# The Super X Divertor (SXD) and High Power Density Experiment (HPDX)

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**Abstract.** A new magnetic geometry, the Super X divertor (SXD), is invented to solve severe heat exhaust problems in high power density fusion plasmas. SXD divertor plates are moved to the largest major radii inside the TF coils, increasing the wetted area by 2-3 and line length by 3-5. 2D simulations show a several fold decrease in divertor heat flux and plasma temperature at the plate. A small high power density device using SXD is proposed, for either 1) useful fusion applications using conservative physics, such as a Component Test Facility or 2) to develop more advanced physics modes for a pure fusion reactor in an integrated fusion environment.

## 1. Introduction

A steady-state fusion reactor will have much higher heating power $P_h$ and pulse length than ITER, which itself is a factor of about six beyond current fusion machines. Invoking the standard measure $P_h/R$ for the severity of the heat flux, we observe: a) the two largest current tokamaks JET and JT-60 each have $P_h/R\sim 7$, b) ITER[1], with $P_h\sim$120MW and R~6.2 m, has $P_h/R\sim 20$, but c) even a moderate fusion reactor[2-5] ($P_h\sim$400-720 MW at R~5-7 m) will have a much larger $P_h/R\sim$ 80-100. Since $P_h/R$ for even compact fusion devices for Component Test Facilities (CTF) will be much larger than ITER's $P_h/R$, the ITER-like standard divertors (SD) cannot be expected to handle such huge increases in heating power density[6].

This huge power density, coupled with the range of scrape-off-layer (SOL) projections, implies that acceptable divertor operation is, perhaps, the most serious roadblock in the march towards economically desirable power densities for fusion. A high SOL power flux leads to operation in the sheath limited regime- a highly undesirable regime associated with high plate erosion, low impurity shielding, low neutral pressures for problematic helium exhaust, and low divertor radiation and high divertor heat fluxes. Attempts to dissipate excess heat via core radiation preclude good confinement, and probably high β[6]. The low power density of ITER gives a $P_h/R$ sufficiently low to allow a standard divertor (SD) to cope, but such SDs are not likely to extrapolate to power densities several times higher.

The Super X Divertor (SXD)[7], created via a redesign of the divertor magnetic geometry, offers a simple and robust, axisymmetric solution to the high heat flux problem. By maximizing divertor power capacity, SXD reduces the core radiation burden, thus enabling a large array of core plasma optimizations needed to attain high fusion power density. SOLPS[8-9] simulations for SXD equilibria generated with CORSICA[10] show striking SXD advantages. We present here the conceptual design of a compact, relatively inexpensive high power density fusion device (HPDX) to show how SXD can enable the required high integrated performance.

## 2. The Super X Divertor (SXD): basic idea, design, and simulations

The basic idea behind the axisymmetric SXD is to move the divertor plates to the largest major radii allowed inside the toroidal (TF) coils while keeping the main plasma geometry unchanged. The new configuration tends to isolate the divertor from the main plasma in many ways. The wetted area and line length increase, respectively, by factors 2-3 and ~3- 5 over and above the best that can be obtained by any of the proposed flux expansion methods (plate near main X-point, extreme plate tilting, X-divertor, snowflake divertor[11], etc.); all of the latter, because of engineering constraints between the field line and divertor plate (found to be 1 degree on ITER), can manage to raise the wetted area above an ITER-like SD by no more than a factor of ~2.

In a standard divertor (SD) configuration like ITER or JET or JT-60, the heat flux concentrates on a very small plasma-wetted area on the divertor plates. Our first attempt, the X divertor (XD), increased the plasma-wetted area by flaring the open magnetic field lines near the plates using extra X points. However, a similar plasma-wetted area expansion can be attained with an SD using a highly tilted plate; it is only the extra line length that makes XD modestly better than SD. Since parallel heat transport along the open field lines to the divertor plates dominates the cross-field transport, the plasma-wetted area $A_w$ on the divertor plates is approximately

$$A_w = \frac{B_{p,sol}}{B_{p,div}} \frac{A_{sol}}{\sin(\theta)} \approx \left[\frac{B_p}{B_t}\right]_{sol} \frac{R_{div}}{R_{sol}} \frac{A_{sol}}{\sin(\theta)} = \left[\frac{B_p}{B_t}\right]_{sol} \frac{2\pi R_{div} W_{sol}}{\sin(\theta)}, \tag{1}$$

where $R_{sol}$, $W_{sol}$, and $A_{sol}$ are the radius, width, and area of the scrape-off layer (SOL) at the midplane, $\theta$ is the angle between the divertor plate and the total magnetic field $B_{div}$, and subscripts p (t) denote the poloidal (toroidal) directions. For plasma with a given $W_{sol}$ and $B_p/B_t$ at midplane sol, $A_w$ can be increased only by reducing $\theta$ or by increasing divertor plate major radius $R_{div}$. Due to engineering constraints, $\theta$ must be greater than about 1 degree, so the only remaining "knob" to increase $A_w$ is to increase $R_{div}$. SXD does just this with simple PF coils.

The surprising discovery is that a large increase in $R_{div}$ can be achieved with relatively small modifications (in positions and currents) to the conventional poloidal field (PF) coils for a standard divertor of a large range of devices[1-5,12-14]; the TF coils do not need to be changed at all. In Figs. 1-3, we show SXD CORSICA equilibria for FDF[12], a low aspect ratio CTF[13], the superconducting (SC) SLIM-CS reactor[3], and the HPDX. The SLIM-CS case shows that an SXD can be obtained with SC coils all outside the TF coils. A common "trick" to generate an SXD from an SD is to create an extra X point which make the separatrix turn from near-vertical to near-horizontal and reach a large $R_{div}$. with a simultaneous significant flux expansion and increase in the line length. Further flux expansion near the plates can be attained by "splitting" the SXD coil as shown in Fig.1 (right); the split coil pair needs nearly the same total current as the single SXD coil in Fig.1 (left).

| Div Type | $R_{div}$ [m] | B Length [m] | Max Aw | Max T eV | Peak MW/m2 |
|---|---|---|---|---|---|
| SD | 2.34 | 27.4 | 3.30 | 150 | 58 |
| XD | 2.51 | 39.7 | 3.54 | 150 | 28 |
| SXD | 4.01 | 73.6 | 5.57 | 10 | 15 |

Table 1. CORSICA and SOLPS (no impurity radiation) results for FDF SXD shown in Fig.1.

Table 1 shows SOLPS results for the FDF-SXD (no impurities) in Fig.1. Details of calculations for SXD configurations for FDF, NSTX, NHTX[14], and CTF are in Ref. 8. All devices show a marked decrease (by ~5 times) in peak heat flux and plasma temperature at plate $T_{div}$ (to <10 eV) for SXD vs SD. SOLPS also shows that lower $T_{div}$ allows more radiation in the SXD than in SD or XD. This reduces core radiation requirements and enables many core plasma optimizations essential for AT reactors.

Note that examples in Figs. 1-3 are not optimized. SXD can also be designed without changing a wide variety of plasma shapes and parameters, using all axisymmetric PF coils, or with modular PF coils as in Ref.6. The change in the net PF coil Amp-m for SXD is only about ±5% from the SD case.

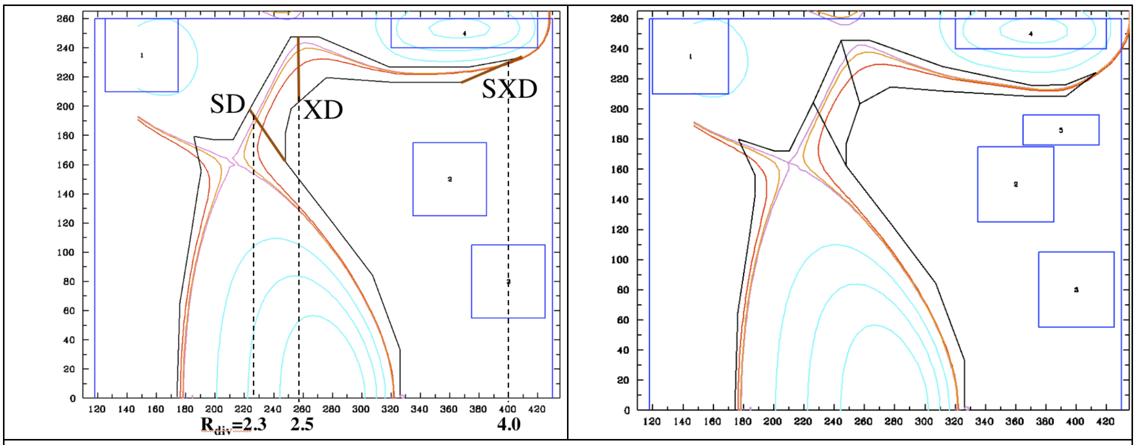


*FIG 1. CORSICA SXD equilibria for FDF: Left: with1 SXD coil (in top right corner of TF), Right: with 2 coils carrying same total current as the single coil.*

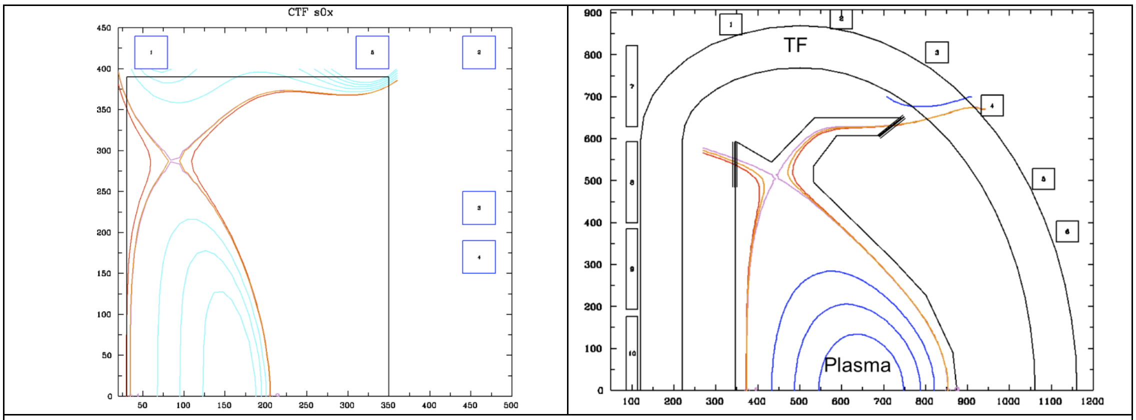


*FIG 2. Left: SXD for low-A CTF, Right: SLIMCS reactor with SC PF coils outside TF.*

In general, the best way to deploy SXD is in a double-null configuration where most of the power exhaust goes to the outer divertor legs. With even a fivefold reduction in peak heat flux on the outer SXD legs, the inner legs do not become the limiting factor.

An important generic feature of SXD is that it is very insensitive to changes in plasma current profiles. Figure 3 (right) shows a CORSICA scan in which the plasma current was changed by ±4%. The main plasma X point moved much more (±3 cm) than the strike point on the SXD plate (±3 mm). Similar insensitivity is seen for small plasma motions. The geometry in the long SXD leg is controlled more by the nearby SXD PF coils than by the distant plasma current. The long SXD leg can be held fixed as the plasma changes or moves.

The overall advantages of SXD accrue from both the increased wetted area, and the longer line length. In addition to FDF and CTF, SOLPS simulations of NHTX-SXD show that it reduces the large outboard heat flux by a factor of 5 - the only method demonstrated (in SOLPS simulations) to bring NHTX peak heat flux below the 10 MW/m$^2$ engineering limit on divertor plates.

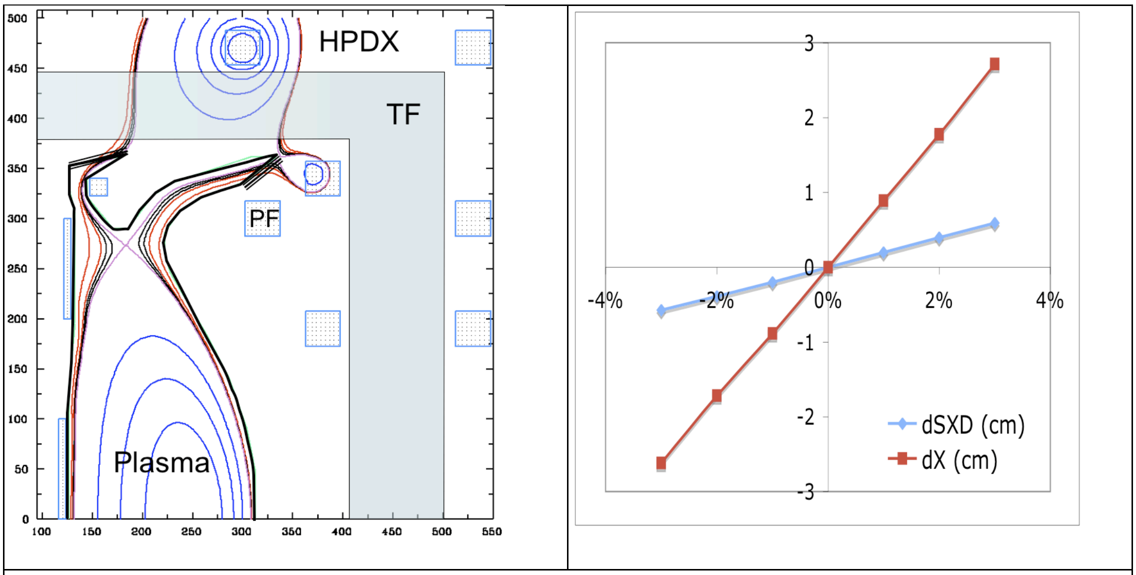


*Fig. 3 Left: HPDX with SXD. Right: Shift in cm (red boxes: main X point, blue diamonds: SXD strike point) vs % change in core plasma current, showing the robustness of SXD.*

The advantages of SXD are particularly striking, in fact, critical for future low aspect ratio (ST) devices, like the ST Component Test Facility (CTF) shown in Fig.2 (left). Even with the outboard divertor channel extending only to R=2.5 m, the divertor surface area jumps up to more than 25 m$^2$ for each divertor (with an assumed SOL width of 0.5 cm). The divertor could withstand the full brunt of all the heating power (without recourse to radiation even at maximum power) to deliver 4 MW/m$^2$ neutron wall loading [13].

Unlike SD, SXD plates are far enough from the plasma so that substantial neutron shielding could be provided; MCNP calculations yield that shielding affects a drastic reduction (to 10%) in the impinging neutron flux. This unique characteristic of the SXD may, by itself, render an SXD unavoidable for any neutron-producing tokamak by letting, for example, an HPDX or a DEMO employ critical divertor materials (Cu and CFCs) which would otherwise undergo severe degradation under simultaneous high heat fluxes and neutron fluences.

SXD is also expected to reduce steady state and peak heat fluxes from ELMs and disruptions, shield the divertor from halo currents, and enable operation at much lower core radiation and edge density – thus reducing disruption probability. An SXD might survive much larger ELMS of a burning plasma by spreading the heat pulse over a larger area and longer time (due to its longer line length). The longer line length may also enable fully detached operation without confinement degradation by holding the detachment front away from main plasma. The SXD is also fully compatible, and sometimes synergistic, with other methods such as using liquid metals (since MHD drag is lower at small B, and the long divertor throat shields evaporated impurities).

**3. A High Power Density Experiment (HPDX) using the SXD**

The 2007 ITER physics basis identifies divertor limitations as the key roadblock to higher fusion power density in steady state scenarios[1] -"The fusion gain in steady state maximizes at low density for constant $\beta_N$. The limitation on reducing the density in next generation tokamaks is set by the impact on the divertor". Yet, studies of possible fusion reactors based on Advanced

Tokamak (AT) modes[2-4] (and also steady state CTF devices[13-14]) postulate up to an order of magnitude higher power density than ITER. The only way to reconcile this apparent contradiction is by using a much "better" divertor than the ITER SD. The SXD was designed precisely to meet such a challenge. The mission of a small high power density experiment (HPDX) using the SXD could be either 1) short term fusion applications using conservative physics, such as a Component Test Facility or 2) to inexpensively develop advanced physics modes for a pure fusion reactor in an integrated fusion environment with high heat fluxes.

To demonstrate, quantitatively, the importance of an SXD for a workable design, we now dwell on a compact reference device by choosing a set of reasonable but definite parameters. For low size, coil mass, and easy maintenance, one considers (similar to ST reactor designs[15-16]) a low aspect ratio A, Cu coil device in which enough room is allowed for a modest neutron shield (~ 0.1 m ) of the inner coil. The shield for this small device constrains the A away from very low values; detailed costing studies also find A~2 to have a lower cost than lower A[17]. We take A=1.8, R=1.35 and elongation $\kappa$=3. This elongation is consistent with other proposed devices. A fusion power of 100MW gives to the HPDX an average neutron wall load of 0.93 MW/m$^2$, as required for a CTF.

To examine the physics and technology requirements of HPDX, we first consider the required values of dimensionless physics parameters $\langle\beta\rangle_N$ and H. (Here $\langle\beta\rangle_N = (\langle p\rangle/\langle B^2\rangle)/(I/aB)$, and $\langle B^2\rangle$ is the volume average *total* $B^2$. The no-wall stability limit is $\beta_N$ ~3 for all aspect ratios for this $\langle\beta\rangle_N$[18]. H is confinement enhancement above ITER98H(y,2).)

To calculate the current drive power for a 100MW fusion power at a given $\langle\beta\rangle_N$, we use an estimated current drive efficiency $I\, n_eR/P_h$= 0.3 x $10^{20}$ ($\langle Te\rangle$/10kev) A/Wm$^2$- very close to what is found in reactor studies and ITER analysis. Numerical VMEC[19] equilibria, with fixed temperature and density profile shapes - characteristic of low collisionality hybrid H-modes - are used to determine $\langle\beta\rangle_N$, the bootstrap current, and the fusion power from thermal cross sections. The total current is the sum of bootstrap and driven currents. Consistent with ITER physics basis, we read from Fig.4 that the current drive power requirement increases strongly with the density, resulting in low fusion gain. The reason why confinement requirement becomes easy to satisfy at high density is because the heating power from current drive ends up being huge; the high density scenario comes with a high cost, of course.

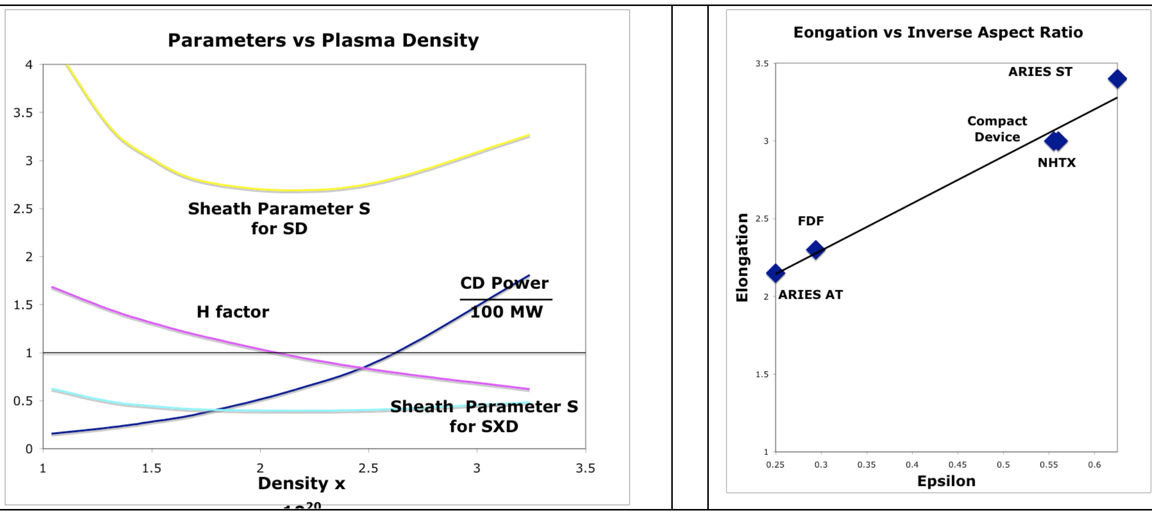


*Fig. 4 Left: Parameters vs Density for the Compact 100MW fusion Device for $\langle\beta\rangle_N$=3. Right: Elongation vs Aspect Ratio for recently proposed devices*

It was also noted in the ITER physics basis that at low density, though the current drive efficiency is high, the divertor operation becomes problematic. To illustrate the severity of the problems that an SD will face in compact high power density machines, we take a small digression to discuss the criteria for a sheath-limited regime, which, as emphasized in the introduction, is unacceptable for many reasons

A simple estimate, to determine if the divertor is in the sheath limited regime, reads $Q_{||u}$ $(B_{div}/B_u)$ $/n^{1.75}L^{\ 0.75} > 10^{-27}$ W/m$^{2.5}$ in MKS units. The original analysis[20] did not included the possibility of significant variations in B in the SOL- we find it can be included through the factor $(B_{div}/B_u)$, where u and div refer to quantities evaluated at the location of the outboard upstream and divertor, respectively. The parallel heat flux $Q_{||u}$ is taken as $P_{SOL}/4\pi R_u\lambda_q$, where $\lambda_q$ is the upstream SOL power width. We have benchmarked this expression using SOLPS simulations for compact devices (NHTX, FDF, HPDX, etc.). When $S'=Q_{||u}$ $(B_{div}/B_u)$ $/n^{1.75}L^{\ 0.75} > 2$ x$10^{-27}$, the SOL is consistently sheath limited with unacceptable divertor electron temperatures (above 100eV while the ITER value is ~ 10eV).

From Fig 4, where we plotted, among others, the sheath parameter $S=S'/2$ x$10^{-27}$ as a function of density, we learn that the SD is always well into the sheath limited regime, whereas the SXD avoids that regime for all densities. The expected benefits of high density operation for the divertor are nullified by the greatly increased heating power from current drive. Since this compact device is in the sheath limited regime, divertor radiation is low; it is a severe loss since such radiation reduces heat fluxes by ~ 50% on ITER. At high density the compact device also has a very high P/R (~70- 120MW/m vs 20MW/m for ITER), so heat flux difficulties with an SD are expected to get severe. With an SD, plate tilting would be required far beyond the ITER engineering limit. Even if this were deemed feasible for some reason, equally debilitating problems like plate sputtering and erosion, low impurity screening, very low helium exhaust, extreme sensitivity to plasma positioning, and neutron damage to divertor components would still remain with the SD option.

We note that for a higher $\langle\beta\rangle_N$ one could make do with lower current drive. But as long as the bootstrap current fraction is below 70% (apparently required for acceptable q profile control in present experiments), S is always above 1. At high density and $\langle\beta\rangle_N \sim 3$, it is possible to assume enough core radiation (> 50%, e.g. from seeded impurities) that S can drop below 1 maintaining H ~ 1. This scenario, however, will require heating power to be well over 100 MW. In addition the core radiation caused peak surface heat flux on the first wall ( ~ 1 MW/m$^2$) will be a serious engineering feasibility issue with near term structural materials. The long preceding discussion clearly indicates that a divertor of the class of SXD, much better than SD, is essentially a necessity for a compact high power density machine like HPDX.

With the severe divertor problem under control, the relatively mild physics requirements of HPDX can be readily satisfied since $\langle\beta\rangle_N \sim 3$ in the required range of H has been demonstrated routinely in hybrid operation on several tokamaks. For these parameters and off axis current drive, it is possible to operate with $q_{min}>2$, removing the threat of the most dangerous tearing modes. Finally, the midrange densities of Fig. 4, are at about a third of the Greenwald limit. Together with low to modest radiation fractions, this should reduce the disruption probability.

As for other engineering feasibility issues, we note that the B field at the center post is 7T, less than or equal to the value in ST reactor[15-16] and CTF studies[12-13]. Preliminary neutron transport calculations indicate that the divertor plate damage can be reduced well below 1 dpa/FPY (full power year) with the SXD, probably enabling operation for over 2 FPY. With 10 cm of steel shielding, neutron damage to the copper center post can be reduced to ~ 2 dpa/FPY, probably enabling operation for 1-2 FPY before replacement.

## 4. Conclusion

By increasing $R_{div}$, SXD maximizes plasma-divertor separation (both physical and magnetic). This increases plasma-wetted area (line length) by factors of 2-3 (3-5) over and above the best that can be obtained by any other flux expansion method. The SXD-engineered extra gains seem essential for AT reactors. SOLPS simulations show that longer line length decreases the plasma temperature at plates, allowing more radiation in the divertor and less in the core. A small, low-cost, high power density experiment using SXD could demonstrate high integrated performance.

This research was supported in part by USDOE Contracts DE-FG02-04ER54742, DE-FG02-04ER54754, and DE-AC05-00OR22725.

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