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Plasma Facing Components: Challenges for Nuclear Materials

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Plasma Facing Components: Challenges For Nuclear Materials

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In current fusion devices, the components located in front of plasma, the so-called plasma facing components (PFCs), sustain severe constraints such as high thermal flux (several MW/m²), erosion, flux of particles. The management of this first material interface is critical from a plasma performance point of view. ITER, as nuclear facility, is initiating a new era for fusion, which will be reinforced for a future fusion power plant which will add specific requirements (sufficient lifetime, a cooling system to produce energy, use of low activation material) while increasing nuclear constraints.

The talk will recall in a first part the main requirements of an actively plasma facing components and the main results obtained with low-Z carbon based PFCs (mainly CFC). Experimental feedback from these challenging components is an essential step for the success of the next generation of components, in particular in term of manufacturing or handling intense heat loads.

Nuclear safety requirements mainly drive the need of new materials for the nuclear phase of ITER. The tritium retention in carbon based PFCs and the strong erosion are expected to be too high in the Deuterium-Tritium phase with CFC targets, justifying the use of high-Z materials. The evolution toward high-Z materials, with tungsten the most promising, becomes a major challenge for fusion research. Large scale experiences with W have only been obtained recently with the operation of ASDEX-upgrade and JET tokamaks but with non-actively cooled PFCs. ASDEX-upgraded is equipped with W coated carbon PFCs while JET includes W coated carbon PFC and inertially cooled solid W, using in all cases a technology not relevant for ITER. Extensive R&D programmes have been performed in Europe to develop reliable actively PFCs for ITER [1-5]. The state of the art will be presented including specific devices needed to fully qualify, at laboratory scale, designs foreseen for ITER. In order to reduce the risks and anticipate any difficulties ITER may face in terms of manufacturing or operation, it is proposed to update Tore Supra with a full W first wall and divertor, benefitting from the unique long pulse capabilities of the Tore Supra platform, the high installed power and the long history of operation with actively cooled high heat flux components [6]. The main goals of the 'WEST' project (W - for tungsten -Environment in Steady-state Tokamak, figure 1) will be presented.

The talk will also address acknowledged gaps in PFCs developments for DEMO, which require extensive studies in different topics, from plasma-surface interaction to engineering including material sciences. Further challenges address simultaneously the higher power density, high-temperature wall, bulk (neutron) and surface (charged particle) accumulated damage. The high neutron fluence expected in a fusion reactor (more than 10 dpa/year) will affect erosion and tritium retention properties of materials. Near-surface material properties will be for instance altered by the neutron damage. Such synergistic effects are expected to be important in the DEMO environment and are difficult to be addressed experimentally. A more robust coupling of materials development, including fundamentally studies, with advanced design is required. If tungsten is the most promising material for the plasma-facing, tungsten also offers less favorable properties (recrystallization, which influences the mechanical properties, embrittlement as a result of neutron-induced damages, He-induced sputtering...) that have to be resolved. Only a global approach, including fundamental science, material development, joining/welding techniques, design innovation and a close link with plasma physics is able to reach the necessary level of credibility for operating such components in a fusion environment in an economically reasonable way.

Fig. 1: The ‘WEST’ project is a major update of Tore Supra to address the issues related to tokamak operation with W actively cooled ITER relevant PFCs. Internal components, presently in CFC will be updated to W components (ITER technology) and the magnetic plasma configuration (presently circular) will be changed to X-point configuration by installing a set of poloidal coils inside the lower and upper parts of the vacuum vessel.

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MINOS workshop (Saclay, 5-7 december 2012)

Plasma facing components: challenges for nuclear materials

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Fusion basics

From present Plasma Facing Components (PFC)...

.. to PFC for ITER

Toward a fusion reactor

On SUN & STARS

Confinement = GRAVITY
high T° & pressure for long time

On SUN & STARS

Confinement = GRAVITY
high T° & pressure for long time

On EARTH

1 - Confine the plasma into a *magnetic bottle* → MAGNETIC fusion

- Characteristic time (time for plasma to exhaust power=confinement time): seconde
- Plasma volume $\sim 1000 m^3$ with only few g of fuel (low density)

JET Tokamak
(Joint European
Torus)

On SUN & STARS

Confinement = GRAVITY
high T° & pressure for long time

On EARTH

- 1 - Confine the plasma into a *magnetic bottle* → MAGNETIC fusion
- 2 - Use fusion reaction with the highest cross section

Plasma does not like materials with high atomic number !!

The fatal fraction of an impurity is the concentration at which the radiated power (impurity line radiation and Bremsstrahlung radiation) exceeds the alpha heating power

$$C \sim 10^{-1}$$

$$W \sim 10^{-4}$$

→ Close links between the surface material and the plasma core

Components in interaction with the plasma = Plasma Facing Components (PFC)

Components in interaction with the plasma = Plasma Facing Components (PFC)

PFC = different designs with a set of different materials

[see H. Burlet's talk]

Confinement is never perfect → management of the deconfined particles leaving the plasma core on a specific component: **divertor** → heat and particles removal

Divertor in JET : without and with plasma

Advantage : Concentrate power and particle exhaust in one location (far from the plasma core)

Disadvantage : Concentrate power and particle exhaust in one location

Confinement is never perfect → management of the deconfined particles leaving the plasma core on a specific component: **divertor** → heat and particles removal

Divertor in JET : without and with plasma

Typical heat flux value:

- steady state → 10 MW/m²
- transient (<ms) → 1-10 GW/m² (ELMs, misalignment of components...)

$$\Phi = \Phi_0 e^{\frac{\delta}{\lambda_q} \sin \alpha}$$

- High heat flux
- **Erosion** in the main plasma interaction area
- **Production of dusts, deposits** in deposition / codeposition area

*Toroidal Limiter
(Tore Supra)*

*Tore Supra : neutraliser finger
Hot deposit (500°C)
Fuel trapping D/C = 1%*

*Tore Supra : outer limiter
Cold deposit (120°C) D/C = 10%*

Dusts/deposits/layers have operational consequences →
Emission of deposits during the plasma startup → to fast plasma termination

PRESENT TECHNOLOGIES

tritium: ~ 100 kg /y
neutrons: 5-20 dpa/y

- Fusion basics

- From present Plasma Facing Components (PFC)...

- ... to PFC for ITER

- Toward DEMO

For fusion guys, current
world is not a nuclear
world 😊

- Fusion basics
- From present Plasma Facing Components (PFC)...
- ... to PFC for ITER
- Toward DEMO

- CFC has a good plasma compatibility (low atomic number),
- a good thermal conductivity : 50-300 W/m.K,
- and no melting, highest sublimation point : 3 900 K
→ C “forgives” occasional overheating

Due to low plasma fluence, drawbacks (erosion) are not an issue for present devices

$\sim 2\text{-}3 \text{ eV} < T_{\text{plasma edge}} < 100 \text{ eV} \rightarrow$
 $15 \text{ eV} < E_{\text{ions}} < 500 \text{ eV}$

Threshold for D

Be 3 eV
 C 8 eV
 W 80 eV

- Copper as structural material due to its high thermal conductivity

A world of CFC and Copper (ex Tore Supra technology)

- Joining is a weak link
- From lab-scale to Industry: never a quiet workflow
- **Carbon retains large amounts of H (D, T) →** the integrated amount of deuterium recovered at the end of a plasma discharge is generally lower than the integrated injected amount

DITS project (Tore Supra, 2007-2010)

- 5h of plasma
- dismantling of part of Tore Supra components

- Post mortem analysis (retention by co-deposition due to erosion dominates)

In carbon → AUG-JET (3-4%), JT-60U~8%, TEXTOR & TS ~10-15%

- Fusion basics

- From present Plasma Facing Components (PFC)...

- ... to PFC for ITER

- Toward DEMO

■ Tritium inventory is limited by Safety Authorities

ITER: 700g inside the vacuum vessel

A full C wall reaches tritium limit (700g) in less than 30-40 ITER nominal plasma discharges

A full W wall reduces tritium problem, but n-effects need to be considered (R&D on-going, modelling...)

CFC is not relevant for the ITER nuclear phase

reminder:

ITER phase 1 : H-H plasma (non nuclear)

ITER phase 2 : D-D plasma (nuclear)

ITER phase 3 : D-T plasma (full nuclear)

A reference design for optimizing plasma performance...and cost

Monoblock concept

Water cooled
(100°C, 3 Mpa)

Divertor cost > 100 M€ (54 cassettes), manufacturing: 6-8 years, installation & commissioning: 1 year

Summary of thermal fatigue testing under High Heat Flux *

* M. Missirlian, PFMC2011
M. Richou (SOFT-2010)
P. Gavila (SOFT-2010)

ITER requirements achieved at lab-scale (with low margin)

- industrial fabrication of large amount of components?
- plasma operation with W-wall ?

To prepare ITER operation, an Iter-like wall was installed in JET

L. Horton (SOFT-2012)

- Demonstrate sufficiently low fuel retention
- Develop integrated ITER compatible scenarios for an all-metal machine
- Effect of transients (ELMs and disruptions) on ILW
- Develop control strategies for detecting and limiting damage...

Reminder : JET components are NOT actively cooled (inertial components)

2880 installable items, 15828 tiles (~2 tones Be, ~2 tones W)

■ 2012 experimental campaign (main results)

- Qualification of new systems including protection wall system
- Beryllium and tungsten **erosion** are **consistent with physical sputtering**
- Measured **fuel retention is more than an order of magnitude lower** with the ILW, consistent with predictions made before the wall was installed
- The accumulation of tungsten into the core plasma **limits the operating space** with the ILW.
Confinement is typically 20% lower than with the carbon wall
Understanding and improving these initial results is a priority for future experiments

■ 2013 experimental campaign → Assessment of ITER operating scenarios

■ **Physics:** operation of the tokamak with a tungsten wall (How to use a plasma in a tungsten environment)

→ ✓ **plasma**, **x components**

■ **Technology:** test under thermal flux, fatigue, etc. of the components

→ **x plasma**, ✓ **components**

WEST : First integral test

Component technology at high flux + long pulse tokamak environment

→ ✓ **plasma**, ✓ **components**

WEST (W Experiment in Steady state Tokamak) → implementation in Tore Supra of an divertor using the ITER technologie and a full W wall

Present configuration

Circular plasma and limiter configuration

WEST configuration

X point plasma and ITER divertor technology

First plasma in WEST configuration in 2015 ?

**'fish' shaping
(h=0.3mm)**

- Fusion basics

- From present Plasma Facing Components (PFC)...

- ... to PFC for ITER

- Toward the reactor

Different approaches (ARIES studies in US, PPCS in EU, ...) and presently no consensus and no clear roadmap

- **attractiveness** (small is beautiful) → need large extrapolations both in physics & technologies
- **credibility** → low extrapolations from ITER

Example of 3 gaps to fill (among a lot) to reach the right level of credibility

PPCS(EU)

PPCS(EU)

→ Neutron fluence will increase

■ Yearly neutron damage in plasma-facing materials (dpa) : 0,5 (ITER) → 5-20 dpa (reactor)

→ material properties will involve under 14MeV fusion neutron: thermal conductivity, swelling, traps for tritium both on *structural materials* and **PFC**

↳ see J. Henry's talk

Defect production in **W** and alpha-particle impingement will create a **new situation** (+ higher T_{wall}) with possible new radiation induced microstructure in subsurface, which will determine the in-service behavior of W as armour material.

→ **New topic which need collaborative work between materials and plasma communities**

→ **Neutron fluence will increase** → new materials (R&D on going)

→ **Higher coolant temperature for energy production**

■ ITER : water (100°C, 3MPa) → DEMO : **water** (in: 325 °C, out: 350°C)

Recent results of water-cooled divertor concepts based on ITER W monoblock and steels (Eurofer) for DEMO application, using nuclear design rules : **HHF is limited to 10 MW/m²**

[A. LiPuma (CEA), SOFT2012]

→ **Neutron fluence will increase** → new materials (R&D on going)

→ **Higher coolant temperature for energy production**

■ ITER : water (100°C, 3MPa) → DEMO : water (in: 325 °C, out: 350°C)
He (600°C, 10 MPa)

He cooled divertor, KIT concept
[A. Norajitra, JNM 2011]

Tested à 10 MW/m²

Main concerns: W as structural material,
number of small components (350 000)
and joining → reliability is questionable

→ **Design drives important R&D on materials**

- Functional gradient material [E. Autissier, CEA, SOFT2012]
- Powder Injection Molding [S. Antusch, KIT, FED 2011]
- ...

→ Heat flux on the divertor will increase

■ P_{fusion} : 500 MW (ITER) → 2500-5000 MW

P_{exhaust} : 150 MW (ITER) → 500-1000 MW

→ x 3 to 7

■ Size (major radius) : 6.2 m (ITER) → 5.5 m (ARIES) to 10 m (Early DEMO)

→ $S_{\text{divertor}} \times 2$

If 10 MW/m² in steady state is a limit for the technology

1) New plasma configuration (ex: **detached plasma**, radiation mantle..)

[Tore Supra team]

1 – plasma on the limiter

2 – start of detachment (and control of the instabilities)

3 – detachment (T_{limiter} decreases)

Radiation not only localized on the divertor (to be tested and validated on ITER)

→ Heat flux on the divertor will increase

■ P_{fusion} : 500 MW (ITER) → 2500-5000 MW

P_{exhaust} : 150 MW (ITER) → 500-1000 MW

→ x 3 to 7

■ Size (major radius) : 6.2 m (ITER) → 5.5 m (ARIES) to 10 m (Early DEMO)

→ $S_{\text{divertor}} \times 2$

If 10 MW/m² in steady state is a limit for the technology

1) New plasma configuration (ex: detached plasma, radiation mantle..)

2) **New divertor configuration** to increase S_{divertor}

**Conventional
Divertor**

Super X : Extend leg on low
field side

Snowflake

and the
field region
and X-point

**Concepts need full validation on small
devices**

- **Plasma facing components are already a challenge but filling the gaps between present start of the art and ITER is on going**
 - to use W in fusion environment
 - to mitigate risks in view of ITER operation (JET, WEST...)
- **Filling the gaps between ITER and a fusion reactor will require a large amount of R&D**
 - assess materials (W, structural material) in the right physical chemistry range taken into account nuclear constraints (14 MeV neutrons, tritium)
 - reach an equilibrium between PFC materials and plasma
- **Plasma facing components are NOT only a material or a plasma physics or a technology issue but **an integrated problem****
 - grand scientific challenge involving both material sciences, plasma physics and innovative design
 - Good driver for material sciences !!!

Thank you for your attention

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