Acknowledgements


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*See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia.
**« Long pulse operation » ?**

Tokamaks operating long pulses have to overcome and control several plasma-core and plasma-wall interaction time scales

- MHD
- E. conf
- Heat loads
- Part + Imp
- Cur. Diff.
- Retention
- Erosion/deposition

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**Outline**

Since the installation of the metallic wall (2011) JET has carried out a series of experiments relevant for « long pulses » and ITER operation.

1. Erosion of Be/W material and redeposition.
2. Fuel retention in metallic environment
3. Particle/impurity transport
4. Thermal load management on metallic components
5. Current profile access to high $\beta_N$ scenario
6. High $\beta$ stability in metallic wall

**Decreasing time scale**
ILW = 2880 installable items, 15828 tiles (~2 tonnes Be, ~2 tonnes W)

Note: JET is not designed for « long pulse » operation
- Inertial PFC components
- Limited external non-inductive current capabilities

Carbon content in JET plasma discharges has been reduced by ~x10

Zeff is lower (~1.2) instead of 2-2.3 in JET-C
- longer resistive time
- Lower dilution
1. Erosion of Be/W material and redeposition.
2. Fuel retention in metallic environment

Flaking & dust in JET-ILW strongly reduced

- Very thick deposits seen in JET-C did not occur with ILW (2011-2012)
- Flakes building up can lead eventually to operational difficulties when they detach
  ➔ see long pulse device recent experience: LHD & Tore supra
Fuel retention reduced by more x 10

From global gas balance:
injected-pumped $D_2 = \text{retained } D\text{ atoms}

S. Brezinsek NF 2013  
K. Schmid PSI 2014

- Retention is mostly independent from the scenario used
- In a C-wall only ~10 ITER 3000s pulses could be run before reaching 700g T limit
- In W/Be wall ~250 pulses could be made.

Fuel retention from gas balance & post mortem

- Total D retention vs gas input
  - JET-C 2001-2004: 66 g, 3.7%  
  - JET-C 2007-2009: ~50 g, 2.1%  
  - JET-ILW 2010-2012: ≤1.3 g, ≤0.3%

- D retention rates
  - JET-C 2007-2009:
    - total 50 g $\rightarrow$ $0.9 \times 10^{20}$ D/s (Gas balance ~ $10-20 \times 10^{20}$ D/s)
    - main chamber 1.8 g $\rightarrow$ $1.3 \times 10^{19}$ D/s (limiter time 12h)
    - divertor 48 g $\rightarrow$ $1.2 \times 10^{20}$ D/s (divertor time 33h)
  - JET-ILW 2010-2012:
    - total 1.3 g $\rightarrow$ $0.6 \times 10^{19}$ D/s (Gas balance ~ $2-20 \times 10^{19}$ D/s)
    - main chamber 0.3 g $\rightarrow$ $4.9 \times 10^{18}$ D/s (limiter time 6h)
    - divertor 1.0 g $\rightarrow$ $6.2 \times 10^{18}$ D/s (divertor time 13h)

- Would still lead to ~3g fuel long term retention in ITER 3000s “long pulses”
  (remaining retention by co-deposition & implantation processes)
PFC temperature can change retention

- In JET W PFCs inertially cooled → changing temperature during plasma sequence.
- In ITER PFC temperature actively cooled.
- The retention increases with decreasing temperature (well known fact)
- The increase is ~50% when the temperature decreases by x2

→ The retention measured in JET-ILW looks applicable to devices with lower wall temperature (i.e. with actively cooled PFCs)
(See WEST: D. Van Houtte Thursday morning)

Eroded Be migrates to inner divertor areas

- Erosion from main wall decreased by a x5 with ILW compared to C
- Be transport to divertor smaller by x4-8 than C in JET-C
- Lower fuel content in Be co-deposits in comparison with C co-deposits
- WALLDYN modelling shows qualitatively the observed deposition pattern on top of tile 1

S. Brezinsek NF 2015
Fuel cannot be fully recovered by conditioning

- Part of the retained fuel can be recovered using recovery techniques such as Ion Cyclotron Wall Cleaning (ICWC).
- ~7x10^22^H recovered with ICWC (JET-ILW) corresponding to the retention in accessible area of the first wall.
- Suggests limited access to co-deposits through plasma isotopic exchange.

Recovery techniques may need be needed in-between long pulses, but cannot access all areas at present.

3. Particle/impurity transport
4. Thermal load management on metallic components
Particle transport: W source / accumulation

W source (sputtering) due to:
- ELMs
- Low Z impurities such as Be, Ne, N₂

W accumulation control
- Increase ELM frequency to flush W (gas puffing/NBI)
- Increase core temperature to screen the W from the plasma core (ICRH)

W source control
- Increase divertor radiation to reduce target temperature (gas puffing/impurity injection)

Core tungsten concentration should stay < 5 × 10⁵

W erosion in H-mode dominated by ELMs

- Intra-ELM sputtering dominates
- Sputtering induced by ion impact D, Be, O, N (seeding gases).
- Total W source between 2 × 10¹⁸ and 3 × 10¹⁹ /s have been observed for the outer divertor
- For the first time effect of prompt deposition observed in tungsten divertor spectroscopy
ELM control essential for W transport control

- Small fuelling pellets reliably trigger ELMs w/o strong impact on density

ICRH can control core W accumulation

- Core W control achieved with central H minority ICRH at 2.5MA/2.7T H-modes.
- Min ICRH for impurity screening is scenario dependent
Stationary conditions can be achieved with N2

![Graph showing stationary conditions with N2]

C. Giroud, IAEA 2014

Near detached conditions achievable with N2

![Graph showing near detached conditions with N2]

A factor of 5 reduction in the plate particle current

A. Jarvinen, IAEA 2014

2.65T, 2.5 MA, P\textsubscript{in} ~ 16 MW
Reduced W erosion obtained in near-detached conditions

- Full detachment not desirable ➔ impact on confinement & plasma control
- Trade off between increasing impurity concentration in the discharge and decreasing $T_{\text{div}}$

$\Rightarrow$ Detachment control is an essential part in extending the duration of the pulse for both W sputtering control and heat load management.

G. Van Rooij, IAEA 2012

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Monitoring temperature on PFCs in metallic walls essential tool for long pulse operation.

- Classical heat up alarm
- Hot spot alarm

- Management of heat loads on PFCs is essential in long pulse discharges to ensure device first wall integrity.
- Response to alarms also an integrated part of the long pulse scenario design.
5. Current profile access to high $\beta_N$ scenario
6. High $\beta$ stability in metallic wall

High $\beta_N$ discharges developed in JET-ILW

- Duration of high performance phase limited by MHD
- Core radiation peaking correlated with tearing modes and $1/1$ islands and impacts on performance – new with ILW!

- Further optimization requires simultaneous control:
  - power to maintain $\beta_N$
  - gas and regular ELMs
  - $q$ profile to minimize MHD

C D Challis 2014
Global confinement increases with increasing $\beta_p$

- Rapid increase in $\beta$ with power due to:
  - Increase with core temperature consistent with transport modelling including fast ion effects. (See J. Garcia oral this afternoon)
  - Increase of density peaking correlated with collisionality
  - Increase of pedestal pressure consistent with peeling-balloning modelling

High $\beta_p$ favorable for steady state operation:
  - Higher bootstrap current
  - Higher pedestal pressure

Power scan in type I ELMy H-mode shows weaker confinement degradation with power than expected from IPB98(y,2) scaling

J. Garcia, IAEA 2014
C. Maggi, IAEA 2014

AT-scenario explored in JET-ILW

- Start-up conditions (BD, Ip ramp) are quite reproducible. High clearance shape chosen for this experiment
- Confinement appears lower at high $q_{min}$
- MHD behaviour consistent with $q_{min}>2$ for early times (i.e. 3/1) and $q_{min}~1$ for later time (i.e. 2/1).
Access to high $q_{\text{min}}$ scenario in JET-ILW appears feasible

Higher initial $q$: higher density target

J. Mailloux / E. Joffrin

MHD spectroscopy used for the first time with the ITER-like wall

- MHD spectroscopy used for the first time in JET-ILW high $\beta$ scenario.
- Preliminary data suggest a somewhat lower no-wall $\beta$ limit compared to previous experiment in the JET-C.
- Assessment of the $\beta$ limit on-going both experimentally and with non-linear MHD stability calculations.

L. Piron / M. Gryaznevitch
JET has contributed to long pulse operation in several key areas:

The metallic (Be/W) has demonstrated important benefit for long pulse operation
- A strong reduction of erosion and flaking \(\rightarrow\) reduces the risks of flake detachment
- A reduced fuel retention \(\rightarrow\) pave the way to repetitive long pulse operation

The metallic (Be/W) is raising additional challenges to long pulse operation
- The control of core W accumulation (with e- heating)
- The minimization of sputtered W sources from the divertor and by ELMs
- Operation and control of detachment with seeding gases.
- The control of transient heat loads on metallic PFCs.

High \(\beta\) and AT-scenario has been explored for the first time in an ITER-like wall.
- Strong link between MHD reconnection event and high Z impurity transport
- High \(\beta_s\) operation benefits from a weaker confinement degradation with power.
- Current profile access with \(q_{min}\) from 1 to 2 appears feasible
- No-wall stability limit has been investigated for the first time with RFA.

JET scenarios have still to overcome key challenges to extend their duration

- Hybrid: 2.5MA/2.9T (\(q_{95}=3.7\))
  - MHD \(\rightarrow\) radiation peaking
  - Control of divertor temperature with seeding gases.
- Baseline: 3.5MA/3.3T (\(q_{95}=3\))
  - Control of the divertor temperature with seeding gases / detachment.
  - Minimisation of sputtered W source by ELMs

These scenarios will be the reference scenario for the future JET DT-phase.
2015/2016 programme

19th October 2015 to 4th April 2016 (198 sessions ~10 pulses each)

N seeding impacts on pedestal confinement

With N
- Radiation reduces target temperature
- Higher $T_{ped}$
- Pedestal pressure loss partially recovered

With Ne
- Radiation reduces target temperature
- Beneficial effect on pedestal not observed

C Giroud – IAEA 2014
Major challenges to achieve DT goals (DT experiments planned for 2017)

Global Confinement
- **Gas puffing (control W accumulation)** → low pedestal temperatures → low core temperatures through stiffness
- **Stronger pumping (density control)** → increases $T_{\text{ped}}$, $T_{\text{e,core}}$ → recovery of confinement
- **High $\beta_N$ (high $P_{\text{IN}}$)** → recovery of confinement

Performance ($P_{\text{fus}}$)
- **Density too high** → Core temperature too low
- **Impurity seeding** → reduction of performance by dilution

Pulse length
- Impurity accumulation (W source due to sputtering) → gas puffing, ICRH and impurity seeding (Ne)
- Target temperature limit → strike point sweeping, impurity seeding
- MHD (hybrid) → $q$ profile optimisation

Controlling power to divertor: impurity injection

**Effect of impurities on core/edge confinement**
- Injection of Ne increases both core, pedestal and divertor radiation
- Ne could be more effective in the divertor for hotter pedestals
- Reduced plasma performance by dilution
- Initial experiments with Ne show no further confinement degradation
- Extrapolations of effect of Ne seeding on $c_W$ and heat loads at 4.5MA ongoing
Divertor heat load control

JET has restrictive operation limits for bulk W and W coated CFC tiles ➔ pulse length at high power at high plasma current

Extrinsic impurities
- $N_2$ not compatible with DT operation ➔ Tritium plant at JET
- Ne: Small reduction of target surface temperature

Strike point sweeping at 4Hz
- 4cm sweeping ➔ strong reduction of surface target temperature
Deposition strongly reduced in JET-ILW

A. Widdowson et al.

Very thick deposits seen in JET-C did not occur with ILW (2011-2012)

Flakes can lead to an increased disruptivity in carbon wall long pulse device: see LHD & Tore supra experience

Test mirrors: Inner divertor

Very thick deposits seen in JET-C did not occur with ILW (2011-2012)

Flakes can lead to an increased disruptivity in carbon wall long pulse device: see LHD & Tore supra experience

Test mirrors: Inner divertor
• M13-34: preheat gas & $n_e$ increased further in order to meet the permit for protection against shine-through with the high voltage PINIs
  • preheat $n_e$ for ILW AT plasmas in C33 is $\sim 5 \times 10^{19}$ m$^{-3}$, twice that for C-wall optimised shear shots
  • Importantly: $n_e$ will have to be increased further in order to use the highest voltage pinis ($6.9 \times 10^{19}$ m$^{-3}$ for 125KV tangential beam) needed for a DT experiment

- As a consequence: even more difficult to obtain plasmas with high $q_0$ – could not obtain $q_0>2$ on 21st Aug sessions
  • A small amount of $P_{\text{LHCD}}$ lead to higher $q_0$, but will need higher $P_{\text{LHCD}}$ for stronger effect – but not available in next campaign
  • Increasing $I_p$ ramp rate also lead to small rise in $q_0$, may be we can push this further?
  • Not tried yet in these plasmas: ICRF heating (but tried in B13-09)
  • What will help in next campaign: evidence that at higher BT higher qmin obtained (Ex-2.1.7)