Burning plasma aspects at ITER

Alberto Loarte on behalf of the Science Division
Science, Controls, and Operation Department
ITER Organization and many collaborators

Outline of talk

- ❑ **ITER mission and main design features**
- ❑ **Overview of ITER Research Plan**
- ❑ **ITER burning plasma scenarios**
  - ➢ ITER Q ≥ 5 burning plasma scenarios
  - ➢ Fast particles in burning plasmas
  - ➢ Access and exit from high Q plasma conditions and burn control
- ❑ **Burning plasma scenario integration issues**
  - ➢ Stationary power exhaust
  - ➢ Helium exhaust
  - ➢ Plasma fueling and DT mix control
  - ➢ Pedestal plasma conditions and Edge MHD control
  - ➢ Core MHD control
  - ➢ Disruptions and disruption mitigation
  - ➢ T retention and removal
- ❑ **Conclusions**

ITER mission and main design features

ITER mission goals

ITER shall demonstrate scientific & technological feasibility of fusion energy:

- **Pulsed operation:**

Q ≥ 10 for burn lengths of 300-500 s inductively driven current

→ **Baseline scenario 15 MA / 5.3 T**

$P_{\alpha}/P_{aux\text{-}heat} \geq 2$

- **Long pulse operation:**

Q ~ 5 for long pulses up to 1000 s

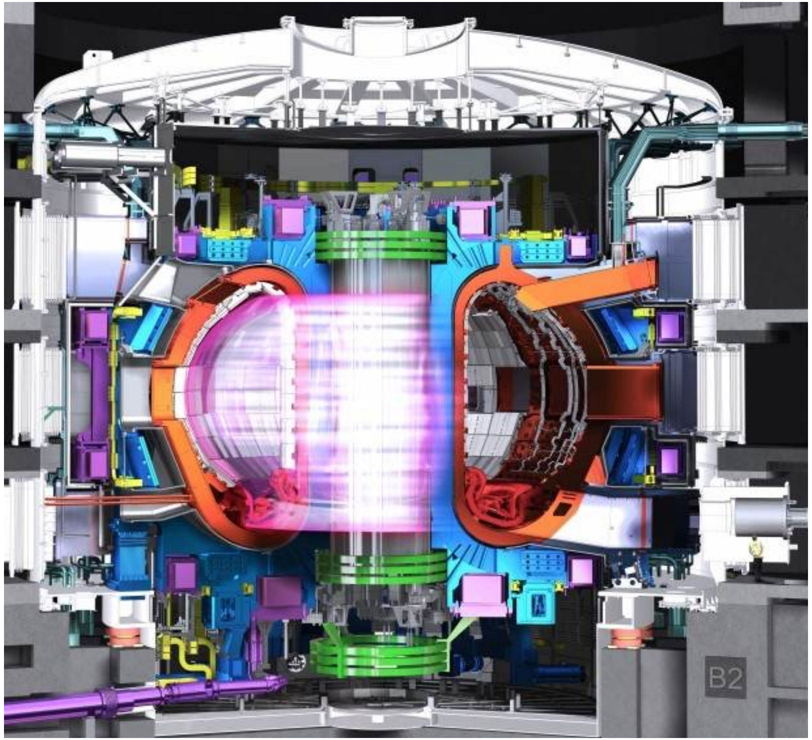
→ **Hybrid scenario ~ 12.5 MA / 5.3 T**

- **Steady-state operation:**

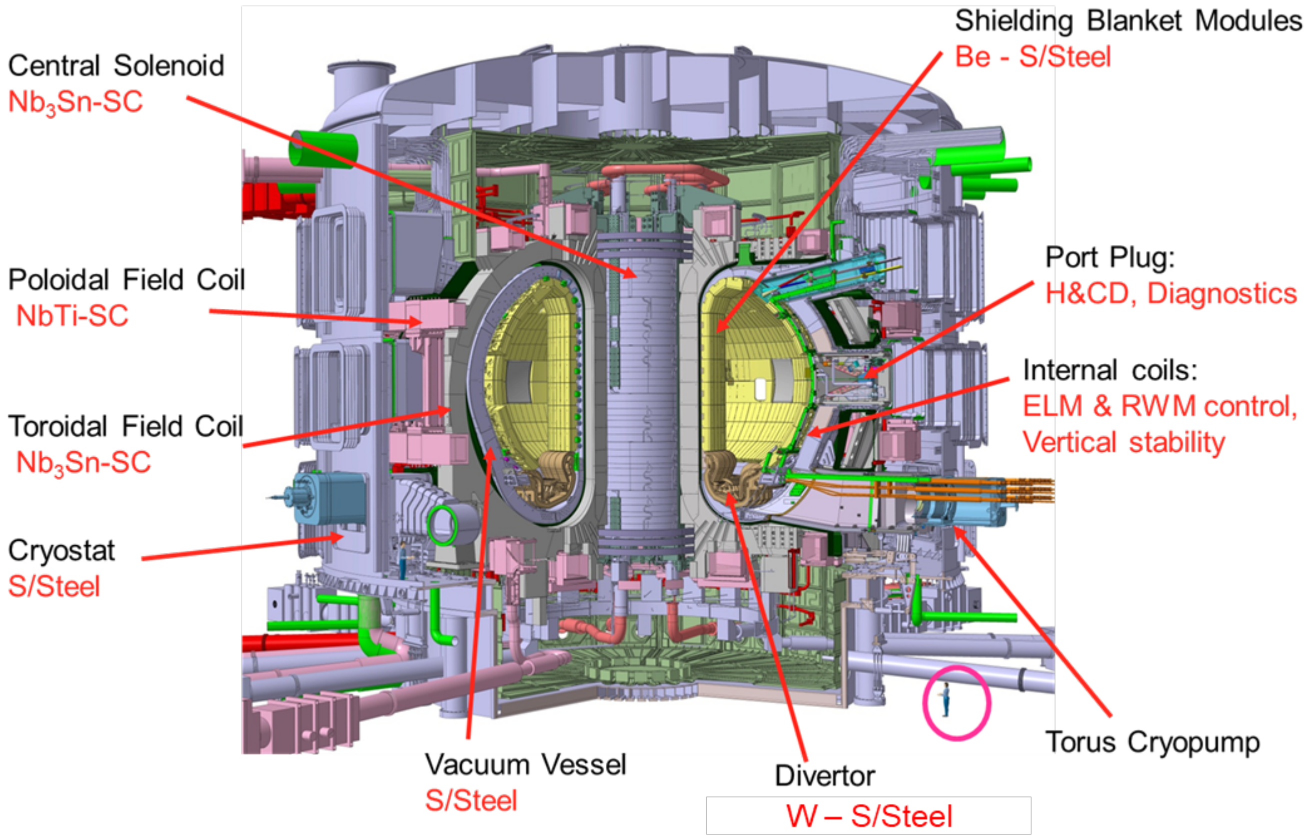
Q ~ 5 for long pulses up to 3000 s, with fully non-inductive current drive

→ **Steady-state scenario ~ 10 MA / 5.3 T**

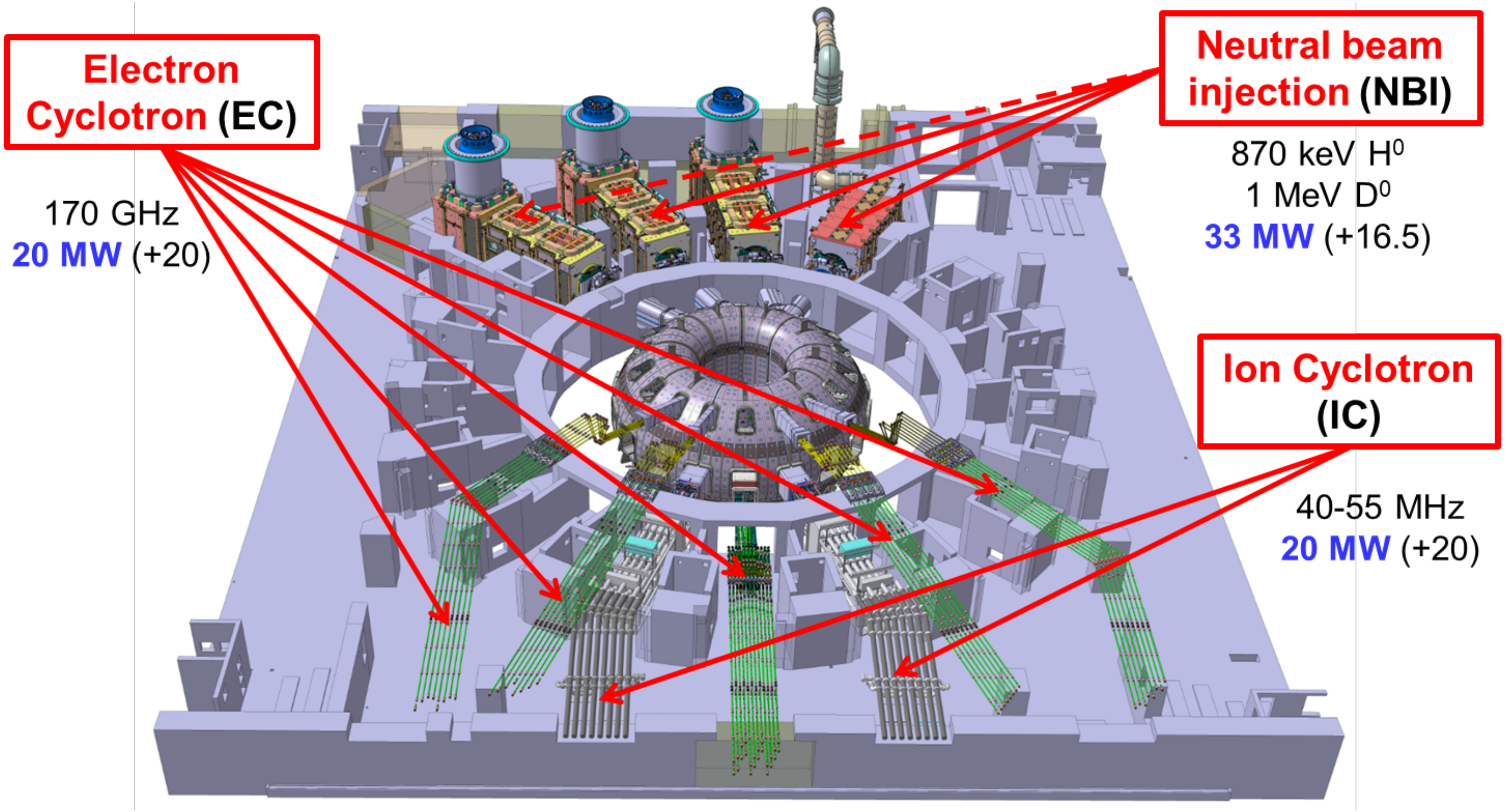
$Q \sim 5 \rightarrow P_{\alpha}/P_{aux\text{-}heat} \sim 1$



ITER Main Design Features

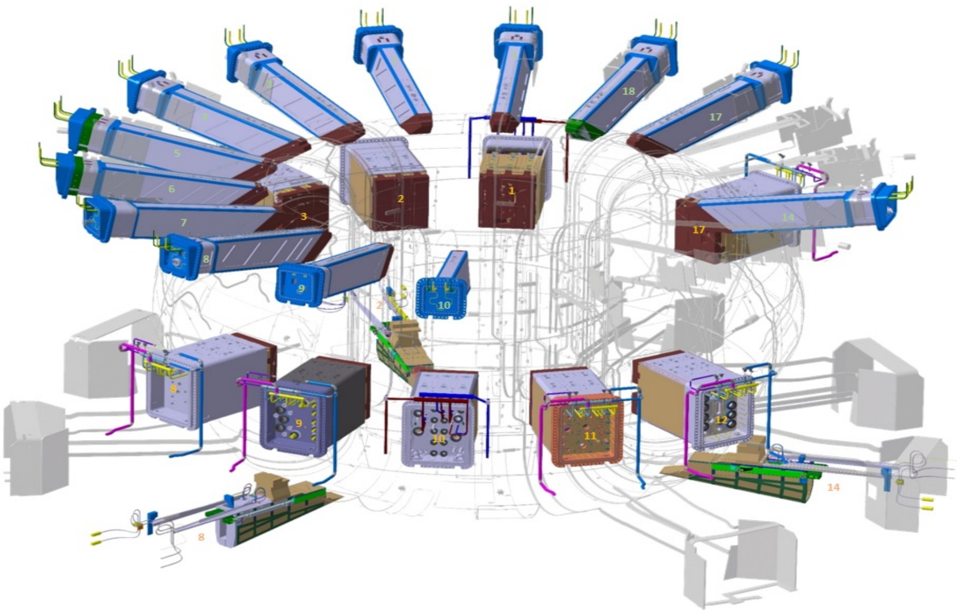


ITER Heating and Current Drive systems

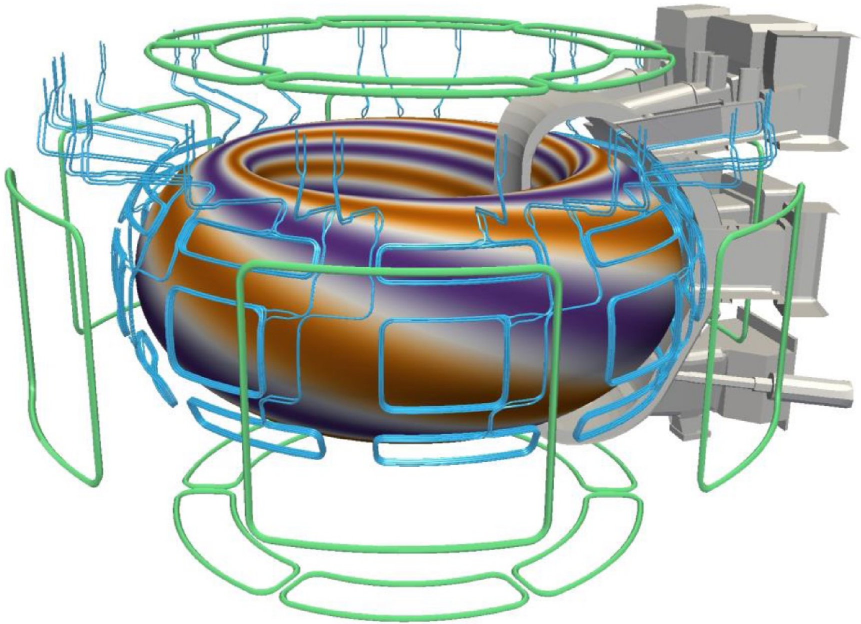


ITER Diagnostics and 3-D coils (Error Field, ELM control)

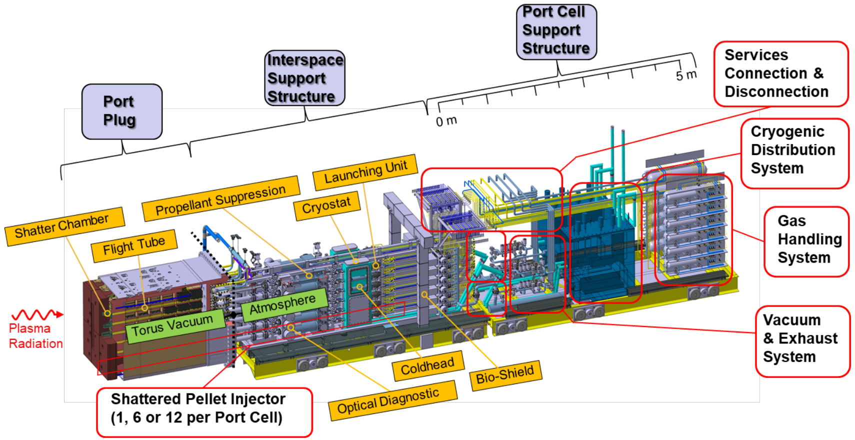
- ❑ **Diagnostics: ~ 60 instruments measuring ~ 100 parameters**

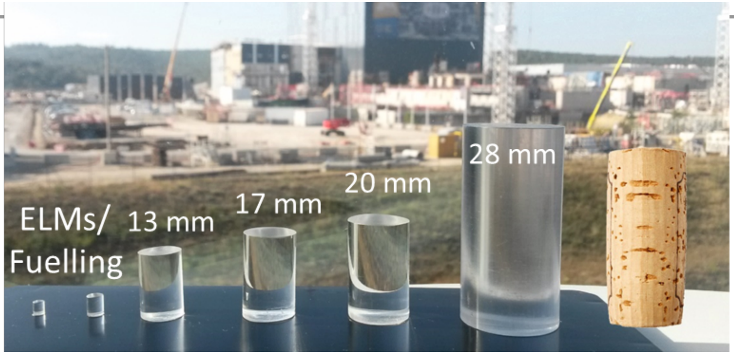


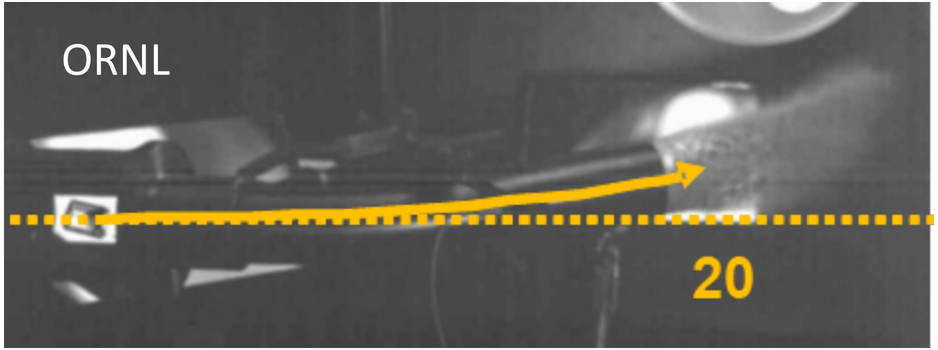
- ❑ **External error field correction coils + internal ELM control coils**

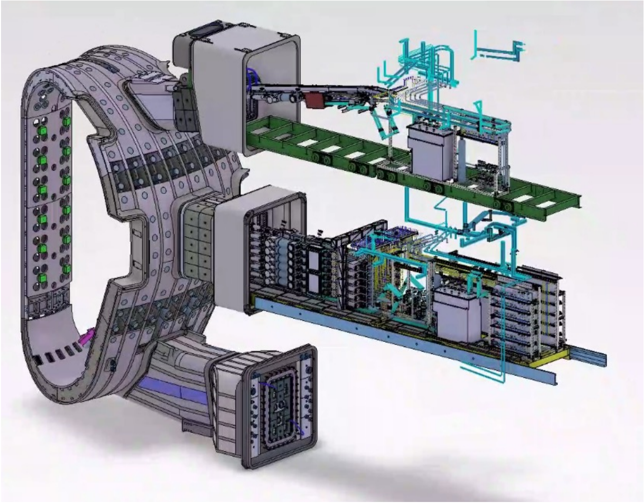


ITER Disruption Mitigation System







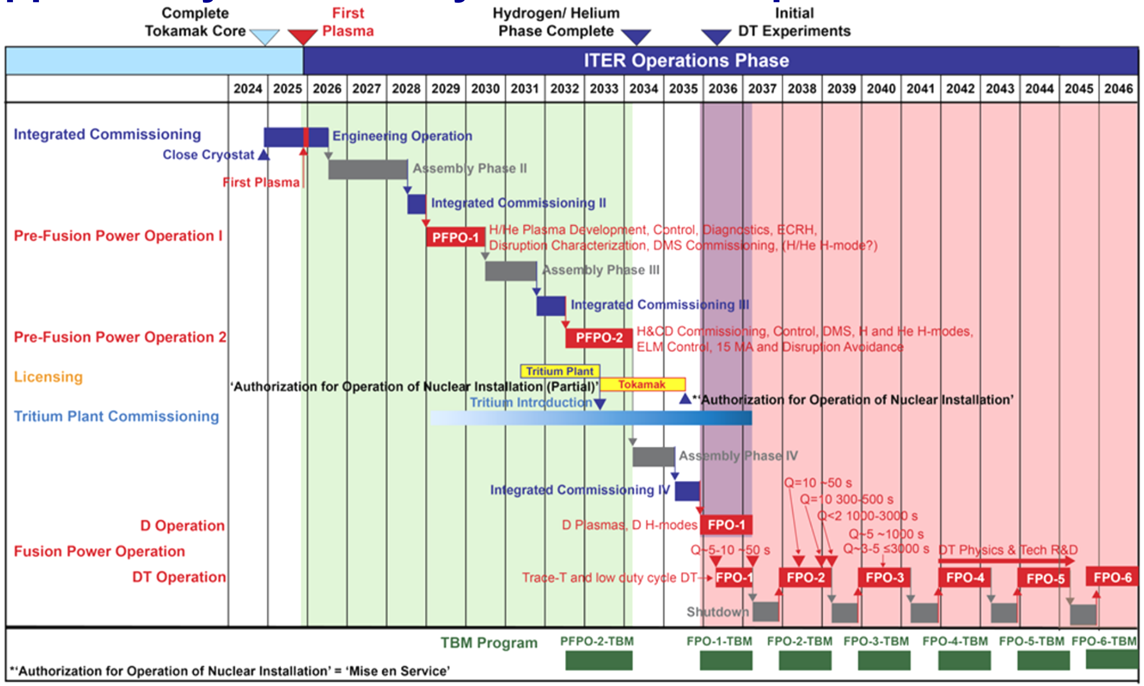


Overview of ITER Research Plan

ITER Research Plan

IRP describes strategy for R&D to achieve Project goals starting from First Plasma :
Q = 10 (300-500 s),Q = 5 (1000 s) & Q = 5 steady-state

Proposed R&D is supported by available systems in each phase



https://www.iter.org/technical-reports

Integrated Commissioning-First Plasma-Engineering Operation

1. Integrated Commissioning

- Integrated commissioning of:
  - Plant systems (central control systems, power supplies, cooling/baking, vacuum, cryogenics etc.)
  - Magnet systems to level required for FP (nominally 50% maximum current)
  - ECRH, diagnostics, fuelling, GDC, PCS systems
- Magnetic diagnostic calibration

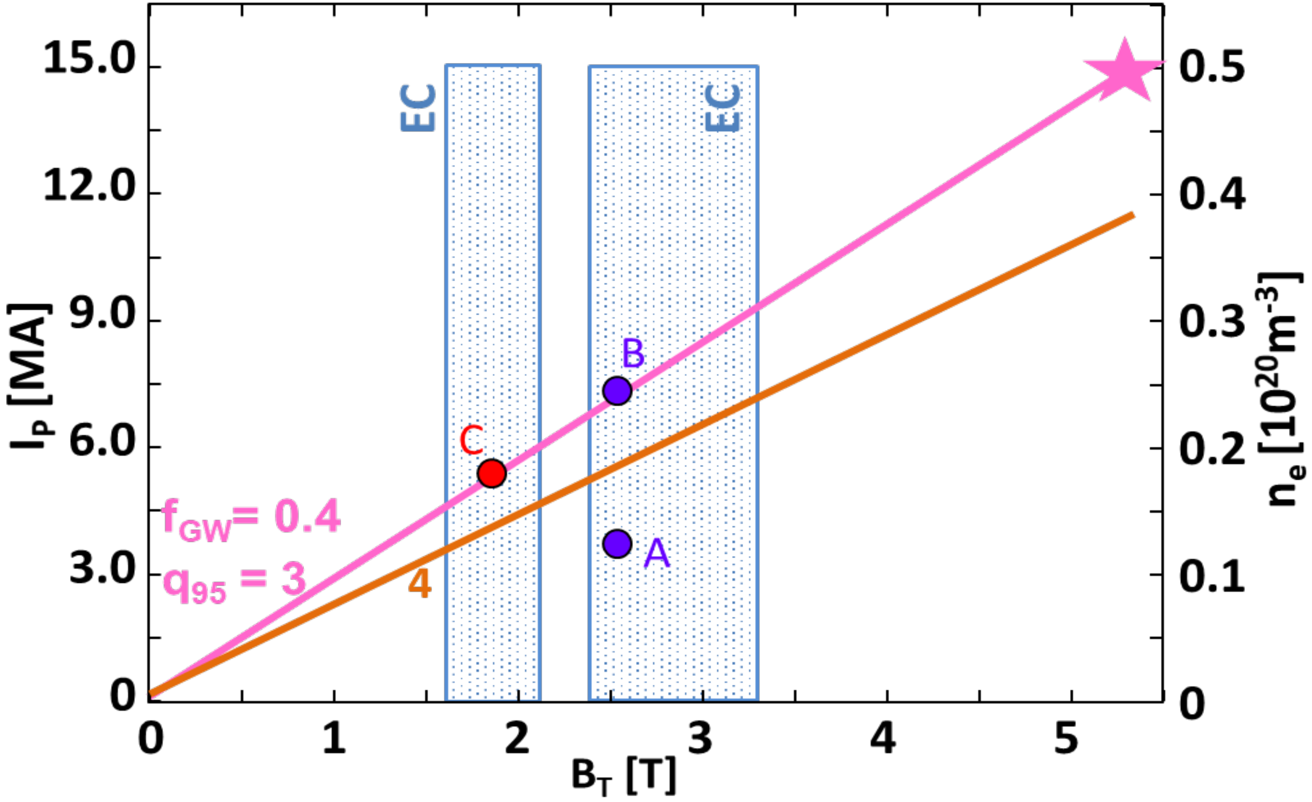
2. First Plasma

- 100 kA/ 100 ms milestone with ECH assisted start-up (P. de Vries, NF 2019)

3. Engineering Commissioning

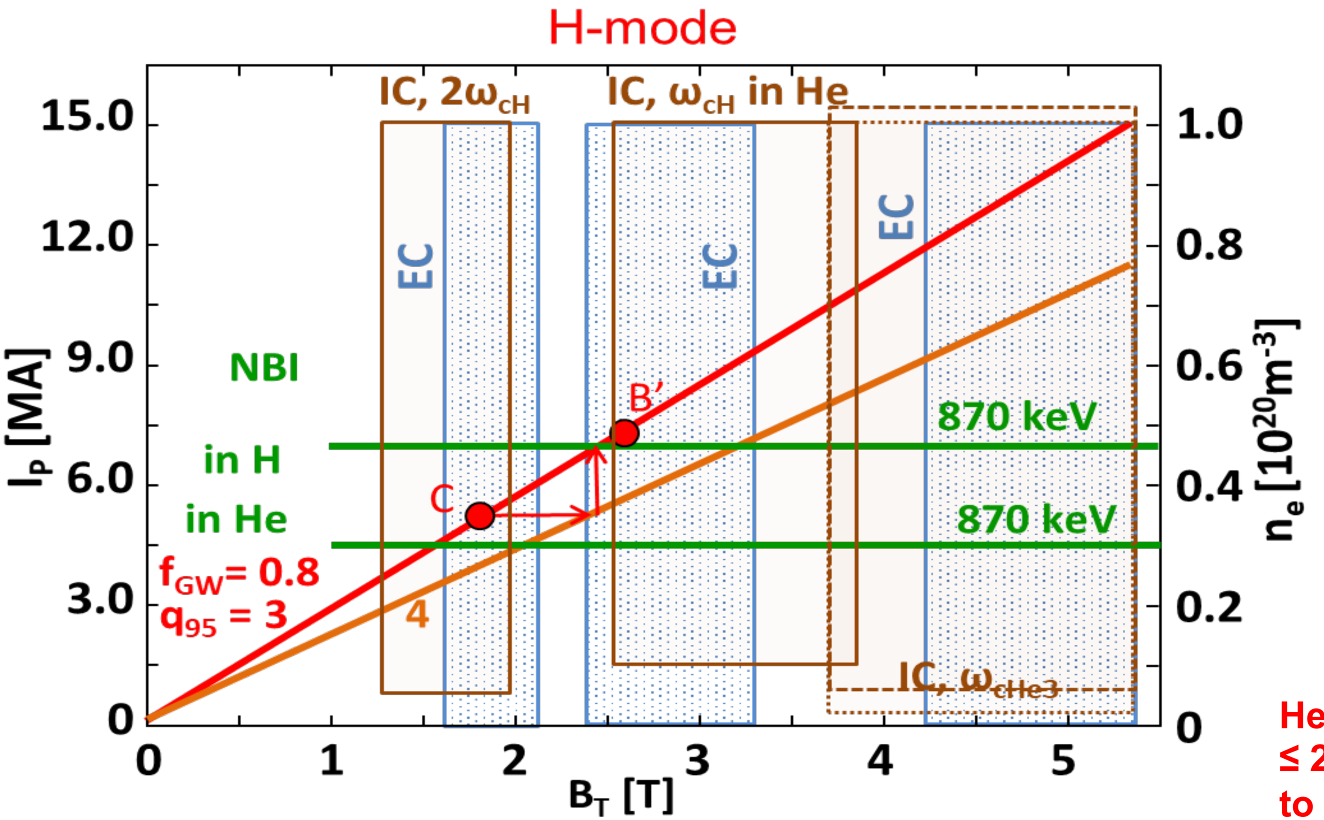
- Performance tests of all Magnet systems to full current
- Definition of strategy to align plasma facing components

PFPO-I



- **A.** 1^{st} divertor plasma
  3.2 MA / 2.65 T
- **B.** q_{95}=3 plasma
  7.5 MA / 2.65 T
  - L-mode in PFPO-I
- **C.** 1^{st} high confinement
  5 MA / 1.8 T

PFPO-II

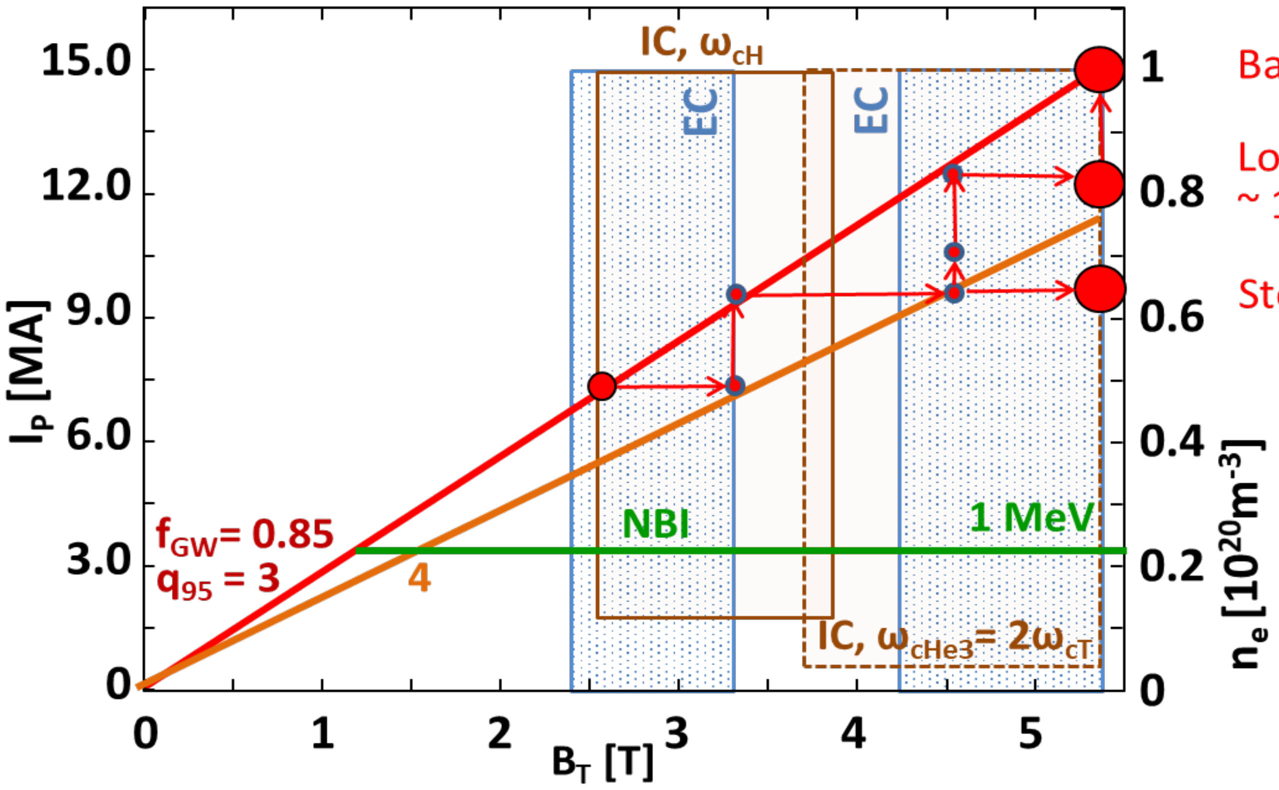


- **C.** *q_{95}=3 plasma*
  5 MA / 1.8 T
  - *H-mode*
- **B'.** q_{95}=3 plasma
  7.5 MA / 2.65 T
  - H-mode

Plasma species	$P^{H\text{-mode}}$ (MW) $n_e = 5\ 10^{19}\ m^{-3}$
H	59
He	42

He H-modes are robust $P_{aux}/P_{LH} \leq 2.0$ → uncertain extrapolation to DT (fuelling, PWI, pedestal, ELM control, …) → R&D required

Fusion Power Operation (D/DT)



Baseline 15 MA / 5.3 T

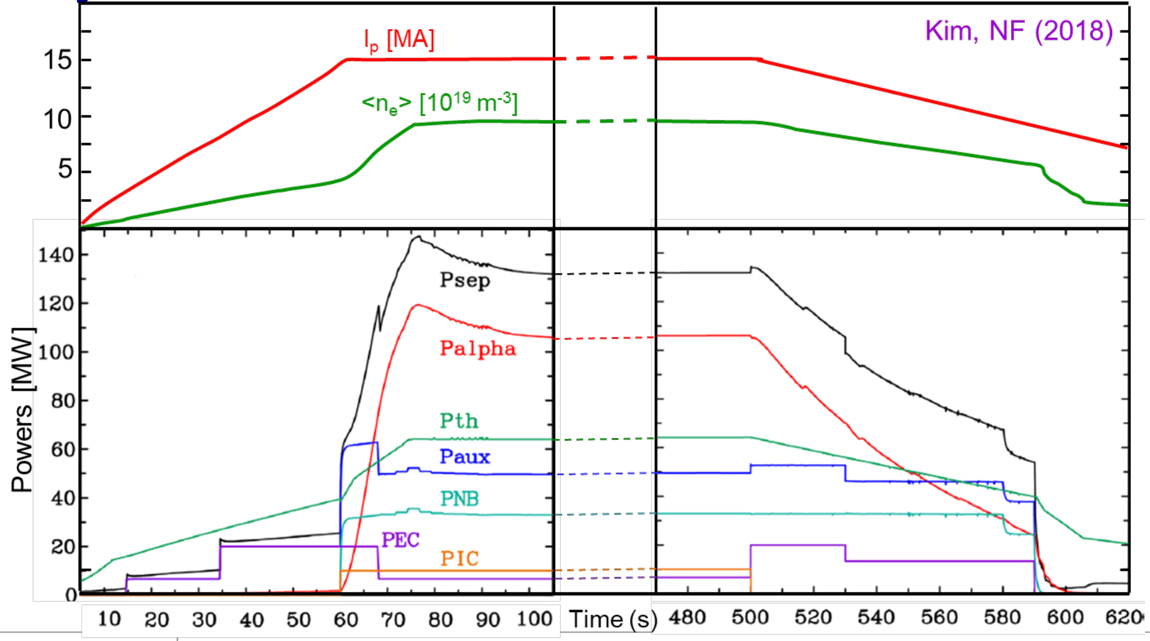
Long pulse hybrid
~ 12.5 MA / 5.3 T

Steady-state ~ 10 MA / 5.3 T

ITER burning plasma scenarios

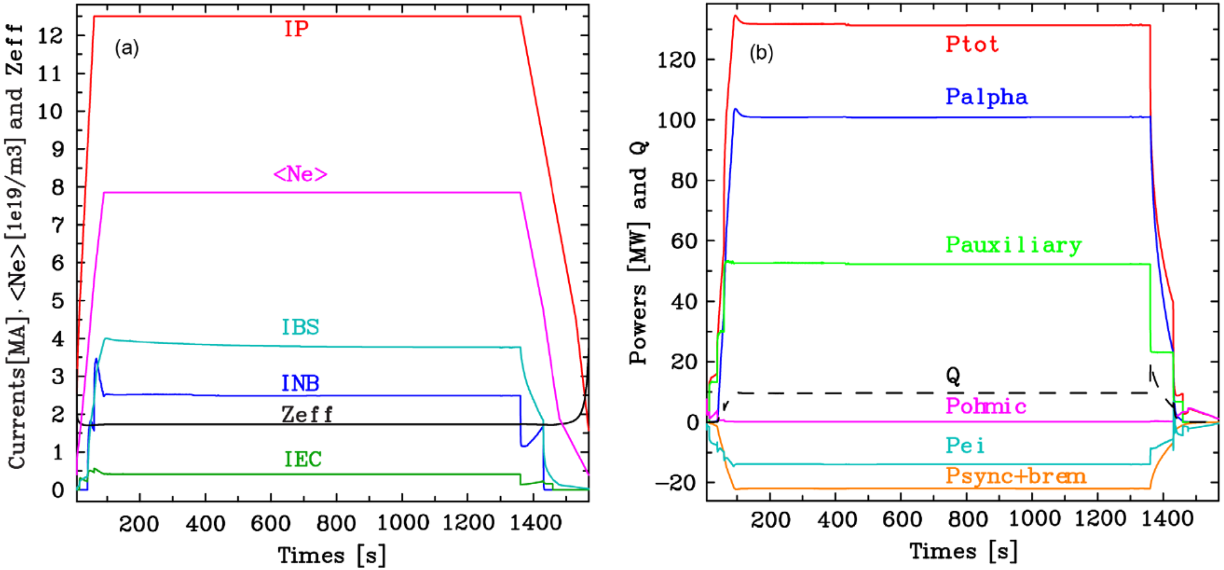
ITER Q = 10 scenario (300 – 500 s burn)

- Based on conventional sawtoothing H-mode with H_{98} = 1 → scenario used for the design of magnets and components (15 MA/5.3 T)
- $P_{aux} = P_{NBI} + P_{ECH}$ (+ P_{ICH}) ~ 50 MW → Alpha-heating dominant scenario with non-inductively driven current ~ 35%



ITER Q ≥ 5 scenario (1000s burn)

- ❑ **Main option is based on improved H-mode/hybrid scenario with q(0) > 1 and H_{98} > 1.2 with burn length limited by q(0) reaching 1 (12.5 MA/5.3 T)**
- ❑ **Obtained with $P_{aux} = P_{NBI} + P_{ECH}$ (+ P_{ICH}) ≥ 50 MW with non-inductively driven current ~ 55%**

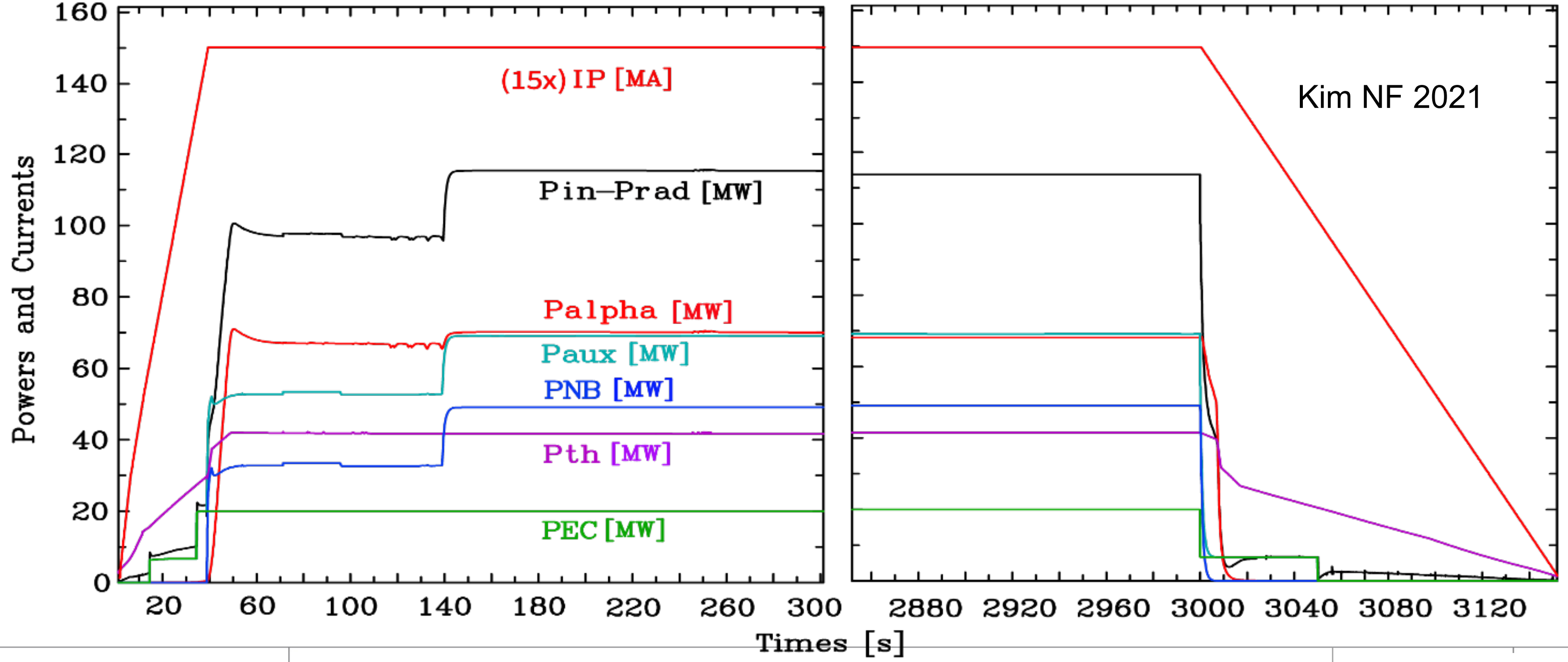


Kim NF 2016

- ❑ **Back-up option based on low density H-mode I_p > 13.5 MA H_{98} = 1**

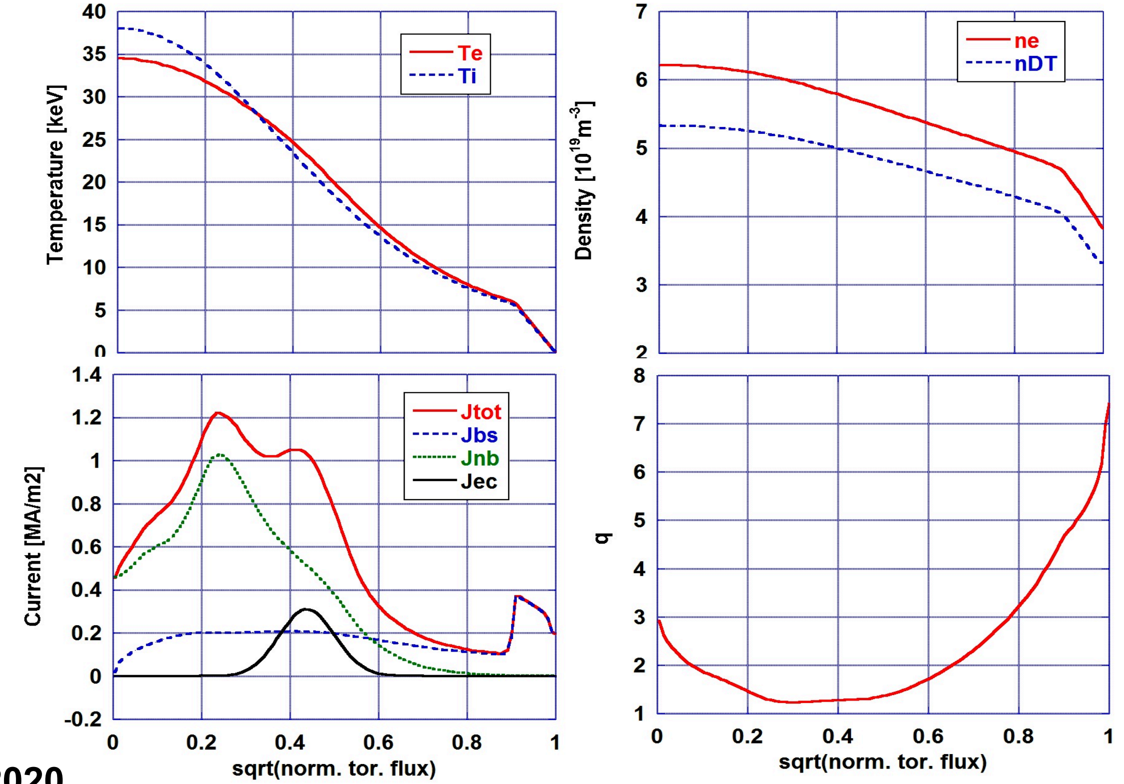
ITER Q ~ 5 scenario (steady-state)

- Based on improved H-mode/hybrid scenario with stationary q profile (q > 1) and H_{98} > 1.5 length limited to 3000s by hardware design (10 MA/5.3 T)
- Obtained with $P_{aux} = P_{NBI} + P_{ECH} \geq$ 70 MW with non-inductively driven current ~ 100%



Q = 5 steady steady-state plasma at 10 MA

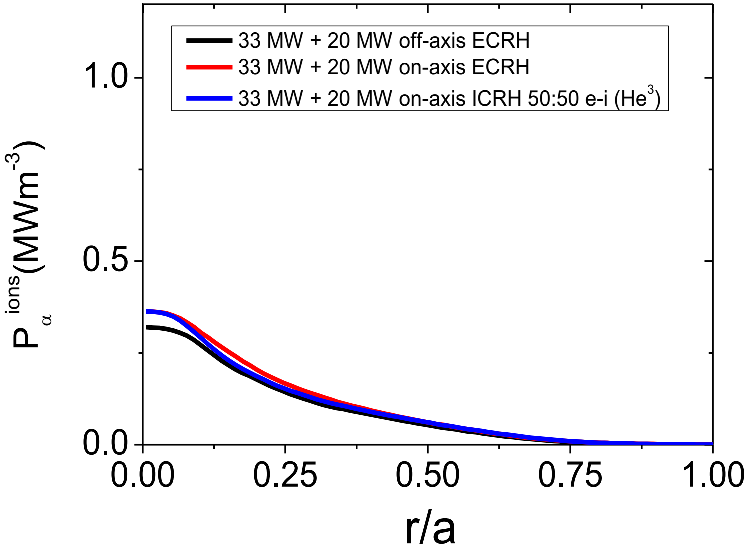
- **Conditions identified by 1.5-D ASTRA modelling**
  - ✓ EPED1+SOLPS used for pedestal and boundary
  - $Q=5.02$, $f_{GW}=0.69$
  - $H_{98}=1.52$, $\beta_N=3.02$
  - $q_{min}=1.23$
  - Relatively high $l_i(3)\sim0.87$ mainly due to 50 MW NBI (+ 20-30 MW ECH)
  - Improved confinement is essential

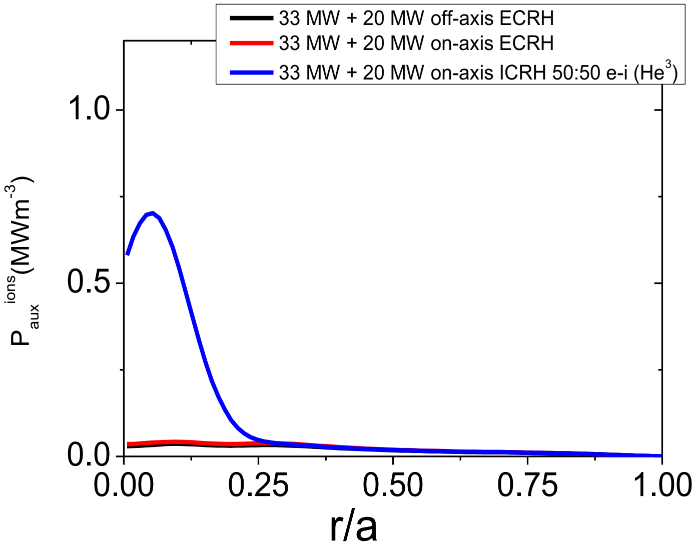


Polevoi – NF 2020

Heating profiles in ITER burning plasmas

- $Q > 5 \rightarrow P_{\alpha} > P_{aux}$ (but locally $q_{aux} > q_{a}$)
- Typical slowing down time for 3.5 MeV α's ~ 1 s → good fast ion confinement required for efficient α heating





Energetic ions in ITER scenarios - I

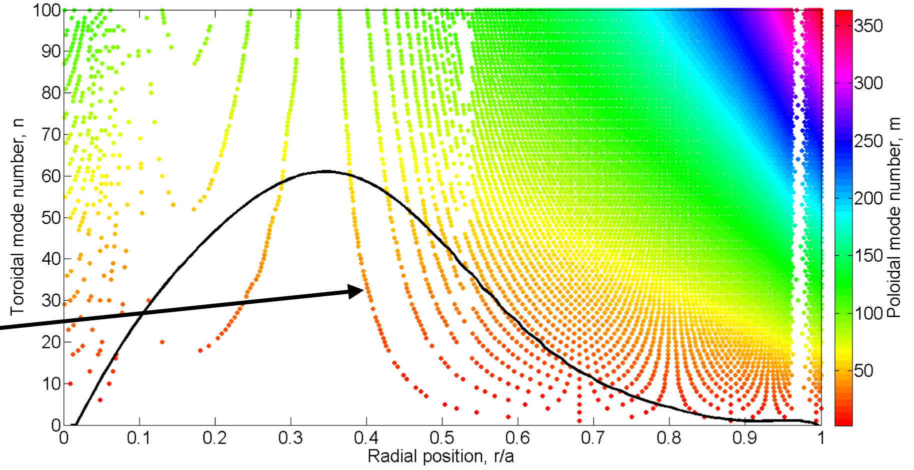
- **Energetic ions impact on ITER burning plasmas**
  - **Can drive MHD Alfvén eigenmodes → energetic ion loss $P_\alpha \downarrow$ ☹**
  - **Can reduce anomalous transport level → higher τ_E → $P_\alpha \uparrow$ ☺**
  - **Can increase core plasma β and thus shafranov shift → increased edge stability/pressure → increased τ_E → P_α ☺**
  - **Alfvén eigenmodes can reduce plasma turbulence → higher τ_E but energetic ion loss → P_α ? ☺**
- **Coupling between all effects difficult to predict in quantitative way for ITER burning plasmas since P_α is dominant**

Energetic ions in ITER scenarios - II

- **Consequences of EP-driven Alfvén eigenmodes range from**
  - **Benign saturation → significant high-amplitude bursting and transport**
- **Extrapolation from present machines difficult due to small** $\rho_\alpha / a \simeq 10^{-2}$

- **Besides loss of heating, ITER first wall loads acceptable for fast ion losses of a few %**
- **Max power transfer from α's occurs when drift orbit width ~ mode width → n ~ 30**
- **Many overlapping AE**

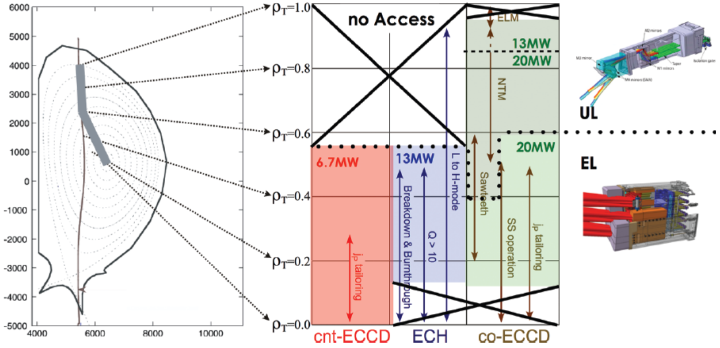
Radial localisation of TAE gaps in ITER Q = 10 plasmas

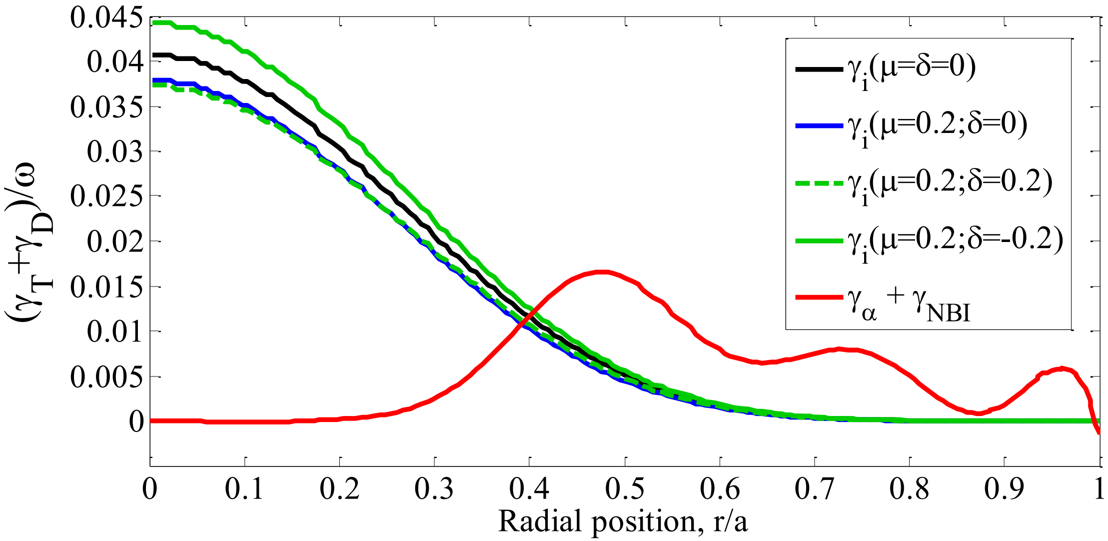


Energetic ions in ITER scenarios - III

- **In Q = 10 core plasma region, thermal Landau damping overcomes drive but not so in edge**

ITER will quantify impact of fast ion instabilities in Q = 10 plasmas and explore means for mitigation and control (e.g. ECH or ECDD used in experiments → impact on Q)





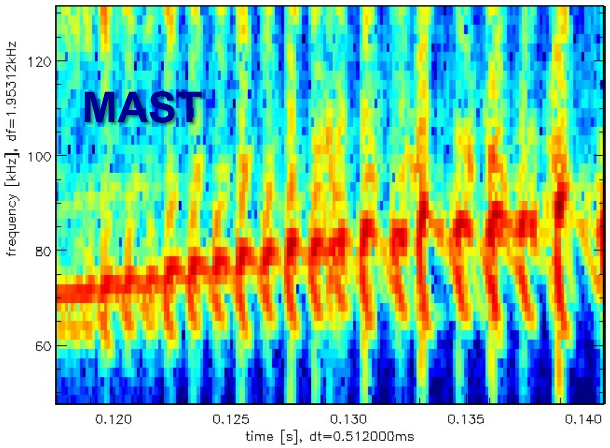
Thermal ion Landau damping

EP drive, NBI + alphas

Energetic ions in ITER scenarios - IV

- **v_α >> $v_{Alfvén}$**

  Balance of strong drive from slowing-down distribution and damping can lead to non-linear behaviour and possible increased transport



Frequency sweeping TAE in MAST #22807 where v_{NBI} > $v_{Alfvén}$

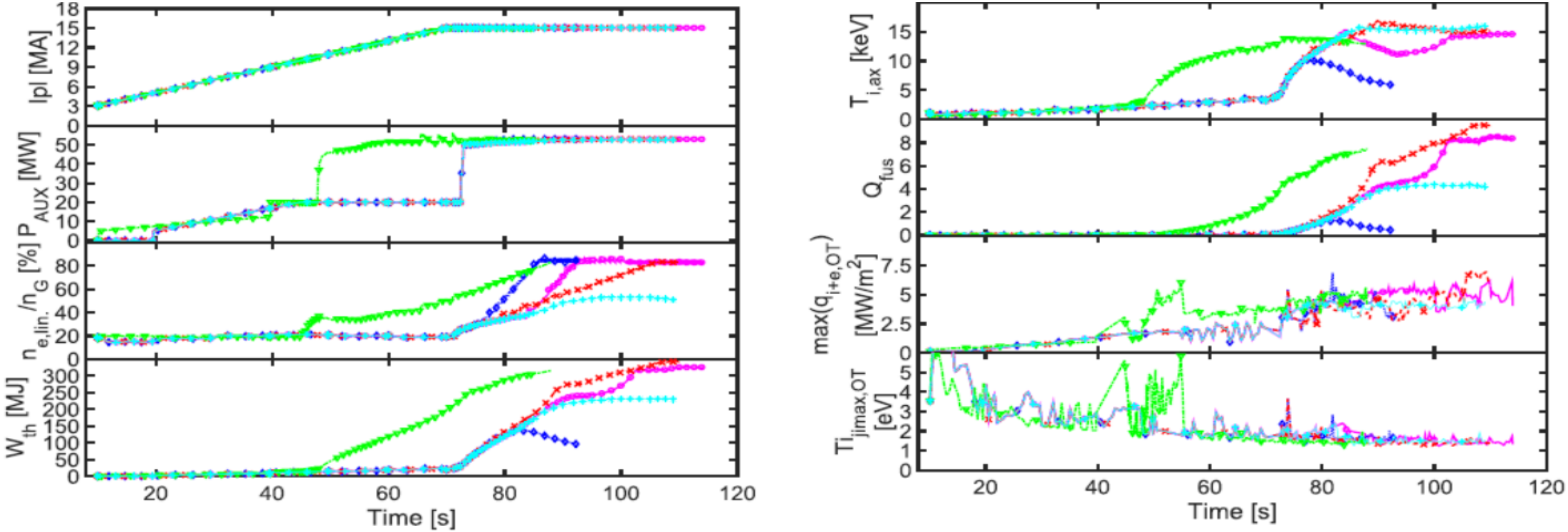
- **Reversed-shear Alfvén Eigenmodes (RSAEs) occur with magnetic shear reversal (Q ~ 5 steady-state plasmas)**
  - **RSAEs located around q_{min} and weakly damped (no/little continuum damping)**
- **Stronger EP drive for higher q (drive ~ q^2) → Higher amplitude RSAEs with higher q_{min}**

Access to high Q conditions

- Access to high Q requires build-up of P_{alpha} since P_{aux} is moderate and $P_{L\text{-}H}$ is high
- Key to high Q access is density control (gas fuelling for n_{sep} and pellet fuelling for n_{core})

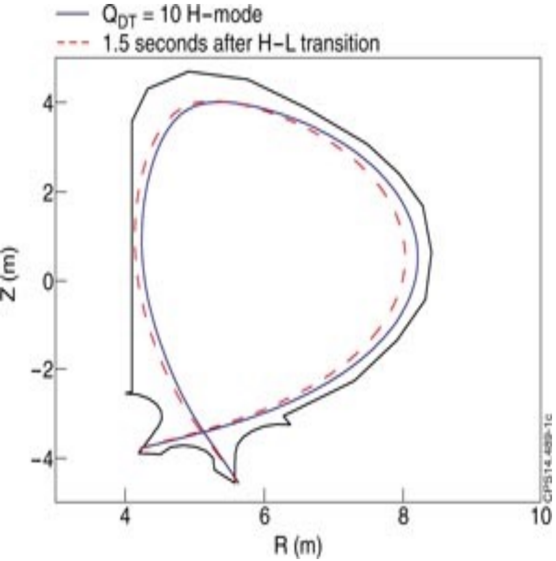
F. Koechl - ITER – JINTRAC - NF 2020

- Access in current ramp and low n_e allows high Q earlier in flat top

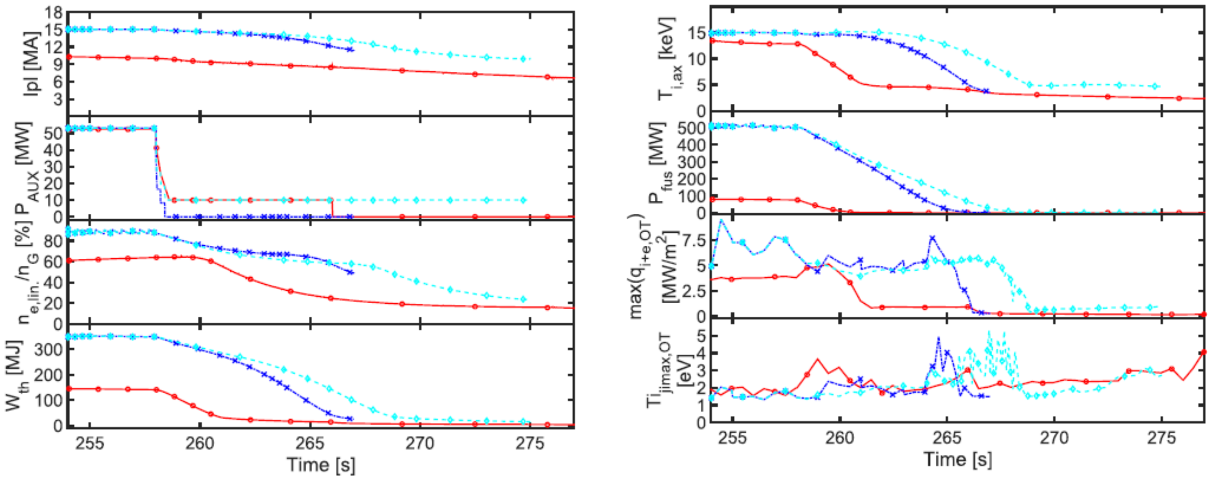


Exit from high Q conditions - I

- Main issue in exit from high Q is to avoid fast H-L transitions → radial plasma movement difficult to control and large power fluxes to divertor
- Adjustment of P_{aux}, fuelling and Ne seeding required to lengthen W_{plasma} decrease phase and avoid too high q_{div} or too deep detachment



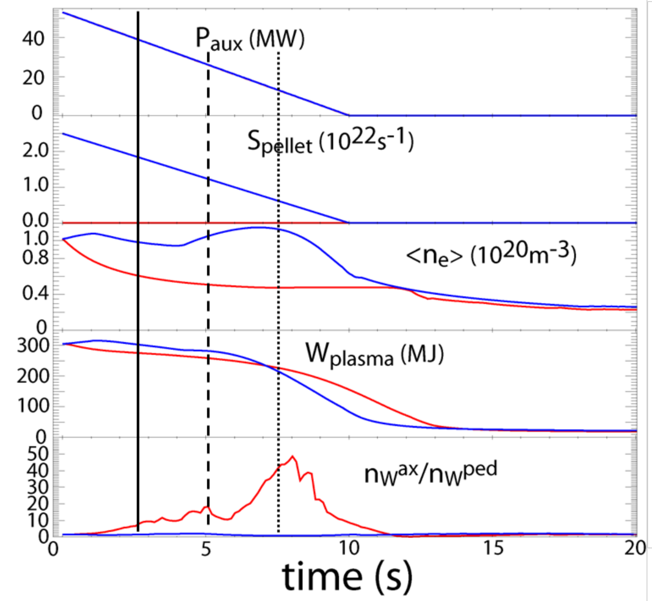
F. Koechl - ITER – JINTRAC - NF 2020

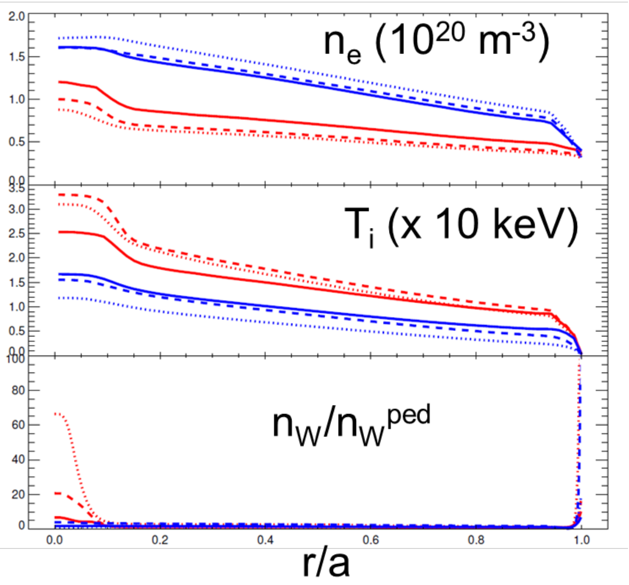


Exit from high Q conditions - II

- W accumulation can take place in exit from due to density/temperature profile evolution if pellet fuelling is quickly switched-off
- P_{rad} remains moderate due to high T_{core}

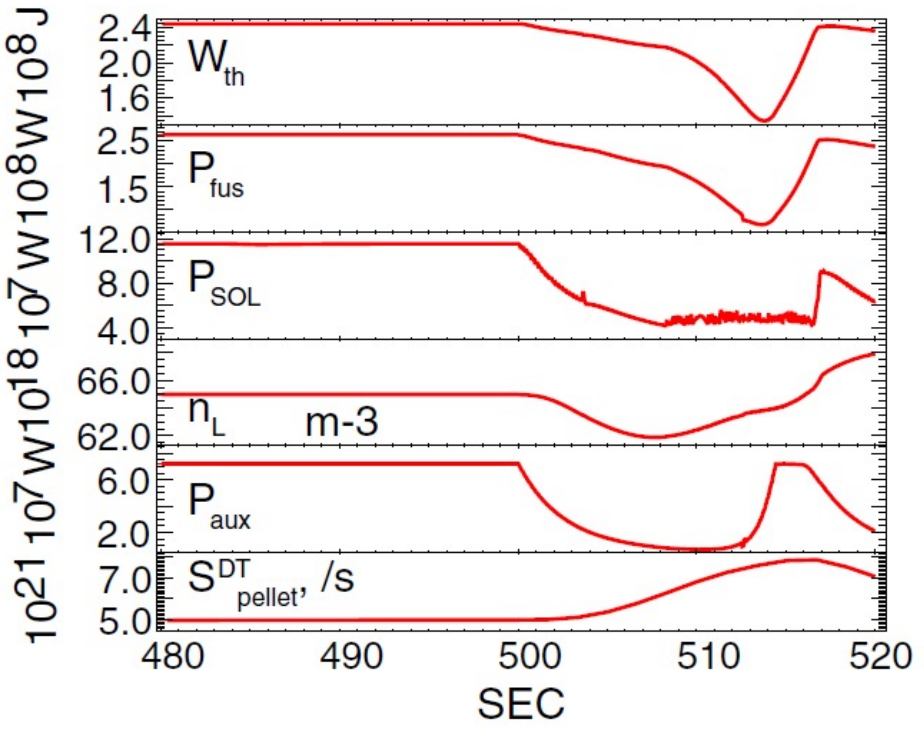
A. Loarte - ITER – JINTRAC – IAEA 2016





Burn control in high Q plasmas

- **Burn control will be studied in ITER burning plasmas**
  - ✓ Effectiveness of control by different actuators:
    - P_{aux}
    - Pellet fuelling
    - DT mix control
    - Pedestal control
    - ….



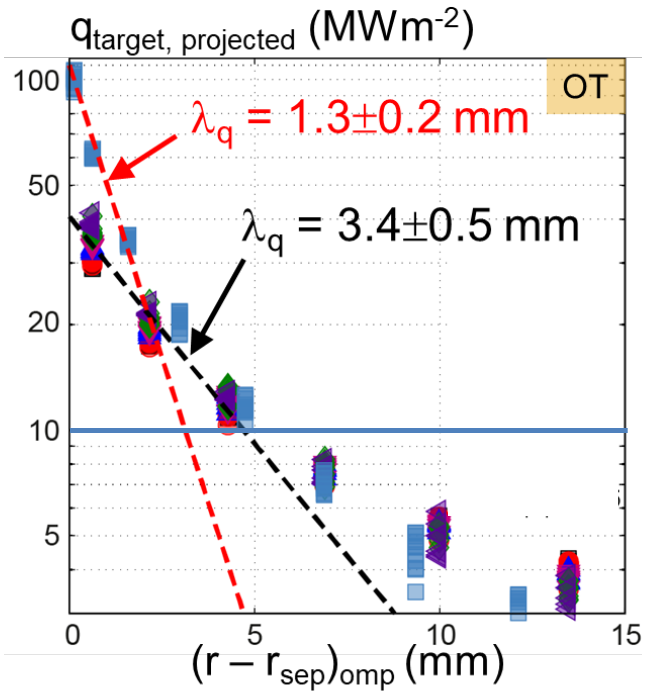
Kessel – NF 2015

➢ P_{tot}/P_{LH} ~ 2-3 and P_{α} ~ 1 – 2 P_{aux} in ITER poses specific challenges

Burning plasma scenarios integration issues

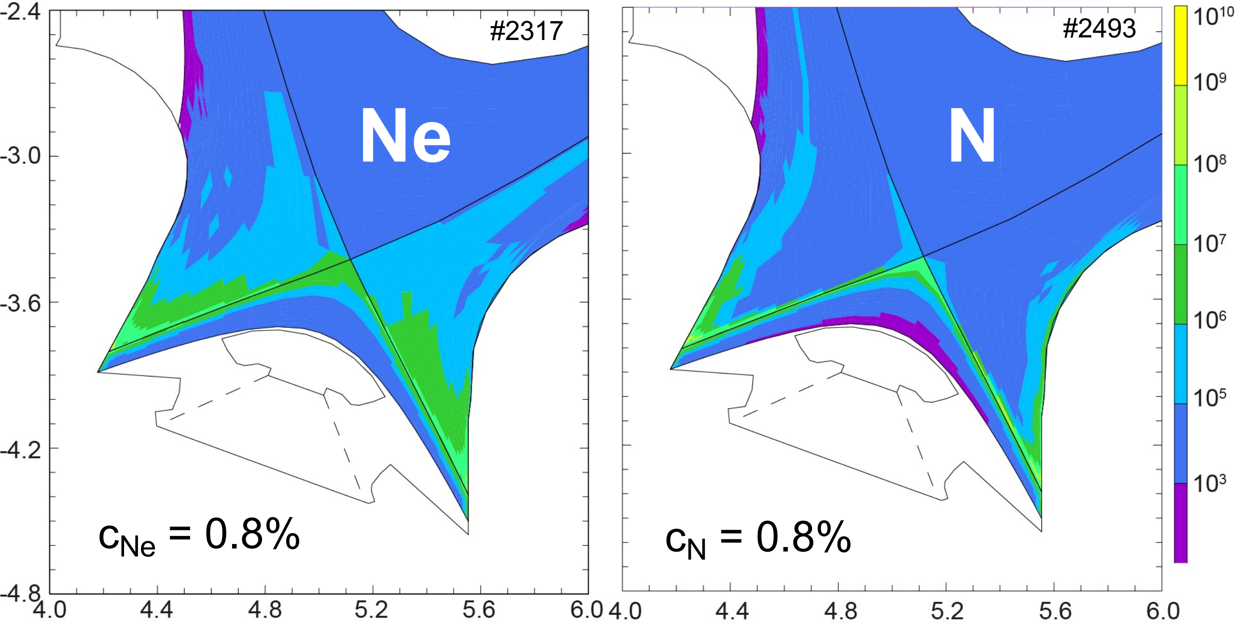
Stationary power exhaust

- Power in charged particles similar for all ITER high Q scenarios
  - Q = 10 with P_{aux} ~ 50 MW → P_{total} ~ 150 MW
  - Q ~ 5 with P_{aux} ~ 70 MW → P_{total} ~ 140 MW
- Expectations
  - Narrow near separatrix e-folding length → 80 – 100 % of P_{SOL} power arrives divertor
  - Broad far SOL e-folding length (+ ELMs) → 20 - 0 % P_{SOL} arrives at first wall
- **Burning plasma divertor power flux must be reduced by factors of 4 – 10 for compatibility with divertor target power handling capability**



R. Pitts NME 2019

SOL radiated power for Q = 10



$P_{RAD,DIV}$ = 56.6 MW
41.3 (Ne) + 15.3 (D)

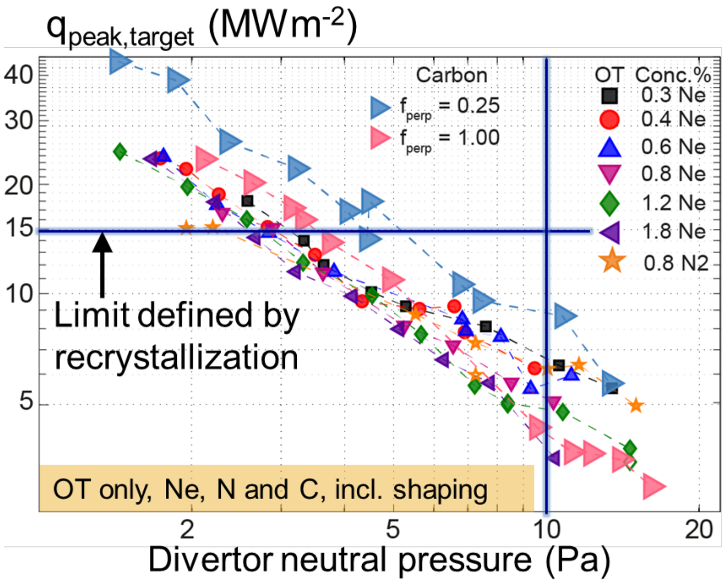
$P_{RAD,DIV}$ = 54.0 MW
38.6 (N) + 15.4 (D)

R. Pitts NME 2019

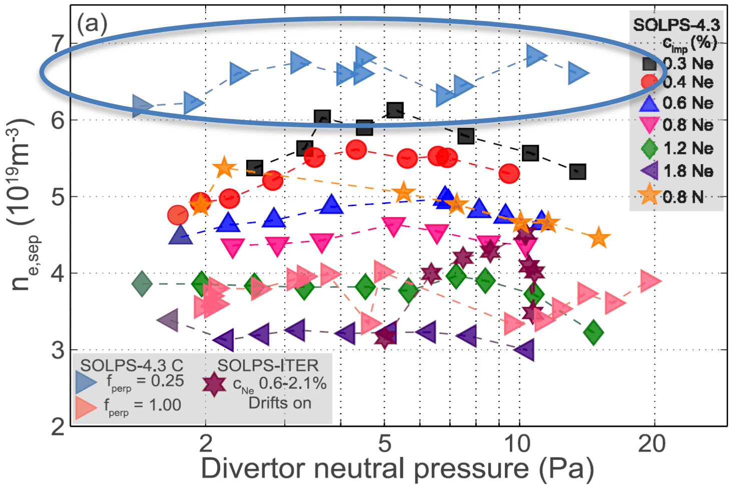
- Both Ne and N_2 provide divertor radiation at ITER scale → Ne favoured for ITER
- Up to 65% of P_{SOL} radiated at divertor → sufficient for power flux control
- **ITER will demonstrate compatibility of burning plasmas with radiative divertor conditions**

Divertor power flux for Q = 10

- Narrower λ_q → higher neutral pressures and detachment levels → edge transport in burning plasmas will be determined in ITER
- n_{sep} ~ 0.5 n_{GW} → impact on SOL and H-mode pedestal transport ?



R. Pitts PSI 2018
NME 2019

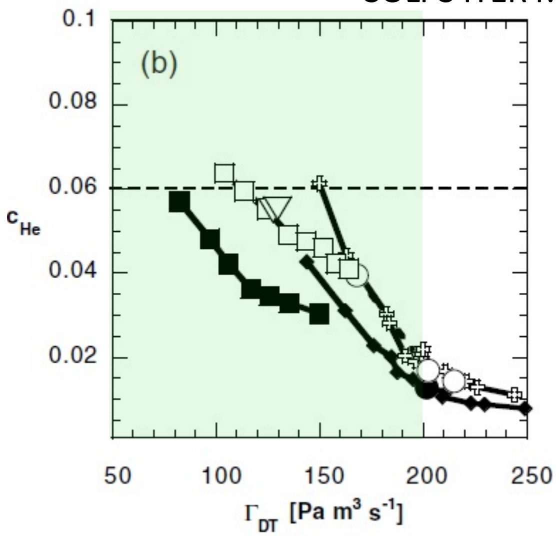


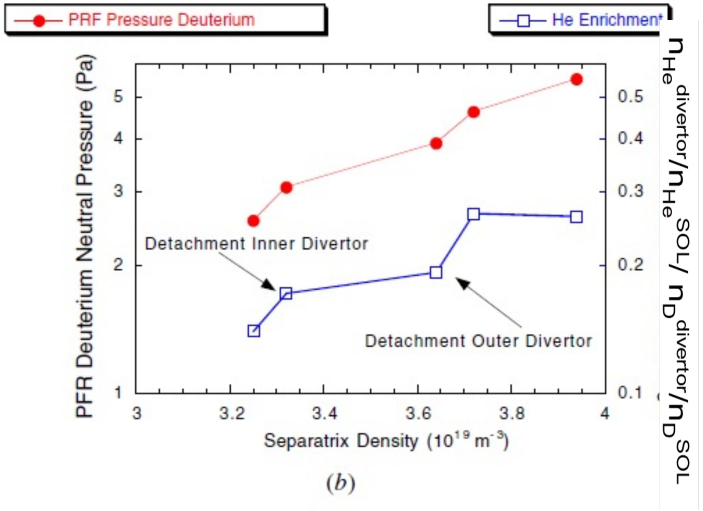
impact of drifts in simulations is qualitatively similar to reduced transport

Helium exhaust

- Experiments and ITER modelling → n_{He}/n_{sep} under 6% within fuelling/pumping capability (200 Pa $m^3 s^{-1}$ ~ 10^{23} DT atoms/s) → impact of He and DT transport in ITER burning plasmas will be quantified (potentially important in steady-state scenario)
- Detached divertor conditions increase He proportion in exhaust gas → power exhaust and helium exhaust aligned for ITER divertor design

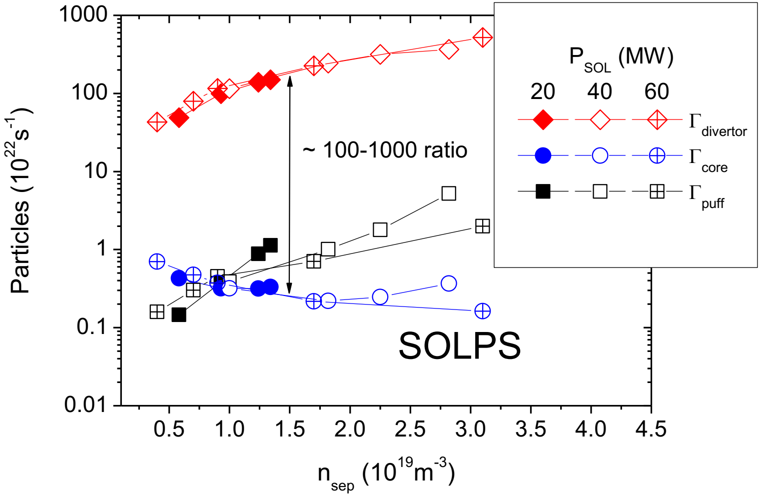
SOLPS ITER Modelling



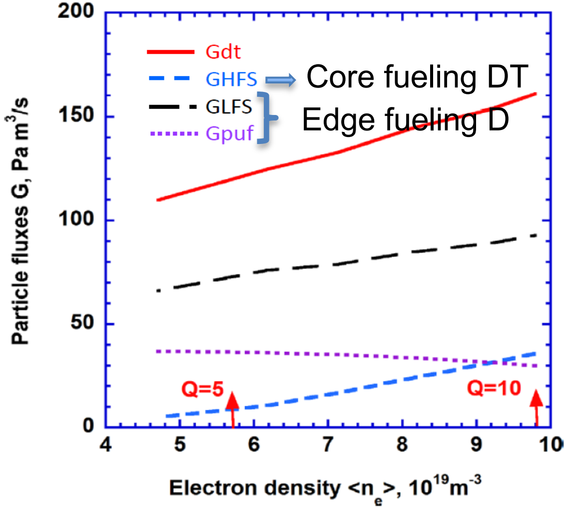


Plasma fueling and DT mix control

- Neutral penetration very ineffective in ITER due to poor neutral penetration
  - Separate control of n_{sep} (gas puffing by D) and n_{ped} (DT pellets)
  - Opportunity to minimize T throughput (10-20 % of total throughput) and increase burn fraction



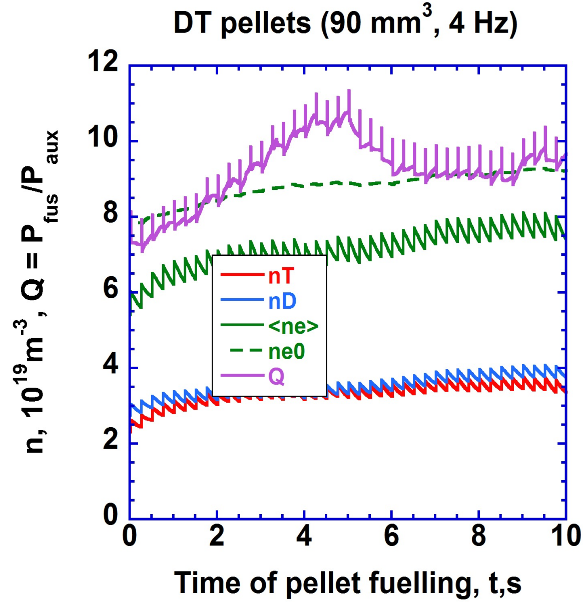
ASTRA+SOLPS b.c. Polevoi NF 2017



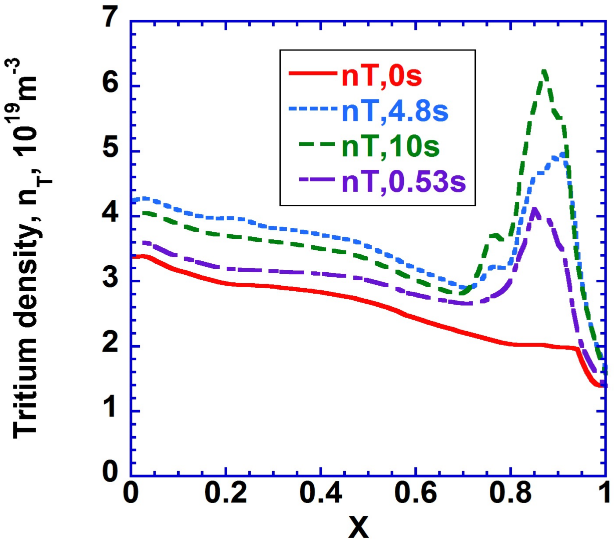
Assumes diffusive-like edge transport (no edge particle inwards pinch) → to be indentified in ITER

Pellet Fuelling and DT mix control

- Pellet deposition in ITER is peripheral due to high T_{ped} → edge source
- Density profiles with "edge bumps" and long relaxation times
- D & T transport to the core is key to DT mix control in burning plasmas

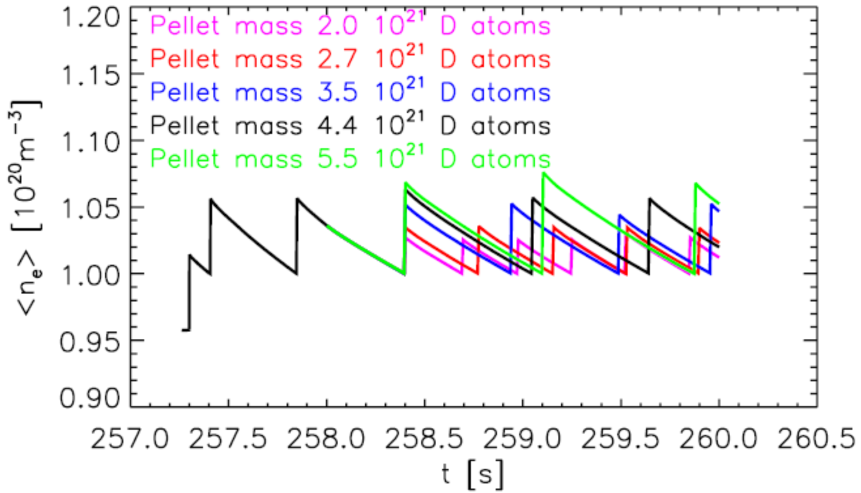


ASTRA-TGLF Polevoi EPS2018

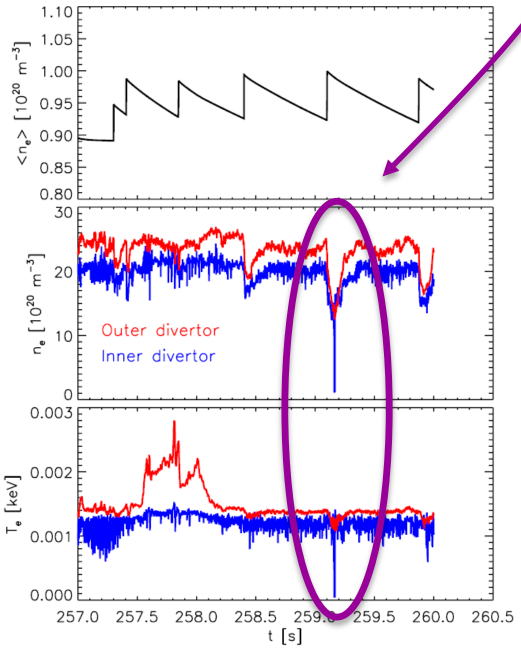


Pellet fuelling integration with radiative divertors

- Peripheral pellet deposition leads to edge n_e oscillations
- Edge n_e oscillations → divertor radiation and detachment
- Optimization of pellet size/frequency for effective fuelling compatible with stable divertor operation



L. Garzotti
ITER - JINTRAC
NF 2019

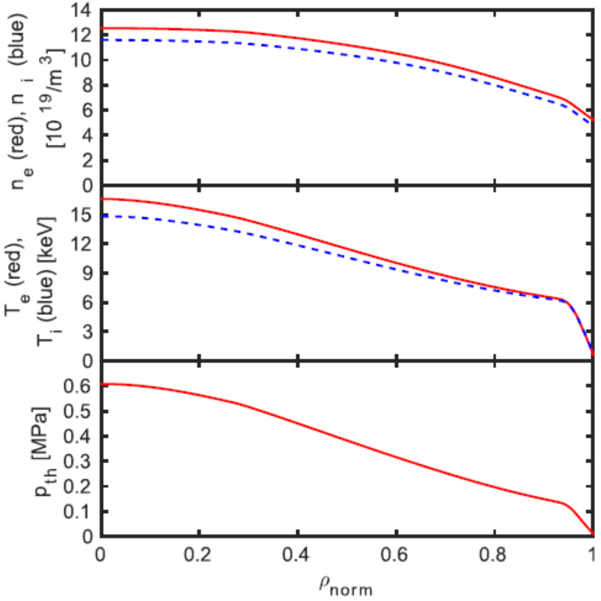


Optimization depends on ITER plasma post-pellet transport mechanisms

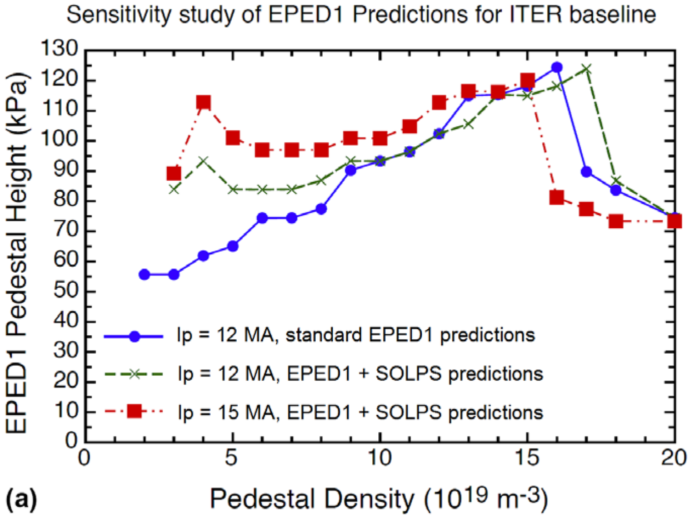
Edge plasma conditions in high Q plasmas

- **High n_{sep} for q_{div} control and moderate grad-n$|_{ped}$ can have large impact on edge MHD stability (lower j_{ped} for same grad-p$|_{edge}$)**
- **Combined with effects on edge transport may lead to new pedestal behaviour in ITER high Q plasmas**

F. Koechl
JINTRAC
NF 2020

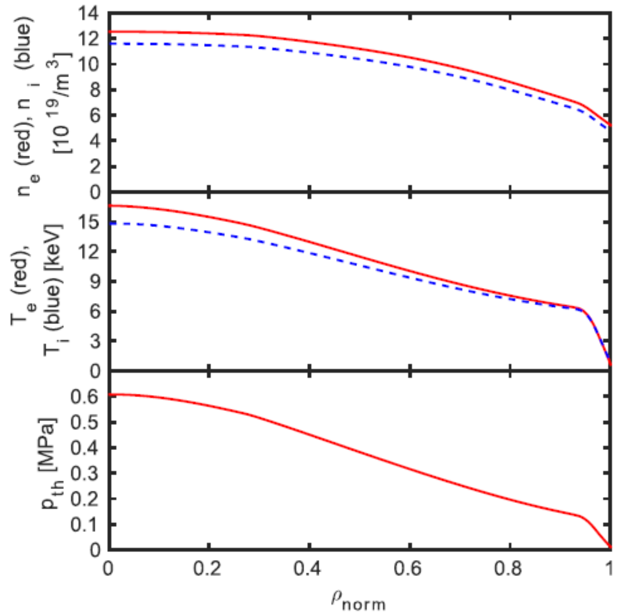


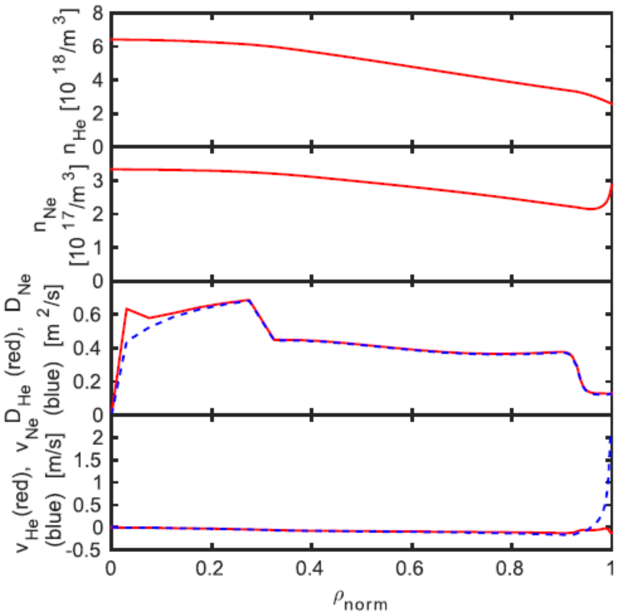
A. Polevoi
ITER - NF 2015



Edge impurity behaviour

- Shallow density gradients in pedestal lead to outwards impurity pinch
- Effect increases with Z → better screening for Ne (and W) and but weak for He → favourable effect in ITER high Q plasmas



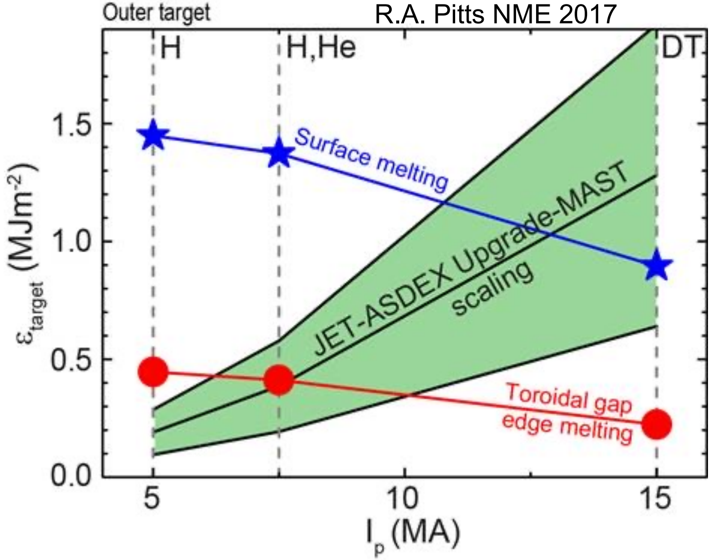


F. Koechl - ITER – JINTRAC - NF 2020

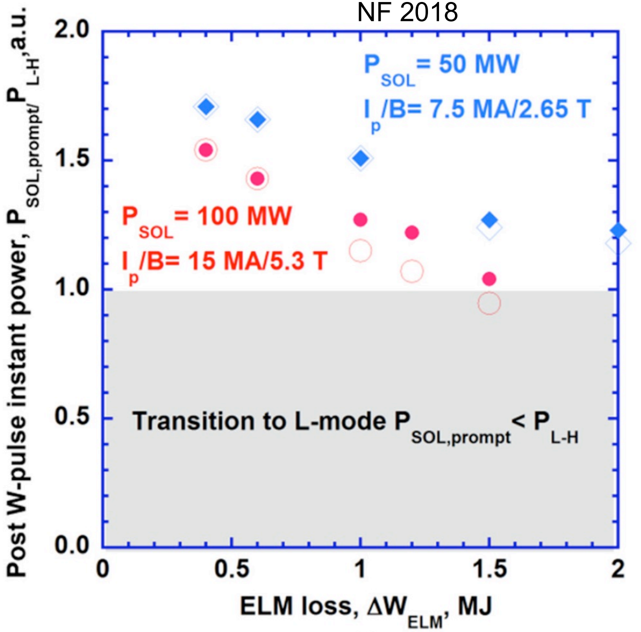
ELM control

- ELM control needed in ITER to avoid divertor melting and W contamination of core plasma
- Operational range with ELMs depends on A_{ELM} and Γ_W by ELMs → mandatory for high Q plasmas

J.P. Gunn NF 2017
R.A. Pitts NME 2017



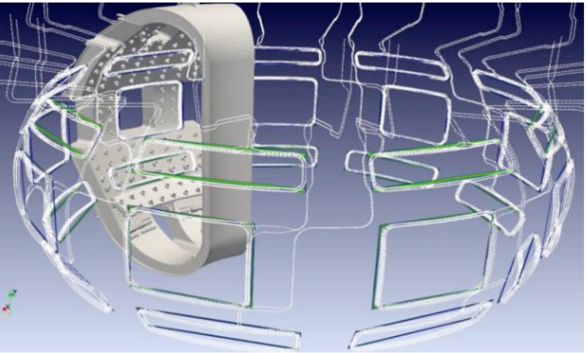
A. Polevoi ASTRA + SOLPS
NF 2018

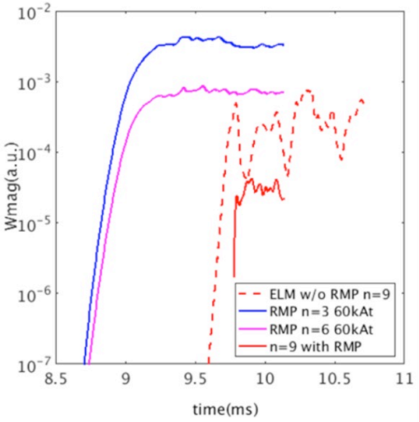


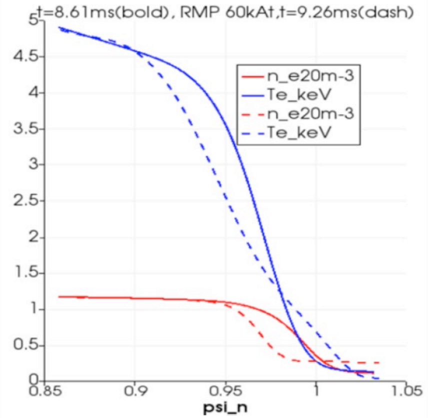
ELM control by 3-D fields

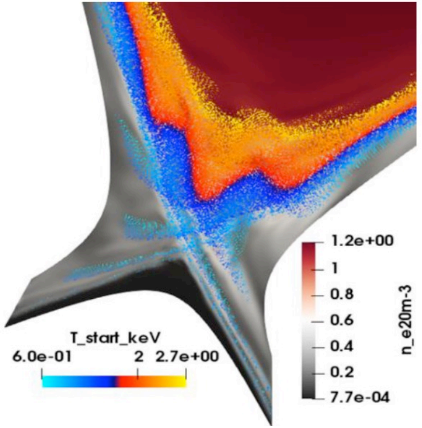
- ELM control by 3-D fields is the main scheme in ITER (27 in-vessel coils)
- Optimum current waveform to achieve ELM suppression and minimize impact on burning plasmas will be demonstrated in ITER

JOREK-Becoulet IAEA 2020(1)





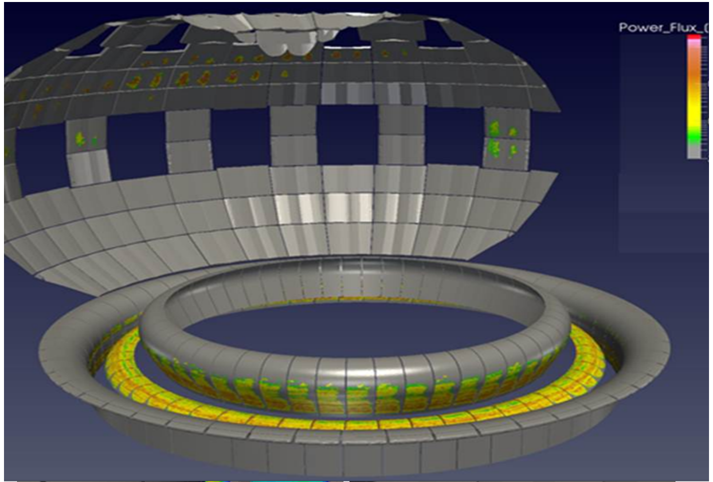




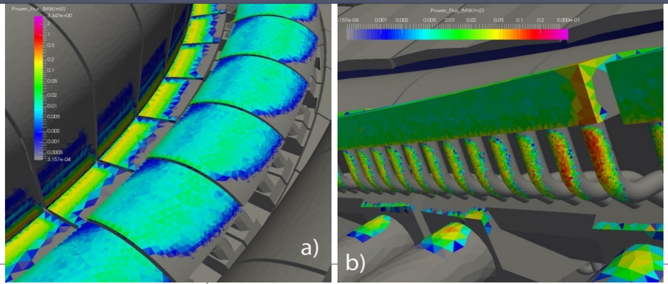
Besides achievement of ELM suppression 3-D fields introduce a range of high Q scenario integration issues

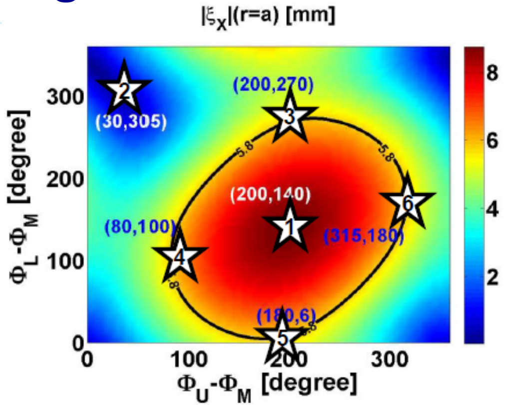
Fast particle loses due to ELM control by 3-D fields

- **3-D fields for ELM control increase fast particle NBI loses due to large edge losses → optimization required for integration with high Q scenario**



ITER-LOCUST
Akers & Ward
IAEA 2016





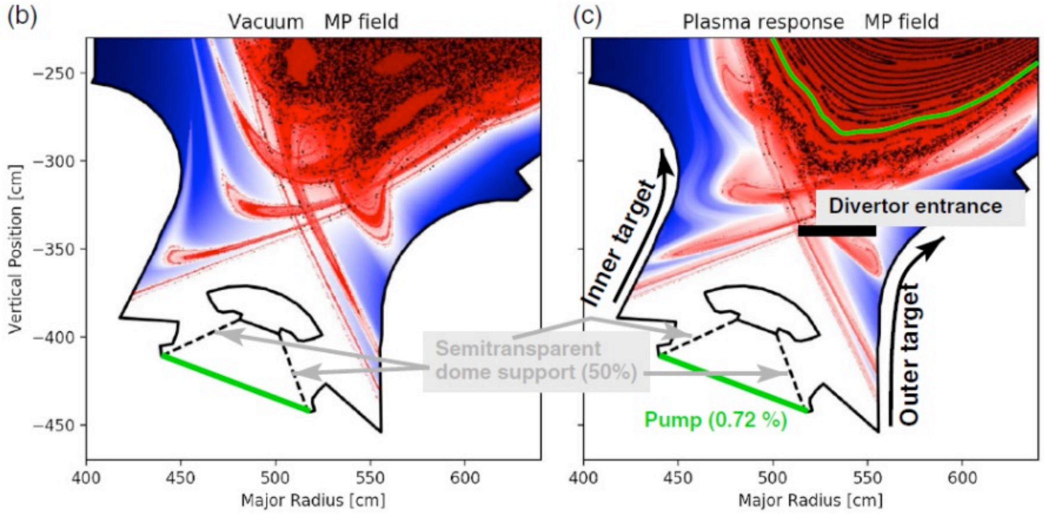
ITER-ASCOT
L. Sanchis
NF 2021

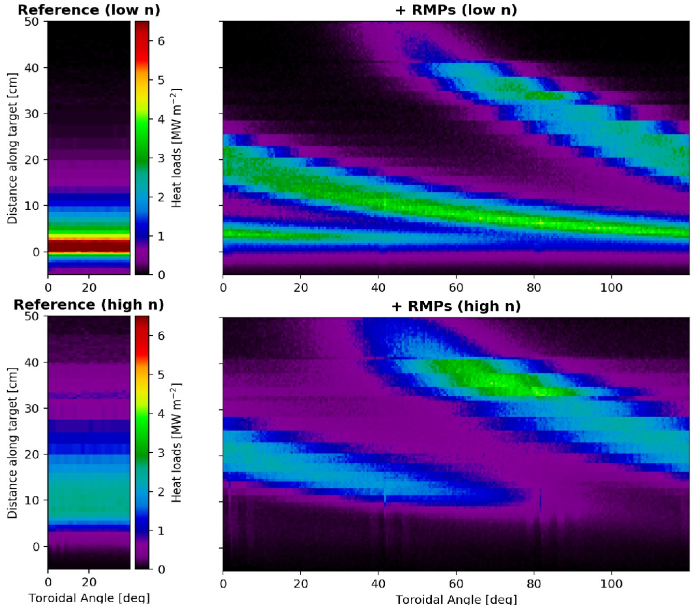
#Case	Max losses (%)
1	13.3
2	0.03
3	8.1
4	8.4
5	9.2
6	12.6

Divertor power fluxes with ELM control by 3-D fields

- Radiative divertor operation with 3-D resonant fields required at high $P_{aux}+P_{\alpha}$ and I_p in ITER
- q_{div} modification by 3-D fields impacts radiative divertor exhaust → ITER high Q will require effective 3-D radiative divertor power exhaust

EMC3-Eirene- H. Frerichs PRL 2020

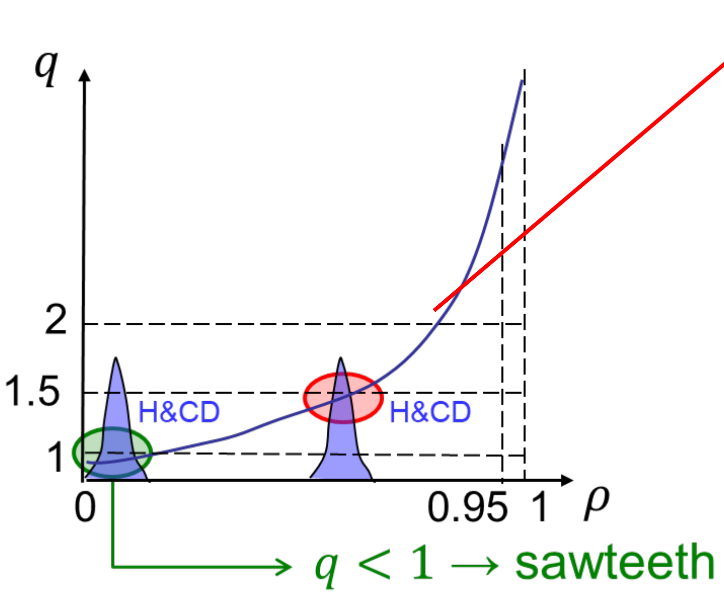


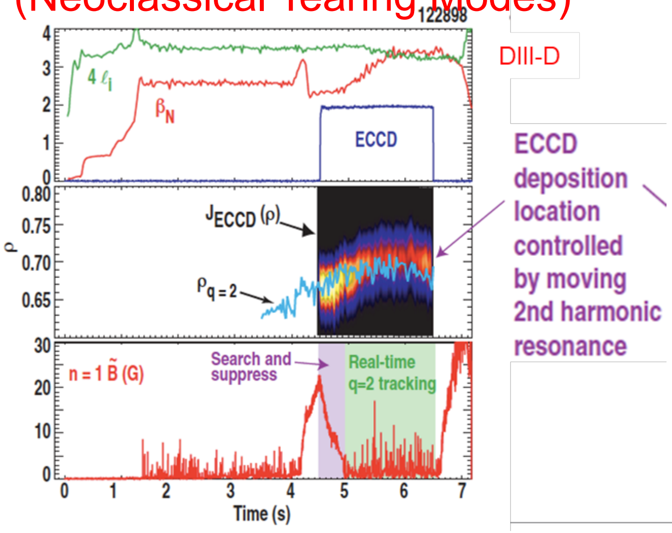


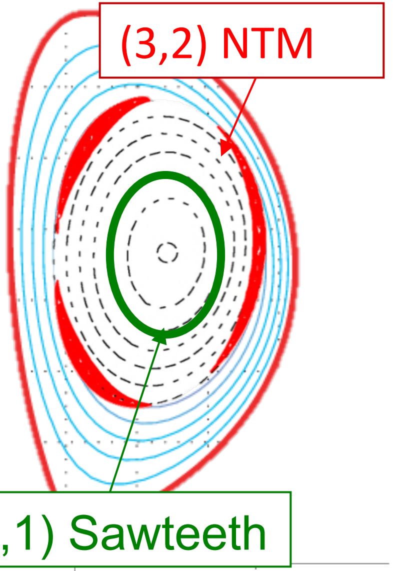
Core MHD Stability control for Q = 10 plasma

- Core MHD instabilities deteriorate plasma confinement and can potentially grow and cause disruptions in ITER → control by H&CD in ITER
- Key issue for ITER is minimizing power for off-axis control → impact on Q

ITER baseline Q=10 magnetic equilibrium



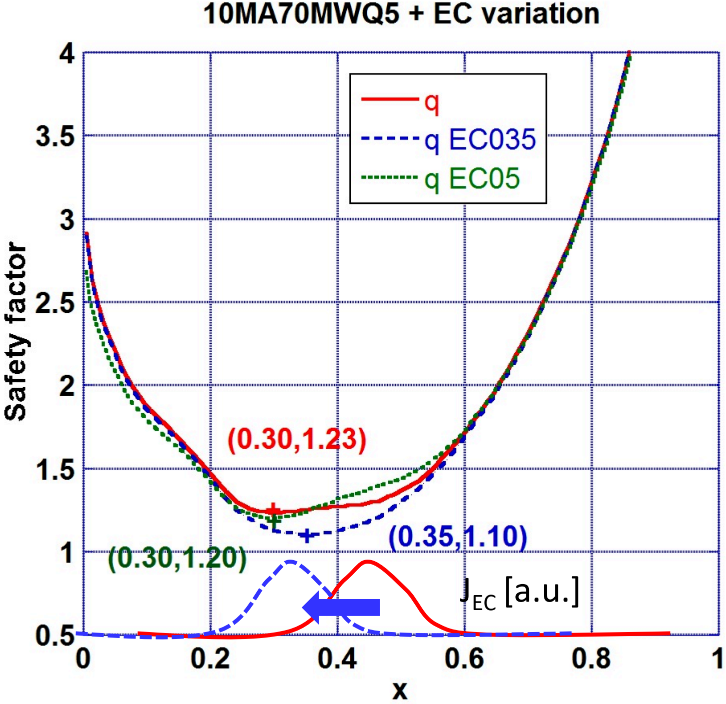


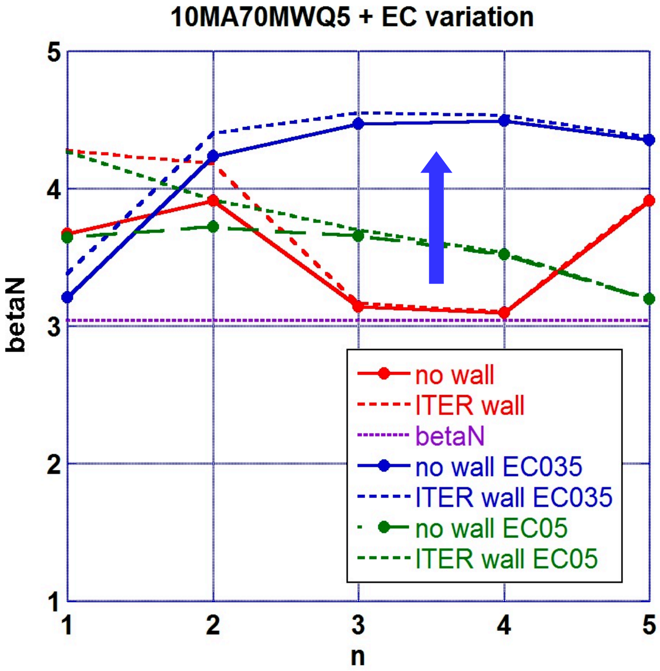


Core MHD Stability control for Q = 5 plasma

- KINX stability analysis shows that low-n (=1-5) ideal MHD modes stability ($\beta_N < \beta_{N,limit}$) can be controlled by varying ECCD location

Polevoi – NF 2020

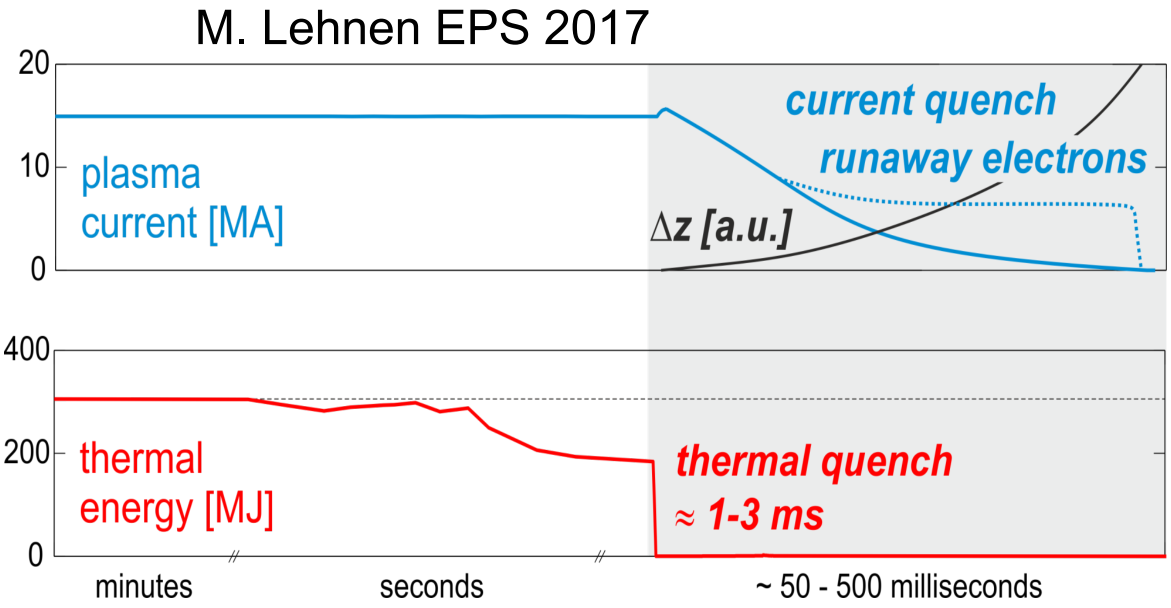


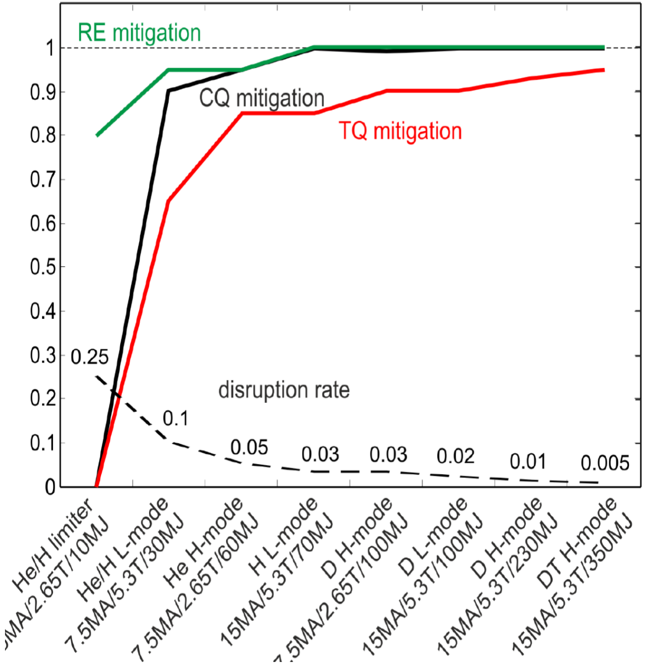


RWM control for disruption avoidance can be provided by ELM control coils

Disruptions

- ITER high Q operation requires very low disruptivity and effective mitigation if disruptions occur → key for tokamak reactors



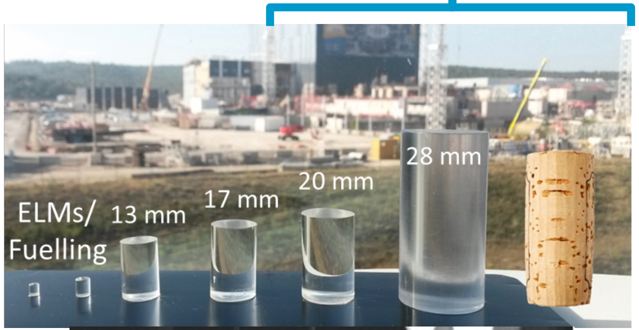


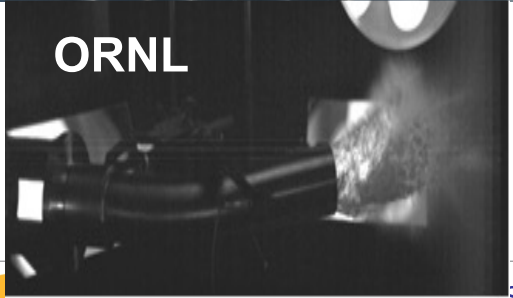
Disruption Mitigation

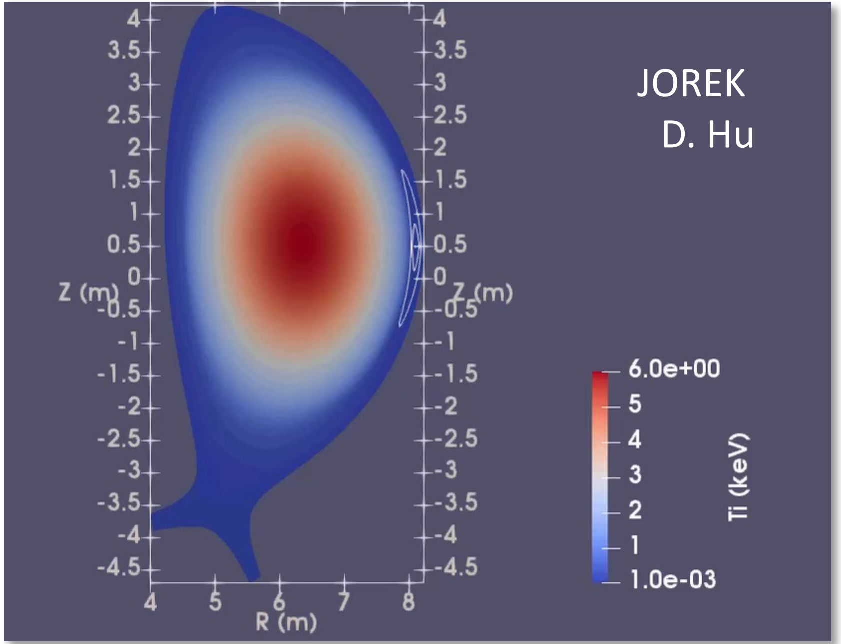
❑ Injection of Shattered Pellets (H_2 and Ne)

- ➢ **Dissipating thermal and magnetic energy → radiation**
- ➢ **Preventing runaway electron formation → increasing plasma density**

Lehnen IAEA FEC 2018

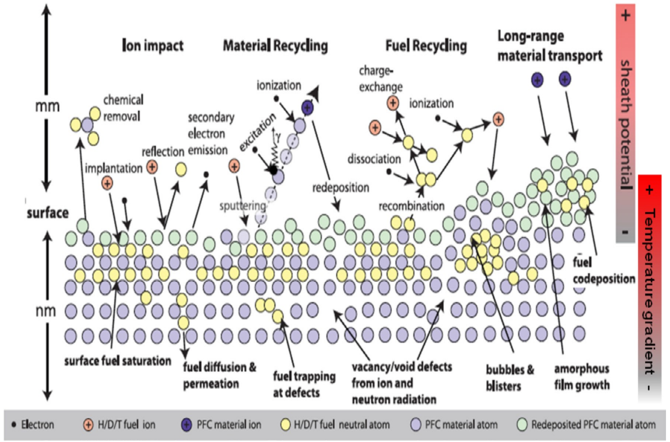


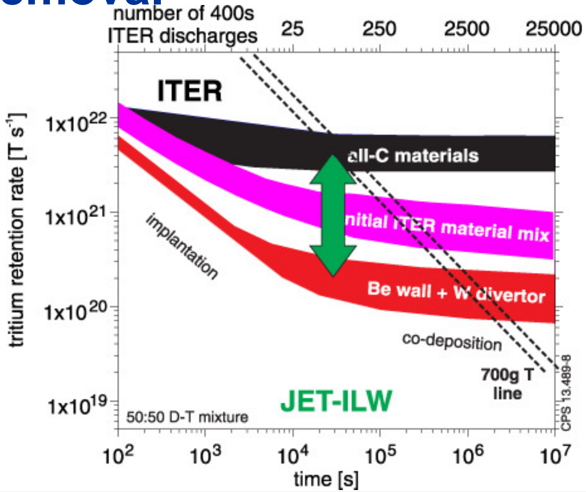




T retention and removal

- Plasma impacting PFCs leads to many processes: recycling, erosion, etc.
- Eroded material can react with plasma ions and trap T semi-permanently
  - Be-wall : Low plasma flux and some T retention (co-deposition)
  - W-divertor : High plasma flux and low T retention (implantation)
- ITER DT operation will determine T retention and investigate strategies to minimize retention and schemes for in-situ removal





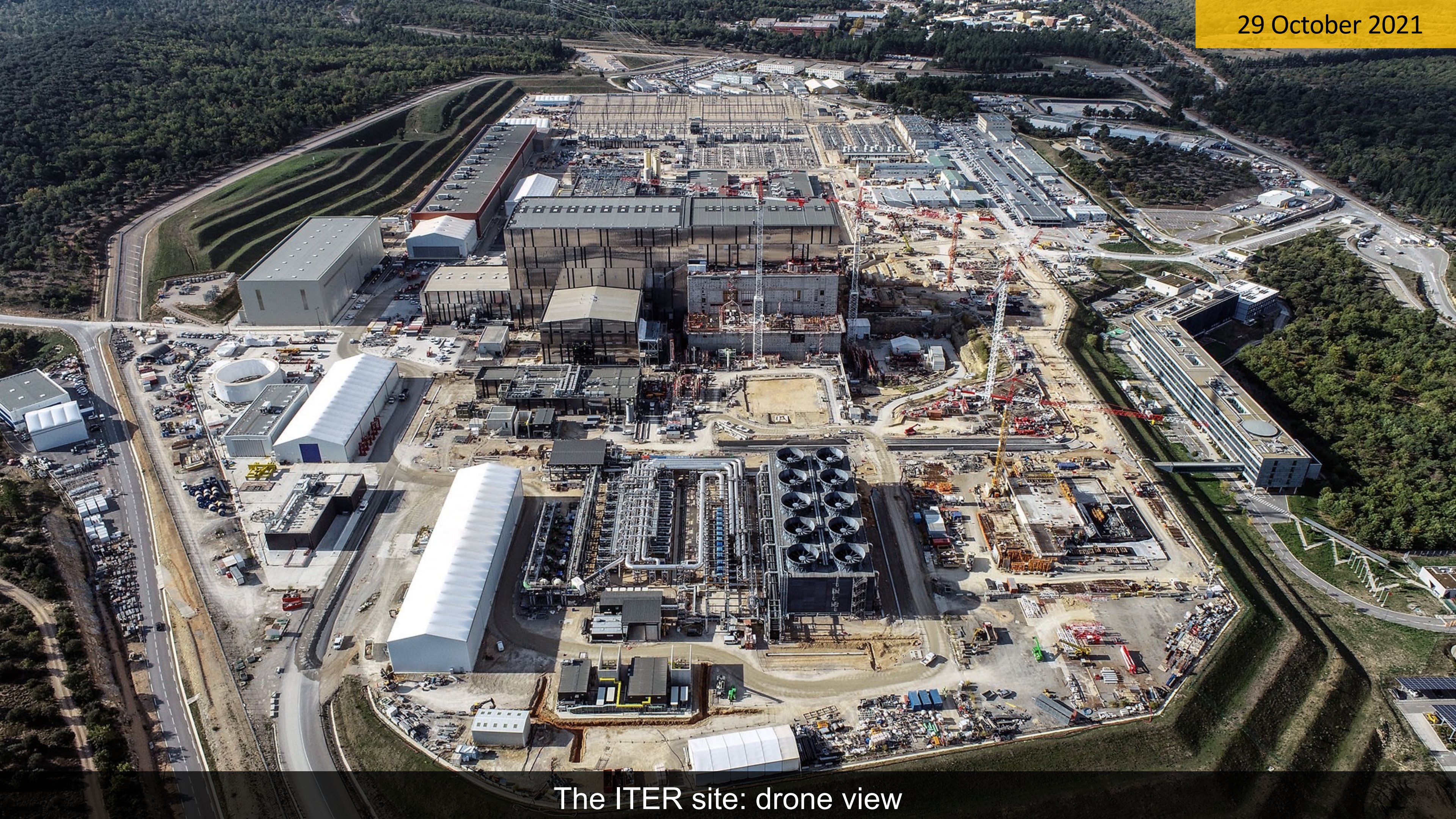
Roth/Schmid – ITER
Brezinsek – JET-NF 2013

Conclusions

- ITER high Q scenarios will allow research on key burning plasma issues for fusion reactor
  - Coupling of physics processes in self-heated plasmas
  - Integration of core-edge physics to achieve burning plasma conditions with acceptable edge plasma conditions
  - Effectiveness of actuators and control schemes for burning plasmas → high Q disruption-free operation
- In addition many fusion reactor technologies will be demonstrated (Tritium cycle, TBMs, H&CD, PFCs, etc.)

Recent news on ITER construction

29 October 2021

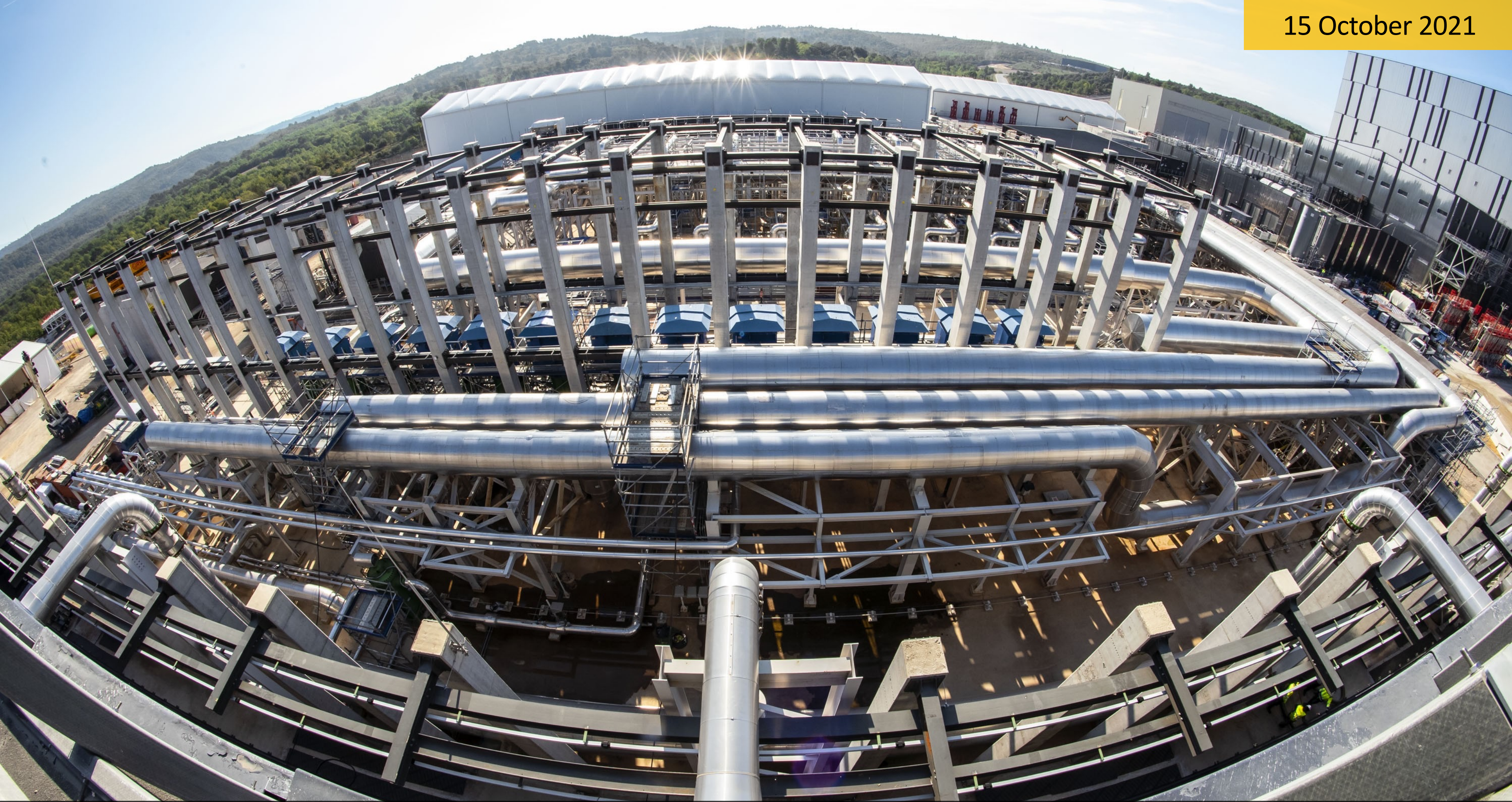


The ITER site: drone view

25 November 2021

Over 5000 t of equipment installed in the cryoplant in 3.5 years → 2 year commissioning period started

15 October 2021

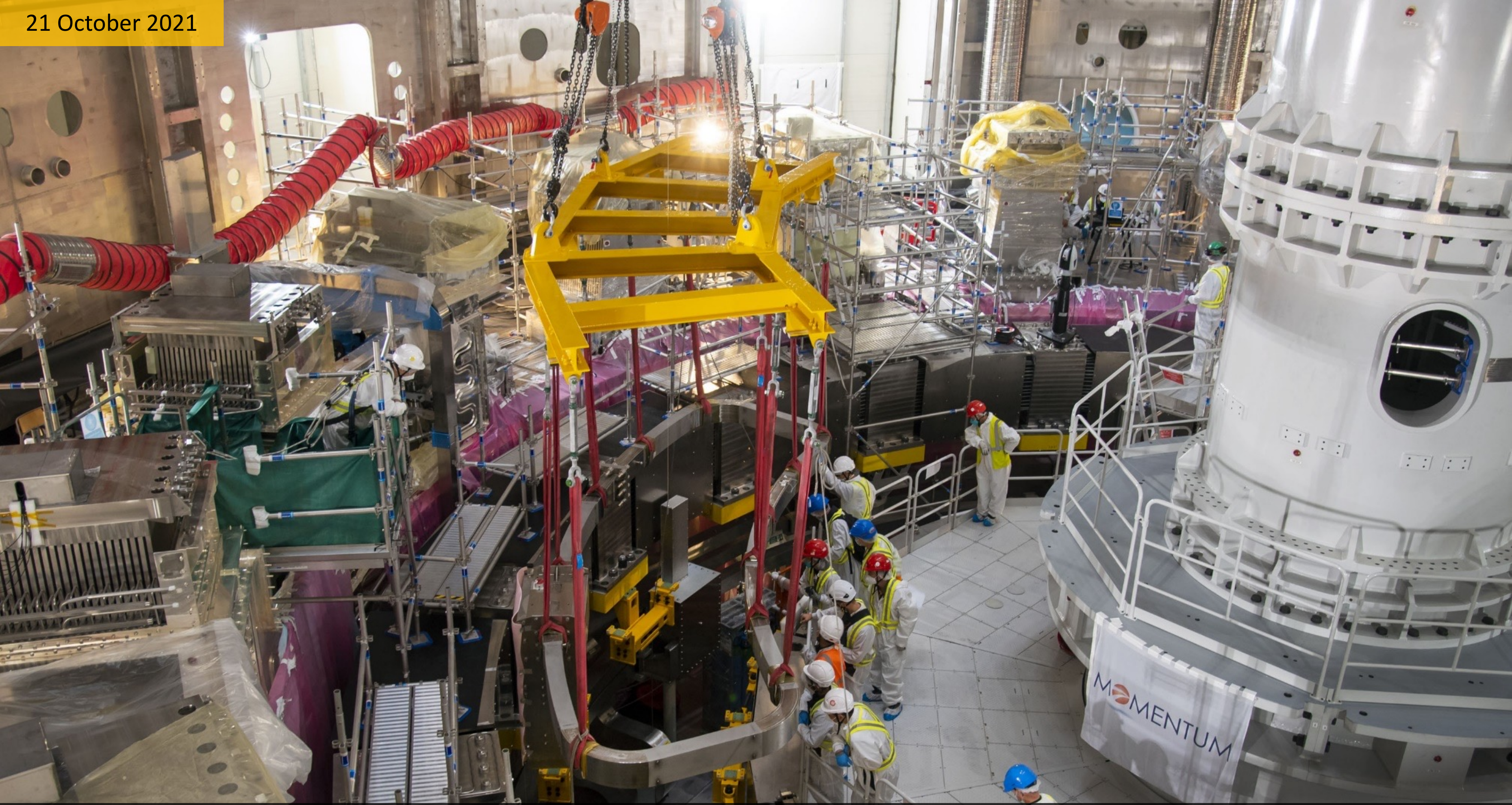


About 6000 t of equipment supplied by India for the cooling plant → plant commissioning now started

8 December 2021

Control Building going up

21 October 2021



Installation of one of the lower (bottom) correction coils (BCC)

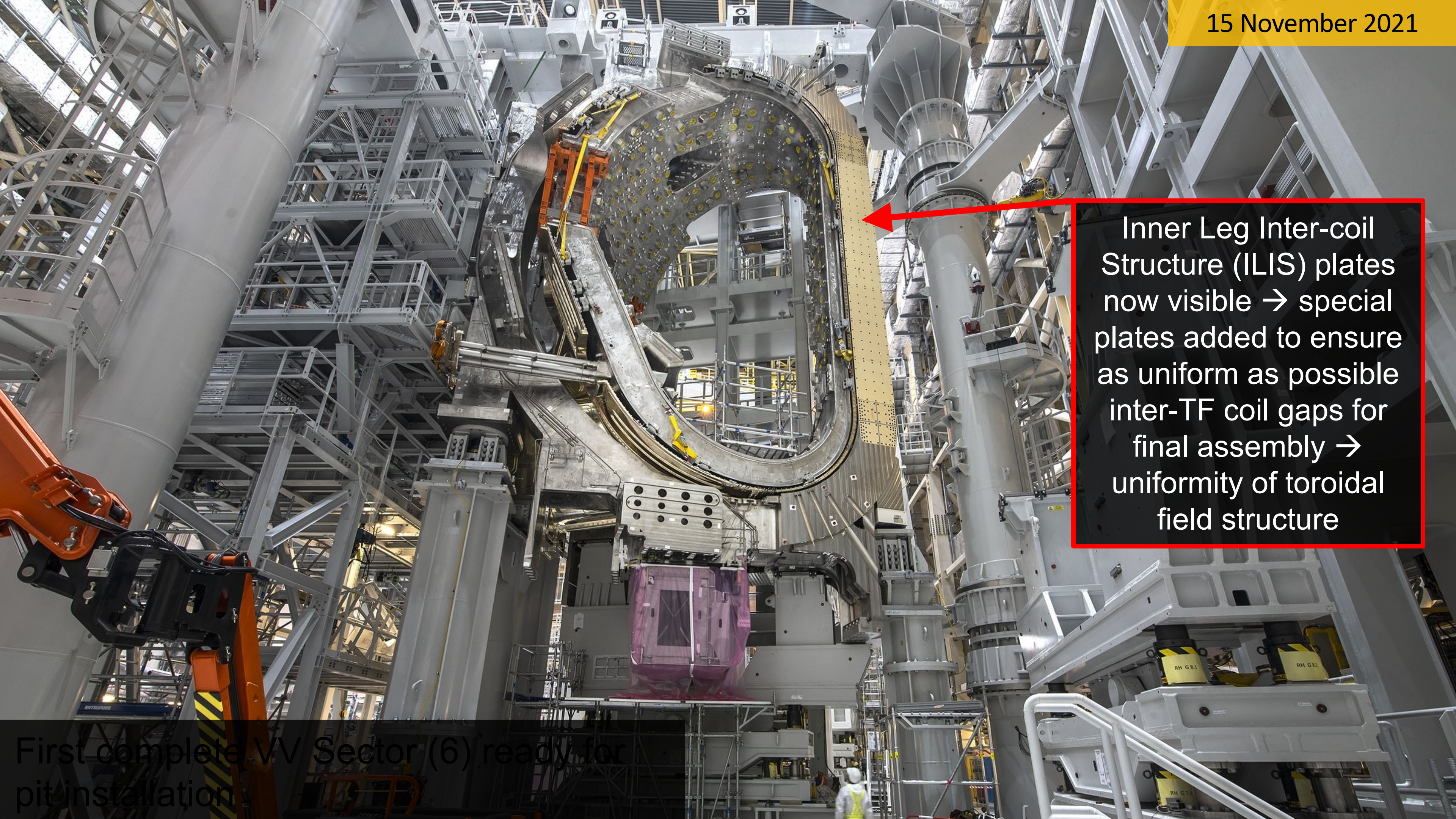
8 December 2021

Metrology and adjustment

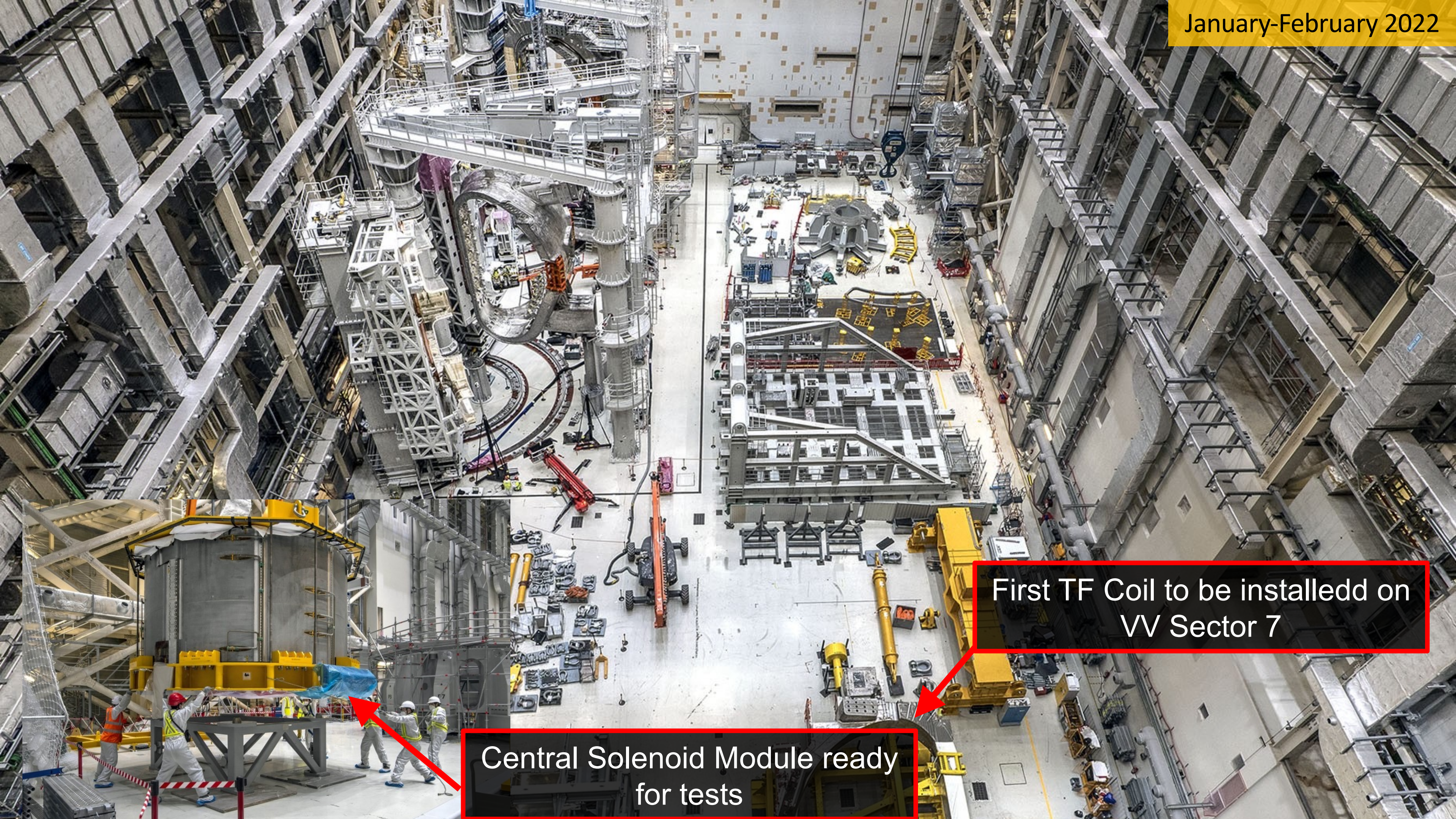
20 December 2021

Tokamak Pit ready for 1st VV sector

15 November 2021



First complete VV Sector (6) ready for pit installation


January-February 2022
First TF Coil to be installedd on VV Sector 7
Central Solenoid Module ready for tests