H&CD application in (burning) plasmas

Ernesto Lerche

15th International ITER School 2026
24 July 2026, Leshan, China

H&CD application in D-T plasmas

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- Laboratory for Plasma Physics, LPP-ERM/KMS, Brussels, Belgium

- Seconded at ITER Organization, H&CD group, St. Paul lez Durance, France

JET tokamak (2009-2023), WEST (… 2023-2026)

- Plasma scenario development & control
- 'Specialized' in heating and fuelling of plasmas (ICRH pilot, expt. coordinator)
- Session leader in JET and WEST ('tokamak pilot')
- Modelling: plasma heating (waves, beams, fast-ion interaction, …)

Disclaimer: The views and opinions expressed herein do not necessarily reflect those of the ITER Organization nor those of the European Commission

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Outline

- Brief introduction to thermonuclear fusion
- Auxiliary heating systems in tokamaks (recap)
  - Introduction
  - Neutral beam injection (NBI)
  - Ion cyclotron heating (ICRH)
  - Electron cyclotron heating (ECRH)
- Overview of JET D-T fusion results
- The role of NBI on JET high performance D-T plasmas
  - Impact of NBI on H-mode access
  - Impact of NBI on flat-top performance
- Impact of ICRH on D-T plasmas & fusion performance
  - H minority heating examples
  - He^3 minority heating examples
  - Fundamental Deuterium and D-NBI heating (D-T fusion record)
  - ICRH for impurity control
- Some ITER predictive D-T simulations
  - Access to burn-through plasmas
  - Impact of ICRH power on fusion performance

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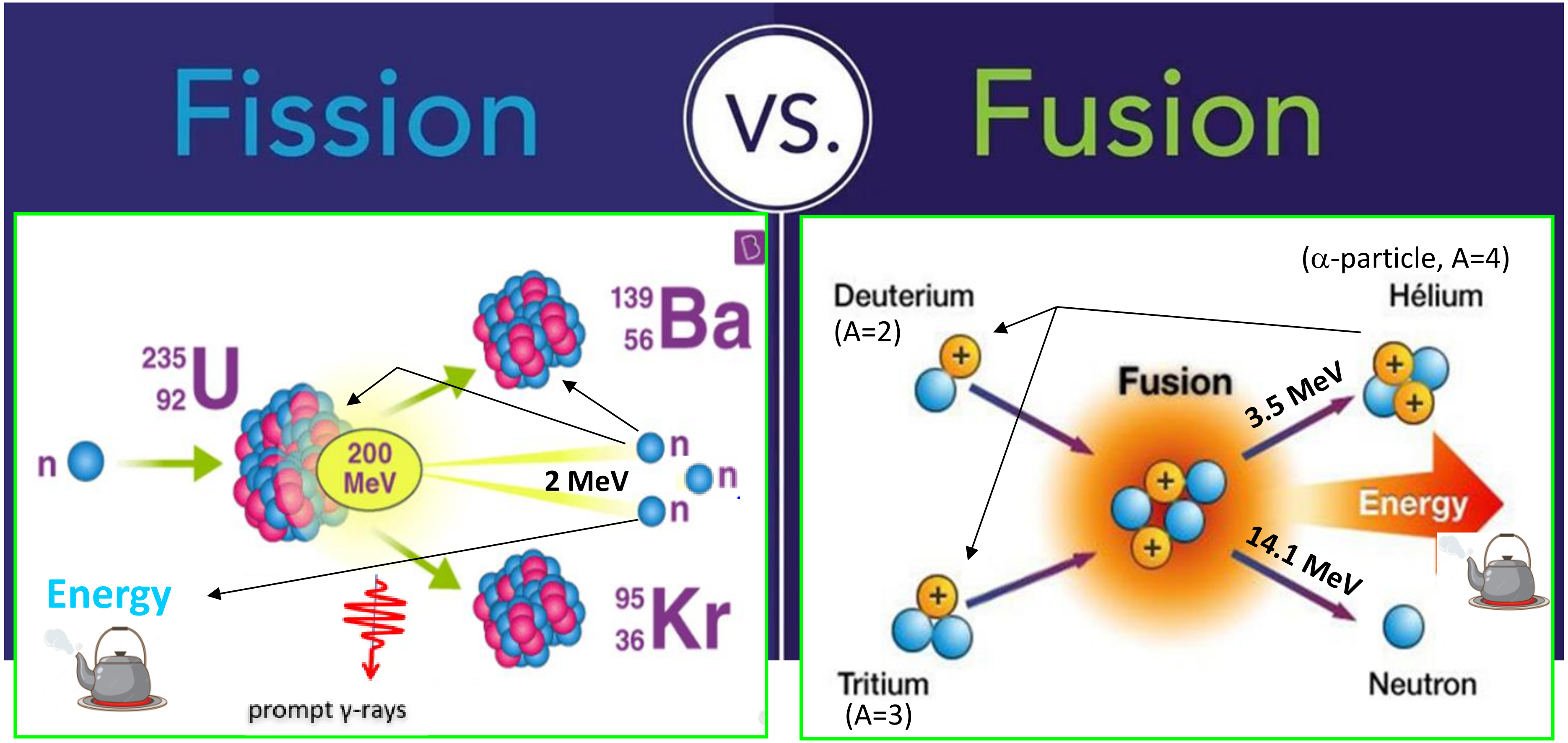
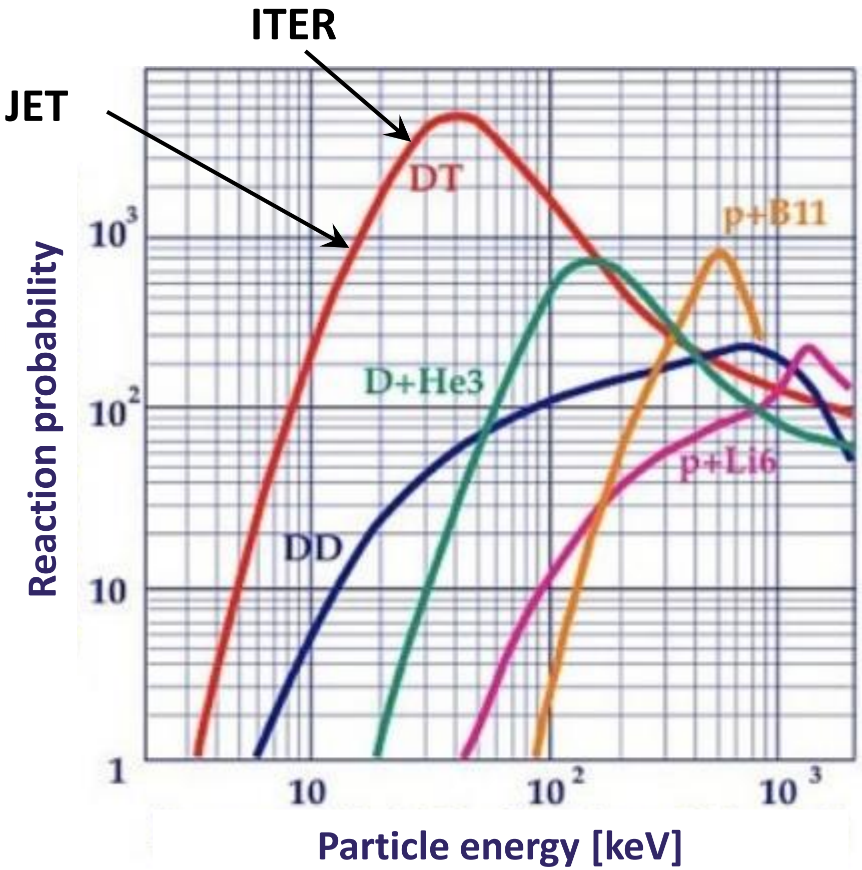
Introduction

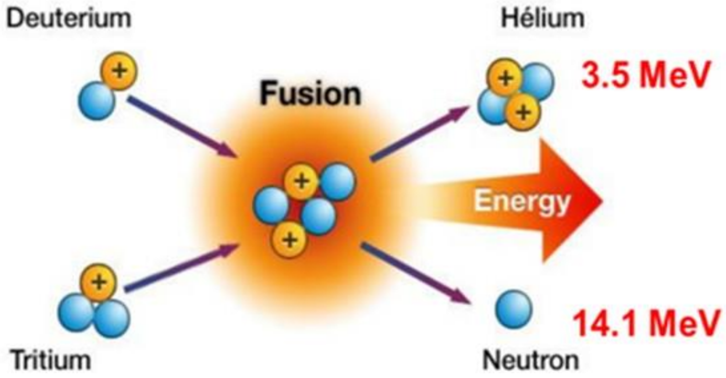


- Output neutrons too energetic:
  - Need moderating bars to slow-down and to control the chain reaction rate
- Strong γ-radiation emission (β,α)
- Long-lasting radioactive waste:
  - 1000's of years

- Helium (α-particle) used directly for self-heating via collisions with D, T ions
- No long-lived radioactive waste:
  - Activation of some internal machine components due to neutrons (10's of years)

Fusion reaction cross-sections





- D:T mixture is the best option
  - Needs E=30-40keV for best fusion performance (hot plasma)

- Intrinsically safe:
  - Self-regulating cross-section

- For JET (E_{bulk}~15keV, E_{fast}~100keV):
  - D-T reactions are split into:
    - (1) thermonuclear (bulk plasma ions)
    - (2) beam-target reactions (fast → bulk ions)

- In ITER and future reactors:
  - Thermonuclear reactions are dominant

The Lawson criterion for sustained fusion

- Triple product: $n\,.\,\tau_E\,.\,T_i$

$$\begin{cases} n = \text{Plasma density} \\ T_i = \text{Ion temperature} \\ \tau_E = W / P_{IN} = \end{cases}$$

Magnetic confinement fusion (tokamaks)

- Large temperature (10’s keV)

- Medium density (~10^{20}/m^3)

- Large τ_E (1-10s)

- Break-even ($Q_{fus} = P_{fus} / P_{IN} = 1$)

→ Losses are compensated by α-heating

$$n\,\tau_E\,T_i > 10^{21}\ \mathrm{keV\,s/m^3} \quad (T_i > 15\mathrm{keV})$$

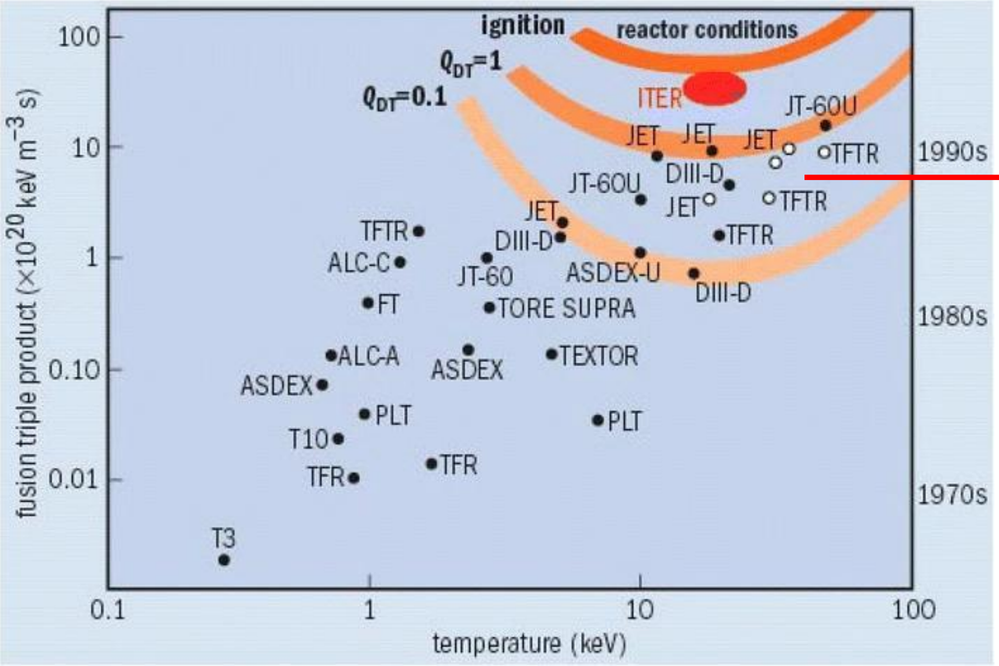
- Ignition ($Q_{fus} = P_{fus} / P_{IN} = \infty$)

→ α-heating alone is enough to sustain the plasma at high temperatures (P_{IN}=0)

$$n\,\tau_E\,T_i > 3.10^{21}\ \mathrm{keV\,s/m^3} \quad (T_i >= 20\mathrm{keV})$$

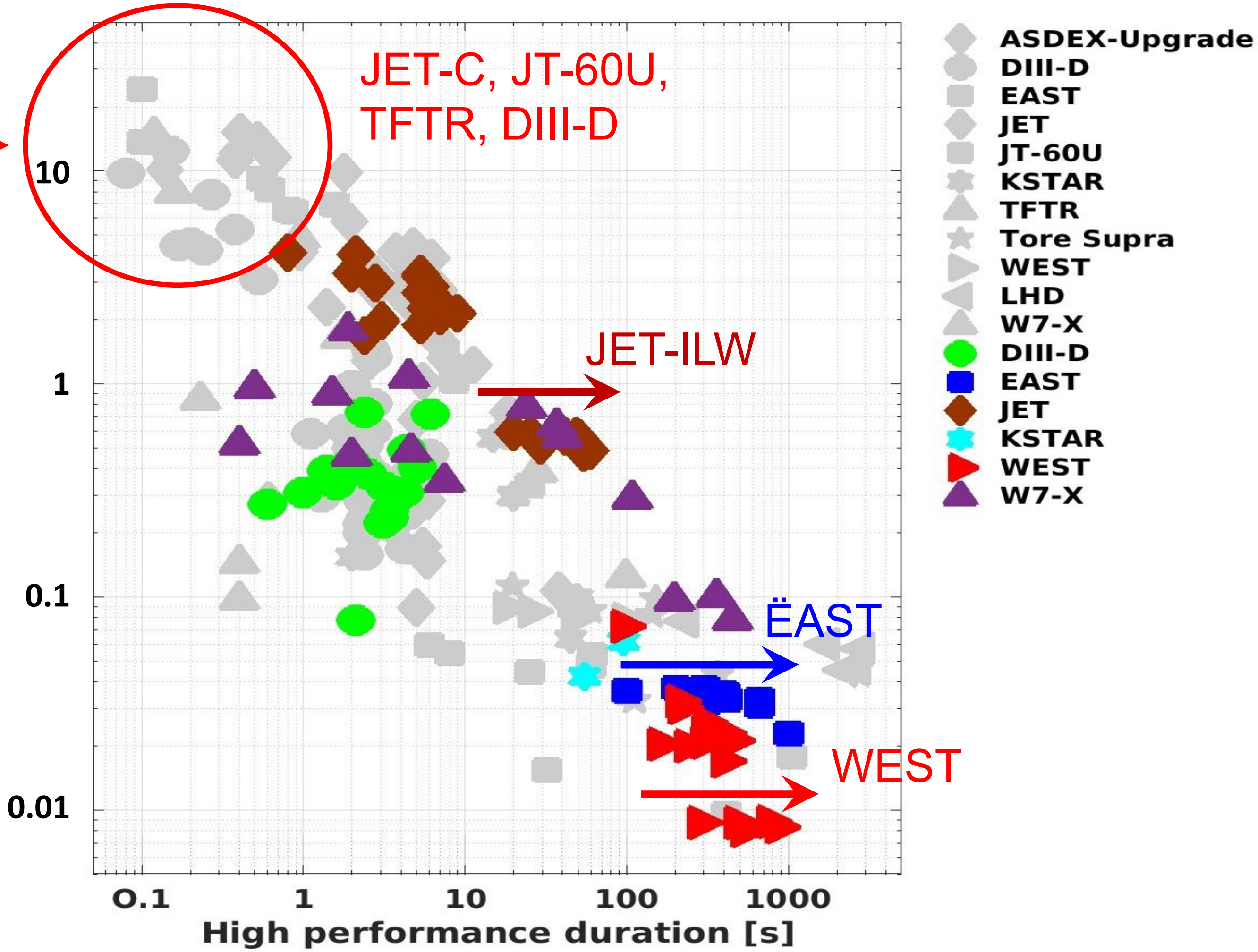
But large triple product is difficult to sustain

Lawson criterion



CICLOP database

(Coordination on Int. Challenges on Long duration OPeration)

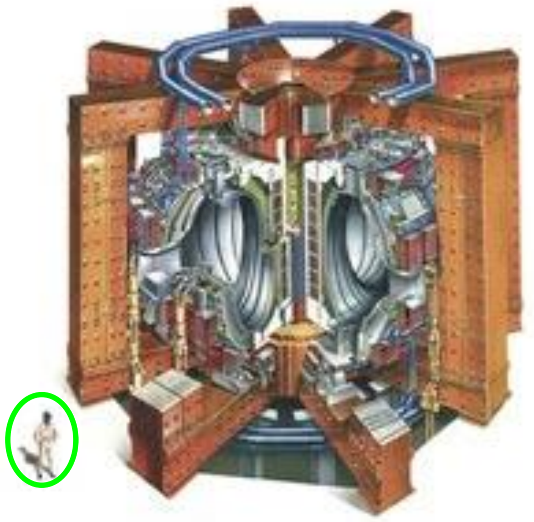


Good progress at constant performance in full-metal devices in the last 5 years

X. Litaudon, NF 2024, 2026
E. Lerche, IAEA-FEC 2024

D-T devices: JET → BEST → ITER → fusion reactor

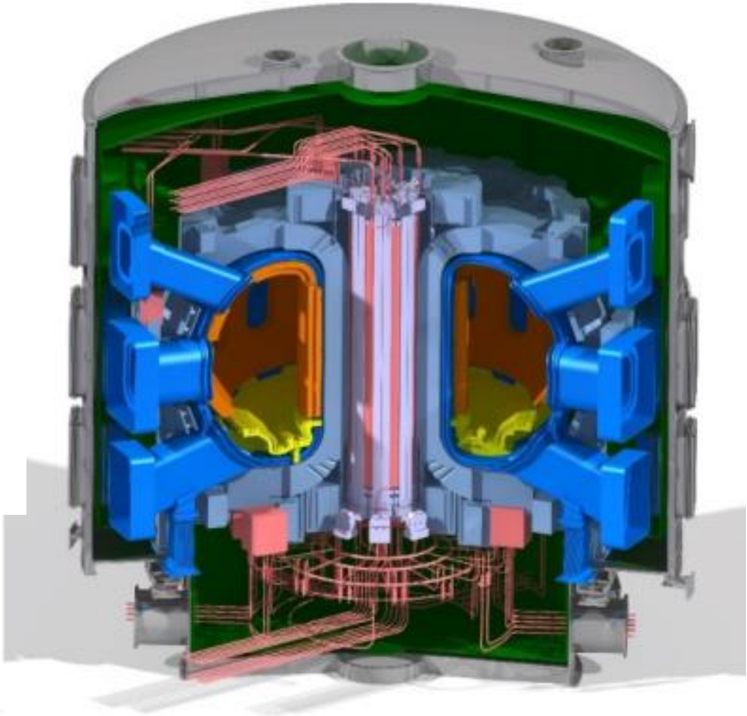
JET Volume = 80 m³



I_P=2-4MA
P_{IN}=30-40MW
P_{fus}=15MW

Q_{fus}=0.2-0.4

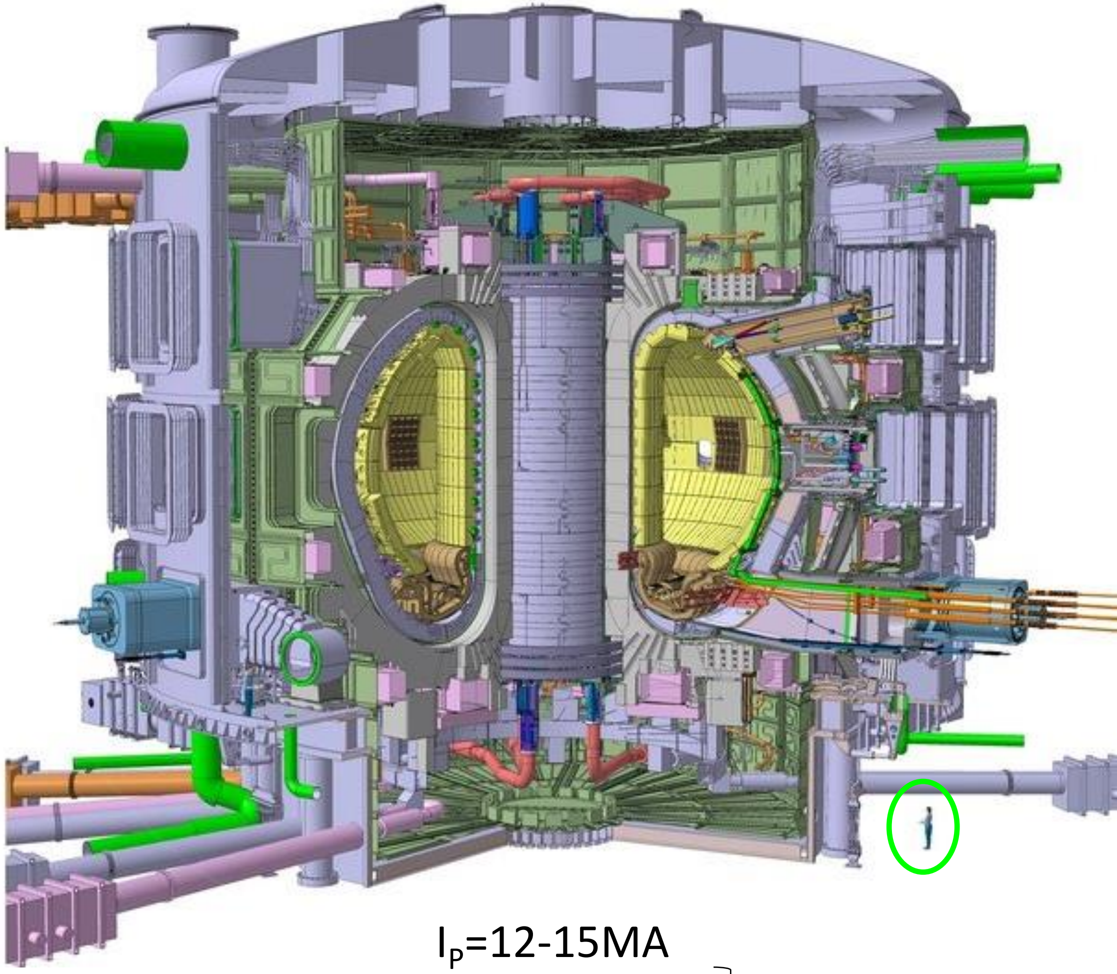
BEST Volume = 140 m³



I_P=6-7MA
P_{IN}=40-50MW
P_{fus}=50MW

Q_{fus}>1

ITER Volume = 800 m³



I_P=12-15MA
P_{IN}=80-90MW
P_{fus}=500MW

Q_{fus}>5

Size (and I_P) matters!

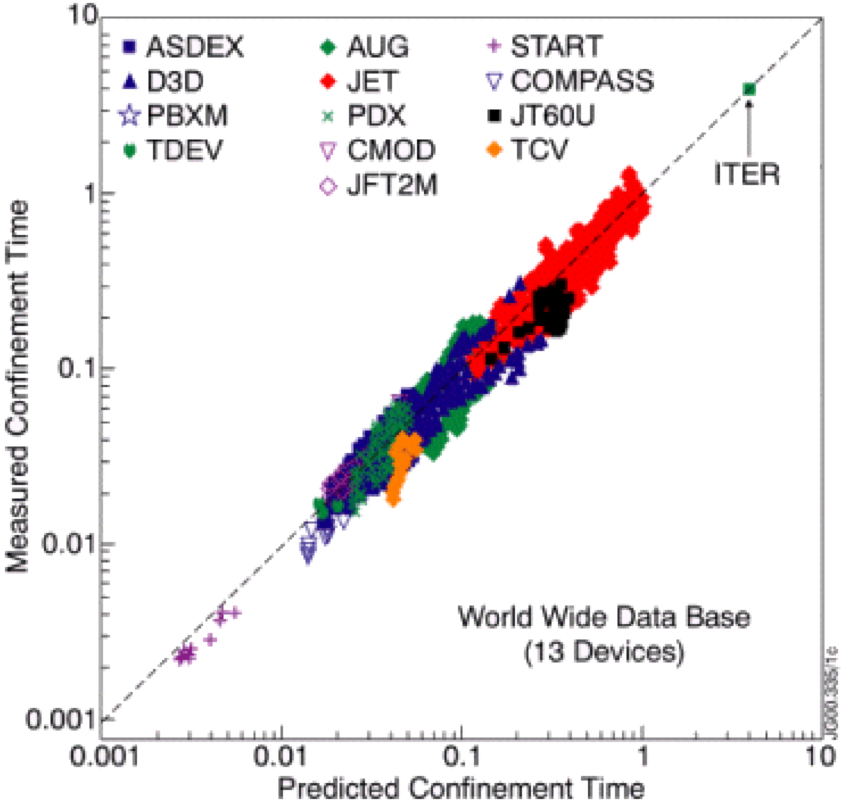
Energy confinement: IPB98 scaling law

Scaling law:

$$\tau_{E\,\mathrm{IPB98}} = 0.056\, I_p^{0.93} B^{0.15} P_{heat}^{-0.69} n^{0.41} M^{0.19} \underbrace{R^{1.39} a^{0.58} k^{0.78}}_{\text{Machine size (plasma volume)}}$$

plasma current

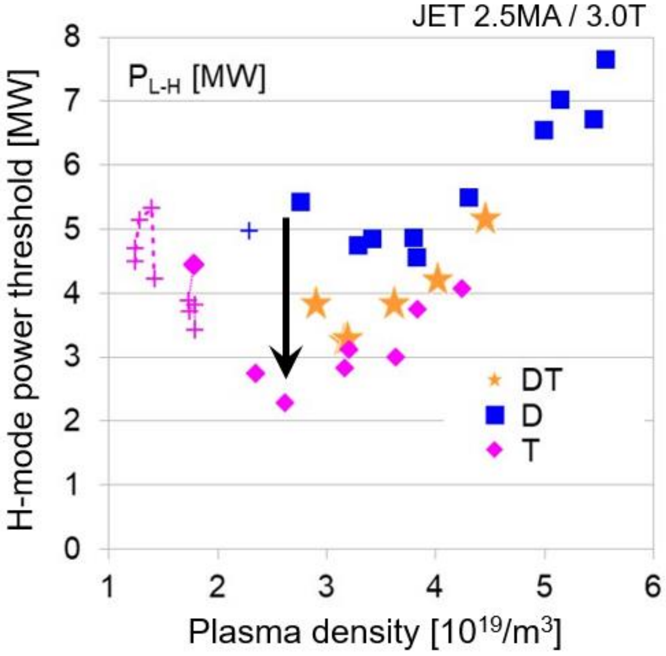
ion mass



ITER PHYSICS BASIS EDITORS, Nucl. Fusion 39 (1999)

- Energy confinement increases with plasma current and plasma volume
- Weak dependence on main ion mass but the power needed to reach H-mode (good confinement) depends on the main ion mass (L-H power threshold) →

H-mode threshold:



E. Solano, Nucl. Fusion 2023

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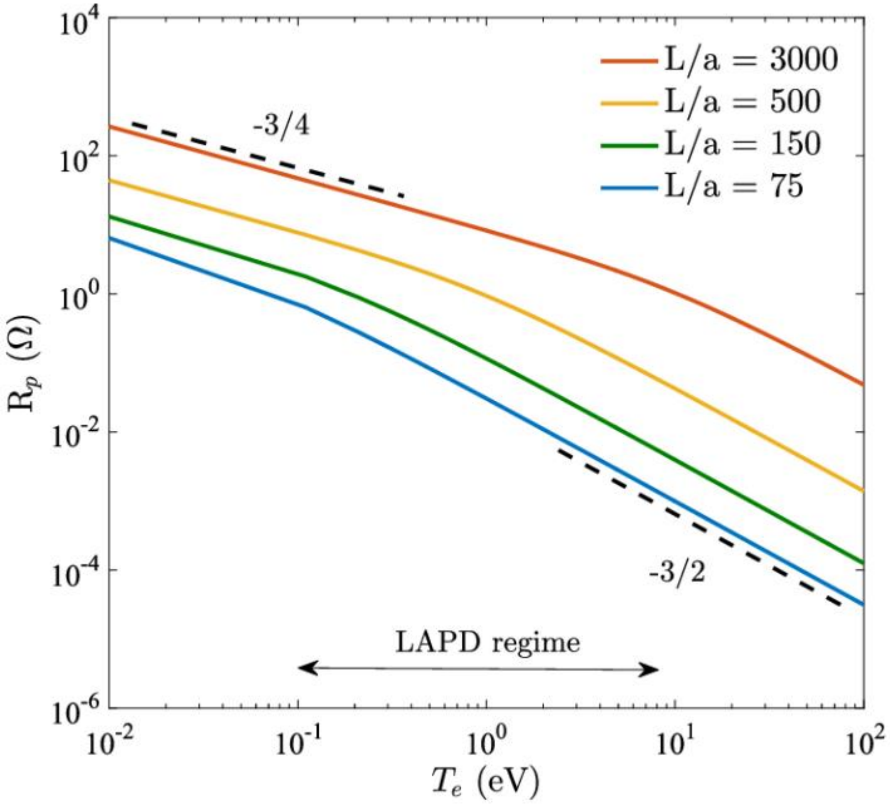
Why the need for auxiliary heating

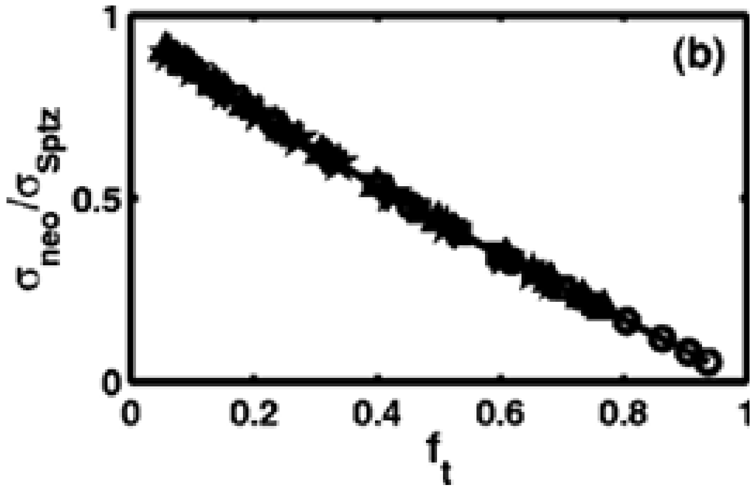
Spitzer resistivity:

$$\eta_{\rm Sp} = \frac{4\sqrt{2\pi}}{3}\frac{Ze^2 m_{\rm e}^{1/2}\ln\Lambda}{(4\pi\varepsilon_0)^2\boxed{(k_{\rm B}T_{\rm e})^{3/2}}}$$



[M. Poulus , PP 2019]

Neoclassical resistivity > Spitzer resistivity (trapped particles)



[O. Sauter et al., PP 1999]

When plasma temperature increases, the plasma becomes less collisional and ohmic heating becomes less efficient

Auxiliary heating systems

- Neutral beam injection (NBI)

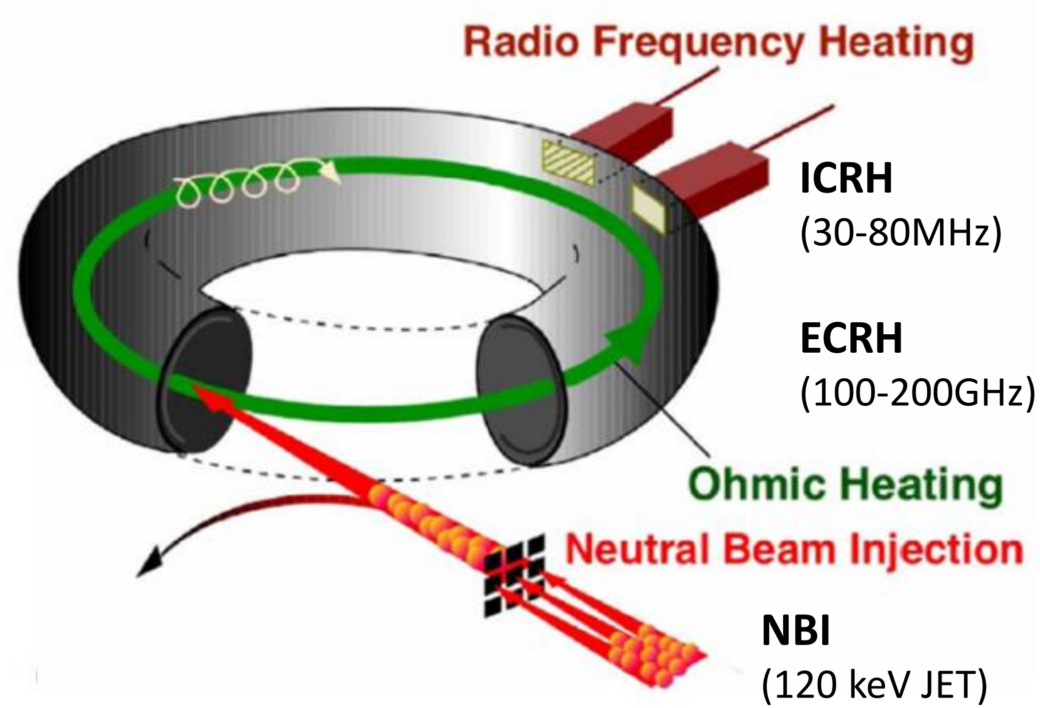
  (high density = high energy)
- Ion cyclotron heating (ICRH)

  (coupling + RF sheaths)
- Electron cyclotron heating (ECRH)

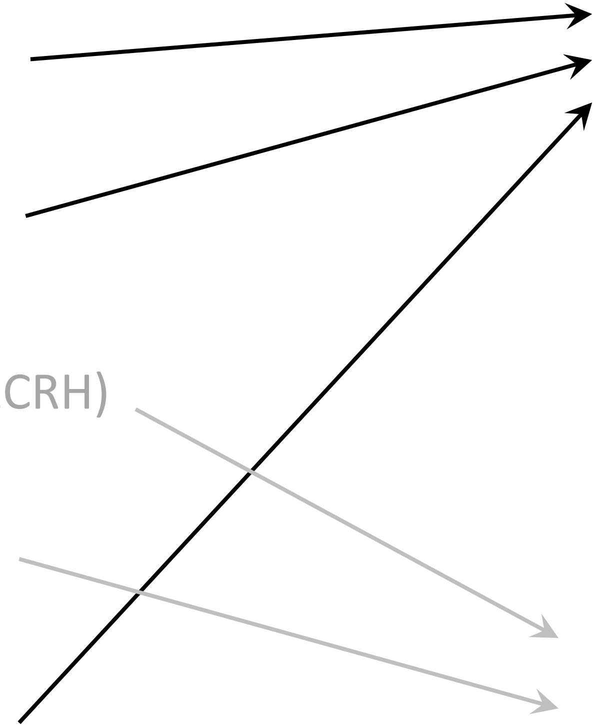
  (exact B_0, polarization)
- Lower hybrid heating (LHCD)

  (penetration at high density)
- Alpha particle (self-) heating

  (confinement + He ash)



Auxiliary heating systems



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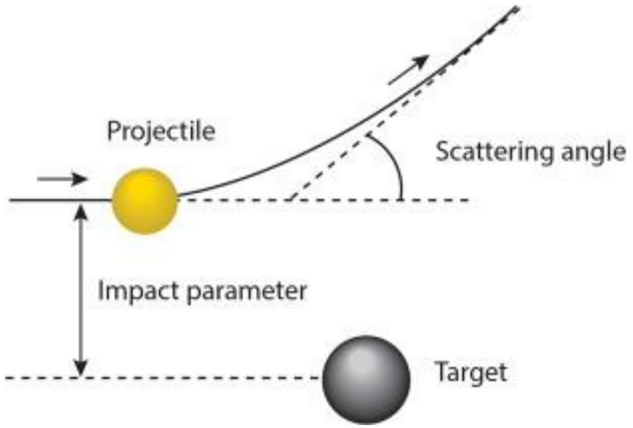
Plasma heating by Coulomb collisions of fast ions



Electron heating → Equipartition
(Thermalization)

Auxiliary heating systems

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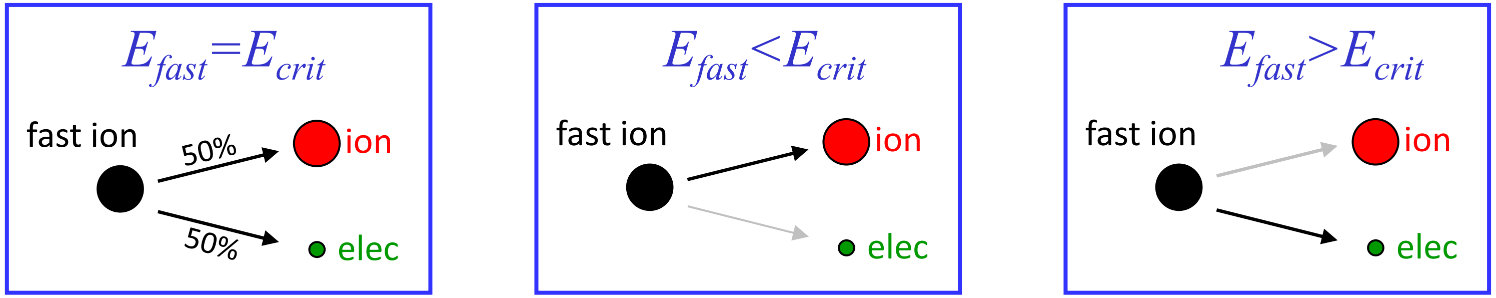
Plasma heating by Coulomb collisions of fast ions

plasma **fast ion** **plasma**

→ Critical energy: $$E_{crit} = 14.8 T_e[keV] m_b \left(\sum_i \frac{N_i Z_i^2}{N_e m_i} \right)^{2/3}$$ (JET~200keV)



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Electron heating → Equipartition

$$\boxed{p_{e-i} = \frac{3}{2} \cdot \frac{q_e \cdot n_e \cdot 10^{19} \cdot (T_e - T_i)}{\tau_{ei}}}$$

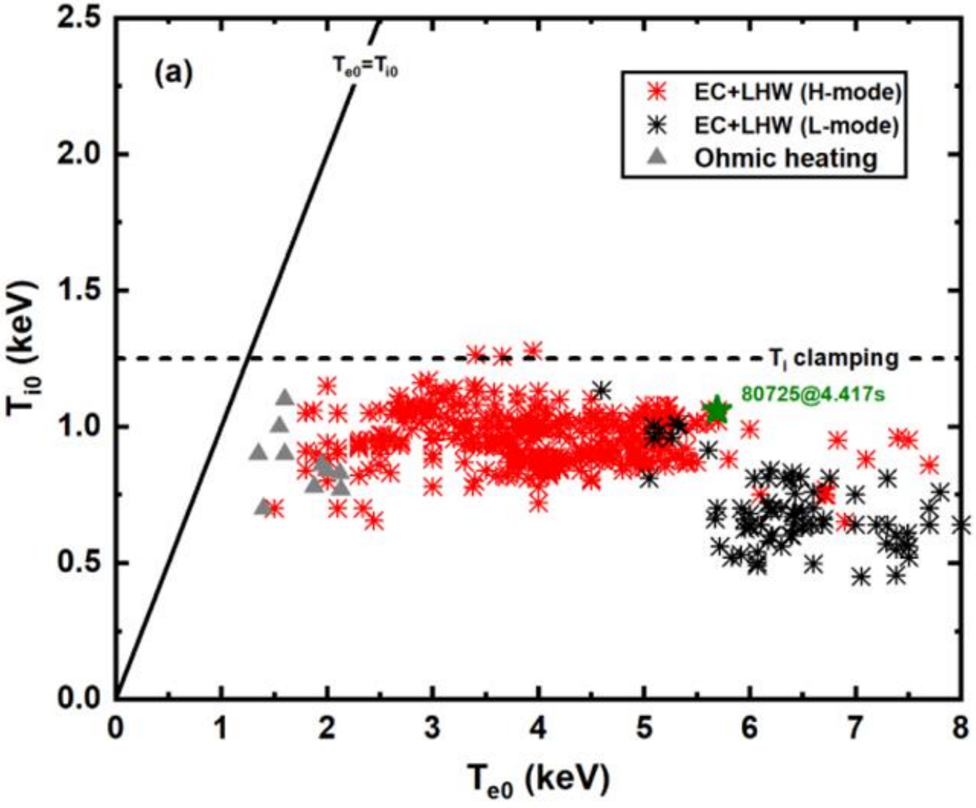
$$\tau_{ei} = \frac{M \cdot m_p}{m_e} \cdot \tau_{ee} \approx \frac{M}{Z^2 \cdot \Lambda_{ee}} \cdot \frac{\boxed{T_e^{3/2}}}{n_i} (s) \longrightarrow$$

Less collisions at high T_e

Equipartition, T_i clamping

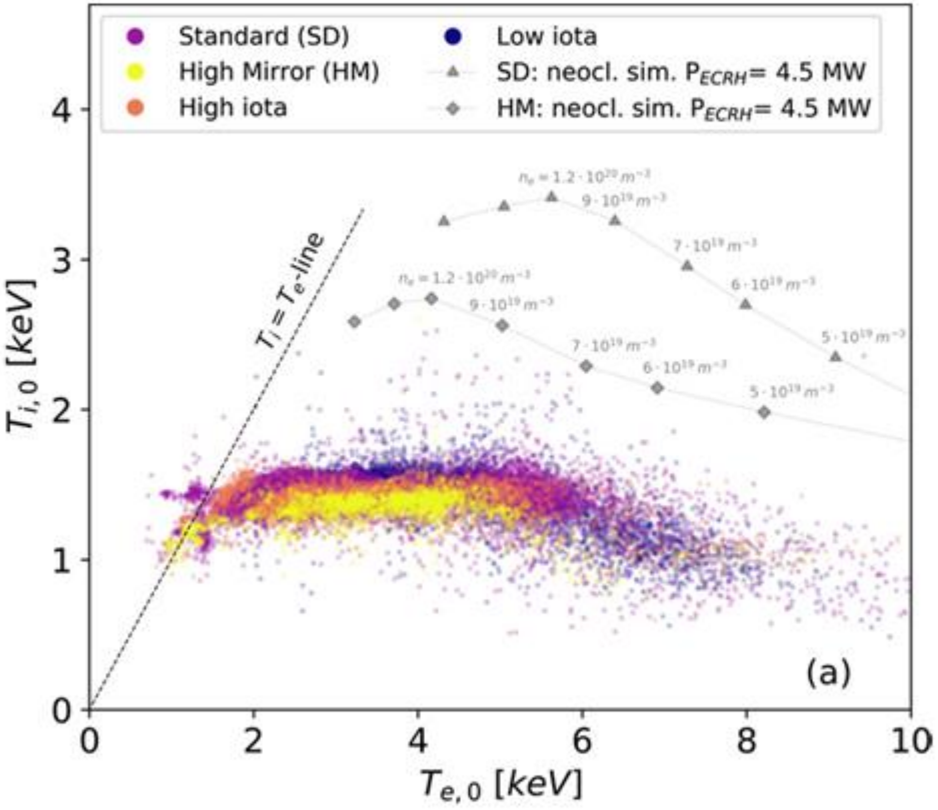
- Dominant electron heated plasmas → non-efficient equipartition → $T_i << T_e$
- "T_i clamping" caused by turbulence driven transport (TEM, ETG)

EAST (EC+LH)



[Jianwen Liu et al., AAPPS 2023]

W7-X (EC)

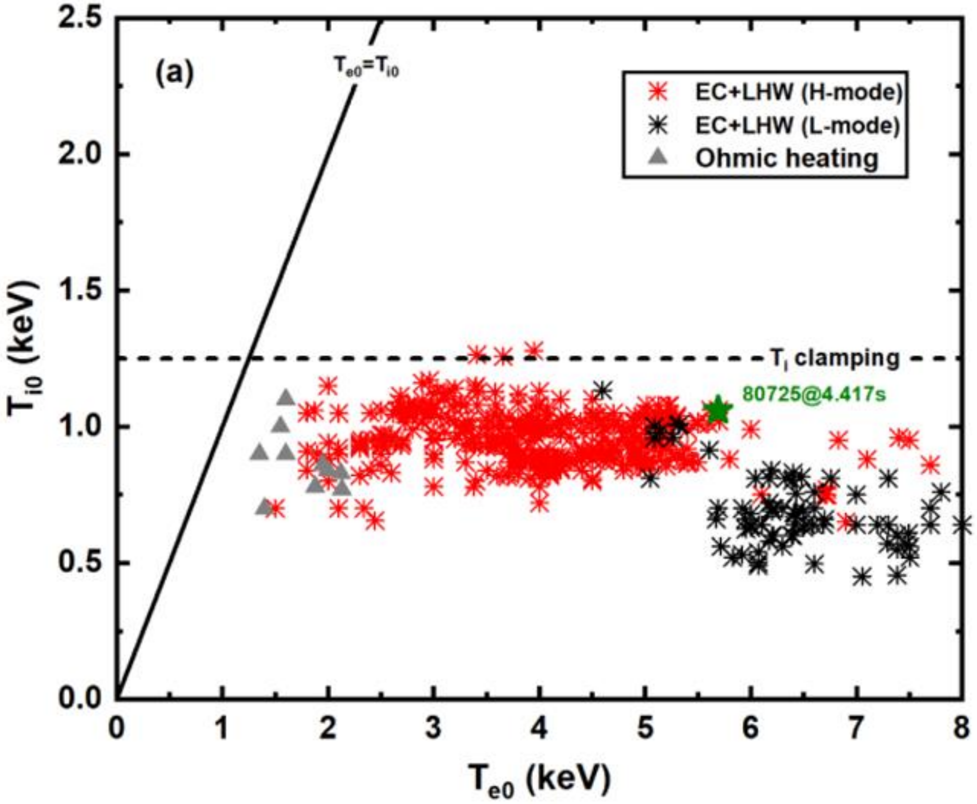


[M Beurskens et al., NF 2021]

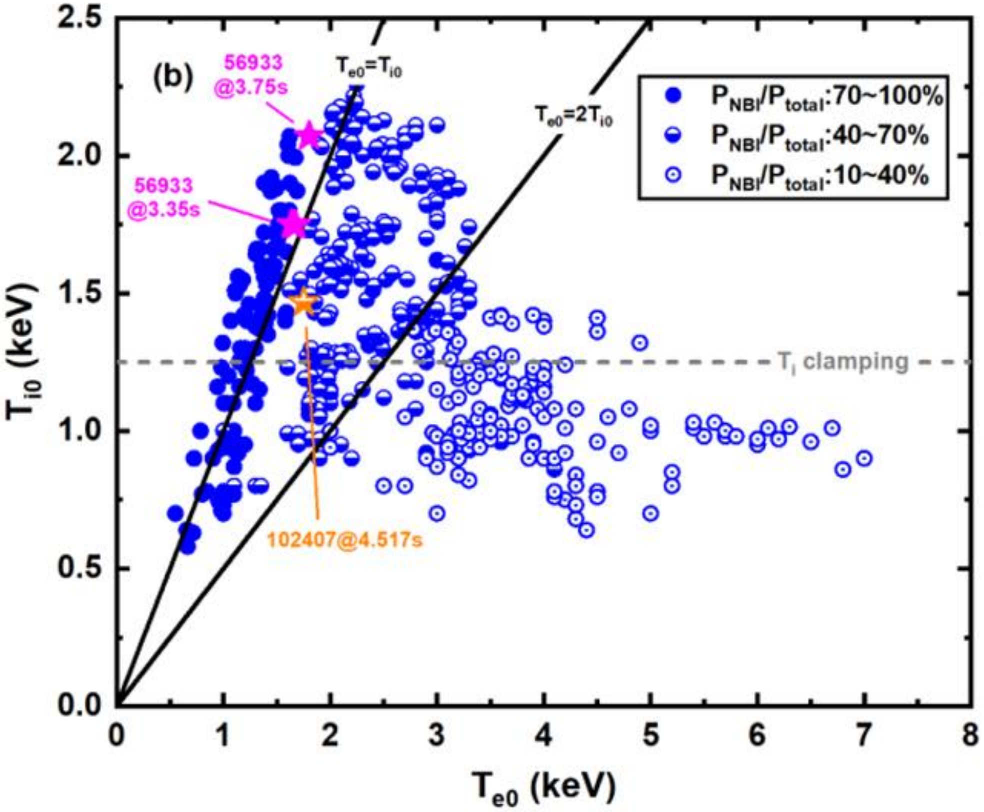
Equipartition, T_i clamping

- In EAST, $T_i \approx T_e$ can only be recovered with dominant ion heating (NBI)
- In W7-X, larger T_i obtained with pellet fuelling (but very peaked density profiles)

EAST (EC+LH)



EAST (EC+NBI)



[Jianwen Liu et al., AAPPS 2023]

NBI - recap

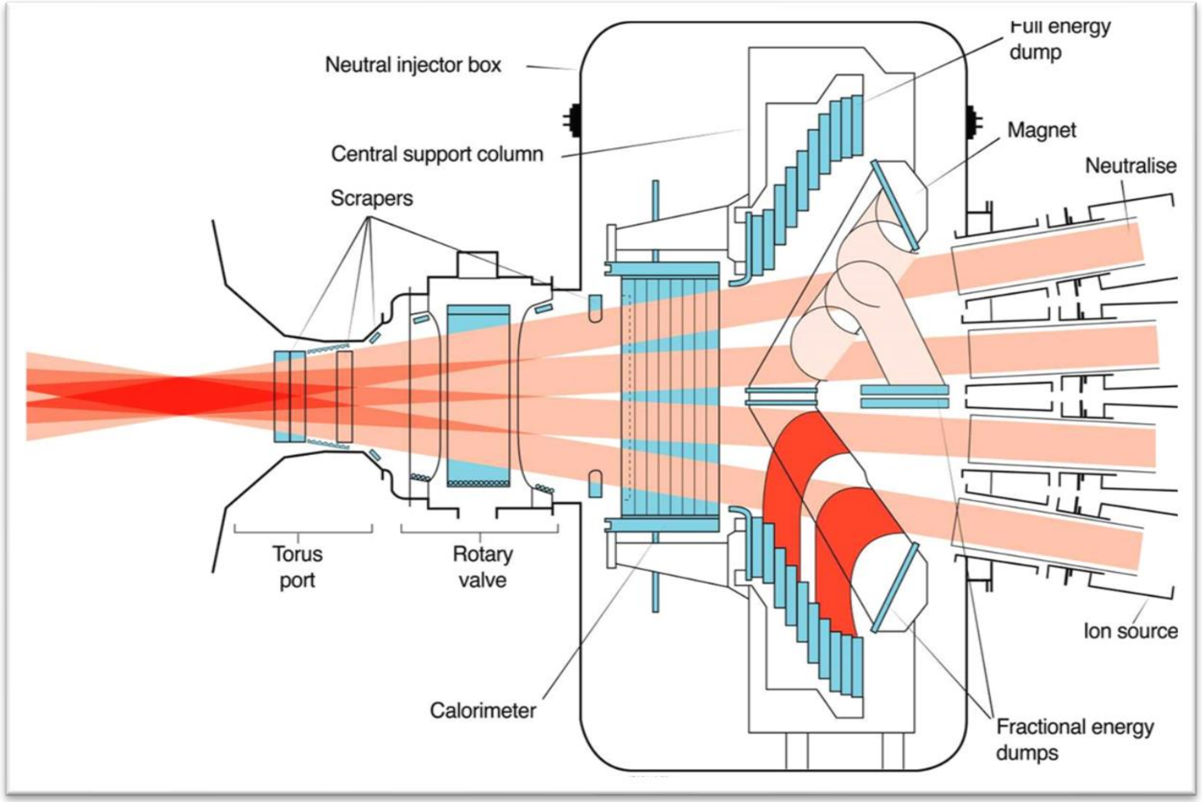
Fast neutral beam particles injected across the magnetic field and are ionised in a given region by interacting with the plasma ions, electrons or neutrons

➔ Once charged, the fast ions become trapped in magnetic field and circulate, transferring energy to plasma ions and electrons by Coulomb collisions

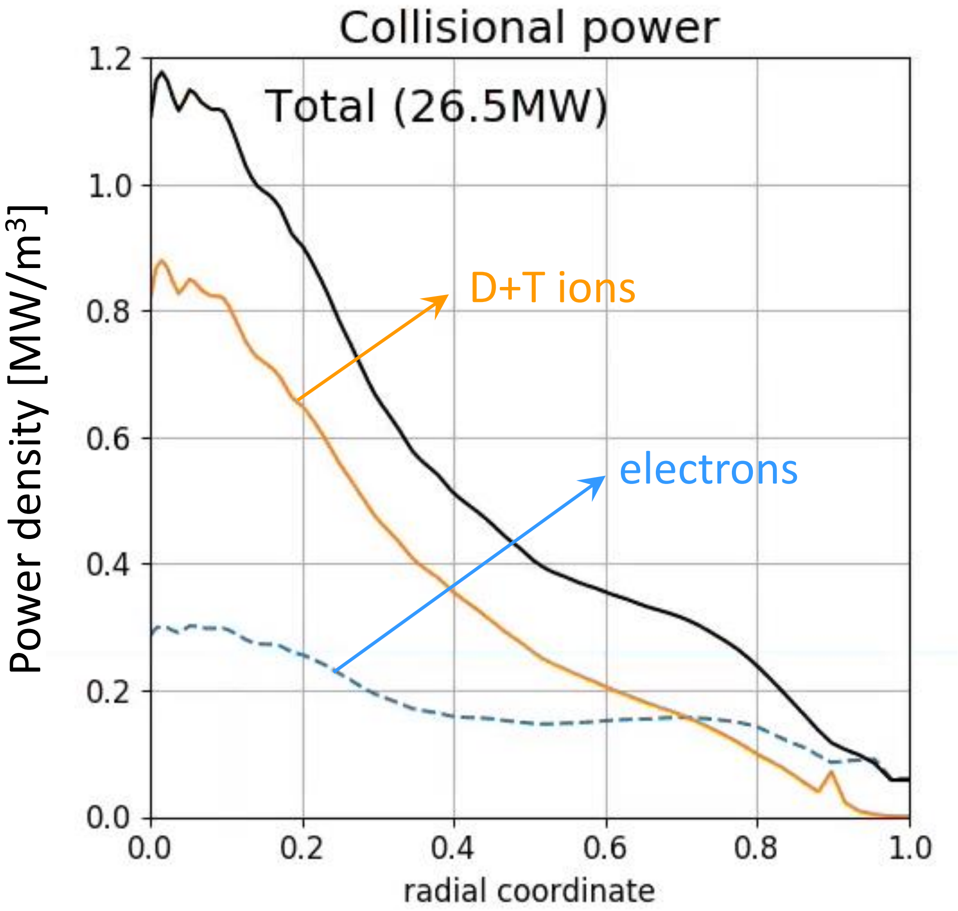
As plasma gets bigger and denser, higher energies are needed to penetrate to the core:
~ 60 keV on MAST,
~ 120 keV on JET,
~ 1 MeV on ITER

Positively- or negatively-charged ions are accelerated and neutralised before injection in the plasma.

JET system

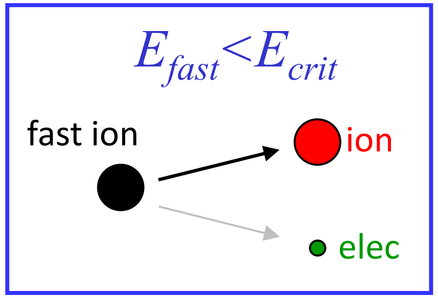


NBI: slowing-down

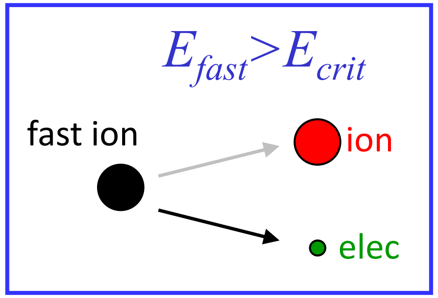


- Energy of injected NB ions is absorbed by collisions with the bulk plasma (electrons and ions)

- In JET, dominant ion heating
  (E_{beam}=120keV < E_{crit} ~ 200keV)

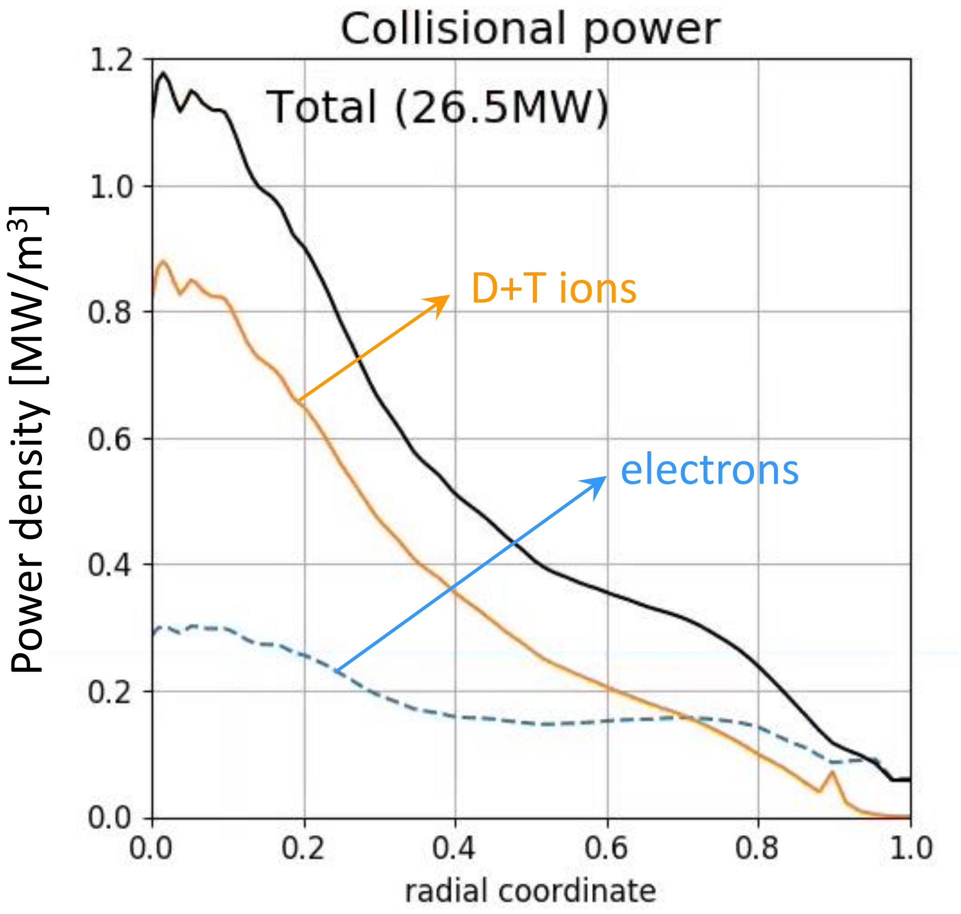


- In ITER, dominant electron heating
  (E_{beam}=1MeV > E_{crit} ~ 400keV)

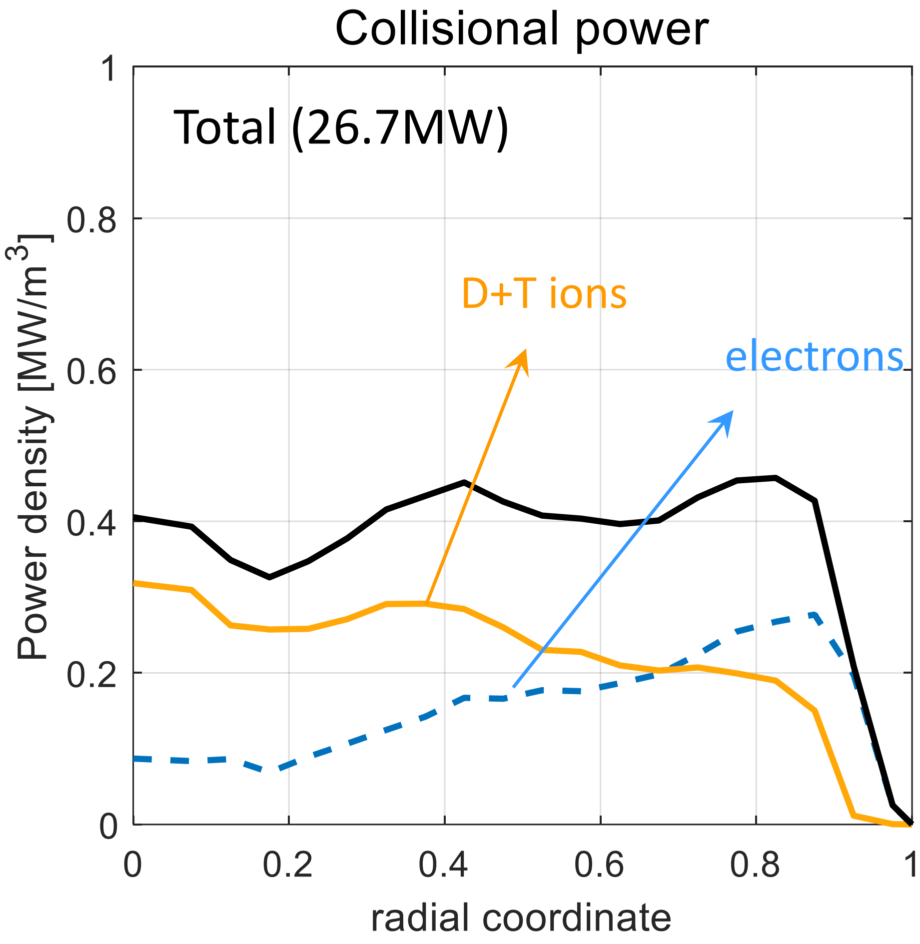


NBI: power deposition

Medium density (hybrid plasmas)

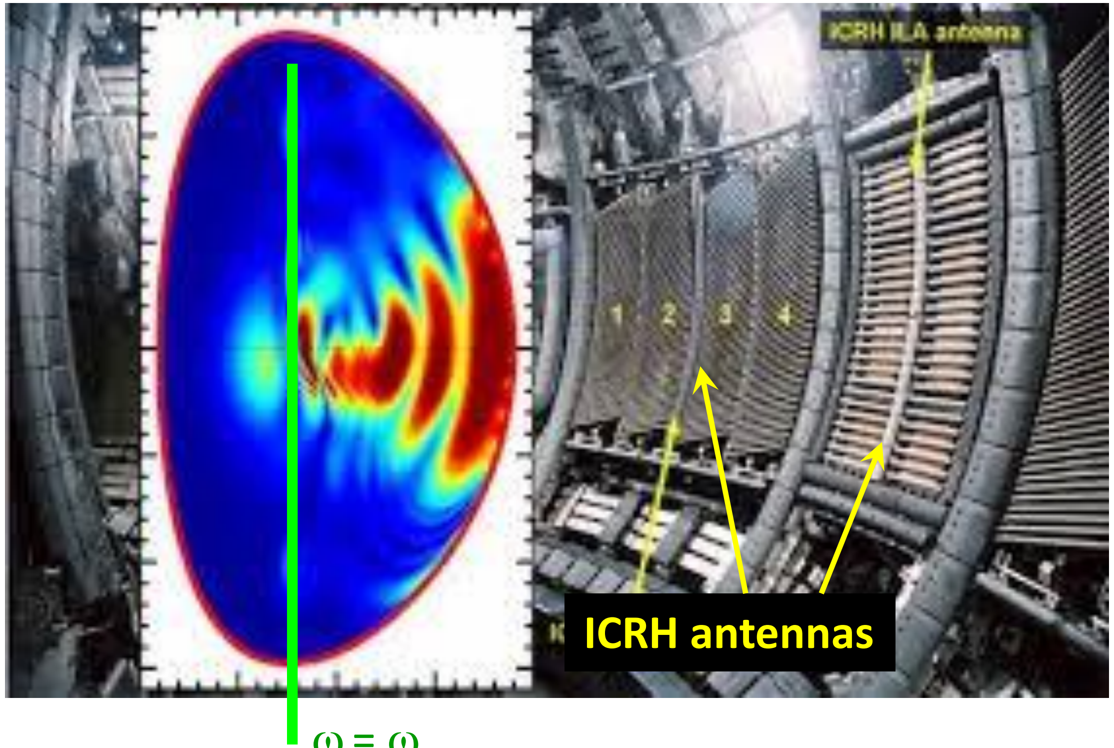


High density (baseline plasmas)



At larger plasma density (baseline plasmas), the beam penetration is less efficient leading to flatter deposition and lower core ion heating

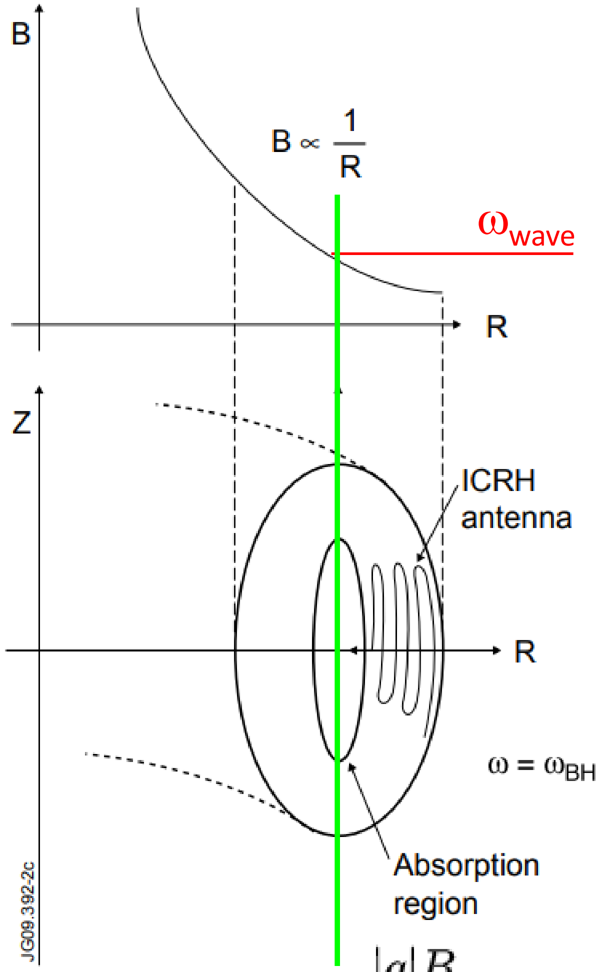
ICRH -recap



- A fast wave is launched into the plasma and propagates towards the centre
- It is absorbed in a narrow region where $\omega_{wave} = \omega_c$
- Two possibilities:

(1) Wave resonates with a minority species → collisions

(2) Wave resonates directly with one of the fuel ions



$$\omega_c = \frac{|q|B}{m}$$

[Jacquet et al, IAEA-FEC 2023]

ICRH: Coupling vs. absorption

Coupling

→ Wave tunnelling through evanescence layer:

- antenna-plasma distance
- SOL density
- frequency, $k_{//}$ spectrum (antenna design, phase)

Heating efficiency

→ Wave absorption in the core:

- ICRF scenario (IC layer, minority conc., $k_{//}$)
- Core plasma profiles

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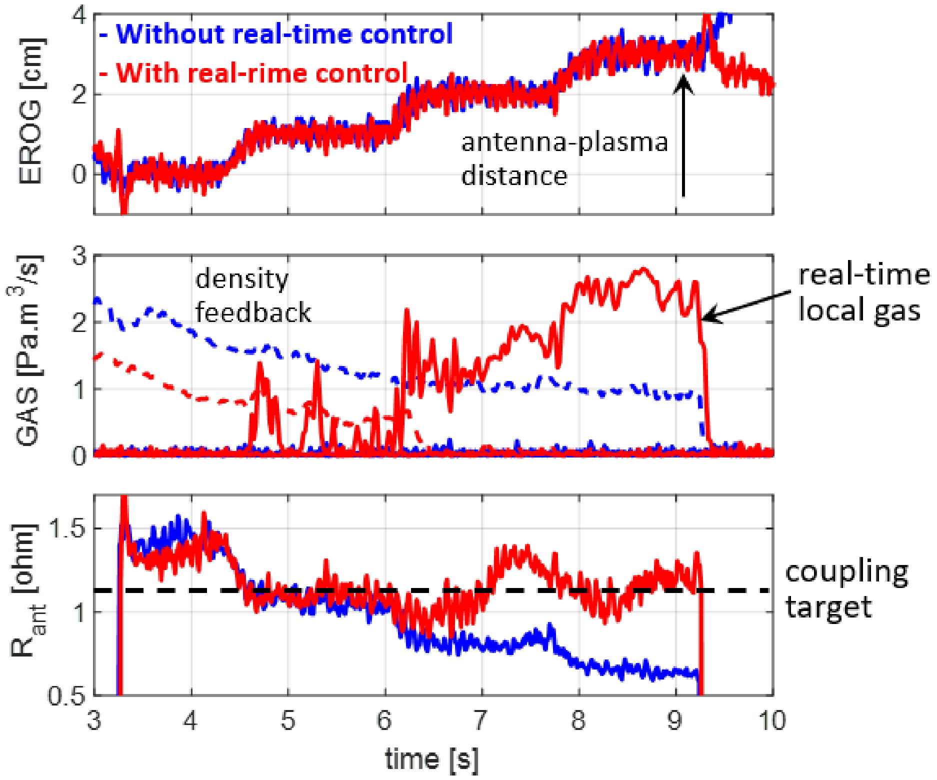
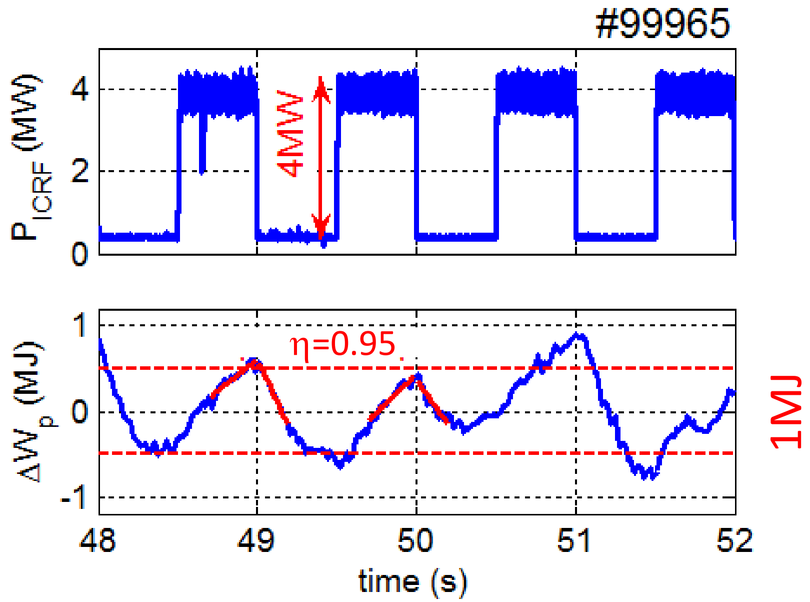
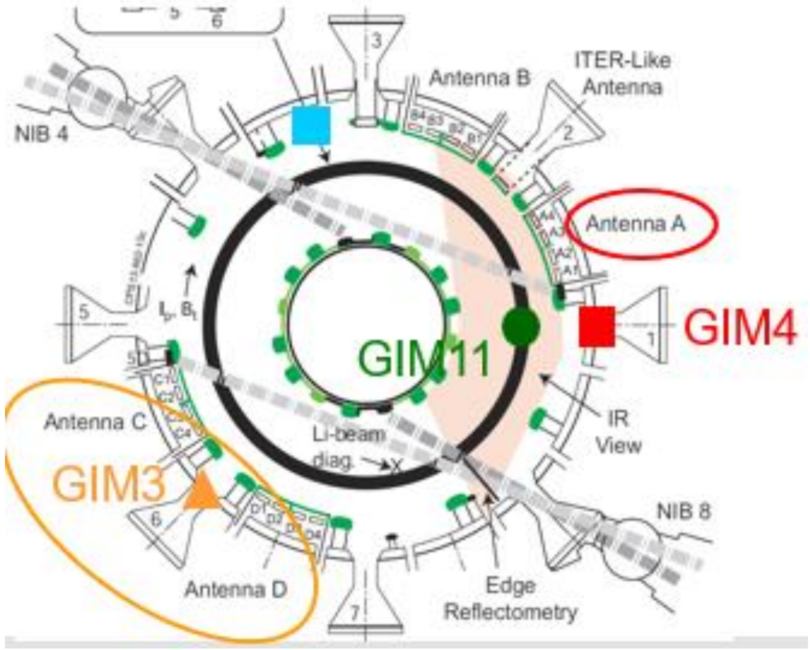


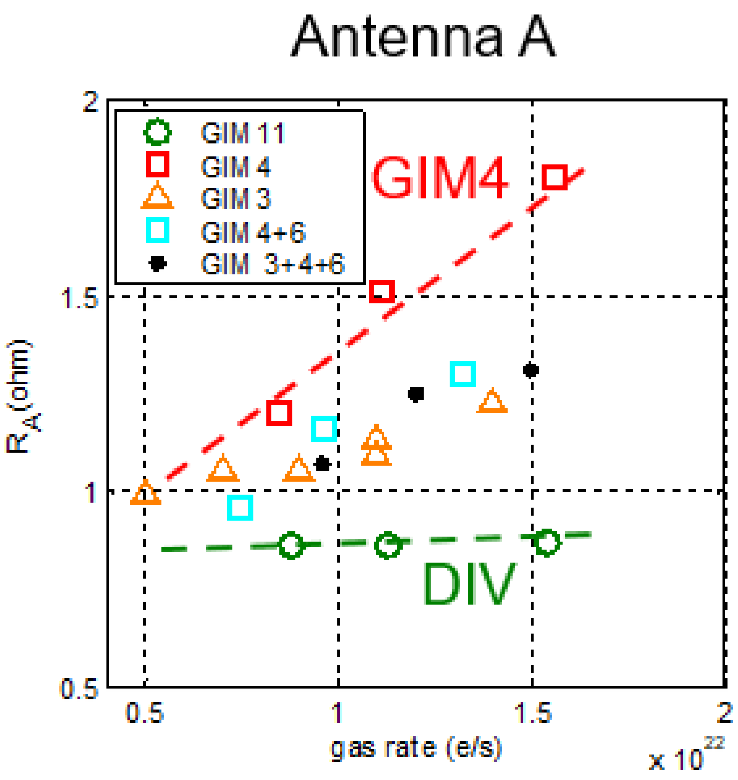


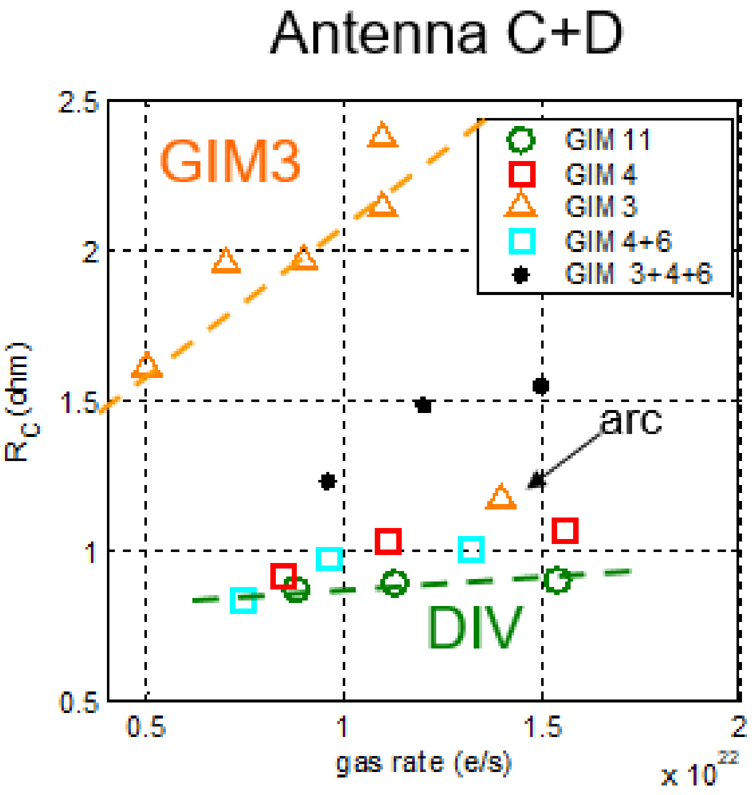
ICRH: Coupling vs. absorption

Coupling

→ Local gas injection = larger RF coupling







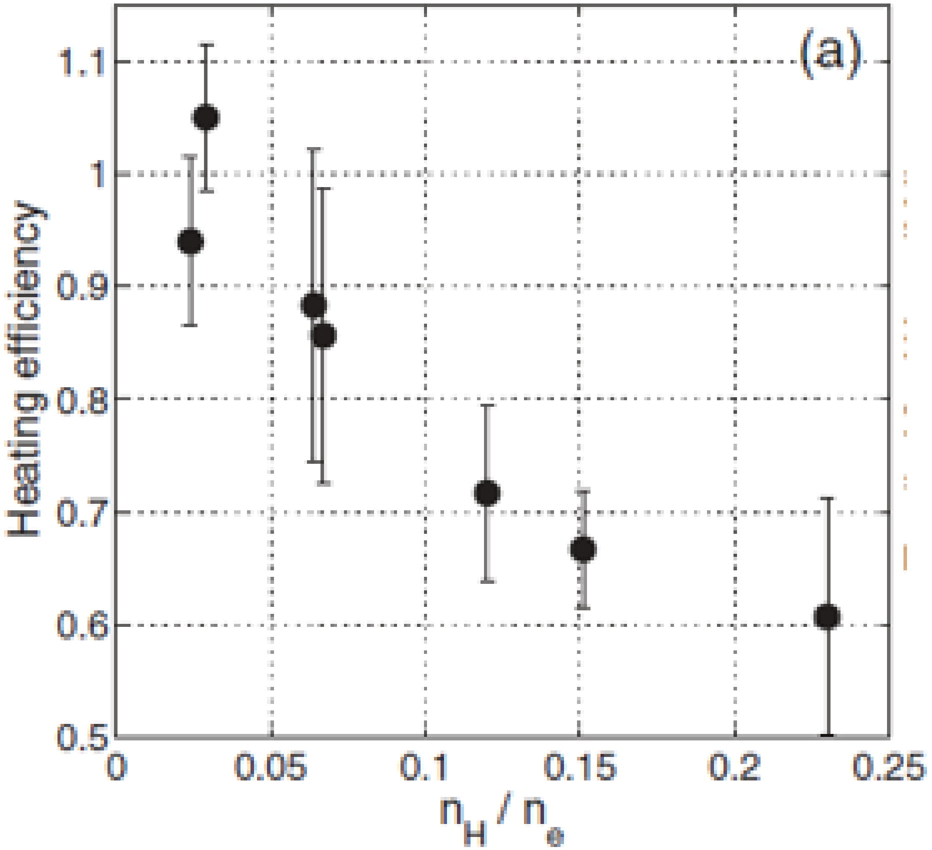
E. Lerche *et al, Journal of Nucl. Materials* 463 (2015) 634–639

Ph. Jacquet *et al* 2016 Nucl. Fusion 56 046001

ICRH: Coupling vs. absorption

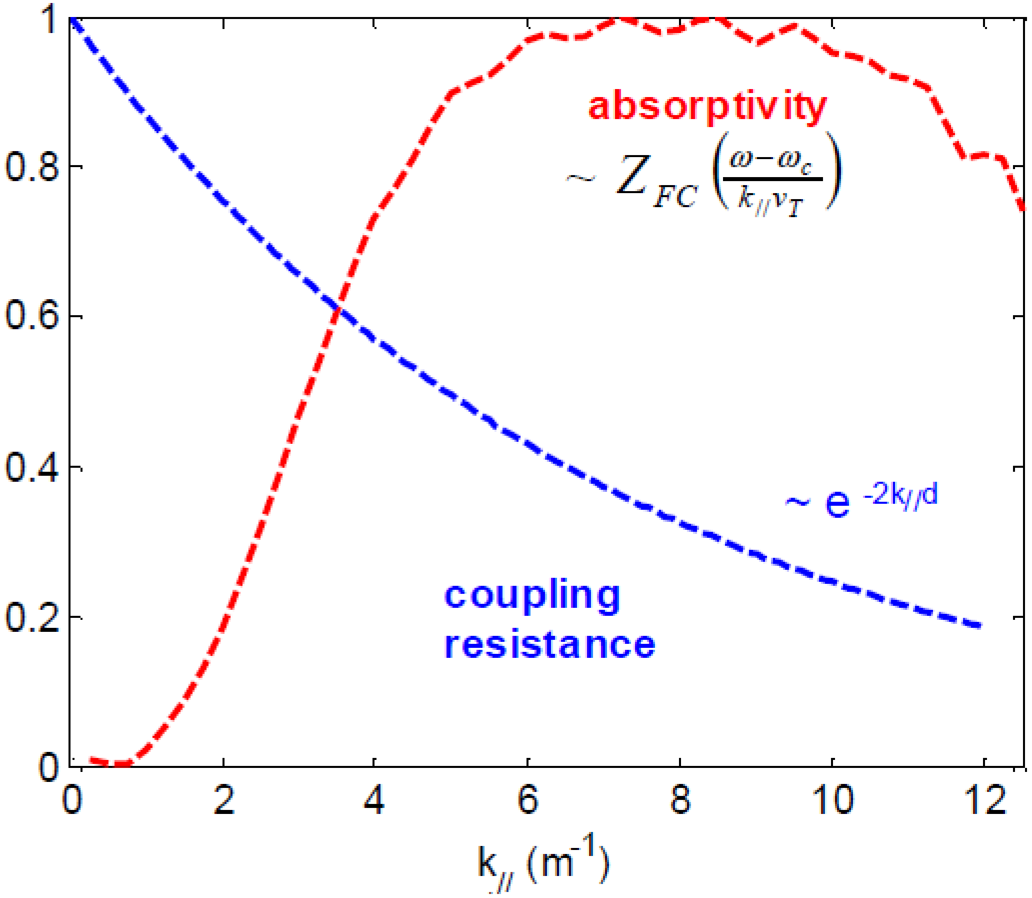
Heading efficiency

→ Minority concentration has strong impact on ICRH heating efficiency

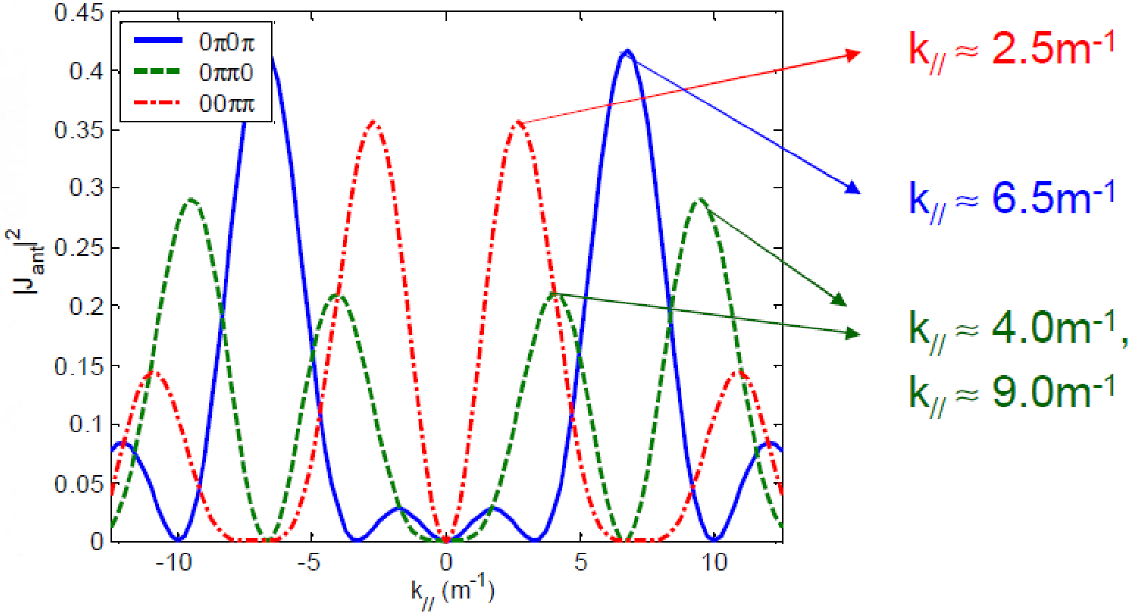


ICRH: Coupling vs. absorption

ICRH performance: trend on $k_{//}$



JET A2 spectra (vacuum)



→ Lower $k_{//}$: ↑ coupling
↓ efficiency

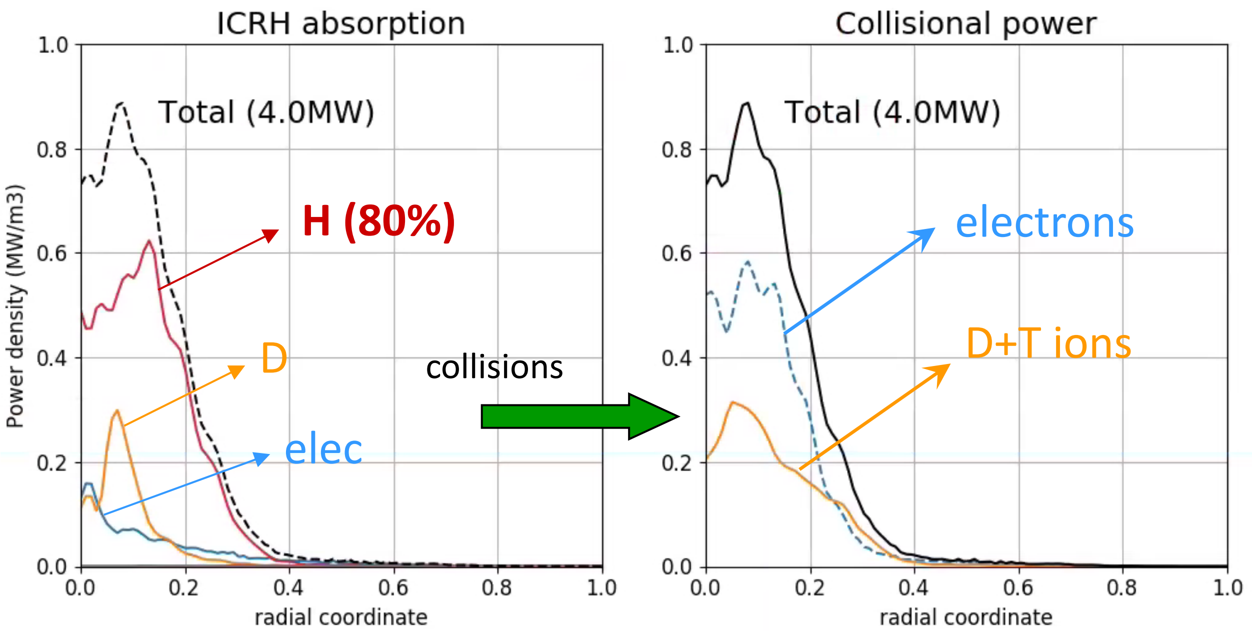
→ Higher $k_{//}$: ↓ coupling
↑ efficiency

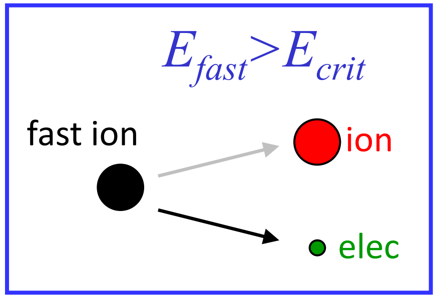
ICRH: minority heating = slowing-down

Hydrogen minority heating (H = 3%):

→ Dominant elec. heating

$E_{fast} > E_{crit}$ (mass=1)

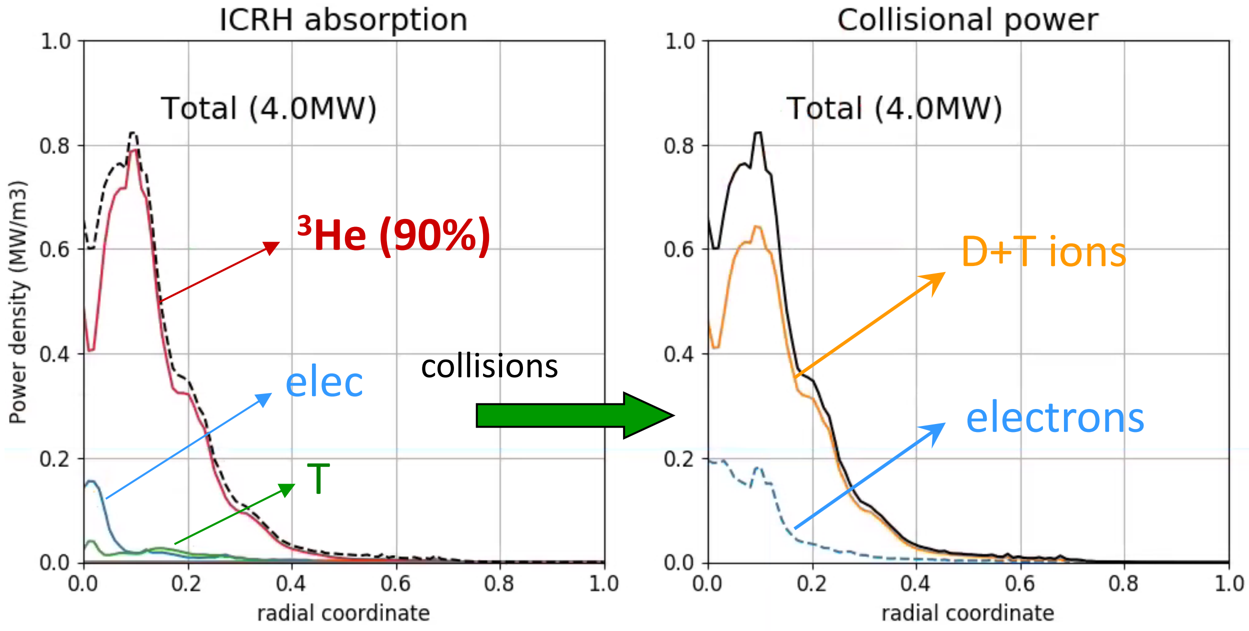


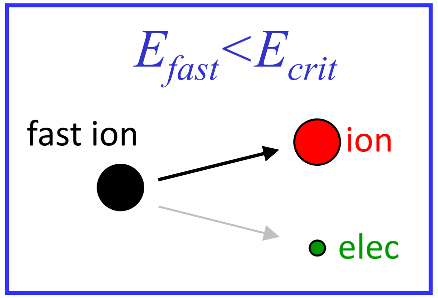


Helium-3 minority heating (^{3}He = 3%):

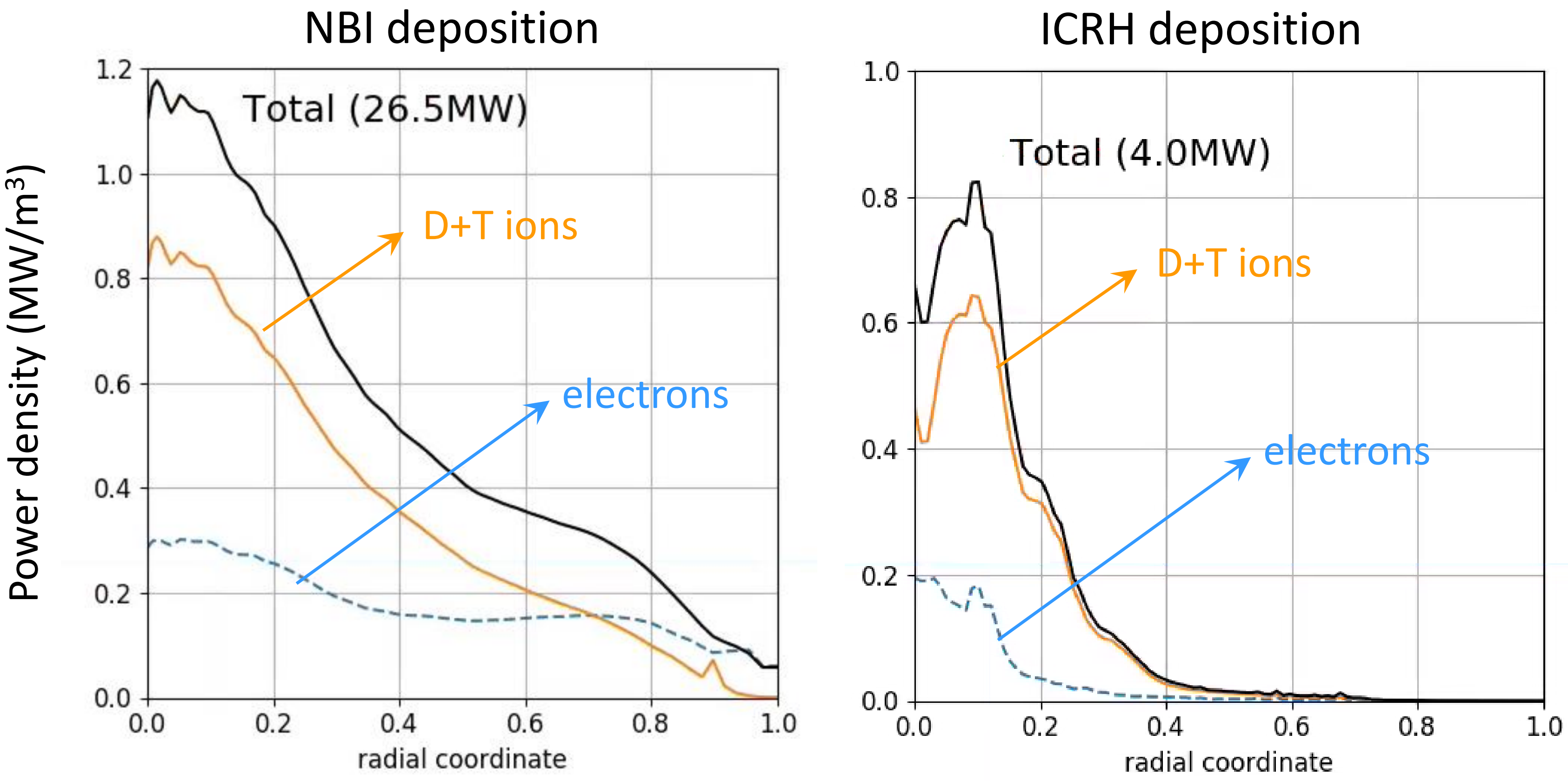
→ Dominant ion. heating

$E_{fast} < E_{crit}$ (mass=3)





ICRH is very peaked compared to NBI heating



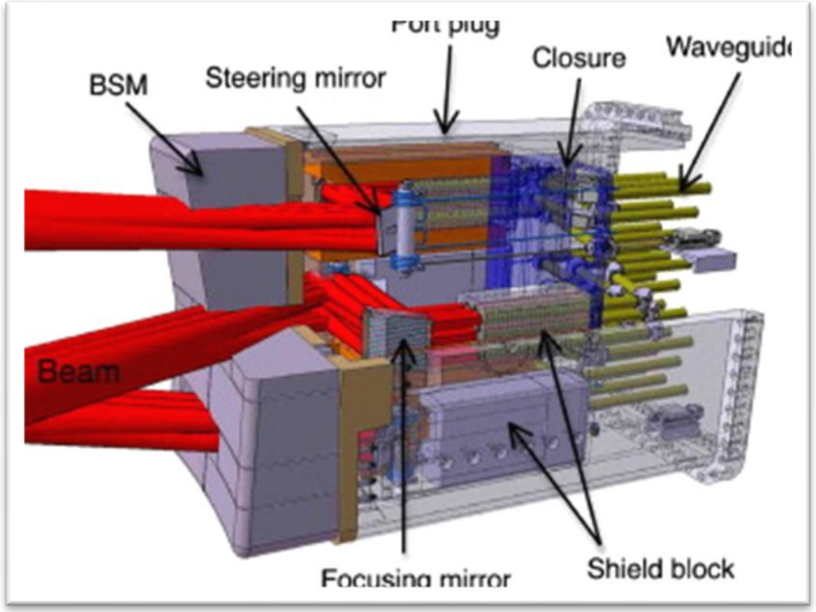
- Central power densities are comparable despite much less ICRH power
- Expect strong impact of ICRH on the core plasma temperatures

Electron cyclotron resonance heating (ECRH)

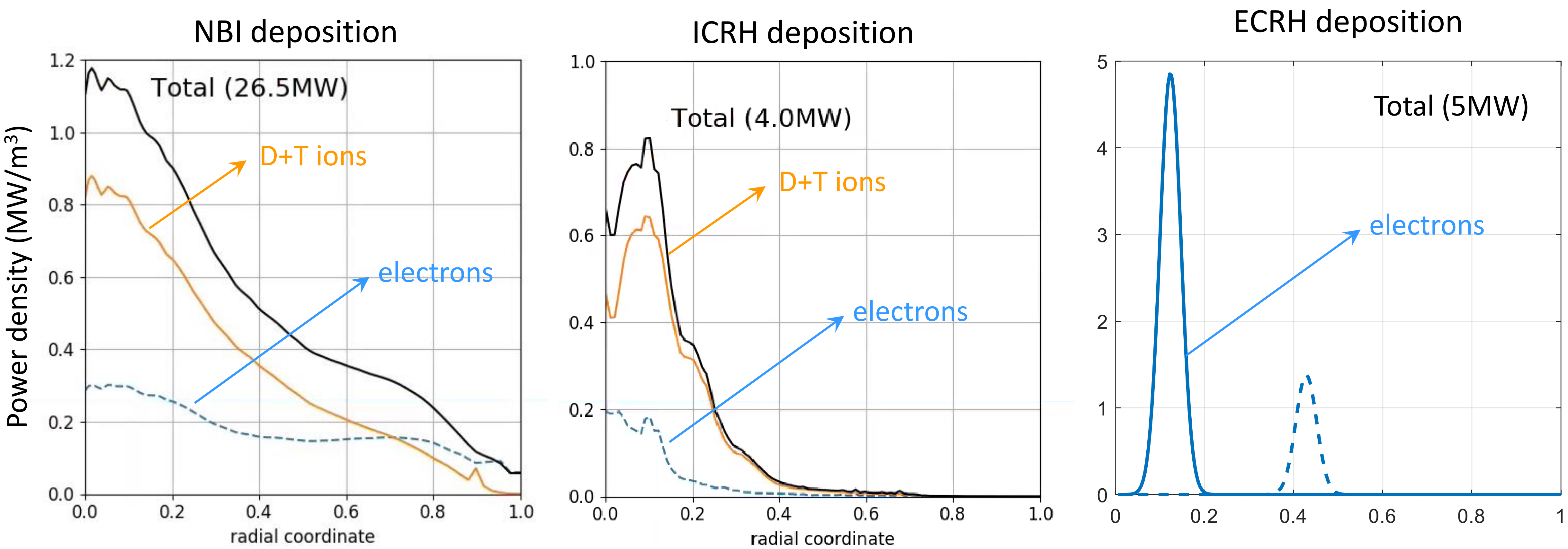
- Needs a small port.
- Can use mirrors to point the beam where you want it = movable parts in tokamak
- Gyrotrons are delicate beasts. The ITER design includes development of a 1 MW gyrotron operating at 170 GHz with a pulse duration of more than 500 s.
- ➔ under development in Europe, India, Japan and Russia

- 40-60 gyrotrons in ITER (up to 60MW)

ITER midplane launcher



ECRH is very peaked compared to anything


NBI deposition

ICRH deposition

ECRH deposition

- ECRH power densities are far higher than anything else (narrow deposition)
- Smaller p_{dens} off-axis (volume effect)

Heating system functions & capabilities

	ICRH	ECRH	NBI	
Electron heating	yes	yes	yes	
Ion heating	yes	no	yes (small device) no (large device)	
Current drive	no	yes	yes	
MHD control	no	yes	no	
Sawtooth control	yes	yes	no	EC more efficient
Fast ion studies	yes	no	yes	
Breakdown	yes	yes	no	
Wall conditioning / T removal	yes	yes	no	IC more efficient
Boronization	Yes*	no	no	

Note: The H&CD system capabilities depend on the system specifics and plasma conditions

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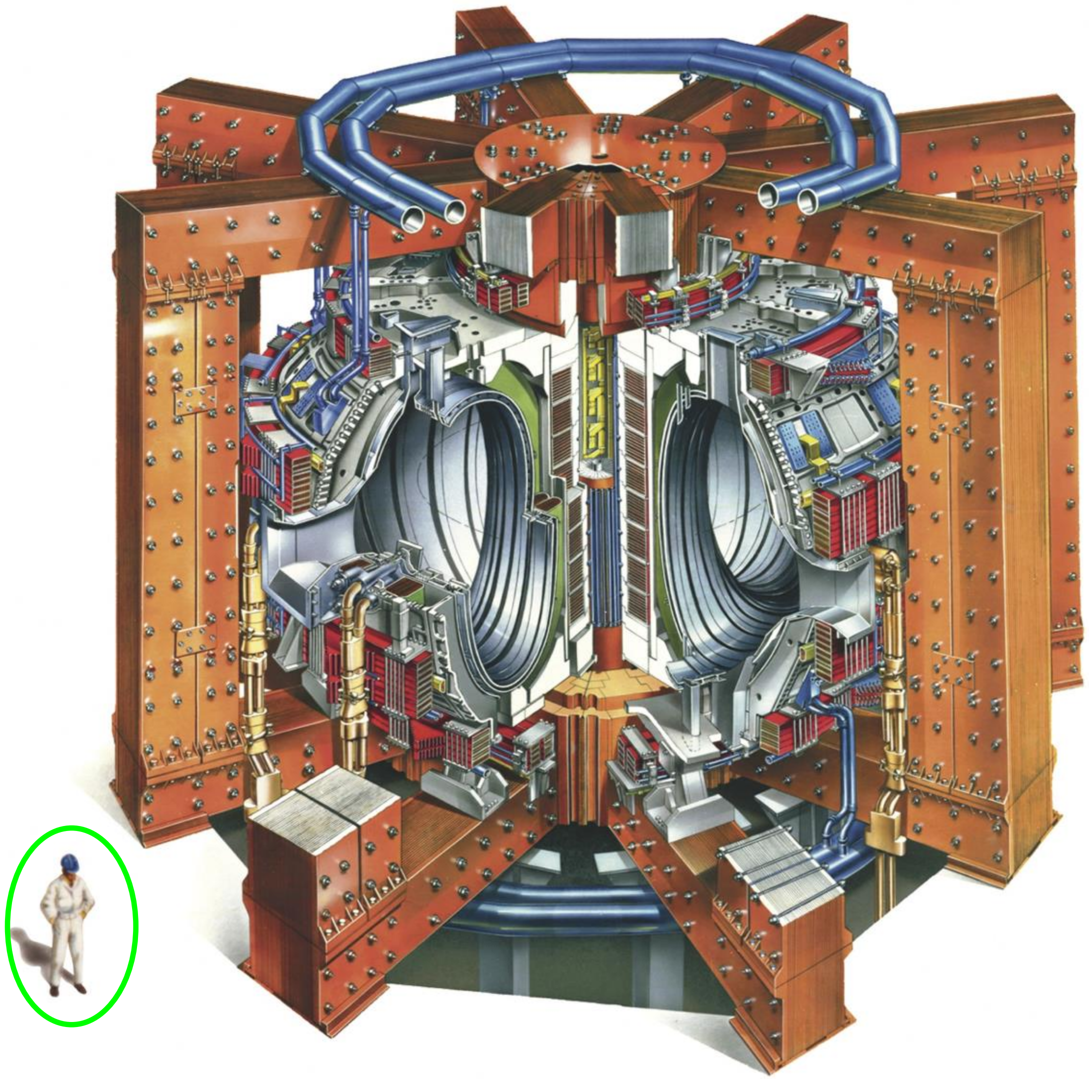
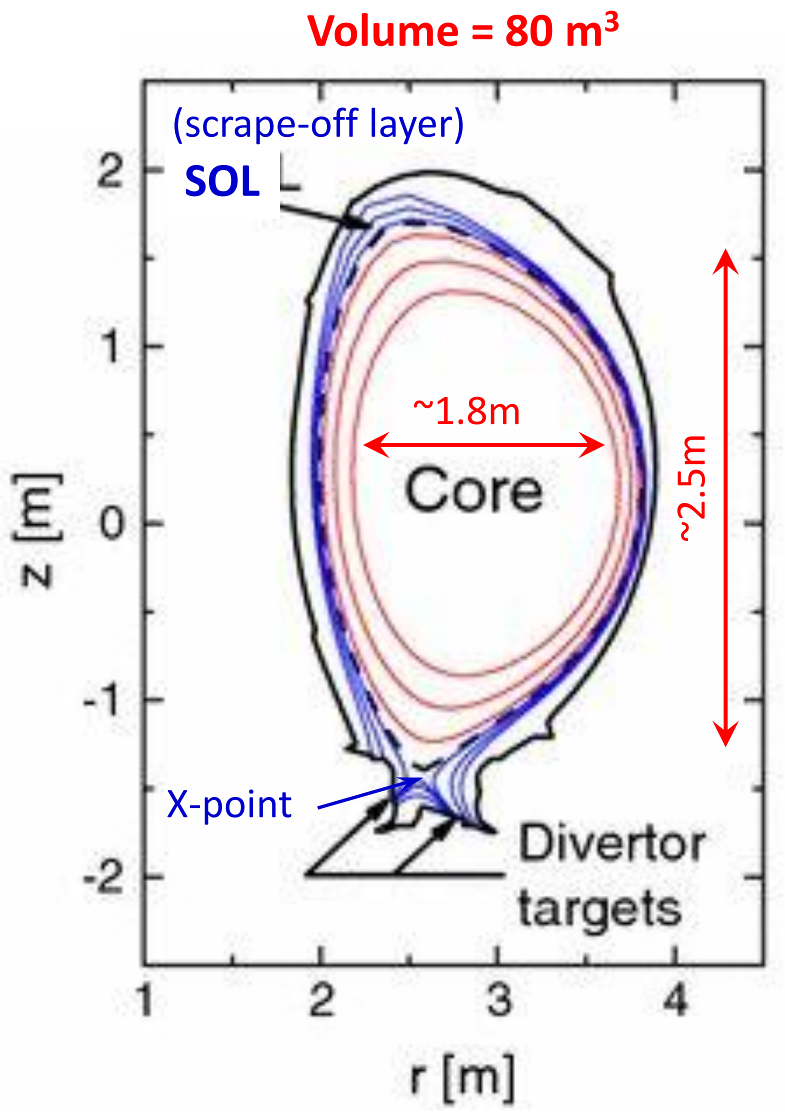
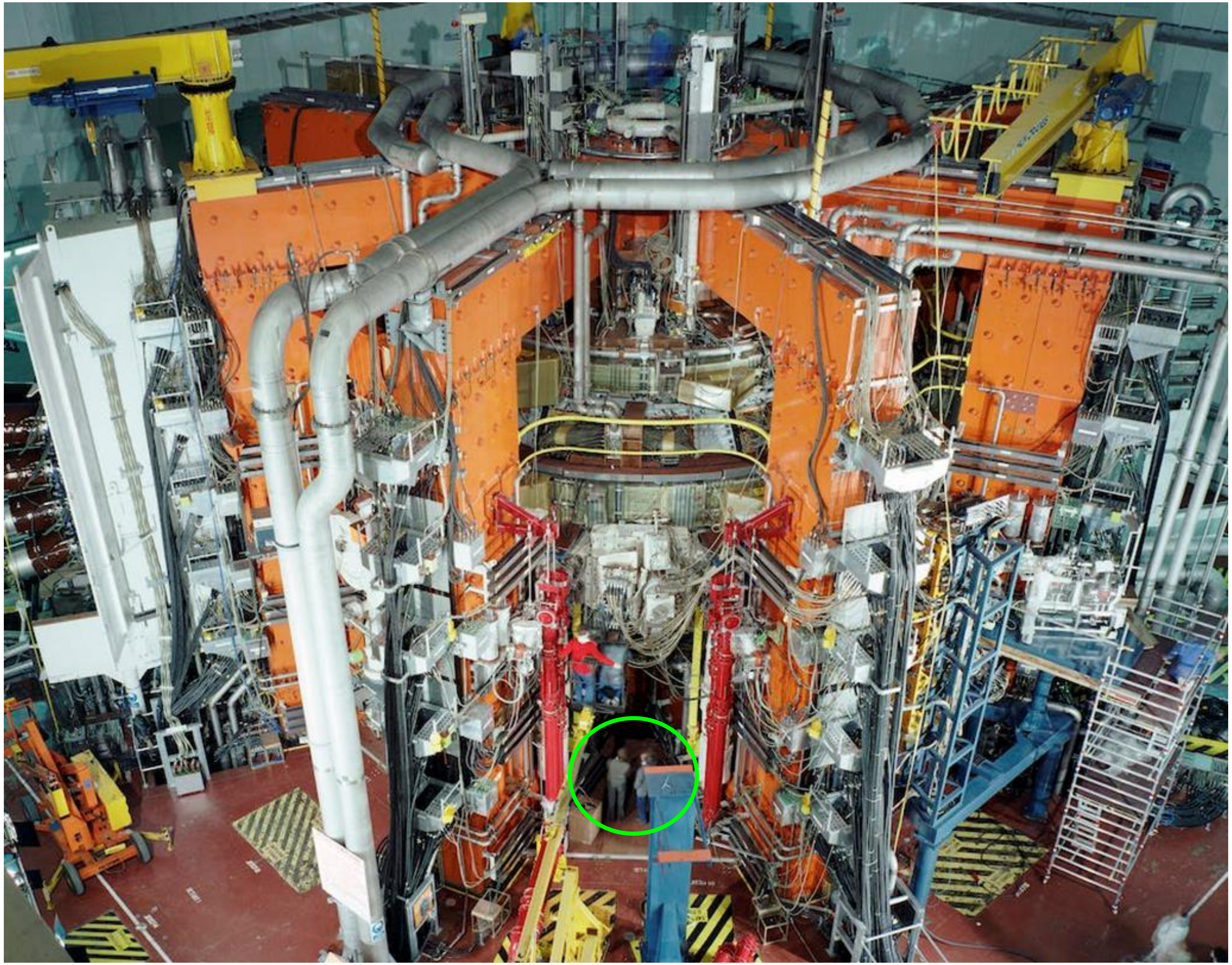
The JET tokamak (Culham, UK)



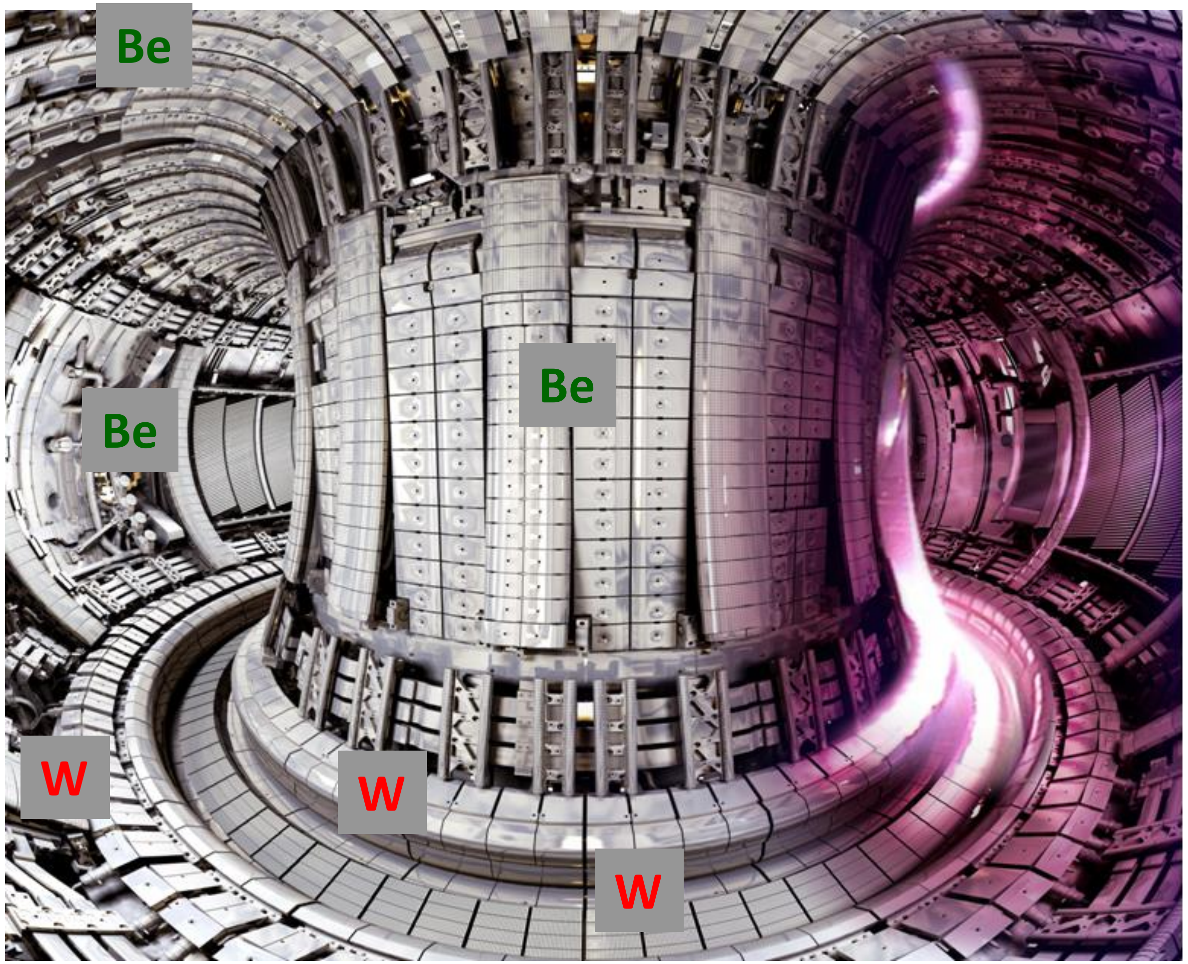


- Magnetic field = 3-4 T
- Plasma current = 2-4MA

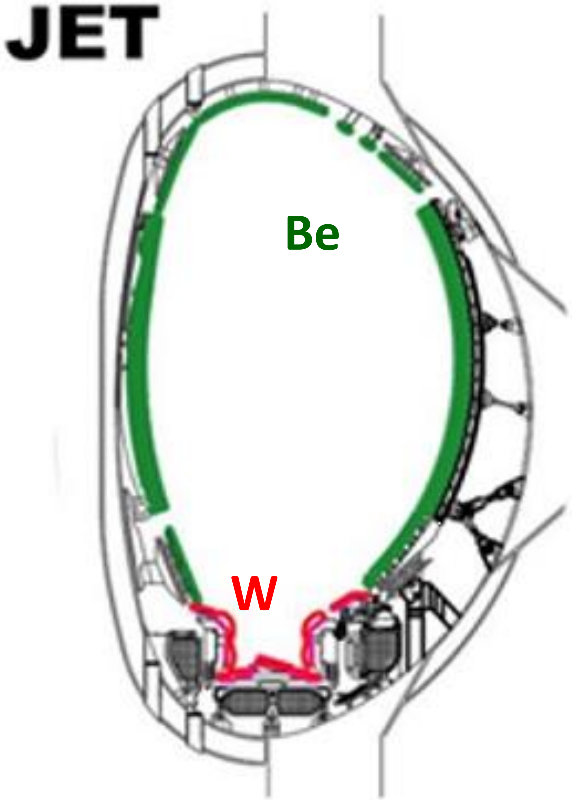
The JET tokamak



The ‘ITER-like’ wall



[Pamela, JNM2007]
[Matthews, Phys.Scr. 2007]



- New challenges appeared with the JET Be/W metallic wall:
  - Plasma-wall interaction with Be (real-time protection for heat loads)
  - Divertor sputtering: If W contamination in the plasma core → large radiation

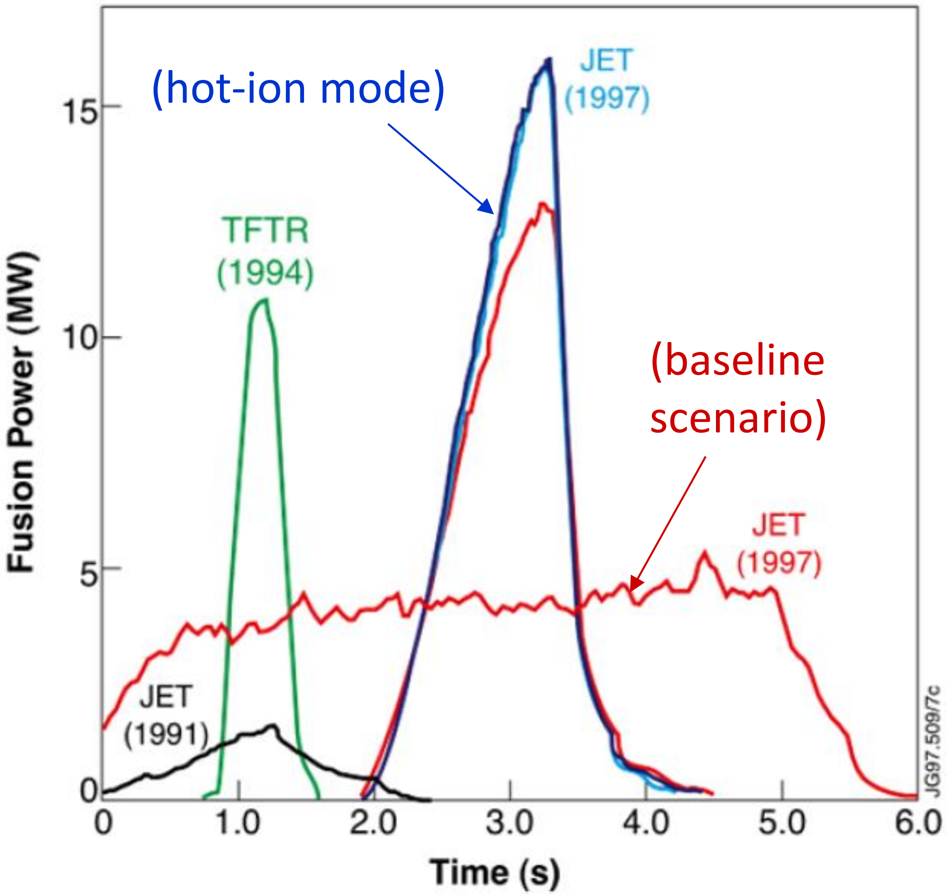
Past D-T experiments in tokamaks

25 years since last D-T experiments

- 1991 (PTE - JET)
- 1994-96 (TFTR – Princeton USA)
- 1997 (DTE1 on JET)

Main results:

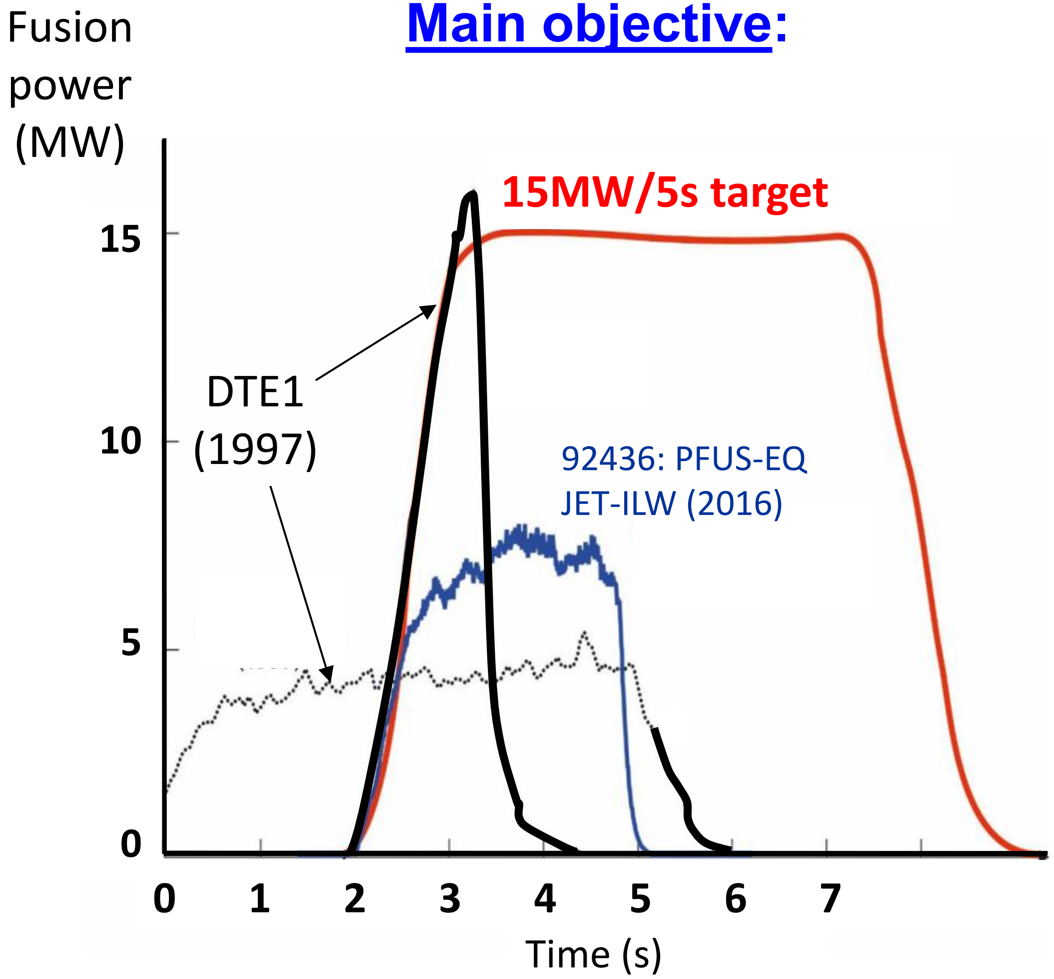
- D-T fusion production (14MeV neutrons)
  - Up to 16 MW transiently
  - P_{fus}~4MW for 4sec
- Plasma physics effects linked to use of D-T mixture (isotope effects)



DTE1: M. Keilhacker et al., 1999 Nucl. Fusion
J. Jacquinot et al., 1999 Nucl. Fusion

Deuterium-Tritium Experiments in JET-ILW

Main objective:



L. Horton and A. Donné, 2016 Fus. Eng. and Design

X. Litaudon et al., SOFT 2016
E. Joffrin et al., IAEA 2018

Upgrades (since DTE1):

- ITER-like wall (ILW)
- Power upgrade
- Diagnostics upgrade

Two D-T campaigns:

- DTE2 (2021) and DTE3 (2023)

Emphasis of was on sustained fusion power, not on peak performance:

- Initial target = 15MW fusion for 5sec

- Final target = 10MW fusion for 5sec

Best stationary pulses achieved in DTE2

→ **World record** on fusion power / energy in tokamaks achieved in DTE2

50:50 D:T JET-C baseline plasma

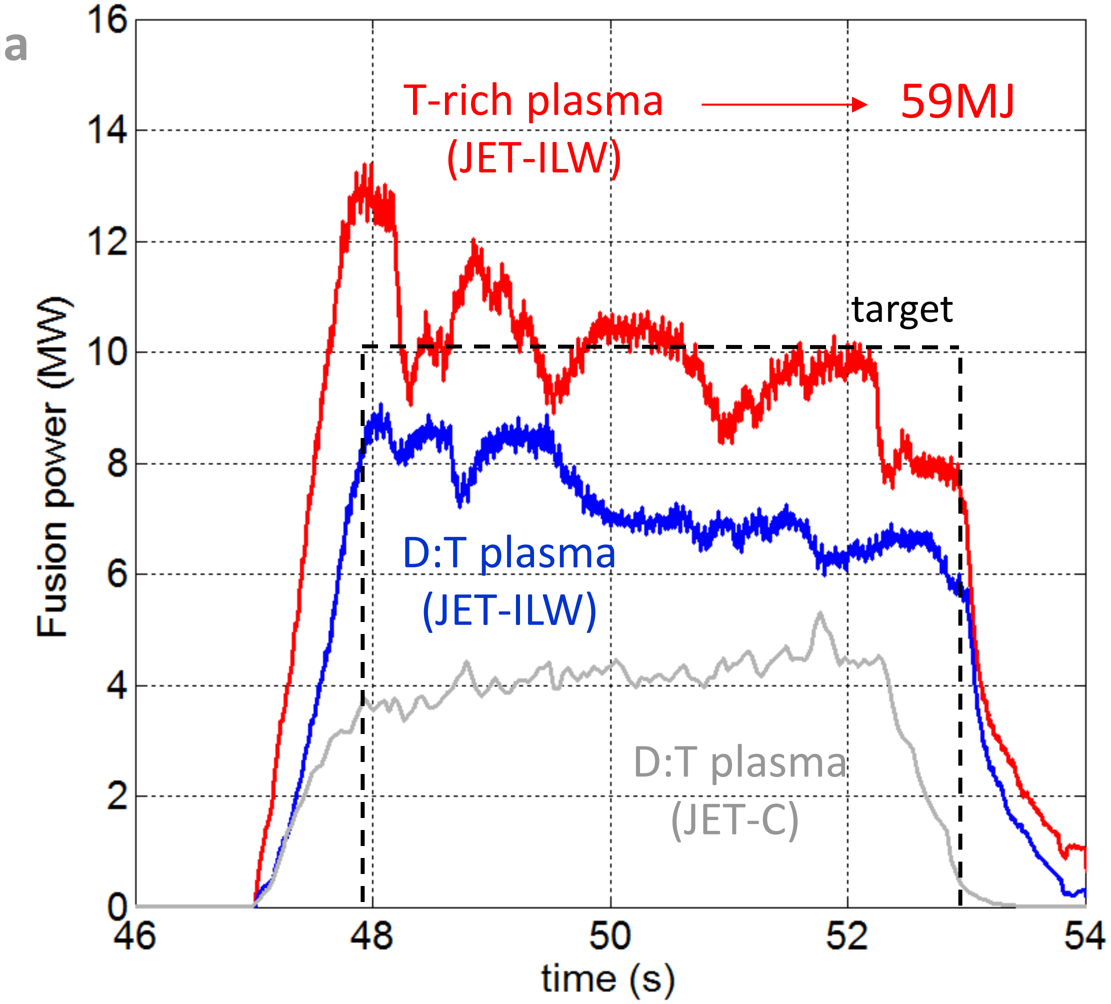
- P_{IN}=24MW
- P_{fus} = 4MW (4s)
- $Q_{fus}= P_{fus}/P_{IN}$ = 0.17 (4s)

50:50 D:T hybrid plasma

- P_{IN}=33MW
- P_{fus} = 7.4MW (5s)
- $Q_{fus}= P_{fus}/P_{IN}$ = 0.23 (5s)

T-rich hybrid plasma

- P_{IN}=33MW
- P_{fus} = 10MW (5s)
- $Q_{fus}= P_{fus}/P_{IN}$ = 0.3 (5s)



(Oscillations due to non-constant input power)

Highest fusion power pulses in DTE2

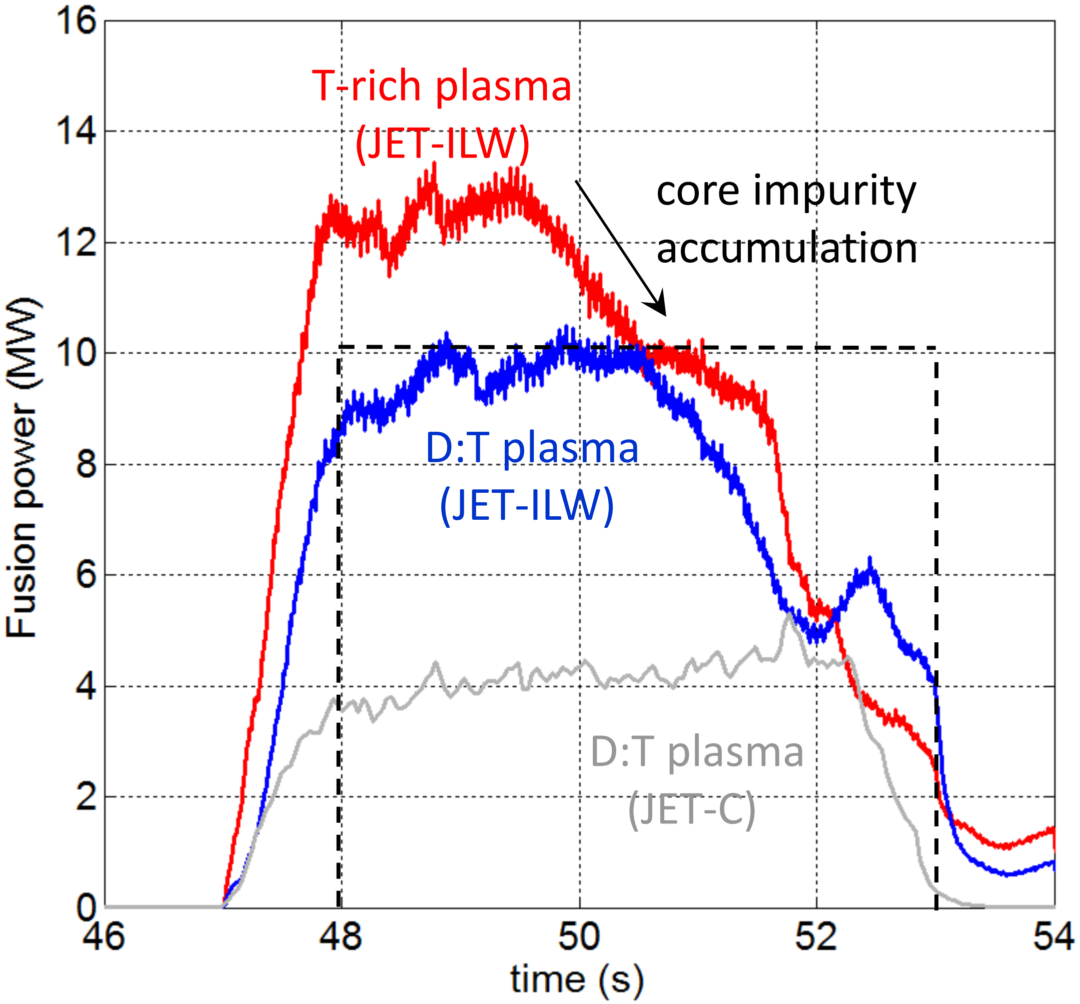
→ Some discharges **exceeded P_{fus}=10MW** for a few seconds but could not be maintained for 5s (auxiliary power reliability, core impurity accumulation, …)

50:50 D-T hybrid plasma

- P_{IN}=33MW
- P_{fus} = 10MW (2s)
- Q_{fus}= P_{fus}/P_{IN} = 0.3 (2s)

T-rich hybrid plasma

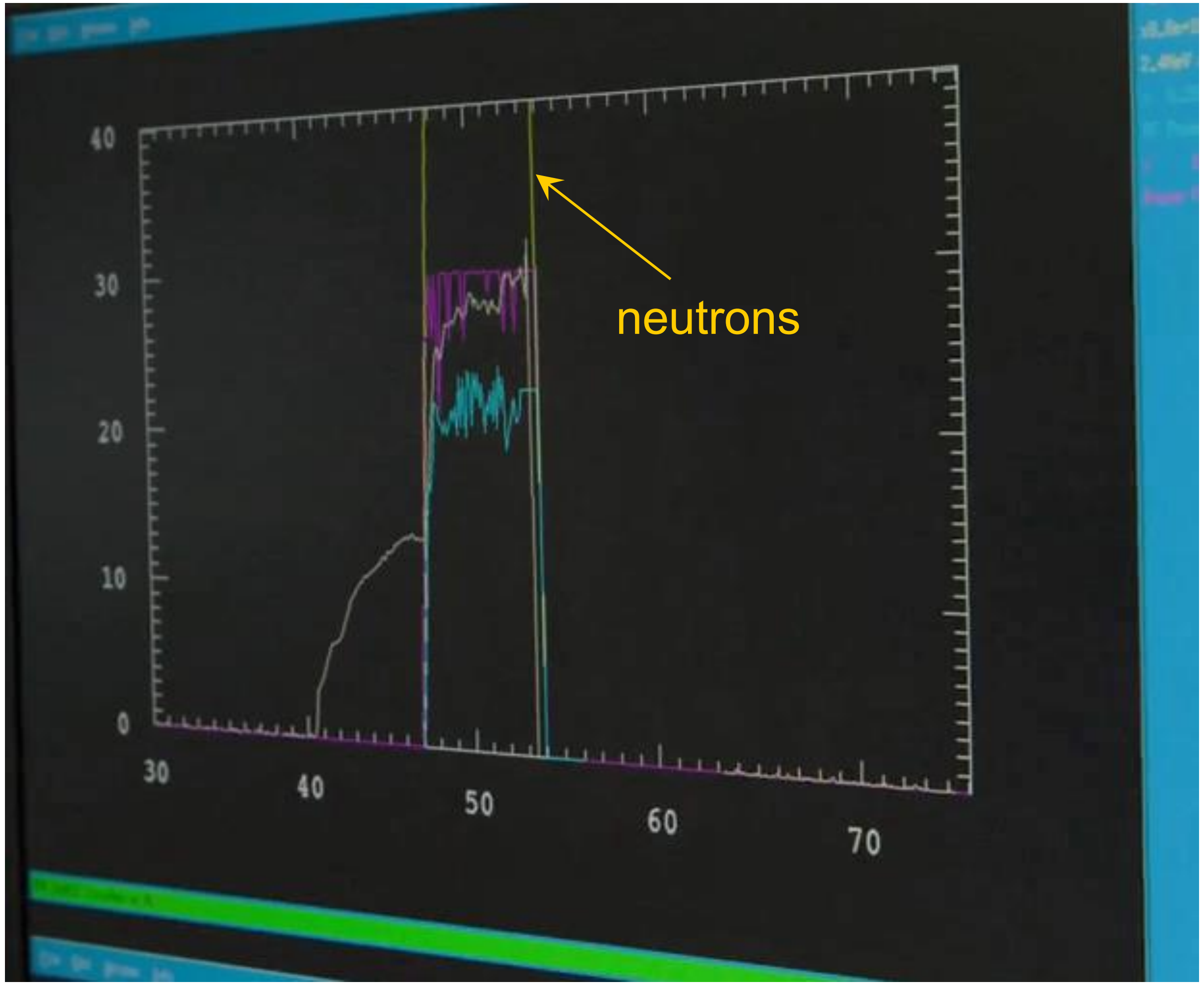
- P_{IN}=33MW
- P_{fus} = 12.5MW (2s)
- Q_{fus}= P_{fus}/P_{IN} = 0.38 (2s)



Additional achievements in DTE3 campaign (2023)

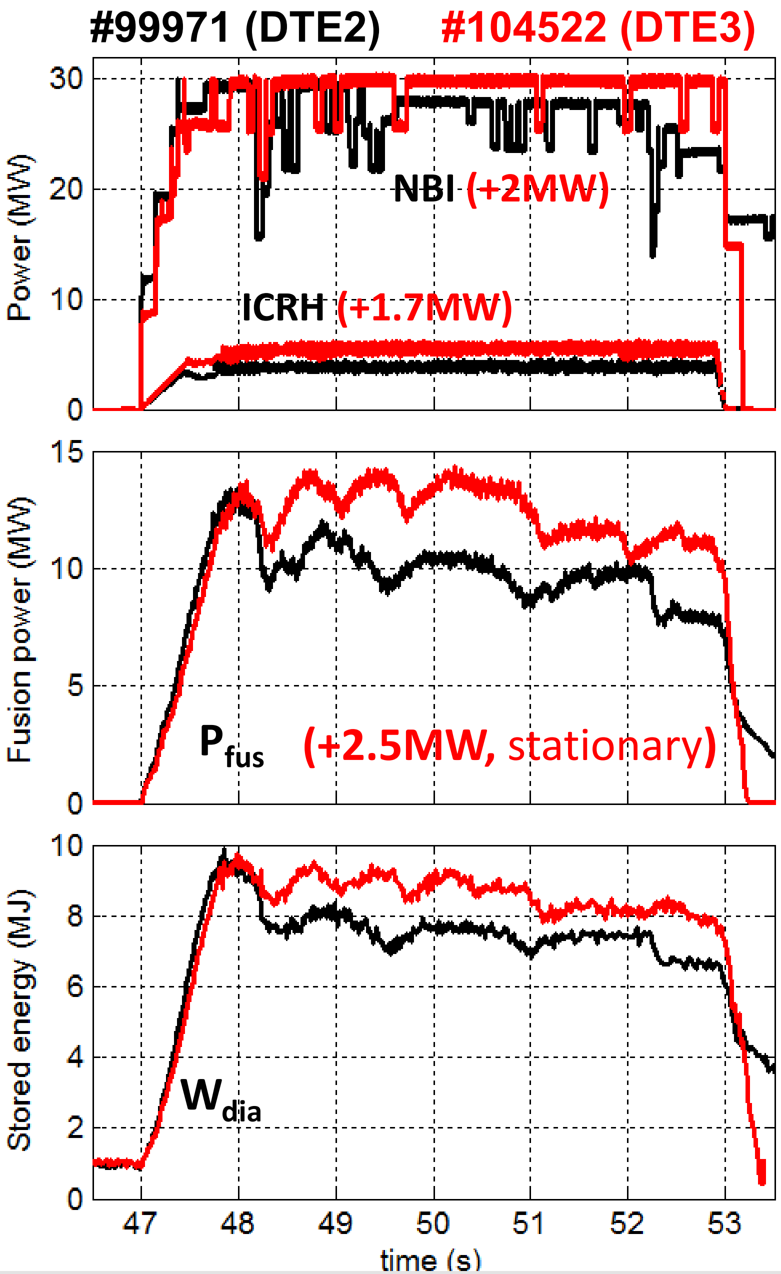
- DTE3: Previous (DTE2) fusion record broken with more ICRH power :

  Larger peak power and more energy (better stationarity): P_{fus}=12.5MW (5s), E_{fus}=69MJ




clideo.com

DTE2 fusion record exceeded in DTE3 (ICRH)



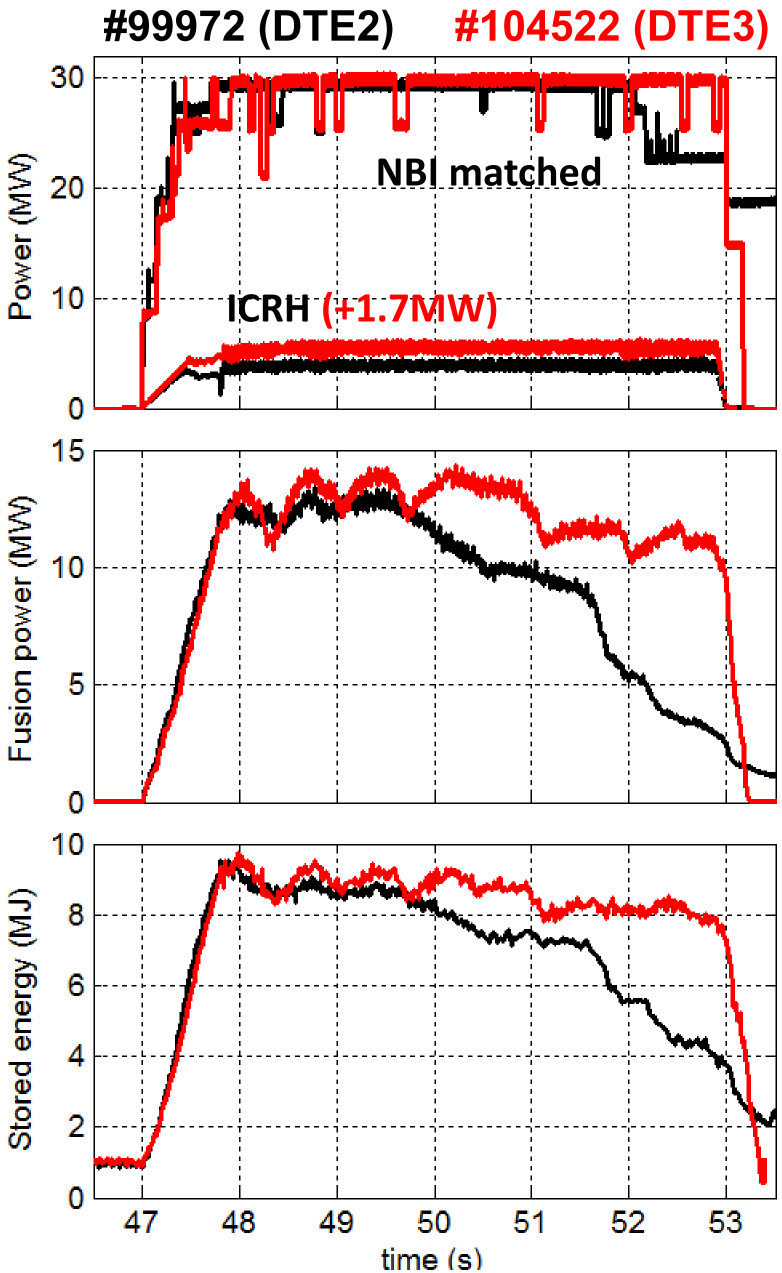
New D-T world-record:
P_{fus} = 12.5MW for 5s
E_{fus}=69MJ

Small additional power leads to larger fusion and better plasma stationarity

→ More central elec. heating

→ Larger NBI+ICRF synergy

DTE2 fusion record exceeded in DTE3 (ICRH)



New D-T world-record:

In particular, +1.7MW ICRF allowed for better stationarity (impurity screening) and slower NBI slowing-down

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Neutral beam injection (NBI)

2 NBI lines injecting ions in the I_P direction (heating + rotation)

- Species: H, D, T
- Power capability = 2 x 16MW = **32MW**
- Energy of injected ions = **100-120keV**

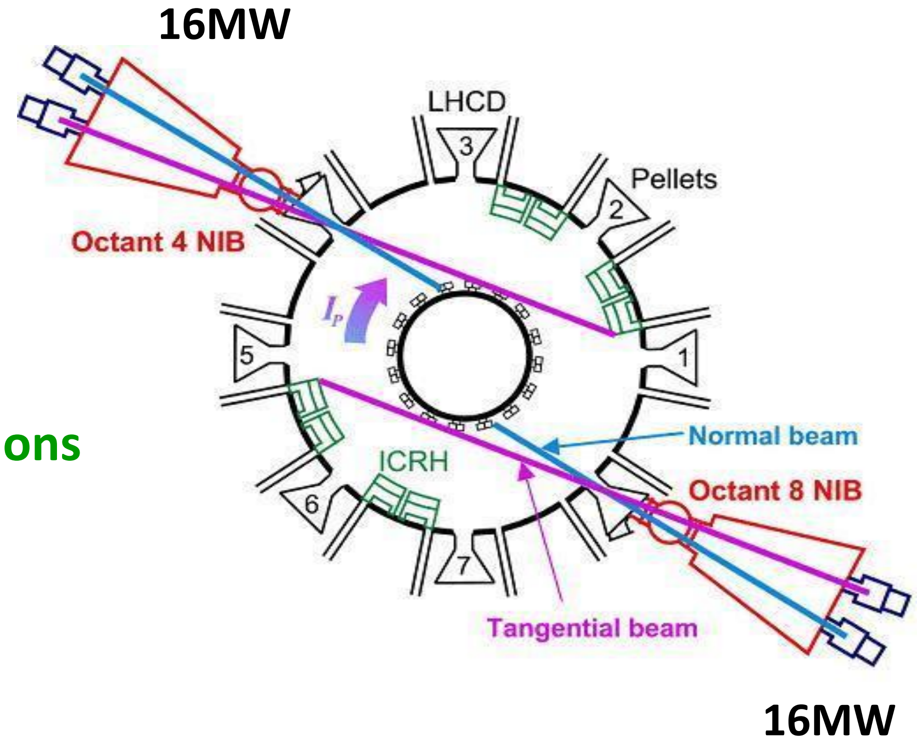
- Below critical energy → ion heating

- Optimal energy for D-T fusion → significant beam-target reactions

- Beam penetration:

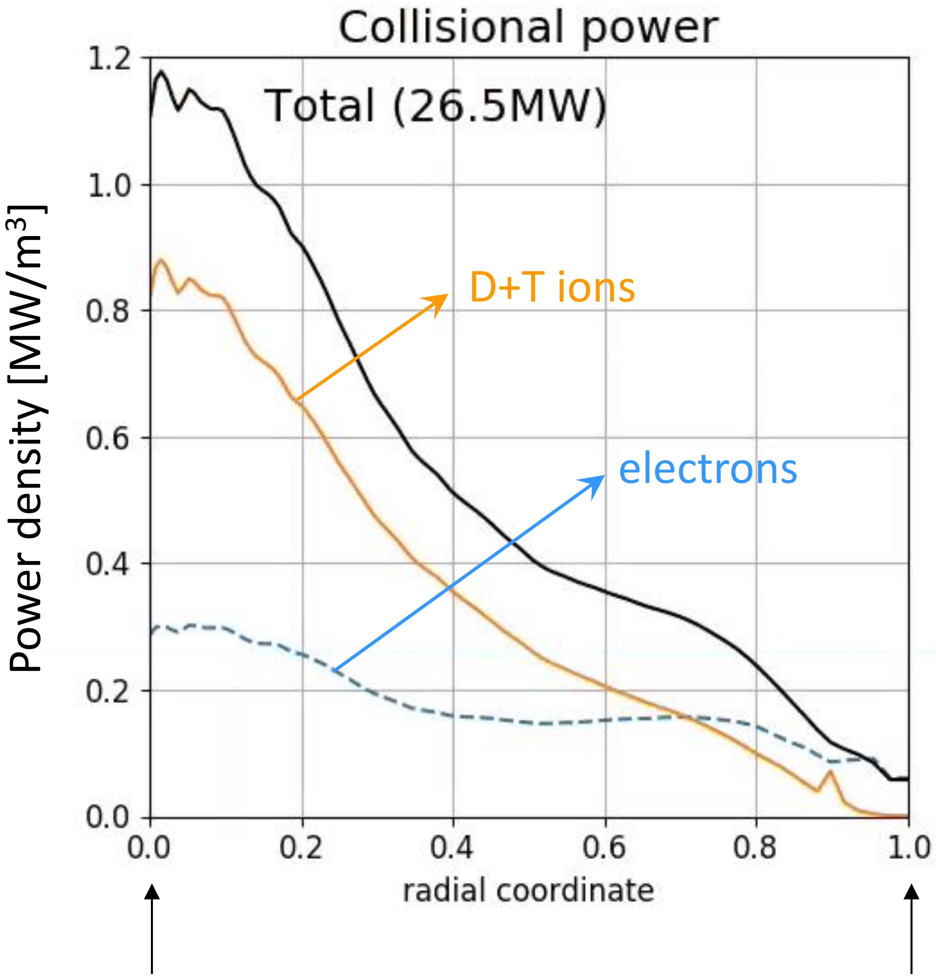
→ Peaked absorption at low collisionality (hybrids): $T_{i0}>T_{e0}$

→ Broad absorption in baseline plasmas: $T_{i0} \sim T_{e0}$



[King et al, Nucl. Fusion, Special Issue on DTE2 2023]

Neutral beam injection (NBI)



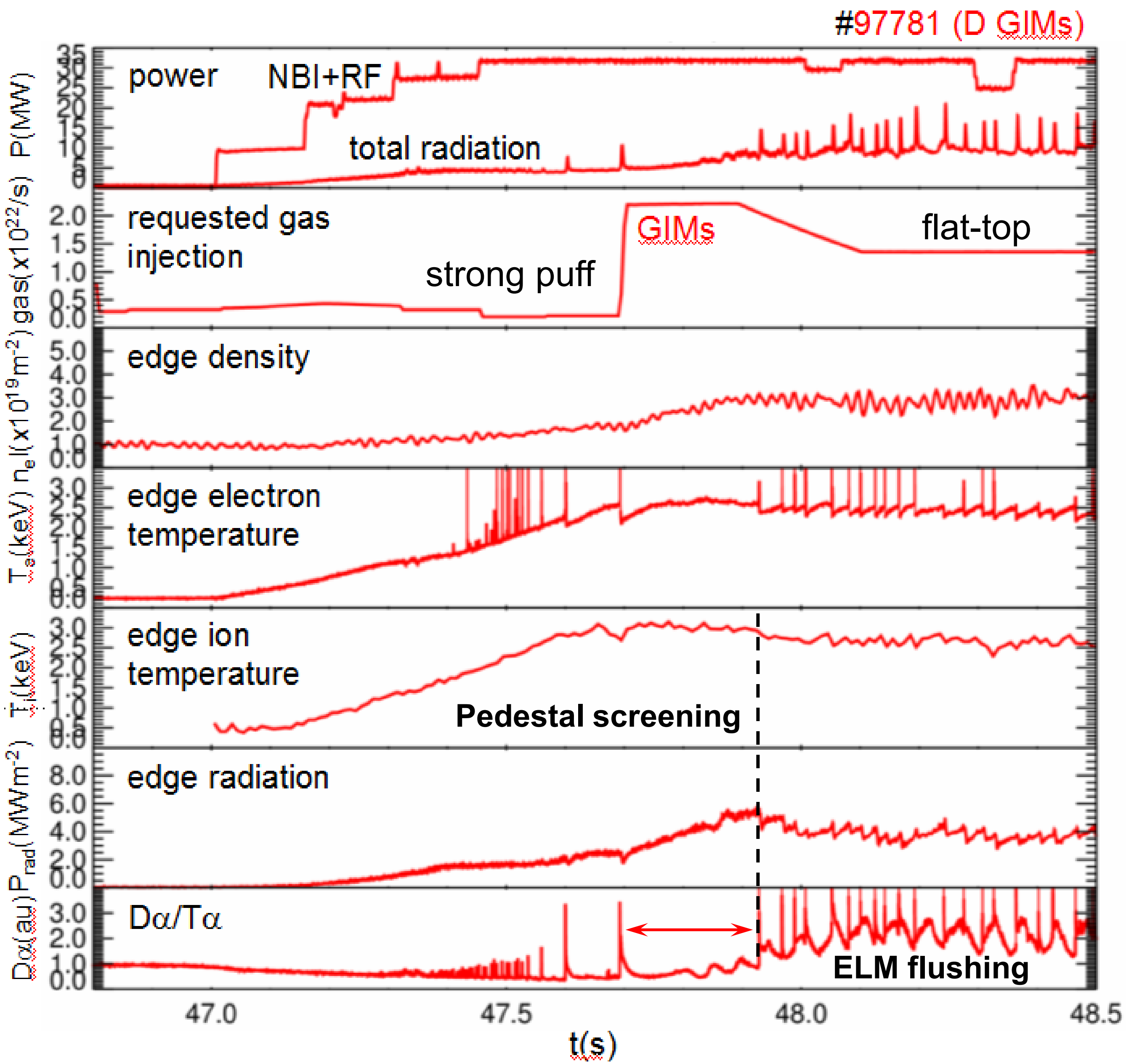
Magnetic axis

Separatrix

- Energy of injected NB ions is absorbed by collisions with the bulk plasma (electrons and ions)
- In JET, dominant ion heating
  ($E_{beam}< E_{crit}$ ~ 200keV)
- At large plasma density, beam penetration is less efficient (flatter deposition)

(In ITER, NBI → dominant electron heating)

Access to H-mode: NBI vs gas

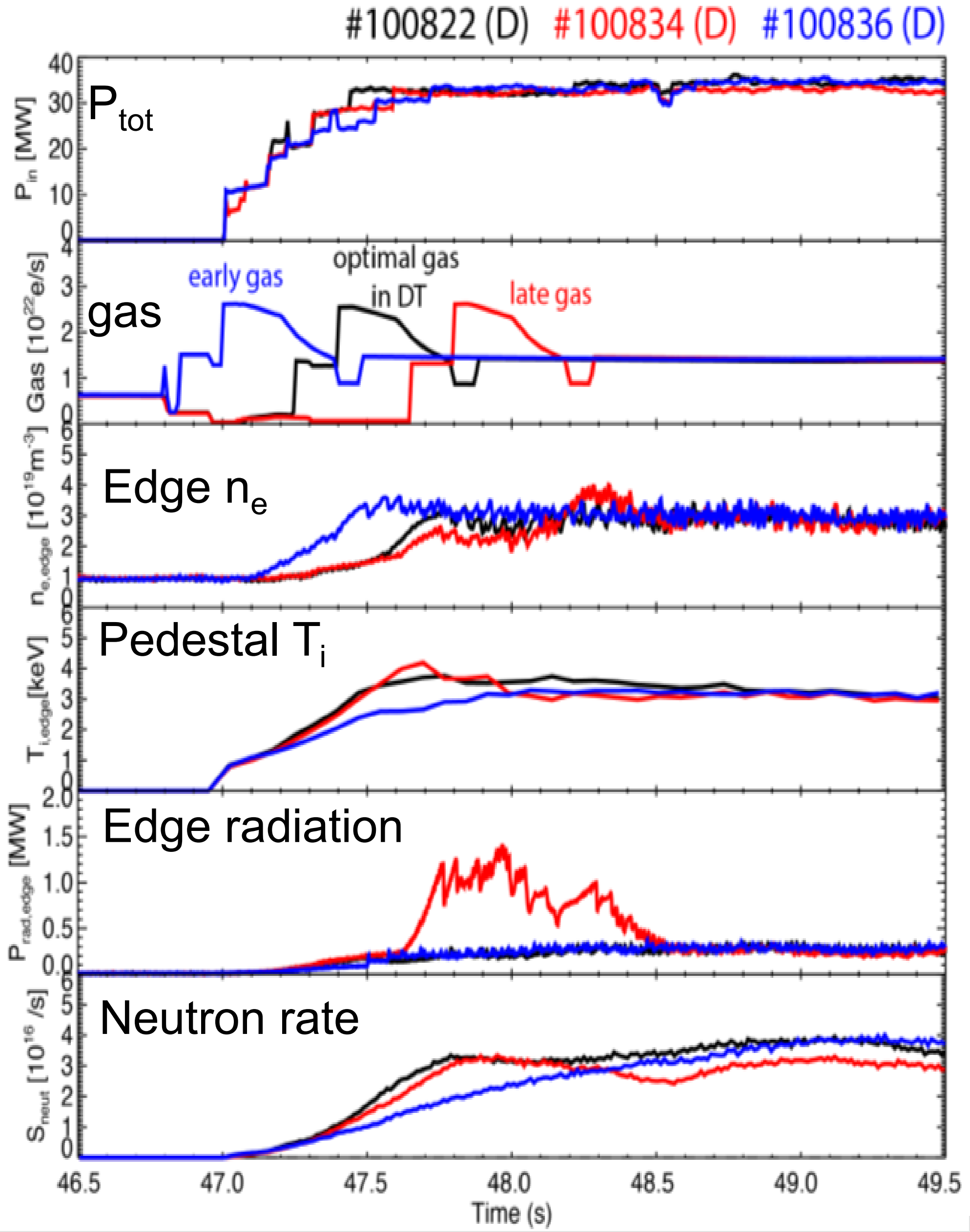


- In best plasmas pedestal temperature built up before significant density increased to establish effective impurity screening (grad T_i)

- ELM-free phase is relatively short, followed by regular type I ELMs that guarantee edge impurity control (flushing)

[A. Field,. NF2022]

Access to H-mode: Optimal gas ‘slug’ vs NBI onset



- A gas puff (‘slug’) is needed for an optimal H-mode entry: transition from hot ion pedestal phase (ELM-free) to the type I ELMy regime (flat-top)
- It has to be well synchronized with the application of NBI power

Too early gas slug:

- Early increase of the plasma density, Low pedestal T_i, (sometimes impurity influx), delay to reach high fusion performance

Too late gas slug:

- Large pedestal T_i but too long ELM-free phase, uncontrolled edge radiation (impurity influx)

Optimal gas slug:

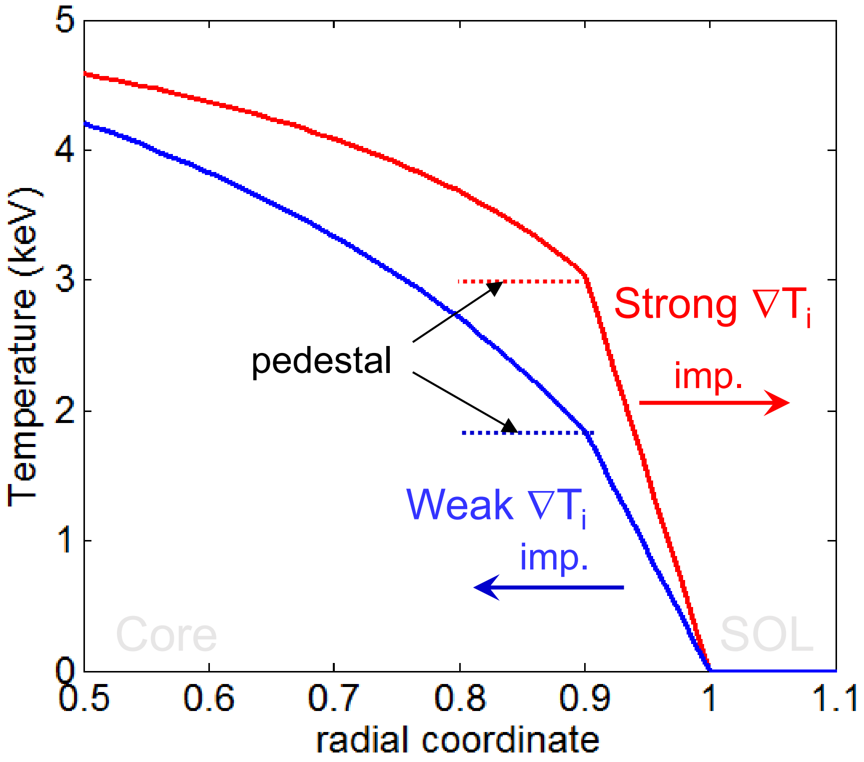
- Large pedestal T_i (comparable to late gas), short ELM-free phase, edge impurity screening, good fusion performance reached early

Edge impurity screening (Ti)

- High performance plasmas with sufficiently hot pedestal feature inter-ELM edge impurity screening

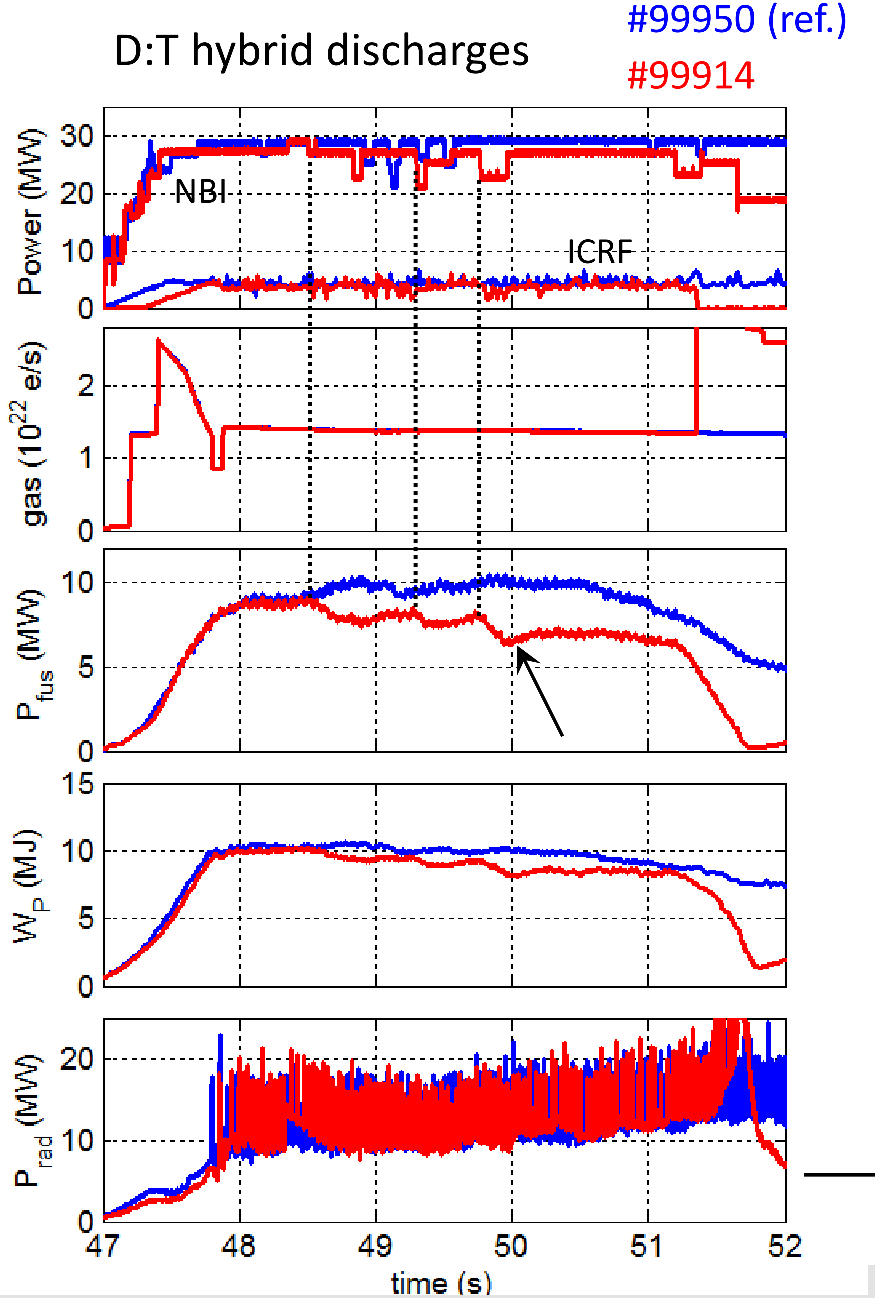
[A. Field et al, Nucl. Fusion 2023]

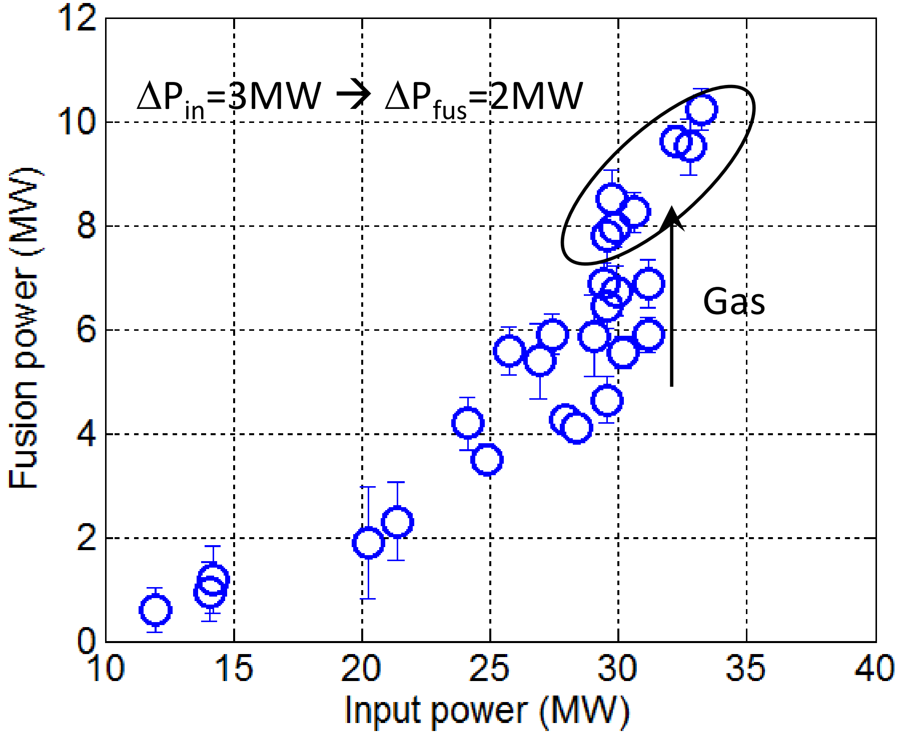
Strong T_i gradients outside the pedestal region ‘block’ impurity influx from SOL



[A. Field,. NF2022]

ELMy H-mode phase: NBI power level





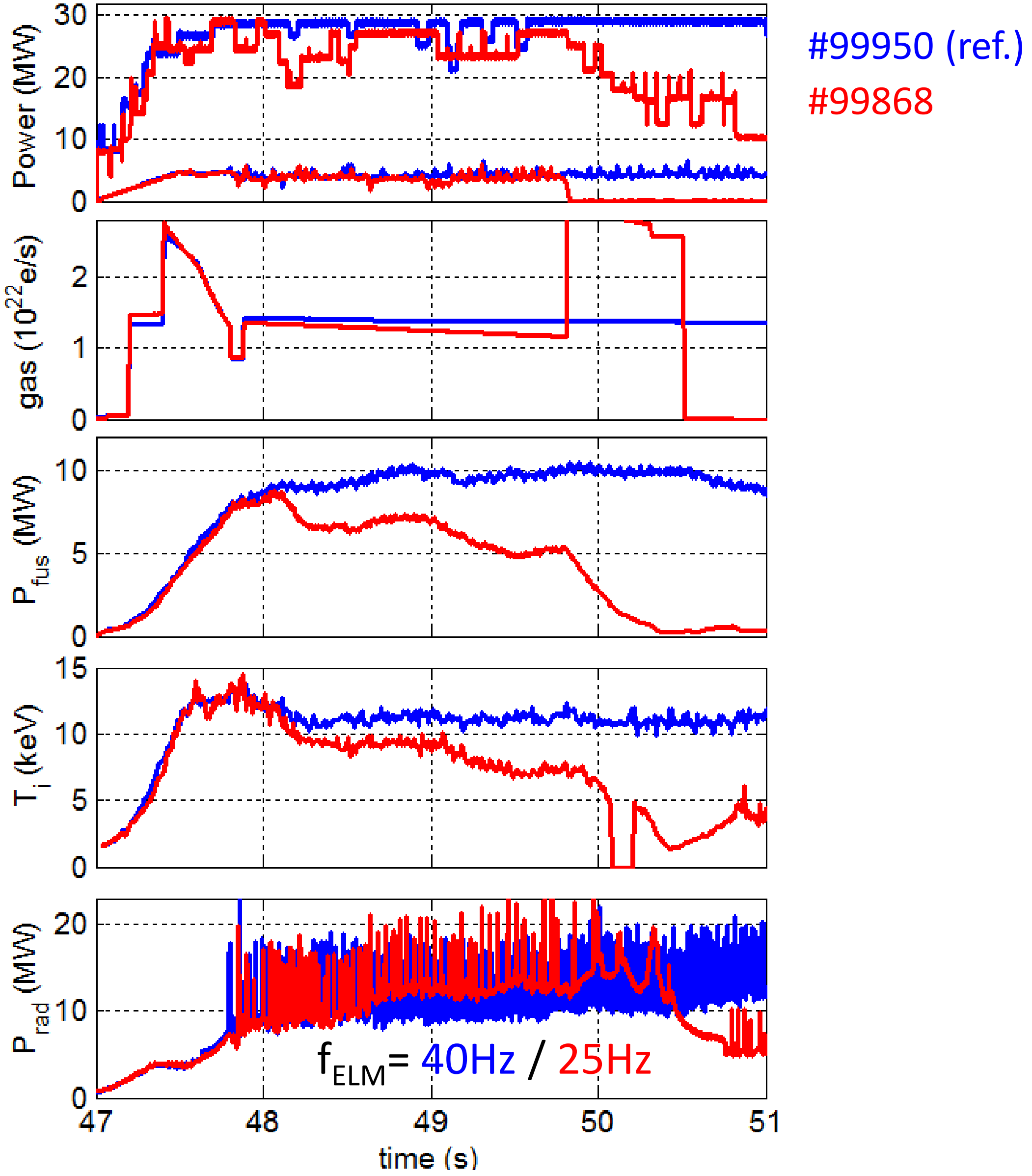
- Large input power is needed for high fusion performance
(ion heating, beam target reactions)

- Steep dependence at high P_{in}

- Performance is very sensitive to NBI power waveform (and power level)

ELMy H-mode phase: NBI power level

D:T hybrid discharges

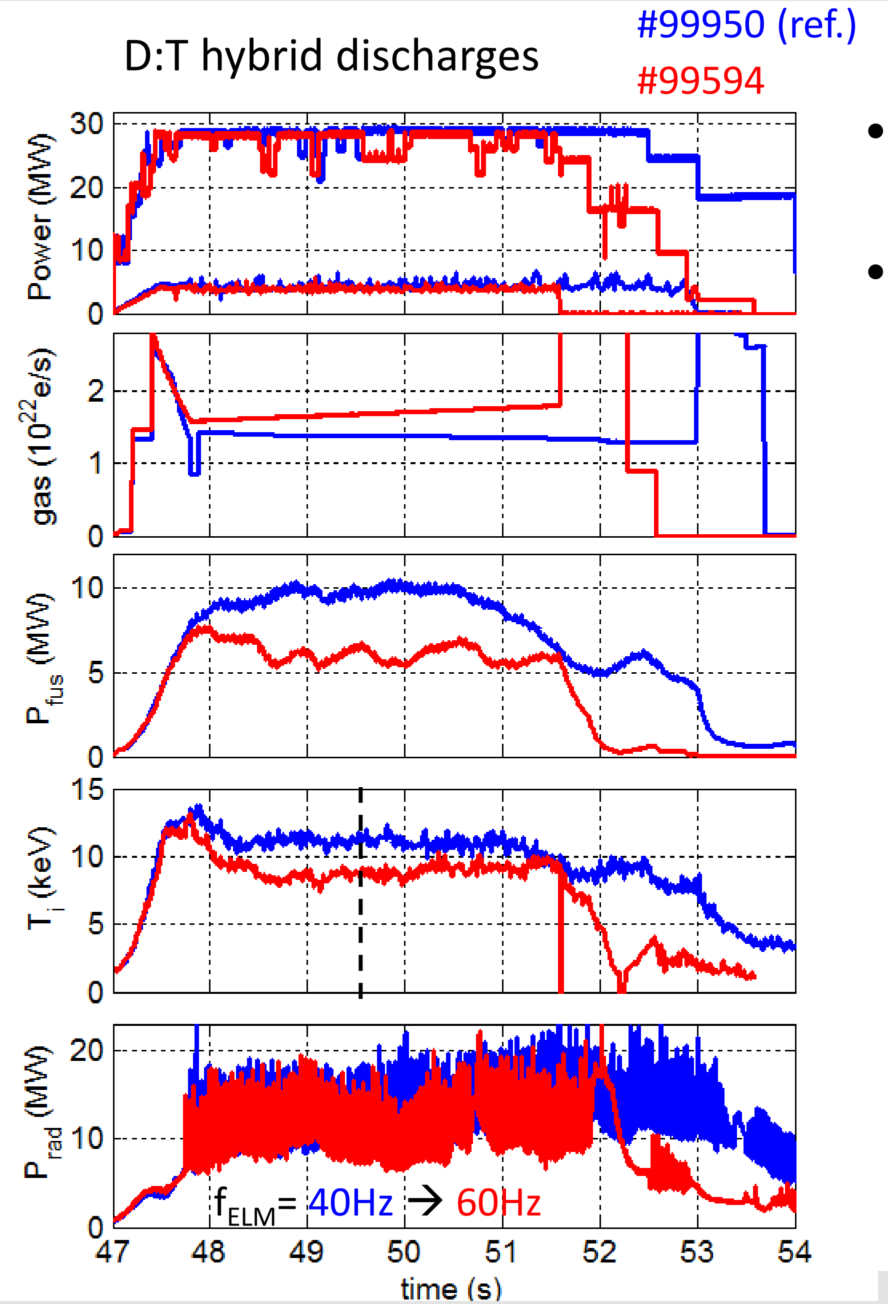


- Ragged NBI input kills plasma performance

→ impurity influx, cold pedestal, slow ELMs, core W accumulation, emergency stop

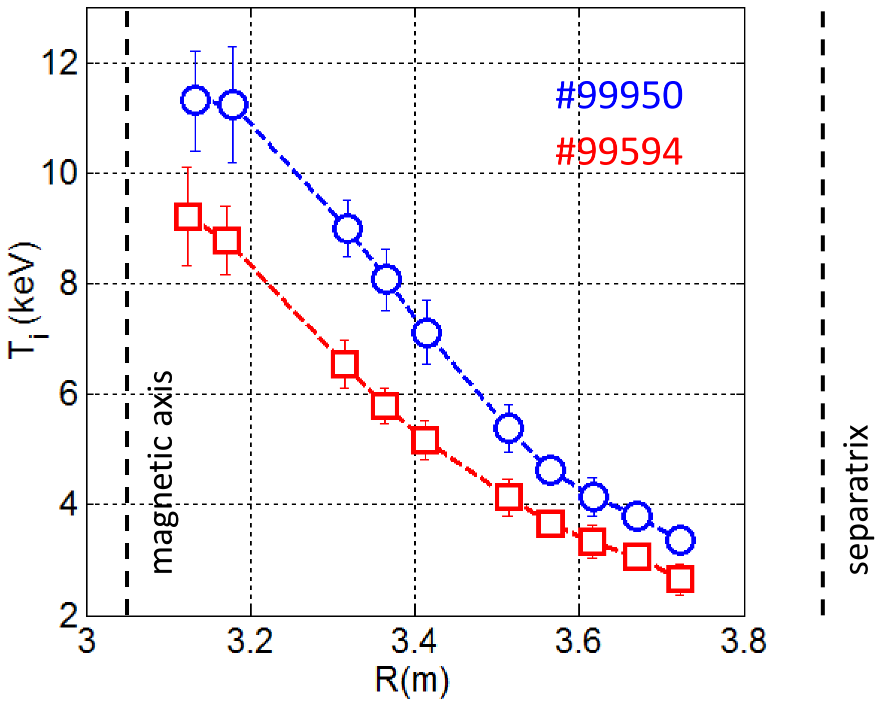
- More gas can be used to drive faser ELMs and avoiding core W accumulation but pedestal remains cold and performance is poor (safety net on ELM frequency)

ELMy H-mode phase: Gas injection level


D:T hybrid discharges

- Too much gas kills plasma performance (pedestal + core degradation)
- Fast ELM's (good for stationarity)

CXRS (49.5s)

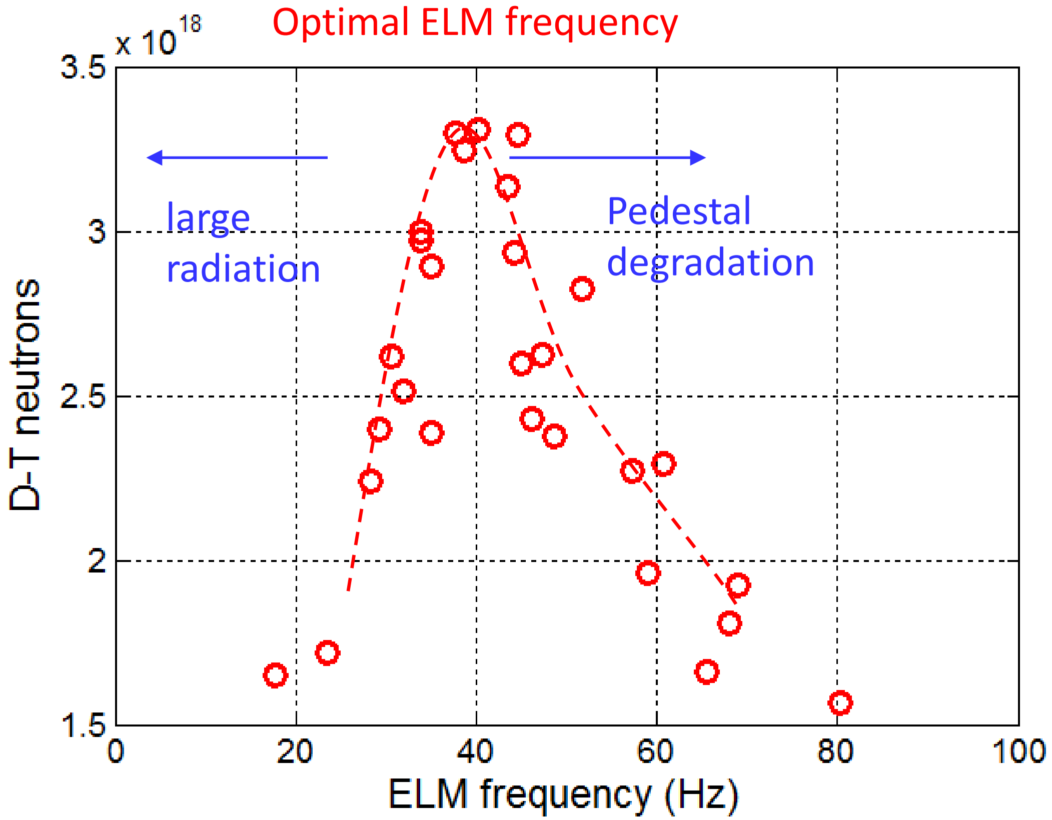


Edge localized modes (ELMs)

- ELMs play a <u>key role</u> in the plasma performance and stationarity

- Too slow ELMs = less efficient impurity flushing (radiation)

- Too fast ELMs = degradation of the pedestal and core performance



Important to control ELMs for good and stable plasma performance

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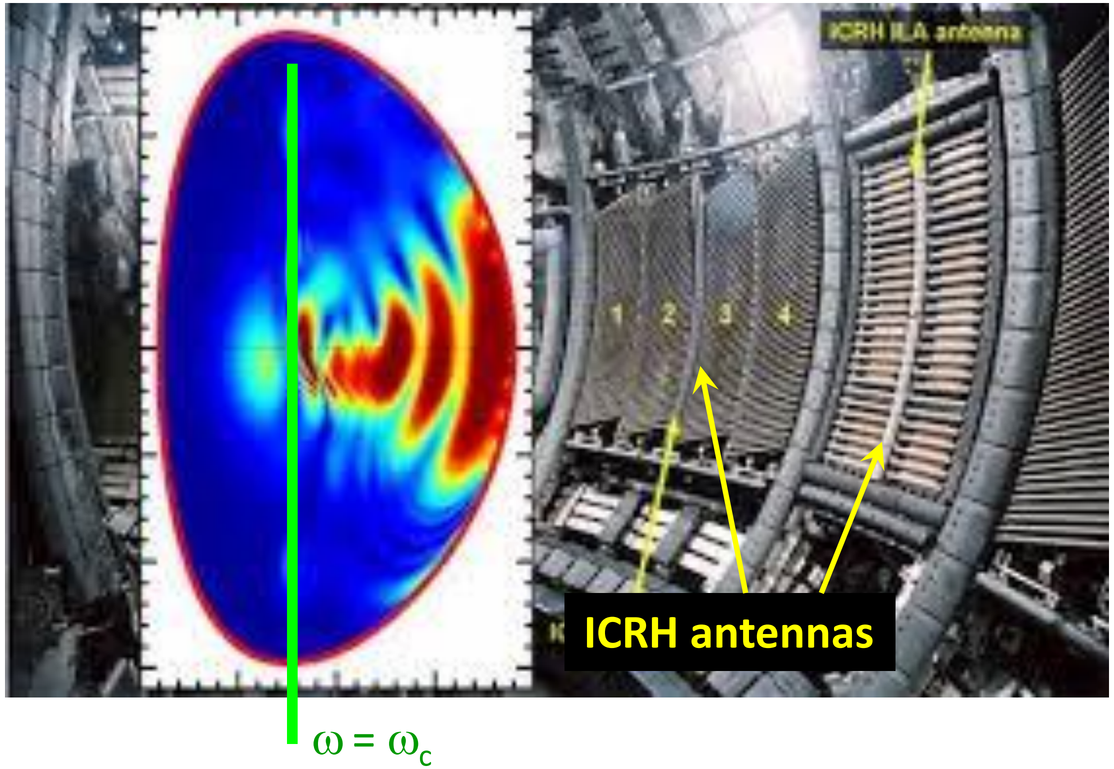
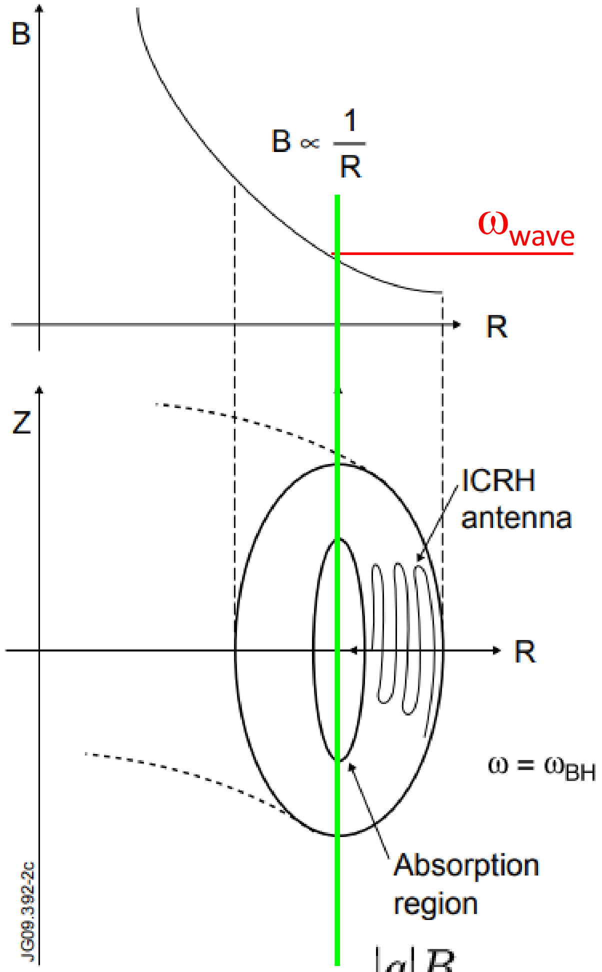
Ion cyclotron resonance heating (ICRH)





$$\omega_c = \frac{|q|B}{m}$$

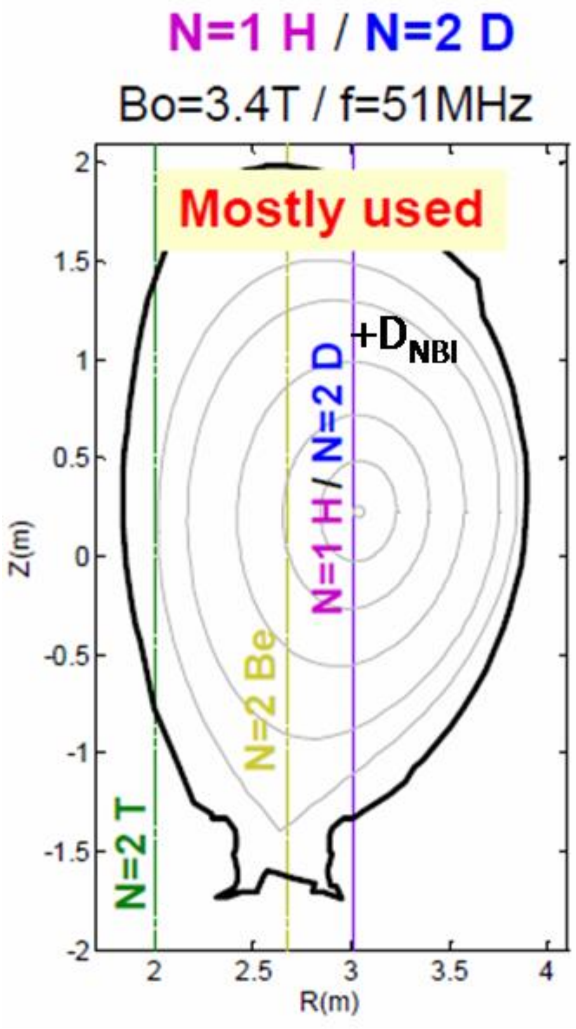
- A fast wave is launched into the plasma and propagates towards the centre
- It is absorbed in a narrow region where $\omega_{wave} = \omega_c$
- Two possibilities:

(1) Wave resonates with a minority species → collisions

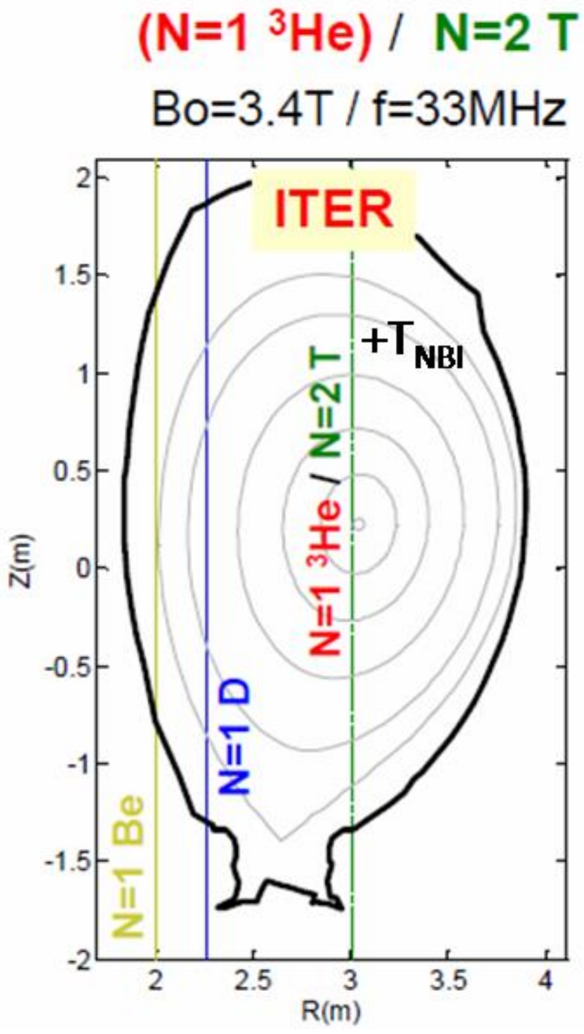
(2) Wave resonates directly with one of the fuel ions

[Jacquet et al, IAEA-FEC 2023]

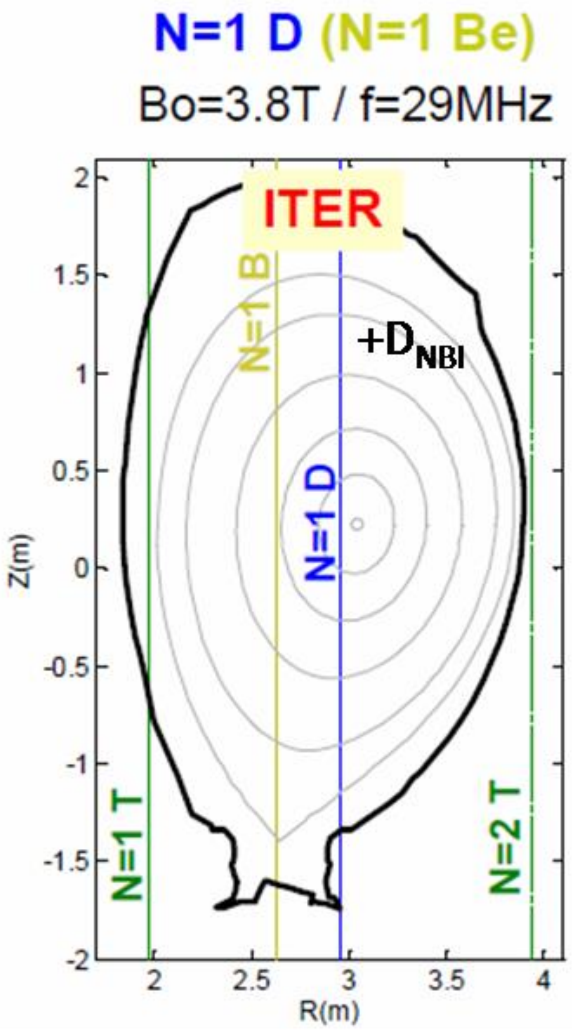
Different ICRH schemes tested in D-T



ICRF electron heating



Dominant ion heating



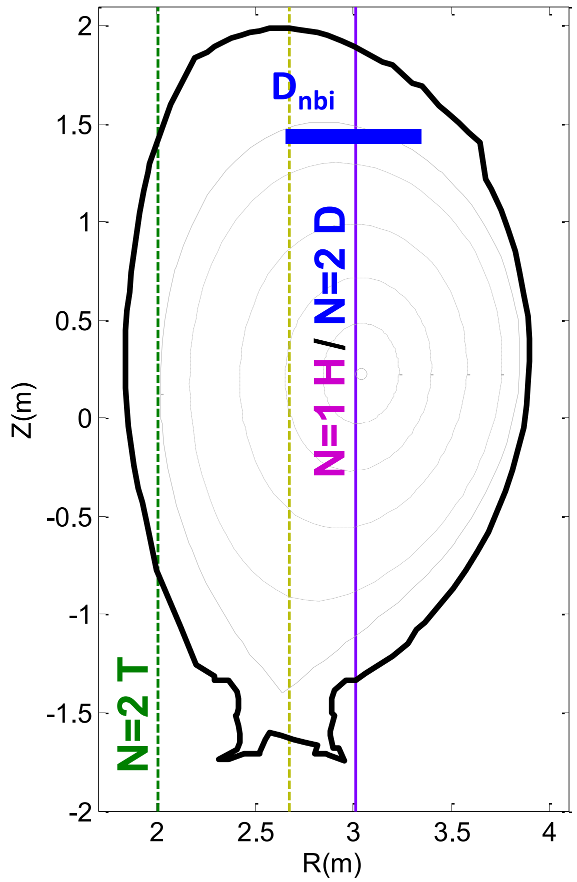
Dominant ion heating + D_{fast} fusion enhancement

- 3-ion ICRH schemes *[Y. Kazakov, this conference]*

Fundamental H and N=2 D_{bulk}+D_{nbi} heating

N=1 H / N=2 D

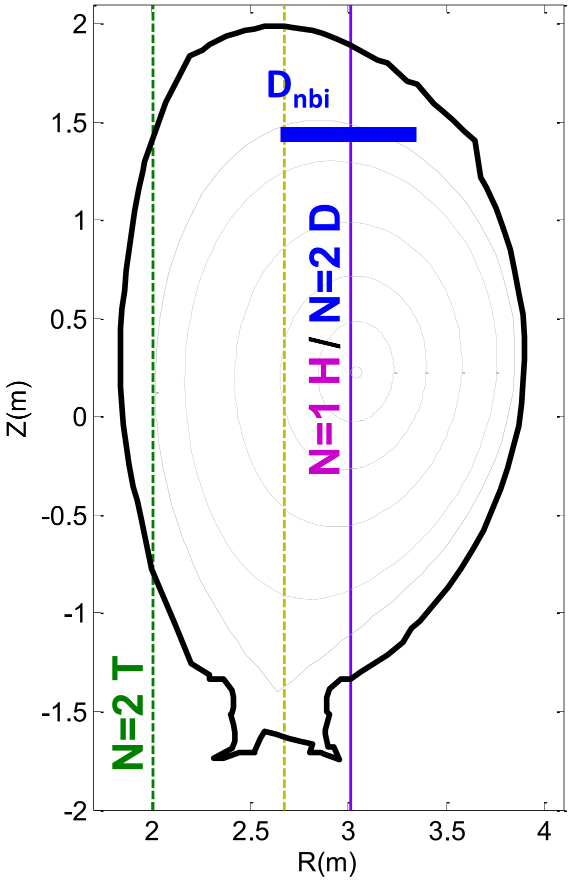
Bo=3.4T / f=51MHz



Fundamental H and N=2 D_{bulk}+D_{nbi} heating

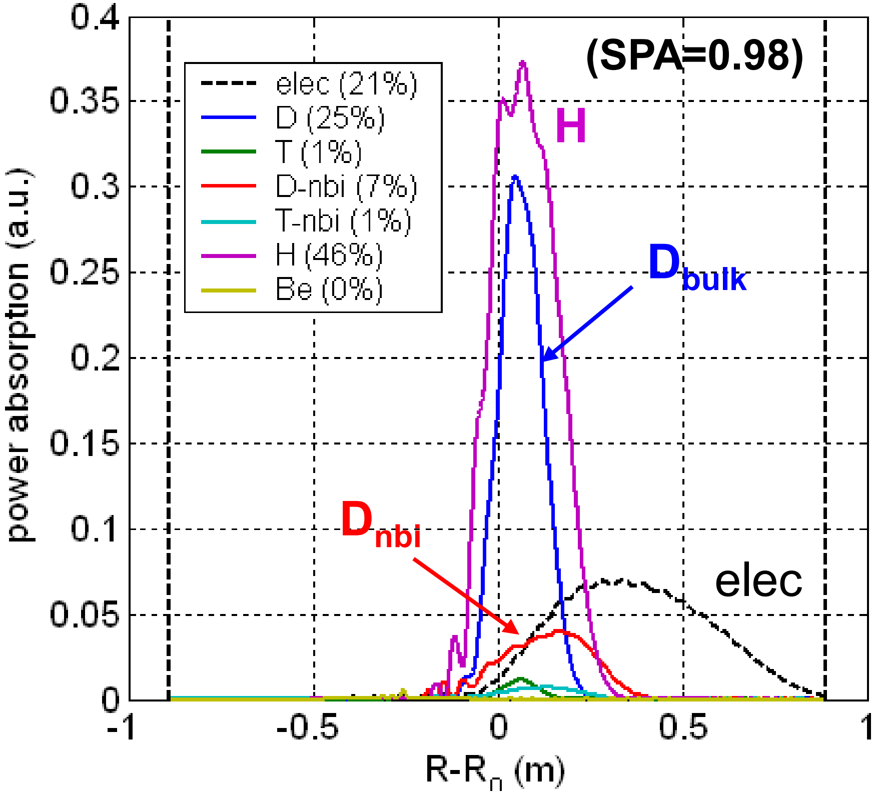
N=1 H / N=2 D

Bo=3.4T / f=51MHz



n_0=8e19/m^3, T_e/T_i=8/10keV

50%D:50%T, $2\%D_{nbi}$, $2\%T_{nbi}$, **2%H**, 1%Be



- Strong SPA
- Dominant H heating:

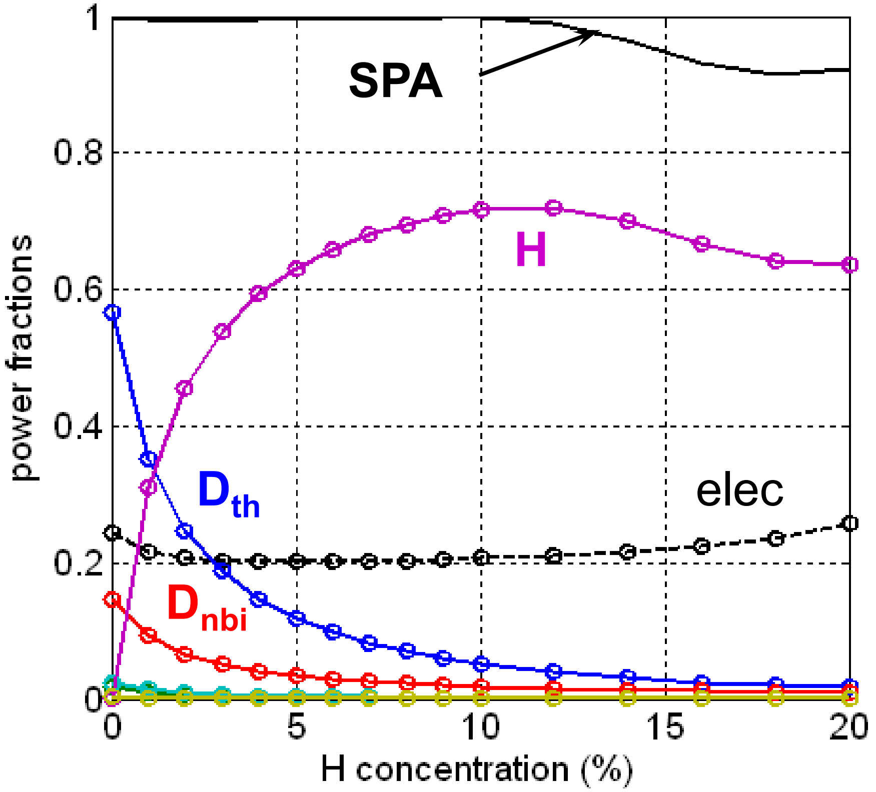
H = 45% (D_{bulk} = 25%; D_{nbi}=7%)

D. Gallart et al.,
2018 Nucl. Fusion

Fundamental H and N=2 D_{bulk}+D_{nbi} heating

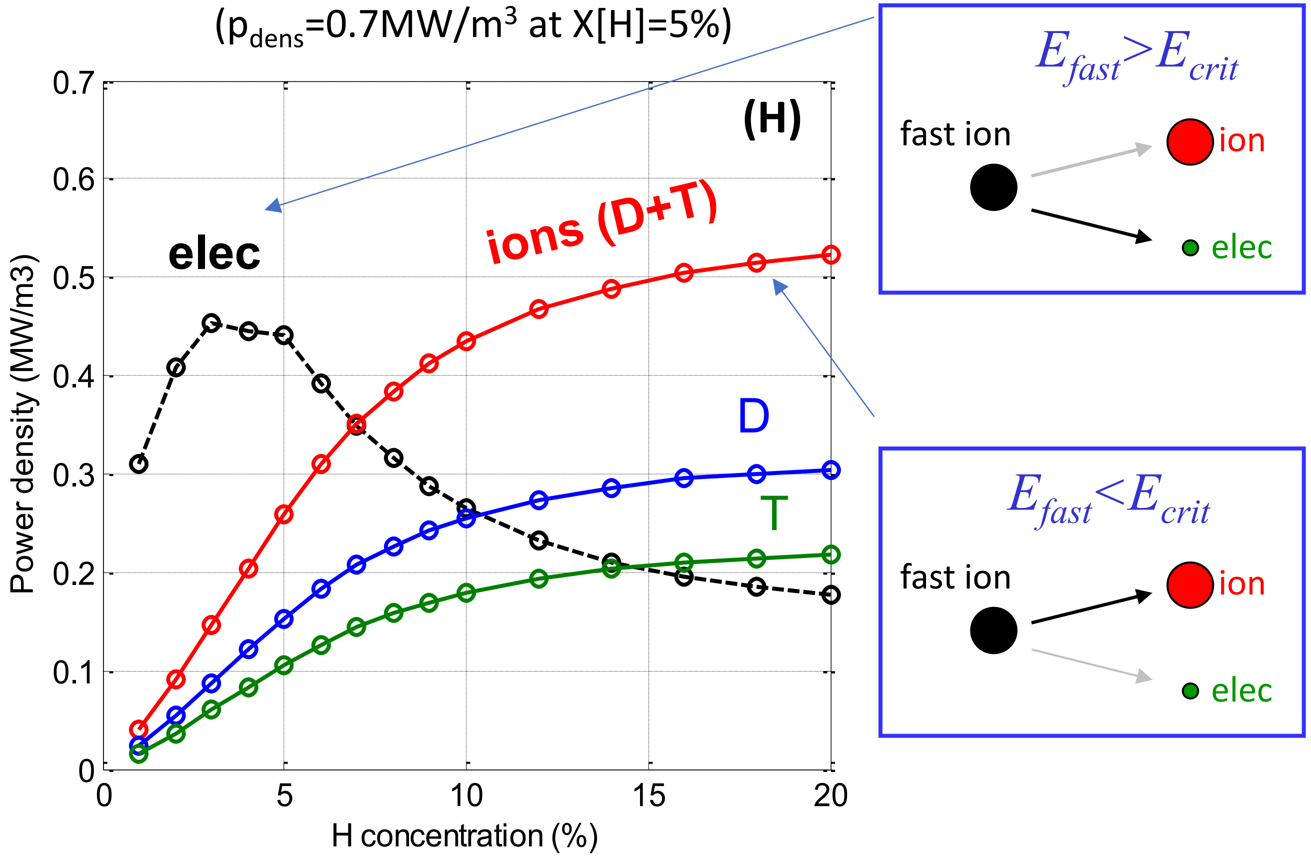
n_0=8e19/m^3, T_e/T_i=8/10keV

D:**T**, 2%D_{nbi}, 2%T_{nbi}, **H scan**, 1%Be



Collisional power redistribution for H min.

(p_{dens}=0.7MW/m^3 at X[H]=5%)

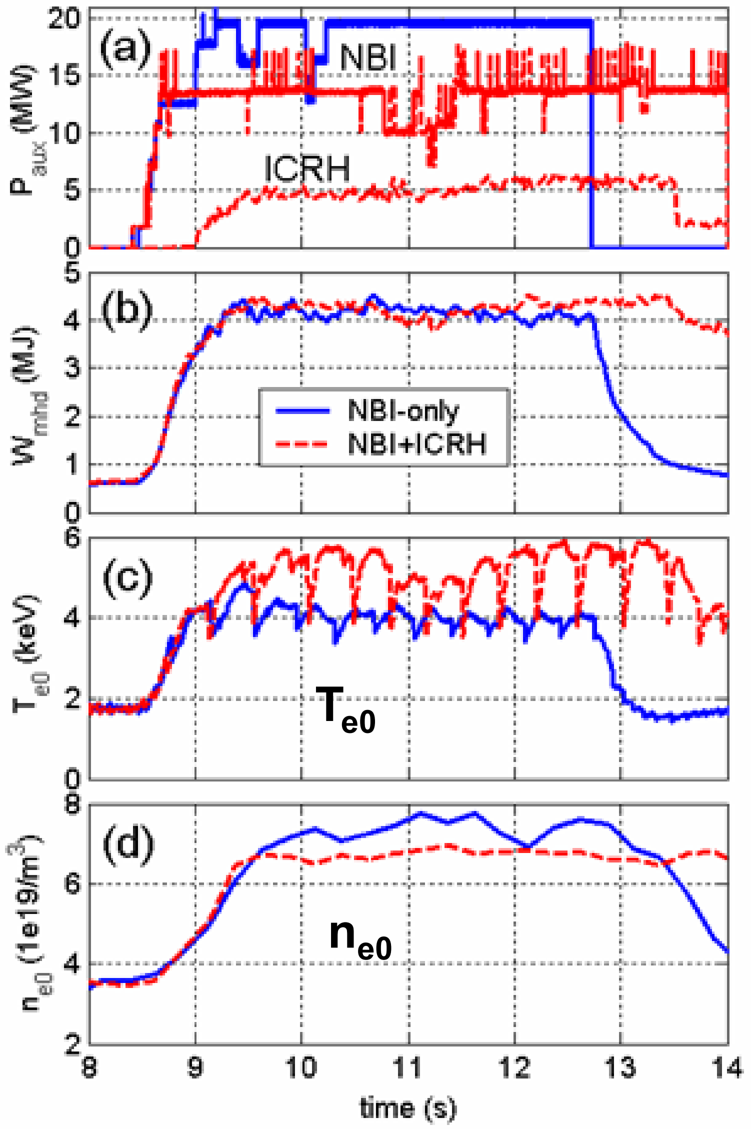


- Always good absorption
- Dominant N=2 **D_{th}**+**D_{nbi}** heating for X[H]<2%:
  → Enhanced DD neutrons
  → Fast (MeV) D ions

- Dominant **electron heating** from H minority up to X[H] ~ 7%:
- When dominant N=2 D absorption, electron heating is still important (fast D ions)

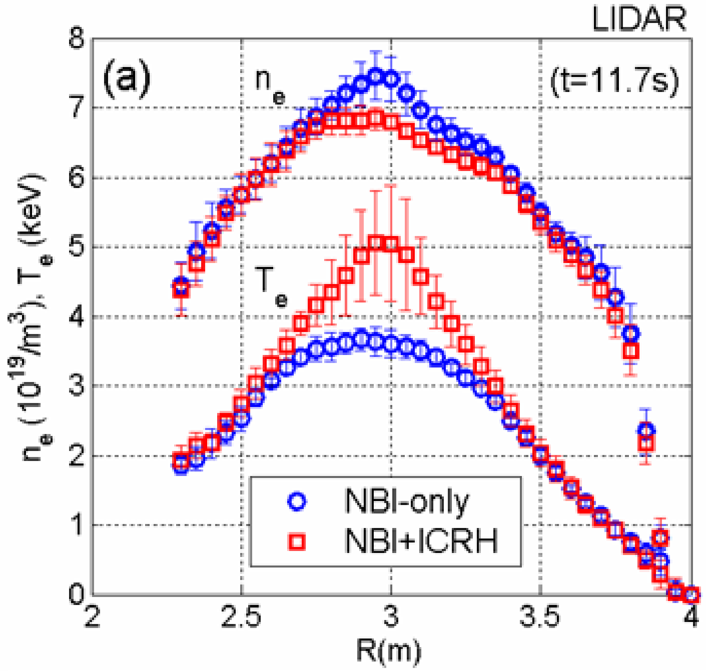
Fundamental H and N=2 D heating→ experiments

JET-ILW baseline H-mode: B_0=2.7T, I_P=2.5MA, P_{aux}=20MW, f=42.5MHz



NBI only (84746)

RF+NBI (84744)



- Larger temperature with ICRH
  (2keV for 6MW)
- Flatter density (turbulence)

E. Lerche et al., 2016 Nucl. Fusion

Collisional electron heating

H min = 3%

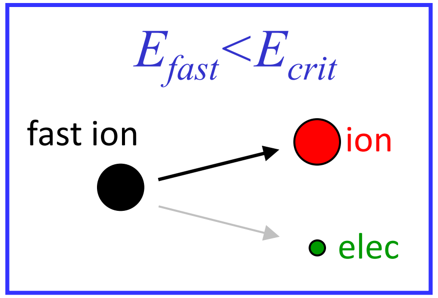
→ Strong electron heating

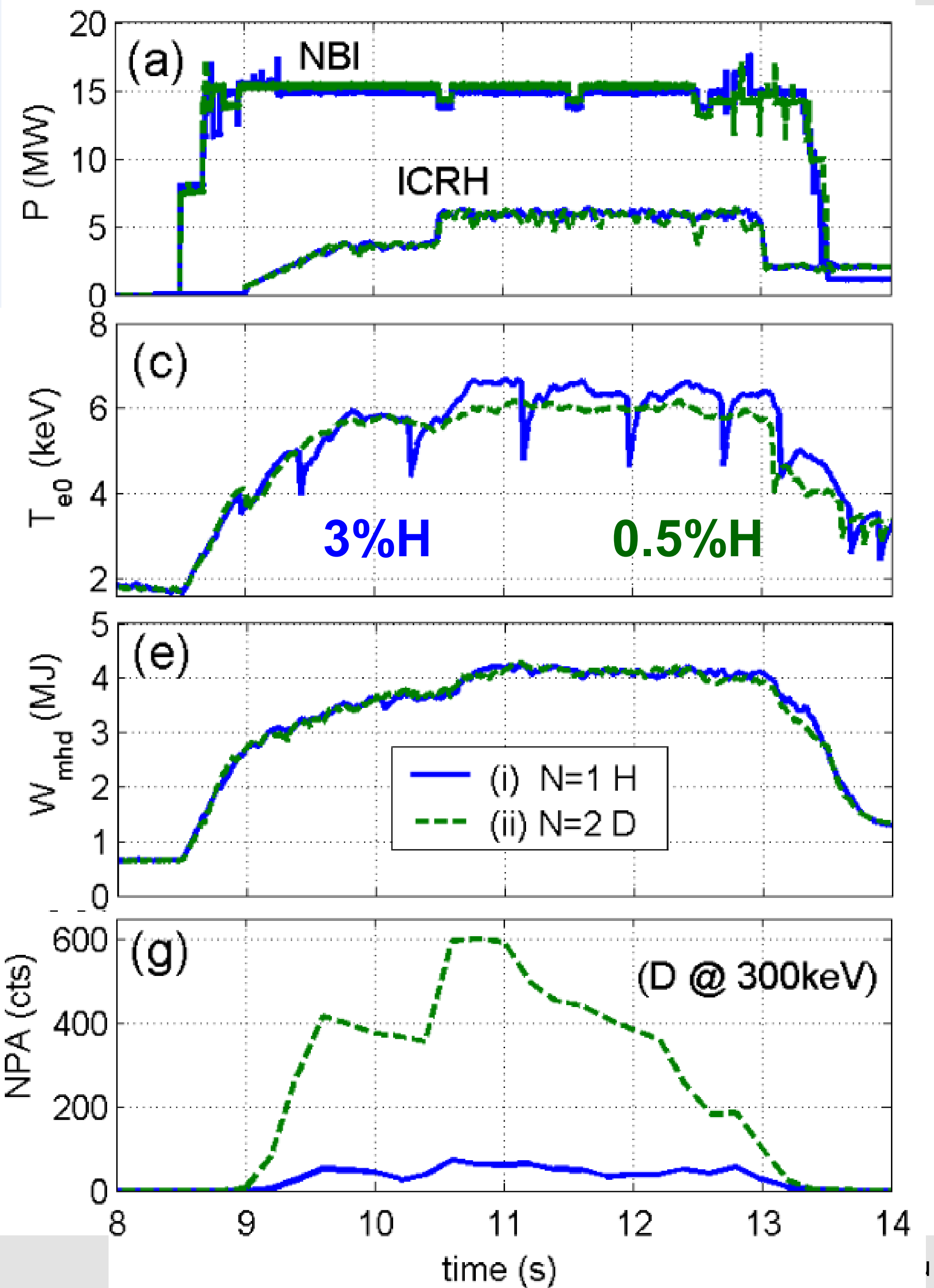
→ Fast H+ ion tails (long sawteeth)

H min ~ 0.5%

→ Strong N=2 D acceleration

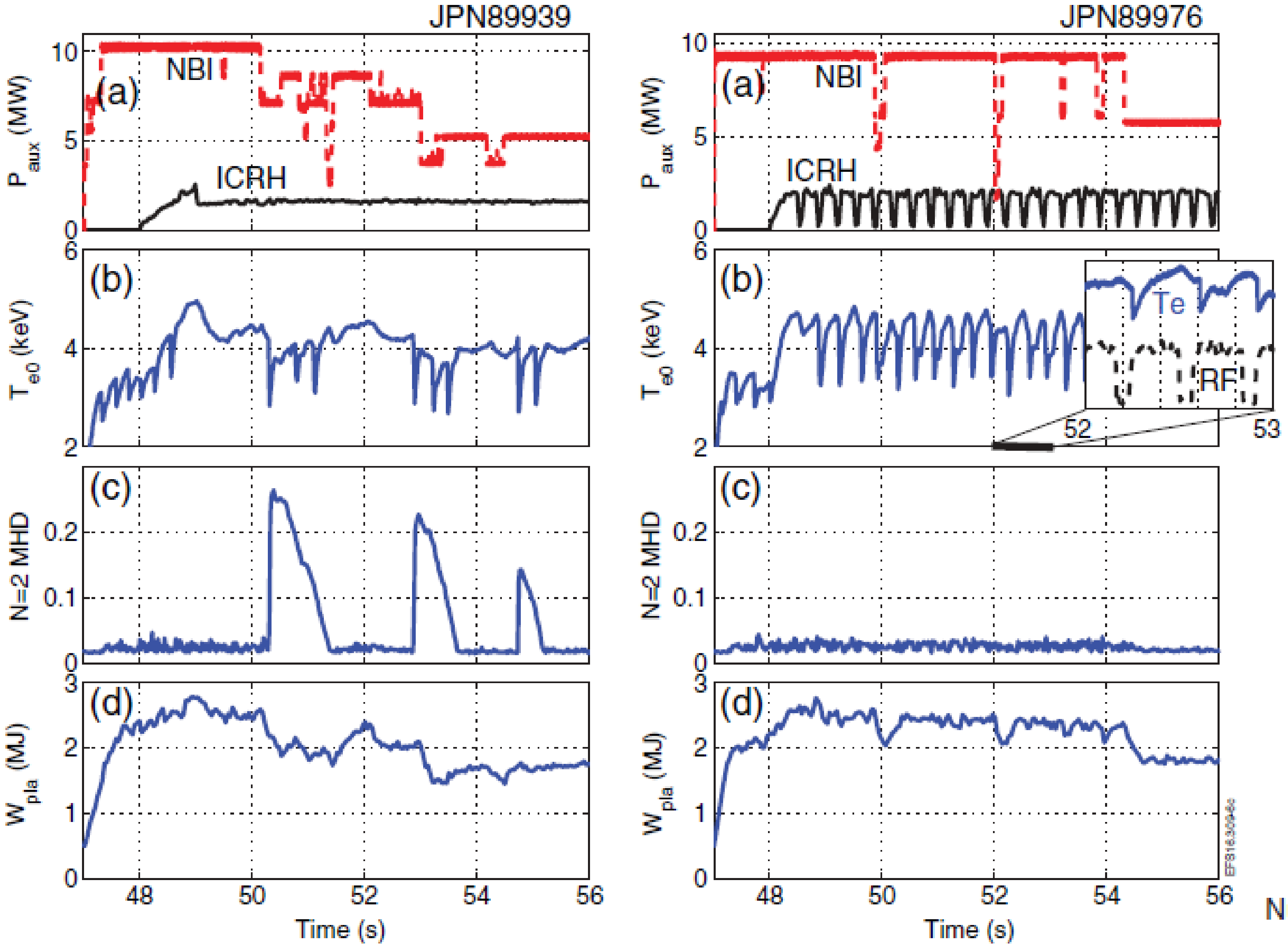
→ Very fast H+ ion tails: sawteeth completely stabilized


$E_{fast}<E_{crit}$



Intermezzo: Sawtooth pacing with ICRH

→ ICRH modulation allows to induce sawtooth crash → Sawtooth pacing

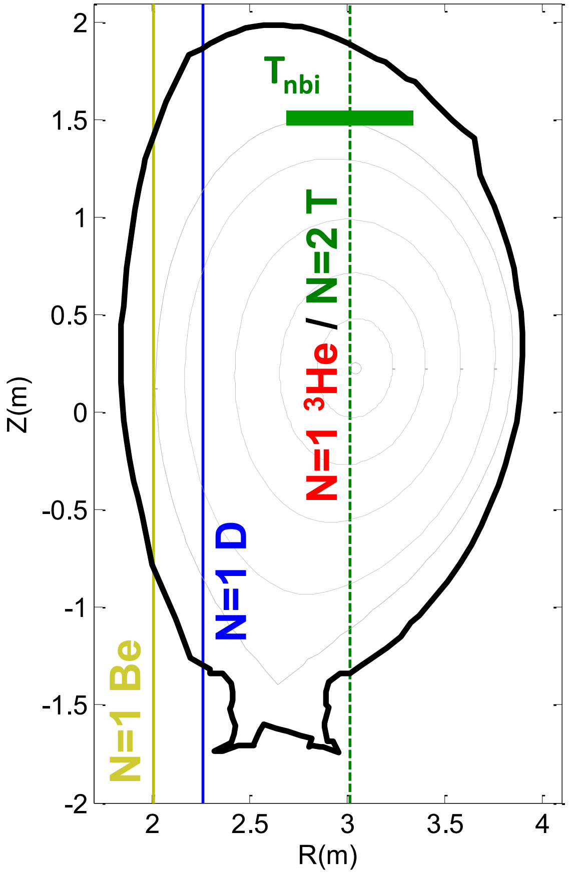


Nucl. Fusion 57 (2017) 036027

Fundamental ^{3}He and N=2 T_{bulk}+T_{nbi} heating

N=1 ^{3}He / N=2 T

Bo=3.4T / f=33MHz

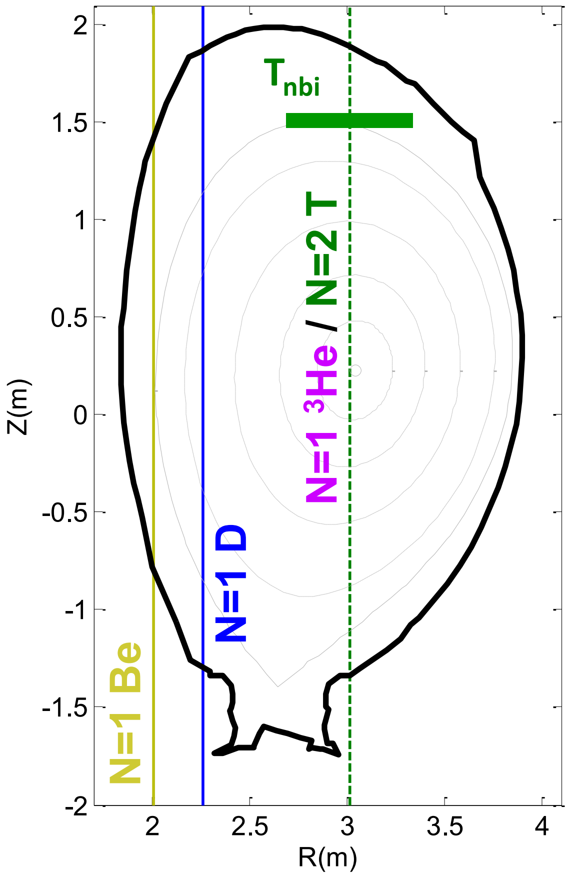


ITER reference
(50:50 D:T)

Fundamental ^{3}He and N=2 T_{bulk}+T_{nbi} heating

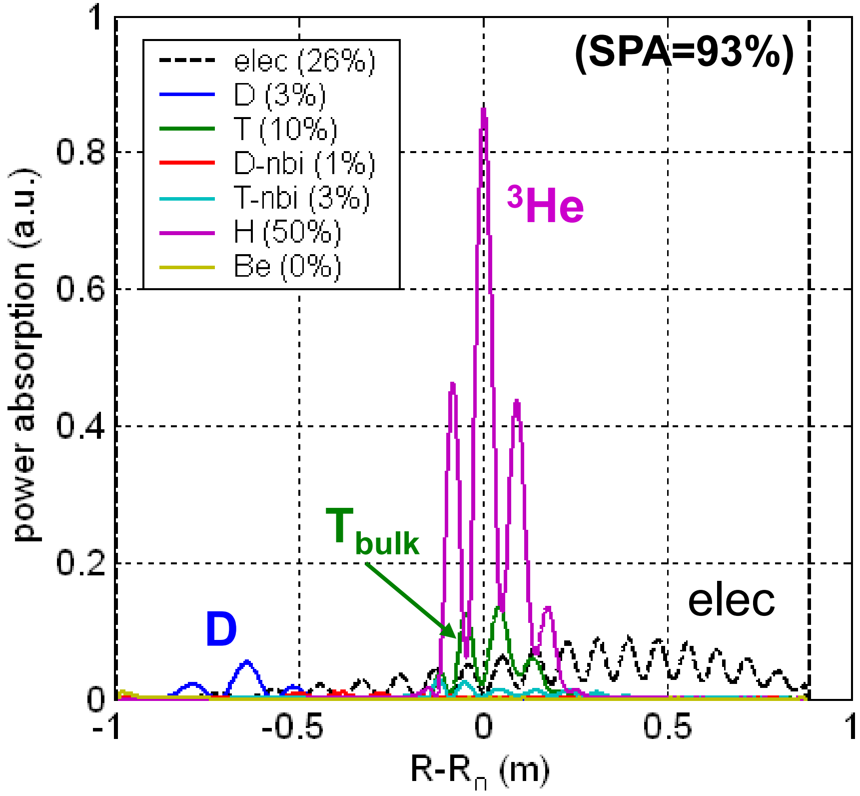
N=1 ^{3}He / N=2 T

Bo=3.4T / f=33MHz



n_0=8e19/m^3, T_e/T_i=8/10keV

50%D:50%T, 2%D_{nbi}, 2%T_{nbi}, **2%^{3}He**, 1%Be

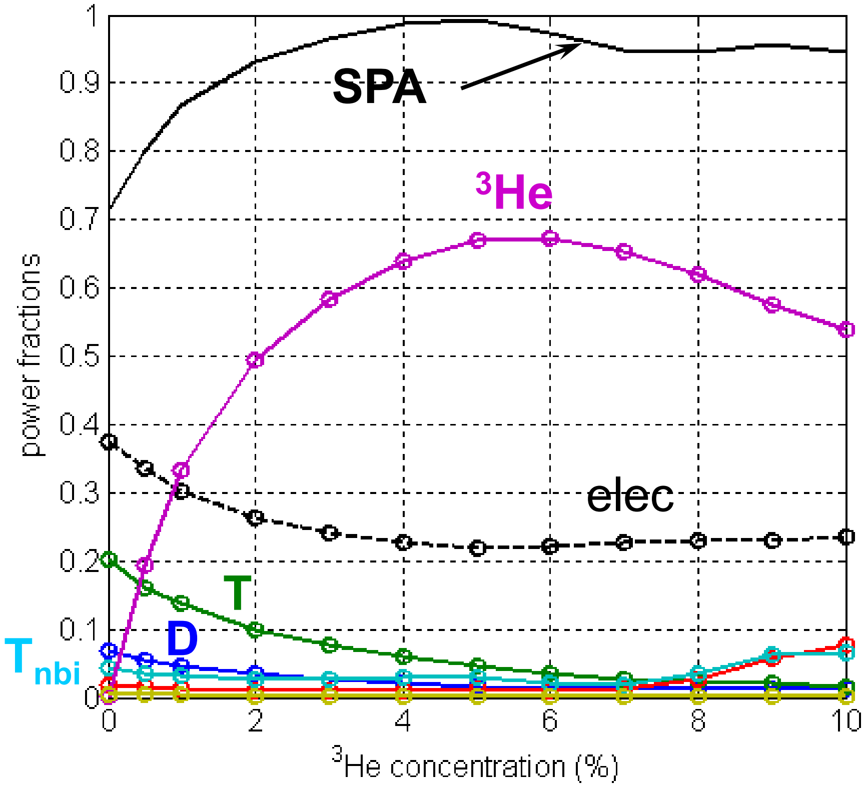


- Strong SPA
- Dominant ^{3}He heating:
  ^{3}He = 50% (T_{bulk} = 10%; elec = 25%)
- HFS absorption reduced

Fundamental ^{3}He and N=2 T_{bulk}+T_{nbi} heating

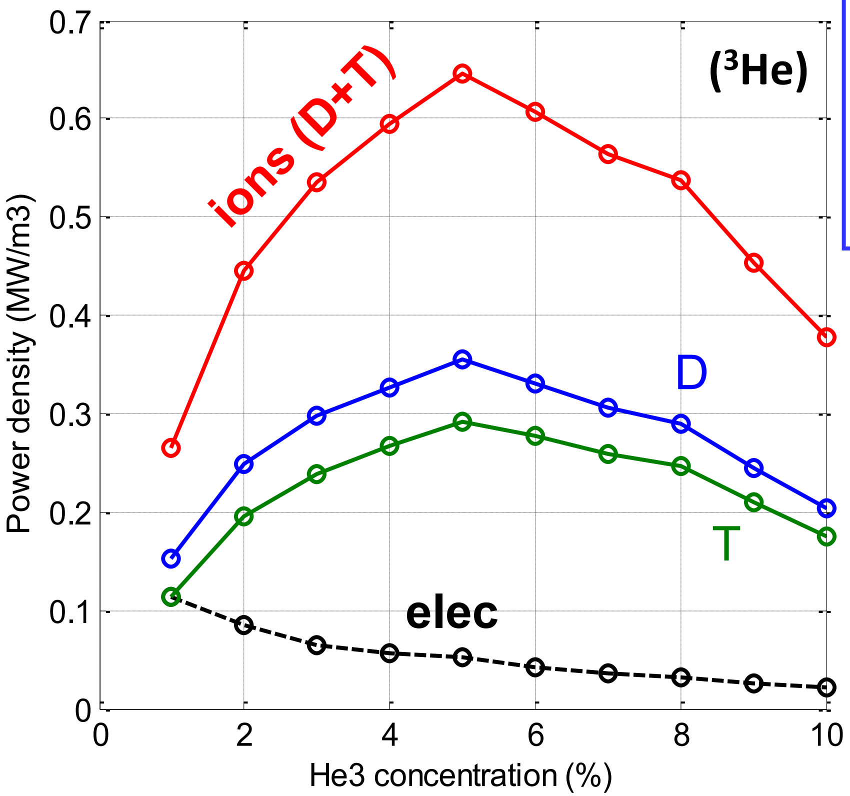
n_0=8e19/m^3, T_e/T_i=8/10keV

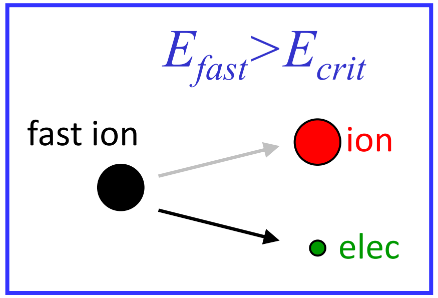
D:**T**, 2%D_{nbi}, 2%T_{nbi}, **^{3}He scan**, 1%Be



Collisional power redistribution of ^{3}He

(p_{dens}=0.7MW/m^3 at X[^{3}He]=5%)





- Good SPA for X[^{3}He]>1-2%
- Dominant N=1 **^{3}He** heating for X[^{3}He]<1%
- Electron LD and TTMP always important

- Dominant ion heating throughout
- Balanced bulk D and T heating

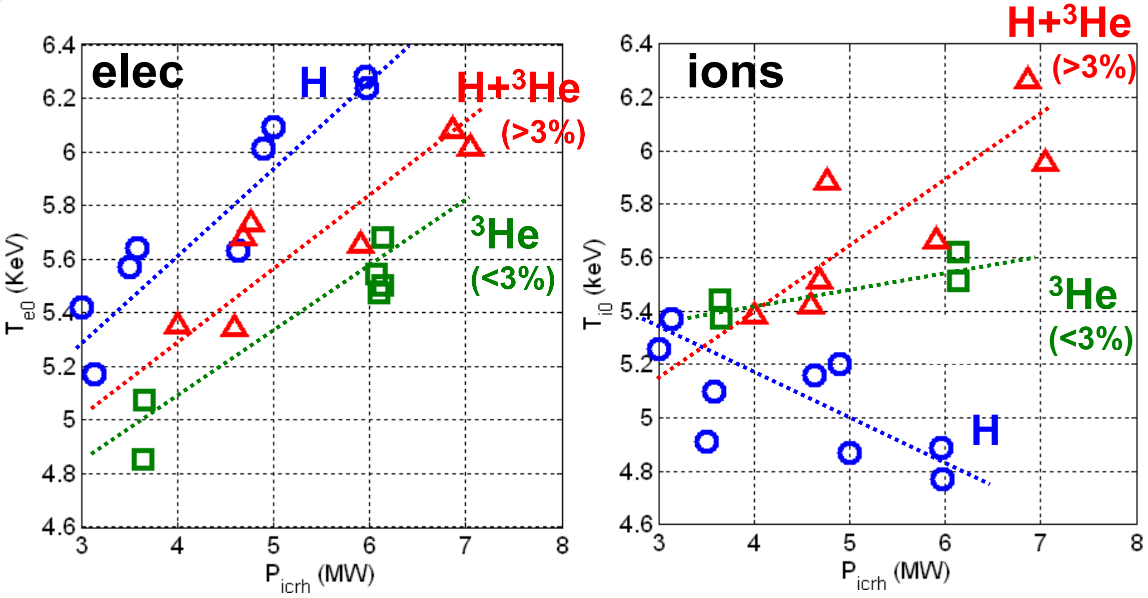
D. Gallart et al., 2018 Nucl. Fusion

Fundamental ^{3}He and N=2 T heating → experiments

JET-ILW: N=1 ^{3}He

D plasma with H, ^{3}He and H+^{3}He min. ICRH

(B_0=3.4T, f=33/51MHz)



- **N=1 ^{3}He** heating can provide **bulk ion heating** if concentration is right
- Combined H+^{3}He min. ICRH provides balanced electron and ion heating

D. Van Eester et al., IAEA 2017

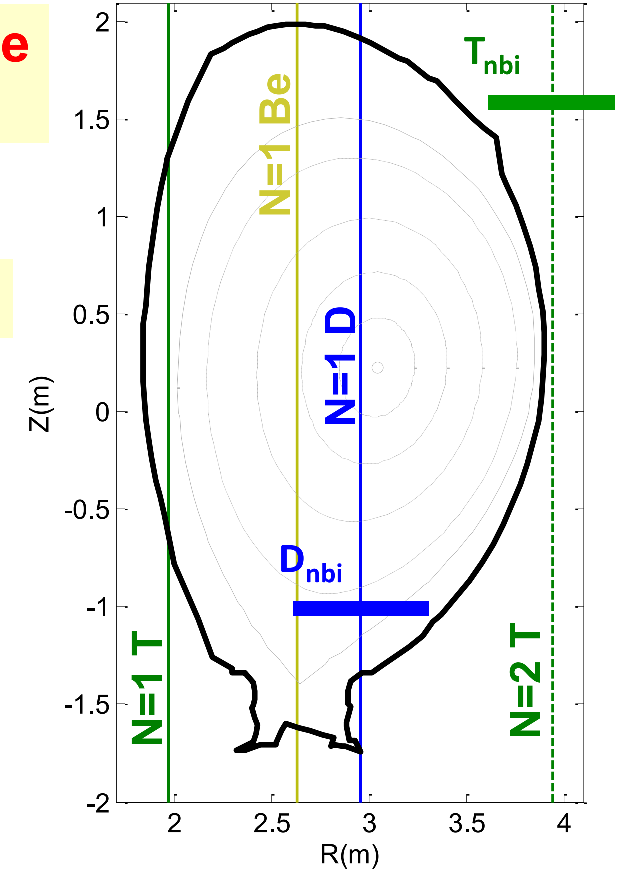
Fundamental D_{bulk} + D_{nbi} heating

ITER alternative
(T-rich)

JET fusion record

N=1 D (N=1 Be)

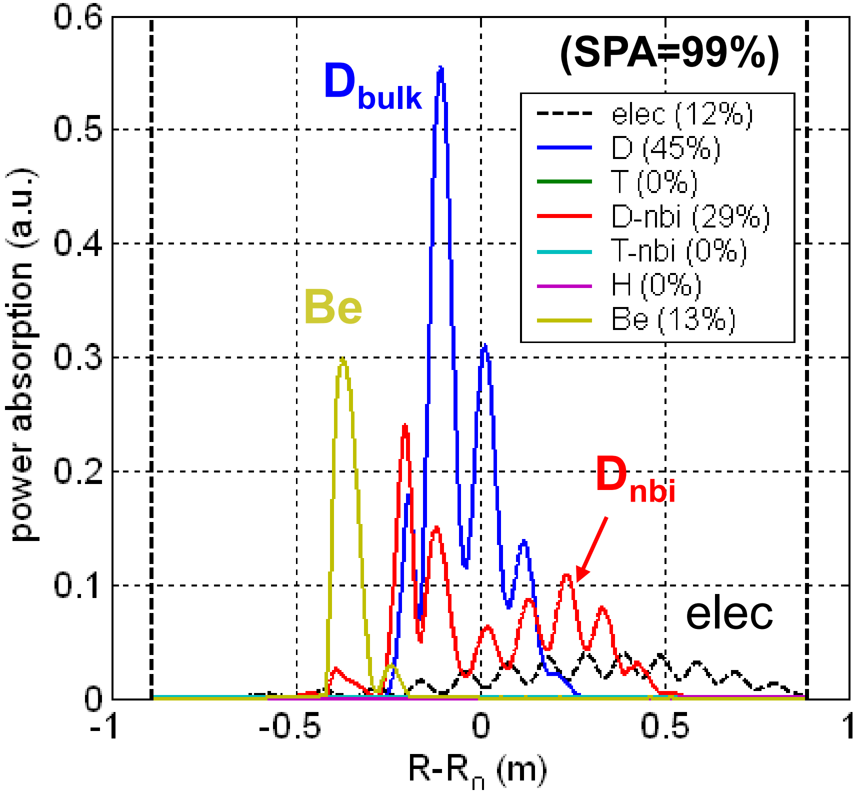
Bo=3.8T / f=29MHz



Fundamental D_{bulk} + D_{nbi} heating

n_0=8e19/m^3, T_e/T_i=8/10keV

15%D:T-rich, 4%D_{nbi}, 0%T_{nbi}, 1%Be

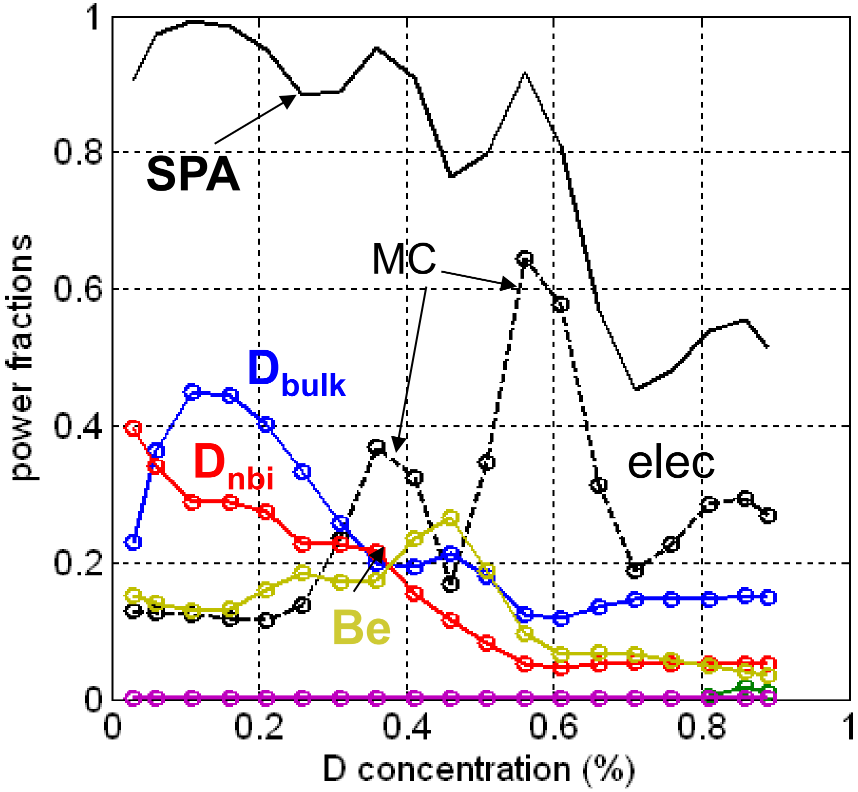


- Strong SPA
- Dominant N=1 D heating:

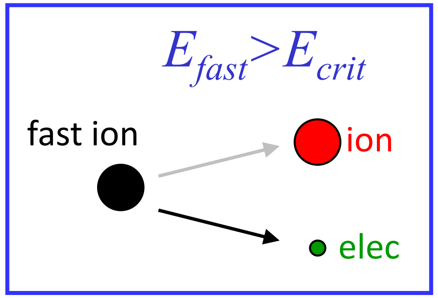
  D_{bulk} = 45%; **D_{nbi}=30%**
- Off-axis **Be** heating

Bo=3.8T / f=29MHz

D:T scan (4%D_{nbi}, 0%T_{nbi}, 1%Be)



- Good SPA up to X[D]=60%
- Dominant N=1 D_{bulk}+D_{nbi} htg. for X[D]<30%
- Some **Be** heating (~20%)



2 ways to achieve high fusion yield

Thermonuclear reactions:

→ Hot D and hot T ions (T>15-20keV)

→ α-heating, requires high performance (good confinement) plasmas

→ Working point for steady-state nuclear fusion reactors (ITER, DEMO, CFEDR, …)

Beam-target reactions:

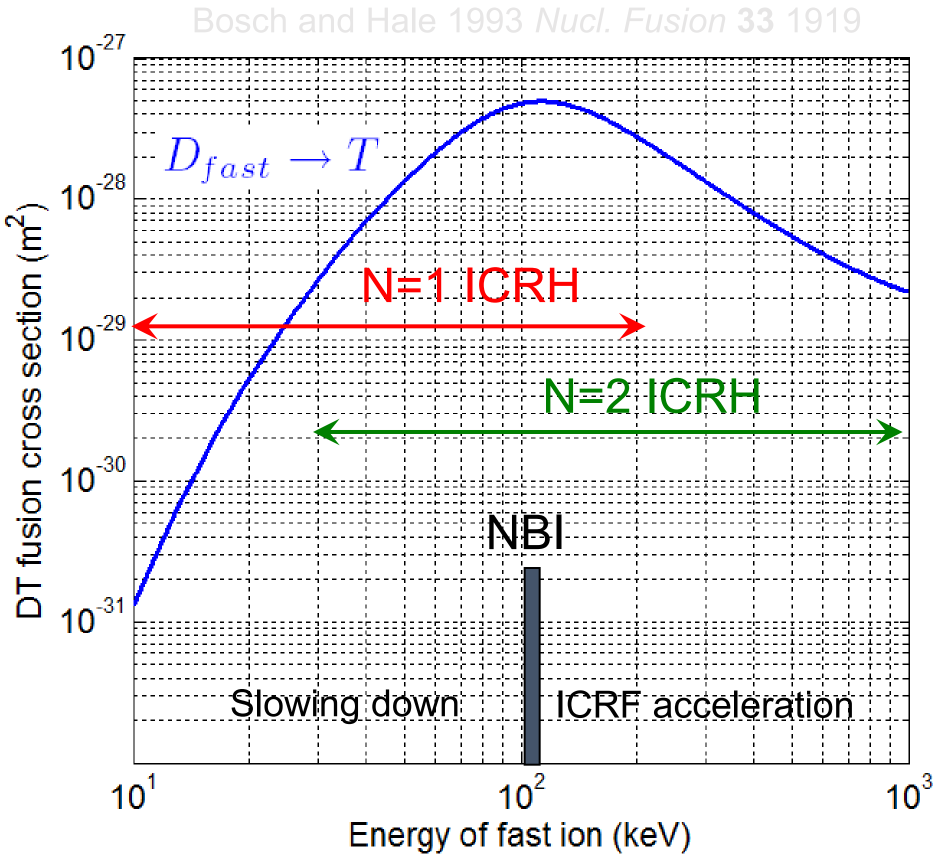
→ Fast D (120keV) colliding with thermal T plasma or fast T (180keV) colliding with thermal D plasma

→ Achieved with NBI or ICRH on medium performance plasma

→ Suitable for devices aiming at generating large D-T neutron fluence for material studies (VNS)

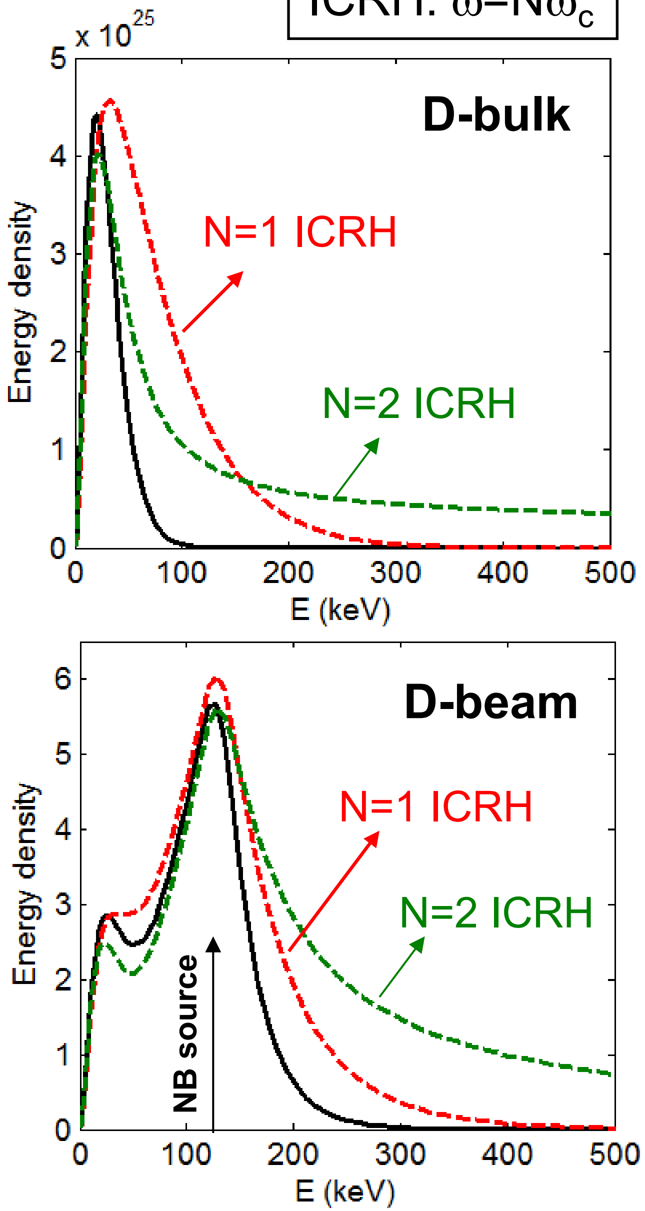
(about 50% of the JET fusion yield)

Beam target D-T fusion cross-section



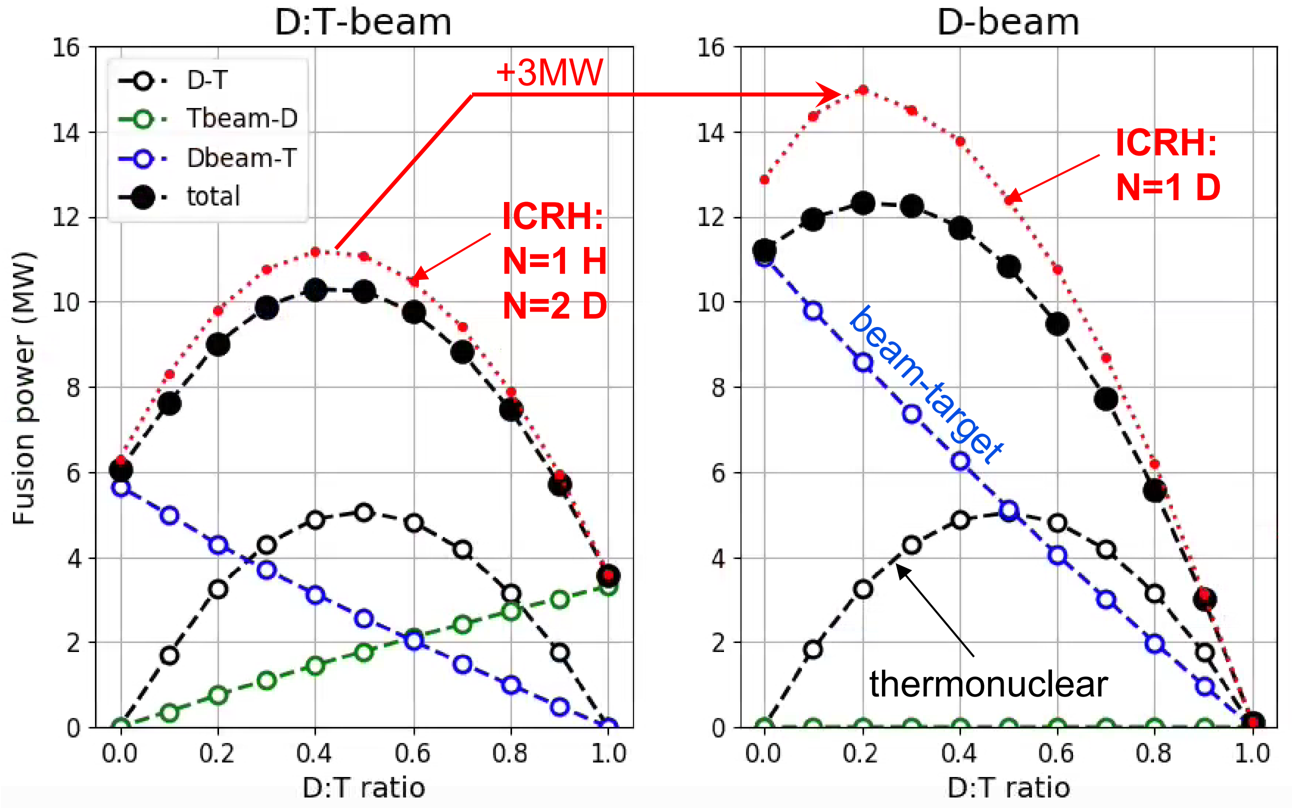
- D-NBI has optimal energy for D-T reactions in JET (120keV)
- N=1 ICRH is preferable to tailor the Deuterium distribution for D-T reactions

ICRH: $\omega = N\omega_c$



Impact of NBI species and isotope mix on fusion power

Hybrid-like parameters: n_{e0}=7.5x10^{19}/m^3, T_{e0}=10keV, T_{i0}=12keV, P_{NBI}=30MW (120keV)



D:T beam:

- Max. fusion with balanced D:T ratio
- For D:T = 50:50
  45% thermonuclear
  55% beam-target

D-beam:

- Max. fusion with D:T ratio < 0.3
- With D:T = 20:80
  25% thermonuclear
  75% beam-target

- T-rich plasmas with D-nbi and N=1 D ICRH allows maximizing fusion power in JET (‘beam’-target reactions)

- Expect about 3MW of fusion enhancement with 5MW ICRH

[Lerche et al., AIP2023]

Impact of NBI species and isotope mix on fusion power

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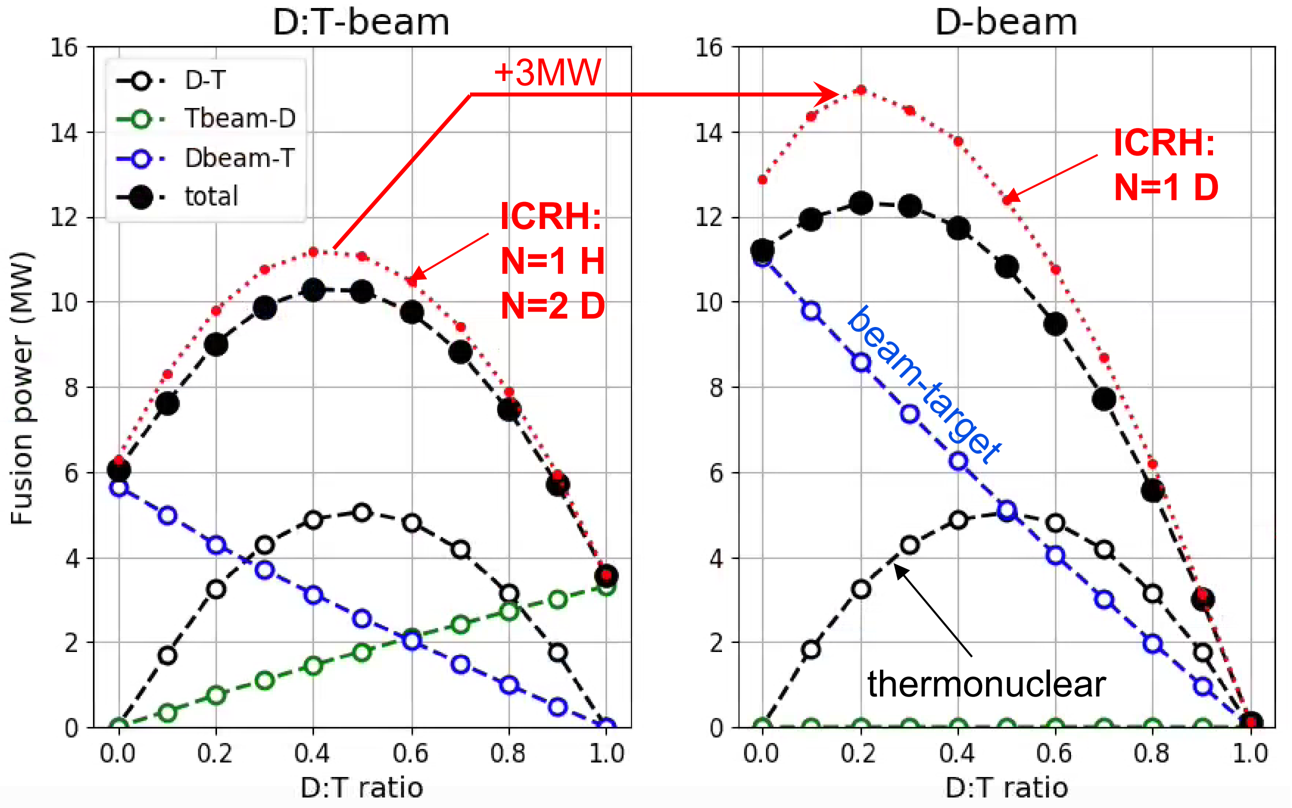

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- With D:T = 20:80
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  75% beam-target

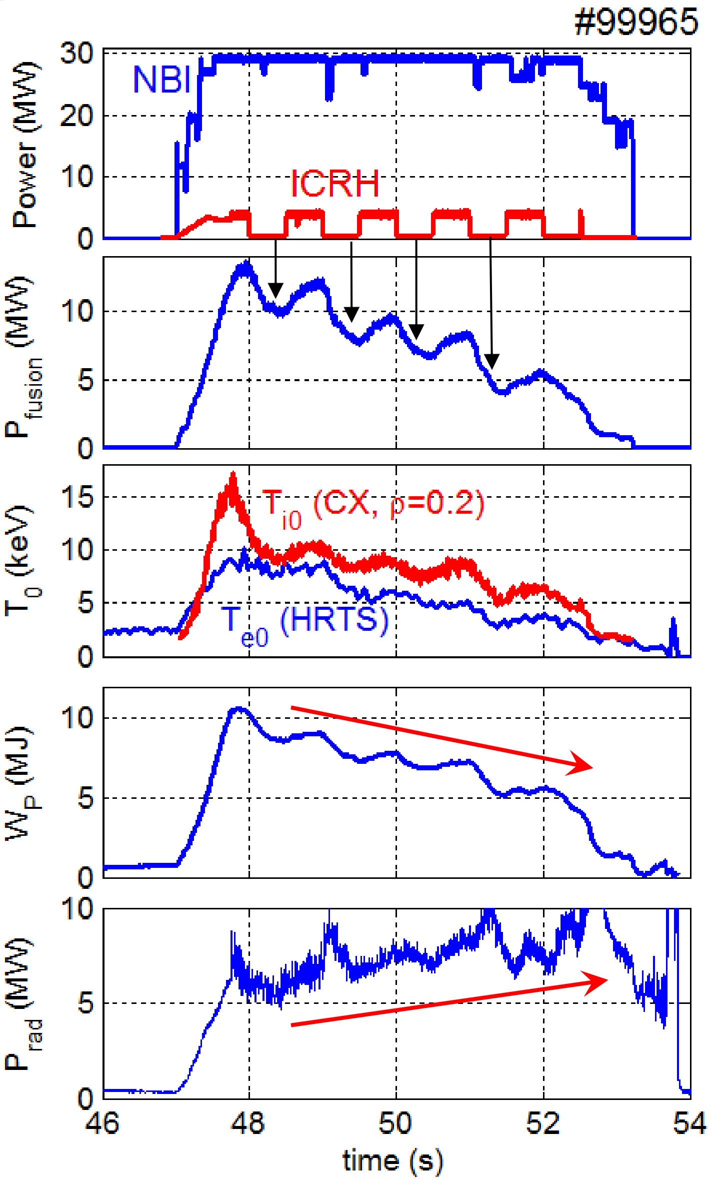
- T-rich plasmas with D-nbi and N=1 D ICRH allows maximizing fusion power in JET ('beam'-target reactions)

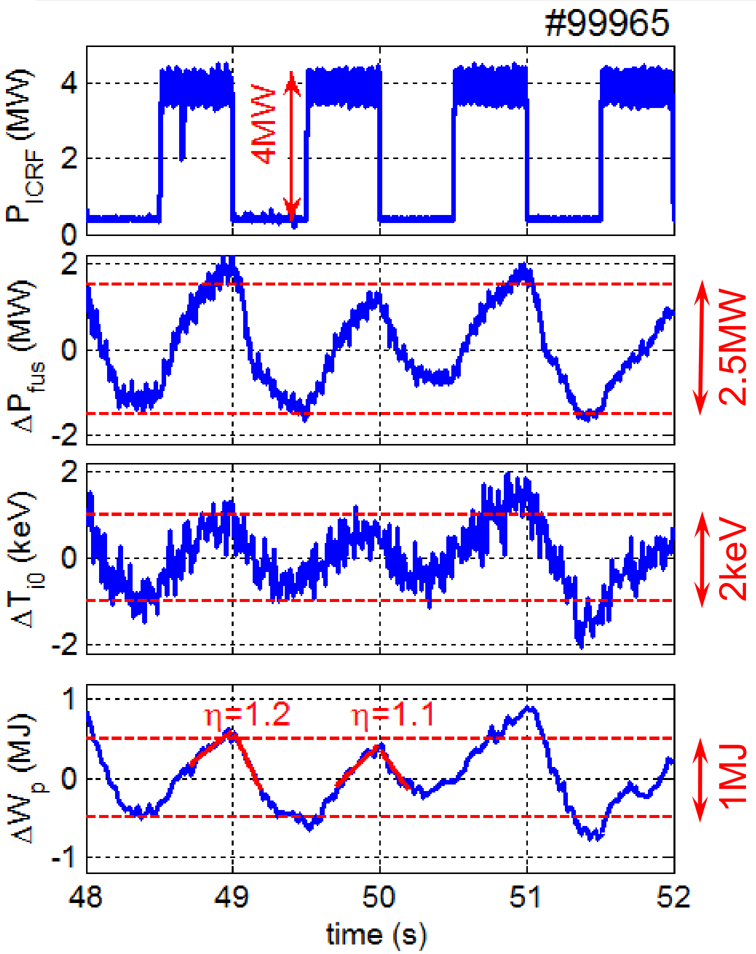
Done prior to the expts.

- Expect about 3MW of fusion enhancement with 5MW ICRH

[Lerche et al., AIP2023]

ICRH impact on D-T fusion power





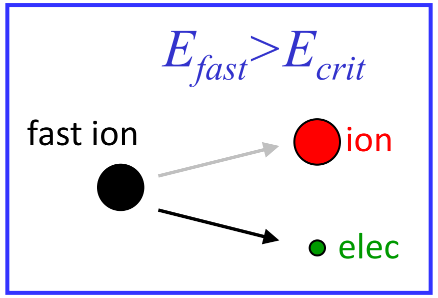
ΔP_{RF} = 4MW

ΔP_{fus} > 2.5MW

(Q_{ICRH}>0.6)

ΔT_{i0} ~ 2keV

ΔW_P ~ 1MJ



$E_{fast} > E_{crit}$

fast ion

ion

elec

- Excellent RF wave absorption with N=1 D heating (break-in-slope)

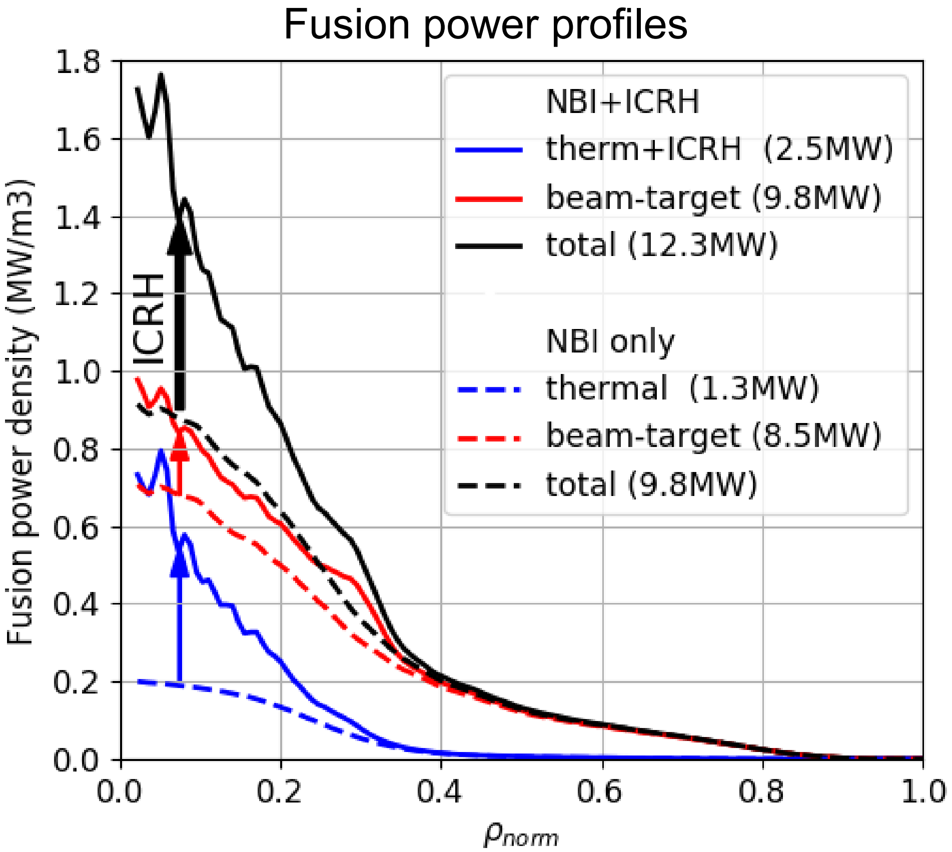
[Mantica et al., IAEA-FEC 2023]

[Lerche et al., AIP2023]

- Fusion power strongly correlated with ICRH power waveform

Neutron enhancement due to ICRH (ETS H&CD modelling)

Pulse #99971 (49s): All ion distributions are non-Maxwellian



P_{fus}= 9.8MW → 12.3MW (+2.5MW with 4MW ICRH)

thermonuclear: 20%
'beam'-target: 80%

Note: In the core (ρ<0.2), the ratio 'thermal' / BT is larger (33 : 67)

ICRF fusion enhancement:

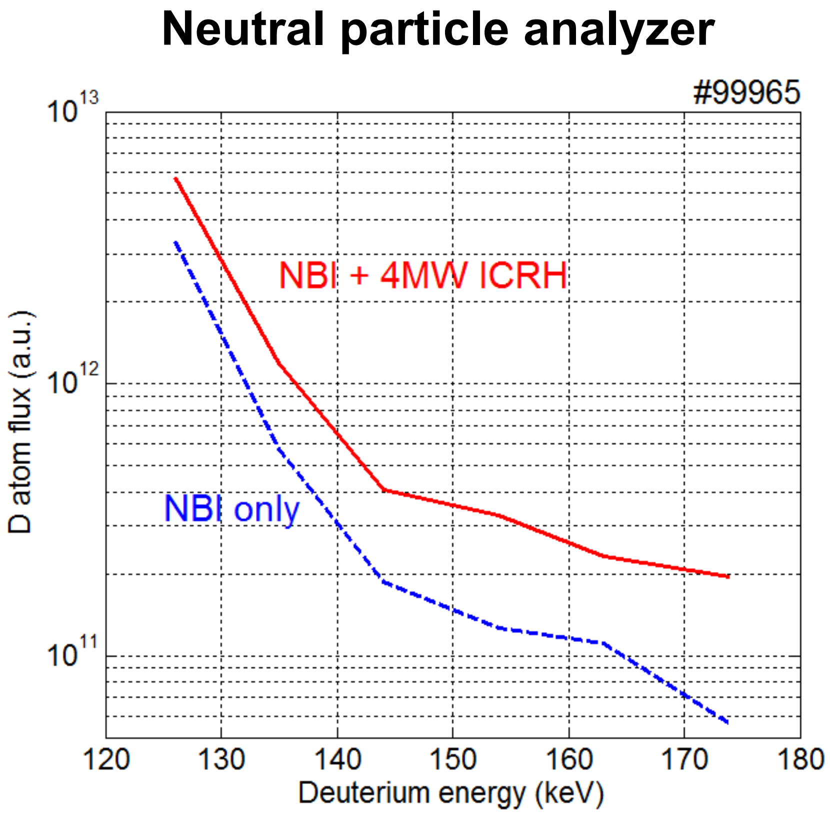
- η_{ICRF}= 2.5/9.8 = **25%**
- Q_{ICRF}= 2.5/4.0 = **0.62**

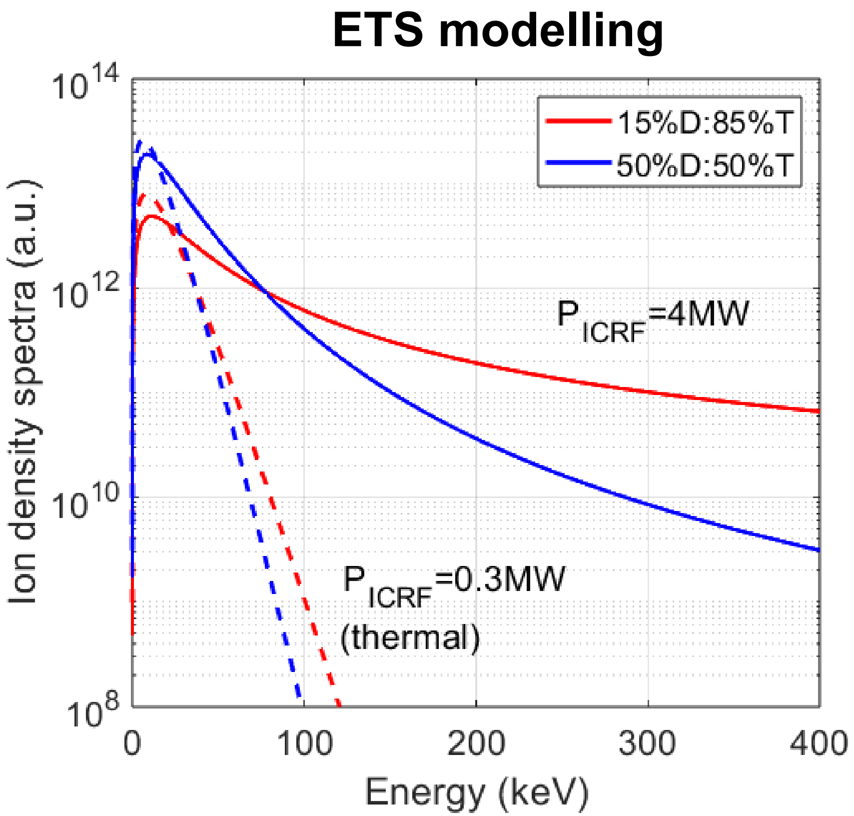
Done after the expts.

- Beam-target reactions are dominant but the ICRH enhancement is stronger on the thermonuclear component in the plasma centre
- Predicted RF fusion enhancement is in very good agreement with experimental measurements (incl. neutron spectroscopy)

[Lerche et al., AIP2023]
[Nocente et al., NF2023]
[Mantica et al., IAEA2023]
[Lerche et al., NF2026]

ICRH effect on fast Deuterium tails





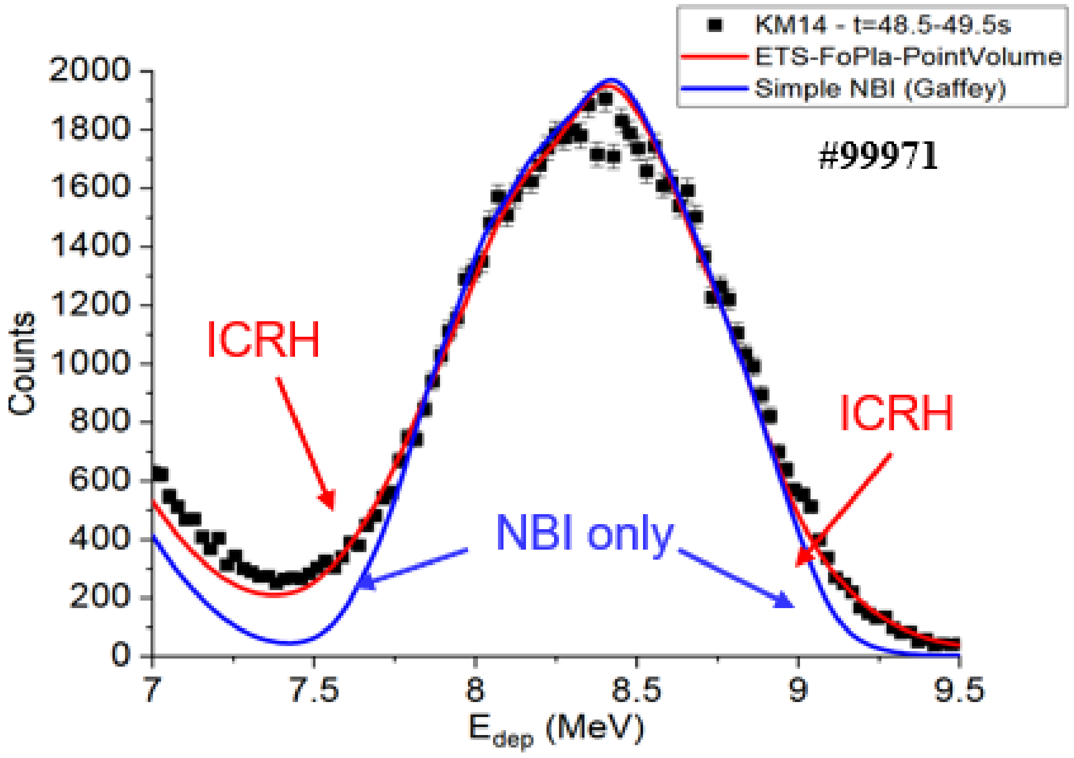
- Substantial increase of D_{fast} population in E=120-170keV range
- Supported by ETS H&CD modelling

[Kiptily et al., IAEA-FEC 2023]

[Lerche et al., NF2026]

ICRH effect on D-T neutron spectra

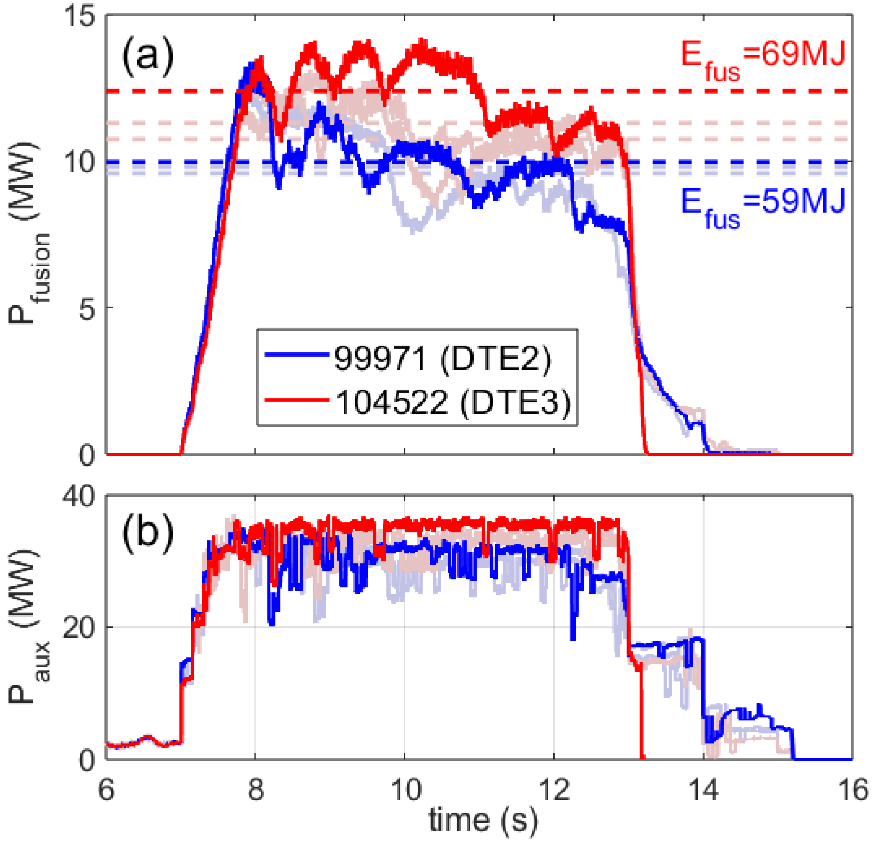
Diamond neutron spectrometer

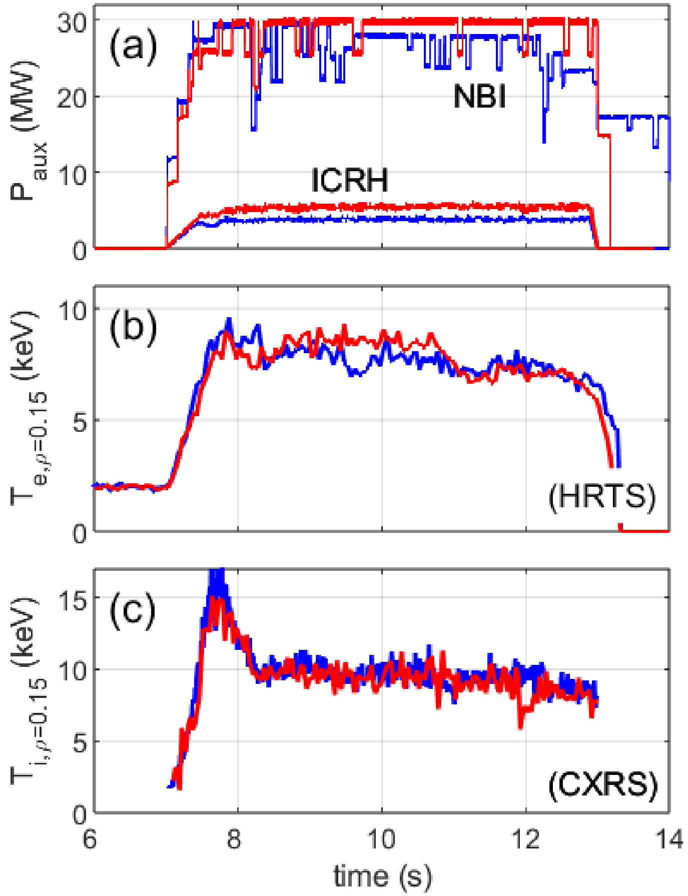


- Neutron spectrometry suggests 20-25% thermal contribution to 14MeV neutrons (75-80% beam-target fusion)
- Spectral broadening of the neutron spectra is consistent with ICRH modelling

[Nocente et al., IAEA-FEC 2023] *[Kiptily et al., IAEA-FEC 2023]*

Best results (record D-T fusion power) require reliable input power

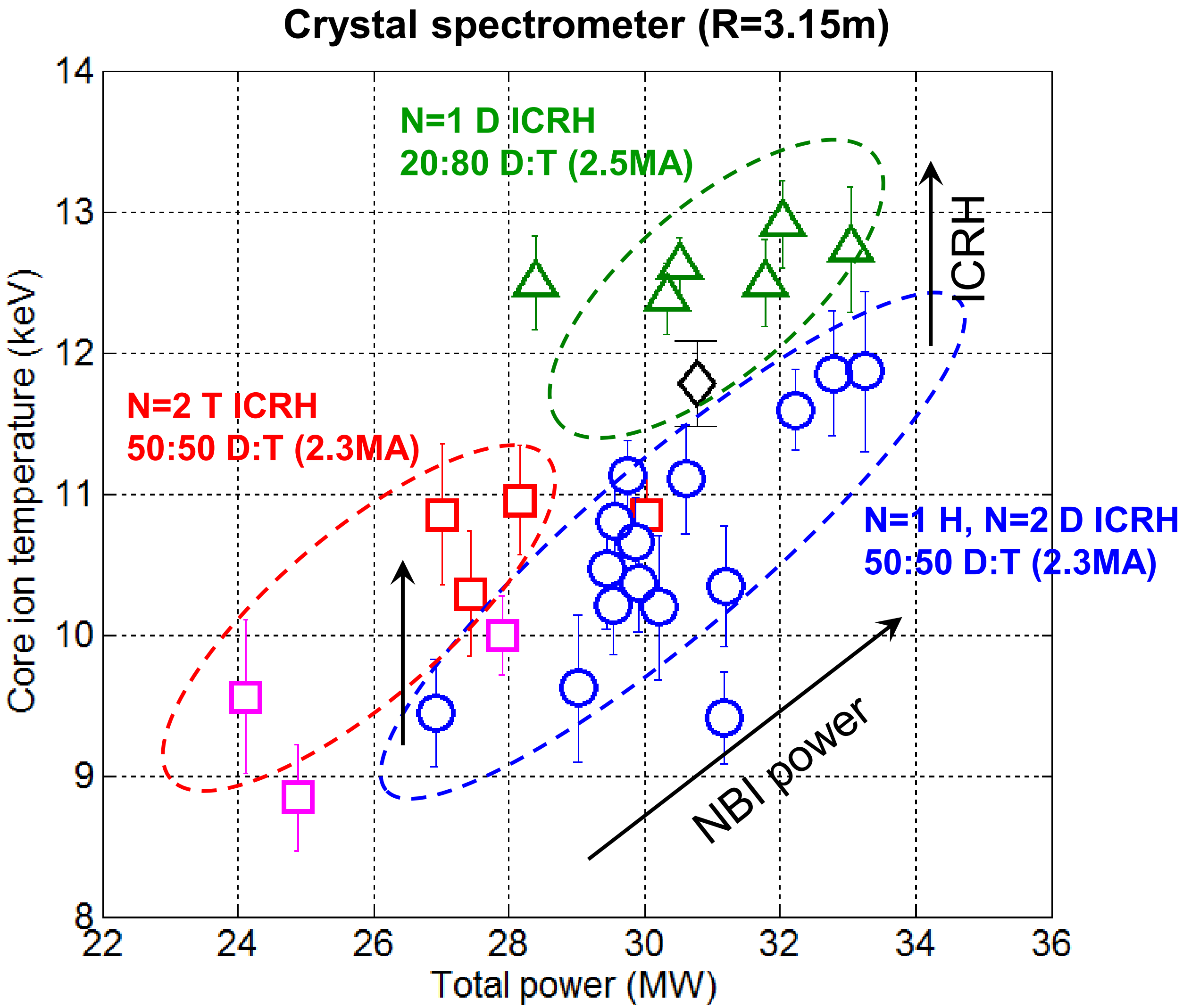




- Steady NBI power → Larger fusion yield
- Extra 1.5MW of ICRH → Larger fusion enhancement + stationarity (impurity screening)

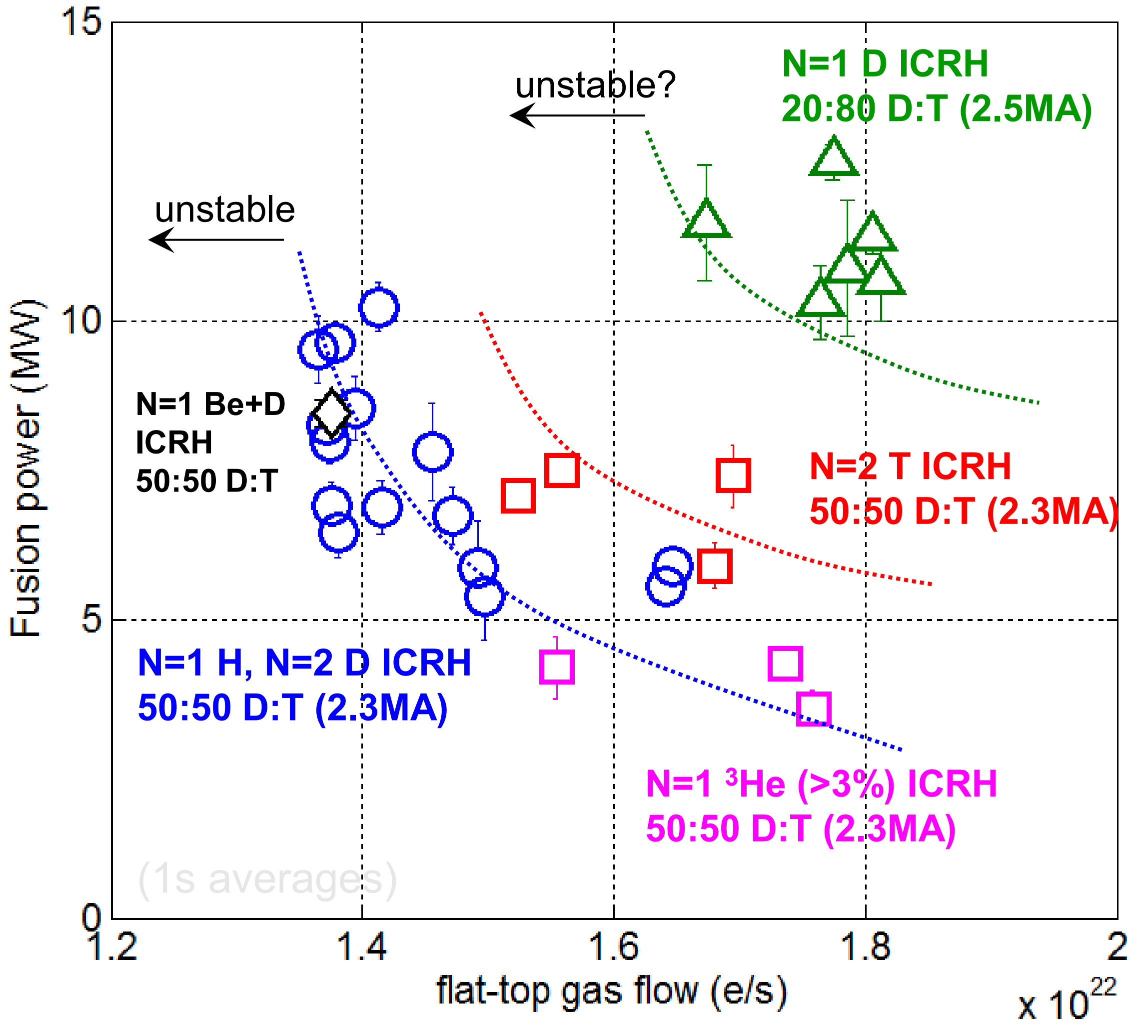
Core ion temperature vs input power

- Core ion temperature increases with total input power (E_{crit}~150-200keV)
- ICRH scenarios with predominant ion heating feature higher core T_i for same input power



Overview: Fusion power vs. fuelling rate

- Increased fusion power as gas flow rate is reduced (larger pedestal t
- Improved P_{fus} with **N=2 T ICRH** heating at medium gas level (but lower NBI)
- Highest P_{fus} in T-rich plasmas with **N=1 D ICRH** despite higher gas injection,



unstable = Slow ELM's, imp. accumul.

Less core electron heating = Higher gas flow threshold for stationarity:

N=2 T ICRH:

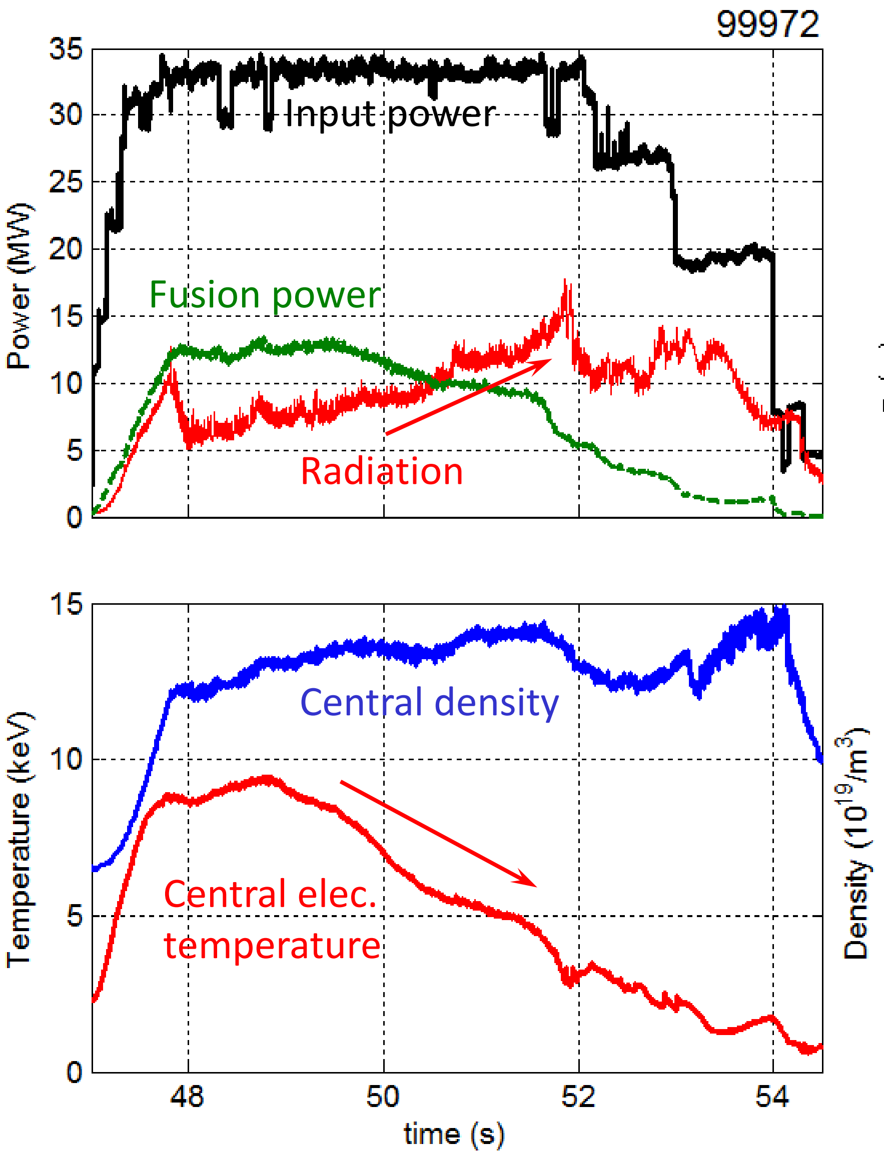
- Marginal core electron heating ?

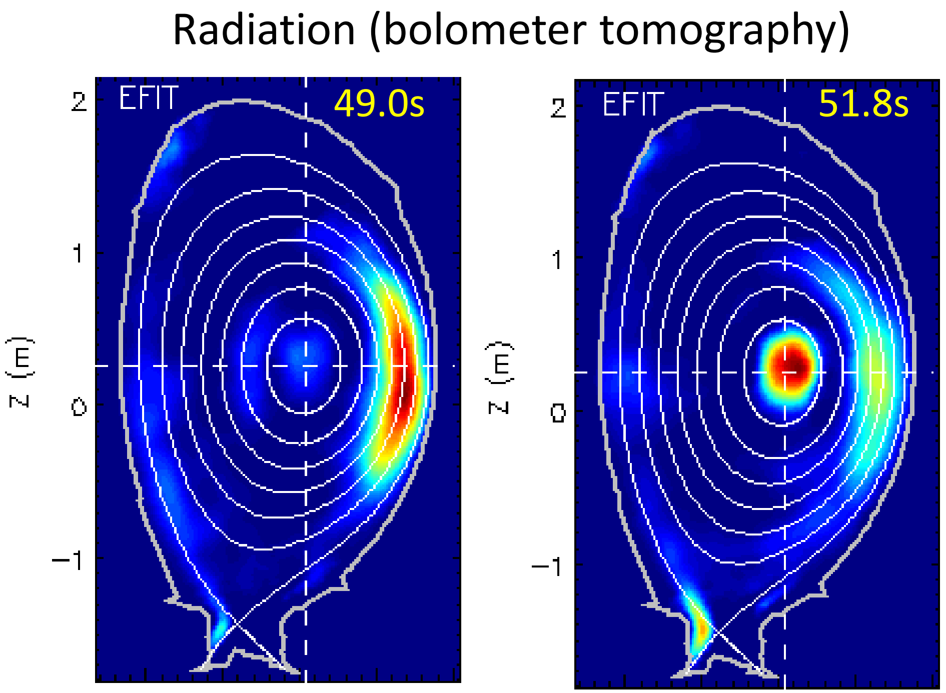
N=1 D ICRH:

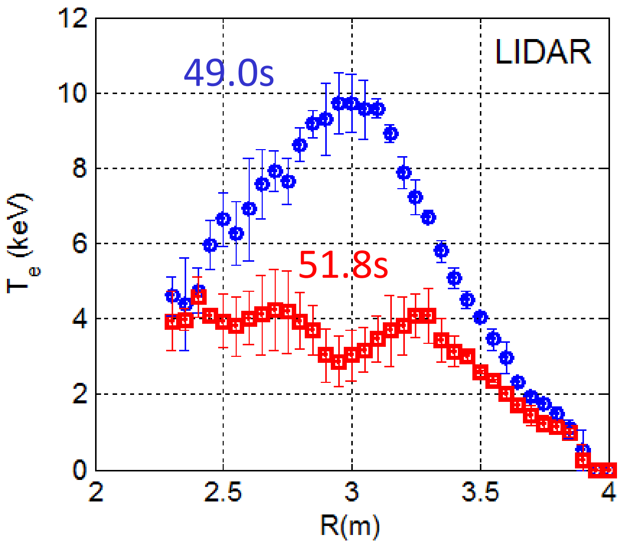
- Higher I_P (needs more gas)
- Marginal core electron heating?

VNS example

Core impurity accumulation

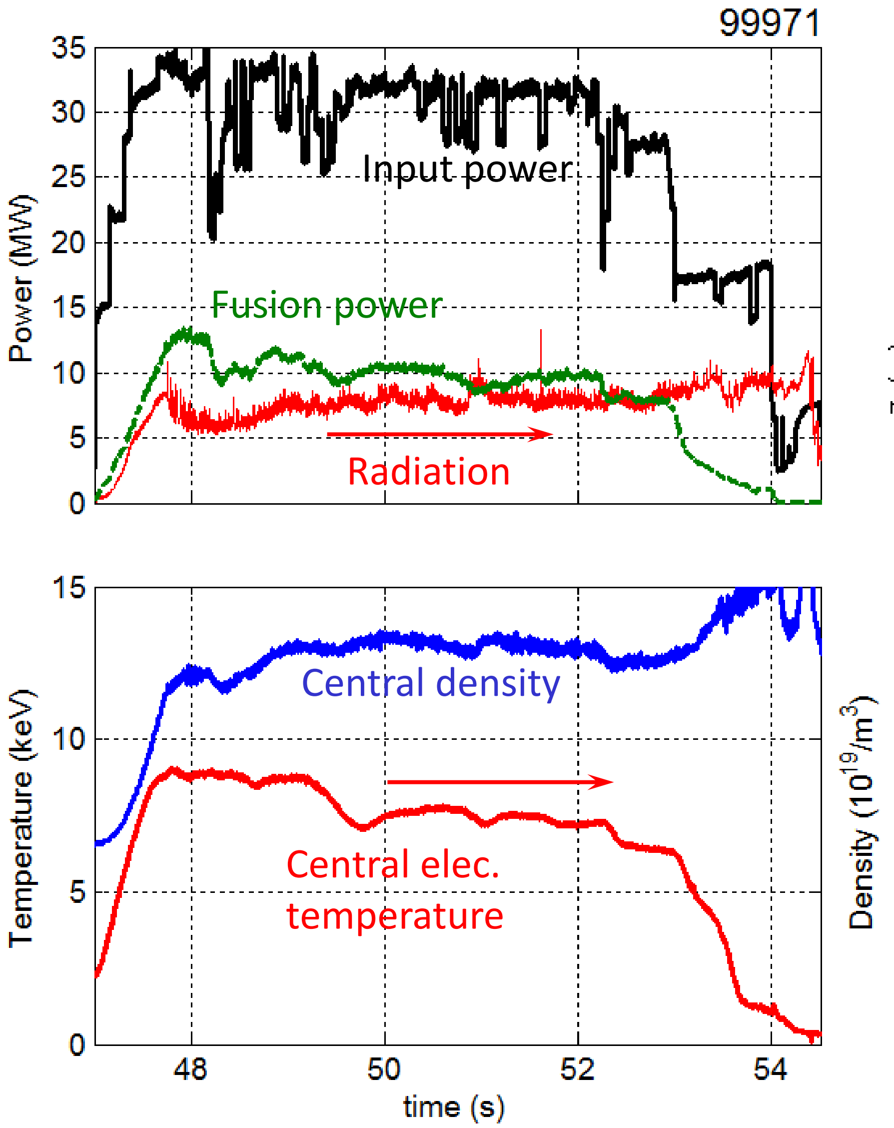




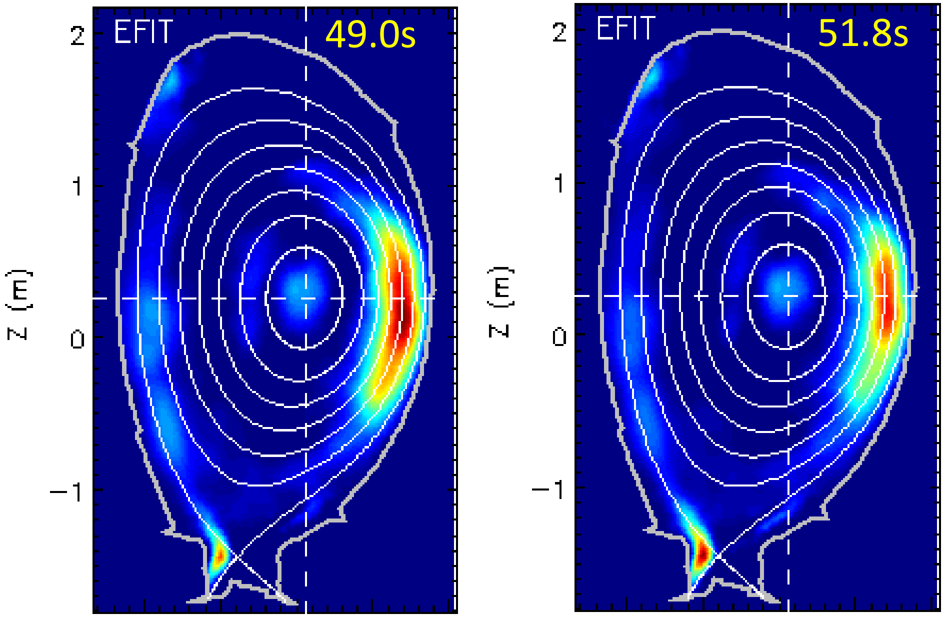


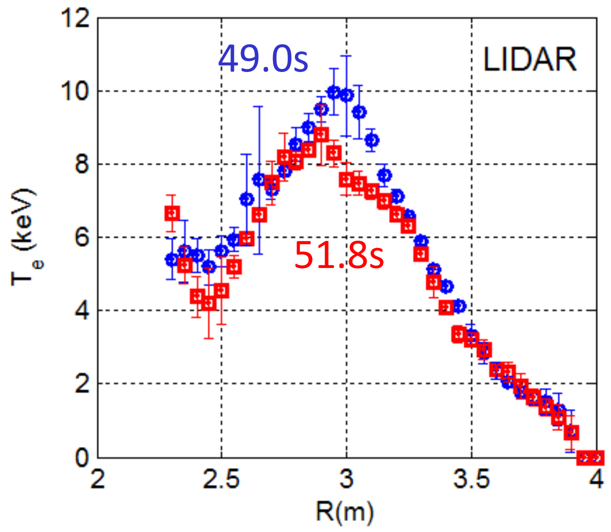
- Core radiation → lower T_{e0} → more radiation → hollow T_e

(No) Core impurity accumulation



Radiation (bolometer tomography)

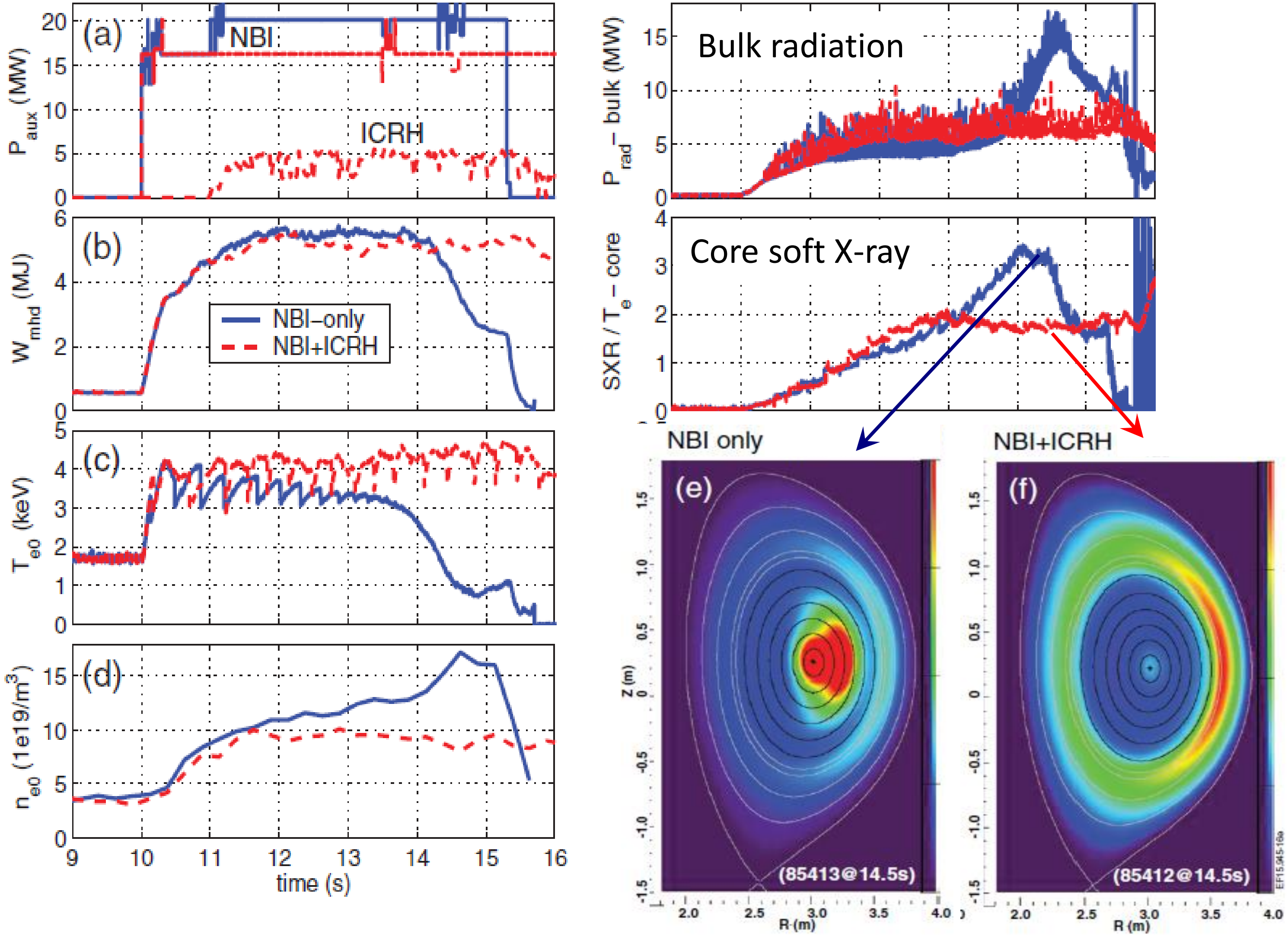




- Radiation roughly constant; kept in the outer part of the plasma → T_e profile stays peaked

Core impurity control using ICRH

- D-D Neon seeded baseline discharge (2.7T/2.5MA)



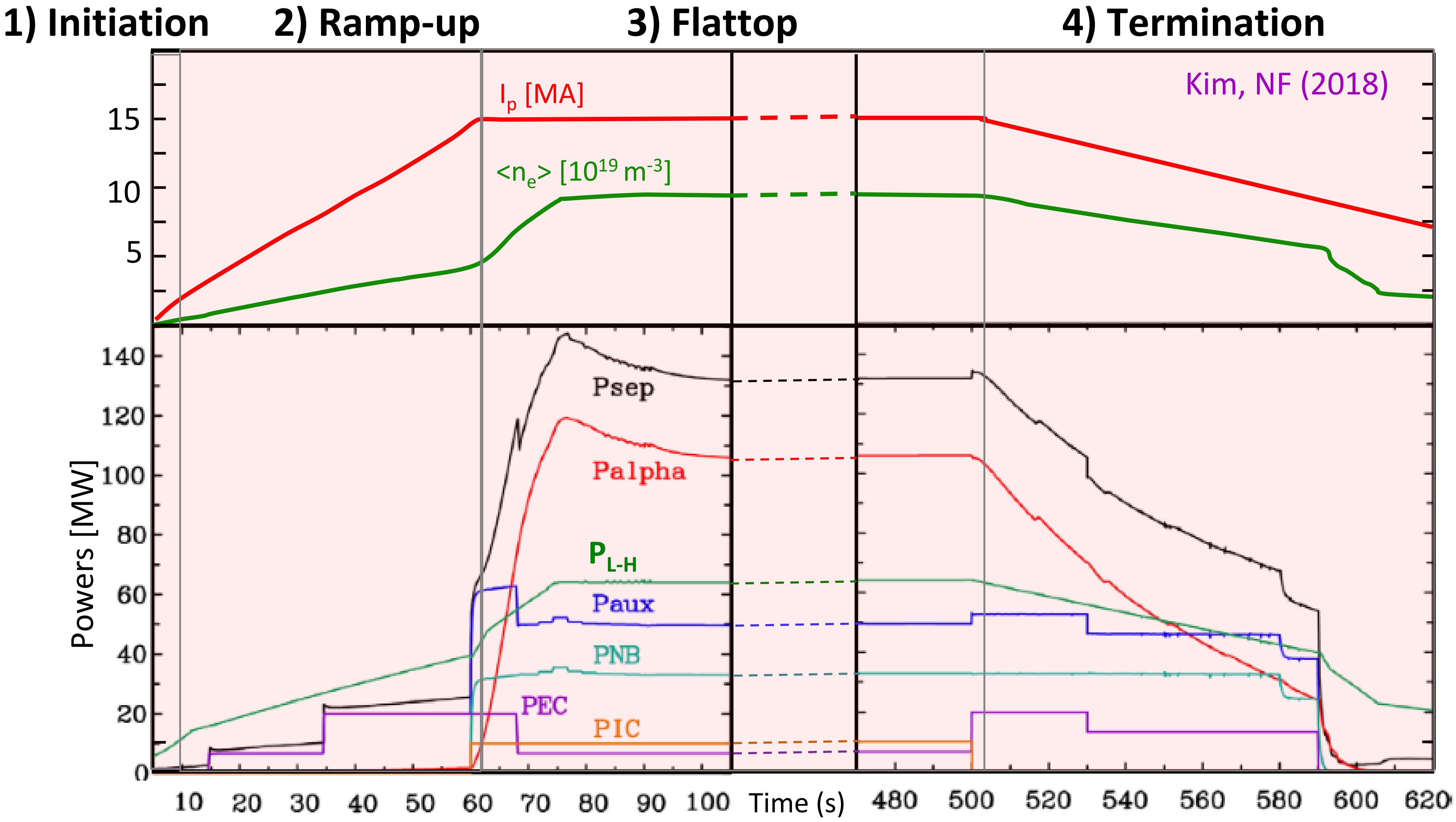
[Lerche, Nucl. Fusion 2016]

- Central ICRH is key for impurity control in high performance JET plasmas (elec. heating)

Outline

- Brief introduction to thermonuclear fusion
- Auxiliary heating systems in tokamaks
  - Introduction
  - Neutral beam injection (NBI)
  - Ion cyclotron heating (ICRH)
  - Electron cyclotron heating (ECRH)
- Overview of JET D-T fusion results
- The role of NBI on JET high performance D-T plasmas
  - Impact of NBI on H-mode access
  - Impact of NBI on flat-top performance
- Impact of ICRH on D-T plasmas & fusion performance
  - H minority heating examples
  - He^3 minority heating examples
  - Fundamental Deuterium and DNBI heating (D-T fusion record)
  - ICRH for impurity control
- Some ITER predictive D-T simulations
  - Access to burn-through plasmas
  - Impact of ICRH power on fusion performance

H&CD in a typical ITER pulse (DT baseline 15 MA / 5.3T)

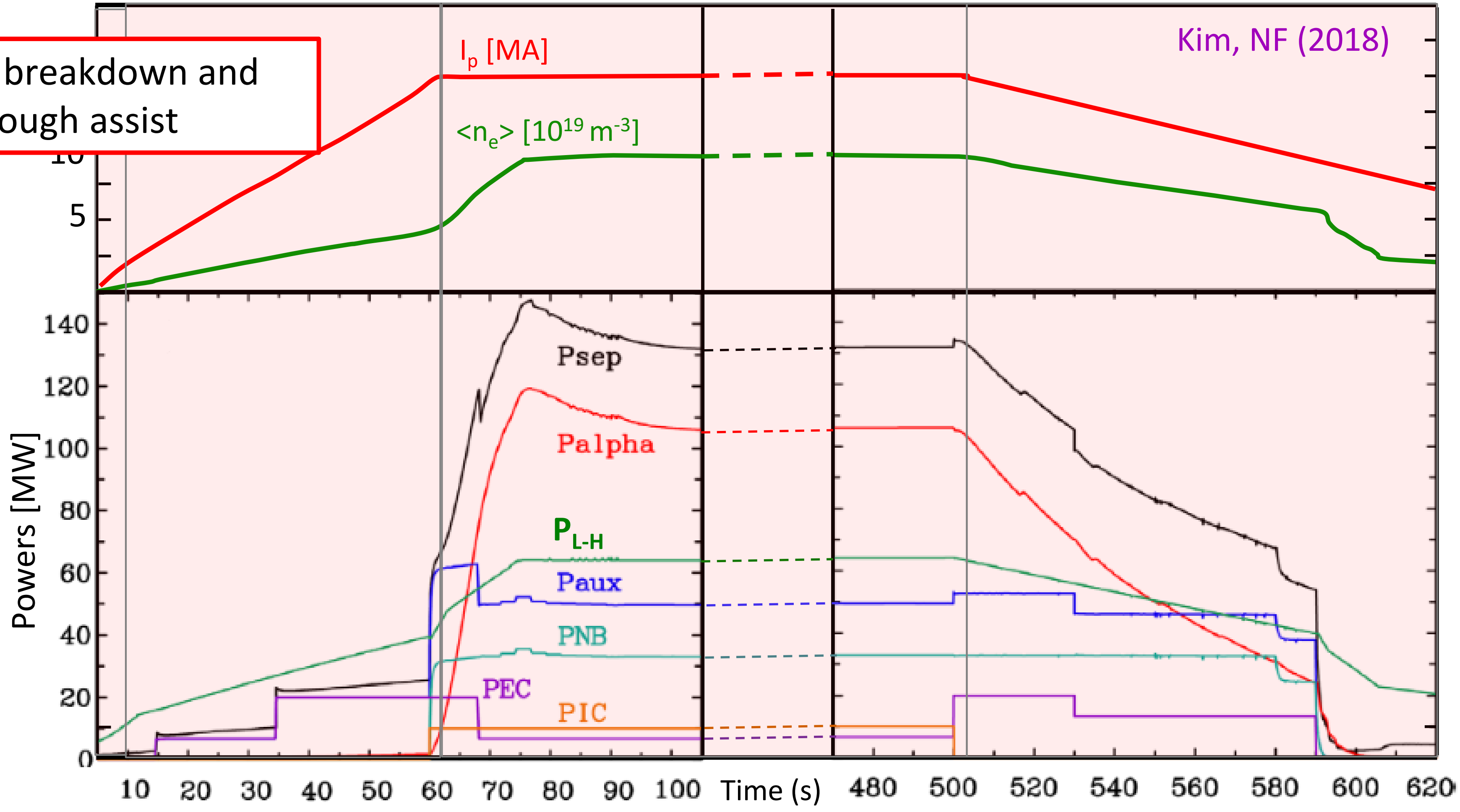


Courtesy of M. Schneider

H&CD in a typical ITER pulse (DT baseline 15 MA / 5.3T)

1) Initiation **2) Ramp-up** **3) Flattop** **4) Termination**

ECH for breakdown and burnthrough assist



Courtesy of M. Schneider

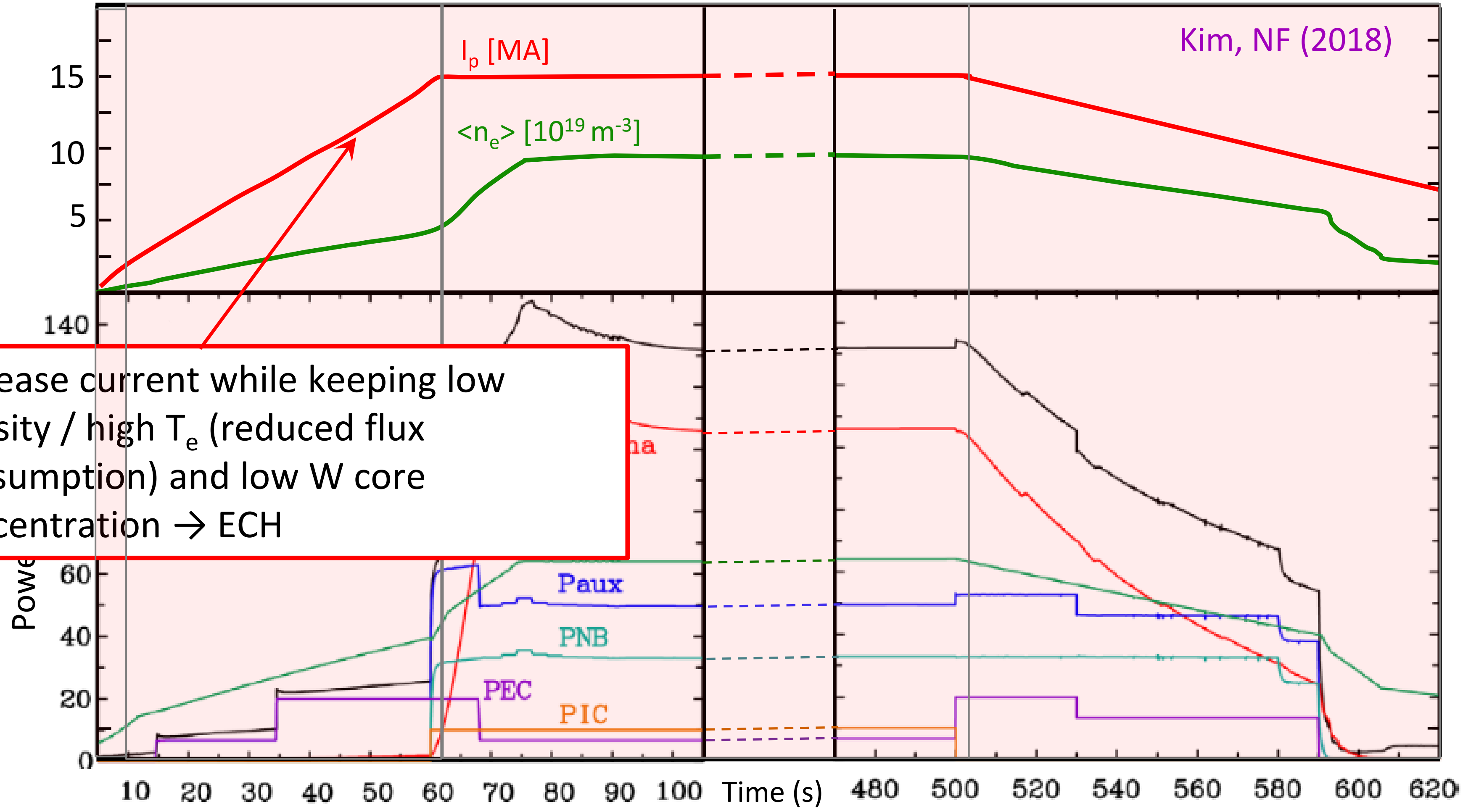
H&CD in a typical ITER pulse (DT baseline 15 MA / 5.3T)

1) Initiation

2) Ramp-up

3) Flattop

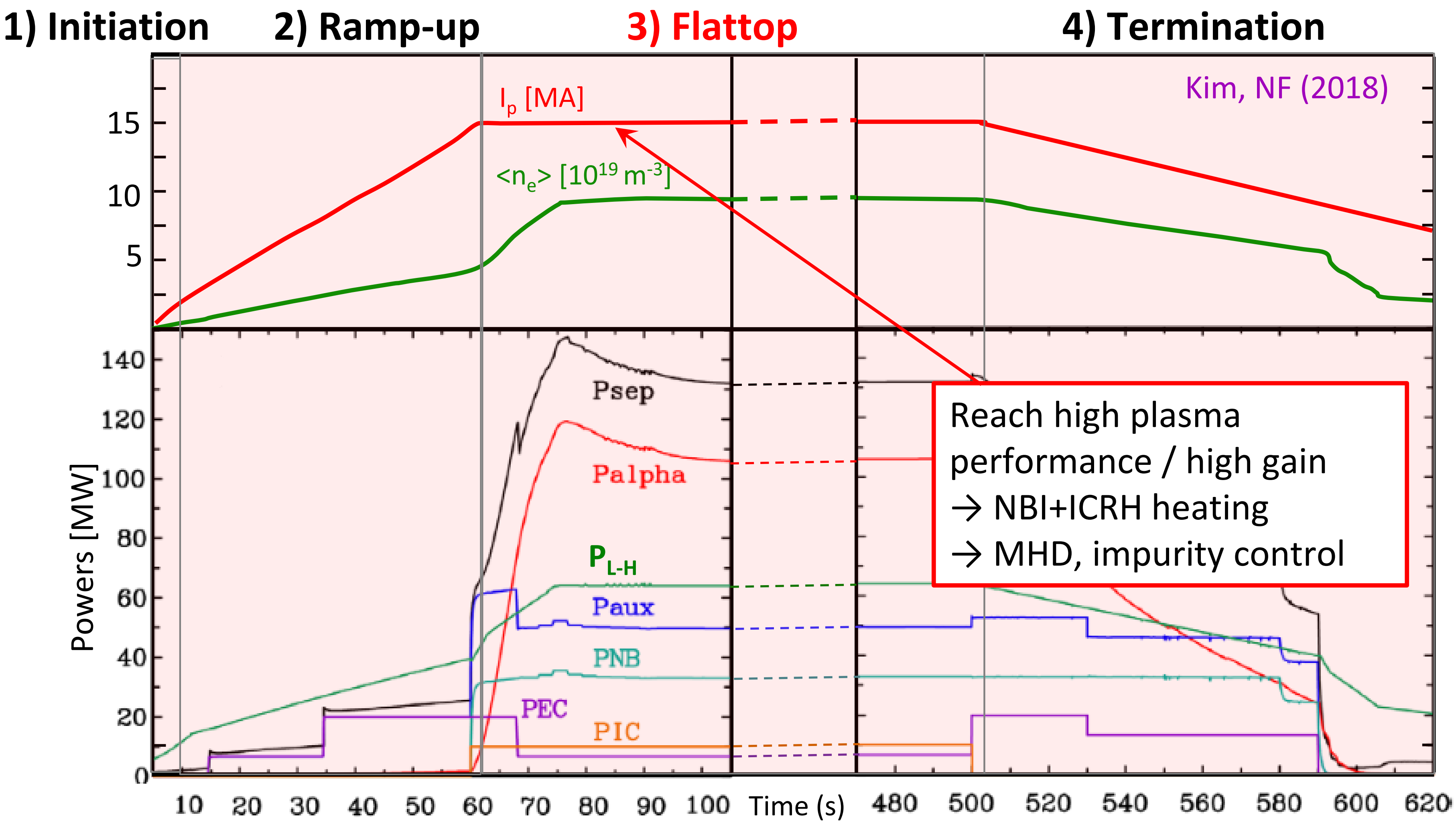
4) Termination



Increase current while keeping low density / high T_e (reduced flux consumption) and low W core concentration → ECH

Courtesy of M. Schneider

H&CD in a typical ITER pulse (DT baseline 15 MA / 5.3T)



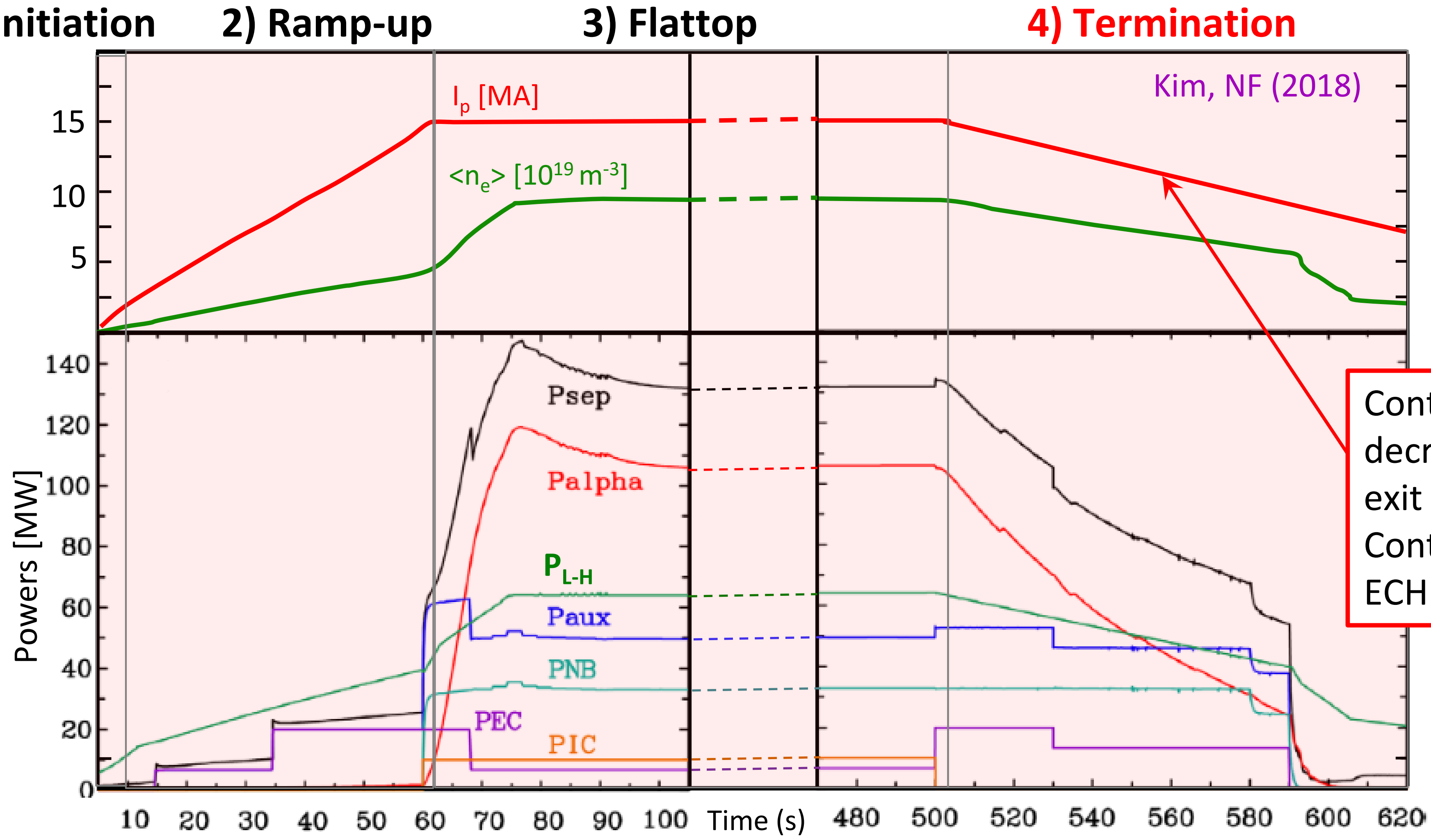
Reach high plasma performance / high gain

→ NBI+ICRH heating

→ MHD, impurity control

Courtesy of M. Schneider

H&CD in a typical ITER pulse (DT baseline 15 MA / 5.3T)

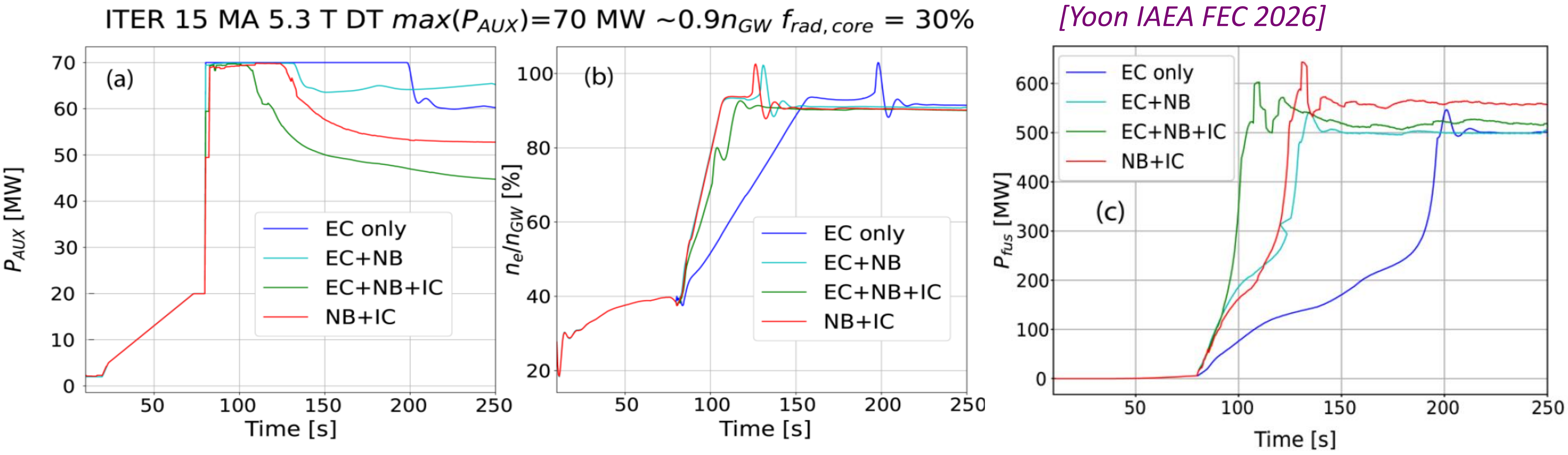


Control plasma energy decrease during H-mode exit to avoid wall contact
Control particle exhaust
ECH to avoid impurities

Courtesy of M. Schneider

Heating mix and fusion gain

- Build-up of alpha heating is essential to access burn plasma conditions
- Density ramp should not be too fast → Tailoring $P_{H\&CD}/\langle n_e\rangle$ is critical
- Too fast n_e ramp → Ti drops → Alpha heating drops → No H-mode access



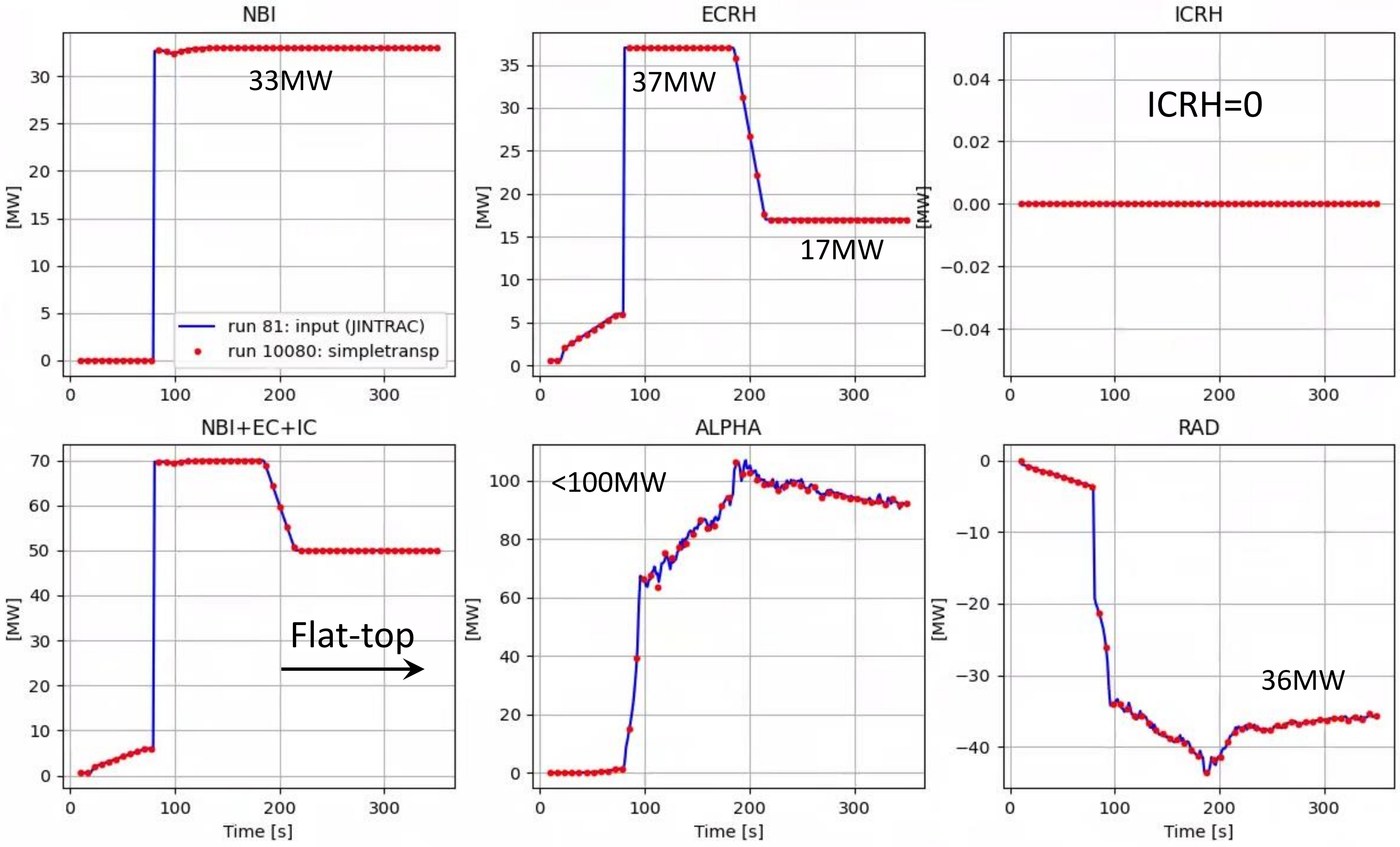
[Yoon IAEA FEC 2026]

- ECRH only needs slow density ramp for equipartition to take place (larger T_i) to ensure large fusion power in the flat-top phase
- NBI+ICH based scenarios allow for higher fusion gain (due to more ion heating)

ITER modelling: D-T shot=53301

JINTRAC (run 80)

ETS H&CD (verification using JINTRAC transport coefficients)



33MW

37MW

17MW

ICRH=0

Flat-top

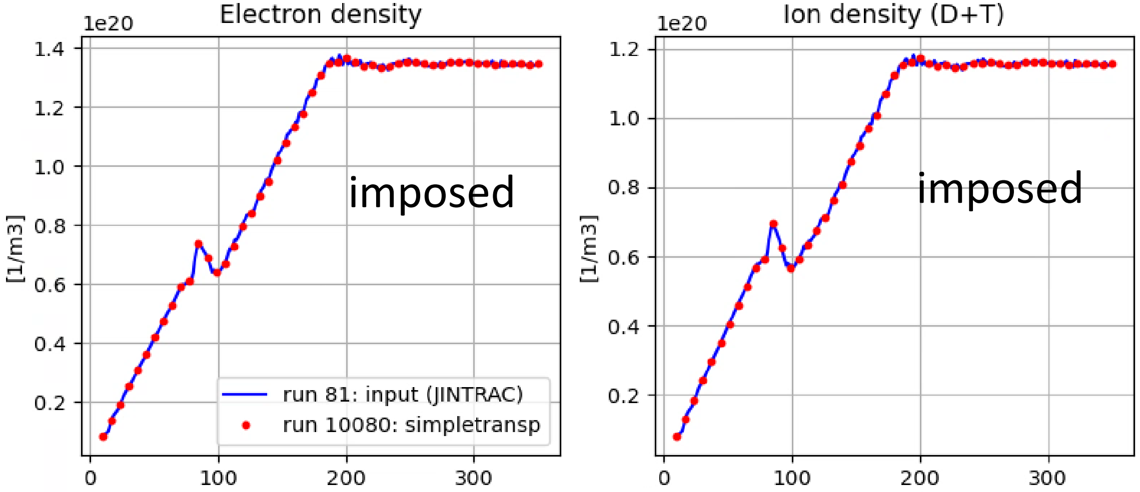
<100MW

36MW

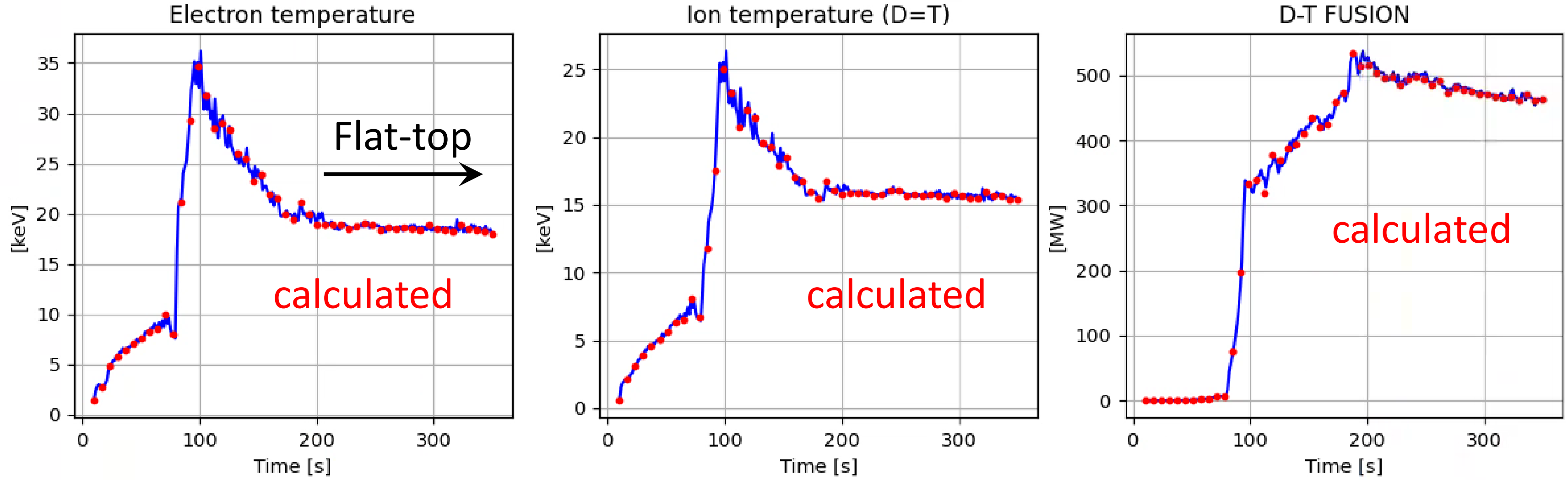
ITER modelling: D-T shot=53301

JINTRAC (run 80)

ETS H&CD (verification using JINTRAC transport coefficients)

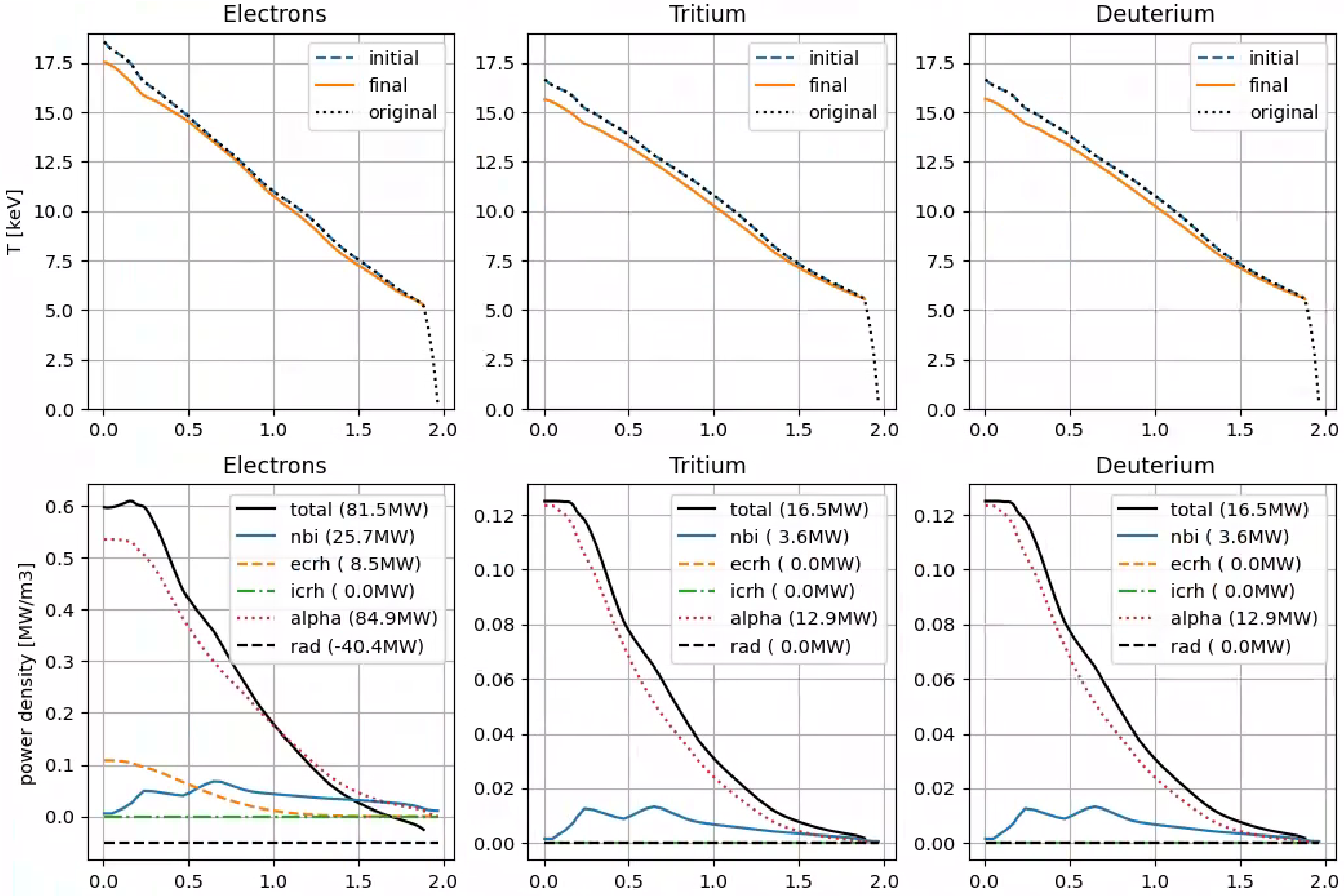


Perfect agreement (by definition)



ITER modelling: P_{icrh} = 0MW

shot=53301 , run_in=481 , run_out=8481 , time=300.0 ,
Bo=5.3T / Ip=15MA Pnbi=33.0MW, Pecrh= 8.5MW, Picrh= 0.0MW, Palpha=110.7MW, Prad=-40.4MW

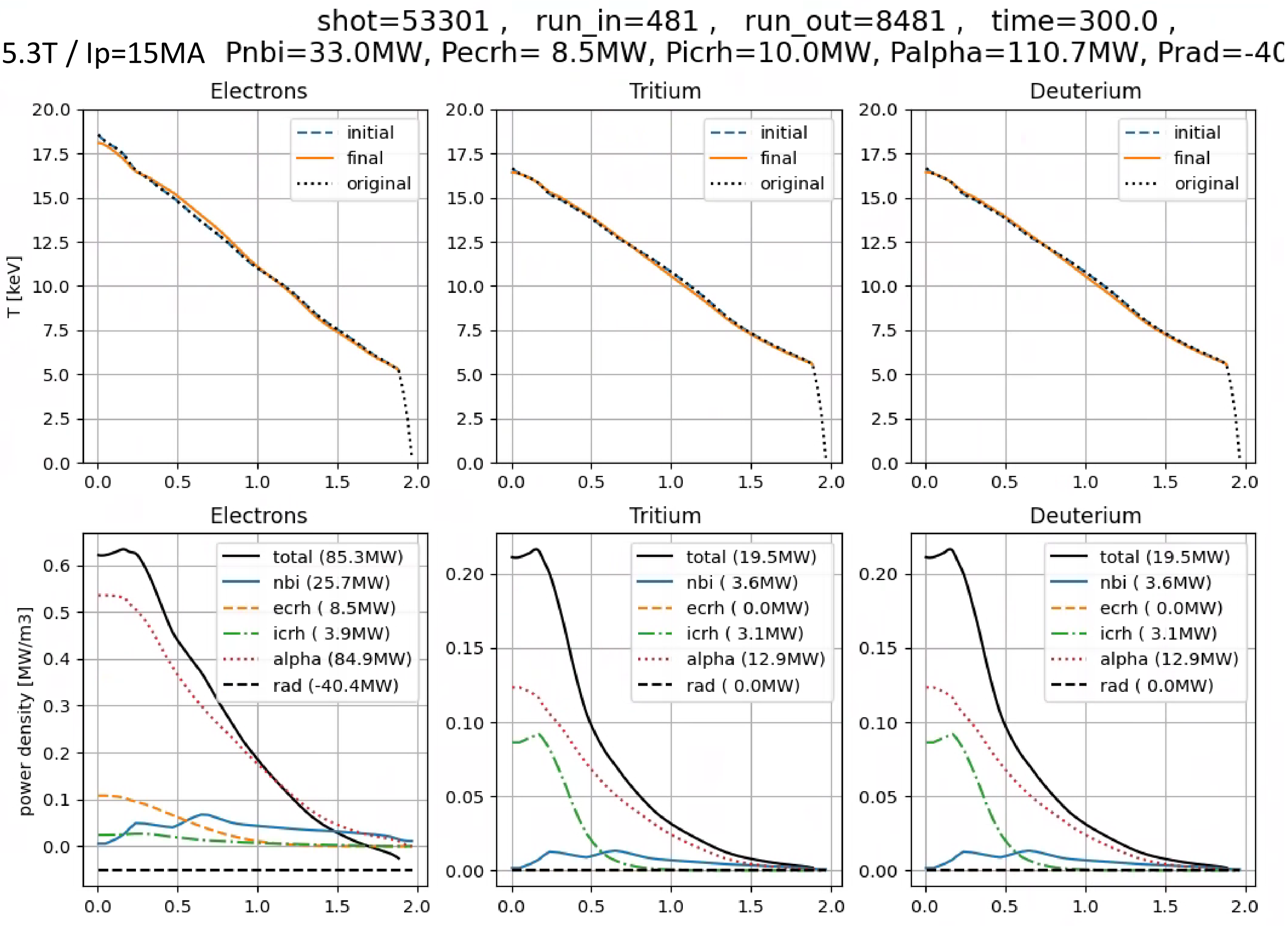


T_i < reference

Very low ion heating

ITER modelling: P_{icrh} = 10MW (reference)

Bo=5.3T / Ip=15MA

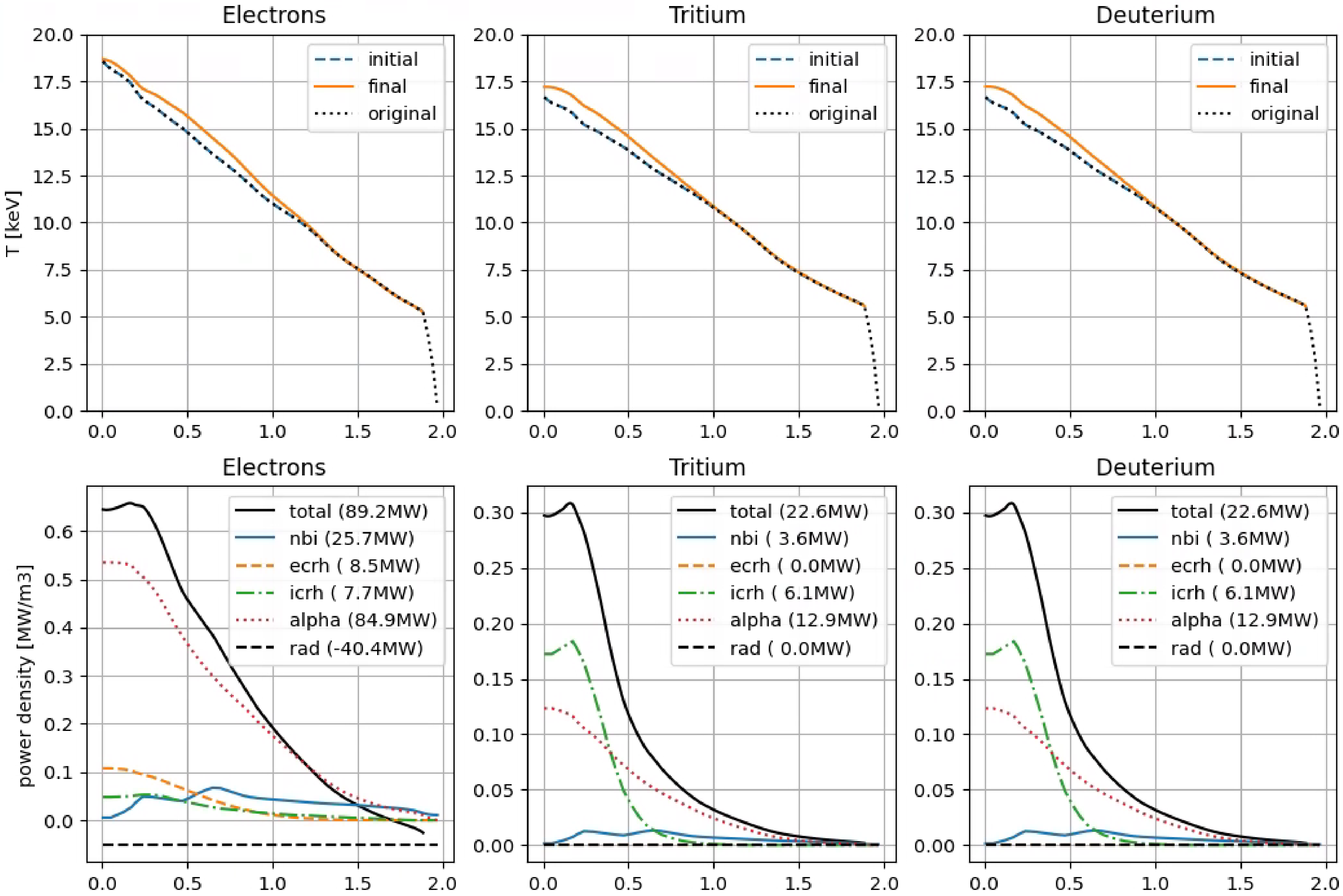


T_i = reference

Some central ion heating with ICRH

ITER modelling: P_{icrh} = 20MW

shot=53301 , run_in=481 , run_out=8481 , time=300.0 ,
Bo=5.3T / Ip=15MA Pnbi=33.0MW, Pecrh= 8.5MW, Picrh=20.0MW, Palpha=110.7MW, Prad=-40.4MW



T_i > reference

More central ion heating with ICRH

Example of non-iterative ICRH power scan (preliminary)

run 81: simpletransp ICRH=0

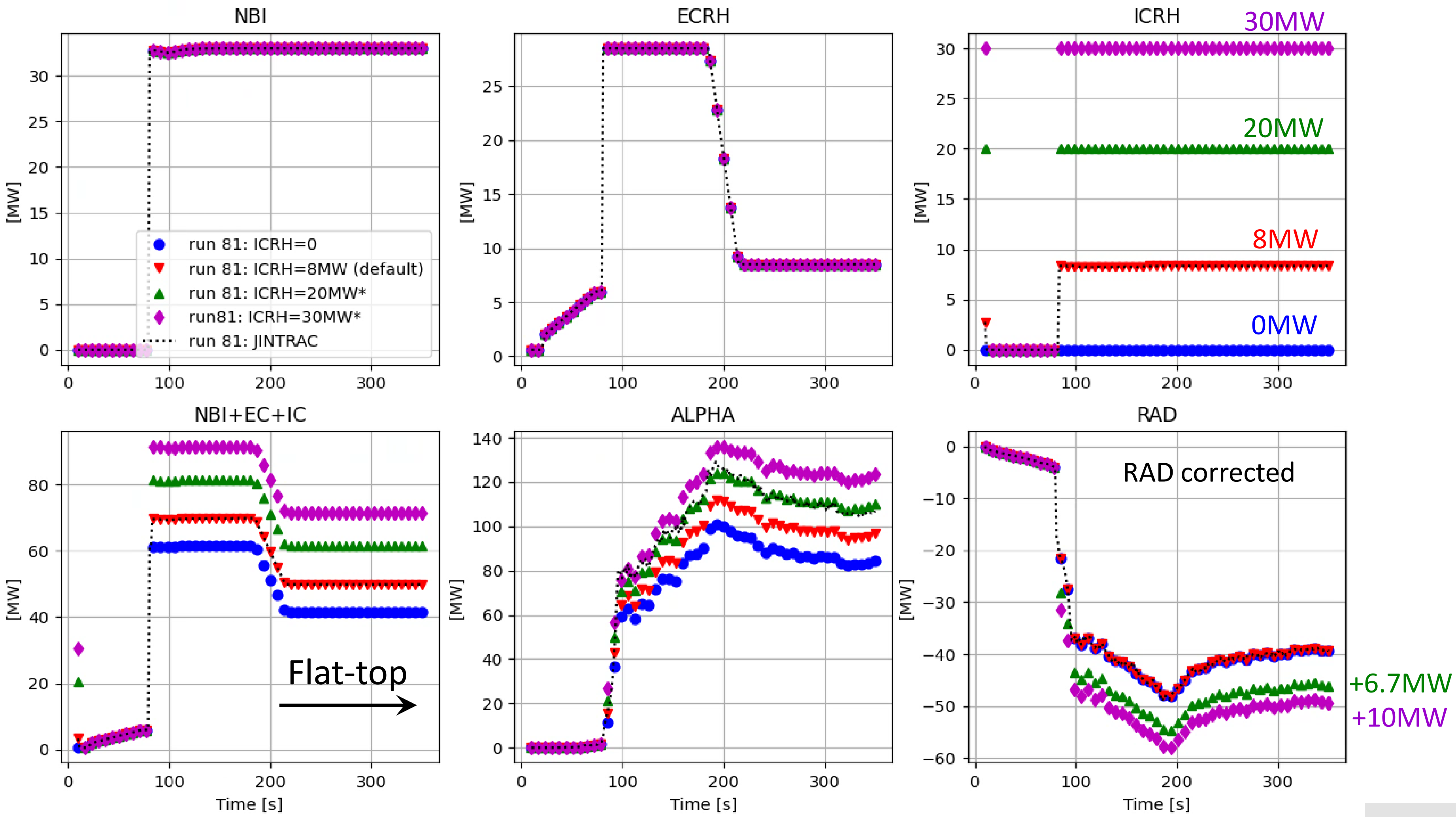
run 81: simpletransp ICRH=8MW (default)

run 81: simpletransp ICRH=20MW*

run 81: simpletransp ICRH=30MW*

(*P_{rad} increased by 33% P_{ICRH})

Fixed transport coefs!



Example of non-iterative ICRH power scan (preliminary)

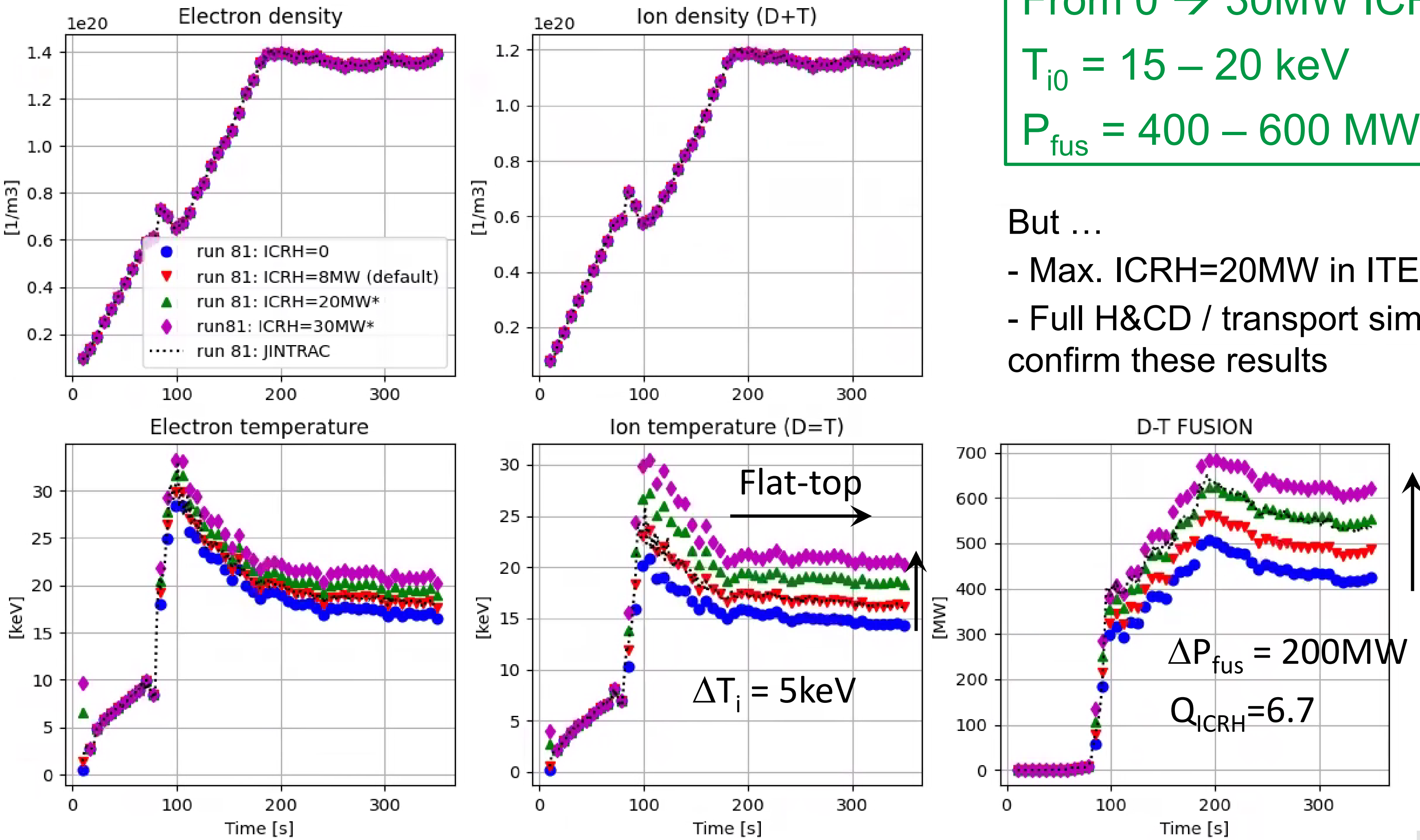
run 81: simpletransp ICRH=0
run 81: simpletransp ICRH=8MW (default)
run 81: simpletransp ICRH=20MW*
run 81: simpletransp ICRH=30MW*

Fixed transport coefs!

From 0 → 30MW ICRH:
T_{i0} = 15 – 20 keV
P_{fus} = 400 – 600 MW

But …

- Max. ICRH=20MW in ITER (one antenna)
- Full H&CD / transport simulation needed to confirm these results



Conclusions

- Auxiliary heating power is key for achieving high performance discharges in tokamaks, and each system should be explored in its best role for given conditions

- Engineering: Periodic system maintenance and systematic conditioning is important for guaranteeing large power levels and reliability

- Physics: Each heating system should be tuned to provide the desired results for a given plasma scenario (based on prior modelling)

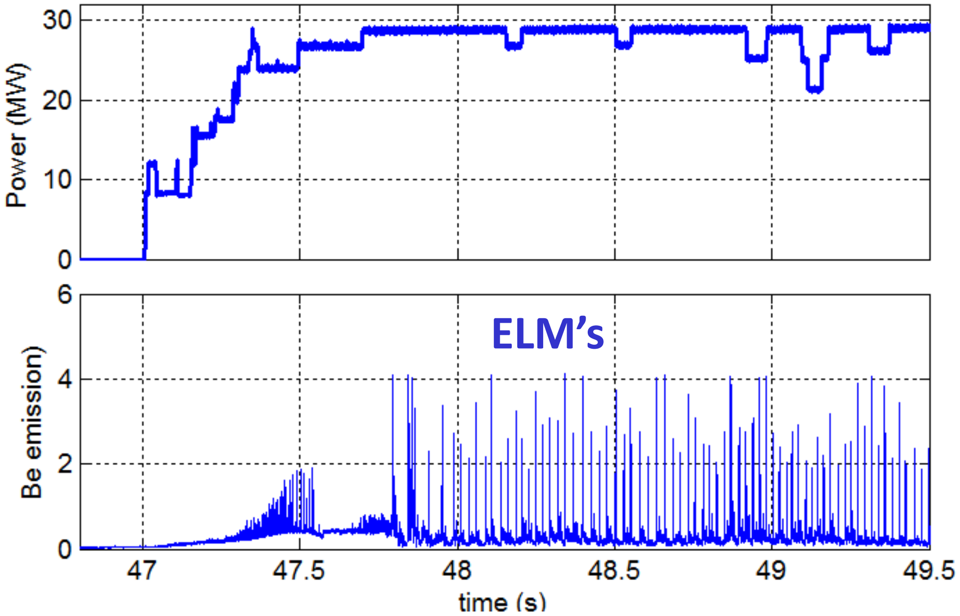
- Never put 'all your money' on a single system:

A combination of various heating systems ensures flexibility for achieving best plasma performance under different conditions (ion heating, electron heating, current-drive, MHD control, impurity control, ...

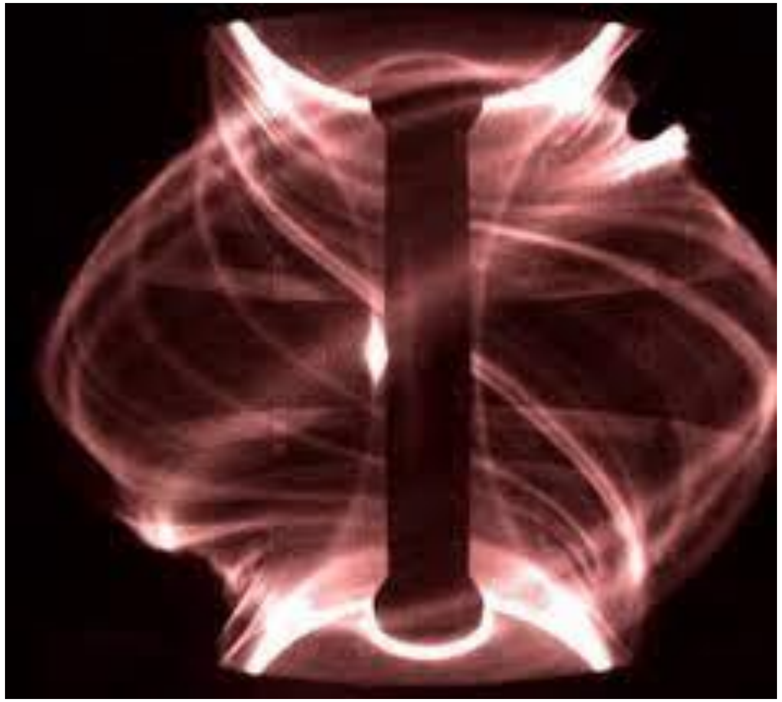
ELMs

Edge localized modes (ELMs)

- ELMs: MHD instability in the plasma edge that periodically expulses particles and energy from the plasma core into the SOL



ELMs in MAST tokamak

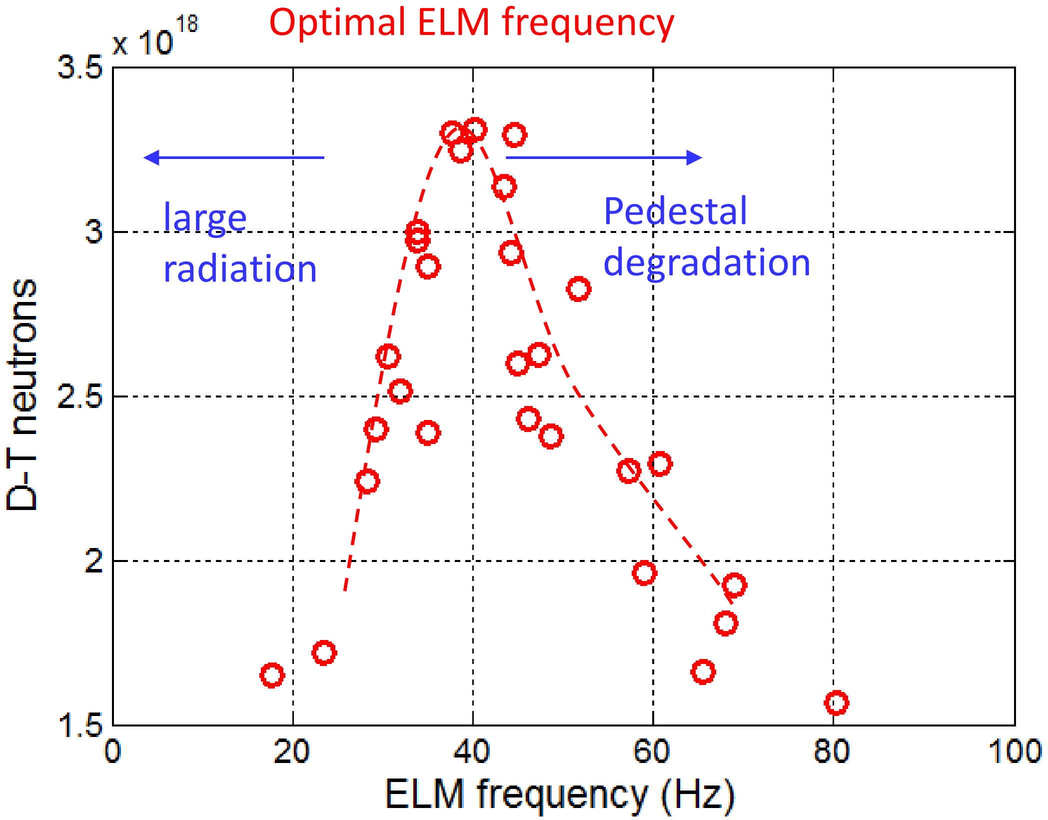


Edge localized modes (ELMs)

- ELMs play a key role in the plasma performance and stationarity

  - Too slow ELMs = less efficient impurity flushing (radiation)

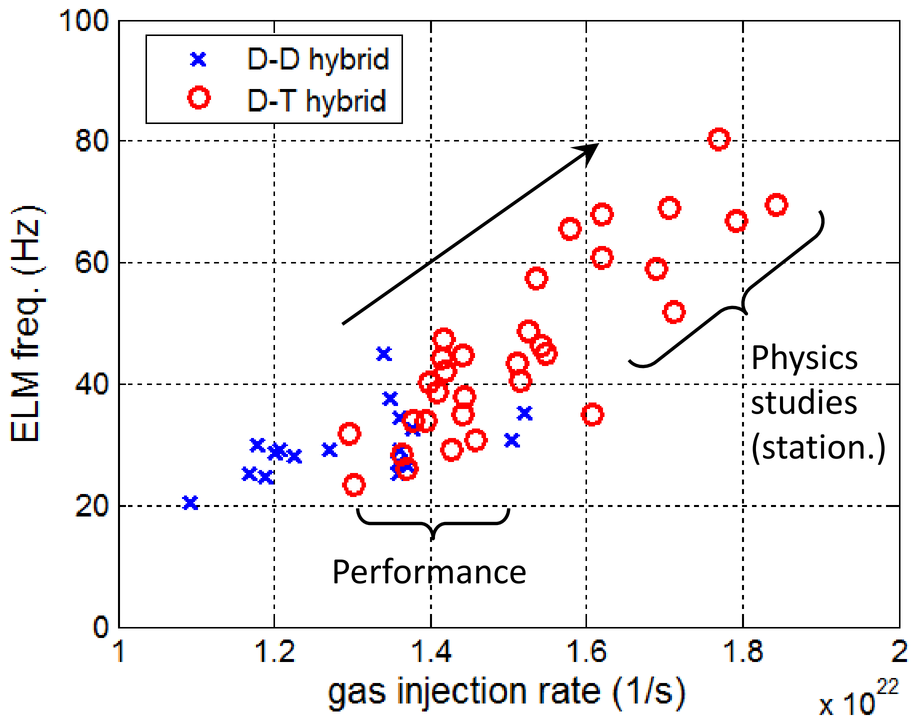
  - Too fast ELMs = degradation of the pedestal and core performance



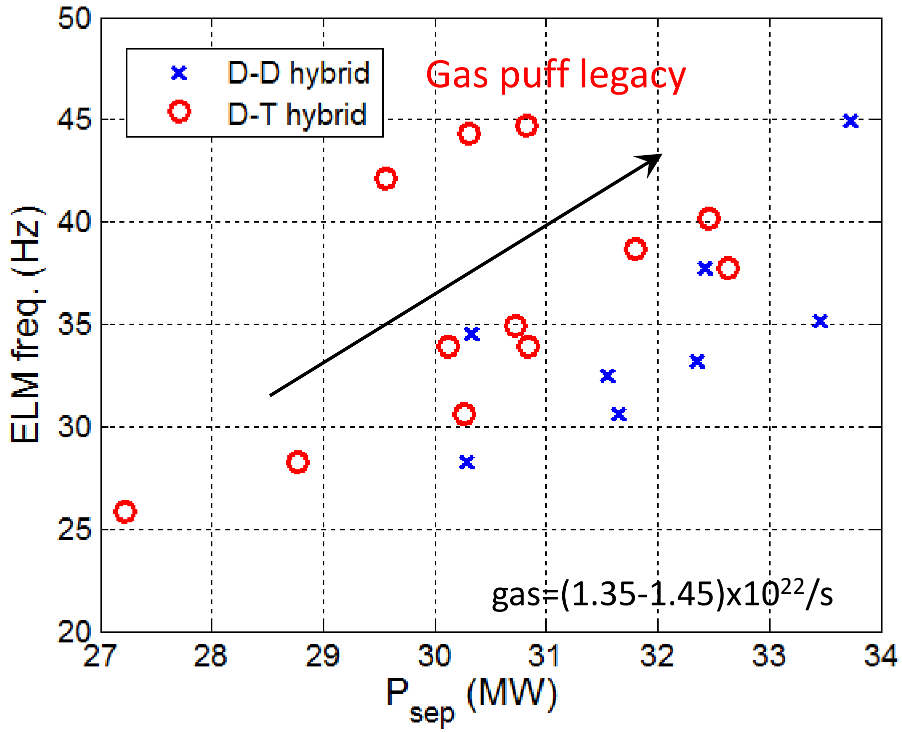
Important to control ELMs for good and stable plasma performance

ELM control: Main actuators

Gas injection rate



Power (through separatrix)



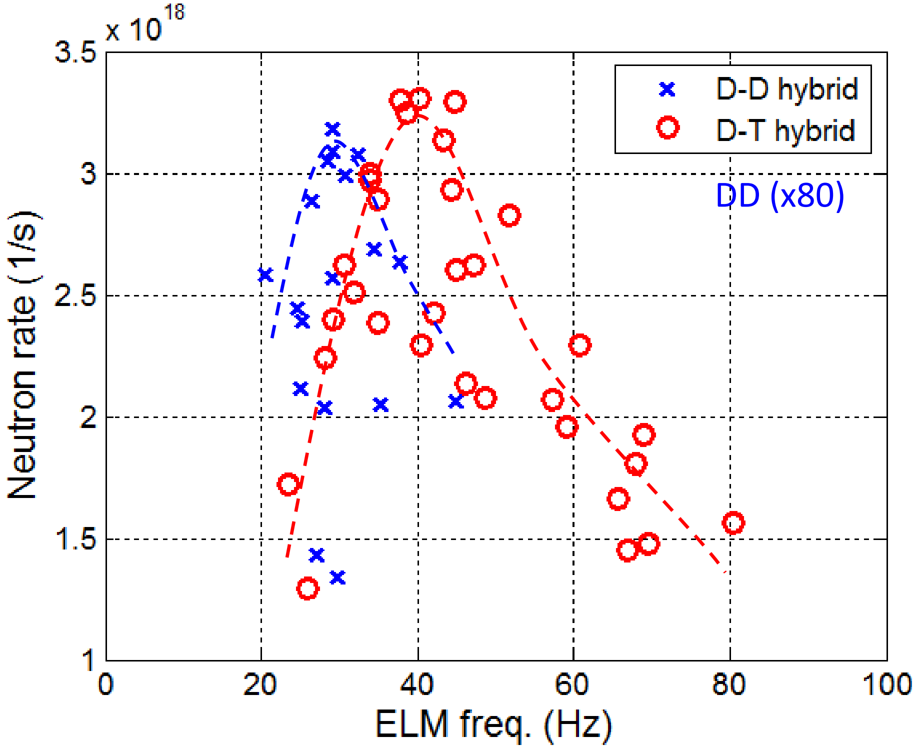
- ELM frequency increases with gas injection

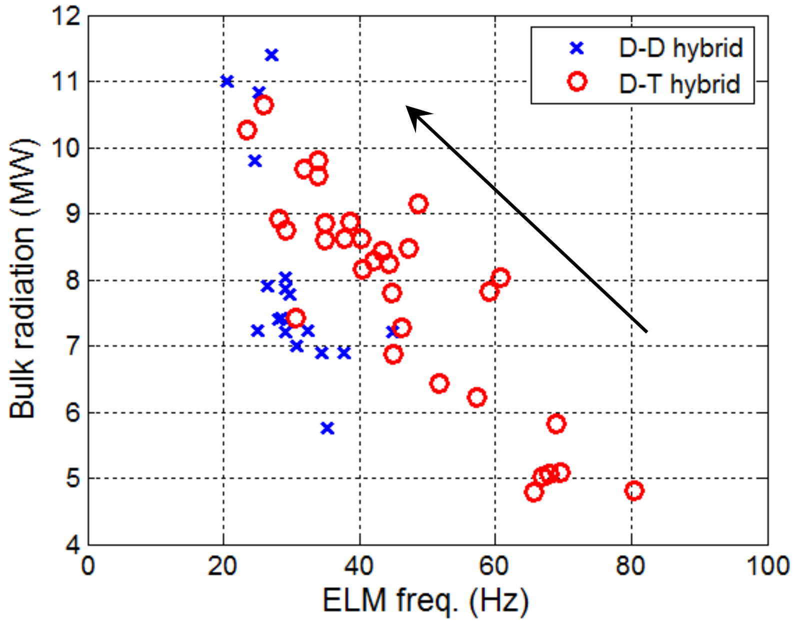
- Similar in D:D and D:T but low gas domain not easily reachable in D:T

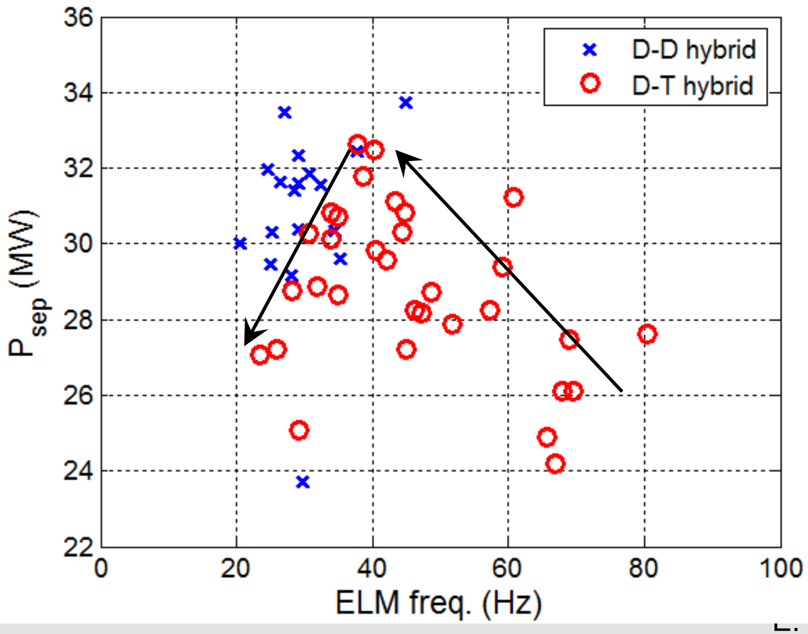
- ELM frequency increases with P_{sep}

- Slightly lower frequency in D:D for given P_{sep}, partly due to lower gas but isotope effects can be at play

ELMs: Key for performance & stationarity





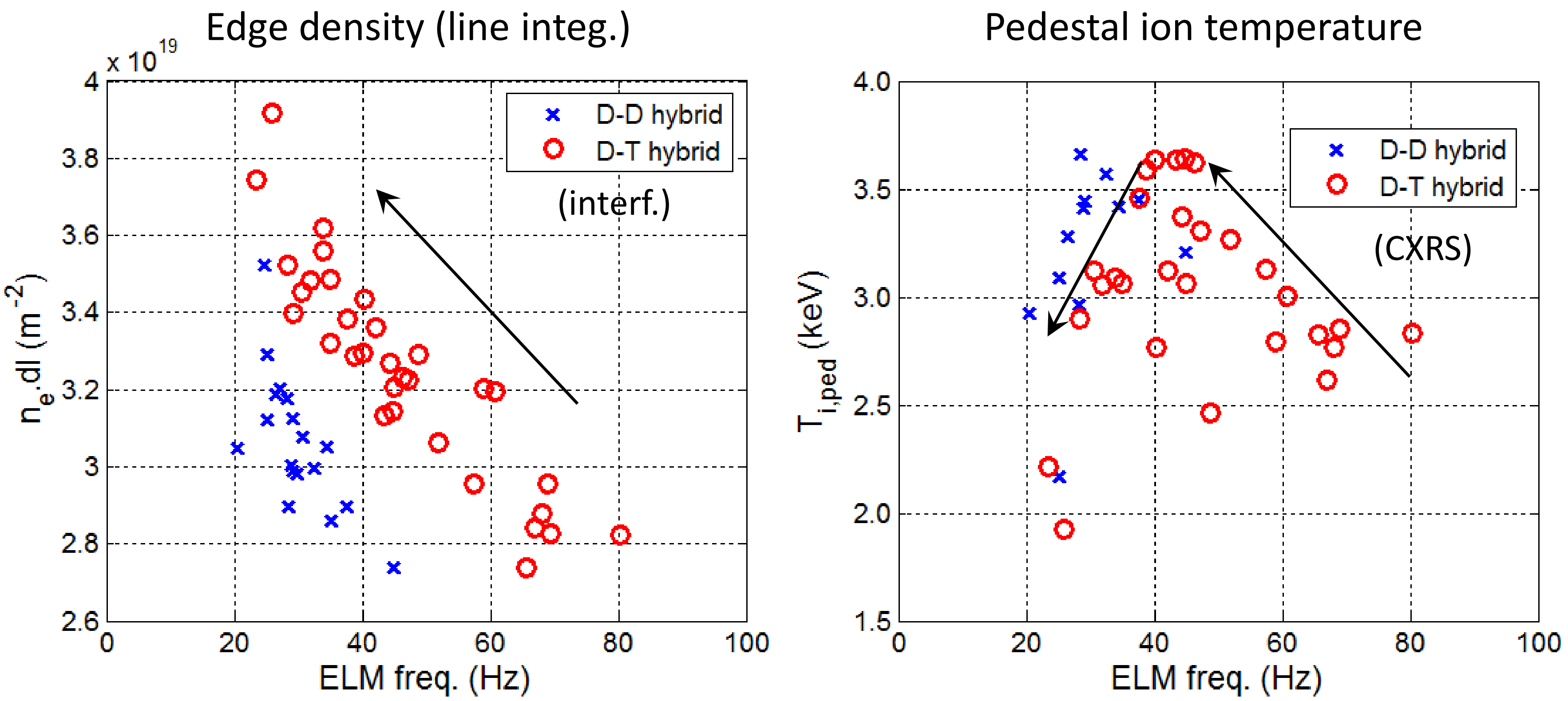


- Optimal for fusion performance vs f_{ELM}: f_{ELM}~40Hz (D:T); f_{ELM}~30Hz (D:D)

- When ELM's get too slow, bulk radiation increases leading to lower core temperature but also lower P_{sep} (which makes things worse …)

- D:D plasmas have lower radiation (larger P_{sep}) = better performance at lower f_{ELM}

ELMs: Relation with pedestal



- Pedestal density increases when ELM frequency is reduced
- Pedestal temperature shows a maximum in ELM frequency (as P_{sep});
  If the ELM frequency gets too low, $T_{i,ped}$ degrades
- D:D pulses feature lower $n_{e,ped}$ at given ELM frequency, so can sustain high ion temperatures down to lower ELM frequencies (~30Hz)

Plasma scenarios

High performance plasma scenarios

Baseline scenario	Hybrid scenario
- Exploit confinement at high plasma current (>3MA)	- Medium plasma current (<3MA)
- Fully diffused current profile	- Less peaked current profile (low shear in the core)
- Moderate β regime (β_N ~2.0): (more robust to high order MHD)	- Exploit confinement in high β regime (more sensitive to MHD)
- Medium/high core collisionality	- Lower core collisionality
- Difficult NBI penetration (wide deposition)	- Easy NBI penetration (peaked deposition)
- $T_i \sim T_e$ (equipartition)	- $T_i > T_e$ (elec-ion decoupling)
- Needs max. power for high current	- Needs max. power for high beta
- Uses gas + pellets to control ELM's	- Uses gas only to control ELM's
- Neon seeded branch: (divertor heat-load and ELM control)	- Tritium-rich branch: (enhanced beam-target reactions)

$\beta = \frac{p}{p_{\text{mag}}} = \frac{n k_B T}{B^2/2\mu_0}$ → Plasma pressure (energy) / magnetic pressure

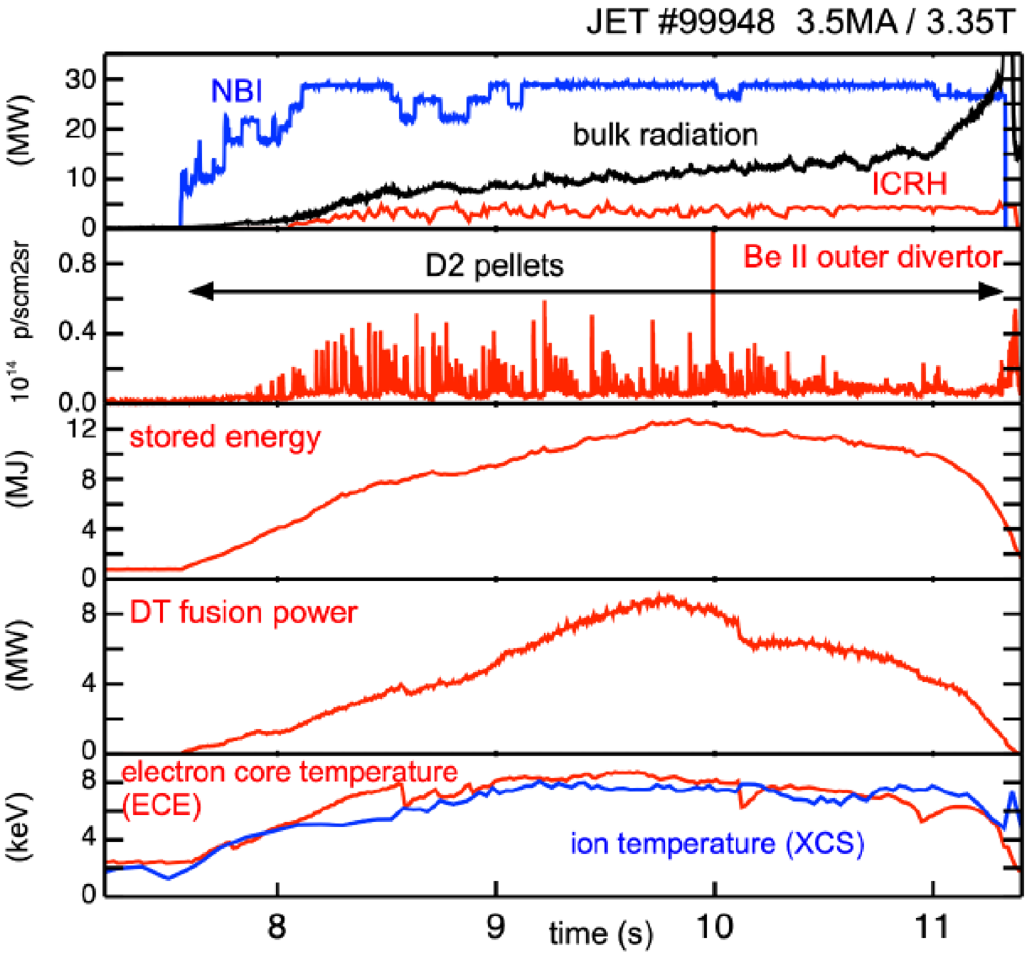
The Baseline scenario - overview

- Main reference scenario for ITER
- Solid H-mode entry, efficient ELM control with pellet injection
- Slow evolution towards maximum performance (~2sec)
- High plasma stored energy (~12MJ)
- $T_i \sim T_e$ (high core collisionality)
- Peak fusion power = 8.3MW
- Stopped by too high radiation:

– Continuous increase in radiation

– Less frequent ELM's (impurity flushing less effective)

– Off-axis radiation runaway leading to a plasma disruption



Not enough time with full NBI power and pellets for optimization in DTE2

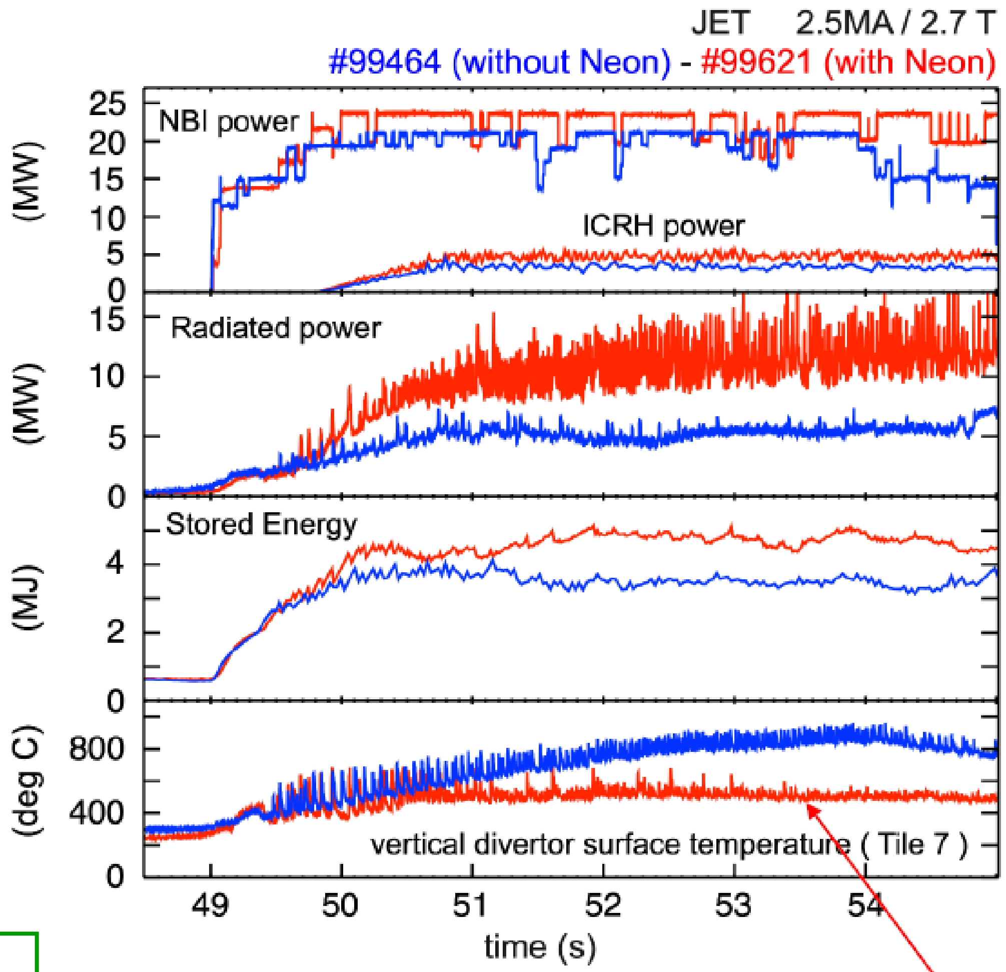
[Garzotti et al, Nucl. Fusion 2019]

The seeded baseline scenario

Integrated scenario with Neon seeding demonstrated for the first time in D-T:

- Well-controlled long pulse
- Good plasma energy confinement
- Large radiation fraction in the X-point region (not in the core)
- Divertor heat-load strongly reduced
- ELM’s partially suppressed without impurity accumulation

Confirms Neon as promising extrinsic radiator for ITER and future reactors



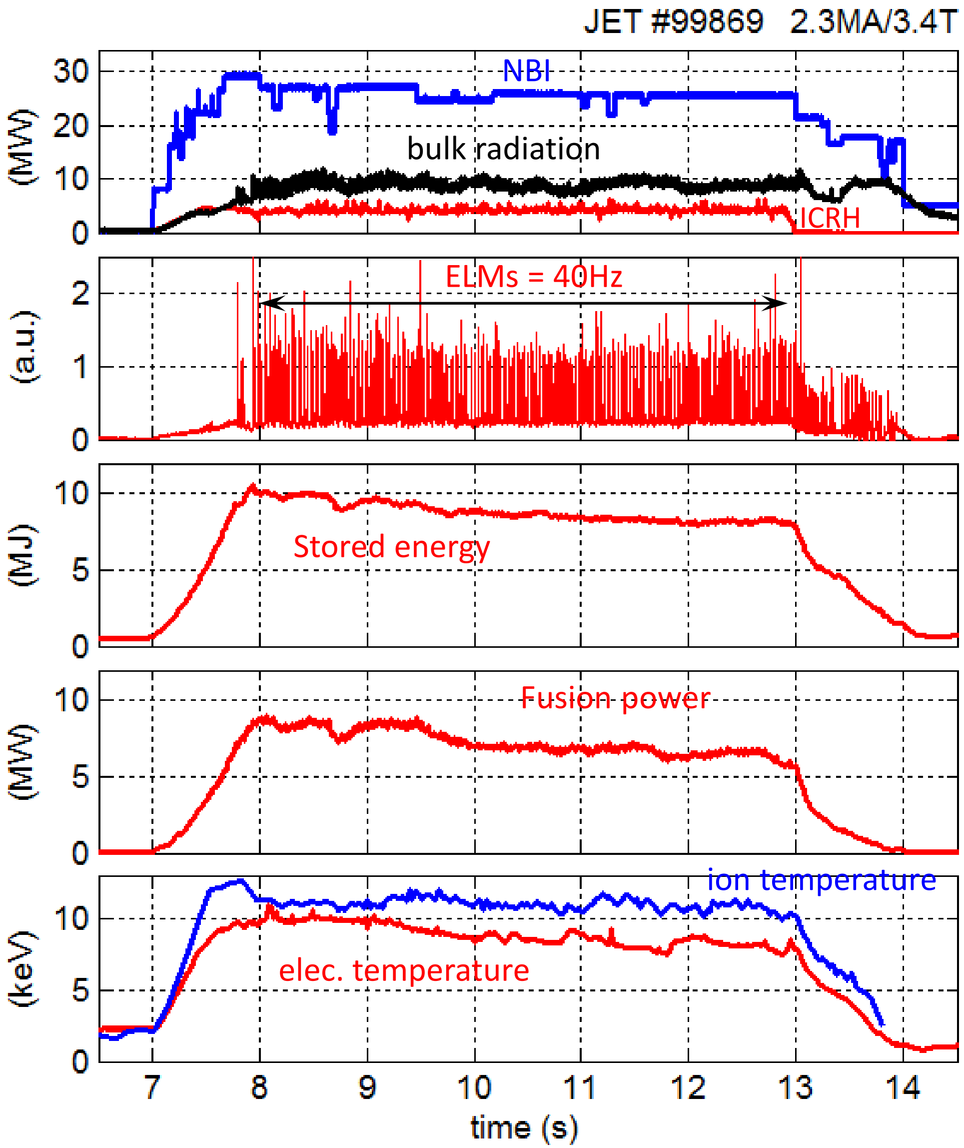
Low divertor temperature

[Giroud et al, Nucl. Fusion, Special Issue on DTE2 2023]

The Hybrid scenario - overview

- Tested for the first time in D-T
- H-mode entry:
  - Edge impurity screening ($T_{i,ped}$)
- Flat-top:
  - ELM control with gas (imp. flushing)
  - Constant radiation throughout
- Fast evolution towards maximum performance (~1s)
- $T_i > T_e$ (low core collisionality)
- Sustained fusion power for 5s
- Occasional performance degradation due to core impurity accumulation or MHD activity

Alternative high performance scenario for ITER

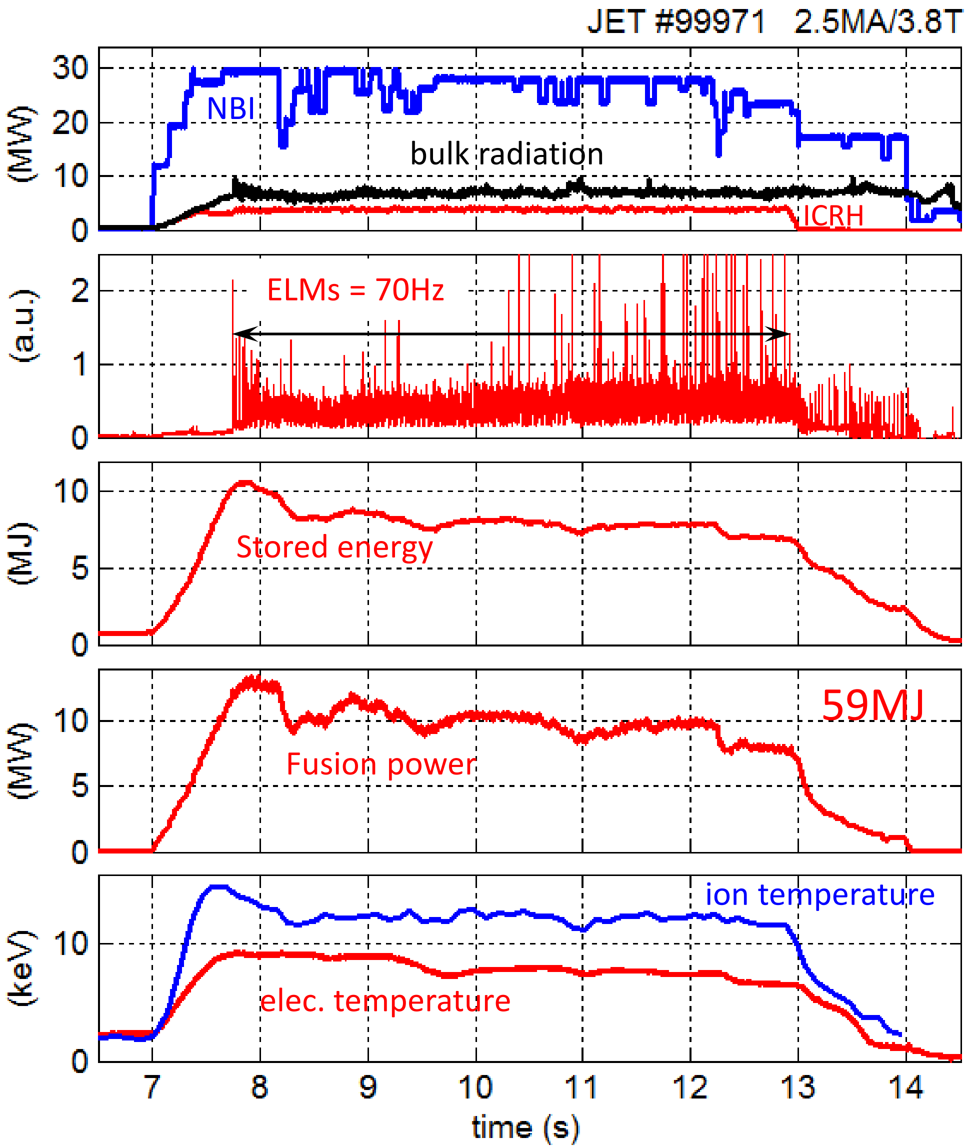


[Hobirk et al, Nucl. Fusion, Special Issue on DTE2 2023]

The T-rich hybrid scenario - overview

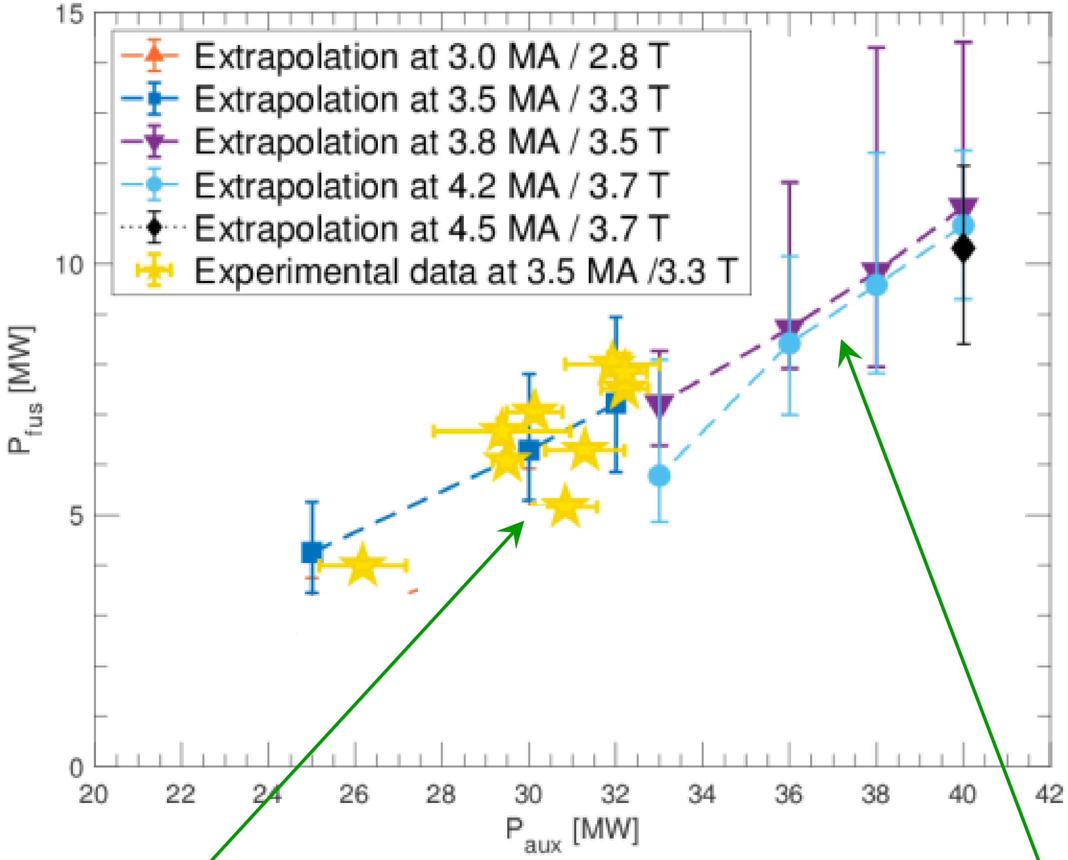
- Tested for the first time ever in H-mode
- Pure D-NBI and fast D ions (ICRH) in a T-rich plasma for enhancing beam-target fusion reactions
- H-mode entry:
  - Edge impurity screening ($T_{i,ped}$)
- Flat-top:
  - Fast and small ELM's (imp. flushing)
  - Constant radiation throughout
- $T_i >> T_e$ (low core collisionality)
- Record fusion power and energy:
  - 10MW for 5sec, 12.5MW for 2sec
- Occasional performance degradation due to core impurity accumulation

Potential scenario for ITER ramp-up phase or for beam-target dominant fusion devices



[Maslov et al, Nucl. Fusion, Special Issue on DTE2 2023]

The Baseline scenario – theoretical predictions

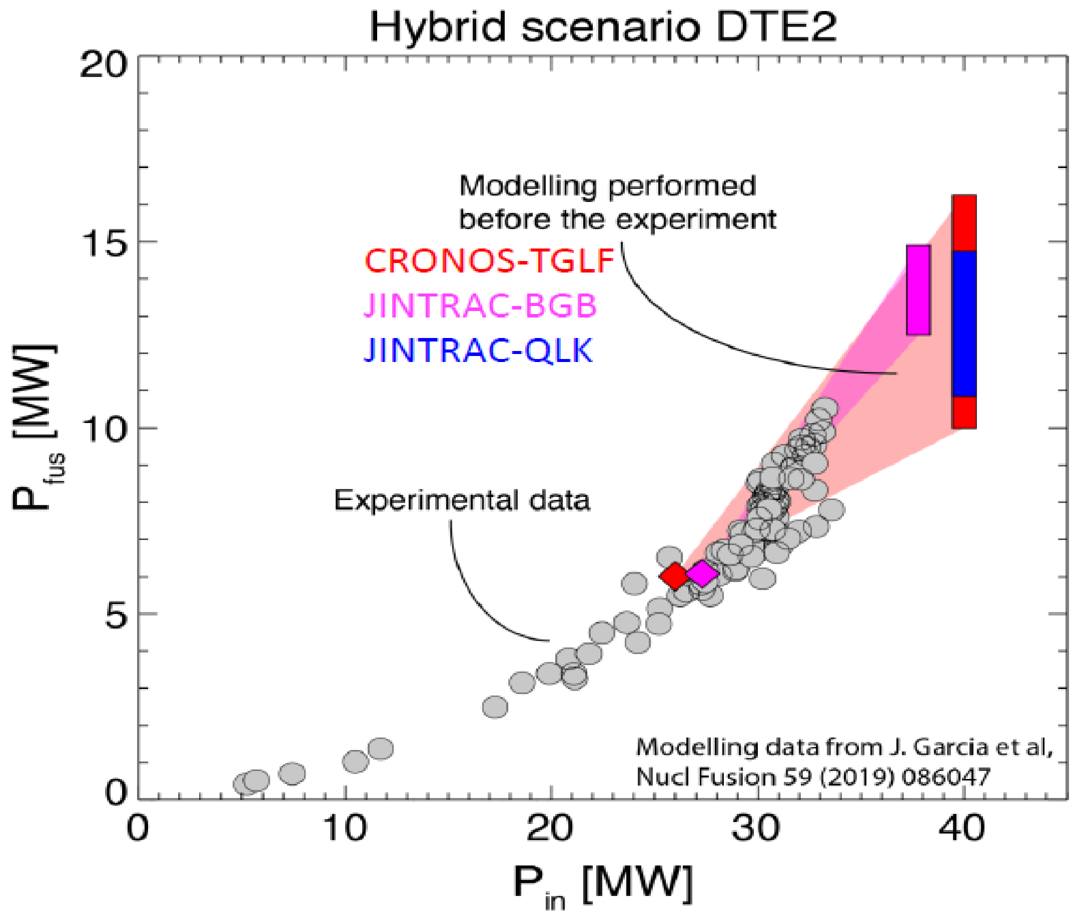


Good agreement between experiments and theoretical predictions for I_P=3.5MA

Simulations confirm that higher power is needed to keep good performance at higher current

[Maggi et al, Nucl. Fusion, Special Issue on DTE2 2023]

The Hybrid scenario – theoretical predictions



Good agreement between experimental results and theoretical predictions with different models