Materials Research in the European Fusion Roadmap

Tony Donné, Sebastijan Brezinsek, Michael Rieth, Marek Rubel

ICFRM-17 Aachen, Germany
EUROfusion coordinates R&D in fusion research

29 Research Units (+ numerous Third Parties) in 27 European countries working together to achieve the ultimate goal of the Fusion Roadmap
The Fusion Roadmap

ITER: the key facility for fusion

Risk mitigation for ITER by doing supporting research in contemporary devices

- JET, Medium Size Tokamaks, JT-60SA
- Plasma Facing Component test devices

Stellarator as a long term alternative to tokamaks

- Bring stellarator line to maturity

Power Plant Physics & Technology

Preconceptual design of a DEMO fusion power plant

DEMO as a single step to commercial fusion power plants that produce electricity and have a closed fuel cycle

Eight missions

**ITER Physics**

1. Plasma Regimes of Operation
2. Heat-Exhaust Systems
3. Neutron Resistant Materials
4. Tritium Self-Sufficiency
5. Implementation of Intrinsic Safety Features of Fusion
6. Integrated DEMO design and system development
7. Competitive cost of Electricity
8. Stellarator

**ICFRM**

**Power Plant Physics & Technology**
Mission 1: Plasma regimes of operation

- Demonstrate and qualify regimes that meet the needs of ITER and DEMO
- High fusion performance with metallic PFCs by improving transport and by controlling MHD instabilities.
- Acceptable power depositions in the W divertor, radiate as much as possible power while keeping high performance
- Develop integrated scenarios with controllers (MHD, detached divertor, dilution…)
- Try to achieve steady state conditions

Preparation on existing devices: JET, MST-devices, JT-60SA + other international collaborations
JET and Medium-Size Tokamaks

JET

ASDEX Upgrade

JT-60SA
(Start in 2019)

ITER

(Our target device)

MAST Upgrade
(Start in 2016)

TCV
(Restart in 2015)
Why not Continue to Operate with a Carbon First Wall?

DT experiment in JET revealed unacceptable safety conditions for ITER or a reactor:
- tritium retention of 20% due to co-deposition in divertor [P. Andrew. et al JNM 1999]
- multistep transport of carbon to inaccessible/remote areas of the divertor => dust

Impossible to breed enough T in a reactor with C walls to compensate for loss in co-deposits!
EU Tokamak operation with a metallic wall

- **ASDEX Upgrade:**
  - conversion to all W PFCs complete Gradually over 7 years
  - in 2014 Massive outer W-divertor and Bare Steel Tiles and new divertor manipulator allowing large area sample insertion

- **JET:**
  - ITER-like Wall Be wall and W divertor change in one shutdown
  - Integrated test with DT scenario compatibility in 2018-19

- **Tore Supra** → **WEST project (2016):**
  - from limiter to divertor configuration, from carbon to W environment,
  - Access to long pulse operation with actively cooled W-monorblocks components
Mission 2: Heat Exhaust Systems

The baseline strategy ‘detached’ divertor together with research in alternative divertor solutions: Super-X, snowflake, liquid metal divertors

- Detachment control for ITER and DEMO
- Efficient PFC operation for ITER and DEMO
- Predictive models for ITER and DEMO divertor/SOL
- Investigate alternative power exhaust solutions
- Research to find more robust materials

Main existing: JET, MST, PFC test devices, JT-60SA + other international collaborations
Potentially a Divertor Test Tokamak?

[Courtesy D. J Campbell, ITER research plan]
Plasma Facilities (Steady-State and Transients)

- (+ e-beam, ion beam facilities)

**Magnum-PSI**
(Restart in 2016 with SC magnet)

**PSI-2**
(operational)

**MAGNUM + PSI 2 days for PFC: ~125 in 2016 (tbc)**
Pilot-PSI not available

**WEST**
(Experiments in 2016+)

**JULE-PSI from 2017**
(Be and T compatible)

**PISCES-B for Be/He, Be/D and Be/N exposure**
(inter. Coll. / EU scientist)
Missions 3-7: DEMO

- Materials
- Early neutron source & design (IFMIF vs. DONES/ENS)

- Breeding blankets
- Safety & environment

- Design Integration and Physics Integration
  - Magnet system
  - Divertor
  - Tritium and fuelling
  - Heating and current drive
  - Diagnostics and control
  - Remote maintenance systems
  - Containment structures
  - Heat transfer, balance-of-plant
Mission 8: Bring the stellarator line to maturity

- Bring stellarator to maturity as a possible long-term alternative to tokamaks, EU programme focuses on the Helical Axis Advanced Stellarator, HELIAS, line

- For 2014-2020: main priority scientific exploitation of the W7-X including theory development & modelling

- Impact on the progress of the basic understanding of plasma physics in support of Mission 1 and 2 and in support of the ITER preparation
Materials - WPMAT

Michael Rieth, S. Dudarev, J. Henry, G. Pintsuk, M. Porton, R. Vila, E. Diegele, F. Groeschel
### Budget (w/o overheads)

#### Hardware (EC/k€)

<table>
<thead>
<tr>
<th>RU</th>
<th>Total Allocated 2014-18</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>2300</td>
</tr>
<tr>
<td>2</td>
<td>2400</td>
</tr>
<tr>
<td>3</td>
<td>2500</td>
</tr>
<tr>
<td>4</td>
<td>2600</td>
</tr>
<tr>
<td>5</td>
<td>2700</td>
</tr>
</tbody>
</table>

#### Manpower (EC/k€)

<table>
<thead>
<tr>
<th>RU</th>
<th>Total Allocated 2014-18</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>273 lab ppy</td>
</tr>
<tr>
<td>2</td>
<td>30 ind ppy</td>
</tr>
<tr>
<td>3</td>
<td>8.74 M€ hw</td>
</tr>
</tbody>
</table>

#### Total Allocated Resources by RU

- **273 lab ppy**
- **30 ind ppy**
- **8.74 M€ hw**

**Note:**
- **EÚAW** 0,3%
- **ÖAW** 0,3%
- **MEEdC** 0,7%
- **MESCS** 2,1%
- **TEKES** 1,2%
- **TEKES-TARTU** 1,0%
- **UL** 0,3%
- **VR** 1,2%
- **KIT** 16,0%
- **IST** 0,4%
- **IPPLM** 0,2%
- **HELENIC** 0,9%
- **HAS** 0,5%
- **FZJ** 4,6%
- **IPP** 1,0%
- **ENEA** 7,4%
- **EÚA-CNR** 0,5%
- **DTU** 0,4%
- **CCFE** 7,2%
- **CEA** 10,3%
- **SCK-CEN** 4,5%
- **Coordinator (IPP / EÚA)** 24,2%
Overall Objectives 2014-2018

- Fill gaps in the database and develop design codes
  - for the baseline materials including irradiation data
- Development of new materials
  - to mitigate requirements of advanced DEMO component designs
- Demonstration of the production of such materials in processes scalable to industrial standards
- Characterization of the properties of such materials
- Develop models for neutron radiation effects
  - specifically microstructural evolution and embrittlement, in iron alloys, steels, tungsten, and degradation of functional materials
Optimization of RAFM steels for possible water cooling

- Various strategies developed (reviews and thermodynamical calculations)
- Specific thermal treatments tested to optimise Ductile-Brittle Transition Temperature
- Alternative heat treatments thus far unsuccessful

Two batches of 76 kg have been produced in 2014 at CSM, Italy. Six batches at OCAS, Belgium. These are under investigation
Adjustment of EUROFER properties by varying heat treatment temperatures

- Austenitisation: 980 °C – 1150 °C
- Tempering: 700 °C – 760 °C

Very successful

Graphs showing:
- Charpy Properties
- Tensile Strength
- Creep Strength

J. Henry, CEA

13 new heats ready, 9 under investigation
Advanced Steels: ODS & RAFM steels

ODS steel: Fabrication & Demonstration
- Production of a 100 kg 14%Cr ODS steel batch by mechanical alloying
  - Plates: thickness 2 mm, size 2 m²
  - Demonstration of applicability to first wall
- Alternatives to mechanical alloying (feasibility studies and industrial large-scale fabrication)

Development of RAFM steels for high temperature applications
- Specific thermal treatments
- Fine tuning of the chemical composition
- Special thermo-mechanical treatments (TMT)

Optimization of RAFM steels for possible water cooling
- Specific thermal treatments (for optimum DBTT)
- Change of chemical composition (for optimum DBTT)
Steel development supported by thermodynamic calculations

→ high temperature steel: nine 80 kg batches produced, four 100 kg batches in production
→ alternative ODS steel production: 23 lab-scale batches produced (250 – 550 g each)
High Heat Flux Materials – Objectives

Develop design options (mainly for the divertor)

(I) Helium Cooled Divertor (HCD)
- Coolant temperature limited to 700-800 °C due to conventional technology
- Main problems: (1) design, (2) structural material

(II) Water Cooled Divertor Structure (WCD)
- CuCrZr: T>300 °C → softening
- laminates, particle and fiber reinforced CuCrZr for possible operation at higher temperatures
- Large-scale industrial manufacturing processes → pipes

(III) Divertor W Armor Parts (e.g. monoblocks, tiles)
- mass fabrication by powder injection moulding (PIM)
- tailoring relevant material properties
- high heat flux testing of materials and (small) mockups

(IV) Blanket First Wall
- safety option against air ingress

Materials
- W-X laminated pipes

Materials
- W-W/fiber composites
- WC & SiC reinforced W
- W alloy development (PIM)
- Cu-W (fiber, particle, laminated) composites
- W/Cu functionally graded
- Self-passivating W alloys

Materials
- Self-passivating W alloys
Achievements – HHFM

30 samples and 16 bars of self-passivating W-alloys produced by HIP and first HHF test in GLADIS

F. Koch, IPP
C. Garcia-Rosales, CEIT
A. Litnovsky, FZJ

Production of WC and SiC reinforced W materials

A. Ivekovič, S. Novak, JSI

Cu-WC composites as thermal barrier

Physical & microstructural characterisation of W plates

A. Galatanu, MEdC

W. Pantleon, DTU
Achievements – HHFM

Mass production of W parts and W alloys development

Monoblocks with various shapes

Samples for ASDEX Upgrade

Green parts vs pre-sintered parts

W Alloys Development

S. Antusch, KIT

Langmuir probes for WEST

A.J.H. Donné | ICFRM – 17, Aachen | 12th October 2015 | Page 22 / 52
HHF Tests in JUDITH: PLANSEE pure tungsten according to ITER specifications ("IGP") compared to PIM W alloys

<table>
<thead>
<tr>
<th>#</th>
<th>T [°C]</th>
<th>$P_{\text{abs}}$ [GW/m$^2$]</th>
<th>$\Delta t$ [ms]</th>
<th>$E_{\text{abs}}$ [MJ/m$^2$]</th>
<th>FHF [MW/m$^2$$^*$$s^{1/2}$]</th>
<th># shots</th>
</tr>
</thead>
<tbody>
<tr>
<td>°C</td>
<td>1000</td>
<td>0.38</td>
<td>1</td>
<td>0.38</td>
<td>12</td>
<td>1000</td>
</tr>
</tbody>
</table>

G. Pintsuk, M. Wirtz, Th. Loewenhoff, FZJ
Achievements – HHFM

W-Cu laminated pipe
length up to 1000 mm

Appl. 1: divertor heat sink

J. Reiser, KIT

Appl. 2: Interface

W
CuCrZr
High Heat Flux Materials

Cu-W(fiber) composite tubes

A. v. Müller, J.-H. You, IPP

W-W(fiber) composite

ErO_x/W

ZrO_x/W

J. Riesch, J.-H. You, IPP

J.W. Coenen, FZJ

Fracture Mechanics

$K_{IQ} [\text{MPa} \cdot \text{m}^{1/2}]$

Normal to RD
Parallel to RD
45° to RD

V. Nikolic, ÖAW

**Objectives**

- Setup of new Group and increase visibility of the topic
- Define and characterise baseline materials
- Develop materials with improved irradiation resistance

**Achievements**

- Two ultra-fine grained alumina produced in cooperation with industry
- Alumina characterisation performed before and after gamma- and heavy ion-irradiation
- Alumina successfully fabricated by SPS (lab scale)
- Assessment of diamond disks from three different suppliers completed (2d loss tangent measurements)
- Irradiation defects models of alumina by DFT established
Ultra-fine grained (UFG) alumina in coop. with industry

GS=0.40 µm
GS=0.14 µm

R. Vila, CIEMAT

Assessment of diamond disks from diff. manuf.

1.2 / 1.8 / 2.3
1.2 / 1.9 / 3.2
1.2 / 1.7 / 2.5

Th. Scherer, KIT

UFG-alumina: unirradiated and after gamma irradiation

loss tangent

optical absorption

Alumina fabrication by SPS

ρ=93.1%, GS=0.63 µm
ρ=99.9%, GS=1.6 µm
ρ=99.9%, GS=8.5 µm

Modelling of radiation defects in Al₂O₃

A. Lushchik, UL

A. Popov, UL
Further Activities

Integrated Radiation Effects Modelling and Experimental Validation (IREMEV)

- phase diagrams for solute segregation to defects
- reactions, recombination and clustering of radiation defects
- helium desorption
- defect production in high energy cascades
- effect of helium and of transmutation products on defect production
- accumulation of helium and hydrogen in the microstructure
- identification of the origin of the synergetic enhancement of swelling

Engineering Data & Design Integration (EDDI)

- design criteria, codes & standards, material handbooks
- priority design needs in terms of material data and performance
- experiments for specific design data
Plasma-Surface Interaction Processes

Processes depend on material mass, projectile mass, material mix and concentration, impact energy ($E_{in}$), impact angle ($\alpha$), surface roughness and temperature ($T_{surf}$).
The Change of the JET Wall

**JET-C**
*until October 2009*

**JET-ITER-Like Wall**
*since May 2011*

- Carbon CFC
- Be on Inconel
- W
- Be
The aim is to have a complete overview of material migration and material damage, not just a number (even large) of analysis points and isolated findings.
Material studies in JET

To determine material migration and fuel retention a large number of plasma-facing components and wall probes have been retrieved and analysed.

Divertor: *set of tiles from the poloidal cross-section*

Limiter: *inner wall guard limiters, outer poloidal limiters, upper dump plates*

**Wall Probes**
DIVERTOR: Be Deposition and Reduced Fuel Retention

Messages:

• Upper part of the inner divertor tiles is the main region of deposition.
• Low deuterium content is measured even in thick deposits.
• Total deuterium retention in the divertor < 1 g.
LIMITERS: Erosion and deuterium retention (Marker Tile)

Messages:
Very low D content on limiters, especially in the central part due to heat-loads.
Erosion of marker layers.

K. Heinola, J. Nucl. Mater. 2015
LIMITERS: Deposition and retention in the castellation
(Need for Beryllium cutting)

Messages:
Shallow deposition in the castellation.
Deuterium content $< 10^{18} \text{ cm}^{-2}$

Total D in the castellation $\sim 9 \times 10^{21} \text{ at.}$
## Total deuterium retention

<table>
<thead>
<tr>
<th>Component</th>
<th>Integrated D content</th>
</tr>
</thead>
<tbody>
<tr>
<td>Inner Divertor</td>
<td>$2.0 \times 10^{23}$</td>
</tr>
<tr>
<td>Outer Divertor</td>
<td>$0.9 \times 10^{23}$</td>
</tr>
<tr>
<td>Divertor (total)</td>
<td>$2.9 \times 10^{23}$ in 13.1 h</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Total operation time (h)</td>
<td>45.1</td>
<td>19.1</td>
</tr>
<tr>
<td>Total retention (g)</td>
<td>~ 50 g</td>
<td>~ 1.3 g</td>
</tr>
<tr>
<td>Retention rate (s⁻¹)</td>
<td>$9.2 \times 10^{19}$</td>
<td>$0.6 \times 10^{19}$</td>
</tr>
</tbody>
</table>

**Message:** Retention rate in JET-ILW is reduced over 15 x.
Comparison of Deposition: JET-C versus JET-ILW

Divertor Tile 4 \((2BNG4C)\)

**Message:** Very thick deposits seen in JET-C did not occur with ILW (2011-2012)
Main source of dust in JET-C: spalling deposits.

No such thick deposits in JET-ILW divertor.

- 2012 dust sent to IFERC Rokkasho (under BA)
- 1.5 g after 2013-2014 operation.
Beryllium flakes of deposits: Composition

E. Fortuna, J. Grzonka, Warsaw University of Technology, IPPLM, Poland
First wall made of Be in JET-ILW vs. C in JET-C

- Sputtering at recessed wall at impact energies ($E_{\text{in}} < 10\text{eV}$) and by charge-exchange neutrals
- 4-5 times smaller primary source with Be
- Absence of low energy erosion in the Be case => chemical erosion in case of C in JET-C!
- Eroded Be transported towards inner divertor
- Reduced step-wise transport and even net-erosion at the inner divertor target plate

Global Material Migration in JET-ILW

Fair balance between Be eroded in main chamber and Be deposited in divertor after first year of ILW operation

S. Brezinsek JNM 2015, NF 2015
A. Widdowson Phys. Scripta 2013
Material Migration in JET-ILW

- Verification of material migration codes (WallDYN and ERO) for ITER

- Deuterium retention determined by co-deposition (2/3) and implantation (1/3)
- Both the pattern of the deposition and the absolute value of the fuel retention in JET-ILW have been reproduced with WallDYN!
- ITER predictions assuming no impact of seeding species and neutrons so far: 700g limit in ITER with Be+W walls (without cleaning) in 3000-20000 discharges
Deuterium Retention in Tungsten under Influence of He and Ar

PSI-2 exposure D+He plasma on W:
- Total deuterium retention is reduced by a factor of 3
- Nano-size bubbles in depth up to ~10 nm

PSI-2 exposure D+Ar plasma on W:
- Total deuterium retention slightly increased
- Change in trapping sites due to material damage by Ar
- N and Ne interaction also modifies the retention

M. Reinhart et al., JNM 2014
Impact of Neutron Damage and Self-damage on Fuel Retention

Comparison self damage (W ion) vs. neutron (fission) defect damage in W

- Identification of base mechanisms as function of dpa
- Effect of plasma (high heat loading; H, He ions) on mechanical properties

- Mono-vacancies, vacancy clusters and dislocation loops for both self-damaged n-irradiated materials
- Saturation of vacancy-like defects in self-damaged W at 0.25dpa. No saturation for n-irradiated W at 0.71dpa.
- TEM of neutron irradiated materials:
  - Low dose (0.22dpa): convoluted dislocation raft
  - High dose (0.71dpa): growing loops split up in small dislocation loops
- Plasma heat loading partly annihilates small dislocation loops and voids from n irradiation

- Defect creation (hardening) by neutrons (heavy ions) interplays against defect removal/recovery (softening) by plasma high heat loading

I. Uytdenhouwen et al., PFMC 2015 (in press)

See O76 v Renterghem Po 3-74 Uytdenhouwen
Tokamak Specific Studies with W: W Prompt Re-deposition in ASDEX Upgrade

**Prompt re-deposition**

![Diagram of plasma, re-deposition process, and thickness of W marker.]

Eroded W is ionised and hits the W PFC within the first Larmor radius

**Main results from AUG:**

- Prompt re-deposition of W on recessed C trench ~30% of net erosion (Qualitative agreement with ERO code modeling)
- Net deposition on both sides of the strike point! W from main chamber contributes to deposition peaks!

![Graphs showing thickness of W marker and re-deposition of W.]

First experiment with AUG divertor manipulator


A.J.H. Donné | ICFRM – 17, Aachen | 12th October 2015 | Page 45 / 52
Synergies in Power and Particle Load Exposures

- Transient power load qualification devices: e-beam, D\(^+\)-beam, Plasma, Laser
- Investigate synergy effect in combined heat and plasma load experiments

**Laser beam**
- 1000 ELM-like events at RT
- Absorbed power density: 0.3 GW/m\(^2\)

**H-Plasma**
- Biasing voltage: -60 V
- Plasma flux: 2.5 – 4.0 \(\times\) \(10^{21}\) m\(^{-2}\)s\(^{-1}\)

**Laser + H-Plasma**

Simultaneous (\(\Delta T \approx 100^\circ C\))

**H-Plasma + Laser**

- Only moderate morphology changes at 400°C – only surface roughness (impacts erosion)
- NEXT step: combine with seeding species (Ne, Ar, N\(_2\)) and He

See O45M. Wirtz
Bulk W Divertor: JET vs. ITER

ITER divertor (actively cooled)

JET ITER-like wall (inertial cooling)

Base temperature
≈ 2800°C
ΔT by ELMs
≈ 800–1000K

Controlled flash melting by ELM induced transient temperature excursions.

J. Linke et al.

J.W. Coenen NF 2015

Surface temp.

melt

recrystallization

brittle

T_{melt}

DBTT

0 450 s

time

J.W. Coenen NF 2015

A.J.H. Donné | ICFRM – 17, Aachen | 12th October 2015 | Page 47 / 52
- ELM size ~ 300 kJ → $q_{||} = 0.5 – 1.0 \text{ GW/m}^2$
- Small W influx events
- Minor W pollution in the core
- No significant impact on plasma operation

Operation with damaged lamella possible
MEMOS benchmark for ITER predictions

MEMOS MODELLING
VALIDATION EXPERIMENT
B. Bazylev et al.
ILW 2011-2012 Dust studies: W and Mo-W

E. Fortuna, J. Grzonka, M. Rubel, Warsaw University of Technology, IPPLM, Poland
Qualification of EUROFER as Potential Plasma-Facing Material for DEMO Wall

- Preferential sputtering of Fe in EUROFER leads to enrichment of W at the surface
- Sputtering experiments under D$^+$ impact reveal a reduction of the effective sputtering yield and increase of the surface concentration of W („thin effective W surface“)

Sputtering of EUROFER by D

In steady state: W enriched at the surface


See I22 W. Jacob
Pedestal and Confinement Degradation with the JET-ILW

- JET-ILW shows degradation of confinement with respect to JET-C operation
- Degradation partially governed by fuelling requirements to allow safe operation in W
- Degradation is not in the plasma core, but determined by changes in the pedestal
- Pedestal recovery possible by increase of $\beta_N$ or by N$_2$ seeding
  [G. Giroud NF2013] [R. Neu JNM2013]
  Research is on-going to solve this before DT operation in JET!

![Diagram of plasma pressure vs. normalised radius]

- H-mode
- L-mode
- edge layer
- SOL
- transport barrier
- pedestal
- $T_e$ (keV)
- $n_e$ $10^{19}$ (m$^{-3}$)

M. Beurskens and J. Schweinzer
NF2014
EUROfusion integrated program on materials

- Basic laboratory experiments and modelling
- Fusion experiments
- Material development and characterisation