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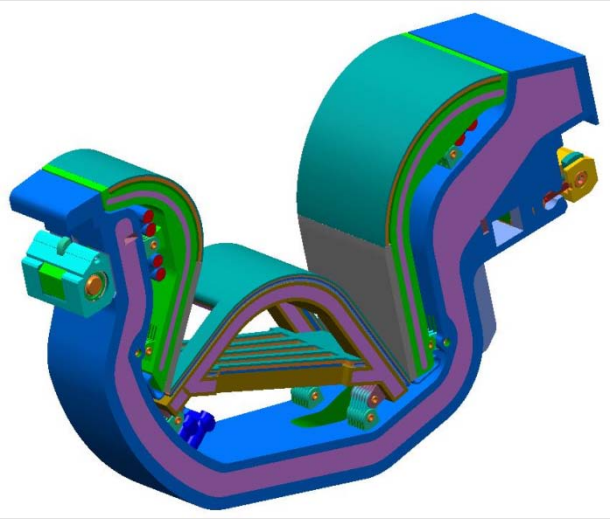
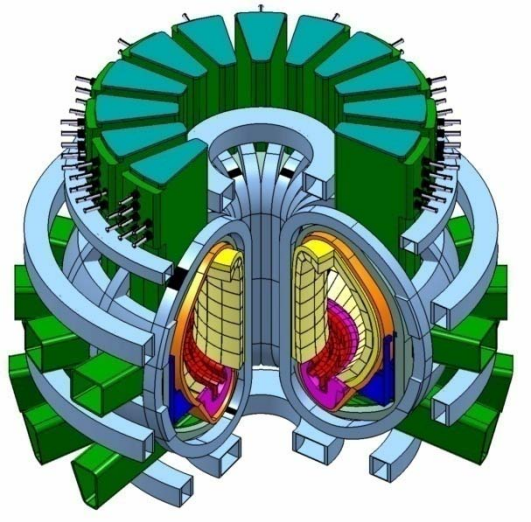
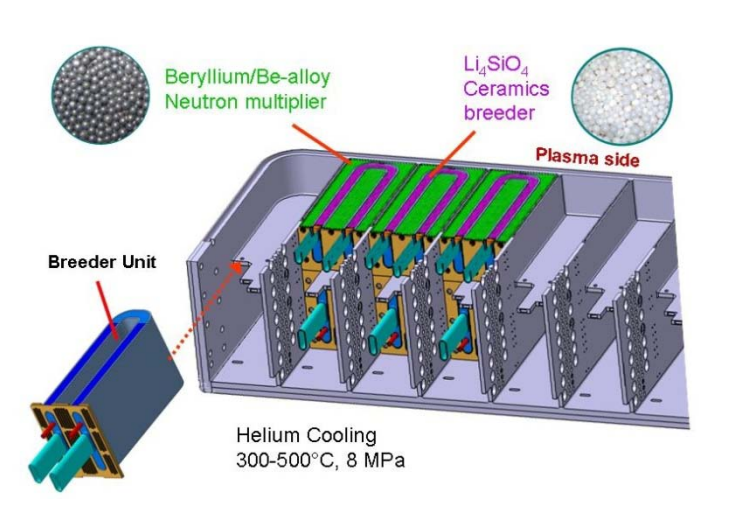
Overview on Fusion Technology development for DEMO in KIT – In vessel component design and integration

L.V. Boccaccini, F. Cismondi and P. Norajitra, Karlsruhe Institute of Technology

Japan-US Workshop on Fusion Power Plants and Related Advanced Technologies, February 22-24, 2011 at National Institute for Fusion Science in Toki, JAPAN







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Presentation Outline

Short introduction to the German fusion programme in EU framework

KIT activities in Fusion Technology

In-vessel component design & integration

- Blanket Development
- Divertor Development
- Reactor integration

Summary and future activities

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The European Fusion Programme

European Commission

FUSION FOR ENERGY

EFDA

Fusion for Energy (F4E) is the European Union's Joint Undertaking for ITER and the Development of Fusion Energy. The organisation was created in order to meet three objectives:

- to provide Europe's contribution to ITER;
- to support fusion research and development initiatives through the Broader Approach Agreement, signed with Japan
- to contribute towards the construction of demonstration fusion reactors.

[http://fusionforenergy.europa.eu/]

The European Fusion Development Agreement (EFDA) is an agreement between European fusion research institutions and the European Commission to strengthen their coordination and collaboration, and to participate in collective activities. Its activities include coordination of physics and technology in EU laboratories, the exploitation of the world's largest fusion experiment, the Joint European Torus (JET) in the UK, training and career development in fusion and EU contributions to international collaborations.

[http://www.efda.org/]

Fusion Programme in Research Centres in the Helmholtz Association in Germany

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HELMHOLTZ ASSOCIATION

Karlsruhe Institute of Technology · IPP Max-Planck-Institut für Plasmaphysik EURATOM Assoziation · JÜLICH FORSCHUNGSZENTRUM

+ 13 others not EFDA Research Centres

Helmholtz Programme for Fusion in 2010-2014

Topic 1 - Stellarator research: completion of the assembly of Wendelstein 7-X including the expected progress in major components (IPP 92%, KIT 4%, FZJ 4% for ~41.2% of budget)

Topic 2 - Tokamak physics for ITER and beyond: experimental tokamak physics investigations (in ASDEX Upgrade, JET, TEXTOR) aimed at preparing ITER and DEMO (IPP 95%, FZJ 5% for ~18.3% of budget)

Topic 3 - Fusion technology for ITER: contribution to ITER construction (e.g. High Power Gyrotrons, Vacuum Pumps) and to JT-60 in BA (High Temperature Superconducting Current Leads) (IPP 15%, KIT 60%, FZJ 25% for ~17.3% of budget)

Topic 4 - Fusion technology beyond ITER: solutions in particular with respect to materials, fuel cycle, in-vessel components, heating & current drive, and power extraction (IPP 10%, KIT 90% for ~10.1% of budget)

Topic 5 - Plasma-wall interactions (PWI): lifetime of wall components, Tritium inventory and removal methods, dust generation, PWI with radiation-damaged materials, surface physics and diagnostics (IPP 38%, KIT 5%, FZJ 57% for ~7.1% of budget)

Topic 6: Plasma theory: to provide a model for the plasma core of a fusion reactor that can be incorporated into design codes for a power plant. (IPP 81%, FZJ 19% for ~6.0% of budget)

Fusion Programme at KIT (1/3)

Topic 3: Fusion technology for ITER

- **ECRH Upper Launcher** (with IPP and other): KIT leads a Consortium for developing and, together with industrial manufacturers, providing the system to ITER.
- **High Power Gyrotrons** (with IPP): development of a 2 MW, 70 GHz coaxial cavity gyrotron for ITER
- **Deuterium-Tritium Fuel Cycle**: procurement of the vacuum pumping to ITER, on the Water Detritiation (WDS) and the Isotope Separation System (ISS).
- **Magnet system:** perform the quality assurance for magnet conductors and structures; support of ITER.
- **Test Blanket Module Systems:** KIT leads a consortium for the development of the EU TBMs, their auxiliary systems and integration in ITER
- **Safety Research:** application in the licencing process of the already performrd experimental investigations of dust explosion characteristics and possible mitigation techniques, and established models for the numerical simulation of dust mobilisation and hydrogen / dust combustion.
- **High Temperature Superconducting (HTS) Current Leads** for JT60-SA

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Fusion Programme at KIT (2/3)

Topic 4: Fusion technology beyond ITER

- **Structural Materials for Blanket and Divertor**: EUROFER, EUROFER-ODS, W and W-alloy including joint technologies.
- **Blanket Development for DEMO**: further development of solid and liquid concepts for high performances, in-vessel integration, development of functional materials (solid breeder and Beryllium).
- **Divertor Development for DEMO**: design, manufacturing and testing of a high temperature, helium cooled divertor.
- **IFMIF:** KIT is leading the engineering design of the entire IFMIF “Test Facilities”, including the test cell, the major test modules (High Flux Test Module: HFTM) and including various instrumented auxiliary rooms.
- **High Temperature Helium Technology:** building and operating of a series of facilities (from 0.2 to 2 kg/s at 8-10 MPa) for the testing of Helium cooled components (Blanket and divertors) and high temperature materials.
- **Advanced ECRH:** Multi-MW and Multi Frequency Gyrotrons (with IPP)
- **High Temperature Superconductors** for Fusion Magnets: e.g. 10-12 mm wide cables for the application regime of 2-5 kA (77 K).
- **Fuel Cycle Technologies:** integration of an outer part to the ITER type fuel cycle where tritium in the form of HT and HTO must be recovered from large He flows; planning of a test facility.

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Fusion Programme at KIT (3/3)

Topic 1: Stellarator research:

- **High temperature superconducting current leads**: development and manufacturing.
- **Microwave heating** (with IPP and others): to supply the entire ECRH system for W7-X including ten 1 MW, 140 GHz gyrotrons, related transmission lines, launching antennae, power electronics and control systems.

Topic 5: Plasma-wall interactions

- **Dust Generation** (with FZJ and IPP): further development of modelling in order to represent the whole range of complex processes of armour material erosion as evaporation, brittle destruction and melt motion.

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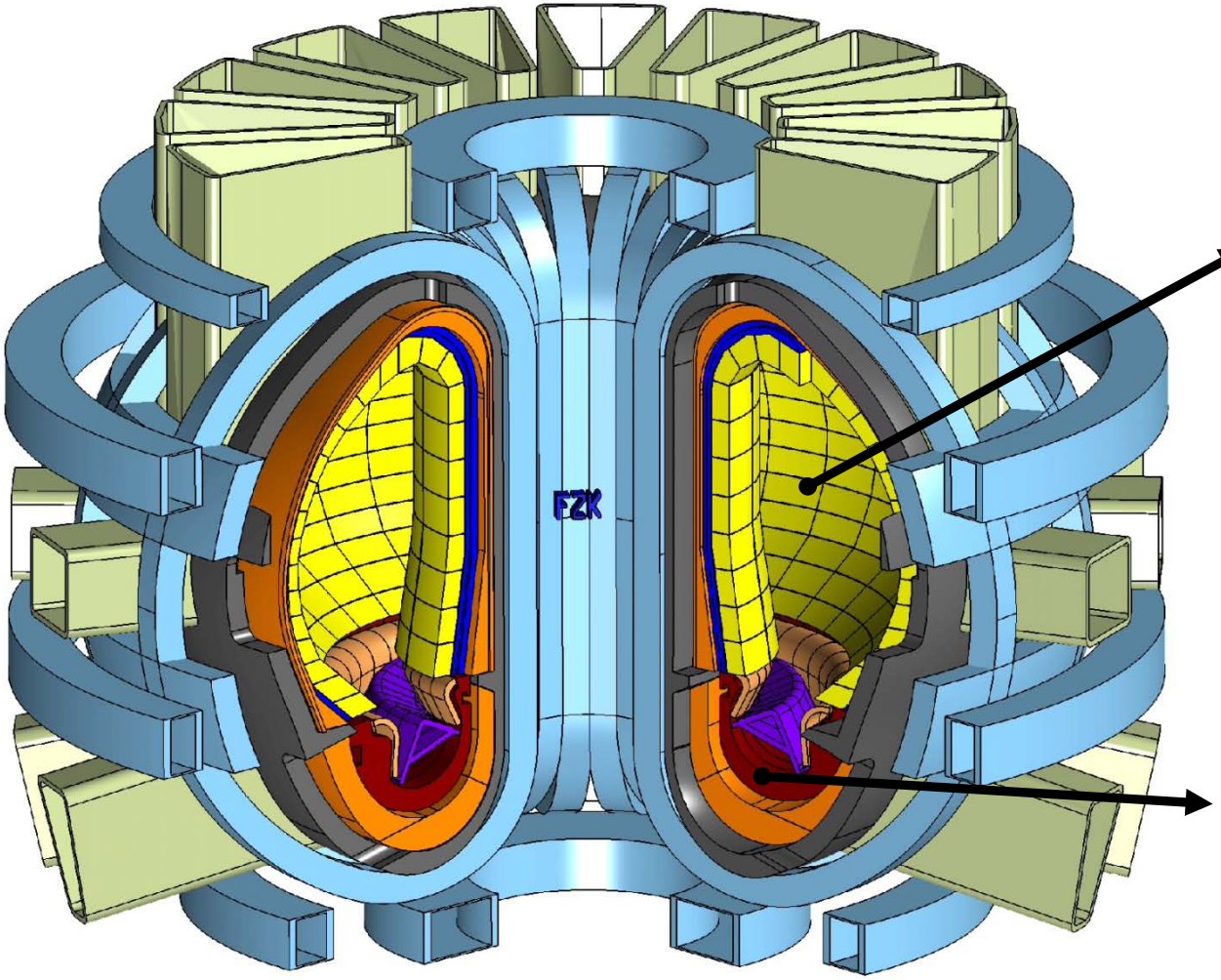
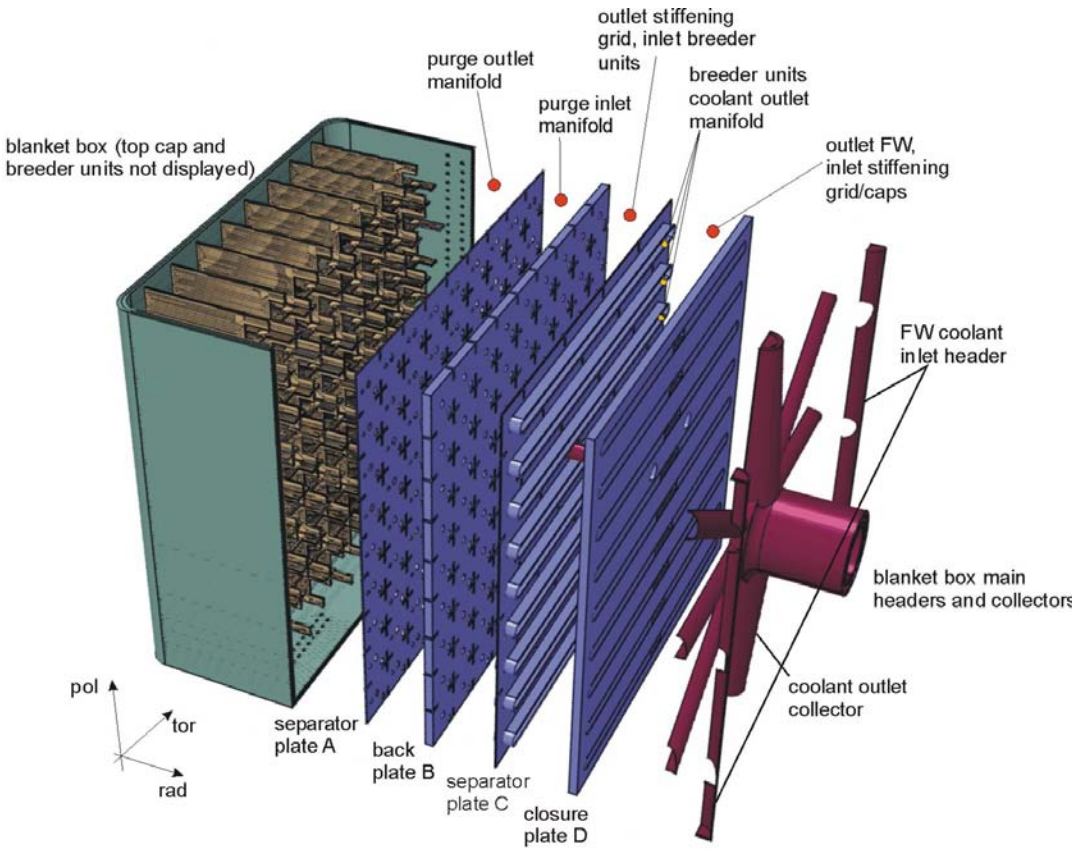
Fusion Reactor Thermonuclear Core

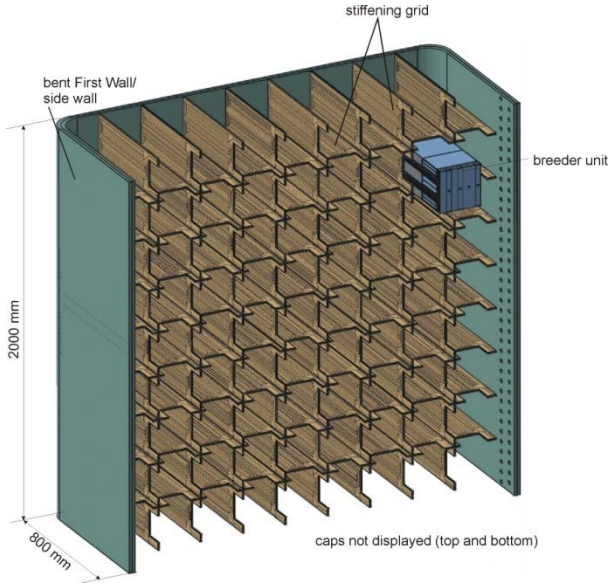


Blanket (HCPB):
- EUROFER as structural material
- Solid Breeder Blanket Concept
- Helium cooling @ 8MPa, 300-500°C
- T extraction with low pressure He purge flow

Divertor:
-W, WL10, ODS EUROFER
- He cooling @ 10MPa, 600-700°C

HCPB Blanket: PPCS-Model B (2003)





Operational parameters	HCPB/HCLL
FW heat flux (peak)	**0.5 MW/m²**
Neutron wall load (peak)	**2.4 MW/m²**
Power Generation System	**Rankine**
Pressure Coolant	**He: 8 MPa**
Temperature Coolant	**He: 300 – 500°C**

S. Hermsmeyer and al: Lay-out of the He-cooled solid breeder model B in the European power plant conceptual study , Fusion Engineering and Design 69 (2003) 281-287

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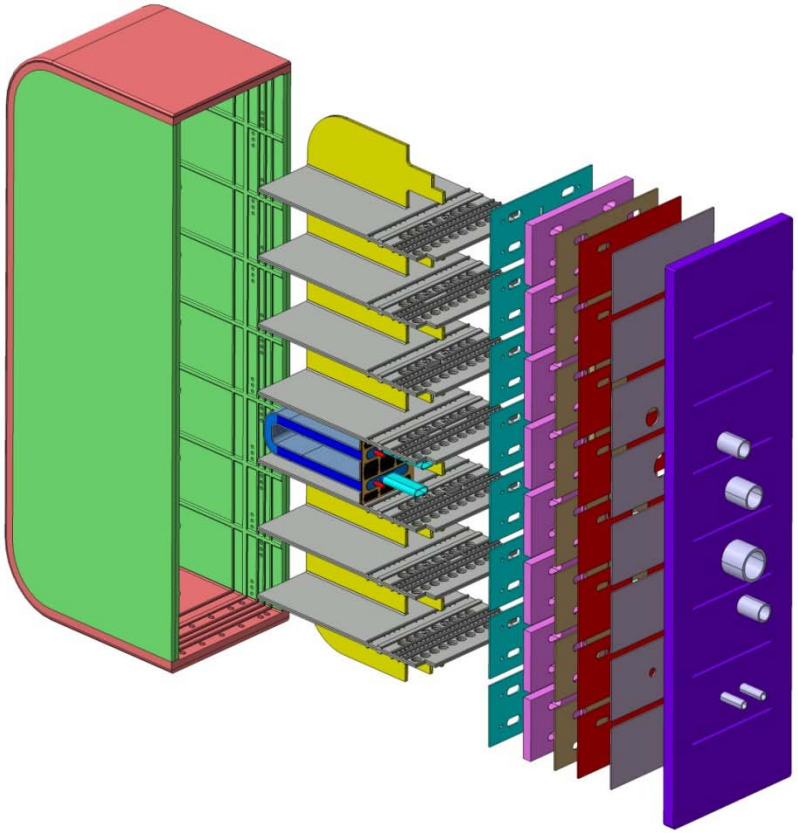
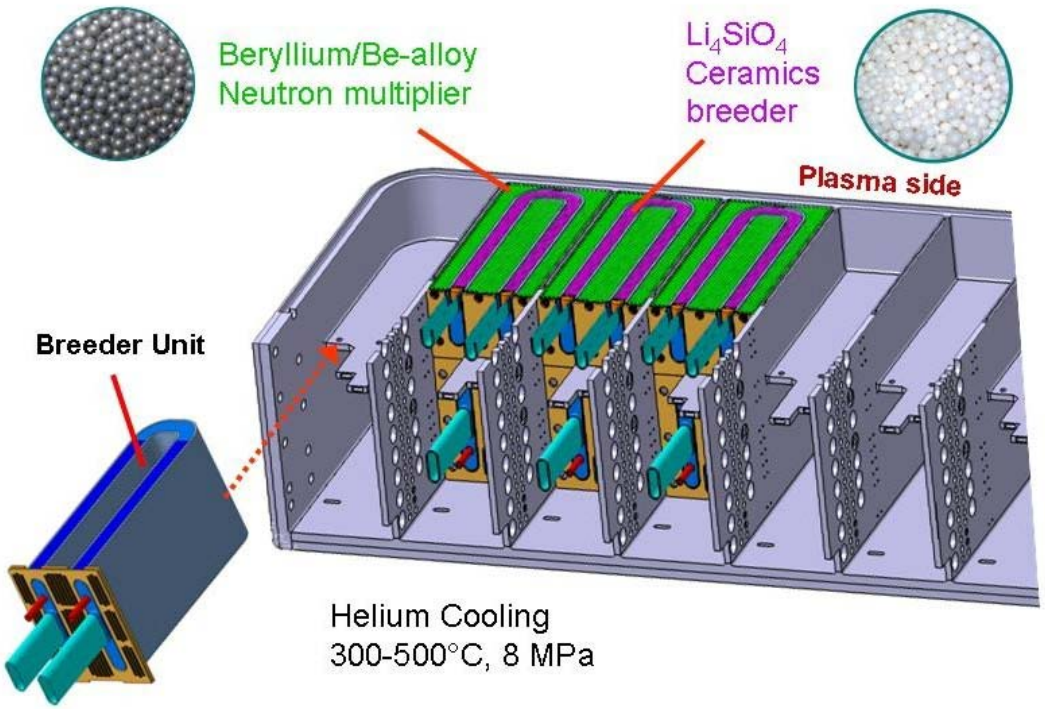
HCPB Blanket: Development under TBM Study





F. Cismondi and al., Design update, thermal and fluid dynamic analyses of the EU-HCPB TBM in vertical arrangement, Fusion Engineering and Design 84 (2009) 607-612.

F. Cismondi, S. Kecskes, G. Aiello, HCPB TBM thermo mechanical design: assessment with respect codes and standards and DEMO relevancy, to be published.

Transient thermo mechanical analyses of the TBM

HCPB simulation of the thermal transient:

- By the end of the pulse the BU works in steady state condition
- Helium outlet temperature at the end of the plasma pulse ~500°C.
- Inversion of the thermal gradient between the FW and the BZ

Time instants significant for the mechanical analyses:

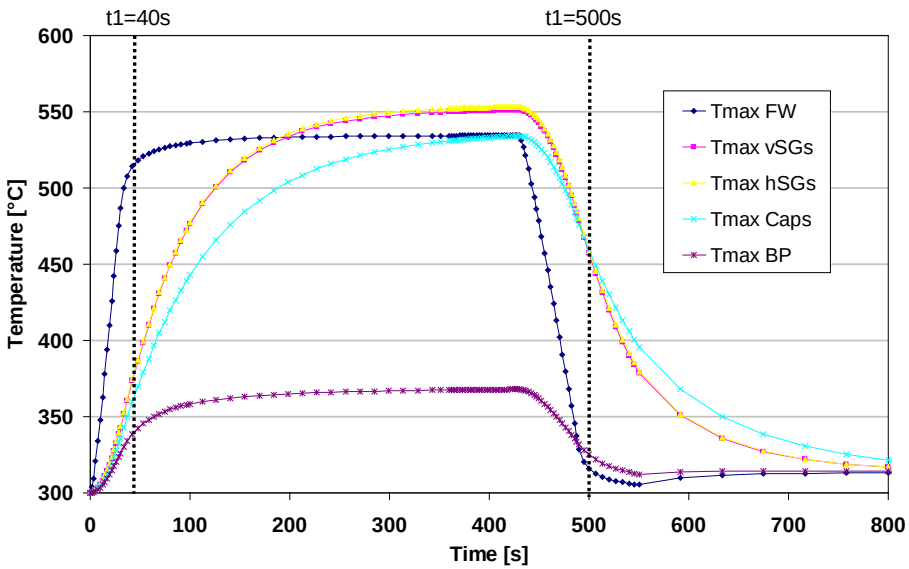
t1=40s, FW reaches nearly stationary conditions

t2=500s: cooling phase, inversion of the temperature difference between FW and Manifold

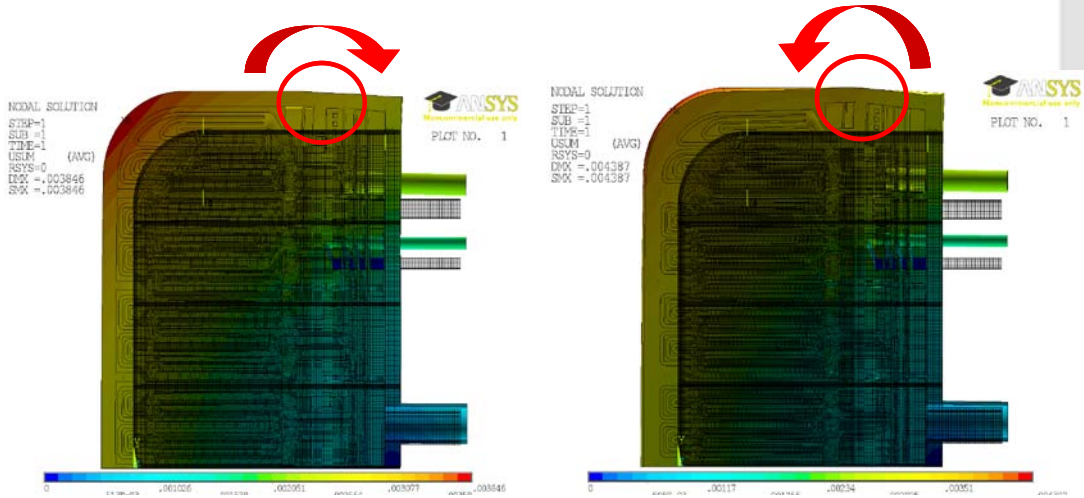
HCPB TBM

Most demanding condition for the structural integrity of the box: inversion of the thermal gradients during plasma ramp-up and ramp-down in the manifold region associated to the weakness of the horizontal SGs.

Evolution of maximum temperature in TBM subcomponents during a typical ITER plasma pulse.



Thermal fields calculated at these instants have been used as input for the thermo-mechanical analysis.



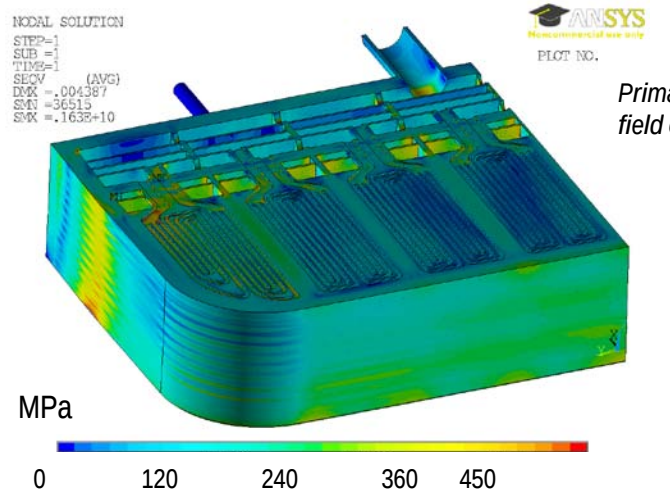
Deformation pattern of the TBM box at 40s (left) and 500s (right) during the plasma pulse

Assessment with respect C&S

HCPB TBM

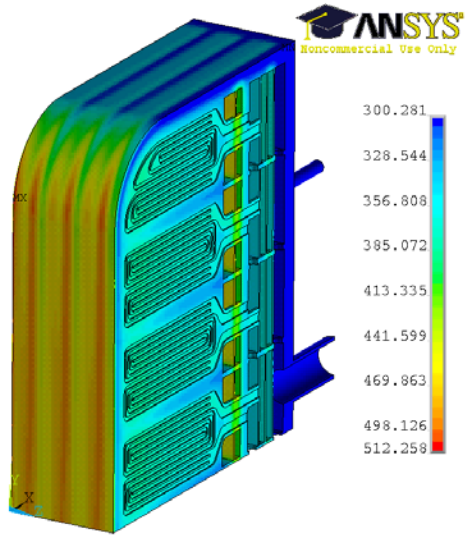
The most demanding condition for the structural integrity of the manifold region is the inversion of the thermal gradients during the plasma ramp-up and ramp-down phases.

Even in steady state conditions the horizontal SGs in the manifold region undergo high stresses (primary stresses) with peak values up to 850MPa.

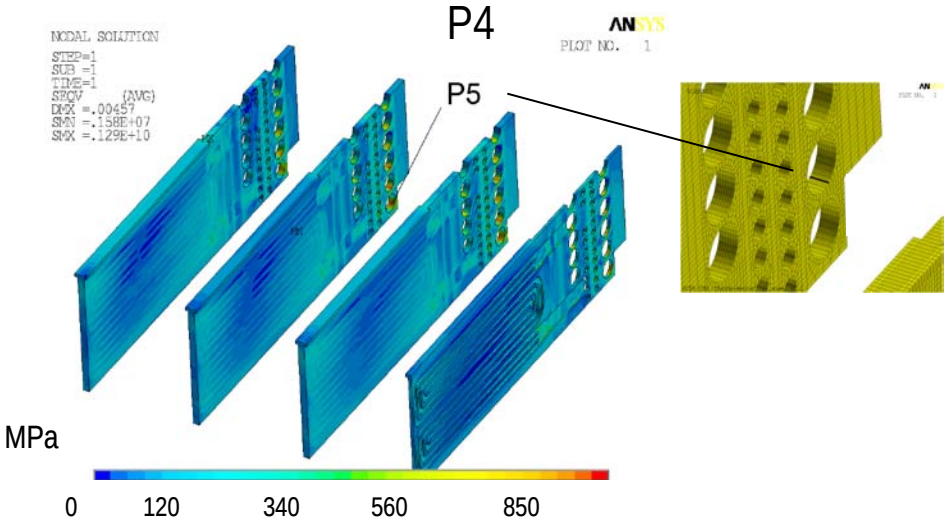


Primary + secondary stress field on the TBM at t2=500s

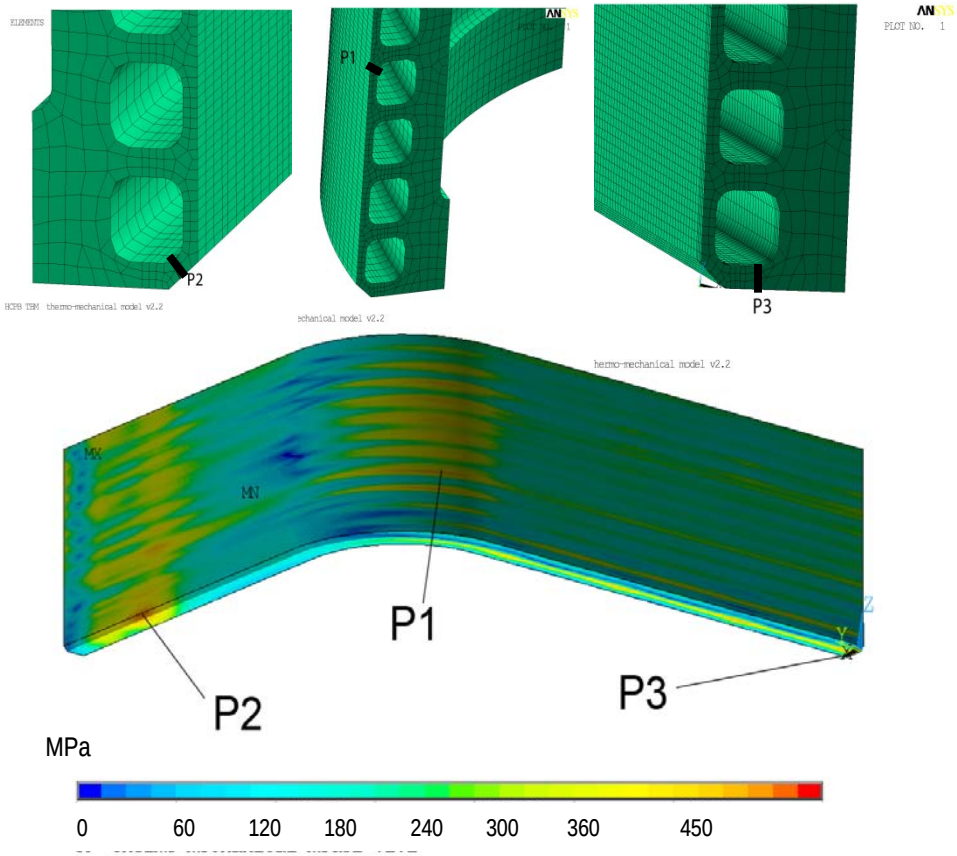
Temperature distribution at t1=40s.



HCPB TBM transient thermo-mechanical model



Assessment with respect C&S



Immediate plastic flow localization

	T (°C)	$\overline{P_m+Q_m}$ (MPa) t_1	t_2	Limit (MPa)	Margin
Line 2	360	169	**442**	200	**-121%**
Line 3	384	70	**248**	174	**-43%**
Line 4	307	551	**673**	258	**-161%**

Ratcheting

	T (°C)	$\overline{P_L+P_b+\Delta Q}$ (MPa) t_1-t_0	t_2-t_0	t_2-t_1	Limit (MPa)	Margin
Line 1	434	252	196	247	495	49%
Line 2	360	198	496	323	543	9%
Line 3	384	265	254	422	528	20%
Line 4	307	593	**736**	246	567	**-30%**

Good behavior of the FW against ratcheting and primary stresses

Problematic behavior of the manifolds region: In P4 limits for immediate plastic collapse and instability and creep are largely exceeded.

Primary stresses + creep

	T (°C)	Immediate plastic collapse and instability $\overline{P_m}$ (MPa) Value	Limit	Margin	$\overline{P_m+P_b}$ (MPa) Value	Limit	Margin	Creep $\overline{P_m}$ (MPa) Value	Limit	Margin	$\overline{P_m+P_b/k_t}$ (MPa) Value	Limit	Margin
Line 1	493	21	147	86%	25	220,5	89%	21	249	92%	36	249	86%
Line 2	458	21	159	87%	27	238,5	89%	21	315	93%	38	315	88%
Line 3	448	22	162	86%	44	243	82%	22	322	93%	36	322	89%

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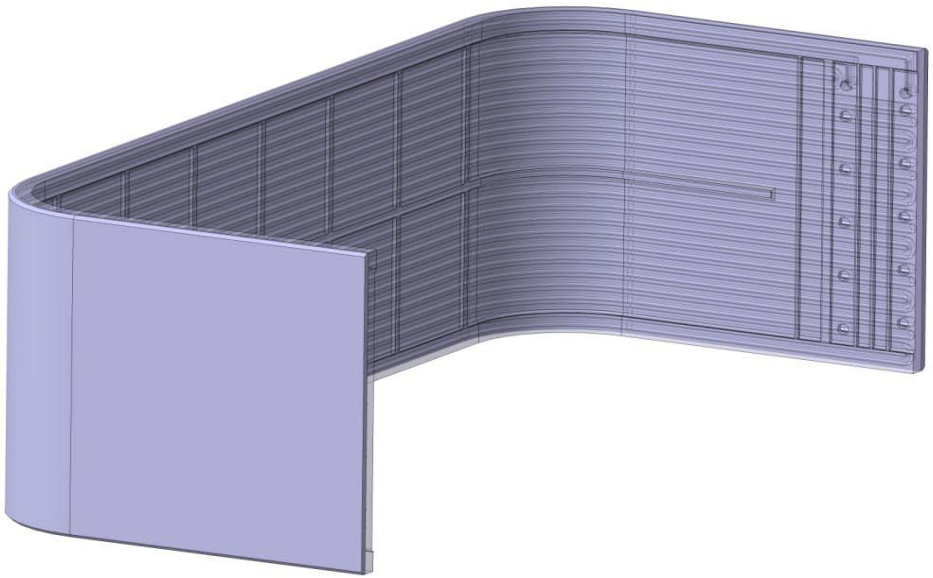
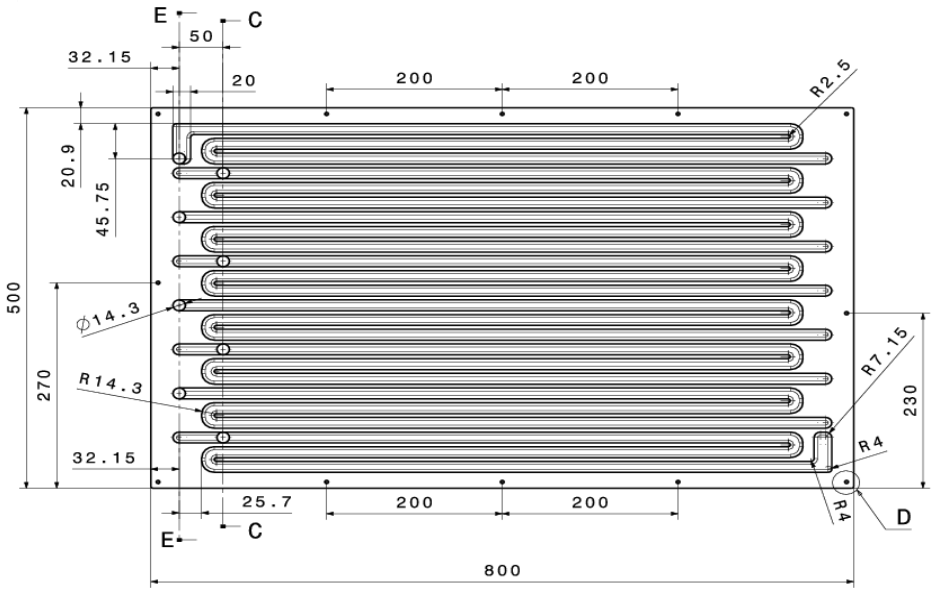
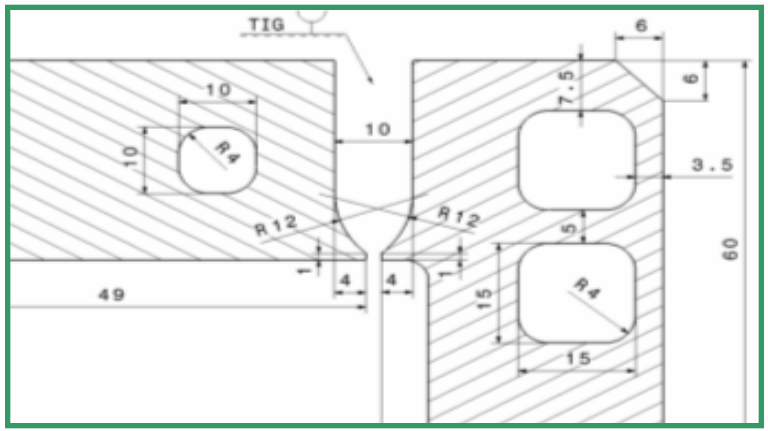
HCPB Blanket: FW Manufactuting





Drawings for a 1/4 mock-up of the First Wall



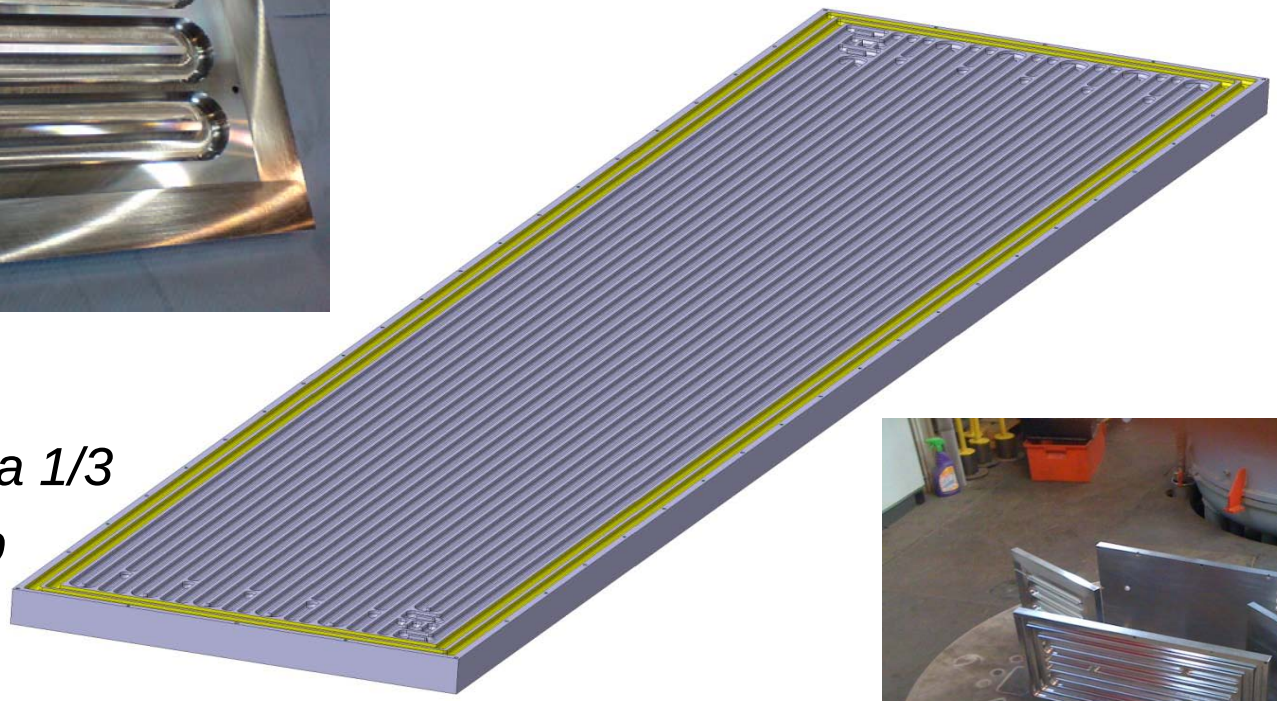
EUROFER97: 9CrWVTa-Ferritic-martensitic steel

TBM FW Dimensions: 3m x 0.6 m x 0.3m

J. Rey, H. Neuberger, A. von der Weth, L. Bankos, P. Freiner (KIT)

Milling of half FW plates and cleaning

Milled EUROFER Plate for a Small scale mock-up



CAD Drawing of a 1/3 FW mock-up

Plasma cleaning of the bounding surface of an EUROFER half-plate

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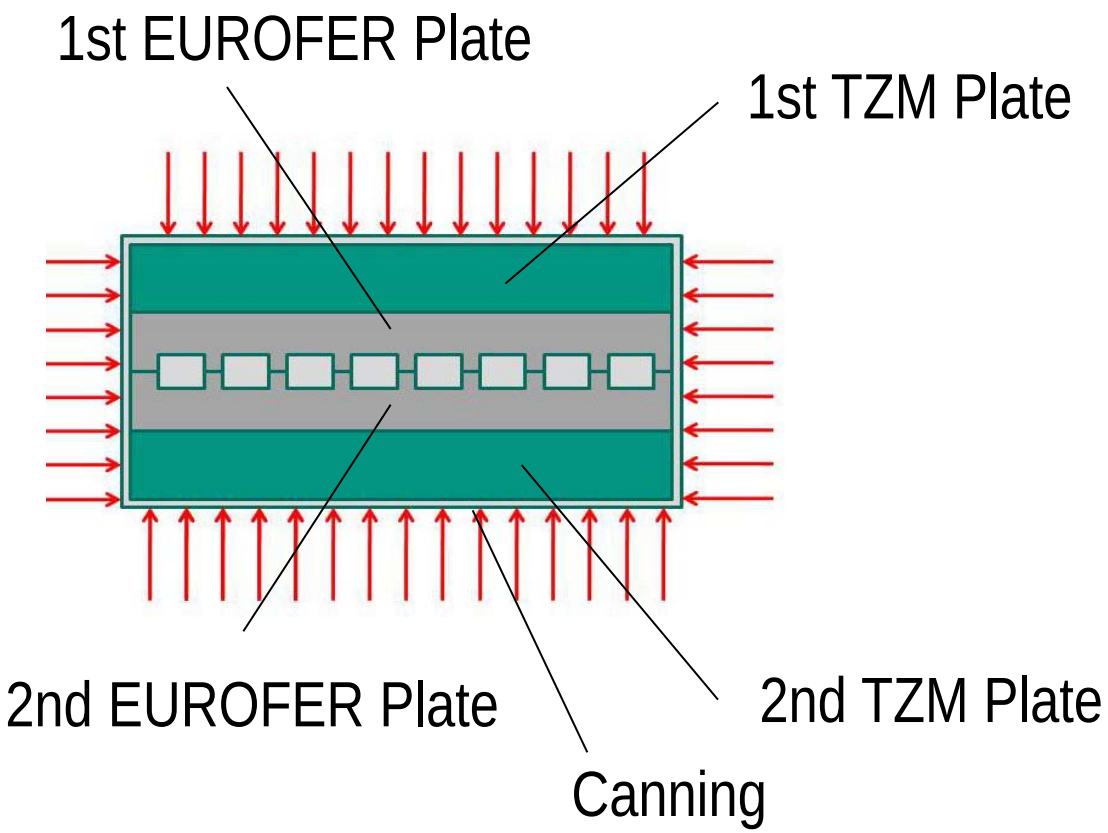
E.g. Hot Isostatic Pressing process for the FW



HIP-ped FW (L=80 cm) under leakage test at 8 MPa

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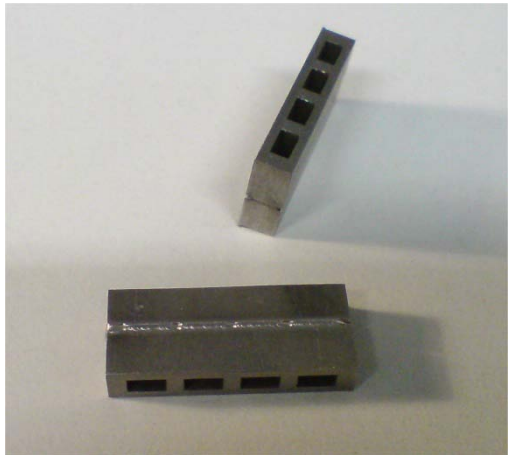
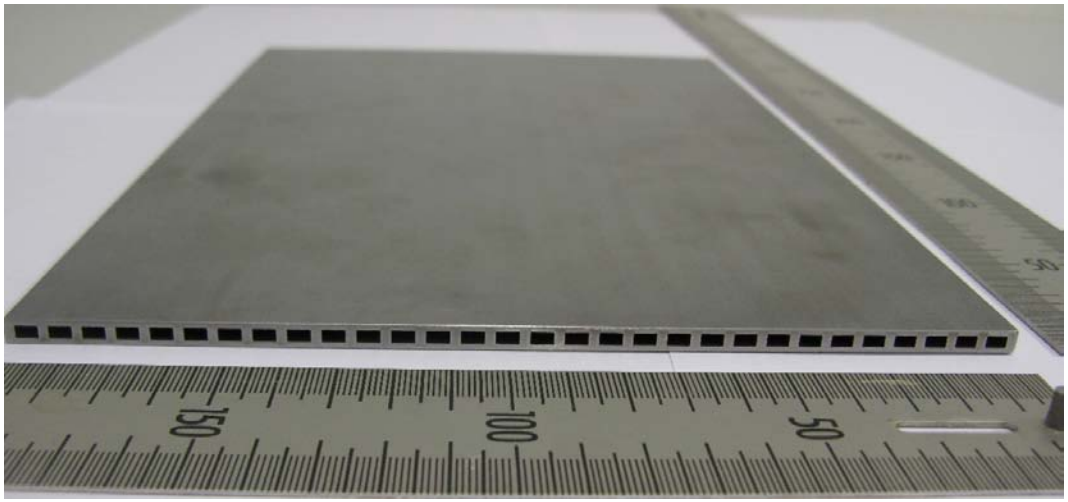
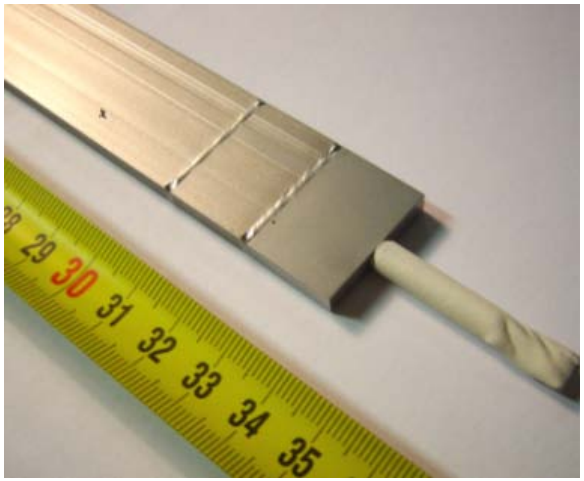
TBM Sub-component fabrication: Cooling plate of the BU



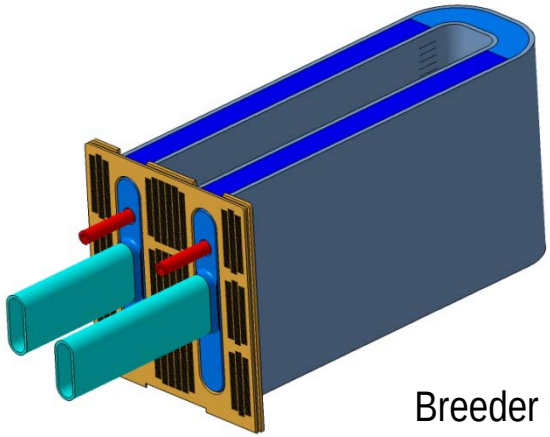
EB Welding optimization for CP assembly



Full Scale BZ Cooling Plate Segment: spark erosion



TBM relevant scale CP



Breeder Units

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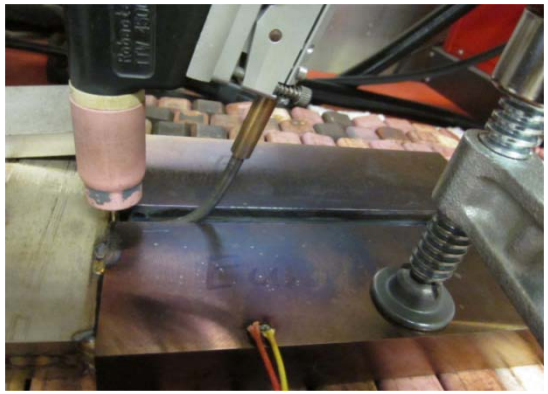
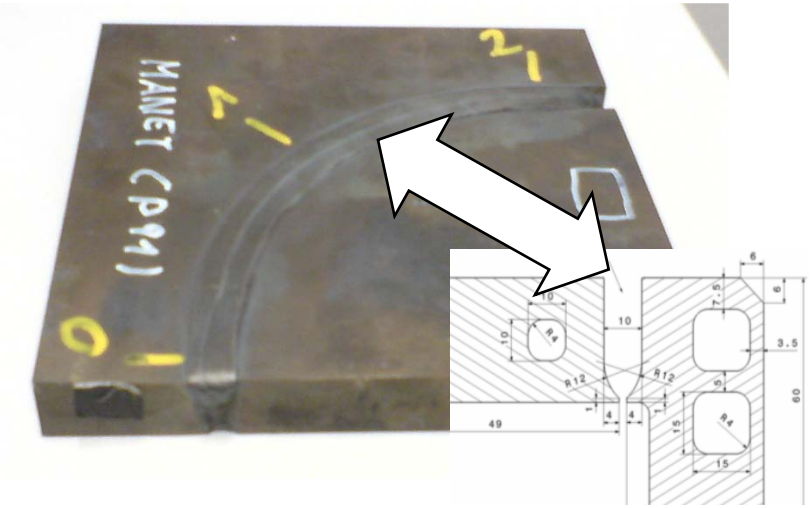
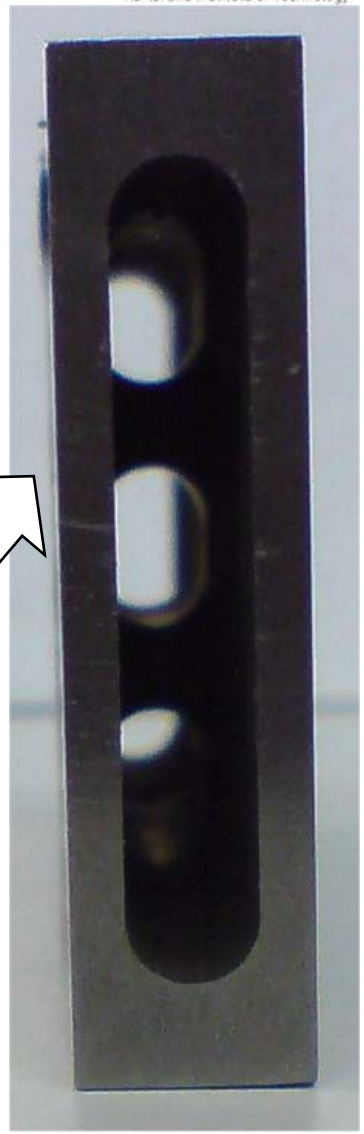
TBM box assembly (some examples):

30 mm FW/Cap-Welding experiments



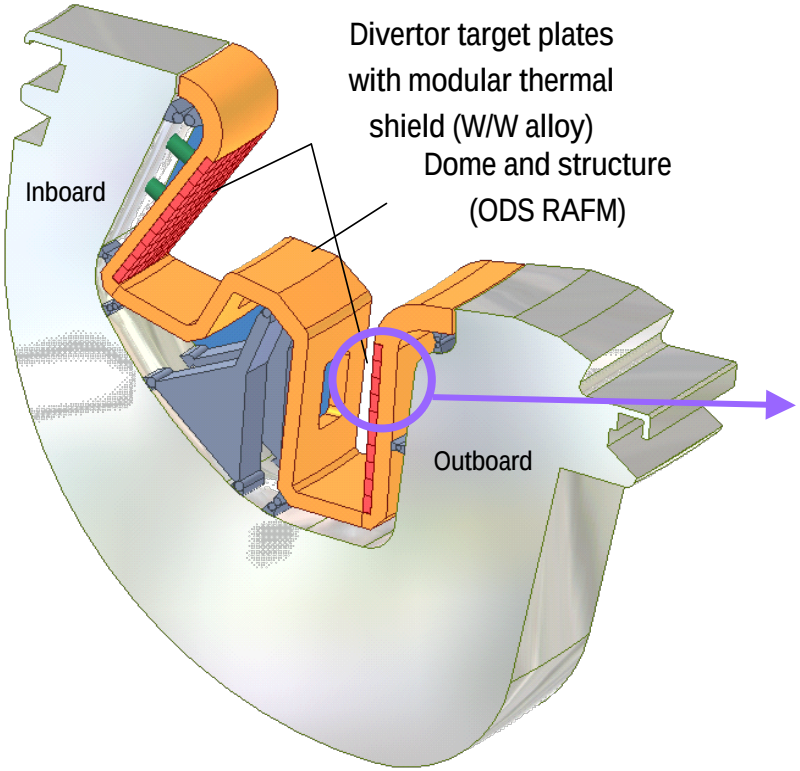




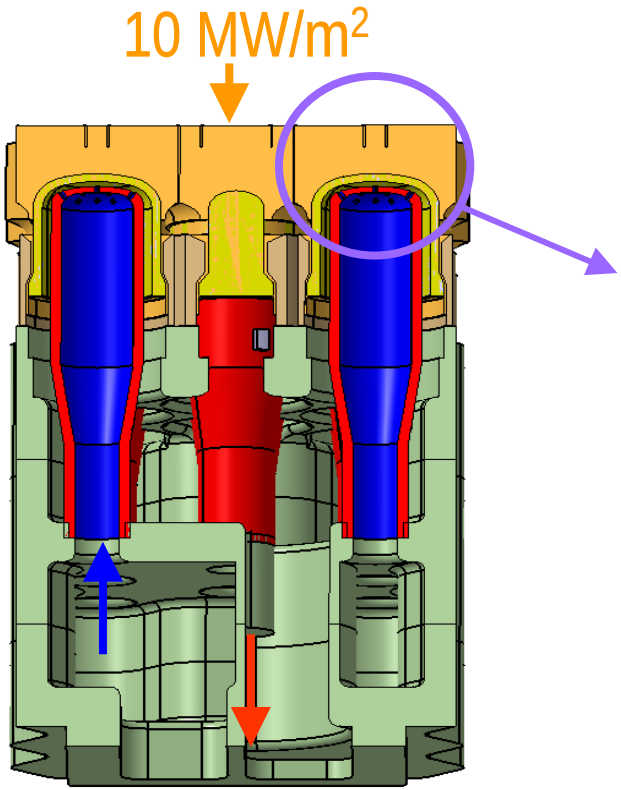
SP assembly welding MUs

The Reference Design: He-cooled modular divertor with jet cooling (HEMJ)

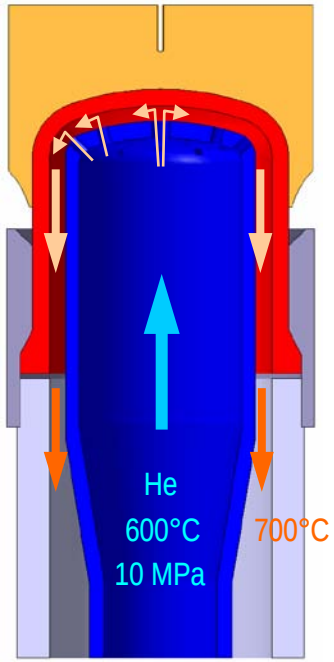
P. Norajitra, KIT



Divertor cassette

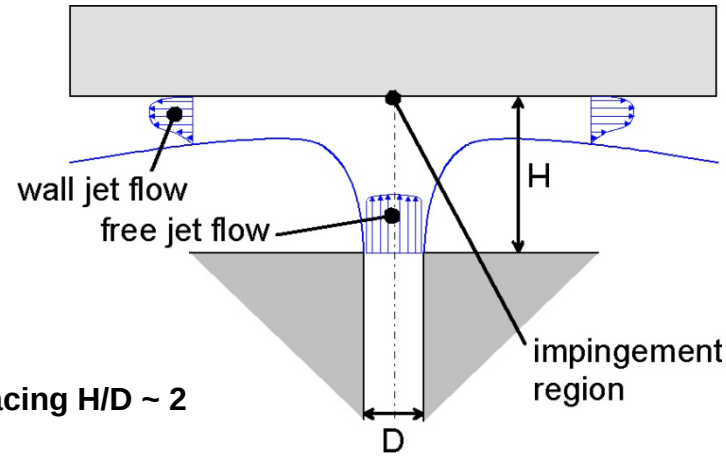


9-Finger module

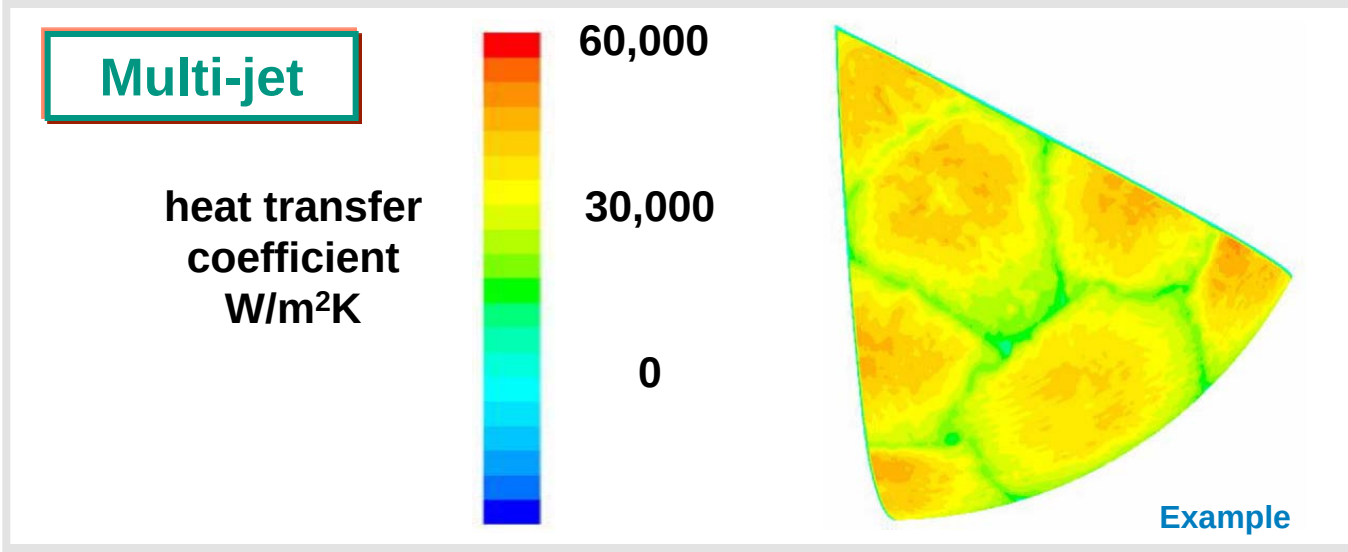


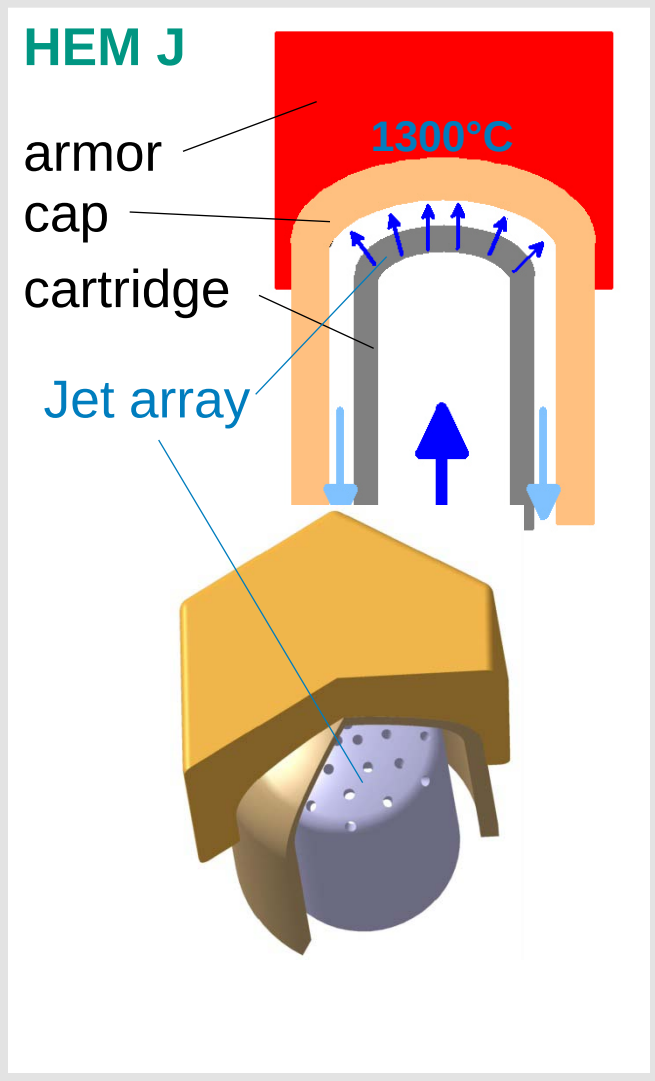
1-Finger module

HEMJ - Principles



Jet-wall-spacing H/D ~ 2





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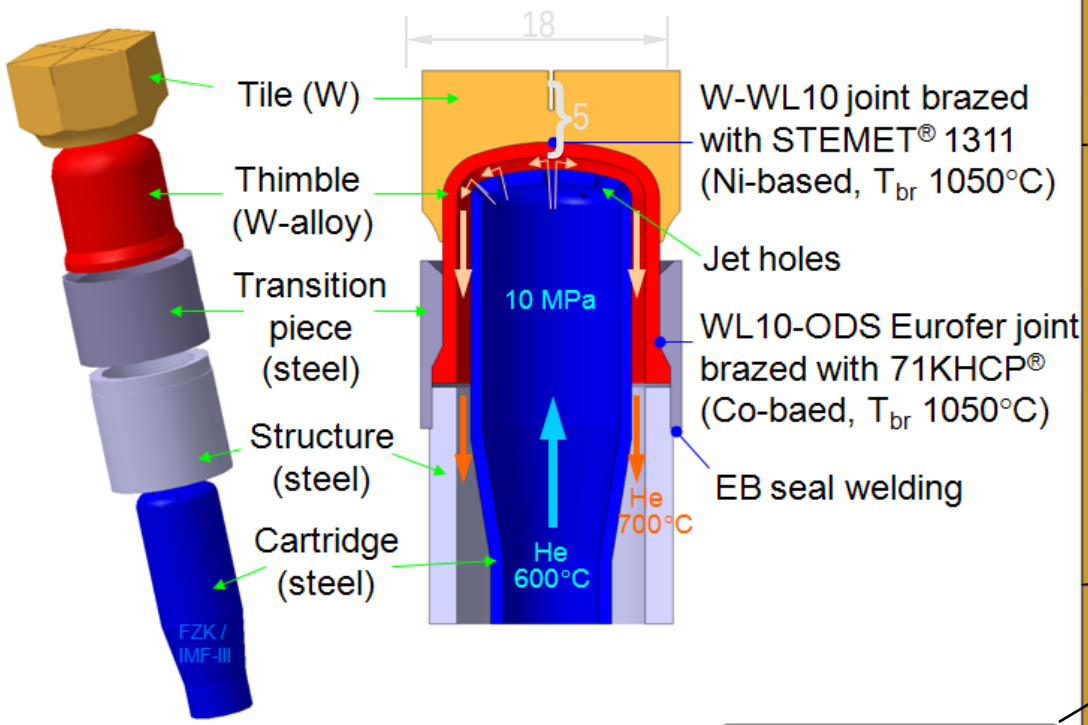
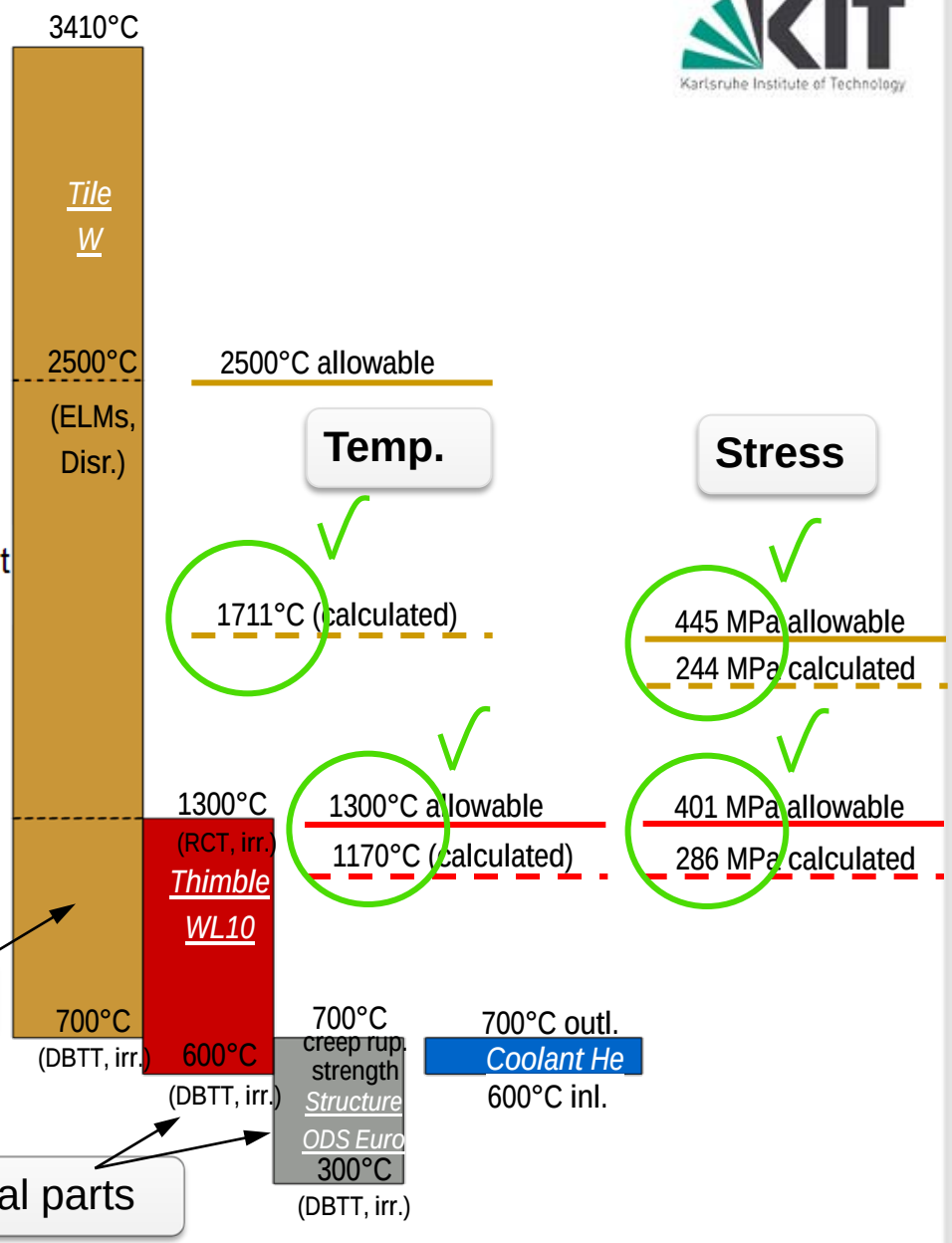
RD: Design temperature windows





Layout reference case:

- 10 MW/m^2
- mfr 6.8 g/s per finger
- pumping power ≤ 10%$Q_{removal}$

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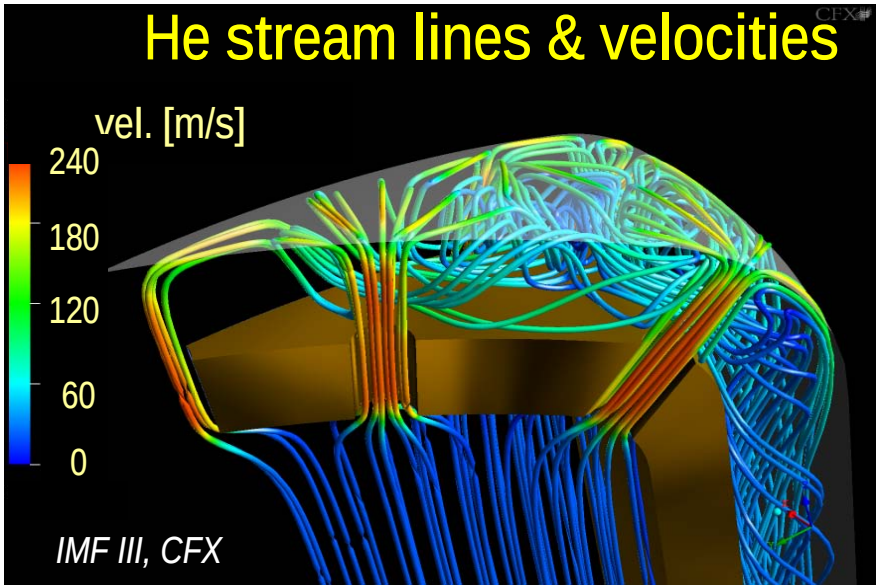
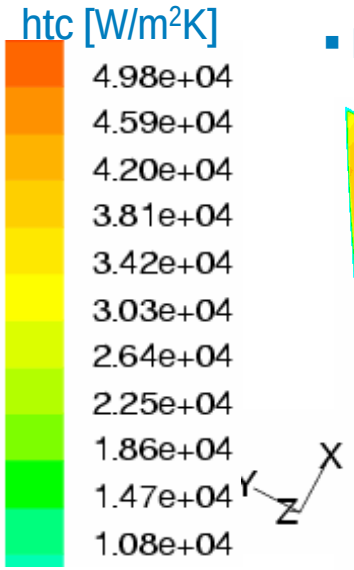
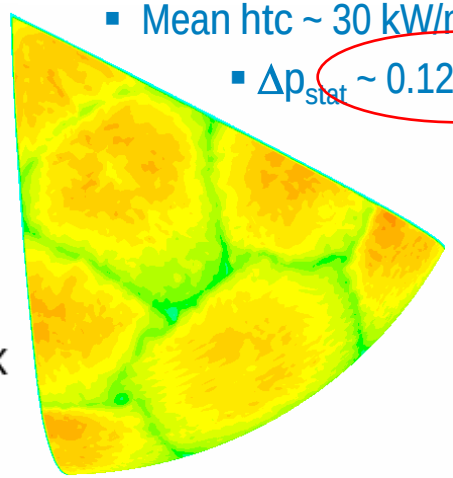
CFD results for RD: 10 MW/m², 10 MPa He, mfr 6.8 g/s

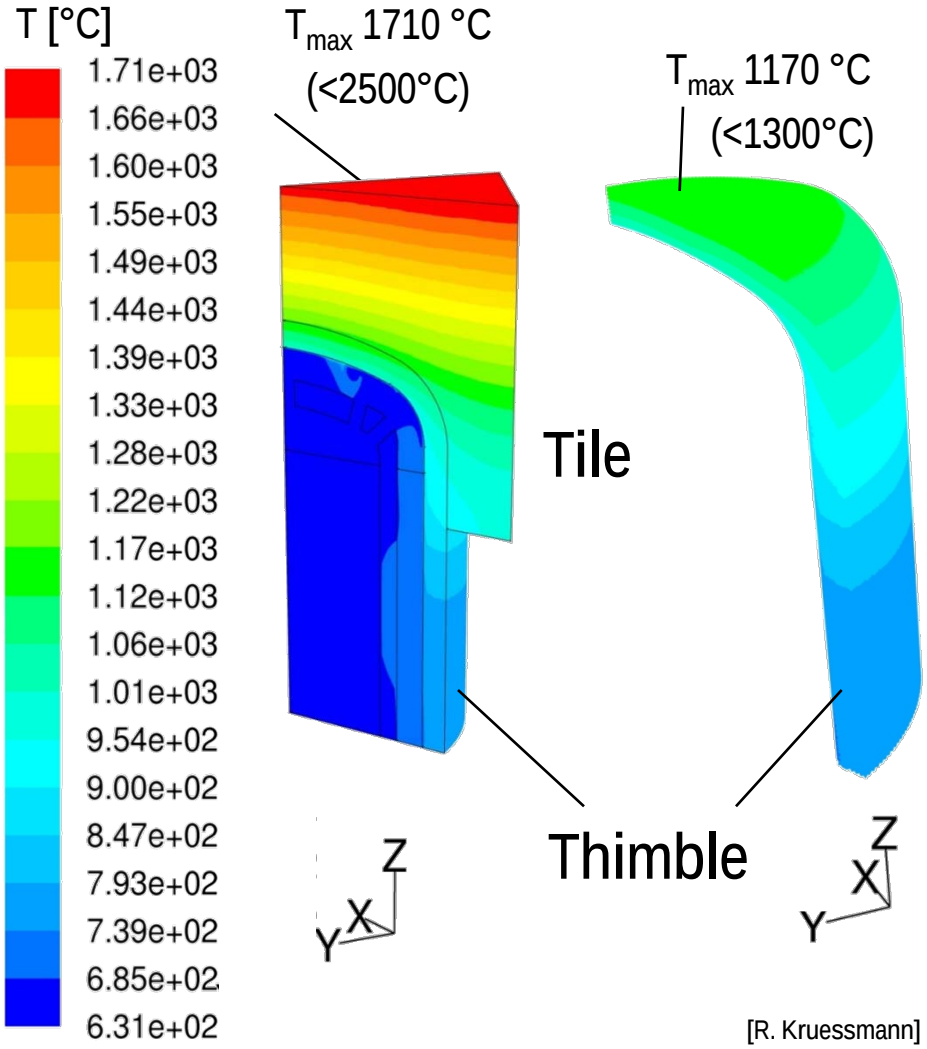




- Max. He velocity ~ 240 m/s
- Mean htc ~ 30 kW/m²K
- Δp_{stat} ~ 0.12 MPa



Temperature distribution



[R. Kruessmann]

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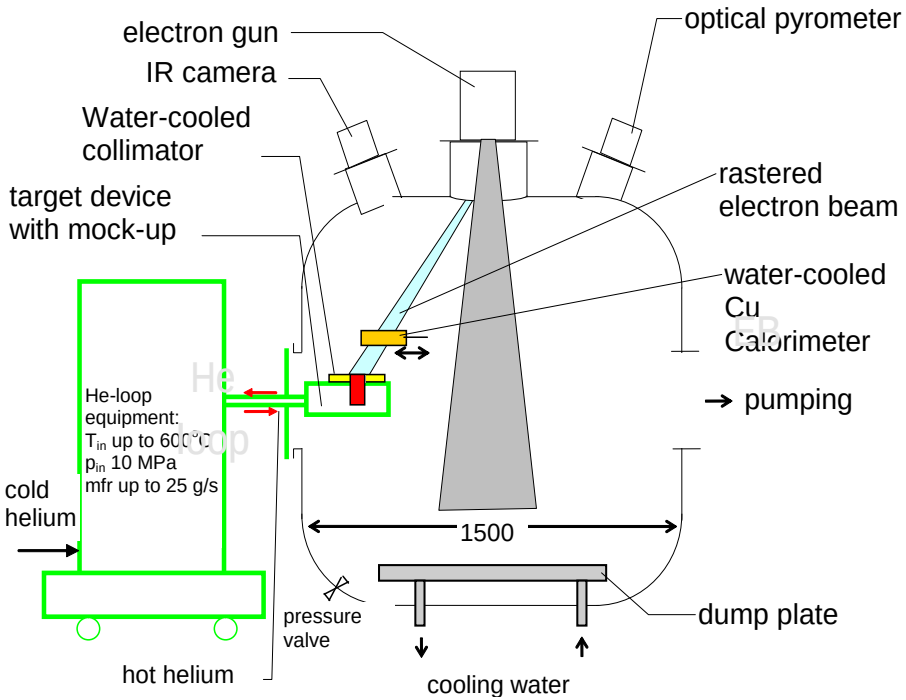
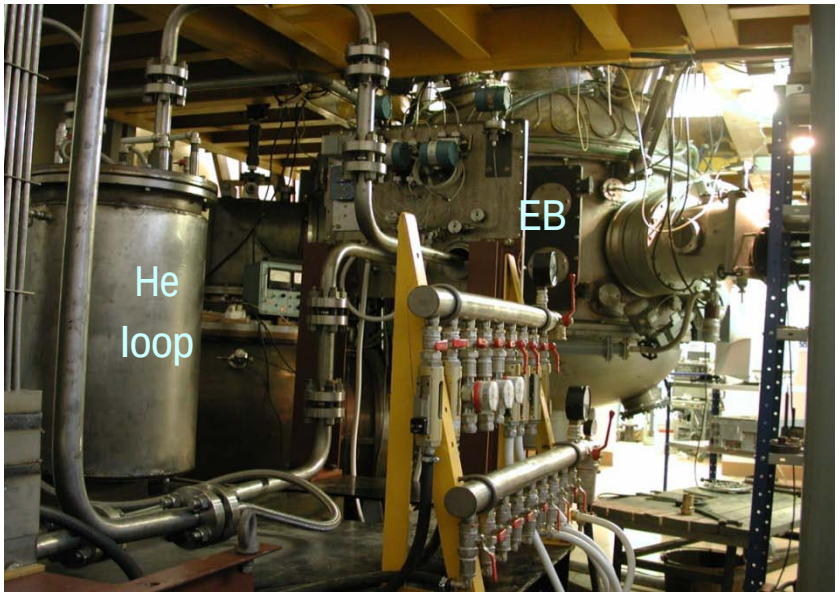
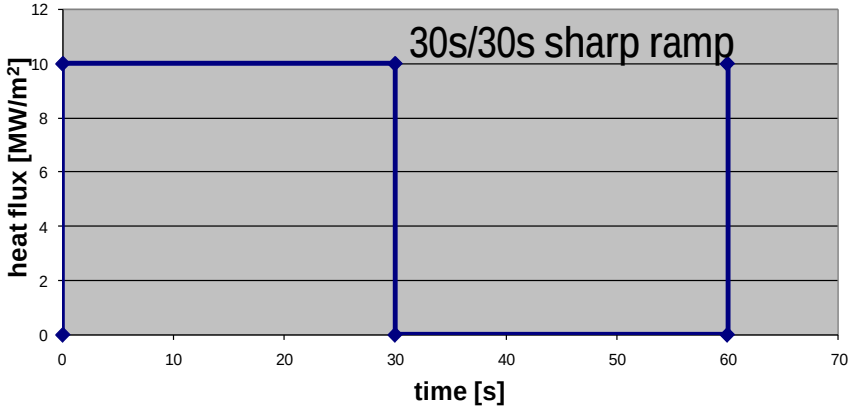
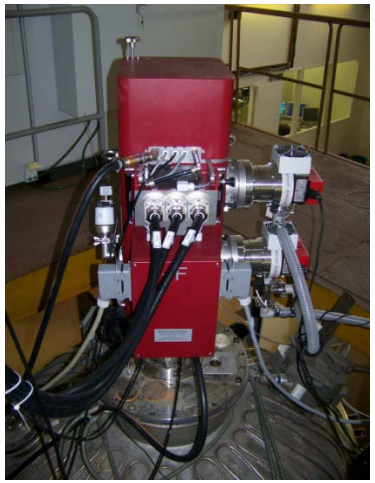
HHF performance tests in a combined He loop & Tsefey EB facility (200 kW, 40 kV) at EFREMOV





EFREMOV under KIT contract

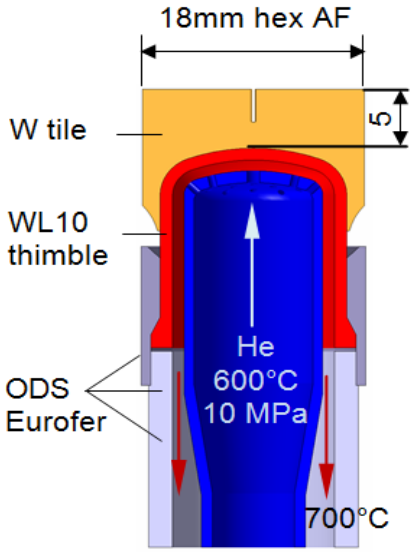


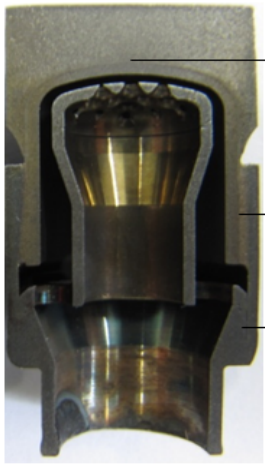


Highlight 2010:

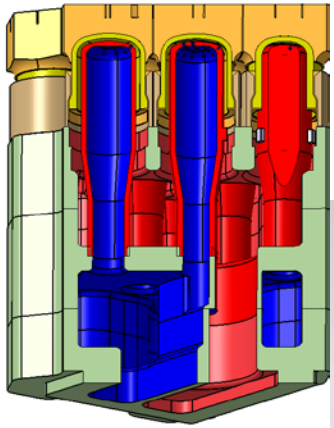
Mockup survived > 1000 cycles at 10 MW/m²

Last KIT self-produced mock-ups sent to Efremov for testing



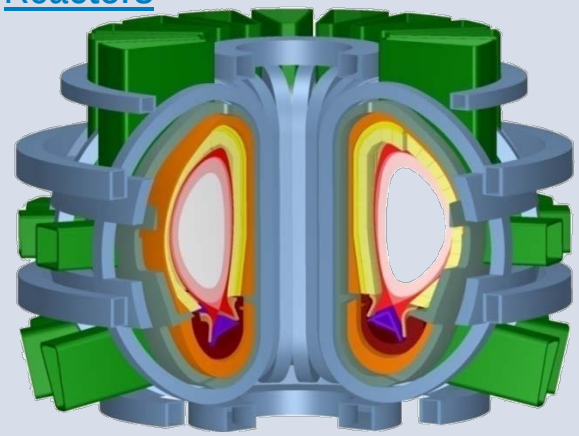


KIT self-produced 9-finger mockups waiting for assembly

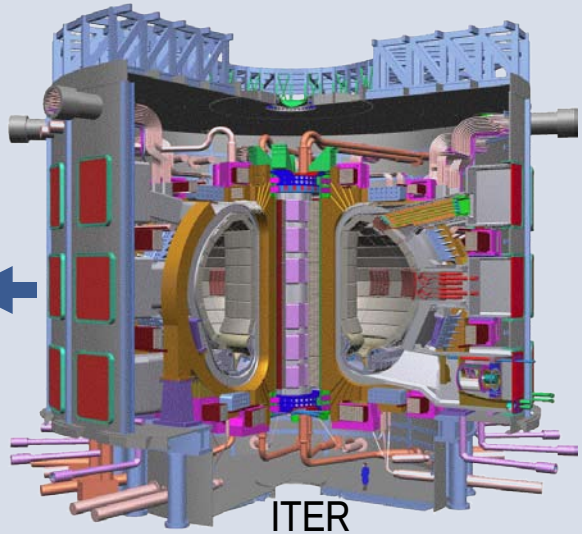


Roadmap Divertor Development towards ITER & DEMO

Reactors



DEMO
2038



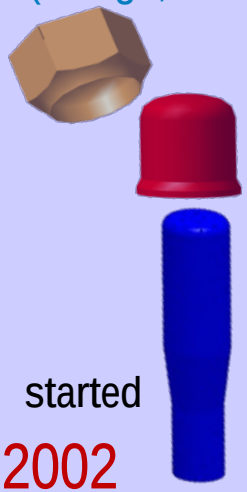
ITER
2020

Big parts develpoment

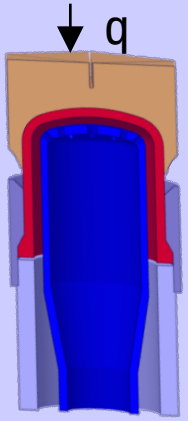
TDM
Cassette

HELOKA-Exp.

Small parts development
(Design, Materials, Joining)



started
2002



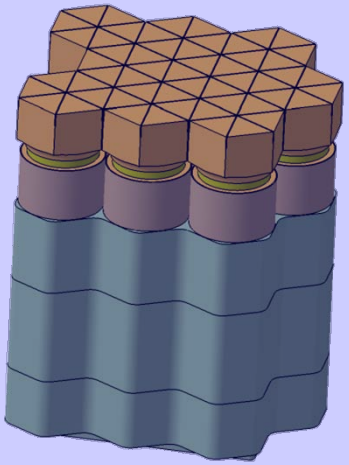
q

1-Finger unit

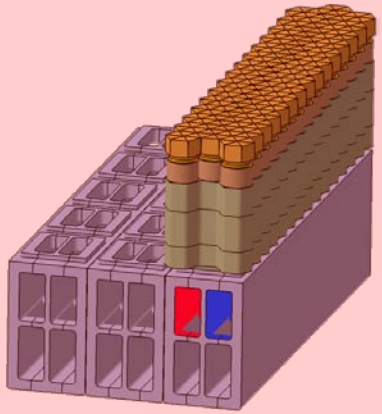
HHF Exp. at
EFREMOV

Reference design
HEMJ

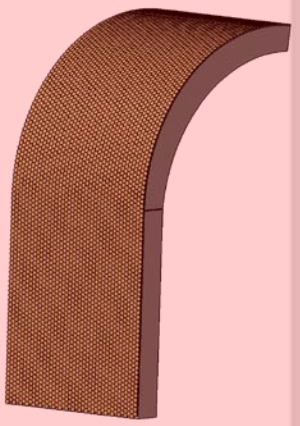
Goal:
q = 10 MW/m^2



9-Finger unit

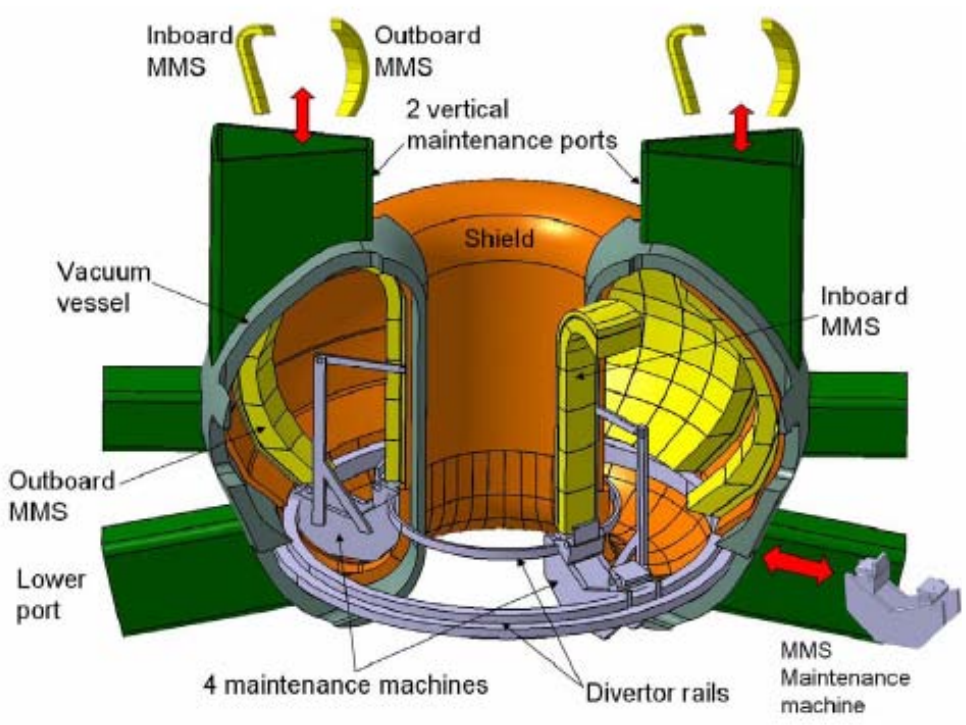


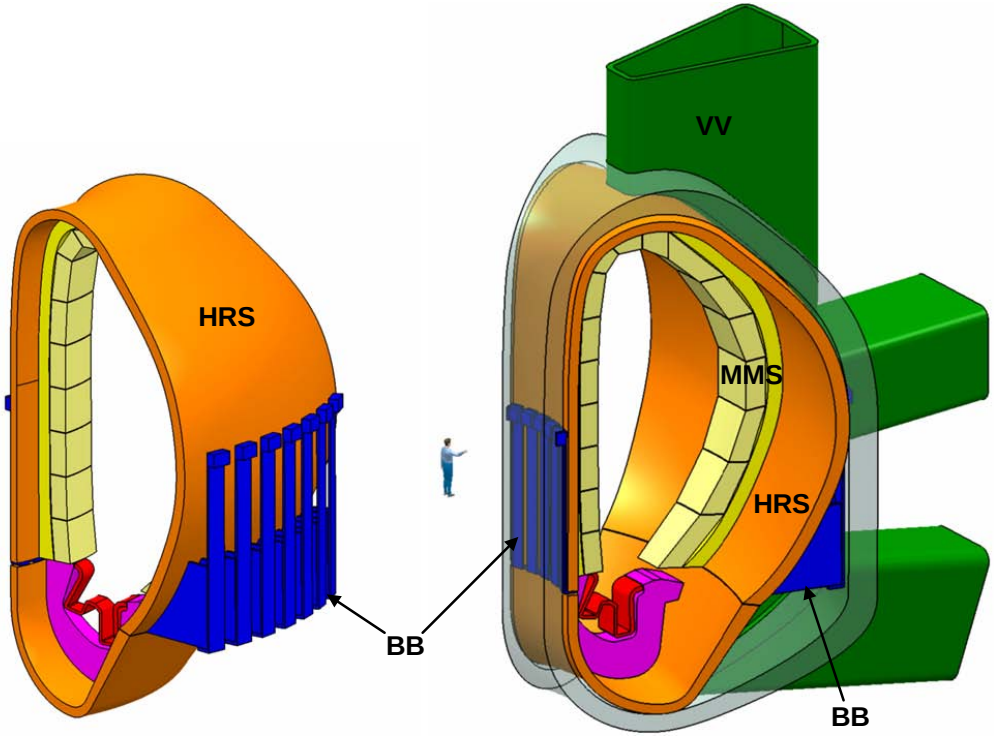
Stripe units



Targets

In vessel component maintenance and itegration





DEMO Multi Module Segment Concept with vertical port maintenance (2007)

See presentation of F. Cismondi

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Summary and future work

- The Germany fusion programme, organised under the Helmoltz Association umbrella, present a well coordinated and comprensive reserch programme covering almost all the topic of Plasma Physic and Fusion Technology.This programme is strongly integrated in the European Fusion Programme (EFDA and F4E).
- The Karlsruhe Institute of Technology the one of the largest center for Fusion Technology in EUROPE. Its activities are aimed to contribute to the ITER construction and, beyond this, to the design and construction of a DEMO reactor.
- In the next year(s) a new activity of Reactor Study is planned under EFDA umbrella (Power Plant Physics and Technology, PPPT) with the objective to develop a Fusion Power Plant conceptual design, integrating the technology and physics aspects for optimal performance, i.e. availability and efficiency by addressing the key issues in physics, technology and system engineering integration.

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