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BY

S. A. COHEN

PLASMA PHYSICS LABORATORY

MASTER

PRINCETON UNIVERSITY
PRINCETON, NEW JERSEY

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Vacuum and Wall Problems in Precursor Reactor Tokamaks*

S. A. Cohen
Plasma Physics Laboratory, Princeton University
Princeton, New Jersey 08540

ABSTRACT

The Princeton Large Torus (PLT) will be completed in 1975 and two other CTR oriented toroidal plasma devices, the Poloidal Divertor Experiment (PDX) and the Tokamak Fusion Test Reactor (TFTR), are planned to be completed in 1977 and 1981. The vacuum systems of these machines must satisfy stringent requirements because of unusual operating conditions, such as, magnetic field induced strains and eddy currents, energetic particle and photon bombardment, large transient gas loads, and the use of 100 Curie quantities of tritium. In addition, novel vacuum wall surfaces and fast moving mechanical or magnetic plasma limiters will be required to minimize the influx of impurities during discharges.

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I. INTRODUCTION

Every discussion of the feasibility of a particular method to achieve thermonuclear fusion starts with the Lawson criterion[1] and a diagram of $n\tau$ versus T, the density-energy confinement time product versus the temperature. Fig. 1 shows the expected regions of operation for several different fusion breakeven schemes. As one can see, the shortest distance from the origin to the break-even line occurs at $n\tau \simeq 7\times10^{13}\ \frac{\text{sec}}{\text{cm}^3}$ and $T \simeq 10$ keV. Though the shortest distance may not be representative of the easiest or most economical method to achieve breakeven, it is the route chosen by the proponents of tokamaks and Θ - pinches. At this point much of the similarity between these methods ends as heating and containment techniques vary considerably in time scale and geometry. Θ-pinches reactors, for example, would work optimally at high densities ($10^{18}/\text{cm}^3$) in a short time scale (10^{-4} sec) pulsed mode. Tokamaks would operate at lower densities ($3\times10^{14}/\text{cm}^3$) and longer confinement times (0.3 sec). Because of this long time scale tokamaks may suffer from certain problems not symptomatic to the other technique. One major problem may be the accumulation of impurity ions in the plasma. Other problems would be caused by the need to vary some of the applied magnetic fields in times as short as a few milleseconds. A feature common to both the magnetic field and impurity problems is the structure of the vacuum vessel wall. Throughout this discussion the point of view is how will the wall affect the plasma, not how will the plasma affect the wall.

These problems are repeatedly encountered in present day tokamaks. If the particle confinement time in larger tokamaks increases as it is hoped, these problems may be exacerbated. Remedies to these problems must involve advances in the understanding and manipulation of plasma behavior, vacuum technology and surface phenomena.

In this paper I will describe the wall and vacuum problems expected in the Princeton Large Torus (PLT), the Poloidal Divertor Experiment (PDX) and the Tokamak Fusion Test Reactor (TFTR), and what solutions are being investigated.

II. THE TOKAMAK

A tokamak is a toroidal, axisymmetric, low β_T ($\beta_T \equiv$ plasma energy/toroidal magnetic field energy) plasma containment device that relies on a strong toroidal plasma current to reduce vertical particle drift motions caused by the curved inhomogeneous toroidal field. In Fig. 2 one can see pictorially how the net helical magnetic field produced by the addition of a toroidal current causes ions to drift back into the tokamak during their transit along the field lines in upper half of the torus. The toroidal current also efficiently heats the plasma at low electron temperatures ($T_e \lesssim$ 5 keV). A flux change inside the torus of about 10 volt seconds is required to drive this ohmic heating current. So as not to "short out" the plasma, no nearby continuous conducting path may encircle the torus. For these reasons both the vacuum vessel containing the plasma and the structure supporting the toroidal field coils must be sufficiently non-conducting or they must contain insulating sections.

Other time-varying magnetic fields are required in tokamaks. A vertical field must be applied to balance the well-known toroidal hoop forces. Complex fields may be used to divert outer field lines from the main plasma volume into a separate volume. The purpose of these diverted field lines (commonly called a "divertor") is to control the species and time behavior of particles that enter the main plasma body. In particular, divertors will be used to remove impurity ions from the plasma edge.

One other important component of a tokamak is its aperture limiter. This is usually a set of refractory metal arms that encircle the plasma and prevent it from expanding out to the vacuum vessel wall. The limiter plays a dominant role in determining the boundary conditions of the plasma. It absorbs the power outflux due to electron and ion thermal conductivity and also neutralizes the ions that impact on it.

A. CHARACTERISTICS OF TOKAMAK DISCHARGES.

Computer calculations of "standard" tokamaks discharge characteristics were carried out to unravel the details of wall associated problems. These calculations were performed with the Princeton Transport Code[2] whose predictions are in good agreement with measurements on existing tokamaks. Table I is a list of relevant parameters for the PLT and TFTR devices. The TFTR parameters are for two different times in the same discharge - one prior to, the other immediately after a strong adiabatic compression. The PLT parameters are for the time in the discharge when the ion temperature is at its peak central value. The influx of impurities due to various machanisms has been set to an arbitrarily low value; thus the plasma temperatures shown in Table I are overly optimistic. Of course, if the plasma is cooler the particle transported energy flux to the wall may be overestimated. In Fig. 3, the radial profiles of the electron and ion density and temperatures are shown for the precompression TFTR calculation. These profiles all have negative gradients; i.e., the density and temperature both decrease monotonically as r increases. Reversed gradient plasmas will be discussed later in this paper. In Table I we see that charge exchange hydrogen, ion thermal conductivity and electron thermal conductivity contribute equally to the energy loss. The energy distributions of charge exchange hydrogen atoms that leave PLT and TFTR are shown in Figures 4 and 5. In contrast with the 5 keV central ion temperatures, the most probable energy of the escaping charge exchange particles is only 300 eV.

It is improbable for impurity atoms to leave the plasma by multiple charge exchange. In quiescent discharges they may either diffuse out or be resonantly pumped out by some rf wave. Those impurities that diffuse out will have an energy equal to the edge ion temperature. (They may gain up to 4 times that energy if there is an electron sheath). Impurities that leave by a resonant process may have several hundred eV of energy.

From examining Table I we can enumerate several mechanisms that could cause impurity influx into the tokamaks during steady state operation. These mechanisms include:

1) Photo-desorption of atoms adsorbed on the first wall.
2) Physical sputtering of wall material or chemisorbed atoms by escaping charge exchange neutrals.
3) Sputtering of wall material by non-hydrogenic ions
4) Sputtering of wall material by 14 MeV neutrons.
5) Chemical sputtering of wall materials by hydrogen.
6) Blistering of wall material by escaping hydrogen or helium atoms.
7) Evaporation or fragmentation of limiter material by the large energy flux.
8) Plasma contact with wall (desorbing loosely bound gases).

More detailed descriptions of some of these mechanisms can be found in References 3, 4, and 5. The impact of runaway electrons on the walls and limiters is an additional mechanism. However, this should not occur in high density discharges.

The relative contribution of each of these to the impurity problem in TFTR is summarized in Table II. The two values shown for neutron sputtering correspond to the cases "chunk emission"[6] and "no chunk emission".[7] The photodesorption efficiency used in Table II is approximately 10^{-4} for photon energies between 1 and 10^3 eV.[8,9] Sputtering yields for stainless steel were taken from References 10 and 11. (This estimate includes the angular distribution of the charge exchange particles).[12] Blister formation may be less important than shown. (Recent observations have shown that 1) blisters do not form when the impacting beam has a wide energy spread,[13, 14] and 2) blisters do not reform on previously blistered regions.)[15] However, the blister formation rates used are from Verbeek and Eckstein.[16] Since all plasma heating in these calculations was "caused" by Ohmic heating and neutral beams, the escaping impurities should leave the plasma at approximately 40 eV, 4x the edge temperature. At this energy, the sputtering yield of Fe on Fe is taken to be ~.05, the same as for Ar impact on Cu.[17] The impurity confinement time was assumed to equal the calculated ion confinement time. Chemical sputtering does not appear to be relevant to stainless steel walled machines. (It may not be relevant to carbon walled devices either, if the carbon temperature is sufficiently low.) The extent of limiter evaporation and fragmentation was calculated using the results of Schivell and Grove.[18]

In addition to the impurity influx during steady state operation, there might be a large influx during the plasma start up. One reason for this is that the helical field is not properly formed at the beginning of the discharge. Consequently particle containment is poor and the wall would experience severe particle bombardment.

The Princeton Transport Code shows that this influx of impurities into the plasma causes the peak ion and electron temperature to drop by about 60%. Considering the impressive predictions in Table I, a 2 keV ion temperature is very disappointing. However, it is not necessary to operate a tokamak in a manner that makes it susceptible to these problems. Specific plans for constructing and operating PLT, PDX and TFTR to avoid the sputtering and desorption problems as well as the aforementioned magnetic field problems will be presented in Section III.

B. THE EFFECTS OF IMPURITIES IN TOKAMAKS

Before presenting the details of the proposed methods for impurity control, I will describe the effects impurities have in tokamaks and why their presence is usually undesirable. Considerations of these effects have greatly helped in formulating solutions. Throughout this discussion, the impurities are described by their mass m_I, charge state Z_I, kinetic temperature T_I, convective motion $\langle v_I \rangle$, and local density n_I.

There are three favorable effects that would occur for a small accumulation of impurities in the plasma core. The first is an enhancement of the Ohmic heating efficiency. Impurities increase the resistivity of the plasma by the factor,[19]

1) $$Z_{eff} = \sum_i n_i Z_i^{\,2} / n_e,$$

where i represents the sum over all ion species present in the plasma, Z_i is the charge of the i^{th} species, n_i is the density of the i^{th} species and n_e is the density of electrons. Z_{eff}, as a function of impurity content, for C, Al, Ti and Mo is shown in Fig. 6. This increase of resistivity in the plasma core would tend to spread the Ohmic heating current over a larger channel, thereby improving the plasma's magnetohydrodynamic stability. Also when impurity ions have a gradient in the same direction as the hydrogen ions certain trapped particle modes that would otherwise cause detrimental plasma losses become stabilized.[20]

These salutary effects of impurities are not compelling reasons to tolerate a contaminated plasma. Even with impurities present Ohmic heating by itself could not bring a tritium -

deuterium plasma to ignition. Other techniques,[21] such as the injection of energetic neutral beams,[22] lower hybrid[23] waves or ion cyclotron resonance waves[24] are necessary heating supplements. There may be other ways to change the current profile - such as the injection of frozen hydrogen pellets - that may not have the detrimental effects of impurities. Similarly, trapped particle effects could be relieved by pellet injection or changes in the plasma cross section.[25] Furthermore, before accumulating in the core, the impurities must be transported across the outer regions of the plasma. Here their effects are completely unfavorable. They cause the current profile to shrink and the trapped particle modes to become unstable. The time scale for a $Z \simeq 20$ impurity to be transported through a "clean" quiescent plasma into the core is approximately

2) $$\tau \simeq \frac{D}{a^2} \simeq 10^{-1} \text{ sec},$$

where $a \simeq 50$ cm,

and D is the neoclassical diffusion coefficient[26] at 50 kGauss magnetic field and 2 keV ion temperature. (This assumes negative gradients for the temperature and density.)

Further anticipated disadvantages of impurity ions in tokamaks include:

1. Enhanced power loss by radiation. Fully stripped ions cause an enhancement of bremsstrahlung by the factor Z_{eff}. This factor increases as $T^{1/2}$. For incompletely stripped ions the problem is more complicated. Recombination losses are a function of ion charge state and

electron temperature. The enhancement factor γ for different oxygen and iron ions, are shown in Fig. 7. γ is proportional to $Z_I^2/T^{1/2}$. Line radiation from ions with at least 3 remaining electrons is given by[28]

$$P \simeq 1.5\times10^{-26} \frac{n_I}{n_e} \quad n_e^2. \tag{3}$$

Fig. 8 shows the calculated power loss spectrum for 1% Fe contamination of a 10^{14} electron cm^{-3}, 1 keV hydrogen plasma.[29] This power loss exceeds planned Ohmic heating input power in PLT by a factor of 5. It is notable that line radiation is the dominant loss term for Fe impurities at this temperature. For fully stripped ions, recombination radiation is the largest loss mechanism.

2. Reduction of the reactivity of a D-T plasma. The reactivity of a D-T plasma is proportional to the product $n_D\ n_T$. A 1% contamination by 26 times ionized Fe lowers the reacting particle densities by 13% each, hence the reactivity by ~ 24%.
3. Enhancement of neoclassical particle transport, and hence energy transport. Particle transport is approximately proportional to $(Z_I)^{1/2}$ at constant Z_{eff}.
4. Shallow energy deposition profiles for injected energetic neutral beams or rf waves. High Z impurities may be quite efficient in ionizing incoming neutral beam atoms by charge exchange[30] or ion impact ionization.[31] Similarly impurities may be resonant at low harmonics of ion cyclotron waves.[32] If impurities are near the edge of the

plasma they would impair the penetration of neutral beams or IC waves into the core.

From the above considerations it is clear that for the same impurity concentration (n_I/n_e) it is far more desirable for the contamination to be fully stripped low Z ions than partially stripped high Z ions. In fact, Meade[33] has calculated that for 0.2%W (Z=74) contamination enhanced radiation losses alone would prevent ignition at any temperature. The same concentration of Mo (Z=42) increases $n\tau$ and T ignition requirements by ~25% each. For the PLT, PDX and TFTR devices the maximum tolerable impurity concentrations are approximately 6% for O, 0.4% for Fe, or 0.10% for W. For concentrations above these levels, the plasma temperature predicted by the Princeton Transport Code falls below one-half that shown in Table I.

III. PLT, PDX, and TFTR

The methods used in PLT, PDX, and TFTR to circumvent the problems discussed in Sec. II are dictated by the prime motivation for each device. These are:[34]

For PLT, to study the physical laws describing plasma transport as a function of magnetic field, plasma current, plasma dimensions, and auxiliary power input.

For PDX, to control plasma boundary conditions by the use of magnetic divertors and magnetic limiters. This specifically includes the reduction of impurity influx and the control of hydrogen recycling.

For TFTR, to demonstrate fusion energy production from magnetically confined reactor grade D-T plasmas using the two-component concept.

Of course, the design and construction of each device also depends on economic factors.

A. The Princeton Large Torus (PLT)

Fig. 9 shows a sketch of PLT. Its main components include the toroidal field (TF) coils, vacuum vessel, pumping system, ceramic sections, Ohmic heating (OH) coils and shaping field (SF) coils. Not shown are the ~30 diagnostic devices that will be attached to PLT to measure the electron density and temperature, ion temperature and density, impurity content, wall conditions, et cetera.

Economy in construction and magnetic field energy were the reasons that the SF and OH coils are compactly nested inside the TF coils. This allows the OH field to use space occupied by the TF coils. In this awkward location their installation was time consuming and tedious.

The vacuum vessel is 130 cm in major radius and about 50 cm in minor radius. It contains two ceramic sections so that the Ohmic heating current will not be shunted through the vacuum vessel. Their dimensions and composition are listed in Table III. The construction of this component was performed in conjunction with the Rutgers' University, Department of Ceramics. Smaller ceramic secitons had been made at Princeton 15 years ago,[35] but the possibility that 40 inch diameter sections could not be made, prompted that the resistance of the vacuum vessel be increased. This was accomplished by fabricating it in alternating sections of

1/4" thick 305 SS cylinders and multicorrugated bellows sections. Each of the 18 pie shaped bellows section contains approximately 25 1" deep corrugations of stainless steel, each 0.022 inches in thickness. With this construction the resistance of the vacuum vessel alone (.01 Ohms) is 10 times greater than the plasma resistance at a 5 eV temperature. This construction of the vacuum vessel also satisfied the time scale requirement ($\tau \simeq .003$ ms) for the penetration of the SF field. In TFTR the vessel is substantially thicker to support larger atmosphere forces Here the field penetration time may be too long for feedback stabilization of the plasma.

It was decided not to attempt novel construction techniques in PLT. In addition, high temperature bakeouts such as were successful at Cern[36] will not be used because of the great expense. It was estimated that the changes required to heat the vacuum vessel to 350° C would increase the construction costs by 25%. In contrast, Russian tokamaks are baked.

The first line for attacking the impurity problem was to get the lowest base pressure possible. For this reason all seals on the vacuum chamber are either gold or copper. Only UHV compatible materials are used. Four of the largest ports on the vacuum vessel are used for connections to the vacuum pumps. These are 10" diameter stainless steel Hg diffusion pumps with multicoolant baffles. Hg pumps were chosen over oil pumps because of the relative ease and speed for detecting and removing Hg from the vacuum vessel in the case of an accident. The total pumping speed is conductance limited to about 4000 liter/second. All sections of the vacuum vessel have been vacuum baked in a separate vacuum oven to about 400° C. This is known[37, 38] to reduce the outgassing rate from 10^{-9} to about 10^{-14} Torr liter/sec-cm^2. However, frequent device opening will probably cause the net outgassing rate to be about 3×10^{-11} Torr liter/sec-cm^2. Due to the large surface area of the torus, 5×10^5 cm^2, this outgassing rate will cause a base pressure of 4×10^{-9} Torr. It is expected that this will be comprised mostly of H_2O, CO, and H_2. Under these conditions multi-monolayer films of absorbed gases will certainly form. Hence desorption during a discharge could be a serious problem.

Five methods for in situ surface cleaning are planned to eliminate the adsorbed gases. The first method is discharge cleaning. Traditionally, this is the main method used. Various gases have been used in discharge cleaning tokamaks with the best results at Princeton being obtained with oxygen[39] or a 30% neon - 70% hydrogen mixture.[40] Discharge cleaning in tokamaks is performed by rapid pulsing the machine at low toroidal magnetic fields. This produces copious photons and 10-50 eV ions which are poorly confined and repeatedly bombard the wall. (The wall temperature may rise to over 100° C.) The major fault with this process is its low duty cycle (10^{-2}). Also sputter cleaning of surfaces is known to increase the amount of CO adsorption by 10^2 -10^3 over annealed surfaces.[41]

The second method to reduce loosely bound gases on the wall is to getter with titanium. This has been used on the ATC[42] tokamak to change hydrogen recycling. Measurements of the vacuum ultraviolet spectrum after gettering have shown that the oxygen impurity content decreased by a factor of 10. (Sublimation was done in only 1/2 the torus). This technique may have a few drawbacks. It is well known that thick titanium films have a tendency to peel. We have not yet experienced problems from this. Also, shutters are required over all windows and ceramic sections during the sublimation - it may be necessary to sublimate between every discharge. The long term effects of titanium sublimations in the main plasma chamber are not known, however, it is disconcerting to be trapping large quantities of gas in areas that are subject to intense particle and energy bombardment.

Thirdly, we have installed the facility to evaporate thin films of other metals in situ. Experiments at PPPL have shown that freshly evaporated Au films do not adsorb appreciable amounts of hydrocarbons or CO for week long exposures to vacuum conditions similar to PLT.[43] In comparison freshly evaporated Al films rapidly oxidize, but then do not adsorb hydrocarbons or CO. Both are possible choices. Aluminum oxide has the additional advantages of having both a lower Z and a lower sputtering yield than gold.[44]

Fourth, we have installed 8 moveable filaments in PLT. These can be positioned on the axis of PLT and used to generate 16 A of 5 keV electrons for electron stimulated desorption (ESD) of gases adsorbed on the vessel wall. At high gas coverages, these electrons have a desorption cross section of about 10^{-17}. This falls to 10^{-20} at partial monolayer coverages.[45,46,47,48] At these efficiencies the ESD technique could clean the walls in times varying from 1 minute to 10 hours. The cleaning time will probably be shorter than 10 hours by a factor of at least 3 because of the high probability of elastic electron backscattering from the surface.[49] In addition ESD efficiency is high down to 100 eV, hence each electron can interact numerous times. ESD has already shown itself to be a viable cleaning technique in the DCX-1 mirror machine.[50]

Recent advances in ESD experiments point a way to modify the surface into, perhaps, a more desirable form. It has been observed that electron bombardment of a CO covered Pt (111) surface causes desorption of more than 95% of the O but only 50% of the C.[51] Recent experiments in our laboratory show essentially the same results for CO on stainless steel.[43] The carbon that is left behind is in the graphitic form as ascertained by Auger line shapes.[43,52] Discharge cleaning of PLT followed by CO adsorption and ESD could coat the entire inner surface with a monolayer film of graphite. This coating does not readily readsorb CO. Detailed measurements[53] have shown that the sticking coefficient of CO and H_2 on bulk graphite is less than 10^{-4}. In addition to this benefit it is possible that a graphite coating may reduce the sputtering of Fe. Calculations on this are in progress.[54] If the sputtering yield of this film is the same as for bulk graphite[12,55] then under standard

PLT operating conditions the graphite film would last for several days before needing to be replenished. This method has the advantage that if the film is hardy it will continue to protect the surface for a long period; on the other hand, if it is readily sputtered (whether chemically or physically) it will be eroded in a short period and no trace be left. This is an important reason to try this approach prior to titanium gettering or Al or Au evaporations.

In Table II, it is noted that little impurity influx is expected from limiter evaporation or fragmentation. This is based on the assumption that the limiter area exposed to the plasma is sufficiently large that the power flux to it never exceeds 1 kW/cm^2.[56,57,58] To satisfy this condition the limiters must be large and the plasma must be well-behaved. The four limiters will each weigh 26 lbs. For a well-behaved plasma the steady state operating temperature (0.5 sec of bombardment followed by 120 sec of no plasma) of the 4 limiters is expected to be about 800° C and the peak surface temperature about 1200° C.

Plasmas with skin-currents would place an unacceptably large thermal load on the limiters. The formation of a skin current may be prevented by initiating the discharge as a very thin toroid (say 10 cm minor diameter) and allowing it to expand out to 45 cm in 50 msec. This would force the current to flow in the core. To carry out this plan one technique would be to have the limiters form a 10 cm diameter constriction in the vacuum vessel and to then withdraw them as the rate 40 cm/50 msec (about 18 mph). Such an expanding limiter is being designed.

Hydrogen gas will be introduced into the tokamak in pulses using valves capable of 5 msec opening and closing time. The hydrogen used in our experiments is purified by passage through palladium.

Related to this is how to reduce the additional amounts of cold neutral hydrogen that accompany the use of the high power neutral beam injectors. The limited space around the neutral beam injectors does not allow ample room for conventional diffusion or turbomolecular pumps. Baffled cryosorption and sublimation pumps are being considered.

In the previous discussion I described plans to lower the influx of desorbed impurities and sputtered impurities by modifying the wall conditions. The question remains: If our efforts are not successful can we alter the plasma so that the impurities will not be as detrimental as expected? One solution is: If impurities would not rapidly diffuse in from the edge it may be possible to "scrape them off" with the limiter. At the present time two different modes of operation are being considered to reverse the neoclassical inward impurity transport.[59,60] The first method (see Fig. 10) uses an axisymmetric neutral gas influx to cool the edge and cause a negative density gradient.[61] Reversing the density gradient reverses the direction of the impurity drift velocity.[2,61,62] This would keep the impurities out of the hot core and possibly allow them to be scraped off. At the least it would delay the penetration of impurities into the core.

Additional advantages of this type of operation include: 1) the reduction of energy flux to the limiter since the energy transported from the hot core will go to heating up the cool blanket, 2) the cool blanket may reduce the energy of charge exchange particles to near the sputtering threshold, 3) the neutral density in the plasma core is actually reduced because neutrals are readily ionized in the cool dense blanket, and 4) reversed gradient profiles can stabilize trapped particle modes.[61] A problem to this approach is that the plasma blanket is only transient structure with perhaps a 50 msec lifetime.

A second method to reverse the normally inward impurity transport was proposed by Ohkawa.[63] He pointed out that a poloidally asymmetric source for the recycling hydrogen could reverse the direction of impurity transport. This could be tested in PLT by gettering only the top half of the vacuum vessel and injecting hydrogen in from the bottom. In Ohkawa's scheme an axisymmetric limiter along the top of the torus is used. Initial plans on PLT do not include this limiter design.

As a final note on vacuum problems we recall the catastrophic vacuum failure that occurred in the TFR tokamak in Fonteney aux-Roses.[64] Numerous holes were burned through the bellows section of the vacuum vessel. It has been shown that these were caused by 50 keV electrons trapped in the toroidal field ripple. These particles drift out of the plasma in superbanana orbits. As protection against this occurring in PLT 1/8" thick sheets of stainless steel will be placed in front of all the bellows sections.

In summary, the vacuum vessel wall has been made thin and with insulating sections because of magnetic field requirements. It is made of UHV materials and will probably have a base pressure of 4×10^{-9} Torr. In spite of this the wall may be the source of sputtered and desorbed impurity atoms. If attempts to reduce the generation of impurities by altering surface conditions are not successful, attempts will be made to modify plasma behavior to flush out impurity ions. If this, too, is unsuccessful, the only recourse in PLT may be to do the heating experiments before impurity ions can penetrate into the hot reactive core.

B. The Poloidal Divertor Experiment PDX

The motivation to construct PDX is to test impurity control schemes in a PLT sized plasma. The main impurity control scheme, the divertor, requires additional large volumes in the toroidal field. For the same motor-generator sets used for the smaller volumed PLT, PDX can only attain a toroidal field of 25kΓ. There are numerous benefits from operating at lower fields. The severe structural problems encountered in the design of PLT will be avoided. For example, the TF coils can be constructed in a less rigid, demountable fashion. (See Fig. 11.) Hence, the SF, OH and divertor field (DF) coils can be prewound and inserted into the demounted TF structure. The vacuum vessel, shown in Fig. 12, is constructed in a similar manner. It is planned that the seals between the various parts of the vacuum vessel be of double viton O-ring construction. A trimetric view of the assembled machine is shown in Fig. 13, and a cross sectional view in Fig. 14. Here the novel features of PDX are seen easily. The plasma can be forced to assume a non-circular cross-section by the divertor field. The DF coils also divert out field lines through a "throat" onto a neutralizer plate in the gettered "burial chamber". A separate vacuum vessel liner surrounds the plasma region. Also the DF and equilibrium field (EF) coils are mounted inside the vacuum vessel.

Ten 10" Hg diffusion pumps and 80 titanium getters will be used for pumping. The diffusion pumps will have a pumping speed of 10^4 liters/sec for air. This should give a base pressure in PDX of less than 2×10^{-9} Torr.

The use of elastomer seals in PDX is necessitated by both its elliptical cross section and its demountable design. Double elastomer seals with a forepumped inner space should alleviate the permeation problem.

It may be possible to bake the elastomer seals in situ to reduce their outgoing rate[65] to 1.3×10^{-9} Torr.liters/cm^2. sec. The exposed elastomer surface area is $4.5 \times 10^3 cm^2$. In comparison, the outgassing rate and surface area of the stainless steel vacuum vessel are 3×10^{-11} Torr.liters/cm^2.sec and 4.0×10^5 cm^2 respectively. Thus the elastomer seals will contribute about half the gas load as the walls. (If necessary, the elastomer seal could be cooled to lower its outgassing rate.) A double elastomer seal and alumina spacers will be used as the "break" in the vacuum vessel. The design calls for an elastomer approximately 1" thick, capable of holding off 500 volts. Because elastomer technology is certainly capable of producing such a seal, the vacuum vessel will not contain bellows sections. In fact, the walls will be 1/2" 305 stainless steel. The problem of field penetration time is avoided by placing some of the coils inside the vacuum vessel.

Having a divertor necessitates that certain coils be inside the vacuum vessel. These will be water cooled and mounted in epoxy filled stainless steel jackets. This type of coil arrangement has been previously used on FM-1, a multipole plasma device at Princeton. Eddy currents in the vacuum vessel and coil jackets appear to be of a manageable level. The net force on each coil turn caused by the poloidal magnetic field is in the vertical direction and of order 5×10^4 Newtons. Supports for the coils will be rigid steel bars that attach to the vacuum chamber.

The liner to the vacuum vessel will be heated to 300-400° C by either electron bombardment, ohmic heating, or the passage of hot gas through embedded tubulation. During operation it is planned to cool the surfaces on which the titanium sublimates. However, the liner will probably be kept at an elevated temperature to reduce adsorbed gases. The choice of materials for the liner is still open. Possible alternatives include titanium, aluminum, beryliumcopper, or graphite, possibly in the form of a curtain.[66] The original liner could, of course, be replaced.

The same cleaning modes proposed for PLT can be used in PDX. It is possible, however, that baking the liner to 400°C will eliminate the need for ESD, discharge cleaning, et cetera.

The most conspicuous feature in this attempt to form and maintain a pure plasma is the divertor. Though a toroidal divertor was used with great success on the C-Stellarator in 1963,[67] it has only been recently that vigorous attempts have been made both experimentally[68] and theoretically[69] to understand the operation of a divertor.

The divertor separatrix defines the plasma size and shape without the need for a mechanical limiter. The diverted field lines carry the escaping plasma from the plasma edge to a separate burial chamber from which only a small percentage of neutralized ions and sputtered impurities can return. This function of the divertor is to "unload" the edge plasma. The divertor is also used to shield the plasma. In this function the divertor serves as an ionization and transport region that prevents energetic sputtered or desorbed neutrals from entering the main plasma body.[70] To operate at 90% ionization efficiency the density-thickness products of the divertor region must be approximately 1×10^{13} cm^{-2}.[71] For obvious reasons the more the divertor is in the "unload" mode, the less efficiently will it be operating in the "shield" mode. An idealized model for divertor action is shown schematically in Fig. 15.

The width of the divertor throat must be sufficient to capture the plasma that escapes across the separatrix. The width of the escaping plasma channel is determined by the cross field diffusion of the plasma as it flows parallel to the field lines into the divertor. Estimates of this width based on classical theory may not be accurate; also it may be difficult to form and maintain the separatrix in the precisely needed position. Hence the divertor throat will be as large as possible (~5 cm) consistent with negligible cold hydrogen reflux.

The function of the neutralizer plate is as its name implies. It bears the same thermal load as the limiter does in PLT. However, the neutralizer plate has more than 6 times the area of the limiter, hence it should not suffer from thermal stress or evaporation problems. W or Ti appear to be the leading choices for the neutralizer material.

The Neutralizer plates sit in the burial chamber. Here 80 titanium sublimation balls will be continuously depositing (except during the time when the toroidal field is on) a fresh layer of titanium on the back side of a cooled liner. Approximately 60kW of power are required for this. The sublimation rate is such that the equivalent weight of 1 ball per day will be used up. Atomic hydrogen impinging on the gettered surfaces is expected to have a 40% sticking coefficient.[72] Molecular hydrogen has a 10 times less probability of sticking. The pumping speed in the burial chamber should be adequate to eliminate the reflux of neutralized cold gas into the plasma.

There should be no need in PDX for an expanding mechanical limiter to suppress skin currents. Proper programming of the divertor fields should constrain the plasma to initiate in a narrow hexapole magnetic null on the vacuum vessel midplane.

If the divertor is successful in removing impurities and cold neutrals from the edge, it will facilitate MHD stable operation in a variety of cross sections. Elongated cross sections are thought to further reduce transport losses.[73]

PDX is designed to be a more versatile tokamak than PLT. Both surface conditions and plasma behavior can, in principle, be modified in many ways not possible on PLT. These extra degrees of freedom may increase the duration of a contamination free hydrogen plasma to several seconds.[74]

C. The Tokamak Fusion Test Reactor (TFTR)

The primary goal for TFTR is to operate for a short interval (0.1 sec) during its 1 sec discharge with a reactor grade D-T plasma in which $Q \gtrsim 1$, where Q is the ratio of released fusion energy (22.5 MeV per fusion event) to injected beam power. It is expected that TFTR will begin operations in the full power D-T mode in late 1982. D-T experiments would continue for 4 years, and a total of 4000 D-T pulses. The pulse repetition period would be 350 sec. with a single discharge persisting for .6 to 1.0 seconds.[75] A schematic of TFTR is shown in Fig. 16. Though its appearance in this drawing is quite similar to PLT there are numerous differences which result from the use of tritium and high power neutral beams.

The basic vacuum vessel is similar in construction to PLT. It would consist of alternating sections of expanded bellows and cylindrical shells. Initial plans call for 16 bellows sections each consisting of 35 convolutions of .040" thick fusion welded rings. The cylindrical shells are approximately 98" I.D. and .287" thick. The connections between the shells and the bellows are 98" ID, 101" OD cylindrical rings that lie in the vertical plane. These rings bear the major portion of the atmospheric pressure on the vessel. The bellows are shielded both inside and outside the vacuum vessel. The inner shield is to protect the bellows from radiation and particle fluxes from the plasma. It would also eliminate chronic tritium escape caused by the diffusion of implanted charge exchange tritium.[76] The outer bellows' shield will be a vacuum tight assembly. The trapped volume could then be forepumped in synchronism with the evacuation of the main chamber. This would reduce buckling strains on the bellows.

The penetration time ($\tau \sim .01$ sec) for the OH and EF fields into this vacuum vessel is not negligible from the standpoint of the dynamic feedback stabilization of the plasma column or magnetic

magnetic loop diagnostics.

TFTR will be constructed so that it can be divided into halves with the separation being along a vertical plane. To accommodate this, the vacuum vessel will have two diametrically opposed locations where it can be cut completely through by grinding. It would be rejoined by welding.

The liner concept that is being tested on PDX will have a rebirth in TFTR. In addition to the first use it will be fabricated and installed in such a manner as to protect the bellows sections from accidental deposition of neutral beam energy. This might happen if the plasma position changed dramatically during beam injection. The intense beam power, 6 kW/cm^2, would melt all but the most refractory materials. For this reason W plates mounted on Mo substrates are chosen as the liner material.

In the initial plans, limiters will be installed at two locations 180^o apart. The limiters will be curved W bars (IR = 100 cm) of 5.1 cm radial thickness, 14.5 cm width and 143 cm arc length, weighing 204 kg. These limiters may bear the greater part of the plasma thermal energy at the termination of the discharge. If 50% of the thermal energy were deposited uniformly in one such limiter its temperature would rise by about 100^{o}C. A uniform energy deposition profile is far too optimistic an assumption. Other methods of terminating the discharge without the energy going into the limiter are being considered. One such technique would be to inject in impurities by either a pulsed gas or a laser produced metal jet[77] technique. Impurities would cause the plasma to cool by radiation instead of conduction.

During the first year of TFTR operation unlimited access to the machine will be possible. Removal and replacement of various diagnostics can be carried out in the traditional fashion - by a man with a wrench. However, after the commencement of D-D or D-T operation remote handling will be the only way to remove or insert diagnostic equipment into the torus.

The pumping system is designed to handle either radiactive or stable working gases. Eight 16" Hg diffusion pumps, mounted in the neutral beam injection lines, will provide a pumping speed of about 10^4 liter / sec (for air). Hg diffusion pumps were chosen over oil pumps because of tritium-hydrocarbon exchange reactions. (Other reasons can be found in reference 75.) The proposed upper limit base pressure for TFTR is 4×10^{-8} Torr. Wall treatment by one of many methods should cause this specification to be easily exceeded.

The neutral beam lines will be about 9 meters long with a 24" Hg diffusion pump attached near the source. Again a severe problem is expected from the cold neutral efflux from the guns. The problem is somewhat alleviated by the long conductance time constant, ~ 900 ms, through the neutralizer tube. This might allow an experiment to be concluded before cold neutrals flood the discharge. As a safeguard, a program is planned to develop the necessary cryopumps or sublimation pumps to eliminate this problem.

Backing the torus diffusion pumps will be an elaborate system to prevent tritium contamination of the laboratory and environs. The main components are shown in Fig. 17, along with the tritium generator and dispenser. The spent tritium will not be recycled. Instead it will be trapped in zirconium aluminum getter cannisters. The zeolite trap - mechanical pump arrangement will be used for non-radioactive gases. A series of ballast tanks will allow testing of all evacuated gases for radioactivity before release to the atmosphere is made.

The tritium generator and dispensing equipment will be enclosed in an inert gas filled enclosure. Connections to TFTR will be made through double vacuum jacketed tubulation. The method chosen for tritium storage is a solid trihydride UT_3, in foil form. A palladium diffuser capability is available for additional purification. A total of 24,000 Curies will be stored in each of two separate tritium generators. Approximately 100 Curies of tritium is needed for each D-T "burn".

The planned operating mode for TFTR is to evacuate the torus to its base pressure, close off the gate valves to the diffusion pumps and inject sufficient tritium to fill the torus to its operating pressure, ~ 3×10^{-4} Torr. The discharge will then be struck, the plasma Ohmically heated and the guns fired into the plasma. After the discharge the gate valves will be opened and the torus evacuated. The pump down time constant is estimated to be 4 seconds.

Not the least problematical in this scenario is the recycling of the tritium during the initial part of the discharge.

Tritium that leaves the plasma will probably bury itself a few Angstroms deep in the liner. Hydrogen and deutrium imbedded during previous discharges may be knocked out. The exact unfolding of the recycling process will thoroughly depend on the past history of the tokamak. Results on ATC have indicated that it may take about 1000 shots in a single working gas for the memory effects from earlier discharges to be wiped out.[78]

TFTR is still in its early planning stage. Its final design, even its final approval, will depend on the degree with which the objective of PLT and PDX are fulfilled.

SUMMARY

The construction of PLT has been along traditional lines: Hg diffusion pumps and stainless steel vacuum vessel. The long duration of PLT discharges may necessitate unconventional modes of wall cleaning or plasma behavior. These include ESD of adsorbed gases or the formation of a cold, transient plasma blanket.

Additional approaches to impurity control will be possible on PDX. An axisymmetric divertor is the most important option. This will require having high current carrying coils inside the vacuum vessel. Developmental programs in surface science and vacuum technology will contribute to PDX through the design of liners, neutralizer plates and a burial chamber .

Experiments performed on PLT and PDX will greatly affect the design ot TFTR. However, numerous problems in TFTR will have no counterparts in any earlier tokamaks. These include the handling of large quantities of tritium and the injection of tens of megawatts of neutral beam power. We will be relying on many laboratories for the technology to handle these.

ACKNOWLEDGEMENTS

It is a pleasure to acknowledge useful conversations with J. Cecchi, H. Garber, D. Grove, D. Jassby, O. Kraus, G. Lewin, G. Martin, D. Meade, D. Mullaney, J. Schivell, F. Seidl, and J. Sinnis.

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TABLE CAPTIONS

Table I. Parameters of PLT and TFTR

*For a deuterium discharge and deuterium beams the flux of neutrons to the wall would be 4×10^9 n/cm^2.sec and the power outflux due to fusion reactions would be 2.8 kW.

**The reason that compression TFTR does not increase the power output due to fusion reactions is that the background plasma is 100% deuterium and the beam is 100% tritium. The beam energy (154 keV) is already at the peak of the fusion cross section. Background plasma comparison with beam - plasma (T-D) fusions. This is the two component concept in action.

TABLE I

Parameters of PLT and TFTR

Parameter	Units	PLT	TFTR Before Compression		TFTR After Compression
Magnetic Field on Axis	(kΓ)	50	41		60
Plasma Current	(MA)	1.0	0.67		1.0
Major Radius of Discharge	(cm)	130	314		295
Minor Radius of Discharge	(cm)	45	56		45
Pulse Length	(sec)	1	~.3		~.6
Hydrogen Isotopes in Plasma		H in plasma & beam	T plasma;	D beam	
Z_{eff}		2	2		2
Time of these parameters	(sec)	.3	.120		.130
Neutral density on axis	(cm^{-3})	8.3×10^{7}	1.6×10^{7}		8.4×10^{6}
Neutral density at edge	(cm^{-3})	3×10^{10}	1.3×10^{10}		1.3×10^{10}
Electron density on axis	(cm^{-3})	6×10^{13}	4.7×10^{13}		1.05×10^{14}
Electron Temperature on axis	(eV)	5260	6040		11,200
Hydrogen Temperature on axis	(eV)	6420	5160		9,960
Ion Confinement Time	(sec)	.04	.06		.1
Energy Confinement Time	(sec)	.09	.09		.09
Ohmic Heating Power	(kW)	62	180		180
Neutral Beam Parameters:			Beam 1	Beam 2	
Beam on	(sec)	.2	.05	.09	None
Beam off	(sec)	.3	.09	.125	
Beam Current	(A)	100	480	155	
Beam Voltage	(keV)	40	83	124	
Power outflux due to:	(kW)				
Fusion Reactions		0*	1420		1360**
Radiation		151	61		97
Charge Exchange		934	550		973
Electron Thermal Conductivity		400	1510		1670
Ion Thermal Conductivity		510	1050		1220
Electron Mass Transport		70	330		318
Ion Mass Transport		107	248		273
Flux of Neutrons to Wall	($cm^{-2}sec^{-1}$)	0*	4×10^{12}		4×10^{12}

TABLE II

Approximate Impurity Concentrations in TFTR at the End of a "Standard" Discharge

Source	Impurity Concentration (NI/Ne)	Impurity
Particle Induced Desorption	.02	C, O
Blistering	(.01)	Fe
Physical Sputtering	.0075	Fe
Neutron Sputtering	5×10^{-4} to 5×10^{-7}	Fe
Photodesorption	5×10^{-4}	C,O
Limiter Evaporation	$<10^{-6}$	W
Limiter Fragmentation	$<10^{-6}$	W
Chemical Sputtering	?	C
Impurity Induced Sputtering	?	Fe

TABLE III

Ceramic Sections in PLT Vacuum Vessel

Composition	96% Al_2O_3	4% SiO_2
ID	37-3/4"	
OD	38-3/4"	
Length	3"	
Connection to vacuum vessel	430 stainless steel flange attached to ceramic using Ti H_2 brazing with Silver Copper Eutectic BT	

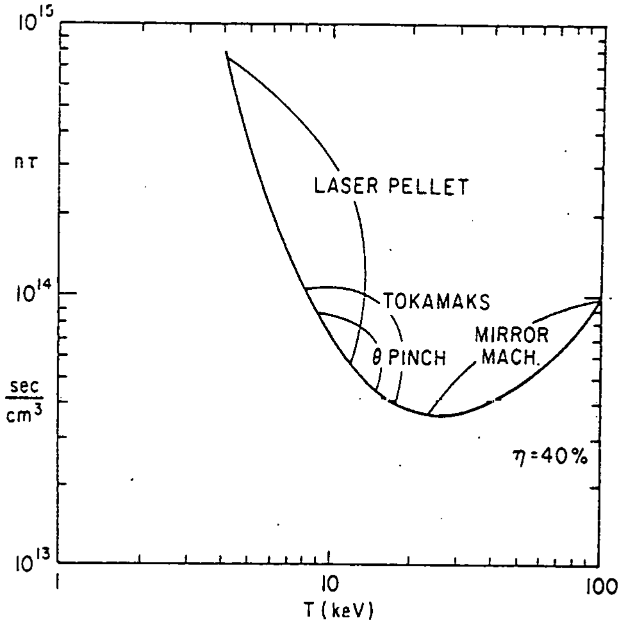

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Fig. 1. The Lawson breakeven criterion for fusion devices with 40% efficiency in energy recycling. Each scheme for reaching breakeven (the solid line) aims for a different region in the nτ vs. T plane. They are constrained to these regions by as diverse reasons as heating technique, collision frequency and economics.

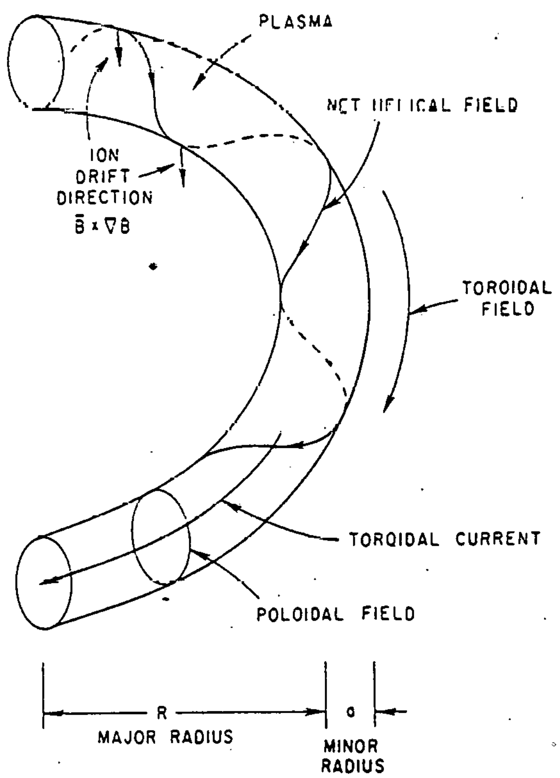

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Fig. 2. Schematic of a tokamak showing the main fields and currents. The addition of a poloidal field to the toroidal field produces a net helical field. This causes ions on the upper half plane of the torus to drift back into the main body of the plasma.

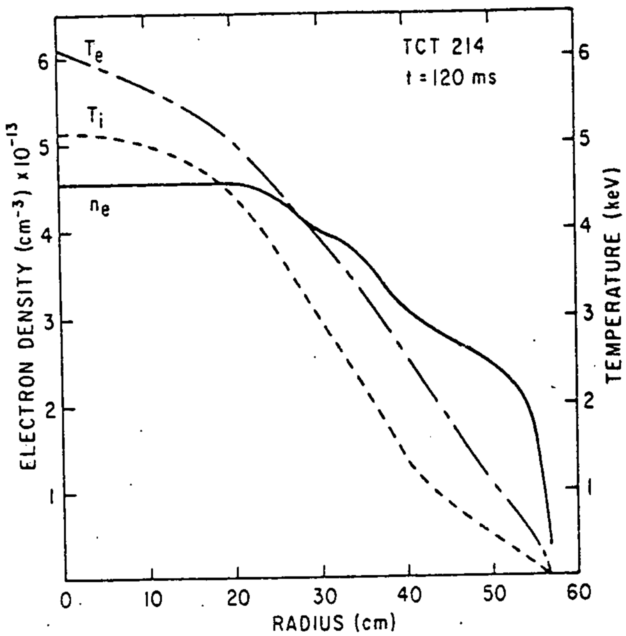

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Fig. 3. Calculated radial distributions of electron density and temperature, and ion temperature for a precompression TFTR plasma. (Other parameters are listed in Table I). Plasmas with up to 85 cm minor radius can be formed. Tokamak discharges with profiles that are peaked on axis are termed "standard". Profiles with density or temperature minima on axis are discussed in the text.

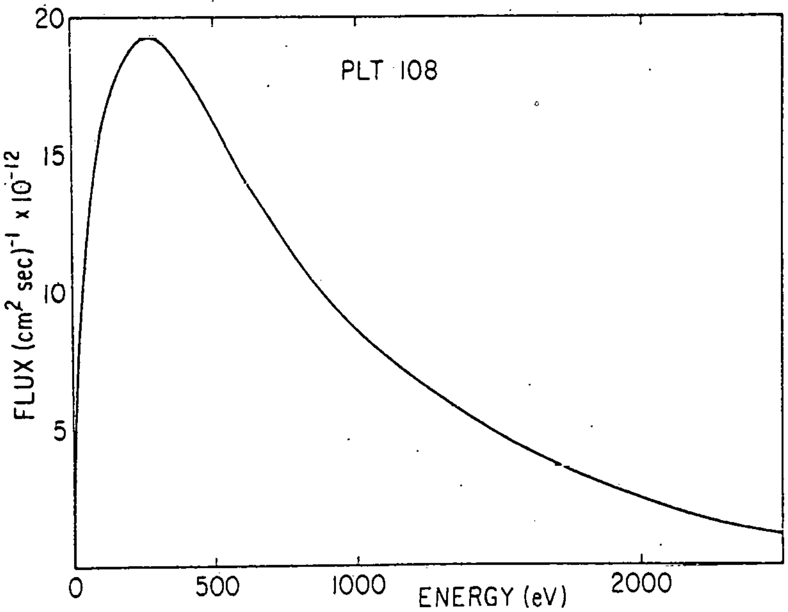

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Fig. 4. Flux to the wall of charge exhange neutrals that escape from PLT at the time of its peak central ion temperature. Though the central ion temperature is over 5 keV the majority of escaping neutrals have a much lower energy. The flux unit is (per eV).

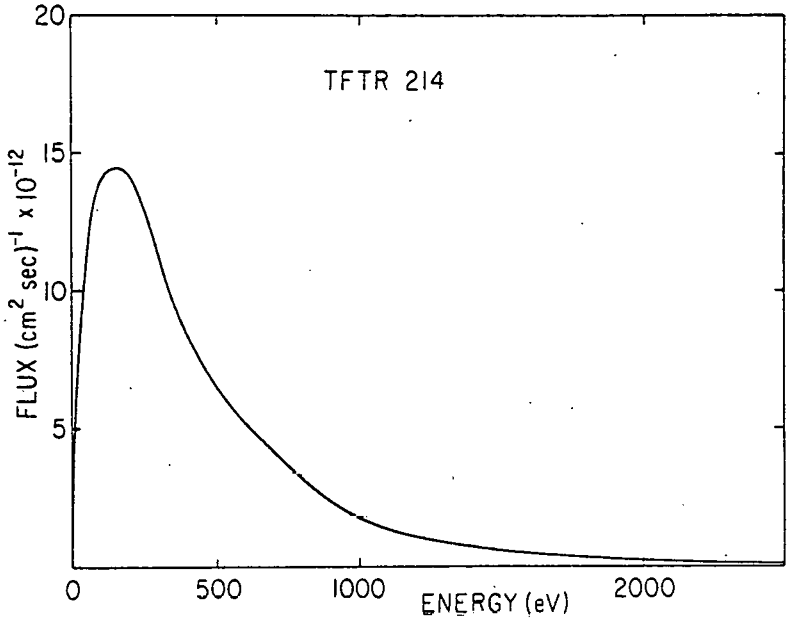

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Fig. 5. Flux to the wall of charge exchange neutrals that escape from TFTR immediately after neutral beam heating. The lower neutral density in TFTR compared to PLT (see Table 1) causes fewer energetic neutrals to bombard the wall. The flux unit is (per eV).

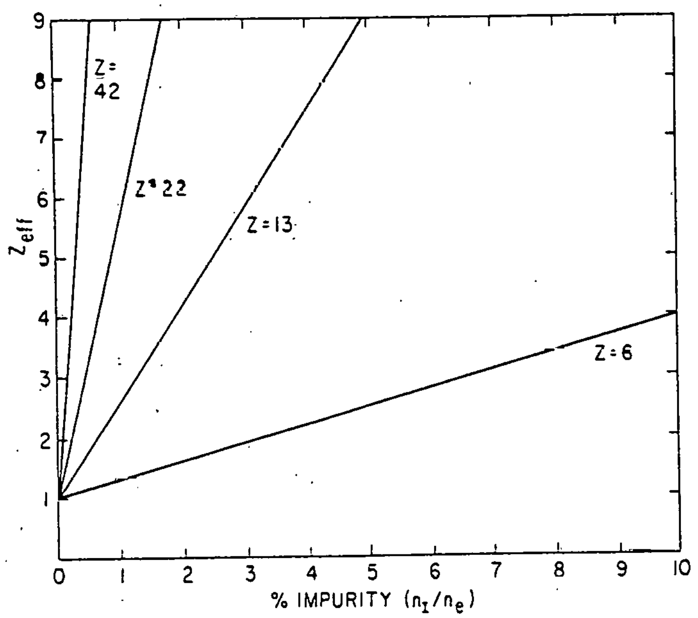

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Fig. 6. Z_{eff} as a function of impurity ion concentration in a hydrogen plasma. The four curves are for fully stripped C, Al, Ti and Mo ions.

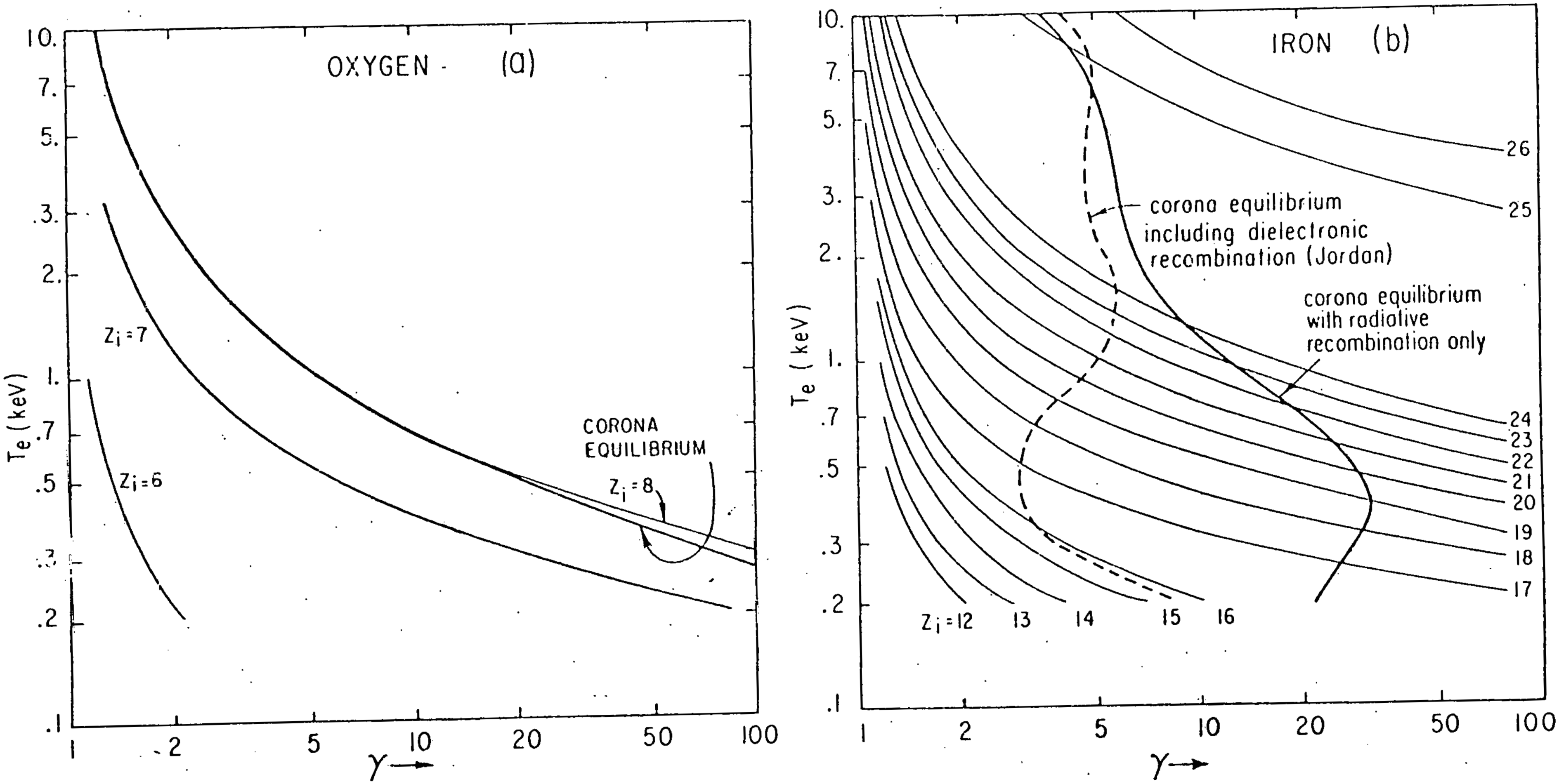

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Fig. 7. Enhancement factor γ for recombination radiation from oxygen and iron ions in a hydrogen plasma. γ is the factor by which the power emitted by recombination radiation exceeds the bremsstrahlung losses. (Ref. 27)

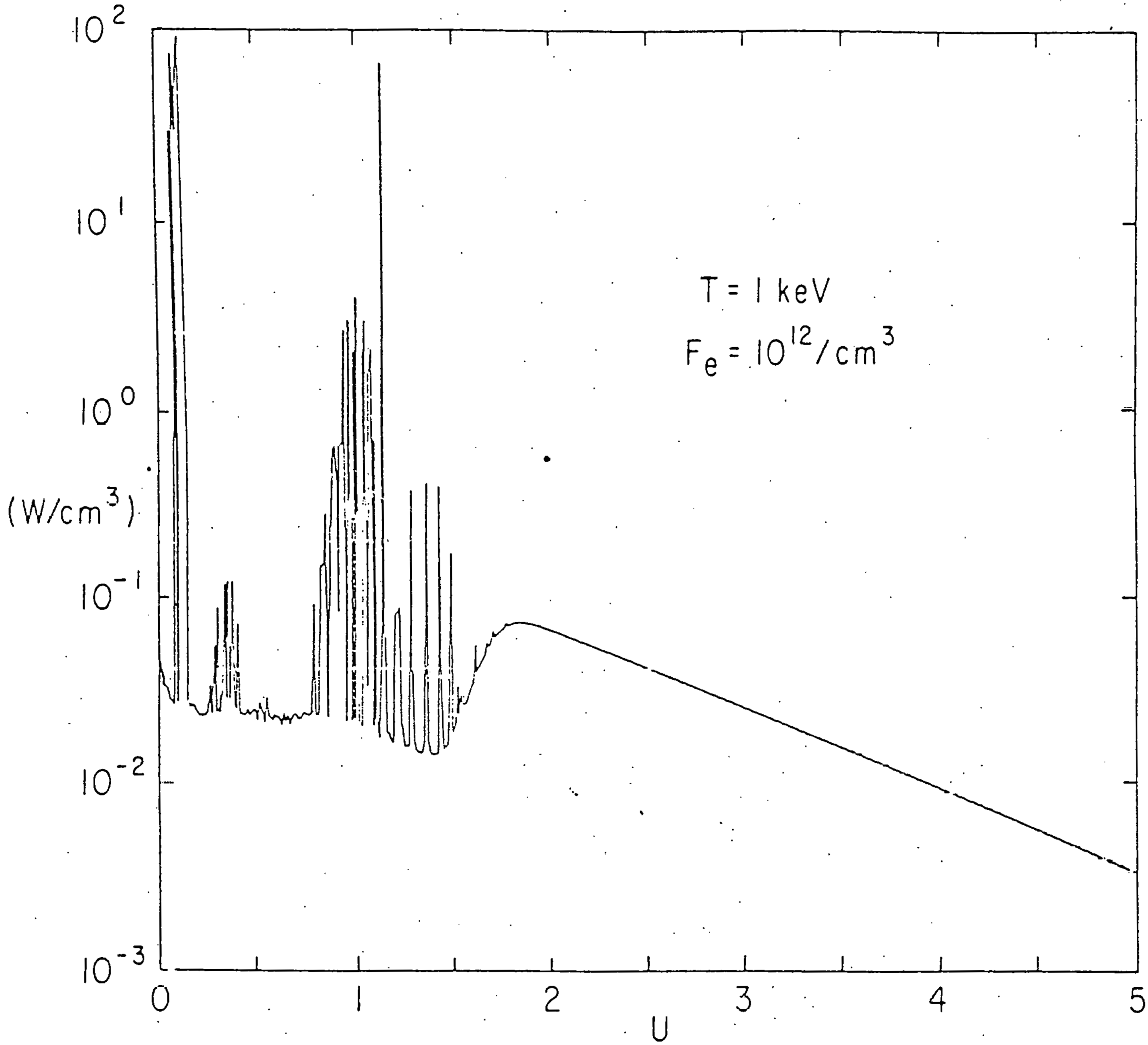

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Fig. 8. Power loss spectrum from a 1% Fe contamination in a 1 keV hydrogen plasma. The scale of the abscissa is in units of 10 keV. (Ref. 29)

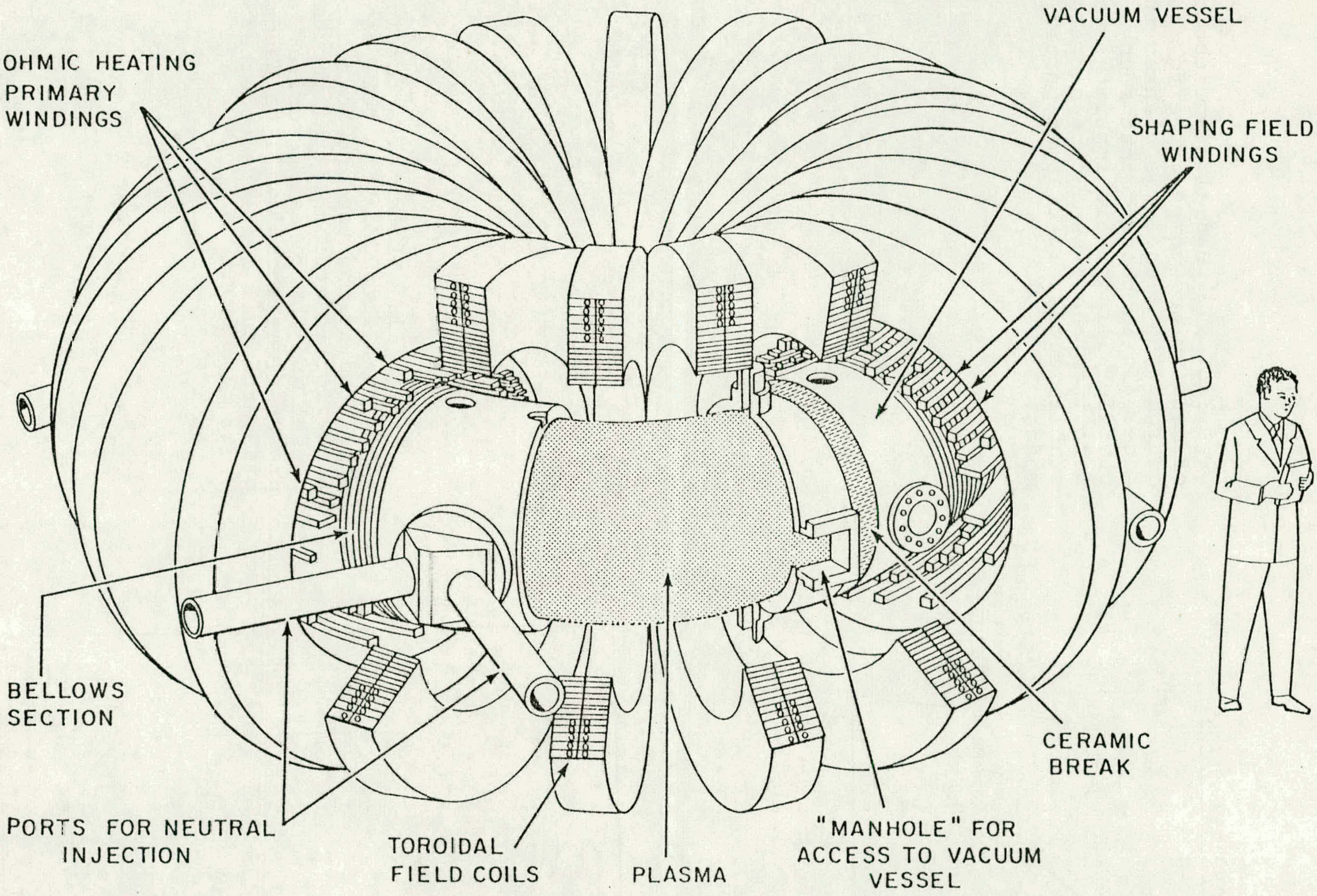

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Fig. 9. A schematic drawing of the Princeton Large Torus. A massive torque frame (not shown) holds the toroidal field coils in position. The Ohmic heating and shaping field coils are nested inside the toroidal field coils. One port on the vacuum vessel is large enough for a man to crawl through. This manhole should greatly facilitate mounting diagnostic equipment inside the tokamak.

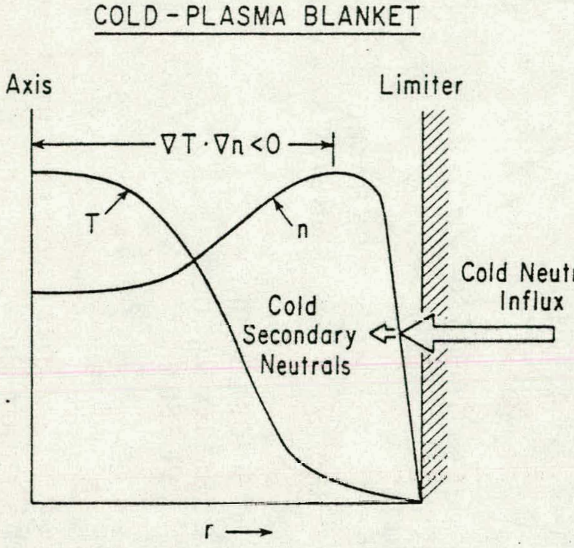

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Fig. 10. Radial distribution of density and temperature for a discharge with a cold gas blanket on its edge. It is thought that such a density profile will inhibit impurity transport into the plasma core and also stabilize certain trapped particle modes.

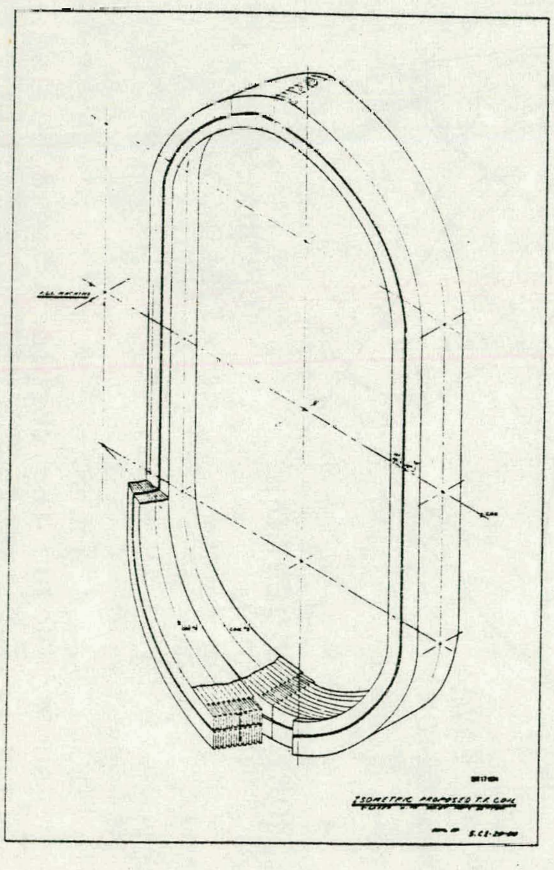

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Fig. 11. Sketch of a toroidal field coil for PDX. The coil can be separated at the top and bottom to allow for insertion of the vacuum vessel and Ohmic heating coils. Large presses (not shown) will be used to clamp the coil sections together.

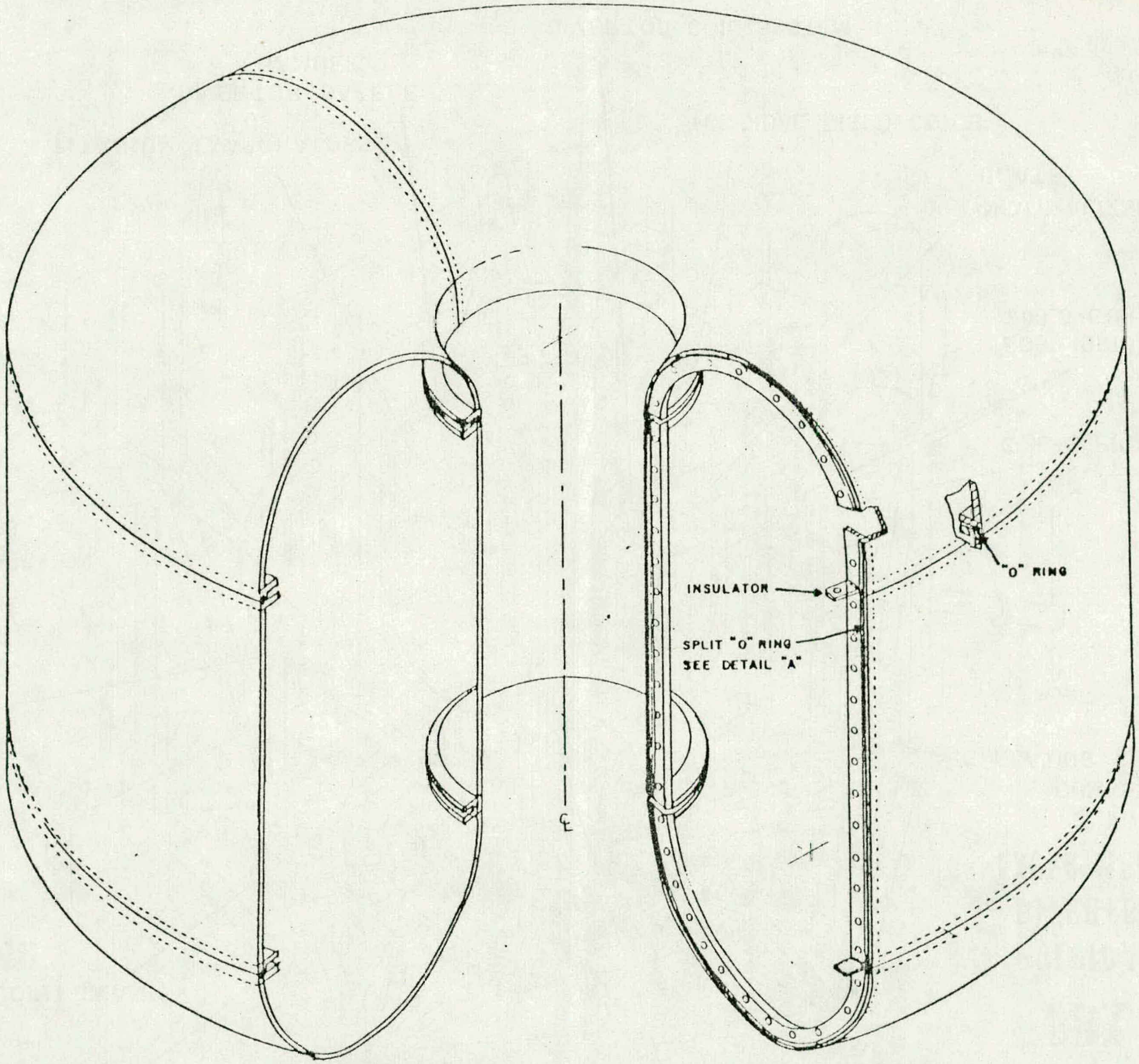

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Fig. 12. Sketch of the PDX vacuum vessel. The vessel consists of four major parts: the top, bottom, inside cylinder and outside cylinder. These will be connected to each other with double viton seals. The inner space between seals will be forepumped to eliminate gas permeation into PDX. The azimuthal insulating brake will be formed by thick viton seals and an alumina spacer.

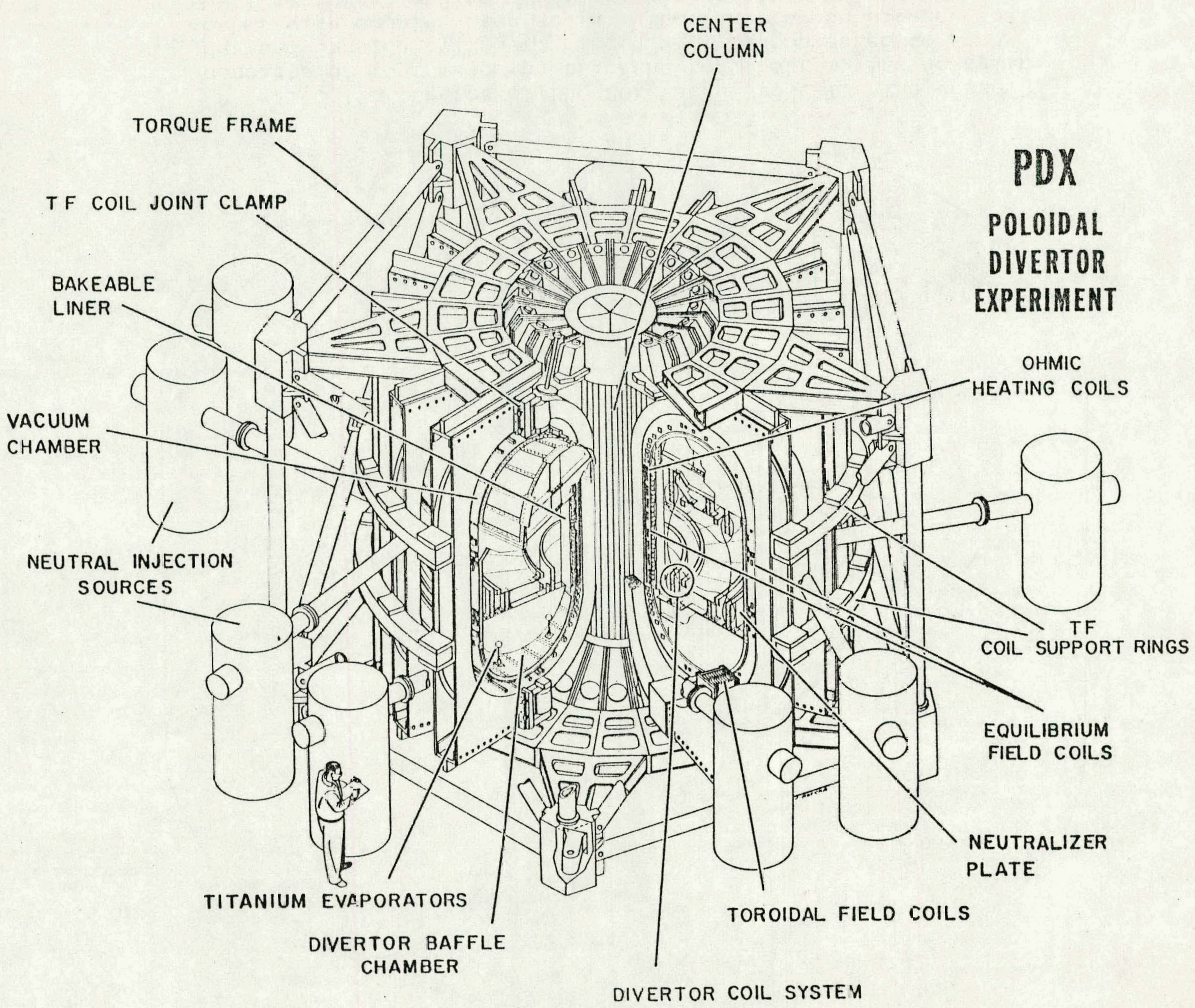

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Fig. 13. Trimetric view of PDX.

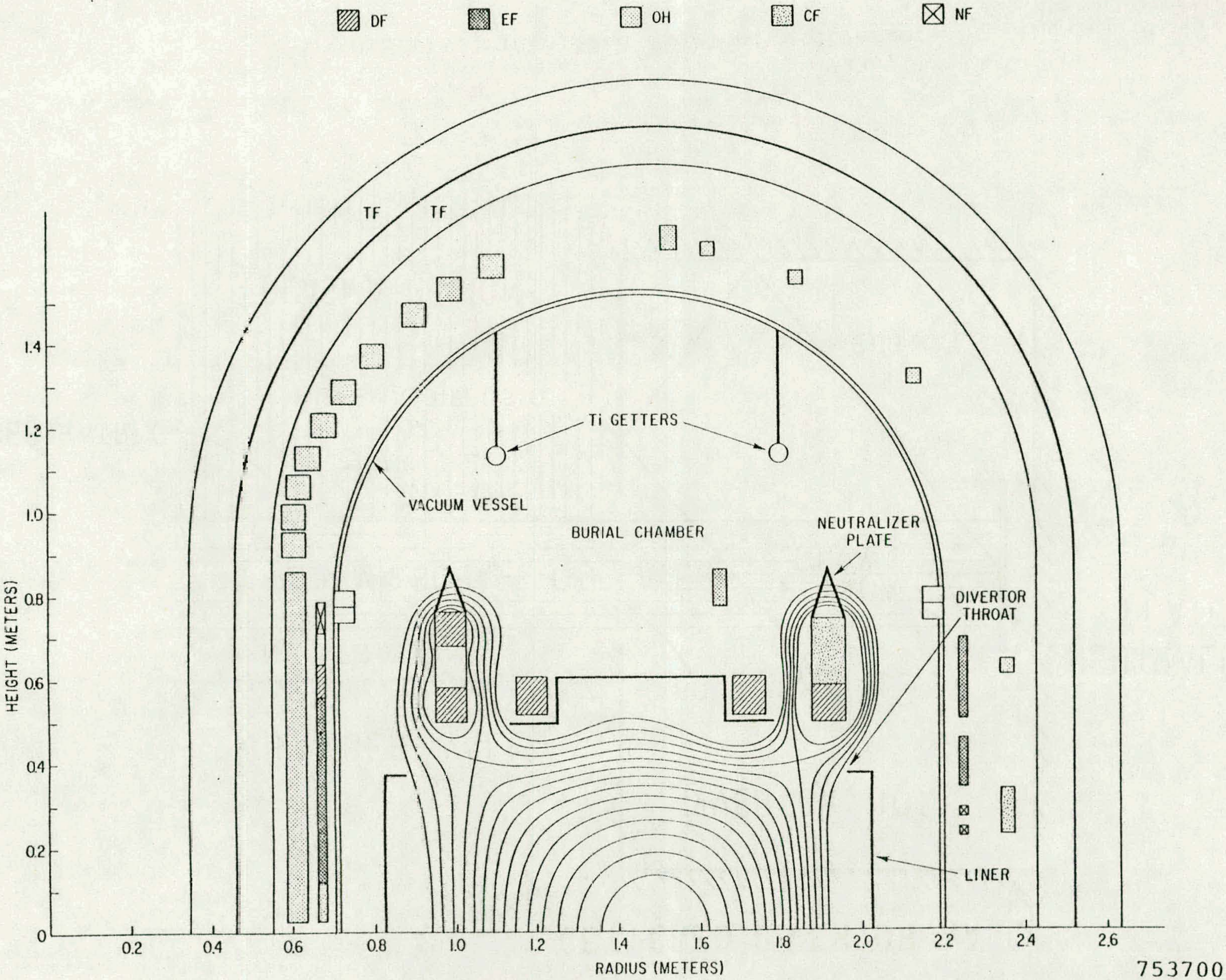

Fig. 14. Cross section of upper half plane of PDX. The DF coils divert magnetic field lines from the plasma edge onto neutralizer plates in both upper and lower (not shown) "burial chambers". The magnetic fields can be programmed for three different divertor configurations: Inner, outer, or both. Getter balls sublimate titanium onto surfaces in the burial chamber for rapid pumping of neutralized hydrogen ions. A 300°C liner surrounds the main plasma volume.

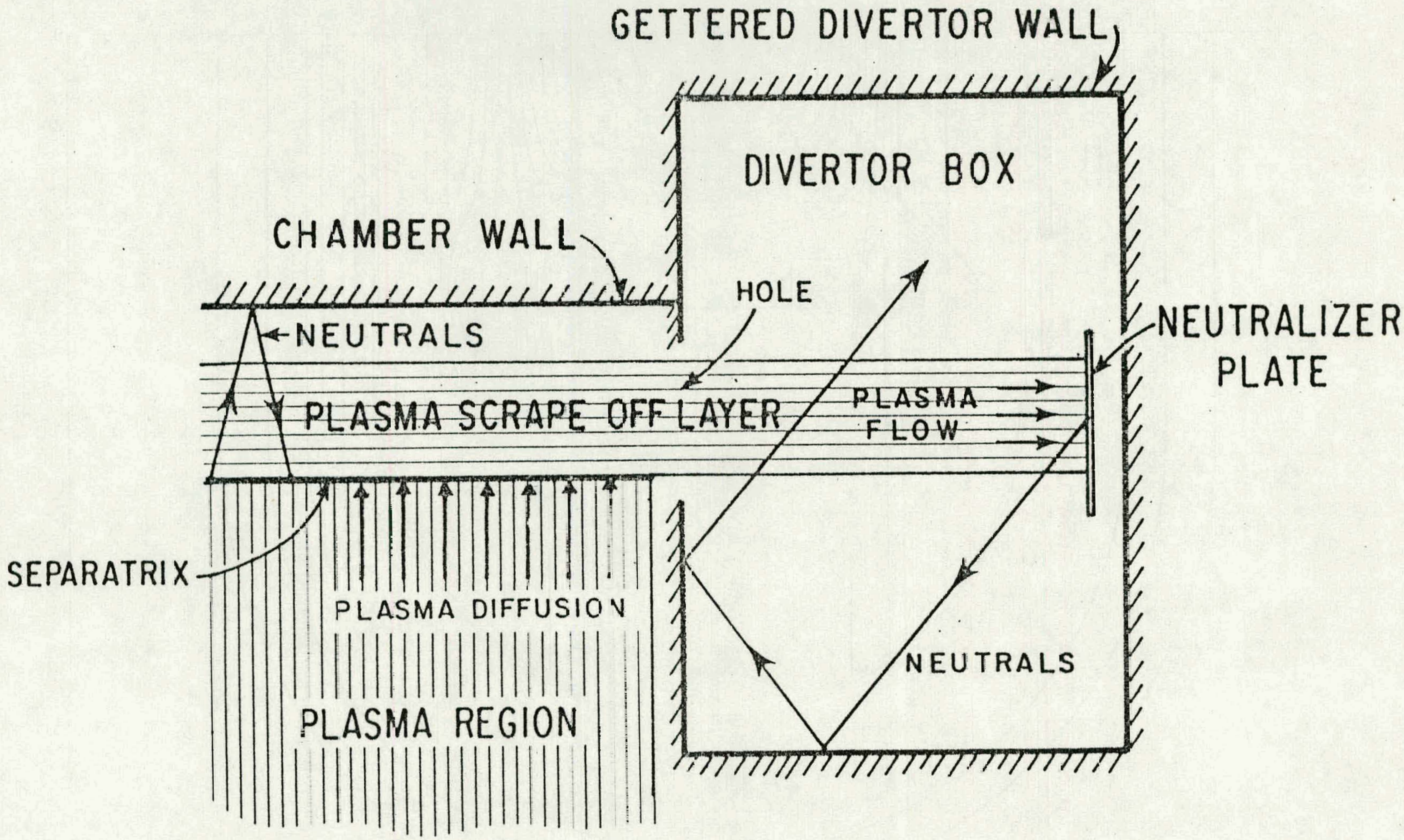

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Fig. 15. Idealized model of a divertor.

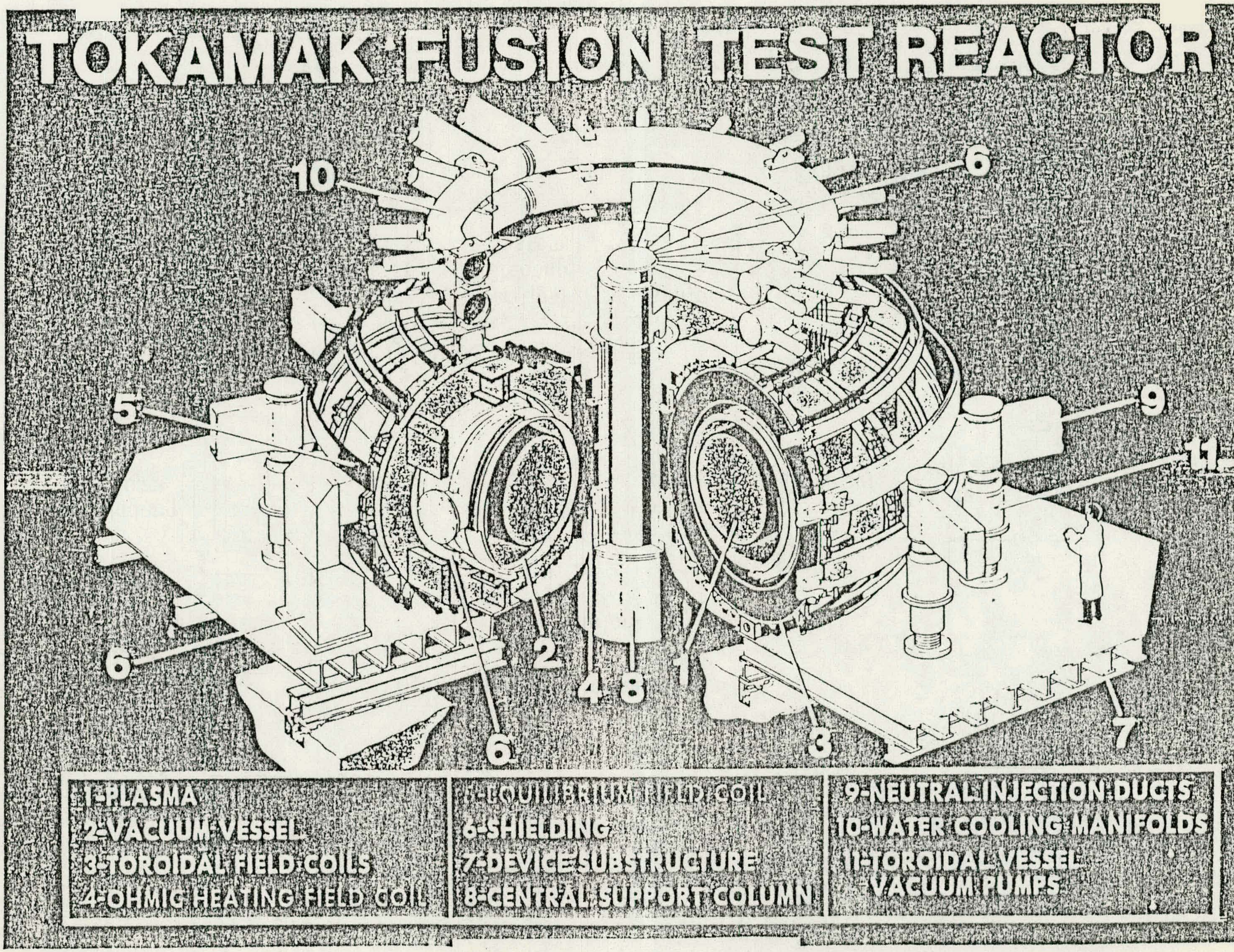

Fig. 16 Trimetric view.

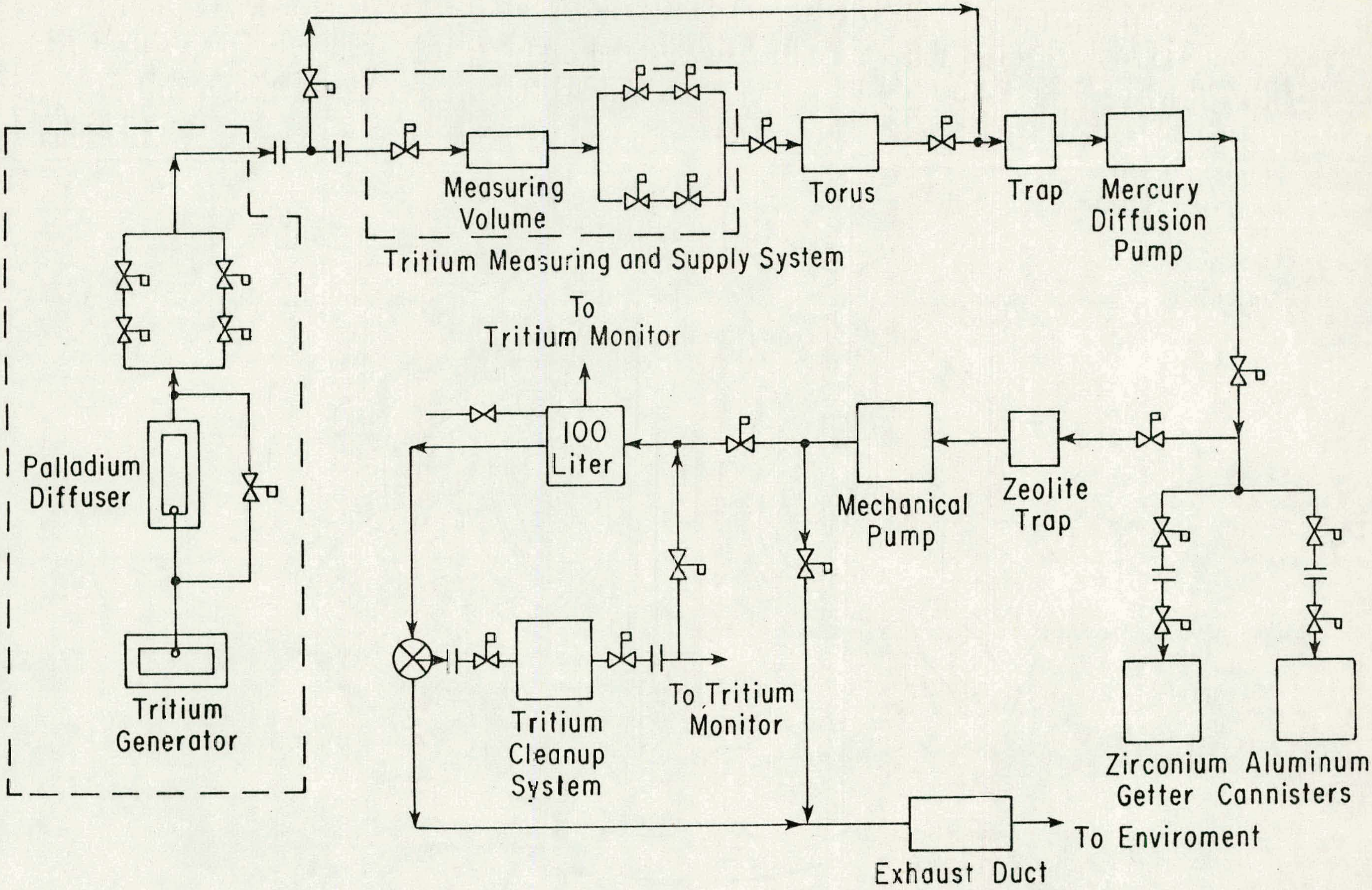

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Fig. 17. Tritium generator and dispenser and TFTR pumping system. (Prepared by H. Garber)