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

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

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Summary and Conclusions: 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).

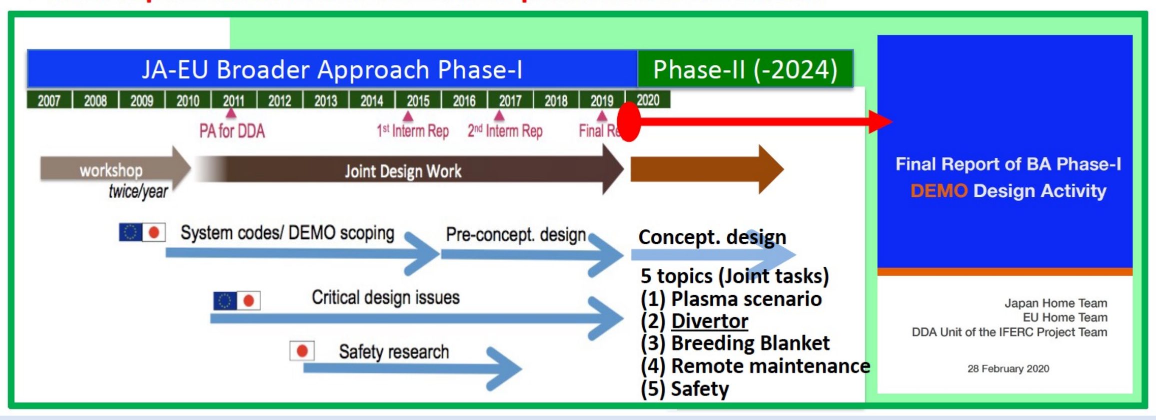
Acknowledgments SONIC simulations were carried out within the framework of the Broader Approach DEMO Design Activity, using the JFRS-1 supercomputer system at CSC, IFERC, Rokkasho, Japan.

1. Introduction: Broader Approach (BA) DEMO Design Activity (DDA)

Europe (EU) and Japan (JA) are undertaking joint DEMO Design Activities (DDA):
JA-HT: Joint Special Design Team for DEMO (Rokkasho, Aomori),
EU-HT: Power Plant Physics and Technology (PPPT) under the EUROfusion Consortium.

The joint design work began in 2010 with the basic goals of:
- identifying the DEMO technical prerequisites and high level requirements;
- identifying the main design and challenges in physics, engineering and technology;
- addressing the challenges and developing the foreseeable technical solutions;
- identifying critical R&D activities and issues to overcome the major design issues.

⇒ **Final report of BA DDA Phase-I was published in 2020 Feb.**

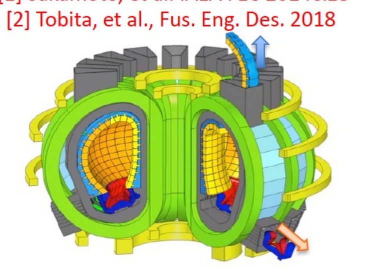


2. Common missions and similar size (8-9m) for JA and EU DEMOs

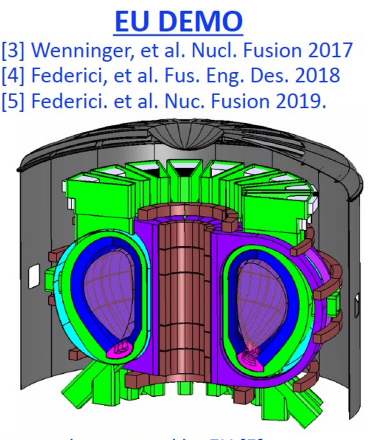
Electric generation, Fuel generation, High duty-cycle, Remote-maintenance in high neutron dose, Safety, Plasma control & Power handling in long-pulse/steady-state.

	Parameters	JA DEMO [1,2]	EU DEMO [4]	ITER(inductive)
Size & Configuration	R_p / a_p (m)	8.5 / 2.4	9.0 / 2.9	6.2 / 2.0
	A	3.5	3.1	3.1
	κ_{95}	1.65	1.6	1.70
	q_{95}	4.1	3.5	3
	I_p (MA)	12.3	18.0	15
	B_t / B_t^{max} (T)	5.94 / 12.1	5.9 / 12.5	5.3 / 12
Absolute Performance	Operation	Steady-state	Pulsed 2 hrs	~400 s
	P_{fusion} (MW)	1462	2000	500
	$P_{el,net}$ (MWe)	250	500	--
	P_{aux} (MW)	83	50	73 (installed)
	Q	18	40	10
	$P_\alpha+P_{aux}$ (=P_{heat} MW)	376	450	173
	P_{sep} (MW)	294	153	~100
Normalized Performance	HH_{98y2}	1.3	1.1	1.0
	β_N	3.4	2.5	1.8
	f_{BS}	0.61	0.39	0.15
	f_{rad}^{main}(=P_{rad}^{main}/P_{heat})	0.22	0.66	~0.33

JA DEMO 2014
[1] Sakamoto, et al. IAEA FEC 2014&18
[2] Tobita, et al., Fus. Eng. Des. 2018



EU DEMO
[3] Wenninger, et al. Nucl. Fusion 2017
[4] Federici, et al. Fus. Eng. Des. 2018
[5] Federici. et al. Nuc. Fusion 2019.



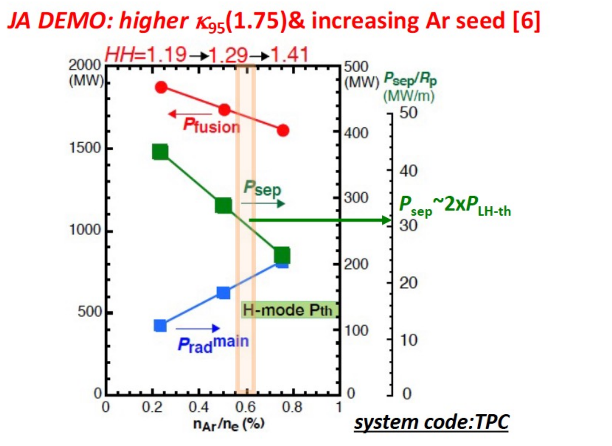
Note) *Concept foreseeing steady-state* (**Flexi-DEMO:** R_p/a_p =8.4/2.7m, P_{fusion} =2GW) is recently proposed by EU [5].

Power exhaust concept and Plasma performance with impurity seeding

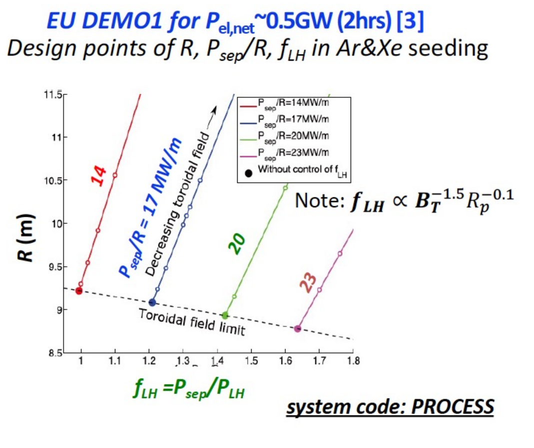
Line-averaged n_e of both DEMOs is lower than that of ITER (1x10^{20}m^{-3}) due to lower Greenwald-density (n^{GW} = 0.68x10^{19}m^{-3}) ⇒ Fusion power (P_{fusion}) is restricted due to fuel dilution by seeding.

- **JA-DEMO (steady-state):** Higher plasma elongation (κ_{95}:1.65→ 1.75) increases I_p(12.3→ 13.5MA), n_e(7.8→ 8.6x10^{19}m^{-3}), P_{fus}(1.5→ 1.7GW), and radiation loss from main plasma (f_{rad}^{main} = P_{rad}^{main}/P_{heat}: 0.22→ 0.4) by increasing Ar seed (n_{Ar}/n_e: 0.25→ 0.6%) with plasma performance of HH_{98y2}~1.3, β_N~3.4.
- **EU-DEMO (pulse):** ITER-level performance (HH_{98y2}~1.1, β_N~2.6) in Xe&Ar seeding achieves $P_{el,net}$~0.5GW ($P_{el,gross}$~0.91GW), and increases f_{rad}^{main} to 0.65 in order to reduce P_{sep} to ~1.2xP_{LH}.

JA DEMO: higher κ_{95}(1.75)& increasing Ar seed [6]



EU DEMO1 for $P_{el,net}$~0.5GW (2hrs) [3]
Design points of R, P_{sep}/R, f_{LH} in Ar&Xe seeding



[6] Asakura, et al. Nucl. Fusion (2017). [7] P_{LH-th} : Martin et al J. Phys.: Conf. Ser. (2008).

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}).

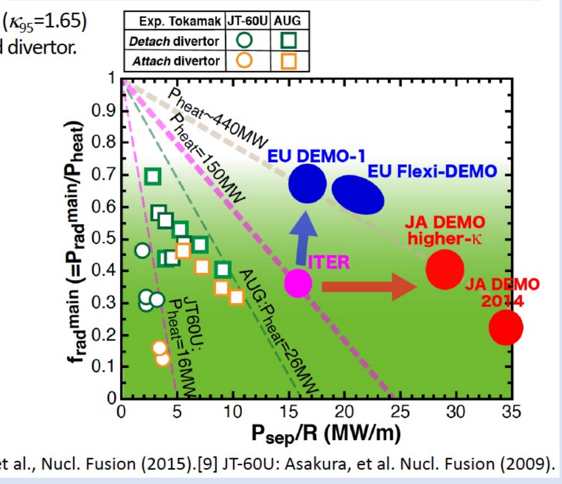
Divertor power handling is determined by requirements of f_{rad}^{main} and the plasma performance.

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

	Parameters	JA DEMO higher-κ [6]	EU DEMO-1 [3]
Power exhaust	line-n_e (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} ($P_\alpha+P_{aux}$ 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



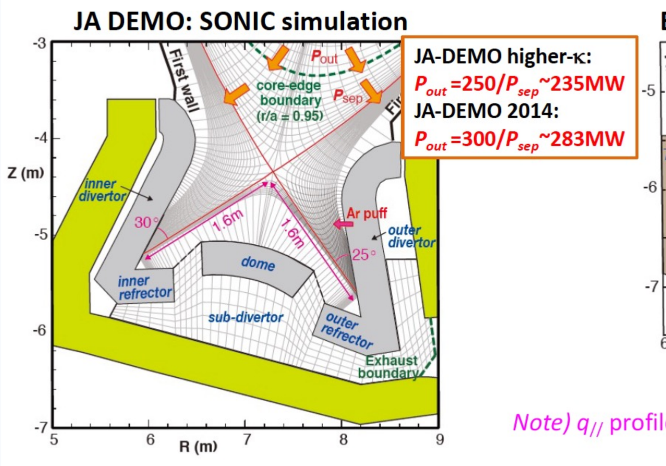
[8] AUG: Kallenbach, et al., Nucl. Fusion (2015).[9] JT-60U: Asakura, et al. Nucl. Fusion (2009).

3. Divertor design and Power exhaust simulation

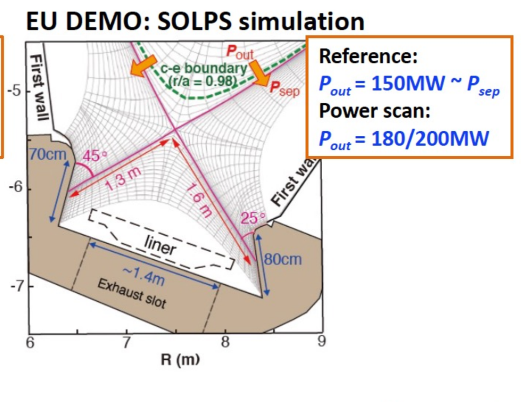
Conventional design concepts in JA & EU are based on the ITER divertor:
- Both DEMOs: Divertor leg is extended (outer L_{div}=1.6 m: 1.6 times longer than ITER).
- JA: Baffles cover divertor plasma for large P_{sep}/R handling ⇔ EU: Open and shallow geometry (ITER-level P_{sep}/R) to increase tritium-breeding 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



EU DEMO: SOLPS simulation



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

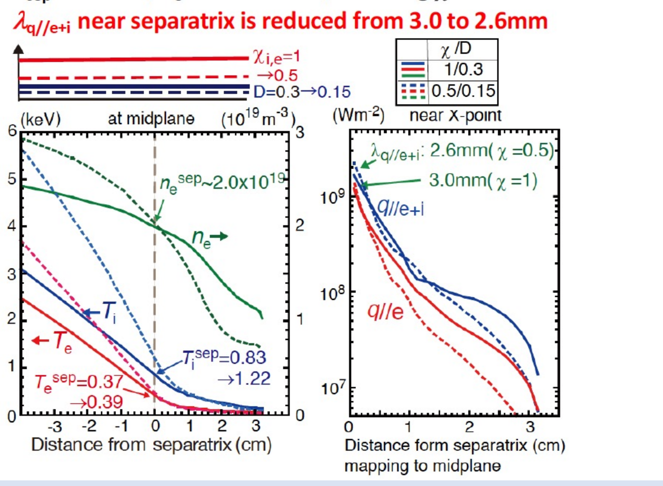
SOL heat flux becomes large and narrow in DEMO simulations

SONIC sim. for JA DEMO: T_e^{sep} & T_i^{sep} increase to 0.37 & 0.83 keV: 2-3 times larger than ITER
⇒ $\lambda_{q//e+i}$ ~3 mm for "standard" ITER values of χ = 1 m^2/s, D = 0.3 m^2/s: $\lambda_{q//}$ = 3.4 mm in ITER [10].
- Reduction to half values (χ = 0.5 m^2/s, D = 0.15 m^2/s) ⇒ $\lambda_{q//e+i}$ is reduced to 2.6 mm[12].

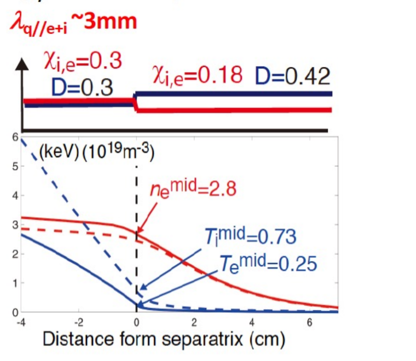
SOLPS sim. for EU DEMO: T_e^{sep} & T_i^{sep} increase to 0.25 & 0.73 keV
⇒ $\lambda_{q//e+I}$ ~3 mm for lower χ (= 0.18 m^2/s), D (= 0.42 m^2/s).

Note) DEMO $q_{//e+i}$ profiles are *still wider than Eich's scaling*[12] (~1.1mm) & *GS model*[13] (~1.5mm).

JA DEMO higher-κ(SONIC)[11]:
P_{sep}~235MW, n_e^{sep}~2x10^{19}m^{-3}, reducing χ & D
$\lambda_{q//e+i}$ near separatrix is reduced from 3.0 to 2.6mm



EU DEMO1(SOLPS):
P_{sep}=150MW, n_e^{sep} ~2.8x10^{19}m^{-3}
$\lambda_{q//e+i}$ ~3mm



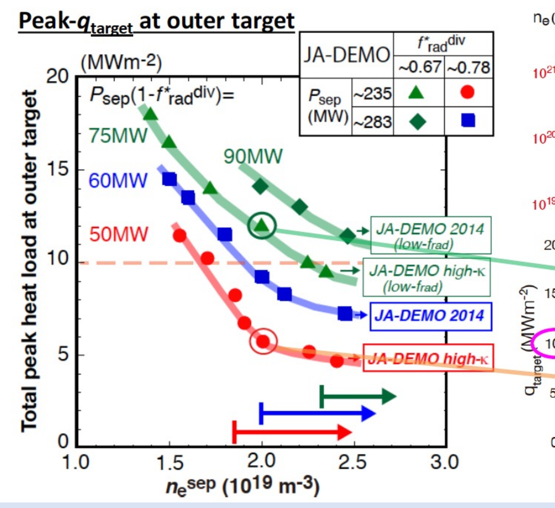
[10] Kukushkin, et al. J. Nucl. Mater. (2013).
[11] Asakura, et al. Fus. Mat. Energy (2021).
[12] Eich, et al. Nucl. Fusion (2013).
[13] R. Goldston, Nucl. Fusion (2012).

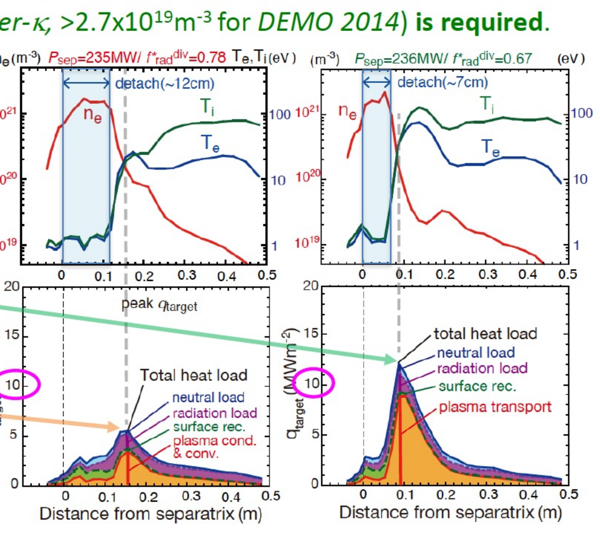
JA DEMO: Divertor operation in low density (n_e^{sep} = 2-3x10^{19}m^{-3})

Heat load can be reduced within the operation range ($q_{target} \leq$ 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} ($\leq$ 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 ($\leq$ 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





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

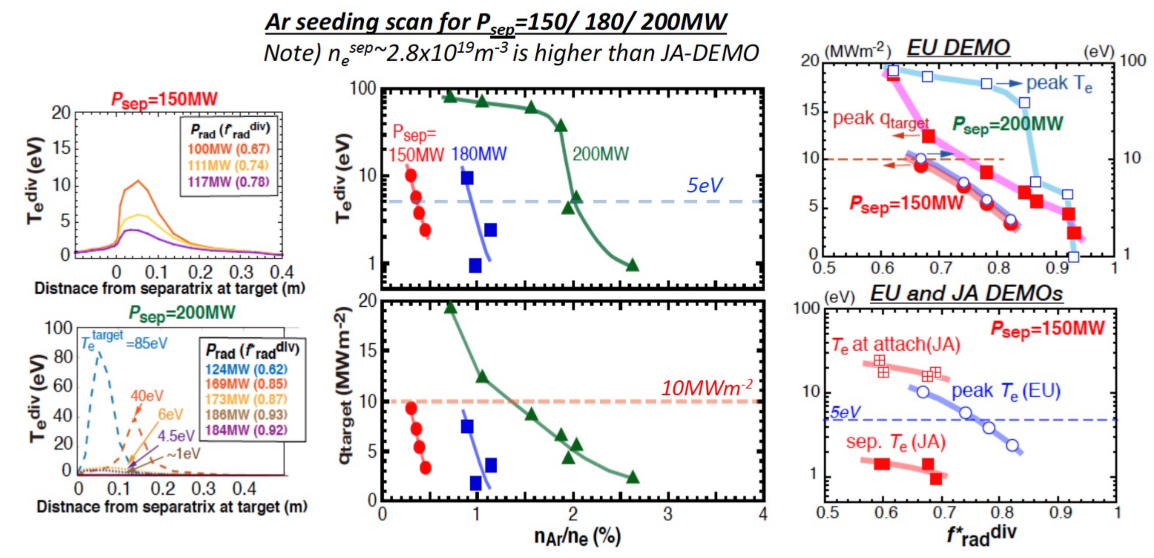
Heat reduction was achieved for all cases by increasing C_{Ar}^{SOL} (= n_{Ar}/n_e)^{SOL} = 0.5-2.5%

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}~2.8x10^{19}m^{-3} is higher than JA-DEMO



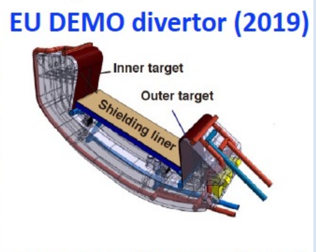
4. Design concepts for water-cooling DEMO divertor

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

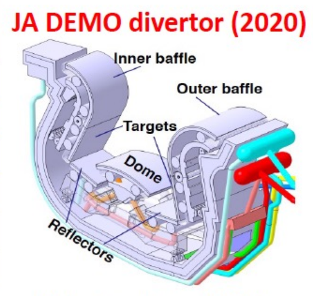
- ITER-like monoblock target is the first candidate for high heat load plasma facing component.
- Remote maintenance concept is also common issue: JA divertor is larger weight.
- Mechanical property of Cu-alloy and interlayer (1-2 dpa) may firstly determine PFC life time (maintenance) under DEMO n-irradiation 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 [14, 15]	JA DEMO [6, 16]
	Number of total 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)

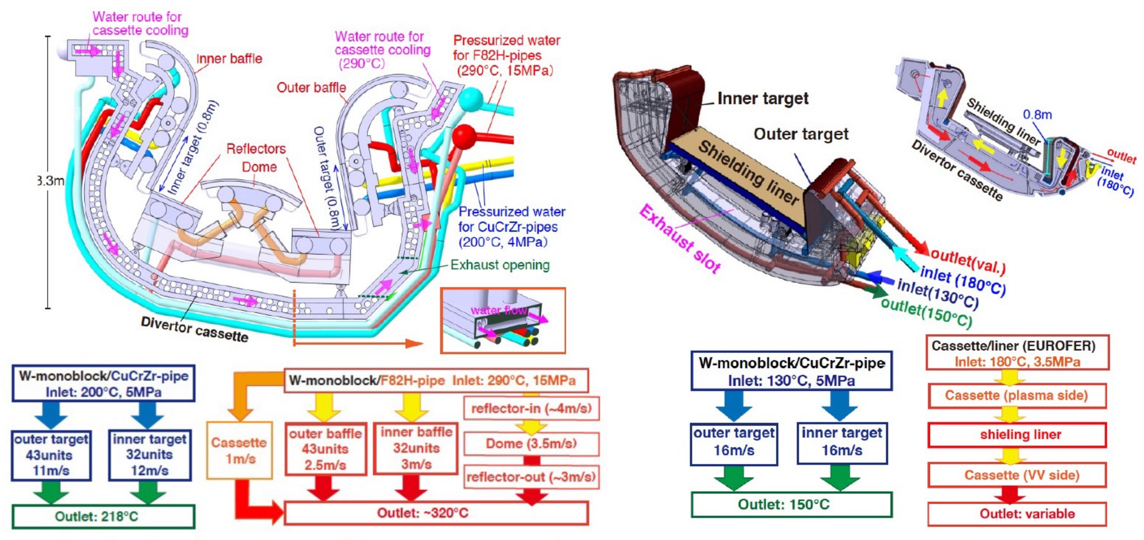


[14] J.H. You, et al., Fus. Eng. Des. (2017). [15] J.H. You, et al., Nucl. Mat. Energy (2018). [16] Asakura, et al. Fus. Eng. Design (2018).

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 [17].
- **EU:** $T_{coolant}$ is reduced (130˚C) to increase the critical heat flux larger than 48 MWm^{-2} (for 150˚C) [18].
$T_{coolant}$(180˚C) for cassette (EUROFER97) to ensure sufficient fracture toughness at n-damage (<6 dpa).



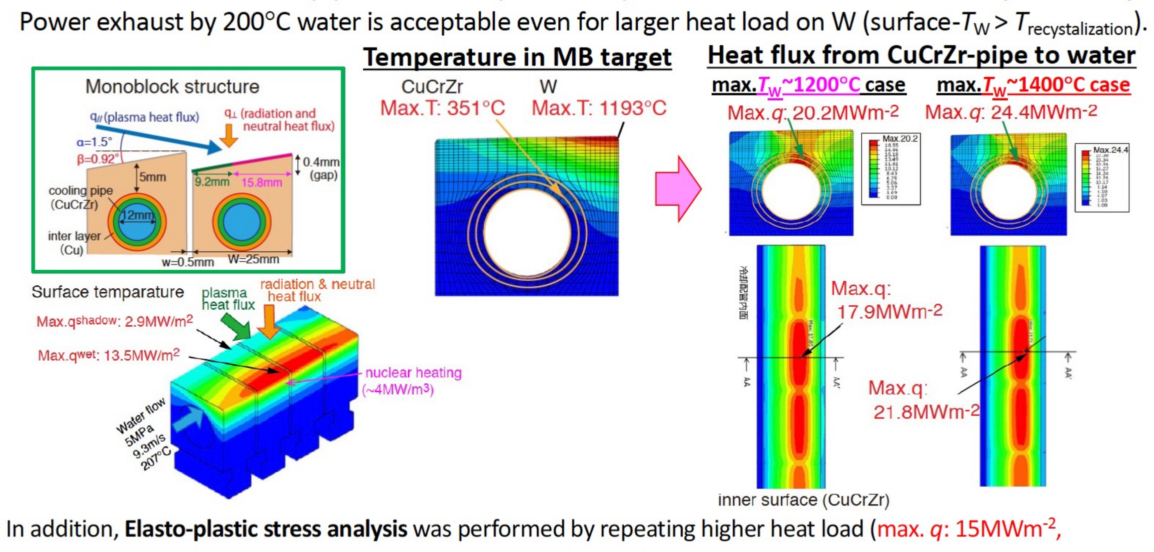
Note) Total $P_{div}^{thermal}$:350MW +$P_{div}^{nuclear}$:120 MW is assumed. [17] Li-Puma, et al, Fus. Eng. Des. (2013). [18] You, et al, Fus. Eng. Des. (2018)

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 > $T_{recystalization}$).



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:
- 5 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

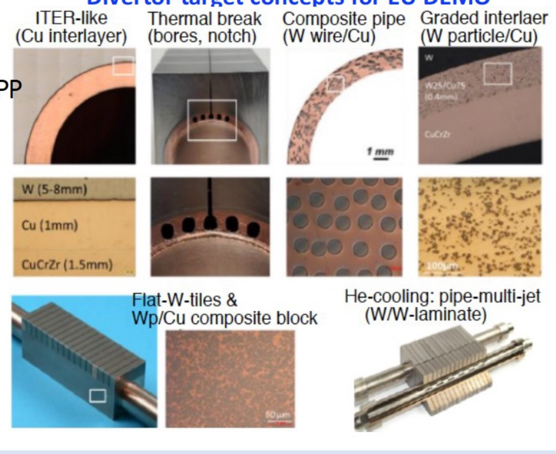
Note) He-cooling by multi-jet pipe /KIT is an option.

- 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



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