University of Wisconsin Fusion Design Studies

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FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

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ABSTRACT

The University of Wisconsin fusion design effort is directed initially to a low β toroidal reactor. The studies are described in terms of parts of the system moving from the center outward with remarks on fuel handling and materials at appropriate points.

The plasma is central geometrically and technically, and self consistent parameters for operating points have been generated in the presence of radiation energy loss enhancement by impurities and particle confinement spoiling. A poloidal divertor produced by both superconducting and regular windings is described. Materials problems of the first wall are discussed with emphasis on radiation effects. Heat removal is discussed with representative results for wall structure, and coolant temperatures and pressure for helium cooling. The special neutronics problems of the high energy neutrons are reviewed and the present state of the calculational model and data are stated. Some aspects of the tritium breeding and handling problem are covered that are likely to be independent of technology. Finally, magnet design is discussed in terms of the costs associated with large high field systems which are dominated by mechanical stress considerations.

THE WISCONSIN FUSION DESIGN EFFORT is initially directed toward a low β toroidal system, but many of the design considerations will apply with little or no modification to other systems. However, it was felt that a more effective contribution to the field would result from initially restricting the study in this way.

The proposed device would consist schematically of a D-T plasma surrounded by a blanket composed of a first wall, coolant, and neutron moderating and energy producing materials. The blanket is surrounded by a shield to attenuate the radiation to a satisfactory level for the cryogenic and superconducting magnet system which surrounds the entire unit. The blanket must contain lithium in sufficient quantities to breed tritium and the shield and blanket must be breached by suitable passages for fuel injection, plasma heating devices, and spent fuel exhaust systems. A schematic of the configuration is shown in Fig. 1. A divertor system may be required to protect the first wall from escaping high energy particles and will require additional magnet coils.

The design of these and other subsystems start naturally with the plasma and its operating parameters. It seems desirable to start up the system and bring it to a steady state operating point where it continues for as long a time as possible. This is referred to as the burn time and is determined in a Tokamak type system by the length of time the stabilizing plasma current can be induced by the imposed field (unless bootstrap currents can stabilize the system). This is estimated to be on the order of a thousand seconds. This interval is much greater than the average particle confinement times that are now predicted (1)* (2); thus the plasma must be refueled in an essentially steady state manner once the operating point is reached.

The operating conditions are determined by energy and particle balances on the different particle species present. Stability criteria such as the limits on β_p, are imposed as constraints, and losses by diffusion processes are included, using confinement times, in the balance equations. The α particles produced by the fusion reactions have been shown by Conn (3) and others (4), (5) to slow down much faster than their expected leakage rate, implying that essentially all their energy goes into heating the plasma particles. Most of the initial alpha energy goes to the electrons and is then divided between radiation losses and heating of the ions.

*Numbers in parentheses designate References at end of paper.

The ions leak from the system or are destroyed by fusion reactions.

Conn and McAlees (6) have attempted to account for each of these effects in an effort to find a self-consistent operating point. Since they include the particle confinement from the beginning, this is an extension of similar work by Rose (7). The operating point is influenced by the plasma radius r_p, the toroidal magnetic field B_T, the ratio of major torus radius, R, to the plasma radius i.e. the aspect ratio A, the stability factor q given (in terms of the poloidal magnetic field B_p and previously defined quantities) by B_T/AB_p, the impurity atomic number, and a spoiling factor which measures the enhancement of particle leakage relative to a neoclassical diffusion coefficient, as modified by Strelkov and Popkov (8) to fit most experimental data. The first group of factors enters the key equations in the particular combination $(r_p^2 B_T^2/q^2 A^{3/2})$ which we have labeled the McAlees number, Mc. The impurity density and atomic number enter the equations through a radiation enhancement factor N_H, which multiplies the bremsstrahlung energy loss term, and is one when there are no impurities. Similarly, the particle confinement spoiling factor S is one when the Shelkov and Popkov fit is used for the diffusion coefficient. On imposing the restriction β_p equal $\sqrt{A}$, where β_p is the ratio of particle to poloidal magnetic pressure, one finds solutions to the coupled balance equations. The lower temperature satisfying these equations is an unstable equilibrium point and without radiation enhancement or particle confinement spoiling, the temperature is low (less than 8 KeV) and the power density is very low. The unstable nature of the point is readily understood if one notes that in the temperature range of interest, the gains are proportional to the temperature and the losses to the square root of the temperature leading to a positive temperature coefficient. Operation at this point can be feedback stabilized in principle and initial calculations indicate time scales on the order of seconds for significant changes so that it should be practical.

The plasma heating by the α particles is proportional to the power. To increase the power level and maintain an equilibrium, the energy losses must increase to balance the gains. This increase of the losses can be accomplished by enhancing bremsstrahlung radiation through introduction of high atomic number impurities into the [illegible] will allo an increased operating temperature and power density as shown in Fig. 2. However, the fractional burn up then increases rapidly. By spoiling particle confinement the fractional burn up of plasma particles can be brought to satisfactory levels. The above discussion holds for neoclassical diffusion. With an appropriate correction factor to fit the current Tokamak experimentally determined confinement time, τ can be taken as

$$\tau = C\, M_c\, T^{1/2} \qquad (1)$$

If Bohm diffusion, as calculated to occur in non axisymmetric toroidal systems (9), controls the confinement time at higher temperatures, it will scale as T^{-1} instead of $T^{+1/2}$. This may make it possible to convert from an unstable to a stable equilibrium point. This possibility has also been noted by Ohta, Yamato, and Mori, and by Mills (10), (11). These concepts are currently under further investigation.

Moving away from the system center the next region is usually referred to as the divertor region. This is the space between the plasma and the first wall which is swept relatively free of particles by the field configuration in order to protect the wall from particles from the plasma which could cause radiation blistering and sputtering (12), (13). It would be much simpler and cheaper to dispense with the divertor and have no special fields for this purpose. As the energy to the first wall for realistic plasma operating conditions is less than originally anticipated it may be possible to allow the particles to simply go to the wall without eroding it at an unacceptable rate. On the other hand, the impurities introduced to enhance radiation losses may also enhance leakage and instability and thus demand an effective divertor. This point is still under investigation.

The particles of the plasma are guided by the field lines which are set up to close on themselves in the plasma region to effect confinement. The field lines in the divertor region before closing must be caused to pass through a passage in the blanket and shield to a chamber which can handle the energetic leakage flux guided there. This is illustrated by the flux lines shown on Figure 3. This diversion of the field lines can be accomplished in theory by extra magnet coils producing either torodial or poloidal fields (around the major or minor radii of the torus). As the poloidal divertor would be less disruptive of the symmetry and thus of stability, this approach has been taken. The divertor field must be programmed in time to hold the proper relationship to the fields due to the induced currents in the plasma. If this is accomplished with ordinary coils placed close to the first wall, the two coils would require on the order of 200 megawatts to drive them in the case of a system with $r_p = 3$ meters and an aspect ratio of 6. This would mean 400-500 megawatts thermal from the plant might be required for the system. Since this is prohibitive, Emmert has suggested placing superconducting coils consuming little energy about two meters back from the first wall near the main coils where they will be adequately shielded. However, the fields would not be able to soak through the relatively high conductivity materials of the blanket and shield on the required time scale. The solution he envisions is to use coils of both the above types with the fields bucking one another out on the short non-burn portion of the cycle. The inner coils then operate on a low duty cycle and thus at a low average power.

The first wall must operate at very high temperatures if high thermal efficiencies are to be achieved and will be exposed to intense radiation fields. The refractory metals niobium, molybdenum, and vanadium are the prime candidates with stainless steel a choice either for a prototype reactor or in the event thermal efficiency can be sacrificed making a lower wall temperature acceptable. The wall problem is essentially specificed when the wall loading is given. Values for neutron wall loading usually mentioned range from the order of one to ten megawatts per square meter. The limitations on it are imposed by heat removal and radiation damage requirements and by achievable plasma conditions. Calculations to date (14), (6) indicate it will be difficult to produce 1 MW/m^2, and probably impossible to achieve 10 MW/m^2, in a low β toroidal system. In addition, while experimental data is very incomplete, radiation damage and impurity production at 10 MW/m^2 are expected to create extreme swelling and embrittlement problems.

The choice between wall material candidates is a trade off between fabrication, compatibility, strength, radioactivity, and radiation effects problems. A decision will require considerable additional data. This is particularly true, as the refractory metals have not received the attention given materials used extensively in fission reactors. In any case, radiation damage will be a major consideration in the assessment. The energy of the incident neutrons as well as their number is important in the evaluation. Kulcinski, Abdou, and Doran (15) have calculated displacement and transmutation rates in potential CTR materials. These calculations compare results for a CTR with a number of other existing or proposed neutron facilities. The HEDL dosimetry data on spectra in the ETR, HFIR, EBR-2, and FFTF were used. Cross sections for flux calculations are from ENDF/B2 as processed to DLC-2 tapes. ANISN was used to obtain spectra, and displacement cross sections as a function of incident neutron energy result from a modified Linhart model of the process. Figure 4 shows the calculated displacement rates for niobium and clearly these can be duplicated in a number of facilities, at least at the lower wall loading. On the other hand, transmutation rates are more difficult to duplicate and the ratio of transmutations to displacements is not available from existing facilities and will require a 14 MeV neutron source preferably of high intensity or a CTR as shown in Figure 5. The ratio can, however, be simulated by two accelerators operating simultaneously, the first for heavy ions and the second for α's or protons.

Heat removal can probably be accomplished up to about 1 MW/m^2, but a neutron wall loading of 10 MW/m^2 would almost certainly deposit more than this and agravate the cooling problems in addition to the wall damage. Thus while a high wall loading is economically desirable, it seems technically unreasonable to allow greater than the order of 1 MW/m^2 even if it can be produced.

The blanket must contain lithium and this has promoted the choice of lithium itself as the coolant or the lithium containing molten salt "flibe". These and gaseous helium are the favored coolants at present. Helium cooling is suggested since it does little to disrupt the neutronic processes which produce the tritium breeding.

A conceptual first wall is shown on Figure 6 which would be suitable for the case of helium cooling. A schematic of the wall coolant flow distribution is also shown in Figure 6. The wall is designed for rigidity against buckling and for ease of heat removal. Sze and Stewart suggest for cooling this wall at a loading of 0.76 MW/m^2, helium at a pressure of 50 atmospheres pumped at a velocity of 8.25 m/sec. In this design Sze calculates inlet and outlet temperatures of 673° K and 1023° K respectively with a pumping power of 0.23 MW.

While our initial calculations have been carried out for helium as the coolant, cooling by lithium and flibe will be considered. Calculations for lithium flowing in a magnetic field are currently being studied.

The neutronics problems that are new and substantially different from those encountered elsewhere are associated with the high energy of the source neutrons. These cause significantly more (n,α) and (n,p) reactions than is caused by fission spectrum neutrons as was mentioned in the radiation damage discussion. In addition, the scattering is very anisotropic. This requires careful treatment of the source geometry, a high order transport approximation and an adequate treatment of scattering anisotropy. Abdou and Maynard have found that a distributed source with a parabolic spatial distribution represents the expected source (16) and that the S6 approximation with P3 scattering anisotropy is adequate to treat neutrons above 8 MeV (17). This is coupled with a mesh spacing of $1/\Sigma_T N$, where N is the order of the S_N approximation. Lower order approximations will likely suffice for neutrons of lower energy.

Another neutronics problem is that of energy deposition. Since most of the energy is carried by the neutrons and the gammas which they produce, the Kerma factors for materials of interest in the blanket must be obtained and used to provide the source distribution for heat transfer calculations (18). These are currently being generated using ENDF/B3 tapes and should be available from RSIC soon.

Tritium production of greater than one per fusion reaction will result from (n,2n) reactions with the wall nuclei and Li^7(n,n'α)T reactions followed by capture of almost all thermalized neutrons in Li^6 through the Li^6(n,α)T reaction. Calculated production per initial neutron ranges from around 1.1 up to about 1.5 depending on the materials, data, geometry, and model employed. To date, these calculations have not been made for realistic geometries and are affected by considerable uncertainty in the data. However, the tritium breeding is not regarded as a major problem at this time.

The tritium recovery and handling problem is a very serious one from the environmental and hazard point of view. A 5,000 MWth plant burns on the order of a kilogram of tritium per day. The fraction of . . . . range from a few to around 20%. If the exhaust gases can be processed in one day an inventory of the order of 10 Kg is necessary with several additional kilograms available depending on the desired reserves for operation. Vogelsang has shown that the external inventory of produced tritium is determined primarily by the breeding ratio T while the internal inventory (i.e. tritium still in the blanket) is fixed primarily by the removal system which is characterized by the mean residence time for tritium in the blanket (19). Without regard to the technology for achieving the result, to keep the internal inventory to less than about one kilogram, which is desirable from hazards considerations, the residence time should be about a day or less. Similarly, in order to produce only an acceptable amount of excess tritium, T must be less than 1.1 and likely as low as 1.02 on the average. Designing to tolerances to insure the proper breeding is undesirable and a system for controlling T by blanket composition variation is a promising approach. The external inventory is shown as a function of time in Figure 7 and this illustrates the start up problem which requires an initial inventory that is greater than will be needed in steady operation in order to allow the internal inventory to be built up.

Over 90% of the neutrons and energy can be absorbed in a blanket consisting of lithium and graphite on the order of one meter thick. Further, a shield of lead, steel, and water about one meter thick will reduce the energy flux to the point where it can reasonably be handled by the cryogenic system for the magnets. The exact choice of materials and dimensions will be dictated by an optimization of the materials, refrigeration, and magnet costs. This optimization has an economic basis and this is a difficult point to deal with as it involves costs and income criteria appropriate to the time when the plant is to be built.

The magnets are among the most expensive parts of the system. Their cost clearly depends on the size of the system and the required magnetic field. However, with magnets of the size needed here, the design and thus the costs are determined by stress problems more than by the superconductors. To insure against local fluctuations which can produce a non-superconducting state causing damage to the windings, the superconductor is encased in copper which serves to stabilize the magnet. In large magnets, the energy storage in the fields is quite large and results in large stresses on the windings. To aid in supporting this stress, stainless steel is included. Finally insulation is provided; thus the entire winding is a composite. A design in which all stresses are kept below the yield stress of copper is conservative and expensive. For example, Boom and Young find the ratio of winding thickness to inner radius varies from 0.1 to 2.4 for central fields of 50 to 150 kilogauss respectively making high fields very expensive at large radii. If the stress is allowed to exceed the yield value for copper but not that of stainless steel, the structure in the winding can be cut in half and the savings are very large. However, there will be large mechanical hysteresis losses in the windings when the fields change as in shut down and start up operations. This is illustrated in Figure 8. This energy must be handled by the refrigeration system partly off setting the gains from this type of design. In the worse case, the yield could cause the windings to loosen up and damage the magnets.

We would like to acknowledge the efforts of the entire Wisconsin design team most of whom have been mentioned in the text.

Fig. 1. - Schematic of Tokamak showing plasma and blanket regions, magnet windings, and a poloidal divertor slot

Fig. 2. - Power versus McAlees number for a reactor with ion density N_i and aspect ratio A for two choices of the radiation enhancement and confinement spoiling factors N_H and S

Fig. 3. - Magnetic field lines closing inside the confinement region and diverted to other parts of the system when started in the divertor region

Fig. 4. - Displacement rates in niobium for a variety of neutron irradiation facilities including a CTR at two neutron wall loadings

Fig. 5. - The ratio of helium to displacement production in different irradiation facilities compared to a CTR

Fig. 6. - A conceptual design of the first wall and a coolant flow header for a helium cooled Tokamak

Fig. 7. - The external tritium inventory as a function of time showing the initial I_o and minimum $I_{ex\ min}$ inventories and the doubling time

Fig. 8. - Mechanical hysteresis in copper when its elastic limits are exceeded repetitively

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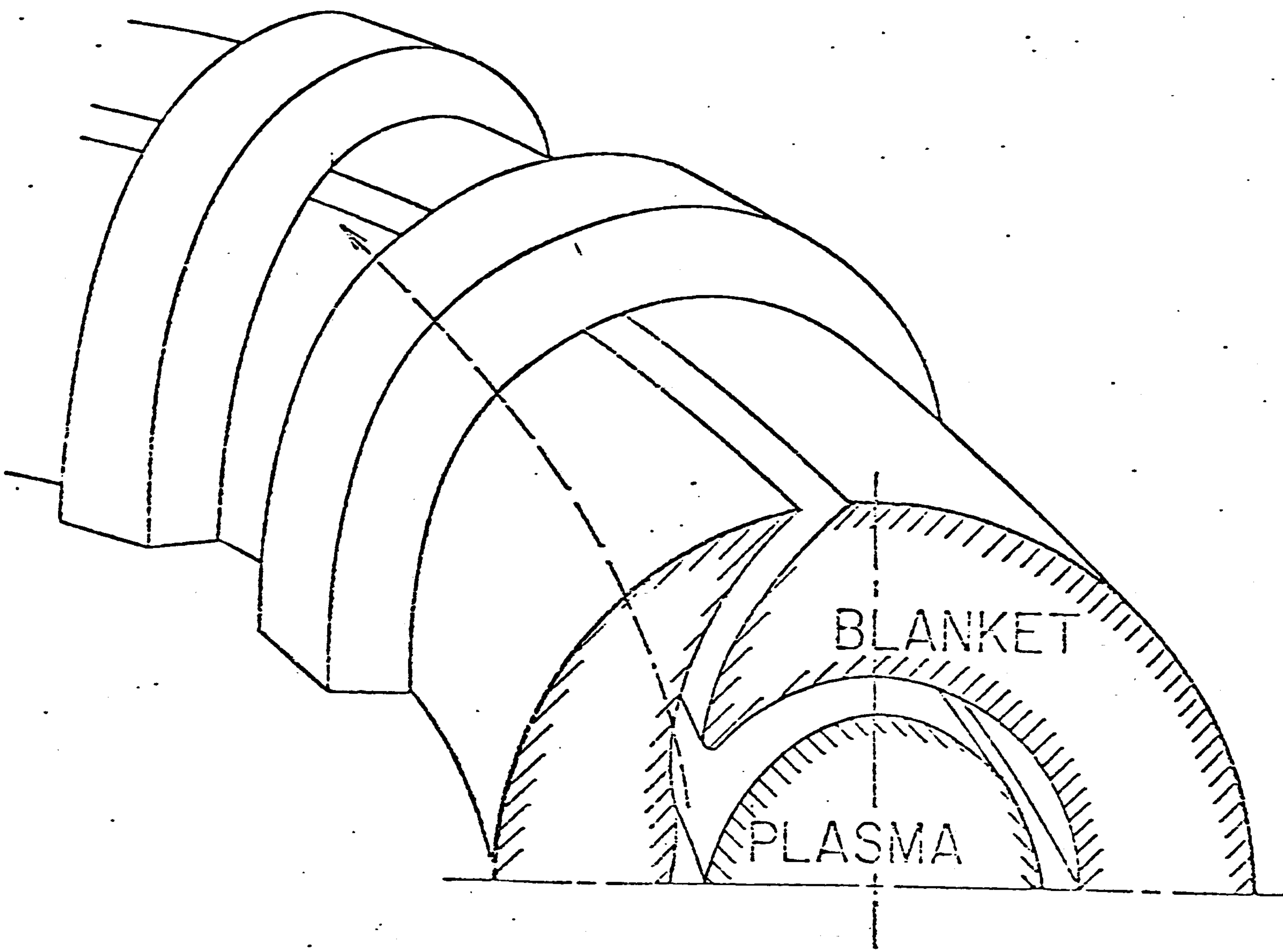
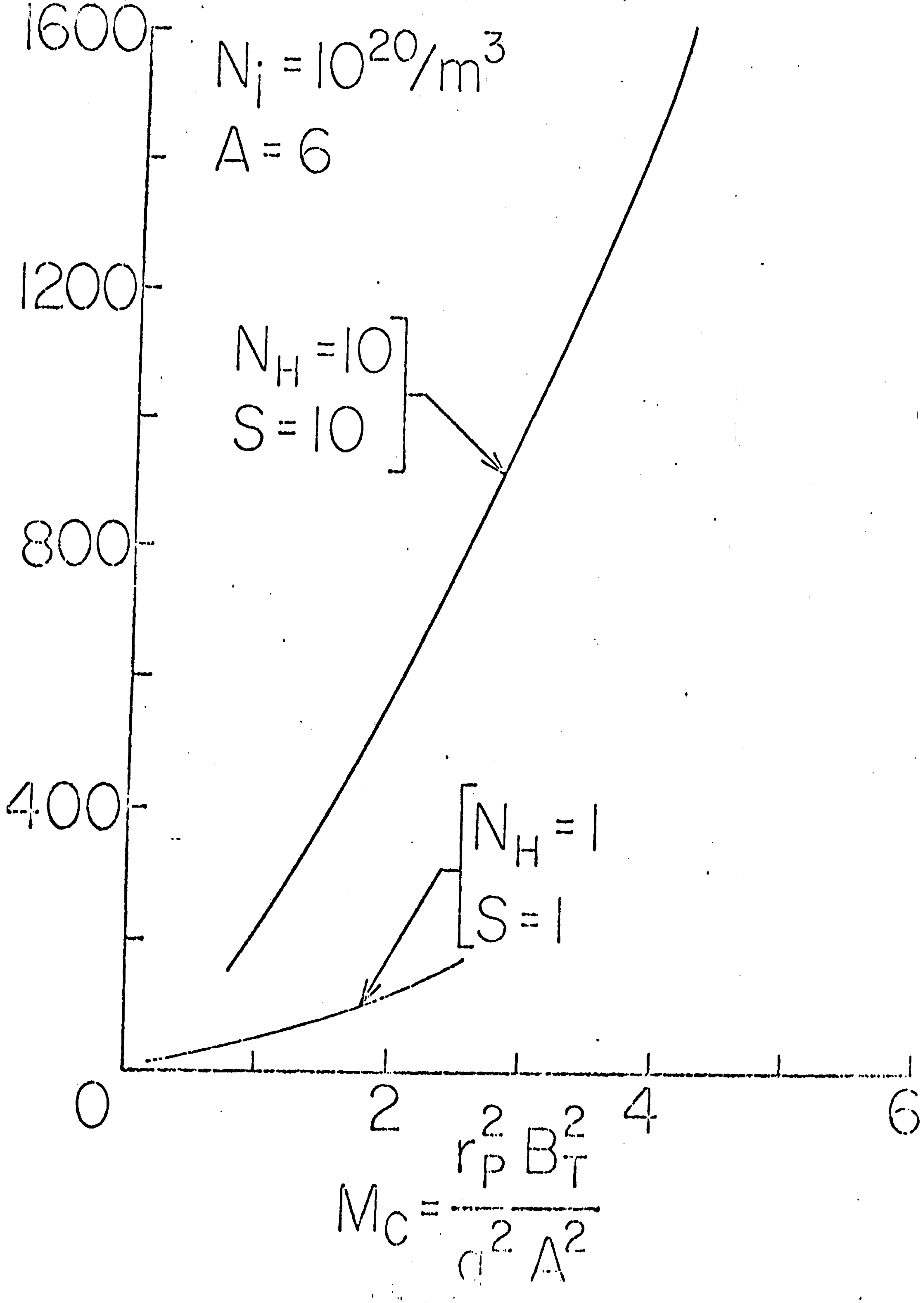
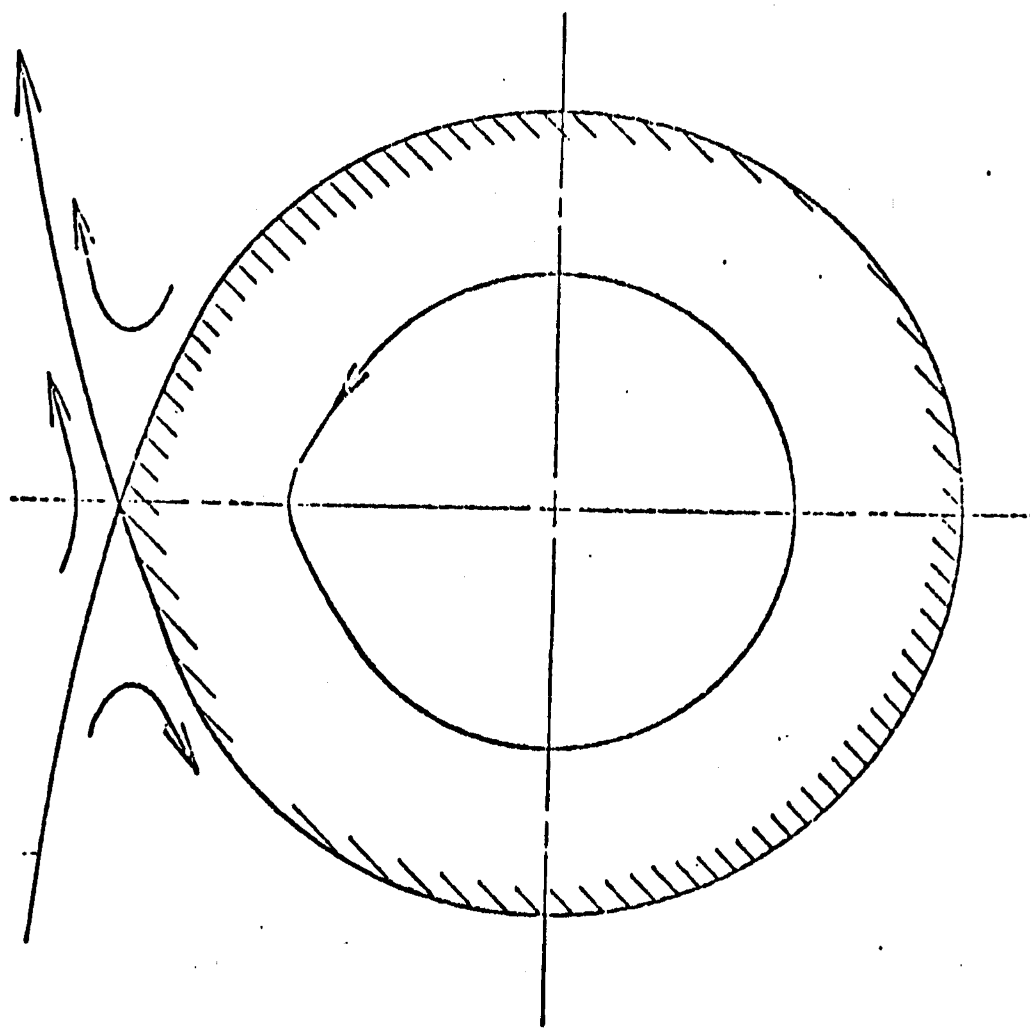
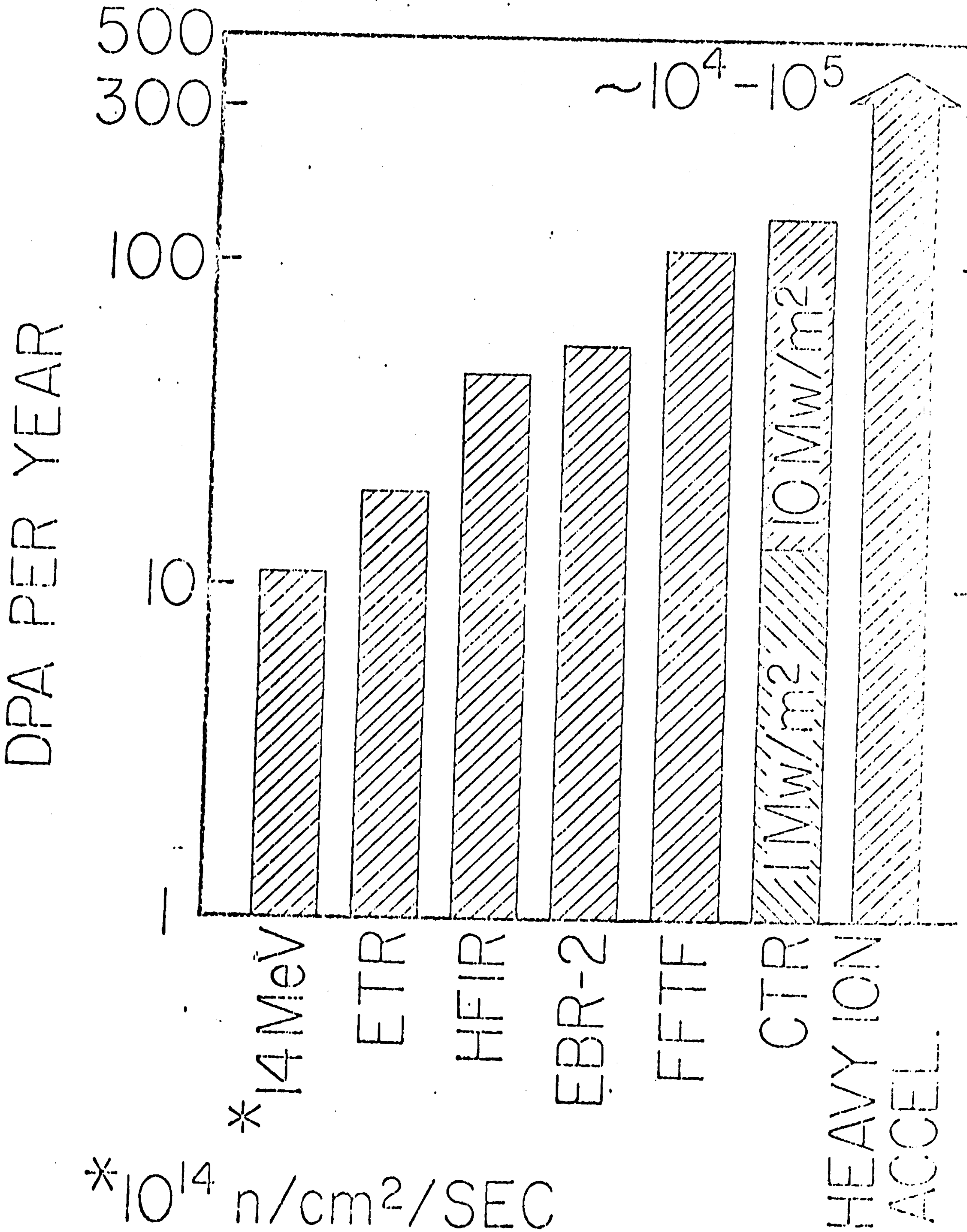


TOKAMAK POLOIDAL DIVERTOR

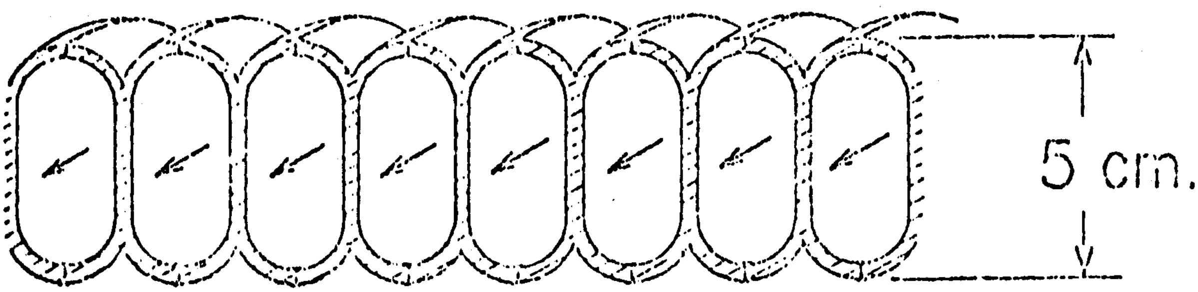


1600
1200
800
400
0
2
4
6
$N_i = 10^{20}/m^3$
$A = 6$
$N_H = 10$
$S = 10$
$N_H = 1$
$S = 1$
$M_C = \frac{r_p^2 B_T^2}{a^2 A^2}$

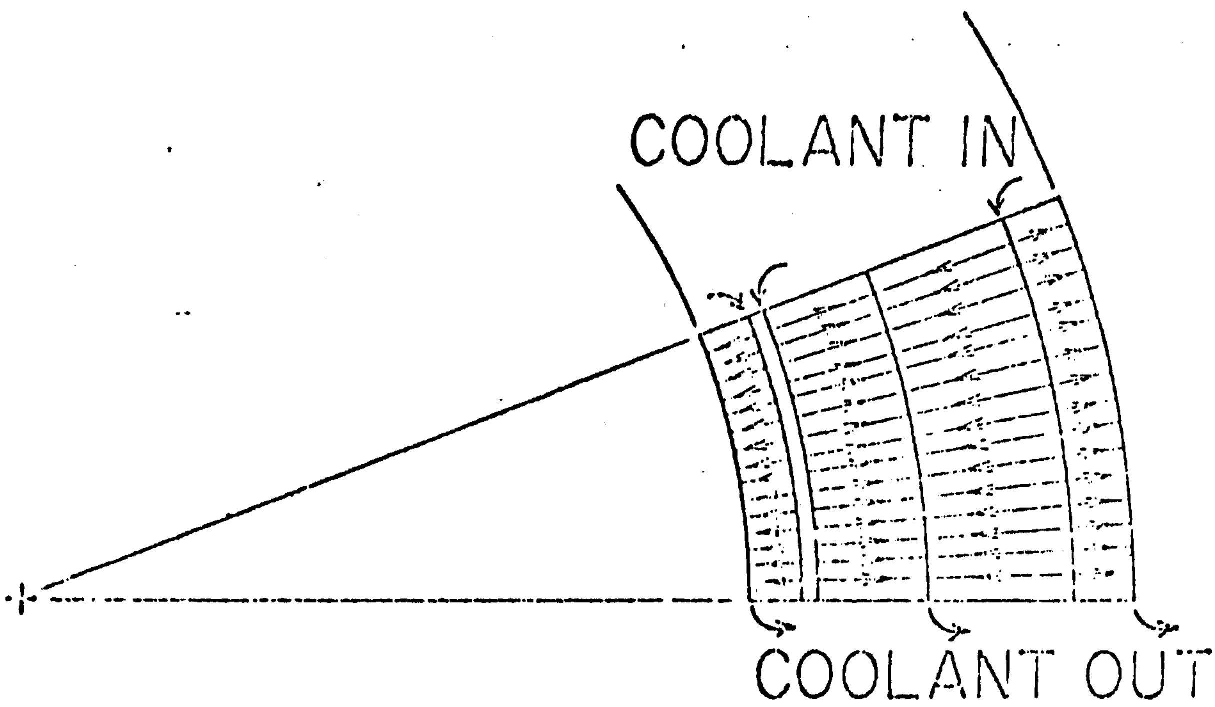


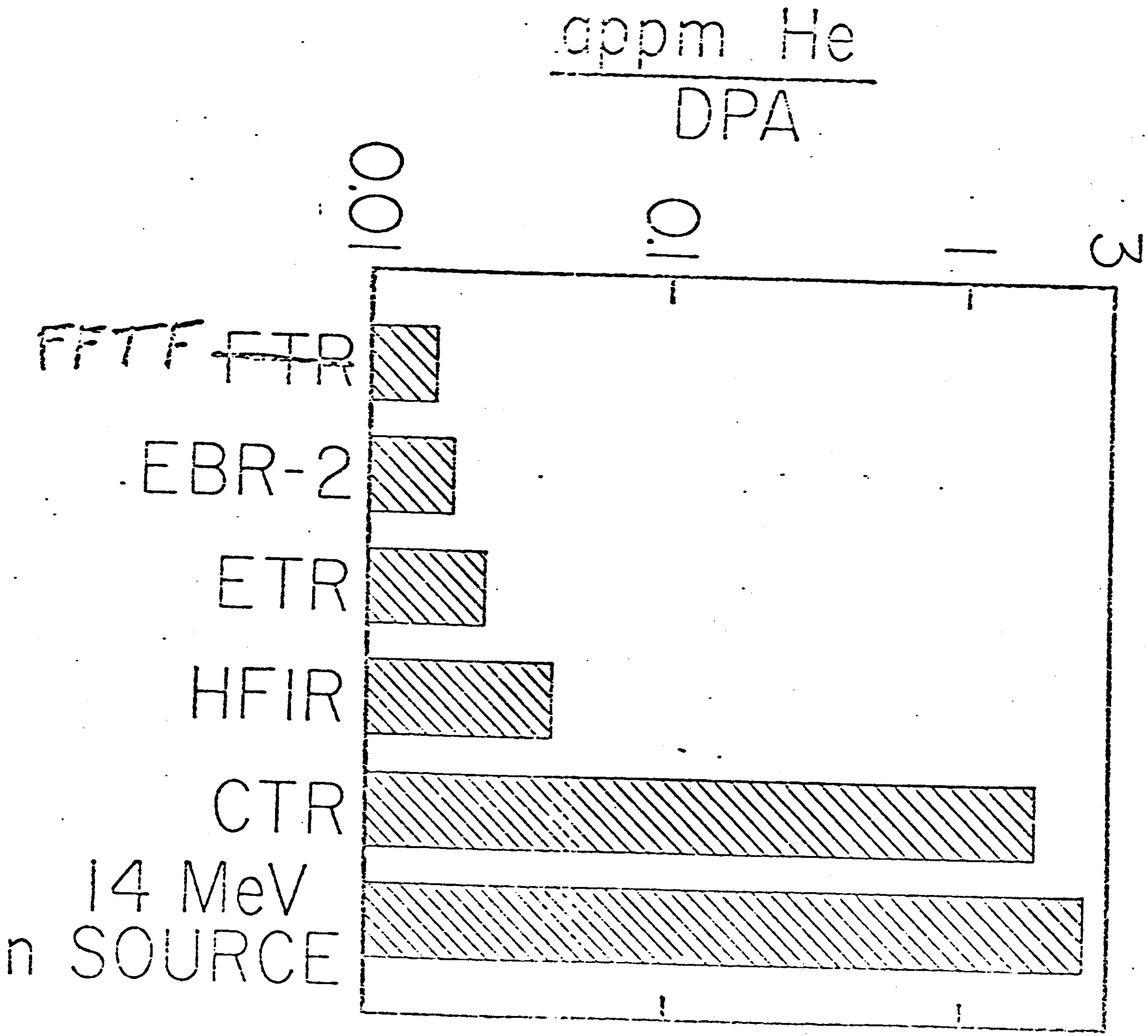

DPA PER YEAR
500
300
100
10
1
~10^4-10^5
1 Mw/m^2
10 Mw/m^2
*14 MeV
ETR
HFIR
EBR-2
FFTF
CTR
HEAVY ION ACEL
*10^14 n/cm^2/SEC

CROSS-SECTION OF FIRST WALL (HELIUM COOLING)



FIRST WALL COOLANT DISTRIBUTION (HELIUM COOLING)




appm He / DPA
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1
3
FFTF ~~FTR~~
EBR-2
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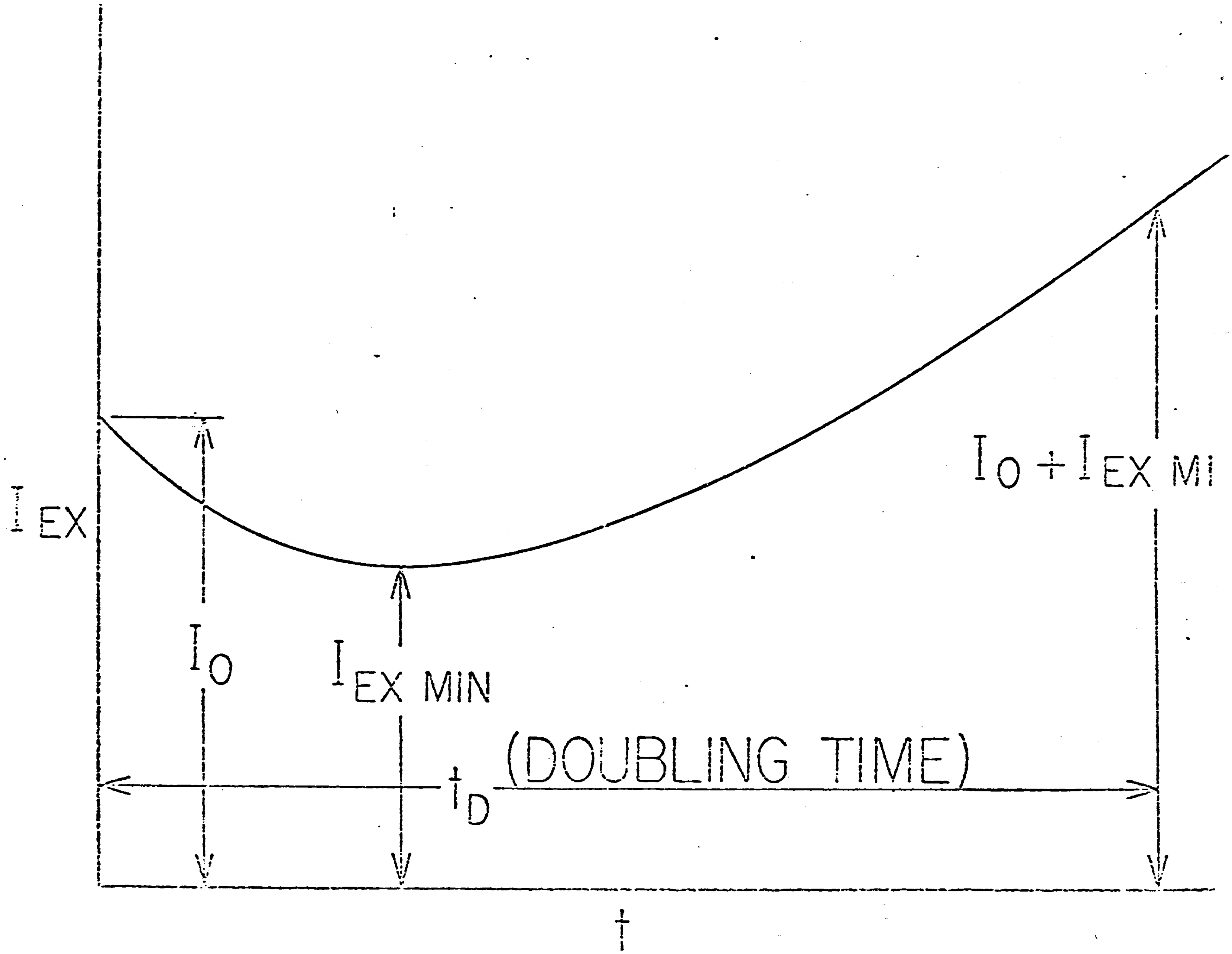
CTR
14 MeV n SOURCE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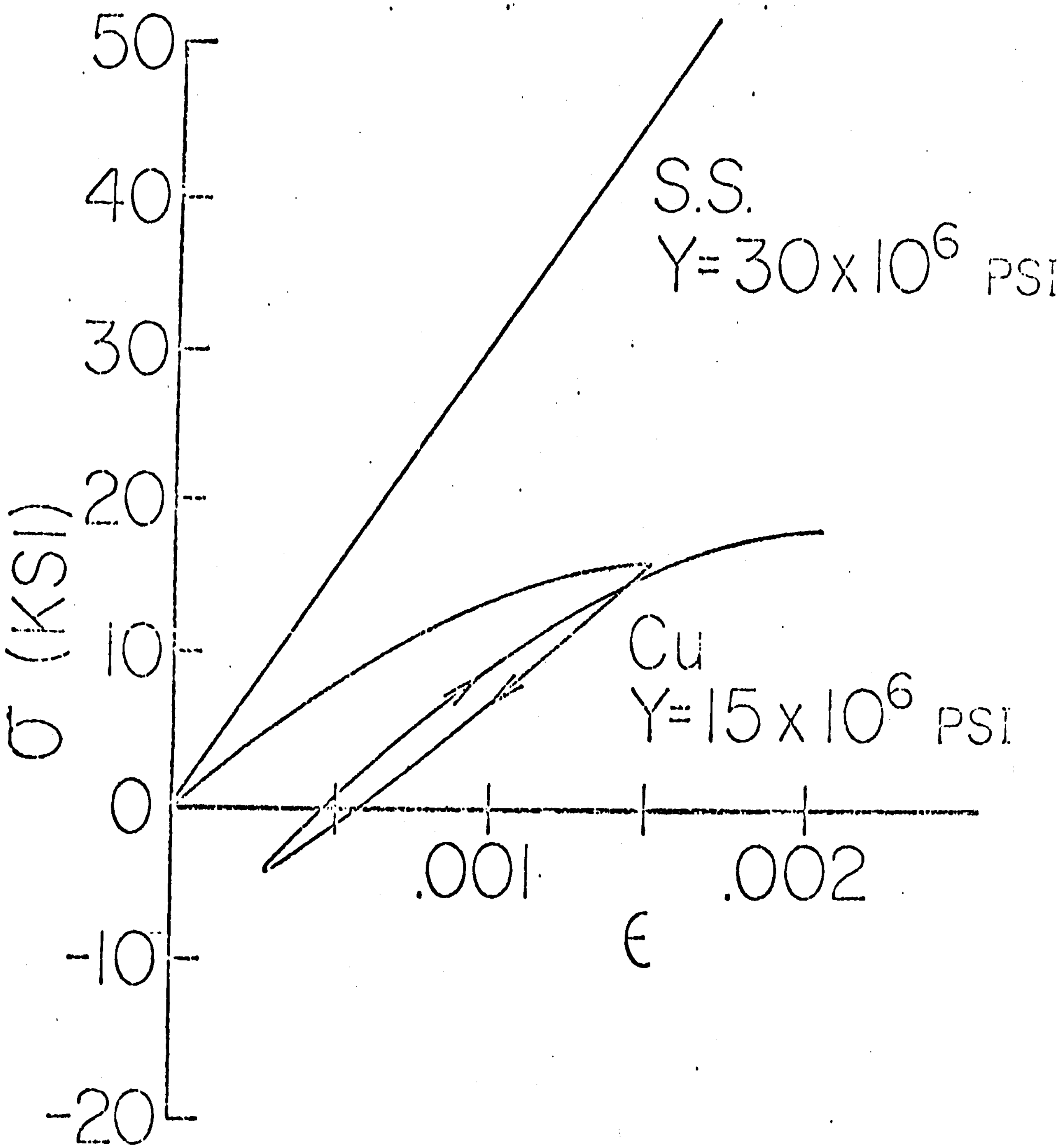

S.S.
Y=30x10^6 PSI
Cu
Y=15x10^6 PSI
σ (KSI)
50
40
30
20
10
0
-10
-20
.001
.002
ϵ