Table I. Plasma parameters determined under the conditions $q(0)\geq 1$ and $q_\psi(a)\geq 3$. Column data depicted with italic letters do not satisfy these conditions. The maximum toroidal field strength B_{max} at the inboard side of the torus is fixed to be B_{max}=12T, keeping the blanket/shielding thickness d of 1.4 m. Given parameters as input data are indicated with * marks.

	$A = 3$		$A = 5$	$A = 7$	
*major radius : R_p(m)	**8.7**	*6.5*	**9.5**	**11.5**	*11.0*
*central safety factor :$q(0)$	**1.0**	*0.5*	**1.0**	**1.8**	*1.0*
surface safety factor :$q_\psi(a)$	**5.35**	*3.52*	**3.44**	**3.0**	*2.75*
*elongation ratio :κ	**2.0**	*2.0*	**1.8**	**1.7**	*1.7*
plasma volume :V(m^3)	**2,889**	*1,205*	**1,219**	**1,042**	*912*
total current :I_{tot}(MA)	**19.8**	*20.0*	**12.3**	**8.9**	*9.2*
bootstrap current : I_{bs}/I_{tot}	**0.42**	*0.27*	**0.49**	**0.72**	*0.53*

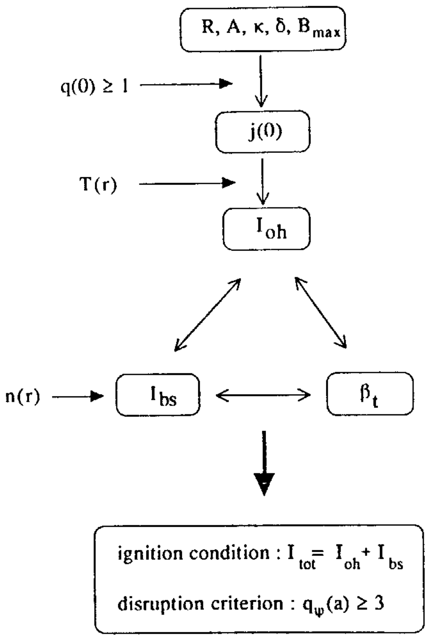

Fig. 1 Flow chart to determine plasma parameters for sawtooth -free plasma.

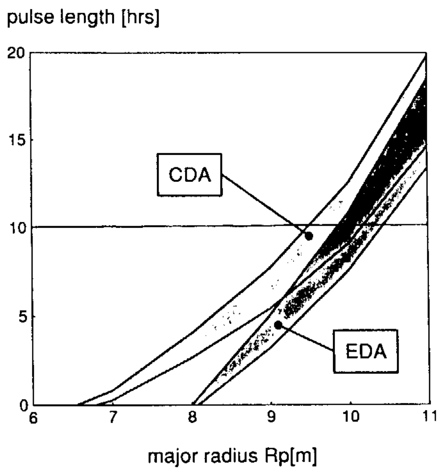

Fig. 2 Pulse length as a function of the major radius for two cases; ITER CDA design(κ=2, f_α=10%) and EDA design (κ=1.6, f_α=24%). The maximum field strength of center solenoid coil is assumed to be 10~13T, shown as lower and upper limits of pulse length.

The Mission and Physics Design of TPX*

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The Mission of TPX

The mission of the Tokamak Physics Experiment (TPX) is to develop the scientific basis for an economic, compact, and steady-state tokamak fusion reactor. This will require the development and demonstration of stable, steady-state tokamak operating regimes, with high β, high confinement, and high bootstrap fraction.

The size and field required for an ignited tokamak reactor scale roughly as:

$$\frac{a(1+\kappa^2)B}{q^*} \propto \frac{\beta}{\beta^* H}$$

where q* is the cylindrical q at the plasma edge, β* is the RMS volume-average β, and H is an enhancement factor over standard L-mode confinement scaling. Thus higher H* (≡ Hβ*/β) at moderate q* permits a more compact and/or lower-field reactor. However at a given economically and technically acceptable neutron wall loading (e.g., ~3MW/m^2), the size and field are also constrained by:

$$\frac{(1+\kappa^2)\kappa^{1/4}a^{1/2}B^2}{q^*(R/a)} \propto \frac{1}{\beta_N^*}$$

where $\beta_N^* \equiv \beta^*/(I/aB)$. Thus size and/or field can only be decreased at fixed wall loading if β_N^* is increased. As the size of a reactor is reduced at fixed neutron wall-loading, the parallel heat flux to the divertor, as measured by P/R, decreases. Furthermore, if the field or size is reduced, the impact of disruptions, as measured by $(W_{kin}+W_{mag})/(Ra\kappa^{1/2})$ or $(W_{kin}+W_{mag})/R$, is also strongly ameliorated.

The requirement for current-drive power in steady state, at fixed T, scales as:

$$\frac{P_{cd}}{P_{fus}} \propto \frac{(1-f_{bs}^*)\, nI_pR/\eta_{cd}}{(\beta_N^* I_p/aB)^2 B^4 Ra^2\kappa} \propto \frac{(1-f_{bs}^*)}{\eta_{cd}} \frac{\beta_N}{(\beta_N^*)^2 a\kappa B}$$

where $f_{bs}^* \propto q^*(R/a)^{1/2}\beta_N$ corresponds to an effective bootstrap current fraction. Thus high values of H* and β_N^* can allow a steady-state fusion reactor with reduced recirculating power, smaller unit size, and lower field, current, stored energy, and heat flux, implying lower cost and higher reliability. Our systems-code studies indicate that if β_N can be increased from 2.5 to 6 and H from 2 to 3 (at α_n = 0.5, α_T = 1.0, β*/β = 1.25), then the Cost of Electricity (COE) from a demonstration tokamak reactor, at fixed unit size, drops by about a factor of two. The minimum unit size of a reactor at a given COE drops by about a factor of four (e.g., from 2000 MWe to 500 MWe). The combination of reduced COE, reduced unit size, and fully continuous operation would make the tokamak much more attractive in the world energy market. For these reasons, the experimental goal of TPX is to use current-profile control, particle recycling control, and strong plasma shaping to demonstrate stable, steady-state modes of tokamak reactor operation with high H* and high β_N^*, high bootstrap fraction, and effective power and particle handling.

The TPX Tokamak

TPX (Fig. 1) employs superconducting TF and PF magnets, making use of cable-in-conduit Nb_3Sn conductor in the TF and most of the PF circuit, and NbTi conductor in the outer PF coils. The major parameters of TPX are R=2.25m, a=0.5m, B=4T, κ_x=0.8, δ_x=0.8, I_p=2MA. The tokamak is capable of steady-state operation, but some external systems are designed for initial 1000 sec operation, with upgradability to steady state. Heating and current drive equipment includes 8MW of 120 keV neutral beam power, achieved by upgrading one of the present TFTR neutral beam lines, 8MW of 40–80MHz fast-wave power, based on TFTR FMIT transmitters, and 1.5 MW of new 3.7 GHz lower-hybrid power, with $n_{||}$ variable between 2 and 3. These systems are chosen to provide both ion (NB) and electron (FW, LH) heating, and bulk (NB), centrally peaked (FW), and variable off-axis (LH) current drive. The systems can be upgraded to 24, 18, and 3 MW respectively, for a maximum simultaneous heating power of 45 MW, corresponding to $0.75 MW/m^2$ of heat flux across the separatrix, as would be present in a tokamak reactor operating with $3MW/m^2$ of neutron flux. The basic tokamak performance (I_pR/a = 9MA) is chosen to permit an acceptable operational window[1], as shown in figure 2, between low densities, where fast-electron diffusion limits control of the driven current profile (resulting in the requirement $\tau_{se}/\tau_E < 0.125$), and high densities, where collisional effects significantly modify the bootstrap current profile (giving the requirement that the fractional collisional correction to I_{bs} be less than 0.15). The heating and current drive system is sized such that the baseline equipment provides access to regimes with β_N~3.3 at full field, assuming H=3, while not overly challenging the divertor[2]. At lower fields, higher values of β_N are accessible with the baseline systems ($\beta_N \propto HP^{1/2}/B$), while heating power upgrades could permit increased flexibility, higher β_N at full field, and/or divertor tests at more reactor-like parameters.

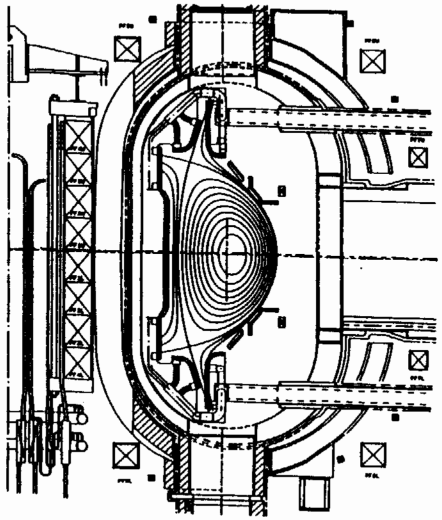

Fig. 1: Cross-section of TPX

The PF system is designed to allow a wide operating range in β_N and l_i: $0 < \beta_N < 5$, $0.5 < l_i(3) < 1.2$, at q~3.3; $0 < \beta_N < 7$, $0.5 < l_i(3) < 1.6$, at q=5. Ideal vertical stability is provided over this full range by internal passive stabilizers, which are connected in a saddle configuration, but grounded to the vacuum vessel through resistive titanium struts. Internal single-turn coils are used for rapid feedback control. While external coils located near the PF ring coils could provide position control, it was found that eddy-current heating of the 4K intercoil cold structure, due to the anticipated AC feedback fields, would be unacceptable. Slow external field-error correction coils are provided to reduce the occurrence of locked-mode disruptions. The strong plasma shaping (κ_x = 2.0, δ_x = 0.8) was chosen on the basis of results on DIII-D, where the highest combined β and τ_E is found in high triangularity, double-null plasmas, and

the theoretical prediction that high edge shear is important for high-β stability. The TPX divertor configuration[2] allows flexibility to accommodate the full range of l_i, which would be very difficult, if not impossible, in a divertor with both inner and outer slots. As l_i increases to its limits, κ_x must be decreased to ~1.6 to maintain vertical stability and acceptable PF current levels, and to provide room for strongly increased flux expansion near the X-point. The high aspect ratio of TPX is chosen to allow high f_{bs} at moderate q and β_N, as in a number of recent reactor studies[3] (including long-pulse inductive designs[4]). TPX will help provide the database required so that reactor optimization as a function of R/a can be possible in the future.

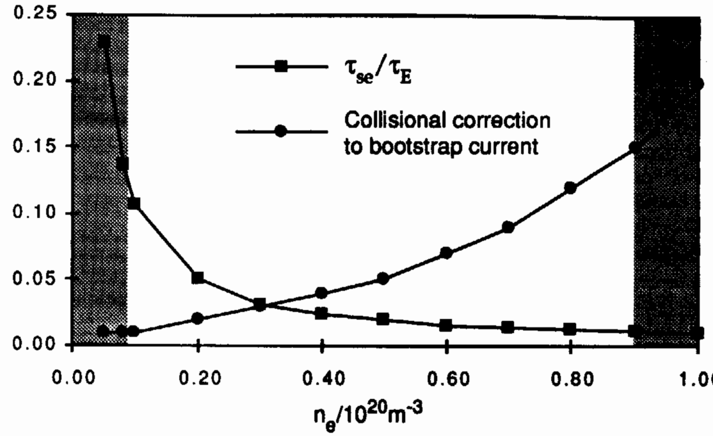

Figure 2: TPX density operating window at I_p = 1.5MA, β_N = 3.3, H = 3.

The TPX double-walled titanium vacuum vessel incorporates neutron and gamma shielding to limit heating of the superconducting coils, and to facilitate hands-on maintenance by reducing activation levels outside of the vacuum vessel. Full remote maintenance will only be necessary within the vessel, and will be provided by manipulators mounted on permanently installed interior tracks. Provision is made, for example, for full remote change-out of the divertor system. In case of failure of a major component (e.g., a TF coil), or the desire for hands-on reconfiguration of the vessel interior, activated components can be removed remotely and all radiation levels become acceptable for hands-on work within one year, even after 10 years at a neutron production rate of 6 x 10^{21} neutrons/year, corresponding to ~200,000 secs/year of full-power-equivalent tokamak operation.

Operational Modes of TPX

Results from a wide range of tokamak experiments[5] indicate that the current profile is a key element controlling both τ_E and β_N. Thus the TPX mission includes a highly synergistic combination of enhanced plasma performance and steady-state current drive (and consequently strong current-profile control). Steady-state operational regimes have been developed for TPX based on 1) standard high-β tokamak conditions, 2) the ARIES-I high bootstrap regime, 3) TFTR supershots, 4) Non-monotonic q profiles, as in JET PEP modes and recent Tore-Supra LHEP modes[6]

5) VH modes in DIII-D and 6) the ARIES-II high f_{bs}, high q, second-stability operating point. Perhaps most interesting is an extension of the non-monotonic-q case, which gives a very high bootstrap current fraction (93%) at $\beta_N^* = 6.3$ (β = 4.8%), at the full 2MA plasma current in TPX (Fig. 3). This regime is stable (up to $\beta_N^* = 7.5$) to modes with n = 1, 2, 3 and ∞, with either a conducting wall at b/a = 1.3, or a conducting structure which could be provided by vertical straps connecting the present passive stabilizers. Experimental evidence from DIII, DIII-D, PBX, and PBX-M suggests that such a structure may provide full stability. Resistive calculations show no sign of double-tearing modes, while quasi-linear drift-wave calculations indicate a strong stabilizing influence of reverse shear, an effect also predicted by the Rebut-Lallia-Watkins model. Current-drive simulations indicate that the necessary "target" q(r) profile can be established at low β, using the baseline heating and current drive system.

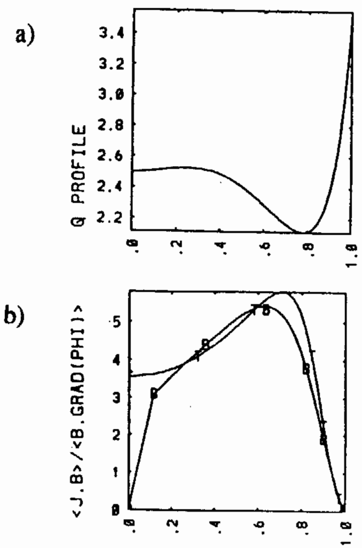

Fig. 3. a) q(ρ) and b) $j_{bs}(\rho)$ and $j_{tot}(\rho)$ for non-monotonic-q case.

TPX and ITER

TPX will make substantial contributions to the ultimate success of ITER. TPX is the only steady-state tokamak planned in the world program that can develop and test operating regimes that address the issues of fully non-inductive, high-bootstrap current drive, combined with reactor-relevant divertor conditions. The ITER Technical Objectives and Approaches specify that ITER must be capable of accommodating current-drive upgrades for steady-state operation. The ability to accommodate major modifications to the divertor system is also being incorporated into the ITER design[7]. The capability of TPX for flexible experimentation will provide very valuable guidance in optimizing the operations of ITER, and in the choice of future upgrades. By the same token, studies performed in the later phases of ITER will provide stepping-stones for advanced regimes developed in TPX to be tested, at least in part, for application in an economic, compact, steady-state demonstration power reactor.

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References

[1] W.M. Nevins et al., IAEA Würzburg, Germany, 1992 / Paper F-1-5
[2] M. Ulrickson et al., this conference.
[3] F. Najmabadi et al., IAEA Würzburg, Germany, 1992 / Paper G-1-1-1(R), J. Wesley et al., IAEA Würzburg, Germany, 1992 /Paper F-1-2, Y. Seki, et al., Rep. JAERI-M 91-081, JAERI, Naka, 1991
[4] N. Inoue et al., IAEA Würzburg, Germany, 1992 /Paper G-1-4
[5] IAEA Würzburg, Germany, 1992, papers from JET, TFTR, JT-60U, Tore-Supra, and others.
[6] G.T. Hoang, "Lower Hybrid Enhanced Performance in Tore Supra," 10th International Conference on Radio Frequency Power in Plasmas, 1993
[7] P.H. Rebut, private communication, 1993

POWER AND PARTICLE CONTROL IN THE TOKAMAK PHYSICS EXPERIMENT DIVERTOR†

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INTRODUCTION

The reference design for the Tokamak Physics Experiment (TPX) includes a double null divertor magnetic configuration. The main motivations for choosing the double null (DN) are the desire to operate with high triangularity (δ), DIII-D experimental results suggesting that high δ is favorable for advanced performance, theoretical indications that high δ is needed for access to the highest β regimes. The DN geometry also provides the flexibility to study the range of ℓ_i required for advanced tokamaks. TPX is being designed to have high plasma heating power (18-45 MW) and long pulse lengths (1000 s to steady state). The configuration of the TPX divertor components is shown in Figure 1. The divertor configuration is capable of handling SN operation and a wide range of different plasma conditions (*e.g.*, ℓ_i up to 1.6 and β_N up to 7). We will discuss the pumping speed required to maintain density control and the methods we have used to predict the particle throughput. Possibilities for improving divertor performance, such as a radiative divertor, are also discussed. The distance from the x-point to the divertor plate has been kept as long as practical (0.57 m in the poloidal plane and 16.3 m along a field line) in order to allow room for a radiative divertor.

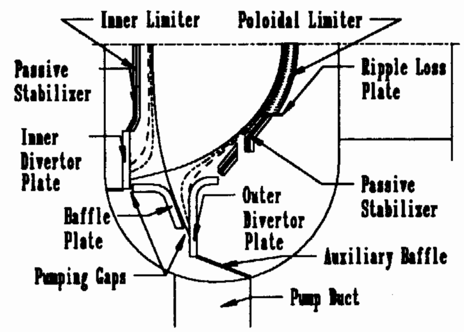

Figure 1 A cross section of the lower divertor of TPX showing the plasma facing components, gas baffles, and pumping duct.

EDGE PLASMA MODELING

We have modeled the boundary layer plasma in TPX using the B2 code. A range of plasma transport and particle recycling coefficients have been considered. Approximately 50 different cases have been studied

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