TECH/3-2Rb

Plasma Exhaust and Divertor Designs in Japan and Europe Broader Approach, DEMO Design Activity

Topics are selected from *Chapter 4: Divertor and Power Exhaust* in final report of Broader Approach (BA) DEMO Design Activity（DDA) Phase-I (2020 Feb.)

Nobuyuki Asakura[1], Kazuo Hoshino[2], Yuki Homma[3], Christian Vorpahl[4], Fabio Subba[5], Hiroyasu Utoh[3], Youji Someya[3], Satoshi Kakudate[1], Satoshi Suzuki[1], Yoshiteru Sakamoto[3], Ryoji Hiwatari[3], Mattia Siccinio[4], Gianfranco Federici[4], Jeong-Ha You[6]

1)National Institutes for Quantum and Radiological Science and Technology (QST), Naka, Japan
2)Graduate School of Science and Technology, Keio University, Hiyoshi, Yokohama, Japan
3) National Institutes for Quantum and Radiological Science and Technology (QST), Rokkasho, Japan
4) EUROfusion Programme Management Unit, Garching, Germany
5) Politecnico di Torino, Torino, Italy
6) Max Planck Institute for Plasma Physics, Garching, Germany

28th IAEA Fusion Energy Conference (virtual), 10-15 May 2021

Power exhaust concepts and challenges for JA and EU DEMOs

- EU and JA BA-DDA study covers common aspects of divertor physics and engineering design: water-cooled single-null divertor and appropriate geometry for plasma detachment.
- Both concepts handle similar thermal heating power (P_{heat}), and require large total radiation fraction ($f_{rad} = P_{rad}/P_{heat} \geq 80\%$) in order to reduce the peak heat load ($\leq$10 MWm^{-2}):

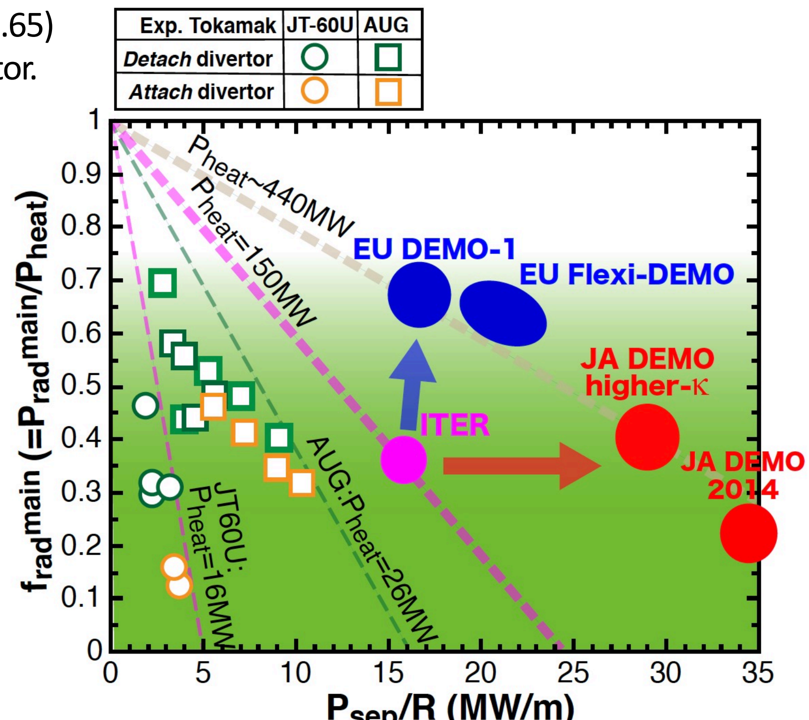
Divertor power handling is determined by requirements of f_{rad}^{main} and the plasma performance.

JA DEMO challenge (steady-state op.):
Lower I_p and higher HH with ITER-level f_{rad}^{main} ⇒
Large divertor power handling: P_{sep}/R ~30MWm^{-1}

EU DEMO challenge (pulse op.):
Higher I_p and ITER-level HH with large f_{rad}^{main} by high-Z seeding ⇒ ITER-level P_{sep}/R =17MWm^{-1}

JA DEMO higher-κ proposal (κ_{95}=1.75) [1] rather than JA DEMO 2014 (κ_{95}=1.65) [2] is shown: having advantages on power exhaust in main plasma and divertor.

	Parameters	JA DEMO [6]	EU DEMO [3]
Power exhaust	**line-n_e^{main} ($10^{20}m^{-3}$)**	**0.86**	**0.87**
	n^{GW} ($10^{20}m^{-3}$)	0.73	0.72
	$n_{imp}^{main}/n_e^{main}$ (%)	0.6 (Ar)	0.039 (Xe)+Ar
	P_{heat} (P_α+P_{aux}, MW)	**435**	**457**
	P_{rad}^{main} (MW)	177	306
	f_{rad}^{main} (=P_{rad}^{main}/P_{heat})	**0.41**	**0.67**
	P_{sep} (MW)	258	154
	P_{sep}/R_p(MWm^{-1})	**30**	**17**



[1] Asakura, et al. Nucl. Fusion (2017), [2] Sakamoto, et al. IAEA FEC 2014&18, [3] Wenninger, et al. Nucl. Fusion 2017.

Divertor design and Power exhaust simulation

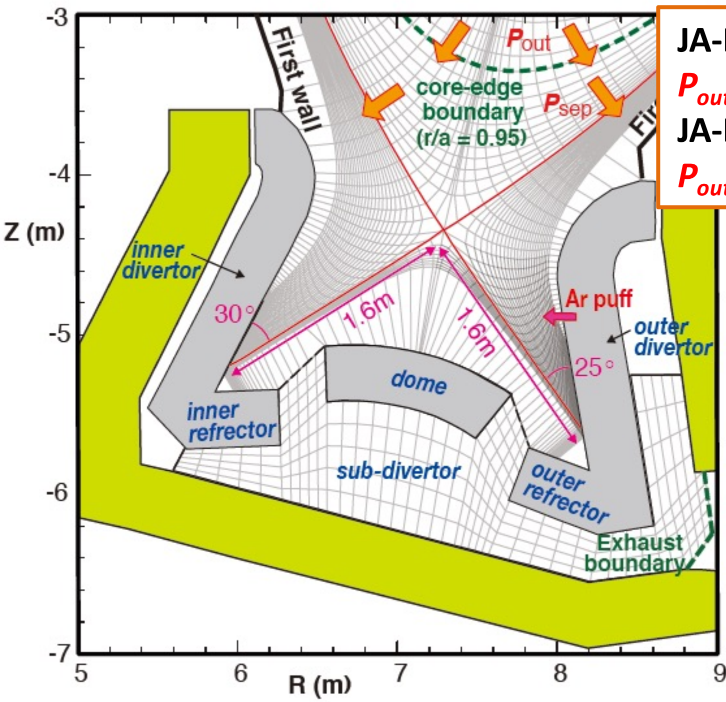
Conventional design concepts in JA & EU are based on the ITER divertor:

- Both DEMOs: Divertor leg is extended (outer L_{div}=1.6 m: 1.6 times longer than ITER).
- JA: Baffles cover divertor plasma for large P_{sep}/R handling ⇔ EU: Open and shallow geometry (ITER-level P_{sep}/R) to increase tritium-breeding area and reduce weight & process for remote maintenance.
- JA: Dome and reflectors are installed to enhance the neutral recycling near the strike-point.
  EU: Dome and reflectors are simplified ("liner") to reduce fast neutron flux to cassette and VV.

SONIC (JA-DEMO) and SOLPS-5.1 (EU-DEMO) simulations have been performed :

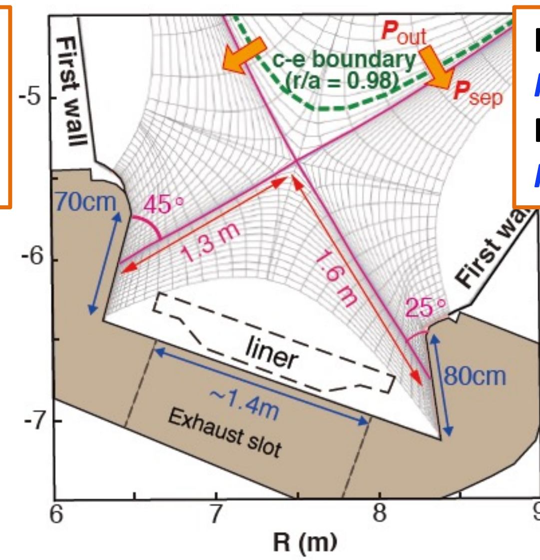
- Exhaust power (P_{out}= JA: 250-300 MW, EU:150-200MW) is given at core-edge boundary.

JA DEMO: SONIC simulation



JA-DEMO higher-κ:
P_{out} =250/P_{sep}~235MW
JA-DEMO 2014:
P_{out} =300/P_{sep}~283MW

EU DEMO: SOLPS simulation



Reference:
P_{out} = 150MW ~ P_{sep}
Power scan:
P_{out} = 180/200MW

Note) $q_{//}$ profile width near separatrix is similar ($\lambda_{q//}$~3mm).

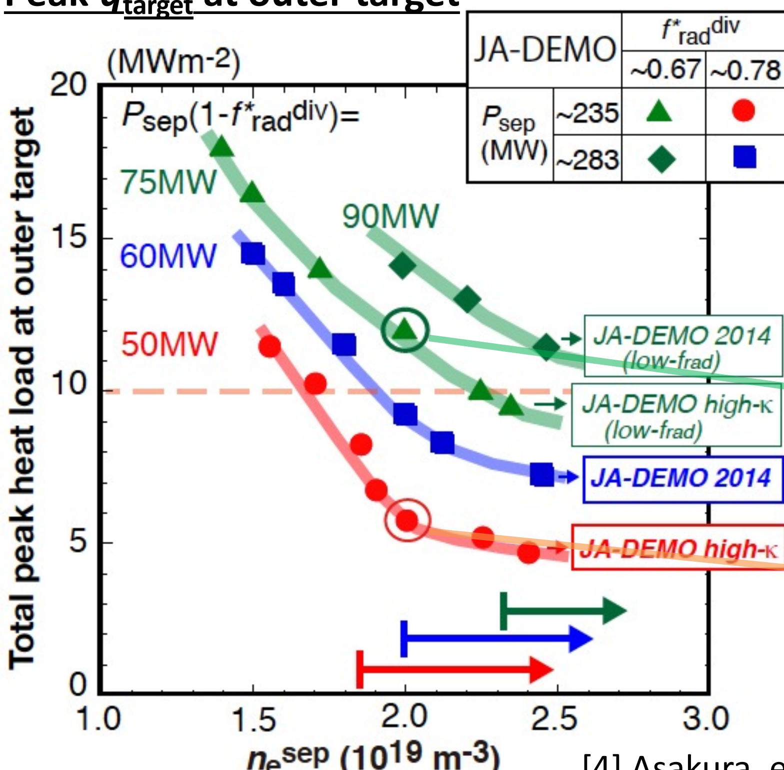
JA DEMO: Divertor operation in low density (n_e^{sep} = 2-3x10^{19}m^{-3})

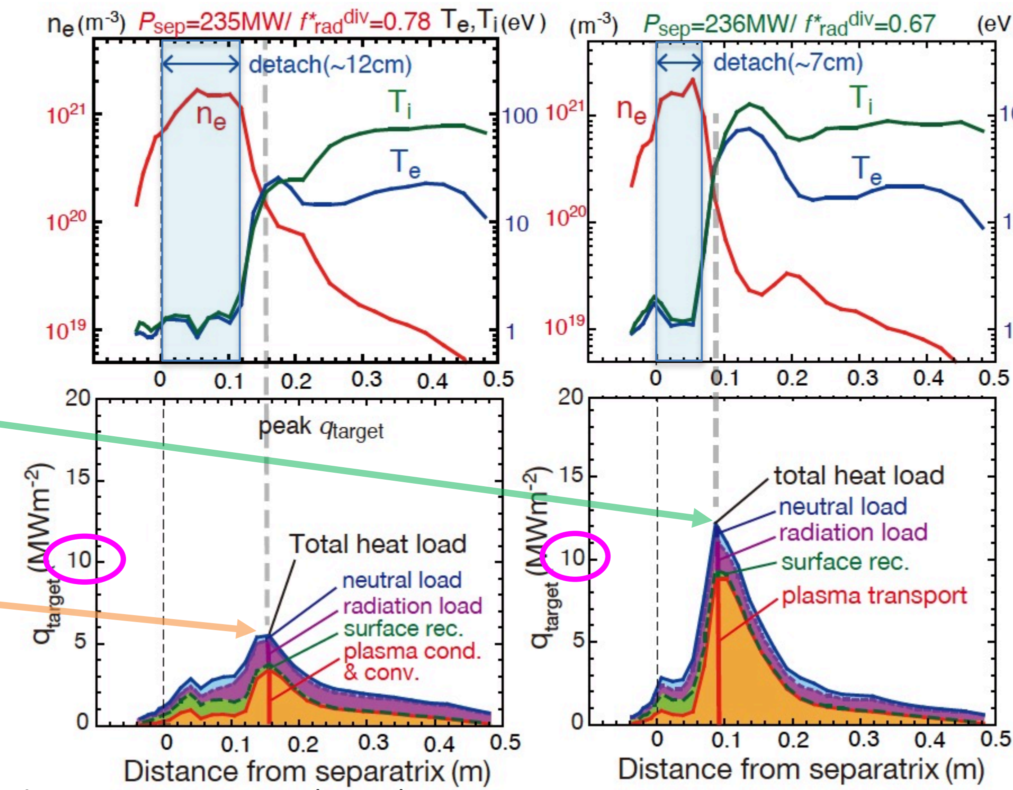
Heat load can be reduced within the operation range (q_{target} ≤ 10 MWm^{-2}) for $f^*_{rad}{}^{div}$~0.8

In each density scan, Ar seeding rate was adjusted to obtain a given $f^*_{rad}{}^{div} = (P_{rad}{}^{div}+P_{rad}{}^{sol})/P_{sep}$.

- **Higher-κ** (P_{sep}~235MW, $f^*_{rad}{}^{div}$ ~0.8) reduces q_{target} (≤ 6 MWm^{-2}), and allow enough operation margin.
- **JA DEMO 2014** (P_{sep}~283MW, $f^*_{rad}{}^{div}$ ~0.8):
  Decreasing *detachment width*, and increasing T_i and T_e *of the attached plasma*.
  ⇒ peak-q_{target} is increased, and margin of the power handling (≤ 10 MWm^{-2}) is reduced.
- ***Lower $f^*_{rad}{}^{div}$ ~0.7*** (P_{sep}~235 and 283 MW) cases:
  ⇒ **higher n_e^{sep}** (>2.3x10^{19}m^{-3} for *DEMO higher-κ*, >2.7x10^{19}m^{-3} for *DEMO 2014*) **is required**.

Peak-q_{target} at outer target





[4] Asakura, et al. Fus. Mat. Energy (2021).

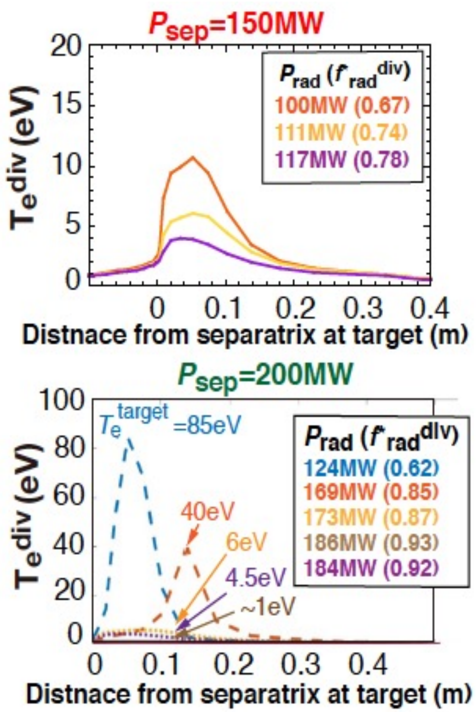
EU DEMO: Divertor power handling by Ar seed for P_{sep}/R =16-22 MWm^{-1}

Heat reduction was achieved for all cases by increasing C_{Ar}^{SOL} (= $n_{Ar}/n_e)^{SOL}$ = 0.5-2.5%

Geometry effect on plasma profile: *partial detachment* was not clearly seen in the open geometry.

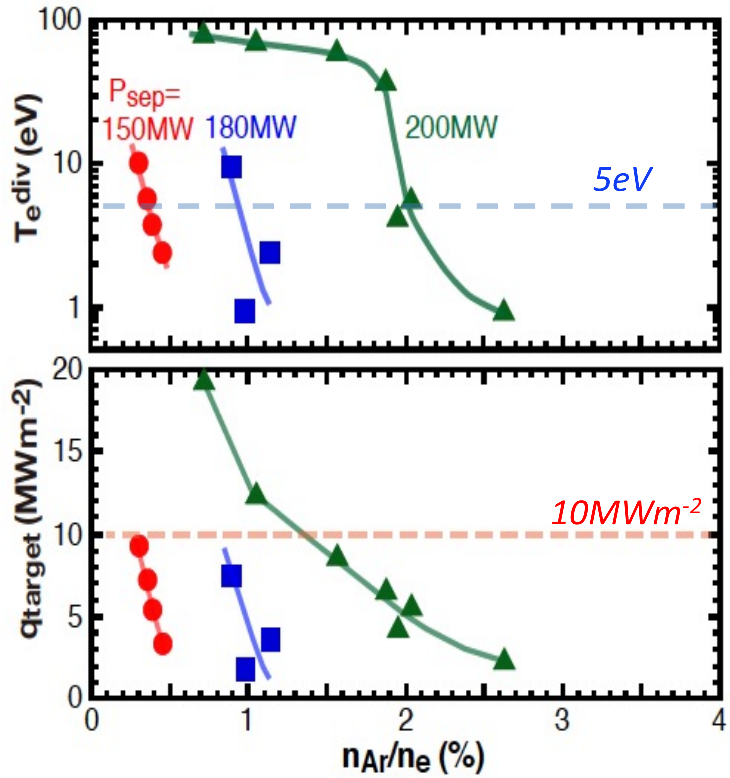
Baseline (P_{sep}=150MW): heat reduction ($q_{target} \leq 10MWm^{-2}$) was achieved by increasing $f^*_{rad}{}^{div} \geq 0.7$.
⇒ Low T_e^{div} ($\leq$ 5eV) was also produced *over wide outer target* for $f^*_{rad}{}^{div} \geq 0.8$ ($C_{Ar}^{sol} \geq 0.8\%$).

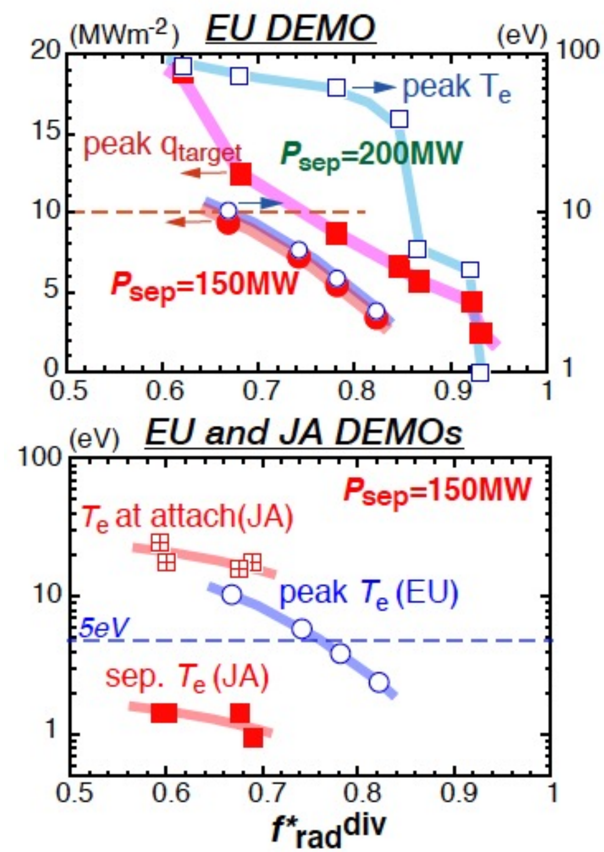
Larger P_{sep} case: q_{target} reduction was achieved ($f^*_{rad}{}^{div} \geq 0.75$) ⇒ low T_e^{div} ($\leq$ 5eV) was required in higher $f^*_{rad}{}^{div} \geq 0.9$ ($C_{Ar}^{sol} \geq 2\%$). Detachment (T_e^{div}~1eV) is seen in very high $f^*_{rad}{}^{div}$ ($\geq$ 0.93).



Ar seeding scan for P_{sep}=150/ 180/ 200MW

Note) n_e^{sep}~$2.8 \times 10^{19} m^{-3}$ is higher than JA-DEMO





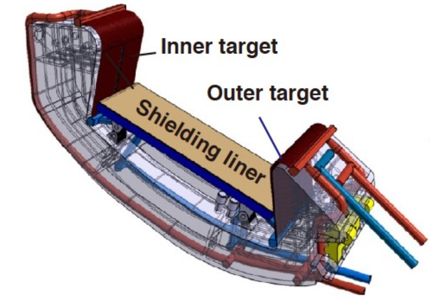
Design concepts for water-cooling DEMO divertor

W-PFC & CuCrZr-pipe is common baseline design based on the ITER divertor.

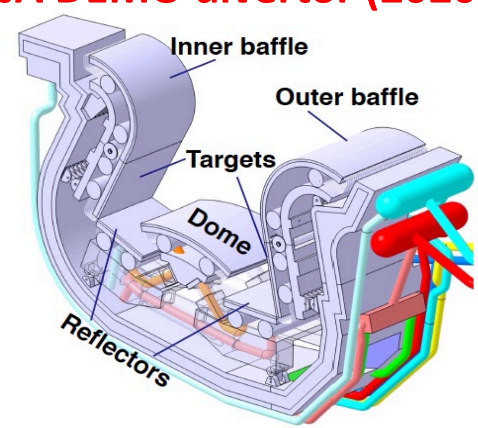
- ITER-like monoblock target is the first candidate for high heat load plasma facing component.
- Remote maintenance concept is also common issue: JA divertor is larger weight.
- Mechanical property of Cu-alloy and interlayer (1-2 dpa) may firstly determine PFC life time (maintenance) under DEMO n-irradiation condition, while coolant-temp. is increased (130-200°C).
  EU: R&D of ITER-like target to reduce stress and to strengthen pipe& interlayer.
  JA: applying ITER-like target *near the strike-points* (lower dose).

		EU DEMO [5, 6]	JA DEMO [1, 7]
Total number of cassettes		48	48
Number of divertor maintenance ports		16	16
Weight of one cassette (ton)		11	23
Target	PFC & Heat sink	**W&CuCrZr**	**W&CuCrZr**
	Water T(°C)/ Pressure(MPa)	**130/5**	**200/5**
	Dose on pipe/fpy (dpa)	<10	<1.5
Dome/Baffle	PFC & Heat sink	W&CuCrZr (***liner***)	W&F82H
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15
	Dose on pipe/fpy (dpa)	<10	<8.5
Cassette	Material	EUROFER97	F82H
	Water T(°C)/P(Mpa)	180/ 3.5	290/ 15
	Dose on struct. material/fpy (dpa)	<6	<3

EU DEMO divertor (2019)



JA DEMO divertor (2020)



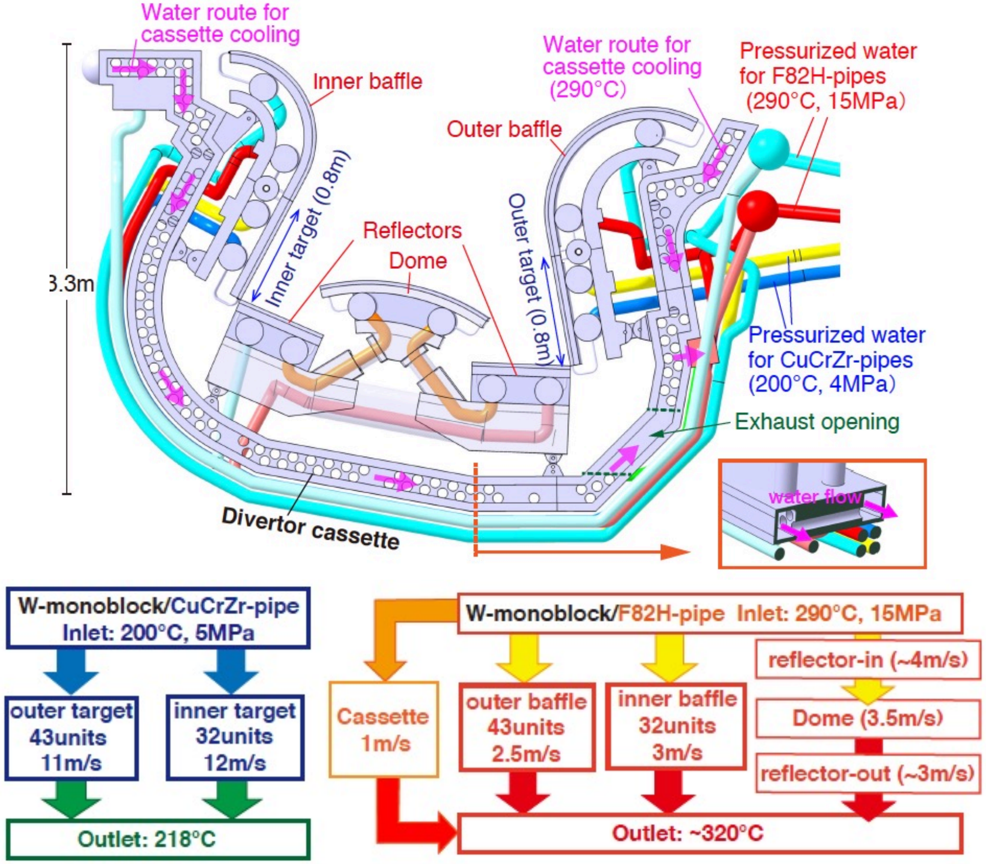
[5] J.H. You, et al., Fus. Eng. Des. (2017). [6] J.H. You, et al., Nucl. Mat. Energy (2018). [7] Asakura, et al. Fus. Eng. Design (2018).

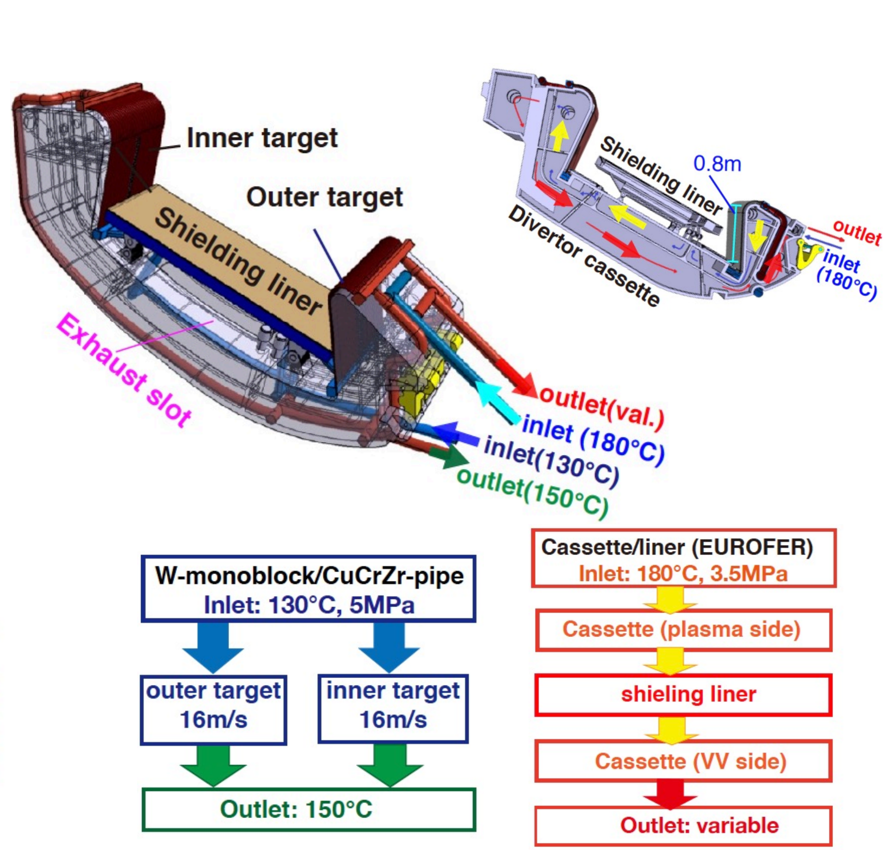
Design concepts of divertor water-cooling for DEMOs:

Optimization of two water routes is required. Coolant-temperature is a design issue.

***Parallel* cooling route for inner and outer targets** is designed to avoid fast flow speed at inboard.

- **JA:** W-MB with CuCrZr/F82H-pipes was arranged for Plasma Facing Components with high/low heat load and low/high n-flux: $T_{coolant}$ =200˚C is used for CuCrZr-pipe to reduce embrittlement [8].
- **EU:** $T_{coolant}$ is reduced (130˚C) to increase the critical heat flux larger than 48 MWm^{-2} (for 150˚C) [9]. $T_{coolant}$(180˚C) for cassette (EUROFER97) to ensure sufficient fracture toughness at n-damage (<6 dpa).





Note) Total $P_{div}^{thermal}$:350MW +$P_{div}^{nuclear}$:120 MW is assumed.

[8] Li-Puma, et al, Fus. Eng. Des. (2013). [9] You, et al, Fus. Eng. Des. (2018)

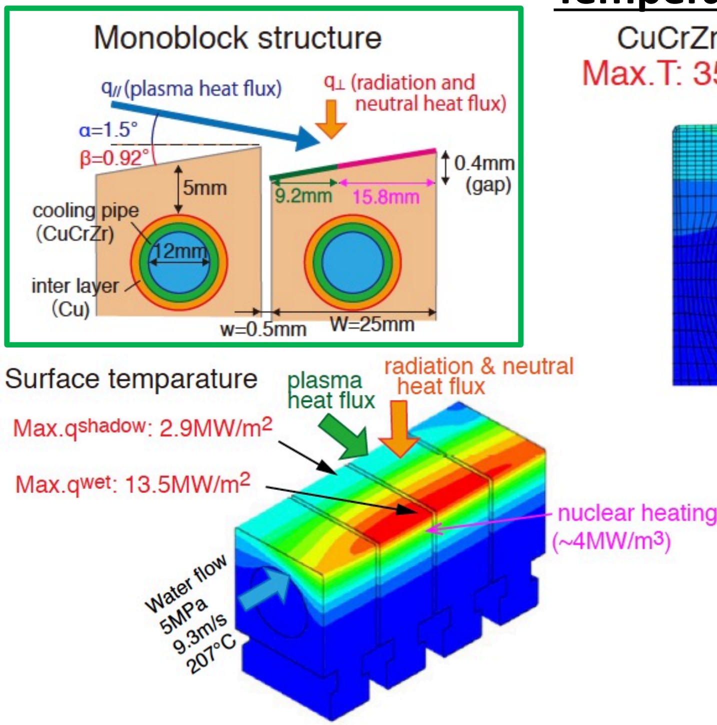
Heat analysis of W-monoblock and CrCrZr heat sink for JA DEMO

Acceptable power load depends on heat load components and target design

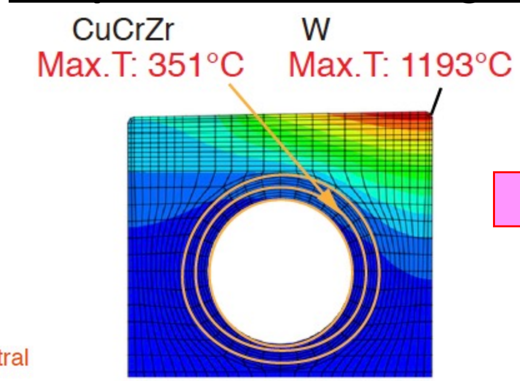
Heat load profile (plasma, radiation&neutral, nuclear heat) is applied to *ITER-like fish scale target*: peak heat load to flat tile (9.1 MWm^{-2}) corresponds to 13.5 MWm^{-2} to the wet area.

- The peak heat load is a critical, i.e. just below recrystallization temperature of W (1200°C). Irradiation-creep/softening of CuCrZr-pipe (351°C) is also anticipated.
- Max. heat flux from the pipe to coolant (18MWm^{-2}) is well below Critical Heat Flux (35MWm^{-2}).

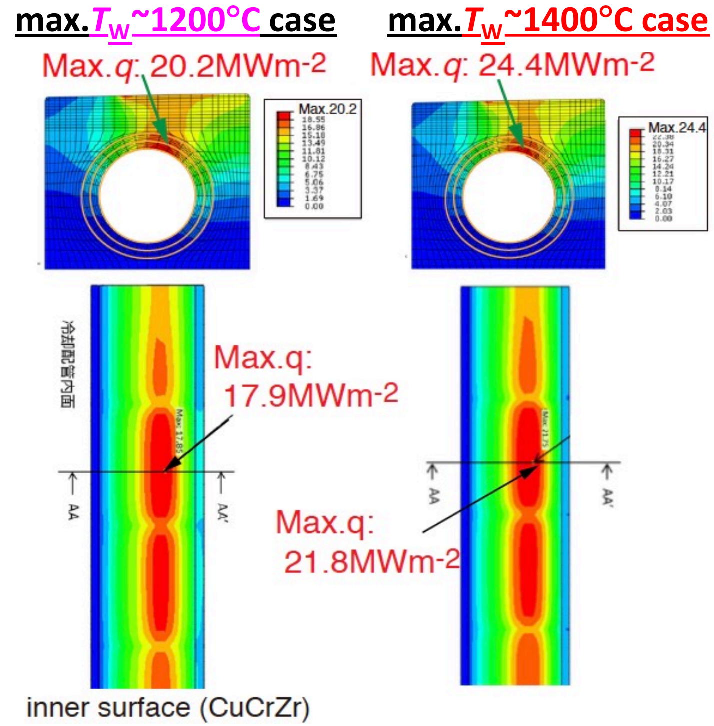
Power exhaust by 200°C water is acceptable even for larger heat load on W (surface-$T_W > T_{recystalization}$).



Temperature in MB target



Heat flux from CuCrZr-pipe to water



In addition, **Elasto-plastic stress analysis** was performed by repeating higher heat load (max. q: 15MWm^{-2}, W surface: 1400°C)⇒ Mechanical strain on CuCrZr pipe (~0.25%) was not critical, while Max. Temp. became 365°C.

Development of water-cooled target components for EU DEMO

Mechanical property of heat sink and joint/interlayer is a key for Cu-alloy application

Development of candidate target concepts based on W-monoblock and Cu-alloy technologies:

- Divertor target concepts are developed for water-cooled targets:
  All are based on a Cu-alloy pipe with swirl tape to increase the heat transfer at the pipe wall:

Baseline:

- **ITER-like** MB & CuCrZr pipe with Cu-interlayer

→ Reducing thickness and width to reduce thermal stresses and prevent vertical cracking.

Reducing stress and strengthen pipe & interlayer:

- **Thermal break** interlayer /CCFC
- W wire-reinforced Cu **composite pipe** /IPP
- **Functionally graded** (W/Cu) **interlayer** /CEA
- W particle-reinforced Cu **composite** heat sink **block** /IPP

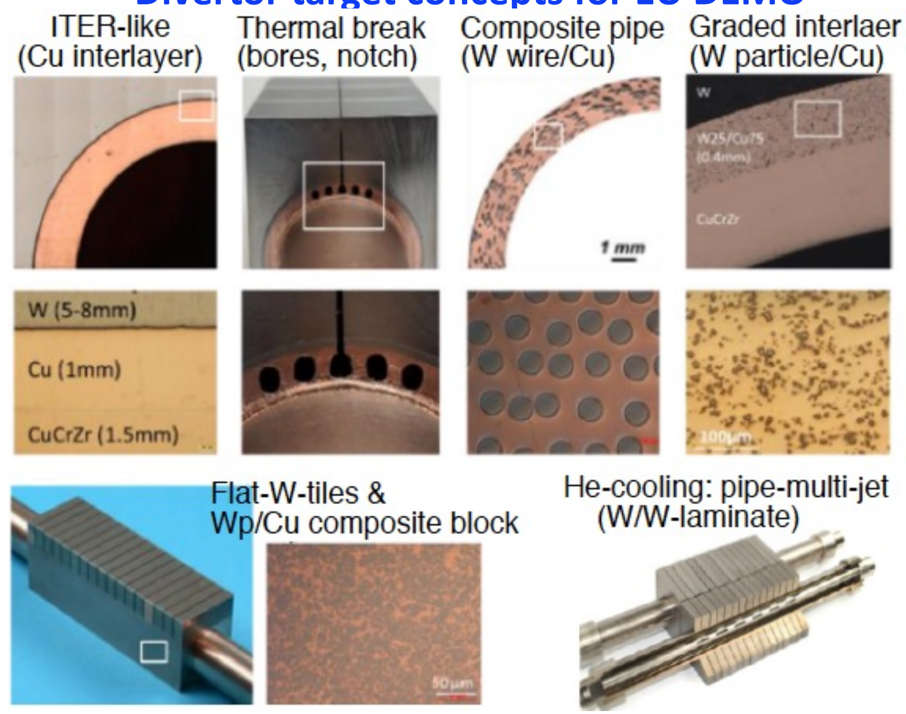
Note) He-cooling by multi-jet pipe /KIT is an option.

- Mock-ups of each concept have been fabricated and 100-level cyclic tested in a high-heat flux facility at 20-25 MW/m² with 20°C water
  → 130°C water

Water cooled target concept

Target concepts	Interlayer	Heat sink
ITER-like (W monoblock)	Cu (1 mm)	CuCrZr pipe
Thermal break (W monoblock)	Cu (1.5 mm) bores	CuCrZr pipe
Composite pipe (W monoblock)	None	Wf/Cu pipe
FGM interlayer (W monoblock)	W/Cu (0.5 mm)	CuCrZr pipe
Composite block (W tiles)	None	W_p/Cu block

Divertor target concepts for EU DEMO



[17] J.H. You, et al., J. Nucl. Mater. (2021).

Summary: BA DDA for Power exhaust concept and Divertor design

Common design issues for Power exhaust and Divertor have been investigated in JA and EU.

- Requirements of f_{rad}^{main} and the plasma performance determined **divertor design concept:**
  Challenges of **JA** (steady-state): ITER-level f_{rad}^{main} (high *HH*) and larger P_{sep}/R =30-34MWm^{-1} and **EU** (pulse): large f_{rad}^{main} (ITER-level *HH*) for ITER-level P_{sep}/R, contribute to optimize future reactor design.
  ⇒ **Same leg length (1.6 m: longer than ITER)** but **different geometry** (**JA:** ITER-like closer baffle, **EU:** rather open without dome and baffle) were proposed as baseline designs.

Power exhaust simulations of P_{sep}~JA: 250-300 MW, EU:150-200MW with Ar seeding have been performed, by using **JA: SONIC** and **EU: SOLPS5.1**, with similar $q_{//}$ profile width ($\lambda_{q//}$~3mm):

- **Large divertor radiation fraction ($f^*{}_{rad}^{div}= P_{rad}^{div}/P_{sep} \geq$ 0.8)** was required to reduce peak-q_{target} ($\leq$ 10MWm^{-2}) and $T_{e,i}$ **in n_e^{sep} range** (JA: 2-3x10^{19}, EU:~2.8x10^{19}m^{-3}) **lower than ITER**.
- Divertor geometry affected *partial detachment profile*.

Integrated design of *divertor target*, *cassette* and *coolant pipe routing* has been developed:
water cooled ITER-like target (W-PFC and Cu-alloy heat sink) is a common baseline design.

- For a year long operation *under DEMO-level n-irradiation*, mechanical property of CuCrZr heat sink and Cu-interlayer is anticipated ⇒ restrictions of q_{target} and $T_{surface}$.
- **EU** has been developing W-MB target components to reduce stress and strengthen pipe& interlayer.
- **JA:** Two coolants to *Cu-alloy heat sink* for Target (200°C) and *F82H heat sink* for Baffle/Cassette (290°C).

⇒ Cu-alloy/interlayer concept and Operation-$T_{coolant}$ for DEMO divertor are common critical issues.

Joint studies on Plasma exhaust and Divertor design are extended to BA DDA Phase-II (-2024).