

Summary of International High Heat Flux Component Workshop, San Diego, December 2008

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**Japan/US Workshop on Power Plant Studies and Advanced
Technologies with Participation of China
University of Tokyo in Kashiwa, Japan
March 16-18, 2009**

ARIES Town Meetings and Workshops

- **The ARIES program organizes town meetings/workshops to provide a forum for discussions between scientists from R&D programs and power plant studies:**
 - **To help guide experimental programs towards solutions that lead to an attractive fusion power plant**
 - **To help design studies develop concepts that are consistent with the understanding of scientists developing those technologies.**
- **Consistent with ARIES mission statement:**
 - **Perform advanced integrated design studies of the long-term fusion energy embodiments to identify key R&D directions and provide visions for the program.**

Past ARIES Town Meetings Have Proven Very Valuable

Mar. 2-3, 1995	ANL	Workshop on Liquid Target Divertors	} <i>Starlite</i>
May 10, 1995	ANL	Starlite Town Meeting on Structural Materials	
Jan. 31, 1996	UCSD	Starlite Town Meeting on Low Aspect Ratio Spherical Tokamaks	} <i>Starlite</i>
June 19, 1997	UW	ARIES Town Meeting on Designing with Brittle Materials	<i>ARIES-RS</i>
May 6-7, 1998	UCSD	ARIES Town Meeting on ST Physics	<i>ARIES-ST</i>
Jan. 18-19, 2000	ORNL	International Town Meeting on SiC/SiC Design & Material Issues for Fusion Systems	<i>ARIES-AT</i>
Mar. 6-7, 2001	Livermore	ARIES Tritium Town Meeting	} <i>ARIES-IFE</i>
May 5-6, 2003	Livermore	ARIES Town Meeting on Liquid Wall Chamber Dynamics	
Sept. 15-16, 2005	PPPL	ARIES Compact Stellarator Physics Town Meeting	<i>ARIES-CS</i>

Background and Goals of Workshop

- **A topic of high current interest is the apparent disconnect or gap between near term and long term concepts for high heat flux components (HHFC, and in particular divertors).**
- **This is the focus of this workshop aimed at:**
 - **better characterizing the international status of current HHFC design concepts for power plants**
 - **comparing it to the present stage of development and experimental information for near term concepts (ITER-like);**
 - **better understanding how to evaluate where we are with respect to the end goal (power plant HHFC concepts) and what needs to be done to get there.**
- **The question carries also of course an important physics aspect in realistically determining the expected physics regime of operation in a power plant and the corresponding heat and particle fluxes on the divertor, and an important material aspect in designing the HHFC's for accommodation of the threats from the fusion environment.**
- **This topic is also now of particular interest in the USA as it relates to the work being done as part of the ReNeW effort.**
 - **The objective of the ReNeW project is to help OFES develop a plan for US fusion research during the ITER era, roughly the next two decades.**
 - **It is hoped that the outcome of the workshop will provide some useful information**

33 Registered Participants

(including 8 from EU and 1 from Japan)

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Agenda

International HHFC Workshop on Readiness to Proceed from Near Term Fusion Systems to Power Plants
December 10-12, 2008
CMRR Auditorium, UCSD, La Jolla, CA

Agenda

Wednesday December 10

8:00-8:30 Coffee and socializing
 8:30-8:35 Welcome/logistics R. Raffray
 8:35-8:45 Welcome from CER M. Tillack
 8:45-8:55 Background and goals of the workshop R. Raffray/R. Nygren/
 D. Whyte

Session 1 Near term HHFC design and R&D (ITER)

Session coordinator: R. Nygren
 8:55-9:30 ITER PFC (Divertor, First Wall) design M. Merola
 9:30-10:05 EU considerations on design and qualification of PFC's for near term machines (ITER) P. Lorenzetto

10:05-10:20 Break

10:20-10:55 Near term to long term PFC/divertor considerations R. Nygren
 10:55-11:30 EU PFC/divertor concepts for power plants P. Norajitra

11:30-11:45 Discussion

11:45-1:00 "Smile" + Lunch

1:00-2:00 Tours (PISCES and Laser Labs)

Session 2 Long term HHFC concepts (power plants)

Session coordinator: R. Raffray
 2:00-2:35 US PFC/divertor concepts for power plants R. Raffray
 2:35-3:10 Japan considerations on design and qualification of PFC's for near term machines (ITER) S. Suzuki
 (presented by S. Konishi)

3:10-3:25 Break

3:25-4:00 Japan PFC/divertor concepts for power plants S. Konishi/Y. Ueda

4:00-4:30 Discussion

Session 3 Material aspects for near term and long term concepts

Session 3.1 Material - Development/Fabrication/Joining

Session coordinator: D. Hoelzer (TBD)
 4:30-5:05 Regular and ODS Ferritic steel as structural material for power plant HHFC's D. Hoelzer
 5:05-5:40 W as structural material for power plant HHFC's M. Rieth

5:40 Adjourn

6:45 Dinner (El Fandango Restaurant, Old Town)

Thursday December 11

8:00-8:30 Coffee and socializing

Session 3 Material aspects for near term and long term concepts

Session 3.2 Material - PMI and High heat flux testing

Session coordinator: M. Roedig
 8:30-9:05 Recent PMI experience in Tokamaks R. Neu
 9:05-9:40 Tritium behavior in different PFC armors from PISCES R. Doerner
 9:40-10:15 High heat flux testing of different armor M. Roedig

10:15-10:30 Break

10:30-11:05 High heat flux and CHF testing in support of ITER F. Escourbiac
 11:05-11:40 Experimental validation of thermal performance of gas-cooled divertors S. Abdel-Khalik

11:40-12:25 Discussion

12:25-1:40 Lunch

Session 4 Physics considerations for near term and long term concepts

Session coordinator: D. Whyte
 1:40-2:15 Near term SOL and pedestal physics issues for ITER R. Pitts
 2:15-2:50 Stellarator as alternate confinement: Benefits and issues R. Maingi
 2:50-3:25 Empirical and modeling scalings of SOL/divertor profiles T. Rognlien

3:25-3:40 Break

3:40-4:15 Controlling SOL & pedestal MHD T. Evans
 4:15-4:50 Physics "gap" issues from ITER to reactors for integrating HHFC and edge plasmas D. Whyte

4:50-5:35 Discussion

5:35 Adjourn

Friday December 12

8:00-8:30 Coffee and socializing

Session 5 Evaluation methodology to evaluate current readiness level vis à vis the long term objective and how to get there

Session coordinator: M. Tillack
 8:30-9:05 Technical Readiness Levels applied to HHFC M. Tillack
 9:05-9:50 Discussion

9:50-10:05 Break

10:05-12:30 Final discussion, wrap-up, summary and writing assignments
 R. Raffray/R. Nygren/D. Whyte

12:30 Adjourn

Observations from the Workshop

- **Time does not allow a detailed summary of each presentation**
- **Example results from different presentations shown to illustrate some of the key observations**
- **More details will be provided in workshop report/publication**

Major Observations from IHHFC Workshop Include:

PFC GAP BETWEEN ITER AND POWER PLANT:

1. **Divertor materials and conditions**
2. **Level of R&D effort to-date**
3. **Steady state and transient loads**
4. **Plasma/Material Interaction conditions**
5. **Technology Readiness Level**

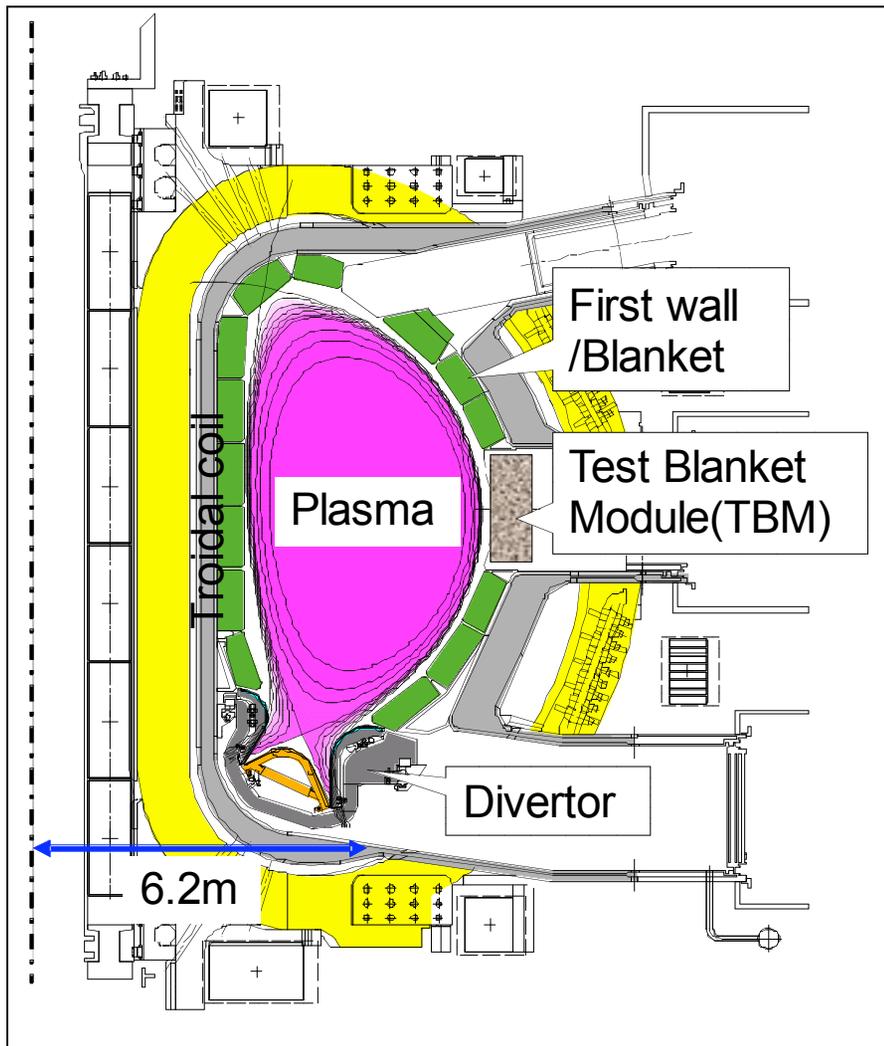
OBSERVATION 1A

ITER divertor is based non-reactor relevant material, coolant and operating conditions:

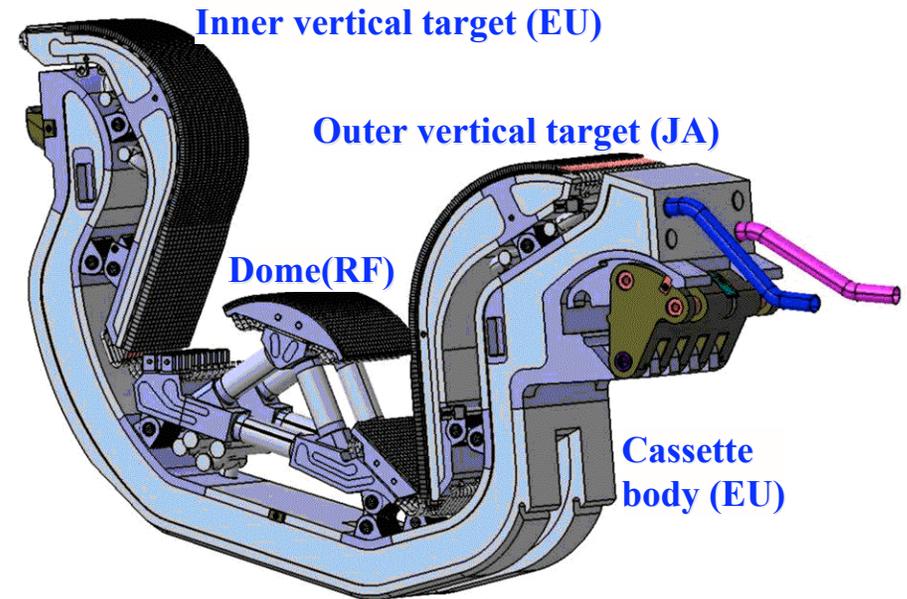
- **Low-temperature subcooled water as coolant**
 - **100°C; 4.2 MPa**
- **Low-temperature CuCrZr as PFC structural material (coolant tube) and austenitic SS for cassette body**
 - **CuCrZr ok for low temperature, very low fluence**
- **CFC or W as armor**

ITER Plasma Facing Components

(from S. Konishi/S. Suzuki's presentation)



Cross sectional view of ITER

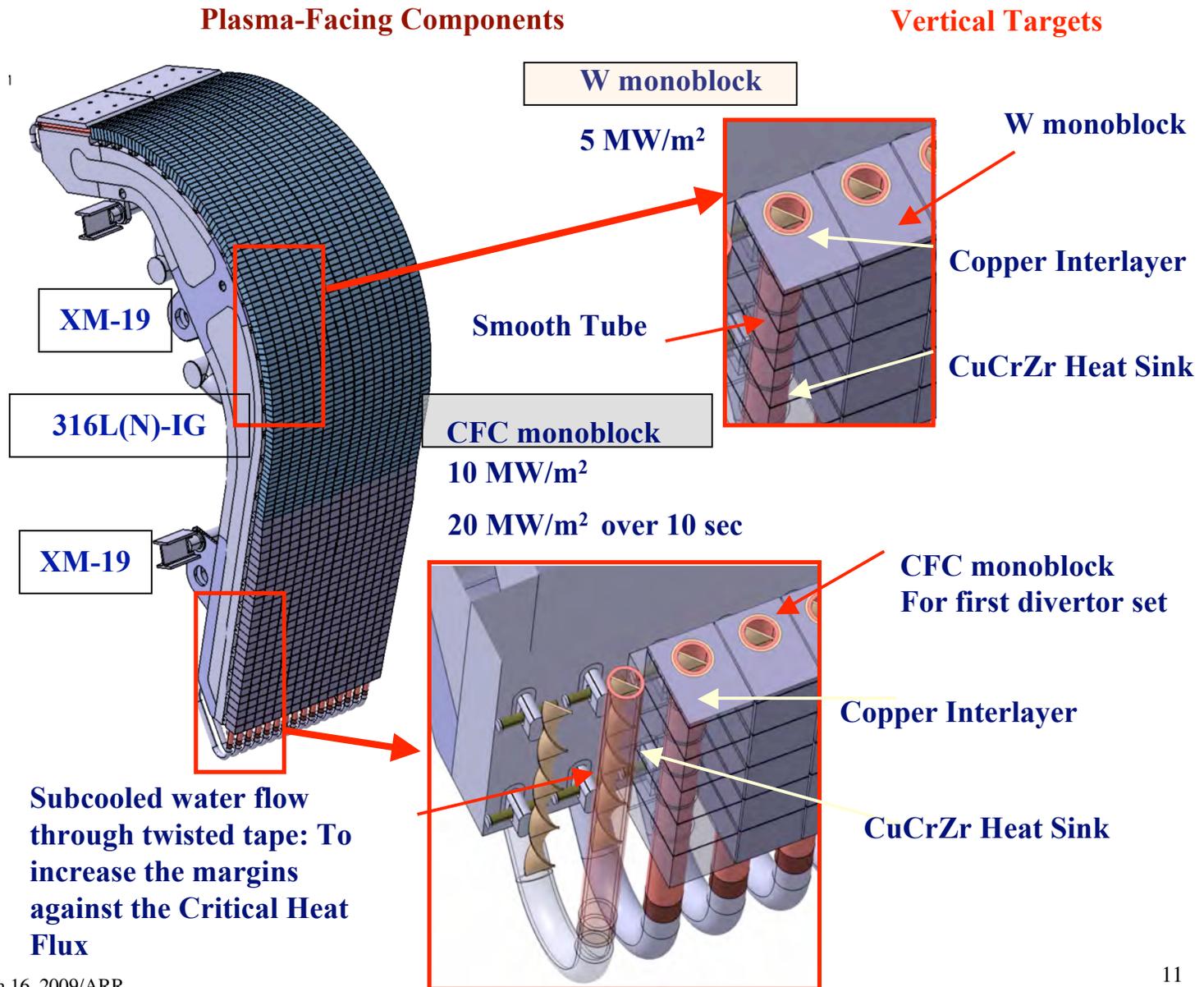


ITER divertor cassette

- 54 Cassettes
- Procurement shared among EU, Japan and RF

ITER Divertor Tube Configuration

(from M. Merola's presentation)



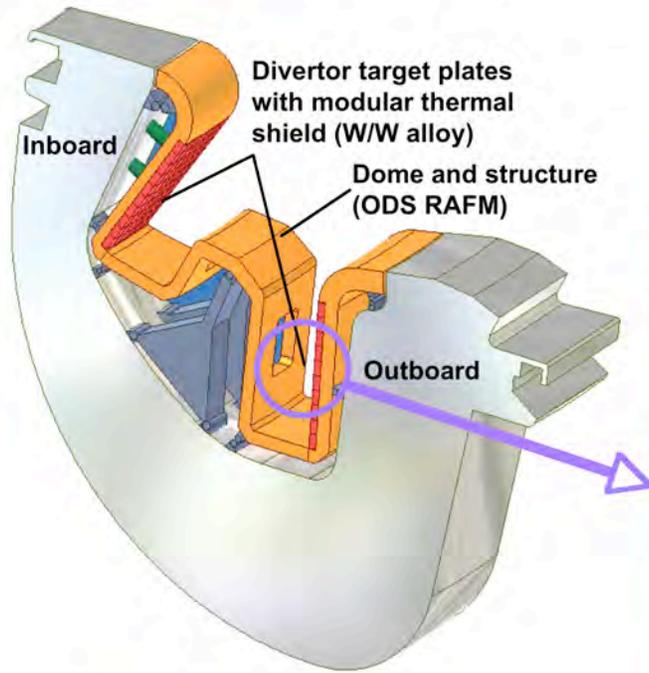
OBSERVATION 1B

Typical tokamak power-plant (DEMO) divertor (at least in EU and US) based on operating conditions and materials very different from ITER:

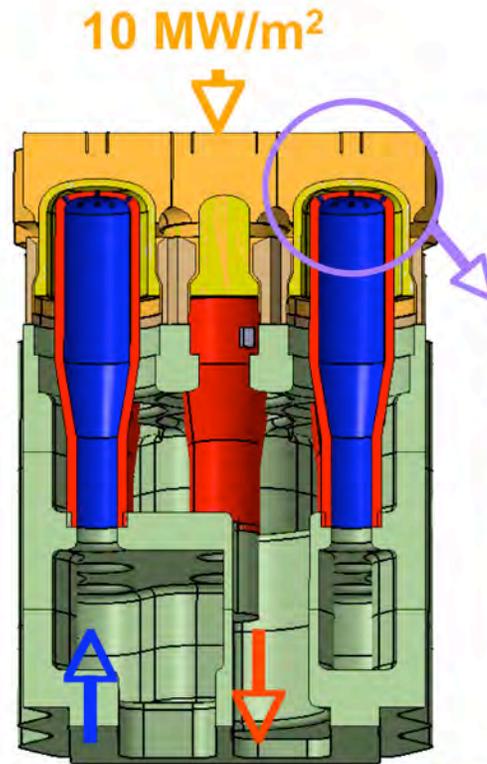
- **He-cooled, W-alloy concepts to accommodate ~ 10 MW/m²**
 - He coolant: ~ 10 MPa; ~ 600 - 700°C
 - W-alloy ~ 700 - 1300°C
 - W armor ~ 1500 - 1800°C
 - W-alloy joined to ODS FS
- **Also advanced design with Pb-17Li + SiC_f/SiC but limited capability to accommodate high heat flux**

He-cooled modular divertor with jet cooling (HEMJ)

(from P. Norajitra's presentation)

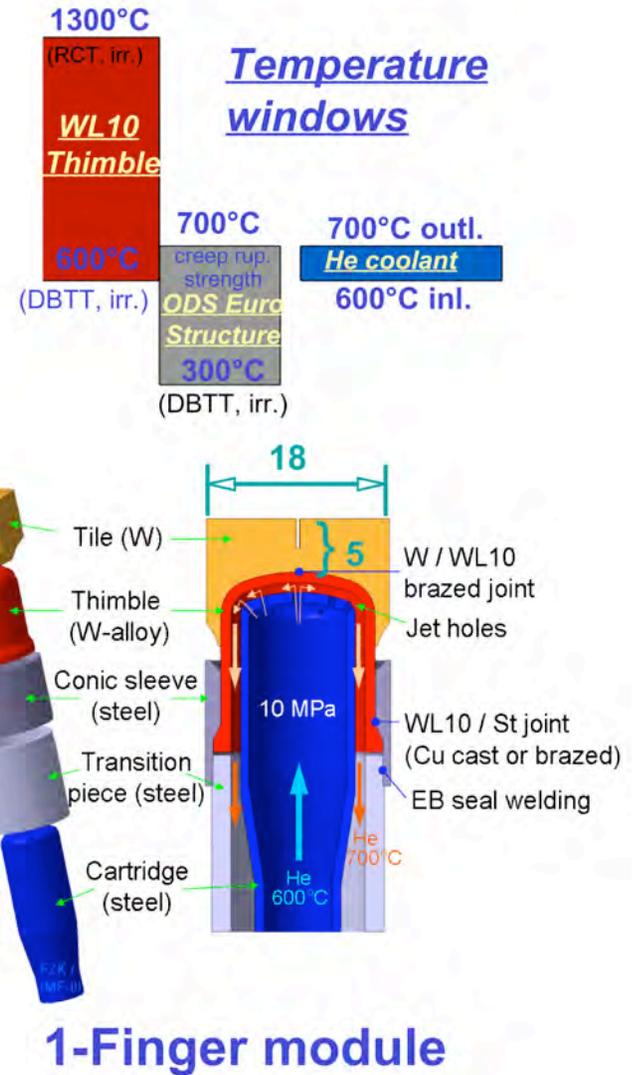


Divertor cassette



9-Finger module

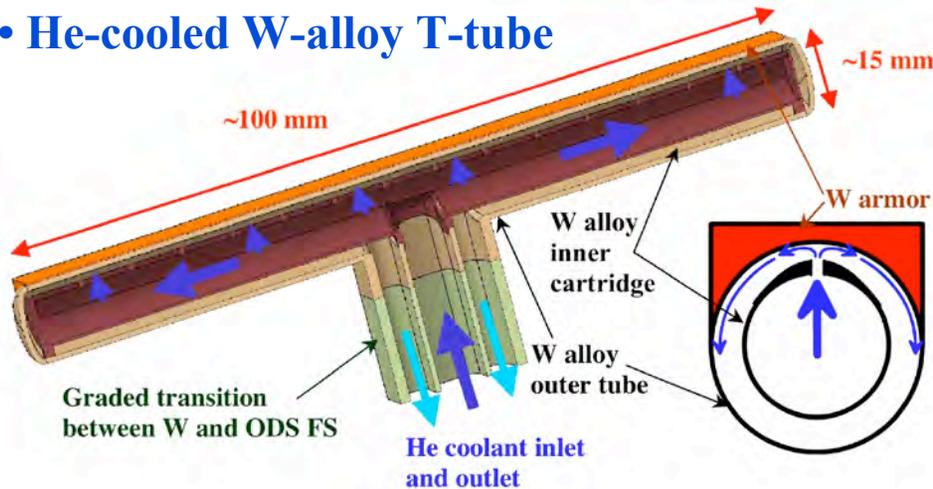
[T. Ihli]



ARIES T-Tube Divertor Design

(from A. R. Raffray's presentation)

- He-cooled W-alloy T-tube



- Design for a max. q'' of at least 10 MW/m^2

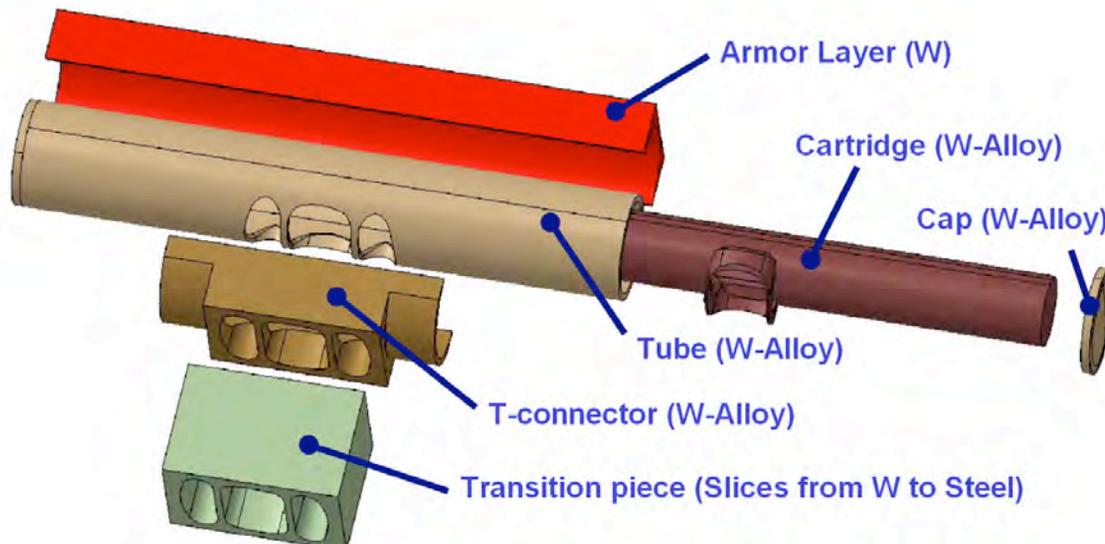
- Mid-size configuration with credible manufacturing and assembly procedures (for CS or Tokamak application).

- Cooling with discrete or continuous jets through thin slots ($\sim 0.4 \text{ mm}$)

- 10 MPa , $\sim 600\text{-}700^\circ\text{C}$ He coolant

- $\sim 600/700^\circ\text{C}$ to 1300°C W-alloy

- A number of such T-tubes can be connected to a common manifold to form desired divertor target plate area.



OBSERVATION 2A

ITER divertor just went through its final design review to evaluate readiness for procurement:

- **Mature and optimized component design and technology**
- **Extensive R&D and testing over last 15 years +**

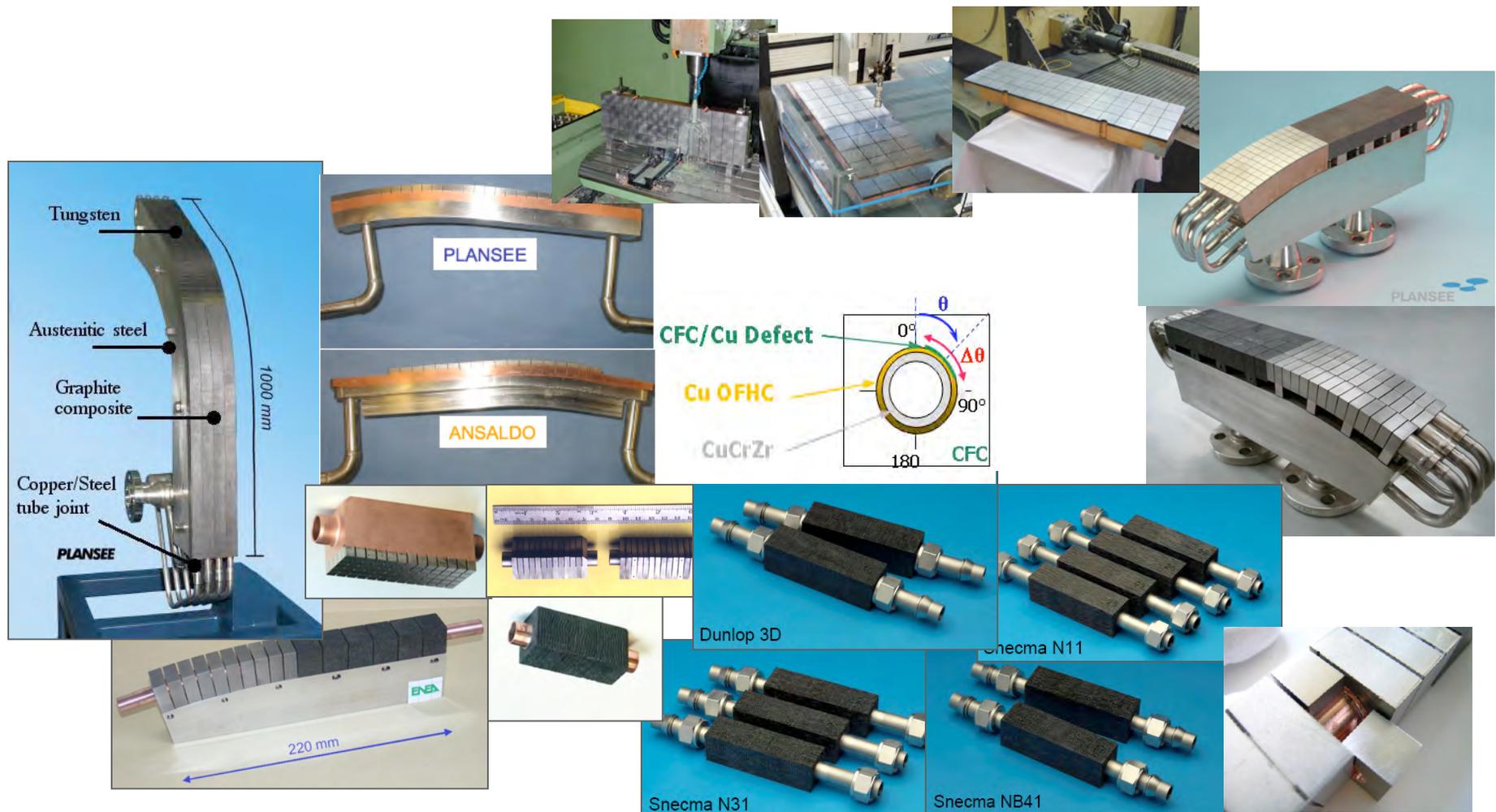
Strategy for Material Selection

(from P. Lorenzetto's presentation)

- The material choice for ITER was performed based on industrially available materials while taking into account their physical and mechanical properties, maintainability, reliability, corrosion performance and safety requirements at the ITER operational conditions.
- Experience from current tokamaks has been taken into account, (plasma facing, diagnostics materials, etc.).
- Knowledge from fission neutron irradiation programs was used.

Example of Extensive R&D Program for ITER Divertor (performed in the EU) (from P. Lorenzetto's presentation)

- Extensive R&D (small to large scale): manufacturing, testing and qualification effort (15+ years).



OBSERVATION 2B

He-cooled W-alloy power plant divertor at the conceptual design stage

- **Promising designs, but not optimized, not mature**
- **Relatively little R&D so far**
- **Extensive material development, R&D and testing needed**

Important Design Criteria

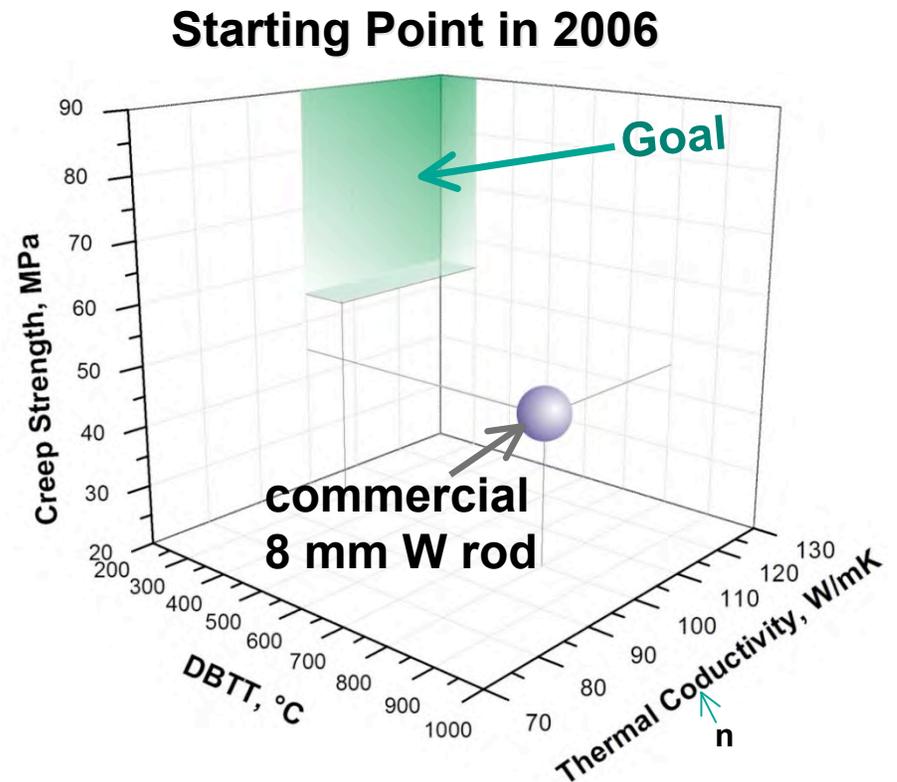
(from M. Rieth's presentation)

Thermal Conductivity → **100 W/mK @ 1200 °C**

Creep Strength → **55 MPa, 20 kh @ 1200 °C**

DBTT (EU Mini Charpy) → **300 °C, un-irradiated**

Recrystallization Temperature → **1300 °C, for 20 kh**



Conclusions on W

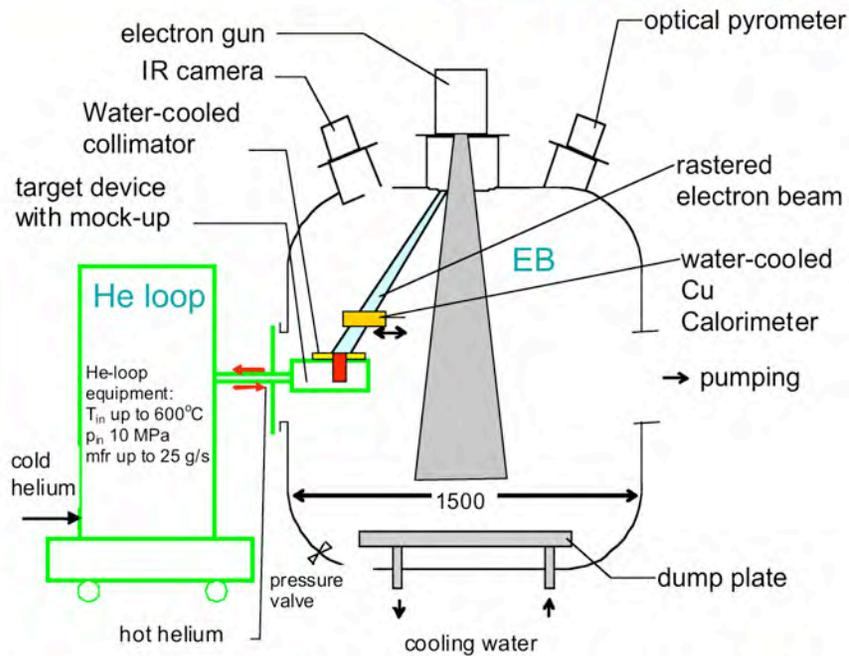
(from M. Rieth's presentation)

- ➔ Long-term creep strength and recrystallization behavior have still to be examined
- ➔ Microstructure significantly defines transition temperatures (rod texture more favorable than that of plates)
- ➔ Oxide particles (and also potassium doping) promote delamination (but they are necessary for stabilizing GB → suppr. re-crystallization)
- ➔ Tungsten materials have a DBTT limit of $\geq 400^{\circ}\text{C}$ (when produced by sintering & deformation, tested according to DIN EN ISO 148-1, ...)
- ➔ Notches/edges have to be avoided in structural parts
- ➔ Optimum fabrication probably only by aligning grains along the contour of the according part → deep drawing, twisting, pressing, ...
 - **Also effect of irradiation on embrittlement**
 - **Still a lot to do to develop W-alloy material with operating temperature window of $\sim 700/800$ to 1300°C and ODS-FS at $700/800^{\circ}\text{C}$ for joining**

Example of initial small-scale HHF tests for He-Cooled W finger unit

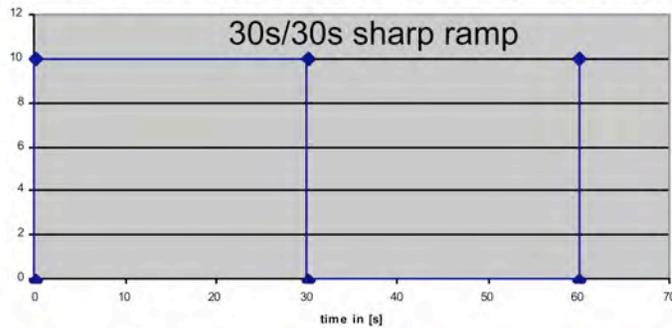
(from P. Norajitra's presentation)

HHF tests in a combined He loop & Tsefey EB facility (60 kW, 27 keV) at EFREMOV

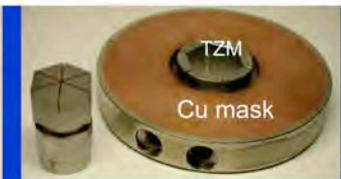


EFREMOV under FZK contract

Successful temperature cycle tests at 10 MW/m²



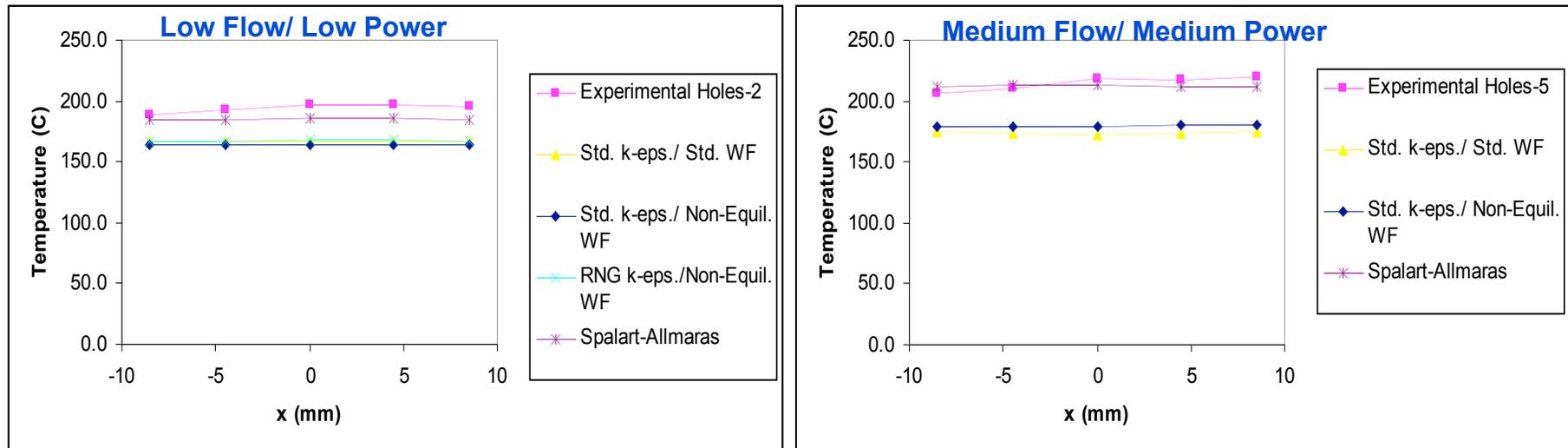
Mock-ups and mock-up holder



Initial Work on Validating CFD Code for Divertor Analysis by Comparison to Scaled Experimental Results Successful

(from S. Abdel-Khalik's presentation)

FLUENT vs. Experimental



- **Mock-up to simulate flow geometry of different divertor concepts**
- **Good agreement between experimental and numerical results**
- **Validated CFD Codes can be used with confidence to predict performance of gas-cooled components with complex geometries**

OBSERVATION 3A

ITER divertor and PFC's designed for demanding quasi steady-state and off-normal events

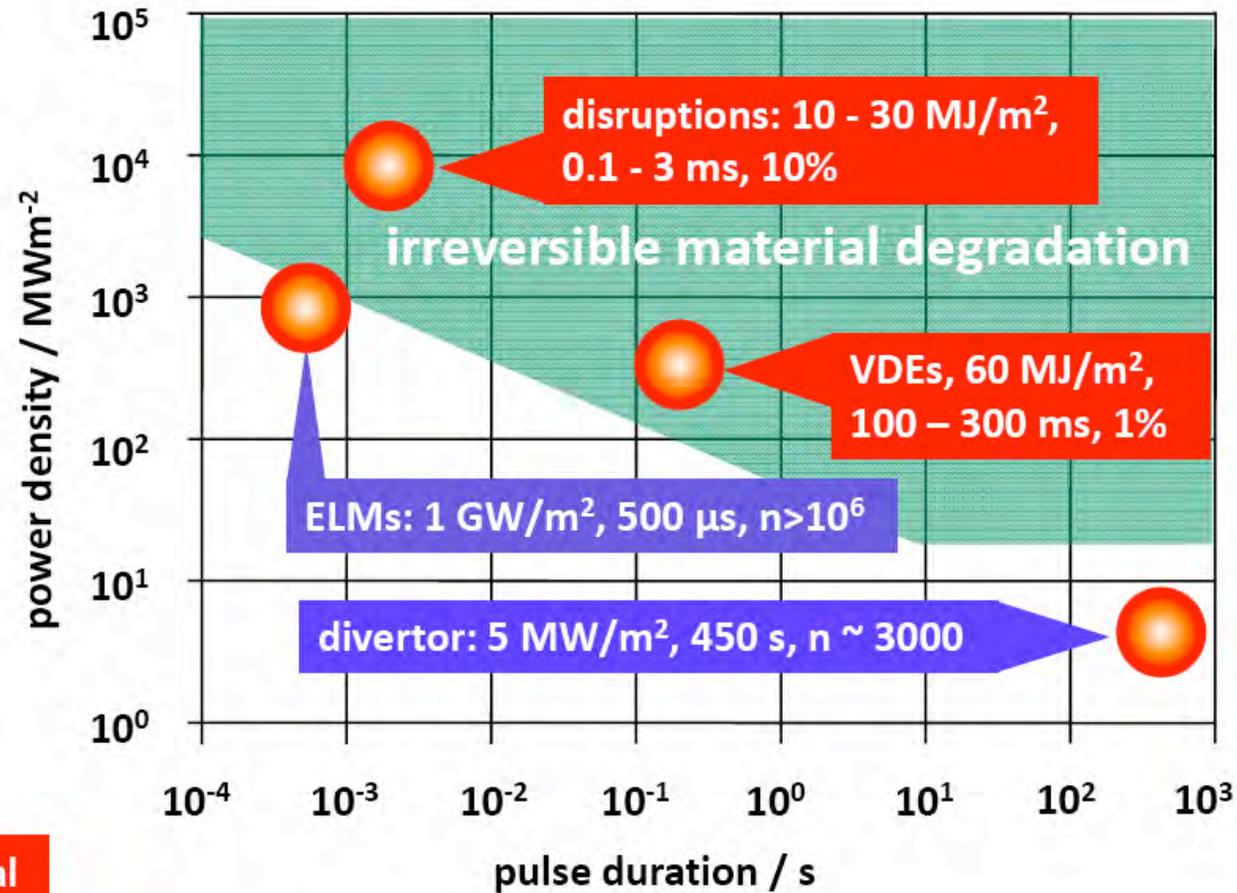
- **Low-temperature provides better accommodation margin**
- **Divertor design load**
 - **Steady state: 10 MW/m²**
 - **Slow transient: 20 MW/m² for <10s**
- **Off-normal or transient events include:**
 - **Disruptions**
 - **VDE's**
 - **ELM's**

Transient Loads from ITER Project Integration and Load Specification Documents and presentations at 2007 ITER WG8 Design Review Meeting

- **Disruptions**
 - Parallel energy density for thermal quench = 28-45 MJ/m² near X-point
 - Deposition time ~ 1-3 ms
 - Perpendicular energy deposition will be lower, depending on incidence angle
 - Parallel energy deposition for current quench = 2.5 MJ/m²
 - No. of Type I/Type II disruptions = 1000/100
- **VDE's (Type I/Type II):**
 - Energy deposition = 30/60 MJ/m²
 - Deposition time ~ 0.05-0.1/0.1-0.2 s
 - Number of VDE's = 50/NA
- **ELMS:**
 - Parallel energy density for thermal quench (controlled/uncontrolled) ~ 0.77/3.8 MJ/m²
 - Deposition time ~ 0.4 ms
 - Frequency (controlled/uncontrolled) = 4/1 Hz

Wall Loads on PFC's in ITER

(from M. Roedig's presentation)

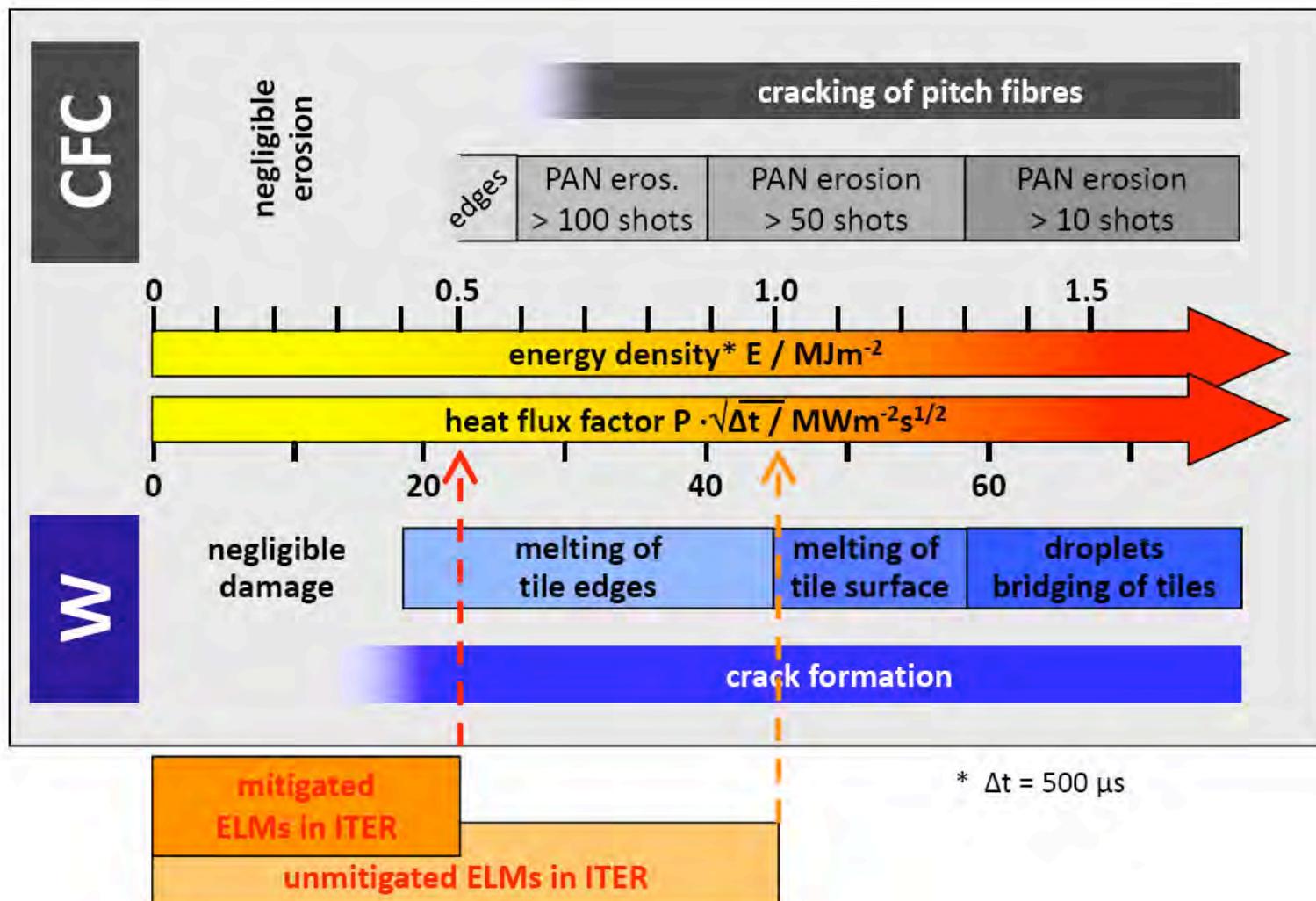


- Large initial armor thickness to help accommodate phase change erosion

ELM Induced Erosion of CFC and W

(from M. Roedig's presentation)

- Tungsten and CFC surfaces experience significant melting/erosion and cracking above 0.5 MJ/m^2
- ELM mitigation/suppression required (*RMP ELM Suppression from DIII-D results*)



OBSERVATION 3B

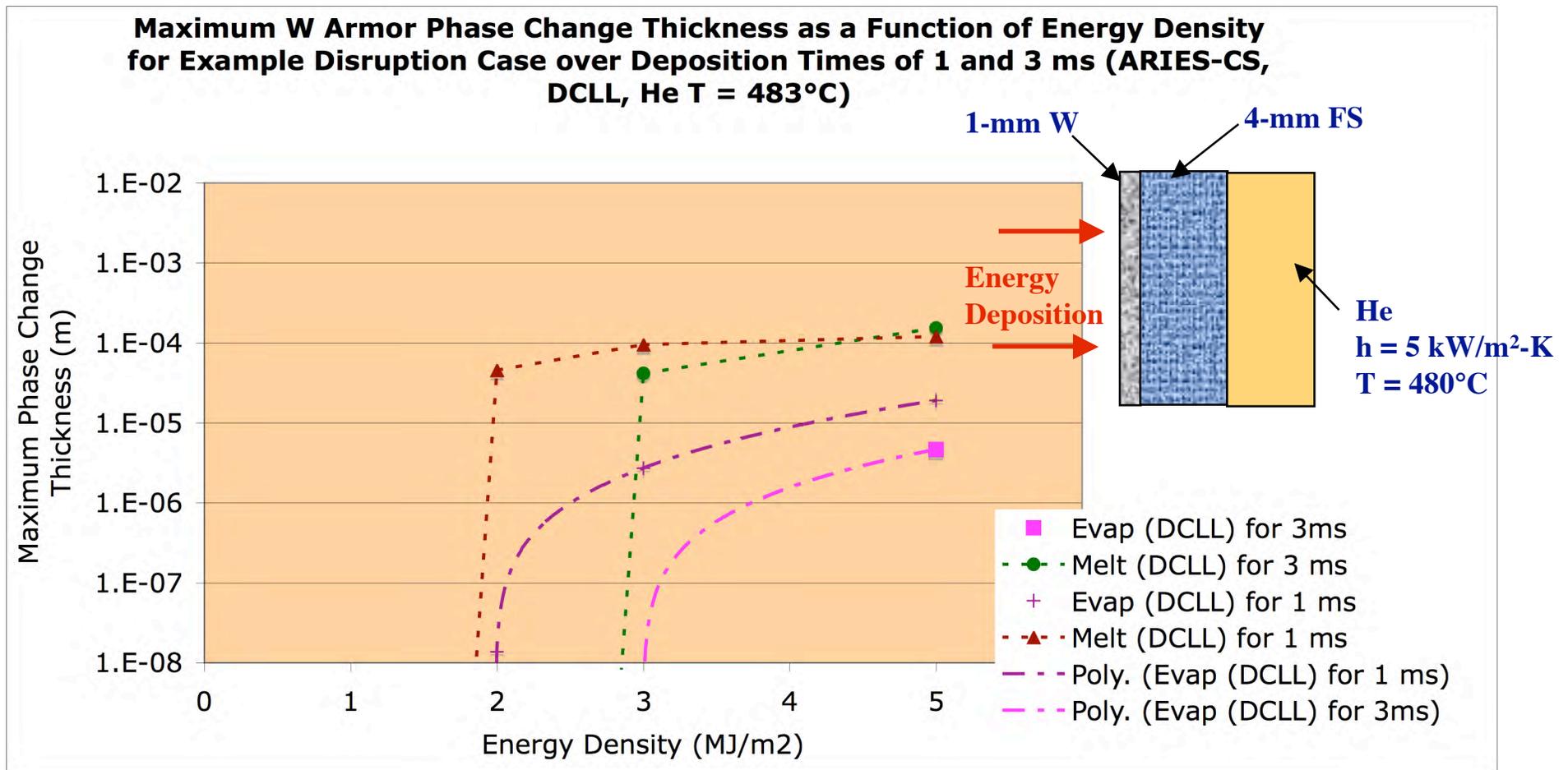
Power plant divertor design load much more limited:

- **Higher operating temperature, higher fluence reduce accommodation margin**
- **Steady state divertor design load ~ 10 MW/m²
(very little margin on this design load)**
- **Very few off-normal events allowed**

Parametric Study of Maximum Phase Change Thickness of a W FW Temperature for Different Disruption Scenarios

(from A. R. Raffray's presentation)

- 1 mm armor (W) on 4-mm FS FW cooled by He at 483°C with $h=5.2 \text{ kW/m}^2\text{-K}$
- Up to $\sim 0.1 \text{ mm}$ melt layer and $\sim 0.01 \text{ mm}$ evaporation loss per event
- Only a few events allowable based on erosion lifetime depending on energy density



Summary of Assessment of Off-Normal Energy Deposition on FW

(from A. R. Raffray's presentation)

- **Focus on thermal effects**
- **EM effects will also be important for FS FW**
- **Only a few disruptions can be accommodated (depending on the energy density)**
- **VDEs cannot be accommodated**
- **Only limited number of uncontrolled ELM cases can be accommodated**
- **Controlled ELMs would drastically limit the lifetime of FS armor (a few days) but might be acceptable for W armor**
- **Avoidance or mitigations of disruptions (and off-normal events) is a key requirement for power plant applications**

OBSERVATION 4A

ITER PMI Conditions

- **3 materials: Be, W and C**
- **Low temperature (~200°C)**

ITER PFM's

(from P. Lorenzetto's presentation)

Armour materials

Compromise: Plasma Performance \leftrightarrow Materials Lifetime \leftrightarrow T retention

~ 680m² Be first wall

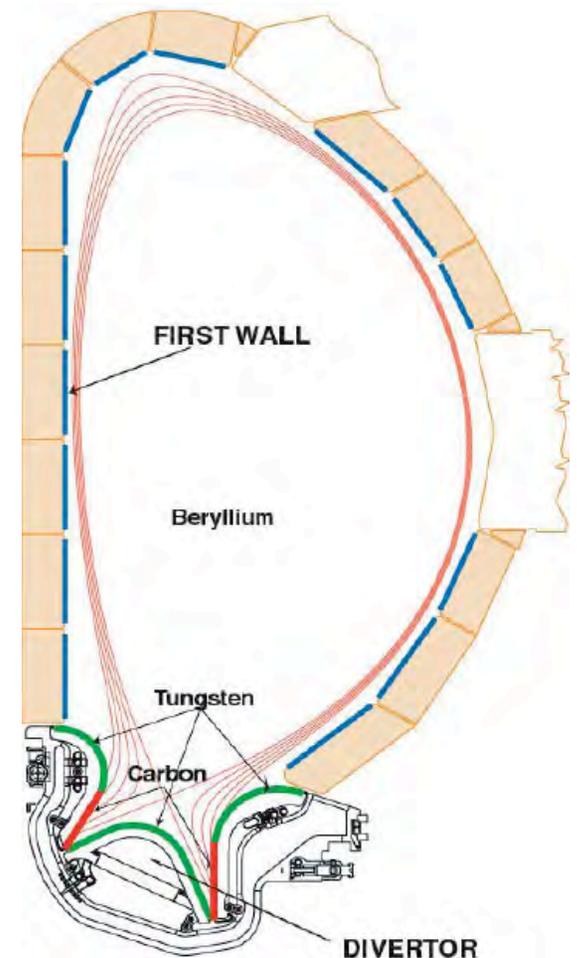
- low Z compatibility with wide operating range and low T retention
- Large experience from JET operation

~ 50 m² CFC Divertor Target (before Tritium phase)

- Good resistance under transients (ELMs and Disruptions)
- Low Z compatibility with wide range of plasma regimes ($T_{e,div} \sim 1 - 100$ eV)
- Large T retention (co-deposition)

~ 100m² Tungsten Baffle/Dome

- Low Erosion, long Lifetime and low T retention
- Less experience



OBSERVATION 4B

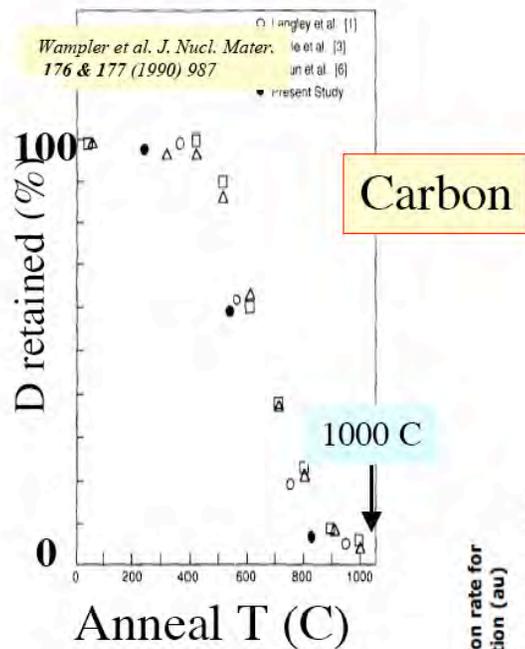
Power Plant (DEMO) Conditions

- **W armor (or ideally bare FS wall)**
- **High temperature (~700°C)**
- **Need fusion testing under these conditions prior to DEMO**

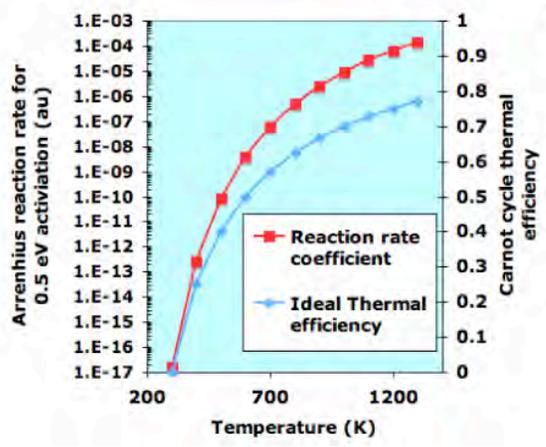
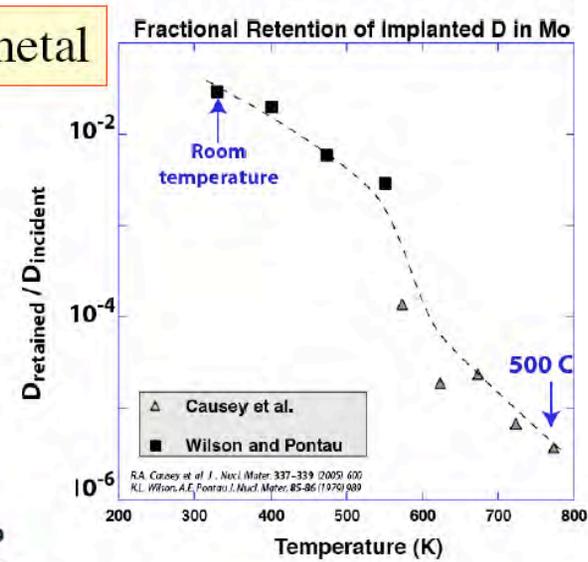
It is Hard to Overstate the Importance of Ambient Temperature for Fuel Control and Tritium Retention

(from D. Whyte's presentation)

- Every major (and minor) modification to the wall surfaces had profound effects on core performance.
 - E.g. lithium layers (TFTR, NSTX), He discharge cleaning (TFTR, DIII-D, etc), boronizations (DIII-D, C-Mod, etc), ad infinitum
- Can we be so naïve that ~10 orders of magnitude modifications to boundary condition of wall will not have profound effects on the core?



High-Z metal



PMI at elevated temperature can influence surface material loss rate

(from R. Doerner's presentation)

- **Material loss rates impact reactor operation**
 - **Impurity content in core**
 - **Lifetime of wall**
 - **Changing thickness will alter thermal gradients in armor which will in turn effect tritium inventory in the armor**
 - **Material mixing**
- **Increases in material loss rates will provide more material available for co-deposition with fuel**
- **Very little PMI data at elevated (DEMO relevant) wall temperature is available**

OBSERVATION 5A

ITER PFC well ahead on development scale

- **TRL 8-9 for specific PFC design**

OBSERVATION 5B

Power Plant (DEMO) PFC at early development stage

- **TRL 2-3 for specific PFC design**
- **Useful to plan for integrated experiment (TRL 6 or above) but need to consider how to proceed through next TRL steps and associated R&D as well**

TRL's Applied to PFC's (from M. Tillack's presentation)

	Issue-Specific Description	Program Elements
1	System studies to define parameters, tradeoffs and requirements on heat & particle flux level, effects on PFC's.	Design studies, basic research
2	PFC concepts including armor and cooling configuration explored. Critical parameters characterized. PMI and edge plasma modeling.	Code development, applied research
3	Data from coupon scale heat and particle flux experiments; modeling of governing parameters and their effects on PFC's as a function of PFC concept. Power plant relevant high-temperature gas-cooled PFC's	Small scale facilities: e.g. Cbam and plasma simulators
4	Bench-scale validation through submodule testing in lab environment simulating heat or particle fluxes at prototypical levels over long times, mockups under representative neutron irradiation level/duration.	Larger-scale facilities for submodule testing, high-temperature + all expected conditions. Neutron irradiation (fission).
5	Integrated module testing of PFC concept in an environment simulating the integration of heat, particle, neutron fluxes at prototypical levels over long times. Coupon irradiation testing of PFC armor and structural material to end-of-life fluence.	Integrated large facility: Prototypical plasma particle + heat flux (e.g. an upgraded DIII-D/JET?) IFMIF?
6	Integrated testing of the PFC concept subsystem in an environment simulating the integration of heat & particle fluxes and neutron irradiation at prototypical levels over long times. Low-temperature water-cooled PFC's for ITER	Integrated large test facility with prototypical plasma particle & heat flux, neutron irradiation.
7	Prototypic PFC system demonstration in a fusion machine.	Fusion machine, e.g. ITER (w/ prototypic divertor), CTF
8	Actual PFC system demonstration and qualification in a fusion energy device over long operating times.	CTF
9	Actual PFC system operation to end-of-life in a fusion reactor with prototypical conditions and all interfacing subsystems.	DEMO (1 st of a kind power plant)

Major Observations from IHHFC Workshop Include:

1. Divertor materials and conditions

- A. ITER: 100°C subcooled water, CuCrZr, W/CFC, austenitic SS
- B. Power plant (Demo): 600-700°C He, W-alloy, ODS FS

2. Level of R&D effort

- A. Extensive R&D for ITER divertor (15 years +): at the edge of procurement
- B. R&D in early stages for power plant divertor material and configuration (must be realistic about time and effort required)

3. Steady state and transient loads

- A. ITER divertor designed for demanding steady-state and off-normal conditions
- B. Power plant q'' on divertor limited to ~ 10 MW/m²; no VDE's; very few disruptions per year.

4. Plasma/Material Interaction Conditions

- A. ITER PMI: 3 materials (Be, C, W), low wall temperature ($\sim 200^\circ\text{C}$)
- B. Power plant PMI: W (bare wall?), high wall temperature ($\sim 700^\circ\text{C}$) (need testing under these conditions)

5. Technology Readiness Level

- A. ITER PFC toward the end of TRL scale
- B. Power plant PFC at early TRL's (providing a guide as to what is needed next)

Please Consult the IHHFC Workshop Website for More Information

<http://aries.ucsd.edu/IHHFC/index.html>