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Divertor design for the tokamak physics experiment ☆

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Abstract

In this paper we discuss the divertor design for the planned TPX tokamak, which will explore the physics and technology of steady state (1000 s pulses) heat and particle removal in high confinement (up to 4 × L-mode), high beta (up to $\beta_N = 5$) divertor plasmas sustained by non-inductive current drive. TPX will operate in the double-null divertor configuration, with actively cooled graphite targets forming a deep (0.57 m) slot at the outer strike point. The peak heat flux on the highly tilted (74° from normal) re-entrant divertor plate (tilted to recycle ions back toward the separatrix) will be in the range of 4–6 MW/m^2 with 17.5 MW of auxiliary heating power. The combination of pumping and gas puffing (D_2 plus impurities), along with higher heating power (45 MW maximum) will allow testing of radiative divertor concepts at ITER-like power densities.

1. Introduction

Work is underway to complete the design of the TPX tokamak (tokamak physics experiment), which is scheduled to begin operation in 2000. The mission of TPX is to explore attractive steady state reactor regimes (i.e., having pulse lengths much longer than the equilibration times for the current profile or the plasma–wall interactions) with high confinement ($H \leq 4$, where H is the enhancement factor over L-mode scaling) and high plasma pressure ($\beta_N \leq 5.0$, where β_N is Troyon factor $\beta/(I/aB)$). Present experiments [1] suggest that such conditions can be best achieved in divertor plasmas with high elongation and triangularity, with good density control to permit efficient current drive and current-profile control. Thus, a double-null (DN) divertor configuration has been selected, though the coil systems have been designed to allow for single-null operation as well. In this paper we present an overview of the TPX divertor including the physics basis for our design choices and a description of the hardware's capability to meet the system requirements.

TPX is being designed as a long-pulse tokamak ($\tau_{pulse} = 1000$ s) with superconducting magnets and auxillary heating suitable for current drive experiments at power levels from 17.5–45 MW. The heating systems available during initial operation will include 8 MW of neutral beams, 8 MW of ion cyclotron rf heating, and 1.5 MW of lower hybrid heating/current drive. Baseline machine parameters and typical initial operating conditions expected for the main plasma are contained

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Table 1
Baseline TPX parameters

$R = 2.25$ m	$a = 0.5$ m	$\kappa_x = 1.6$–2, $\delta_x = 0.8$
$B_T = 4$ T	$I_p = 1.6$–2 MA, $q_{95} = 3$–5	$\tau_{pulse} = 1000$ s
$V_p = 20$ m^3	$A_p \simeq 60$ m^2	$A_{div} \simeq 1.2$ m^2
$P = 18$ MW	$\langle n \rangle \geq 5 \times 10^{19}$ m^{-3}	$Z_{eff} \leq 2$

in Table 1. D–D neutron fluence will be high enough to necessitate the use of low activation materials such as titanium for the vacuum vessel, sheilding to limit the nuclear heating of the coils, and remote handling techniques such as an in-vessel robot manipulator after the first two years of operation.

We have adopted a conservative design for the TPX divertor, as shown by the cross section of the lower divertor in Fig. 1. The double-null configuration desired for MHD stability at high elongation has the advantage that it minimizes the power and particle flux at the inner target, where access is difficult for placement of large cooled structures, such as slots, in highly triangular discharges. Thus, a flat inner target is possible, which provides greater flexibility in plasma shaping. At the outer target an inclined plate (74° from normal to the separatrix flux surface) maximizes the surface area available for heat removal so that no radiative divertor operation is required until the heating power exceeds 27 MW. The central baffle in the private region helps gas stay trapped near the outer target and improves particle exhaust by minimizing back-streaming from the pumping plenum. Rapid feedback control of the particle throughput (for density control and regulation of divertor radiation) is obtained by slightly shifting the separatrix intercept relative to the plenum entrance slot, or by throttling the external vacuum pumps. All of the plasma facing surfaces shown will be fabricated from carbon fiber composites to minimize their susceptibility to severe damage from off-normal heat loads due to disruptions. Finally, the inclusion of gas valves for deuterium or impurity gas puffing combined with a strong pumping capability, will allow us to increase local radiative cooling of the SOL plasma in order to operate at the highest heating powers.

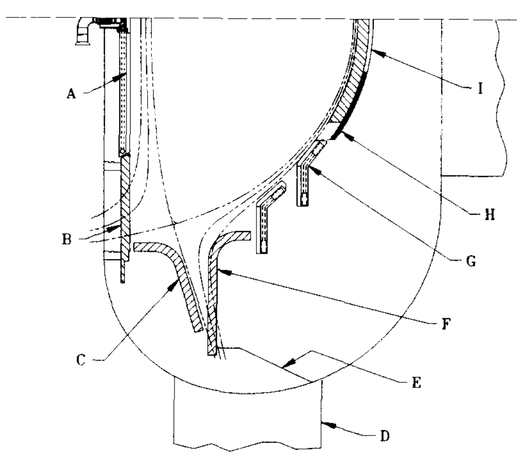

Fig. 1. Cross section of the lower half of TPX showing plasma facing components. A – inner limiter; B – inner divertor target plate; C – divertor baffle plate; D – pump duct; E – gas baffle; F – outer divertor target plate; G – passive stabilizer; H – ripple-loss armor; I – poloidal limiter.

Table 2
Input assumptions for modeling

Parameter	Baseline values	Range of simulations
P_{heat} (MW)	18	18–45
P_{SOL} [a] (MW)	4.7	1.8–21
$P_{d,out}/P_{d,in}$	4/1	2/1–4/1
n_{sep} (10^{20} m^{-3})	0.165 ($\langle n_e \rangle/3$)	0.165–0.60
Z_{eff}	1.84	1.67–2.0
χ_e, χ_i (m^2/s)	2, 0.67	1–4, 0.67–2.7
v_{conv} (m/s)	0	0, −2
$L_{\parallel}$ (m)	72 [c]	55–75
θ_{plate}	90	90
R_{plate} [#]	0.999	0.95–1.000

[a] $P_{SOL} = 0.6 \times 0.55 \times 0.8 \times P_{heat}$.
[b] For SN discharges only.
[c] Determined on field line 0.5 cm in SOL.
[d] Particle recycling coeff. at the target plate.

2. Expected parameters for divertor operation

The operating points used for the divertor design are based on modeling of the scrape-off layer plasma using 1-d (NEWT-d) and 2-d (UEDGE, b2, b2.5) codes [2], along with Monte Carlo simulation of the neutral gas transport (DEGAS). The peak heat flux predicted by the simulations was also compared with interpolation/extrapolation of data from the DIII-D and JT-60U tokamaks. Overall, this approach is very similar to that of the ITER CDA team [3].

The models require certain input assumptions, which we list in Table 2. The distribution of power among the four divertor targets is based on DN data from PDX, ASDEX, and DIII-D experiments. The column labeled Baseline Values represents the choices for simulating the nominal TPX operating conditions with P_{SOL}/P_{heat} = 18 MW, $\tau_E = 2 \times$ L-mode, and $I_p = 2$ MA. ELMing H-mode conditions were generally assumed (e.g., $n_{sep}/\langle n_e \rangle = 0.33$, $P_{rad,core}/P_{heat} = 0.30$, so that $P_{SOL}/P_{heat} = 0.70$). The radial transport coefficients $D\perp$ and $\chi_{e,i}$ used here fall in the range reported from model validation work by various tokamak groups, though more recently, smaller values in the range of 0.1–0.5 m^2/s

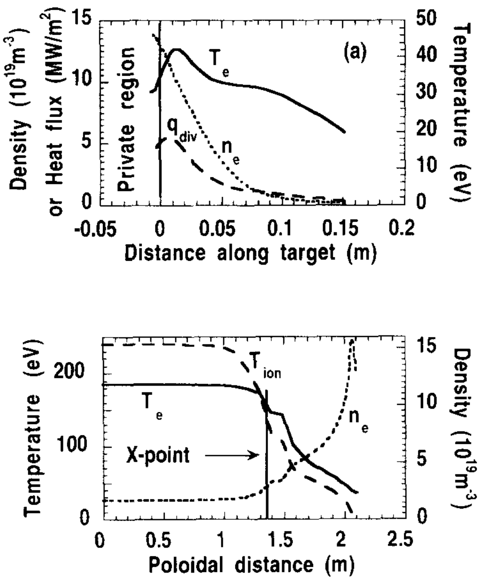

Fig. 2. (a) Radial profiles of $n_{e,\,div}$, $T_{e,div}$, and q_{div} across the outer divertor target, (b) profiles along a SOL field line just outside the separatrix.

have been reported by the DIII-D group [4].

The plasma density on the separatrix is a very important factor for determining the peak heat flux and electron temperature at the divertor targets, so we have based our assumption in Table 2 on the published data for ELMing H-mode discharges [5]. Low density on the speparatrix implies high edge temperatures, high parallel thermal conductivity, and a narrow scrape-off layer. TPX current drive experiments will require operation at relatively lower density than is usually obtained in H-mode discharges: $\bar{n}_e = 0.4\,\bar{n}_{Greenwald}$ versus $\bar{n}_e \geq (0.5\text{–}0.8)\bar{n}_{Greenwald}$. Recent experiments in DIII-D with divertor cryopumping have achieved such densities and have observed an increase in T_e on the separatrix near the midplane and at the inner strike point in a single-null configuration [6].

The numerical simulations show that acceptable divertor conditions ($T_{e,div} = 40$ eV, and $\hat{q}_{div} = 4\text{–}6$ MW/m^2) should be obtained in the TPX divertor when running non-inductive current drive experiments with the baseline heating power of 18 MW. Fig. 2a shows the $n_{e,div}$, $T_{e,div}$, and q_{div} profiles across the outer divertor target for such a discharge. Here we have expanded the radial scale to account for the 74° plate tilt, since the B2 simulations are presently operating with orthogonal target plates. The total particle flux across the separatrix computed by B2, about 10^{21}/s, is consistent with a particle confinement time of $\tau_p = 2\tau_E = 0.4$ s, where τ_E is twice L-mode. The variation of n and T along a SOL field line just outside the separatrix appear in Fig. 2b. These results were confirmed with a similar set of UEDGE runs.

Comparison between the numerical simulations and experimental data suggest that the codes give pessimistic results for the predicted TPX divertor heat flux. Itami has published [7] a scaling relation for the peak heat flux in JT-60U single-null discharges, as has Hill for DIII-D discharges [5]. If we apply their results to TPX, correcting for the differences in machine size, the SOL power distribution (DN versus SN), and the magnetic flux expansion, we obtain about 3–4 MW/m^2, which is lower than our simulations would predict. Now that Alcator C-mod is running, we have SOL data at higher magnetic fields (5 T); they find a power fall-off length of approximately 4 mm for ohmic plasmas, which is just about what was predicted based on connection length scaling from Alcator C. As far as the in/out and up/down distribution in double-null plasmas, the only recent data is from DIII-D, which has reported that the peak heat flux on the inner target is less than 10% of that at the outer [8]. Earlier data from ASDEX and PDX showed a 4:1 out:in power split for DN plasmas [9].

3. Hardware design and capability

The actively-cooled plasma facing components in TPX will be made from carbon fiber composite graphite. The centerpost armor consists of carbon tiles bolted to a cooled titanium substrate with a power handling capability of about 1 MW/m^2. The outer toroidal limiters located above and below the plasma midplane actually form part of a conducting shell used to enhance MHD stability, and as such, have carbon tiles fastened to a copper structure. The vertical straps which connect the upper and lower parts of this shell will form a set of discrete poloidal limiters, in addition to the three main poloidal limiters used for plasma start-up and ramp-down.

The divertor structure in TPX is designed primarily to spread the heat flux over as large an area as possible. The vertical inner and outer target plates accommodate a wide range of plasma elongation and triangularity and provide a "re-entrant" geometry for recycling so that neutrals recycled at the plate are returned to regions of higher plasma density and ionization probability, and thus stay better trapped in the divertor region. The nominal distance from the X-point to the strike point is 0.57 m. This divertor shape will also allow for considerable variation in flux expansion near the X-point produced by changes in the plasma current profile (internal inductance and elongation). Details of the proposed design can be found in Ref. [10].

The high heat flux surfaces on the vertical divertor targets are designed to handle 7.5 MW/m^2 peak steady state heat flux, which is nearly twice that predicted by the plasma models. The central divertor baffle has a design limit of 4 MW/m^2, set by the expected maximum radiative heat flux of 1.7 MW/m^2 (assuming line radiators near the X-points dissipating a total of 70% of the heating power). The peak surface temperature will be about 1400°C for the outer target tiles, which will consist of carbon blocks brazed onto a water-cooled copper tube fastened to the support structure. The design of the cooling system is such that, if the peak heat flux were to exceed the design limits by a factor of 2 or more, catastrophic failure should not result because the graphite surface will ablate and limit the temperature rise before there is bulk coolant boiling.

The need to provide remote handling capability is an important element for the TPX first-wall design, since the expected neutron flux and pulse length will quickly make human access to the interior of the tokamak impractical once deuterium operation begins. Present plans call for remote manipulator arms traveling on rails mounted behind the toroidal limiters to replace the plasma facing components such as the limiters, divertors, and centerpost armor tiles. In order to fit through the access ports, each of the 32 divertor modules will consist of two major parts: the outer target/central baffle, and the inner target. All the high heat flux components will be mounted to internal continuous rails to ensure reproducible alignment relative to the magnetic field. The remote maintenance hardware will be tested early in the operational phase, when hygrogen fueling will be used to minimize activation and still allow human access.

Particle control in TPX will be provided by eight external cryopumps connected to the upper and lower divertors via 16 vertical ports. The central divertor baffle below the X-point defines a toroidally continuous pumping plenum isolated from the plasma and connected to the pumping ducts. Neutrals enter the plenum through a 6.5 cm toroidal gap (entrance slot) at the bottom of the divertor slot. No pumping is provided at the inner strike point since the pressures there are expected to be much lower. The particle throughput (70 Torr l/s) has been calculated using the DEGAS code to track the probability that recycled particles launched from the outer target plate will reach the pump duct given the plasma conditions at the plenum entrance. For typical conditions, only about 5% of the atomic source is pumped, with the rest being reionized by the divertor plasma; the pumped fraction increases strongly as the slot is made narrower, as can be seen from a simple flux balance argument first used in the design of the DIII-D divertor pump [11]. The throughput to the pumps can be varied by changing the strike point position or by throttling the cryopumps, both of which can be accomplished rapidly during the discharge.

Erosion of the target plate materials should not be a limiting factor for TPX, which is expected to operate a total of about 2×10^5 s per year. Using the REDEP code [12] we have examined the expected net erosion of the outer target plate for standard operating conditions. These results are summarized in Table 3. Note that the lifetime to 50% erosion assumes that the strike point remains at a fixed position, which is unlikely given the wide range of experiments planned.

Table 3
Erosion lifetimes

Material	Max. rate (cm/s)	Lifetime to 50% (s)
1 cm carbon	1.1×10^{-6}	4.5×10^5
0.35 cm beryllium	7.6×10^{-7}	2.2×10^5
Tungnsten	3.0×10^{-9}	machine lifetime

4. Radiative divertor operation

TPX operation at its highest heating power (45 MW) will require a factor of 3–5 reduction in the peak divertor heat flux in order to stay safely below the thermal limits of the cooled structure, since B2 modeling predicts a peak heat flux exceeding 15 MW/m^2, and scaling from experiments suggests values above 10. As in ITER, we plan to reduce the heat flux by increasing the radiative losses in the edge, scrape-off layer, and divertor plasmas through impurity plus deuterium gas fueling in the divertor region. Significant heat flux reduction by gas injection has already been demonstrated in a number of divertor tokamaks [13]; the challenge in TPX is to do so while maintaining good core energy confinement and current drive efficiency (low Z_{eff}, moderate density, and high T_e). Early experiments at 18 MW will provide valuable data on steady state divertor operation for ITER. At 45 MW, the edge plasma temperatures and parallel heat flux will approach values predicted for ITER ($P_{sol}/A_{surf} \leq$ 0.5 MW/m^2), making this a relevant test of the physics and technology of the radiative divertor concept as a solution for ITER's divertor problem.

So far, we have begun to look at radiative divertor scenarios for TPX using the B2.5, UEDGE, and NEWT-1d codes. In these, the model of the radiation efficiency for impurities allows for non-equilibrium charge-state distributions to develop due to charge-exchange recombination and finite impurity lifetime in the SOL plasma. Both the NEWT-1d and B2.5 codes solve for the self consistent impurity transport in the presence of the background plasma, and the resulting impurity radiation can affect the thermal forces on the impurity ions.

All three codes predict that increasing the impurity concentration in the divertor will reduce the expected divertor heat flux. We focused on neon and argon because they should be efficient radiators at the low temperatures expected in the divertor region, but poor radiators in the very high temperature core plasma ($T_e(0) \simeq 8$–20 keV). With the B2.5 code we are finding that concentrations of neon and argon below 1% at the plasma midplane can produce significant heat flux reductions, though in this case the upstream density on the separatrix had to be increased to greater than 4×10^{19} m^{-3}.

The NEWT-1d calculations show that it is possible to use argon injection to radiate nearly all the power flowing in the SOL plasma, though again, the impurity concentration is higher than desired. The impurity concentration upstream depends sensitively on the balance between the thermal forces ($\alpha \nabla T_{e,i}$) pushing toward the midplane and the frictional forces due to deuteron flow toward the target plates. Our modeling shows that the addition of a midplane D_2 gas puff or pellet fueling to increase the flow velocity in the SOL can reduce the midplane impurity concentration, though at the penalty of higher edge density. Recent experiments on DIII-D have shown that the combination of divertor pumping and midplane fueling can reduce core plasma impurity contamination by factors of 3 or more [14].

The hardware configuration of the TPX divertor is not designed specifically to produce radiative divertor solutions, since experimental results show that large reductions can be obtained in a wide variety of divertor geometries. However, it is our belief that the re-entrant slot geometry, which has been shown in calculations to promote high recycling, is likely also to be optimal for radiative divertor operation. It concentrates neutrals in the high heat flux region of the SOL where they are most needed and limits the escape of gas out of the divertor back to the main plasma. Furthermore, having the ability to quickly (≤ 100 ms time scale) adjust the particle throughput by shifting the strike point away from the entrance to the pumping slot will allow us to better maintain the radiative equilibrium as the heating power or main plasma conditions vary. Divertor plasma models are just now reaching the necessary sophistication to guide designers in determining how the machine geometry can be optimized to increase the heat flux reduction attainable with impurity or deuterium gas injection; the flexibility of the TPX divertor design will allow future hardware modifications to test any new concepts.

5. Summary

In this paper we have presented the design philosophy and physics basis for the TPX divertor. As shown, the actively cooled carbon divertor targets should handle the expected peak divertor heat flux for long pulse operation with 18 MW of heating and current drive power. In terms of scrape off layer power density, TPX at first represents only a modest extrapolation from the presently operating DIII-D, JET, and JT-60U, but in terms of cooling technology and particle control requirements, TPX represents a large step towards testing concepts for future reactor-size devices such as ITER and DEMO. Eventually, at its highest heating power, TPX will push edge temperatures and power densities well into the ITER&DEMO-like regimes, where radiative divertor concepts for heat flux reduction need to be tested.

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