RF Heating and Current Drive in ITER: Challenges and Needs

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- and many other colleagues in the ITER Organization, ITPA and the international fusion programme

Synopsis

- **The ITER Project:**
  - ITER mission goals
  - ITER Research Plan – rationale and structure
  - Baseline plasma scenarios
  - H&CD systems
- **Role and challenges for RF H&CD in fulfilling ITER's mission:**
  - Plasma initiation and formation
  - Achieving H-mode
  - Establishing advanced plasma scenarios
  - Plasma stability control
- **ITER construction status**
- **Conclusions**

The ITER Project

ITER Mission Goals

Physics:

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

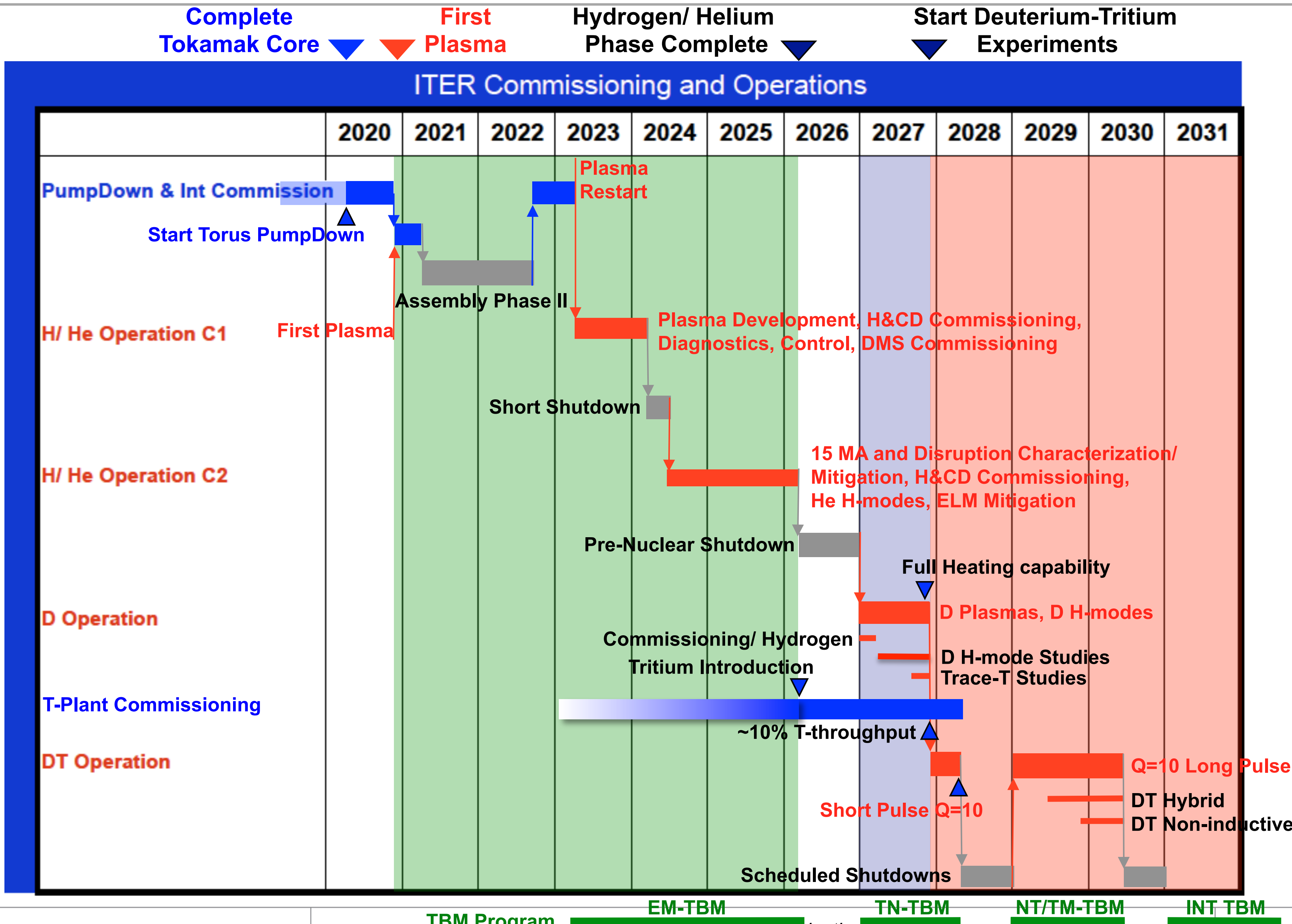
Technology:

- 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



ITER Research Plan – Structure

2020 | 2021 | 2022 | 2023 | 2024 | 2025 | 2026 | 2027 | 2028 | 2029 | 2030

Hydrogen/ Helium

- Develop plasma scenarios
- Commission Control, H&CD, Diagnostics
- Disruptions, Disruption Mitigation
- H-modes (helium)
- ELM control

Deuterium

- Deuterium H-modes
- DT scenario preparation
- Trace tritium experiments

DT

- Fusion power production
- Q=10
- Towards long pulse

more DT

Towards the future

ITER Scenarios

- **Baseline scenarios:**

Single confinement barrier

- ELMy H-mode:
  - Q=10 for ≥300s
  - well understood physics extrapolation to:
    - control
    - self-heating
    - α-particle physics
    - divertor/ PSI issues
  - physics-technology integration
- Hybrid:
  - Q=5 - 50 for 100 - 2000s
  - conservative scenario for technology testing
  - performance projection based on extension of ELMy H-mode

- **Advanced scenarios:**

Multiple confinement barriers

- satisfy steady-state objective
- prepare DEMO
- develop physics in a range of scenarios:
  - extrapolation of regime
  - self-consistent equilibria
  - MHD stability
  - controllability
  - divertor/ impurity compatibility
  - satisfactory α-particle confinement

ITER Reference Scenarios

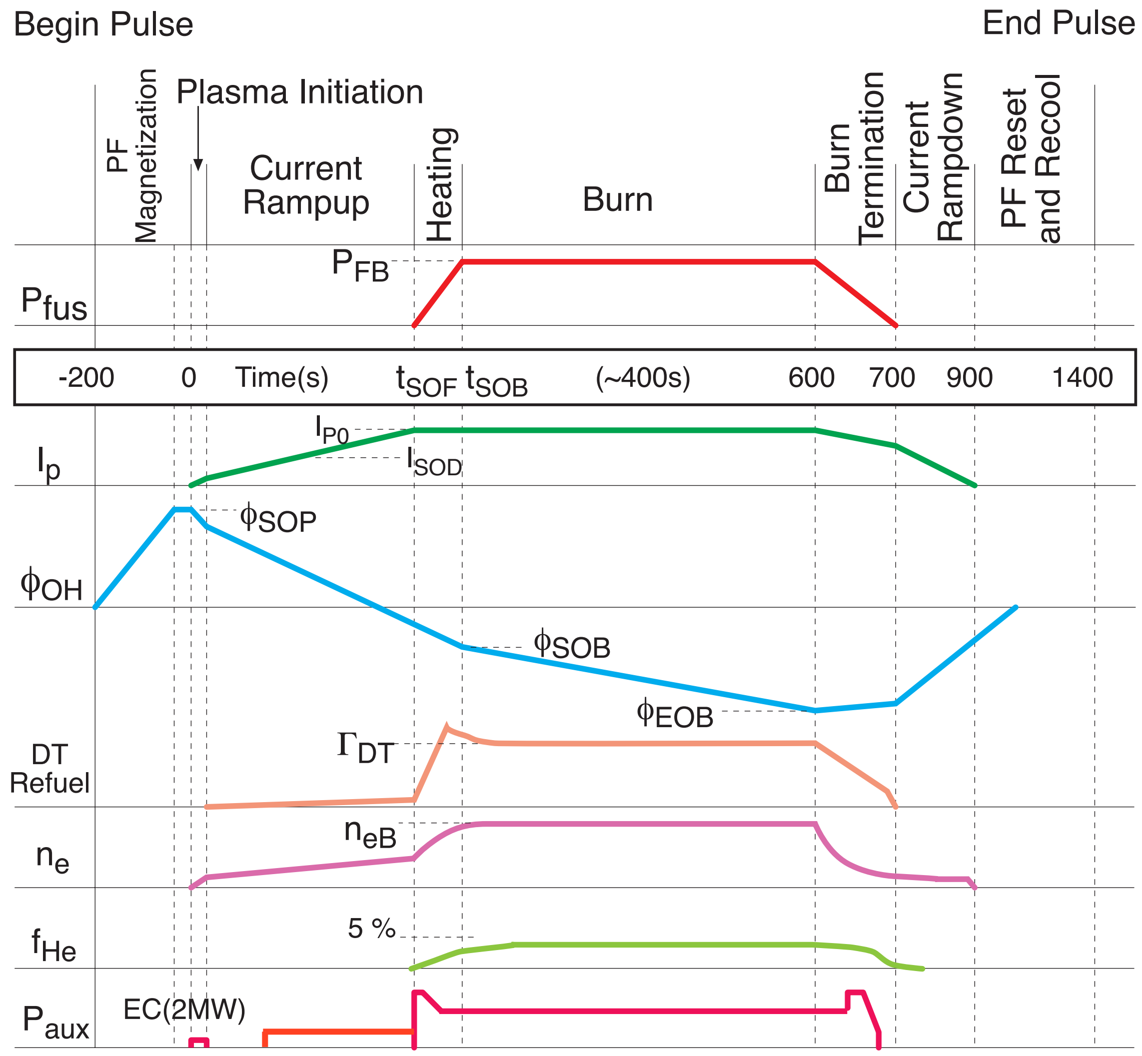
- The set of DT reference scenarios in ITER is specified via illustrative cases in the *Project Requirements*:

Parameter	1. Inductive operation	2. Hybrid operation	3. Non-inductive operation
R/a (m/m)	6.2 / 2.0	6.2 / 2.0	6.35 / 1.85
Toroidal field, B_T (T)	5.3	5.3	5.18
Plasma current, I_P (MA)	15.0	13.8	9.0
Elongation, κ_x/κ_{95}	1.85 / 1.7	1.85 / 1.7	2.0 / 1.85
Triangularity, δ_x/δ_{95}	0.48 / 0.33	0.48 / 0.33	0.6 / 0.4
Fusion power, P_{fus} (MW)	500	400	356
P_{add} (MW)	50	73	59
Energy multiplication, Q	10	5.4	6
Burn time (s)	300 - 500	1000	3000
Minimum repetition time (s)	1800	4000	12000
Total heating power, P_{TOT} (MW)	151	154	130
L-H transition power, P_{L-H} (MW) (note 1)	76	66	48
Plasma thermal energy, W_{th} (MJ)	353	310	287
Maximum fuelling input (Pa-m^3/s)	200	160	120

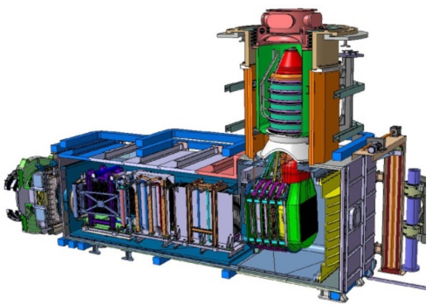
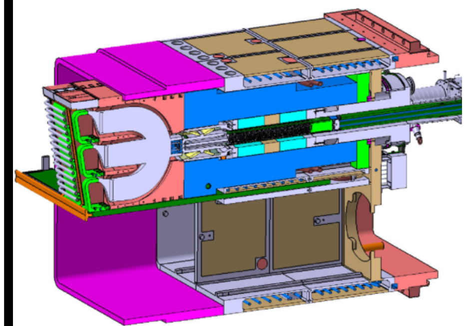
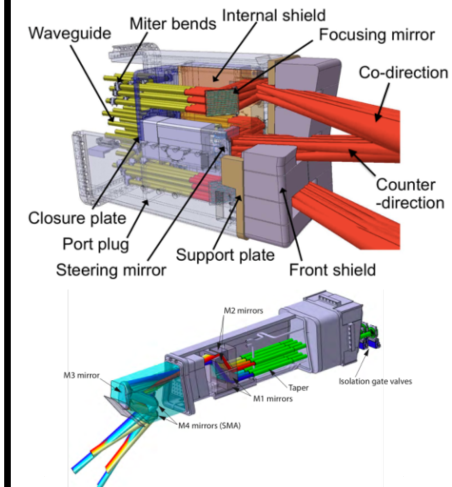
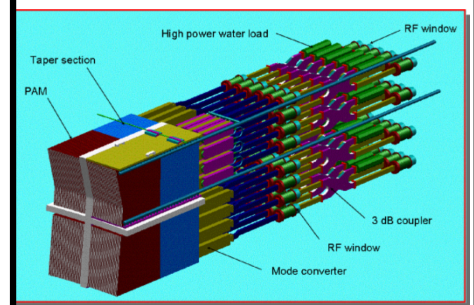
- In addition, a range of non-active (H, He) and D plasma scenarios must be supported for commissioning purposes to support rapid transition to DT operation

15MA Inductive Scenario - Schematic

- **Typical 15MA Q=10 inductive scenario has:**
  - current ramp-up phase of 70-100s
  - heating phase of ~50s
  - burn phase of 300-500s
  - shutdown phase of 200-300s
- **Typical pulse repetition time ~1800s**
  - based on burn duty cycle of 25%



ITER Heating and Current Drive Systems

NB	IC	EC	LH
Neutral Beam - 1 MeV	Ion Cyclotron 40-55MHz	Electron Cyclotron 170GHz	Lower Hybrid ~5 GHz
			
33MW* +16.5MW#	20MW* +20MW#	20MW* +20MW#	0MW* +40MW#
Bulk current drive limited modulation	Sawtooth control modulation < 1 kHz	NTM/sawtooth control modulation up to 5 kHz	Off-axis bulk current drive

***Baseline Power**

#Possible Upgrade

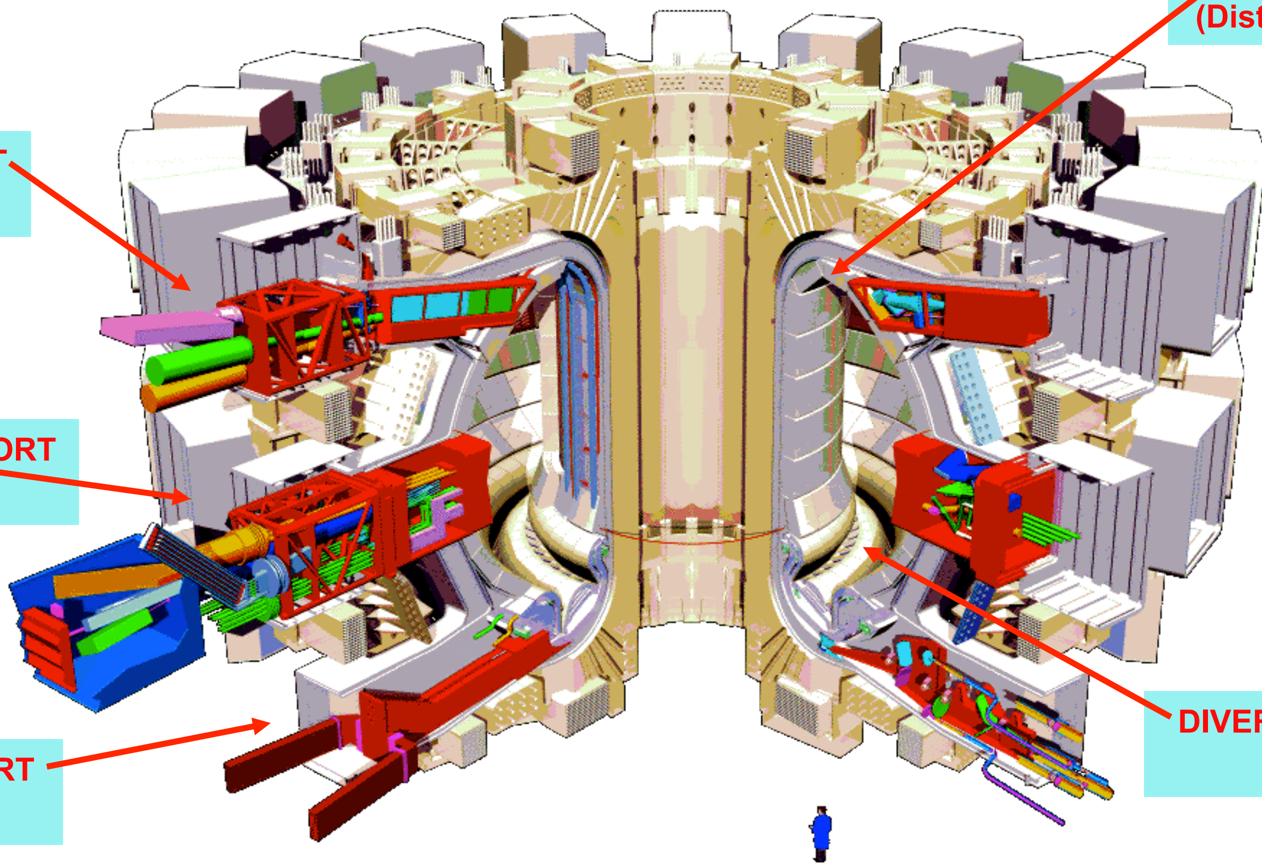
ITER Heating and Current Drive

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



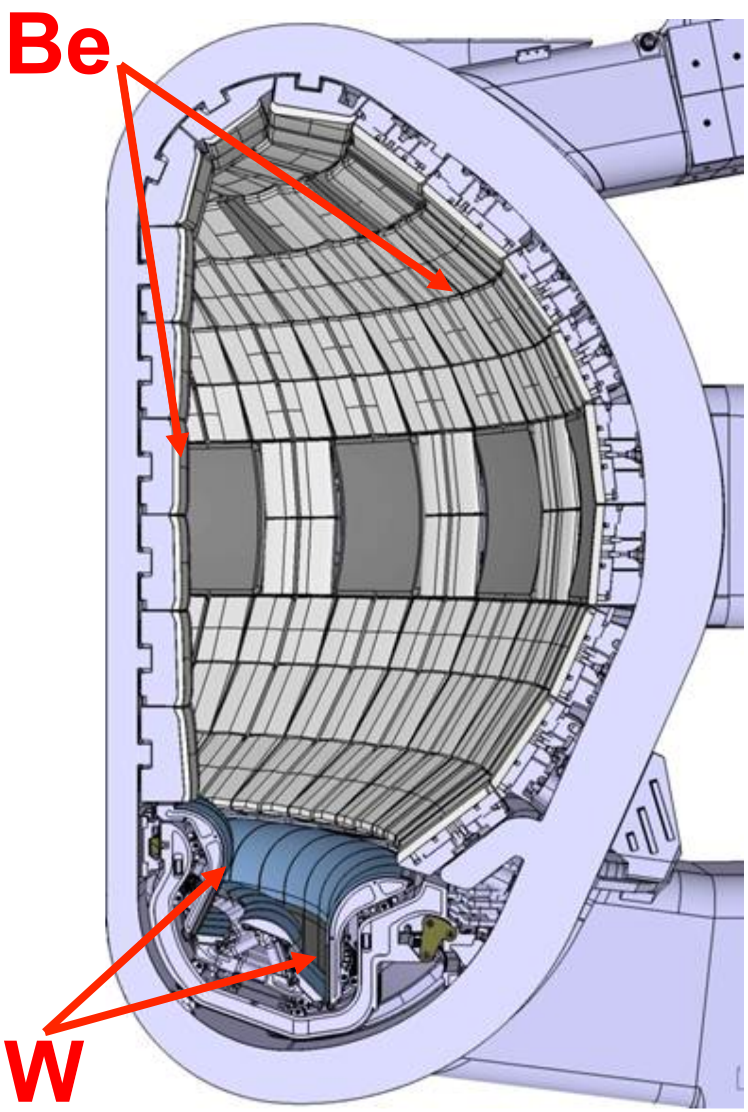
- **About 40 large scale diagnostic systems are foreseen:**
  - Diagnostics required for protection, control and physics studies
  - Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  - Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ….)

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

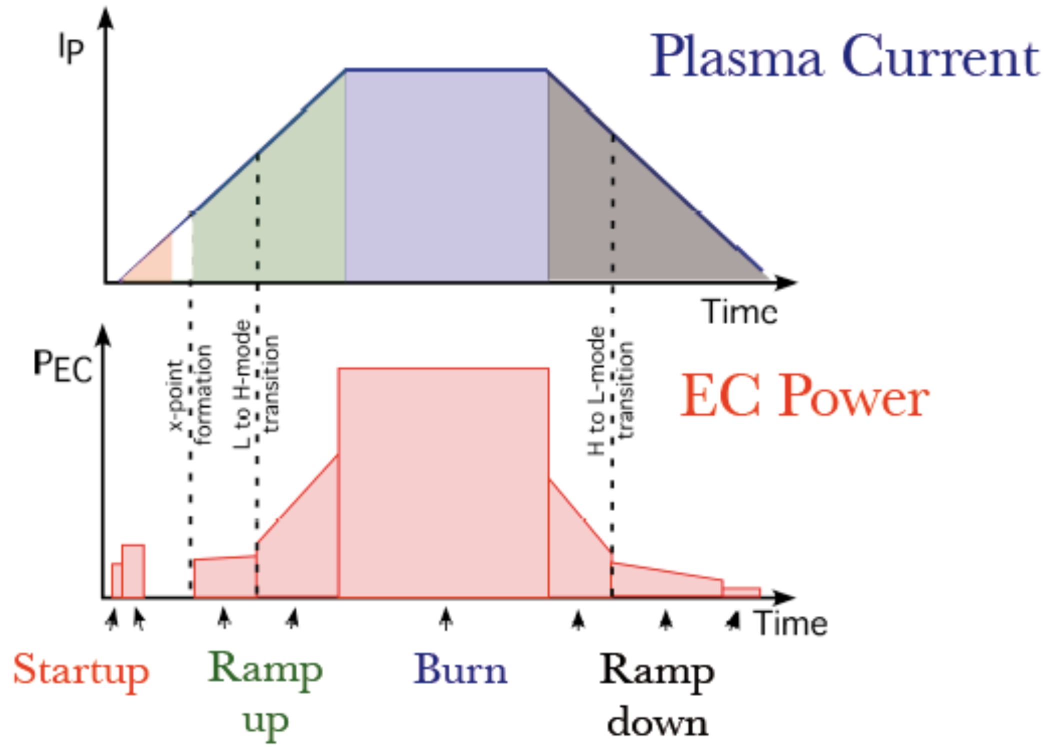
- **W divertor (~150m²):**
  - resistant to sputtering
  - limits fuel retention (but note Be)
  - melting at ELMs, disruptions, VDEs
  - W concentration in core must be held below ~ 2.5×10^{-5}



ITER H&CD Systems - Functions

- **Variety and total power of H&CD reflects required functionality:**
  - assist wall conditioning
  - support plasma breakdown and burnthrough
  - optimize current ramp-up and ramp-down (flux consumption, stability)
  - assure access to H-mode and heating to temperatures required for plasma burn
  - maintain burn point in DT plasmas
  - provide localized current drive for control of MHD instabilities (NTMs, sawteeth)
  - provide localized and global current drive to support establishment of advanced plasma scenarios with long-pulse capability

Example: ECRH in an Inductive Plasma



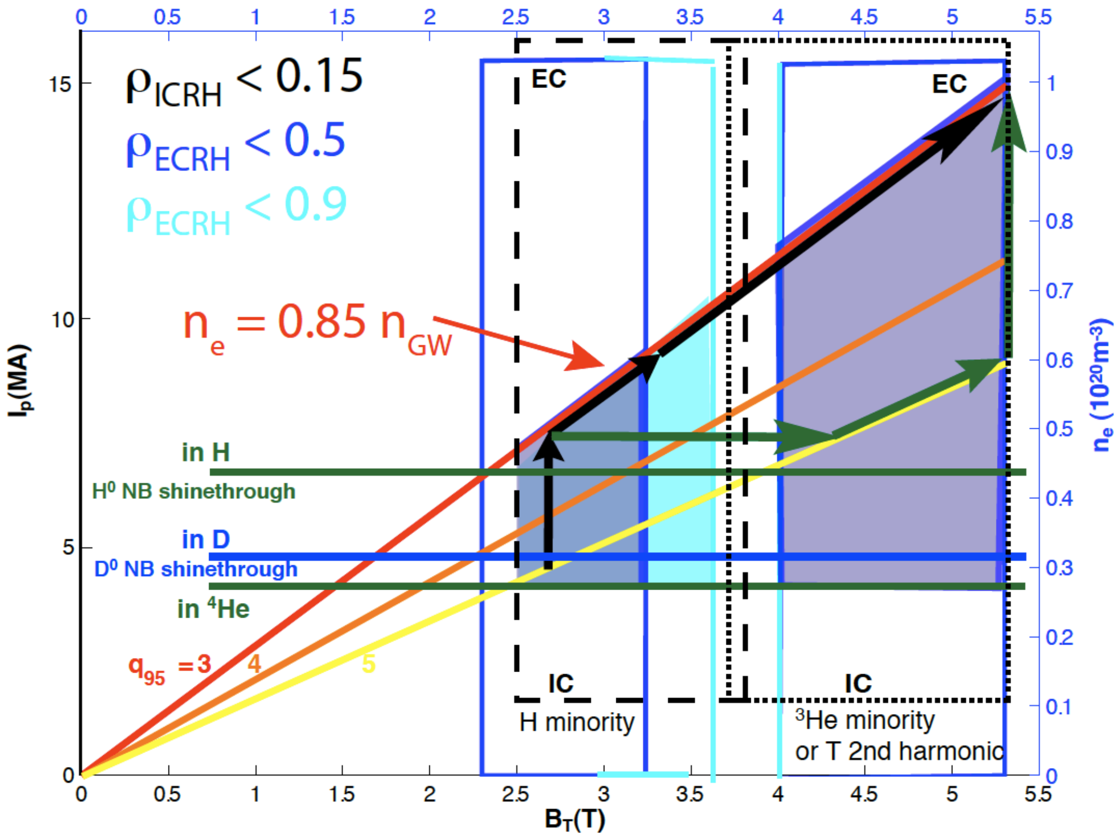
Start-up:
- Breakdown
- Burnthrough

Ramp-up:
- Current ramp-up assist
- L-to-H transition
- Central heating
- Profile tailoring

Burn:
- Heating
- Impurity control
- MHD control
- Profile tailoring
- Disruption mitigation
- ELM control

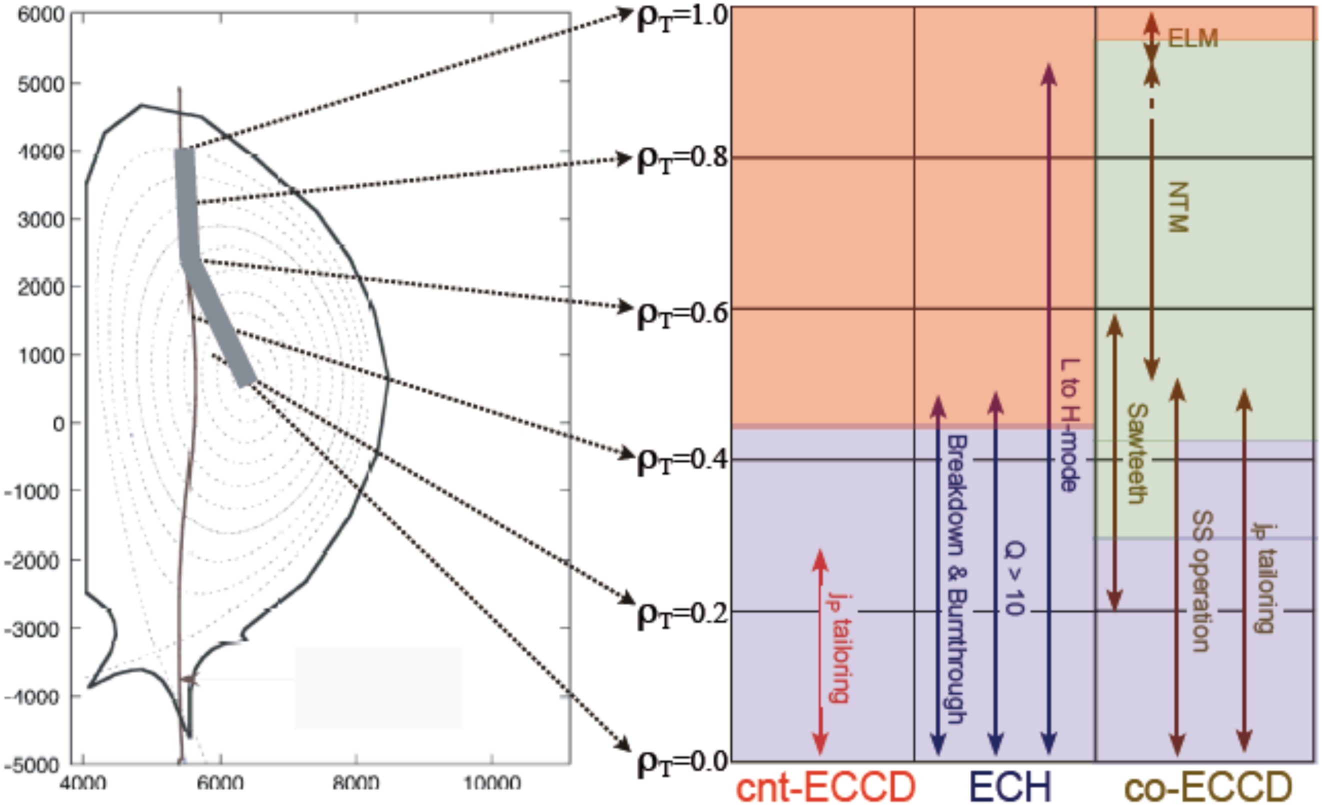
Ramp-down:
- H-to-L transition
- Current ramp-down assist
- Profile tailoring

H&CD System Flexibility



- **To satisfy the requirements of the ITER Research Plan, the H&CD systems must be capable of operating over a wide range of parameters:**
  - dark shaded region illustrates the range of toroidal fields over which a majority of injected power can be deposited at $\rho \leq 0.5$

ECH&CD Flexibility

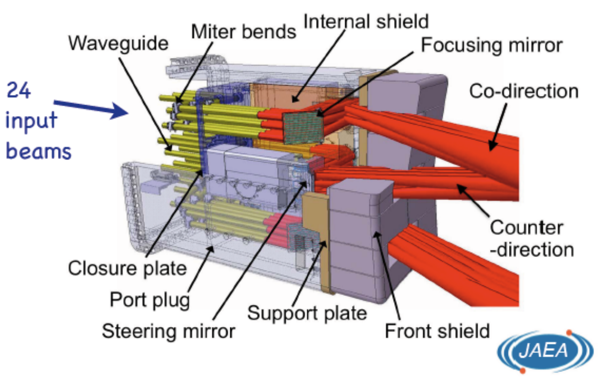


Upper Launcher:

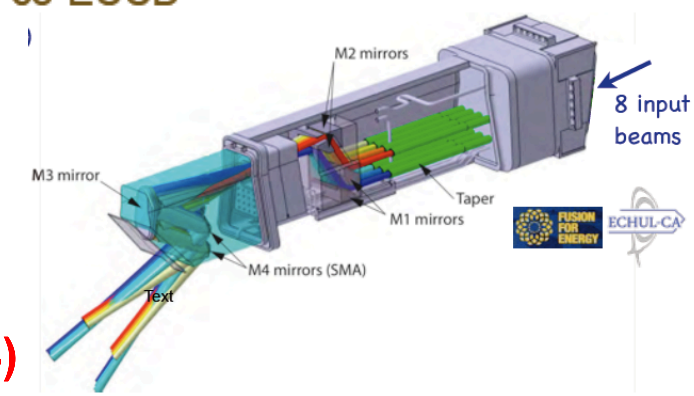
- access $\rho > 0.3$
- 20 MW co-ECCD
- MHD control
- mid-radius ECCD

Equatorial Launcher:

- access $\rho < 0.45$
- 13.3 MW co-ECCD
- 6.7 MW ctr-ECCD
- Central H&CD

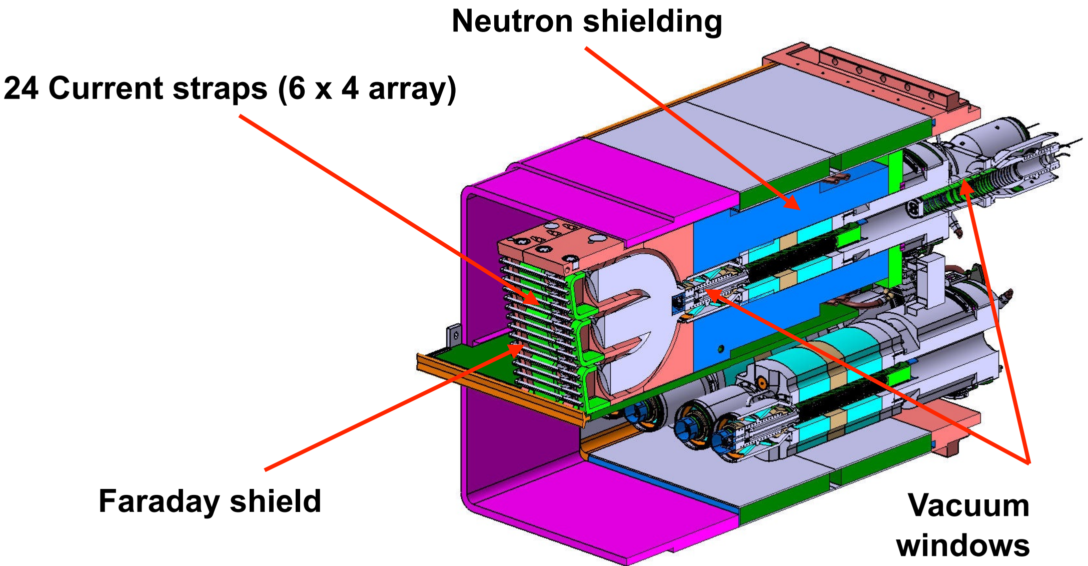


Equatorial Launcher (1)



Upper Launcher (4)

ICH&CD Flexibility



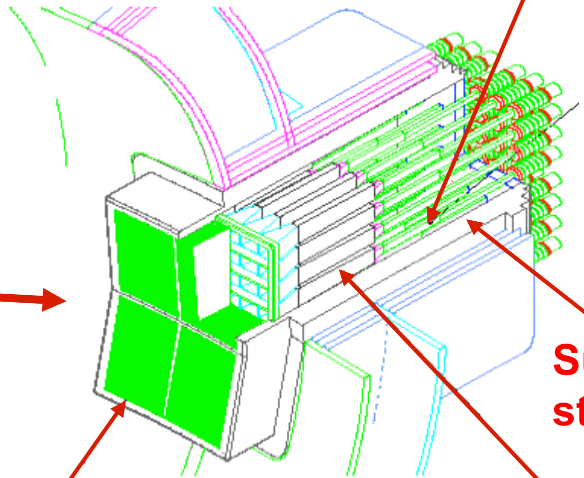
The baseline ICRF system must provide 20 MW of power from two antennas

Resonance	MHz	Comments
$2\Omega_T=\Omega_{3He}$	53	Second harmonic tritium + minority heating of ^3He to optimize ion heating
FWCD	55	On axis current drive
Ω_{3He}	45	Minority ion current drive at sawtooth inversion radius (outboard)
Ω_{3He}	40 - 55	Minority heating of ^3He in H, D, ^4He or DT (B_T= 3.7 to 5.3T)
Ω_H	40 - 55	Minority heating of H in D, He or DT at reduced magnetic field (2.5 to 3.8T)

Nominal B_T =5.3T (first three rows)

LHCD System (Possible Upgrade)

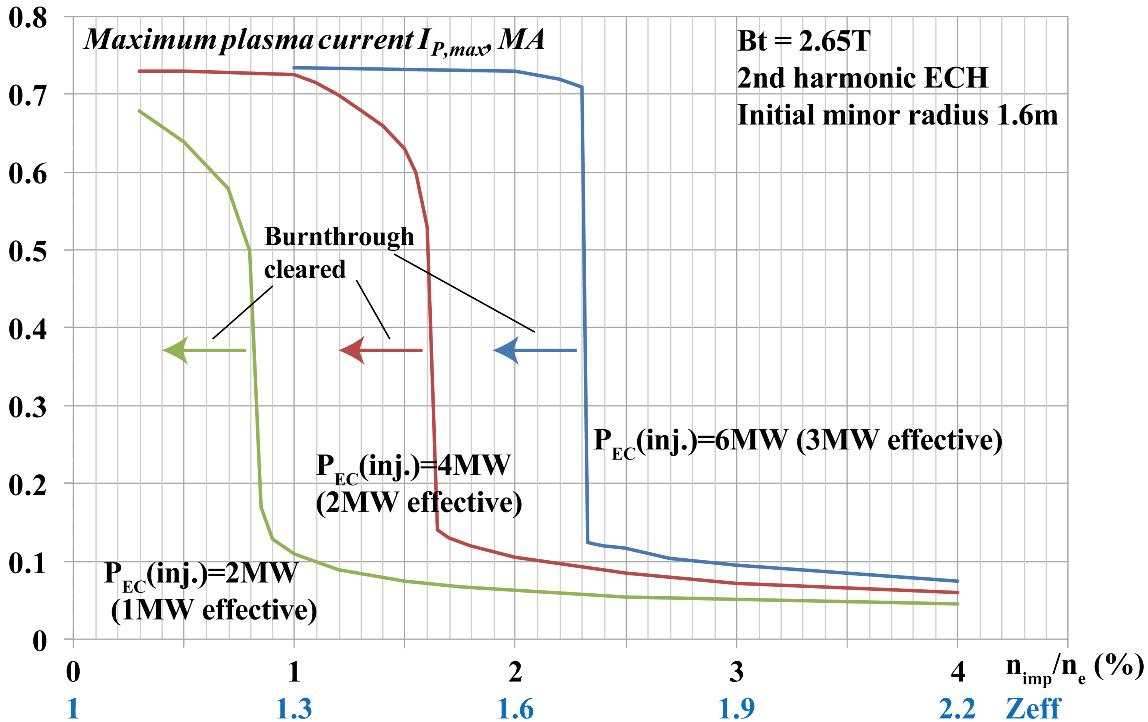
- **A lower hybrid current drive system has been proposed as a possible upgrade to ITER H&CD:**
  - highly efficient off-axis current drive would be beneficial in advanced scenarios
- **20MW 5GHz radiofrequency system:**
  - high power density RF multijunction launcher with vacuum transmission and matching components
- **Development of high power density passive/active multijunction (PAM) launcher is key R&D activity**
- **R&D on source and transmission components also required**



Role and Challenges for RF H&CD

ECRH-assisted Plasma Initiation

- **Plasma breakdown in non-active operation at B_T=2.65 T was assessed with the SCENPLINT code:**
  - showed strong dependence of achievable I_p on Z_{eff} and plasma size [1]
  - for 2nd harmonic, 50% efficiency assumed [2]: (6 MW equivalent to 3 MW at 5.3 T)
  - smaller initial plasma minor radius makes plasma initiation more difficult and decreases allowable impurity content

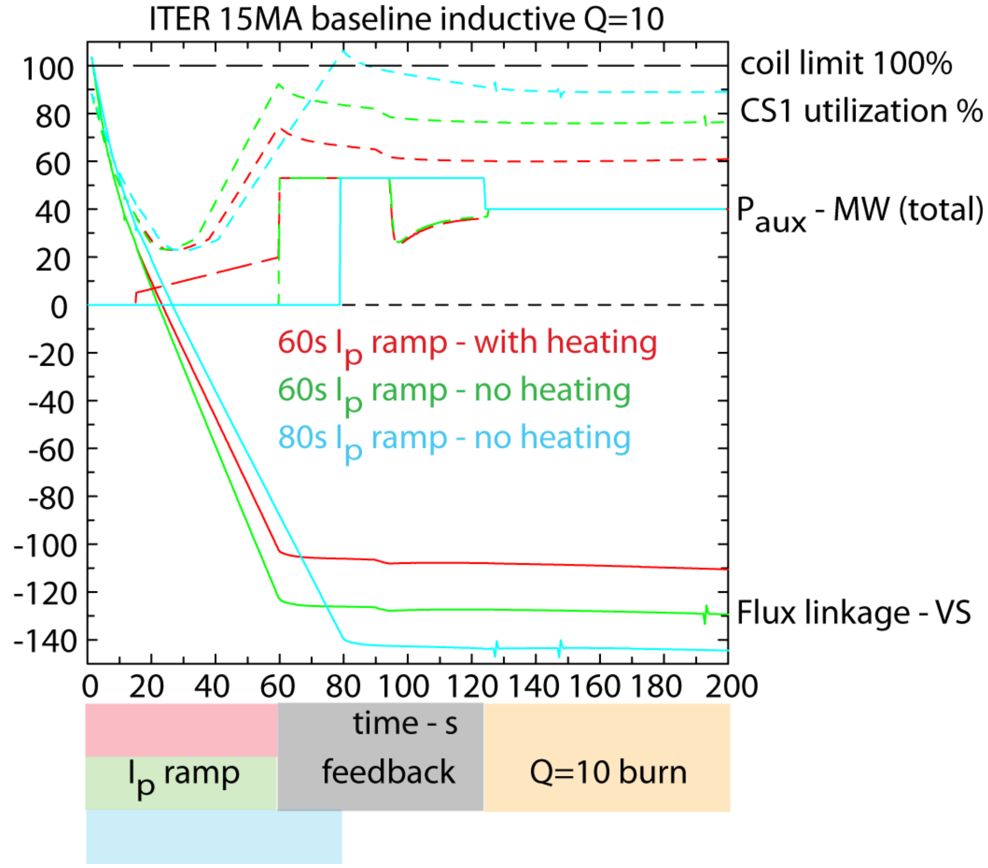


[1] A.A.Kavin, K.M.Lobanov, A.B.Mineev, Intermediate Report for contract "Study of plasma initiation with 0-D plasma transport and equilibrium models"

[2] K. Kajiwara et al. Nucl. Fusion 45 (2005) 694

Flux Saving during Current Ramp-up I

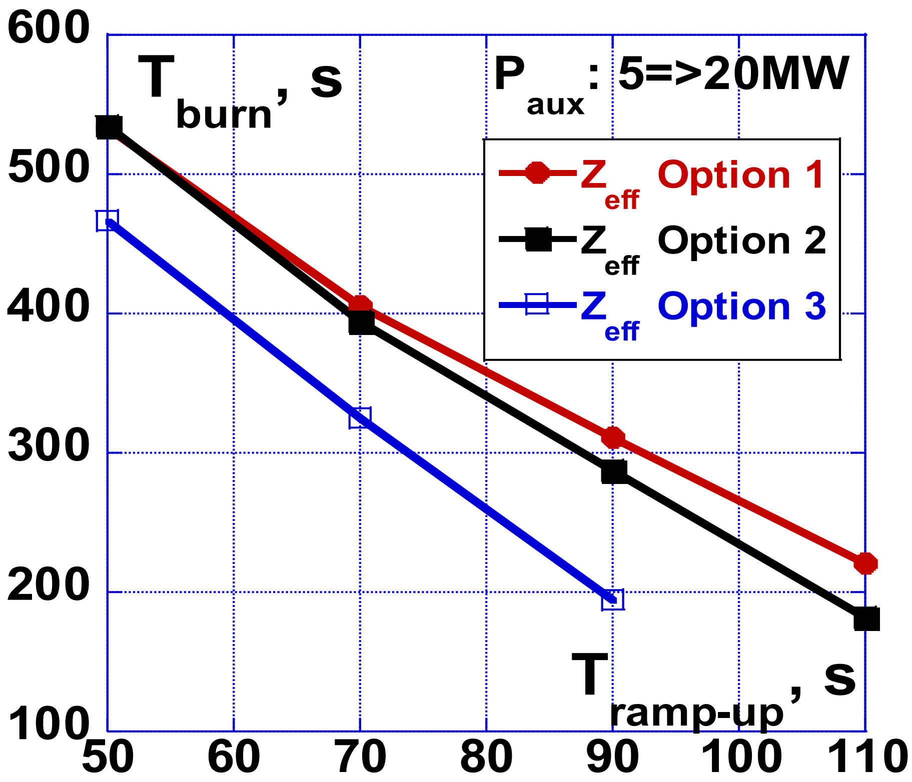
- **CORSICA simulations demonstrate ECRH during ramp-up enables faster current ramp-up with lower flux consumption and lower CS1 (magnetic flux) utilization:**
  - this provides a longer burning phase and a target q-profile with higher q_{min} favorable for sawtooth control and advanced operation scenarios
  - requirements on EC system: CW 5-20MW inside ρ ~ 0.4

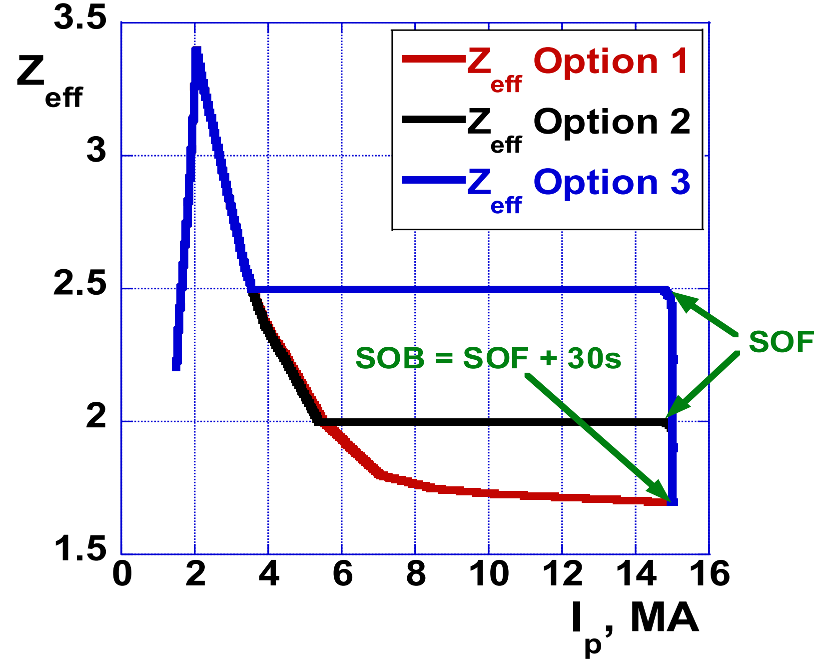


CORSICA - T Casper

Flux Saving during Current Ramp-up II

- **DINA (and CORSICA) simulations show sufficient capability to maintain Q_{DT} = 10 for 300-500 s as required by ITER mission goal**
  - DT scenario utilizes ramp-up assist with ECRH

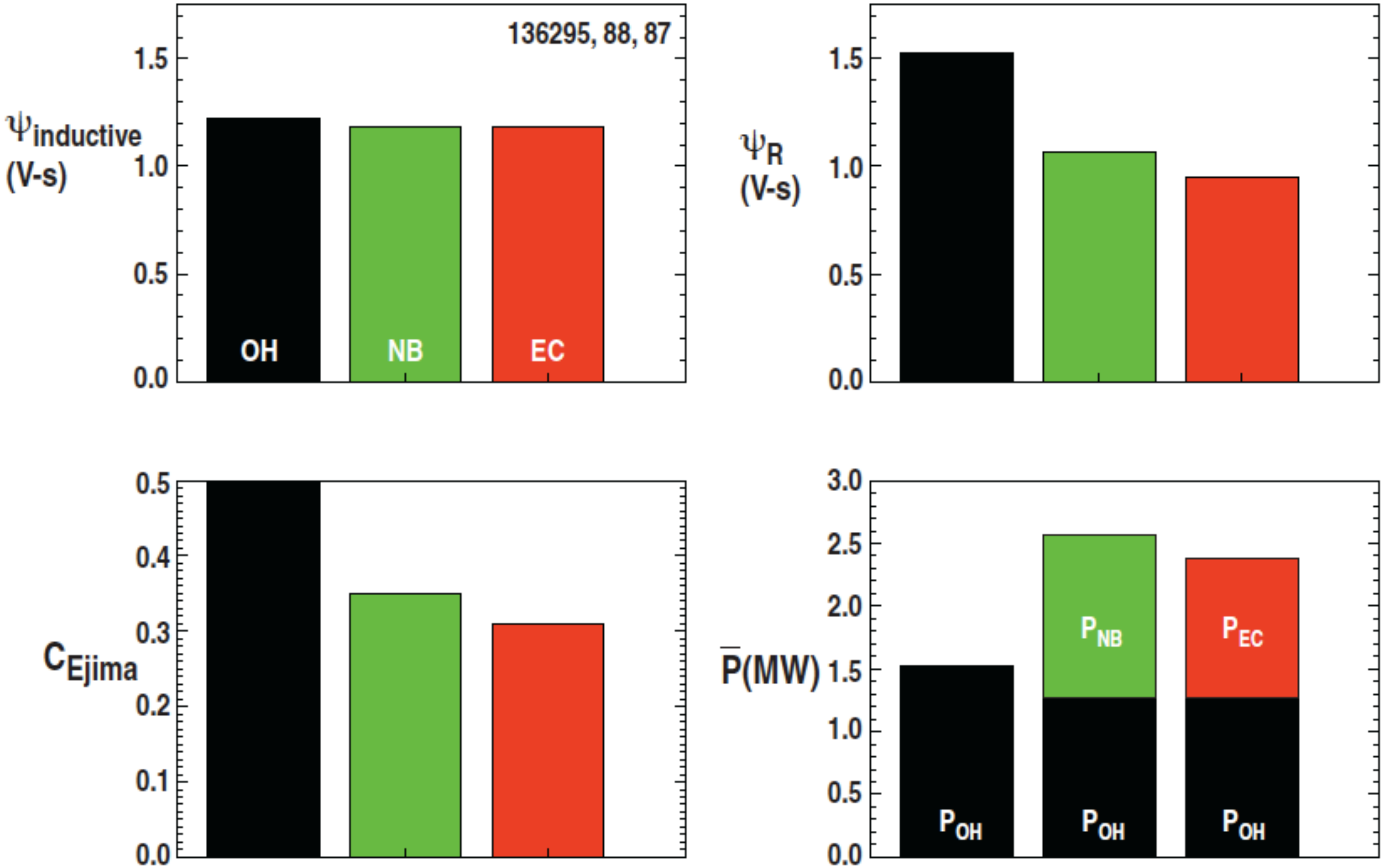




DINA – A Kavin et al

Flux Saving during Current Ramp-up III

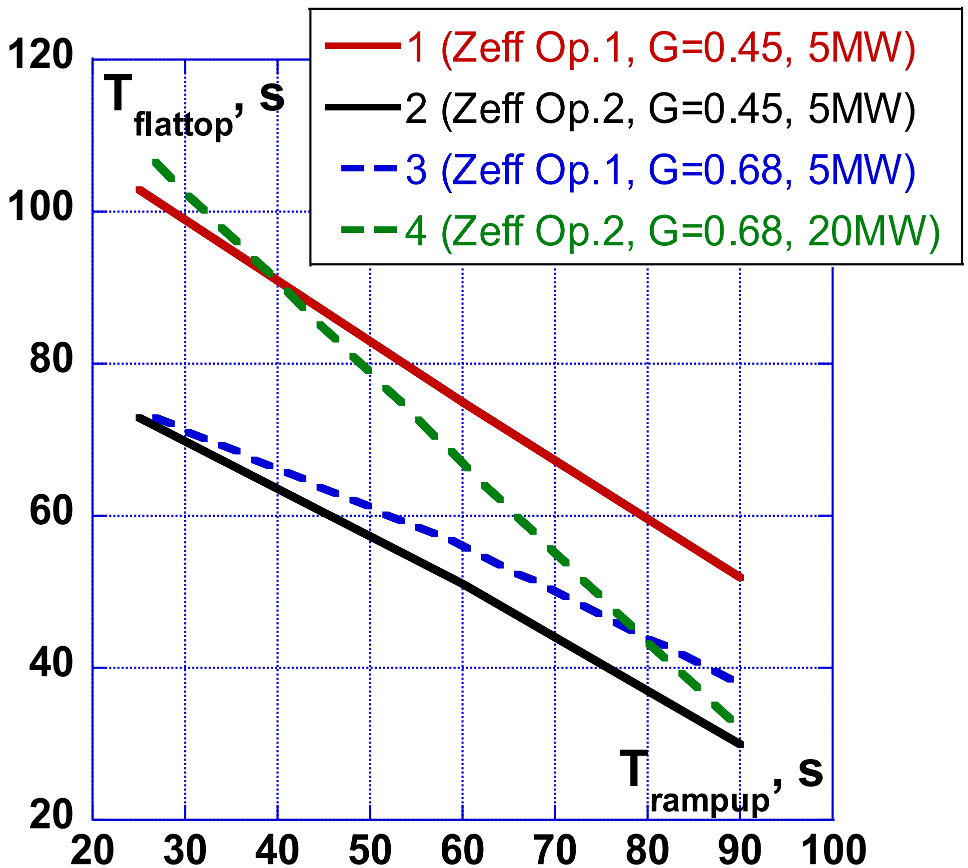
- **Such a reduction in flux consumption has been demonstrated in DIII-D ITER-simulation experiments**

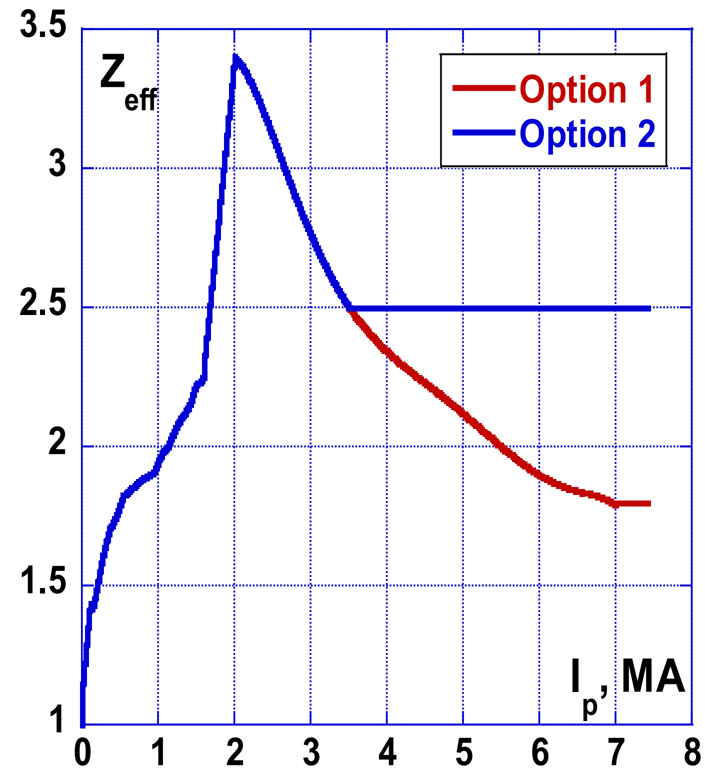


DIII-D – GL Jackson et al, APS-DPP Meeting, Orlando 2009

Flux Saving during Current Ramp-up IV

- **ECRH can play an important role in "conserving" flux consumption in ITER plasma scenarios:**
  - long 7.5MA (2.65T) L-mode hydrogen scenarios with 50% reduction of internal stress in all central solenoid conductors – contributes to fatigue lifetime





DINA – A Kavin et al

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

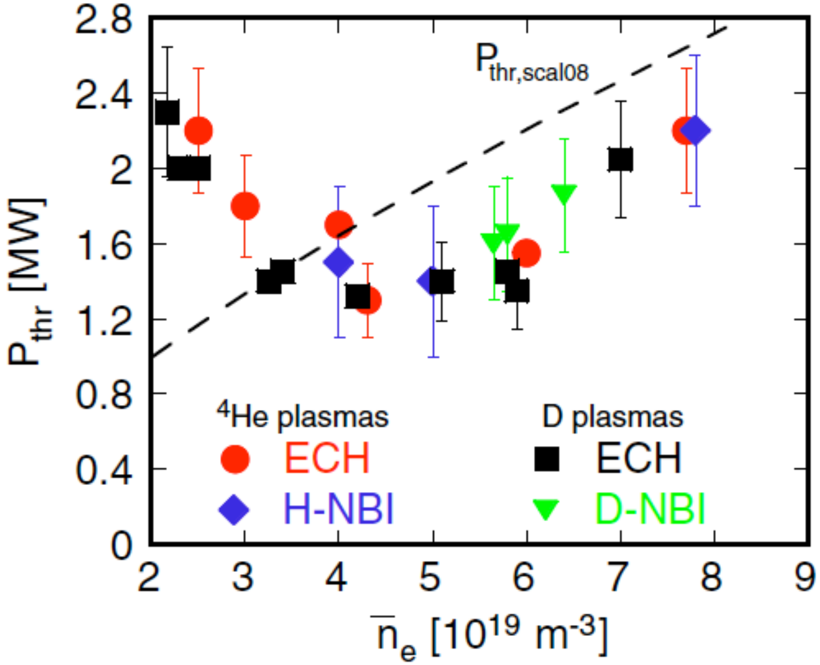
No H-mode access in H at full field

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

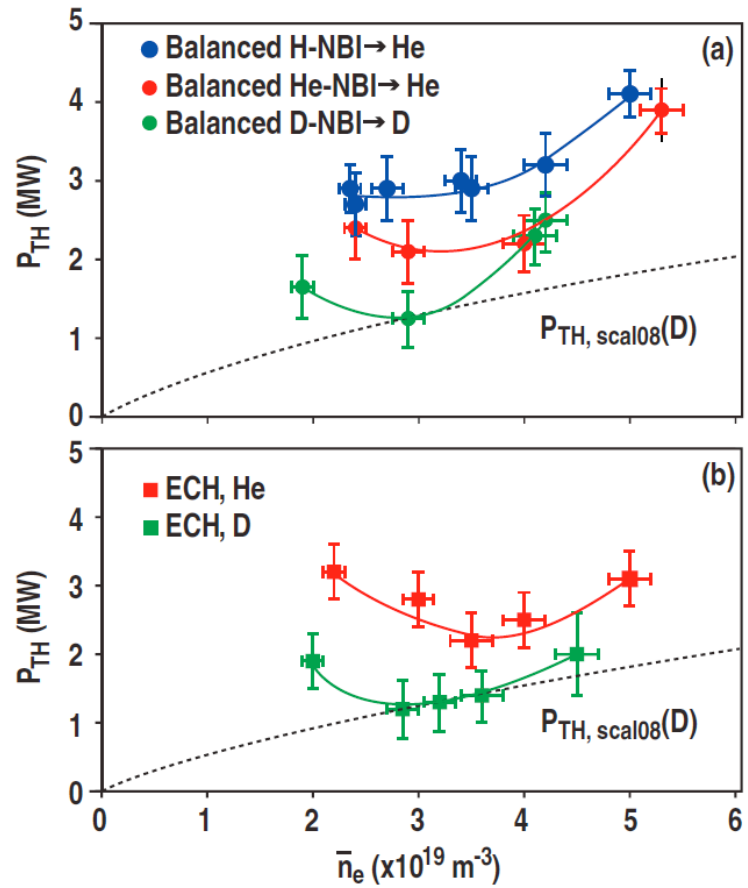
- **Note: margins may be required for (i) core radiation and (ii) access to good confinement (H_{98} = 1)**

H-mode Access with ECRH vs NBI

AUG, F. Ryter et al., NF 49 (2009) 062003



DIII-D, P. Gohil et al., NF 51 (2011) 103020



- **Recent H-mode power threshold experiments have compared the threshold with ECRH and NBI (and H, He, D):**
  - within the experimental uncertainties, the power threshold is found to be equal in the two cases

ITER H-mode Threshold - Implications

- **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
  - IRP 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

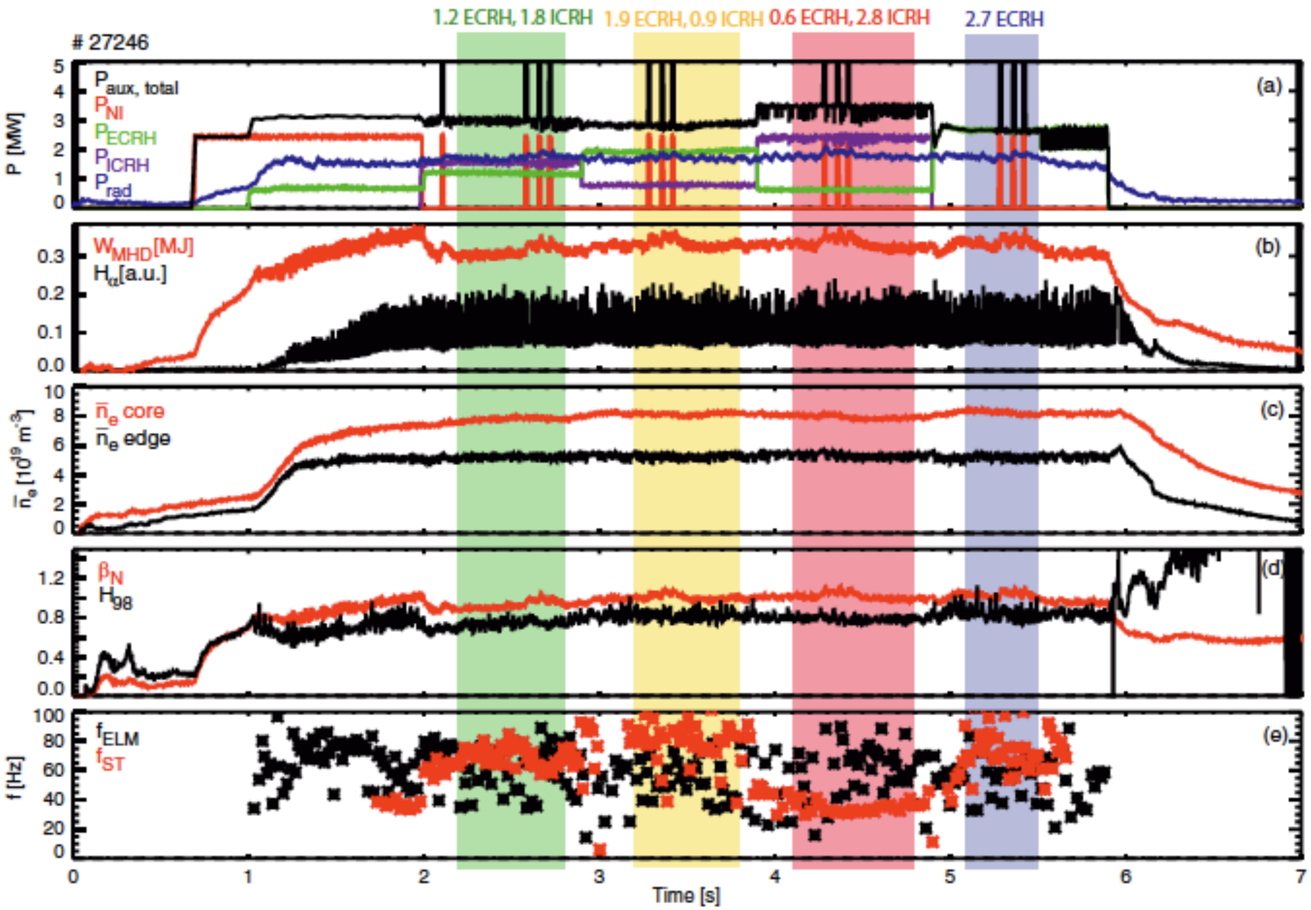
- **Deuterium Operations:**
  - at least 30 MW required for H-mode access at 7.5 MA/ 2.65 T

- **Deuterium-Tritium Operations:**
  - P_{inj} ~ 50 MW required for access to H-mode at 15 MA/ 5.3 T, but alpha heating power ensures H-mode operation at Q=10 operating point

⇒ It is essential that ECRH and ICRF couple power efficiently across a wide range of plasma operating conditions to support H-mode operation throughout development of Research Plan

H-mode Performance with ECRH vs ICRF

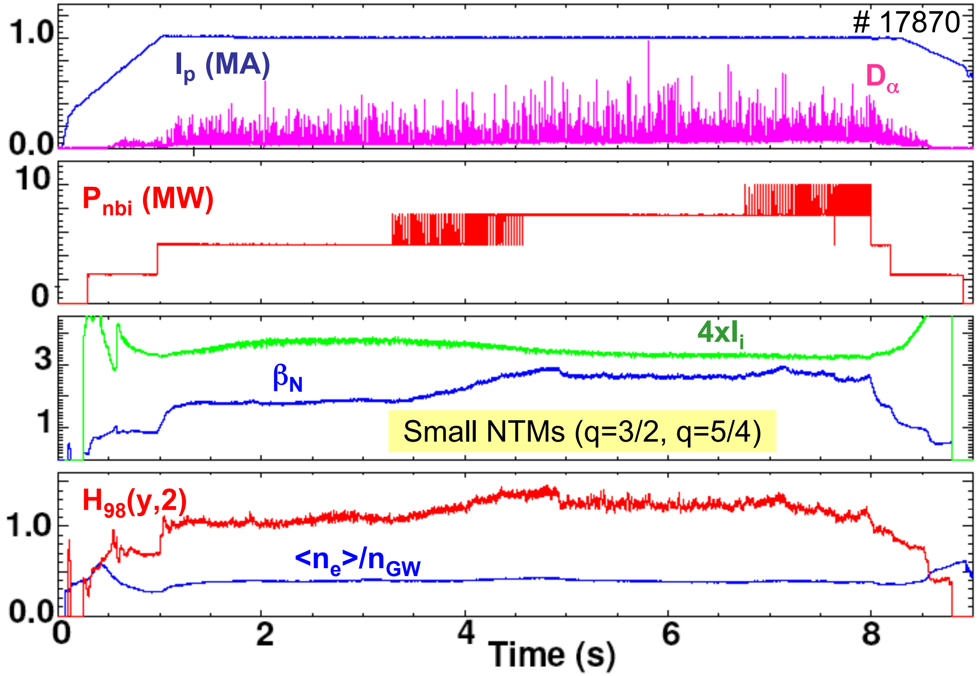
AUG, F. Sommer et al., NF 52 (2012) 114018



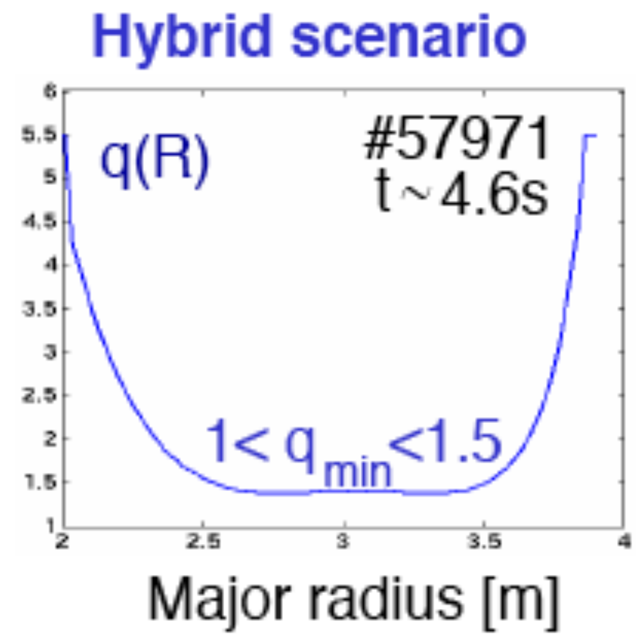
- **Experiments with ECRH and ICRF (and NBI) heated H-modes in AUG showed that, although detailed profile differences were observed, overall H-mode performance was very similar**

ITER Hybrid Scenario Operation

AUG - O Gruber et al, 46th APS-PPD Meeting, Savannah, 2004



JET - M L Watkins et al, 21 IAEA FEC, Chengdu, 2006

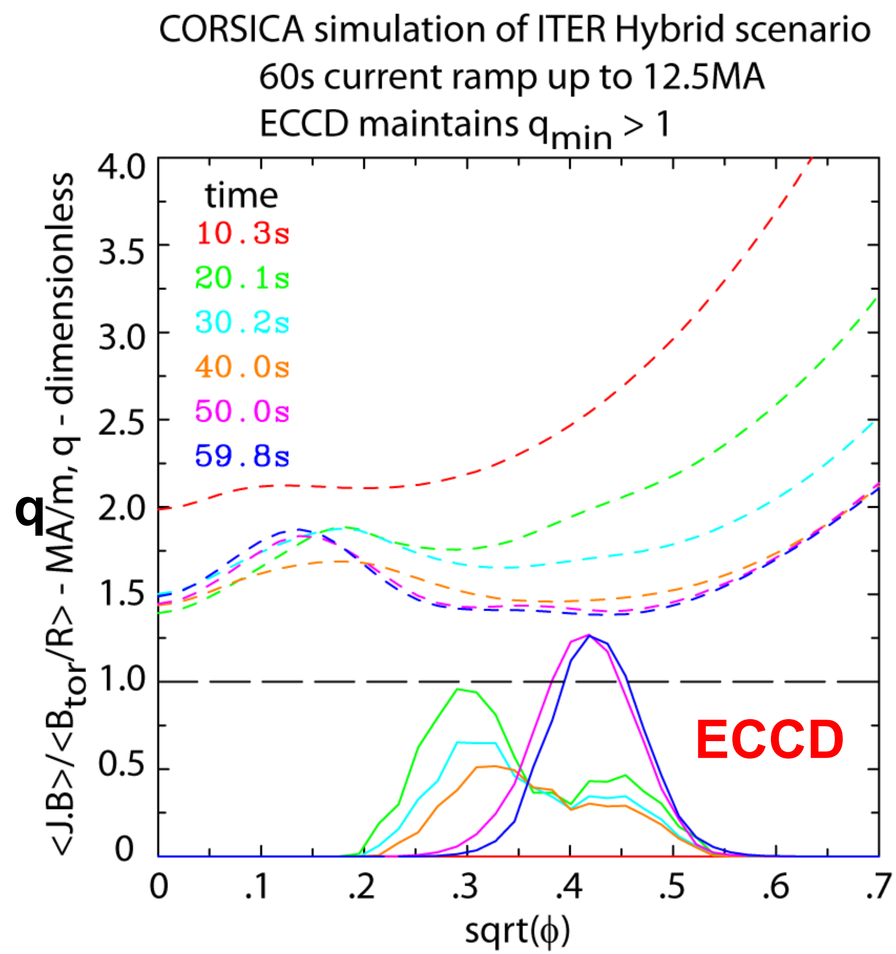


- **The so-called "hybrid" mode (improved H-mode) developed in recent years may allow ITER both to operate at higher fusion performance and for longer durations:**
  - flat central q-profile with q(0) ~ 1 appears critical
  - R&D is ongoing to demonstrate extrapolability of regime to ITER

q-Profile Control for Hybrid Scenarios

- **CORSICA has simulated a hybrid scenario using 20MW off-axis ECCD:**
  - sustains hybrid scenario with q_{min}>1 for > 400 s against current penetration and NBCD

- **The well-localized off-axis ECCD capability is a powerful tool for current profile control**

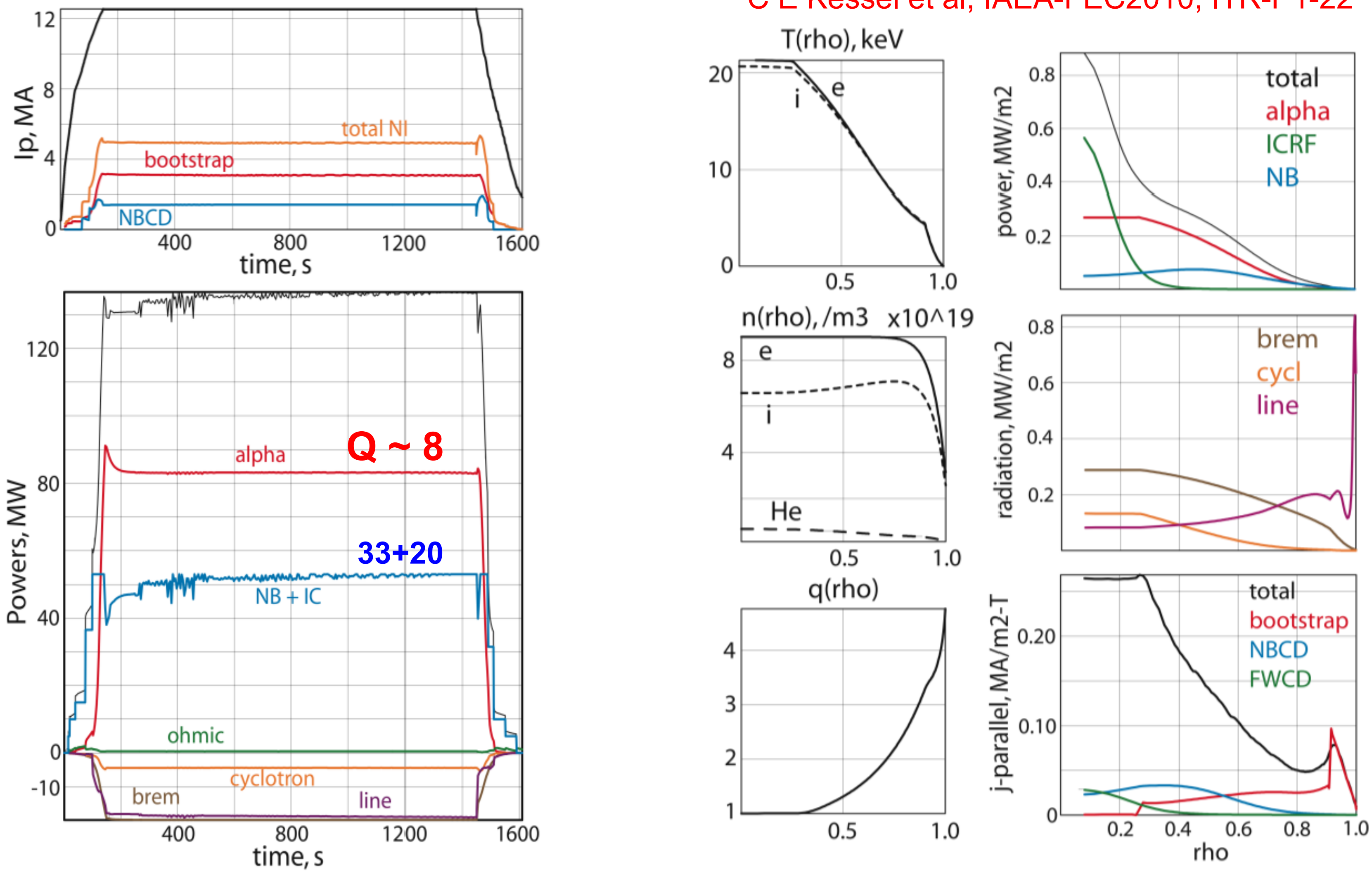


CORSICA - SH Kim and T Casper

End-to-End Hybrid Scenario

TSC

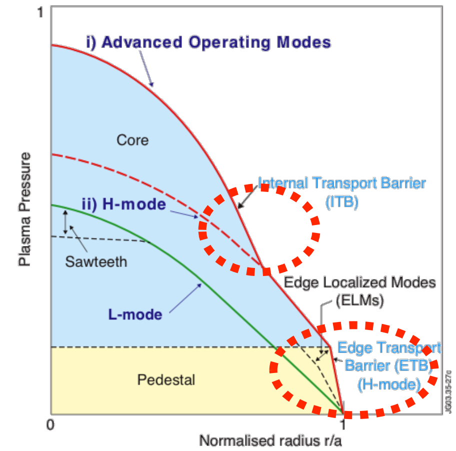
C E Kessel et al, IAEA-FEC2010, ITR-P1-22



- **Improved H-mode hybrid with burn duration of ~1300 s at I_p = 12 MA, H_{98} = 1.25, P_{aux} = 33 MW NB + 20 MW IC**

Steady-State Operation

- **Discovery of internal transport barriers ⇒ "advanced scenarios"**



plasma with reversed central shear + sufficient rotational shear

↓

internal transport barrier
⇒ enhanced confinement

↓

reduced current operation +
large bootstrap current fraction

↓

reduced external current drive +
current well aligned for mhd stability and confinement enhancement

↓

Steady-state operation +
High fusion power density

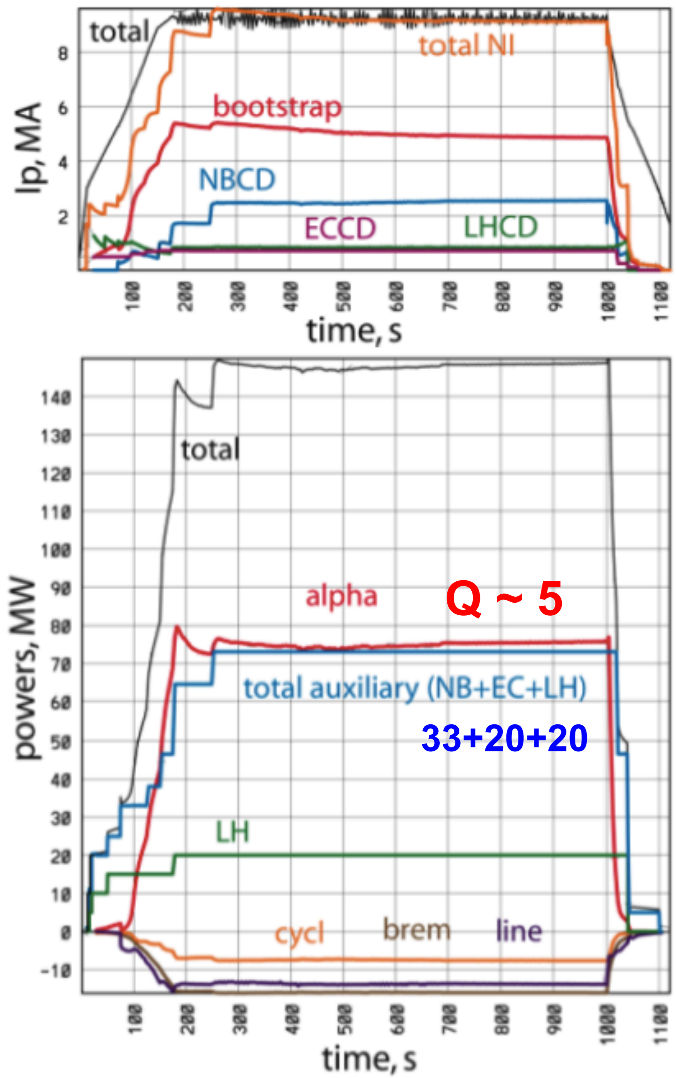
active mhd control

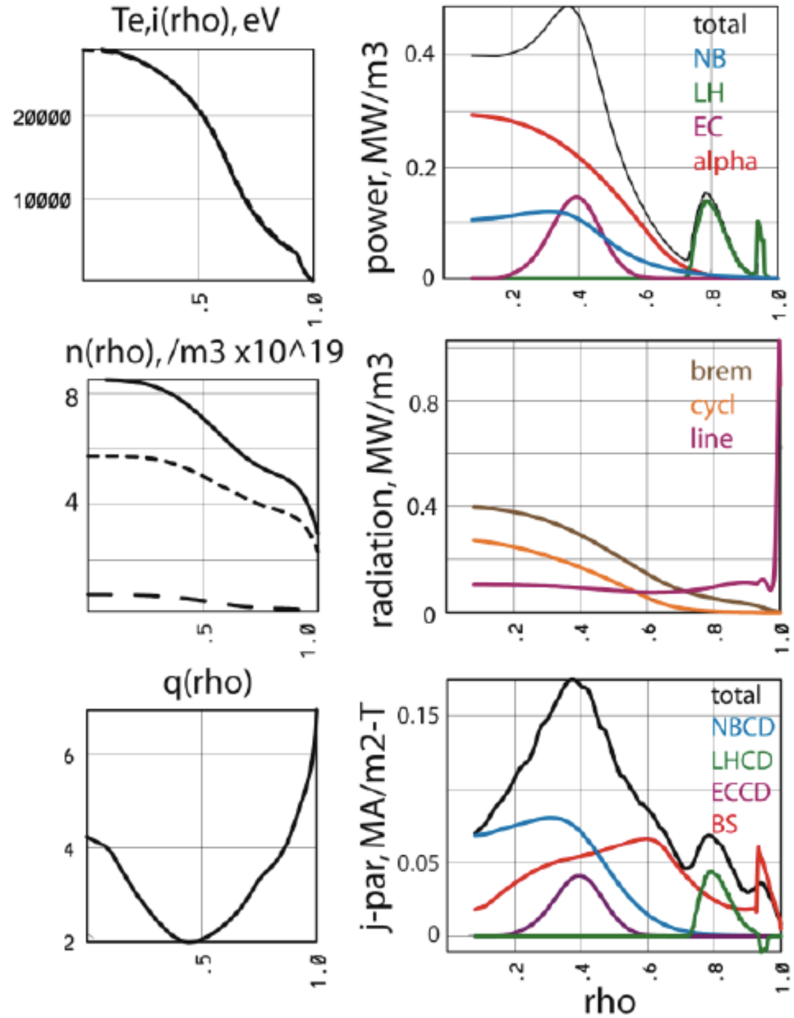
- **But development of an integrated plasma scenario satisfying all reactor-relevant requirements remains challenging**

End-to-End Steady-State Scenario

TSC

C E Kessel et al, IAEA-FEC2010, ITR-P1-22

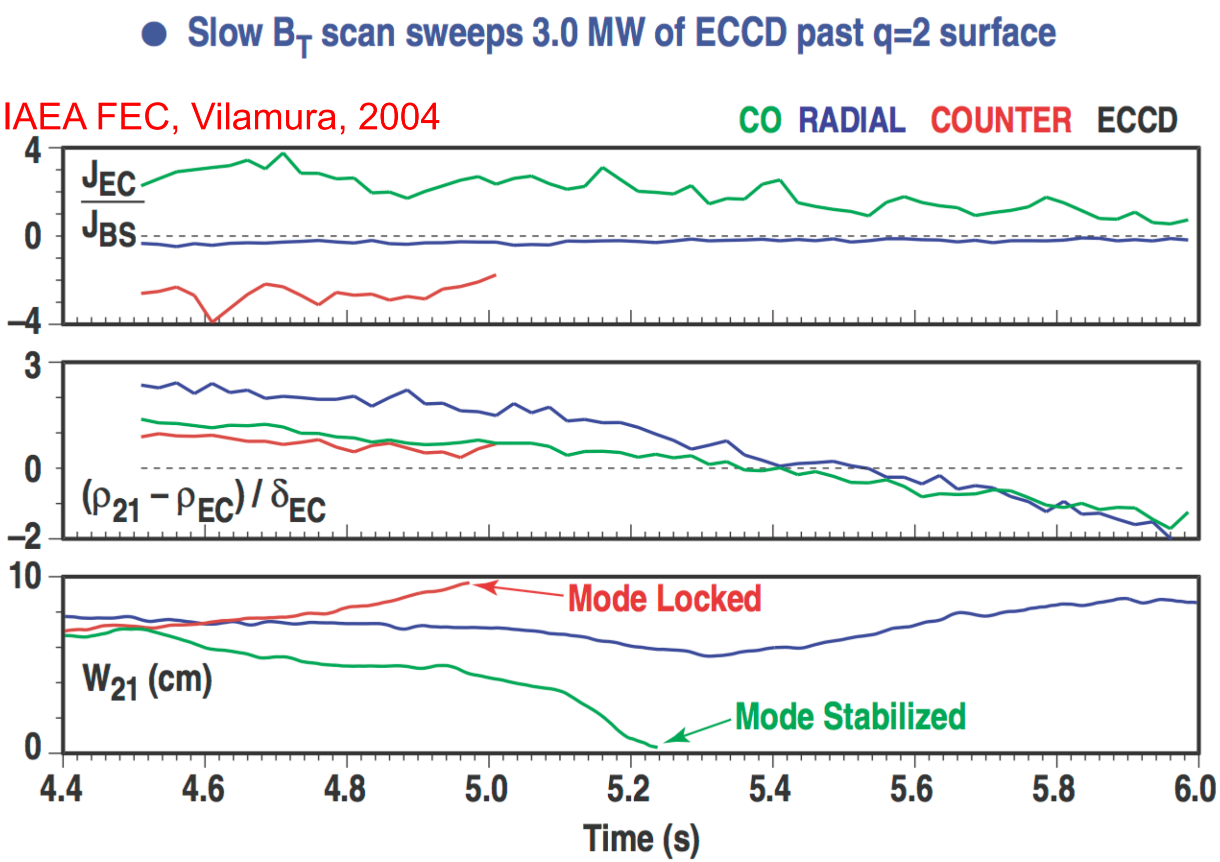




- **Fully non-inductive steady-state scenario at I_p = 9.25 MA, H_{98} = 1.7, β_N = 2.8, P_{aux} = 33 MW NB + 20 MW EC + 20 MW LH**

Neoclassical Tearing Mode Control by ECCD

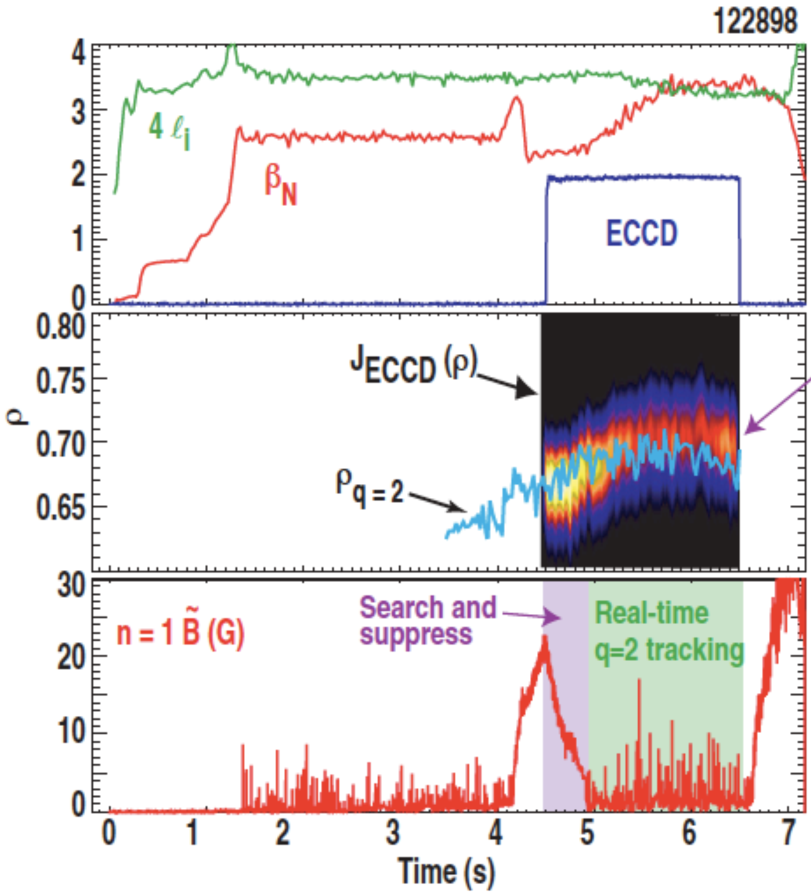
- NTMs are expected to place the principal limit on ITER performance in the Q = 10 inductive scenario: principal modes are m/n = 3/2 and 2/1
- DIII-D experiments demonstrated co-ECCD on q=2 surface has a stabilizing effect on m/n=2/1 NTM, while ECH does not have much effect
- co-ECCD capability is required for NTM control in ITER



CC Petty et al., 20th IAEA FEC, Vilamura, 2004

Control of NTMs: Steerable ECCD

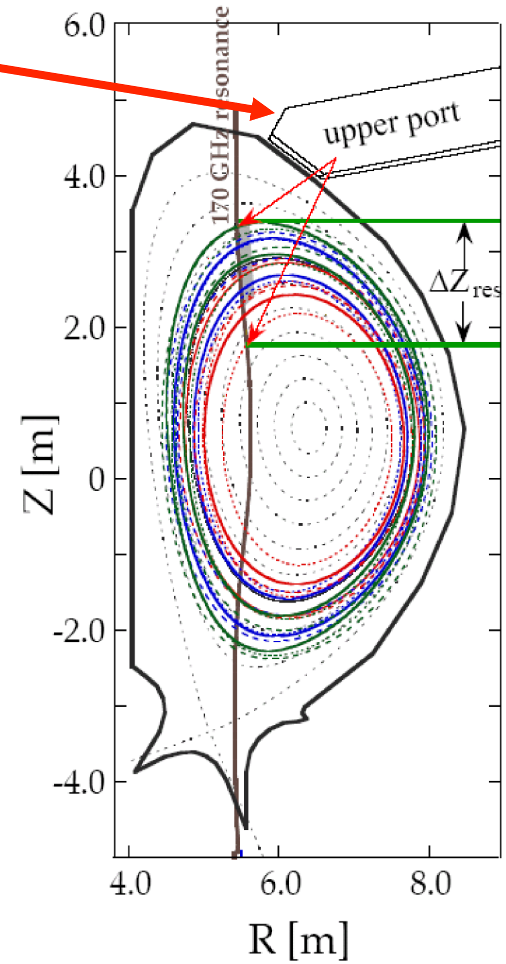
- **An MHD instability is detected (magnetically, SXR, ECE …):**
  - localized electron cyclotron current drive is used to suppress the instability
  - ITER has 4 steerable upper ECH&CD launchers launching 20 MW



M R Wade et al, FEC-2006, OV1-4

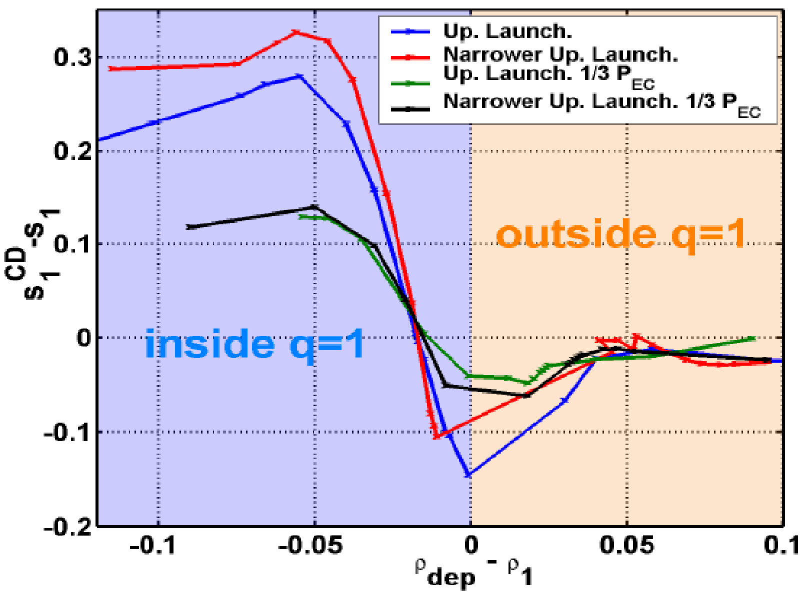
DIII-D

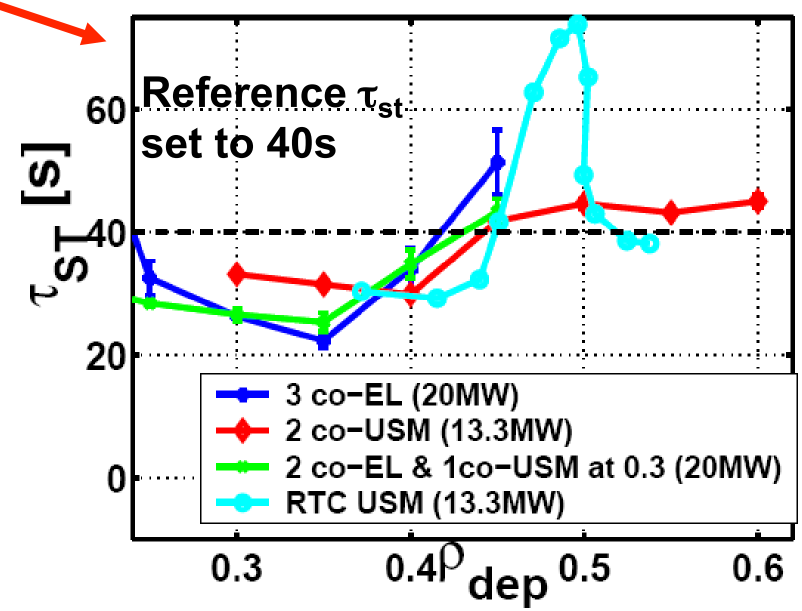
ECCD deposition location controlled by moving 2nd harmonic resonance



Control of Sawteeth I

- Sawtooth control is required to prevent island seeding that can trigger NTMs (and possibly also to control core impurity accumulation)
- When complete elimination of q=1 during burning phase is difficult, sawtooth destabilization using on-axis co-ECCD and off-axis ctr-ECCD is useful to limit sawtooth crash to allowable level for avoiding NTMs
- Simulations assuming different EC launchers show effectiveness of 13-20MW ECCD in affecting the sawtooth period τ_{st}





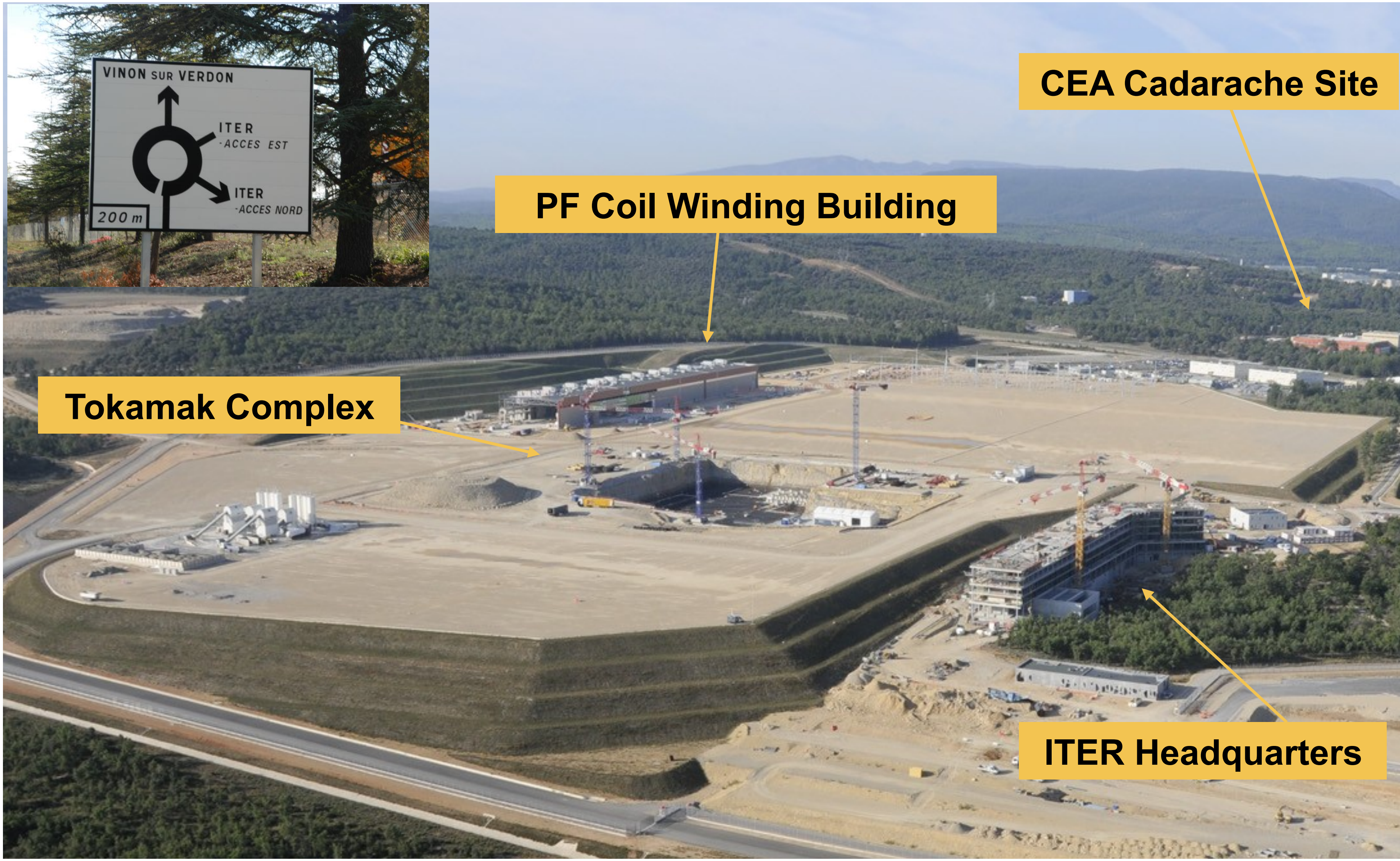
IT Chapman et al. ITPA MHD WG3 final report on power requirements sawtooth control in ITER

Control of Sawteeth II

- **Energetic ions, in particular, α-particles, are expected to stabilize sawteeth, causing longer periods between sawtooth crashes:**
  - this is known to enhance the seeding of NTMs at a sawtooth crash
- **A proposed technique for the "destabilization" of sawteeth is the use of co-ECCD inside the q=1 surface:**
  - analysis indicates that sufficient ECCD power is available in ITER to influence the sawtooth period
  - experiments (eg AUG) have confirmed the influence of ECCD on the sawtooth period in the presence of energetic ions (produced by ICRF)
- **An additional effect on the sawtooth period can be produced by ICRF deposition outside the q=1 surface in which energetic ions contribute to destabilization:**
  - a combination of the two effects under real-time control may be necessary in ITER

ITER Construction Status

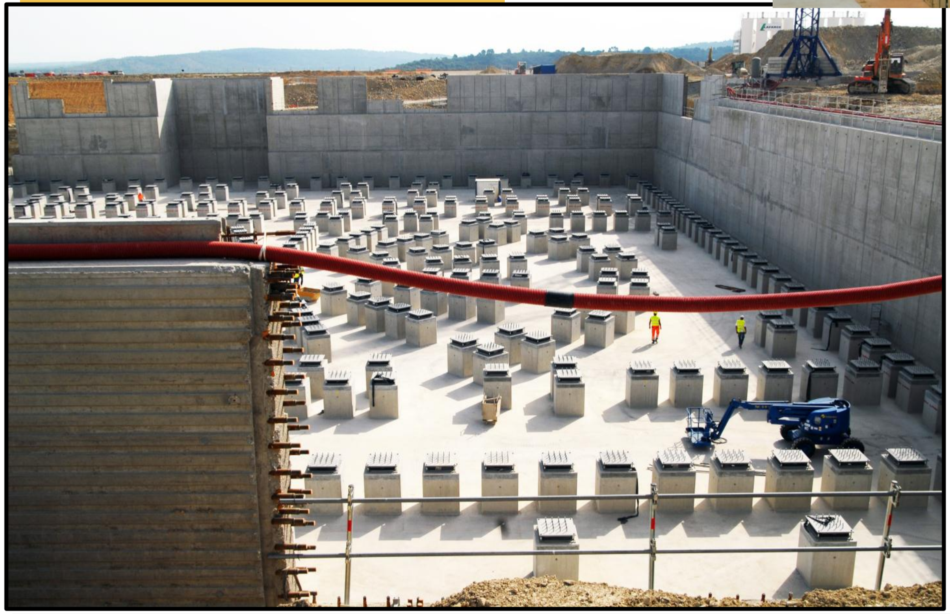
ITER Construction at Cadarache



Construction Status at Cadarache

Tokamak Complex Construction with new ITER Headquarters Building

Tokamak Complex Foundations



PF Coil Winding Building

Conclusions I

- **Achievement of high fusion gain DT plasmas in ITER will require the integration of several challenging aspects of plasma operation:**
  - ITER's H&CD systems are designed with considerable flexibility in order to address these challenges
  - the H&CD systems must be able to provide very reliable high power, long pulse operation in order to fulfill their key role in ITER plasma scenarios
- **Many of the key principles underlying the application of RF H&CD systems in ITER have already been demonstrated in present experiments**
  - extrapolation of physics basis to ITER scale has associated uncertainties
  - scenarios call for multiple applications in a single pulse – we need to learn how to implement this
- **Development of long pulse advanced scenarios relies heavily of application of RF H&CD systems**
  - there is still considerable R&D to be accomplished to establish an adequate physics basis

Conclusions II

- **Key Physics R&D needs which must be addressed in preparation for ITER operation include:**
  - development of efficient coupling methods for ICRF and LHCD power
  - further exploration of ICRF H&CD scenarios for application to ITER
  - optimization of flux consumption and current ramp-up (including MHD stability) with RF H&CD
  - characterization of H-mode access and performance over a wider range of parameters
  - detailed studies of MHD instability control in ITER-relevant scenarios to improve predictions of power requirements
  - exploration of use of RF H&CD systems to sustain long pulse advanced scenarios in preparation for fully non-inductive operation