FIP/3-4Rb

17(Friday) am

DEMO Concept Development and Assessment of Relevant Technologies

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Japan Atomic Energy Agency

FIP/3-4Ra

17(Friday) pm

Physics and Engineering Studies of the Advanced Divertor for a Fusion Reactor

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Introduction: Demo concept design and Advanced divertor study

DEMO is a bridge from ITER to a commercial reactor, and will demonstrate

Electric power generation, **Tritium self-sufficiency**, **Steady-state operation**.

Breeding blanket and large power exhaust are principal design issues.

Design parameters for DEMO have been studied with considering,

Medium size (R_p > 8m) for full inductive I_p ramp up by CS coil ($\Delta\Psi_{CS} \propto R_{CS}^2$)

Fusion power (P_{fus} < 2 GW) compatible with power handling in divertor.

To minimize the development subjects, it is designed by utilizing existing technologies from Tokamaks (ITER, JT-60SA, ….) and Nuclear reactor technologies (PWR).

Advanced divertor study will provide new options of magnetic configuration.

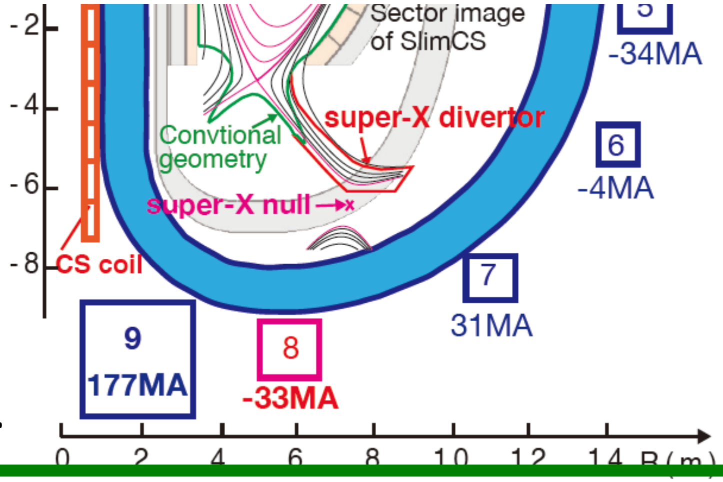
Advanced divertor is produced by driving reverse current in one of the divertor coils

⇒ Coil currents and number are increased.

Physics and Engineering issues were investigated in a super-X divertor with a short divertor leg, comparable to the conventional divertor size.

Divertor performance was compared in SlimCS (R = 5.5m, P_{fus} = 3GW, I_p=16.7MA) in order to compare the previous results in the conv. divertor.

Super-X divertor image for SlimCS



Divertor heat load and compatible heat removal technology are important key for reactor design

FIP/P8-11 K. Hoshino et al.

Basic concept of divertor

- Water-cooling and W mono-block target design: Lower peak heat load (< 5MW/m^2) is required for **W-target&F82H(RAFM)-cooling tube design**.

Assessment of reduction in heat load by SONIC

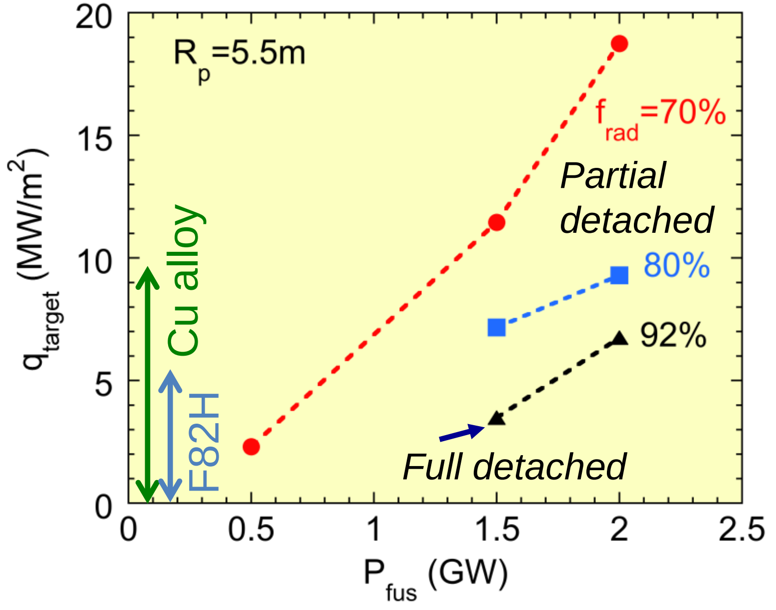
- Simulation study for R_p =5.5 m SlimCS indicated: q_{target} < 10 MW/m^2 is obtained with large radiation fraction (f_{rad} =P_{rad}/P_{out} > 80%, P_{out} = 300-400 MW) for P_{fus}=1.5-2 GW,
- suggesting q_{target} < 10 MW/m^2 in **larger R_p** with **P_{fus}=1.5 GW and f_{rad} = 70%** ($q_{target} \propto P_{out}/R_p$).

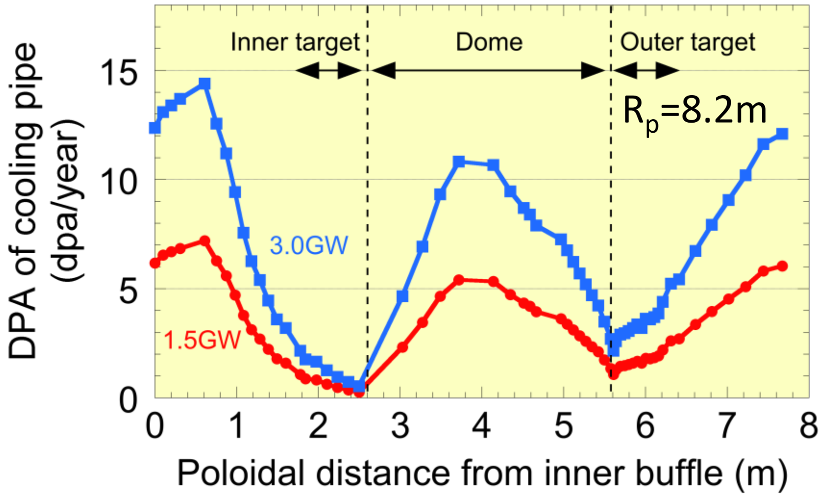
Assessment of heat removal capability

- P_{fus} = 1.5 GW operation reduces dpa/year < 1.5 near the strike-point: **W-target&Cu-alloy-cooling tube** will be applied at inner and outer targets.

⇒ Replacement of the divertor target is expected in 1-2 years.

Peak heat load at outer target, incl. plasma, surf. recomb., radiation and neutral loading

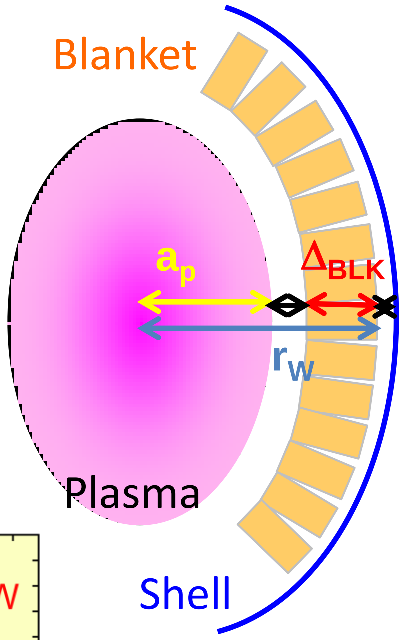


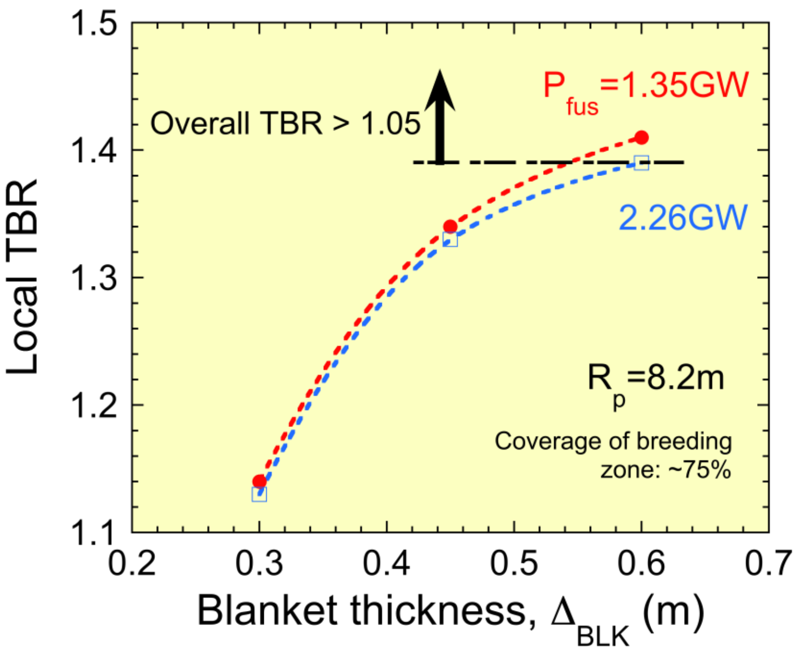


Impact of blanket thickness for overall TBR~1.05 on plasma elongation of ~1.65 by vertical stability

Assessment of blanket thickness

- Water cooled solid breeder based on ITER-TBM
  - ✓ PWR condition (~300℃, 15.5 MPa)
  - ✓ $Be_{12}Ti$ and Li_2TiO_3 pebbles
- Overall TBR >1.05 is evaluated for **blanket thickness of 0.6m** and P_{fus} < 2.0 GW.

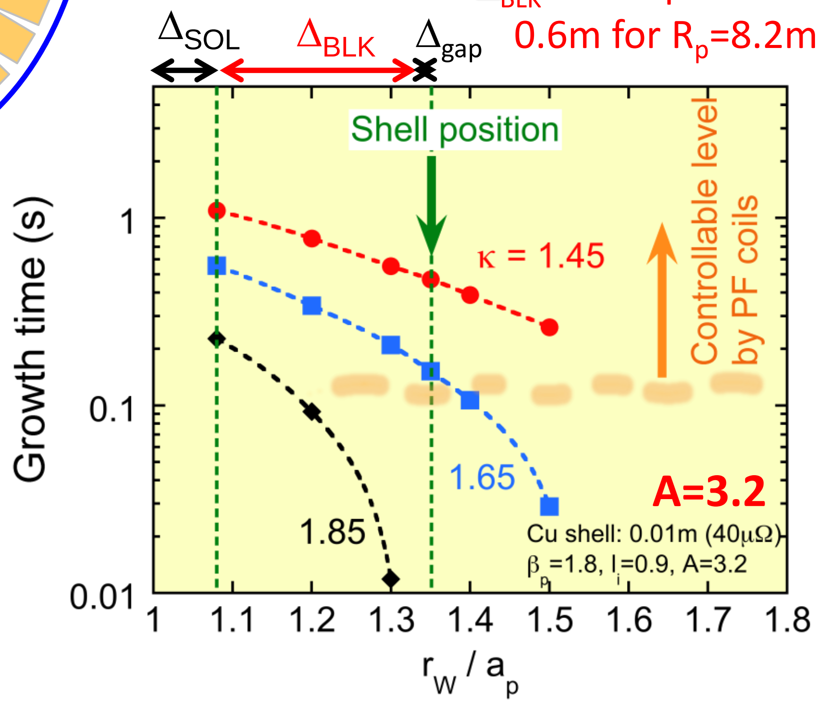




Assessment of vertical stability

- No use of in-vessel coil in DEMO
- Stabilize plasma by conducting shell, typically at r_W/a_p ~ 1.35
- Vertical stability analysis indicates design elongation of κ_{95} ~ 1.65.

Δ_{BLK} corresponds to 0.6m for R_p=8.2m



DEMO scoping study: a concept design with R_p~8.5m & P_{fus}~1.5GW based on technology assessments

Based on the assessment, possible design/plasma parameter sets are evaluated by systems code (TPC).

Key concepts

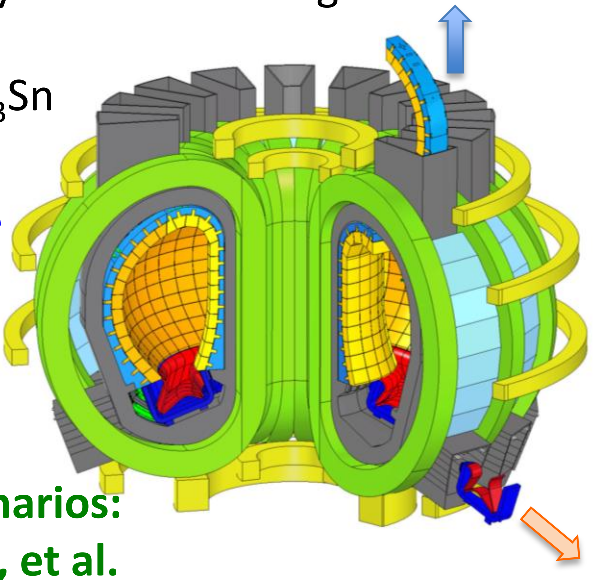
- **R_p > 8 m** for full inductive I_p ramp up.
  - ✓ Operation flexibility from pulse to steady-state
- **P_{fus} ~ 1.5 GW and P_{gross} ~ 0.5 GW** based on the assessments of divertor heat removal and overall TBR > 1.05
- **κ_{95} = 1.65** for vertical stability with conducting shell.

Poster

- **B_T^{max} > 12 T** based on Nb_3Sn or Nb_3Al, S_m = 800 MPa
- **Segmented maintenance** scheme:

  Segment RM image for blanket and divertor

- **Analysis of Accident Scenarios: SEE/P5-10 M. Nakamura, et al.**



	Parameters	DEMO (Steady state)	Ref. ITER
Size & Configuration	R_p (m)	8.5	6.35
	a_p (m)	2.42	1.85
	A	3.5	3.43
	κ_{95}	1.65	1.85
	q_{95}	4.1	5.3
	I_p (MA)	12.3	9.0
	B_T (T)	5.94	5.18
	B_T^{max} (T)	12.1	11.8
Absolute Performance	P_{fus} (MW)	1462	356
	P_{gross} (MWe)	507	-
	Q	17.5	6
	P_{ADD} (MW)	83.7	59
	n_e ($10^{19}m^{-3}$)	6.6	6.7
	NWL (MW/m^2)	1.0	0.35
Normalized Performance	HH_{98y2}	1.31	1.57
	β_N	3.4	2.95
	f_{BS}	0.61	0.48
	n_e/n_{GW}	1.2	0.82
	f_{He}	0.07	0.04

1. Concept design study of advanced divertors for DEMO

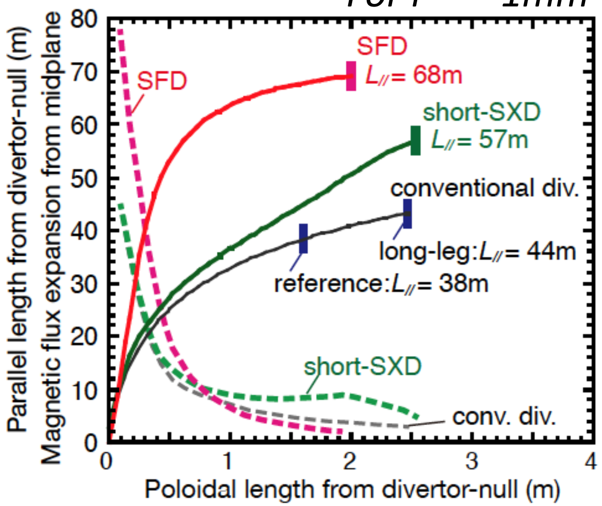
Study showed large current (>100 MAT) is required for the divertor coils outside TFC

⇒ **Divertor coils should be installed <u>inside TFC</u>: "interlink-winding"**

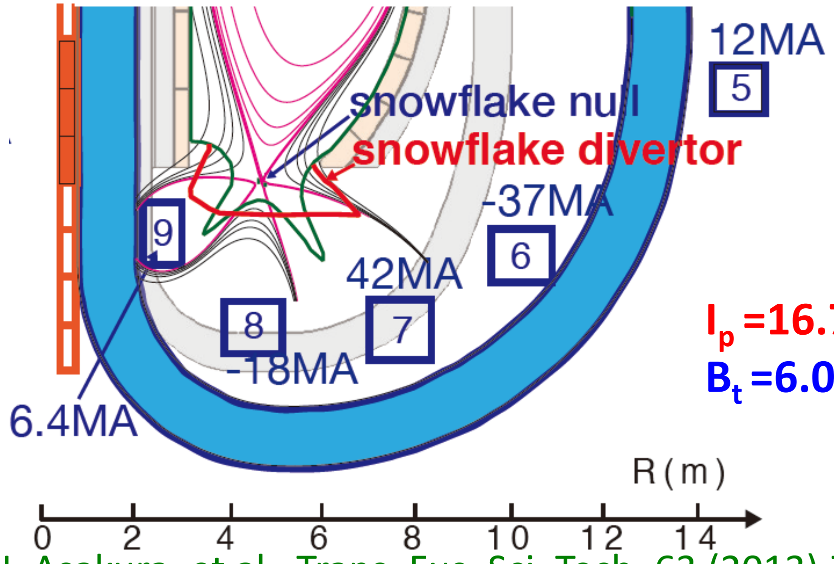
- **Snowflake**: Flux expansion (f^{exp}) increases near SF-null and Connection length ($L_{//}$) is 1.5-1.7 times, while f^{exp}_{div} and *target wet area* are smaller than conv. divertor,
  ⇒ **appropriate for compact divertor concept.**

- **Short Super-X**: f^{exp} and $L_{//}$ increase along divertor leg
  ⇒ **radiation and detachment control in divertor.**
  Interlink coil current and number are less than SFD.

For r^{mid} =1mm

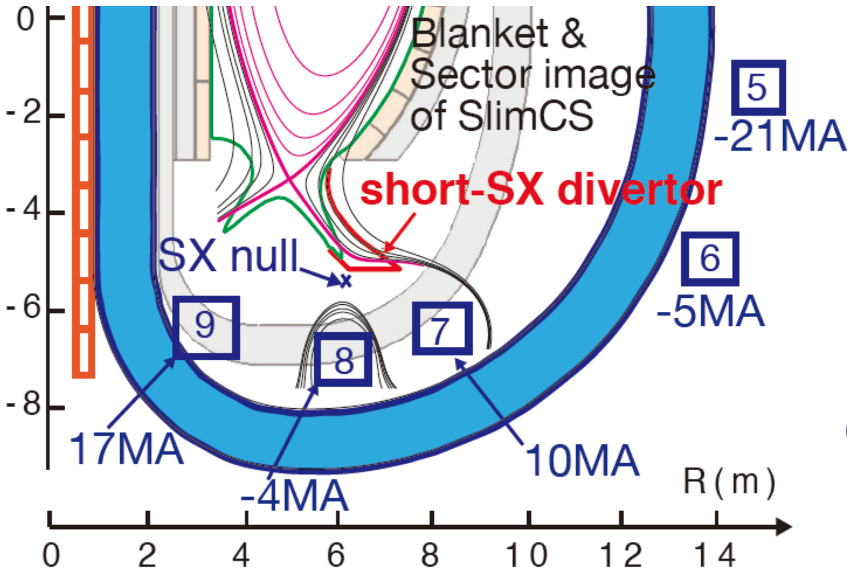


Snowflake divertor (SFD)



I_p =16.7MA
B_t =6.0T

Short super-X divertor (SXD) in outer leg



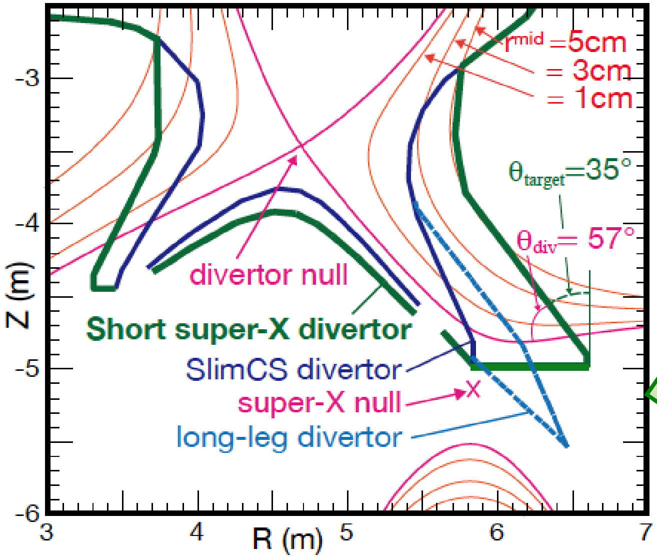
N. Asakura, et al., Trans. Fus. Sci. Tech. 63 (2013) 70.

2. Conceptual design of Short super-X divertor

Interlink divertor coil and the short SXD are arranged *under Engineering restrictions*:

- **Arrangement of PFCs with 2 interlink-coils:**
  Interlink coils outside neutron shield and vessel.
- **Divertor cassette and its replacement:**
  same height as SlimCS divertor
- **Superconductor interlink design: Nb_3Al**
  maximum current 25MAT and 1.6m size

Magnetic configuration of short-SXD and conventional divertor geometries

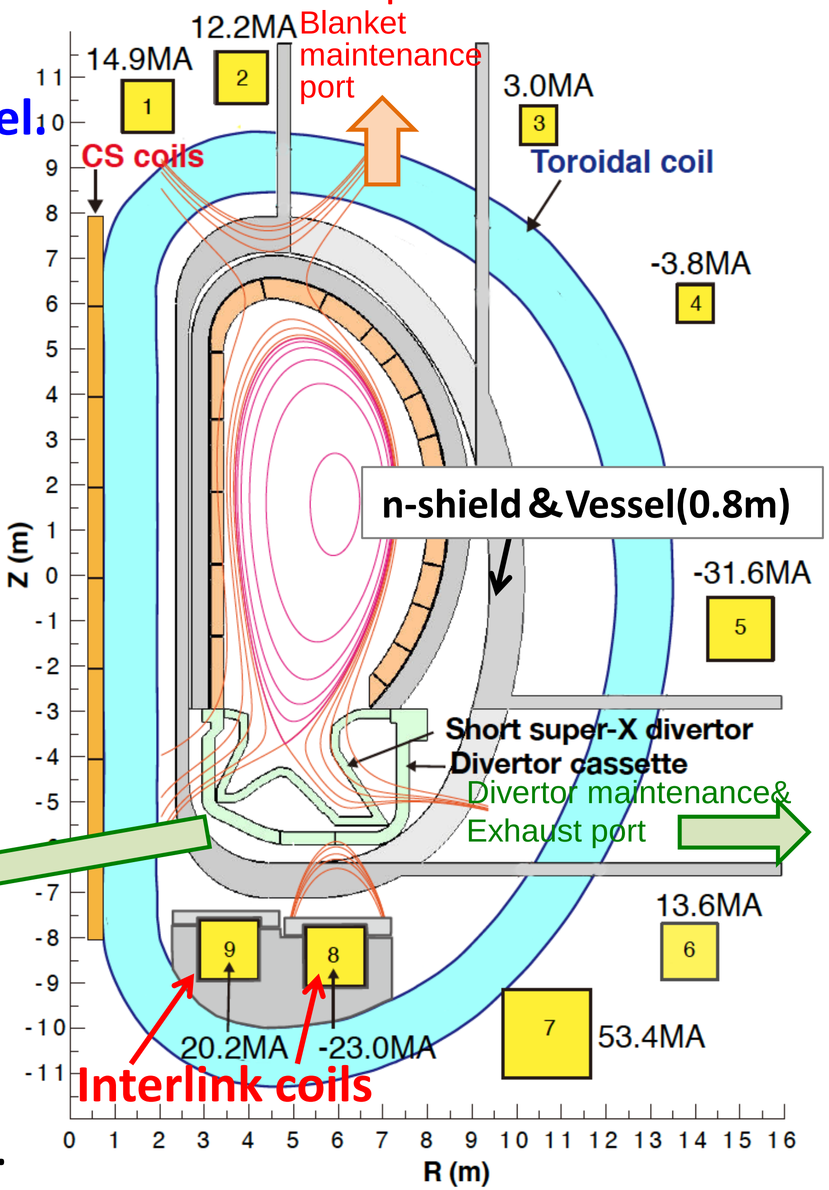


$f_{SXD} = (\Psi_{SX\text{-}null} - \Psi_{ax})/(\Psi_{Div\text{-}null} - \Psi_{ax}) = 0.99$:

max f^{exp} and $L_{//}$ increases to 19 and 2 times than conv. div.

2 interlink arrange for SlimCS

I_p =16.7MA, B_t =6.0T



Engineering design and issues for Interlink divertor coil

Nb_3Al Superconductor is preferable for Interlink divertor coil than Nb_3Sn:

(1) React and Wind

(2) Stress analysis (< 250MPa): lower load ratio (<50%) of allowable stress (500MPa)

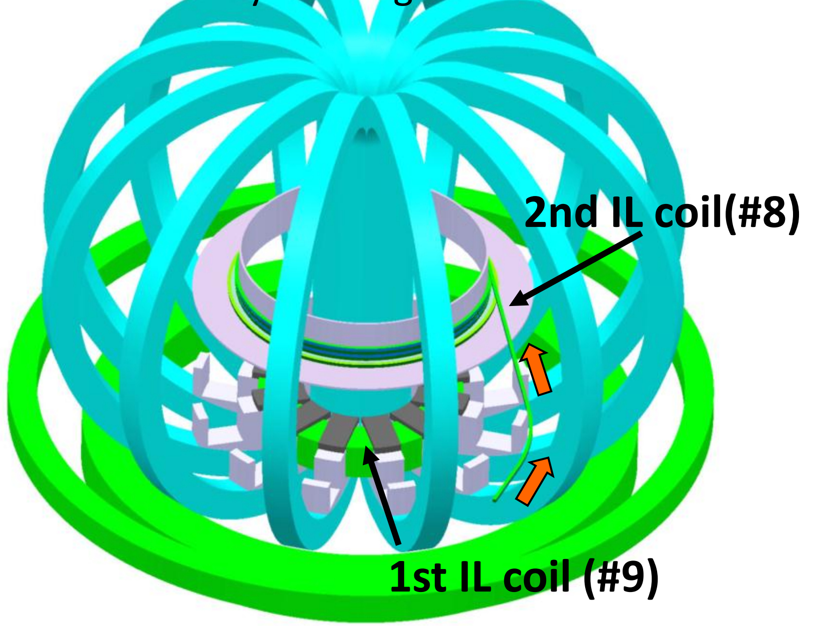
Design issues and Development:

- **SC filament is reduced from 60 to 1μm (equivalent to Nb_3Sn) to decrease AC losses.**
- **EM-force on IL-coil (-23 MAT) becomes 500-600 MN under average Br (0.67T)**

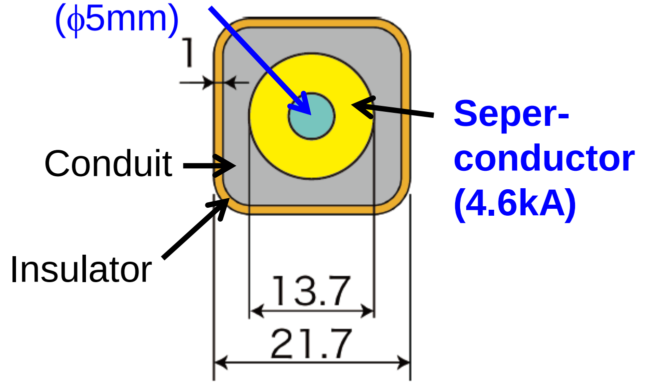
⇒ additional load on TFCs ⇒ support of IL-coil is necessary.

Winding image of Nb_3Al conductor:

Superconductor coil is inserted though TF coils, and is winded by rotating SC bobbin.



Interlink superconductor is designed, based on ITER poloidal coil conductor:



- **25 MAT corresponds to coil size of 1.6mx1.6m**

H. Utoh, et al. Fusion Eng. Des. 89 (2014) 2456.

3. Divertor plasma simulation of short SXD by SONIC code

SONIC simulation for short SXD :

Input parameters are the same as conventional divertor:

P_{out}= 500 MW, n_i=7x10^{19} m^{-3} at core-edge boundary,

$\chi_i = \chi_e$ = 1 m^2s^{-1}, D = 0.3 m^2s^{-1} :same as ITER simulation

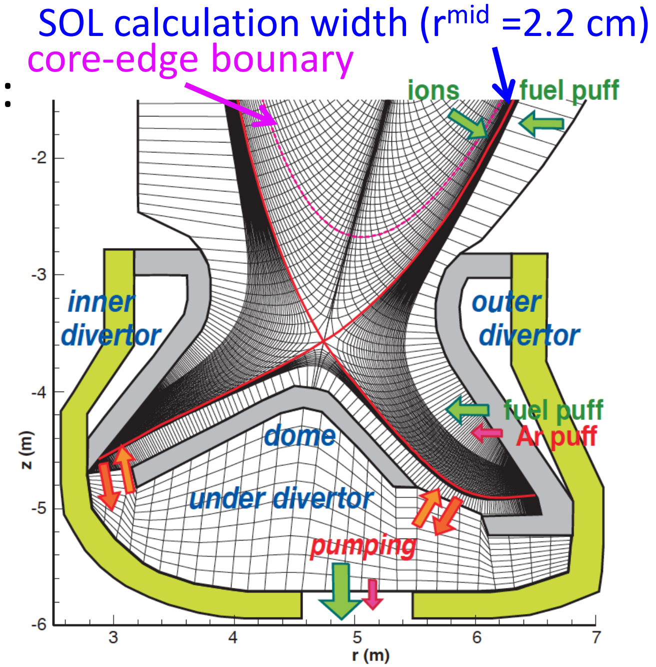
Radiation power loss is increased by **Ar seeding**

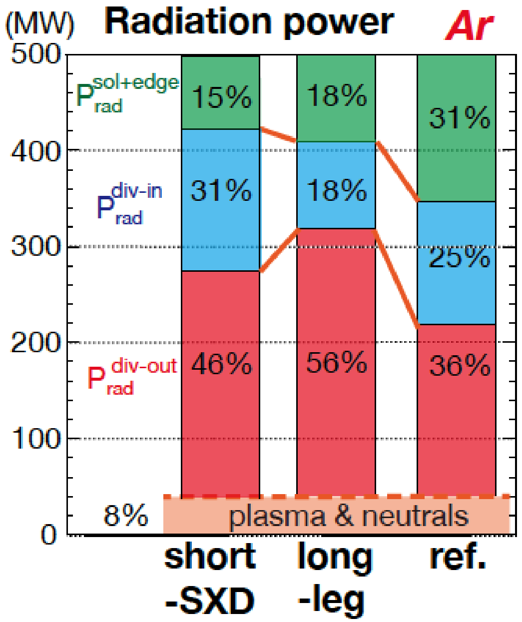
at the same total radiation fraction (P_{rad}/P_{out}) = 0.92

as SlimCS divertor analysis (IAEA FEC2012).

Radiation power is increased in the divertor, compared to reference divertor

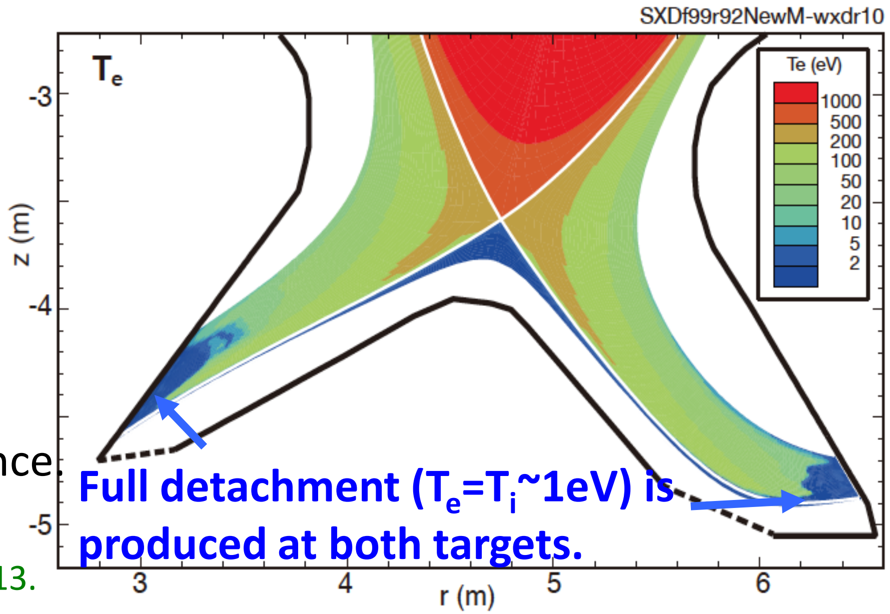
⇒ Impurity retention is improved.





Note:
n_Z/n_i(SOL) ~ 1.5%
in short-SXD and LL,
while ~2% in reference

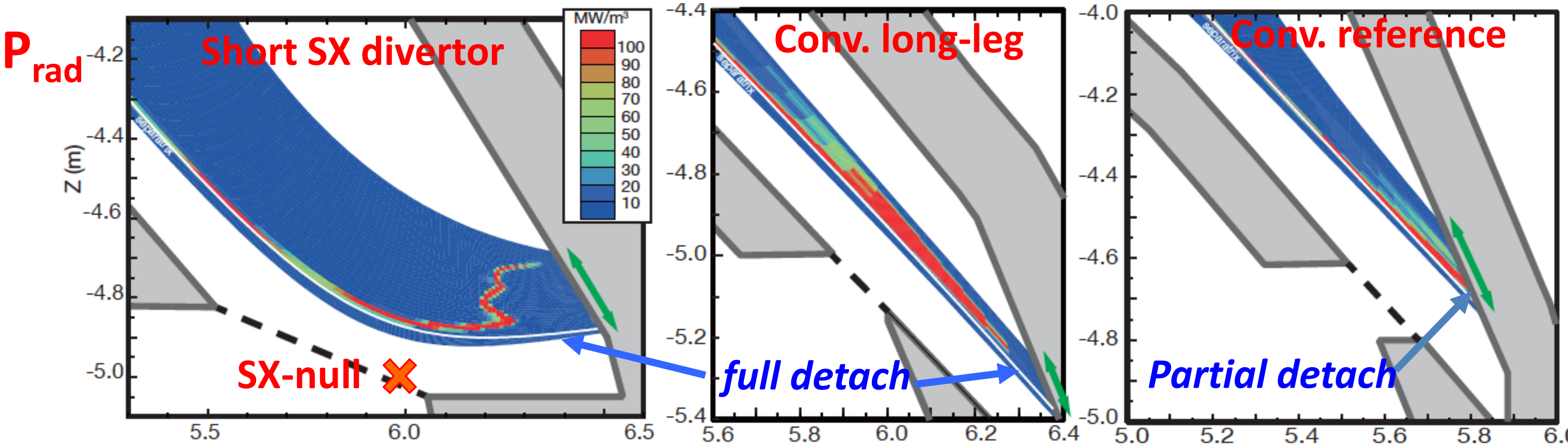
N. Asakura, et al. Nucl. Fusion, 53 (2013) 123013.



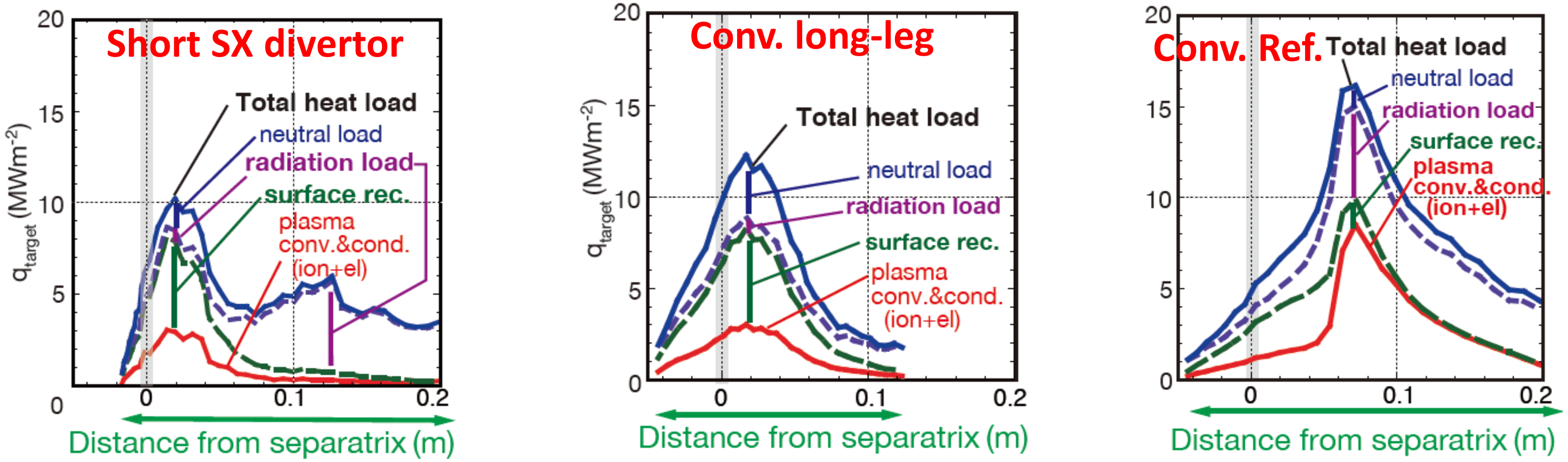
Detachment is produced near SX-null in short SXD

Radiation is enhanced *near the SX-null* (along the separatrix). At the same time, high temperature plasma (>100 eV) is maintained near the SX-null (in Poster).

⇒ Radiative area in the poloidal direction is narrow due to longer fieldline length.



- **Maximum total heat load ~10 MWm^{-2} in the full detached divertor ($T_e = T_i$ ~ 1eV)**
  ***Surface recombination* is dominant** near the separatrix, due to large ion flux.



Summary: Demo concept design and Advanced divertor study

DEMO concept is considered through the assessment of relevant technologies.

- R_p > 8.0 m for full inductive I_p ramp up by CS coil
  - ✓ Operation flexibility from pulse to steady-state
- P_{fus} = 1.5 GW and P_{gross} ~ 0.5 GW are foreseeable from the viewpoints of
  - ✓ Divertor heat removal capability and tritium self-sufficiency in blanket.

By considering above and other assessments, DEMO concept design study shows,

- ✓ f_{rad} = 70% will be compatible with P_{fus}=1.5 GW, R_p > 8.0 m and partial use of Cu-alloy as cooling pipe near high q_{target} and lower dpa/year region
- ✓ Water cooled solid breeder blanket with its thickness of 0.6 m for TBR > 1.05
- ✓ κ_{95} = 1.65 for vertical stability with conducting shell

Segmented maintenance scheme. Re-use & recycle of components.

B_T^{max} > 12 T is achieved, based on both Nb_3Sn or Nb_3Al, S_m = 800 MPa

Poster

Advanced divertor study will provide new options of the divertor configuration.

Physics and Engineering issues of *Short-SXD* has been studied in SlimCS:

- Interlink divertor coils are required: Nb_3Al SC is preferable for React&Wind

⇒ SC filament size should be reduced, and IL-coil support for EM-force is required.

- f_{exp} and $L_{//}$ *to target* are increased along the divertor leg: max. 19 and 2 times.
- Power handling has been investigated by SONIC for P_{FP}= 3GW reactor(P_{out}=500MW)

⇒ Radiative area is narrow poloidally, and efficient to produce full detachment.

Surface recombination is dominant near the separatrix due to large ion flux.

Conv. divertor is the first choice: Advantages and issues in adv. divertors are studied as alternative.