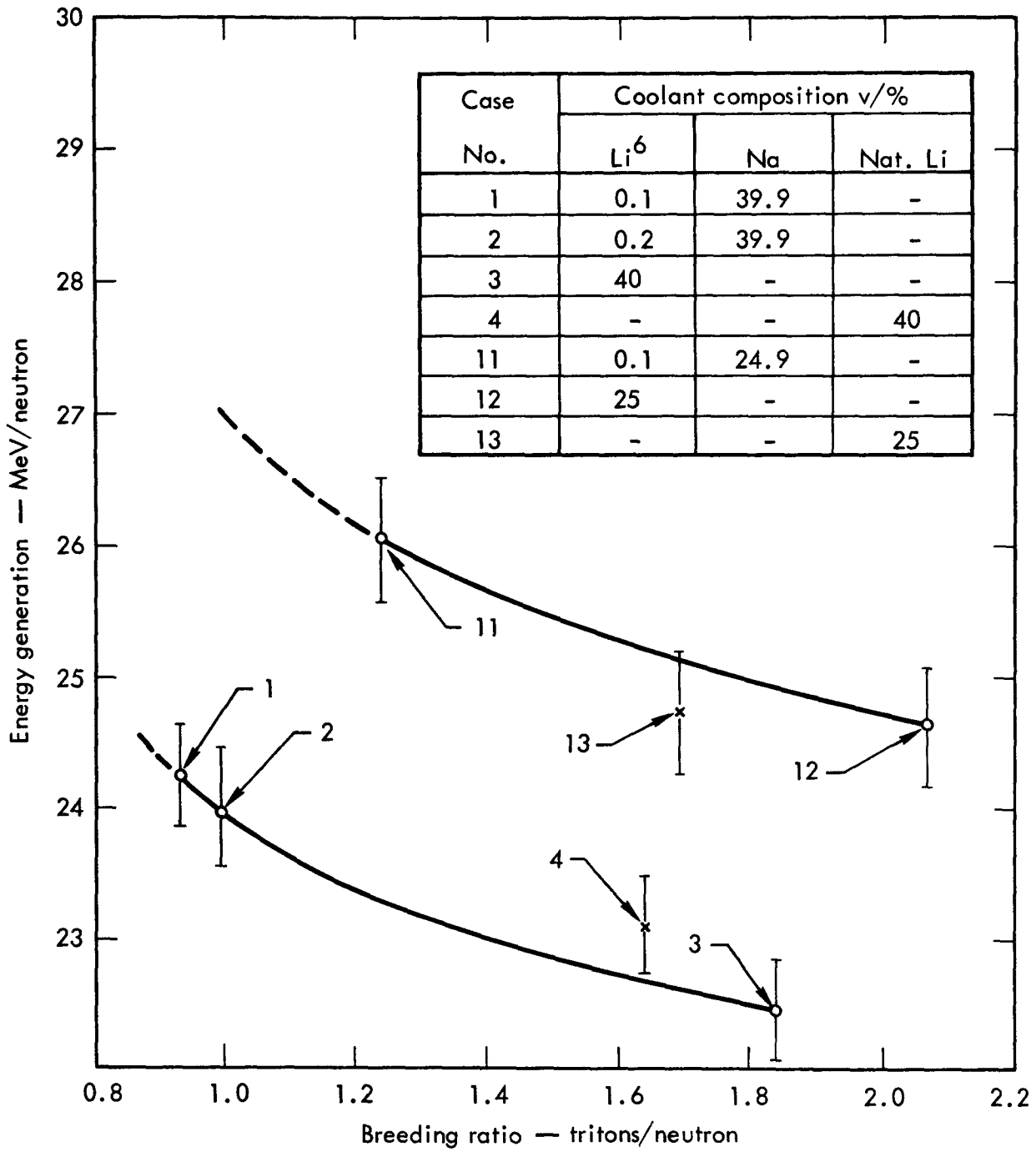


Fig. 11 Energy generation vs. breeding ratio for varying coolant mixtures.

Cases 1 through 4 and cases 11 through 13 are blankets with total coolant volume fractions at 40 percent and 25 percent, respectively. It is interesting to note the natural lithium coolant cases (4 and 13) compared to the lithium-6-sodium mixture cases. It does not appear worthwhile to separate lithium-6 and mix it with sodium.

More detailed results from case 4 are listed in Table 10.

TABLE 10  
Tritium Breeding and Energy Generation By Zone for Blanket, Case #4

Region No.	Outer radius $R_o$ (cm)	Tritons per fusion neutron			Energy/fusion neut. $E/n$ (MeV)	Std. dev. %
		$T^6/n$	$T^7/n$	$T/n$		
1	280	- Plasma Region				
2	285	0.100	0.068	0.168	1.92	2.4
3	290	0.117	0.066	0.183	1.98	2.4
4	293	0.085	0.033	0.118	2.50	1.9
5	300	0.100	0.030	0.130	2.41	2.3
6	320	0.428	0.083	0.511	7.71	1.6
7	340	0.286	0.036	0.322	4.13	4.0
8	380	0.187	0.020	0.207	2.43	2.4
Totals		1.303	0.336	1.639	23.07	

The  $T^6/n$ ,  $T^7/n$ ,  $T/n$  and  $E/n$  are again the lithium-6, lithium-7 and total triton producing reactions and energy generation per fusion neutron in the seven blanket regions and their totals.

It is apparent the presence of beryllium in the blanket (case 4 has 35 percent by volume) causes a marked increase in energy generation, 23 vs. 18 MeV per 14 MeV neutron or 28 percent, over the all lithium blanket discussed earlier.



In another Culham paper, Steiner<sup>13</sup> compared the nuclear behavior of two designs differing only in their vacuum wall coolants. One design employs lithium, the second lithium-beryllium fluoride (flibe). Calculations were performed using the ANISN one-dimensional discrete ordinates transport code employing 100 neutron and 21 gamma energy groups, an eight order quadrature and a third order Legendre expansion.

Blanket geometry was a slab with perfect reflection at the inner surface. Blanket configuration contained a  $\frac{1}{2}$  cm niobium vacuum wall, 3 cm vacuum wall coolant (lithium in design 1, flibe in design 2),  $\frac{1}{2}$  cm niobium second wall, 60 cm lithium coolant diluted by 6 percent niobium structure, 30 cm graphite moderator, and 6 cm lithium coolant diluted by 6 percent niobium structure. Total blanket thickness was 100 cm.

As shown in Table 11 tritium breeding in Steiner's blanket are 1.35 and 1.22 for designs 1 and 2. These are attractive but somewhat lower than the 1.77 and 1.64 breeding ratios the SORS-N Monte Carlo calculated for the lithium and lithium plus beryllium blankets discussed earlier.

TABLE 11 SUMMARY DESCRIPTIONS OF DESIGNS 1 AND 2 (STEINER)

REGION NUMBER	DESCRIPTION OF REGION	THICKNESS OF REGION (CM)	DESIGN 1	DESIGN 2
1	VACUUM WALL	0.5	NIOBIUM	NIOBIUM
2	VACUUM WALL COOLANT	3.0	LITHIUM	FLIBE
3	SECOND WALL	0.5	NIOBIUM	NIOBIUM
4	COOLANT + STRUCTURE	60.0	94% LITHIUM 6% NIOBIUM	94% LITHIUM 6% NIOBIUM
5	MODERATOR	30.0	GRAPHITE	GRAPHITE
6	COOLANT + STRUCTURE	6.0	94% LITHIUM 6% NIOBIUM	94% LITHIUM 6% NIOBIUM
TOTAL		100 CM		

NOTE: ALL PERCENTAGES ARE VOLUME PERCENTAGES

SUMMARY OF TRITIUM BREEDING CALCULATIONS\* (STEINER)

DESIGN	$T_6$	$T_7$	T	L
1	0.89	0.46	1.35	0.023
2	0.86	0.36	1.22	0.020

\* BASIS: ONE 14-MEV NEUTRON INCIDENT ON THE VACUUM WALL.  $T_6$  IS THE TRITIUM BREEDING TRITIUM BREEDING RATIO IN  ${}^6\text{Li}$ ,  $T_7$  IS THE TRITIUM BREEDING RATIO IN  ${}^7\text{Li}$ , T IS THE SUM OF  $T_6$  AND  $T_7$ , AND L IS THE NEUTRON LEAKAGE.

Steiner did an excellent job for analysing heating in his blankets. Figure 12 shows the striking difference in heating rates in the first three blanket regions between having lithium or flibe as vacuum wall coolants, designs 1 and 2, respectively.

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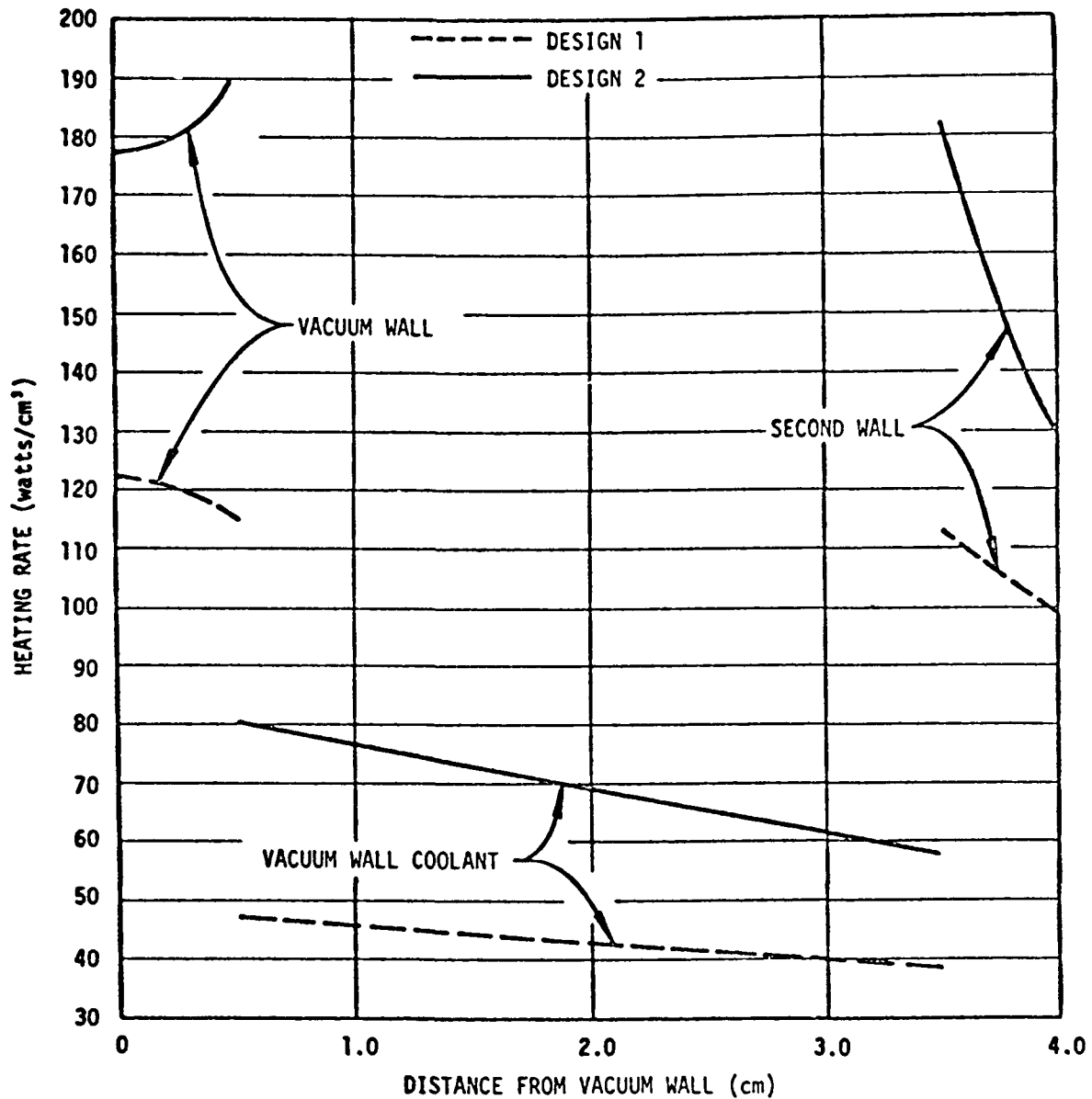


Fig. 12 Spatially-Dependent Heating Rates for the First Three Regions of Designs 1 and 2. (Steiner)

Heating rates are based on a wall loading of  $10 \text{ MW/m}^2$  of 14 MeV neutrons incident on the vacuum wall.

Figure 13 shows heating rates over the entire blanket design No. 1. Heating rates here are also based on a wall loading of  $10 \text{ MW/m}^2$ .

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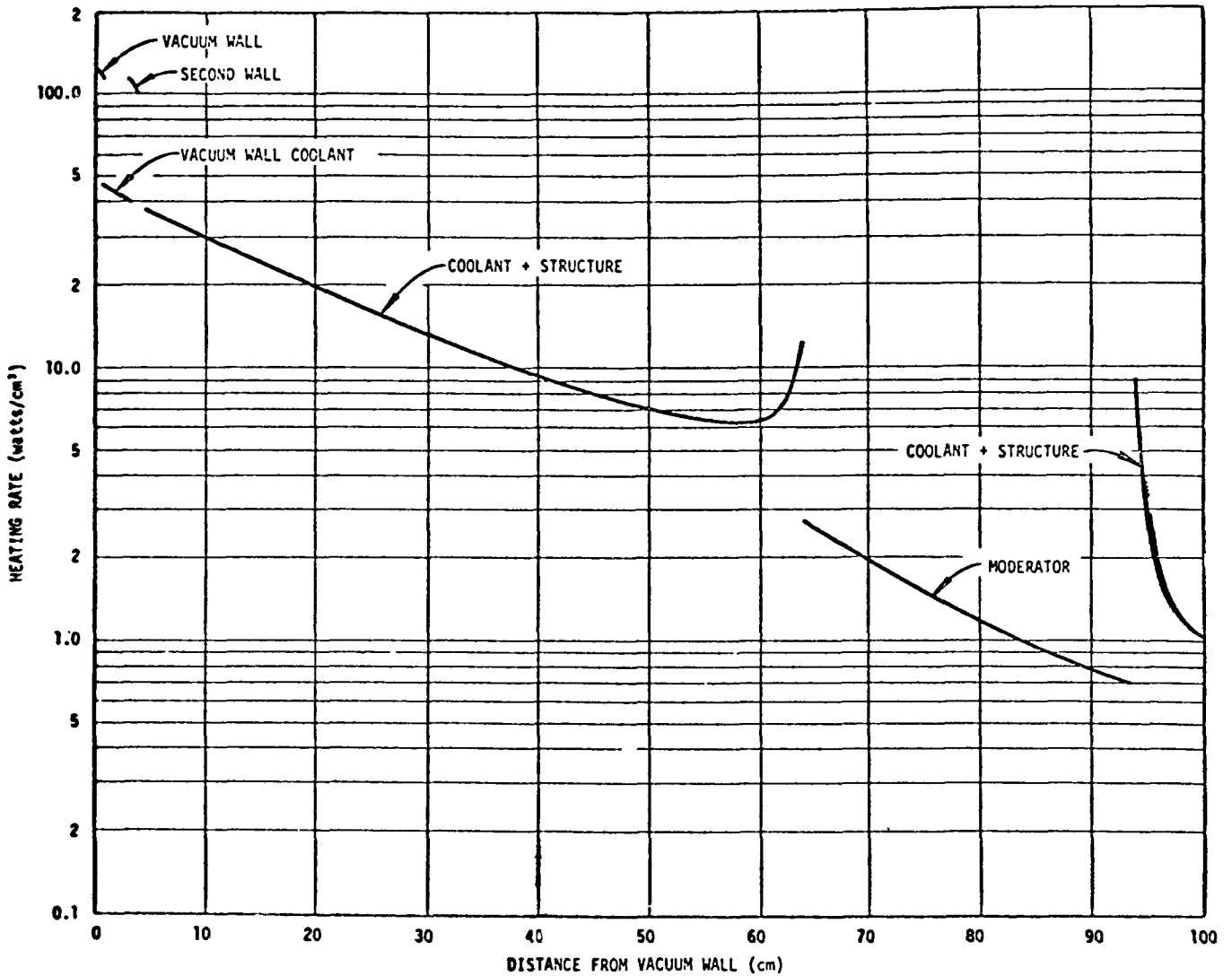


Fig. 13 Spatially-Dependent Heating Rates for Design 1. (Steiner)

Based on the preceding discussions, I believe it is safe to say that required tritium breeding from DT fusion neutrons is indeed achievable in realistic blankets. Blankets that meet structural and heat transfer requirements. What remains to be seen is the economics of real blankets.

Before leaving the subject of tritium breeding, I would like to make some comments about an intriguing concept reported by Lidsky in his Culham paper <sup>14</sup> entitled "Fission-Fusion Symbiosis." In this paper he suggests we could use the excess tritium breeding capacity that blankets have to produce fissile U233 or Pu239 by capturing the excess neutrons in TH232 or U238. This fissile fuel would then be used to augment the fuel production in a converter fission reactor to give a much shorter doubling time than the envisioned fission breeder reactor will give.

The suggestion is to combine a DT fusion reactor and a fission converter reactor into a system where the advantages of each offset the disadvantages of each. The neutron rich but power balance poor DT fusion reactor and neutron poor but power rich fission converter reactor could make a very attractive couple.

In his paper Lidsky sites as an example a molten salt converter fission reactor and a toroidal fusion reactor combination. The fission reactor had a conversion ratio of 0.96 and the fusion reactor blanket produced 1.126 tritons and 0.325 fissile atoms per DT fusion neutron. The overall fuel doubling time was 5 years and the capital cost per net electrical kW was estimated to be  $\approx 180$  \$/kW<sub>e</sub>.

If such a system is viable, I see two major advantages. First, it has a much shorter fuel doubling time, 5 years, than an all fission breeder plant which Lidsky referenced as being a minimum of 15 years. When compared with a demand doubling time of more like 10 years, the advantage is obvious. Second, if viable, such a system would negate the need for development of fission breeder reactions, which in turn would allow more effort to be placed on fusion research and development hastening the day of pure fusion power.

## RADIOLOGICAL ASPECTS

When considering a new power source to meet the future power requirements of our planet, its environmental implications must be considered. Nuclear energy's advantages over fossil fuels are obvious: No burning of our oxygen and hydrocarbon resources, and no release of carbon dioxide and other combustion products to name the most obvious.

On the other side of the coin, nuclear energy poses radiological problems that must be dealt with. In my opinion, the radiological problems of nuclear energy can be dealt with while the burning of oxygen and release of carbon dioxide etc. cannot long be dealt with assuming the energy demand continues to grow as projected.

What are the radiological problems of fusion and how do they compare with those of fission?

In my comparison of fusion and fission, I do not mean to imply fission is hazardous and fusion is not. I only want to show that dealing with the radiological problems of fusion systems should be much easier than dealing with the radiological problems of fission systems. Two obvious advantages of fusion are:

1. Fusion reactions produce stable helium while fission products are a multitude of highly radioactive elements.
2. Fusion reactors will be incapable of a nuclear run away. The fusing plasma is so tenuous there is never enough fuel present at any one time to support a nuclear excursion. A fission reactor must contain a critical mass of fissionable material containing an extremely large amount of potential energy.

Both the DD and DT fusion reaction produce fast neutrons and produce and consume tritium, respectively. These neutrons and the tritium are the primary causes of the radiological problems fusion reactor system designers must deal with.

The DT reaction with its resulting 14.1 MeV neutron and required tritium and tritium breeding poses a greater radiological problem than DD fusion. The radiological problems of a DT fusion reaction will be:

1. Shielding primary and secondary neutrons and gammas.
2. Containing radioactive materials during normal operation.
3. Containing radioactive materials in case of an accident.
4. Storage of radioactive waste products.

To get some feeling for the magnitude of these problems, I would like to crudely compare a DT fusion reactor and a fission reactor.

For this evaluation, the tritium producing DT reactor blanket is assumed to be composed at 75% lithium, 5% niobium structure, and 20% void (by volume). A 1.5 meter thickness and lithium depleted to 4% Lithium-6 was found to give the desired breeding ratio of 1.3. Results of the blanket neutronic calculations follow:

TABLE 12

<u>Reactions:</u>	<u>Per 14.1 MeV Neutron</u>
Tritium breeding	1.31
Energy	17.1 MeV
$\text{Li}^7(n, n'\alpha)\text{T}$ reactions	0.704
$\text{Li}^6(n, \alpha)\text{T}$ reactions	0.602
$\text{Nb}^{93}(n, \gamma) \text{Nb}^{94}$ reactions	0.232
$\text{Li}^7(n, \gamma) \text{Li}^8$ reactions	$1.79 \times 10^{-4}$
$\text{Nb}^{93}(n, 2n) \text{Nb}^{92}$ reactions	0.125

The 14.1 MeV neutrons produced by the DT fusion reaction is the primary source of the radiological hazard.

These neutrons are the major components of the penetrating radiation that must be shielded. These neutrons also produce secondary gammas in

the blanket due to inelastic and capture reactions. These reactions in turn cause a contamination hazard by activation of the blanket materials.

1. Since about 80% of a DT reactor's power is born as fast neutrons and gammas compared to about 10% in a fission reactor, neutron and gamma shielding problems in a DT reactor should be more difficult than in a fission reactor.
2. During normal operation, tritium leakage should be the major contamination problem for a DT reactor. Morley and Kennedy in another Culham paper, "Fusion Reactors and Environmental Safety,"<sup>15</sup> compare dose rates from a tritium leakage rate of 0.1% ( $5 \times 10^5$  Ci/y/1000 MW<sub>e</sub>) of tritium burned to dose rates from the noble gas fission product krypton-85 (half life = 10.8y) which is currently discharged to the atmosphere. At a total power level of  $10^6$  MW<sub>e</sub> the tritium world dose rate at equilibrium is stated to be 0.12 mrem/y vs. 1.8 mrem/y for krypton 85.
3. In comparing the potential accidental contamination problems of fusion and fission, two factors are considered.
  - (1) Amount and kinds of radioisotopes present.
  - (2) Radiological tolerance to these radioisotopes.

I have estimated the radioisotope inventory based on a 2000 MW(th) plant and the blanket reactions shown in Table 12. The tritium inventory in a 2000 MW fusion plant has been estimated<sup>16</sup> at  $\approx 1$  kg or  $9.48 \times 10^6$  curies. Luckily the by-product of the required tritium breeding neutron-lithium reactions is stable helium. Lithium-7 does undergo a neutron capture reaction but its probability is so low and lithium-8's half life is so short (0.085 sec) it is of no consequence.

The remaining blanket material, niobium-93 structure, undergoes both an n,2n reaction producing niobium-92 (half life 10.1 days) and a capture reaction producing two isomeric states of niobium-94 (half lives 6.6 min and  $2 \times 10^4$  years).



Inventory calculations of these materials assume the short half life materials are in equilibrium and the long half life niobium-94 isomer has been building up for 1 year. Results are:

Material	Half Life	Inventory (Curies)
Lithium 8	0.0855	$3.27 \times 10^6$
Niobium <sup>92</sup>	10.1d	$2.0 \times 10^9$
Niobium <sup>94</sup>	$2 \times 10^4$ y	$1.13 \times 10^4$
Niobium <sup>94*</sup>	6.6 m	$3.43 \times 10^9$

For comparison a 2000 MW fast fission reactor would contain  $\approx 1000$  kg of plutonium<sup>17</sup> which is  $6.13 \times 10^4$  curies. To get a feeling for the relative hazard potential of the fusion reactor radioisotope inventory compared to the plutonium inventory of a fast fission reactor each are weighted by the reciprocal of their Continuous Maximum Permissible Concentration<sup>18</sup> (MPC's) in air.

$$\begin{aligned}
 \text{Relative Hazard} &= \frac{\frac{A(T)}{\text{MPC}(T)} + \frac{A(\text{Nb}^{92})}{\text{MPC}(\text{Nb}^{92})} + \frac{A(\text{Nb}^{94})}{\text{MPC}(\text{Nb}^{94})}}{\frac{A(\text{Pu239})}{\text{MPC}(\text{Pu239})}} \\
 &= \frac{4.7 \times 10^{12} + 4.5 \times 10^{12} + 1.1 \times 10^{12}}{1.0 \times 10^{17}} \\
 &= \frac{1.0 \times 10^{13}}{10^{17}} = 10^{-4}
 \end{aligned}$$

While the analysis was quite crude, I believe it gives a good indication that the relative biological hazard potential of the material contained in a fusion reactor is 3 to 4 orders of magnitude less hazardous than the plutonium in a fission reactor.

4. To compare the relative difficulty of long term storage of radioactive waste products both the chemical-physical problems of containment and hazard potential of the materials stored must be compared. I have estimated the latter by comparing niobium activated in a DT reactor

blanket to the fission products of a fission reactor in the same manner as above. Results indicate the hazard potential of the fusion blanket waste products to be  $\approx$  20 times less than fission products.

Even though the analysis is crude and the use of MPC values are questionable, I think the results are useful in getting a feeling for the relative difficulty of dealing with the radiological problems.

Let me now sum up what I think are the radiological advantages of a DT fusion system:

1. The basic DT reaction does not produce radioactive waste products.
2. The necessary tritium breeding reactions do not produce radioactive waste products.
3. Tritium is less hazardous than plutonium.
4. Tritium should be easier to contain than the noble fission gases.
5. Blanket activation can be controlled by proper choice of materials while one is stuck with fission products.
6. DT fusion can lead the way to even cleaner cycles such as  $D He^3$  fusion.
7. Fusion reactors can not have a nuclear excursion.

## CONCLUSION

Blanket neutronic calculations to date show that attractive tritium breeding is possible making the DT fusion cycle an attractive choice. In my opinion the simplest blanket, an all lithium blanket, is also the best. Lidsky's proposal of a fusion-fission system should be given serious consideration in the hope that fission breeder development can be skipped and the day of fusion hastened.

While we have made a good start, much more fundamental neutronics work remains to be done, for example:

1. Cross section measurement, theory, evaluation and library development is needed in neutronics and photonics.
2. Highly efficient shielding methods need to be developed to limit neutron and gamma heating problems in the superconducting magnets required to contain the plasma.
3. Rigorous methods must be developed and used to analyse and deal with the radiological hazards involved.
4. Neutron damage to materials subjected to 14 MeV neutrons must be understood and dealt with efficiently.
5. Neutronics must be integrated into an overall system analysis procedure dealing with heat transfer, structure, and costs, etc.

I hope this presentation has been informative. For those of you who wish to pursue the subject further, the Culham Conference papers<sup>19</sup> would make excellent reference material.

## REFERENCES

- 1 J. R. Stehn, "Neutron Cross Sections," Report BNL-325, Vol. 1, Supplement 2, Brookhaven National Laboratory, May 1964.
- 2 A. S. Bishop, Project Sherwood, Addison-Wesley Publishing Co., Reading, Massachusetts, 1958.
- 3 S. Glasstone and R. H. Lovberg, Controlled Thermonuclear Reactions, D. Van Nostrand Co., Princeton, New Jersey, 1960.
- 4 D. J. Rose and M. L. Clark, Plasmas and Controlled Fusion, The M.I.T. Press, Massachusetts Institute of Technology, Cambridge Massachusetts, 1961.
- 5 W. B. Myers, et al, "Tritium Regeneration in a D-T Thermonuclear Reactor Blanket," UCID-4480, Lawrence Radiation Laboratory, Livermore, Calif., 1962.
- 6 A. J. Impink, Jr., "Neutron Economy in Fusion Reactor Blanket Assemblies," Technical Report 434, Massachusetts Institute of Technology, Cambridge, Massachusetts, 1965.
- 7 W. G. Homeyer, "Thermal and Chemical Aspects of the Thermonuclear Blanket Problem," Technical Report 435, Massachusetts Institute of Technology, Cambridge, Massachusetts, 1965.
- 8 G. I. Bell, "Neutron Blanket Calculations for Thermonuclear Reactors," Los Alamos Scientific Laboratory, Los Alamos, New Mexico, LA-3385-MS, 1965.
- 9 J. D. Lee, "Tritium Breeding and Energy Generation in Liquid Lithium Blankets," UCRL-71757, Lawrence Radiation Laboratory, Livermore, California, 1969.
- 10 E. Plachaty, "SORS Monte Carlo Neutron Transport Code for the CDC 6600," Report UCRL-50532, Lawrence Radiation Laboratory, Livermore, Calif., 1968.
- 11 W. Engle, "A Users Manual for ANISN," Report K-1693, Union Carbide Corp. Nuclear Division, Oak Ridge, Tennessee, 1967.
- 12 R. W. Werner, et al, "Preliminary Design Considerations for an Astron Power Reactor System," UCRL-71768, Lawrence Radiation Laboratory, Livermore, Calif., 1969.
- 13 D. Steiner, "Neutronic Behavior of Two Fusion Reactor Blanket Designs," Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1969.

## REFERENCES (con't)

- 14 L. M. Lidsky, "Fission-Fusion Symbiosis: General Considerations and A Specific Example," Research Laboratory of Electronics & Dept. of Nuclear Engr., Massachusetts Institute of Technology, Cambridge, Mass., 1969.
- 15 F. Morley & M. W. Kennedy, "Fusion Reactors and Environmental Safety," Radiological Protection Division, UKAEA Health and Safety Branch, England, 1969.
- 16 E. F. Johnson, "Overall Tritium Balances in Fusion Reactors," Princeton University, Princeton, New Jersey, 1969.
- 17 Etherington, "Nuclear Engineering Handbook," McGraw-Hill, 1958 p 12-23.
- 18 Report of Committee II Permissible Dose for Internal Radiation, 1959.
- 19 Nuclear Fusion Reactor Conference, September 17-19, 1969, UKAEA, Culham Laboratory, Abingdon, Berkshire, England.

A THERMAL ENERGY CONVERSION  
SYSTEM FOR A FUSION REACTOR\*

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ABSTRACT

The blanket of a fusion reactor performs several functions. The primary function is the conversion of neutron kinetic energy to thermal energy and the transfer of this energy to a thermal power plant. Another important function, unique to the deuterium-tritium (DT) fuel cycle, is the regeneration of tritium at a rate that exceeds its consumption in the fusion reaction.

An analysis is presented which describes a self consistent, workable blanket for a DT fusion reactor. The very high neutron energy flux at the first wall is effectively distributed throughout a blanket constructed of niobium, using lithium as a moderator-coolant and sodium heat pipes to transfer heat radially. The lithium also provides adequate tritium breeding. The blanket is made up of modules containing small diameter tubes which must be designed to withstand only small internal pressures. The heavily loaded vacuum wall is external to the blanket, in a low temperature, neutron-free environment.

The analysis includes heat transfer calculations for the first wall, the heat pipes, and the lithium coolant. Fluid flow calculations include consideration of magnetically induced pressure gradients.

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## INTRODUCTION

As with any power plant, the production of useful power from a fusion reactor plant requires conversion of the energy of the reaction products into electricity. In general, the reaction products of a fusion reactor are energetic charged particles and energetic neutrons. The fusion source also emits considerable electromagnetic radiation.

Several fuel cycles are under consideration for fusion reactors, and the fraction of reaction energy present in each of the forms is a function of the particular fuel mix and operating conditions. Reaction energy in the form of neutron kinetic energy is converted by a neutron moderator into thermal energy for use in a thermal power cycle. Charged particle energy may also be converted into thermal energy, but direct conversion is an attractive alternative. Direct conversion of the energetic charged particles would be accomplished through electrostatic deceleration and subsequent collection of the particles on a set of electrodes. Such a system has been proposed by Post<sup>1</sup> and has been treated from an engineering standpoint by Werner.<sup>2</sup> This paper will consider the design of a thermal conversion system for the recovery of fusion neutron energy.

A particular fuel cycle and a particular set of operating conditions which are believed to be representative of a typical fusion reactor are chosen. Most of the discussion is devoted to the fluid mechanical and thermal aspects of the conversion system, but structural aspects are also considered. It will be seen that some special problems arise in the design of the conversion system. It is believed that these problems are all amenable to solution and that a workable thermal conversion system for a fusion reactor is indeed possible.

## FUSION FUEL CYCLES

The primary reactions which are considered for a fusion reactor are shown in Table 1. The numbers in parentheses are the kinetic energies, in MeV, of the reaction products.

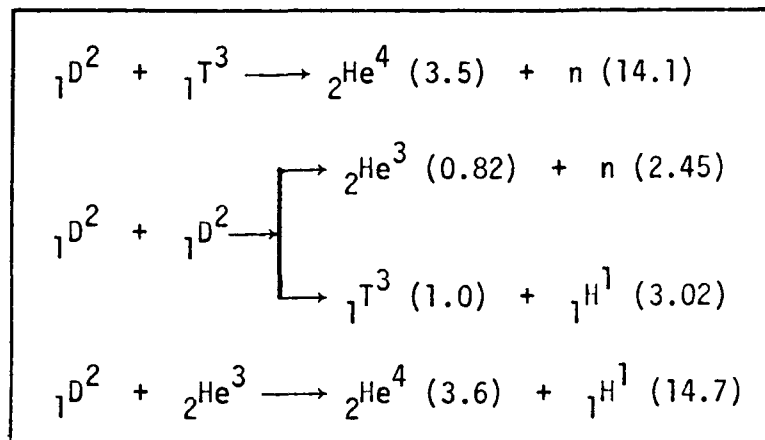


Table 1 - Primary Fusion Reactions

The deuterium-helium reaction (DHe<sup>3</sup>) produces only charged particles and, therefore, is an attractive candidate for use with only a direct energy conversion system. The deuterium-deuterium cycle (DD) produces neutrons carrying 34% of the total reaction energy and charged particles carrying the other 66%. (The two DD reactions occur with almost equal probability). The deuterium-tritium cycle (DT) has 80% of its reaction energy available as 14.1 MeV neutrons. DD and DT reactors may employ both direct and thermal energy conversion systems. Because of the 80% neutron energy for DT, however, it appears that thermal conversion systems are essential for DT reactors.

In order to obtain net electrical power from any of these fuel cycles it is necessary that the recoverable fusion power exceed the power losses. Power losses are of two types, direct and indirect. Direct losses are those associated with the escape of either radiation or reactant particles from the plasma. Indirect losses are those associated with operation of auxiliary equipment such as magnets and particle injectors.

Power balance against direct losses is primarily a question of the rates of energy gain and loss for each unit volume of the reacting plasma. The rates of the fusion reactions are strong functions of the mean energy of the reactants. Figure 1 shows fusion reaction cross sections (proportional to reaction rate) as a function of the energy of deuterons striking stationary targets. Observe that the DT reaction has the largest



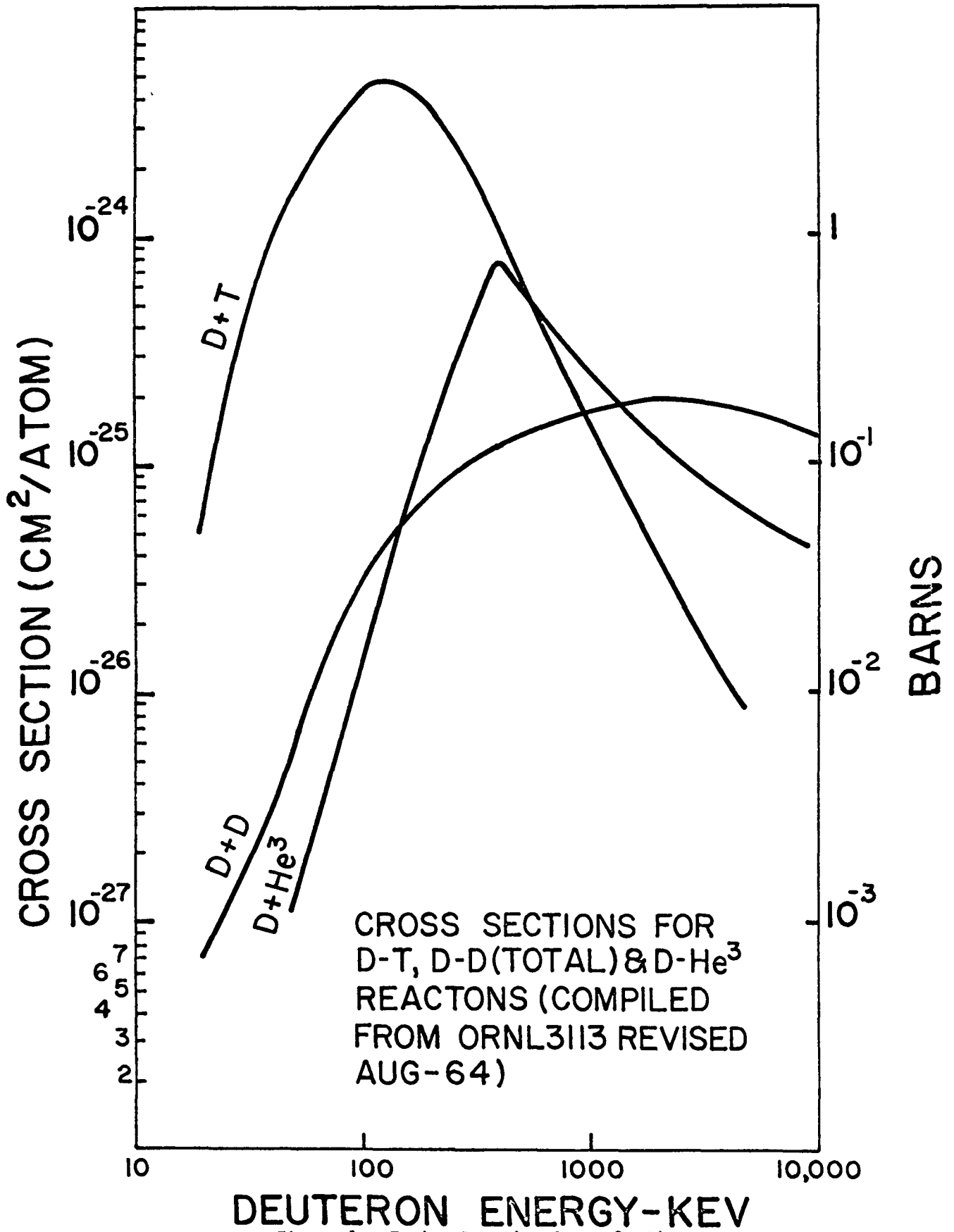


Figure 1 - Fusion Reaction Cross Sections

cross section for energies below about 500 keV. For energies below 100 keV the DT cross section is more than two orders of magnitude larger than that for DD or DHe<sup>3</sup>. Studies have shown that a DT reactor can produce net power at reactant energy levels substantially below those required for other fusion cycles.<sup>3</sup>

### PARTICULAR SYSTEM

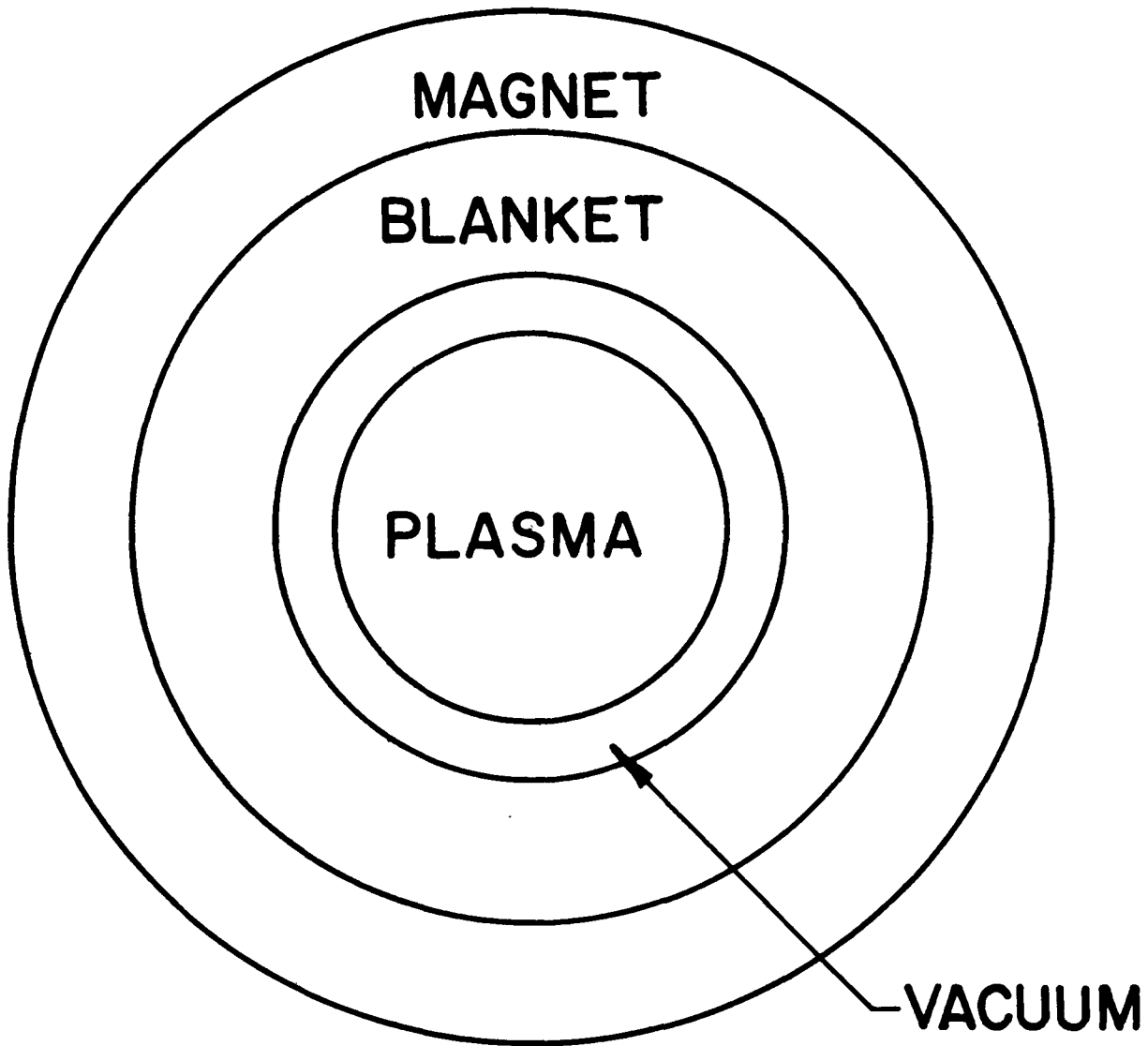
A DT fusion reactor was chosen for this study for two reasons. First, DT absolutely requires a thermal energy conversion system because 80% of the reaction energy appears as energetic neutrons. The second reason is that the first fusion reactors to be built will probably operate on the DT cycle because of its lower required reactant energy.

The assumed plasma parameters are shown in Table 2. It is believed that these parameters are appropriate for a DT fusion reactor. The lower electron temperature is typical of the mirror type containment system.

Mean plasma ion energy, $\bar{W}_i = 90$ keV
Ion temperature, $T_i = (2/3) \bar{W}_i = 60$ keV
Electron temperature, $T_e = 20$ keV
Plasma radius = 1.0 m
Plasma ion density, $n_i = 1 \times 10^{15}$ ions/cm <sup>3</sup>
Plasma neutron power density = 100 w/cm <sup>3</sup>

Table 2 - Plasma Parameters

The plasma is assumed to be magnetically contained in a cylindrical volume inside the coils of a superconducting magnet. Moderation of the fusion reaction neutrons takes place in a region called the blanket, a cylindrical annulus placed between the plasma and the magnet. The applied magnetic field is assumed to be axial and about 70 kgauss in strength. The physical arrangement of the plasma, blanket and magnet are shown in Figure 2.



## FUSION REACTOR CROSS SECTION

Figure 2

The radii of the blanket are 2.2 m inside and 4.1 m outside. The inside radius is chosen such that power flux at the first wall is not excessive. The thickness of the blanket is determined by the requirement to reduce the neutron energy to a negligible fraction of 14 MeV. The blanket is assumed to consist of several (5 or more) axial units of length 5 m each. The units are structurally independent, but coolant tubes are assumed to make a single axial pass through all of the units. We shall design a blanket unit to convert fusion neutron power into 2000 Mw of blanket thermal power.

### FUNCTIONS OF THE BLANKET

Before we consider the details of blanket design let us consider the several important functions of the blanket. The primary function of any fusion reactor blanket is the conversion of the neutron kinetic energy to thermal energy and the transfer of this energy to a thermal power plant. To perform this function the blanket must contain a fast neutron moderator material and a fluid coolant.

An equally important blanket function, unique to the DT cycle, is the regeneration of tritium by nuclear reaction at a rate that exceeds its consumption in the fusion reaction. Tritium breeding is necessary because the isotope does not appear in nature. Tritium is radioactive and beta decays to helium-3 with a 12.3 year half life. Both naturally occurring isotopes of lithium, lithium-6 (7.4%) and lithium-7 (92.6%), react with energetic neutrons to produce tritium. The reactions are shown in Table 3. Thus, it appears that the blanket must contain lithium in order to accomplish the tritium breeding function.

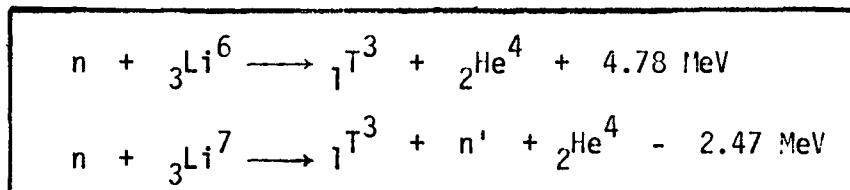


Table 3 - Tritium Breeding Reactions

The third blanket function is structural: containment of the various blanket materials and exclusion of the atmosphere from the blanket and plasma regions. The structural material must have good neutron economy and an acceptably low rate of radiation damage.

#### DESCRIPTION OF A PARTICULAR BLANKET

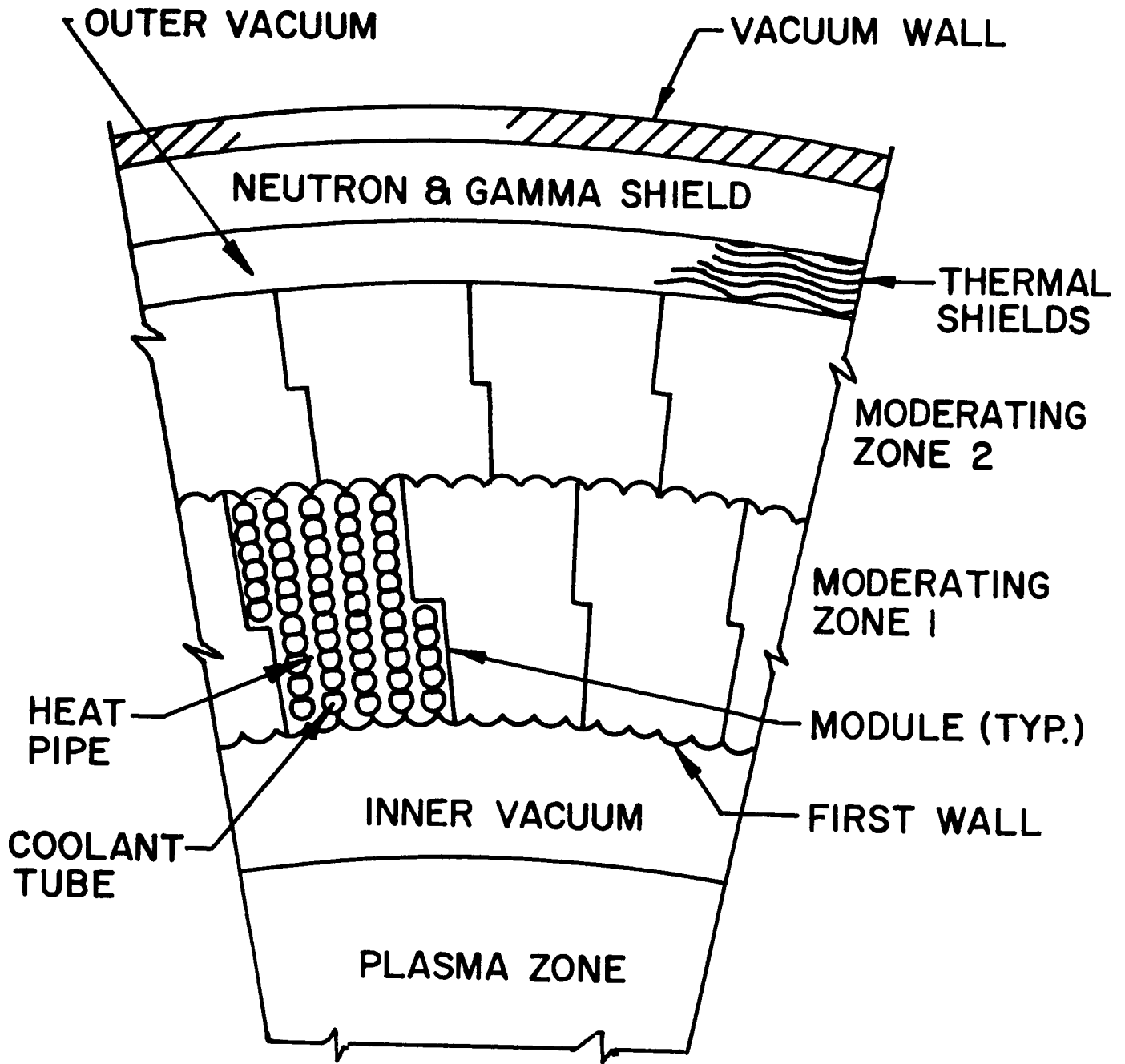
The blanket for this study uses liquid natural lithium both as the primary neutron moderator and as the blanket coolant fluid. The lithium also provides the required tritium breeding. The blanket structural material is taken to be niobium. The coolant temperature is assumed to increase from 1000°K at the blanket entrance to 1150°K at the exit.

A cross section through the blanket is shown in Figure 3. This blanket is essentially the one considered by Werner at the Culham Fusion Conference.<sup>4</sup> The blanket is made up of heat transfer modules that form an interlocking structure. Radially, the blanket is divided into a number of moderating zones. In this case, two zones are assumed; each with a radial depth of about 1 m. It will be seen that the bulk of the neutron energy is deposited in the first zone, and thus the more severe heat transfer problems occur there. Circumferential division is primarily for fabrication and assembly convenience.

Note that the first wall of the blanket is not a "vacuum wall" as it is in some fusion blanket concepts. In this blanket the highly loaded vacuum wall is moved outside the blanket where it enjoys several advantages:

1. It exists in a region of essentially zero neutron flux and therefore is not subject to radiation damage and neutron heating.
2. The temperature is low and the problems of structural creep are eliminated.
3. The wall material and the maximum thickness of the wall are no longer influenced by requirements for neutron economy. The wall can be as thick as necessary for structural rigidity and constructed of a common material such as stainless steel.

# MAGNET ZONE



# BLANKET CROSS SECTION

Figure 3

The blanket proper contains two primary elements: axially directed tubes for the flowing lithium coolant and radially directed heat pipes for radial heat transport. We choose sodium as the appropriate heat pipe working fluid. In sections to follow, we shall consider the thermal and fluid mechanical operation of these two elements in some detail.

### THE FIRST WALL

The fusion reaction takes place inside the volume enclosed by the first wall. The energetic neutrons to be moderated in the blanket must pass through the first wall, causing neutron heating. Radiation from the plasma also heats the first wall. We must ensure that the total heat load on the first wall is not excessive.

#### Neutrons

The neutron energy flux at the first wall may be calculated from the plasma power density and the dimensions of the system. In this case, the energy flux is  $2.20 \text{ kw/cm}^2$ . Note that this neutron flux results in a neutron power less than the 2000 Mw for which we are designing the blanket unit:

$$\begin{aligned} \text{Neutron power} &= (2.20 \times 10^7 \frac{\text{W}}{\text{m}^2}) (2\pi (2.2) (5.0) \text{ m}^2) \\ &= 1520 \text{ Mw} \end{aligned}$$

The additional 480 Mw arises from nuclear reactions in the blanket itself.

The energy lost by the neutrons traversing the first wall has been calculated using SORS-N, a Monte Carlo neutron-transport code.<sup>5</sup> For an 0.1 cm thick niobium first wall the neutron energy loss is 0.17 MeV. The energy loss is very nearly proportional to wall thickness for walls thinner than 0.1 cm.

The actual required thickness of the first wall may be calculated based on allowable stress. Recall from Figure 3 that the first wall is

actually made up of the end caps of many heat pipes, and thus the stress in the wall is determined by the vapor pressure in the pipe. Assuming a cylindrical end cap of radius 3 cm, a vapor pressure of about one atmosphere (sodium at 1150°K) and an allowable stress of 5000 psi results in a first wall thickness of less than 0.01 cm.

To be conservative from a stress standpoint and realistic from a fabrication standpoint, we shall take the first wall thickness to be 0.1 cm, and thus absorb 0.17 MeV per incident neutron. This results in a neutron heating load of 26 w/cm<sup>2</sup> for the first wall.

### Charged Particles

It should be mentioned that the charged particles of the fusion reaction, which are carrying 20% of the reaction energy or 380 Mw per blanket unit, are prevented from impinging on the first wall by the magnetic containment system. This is fortunate because impingement of the charged particles on the first wall would contribute an unacceptable heating load of 550 w/cm<sup>2</sup>. The charged particles leak from the system on magnetic field lines, and may be dealt with outside the reactor by another thermal conversion system, or perhaps a direct converter.

### Radiation

If the fusion plasma were a black body throughout the spectrum, it would radiate according to the Stefan-Boltzman law at a rate:

$$q_R = 5.67 \times 10^{-12} T^4 \text{ w/cm}^2$$

At  $T = 60 \text{ KeV}$  ( $7 \times 10^8 \text{°K}$ ),  $q_R = 1.4 \times 10^{24} \text{ w/cm}^2$ . Fortunately, this stupendous power flux does not occur because the plasma is not a black body. The fusion plasma does emit two types of radiation which must be considered in the heating of the first wall. These are bremsstrahlung radiation and electron synchrotron radiation.



Bremsstrahlung occurs when an electron passes close to a nucleus and is accelerated by the nuclear charge. In the fusion reactor, the bremsstrahlung lies principally in the x-ray region of the spectrum. The bremsstrahlung power density in the plasma has been calculated by Glasstone and Lovberg<sup>6</sup> and may be written:

$$P_B = 2.14 \times 10^{-30} n_D n_T T_e^{1/2} \text{ w/cm}^3$$

where  $T_e$  is the electron temperatures in KeV and  $n_D$  and  $n_T$  are the number densities of deuterium and tritium, respectively, in ions/cm<sup>3</sup>. For the present case, the bremsstrahlung energy flux on the first wall is:

$$\begin{aligned} q_B &= (2.14 \times 10^{-30}) (0.5 \times 10^{15})^2 (20)^{1/2} \frac{(100.0)^2}{2(220.0)} \\ &= 54 \text{ w/cm}^2 \end{aligned}$$

Because of the x-ray frequency of bremsstrahlung, this total energy flux is absorbed by the first wall.

Electron synchrotron radiation emerges from the fusion plasma due to electrons gyrating in the applied magnetic field. Since this radiation lies in the infrared and far infrared regions of the spectrum, reflection by the first wall surface significantly reduces the heat load. The synchrotron radiation absorbed by the first wall is a function of the thickness of the plasma (important because reabsorption is significant), the electron temperature and number density, the magnetic field strength, and the coefficient of reflectivity of the wall. Mills<sup>7</sup> has shown that the absorbed synchrotron radiation is more than two orders of magnitude less than the bremsstrahlung heat load for the plasma parameters of this study and magnetic fields as strong as 75 kilogauss. The synchrotron radiation heat load on the first wall will therefore be neglected.

An additional heat load on the first wall may arise from gamma radiation backscattered from neutron-gamma reactions in the blanket. This effect has not been calculated for this blanket because SORS-N, the neutron transport code, does not consider the transport of gamma radiation. SORS-N assumes the energy of neutron-gamma reactions to be deposited at the site of the reaction. An investigation by Steiner<sup>8</sup> which included gamma transport concluded that backscattered gamma was not significant for a lithium cooled blanket. (Steiner did find significant first wall heating from backscattered gamma for a blanket cooled with lithium-beryllium fluoride (flibe).) Therefore, with some justification, we ignore backscattered gamma radiation.

### Total First Wall Heating

The total heat load on the 0.1 cm thick first wall is the sum of the neutron energy deposition and the bremsstrahlung radiation, a total of  $80 \text{ w/cm}^2$ . For niobium the temperature difference across the wall is  $15^\circ\text{K}$ . The  $80 \text{ w/cm}^2$  to be removed from the outside surface of the first wall is well within the capabilities of high performance liquid metal heat pipes. Heat pipe fluid will vaporize in the wick which covers the outside surface of the first wall, and the vapor will carry the heat radially outward. The maximum heat removal capability at such a wall is determined by the flux at which nucleate boiling occurs in the wick structure. For typical sodium heat pipes it appears that  $200 \text{ w/cm}^2$  is a reasonable limit.

### POWER DENSITY IN THE BLANKET

The neutron energy deposition in the blanket has been calculated using SORS-N, the same neutron transport code used to calculate energy deposition in the first wall. For this calculation, the blanket is assumed to be a radially uniform homogeneous mixture made up of its various component materials in proper proportions. The volume fractions for this blanket have been estimated to be 0.75 natural lithium, 0.05 niobium and 0.20 heat pipe void. Although the heat pipes contain a working fluid, in this case sodium, the amount is negligibly small from a neutronics standpoint.

Before we consider the radial variation of power density in the blanket let us look at some overall results of the neutronic calculations. The average total energy deposited per incident 14.1 MeV neutron is 18.6 MeV. It is this energy multiplication, due to various nuclear reactions in the blanket, which increases the 1520 Mw of neutron power per blanket unit to 2000 Mw of blanket thermal power. The energy multiplication can be further increased by increasing the fraction of niobium in the blanket or adding other materials which undergo neutron multiplication reactions (such as beryllium) or exothermic neutron capture reactions (such as sodium).

The tritium breeding reactions of lithium-6 and lithium-7 result in a tritium breeding ratio (tritons generated per triton used) of 1.66, more than sufficient to sustain the reaction if an effective recovery method can be devised. One approach investigated by Werner<sup>9</sup> indicates that tritium recovery should not be a problem. Recall that this blanket uses natural lithium. Although lithium-6 constitutes only 7.4% of natural lithium, it accounts for somewhat more than half of the tritium produced because of a higher average reaction cross section. Although it might seem that tritium breeding could be greatly increased by using lithium-6 enriched lithium, such is not the case because the lithium-6 reaction is a neutron capture reaction while the lithium-7 reaction is  $(n,n'\alpha)$ . Thus, replacing lithium-7 atoms with lithium-6 increases the probability of lithium-6 reactions, but decreases the probability of lithium-7 reactions followed by lithium-6 reactions, a process which yields two tritons per source neutron. It turns out that natural lithium is very nearly the optimum mix for maximum tritium breeding.

The various tradeoffs concerning tritium breeding and blanket energy generation have been considered in some detail by Lee.<sup>10</sup>

The average energy which leaks beyond the outside radius of the blanket per incident 14.1 MeV neutron is 0.0752 MeV. This amounts to a neutron power loss of 8 Mw per blanket unit, only 0.4% of the blanket thermal power. Although the power loss is trivial from an energy utilization

standpoint, it is necessary to deny its access to the superconducting magnet with neutron and gamma radiation absorption shields.

The radial variation of blanket power density is shown in Figure 4. The maximum power density is about 46 watts/cm<sup>3</sup>. Approximately 90% of the total thermal power appears in the first 100 cm of the blanket, which we have previously designated as the first radial moderating zone. The average power density in this zone is 21 watts/cm<sup>3</sup>. We shall deal only with the first zone in our consideration of blanket heat transfer and fluid flow.

### HEAT TRANSFER

The consideration of heat transfer in the first radial zone of the blanket may be separated conveniently into two elements. First is the operation of the heat pipes which transport heat radially outward from the region in the zone where the power density exceeds the average to the region deficient in power. Second is the transfer of heat into the axially flowing lithium coolant in the deficient power region and out of the coolant in the excess power region.

#### Heat Pipes

Heat pipes are self-contained thermal conductance devices which can transfer large quantities of heat as latent energy by evaporating a working fluid in a heating zone and condensing the vapor thus produced in a cooling zone. The condensate is returned to the heating zone by capillary pumping. The heat transfer is essentially isothermal. In the present application the heat pipes consist of radial passages between radial rows of axially directed coolant tubes. Capillary wicking covers the outside surfaces of the coolant tubes.

The purpose of the heat pipes is to redistribute the energy deposited in the blanket zone so that the coolant everywhere handles exactly the average power density (21 watts/cm<sup>3</sup>). As shown in Figure 4,

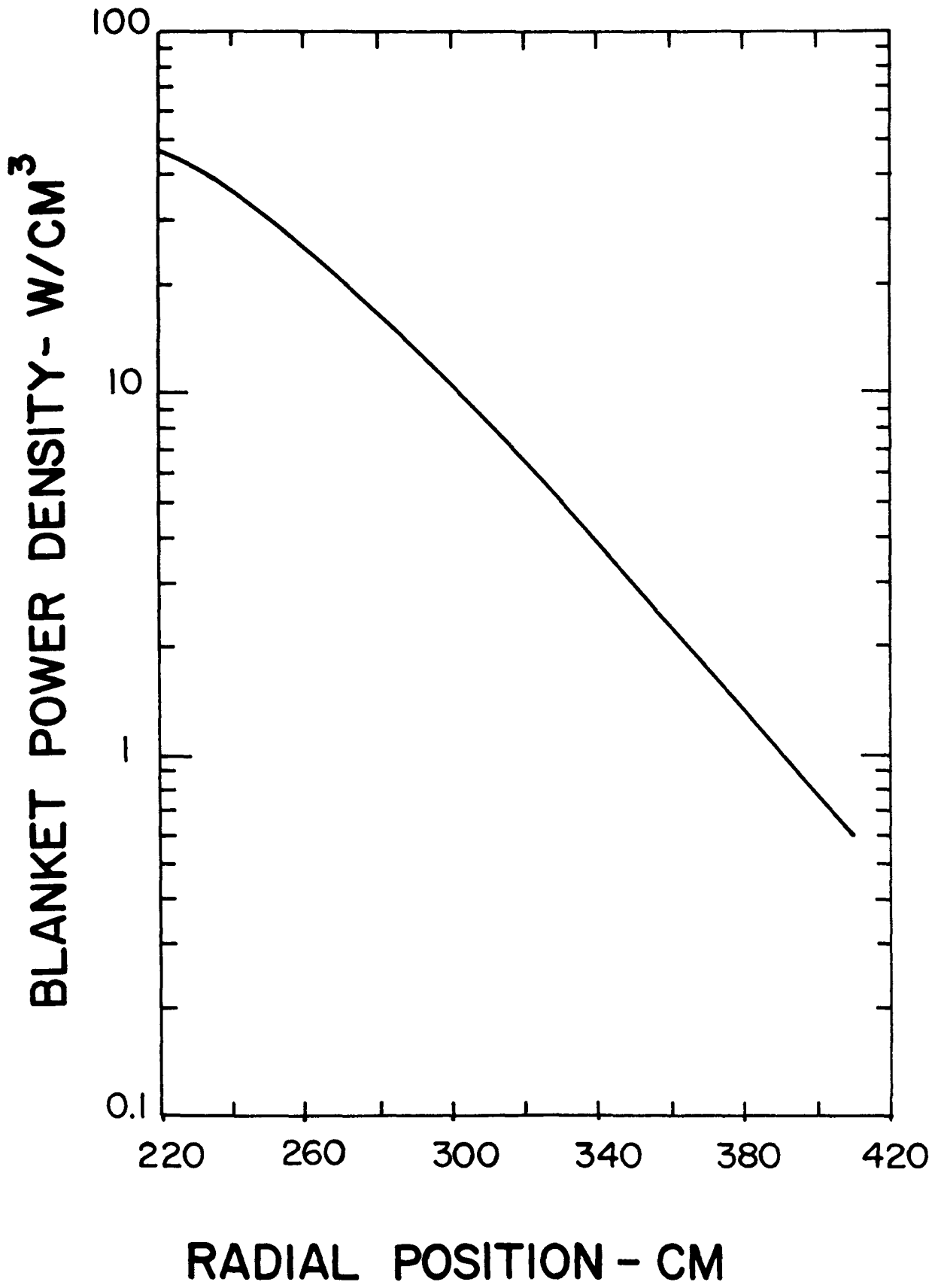


Figure 4 - Blanket Power Density

the power density of the blanket equals  $21 \text{ watts/cm}^3$  at about  $r = 268 \text{ cm}$ , or 48 cm from the first wall. The average power density in the first 48 cm of the blanket is  $31 \text{ watts/cm}^3$ ; so the "excess" blanket power to be transported radially outward is 368 Mw. If we also distribute the first wall heat load (55 Mw) uniformly throughout the first blanket zone the total heat to be transported radially outward past the  $r = 268 \text{ cm}$  station is increased to 400 Mw.

The peak axial heat flux in the heat pipes (radial direction in the blanket) is 400 Mw divided by the total heat pipe cross sectional area at  $r = 268 \text{ cm}$ . Estimating the flow area as 20% of the available circumferential area results in a peak axial heat flux of  $2.4 \text{ kw/cm}^2$ . This is within the capability of sodium heat pipes, at least in zero magnetic field applications. In the present application, however, the heat pipes are perpendicular to the magnetic containment field; so the electrically conducting heat pipe fluid is subjected to magnetically induced pressure gradients. This effect is discussed in the section on fluid flow.

### Lithium Coolant

The power handled per unit volume of lithium is the average power density divided by the lithium volume fraction, or  $21/0.75 = 28 \text{ w/cm}^3$ . The heat to be transferred between the lithium coolant and the heat pipe fluid depends on the difference between the local power deposition in lithium and the average power handled. Unfortunately, the exact power deposition in the lithium is unknown because the neutronics calculations assumed a homogeneous blanket. As an estimate, we take the local lithium power to be the blanket power density divided by 1.0 minus the void fraction. This results in a maximum lithium power density of  $58 \text{ w/cm}^3$  at the first wall and a minimum of  $6.5 \text{ w/cm}^3$  at the outside radius of the first zone. Thus, the greatest heat transfer to or from the lithium coolant occurs at the first coolant tubes, where the excess power deposition is  $58 - 28 = 30 \text{ w/cm}^3$ . We shall consider the heat transfer for these critical tubes.

An energy balance for a circular tube with uniform internal heat generation yields:

$$q_s = R q_G/2$$

where  $q_s$  is the surface heat flux,  $R$  is the tube radius, and  $q_G$  is the volumetric heat generation (in this case,  $q_G$  is the excess power deposition,  $30 \text{ w/cm}^3$ ). Since we must limit  $q_s$  to  $< 200 \text{ w/cm}^2$  to prevent nucleate boiling in the heat pipe wick, the maximum tube radius we can consider is  $R \approx 13 \text{ cm}$ . To be conservative we shall take  $R = 3 \text{ cm}$ .

The radial temperature profile inside the lithium tube depends upon the characteristics of the flow. Turbulent convection would result in the smallest temperature variation. Laminar flow, which may persist even at high Reynolds numbers because of the parallel magnetic field,<sup>11</sup> results in greater temperature variation. Here we calculate the worst case by assuming solid body conduction. The centerline to wall temperature difference is given by:

$$\Delta T = \frac{q_G R^2}{4k}$$

where  $k$  is the thermal conductivity of the lithium, approximately  $0.63 \text{ w/cm}^\circ\text{K}$ . For  $R = 3 \text{ cm}$ , the temperature difference is  $\Delta T \approx 110^\circ\text{K}$ . If the minimum operating pressure of the lithium is 1 atmosphere the boiling point is over  $1600^\circ\text{K}$ ; so the maximum lithium temperature,  $1150 + 110 = 1260^\circ\text{K}$ , is of no concern.

#### FLUID FLOW

Both the heat pipes and the lithium coolant tubes are inside the magnetic field region. Thus, in addition to normal flow losses, the electrically conducting fluids are subjected to magnetically induced pressure gradients.

## Heat Pipes

Fluid flow in a heat pipe consists of vapor flow in the core of the pipe and liquid flow in the capillary structure lining the inside of the pipe walls. The two flows are parallel but in opposite directions. The maximum fluid circulation in a heat pipe and the accompanying maximum heat transfer capability are calculated by equating the sum of the liquid and vapor flow pressure drops to the pressure rise across the capillary meniscus.

In the present application the vapor and liquid flows in the heat pipe are perpendicular to the applied magnetic field. The magnetic effect on the vapor flow is negligible because of the low electrical conductivity of the vapor. However, the magnetic effect on the liquid flow is very important, and must be considered in the design of the heat pipe. Electrical eddy currents flow in the liquid in a plane perpendicular to the fluid velocity, causing thinning of the side-wall boundary layers with accompanying increased viscous losses. These electrical currents are increased if the channel walls are electrically conducting, resulting in a magnetic body force in the liquid which opposes its motion.

Modification of the heat pipe equation to include the magnetic effect on liquid flow has been reported by Carlson and Hoffman.<sup>12</sup> The report lists three important steps for minimization of the magnetic effect:

1. The wick structure should have as small a wall electrical conductance as possible.
2. There is a great advantage to a compound wick structure with large liquid flow channels covered with a fine mesh or screen to provide the capillary pumping.
3. Maximum performance requires optimizing the liquid-vapor area apportionment in the heat pipe for the particular magnetic field to be encountered.



For a particular heat pipe with length 1 m and diameter 2 cm using sodium at 1000°K as the working fluid, the following maximum axial heat fluxes are calculated in Ref 12. The parameter C is the ratio of wall to fluid electrical conductance.

Zero magnetic field:  $q = 5.1 \text{ kw/cm}^2$

Magnetic field = 70 kilogauss :

$$C = 0 \quad q = 4.4 \text{ kw/cm}^2$$

$$C = 0.001 \quad q = 2.5$$

$$C = 0.01 \quad q = 0.8$$

It appears that such a heat pipe could satisfy the requirements of the present application ( $q = 2.4 \text{ kw/cm}^2$ ) if C can be maintained sufficiently close to zero. The nonconducting wall case,  $C = 0$ , is especially promising, but whether suitable insulators compatible with high temperature sodium can be found is not known.

For the non-zero C cases, if we neglect the effect of contact resistance, the conductance ratio may be written:

$$C = \frac{\sigma_w t_w}{\sigma_L a}$$

where  $\sigma_w$  and  $\sigma_L$  are the electrical conductivities of the wall and liquid, respectively, a is the half-width of the flow channel, and  $t_w$  is the effective channel wall thickness. At 1000°K the conductivities of niobium and liquid sodium are about equal; so C is approximately  $t_w/a$ . While  $C = 0.01$  may be possible for thin-walled niobium heat pipes with  $a \approx 1 \text{ cm}$ , it appears that  $C = 0.001$  is impossible.

Electrical contact resistance between the liquid metal and a metallic heat pipe wall would decrease the value of C. This effect has not been investigated. Another possible solution to the problem is a sandwich

construction of the heat pipe wall. A very thin metal layer (small  $t_w$ ) would be bonded to the primary metal structure with an intermediate oxide insulator.<sup>13</sup> If all else fails, it may be necessary to investigate the possibility of an electrically non-conducting working fluid for the heat pipes. Then, of course, the magnetic effect would be completely eliminated.

### Lithium Coolant

The lithium coolant makes a single axial pass through the blanket. Assuming 5 axial blanket units and a lithium temperature rise of 150°K yields a volumetric flow rate of 33 m<sup>3</sup>/s in the first radial zone. For the given cross section of the blanket the lithium flow velocity is a moderate 2.5 m/s. The Reynolds number for a tube with radius 3 cm is  $3 \times 10^5$ .

The flow of the lithium within the blanket is parallel to the magnetic field and therefore not subjected to the magnetic pressure drop considered for the heat pipe liquid flow. For normal turbulent flow in smooth walled pipes, the pressure drop of the lithium is only  $8.7 \times 10^3$  N/m<sup>2</sup>, or about 1.3psi. This pressure drop may be further reduced if the magnetic field is effective in laminarizing the flow. This effect has been reported by Globe<sup>11</sup> but only for Reynolds numbers up to 4000.

Although the lithium flow is parallel to the magnetic field within the blanket, it must cross the field as it enters and exits the blanket region. In these entry regions, the flow is subjected to two magnetic effects. First is the Hartmann effect: electrical eddy current flow in the liquid in a plane perpendicular to the fluid velocity. This is the same effect discussed previously in the section on heat pipe fluid flow. Second is the "end loop effect": electrical eddy current flow in a plane perpendicular to the magnetic field. These currents are caused by magnetic field gradients. Unfortunately, the two effects can be evaluated separately and the results superimposed only for the case of nonconducting walls. With conducting walls the two effects are coupled because both types of eddy currents flow through the walls.

For the case of nonconducting walls the entry region pressure drop due to the Hartmann effect is given by:

$$\Delta P_H = u \bar{B} \sqrt{\sigma \eta} \frac{\ell}{R}$$

where  $u$  is the flow velocity,  $\bar{B}$  is the average magnetic field strength,  $\sigma$  is the electrical conductivity,  $\eta$  is the viscosity,  $\ell$  is the entry length, and  $R$  is the tube radius. The entry region pressure drop due to the end loop effect is given by Shercliff<sup>14</sup> for the special case  $\ell = 0$  (step change in field strength):

$$\Delta P_E = 0.27 \sigma u B^2 R$$

where  $B$  is the strength of the magnetic field to be entered. The constant in this equation is a decreasing function of  $\ell/R$ . Hoffman<sup>15</sup> has calculated 0.16 for  $\ell/R = 1$  and 0.038 for  $\ell/R = 8$ .

The way in which the fluid should be brought into and out of the field is not obvious. An abrupt entry through a large diameter manifold at a position of high magnetic flux density would minimize the Hartmann effect. On the other hand, a gradual entry through small diameter tubes at a position of weak magnetic field gradients would minimize the end loop effect. Furthermore, the degree to which either of these approaches can be adopted is uncertain because a detailed description of the magnetic field is not available.

In order to calculate a "worse than real" example, we make the following assumptions: For the Hartmann effect assume  $R = 3$  cm and  $\ell = 3$  m. Then, for  $\bar{B} = 35$  kgauss,

$$\Delta P_H = 2.1 \times 10^4 \text{ N/m}^2 = 3.0 \text{ psi}$$

For the end loop effect assume  $R = 3$  cm and  $\ell/R = 8$ . Then, for  $B = 70$  kgauss,

$$\Delta P_E = 3.5 \times 10^5 \text{ N/m}^2 = 51 \text{ psi.}$$

Doubling the sum of  $\Delta P_H$  and  $\Delta P_e$  to account for both entry and exit and adding the normal viscous pressure drop (1 psi) yields a total lithium pressure drop of 110 psi. Such a pressure drop requires neither a high operating pressure nor excessive pumping power. The pumping power in this case would be 25 Mw, or only 0.3% of the 5 unit blanket thermal power.

The consideration of the lithium pressure losses in the magnetic field entry and exit regions for the case of conducting wall tubes is beyond the scope of this paper. It appears, however, that losses in this case may be excessive unless the conductance ratio  $C$  can be very small. Insulated walls may therefore be a necessity in the entry regions.

#### THERMAL POWER PLANT

This paper does not include any calculations for the thermal power plant facilities external to the fusion reactor blanket. However, the total system flow diagram is imagined to look like that shown in Figure 5. The binary, potassium-steam Rankine cycle shown with the temperatures indicated is considered to be within current technology. The binary cycle was chosen over a single fluid cycle to illustrate increased cycle efficiency. At the conditions cited the system yields a thermal efficiency of 50%. Details of the cycle analysis may be found in Ref. 16.

#### CLOSURE

The DT fusion reactor blanket discussed herein appears to be a workable system for the conversion of neutron kinetic energy into thermal energy. The very high neutron energy flux at the first wall is effectively distributed throughout a blanket constructed of niobium, using lithium as a moderator-coolant and sodium heat pipes to transfer heat radially. The lithium also provides tritium breeding adequate to sustain the reaction. The blanket is made up of modules containing small diameter tubes which must be designed to withstand only small internal pressures. The heavily loaded vacuum wall is external to the blanket, in a low temperature, neutron-free environment. While the heat transfer

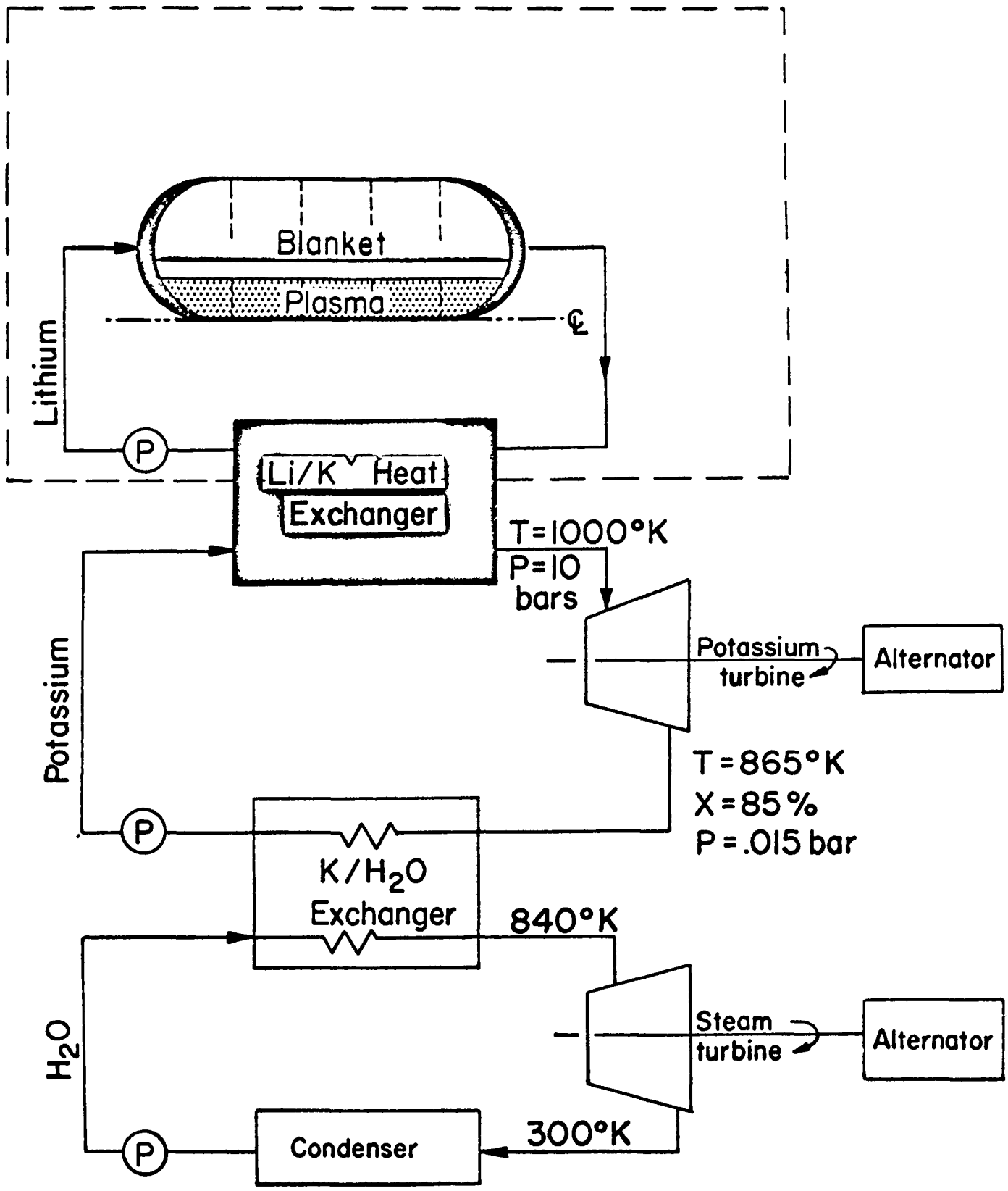


Figure 5 - Power Plant Flow Diagram

and fluid flow calculations presented describe a self consistent, workable system, no claim is made as to optimum design.

One problem which requires much more investigation is the apparent necessity for the heat pipes and the lithium coolant inlet and outlet tubes to have walls which are electrically nonconducting, or at least of very low conductance. Additional analyses and experiments are required to determine if this apparent necessity is real, and material compatibility studies will be required to determine if the nonconducting wall is feasible for high temperature liquid metal flows.

## REFERENCES

- 1 R. F. Post, "Mirror Systems: Fuel Cycles, Loss Reduction and Energy Recovery," Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, England, 1969.
- 2 R. W. Werner, "Mirror Machines and Direct Conversion," UCRL-72487 Lawrence Radiation Laboratory, 1970.
- 3 R. F. Post, "Critical Conditions for Self-Sustaining Reactors in the Mirror Machine," Nuclear Fusion, 1962 Supplement, Part 1, p. 99.
- 4 R. W. Werner, "The Module Approach to Blanket Design - A Vacuum Wall Free Blanket Using Heat Pipes," Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, England, 1969.
- 5 E. Plachaty, "SORS Monte Carlo Neutron Transport Code for the CDC 6600," UCRL-50532, Lawrence Radiation Laboratory, Livermore, California, 1968.
- 6 S. Glasstone and R. Lovberg, Controlled Thermonuclear Reactions, D. Van Nostrand Co., Inc, Princeton, New Jersey, 1960.
- 7 R. G. Mills, "Synchrotron Radiation from Fusion Reactors," MATT-658, Plasma Physics Laboratory, Princeton University, New Jersey, 1969.
- 8 D. Steiner, "Neutronic Behavior of Two Fusion Reactor Blanket Designs," Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, England, 1969.
- 9 R. W. Werner, "Heat Pipes for Recovery of Tritium in Thermonuclear Reactor Blankets," UCRL-72329, Lawrence Radiation Laboratory, Livermore, California, 1970.
- 10 J. D. Lee, "Some Neutronic Aspects of DT Fusion Reactors," UCRL-72493 Lawrence Radiation Laboratory, Livermore, California, 1970.
- 11 S. Globe, "The Effect of a Longitudinal Magnetic Field on Pipe Flow of Mercury," ASME Trans. J. of Heat Transfer, Nov., 1961, p 445.
- 12 G. Carlson and M. Hoffman, "Effect of Magnetic Fields on Heat Pipes," UCRL-72060, Lawrence Radiation Laboratory, Livermore, California, 1969.
- 13 W. M. Wells, "Experiments and Calculations on the Feasibility of Pumping Liquid Lithium in a Thermonuclear Reactor Blanket," UCRL-50544, Lawrence Radiation Laboratory, Livermore, California, 1969.
- 14 J. A. Shercliff, "Edge Effects in Electromagnetic Flowmeters," J. Nuclear Energy I, 1956, Vol 3, p 305.
- 15 M. Hoffman, private communication.
- 16 P. B. Mohr, "Fusion Reactor Power Conversion Studies," Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, England, 1969.

# MIRROR MACHINES AND DIRECT CONVERSION\*

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## ABSTRACT

Direct conversion of fusion reactor charged particle energy to electrical form is discussed. The method of conversion, using a series of electrostatically focused collector electrodes permits efficiencies in excess of 90%. The source of charged particles are the "classical losses" in mirror machines, i.e. the loss of charged particles into the escape cone of the mirrors.

The concept is one proposed by Post of the Lawrence Radiation Laboratory.

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## INTRODUCTION

In the normal context of the definition, "direct conversion" is concerned with those processes by which energy in its various forms is converted directly to electricity without the necessity of an intermediate step, or more accurately, using fewer intermediate steps than conventional converters. Direct conversion also implies the elimination of moving parts. Different forms of direct converters come readily to mind: thermionic generators, thermoelectric generators, semiconductors, magnetohydrodynamics, etc. The development of these converters to date has been such that few of them have achieved or even have the potential of achieving efficiencies that approach the more conventional, traditional methods of conversion such as a Rankine or Brayton cycle. Since traditional systems themselves rarely achieve efficiencies as high as 50% and more commonly average out at about 33% because of heat engine limits, this is a serious shortcoming for direct converters as a class of mechanisms. Nevertheless, they hold a strong position in current technology because of their unique characteristics and special applications, particularly in space technology.

To this list of direct converters we would now like to add a direct converter for fusion reactors which need not have the limits of efficiency that we associate with the other converters. This is one proposed by Dick Post of the Lawrence Radiation Laboratory.<sup>1</sup> In the direct converter as it relates to fusion reactors and to mirror machines in particular, it is the conversion of kinetic energy of the charged particles of the reaction products with which we will be concerned and it will be shown that in principle, and in contrast to the other converters where efficiencies are limited, direct conversion of plasma energy to electrical form can be done at exceedingly high efficiency and that this efficiency can be in excess of 90%. That this is possible is based on the fact that the usual Carnot limit as it applies to mirror machines and charged particles does not control efficiency and on the added fact that the particle flux escaping from a mirror system does so in a thermodynamically orderly way. Carnot limits state that no heat engine can be more efficient than the efficiency defined by the ratio of the difference in the

maximum temperature of the working substance minus the rejection temperature divided by the maximum temperature. In our case, working fluid temperatures, the charged particle kinetic energy, are millions of degrees Kelvin and rejection temperatures are only hundreds of degrees Kelvin so that efficiency by Carnot standards is effectively 100%. The relation between energy and temperature is that  $1 \text{ ev} \approx 1 \times 10^4 \text{ }^\circ\text{K}$ . The thermodynamically orderly escape of charged particles is due to the fact that they follow magnetic field lines. Before the principles of this direct converter can be discussed, it is necessary to consider the fuel cycles available to then provide some brief background on mirror machines and to review what are termed "classical losses" in reactors of this type (which will be seen to be the source of the charged particle energy). As a preamble, a comment on "Why the interest in direct conversion" is in order.

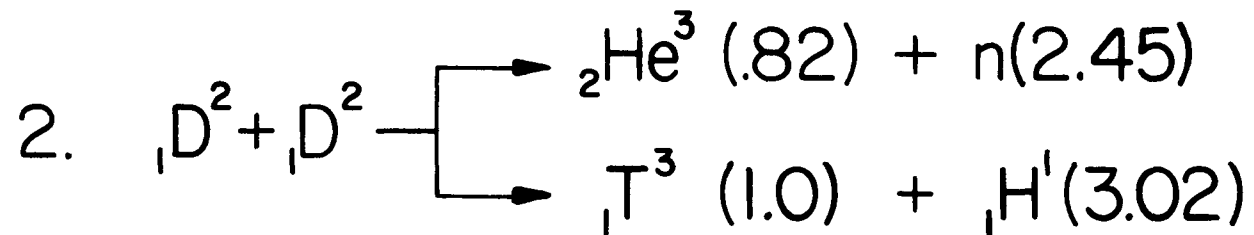
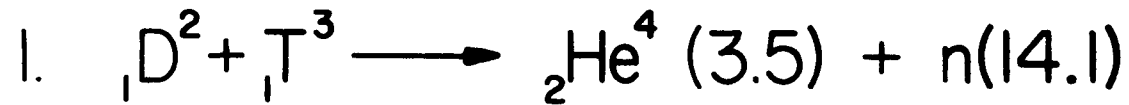
#### WHY DIRECT CONVERSION FOR MIRROR MACHINES

It is a reasonable question to ask why, aside from the engineering advantages already mentioned, one should be concerned with direct conversion for fusion at a time when a stable plasma has not yet been established. It seems that there are a number of compelling reasons for this interest. First, it has long been recognized that "classical losses" in mirror machines, i.e. the loss of charged particles into the escape cone of the mirrors, produce a rather tight race between nuclear power production and collisional loss. The reaction mean free path in a system containing say  $10^{14}$  or  $10^{15}$  particles/cm<sup>3</sup> has a value like 6000 miles. This is an astronomical number compared to the container size and quite evidently many reflections must occur in a mirror machine before a reaction takes place, and ample opportunity is afforded for small angle scattering to place a particle in the loss cone. The loss into the mirrors determines directly the power that must be injected. To relate nuclear power produced to power injected a figure of merit has been introduced and is called "Q". The value of "Q" quite clearly must be greater than one to produce net power output and must be several times one to produce net power output economically. If direct conversion can be utilized in

mirror machines at high efficiency and the energy recovered and fed back for reinjection then the effective value of "Q" will rise markedly. Direct conversion also allows consideration of fuel cycles such as D-He<sup>3</sup> where all the energy is in the form of charged particles. For these two reasons direct conversion is not only of interest in its own right but must influence present plasma investigations and direction of research.

### FUSION REACTIONS

The principal fusion reactions are shown in Figure 1. Generally speaking, the reaction or the plasma considered with the greatest emphasis at this time at this state of development in plasma physics, is the one involving deuterium and tritium. This is because that particular reaction has the largest cross section and by that yardstick is probably the least formidable to attain. By attainment is meant not merely the production of fusion neutrons which has been done but the sustaining of a viable plasma from which a positive power balance can be achieved. The numbers in parentheses are the energies in MeV of the fusion reaction products. For example, in the D+T reaction, the energy that is produced takes two forms, three and one half MeV is in the form of charged particle energy and 14.1 MeV is evidenced as a high energy neutron. The excess energy, of course, is due to the mass deficiency of the reaction products. The utilization of the neutron energy and the conversion of its kinetic energy to thermal energy has been discussed elsewhere. From a direct conversion standpoint, our interests center on the charged particles in the reaction products. The reaction in which all of the energy released is in the form of charged particles is the D-He<sup>3</sup> reaction. The energy released is quite large, 3.6 MeV as He<sup>4</sup> nuclei and 14.7 MeV as protons. It is this reaction to which we will principally direct our attention. The other two reactions, the D-T and the D-D, remain of strong interest for direct conversion influenced, of course, by the proportion of charged particle energy to total energy that each produces and also by the probability of successful attainment of plasmas of that particular composition. The latter two reactions would have to be used in conjunction with an appropriate thermal cycle to recover the neutron energy. The cross sections for these three reactions are shown in Figure 2. Notice that



THE PRINCIPAL FUSION REACTIONS

Figure 1

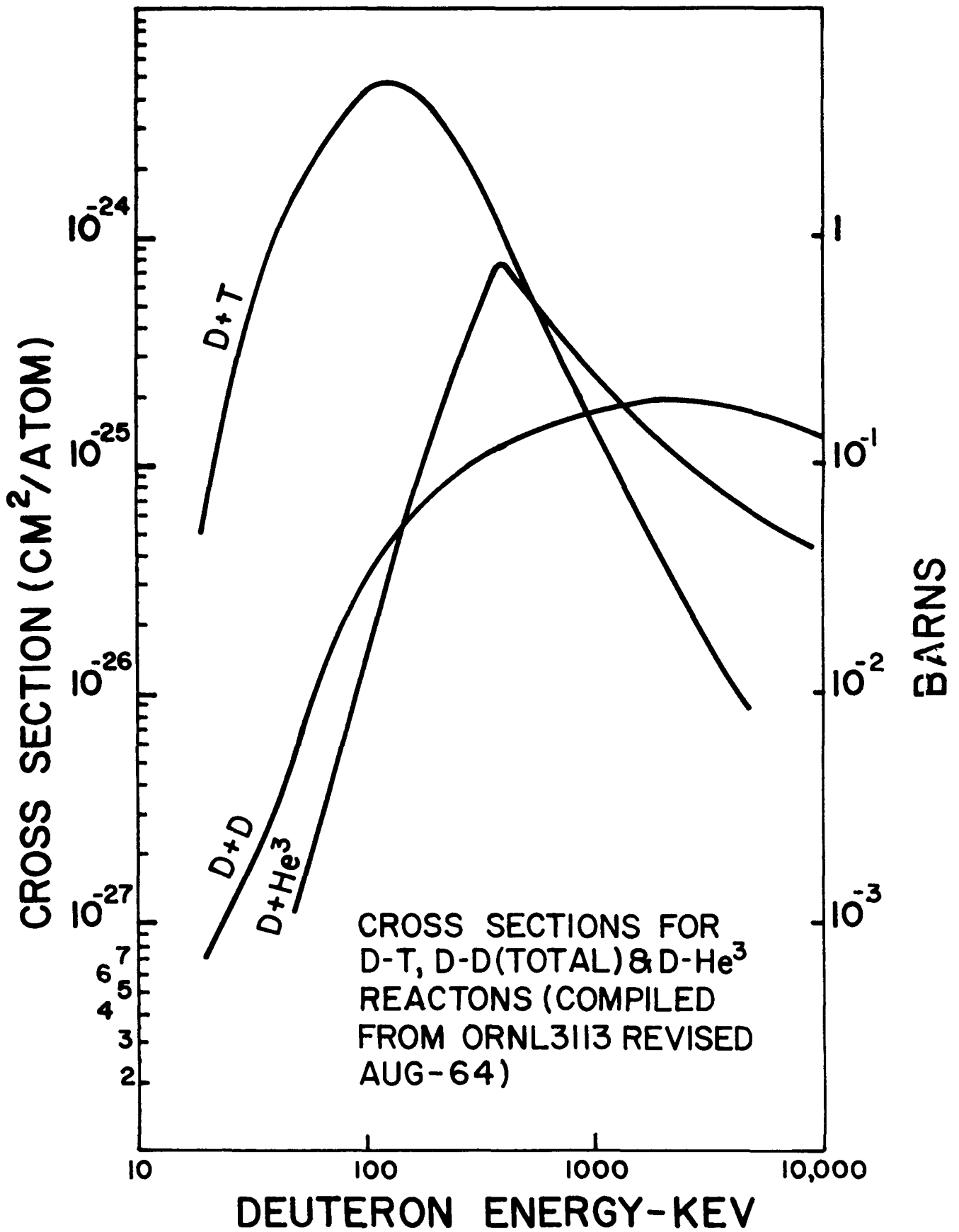


Figure 2

at energies less than about 100 KeV the D-T cross section is about two orders of magnitude higher than the sum of the D-D reactions and about three orders of magnitude higher than the D-He<sup>3</sup> reaction. However, as energy is increased, the D-T cross section diminishes markedly while the D-D and the D-He<sup>3</sup> continue to rise until at about 1000 KeV the three cross sections are about equal. Plasma temperatures in excess of 100 KeV will be of interest for the mirror machine direct converter.

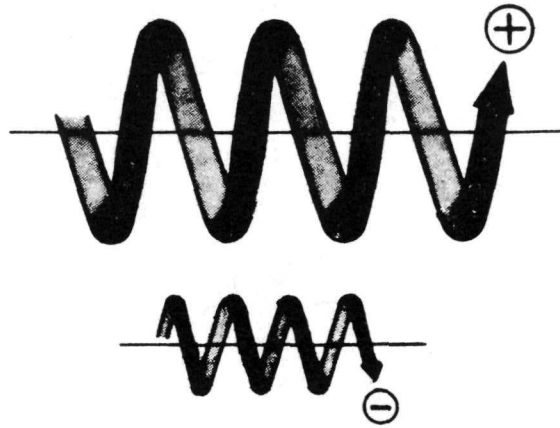
### END LOSSES IN A MIRROR MACHINE

At the Lawrence Radiation Laboratory a major effort in fusion research for many years has been directed towards "mirror machines" and, as everyone probably realizes, a "mirror machine" is a magnetic bottle in which plasma particles are contained within a cylindrically shaped chamber. These particles are contained by being reflected back and forth between regions of magnetic field at each end of the container which are higher than the magnetic field in the central region. This is shown schematically in Figure 3.

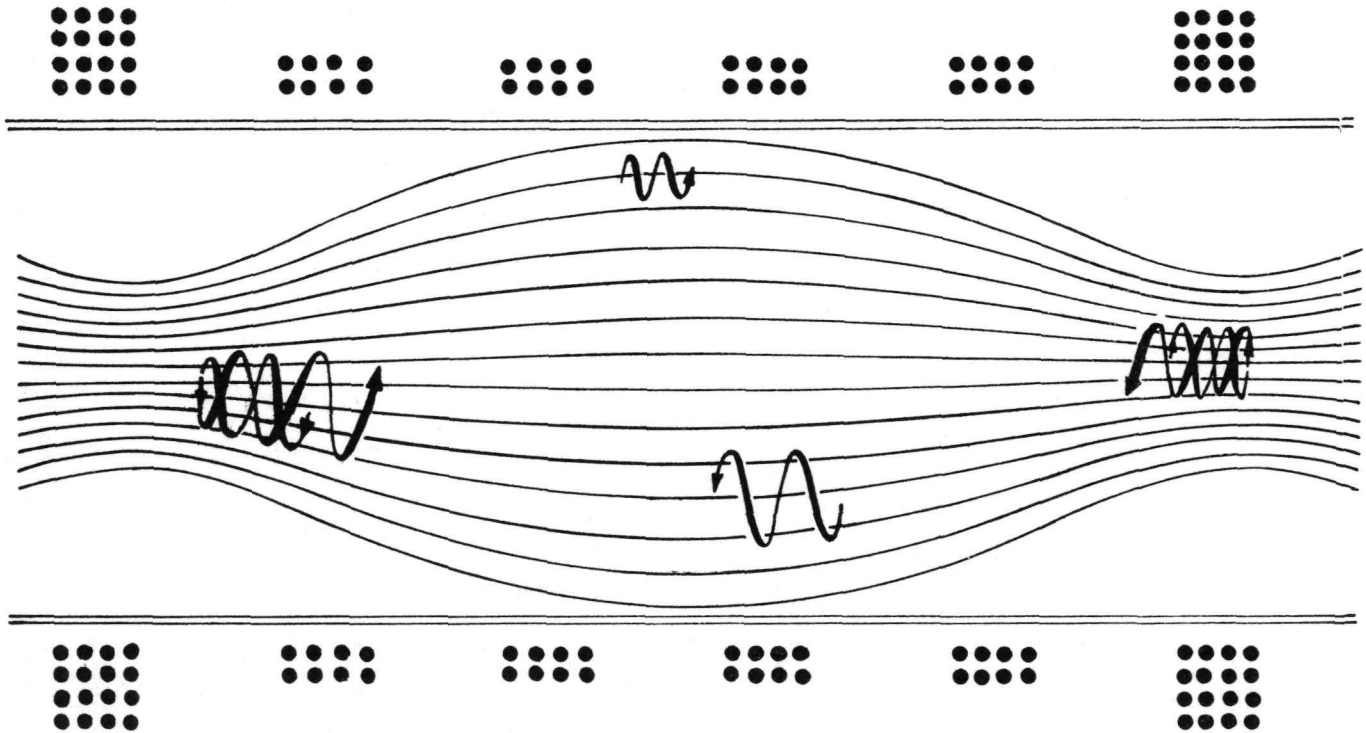
The dominant direct loss mechanism, the "classical loss," in a mirror machine is that which is associated with scattering of charged particles into the escape cone of the mirrors. This loss is substantially higher than all other direct loss mechanisms that arise such as bremsstrahlung radiation and synchrotron radiation. The rate of end loss depends on the density of the plasma, the mean collisional (scattering) cross section between the charged particles and on the charged particles mean relative velocity. End losses arise from binary collisions between the ions of the plasma and, therefore, depend on the square of the plasma density. The end loss rate per unit volume can be represented by an equation of the form:

$$\frac{dn}{dt} = -n^2 \langle \sigma v \rangle_s f(R_M) \quad (1)$$

where  $\langle \sigma v \rangle_s$  is a mean scattering rate parameter  $\sigma$  multiplied by the mean relative velocity  $v$  and  $f(R_M)$  is a function expressing the effect of the



PLASMA PARTICLES IN MAGNETIC FIELD tend to travel in helices around lines of magnetic force. Electrons (-) and positive ions (+) rotate in opposite directions.



MAGNETIC MIRROR EFFECT is produced by a "magnetic bottle" in which the magnetic field increases at each end. The stronger field bends the path of the approaching particle into even tighter circles and exerts a force that reflects it away from the ends of the bottle .

Figure 3

mirror ratio  $R_M = B_M/B_0$ . This is discussed in some detail by Post.<sup>2</sup>

The probability of particle loss into the loss cone as a function of  $R_M$ , the mirror ratio, comes about in the following way. Assuming that no collisions with other charged particles occur, then a particle moving under the influence of a magnetic field must conserve its total kinetic energy. That is, for a particle moving along a line of force, the sum of the perpendicular energy  $W_{\perp}$  and the parallel energy  $W_{\parallel}$  must be a constant. Therefore, for two discrete points a and b on a given line of force, the following relation must hold:<sup>3</sup>

$$W_{\perp a} + W_{\parallel a} = W_{\perp b} + W_{\parallel b} \quad (2)$$

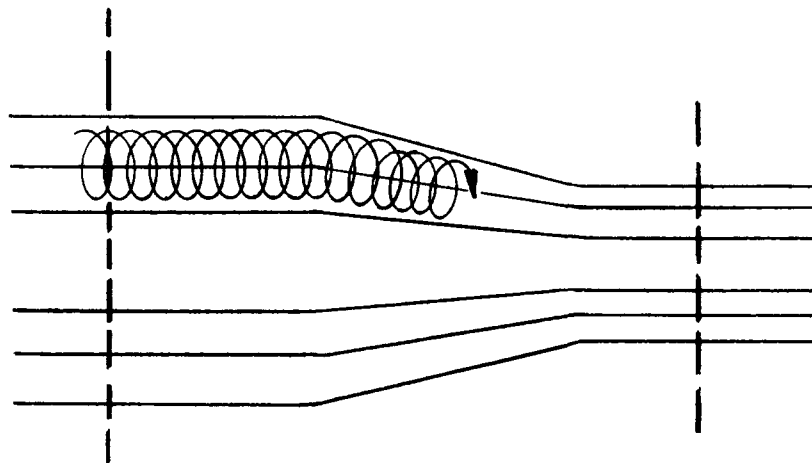


Figure 4 - Constancy of Particle Kinetic Energy Along a Line of Force

Equation (2) represents one of the adiabatic invariants of the motion of charged particles in magnetic fields. Adiabatic conditions are said to exist when the magnetic field varies slowly in time compared to the gyromagnetic frequency and varies slowly in space over a distance of the same order as the gyromagnetic radius. The gyromagnetic frequency is:

$$\omega_g = \frac{eB}{mc} \quad (3)$$



And the gyromagnetic radius:

$$r_g = \frac{mv_{\perp}c}{eB} \quad (4)$$

Another adiabatic invariant is that a particle moves in a magnetic field so as to maintain its magnetic moment  $\mu$  equal to a constant. That is:

$$\mu = \frac{IA}{c} \quad (5)$$

where  $I$  is the current produced by a particle moving in a circular orbit of area  $A$ . Since  $A$  can be expressed in terms of  $r_g$  from (4) and the current is equal to  $e/\tau_g$ , where  $\tau_g$  is the gyromagnetic period, (the time for one particle rotation) which is equal to  $1/2\pi \omega_g$  from (3) then:

$$\mu = \frac{\frac{1}{2}mv_{\perp}^2}{B} = \frac{W_{\perp}}{B} \quad (6)$$

Consider the situation shown in Figure 5:

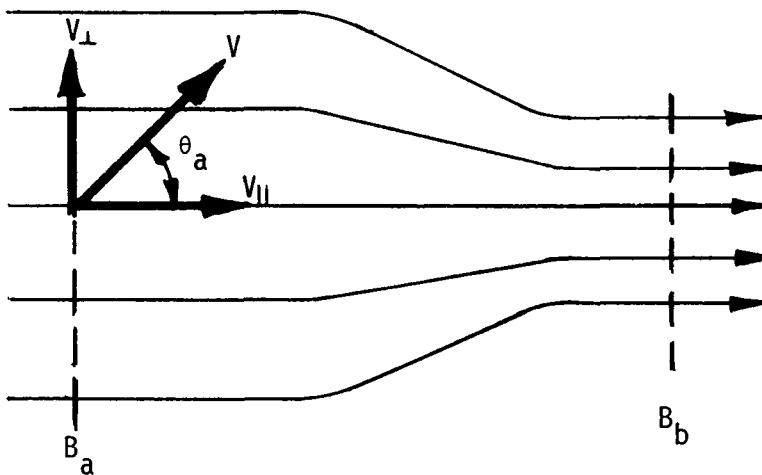


Figure 5 - Pitch Angle  $\theta$  of a Particle in a Magnetic Field

The field is assumed to increase from its value at a to a maximum value at b. The velocity of the particle towards b is:

$$v_{||} = v_a \cos \theta_a \quad (7)$$

and its perpendicular component:

$$v_{\perp} = v_a \sin \theta_a \quad (8)$$

From equation (2) the total kinetic energy is conserved and from equation (6) the magnetic moment is invariant.

If we rewrite equation (2) in the following form:

$$\frac{1}{2} m v_{||}^2 + \frac{1}{2} m v_{\perp}^2 = \frac{1}{2} m v^2 \quad (9)$$

and the constancy of the magnetic moment as:

$$\frac{mv_{\perp}^2}{B} = \frac{mv_{\perp}^2 a}{B_a} = \frac{mv^2}{B_a} \sin^2 \theta_a \quad (10)$$

then dividing (9) by B and substituting from (10) the following is obtained:

$$\frac{v_{||}^2}{B} = \frac{v^2}{B} - \frac{v^2 \sin^2 \theta_a}{B_a} \quad (11)$$

So that,

$$v_{||}^2 = v^2 \left(1 - \frac{B}{B_a} \sin^2 \theta_a\right) \quad (12)$$

For particle reflection to occur  $v_{||}$  must be zero and it then follows:

$$\sin^2 \theta_a = \frac{B_a}{B} \quad (13)$$

If  $B = B_M$ , the field at the mirror, and consequently  $B_M/B_a = R_M$  (the mirror ratio), then for a particle to be bound between mirrors:

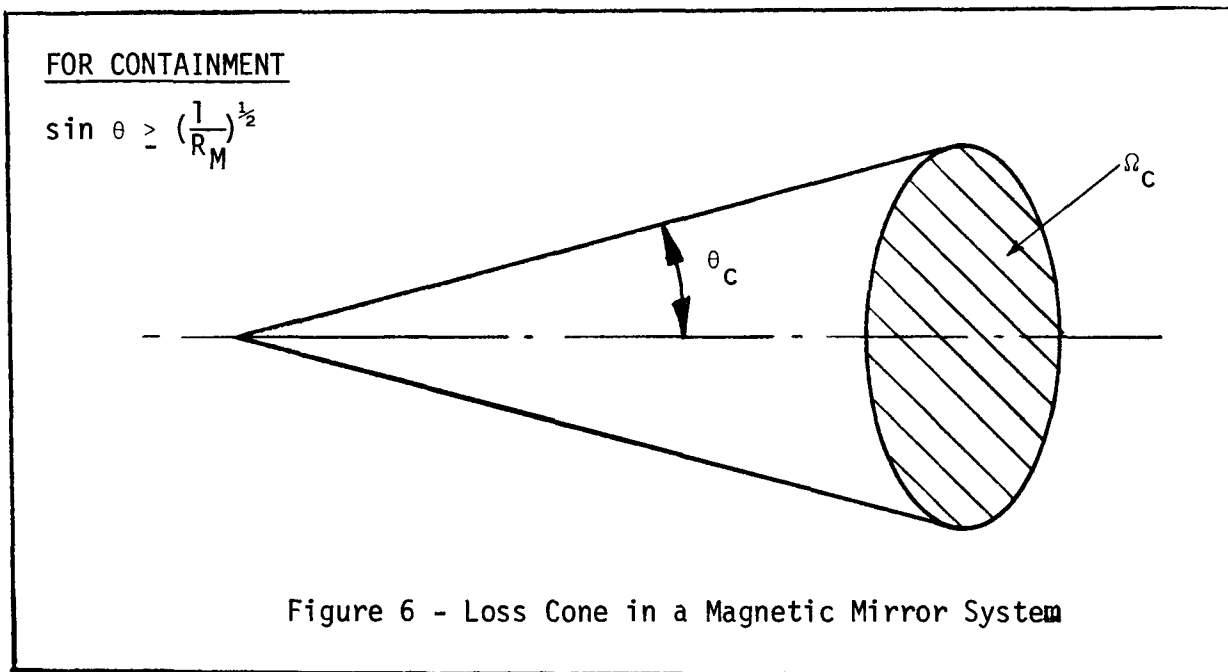
$$\sin \theta_a \geq \left(\frac{1}{R_M}\right)^{\frac{1}{2}} \quad (14)$$

Equation (14) states that any particle with an initial velocity vector which is at a pitch angle smaller than  $\theta_a$  will be lost out the end(s) of the mirror machine.

If we now refer back to equation (1) it can be seen that the term  $f(R_M)$  represents the probability  $P$  that a particle, having scattered, will fall into the loss cone of the mirror machine and escape. The loss cone is defined as the region in velocity space within which particles will be lost. (See Figure 6). This can be stated as:

$$P = \frac{\Omega_c}{2\pi} = \int_0^{\theta_c} \frac{2\pi \sin \theta \, d\theta}{2\pi} \quad (15)$$

$$= 1 - \cos \theta_c$$



Since  $\cos \theta = (1 - \sin^2 \theta)^{1/2}$  and using (14) we obtain that:

$$P = 1 - \left( \frac{R_M - 1}{R_M} \right)^{1/2} \quad (16)$$

Having established that the loss of particles is one which is inevitable and that the probability of escape is as stated in equation (16), it now remains to be established how these particles may best be utilized.

### THE GENERAL PROCESS OF DIRECT CONVERSION

The process of conversion of charged particle energy to electrical energy may be imagined to take place in a sequence of four steps:

1. Expansion
2. Charge separation
3. Deceleration and collection
4. Conversion to a common potential

As shown pictorially in Figure 7 the reaction products escaping from the mirrors at already low ion density<sup>4</sup> (typically  $10^8/\text{cm}^3$ ) would be allowed to further decrease their density by expansion into a larger chamber where the density would be reduced to about  $10^6/\text{cm}^3$ . The expansion process would be done adiabatically similar to an expansion in a nozzle. Adiabatically in fusion terminology means that the change in magnetic field with distance is sufficiently low so that particles still stay on field lines. In this case, the expansion is controlled by coupling an external radial magnetic field to the mirror field and allowing the resultant field to decrease from its high level at the mirrors (perhaps 150 kilogauss) to levels of about 500 gauss. The effect of this expansion is to decrease the ion density and also to convert the particles' rotational energy to translational energy. At the end of this expander field, the electrons are separated off electromagnetically and the positive ions, which contain the bulk of the energy compared to the electrons, yield this energy to a series of electrostatic collectors. Each collector is

# FUSION REACTOR WITH DIRECT CONVERSION

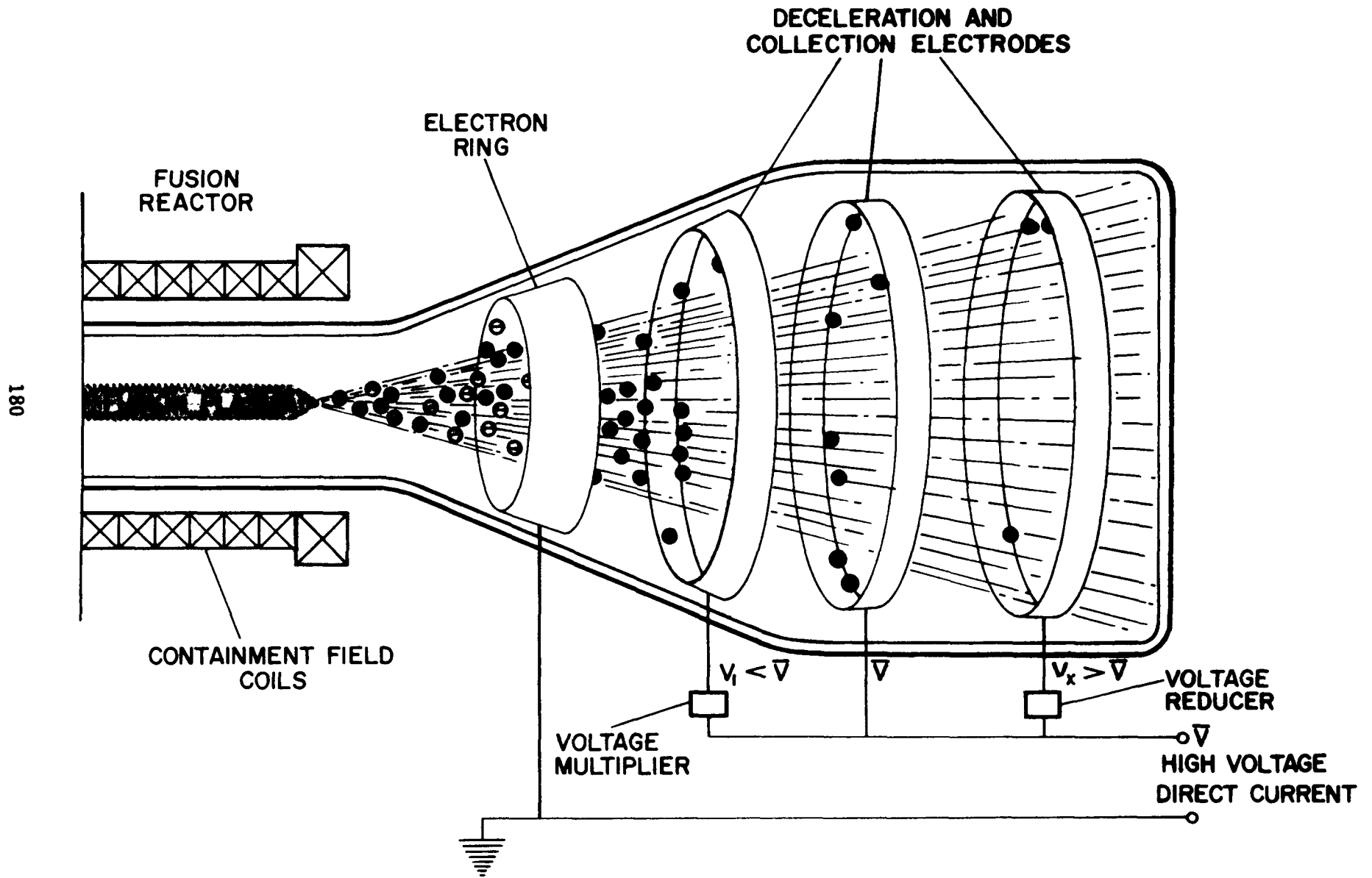


Figure 7

kept at a different potential, the potential of the first collector being lower than the average potential of the particles and the last collector being higher than the average particle potential. This collector system is in effect an electrostatically focused linear decelerator. Each ion slows down until it loses almost all of its kinetic energy at which point it's deflected into a collector electrode. Collection efficiency is proportional to the number of collectors. That is, as the number of collectors increases, the potential difference between collectors decreases and the particles can be collected more closely to the point where they have lost all their kinetic energy to the collectors which they have passed. By a series of external inverters and rectifiers coupled to the collectors useful power is produced in the form of high voltage dc.

#### THE EXPANDER

The expander section of a direct converter can be represented by a fan shaped enclosure which contains a magnetic field winding producing a radial field. This is shown in Figure 8. The lines of this field couple to the fringing field of the mirror and within the expander chamber the field is allowed to weaken as a function of radius until it reaches a low value. Also, the initial bundle of flux lines is transformed to a flat fan-shaped pattern.

From the continuity of magnetic flux:

$$r_E h_E B_E = \text{constant} \quad (17)$$

the first relationship for the expander is established.

The function of the expander is twofold: it must reduce the density of the particle stream and it must convert the particles perpendicular energy to translational energy. The density reduction is required because it appears that space charge effects will set the upper limit on the power handled by the collector structure of the direct converter.

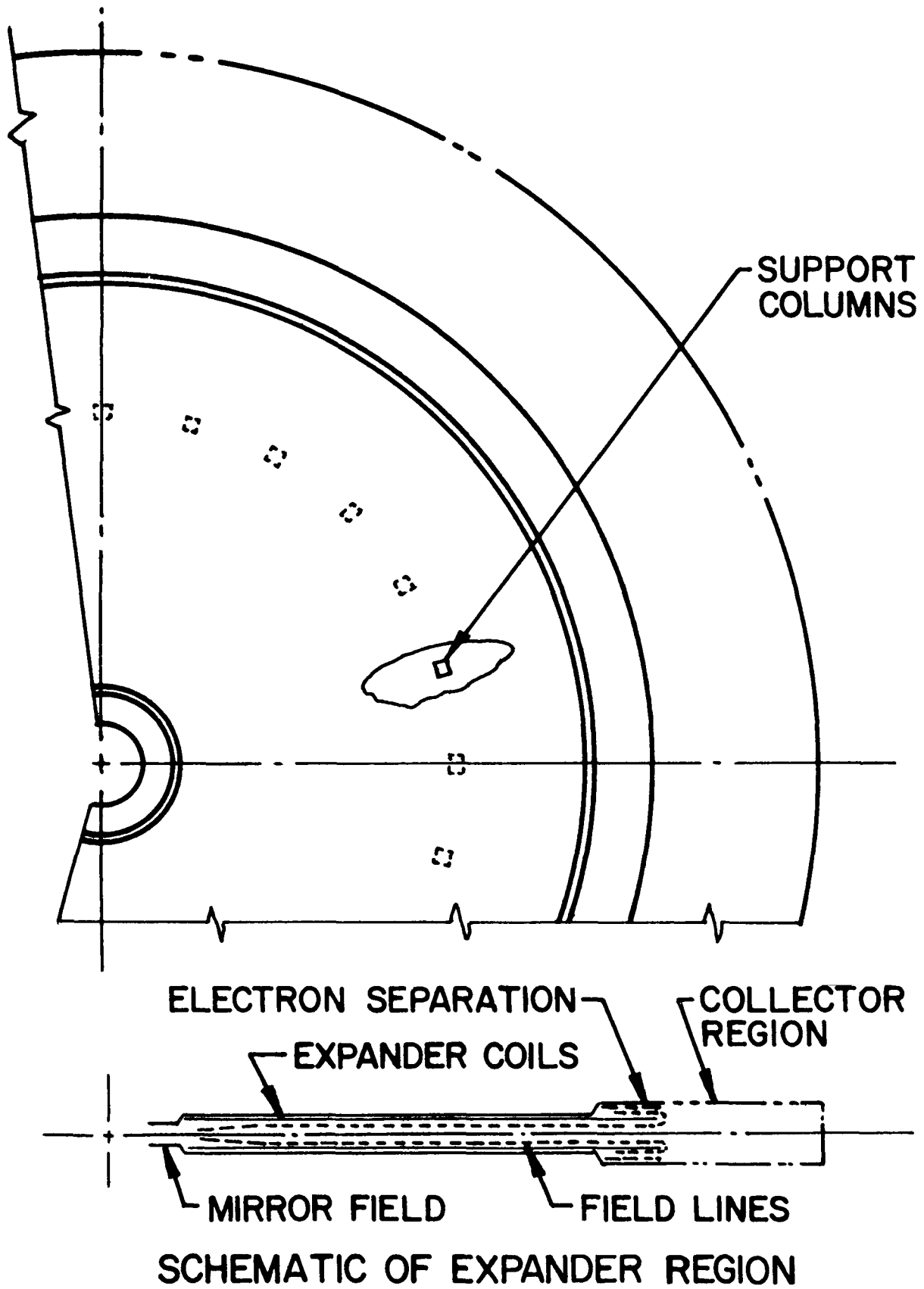


Figure 8

Therefore, space charge limits imposed by the collector must establish the proportionate scale of the expander which precedes it. The power handling limit can be scaled from the relationship:

$$\omega_p \times \tau = \text{constant} \quad (18)$$

where:

$\omega_p$  = plasma frequency in the particle beam at entrance to the collector and  $\tau$  = the transit time through a focusing length of the collector structure.

The plasma frequency is defined by:

$$\omega_p \equiv \left( \frac{4\pi n e^2}{m} \right)^{\frac{1}{2}} \quad (19)$$

So that the influence of the plasma frequency term as it relates to space charge is due to the number density,  $n$ , of the particles.

The total power to be handled by the collector is proportional to the energy flux in the beam and to the entrance area of the collector. If the collector entrance and the expander exit regions are contiguous and their areas are equal then:

$$P \propto (J \cdot W \cdot A) \quad (20)$$

or 
$$P \propto [(nv) (v) (hr_E)]$$

where  $h$  is a typical collector dimension (the particle stream height) and  $r_E$  is the radius of the expander.  $W$  is the particle energy. From  $(\omega_p \tau)$  requirement in (18) it is seen that  $\omega \propto (\sqrt{n})$  and  $\tau \propto (\frac{h}{v})$  and that the number density  $n$  is:

$$n \propto \left( \frac{v^2}{h^2} \right) \quad (21)$$



By substitution of (21) into (20) it is seen that:

$$P \propto \left[ W^{5/2} \left( \frac{r_E}{h} \right) \right] \quad (22)$$

The 5/2 power scaling with energy makes it advantageous to operate at high particle energies. Since power is inversely proportional to  $h$ , then subdivision of a single flux fan into a number of stacked fans might permit operation at lower energies, other limits permitting. Thus in a stacked system:

$$P \propto \left[ W^{5/2} N^2 \right] \quad (23)$$

where  $N$  is the number of stacks.

It is assumed conservatively that on emergence from the mirror the particle energy is all rotational. The conversion of rotational energy to translational energy is inversely proportional to the field change from the mirror to the expander terminus. The loss due to efficiency of conversion of rotational energy to translational energy is given by:

$$\frac{E_{\perp}}{E_{\text{tot}}} = \frac{B_E}{B_M} \quad (24)$$

If the field at the expander is relatively small, for instance  $B_E \approx 500$  gauss and  $B_M$ , the field at the mirror, is quite large,  $\approx 150,000$  gauss, it can be seen that the loss due to incomplete conversion is extremely small.

Having reduced the density of the plasma and having also given the particle high translational energy as a function of expander radius, the next step in the process is to separate the electrons from the ions. This will be seen to introduce an additional loss. It has been proposed by Post that this separation can be accomplished by rapidly diverting the field lines at the expander exit in a direction perpendicular to the plane of the fan. This was shown in Figure 8. The electrons, behaving

adiabatically, will continue to be guided by the field lines while the ions will cross the field lines and enter the collector. The reason for this behavior difference is that the condition for adiabatic invariants for electrons is more easily satisfied than for positive ions. After separation, recovery of energy from both electrons and ions is possible by the same general technique. However, the electrons carry only a small fraction of the total energy and we will be concerned only with the recovery of the ion energy. The ions, in their non-adiabatic behavior as they leave their guiding centers and proceed into a field free zone, receive a small transverse momentum kick in the plane of the fan because they must cross field lines as a condition for emergence. Busch's Theorem<sup>5</sup> states that the velocity imparted to the charged particle in a direction transverse to the field is given by:

$$|v| = \frac{e}{m} B_x |y| \quad (25)$$

where  $y$  is the height above the plane of symmetry (See Figure 9).

Since the transverse energy component  $W_{\perp z} = mv_z^2/2$  then:

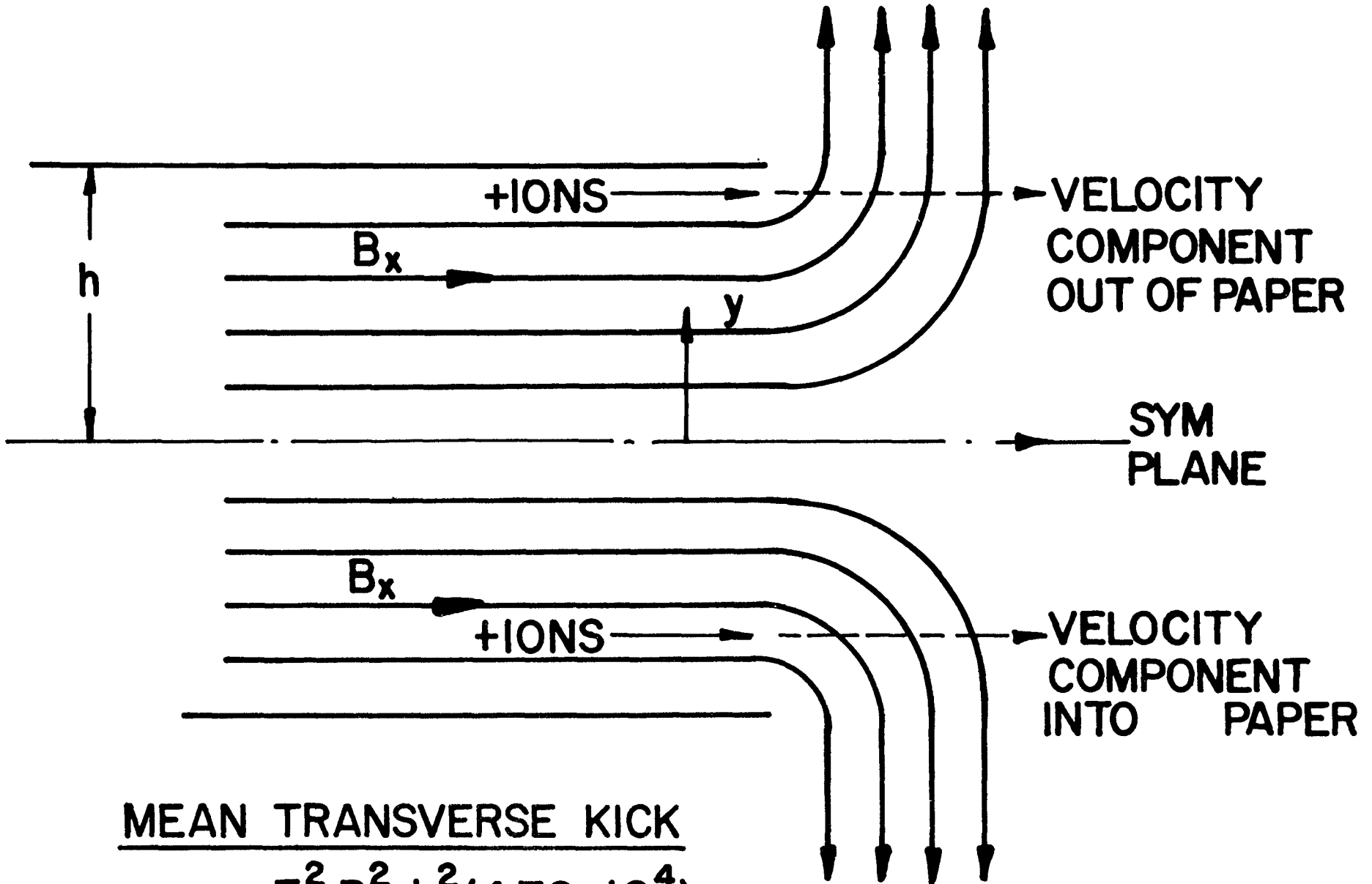
$$W_{\perp z} = \frac{e^2}{2m} B_x^2 y^2 \quad (26)$$

If this is normalized to a proton and expressed in KeV then:

$$W_{\perp z} = 4.79 \times 10^4 \frac{Z^2}{A} B_x^2 y^2 \quad (27)$$

where  $A$  is the atomic mass,  $Z$  is the atomic charge (or atomic number) of the charged particle and  $B$  is in tesla ( $10^{-4}$  gauss). From the Figure 9 it can be seen that the mean transverse kick is:

$$W_{\perp z} = \left[ 4.79 \times 10^4 \frac{Z^2}{A} B_x^2 \right] \times \int_0^h \rho(y) y^2 dy \quad / \quad \int_0^h \rho(y) dy \quad (28)$$



MEAN TRANSVERSE KICK

$$\bar{W}_{\perp z} = \frac{z^2}{A} \frac{B_x^2 h^2}{3} (4.79 \times 10^4)$$

Figure 9

where  $\rho(y)$  is the particle density. If the particle density is assumed to be constant, then  $\rho(y) = \frac{1}{3}$  and:

$$\bar{W}_{\perp z} = \frac{Z^2}{A} B_x^2 \frac{h^2}{3} (4.79 \times 10^4) \quad (29)$$

For a representative example:

$$\begin{aligned} \bar{W}_{\perp z} &= \left(\frac{1}{2}\right) (5. \times 10^{-2})^2 \frac{.5^2}{3} (4.79 \times 10^4) \\ &= 5 \text{ KeV} \end{aligned}$$

Mean particle energies are  $\sim 800$  KeV. Thus this "kick" loss amounts to approximately 0.4%.

#### THE COLLECTOR SYSTEM

As the positively charged particles emerge from the expander and begin to enter the collector, they do so with a spread in energy, the lower end of which is set by the plasma potential the upper end by the exponential fall off of the energy distribution. This distribution function is discussed by Fowler and Rankin.<sup>6</sup> Suffice it to say that in order to recover the energy of the particles in a highly efficient manner it is necessary to provide some systematic means of energy sorting. This can be done by directing the particles through a series of decelerating electrodes so that each particle is decelerated until its energy is small compared to its original energy at which time the particle is diverted into a collector element which closes the current loop. This means that the currents are delivered over a wide range of potentials. The first electrode must therefore be at the plasma potential and successive electrodes each at some increment higher until the maximum potential is reached. Clearly a system must be used which does not allow the particles to be prematurely diverted into a collector when they still have significant energy. The collector system must be a region of retarding force, slowing down the particle stream. Space charge effects will tend to defocus

the stream and cause particles to divert to the collector early. This must be compensated for. The system of electrodes must collect the low energy particles first while continuing to focus the higher energy particles. A system which has been proposed and experimental work initiated on, uses the focusing properties of periodic electrostatic lenses whose focusing ability is energy sensitive. As the particles energy decreases, it will reach a point where the lense system potentials will cause the particle to be overfocused or diverted. An example of a collector system is shown in Figures 10 and 11. The first figure shows several unit cells of a collector and the next shows some relative potentials that might be applied to a larger number of collectors. Here the system is made up of two main elements: (1) fins at potentials ranging from  $\phi_0$  (the plasma potential) to  $\phi_n$ , with their potential increasing in the y direction and (2) diverter grids at potentials alternately higher on one side of the plane of symmetry than the other and also generally increasing in the y direction. In the example shown in Figure 11 the fin potential is equal to the average potential of the diverter grids on either side of it and opposed grids are alternately positive or negative with respect to each other. It is assumed that the fins accomplish the focusing of the charged particles. At a point where the charged particles velocity in the y direction is sufficiently small and close to zero the potential difference between opposing diverter grids will cause the particle to be accelerated in the x direction towards the more negative grid. It is possible at this point to allow the particle to be collected at the grid itself. However, if the grid has sufficiently high porosity it is then advantageous to allow the particle to proceed through the grid structure and be collected by the aft segment of the up-stream fin and therefore at a more favorable potential.

#### HVDC VIA INVERTER AND RECTIFIERS

After collection of the particles over the energy range from  $\phi_0$  to  $\phi_n$  it is necessary to bring all potentials to a common potential,  $\bar{v}$  so that it is available at the bus bar in a form that has some commercial utility. Post has proposed that this can be done in a highly advantageous and economical way by using a system of inverters and rectifiers. He

# COLLECTOR ELECTRODE CONFIGURATION

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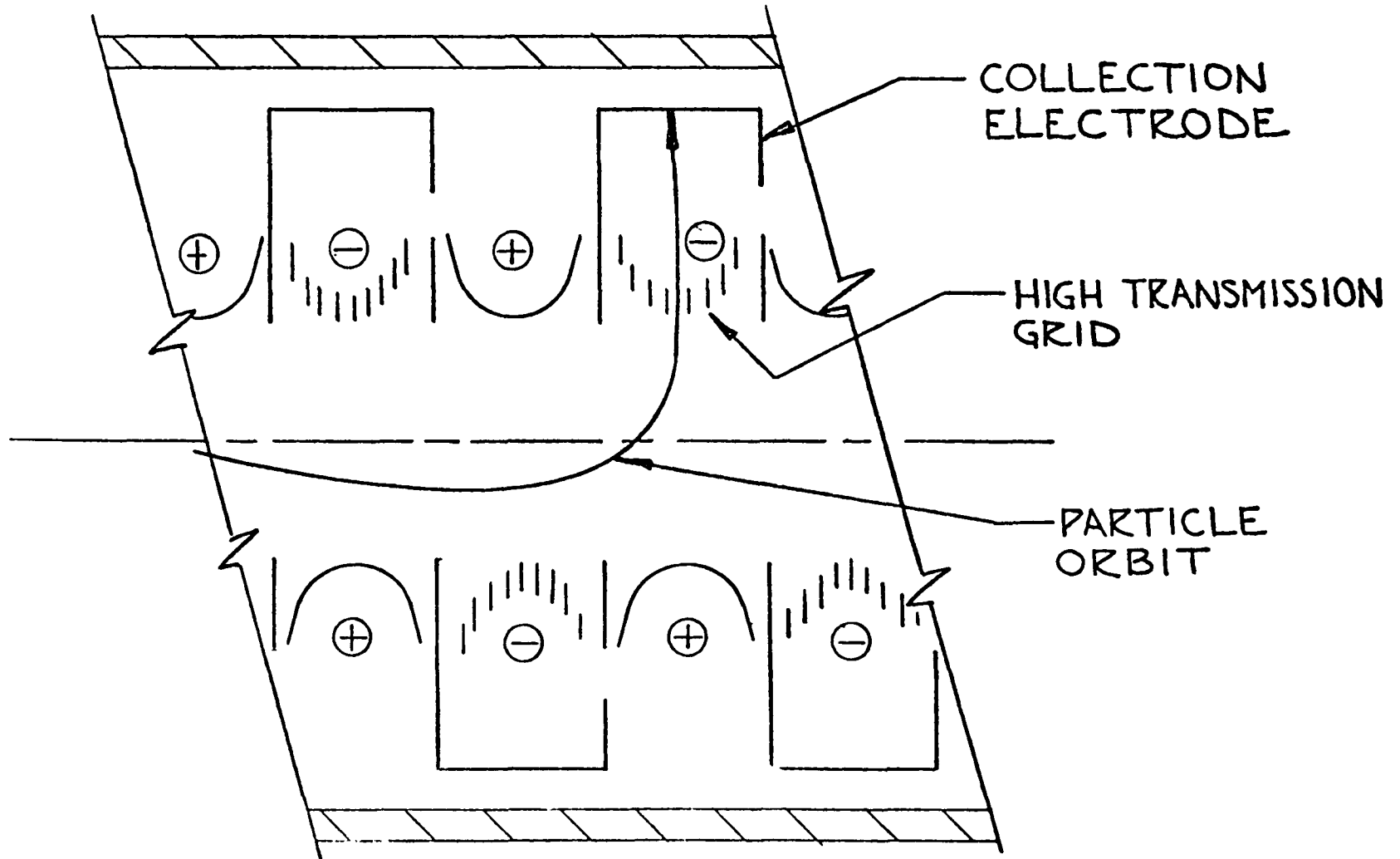
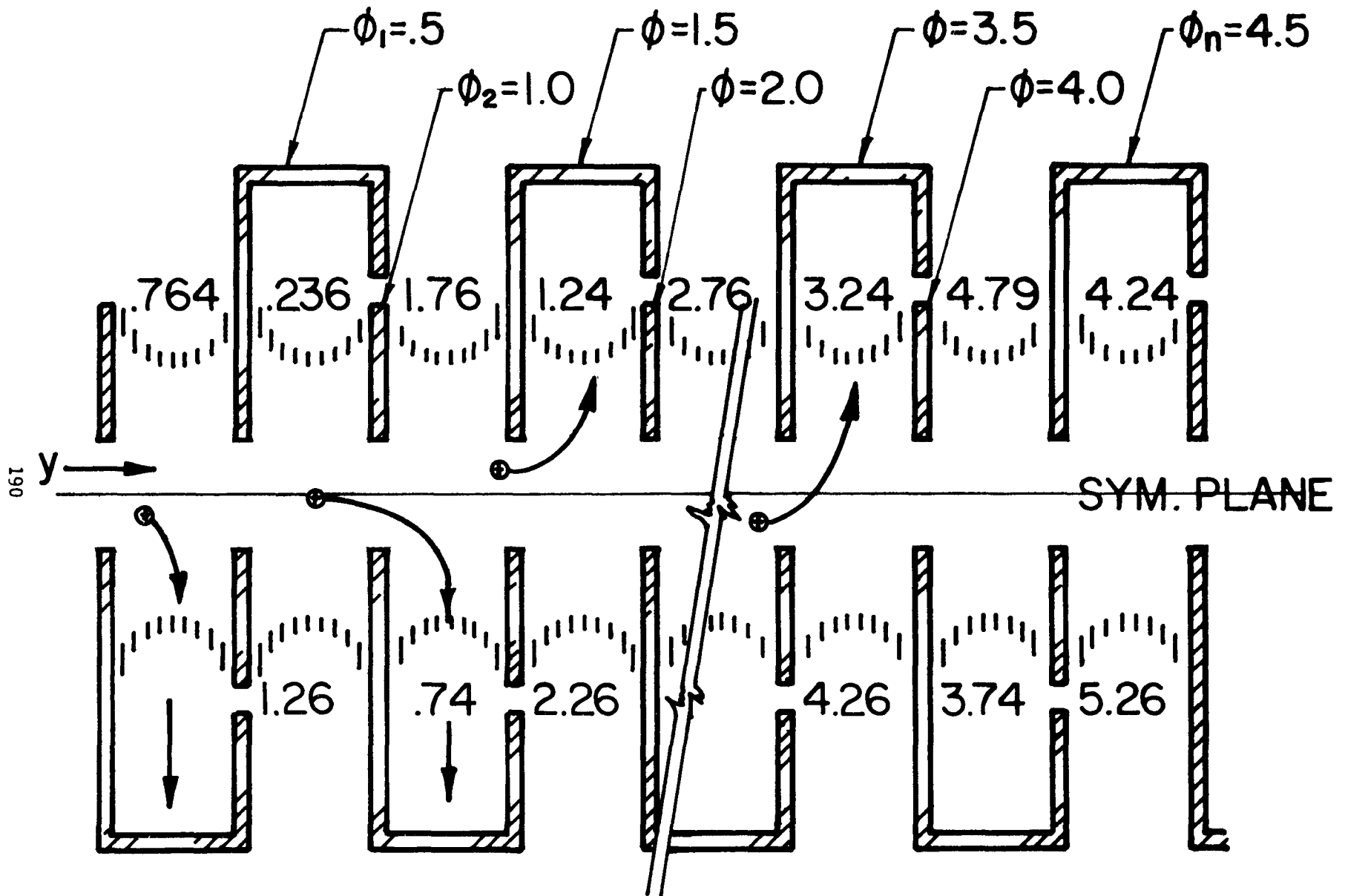


Figure 10



# COLLECTOR STRUCTURE FOR A DIRECT CONVERTER

Figure 11

states that: "The expander-collector system accomplishes the act of direct conversion of particle kinetic energy to dc electrical energy, but delivers this energy at several different potentials. While inverter circuits could be used to convert this dc energy to ac, a more attractive possibility is to convert to a common dc potential, for direct connection to a HVDC transmission line. A circuit to accomplish this operation is shown in Figure 12. A common potential is defined in terms of the several collector currents and potentials,  $I_j$  and  $V_j$ :

$$\bar{v} = \frac{\sum_{j=1}^N I_j V_j}{\sum_{j=1}^N I_j} \quad (30)$$

As shown, inverters are used to diminish all potentials  $> \bar{v}$ , and rectifiers to raise all potentials  $< \bar{v}$ . Currents are directly summed. From the definition of  $\bar{v}$  it follows that the ac power from the inverters is just sufficient to power the rectifiers. However, the total power handled (sum of inverter and rectifier power) is relatively small. The total power handled by the I-R system is given by:

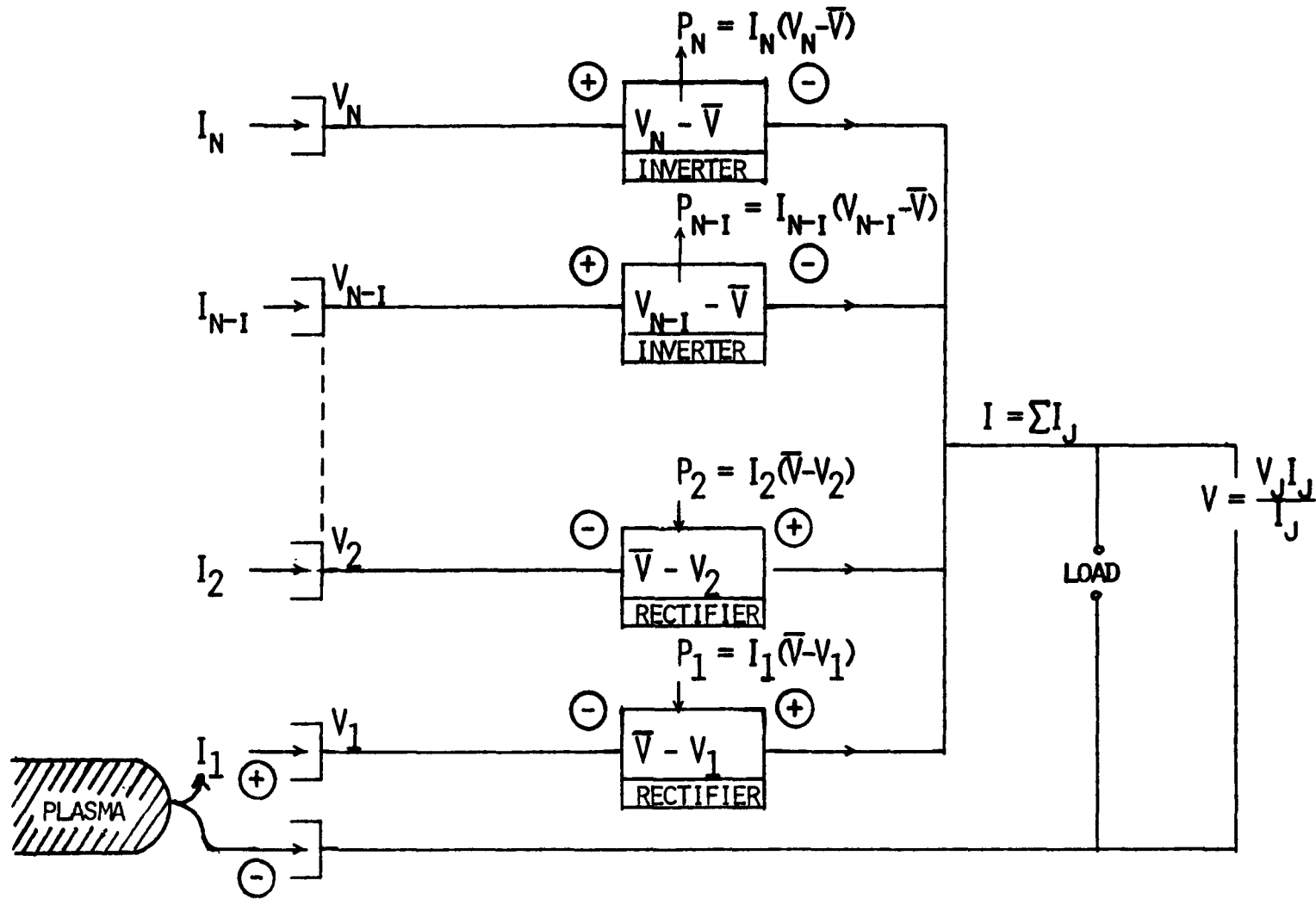
$$P_{I-R} = P_I + P_R = \left\{ \bar{v} \left[ \sum_{V_j < \bar{v}} I_j - \sum_{V_j > \bar{v}} I_j \right] - \left[ \sum_{V_j < \bar{v}} I_j V_j - \sum_{V_j > \bar{v}} I_j V_j \right] \right\}$$

For the example given earlier (calculated for the Fowler-Rankin distribution function)  $P_{I-R} = 0.28 P_0$ , i.e. the inverters and rectifiers each handle 14% of the total power. Since the efficiency of these devices, as established in HVDC practice, is about 0.99, the net loss thus introduced will be only about 0.3%."

#### CHARGE TRANSFER AND IONIZATION LOSSES

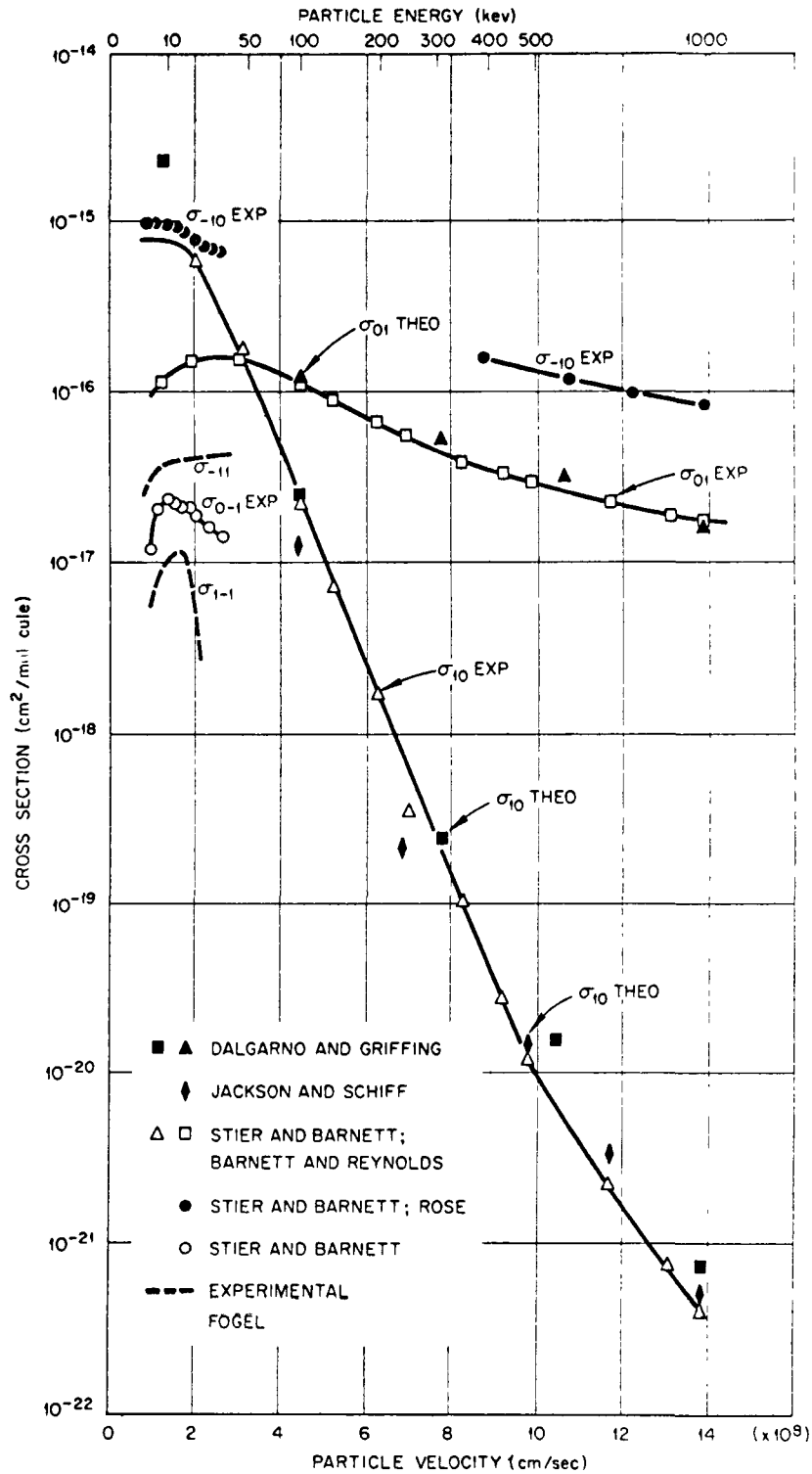
A loss mechanism in the direct converter is that due to charge transfer and ionization loss. Charge transfer loss calculations<sup>7</sup> were based on cross sections of hydrogen ions in hydrogen gas, (See Figure 13) and in helium gas (See Figure 14). Deuteron cross sections were assumed to be the same as hydrogen ions at 1/2 the deuteron energy.





CHARGED PARTICLE DIRECT CONVERSION POWER CIRCUITRY

Figure 12



**Figure 13**  
Charge Transfer Cross Sections of Hydrogen Atoms and Ions in Hydrogen Gas.

Ref. Barnett, C.F., Ray, J.A., & Thompson, J.C., "Atomic and Molecular Collision Cross Sections of Interest in Controlled Thermonuclear Research," ORNL-3113, Revised UC-20, Aug. 1964, p. 60

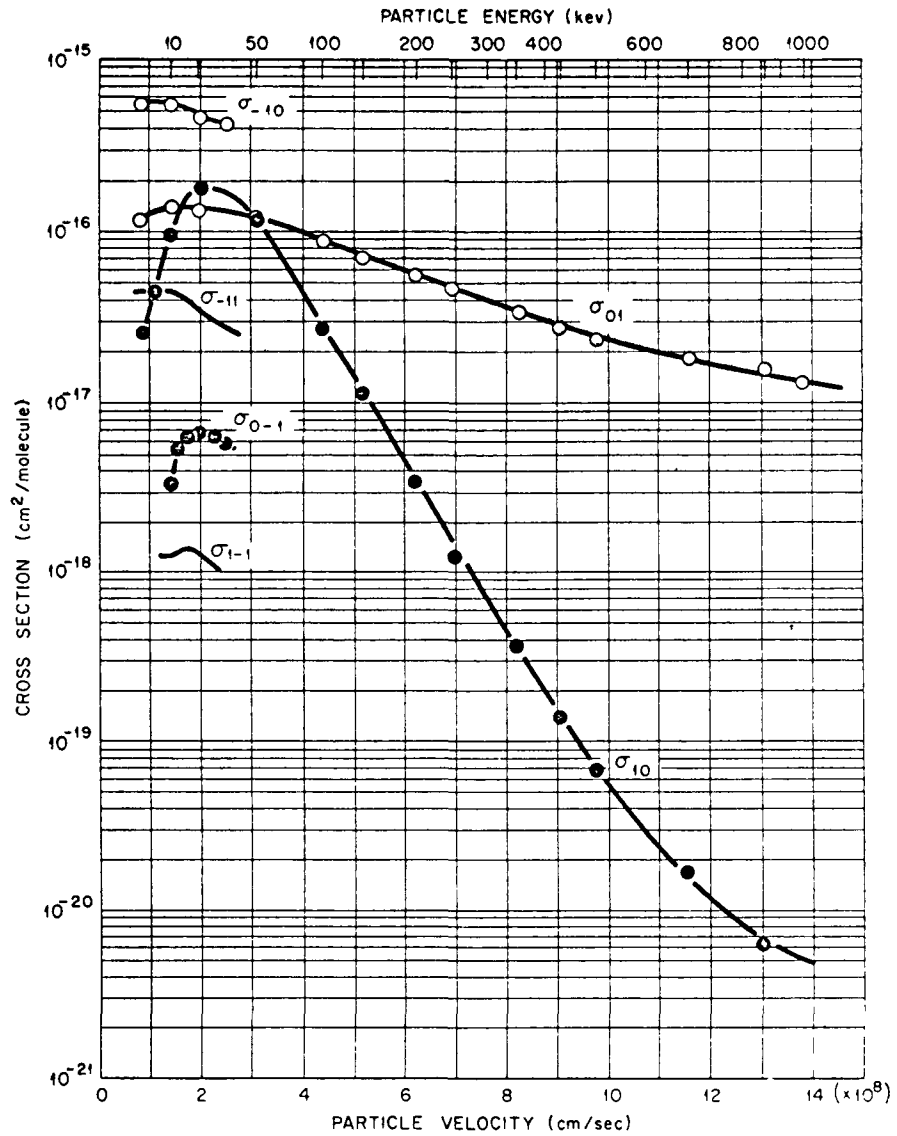


Figure 14

**Charge Transfer Cross Sections of Hydrogen Atoms and Ions in Helium Gas.**

Ref. Barnett, C.F., Ray, J.A., & Thompson, J.C., "Atomic and Molecular Collision Cross Sections of Interest in Controlled Thermonuclear Research," ORNL-3113, Revised UC-20, Aug. 1964, p. 60

Charge transfer in expander: Deuterons at 800 KeV were assumed to transit a 75 meter expander. The probability of charge transfer is:

$$P_{CT} = 1 - \exp(-\Sigma_{CT} \cdot \ell) \quad (31)$$

where:

$$\begin{aligned} \Sigma_{CT} &= \sigma_{CT} \cdot 2.58 \times 10^{19} \frac{\text{mol}}{\text{cm}^3(\text{STP})} \cdot \frac{P(\text{torr})}{76} \\ &= 3.54 \times 10^{16} \sigma_{CT} \cdot P \end{aligned}$$

at the 800 KeV deuteron level (400 KeV H<sub>2</sub>) the cross sections are:

$$\begin{aligned} \sigma_{CT} &= 4 \times 10^{-20} \text{ cm}^2/\text{mol} (\text{H}_2) \\ &= 2 \times 10^{-19} \text{ cm}^2/\text{atom} (\text{He}) \end{aligned}$$

and the length as stated is:

$$\ell = 7.5 \times 10^4 \text{ cm}$$

The following table shows the probability of charge transfer in the expander as a function of pressure and assumed background gas:

Pressure (torr)	P <sub>CT</sub> (in H <sub>2</sub> )	P <sub>CT</sub> (in He)
1 x 10 <sup>-3</sup>	.0104	.0517
1 x 10 <sup>-4</sup>	.00106	.00530
1 x 10 <sup>-5</sup>	.000106	.000531

Table 1 - Probability of Charge Transfer in the Expander

Charge transfer in collector: Deuterons at 800 KeV were assumed to transit 10 elements of a 20 element, 20 meter long collector. Because the cross section is strongly energy dependent the deuteron's transit

through the 10 collector elements was treated as 10 one meter transits at energies between 760 KeV and 40 KeV in 10 equal steps of 80 KeV each. The charge transfer loss results are sensitive to minimum ion energy (40 KeV assumed above) because the cross section is a strong function of energy. Because of this, real collector dynamics should be used when more refined collector charge transfer losses are calculated. Table 2 shows the probability of charge exchange in the collector

Pressure (torr)	Max. Loss		Min. Loss	
	In H <sub>2</sub>	In He	In H <sub>2</sub>	In He
10 <sup>-4</sup>	.242	.159	.0184	.0152
10 <sup>-5</sup>	.0273	.0171	.0020	.00160

Table 2 - Probability of Charge Exchange in Collector

Ionization losses in the collector: were based on an average cross section ( $\sigma_i$ ) in hydrogen and helium of  $10^{-16}$  cm<sup>2</sup>/mol. This average value was used because the ionization cross section is a weak function of energy. The probability of an ion causing an ionization,  $P_i$ , is equal to the path length ( $\ell$ ) times the mean free path ( $\Sigma_i$ ).

$$P_i = \Sigma_i \cdot \ell \quad (32)$$

where

$$\bar{\Sigma}_i = \bar{\sigma}_i \cdot 2.69 \times 10^{19} \left( \frac{\text{mol}}{\text{cc(STP)}} \right) \frac{P(\text{torr})}{760}$$

$$P_i = 3.54 \cdot P \cdot \ell$$

The probability of ionization loss in the collector is shown in Table 3.

Pressure (torr)	Max. Loss		Min. Loss	
	In H <sub>2</sub>	In He	In H <sub>2</sub>	In He
10 <sup>-5</sup>	.0708	.0708	.0035	.0035

Table 3 - Probability of Ionization Loss in the Collector

The maximum loss case assumes all the initial energy of a primary ion undergoing charge exchange is lost and for each ionization the ion and electron produced are accelerated out opposite ends of the collector taking 1600 KeV with them. The minimum loss case assumes only the kinetic energy remaining in the primary ion at the time of charge exchange is lost, and for each ionization only the energy picked up by acceleration between 2 adjacent electrodes is lost, which in this case is 80 KeV. Fortunately, a collector design that efficiently collects the primary ions should also efficiently collect the secondary ions and electrons born in the collector, making the minimum loss case the more realistic.

The total charge transfer and ionization losses estimated in the expander-collector operating at the described conditions are  $0.106\% + 0.554\% = 0.660\%$  in hydrogen and  $0.530\% + 0.515\% = 1.045\%$  in helium. In this low pressure range the losses are approximately proportional to pressure.

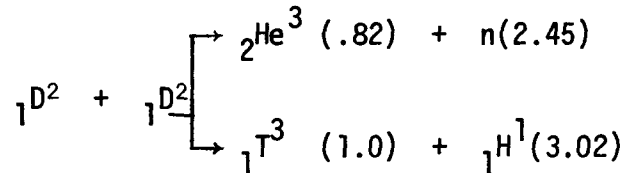
#### SOME COMMENTS ON THE FUEL CYCLES

It is evident that for direct conversion three fuel cycles are possible; the D-T cycle, in which direct conversion is used in conjunction with thermal conversion from a blanket which provides neutron multiplication and tritium breeding; the D-D cycle, where tritium breeding is not required but thermal conversion is a necessary contribution to the total power output; and the D-He<sub>3</sub> cycle, in which charged particles are the only product and consequently no blanket is necessary. However, within this cycle some D-D side reaction is inevitable (in proportion to the relative cross sections) and therefore a nominal blanket would be required to cope with perhaps 5% of the total energy release.

It is also evident that within these three fuel cycles two parameters are subject to control: (1) the composition of the particular fuel and (2) the mean energies of the various fuel ions.

The source of deuterium and tritium for the fuel is by straightforward separation processes for deuterium and by breeding in the blanket for

tritium. The source of  $\text{He}^3$  fuel is either from the radioactive decay of tritium as an external means or from the accompanying D-D reactions as an internal source. Post has shown that if the latter method is employed exclusively, then the fractional density of  $\text{He}^3$  fuel ions can be  $\approx .25$  at high temperature. The D-D reaction is repeated here to show the  $\text{He}^3$  source.



### OVERALL EFFICIENCY

The overall efficiency of the direct converter can best be illustrated by the use of a reference case. We will assume that:

- a. Mirror Field,  $B_M = 150$  kilogauss
- b. Expander Field  $B_E = 500$  gauss
- c. Radius of Expander  $R_E = 75$  meters
- d. Height of the particle flux  $h_E = 1$  meter
- e. Fan Angle  $\beta = 240^\circ$
- f.  $E_{\text{tot}} (\text{ave}) = 800$  KeV
- g. Number of collectors  $\approx 20$

A. The loss due to efficiency of conversion of rotational energy to translational energy is:

$$\frac{E_{\perp}}{E_{\text{tot}}} = \frac{B_E}{B_M} \quad (33)$$

B. The transverse "kick" loss the particles receive on exiting the expander field free zone is:

$$\frac{E_K}{E_{\text{tot}}} = \frac{Z^2 B_E^2}{3A} \frac{(4.79 \times 10^4)}{E_{\text{tot}}} \left(\frac{h}{2}\right)^2 \quad (34)$$

or  $\eta_K \sim .996$

C. The total charge transfer and ionization losses in the expander and collector were shown to be  $\sim .0066$  in hydrogen and  $.0104$  in helium. If we use an average of these two the loss is  $\sim .0085$ , and,

$$\eta_{CTi} \approx .99$$

D. The fraction of particles intercepted by the diverter grid is simply:

$$P_{IG} = \Gamma \quad (35)$$

where  $\Gamma$  is the grid porosity of  $\eta_{IG} = 1 - \Gamma$   
 $\approx .97$

E. Some fraction of the charged particles are intercepted by columns intruding into the expander zone. It can be shown<sup>8</sup> that a representative value for this fraction is:

$$P_{IE} \approx .025 \quad (36)$$

or  $\eta_{IE} = .975$

F. The efficiency of the inverter-rectifier system to bring potentials to a mean value  $\bar{V}$  is:

$$\eta_{IR} = .997$$

G. We assume 20 collectors. The collector efficiency is:

$$\eta_c \approx 1 - \frac{1}{N}$$

where  $N$  is the number of collectors

$$\eta_c = .95$$



The various efficiencies are indicated in the following table:

Conversion $\eta_c$	99.7
Field exiting $\eta_K$	99.6
Avoidance of Charge Transfer & Ionization $\eta_{CT}$	99.
Diverter grid $\eta_{IG}$	97.
Expander columns $\eta_{IE}$	97.5
Inverter Rectifier	99.7
Collector	95.
Overall	88%

Table 4 - Direct Converter Efficiency

The overall efficiency assumes that losses occur sequentially and is therefore the product of all efficiencies.

### CONCLUSIONS

We have attempted to show in the preceding that direct conversion has some rather important advantages. It seems evident that high efficiencies are attainable and the usual Carnot limits are not governing. The implication of 88% efficiency compared to the efficiencies realized with conventional systems, (say 40%) means that waste heat load is reduced by a factor of 5 (60% vs. 12). If more collectors are used or the diverter grid made more tenuous for instance, then waste heat can be reduced even further.

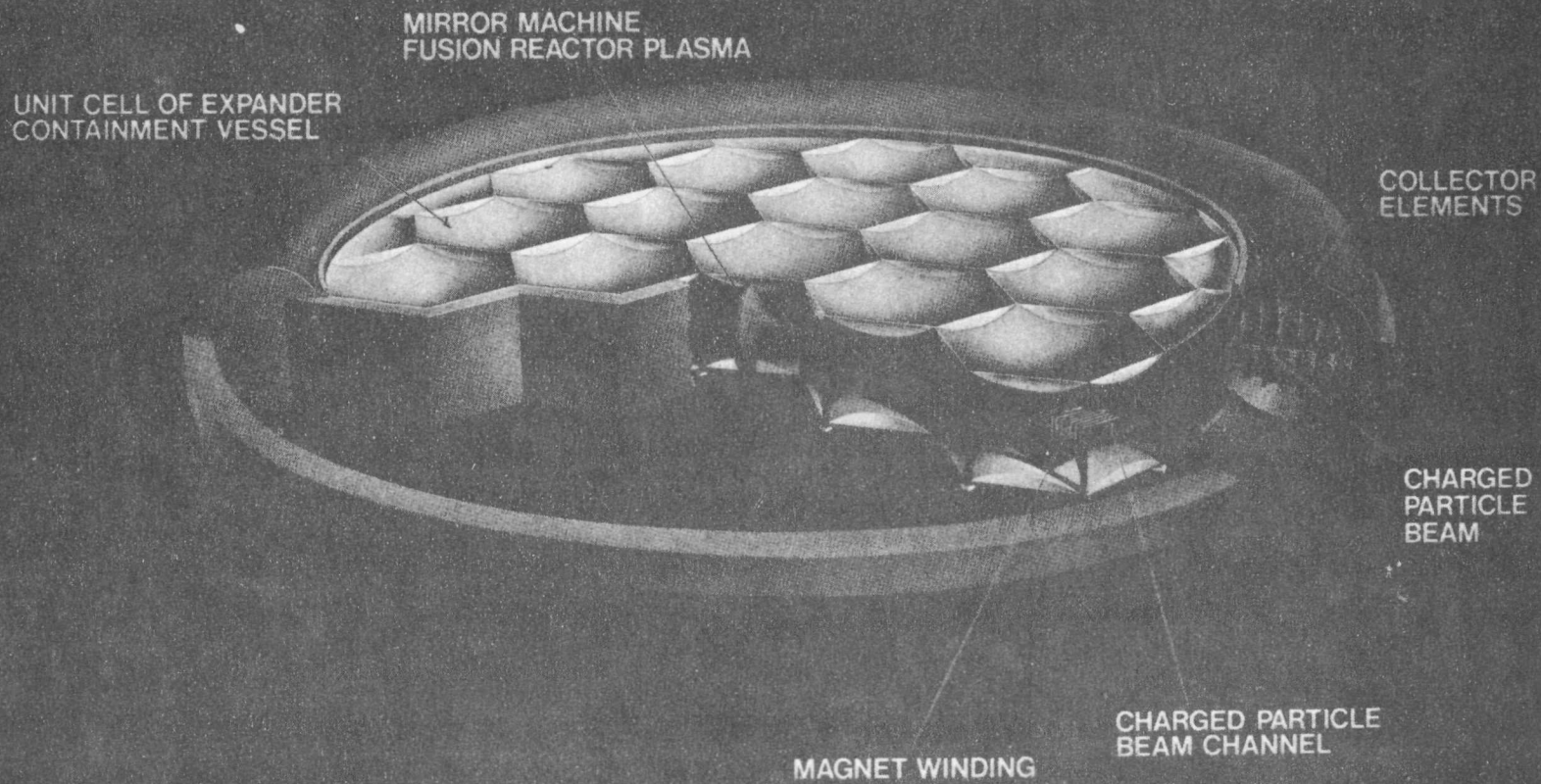
We have examined in a preliminary way the engineering problems associated with a direct converter and can say unequivocally that none presents a problem that is not resolvable within current technology. As far as costs are concerned, the direct converter, although admittedly large, has no elements within it that have critical dimensions. It is our

opinion that tolerances in fabricated parts are not at all critical. Consequently, costs are attractive and we have estimated that for a 1000 MW system, the direct converter can be produced for approximately \$20/kw. This compares with \$70-80/kw for conventional systems. The final figure shows a conceptual 1000 MW direct converter.

#### REFERENCES

- 1 R. F. Post, "Mirror Systems: Fuel Cycles, Loss Reduction and Energy Recovery," Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, England, 1969.
- 2 R. F. Post, "Critical Conditions for Self-Sustaining Reactions in the Mirror Machine," Nuclear Fusion: 1962 Supplement, Part 1.
- 3 S. Glasstone and R. Lovberg, Controlled Thermonuclear Reactions, D. Van Nostrand Co., Inc, Princeton, New Jersey, 1960.
- 4 R. F. Post, et al, "Preliminary Report of Direct Recovery Study," UCID-15650, May, 1970. Lawrence Radiation Lab., Livermore, Calif.
- 5 J. R. Pierce, Theory and Design of Electron Beams, D. Van Nostrand Co., Inc, Princeton, New Jersey, 1954.
- 6 T. K. Fowler and M. Rankin, "Fusion Energy Balance in Mirror Machines," Plasma Physics (Journal of Nuclear Engineering), 1966, Vol 8 p 121-128 Pergamon Press.
- 7 J. D. Lee - Personal Communication
- 8 Engineering Considerations of an Expander Collector Structure, University of California, Lawrence Radiation Laboratory internal report.

# FUSION REACTOR DIRECT ENERGY CONVERTER



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Figure 15

HEAT PIPES FOR RECOVERY OF TRITIUM IN  
THERMONUCLEAR REACTOR BLANKETS\*

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Abstract

Controlled thermonuclear reactors, using deuterium-tritium as a fuel for the fusion reaction, require a means of regenerating tritium so that cycle **continuity** is maintained. This paper suggests a unique way for satisfying the tritium needs. It is proposed that heat pipes using sodium as a working fluid be used as tritium transporters in the blanket structure of a fusion reactor. The tritium produced by the reactions  $(n + {}^6\text{Li} \rightarrow \text{T} + {}^4\text{He} + 4.8 \text{ MeV})$  and  $(n + {}^7\text{Li} \rightarrow n^1 + \text{T} - 2.47 \text{ MeV})$  in the lithium moderator of the blanket would be diffused through the heat-pipe tube wall and transported within the heat-pipe body to an accessible processing point outside the blanket. By diffusion through the tube wall to a vacuum environment for the second step, the tritium would be brought outside the heat-pipe body and then processed for re-cycling. Heat pipes are explained as heat-transfer devices and as gas handlers in a fusion reactor environment.

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\*This work was performed under the auspices of the U. S. Atomic Energy Commission.

# FUEL BURNUP AND DIRECT CONVERSION OF ENERGY IN A DT PLASMA\*

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## Abstract

In feasibility studies for thermonuclear reactors it is of interest to set up mathematical models for studying the various physical phenomena which take place in the fuel plasma. Among these phenomena are the following: 1) The rate of fuel burnup, 2) the heating of the plasma by the reaction products, 3) direct conversion of energy by the expansion of the plasma against the containing field, and 4) the possible development of instabilities in the plasma at various stages in the burn process. To get a detailed theoretical understanding of these phenomena it is necessary in some sense to solve for the distribution functions of the various species present in the plasma. The numerical methods which have been developed for this purpose fall roughly into two categories: 1) solving the transport equations numerically as integro-differential equations and 2) the computer simulation, or Lagrange-particle method. Application of these methods to various problems are discussed in Methods of Computational Physics, Vol. IX.

For plasmas, the first method usually involves the numerical solution of the Fokker-Planck equation. A problem ideally suited to the use of this method is the Fuel Burnup and Direct Conversion of energy problem. In this problem one may assume to a reasonable approximation that the distribution function of the various species are isotropic in velocity space. Under

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\*This work performed under the auspices of the U. S. Atomic Energy Commission

this assumption the only independent variables remaining in the equation are the velocity magnitude  $v$ , and the time  $t$ . The resulting system of coupled, nonlinear equations is easily solved by standard numerical techniques. Calculations of this type were reported at the CTR conference at Culham in September 1969. Calculations with an additional independent variable in phase space can be done by such methods, but already the calculations become considerably more involved and further increases in dimensionability seem to 'be out of reach' at present.

There are many situations in which one needs to compute to some approximation, problems with more than two independent phase space variables. The Lagrange particle or simulation model allows the numerical solution of such problems. The plasmas treated this way have been for the most part collisionless plasmas, or plasmas which may reasonably be assumed to obey the Vlasov equation. In problems relevant to the full time scale of thermonuclear burn phenomena the collisionless model is not adequate. In such cases it is necessary to use methods for simulating binary collision effects. Such methods are in the process of being developed. In particular, it has been found possible to set up a simulation of classical resistivity and electron runaway. This method works for any degree of approach to equilibrium including highly nonequilibrium cases.

STATUS REPORT ON FAST BREEDER REACTORS (Why Breeders?)

by

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This paper was presented at the Symposium on Fusion  
Reactor Design, June 2-5, 1970, at Texas Tech  
University, Lubbock, Texas

## STATUS REPORT ON FAST BREEDER REACTORS (Why Breeders?)

World reserves of low cost uranium and growth of nuclear power require commercial fast breeder reactors some time between 1980 and 1990. High gain breeders consume small amounts of uranium and therefore will extend existing resources of economic fission fuels by orders of magnitude. While development work on fast breeders has been going on for approximately two decades, large electric power producing breeders are still some years in the future. Technical feasibility of fast breeders has been proven, but competitiveness of large breeders with more conventional fossil fired or thermal reactor plants is still to be demonstrated. Advanced breeders, such as gas-cooled fast reactors with carbide fuels and direct cycle gas turbines, promise to have power costs comparable to the most optimistic estimates made for fusion reactors by the end of this century.



## INTRODUCTION

It has been recognized by early nuclear power pioneers such as E. Fermi and W. Zinn that fast neutron fission reactors could breed more fissile material than they consume, and that such breeding would eventually become necessary to extend the known resources of uranium for economic electric power. As a matter of fact, the first fast reactor, Clementine, was operated in 1946 at Los Alamos with mercury cooling. It is interesting to remember that the first electric power from fission came from the first fast breeder reactor: EBR I produced 100 Kw of electrical power in 1951 at Arco, Idaho.

Since that time, a large amount of research and development has been carried on around the world in the fast breeder field, mostly in USA, USSR and Western Europe. While a number of experimental facilities have been operated, the first large electric power producing demonstration plants are only now being built. A significant amount of development is still necessary to transform those fast breeders into commercial power plants. The timetable is still somewhat uncertain and varies from one country to another, but it is generally agreed that by the mid-1980s, large fast breeder plants should be economically competitive.

In our report, we are going first to look at the requirements for electric power and more specifically for nuclear power and consider the fuel requirements and the resources available. We will then briefly mention the principles of fast and thermal breeders, and compare the two main types of fast breeders (LMFBR and GCFR). We will try to explain the main reasons for worldwide interest in FBR and state the main remaining development problems. We will then recall the historical evolution of FBR and describe the existing programs. Finally we shall make an attempt at cost comparison between fission and fusion reactors and suggest a timetable for those various concepts.

## ELECTRIC POWER REQUIREMENTS

The increasing needs for electric power around the world lead to doubling times for power generation between 7 and 10 years. The fraction of total power requirements corresponding to electric power is also rapidly increasing. In the next thirty years (1970-2000), the US electric industry will build nearly ten times as much generating capacity as exists today. The present thermal generating capacity in the US is expected to increase from 270 000 MWe in 1970, to over 1 100 000 MWe by 1990 and 2 300 000 MWe by 2000 [Ref. 1]. The electrical energy generated in thermal plants which was  $600 \times 10^9$  Kwh in 1960, and is  $1300 \times 10^9$  Kwh in 1970, is expected to increase to over  $5500 \times 10^9$  Kwh by 1990. Therefore the level of fuel consumption in 1990 will be over four times the level in 1970. The annual fuel needs in millions of tons of coal (or coal equivalent) will increase from about 535 in 1970 to over 2200 by 1990.

The present fuels for thermal plants producing electric power are now gas, oil, coal and uranium. In the US in 1970, most of the electrical energy comes from coal (about  $300 \times 10^6$  ton/year), oil (about  $270 \times 10^6$  bbl/year), and natural gas ( $3 \times 10^{12}$  ft<sup>3</sup>/year). Present forecasts show a decrease in coal uses by 1990 ( $\sim 226 \times 10^6$  ton/year) and an increase in oil needs (by a factor of about 2.4) and gas needs (about doubled). This forecast assumes that by 1990 approximately half of the electrical generation capacity of thermal plants in the US will come from nuclear energy (500 000 to 600 000 MWe nuclear) and that about 70% of the total electric energy generated will come from nuclear plants. The nuclear fuel requirements in net tons of  $U_3O_8$ , increase from approximately 8000 in 1970, to 52 000 in 1980 and 90 000 in 1990. By the end of this century, it is expected that about half of the total US energy requirements will be provided by electricity, i.e., more than one-third of the total energy produced will come from nuclear power.

While the previous data correspond to US needs, similar requirements exist in the Western world. For instance, the annual needs in EEC countries will be about 30 000 metric tons of natural uranium in 1975; by 1980, they should be between 56 000 and 80 000 tons [Ref. 2]. For 310 000 to 410 000 MWe of installed nuclear power in the year 2000 in EEC countries, a total

of 325 000 to 525 000 metric tons of natural uranium will be required between 1970 and 2000.

Let us recall that the forecasts for nuclear power growth in the US show about 150 000 MWe installed by 1980, 500 000 MWe by 1990 and  $10^6$  MWe by the year 2000 [Ref. 3]. Figure 1 from the EEI Fast Breeder Reactor Report for 1968 shows that the US resources at less than \$10/pound of  $U_3O_8$  are about  $0.66 \times 10^6$  short tons. Those low cost resources should be exhausted by about 1985, even with plutonium recycle in light water reactors. The total estimated resources below \$15/pound of  $U_3O_8$  of about  $10^6$  short tons would be reached some time around 1990. While those 1968 AEC estimates are subject to increases as new reserves are discovered, large investments in exploration, mining, milling and diffusion plants would be needed to meet the increasing ore requirements of LWR; without breeders, a total of about  $3 \times 10^6$  short tons of  $U_3O_8$  would be needed by the year 2000. With the introduction of advanced converters such as HTGR in the 1970s, the need for uranium ores will already be reduced significantly since new fissile material (U-233) is bred in another material (thorium) and the total uranium requirements of HTGR are about half those of LWR (see Table I from Ref. 4). Furthermore, HTGR fuel cycle costs are not as sensitive to increase in ore costs as are LWR fuel cycle costs and more expensive ores could therefore be used economically in HTGRs. But, even with a mixture of HTGR, LWR and fossil-fueled plants, it is estimated that about  $2.5 \times 10^6$  tons of  $U_3O_8$  would be used in the USA in the 50-year period between 1970 and 2020. Introduction of fast breeder reactors in 1984 is estimated to reduce those requirements to  $1.1 \times 10^6$  tons of  $U_3O_8$ , which is equivalent to a saving of  $4 \times 10^{11}$  tons of coal [Ref. 5]. The annual ore requirements would peak below  $10^5$  tons of  $U_3O_8$  by 1990 and would decrease to about half this number by the year 2020.

The previous data correspond to US needs; we have already seen that similar requirements exist for Western Europe. The situation is worse there since the uranium resources are rather meager (only about 50 000 tons of natural uranium reserves in France), while the requirements are much larger. For instance, the annual needs of EEC countries by 1980 will already be

between 56 000 and 80 000 tons [Ref. 2]. It is also estimated that nuclear energy in Japan will provide 30% of the total electric energy production by 1978 [Ref. 6]. Furthermore, those countries have no industrial enrichment facilities and now have to import enriched uranium from the US, unless they build natural uranium reactors (Magnox or HWR) which do not appear to be competitive in those countries at the present time, or build their own commercial enrichment plants. Therefore breeding appears to be even more urgently needed in Western Europe and in Japan than in the USA.

### FAST BREEDER REACTOR PRINCIPLES

Natural uranium contains only 0.7% of the fissionable material U-235, the remainder being U-238. In such reactors as LWR or HTGR, the uranium fuel is enriched through the gaseous diffusion process, to about 4% of U-235 in LWR for instance. While U-235 fission occurs in the reactor, a small fraction of the fertile material is transformed into new fissile material which itself is partly consumed in the reactor: U-238 yields Pu-239 in LWR and Th-232 yields U-233 in HTGR. But, in a light water reactor, less than 5% of the total weight of fissile and fertile material in the core is fissioned before the core is removed for reprocessing; furthermore, a large amount of natural uranium depleted in U-235 has been used in the diffusion plant to produce the 4% enriched uranium. Fuel utilization is better in an advanced converter such as HTGR, approximately by a factor of 2 (see Table I), but the fission fuel is still not used as efficiently as it would be in a breeder. The breeding principle consists of transforming the fertile material (U-238, Th-232) into new fissile material (Pu-239, U-233). A breeder produces more new fissionable material than it consumes, thus leading to enormous extension of the reserves of uranium and thorium: from a few decades with converter reactors, to hundreds or perhaps even thousands of years with fast breeders [Ref. 5].

Before we look in more detail at fast breeders, let us mention that breeding is thought to be feasible in thermal reactors operating with the Th-232/U-233 cycle. The ratio of neutrons produced to neutrons absorbed

in the artificial fissile material U-233 is about 2.2 in a thermal spectrum, which gives a maximum theoretical conversion ratio of 1.2 (ratio of new fissile material to fissile material consumed). Some of the excess neutrons leak out of the reactor or are absorbed in structural materials, and thus breeding will be at best marginal in thermal reactors. There are several types of such near breeders or low gain thermal breeders. The most advanced is the High Temperature Gas Cooled Reactor moderated by graphite and cooled by helium, which now operates on the U-235/Th-232/U-233 cycle: a 40 MWe prototype has been in operation at Peach Bottom since 1967, a 330 MWe plant will be operating in Colorado in 1972 and large HTGRs (600 to 1100 MWe) are now being marketed in Europe and in the US. The existing HTGRs are non-breeders, but if the ore costs justify it, their fuel cycle could be adapted to higher conversion ratios, perhaps by addition of some BeO to the graphite moderator. Two thermal breeder concepts are under development in the US, the Light Water Breeder Reactor (at Bettis Laboratory) and the Molten Salt Reactor (at Oak Ridge National Laboratory). While their breeding gain will always be very low, economic thermal breeders could significantly extend the ore resources by utilizing the important thorium reserves (comparable to the uranium reserves). The thorium cycle has also been considered with heavy water moderation in Canada. Those thermal reactors will probably only develop as high efficiency advanced converters since, at least for the time being, their potential marginal breeding is not justified by current economics.

What are now the main characteristic features of fast breeder reactors? As indicated by their name, most fissions occur at high energies (fast spectrum), and they produce more fissile material than they consume (breeding). Compared to existing thermal nuclear reactors where most fissions occur at thermal energy, the main differences with fast breeders are

- No moderator: in order for most fissions to occur at energies from, say 0.1 Mev to a few Mev, only fuel (fertile and fissile), structural materials (fuel clad, boxes, control rods) and coolant are found in the core.

- A large number of neutrons are produced per neutron absorbed in fissile material: approximately 2.3 neutrons are produced for each neutron absorbed in Pu-239, or about 2.65 neutrons if fast fission in U-238 is taken into account.
- Some of the neutrons produced in the core are absorbed in U-238 to produce Pu-239. Others are absorbed in natural uranium or depleted uranium surrounding the core (blanket) to produce more Pu-239.
- Most nonfuel materials have low neutron absorption cross section in a fast flux as compared to fuel materials (fissile and fertile). Thus, in a large reactor surrounded by blankets, losses by leakage or parasitic absorption may be kept very low.
- The breeding ratio could thus be high, 1.3 to 1.5 with oxide fuels, 1.5 to 1.6 with carbide fuels. Hence rather low doubling times could be obtained (the doubling time is the time required by the reactor to produce enough fissionable material to refuel itself and another reactor of the same power). The doubling time of large fast breeders could be made comparable to the doubling time of electric power generation needs, i.e., 7 to 10 years.
- Since fast reactors are not moderated, their fuel inventory is high (compared to thermal reactors) and their power density is of necessity very high (an order of magnitude higher than thermal reactors).
- Control of fast reactors must also take into account the fact that the characteristic times (neutronic and thermal) are smaller than for thermal reactors.
- Fast breeders are also more sensitive to small dimensional changes in the core, as they affect both reactivity and cooling.

As shown in Table II, a number of fast breeder concepts could be considered with various types of fuel and fertile materials, coolants, reactor

arrangement and power cycles. For various technical and historical reasons, the main types of fast breeder presently under consideration for power reactors are (see Fig. 2)

LMFBR: Liquid metal cooled (Na) with an intermediate sodium circuit (nonactive) and a steam cycle; ( $\text{UO}_2\text{-PuO}_2$ ) fuel in the core and depleted  $\text{UO}_2$  in the blankets.

GCFR: Gas cooled (helium) with a steam cycle and the same type of fuel as LMFBR

Table III derived from Ref. 3 gives a comparison of advantages and disadvantages of the two main coolant approaches, namely sodium cooling and gas cooling. Table IV shows a simplified comparison of reactor coolants [Ref. 4]: water and helium for thermal reactors, helium and sodium for fast breeders. For typical design conditions of large LMFBRs and GCFRs, the heat transport per unit core frontal area with helium is about 80% of its value with sodium, while the heat transfer per unit fuel surface area is about 60%. Thus, while from a heat transfer viewpoint, pressurized helium is not quite as good as sodium, the gas still can effectively and economically (i.e., with reasonably low pumping power) remove the nuclear heat from the core. A comparison of typical performance of those two types of fast breeder is shown on Table V both for demonstration plants ( $\sim 300$  MWe) and large reactors ( $\sim 1000$  MWe). Comparable doubling times and cycle efficiencies are obtained with the two systems in both cases (demonstration plant and commercial reactor) and with two types of fuel (oxide and carbide). The specific inventory is lower for LMFBR (because of good sodium cooling properties), but the breeding ratio is also lower (because sodium acts as moderator); thus the doubling time is about the same for GCFR and for LMFBR.

#### INCENTIVES FOR FAST BREEDER REACTOR DEVELOPMENT

Let us restate the main reasons for the world-wide interest in the development of FBRs.

- The reserves of low cost uranium ores in the USA will probably be exhausted some time between 1985 and 1990 (see Fig. 1).

While existing light water reactor costs are very sensitive to increase in ore costs, the fuel cycle costs of advanced converters, and *a fortiori* breeders, are rather insensitive to such increase. Furthermore, the uranium requirements of fast breeders are very small, since, in principle, 100% of the uranium could be burnt, rather than only a fraction of a percent in LWRs.

- FBRs, as well as HTGRs, will have a much higher thermal efficiency than LWRs: 39 to 42% versus 32%, which reduces the thermal effects of the nuclear power plants. The steam conditions will be comparable to the most modern fossil-fueled plants.
- FBRs will offer a premium market for all the plutonium produced in LWRs. As shown on Fig. 3 (from Ref. 3), about 250 metric tons of plutonium will come out of US light water reactors by 1984. At a price of \$8/gram, it represents two billion dollars worth of plutonium.
- The large stockpiles of depleted uranium, estimated to be approximately 400 000 tons of U-238 by 1980, could be used as fertile material in FBRs: in a 30-year period, about 50% of the U-238 in a FBR will be converted into fissile plutonium. Thus the otherwise wasted depleted uranium could greatly increase the world energy resources.
- The projected power costs of FBRs will be lower than those of other reactors (which are already competitive with fossil fuel plants). Table VI adapted from Ref. 7 shows the range of projected costs of 1000 MWe plants after appropriate R&D programs. LMFBRs should give cheaper power than LWRs some time between 1984 and the year 2000 according to the various estimates. Similarly, LMFBRs could become competitive with HTGRs as early as 1986 by assuming high costs for HTGR and low costs for LMFBRs. By taking median values, the data from Table VI show that LMFBRs



should become competitive with LWRs by about 1989 and with HTGRs by about 1995. Similar results obtain with the Gas Cooled Fast Reactor which should be the cheapest system by the end of this century [Ref. 7] since the running costs of GCFR and LMFBR are the same and the plant costs of GCFR should be lower.

A Cost-Benefit Analysis of the US Breeder Program has been performed by the US AEC [Ref. 8]. This study compares the anticipated R&D costs of the breeder program to the expected savings in energy costs. This evaluation mainly considers the Liquid Metal Fast Breeder Reactor but it also mentions the effect of a parallel breeder program. With LMFBRs introduced in 1984, rising uranium costs, constant fossil fuel costs and a 6.3% per year growth rate of electrical energy demand (11.4 years doubling time), the benefit-to-cost ratio is 3.64 (for a discount rate of 7%). This is the ratio of anticipated savings to development costs, discounted to 1970. If GCFRs are introduced at the same time as LMFBR, the benefit-to-cost ratio may be shown to increase to 4.47, assuming a LMFBR installed capacity double that of the GCFR. Those results correspond to gross discounted savings compared to a non-breeder economy from 9 to 12 billion dollars, or more than 200 billion (1970) dollars benefits from 1984 to 2020 [Ref. 5]. In this analysis, the electrical energy demand increases from  $2000 \times 10^9$  Kwhr per year in 1980 to  $18,500 \times 10^9$  Kwhr in 2020. These studies assume a mixture of LWRs, HTGRs and FBRs introduced at different rates and dates.

#### DEVELOPMENT PROBLEMS OF FAST BREEDER SYSTEMS

The development problems of the two FBR systems are somewhat different, except for fuel development since both LMFBR and GCFR use the same type of  $UO_2$ - $PuO_2$  stainless steel clad fuel pins. Tables VII and VIII show the status of FBR technologies [Ref. 9] and the major development areas for LMFBR and for GCFR [Ref. 3]. No basic feasibility problem is involved, and those developments are only required to obtain economic power from fast breeder power plants. FBR developments, system and component design, fuel

considerations, and core heat removal were discussed at the 1968 American Nuclear Society Meeting in Washington and are reported in Refs. 10 and 11. Before we emphasize the most important remaining problems, let us briefly describe these two FBR systems.

Figure 2 shows very schematically the sodium and the gas fast systems. In the LMFBR, the reactor core is cooled by a flow of primary sodium which exchanges its heat in a primary heat exchanger with a secondary sodium coolant. This nonradioactive secondary sodium raises steam in another heat exchanger and the steam expands in a steam turbine driving a generator. The primary sodium loop which has to be heavily shielded is sometimes contained in a large pool of sodium where one finds the reactor core, blankets and shields, the primary pumps, and the primary heat exchangers. In a GCFR, the primary loop comprising the reactor core and internals, the helium circulators and the steam generators, is contained within a Prestressed Concrete Reactor Vessel which holds the gas pressure (70 to 120 atm). The steam raised in the heat exchanger then expands in a steam turbine. In some advanced concepts, the hot gas out of the reactor expands directly into gas turbines which drive both helium compressors and generators [Ref. 12]. As previously mentioned, the cores of the LMFBR and GCFR are very similar, consisting of assemblies of vertical fuel elements, each containing a number (200 to 300) of fuel pins cooled by axial flow of sodium or helium respectively.

The advantages claimed by the proponents of these two systems, which both have high thermal efficiency and potentially low doubling times, are mainly

- LMFBR: a) Good heat transfer characteristics with low coolant pressure
- b) Potential for low specific fissile inventory
- c) Good emergency cooling possibilities, especially if the core can always be kept under sodium.
  
- GCFR: a) Inert, and transparent coolant which does not interfere with core neutronics
- b) Potential for high breeding gain

- c) Simplified operation and maintenance, and no need for intermediate coolant loop.

The GCFR design contemplates commonality of fuel with LMFBR, obtained by equalizing the pressure on both sides of the cladding [Ref. 13]. GCFR fuel will use artificial surface roughening to improve the surface heat transfer coefficient [Ref. 14]. Furthermore, as indicated in Table VII, a great deal of the HTGR technology is applicable to GCFR.

Besides the fuel problems which are common to both systems and will be discussed later on, the special development problems seem to be in the area of steam generators for sodium cooling and helium circulators for gas cooling. In view of the chemical reactions between sodium on one hand, water, steam and air on the other hand, the design of reliable, safe and economic sodium-cooled steam generators is a difficult problem which is now under development in a number of countries, such as France, Germany, Japan, UK, US and USSR [Refs. 10 and 11]. Core heat removal does not appear to be a problem in helium-cooled fast breeders under normal operation, but cooling must be insured under all circumstances, including the removal of afterheat once the reactor is shut down. Therefore reliable and redundant helium circulators (and drives) must be provided. A large amount of circulator design and testing is now taking place for thermal helium cooled reactors (HTGR) which should also be applicable to fast systems [Ref. 15]. Extension of the PCRV experience is also required since helium pressure in a GCFR is about double that found in thermal systems [Ref. 12]. Both LMFBR and GCFR require fast and reliable in-core instrumentation, although of a different nature: thermocouples are needed in both systems for measurement of coolant temperature at the exit of the fuel elements. But sodium might boil and be ejected out of the core in case of supersaturation, and acoustic monitors may be needed for early detection of boiling in order to avoid failure propagation and the effects of a positive void coefficient of reactivity. Detailed experimental verifications of core thermal and mechanical behavior are required for both FBR systems.

The maximum fuel burnup required for large economic fast breeders varies from 7-8% to about 10%, according to European or US estimates [Ref. 7]. Satisfactory experience has been obtained in fast test facilities in the US (EBR II) and in Europe (Rapsodie in France and Dounreay in UK), but only up to 5 to 8% with mixed oxide fuels. While the desired burnup has nearly been reached, the fast fluence (which indicates the amount of radiation damage) was usually a factor 3 to 4 lower than the fluence desired for large breeders ( $2-4 \times 10^{23}$  n/cm<sup>2</sup> > 0.1 Mev). Even with demonstration plants of approximately 300 MWe, the fast flux will be only about 1/3 and the fast fluence/burnup ratio only about 1/2 of the values for 1000 MWe reactors. The Fast Flux Test Facility (FFTF) will have a fast flux comparable to that of large FBRs ( $\sim 7 \times 10^{15}$  n/cm<sup>2</sup>-sec above 0.1 Mev), but this large test reactor (400 Mwt) will only start operation in the US in 1974-75.

As shown in Table IX the design characteristics of fuel elements for fast breeder reactor prototypes are very similar, even with gas cooling. The fuel pin is small (0.5-0.6 cm pellet diameter), long ( $\sim 100$  cm) with a thin stainless steel clad (OD/ID: 1.12-1.16). The maximum linear rating is 13-18 Kw/ft (425-585 w/cm), the design hot spot clad temperature is 665-730C and the maximum burnup is 50-100 Mwd/Kg (fuel). A core plan for a 300 MWe GCFR demonstration plant is shown on Fig. 4 which is very similar to a LMFBR core.

Although fuel swells at high burnups, swelling of the stainless steel clad under high fast neutron fluences may even be more limiting. This radiation swelling of the steel depends both on fluence and temperature and may therefore cause serious distortion and bowing of the fuel elements. Close to 10% volume expansion of 20% cold-worked 316 stainless steel could be expected at fluences of about  $10^{23}$  n/cm<sup>2</sup> (>0.1 Mev) at 500C. Figures 5 and 6 show the strong dependence of radiation swelling upon fluence (significant only above  $\sim 0.5 \times 10^{23}$  n/cm<sup>2</sup>) and upon temperature (maximum for stainless steels around 450C). Some metallurgical improvements could be expected, but the nonuniform swelling of the fuel elements may have to be accommodated by leaving a clearance between elements (less detrimental to

the neutron economy in GCFR than LMFBR) and by periodic rotation of the fuel elements [Ref. 16].

#### HISTORICAL FBR DEVELOPMENTS AND WORLD PROGRAMS

As previously mentioned, the first fast reactor (Clementine) was operated in the US in 1946. Since then a number of fast reactors have been operated or are being built in the US and abroad as seen on Table X (from Ref. 8). The first fast breeder, EBR-I was started in the US in 1951 while the first sodium cooled breeders in the USSR and in the UK were started in 1959 (BR 5 and Dounreay respectively). The first fast reactor in France, Rapsodie, went critical in 1967 and has recently been uprated from 20 to 40 Mwt. Some characteristics of those first generation fast breeder reactors are shown in Table XI (from Refs. 10 or 11). Most of these first reactors use metallic fuel, except for Rapsodie which uses  $\text{PuO}_2\text{-UO}_2$  fuel, and BR 5 which had a  $\text{PuO}_2$  core and also a UC core; those fast reactors are cooled by liquid metal, Hg, NaK or Na.

Table XII updated from Ref. 3 shows the milestones of the LMFBR programs around the world. Germany, together with Belgium and Holland, Italy and Japan have started ambitious FBR programs which follow by a few years the programs already started in France, UK, USA and USSR. It may be noted that, although the first demonstration plant went critical in 1963 (Enrico Fermi in the US), the next demonstration plant is not scheduled to start until 1970 (BN 350 in the USSR). Figure 7 shows the Fermi reactor which has not yet operated at its design power (200 Mwt) since some of the fuel elements experienced melting in 1966. Table XIII shows the second generation of LMFBRs, which are demonstration plants with electric power production in the 250 to 600 MWe range, and not commercial reactors. All of them will have stainless clad  $\text{UO}_2\text{-PuO}_2$  fuel cooled by sodium. As shown in this table, the net cycle efficiency is high (35 to 42%), and the power density, specific power and linear rating are very similar for all plants. In most cases, the breeding ratio is rather low and therefore the doubling time is high. The assumed maximum burnup varies from 50 to 100 Mwd/Kg. Both pool and loop designs are utilized. Figure 8 shows an example of the pool concept

for the PFR under construction in the UK. Figure 9 from Ref. 17 shows a concept of a 1000 MWe sodium-cooled fast breeder with the same pool design, which should have a net efficiency of 41%, a specific power of 1.3 MWt/Kg(fissile) and an average power density of  $\sim 0.6$  MWt/liter (of core). With a breeding ratio of 1.4 with  $\text{PuO}_2\text{-UO}_2$  fuel, the doubling time should be approximately 8 years.

Rather large sums of money are being spent now on LMFBR programs: close to \$125 millions per year in the US and about the same amount abroad, mainly in France, Germany, Italy, Japan and UK. The spending level is also high in the USSR where two large reactors are presently under construction, BN 350 and BN 600. It is felt that total expenditures of 2 to 3 billion dollars would have to be spent in the US before the advent of true commercial breeders some time around 1990. As could be seen in Table XIII, the development philosophy is quite different in the US and abroad. A conservative approach is taken by the US AEC [Ref. 18] whereby large component development and fuel testing programs (including full-size loops in a Fast Flux Test Facility) precede the construction of one or more demonstration plants. In other countries, such as France, Germany, UK and USSR, construction of demonstration plants follows closely the operation of first generation reactors (Rapsodie, DFR); a great deal of the fuel testing will be performed in the demonstration plants which therefore are expected to go up to full power rather slowly. This latest approach should lead to earlier commercial development of fast breeders as shown on Table XIV established from recent estimates [Refs. 7, 11, 18, 19, 20]. The first large FBR should operate in USSR before the first US demonstration plant. There now appears to be a lag of several years between the US and the European LMFBR programs.

A review of the US fast breeder reactor program was recently published in *Energie Nucleaire* [Ref. 21]. The US FBR program was discussed by M. Shaw and J. Yevick of the US AEC and by H. Fenech of the University of California at Santa Barbara. Four US manufacturers described their LMFBR programs, and gas cooled fast reactor developments were also presented (see also Ref. 4) Reports on the US and European GCFR work were recently given at Oak Ridge National Laboratory [Ref. 22] and are discussed by C. Rennie [Ref. 23].

The potential of the GCFR system has previously been discussed in two US AEC reports on alternate breeders [Refs. 24 and 25]. The utilization of a closed cycle helium turbine coupled with a GCFR is presented in Refs. 12 and 26. Figure 10 shows a conceptual design of a GCFR 300 MWe demonstration plant [Ref. 16] whose characteristics are given in Table XV, together with typical design parameters for a 1000 MWe plant. These GCFR designs are based on LMFBR fuel technology and, as shown on Table IX, the fuel element parameters are very similar. The Prestressed Concrete Reactor pressure Vessel containing the core, blankets and shields in a central cavity and the steam generators and helium circulators in satellite cavities is similar to designs for other thermal gas-cooled reactors such as AGR or HTGR. Design and safety studies for a GCFR demonstration plant are being pursued by Gulf General Atomic Incorporated together with 41 US and one foreign utility company. Some fuel irradiation is being performed under US AEC sponsorship and work in heat transfer and fluid flow is under way in a joint program with the Swiss Federal Institute (EIR). Besides government studies in Europe, a Gas-cooled Breeder Reactor Association has been set up in Brussels in 1969 by 11 industrial groups, utilities, and government organizations with members from seven European countries. It appears that the development of an alternate breeder system could proceed with a timetable not very different from that of the LMFBR, i.e., a large commercial GCFR with steam cycle could be in operation some time between 1985 and 1990.

## CONCLUSIONS

We have tried to show the status of fast breeder reactors which have now been under development for about 20 years in this country and will probably be built in large numbers by utilities some time in the coming twenty years. The timetable for introduction of FBRs is different in Europe and in the US because of different needs. Uranium and thorium reserves are much larger in the US where enrichment facilities are also readily available and therefore it does not appear that there is an urgent need for breeders in this country before about 1990, while they may be needed in Europe a few years before (1980-1985). Most of the development work up to now has been performed for Liquid Metal Fast Breeder Reactors, but Gas-Cooled Fast breeder Reactors based on similar fuel technology will benefit from the advent of High Temperature Gas-cooled Reactors and should be competitive at about the same time. Fast breeders will considerably extend the use of fission fuels (uranium and thorium) since they make a very efficient use of fissile material and breed new fissile material from fertile material. The plutonium produced by existing Light Water Reactors will be used in FBRs and thermal and fast reactors will probably coexist for a number of years, together with fossil fuel plants (especially for small plant sizes).

It is interesting to speculate, now that we have attempted to answer the question *why breeders?* how those advanced fission reactors will fit in the timetable for fusion reactors. In an article published in the June 1970 issue of Fortune [Ref. 27], entitled "The Hot New Promise of Nuclear Power," Mr. T. Alexander writes "The closest to hand is the fast breeder fission reactor, the technical and economic feasibility of which seems assured. The other, and potentially far better, is controlled thermonuclear fusion. In some ways complementary to the breeder, in some ways competitive, the fusion principle is inherently safer and cleaner. And if direct conversion of fusion energy to electricity could be achieved, as many now believe, civilization might obtain its energy at a price far lower than ever imagined." He continues by stating that a Controlled Thermonuclear Reactor has not yet reached the stage of the first fission reactor in 1942.



The timetable for fusion reactors varies according to the degree of optimism, but it was pretty much agreed at the Culham meeting [Ref. 28] that a true controlled nuclear reaction (producing more power than is provided to it) could be expected to occur during this decade. In view of the difficult problems of economic fuel injection and material behavior under high neutron bombardment [Ref. 29], especially for the most promising (D-T) reaction, it is questionable that an acceptable commercial fusion reactor could be built before the end of this century. As we have seen, there is no immediate need for fusion reactors from the viewpoint of fuel resources. It is true that the inventory of radioactive materials will be smaller in a fusion reactor than in a fast breeder of the same power, and also that a fusion reactor does not contain several critical masses. But a large amount of radioactive tritium circulates in the various loops of a fusion reactor and the materials in the blankets (niobium or molybdenum) become quite active. Therefore fusion reactors will also most probably require a secondary containment like fast breeders. The much lower thermal pollution ( $\sim 90\%$  overall efficiency) claimed for fusion reactors will only be obtained in the next century with direct conversion of thermonuclear power to electricity from such reactions as (D-He<sup>3</sup>) which require a much higher ignition temperature than (D-T) reactions. In the present systems with intermediate fluids such as lithium [Refs. 30, 31] or helium [Ref. 32], and a steam turbine or gas turbine cycle, the net efficiency of fusion systems is comparable to that of advanced fission reactors ( $\sim 40\%$ ) when the energy required for coolant pumping and fuel injection is taken into account.

At this point, it may be interesting to look briefly at the world reserves of fission and fusion fuel and also at a brief comparison of extrapolated power costs. We first remark that approximately the same amount of energy is obtainable per unit mass of fission fuel (U or Th) or fusion fuel (lithium needed to breed tritium): about 1 MW-day/gram. Table XVI derived from Ref. 33, shows that the energy content of (U + Th) and of lithium are about the same; even if more lithium were discovered, the content of natural lithium in the earth's crust (65 ppm) is only four times that of U + Th (16 ppm). Therefore the amount of energy obtainable from

the (D-T) fusion reaction is of the same order of magnitude as the energy obtainable from fission reactions, i.e., sufficient for a large number of centuries. Deuterium out of the oceans will then carry on the burden for millenia. The last remark concerns power costs; which, it is recognized, are much harder to ascertain for an unknown technology such as fusion reactors than for an extrapolation of known technology, namely fast breeders. Using numbers presented by Werner et al. in Ref. 31, the capital cost of a 4200 MWe fusion reactor is about \$235/Kwe, while the running costs, including fuel, are about 1.75 mill/Kwh: therefore the net power costs are approximately 6.45 mill/Kwh. The data given by the latest EEI report [Ref. 7] for 1000 MWe GCFR plants in the year 2000 are \$150-180/Kwe and 0.7-0.9 mill/Kwh running costs, leading to total power costs of only 3.7 to 4.5 mill/Kwh. Even using the more optimistic data from Carruthers et al. [Ref. 33], the capital costs for a 2100 MWe fusion reactor is 175-188 \$/Kwe and the running costs are about 0.21 mill/Kwh, leading to power costs of 3.71-3.97 mill/Kwh, compared to about 3.73 mill/Kwh for a fast breeder of the same electrical output. With the previous data it appears that the first large (D-T) fusion reactors could only be marginally competitive with fast breeder reactors by the year 2000. There most probably will be room for advanced thermal fission reactors, fast breeders and fusion reactors for many years, with possibly even symbiosis between the various types of reactors, such as breeding of thermal fission fuel (U-233) in fast breeders, or breeding of fast breeder fuel (Pu-239) in fusion reactors [Ref. 34].

Let us conclude by quoting from Chairman Seaborg's speech of November 26, 1969 [Ref. 5]. "If the energy potential of breeders is so good, why bother with trying to achieve controlled thermonuclear fusion?...I mentioned earlier that fission through breeders would supply us with energy for centuries or perhaps thousands of years. With the successful utilization of controlled fusion, man will have virtually an unlimited energy resource at hand. Even at a power consumption rate many times that of today's, he will have an energy reserve that will last for millions of years."

## REFERENCES

- [1] G. GAMBS. The Electric Utility Industry, Future Fuel Requirements 1970-1990. *Mechanical Engineering*, April 1970
- [2] *Nuclear Engineering International*, p. 314, April 1970
- [3] Edison Electric Institute. Fast Breeder Reactor Report, April 1968
- [4] G. MELESE-d'HOSPITAL. The Gas Cooled Fast Breeder Reactor. Gulf General Atomic Report GA-9390 (1969)
- [5] G. SEABORG. Fission and Fusion -- Developments and Prospects. US AEC Report S-4269. Speech at University of California, Berkeley, 20 November 1969
- [6] *Nuclear Engineering International*, p. 166, March 1970
- [7] Edison Electric Institute Report of the EEI Reactor Assessment Panel, April 1970
- [8] Cost-Benefit Analysis of the US Breeder Reactor Program. US AEC Report WASH 1126, April 1970
- [9] M. BLUMEYER and J. PEAK. The Breeder Reactor in Your Future -- The Gas-Cooled Fast Breeder Reactor. Gulf General Atomic Report GA-9960 (1970)
- [10] Proceedings of the International Conf. on Constructive Uses of Atomic Energy, American Nuclear Society meeting, Washington, D.C. (1968)
- [11] *Reactor Technology*, 13, No. 1, Winter 1969-1970 (Superintendent of Documents)
- [12] G. MELESE-d'HOSPITAL and P. FORTESCUE. Fast Breeder Reactors with Direct Cycle Gas Turbines. American Society of Mechanical Engineers paper 69-WA/NE-17 (1969)
- [13] P. FORTESCUE, et al. A Developmental Gas-Cooled Fast Reactor Plant. Gulf General Atomic Report GA-9413, September 1969
- [14] G. MELESE-d'HOSPITAL. Factors Affecting the Design of Gas Cooled Fast Breeder Reactors. Gulf General Atomic Report GA-8583 (1968)

- [15] *Nuclear Engineering International*, pp 1069-1093, November 1969
- [16] P. FORTESCUE and W. I. THOMPSON. The GCFR Demonstration Plant Design. Gulf General Atomic Report GA-10036 (1970)
- [17] Proceedings of the Int. Conf. on Sodium Technology and Large Fast Reactor Design. Argonne Natl. Lab. Report ANL-7520, Part II (1968)
- [18] *Nuclear News*, pp 49-53, March 1970
- [19] F. STASZESKY and C. PRUSEL. Fast Breeder Reactor Development in the United States from a Utility Point of View. American Power Conference (1970)
- [20] Proceedings of NUCLEX 69 Conference, Session No. 4 on Fast Breeder Reactors, Basel, Switzerland (1969)
- [21] *Energie Nucleaire*, pp 461-518, November 1969
- [22] *Nuclear News*, pp 23-27, June 1970
- [23] COMPTON A. RENNIE. Fast Breeder Reactors. Not LMFBR or GCFR — but FBR. April 30, 1970
- [24] Evaluation of Gas Cooled Fast Reactor. US AEC Report WASH-1089 (1969)
- [25] An Evaluation of Alternate Coolant Fast Breeder Reactors. US AEC Report WASH-1090 (1969)
- [26] L. LYS and J. BROIDO. Principal Considerations in the Design of Gas-Cooled Fast Reactors with Direct Cycle Gas Turbines. Symposium on Advanced and High Temperature Gas Cooled Reactors, Julich, October 1968. Paper SM-111/69
- [27] TOM ALEXANDER. The Hot New Promise of Thermonuclear Power. *Fortune*, pp 94-97, pp 126-132, June 1970
- [28] Proceedings of the British Nuclear Energy Society conference on Nuclear Fusion Reactors, Culham, September 1969

- [29] B. MYERS. Some Observations on 14 MeV Neutron Radiation Effects on Reactor Materials. Ibid
- [30] A. P. FRAAS and D. J. ROSE. Fusion Reactors as Means of Meeting Total Energy Requirements. American Society of Mechanical Engineers report 69-WA/Ener 1 (1969)
- [31] R. W. WERNER et al. Controlled Thermonuclear Power, UCRL-71500. Presented at IECEC Conference, Washington D.C. (1969)
- [32] G. MELESE-d'HOSPITAL and G. R. HOPKINS. Gas Cooling for Fusion Reactor Blankets. Gulf General Atomic Report GA-10004, May 1970
- [33] R. CARRUTHERS, et al. The Economic Generation of Power from Thermonuclear Fusion. CLM-R 85. (UKAEA October 1967)
- [34] L. M. LIDSKY. Fission-Fusion Symbiosis. Proc. of the BNES Conf. on Nuclear Fusion Reactors, Culham, September 1969.

TABLE I

 URANIUM UTILIZATION FOR 1000 MWe PLANTS\*  
 AT 80% CAPACITY FACTOR

	Gas Cooled Reactors				
	LWR	HWR†	Magnox†	HTGR	GCFR**
Natural U required for first core, tonnes $U_3O_8$	580	300	1130	450	700
Natural U required for yearly feed, tonnes $U_3O_8$	170	150	330	55	—
Natural U equivalent of yearly U discharge, tonnes $U_3O_8$	40	—	—	5	—
Natural U equivalent of yearly Pu discharge, tonnes $U_3O_8$	20	—	—	—	+75
Net yearly requirements tonnes $U_3O_8$	110	150	330	50	-75
Total 30-yr commitment tonnes $U_3O_8$	3770	4650	10,700	1900	-1500

Similar results obtain for LMFBRs.

- \* LWR = light water cooled and moderated reactor  
 HWR = heavy water cooled and moderated reactor  
 Magnox = CO<sub>2</sub> cooled, graphite moderated, natural uranium reactor  
 HTGR = high temperature gas cooled reactor (thorium cycle)  
 GCFR = gas cooled fast breeder reactor (PuO<sub>2</sub>-UO<sub>2</sub> fuel)

†Throw-away cycle

- \*\* Assuming fissile plutonium equal to 100% enriched U-235.  
 (Fertile material: depleted uranium ~1 tonne  $U_3O_8$ /yr)

TABLE II  
DESIGN VARIABLES FOR FAST BREEDERS

Fissile material:	U-235, Pu mixtures
Fertile material:	U-238, depleted U, Th-232
Type of fuel:	metal, oxide, carbide
Reactor coolant:	liquid metal (Na, NaK), gas (He, CO <sub>2</sub> ) steam
Reactor arrangement:	integrated concept (pool), loop design
Power cycle:	direct (gas turbine), indirect (gas/steam) intermediate loop (Na/Na/steam)

TABLE III

Comparison of FBR Coolant Approaches  
(from EEI, April 1968)

Sodium-Cooled	Gas-Cooled
<b>ADVANTAGES</b>	
1. Good emergency and post-accident cooling	1. Chemically and neutronically inert
2. Low pressure	2. No coolant phase-change void effect
3. Extensive fast reactor fuel and component experience	3. High internal conversion ratio and long refueling intervals
4. Good heat transfer characteristics	4. Potential high breeding ratio and short doubling time
5. Vented fuel possible	5. Vented fuel possible
6. Low specific inventory potential	6. Small positive void coefficient
<b>DISADVANTAGES</b>	
1. Opaque fluid	1. No fast reactor fuel and component experience
2. Chemically and neutronically active	2. Emergency cooling problem
3. Phase change-void effect	3. Heat transfer limited
4. Secondary heat transport system required.	4. Stringent leak requirements
5. Component development required	5. Component development required
6. Lack of electric utility experience with sodium	6. High coolant velocity
	7. High pressure



TABLE IV  
SIMPLIFIED COMPARISON OF REACTOR COOLANTS

COOLANT	WATER		HELIUM		SODIUM
	(PWR)	(HTGR - GCFR)			(LMFBR)
Reactor Type					
Average temperature (°C)	300	600	450		450
Average pressure (atm)	150	50	100		10
$\Delta T$ (coolant rise) °C	25	400	300		165
Density $\rho$ (g/cm <sup>3</sup> )	0.70	0.003	0.0062		0.80
Specific heat $C_p$ (wsec/g-°C)	5.2		5.2		1.25
Flow velocity $V$ (cm/sec)	400	5000	8000		600
Heat transport per unit frontal area $\rho V C_p \Delta T$ (w/cm <sup>2</sup> )	36,000	31,000	78,000		100,000
Heat transfer coefficient $h$ (w/cm <sup>2</sup> -°C)	3	0.3	1.5*		10
Average film drop $\Delta t$ (°C)	25	200	100		25
Heat transfer per unit surface area $h \Delta t$ (w/cm <sup>2</sup> )	75	60	150		250

\*With artificial surface roughening doubling the heat transfer coefficient.

TABLE V

## TYPICAL PERFORMANCE OF FAST BREEDER REACTORS

	DEMONSTRATION PLANT*		COMMERCIAL REACTOR**			
	Na	He	Na	He	Na	He
Coolant						
Fuel	UO <sub>2</sub> -PuO <sub>2</sub>		UO <sub>2</sub> -PuO <sub>2</sub>		UC-PuC	
Power (MWe)	250-500	310	1000	1000	1000	1000
Inventory $\frac{\text{Kg-fiss(core)}}{\text{MWe}}$	2.5-3.5	4.4	1.55	2.6	1.45	2.0
Breeding ratio	1.1-1.3	1.33	1.31	1.51	1.50	1.60
Doubling time (year)	15-20	~20	8	8	4.5	4.5
Coolant exit temp. (C)	540-590	540	650	635	585	590
Cycle efficiency	0.39-0.41	0.38	0.43	0.40	0.39	0.38
Capital costs (\$/Kwe)			150	130	150	130
Fuel costs (mill/Kwh)			0.8	0.8	0.4	0.4
Power costs (mills/Kwh)			4.2	3.8	3.8	3.4

\* LMFBR demonstration plants in France, Germany, UK and USA. GCFR designs from GGA.

\*\* The costs are taken from US AEC studies (1969)

TABLE VI  
PROJECTED NUCLEAR POWER COSTS (1970 dollars)\*

Year of Startup	LWR	HWR	HTGR	LMFBR	GCFR	MSBR
<u>1975</u>						
Plant (\$/Kwe)	200-240	280-340	230-270			
Energy (mill/Kwhr)	6-7	6.8-8.2	6.1-7.1			
<u>1980</u>						
Plant (\$/Kwe)	170-210	270-330	180-220	Demon- stration plant		
Energy (mill/Kwhr)	5.2-6.2	6.6-8.0	5.1-6.1			
<u>1985</u>						
Plant (\$/Kwe)	160-190	220-260	160-190	220-260	Demon- stration plant	
Energy (mill/Kwhr)	4.9-5.7	5.5-6.5	4.6-5.4	5.5-6.5		
<u>1990</u>						
Plant (\$/Kwe)	155-185	200-240	150-180	180-220	225-275	Demon- stration plant
Energy (mill/Kwhr)	4.8-5.6	4.8-5.8	4.3-5.1	4.5-5.5	5.6-6.8	
<u>2000</u>						
Plant (\$/Kwe)	150-180	180-200	135-165	160-190	150-180	150-180**
Energy (mill/Kwhr)	4.7-5.5	4.2-5.2	4.0-4.8	3.9-4.7	3.7-4.5	3.7-4.5

\* from Ref. 7, Edison Electric Institute 1970 report

\*\* beyond 2000

TABLE VII  
Status of FBR Technologies

Technological area under study or development	GCFR	LMFBR
Coolant	Considerable operating experience with British reactors (CO <sub>2</sub> ) and US and European HTGR (helium)	Operating experience with EBR-II, SEFOR and Fermi in the US, DFR in England and Rapsodie in France
Fuel	GCFR fuel design will be based primarily on LMFBR experimental data. Heat transfer results with roughened surfaces available mostly from European GCRs.	Experimentation is under way. High burnup on fuels will be tested upon completion of the Fast Flux Test Facility and the demonstration plant.
Pressure vessels	Considerable PCRV experience in England and Europe. Fort St Vrain should provide any added US experience required.	High pressure primary containment not required.
Pumps and circulators	HTGR experience will be directly applicable to GCFR. Fort St Vrain circulator development program has been completed. Also, UK and European gas-cooled reactors from which to draw experience.	Both mechanical and electromagnetic pumps are under study. Mechanical pumps seem to be favored. Pumps to about 20,000 gpm will be needed for demonstration plants. Larger sizes will be needed for 1000 MWe plants. At present, 12,000 gpm is largest size operated.
Steam generators	Steam generators for Peach Bottom HTGR are operating satisfactorily at design output. Steam generators for use at Fort St Vrain are of a different design. Final proof testing of the design will occur when Fort St Vrain starts up in 1971. Since GCFR steam generators will be the same design principles as the Fort St Vrain generators, Fort St Vrain will provide most of the necessary steam generator experience.	Present steam generators use bi-metallic tubes to mitigate corrosion problems. Only the Fermi reactor has a single tube steam generator and it has a history of leakage. However, successful single tube steam generators will probably be required for economical operation. Considerable amount of sodium steam generator testing now under way.

TABLE VIII  
 MAJOR DEVELOPMENT AREAS FOR LMFBR AND GCFBR\*  
 (from Ref. 3)

Item	Na	Gas		Na	Gas
<u>Physics</u>			<u>Fuel and Materials</u>		
Large Pu criticals	X	X	Fuel irradiation	X	X
Doppler confirmation	X	X	Clad, structure swelling	X	X
Data methods	X		Fuel/cladding interaction	X	X
Coolant reactivity			Cladding support/cycling		X
Void coefficients	X				
<u>Safety</u>			<u>Instrumentation/Control</u>		
Realistic DBA**	X	X	Fuel failure monitor	X	X
Loss of coolant		X	Impurity monitoring	X	
Emergency decay cooling		X	Boiling detection	X	
Coolant voiding effects	X				
Fuel failure propagation	X	X	<u>Coolant Technology</u>		
			Chemistry and Purification	X	
			Na-H <sub>2</sub> O reaction	X	
<u>Core Design</u>			<u>Fuel Recycle</u>		
Fuel element structure		X	Fuel fabrication	X	X
S/A irradiation testing	X	X			
Coolant boiling effects	X				
Heat transfer/hydraulics		X			
Core clamping and hold-down	X	X			
<u>Plant Design</u>					
Emergency decay coolant		X			
Temperature optimization	X	X			
Fuel transfer system	X	X			
System transients	X	X			
<u>Components</u>					
Steam generator	X	X			
PCR/V		X			
Pumps/blowers	X	X			
Fuel handling	X	X			

\* Areas where effort is required are indicated by an X.

\*\* Design Basis Accident

TABLE IX

CHARACTERISTICS OF STAINLESS STEEL CLAD, MIXED OXIDE FUEL ELEMENTS  
FOR FAST BREEDER REACTOR PROTOTYPES

	PFR UK	Phenix France	SNR Germany	BN-350 USSR	BN-600 USSR	GE USA	AI USA	Westinghouse USA	GGA USA
Coolant	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Sodium	Helium
Power, MWe	250 (275)*	233	295	350	600	310	500	300	~310
Pellet diameter, cm	0.50	0.55	0.52	~0.52	0.60	0.51	0.54	0.56	0.62
Cladding OD/ID	1.15	1.16	1.15	1.13	1.135	1.16	1.135	1.125	1.15
237 Maximum linear rating, Kw/ft	13.5	~14.5	~17.5	14	13 (av)	15	15	13.5	13.8
Hot spot cladding temperature at mid-radius, C	650 (700)*	700 (730)*	700	710	690	~700	~700	665	700
Maximum burnup, 10 <sup>3</sup> MWd/tonne	70	≥50	85	≥50	100	100	75 (av)	100	50-100

\* For later cores.

TABLE X  
LIQUID METAL COOLED FAST REACTOR PROJECTS (from Ref. 8)

<u>Name</u>	<u>Country</u>	<u>Power (Mwt)</u>	<u>Coolant</u>	<u>Initial Operation</u>
Clementine	United States	.025	Hg	1946
EBR I	United States	1.0	NaK	1951
BR 2	USSR	0.1	Hg	1956
BR 5	USSR	5.0	Na	1959
Dounreay	Great Britain	60.0	NaK	1959
LAMPRE	United States	1.0	Na	1961
EBR II	United States	60.0	Na	1963
Fermi	United States	200.0	Na	1963
Rapsodie	France	20.0 <sup>1</sup>	Na	1967
SEFOR	United States <sup>2</sup>	20.0	Na	1969
BR 60 (BOR)	USSR	60.0	Na	1970
BN 350	USSR	1000.0	Na	1970
PFR	Great Britain	600.0	Na	1972
JEFR	Japan	100.0	Na	1973
Phenix	France	600.0	Na	1973
FFTF	United States	400.0	Na	1974
PEC	Italy	140.0	Na	1975
NA 2	W. Germany <sup>3</sup>	750.0	Na	1975
BN 600	USSR	1470.0	Na	1976

<sup>1</sup>Being increased to over 40 Mwt in 1970.

<sup>2</sup>With Germany and Euratom.

<sup>3</sup>With Belgium and Netherlands.

TABLE XI

## First Generation Fast Breeder Reactors

	USA				USSR			UK	France
	CLEMEN- TINE	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR	RAPSODIE
<i>Reactor Power</i>									
Thermal, MW(th)	0.025	1.2	62.5	200	0	0.1	5	72	20
Electrical, MW(e)	0	0.2	20	66	0	0	0	15	0
<i>Core</i>									
Fuel	Pu-Metal	U-Metal	U-Metal	U-Metal	Pu-Metal	Pu-Metal	PuO <sub>2</sub>	U-Metal	PuO <sub>2</sub> /UO <sub>2</sub>
Core volume, liters	2.5	6	65	420	1.7	1.7	17	120	54
Fuel rating av, MW(th)/kg fissile	0.0016	0.02	0.3	0.37	0	0.008	0.1	0.24	0.14
Power density av, MW(th)/liter	0.01	0.17	0.8	0.45	0	0.06	0.3	0.5	0.32
Linear rod power max, W/cm (av 50)	300	300	450	250	0	150	200	(av 320)	(av 210)
Neutron flux max, n/(cm <sup>2</sup> sec) (av 5 × 10 <sup>12</sup> )	1.1 × 10 <sup>14</sup>	1.1 × 10 <sup>14</sup>	3.7 × 10 <sup>15</sup>	4.7 × 10 <sup>15</sup>	5 × 10 <sup>10</sup>	1 × 10 <sup>14</sup>	1 × 10 <sup>15</sup>	2.5 × 10 <sup>15</sup>	1.8 × 10 <sup>15</sup>
<i>Primary Heat Transfer System</i>									
Coolant	Hg	NaK	Na	Na	-	Hg	Na	NaK	Na
Coolant temperature									
Core inlet, °C	40	230	370	290	-	30	375	200	410(450)
Core outlet, °C	120	320	470	430	-	60	450(500)	350	500(540)
Coolant mass flow, m <sup>3</sup> /h	0.6	80	2200	5500	-	6	240	1800	800
Number of coolant loops	1	1	2	3	-	1	2	24	2
<i>Time Schedule</i>									
Design	1945	1945					1956		1958
Construction	9/1946	1949	1957	8/1956			1957	3/1955	1962
First criticality	11/1946	8/1951	10/1961	8/1963			6/1958	11/1959	1/1967
Full operation	3/1949	12/1951	4/1965	8/1966	1955	1956	7/1959	7/1963	3/1967
Shutdown	6/1953	1963	-	-	1956	1957	-	-	-
<i>Remarks</i>									
	1. fast reactor, 1. Pu-fueled reactor	1. nuclear electricity generation Pu-Core since 1962	Reactor Plant with integral fuel processing facility	Since 10/1966 out of operation			UC-Core since 1965		(RAPSODIE is not really a reactor of the first generation, it belongs, to a large extent, to the second generation)



TABLE XII  
FBR PROGRAMS — Milestones (from Ref. 3)

	USA	USSR	UK	France
Start initial studies	1945	1949-1950	1951	1956
First critical :small	1955	1955	1954	--
facility operation :large	1968	1961	1962	1966
Experimental reactor critical	1946 (Clementine) 1951 (EBR-I) 1961 (LAMPRE) 1963 (EBR-II) 1963 (EVESR)	1956 (BR-2) 1958 (BR-5)	1959 (DFR)	1967 (Rapsodie)
5% burnup in :thermal flux	1960			
FBR pins :fast flux	1967	1961	1964	1968
Test reactor critical	1969 (SEFOR) 1974 (FFTF)	1969 (BOR) (60 MWt)	--	--
Large test :(>5 MWt or loops operation : >5000 gpm)	1966 (SCTI) 1970 (SPTF)	1966	1963 (7000 gpm pump) 1967 (large Na-H 0)	1962 (10 MWt, IHX, pump) 1964 (5 MWt SG) 1969 (20 MWt SG)
Demonstration reactor critical	1963 (Fermi) ~1978 (Na FBR)	1970 (BN 350)	1972 (PFR)	1973 (Phenix)
Large reactor critical (estimate)	~1984	1975 (600 MWe)	1979	1980
Professional Staff 1966-1967	1000	?	~650	~700
R&D funds :through 1967 millions \$ :1967 annual :70 - 75 annual	400 ~70 big increase?	?	? ~35 increase	? ~30 ~30
Reactor construction millions \$ :already built	EBR I-II LAMPRE 90 Fermi	?	24 DFR	45 Rapsodie
:being built or discussed	330 SEFOR FFTF 3 Dems.?		70 PFR	100 Phenix

Table XII contd.

	Germany/Belgium/Holland	Italy	Japan
Start initial studies	1956 (Belgium), 1960 (Germany)	1962	1962
First critical :small facility operation :large	1966 --	--	-- 1967
Experimental reactor critical	1970 (KNK-Na)	--	1973 (JEER)
5% burnup in :thermal flux FBR pins :fast flux	1968	1968 1968	1969
Test reactor critical	--	1975 (PEC) (140 Mwt)	1972 (~100 Mwt)
Large test :(>5 Mwt or loops operation :>5000 gpm)	1965 (5 Mwt SG) 1969 (88000 gpm pump) 1968 (large Na-H <sub>2</sub> O) 1970 (50 Mwt SG)	1972 (50 Mwt SG)	(under consideration)
Demonstration reactor critical	1975 (Na)	1978?	1977?
Large reactor critical (estimate)	~1980	1983?	1987
Professional staff 1966-1967	200-250 (Germany) 150-200 (Belgium + Holland)	~100	~150
R&D funds millions \$ :through 1967 :1967 annual :70 - 75 annual	~60 (Germany), ~10 (Belgium & Holland) ~40 (+15 Belgium & Holland) increase	~15 ~10 big increase	? ~10 big increase
Reactor construction millions \$ :already built :being built or discussed	0 HNK , Na Dem. (300)	0 PEC Na Dem. (140)	0 JEER Na Dem. (150)

TABLE XIII  
LMFBR Demonstration Plants\*

	USA			USSR		UK	France	Germany
	GE	Westing- house	AI	BN-350**	BN-600	PFR	Phenix	SNR
<u>Reactor Power</u>								
Thermal, Mwt	750	770	1250	1000	1470	600 (670)	563	730
Electrical, MWe	310	300	~500	350	600	250 (275)	233	295
Net efficiency (%)	41.4	39	40	35	40.8	~41	41.5	40.4
<u>Core (reference)</u>								
Fuel: pin length (cm)	90	76	125	105	70	91.5	85	95
diameter (cm)	0.58	0.64	0.64	~0.60	0.69	0.585	0.66	0.60
Core volume (liter)	2000	~2000	3100	1900	~2300	1320	1225	1750
Fuel rating av, Mwt/kg fissile	0.82	0.85	0.93	~1.28	1.22	~0.7	0.78	0.84
Power density av Mwt/liter (core)	0.31	0.39	~0.37	0.5	0.55	~0.4	0.43	0.37
Linear rod power max, W/cm	500	450	490	470	~500	450	~480	580
Breeding ratio	1.2	1.22	1.3	1.5	~1.35	1.15-1.25	1.16	1.33
Burnup, Mwd/Kg (max)	100	~100	75 (av)	50	100	70	>50	85
<u>Primary Heat Transfer System</u>								
Type	pool	loop	loop	loop	pool	pool	pool	loop
Number of coolant loops	3	2+1	3	5	3	3	3	3
<u>Coolant temperature</u>								
Core inlet, C	425	400	405	300	380	400 (430)	405 (420)	375
Core outlet, C	590	550	570	500	550	560 (600)	565 (590)	545
<u>Steam conditions</u>								
Temperature, C	510	480	~480	435	505	515 (565)	510 (540)	505
Pressure, atm	170	170	~170	45	140	162	163	165
Date of Operation				1970	1976	1972	1973	1975

\*PuO<sub>2</sub>-UO<sub>2</sub> fuel, stainless steel clad.

\*\*UO<sub>2</sub> for the first load (950 Kg of U-235)

TABLE XIV

## ESTIMATED LMFBR POWER PLANTS TIMETABLE

	<u>Demonstration Plant</u> (250 to 500 MWe)	<u>Large Plant</u> (600 to 1000 MWe)	<u>Commercial Plant</u> ( $\geq$ 1000 MWe)
France	1973*	1980	~1984
Germany	1975*	1980	~1986
Japan	1977		1987?
UK	1972*	1979	~1983
USA	1977-78	1984	~1990
USSR	1970*	1976*	~1980

---

\* Plants under construction or committed

TABLE XV

## TYPICAL PARAMETERS FOR GAS-COOLED FAST BREEDER REACTORS\*

Plant	Commercial Reactor	Demonstration Plant
Electrical power (Mw)	1000	311
Thermal power (Mw)	2530	826
Net cycle efficiency (%)	39.5	37.6
Helium pressure (atm)	85	85
Reactor inlet temperature (C)	340	312
Reactor outlet temperature (C)	635	541
Midclad hot spot temperature (C)	800	700
Maximum linear rating (w/cm)	580	450
Maximum fuel burnup (Mwd/t)	100,000	50,000-100,000
Active core volume (liters)	8600	3200
Active core length (cm)	160	100
Fuel pin outside diameter (cm)	0.8	0.715
Clad OD/ID	1.10	1.15
Fuel volume fraction	0.30	0.30
Coolant volume fraction	0.54	0.45
Reactor pressure drop (atm)	3.4	2.7
Average rating Mw/kg (fissile)	0.90	0.60
Average power density kw/l (core)	270	240
Total pumping power (Mw)	~100	~50
Average fuel enrichment (%)	18	12
Fuel life (year)	2.3	2.4
Doubling time (year)	9	21
Conversion ratio	1.5	1.33
Steam conditions:		
Temperature, C (F)	538 (1000)-538 (1000)	468 (875)-496 (925)
Pressure, atm (psi)	163 (2400)	196 (2900)

\*PuO<sub>2</sub>-UO<sub>2</sub> fuel with stainless steel clad and surface roughening.

TABLE XVI

ENERGY CONTENT OF WORLD FUEL RESERVES  
 (1Q =  $10^{18}$  Btu)

	<u>High Grade Ores</u>	<u>Low Grade Ores (estimate)</u>
Uranium	76	$5 \times 10^6$
Thorium	48	$7 \times 10^6$
U + Th	124	<u><math>12 \times 10^6</math></u>
Lithium	18*	$5 \times 10^6$
(Li <sup>6</sup> in oceans)		$6 \times 10^6$
Total Li		<u><math>11 \times 10^6</math></u>

Assumptions: 100% utilization of U and Th by breeders;  
 Li<sup>6</sup> content (7.4%) burnt in D-T reactors.

\* very conservative figure because of low demand. (From  
 Ref. 33)

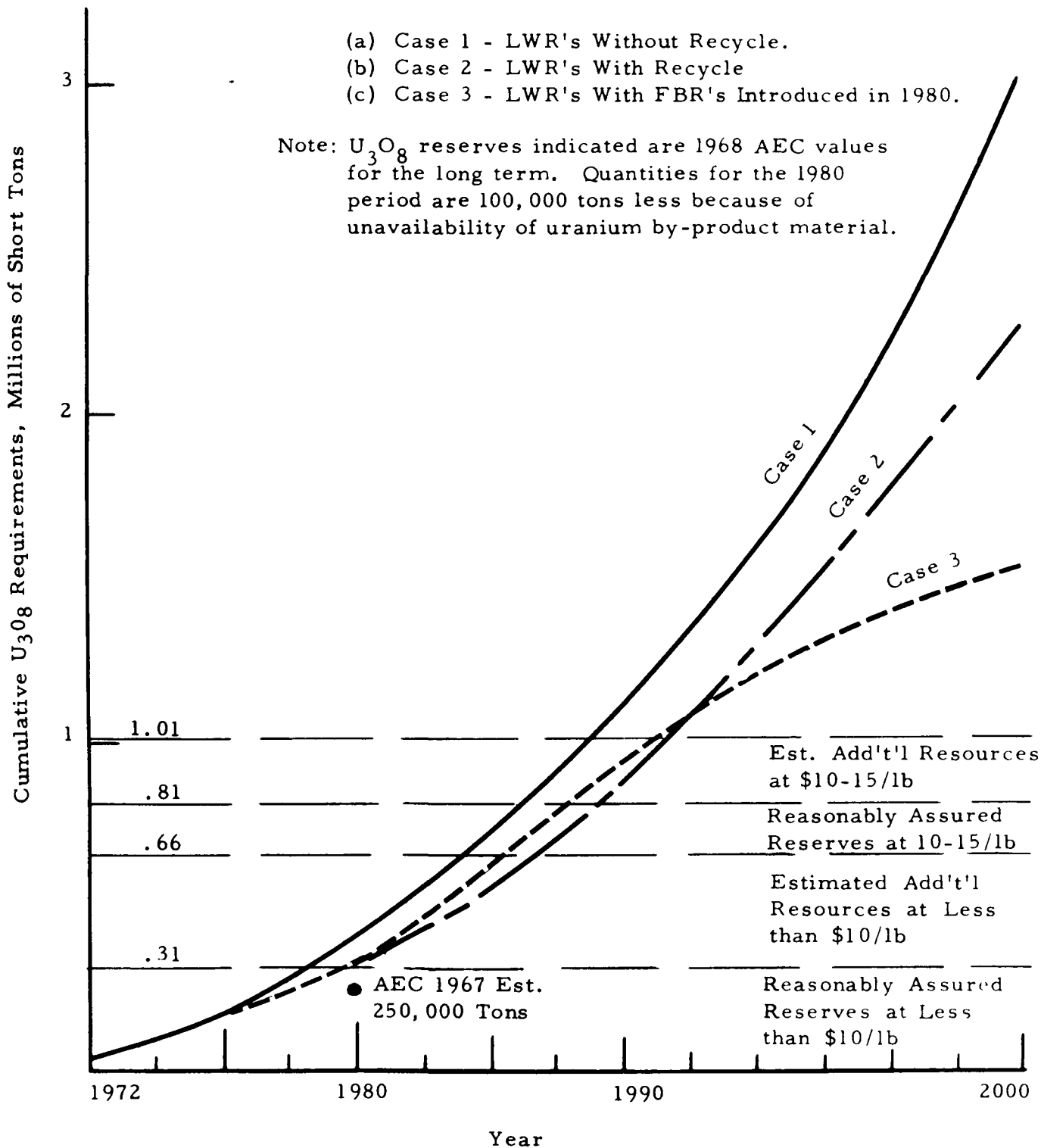
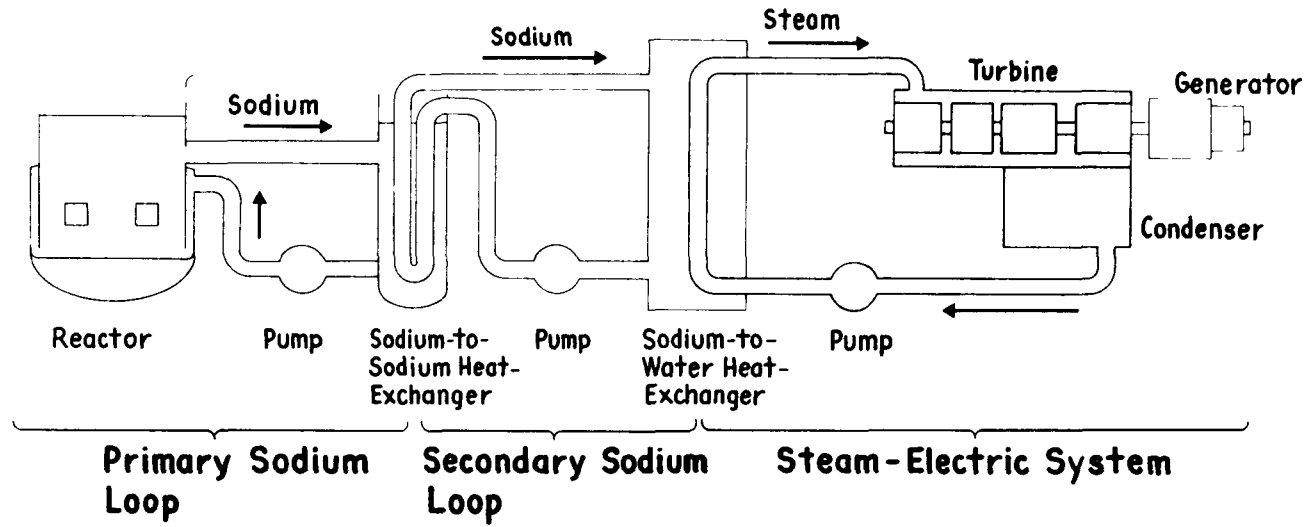


Fig. 1 — Cumulative  $U_3O_8$  Requirements.

# SODIUM FAST SYSTEM



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# GAS FAST SYSTEM

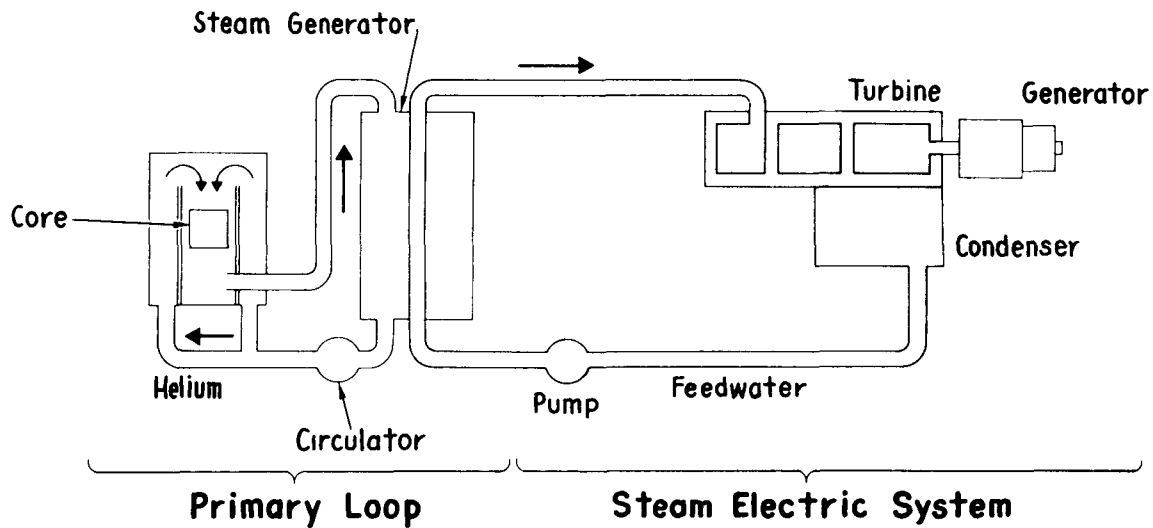
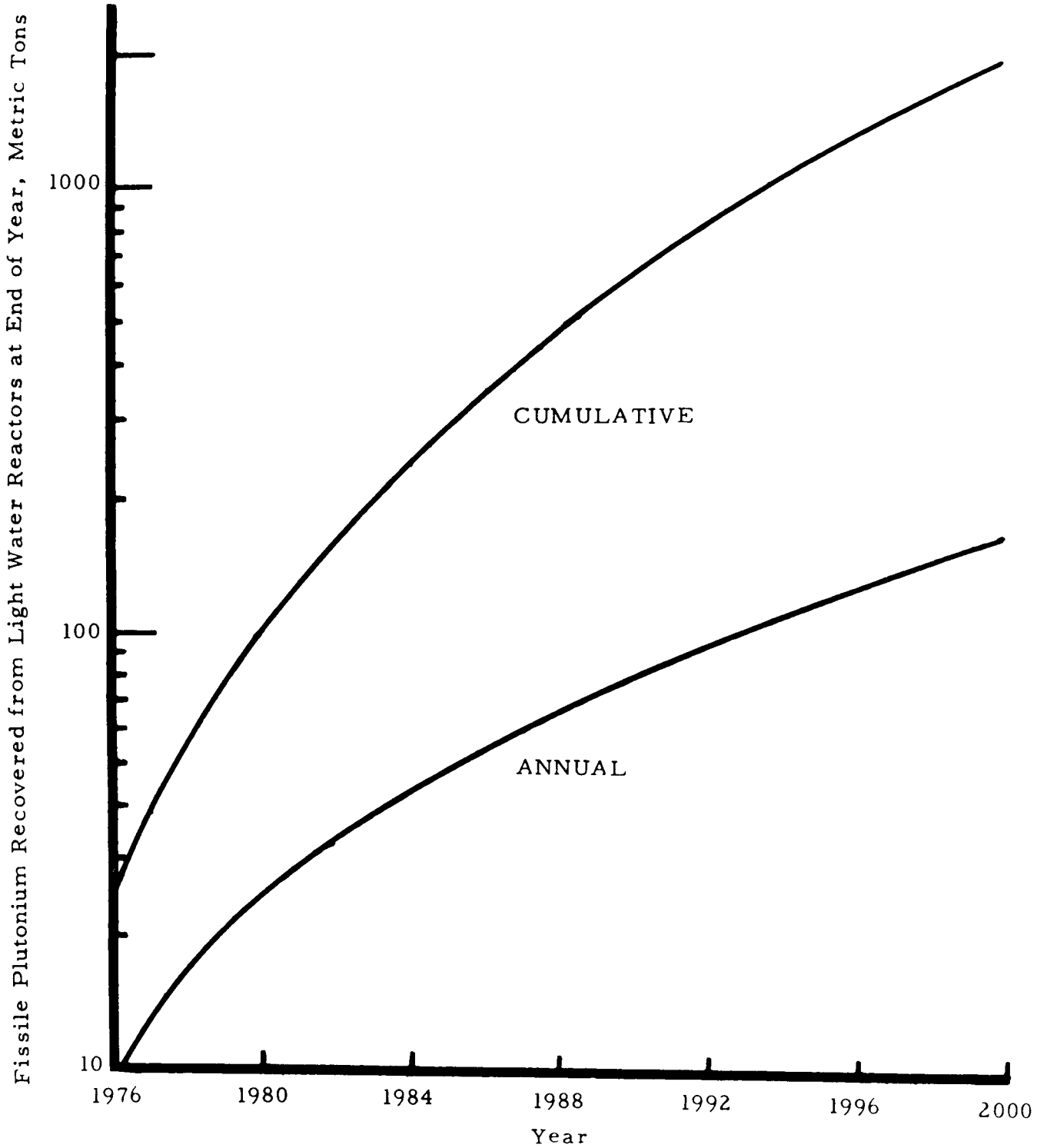
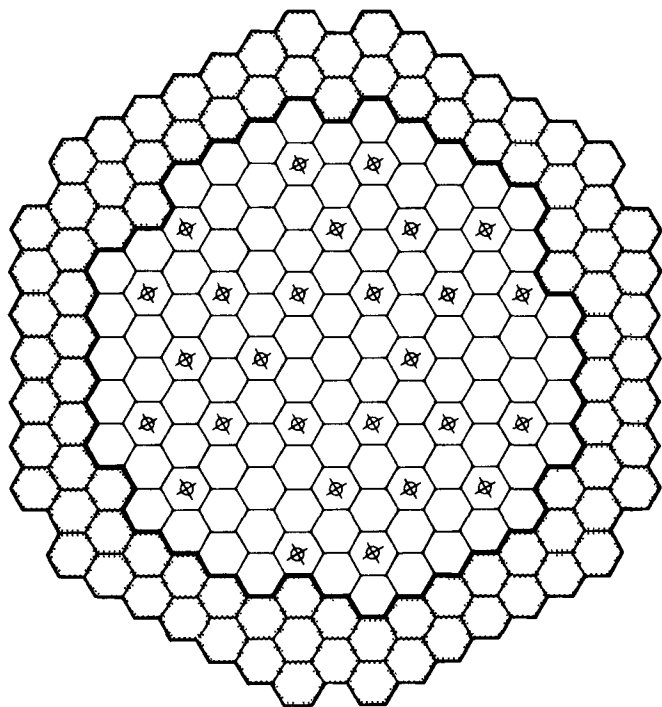


FIGURE 2. Fast breeder reactor systems








**Fig. 3 — Fissile Plutonium Recovered from Light Water Reactors.**



CORE PLAN

-  STANDARD FUEL ELEMENTS
-  CONTROL FUEL ELEMENTS
-  BLANKET ELEMENTS

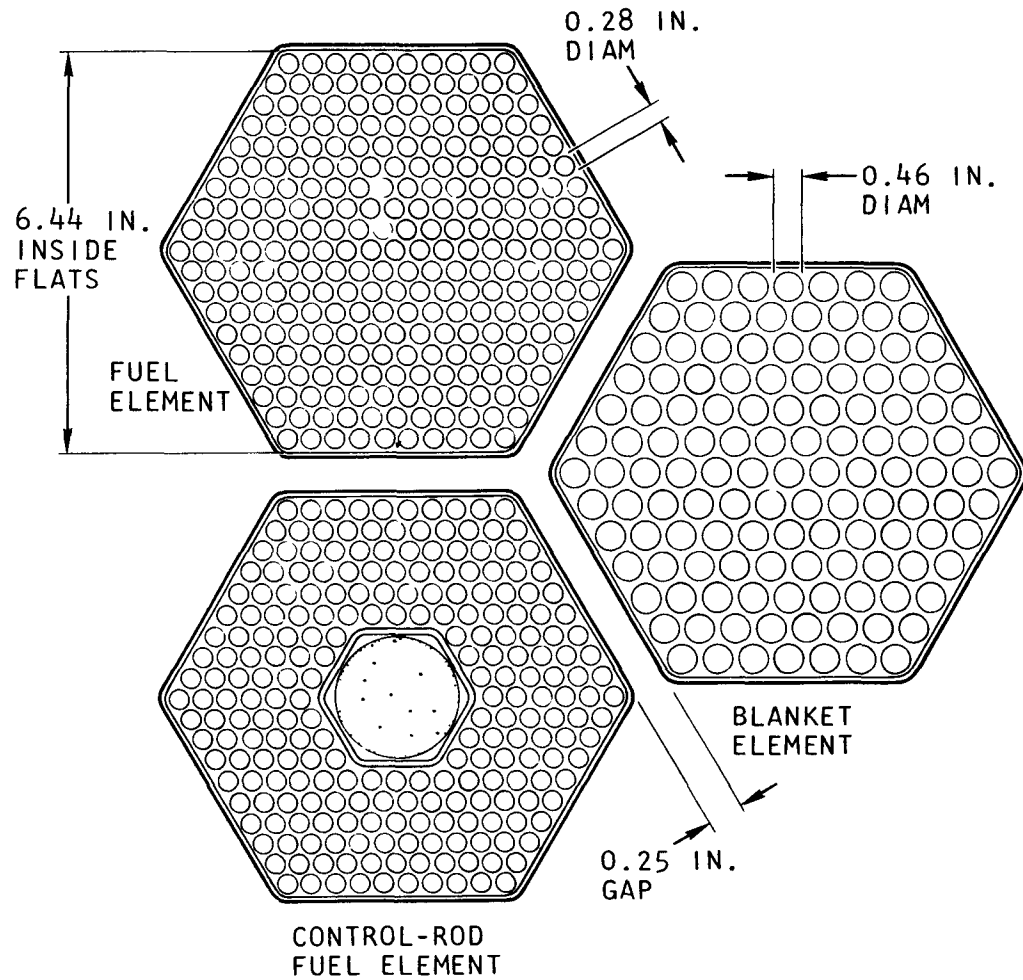


FIGURE 4. Plan view of 300 MWe GCFR core

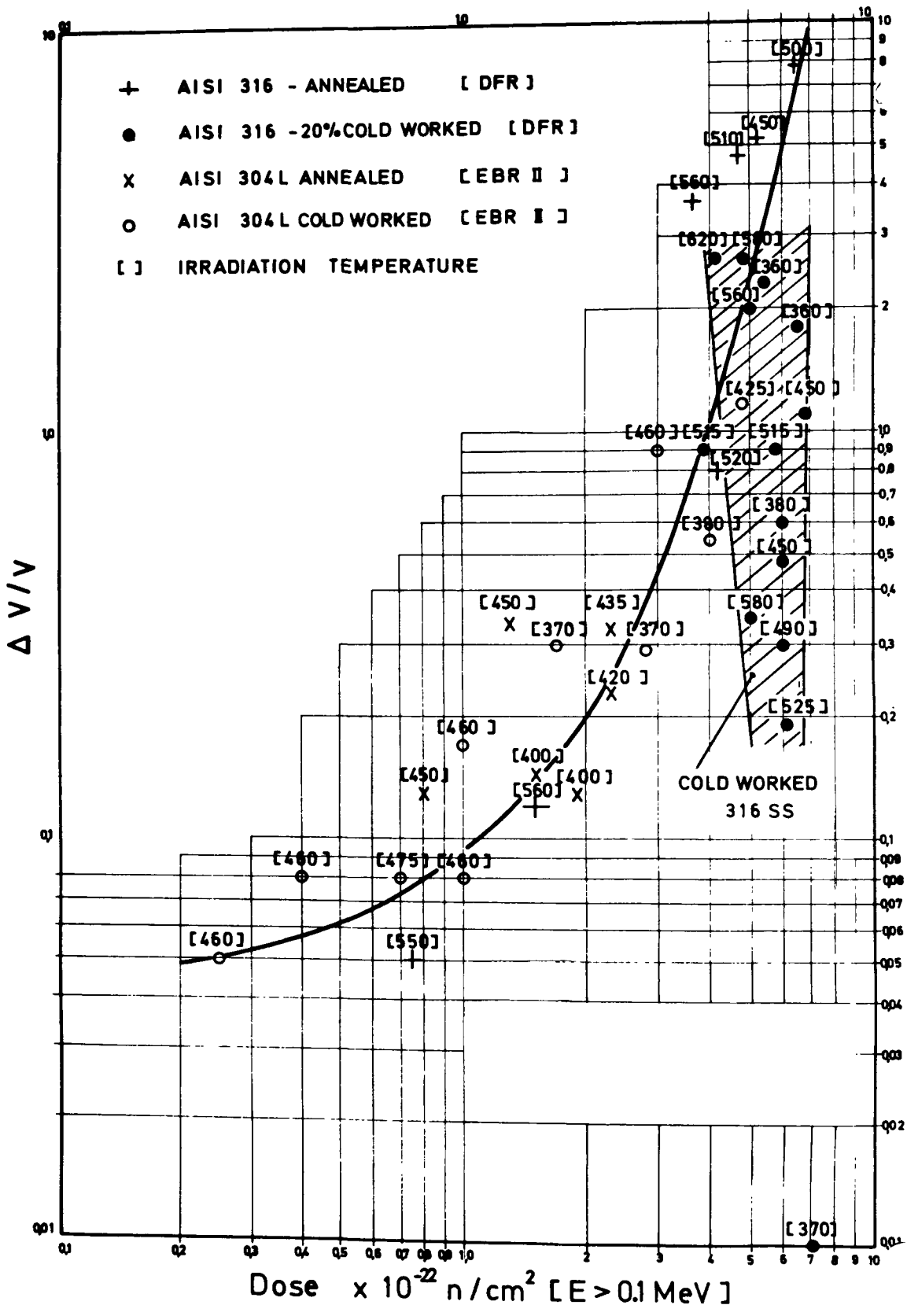


FIGURE 5. Swelling of stainless steel. [Ref. 10]

# DEPENDENCE OF VOLUME EXPANSION AT A CONSTANT FLUENCE ON IRRADIATION TEMPERATURE (EXPERIMENTAL DATA - AUSTENITIC STAINLESS STEELS)

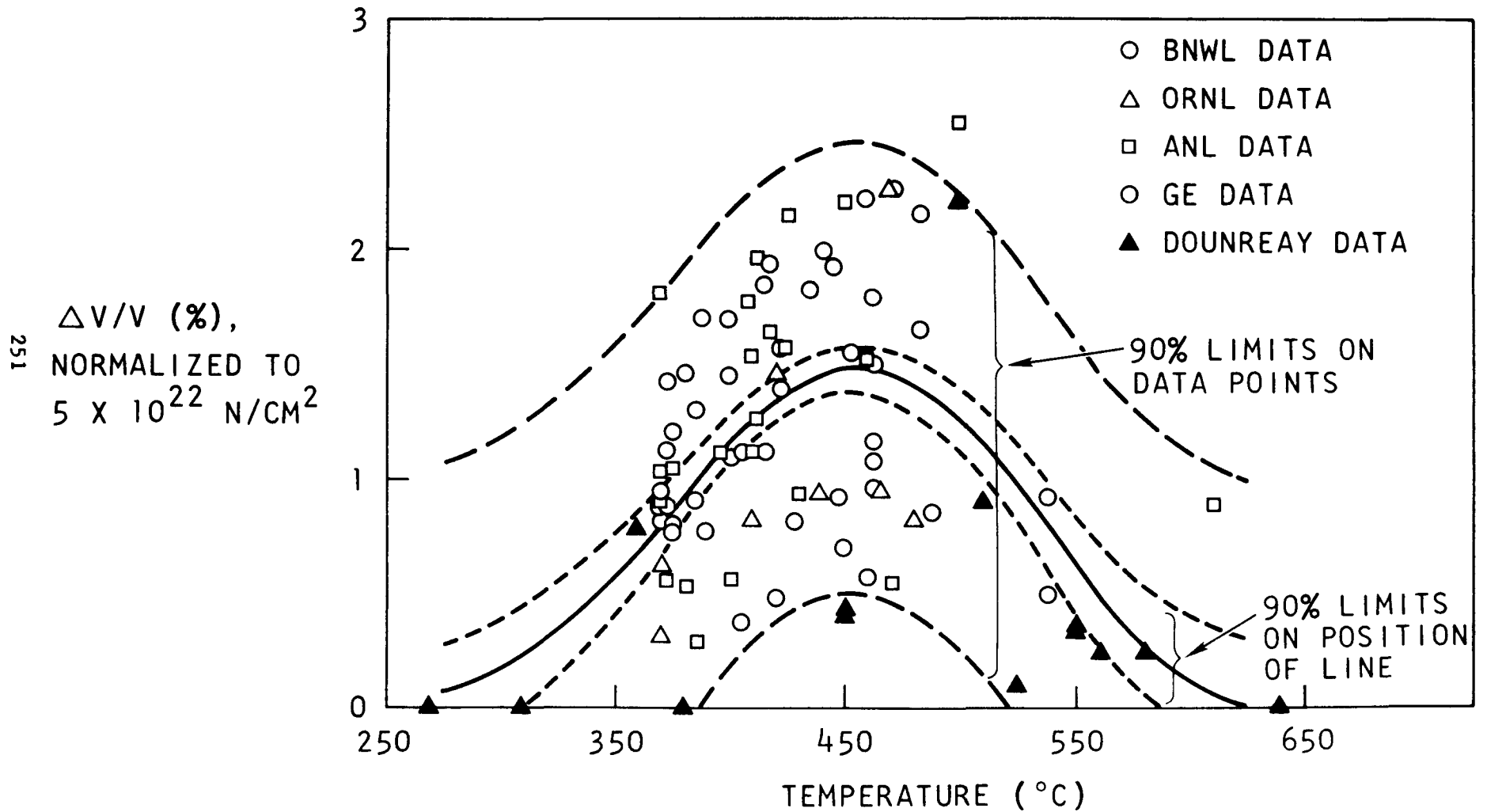


FIGURE 6

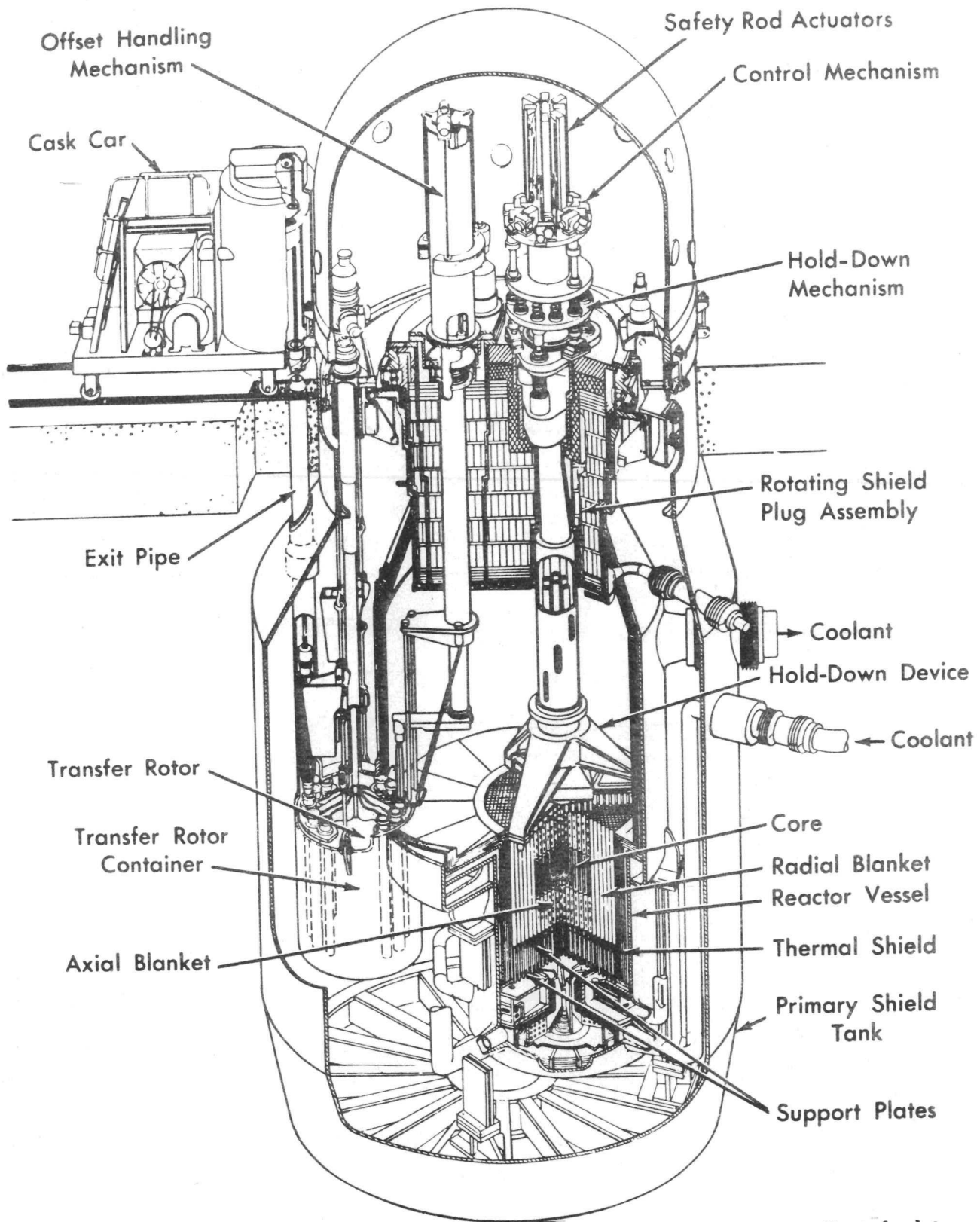


FIGURE 7. The Enrico Fermi fast breeder reactor has an offset fuel handling machine permanently installed within the reactor vessel. (Courtesy "Solid Fuel Reactors," by J. R. Dietrich and W. H. Zinn, published for the USAEC by the Addison-Wesley Publishing Co., Inc., Reading, Mass., 1958.)

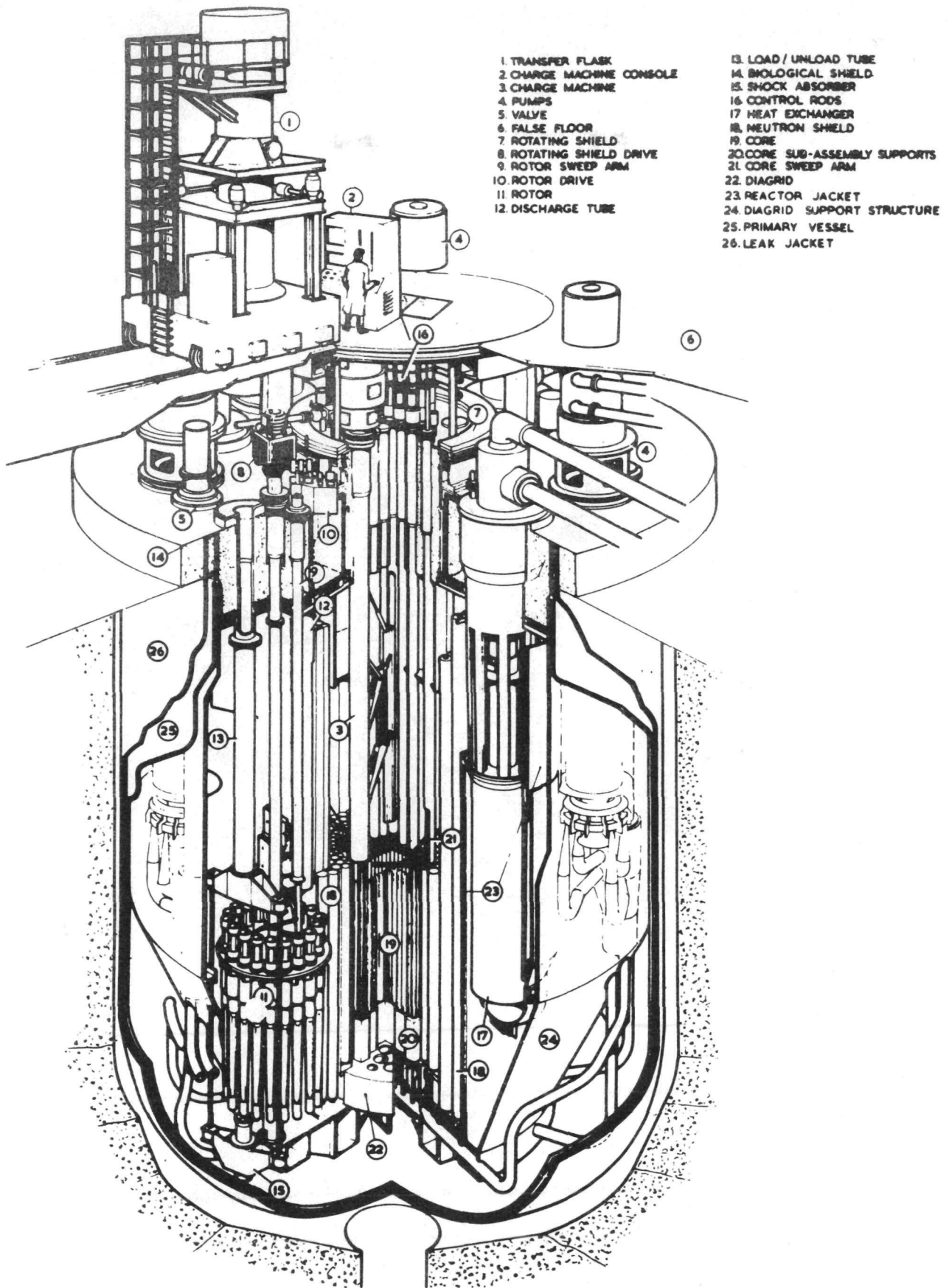


FIGURE 8. PFR Primary Circuit [Ref. 20]

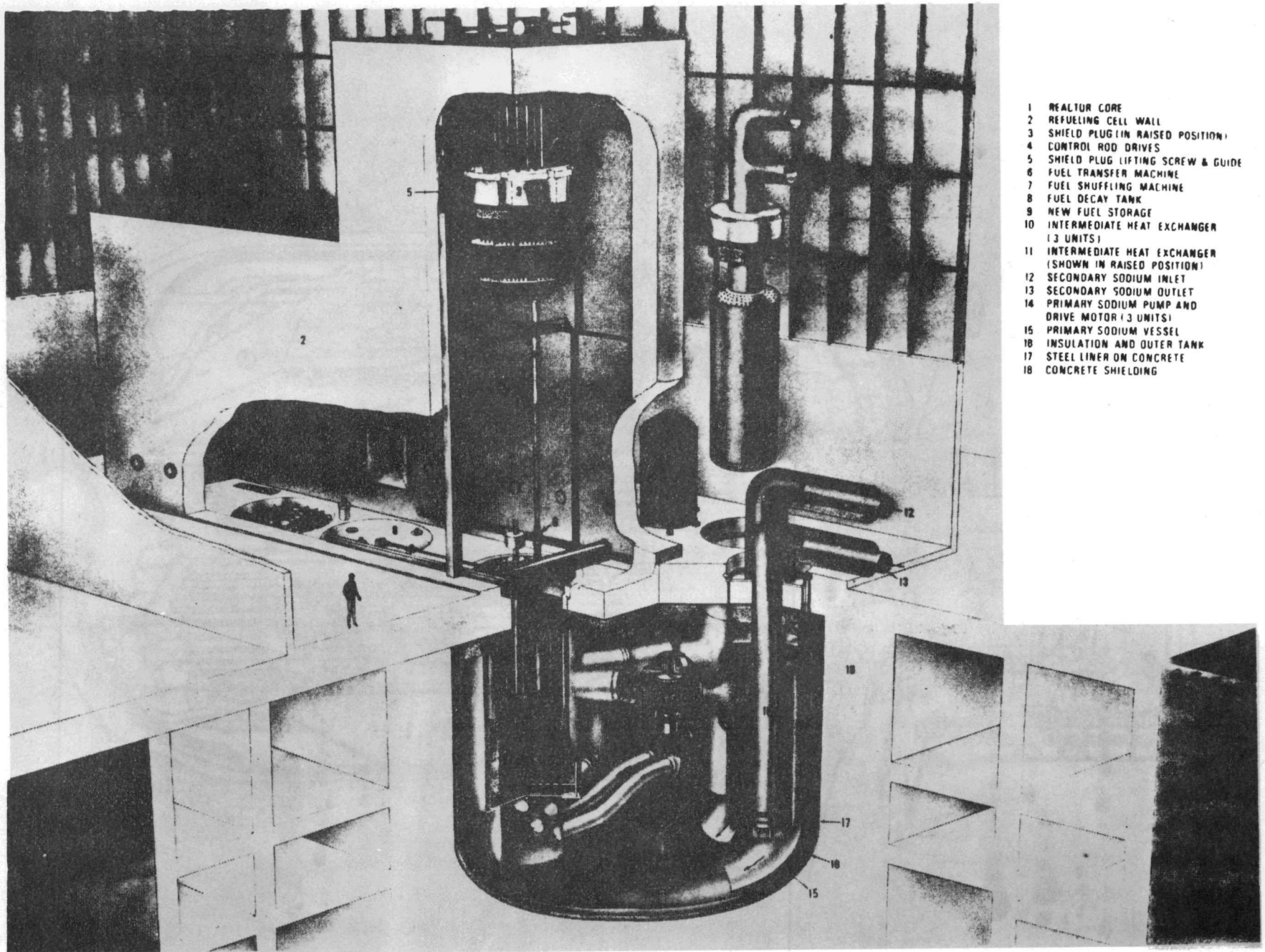


FIGURE 9. Sodium Breeder Reactor Plant. [Ref. 17]

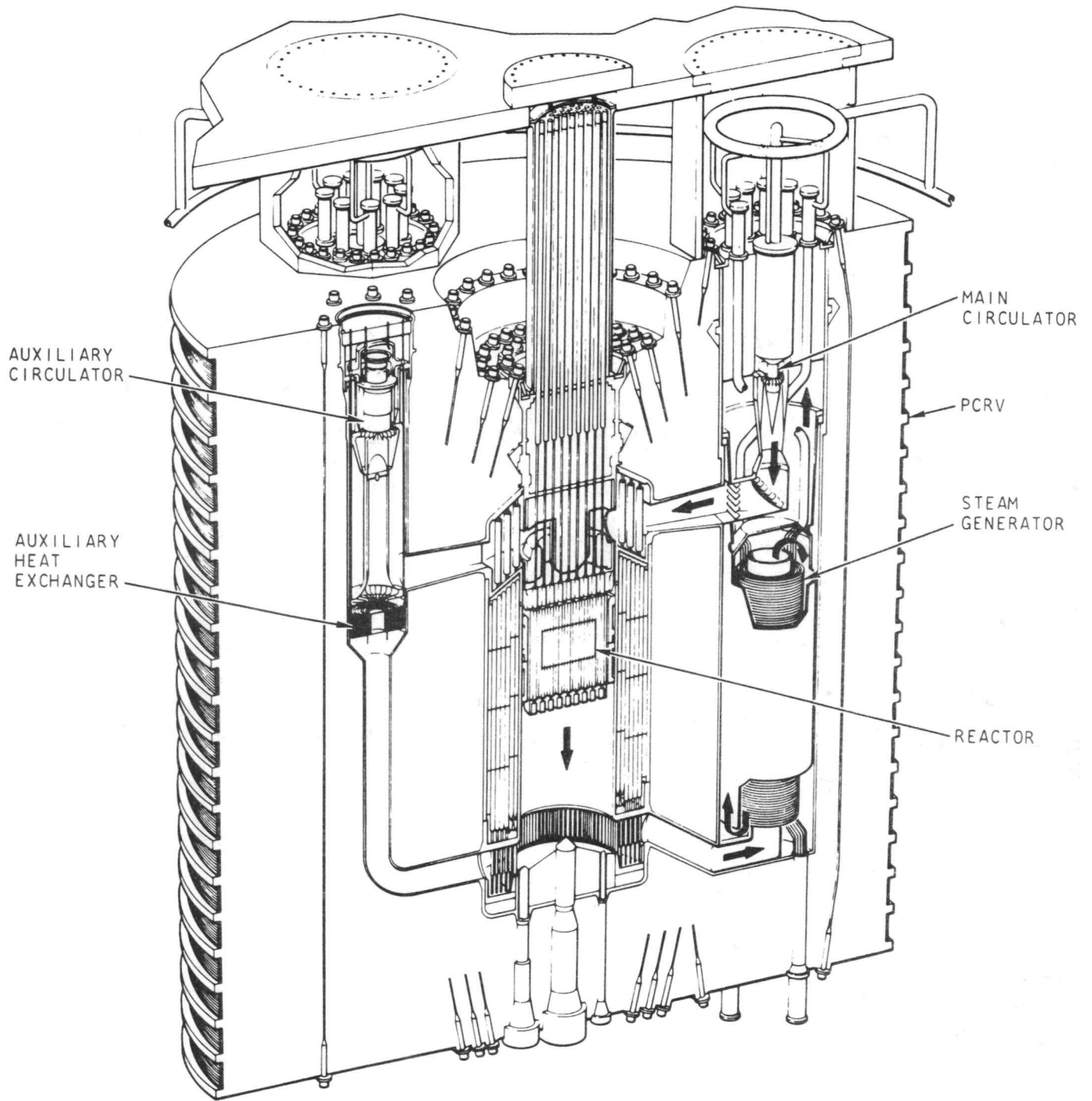


FIGURE 10. Nuclear steam supply of 300 MWe GCFR



## WHY FUSION?\*

by

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### INTRODUCTION:

Could controlled fusion power change our future? To answer this question we must understand the relationships that exist between population expansion, living standards, raw materials, pollution and energy resources. In Figure 1, the apartment house represents the size of the earth, with its limited amount of space.<sup>1</sup> The man sitting on the top floor represents North America. The three men down below on the second floor represent the other developed countries -- primarily Europe, Russia, and Japan. The eight people down on the bottom floor represent the less-developed world - Asia (less Japan), Africa and South America. That is essentially the ratio of population in these groups -- 1:3:8.<sup>2</sup> On the right side of Figure 1 is an energy source. It represents our present sources of energy which are primarily the fossil fuels. Those who use most of the energy live on the upper floors and they are getting pollution from their energy sources. The raw materials necessary to maintain the life of ease on the top floor are being placed on the dumbwaiter (shown at the left of the figure) by those on the bottom floor. Since nothing is really used up but only transformed, almost everything ends up as pollution on the upper floors. This piles up and trickles down a little bit on those living below. Also, the tenants of this apartment house have developed the habit of keeping numbers of bombs to secure a more livable environment.

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\*From the transcribed tape of the speech delivered June 5, 1970, at the Fusion Reactor Design Symposium, Lubbock, Texas.

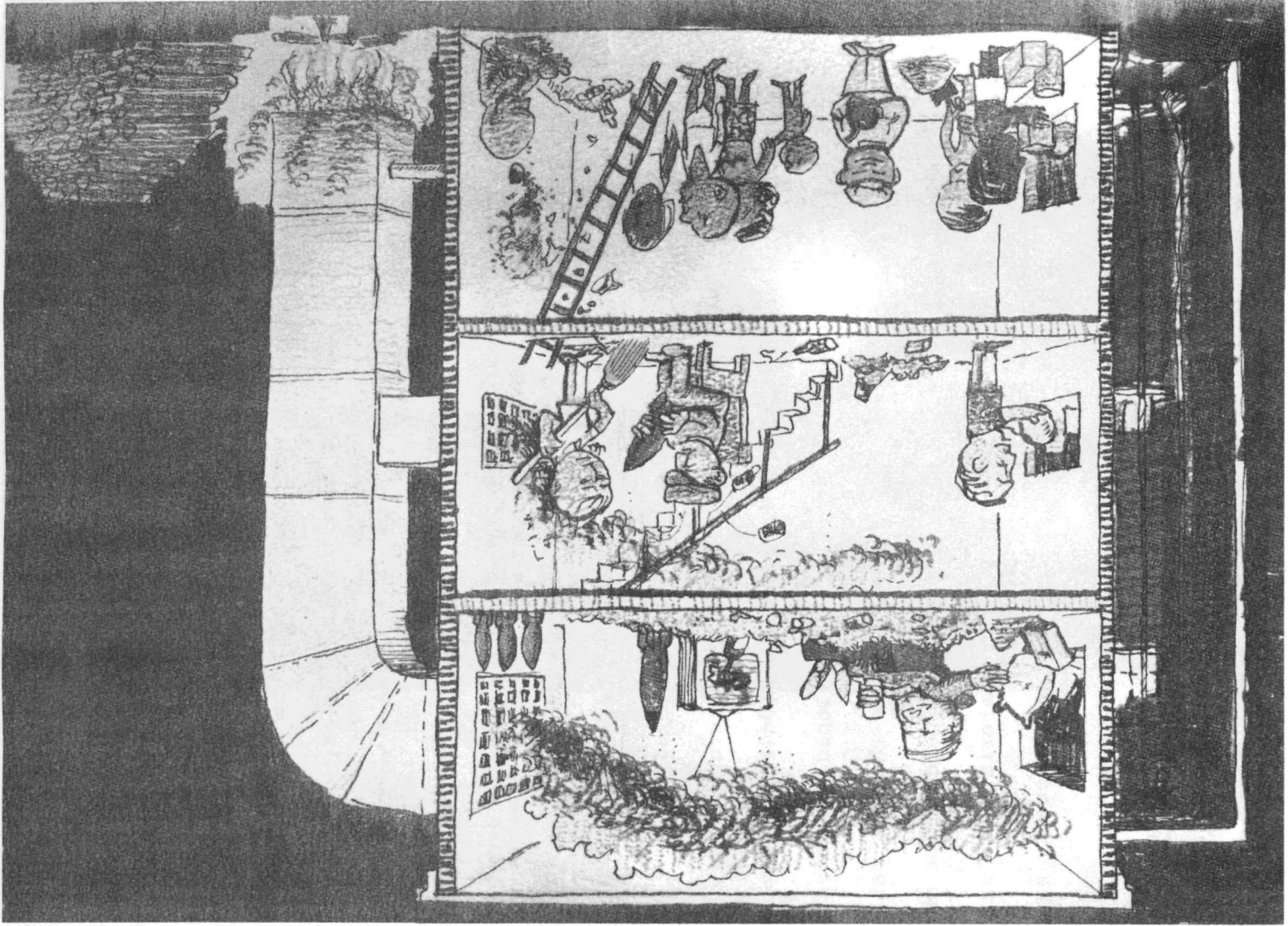


Figure 1

Now for some actual data. Figure 2 is a plot of energy consumption versus the Gross National Product on a per capita basis for nations of the world.<sup>3</sup> Although imperfect, the GNP is a good measure of living standards. And you see there is a very close relationship between energy use and living standards. The developed countries, such as the U.S. and Canada are up on top, the rest of the developed world is in the middle and the underdeveloped countries, which use practically no energy, are down in the lower left hand corner accompanied by very poor living standards. Let's look at use of materials versus energy consumption as shown in Figure 3. The only data that I could find on the use of materials was for steel, which is a good indicator.<sup>4</sup> Again, if we look at who's using all the materials we see the U.S. sitting up on top, all the other developed countries tend to be in the middle and if you look down in the corner, we find all of the less developed countries. Now I would project that pollution will probably follow the same pattern. This is shown in Figure 4.<sup>5</sup> If you go to Guinea, you wouldn't see much pollution, but in the U.S., large quantities of raw materials are being used and disposed of accompanied by the pollution from the energy sources.

#### POPULATION:

To obtain total worldwide figures, we must use population as a multiplier factor for all of the above data. Figure 5 shows how this multiplier is projected to change in the future.<sup>6</sup> It's a very steep rising curve showing a world population of 6 or 7 billion and a U.S. population of over 300 million by the year 2000.

GROSS NATIONAL PRODUCT  
(DOLLARS PER CAPITA)

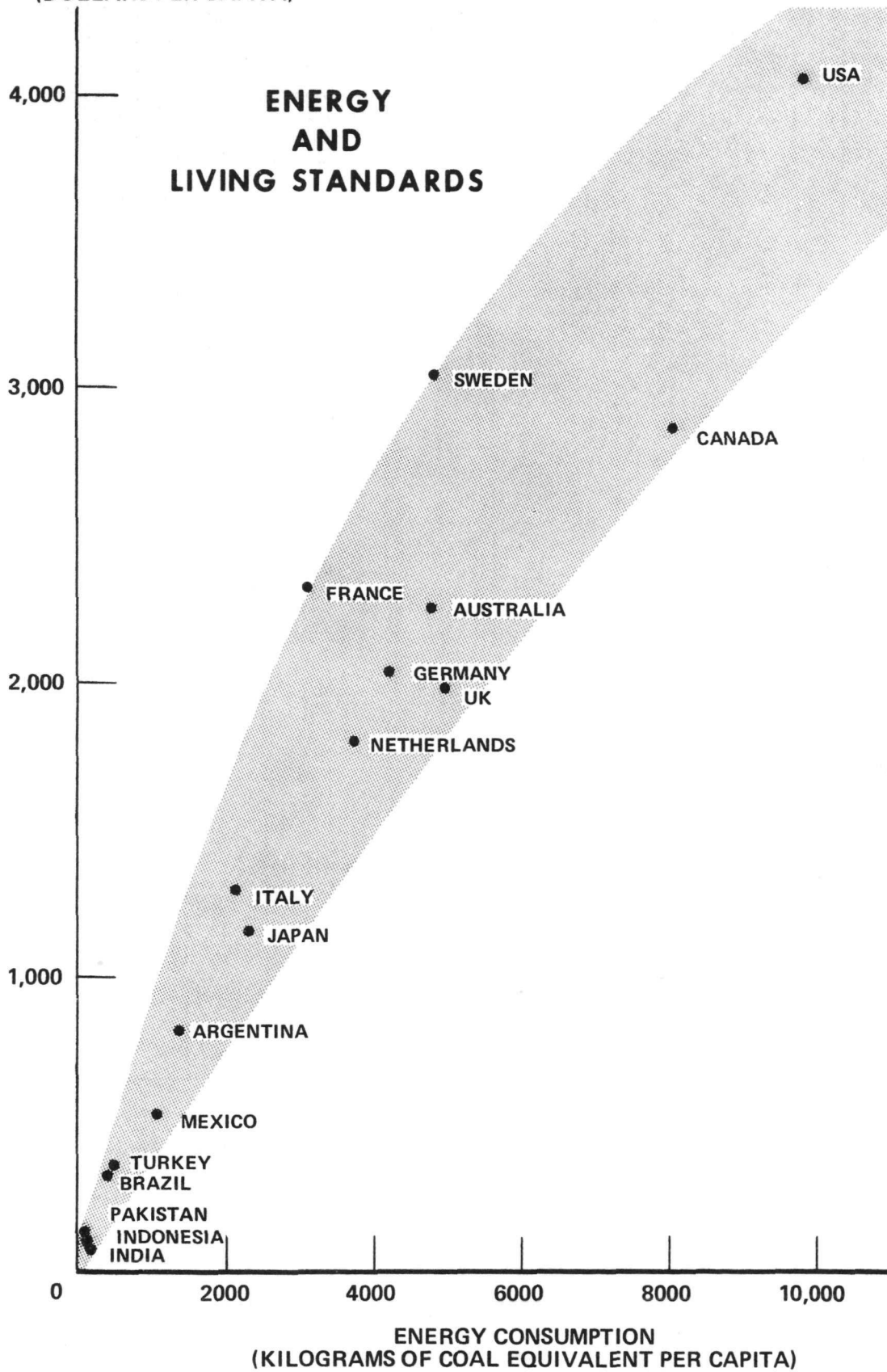
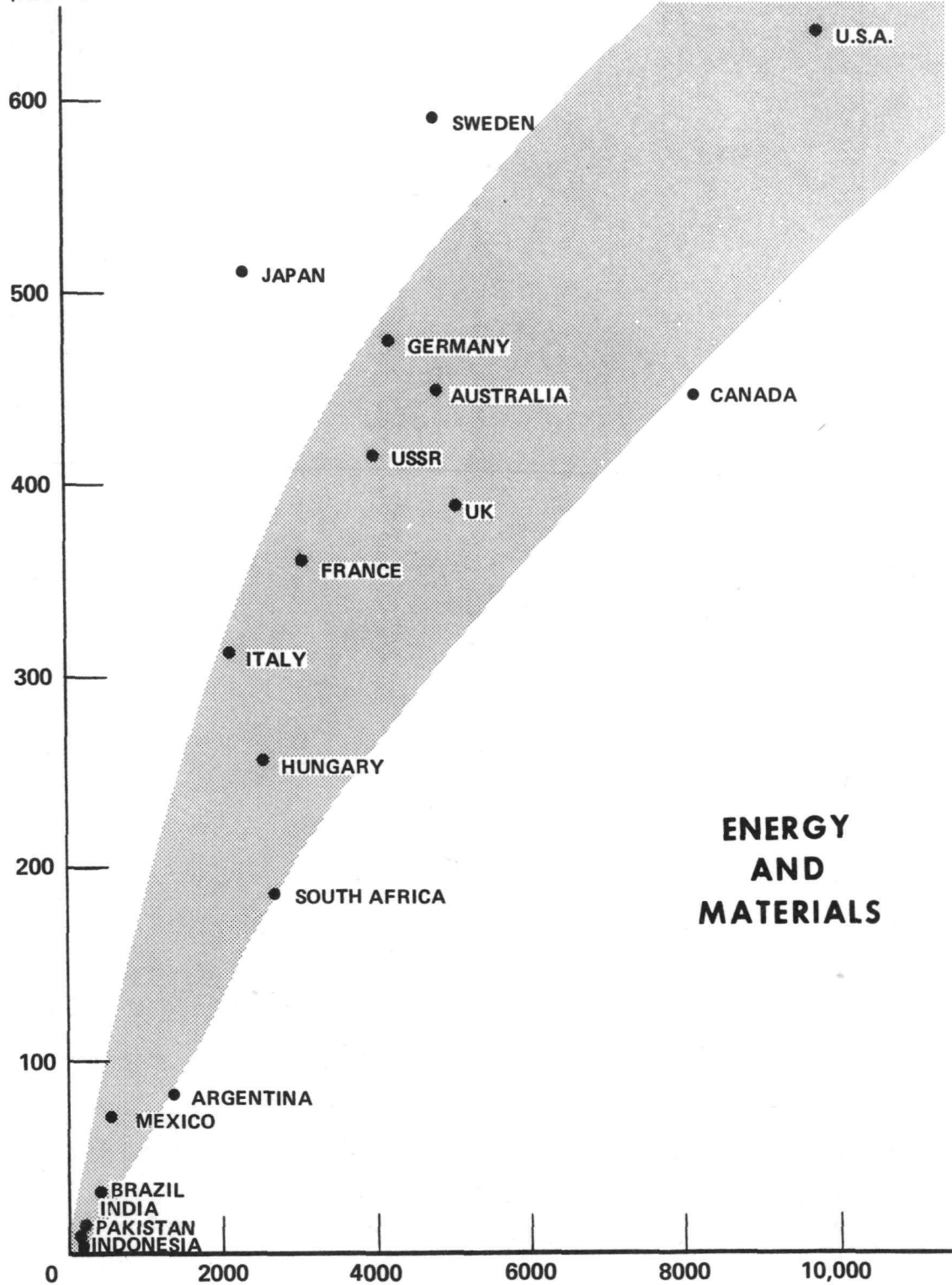


Figure 2

STEEL CONSUMPTION  
(KILOGRAMS PER CAPITA)



**ENERGY  
AND  
MATERIALS**

ENERGY CONSUMPTION  
(KILOGRAMS OF COAL EQUIVALENT PER CAPITA)

Figure 3

**AVERAGE SOLID WASTE COLLECTED  
(POUNDS PER PERSON PER DAY)**

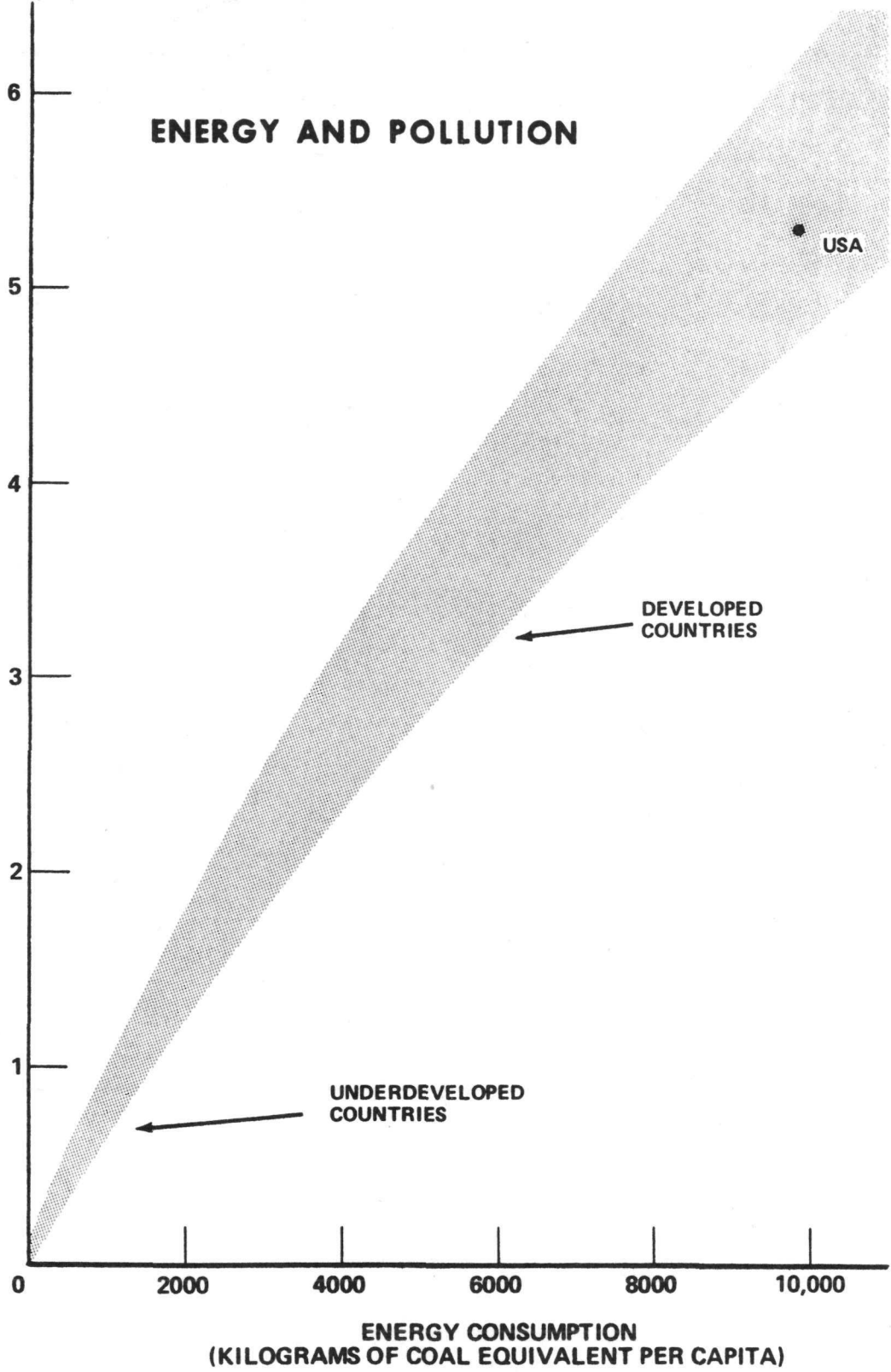


Figure 4

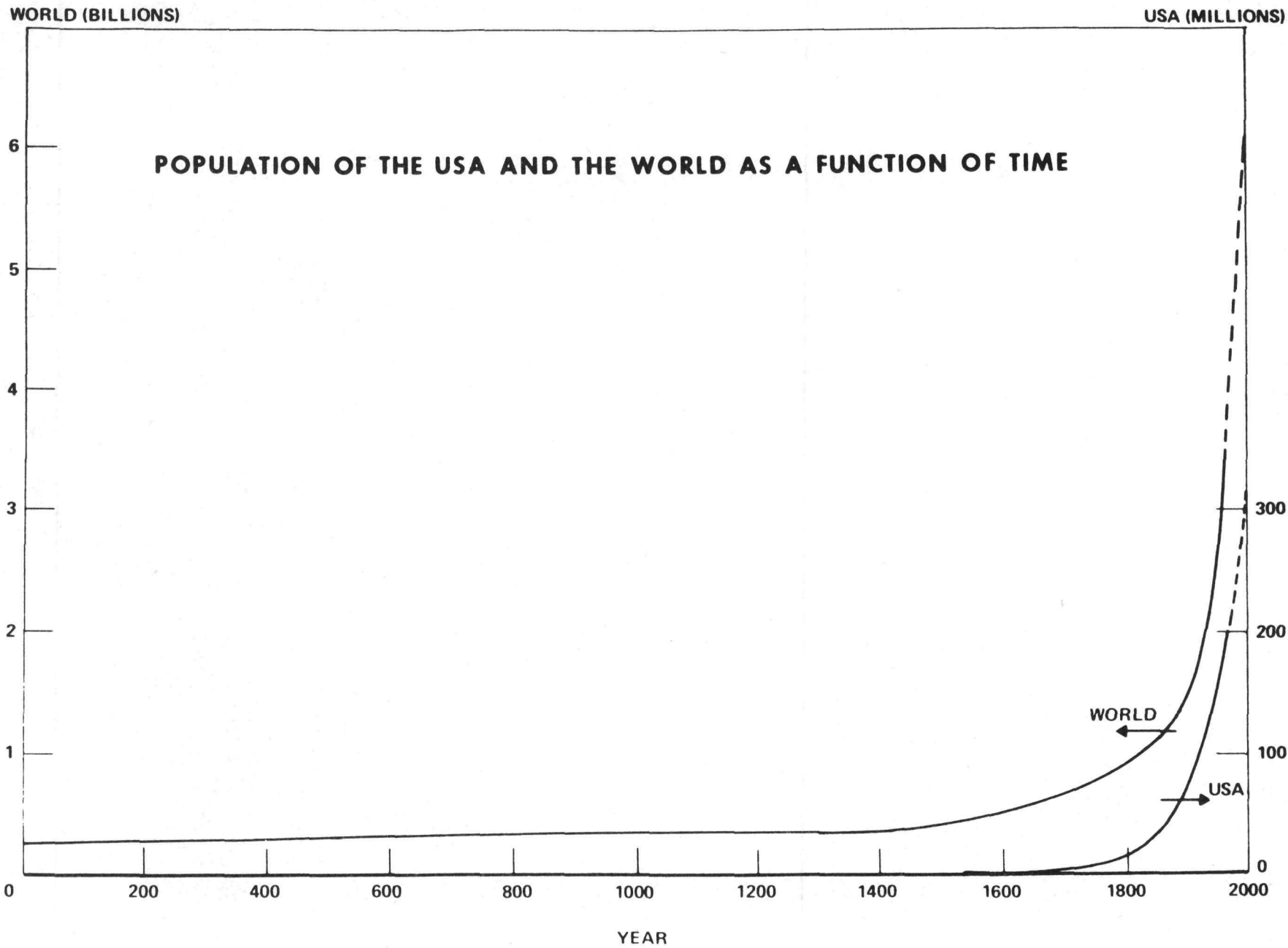


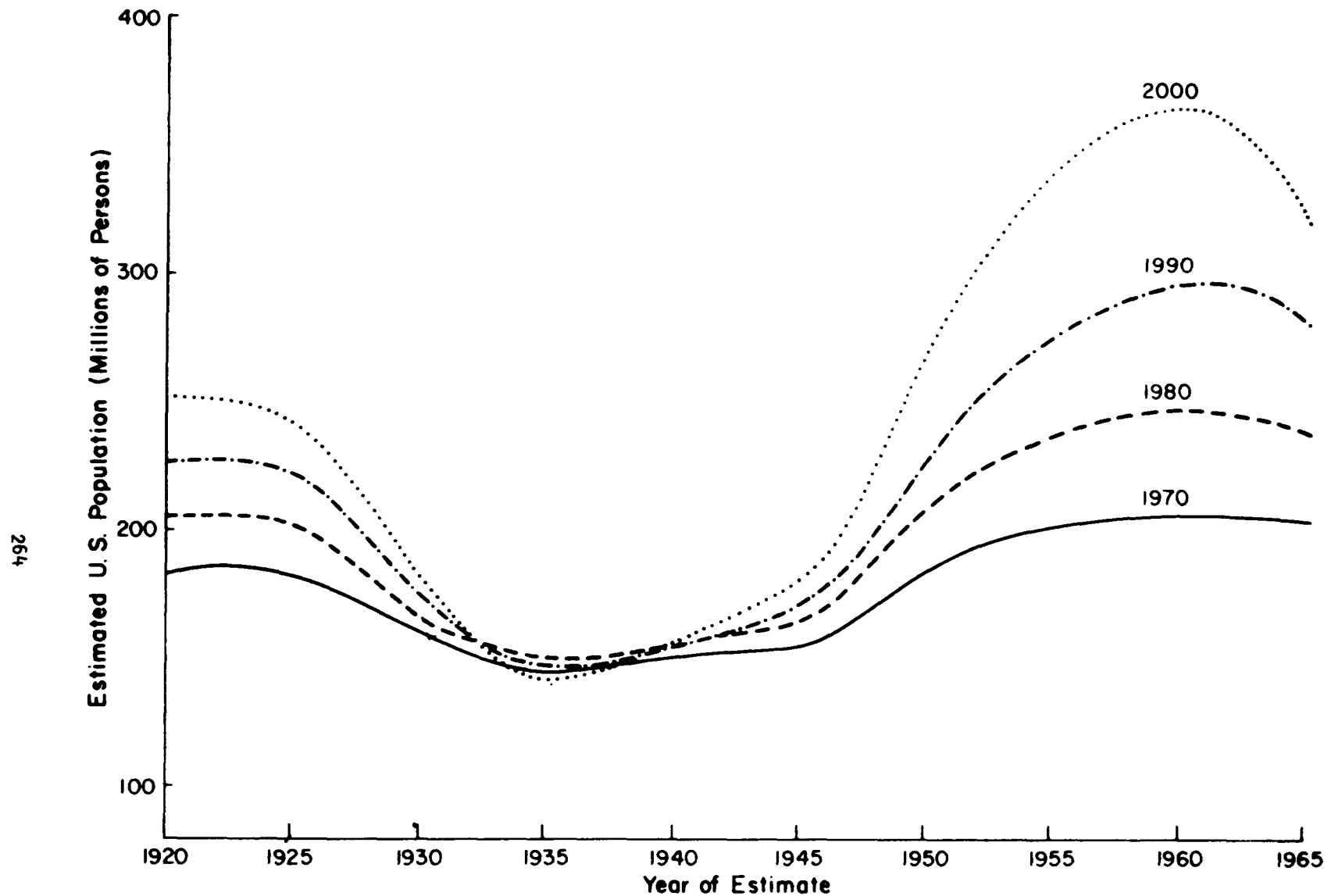
Figure 5

But population estimates are predictions and can change. Figure 6 illustrates this.<sup>7</sup> Where you are in time can have a large effect upon what the future population is projected to be. For example, if you were back in the depression days of the 1930's and you said, "What is the population of the U.S. going to be in the year 2000", you would find it was going to decrease. If you were predicting it in the year 1960, you would estimate over 400 million by 2000. Right now we're on a downward trend. Maybe people are seeing the future a little bleaker than they were before or maybe they are consciously attempting to create a more desirable future. Who are the people making the decisions that will determine what our population will be? They are the women, the mothers, who are having children. They fall over a narrow age range. Figure 7 shows the distribution of the U.S. population versus age.<sup>8</sup> You find that about 90% of the births occur for women in the range between 16 and 35. So it is this age group that determines what our future population will be. From now on I'm going to assume that the population that we will be dealing with in the year 2000 and the demands that we'll be talking about will be based upon a U.S. population of 315 million and a world population of between 6 and 7 billion. But remember, we have made an assumption and a small group of people, if persuaded, could change the entire population question.

MATERIALS:

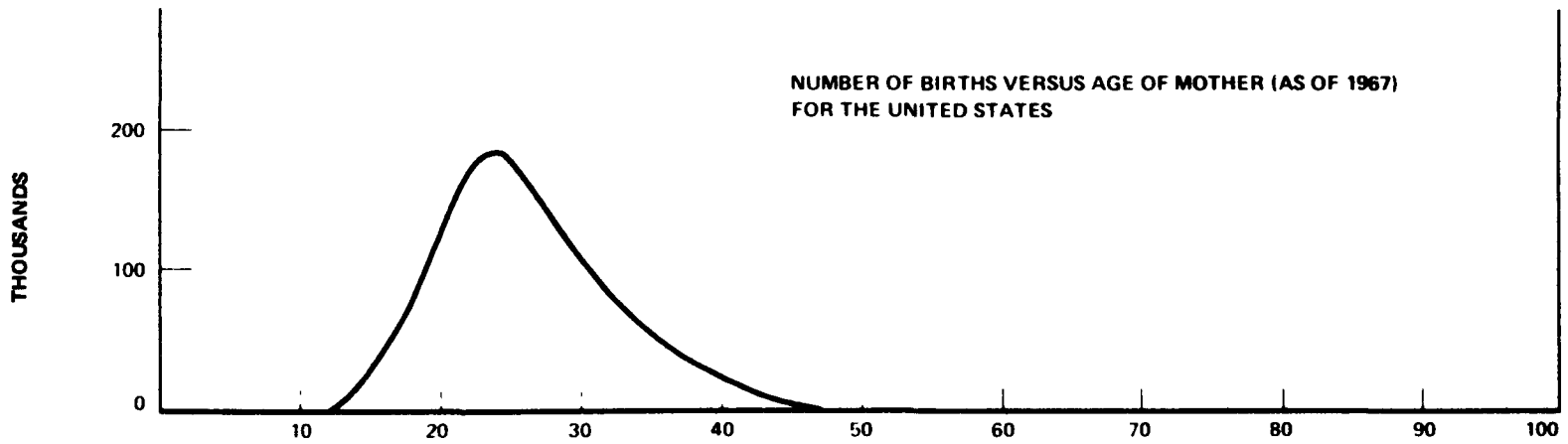
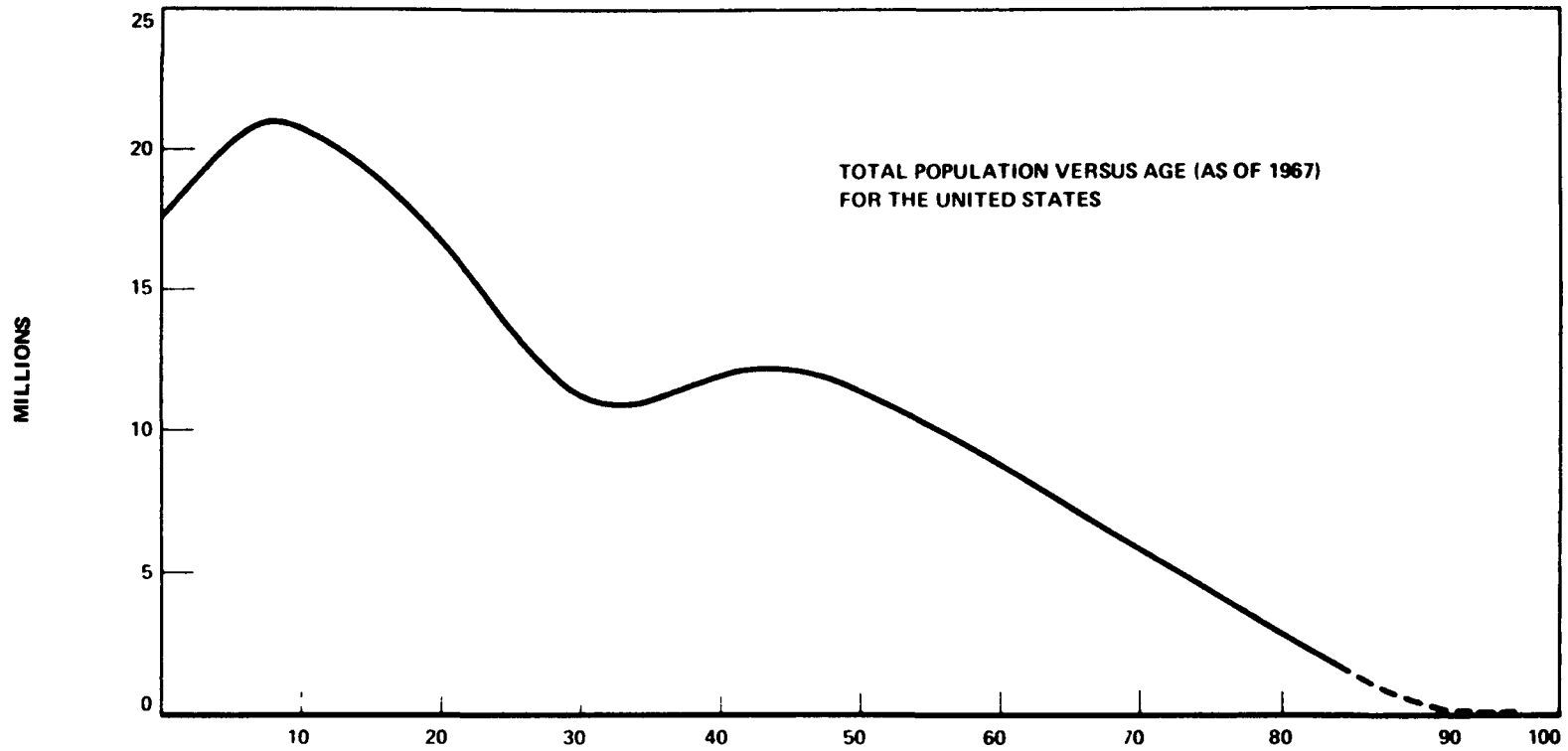
As the population and standard of living rise so do the world's needs for raw materials. The upper portion of Figure 8 shows the recent





Comparison of population estimates for the United States. Curves for the four terminal years indicated are generated by connecting points predicted on the assumption of simple constancy of trends, starting from the years indicated on the horizontal axis. Births are adjusted for underregistration.

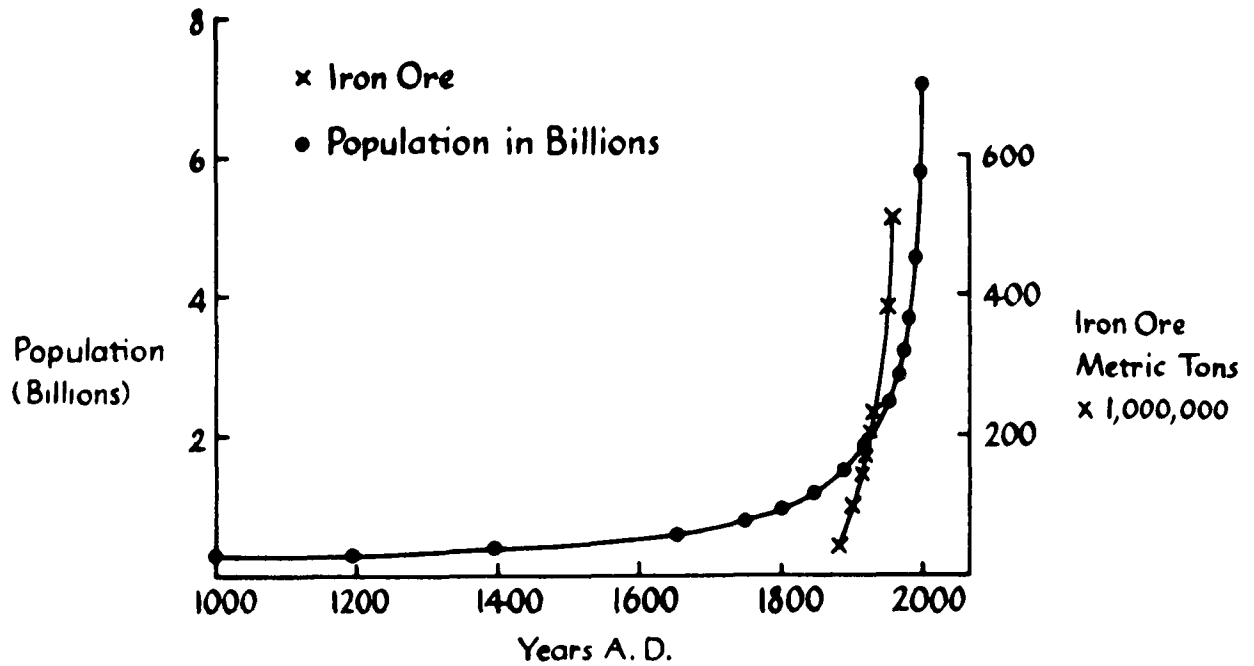
Figure 6



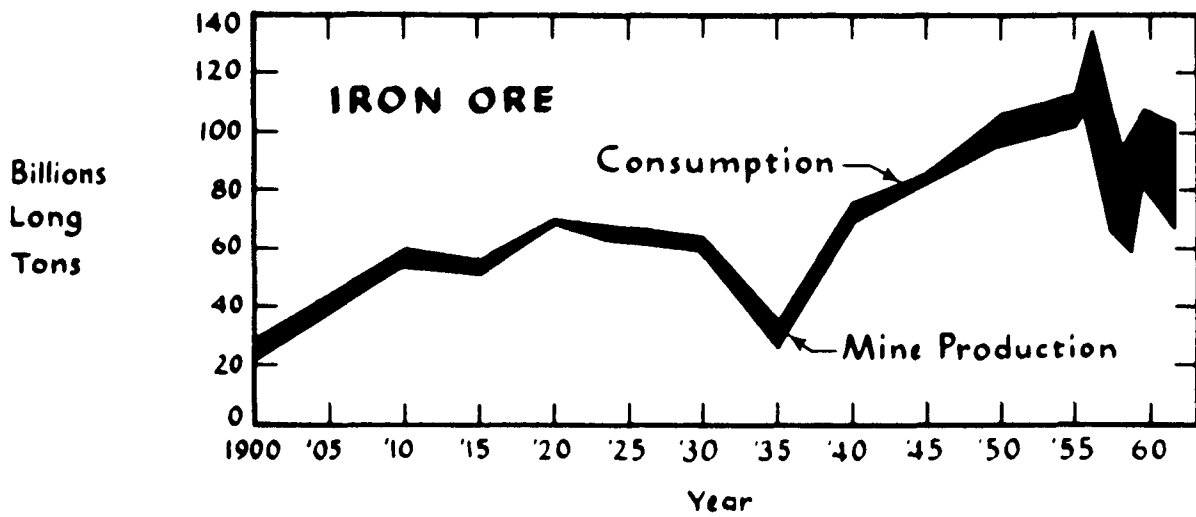
sharp rise in the world's population and a corresponding very sharp rise in the world's demand for iron. For a developed country, like the U.S., the bottom portion of Figure 8, shows that we are now importing a portion of our raw materials. In Figure 9, you see that we are importing copper. In fact, we are importing a portion of almost all of our materials from overseas.<sup>9,10</sup> We are producing less than we are using. We are, along with the other developed countries of the world, now dependent upon external sources to maintain our standard of living. Let's recognize that the mineral resources of the world are not equally distributed among the continents and the nations of the world, and that during the current period of history we are depleting the richest mineral deposits and districts that ever existed. These were created over billions of years, and we're depleting them in decades.

The developed nations are becoming increasingly dependent upon the less developed nations to supply materials. Figure 10 shows a picture of a new Buick, however, the metal composition of the car happens to be for an old Buick, because I had to get it from scrap data.<sup>11</sup> Where is the material for U.S. automobiles coming from? You see that about 3/4 of a ton is coming from foreign sources.<sup>12</sup> We as individuals in this country, are creating these demands. And thus, in a way, we are creating a mandate upon our government to secure by political or military means these raw materials obtainable only beyond our own boundaries.

To better understand the magnitude of the problem, I took the Bureau of Mines' figures for commercial grade ore reserves -- both known and assumed

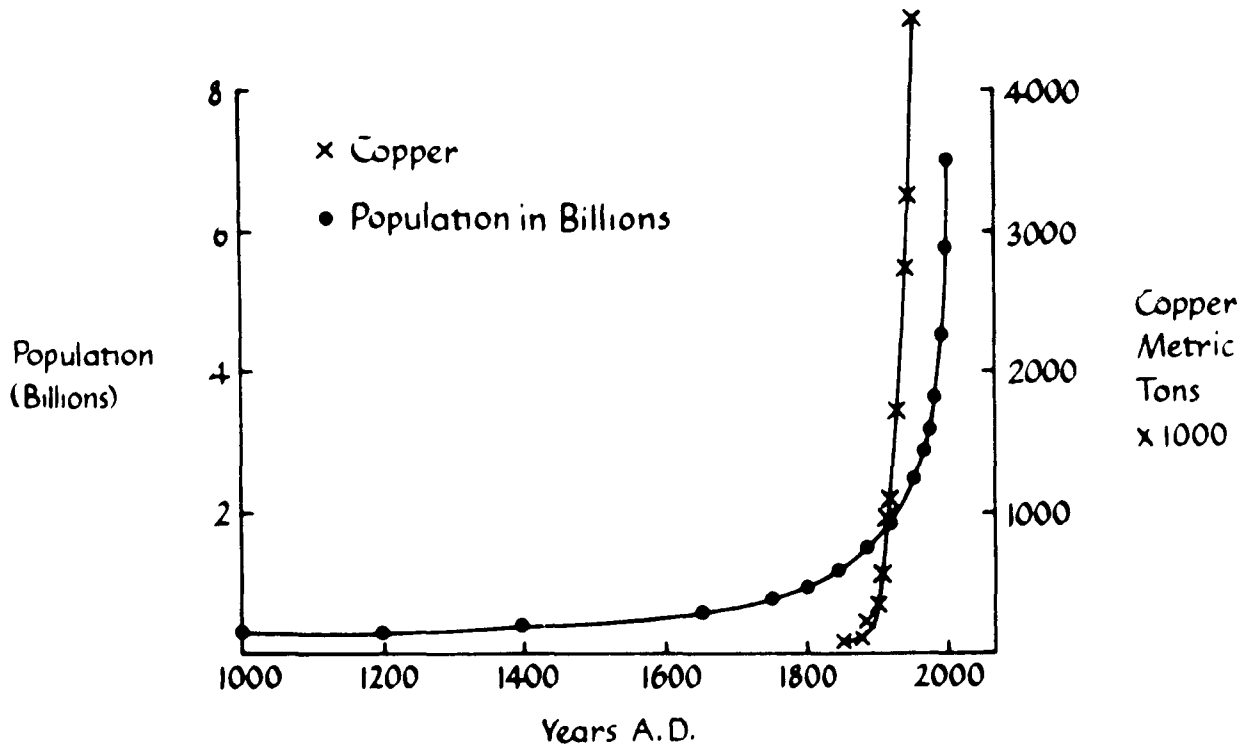


WORLD PRODUCTION

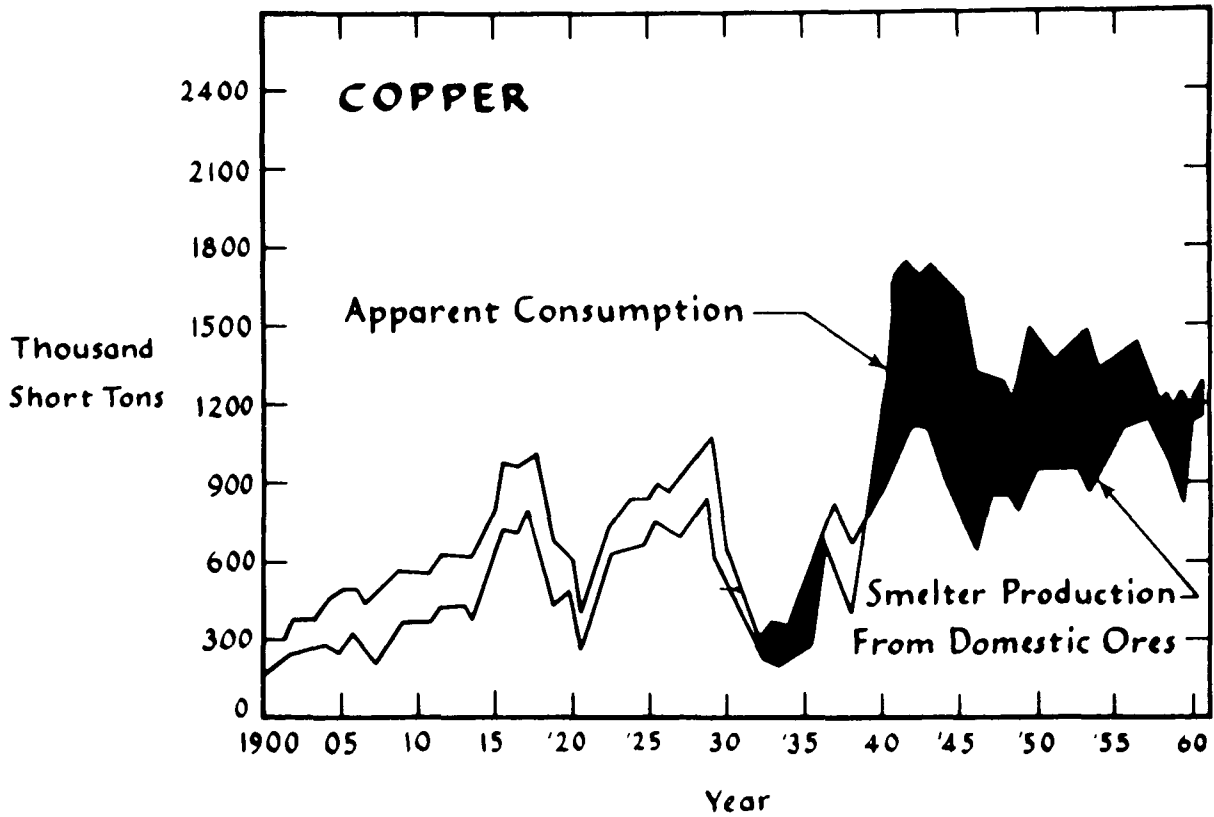


U.S. INCREASING DEPENDENCE ON IMPORTS

Figure 8



**WORLD PRODUCTION**



**U.S. INCREASING DEPENDENCE OF IMPORTS**

Figure 9

## FOREIGN METALS IN A U.S. AUTOMOBILE

	<u>TOTAL POUNDS</u>	<u>FROM FOREIGN COUNTRIES</u>	<u>POUNDS OF FOREIGN METAL</u>
IRON	3,705	36%	1,334
COPPER	52	38%	20
LEAD	24	58%	14
ALUMINUM	48	89%	44
ZINC	123	59%	77
OTHER MATERIALS	415	—	—
	<u>4,367</u>		<u>1,489</u>

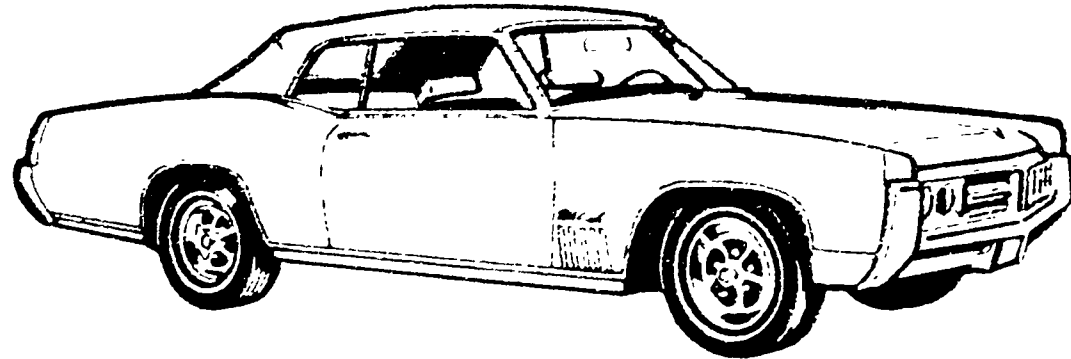


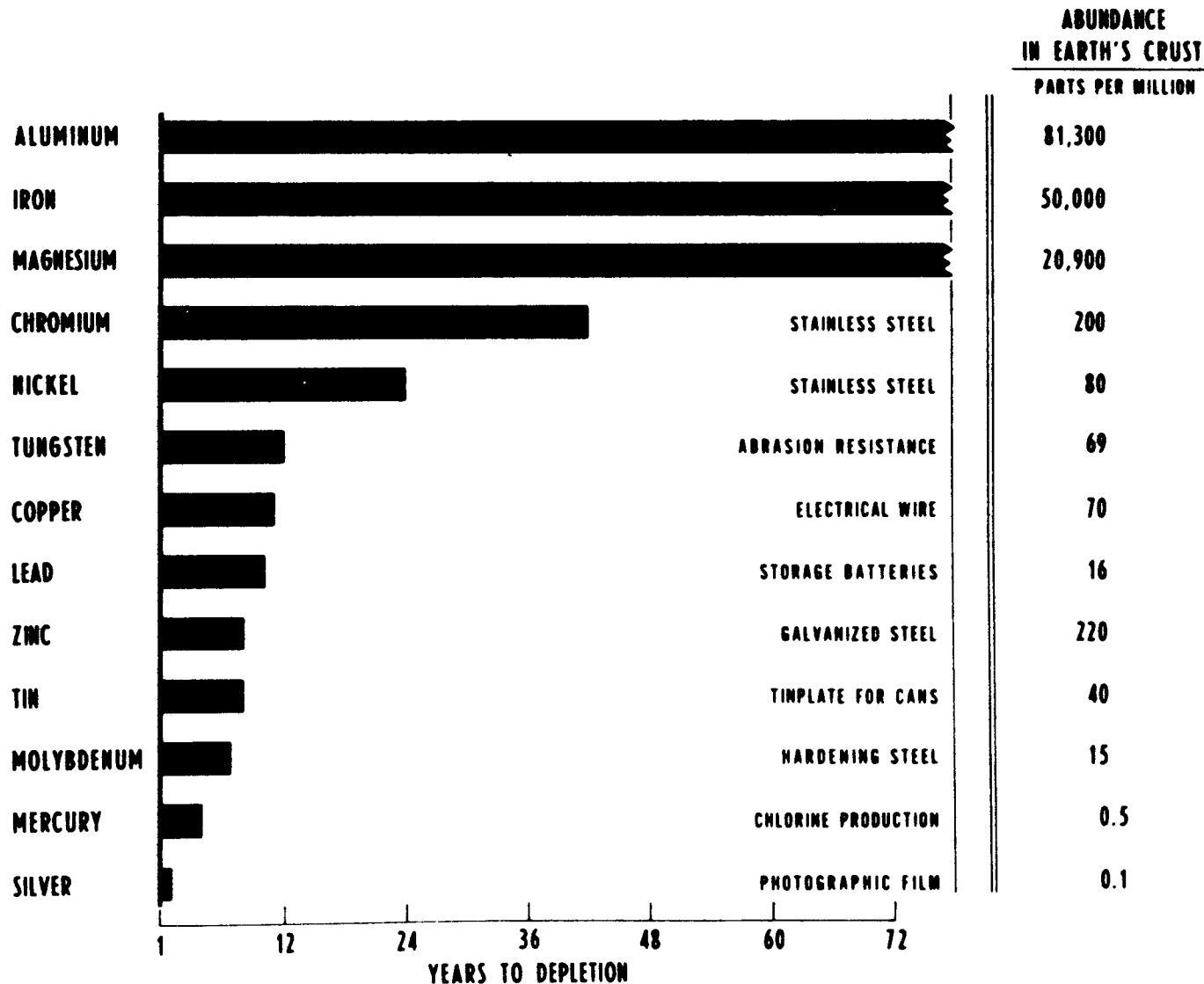
Figure 10

reserves.<sup>12</sup> Then I projected that the present population of the world had our standard of living and asked how long would these reserves last. Figure 11 shows what would happen during the lifetime of an individual, in say 70 years. We see that within decades we would really be depleting all of what you might call the vitamins of our industrial society. The materials needed for the galvanized steel, the stainless steel, the copper wire, the photographic film, and so on -- and you must also recognize that these metals occur in the earth's crust in very few parts per million.<sup>13</sup> Once the high quality ore is gone, we will have a very hard job obtaining additional new metal. Now there are certain metals that are relatively abundant such as aluminum, iron, and magnesium that we don't have immediate worries about. Figure 12 shows that the grade of ore we have been mining, for instance in copper, has been decreasing in the U.S.<sup>14</sup> Figure 13 shows that although our production of ore is changing only slowly, the amount of energy being used to obtain these ores has recently rapidly risen.<sup>14</sup> Increased energy will be required as one goes to the lower grade ores.

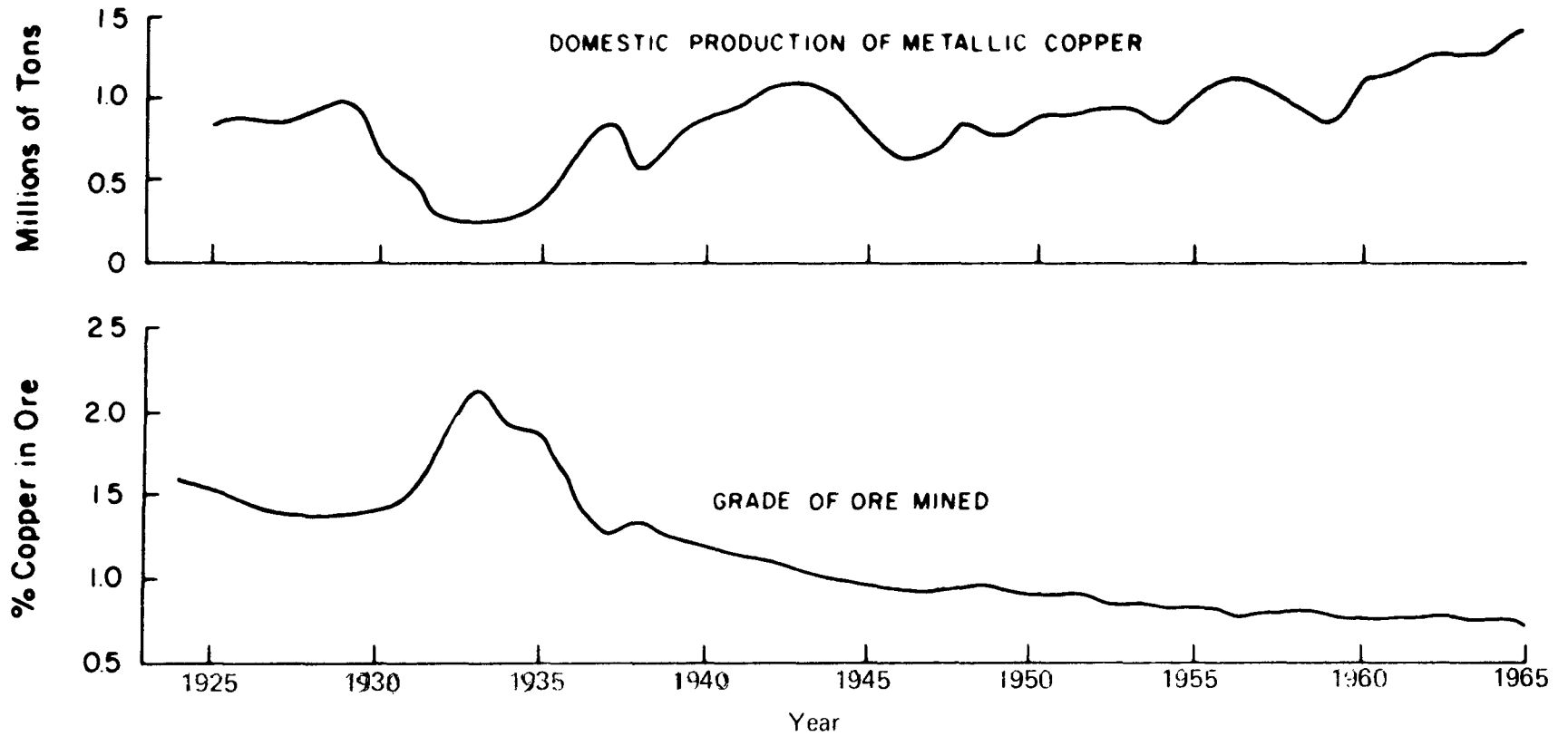
#### POLLUTION

Lets look at the question of pollution. We really don't use anything up; actually all we do is alter its form. Figure 14 is a plot of the refuse production per person in the U.S. versus time.<sup>15</sup> These data were obtained before the Solid Wastes Program of HEW made a careful study.<sup>16</sup> That study showed we are already generating 7 pounds of household, commercial, and municipal wastes per person per day. And that if you add on the wastes being produced by the industries that provide our high standard of living you must add another 3 pounds, so each person in this country is now

# DEPLETION OF WORLD RESERVES OF COMMERCIAL GRADE ORES IF WORLD POPULATION HAD U.S. LIVING STANDARD

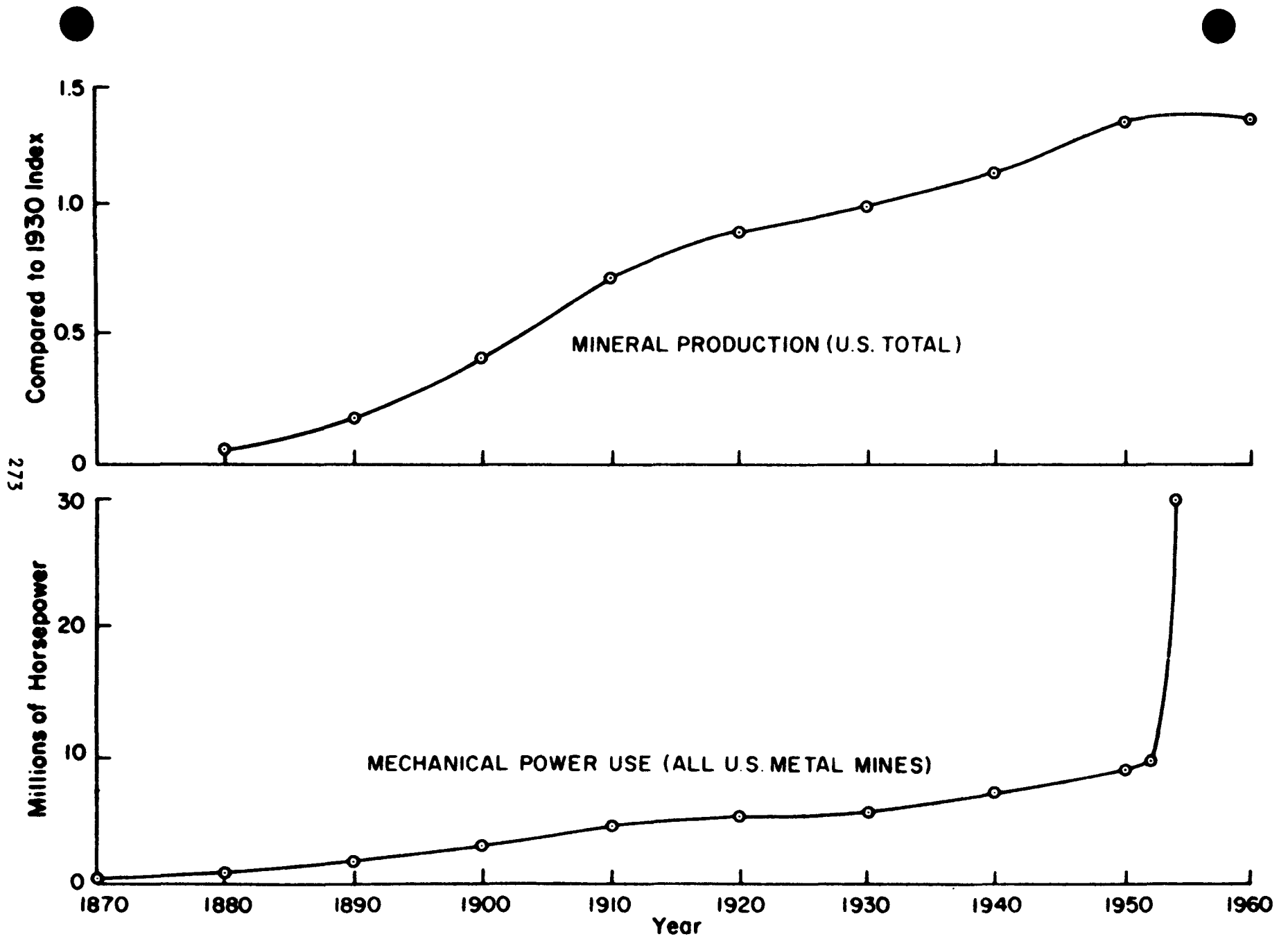






GRADE OF COPPER ORE MINED IN U. S.

Figure 12



INCREASING ENERGY USE FOR METAL PRODUCTION

Figure 13

responsible for about 10 pounds of waste per day. Some interesting numbers for total waste production in the United States were given for 1967: household, commercial and industrial total 360 million tons per year. To that add agriculture waste which is 550 million tons, animal waste (for all the fine steaks we eat) -- 1 1/2 billion tons, and mining wastes - over 1.1 billion tons. So the grand total is something like 2 1/2 billion tons per year.

Now what do we do with all this? There are really three sinks we can put it in. We can put it in the ocean, and that's what we do with a lot of it. We essentially use the rivers as a means of transporting the wastes to the ocean. This creates what is commonly known as water pollution. Another way is to burn it and this, of course, releases particulate matter and CO<sub>2</sub> and other gases to the air. We looked at how much CO<sub>2</sub> would be released in the year 2000 if all of the municipal refuse was burned and it was 444 million tons per year, just from the U.S. alone.<sup>17</sup> Another alternative is to bury it. From Figure 15 we see that between 1965 and 2000, 10 billion tons would be accumulated just from the municipal refuse.<sup>15</sup> If all this were compacted and disposed of by sanitary landfill, it would require burial to a depth greater than 10 feet in a land area the size of the State of Delaware. If a burial depth of 20 feet were used, the land area could be reduced to the size of the State of Rhode Island. The average composition of municipal refuse is shown in Figure 16.<sup>15</sup> The refuse is mostly paper, but it does contain much valuable material such as metal. Practically everything that we use reappears in our wastes. From Figure 17, you see that the municipal refuse is not too bad an ore with almost 7% iron plus a number of other valuable elements, if they only could be

# REFUSE PRODUCTION PER PERSON IN THE U.S.

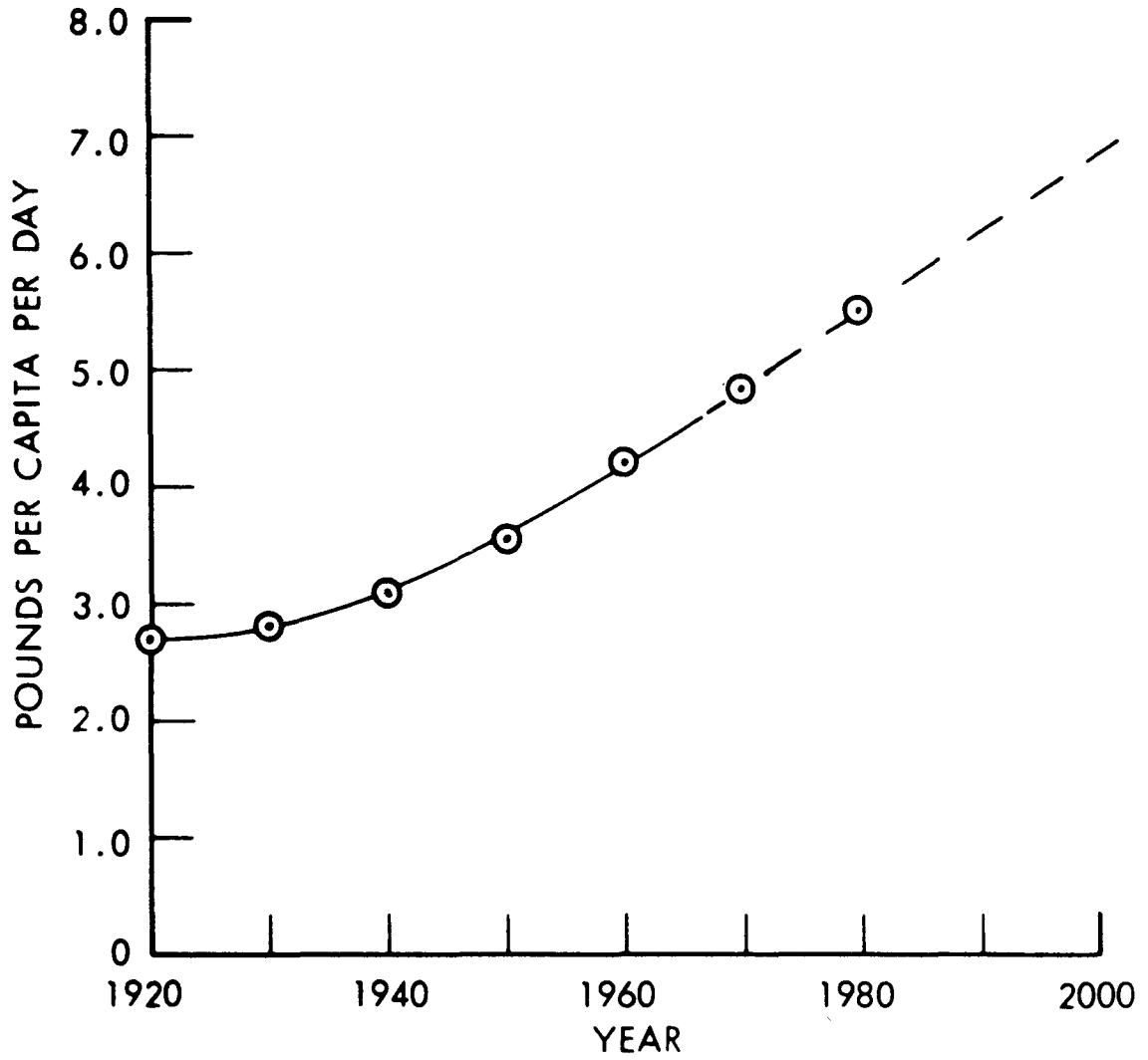


Figure 14

LAND AREA REQUIRED IF ALL REFUSE PRODUCED IN THE  
UNITED STATES IS DISPOSED OF BY SANITARY LANDFILL

YEAR	ACCUMULATED SOLID WASTE (10 <sup>9</sup> TONS)	SQUARE MILES OF LAND AT FINAL DEPTHS OF <sup>(1)</sup>		
		10 FT	15 FT	20 FT
1965	0	0	0	0
1970	0.85	187.5	140.6	93.8
1975	1.84	406.3	304.7	203.1
1980	2.99	671.9	515.6	343.8
1985	4.31	968.8	726.6	484.4
1990	5.82	1296.9	984.4	656.2
1995	7.55	1687.5	1265.6	843.8
2000	9.49	2125.0	1593.8	1062.5

(1) BASED ON FINAL COMPACTION TO DENSITY OF  
32 LBS/FT<sup>3</sup>.

Figure 15

**AVERAGE COMPOSITION OF MUNICIPAL REFUSE (% BY WEIGHT)**

<u>RUBBISH (64 PER CENT)</u>	
PAPER, ALL KINDS	42
WOOD AND BARK	2.4
GRASS	4.0
BRUSH	1.5
CUTTINGS, GREEN	1.5
LEAVES, DRY	5.0
LEATHER GOODS	0.3
RUBBER	0.6
PLASTICS	0.7
OILS, PAINT	0.8
LINOLEUM	0.1
RAGS	0.6
STREET REFUSE	3.0
DIRT, HOUSEHOLD	1.0
UNCLASSIFIED	0.5
<u>FOOD WASTES (12 PER CENT)</u>	
GARBAGE	10.0
FATS	2.0
<u>NON COMBUSTIBLES (24 PER CENT)</u>	
METALS	8.0
GLASS AND CERAMICS	6.0
ASHES	10.0
	<hr/> 100.0

Figure 16

ELEMENTS RECOVERED % BY WEIGHT

---

C	33.345
O <sub>2</sub>	44.700
H <sub>2</sub>	4.800
Si	4.610
Al	3.120
Ti	.112
Fe	6.645
Cu	1.377
Mg	.240
Na	.730
S	.202
Pb	.019
Mn	.011
Ni	.005
Sn	.018
B	.003
Zn	.063
	<hr/>
	100.000

Figure 17

recovered at low cost.<sup>17</sup>

ENERGY:

To better understand the importance of energy, return to Figure 2 which shows the per capita energy use by country versus the standard of living. However, in studying a curve like this, we often forget that it presents an incomplete picture of the total energy we use. The main source of energy for the world is the sun rather than the fossil fuels, water power, and nuclear power which serve as auxiliary energy. This auxiliary energy is part of a feedback loop that determines the efficiency with which the sun's energy is used by mankind as shown in Figure 18. Thus, this auxiliary energy is the true currency of the world for it determines your standard of living -- its not dollars or rubles, it's energy. Without energy we would quickly return to a primitive state of existence.

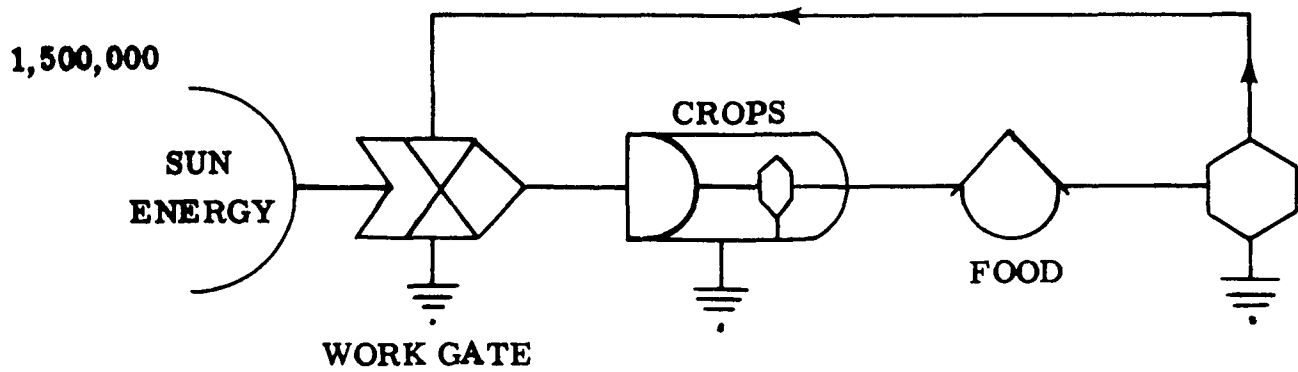
If you have an underdeveloped society, only the sun's energy is available to help the man push his plow. He's able to raise enough food to feed himself and his family who also have to work. You don't have much multiplication in the feedback loop. In an advanced industrial country, what you do is add, as you see in Figure 8, auxiliary energy. You essentially open the gate to allow more efficient use of the sun's energy and you do it by producing fertilizers, pesticides, tractors that can plow, refrigerators that can store the food, transportation means and so on. This enables you to support a large population that lives in cities and thinks of new ways to multiply



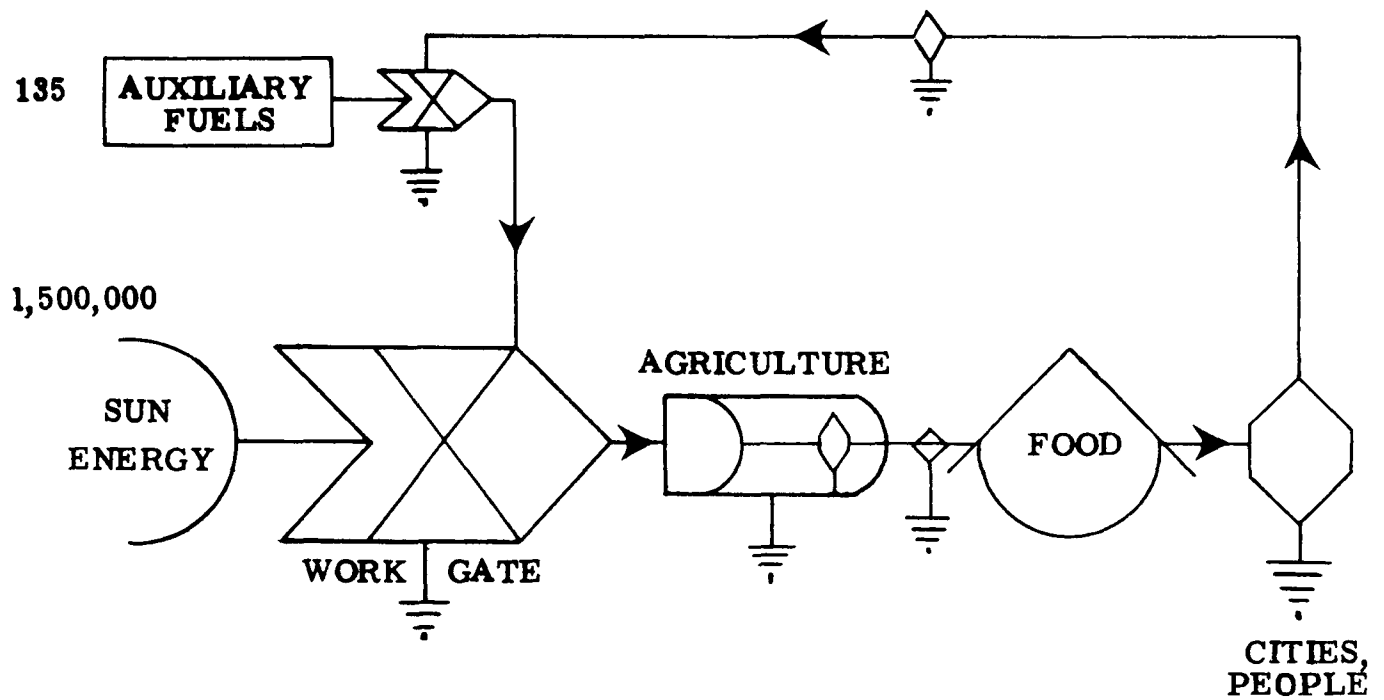
# SIGNIFICANCE OF AUXILIARY ENERGY

## OPENING THE GATE FOR THE SUN'S ENERGY

KCAL/m<sup>2</sup>/YR



## UNDERDEVELOPED COUNTRIES



## ADVANCED INDUSTRIAL COUNTRY

Figure 18

the efficiency of the feedback loop. You can also see what happens in those food programs where we send food to a country that has not opened its energy gate very far. What you do is bulge the population in the cities. They still don't have the auxiliary energy needed in the feedback loop to handle this population that becomes dependent upon you. And both sides get angry. How does an underdeveloped country get out of this situation? Figure 19 shows the building of a dam in India.<sup>19</sup> These specks are people. And they are crawling up and down building the dam. It's not like the pyramids - when they finish, this dam will produce 460 megawatts of power and 3,000,000 acres of irrigated land. This is the first step on building up an industrial society -- the use of human power (sun's energy) to build the auxiliary energy necessary to amplify the amount of the sun's energy that can be harnessed by civilization. This is about the only way you can do it. Figure 20 shows that you can go up orders of magnitude in the amount of food produced by the use of auxiliary energy.<sup>18</sup> In what form do we use auxiliary energy in the U.S.? Figure 21 shows that most of the energy is not electrical, although we seem to talk as if electrical was the only form of energy.<sup>20</sup> In 1960 only about 1/4 was electrical, in 1980 about a third and we're projecting that in the year 2000 something like a 50-50 split. But let us not forget that we are talking in the controlled fusion program of creating a new source of energy -- not just a source of electricity.

What energy sources are possible -- some are limited while others are "infinite". The ones that are limited are shown in Figure 22.<sup>21</sup> Fossil fuels -- these are limited, irreplaceable, and they're being used up at a fast rate. The questions about their longer term use

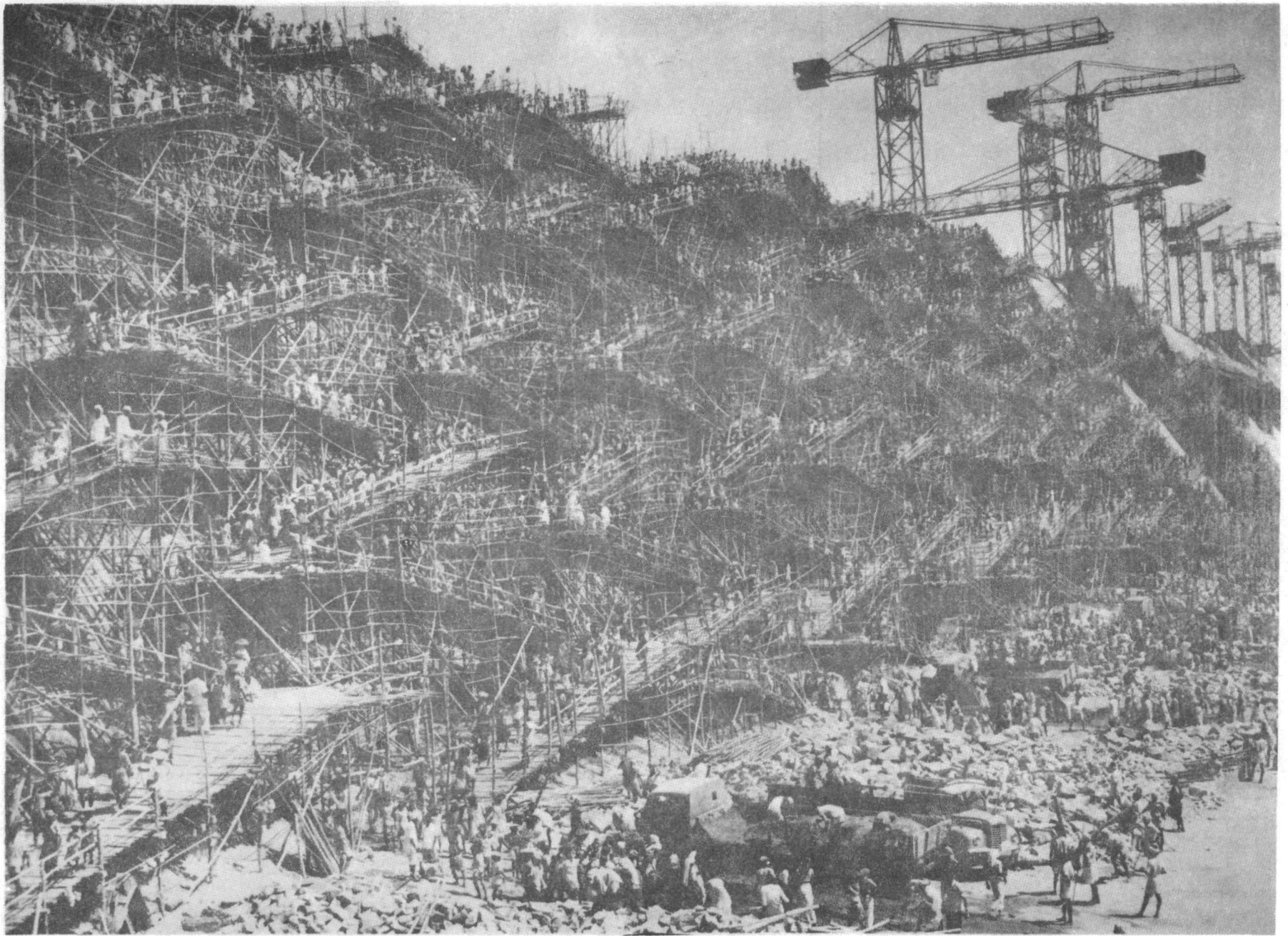


Figure 19

# ENERGY AND FOOD

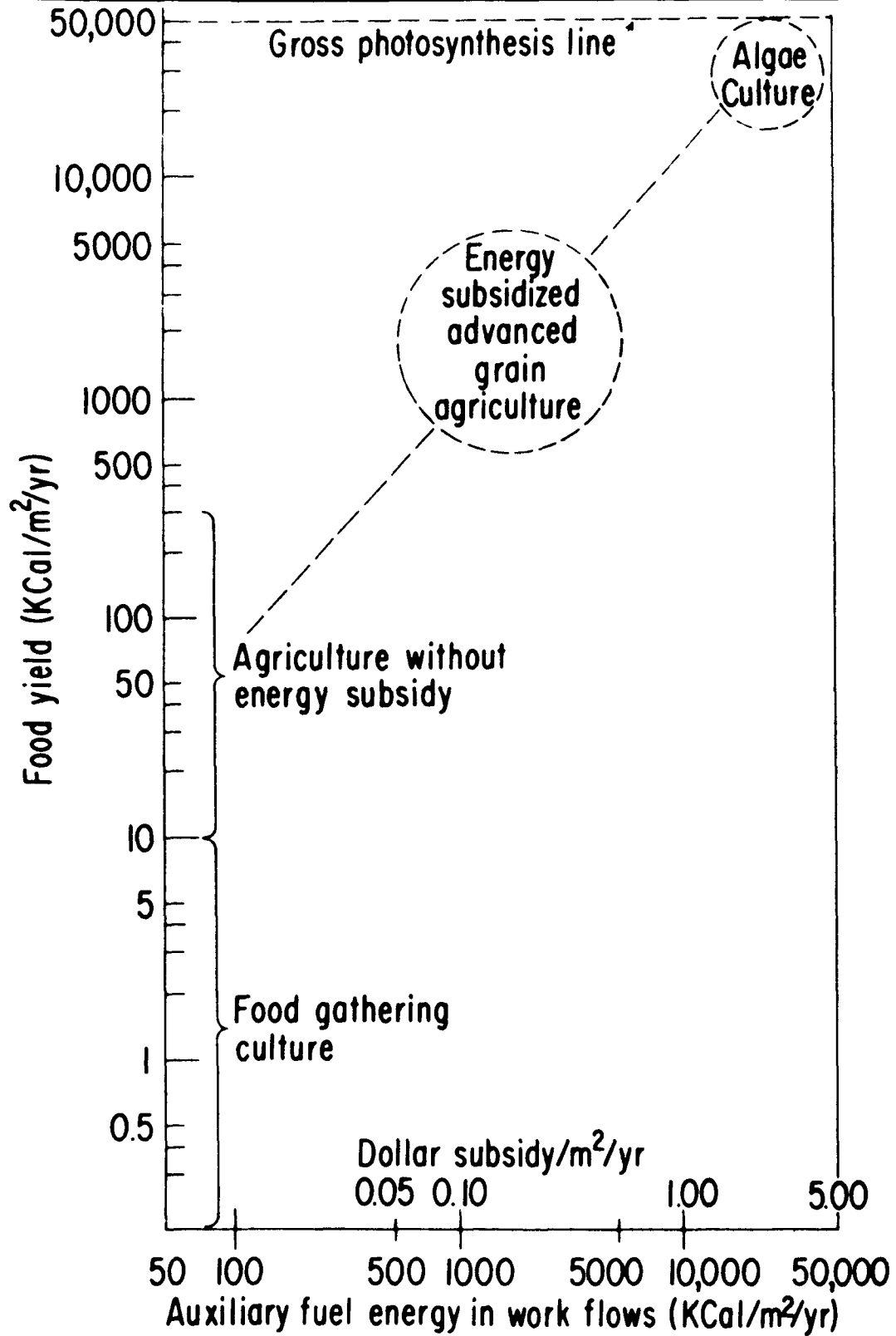


Figure 20

# U.S. ENERGY CONSUMPTION 1920 - 2000

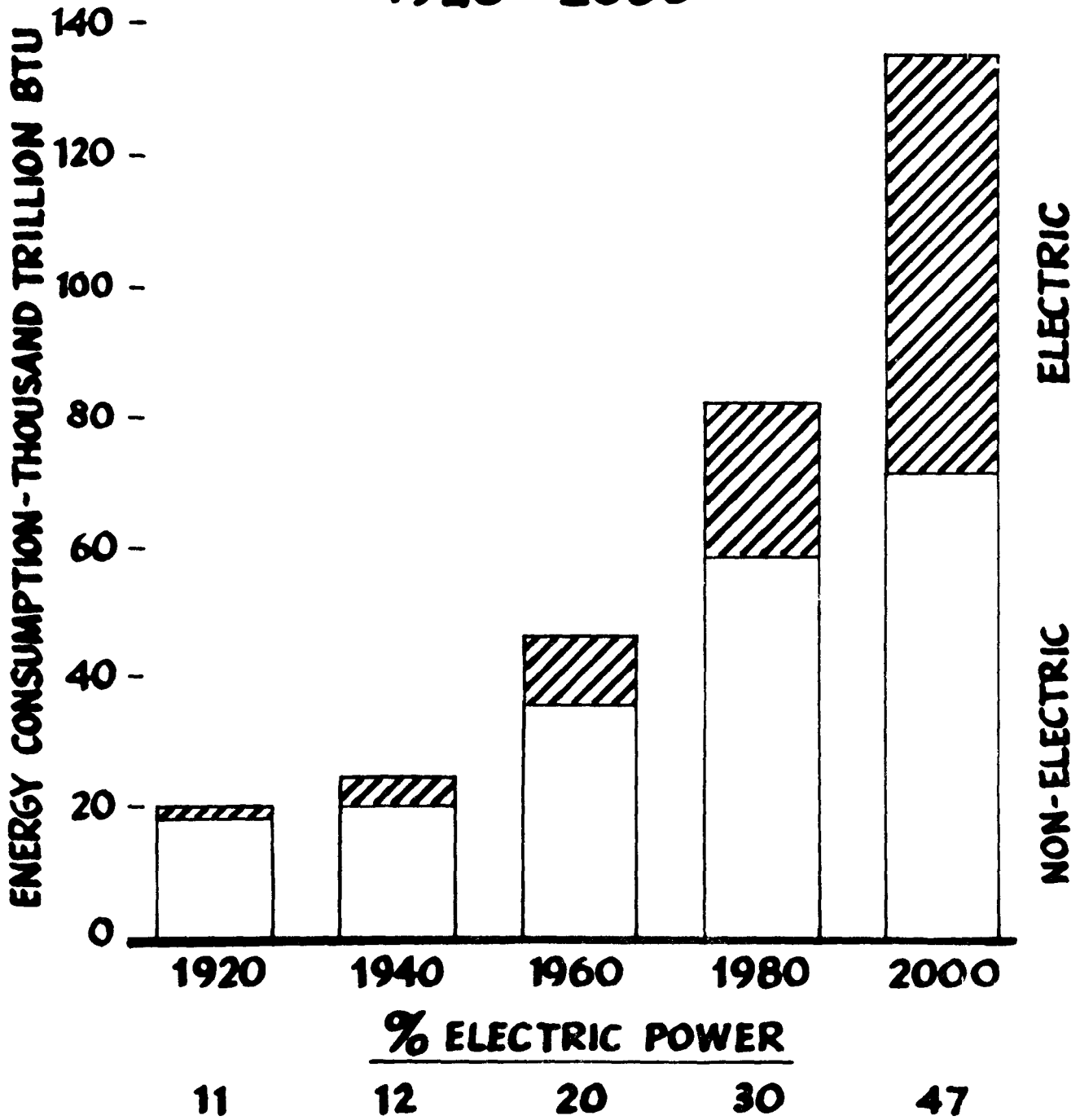


Figure 21

center upon the CO<sub>2</sub> build up in the atmosphere. How does CO<sub>2</sub> affect the weather? What's the affect on photosynthesis? Does it affect the pH in the first layers of the ocean and hence sea life?

The conventional nuclear fission reactor is another auxiliary source. These are the converter reactors such as the pressurized water reactors. They have a very inefficient conversion - 1 or 2% efficiency - for recovery of the fuel heat content from the uranium. And they are using up the U-235. Now that raises the question brought up in the preceeding paper about fission fuel availability at low cost and the need for moving on to more efficient fission reactors -- breeders. If you look at our fission fuel reserves in the U.S., you can see that the "infinite" reserves are very expensive to recover. Take the projected nuclear plants to be built by the year 2000 in the U.S. The fuel needs over their expected life would raise the cost of uranium into the region of \$30-\$50/pound, if breeders do not come in.

In Figure 23, we have what I will call the "infinite energy sources"<sup>22</sup>. First, there will be the fission breeder reactors whose development is further advanced than for any of the other major "infinite energy sources". The questions raised for long term use are on handling of fission products since every ton of uranium produces a ton of fission products. There is safety, which is being worked on hard to eliminate any potential hazards. The radiological concentration processes -- an area where scientific data is still not fully known. Also, the proliferation of nuclear material requires careful study.

In solar energy, the main question that remains is our ability to

## LIMITED ENERGY SOURCES

### FOSSIL FUELS:

**CONCERN:** A LIMITED AND IRREPLACABLE NATURAL RESOURCE

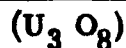
**QUESTION:** ATMOSPHERE CO<sub>2</sub> BUILDUP - EFFECT ON WEATHER, PH OF OCEAN, PHOTOSYNTHESIS

### FISSION WATER REACTORS:

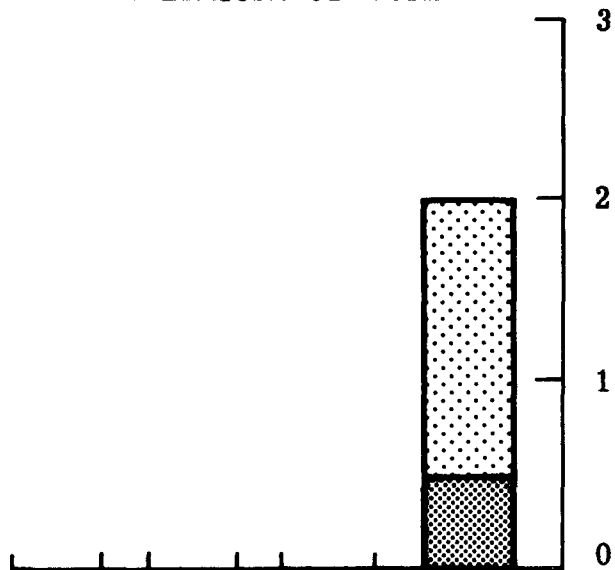
**CONCERN:** USE OF IRREPLACABLE U-235 "SEED - CORN" - ONLY 1-2% EFFICIENCY FOR RECOVERING FUEL'S HEAT CONTENT

**QUESTION:** FUEL AVAILABILITY AT LOW COST

#### U. S. URANIUM RESERVES

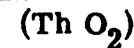


MILLIONS OF TONS

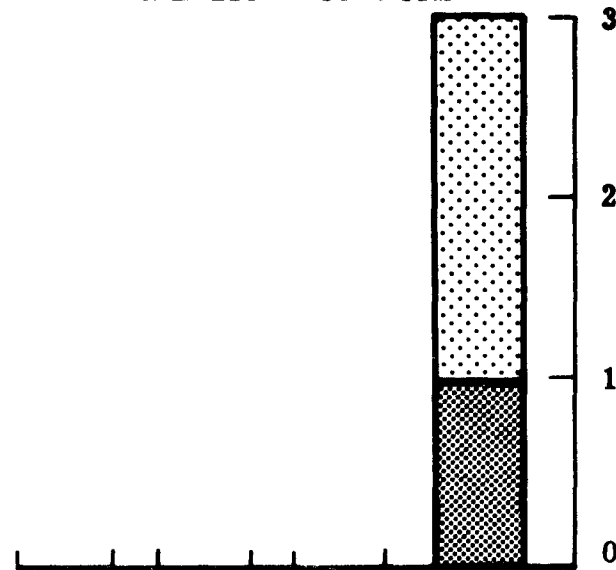


\$/lb.	5-30	30-50	50-100	100-500
TOTAL	1.5	8.0	15.0	2000
ASSURED	(.8)	(5.0)	(6.0)	(500)

#### U. S. THORIUM RESERVES



MILLIONS OF TONS



MILLIONS OF TONS	5-30	30-50	50-100	100-500
TOTAL	.6	10.0	25.0	3000
ASSURED	(.2)	(3.0)	(8.0)	(1000)

Figure 22

"INFINITE" ENERGY SOURCES

MAJOR:

FISSION BREEDER REACTORS:

QUESTION: FISSON PRODUCTS, SAFETY, AND RADIOECOLOGICAL CONCENTRATION PROCESSES, PROLIFERATION OF NUCLEAR MATERIAL

SOLAR ENERGY:

QUESTION: TECHNOLOGICAL ABILITY TO ECONOMICALLY AND EFFICIENTLY CONCENTRATE THE LOW RADIATION ENERGY DENSITY

FUSION REACTOR:

QUESTION: ADEQUATE SCIENTIFIC UNDERSTANDING OF PLASMA PHYSICS (CONFINEMENT AND SCALING LAWS)

MINOR:

OTHER ENERGY SOURCES:

INSUFFICIENT TO MEET FUTURE DEMANDS

WORLD POWER CAPACITY

THOUSANDS OF MW

WATER	2,900
TIDAL	1,100
GEOHERMAL	10
WIND	NEGLECTIBLE
BUT: WORLD ENERGY REQUIREMENTS	= 30,000
BY YEAR 2000	

Figure 23



economically and efficiently concentrate the low radiation energy density that is coming in from the sun. There are proposals for the use of satellites to avoid problems created by weather on earth.<sup>23</sup> Maybe there could be research and development programs on how to obtain more efficient solar energy. Almost no effort is being devoted to harnessing solar energy.

For controlled fusion we need a better understanding of the plasma to control confinement and establish scaling laws. Right now we're getting confinement times which are better than needed, but we have to put all three properties - confinement, temperature and density -- into one device and then check the scaling laws. This is the current world objective. Each of these three "infinite" energy sources requires extensive engineering and materials' development -- and fusion is at an early stage.

Looking at other energy sources you find ones that are infinite but insufficient to meet future annual demands. Water -- if you took all the water power that could be developed in the world you could produce only about 1/10 of the world's energy requirements for the year 2000. Geothermal, or wind power, will not meet or come close to meeting the needs that we are going to be facing in this world. We have a very limited choice. We don't have many options. And none of these options are guaranteed at this moment. Environmental limitations could further restrict us. If we don't get an unlimited energy source that is relatively inexpensive or at least somewhere close, we will be in trouble. We will be unable to support the large world population at a standard of living anywhere near what we have now in this country or even what less fortunate countries now hope to obtain.

## Fusion Research

Most of our efforts in fusion have been in trying to advance the plasma parameters towards the regime needed for fusion reactors.<sup>24a</sup> Figure 24 shows how plasma temperature for fusion like densities have moved up. When the fusion program was in its formative stages, there were experts who said we could never reach this minimum ignition temperature of 50 million degrees. Temperature was slowly increased and then suddenly the ignition point was passed and nobody noticed it too much. Now we are well past ignition temperature for fusion reactors and can produce copious amounts of neutrons at will. Incidentally, the temperature of the center of the sun is down there at a tepid 20 million degrees. You barely can see the temperature of a conventional arc on the figure, it is so cold.

In Figure 25 confinement time for toroidal systems is shown. Again we have a long period with not too much change. And then the curve breaks sharply upward. As you heard in the talks earlier this week, the confinement time needed is affected by the size of the system. The Rose study for instance shows that for large systems less than 200 Bohm times are adequate.<sup>24b</sup> The confinement time needed for a fusion reactor is not a point but a broad band region and we're now in that region. In fact we are above it. The toroidal multipole experiment at Gulf General Atomics, although at low temperature and low density, has recently shown nearly classical confinement.<sup>25</sup> Incidentally, the Soviet Tokamak T-3, that appears on the chart operates at about 10,000,000°C.

TEMPERATURE (°C)

PLASMA HEATING

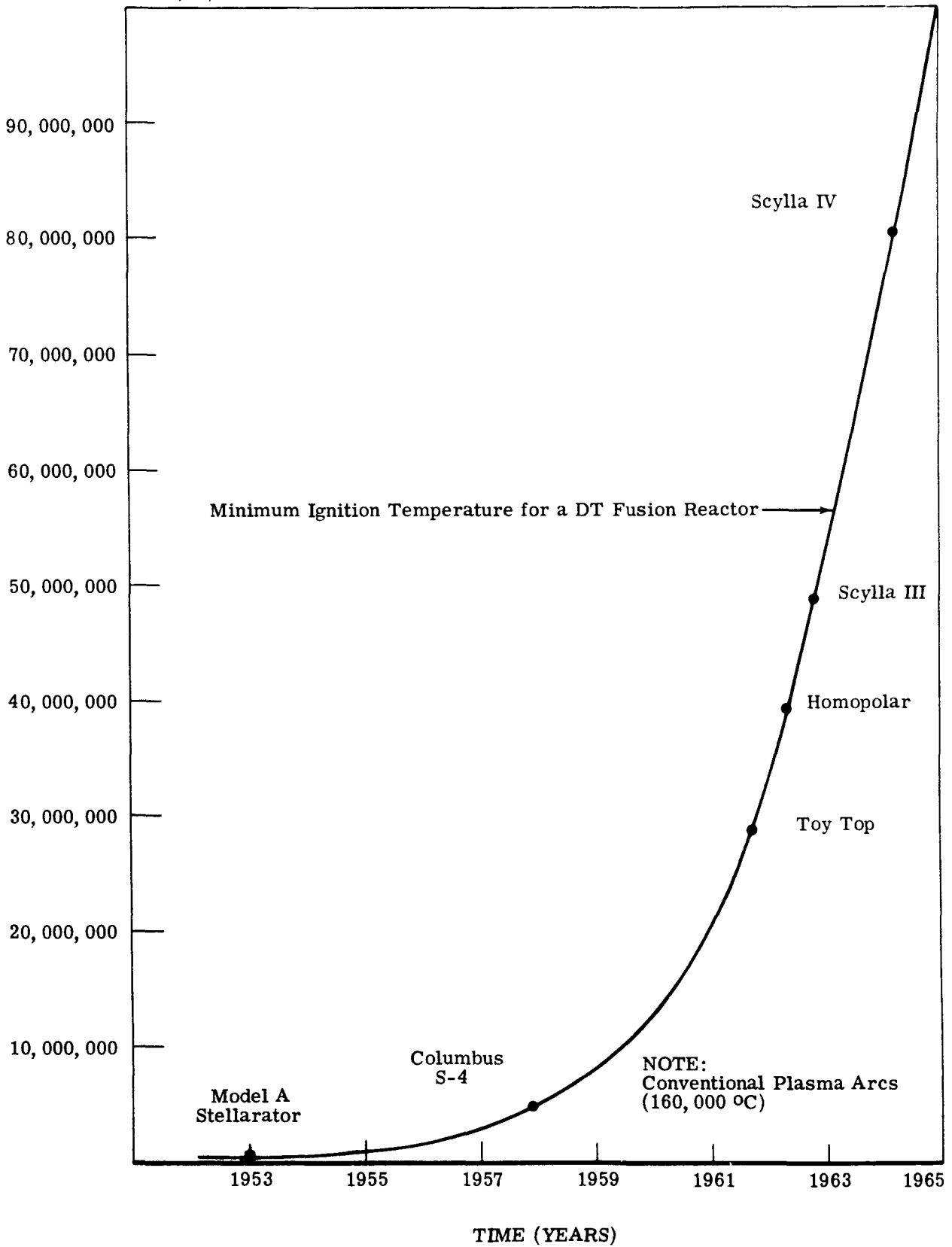
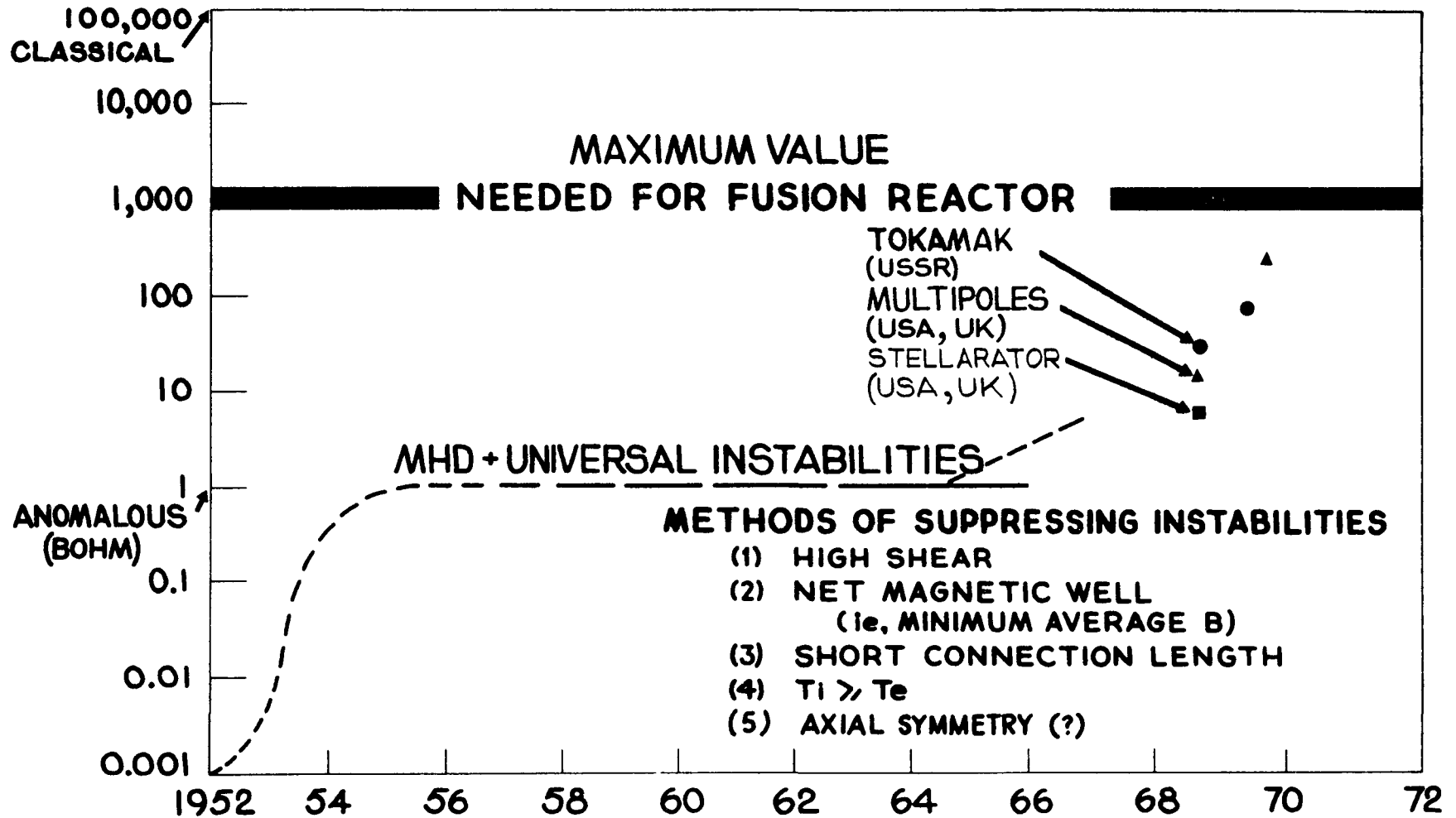


Figure 24

# CONTAINMENT TIME IN TOROIDAL DEVICES (RELATIVE SCALE)



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Figure 25

Figure 26 indicates the improvement in an open system. This is the 2X experiment at the Lawrence Radiation Laboratory that is running at 80 million degrees. The plot by years, shows that we have moved closer and closer to what we call classical end loss confinement. We're within a factor of two of this which is probably good enough for reactors. Fast pulsed systems such as the theta pinch likewise are exhibiting excellent confinement properties. In Figure 27, the experimental points for a British theta pinch experiment show that the cross-field plasma loss, instead of being the fast Bohm rate or even 1/10 of that, were at the slow classical rate. The Scylla experiments at the Los Alamos Scientific Laboratory are operating in the range of 50-80 million degrees. Our next objective is to close the open-ended Scylla type experiments and make a toroidal system (Scyllac) that hopefully will maintain this good confinement. These are some points that have given optimism over the last three years. Most of this data is new. It is the reason we started technology studies, and the reason for the Culham conference on Nuclear Fusion Reactors in September 1969, and why we are meeting here today.

#### FUSION REACTORS:

First, I want to point out that serious fusion reactor technology studies have been underway for less than 3 years. At present these studies represent only a very small amount, something like 2%, of our annual effort. Over three years we have spent about  $1\frac{1}{2}$  million dollars on this work. We really do not have details. Many of you may see better ways to accomplish our technological objectives. Don't be afraid to express your ideas because the amount of effort that has gone into this area is small compared to the magnitude of the problem.

# DEUTERIUM PLASMA CONFINEMENT IN 2X FACILITY

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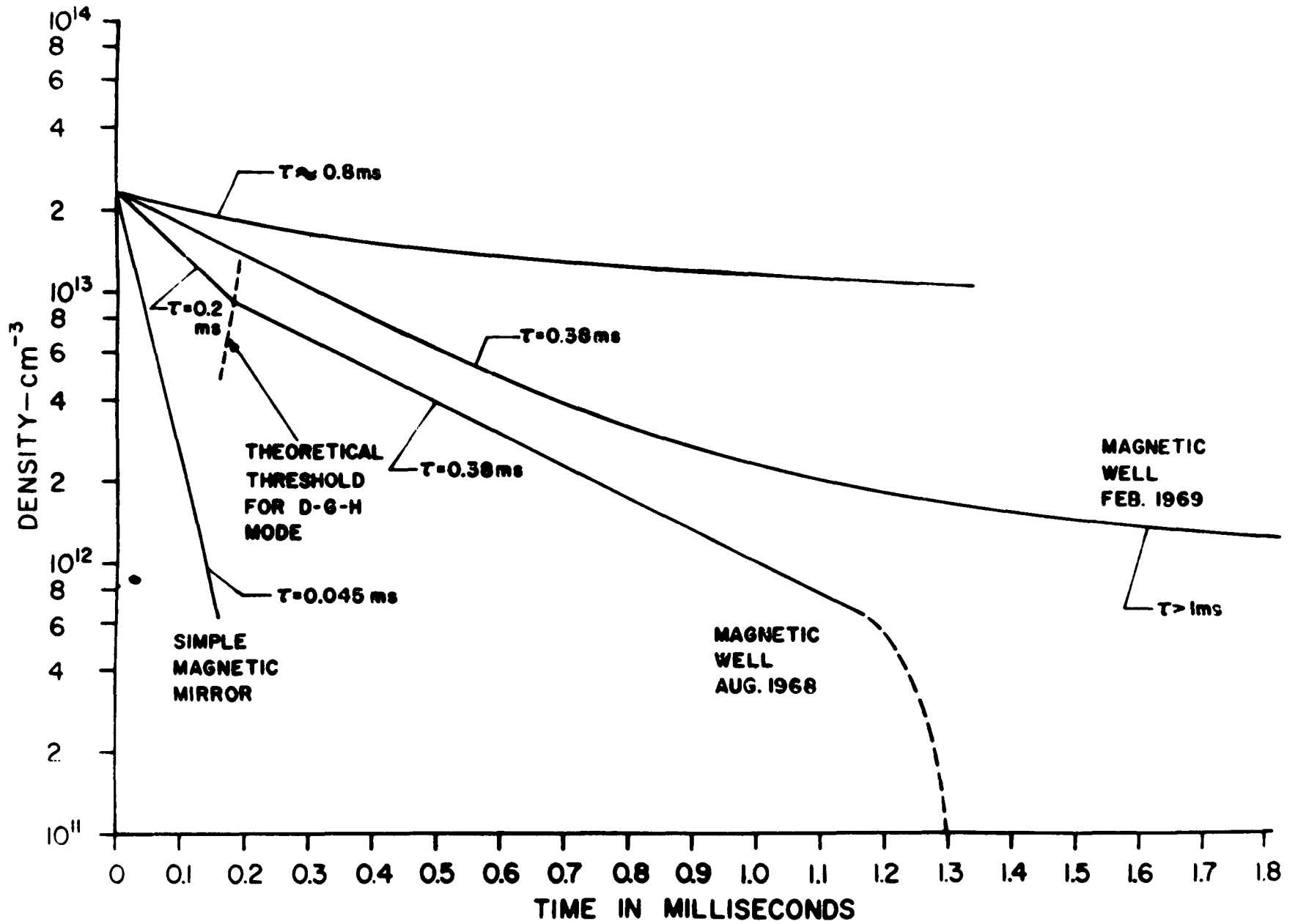


Figure 26

# RADIAL DENSITY DISTRIBUTION IN BRITISH 8-METER THETA-PINCH

(40KV; 20 mTorr; 19.5  $\mu$ sec)

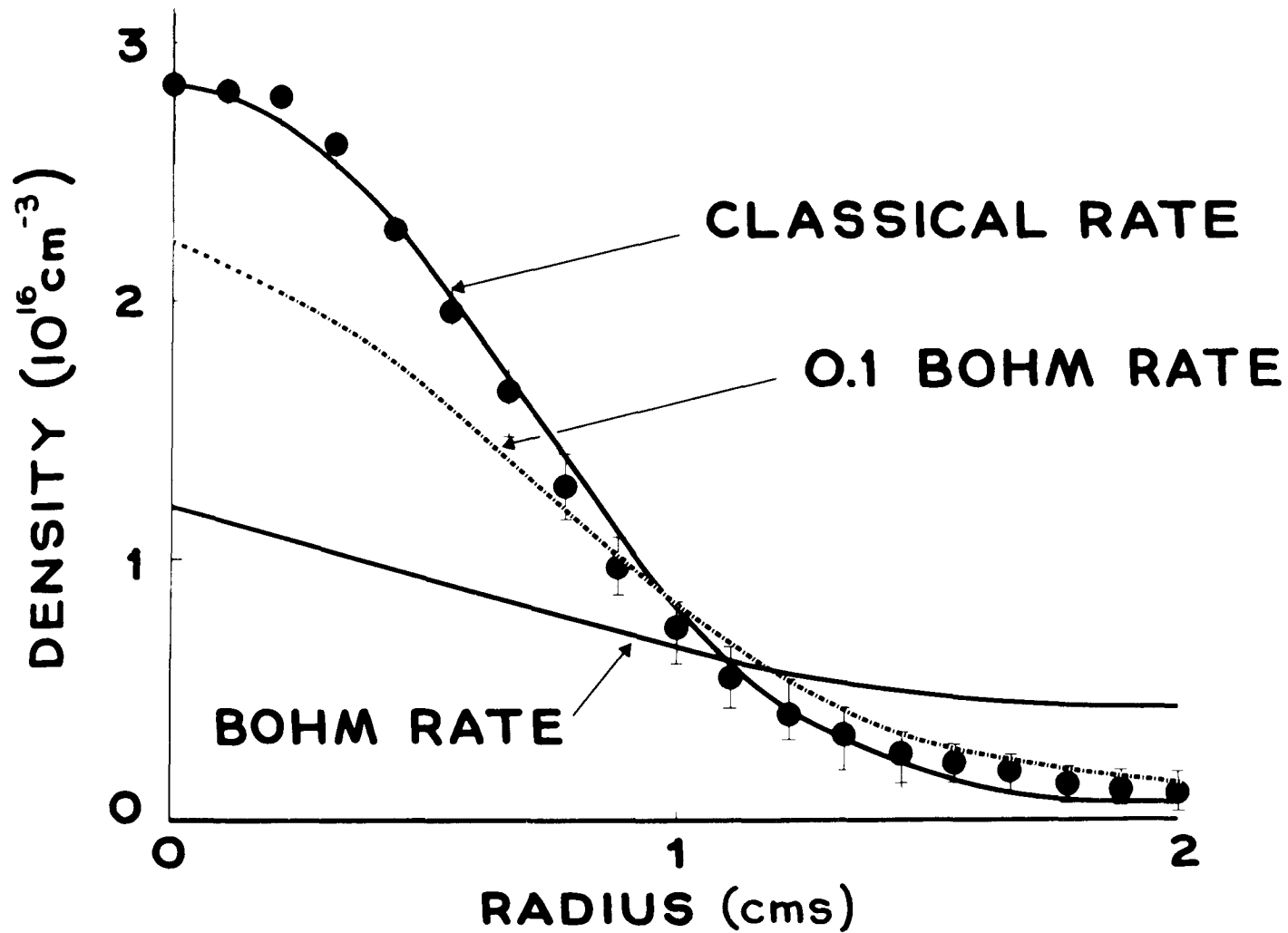


Figure 27

Deuterium-tritium (D-T) is the fuel cycle usually considered in fusion technology studies since it requires lower temperatures and fewer engineering unknowns before design of a reactor. But the true potential of fusion eventually may lie elsewhere. The exploration of other fuel cycles has only just begun. Although their unknowns are larger, the effort devoted to resolving these uncertainties is as yet miniscule.

Figure 28 gives the possible fuel cycles for fusion reactors. The first three cycles produce neutrons. The only one with a radioactive fuel is D-T. The fourth cycle, p-Li<sup>6</sup> has been proposed by Dick Post at LRL. As we move into the possibilities of very high efficiency converters and very good plasma confinement, it becomes possible to consider such cycles, for the future. The main advantage of this cycle is that no matter what you do you get no neutrons. Its all charged particles -- excellent for direct conversion. Maybe its the cycle to consider for those advanced space propulsion missions that have been discussed at this conference. It looks, in any case, as if the p-Li<sup>6</sup> cycle could be made self-sustaining although it would be an extremely difficult task.<sup>26</sup>

Figure 29 lists the environmental features of a fusion reactor. I'd like to comment on each of these since they are dependent on the fuel cycle and type of fusion system used. Lack of radioactive waste products: The p-Li<sup>6</sup> cycle would have no radioactivity because there would be no neutrons and no radioactive fuel. If you used the D-He<sup>3</sup> cycle there would be some radioactivity induced in the wall -- nothing



# FUSION FUEL CYCLES

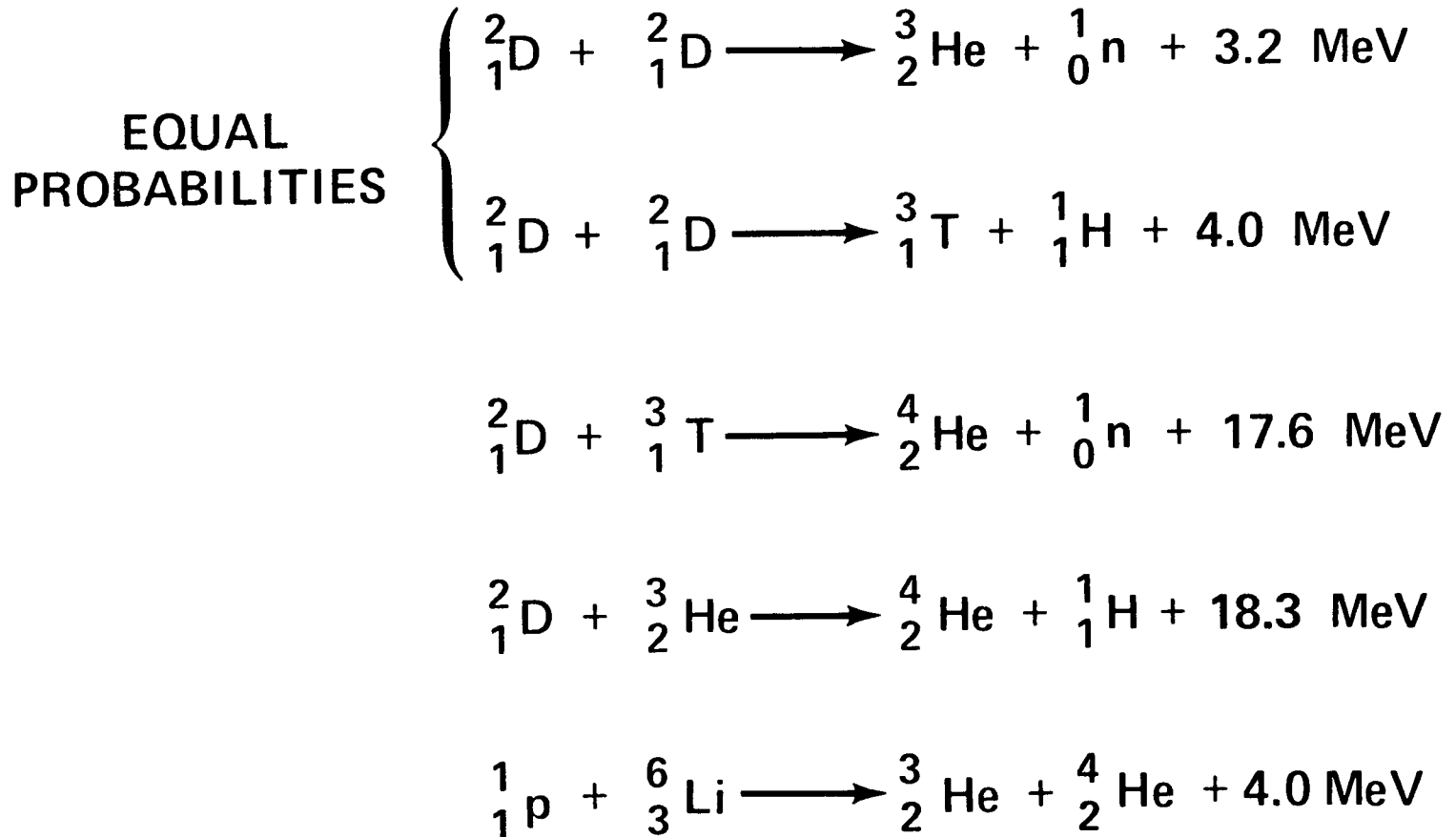


Figure 28

like a D-T, but there is some. There's little or no radioactive fuel because tritium formed from any D-D reaction is burned up very quickly. As you move over to the D-T cycle you have the question of radioactive fuel -- the tritium and how you handle it -- and the intense induced radioactivity in the walls and blanket. Now this is still many orders of magnitude less hazard potential than is present in a fission system.<sup>27</sup> However, when you speak about eventually storing this radioactive structural material, the degree of the problem can vary. If you make the assumption that you have to change the walls every year in a D-T reactor as would be possible with a modular design and you assume a wall material that is going to become highly radioactive you are moving closer to the radioactive waste disposal problems of a fission system -- maybe only two orders of magnitude away.

Inherent safety against nuclear explosion: There is no question that fusion reactors using magnetic confinement will have an inherent nuclear safety that is independent of engineering design. In our discussions the other night, we were talking about very fast pulsed systems which use inertial confinement. There you really have a little explosion. I don't know what can be said about the inherent safety of such systems.

Absence of after heat problem: Possible melt down of a fusion system due to after heat is normally not considered a problem since the fusion nuclear process does not involve the concept of a "critical mass". However, cooling for after heat will have to be provided for fusion systems with intense induced radioactivity.

Low biological hazard in the event of sabotage or national disaster: Again, if you assume the p-Li<sup>6</sup> cycle we have no hazard. For the D-He<sup>3</sup> cycle we have practically no hazard because there are no volatile radioactive

# FAVORABLE ENVIRONMENTAL FEATURES OF FUSION REACTORS

1. LACK OF RADIOACTIVE WASTE PRODUCTS.
2. INHERENT SAFETY AGAINST NUCLEAR EXPLOSIONS.
3. ABSENCE OF AN AFTER-HEAT PROBLEM.
4. LOW BIOLOGICAL HAZARD IN EVENT OF SABOTAGE OR NATURAL DISASTER.
5. REDUCED DANGER OF DIVERSION OF WEAPONS-GRADE MATERIAL FOR CLANDESTINE PURPOSES.
6. RELATIVELY LOW WASTE HEAT, PLUS POTENTIAL OF DIRECT CONVERSION.
7. POTENTIALITY OF "FUSION TORCH."

materials or fuel. For the D-T fuel cycle you must compare the biological hazard for the possible release of tritium to the atmosphere with say volatile fission products from nuclear systems, and there seems to be a very wide margin in favor of fusion.<sup>27</sup>

Reduced danger of diversion of weapon grade material for clandestine purposes: This is a serious problem for the future.<sup>28</sup> If you look at fusion reactors you see that this is no problem with p-Li<sup>6</sup>, D-He<sup>3</sup>, or even the D-D. Now I want to qualify those latter two, because these systems produce neutrons and although they could be designed so that the production of any weapon grade material would be extremely difficult, it would require a careful reactor design and an assumption that the nations of the world have agreed to international control and inspection. When you come to the D-T there is the same qualification but with the additional consideration of tritium. Tritium requires a fission trigger for any weapon potential. But even if it should ever become a weapons' material, it circulates only within the fusion reactor as shown in Figure 30. It does not travel out of the plant. There is one initial fuel shipment of tritium and then the plant breeds its own tritium, which is separated on site. Only non-radioactive lithium and deuterium need be shipped to provide fuel. This greatly reduces transportation and material diversion hazards.

The relatively low waste heat from fusion reactors: Again, you are dependent upon the fuel cycle and the type of energy conversion used. You go all the way from pulse D-T systems, where the conversion efficiencies didn't look any better than conventional reactors, to the high temperature D-T steady state systems where you obtain efficiencies equivalent to the

# D-T FUSION POWER PLANT

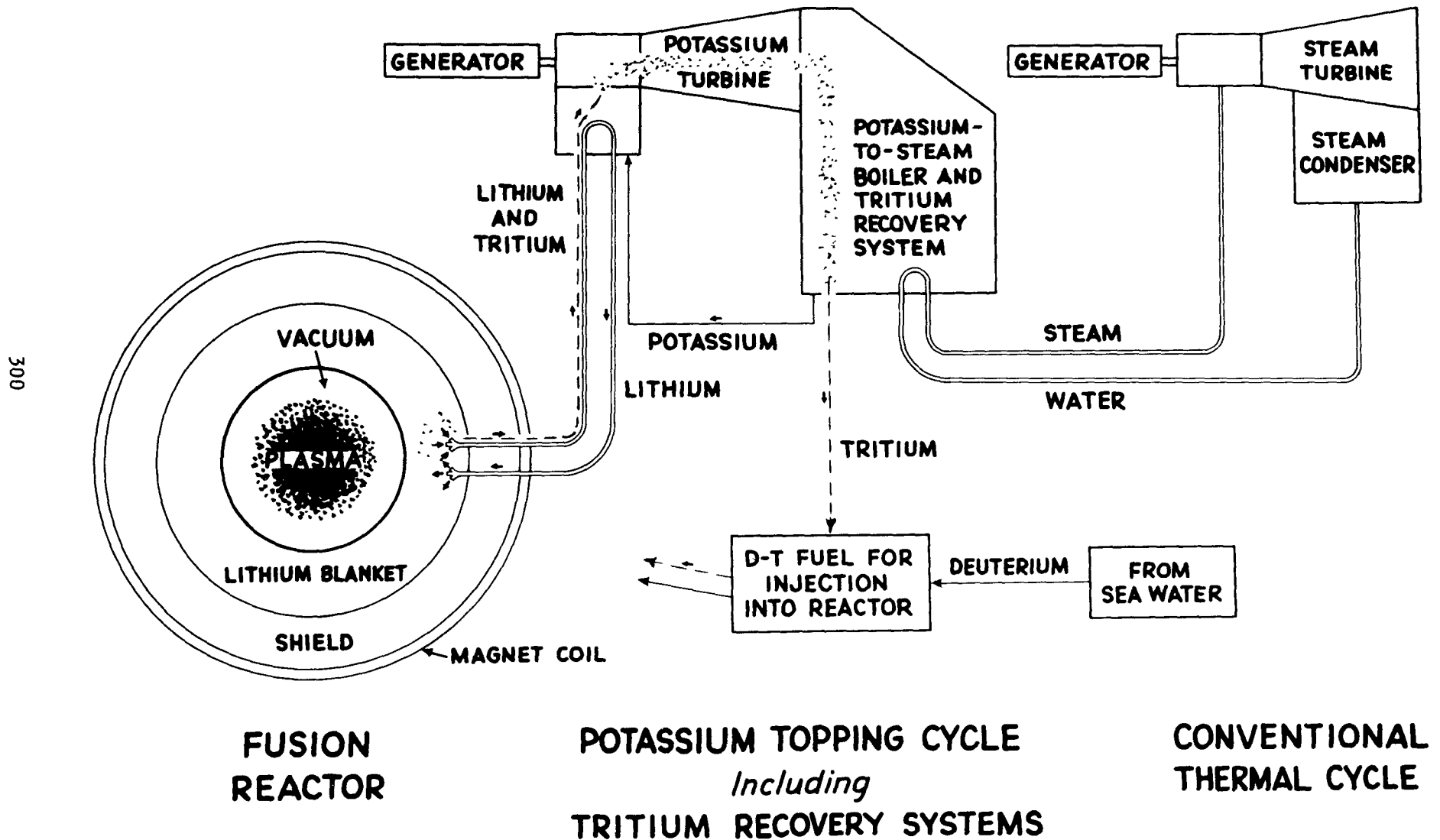
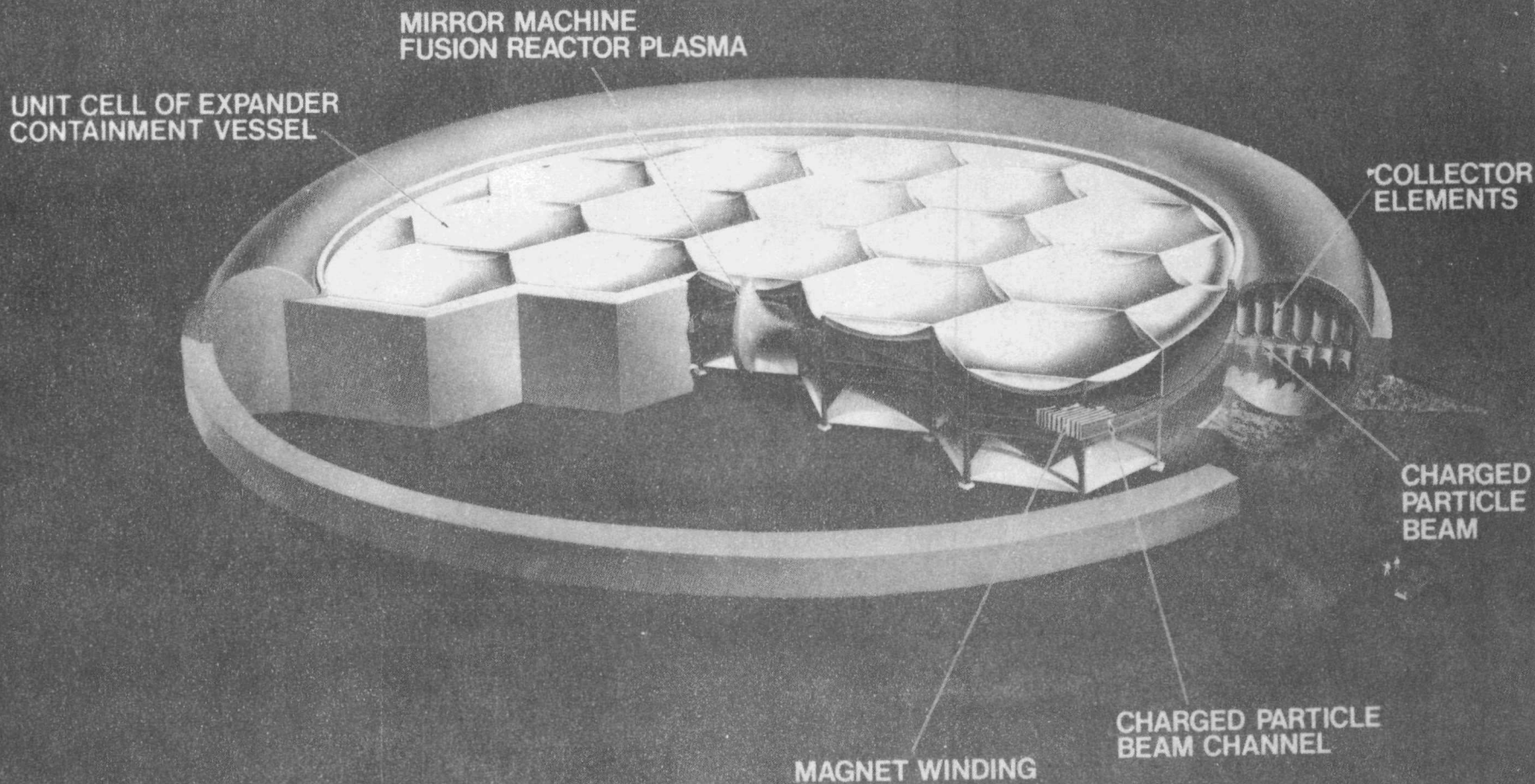


Figure 30

best hoped for in breeders or even higher. And on to the direct conversion cycles of D-He<sup>3</sup> and p-Li<sup>6</sup> where efficiencies of 80-90% are possible<sup>26</sup> that will be very hard to match. Figure 31 is an artists drawing of a 1000 megawatt D-He<sup>3</sup> fusion power plant using a direct energy converter as described in the talk by R. W. Werner of LRL. To see the relationship between high efficiency and thermal pollution go to Figure 32. The black represents the amount of energy that has to dissipate into the environment if no use can be found for the low grade heat. Present fission reactors run at efficiencies slightly better than 30%, conventional fossil fuel plants operate at about 40% as will first generation breeder reactors, Advanced breeder reactors, MHD converters and steady state D-T fusion reactors should be in the 50-60% range. Direct conversion fusion reactors should be in the 80-90% range. If achieved, this would represent a truly significant improvement in the reduction of waste heat, approaching an order of magnitude over present day systems. This means you could easily locate large plants of the future right in the center of cities. Lower efficiency plants could also be located in the center of a city if a use for the waste heat could be found, because thermal pollution is nothing more than energy in the wrong place. If a good use is found for the waste energy, such as the heating of buildings or the distillation of sewage, it becomes a valuable asset.<sup>29</sup>

In Figure 33, I show that the trend of population is to urban centers.<sup>30,31</sup> We have got to consider how to get energy there as

# FUSION REACTOR DIRECT ENERGY CONVERTER



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Figure 31

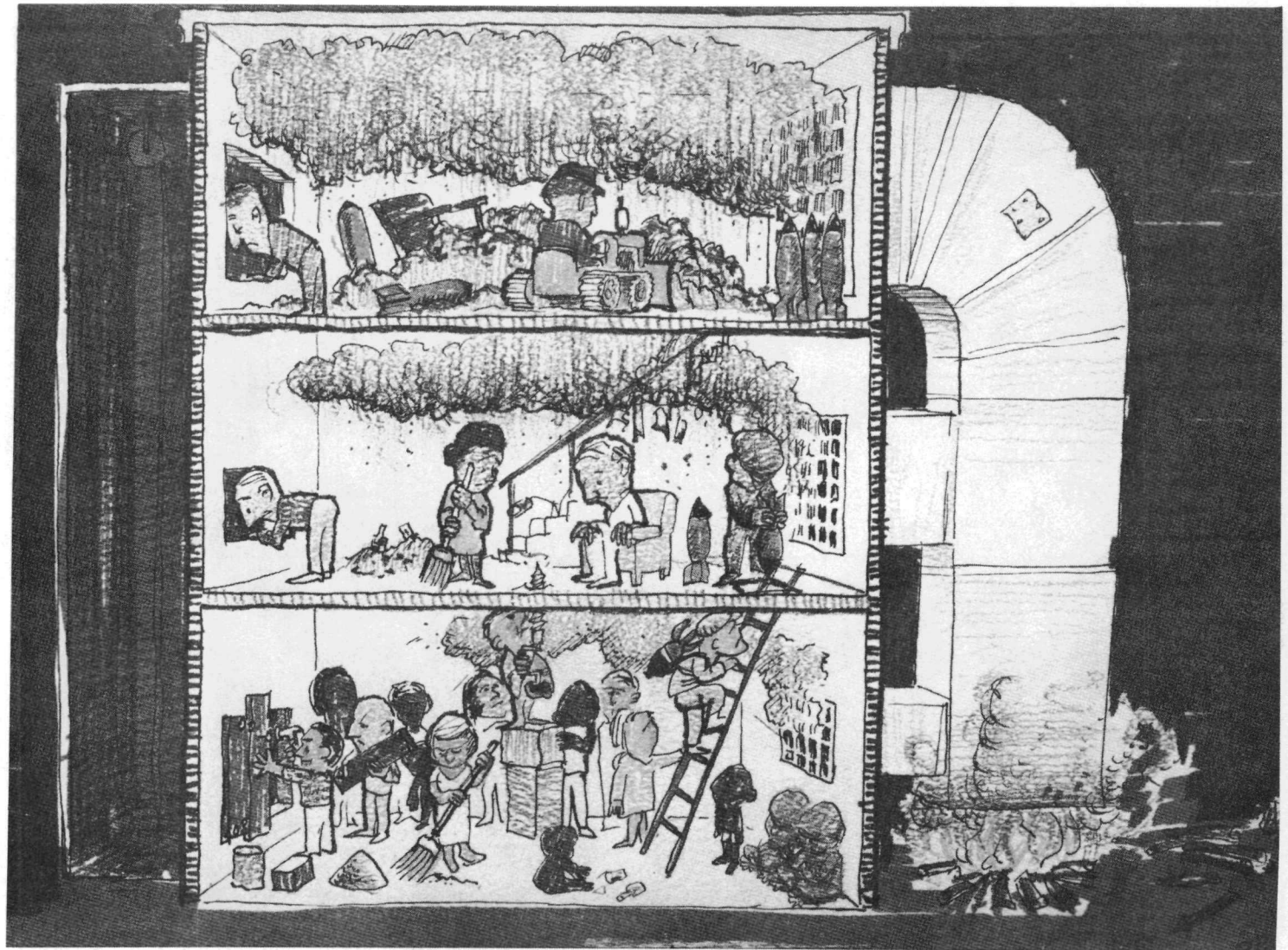
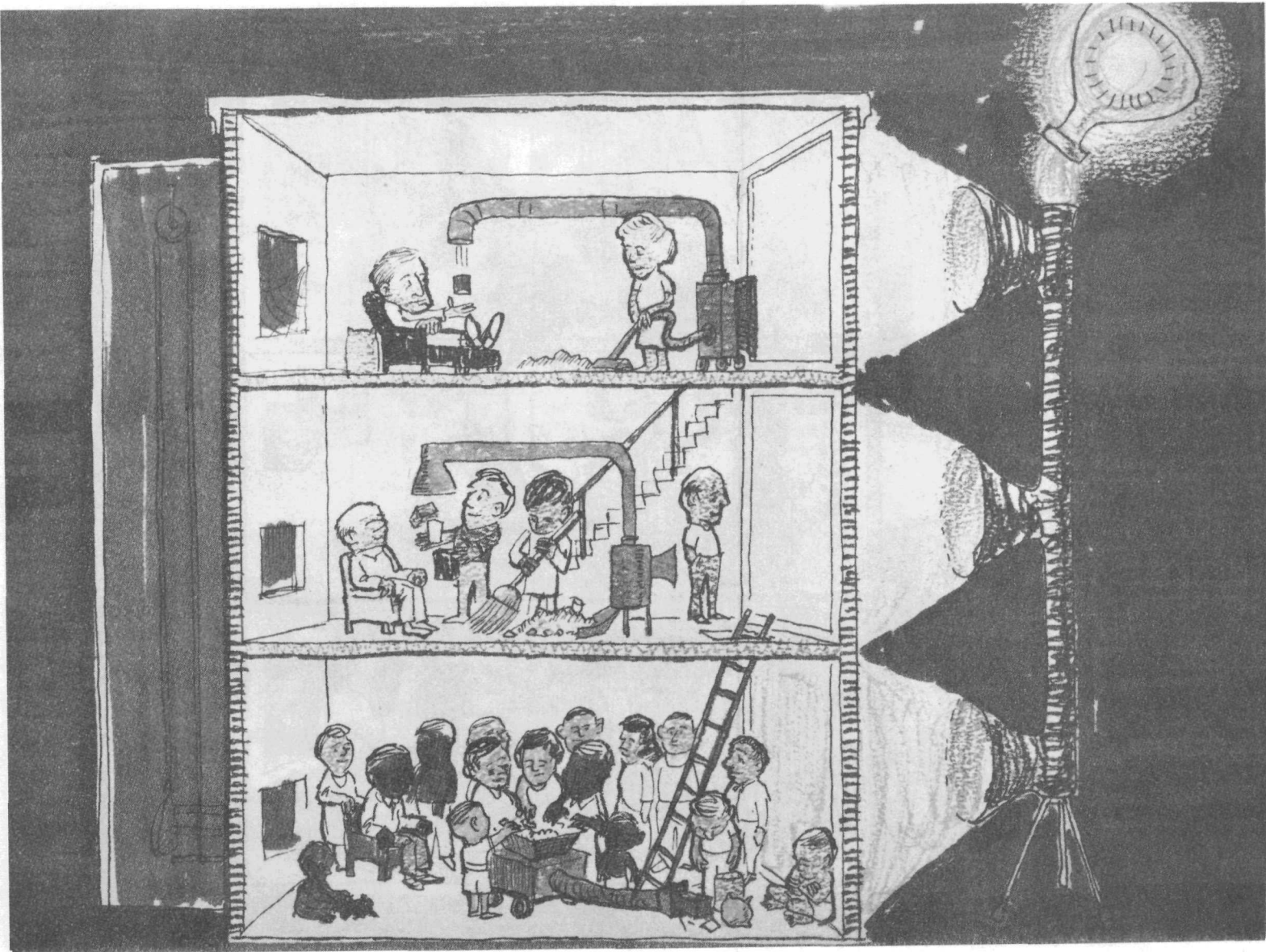


Figure 45





efficiently as possible while producing as little unwanted energy as possible. All the energy we use will eventually appear as heat. The combination of a large population and their high standard of living means that in future metropolitan areas large amounts of energy will be released over a comparatively small, densely populated area. Thus, these megalopolises will face consequences, such as increased temperatures and weather disturbances, from even the normal "desirable" uses of energy.<sup>32</sup>

#### FUSION TORCH

I want to talk now about a concept of Ben Eastlund's and mine called the "Fusion Torch".<sup>17,33,34</sup> Figure 34 is a log-log plot of plasma parameter space in terms of plasma density versus temperature. Almost all industrial processes using plasmas now fall in the upper left hand quadrant. What fusion research has done is to make available the entire right hand quadrant for exploitation. It's an unexplored area industrially. With fusion reactors, when they become available, we will have large volumes of very inexpensive plasmas in these regions. Some commercial application using existing plasma technology could actually begin now in this region.

Figure 35 is a schematic of the Fusion Torch. There are two concepts involved. The first is returning materials to their basic elements. Here the kinetic energy of the plasma is used to vaporize, dissociate, and ionize any solid. The plasma source could be a fusion

NUMBER DENSITY  
IN PARTICLES /  $\text{cm}^3$

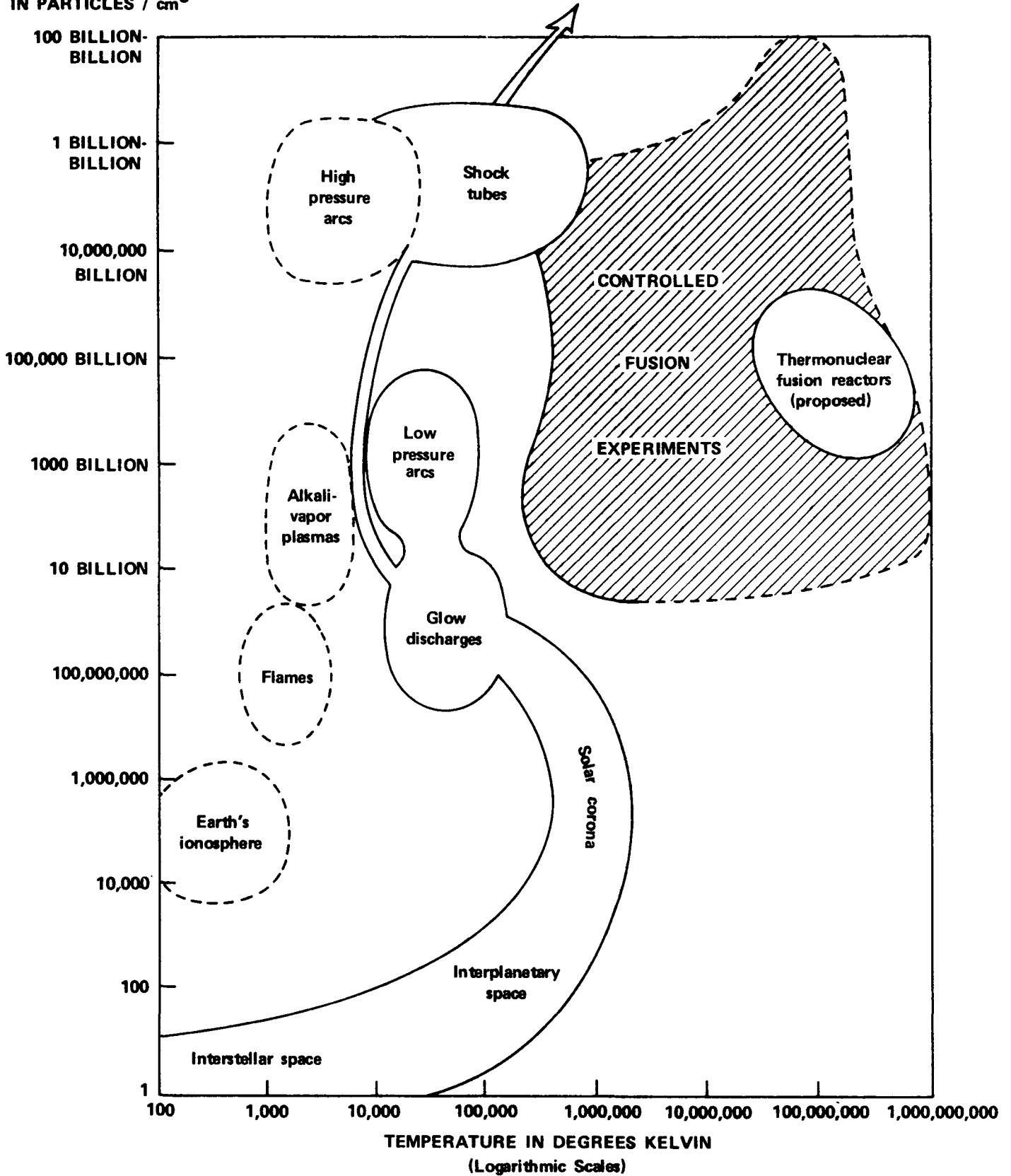


Figure 34

# THE FUSION TORCH

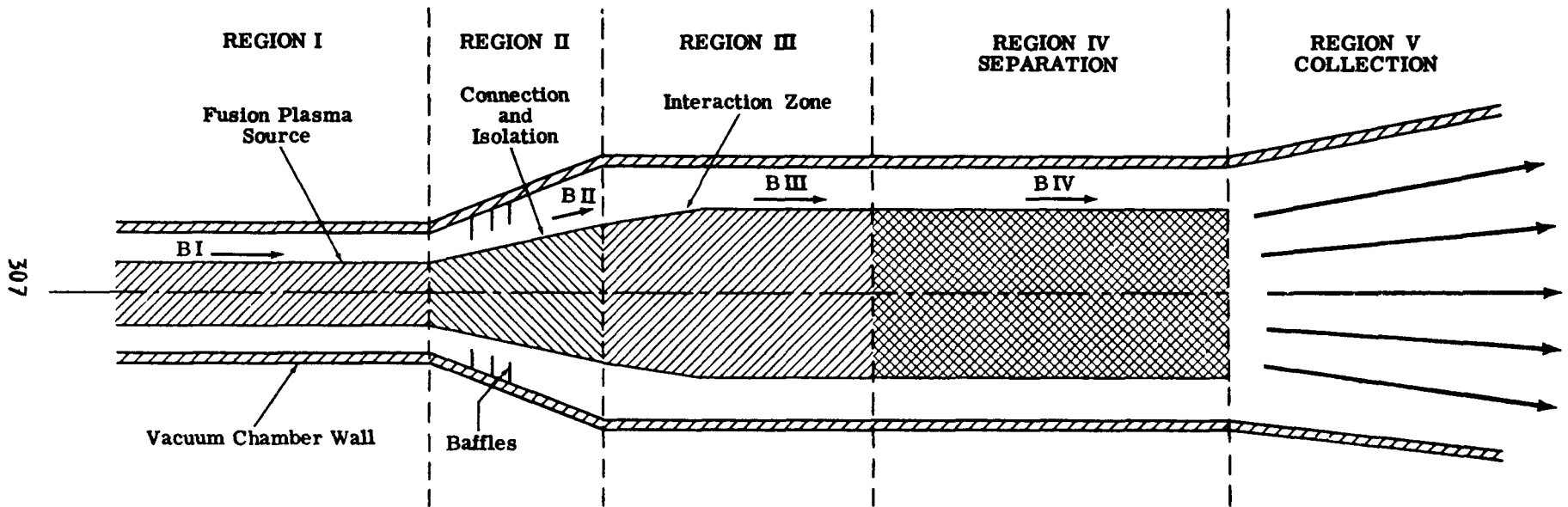


Figure 35

reactor or just a plasma generator. Next, is an isolation region. Then the inter-action zone where material is injected into the very hot plasma which is at millions of degrees. The region following represents the various separation schemes that could be used to collect the material. We are showing in the schematic that these are possibilities; much work would have to be done before one could guarantee what would occur in a particular process. In fact, much of the data needed for analysis does not yet exist. Nevertheless, the Fusion Torch concept does open up the possibility of closing the materials' cycle as shown in Figure 36. As discussed earlier, raw materials are limited and being depleted; we are going to have to use more energy to mine the lower quality ores; we are also going to have to use more energy to reduce pollution. The Fusion Torch is an energy-intensive process. It takes energy to ionize and dissociate material. But at the moment we see no way to completely close the materials' cycle other than to put matter into the plasma state. Figure 37 is a schematic of a fusion reactor with a Fusion Torch unit.<sup>35</sup> The D-T fuel cycle was used (as is normally done) although it creates numbers of problems that are much less serious in other fuel cycles. Although the Fusion Torch is an energy-intensive process, you are really cascading the energy created in the controlled fusion reactor downward via the Fusion Torch to very high grade heat at its walls which can then be used to produce steam. So, you have not really expended any energy since the heat can be recovered at a high efficiency and used to generate electricity. Although Figure 37 shows the amount

## CLOSING THE MATERIALS' CYCLE

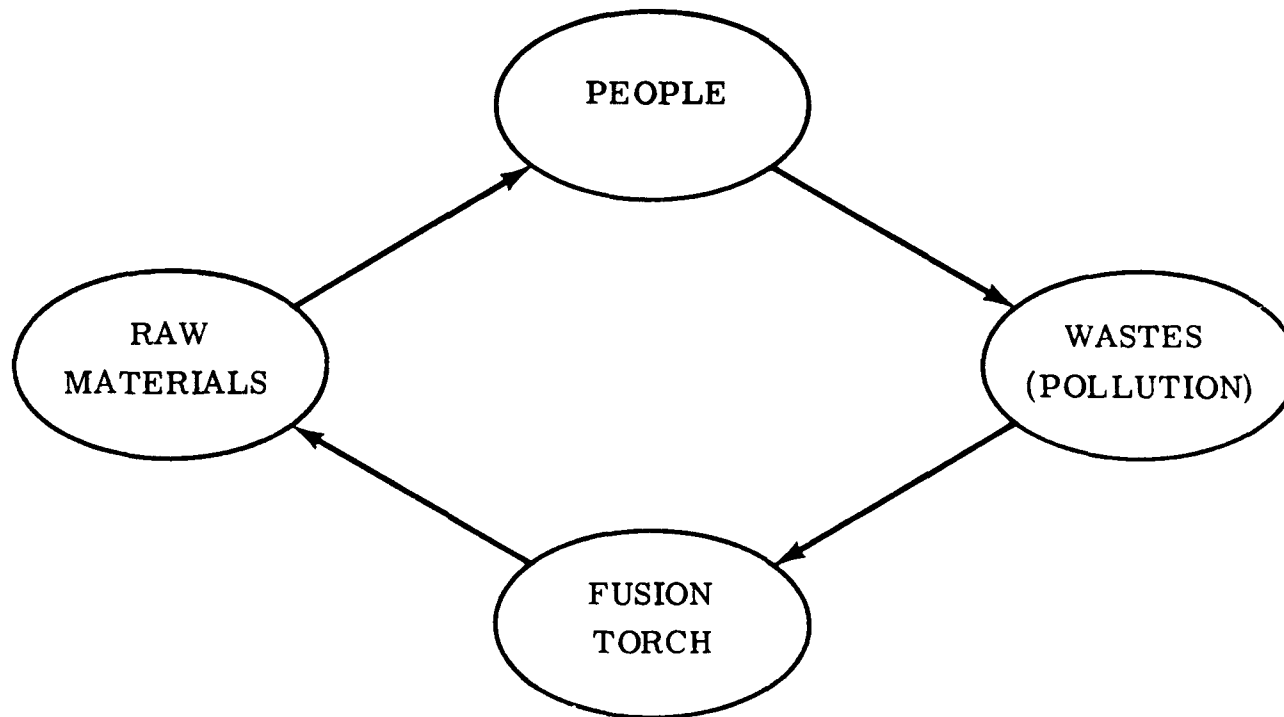
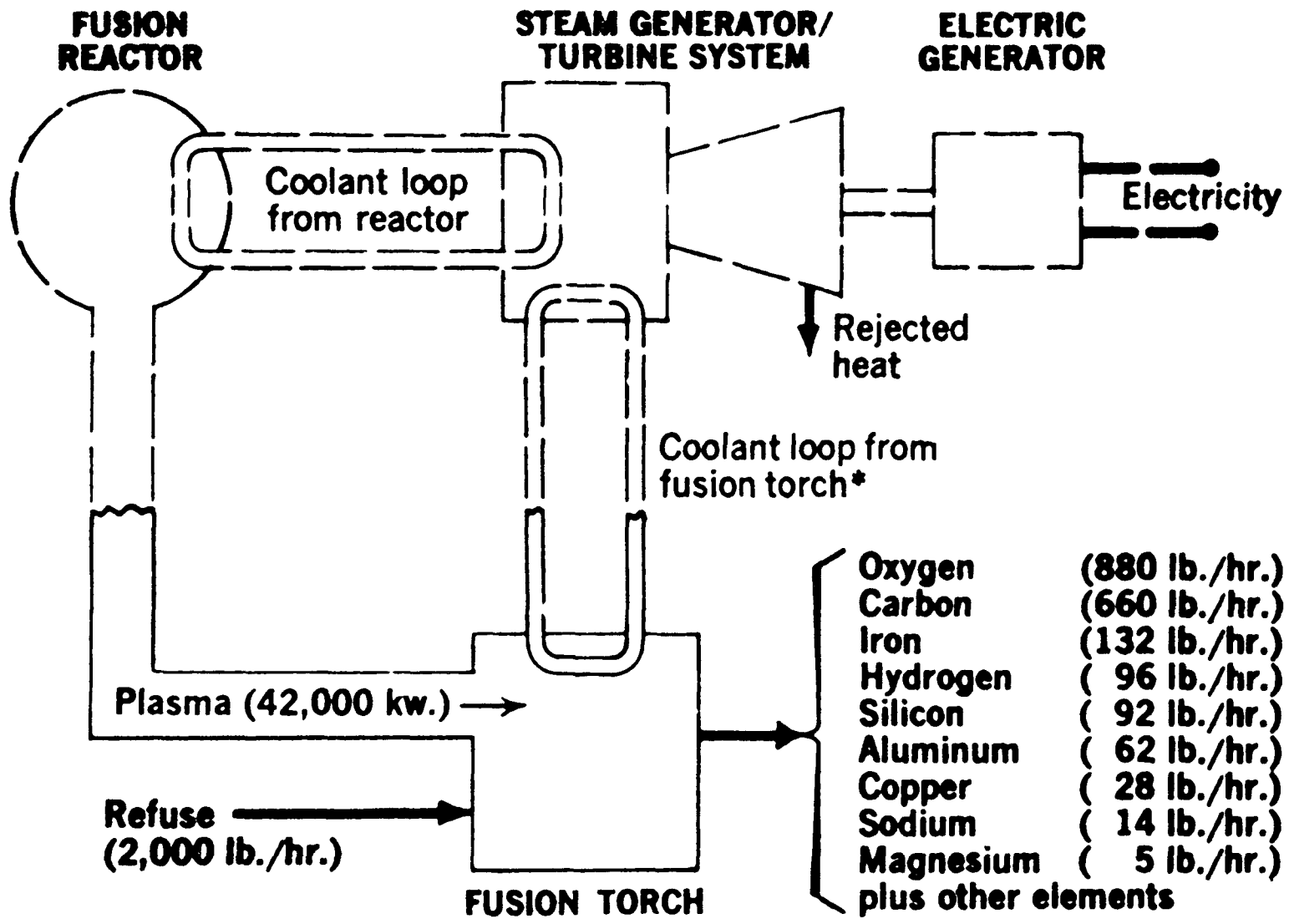


Figure 36



\*The thermal energy in the coolant stream from the fusion torch can be converted to electricity at about 50% efficiency.

**FUSION TORCH** can be part of a deuterium-tritium fusion power system.

Figure 37

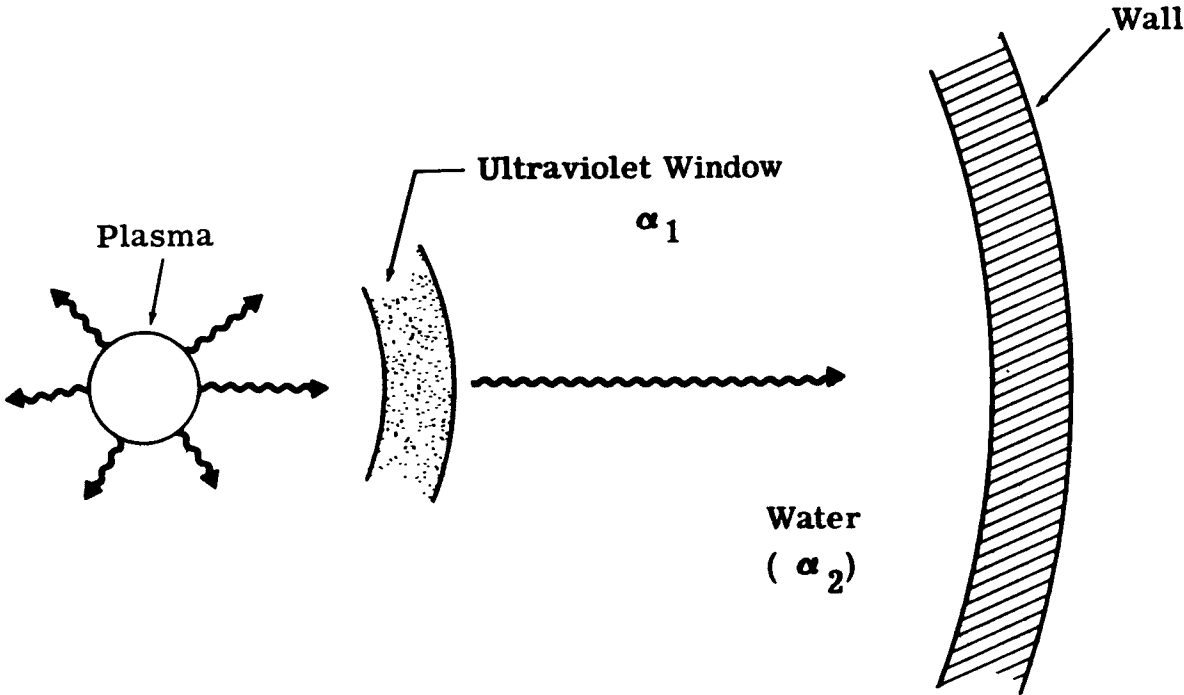
of material that would come out of such a system if typical refuse was processed, you probably would not operate like this but would pre-separate the refuse to optimize costs. In the near term applications you could use existing plasma generators for processing. For example, we are looking at the separation of aluminum from aluminum oxide using ultra-high temperature plasmas.<sup>36</sup> This is a good candidate since aluminum takes a great deal of energy to process and some low temperature plasma separation has been done.<sup>37</sup>

The second concept involved in the Fusion Torch is transforming the kinetic energy of the plasma into ultraviolet radiation by the injection of trace amounts of high Z impurities into the plasma. The energy is then transmitted through an ultraviolet quartz window as shown in Figure 38. By this technique it is possible to generate large quantities of ultraviolet energy. This opens up the possibilities for bulk heating and new chemical processing techniques. We know we can successfully generate ultraviolet energy because this was one of the limitations that held back the fusion program in the early days. The impurities getting into the plasma prevented us from reaching high temperatures because of the ultraviolet radiation emitted. The Fusion Torch concept is still in its inception with no specific programs being supported by the AEC. The concept is intended as a hunting license for those who think they can come up with ideas to use industrially the plasmas that have been developed in the fusion program and hopefully will be available in large volumes with future fusion reactors.



RADIATIVE BULK HEATING  
CHEMICAL PROCESSING

$$I = I_0 e^{-\alpha x}$$



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Figure 38

### FUSION POWER TIMESCALE:

There are three principal factors that determine the timescale to obtaining power producing fusion reactors. These are listed in Figure 39. First, is the combination of the physics, technology, and economics that establish what we call feasibility. Next, is the scientific and man power base upon which the research and development program can build. And third are the financial resources available to support the R&D program. We should recognize that it isn't the United States who is determining this time scale because we're less than a fifth of the world effort in fusion.<sup>30</sup> We do, however, make a major contribution.

Let's look at the first question on reactor feasibility as summarized in Figure 40. Fusion, unlike fission, does not require the proof of any new physical process such as the fission chain reaction which is commonly considered the proof of "scientific feasibility" for fission. There is no need for a chain reaction in a controlled fusion reactor. Our sun and the stars demonstrate that we can get net energy from fusion. However, to obtain useful fusion energy on earth, requires that we generate the hot plasmas and confine them for long enough. To accomplish this an extensive effort in fusion plasma physics has been underway. For confinement we speak of using electromagnetic fields rather than gravitational fields. The task is difficult, for to control the fusion process requires the solving of a set of equations that are quite intractable. A solution will require many approximations, plus experimental verifications. Such efforts have been carried out over the years through a series of experiments on one side with analytical work on the other. We have now added to these a third, numerical simulation technique brought about by the advent of very fast computers.<sup>39</sup> These three tools are being used together to bring us closer to an understanding of the fusion process. We have what

**FACTORS THAT DETERMINE  
THE TIMESCALE FOR FUSION POWER**

- 1. THE COMBINATION OF PHYSICS, TECHNOLOGY, & ECONOMICS  
THAT ESTABLISH "FEASIBILITY"**
- 2. THE SCIENTIFIC & MANPOWER BASE UPON WHICH THE R & D  
PROGRAM CAN BUILD**
- 3. THE FINANCIAL RESOURCES AVAILABLE TO SUPPORT THE  
R & D PROGRAM**

**Figure 39**

REACTOR "FEASIBILITY"

NUCLEAR FISSION

PROVE CHAIN REACTION POSSIBLE

SCIENTIFIC FEASIBILITY

NUCLEAR PHYSICS

REACTOR TEMPERATURES

SUITABLE MATERIALS

NEUTRONICS

STARTUP AND CONTROL

SYSTEM COMPONENTS (PUMPS, ETC.)

COMPATIBLE ENERGY CONVERTER

RADIOACTIVE WASTE HANDLING

TEST OF REACTOR CONDITIONS

PROTOTYPE POWER PLANT

CONTROLLED FUSION

FUSION PLASMA PHYSICS

REACTOR TEMPERATURES

SUITABLE MATERIALS

NEUTRONICS

STARTUP AND CONTROL

SYSTEM COMPONENTS (PUMPS, ETC.)

COMPATIBLE ENERGY CONVERTER

TEST OF REACTOR CONDITIONS

SCIENTIFIC FEASIBILITY

PROTOTYPE POWER PLANT

ECONOMIC POWER PLANT

Figure 40

amounts to an iterative process and progress of recent years shows we are rapidly converging to an understanding of the relevant physics.

To reach the plasma conditions that we need to demonstrate the "scientific feasibility" of a fusion reactor, we will have to establish beyond reasonable doubt the solution to technological problems to be faced in commercial power plants. To demonstrate the "economic feasibility" of a fusion reactor, we will have to show that the costs are low. In certain areas, we must accept the existing state of knowledge because additional development would be too expensive. To reach reactor temperatures the fusion program has developed vacuum technology, injection techniques, and plasma heating technology. For suitable magnet materials for a fusion reactor we need superconducting magnets. We are relying upon people outside of the fusion program for these developments. If you look at the cost of fusion core you find that most of the cost is in the superconductor. There are new superconductors available that could be developed that might be cheaper and give higher fields, but those developments are not being pursued. When you look at the neutronics, we rely very heavily on all of the work that has been done in the fission field. For start-up and control we will have to do additional work. System components, pumps, and so forth, have been developed and will be developed to the extent necessary to do the plasma feasibility tests. From the talk on direct conversion, it is clear that the energy converters themselves can determine what is a feasible fusion reactor. So we are now at a point in fusion plasma research where developments in technologies can bring us closer to feasibility. Fusion research is a combination of technology and physics, and when you reach "scientific feasibility" you are at a different point than you were

when you achieved the first chain reaction which had none of the technology conditions necessary for a power plant or a test of reactor conditions for commercial use.

Figure 41 illustrates the strong scientific base which now supports R & D programs in the U.S.<sup>40,41,42</sup> The figure is a cumulative plot of the dollars the U.S. Government has spent on research and development (operating, equipment, and construction) over the last 70 years. Note that from 1900 to 1942, the U.S. spent very little on R & D. This was the scientific base available when it was first proven that net energy could be produced from nuclear fission. It took over twenty years before nuclear fission became economically competitive. If you look at the year 1960, about the time we began the space program, you can see we were building upon a much stronger scientific base, and in 10 years we had a man on the moon. In 1970, as controlled fusion faces its technological problems, we have a base of almost 200 billion dollars of government R & D. The cumulative amount spent by the U.S. controlled fusion program from its inception is about 1/3 billion dollars as indicated by the line in the bottom right hand corner of Figure 41. Over the next 30 years, the U.S. will probably spend somewhere between 500 billion and 700 billion dollars on research and development, using the scientific manpower that we have available and are training in this country. What we do with that research and development will determine what will be the benefits to our society at the end of that time. We are making our commitments now on how that money is to be used and what programs

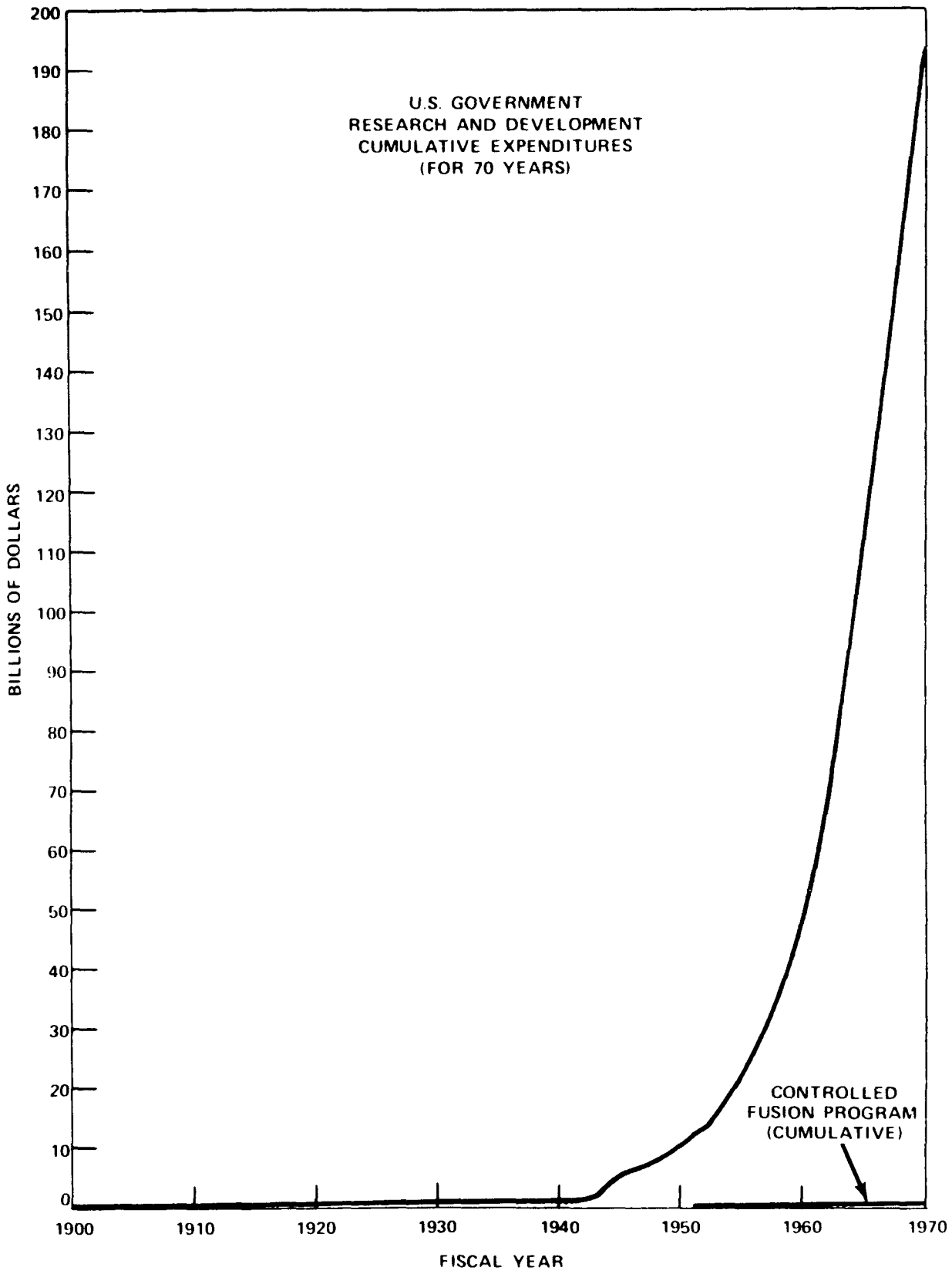


Figure 41  
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it is to support. We have started on supersonic aircraft; we have started on antiballistic missiles.

Figure 42 gives the annual expenditures for the fusion program for operating and equipment funds. The dashed curve uses the consumer price indexes to adjust these dollars for inflation.<sup>43</sup> For the last seven years, the real purchasing power for the fusion program has remained essentially constant. As we pass into a stage where more technology must be done in order to prove feasibility we are forced into a tradeoff. To hire additional engineers to do the technology we have to reduce the number of physicists working on the program. Figure 43 gives some time scales to a controlled fusion power plant. The assumption we normally make is number two. This assumes an expansion of the world-wide fusion program as the reactor regime is reached in fusion experiments. At that rate we would be almost 25 years away. Now if we had to make the assumption that our effort and all the rest of the world's remains constant you could come up with numbers in the order of 50 years. Or we could make the assumption that we, or some other nation, expands and really tries to obtain fusion power from fusion as rapidly as possible. People have given me estimates that vary between 7 and 12 years.

CONCLUSION:

Lets take a look at our past, the present, and into the future. Figure 44 is called the great transition.<sup>44</sup> Our time scale is in terms of thousands of years. And we're on that steep slope. We're in a period of very rapid growth and change - probably the most abnormal



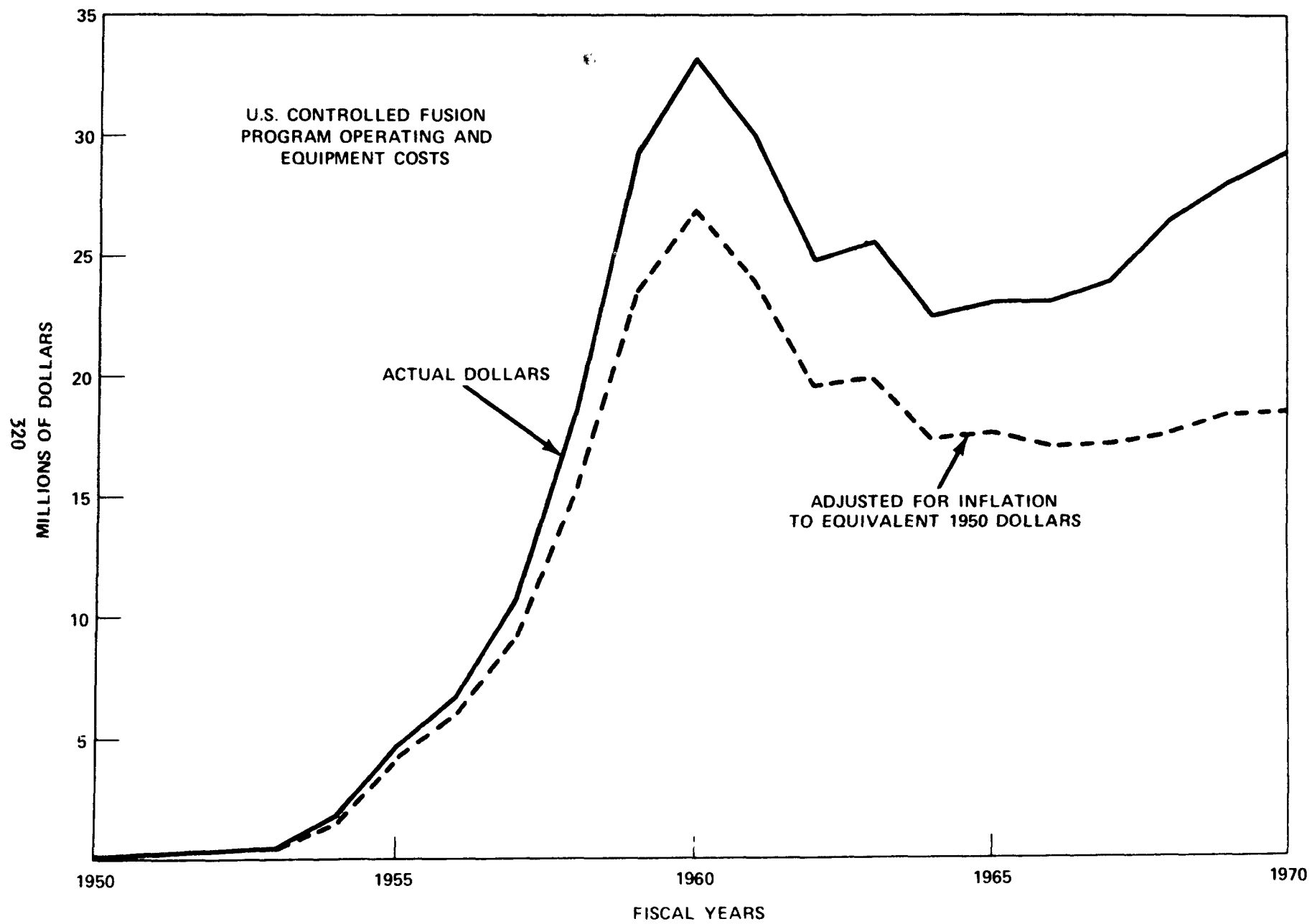


Figure 42

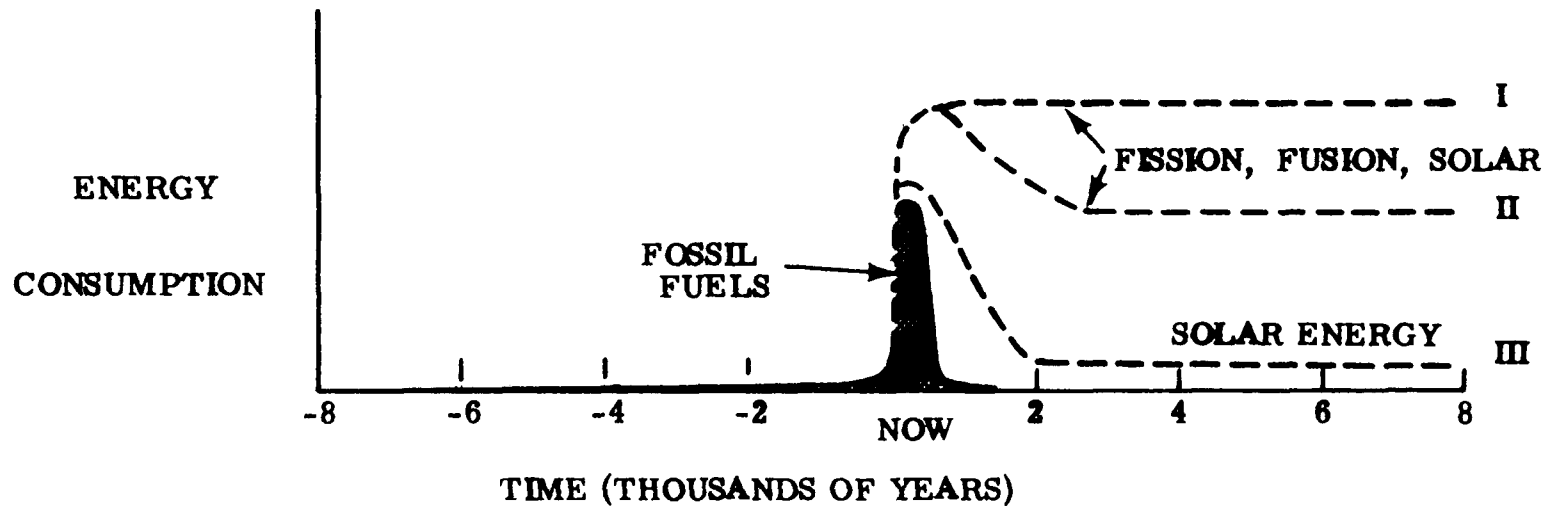
## TIMESCALE TO A CONTROLLED FUSION POWER PLANT

	<u>YEARS</u>
ASSUMPTION 1.	
FUNDING AT PRESENT RATE	~ 50
ASSUMPTION 2.	
FUNDING AT PLANNED ACCELERATED RATE	~ 25
ASSUMPTION 3.	
FUNDING AS "NATIONAL OBJECTIVE"	~ 10

Figure 43

in human history and probably the greatest ecological and biological upheaval that mankind or this earth has ever seen. And we're in it, now. We have certain traps that we can fall into.<sup>45</sup> Population is one. The rapid rises seen in Figure 5 could bring about food shortages which could bring political upheaval, dictatorship and loss of freedom to people. Energy is one way of postponing some of those problems. Another trap is limited resources. The only way to get around these limitations is to close the material cycle. This requires energy. It must be a clean energy source, one that doesn't pollute the environment. The third trap is war. A recent U.N. journal, stated that there is the equivalent of about 20 tons of TNT for every man, woman, and child on earth.<sup>46</sup> The security of this country, the Soviet Union, and possibly the world, is based upon a nuclear deterrent system.<sup>47</sup> And that deterrent system could go unstable - as every nation obtains nuclear weapons which is something that one must assume will occur over the next thirty years. During this period of rapid change there will be stress on the political systems of the countries of the world. The question is whether the political systems can adjust this rapidly. Also, you will have stresses on the social system which therefore, must be very flexible. There are questions on economics. We're going to have to begin to adjust to a non-growth economy and with limited resources. We'll have to take into account the quality of life in the economic analysis. Figure 44 lists three assumptions. First, successful control is assumed. Second, an overshoot in population is assumed and we go into temporary chaos and the population of the world levels off at

## THE GREAT TRANSITION



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### THREE TRAPS:

1. POPULATION (FOOD)
2. ENTROPY (RESOURCES, ENERGY, POLLUTION)
3. WAR (HUMAN NEEDS AND BEHAVIOR)

### THREE ASSUMPTIONS

- I. SUCCESSFUL CONTROL - AVOIDANCE OF TRAPS
- II. SERIOUS OVERSHOOT OF POPULATION - TEMPORARY STATE OF CHAOS
- III. SERIOUS FOOD, RESOURCES, AND ENERGY SHORTAGES - CONFUSION AND CHAOS - NUCLEAR WAR

Figure 44

a more desirable level. Third, things do not go well and we end up with a serious catastrophe. We are back where we started from with a small world population dependent on the sun.

Returning to Figure 1, we can recall where we are today. Three floors of people all in one building, with our little bombs, our limited supplies being sent up to those on top, pollution building up, and our fuel being burned up rapidly. In Figure 45, we look in at our apartment house as it would appear in the year 2000 if we continue in the same way as we have in the past.<sup>1</sup> By the year 2000, our energy sources will be running down, the ratio of population will be 2:4:16.<sup>2</sup> Living conditions will be quite crowded down on that bottom floor. There will be tremendous uses of energy. There will be tremendous numbers of weapons. We'll be handling pollution with mechanized methods. We'll be wondering why these fellows on the bottom are not sending up supplies anymore. There will be some unsatisfied people trying to break out of that bottom floor. That's the way things might be.

In Figure 46 we have a happy picture. We must make some assumptions of course. We'll make the assumption that you have not only an unlimited source of energy but also a very inexpensive source of energy. The impact of that is almost unimaginable. Let us assume that you can recycle material. Of course on the top floor they use an automated machine, while on the bottom floor they do it by hand. But you've got the idea. Everybody is happily looking out and basking in the sunshine of abundant energy. We've taken out the bombs -- maybe that would be a good idea. The population growth hasn't been changed although that could be done, too.

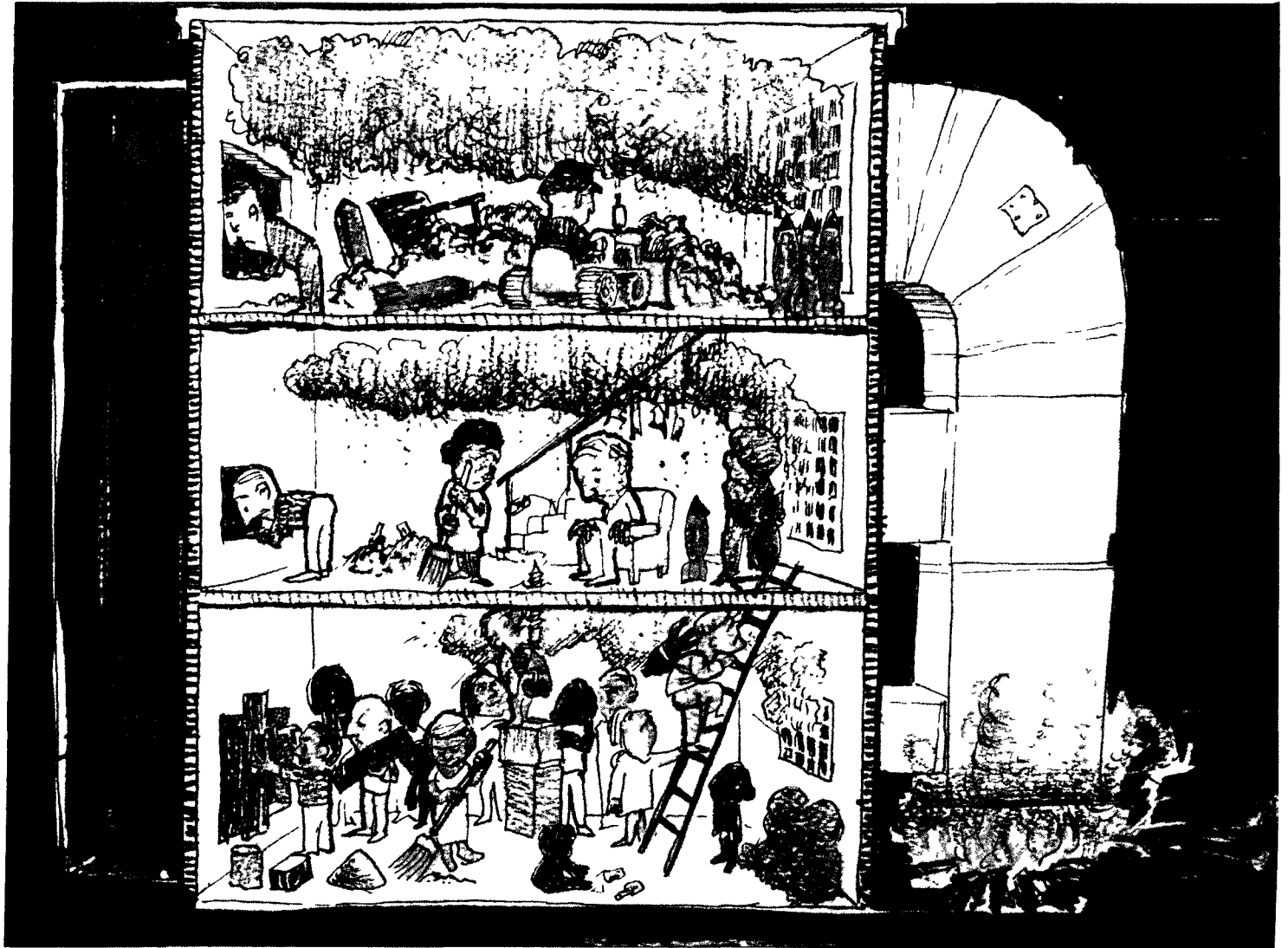
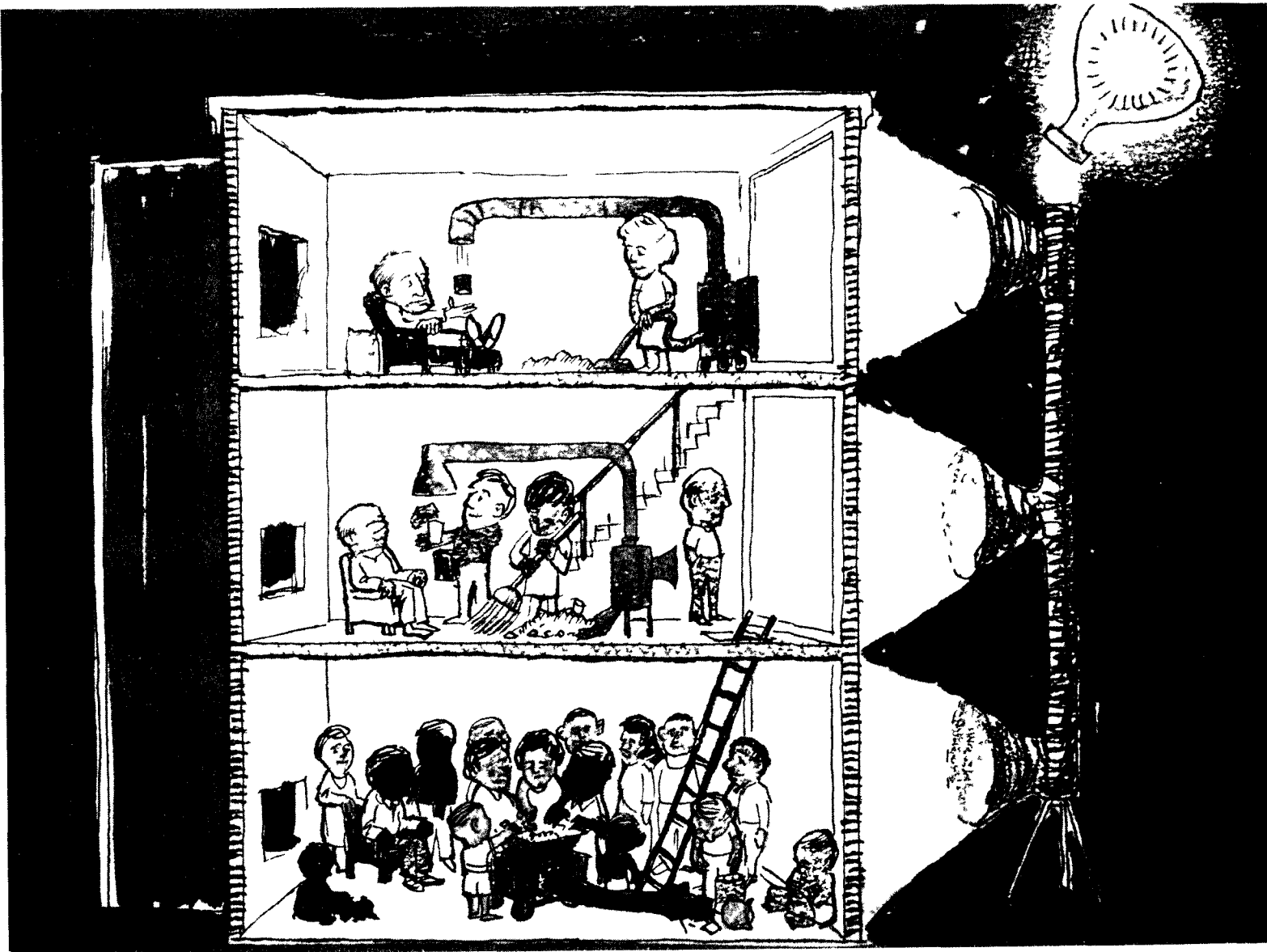


Figure 45



Will this happy picture become a reality? We are in a race with time. At present about one tenth of one percent of our entire federal budget is going towards solving the problems of infinite energy sources, (solar energy, breeders, and fusion), materials' recycling, disarmament, and population control. I've been talking about the technology for solving many major problems that are facing mankind. But one does not cure all these ills of society by technology alone. We're all well aware of that. Human behavior must change and that requires a vast effort in education and the development of a value system that can permit the survival of mankind. Thank you.



1. Figures 1, 45, and 46 are the work of freelance artist Robert Bordeaux, 26817 Dix Court, Damascus, Maryland.
2. Herman Kahn and Anthony Wiener, The Year 2000 (The MacMillan Company, New York, 1967).
3. United Nations 1968 Statistical Yearbook, Statistical Office of the U.N., Department of Economic and Social Affairs, New York 1969.
4. Ibid.
5. Ralph J. Black, Anton J. Muhich, Albert J. Klee, H. Lanier Hickman, Jr., and Richard D. Vaughn, "The National Solid Wastes Survey -- An Interim Report", Presented at the 1968 Annual Meeting of the Institute for Solid Wastes of the American Public Works Association, Miami Beach, Florida, October 24, 1968 (HEW, Environmental Control Administration).
6. From unpublished study (January 6, 1970) by A. S. Bishop, Director of Environment for Economic Commission of Europe.
7. Figures are adaptations from "Resources and Man", A Study and Recommendations by the Committee on Resources and Man, National Academy of Sciences - National Research Council, (W. H. Freeman and Company, San Francisco, 1969).
8. Bishop, op. cit., ref. 6.
9. Mineral Resources. Report to the committee on Natural Resources, publication 1000-C, (Washington: National Academy of Sciences-National Research Council, 1962).
10. Charles F. Park, Jr., Affluence in Jeopardy-Minerals and the Political Economy, (Freenan, Cooper, and Co., San Francisco, 1968).
11. "Automobile Disposal, A National Problem"(Washington: U.S. Department of Interior, Bureau of Mines, 1967), (Case studies of factors that influence the accumulation of automobile scrap).
12. "Commodity Statements", U.S. Department of the Interior, Bureau of Mines, January 1969, unpublished data.
13. Robert C. Weast, Editor, Handbook of Chemistry and Physics, The Chemical Rubber Company, 50th Edition 1969-1970.
14. Resources & Man, op. cit., ref. 7.

15. "New Utility Concepts for New Cities", Annual Progress Report-Civil Defense Research Project, ORNL-4284, Part I, November, 1968, pp. 109-123.
16. Black, op. cit., ref. 5.
17. Bernard J. Eastlund and William C. Gough, "The Fusion Torch: A New Approach to Pollution and Energy Usage", Presented at the American Institute of Chemical Engineers 62nd Annual Meeting, Washington, D. C., November 16-20, 1969.
18. Howard T. Odum, "Energetics of World Food Production", from The World Food Problem, Vol. II, Report of the Panel on the World Food Supply, President's Science Advisory Committee, (The White House, Washington, D. C., May 1967), pp. 55-94.
19. Photograph courtesy of Water Power, an international journal for hydroelectric development, "The Nagarjunasagan Project", Water Power, January 1970.
20. National Power Survey, Part 1, Report by the Federal Power Commission (Washington: U.S. Government Printing Office, October 1964).
21. AEC's Division of Raw Materials, private communication, March 21, 1969.
22. Energy Resources, Report to the Committee on Natural Resources, publication 1000-D, (Washington: National Academy of Sciences-National Research Council, 1962).
23. Peter E. Glaser, "Power from the Sun", Science, Vol. 162, No. 3856, (November 22, 1968), pp. 857-861.
- 24a. Amasa S. Bishop, "The Status and Outlook of the World Program in Controlled Fusion Research", presentation before the National Research Council--National Academy of Sciences, March 11, 1969.
- 24b. D. J. Rose, "Engineering Feasibility of Controlled Fusion - A Review", Nuclear Fusion, 9, pp. 183-203 (1969).
25. Ohkawa, M. Yoshikawa, Kribel, Schupp, and Jensen, "Plasma Confinement in D.C. Octopole", Phys. Review Letters, Vol 24, p. 95, January 19, 1970.

26. R.F. Post (Chairman), " Preliminary Report of Direct Recovery Study", Lawrence Radiation Laboratory, report UCID-15650, May 15, 1970.
27. A. P. Fraas and H. Postma, "Preliminary Appraisal of the Hazards Problems of a D-T Fusion Reactor Power Plant", ORNL-TM-2822, April 29, 1970.
28. T. B. Taylor, "The Rapid Growth of Nuclear Technology -- Implications for Nuclear Safeguards,"IR&T Journal (International Research and Technology Corporation), Vol. 1, No. 3, (January 1968).
29. A. P. Fraas, "Conceptual Design of a Fusion Power Plant to Meet the Total Energy Requirements of an Urban Complex", Proceedings of the Conference on Nuclear Fusion Reactors, Culham, England, September 17-19, 1969.
30. Jerome P. Pickard, Dimensions of Metropolitanism, research monograph 14, (Washington, D.C. Urban Land Institute, 1967).
31. Jerome P. Pickard, Appendixes to Dimensions of Metropolitanism, research monograph 14 A, (Washington, D.C. Urban Land Institute, 1967).
32. William A. Porter and Marion O. Hagler, "Some Energy Considerations of the Use of Fusion Energy and the Fusion Torch", unpublished paper, Texas Tech. University, May 1970.
33. Bernard J. Eastlund and William C. Gough, "The Fusion Torch -- Closing the Cycle from Use to Reuse", U.S.A.E.C. Report WASH-1132, May 15, 1969.
34. Bernard J. Eastlund and William C. Gough, "The Fusion Torch -- Unique Applications of Ultra-High Temperature Plasmas", Proceedings of the Conference on Nuclear Fusion Reactors, Culham, England, September 17-19, 1969.
35. "Nuclear Fusion Dividend: Solid Wastes Treatment?", Chemical Engineering, pp. 56-57, December 15, 1969.
36. Bernard J. Eastlund, "Pollution Control is Elementary -- with Fusion Plasmas", to be published in the proceedings of the Conference on Industry and Tomorrow's Society held Montreal, Canada, June 9, 1970.
37. Roger K. Rains and Robert H. Kaalec, "The Reduction of  $Al_2O_3$  to Aluminum in a Plasma", Metallurgical Transactions, Vol. 1, June 1970.

38. Amasa S. Bishop, "World Survey of Major Facilities in the Field of Controlled Fusion" U.S.A.E.C. report, Second Edition, November 1969.
39. William C. Gough, "Survey of Computer Usage in the A.E.C.'s Controlled Thermonuclear Research Program", internal AEC report, February 1969.
40. A. Hunter Dupree, Science in the Federal Government - A History of Policies and Activities to 1940 , Harper and Row, New York and Evanston, 1957, 1964.
41. Basic Research and National Goals, a report to the Committee on Science and Astronautics, U. S. House of Representatives by the National Academy of Sciences, March 1965.
42. Special Analyses, Budget of the United States, Fiscal Year 1970, U.S. Government Printing Office, Washington, 1969.
43. 1969 Business Statistics, 17th Biennial Editon, A Supplement to the Survey of Current Business, Office of Business Economics, Department of Commerce.
44. Energy Resources, op. cit., ref. 22; adaptation of drawing by M. King Hubbert.
45. Kenneth E. Boulding, The Meaning of the 20th Century - The Great Transition, (Harper and Row, N.Y. 1964).
46. Robert Humphries, "The Imperiled Environment", Vista, Vol. 5, No. 5, (March-April, 1970).
47. William C. Gough, "Presidential Communications During Time of Crisis -- The Story of the Moscow - Washington 'Hot Line'", Kennedy School of Government, Harvard University, January 1967.

ON THE CONCEPT OF PULSED THERMONUCLEAR  
MHD ENERGY CONVERSION

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Old Dominion University

Abstract

It is likely that micro-thermonuclear detonations, as discussed by Kidder<sup>1</sup> and Winterberg<sup>2</sup>, will be fully realizable in the very near future. How can the nuclear energy which is released be converted to useful work? An interesting answer is to have the intense plasma burst pass through a MHD channel, thus converting some part of the plasma enthalpy directly to electrical work.

What is the nature of this fusion plasma, how is it formed, and is it practically feasible to harness the energy? These last questions are the subject of the present report. A physical description is given based on some analysis and a brief literature survey.

<sup>1</sup> Kidder, Ray E. The application of lasers to the production of high-temperature and high-pressure plasma. Nuclear Fusion 8, No. 1, 3-12 (1968).

<sup>2</sup> Winterberg, F. The possibility of producing a dense thermonuclear plasma by an intense field emission discharge. Phys. Rev. 174, No. 1. 212-20 (5 Oct. 1968).

# INTENSE RELATIVISTIC ELECTRON BEAMS AND CONTROLLED FUSION RESEARCH

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## Abstract

The different methods for igniting a small thermonuclear explosion by irradiating a target with a beam of energetic particles are discussed with regard to their relative merits and promise of success. On the basis of this analysis, the scheme in which the target heating is effected by an intense relativistic electron beam seems particularly promising as a method for achieving the desired goal. Beams of the required intensity can be produced with high efficiency by the high voltage Marx-circuit Blumlein technique. A second method using a charged, levitated, highly magnetized superconducting ring may produce electron beams of substantially higher voltage and total energy output than in the Marx-circuit technique. The target heating is brought about by collective plasma instabilities. In case the collision-free beam dissipation should pose unforeseen difficulties, it is shown that one may alternatively irradiate the thermonuclear target by an intense beam of ions. In this case the stopping power range ensures complete collisional beam energy dissipation. A further distinct advantage of the methods described is the reduction of the critical ignition size, by which it may even be possible to extract energy from a D-D thermonuclear reaction. The energy produced by a chain of thermonuclear micro-explosions occurring inside a spherical container can be converted into useful electrical energy.

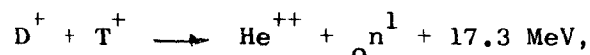
A modification of this scheme can be used for rocket propulsion by having the explosions take place at the center of a spherical reflector open on one side.

Wall Erosion by Sputtering in Controlled  
Thermonuclear Fusion Devices

James M. Daniel and C. R. Finfgeld  
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In any controlled thermonuclear fusion reactor, ions escaping from the plasma will strike the walls of the vacuum enclosure. Sputtering is among the consequent deleterious processes, resulting in contamination of the plasma and erosion of the wall. This paper reviews existing data relevant to the problem, and estimates the wall lifetime as a function of burnup fraction using our best conjectures for yields. Those yields which will be the primary determinants of erosion rates are also identified, and recommendations as to which quantities ought to be determined more accurately are made. It is hoped that these estimates will be useful for design engineering until more accurate measurements become available. We consider the reaction



take niobium as the wall material, and use the other assumptions of TM-2204<sup>1</sup> and TM-2692.

Presently available yield data for hydrogen ions is summarized in Fig. 1. Yonts<sup>2</sup> reported a yield of 0.0042 atoms/ion for  $D^+$  at 20 keV on niobium at 1100 °C, which is somewhat less than the yield of approximately 0.01 atoms/ion which would be estimated on the basis of other light ion yield data. Agreement within a factor 2 is encouraging, and an average value of 0.0071 atoms is taken for these purposes.

There is no experimental data for sputtering by tritons. However, KenKnight and Wehner<sup>3</sup> found that, for molecular hydrogen ions,



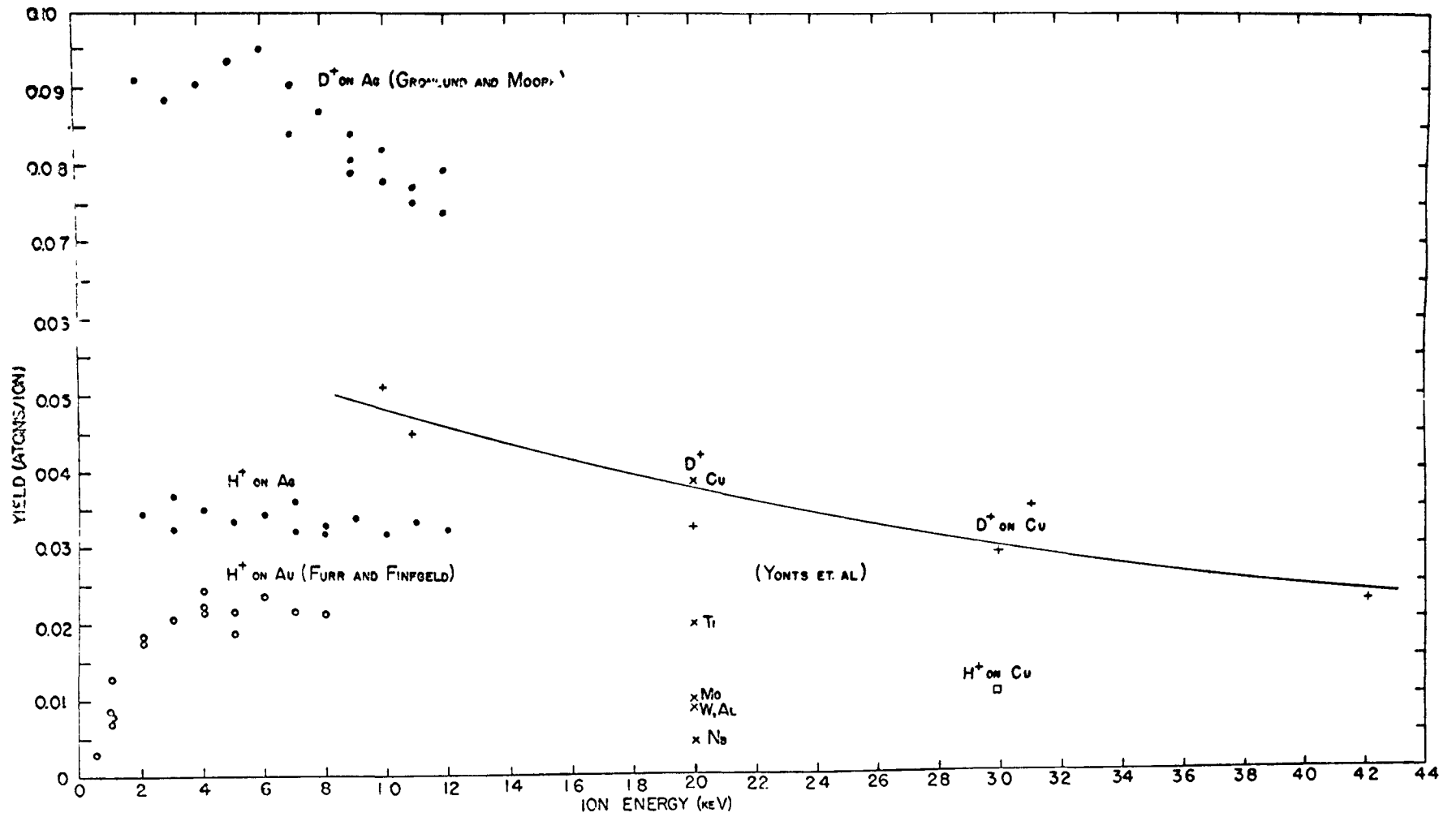


Fig. 1

the ratios of yields for  $H_2^+$  and  $H_3^+$  were close to 1.5. Applying this factor to the  $D^+ - T^+$  system, we estimate a sputtering yield of 0.011 atoms/ion for  $T^+$  at 20 keV. It is assumed that the peak of the ion energy distribution will be 20 keV, and variation of sputtering yield over the energy distribution is ignored.

At low burnup fractions, sputtering due to  $D^+$  and  $T^+$  will almost certainly be the predominant erosive mechanism. As burnup fractions are improved, however, sputtering by  $He^{++}$  will become increasingly significant. Unfortunately, there is no yield data for  $He^{++}$ , and data for multiply charged ions in general is exceedingly scarce. For this reason, we begin with Yonts' value of 0.051 atoms/ion for  $He^+$  at 20 keV. We would expect second ionization to reduce yields by reducing the hard-sphere collision diameter, and to increase them through strengthened Coulomb forces. However, lacking information on which to base estimates of these competing effects, we ignore the consequences of second ionization. Making a crude energy correction from 20 keV to 3.5 MeV, we use Kaminsky's results<sup>4</sup> for  $D^+$  on silver. He found that yields decreased by about a factor 5 between 100 keV and 1 MeV, and in rough accord with this and the energy dependence for  $D^+$  on copper of Fig. 1, we arbitrarily take  $1 \times 10^{-2}$  atoms/ion as a guess for the alpha-particle sputtering yield.

For the sake of completeness, the very slight effect of  ${}_0n^1$  is also included. Garber, Dolya, Kolyada, Modlin and Fedorenko<sup>5</sup> give the sputtering yield for monoenergetic 14 MeV neutrons on gold as  $3 \times 10^{-3}$  atoms/neutron. Norcross, Fairand and Anno<sup>6</sup> found a yield of  $1 \times 10^{-4}$  for a proton-moderated fission spectrum of fast neutrons on gold. Since neutron sputtering involves nuclear rather than atomic collisions, we take the relative yields for gold and niobium to be proportional to their total cross sections for 14 MeV neutrons, which are 5.25 and 4.0 barns respectively. This indicates yields as high as  $2.3 \times 10^{-3}$ , or as low as  $7.6 \times 10^{-5}$  atoms/neutron. We chose  $1 \times 10^{-3}$  atoms/neutron, a compromise weighted somewhat in favor of the monoenergetic result.

Proceeding to calculate wall erosion rates, we begin with the quantities of TM-2204. At a power level of 30,000 MW (thermal), approximately  $10^{22}$  D<sup>+</sup>-T<sup>+</sup>/sec =  $S_i - L_i$  are burned, where  $S_i$  and  $L_i$  are rates of ion injection and loss, respectively, and the wall area A is taken to be 3,000 M<sup>2</sup>. The total erosion rate R(atoms/cm<sup>2</sup>-sec) is given by

$$R = \phi_{\text{He-n}}(Y_{\text{He}} + Y_n) + \phi_{\text{D-T}}(Y_D + Y_T),$$

where the Y's are sputtering yields,

$$\phi_{\text{He-n}} = (S_i - L_i)/A = 3.3 \times 10^{14}/\text{cm}^2\text{-sec}, \quad \text{and}$$

$$\phi_{\text{D-T}} = L_i/A = (S_i - L_i)\left(\frac{1}{f_b} - 1\right)/A = 3.3 \times 10^{14} \left(\frac{1}{f_b} - 1\right)/\text{cm}^2\text{-sec}.$$

Erosion rate as a function of burn-up fraction for D-T, and total erosion rates, both in cm/year, are given in Table 1, along with wall lifetime. As in TM-2204, 20% removal of a 1.0 cm. wall is taken as terminating wall life.

Table 1 - Erosion rates and wall lifetime as a function of burn-up fraction.

$f_b$	D-T Erosion Rate (cm/year)	Total Erosion Rate (cm/year)	Wall Lifetime (years)
0.5	0.67	0.67	0.18
1.0	0.33	0.33	0.36
2.0	0.17	0.17	0.72
3.0	0.11	0.11	1.1
4.0	0.081	0.083	1.5
5.0	0.064	0.066	1.8
10.0	0.030	0.032	3.7
20.0	0.014	0.017	7.7
30.0	0.0078	0.0099	12
40.0	0.0051	0.0071	17
50.0	0.0034	0.0054	22
60.0	0.0023	0.0043	28
70.0	0.0014	0.0035	34
80.0	0.00084	0.0029	41
90.0	0.00037	0.0024	49
100.0	0	0.0021	58

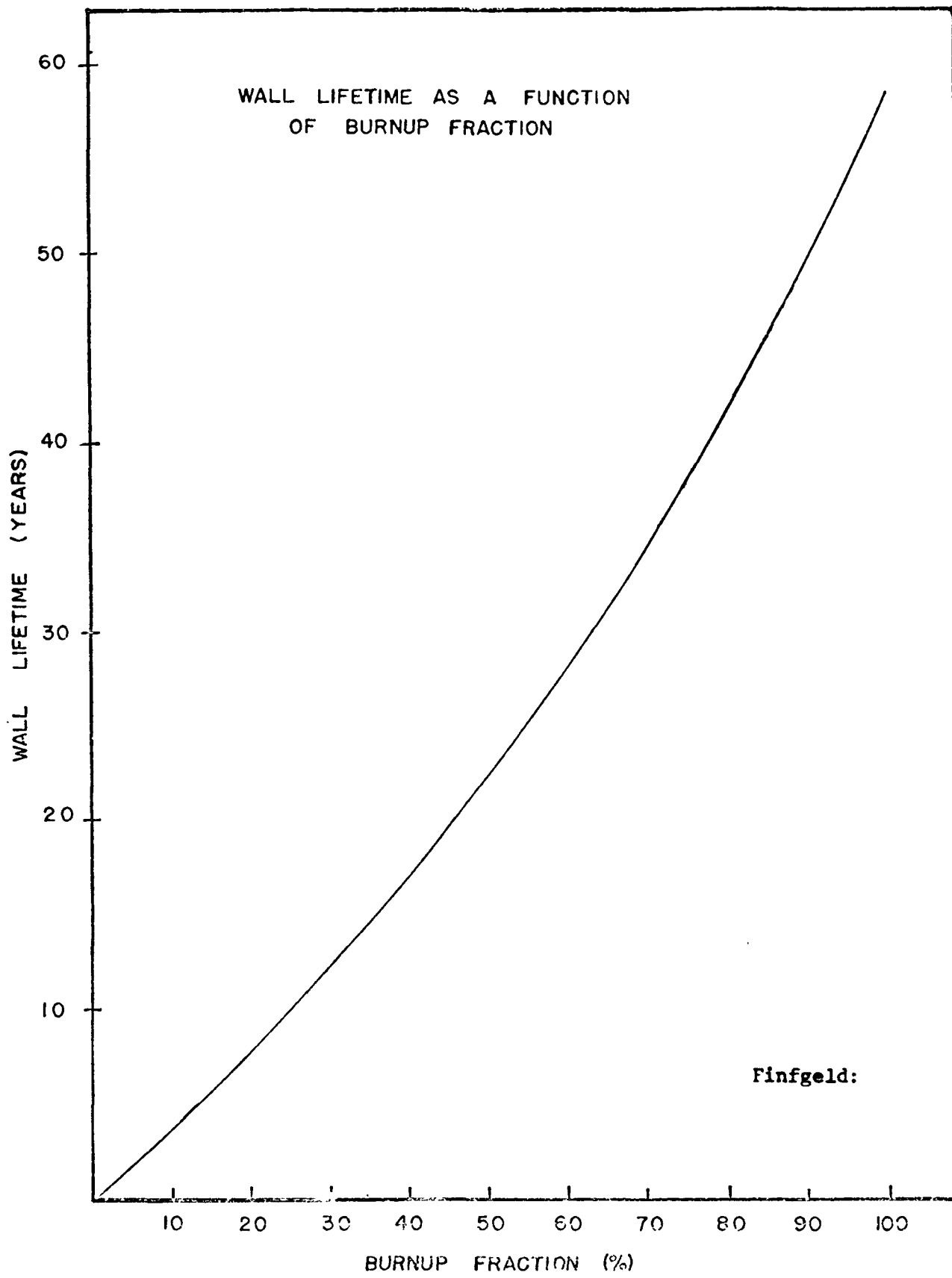


Fig. 2

## CONCLUSIONS

For burnup fractions of 3% or less, D-T sputtering appears to be the only significant mechanism of wall erosion. At approximately 15% burnup, sputtering by He-n contributes 10% to the total erosion rate, and would become the primary determinant of wall lifetime if burnup fractions of 60% or more were ever achieved.

The yields used here are all for clean surfaces, and will probably prove to be higher than those for actual reactor walls with at least monolayer gas contamination. In fact, continuous re-cycling of the hydrogen working fluid by light-ion impart desorption should prove highly advantageous, for both wall lifetime and plasma purity. Bombardment by heavier ions, however, would be quite detrimental.

The sputtering yields upon which these conjectures are based are almost all uncertain, or worse. Yields for  $D^+$  are most reliable, and those for  $T^+$  will become better known when data for protons becomes available. Effects of elevated target temperature and energy variation, as well as yields for several different materials, need to be established.

Thereafter, yields for multiply charged ions, and for  $He^{++}$  in particular, need to be measured. Our data in this area is most inadequate, the yield used herein being simply a guess, and the undesirable effects of  $He^{++}$  bombardment outlined above may be

badly under-estimated. It is expected that, with the achievement of improved burn-up fractions, this information will become increasingly significant in the accurate estimate of wall lifetimes.

#### REFERENCES

- 1 D. J. Rose, ORNL-TM-2204.
- 2 O. C. Yonts, ORNL-TM-2692.
- 3 C. E. KenKnight and G. K. Wehner, J. Appl. Phys. 35, 322 (Feb. 1964).
- 4 M. Kaminsky, Bul. Am. Phys. Soc., 8, 339 (1963).
- 5 R. I. Garber, G. P. Dolya, V. M. Kolyada, A. A. Modlin, and A. I. Fedorenko, ZhETF Pis'ma 7, No. 10, 375-378 (20 May 1968).
- 6 D. W. Norcross, B. P. Fairand, and J. N. Anno, J. Appl. Phys. 37, 621 (1966).

THE NECESSITY OF STUDYING THE IMPLICATIONS  
OF FUSION POWER ON ALL ASPECTS OF SOCIETY

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Abstract

Important technological advances have major repercussions on our daily lives. Controlled fusion power has the potential of being one of those few developments that completely restructures society. Energy is one of the most fundamental factors in a society and these reactors will have major environmental and social effects. It is necessary to consider the wider implications at an early stage of this program so that these considerations can be incorporated into the technological evolution of the fusion reactor, and so that the program will be funded and accepted by those groups concerned with it. It is not enough to have scientists and engineers concerned with these issues. Too many painful experiences of the past have shown the need to have the participation of these in the humanities and in the social, political, and economic sciences. These people must be incorporated into symposia such as this one. The hostility and fear many feel toward science and technology is due in part to the fact that it is often forgotten that ultimately the goal of science and technology must be to serve man, not vice-versa.



ACCOMMODATING FOR UNCERTAINTIES AND  
INTERPLANT VARIATIONS IN THE  
ENGINEERING OF A FUSION  
PLANT-TORCH COMPLEX

by

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## Abstract

Uncertainties in the design parameters of a fusion plant are investigated. The plant is assumed to consist of a power reactor supplying a part of a city with electrical energy. The waste energy from the reactor is to be utilized by a fusion torch. The torch processes scrap material to reduce it to its basic elements.

Sensitivity of the power output of the reactor and the cost per unit energy to uncertainties in confinement time and plasma density is evaluated. A proper plasma density is obtained in case of uncertainty in plasma temperature.

The design criterion for a feasible self-heated fusion reactor is obtained taking into account uncertainties in the parameters involved. The expected value of the density-confinement time product is found to differ in some extent from that evaluated using average values for the parameters where uncertainty resides.

The torch is considered next. The variation in the waste energy throughput to the torch is taken into account. A utilization factor for the energy supplied to the torch is derived. Also obtained is the optimal capacity of the torch that gives maximum venture profit.

The case of pulsed supply of scrap material in the presence of a variable processing rate is finally considered. Utilization factors on daily basis are obtained for the supply and the process. The optimal capacity of the storage and the effect of overflow are discussed.

## 1. Introduction

The outputs of a fusion reactor consists of: useful energy  $W_o$  , waste energy  $W_w$  and escaping plasma particles. The last two are proportional to the inputs to the fusion torch. All outputs are given by:<sup>1,2,3</sup>

$$W_o \approx 1.0 \times 10^{-33} p(1-p) T^{2.36} N^2 \tau / V \quad \text{kWhr. (1-1)}$$

$$W_w \approx 1.0 \times 10^{-3} V / (NT^{1.36} \tau) \quad \text{kWhr. (1-2)}$$

for  $8 \leq T(\text{keV}) \leq 16$ . Here  $N$  is the number of ions or electrons since both species are used at equal numbers,  $\tau$  is the reactor confinement time,  $T$  is the plasma kinetic temperature,  $V$  is the plasma volume and  $p$  is the ratio of tritons to the fuel mixture (deuterons + tritons).

There is a great deal of uncertainty to what will be the confinement time for a specific design. The fact that the confinement time depends in some way or another on the magnetic field strength  $B$ , the plasma density  $n$  and the plasma temperature  $T$ ; results in uncertainties evolving from uncertainties in these parameters. It is likely that the predicted values for  $B$ ,  $n$  and  $T$  will deviate from the actual values after the reactor is put in operation. In addition these variables are liable to changes during operation.

Since the core of the reactor is to be integrated into an injection device, a magnetic field system and a heating device; any uncertainty in the rate of injection of particles or energy will propagate through the design of the reactor. On the other hand

unpredicted changes in the reactor output will have their effect on the design of the torch and the whole plant. These changes have to be considered in the design of the industrial complex which is supplied by the electrical power from this reactor.

Uncertainties in the design parameters of a fusion reactor have their impact on the price of the power produced and their effect may become dangerous to the extent that the fusion plant become unfeasible. This is because a fusion reactor has to have values of  $N, T$  and  $\gamma$  which exceed some low bounds in order the reactor produces useful power. In the case of D-T reactors the feasibility criteria are:

$$N\gamma/V > 10^{14} \text{ sec/cm}^3 \quad (1-3)$$

and  $T > 6 \text{ keV} \quad (1-4)$

In this paper the techniques introduced by Rudd and Watson are used to investigate the following problems:

- I. The sensitivity coefficients of the useful energy and the waste energy due to uncertainties in  $p, N$  and  $\gamma$ ; and the effect of such changes on the cost of the power produced. The propagation of uncertainty in partially controlled parameters through the design is investigated.
- II. The expected value criterion is used to select a proper plasma density in the presence of uncertainty in the plasma temperature.
- III. A great deal of work has been done to develop a design criterion which is given numerically in Eq.(1-3). The values of parameters used to evaluate the value of  $N\gamma/V$  are taken as the average values. Thus the expected value criterion is used to give the desired condition on the product of the plasma

density and the confinement time in the presence of uncertainties.

- IV. The problem of the design of the torch in the presence of variation in the input energy is then treated. A linearized curve of the energy leaving the fusion reactor is used to evaluate the optimal capacity of the torch for maximum venture profit.
- V. The effects of storage on a pulsed supply of material for processing in the torch is investigated taking in consideration the variable but continuous process. Variable energy input is likely to effect the rate of processing.

At the end of this work some of the points needed to be investigated are pointed out.

## 2. Propagation of Uncertainty Through Fusion Reactor Design

### 2-1 UNCERTAINTY IN THE ENERGY OUTPUTS:

Let us test the sensitivity of the energy outputs  $W_w$  and  $W_o$  to changes in the parameters  $N$ ,  $\bar{\tau}$  and  $p$ . These parameters are controlled by the injection device and by the magnetic field configuration. The sensitivity coefficients for  $W_o(p, N, \bar{\tau})$  are:

$$S_p = \frac{\partial W_o}{\partial p} = W_o (1-2p)/(p(1-p)) \quad (2-1)$$

$$S_N = \frac{\partial W_o}{\partial N} = 2W_o/N \quad (2-2)$$

$$S_{\bar{\tau}} = \frac{\partial W_o}{\partial \bar{\tau}} = W_o / \bar{\tau} \quad (2-3)$$

If in the base design  $\bar{W}_o$  is the thermal energy output available for electricity, then the output energy after a slight change in  $p$ ,  $N$  and  $\bar{\tau}$  is given by:

$$W_o = \bar{W}_o + W_o ( 2 + (1-p)(1-\bar{p}/p)/(1-p) - \bar{N}/N - \bar{\tau}/\tau ) \quad (2-4)$$

where the bars indicate that the parameters are those of the base design. From Eqs.(2-1) through (2-3) it is observed that  $W_o$  is not sensitive to slight changes in the value of  $\bar{\tau}$  while a change in  $N$  induces a change of the same order in  $N$  since  $\frac{\partial W_o}{\partial N} \sim 2N$  while  $\frac{\partial W_o}{\partial \bar{\tau}}$  is independent of  $\bar{\tau}$ . The effect of changing  $p$  is milder than that due to changes in  $N$ . The effect on the value of  $\bar{W}_o$  due to changes in  $p$ ,  $N$  and  $\bar{\tau}$  is expressed approximately by Eq.(2-4). As an example, let:  $\bar{p}=0.5$ ,  $p=0.4$ ,  $\bar{\tau}=1$ ,  $\tau=0.8$ ,  $\bar{N}=10^{23}$  and  $N=2 \times 10^{23}$ , then  $W_o/\bar{W}_o \approx 1.2$ .

A word of caution here is that  $\bar{\tau}$  is a function of  $T$ , however, the change is considered to be independent of  $T$ . The effect of changes in  $T$  on  $\bar{\tau}$  is considered afterwards.

Now, let us consider the sensitivity of  $W_w$  to changes in  $\bar{\tau}$  and  $N$ :

$$S_T = \frac{\partial W_w}{\partial T} = -W_w / T \quad (2-5)$$

$$S_N = \frac{\partial W_w}{\partial N} = -W_w / N \quad (2-6).$$

We see that the sensitivity coefficients are negative i.e. an increase in  $T$  or  $N$  causes a decrease in the base design value of  $W_w$ . If we consider the parameters given in the above example, we get a change in  $W_w$  given by:  $W_w / \bar{W}_w = 0.8$ . This result is no surprise since an increase in the useful energy  $W_o$  causes a decrease in the waste energy  $W_w$ .

## 2-2 UNCERTAINTY IN THE COST PER UNIT ENERGY:

Assuming that the amount of money invested  $I$  in the power plant does not change with slight change in  $W_o$ , the cost per kWhr is given approximately by:

$$c = kII/(aP_o) \quad \text{mills/kWh} \quad (2-7)$$

where  $a$  is the plant availability factor,  $i$  is the annual charges,  $P_o = W_o / T$  kW and  $k$  is a conversion factor to express  $c$  in mills per kWh. Now  $T$  does not appear in the equation for  $c$ . The cost per unit power is therefore a function of  $p, N, V$  and  $T$  only. Thus the sensitivity coefficients are:

$$S_p = -c(1-2p)/(p(1-p)) \quad (2-8)$$

$$S_N = -2c/N \quad (2-9)$$

$$S_T = -2.36 cT \quad (2-10)$$

$$S_V = c/V \quad (2-11)$$

Here we included changes in  $T$  and  $V$  since after startup and during operation of the reactor the plasma is likely to expand or contract due to variations in the pressure of the magnetic field and consequently  $V$  and  $T$  will be changed within some uncertainty.

The engineer has a full control of the injection rate, the magnetic field strength and the initial volume of the plasma but he may not be able to decide on  $N$ ,  $V$ ,  $T$ , and  $p$  with 100% certainty.

The signs of Eqs.(2-8) and (2-9) are opposite to those of Eqs.(2-1) and (2-2) although they depend on  $p$  and  $N$  in the same fashion. This is because  $c$  is inversely proportional to  $P_0$  as seen from the approximate relation of Eq.(2-7).

The best guess of the cost per unit electrical energy produced is 4 mills per kWh, taking into account the risk factor proper to the adventure of using a new technology. This best guess of  $c$  corresponds to  $\bar{p}=0.5$ ,  $\bar{N}=1.25 \times 10^{23}$ ,  $\bar{V}=2.5 \times 10^8$  cm<sup>3</sup> and  $\bar{T}=10$  keV; where the overbars are assigned for the best guess values. The effects of the uncertainties on  $c$  is approximately given by:

$$c - \bar{c} = - 6.4 \times 10^{-23} (N - 1.25 \times 10^{23}) - 94.4(T - 10) + 1.6 \times 10^{-8} (V - 2.5 \times 10^8) \quad (2-12)$$

where the sensitivity coefficients are evaluated using the best guess values and  $N$ ,  $T$ , and  $V$  are values near these best guess values. Figure 2-1 shows the relation between  $c$  as a function of  $T$  in a wider range of temperature than that considered above. The propagation of uncertainty in  $T$  through the design is also demonstrated. In situation A, the assumed uncertainty in the temperature of the plasma ions falls in a range in which the cost is sensitive to the temperature. This is the range for which the approximate forms given in Eqs.(1-1) and (1-2) are valid. In this situation the uniform distribution function describing an uncertainty in the ion temperature is mapped in a distorted distribution in the uncertainty of the cost with the range greatly expanded.



The situations designated B and C are at higher temperatures than the popular situation A. In the case of B the assumed uncertainty falls within a range in which the cost is not much influenced by T. Thus a uniform distribution of uncertainty in T is mapped into a nearly uniform distribution of uncertainty in the cost, with the range of uncertainty greatly suppressed. Situation C is an intermediate case between the expanded uncertainty in situation A and the suppressed uncertainty in situation B. Thus the cost is greatly sensitive to the variable T in which the initial uncertainty resides if T is selected around 10 keV.

### 3. Use of Expected Value Criterion to Select the Plasma Density

Consider the capacity of a fusion reactor of volume V and plasma density n (=N/V) fueled by equal portions of deuterium and tritium; the energy conversion from thermal to electrical is at an efficiency  $\eta$ . The power output is given by:

$$P(n,T) = 2.5 \times 10^{-34} \eta^2 V n^2 T^{2.36} \quad (3.1)$$

Assume that the engineer has full control on the design parameters except from the plasma temperature in which uncertainty is encoded as the uniform distribution function:

$$p(T) = \begin{cases} 1/(T_H - T_L) & T_L \leq T \leq T_H \\ 0 & \text{otherwise.} \end{cases} \quad (3-2)$$

The situation is realistic in a steady-state reactor if the injection device is efficient enough to give the required ion density. The

required decision is to find the proper plasma density  $n$  which renders a minimum cost. The costs are assumed to be :

$$C(n,T) = \begin{cases} c_1 P & T \geq T_0 \\ c_1 P + c_2 (Q - P) + C_s & T < T_0 \end{cases} \quad (3-3)$$

where  $c_1$  and  $c_2$  are the costs per unit power of the initial capacity and of the additional power respectively,  $Q$  is the required capacity,  $C_s$  is a penalty for the shutdown to be paid to a substituting power plant which carries on the service during the outage of the plant under consideration and  $T_0$  is the plasma temperature corresponding to a capacity  $Q$ , that is:

$$T_0 = (4 \times 10^{33} Q / n V)^{1/2.36} \quad (3-4)$$

$c_2$  is not necessarily equal to  $c_1$  but is most likely larger, since raising the reactor capacity may require the replacement of some parts

The expected cost is then:

$$\begin{aligned} \overline{C(n)} = & 2.5 \times 10^{-34} \eta^2 V n^2 (c_1 - c_2) (1 / (T_H - T_L)) \int_{T_L}^{T_0} T^{2.35} dT \\ & + (c_2 Q + C_s) / (T_H - T_L) \int_{T_L}^{T_0} dT \\ & + 2.5 \times 10^{-34} \eta^2 V n^2 c_1 / (T_H - T_L) \int_{T_0}^{T_H} T^{2.36} dT \end{aligned} \quad (3-5)$$

Thus,

$$\overline{C(n)} = (T_H - T_L)^{-1} (.67 \times 10^{-34} V n^2 (c_1 T_H^{3.36} + (c_2 - c_1) T_L^{3.36}) - (c_2 Q + C_s) T_L + (4 \times 10^{33} Q / n V)^{2.36} (c_2 Q + C_s - Q/3.36)) \quad (3-6)$$

$$\text{for } (4 \times 10^{33} Q / T_H V)^{2.36 \frac{1}{2}} \leq n \leq (4 \times 10^{33} Q / T_L V)^{2.36 \frac{1}{2}}.$$

Differentiating with respect to  $n$  and equating the result to zero the minimum expected cost is reached when:

$$n^* = (4 \times 10^{33} Q / T_H V)^{2.36 \frac{1}{2}} (c_2 / c_1)^{59/168} (1 + 84 C_s / 59 c_2 Q)^{59/168} \times (1 + (c_2 - c_1) (T_L / T_H)^{3.36} / c_1)^{-59/168} \quad (3-7)$$

The cost  $C_s$  is actually much less than  $c_2 Q$ . In addition if  $(c_2 - c_1) (T_L / T_H)^{3.36} / c_1 \ll 1$ , Eq.(3-7) reduces to:

$$n^* = (4 \times 10^{33} Q / T_H V)^{2.36 \frac{1}{2}} (c_2 / c_1)^{59/168} (1 - 59 (c_2 - c_1) (T_L / T_H)^{3.36} / 168 c_1) \quad (3-8)$$

If we define an overdesign factor  $f$  as the ratio of the recommended design density to the optimistic design density  $(4 \times 10^{33} Q / T_H V)^{2.36 \frac{1}{2}}$ , then:

$$f = (c_2 / c_1)^{59/168} (1 - 59 (c_2 - c_1) (T_L / T_H)^{3.36} / 168 c_1) \quad (3-9)$$

$$\text{for } f < (T_H / T_L)^{1.18}$$

$$f = (T_H / T_L)^{1.18} \quad \text{otherwise.}$$

If  $c_2 = c_1$ ,  $f$  can be obtained from Eq.(3-8) as:

$$f = (1 + C_s / 2 c_2 Q) \quad (3-10)$$

on the assumption that  $C_s \ll c_2 Q$ . Since this is the case in practical situations, it is not dangerous to gamble on possible underdesign, although it is preferable to overdesign by the factors given above.

Actually, in the construction of a huge power plant some money is assigned from the beginning to startup problems which may be due to inadequate estimation of parameters not completely under the control of the engineer.

#### 4. The Expected Design Criterion in the Presence of Uncertainties

It has been shown that the fuel in the steady state reactor under investigation can be heated internally thus saving the large investment which could have been put in the construction of a heating system. The amount of heat which can be retained per unit time in the plasma reaction vessel is calculated from an energy balance on the plasma and is given by:

$$P_h = P_F - P_R \quad (4-1)$$

where  $P_F$  is the fusion energy released per unit time to the alpha ions produced from the fusion reaction and  $P_R$  is the energy radiated from the plasma per unit time. The amount of energy required to heat the plasma per unit time is:

$$P_h = 2.4 \times 10^{-19} n^2 T / \zeta \quad \text{kW} \quad (4-2)$$

Equating Eqs.(4-1) and (4-2) yields the following equation for the product  $n\zeta$  which we will call  $x$ :

$$x(c, E, T) = 12 T / \overline{\beta(T)} cE \quad \text{sec/cm}^3 \quad (4-3)$$

where  $\overline{\beta(T)}$  is the product of the fusion cross section times the velocity of the incident particle averaged over the velocity distribution of the deuterons and tritons and is a function of  $T$  only. In

the range  $8 \leq T(\text{keV}) \leq 16$   $\bar{\rho}(T)$  can be written as:

$$\bar{\rho}(T) = 4.7 \times 10^{-19} \frac{T^{2.36}}{T} \text{ cm}^3/\text{sec} \quad (4-4)$$

where  $T$  is in keV. However,  $\bar{\rho}(T)$  is used for generality. In Eq.(4-3) it is assumed that the electron temperature is almost equal to the ion temperature.  $E^*$  is the fraction of energy released to the alphas per fusion reaction in keV and  $c$  is the fraction of energy retained in the plasma. The parameter  $c$  is liable to uncertainty since the amount of energy loss is questionable and only rough calculations can give the order of magnitude of the losses. The amount of energy released per reaction is usually taken as the most probable energy of a normal distribution of the alphas produced, that is:

$$p(E^*) = \left( \frac{1}{\sqrt{2\pi} \sigma_E^*} \right) \exp\left(-\frac{(E^* - \bar{E}^*)^2}{2 \sigma_E^{*2}}\right) \quad (4-5)$$

where  $\bar{E}^*$  is the most probable energy of the alpha particles and  $\sigma_E^*$  is the variance. There is a 96 per cent chance that the uncertainvalue of  $E^*$  will fall within  $(\bar{E}^* + 2T)$  and  $(\bar{E}^* - 2T)$ , thus:  $\sigma_E^* = T$ ; and the distribution becomes:

$$p(E^*) = \left( \frac{1}{\sqrt{2\pi} T} \right) \exp\left(-\frac{(E^* - \bar{E}^*)^2}{2T^2}\right) \quad (4-6)$$

The expression for the normal distribution is different from that given in Eq.(12.2.6) of reference 5 by a numerical factor for normalization purposes.

The uncertainty in  $c$  encodes a uniform distribution:

$$p(c) = \begin{cases} \frac{1}{(c_H - c_L)} & c_L \leq c \leq c_H \\ 0 & \text{otherwise} \end{cases} \quad (4-7)$$

where  $c_H$  is the optimistic value and  $c_L$  is the pessimistic. This distribution is adequate for  $c$  since its value has a constant likelihood to lie between two bounds.

Now the problem is to find the expected value of the parameter  $x$  which gives the criterion for a feasible self-heated reactor.

The expected value of  $x(c, E^*, T)$  is:

$$\overline{x(T)} = (12 / \sqrt{2\pi} \overline{\rho(T)} (c_H - c_L)) \int_0^\infty dE^* \int_{c_L}^{c_H} dc (cE^*)^{-1} \exp(-(E^* - \overline{E}^*)^2 / 2T^2) \quad (4-8)$$

Carrying out the integral over  $c$  we get:

$$\overline{x(T)} = (12 \ln(c_H/c_L) / \sqrt{2\pi} \overline{\rho(T)} (c_H - c_L)) \int_0^\infty dE^* E^{*-1} \exp(-(E^* - \overline{E}^*)^2 / 2T^2) \quad (4-9)$$

At very low plasma temperatures, i.e. for  $T \ll \overline{E}^*$  the Gaussian is very narrow around  $\overline{E}^*$ ; consequently the bulk of the integral comes from the region in the immediate vicinity of  $E^* = \overline{E}^*$ , and the denominator in the integral can be extracted from the integral as  $1/\overline{E}^*$  and the result is:

$$\overline{x(T)} \simeq 12T \ln(c_H/c_L) / (\overline{\rho(T)} (c_H - c_L) \overline{E}^*) \quad (4-10)$$

for  $T \ll \overline{E}^*$ . In this special case  $E$  in the original expression has just been replaced by the most probable energy.  $\overline{E}^* \simeq 3,520$  keV and the predicted range of plasma temperature is around 10 keV in the case of D-T plasma. In this case Eq.(4-10) is applicable and it can be written explicitly as:

$$\overline{x(T)} \simeq 2.55 \times 10^{19} \ln(c_H/c_L) / (T^{1.36} (c_H - c_L) \overline{E}^*) \quad (4-11)$$

At high temperatures, on the other hand, the normal distribution is broad, and the integral in Eq.(4-9) is now dominated by the term  $1/E^*$ . Since this term is peaked at  $E^* < \overline{E}^*$ , the exponential can be

extracted as  $\exp(-\bar{E}^*/2T)$ . Carrying out the integral over the limits assigned above leads to undesirable diversion and the answer is impractical. To avoid such situation we may bound the uncertainty in  $E^*$  to the range between a low value  $E_L^*$  and an upper bound  $E_H^*$  on the account that the uncertainty below  $E_L^*$  and above  $E_H^*$  is almost zero. This is the case of interest since particles of  $E^* \gg \bar{E}^*$  are likely to leave the reaction vessel and will not contribute to the heating of the plasma. Ions emerging from the reaction with energy close to the ionization energy will rapidly neutralize. Hence  $\bar{x}(T)$  is approximately given by:

$$\bar{x}(T) \simeq 12 \ln(c_H/c_L) \ln(E_H^*/E_L^*) \exp(-\bar{E}^{*2}/2T^2) (\sqrt{2\pi} \bar{f}(T) (c_H - c_L))^{-1} \quad (4-12)$$

for high T.

Equation (4-3) is well known as the design criterion for a feasible thermonuclear reactor. However, Eq. (4-10) gives the expected design criterion taking into account the uncertainties existing in the values of c and  $E^*$ . Such uncertainties have been disregarded and the most probable values were used as a way of simplifying the treatment. Thus Eqs. (4-10) and (4-12) are considered to be more adequate than Eq. (4-3) since the uncertainties do in fact exist.

To complete the picture let us check the sensitivity of  $\bar{x}(T)$  to uncertainty in T. The sensitivity coefficient is :

$$S_T = d\bar{x}(T)/dT = -1.36 \bar{x}(T)/T \quad (4-13)$$

for  $8 \leq T(\text{keV}) \leq 16$  and D-T plasma. A negative sensitivity coefficient is to the advantage of the designer. An underestimated T leads

to a larger  $\bar{x}(T)$ . A larger confinement time means more reaction will take place and more energy will be available to heat the plasma, thus raising T. An overestimated T leads to a less  $\bar{x}(T)$  thus decreasing the heat available for the plasma and consequently reducing T.

This observation is useful in controlling the reactor during operation, since  $S_T$  can be regarded as a temperature coefficient for  $\bar{x}(T)$ , i.e.:

$$\alpha_T = S_T/x(T) \simeq -1.36/T \quad (4-14)$$

for D-T plasma and  $8 \leq T(\text{keV}) \leq 16$ .  $\alpha_T$  is the temperature coefficient of  $\bar{x}(T)$ . For a critical parameter as  $\bar{x}(T)$  it is to the designer's advantage to have a negative temperature coefficient.

The value of  $\bar{x}(T)$  as a matter of fact plays as a milestone in the design of a self-heated steady state fusion reactor ; since a value of  $\bar{x}(T)$  less than that given by Eq.(4-10) requires the use of energetic injection or external heating mechanism which is a burden on the reactor economy. A larger  $\bar{x}(T)$  than that of Eq.(4-10) is a burden on the technology which is trying hard to reach a reasonable value of x at which any fusion could take place.

## 5. Design of the Fusion Torch in the Presence of Variations

Figure 5-1 shows a simplified block diagram of a fusion reactor-torch assembly. The function of the unit is discussed in reference 1. The torch is to process scrap material or raw material



using the waste energy diffusing out of the reactor. The energy is transmitted from the plasma to the torch via a divertor with an efficiency  $\eta$  less than one. The energy input to the torch is thus:

$$W_T \approx 1.0 \times 10^{-3} \eta_D \frac{V}{N} \rho T^{1.36} \quad \text{kWh} \quad (5-1)$$

for D-T plasma at  $8 \leq T \text{ (keV)} \leq 16$ .

The energy  $W_T$  available to the torch is not constant but varies throughout the day and the seasons. In addition uncertainties in the design parameters of the fusion reactor propagates through the operation range of the torch. There is no means of controlling  $W_T$  without interfering with the performance of the reactor and since the torch is mainly handling the energy refuse from the reactor it has to be flexible to ride out reasonable deviations from the average  $W_T$ .

The energy  $W_T$  is used to process  $w_{s1}$  ion pairs per hour of scrap material. In order to exhaust the energy  $W_T$  efficiently the scrap material must be fed approximately at a rate:

$$w_{s1} = W_T r_1 / W_1 \quad (5-2)$$

where  $W_1$  is the energy per ion pair and  $r_1$  is the processing rate. Assuming  $r_1$  and  $W_1$  are constant for a specific type of scrap material, the rate of feeding the scrap must closely match the variations in  $W_T$ . In the case of power stations deviations from the mean tend to be the rule rather than the exception.

Figure 5-2 shows an idealization of the typical daily variation curve for the energy input to the torch. This reactor waste energy is assumed to vary in the same fashion as the electrical energy produced from the reactor  $W_0$ . The electrical energy is considered to match the load of power consumption.

Consider that every energy unit utilized yields a profit  $P_p$  \$ per unit energy, that each unit unutilized by the torch costs the company  $L$  \$/unit to dispose of as waste and that the investment in the torch is given by the power law:

$$I = I_0 (Q/Q_0)^M \quad (5-3)$$

The total amount of energy units supplied in  $T$  hours ( a day ) are given by :

$$Q_t = Q_L ((Q_b - Q_L)/(Q_p - Q_L))T/2 + Q_{H1} T_1/2 + Q_{H2} T_2/2. \quad (5-4)$$

The total amount of energy units lost are:

$$Q_l = \frac{1}{2} Q_{H1} T_1 + \frac{1}{2} Q_{H2} T_2 + \frac{1}{2} T (Q_p - Q_D)^2 / (Q_p - Q_L) - \frac{1}{2} Q_p T (Q_p - Q_b) / (Q_p - Q_L). \quad (5-5)$$

The amount of energy units utilized are:

$$Q_u = \frac{1}{2} (Q_p + Q_L) T - \frac{1}{2} Q_b T - \frac{1}{2} T (Q_p - Q_D)^2 / (Q_p - Q_L) \quad (5-6)$$

for  $Q_L < Q_D \leq Q_b$ .

Since the economy of the torch is different in nature from the economy of the utilities, one can write the venture profit as:

$$V = \frac{1}{2} P_p T (Q_p + Q_L - Q_b - (Q_p - Q_D)^2 / (Q_p - Q_L)) - \frac{1}{2} L (Q_{H1} T_1 + Q_{H2} T_2 + (Q_p - Q_D)^2 T / (Q_p - Q_L) - Q_p T (Q_p - Q_b) / (Q_p - Q_L)) - I_0 (Q_D / Q_0)^M. \quad (5-7)$$

Differentiating  $V$  with respect to  $Q_D$  and setting the derivative to zero, the optimal capacity can be obtained from the relation:

$$(Q_p - Q_D) / (Q_p - Q_L) = I_0 M (Q_D / Q_0)^{M-1} / ((P_p + L) Q_D T) \quad (5-8)$$

for  $Q_L < Q_D \leq Q_b$ .

If  $Q_D > Q_b$ , Eq.(5-4) for  $Q_t$  is still the same and Eq.(5-5) for  $Q_1$  becomes:

$$Q_1 = \frac{1}{2}T_1 (Q_{H1} - Q_D)^2 / (Q_{H1} - Q_b) + \frac{1}{2}T_2 (Q_{H2} - Q_D)^2 / (Q_{H2} - Q_b) \quad (5-9)$$

Consequently, the utilized energy is:

$$Q_u = \frac{1}{2}Q_L T_b (Q_{H1} - Q_D) / (Q_{H1} - Q_b) + \frac{1}{2}T_1 (Q_{H1} - (Q_{H1} - Q_D))^2 / (Q_{H1} - Q_b) + \frac{1}{2}T_2 (Q_{H2} - (Q_{H2} - Q_D))^2 / (Q_{H2} - Q_b) \quad (5-10)$$

The venture profit is then given by:

$$V = \frac{1}{2}P_L (Q_{H1} T_b (Q_{H1} - Q_D) / (Q_{H1} - Q_b) + T_1 (Q_{H1} - (Q_{H1} - Q_D))^2 / (Q_{H1} - Q_b) + T_2 (Q_{H2} - (Q_{H2} - Q_D))^2 / (Q_{H2} - Q_b)) - \frac{1}{2}L (T_1 (Q_{H1} - Q_D)^2 / (Q_{H1} - Q_b) + T_2 (Q_{H2} - Q_D)^2 / (Q_{H2} - Q_b)) - I_o (Q_D / Q_o)^M \quad (5-11)$$

and the optimal  $Q_D$  is obtained from the relation:

$$\frac{T_1 (Q_{H1} - Q_D) / (Q_{H1} - Q_b) + T_2 (Q_{H2} - Q_D) / (Q_{H2} - Q_b)}{1} = \frac{1}{2} I_o M (Q_D / Q_o)^{M-1} / ((P + L) Q_D) \quad (5-12)$$

Similarly if  $Q_D < Q_L$ :

$$Q_u = Q_D T \quad (5-13)$$

$$Q_1 = Q_t - Q_D T \quad (5-14)$$

$$V = P Q_D T - L (Q_t - Q_D T) - I_o (Q_D / Q_o)^M \quad (5-15)$$

and  $Q_D$  optimum is obtained from :

$$1 = i I_o^M (Q_D / Q_o)^M / ((P_p + L) Q_D T). \quad (5-16)$$

The energy utilization factor is simply :

$$\bar{\Phi}_E = Q_u / Q_t \leq 1 \quad (5-17)$$

and the venture profit could have been expressed in terms of  $\bar{\Phi}_E$  as:

$$V = (P_p \bar{\Phi}_E - L(1 - \bar{\Phi}_E)) Q_t - i I_o (\bar{\Phi}_E Q / Q_o)^M. \quad (5-18)$$

## 6. Pulsed Supply of Scrap Material to a Torch of Variable Processing Rate

Consider that the scrap material arrives to the torch for processing in discrete batches as truck loads. The processing rate varies in the same manner as the energy supplied Fig. 5-2 . Figure 6-1 shows the rate of arrival of the material to be processed. On the same plot the maximum rate of operation of the torch in processing this type of material is shown to the same scale. The utilization of energy from the reactor is taken as  $\bar{\Phi}_E = 1$ . A storage compartment is provided for the arriving material.

Due to space limitation the capacity of the storage compartment is finite and relatively small. Any overflow has to be reshipped back or sent to a remote storage place. The effect of the storage limited capacity on the process is to be investigated. This problem arises if the fusion plant complex is located in a heavily populated area as have been suggested<sup>1,2</sup>.

Trucks unload the material continuously in the  $j$ th cycle for  $A_j$  hours at an idealized rate of  $H_j$  units per hour. No delivery takes place in the next  $B_j$  hours and then the cycle repeats itself. In a period of twenty four hours four cycles take place. The pattern of variable operating rate of the torch is repeated every day. During delivery  $H_j > Q(\Theta)$  at any time  $\Theta$ , where  $Q(\Theta)$  is the processing rate with maximum energy utilization factor.  $Q(\Theta)$  is approximately given by:

$$Q_i(\Theta) = m_i \Theta + q_i \tag{6-1}$$

where  $i$  refers to the segment of the linearized operating rate curve under consideration,  $m_i$  is the slope of the segment and  $q_i$  is the intersection of the  $i$ th segment with the vertical axis. The values of  $m_i$  and  $q_i$  takes four different values over the day period, since the operating rate curve Fig. 6-1 is idealized by four straight line segments labelled as I to IV.

The supply utilization factor  $\Phi_s$  is defined as the fraction of the supply that can be utilized by the torch. This will be calculated on a daily basis. In the case of no storage:

$$\bar{\Phi}_s = \frac{\sum_{j=1}^n (H_j A_j - (H_j A_j - \int_{\theta_j}^{\theta_j + A_j} Q(\theta) d\theta))}{\sum_{j=1}^n H_j A_j}$$

$$= \frac{\sum_{j=1}^n \int_{\theta_j}^{\theta_j + A_j} Q(\theta) d\theta}{\sum_{j=1}^n H_j A_j} \quad (6-2)$$

The integration is over a time duration of  $A_j$  hours. The time  $\theta_j$  is the time at which a supply arrives. In the situation demonstrated in Fig. 6-1, we have:

$$4HA \bar{\Phi}_s = (m_{I1} A_1^2 + m_{II2} (A_2^2 + 2A_2 \theta_2) + m_{III3} (A_3^2 + 2A_3 \theta_3) + m_{IV4} (A_4^2 + 2A_4 \theta_4) + 2(q_{I1} A_1 + q_{II2} A_2 + q_{III3} A_3 + q_{IV4} A_4)) \quad (6-3)$$

where the initial time  $\theta_1$  is taken as the zero time reference. Arabic numeral subscripts refer to the order of the supply energy cycle. In the absence of storage  $(H_j A_j - \int_{\theta_j}^{\theta_j + A_j} Q(\theta) d\theta)$  units of material must be shunted to the overflow every cycle; where  $\theta_j$  is the time at which the cycle to the supply starts.

In the period  $0 \leq \theta \leq A_j + \theta_j$ , the amount  $S(\theta)$  of material stored increases as follows:

$$dS(\theta)/d\theta = \begin{cases} (H_j - Q(\theta)) & \text{for } S(\theta) \leq C \\ 0 & S(\theta) = C \end{cases} \quad (6-4)$$

where  $C$  is the storage capacity.

Integrating Eq.(6-4) with the assumption that  $S(\theta_j) = 0$ , we get:

$$s(\theta) = \begin{cases} H_j(\theta - \theta_j) - \int_{\theta_j}^{\theta} Q(\theta') d\theta' & \text{for } S(\theta) \leq c \\ c & \text{otherwise} \end{cases} \quad (6-5)$$

which is valid as long as  $\theta_j \leq \theta \leq \theta_j + A_j$ .

The overflow first occurs at  $\theta_{oj}$  which can be obtained by setting  $\theta = \theta_{oj}$  in Eq.(6-5), that is:

$$s(\theta_{oj}) = H_j(\theta_{oj} - \theta_j) - \int_{\theta_j}^{\theta_{oj}} Q(\theta') d\theta' = c \quad (6-6)$$

or:

$$\theta_{oj} = \left( \frac{(H_j - q_1)}{m_1} \right) \left( 1 \pm \left( 1 - \frac{(2m_1((H_j - q_1)\theta_j + c - \frac{1}{2}m_1\theta_j^2))}{(H_j - q_1)^2} \right)^{\frac{1}{2}} \right) \quad (6-7)$$

Let us assume that  $H_j > q_1$  and that:

$$2m_1 \left( (H_j - q_1)\theta_j + c - \frac{1}{2}m_1\theta_j^2 \right) < (H_j - q_1)^2 \quad (6-8)$$

therefore Eq.(6-7) reduces to the two solutions :

$$(\theta_{oj} - \theta_j) = (c - \frac{1}{2}m_1\theta_j^2) / (H_j - q_1) \quad (6-9)$$

or

$$(\theta_{oj} - \theta_j) \simeq \left( \frac{2(H_j - q_1)}{m_1} \right) - 2\theta_j - \frac{(c - \frac{1}{2}m_1\theta_j^2)}{(H_j - q_1)} \quad (6-10)$$

The second solution is not the proper one in the limit of constant processing rate, that is  $Q(\theta) = q_1$  and  $m_1 = 0$ ,  $\theta_{oj} - \theta_j = \infty$ , while the solution of Eq.(6-9) reduces to the solution given by

Rudd and Watson (Eq.(14.2.4) reference 5) where  $\theta_j$  is taken as zero. Consequently Eq.(6-9) is taken as the solution.

If any overflow takes place in the  $j$ th cycle :

$$(\theta_{oj} - \theta_j) = (C - \frac{1}{2}m_1 \theta_j^2) / (H_j - q_1) \leq A_j \quad (6-11)$$

However, it is inconvenient in the case under study to have overflow otherwise the scrap material has to be discarded causing loss to the processing company. Consequently there is a critical value  $C^*$  for which the supply will never exceed the processing capacity plus the storage. This value is:

$$C^* > A_j(H_j - q_1) + \frac{1}{2}m_1 \theta_j^2 \quad (6-12)$$

If the engineer has a priori knowledge of  $H_j$  and  $A_j$  before constructing the storage space; he can estimate the storage place by calculating the value of  $C^*$  for each pulse and selecting the largest value. If on the otherhand, the engineer can control the values of  $H_j$  and  $A_j$  with the presence of a limited  $C$ ; he can select them to match the values of storage capacity. In the case when  $H_j$ ,  $A_j$  and  $C$  are neutral to the control of the engineer, the overflow occurs when:

$$\theta_{oj} - \theta_j \leq A_j \quad (6-13)$$

The amount of scrap material that must be shunted to the overflow in one day is :

$$\sum_{j=1}^4 (H_j (A_j + \theta_j - \theta_{oj}) - \int_{\theta_{oj}}^{\theta_j + A_j} Q(\theta) d\theta) \quad (6-14)$$

Thus, the supply utilization factor in the presence of the storage is:



$$\bar{\Phi}_s = \frac{\sum_j (H_j A_j - (H_j (A_j + \theta_j - \theta_{oj}) - \int_{\theta_j}^{\theta_j + A_j} Q(\theta) d\theta))}{\sum_j A_j H_j} \quad (6-15)$$

in one day.

Substituting for  $(\theta_{oj} - \theta_j)$  from Eq.(6-9) and evaluating the integral we can write the supply utilization factor as:

$$\begin{aligned} \bar{\Phi}_s = & \left( \sum_j H_j A_j \right)^{-1} \left( \sum_j \left( H_j \left( C - \frac{1}{2} m_1 \theta_j^2 \right) / (H_j - q_1) \right) \right. \\ & + \frac{1}{2} m_1 \left( A_j^2 - \left( C - \frac{1}{2} m_1 \theta_j^2 \right) / (H_j - q_1) \right)^2 \\ & + 2 \theta_j \left( \left( C - \frac{1}{2} m_1 \theta_j^2 \right) / (H_j - q_1) - A_j \right) \\ & - 2 A_j \left( C - \frac{1}{2} m_1 \theta_j^2 \right) \\ & \left. + q_1 \left( A_j - \left( C - \frac{1}{2} m_1 \theta_j^2 \right) / (H_j - q_1) \right) \right) \quad (6-16) \end{aligned}$$

Equation (6-16) reduces again to the value of  $\bar{\Phi}_s$  in the case of constant operation, considering that we are calculating  $\bar{\Phi}_s$  here for several cycles per day (4).

If we define the process utilization factor  $\bar{\Phi}_p$  as the fraction of possible production capacity per day that is actually utilized, then:

$$\bar{\Phi}_p = \left( \bar{\Phi}_s \sum_j H_j A_j \right) / \left( \int_0^{T=24 \text{ hrs}} Q(\theta) d\theta \right) \quad (6-17)$$

where the denominator is given by the area under the operation rate curve. Assuming that the operation rate follows very closely the rate of energy variation given in Fig. 5-2 which is practically the

case if we consider that the operation rate is directly proportional to the energy available for the process. If the constant of proportionality is taken as  $g$  then:

$$\int_0^T Q(\theta) d\theta = gQ_t \quad (6-17)$$

where  $Q_t$  is given by Eq.(5-4).

The economically optimal storage capacity can be found as follows:

If we assume that the net profit to be earned by the processing of a unit of scrap material in the torch is  $P_p$ . If the material which is shunted to the overflow costs the company a penalty of  $L_p$  \$/unit the total net profit for the torch is:

$$\sum_j H_j A_j (\bar{\Phi}_s P_p - (1 - \bar{\Phi}_s) L_p) \quad (6-18)$$

The investment can be estimated as:

$$I = I_o (C/C_o)^M + I_1 \quad (6-19)$$

where  $I_1$  is the constant investment in all equipment except storage and  $\alpha$  is the collection of interest terms and the like required to amortize the investment on a day basis, the venture profit to be maximized is:

$$\max_{\{C\}} \left( \sum_j H_j A_j (\bar{\Phi}_s P_p - (1 - \bar{\Phi}_s) L_p) - \alpha (I_o (C/C_o)^M + I_1) \right). \quad (6-20)$$

The values of  $\bar{\Phi}_s$  can be easily modified to take into account values of  $\bar{\Phi}_E$  less than one. Seasonal variation of supply material and energy can also be considered.

## 7. Concluding Remarks and Recommendation for Further Work

The uncertainties that reside in the design parameters make it impractical to rely upon average values only. Although in this paper we attempted to tackle some of the important design problems as the design criterion for a useful power fusion reactor and the variation in the processing rate of the torch; an intensive numerical evaluation is needed in order to reach decisive conclusions about the optimal capacity, optimal cost and so forth. Variation in the enrichment of some isotopes in the fuel may reveal valuable information.

The account for a tolerance to failure has to be considered. Failure tolerance is actually a complicated problem in the case of plasma since we are dealing with hydrogen and radioactive material. Vulnerability to sabotage and possible radiation hazard makes it necessary to study disaster propagation in a power plant consisting of storage places for hydrogen isotopes and of several operating reactors. A three dimension evaluation may be required in this case since optimal segregation distances are effected to a great extent by the difference in magnitude and the size difference.

The general residence time concept<sup>7</sup> can be used to calculate the burnup rate of the fuel. Residence time corresponds to the reactor confinement time and the burnup corresponds to the activity of the catalyst in chemical reactors.

The transfer function of the reactor is already evaluated<sup>8</sup>. Thus by developing a local linearized model of the plasma process, the iterative optimization technique introduced by Rudd, Aris and Amundson<sup>9</sup> can be used to optimize the design of the system under arbitrary constraints.

## REFERENCES

- 1) Z. A. Sabri, University of Wisconsin, PLP 338, Nov. 1969
- 2) B. J. Eastlund and W. C. Gough, "The Fusion Torch", WASH-1132, 1969.
- 3) D. J. Rose, "The Need for Fusion", Proceedings of the International Conference on Nuclear Fusion Reactors, Culham, U. K., 1969.
- 4) J. D. Lawson, AERE GP/R 1807, 1955.
- 5) D. F. Rudd and C. C. Watson, Strategy of Process Engineering, John Wiley & Sons, Inc., New York, 1968.
- 6) R. G. Mills, Nuclear Fusion, 7, 1967.
- 7) Dale F. Rudd, The Canadian J. of Chem. Eng., 40, 1962.
- 8) Z. A. Sabri, A Research Progress Report, Nuc. Eng. Dept., U. of Wis., 1968.
- 9) D. F. Rudd, R. Aris and N. R. Amundson, A.I.Ch.E., Journal, Sept. 1961.

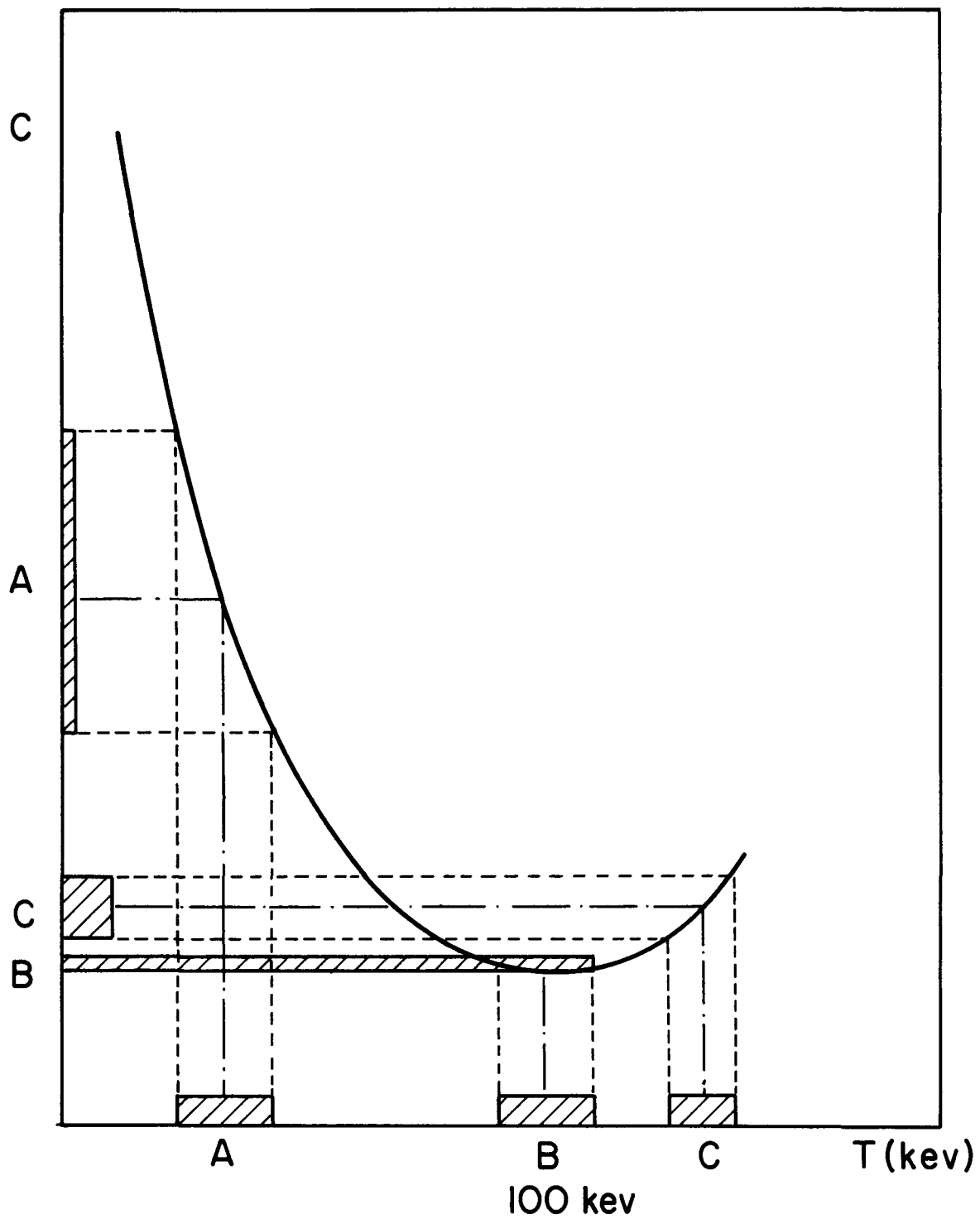


Fig.2 -1 Propagation of uncertainty through design

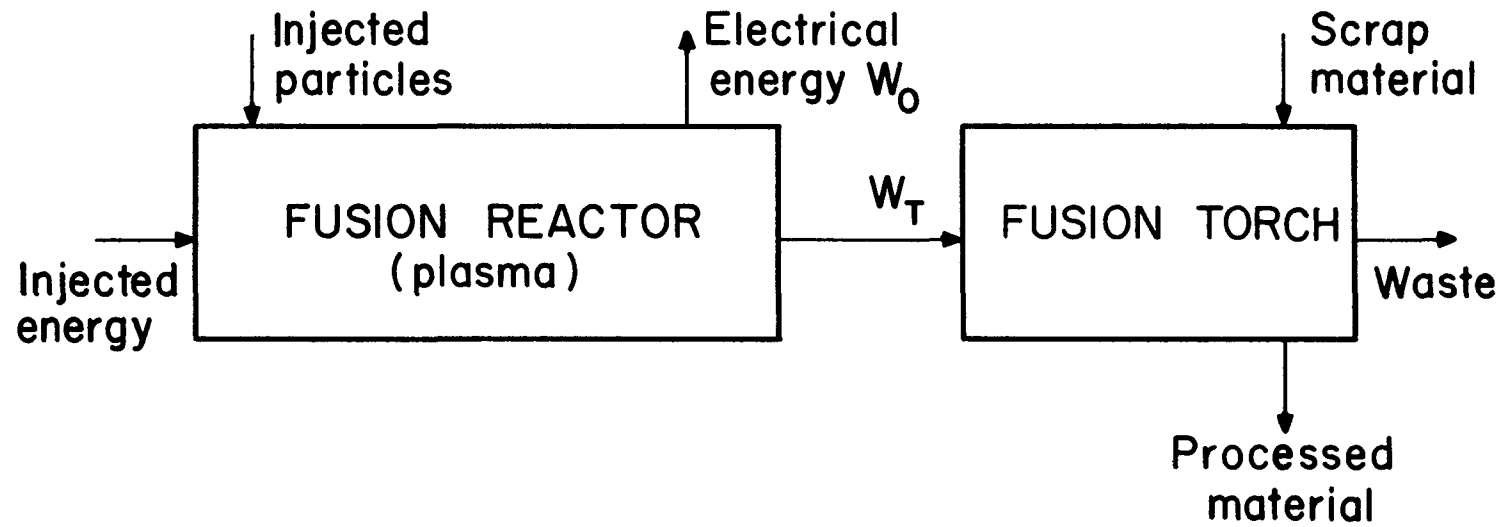


Fig. 5-1 FUSION REACTOR TORCH COMPLEX

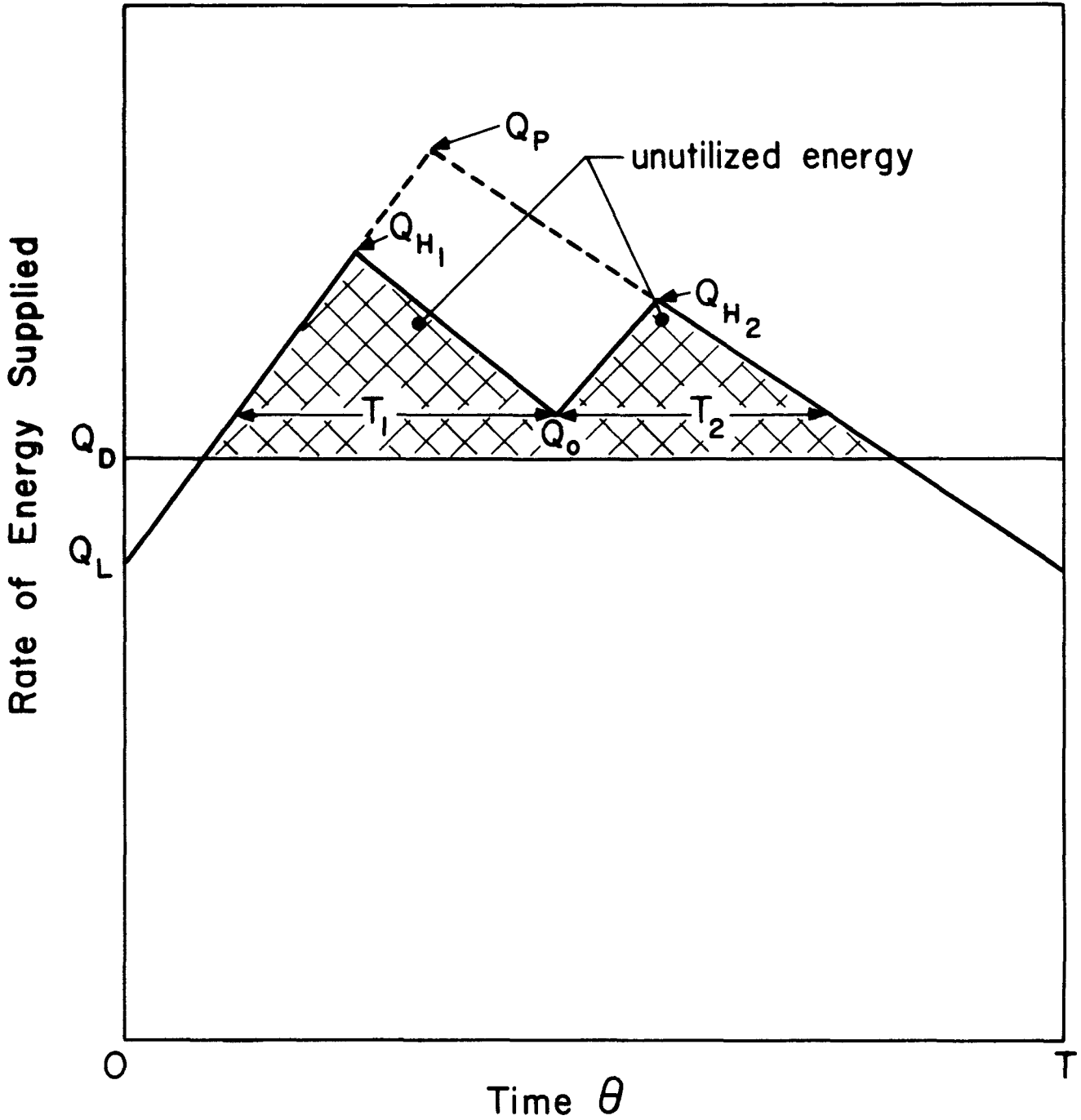


Fig. 5-2 Idealized Daily Variation in the Energy Available for the Torch.

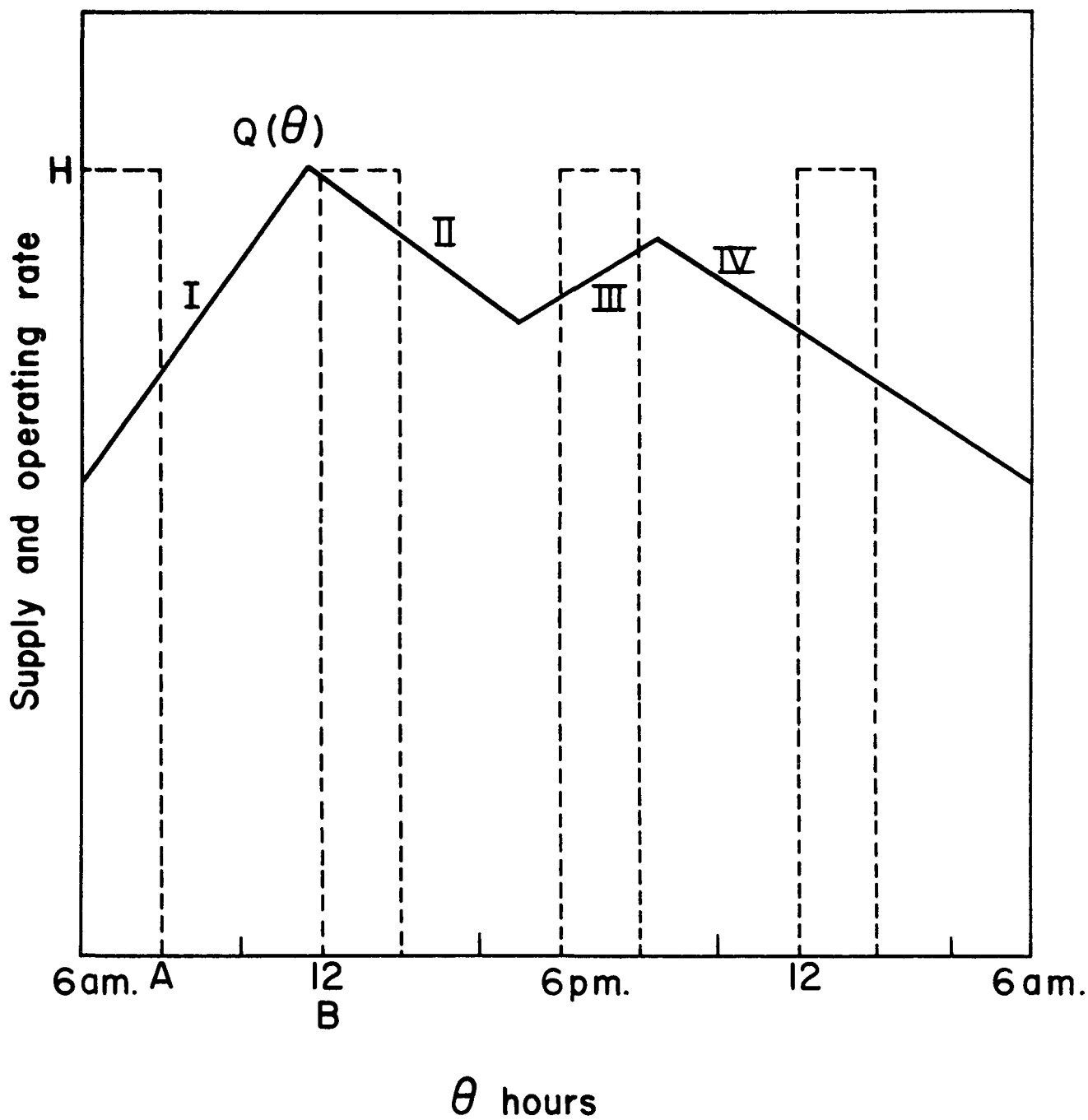


Fig. 6-1 Rate of scrap material supply and maximum processing rate.



Experience Teaching A  
Fusion Design Seminar

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A fusion reactor design seminar was offered during the spring semester at Cornell on a trial basis. Six graduate students were involved, two of whom had completed courses in plasma physics while the others had reactor physics and engineering backgrounds<sup>†</sup>. The class was run in a true seminar fashion; while lectures were presented by the instructor and other invited lecturers<sup>††</sup>, much of the class involved progress reports and summaries of individual investigations by class members. As described later, the climax of the class was a presentation in an open seminar attended by a variety of persons with backgrounds ranging from fish science and economics to plasma physics.

The assignment was to design a 5000 MWe CTR for siting in or near New York City. Sufficient detail was required to enable a reasonable cost estimate, and a discussion of sociological implications was requested. Somewhat arbitrarily, the class was divided into two groups: one was to work on a closed, steady-state device similar to a Stellarator or Tokomak while the second was assigned a pulsed mirror-type device. After some discussions, the latter group was permitted to assume  $\beta = 1.0$  while because of stability considerations the first group was restricted to  $\beta \leq 0.15$ . Both were to use the D-T reaction. At this point it was thought that the pulsed system might be competitive because of the high  $\beta$ -value and also the ability to inject fuel and pump liquid metal coolant during the period between pulses when magnetic fields were small or absent.

† The students were: John Glancy, Tom Holleman, Tom Johns, Massoud Navidi, Dave Strobel, and David Thompson.

†† Invited lecturers included: Professors Norman Rostoker; Charles Wharton; Hans Fleischmann; and also Mr. G. Staley of New York State Gas and Electric Company.

Previous conceptual design studies by Rose, Carruthers et al; Mills; and Ribe et al were not distributed to the class until the third week. Thus, the students were forced to begin thinking about the task without being prejudiced by past approaches. Rather than slowing progress, this is thought to have instilled more enthusiasm and a deeper insight, as often occurs when one wrestles with a difficult problem without the aid of "ready made" solutions.

In the space available, it would be impossible to present the two class designs in any detail. However, some comments about the results may transmit the "flavor" of the studies. Some typical parameters are given in Tables I and II.

As anticipated, the steady-state design turned out to have a reasonable size, and based on an advanced thermal energy conversion cycle (not direct conversion), a respectable 50% overall efficiency was predicted. Several interesting points might be noted. The torus radius, it turned out, was determined by space requirements for the blanket and diverters rather than the wall thermal loading. In contradiction to previous ORNL calculations the power required to pump the lithium coolant across field lines was found to be excessive if conducting tube walls were employed. To get around this, EM pumps were inserted at each point where flow crossed lines (the pumps use the confining B-field for their operation), and flow along field lines was utilized over a majority of the path.

Despite its efficiency, the sheer size of the plant represents a problem for waste heat disposal, e.g. assuming a 20°F temperature rise, 1.8 million gpm would be used for cooling (about six times that used at the Shoreham Nuclear Power Station on Long Island Sound). Unless some use can be found

TABLE I

Some Parameters for the Steady-State Design

<u>Components</u>		<u>Parameters</u>	
Magnet	Nb <sub>3</sub> Sn-copper super-conducting 6m major dia., 2.43 <del>minor</del> dia.	Minor Radius of Torus	1 meter
		Major Radius	5.43 meters
Diverters	Four (0.1 MW power)	Thermal Load on Vacuum Wall	3.86 MW/m <sup>2</sup>
Vacuum Pumps	Ten 32-inch diffusion pumps	Average Ion Energy	10 keV
Vacuum Wall	Niobium web	Particle Density	5 x 10 <sup>14</sup> cm <sup>-3</sup>
Coolant	Lithium	β	0.15
Coolant Pumping Power	76 MW	Magnetic Field	120 kG (inner windings) 34.6 kG (outside)
Ion Injection Power	312 MW	Total Thermal Power	10,600 MW
Energy Conversion Plant	Potassium-steam binary vapor cycle	Output Power	5280 MWe
		Overall Efficiency	49.8%
		Breeding Ratio	1.35

TABLE II

Some Parameters for the Pulsed Design

<u>Components</u>		<u>Parameters</u>	
Magnet	Lithium cooled TZM, 7-cm thick	Plasma Radius	1 meter
Magnet Power	Superconductor storage system	Magnet Coil Radius	10 meters
		Length	16 meters
Coolant	Lithium	Mirror Ratio	2.5
Coolant Pumping Power	143 MW	Peaked Field Strength	100 kG
Energy Conversion Plant	Potassium-steam binary vapor cycle	Ion Energy (compressed)	58 keV
		Electron Energy (compressed)	39 keV
		Particle Density (compressed)	$2.6 \times 10^{15} \text{ cm}^{-3}$
		Duty Cycle	<b>1/9</b>
		Fuel Burnup Fraction	0.073
		Pulse Length	70 msec.
		Total Thermal Power	21,800 MW
		Plant Feedback Power	5,900 MW
		Output Power	5,000 MW
		Overall Efficiency	23%
		Breeding Ratio	1.15

for this (space heating, etc.), it would be necessary to pipe it to a disposal area in the Atlantic, adding to plant costs.

The pulsed reactor did not turn out to be as favorable, as seen by the low efficiency found. The difficulty seems to have been the following: In order to obtain a reasonable value of  $\psi$  (probability of escape per collision) it was necessary to go to relatively high ion temperatures. This in turn, reduced the amount of direct conversion energy recoverable during the expansion phase of the pulse since this conversion is inversely proportional to temperature. Concurrently, the magnet design (a conventional design inside the blanket) involved considerable energy losses, particularly due to Joule heating.

This design did not attempt to use direct collection methods to capitalize on the mirror leakage, and while this might help, it would not completely cure the problem.

Clearly the two designs were strongly influenced by the previous design studies noted earlier. Still many changes have been incorporated, the scaling is different, etc. as the class faced each problem anew.

Both concepts left a number of unanswered questions. For example, advances in fuel injection techniques of several orders of magnitude were assumed without a firm basis. Reasonable extrapolations of present technology simply would not be sufficient. Questions about radiation damage to materials, etc., were simply given token recognition.

Both groups devoted considerable time to the cost estimates summarized in Table III. It is seen that the fusion plant cost estimates are somewhat higher than reported in previous studies. Although the same basic unit cost figures were used (e.g. the cost per foot of magnet, etc.) many more details were

TABLE III

Cost Estimates

	Steady-State Fusion	Pulsed Fusion	Current Fission <sup>†</sup>
Fixed Charges [Mills/kW-hr]	5.2	4.46	4.07
Fuel Charges [Mills/kW-hr]	0.0056	0.02	1.70
Operating and Maintenance [Mills/kW-hr]	0.30	0.79	0.30
Insurance [Mills/kW-hr]	0.10	0.10	0.10
	<hr/>	<hr/>	<hr/>
Total	5.61	5.37	6.17

† From Philip Sporn, "Economics of Nuclear Power Badly Eroded," Nuclear News, March 1970.

included, and perhaps less optimistic assumptions were made relative to escalation costs, etc. (Such a view was taken based on the history of fission plants where early estimates were a factor of two lower than those reported by Sporn in Table III.) Further the two fusion plant estimates are not entirely self-consistent. The pulse design appears to have a lower fixed charge, largely due to the reduced cost of the magnets and perhaps an overly optimistic estimate for a superconducting energy storage system (yet to be developed). The operating and maintenance costs for this design were taken to be fairly high (ballpark of estimates for fast breeder reactors) since it is complex. In retrospect, it appears that the steady state costs are likely to have been set too low.

In conclusion, it is obvious that the costing of a plant still contains large uncertainties. Still one boundary condition all agree on is that the small fuel cost relative to fission reactors should lead to a lower cost since other charges don't appear to be that much different.

Another interesting point about the estimates is that the major capital item in both designs (about 2/3's of the cost) was the turbine-electric section, while magnets and coils were second. Since the former involves a fairly well established technology, the estimate for it should be expected to be reasonably accurate. In the pulsed design, the magnetic storage unit was also found to represent a major cost item. This automatically introduces a larger uncertainty since the storage system, especially switching gear, has yet to be developed.

An important aspect of the class was a presentation of the results at a final seminar. It was listed in the university calendar as an open meeting, and in addition a special review panel was invited. It included persons from



Radiation Biology and also Fish Sciences (both of whom had been active in the recent protest against the Bell Nuclear Station proposed for Lake Cayuga); a representative from New York State Gas and Electric Company; a member of the Electrical Engineering staff interested in power generation; and a plasma physicist. The wide variety of backgrounds represented promoted an interchange of thoughts that effectively tied together and highlighted the semester's work. Again, it is difficult to provide adequate coverage here, and only a few representative remarks are possible.

Persons with non-nuclear backgrounds were somewhat skeptical. They had read news releases about the safety of fusion plants, the reduction of pollution, and the fusion torch concept. This all sounded too good to be true and the impression seemed to hold that it was "overselling" ..... something must be wrong or overlooked! One person pointed out that the study was entirely based on AEC reports and data, and expressed an inherent distrust since this agency is charged with development of the CTR. (An argument familiar to persons in the fission field.) The thrust of questions from these people centered around pollution problems and arguments about the large size of the plant. The waste heat rejection was noted earlier. The audience felt that this problem has been "skipped over" ..... only token thought had been given to possible uses, and the cost suggested for dumping in the ocean and the dump site, connected ecology, etc. only given lip service. It was generally concluded that more urgency should be given to research on D-He<sup>3</sup> approaches so that higher efficiencies might be achieved. One thought, not generally agreed to, was that "demonstration" devices using D-T represented a political approach which, if by-passed, would save the

public much money and distress in the long run.

The cost estimate was also viewed with some suspicion in light of the many uncertainties involved. It was suggested that what was needed is to form an unbiased observer (group) to make a comprehensive cost analysis of the various competing sources and concepts. This way, at least, all would be on an equal footing, using the same ground rules and assumptions, etc. Carrying this one step further, it was suggested that an outside group, not directly associated with the AEC development, might be asked to undertake an entire conceptual study in order to evaluate the sociological implications from "an unbiased point of view."

Radioactivity questions also consumed much time. Sure fusion would be better than fission in this respect, but would it really be good enough "to save mankind?" Some arguments about maintenance problems due to induced radioactivity in the structure were raised, but the most important questions centered around tritium control. Tritium recovery by diffusion through the niobium pipe walls in the main heat exchangers was proposed in the designs. Might it also diffuse through other walls and escape? Class calculations indicated that the permeation rate through other components (largely made of stainless steel and operating at much lower temperatures) would be well below tolerable limits. The audience questioned the accuracy of this calculation (questioned the validity of the diffusion coefficient) and also pointed to the rather large inventory involved as a potential hazard. (This was a failure of the present design; it was estimated that over 100 kG of tritium would be dissolved in the rather large volume of lithium used for coolant and in the blanket. Other approaches, e.g. the use of heat pipes, would greatly reduce this inventory.) The argument that tritium would be contained and reused in

the same plant was warmly applauded.

Without going further, it is clear from this sampling of comments that continuing discussions with concerned persons from other areas are desirable if a general acceptance of the fusion approach is expected. The average person is simply distrustful at this stage. Hopefully, this seminar helped a little in this direction, but much more is needed to obtain effective communication.

In conclusion, it can be stated that a seminar like this is extremely effective in exciting a student's imagination and attracting him to an area. Design seminars are not uncommon; the fusion problem is especially attractive in the present stage since the class will typically gain the feeling that they are undertaking an "open" problem in parallel with other conceptual design groups at the various AEC laboratories. The present studies worked out very well despite the variations in backgrounds among the students. Those with nuclear backgrounds originally feared that a stress on plasmas would "snow" them. However, they found that there were plentiful engineering challenges so that a knowledge of plasmas, stability, etc. was not absolutely necessary to make a contribution. However, assignments were varied on purpose so that each student was forced to look at a variety of topics. As it turned out, the engineers were quick to pick up some basic plasma background, and vice versa. (Appropriate lectures were also included to aid this process.) It is the author's opinion that the mix of backgrounds was, in fact, an aid rather than detriment. In fact, were the course offered again, an attempt would be made to interest students from other departments, both in and outside engineering.

SOME THOUGHTS ON THE DESIGN OF A THERMONUCLEAR SPACE POWER SYSTEM\*

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INTRODUCTION

Since the dawn of the nuclear era the Air Force has shown interest, both officially (ANP-Project) and unofficially (various publications by its members) in the exploitation of nuclear energy. As I am not the official Air Force historian I shall mention only some of the early events with which I have been intimately involved.

To the best of my memory the earliest nuclear plasma engine configuration considered was after the publication of Alfvén's book<sup>1</sup>. This book showed the usefulness of magnetic fields as confinement walls for a plasma medium. The plasma was then assumed to be of fission origin as it was discussed in detail by the author<sup>5</sup>. The first electric thrust generators were mentioned in Alexander's book<sup>2</sup>.

The first performance analysis of a nuclear (fission) electric space propulsion system was done and reported in 1952<sup>3</sup>. When in 1954 it was found that a thermonuclear plasma is not in thermodynamic equilibrium and the bremsstrahlung losses ( $\propto T^{1/2}$  rather than  $\propto T^4$ ) are predominant the first performance analysis of a thermonuclear power plant was made (1955). It was a crude analysis because it employed low accuracy thermonuclear cross-sections published in the Physical Review between 1951 and 1954. In these calculations only bremsstrahlung losses were considered.

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\*The views and opinions expressed by the author do not necessarily represent those of the AFOSR or the USA.

After the publication of more accurate cross-sections these calculations were drastically revised and extended. A more drastic revision and extension followed the first Geneva Atoms for Peace Conference (1953).

In 1956 an electric plasma thrust generator was designed and built by my associate and student (OSU Graduate Center) Dr. Kenneth Kissell. It was subsequently tested successfully. A copper thrust chamber was used as anode and a coaxial tungsten electrode as cathode. The electric power supply was a rotary arc welder. Argon and helium gas were employed as working media.

In 1957 Col. Paul Atkinson started the present AFOSR Electric Propulsion Program. The first project scientist was Milton Slawsky. This program stressed low temperature plasmas, electrostatic thrust generators, colloids and MHD power generators.

In 1962 when Samaras took over the project (9752) a change from low to high temperature plasmas was initiated.

Whereas the AEC work on thermonuclear power was based on microscopic approaches and Plasma Physics, the early Air Force work<sup>5</sup> was mostly based on macroscopic approaches and Plasma Engineering. It is not surprising that little overlapping occurred and the information from both may be considered as complementary.

It should be well understood that a thermonuclear space power plant, although based on the same fundamental plasma physics as a stationary terrestrial power plant, is vastly different from a stationary one. There are many fundamental differences between the aforementioned:

- a. A space power plant requires extremely high thermodynamic

efficiency in order to minimize the weight of the radiator and increase the payload.

b. It demands a direct transformation of thermonuclear into electrical energy.

c. It should have a very wide range of efficient operation, e.g., the ratio of maximum to cruising power may be  $10^2$  to  $10^3$ .

d. A very advanced and reliable control system is mandatory.

#### THE APPROACH TO DESIGN STUDIES

The philosophy of early investigations was based on zero order effects and the experience gained from Jet Propulsion in World War II. This approach is based upon a given (rather general) configuration and a thermodynamic cycle. In most cases the configuration does not have to be very detailed although it may be more specific than a system consisting of black boxes.

A typical example of a black box configuration is shown in Fig. 1 taken from Ref. 5, p.453. While this may be suitable for early design investigations a more detailed one is required even for a preliminary design study.

The thermodynamic cycle is shown in Fig. 2. In the most general case, the thermodynamic cycle will represent the following processes:

a. Polytropic compression (energy addition from external sources, e.g., turbulent heating).

b. Energy release caused by the nuclear reaction

c. Polytropic expansion (energy extraction and transformation).

The thermodynamic medium in which the thermodynamic changes occur is usually a plasma flow.

It is well known that the thermodynamic cycle represents mean values of the parameters and one-dimensional flows. Again it is understood that the thermodynamic diagram varies little from steady functioning to cyclic operating machines.

The thermodynamic cycle is plotted in two coordinates: total enthalpy (including both kinetic and electromagnetic)  $\chi$  and entropy  $S$ . Any other suitable coordinates may be used, for example, kinetic temperature, pressure and others; experience with jet propulsion systems, however, suggests that the former are extremely useful for cycle performance analysis.

In Fig.2 line 1-2 indicates a diabatic compression, namely, a compression accompanied by energy (heat) addition. The ignition point lies on this line and is not far away from point 2.

Line 2-3 corresponds to a rapid energy release process which, most of the time, may be considered as isobaric.

Line 3-5 represents adiabatic expansion which under certain conditions may approach an adiabatic one.

Sometimes both the compression and expansion processes of the plasma may approach adiabatic conditions. For comparison, the ideal case of an isentropic variation may be invoked in all cases.

As the cycles under consideration are not markedly different from those of other power-producing devices, similar efficiencies may be defined. On the other hand, the losses occurring in each part of the cycle are different; consequently, they require a detailed analysis and appraisal.

Generally speaking, the losses appear as energy losses and as particle losses. The energy losses are either associated with radiation or with energy carried away by mass. The most important radiation losses are:

- a. Bremsstrahlung
- b. Gyromagnetic radiation
- c. Excitation radiation
- d. Cherenkov radiation.

The energy losses associated with mass are:

- a. Heat conduction and convection
- b. Energy carried by runaway particles
- c. Energy lost by charge exchange collisions.

The loss of mass is caused by:

- a. Diffusion
- b. Runaway particles
- c. Neutral particles which cannot be confined by the magnetic field.

The preceding losses may cause a number of effects on the plasma and the surrounding solid walls. These are contamination of the plasma resulting from thermal evaporation of solid walls and sputtering.

The compression and expansion processes have been discussed in detail before<sup>5</sup>, however, a few fundamentals on the direct transformation of nuclear into electrical energy may be given. Before this it is advisable to compare thermonuclear with chemical power plants.

In today's chemical power plants, such as gasoline and diesel engines, gasturbines and rockets, the time of confinement of the reacting medium (time of flame propagation) is about one (1) millisecond. The corresponding



temperatures are 0.12 - 0.5 ev (1200-6000° K). In thermonuclear power plants the temperatures are expected to be 10 to 100 kev, i.e., five (5) orders of magnitude higher. From this it may be concluded that plasma confinement times larger than the aforementioned (millisecond) cannot be seriously supported. To explain this contention, in the following the time required for energy release and the time available from diffusion will be calculated.

### 1. The time of energy release

In 1957 Lawson's rule was announced: the plasma density multiplied by the time of confinement is constant

$$n\tau = \text{constant} \quad (1)$$

Immediately after (1958) it was shown by Samaras<sup>5</sup> (p.326 and Fig. 4.43) that  $n\tau$  is not constant but varies enormously with initial  $\theta_2$  and final  $\theta_3$  temperatures (substitute time of energy release  $t_{23}$  for  $\tau$ ) as follows:

$$nt_{23} = f(\theta_2, \theta_3) \quad (2)$$

Figures 3 and 4 (taken from Ref. 6) show the energy release time as a function of the initial  $\theta_2$  and final  $\theta_3$  temperatures for D-T and D-He<sup>3</sup> reactors.

### 2. The time of confinement

To obtain the time of confinement the plasma diffusion coefficients should be known. This is easily done by non-dimensionalizing the laminar and turbulent diffusion coefficients as a diffusion parameter:

$$\text{laminar} \quad C_d = \frac{q_e B D \ell}{k_B T} = (\omega \tau)^{-1} \quad (3)$$

$$\text{turbulent } C_d = \frac{q_e B D_t}{k_B T} = K \quad (4)$$

where:

$q_e$  = charge of the electron

$B$  = magnetic field intensity

$D_l, D_t$  = laminar and turbulent diffusion coefficient, respectively

$k_B$  = Boltzmann's constant

$T$  = temperature

$\omega\tau$  = Hall's parameter

The experimental results of various diffusion experiments are shown in Fig. 5. This picture is similar to that of the friction coefficient in the flow through a circular pipe with walls of various degrees of roughness. The laminar and turbulent regions are obvious and the roughness coefficient has an analog in the type of boundary layer existing between the plasma and the confining magnetic field.

The diffusion coefficient  $D_{\perp}$  may be correlated with a characteristic length  $L$  of the power plant and a diffusion time which for convenience may be called as  $t'_{23}$  as follows:

$$D_{\perp} = f \frac{L^2}{t'_{23}} \quad (5)$$

where:

$f$  = constant

Another important parameter is the kinetic over magnetic pressure ratio.

$$\beta = \frac{2n k_B \theta}{B^2} \cdot \mu_m \quad (6)$$

From the foregoing assuming a reference ion density  $n_r = 10^{20} \text{ m}^{-3}$  the following results:

$$t'_{23}(n \times 10^{-20}) = 24.65 \left( \frac{f\beta L^2}{C_d} \right) \left( \frac{B_2}{10} \right)^3 \cdot \theta_2^{-2} \quad (7)$$

The most pessimistic case, namely, of Bohm diffusion is considered ( $C_d = 1/16$ ). For simplicity the following values of the parameters may be assumed:  $f = 1$ ,  $\beta = 1$ ,  $L = 1$  meter. Then Eq. (7) may be plotted as broken line in Figs. 3 and 4 for various values of the magnetic field B.

Examination of these figures suggests that for magnetic fields realizable in the future ( $B = 40$  Tesla) and final temperatures  $\theta_3 = 100$  kv, the confinement time is larger than the energy release time  $t'_{23} \gg t_{23}$ .

Taking as an example  $\theta_2 = 20$  kv,  $\theta_3 = 50$  kv and an ion density  $n = 10^{23} \text{ m}^{-3}$  the required time for energy release is  $t_{23} = 1.3$  msec. The corresponding diffusion time for a magnetic field  $B = 15$  Tesla is  $t'_{23} = 3.3$  msec.

Assuming a diffusion coefficient one order of magnitude lower than Bohm (see Fig. 5) the corresponding diffusion time is  $t'_{23} = 33$  msec.

Increasing the magnetic field to  $B = 30$  Tesla the diffusion time becomes  $t'_{23} = 27$  msec and for  $B = 40$  Tesla is  $t'_{23} = 64$  msec.

The foregoing magnetic fields are not fictitious. Existing superconductors produce magnetic fields of 20 Tesla, whereas those in the experimental stage (see Dallas Meeting of the APS) are around 40 Tesla. The high magnetic field intensity superconductive magnets will need a force-free design as it was discussed by M. Levine of AFCL sometime ago. One of the most important components of the system is the radiator which radiates the losses. Preliminary calculations indicate reasonable sizes at temperatures of 1100 to 1200° K. While these temperatures seem excessive for maximum

power they are considerably decreased for cruising conditions which may correspond to  $10^{-2}$  to  $10^{-3}$  of the maximum power. Then corresponding temperatures may be calculated

$$\text{Rel } T = 10^{-2/4} \text{ to } 10^{-3/4} = 0.316 - 0.178$$

and for  $T_m = 1200^\circ\text{K}$ ,  $T_{cr} = 379 \text{ to } 213.5^\circ\text{K}$  which are acceptable.

The high temperatures of maximum power will appear in the flight corridor where aerodynamic cooling may have a significant effect. Some early calculations (Fig. 6) indicate the altitude speed characteristics of the spacecrafts with large, medium, and small engines.

#### DIRECT TRANSFORMATION OF NUCLEAR INTO ELECTRICAL ENERGY

The energy transformation between the plasma and its surroundings may be easily understood by examination of the energy equation.

The mechanism of energy transformation may be examined in various ways, and the methods of approach used depend upon the investigator's viewpoint.

A plasma in a magnetic field behaves as a diamagnetic medium  $k_m < 1$ ; this diamagnetism of course depends upon its conductivity. Each charged particle of the plasma rotates around a magnetic line of force and thus possesses a magnetic moment  $M_d$ . The sum of these magnetic moments is the total magnetic moment of the plasma in the magnetic field. It is well known also that an electric current loop possesses a magnetic moment  $M_d$ . From the foregoing, the similarity between a plasma in a magnetic field and an electric current is obvious. Taking as an example a cylindrical plasma in a magnetic field, it is seen that the plasma may be represented by a current layer,  $i$ , whose magnetic moment per unit length

is the same as that of the plasma. With increasing plasma temperature which may be caused by the energy release process, the magnetic moment  $M_d$  of the plasma increases and with it the current,  $i$ . The rising current induces electromotive forces in the electromagnetic circuits which surround the plasma. This process continues until the maximum temperature of the cycle is attained. Further transformation of plasma mechanical enthalpy may occur by an expansion of the plasma. This may be effected by a reduction of the external magnetic field.

The system of plasma and external magnet coils may also be considered as a transformer. The primary of the transformer is the plasma and the secondary is the magnet coils. This is shown in fig. 7 both schematically and in a circuit form.

The external magnetic field interacts with the plasma and this interaction is shown as the magnetic pressure upon the plasma caused by the external magnetic field  $p_m = B^2/(2\mu_m)$ . It is known that an increase of the external magnetic field  $B$  compresses the plasma, that is, transfers energy from the external coils to the plasma. Again an expansion of the plasma transfers electric energy to the external coils.

Some typical results of calculations are shown in Figs. 8 and 9.

#### POWER EQUILIBRIUM, STARTING AND STARTERS

The power equilibrium in a fusion reactor is not markedly different from that of any other heat engine. Thus the power equilibrium equation for a continuous reactor will be similar to that of a gas turbine and the power equilibrium of an impulsive reactor will be similar to a reciprocating engine. It should be stressed, however, that the losses will

be different in each case.

For a continuous reactor in a steady state of operation, the power generated by the reaction and the power of the starter (external power introduced into the system) will be equal to the excess power and the losses; In an unsteady state of operation, the energy required for the acceleration or deceleration should also be considered. Then the power equilibrium equation may be written

$$P_f + P_{st} = P_{ex} + P_{ac} + \sum (P_L) + P_{aux} \quad (8)$$

where  $P_f$  = power produced by the fusion reaction

$P_{st}$  = power of the starter,

$P_{ex}$  = excess power,

$P_{ac}$  = power of acceleration (negative in the case of deceleration),

$\sum (P_L)$  = sum of the power losses,

$P_{aux}$  = power required for the auxiliaries.

The power losses may be analyzed into

- a. Radiation losses (bremsstrahlung and gyromagnetic)
- b. Leakage power losses
- c. Joulean losses
- d. Heat conduction losses

and other losses peculiar to the system under consideration.

For an impulsive reactor in an unsteady state of operation, the energy balance per cycle may be written

$$\int_0^{t_{cy}} (P_f + P_{st}) dt = \int_0^{t_{cy}} [P_{ex} + P_{ac} + \sum (P_L) + P_{aux}] dt \quad (\text{joules}) \quad (9)$$

where  $t_{cy}$  = duration of the cycle in seconds

and the other symbols have their usual meaning.

Depending upon the cycle of operation, a number of simplified assumptions may be made and simple zero order effects may be obtained.

A starting process is necessary to put a nuclear energy release system

into operation. This is not a peculiar requirement of nuclear systems because a similar process is also necessary in chemical energy release systems, such as gas turbines and reciprocating internal combustion engines.

In the special case of nuclear energy release systems installed in a spacecraft, there may be two main types of starting: (1) starting at ground level; (b) starting in space. In ground level starting, large power external starters may be available most of the time. In space starting, the low density of the surrounding space helps considerably in creating the desirable vacuum conditions in the system.

Depending upon the type of the system, the power required by the starter is given by Eq. (8) or by Eq. (9).

A large number of starting systems for fusion energy release engines may be proposed; because of the extreme requirement upon power, however, the solid fuel MHD generator seems to have certain merits.

Depending upon the starting system selected and the type of the engine, various methods of starting may be developed.

#### WATCHING AND CONTROL

Nuclear energy release systems may be divided into two basic types: (a) those which can be analyzed into individual components; (b) those which cannot be analyzed into individual components. The first type may be considered as comparable to a gas turbine system; the second, to a reciprocating engine.

The main advantage of the first type over the second is its capability

of being analyzed into individual components which can be investigated and tested separately. Generally speaking, each one of these components performs a discrete function and is represented by a separate line in the thermodynamic cycle. As always, this great advantage of component separation and testing is accompanied by the disadvantage of the necessity of matching the various components during the synthesis stage of the energy release system. This situation may be aggravated by the appearance of narrow operational characteristics of the different components and the incompatibility of different requirements, such as easy starting, low weight, high efficiency, and high reliability.

Past experience with terrestrial jet propulsion engines indicates that some of the greatest operational difficulties may be attributed to unsatisfactory matching of the components.

To perform the matching of the components during the synthesis process, a satisfactory understanding of the component characteristics is necessary. Again, to satisfy the performance requirements, certain rules for changing the performance characteristics are needed.

To a certain degree, existing information may be considered sufficient to allow us to calculate the performance of each of the components at the design point and its vicinity; today, however, there is little knowledge about the theoretical evaluation of the instabilities and performance away from the design point. In this case, experimental data should be provided, if possible.

By using similarity and non-dimensional analysis methods, steady state performance characteristics may be developed. It is anticipated



that the characteristics during unsteady operation will be different from those of steady operation.

During the development of the performance characteristics, considerable help and inspiration may be obtained by employing information available from terrestrial jet propulsion engines. Then instead of time, such as cycle time,  $t_{cy}$ , its inverse may be used, namely, the number of cycles per unit time which corresponds, to a certain degree, to the rotational speed  $N$ .

The equation for the excess power may be written in a functional form

$$P_{ex}/(n_2\theta_3N) = f_2\{\theta_2/\theta_3, \theta_1/\theta_2, \theta_0/\theta_2, \eta_{cx}, a\} \quad (10)$$

Taking into consideration the linearity characteristics (of Fig.4.43 of Ref. 5) the following results

$$P_{ex}/(n_2\theta_3N) = f_3\{(N/n_2), (\theta_1/\theta_2), (\theta_0/\theta_2), \eta_{cx}, a\} \quad (11)$$

Equation (11) represents the non-dimensional excess power of an engine as a function of five parameters. Fortunately, only some of these can be varied independently; thus, the compression-expansion efficiency is a function of the other parameters. As the injection temperature  $\theta_0$  is usually fixed and the relative mass addition parameter  $a$  can be kept constant, only two parameters may be varied and the performance of the engine may be plotted as

$$P_{ex}/(n_2\theta_3N) = F[(N/n_2), (\theta_1/\theta_2)] \quad (12)$$

In a similar way, the fuel consumption and the specific fuel consumption may be expressed in terms of the parameters just cited.

When an engine is functioning, the operator must always be able to control it to the desired conditions. Thus a control system should be incorporated with the energy release system.

The control system should be designed for both steady and unsteady (accelerating, decelerating) conditions.

Experience has shown that the energy release system and its controls should not be examined separately but always as a system. The main reason for this requirement is that one of the main functions of the controls is to protect the engine at any marginal operational condition. Experience with nuclear reactors suggests that the protection afforded by automatic control considerably increases the life and reliability of the system.

Control systems may be classified according to the level of intelligence available in the loop. Thus the lowest level of intelligence is held by the open-loop control system. The next is the closed-loop control system. Above this is the adaptive control system.

The next three steps are held by artificial intelligence control systems, namely, automata with various degrees of sophistication which are adept at learning and pattern recognition.

For the sake of convenience human intelligence was also subdivided into three (3) levels. Animal intelligence may be considered as overlapping between artificial and human intelligence.

From the preceding, it is obvious that artificial intelligence automatic controls and protecting devices should be incorporated in all thermonuclear energy release systems. The protecting devices should be incorporated in complete electronic control systems and they should

contain electronic temperature limiters, pressure limiters, and others.

The control system may be subdivided into two parts: controls for steady operation and controls for acceleration.

The variables to be controlled may be subdivided into two groups: dependent and independent variables. These are shown in the following table

DEPENDENT AND INDEPENDENT VARIABLES OF A NUCLEAR ENERGY RELEASE SYSTEM	
<i>Independent Variables</i>	<i>Dependent Variables</i>
Time of the reaction or rps	Power extraction
Plasma density ( $n_D + n_T$ )	Specific fuel consumption
Fuel mixture ratio $n_T/(n_D + n_T)$	
Temperature	

For a simple system, all four independent variables shown in the table may be varied independently; in many cases, however, it is advisable to reduce the number to one or two. Such a system will employ a single lever and the variation of the other variables will be such as to guarantee optimum power or minimum specific fuel consumption.

The measuring instruments of the controlled quantities are numerous and have been discussed elsewhere<sup>2</sup>.

For design of the control system, methods similar to those discussed before<sup>5</sup> may be employed. The equations to be used for these are

- a. The energy conservation equation
- b. The mass conservation equations.

Depending upon the type of the control system, these equations may be simplified considerably. This is true for the mass conservation equations which may be reduced to a single one.

An increase in engine power is effected by increasing the fuel injection rate beyond and above that required for steady state operation.

Assuming constant reaction time  $t_{23}$  or its inverse  $N$ , the maximum temperature  $\theta_3$  increases. Under difficult cooling conditions, the increased maximum temperature may affect the engine life adversely; consequently, due consideration must be given to limiting the maximum temperature  $\theta_3$  below a certain level during the acceleration. At the same time, the rate of acceleration must be fast enough to give satisfactory system operation. This may be accomplished by limiting the top temperature to a reasonable maximum and making the period of acceleration **short** enough to satisfy the engine operation requirements.

The automatic acceleration control which satisfies the conditions just mentioned should be an integral part of the automatic fuel injection system.

#### AUXILIARIES

The energy release system requires extra power to drive a large number of components collectively called auxiliaries. The auxiliaries may be subdivided into two main groups:

- a. Those directly connected with the operation of the engine, such as fuel pumps, vacuum pumps, and others.
- b. Those which may be required in the immediate environment of the engine.

The various installations in a spacecraft are serviced by auxiliaries:

- a. Electric requirements, such as generators, transformers, control panels, lighting, radar, and others.
- b. Navigation equipment
- c. Heating and airconditioning.

Various methods of extracting power from the engine may be devised. These may fall within either of the following categories; direct extraction and utilization and indirect extraction.

The direct methods utilize the electric energy in the form produced by the engine, whereas the indirect use transformers and other intermediate equipment.

In conclusion it is felt that the time is ripe to train the designers who will be able to initiate design studies leading to the final design for the development of a successful prototype thermonuclear power plant.

#### LITERATURE

1. Alfvén, H. "Cosmical Electrodynamics" International Monographs in Physics, Oxford, Clarendon Press, 1950.
2. Alexander, H. "Tomorrow's Air Age" Rinehart, New York, p. 119 1953.
3. Samaras, D. G. "Some Future Trends in Aviation 1970 - 2000 A.D." WADC Staff Meeting, June 19, 1952, USAF, WADC, TMR, No. 52-34, 1952
4. Samaras, D.G. "Theory of Ion Flow Dynamics" Prentice Hall, Inc. Space Technology Series, Englewood Cliffs, New Jersey, 1962.
5. Samaras, D. G. "Applications of Ion Flow Dynamics" Prentice Hall, Inc. Space Technology Series, Englewood Cliffs, New Jersey, 1962
6. Samaras, D.G. "Thermodynamic Considerations of Thermonuclear Space Propulsion" Proc. 16th Intern. Astronautical Congress, p.305, Athens (Greece), September 1965

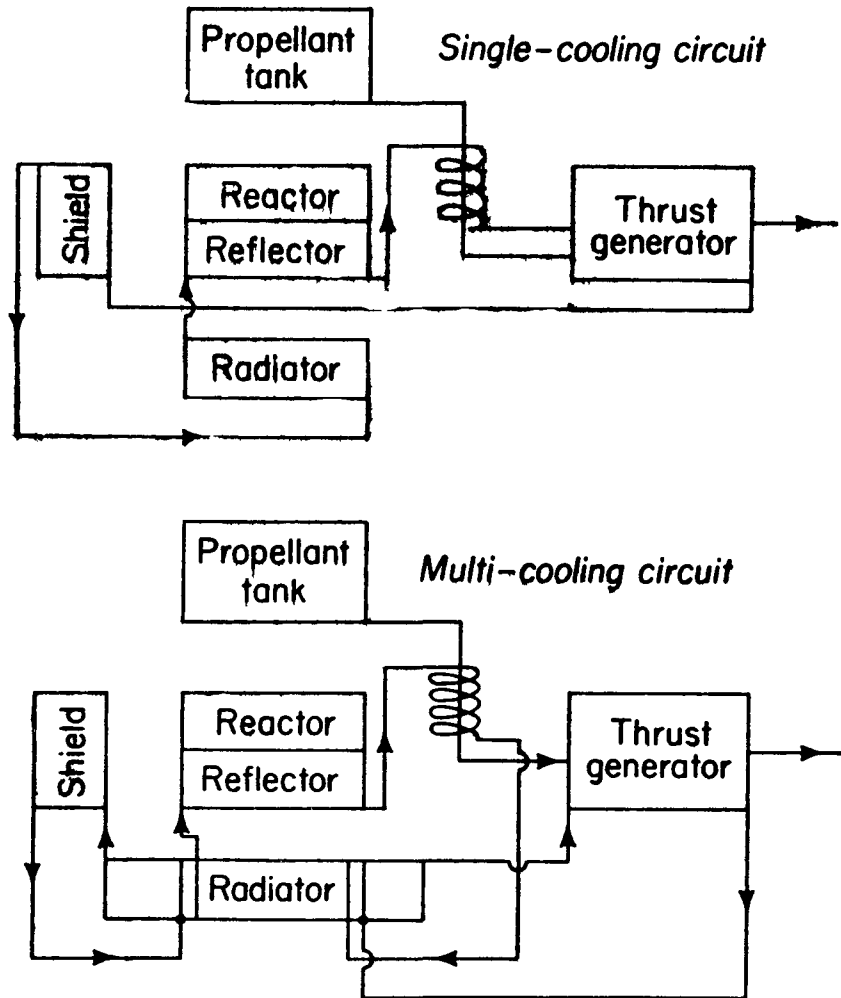


Fig. 1 Single and Multi-cooling Circuits of a Nuclear Space Propulsion System

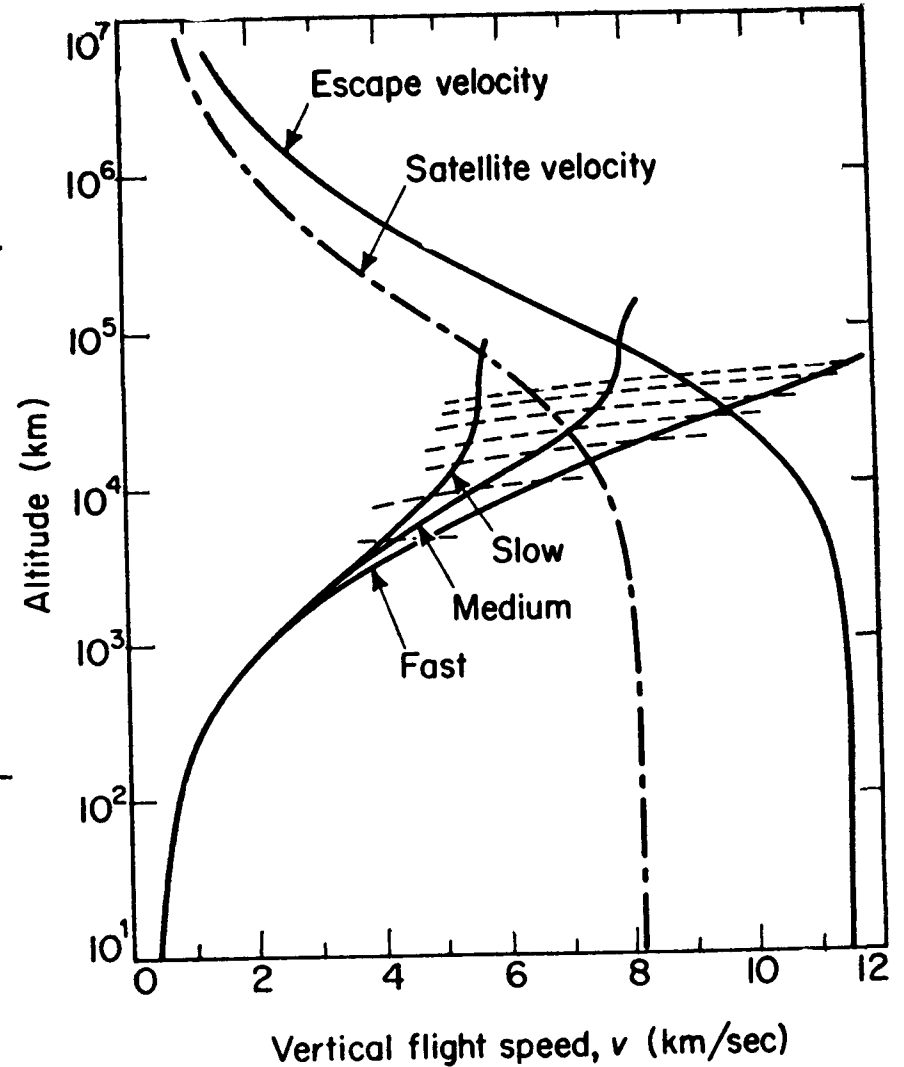


Fig. 6 Variation of Altitude with Vertical Flight Speed of Three Thermonuclearly Driven Spacecraft

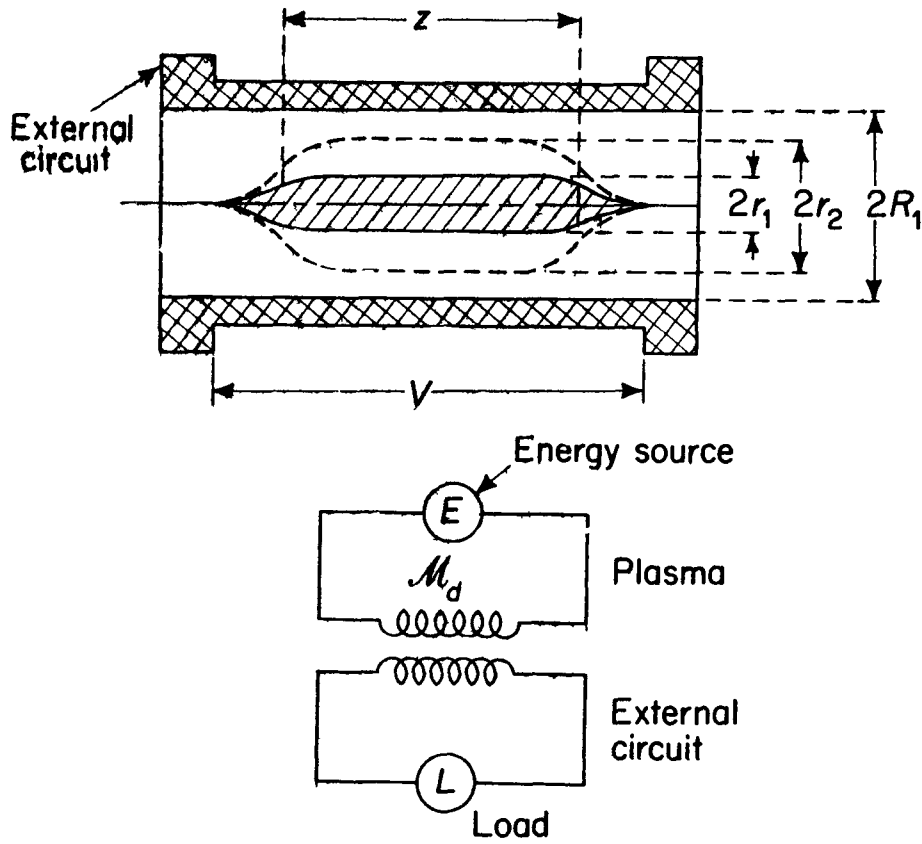


Fig. 7 Schematic Diagram Circuit Equivalent of Energy Transfer from a Plasma to Surrounding Coil

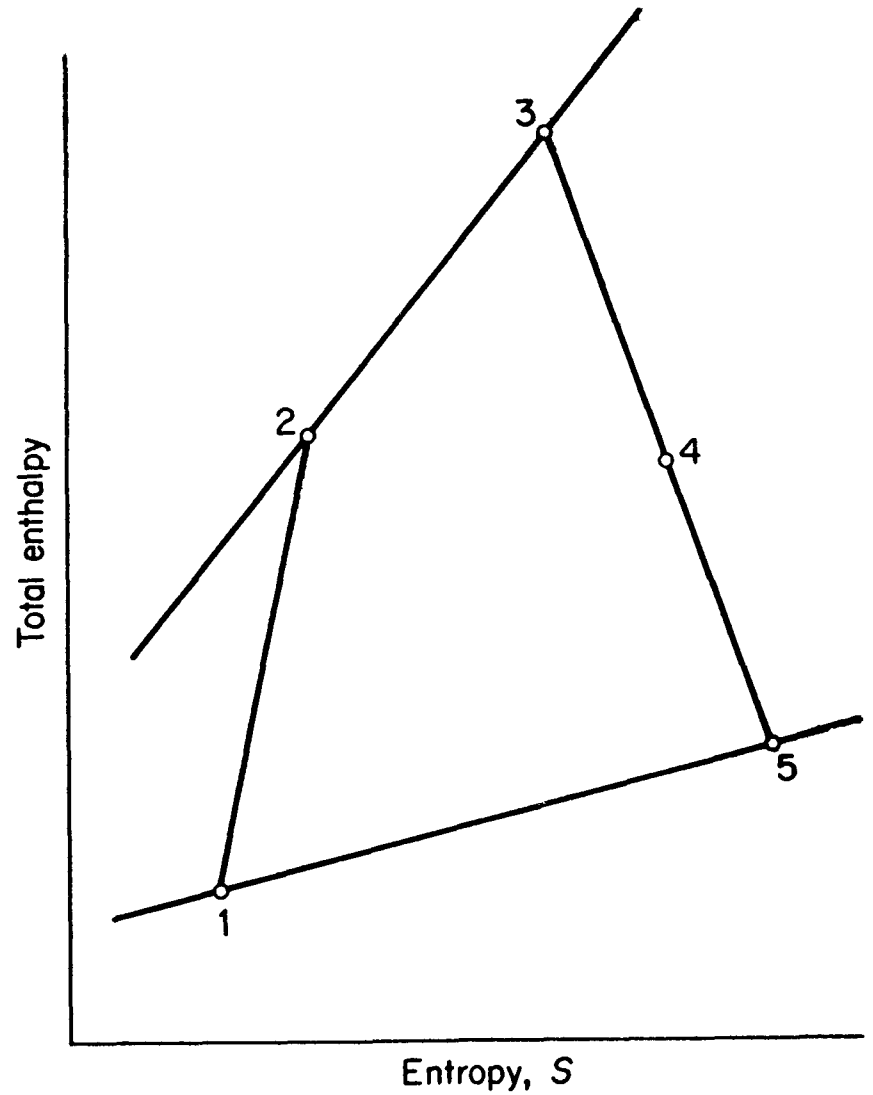


Fig.2 Thermodynamic Cycle of a Nuclear Energy Release System

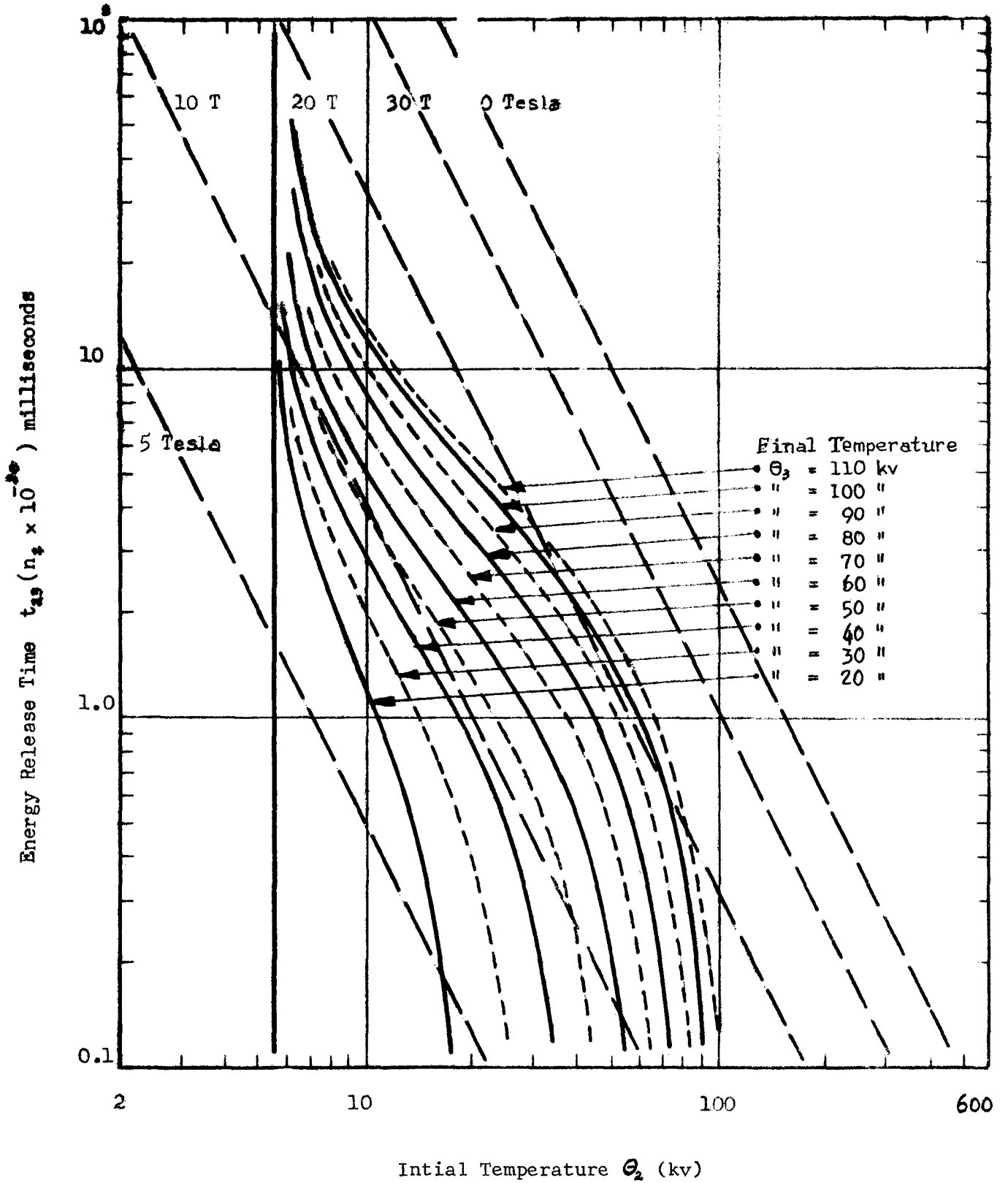


Fig.3 Variation of the Energy Release and Confinement Times with the Initial and Final Temperature and Magnetic Field for D-T



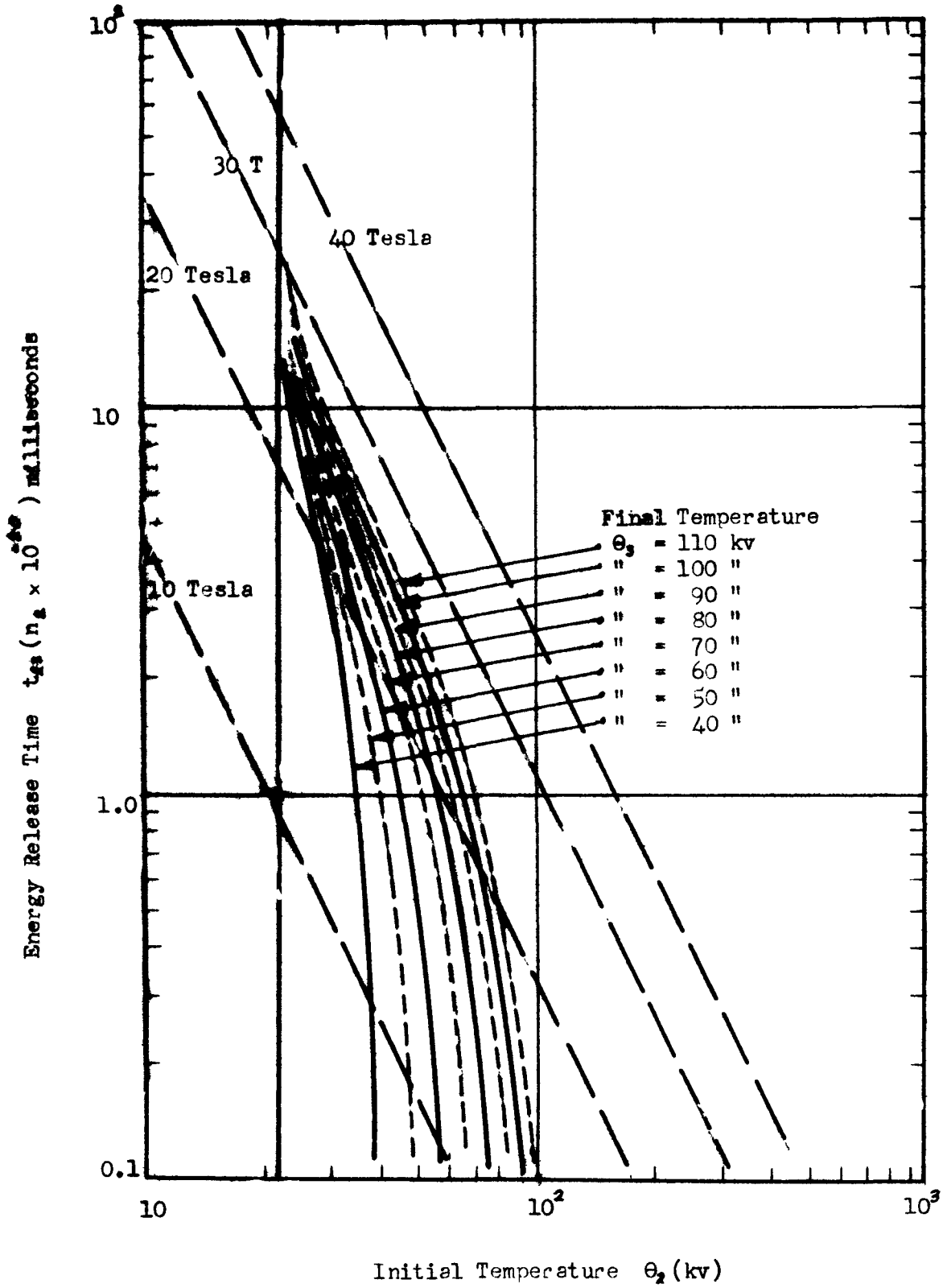


Fig.4 Variation of the Energy Release and Confinement Times with the Initial and Final Temperature and Magnetic Field for D-He<sup>3</sup>

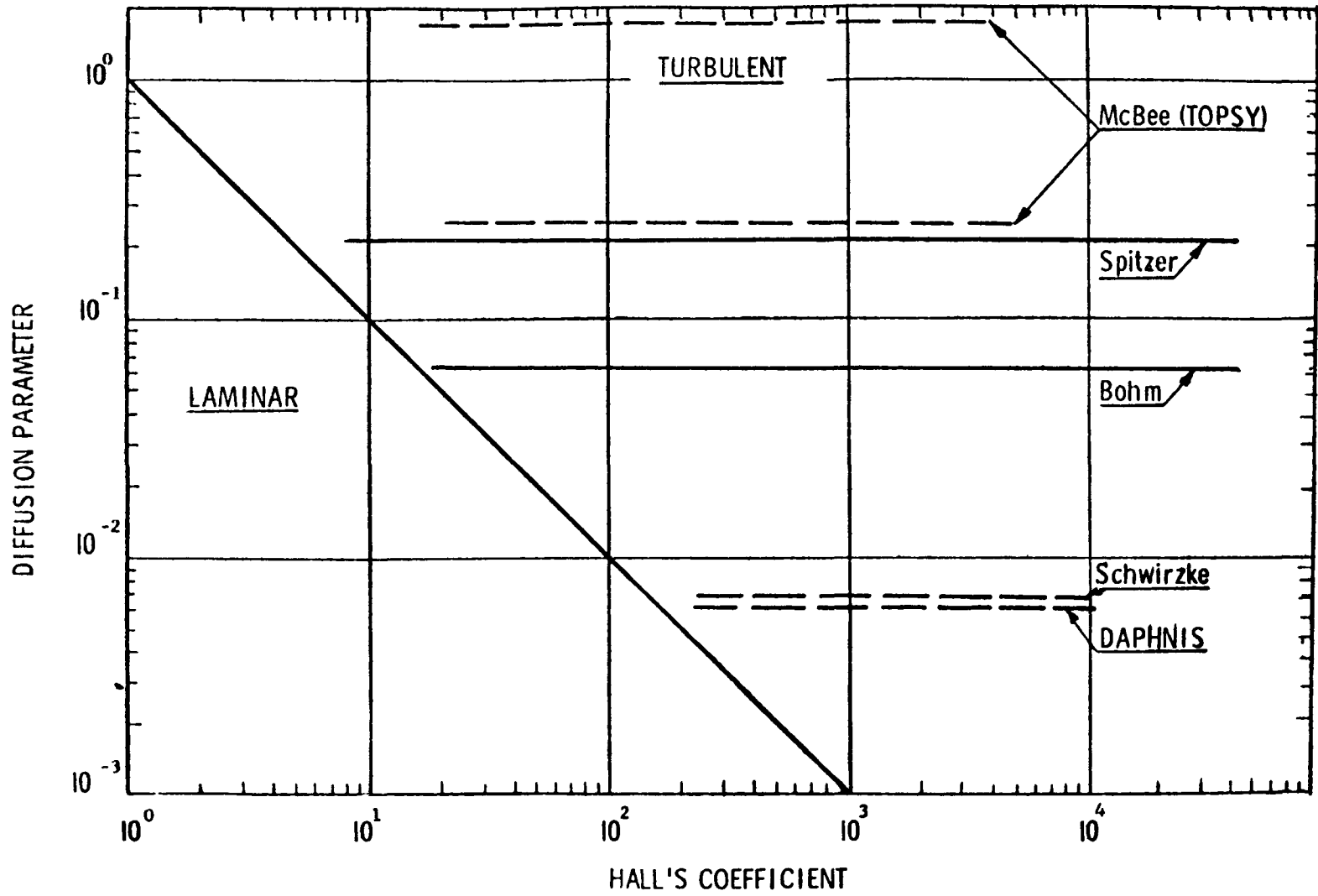


Fig. 5 Plasma diffusion through a magnetic field

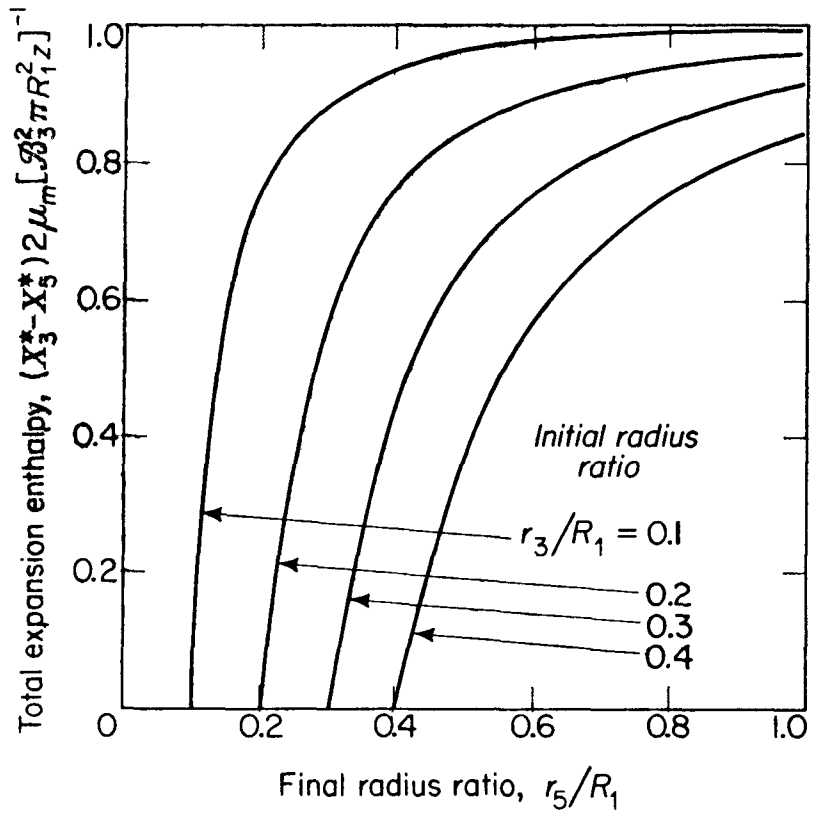


Fig. 8 Variation of the Total Two-dimensional Enthalpy During Adiabatic Radial Expansion

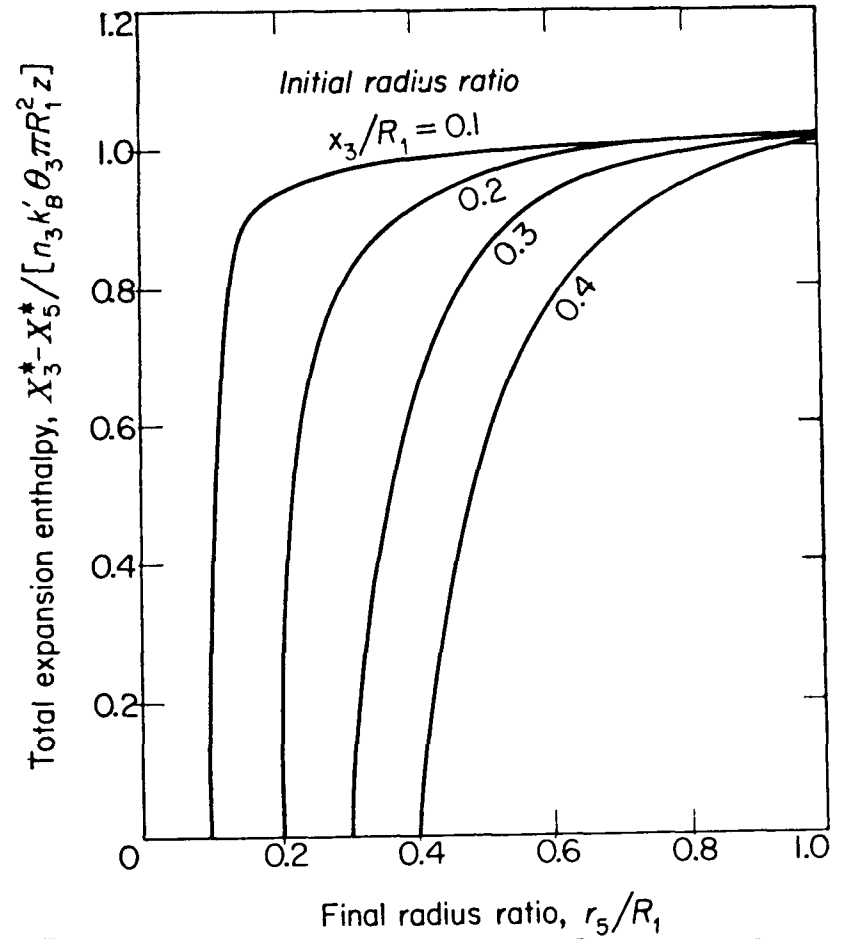


Fig. 9 Variation of the Total Three-Dimensional Enthalpy During Adiabatic Radial Expansion

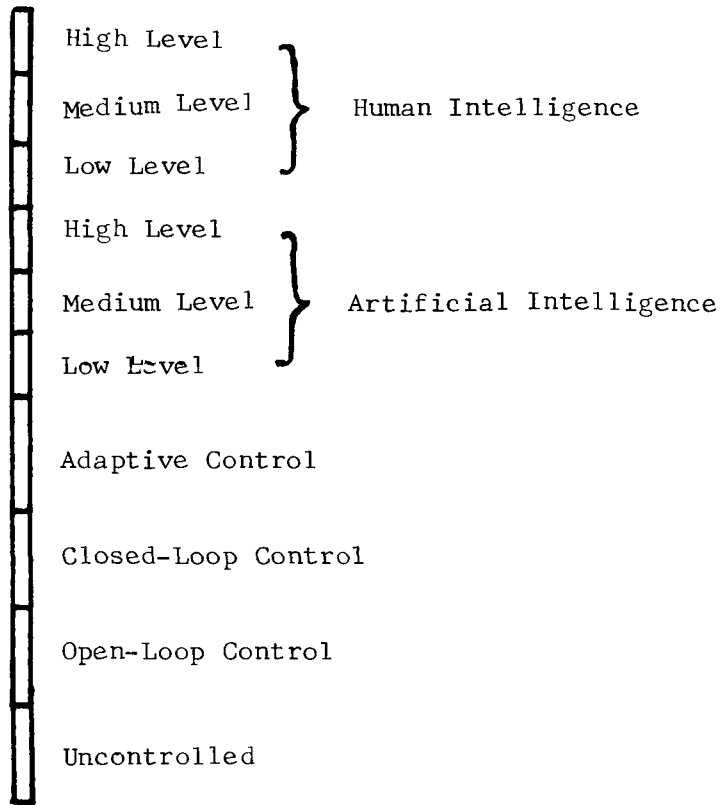


Fig.10 Classification of Control Systems. According to the Level of Intelligence in the Loop

# PROBLEMS POSED BY SPACE APPLICATIONS OF FUSION REACTORS \*

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## Summary

Some of the potential advantages of fusion propulsion systems include the following:

1. Can operate at optimum exhaust velocity
2. Gas temperature not materials limited
3. Direct conversion of plasma enthalpy to thrust or electrical power
4. Lower round trip time and initial mass for fixed payload than competing systems
5. If D-D reaction used, fuel should be inexpensive enough to exhaust directly to space

Many of the foreseeable problems associated with space and ground-based applications of fusion reactors appear to be common to the two applications, while some are unique to the respective application. A similar degree of commonality and uniqueness seems to hold for existing developmental programs leading to space and ground-based applications of fission reactors. Some of the basic physics problems most troublesome for space applications of fusion reactors include the following:

1. Must be self-sustaining: no steady-state injection
2. Must use D-D or D-He<sup>3</sup> reaction
3. Control of synchrotron radiation

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\* Abstracted from "Technological Problems Anticipated in the Application of Fusion Reactors to Space Propulsion and Power Generation", by J. R. Roth, W. D. Rayle, and J. J. Reinmann, NASA TMX 52804, presented at 5th Intersociety Energy Conversion Engineering Conference, Sept. 20-25, 1970.

4. Ions escaping from plasma must be made unidirectional
5. Mixture with propellant to form exhaust jet with optimum velocity
6. Development of a liquid metal magnetofluid

Some of the engineering problems associated with space-borne fusion reactors include the following:

1. Development of light weight components
2. Energy conversion system for thruster
3. Development of heat transfer and radiator system
4. Development of a light weight liquid helium refrigerator
5. Zero-G cryogenic heat transfer
6. System for repeated startup in space

If self-sustaining fusion reactors based on the  $D-He^3$  (or possibly the  $D-D$ ) reaction can be achieved, it is possible that fusion reactors will see their first large scale applications in space, rather than for ground-based electrical power generation. Studies have shown that fusion reactors may be marginally competitive economically with other projected power generating systems (such as advanced fast breeder reactors) on the ground, while mission analyses indicate that fusion reactors may be much superior to other competing space power and propulsion systems.

Some of the major research areas related to space applications that should receive attention are:

1. A more detailed study of the  $D-He^3$  reaction characteristics
2. The study of energy balance at the higher plasma temperatures involved

3. Systems studies of D-D and D-He<sup>3</sup> reactors, with loss of neutrons to space
4. Experimental and theoretical work on the direct conversion of the plasma enthalpy to power or thrust
5. Studies of neutron and radiation shielding methods for superconducting coil protection
6. Development of light weight, high current density superconducting magnets, light weight cryoplants, and associated systems components
7. Development of a liquid metal ferrofluid suitable for space applications of magnetocaloric pumping and power generation.

Work is also needed on methods of collecting the radially diffusing plasma from toroidal systems, and converting it into a unidirectional exhaust beam. One of the most important unknowns, the space-restart system, cannot be adequately specified until controlled fusion has been achieved.

## Fusion Power for Interstellar Flight

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### Abstract

A fusion drive can deliver an exhaust velocity of several thousand km/sec. This is much larger than the optimum exhaust velocity for any mission inside the Solar System. Thus fusion drives should be considered for interstellar flight.

An analysis is made of the reaction equilibrium and energy balance in a spatially-uniform, steady-state plasma. A particle-conservation equation is written for each of the ionic species in the plasma. For given ion temperature, deuteron number density, particle confinement time, and fuel composition, this set of equations can be solved for the plasma composition and the fuel feed rate.

Separate energy balance equations are written for the ions and for the electrons. By simultaneous solution of the two equations, the electron temperature and the rate at which power must be removed from the plasma can be determined. For given values of the energy-conversion and plasma-acceleration efficiencies, the specific thrust and the thrust per unit plasma volume can be calculated. For a given drive mass/blast power ratio, the thrust/



mass ratio can be calculated.

An optimization analysis is carried out for a two-stage starship, both stages being assumed to have the same initial acceleration, specific thrust, and drive mass/blast power, payload/initial mass, and fuel mass/tank mass ratios. If all these quantities are given except the initial acceleration, one can determine what acceleration results in minimum flight time for a given distance. Specific thrust and acceleration can also be optimized simultaneously; if no upper limits are placed on the specific thrust, this determines the capabilities of a drive that is not energy-limited, but acceleration-limited.

Numerical calculations indicate that, for an overall payload fraction of 0.01 and for drive mass/blast power ratios higher than 100 kg/mw, the optimum specific thrusts for flights to the nearer stars can be delivered by fusion drives--that is, fusion drives are acceleration-limited, not energy limited. Unless very low mass/power ratios can be achieved, the total-mass-conversion drive and the interstellar ramjet--both much more difficult technically than the fusion drive--would not give shorter flight times.

What power/mass ratios can be achieved, is quite uncertain. No attempt is made to carry out an engineering design study for a fusion drive. However, rough estimates of the masses of some of the major subsystems indicate that it will be hard to do better than

100 kg/mw. A fusion drive can burn deuterium or a mixture of deuterium and helium 3. A drive burning deuterium and helium 3 can deliver a higher specific thrust, at a lower power/mass ratio, than a drive burning pure deuterium. However, helium 3 must be synthesized. It is not possible to say at present whether the higher cost of the deuterium-helium 3 fuel combination will be justified by its performance advantage. Fission power for interstellar flight can be ruled out on the basis of fuel cost.

Unless power/mass ratios much smaller than 100 kg/mw can be achieved, not even the nearest stars can be reached with flight times less than one human lifetime. This does not preclude flights by unmanned probes.

## SYMPOSIUM SUMMARY

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Proposed fusion reactors have traditionally been classified as either steady state or pulsed. Steady state reactors would seem to permit more careful adjustment of the plasma parameters to optimum values than pulsed reactors and to permit the use of superconducting magnets to generate the confining magnetic field. Pulsed reactors place less stringent requirements on plasma stability and confinement than steady state reactors and offer a potential advantage in permitting the reactor to be refueled between pulses when there is no fusion plasma present. The equilibrium condition for steady-state reactors (the point at which the released energy remaining within the plasma is equal to that needed to heat the incoming cold fuel to the equilibrium plasma temperature) is unstable and will require a control system, which may be complex, to maintain equilibrium. Pulsed reactors based on the theta-pinch concept require the coil to be placed close to the cylindrical plasma. This configuration reduces the magnetic field volume and may thus permit substantial savings in the cost of the reactor core. A disadvantage of this system is that the coil is subjected to the 14 Mev neutrons, bremsstrahlung, and cyclotron radiation. Radiation heating, ohmic heating, and structural requirements place severe constraints on the coil design. Steady state designs must include shielding of the superconducting magnets from neutron and gamma radiation. At present, a major problem in pulsed designs is energy storage. Magnetic energy

storage appears to be the only possibility since capacitive energy storage seems to be far too costly to be economically competitive. It is not yet clear whether or not magnetic energy storage elements will be able to deliver sufficient energy at fast enough rates.

Requirements for fuel injection into a DT fusion reactor depend upon  $\alpha$ -particle heating rates and on whether the reactor is pulsed or steady state. Some calculations indicate that it may be difficult to fill the reactor volume of a steady state reactor with fuel because of the difficulty in penetrating a fusion plasma with beams or pellets. Pulsed reactors permit refueling in the off period, but consequently also require higher peak currents from the injectors. In a steady state reactor with no  $\alpha$ -particle heating, the Lawson criterion was used to estimate the maximum current required to maintain a steady state 1500 MW(th) reactor, with a fractional burnup of 5%, as about 1000 amperes. With foreseeable beam technology, this current will be difficult to provide economically. Furthermore, the wall area required by the injectors may be large enough to affect the neutron economy.

Several important advances in blanket design have been reported. For example, it now appears that a simple natural liquid lithium blanket will provide adequate tritium breeding without enrichment. Second, a new modular design permits the vacuum wall to be located outside the blanket. This configuration relaxes the structural requirements which arise when the vacuum wall is also the first wall. This design also uses modified heat pipes both to equalize the radial temperature distribution and to separate out tritium which has been generated in the lithium blanket.

An interesting proposal [Lidsky, MIT] for utilizing fusion neutrons, that was only mentioned briefly at the symposium, is to produce fissile  $U^{233}$  or  $Pu^{239}$  by capturing excess fusion neutrons in  $Th^{232}$  or  $U^{238}$  placed in the blanket of a fusion reactor. This fissile fuel can then be used in a converter fission reactor to give a much shorter fuel doubling time than fission breeder reactors. The neutron rich but power balance poor DT fusion reactor and the neutron poor but power rich fission converter reactor could make a very attractive couple.

Direct conversion of fusion reactor charged particle energy to electrical energy by using a series of electrostatically focused collector electrodes, rectifiers, and inverters provides a means of achieving overall conversion efficiencies of more than 90%. For a 1000 MW system, the direct converter should cost approximately \$20/kW. This cost compares with \$70-\$80/kW for conventional conversion systems. This energy conversion scheme is particularly suited for mirror type reactors where it increases the "usual" Q by an order of magnitude.

Considerable interest in the use of fusion reactors for space power and space propulsion systems was evident among several of the symposium participants. As in the case of using fusion reactors for electric power generation, there was disagreement, among those who attended the special session on Space Applications of Fusion Reactors, concerning the relative merits of pulsed and steady state reactors for space applications. The fact that agreement could not be reached simply indicates that neither pulsed nor steady state reactors can be shown to be inherently unsuited for space applications at the

present stage of fusion reactor design and technology. However, several factors are especially important for space reactors regardless of whether they are steady state or pulsed. For example, since the isotropic neutron distribution from a D-T reactor is not directly useful for propulsion or heating the plasma and conversion will require components with considerable mass, reactors in which the fusion energy appears in charged particles (D-He<sup>3</sup> or P-Li<sup>6</sup>) are especially attractive for space applications. These cycles require higher ignition and operating temperatures, of course, and thus lead to other problems. Mass considerations also seem to require that space borne reactors be self sustaining so that massive supplementary fuel heating systems are not required. This restriction requires the charged reaction products to be capable of heating the fuel to temperatures high enough to produce at least one fusion reaction. In a D-T reactor, such a reaction would have to be sustained by the relatively small amount of fusion energy which is not carried away by the neutrons, that is, by  $\alpha$ -particle heating. In D-He<sup>3</sup> and p-Li<sup>6</sup> reactors, most of the fusion energy is given off in charged reaction products which can be confined and used to heat the injected fuel as well as to supply power. Provided that the ignition temperatures and adequate confinement can be achieved, therefore, the self sustaining requirement could be more easily satisfied in D-He<sup>3</sup> or p-Li<sup>6</sup> reactors than in D-T reactors.

Quite a lot of discussion centered on environmental, economic, and social considerations, which are all, of course, interrelated. Fusion reactors are expected to introduce relatively few environmental problems. In D-T systems, for example, no radioactive wastes are

formed, and tritium, which is radioactive, is less hazardous than the plutonium in fission reactors and should be easier to contain than the noble fission gases. Moreover, fusion reactors cannot run away and also have low after heat.

Most of the fusion reactor design studies to date have been based on the D-T fuel cycle since it has the lowest ignition requirements. It has also usually been assumed that at least a large part of the tritium breeding blanket is lithium. The world reserves of lithium are not well known, because of present lack of demand, but estimates were given showing that, from an energy viewpoint, they are comparable to fast fission breeder fuels, U and Th. One of the arguments for fusion reactors has been the "unlimited" amounts of fuel. A fusion reactor based on the D-T fuel cycle may be limited by the available amounts of lithium, however. This is then also an incentive for considering other fuel cycles. Estimates were also made to show that electric power from fast fission breeder and fusion reactors are comparable in cost.

Economic calculations concerning fusion reactors have traditionally used the projected costs of producing electrical power by fast fission breeder reactors as a target cost which fusion reactors must meet. The rationale for this approach is that the design details of fusion reactors are not yet specified in enough detail to permit accurate cost estimates, while the cost of fusion reactors' major competitor, fast fission breeder reactors, are known well enough to give a reasonably accurate target cost which must be met in order to be competitive. Although these estimates seem to show that there is no reason why fusion reactors cannot compete economically with fast

fission breeder reactors in producing cheap electric power, there are at least two factors which may make these analyses less favorable to fusion reactors than they should be. First, the fast fission breeder reactor cost estimates used for fusion reactor target costs do not include an effective cost for actual and potential hazards and environmental pollution. Of course it is presently impossible to quantify these costs; however, it is certainly true that these costs will be more important in coming years and that the effective fusion reactor costs will be less than the corresponding costs for fast fission breeders. Second, the use of projected fast fission breeder costs as a target cost assumes that the major part of the produced power will be converted to electricity. In fact, as the population increases, it seems that more and more energy will be required for recycling waste materials for re-use, for desalinating seawater and for other bulk processes. It is not at all clear that electrical energy will be the most appropriate form for these processes. These processes may require sufficient energy at one location to permit more direct utilization of energy from the fusion plasma. (The main advantage of converting energy to electrical form is the ease with which electrical energy can be transported). It may be possible to utilize the fusion plasma in a Fusion Torch to vaporize and ionize wastes and permit them to be separated into useful raw materials. The Fusion Torch concept might also be applied by injecting high atomic number impurities into the plasma to produce ultraviolet radiation which would be used to bulk heat liquids for chemical processing and disalination of water. If these applications actually develop and the Fusion Torch is even



partly feasible, fusion reactors could offer significant advantages over fast fission breeder reactors in this area.

Very little study has been given to the social implications of the introduction of fusion power, although work is beginning, as evidenced by the attendance of sociologists and economists during parts of the symposium. Understanding of the social effects, "good" or "bad", will be important for considering factors in the design, location, and use of the fusion reactors by those directly concerned with fusion reactor design and planning and for evaluation of fusion reactors by the society at large.

It was well documented at the symposium that the fast fission breeder reactor development is progressing well and that they probably will be "on line" within twenty years. With the current rate of world wide funding it does not appear that one will have an "on line" fusion reactor in this century.

SELECTED BIBLIOGRAPHY FOR  
FUSION REACTOR DESIGN STUDIES

The following bibliography is not complete but contains some of the most pertinent references for anyone interested in pursuing research in this area. Many of the references were provided by W. C. Gough, USAEC. It will be seen that the reference format is not uniform all through the list. This was not found important enough to take the time to change it. The references are also not listed in any particular order.

D. A. Baker and L. W. Mann, "Curves for Predicting Stresses in Controlled Fusion Pulsed Magnetic Field Systems," Report No. LA-4186; Los Alamos Scientific Laboratory, Los Alamos, New Mexico, May 23, 1969.

C. J. Barton and R. A. Strehlow, "Blankets for R. A. Strehlow, "Blankets for Thermonuclear Reactors," Oak Ridge Nat. Lab., Oak Ridge, Tennessee, ORNL-3258, July 1962.

G. I. Bell, "Neutron Blanket Calculations for Thermonuclear Reactors," Los Alamos Sc. Lab., Los Alamos, New Mexico, LA-3385-MS, September 20, 1965.

George I. Bell, "Neutron Blanket Calculations for Thermonuclear Reactors, II," Report No. LA-4131-MS; Los Alamos Scientific Laboratory, Los Alamos, New Mexico, March 11, 1969.

\* G. I. Bell, W. H. Borkenhagen and F. L. Ribe, "Feasibility Studies of Pulsed, High- $\beta$  Fusion Reactors," Report No. LA-DC-10618; Los Alamos Scientific Laboratory; Los Alamos, New Mexico, September 1969.

R. Carruthers, "2000 MW Fusion Reactors Forecast," "Electrical Review," UKAEA Research Group, Culham Laboratory, Abingdon, Berkshire, U.K., October 3, 1969.

\* R. Carruthers, P. A. Davenport, and J. T. D. Mitchell, "The Economic Generation of Power from Thermonuclear Fusion," Culham Report R. 85., Culham Laboratory, Abingdon, Berkshire, U.K., Oct. 1967.

R. A. Dugdale, "Thermal Shock and Thermal Stress Fatigue in a Stabilized Pinch Fusion Reactor," AERE R. 2955, June 1959.

\* B. J. Eastlund and W. C. Gough, "Closing the Cycle From Use to Reuse," Report No. WASH-1132; Division of Research, United States Atomic Energy Commission, Washington, D. C., May 15, 1969.

\* B. J. Eastlund and W. C. Gough, "The Fusion Torch: A New Approach to Pollution and Energy Usage," Division of Research, U.S. Atomic Energy Commission, Washington, D. C., November 7, 1969.

B. J. Eastlund and W. C. Gough, "The Fusion Torch - Unique Applications of Ultra-High Temperature Plasmas," Division of Research, U. S. Atomic Energy Commission, Washington, D. C. 20545, August 18, 1969.

W. G. Homeyer, "Thermal and Chemical Aspects of the Thermonuclear Blanket Problem," Technical Report No. 435; Research Laboratory of Electronics, Massachusetts Institute of Technology, Cambridge, Massachusetts, June 29, 1965.

D. G. Jacobs, "Sources of Tritium and its Behavior Upon Release to the Environment," Report No. TID-24635; Nuclear Safety Information Center, Oak Ridge Nat. Lab., Oak Ridge, Tennessee, December 1968.

E. F. Johnson, "Tritium Recovery from Fusion Reactor Blankets," Report No. MATT-688; Plasma Physics Laboratory, Princeton University, Princeton, New Jersey.

O. Kofoed-Hansen, "Introduction to Controlled Thermonuclear Research," Riso Research Est., Riso, Denmark, Report No. 18 (Summer School) 1960.

J. G. Linhart, "Power from Thermonuclear Reactions," Nuclear Engineering, Feb. 1967, pp. 60-65.

L. N. Lontai, "Study of a Thermonuclear Reactor Blanket with Fissile Nuclides," Technical Report No. 436; Research Laboratory of Electronics, Massachusetts Institute of Technology, Cambridge, Massachusetts, July 6, 1965.

G. M. McCracken and J. H. C. Maple, "The Trapping of Hydrogen Ions in Molybdenum, Tantalum, Titanium and Zirconium," Culham Preprint p. 118, Culham Laboratory, Abingdon, Berkshire, U.K., Aug. 1966.

G. M. McCracken and D. K. Jefferies, "The Temperature Dependence of Hydrogen and Deuterium Ion Trapping in Reactive Metals," 5th Symposium of Fusion Technology, Paper No. 49, July 1968.

R. G. Mills, "Four Lectures on Thermonuclear Power," Plasma Physics Lab., Princeton University, Princeton, N. J., MATT 145, Sept. 1962.

R. G. Mills, "Some Engineering Problems of Thermonuclear Reactors," Nuclear Fusion, 7, No. 4, pp. 223-236, Dec. 1967.

R. G. Mills, "The Problem of Control of Thermonuclear Reactors," Report No. LA-4250; Los Alamos Sci. Lab., Los Alamos, N. M., April 8-11, 1969.

R. G. Mills, "A cursory Look at Tokamak Fusion Reactors," MATT-659; Plasma Physics Laboratory, Princeton University, Princeton, New Jersey, December, 1968.

A. J. Impink, Jr., "Neutron Economy in Fusion Reactor Blanket Assemblies," Technical Report 434; Research Laboratory of Electronics, Massachusetts Institute of Technology, Cambridge, Massachusetts, June 22, 1965.

- R. F. Post, "Impurity Radiation Losses from a High Temperature Plasma," Riso Report No. 18, p. 313, Riso, Denmark, 1960.
- Ribe, Gilmer, Hoyt, G. Hain and K. Hain, "Comparison of Computed and Measured Behavior of Fast Theta Pinches," LAMS-2911, Los Alamos Scientific Lab., Los Alamos, N. M., May 1963.
- Anon., "Plasmas and Magnetic Fields in Propulsion and Power Research," SP-226, 1970, NASA Washington, D. C.
- Kelley, Lazar, et al, "High Current, 600 kv Accelerator for DCX-Type Injection," Am. Phys. Soc. Mtg., 11/2-4/60, Tenn.
- Mackin, "Critical Parameters for Burnout in a Device Fed by Energetic Ions," Am. Phys. Soc. Mtg., 11/2-4/60, Tenn.
- Morgan, Kelley, et al, "Mass Analysis at Large Ion Currents from the Duo-Plasmatron Ion Source," Am. Phys. Soc. Mtg., 11/2-4/60, Tenn.
- Neufeld and Doyle, "Electromagnetic Interaction of a Beam of Charged Particles with Plasma," Am. Phys. Soc. Mtg., 11/2-4/60, Tenn.
- Simon, "High Energy Injection," International Course in Plasma Physics, Riso, Denmark, 8/1-12/60.
- Gauster and Potts, "Some Engineering Problems in the DCX Approach to Fusion Research," Symposium on Engineering Problems in Thermonuclear Research, England, Sept. 6-8, 1960.
- McNally, "Fusion Chain Reactions," Semi-Annual Progress Report, ORNL-3760, p. 98, 1964, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- Members of the Project Matterhorn Staff, "The Proposed Model C Stellarator Facility," NYO-7899, Plasma Physics Laboratory, Princeton University, Princeton, N. J., Aug. 1957.
- Mills, "Power Production by Stellarators," Princeton Tech. Memo No. 770, Plasma Physics Laboratory, Princeton University, Princeton, N. J., April 1959.
- Mills, "Fusion Power from Magnetically Confined Plasma," MATT-34, Plasma Physics Laboratory, Princeton University, Princeton, N. J., March 1960.
- Mills, "Economic Prospects for Thermonuclear Reactors," MATT-60, Plasma Physics Laboratory, Princeton University, Princeton, N. J., Feb. 1961.
- Mills, "Energy Storage by Superconducting Devices," Princeton Tech. Memo. No. 115, Plasma Physics Laboratory, Princeton University, Princeton, N. J., (1960).
- D. J. Rose, "Basic Concepts for a Steady-State Fusion Reactor," Trans. Am. Nuclear Society, 5, No. 1, p. 98, 1962.
- D. J. Rose, et al., "Engineering Calculations for Barely Conceivable Reactors," 50 pages, unpublished class notes of Spring, 1962, MIT, Cambridge, Mass.

- W. M. Wells, "Experiments and Calculations on the Feasibility of Pumping Liquid Lithium in a Thermonuclear Reactor Blanket," Report No. UCRL-50544; Lawrence Radiation Laboratory, University of California, Livermore, California, February 3, 1969.
- L. M. Petrie, Jr., "Gamma-Ray Spectra in Fusion Blanket Mock-ups," Technical Report 438; Research Laboratory of Electronics, Massachusetts Institute of Technology, Cambridge, Massachusetts, July 20, 1965.
- J. E. Powers and F. L. Ribe, "Computer Program for Solution of the Energy Balance and Heat Transfer of a Pulsed Thermonuclear Reactor," Report No. LA-3347-MS; Los Alamos Scientific Laboratory, Los Alamos, New Mexico, June 1, 1965.
- \* F. L. Ribe, "Pulsed Systems," Report No. LA-DC-11045; Los Alamos Scientific Laboratory, Los Alamos, New Mexico, September, 1969.
- F. L. Ribe, "Effects of Gamma-Ray Heating Estimates on LASL Pulsed-Reactor Model," Report No. LA-DC-10552; Los Alamos Scientific Laboratory; Los Alamos, New Mexico, May 23, 1969.
- \* F. L. Ribe, T. A. Oliphant, Jr., and W. E. Quinn, "Feasibility Study of a Pulsed Thermonuclear Reactor," Report No. LA-3294-MS; Los Alamos Scientific Laboratory; Los Alamos, New Mexico, April 23, 1965.
- D. J. Rose, "On the Feasibility of Power by Nuclear Fusion," ORNL-TM-2204, Oak Ridge Nat. Lab., Oak Ridge, Tennessee, March 1968.
- D. J. Rose, "Hazards of Tritium From Controlled Fusion," Oak Ridge National Laboratory, Oak Ridge, Tennessee, 37830.
- K. Simonyi and M. Uzsoy, "About the Star-Like Fusion Reactor," Periodica Polytechnica (Budapest), Electrical Engineering, 1, no. 3, pp. 243-257, 1957.
- P. S. Spangler, "Fusion Reactor Blanket Experiment," Technical Report 437; Research Laboratory of Electronics, Massachusetts Institute of Technology, Cambridge, Massachusetts, July 13, 1965.
- L. Spitzer, D. J. Grove, W. E. Johnson, S. L. Tonk, and W. F. Westendorp, "Problems of the Stellarators as a Useful Power Source," NYO 6047, Plasma Physics Laboratory, Princeton University, Princeton, N. J., Aug. 1954.
- R. A. Strehlow and D. M. Richardson, "Chemistry of Tritium in Controlled Fusion Devices," Oak Ridge Nat. Lab., ORNL 3836, Semi Annual Progress Report to April 1965, Section 9, 6p., Oak Ridge, Tennessee.
- P. C. Thonemann, G. Francis, J. D. Jukes, D. W. Mason, B. McNamara, D. R. Sweetman, J. B. Taylor, C. J. H. Watson, F. A. Julian, "Thermonuclear Reactors based on Mirror Machine Confinement," A report from the Culham Mirror Study Group, 1967-68, Culham Report CLM-R 94, 1969, Culham Laboratory, Abingdon, Berkshire, U.K.
- F. L. Parker, "Radioactive Wastes from Fusion Reactors," Science, 159, 83; January 11, 1968.

S. H. Maslen, "Fusion for Space Propulsion," IRE Transactions on Military Electronics, Vol. MIL-3, No. 2, pp. 52-57, April 1966.

W. E. Moeckel, "Propulsion Systems for Manned Exploration of the Solar Systems," Astronautics and Aeronautics, Vol. 7, No. 8, pp. 66-67, Aug. 1969.

J. R. Roth, "A Preliminary Study of Thermonuclear Rocket Propulsion," Journal of the British Interplanetary Society, Vol. 18, Pt. 3, pp. 99-180, June 1961.

D. J. Rose, "Engineering Feasibility of Controlled Fusion - A Review," Nuclear Fusion, Vol. 9, pp. 183-203, 1969.

A. P. Fraas, "A Diffusion Process for Removing Tritium from the Blanket of a Thermonuclear Reactor," ORNL-TM-2358, Dec. 1968, Oak Ridge National Lab., Oak Ridge, Tennessee

W. R. Chambers, A. P. Fraas, M. N. Ozisik, "A Potassium-steam Binary Vapor Cycle for Nuclear Power Plants," ORNL-3584, May 1964, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

N. C. Christofilos, "General Parameters for a 500-MW Astron Power Reactor," UCRL-5095, Lawrence Radiation Laboratory, University of California, Jan. 22, 1958.

N. C. Christofilos, Cook, Myers, Taylor and Wells, "Engineering Study of an Astron Power Reactor," UCRL-5115 and TID-7558 (Suppl. 1), Lawrence Radiation Laboratory, University of California, Feb. 1958.

N. C. Christofilos, "Energy Balance in the Astron," UCRL-6390, Lawrence Radiation Laboratory, University of California, June 15, 1961.

R. F. Post, "Some Aspects of the Economics of Fusion Reactors," UCRL-6077, Lawrence Radiation Laboratory, University of California, August 1960.

R. F. Post, "Fast Neutral Particle Injection into a Mirror Machine," Lawrence Radiation Laboratory, University of California, UCRL-6078, (1960).

R. F. Post, "Critical Conditions for Self-Sustaining Reactions in the Mirror Machine," Salzburg Conf. Paper CN-10/153, 1961; or UCRL-6377; Lawrence Radiation Laboratory, University of California, Nuclear Fusion, 1962 Supplement, Part 1, p. 99.

I. N. Golovin and V. I. Pistunovich, "The Production of Thermonuclear Plasma by the Injection of Fast Particles into a Magnetic Trap," Kurchatov Inst., Moscow, Report IAE 676, Moscow, 1964.

R. Hancox, "Wall Materials for Thermonuclear Devices Limitations due to Evaporation," AERE R. 2957, Aug. 1959.

J. L. Hilton, "Economics of a Hypothetical He<sup>3</sup>-D Fusion Reactor for Central Station Power," IEEE Transactions on Nuclear Science, Vol. 16, No. 4, pp. 20-28, August, 1969.

Berman, "Impurity Radiant Energy Loss," Space Technology Laboratories Report STL/TR-60-0000-GR 435 (1960).

Geller, Kupp and Koslov, "Engineering Aspects of a Thermonuclear Power Plant," Vitro Engineering Company, unpublished, 1958.

California Research and Development Corp., "Proposal for a Driven Fusion Reactor," submitted to the Division of Research, USAEC, Unpublished, 1957.

Hurwitz, "Electrical Power Problems in Fusion Research," GP-96, General Electric Co., Sept. 1958.

BenDaniel, "A Brief Discussion of Some Factors Related to the Economics of Steady-State Deuterium Mirror Machine Operation," 61-RL-(2725E), General Electric Co., May 1961.

H. M. Mott-Smith, Characteristics of Thermonuclear Plasmas, Gibbs & Cox, Inc., New York, N.Y., Unpublished.

Fry, "A Review of Work Towards Nuclear Energy from Controlled Thermonuclear Reaction," Paper No. 2903, I.E.E. Convention, April, 1959.

Jukes, "Possibilities of Direct Energy Conversion from Fusion Reactors," Paper No. 2902, I.E.E. Convention, April 1959.

Carruthers, "Energy Storage for Thermonuclear Research," Paper No. 2915, I.E.E. Convention, April 1959.

Lawson, "Some Criteria for a Useful Thermonuclear Reactor," AERE-GP/R-1807, Dec. 1955, and Proc. Phys. Soc. London, B, 70, 6 (1957).

Thompson, "Thermonuclear Reaction Rates," Proc. Phys. Soc., London, B, 70, 1 (1957).

Liley, "The Circuit Dynamics of Plasmas," Proc. of IEE, London, A, 106 (1959).

Robson, "Choice of Materials and Problems of Design of Heavy-Current Toroidal Discharge Tubes," Proc. of IEEE, London, A, 106 (1959).

Hesselberg-Jensen, Kofoed-Hansen, Sillesen and Wandel, "Some Criteria for a Self-Sustaining Steady State Thermonuclear Reaction," Riso Report No. 2, May 4, 1958, Riso, Denmark.

Hesselberg-Jensen, Kofoed-Hansen and Wandel, "A Survey of the Energy Balance in a Thermonuclear Reacting Plasma, Containing Deuterium, Tritium and Reaction Products under Isothermal Pulsed or Steady-State Conditions," A Riso preprint, July 18, 1958, Riso, Denmark.

Kofoed-Hansen, Hesselberg-Jensen, Jensen, Sillesen, Vinther and Wandel. "A Compilation of Cross Sections and Rates of Interest in Controlled Thermonuclear Research," A Riso preprint, November 1958, Riso, Denmark.

Wandel, "Energy Exchange by Coulomb Encounters for Maxwellian Ion Mixtures in the Plasma State," Riso Report No. 4, January 1959, Riso, Denmark.

Hesselberg-Jensen and Vinther, "On the Effects of a Steady State Thermonuclear Reaction of Finite Energy Transfer Rates between the Charged Particles," Riso Report No. 5, March 1959, Riso, Denmark.

Hesselberg-Jensen, "On the Optimization of the Tritium Enrichment with Respect to the Energy Balance in Steady State Thermonuclear Reactions," Riso Report No. 6, August 1958, Riso, Denmark.

Kofoed-Hansen and Wandel, "On the Energy Balance and Reaction Kinetics for Isothermal Thermonuclear Reactions Involving Deuterium and Tritium," Riso Report No. 7, December 1958, Riso, Denmark.

Kofoed-Hansen, "On the Self-Heating of a Plasma by Thermonuclear Reactions," Riso Report No. 15, May 1960, Riso, Denmark.

Mitsuo Imoto, "Essential Limitations and T-Nt Diagram for Self-Sustaining Fusion Systems," J. Nuc. Energy, Part C, 1, 19 (1959).

D. M. Gruen and S. Fried, "Fused Salts as Thermonuclear Reactor Breeder Blankets," ANL-5840, March, 1958, Argonne Nat. Lab., Argonne, Ill.

Hinterman and Wideroe, "Tritium Production and Cycling in a Fusion Reactor with Lithium Blanket," Proc. Second Int. Conf. on the Peaceful Uses of Atomic Energy; Vol. 32, 440 (1958).

L. G. Barrett, "A Fusion-Fission Reactor," General Electric Company, June 27, 1957.

W. B. Myers, W. M. Wells, and E. H. Canfield, "Tritium Regeneration in a D-T Thermonuclear Reactor Blanket," LRL UCID-4480, May 22, 1962, Lawrence Radiation Laboratory, University of California.

H. M. Agnew, "(n,2n) Cross Sections of Be<sup>9</sup>," LA-1371, Feb. 19, 1952, Los Alamos Scientific Lab., Los Alamos, N.M.

Rosen and Stewart, "Neutron Emission Probabilities from the Interaction of 14-Mev Neutrons with Be, Ta, Bi, and U<sup>238</sup>," LA-2111, Jan. 1957, Los Alamos Sci. Lab., Los Alamos, N.M., Jan. 1957.

Rosen and Stewart, "The Neutron-Induced Disintegration of Li<sup>6</sup> and Li<sup>7</sup> by 5 - to 14-Mev Incident Neutrons," LA-2643, Written Jan. 1956, distributed Dec. 1961.

Dunn, "Plasma Feasibility of a Pulsed Fusion Device," Unpublished MIT thesis, 1965.

Caswell, Gabbard, Padgett and Doering, "Attenuation of 14.1 MEV Neutrons in Water," Nuc. Sci. and Eng. 2, 143 (1957).

R. E. Clausing, "Some Thoughts Concerning Materials Problems for Sherwood Devices," ORNL-CF-58-1-33, Dec. 13, 1957, Revised Jan. 8, 1958, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

Mark T. Robinson, "On the Chemistry of Thermonuclear Reactor Breeder Blankets," ORNL-CF-57-12-39, Feb. 7, 1958, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.



W. Davis, Jr., "Some Chemical Problems Associated with Thermonuclear Reactors," ORNL-CF-58-11-95, November 10, 1958, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

Barton and Strehlow, "Blankets for Thermonuclear Reactors," ORNL-3258, July 12, 1962, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

W. E. Kinney, "A Calculation of the H<sub>3</sub> Breeding Ratio in a Stellarator Blanket," ORNL-CF-61-6-63, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

E. F. Johnson, "Recovery of Tritium from Dilute Solutions of Lithium Tritide in Lithium," NYO-6370 or Tech. Memo. No. 25, Undated, Plasma Phys. Lab., Princeton Univ., Princeton, N. J.

E. F. Johnson, "Appraisal of Possible Stellarator Blanket Systems," NYO-7900 or PM-S-30, September 20, 1957, Plasma Phys. Lab., Princeton Univ., Princeton, N. J.

Camp and Johnson, "The Effect of Strong Magnetic Fields on Chemical Engineering Systems," MATT-67, March, 1961, Plasma Phys. Lab., Princeton Univ., Princeton, N. J.

Gastwirt and Johnson, "The Thermal Decomposition of Lithium Nitrate," MATT-98, Plasma Phys. Lab., Princeton Univ., Princeton, N. J.

Johnson and E. F. Pardee III, "Electrical Conductivity of Fused Salts," MATT-99, August 1961, Plasma Phys. Lab., Princeton Univ., Princeton, N.J.

F. W. Camp and E. F. Johnson, "Determination of Diffusivity and Solubility in Two Phase Systems," MATT-115 and MATT-148, March 1962 and September 1962, Plasma Phys. Lab., Princeton Univ., Princeton, N.J.

Camp and Johnson, "Magnetic Effects in Certain Systems of Chemical Engineering Interest," MATT-235, September, 1963, Plasma Phys. Lab., Princeton Univ., Princeton, N. J.

E. F. Johnson, "Thermonuclear Power Reactors," Chem. Eng. Progress, 58, 62, August 1962. (A semi-popular treatment of blanket problems.)

Christofilos, Cook, Myers, Taylor and W. M. Wells, "Engineering Study of an Astron Power Reactor," TID-7558 (Suppl. 1), p. 18-30 of Report, Controlled Thermonuclear Conference, Washington, Feb., 1958.

TID-7558 (Suppl. 1) "Papers Presented at the Controlled Thermonuclear Conference," and Washington, D. C., 2/3-5/58.

S. G. Alikhanov, I. K. Konkashbaev, P. Z. Chebotaev, "The Energy Balance in a Dense Fusion Plasma Contained by Walls," Nuclear Fusion 10 13 (1970).

W. Butler, A. J. Hatch and A. J. Ulrich, "Radio-frequency Thermonuclear Machines," Proc. 2nd Int. Conf. on Peaceful Uses of Atomic Energy, 32 324, Geneva, 1958.

I. J. Spalding, "Cusp Containment and Thermonuclear Reactors," Nuclear Fusion 8, 161 (1968).

B. Lehnert, "Screening of a High-Density Plasma from Neutral Gas Penetration," Nuclear Fusion 8, 173 (1968).

C. M. Braams, "On the Stability of Plasma Confined by a Cold-Gas Blanket," I.R. 66/017, Associatie Euratom-Fom, Fom-Instituut Voor Plasma-Fysica Rijnhuizen, Jutphaas.

C. E. Nielsen, "Some Thoughts on How to Build a Gas-Confined Thermo-nuclear Reactor," September 21, 1966, [Tentative and informal material distributed for discussion within the Thermonuclear Division], Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

A. A. Hussein, "A Study of the Feasibility of Steady State Self-Heated Controlled Thermonuclear Power Reactors," PLP 323, November 1969, University of Wisconsin, Madison, Wisconsin.

W. J. Lange, "Ultrahigh Vacuum Research in Support of the Thermonuclear Power Program," Scientific Paper 403-FF-312-P1, August 8, 1958, Westinghouse Research Laboratories, Beulah Road, Churchill Borough, Pittsburgh 35, Pennsylvania.

J. N. Grace and M. A. Schultz, "Fusion Reactor Kinetic Equations and Transfer Function," WCAP-416, January 14, 1957, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

E. D. Simon and G. Bronner, "A One-Megajoule Inductive Energy Storage System Using Ignitron Switching," MATT-741, April 1970, Plasma Physics Laboratory, Princeton University, Princeton, N. J.

Don Steiner, "Neutronics Calculations and Cost Estimates for Fusion Reactor Blanket Assemblies," ORNL-TM-2360, November 22, 1968, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

A. P. Fraas, "A Potassium-Steam Binary Vapor Cycle for a Molten-Salt Reactor Power Plant," Paper No. 66-GT/CLC-5, Oak Ridge Nat. Lab., Oak Ridge, Tennessee.

J. Rand McNally, Jr., "A Novel Concept for Start-Up of Controlled Fusion Reactions," Oak Ridge National Laboratory, Oak Ridge, Tennessee. Paper presented at Rochester Meeting, American Physical Society, June 18-20, 1969.

F. L. Ribe, "Pulsed Systems," (With a contribution from R. L. Hirsch, U.S. Atomic Energy Commission, Washington, D. C. USA), LA-DC-11045, A Summary of Topics from the Working Study Group of the Nuclear Fusion Reactor Conference, Culham, England, 22-26, Sept. 1969. Los Alamos Scientific Laboratory, Los Alamos, New Mexico.

A. I. Morozov, "Stationary Plasma Accelerators and Their Applications in Thermonuclear Research," I. V. Kurchatov Institute of Atomic Energy, Moscow (Translated at Los Alamos Scientific Laboratory).

Richard W. Werner, "The Generation and Recovery of Tritium in Thermo-nuclear Reactor Blankets Using Heat Pipes," UCID-15390, October 3, 1968, Lawrence Radiation Laboratory, University of California, Livermore.

\* R. W. Werner, "Heat Pipes for Recovery of Tritium in Thermonuclear Reactor Blankets," UCRL-72329 Preprint, March 18, 1970, Lawrence Radiation Laboratory, University of California, Livermore.

Richard W. Werner, "Heat Pipes as a Means of Energy Removal from Thermonuclear Reactor Vacuum Walls," UCID-15386, July 24, 1968, Lawrence Radiation Laboratory, University of California, Livermore.

Stephen O. Dean, "Confinement of Laser-Produced Plasma in Resonant Cavities by RF Electromagnetic Fields," Oct. 1968, U. S. Naval Research Laboratory.

Tien Sun Chang, "Feasibility Study of Vortex Confinement of Fusion Plasmas," ORNL-PR-2-1, (1966), Oak Ridge National Laboratory, Oak Ridge, Tennessee.

P. L. Kapitsa, "Thermonuclear Reactor with a Plasma Column Hovering Freely in a High-Frequency Field," JPRS 50255, April 7, 1970, Clearinghouse for Federal Scientific and Technical Information, Springfield, Va. 22151.

R. F. Post, "High Temperature Plasma Research and Controlled Fusion," Annual Review of Nuclear Science 9, 367 (1959).

M. J. Eden, P. A. H. Saunders, "Triggering Requirements for Pulsed Fusion Reactors," CLM-P226 (1969), Culham Laboratory, Abingdon, Berkshire, U.K.

Yu. I. Ostroushko, P. I. Buchikhin, V. V. Alekseeva, T. F. Naboishchikova, G. A. Kovda, S. A. Shelkova, P. N. Alekseeva, and M. A. Makovetskaya, "Lithium, Its Chemistry and Technology," AEC-tr-4940 Chemistry, United States Atomic Energy Commission, Division of Technical Information.

K. A. Ehrlicke, "Synoptic Comparison of Advanced Propulsion Systems for Maneuvering Operations Associated with Several Employment Modes in Geolunar Space," April 1968, Autonetics Division of North American Rockwell Corporation, 3370 Miraloma Avenue, Anaheim, California 92803.

J. G. Linhart, "Some Effects of  $\alpha$ -Particles in a low- $\beta$  fusion Reactor," Nuclear Fusion 9, 269 (1969).

Proceedings of Third Symposium on Advanced Propulsion Concepts, 2-3-4 October 1962, Sheraton-Gibson Hotel, Cincinnati, Ohio. Sponsored by United States Air Force Office of Scientific Research and General Electric Company, Flight Propulsion Division, Vol. 1, Gordon and Breach Science Publishers. (See following 3 references).

John S. Luce, "Controlled Fusion Propulsion," Research Division, Aerojet-General Nucleonics.

William I. Linlor and Milton U. Clauser, "Fusion Plasma Propulsion System," Hughes Research Laboratories.

Dwain F. Spencer and Leonard D. Jaffe, "Feasibility of Interstellar Travel Jet Propulsion Laboratory.

Proceedings, Nuclear Fusion Reactor Conference, UKAEA, Culham Laboratory, Abingdon, Berkshire, England, September 17-19, 1969. (See following 41 references).

A. P. Fraas, "Conceptual Design of a Fusion Power Plan to Meet the Total Energy Requirements of an Urban Complex."

P. J. Searby and L. G. Brookes, "The Economic Factors Affecting Research into Fusion Power."

J. L. Tuck, "On Nuclear Fusion Objectives."

L. M. Lidsky, "Fission-Fusion Symbiosis: General Considerations and a Specific Example."

F. Morley and J. W. Kennedy, "Fusion Reactors and Environmental Safety."

R. S. Pease, "Plasma Confinement by Magnetic Fields."

R. F. Post, "Mirror Systems: Fuel Cycles, Loss Reduction and Energy Recovery."

D. R. Sweetman, "Mirror Reactors: Some General Considerations."

J. C. Cordey and C. J. H. Watson, "The Geometry of a Minimum-B Reactor."

L. G. Kuo-Petravic, M. Petravic and C. J. H. Watson, "Alpha Particle Heating and the Energy Balance in a Mirror Reactor."

N. C. Christofilos, "ASTRON Plasma Parameters Confined in the Closed Magnetic Well of a Proton E-Layer."

A. A. Newton, "Fusion Reactors and Plasma Flow."

I. N. Golovin, Yu. N. Dnestrovsky, and D. P. Kotomarov, "Tokamak as a Possible Fusion Reactor - Comparison with Other CTR Devices."

P. Hubert, "Thermonuclear Future of the Tokamak Magnetic Confinement Device."

A. Gibson, "Permissible Parameters for Economic Stellarator and Tokamak Reactors."

G. I. Bell, W. H. Borkenhagen, and F. L. Ribe, "Feasibility Studies of Pulsed, High-Beta Fusion Reactors."

H. A. B. Bodin, T. E. James and A. A. Newton, "A Pulsed Fusion Reactor Based on a Toroidal Pinch."

E. P. Butt, "Feasibility Studies of Pulsed Toroidal Reactors."

T. E. James, A. A. Newton, and H. A. B. Bodin, "Some Design Aspects of the Field System and Associated Power Supply of Pulsed Toroidal Fusion Reactors."

- \* T. A. Oliphant, "Fuel Burnup and Direct Conversion of Energy in a D-T Plasma."
- \* R. G. Mills, "Time-Dependent Behavior of Fusion Reactors."
- R. Carruthers, "Engineering Parameters of a Fusion Reactor."
- A. J. Summers, N. J. Freeman, and N. R. Daly, "Sputtering Coefficients of Niobium."
- G. M. McCracken and S. K. Erents, "Ion Burial in the Divertor of a Fusion Reactor."
- M. T. Robinson, "The Energy Dependence of Neutron Radiation Damage in Solids."
- B. Myers, "Some Observations on 14 MeV Neutron Radiation Effects on Reactor Materials."
- D. G. Martin, "An Assessment of Some Radiation Damage Effects in the Containment Vessel of a Thermonuclear Reactor."
- B. J. Eastlund and W. C. Gough, "The Fusion Torch - Unique Applications of Ultra-High Temperature Plasmas."
- O. C. Yonts, "Sputtering of Niobium by  $D^+$  and  $He^+$  Ions."
- D. J. Rose, G. L. Flint, Jr., and F. B. Marcus, "Some Topics on Pulsed Systems and Mirrors."
- E. F. Johnson, "Overall Tritium Balances in Fusion Reactors."
- R. W. Werner, B. Myers, P. B. Mohr, J. D. Lee, and N. C. Christofilos, "Preliminary Design Considerations for an Anstron Power Reactor System."
- J. D. Lee, "Tritium Breeding and Energy Generation in Liquid Lithium Blankets."
- D. Steiner, "Neutronic Behavior of Two Fusion Reactor Blanket Designs."
- S. Blow, V. S. Crocker and B. O. Wade, "Neutronics Calculations for Blanket Assemblies of a Fusion Reactor."
- H. C. Cole, "Hall Accelerators in Fusion Research."
- D. J. H. Wort, "Fusion Reactor Ignition Using RF Power."
- G. R. Hopkins and G. Melese-d'Hospital, "Direct Helium Cooling Cycle for a Fusion Reactor."
- R. W. Werner, "Module Approach to Blanket Design--A Vacuum Wall Free Blanket Using Heat Pipes."
- P. B. Mohr, "Fusion Reactor Power Conversion Studies."
- A. S. Bishop, "Conference Summary."

V. S. Crocker, S. Blow, C. J. H. Watson, "Nuclear Cross-Section Requirements for Fusion Reactors," CLM-P 240, Culham Laboratory, Abingdon, Berkshire, U. K.

Papers to be Presented at the 5th Intersociety Energy Conversion Engineering Conference, Las Vegas, Nevada, Sept. 21-29, 1970  
(Next 18 References)

Bob Mills, "Ignition of Toroidal Fusion Reactors," Princeton University, Plasma Physics Laboratory.

Fred Ribe, "Economic Considerations and Magnetic Energy Storage for High- $\beta$  Pulsed Reactors," Los Alamos Scientific Laboratory.

Dick Post, "Direct Conversion of Fusion Energy to Electricity," Lawrence Radiation Laboratory.

Don Steiner, "A Review of ORNL Fusion Feasibility Studies," Oak Ridge National Laboratory.

Nick Christofilos, "Astron Plasma Parameters Confined in the Closed Magnetic Well of a Proton E-Layer," Lawrence Radiation Laboratory.

J. D. Lee et al., "Some Engineering Aspects of an Astron Fusion Power Reactor System," Lawrence Radiation Laboratory.

Ernie Johnson, "Tritium Recovery in Fusion Power Reactors," Princeton University, Department of Chemical Engineering.

Dick Werner, "Heat Pipes for Recovery of Tritium in Thermonuclear Reactor Blankets," Lawrence Radiation Laboratory.

J. D. Lee, "Some Observations on the Radiological Hazards of Fusion vs. Fission," Lawrence Radiation Laboratory.

L. Lidsky, "Hybrid Fission-Fusion Reactors", M.I.T.

Gilbert Melese-d'Hospital and George Hopkins, "Gas Cooling for Fusion Reactor Blankets," Gulf General Atomic

Blake Myers, et al, "Structural Effects of Niobium Ion Bombardment on Niobium Foils, I," Lawrence Radiation Laboratory.

Edward Lady and Darrell Call, "High Voltage Test of a Large Cryogenic Coil for Magnetic Energy Storage System," Los Alamos Scientific Laboratory.

Clyde Taylor and Ralph Moir, "Magnets for Fusion Reactors," Lawrence Radiation Laboratory.

Carl Henning, "Stability in Superconducting Magnets," Lawrence Radiation Laboratory.

Blake Myers, "Thermonuclear Reactors and Electric Power Generation, a Survey," Lawrence Radiation Laboratory.

Tom Batzer, "Ultra High Vacuum Techniques for Controlled Fusion Research," Lawrence Radiation Laboratory.

Pete Mohr, "Some Economic Aspects of Power Conversion for Fusion Reactors," Lawrence Radiation Laboratory.

\* Roth, J. R., Rayle, W. D., and Reinmann, J. J.

"Technological Problems Anticipated in the Application of Fusion Reactors to Space Propulsion and Power Generation," NASA Technical Memorandum, NASA TM X-52809, Lewis Research Center, Cleveland, Ohio.

\* Mills, R. G.

"Synchrotron Radiation from Fusion Reactors," Report No. MATT-658, Plasma Physics Laboratory, Princeton University, Princeton, N. J.

---

\* Papers that were distributed at the Symposium.

## LITERATURE SEARCH REFERENCES

"Controlled Thermonuclear Processes," A Literature Search compiled by Raymond L. Scott and Sidney F. Lanier, November 1960, TID-3557, Office of Technical Information, United States Atomic Energy Commission.

"Controlled Fusion and Plasma Research," A Literature Search compiled by Sidney F. Lanier, August 1962, TID-3557 (Suppl. 1), Division of Technical Information, United States Atomic Energy Commission.

"Controlled Fusion and Plasma Research," A Literature Search compiled by Sidney F. Lanier, January 1965, TID-3557 (Suppl. 2), Division of Technical Information, United States Atomic Energy Commission.

"Research on Controlled Thermonuclear Fusion," Bibliographical series, No. 7, International Atomic Energy Agency, Vienna 1962.

"Bibliography on Plasma Physics and Magnetohydrodynamics and their Applications to Controlled Thermonuclear Reactions," Compiled by James D. Ramer, Engineering and Physical Sciences Library, University of Maryland, College Park, Maryland.

"Controlled Thermonuclear Processes," A Selective Bibliography Compiled by Thomas W. Scott, May 1958, Technical Information Service, U. S. Atomic Energy Commission.

"Bibliography on Plasma Physics," Compiled by N. Robert Nilsson, Institute of Physics, Uppsala, Sweden, December 1960.

E. Atlee Jackson, "A Survey of the Research of Ionized Media," AFCRC-TR-59-174, August 1959, Electronic Material Sciences Laboratory, Electronics Research Directorate, Air Force Cambridge Research Center, Air Research and Development Command, United States Air Force, Bedford, Massachusetts.

"A Survey of Some Plasma Acceleration Devices for Space Propulsion Applications," Prepared by D. M. Dix, Plasma Research Laboratory, Report No. TDR-169(3713-01)TR-1, Laboratories Division, Aerospace Corporation, Contract No. AF 04(695)-169, May 17, 1963.

Bibliographical Series No. 35, "Laser Applications in Plasma Physics," (1962-1968), International Atomic Energy Agency, Vienna, 1969.

"Plasma Physics & Magnetohydrodynamics," A Report Bibliography prepared by Armed Services Technical Information Agency, March 1962, AD-271 170.

"Plasma Physics & Magnetohydrodynamics," (Supplement to AD 271 170), A Report Bibliography prepared by Defense Documentation Center, June 1963, AD-405 732.



# FUSION REACTOR DESIGN SYMPOSIUM

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