International HHFC Workshop on Readiness to Proceed from Near Term Fusion Systems to Power Plants

Summary

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

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.
- This topic includes of course important physics and material aspects.
- It is also 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

Observations from the Workshop

- 33 Registered Participants (including 8 from EU and 1 from Japan)
- 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

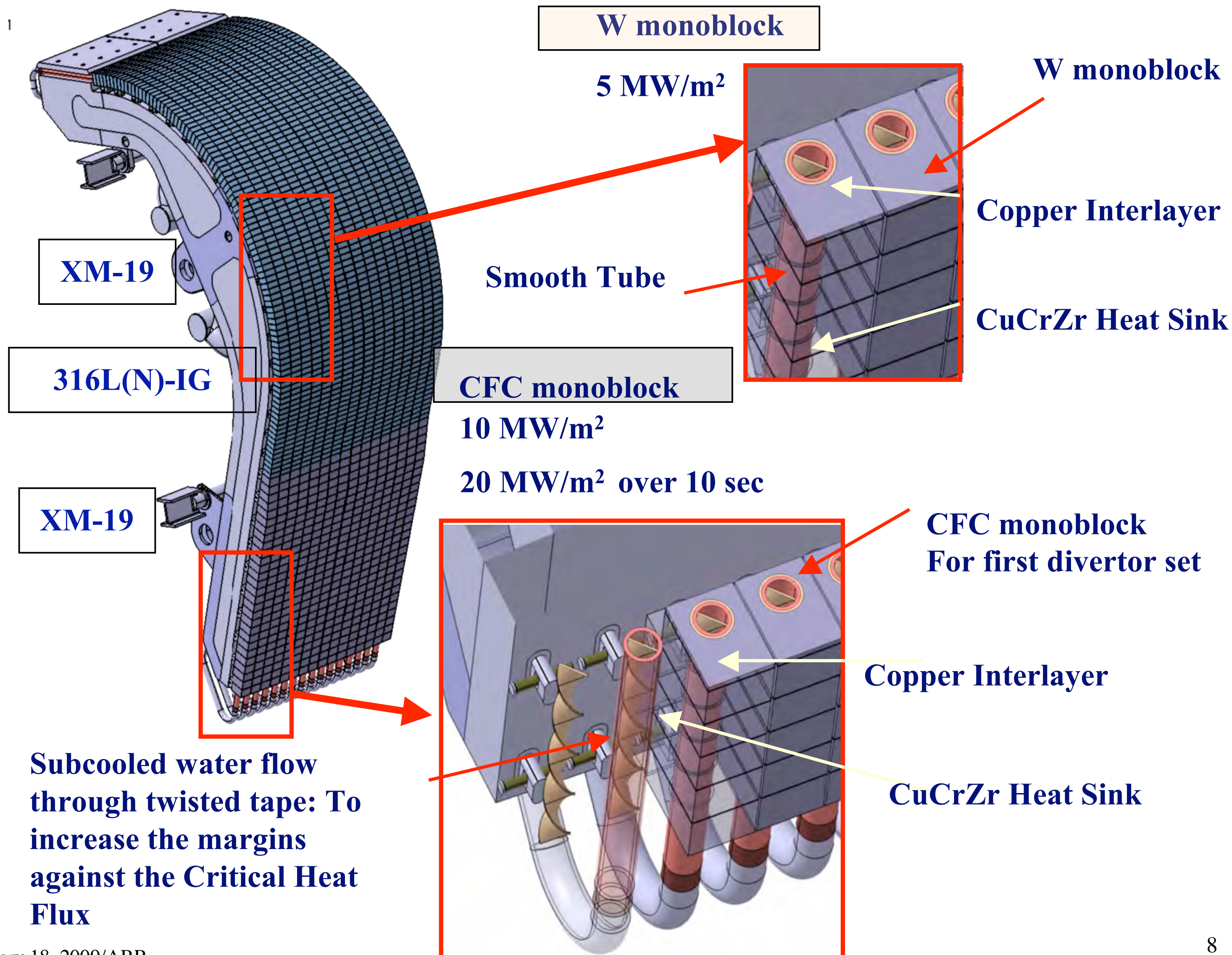
ITER divertor is based non-reactor relevant material, coolant and operating conditions:

- **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 Divertor Tube Configuration

(from M. Merola's presentation)

Plasma-Facing Components **Vertical Targets**



OBSERVATION 1B

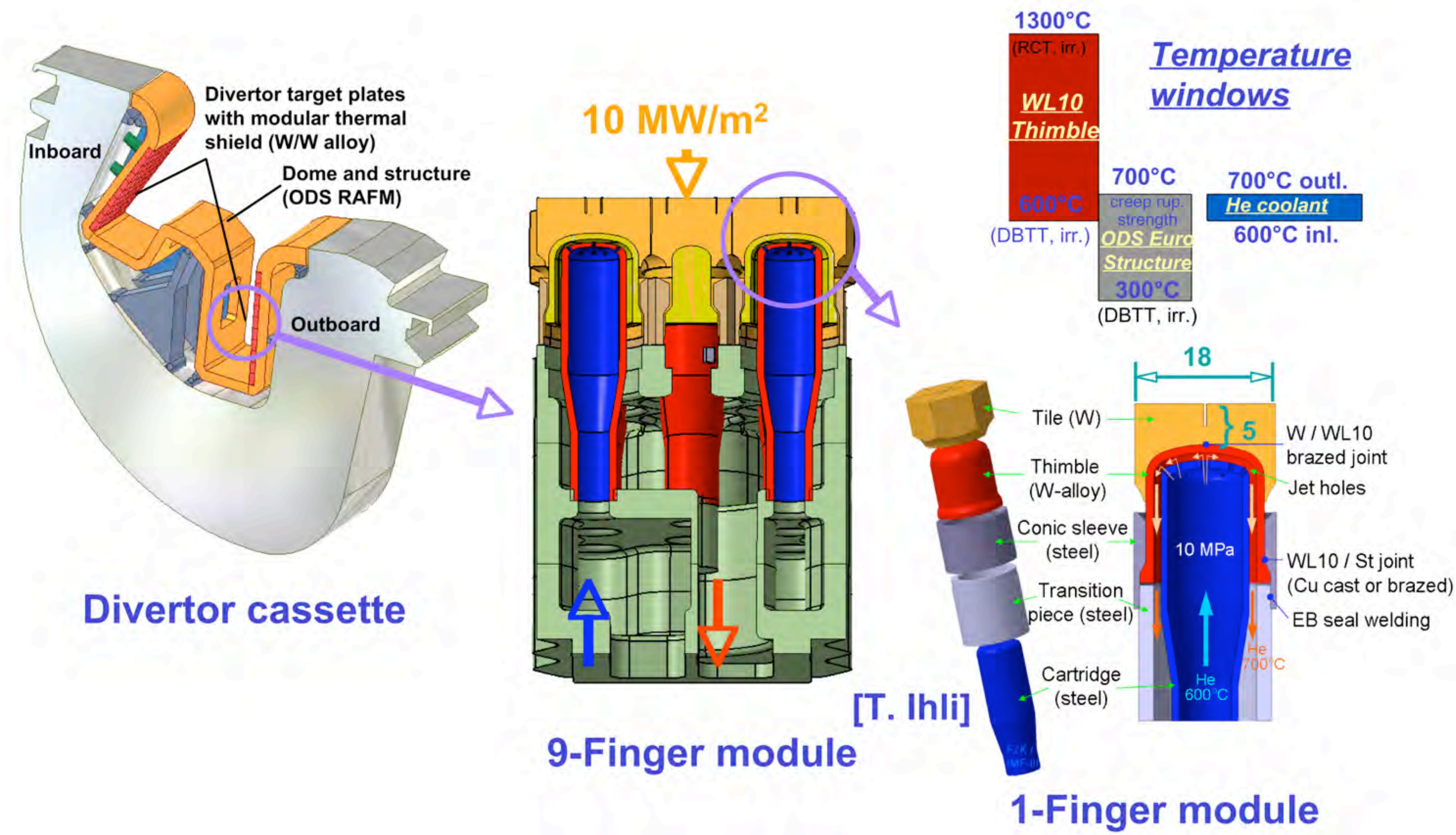
Typical tokamak power-plant (DEMO) divertor (at least in EU and US) based on operating conditions and materials very different from ITER:

- **He-cooled, W-alloy concepts to accommodate ~10 MW/m^2**
  - **He coolant: ~10MPa; ~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**

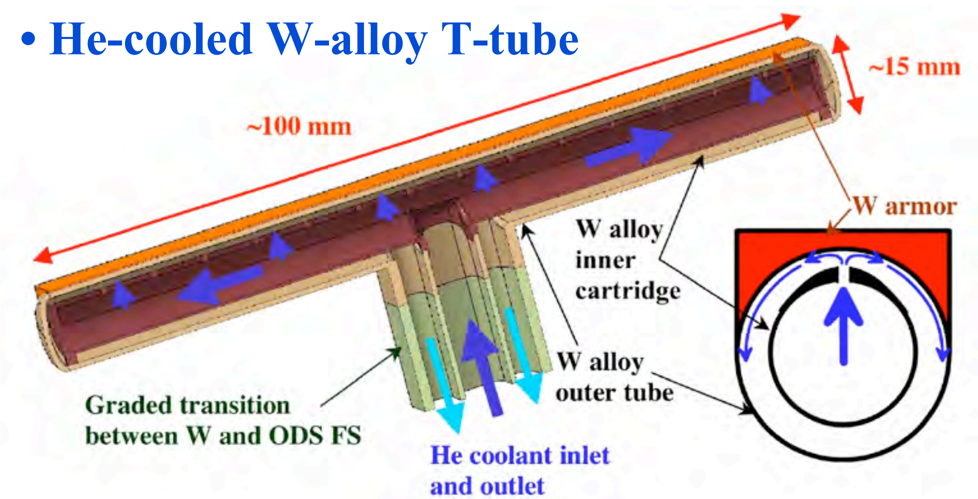
He-cooled modular divertor with jet cooling (HEMJ)

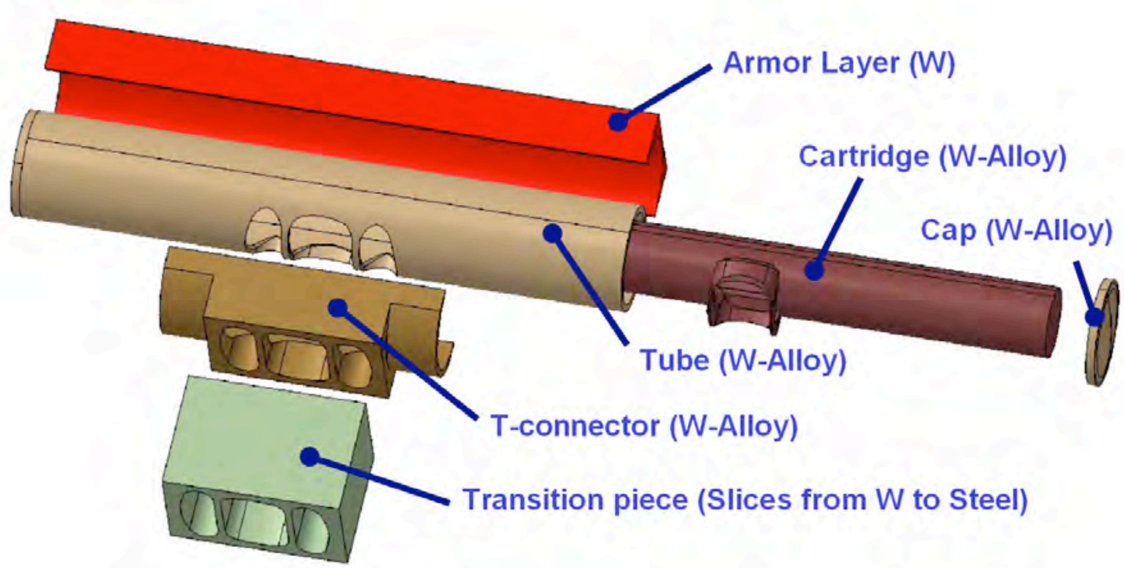
(from P. Norajitra's presentation)



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 (~0.4 mm)
- 10 MPa, ~600-700°C He coolant
- ~600/700°C to1300°C W-alloy
- A number of such T-tubes can be connected to a common manifold to form desired divertor target plate area.

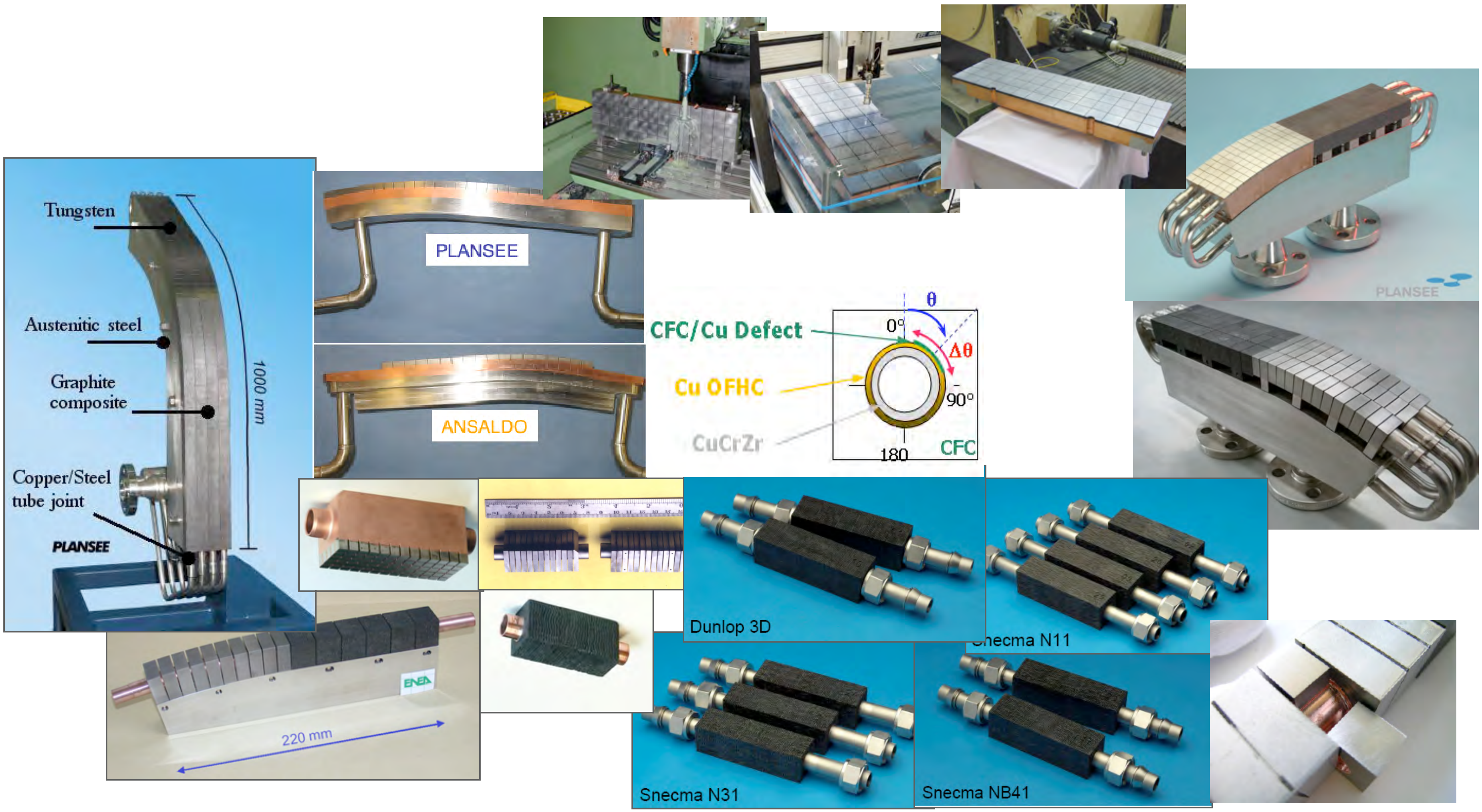
OBSERVATION 2A

ITER divertor just went though 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 +**

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

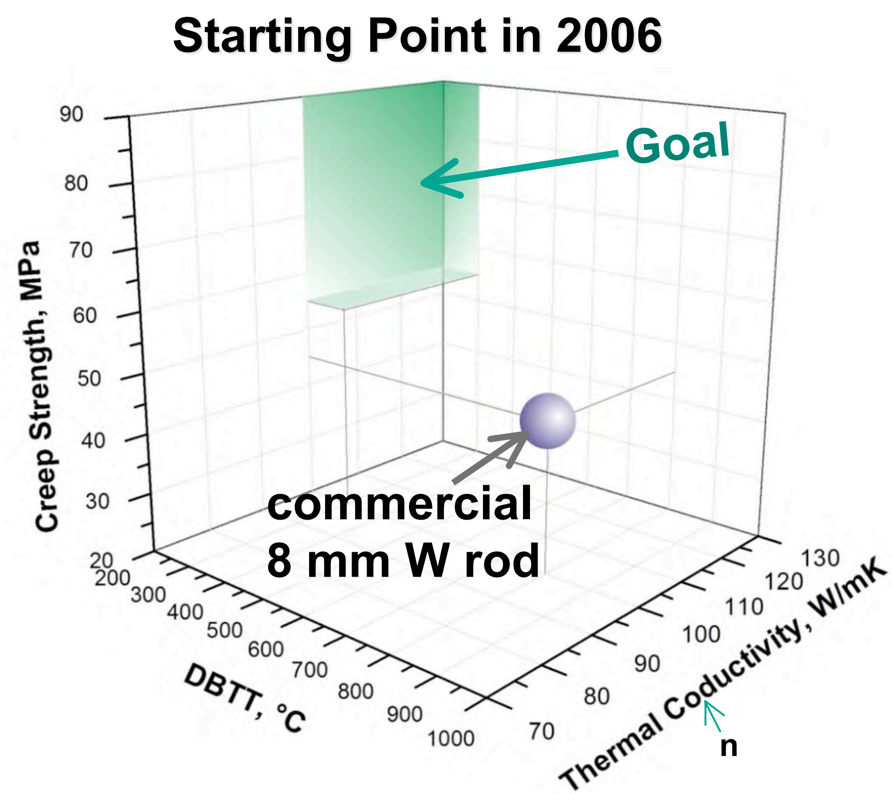
He-cooled W-alloy power plant divertor at the conceptual design stage

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

Important Design Criteria for W

(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

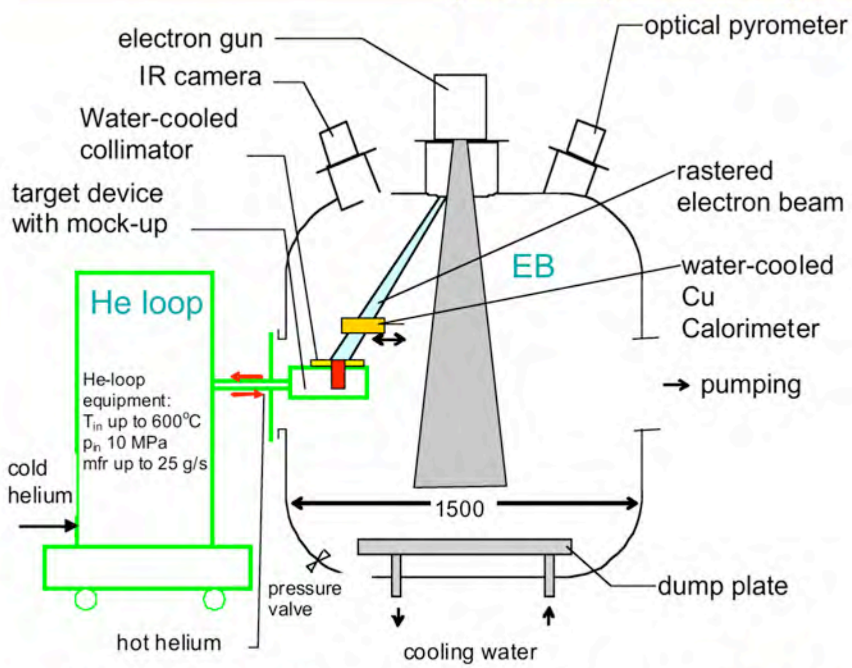


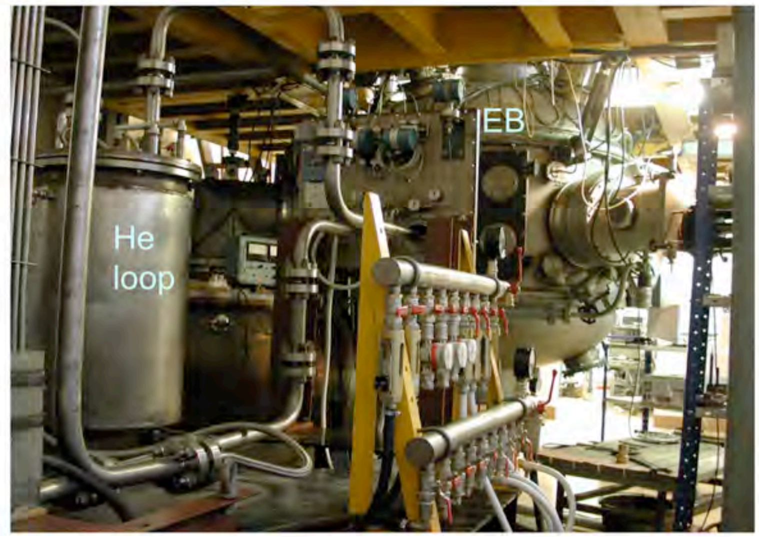
• **Effect of irradiation on embrittlement**

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

(from P. Norajitra's presentation)

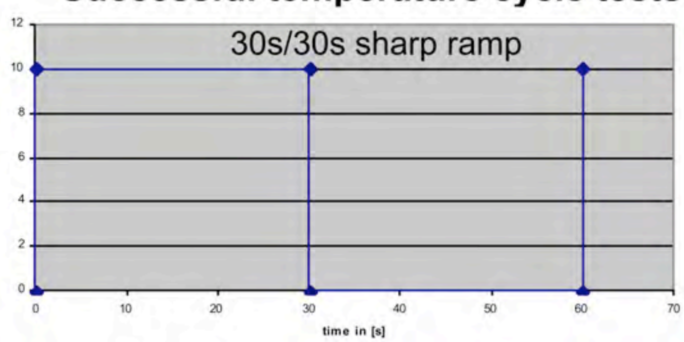
HHF tests in a combined He loop & Tsefey EB facility (60 kW, 27 keV) at EFREMOV

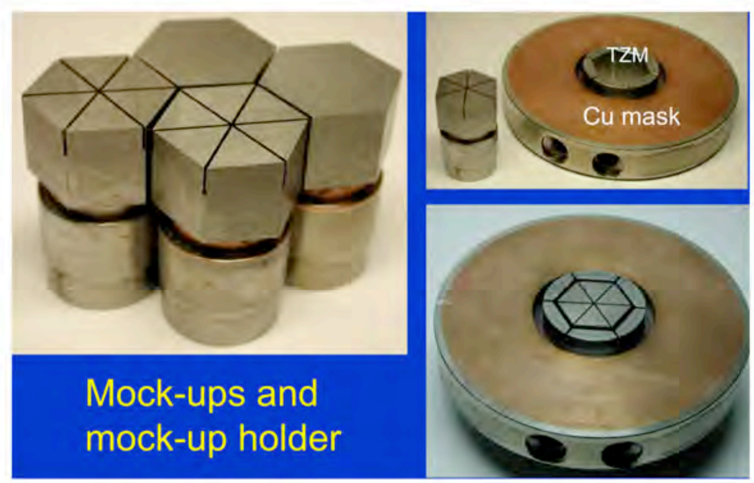




EFREMOV under FZK contract

Successful temperature cycle tests at 10 MW/m²





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/m^2**
  - **Slow transient: 20 MW/m^2 for <10s**

- **Off-normal or transient events include:**
  - **Disruptions**
  - **VDE's**
  - **ELM's**

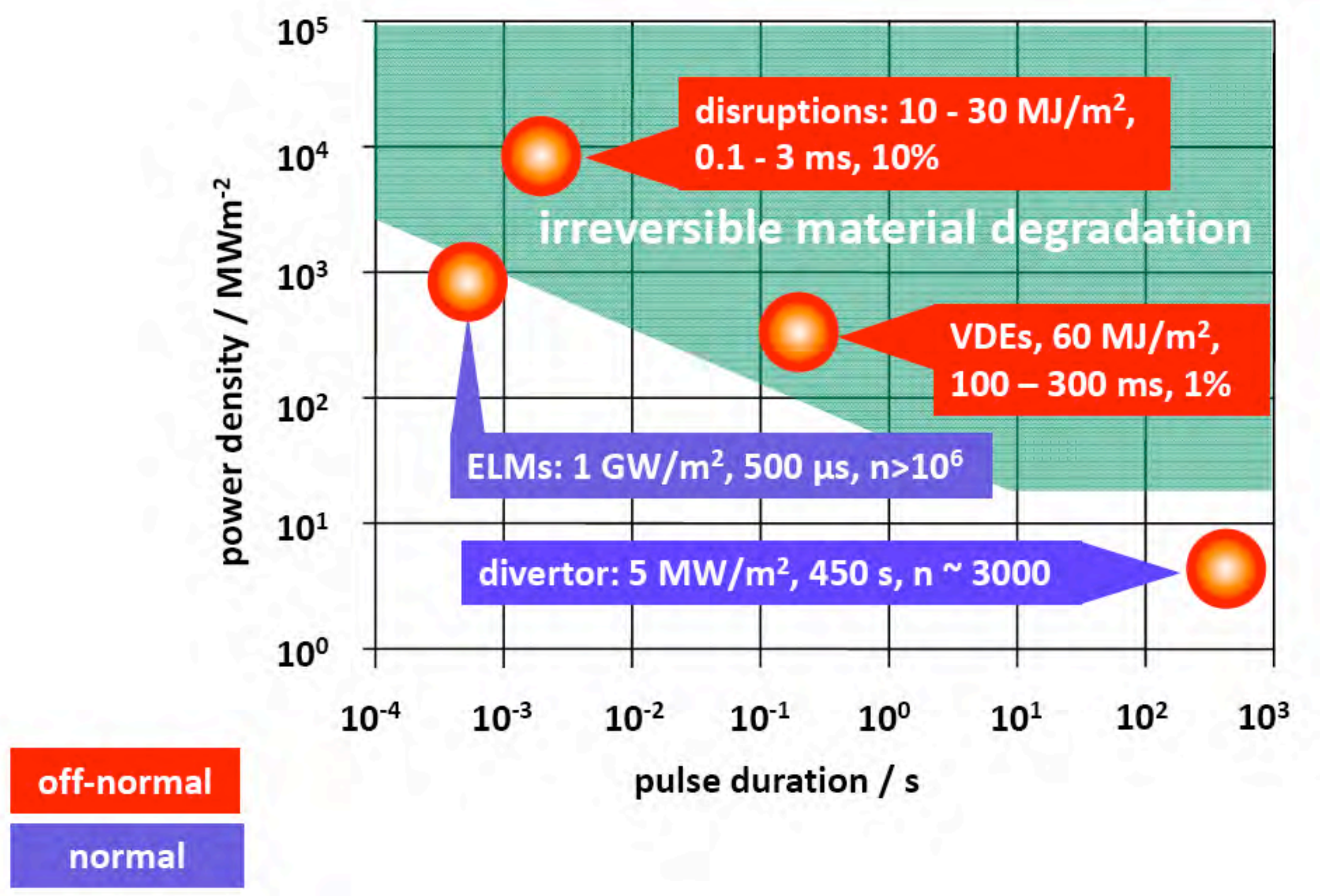
Transient Loads from ITER Project Integration and Load Specification Documents and presentations at 2007 ITER WG8 Design Review Meeting

- **Disruptions**
  - Parallel energy density for thermal quench = 28-45 MJ/m^2 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/m^2
  - No. of Type I/Type II disruptions = 1000/100

- **VDE's (Type I/Type II):**
  - Energy deposition = 30/60 MJ/m^2
  - 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/m^2
  - 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**

OBSERVATION 3B

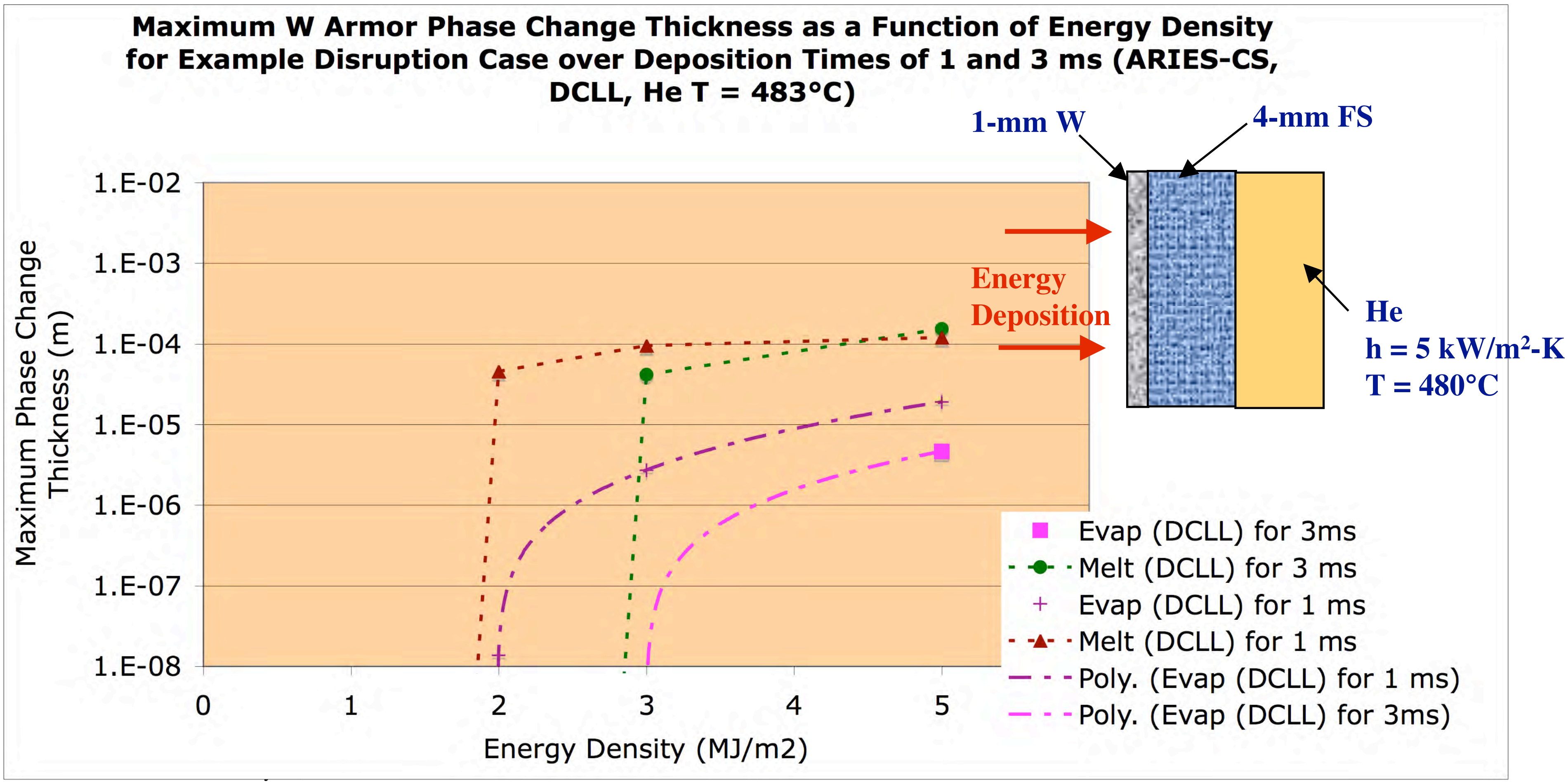
Power plant divertor design load much more limited:

- **Higher operating temperature, higher fluence reduce accommodation margin**
- **Steady state divertor design load ~ 10 MW/m^2 (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 kW/m^2-K
- Up to ~0.1 mm melt layer and ~0.01 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 ←→ Materials Lifetime ←→ 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)**

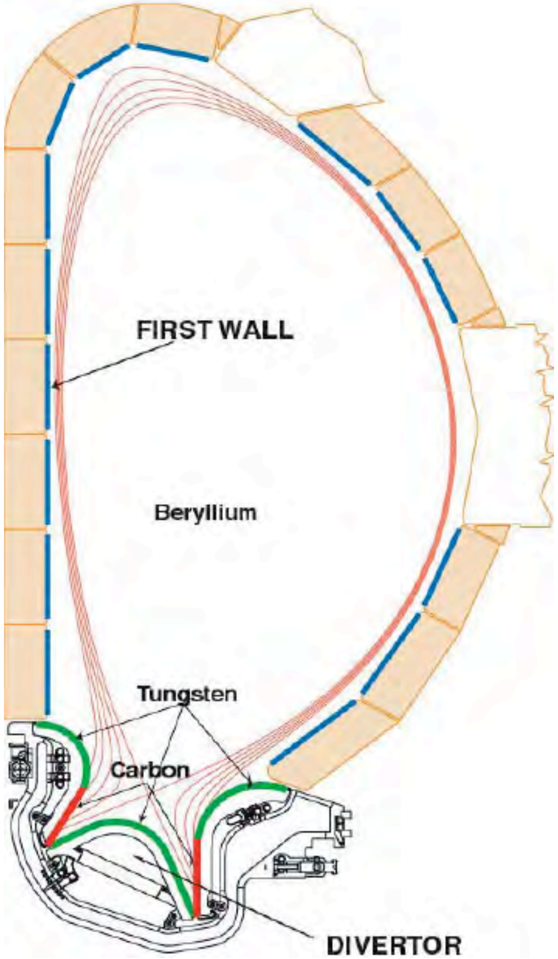
→ **Low Z compatibility with wide range of plasma regimes ($T_{e,div}$ ~ 1 – 100 eV)**

→ **Large T retention (co-deposition)**

~ 100m² Tungsten Baffle/Dome

→ **Low Erosion, long Lifetime and low T retention**

→ **Less experience**



OBSERVATION 4B

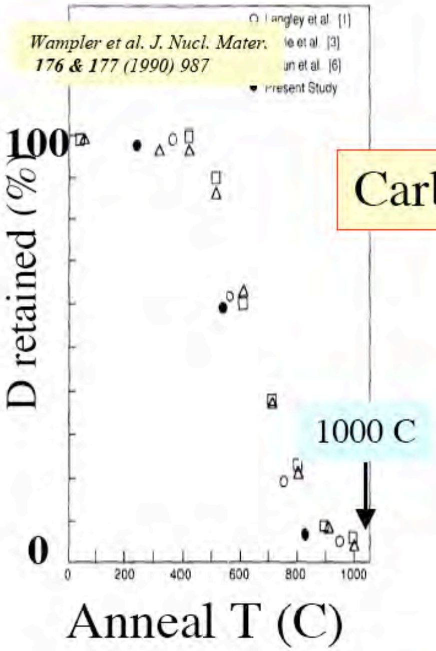
Power Plant (DEMO) Conditions

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

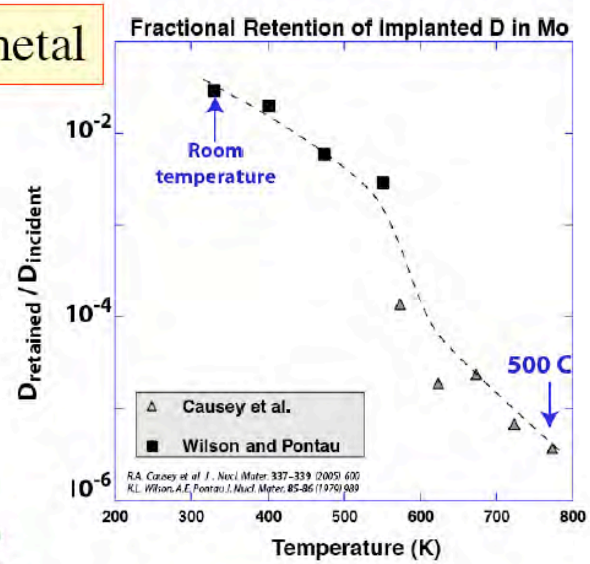
It is Hard to Overstate the Importance of Ambient Temperature for Fuel Control and Tritium Retention

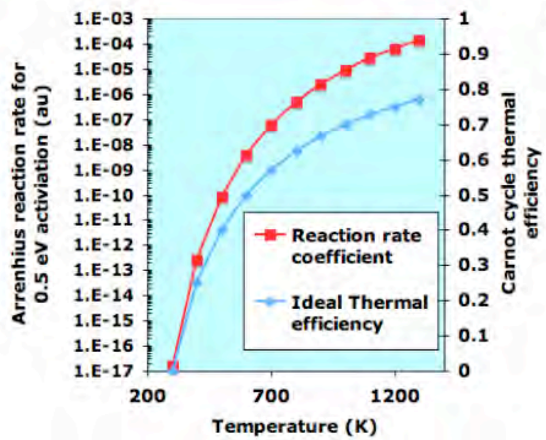
(from D. Whyte's presentation)

- **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





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 though 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 heat and mass transfer processes as demonstration of function of PFC concept.	Small-scale facilities: *e.g.,* e-beam 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.	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 (1st of a kind power plant)

Power plant relevant high-temperature gas-cooled PFC's

Low-temperature water-cooled PFC's

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 (~200°C)
   - B. Power plant PMI: W (bare wall?), high wall temperature (~700°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