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EFFECT OF PLASMA PHYSICS ON CHOICES OF FIRST WALL MATERIALS AND STRUCTURES FOR A THERMONUCLEAR REACTOR

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ABSTRACT

Impurity ions adversely affect the behavior of present-day tokamaks, and control of impurities is expected to be a key element in determining the feasibility of thermonuclear fusion reactors. In this paper, we describe the plasma-surface interactions for tokamaks and several techniques for controlling impurities. The plasma-surface problem of next generation devices PLT, PDX, DIII and TFTR is expected to be similar to those encountered in a reactor. For these devices calculations indicate that most of the particle energy efflux will be in the 1 keV region. Ironically this energy region has not yet been investigated thoroughly by the surface physicists.

I. INTRODUCTION

The three major plasma physics problems associated with developing a controlled thermonuclear reactor are:

(1) providing adequate plasma confinement, that is $n\tau \gtrsim 10^{14}$ sec - cm^{-3},

(2) heating the plasma into the 10-20 keV range and

(3) maintaining sufficiently low plasma impurity levels.

Solutions to the first two problems appear to be in hand with the very encouraging tokamak results of good confinement [1,3] and plasma heating using neutral beams [4] and lower hybrid resonance waves [5]. However, the most advanced present-day tokamak [6], TFR, has ominously large impurity concentrations which produce $\bar{Z} = \Sigma_j Z_j^2 n_j / n_e \approx 5$. Impurities are of two types [7], low-Z ions (O,C) which are desorbed from the first wall surface by plasma bombardment and high-Z ions, (Fe, Mo and W) which are probably produced by sputtering of the limiter and the first wall. Sputtering of the first wall and the limiter will become a more severe problem in the next generation devices such as PLT, T-10 and TFTR as the ion energy increases. High-Z impurity ions are particularly bad since they are not fully stripped in the hot plasma core and therefore radiate energy copiously. On the other hand low-Z impurities are somewhat less troublesome since they emit line radiation only at the plasma surface. The basic principles of tokamaks are described in Section II. In Section III we describe the particle and energy fluxes incident on the first wall of tokamak reactors, followed by the effects of impurities on plasma and reactor behavior in Section IV, proposed methods of impurity control are described in Section V followed by areas of surface material research in Section VI.

II. BASIC PRINCIPLES OF TOKAMAK OPERATION

In this report, we will be concerned mainly with plasma surface interactions in a tokamak. The important features of a tokamak are illustrated in in Fig. 1. Plasma confinement is provided by the poloidal magnetic field, B_p, which is produced by the toroidal plasma current I_p. The confinement capability of a tokamak is specified mainly by the magnitude of the plasma current. Plasma current is induced by passing a time-changing magnetic flux, I_{OH}, through the plasma current loop. For this purpose, the vacuum vessel must have an insulated gap or high resistivity to prevent the vacuum vessel from shorting out the induced electric field. Because of this induction technique, the pulse length for plasma current is limited; for typical reactor parameters, this can be as long as 10^4 seconds. In addition a vertical field, B_V, must be supplied by external coils to provide toroidal equilibrium for the plasma ring. Since the plasma current varies in time, the vertical field must also vary during the plasma current rise. This introduces an additional constraint on the allowed conductivity of the vacuum vessel.

One of the more severe constraints on a tokamak is the large toroidal field, B_T. Both experimentally and theoretically, it has been demonstrated that plasma stability requires that the toroidal field must satisfy

$$B_T = q\left(\frac{R}{a}\right) B_p = \frac{qR}{2\pi a^2} \mu_o I_p \ , \tag{1}$$

where q the safety factor is ≥ 1, a is the plasma minor radius and R is the plasma major radius. We desire the largest possible I_p for plasma confinement and therefore we work at toroidal fields that are limited by mechanical strength considerations which gives $B_t \simeq 50$ kG in the plasma.

An important feature of present-day tokamaks is the plasma limiter which is designed to prevent plasma from hitting and destroying the vacuum vessel. Most tokamaks use simple disk shaped poloidal limiters which are made from refractory metals.

Tokamak plasmas are heated into the 1-2 keV region by the ohmic heating produced by the plasma current. Heating to reactor temperatures will require intense neutral beams or radio-frequency heating.

III. PARTICLE AND ENERGY FLUXES TO THE FIRST WALL OF A TOKAMAK REACTOR

Particle and energy fluxes to the first wall of a tokamak reactor depend on the methods used for plasma heating, impurity control and refueling. Next generation devices such as PLT, PDX, DIII, and TFTR have plasma parameters (Table I) and use heating and impurity control techniques that will be similar to those of reference reactor designs such as the Princeton Reference Design [8] or the U-MAK [9] design. An operating tokamak reactor will be heated to reacting temperatures by neutral beams and or radio-frequency waves. Plasma refueling will take place by either injection of solid D-T pellets, injection of energetic neutral beams as in the proposed Colliding-Ion-Torus [10] or by cold gas at the plasma surface as is done in present-day tokamaks.

Plasma heating interacts with first wall materials choices in a rather minor way. Energetic neutral beams of $\gtrsim$ 200-300 keV are required to penetrate reactor-size plasmas, damage to the first wall near the injector port may be significant due to blister formation. On the other hand RF heating requires wave launching structures near the first wall in the case of ion-cyclotion resonance heating. The method of plasma refueling has a rather large effect on the particle and energy fluxes to the first wall. The present refueling technique simply allows the escaping plasma to be neutralized at the wall or limiter and the resultant cold neutrals flow back to the surface of the plasma where they are reionized. Calculations indicate that this process produces a substantial flux of warm (0.1 to 2.0 keV) charge-exchange neutrals at the first wall when the central ion temperature is $\simeq$ 10 keV. These neutral atoms are in the range of maximum sputtering coefficient and can produce significant amounts of impurity influx. Perhaps the charge exchange problem can be reduced by the injection of D-T pellets into the reacting plasma core or by the injection of energetic beams. A variety of techniques have been proposed to extract the plasma energy without poisoning the plasma with impurities, these methods will be discussed in Section V.

The particle flux, assuming uniform deposition to the first wall of a reactor without a divertor, can be estimated as follows:

$$\phi_p = \frac{n_e V_p}{\tau_p A_w} \doteq \frac{n_e^2 a}{2 n_e \tau_p} \sim 5 \times 10^{15}/\text{cm}^2\text{-sec} \qquad (2)$$

where we have taken typical values [9] of $\tau_p \simeq 6$ sec, $n_e = 10^{14}\,\text{cm}^{-3}$ and a = 500 cm. Similarly, the energy flux on the first wall due to escaping plasma is given by

$$\phi_E = \frac{3 n_e T_p V_p}{\tau_E A_w} = 3\phi_p T_p \frac{\tau_p}{\tau_E} \sim 72\text{w/cm}^2 . \qquad (3)$$

This energy flux is comfortably low compared to thermal cooling and thermal stress limits of ~ 1 kw/cm^2. However, we must be concerned about non-uniform deposition particularly during startup and termination of the discharge. Present-day tokamaks always have metallic limiters to protect the vacuum vessel from destruction due to disruptive instabilities or runaway electron beams. These limiters can be either a disc in the poloidal direction or the toroidal direction. Present-day tokamaks have poloidal limiters, For this type of limiter the power density is given by

$$\phi_{E,L} = \frac{3n_e kT_p V_p}{\tau_E 2\pi a\lambda} \tag{4}$$

where λ, the plasma penetration distance into the limiter shadow, is $\lambda 2 = D_\perp \tau_{\parallel}$, $\tau_{\parallel} = \pi R/V_s$ is the time for a particle to flow parallel to the field into the limiter and R is the major radius of the tokamak. For typical reactor parameters, the energy density if entirely absorbed on one poloidal limiter would be $\simeq 10$ kw/cm^2. This power loading is very high and presents one of the most severe materials problems for today's tokamaks. Limiters from the ST tokamak were severly damaged in many cases probably due to runaway electron bombardment or nonuniform plasma deposition during discharge termination. Next generation tokamaks such as PLT propose to solve this problem by using spinning or specially shaped limiters to distribute the heat load over larger areas. In divertor tokamaks, such as PDX, the effective limiter area is increased by a factor of 16 over PLT so that the heat deposition problem is manageable. Another solution to the limiter heat problem may be to use flowing liquid lithium films [9]. However, the possible high vapor pressure of these films remains as worry.

The tokamak radial transport computer code [11] has been used to estimate, in more detail, the energy distribution of the particles that are incident on the first wall of the TFTR tokamak. For this particular calculation the plasma parameters were plasma current = 2.5 MA, major radius = 2.5 m, minor radius = 0.85 m, toroidal field = 52 kG, a constant low-Z impurity level corresponding to $\bar{Z} = 2$ with a deuterium beam injected at 154 keV and 12 MW into a tritium plasma. The boundary temperature is arbitrarily maintained at 10 eV and the plasma transport is determined from a six-regime turbulent diffusion model. The profiles of electron density, electron and triton temperature and neutral gas density are shown in Fig. 2 at t = 384 msec after discharge initiation. Notice that the edge region temperatures are considerably lower than the central values. The edge region (~ 10 cm) of TFTR is probably very similar to the outer 10 cm of today's tokamaks. That is, the plasma deposits energy into the limiter with electron temperature of ~ 50-100 eV and therefore, due to the sheath effect, ions hit the limiter at an energy $kT_i + ZeV_s$ = 200-400 eV. Note also that the charge exchange neutrals produced during the recycling and ionization processes in TFTR are estimated to have a spectrum (Fig. 3) that is peaked just below 1 keV, which is close to the peak of most sputtering coefficients as a function of energy. The fast beam ions should be slowed from 154 keV to near the background plasma temperature before they hit the wall or are lost by charge exchange. In TFTR, roughly 10% of the fast ions are lost by charge exchange, that is, we have in addition to the background plasma energy fluxes on the wall an additional $\simeq$ 10% due to $\simeq$ 60 keV deuterons. The uniformity of these losses remains to be determined. In a reactor, the neutral beam problem will be similar. Alpha particles will be formed in the central hot core region and due to the high plasma current required for the plasma confinement, the alphas will also be well confined and should slow down from 3.5 MeV to near the background plasma regime. As a result, the concentration of research effort on blistering due to 3.5 MeV alphas seems unwarranted. The estimates of particle fluxes to the first wall of TFTR are given in Table II.

IV. EFFECTS OF IMPURITIES ON PLASMA AND REACTOR BEHAVIOR

The primary effects of impurities on plasma behavior are:

(1) enhancement of energy losses,

(2) increase of plasma collision frequencies, and

(3) reduction of fuel ion density.

Plasma energy losses are enhanced by impurities due to:

(1) ionization of impurities,

(2) line radiation from partially stripped ions,

(3) recombination radiation from fully stripped ions, and

(4) bremsstrahlung radiation

The power loss due to ionization of impurity ions can be expressed as $P_I = n_z W_z/\tau_z$ where n_z is the impurity density, W_z is the energy required to ionize the ion and is at most a few kT_p and τ_z is the confinement time or influx time for the impurity ions. This power will be compared with the power due to plasma losses, $P_p = 3n_e kT_p/\tau_p$. The ionization power is typically only 10^{-2} of the plasma loss. The ionization state of impurity ions in a large tokamak is determined by coronal equilibrium in which electron collisional ionization is equal to radiative recombination. In coronal equilibrium the impurity level is a function mainly of electron temperature, neglecting the effects of dielectronic recombination. For reactor plasmas with T_e = 15 keV. we find that ions with $Z < 42$ are effectively fully stripped. The radiation from highly stripped high-Z ions is not accurately known and we will use some simple estimates for radiation losses. The radiation from ions with at least three electrons remaining is estimated [1,2] to be given by

$$P_L \doteq 2 \times 10^{-26} n_e n_z \quad w/cm^3 \ . \tag{5}$$

As the ions are stripped into helium-like and hydrogen-like ions, the line radiation drops significantly. However, the recombination radiation becomes important for highly-stripped, high - Z ions. For highly-stripped ions the hydrogen approximation [13] gives the recombination power radiated as

$$P_R = 1.3 \times 10^{-32} Z^4 n_e n_z T_e^{-1/2} \quad w/cm^3 \ , \tag{6}$$

where T_e is in keV, and Z is ionization state. Recombination to form lithium-like ions emits roughly 50% less energy for the cases of interest. For fully ionized impurities, the recombination radiation is more important than free-free bremsstrahlung from impurities when $Z^2 > 38T_e$. In the fusion reactor temperature range of 15 keV, recombination is more important than bremsstrahlung when Z > 25. Total radiation power losses and charge state for Fe impurities in a thermnuclear plasma are illustrated Fig. 4 as a function of electron temperature [14].

Impurity ions also increase the coulomb collision frequency for the electrons, this increase is proportional to $\bar{Z} = \Sigma Z_j^2 n_j/n_e$ and is typically 3-5 in existing tokamaks. The radial distribution of impurity ions can affect the radial current distribution and hence the plasma micro and MHD stability. Ohmic heating power input is increased by the factor $\bar{Z}$ since the plasma current is limited by stability considerations. Impurities will also have an impact on neutral beam heating. For example, fast ions injected parallel to the field are scattered into trapped particle orbits Z times faster than in a pure plasma [15]. But most importantly, in a reactor-like plasma, when the injected neutral beam energies are 100-300 keV, ionization of the beam by impact ionization on the impurity ions [16] may cause the beams energy to be deposited on the edge of the plasma. If this is so then large fluxes of 100 keV particles may be striking the wall of the reactor near injector ports.

In the case of RF heating, it is possible for impurity ions near the plasma surface to be heated efficiently and to then bombard the limiter or wall at relatively high energies thereby releasing more impurities. This phenomenon may have led to the energy limits on ICRH heating experiments in the ST tokamak [17].

One of the most damaging effects of impurities is the depletion of fuel ions. Since the total plasma pressure is limited, and since every impurity ion with charge Z must be neutralized by Z electrons, the fuel ion density is substantially reduced. The reduction in fuel ion density is given by:

$$n_i/n_i(f = 0) = (1 - fZ)/(1 + f) \quad , \quad f = n_z/n_e \ , \tag{7}$$

and the power production goes down as the square of this ratio. Again, in typical operating tokamaks, the fuel ion density is reduced by a factor of 2.

Impurities have a significant effect on reactor behavior, high gain reactors or ignition devices are most severly affected. The effects of high-Z impurities on the $n\tau$ required for ignition has been calculated [18] for a homogeneous steady-state 50-50 D-T reactor (Fig. 5). The curve labeled 0 is the $n\tau$ required to ignite a pure plasma. An impurity level of 0.1% Mo is sufficient to substantially increase the required $n\tau$ in the 12-15 keV range where many reference reactors are scheduled to operate. The effect is even more drastic for higher Z impurities such as tungsten. These calculations indicate that ignition was impossible above impurity levels of 0.8 Mo and 0.25% W. Impurities do not effect sub-ignition devices as much, and it is possible (Fig. 6) to run the reactor as a simple power amplifier with large circulating power in the presence of relatively high impurity levels. This point has also been emphasized by Conn et al [19] in connection with a two-component power amplifier where the impurities have little effect for Q (energy amplification) ≈ 1, but become very important for Q ≈ 5-10. So that in any real power producing reactor, where Steiner [20] has estimated that Q must be ≈ 10, impurity control will be necessary while impurities will not be as important in a small scale physics experiment such as TFTR which has Q ≈ 1.

V. METHODS OF IMPURITY CONTROL IN TOKAMAKS

The previous section described the need to maintain impurity levels at rather low operating levels during the tokamak pulse which lasts several (~ 1000) plasma particle confinement times.

The impurity level in a steady-state confined plasma without a divertor, due to charge-exchange neutral and ion sputtering of the walls and limiter, is given approximately by

$$f = \frac{n_z}{n_e} = \frac{\tau_z}{\tau_p}\left[\frac{S_n\langle\sigma_{cx}V_i\rangle}{(2\langle\sigma_i V_e\rangle + \langle\sigma_{cx}V_i\rangle)} + S_i\right](1-e^{-t/\tau_z}) \qquad (8)$$

where τ_z and τ_p are the impurity and particle confinement times, S_n and S_i are the charge-exchange neutral and ion sputtering coefficients, and $\langle\sigma_{cx}V_i\rangle$ and $\langle\sigma_i V_e\rangle$ are the charge-exchange and ionization rates, and we have assumed that 1/2 of the fast charge-exchange neutrals are absorbed. When the edge region plasma temperature is in the 100 eV range, the rate coefficients give an impurity level due to charge-exchange of

$$f(CE) = 0.4\frac{\tau_z}{\tau_p}S_n(1-e^{-t/\tau_z}) , \qquad (9)$$

which is similar in magnitude to the results of the complex computer code. Note that for $\tau_z \approx \tau_p$, the equilibrium impurity level is $0.4\ S_n$, which for the charge-exchange energies expected for a reactor, namely ~ 1 keV, predicts impurity levels of 0.1 - 1.0%. This impurity level would be large enough to significantly reduce reactor performance. If $\tau_z/\tau_p \gg 1$ as classical collisional plasma transport predicts, then some means must be found to reduce the impurity generation, the inward transport of impurities or the effect of the impurities on reactor behavior.

Impurity generation can be reduced by lowering the temperature of the plasma edge. Perhaps this can be done in a tokamak with a cold gas blanket surrounding the tokamak. Present studies [11] of these gas blankets using the transport codes indicate that the edge plasma (10-20 cm from the wall) can be kept cold (below the sputtering threshold of 50 eV) for times the order of one energy confinement time. This process might be helpful for short pulse devices such as TFTR but will not allow long pulses as envisioned in tokamak reactor design [9].

Other techniques for reducing impurity generation involve using materials with a low sputtering coefficient or using special wall structures such as the honeycomb structure proposed by ORNL [21]. These techniques are estimated to reduce the impurity generation by factors of 3 - 5 and will be tested in the ISX device [22] now under construction.

Impurities are present very early in the discharges of present-day tokamaks, these early impurities are oxygen, which is probably due to desorption during the initial gas breakdown phase, and wall or limiter material which is probably due to the discharge running into the limiter or wall during the first stages of current rise. The desorption of oxygen can probably be reduced with lower base pressures or better surfaces for the vacuum wall, or by evaporating titanium over the vacuum wall surface just prior to the plasma pulse and thus burying or chemically pumping O_2, CO_2 and H_2O. This latter technique was very successful in reducing impurities in the ATC tokamak [23]. However, sputtering or evaporation due to beams of runaway electrons formed during the initial part of the discharge is still a problem, this may be solved by using a magnetic limiter to force the initial plasma discharge to form away from the metallic limiters and vacuum wall. In addition, the magnetic limiter can be programmed to expand with increasing plasma current so that the plasma cross-sectional area is proportional to the plasma current. In this case, the plasma current density is constant in space and the skin effect and associated MHD instabilities which drive plasma into the wall and limiter during plasma startup are avoided.

One of the most powerful methods to reduce impurity generation is the magnetic divertor, which can be combined with the magnetic limiter previously described. A poloidal divertor for a tokamak is shown in Fig. 7. In this device the escaping plasma is channeled magnetically into a remote burial chamber where the neutralized plasma is rapidly pumped on titanium surfaces. In this case, sputtering due to plasma ion and charge-exchange neutral bombardment of the vacuum vessel is avoided, all of the sputtering and surface materials problems are transferred to the remote burial chamber. The divertor concept was demonstrated to operate effectively on the Model C stellarator [24] and on the FM-1 device [25]. The divertor also creates a scrape-off or shield plasma (Fig. 7) which can ionize wall originating impurities and then sweep the ionized impurities into the burial chamber before the impurities can migrate into the hot confined plasma. This shielding effect can be very efficient and allows one to separate the hot confined plasma from a secondary shield plasma which interacts with the vacuum wall and limiters. Also, the shielding divertor allows one to refuel the plasma near the surface and to run the tokamak for long pulses as envisioned by the reference reactor designs. The primary disadvantage of the divertor system is that roughly 0.3 of the volume inside the toroidal magnet must be devoted to the divertor. The effectiveness of magnetic limiters and poloidal divertors in controlling impurities will be determined with the PDX (Poloidal Divertor Experiment) now under construction at Princeton. This device will have a plasma size very similar to PLT, namely; $a = 45$ cm, $R = 145$ cm with a 25 kG toroidal field and a plasma current of 500 kA.

All of the methods described for reducing impurity generation, while very efficient, still allow some small fraction of impurities to penetrate into the plasma core either during startup or during the burn cycle. Therefore, it is important to devise schemes to remove impurities from the central hot plasma. Ironically, the natural tendency of classical diffusion is to drive impurities and alpha ash toward the center of the hot plasma. Basically, this arises because of a frictional force, F_f between the fuel ions and the impurities. The frictional force arises because of the fundamental diamagnetic drifts, V_d, which are required for plasma equilibrium. This frictional force $\vec{F} \times \vec{B}$ drifts impurities inward toward the peak of the fuel ion density while driving the fuel ions outward to the wall. A variety of techniques have been proposed to reverse the direction of the net frictional force on the impurity ions and to thereby drive the impurity ions out of the discharge. Ohkawa [26] has proposed using cold plasma streams near the surface of the plasma which because of the toroidal geometry drive impurities outward. Travelling electromagnetic waves [27] moving anti-parallel to the plasma current exert a frictional force which can also drive impurities outward. Also by manipulating the density and temperature profile, e.g., with a divertor, it may be possible to reduce or even reverse the classical inward impurity transport [28].

The last area of impurity control involves reducing the effect that a specific impurity concentration has on plasma behavior. Clearly low-Z impurities have smaller effects than high-Z materials, with this in mind, the Wisconsin group has proposed using a carbon curtain [29] or an internal carbon shield while the General Atomic group [30] has proposed a Si C limiter system. Success in this area will depend on our ability to find a low-Z material with a low physical and chemical sputtering coefficient that has ability to withstand substantial heat loads such as those incident on a limiter. At this time, it appears as though the ultimate solution to the impurity problem will be a combination of these various techniques. The Russian T-10 tokamak and PLT will be initially operated with high-Z refractory metal limiters.

VI. AREAS OF SURFACE RESEARCH

In order to evaluate materials for tokamak vacuum walls or shields, limiters and divertor pumping a number of surface data are required which are not readily available in the literature. The following list reflects the author's interest and is by no means a complete list.

Physical sputtering coeffients are needed for hydrogenic ions (H^+, D^+) as well as impurity ions (e.g. O^+, C^+) on standard wall, liner and limiter materials such as stainless-steel, niobium, various types of carbon, silicon carbide, titanium, tantalum, molybdenum, tungsten, liquid lithium, etc. Previous sputtering measurements were nearly aways made in the 5-10 keV range, since most of the particle efflux in a tokamak is estimated to be near 1 keV, the new measurements should extend into the ~ 300-500 eV range. It would also be desirable to determine the sputtering yields near threshold for these various materials, so that the plasma physicists know how low the edge temperatures must be kept with a gas blanket in order to reduce the sputtering to a negligible value. The yield should be measured as a function of incident ion angle from normal to $\approx 45^{\circ}$ since much of the plasma loss is concentrated in this interval. In addition, the energy distribution of the sputtered products needs to be measured so that the penetration depth of impurity neutrals into the plasma can be estimated and used for the design of shielding divertors and other impurity control methods. Also, self sputtering coefficients for these materials at ion energies ~ 1 keV as well as chemical sputtering coefficients over a temperature range of 300°K to 1500°K would be important in determining the total impurity generation.

An important aspect of these measurements should be not only to measure properties under ideal surface conditions, but to try to simulate the operating enviroment of a tokamak system where surfaces become contaminated under operating conditions. In this regard, it will be necessary for some measurements to be done on operating tokamak devices to determine their real surface constituents.

The design of limiter or neutralizer plates in a tokamak appears to be a very serious heat deposition problem. Materials contemplated for limiters should be evaluated for their ability to withstand thermal shock and evaporation as well as sputtering.

Divertor designers are in need of data on the trapping of hydrogen ions and neutrals in titanium, zirconium, liquid lithuim, etc. The objective is to develop a high speed getter pump capable of handling the large gas loads in the divertor burial chamber. Trapping efficiencies as a function of neutral atom energy from 1 keV to several eV, material temperature from 77°K to 1000°K, neutral fluence and incident angle. The overall information could be obtained in a simple divertor modeling chamber where an incident hydrogen ion beam enters a gettered chamber through a small orifice, strikes a neutralizer plate and the resultant neutrals are pumped by a gettered surface. The neutral pressure in the chamber gives the overall pumping speed of the system and is the quantity of interest.

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TABLE I

Tokamak Parameters

	TFR	PLT	PDX	DIII	TFTR
Major radius (cm)	98	130	145	140	245
Minor radius (cm)	20	45	47	45	85
Elongation	1.0	1.0	1.0	3.3	1.0
Toroidal field (kG)	60	50	24	26	50
Plasma Current (MA)	0.35	1.0	.52	3.2	2.1
Safety factor (q)	3.5	3.5	3.5	3.5	3.5
Pulse length (sec)	0.5	1.0	1.0	1.0	0.5
Auxiliary Heating (kw)					
neutral beam	0.5	2	4	5	12
radio frequency	-	4	4	-	-
Plasma density (cm^{-3})	$5x10^{13}$	$5x10^{13}$	$5x10^{13}$	$5x10^{13}$	$5x10^{13}$
Plasma temperature (keV)	1-2	2-4	2-4	2-4	4-6
Confinement time (sec)	.03	~0.1	~0.1	~0.1	~0.1
First wall	SS	305-SS	Ti,Ta,--	316-SS	305-SS
First wall temperature	$400^{o}C$	$20^{o}C$	$200^{o}C$	$\sim 300^{o}C$	$20^{o}C$
Limiter	Mo,W	W,Mo	divertor	SiC	W,Mo
Vac. Vessel Vol. (cm^3)	$1x10^6$	$6.9x10^6$	$3.6x10^7$	$2.7x10^7$	$7x10^7$
Pumping system	turbo	diff	diff,gett	turbo	diff.
Base pressure (torr)	$5x10^{-10}$	$\sim 10^{-8}$	10^{-8}	10^{-8}	$4x10^{-8}$
Operation date	Mar.'73	Nov.'75	Oct.'77	Feb.'78	Jan.'80

TABLE II

Energy Fluxes to the First Wall of TFTR

Species	Flux W/cm^2	%	Energy (keV)
Plasma electrons	3.7	44	~ 0.1
Plasma ions	3.1	37	~ 0.4
Radiation	0.3	3	~ 0.01-0.1*
Charge exchange neutrals	0.6	7	~ 0.5
beam neutrals	0.7	9	~ 60

* radiation near 10 eV for pure hydrogen and ~ 100 eV for line radiation from impurities. Bremsstrahlung is negligible.

FIGURE CAPTIONS

Fig. 1 Schematic view of a Tokamak. The plasma current I_p is induced by a changing flux ϕ_{OH}, held in toroidal equilibrium by the vertical field B_V and is stabilized by the toroidal field B_T. The insulated gap prevents unidirectional currents from flowing in the vacuum vessel.

Fig. 2 Density and temperatures for the TFTR tokamak computed using the six regime transport code. The assumed conditions are: B_T = 52kG, I_p = 2.5 MA, constant impurities with $\bar{Z}$ = 2.0, 12 MW deuterium beam at 154 keV injected into triton plasma with no compression.

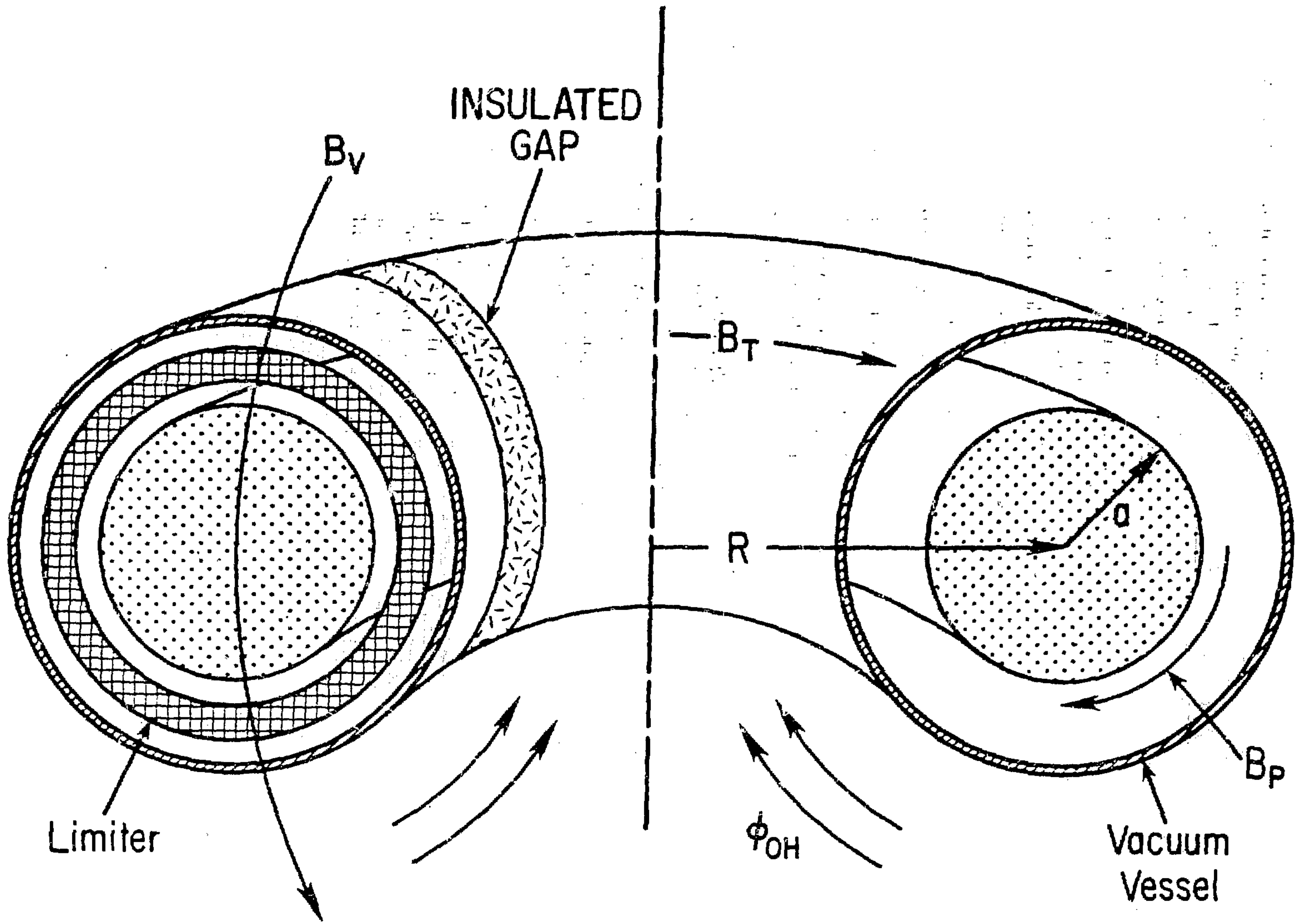
Fig. 3 The charge exchange neutral spectrum due to thermal plasma ions. The fast neutrals due to beam ion charge exchange is not included.

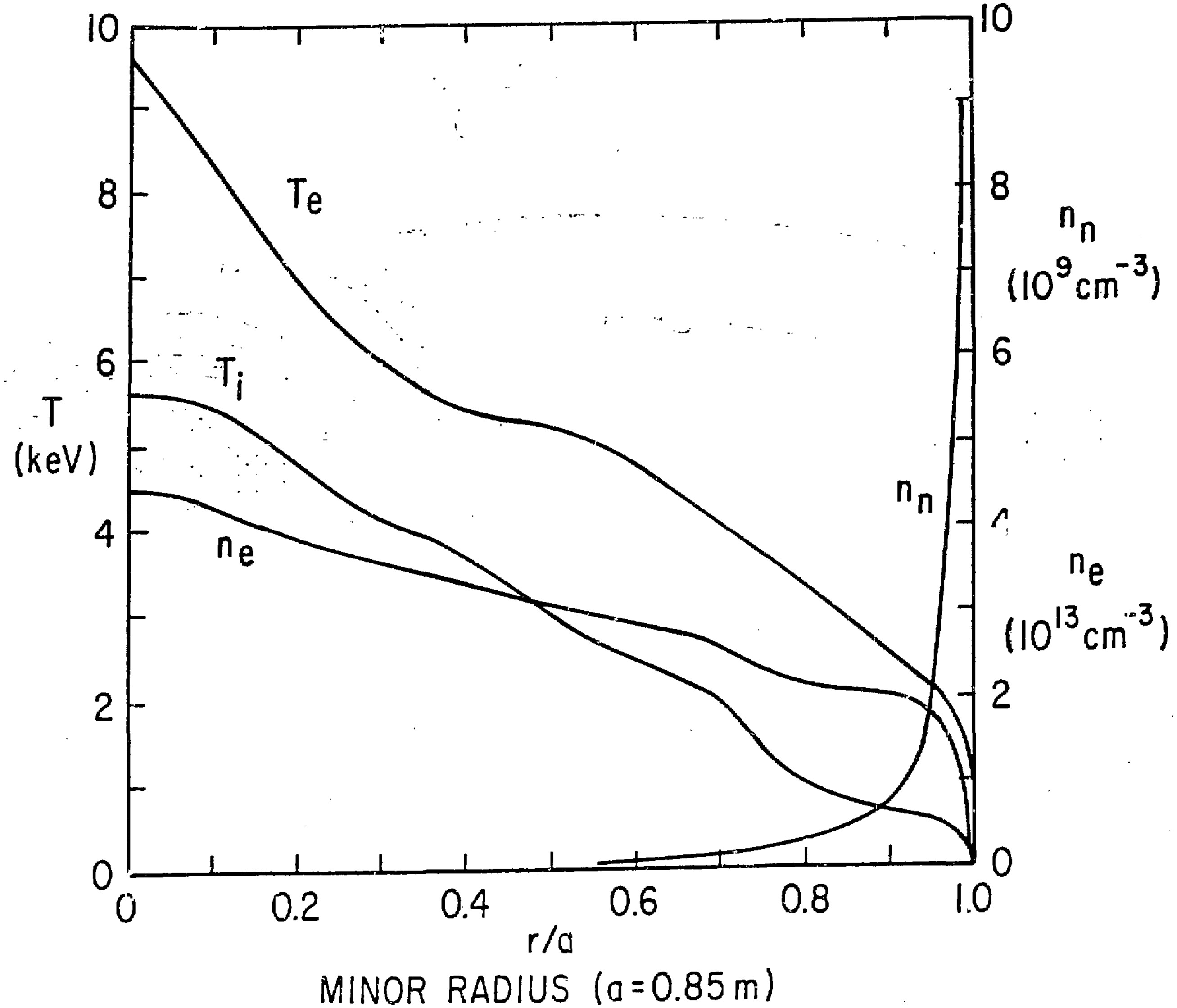
Fig. 4 The total power radiated by iron impurities versus electron temperature. For these calculations $n_e \sim 10^{14}$ cm^{-3}. The calculations of Huebner et al reference 14 is compared with the radiation from a simple lithium-like ion given by Post reference 13.

Fig. 5 Effect of high-Z impurity ions on the ignition condition for a 50-50 DT reactor. The curve labelled 0 is the $n\tau$ required for a pure plasma while the other curves have impurity fractions, $f = n_z/n_e$.

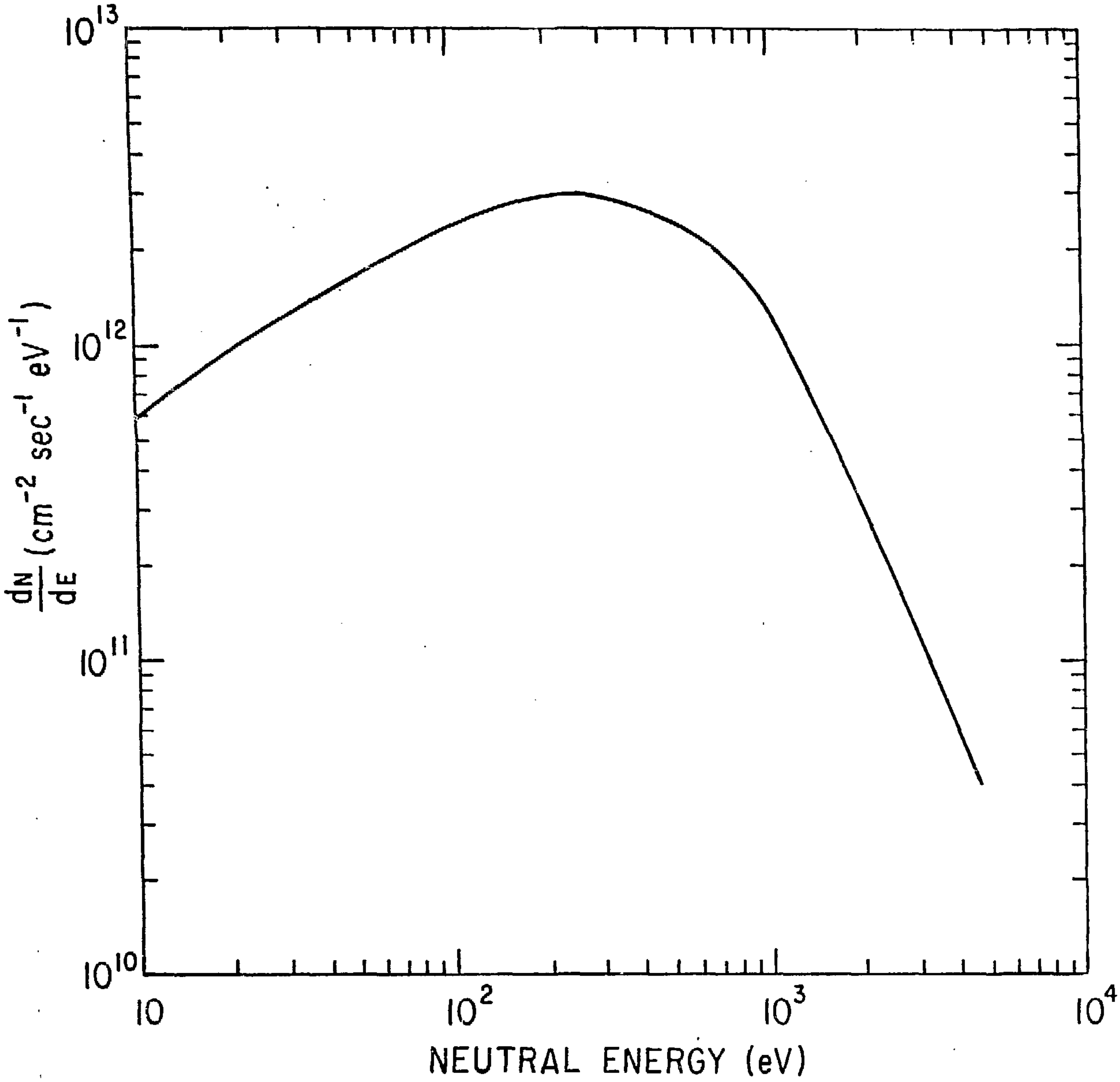
Fig. 6 Effect of high-Z impurity ions on the Lawson condition for a 50-50 DT reactor. The energy recycling efficienty is assumed to be 40%.

Fig. 7 Cross-sectional view of the PDX (Poloidal Divertor Experiment). Plasma, that escapes from the main confinement region, is channeled into the neutralizer plates and the resultant neutrals are pumped by titanium gettered surfaces in the burial chamber. The liner is replacable and is capable of being heated to 200°C. The primary vacuum vessel is 305-SS.

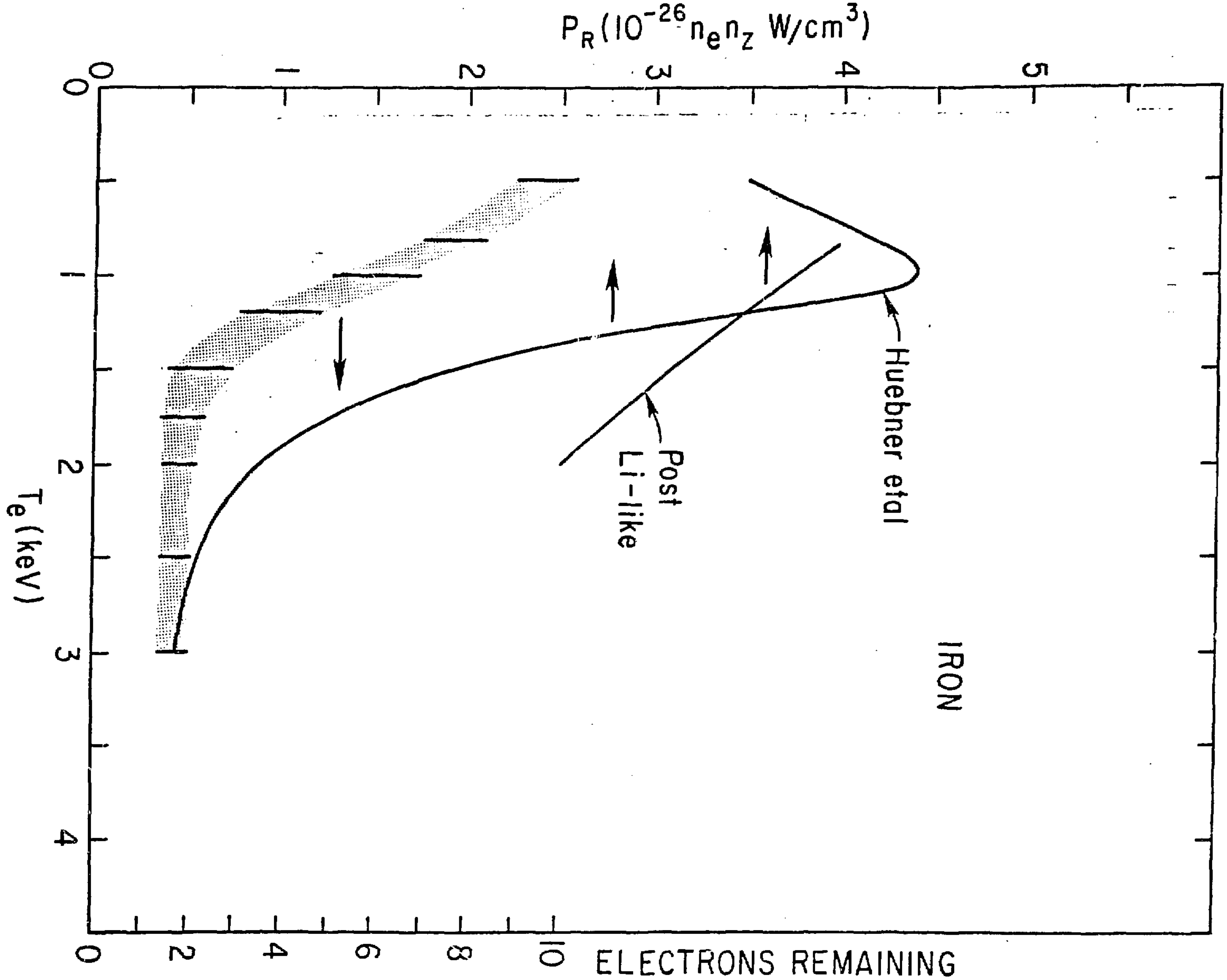

INSULATED
GAP
B_V
B_T
R
a
Limiter
ϕ_{OH}
B_P
Vacuum
Vessel



10
8
6
4
2
0
T
(keV)
Te
Ti
ne
nn
nn
(10^9 cm^-3)
ne
(10^13 cm^-3)
0
0.2
0.4
0.6
0.8
1.0
r/a
MINOR RADIUS (a=0.85 m)

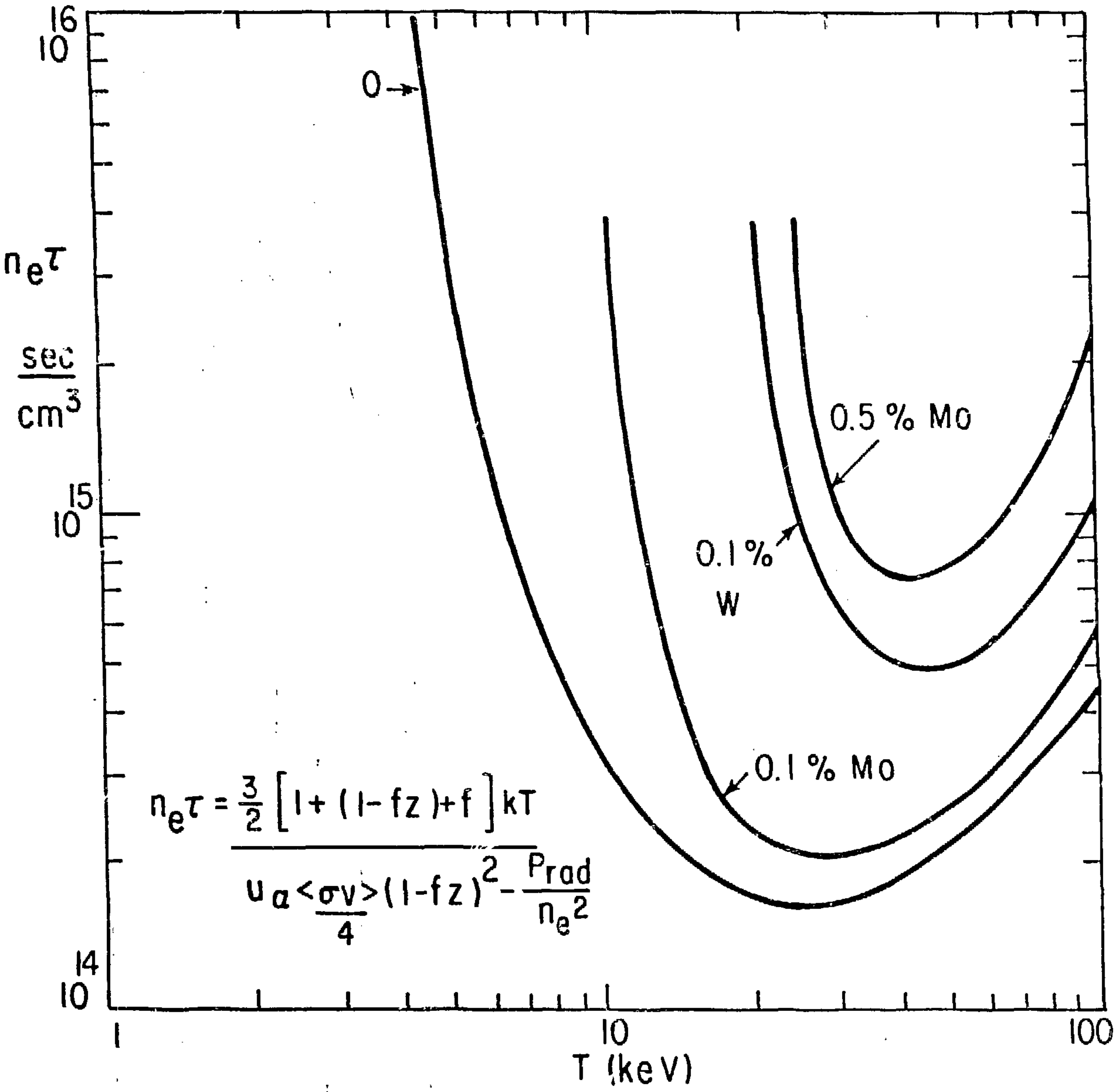


10^{13}
10^{12}
10^{11}
10^{10}
$\frac{dN}{dE}$ (cm^{-2} sec^{-1} eV^{-1})
10
10^2
10^3
10^4
NEUTRAL ENERGY (eV)

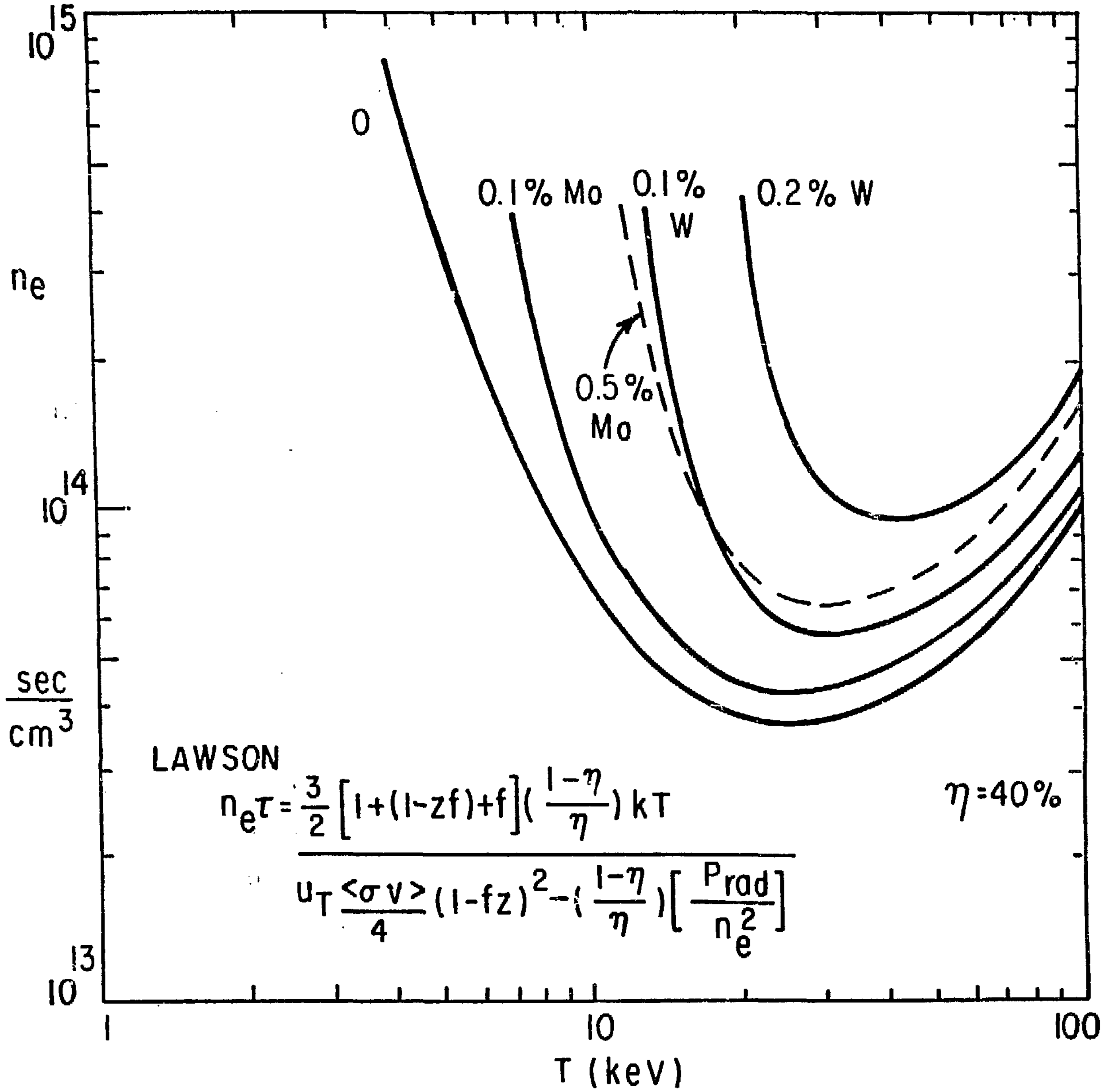


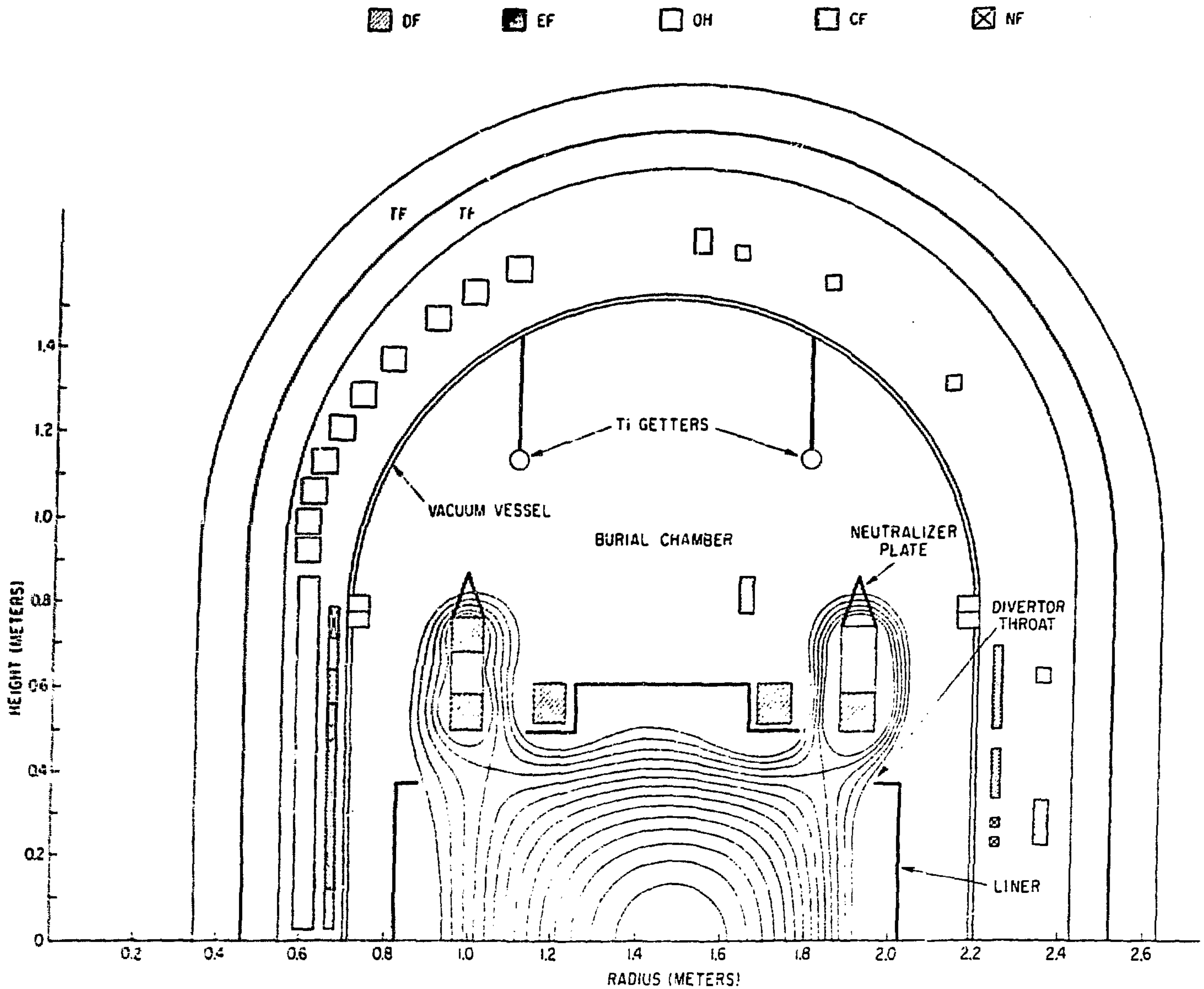
$P_R(10^{-26} n_e n_Z\ W/cm^3)$
0
1
2
3
4
5
Huebner etal
Post
Li-like
IRON
T_e (keV)
0
1
2
3
4
0
2
4
6
8
10
ELECTRONS REMAINING

EFFECT OF HIGH Z IMPURITY ON THE IGNITION CONDITION



EFFECT OF HIGH Z IMPURITY ON THE LAWSON CONDITION





DF
EF
OH
CF
NF
TF
TF
Ti GETTERS
VACUUM VESSEL
BURIAL CHAMBER
NEUTRALIZER PLATE
DIVERTOR THROAT
LINER
HEIGHT (METERS)
RADIUS (METERS)
0
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
2.0
2.2
2.4
2.6