Application of Heating and Current Drive systems in the ITER Research Plan

Mireille SCHNEIDER - 23 July 2026
15th ITER International School on Physics and engineering of H&CD systems for magnetic fusion plasmas

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

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china eu india japan korea russia usa

Scope of the lecture

- The ITER mission goals
- How do H&CD systems help achieve ITER mission goals?
- Global view of ITER H&CD systems
- H&CD mix in the ITER Research Plan
- The ITER EC system
- The ITER IC system
- The ITER NBI system
- Operation of H&CD systems in the ITER Research Plan
- To wrap it up: H&CD across the phases of an ITER pulse
- Summary

The ITER mission goals

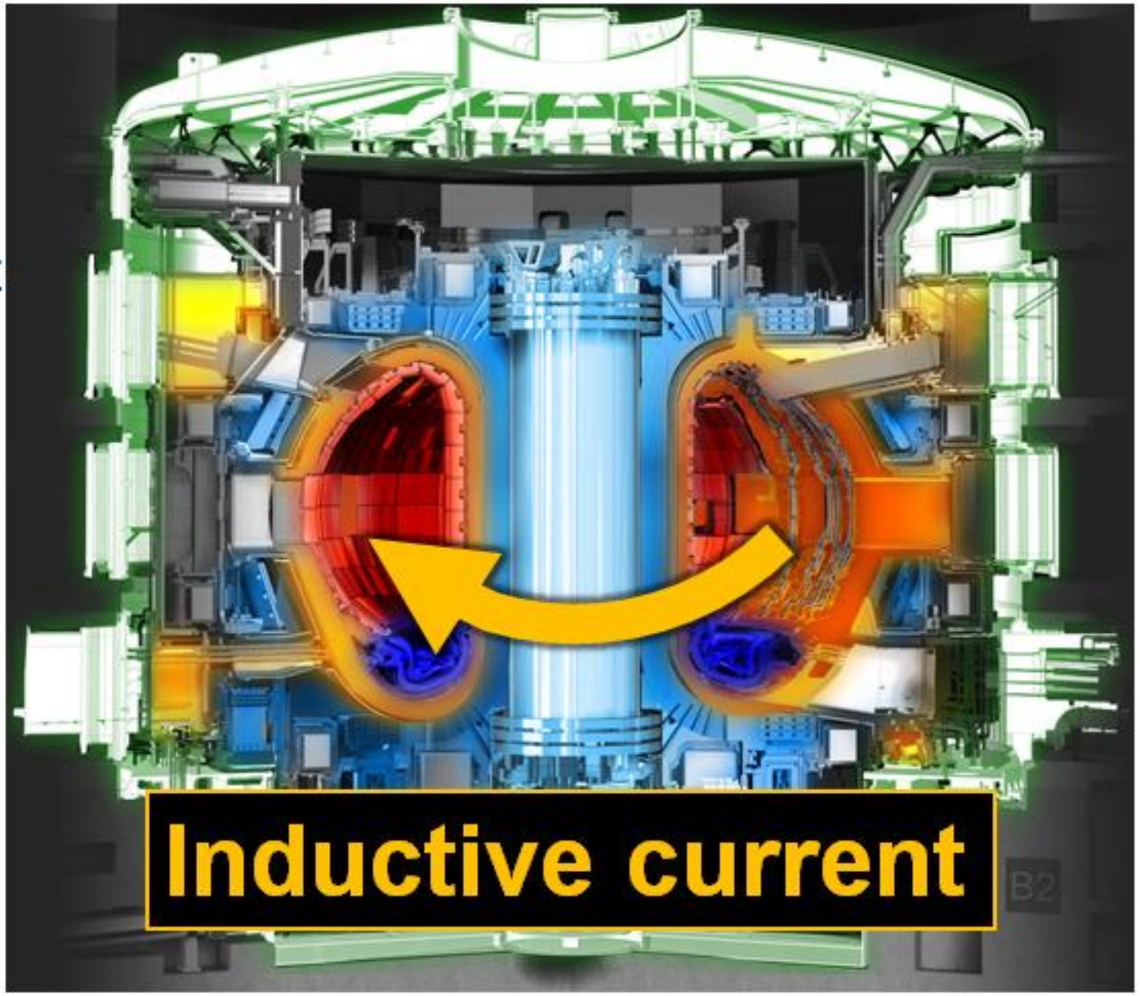
The ITER Research Plan

ITER shall demonstrate scientific and technological feasibility of fusion energy:

- **Pulsed operation:**
  $Q \geq 10$ for burn of 300-500 s, with inductively driven current.
  → Baseline scenario 15 MA / 5.3 T

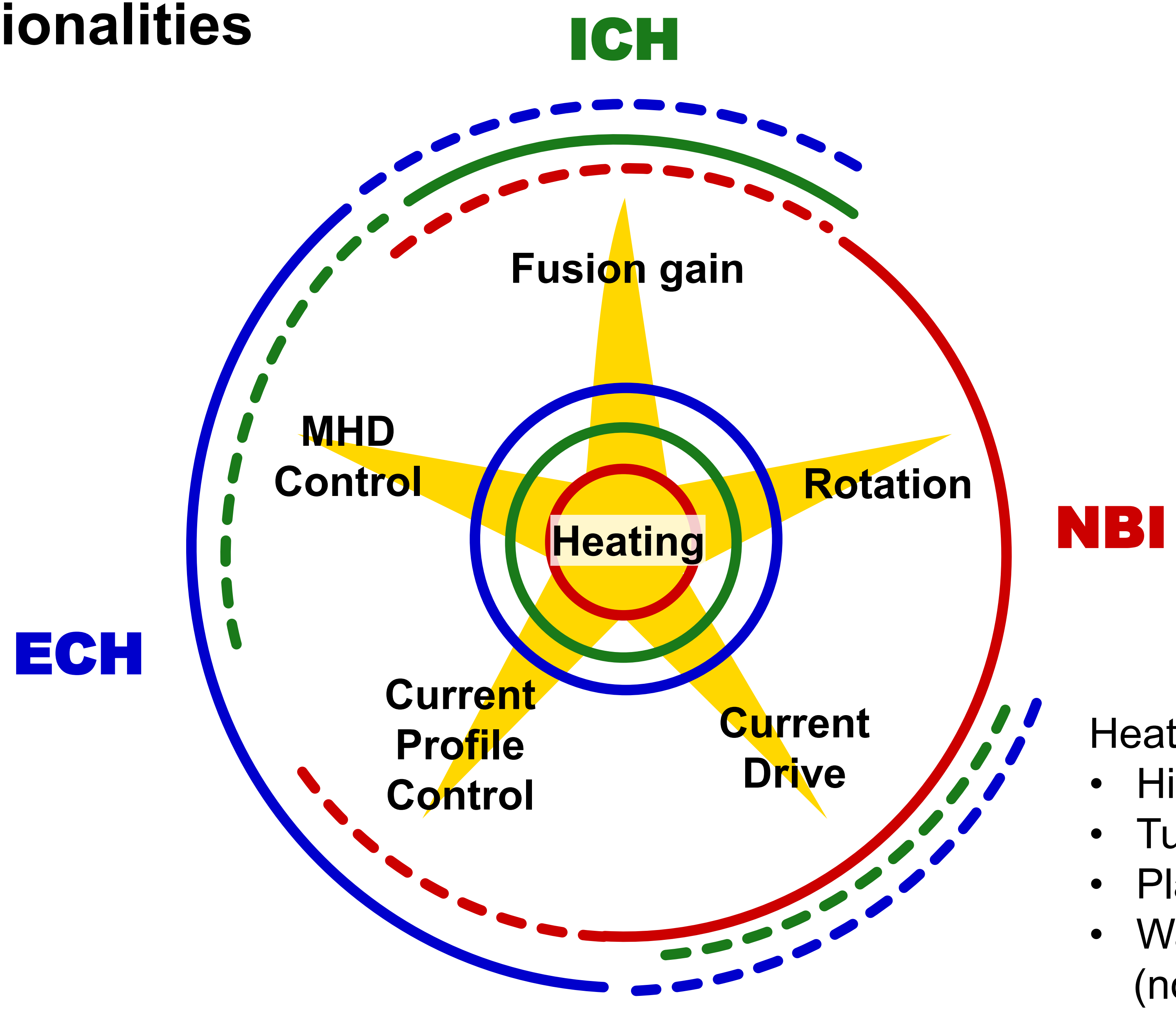
- **Long pulse operation:**
  $Q \sim 5$ for long pulses up to 1000 s, supported by non-inductive current drive.
  → Hybrid scenario ~ 12.5 MA / 5.3 T

- **Steady-state operation:**
  $Q \sim 5$ for long pulses up to 3000 s, with fully non-inductive current drive
  → Steady-state scenario ~ 10 MA / 5.3 T.



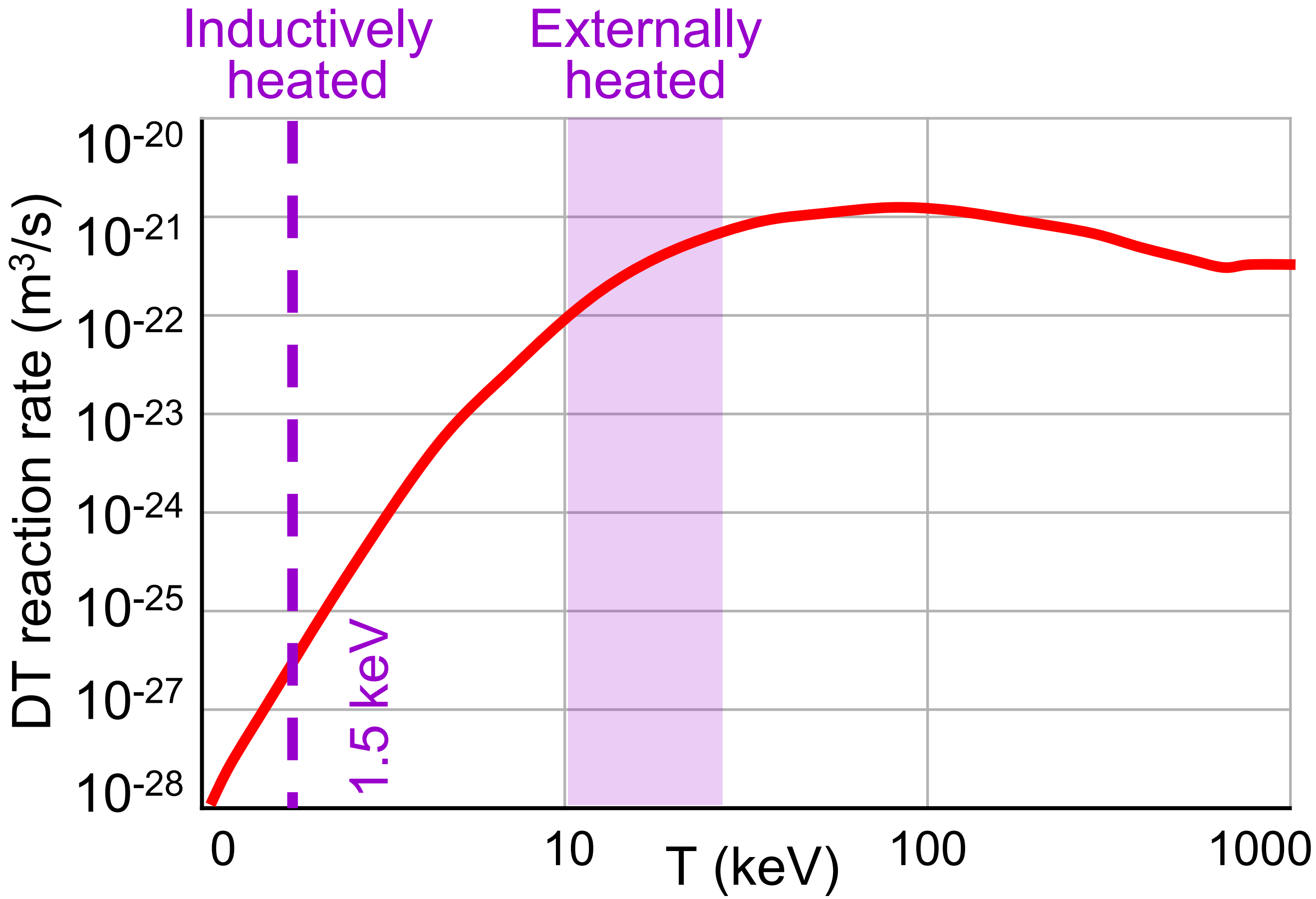
How do H&CD systems help achieve ITER mission goals?

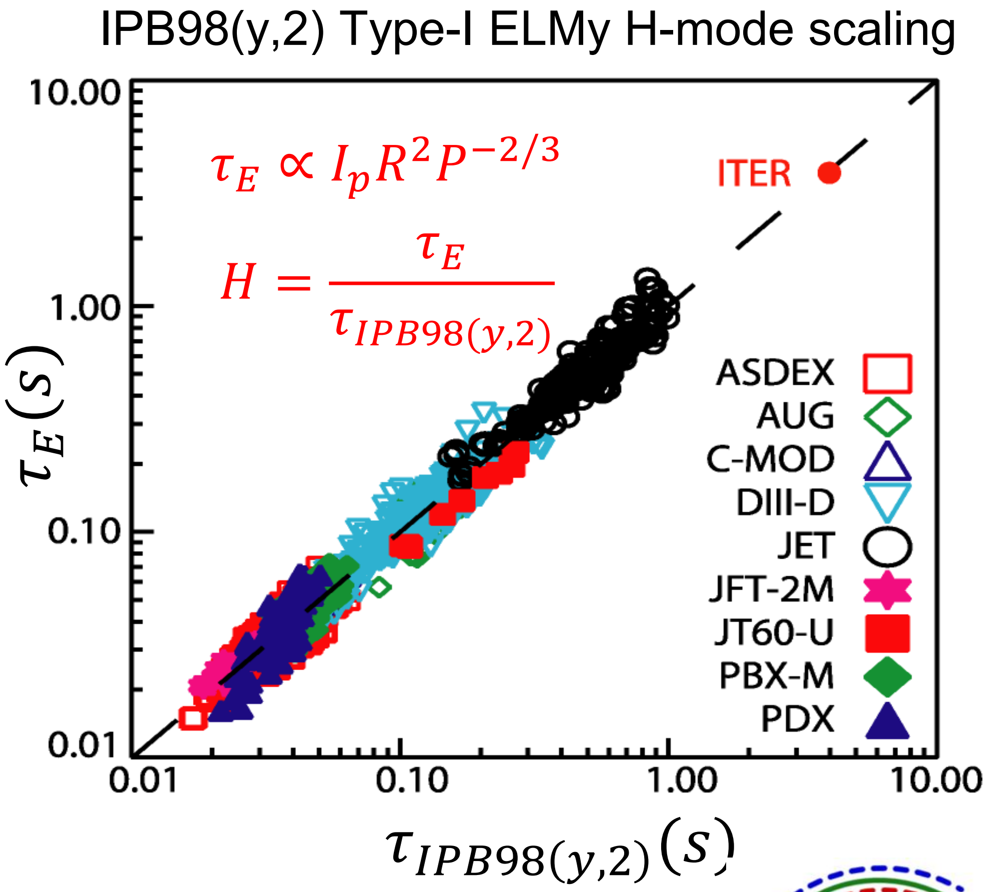
H&CD functionalities



Heating also good for:
- High confinement
- Tungsten removal
- Plasma startup assist
- Wall conditioning
  (not addressed in this talk)

H&CD to reach fusion burn conditions





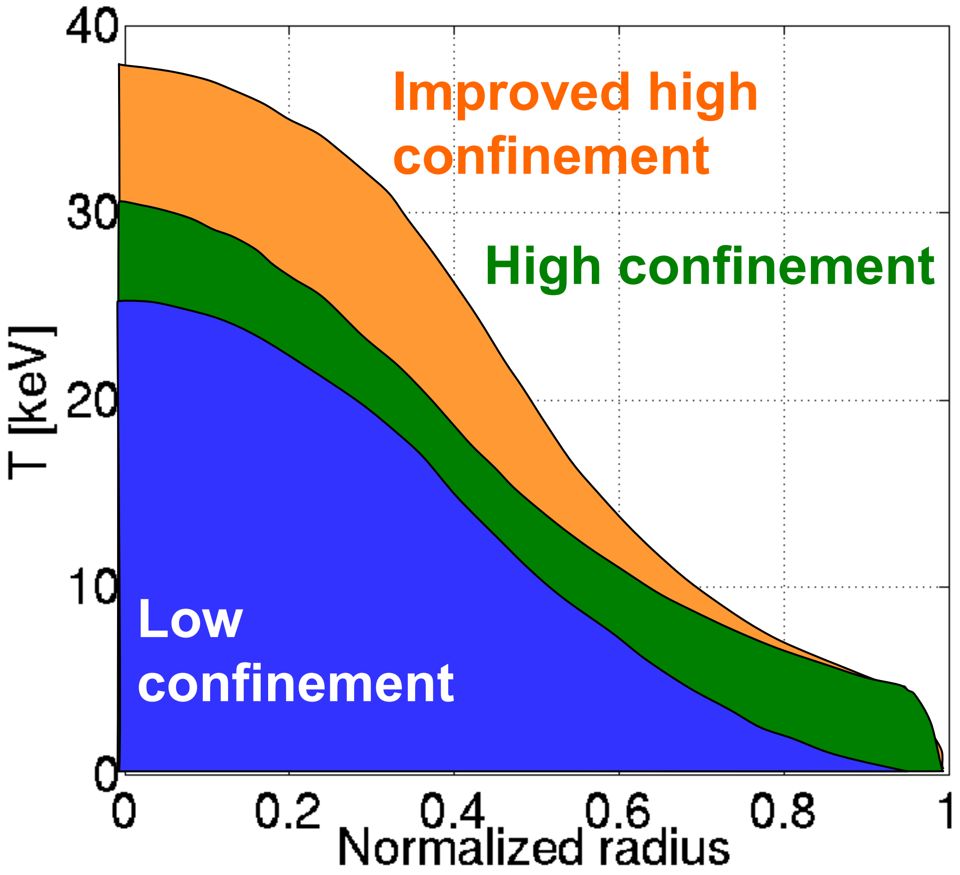
- <T> ~ several 10 keV required for high DT reactivity.
- Inductive current in ITER can ohmically heat up to <T> ~ 1.5 keV ($\eta \sim T^{-3/2}$)

→ External heating required to reach:
High confinement → high temperature → high Q → α self-heating

H&CD to reach high confinement

Three confinement regimes:

- Low confinement (L-mode): pulsed operation ($H \sim 0.5$)
- High confinement (H-mode): pulsed operation ($H \sim 1$)
- Improved high confinement: long pulse ($H \sim 1.2$) / steady-state ($H \sim 1.6$)



Power threshold for transition from L to H-mode:

$$\boxed{P_{L-H} \propto n_e^{0.7} \times B^{0.8}}$$

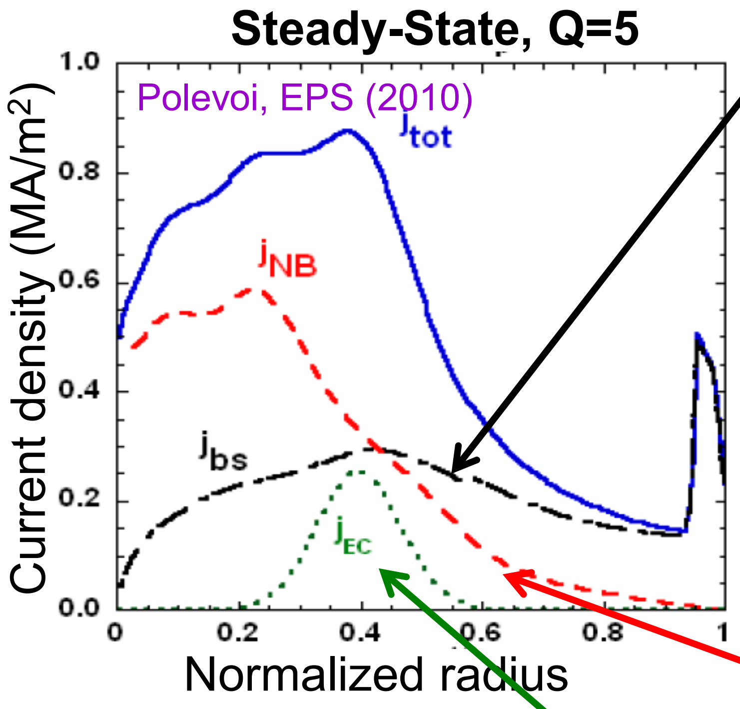
[Y.R. Martin, J. Phys. Conf, 2008]

→ Low density and B-field operation when H&CD is limited.

$$P_{L-H}(\boldsymbol{T}) < P_{L-H}(\boldsymbol{D}) < P_{L-H}(\boldsymbol{H})$$

H&CD to drive current for long pulse operation

- The inductive flux of a tokamak is finite → pulsed, limited in time.
- Long pulse operation requires that I_p be driven by other means:



Bootstrap current $\mathbf{j_{bs}}$: self-driven from gradients; increases with β:

- Plasma performance: $\beta = \frac{p_{plasma}}{p_{magnetic}} = \frac{\langle p \rangle}{B^2/2\mu_0}$
- Plasma stability limit (~3-4): $\beta_N = \beta \frac{aB_T}{I_p}$

Neutral Beam Current Drive $\mathbf{j_{NB}}$ (NBCD)

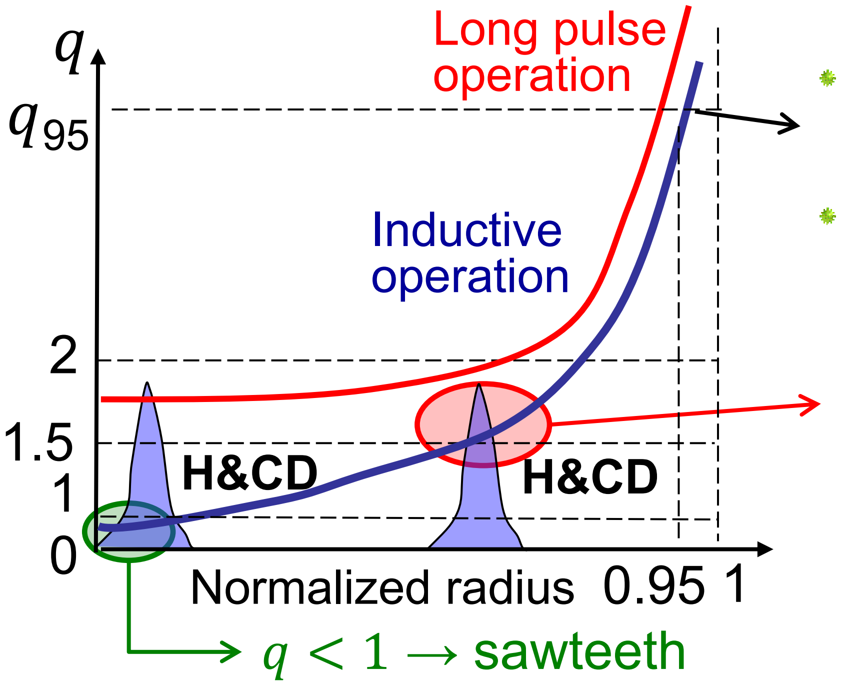
Electron Cyclotron Current Drive $\mathbf{j_{EC}}$(ECCD)

External sources

H&CD to control MHD instabilities

Instabilities can deteriorate plasma confinement:

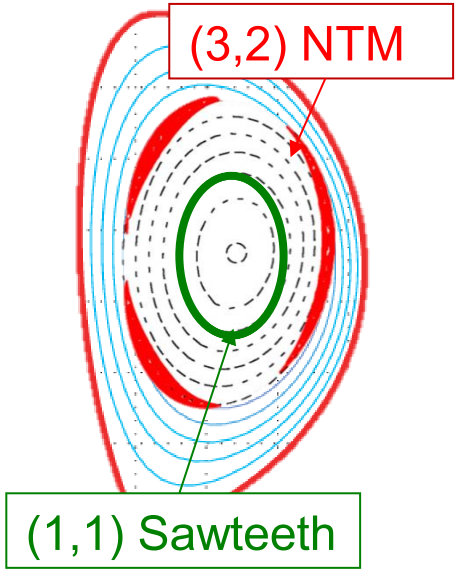
- Characterized by safety factor, $q = toroidal/poloidal$ turns of magnetic field lines
- Instabilities on rational q surfaces, driven by pressure and current gradients
- ITER H&CD systems can influence both.



- Plasma can disrupt if $q_{95} < 2$
- ITER routine operation: $q_{95} \geq 3$

$q = 3/2 \rightarrow$ NTM
(Neoclassical Tearing Modes)

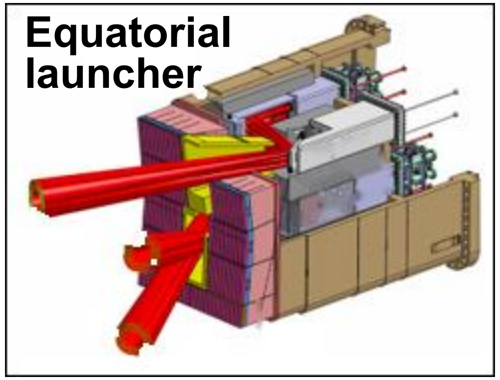
ITER baseline Q=10
magnetic equilibrium

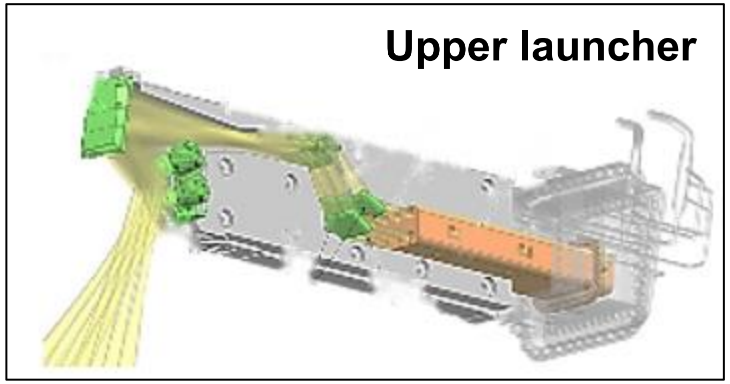


Global view of ITER H&CD systems

Three H&CD systems on ITER

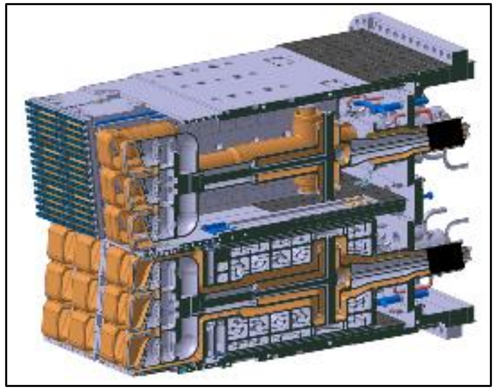
1) Electron Cyclotron Heating (ECH)





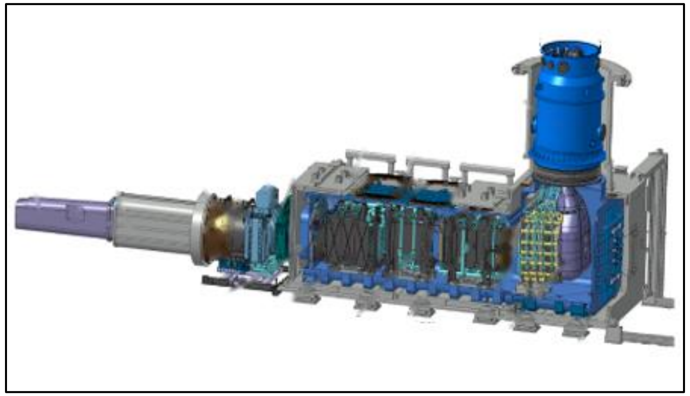
- mm waves launched through steerable mirrors
- Highly localised and flexible deposition
- Heats electrons
- Ideal for MHD control and core W removal
- Provides wall conditioning

2) Ion Cyclotron Heating (ICH)



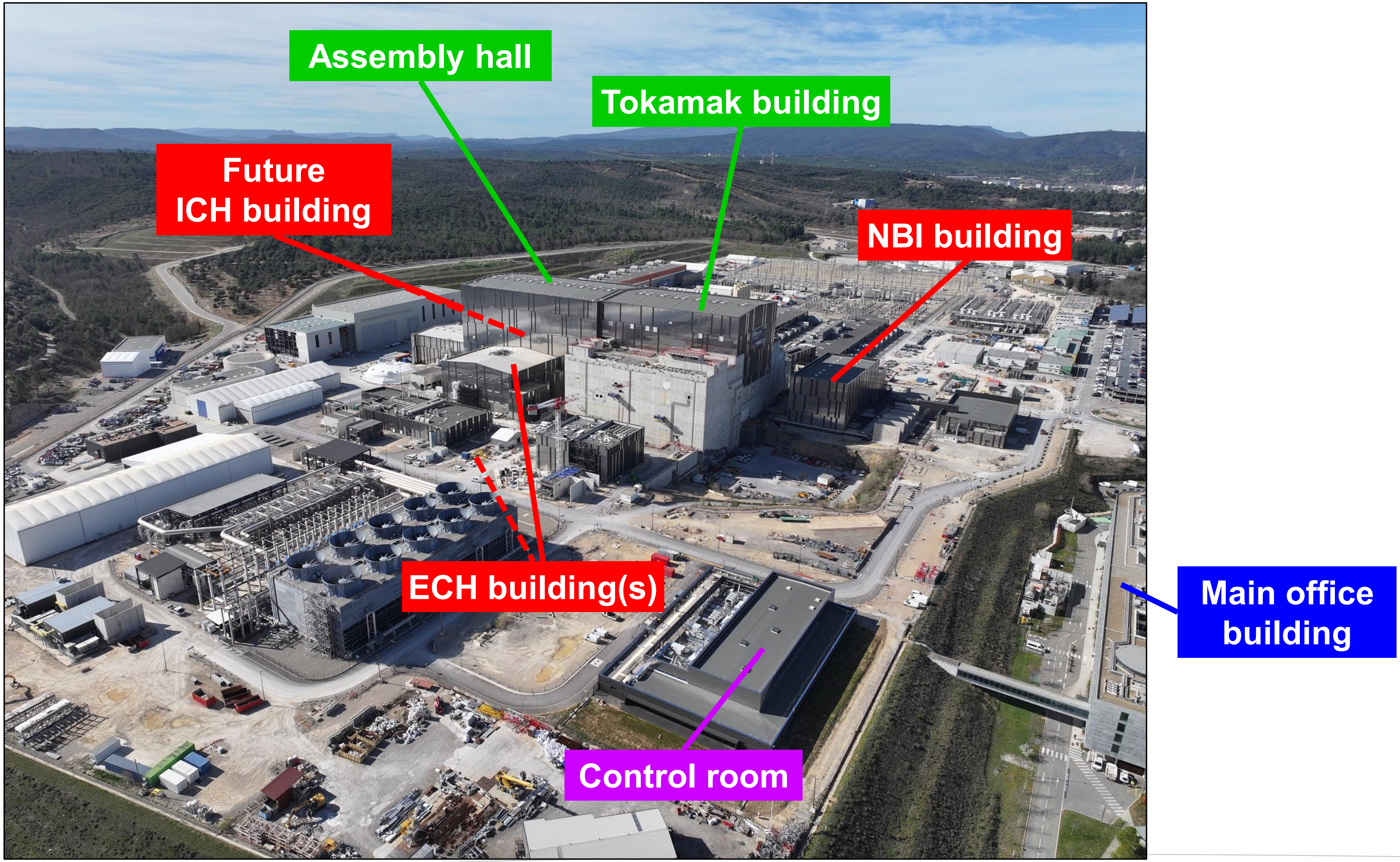
- RF waves from single multi-strap antenna
- Direct heating of bulk ions
- Generates alpha-like hot ions (^{3}He fast ion tail)
- Provides wall conditioning

3) Neutral Beam Injection (NBI)



- High energy neutral atoms fired tangentially into the plasma
- Main source of external current drive

H&CD systems within the ITER worksite



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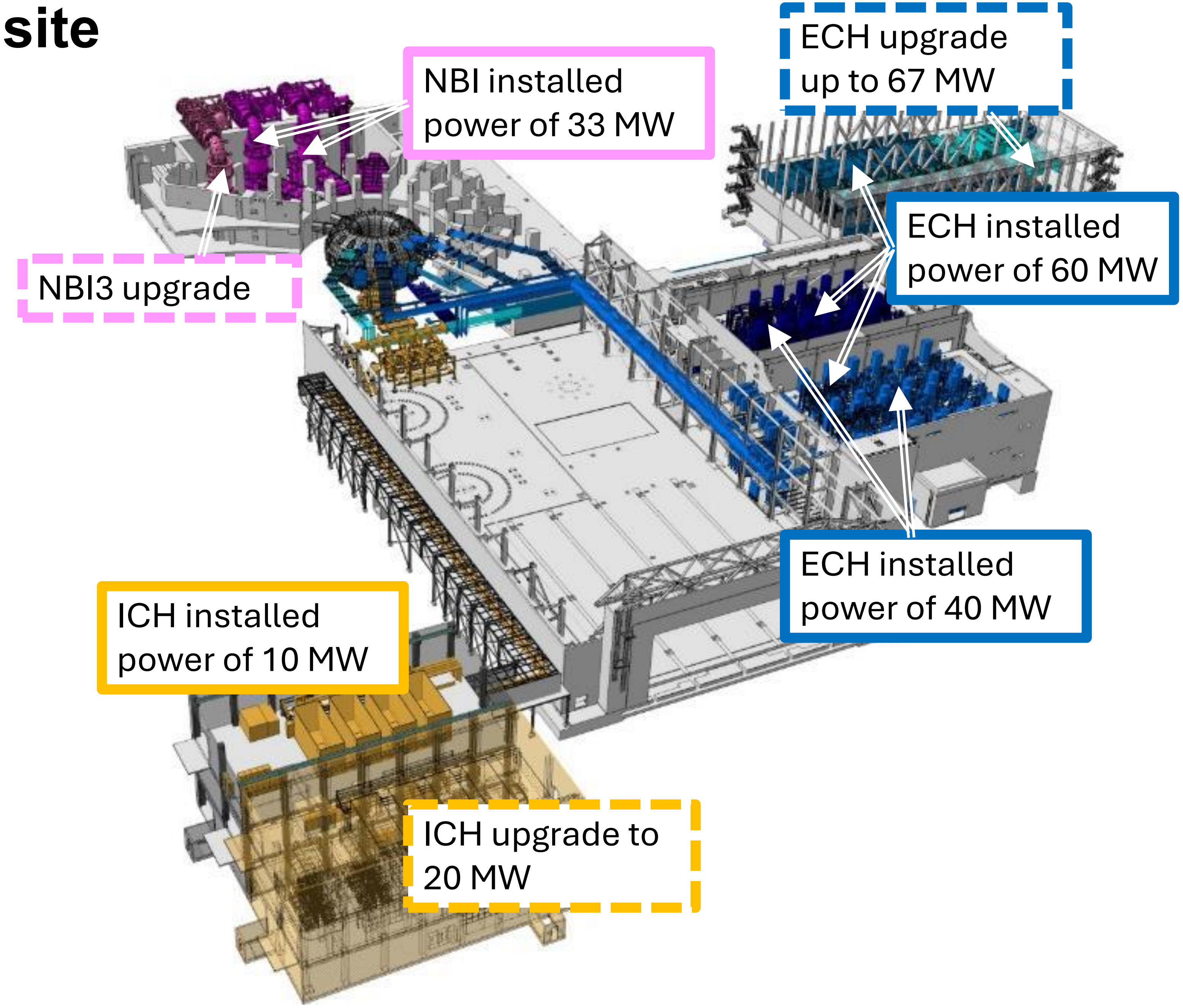
H&CD installation on ITER site

Electron Cyclotron Heating

Neutral Beam Injection

Ion Cyclotron Heating



H&CD systems on ITER tokamak

- **ECH system**
  170 GHz
  - 3-4* Upper Launchers (UL)
  - 2 Equatorial Launchers (EL)

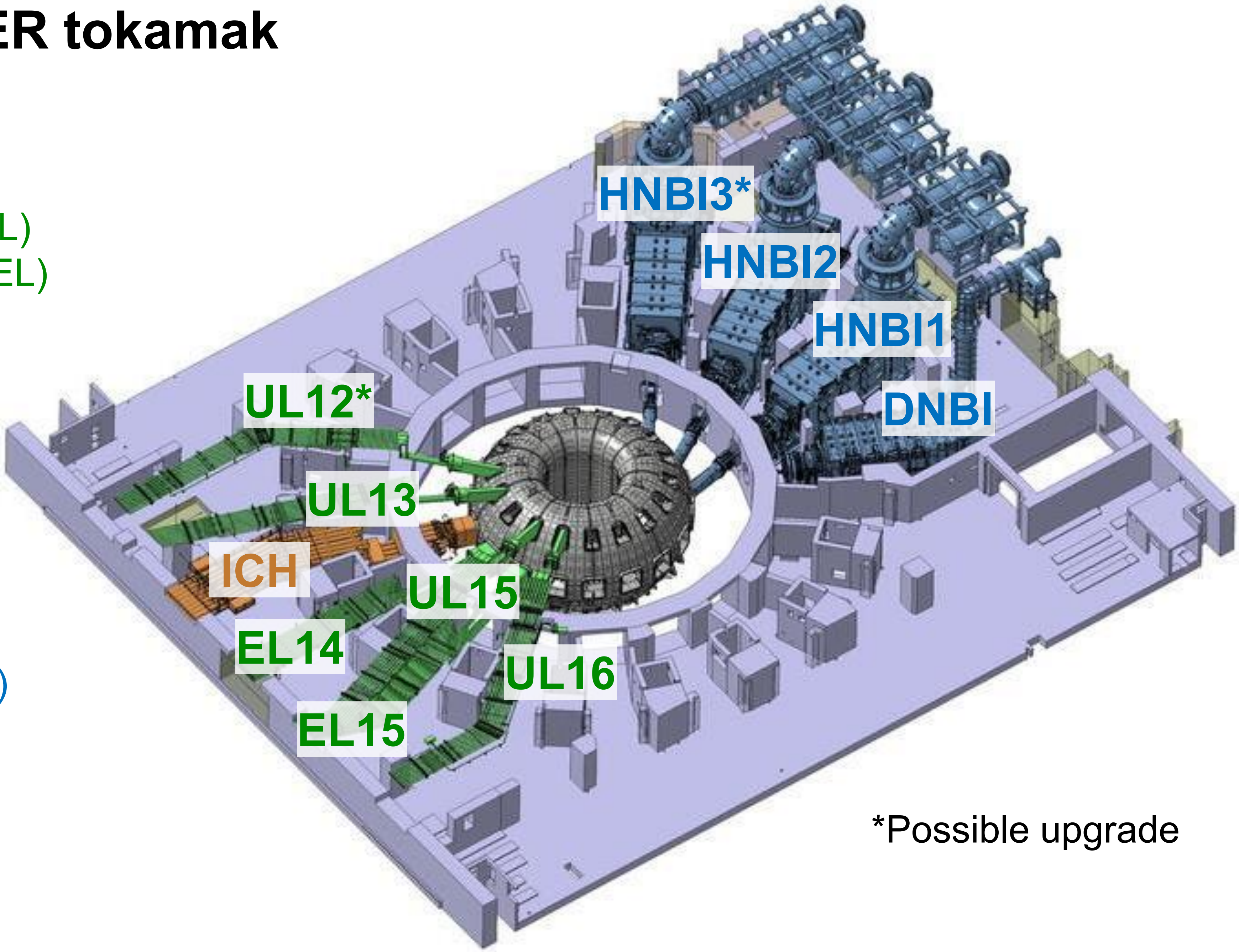
  → Up to 60 - 67 MW

- **ICH system**
  [40-55] MHz
  - 1 antenna

  → Up to 10 - 20* MW

- **NBI system**
  - 2-3* heating boxes (HNB)
    - 870 keV H beams
    - 1 MeV D beams
  - 1 diagnostic box (DNB)
    - 100 keV H beam

  → Up to 33 - 49.5 MW



*Possible upgrade

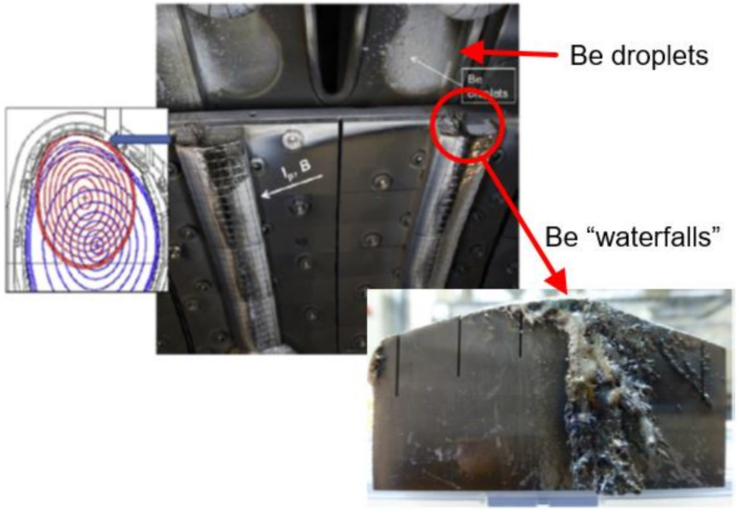
H&CD mix in the ITER Research Plan

Why going from Be to W Blanket First Wall (1/2)

- Be melts at low temperature (~1450°C)
  → Less resilient to melting events (melting starts already during 7.5MA Current Quench = CQ)

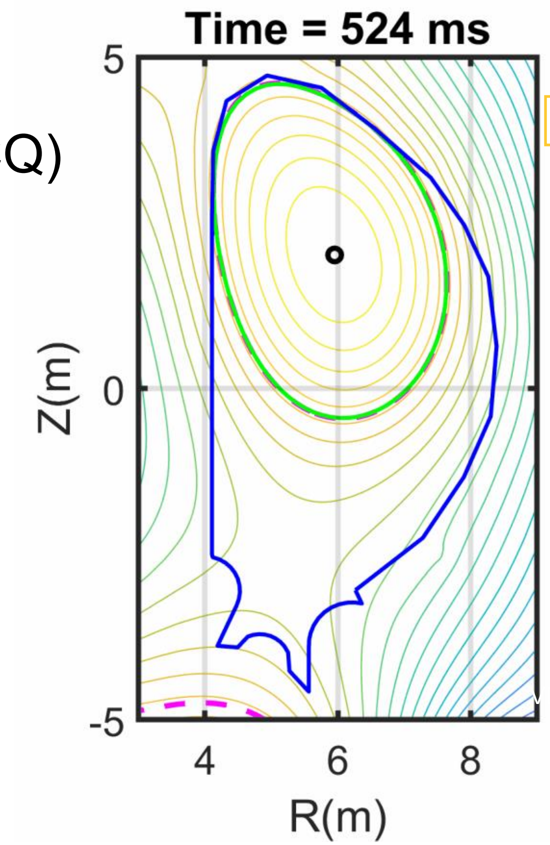
- Worst possible unmitigated 15 MA VDE CQ:
  - Deposits ~600MJ over ~400ms

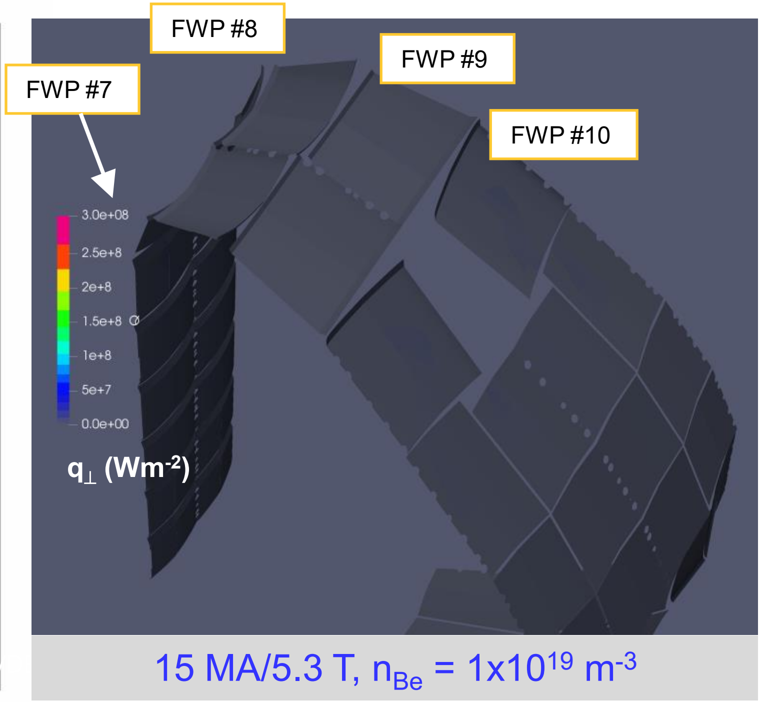
- Deliberate unmitigated VDE at JET:
  - Deposits ~10MJ over ~10ms



- W melts at much higher temperature (~3400°C)
  → More robust to unmitigated disruptions at high current (melting expected from ~12 MA)

[J. Coburn et al. NF 2022]





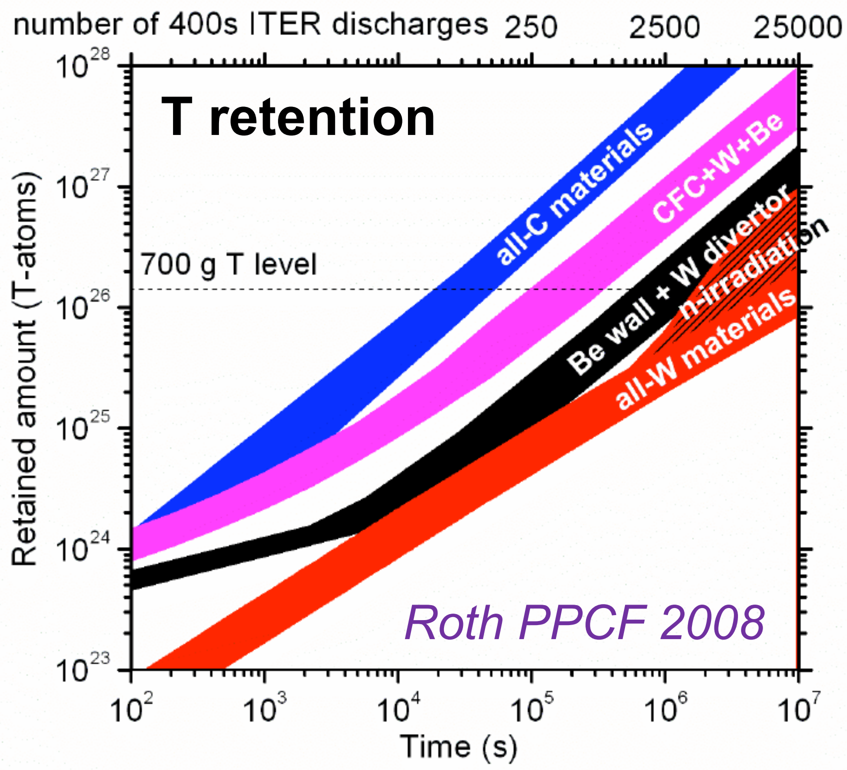
15 MA/5.3 T, n_{Be} = 1x10^{19} m^{-3}

- Runaway Electrons are a serious issue for both Be and W:
  → No active FW cooling in 1st operation phase (SRO)
  → Increased W wall thickness planned to make it more RE-resilient

Why going from Be to W Blanket First Wall (2/2)

- Other concerns with Be:
  - T retention by co-deposition
  - Potential for significant quantities of Be dust to be generated
  - Health and safety concerns with Be, with associated licensing challenges

- W operation:
  - Eliminates Be health risks
  - Reduces T retention
  - Produces less dust
  - Is more reactor relevant



- Drawbacks for using W instead of Be:
- Increased P_{rad} (by 50% in H-mode)
- Needs boronization (via GDC) to getter oxygen to reduce W influx to improve plasma startup
- Maximum boronization cycle every 2 weeks, set by the need to avoid dust generation and T retention

The ITER new baseline

** Site electrical-power limits cap injection in DT-2 at ~110 MW*
TBD = To Be Defined

IC-I
(~18m)

SRO
Start of Research Operation

Inertially cooled FW

27 months

H, D plasmas

ECH	40 MW
ICH	10 MW
Total	**50 MW**

- H-mode in D (5-7.5MA / 2.65T)
- L-mode in H at full W_{mag} (15 MA / 5.3T)
- Neutrons < $1.5 \cdot 10^{20}$

IC-II
(~10m)

DT1
Fusion Power Operation

Water-cooled FW

5 x 2 years

H, D, DT plasmas

ECH	60-67 MW
ICH	10-20 MW
NBI	33 MW
Total	**103-120 MW**

- $Q \geq 10$ and $P_{fus} = 500\ MW$ for 300-500s
- Neutrons < $3.5 \cdot 10^{25}$

IC-III
(TBD)

DT2
Fusion Power Operation

Water-cooled FW

TBD

DT plasmas

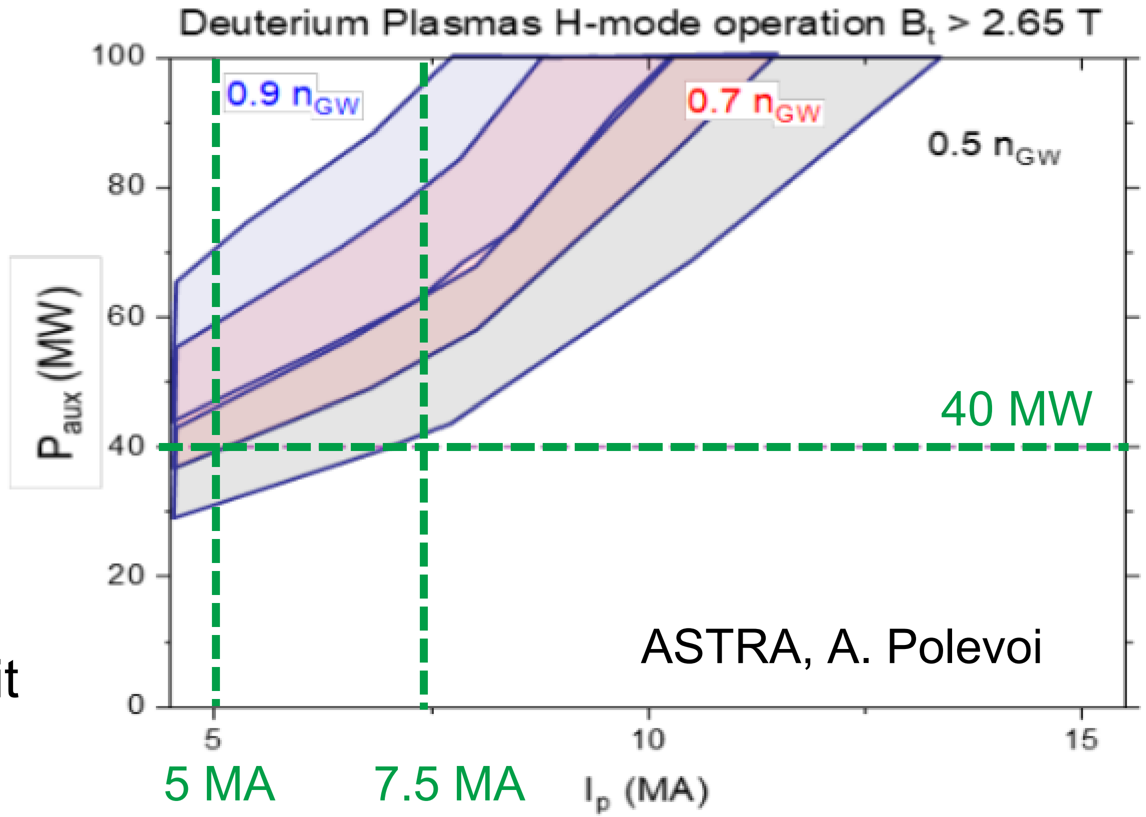
ECH	60-67 MW
ICH	10-20 MW
NBI	49.5 MW
Total	**120-137* MW**

- Long pulse, steady-state $Q \geq 5$ for 1000-3000s
- Neutrons < $3 \cdot 10^{27}$

Main motivations for the chosen heating mix

- Early H-mode access: large power in SRO (ECH 40 MW)
- W wall: high W radiation from core accumulation
  → Central heating to expel W from the core, mostly ensured by ECH (at low density / early ramp-up)
- Neutron budget: limited fluence in SRO and DT-1
  → Scenarios developed in D to minimise neutron production while developing path to Q ≥ 10
  → D H-mode plasmas need more external power than DT (to replace missing alpha power)
- Boronization: W wall → Oxygen to be removed via boronization, but larger T retention → ICWC to remove it (at least 5 MW ICH)

Auxiliary power for H-mode operation at various densities and I_p at 2.65T

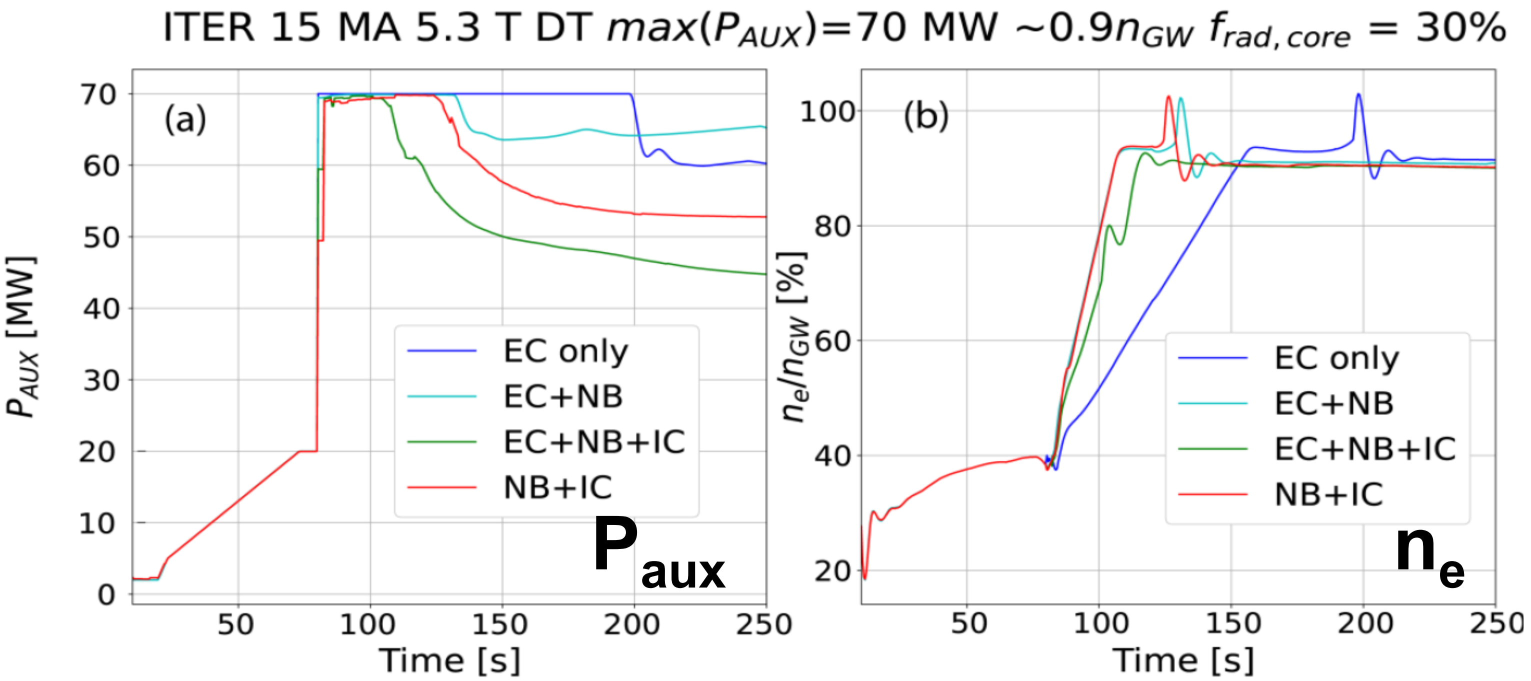


[Loarte PPCF 2025]

- ICH-induced W sputtering:
  Antenna phasing to be optimized to reduce RF-induced impurity generation
  → Suitability to be proven with a full W wall on ITER with 10 MW ICH
  → Decision to update ICH power to 20 MW based on SRO results (coupling, PWI, heating)

Heating mix and fusion gain

- Build-up of alpha heating essential to access burn conditions
- L-H threshold rises with density → 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

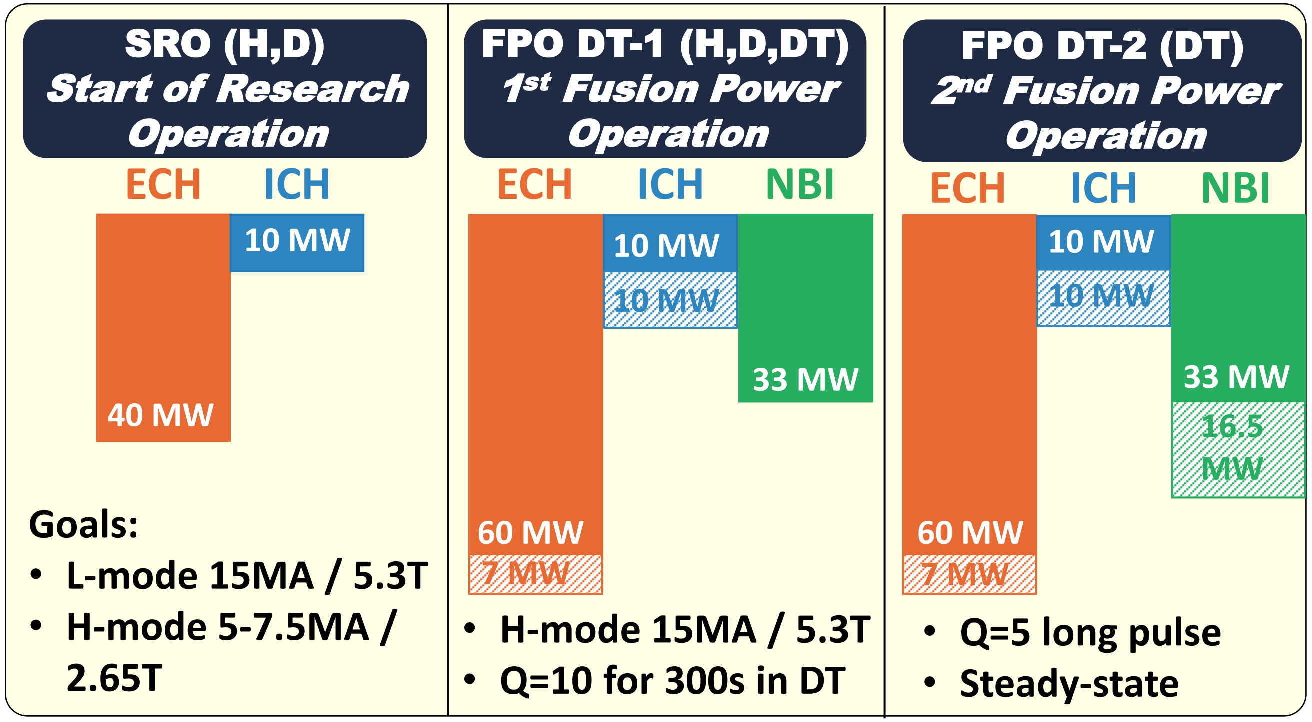


[Yoon IAEA FEC 2026]

- Fusion needs hot ions, ECH heats only electrons → slower density ramp for collisional processes to ultimately heat ions and ensure enough fusion power to sustain H-mode
- Slightly higher fusion gain with NBI+ICH heating (due to more ion heating)

H&CD in the ITER new baseline

Phase	H&CD mix	Task
SRO	40 MW of ECH	Enable H-mode operation in SRO
	10 MW ICH operation to be validated before any upgrade to 20 MW for DT-1	ICH operation in full W wall: ICH coupling & ICRF-specific sputtering tests
		Fuel removal tests with ICWC
DT-1	+60-67 MW ECH +10 MW ICH? +33 MW NBI	Enable H-mode development at low neutron fluence to high I_p in DT-1
DT-2		Enable access to high Q for P_{rad}/P_{tot} > 0.3
	HNB-3 +16.5 MW NBI	Enable Q=5 steady state operation for DT-2



[Loarte PPCF 2025]

The ITER EC system

EC global principle, functionalities and operational constraints

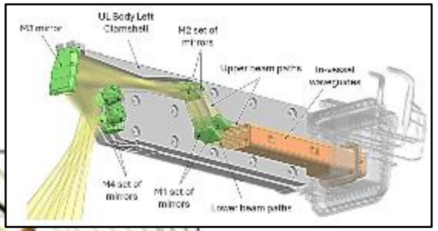
Wave frequency ω to match electron cyclotron frequency Ω_{ce} for energy transfer:

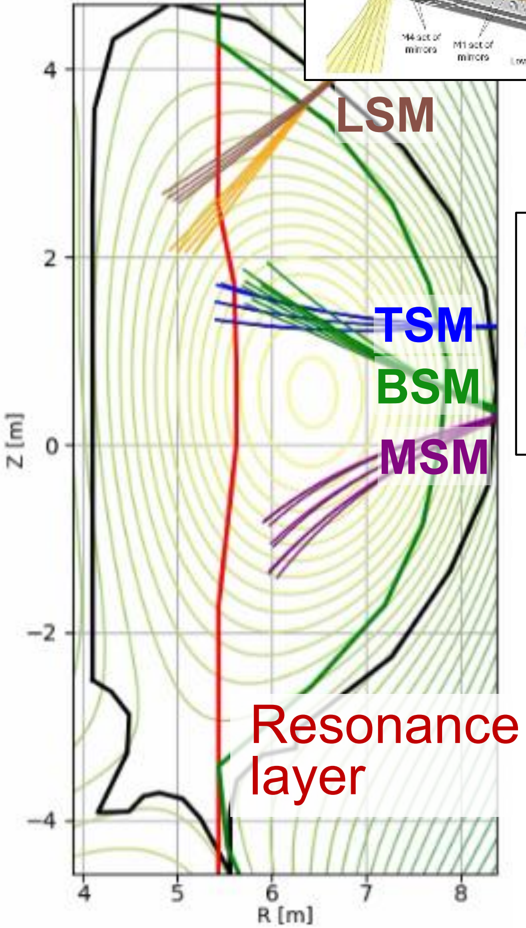
$$\omega - n\Omega_{ce} - \vec{k} \cdot \vec{v} = 0$$

- $\Omega_{ce} = eB/m$ → Vertical resonance layer
- $\omega \sim 170\ GHz$ → No evanescence region

EC functionalities
• Assisted plasma startup • Heating (electrons) • Current drive • MHD control (NTM, sawteeth, Alfven) • Core tungsten removal • Current profile control • Wall conditioning (ECWC)
Operational constraints
• Magnetic field (resonance condition) • Local pressure ($T_e n_e$ for absorption) • Local density (n_e X1 cutoff at 5.3T)

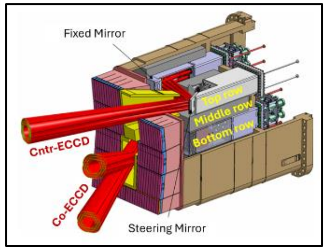
Upper launchers





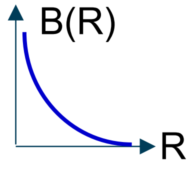
Upper Launcher (UL):
- Off-axis H&CD
- (N)TM control

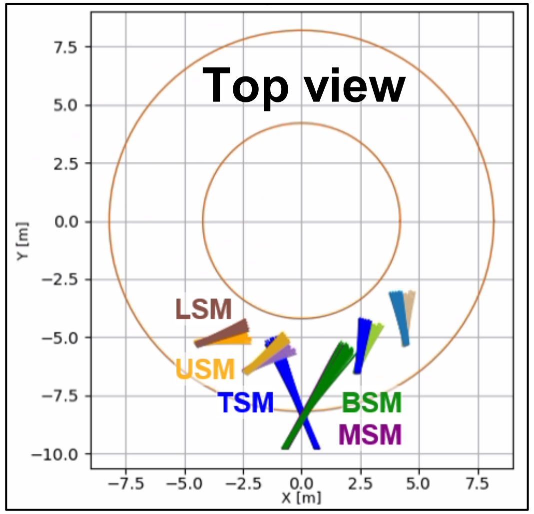
Equatorial Launcher (EL):
- Breakdown/Burn-through
- Central H&CD
- Sawtooth, Alfven control
- Current profile shaping



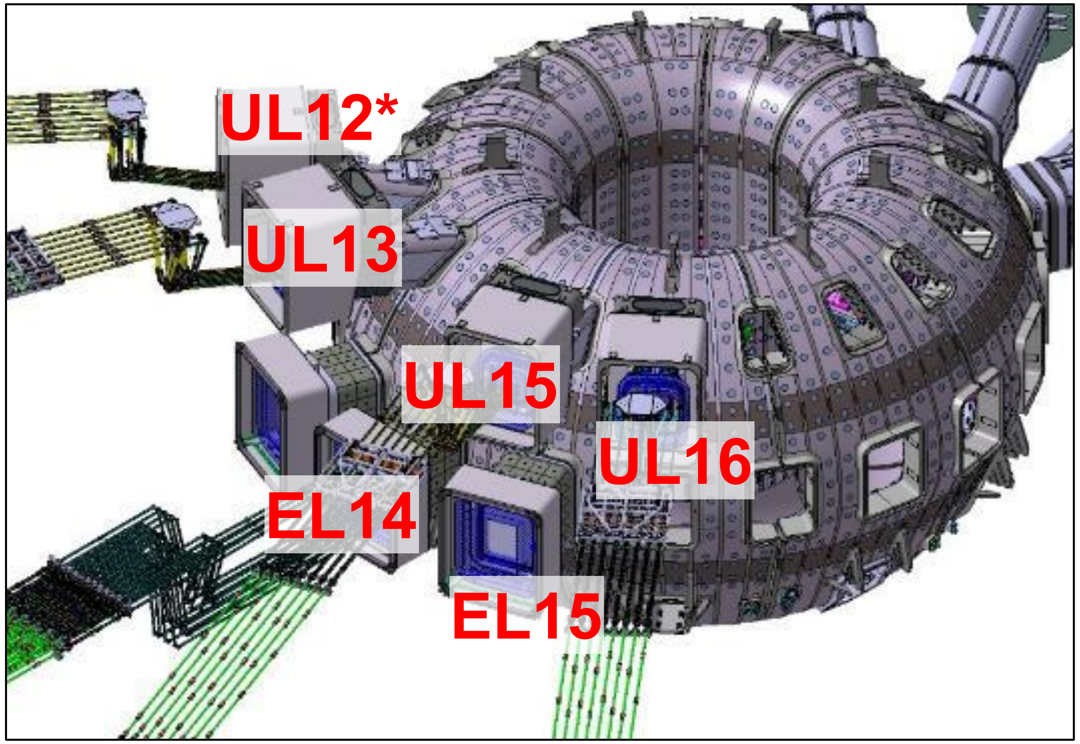
Equatorial launcher

$$B = \frac{B_0 R_0}{R}$$





The EC system in the ITER new baseline



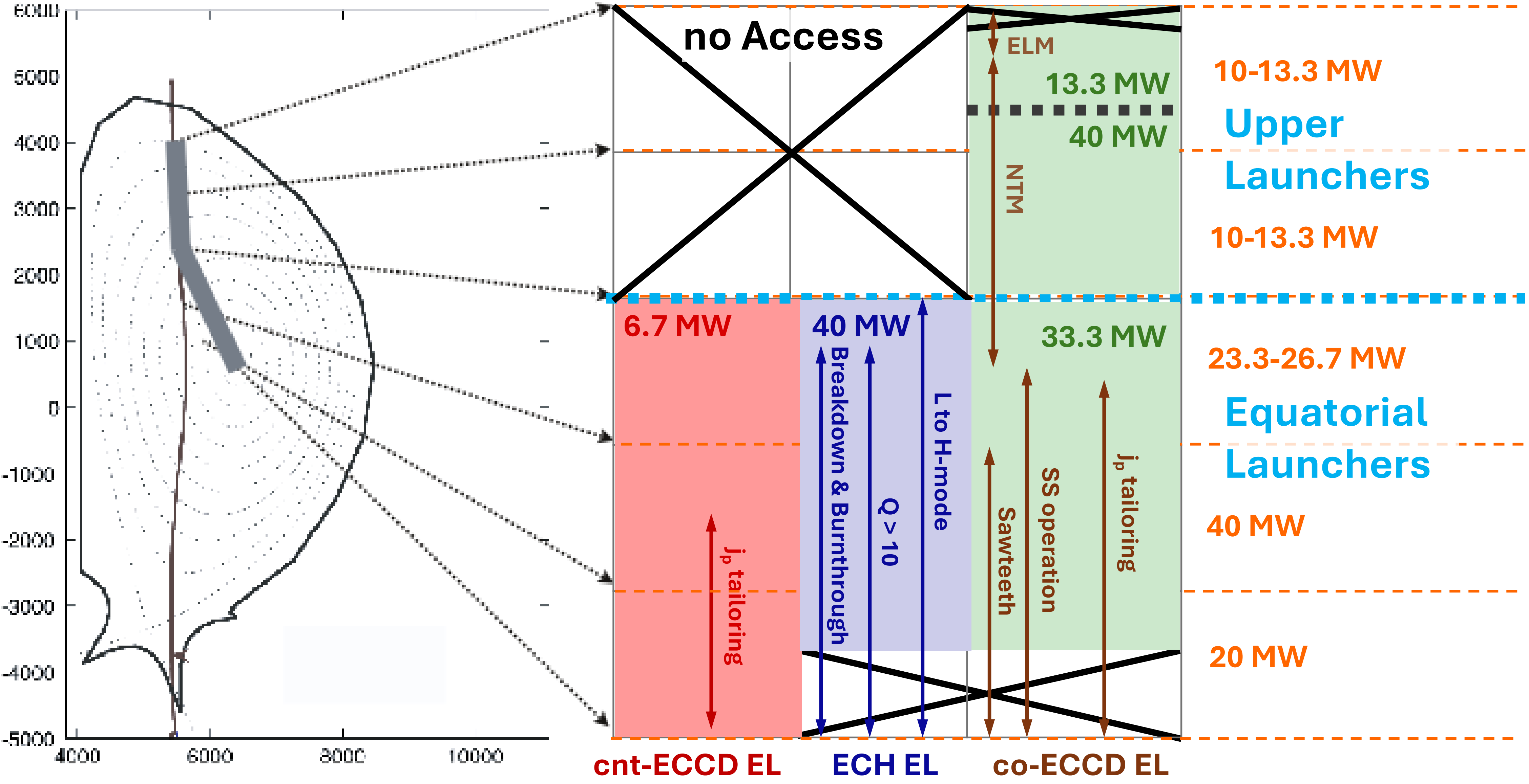
- Equatorial Launcher (EL14): 3 mirrors (x 8 beams)
  - EL/TSM (counter-current!)
  - EL/MSM
  - EL/BSM

  (EL15: all mirrors co-current)
- Upper Launcher (UL): 2 mirrors (x 4 beams)
  - UL/USM
  - UL/LSM
- Power to the plasma: **0.83 MW / beam**
- Frequency: **170 GHz**

	Start of Research Operation	Fusion Power Operation DT
Equatorial	**1 EL** → 24 beams → 20 MW	**2 EL*** → 48 beams → 40 MW
Upper	**3 UL (4* installed)** → 24 beams → 20 MW	**3-4* UL** → 24-32 beams → 20-27 MW
Total	48 beams (56 installed) → **40 MW**	72-80 beams → **60-67 MW**

UL*, EL* = (possible) upgrades

EC functionalities on ITER



Operability of the EC system

- The Plasma Control System will control in real-time the EC wave power, polarization and steering angles

Functionality	Description
Latency	≤ 3ms
Power range	0 to 40 MW in SRO, 0 to 60-67 MW in DT-1 and DT-2
Power response time	10 ms without modulation
Power modulation	Full power modulation up to 5 kHz with a variable duty cycle
Steering angle	Mirrors poloidally steerable with full swing in ~3s (slew rate: 6°/s) Mirrors NOT toroidally steerable
Polarization	O-mode or X-mode, ~3s to transit from one to the other

- Important operational constraint: 2 gyrotrons connected to 1 power supply → power granularity = 0.83 x 2 = 1.67 MW without modulation (disconnecting 1 gyrotron during operation takes ~3s)
  → To be taken into account in scenario design to minimize the utilization of the duty cycle

Applications of the ITER EC system in the ITER Research Plan

Functionality	How the EC system will help
Robust plasma initiation	EC-assisted plasma breakdown and burn-through: • Help to ionize the prefill gas • Accelerate electrons for Townsend avalanche driven by E-field • Increase plasma temperature during burn-through
Optimization of current ramp-up	ECH helps to avoid W radiative collapse and reduce magnetic flux consumption
Current drive	• ECCD as non-inductive current drive • Current profile shaping to enhance bootstrap current for hybrid and steady state operation
Minimize core W	ECH as central heating to expel W from the core
H-mode access	ECH (+ other H&CD) such that $P_{sep} > 1.5\ P_{L\text{-}H}$
High fusion gain	ECH (+ other H&CD) to increase plasma temperature
MHD control	(N)TM control, Sawtooth control and Alfven Eigenmodes
Condition First Wall	Electron Cyclotron Wall Conditioning (ECWC)

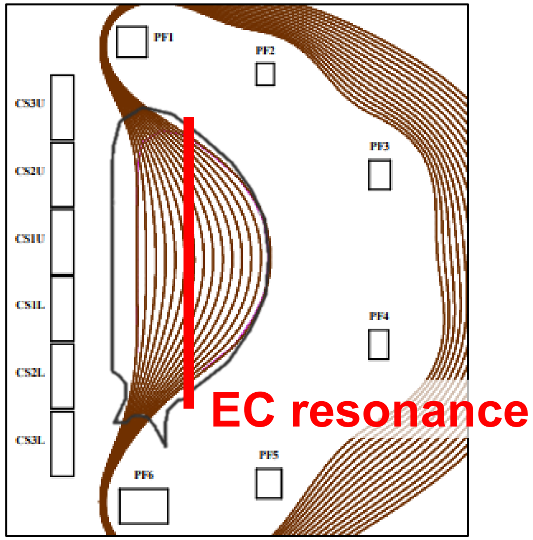
EC-assisted plasma startup

Initiation = **breakdown** (ionization) + **burnthrough** (heating)

- **Breakdown:**

j_{coil} variation → E-field formation → Acceleration of electrons → Ionization (avalanche) → Plasma

- **Burnthrough:**
  → Temperature increase → **Enhanced by ECH**



→ EC will assist plasma initiation to ensure robust start-up for a wider range of plasma conditions.

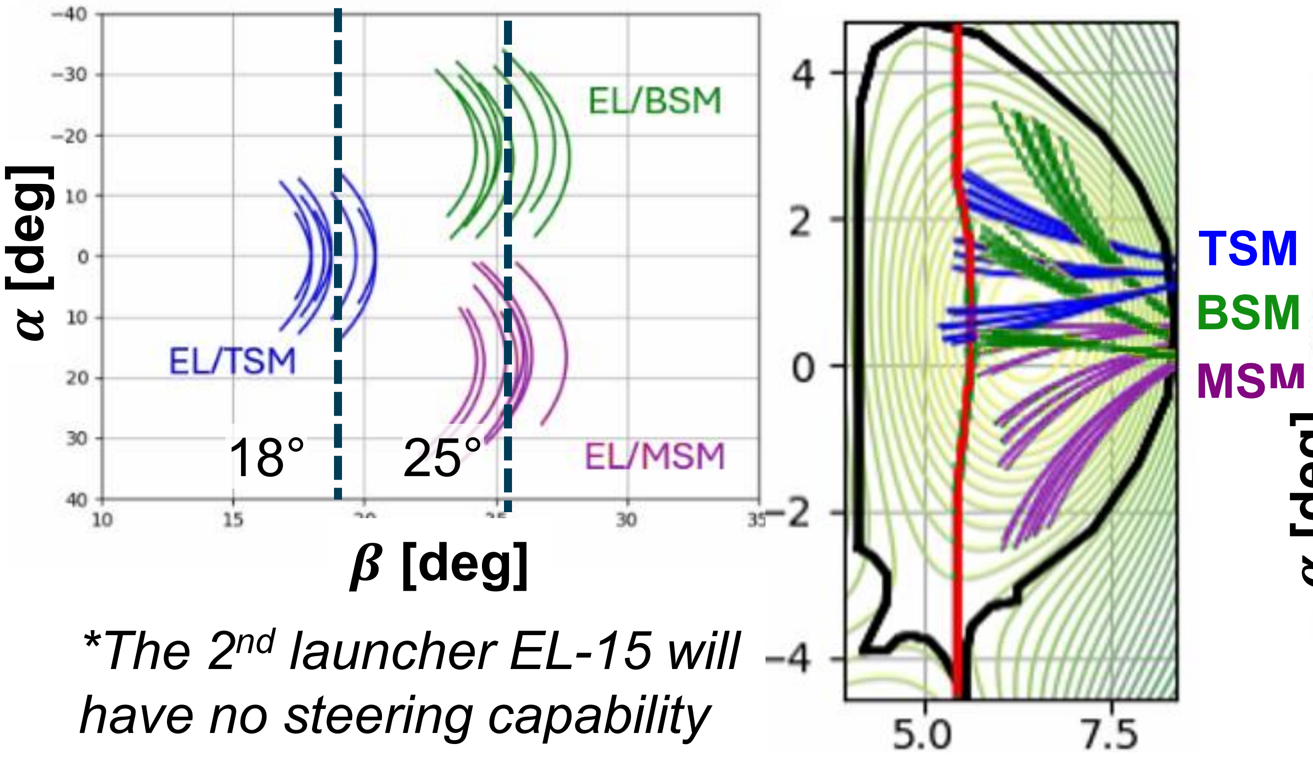
Steering angles of EC mirrors

Mirrors of ITER EC launchers are poloidally steerable but toroidally roughly fixed:

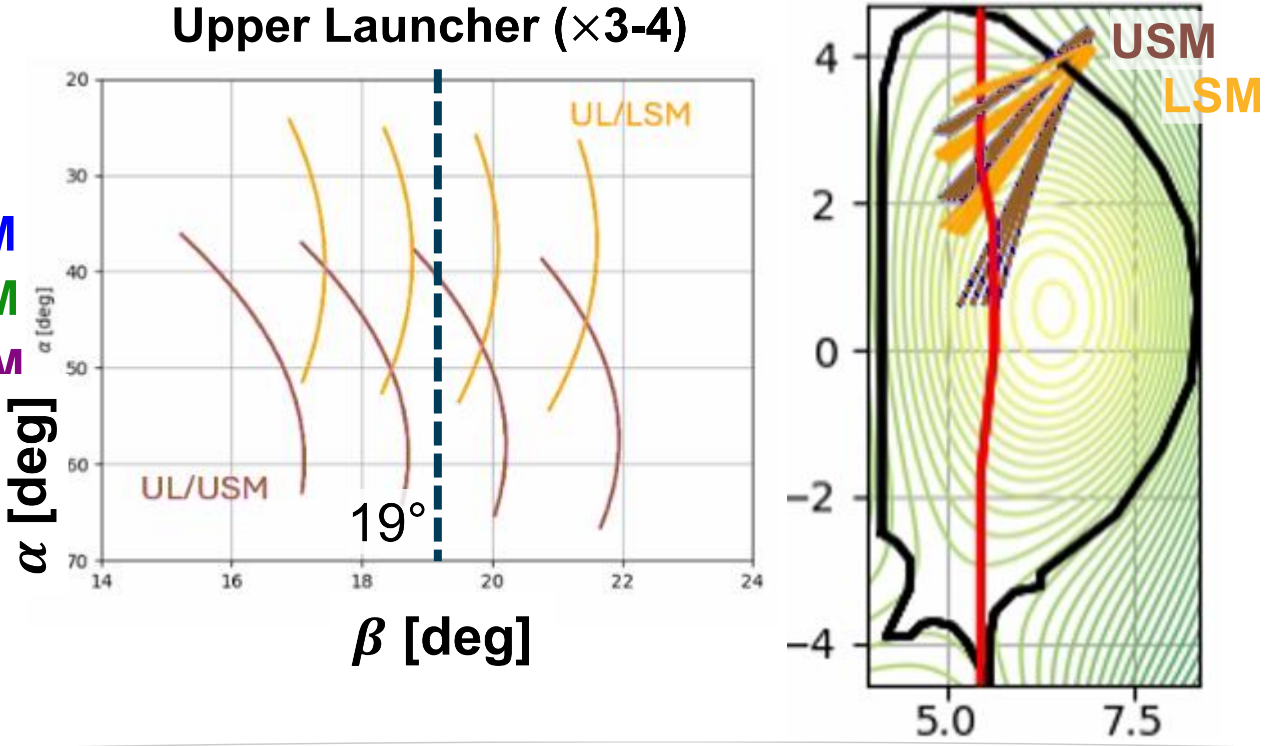
- Poloidal steering angle: α
- Toroidal steering angle: β

	α poloidal steering range (°)	r/a radial coverage
EL/TSM	[-10.0, 9.9]	[0 - 0.4]
EL/MSM	[5.2, 30.3]	[0.05 - 0.6]
EL/BSM	[-30.2, -5.1]	[0.05 - 0.6]
UL/USM	[37.43, 64.8]	[0.4 – 0.8]
UL/LSM	[25.45, 53.0]	[0.65 – 0.88]

Equatorial Launchers (EL14)*



Upper Launcher (×3-4)



**The 2nd launcher EL-15 will have no steering capability*

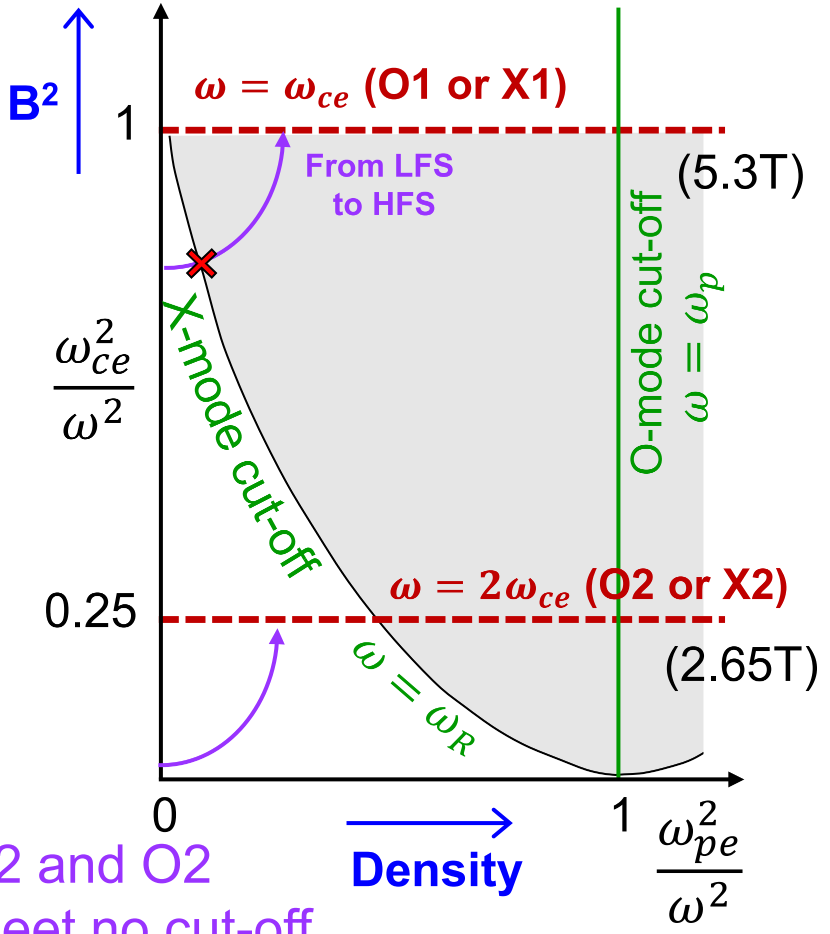
EC operation in ITER: the CMA diagram

[Schneider RFPPC 2025]

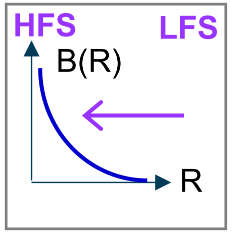
Resonance $\omega_{ce} = \frac{eB}{m_e}$

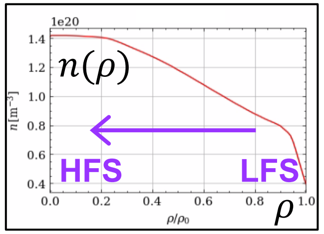
X-mode cutoff $\omega_R \approx \sqrt{\left(\frac{eB}{2m_e}\right)^2 + \frac{n_e e^2}{\varepsilon_0 m_e}} + \frac{eB}{2m_e}$

- Effect of density profile: what occurs when the beam travels from LFS to HFS?

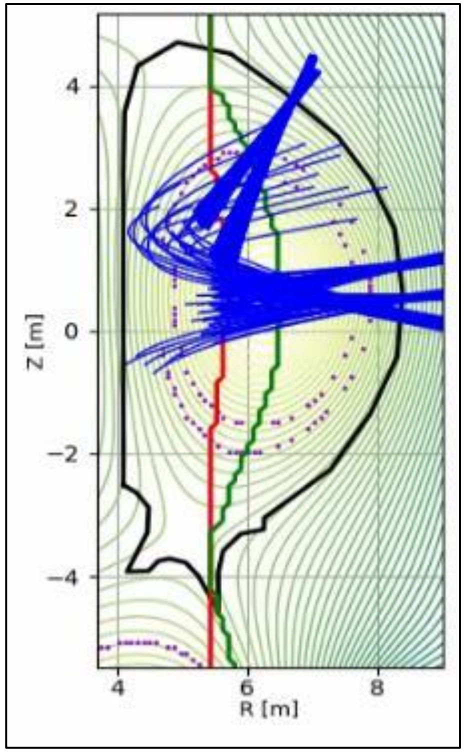


X2 and O2 meet no cut-off



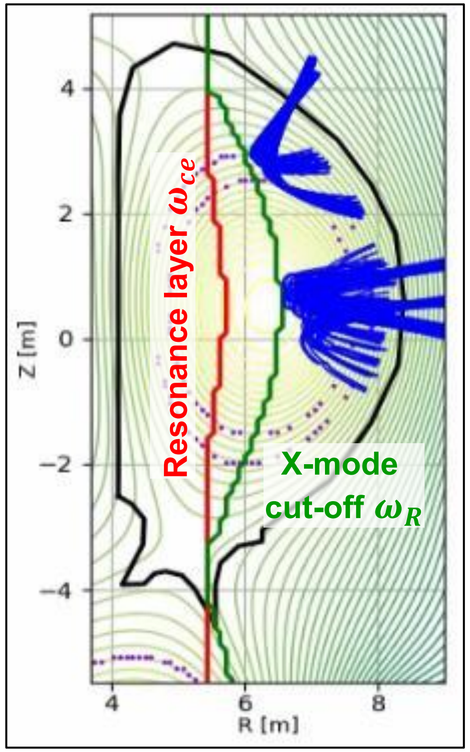


O1 beams



O1 goes through and absorbs at the resonance

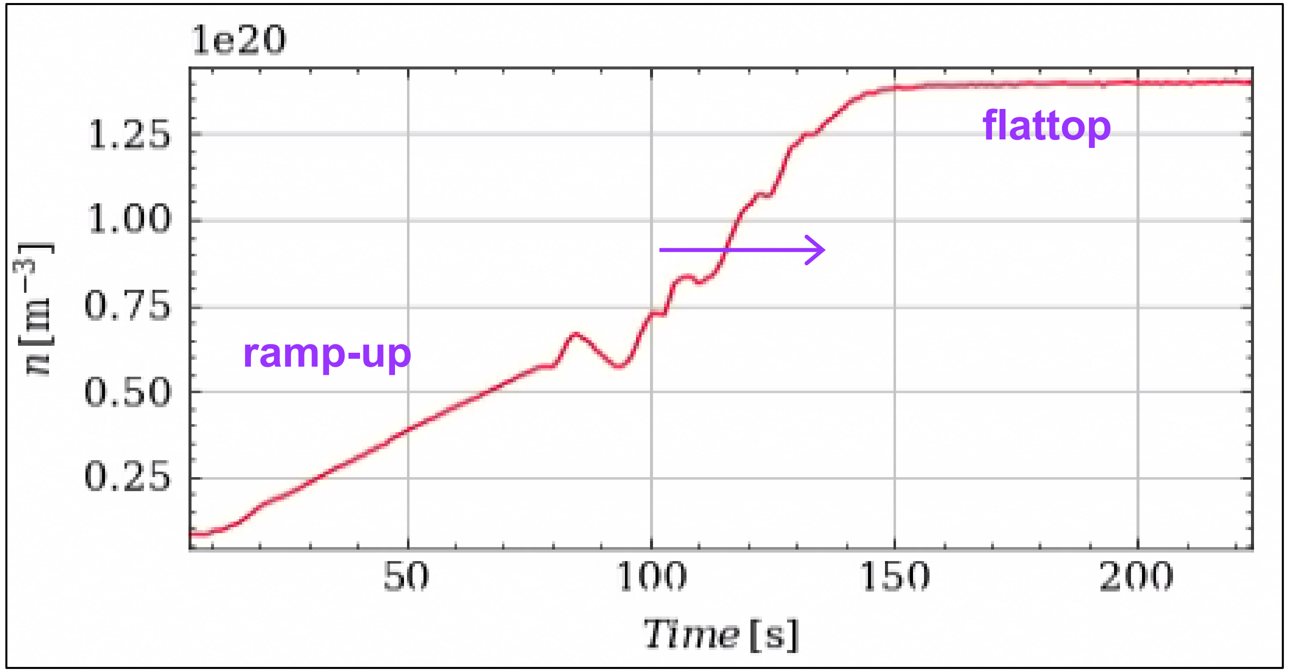
X1 beams



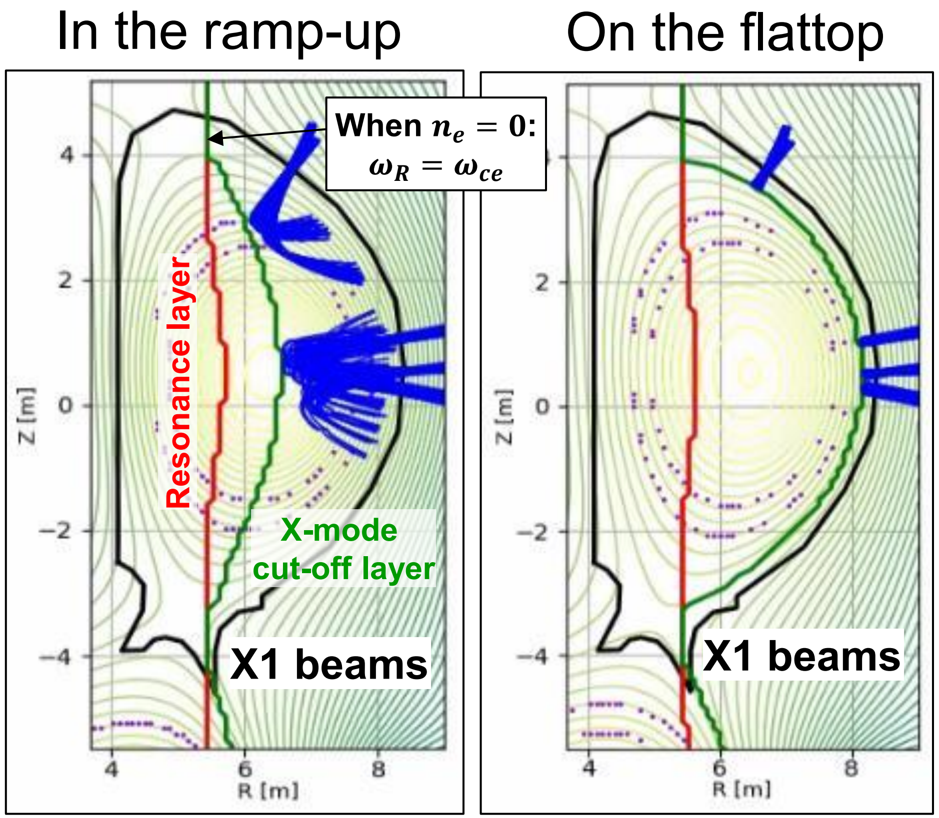
X1 reflects at cut-off

EC operation in ITER: dynamics of the density cutoff at 5.3T

- Effect of time evolution: how does the X1 cut-off layer evolve during the pulse?



[Schneider RFPPC 2025]



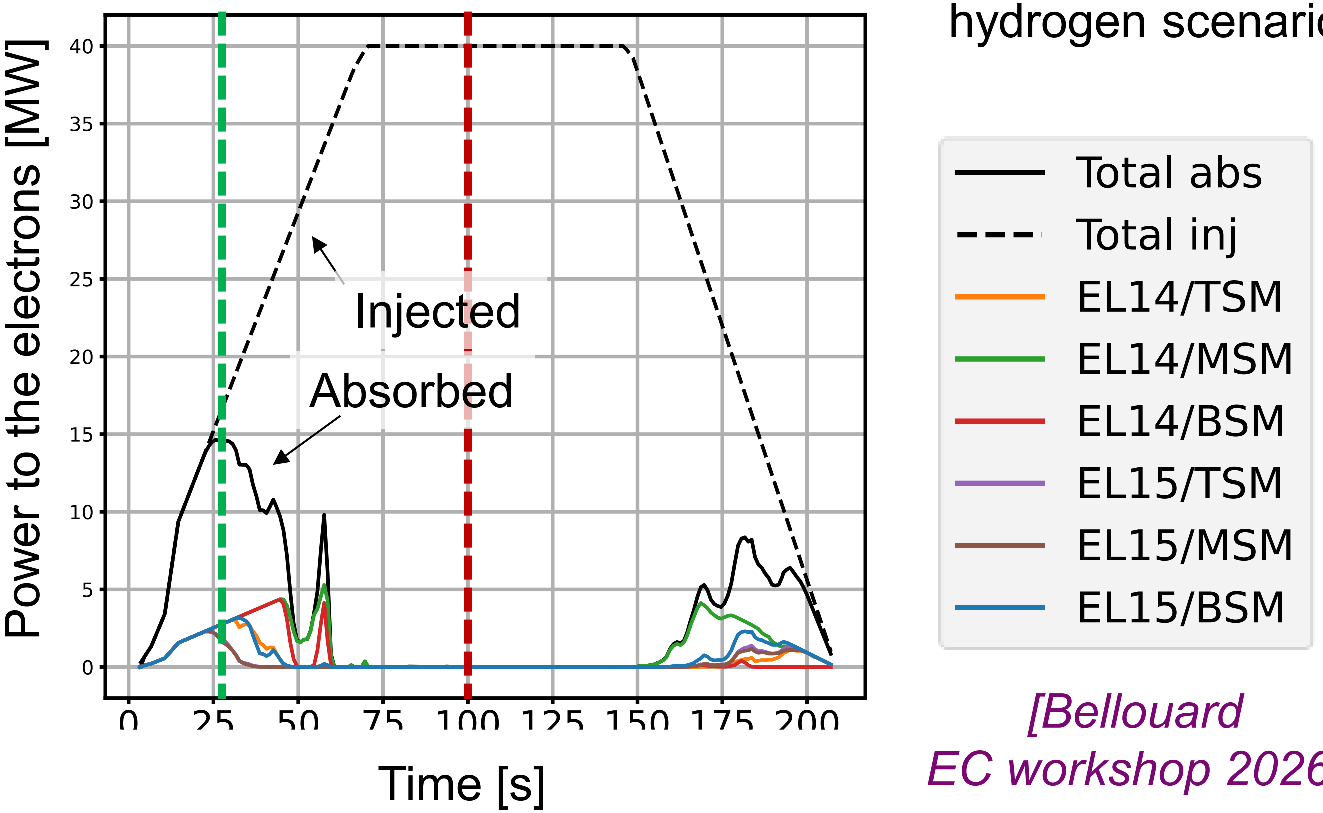
The X1 cut-off moves to the LFS when the density increases → At high density X1 completely reflected

→ At 2.65T, no cutoff involved, both X2 and O2 polarizations are possible
→ At 5.3T, O1 workhorse, but X1 can be foreseen during early ramp-up (cutoff close to resonance)

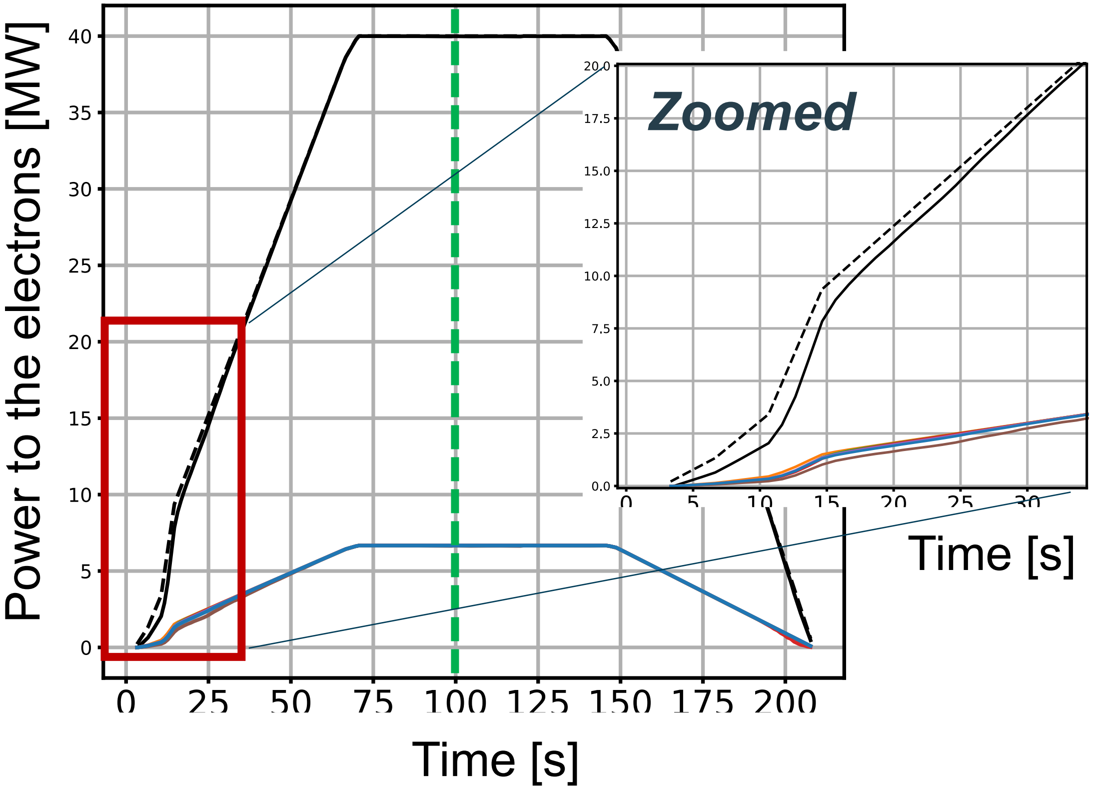
X1 and O1 polarization in early ramp-up and late ramp-down at 5.3T

SRO L-mode 15MA/5.3T hydrogen scenario

X1 absorption



O1 absorption



[Bellouard EC workshop 2026]

→ X1 can be used until ~25s in the early ramp-up and during the late ramp-down

→ O1 absorbed by 75% to 95% until ~20s ramp-up and late ramp-down

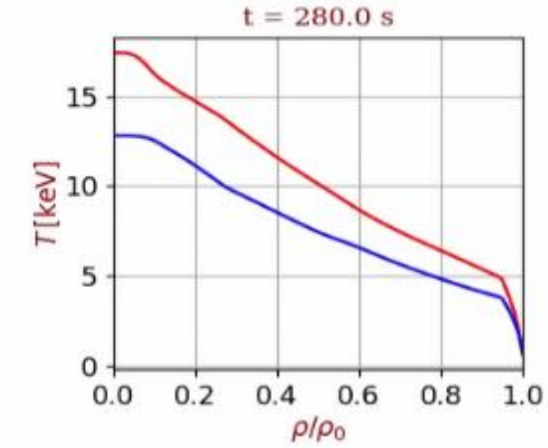
Non-absorbed EC power from O1 shine-through losses may be detrimental for the ITER first
→ Using X1 polarisation for early ramp-up and late ramp-down seems optimal.

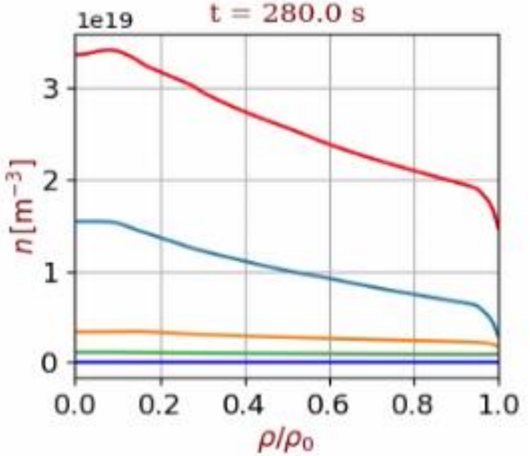
3rd harmonic parasitic edge absorption at 2.65T in H-mode

- At 2.65T: central X2, with X3 at the edge in the plasma
- H-mode : large edge $n_e T_e$ → Edge absorption!

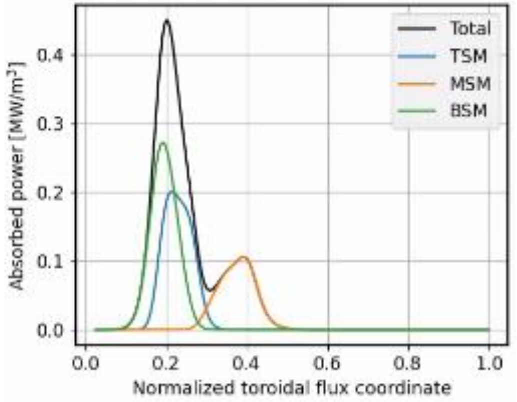
} Edge X3 parasitic absorption!

- Example: JINTRAC 2.65 T / 7.5 MA, H-mode (IMAS 104104/33/public/ITER)

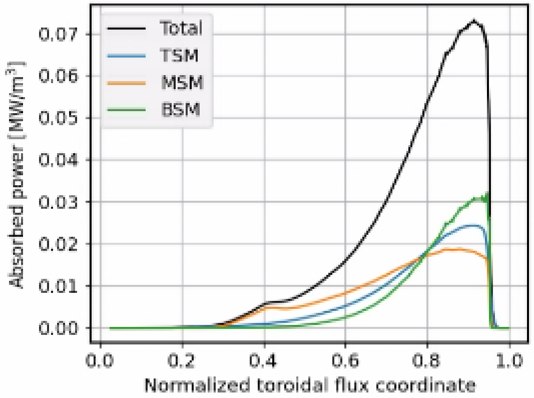




Without X3



With X3



- In L-mode: small edge $n_e T_e$ → negligible X3: operation in X2 is the most appropriate

- Increasing the B-field would be detrimental for central H&CD with EL and for (N)TM control with UL

→ Solution: switch from X2 to O2 polarization during an L-H transition

[Schneider RFPPC 2025]

Heating

iter

Optimal polarization for EC operation in ITER

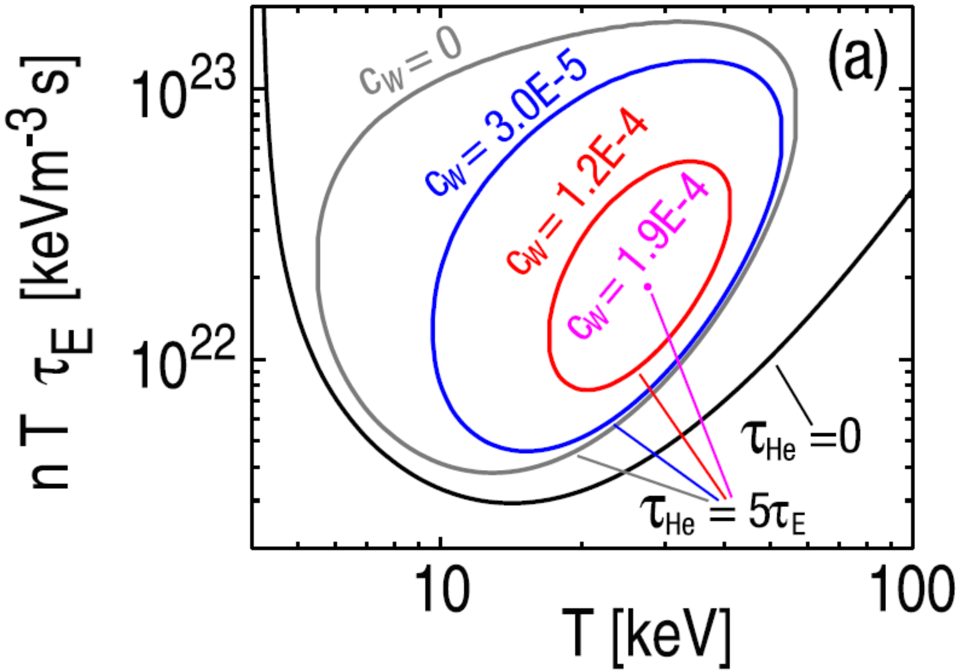
	Ramp-up (L-mode)	L-mode on flattop + ECWC	H-mode on flattop
2.65T	**X2** Better absorption in X-mode at low n_e & T_e	**X2** Better absorption in X-mode and no edge X3 due to low edge T_e and n_e	**O2** X-mode prohibited due to edge parasitic X3 from high T_e^{ped} and n_e^{ped}
5.3T	**X1** (only early ramp-up) Better absorption than O1 at low n_e & T_e but cutoff	**O1** Density cut-off with X1	**O1** Density cut-off with X1

[Schneider RFPPC 2025]

Why is tungsten detrimental for fusion plasma performance?

- Tungsten (W) accumulated in the core significantly increases radiation losses
- Lawson criterion fulfilled when α heating compensates for energy losses → Minimum $nT\tau_E$ for ignition

T. Pütterich et al, NF 50 (2010) 025012

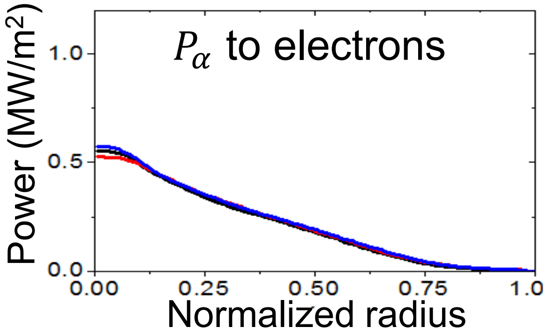


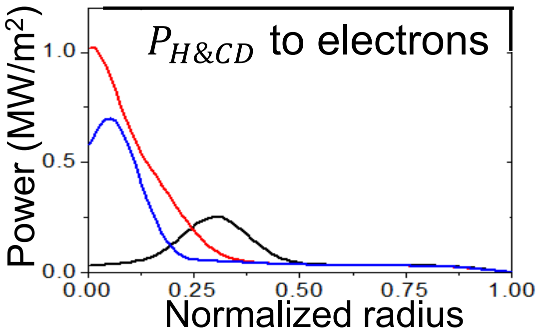
- n = Plasma density
- T = Plasma temperature
- τ_E = Energy confinement time
- τ_{He} = Helium confinement time
- C_w = W concentration

→ The operational space to get plasma ignition is considerably reduced when the W concentration increases.

Tungsten accumulation removal

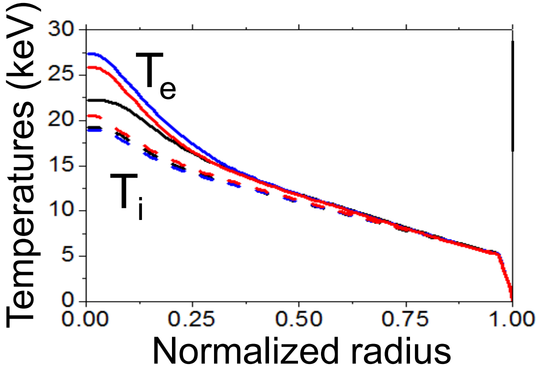
- W divertor and first wall → W accumulation in the core
- In DT, even if P_{α} /$P_{H\&CD}$ ~ 2 for Q=10, ITER H&CD schemes can modify plasma parameters due to higher power density in the core:

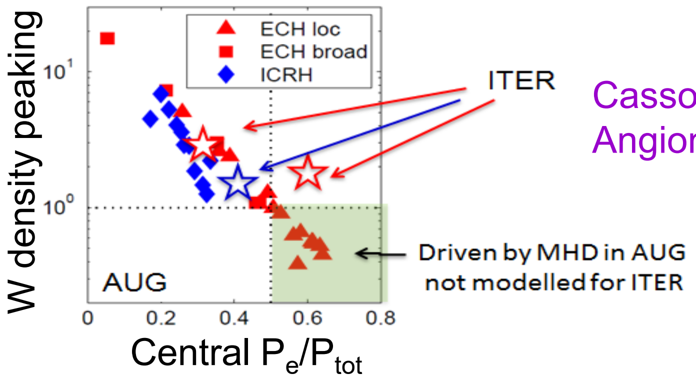




33 MW NBI +
— 20 MW off-axis EC
— 20 MW on-axis EC
— 20 MW on-axis IC

Heating the centre affects W neoclassical transport:





Casson, PPCF (2015)
Angioni, IAEA (2016)

→ Central heating prevents tungsten accumulation in the core.

Tungsten transport

- Which mechanisms lead to W accumulation?
- W particle flux:

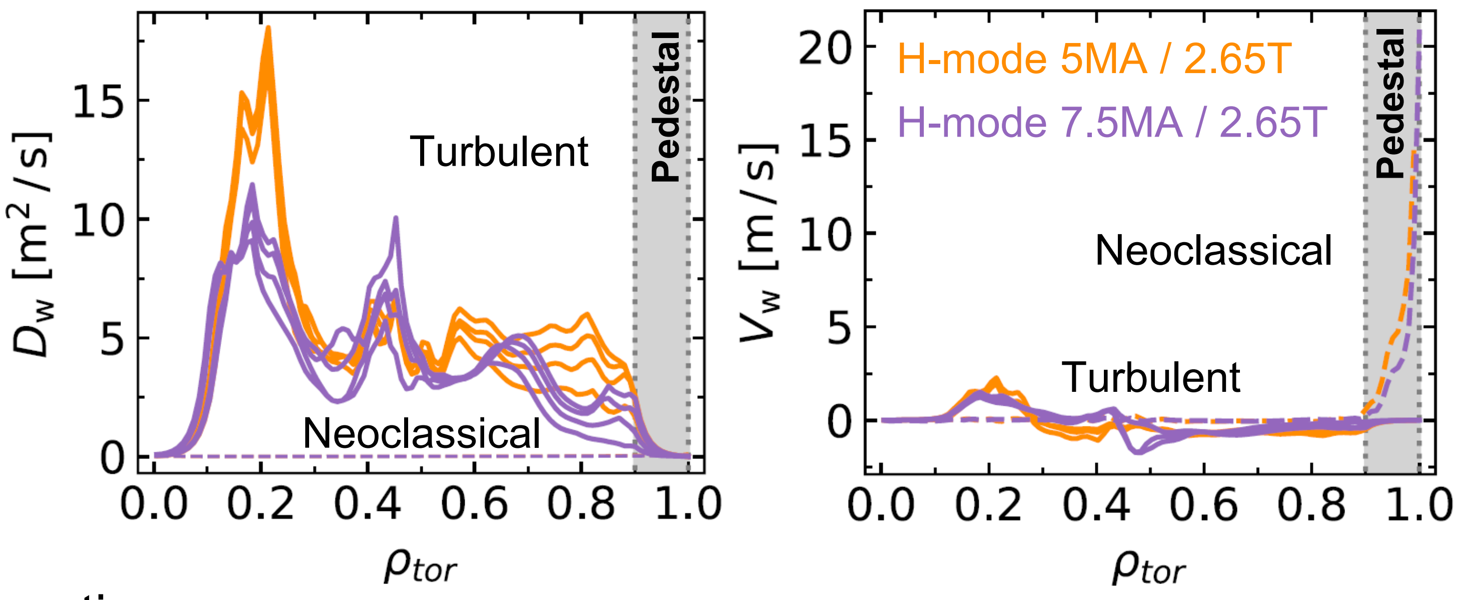
$$\Gamma_W = -D_W \nabla n_W + V_W n_W$$

n_W = W density

Diffusion

Convection

D. Fajardo, PPCF 67 (2025) 015020



$$D_W = D_{turb} + D_{neo} \qquad V_W = V_{turb} + V_{neo}$$

Neoclassical: induced by collisions

Turbulent: induced by microscopic instabilities

How does tungsten travel to the plasma core?

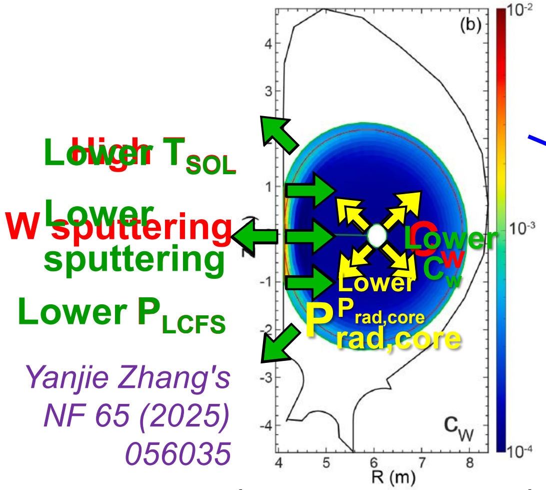
- Mechanisms leading to W transport:

<table>
<tr><th></th><th>SOL</th><th>Edge / Pedestal</th><th>Core</th></tr>
<tr><td></td><td rowspan="3">Transport assumed turbulent</td><td colspan="2">Competition turbulent / neoclassical</td></tr>
<tr><td>Today's devices</td><td>• $\left|\frac{R}{L_n}\right| > \left|\frac{R}{2L_T}\right|$: dominant inwards neoclassical pinch
• Efficient W exhaust with small ELMs</td><td>• L-mode: low confinement, high core-edge transport (no barrier), low power
→ More C_w in H-mode</td></tr>
<tr><td>ITER (+ high T_i JET)</td><td>• $\left|\frac{R}{L_n}\right| < \left|\frac{R}{2L_T}\right|$: outwards neoclassical pinch
• W screening in H-mode
• ELMs induce higher W core accumulation</td><td>• L-mode: more neoclassical transport, locally high W radiative losses
• H-mode: dominant turbulent transport
→ More C_w in L-mode</td></tr>
</table>

W source and transport in limiter phase of ITER L-mode plasmas

Highest risk of W radiative collapse in limiter phase

SOLPS-ITER

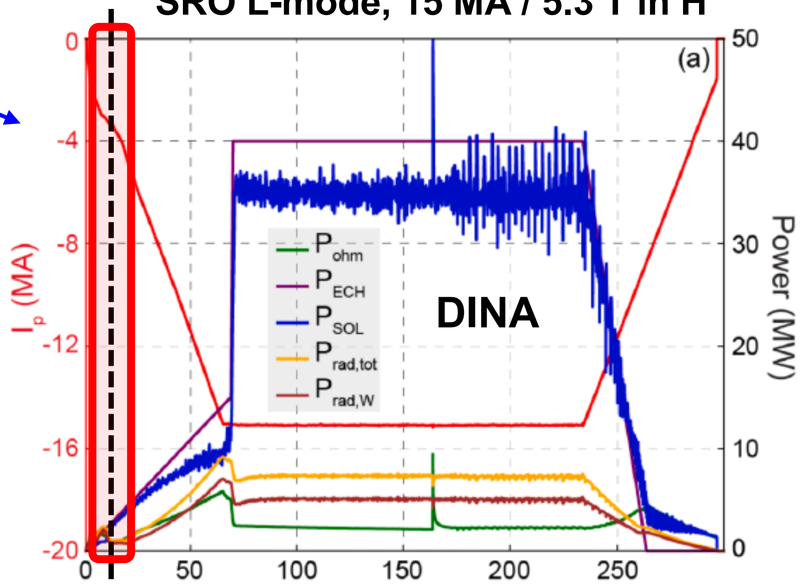


Yanjie Zhang's NF 65 (2025) 056035

(No Ne seeding)

→ Self-regulated mechanism
→ Thermal stabilization

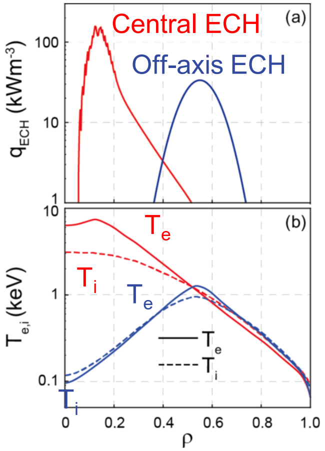
SRO L-mode, 15 MA / 5.3 T in H

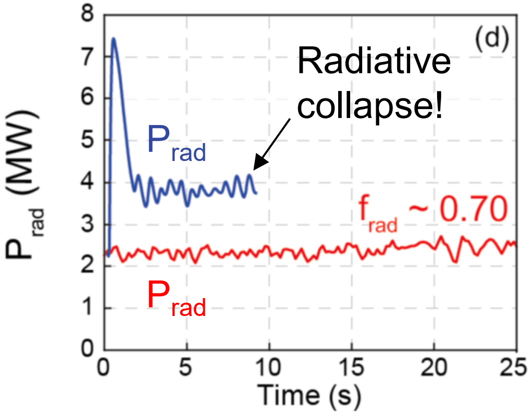


X-point at t = 10 s

R. Pitts et al., NME 42 (2025) 101854

- **Central ECH**
  → T_e rise → Enhanced heat turbulent transport
  → Outwards convective W flux





R. Pitts et al., NME 42 (2025) 101854

- **Density tailoring**
  → Flattened or hollow density profile
  → Increased outwards W turbulent transport

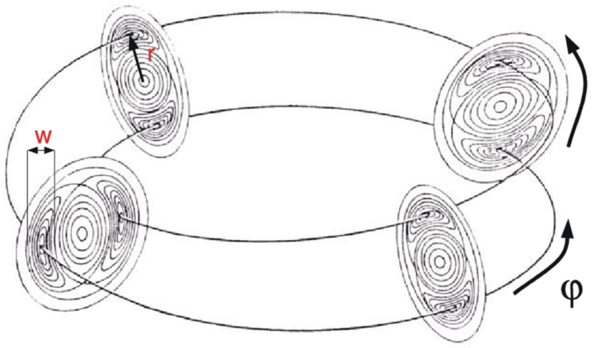
- **Low-Z impurity injection (Neon)**
  → Cool down plasma edge
  → Reduced W sputtering

Heating

But how do we avoid W radiative collapse in this early ramp-up phase?

NTM control with ECCD (1/3)

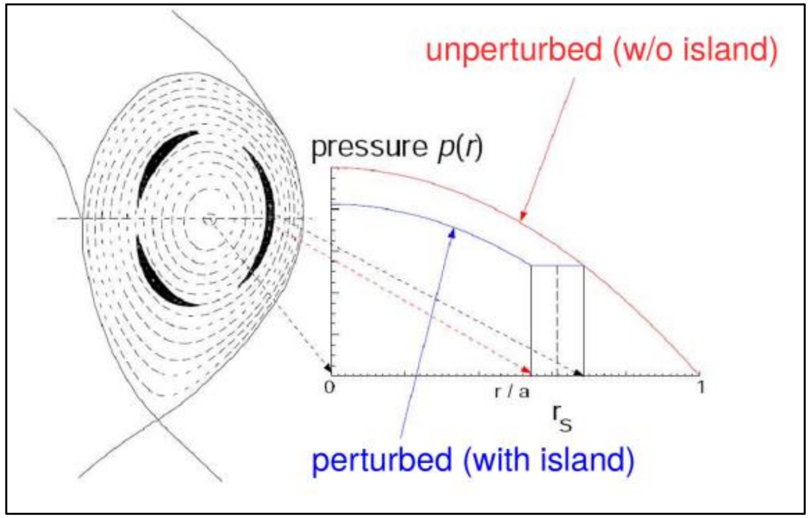
- (Neoclassical) Tearing Modes are current-driven resistive MHD instabilities:
  - Classical tearing modes (TM) reconnect magnetic field lines at rational surfaces → magnetic islands typically at q = 2 or 3/2:



$$\tau_R \frac{dw}{dt} = r_s^2 \Delta'(w)$$

w = island width [m]
τ_R = resistive timescale [s]
r_s = radius of rational surface [m]
Δ' = classical tearing instability [m^{-1}]

  - Island evolution:
    - $\Delta' > 0$: instable and grows
    - $\Delta' < 0$: stable and shrinks



  - Neoclassical tearing modes (NTM) are TM driven by the loss of bootstrap current inside the island:

$$\tau_R \frac{dw}{dt} = r_s^2 (\Delta'(w) + \Delta'_{bs}(w)) \qquad \Delta'_{bs} \sim \frac{\beta_{pol}}{w}$$

→ Degrade energy confinement

NTM control with ECCD (2/3)

- NTMs are harmful:
  - Degrade energy confinement → reduce plasma performance
  - Above critical width (~5-10 cm) → lock mode → disruption

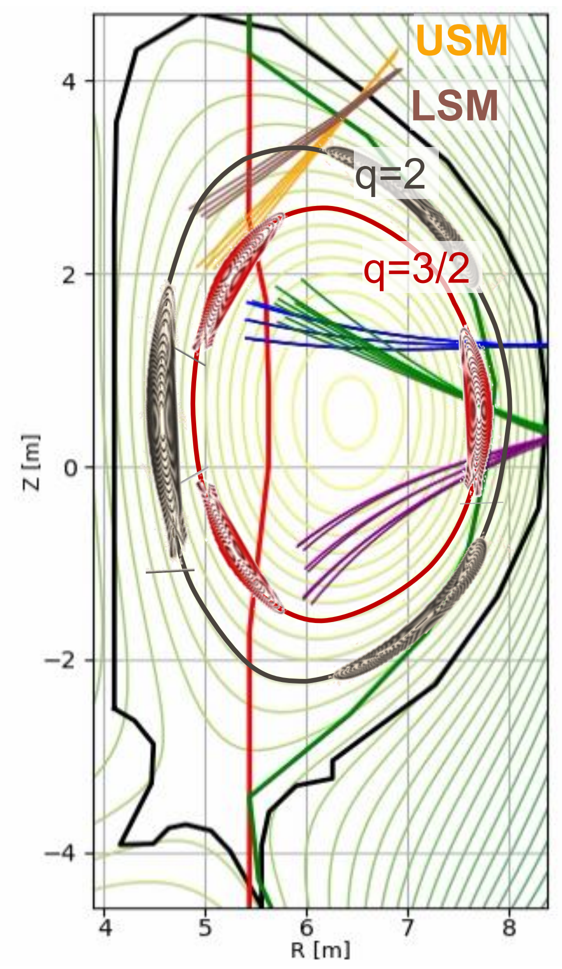
- NTM stabilization via ECCD:

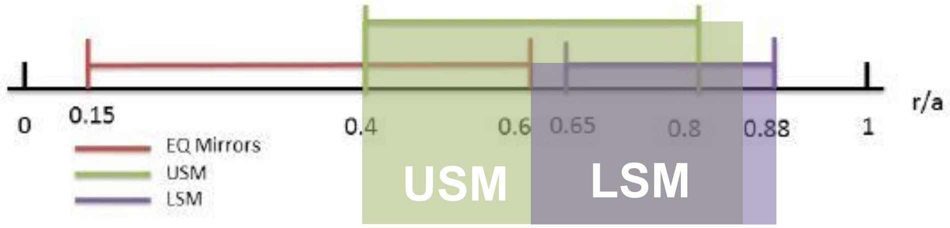
$$\tau_R \frac{dw}{dt} = r_s^2(\Delta'(w) + \Delta'_{bs}(w) + \Delta'_{cd}(w)) \qquad \Delta'_{cd} \sim \frac{j_{cd(w)}}{w}$$

- NTM on ITER:
  - High β amplifies the bootstrap drive
  - Lower rotation leads to faster mode locking
  - Low resistivity leads to slower growth rate and more efficient ECCD

- The Plasma Control System sends the power to be applied and radial location to hit

- Default assignment:
  - USM mirrors on q=3/2
  - LSM mirror on q=2

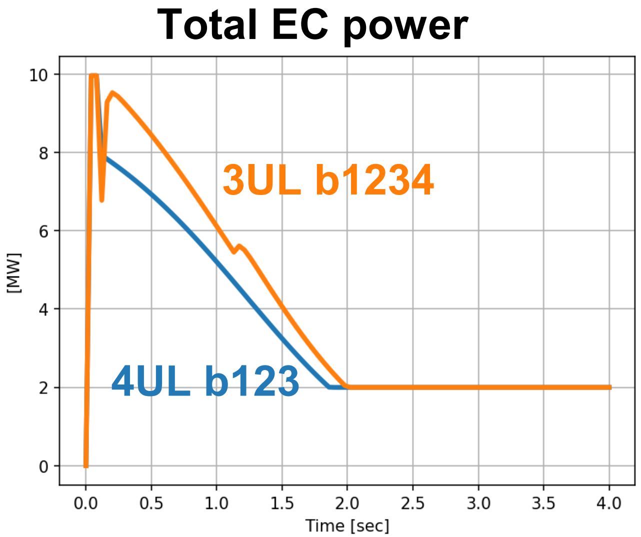


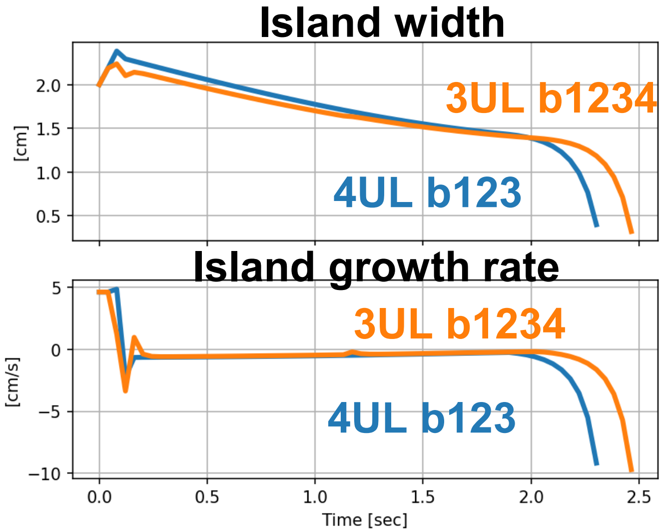


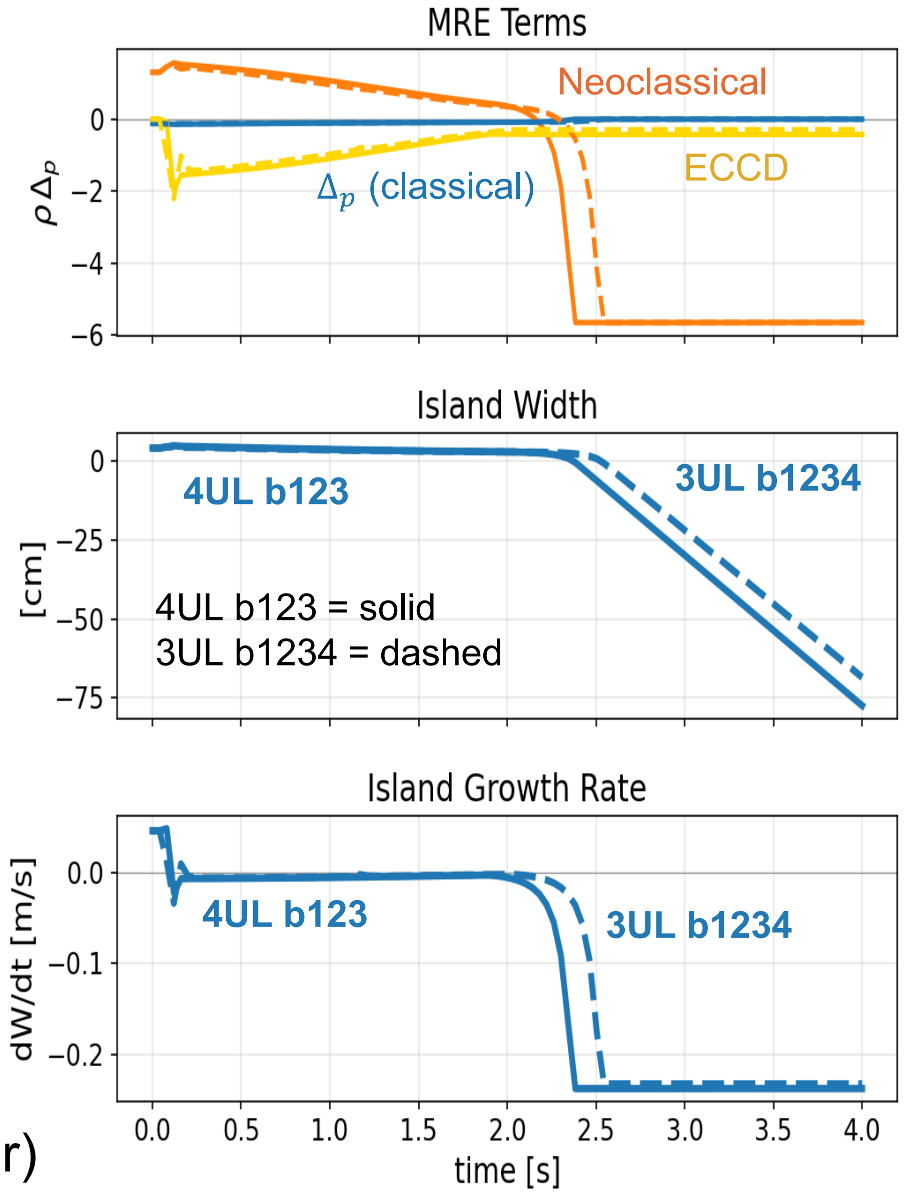
NTM control with ECCD (3/3)

[Schneider EC workshop 2026]

- Using NTM controller in ITER Pulse Design Simulator
- Example of SRO 15 MA / 5.3 T scenario
- Compare 4 USM x 3 beams vs. 3 USM x 4 beams for q=2:







- Using 4 ULs x 3 beams is marginally more efficient than using 3ULs x 4 beams
- In both cases the island is stabilized
- Of course 4 ULs x 4 beams remains more efficient (more power)

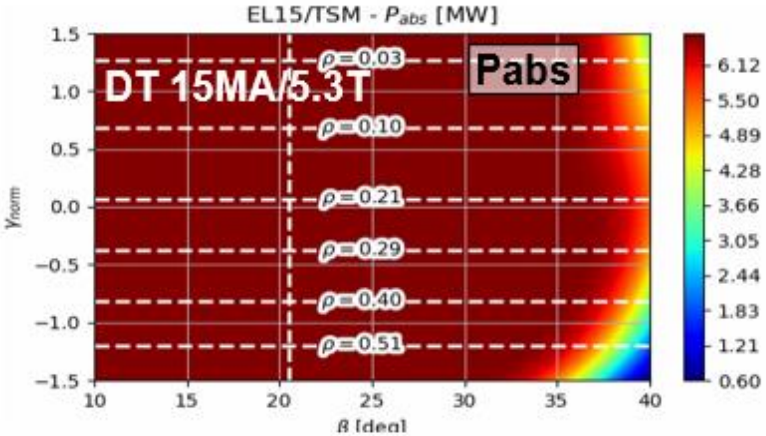
The 2nd Equatorial Launcher (1/2)

- To be installed in EP-15 to inject an extra 20 MW towards the plasma centre
- It will be non-steerable to simplify its design and procurement
- Contrary to the first EL in EP-14, all mirrors will inject co-current
- Preliminary design for optimal fixed β toroidal and α poloidal angles (scan with TORBEAM code)
  - Optimal β chosen to maximise ECCD
  - Optimal α wrt desired ρ location of ECH/ECCD: tune mirror angles for relatively broad ECH slightly off-axis to avoid triggering core instabilities
  - Priority to DT 15 MA / 5.3 T baseline scenario, then compatibility checked on other scenarios
  - Downwards poloidal injection to remain central during plasma termination

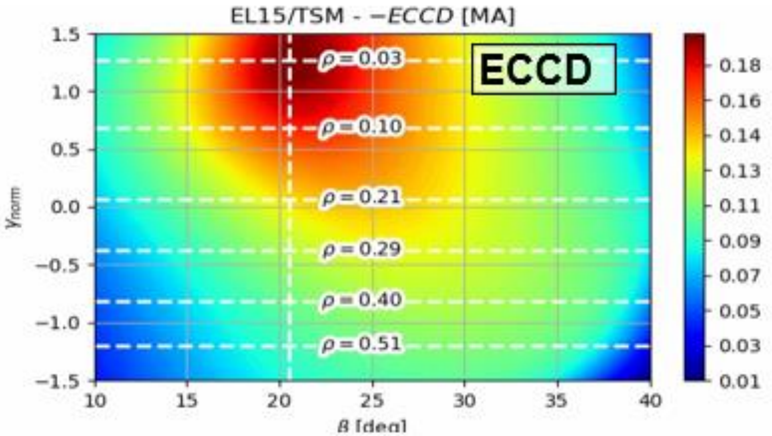
The 2nd Equatorial launcher (2/2)

[Schneider RFPPC 2025]

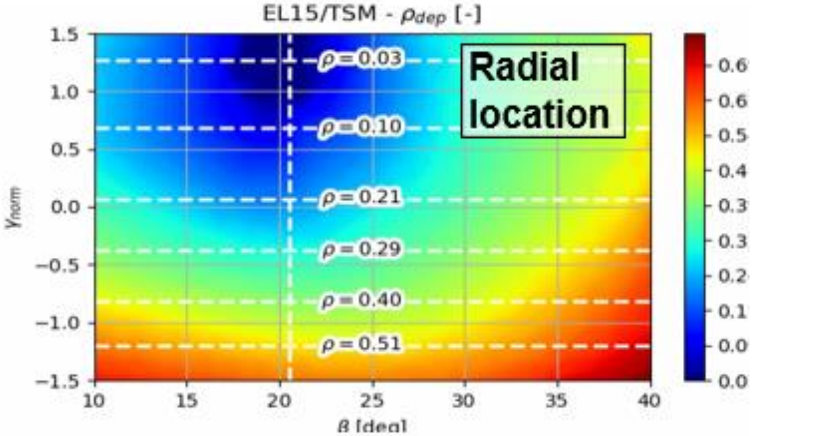
Absorbed power



Current Drive

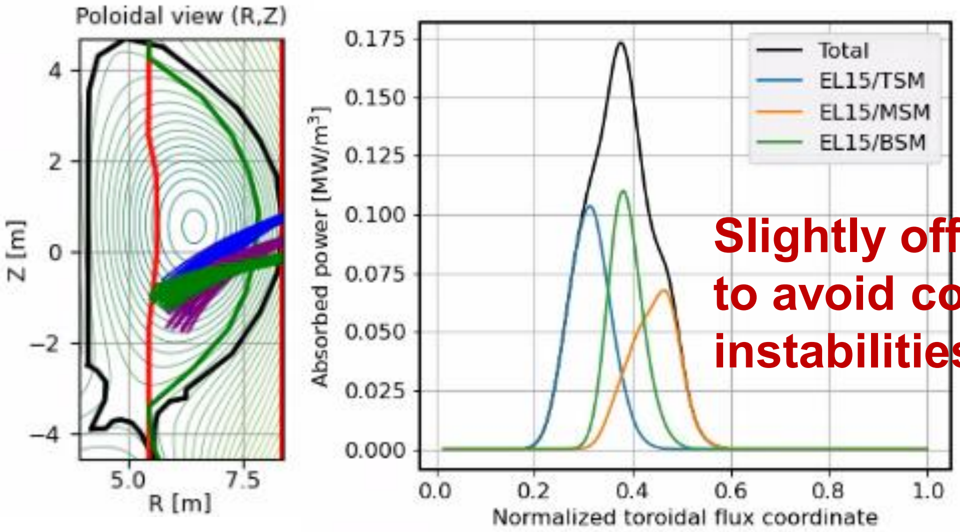


Radial location



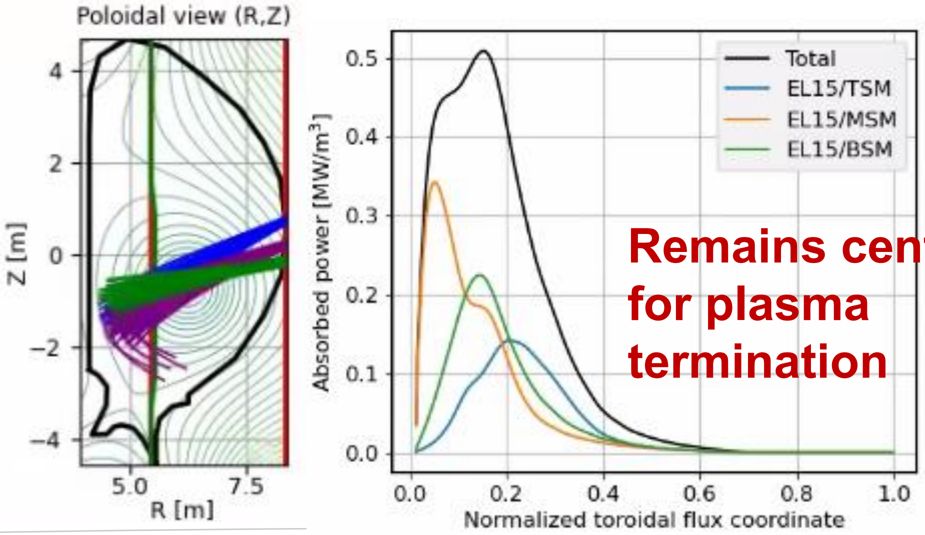
→ Results: toroidal angle $\beta = 20 - 22°$ and poloidal steering such that $\rho = [0.2\ - 0.5]$

Flattop ITER DT baseline 15 MA



Slightly off-axis to avoid core instabilities

End of ramp-down ITER DT baseline 15 MA



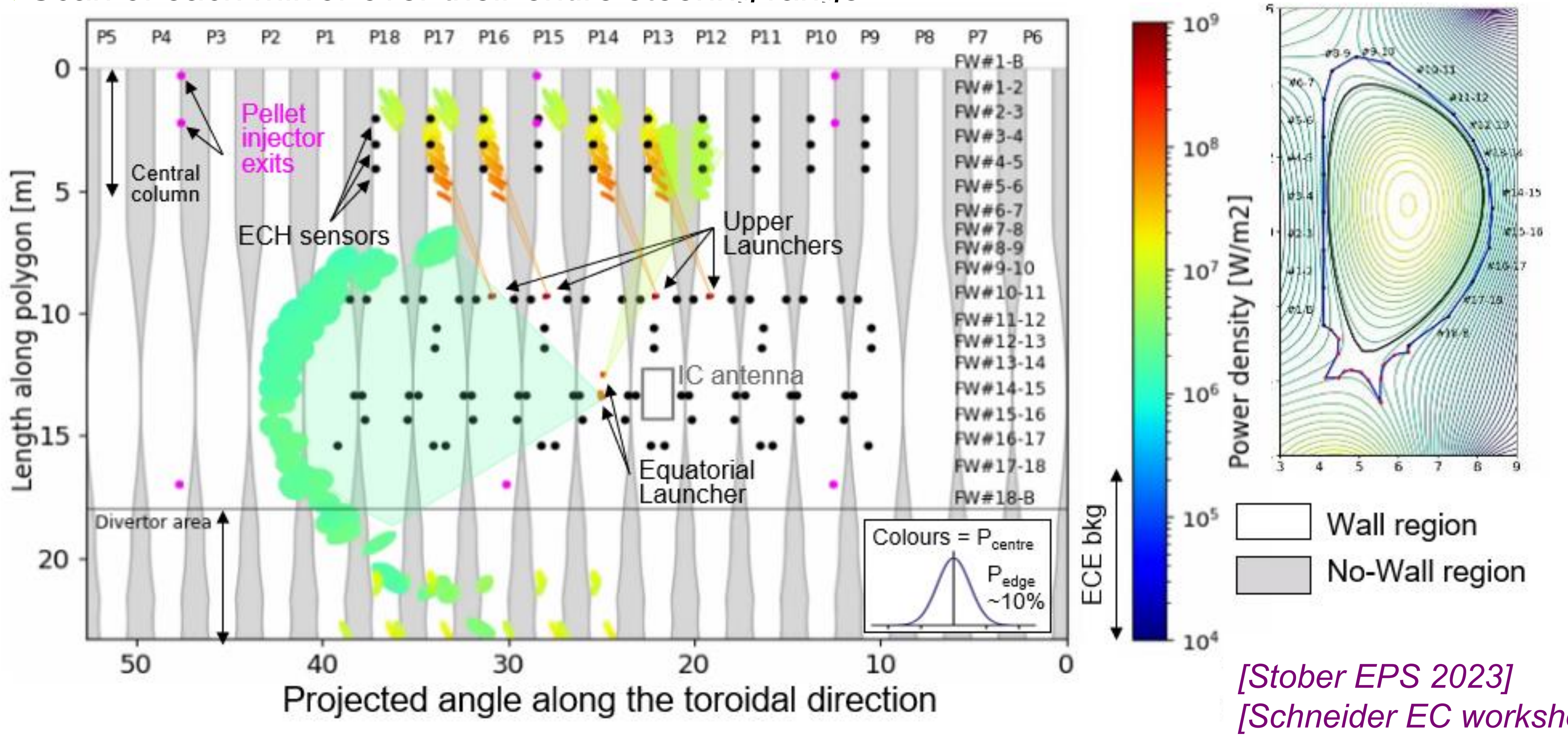
Remains central for plasma termination

Applicable EC power for each radial location inside the plasma

<table>
<tr><th></th><th>P (ρ ~ 0,0.2)</th><th>P (ρ ~ 0.2,0.4)</th><th>P (ρ ~ 0.4,0.6)</th><th>P (ρ ~ 0.6,0.8)</th></tr>
<tr><td rowspan="3">Max applicable with 2EL and 3 UL or 4 UL</td><td>20 MW
• 3 mirrors, EL14 (20 MW)</td><td>40 MW
• 3 mirrors, EL14 (20 MW)
• 3 mirrors, EL15 (20 MW)</td><td>23.33 MW
26.67 MW
• 2 mirrors, EL14 (13.33 MW)
• 1 mirror, 3 UL (10 MW)
• 1 mirror, 4 UL (13.33 MW)</td><td>10 MW
13.33 MW
• 1 mirror, 3 UL (10 MW)
• 1 mirror, 4 UL (13.33 MW)</td></tr>
<tr><td colspan="2">40 MW
• 3 mirrors, EL14 (20 MW)
• 3 mirrors, EL15 (20 MW)</td><td colspan="2">33.33 MW
40 MW
• 2 mirrors, EL14 (13.33 MW)
• 2 mirrors, 3 UL (20 MW)
• 2 mirrors, 4 UL (26.67 MW)</td></tr>
<tr><td colspan="4">60 MW
66.67 MW
• 3 mirrors, EL14 (20 MW)
• 3 mirrors, EL15 (20 MW)
• 2 mirrors, 3 UL (20 MW)
• 2 mirrors, 4 UL (26.67 MW)</td></tr>
</table>

EC heat loads in vacuum

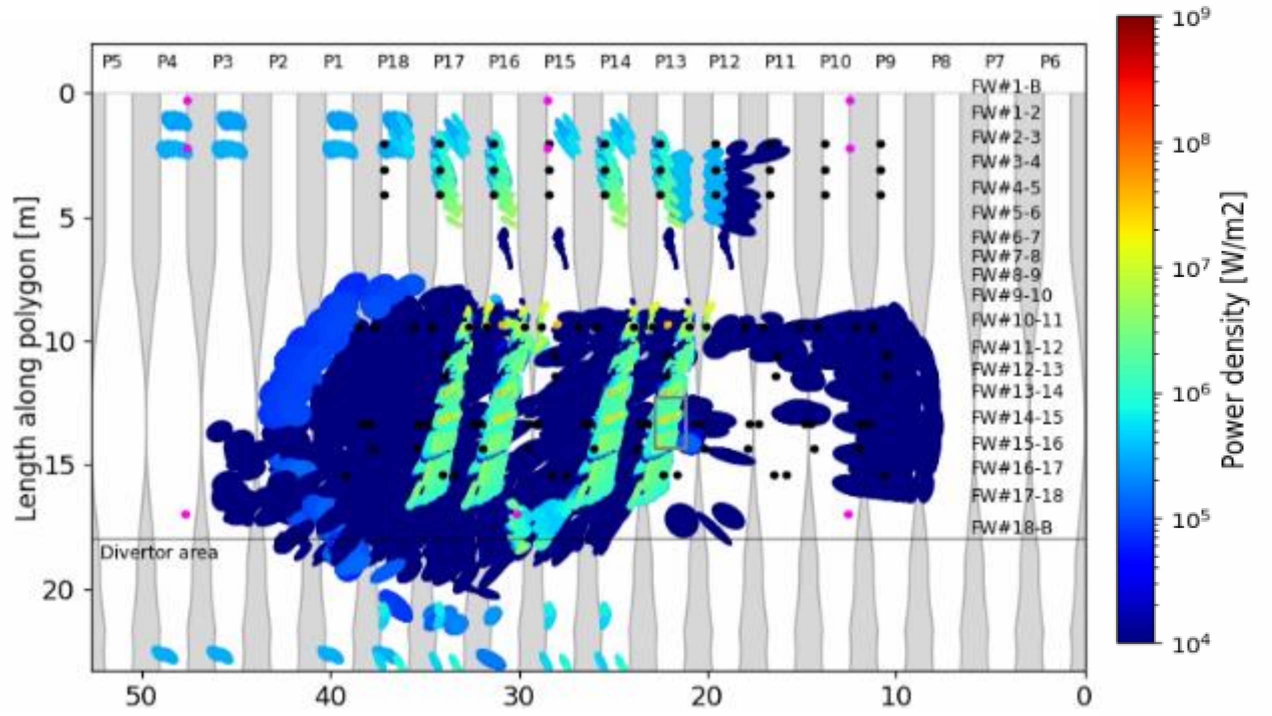
- All 56 beams with a power of 1 MW each → to be rescaled to chosen operational conditions
- Scan of each mirror over their entire steering range



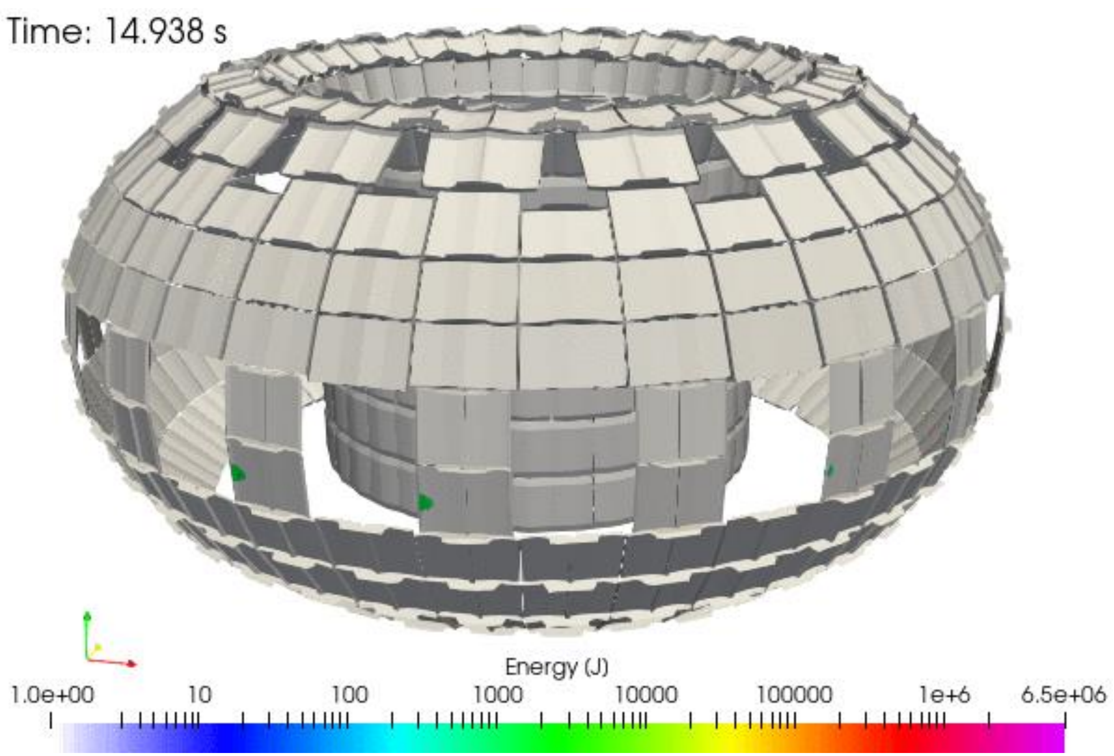
[Stober EPS 2023]
[Schneider EC workshop 2024]

EC heat loads from 5% X1 cross-polarization at 5.3T

- EL + UL all together with steering angle scan, in time range [15-115] s - 5% X1 assumed



Projected angle along the toroidal direction



- EC heat loads up to ~20 MW/m^2 but mostly transient (early ramp-up) and from ULs which will be switched off during the ramp-up`(+ ~98% W reflection coefficient)

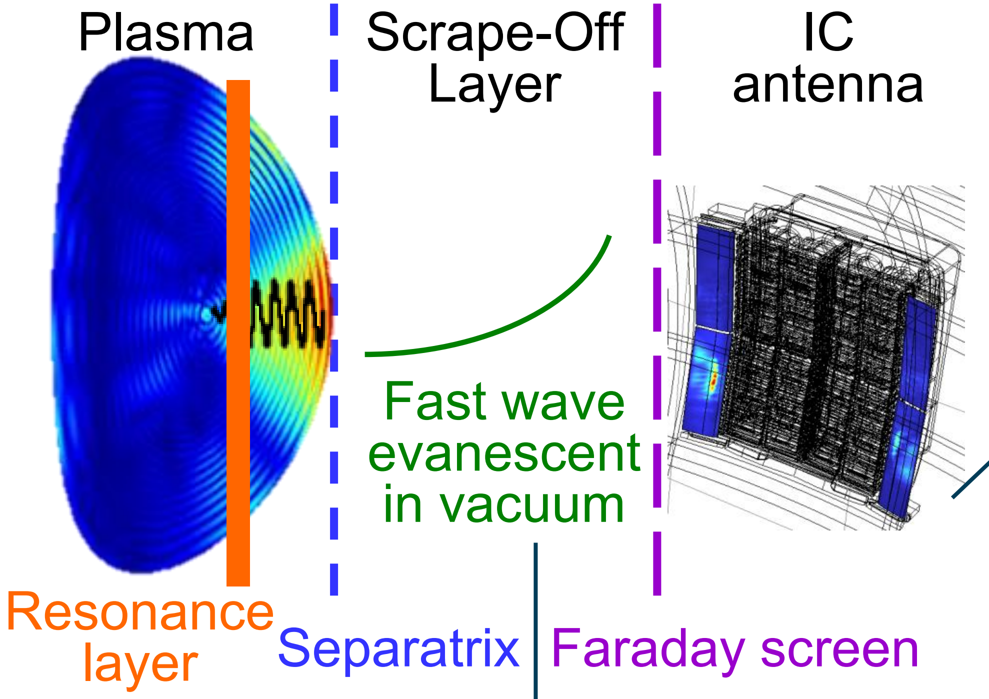
[Stober EPS 2023]
[Schneider EC workshop 2024]

- Reasonable heat loads from EL with appropriate steering

The ITER IC system

IC global principle, functionalities and operational constraints

- Also based on resonance wave absorption mechanisms.
- Heats specific ions according to schemes (minority/majority)
- $\omega \sim [40 - 55]\ MHz \rightarrow$ Fast wave evanescent at low density:



RF sheaths mitigated by proper phasing and voltage balance

Mitigated by local midplane gas puff

2 criteria for IC coupling:
- Sufficient density near antenna
- Moderate edge density gradients

IC functionalities
• Heating (ions/electrons) • Sawtooth control • Wall conditioning (ICWC) • Current Drive (small)
Operational constraints
B-field Minimum density Plasma composition

The IC system in the ITER new baseline

- Non-optimized ICRF antennas operation is challenging with W wall

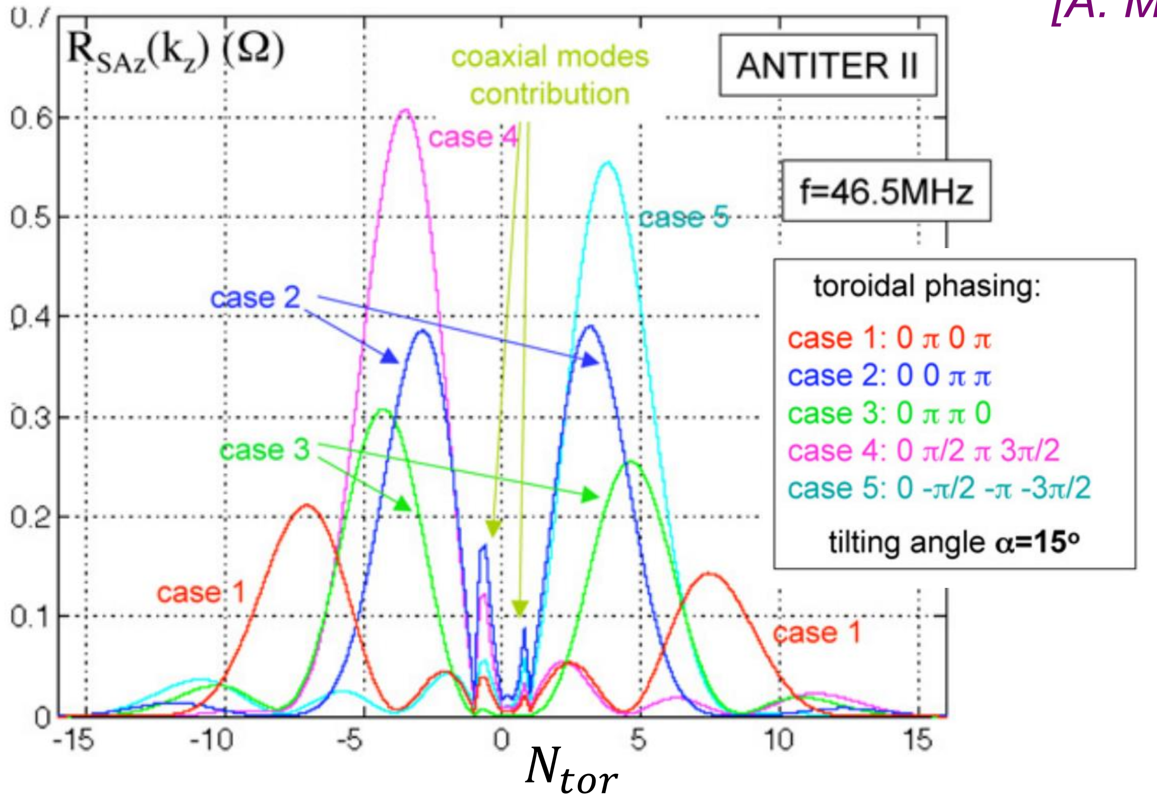
- AUG demonstrated that RF impurity generation can be much reduced with optimized antenna array
  → Same optimization strategy for ITER

- Modelling shows high power ICH (up to 20 MW per 1 antenna) can be operated with low ICRF-specific W-sputtering (below 1-10% of the total W sputtering) → To be demonstrated experimentally

- 10 MW ICH in SRO for ICH coupling tests and ICWC (needs 5 MW installed power), later used for T removal in DT-1)

→ Upgradable to 20 MW ICH in DT-1

The phasing of ITER IC antenna



[A. Messiaen 2010 Nucl. Fusion (2010)]

- **Heating phasings**:
  - $[0.\pi.0.\pi]$ = dipole = standard – high N_{tor}, high $k_{\parallel}$, lower antenna coupling (but better absorption)
  - $[0.0.\pi.\pi]$ = super-dipole
  - $[0.\pi.\pi.0]$ = reference phasing in ITER
- **CD phasings**: (e.g. for ST control)
  - $\pm[0.\pi/2.\pi.3\pi/2]$, i.e. $+90°$
  - $\pm[0.-\pi/2.-\pi.-3\pi/2]$, i.e. $-90°$

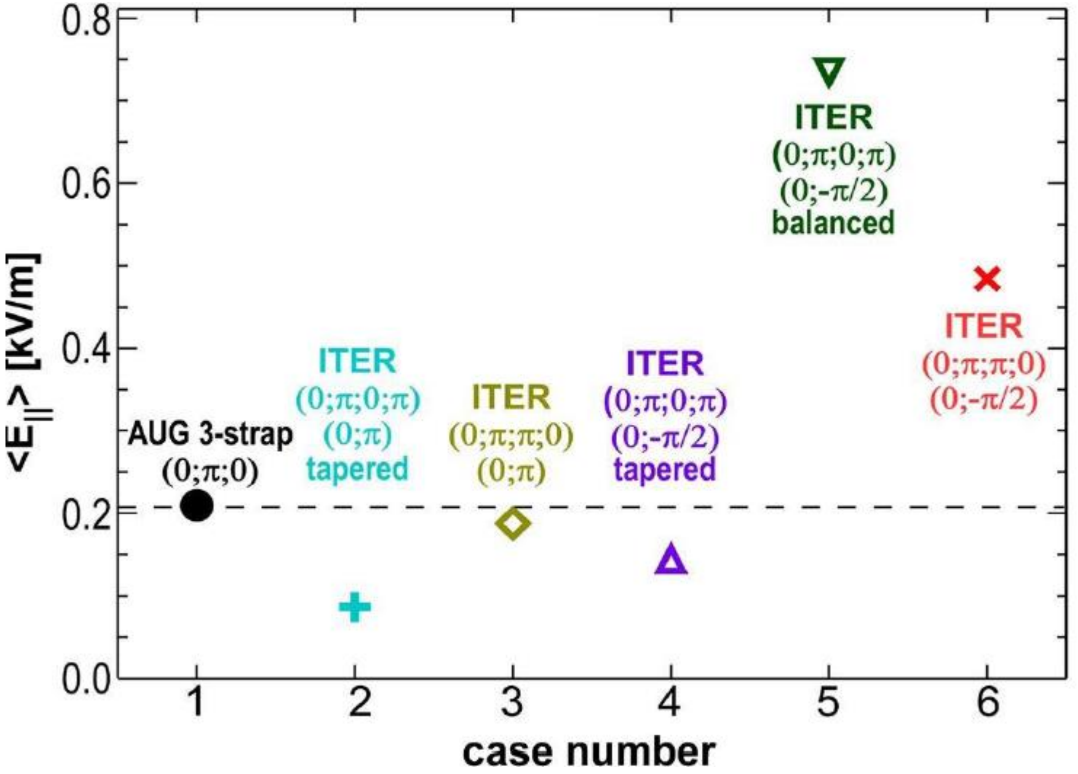
(Poloidal phasings: top/bottom = [0,-$\pi/2$] fixed)

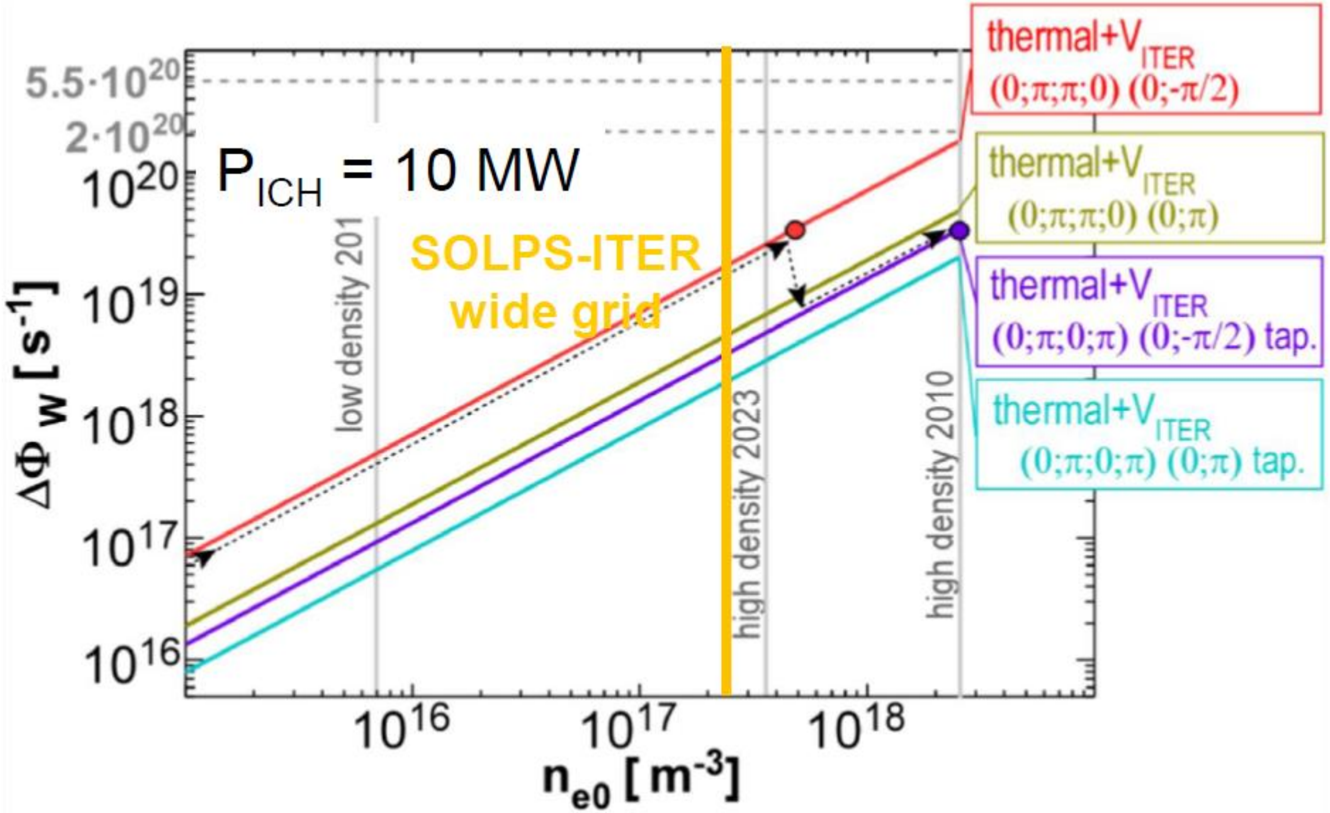
- High N_{tor}, high $k_{\parallel}$:
  → **Higher electron damping**, lower v_{φ} ($v_{\varphi} = \omega/k_{\parallel}$) → more electrons resonant with the wave
  → **Lower coupling** but **better absorption**
- Low N_{tor}, low $k_{\parallel}$:
  → **Higher coupling** but **potentially lower absorption**

Phasing of IC antenna and W source

- ICH antenna flexibility used to achieve ASDEX-U 3-strap like parallel E-field

V. Bobkov NME 2024 & L. Colas NME 2025





→ Adapted phasing between the 4 straps leads to reduced RF sheath effects

→ Predictions indicate that ICH-induced W source is << W wall source (~10^{21} W/s)

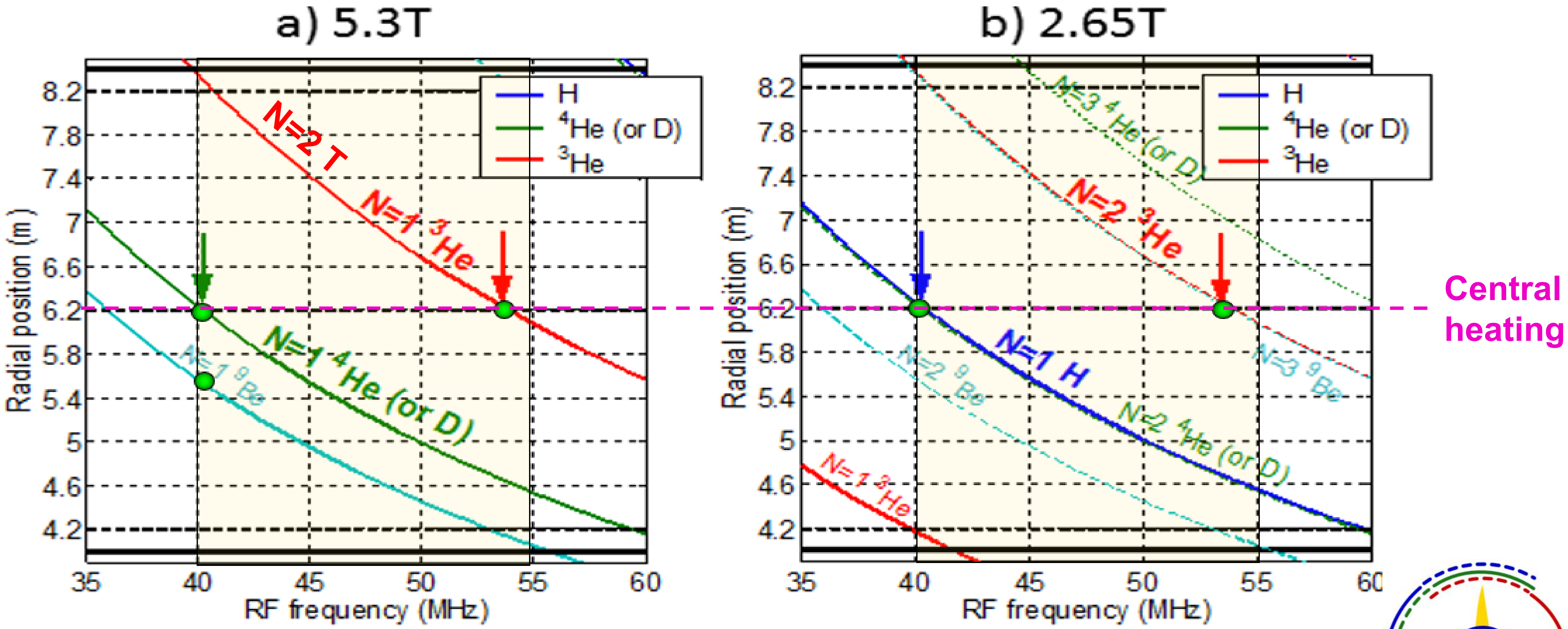
→ To be experimentally confirmed in SRO to decide for power upgrade from 10 MW to 20 MW for DT

Operability of the ITER IC system

- The Plasma Control System will control in real-time the IC power, wave frequency and phase.

Functionality	Description
Latency	≤ 2ms
Power range	Deliver power from 0.02 to 10-20 MW for the IC antenna
Power dynamics	Full power slope from 0 to 10-20 MW (no modulation): • in 200 ms at the beginning of each pulse • in 10 ms for the rest of the discharge
Power modulation	• Modulation from $P_{max}/2$ to P_{max} shall be up to 50 Hz. • Modulation from 0 to full power shall be up to 2.5 Hz.
Wave frequency	Frequency range [40 – 55 MHz] (changed between pulses).
Frequency dynamics	Real-time frequency change from ±1 kHz to ±1 MHz in 10 ms but a few secs for the Matching System to adjust while the coupled IC power drops.
Antenna phasing	Poloidal straps with real time toroidal phase changes from 0 to 2π (to change the radiated power spectrum) in 10 ms but few secs for MS
Coupling measurement	Real-time estimate of the antenna coupling resistance and RF losses (in a few ms) such that the PCS moves the plasma or injects gas to improve it (~ 1 s response)

IC heating schemes for ITER (1/2)

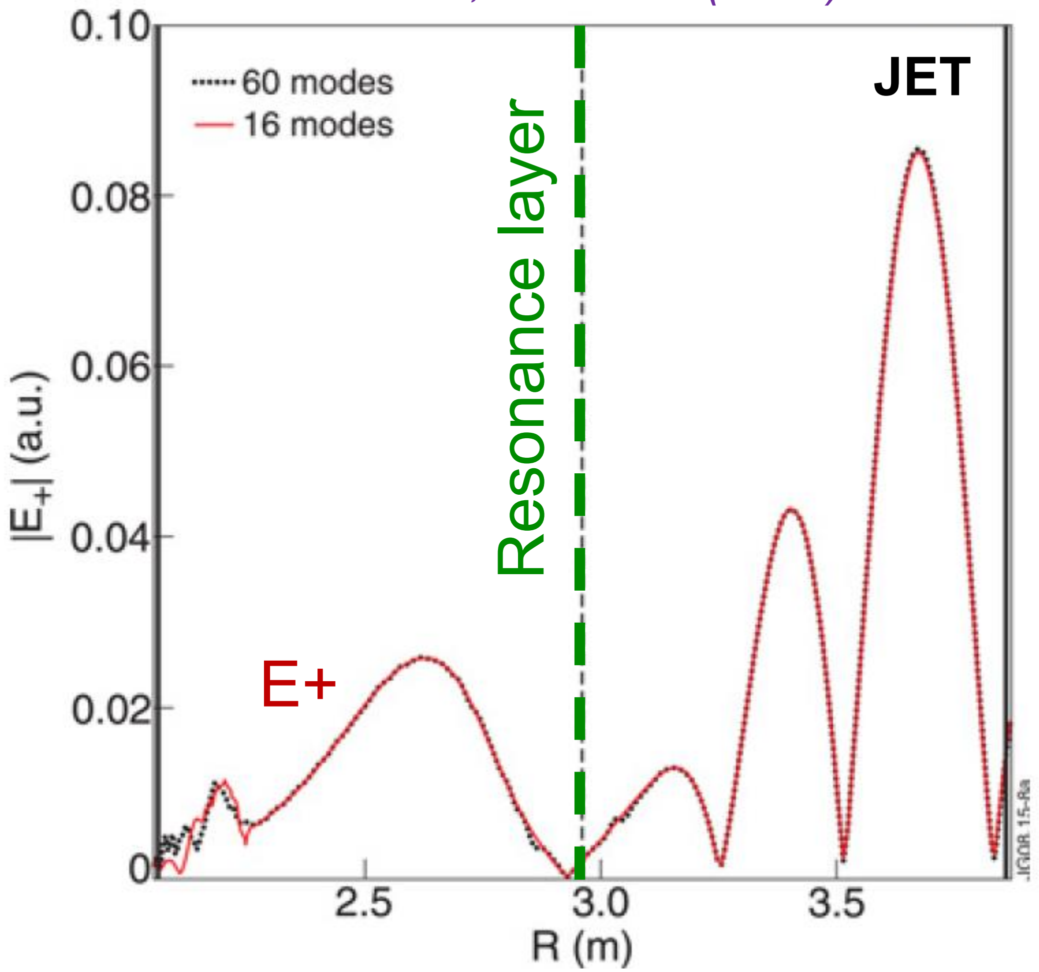


The chosen scheme depends whether the heated ion is majority or minority.... due to the E+ screening effect: fundamental heating of majority ion is not possible

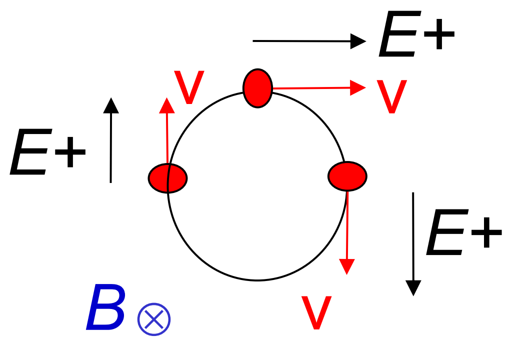
E+ screening effect for fundamental heating of majority ion

- Fundamental heating of majority ion not possible due to E+ screening effect:

E. Lerche et al, PPCF 51 (2009) 044006



Resonance condition: $\omega - n\Omega_c - \vec{k}\cdot\vec{v} = 0$



- Majority ion: strong collective plasma response which induces RF current that screens E+
- Minority ion: only a small fraction of ions is resonant → collective response not strong enough to screen E+

E+ screening effect translated into equations

- RF electric field: $E = E_+ e_+ + E_- e_-$

- How much current does the plasma produce?
  → Ions respond to the E-field by creating a current (Ohm's law): $J_+ = \sigma_+ E_+$

- At cyclotron resonance: $\omega = \omega_c$
  → Majority ions respond strongly → Conductivity becomes huge: $\sigma_+ \rightarrow \infty$

- How much polarization does the plasma produce?
  → Ampere's law:

$$\nabla \times B = \underbrace{\mu_0 J}_{\text{Plasma response}} + \underbrace{\mu_0 \varepsilon_0 \frac{\partial E}{\partial t}}_{\text{Vacuum response}} = \mu_0 \sigma_+ E_+ + \mu_0 \varepsilon_0 \frac{\partial E}{\partial t} \sim \mu_0 \sigma_+ E_+ \Rightarrow E_+ \sim \frac{1}{\sigma_+} \rightarrow 0$$

At resonance $\sigma_+ \gg \varepsilon_0 \omega$

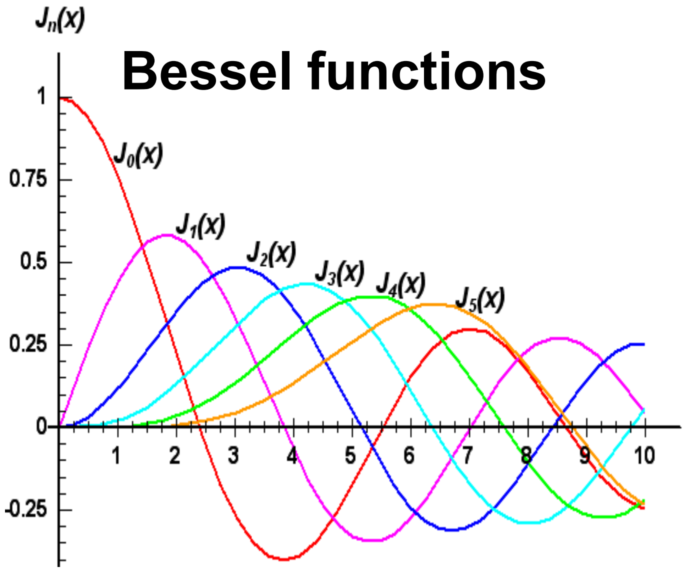
→ A huge plasma current creates its own E-field opposed to the applied RF field
→ Effective E+ polarized field tends to zero.

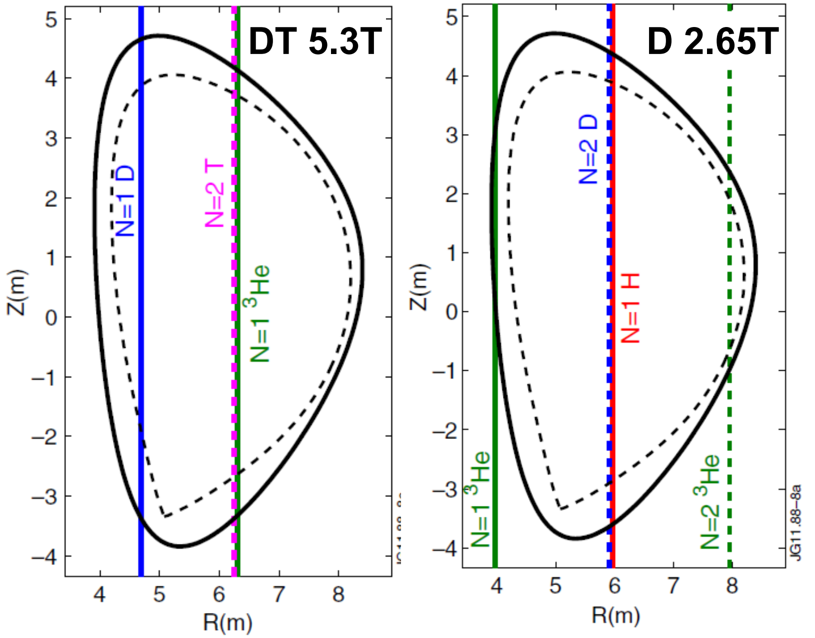
Choice of ICH schemes for the two ITER B-fields

[Lerche, PPCF (2012)]

- Absorption strength (diffusive in velocity space):

$$D_{RF} \sim \left| E_+ J_{n-1}\left(\frac{k_\perp v_\perp}{\Omega_c}\right) + E_- J_{n+1}\left(\frac{k_\perp v_\perp}{\Omega_c}\right) \right|^2$$

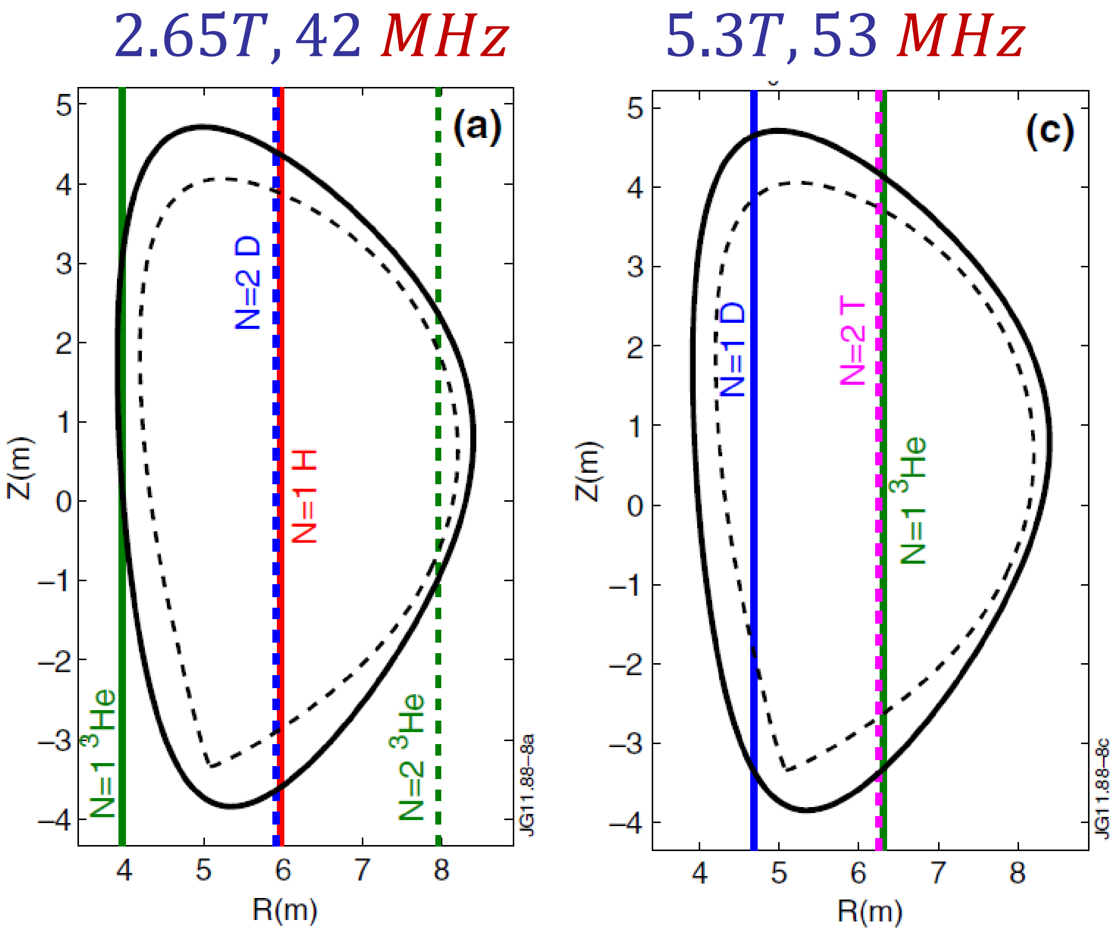




- Minority heating scheme:
  - Fundamental resonance ($n = 1$):
    → $J_0\left(\frac{k_\perp v_\perp}{\Omega_c}\right)$ dependence
    → Mostly heat thermal ions → Scheme: Fundamental He^3 minority (<2%) at 5.3T
    Fundamental H minority at 2.65T

- Majority heating scheme: (screening effect at n=1)
  - Harmonic resonances ($n > 1$), FLR effect:
    → $J_{1,2,\ldots}\left(\frac{k_\perp v_\perp}{\Omega_c}\right)$ dependence
    → Mostly heat hot bulk ions → Scheme: 2nd harmonic T majority at 5.3T
    2nd harmonic D majority at 2.65T

IC heating schemes for ITER (2/2)



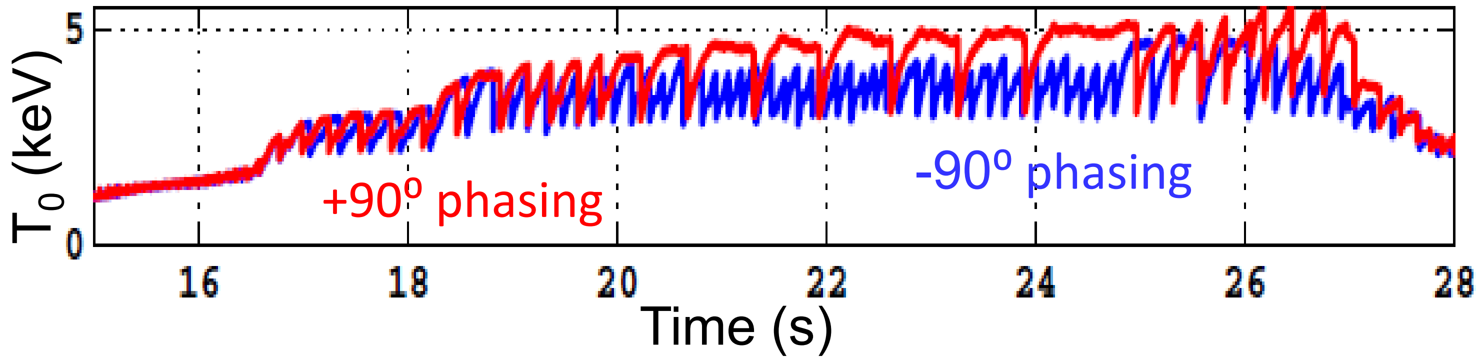
B-field	Main plasma species	SRO	DT-1 and DT-2
2.65T	D	H minority (N=1) or D majority (N=2), at **42 MHz**	
	H	No efficient scheme	
	DT	ø	H minority (N=1) or D majority (N=2), at **42 MHz**
5.3T	D or H	^{3}He minority (N=1) at 53 MHz (1-5%)	
	DT	ø	^{3}He minority (N=1) followed by T majority (N=2) both at **53 MHz**

[Lerche PPFC 2012]

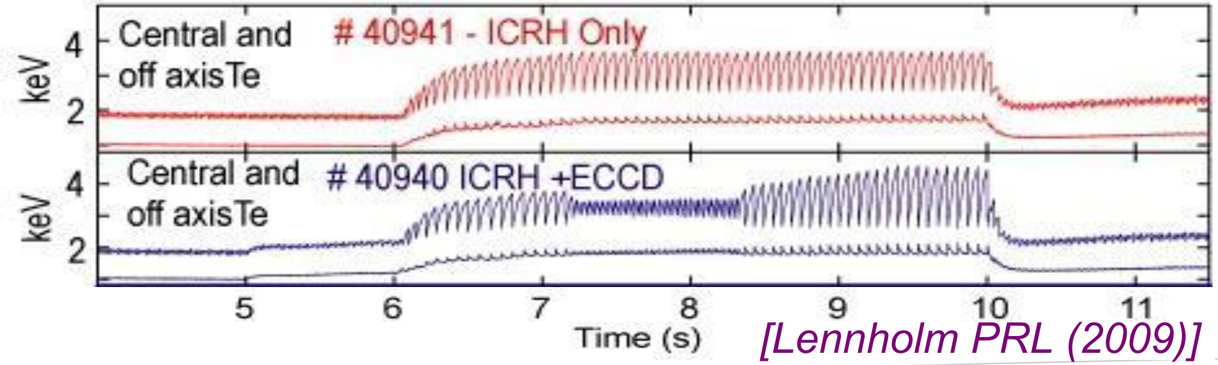
Sawtooth control

- Sawtooth crashes when safety factor $q < 1 \rightarrow$ Periodic variation of density & temperature
- Large sawteeth may trigger Neoclassical Tearing Modes (NTM)
- Sawtooth control tools available:
  - NBI: on-axis is stabilizing $\rightarrow$ longer period
  - ECCD: local modification of magnetic shear
  - ICH: kinetic (de-)stabilization of the kind mode: stabilizing (+90º) or destabilizing (-90º)

Sawtooth control by ICH at JET *[Chapman, NF (2010)]*



Sawtooth destabilization by ECCD in Tore Supra

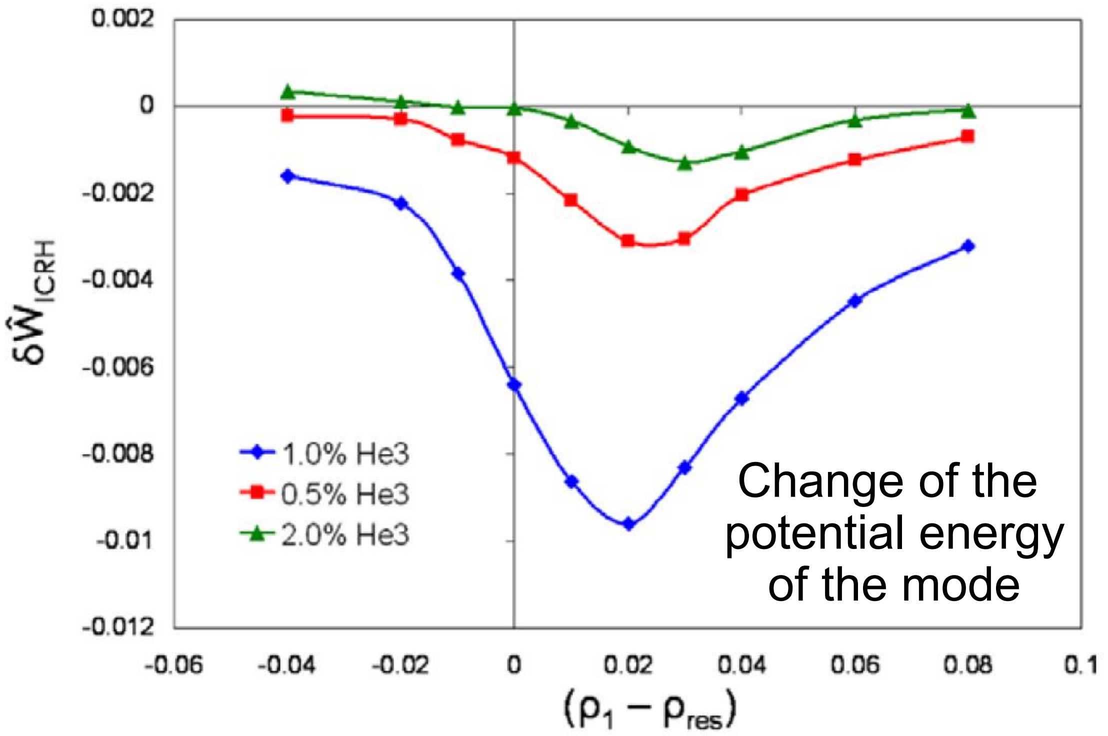


[Lennholm PRL (2009)]

$\rightarrow$ Needs real time control of ECCD location ($0.15 < r/a < 0.88$) and ICH frequency ($\pm 1\ MHz$).

Sawtooth control on ITER (1/2)

- Purpose: to reduce the sawtooth period to prevent NTMs.



Off-axis low field side He^3 minority heating; 20 MW at $52\ MHz$, $-90º$ phasing (SELFO+HAGIS)

Region in which the sawteeth are sensitive to the destabilizing influence of ICH ions ~ 5 cm.

→ Real-time control of IC frequency is essential.

→ Accurate reconstruction of the q-profile is essential.

Note: ±1 MHz sweeping moves IC resonance to ~24cm.

[Chapman NF 2013]

- Too low He^3 concentration → Too low absorbed power, broader distribution
- Too high He^3 concentration → Too low energy in fast ion tail, higher dilution

→ Strong sensitivity to minority concentration

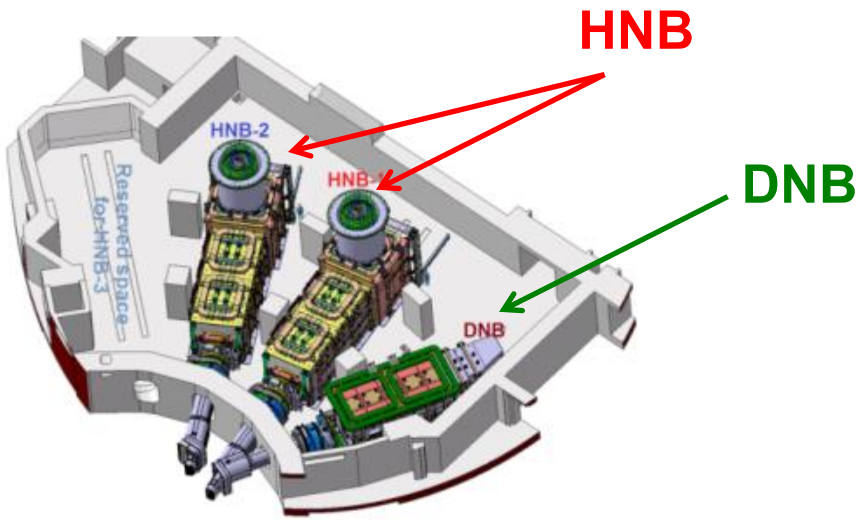
→ Accurate control of minority concentration is essential.

Sawtooth control on ITER (2/2)

- 2 possible opposite approaches:
  - Stabilization of the sawteeth + pre-emptive NTM control via ECCD near $q = 2$
    → Early heating during ramp-up (with reduced auxiliary heating e.g. NBI+ICH)
    → Early ignition with further stabilization due to alpha particles
    → ECCD applied just before each crash to tackle NTM when it is still small
    → Maximization of fusion gain due to reduced auxiliary heating (paced ECCD)
  - Destabilization of the sawteeth:
    → Better avoidance of potentially disruptive NTMs
    → Frequent expulsion of on-axis accumulated impurities
    → Helium ash removal

The ITER NBI system

Overview of NBI requirements



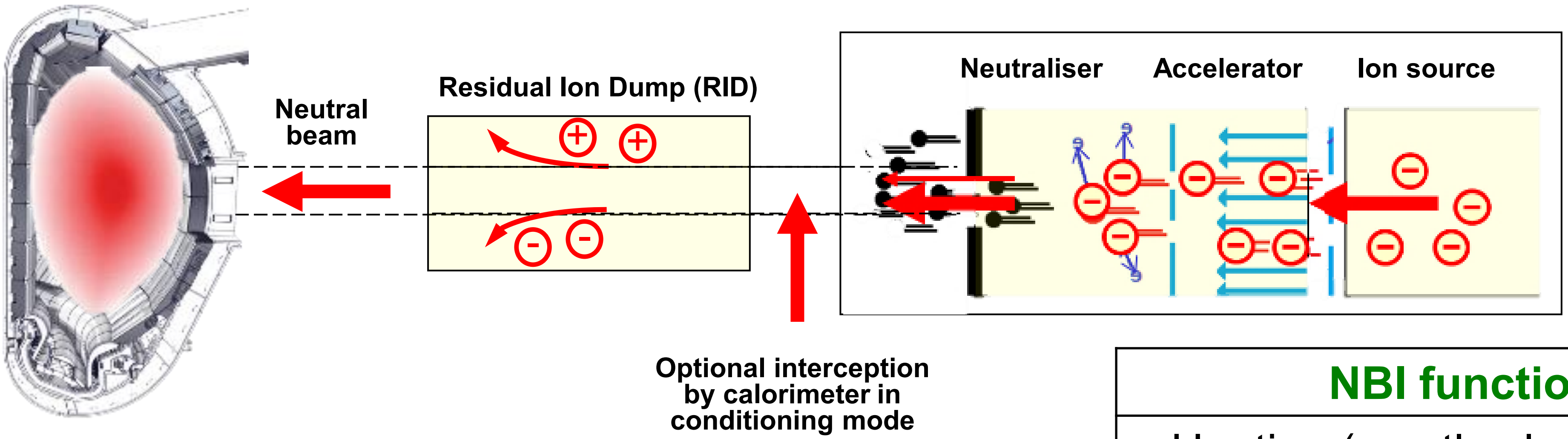
Beams at ITER

- **Heating beam (HNB)**
  - Heating
  - Current drive
  - Plasma rotation
- **Diagnostic beam (DNB)**
  - He ash, Ti, density of light impurities, rotation

Parameter	Unit	HNB	DNB
Injected neutral beam power	MW	16.5 per beamline	1.4 (unmodulated)
Beam energy / species (H phase)	MeV	0.87 / H	0.1 / H
Beam energy / species (D/DT phase)	MeV	1 / D	0.1 / H
Accelerated current (H phase)	A	46 / H	60 / H
Accelerated current (D/DT phase)	A	40 / D	60 / H
Beamlet divergence	mrad	< 7	< 7
Beam-on time (H, D/DT phase)	s	1000 / 3600	3600 (modulated)
Injection geometry	-	Tangential	Normal
NBI axis vertical inclination angle	mrad	–49.7 ± 10	15 (fixed)

The NBI system in the ITER new baseline

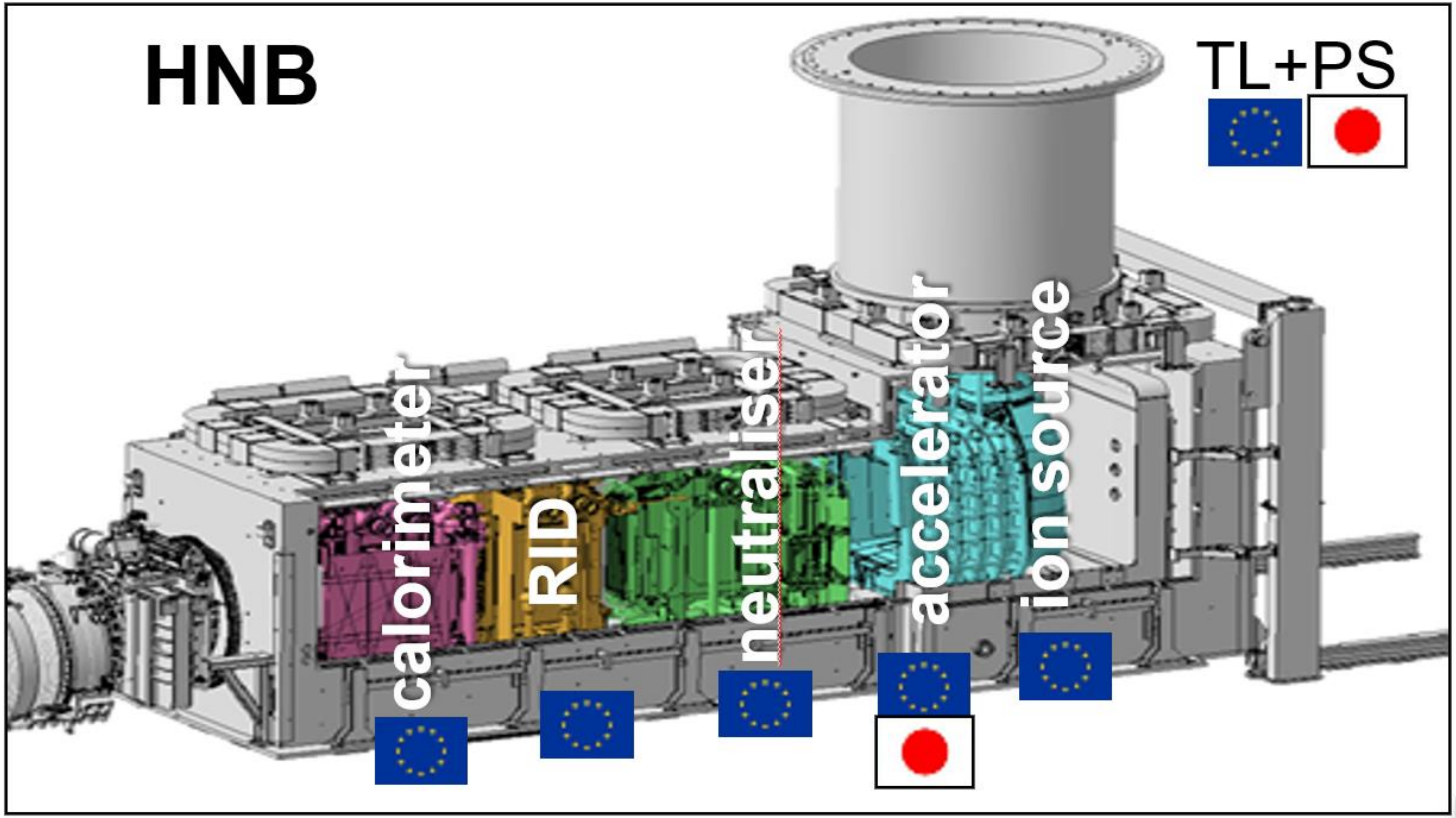
ITER has a large, high density, hot plasma.
→ Needs high energy neutral beams to penetrate up to the core
→ Achievable by accelerating negative ions up to 1 MeV.



- The source of negative ions is generated
- Ions are accelerated on a grid
- Ions are neutralized (55% power reduction)
- Residual ions are filtered out → re-ionization losses
- Neutral beam injected through a beamline

NBI functionalities
• Heating (mostly electrons) • Main source of current drive • Torque source (rotation) • Current profile shaping • Current and q-profile measurements
Operational constraints
Minimum density (shinethrough)

Detailed of an Neutral Beam Injector

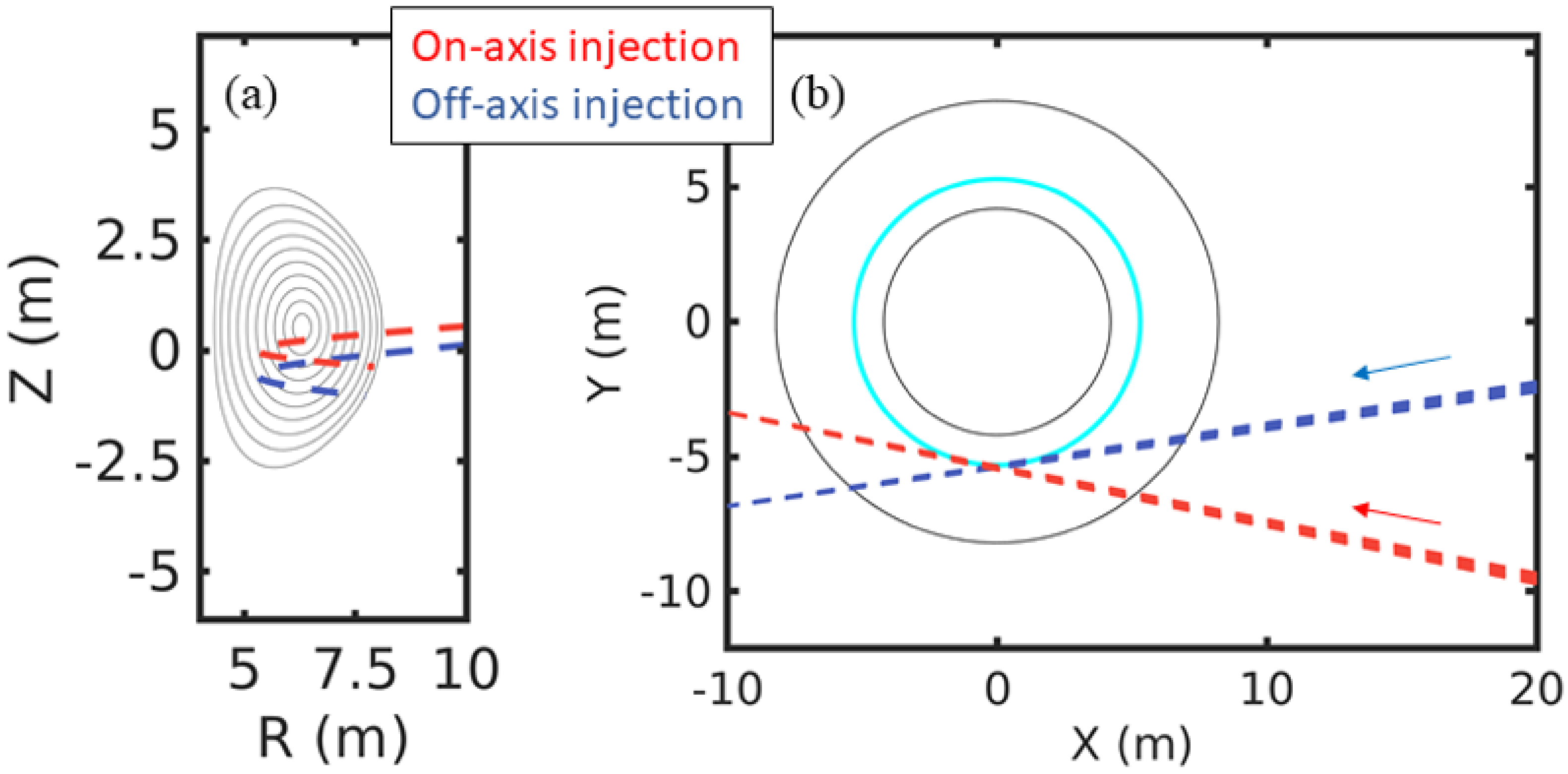


Operability of the ITER NBI system

- The Plasma Control System will control in real-time the NBI power and energy.
- *Beam perveance matching criteria: unique relationship between beam energy (acceleration voltage) and power: $P_{NBI} \sim E_{NBI}{}^{2.5}$

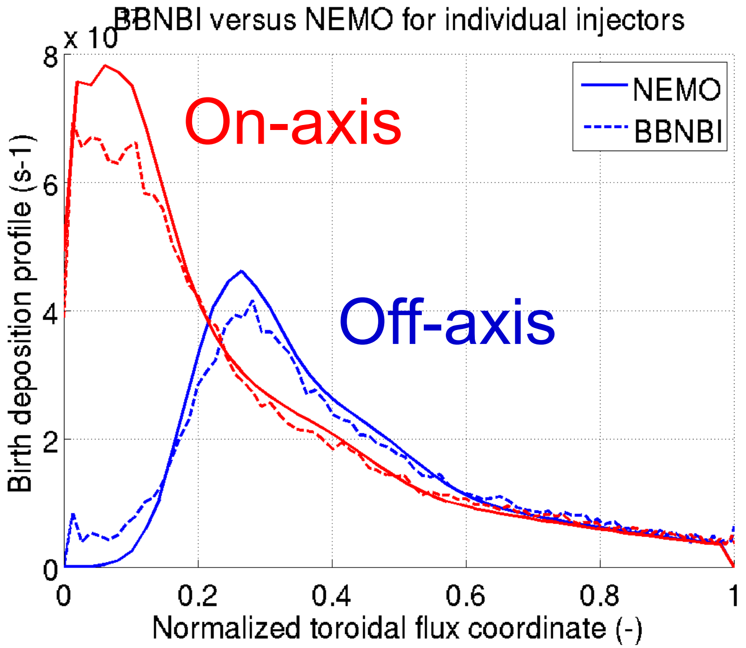
Functionality	Description
Latency	≤ 1 ms
Injected species	H (up to 870 keV) or D (up to 1 MeV)
Injected energy*	Up to 870 keV for H beams, up to 1 MeV for D beams
Energy variation	From 500 keV (for good beam divergence) to nominal energy with minimum 1keV steps
Injected power*	Up to 16.5 MW to the vessel per injector (controlled independently)
Power modulation	• ±25% power variations with max frequency of 0.3 Hz • Full power modulation in frequency range [5-7] Hz (limited by pulsed power network) • Dwell time to turn beam power on and off again between 10 ms and 100 ms
Power dynamics	• Minimum time of 80 ms to go from 0 to nominal power / energy • Minimum time of 1 ms to go from nominal power / energy to 0
Duty cycle	• For durations < 150 ms: max one pulse every 600s • For durations > 150 ms: 25% (ratio pulse burn duration / pulse repetition)
Beam geometry	On-axis or off-axis for each beam injector (can be modified only during maintenance)

NBI beam geometry: on-axis or off-axis



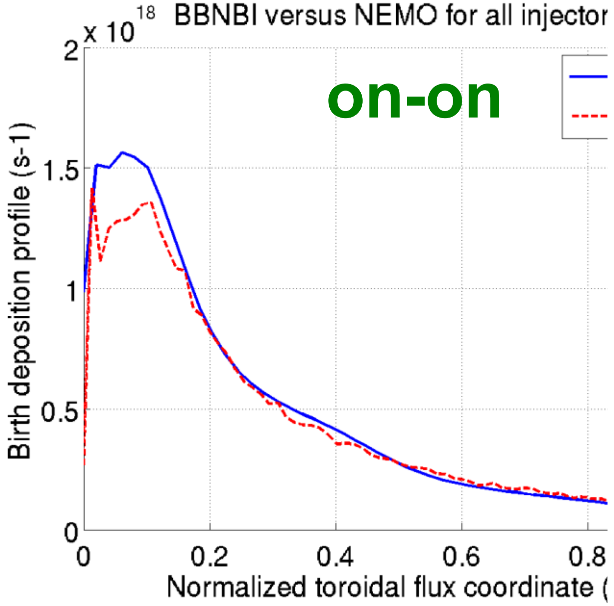
[Vincenzi NF 2024]

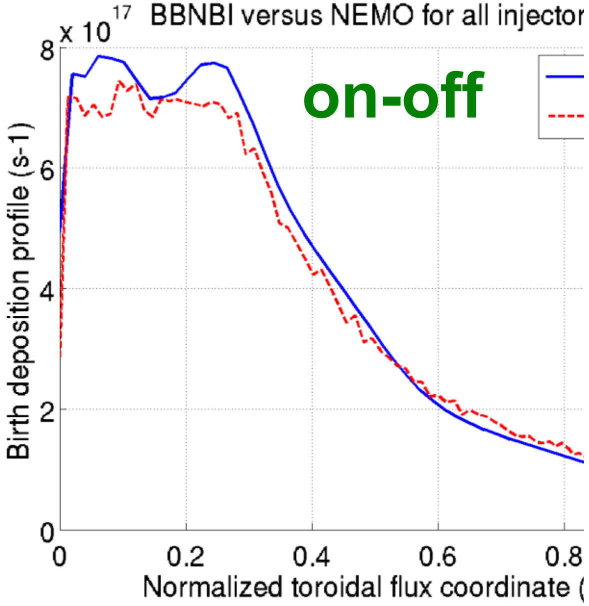
[Schneider ITPA-EP 2015]

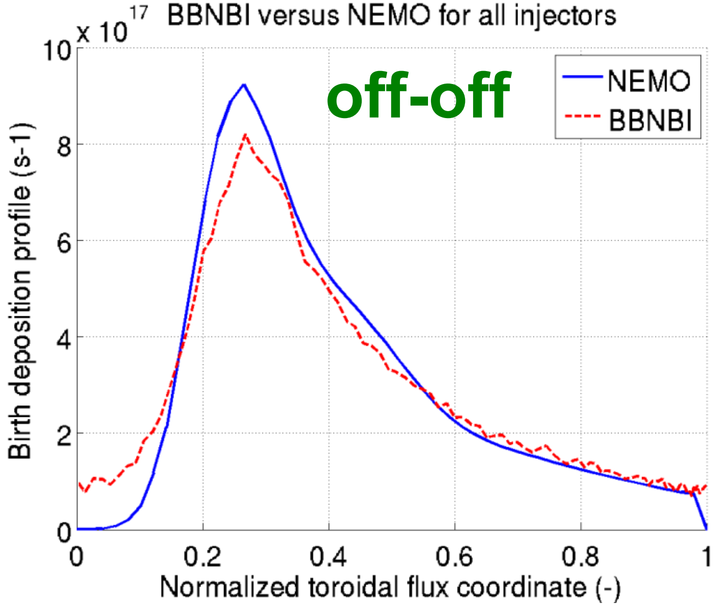


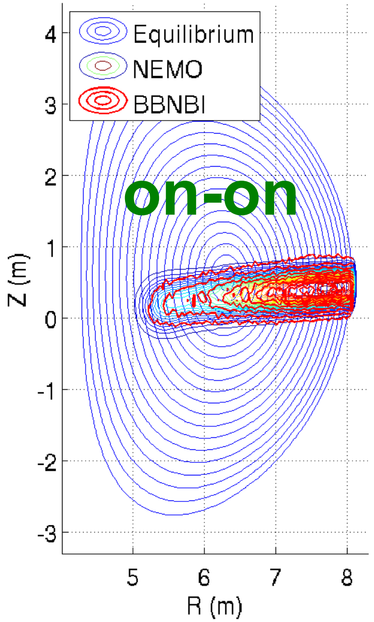
Each beam injector can be positioned on-axis or off-axis
(changes possible only during maintenance)

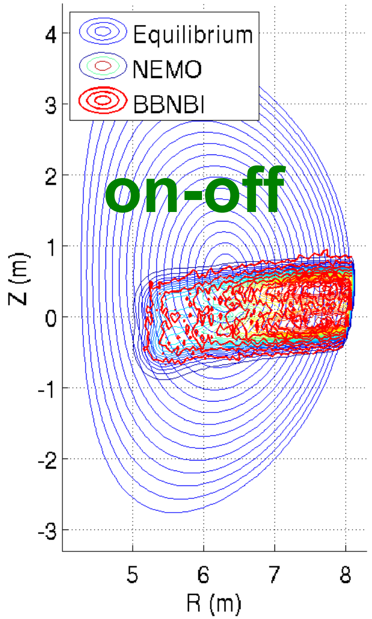
NBI deposition for the three possible configurations

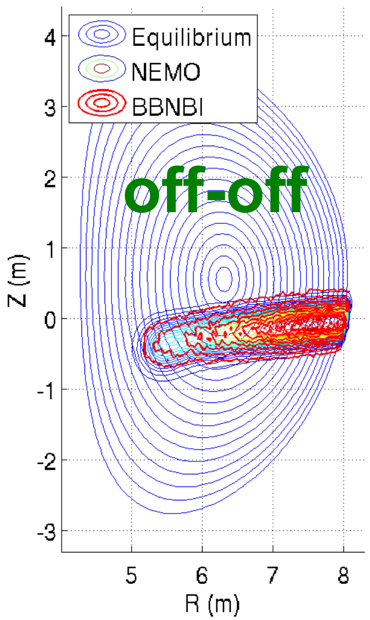








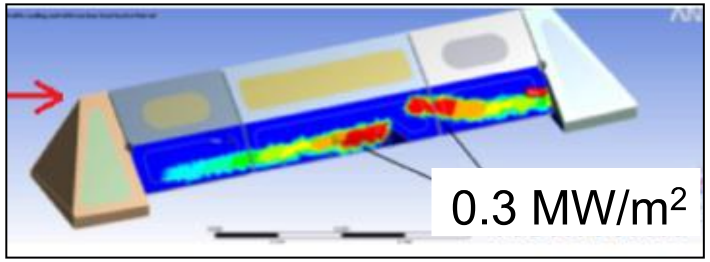




[Schneider ITPA-EP 2015]

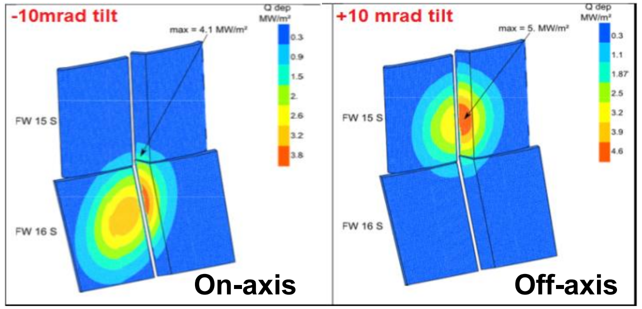
NBI shine-through losses

- Max power loads on shield block: **0.3 MW/m²** *[Singh NJP 2017]*
(→ 30000 cycles for 100s pulse)

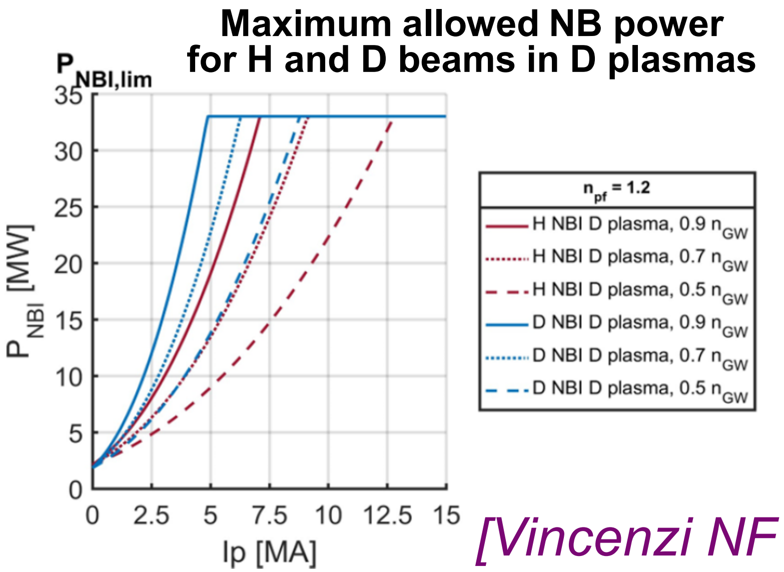


But angled surface → factor 2.65 on tolerable actual incident power: 0.3 x 2.65 = **0.8 MW/m²**.

- But tilted beams (±10mrad): power peak shifted
→ only tail on shield block→ factor 2.8.



→ Actual tolerable incident shinethrough power: 0.8 x 2.8 = **2.2 MW/m²**.



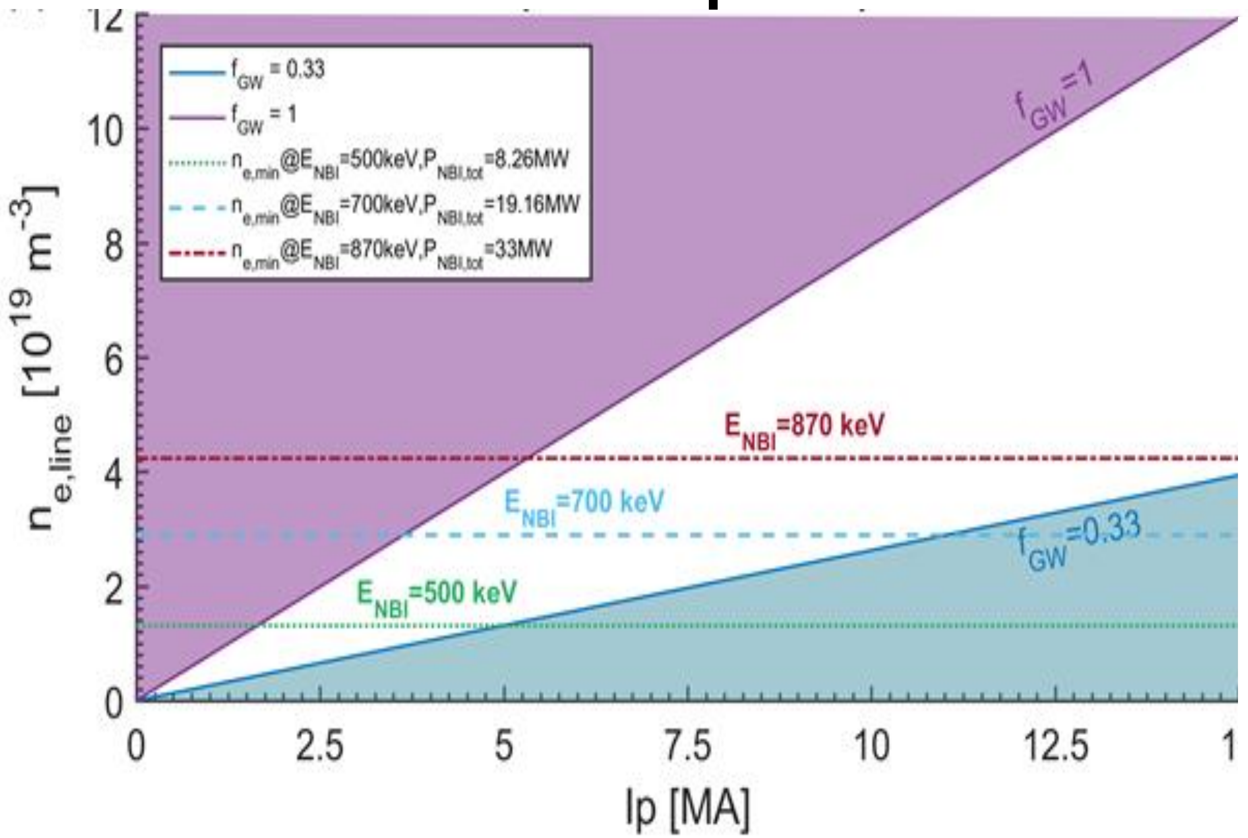
[Vincenzi NF 2024]

→ Evaluate optimal energy E_{NBI} for given density to remain below shine-through limit.

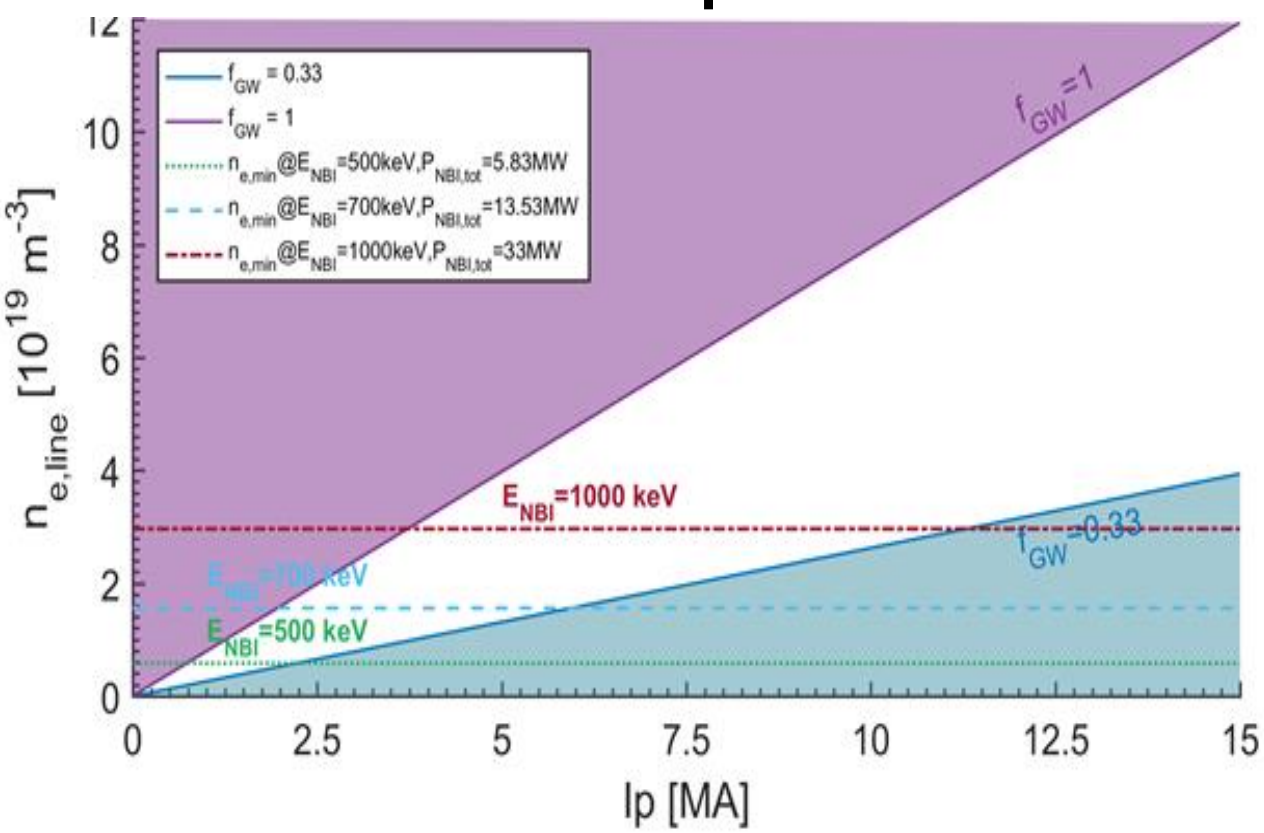
→ Associated constraint on power due to beam perveance criteria: $P_{NBI} \sim E_{NBI}^{2.5}$

Domain of operation of NBI system in plasma current and density

H beam in D plasma



D beam in D plasma

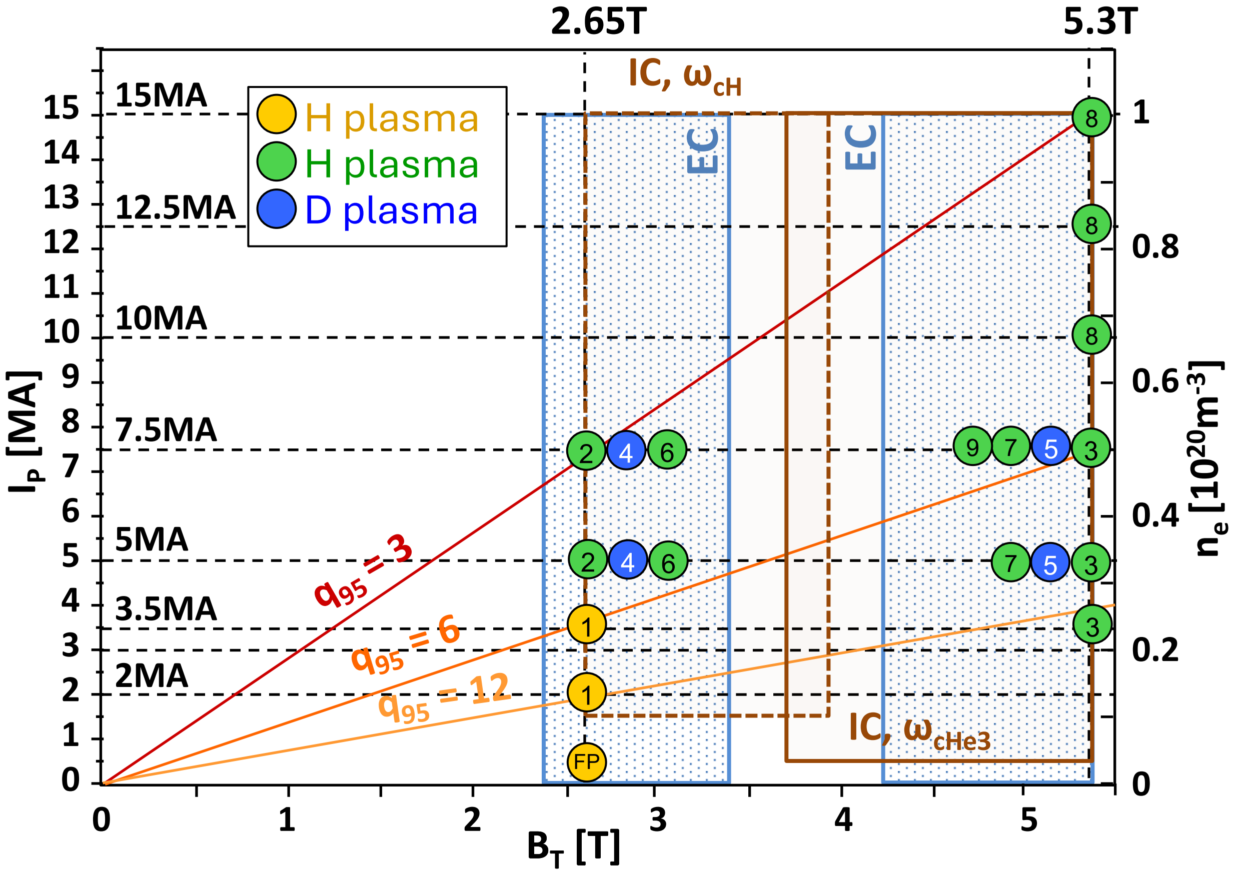


[Vincenzi NF 2024]

Operation of H&CD systems in the ITER Research Plan

H&CD and SRO research plan

Start of Research Operation (SRO)



[Loarte PPCF 2025]

~ 3 months
1 FP, lim., div. ≤3.5 MA
Hydrogen

~ 6.5 months
2 5-7.5MA / 2.65 T
Hydrogen

~ 0.75 month
3 3-7.5 MA / 5.3 T
Hydrogen (DMS)

~ 6 months
4 5-7.5 MA / 2.65 T
Deuterium, H-mode

~ 0.5 month
5 5-7.5 MA / 5.3 T
Deuterium, L-mode

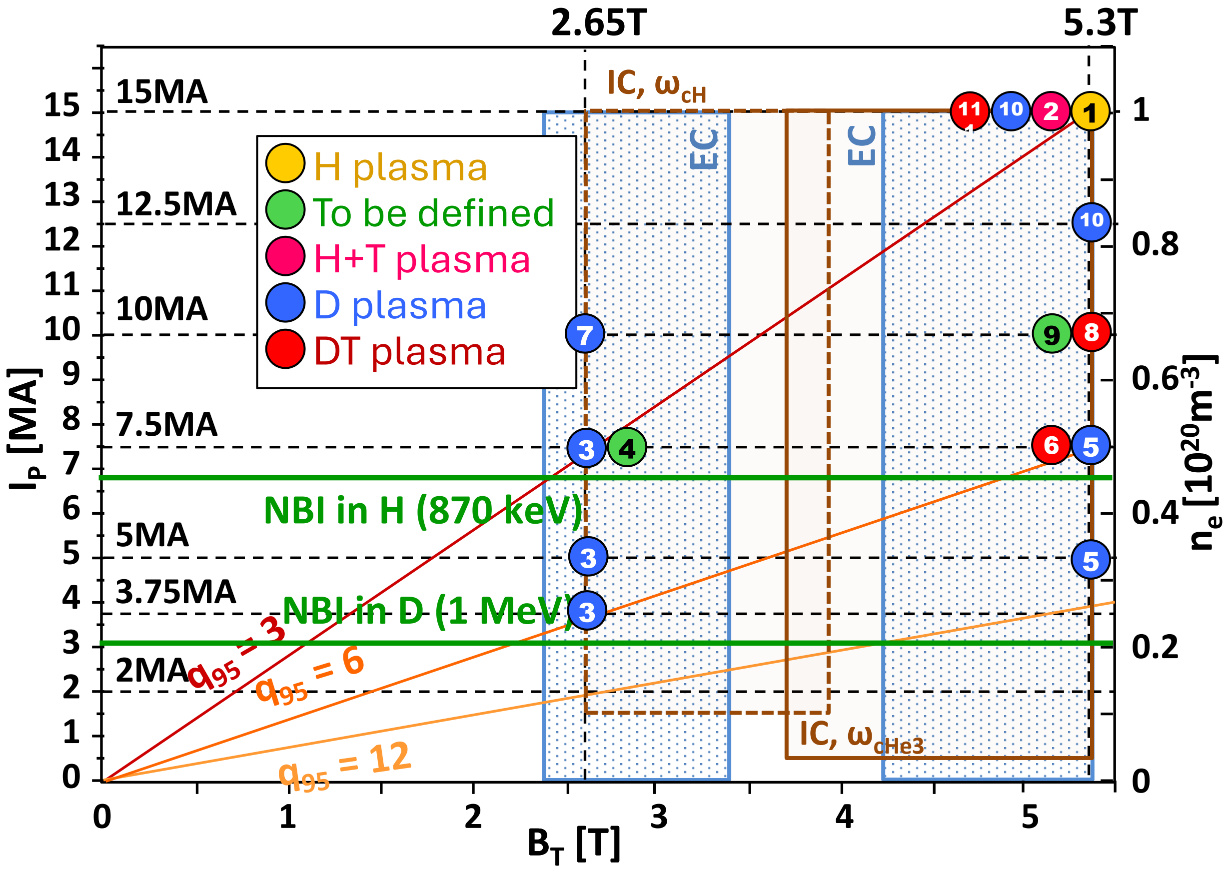
~ 2.7 months
6 5-7.5 MA / 2.65 T
Hydrogen

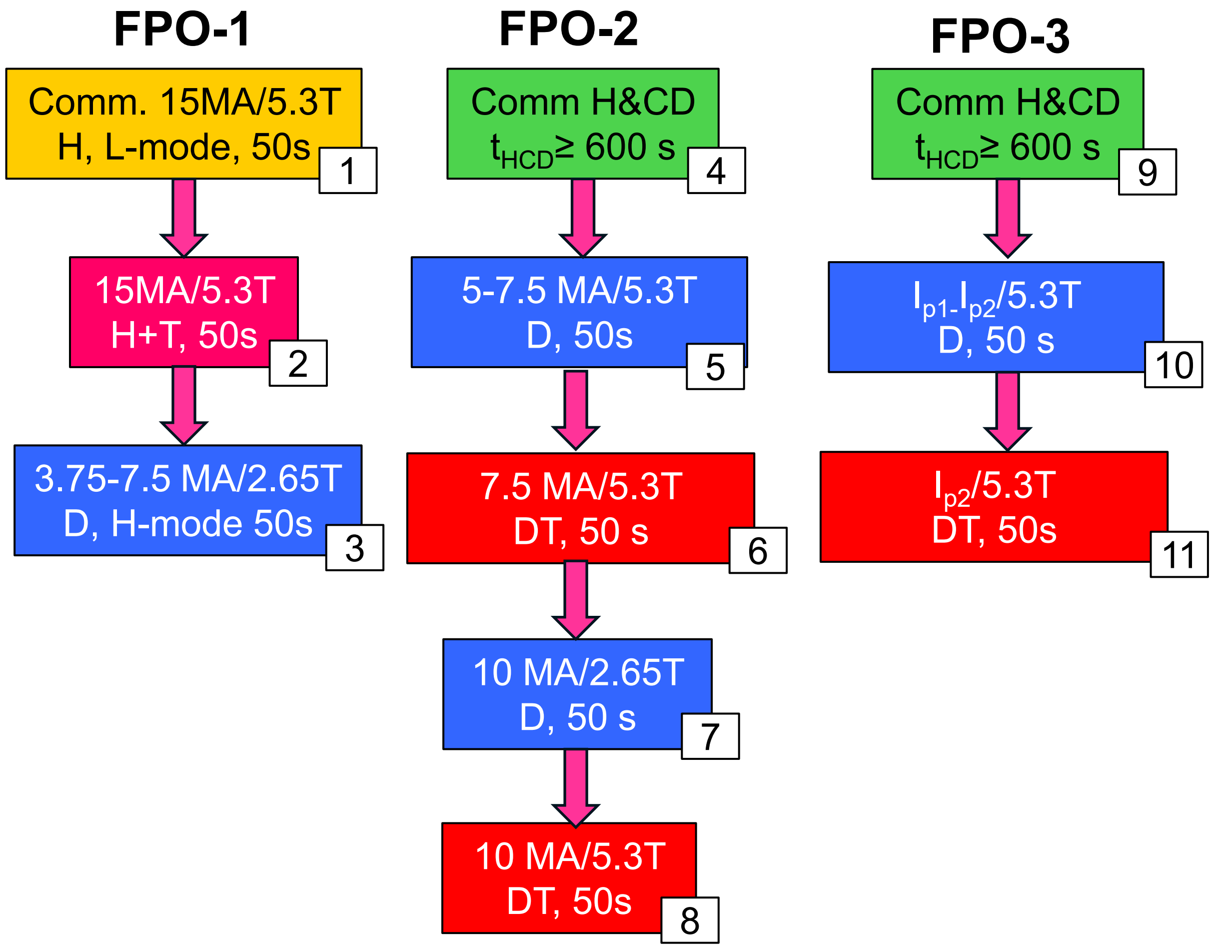
~ 4.4 months
7 5-7.5 MA / 5.3 T
Hydrogen

~ 3.2 months
8 15 MA / 5.3 T
Hydrogen

5 days
9 7.5 MA / 5.3 T
Hydrogen
(for post SRO PM analysis)

H&CD and DT-1 research plan





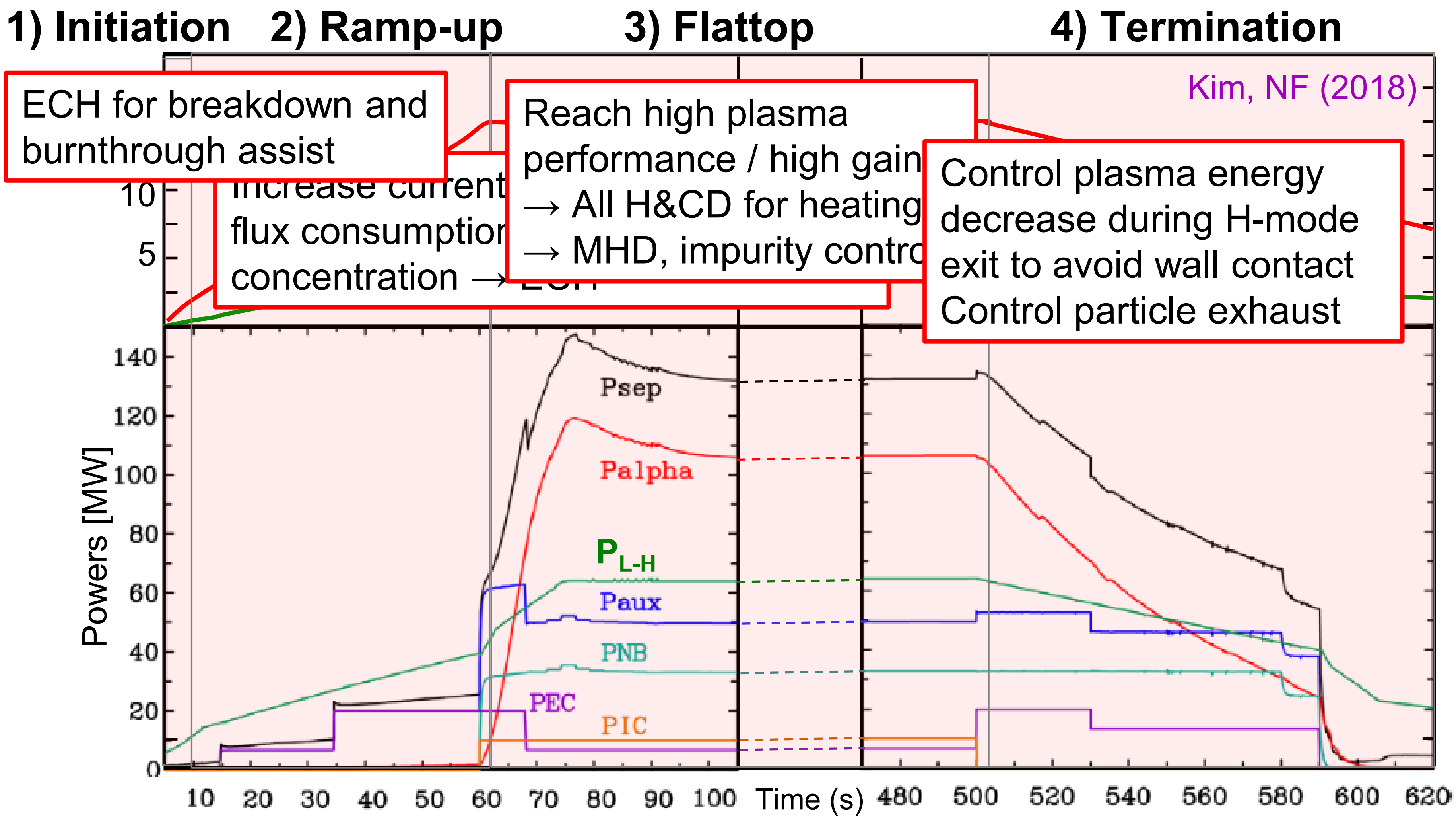
Operational points:

- FPO-1, FPO-2, FPO-3 displayed
- FPO-4, FPO-5: all at 15MA/5.3T

Assumption:
I_{p1} = 12.5MA, I_{p2} = 15MA

To wrap it up: H&CD across the phases of an ITER pulse

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



Summary

- **ECH** is the most flexible H&CD system in terms of operational conditions and dynamics → a priori the only one to be applied during ITER transient current ramp-up and plasma termination
  - EC-assisted startup
  - Workhorse for central heating → W removal
  - Best suited for real-time MHD control

- **ICH** indicates uncertainties in operating in W environment → Needs optimized phasing to reduce RF impurity generation.
  - ICH SRO tests will help deciding of its possible upgrade from 10 to 20 MW
  - ICWC is essential for wall cleaning without reducing the B-field
  - Heating ions directly enhances fusion gain

- **NBI** is essential for hybrid and steady-state operation:
  - It is the main source of non-inductive current drive
  - Significant constraints with plasma density due to shine-through losses


Thank you for your attention