The ITER Research Plan

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Acknowledgements:

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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 ($Q \geq 5$/ ≤ 3000 s)
- retain the possibility of exploring 'controlled ignition' ($Q \geq 30$)

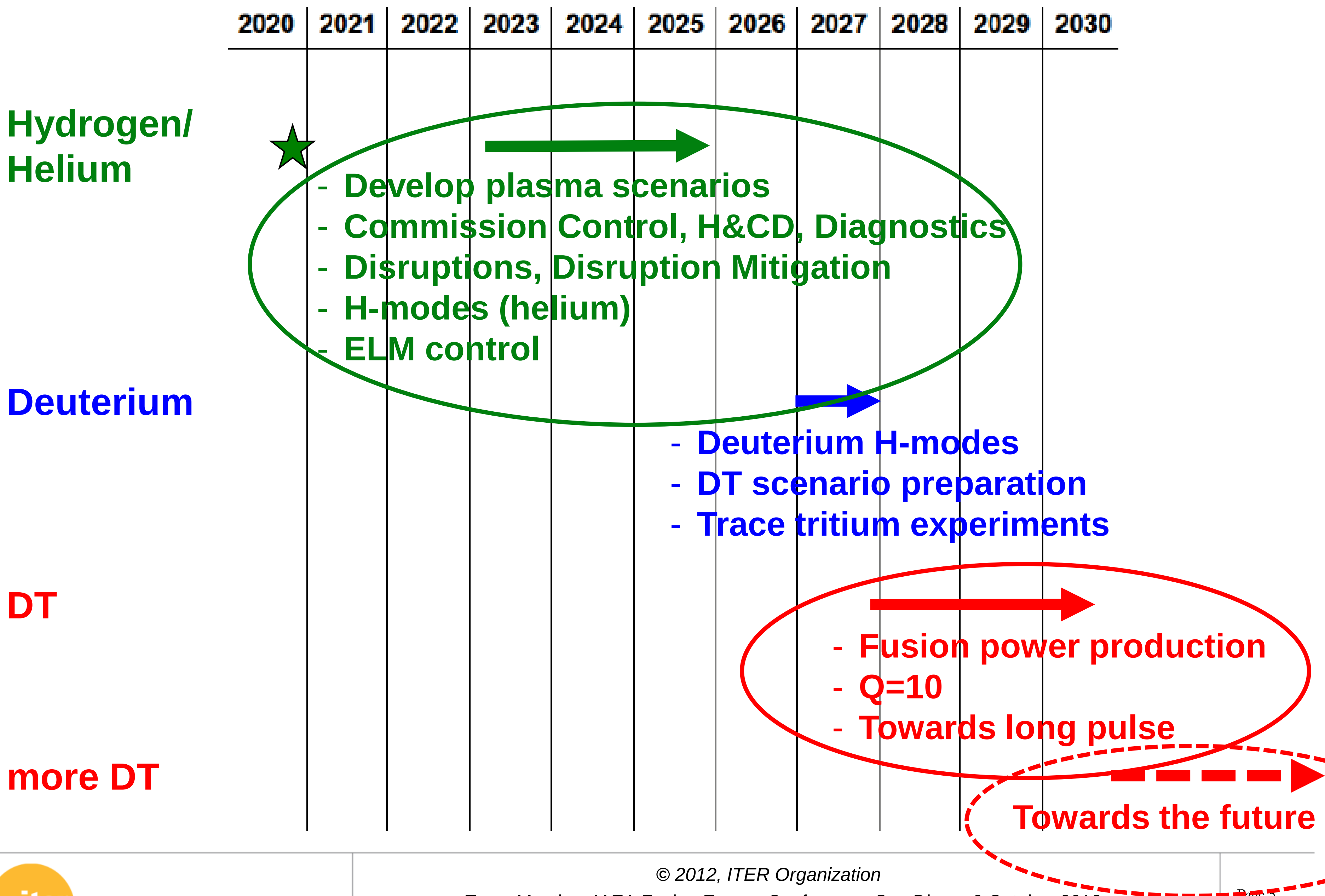
Technology:

- 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

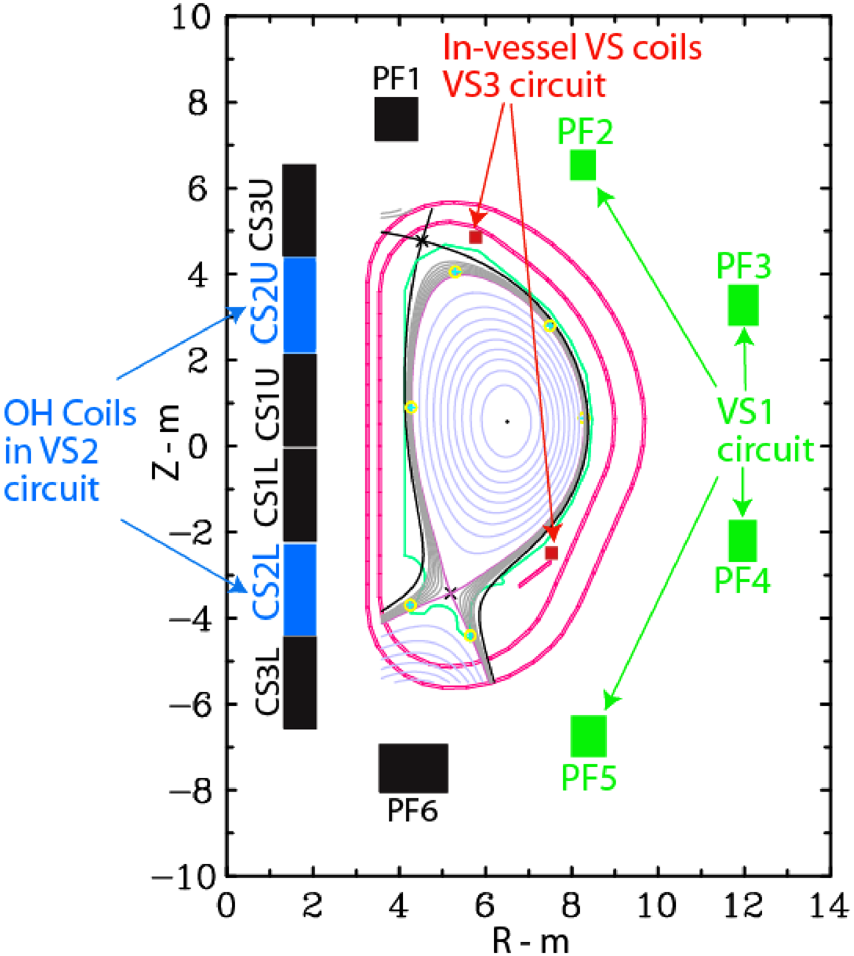


Risk Assessment ⇒ Key R&D Needs

- **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

ITER PF layout



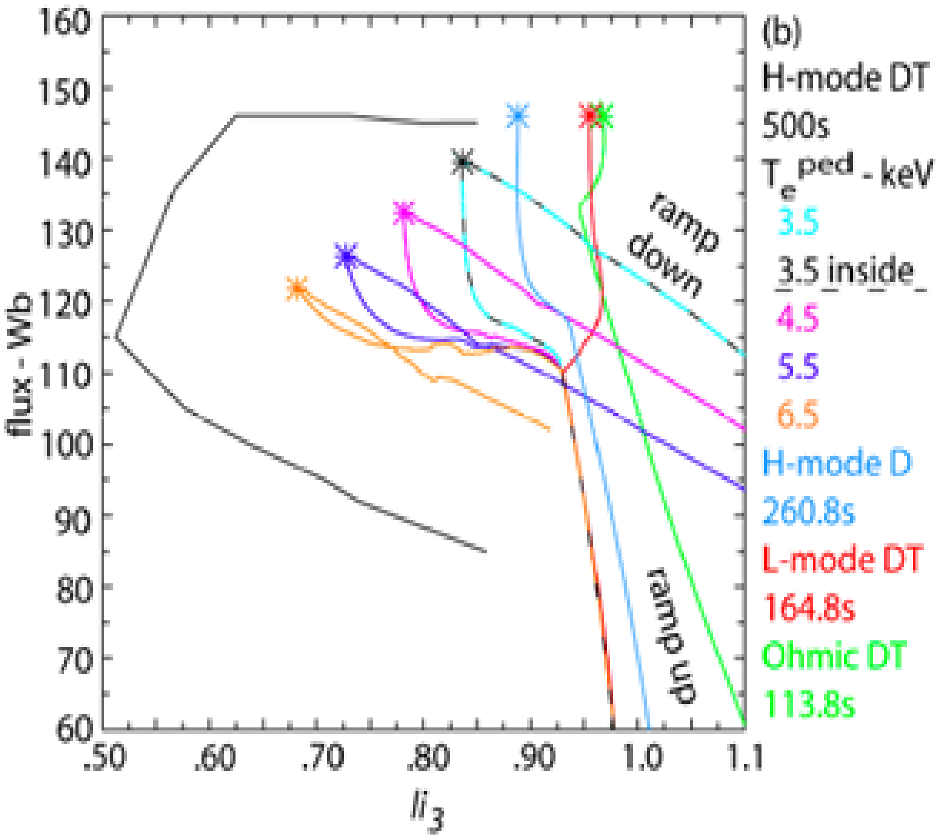
- **In ITER, care must be taken in developing scenario:**
  - avoid coil current saturation
  - minimize flux consumption during current ramp-up
  - maintain plasma position control during transients
  - maintain vertical stability during current ramp-down

  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

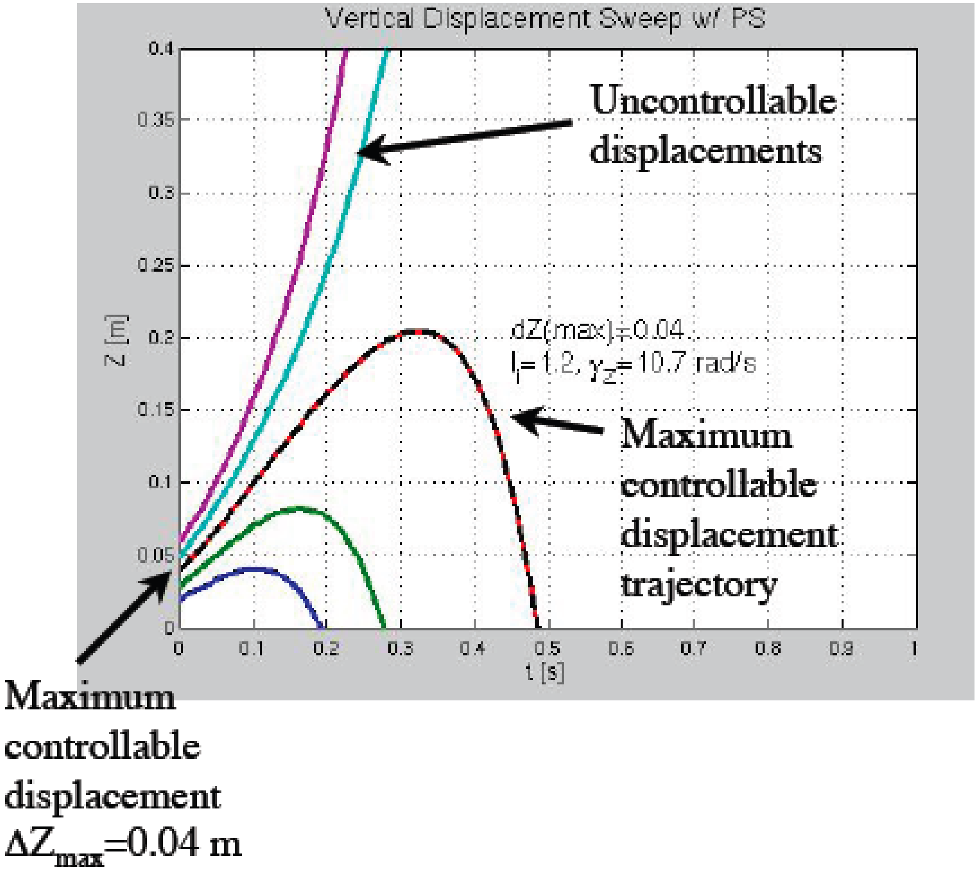


Vertical Stabilization Performance

- **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}



⇒ Internal coils for vertical stabilization to meet requirements

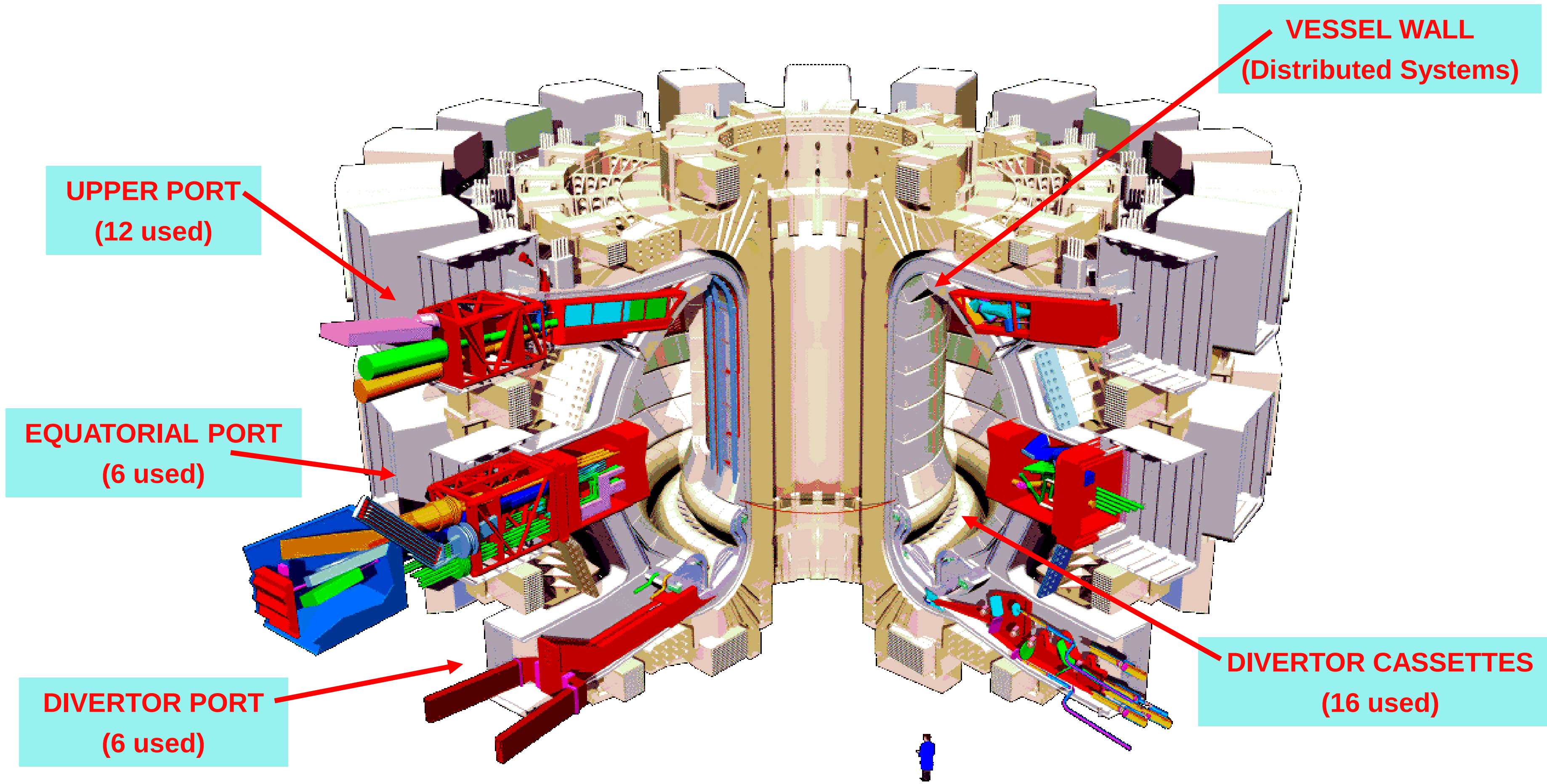
Heating and Current Drive

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)

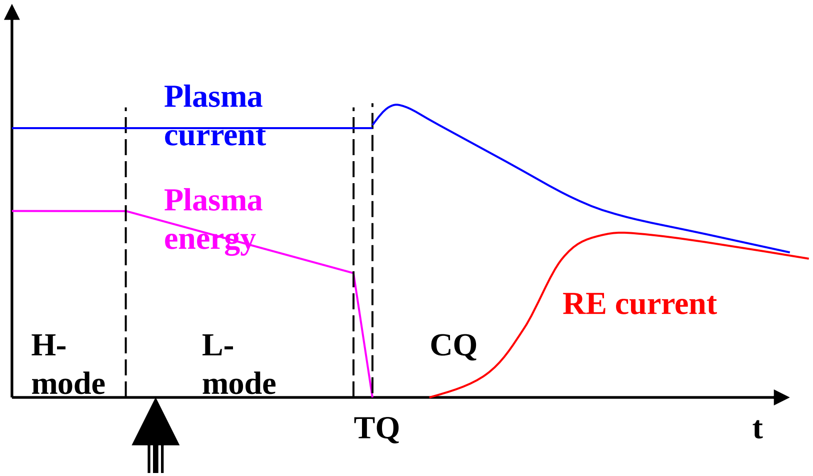
* 10 MW available in non-active phase – only one ICRF antenna installed

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



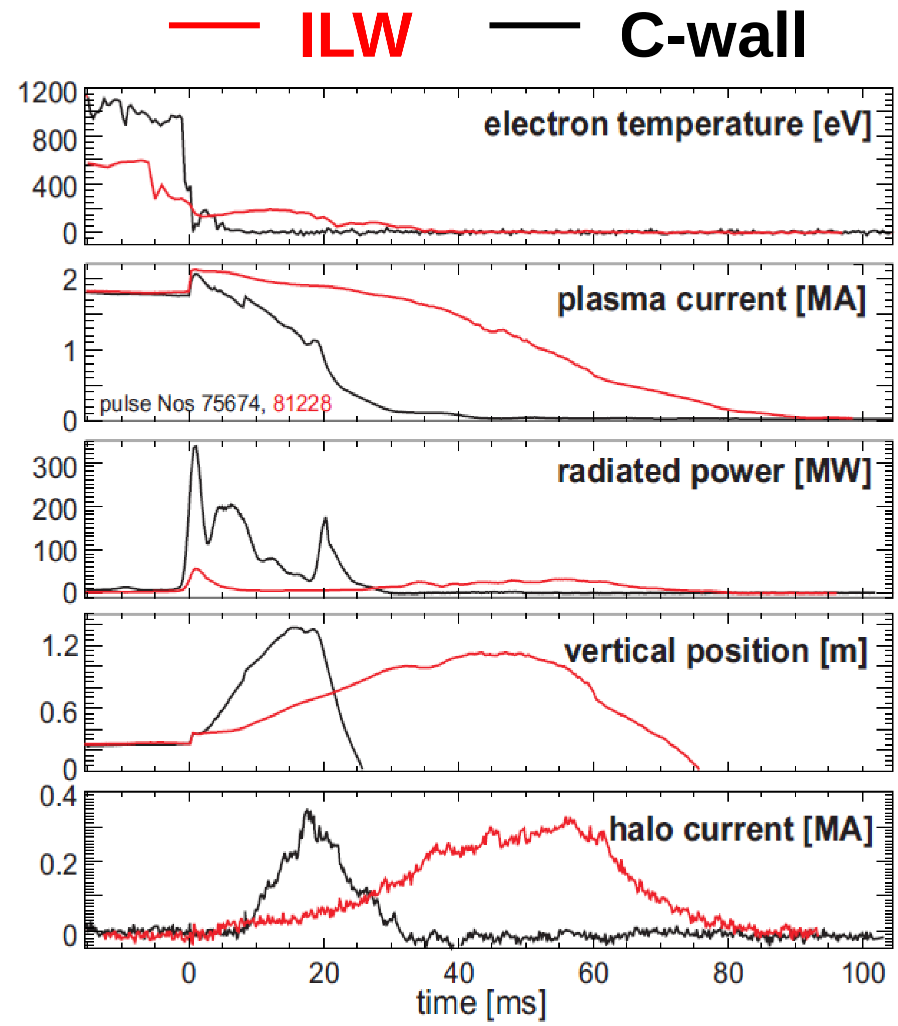
Typical chain of events during plasma disruption

Disruption/ VDE/ RE mitigation is essential for reliable operation of ITER
⇒ Massive material injection (MMI) is the most likely solution

- **Most serious thermal loads occur during Thermal Quench**
  ⇒ 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 ⇒ eddy currents, "halo" currents**
  ⇒ Need to reduce by factor of at 2-3 to improve load margins
- **Runaway electrons can be generated during Current Quench**
  ⇒ Need to reduce <u>intensity</u> and <u>energy</u> factor of at least 10

Disruption/ Mitigation

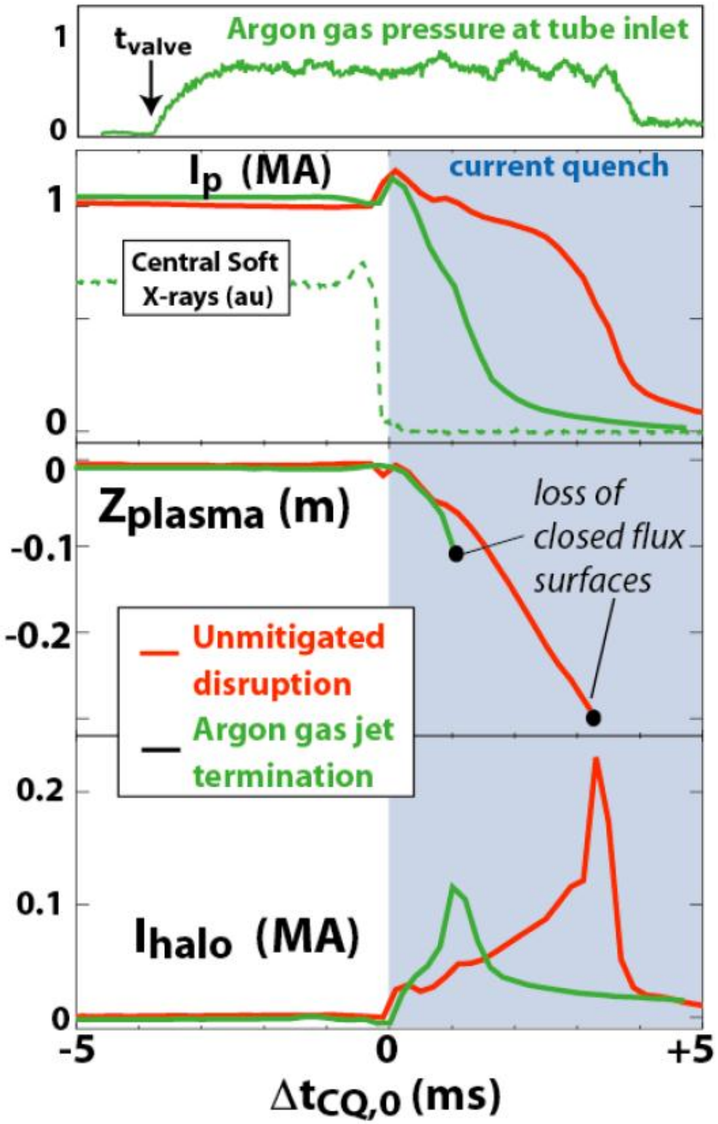
- **Well recognized issue for ITER with all-metal walls (W_{th}, W_{mag} >> than current devices):**
  - JET ILW clearly demonstrated expected low radiation in unmitigated TQ and CQ (cf. C walls)
  - hotter CQ plasma, slower current decay, slower vertical displacement, longer halo current phase
  - energy 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

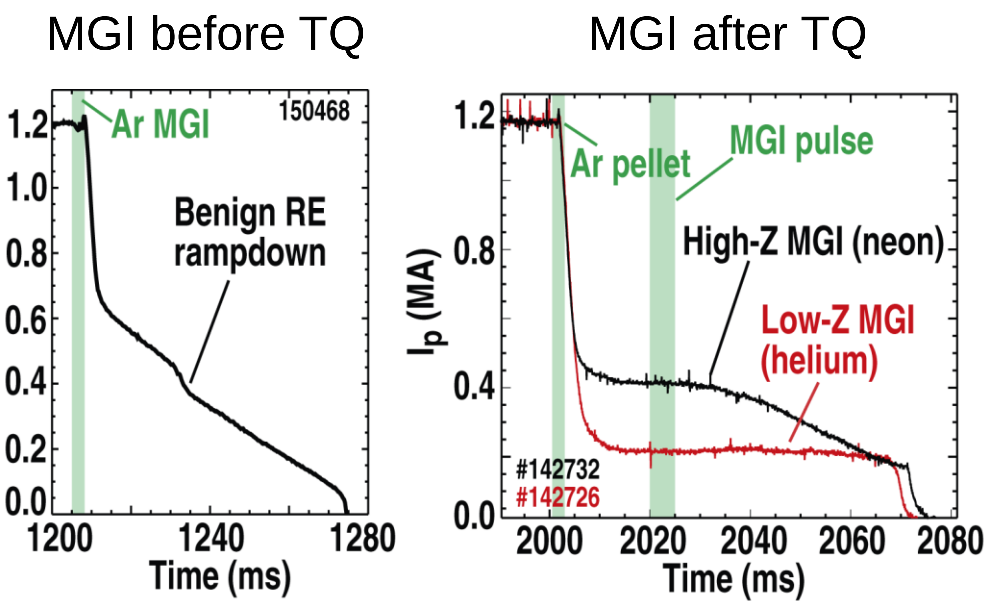
D Whyte, PSI-2006



- **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)

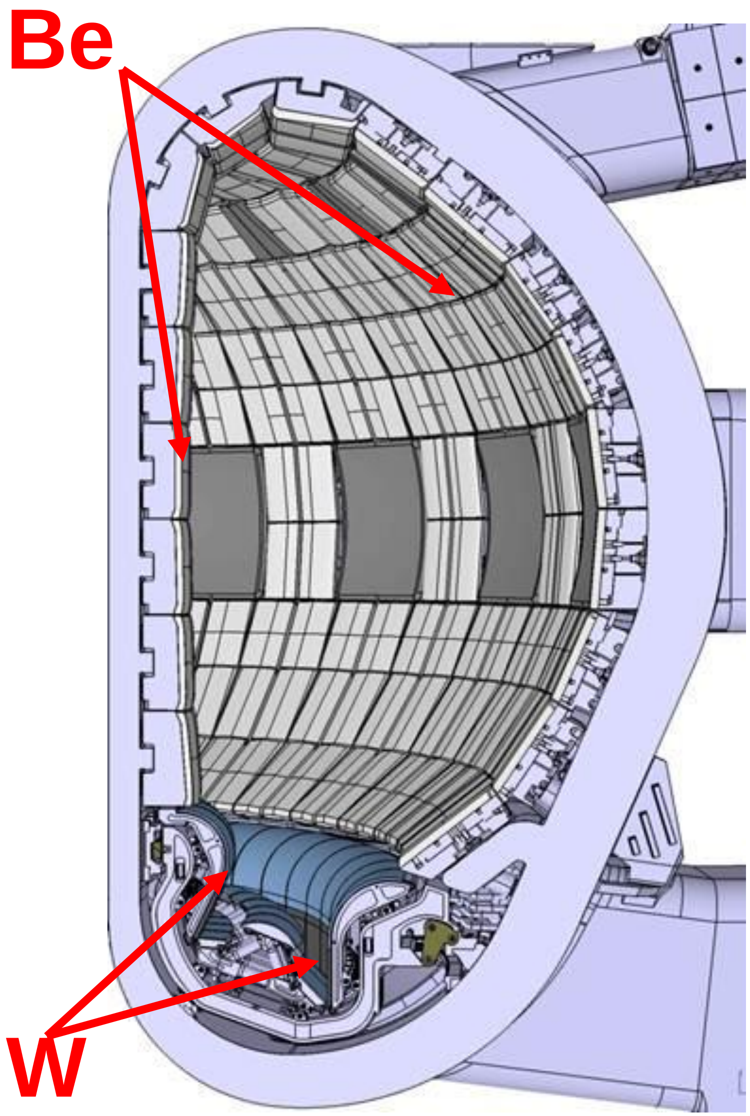
- **DMS must also incorporate flexibility to allow for learning and tuning during non-active phase of operation**

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

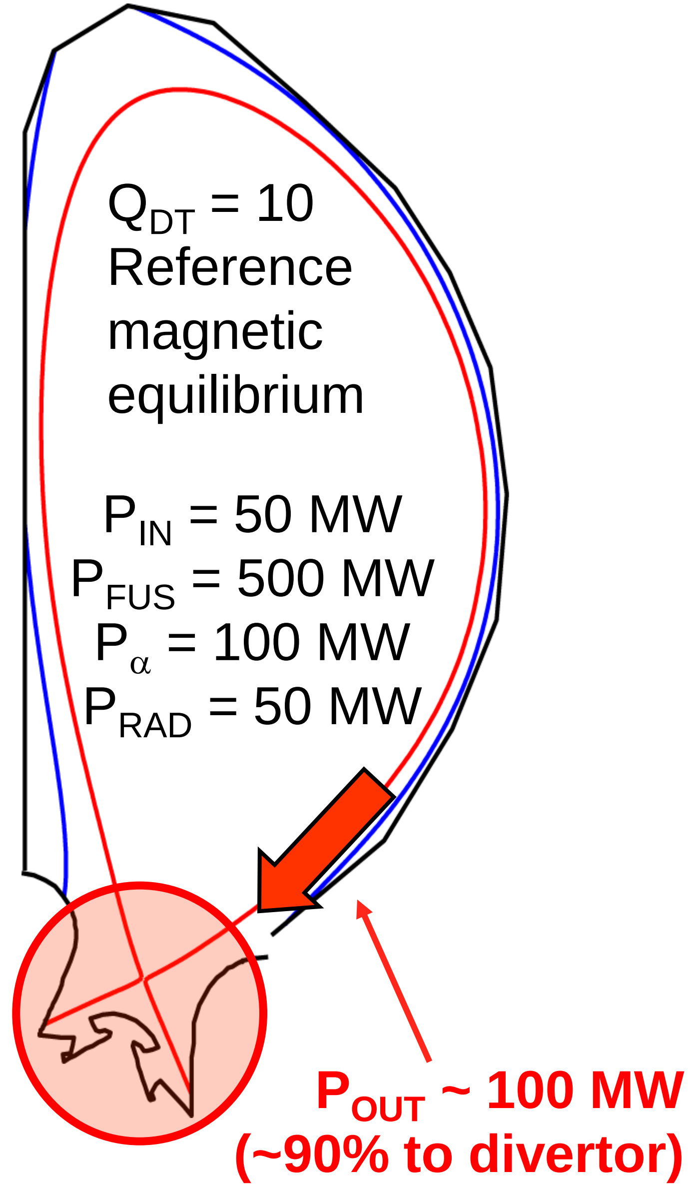
Stationary power handling:

- **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 !**

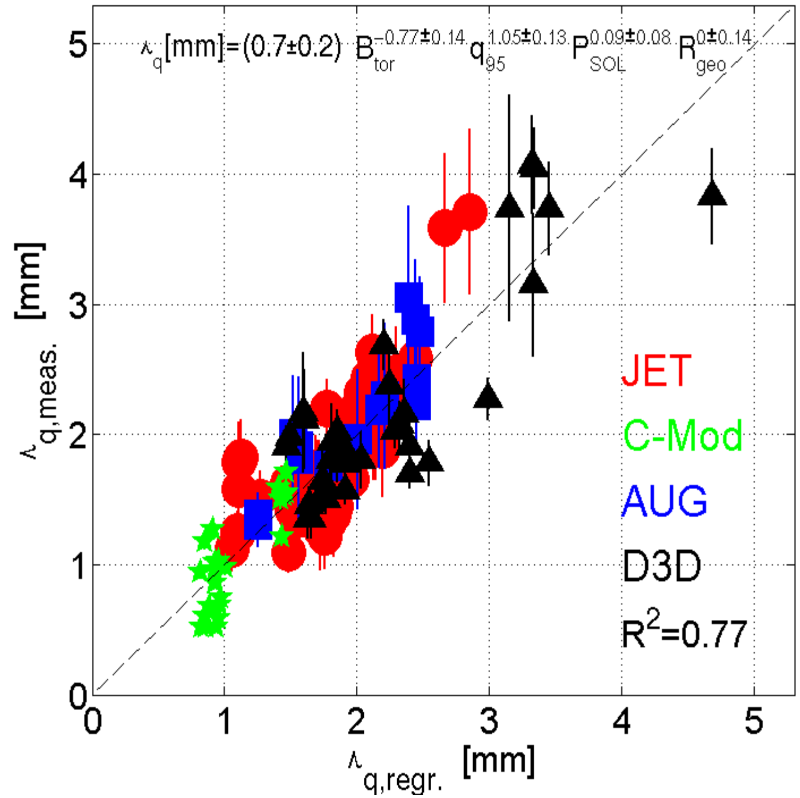


Power and Particle Exhaust

Heat Flux Width:

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

- **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



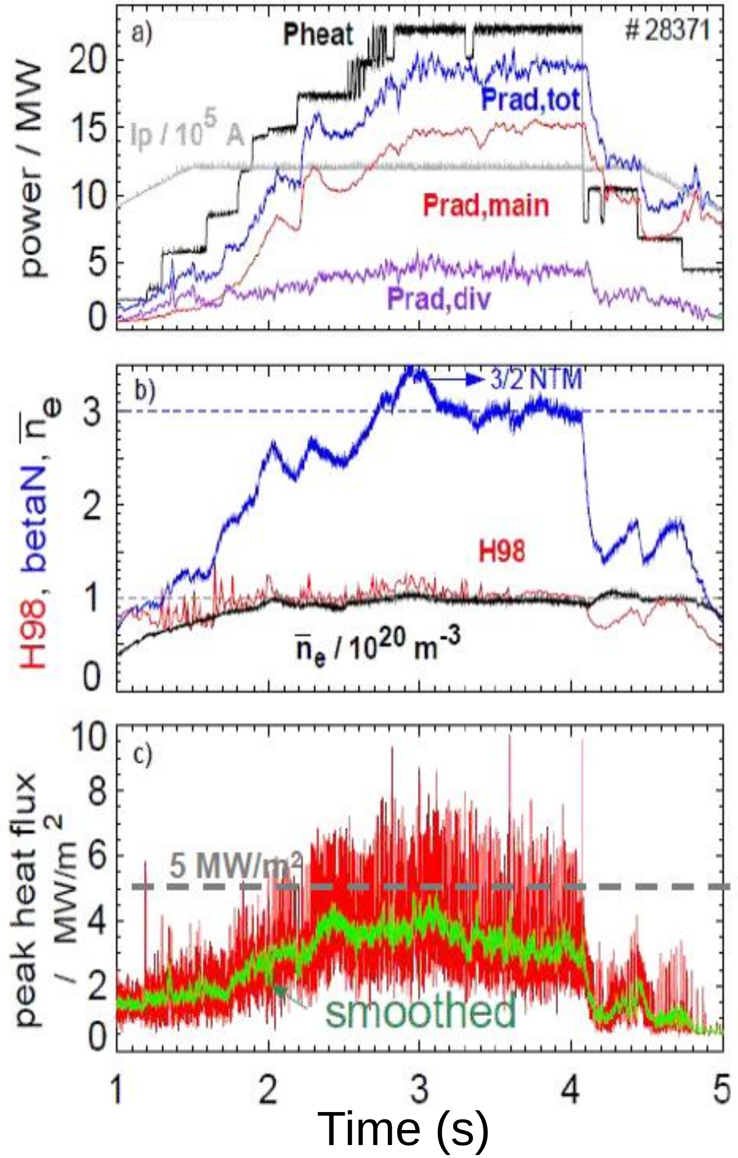
T. Eich et al., IAEA 2012

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)!



iter china eu india japan korea russia usa

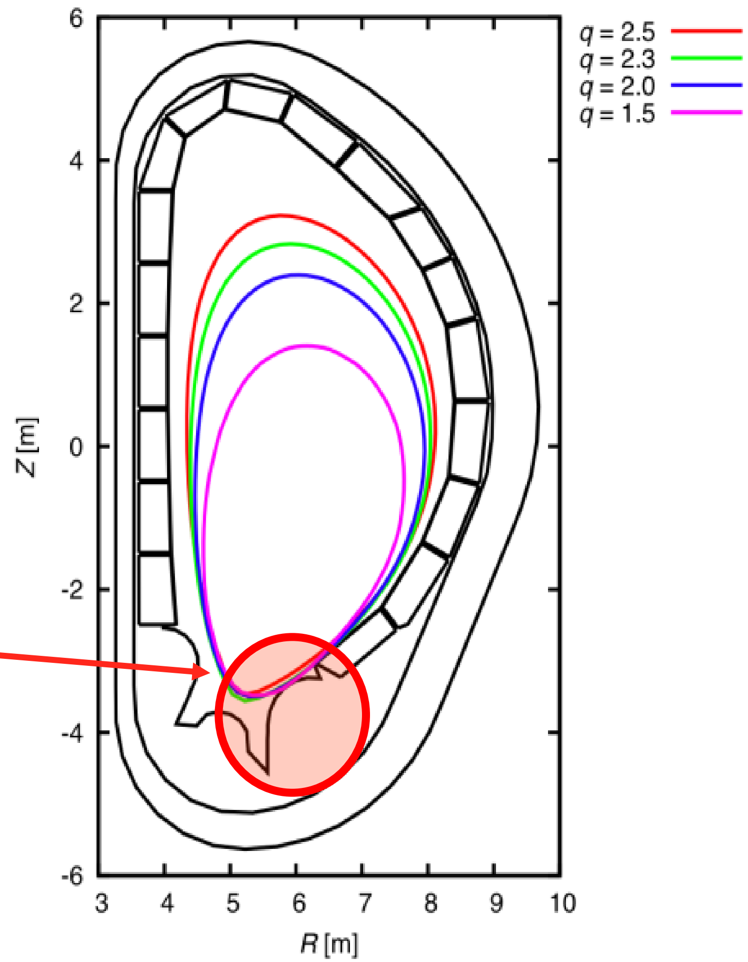
Power and Particle Exhaust

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!



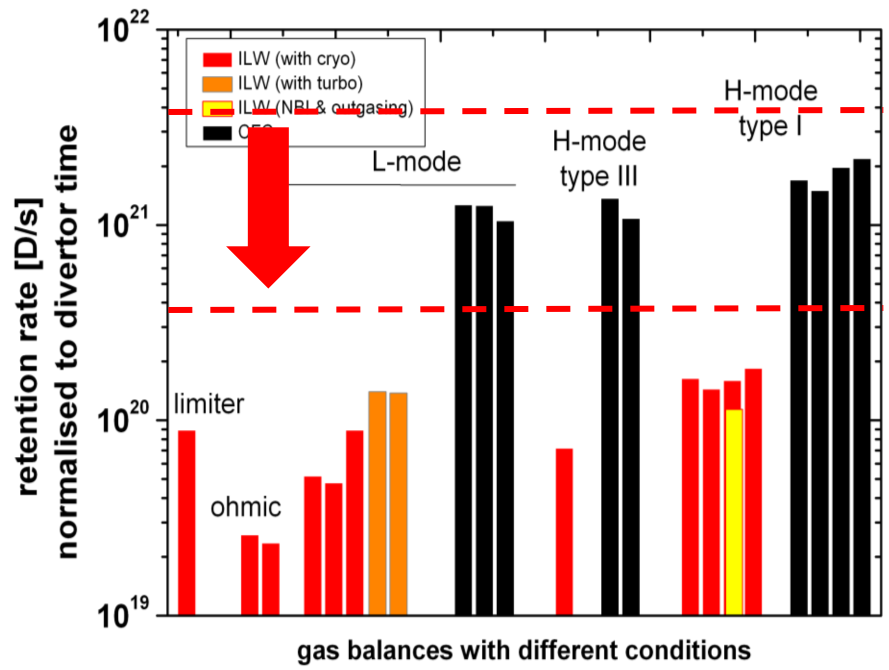
DINA simulation of 15 MA VDE

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>

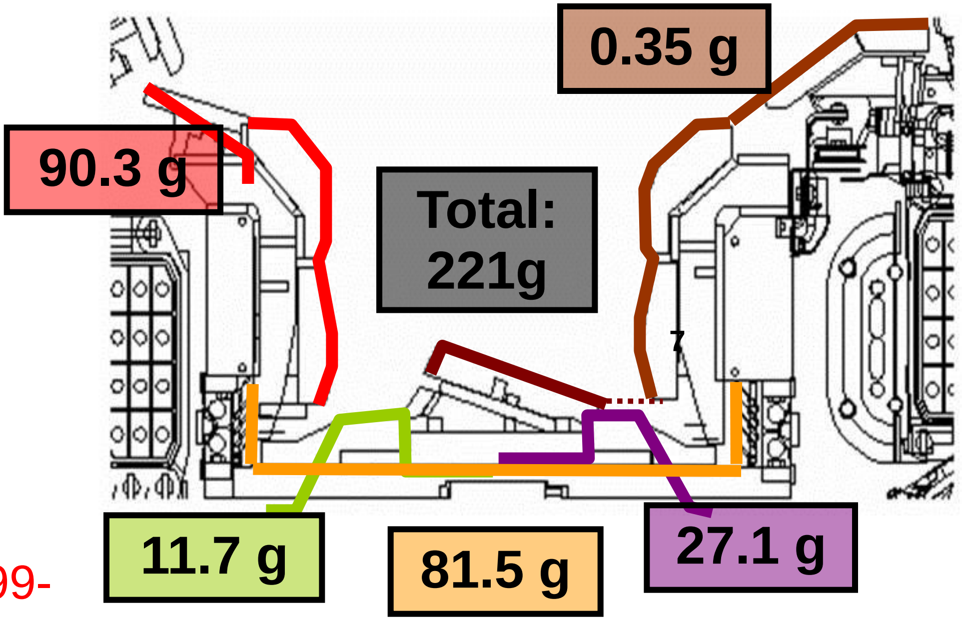
⇒ **ITER must demonstrate capability to characterize fuel retention and to remove retained fuel (divertor baking at 350° C)**

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

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

half-field/ half current H-mode development

DT H-mode access

Q=10

Full-field/ full current H-mode development

No H-mode access in H at full field

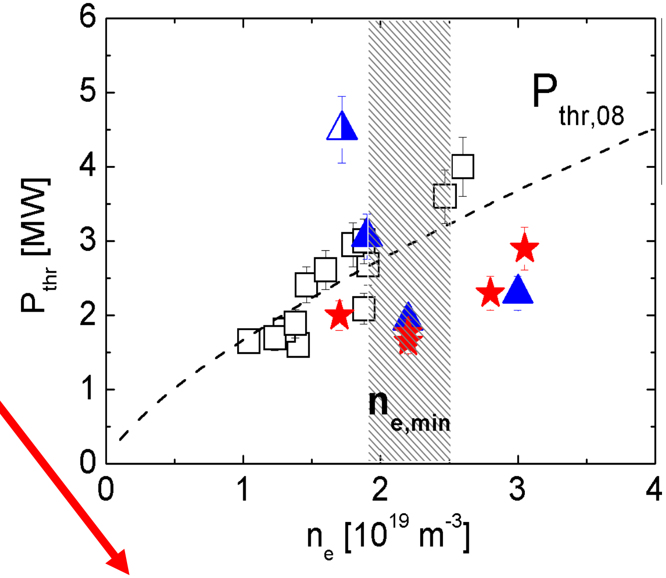
No H-mode access in D for full Q=10 simulation

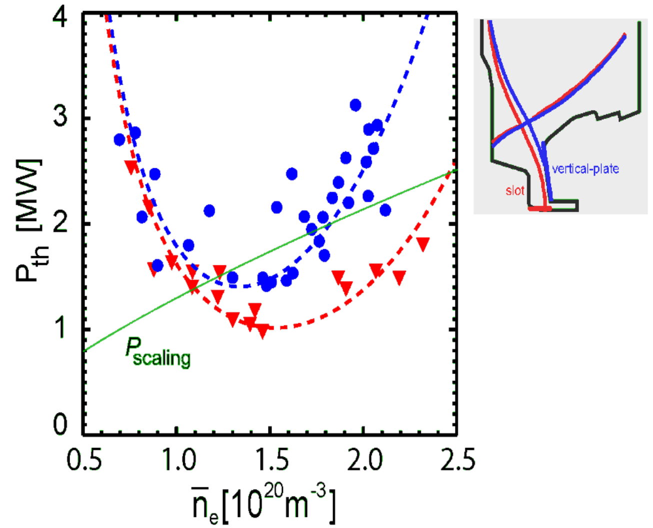
- **Note: margins may be required for (i) core radiation and (ii) access to good confinement (H_{98} = 1)**

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

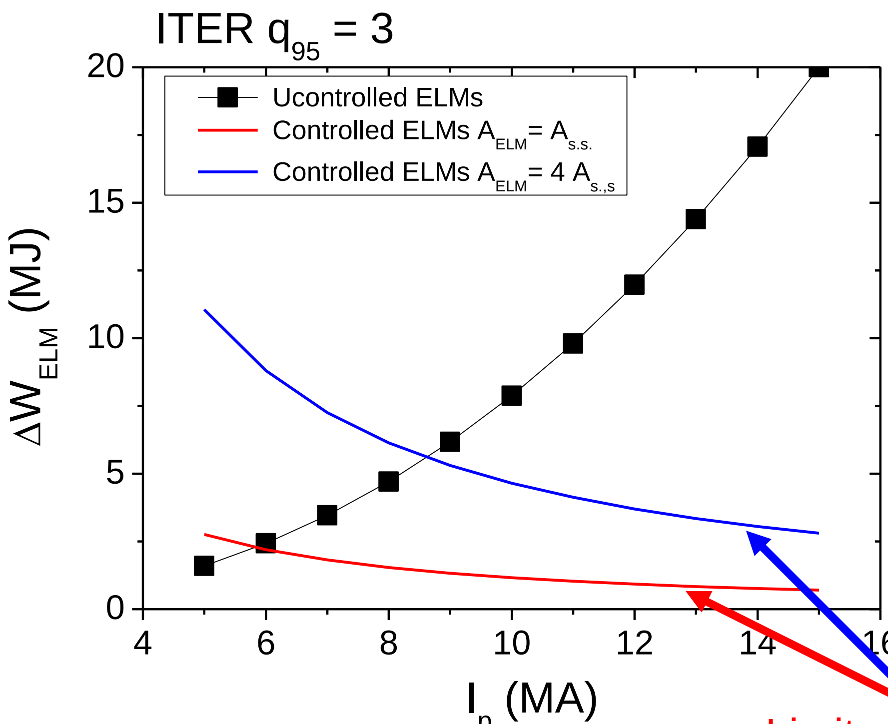
- **Hydrogen/ Helium operations:**
  - it has long been recognized that achievement of H-mode in hydrogen is at best marginal, requiring essentially full (100%) H&CD power routinely
  - ITER Research Plan plans call for initial studies of H-modes and ELM control in helium plasmas: ~ 50 MW required for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

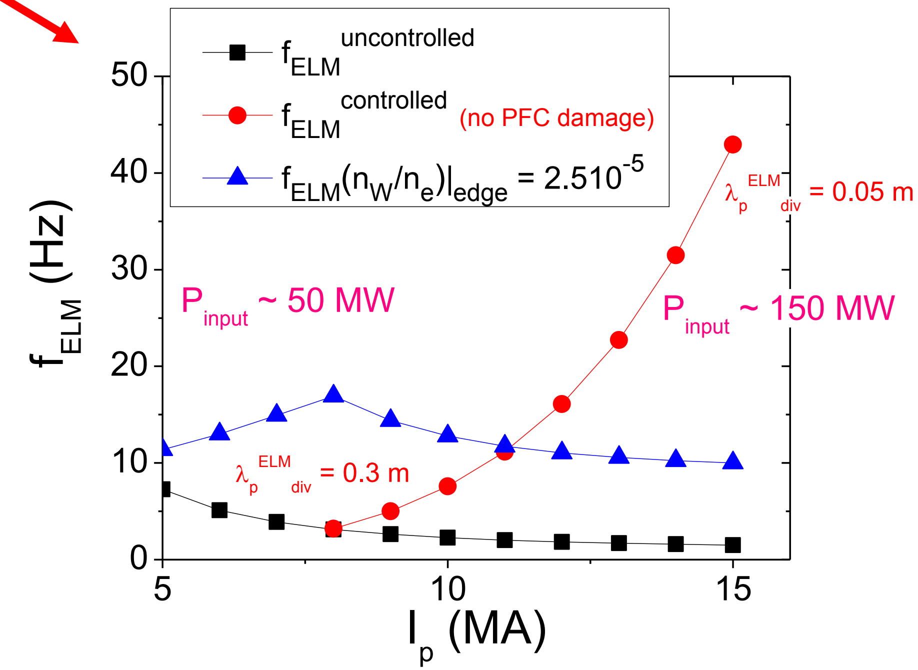
Uncontrolled ELMs Operation limited to: $I_p \leq 6$ - 9MA

- **In ITER, uncontrolled ELM operation with low erosion possible up to I_p = 6.0–9.0 MA depending on $A_{ELM}(\Delta W_{ELM})$**

  $\Rightarrow$ 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**



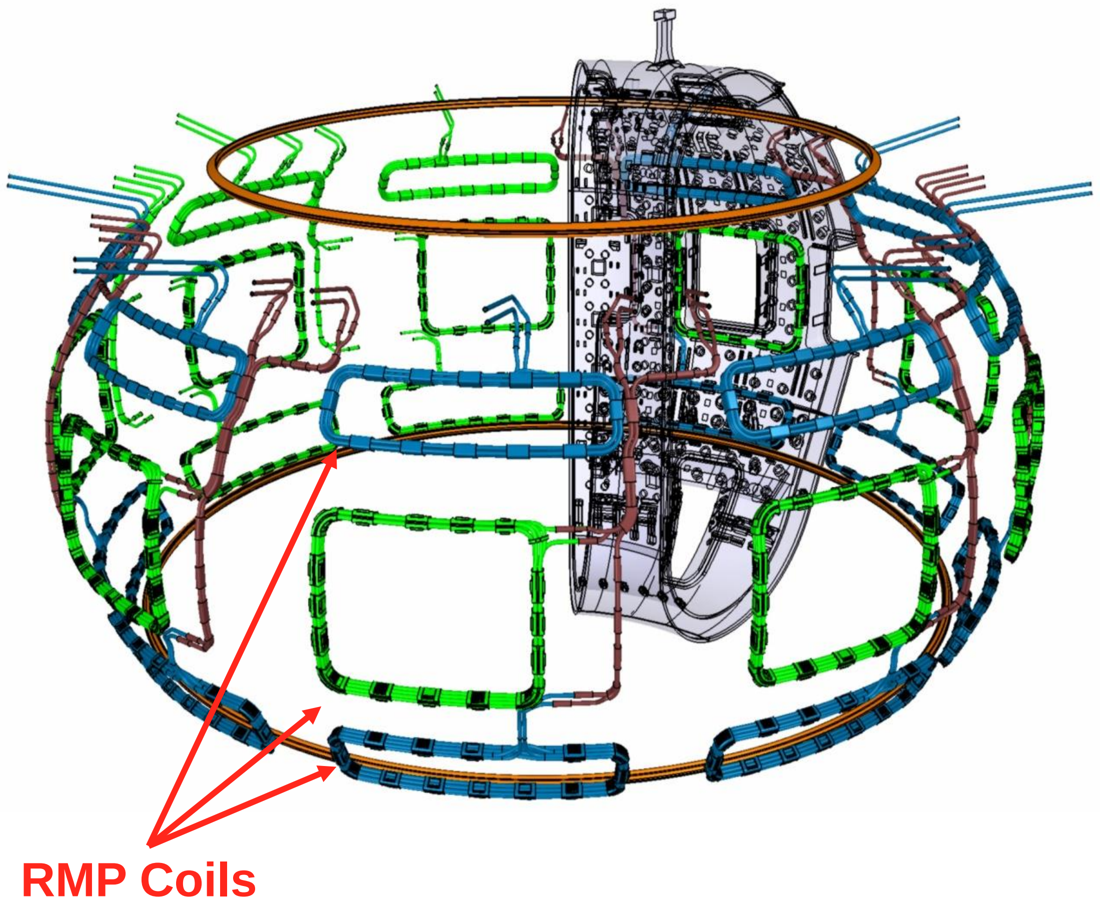


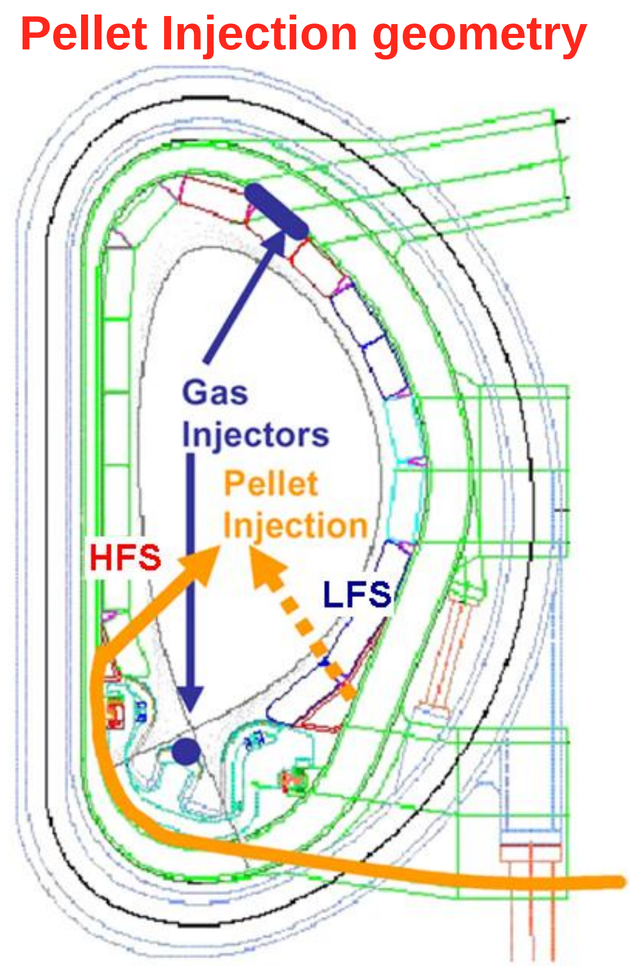
Limits for acceptable rates of erosion

A Loarte, IAEA-FEC2010

A Loarte, IAEA-FEC2012

ITER ELM Control Techniques

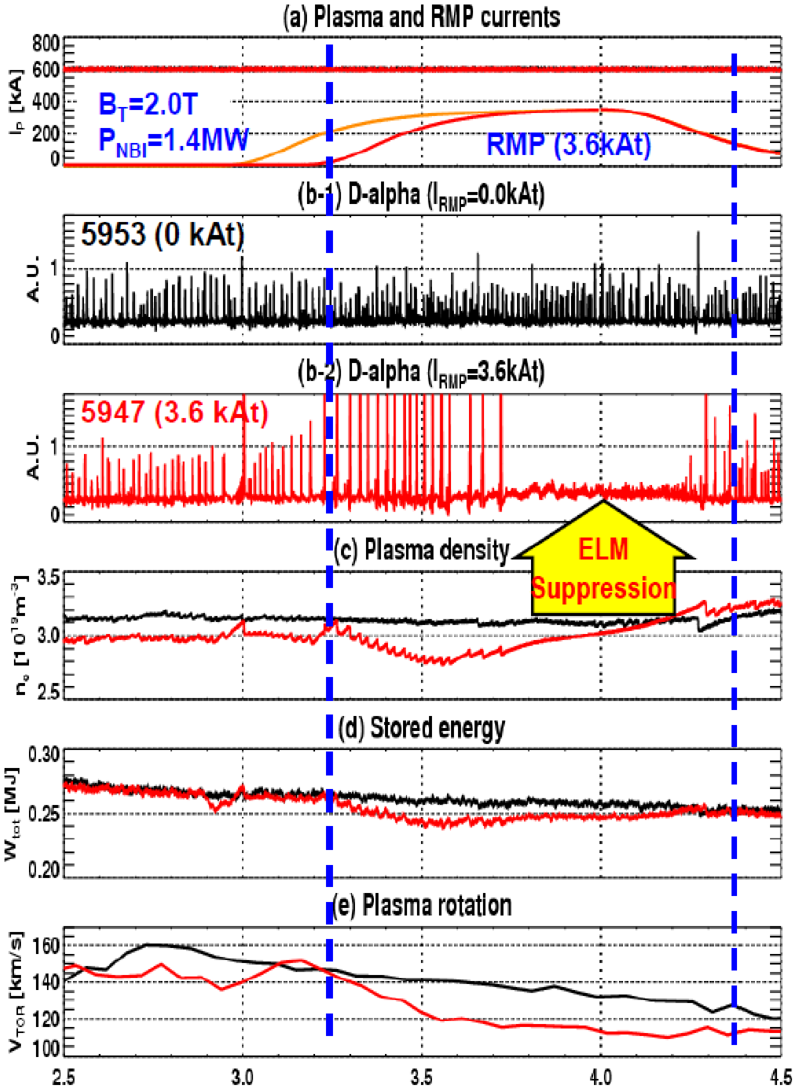




- **Two principal techniques under development:**
  - 3 × 9 array of RMP coils, launching mainly n=4, with 90 kAturn capability
  - high frequency (f ≤ 16 Hz) pellet injection system, allowing f_{inj} ~ 50 Hz

Type-I ELM Mitigation/ Suppression

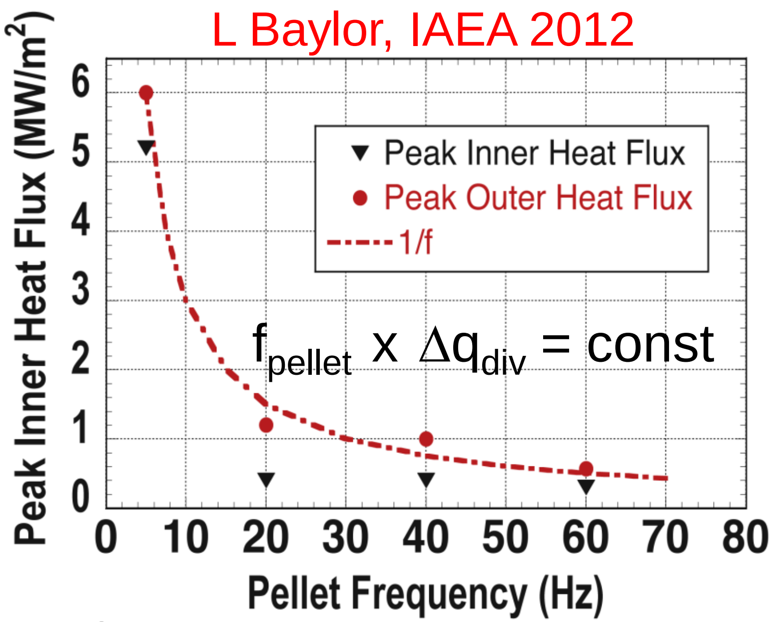
- Suppression seen very recently on KSTAR with n = 1, +90º phasing
- DIII-D - suppression at n = 3 and now suppression at n = 2 at low collisionality (but low density)
- AUG - suppression at n = 2
- JET – suppression at n = 2 (ex-vessel coils)
- MAST – mitigation (but not yet suppression) at n = 4 or 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

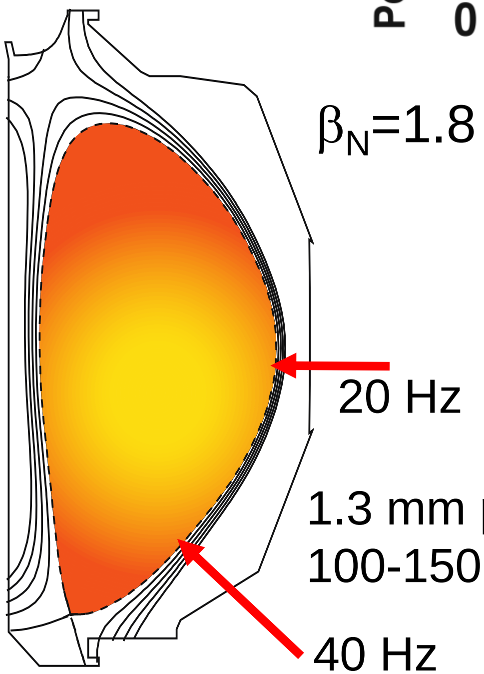


KSTAR, Y. Jeon, IAEA 2012

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





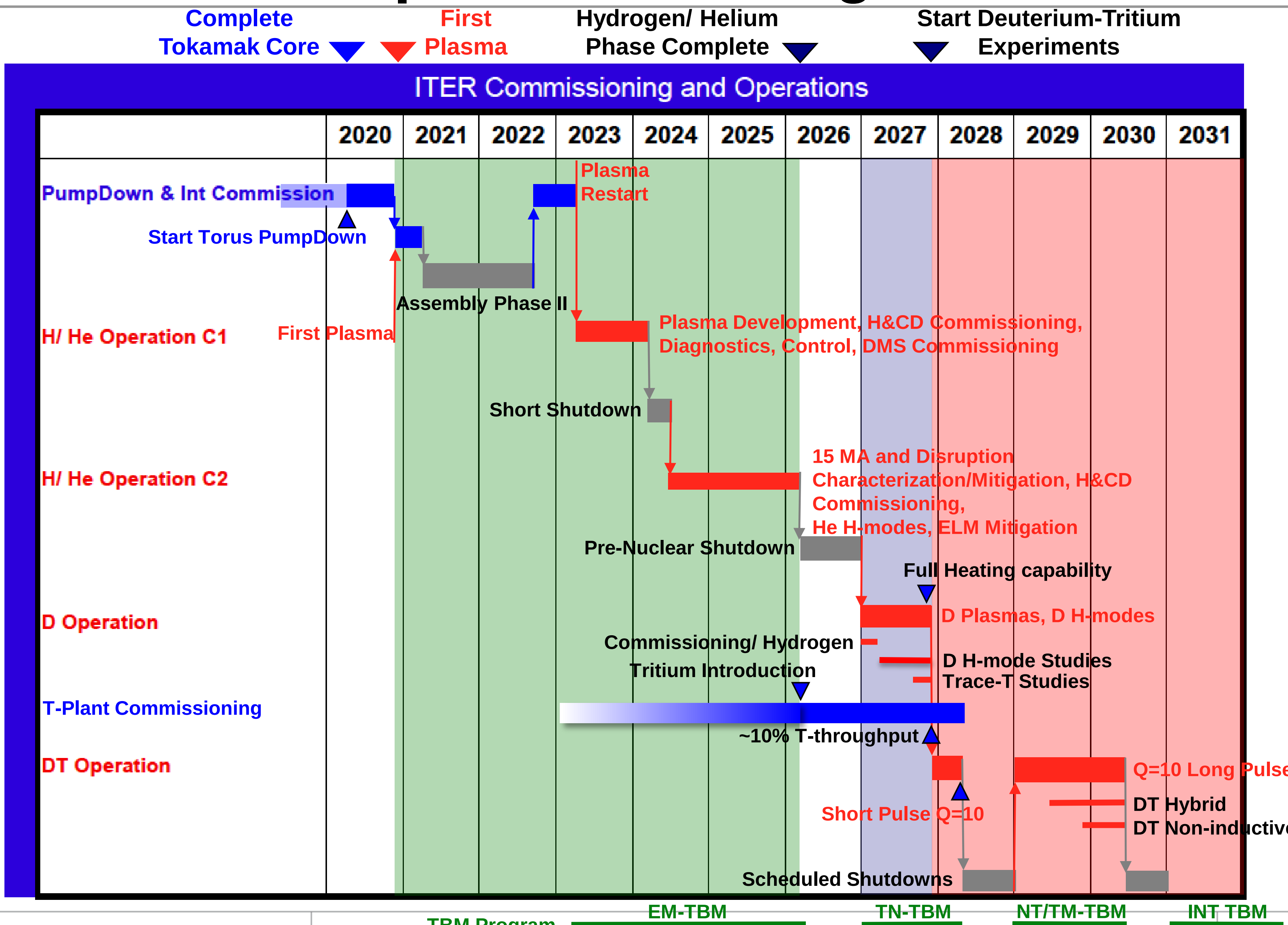
ITER shape, launch geometry

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

⇒ Fusion community is an integral part of the ITER project