Session: Divertors in next step devices (5 talks)

- **ITER design issues**

⇒ **conventional DEMO design(EU-DEMO1/Flex-DEMO, JA-DEMO, CFETR, K-DEMO, Helical devices):**
Power handling scenario (continuing from yesterday DEMO session)

- **Compact Pilots Plant:** Expecting P_{fusion}, thermal power handling in core&divertor with existing technology?
- Modelling issues (impurity seeding ⇔ detachment ⇔ divertor size & geometry) for ITER/DEMO condition
- Physics and Simulation issues are mostly presented. How about Engineering and Technology issues?

ITER design and operation:

[82] The first ITER tungsten divertor: what do we hope to learn? (SOLPS-ITER and design)
PITTS, Richard (ITER Organization)

DEMO design (of CFETR):

[52] Recent progress on divertor physics design of CFETR (SOLPS)
DING, Rui (Institute of Plasma Physics, Chinese Academy of Sciences)

Compact Pilot Plant (CPP) concepts (US activity):

[54] A strategy to develop power exhaust solutions for tokamaks beyond ITER
CANIK, John (Oak Ridge National Laboratory)

Contributions to conventional and alternative divertor designs:

[83] Power exhaust studies in the Divertor Tokamak Test facility (SOLEDGE2D+Eirene)
VIANELLO, Nicola (Consorzio RFX, Associazione Euratom-ENEA sulla Fusione)

[14] The physical design of EAST lower tungsten divertor by SOLPS modeling (SOLPS)
SANG, Chaofeng (Dalian University of Technology)

Divertors in next step devices (1/3)

ITER design and operation:

[82] The first ITER tungsten divertor: what do we hope to learn? (SOLPS-ITER)
PITTS, Richard (ITER Organization) Note: topics from Nucl. Mater. Energy 20 (2019) 100696

Impurity seeding with relatively low-Z: N2/Ne is baseline scenario: Ar, Kr, Xe to increase P_{rad}^{main}?

Target heat load scaling in (partial) detachment

- q_{target} & Γ_{target} as a function of p_n (at divertor exhaust slot): extending to DEMO
- P_{rad}^{div}, P_{rad}^{sol} and the ratio as a function of p_n: it is different in higher Z
  p_n (// transport model can explain? Effects of Dome/Reflector/Baffle geometry.

Model of χ and D (value and profile), and scaling for $\lambda_{q//}$ and S (dissipation in divertor).

- $T_{e,i}^{target}$, $n_{e,i}^{target}$, p^{target} profiles in partial detachment, and what determines the detachment width

Divertor size (length) and the geometry (baffle/dome) to DEMO can be determined from ITER?
P_{sep}/R~16MW/m: closer geometry can be simplified? P_{sep}/R~30MW/m: Leg length can be reduced?

Detachment modelling: plasma pressure drop vs T_e in modelling

- Elastic collision with molecular is necessary to reduce T_e to 0.5eV-level?
- Modelling of volume recombination & MAR? Photon transport/absorption model?
- Drift effects are necessary to simulate experiment profile of detachment plasma?

Long operation lifetime issue:

- Restrictions of max. q_{target} :W-recrystallization, edge & shaping design, Net-surface erosion (DEMO)

Transient heat loading and mitigations→ DEMO will be designed based on ITER experiences.

- ELM mitigation and suppression scenario to DEMO (QH etc., RMP, pellets)
- Disruption mitigation/avoidance

→ Influence on DEMO design: design of baffle coverage & geometry, and limiter.

Divertors in next step devices (2/3)

DEMO design:

[52] Recent progress on divertor physics design of CFETR (SOLPS)

DING, Rui (Institute of Plasma Physics, Chinese Academy of Sciences)

- Simulation results for P_{fusion}~2GW, Psep~200MW (P_{sep}/R~28MW/m)
- Divertor size and geometry: ITER-like and Long leg geometry (1.7, 2.4m), SAS geometry
  - detachment profile (partial detachment)

→ Divertor size and geometry (baffle, dome, SAS: target geometry or tightness?) for optimization.

Compact Pilot Plant (CPP) concept (US): SPARC …

[54] A strategy to develop power exhaust solutions for tokamaks beyond ITER

CANIK, John (Oak Ridge National Laboratory)

- High Bt and Compact fusion concept P_{fus}~? (>50MW)
- High confinement (HH=1.5-1.8) and β_N (f_{BS}~qβ_N) with high n_e and large radiation fraction

→ large/small Gap? Control (heating, CD, momentum, impurity etc.) for high performance plasma?

→ P_{sep}/R and Divertor size and geometry optimization (conventional double null or advanced mag. Geometries?) in compact space?

Divertors in next step devices (3/3)

Contributions to conventional and alternative divertor designs:

[83] Power exhaust studies in the Divertor Tokamak Test facility (SOLEDGE2D+Eirene)
VIANELLO, Nicola (Consorzio RFX, Associazione Euratom-ENEA sulla Fusione)

- SF configuration able to reach pure D2 detachment at higher PSOL
- Detachment is obtained in all the configurations with reasonable seeded impurity concentration: snow-flake solutions providing lower concentration at the separatrix.

Performance of power exhaust beyond conv. magnetic concepts such as Double null, Longer-leg, SAS:

- Reduction in T_e^{div} & T_i^{div} *over whole target area* ("full detach") more than $q_{target} \leq 10$ MW/m^2.
- Stable control of *Radiation peak (radiation volume) and Impurity* in the divertor leg.
- Enhancement of *energy and particle Diffusions* in the divertor.
- Robust control of *the magnetic null position* and *the plasma shape*.

In addition, good effect on edge plasma control such as mitigating ELMs (particularly for SFD)

[14] The physical design of EAST lower tungsten divertor by SOLPS modeling (SOLPS, DIVIMP for W)
SANG, Chaofeng (Dalian University of Technology)

Locations of impurity puff (SOL or Private), and Ar and Ne seeding for the power exhaust.

W sputtering and W impurity transport(DIVIMP, and SOLPS).

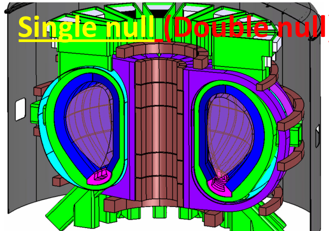
(Quasi-snowflake is assessed)

Power exhaust (P_{heat} = 300-500MW) is required for recent DEMO design

From ISFNT-13 Asakura (2019) PL7

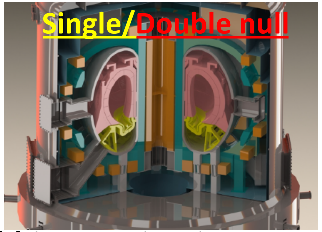
Conducting shell/Feedback coils for vertical stability is necessary for **high-κ. For steady-state operation, **P_{aux}** is increased with **I_p**.*

	Parameters	EU DEMO1**	JA DEMO	CFETR (2nd step)	K-DEMO (1st ph.)	ARIES-ACT1	ITER (inductive)
Size & Configuration	R_p / a_p (m)	**8.9 / 2.9**	**8.5 / 2.42**	**7.2/ 2.2**	**6.8/ 2.1**	**6.3/ 1.6**	6.2 / 2.0
	A	3.1	3.5	3.3	3.2	3.9	3.1
	I_p (MA)	**19.1**	**12.3**	**13.8**	**12.3**	**11**	**14**
	B_T / B_T^{max} (T)	4.9 / 12.2	5.94 / 12.1	6.5/ 14	7.4/ 16	6.0/ 11.8	5.3 / 12
	κ_{95}	1.65	1.65	2.0	1.8	2.2	1.7
	q_{95}	3.5	4.1	5.5	7.3	4.5	3
Heating	**Operation**	Pulsed 2 hrs	Steady-state	Steady-state	Steady-state	Steady-state	~400 s
	P_{fusion} (MW)	**1998**	**1462**	**974/ 2192**	**1488**	**1813**	500
	P_{aux} (MW)	50	84	82/ 78	119	45	73 (installed)
	P_{heat}:P_α+P_{aux} (MW)	**450**	**376**	**277/ 516**	**416**	**408**	~150
	Av. Neutron (MWm^{-2})	~1	~1	~1/~2.2	~2	~2.5	0.5



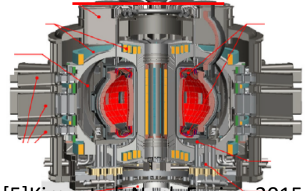
** [1] EU-DEMO PDD 2019

[2]Sakamoto, et al. IAEA FEC 2014,
[3]Tobita, et al. Fus. Sci. Tech. (2018)



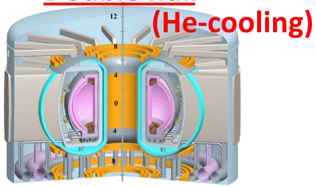
[4]Zhuang, et al. Nucl. Fusion 2019

Double null



[5]Kim, et al. Nucl. Fusion 2015
[6]Park, et al. Nucl. Fusion 2019

Double null
(He-cooling)



[7] Najmabadi, et al., Fus. Sci. Tec. 2013

Approaches of increasing f_{rad}^{main} and f_{rad}^{div} *in larger* P_{sep}/R are necessary

From ISFNT-13 Asakura (2019) PL7

Divertor power handling is determined by requirements of f_{rad}^{main} and the plasma performance:

- Development of *larger-size* and *closer divertor geometry* is a conventional approach.
- Double null or Advanced magnetic geometries will significantly affect engineering & technology issues.

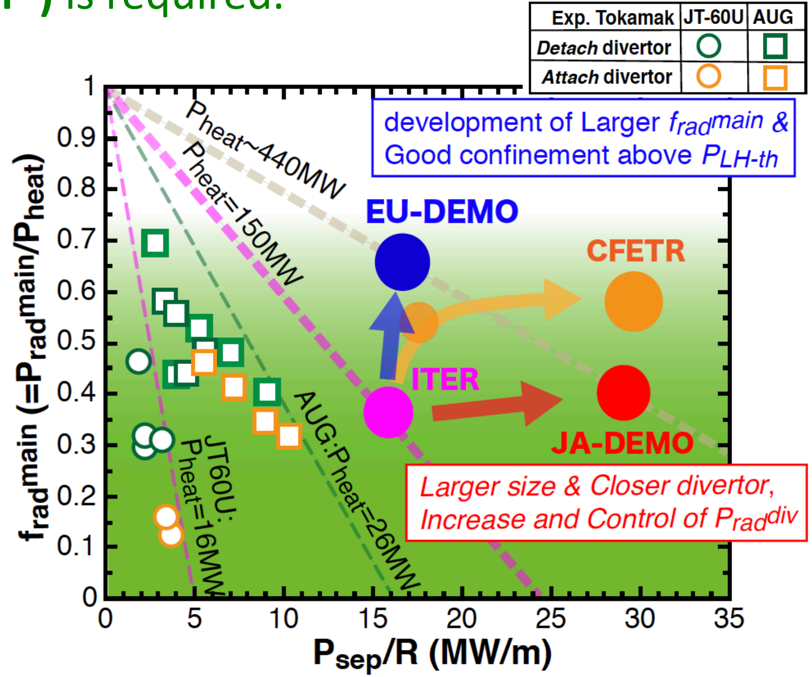
Line-ave. n_e for DEMO is lower than that of ITER ($1\text{x}10^{20}\text{m}^{-3}$) due to lower Greenwald-densities:

- Plasma detachment at **low $n_e^{sep} \sim n_e^{ped}/3$ ($2\text{-}3\text{x}10^{19}\text{m}^{-3}$)** is required.

Single null divertor DEMO design

	Parameters	JA-DEMO [8] High-k	EU-DEMO1 [9] 2017	CFETR [4] P_{fus}~2GW
Power exhaust	**line-n_e^{main} (10^{20}m^{-3})**	**0.86**	**0.87**	**0.87**
	n^{GW} (10^{20}m^{-3})	0.73	0.72	0.91
	n_{imp}/n_e (%)	0.6 (Ar)	0.039 (Xe)	0.5 (Ar)
	P_{heat} (MW)	**435**	**457**	**516**
	P_{rad}^{main} (MW)	177	306	295
	P_{rad}^{main}/P_{heat}	**0.41**	**0.67**	**0.57**
	P_{sep} (MW)	258	154	221
	P_{sep}/R_p (MWm^{-1})	**30**	**17**	**31**

Note: n_e^{ped} was required less than $0.9\text{x}n^{GW}$ in JET-ILW & AUG experiments [15]



[11] JT-60U: Asakura, et al. Nucl. Fusion (2009). [12] AUG: Kallenbach, et al., Nucl. Fusion (2015).
[13] A. Huber, et al., Nucl. Matter. Energy (2017).

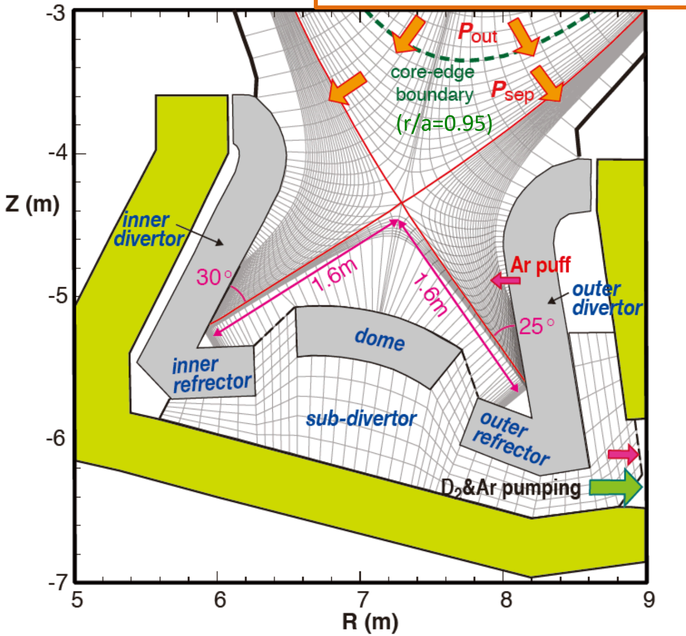
Power exhaust simulation in DEMO divertors

From ISFNT-13 Asakura (2019) PL7

Conventional design concepts are based on the ITER divertor: θ_{div}~38°(in)/24°(out)

- ***Outer leg lengths* are similar, L_{div}=1.6-1.7 m (~1.6 times longer than ITER).**
- Baffles cover divertor plasma for high P_{sep}/R design ⇔ Open and Shallow geometry for EU-DEMO1 (ITER-level P_{sep}/R) to increase tritium-breeding area and reduce weight & process for remote maintenance.

JA-DEMO: SONIC ***P_{out}* =250/*P_{sep}*~235MW**

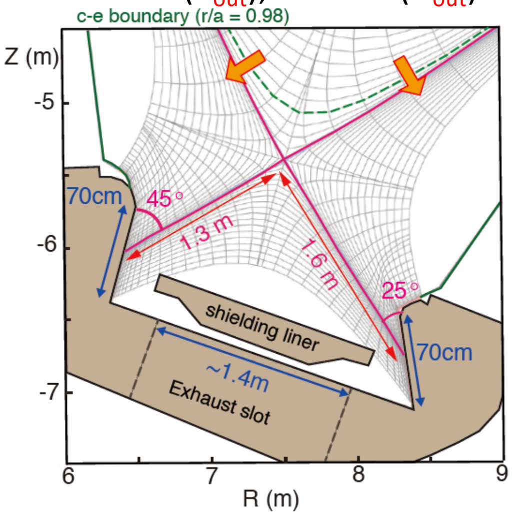


[8] Asakura, et al. Nucl. Fusion (2017).

EU-DEMO1: SOLPS

***P_{sep}~P_{out}* =150MW**

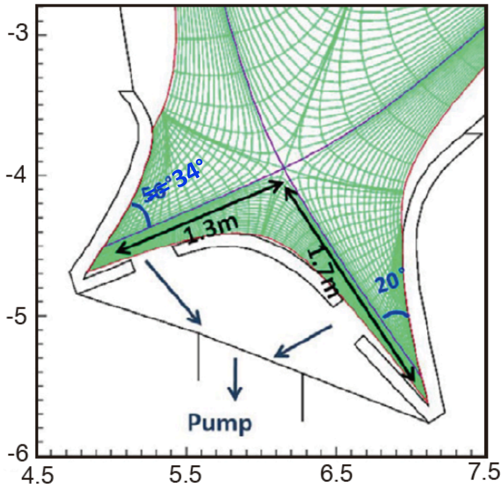
Power (P_{out}), Particle (Γ_{out}) fluxes are given at core-edge boundary.



[14] Subba, et al. Plasma Phys. Control. Fusion (2018).
[15] Subba, et al. Final Report 2019

CFETR: SOLPS

***P_{out}*~300MW?**



[4] Zhuang, et al. Nucl. Fusion 2019

SOL heat flux profile becomes large and narrow in DEMO

From ISFNT-13 Asakura (2019) PL7

T_e^{sep} & T_i^{sep} increase to 0.37 & 0.83 keV (in SONIC sim. for JA-DEMO): 2-3 times larger than ITER

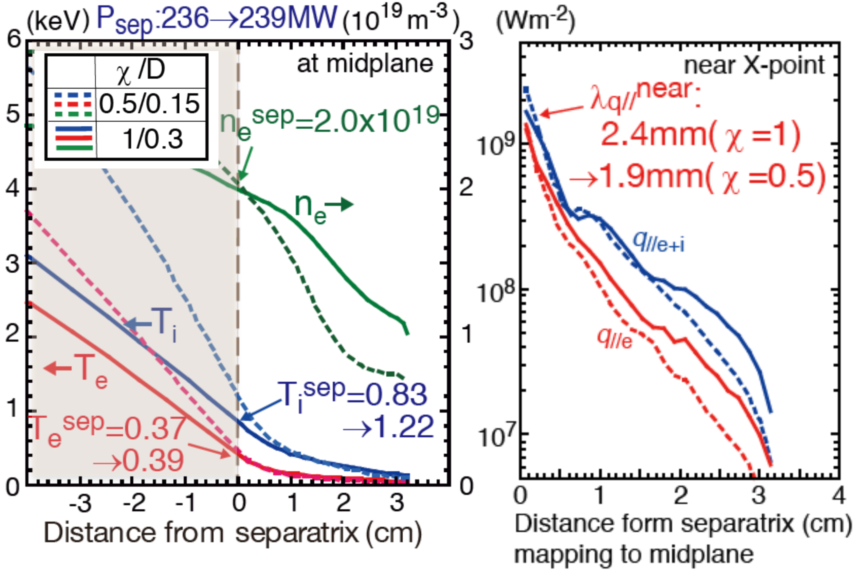
⇒ $\lambda_{q//}$ = 2.4mm for the same χ (=1m²/s) and D (=0.3m²/s) as ITER ($\lambda_{q//}$ = 3.4 mm)[16].

- Reduction in χ and D to half values（χ = 0.5m²/s, D = 0.15m²/s）⇒ $\lambda_{q//}$ is reduced to 1.9mm.

$q_{//}$ profiles in DEMOs are *wider than Eich's scaling* (~1mm)[18] *and Goldston's model* (~1.5mm)[19].

JA-DEMO(SONIC)[8]: P_{out}=250MW, low n_e^{mid}

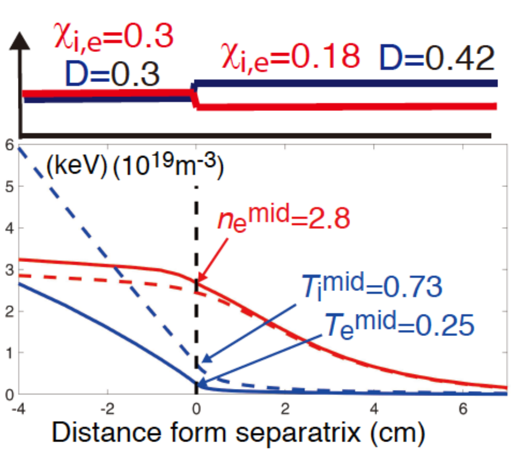
$\lambda_{q//}^{near}$ is reduced from 2.4 to 1.9mm



EU-DEMO1(SOLPS)[15]

P_{out}=150MW, low n_e^{mid}

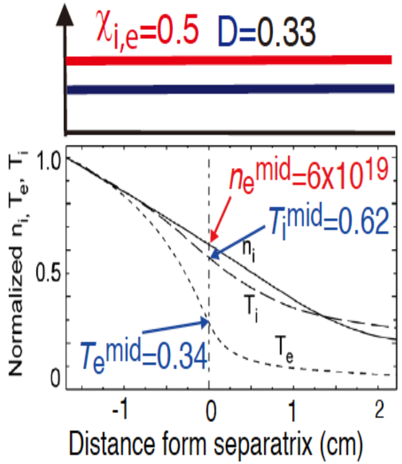
$\lambda_{q//}^{near}$ ~3mm



ARIES-ACT1(UEDGE)[17]

P_{out}=320MW, high n_e^{mid}

$\lambda_{q//}^{near}$ ~2mm



[16] Kukushkin, et al. J. Nucl. Mater. (2013) [17] Rensink, et al. Fus. Sci. Tec. (2015) [18] Eich, et al. Nucl. Fusion (2013). [19] R. Goldston, Nucl. Fusion (2012).

Design concepts for water-cooling DEMO divertor:

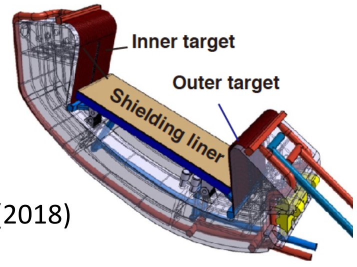
From ISFNT-13 Asakura (2019) PL7

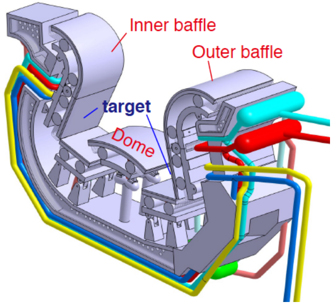
W-PFC & CuCrZr-pipe is a common baseline design. Divertor weight is increased.

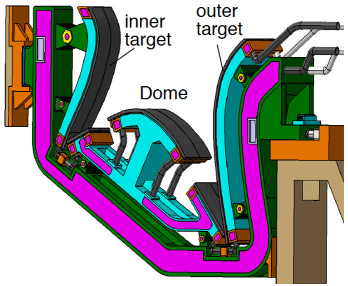
		EU DEMO1 [1]	JA DEMO[8, 20]	CFETR (2^{nd}step) [4]	K-DEMO (1^{st}ph.) [21]
	Number at units in a cassette	48	48	80	Upper: 32/Lower: 32
	Weight of one cassette (ton)	11	23	11	TBD
Target	PFC & Heat sink	**W&CuCrZr***1	**W&CuCrZr**	**W&(CuCrZr/ODS-Cu**/RAFM)*2	**W&(CuCrZr** or RAFM)
	Water T(°C)/P(Mpa)	130/ 3.5	200/5	140/5	290/ 15
	Dose on pipe/fpy (dpa)	<10	<1.5	TBD	<1.2
Dome/Baffle	PFC & Heat sink	**W&CuCrZr (liner)**	W&F82H	**W&(CuCrZr/ODS-Cu**/RAFM)	W&RAFM
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15	140/5	290/ 15
	Dose on pipe/fpy (dpa)	<10	<8.5	TBD	< 10.9
Cassette	Material	EUROFER97	F82H	RAFM	RAFM
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15	140-180/ 5	290/ 15
	Dose on struct. material/fpy (dpa)	<6	<3	TBD	TBD

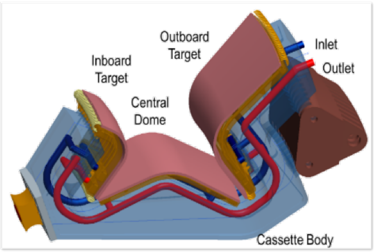
*1: Alternative target concepts: liquid metal and He-cooling.

*2: He-cooling is an option.









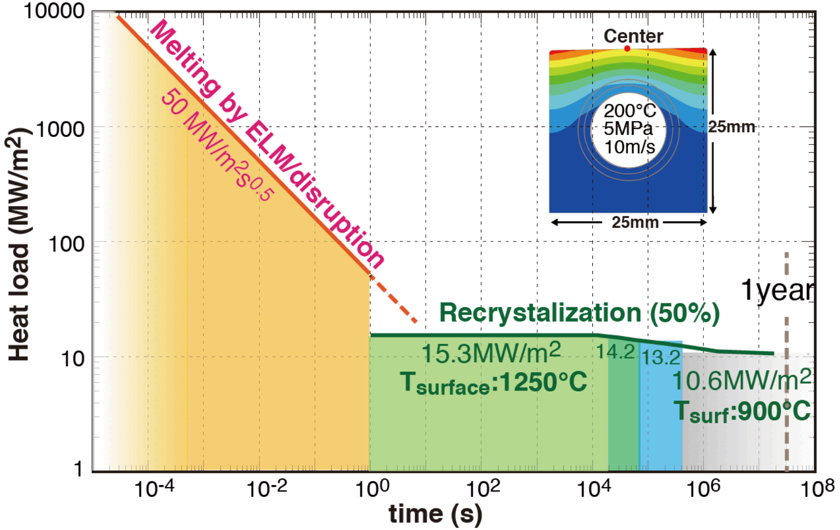
[20] Asakura, et al. Fus. Nucl. Design (2018)
[21] Kwon, et al. ISFNT14, O1-2.2.

Power handling of W-PFC target for year-long operation:

Reduction in recrystallization temperature and *net-erosion in partial attached plasma*

- **W-recrystallization will progress even at lower temperature (~900˚C)** [22,23]:
  Peak q_{target} should be reduced to <10MWm^{-2} for the coolant temperature of 200˚C.
- **Net erosion will be increased to *a few mm level* (if T_e^{div}~20eV at attached area):**
  Reduction in T_e & T_i of attached plasma is necessary such as "pronounced detachment: AUG"[12]
  Experiment data and Modeling of erosion & transport (finite-Larmor effect[24]) must be improved.

Operation limit of steady-state and transient heat loads on W-PFC



Simple estimation of net erosion: 90% re-deposition

- **Net erosion (Δd) becomes a half of W-width（d:5mm）**

Net erosion/year(mm)	T_e=5eV	10eV	20eV
DEMO (steady state)	0.15	1	**2.5**
ITER(400s, 2000 shots)	0.004	0.026	**0.064**

attach plasma area Γ_i~10^{23} m^{-2}s^{-1}, ~20eV <Z>=4,
n_{Ar}/n_i =0.2%, assuming net erosion: R_{net}=0.1

Erosion yield with Ar imp. Y_iC_i ~4x10^{-4} (at 20eV)[25]

$$\Delta d\ (\mathrm{mm}) = 4.95\times10^{-19} R_{net} * Y_iC_i * \Gamma_i * t(\mathrm{year})$$

[22] Alfonso et al., J. Nucl. Mat. (2014). [23] Alfonso et al., Fus. Eng. Des. (2015).
[24] Homma, et al., Nucl. Mater. Energy. (2017). [25] Kallenbach, et al., J. Nucl. Mat. (2011).

Design constrains of W and Cu-alloy heat sink under neutron irradiation

From ISFNT-13 Asakura (2019) PL7

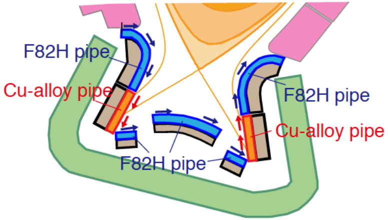
Design constrains of the power handling:

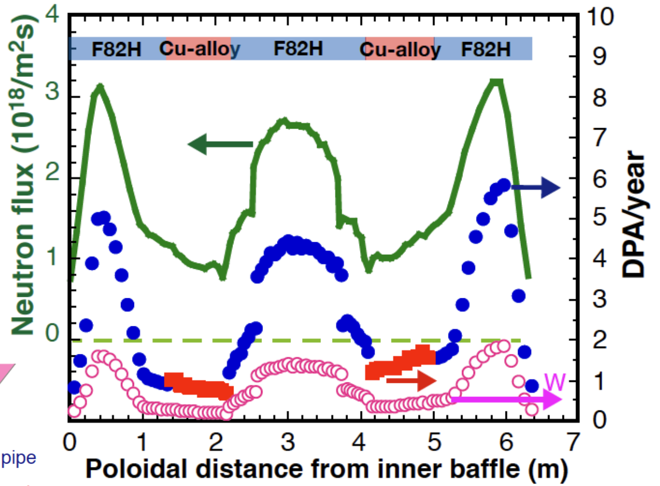
- **firstly determined by mechanical property of Cu-alloy**

⇒ it is applied at high heat flux and low neutron flux area:
1~1.5 dpa/fpy near the strike points:
ITER technology (W&CuCrZr target) can be applied, while replacement will be 1-2 years.

⇒ Design criteria, systematic database of the properties, and their improvement.

- Reduction in W thermal conductivity will be acceptable up to several dpa (~3 years).
1 dpa (W)=2.87 dpa (CuCrZr&ODS-Cu)





Neutronics analysis for JA-DEMO [8]
(MCNP-5 code, FENDL-2.1 nucl. database)

Softening → Embrittlement → Thermal cond. reduction

Heat sink/ Coolant pipe	Yield strength at RT (MPa)	T-threshold (˚C)	Radiation-induced (dpa) hardening	Radiation-induced (dpa) softening	Embrittlement by transmuted He (dpa)	Reduction (20%) in Thermal cond. by transmuted product (dpa)
Pure-Cu	~60 MPa	---	~0.1	---	**6 (at 350˚C)**	10
CuCrZr	**>400 MPa**	**280**	**~0.2**	**~1**	40appm limit with 7appm/dpa	**10**
ODS-Cu(GlidCop®[26])	>400 MPa	300	~0.2	1~2		10

[26] Tokitani, et al. ICFRM-19 (2019). [27] S.J. Zinkle et al., Fusion Materials DOE/ER-0313/16 (1994), [28]B.N. Singh et. al, J. Nucl. Mater. (1993).

1. Introduction: Power exhaust and divertor concepts for DEMO design

Conventional divertor concepts for recent DEMO design (P_{fusion}=1.5-2GW, R_p=7-9m):

Large radiation fraction ($f_{rad}= P_{rad}/P_{heat} \geq 0.8$) is required by impurity seeding to reduce $q_{target} \leq$ 10MW/m^2, which is handled by ITER technology (W-Plasma Facing Component & Cu-alloy pipe with water cooling).

Thermal power exhaust

Radiation loss (imp. seeding)

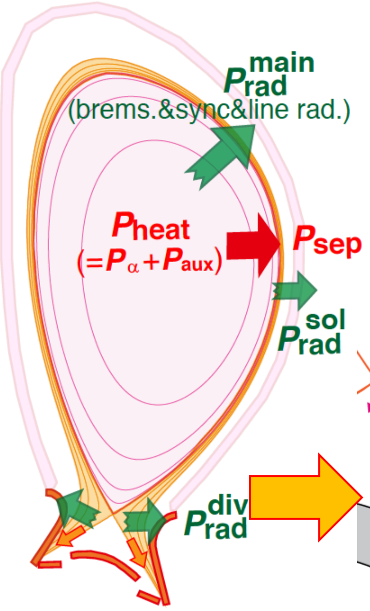
main plasma ***SOL & divertor***

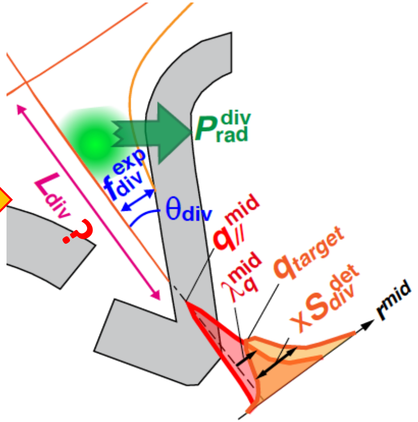
Geometry design

$$q_{target} = \frac{P_{heat}}{R}(1 - f_{rad}^{main}) \frac{(1 - f_{rad}^{div})}{4\pi \lambda_q^{mid} S_{div}^{det}} \frac{\sin\theta_{div}}{f_{div}^{exp}}$$

Single null divertor

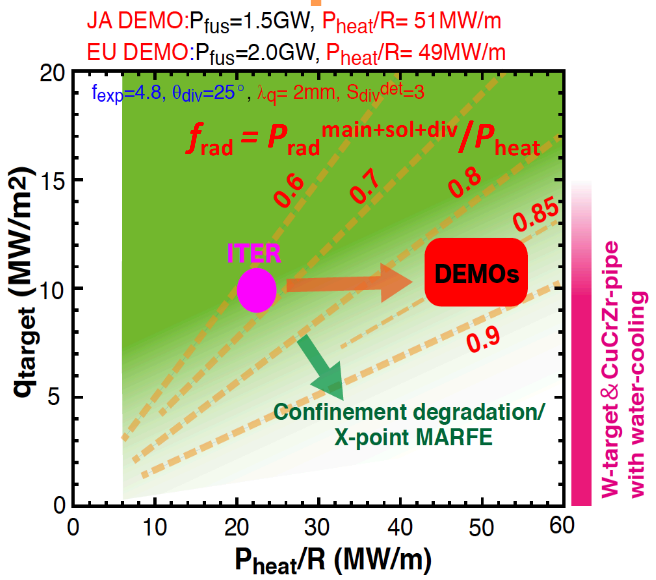
transport & detachment





$f_{rad}^{main} = P_{rad}^{main}/P_{heat}$

$f_{rad}^{div} = P_{rad}^{div}/P_{sep}$



4. Advanced magnetic configurations for DEMO (short note)

Introducing extra-divertor coil(s) with driving I_{div} in the reversal-I_p direction, *fieldline length* and *flux expansion* are increased in the divertor and target ⇒ enhance P_{rad}^{div} and plasma detachment.

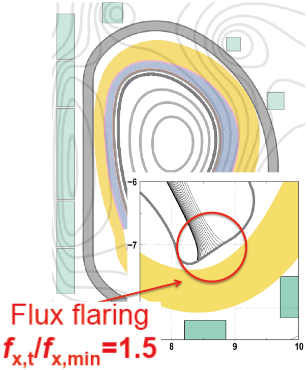
(1) X-Divertor: increases flux expansion near the outer target.

(2) Super-X Divertor: fliedline is extended to outboard to increase R_{target} (increasing wet area).

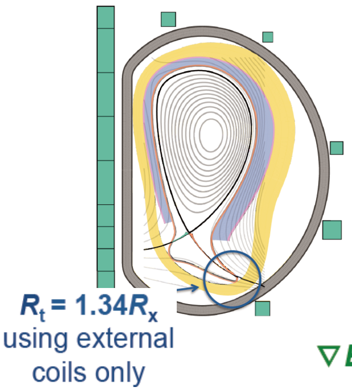
(3) Snowflake Divertor: fieldline and flux expansion near the X-point are increased. Enhancement of P_{rad}^{div} volume and plasma diffusion will be also expected locally in low ∇B_θ (near X-point).

Costs: Poloidal Field Coil currents are significantly increased for the external-TFC design
⇔ Installation of “Interlink-coil (ILC)” will increase cost on engineering and technology (extending TFC size, SC-coil React&Wind, ILC-fabrication and fixing for large vertical-force, etc.)

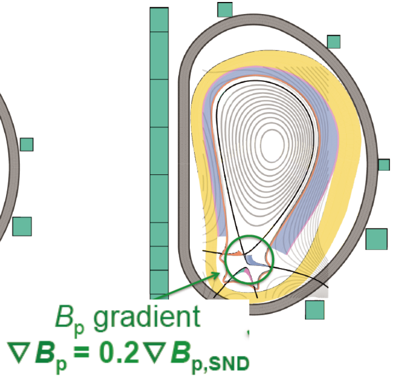
X-Divertor (XD)



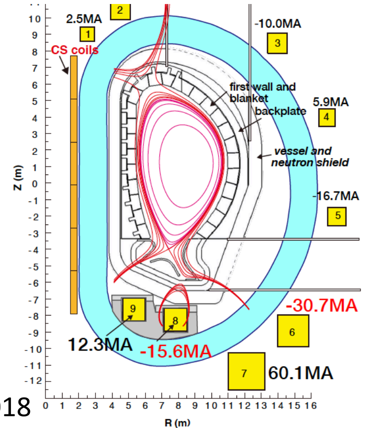
Super-X Divertor (SXD)



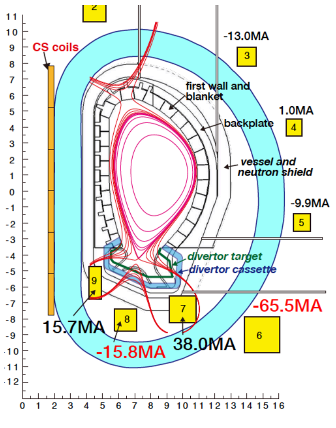
Snowflake Divertor



Short-SXD (2 interlinks)



SFD (3 interlinks)



[35] Reimerdes, et al., EUROfusion Consortium (2016)

[36] Asakura, et al, SOFT 2018

Advantages of power exhaust and control will be confirmed in exp.& sim.

From ISFNT-13 Asakura (2019) PL7

Performance of power exhaust <u>beyond conv. magnetic concepts</u> such as Double null, Longer-leg, SAS:

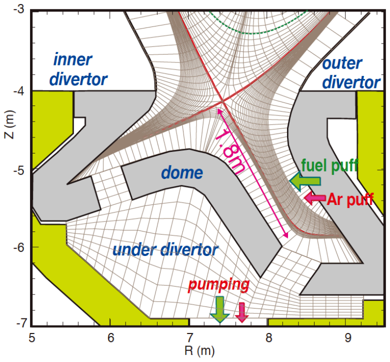
- Reduction in T_e^{div} & T_i^{div} *over whole target area* ("full detach") more than $q_{target} \leq 10$ MW/m^2.
- Stable control of *Radiation peak (radiation volume) and Impurity* in the divertor leg.
- Enhancement of *energy and particle Diffusions* in the divertor.
- Robust control of *the magnetic null position* and *the plasma shape.*
- Good effect on edge plasma control such as mitigating ELMs (particularly for SFD)

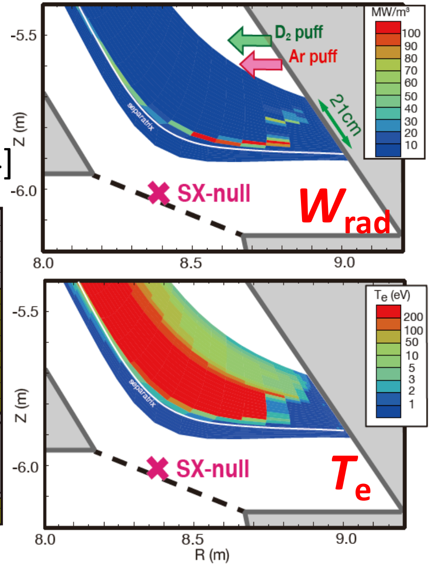
<u>Short-SXD:</u>

High P_{sep}=285MW,

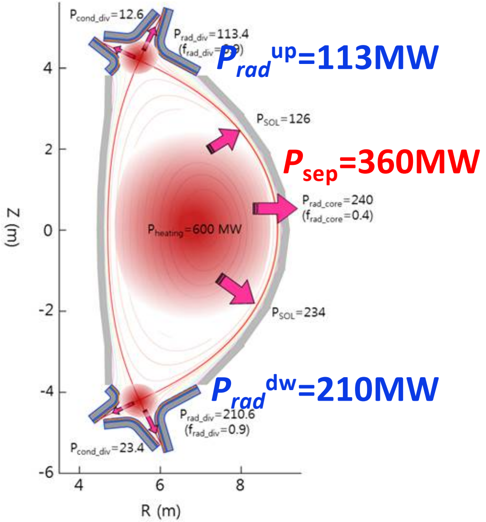
f_{rad}=0.78,

low n_e =2x10^{19}m^{-3}[34]

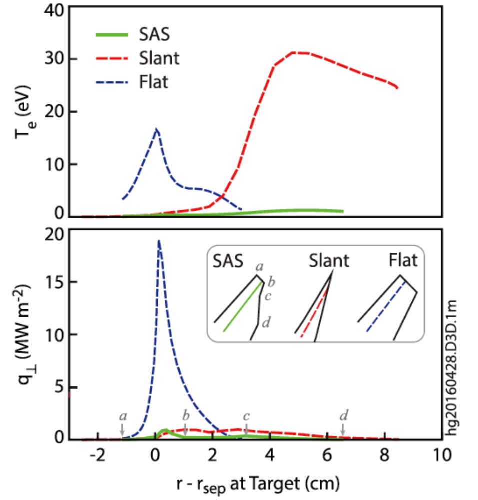




<u>Double-null div.(K-DEMO)</u>[21]



<u>Small Angle Slot divertor</u>[36]



[37] Guo, et.al., Nucl. Fusion (2017)

Summary: Power exhaust and Divertor design for water-cooling concepts

Conventional divertor concepts for recent DEMOs (P_{fus}=1.5-2GW, R_p=7-9m) were summarized:

- Requirements of f_{rad}^{main} and the plasma performance will determine **divertor design concept**. Approaches of two concepts, i.e. increasing f_{rad}^{main} (for ITER-level P_{sep}/R) and f_{rad}^{div} (for larger P_{sep}/R ~30MWm), will contribute to optimize future DEMO and power plant designs.

Power exhaust simulations for DEMO divertor:

Simulation studies suggested that the total radiation fraction ($f_{rad} = P_{rad}/P_{heat} \geq 0.8$) is required to reduce both peak-q_{target} and $T_{e,i}$ $\Rightarrow$ improvements of λ_q (χ) and detachment models are required.

- Outer leg length is similar: L_{div}=1.6-1.7 m and Width of $q_{//}$ profile is $\lambda_{q//}$=2-3mm.
- Geometry effects (ITER like closer baffle/ without baffle) on plasma detachment profile and the required radiation will be important key to operate the divertor in the low n_e^{sep} range.

ITER-like target (W-PFC and Cu-alloy heat sink) is a common baseline design:

- For a year long operation, Re-Crystallization and Net-Erosion on W, and Mechanical property of CuCrZr heat sink under n-irradiation will be anticipated $\Rightarrow$ restrictions of q_{target} , $T_{e,i}$ and $T_{surface}$.
- Integrated design of divertor target, cassette and coolant pipe routing has been developed: Two routes for W-PFC&Cu-alloy heat sink (lower-T) and RAFM heat sink for Baffle/Cassette (higher-T) $\Rightarrow$ Coolant-T (130-200°C) and Cu-alloy property under n-irradiation are design issues.
- Water-cooled target components (incl. joint/interlayer) for high n-irradiation should be developed.

Summary (continue)

Comment on Advanced magnetic configurations for DEMO:

Performance of power exhaust beyond conventional magnetic concepts such as Double null, Longer-leg, Small Angle Slot divertors, is expected in experiments and simulations.

5. Summary: Power exhaust and divertor design for JA DEMO

DEMO DESIGN
JOINT SPECIAL TEAM

Recent progress of Japanese DEMO design and Divertor concept were summarized.

High plasma performance of HH_{98y2}~1.3, β_N~3.4, f_{BS}~0.6, n_e/n^{GW} ~1.2 is expected with $(n_{Ar}/n_e)^{main}$ = 0.6% by impurity (Ar) seeding (**P_{rad}^{main}/P_{heat}=0.41**, slightly larger than ITER).

- Divertor power handling of reference concept (P_{sep}~250 MW, P_{sep}/R ~29 MW/m) and under sever conditions (P_{sep}, $P_{rad}^{sol+div}/P_{sep}$, χ) was studied in the *expecting low SOL n_e* (~1/3x n_e^{main} =2-3x10^{19}m^{-3}).

Plasma performance in the long-leg divertor by SONIC simulation:

- Partial detachment (outer) was produced for $P_{rad}^{SOL+div}/P_{heat}$ = 0.43 ($P_{rad}^{SOL+div}/P_{sep}$= 0.78)
  ⇒ large $q_{//}$ near SOL (r^{mid} < 1 cm) can be reduced by the partial detachment, and peak-q_{target} at attached region is also reduced less than 10 MWm^{-2}, which was simulated under *sever conditions, i.e.* increasing P_{sep} by 20% or reducing $P_{rad}^{SOL+div}/P_{sep}$ by 10%.
- Heat flux profile reducing χ = 1 ⇒ 0.5 m^2/s: λ_q^{SOL} (~2mm) is still larger than Eich's scaling
  ⇒ Impact of reducing χ, particularly for smaller $P_{rad}^{SOL+div}/P_{sep}$, is serious.
- Net-erosion in the *partially attached area* (T_e=20-30eV) will be a critical life-time issue of W-target in year-long operation ⇒ improvement of W transport model is on going.
- Impurity concentration in SOL : c_{Ar}^{SOL}（0.4-0.6%）is so far comparable to c_{Ar}^{main} in system code. Increasing $P_{rad}^{sol+div}$ with controlling dilution of the core plasma is required.

DEMO DESIGN
JOINT SPECIAL TEAM

Summary (2): Some issues in SONIC simulation and modelling

SONIC code (re-structuring to *Multi-Process Multi-Data*, i.e *multi-species*, renewing *plasma fluid-code including drifts*) and **modelling for DEMO plasma** (*erastic collision of atom and molecule, photon absorption, thermal force on impurity in low-collisional SOL*) are developped.

⇒ Power exhaust and divertor design, consistent with He exhaust, will be revised.

⇒ Restructure of the plasma fluid code (**SOLDOR in SONIC**) **incorporating drifts** is on going.

Improvement of simulation on the heat load profile *at the partial detachment* is necessary:

- Plasma modelling : distributions of diffusion coefficients, momentum loss process, etc.
- Empirical scaling of the detached heat load and the peak value will be used for design.

Control of radiation peak and detachment front *<u>in the long-leg</u>* is high priority issue:

- Impurity transport in SOL (low collision) - divertor (high collisional), and the shielding efficiency (thermal force vs friction force) are key issues to design the seeding scenario and divertor:

AUG

P_{rad}^{div}/P_{heat} >0.6