IMPURITY AND PARTICLE CONTROL FOR INTOR*

INTOR Group

Prepared by D. Post, IAEA, Vienna

PPPL--2192

DE85 007136

ABSTRACT

The INTOR impurity control system studies have been focused on the development of an impurity control system which would be able to provide the necessary heat removal and He pumping while satisfying the requirements for (1) minimum plasma contamination by impurities, (2) reasonable component lifetime (~ 1 year), and (3) minimum size and cost. The major systems examined were poloidal divertors and pumped limiters. The poloidal divertor was chosen as the reference option since it offered the possibility of low sputtering rates due to the formation of a cool, dense plasma near the collector plates. Estimates of the sputtering rates associated with pumped limiters indicated that they would be too high for a reasonable system. Development of an engineering design concept was done for both the poloidal divertor and the pumped limiter.

*Presented at the Tenth International Conference on Plasma Physics and Controlled Nuclear Fusion Research, London, September 12-19, 1984, IAEA CN/44-G-I-3.

1. INTRODUCTION

The INTOR impurity and particle control system must be able to both absorb the 120 MW of alpha particle heating power and to remove the alpha particles at the rate they are produced. This must be accomplished without contamination of the plasma by impurities and without a large erosion rate of the first wall components. The major candidate systems studied were a poloidal divertor and a pumped limiter. The studies have consisted of an assessment of the experimental and engineering data base for impurity control systems based on current experiments, the use of sophisticated computational models to extrapolate to the operating parameters and performance for INTOR, and the development of an engineering concept for the design of a system for INTOR. The major systems studied have been the poloidal divertor and the pumped limiter (Fig. 1).

The major impurity control problem is likely to be sputtering of collector plate materials by energetic plasma ions and charge exchange neutrals. The energy of the plasma ions that strike the collector plate is largely determined by the sheath potential which is several times ($\sim$ 2-4) the electron temperature of the plasma near the collector plate. Based on extrapolations from experiments and the use of computational and analytic models, it is expected that the temperature at the plasma edge of INTOR should be in the 100-200 eV range. The density should be in the 10^{13} - 3×10^{13} cm^{-3} range. Thus the sheath potential and ion energy for the plasma that is incident on a limiter should be in the 300-800 eV range. This will lead to large sputtering rates for the limiter.

If the temperature of the plasma near the collector plate can be reduced to 20-30 eV, then materials can be found which have sputtering thresholds above the incident ion energy. One method for lowering the plasma edge temperature is the use of large impurity radiation losses from the plasma edge. This is observed on current tokamaks where low-Z radiation losses from the edge can exceed 70° of the heating power [1]. However, the fraction of power radiated in experiments with high power auxiliary heating is usually less than this. Modelling studies indicate that the production of a very cool edge by impurity radiation requires significant levels of impurities at the plasma edge. If these impurities were present in the plasma center, they would cause significant energy losses, and are thus unacceptable.

A second way of producing a low temperature plasma is to increase the recycling rate. Neglecting radiation losses, it can be shown from sheath theory [2] that the temperature and density near the collector plate are largely determined by the localized recycling coefficient R, the number of times an ion-electron pair strikes the collector plate after leaving the main plasma. In this formalism, $T=To/R$ and $n=no\ R^{3/2}$, where To and no are the edge temperature and density with no recycling ($\sim$ 600 eV and 10^{11} cm^{-2}). Both modelling calculations [3-6] and experiments on ASDEX [7], D-III [8], and PDX [9] indicate that this can be accomplished by the use of a suitably designed poloidal divertor. A cool, dense plasma ($n_e \gtrsim 10^{14}$ cm^{-3}, $T_e \lesssim 30$ eV) can be produced near the collector plate by intense localized recycling of the plasma and neutral gas. The low temperature of the diverted plasma minimizes the erosion, and the high density of the plasma provides a high neutral density which eases the helium pumping speed requirements.

2. GENERAL MATERIALS CONSIDERATIONS

Impurity control components are exposed to high particle and heat fluxes that can result in high sputtering erosion rates and high fluxes of 14 MeV neutrons which will degrade the bulk properties. The materials used for impurity control must therefore be resistant to erosion losses and radiation damage, capable of operating at elevated temperatures, and at the same time not be a source of contamination to the plasma.

Among the properties important for material selection are the thermophysical properties, mechanical strength and ductility, fatigue and crack growth behavior, coolant and hydrogen compatibility, radiation swelling and creep, and sputtering erosion behavior [e.g., 10]. The desired thermophysical properties are those that minimize the thermal stresses. The mechanical strength and ductility should be adequate to accommodate the weight loads, coolant pressures, thermal stresses, and electromagnetic forces. For high-cycle machines like INTOR, the materials should exhibit favorable fatigue, crack growth, and stress corrosion behavior. The materials should also exhibit low radiation swelling and creep rates. The surface sputtering rate should be low enough to provide extended lifetimes and to keep impurities to an acceptable level in the plasma.

A survey of available materials indicates that no one material satisfies both the surface sputtering and structural requirements. Hence, the design of impurity control components incorporates separate plasma side materials which are attached to a structural material selected to meet the strength and radiation damage requirements. The division of plasma side and structural materials allows greater flexibility in the selection of materials but also creates additional difficulties associated with attachment.

The candidate materials considered for impurity control are listed in Table 1. These materials were selected from a larger pool of possible materials based upon the property requirements listed above. The plasma side materials are divided into low-Z, medium-Z, and high-Z materials. At low plasma edge temperatures, ($\lesssim$ 50 eV) all materials may be used but high-Z materials are expected to exhibit very low sputtering erosion, and therefore they are predicted to have the greatest lifetimes. At higher edge temperatures, both medium- and high-Z materials are unacceptable due to excessive self-sputtering. The permissible plasma side materials are those whose self-sputtering coefficients never exceed unity, which limits the selection to materials whose atomic weights are at or below the atomic weight of SiC. The candidate heat sink materials are copper alloys and transition metal alloys.

Material selection for INTOR has focused on the use of low-Z materials Be and C, for plasma edge temperatures $\gtrsim$ 100 eV and the use of high-Z materials, W and Ta for plasma edge temperatures $\lesssim$ 50 eV. Be is favored over C because C is known to exhibit enhanced chemical sputtering and because C has rather limited irradiation lifetimes. W is favored over Ta because Ta is susceptible to hydrogen embrittlement and thus does not appear to be compatible with the DT environment. Copper alloys have received the most attention as heat sink materials since they are readily available, are easily fabricated, and are capable of operating at the anticipated operating temperature ($100 < T < 300°C$).

3. POLOIDAL DIVERTOR

A. Divertor Physics

The INTOR poloidal divertor has a short, compact configuration of the "expanded boundary" type [11] (Fig. 1a). The need for a blanket, shielding, and remote handling has led to the placement of the poloidal field coils outside the toroidal field coils. The major attraction of the divertor is the possibility of a low-temperature, high-density diverted plasma near the collector plate. The key physics questions are the credibility of (1) producing such a cool, dense plasma in the divertor chamber, and (2) producing that plasma in the "open" geometry which follows from having the poloidal field coils outside the toroidal field coils. These issues have been addressed both by an assessment of experimental data from divertor experiments on D-III [8], ASDEX [7], PDX [9], and PBX [12], and by the use of large-scale computational models [3-6] to extrapolate from these experiments to an INTOR sized device.

The experimental results from the divertor experiments indicate that a dense, cool plasma can be produced near the divertor plate by intense, localized recycling [7-9, 12]. Temperatures as low as 5 eV and densities as high as $3 \times 10^{14} cm^{-3}$ have been produced with several megawatts of auxiliary heating. There are substantial density and temperature gradients along the field lines, with the temperature at the tokamak edge near the main plasma often being a factor of 10 or more higher than the temperature on the same flux surface in the divertor.

With regard to the viability of such a "high-recycling" divertor in an "open" geometry, experiments on D-III [8] and PBX [12] with such an open geometry indicate that if the diverted plasma is sufficiently wide so that the neutrals formed at the collector plate are ionized in the diverted plasma, a high-recycling divertor is formed.

Large-scale computational models which calculate the plasma transport in the plasma edge both along and across the flux surfaces self-consistently with particle and energy sources due to the recycling neutral gas [3-6] have been used. The neutral transport is usually computed in realistic two-dimensional geometries with detailed models for atomic processes and wall reflection and sputtering. These codes have been used to carry out modelling analyses of divertor experiments with reasonable agreement [e.g., 6]. These models indicate that the INTOR divertor should be able to operate in the "high-recycling" regime. The calculations of each delegation indicate that the peak temperature of the diverted plasma at the collector plate should be 30 eV or less and that the peak density should be $10^{14} cm^{-3}$ or greater (Fig. 2). The low temperature eases the erosion problem, and the high density indicates that the helium ash can be exhausted with modest sized pumping systems.

One major result from these studies is the evolution of poloidal divertors from the very large systems envisaged in many early reactor designs (Fig. 3) to divertors of the "PDX" and "ASDEX" type with internal coils, and then to the short, compact divertors of the type planned for INTOR (Fig. 1). The ratio of the volume of the divertor chamber to the volume of the plasma has been reduced from ~ 2 to ~ 0.15. The method of impurity control has shifted from high speed exhaust requiring massive pumping systems, to "high-recycling" divertors with very low pump speed systems, and relatively stagnant flows ($v_{\parallel} / v_{sound} \ll 1$) into the divertor. The emphasis on impurity control has shifted from "impurity shielding" by the plasma edge to minimizing the impurity production at the collector plate where the plasma comes into contact with the wall and where most of the power falls.

B. Divertor Collector Plate Design

The collector plates receive most of the particle flux and power which enters the divertor, and hence experience the most severe environment of any of the plasma side components. The goals of the design studies have been to develop collector plate designs that can safely and reliably remove the power deposited on the surface, that have extended lifetimes (> 1 y), and that can also satisfy the physics requirements.

The consequence of the low plasma temperature at the divertor plate is that sputtering erosion is completely eliminated for high-Z materials (Table II). The use of tungsten at the plasma side material provides an additional benefit since no vaporization or melting is predicted to occur for the reference disruption conditions. The elimination of erosion on the collector plates means that plasma side material can be a thin layer (~ 1 mm) rather than a thick plate (1-2 cm). Thermomechanical analyses performed during the INTOR studies suggest that a heat load of ~ 5 MW/m^2 represents a practical upper limit for impurity control systems. The present Phase Two A specifications (Table II) are at or above this limit. Work is in progress to determine if design changes to reduce the peak heat load such as a reduction in the angle between the plate and the magnetic field surfaces are needed.

Potential failure modes for the collector plates are erosion, excessive dimensional change due to radiation swelling or creep, debonding between the plasma side material and heat sink, and severe embrittlement which prevents the system from withstanding off-normal events. The current design eliminates erosion as a life limiting concern, but other concerns such as radiation damage could result in a short lifetime. Unfortunately, the data base for the impurity control materials is sparse, and it is not possible to characterize the long-term response of the collector plates adequately. Since the divertor lifetimes could be much shorter than the other nuclear systems, provision is made to replace it independently of the rest of the reactor.

4. PUMPED LIMITER

A. Physics studies

The pumped limiter was examined during Phase Two-A, Part One because it offered the potential for a reduced cost device that might still provide adequate impurity control. The basic configuration is a shaped, double-sided pumped limiter located at the bottom of the vacuum vessel (Fig. 1b). The potential performance of the pumped limiter was studied by an assessment of pumped limiter experiments [13-16] and by using large-scale computational models to extrapolate from these experiments to INTOR.

It is to be expected that limiters have less potential for impurity control than divertors, due to higher temperatures (~ 100-150 eV for INTOR) than divertors (~ 25 eV) near the collector plate. However, near-term experiments on JET, TFTR, and T-15 with high power auxiliary heating and limiter operation will provide data on impurity control with limiters. At the present time, carbon limiters are able to provide adequate impurity control on most present tokamaks with high power auxiliary heating. A second issue is the lifetime due to erosion. For a long pulse, high duty factor experiment such as INTOR, erosion will be an issue even if tolerably clean plasmas can be produced with limiters, since the lifetime of the limiter must be of the order of a year or great[illegible] Experimental data will not be soon forthcoming since long pulse, high duty factor machines are not likely to precede INTOR in the immediate future. Predicted sputtering rates for the pumped limiter are in the 5-50 cm/year range. These rates may be reduced by the redeposition of the sputtered material back onto the limiter. Model calculations of this indicate that the net erosion rate may be marginally acceptable in some designs. However, the confidence in our understanding of the physics of the transport of impurities in the plasma edge is not sufficiently high to base the INTOR design on a pumped limiter.

The pumping of helium is a key issue. Very promising early small-scale pumped limiter experiments [17] have been followed by large-scale modular pumped limiter experiments with high power auxiliary heating on ISX [13], PDX [14], and PLT [15] and with ohmic heating on TEXTOR [16]. With auxiliary heating ($\gtrsim$ 2 MW), neutral pressures of 1-5 $\times$ 10^{-3} torr in the pumping chamber and particle removal efficiencies of 2-5% were measured. An axisymmetric limiter will be required for INTOR, so a key question is how these pressures and particle removal efficiencies will scale when the particle exhaust is spread out on an axisymmetric structure instead of localized on one or two limiters. Experiments on this question are needed.

The performance of these limiter experiments has been modelled with reasonable success [14,18] using Monte Carlo neutral transport codes. These codes have also been used to model the INTOR limiter performance [19]. These models show that for a large area limiter such as the INTOR limiter, the neutral mean-free path is small compared to the limiter. Thus, the neutral recycling is localized on the front face of the limiter, and on the "neutralizer plate" underneath the limiter. The charge exchange flux falls almost entirely on the limiter, and on the first wall near the limiter tips and near the neutralizer plate. Thus, the erosion due to charge exchange neutrals is localized there. The first wall away from the limiter will have a very low charge exchange flux and therefore will have a very long erosion lifetime for sputtering, perhaps as long as the machine lifetime. This greatly eases the general remote maintenance requirements. There is evidence from PDX and TFTR limiter experiments that this type of localized recycling is a real effect [14]. The small number of neutrals that go behind the limiter will also reduce the pumping speed.

B. <u>Limiter Design</u>

The limiter occupies the same location as the divertor, and it takes up somewhat less space than the divertor. The particle and heat flux requirements for the limiter (Table II) are similar to those for the divertor. The limiter must have adequate heat removal capacity and should have a lifetime exceeding ~ 1 y, just like the divertor.

There are some important engineering differences between the limiter and divertor, however. The plasma temperature at the collector plate is 150 eV for the limiter compared with 25 eV for the divertor. At 150 eV, low-Z materials must be used to avoid runaway self-sputtering. The sputtering erosion, particularly at the leading edge, is predicted to be high. The erosion lifetime will be limited by the maximum allowable thickness of the plasma surface material. This thickness is usually limited by the thermal stresses and fatigue strain that can be tolerated in the structure. Beryllium is the favored low-Z material. Its erosion lifetime is approximately 2 y on the top surface but is only ~ 0.15 y at the leading edge. A possible solution to the short lifetime is to replace the beryllium with tungsten at the leading edge. The plasma temperature is < 50 eV at this position, which is acceptable for the use of sputtering high-Z materials. The use of another material creates additional interface problems, however. The plasma edge is predicted to have short power and e-folding distances, which would result in high peak heat loads on a flat limiter. In order to reduce peaking, the limiter surface is shaped to spread the power uniformly over the surface. Unfortunately, a shaped limiter would be susceptible to nonuniform heating if the plasma shifts position. The leading edges appear to have lower heat loading limits than the top surface, and therefore the edges have been placed at positions where the peak load is 1 MW/m^2. At this power level, a double-edged limiter is needed to maximize the pumping capability of the system.

Aside from the sputtering erosion, the lifetime concerns are similar to those of the divertor. Again, since the data base is sparse, it is not possible to characterize the long-term response of the limiter adequately.

5. SUMMARY

Based on these studies, the INTOR impurity control group has selected the poloidal divertor as the reference impurity control option. In view of its reduced complexity, the pumped limiter has been retained as the back-up option. The major advantage of the poloidal divertor is the demonstrated ability to produce a cool, dense plasma in contact with the collector plate, thus greatly reducing the impurity production at the plate (Table III). There are some theoretical and experimental indications that impurities generated at the divertor plate are returned to the plate because of, among other effects, the large proton flux on the plate. Other items of comparison are listed in Table III. Several other systems, including a bundle or hybrid divertor, and an ergodic magnetic limiter were considered, but were not felt to have the level of credibility of the divertor or pumped limiter.

Future work on the physics of the impurity control system for INTOR will concentrate on theoretical studies and continued assessment and encouragement of divertor and limiter experiments. The theoretical studies of divertor systems will center on two-dimensional calculations of plasma and impurity transport in realistic geometries and a better study of the pumping efficiencies. The theoretical studies of limiters will concentrate on self-consistent, two-dimensional studies of the plasma conditions around limiters. Two key questions are the extent to which localized recycling can lower the temperature of the plasma in contact with the limiter front face, and to what extent localized recycling in the scrape-off plasma can increase the pumping efficiency. On the experimental side, experiments on "expanded boundary" divertors and on axisymmetric pump limiter systems with high power auxiliary heating are to be encouraged.

Future emphasis for engineering will be on divertor design refinements.

The work will include examination of the benefits of a shortened divertor channel on operation and reactor design, analysis of the effects of sputtered first wall material on divertor operation, and additional predictions of the thermal-hydraulic and stress response of the divertor collector plates. In addition, the INTOR participants are exploring alternate concepts for impurity control and are continuing to examine limiter design trade-offs.

In the engineering area, there is a need to perform R&D that provides needed materials data for design. In particular, fabrication methods for impurity control components should be developed, and the effects of radiation damage on impurity control materials should be determined. The effects of sputtering erosion under prototypical plasma edge conditions also needs to be determined. Important areas to be addressed are redeposition of sputtered particles and the edge temperature limits for high-Z materials. Finally, data on the effects of disruptions on erosion are needed.

ACKNOWLEDGMENT

The INTOR group gratefully acknowledges the work of the many scientists of the EC, Japan, the U.S., and the USSR which contributed to the design considerations for the INTOR impurity control system.

This work is partially supported by the U.S. Department of Energy Contract No. DE-AC02-76-CHO-3073.

REFERENCES

[1] BOL, K., et al., Plasma Phys. Controlled Nuclear Fusion, 1978 1, (1979), 11, IAEA, Vienna.

[2] HOBBS, G. and WESSON, J., Plasma Phys. 9, 85 (1967).

[3] PETRAVIC, M., et al., Phys. Rev. Lett. 48, 326 (1982).

[4] BRAAMS, B., et al., J. Nucl Mater. 121, 75 (1984).

[5] SAITO, S., SUGIHARA, M., and FUJISAWA, N., J. Nucl. Mater. 121, 199 (1984).

[6] IGITHHANOV, Y., et al., in Proc. of Eleventh European Conference on *Controlled Fusion and Plasma Physics*, Aachen (FRG) 2, 377 (1983).

[7] KEILHACKER, M. and Asdex Team, Proc. of Ninth International Conference on Plasma Physics and Controlled Nuclear Fusion, Baltimore, USA, (1982), IAEA-CN-41/R-2.

[8] SHIMADA, M., et al., Phys. Rev. Lett. 47, 796 (1981).

[9] OWENS, D. K., et al, J. Nucl. Mater. 121, 19 (1984).

[10] MATTAS, R.F., SMITH, D., and ADBOU, M., J. Nucl. Mater. 122&123, 66 (1984).

[11] OHYABU, N., Nucl. Fusion 21, 519 (1981).

[12] JARDIN, S., et al., Tenth International Conference on Plasma Physics and Controlled Nuclear Fusion, London, UK, (1984) IAEA-CN-44/A-IV-3.

[13] MIODUSZEWSKI, P., et al., J. Nucl. Mater. 121, 285 (1984).

[14] BUDNY, R., et al., J. Nucl. Mater. 121, 294 (1984).

[15] COHEN, S., et al., Sixth PSI Conference, Nagoya (to appear J. Nucl. Mater. 1985).

[16] PONTAU, A., et al., J. Nucl. Mater. 121, 304 (1984).

[17] MIODUZEWSKI, P., J. Nucl. Mater. 111&112, 253 (1982).

[18] EVANS, K., et al., Sixth PSI Conference, Nagoya (to appear J. Nucl.

Mater. 1985).

[19] HEIFETZ, D., et al., J. Nucl. Mater. 111&112, 2981 (1982).

TABLE I. Candidate Impurity Control Materials

Plasma Side Materials		Heat Sink Materials
Low-Z:	C, Be, B, TiC, SiC, B_4C, BeO	Copper Alloys
Medium-Z:	Stainless Steel, Vanadium	Vanadium Alloys
High-Z:	W, Ta, Nb	Niobium Alloys

TABLE II. Limiter and Divertor Operating Conditions

	Divertor	Limiter
Total power to collector plates	70 MW	84 MW
Particle	53 MW	80 MW
Radiation	17 MW	4 MW
Presheath ion energy	25 eV	150 eV
Sheath potential	60-80 eV	300-800 eV
Peak power	4.7-7 MW/m^2	2.4 MW/m^2

TABLE III. Comparison of Divertor and Pumped Limiter

ITEM	DIVERTOR	PUMPED LIMITER
1. Plasma parameters in front of collector plate	high probability of high density ($>10^{14}cm^{-3}$) and low temperature ($\lesssim$ 30 eV)	high probability of medium density (5×10^{12}-$5\times10^{13}cm^{-3}$) and medium temperature (100-200 eV)
2. Impurity control	low sputtering rates and possibility of trapping impurities in divertor chamber	large sputtering rates and easier access to main plasma for impurities
3. Collector plate materials	low- or high-Z	low-Z
4. First wall erosion	concentrated near divertor (question about impurity shielding performance of scrape-off plasma)	concentrated near limiter
5. Heat flux limits	2-5 MW/m^2	2-5 MW/m^2 for plate and ~ 1 MW/m^2 at plate tip
6. Component lifetime	very long (for erosion), redeposition of first wall material potential limitation	short (on the order of 1 year with high degree of redeposition, much less with less lower redeposition)
7. Pumping requirement	$(1-10) \times 10^4$ ℓ/sec	$(1-5) \times 10^5$ ℓ/sec
8. Effects on energy confinement	H-mode	L-mode
9. Torus size	increased torus size due to null points and divertor chamber	lesser increase due to need for pumping chamber
10. Poloidal coil power requirements	increased compared to limiter	
11. Relative cost	~ 7% more expensive than limiter	

FIGURE CAPTIONS

FIG. 1. Schematic outline of (a) poloidal divertor and (b) pumped limiter system for INTOR.

FIG. 2. Calculations of the electron density in the INTOR poloidal divertor.

FIG. 3. Evolution of poloidal divertor designs from early reactor concepts to the INTOR divertor.

#84P0265

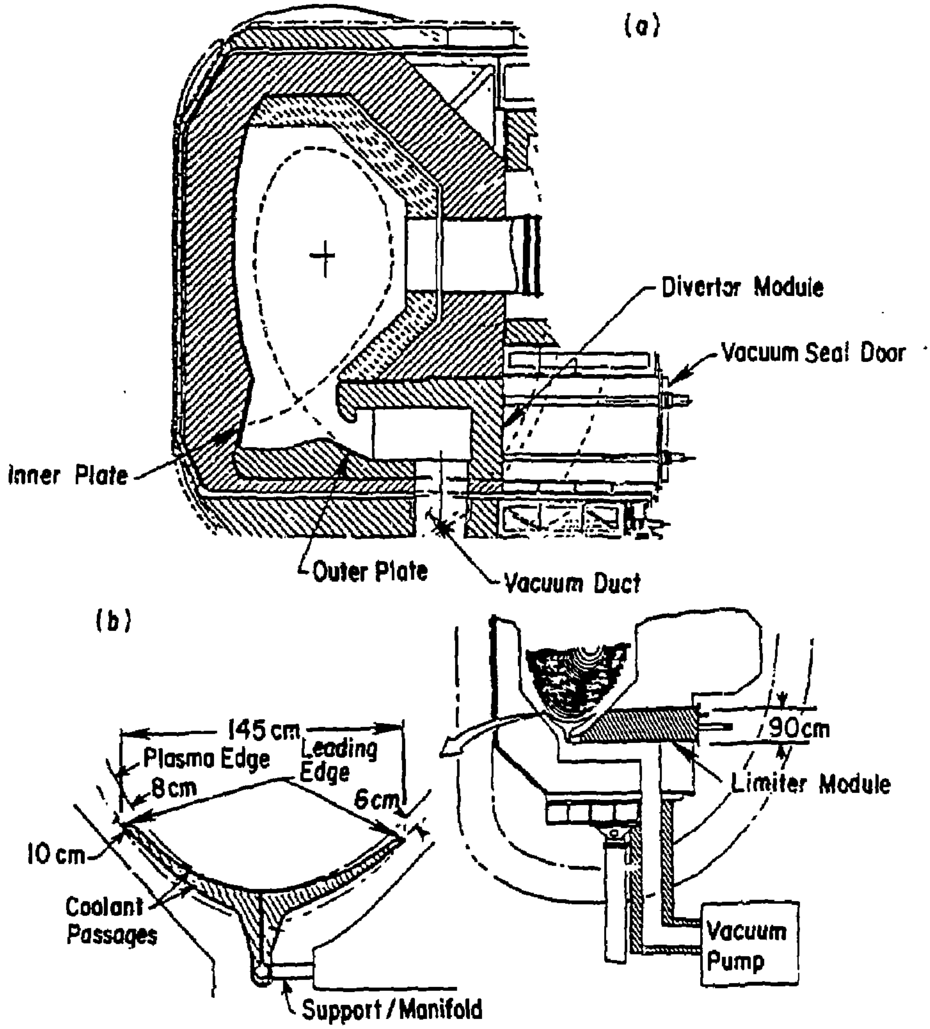


FIG. 1

#84P0348

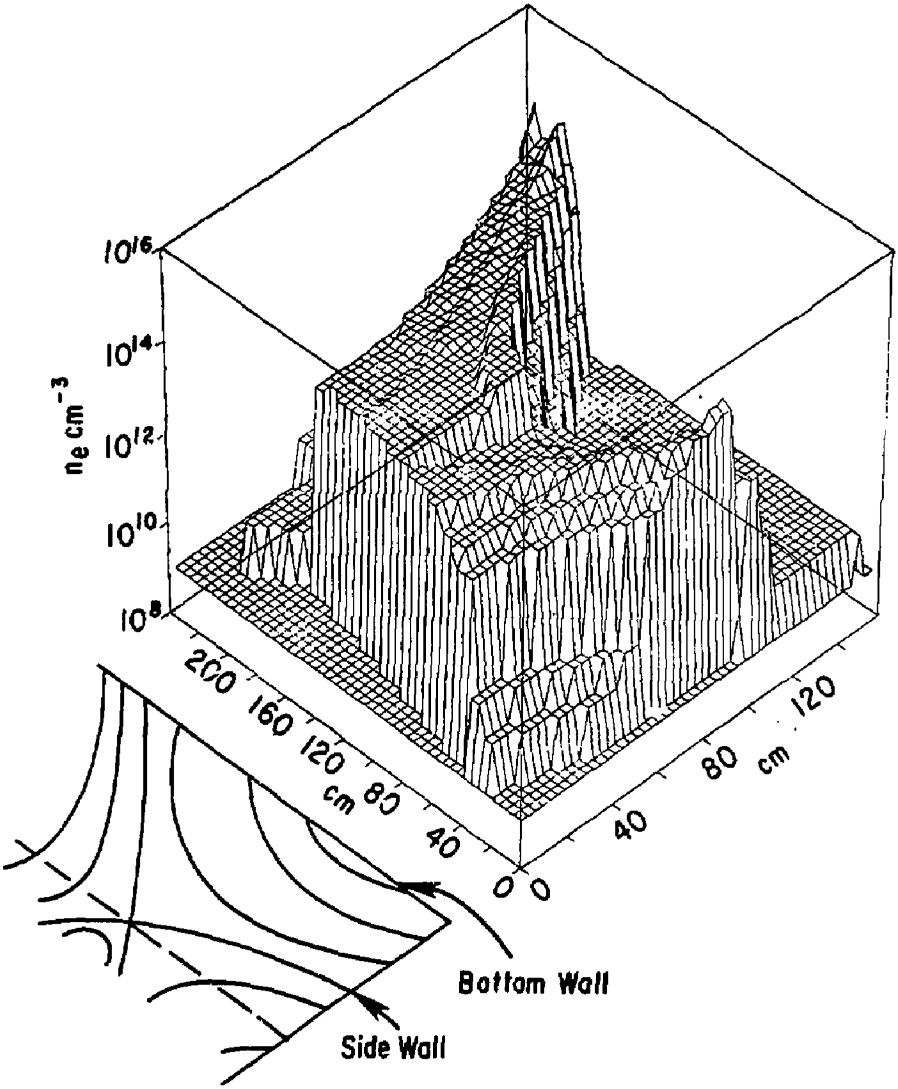
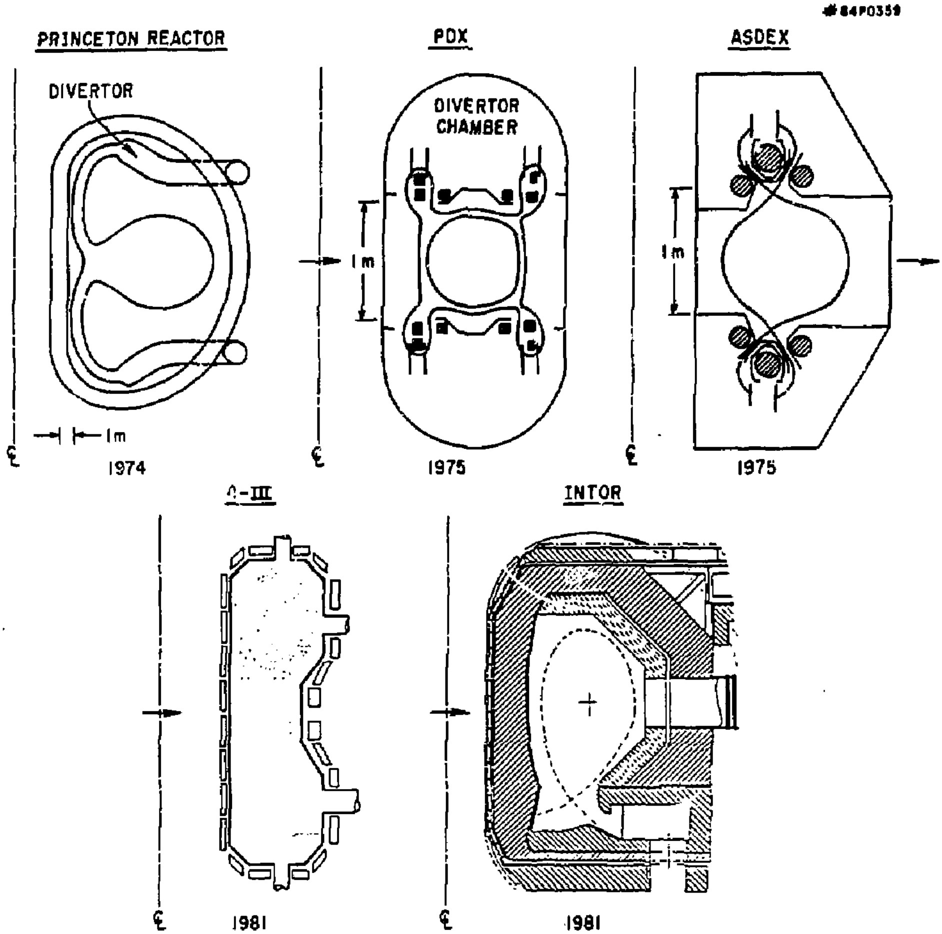


FIG. 2



#84P0359
PRINCETON REACTOR
DIVERTOR
1m
1974
PDX
DIVERTOR
CHAMBER
1m
1975
ASDEX
1m
1975
Ω-III
1981
INTOR
1981

FIG. 3

EXTERNAL DISTRIBUTION IN ADDITION TO UC-20

Plasma Res Lab, Austra Nat'l Univ, AUSTRALIA
Dr. Frank J. Paoloni, Univ of Wollongong, AUSTRALIA
Prof. I.R. Jones, Flinders Univ., AUSTRALIA
Prof. M.H. Brennan, Univ Sydney, AUSTRALIA
Prof. F. Cap, Inst Theo Phys, AUSTRIA
Prof. Frank Verheest, Inst theoretische, BELGIUM
Dr. D. Palumbo, Dg XII Fusion Prog, BELGIUM
Ecole Royale Militaire, Lab de Phys Plasmas, BELGIUM
Dr. P.H. Sakanaka, Univ Estadual, BRAZIL
Dr. C.R. James, Univ of Alberta, CANADA
Prof. J. Teichmann, Univ of Montreal, CANADA
Dr. H.M. Skarsgard, Univ of Saskatchewan, CANADA
Prof. S.R. Sreenivasan, University of Calgary, CANADA
Prof. Tudor W. Johnston, INRS-Energie, CANADA
Dr. Hannes Barnard, Univ British Columbia, CANADA
Dr. M.P. Bachynski, MPB Technologies, Inc., CANADA
Chalk River, Nucl Lab, CANADA
Zhengwu Li, SW Inst Physics, CHINA
Library, Tsing Hua University, CHINA
Librarian, Institute of Physics, CHINA
Inst Plasma Phys, Academia Sinica, CHINA
Dr. Peter Lukac, Komenskeho Univ, CZECHOSLOVAKIA
The Librarian, Culham Laboratory, ENGLAND
Prof. Schatzman, Observatoire de Nice, FRANCE
J. Radet, CEN-BP6, FRANCE
AM Dupas Library, AM Dupas Library, FRANCE
Dr. Tom Mual, Academy Bibliographic, HONG KONG
Preprint Library, Cent Res Inst Phys, HUNGARY
Dr. S.K. Trehan, Panjab University, INDIA
Dr. Indra Mohan Lal Das, Banaras Hindu Univ, INDIA
Dr. L.K. Chavda, South Gujarat Univ, INDIA
Dr. R.K. Chhajlani, Vikram Univ. INDIA
Dr. B. Dasgupta, Saha Inst, INDIA
Dr. P. Kaw, Physical Research Lab, INDIA
Dr. Phillip Rosenau, Israel Inst Tech, ISRAEL
Prof. S. Cuperman, Tel Aviv University, ISRAEL
Prof. G. Rostagni, Univ Di Padova, ITALY
Librarian, Int'l Ctr Theo Phys, ITALY
Miss Clelia De Palo, Assoc EURATOM-ENEA, ITALY
Biblioteca, del CNR EURATOM, ITALY
Dr. H. Yamato, Toshiba Res & Dev, JAPAN
Direc. Dept. Lg. Tokamak Dev. JAERI, JAPAN
Prof. Nobuyuki Inoue, University of Tokyo, JAPAN
Research Info Center, Nagoya University, JAPAN
Prof. Kyoji Nishikawa, Univ of Hiroshima, JAPAN
Prof. Sigeru Mori, JAERI, JAPAN
Library, Kyoto University, JAPAN
Prof. Ichiro Kawakami, Nihon Univ, JAPAN
Prof. Satoshi Itoh, Kyushu University, JAPAN
Dr. D.I. Choi, Adv. Inst Sci & Tech, KOREA
Tech Info Division, KAERI, KOREA
Bibliotheek, Fom-Inst Voor Plasma, NETHERLANDS
Prof. B.S. Liley, University of Waikato, NEW ZEALAND
Prof. J.A.C. Cabral, Inst Superior Tecn, PORTUGAL
Dr. Octavian Petrus, ALI CUZA University, ROMANIA
Prof. M.A. Hellberg, University of Natal, SO AFRICA
Dr. Johan de Villiers, Plasma Physics, Nucor, SO AFRICA
Fusion Div. Library, JEN, SPAIN
Prof. Hans Wilhelmson, Chalmers Univ Tech, SWEDEN
Dr. Lennart Stenflo, University of UMEA, SWEDEN
Library, Royal Inst Tech, SWEDEN
Centre de Recherchesen, Ecole Polytech Fed, SWITZERLAND
Dr. V.T. Tolok, Kharkov Phys Tech Ins, USSR
Dr. D.D. Ryutov, Siberian Acad Sci, USSR
Dr. G.A. Eliseev, Kurchatov Institute, USSR
Dr. V.A. Glukhikh, Inst Electro-Physical, USSR
Institute Gen. Physics, USSR
Prof. T.J.M. Boyd, Univ College N Wales, WALES
Dr. K. Schindler, Ruhr Universitat, W. GERMANY
Nuclear Res Estab, Julich Ltd, W. GERMANY
Librarian, Max-Planck Institut, W. GERMANY
Bibliothek, Inst Plasmaforschung, W. GERMANY
Prof. R.K. Janev, Inst Phys. YUGOSLAVIA