The Super X divertor to handle the enormous divertor challenge of next step devices

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Proposed Nest Step devices-probably much more challenging divertor operation than ITER

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Device	Heating Power (MW)	R_{major} (m)	P/R (MW/m)
ITER	120	6.2	19
NHTX	40	1	40
ST-CTF	60	1.2	50
FDF	110	2.5	44
HPDX	120	2.5	48
ARIES-AT	390	5.2	74
ARIES-RS	510	5.4	93
ARIES-ST	620	3.2	195

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In steady state, even ITER's divertor appears problematic

2007 ITER Physics Basis:

"The fusion gain in steady state maximizes at low density for constant β_N. The limitation on reducing the density in next generation tokamaks is set by the impact on the divertor."

"It should be noted that presently developed advanced scenarios have not yet provided fully integrated scenarios and several issues remain to be solved, such as edge compatibility with the divertor"

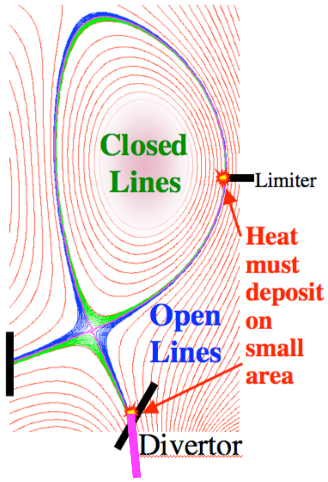
- **So for ITER:**
  - **Advanced scenarios which might lead to high power density are already at or beyond the expected limit of standard divertor heat capacity**
- **Other next step devices also operate in steady state with higher P/R, and so are even more challenged than ITER**

Hence divertor must evolve to proceed beyond ITER

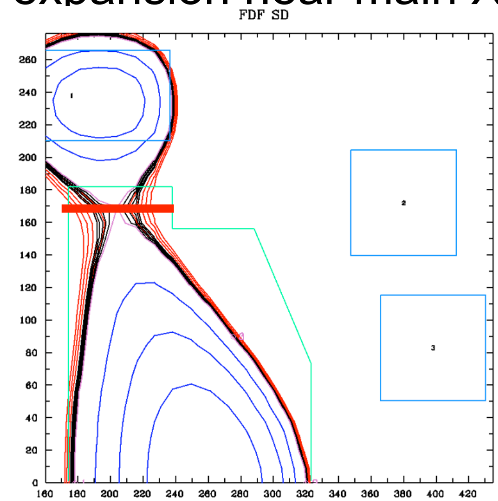
- Three types of problem:
  - High divertor heat flux.
    - **Solution: spread the heat out**
  - High divertor plate temperature.
    - Plate temperatures can easily exceed 100 eV, leading to high sputtering (erosion, dust, plasma impurities, etc.), and low radiation.
    - **Solution: longer line length**
  - Divertor neutron damage.
    - ITER divertor technology: serious degradation at ~ 1 dpa
    - CTF: must test to dozens of dpa
    - Reactor: must run at ~ 100 dpa
    - **Solution: have the divertor plate shielded from the neutrons.**

Limiters to Divertors to X-Divertors to Super-XD

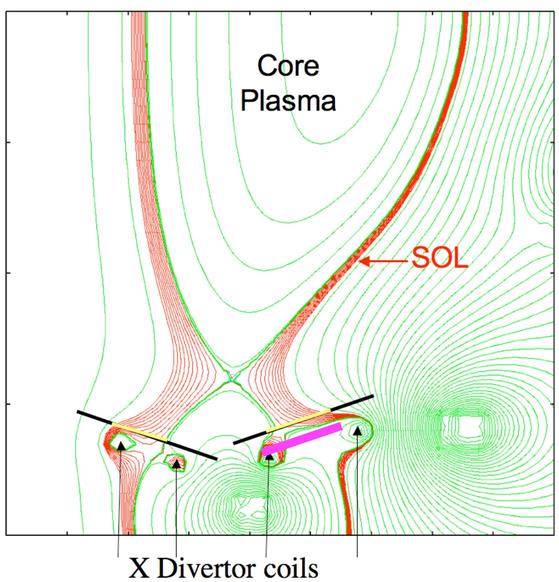
Limiter & Standard Divertor



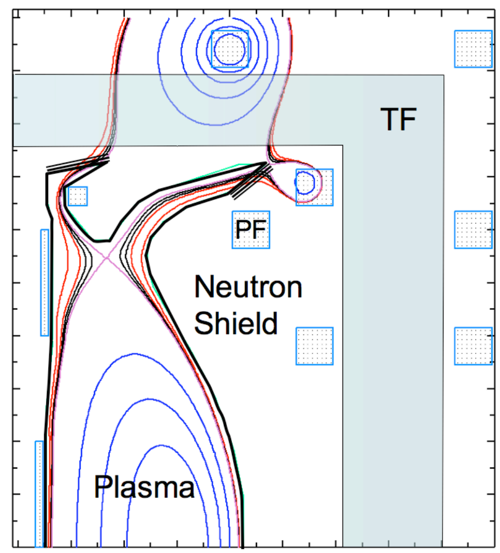
Flux expansion near main X-point



XD/snoflake to expand flux



Super X-Divertor at Large R



***All* flux expanders equally limited by 1 deg tilt limit**

Preview of the talk

- How bad is this problem? - The magnitude of the divertor heat flux problem critically depends on SOL width
  - Review SOL width projections from ITER physics basis for next step devices- much uncertainty
- Are there new physics “insights” which can reduce uncertainty in the SOL width?
  - Recent experimental findings imply a connection between H-mode barrier transport and SOL transport
  - We use these to determine a plausible range of SOL widths for next step devices
  - Basic point: a good H-mode will necessarily have a small SOL width-small SOL width creates divertor challenge- so a solution of the divertor problem is inextricably linked with a high quality edge barrier
- The Super X divertor (SXD) as a solution to these problems
  - Review recent results on application of the SXD to STs
  - New magnetic equilibria
  - 2D SOLPS simulations by John Canik for SXD on NHTX

Part 1 - Projecting SOL width

- Review the methods mentioned in the 2007 ITER physics basis which are complete enough to make projections
- Revisit the 1999 physics basis
- The projections of 2007 basis differ strongly from those of 1999 for ITER and for others!

Part 1 - Projecting SOL width using methods highlighted in the ITER Physics Basis(2007)

- 2007 ITER physics basis gives quantitative estimates of SOL power width λ_q from
  - Projections based on JET data
  - Fluid Modeling (B2-Eirene with $\chi \sim 1\ m^2/sec$)
- It also mentions physics based scalings of λ_q which have been found to fit some datasets well (Connor 1999)
  - JET & C-mod data, dissipative MHD turbulence with a collisional SOL
  - COMPASS data, dissipative MHD turbulence with a collisionless SOL
  - These can give quantitative projections by normalizing to some data- we use the most recent JET data
- The 1999 ITER physics basis highlights empirical scaling laws due to Loarte which are being invoked, for instance, by FDF
  - These λ_q scaling laws are NOT mentioned in the 2007 ITER physics basis

Physics underlying these scalings:

- Projections from JET data (Fundamenski):
  - Empirical scaling of λ_q with P,B,q,n on JET (the machine closest to ITER and some next step devices)
    - With restriction that SOL width must be at least a poloidal gyroradius
  - Scaling most similar to expectations based on classical/neoclassical transport
    - Interpreted to mean that turbulence suppression in the H-mode extends a few mm into the near SOL
  - An R scaling is inferred (both heuristically and from assumption of classical transport scaling)
- B2-Eirene with $\chi = 1$ m^2/sec
  - 2d code community best guess for χ - usually predicts SOL width within a factor of two for standard A
- Connor (1999) derived possible λ_q scalings based on anomalous χ scalings (dimensional analysis from various physics models)
  - tested against data sets from C-mod & JET (collisional) and COMPASS (low collisionality) to find the best model
  - These can give quantitative projections by normalizing to some data- we use the most recent JET data
  - Provides an alternative SOL projection from JET without neoclassical assumption
- 1999 λ_q regression from a data set of ASDEX, DIII-D and JT-60U- two formulas
  - Highly heterogeneous data (operating conditions, degree of detachment, diagnostics..)

Extrapolation results for SOL width-ITER

- **2004 JET extrapolation: 4 mm**
- **B2-Eirene extrapolation: 5 mm**
- **Connor-JET collisional 5 mm**
- **Connor-JET low collisionality 7.5 mm**
- **1999 λ_q regression 14-23 mm**
- The 1999 ITER physics basis formula is a severe outlier
- Though the other methods have significantly different parameter dependencies, their projection for ITER agree remarkably well
  - ~ 5mm is the inferred λ_q in the ITER physics basis
- 2007 ITER physics basis makes a strong point of noting that the ITER divertor works for the more pessimistic λ_q estimates of 4-5 mm
  - Workability should not depend on using only the most optimistic 1999 result

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Applying these methods to other Devices

- Extrapolated λ_q in mm (calculated by assuming 50% core radiation fraction)

Device	JET extrap.	B2 (scaled)	Connor Low, high ν	1999 regression
ITER	4	5	5 7.5	14 23
NHTX	2.3	1	8 3	18 39
ST-CTF	1	1	8 3	19 53
FDF	2	2	6 3	11 32
ARIES AT	2.5	1.5	7 4	16 38

The range of these projections leads one to try to find some new physics insight to reduce the uncertainty

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Comments

- **The extrapolation from JET data and B2 Eirene often give λ_q ~ 1- 2 mm**
- **The collisionality is typically not strongly high or low, so neither Connor formula is obviously inapplicable**
- **The 1999 ITER extrapolation is consistently much larger than any other projection**
  - **NOTE: if the ITER λ_q were as given by the 1999 formula, ITER would have no divertor problems in steady state mode (unlike the statements from the 2007 physics basis)**

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Insights: recent experimental results

- The power comes out in the near SOL, usually within a few mm of the separatrix (outer midplane)
- Recent experimental results: fluctuations a few mm outside the separatrix are nearly indistinguishable from those inside
  - GPI data - turbulence has a scale size larger than the power width
  - Probes see no apparent transition in turbulence at the separatix
- Perhaps, from these findings, a guiding principle seems to emerge :

the transport in the near SOL

is roughly as strong as the transport in the pedestal

- First: we test this hypothesis on the data for λ_q.
- Then: results greatly help in projecting λ_q

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How to quantify this hypothesis and compare with SOL width data?

- Parameterize transport two ways:

1. Assume a diffusion process (i.e., χ) operating in both the SOL and pedestal
   - compare the magnitude of χ in both regions
   - From the experimental SOL width, estimate the corresponding χ
   - Estimate the χ in the pedestal using power balance and experimental data
2. Presume there is a marginal stability process from pressure gradients (as indicated by C-mod results)
   - Estimate dp/dx for the pedestal
   - Estimate dp/dx in the SOL
   - Compare the two

How to determine χ?

- In the pedestal:
  - For χ, use power balance with an empirical estimates of necessary quantities:

Fick's law: χ_{ped} n dT/dr = Power/Surface Area

Estimate: ndT/dr ~ $(nT)_{Ped}/\lambda_{Ped}$, so

Thus: $\chi_{PEDESTAL}$ ~ (Power/Surface Area) / (p_{Ped}/λ_{Ped})

  - In lieu of data use:
    - Pedestal pressure = 1/3 stored energy, estimate stored energy from the H-mode scaling law ITER98H(y,2)
    - Pedestal width ~.03 a for normal aspect ratio, but proportional to A = 1.5
- In the SOL:
  - Determine χ from the SOL width using standard balance of perp and parallel transport
  - If the power is assumed to come across the separatrix via the electrons, balance perpendicular transport with parallel Spitzer conduction (see Stangby)
  - If power is assumed to come across the separatrix via ions, balance perpendicular transport with parallel ion streaming

How to determine dp/dr for the marginal stability ansatz?

- In the pedestal:
  - For χ, use power balance with an empirical estimates of necessary quantities:

    Estimate: $dp/dr \sim (nT)_{Ped}/\lambda_{Ped}$

  - Estimate as before
- Determine nT in the SOL needed to carry away the P_{SOL} by transport along field lines
  - Assume parallel heat transport due to ion streaming if ions carry the heat into the SOL

For the experimental SOL data

Device	$\chi_{electron}$ (m^2/s)	χ_{ion} (m^2/s)	$\chi_{electron}/\chi_{ped}$	χ_{ion}/χ_{ped}	$dp/dr_{SOL}/dp/dr_{ped}$
JET	1-4	0.1 - 0.2	11- 13	0.4 - 1.	0.35 - .8
NSTX	12-27	1.9 - .4	10-28	0.3 - 1.2	.13 - .5

- If energy comes out through electrons, χ would have to increase by an order of magnitude in the near SOL as compared to the pedestal
- If energy comes out through the ions, χ would be about the same in the near SOL as the pedestal

Supports energy transport via through ions

- dp/dr in the SOL is typically ~ 1/2 the value in the pedestal for JET and 1/4 for NSTX

marginal stability possible but with some scatter

Other evidence that power across separatrix is due to perp ion tranport

- Measured values of midplane electron temperature width λ_{Te} are too narrow to be consistent with λ_q - if electron conduction dominates

Since $\chi_{||} \sim T^{7/2}$, $\lambda_q \sim (2/7)\ \lambda_T$

But experimentally, $\lambda_q \sim \lambda_T$

- This can be explained if ion perpendicular transport carries heat across separatix:
  - SOL electrons are only heated by equilibration with the hotter ions
  - most of the parallel heat conduction is via ion streaming
- Analysis shows: $\lambda_{Te} \sim \lambda_q$ ($\sim \lambda_{Ti}$), like experiments
  - Full verification requires ion SOL temperature measurements

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Relationship between pedestal and SOL transport: a tool for discrimination

- Values of λ_q which are very small require values of χ_{SOL}/χ_{ped} that are far below 1. It seems unlikely that
  - the SOL transport is much less than the pedestal transport (one generally expects the reverse)
  - the SOL stability (critical value of $dp/dr_{SOL)}$) is much better than the pedestal- again, one would expect the reverse.

- Values of λ_q which are very large require values of χ_{SOL}/χ_{ped} that are much greater than 1.
  - A large χ_{SOL} is likely to "leak" into the pedestal, resulting in ruined H-mode transport
  - It is well known that it is possible to operate with superlative power exhaust at the expense of severely degrading the H-mode pedestal

- We can consider that plausible values of λ_q correspond to the experimental range of χ_{SOL}/χ_{ped} or dp/dr_{SOL} / dp/dr_{ped}

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The “plausible” range of λ_q values for next step experiments

- λ_q in mm:

Device	χ_{SOL} method	dp/dr_{SOL} method
ITER	3.1 - 5.4	3.2 - 4.3
NHTX	4.1 - 9.4	4.1 - 9.9
ST-CTF	2.7 - 6.2	1.8 - 4.4
FDF	5.2 - 9	3.0 - 4.9
ARIES AT	5.0 - 8.7	3.0 - 5.0

- Within the range of uncertainty, $\lambda_q \sim 5$ mm, like ITER, for next step devices
- Hence P/R is a reasonable measure of divertor challenge
- Since ITER is very worried about divertor operation in their steady state scenarios, such operation on other next step devices requires a substantially better divertor

Further Comments

- Extrapolations with $\lambda_q \sim$ 1-2 mm appear implausibly pessimistic
- Extrapolations with λ_q >10 mm appear implausibly optimistic (1999 regression not mentioned in 2007 physics basis)
- The ITER physics basis notes that divertor operation seems feasible for projections at the plausibly pessimistic end of the range- which appears prudent
- A "prudently pessimistic" value to assume for most next step machines is 2-4 mm, depending on the machine
- This would lead to many times higher parallel heat fluxes than ITER for next step reactors

How feasible is high divertor radiation?

- Divertor radiation fractions decrease with increasing $Q_{||}$, shorter line lengths, lower density
  - Results from UEDGE, 1d models, elementary physics considerations, etc.
  - Maximum ITER SOL radiation fraction ~ 60% (extensive community analysis, similar to present experiments)
- Compared to ITER, next step devices have:
  - Substantially higher $Q_{||}$
  - Substantially lower line length
  - Density about the same
- Hence, elementary considerations imply that radiative divertor solutions, alone, are unlikely to solve the heat flux problem on next step devices

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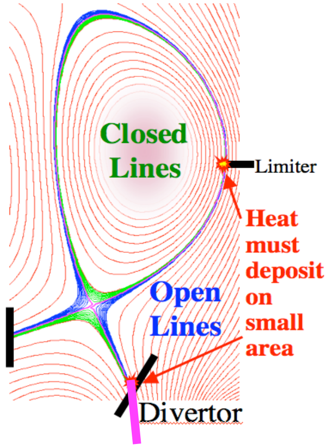
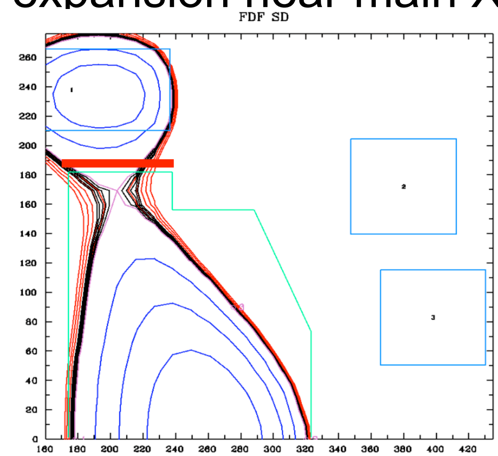
Let us now turn to a magnetic geometry which can handle these enormous parallel heat fluxes

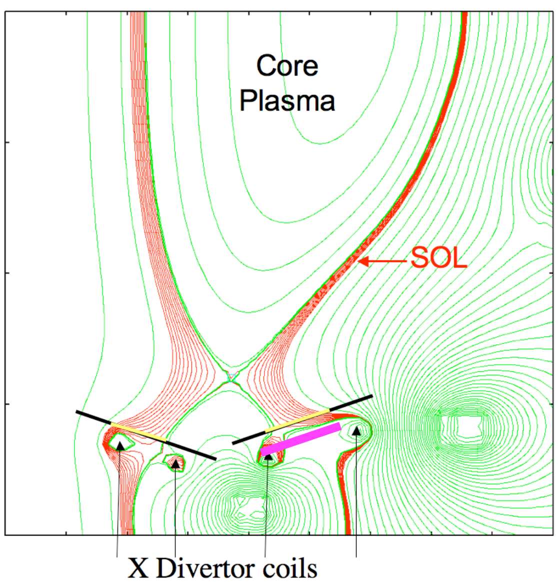
Limiter & Standard Divertor



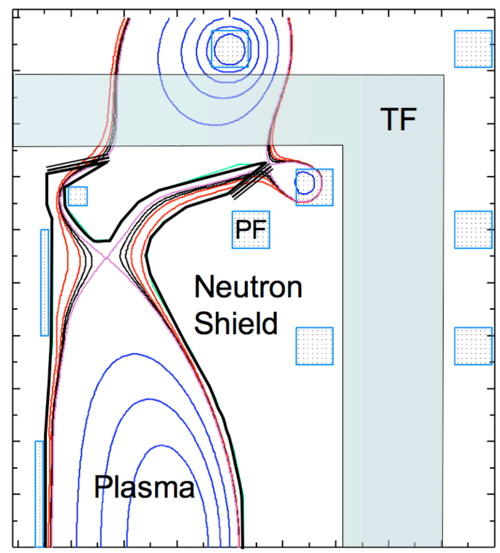
Flux expansion near main X-point



XD/snoflake to expand flux



Super X-Divertor at Large R



***All* flux expanders equally limited by 1 deg tilt limit**

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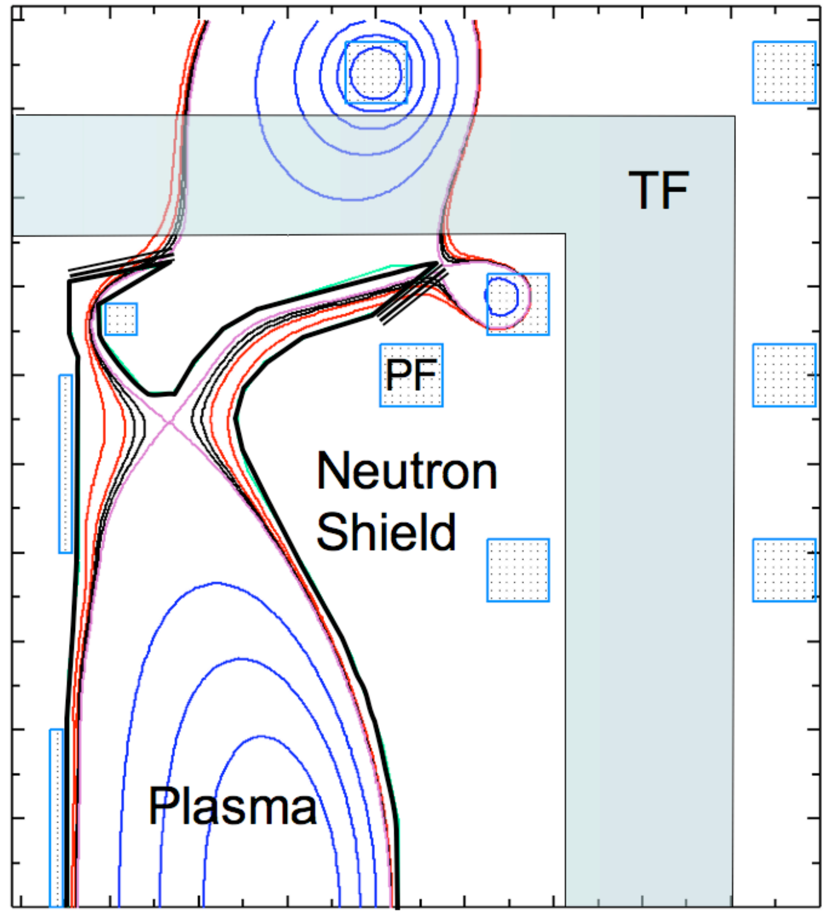
1 deg limit => SXD is the *only* way to increase A_w

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

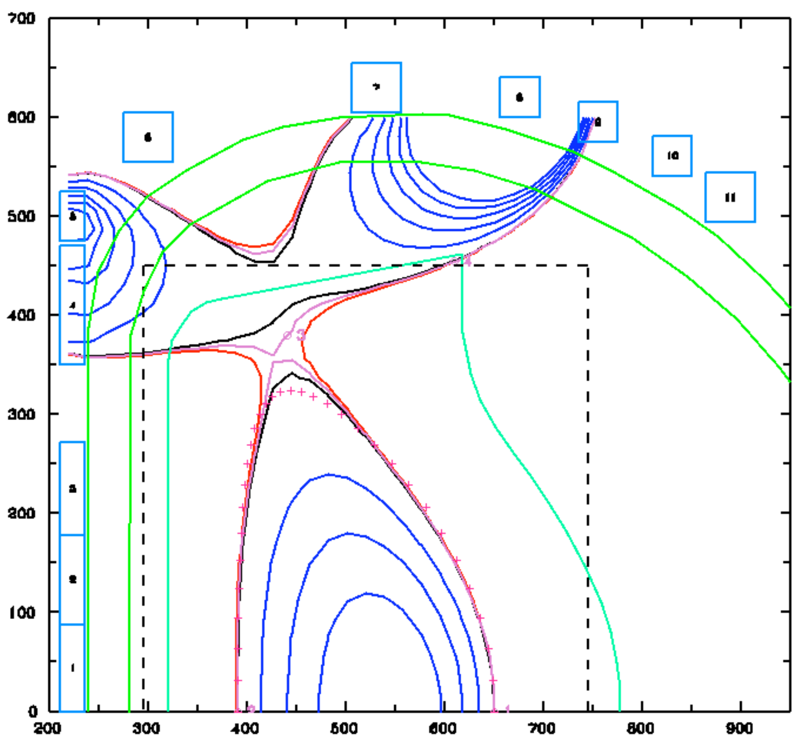
- Angle between total B and plate *must be* more than 1 degree, so
- Flux expansion gains via any route (tilting plate, XD, snowflake--) are **equally** limited
- SOL width (so A_{sol}) is a given by upstream physics
- So *only* knob left is R_{div}/R_{sol} --- maximization of which is a crucial SXD strategy. Direct gain of ~2 in A_w

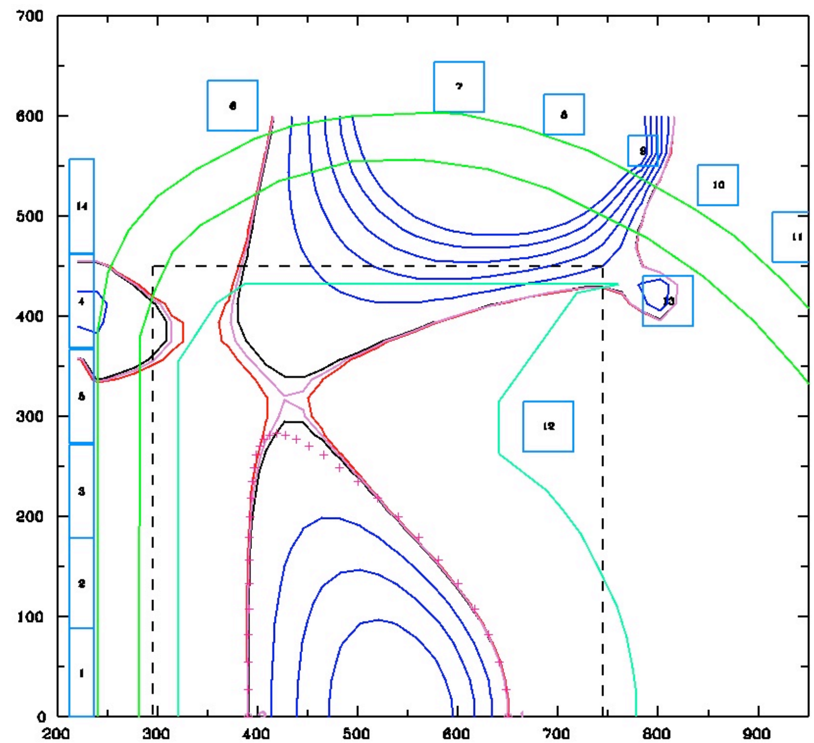


HPDX - CORSICA Equilibrium

SXD for Superconducting ARIES-AT Reactor

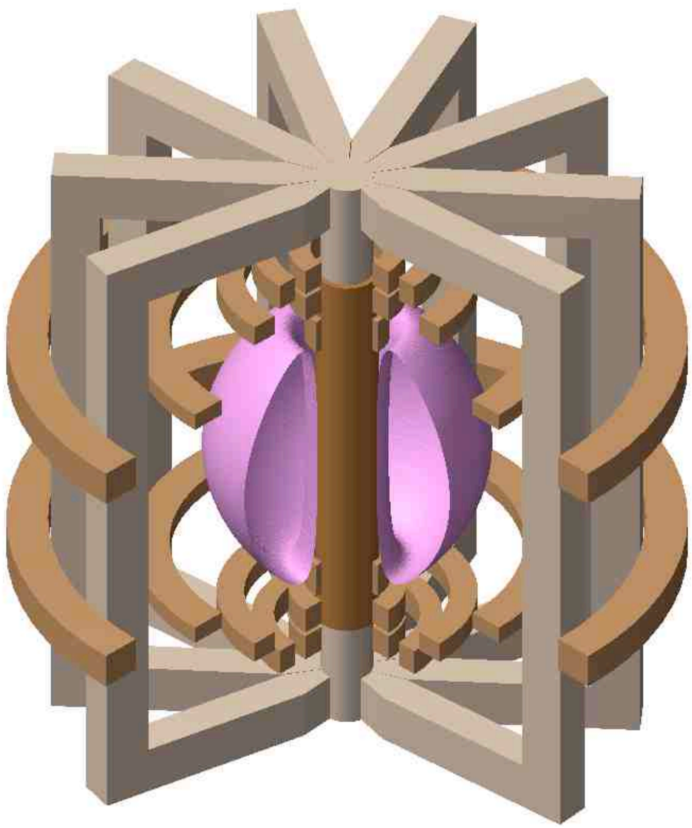
- SXD has been implemented within existing TF coils of SC reactors:
  1. Either with all axisymmetric PF coils outside TF coils (left fig.), or
  2. Or with modular coils not linked with TF coils (right fig.)
- No extra TF "real estate" is needed even for existing reactor designs
- The net MA-m in PF coils and their locations are similar to the standard divertor case



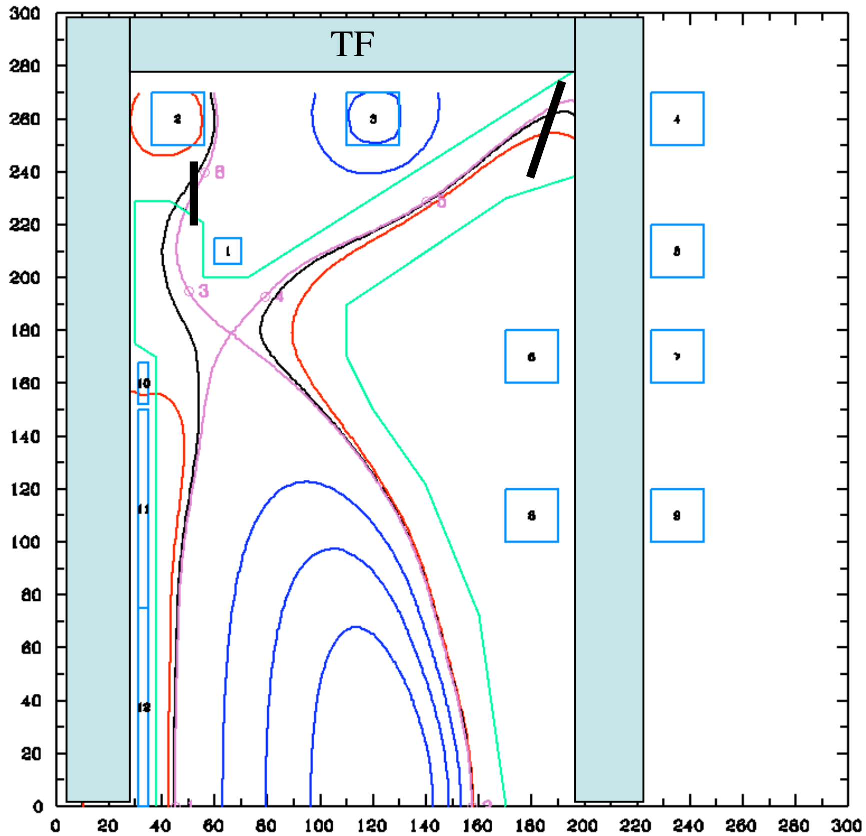


SXD fits in TF corners - no TF real estate issues

- For NHTX, FDF, and Reactors the Super-XD does not require larger TF coils
- SXD uses available space (in the corner of TF coils) which is normally unused
- FDF, ARIES RS, ARIES AT, and ARIES ST are similar in this respect
- SXD coils & currents very similar to NHTX coils with standard divertor



NHTX (PPPL/ORNL)



A CORSICA Equilibrium for NHTX-SX

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Other advantages of the SXD strategy

- Spectacular Increase in Line Length - A significantly lowering of B_{pol} in the long leg => increase in line length by up to 5x. Long line length leads to:
  - A strong lowering of plasma temperature at plate from ~ 100 eV to ~ 10 eV
  - this greatly lowers high Z impurities, which could be crucial for a tungsten divertor plate and an AT mode
  - A jump in divertor radiation fraction from low Z impurites- from insignificant to substantial- 10-15% to > 50%.

  A result found in calculations using an elaborate 1D model. 2-D runs (SOLPS) are underway to verify these advantage (IFS collaboration with ORNL/PPPL)

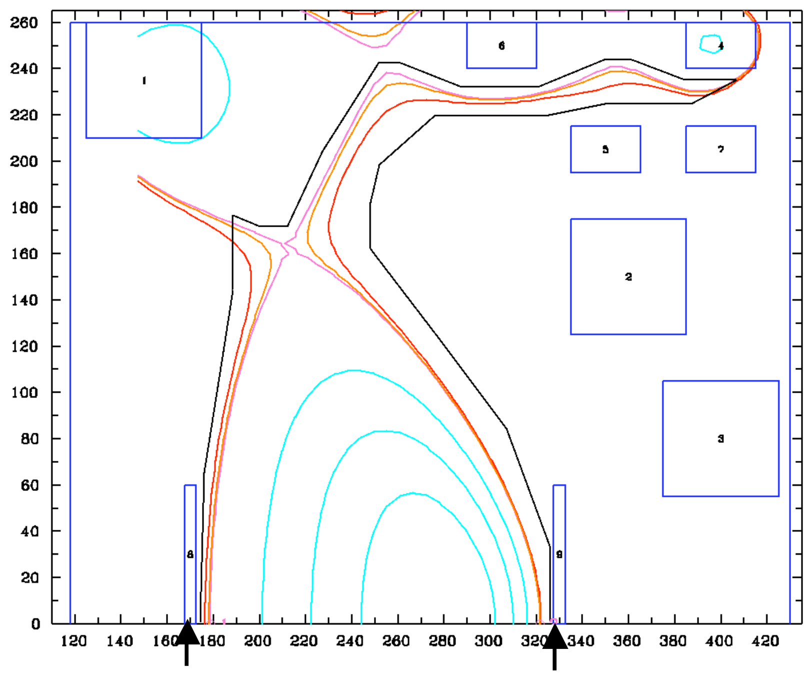
  - A widening of SOL width - longer line allows more time perpendicular diffusion to spread heat
- **Direct A_w gain, widening of SOL, and enhanced radiation working in unison boosts up the maximum tolerable P_{SOL} by a factor over 5**
- **Decreases need to radiate power from core: allows better core performance-for optimistic λ_q "no" core radiation is possible**

SXD: Easy and Robust

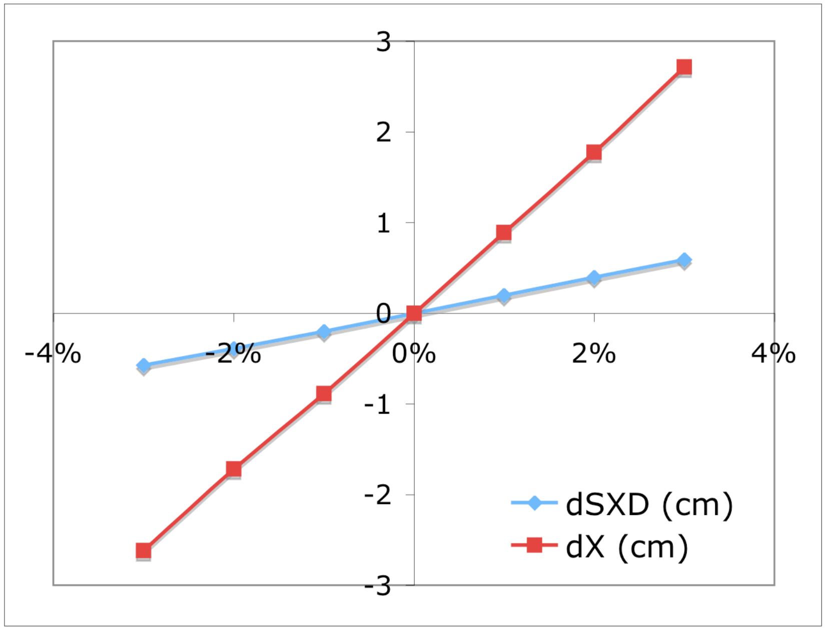
- **Surprisingly, the SXD can be implemented by just moving the PF coils around a bit**
  - Coil currents & locations are not very different from standard divertor case. This is so for a variety of machines that we have investigated
  - Increased distance from plasma isolates SXD from plasma changes. By the same token, the main plasma is also somewhat immune to what may be happening in the vicinity of the plate. One may, for instance, be able to operate in a fully detached mode without damaging the main plasma.
  - The relative isolation makes SXD strike point insensitive to plasma fluctuations- we have tested this in a variety of studies.

SXD is *very insensitive* to plasma changes

- In general (for NHTX, FDF …), SXD strike point, wet area, line length, B line angle, ALL are insensitive to sudden changes in plasma current
- Possible reason: plasma is far, while SXD coils are near the SXD plate
- Preliminary snowflake studies (NHTX case) show greater sensitivity
  - higher-order main X point near plasma easier to perturb?
- Simulated by adding two "wall simulator coils" & fixing all others
- Vary I_{plas}, R_0, a etc. by ±3% each and record main X and SXD shifts



FDF 7L0 with "wall coils"



Main X & SXD Shift (cm) vs dI_{plas} ±3%

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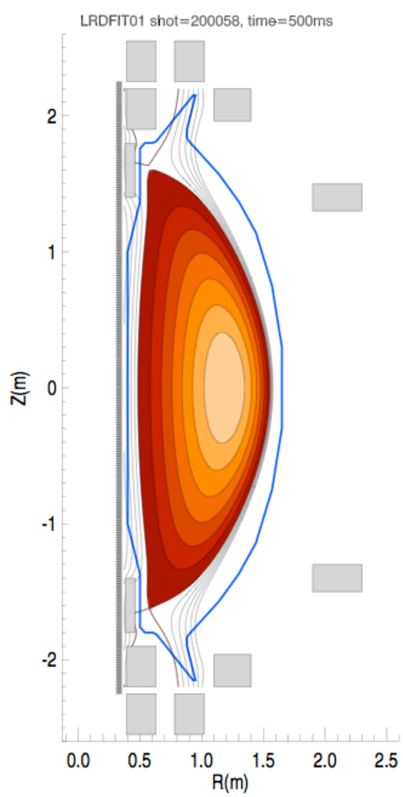
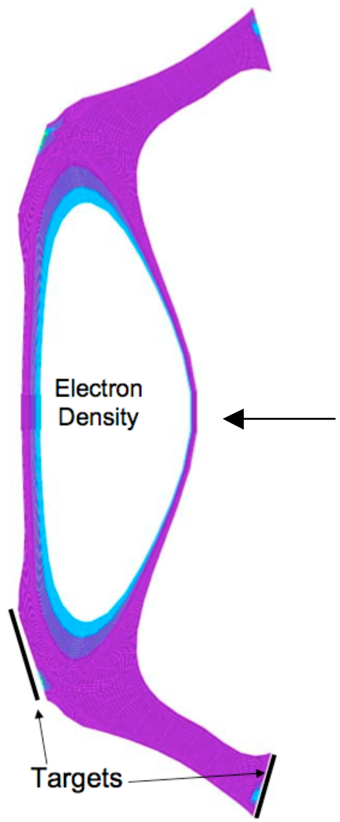
Example: SXD can save NHTX from heat flux menace

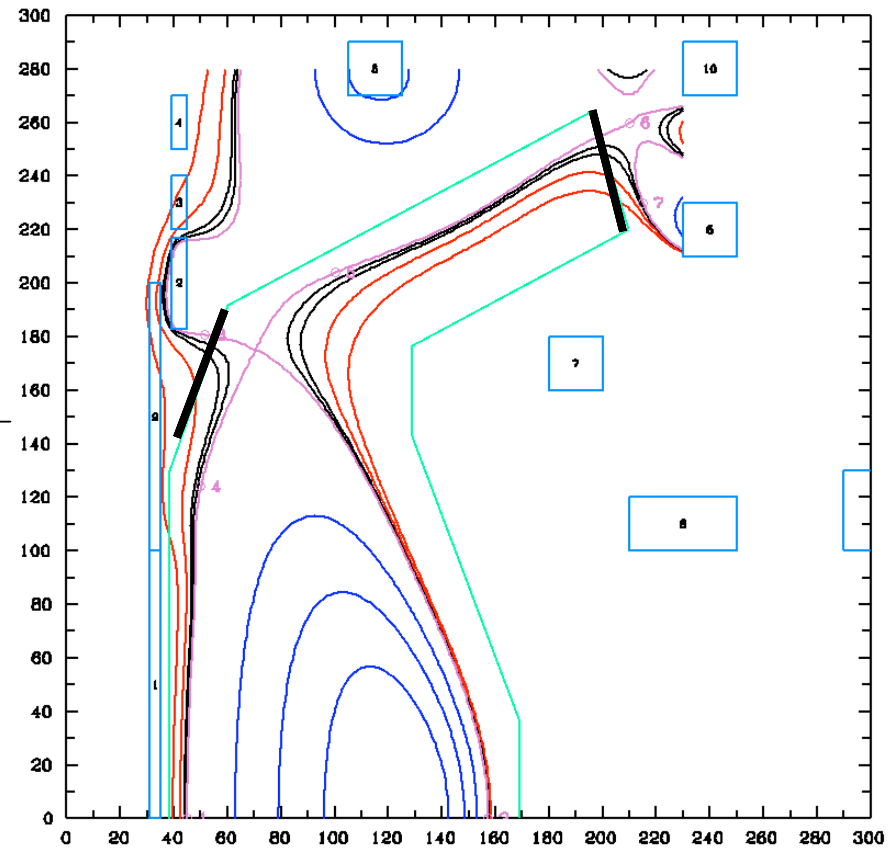
- **With SXD & 30 MW, peak heat flux can be kept under 10 MW/m^2**
- **Not possible with standard divertor (peak stays at 30-40 MW/m^2)**
- **SOLPS 2-D calculations (Canik & Maingi) confirm 1-D code expectations**



NHTX Standard Divertor



SOLPS SXD Calculation



NHTX SXD (A Corsica Equilibrium)

SXD: Surviving Disruptions & ELMs

- **SXD can improve survivability to disruptions or ELMs:**
  - Heat flux is spread over a larger area further from plasma
  - Ions travel a much longer distance, so heat pulse could also be spread out significantly in time (material damage ~ $1/\text{time}^{1/2}$)
  - The divertor plate is not in the way of halo currents from a VDE
    - Wall can probably be made to be a more mechanically robust structure than a divertor

Neutron damage to divertor - critical issue

- **Tungsten “armor” on a high thermal conductivity actively cooled substrate**
  - High conductivity substrates (Cu or C) severely deteriorate after only a few dpa
  - Reactor walls must tolerate ~ 100 dpa (but at heat flux much less than divertor)
  - Promising main chamber wall materials must be tested at ~100 dpa
- **Only hypothetical *high heat flux divertor materials* might tolerate ~ 100 dpa**
  - Decades away with much material development effort in the EU and Japan
  - The US virtually does not have a fusion material development program anymore
  - Slow development would hamstring any high duty cycle DT device (CTF, DEMO)
  - Cannot credibly field a high duty cycle DT device without a divertor with a high chance of survival under copious fusion neutron *and* SOL heat fluxes.
- **SXD: substantial shielding of divertor plates for future CTF, DEMO**
  - With SXD, ITER divertor technology may well suffice for high duty cycle DT
- **This alone may make SXD essential for next generation fusion devices**

Summary

SXD simultaneously and uniquely:

- Spreads heat over ~ 2-2.5 times more area than any other concept (within an engineering constraint for B angle with the plate)
- Greatly increases the line length beyond other concepts, lowering plate temperatures
  - Plate temperatures ~ 100 eV with other schemes could be a show stopper for long duty cycle operation due to high Z impurity generation, plate lifetime, dust generation
- Shields the divertor hardware from neutrons
  - Lifetime of divertor hardware in dpa could be much less than the first wall
- Improves disruption or ELM survival
  - A critical issue as device stored energy becomes large
- Offers robust operation to plasma perturbations
- The SXD is synergistic with liquid divertor options
- **We believe the SXD essential for next generation fusion devices**

The SXD on NSTX

- We are working to make it possible to implement the SXD concept on NSTX within budget and manpower constraints
  - Same vacuum vessel
  - Minimal (perhaps no) outer PF coil modifications
- By experimentally testing the SXD, NSTX can demonstrate a crucial component needed to go to the next steps in the fusion program, for both STs and for normal A tokamaks