Advanced divertor magnetic configurations for tokamaks: concepts, status, future.

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- Dale Meade, Princeton, NJ

This talk to discuss poloidal magnetic divertor configurations for tokamaks

Outline

- Introduction
  - Poloidal magnetic divertor configuration
  - Perspectives on advanced magnetic divertor configurations: physics, engineering, history
- Status of experiments
- Research plans

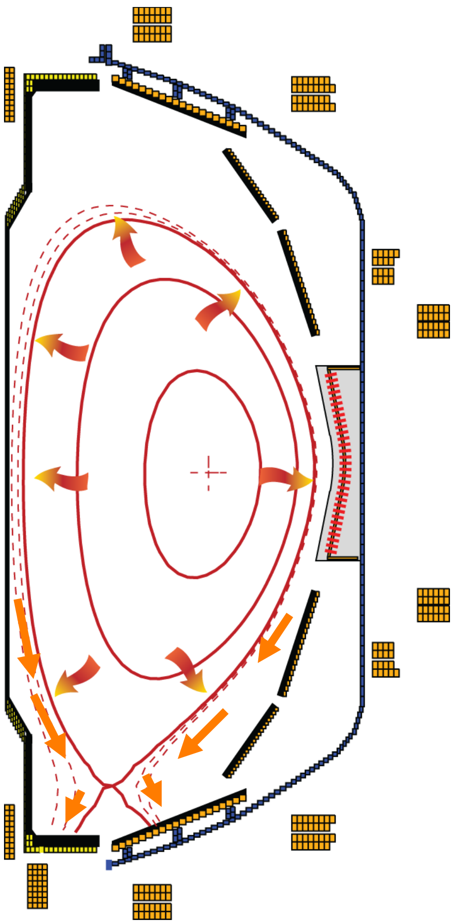
Other Plasma-Material Interface areas are as important, however, not discussed in this talk

- Core and pedestal integration
- Plasma facing components
  - Divertor target geometry
  - Continuously moving divertor plates
  - Liquid metal divertors
  - Pebble divertors
  - Solid divertor targets with active cooling
- Operating scenarios
  - Particle control with cryo
  - Radiative regimes
  - Ergodic divertors
  - 3D fields
- Numerical plasma and material models

Poloidal X-point divertor enabled progress in tokamak physics studies in the last 30 years

- Critical divertor tasks
  - Power exhaust
  - D/T and He pumping
  - Impurity source reduction
  - Impurity screening

Advanced magnetic configurations: potential to perform the divertor tasks better than the standard X-point divertor



National Spherical Torus Experiment, Princeton Plasma Physics Lab

C. S. Pitcher and P. Stangeby, PPCF 39, 779 (1997)

Significant gaps exist between present divertor solutions and future device requirements

- P_{heat}/R
  - Present experiments: ≤ 14 MW/m
  - ITER: ≤ 20 MW/m DEMO: 80-100 MW/m
  - Proposed solution: radiate up to 80-90%
- Steady-state heat flux
  - Technological limit q_{peak} ≤ 5-15 MW/m^2
  - ITER: q_{peak} ≤ 10 MW/m^2 (Mitigated)
  - DEMO: q_{peak} ≤ 150 MW/m^2 (Unmitigated)
- ELM energy, target peak temperature
  - Melting limit 0.1-0.5 MJ/m^2
  - DEMO: Unmitigated, ≥ 10 MJ/m^2
- Impurity erosion (divertor target plasma temperature)

Greenwald report, Toroidal Alternate Panel Report, ReNeW IAEA DEMO Workshops

Advanced magnetic divertor configurations can improve standard divertor properties and performance

Radiated power loss Increase via V_{div}, $L_{||}$

Poloidal Target Inclination, etc

$$q_{peak} \simeq \frac{P_{div}}{A_{wet}} = \frac{P_{div}(1 - f_{rad}) f_{geo}}{\mathbf{N}\; 2\pi R_{SP} f_{exp} \lambda_{q_{||}}}$$

Increase plasma-wetted area

Number of Divertors / legs

Increase divertor area at large R_{SP}

Increase λ_q via increased radial transport, $L_{||}$

Increase plasma-wetted area via increasing f_{exp}

- **Divertor physics is inherently 2D or even 3D**
  - Parallel / cross-field transport and turbulence
  - Radiation front (detachment) stability
  - Neutral pressure / density distribution

Engineering and technology aspects define divertor configuration options

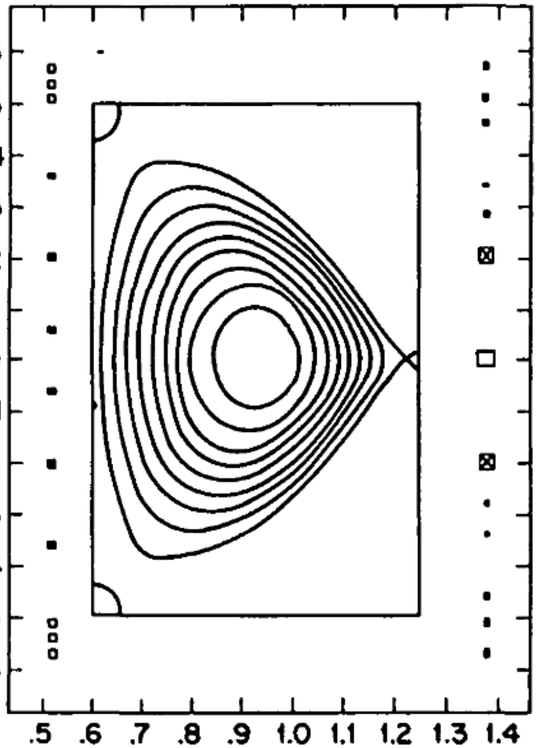
- Plasma equilibria, shaping and control
- Magnetic coils – inside or outside TF magnet
  - Neutron shielding
  - Cooling
  - Electromagnetic forces
  - Maintenance and remote handling

Advanced divertor magnetic configurations (classified by appearance)

1. **Multiple divertors, each with one X-point**
2. **Higher order (2^{nd}, 3^{rd}) null divertors**
3. **Divertor with multiple X-points**
4. **Long-legged divertors with multiple X-points**

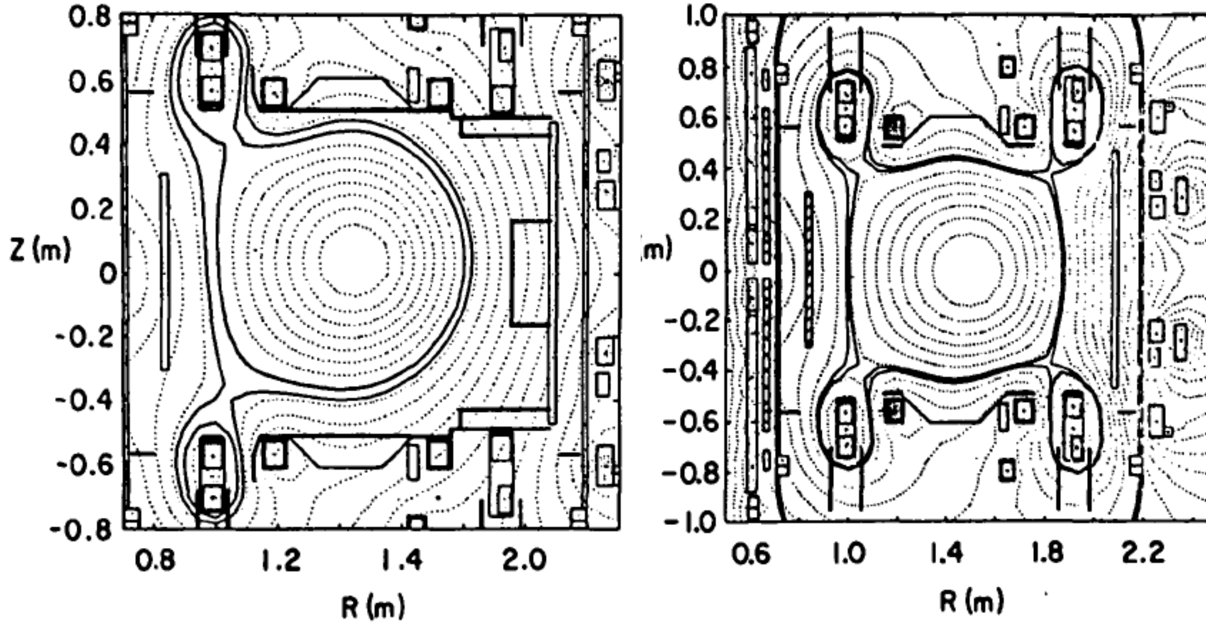
- Note on early concepts
  - Envisioned before H-mode discovery (1982)
  - Some concepts envisioned divertor for particle and impurity control **only**

1. Multiple divertors: triple-null, quadruple-null to share heat and particle fluxes



- D-shaped plasma, high triangularity
- Increased local shear
- Enhanced kink stbility

J. Kesner, Nucl. Fusion 30, 548 (1990)

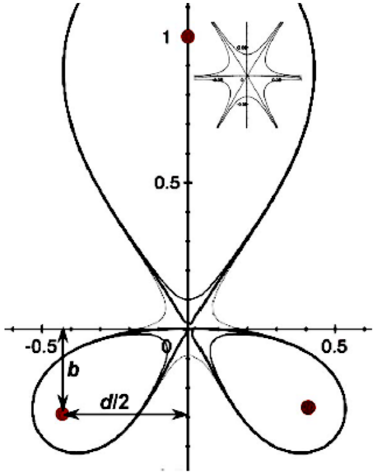


- Poloidal Divertor Experiment tokamak at PPPL (1979 – 1983)
- Significant contribution to divertor physics with double-null configuration

K. Bol et.al, Nucl. Fusion 25, 1149 (1985)

2. Higher order null divertors : larger region of very low B_p to affect geometry and transport

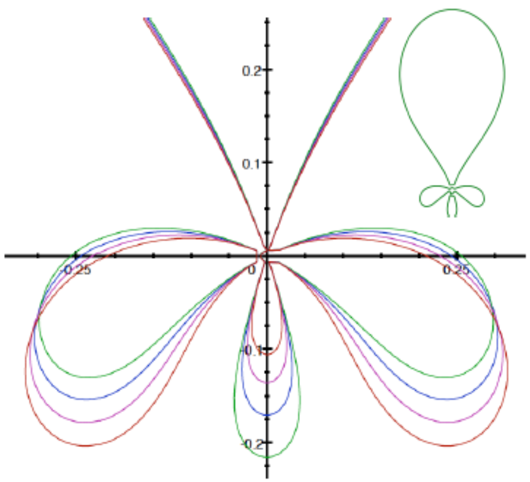
- Snowflake, 2nd order null
  - $B_p \sim 0$, *grad* $B_p \sim 0$ (Cf. first-order null: $B_p \sim 0$)
  - $B_p(r) \sim r^2$ (Cf. first-order null: $B_p \sim r$)
  - Four divertor legs



D. D. Ryutov, Phys. Plasmas 14 (2007), 064502

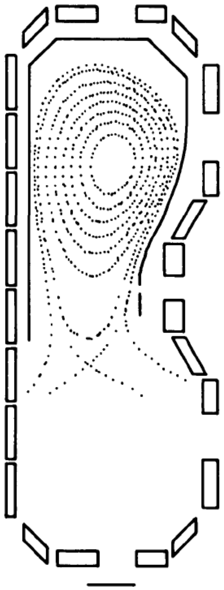
- Cloverleaf, 3rd order null
  - $B_p(r) \sim r^3$
  - Six divertor legs

- Strong plasma convection



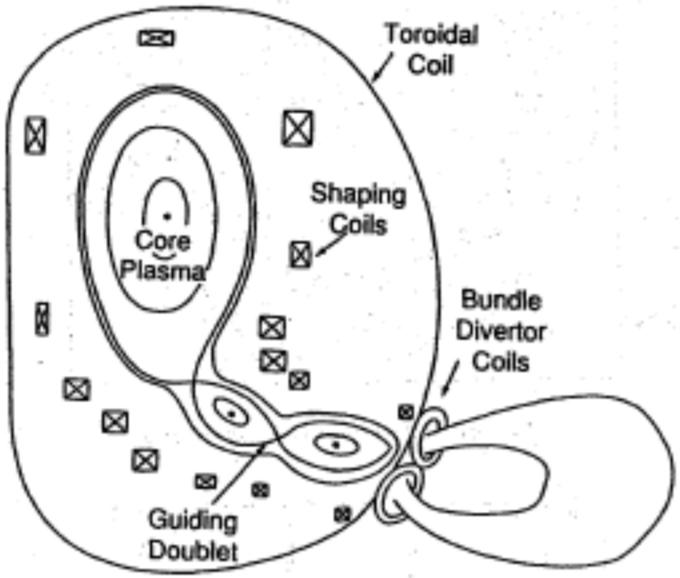
D. D. Ryutov et. al, Phys. Plasmas 20 (2013), 092509

3. Divertor with multiple X-points: expand SOL in the divertor region



N. Ohyabu et. al, Nucl. Fusion 5 519 (1981)

- Doublet III Expanded Boundary



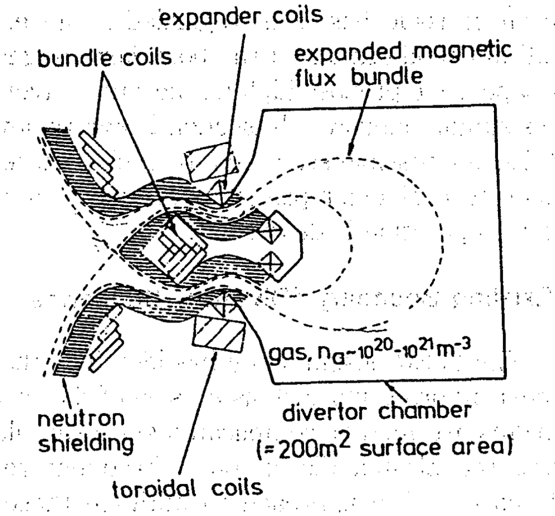
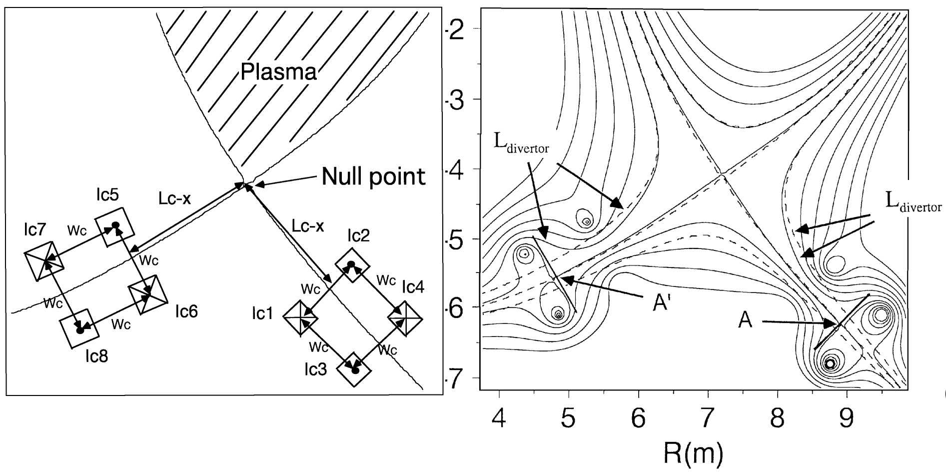


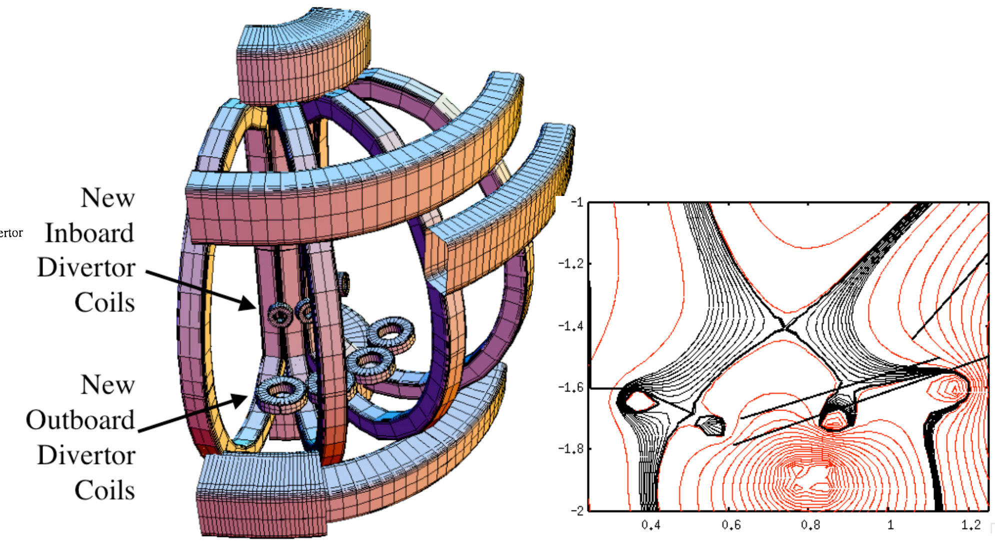
Fig. 3. Bundle divertor proposed for INTOR (M. F. A. Harrison *et al.*: J. Nucl. Mater. 93&94 (1980) 454).

N. Ohyabu, J. Plasma Fus. Res. 5 525 (1991)

- Poloidal Bundle + Expanded Boundary
- Analysis of radiation, H-mode compatibility, neutron shielding, coil currents

3. Divertor with multiple X-points: expand SOL in the divertor region





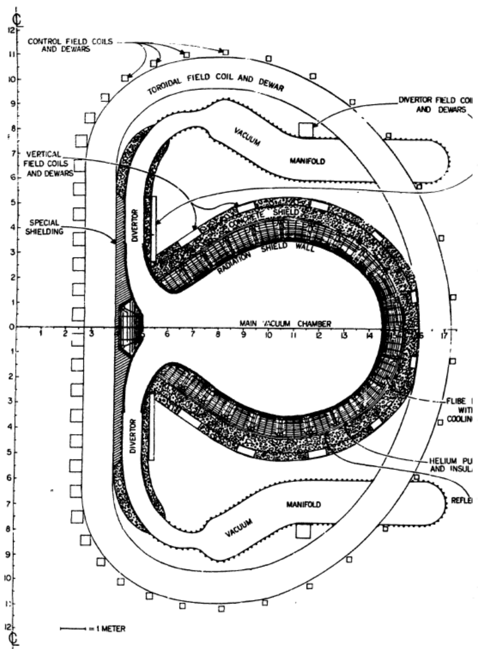
H. Takase, J. Phys. Soc. J. 70, 609, 2001

M. Kotschenreuther et. al, IAEA FEC 2004; Phys. Plasmas 14, 072502 (2007)

- Cusp-like divertor configuration
  - Coil currents acceptable for ITER-like parameters

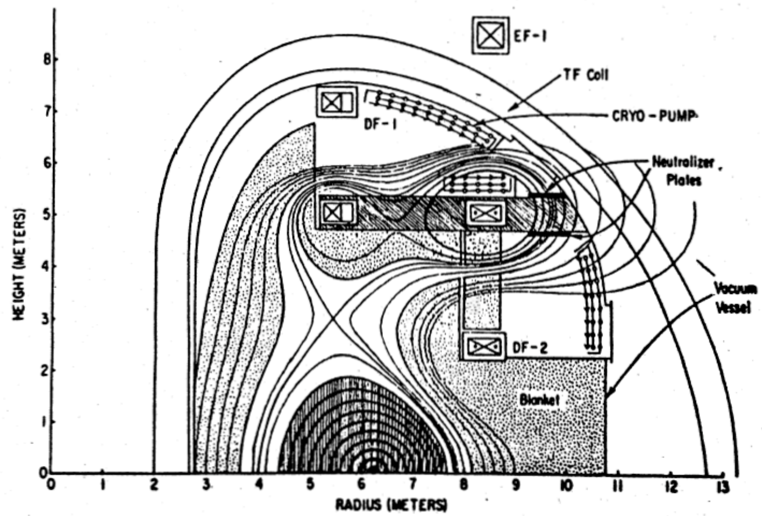
- X-Divertor
  - Small dipole coils under each divertor leg, inside the TF
  - Potential to stabilize rad. front

4. Long-legged divertors with multiple X-points: increase connection length, expand SOL



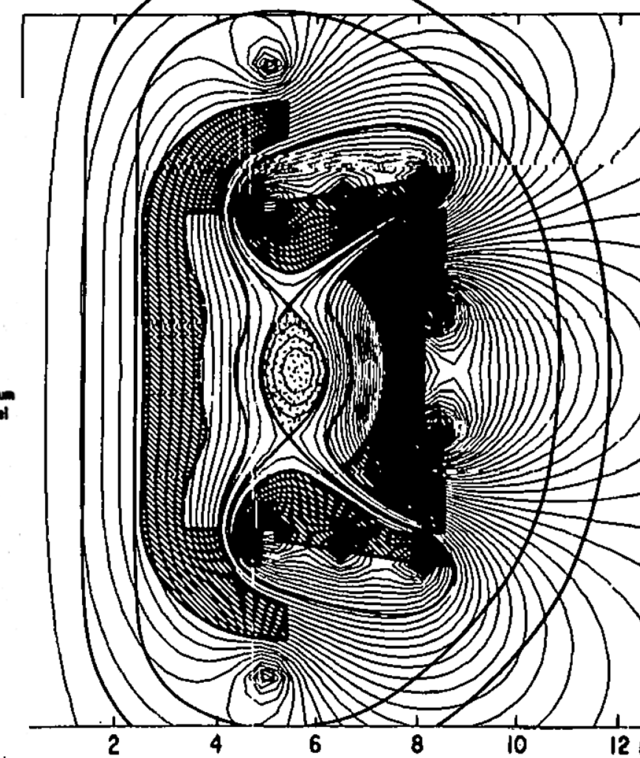
F. H. Tenney et.al,
J. Nucl. Mater. 53 43 (1974)

Conceptual divertor design for Princeton Reference Design Reactor.



A.V. Georgievsky et.al, 6th Symp Eng Prob of Fus Energy, p 583, 1975, IEEE 75CH1097-5-NPS, Copyright 1976

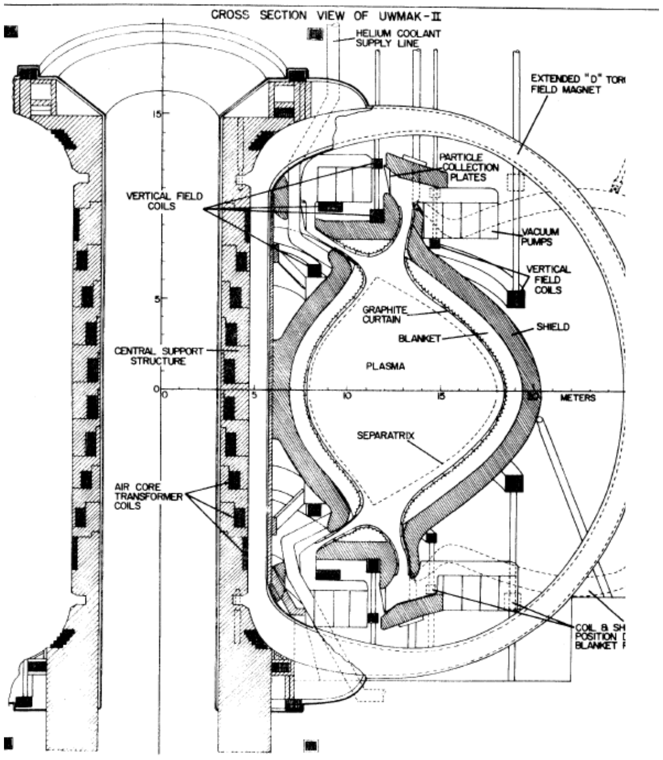
Long-legged high flux expansion poloidal divertor

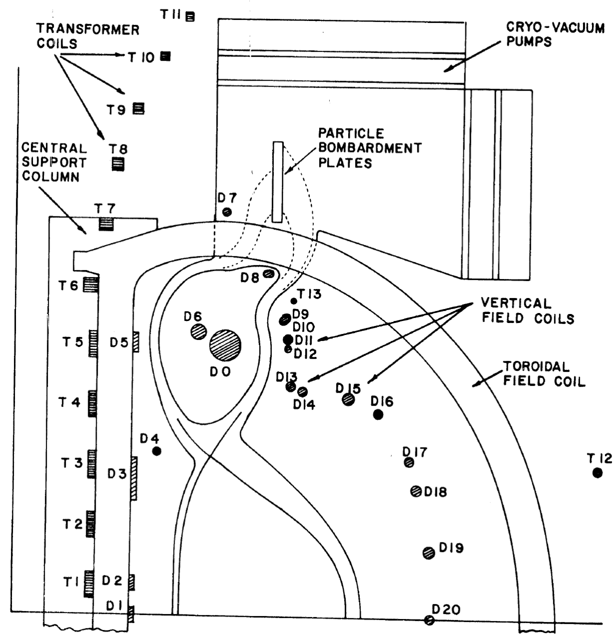


F. H. Tenney,
PPPL Report 1284, 1976

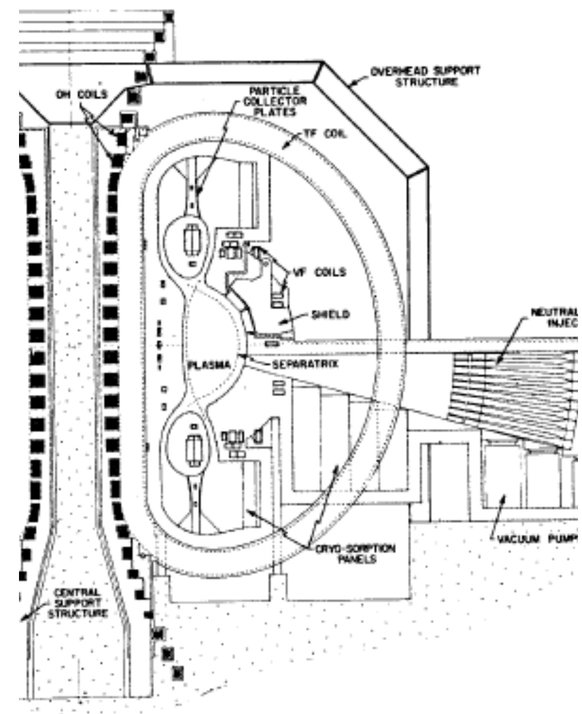
Long-legged double null poloidal divertor

Long-legged divertors with multiple X-points





CROSS SECTION VIEW OF TETR



R. W. Conn et.al, U. Wisc. Report UWFDM-114, 1974

B. Badger et.al, U. Wisc. Report UWFDM-150, 1975

G. L. Kulcinski et.al, U. Wisc. Report UWFDM-173, 1976

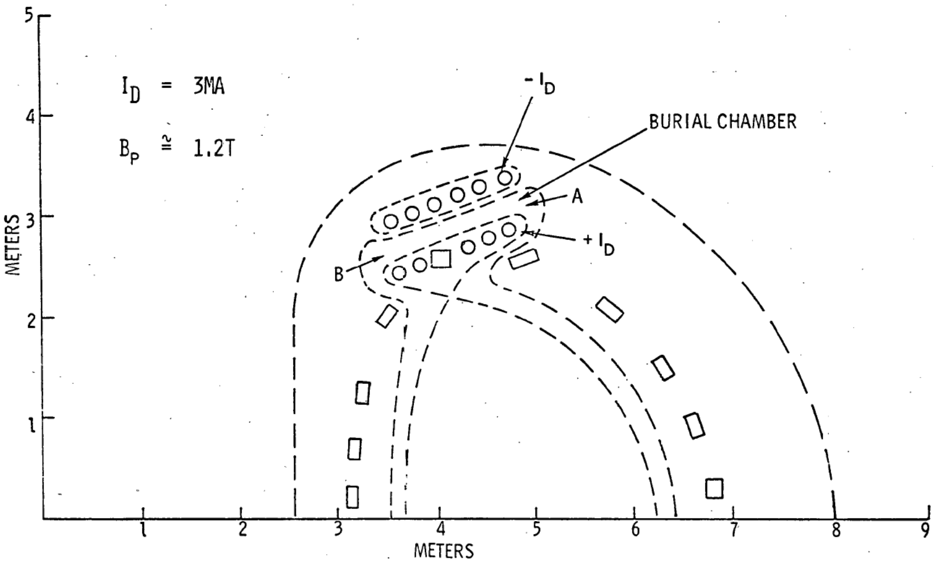
UWMAK-II

UWMAK-III

Tokamak Enginering Test Reactor

UW Fusion Technology Institute conceptual reactor systems studies

4. Long-legged divertors with multiple X-points

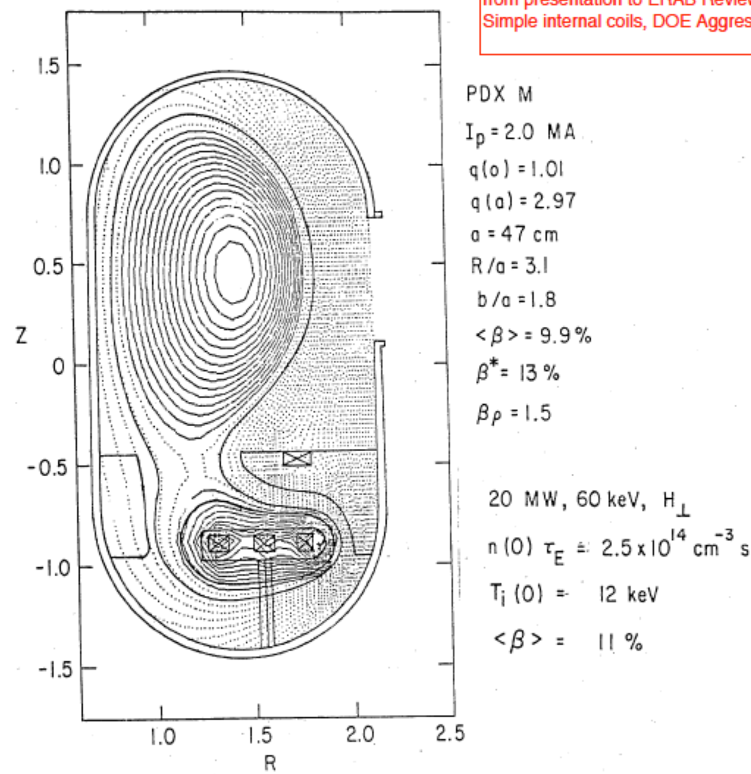


T. F. Yang et.al, Westinghouse Corp. WFPS-TME-055 1977

TNS reactor (w/ ORNL)

- Divertor heat flux 1-3 MW/m^2
- Flowing liquid lithium targets for heat and particle removal

Nov 1978 -Original Proposal for Modification of PDX from presentation to ERAB Review Comittee Simple internal coils, DOE Aggressive Proposals

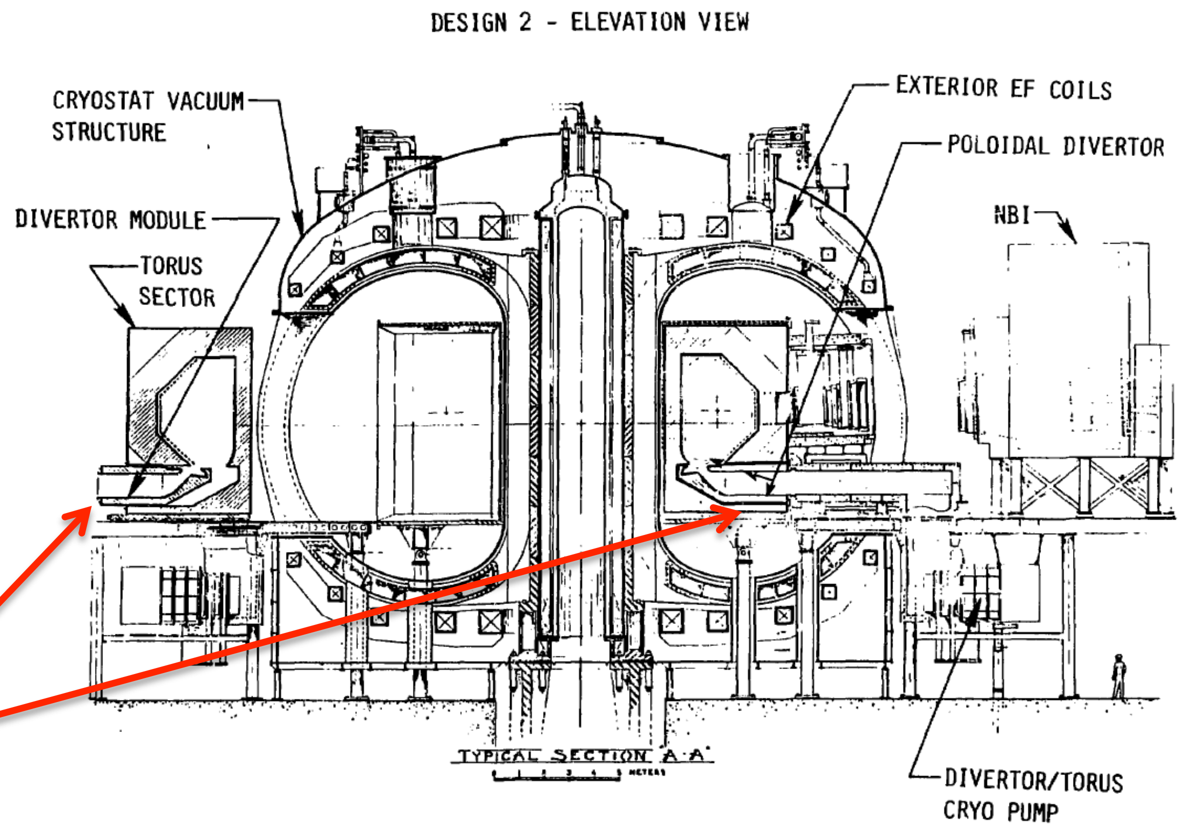


D. Meade, Private Communication

PDX Modification proposal

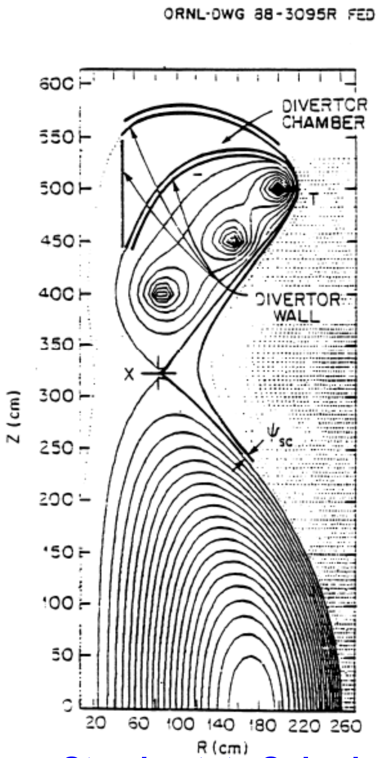
4. Long-legged divertors remove interaction zone away from plasma

- Conceptual design Engineering Test Facility
- R=5.6 m
- a=1.3 m
- B_t=5.5 T
- I_p=6.1 MA
- P_{NBI}=60 MW
- Poloidal divertor
  - W target plates
  - Cryo-panels for particle control

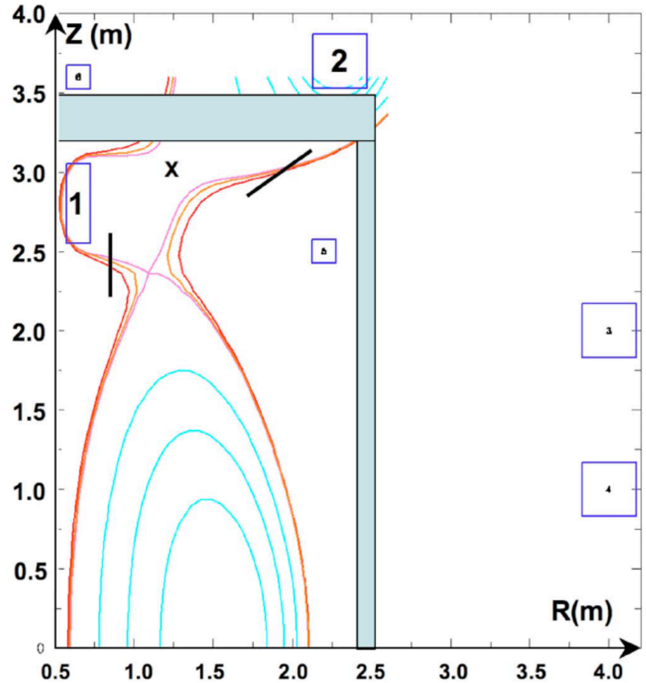


P. H. Sager et. al, J. Vac. Sci. Tech. 18, 1081 (1981)

4. Long-legged divertors with multiple X-points

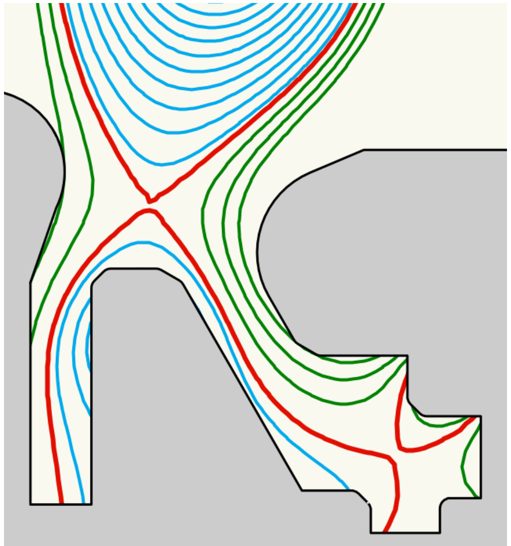


M. Peng, Steady-state Spherical tokamak TST, Workshop on Edge Plasma for BPX and ITER, 1991



P. M. Valanju et.al, Phys. Plasmas 16, 056110 (2009)

Super-X divertor



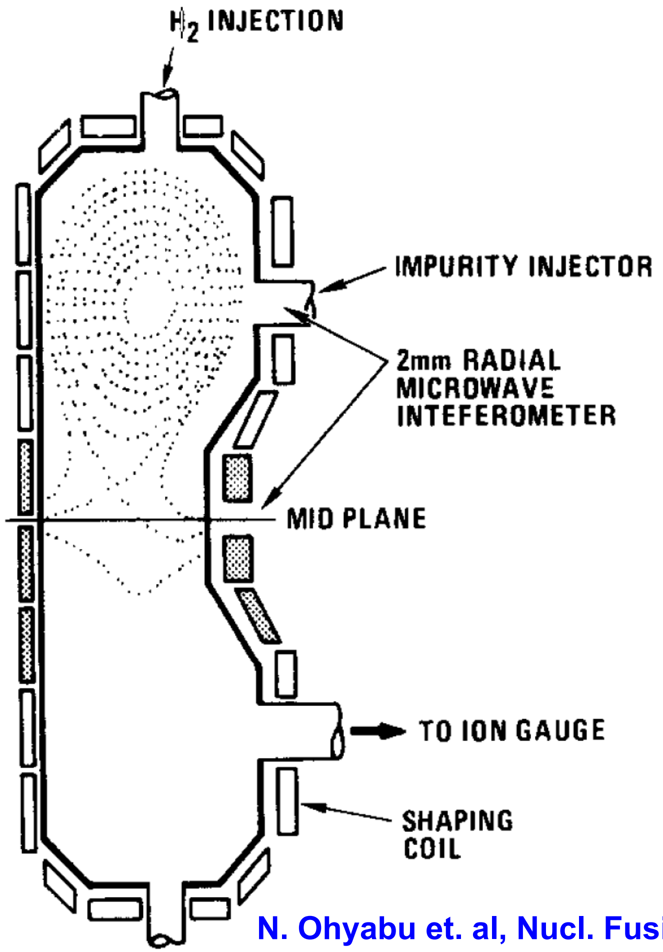
B. LaBombard et.al, APS 2013, IAEA 2014

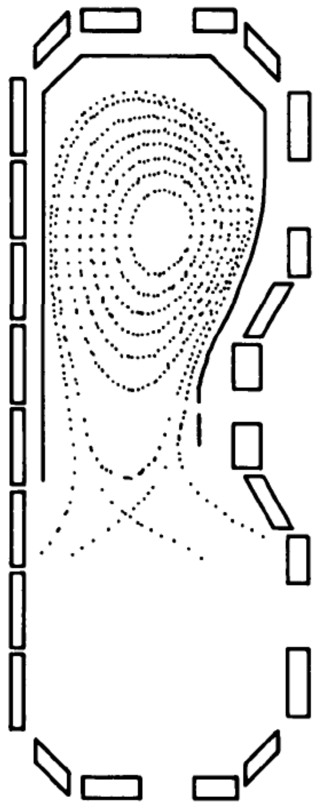
X-point target divertor
ADX tokamak proposal

Advanced magnetic divertor configurations: status of experiments

1. **Dedicated experimental devices (1979-1986)**
   - PDX
   - Doublet III Expanded boundary
2. **Snowflake divertor configuration (existing devices with existing coils, 2008-present)**
   - TCV, 2008
   - NSTX, 2009
   - DIII-D, 2012
   - EAST, 2014
3. **Long leg divertor physics (2010-present)**

1. Doublet III Expanded Boundary was the first advanced magnetic divertor configuration experiment

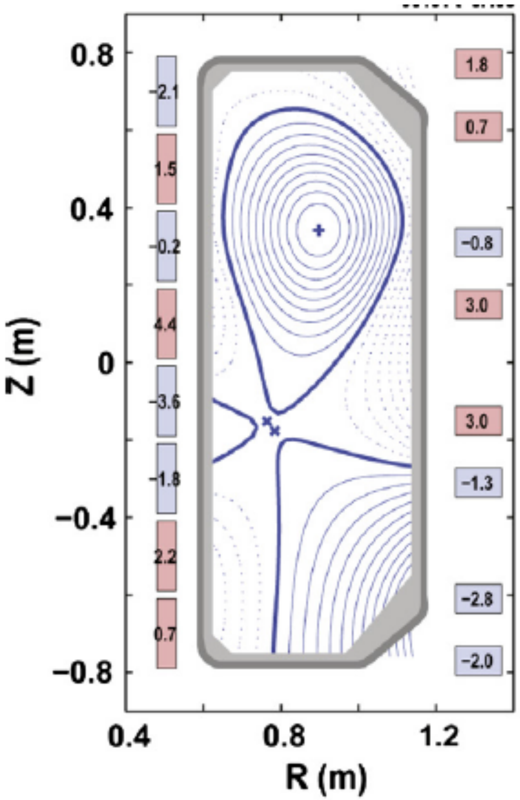


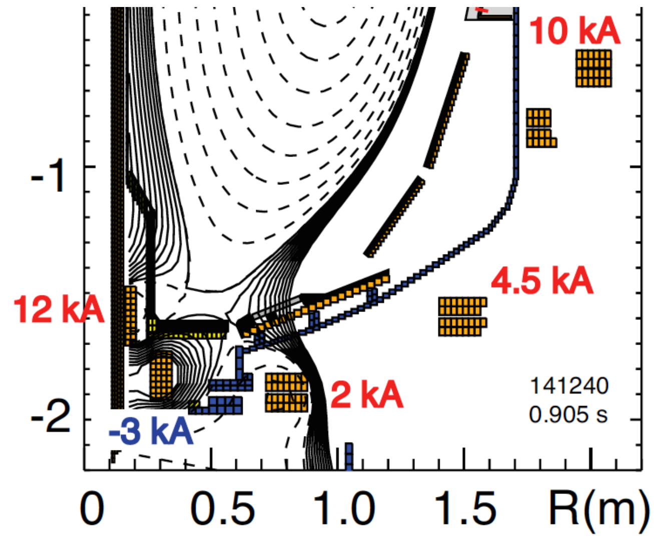


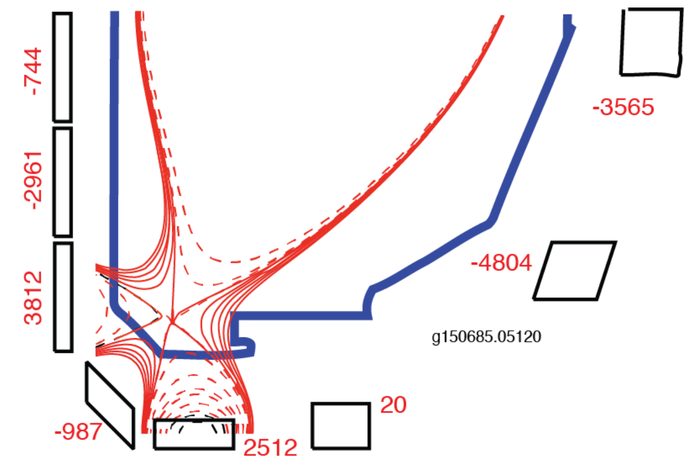
N. Ohyabu et. al, Nucl. Fusion 5 519 (1981)
A. Mahdavi et. al, J. Nuc. Mater. 111 (1982) 355

- Stability of divertor configuration with outside PF coils
- Integrated power and particle exhaust
  - Reduction of intrinsic impurities in the core
  - Radiative divertor plasma cooling with seeded argon
  - Argon screening from main plasma
- Compatibility with core confinement

2. Snowflake configuration has been realized in several devices using existing PF coils

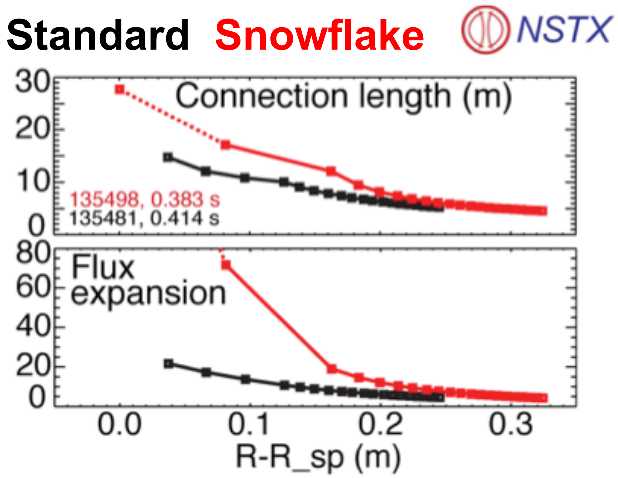


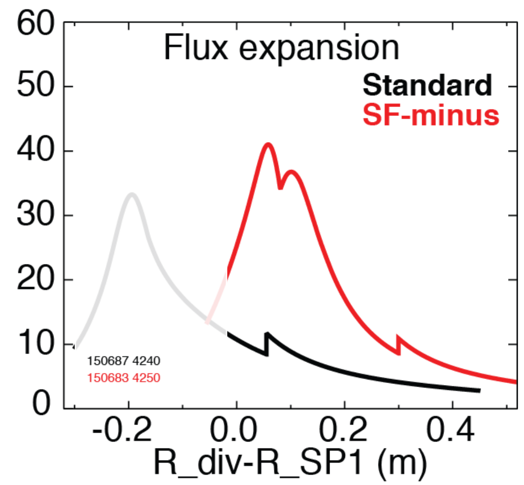


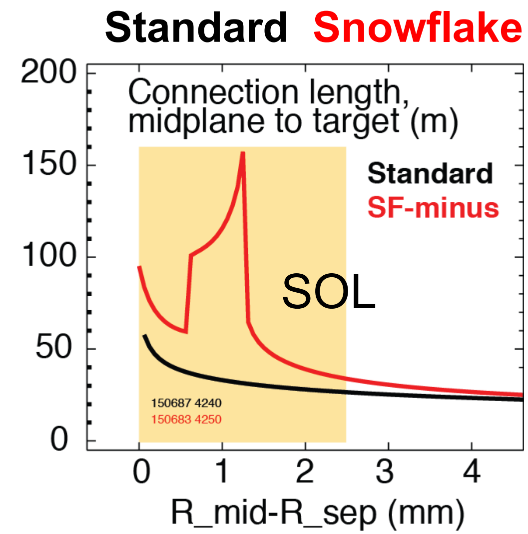


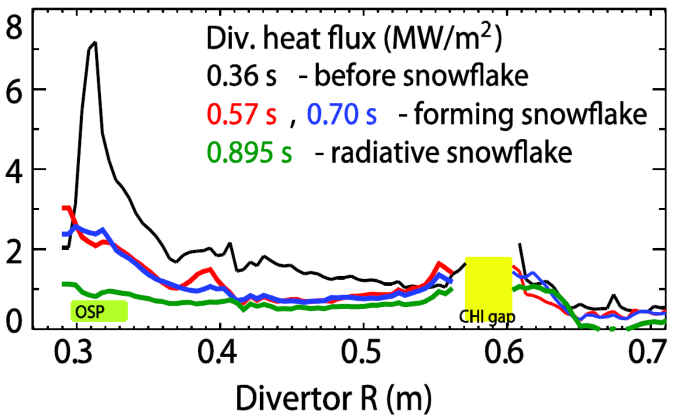
- TCV
  - Ip≤1 MA
  - B_t≤1.5 T
  - 16 PF coils, Pre-programmed currents
- NSTX
  - I_p=0.8-1.0 MA
  - B_t≤0.45 T
  - 3 divertor coils, pre-programmed currents
- DIII-D
  - I_p=0.8-1.0 MA
  - B_t=2 T
  - 3 divertor coils w/ control

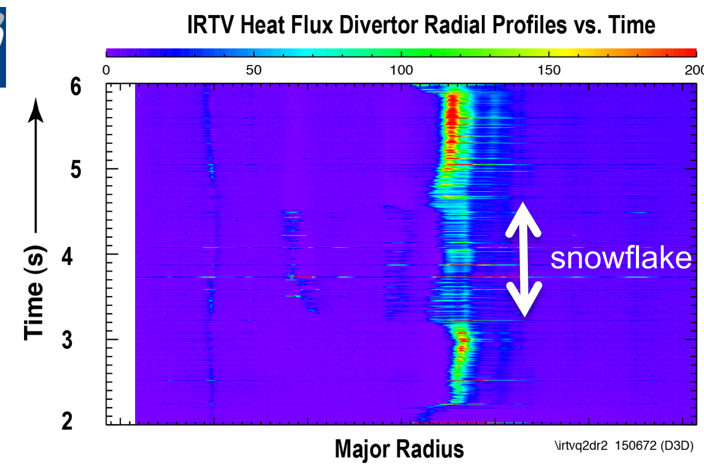
Divertor heat flux significantly reduced due to snowflake divertor geometry effects





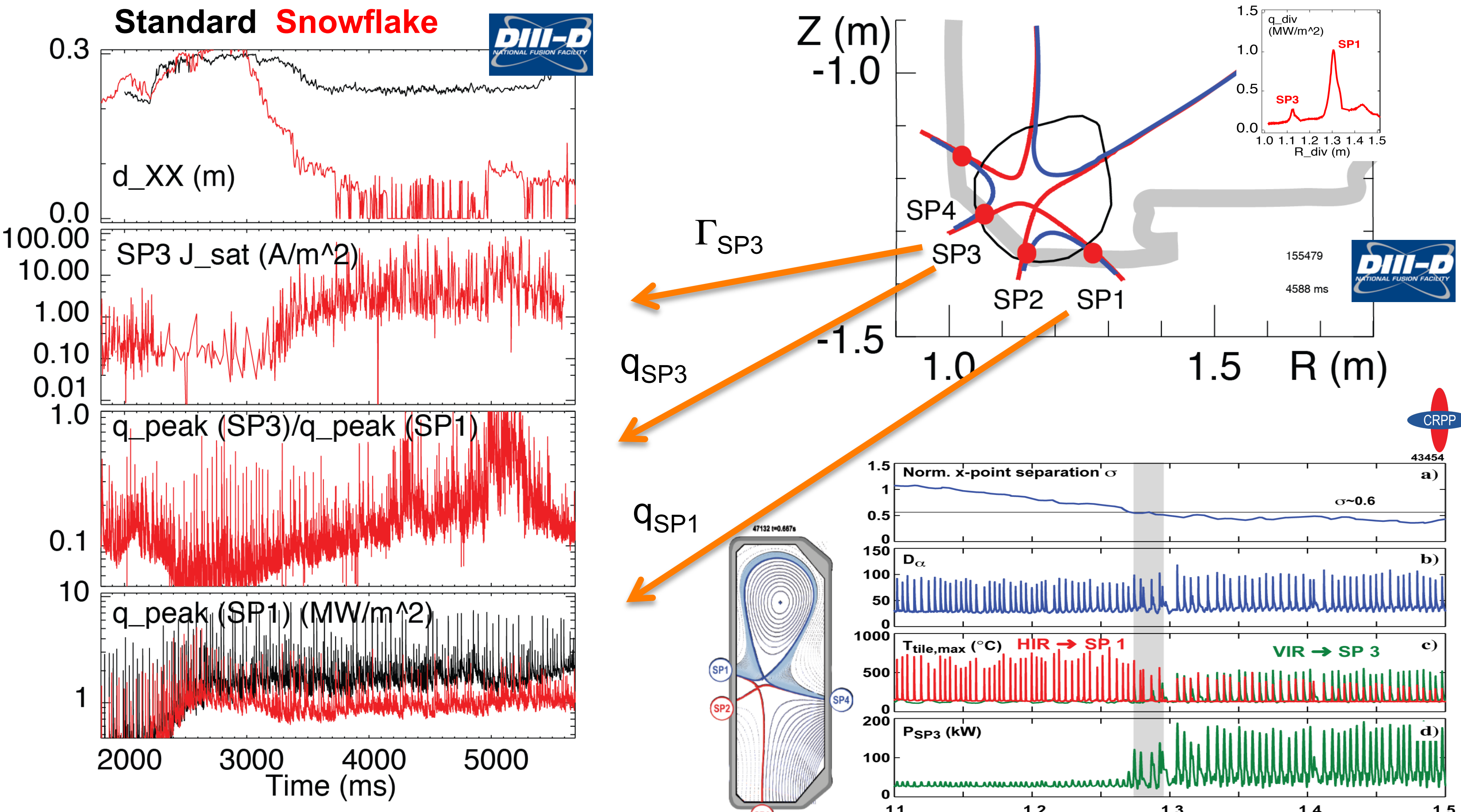






- NSTX
- DIII-D

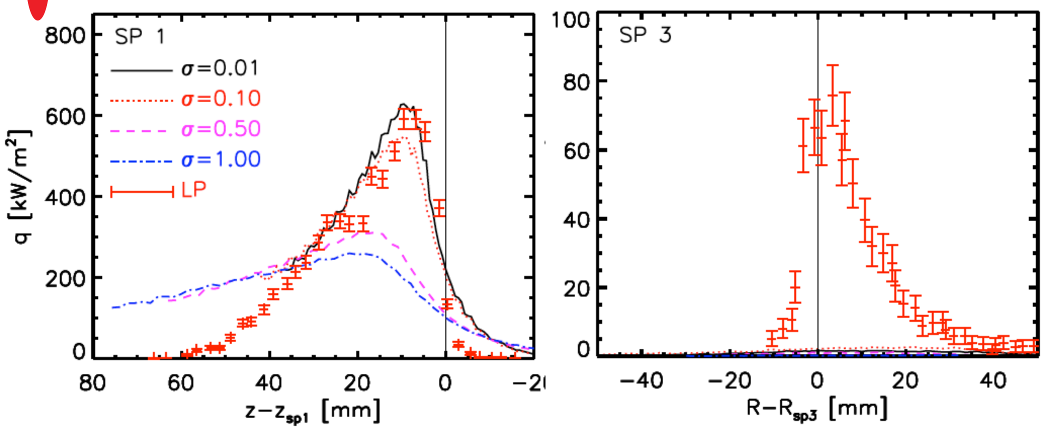
Snowflake divertor enables power and particle sharing over multiple strike points

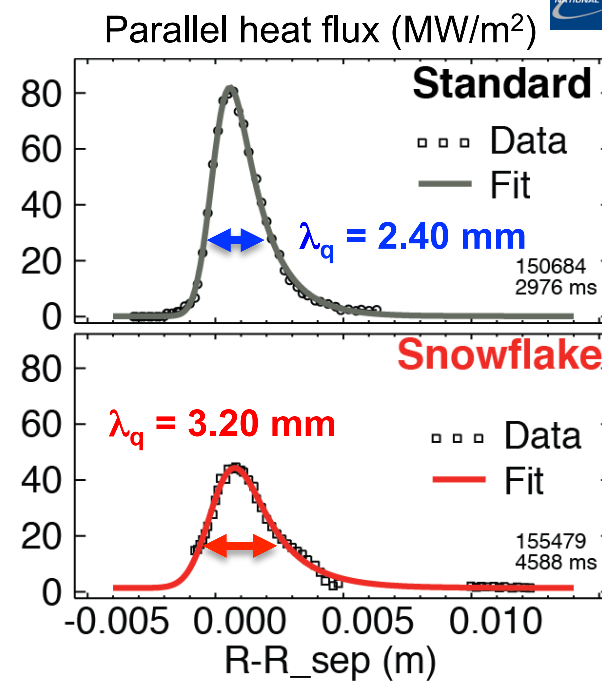


V. A. Soukhanovskii et.al, IAEA FEC 2014

W. Vijvers et.al, IAEA FEC 2012

Divertor heat transport affected by snowflake geometry





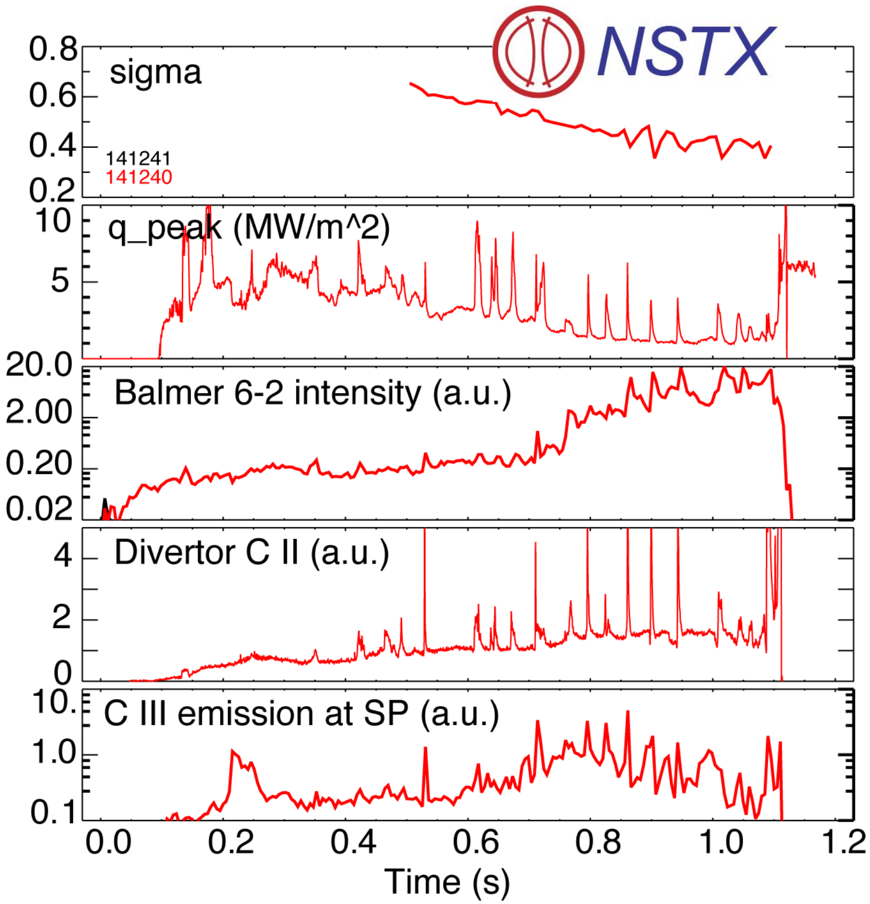
- EMC3-IRENE modeling under-predicts power in additional strike points
- Suggests additional transport channel in the null region

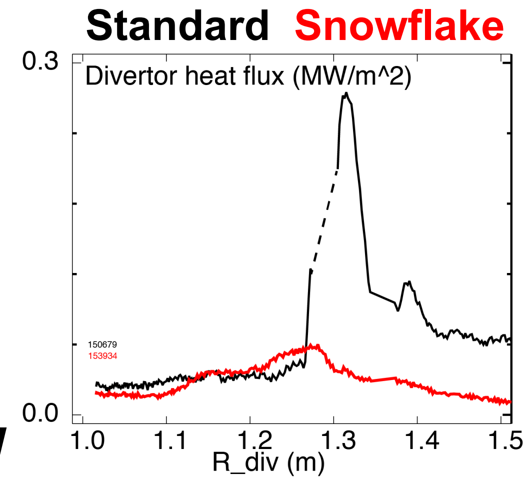
H. Reimerdes et al. PPCF 2013, T. Lunt et al. PPCF 2014

- Increased λ_q may imply increased transport
  - Increased radial spreading due to $L_{||}$
  - SOL transport affected by null-region mixing

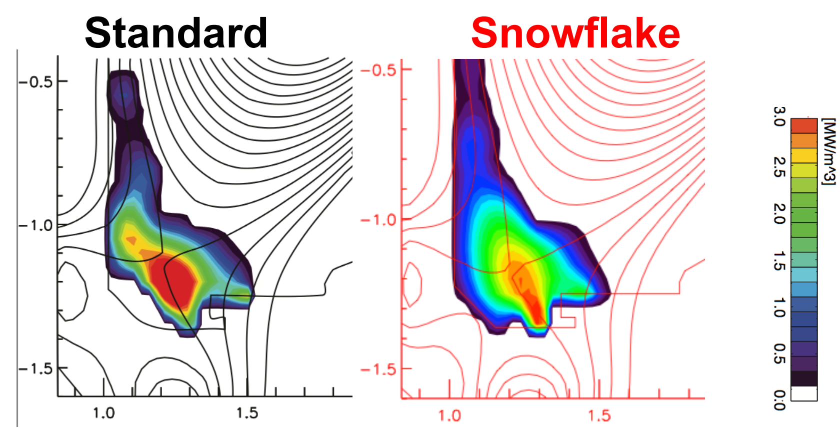
V. A. Soukhanovskii et.al, IAEA FEC 2014

Snowflake configuration favorably affects radiative divertor and detachment





P_{SOL} = 3-4 MW



- Natural partial detachment in NSTX snowflake otherwise inaccessible with standard divertor

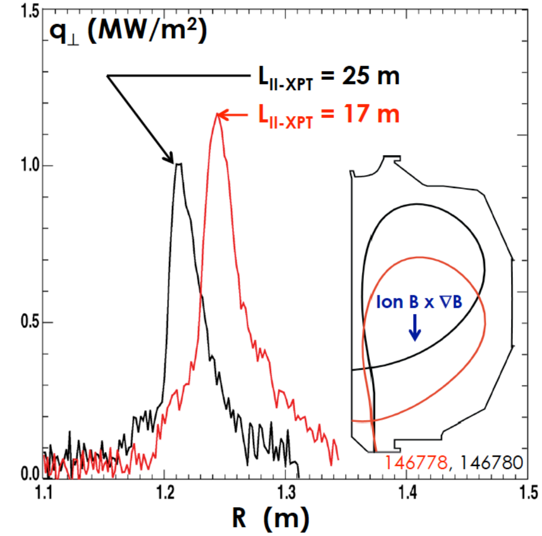
V. A. Soukhanovskii et.al, NF 2011; POP 2011

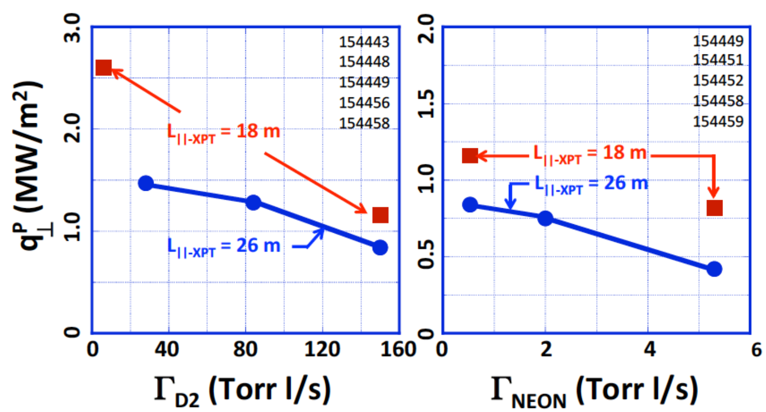
- Broader radiated power distribution, nearly complete power detachment in DIII-D

V. A. Soukhanovskii et.al, IAEA FEC 2014

3. Recent DIII-D experiments demonstrated benefits of the increased connection length

- Longer connection length vs shorter connection length
  - Divertor peak heat flux reduced
  - Divertor T_e reduced
  - Divertor radiation increased





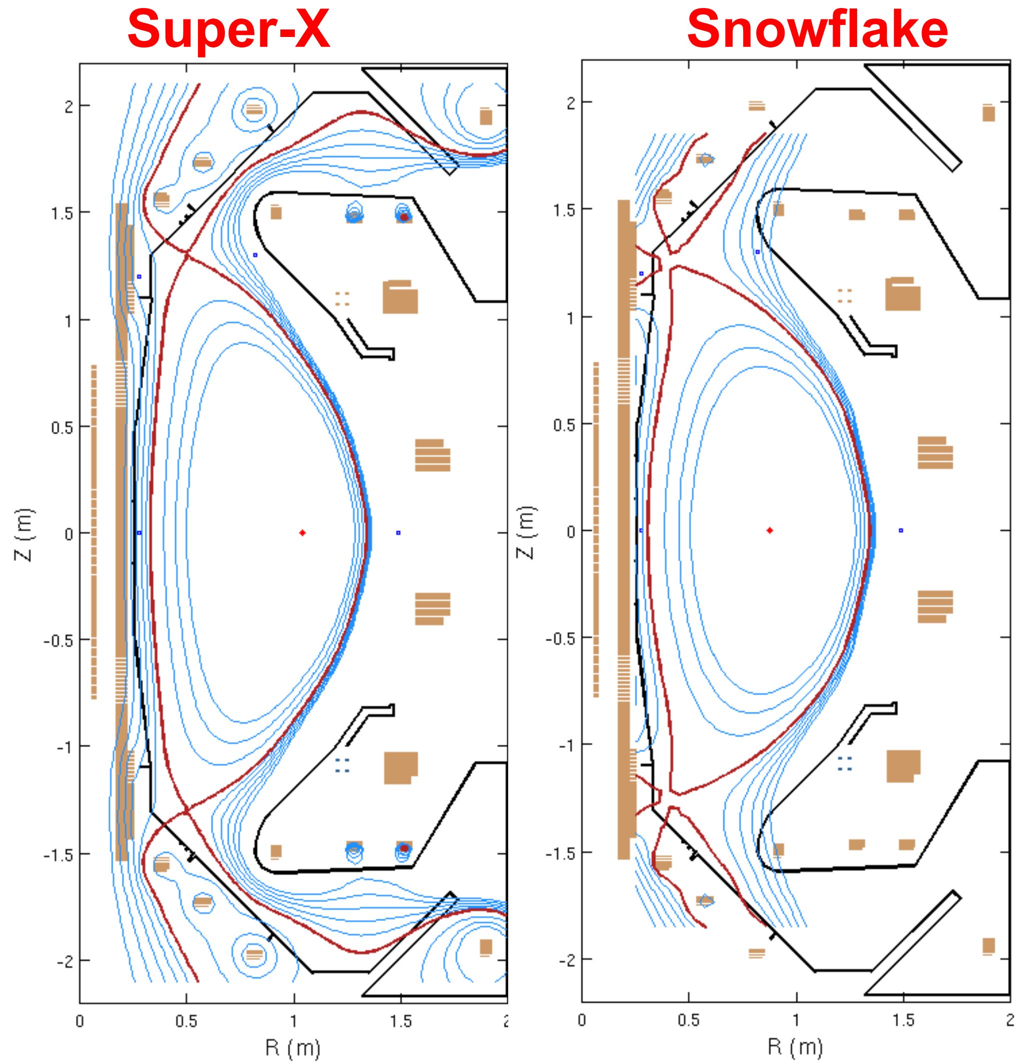
T. Petrie et.al, APS 2013, PSI2014, IAEA FEC 2014, APS 2014

Advanced magnetic divertor configuration development

- Near-term plans (5 years)
  - Clarify effects of 2nd order null, extra X-points and long legs on
    - Pedestal stability
    - Steady-state and transient transport (heat, ion, impurity)
    - Impurity radiation limits
- Tokamaks
  - Upgraded: TCV, NSTX-U, HL-2M, MAST-U
  - Existing: DIII-D, EAST
- Long-term plans (5-15 years) ?

MAST Upgrade to test advanced divertor configurations

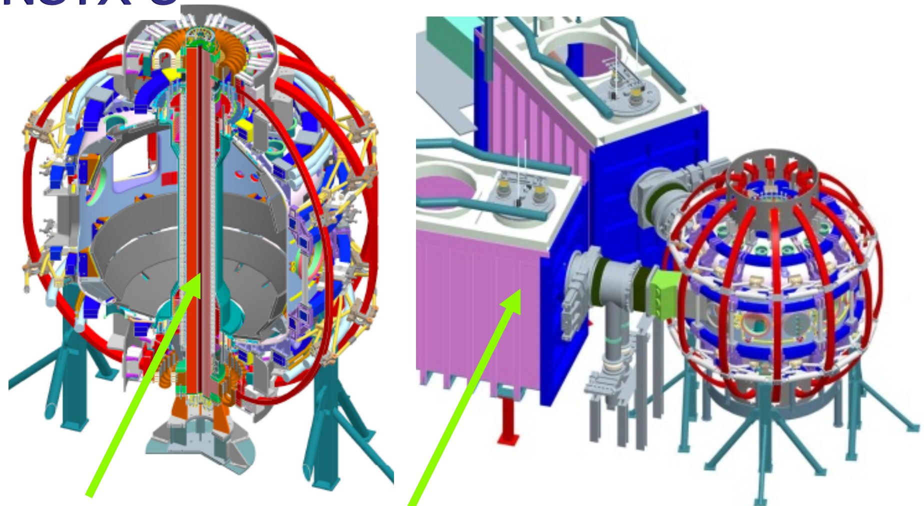
- B_t=0.8 T
- $I_p \leq$ 2 MA
- $P_{NBI} \leq$ 7.5 MW
- 8 divertor PF coils
- Extensive diagnostic set
- Radial and parallel transport, stability of detachment
- Pedestal formation, structure, 3D fields



G. Fishpool et.al, J. Nucl. Mater. 2013

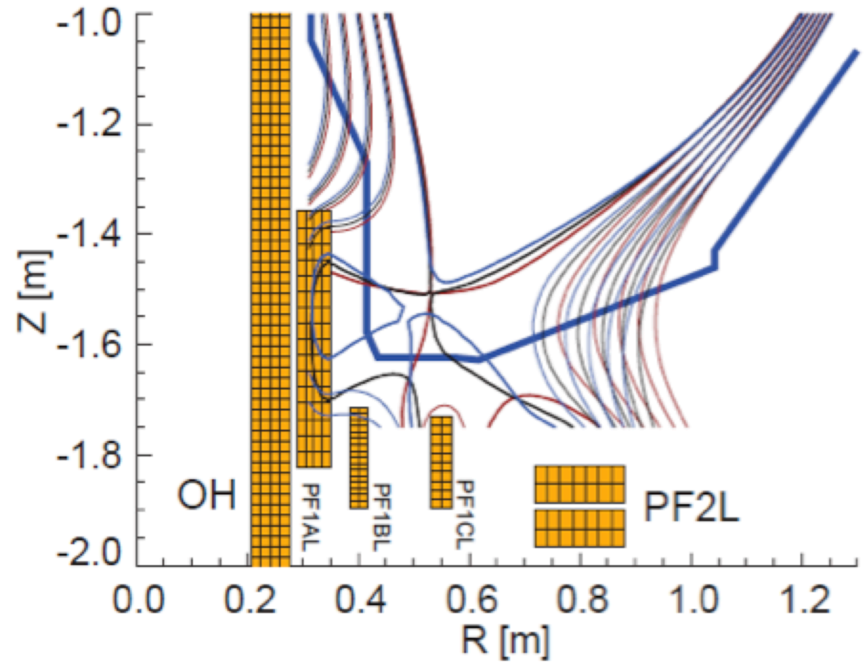
Snowflake divertor is a leading heat flux mitigation candidate for NSTX-Upgrade

NSTX-U



New center-stack $\quad$ 2nd neutral beam

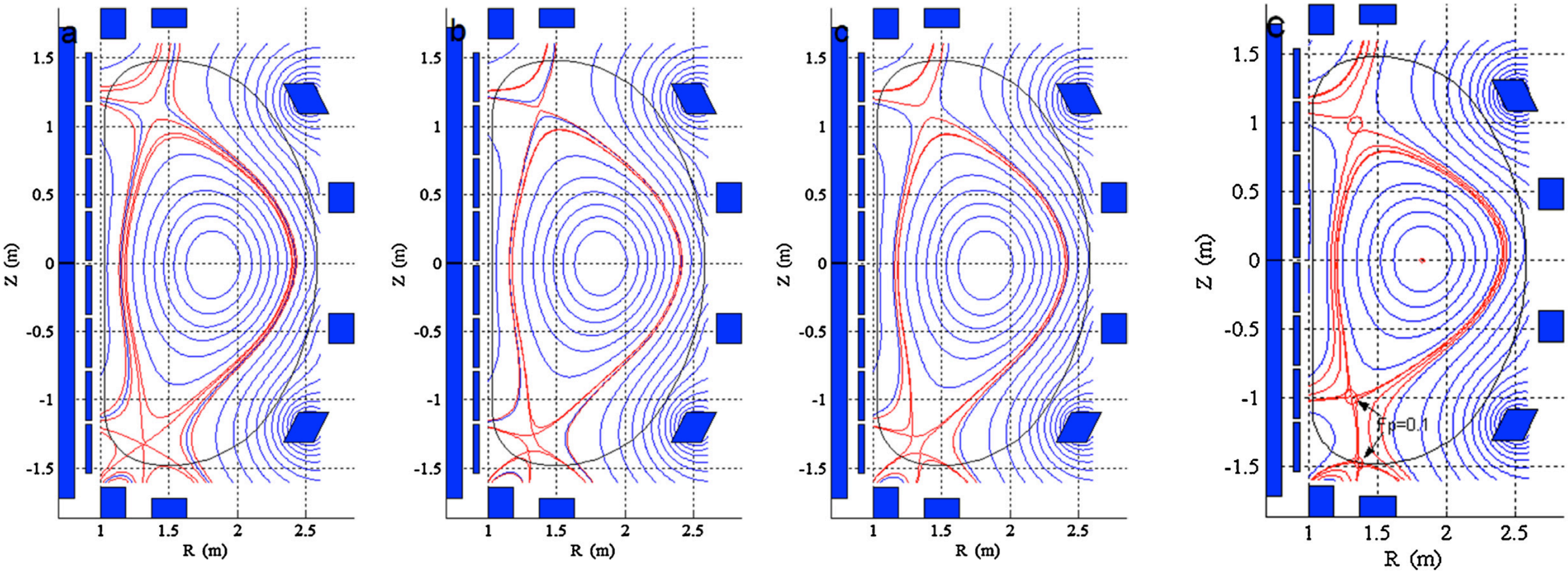
B_T	$\longrightarrow$ 1 T	P_{NBI}	$\longrightarrow$ 12 MW
I_p	$\longrightarrow$ 2 MA	pulse	$\longrightarrow$ 5 s



- NSTX-U Mission elements:
  - Advance ST as candidate for Fusion Nuclear Science Facility
  - **Develop solutions for the plasma-material interface challenge**
  - Explore unique ST parameter regimes to advance predictive capability for ITER
  - Develop ST as fusion energy system

J. E. Menard et. al, Nucl. Fusion 52 (2012) 083015

HL-2M tokamak enables a number of advanced magnetic configurations

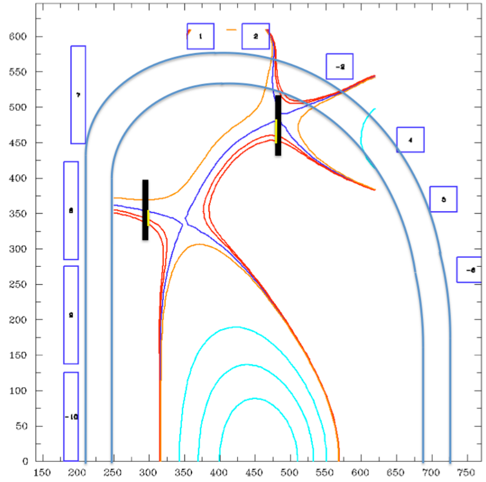


Snowflake **Snowflake-plus** **Snowflake-minus** **Tripod**

- Major radius R = 1.78 m
- Minor radius a = 0.65 m
- Toroidal field B_t = 2.2 T
- Plasma current I_P = 2.5 MA
- P_{in}=32 MW (Design)

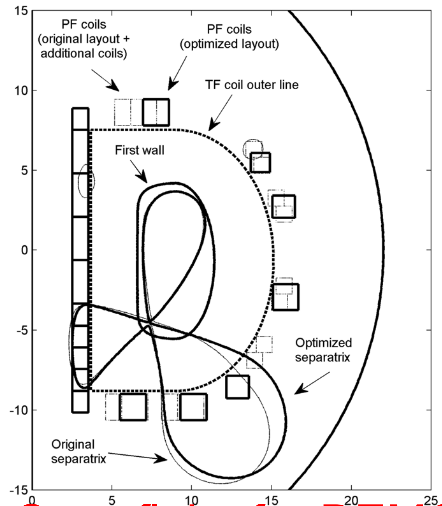
G.Y. Zheng et.al, Fus. Eng. Design 89 (2014) 2621

Many pre-DEMO and DEMO designs include snowflake and Super-X configurations



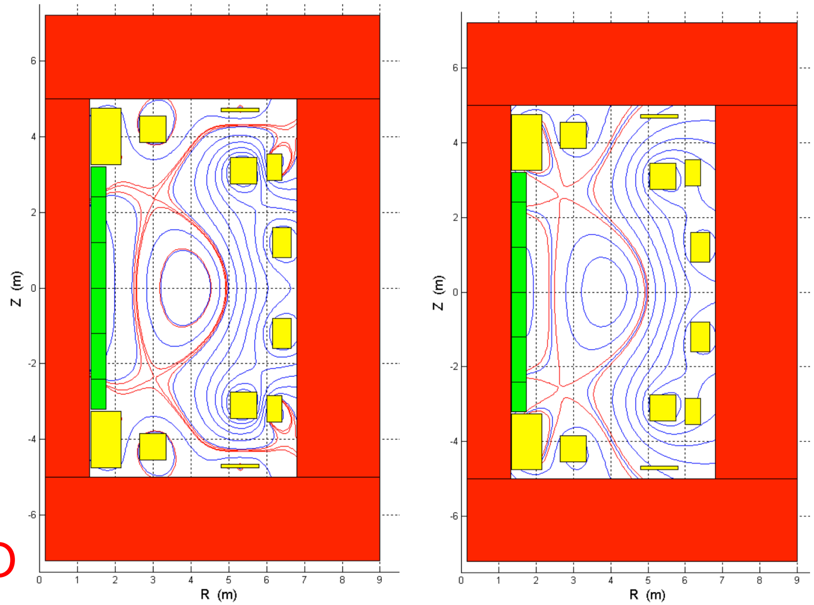
Super-X for Aries Slim CS

M. Kotschenreuther et. al, ARIES Workshop 2010



Snowflake for DEMO

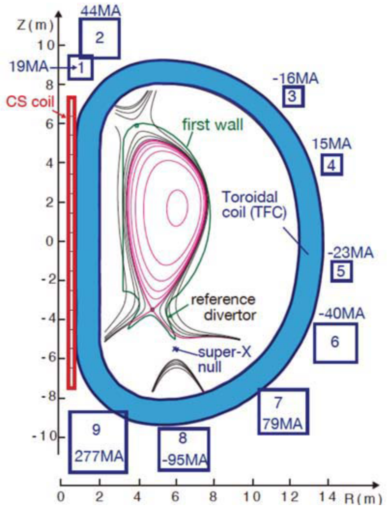
R Albanese et. al,
PPCF, 56 (2014) 035008

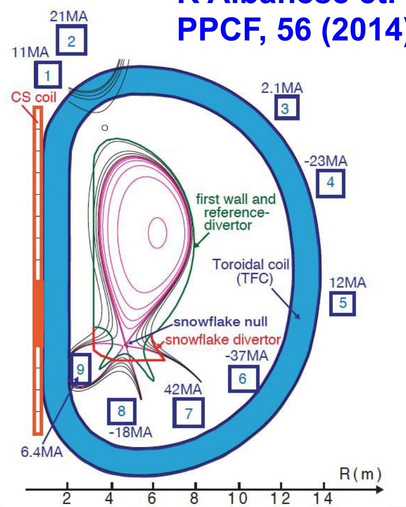


China Fusion Experimental Reactor

Y. Wan, SOFE 2013

Z. Luo et.al, IEEE TRANSACTIONS ON PLASMA SCIENCE, V42, 2014, 1021





Super-X and Snowflake for DEMO

N. Asakura et.al, Trans. Fus. Sci. Tech. 63, 2013, 70.

Everything New Is Actually Well-Forgotten Old (Новое – это хорошо забытое старое, the Russian saying).

- A number of potentially attractive magnetic divertor configurations exist
- Much research remains to be done to qualify them as
  - Advanced magnetic divertor configurations
  - Divertor candidates for a fusion reactor

Backup

Abstract

The present vision for controlling the plasma–material interface of a tokamak is an axisymmetric poloidal magnetic X-point divertor. The divertor must enable access to high core and pedestal plasma performance metrics while keeping target plate heat loads and erosion within the operating limits of plasma-facing component cooling technology and target plate materials. The proposed ITER divertor is based on standard X-point geometry designs tested in large tokamak experiments and uses tilted vertical targets to generate partial radiative detachment of the strike points. However, the standard divertor approach is likely to be insufficient for next step advanced tokamak and spherical tokamak devices such as the proposed fusion nuclear science facilities and for the DEMO reactor.

Novel magnetic divertor configuration development and optimization has always been an active area in fusion plasma research. In this talk, advanced poloidal divertor concepts and experimental performance will be reviewed. Advanced divertors have the capability to modify steady-state and transient power exhaust via modifications to parallel and perpendicular transport and dissipative loss channels. The basic physics principles of these concepts will be summarized, from the first divertor for impurity control proposed by L. Spitzer for the stellarator, to long legged divertors, expanded boundaries, multiple X-point divertors, and multipole divertors for tokamaks. Many of the these divertor configurations face practical limitations on magnetic coil layout and construction. In recent years, two advanced divertor concepts have been pursued experimentally: snowflake (2nd order null) divertors, implemented in the TCV, NSTX, DIII-D and EAST tokamaks with existing magnetic coils, and the (long-legged) Super-X divertor, which is presently being implemented in MAST Upgrade using specially designed additional coils. The status and plans for research in these areas will be summarized.

Several outstanding physics and engineering problems need to be addressed in order to qualify an advanced divertor concept for a next step tokamak reactor. This talk will discuss the general issues and motivations, including coil design and placement, equilibria design and plasma real-time control, plasma-facing component design, compatibility with highly radiative scenarios, and integration with high-performance core and pedestal plasma.

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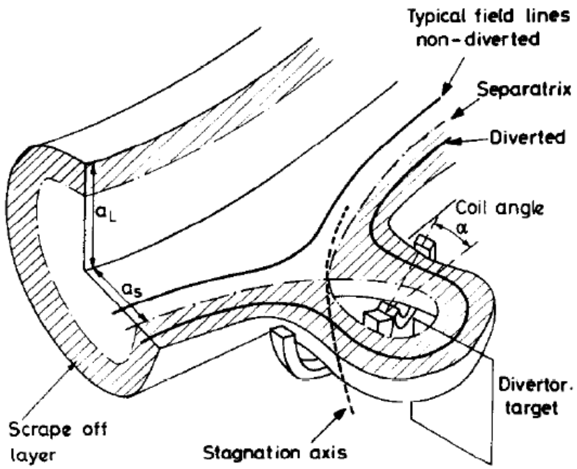
Conventional divertor history…

Poloidal bundle divertor



A. D. Sanderson et.al, J. Nucl. Mater. 76 530 (1978)

- First proposal of a magnetic divertor
  - L. Spitzer, Phys. Fluids 1 (1958) 253 (for impurity control in a stellarator)
- First use of a poloidal divertor
  - JFT-2a (DIVA), Japan, 1975 (approx.)
- First demonstration of impurity control using a divertor
  - JFT-2a, Japan, 1975 (approx.)
- First demonstration of H-mode with a poloidal divertor
  - ASDEX, Germany 1982