The ITER Research Plan

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Acknowledgements:

Many colleagues in the ITER Organization, ITPA and the international fusion programme

– Particular thanks to the major fusion facilities for sharing their latest results in the run-up to the IAEA Conference

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

Synopsis

- **ITER mission goals**
- **ITER Research Plan – rationale and structure**
- **Challenges on the way to producing fusion power in ITER:**
  - establishing the plasma scenarios
  - disruptions and disruption mitigation
  - power handling
  - achieving H-mode
  - ELM control
- **Summary of the Research Plan**

ITER Mission Goals

Physics:

- ITER is designed to produce a plasma dominated by α-particle heating
- produce a significant fusion power amplification factor ($Q \geq 10$) in long-pulse operation (300 – 500 s)
- aim to achieve steady-state operation of a tokamak ($Q \geq 5$/ $\leq$ 3000 s)
- retain the possibility of exploring 'controlled ignition' ($Q \geq 30$)

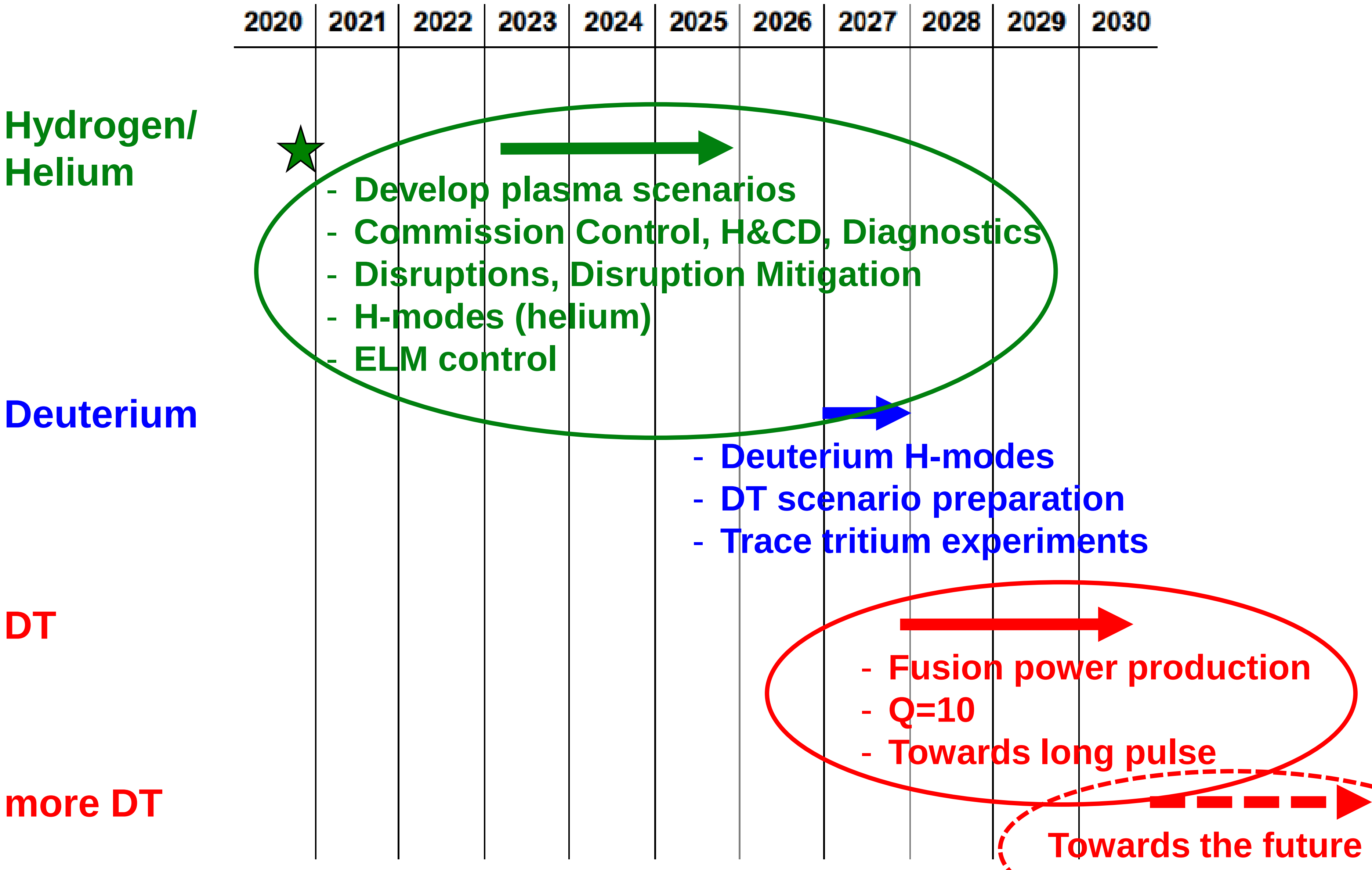
Technology:

- demonstrate integrated operation of technologies for a fusion power plant
- test components required for a fusion power plant
- test concepts for a tritium breeding module

ITER Research Plan – Rationale

- **The ITER Research Plan has been developed to analyze the programme towards high fusion gain DT operation:**
  - allows programme logic to be developed and key operational challenges to be identified and addressed during ITER construction
  - supports planning of installation and upgrade programme accompanying operation
  - provides insight into principal physics risks impacting on experimental programme
    $\Rightarrow$ R&D priorities in current research programmes
  - encourages exploration of issues in burning plasma physics which are likely to be encountered on route to Q = 10 and beyond

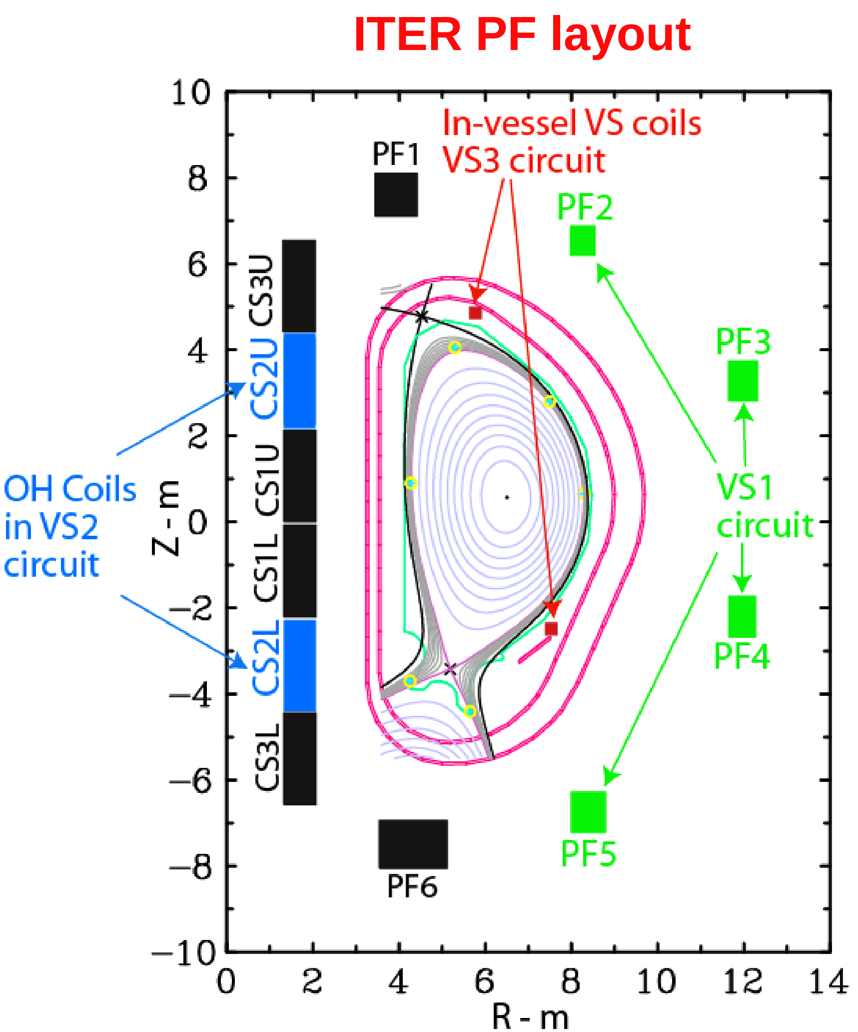
ITER Research Plan – Structure



Risk Assessment ⇒ Key R&D Needs

- **Top 12 risks associated with plasma operation and their potential consequences have been identified; mitigation strategies (and implications) have been developed – top 6 are:**
  - Disruption loads and effectiveness of disruption mitigation
  - Uncertainty in H-mode power threshold scaling
  - Effectiveness of ELM mitigation schemes
  - Vertical stability control limited by excessive noise (or failure of in-vessel coils)
  - Availability of reliable high power heating during non-active phase of programme (⇒ H-mode access)
  - Acceptable "divertor" performance with tungsten PFCs over required range of plasma parameters

Establishing the Plasma Scenarios



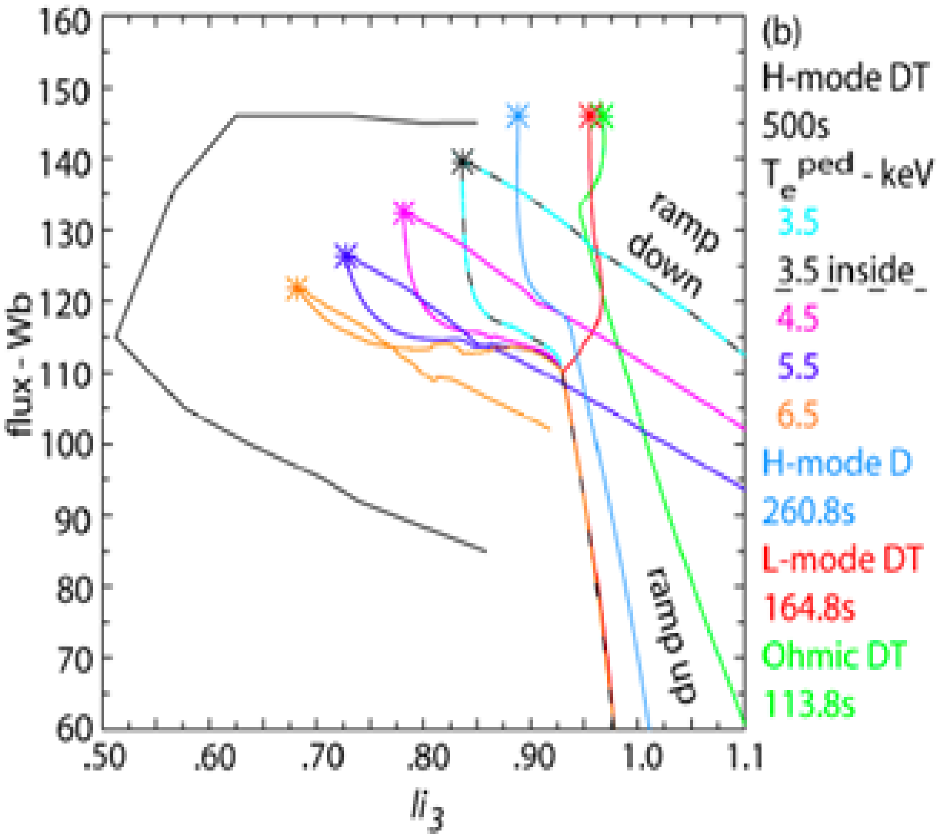
- **In ITER, care must be taken in developing scenario:**
  - avoid coil current saturation
  - minimize flux consumption during current ramp-up
  - maintain plasma position control during transients
  - maintain vertical stability during current ramp-down

NB: very long pulses require particular care to avoid drifts in magnetic diagnostic signals

Flux Consumption in ELMy H-mode

- **Optimization of magnetic flux consumption is key issue for long-pulse operation in ITER:**
  - several limits must be respected in scenario development:
    - PF/CS coil current and field limits
    - saturation of PF6 (“divertor”) coil at low values of l_i
    - consumption of excessive magnetic flux during ramp-up at high l_i
    - Central Solenoid force limits
  - a wide range of scenarios has now been developed for 15MA operation in non-active and DT phases of operation, allowing up to 500 s burn duration

T A Casper, IAEA 2010

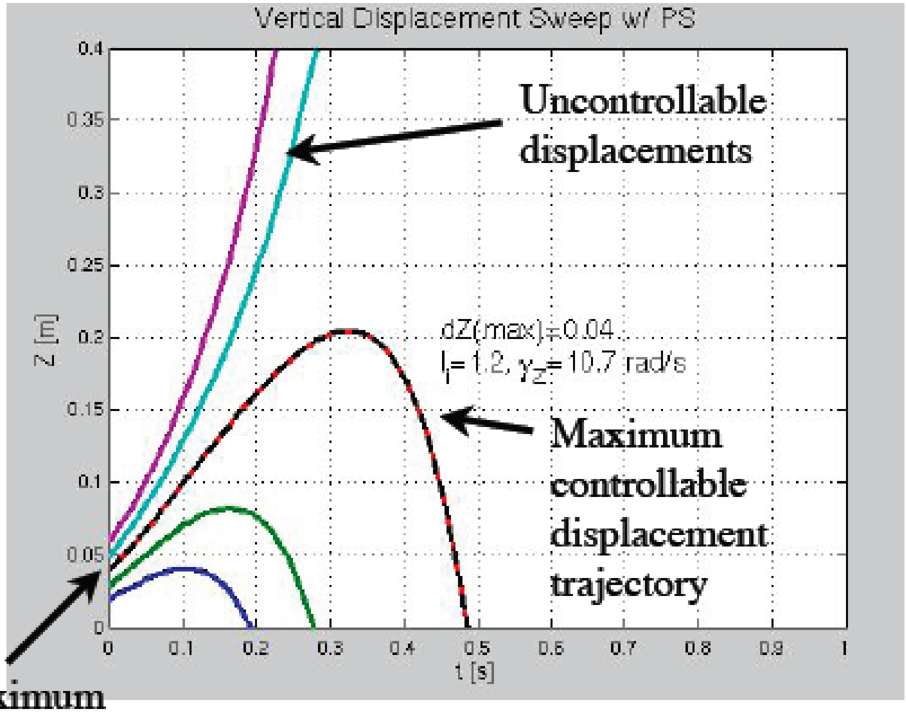


Vertical Stabilization Performance

- **Performance of VS system characterized by ΔZ_{max}**
  - maximum controllable "instantaneous" vertical displacement
- **Experiments suggest that:**
  - $\Delta Z_{max}/a > 5\%$ is "reliable"
  - $\Delta Z_{max}/a > 10\%$ is "robust"
- **For "worst case" conditions ($l_i(3) = 1.2$), original ITER system:**
  - $\Delta Z_{max}/a = 2\%$
  - large overshoot in ΔZ due to vessel time constant

D Humphreys et al, IAEA-FEC2008, IT-2-4b
A Portone et al, IAEA-FEC2008, IT-2-4a

Example of Analysis and Gedanken Experiment to Calculate ΔZ_{max}



Maximum controllable displacement ΔZ_{max}=0.04 m

⇒ Internal coils for vertical stabilization to meet requirements

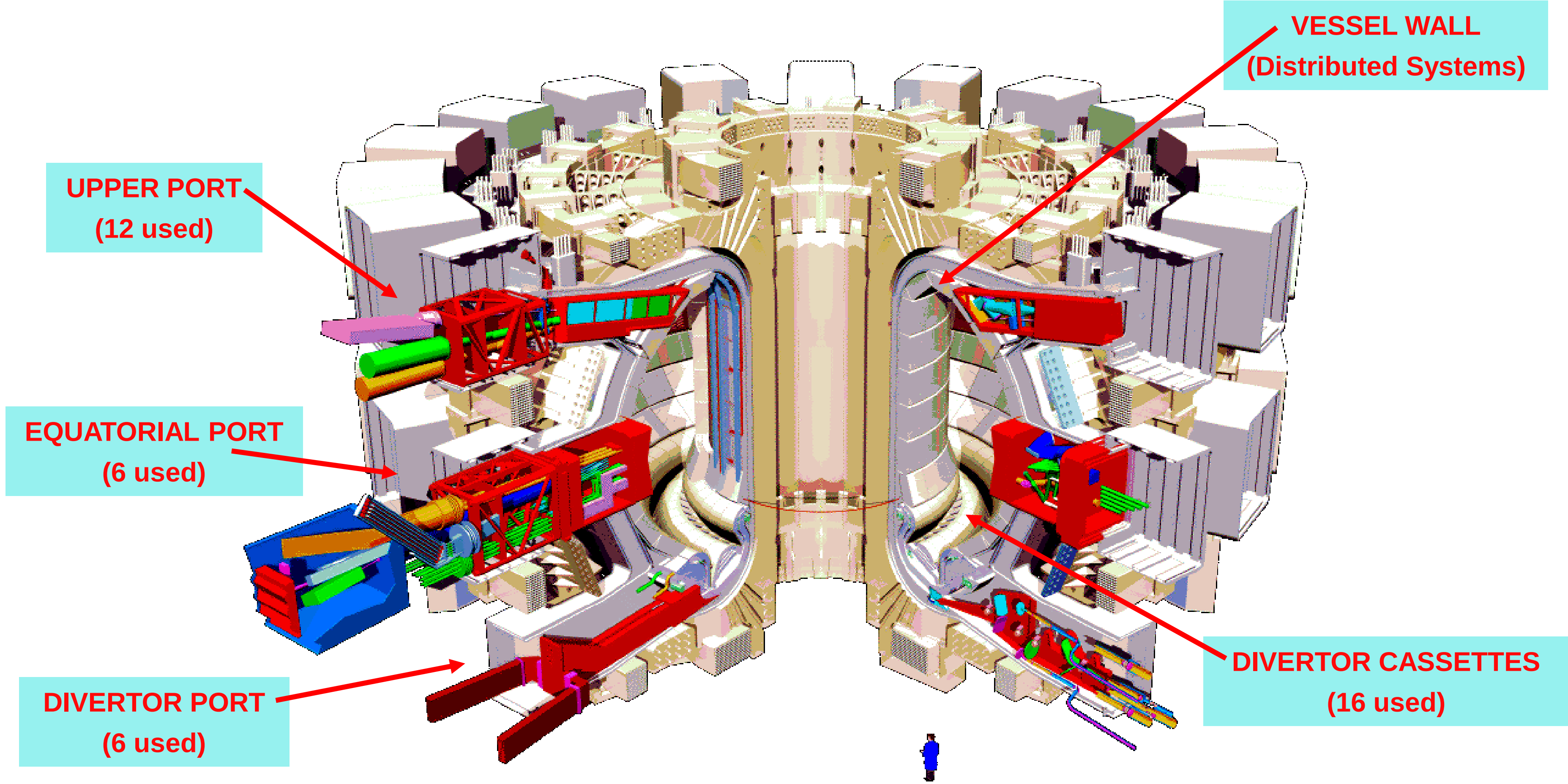
Heating and Current Drive

ITER is equipped with a flexible H&CD system with extensive functionality

Heating System	Stage 1	Possible Upgrade	Characteristics
NNBI (1 MeV D^0) (870 keV H^0)	33	16.5	Vertically steerable for CD
ECH&CD (170 GHz)	20	20	Equatorial and upper port launchers with steerable mirrors
ICH&CD (40 - 55 MHz)	20*	20	$2\Omega_T$ or Ω_{He3} (H minority at 2.65 T)
LHCD (5 GHz)	0	40	$1.8 < n_{par} < 2.2$ off-axis CD
Total	73 →	130	(110 simultaneously)

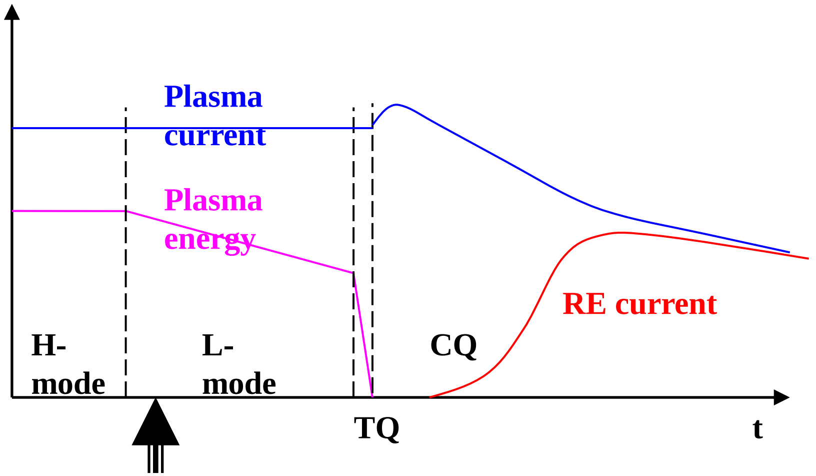
* 10 MW available in non-active phase – only one ICRF antenna installed

Analyzing the Plasma - ITER Diagnostics



- **About 40 large scale diagnostic systems are foreseen:**
  - Diagnostics required for protection, control and physics studies
  - Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  - Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ….)

Disruptions, VDEs, Runaway Electrons



Typical chain of events during plasma disruption

Disruption/ VDE/ RE mitigation is essential for reliable operation of ITER

$\Rightarrow$ Massive material injection (MMI) is the most likely solution

- **Most serious thermal loads occur during Thermal Quench**

  $\Rightarrow$ Need to reduce by factor of at least 10 to limit impact on PFCs

- **Major mechanical forces act on VV and PFCs during Current Quench $\Rightarrow$ eddy currents, "halo" currents**

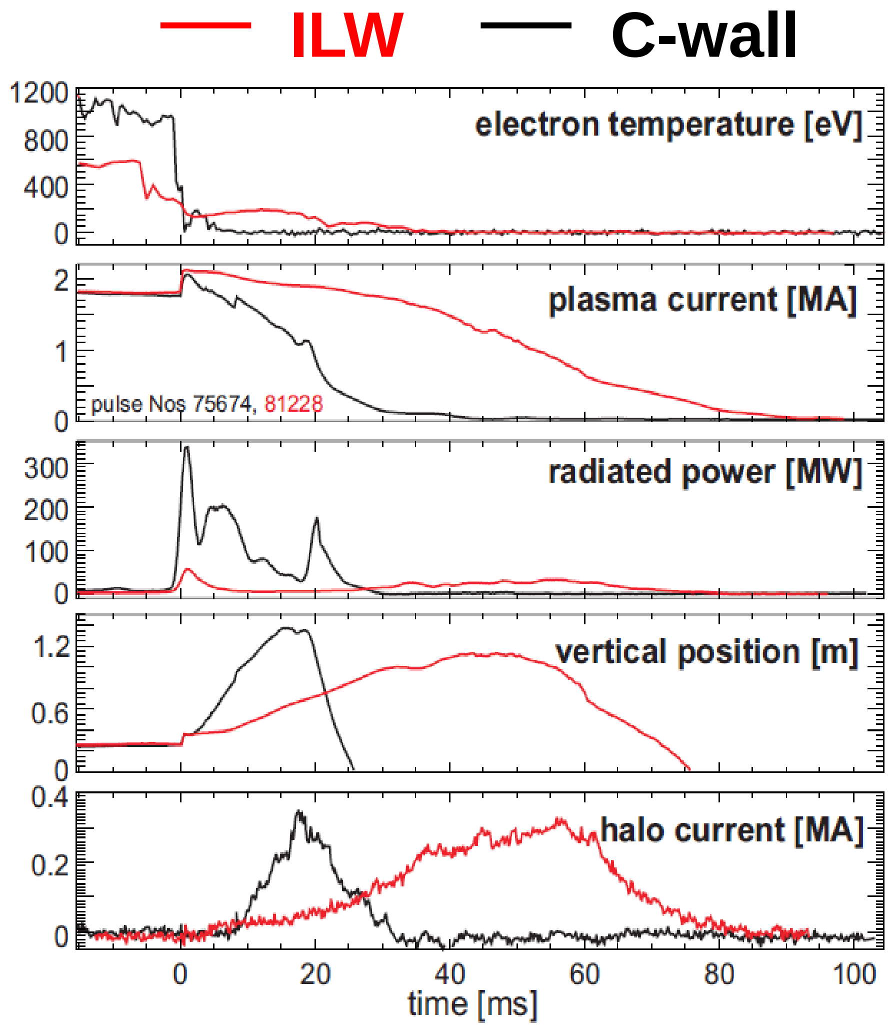
  $\Rightarrow$ Need to reduce by factor of at 2-3 to improve load margins

- **Runaway electrons can be generated during Current Quench**

  $\Rightarrow$ Need to reduce <u>intensity</u> and <u>energy</u> factor of at least 10

Disruption/ Mitigation

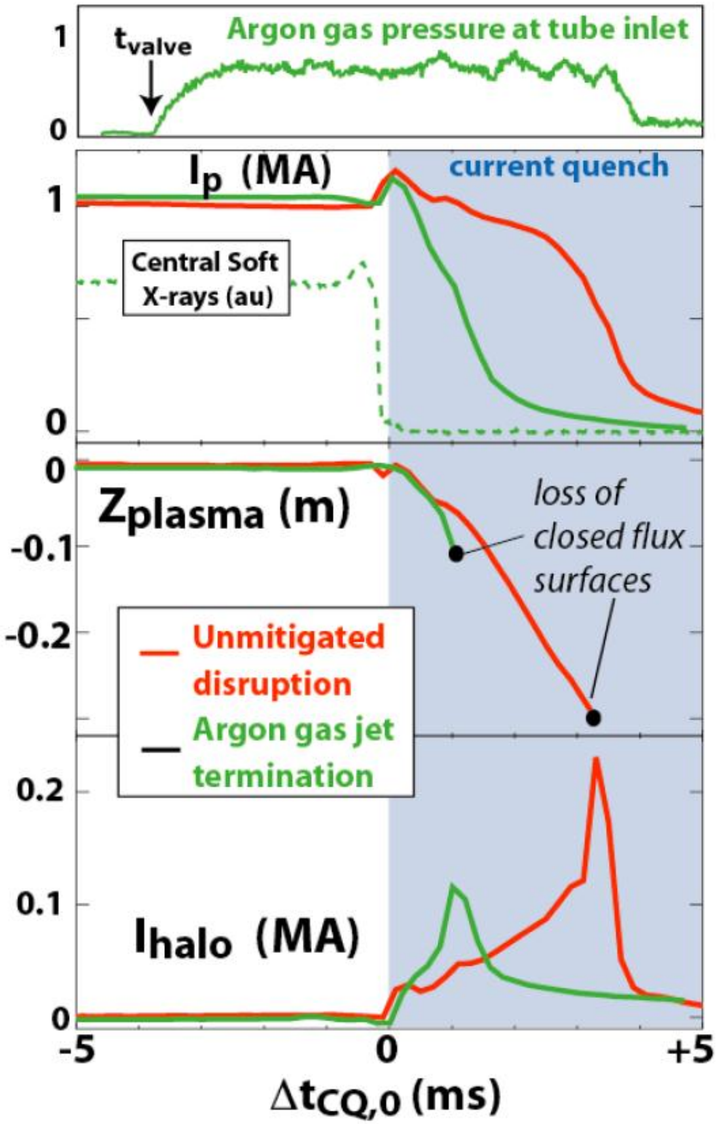
- **Well recognized issue for ITER with all-metal walls (W_{th}, W_{mag} >> than current devices):**
  - JET ILW clearly demonstrated expected low radiation in unmitigated TQ and CQ (cf. C walls)
  - hotter CQ plasma, slower current decay, slower vertical displacement, longer halo current phase
  - energy dissipation through convection/conduction dominates
  - longer time to transfer W_{mag} to CQ plasma → higher thermal loads
  - stresses on VV increased due to longer impact time of forces



M Lehnen, IAEA 2012

Disruption/ VDE Mitigation

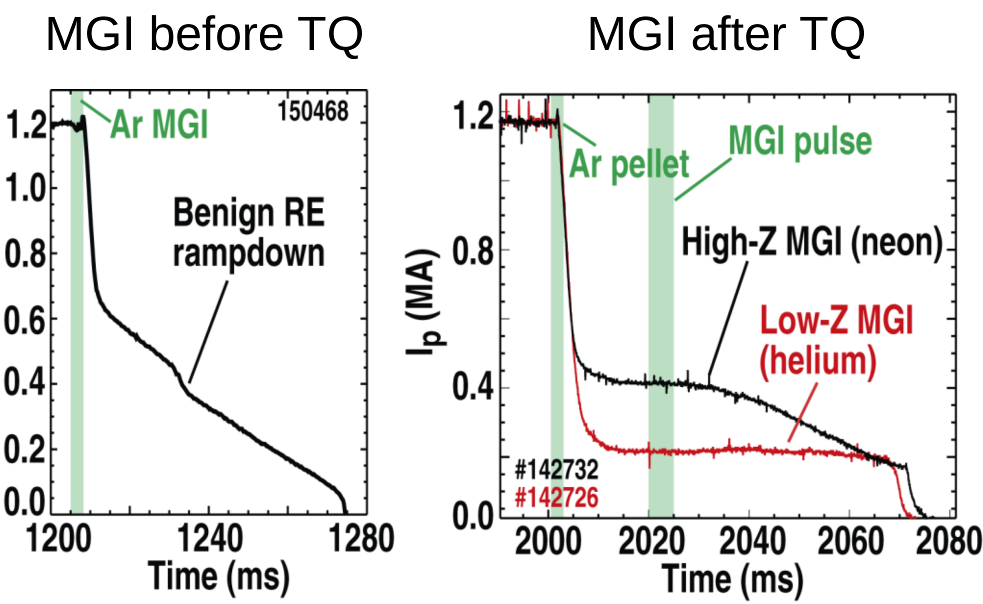
D Whyte, PSI-2006



- **The development of** **high pressure impurity gas injection** **looks very promising for disruption/ VDE mitigation:**
  - efficient radiative redistribution of the plasma energy - reduced heat loads
  - reduction of plasma energy and current before VDE can occur
  - substantial reduction in halo currents (~50%) and toroidal asymmetries

Techniques required for RE Mitigation

- **Suppression of post-disruption runaway electrons is perhaps most challenging aspect of disruption mitigation:**
  - basic principle involves MMI to deconfine or decelerate REs
- **Recent progress in RE suppression:**
  - excellent new experiments on DIII-D: radial stabilization of RE beam then decelerate it with MGI
  - effectively suppressed on KSTAR with D_2 MGI but only below B_T ~3 T
  - not seen at all yet on JET in the ILW!
- **More work required in general on RE suppression and on disruption avoidance, prediction and mitigation efficiency**



E Hollmann, IAEA 2012

Overall strategy for Disruption Mitigation/ Avoidance

Disruption mitigation in ITER involves a multi-facetted approach:

- **Disruption detection and avoidance to ensure identification of approaching disruption with high success rate:**
  - Plasma Control System can trigger “rapid shutdown|” if time permits
  - alternatively, PCS triggers interlock system to fire DMS

- **DMS subsystem for thermal quench mitigation:**
  - mitigates thermal loads and EM loads of disruptions/VDEs
  - injected from 3 Upper and 1 Equatorial Port
  - high pressure gas, shattered pellets, or solid pellets are candidates
  - Ne, Ar, or D_2/He at up to 2 kPa.m^3; 0.5 – 2.5 g of solid/ dust material

- **DMS subsystem for RE suppression/ mitigation**
  - may involve both control of RE beam and MMI to provoke either deconfinement or deceleration
  - multiple injectors from single Equatorial Port
  - Ne, Ar, or D_2/He at up to 2 kPa.m^3

Overall Performance for DMS Subsystems

- **Each element of DMS must achieve high reliability during non-active phase of operation**
- **Reliability figures based on analysis of targets for PFC lifetime**
- **Substantial R&D needed to approach these reliability requirements**

	Energy load on divertor target	Energy load on first wall (VDEs)	EM load due to halo currents (VDEs)	Runaway electrons
Disruption rate (Avoidance)	$\leq$ 5 %	$\leq$ 1-2 %	$\leq$ 1-2 %	<< 1 %
Prediction success	$\geq$ 95 %	$\geq$ 98 %	$\geq$ 98 %	~ 100 %
Mitigation performance	$\leq$ 1/10	$\leq$ 1/10	$\leq$ 1/2	$\leq$ 2 MA

M Sugihara, IAEA 2012

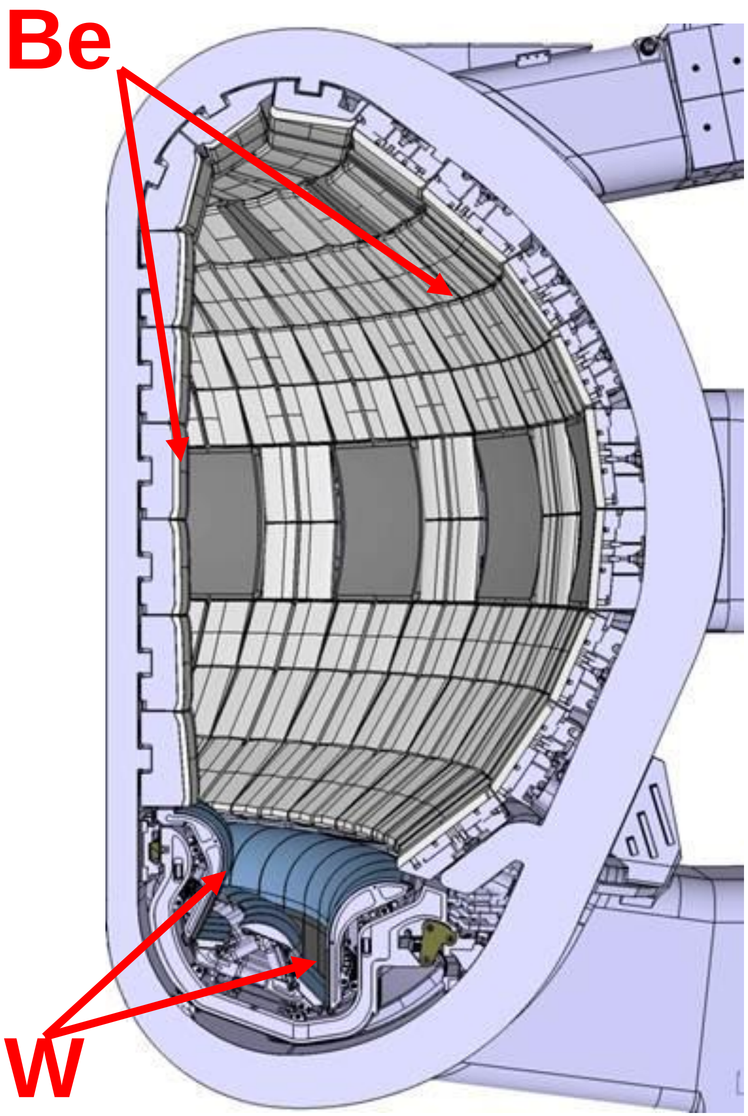
(DT burning phase)

- **DMS must also incorporate flexibility to allow for learning and tuning during non-active phase of operation**

ITER Plasma Facing Components

For DT phase, ITER will operate with all metal PFCs – also in working basis for initial plasma operation

- **Be first wall (~700m²):**
  - low-Z limits plasma impurity contamination
  - low melting point
  - erosion/ redeposition will dominate fuel retention
  - melting during disruptions/ VDEs
  - dust production
- **W divertor (~150m²):**
  - resistant to sputtering
  - limits fuel retention (but note Be)
  - melting at ELMs, disruptions, VDEs
  - W concentration in core must be held below ~ 2.5×10^{-5}



Power and Particle Exhaust

Stationary power handling:

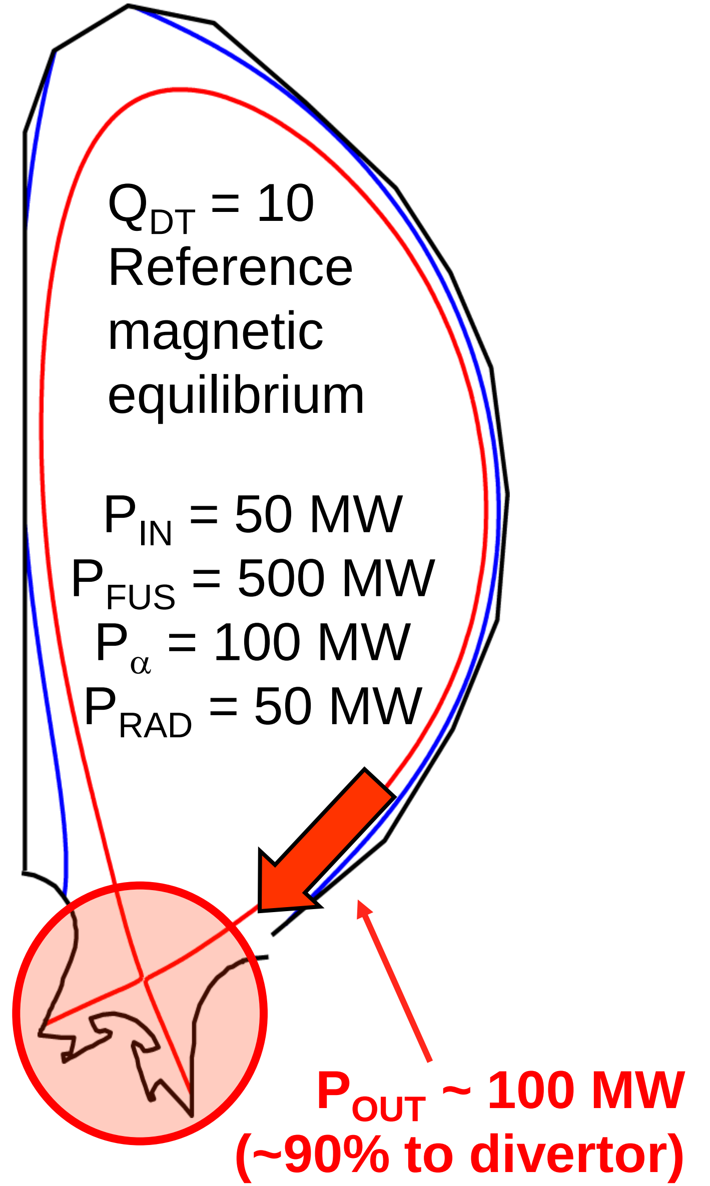
- **Must limit power flux density to (steady-state) engineering limit for plasma facing surfaces of 10 MWm^{-2}:**

- but λ_q may be very narrow
- extract helium from core plasma to limit concentration to below ~6%
- prevent impurities from walls penetrating to plasma core
- ensure adequate PFC lifetime

⇒ **use injected impurities to radiate a sufficiently large fraction of the exhaust power – radiative divertor/ partial detachment**

⇒ **should be effective even with narrow scrape-off layer**

⇒ **but must limit core impurity contamination !**

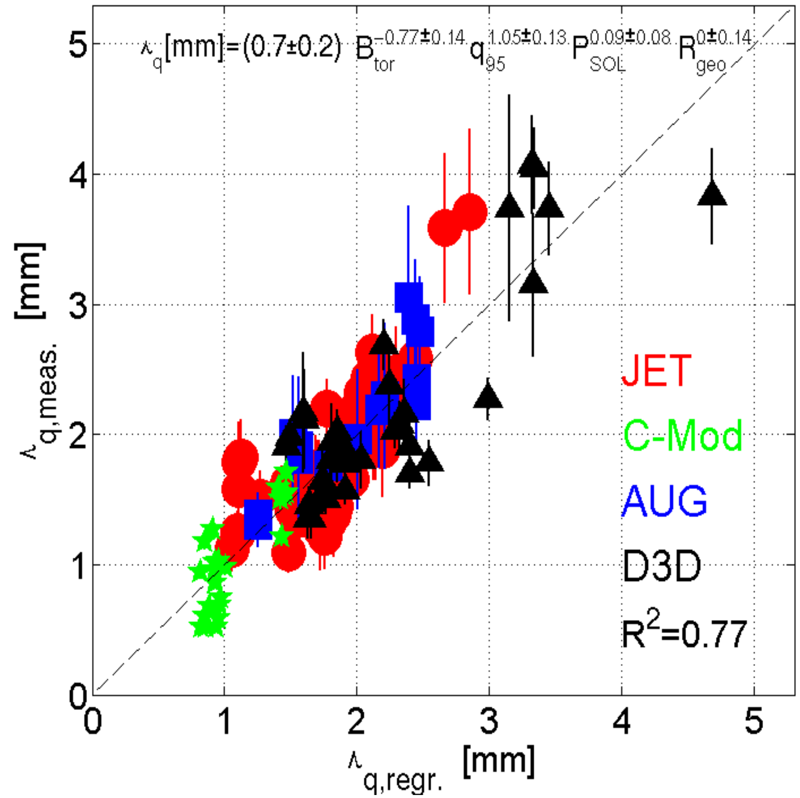


Power and Particle Exhaust

Heat Flux Width:

$$\lambda_q(mm) = (0.7 \pm 0.2) \times B_{tor}^{-0.8\pm0.1} \times q_{95}^{1.05\pm0.2} \times P_{SOL}^{0.1\pm0.1} \times R_{geo}^{0\pm0.1}$$

- **Example of how improved research tools (new high time and space resolution IR cameras) can reveal unexpected (not always favourable) new findings:**
  - width of near SOL channel for parallel heat flow appears to be much narrower than we thought
  - good example of how ITPA has rallied to assist
  - looks possible for ITER to live with it (strong divertor dissipation), but may require dual radiation feedback control (see next).
  - community still debating if narrow width compatible with pedestal stability



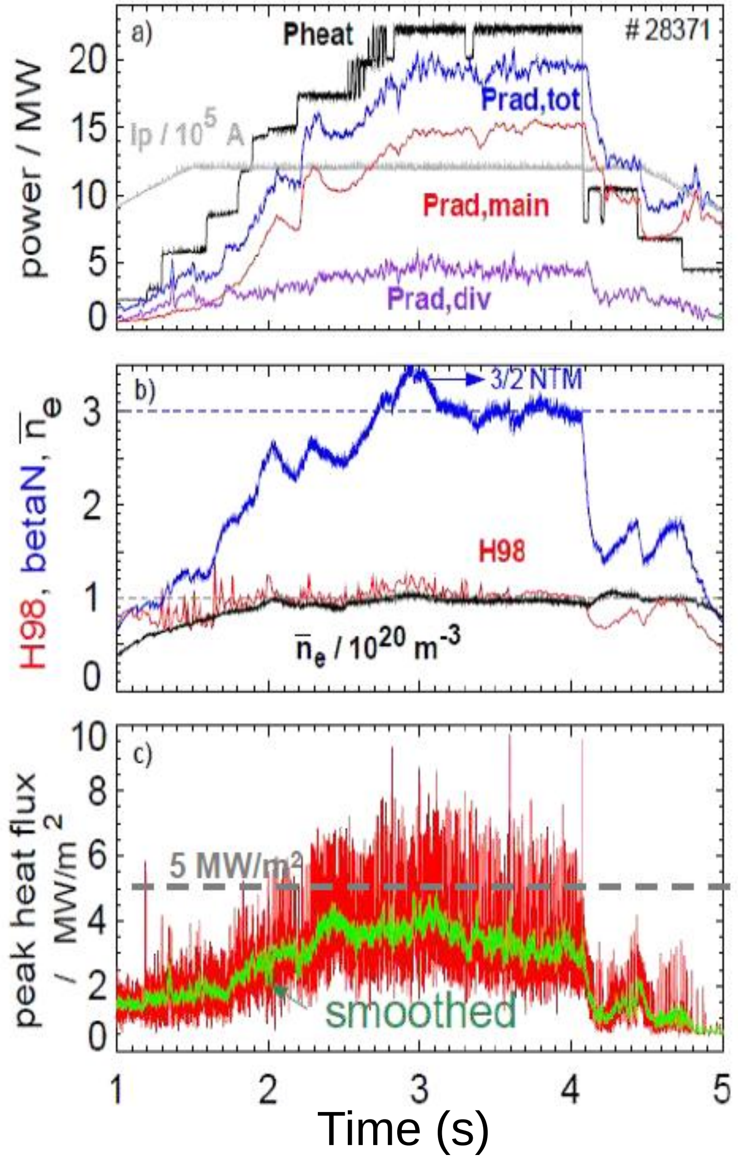
T. Eich et al., IAEA 2012

Power and Particle Exhaust

A Kallenbach et al, IAEA 2012

Integrated Power Flux Control:

- **High power operation in ITER on actively cooled metal PFCs will require robust, reliable heat load control (ELMs and stationary loads)**
  $\Rightarrow$ especially with narrow λ_q
  (lower margins for reattachment)
  - almost certainly needs simultaneous edge and divertor seeding (e.g. Ar (edge), Ne or N_2 (divertor))
  - simplest possible diagnostic signals for reliability (e.g. bolometer chords for radiation control in combination with hotspot detection)
  - maintain high confinement → but has to be compatible with $P_{L\text{-}H}$
  - now demonstrated on AUG with Ar+N_2
  - NB: would need to be combined with ELM control on ITER (ELMs not an issue on AUG)!



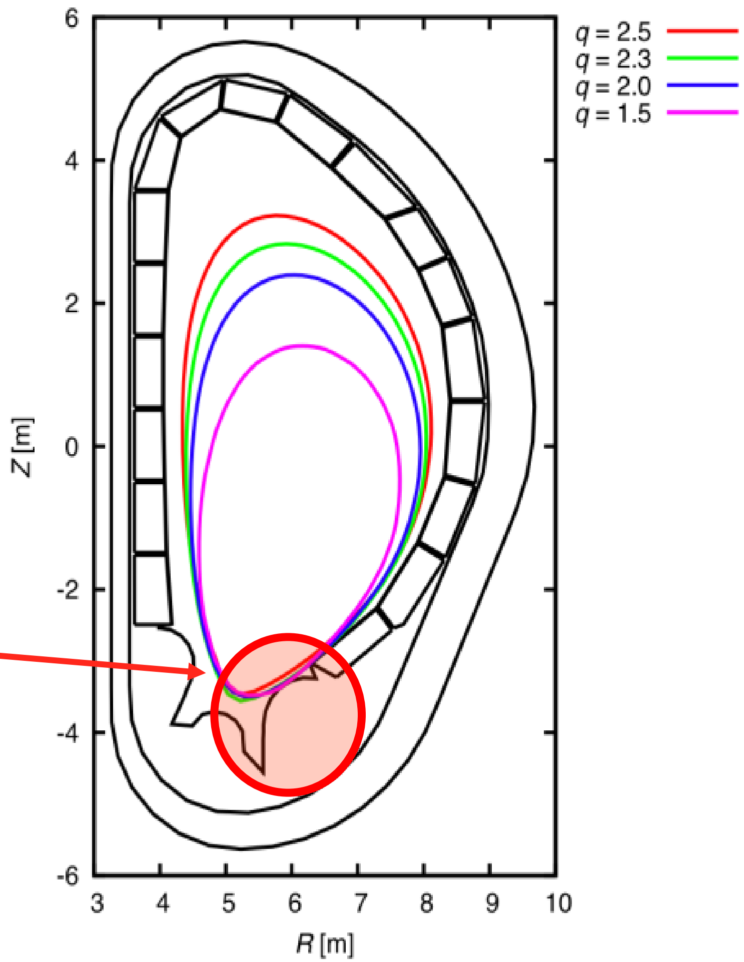
Power and Particle Exhaust

Transient power loads:

- **Energy loads at transients can cause W melting even in non-active phase:**
- unmitigated major disruptions in non-active phase can produce energy loads above 50 $MJm^{-2}s^{-1/2}$ melting limit for W (although uncertainties are large)
- type-I ELMs at 7.5 MA in helium plasmas might produce energy loads in this range
- outer baffle must be carefully shaped to mitigate possibility of melting during VDEs

- **Melting of Be surface can occur during current quench and VDEs**

⇒Early development of reliable disruption/ VDE and ELM mitigation methods essential!



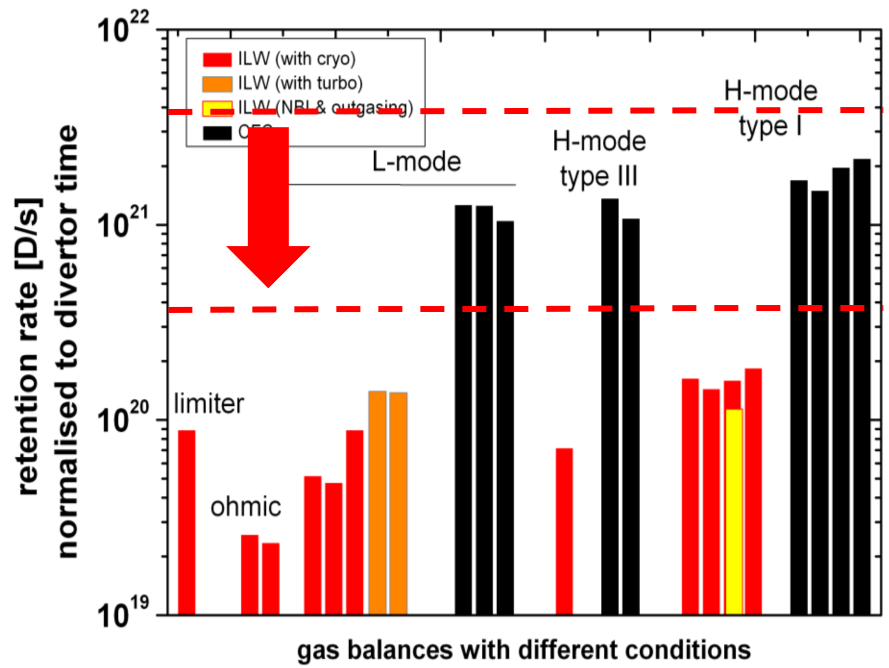
DINA simulation of 15 MA VDE

Tritium Retention

- **Gas balance in JET shows long term fuel retention reduced by <u>at least 10 × </u> in Be/W compared with C-walls**
  - as expected from laboratory studies on Be co-deposits before ILW experiments – now demonstrated on large tokamak scale
  - residual retention consistent with co-deposition in Be layers
  - <u>material migration model </u>used for ITER nuclear phase retention and dust generation estimates <u>fully supported by ILW experiments</u>

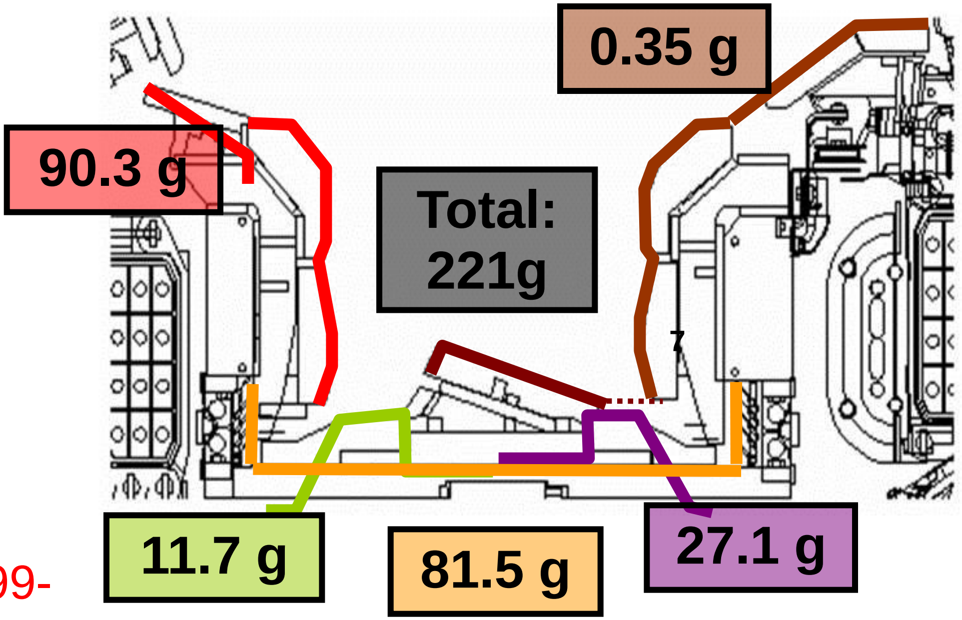
⇒ ITER must demonstrate capability to characterize fuel retention and to remove retained fuel (divertor baking at 350° C)

F. Romanelli, IAEA 2012



Dust

- **Erosion and redeposition processes in plasma environment produce microparticles and redeposited layers ⇒ dust formation**
- **Recent dust collection from JET after ~6 years → dominated by C but Be rich due to Be wall evaporation**
- **In ITER, dust production will be substantially higher than JET:**
  - long pulses and high particle fluxes: 1 ITER pulse ~ 6 years JET operation in terms of divertor fluence (based on 1999-2001 JET campaigns)
  - high transient heat loads at ELMs and disruptions



J. P. Coad,
A. Widdowson, JET

⇒ ITER must demonstrate capability to characterize dust production and to remove dust if excessive accumulation detected

Access to Good Confinement:

H-Mode Power Threshold

- **The latest H-mode threshold power scaling for deuterium plasmas:**

$$P_{thresh} = 0.05\bar{n}_e^{0.72} B_T^{0.8} S^{0.94}$$ (Y Martin, HMW-2008)

- **The isotope dependence based on JET results in H, D, and DT indicates that $P_{thresh} \propto 1/A$ for hydrogen isotopes**

Possible helium H-mode access

half-field/ half current H-mode development

n_e (10^{20} m^{-3})	B_T (T)	S (m^2)	P_{th} - H_2 (MW)	P_{th} - D_2 (MW)	P_{th} - DT (MW)
0.5	2.65	683	61	31	24
0.5	5.3	683	106	53	43
1.0	5.3	683	175	88	70

DT H-mode access

Q=10

Full-field/ full current H-mode development

No H-mode access in H at full field

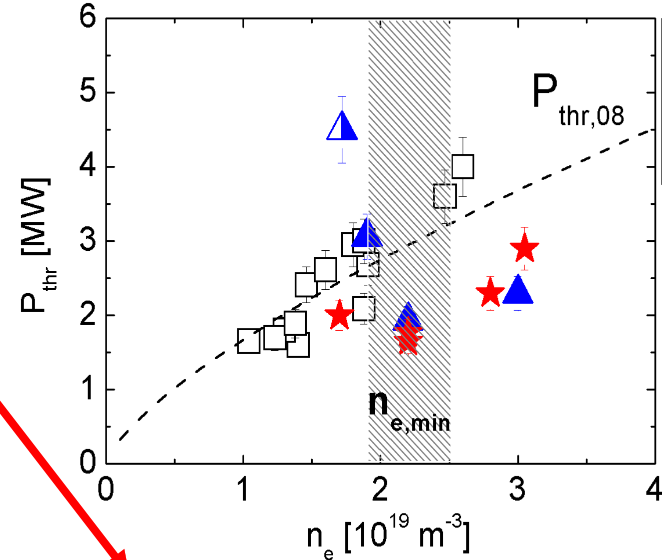
No H-mode access in D for full Q=10 simulation

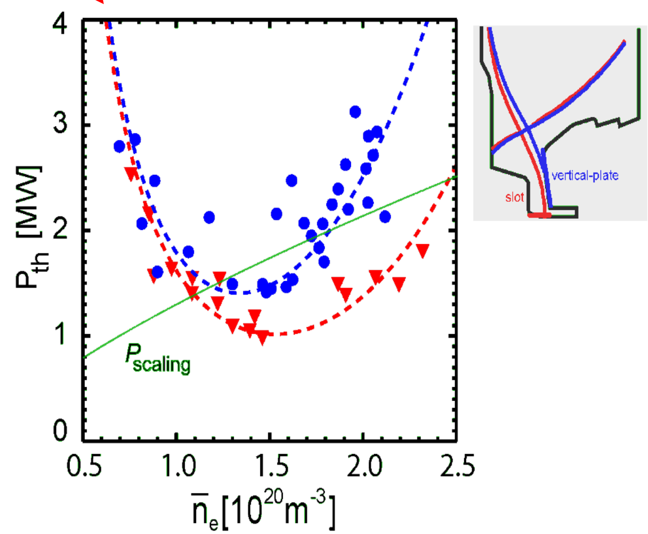
- **Note: margins may be required for (i) core radiation and (ii) access to good confinement (H_{98} = 1)**

L-H Transition

- **Power threshold clearly reduced in AUG (20%), JET (30%) after change from C to metal walls (but higher at low density in JET!) $\Rightarrow$ potential gain for ITER**
  - KSTAR (with C-walls) confirms existing P_{th} scaling
  - new C-Mod results demonstrate strong effect of divertor magnetic geometry on P_{th} (also at JET)
  - NB: lower P_{th} on JET in ILW does not appear to bring much advantage → much higher P_{net} required for H_{98} = 1 in the ILW (c.f. C walls) → pedestal pressure reduced in ILW
  - new result from AUG: ion pressure gradient separates L & H-mode ($\nabla p_i/en_i$) → use ECH and low n_e to decouple T_i, T_e → P_{LH} rises at low n_e due to reduced ion heating (we saw something similar at TCV).
  - new DIII-D results on links between high frequency turbulence and low frequency turbulent driven flow at the transition

JET, Beurskens, IAEA 2012





C-Mod, J. Hughes, IAEA 2012

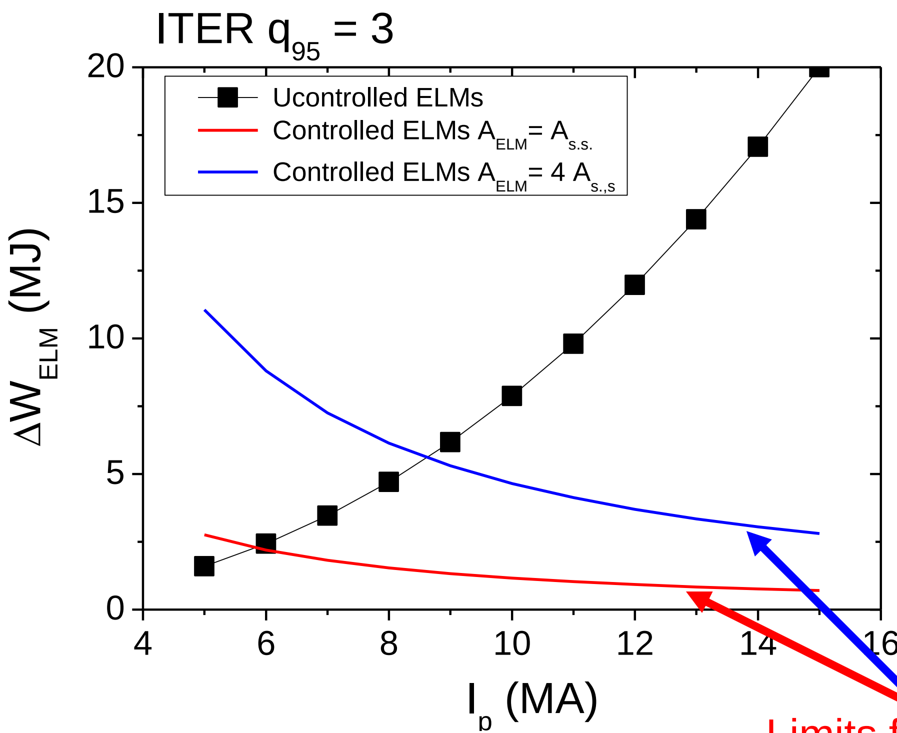
ITER H-mode Threshold - Implications

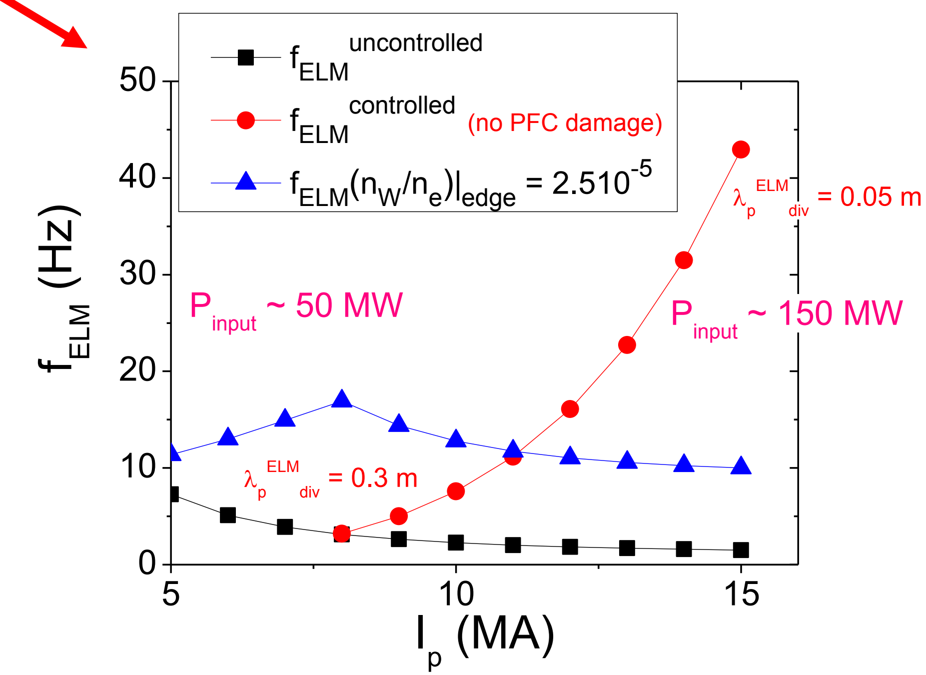
- **Uncertainties in prediction of H-mode threshold power remain substantial:**
  - recent experiments are identifying more clearly some of the "hidden variables" in the database: X-point height, PFC material …
  - but interpretation not always obvious
  - scaling of density minimum also an issue for ITER
  - access conditions for $H_{98} = 1$ confinement still ill-defined
  - observed reduction in threshold with all-metal walls intriguing and potentially beneficial

- **Hydrogen/ Helium operations:**
  - it has long been recognized that achievement of H-mode in hydrogen is at best marginal, requiring essentially full (100%) H&CD power routinely
  - ITER Research Plan plans call for initial studies of H-modes and ELM control in helium plasmas: ~ 50 MW required for reliable H-mode access at 7.5 MA/ 2.65 T

Uncontrolled ELMs Operation limited to: $I_p \leq 6$ - 9MA

- **In ITER, uncontrolled ELM operation with low erosion possible up to I_p = 6.0–9.0 MA depending on $A_{ELM}(\Delta W_{ELM})$**
  - ⇒ Mitigation of heat loads by factor of 10-20 required
- **Use of a tungsten divertor sets a lower limit on acceptable ELM frequency (or equivalent transport process) to limit W in core**



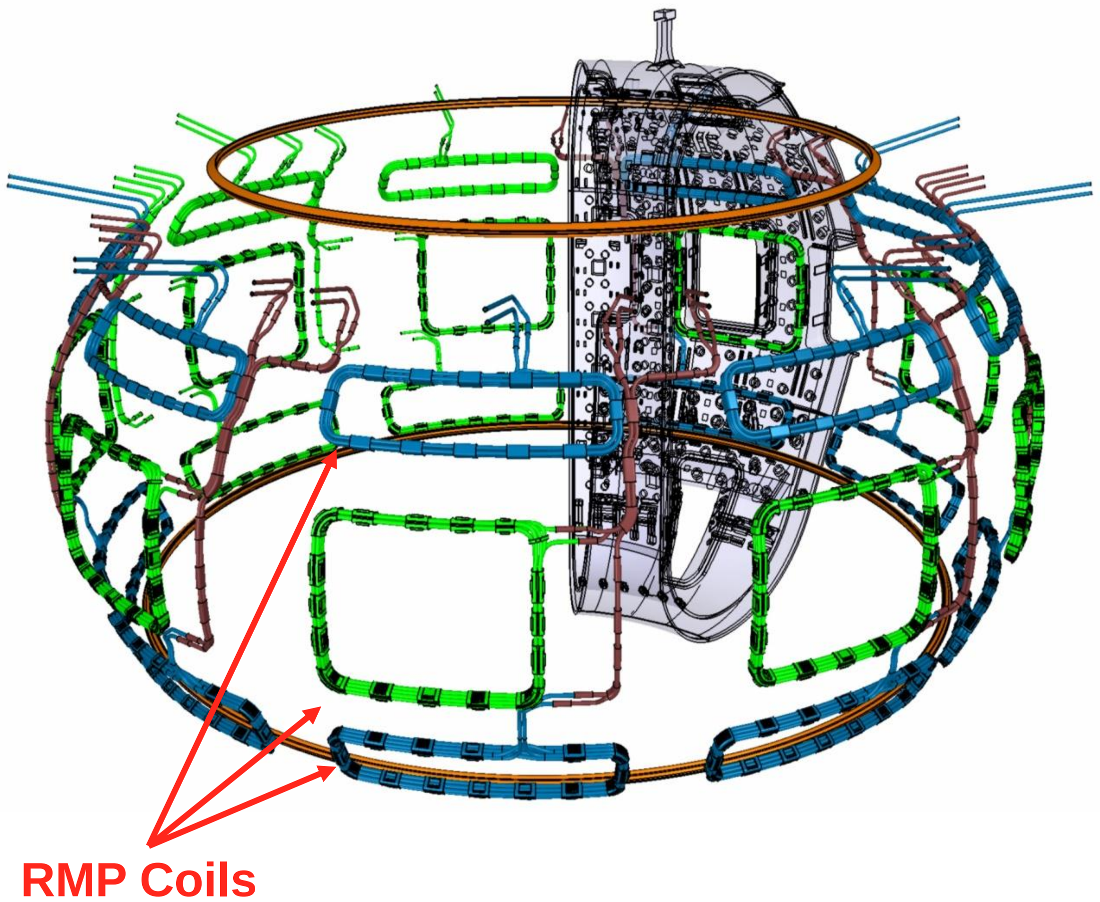


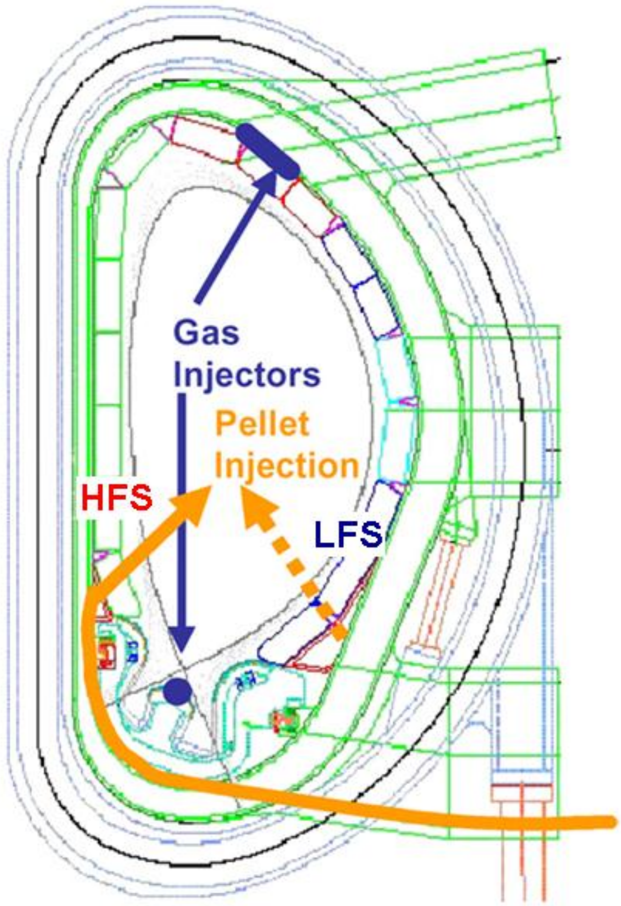
Limits for acceptable rates of erosion

A Loarte, IAEA-FEC2010

A Loarte, IAEA-FEC2012

ITER ELM Control Techniques

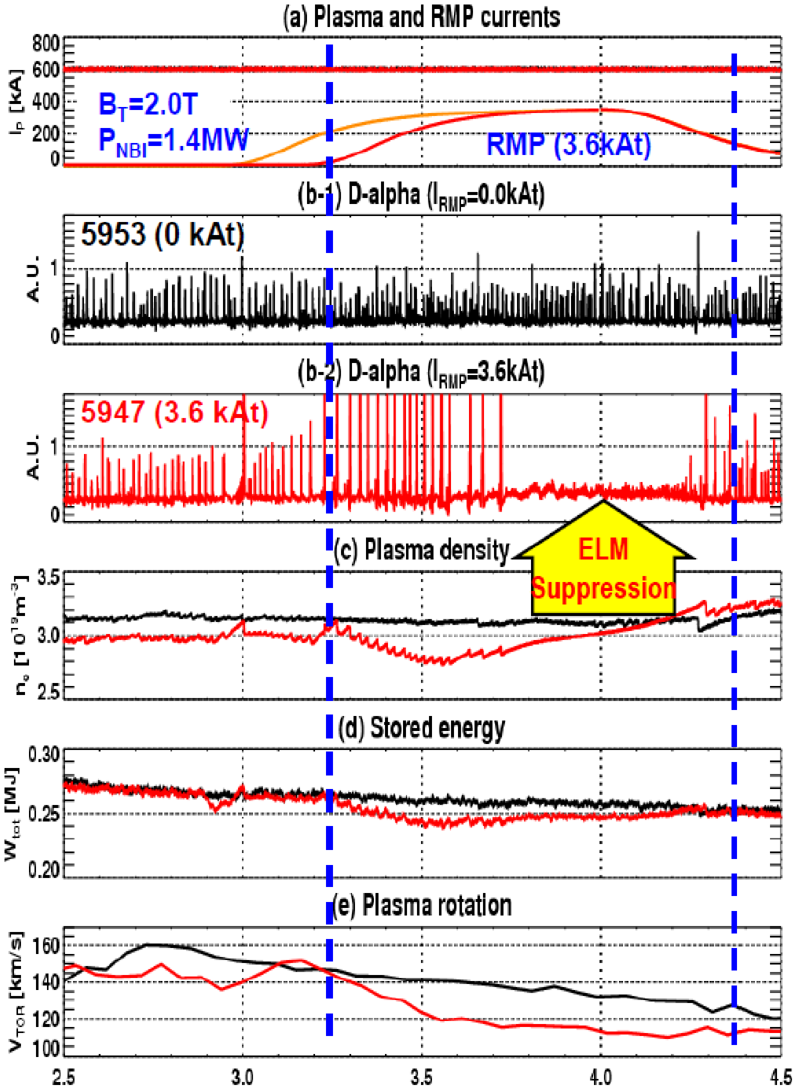




- **Two principal techniques under development:**
  - 3 × 9 array of RMP coils, launching mainly n=4, with 90 kAturn capability
  - high frequency (f ≤ 16 Hz) pellet injection system, allowing f_{inj} ~ 50 Hz

Type-I ELM Mitigation/ Suppression

- Suppression seen very recently on KSTAR with n = 1, +90º phasing
- DIII-D - suppression at n = 3 and now suppression at n = 2 at low collisionality (but low density)
- AUG - suppression at n = 2
- JET – suppression at n = 2 (ex-vessel coils)
- MAST – mitigation (but not yet suppression) at n = 4 or n = 6 in LSN, and n = 3 in DN – f_{ELM} has been increased by up to a factor 9
- Suppression/mitigation is usually accessible with small penalty on H-mode pedestal pressure, confinement
- Perturbations do not necessarily have to be resonant

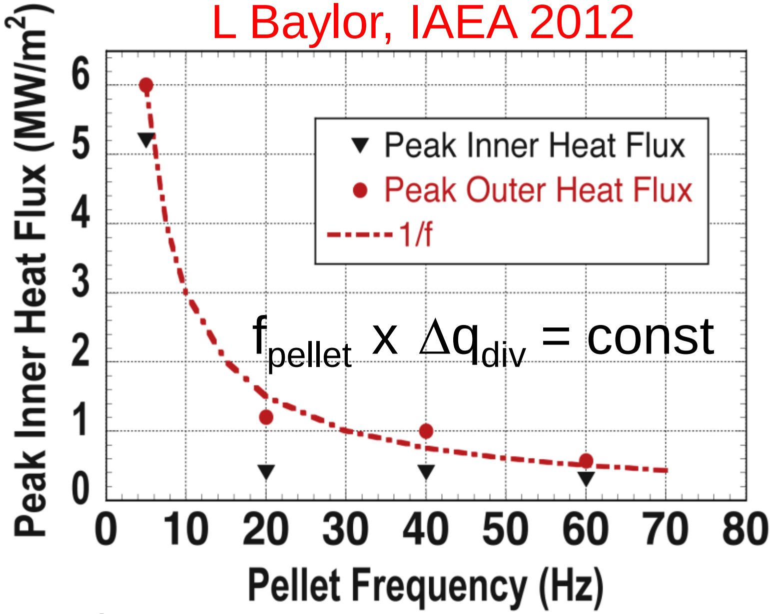


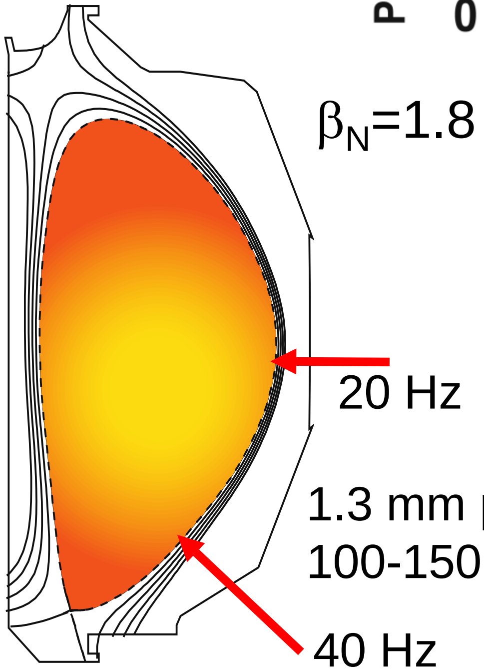
KSTAR, Y. Jeon, IAEA 2012

Type-I ELM Mitigation/ Suppression

- **Excellent new pellet pacing results from DIII-D:**
  - LFS injection up to 60 Hz
  - reduced ELM energy loss (reduced divertor heat flux) ⇒ seems to contradict JET divertor heat load findings
  - very little change in confinement
  - no increase in density

L Baylor, IAEA 2012





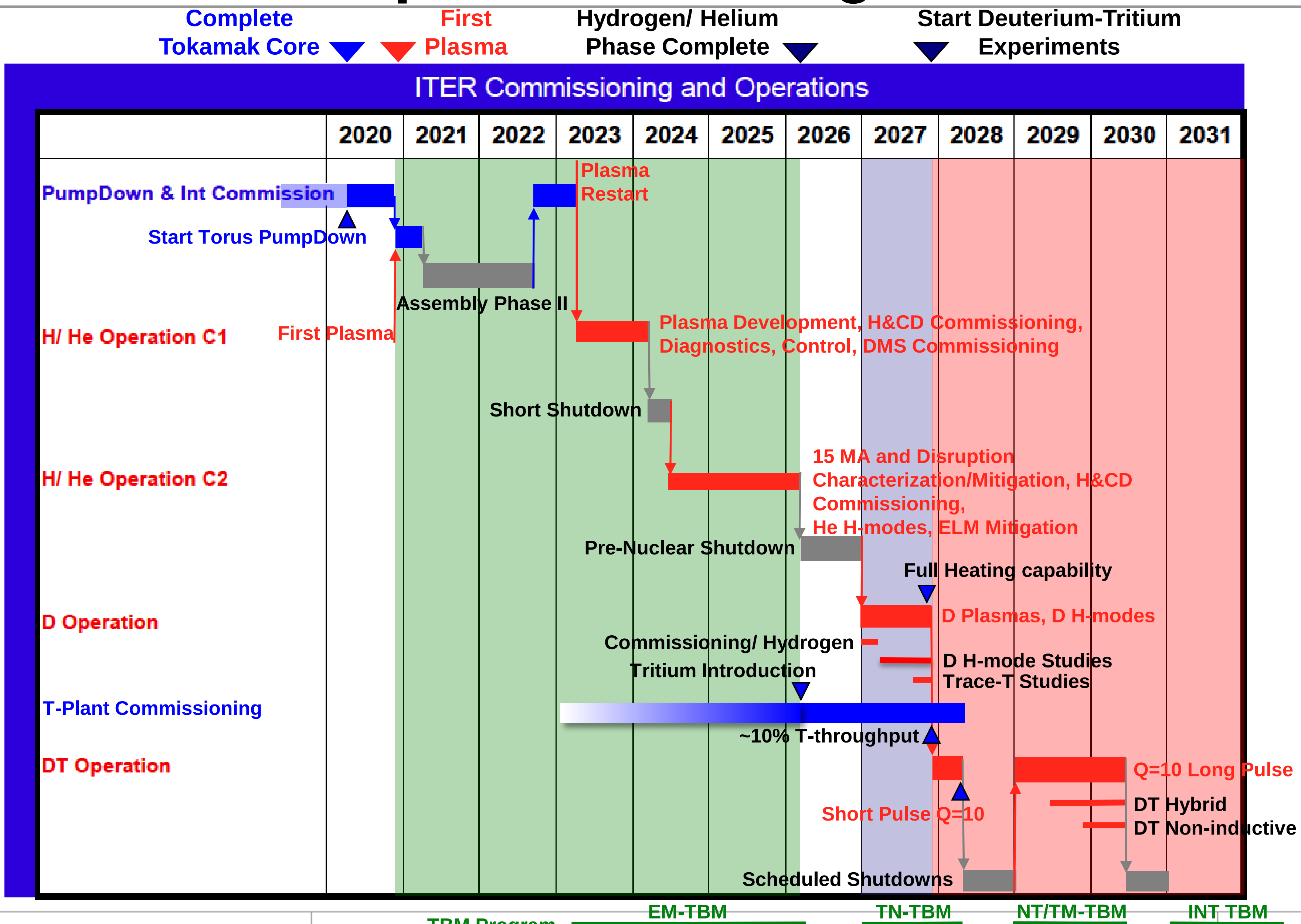
ITER shape, launch geometry

Type-I ELM Mitigation/ Suppression

- **Type I ELM suppression/mitigation using magnetic perturbations now demonstrated on 6 tokamaks equipped with coil systems:**
  - DIII-D, AUG, KSTAR, MAST, NSTX (in-vessel coils)
  - JET ILW (ex-vessel Error Field Correction Coils)
- **Type I ELM pellet pacing demonstrated in 3 tokamaks:**
  - DIII-D, JET, AUG
  - Latest DIII-D experiments access ITER relevant range of pellet ELM control (LFS injection, f_{ELM} up to ~60 Hz)
- **Vertical kicks as ELM control method demonstrated on 3 tokamaks:**
  - TCV, AUG, JET → an option for ITER at low plasma current (e.g. potential route towards minimizing W impurity build-up during early H-mode phases on ITER
- **Major progress across the world's tokamaks:**
  - considerably strengthens confidence that ITER's mitigation strategies are sound

⇒ **R&D should continue to better assess impact of ELM mitigation methods on relevant scenarios (confinement, H-mode threshold, stability etc)**

ITER Experimental Programme



Conclusions

- **Achievement of high fusion gain DT plasmas in ITER will require the integration of several challenging aspects of plasma operation:**
  - this capability will be built up through a multi-annual research programme
  - flexibility in design of tokamak and auxiliary systems are fundamental to successful implementation of this programme

- **The ITER Research Plan has allowed us to develop the major steps on the path towards DT fusion power production:**
  - identification of the principal challenges and risks

- **R&D activities in present experimental, theory and modelling programmes will make a significant contribution to providing the physics basis and methodology for resolving the key challenges:**
  - cost effective use of the fusion programme's resources

$\Rightarrow$ **Fusion community is an integral part of the ITER project**