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CONCEPTUAL DESIGN OF A DIVERTOR FOR A TOKAMAK EXPERIMENTAL POWER REACTOR

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A. M. Stephanovsky,*** and F. H. Tenney
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Princeton, New Jersey 08540

Summary

A design for a Double-Null Poloidal Divertor for a Tokamak EPR is presented which allows remote assembly of the torus and utilizes a standard neutral pumping and heat removal system.

I. Introduction

The present Experimental Power Reactor EPR designs by the Oak Ridge,[1] Argonne[2] and General Atomic[3] groups consider reactors which approach the condition of net electrical power and as such require relatively clean plasmas for burn times ~ 50-100 sec. Impurities have a number of deleterious effects on tokamak fusion reactors such as:

1. Enhanced radiation losses due to line radiation from partially-stripped impurities, recombination and bremsstrahlung radiation from fully stripped impurities.

2. Reduction of reacting fuel ion density when the total plasma pressure is limited.

3. Energetic neutral beams (~ 200 keV) used for plasma heating are ionized by ion-impact ionization which is multiplied by $\bar{Z}$. This increased ionization forces the beam to be deposited near the surface for Z ≈ 3, or requires that the beam energy be increased substantially for adequate beam penetration.

4. The current profile is also affected by the presence of impurities. Low Z impurities at the outer surface can cause the plasma column to shrink and have a disruptive instability while high-Z impurities near the plasma center can cause the temperature profiles to be hollow.

Power producing reactors operating at thermonuclear ignition are severely affected by impurities, particularly high-Z impurities. A number of studies[4] have indicated that high-Z impurity concentrations (e.g. Mo) at n_z/n_e ~ 0.1 to 0.5% will significantly increase the $n\tau$ required for ignition. If ignition is not possible due to the presence of impurities, energetic neutral beams can be used to produce reactions and plasma heating. For low thermonuclear gain devices, Q ~ 1, such as TFTR, the major effect of the impurities is to limit neutral beam penetration. However, a beam driven reactor would require Q ~ 5-10 to produce net electrical power[5] and in this case even low-Z impurities producing $\bar{Z} = \Sigma\, n_j Z_j^2/n_e \approx 2\text{-}3$ can significantly increase the plasma current[6] and hence the cost of the reactor required to achieve a given Q.

In addition to the impurity problem at the plasma-wall boundary, the pumping of cold neutral gas in an ... beam driven reactor is a significant problem, ~ 10^{11} cm^{-3} to recycle the escaping plasma. Als... neutral density in the neutral beam injection du... must be ≤ 10^{11} cm^{-3} to avoid impact ionization o... neutral beam. Therefore the required pumping sp... the vacuum system is S ~ 10^8 liters/sec. The pu... orifice area required is

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where K takes into account the reduction in conductance due to the length of the pumping lines between the vacuum vessel and pumps. Consider a pumping duct extending radially outward all around the torus with a height equal to the plasma diameter, in this case the orifice area is $A = 2\pi R 2a \approx 5 \times 10^5$ cm^2 or roughly an order of magnitude less than the required pumping area, and the blanket area has been severely compromised. The solution to the problem is to pump the escaping particles at a higher velocity e.g., 100 eV instead of room temperature.

A viable solution to both the impurity and neutral pumping problem appears to be a poloidal magnetic divertor. In this report we describe a conceptual design of a divertor for an EPR device similar to the ORNL and ANL EPR devices. The divertor design described here is the result of work carried out during a two week exchange visit by Soviet scientists to the Princeton Plasma Physics Laboratory and as such represents only the initial thoughts on an EPR divertor and it is hoped that further study and optimization will be carried out in the future.

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4. a configuration with superconducting divertor coils that are shielded from the neutrons and with the entire assembly capable of remote non-destructive disassembly.

The proposed configuration is a Double-Null Poloidal Divertor which provides a reacting plasma that is essentially identical to the ORNL EPR design and is similar to the ANL EPR and Russian T-20 design, Table I.

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In addition to the impurity problem at the plasma-wall boundary, the pumping of cold neutral gas in an intensely beam driven reactor is a significant problem, particularly in neutron producers with $n\tau \approx 10^{13}$ sec-cm^{-3}. For these devices, the particle throughput due to beam injection becomes ~ 10^{22} particles/sec for a reactor with $a \approx 1_m$, R_o = 3m and $\bar{n}_e = 7\times10^{13}$ cm^{-3}. This particle input is ~ 25% of the natural recycling flux and would cause the plasma density to double in ~ 4 particle confinement times and thereby causing the plasma pressure to increase beyond the MHD equilibrium limit. Near the surface of the confined plasma, the neutral density for 1 eV D^o must be in the range of the vacuum system is S ~ 10^8 liters/sec. The orifice area required is

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Also we have chosen to consider a 50 kG toroidal field since this allows a 7 MA plasma current for q(o)= 1-2 and q(a) ≅ 4 in our design. This large plasma current along with impurity reduction by the divertor increases the prospects for operating the reactor at ignition or at high gain near ignition and thereby achieving substantial thermonuclear power perhaps even net power output.

The Tokamak Reactor Divertor design is shown in Fig. 1. The main features of the design are:

1. Double-Null poloidal divertor inside the toroidal field coils with a slightly "D" shaped confined plasma with a 7MA plasma current,

2. extended divertor exhaust channels and pumping chambers located at the outer surface of the torus allowing a standard pumping and heat removal system, and

3. a divertor coil system with zero net ampere turns, allowing the poloidal coils to be placed inside the toroidal coil without topologically linking the toroidal field coil. This feature allows remote disassembly of the entire coil system.

B. Divertor Physics Requirements. While many of the detailed plasma physics questions concerning the divertor scrape-off region are unanswered at the present time, we can nonetheless make reasonable estimates of the two most important parameters of the divertor scrape-off;

1. the width of the divertor exhaust channels and
2. the length of the exhaust channels.

A shielding divertor would have a moderately wide (~ 5 cm) dense (5×10^{12} cm^{-3}) plasma which ionizes incoming impurity atoms and sweeps the resultant impurity ions into the divertor burial chambers. A shielding divertor allows neutral recycling and refueling at the surface of the confined plasma with negligible impurity influx and therefore permits the long burn times required for EPR. The present design allows a 20 cm scrape-off channel which should easily provide for a shielding divertor with essentially no plasma wall contact.

The length of the exhaust channels was designed to be the maximum allowed inside the toroidal field coil bore so as to minimize back flow of neutral gas by enhancing the plasma pumping effect and increasing the pumping area. Also the neutralizer plates are at the largest possible major radius with a large surface area thereby reducing the energy removal problem.

C. Poloidal Field Design.

1. General Considerations. The proposed design combines favorable features of both the long exhaust channel Single-Null Divertor in the Princeton Reference Design[7] and the short channel Double-Null Divertor in the Wisconsin UMAK-1 design,[8] that is the design has long exhaust channels with a stable plasma column placed in the high toroidal field region near the inside bore of the toroidal field coil. Since the divertor poloidal field design determines the requirements for all other systems pumping, mechanical structure and toroidal field coil size, considerable effort during the exchange period was directed toward a study of the poloidal field configuration. While a number of parameters remain to be optimized, this design should enable some engineering design estimates to be made.

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3. Plasma Current Startup Considerations. Impurity control is also essential during the plasma startup phase. This can be accomplished magnetically by:

a. initiating the plasma discharge at a multipole null in the poloidal field located away from metallic limiters,

b. creating an expanding magnetic limiter that reduces the plasma-wall interaction due to skin-effect related MHD instabilities, and

c. providing proper divertor action during plasma startup.

In this reference design three possible startup sequences have been considered.

a. Fixed flux plot. The DF and EF currents are increased proportional to the plasma current. This maintains divertor action and separatrix positioning during plasma startup but does not provide an expanding magnetic limiter to eliminate the skin effect. In this scheme, 43 volt-sec is generated by the DF and EF fields.

b. Fixed DF-hexapole null. This mode is very similar to the startup sequence in PDX. The DF currents are fixed in time; at t = 0, the EF current is + 10.4 MA which generates a hexapole null at R = 6.9 m. As the plasma current increases, the EF coil current is driven toward - 2.8 MA providing the proper equilibrium transverse field. This mode provides 76 volt-sec capability. Studies of flux plots during start-up for this simple programming show poor divertor throat tracking, clearly we have not yet optimized the present configuration for startup.

c. Fully programmed poloidal coils. Future studies will involve programming all coil systems OH, DF and EF. Since in a reactor, the OH coil current is very nearly constant during the burn phase, we will investigate coupling the OH transverse field into the poloidal field design.

D. Engineering Description.

1. Poloidal coil design. An important feature of this divertor EPR reference design is the demountable internal poloidal coil. The internal coils are in the form of 180° loops which carry positive current half-way around the torus, crossover and return

toroidal coil without topologically linking the toroidal field coil. This feature allows remote disassembly of the entire coil system.

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2. Results of Poloidal Magnetic Field Studies. The configuration shown in Fig. 1 satisfies the general criteria stated above. The net vertical field for toroidal plasma equilibrium is in agreement with the magnitude required by the Shafranov formula and has radial positional stability. However, the vertical positional stability is probably marginal and further optimization is required. Each of the poloidal coils has a specific function. The positive coil (R = 5.4, Z = 5.1) produces the primary divertor stagnation point at R = 5.4, Z = 3.1, while its return current at R = 5.4, Z = 7.2 bends the exhaust channel down from the toroidal coil and diverts it toward the outside of

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a. Fixed flux plot. The DF and EF currents are increased proportional to the plasma current. This maintains divertor action and separatrix positioning during plasma startup but does not provide an expanding magnetic limiter to eliminate the skin effect. In this scheme, 43 volt-sec is generated by the DF and EF fields.

b. Fixed DF-hexapole null. This mode is very similar to the startup sequence in PDX. The DF currents are fixed in time; at t = 0, the EF current is + 10.4 MA which generates a hexapole null at R = 6.9 m. As the plasma current increases, the EF coil current is driven toward - 2.8 MA providing the proper equilibrium transverse field. This mode provides 76 volt-sec capability. Studies of flux plots during startup for this simple programming show poor divertor throat tracking, clearly we have not yet optimized the present configuration for startup.

c. Fully programmed poloidal coils. Future studies will involve programming all coil systems OH, DF and EF. Since in a reactor, the OH coil current is very nearly constant during the burn phase, we will investigate coupling the OH transverse field into the poloidal field design.

D. Engineering Description.

1. Poloidal coil design. An important feature of this divertor EPR reference design is the demountable internal poloidal coil. The internal coils are in the form of 180° loops which carry positive current half-way around the torus, crossover and return as negative current. This feature allows remote disassembly and zero coupling to the OH field and small coupling to the plasma field. The cross-over design requires some care since they are subject to large forces, interfere with plasma motion into the divertor and generate magnetic field errors. These problems appear to be soluble and are discussed in more detail in the appropriate sections. With the present design we have only 3 types of DF, EF coils. The ohmic heating coils are similar to those of the ORNL and ANL design and are located outside the toroidal field coil.

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2. Poloidal coil conductor. We consider the DF coils to be fabricated from stabilized NbTi cable similar to that described by ORNL or ANL. However, space considerations at the DF-1 coil lead us to consider a high current density coil in which the hoop stress is taken by an external stainless steel ring. For DF-1, we assume a core current density of 3 kA/cm^2 with a 10 cm dewar structure to arrive at the sizes illustrated in Fig. 1. Clearly, this is an area where careful expert design is required. The DF-2 and EF-1 coils have less stringent requirements and the coils shown have 2 kA/cm^2.

3. Poloidal coil support structure. A poloidal structure problem is the support for the vertical forces on DF-1, the net force on the lower (positive) half-turn of DF-1 is 5 x 10^6 lbs. This force can be supported by a cantilevered steel plate (shown in cross-section in Fig. 1) with a modest deflection of ≈ 1 cm for a 50 cm thick plate clamped near the vacuum wall. As mentioned previously, the hoop force of the superconducting coil will be taken up on a stainless steel backing ring.

4. Toroidal field coil. The torque on the toroidal field (TF) coils due to the divertor field should also be investigated as well as the effect of the time changing poloidal fields on the superconducting toroidal field coils.

5. Vacuum system. The pumping system for the proposed reactor must handle a particle throughput of ≈ 10^{22} particles/sec and maintain a neutral pressure of ≈ 10^{-5} torr at the surface of the plasma. Since the plasma in the divertor channels behaves somewhat as a diffusion pump, the neutral pressure in the divertor chambers is expected to be ≈ 5x10^{-5} torr. We propose to supply the required pumping speed of ≈ 10^7 ℓ/sec for D^o, T^o with internal cryopumps having an area of 3-5x10^{2}m^2. Helium will be pumped with external compression pumps having a speed of 5x10^5 ℓ/sec.

One of the important features of the present design is that no special technology, such as flowing liquid lithium, is necessary to remove the thermal energy of the plasma flowing into the divertor. The power density on the neutralizer plate is ≈300 w/cm^2 for flat plates and can be reduced to ~ 150 w/cm^2 with a corrugated plate. Low power densities such as this can be handled by "standard" cooling techniques.

Conclusion

The proposed divertor design, appears to be a viable solution to the need for impurity control and neutral pumping in tokamak reactors and as such can be used for scoping a variety of engineering problems peculiar to divertors. However, the present design does not represent an optimized system with regard to device cost. The proposed toroidal field coil could probably be reduced in size from an inside height of 7.8 m to 6.5 m and from an outer midplane radius of 12.8 m to 11.3 m without seriously deteriorating the divertor performance. In this latter case, the toroidal field coil for the divertor EPR is only slightly larger than the toroidal coils for the ANL and ORNL

References

1. M. Roberts, Oak Ridge Tokamak EPR Referenc
paper C 2-3, these proceedings.

2. W. M. Stacey, Jr., et al, Argonne Tokamak
mental Power Reactor Studies, paper C 2-1,
proceedings.

3. C. C. Baker, et al, General Atomic Experim
Power Reactor Design, paper C 2-7, these p

4. D. M. Meade, Nuc. Fusion 14, 289 (1974).

5. M. Nozawa and D. Steiner, Oak Ridge Nation
tory Rep. ORNL TM-4421 (1974).

6. R. W. Conn and J. Kesner, Nuc. Fusion 15,

7. R. G. Mills, Princeton Plasma Physics Labo
Rep. MATT-1050 (1974).

8. G. L. Kukinski, et al, University of Wisco
Nuclear Engineering Rep. UW FDM-68 (1973).

TABLE I

EPR Parameters

	T-20	ORNL	ANL
a(m)	2	2.25	2.1
R(m)	5	6.75	6.25
B_T(kG)	35	48	34
I(MA)	6	7.2	4.8
q(o)			
q(a)	2.3	2.5	2.5
β_θ	1	2	2.2
τ_p(s)	2	~2	2
τ_b(s)	20	100	20-50
P_{LOSS}(MW)	50	~100	130

3. Poloidal coil support structure. A poloidal structure problem is the support for the vertical forces on DF-1, the net force on the lower (positive) half-turn of DF-1 is 5 x 10^6 lbs. This force can be supported by a cantilevered steel plate (shown in cross-section in Fig. 1) with a modest deflection of ≈ 1 cm for a 50 cm thick plate clamped near the vacuum wall. As mentioned previously, the hoop force of the superconducting coil will be taken up on a stainless steel backing ring.

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***I. V. Kurchatov Institute of Atomic Energy, Moscow, USSR.

5. M. Nozawa and D. Steiner, Oak Ridge National Labo tory Rep. ORNL TM-4421 (1974).

6. R. W. Conn and J. Kesner, Nuc. Fusion 15, 775 (19

7. R. G. Mills, Princeton Plasma Physics Laboratory Rep. MATT-1050 (1974).

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B_T(kG)	35	48	34	50
I(MA)	6	7.2	4.8	7.0
q(o)				1-2
q(a)	2.3	2.5	2.5	3-4
β_θ	1	2	2.2	2
τ_p(s)	2	~2	2	2
τ_b(s)	20	100	20-50	
P_{LOSS}(MW)	50	~100	130	~80

2

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References

1. M. Roberts, Oak Ridge Tokamak EPR Reference Design, paper C 2-3, these proceedings.
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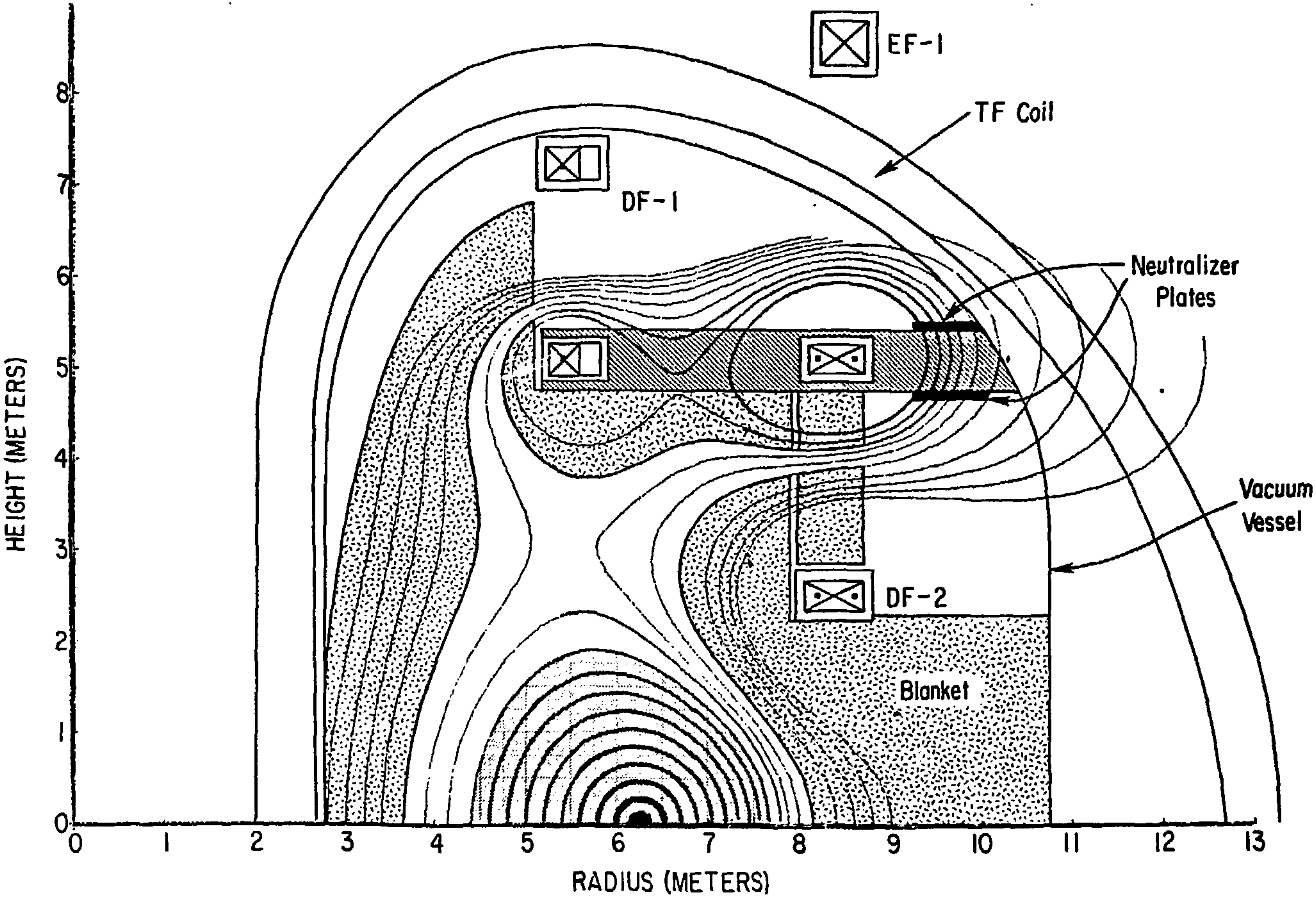
τ_b(s)	20	100	20-50	
P_{LOSS}(MW)	50	~100	130	~80

3

Figure Caption

Fig. 1. Cross-sectional view of the tokamak EPR divertor design showing the toroidal field (TF) coil, divertor field (DF) coils and equilibrium field (EF) coil.


EF-1
TF Coil
DF-1
Neutralizer Plates
Vacuum Vessel
DF-2
Blanket
HEIGHT (METERS)
RADIUS (METERS)
0
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