



# **Some Technological Problems in Fusion Reactor Design**

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## SOME TECHNOLOGICAL PROBLEMS IN FUSION REACTOR DESIGN

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### ABSTRACT

A review of possible fusion reaction cycles with the associated neutron production rates and energies is presented. Some of the problems in having to thermalize and capture these neutrons for D-T cycles are outlined in terms of likely neutron wall currents and resulting neutron fluxes. Due to the energy of these neutrons other problems arise because of elastic and inelastic collisions. Such collision rates and dpa for a "standard blanket" are presented. Problems in thermal and transient magnetic stress loading are also outlined.

## Introduction

Progress in the physics of plasma confinement during the past several years has led to speculation as to when one might be able to generate electricity by the fusion process. Several conferences <sup>1-3</sup> have been conducted to explore the technological aspects of fusion and almost all research oriented meetings devoted to any aspect of energy, its production, or distribution contain some elements of the fusion problem. It is, therefore, important that conferences such as this look toward the problems of fusion reactor design. In particular, the materials problems of such reactors look formidable and it is good to bracket these problems as well as possible in order to put them in perspective and keep them before us. In this paper we attempt to quantify some of the problems in fusion reactor designs and in particular those problems that are related to the neutron production, transport, and collisions in reactor blankets. Since this will be fundamental in seeking solutions to the materials problems of radiation damage; fatigue; stress cycling; hydrogen permeation, diffusion and solubility; helium embrittlement and so on, zeroth order solutions are instructive.

All near term approaches to fusion are concerned with burning deuterium either with itself or with another isotope of hydrogen (T) or helium ( $\text{He}^3$ ). Of the three reactions possible each has its advantage and one might think that the technological problems are dependent on the particular choice. For the area

of interest in this paper, such is not the case as we will see. Table I lists the reactions discussed along with two fission reactions useful in breeding tritium (T) in a blanket. Also listed are the ignition temperatures for the reactions where it is defined as that temperature at which the power radiated by bremsstrahlung is just balanced by the power produced from fusion. This requirement on power has its equivalent energy balance which leads to the Lawson condition<sup>4</sup> for the required minimum production of plasma density ( $n$ ) and confinement time ( $\tau$ ) and this too is listed for the various reactions.

From considerations on temperature and  $n\tau$  we see that the D-T reaction appears least demanding on the plasma physics and as a result, the first fusion reactors are likely to run on this cycle. One sees that this reaction and the D-He<sup>3</sup> reaction burns isotopes which are not naturally occurring. This is in fact a problem but T is easily produced by neutron captures in Li and He<sup>3</sup> is not so easily produced either by D-D reactions or the radioactive decay of T which has a halflife of 12.6 years. Most approaches to D-He<sup>3</sup> fusion are designed around D-D systems with the reaction product He<sup>3</sup> recycled as fuel.

The point of this is that any system employing deuterium will produce D-D neutrons and these must be thermalized and exothermically captured to be most useful. To do this, most conventional approaches to fusion assume a system similar to that shown in Fig. 1 where the hot plasma is coaxial to a

liquid metal blanket, shield and superconducting magnet. The blanket and shield are to thermalize and capture neutrons, and to produce heat and possibly fuel for the system. In the D-D system, Na with a capture energy or  $Q$  of 12.6 MeV is envisioned as the liquid metal coolant but in D-T systems Li is required to produce T and  $Q_{Li} = 4.8$  MeV.

The first wall facing the plasma not only receives the photon and gamma radiation load but all neutrons produced in the system must pass through this wall. Because of the possible high temperatures within which this wall must operate and because of the corrosive nature of the liquid metals, most reactor designs only consider the bcc metals, Mo, Nb and V together with their alloys, as candidate materials. Similar arguments hold for the structural materials within the blanket and shield region but perhaps these requirements are less severe.

### Neutron Wall Current

The three approaches listed in Table I can be compared with each other in terms of the energy and rate of production of reaction neutrons. To do this we use somewhat standard notation for the reaction rate per unit volume as

$$R_{DT} = n_D n_T \langle \sigma v \rangle_{DT} = \frac{n^2}{4} \langle \sigma v \rangle_{DT} \quad (1)$$

where  $\langle \sigma v \rangle_{DT}$  is the cross section of the D-T reaction averaged over the velocity distribution of the reactants. In these simple

calculations we take the plasma density ( $n$ ) as a constant for all the approaches and here  $n_D = n_T = n/2$ . Because D-D reactions can also occur in such reactors, we have the total reaction rate in a D-T system given by

$$R_{DT} = \frac{n^2}{4} \langle \sigma v \rangle_{DT} + \frac{n^2}{8} \langle \sigma v \rangle_{DD} \quad (2)$$

where the second term represents both branches of the D-D reaction. That is,  $\langle \sigma v \rangle_{DDn} + \langle \sigma v \rangle_{DDp} = \langle \sigma v \rangle_{DD}$  or the reaction rate for the branch which produces a neutron and the one for the branch to produce a proton combine to give the total reaction rate and the branching ratio is about equal for the temperatures considered. From this we know that each D-D reaction produces on the average, only half a 2.45 MeV neutron. We can combine Eq. (2) to give

$$R_{DT} = \frac{n^2}{4} \langle \sigma v \rangle_{DT} \left[ 1 + \frac{\langle \sigma v \rangle_{DD}}{2 \langle \sigma v \rangle_{DT}} \right] \quad (3)$$

and the first term represents the production rate of 14.1 MeV neutrons and the second term represents twice the production rate of 2.45 MeV neutrons.

Similar calculations can be made for D-D and D-He<sup>3</sup> reactions where one must determine the T and He<sup>3</sup> production rate from D-D reactions in a steady state to give

$$n_T = n_3 = \frac{n_D}{4} \frac{\langle \sigma v \rangle_{DD}}{\langle \sigma v \rangle_{DT}} \quad (4)$$



Using this the D-D reactions are obtained

$$R_{DD} = \frac{n^2}{2} \langle \sigma v \rangle_{DD} \left[ 1 + \frac{1}{2} \right] \quad (5)$$

where the first term is twice the 2.45 MeV neutron production and the second term is the 14.1 MeV neutron production from burning tritium of a density given by Eq. (4).

For the D-He<sup>3</sup> reactor, where we denote He<sup>3</sup> as 3, we find

$$R_{D3} = \frac{n^2}{4} \langle \sigma v \rangle_{D3} \left[ 1 + \frac{\langle \sigma v \rangle_{DD}}{2 \langle \sigma v \rangle_{D3}} \left( 1 + \frac{1}{2} \right) \right] \quad (6)$$

Here the first term represents the D-He<sup>3</sup> reactions which produce no neutrons, the second represents the D-D reactions which is twice the 2.45 MeV neutron production and the third term the D-T reactions which produce 14.1 MeV neutrons.

Viewing the results of Eqs. (3), (5) and (6) we can compare the production rate of 2.45 MeV neutrons to the total neutron production. For temperatures below 100 keV,  $\langle \sigma v \rangle_{DT} \gg \langle \sigma v \rangle_{DD}$  and we find that in D-T systems 99% of the neutrons are fast or 14.1 MeV. In the D-D and D-He<sup>3</sup> systems the ratio of slow or 2.45 MeV neutrons to the total is independent of temperature and because of the factor of 1/2 neutron per D-D reaction, we find these systems produce equal amounts of fast and slow neutrons.

Now that we know the ratio of fast to slow neutrons, let us investigate the total neutron production per unit of power for the three cycles. This cannot be done

for fixed density alone since it becomes a comparison of the coefficient in front of the brackets in Eqs. (3), (5) and (6) and essentially this amounts to a comparison of the various cross sections. We can, however, compare the neutron production per unit of power produced and to do this we will have to use the  $Q$  values for the various reactions and neutron captures. In the D-T reactor we use  $Q_{DT} = 17.1$  MeV and assume all neutrons are captured in  $\text{Li}^6$  where  $Q_{\text{Li}} = 4.8$  MeV. For the D-D and D-He<sup>3</sup> reactors we have  $Q_{DD} = 3.65$  MeV, averaged between the two branches, and  $Q_{D3} = 18.3$  MeV. Since no breeding is required here, we capture all neutrons in Na with  $Q_{\text{Na}} = 12.6$  MeV. Table II shows the power produced per unit volume for the three cycles discussed assuming a plasma density of  $1 \times 10^{14}/\text{cm}^3$  and various temperatures below 100 keV. Also shown is the number of 14.1 MeV neutrons, 2.45 MeV neutrons and total neutrons produced per sec, per thermal watt of reactor power. From this we see that the total neutron production is only about an order of magnitude different from one extreme to the other.

Let us now move on to determine the magnitude of the neutron current and limit our comments to the D-T reactor since it is the most abundant producer of neutrons. Whether we have an open system (mirror) or closed system (torus) the average 14.1 MeV neutron current ( $\phi$ ) per unit length for a cylindrical plasma of radius  $r_p$  and duty factor  $\eta$  can be written as

$$\phi 2\pi r_w = \frac{n^2}{4} \langle \sigma v \rangle_{DT} \pi r_p^2 \eta \quad (7)$$

where  $r_w$  is the wall radius and generally is related to the plasma radius by a constant such that  $r_p = \gamma r_w$ . We can assume for simplicity that both the plasma density and temperature are constant over the radius  $r_p$ . If it is not, one must integrate  $n^2 \langle \sigma v \rangle$  over the area and this can lead to reductions of the current by factors of up to 4 for parabolic  $n$  and  $\langle \sigma v \rangle$ .

Equation (7) can be put into more useful units and plotted as a function of the variable  $n^2 r_p \eta$  with the temperature or  $\langle \sigma v \rangle_{DT}$  as a parameter. Figure 2 is such a plot where the neutron current in neutrons/cm<sup>2</sup>-sec is given as a function of the plasma density in (particles/cm<sup>3</sup>)<sup>2</sup>, plasma radius in m and reactor duty factor ( $\eta$ ). Low  $\beta$  toroidal systems such as tokamaks<sup>5</sup> or stellarators<sup>6</sup> will operate at lower temperatures than mirrors and to distinguish the regions we crosshatch the various operating regimes. High  $\beta$  experiments such as the 0-pinch<sup>7</sup> and laser fusion devices<sup>8</sup> operate at higher densities and modest temperatures but have low duty factors. Estimates of these reactor systems are also indicated.

From the results of Fig. 2, it is easy to see that low  $\beta$  toroidal systems are likely to operate with wall currents somewhat lower than other fusion approaches. This is essentially because the reaction rate per unit volume is less - a fact which suggests this approach may be more expensive per unit power output than other alternatives.

breeding ratios as low as 1.05 may be acceptable if the tritium recovery time from the blanket is sufficiently short, and we will return to this point later. To further reduce the leakage flux to the superconductors surrounding the device, the shield will have to attenuate the total neutron flux by an additional factor of  $10^5$  or more.

The flux versus energy is plotted in Fig. 4 for the blanket of Fig. 3. Here the 14.1 MeV neutron current is 1 neutron/cm<sup>2</sup>-sec giving a total flux of about 10 neutrons/cm<sup>2</sup>-sec and an isotropic shell neutron source was assumed for this calculation. Three regions are shown, the first wall, the center of the 60 cm coolant region and the third wall. We see the spectrum softens appreciably in only about 30 cm of the blanket but is still almost as hard as that found in proposed LMFBRs.<sup>13</sup> Also note that the flux versus energy reverses below about .01 MeV and the neutron flux below this is higher in the third wall than that in the first wall.

Transmutation collisions are also important because the transmutation products may not be soluble in the host material. There are two major transmutation reactions in Nb: Nb(n, $\alpha$ )Y and Nb(n,p)Zr. The production of Y is small but it is marginally soluble in Nb. The production of Zr can be large and while it is soluble in Nb, the real problem in either case is the hydrogen and helium because of embrittlement and bubble formation.<sup>14</sup> In view of the possible problems from these elements, we list the (n,p) and (n, $\alpha$ ) reaction rates in Table III.<sup>15</sup>

### Blanket Activation Calculation

Some form of blanket must surround the reacting plasma to thermalize and capture the neutrons. A "standard blanket" configuration has been adopted for the D-T systems in order to compare calculational techniques and cross section data.<sup>9</sup> This design may not be applicable to the blanket systems for other fuel cycles, such as those employing Na, but it is appropriate for what we seek to do here. This standard blanket, shown in Fig. 3, is cooled by natural Li metal consisting of 94% Li<sup>7</sup> and 6% Li<sup>6</sup>. A first and second wall of 0.5 cm thick Nb are used for vacuum containment and to provide coolant channels. Niobium structural material is assumed to be homogeneously dispersed throughout the coolant on a 6 a/o basis. For calculational purposes a 0.5 cm thick Nb wall has been inserted just before the graphite reflector and this is referred to as the third wall.

The neutron flux and heating rate per unit volume shown in Fig. 3 are normalized to a neutron wall loading of 10 MW/m<sup>2</sup> or a current of  $4.4 \times 10^{14}$  neutrons/cm<sup>2</sup>-sec. These values are about a factor of 10 larger than anticipated in low  $\beta$  toroidal systems but close to those which may be found in large mirror systems. An anisotropic scattering Sn code (ANISN)<sup>10</sup> was used for these calculations along with ENDF/B cross section data files.<sup>11</sup> This blanket design has not been optimized but calculations suggest that the 100 cm thick blanket has a tritium breeding ratio of 1.3 and a leakage of 0.02. Vogelsang<sup>12</sup> has suggested that

Also listed in Table III is the dpa (displacements per atom) value in the corresponding regions. The dpa value is essentially the neutron flux times the damage cross section. This quantity is calculated from the elastic and inelastic damage energy deposition values given by Robinson<sup>16</sup> who uses a Kinchen and Pease model<sup>17</sup> for calculating the displacement efficiency. The dpa unit represents the numbers of times each atom is theoretically displaced from its lattice site. The actual number of displaced atoms remaining after an extended, high temperature irradiation is obviously much less than the dpa value but since we are looking at only relative effects, such a unit is useful.

Our calculations reveal that each atom of the Nb will be displaced 50 times per year or about the same as in proposed fast breeder reactors. Notice that the H and He production drops significantly in the third wall and the dpa scales about like the fast neutron flux of Fig. 4. These numbers are somewhat less than those presented by Martin<sup>14</sup> and Steiner<sup>18</sup> and are a consequence of two factors. The first is that Martin used a wall loading of  $3.7 \times 10^{14}$  n/cm<sup>2</sup>-sec and Steiner used  $4.4 \times 10^{14}$ /cm<sup>2</sup>-sec which are factors of about 5 larger than assumed here. A second reduction comes from the neutron source distribution assumed in the plasma. The work of Abdou and Maynard<sup>15</sup> suggests that differences in the source geometry can make variations as great as 50% in the inelastic collisions and over 100% in the dpa.

The previous results have all been for Nb because it is one of the first bcc metals where high energy cross section data have been developed for use in ANISN. Similar data will be available for other material and equivalent calculations are underway. In the meantime, some indication of the problem in other bcc metals can be gained by comparing the  $(n,\alpha)$  and  $(n,p)$  cross sections with those of Nb at a well developed energy like 14.1 MeV. Unfortunately, Mo has many isotopes but we find that  $\text{Mo}^{92}$  has the largest  $(n,p)$  cross section of  $100 \text{ mb}^{19}$  and several isotopes have  $(n,\alpha)$  cross sections of about 20 mb. We find the  $(n,p)$  cross section for V is about 36 mb and the  $(n,\alpha)$  is about 24 mb. Niobium, on the other hand, has a cross section for  $(n,p)$  of 25 mb and  $(n,\alpha)$  of 10 mb. Therefore, one might assume that Mo will contain a much higher a/o of neutron induced H after a modest lifetime when compared to Nb or V. This is especially true in view of the fact that the permeability of Mo for hydrogen is very low compared to that for Nb or V. On the other hand, V would have a larger helium concentration than Nb or Mo and this could present a more severe embrittlement problem. Before leaving the subject of transmutation reactions it should be mentioned that most of these reactions have an energy threshold which is above 3 MeV. For this reason, equivalent calculations for 2.45 MeV neutrons are not included. Generally the displacement damage for these neutrons is about half that for 14.1 MeV neutrons. On the other hand, the final material choice may not be made on any of the above considerations but on consideration of afterheat.<sup>20</sup>

Since hydrogen is one of the subjects of our concern we should explore the problem of the production of tritium and helium in D-T reactors by  $\text{Li}(n,\alpha)\text{T}$  reactions. For radiological and economic purposes it will be essential to remove the T as rapidly as possible to maintain a minimum inventory of fuel.<sup>12</sup> However, it is good to point out that the ease of recovery of hydrogen at low concentrations in lithium is an unknown quantity.<sup>21</sup> If we are able to reduce the T concentration in the Li blanket to 10 ppm, this would also be the limiting concentration in the structure because of the high temperatures and generally large diffusivity of T. Therefore, the minimum concentration of this fuel in the breeding Li metal may be a major materials consideration. If the fused salt  $2\text{LiF}-\text{BeF}_2$  (FLIBE) is used as the blanket material, the breeding ratio is reduced but probably acceptable, and the minimum concentration of T is also reduced because of the difference in activity. On the other hand, the corrosion problems in a FLIBE system are probably increased.

The heat deposited in the blanket varies with radius as in Fig. 3 and consequently the temperature gradient within the blanket will always be towards the plasma. Logically, the first wall will be at the highest temperature. If we are considering a toroidal system, there will not be equal heating around the poloidal direction and the inner surface first wall will likely be warmer than the first wall at a larger major radius. This is due to two effects. The first is that the inner surface, with a lower average major radius, is exposed to an isotropic neutron source while the outer surface is not. Consequently,



neutron heating should be greater as pointed out by Abdou and Maynard.<sup>15</sup> A second problem arises from the bremsstrahlung and  $\gamma$  radiation. These sources are isotropic but again, for small aspect ratio toroids, the surface area of the inner wall is significantly less per unit volume of plasma than that of the outer wall. Estimates of the first wall bremsstrahlung heating load suggest that it will likely be several times the heating due to neutrons. Consequently temperature gradients will exist both within the first wall and in the poloidal direction around the wall. These problems may increase the buckling stress but perhaps more importantly will increase the bubble migration within the material as well as the hydrogen and tritium concentrations within the warmer zones. Both processes lead to embrittlement and suggest severe limitations on temperature cycling.

#### Miscellaneous Reactor Considerations

There are any number of reactor problems that we do not foresee at the present time but which will eventually become important as systematic designs are undertaken. A whole host of special problems occur at the first wall because of energetic particles and waves which strike it. Such problems have been summarized and we will not review them here.<sup>22</sup> On the other hand, there are some problems that relate more to the purpose of this paper and should be mentioned.

One such problem arises in pulsed toroidal systems where the plasma is required to carry a current and perhaps the worst case occurs in pinches. The pinch reactors of Los Alamos<sup>7</sup> or Culham<sup>23</sup> envisions carrying sufficient current in the transient coil to produce fields of 141 kG for half cycle periods of 0.067 sec. This represents a transient expanding force of 800 atmospheres and is a major concern. Tokamaks do not require the first wall to carry a driving current but the wall must carry the plasma image currents. These may have periods of  $10^2$  to  $10^4$  longer so the field will soak through the first wall and the load will be distributed within the blanket. Rough estimates suggest this expanding pressure will be less than 4 atmospheres but in these systems the wall area is ten times larger and this represents a significant force. This is especially true when one considers that between pulses, atmospheric pressure reverses the stress.

The driving or image currents in the first wall also represent a significant heat load to the wall. Ribe<sup>7</sup> has calculated this for a 0-pinch reactor and has employed special cooling systems to keep the resistance down. In tokamak systems<sup>5</sup> approximately  $0.9 \times 10^6$  A/m image current will be transiently induced into the first wall and blanket. The L/R time for the inner wall is less than the outer wall but the image current is greater in the inner

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wall. Thus it seems the inner wall will operate significantly warmer than the outer wall for this and other reasons.

To get some idea of the importance of this type of problem, in the tokamak reactor of Golovin et al.<sup>5</sup> a current of  $1.1 \times 10^7$  amperes is induced in the wall and blanket. If the pulse length of the system is 20 sec, the skin depth is about 1 m and some  $1.2 \times 10^9$  joules are dissipated in the structure. This also suggests that the possibility of aligning liquid metal cooling channels parallel to the magnetic induction is virtually impossible because of the time dependent soak in of the field.

The MHD pumping problem with liquid metals has suggested that He gas cooling should be a solution. Unfortunately, the compatibility of bcc refractory metals with the impurities that will surely be carried by the He is not good. J.H. DeVan of ORNL reminds us of the work at JPL where 1 ppm impurities of  $O_2$  and  $H_2O$  in a recirculating argon loop using Nb-1%Zr was found to contain 4480 ppm oxygen plus nitrogen after 1000 hours of operation at  $1093^\circ C$ .<sup>24</sup> These and other yet unforeseen problems will surely make the task of designing fusion reactors challenging.

## Conclusions

Fusion reactors will present a new set of design problems for reactor engineers. However, before such designs can be undertaken, a large amount of additional information will be necessary to determine radiation, temperature, impurity and stress effects on reactor materials. In this survey we have found that the neutron flux can be expected to range from about  $10^{14}$  to  $10^{16}$  neutrons/cm<sup>2</sup>-sec depending on the confinement scheme. The operating temperatures for material components is expected to be between  $0.3 T_m$  and  $0.5 T_m$  in various regions of reactor first walls. Modest to large concentrations of H, T and  $\alpha$  particles may cause severe problems in structural members of the blanket that will be repeatedly cycled both in stress and temperature.

While bcc refractory metals are prime candidates for fusion reactor materials, little is known about these metals under high neutron fluences. Alloys of these metals, and others that bring together the desirable properties of each, need to be explored for both the problems considered above, corrosion compatibility and the ease of fabrication. Reasonable efforts begun now in parallel with light water and fast breeder reactor materials can have a significant payoff in the future.

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Table I

<u>Reaction</u>	<u>Ignition Temperature</u>	<u>nr</u>
$D + T \rightarrow He^4(3.5) + n(14.1)$	4.5 keV	$5 \times 10^{13} \text{ cm}^{-3} \text{ sec}$
$D + D \rightarrow He^3(0.8) + n(2.6)$	28 keV	$10^{15} \text{ cm}^{-3} \text{ sec}$
$\quad \quad \rightarrow T(1.0) + p(3.0)$		
$D + He^3 \rightarrow He^4(3.7) + p(14.7)$	22 keV	$7 \times 10^{14} \text{ cm}^{-3} \text{ sec}$
$n + Li^6 \rightarrow He^4(2.1) + T(2.7)$	--	--
$n + Li^7 \rightarrow n + He^4 + T(2.5)$	--	--



Table II

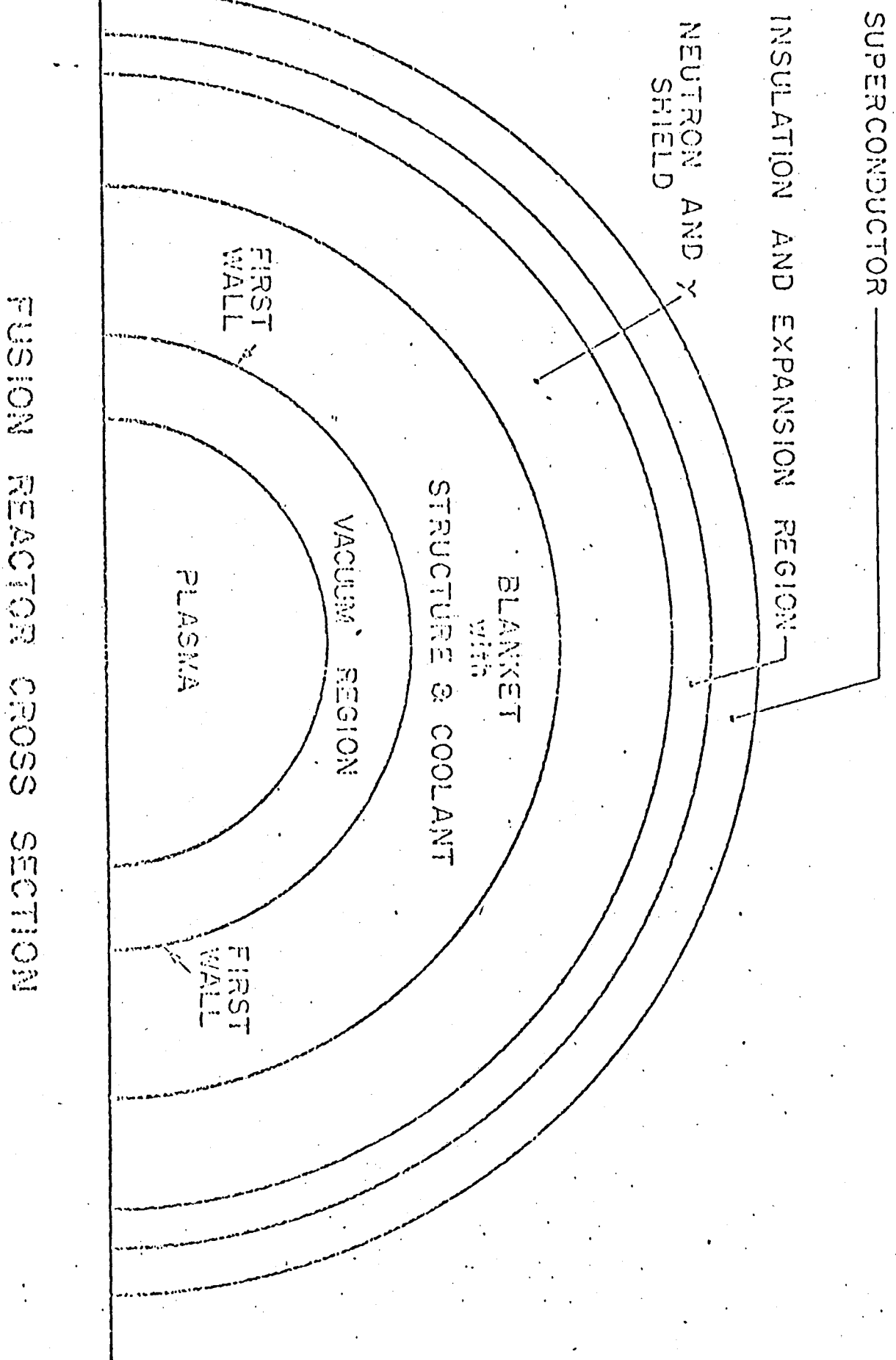
Power Densities and Neutron Production Rates for Various Cycles and Temperatures

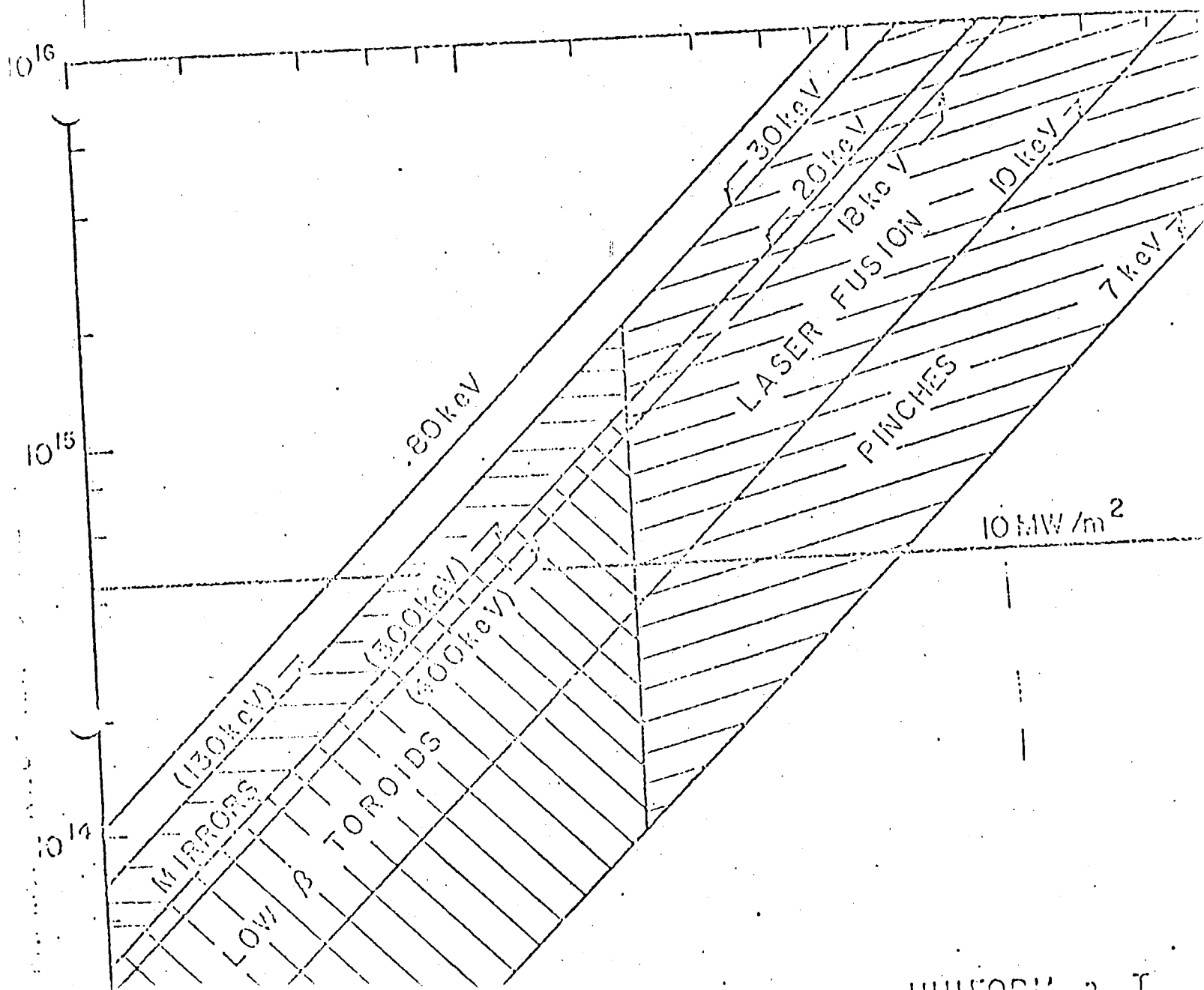
<u>T(kev)</u>	<u>Cycle</u>	$\frac{P}{V} \left( \frac{\text{watts}}{\text{cm}^3} \right)^*$	$\frac{14.1 \text{ MeV n/sec}}{\text{watt}}$	$\frac{2.45 \text{ MeV n/sec}}{\text{watt}}$	$\frac{\text{Total n/sec}}{\text{watt}}$
10	D-T	0.98	$2.8 \times 10^{11}$	$5.5 \times 10^8$	$2.8 \times 10^{11}$
20	D-T	3.84	$2.8 \times 10^{11}$	$5.9 \times 10^8$	$2.8 \times 10^{11}$
	D-D	0.01	$9.1 \times 10^{10}$	$9.1 \times 10^{10}$	$1.8 \times 10^{11}$
	D-3	0.04	$5.6 \times 10^{10}$	$5.6 \times 10^{10}$	$1.1 \times 10^{11}$
60	D-T	7.78	$2.8 \times 10^{11}$	$1.3 \times 10^9$	$2.8 \times 10^{11}$
	D-D	0.44	$9.1 \times 10^{10}$	$9.1 \times 10^{10}$	$1.8 \times 10^{11}$
	D-3	0.59	$1.7 \times 10^{10}$	$1.7 \times 10^{10}$	$3.4 \times 10^{10}$
100	D-T	7.27	$2.8 \times 10^{11}$	$2.6 \times 10^9$	$2.8 \times 10^{11}$
	D-D	0.82	$9.1 \times 10^{10}$	$9.1 \times 10^{10}$	$1.8 \times 10^{11}$
	D-3	1.40	$1.3 \times 10^{10}$	$1.3 \times 10^{10}$	$2.6 \times 10^{10}$

\* For a plasma density of  $1 \times 10^{14} / \text{cm}^3$

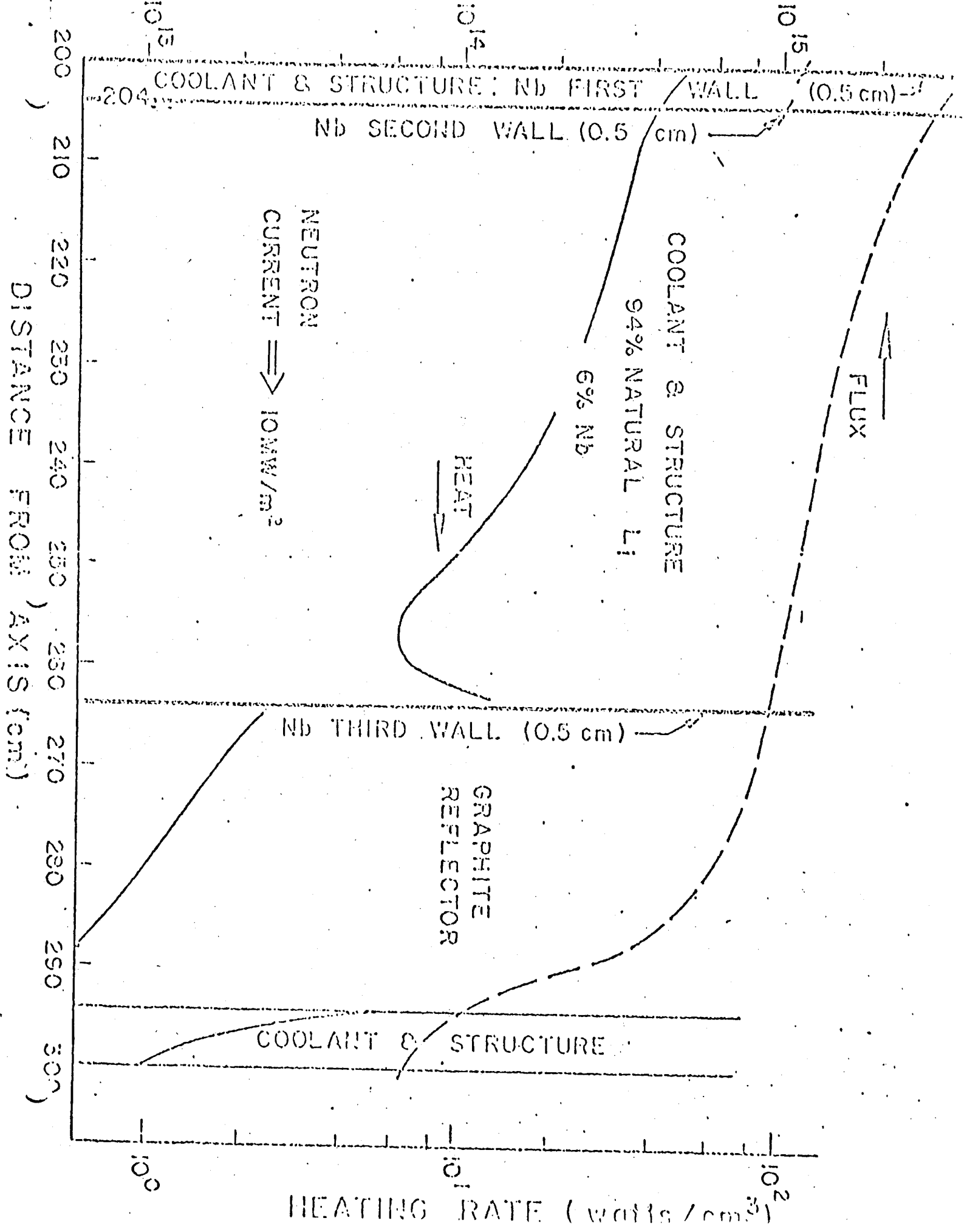
Table III

<u>Region</u>	<u>(n,p)</u>	<u>(n,<math>\alpha</math>)</u>	<u>dpa</u>
1st wall	$1.76 \times 10^{-3}$	$5.31 \times 10^{-4}$	$14.8 \times 10^{-21}$
2nd wall	$1.14 \times 10^{-3}$	$3.36 \times 10^{-4}$	$11.1 \times 10^{-21}$
3rd wall	$0.43 \times 10^{-3}$	$0.13 \times 10^{-5}$	$0.93 \times 10^{-21}$





NEUTRON FLUX (n/cm<sup>2</sup>-sec)



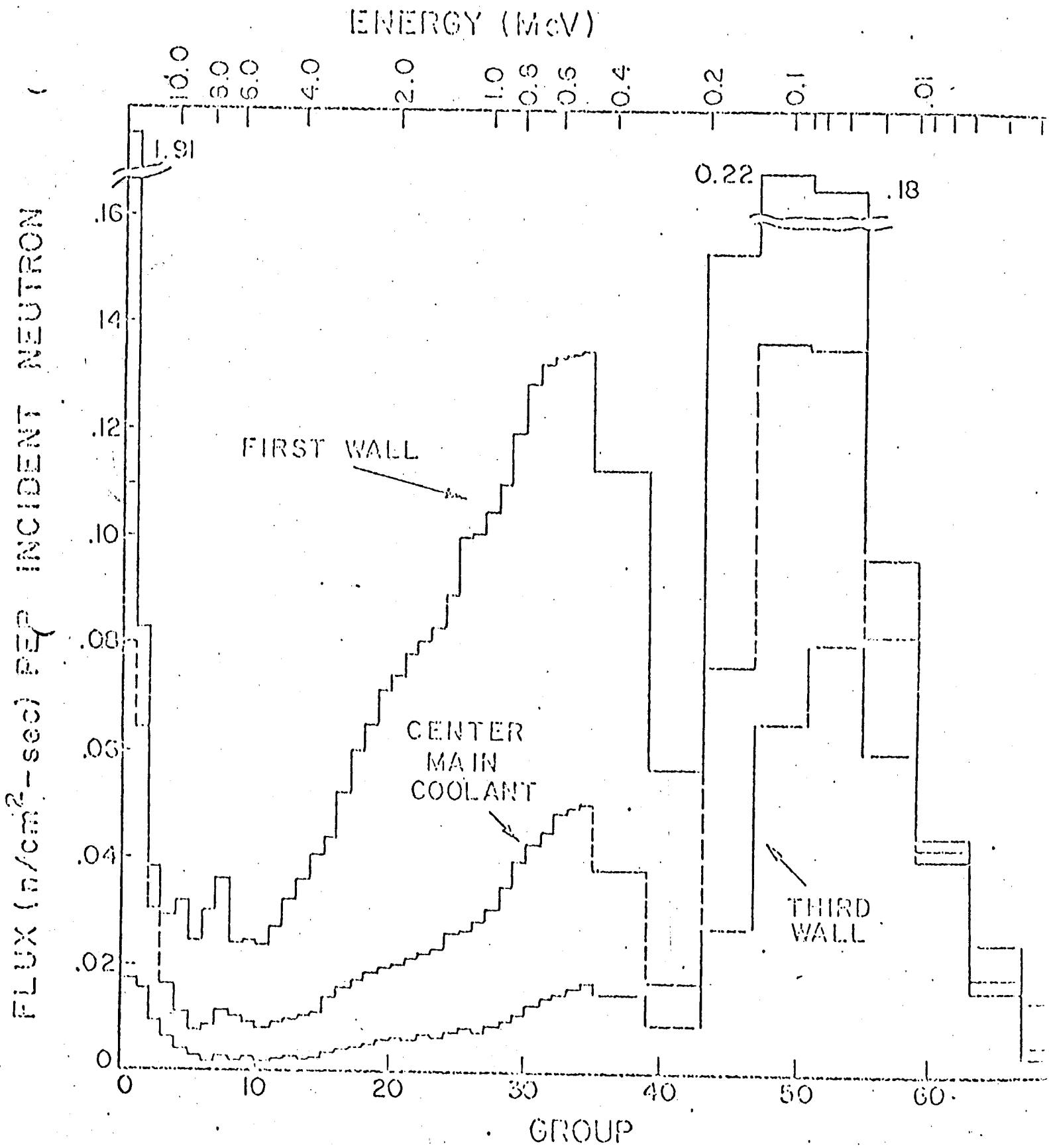


Figure 1 Typical fusion reactor cross section showing the plasma, vacuum or divertor region, first wall, lithium containing blanket with its support structure, neutron and  $\gamma$  shield for the magnets, a free space for insulation or expansion and the superconductors with their supporting structure.  
(first listed - page 2)

Figure 2 14.1 MeV neutron current to the first wall as a function of the reactor parameters for a D-T system. The reactor is characterized by the product of the square of the plasma density ( $\text{cm}^{-3}$ )<sup>2</sup>, plasma radius (m), ratio of plasma to wall radius ( $\gamma$ ) and duty cycle ( $\eta$ ). Plasma temperature is also a parameter where uniform density and temperature are assumed across the plasma. For mirror reactors a mirror ratio of 3.3 is assumed and a reference value of  $10 \text{ MW/m}^2$  is also shown. (first listed - page 7)

Figure 3 Total neutron flux and energy deposition rate for a "standard blanket" of 100 cm thick. For this blanket a 14.1 MeV neutron current corresponding to  $10 \text{ MW/m}^2$  is assumed. A third wall is added for calculation of damage rates. (first listed - page 8)

Figure 4 Total neutron flux as a function of energy in the first wall, center of the main coolant region and the third wall for a 14.1 MeV neutron current of  $1 \text{ neutron/cm}^2\text{-sec}$ . The calculation is for a "standard blanket" as modified in Figure 3 for a third wall.  
(first listed - page 4)