Tokamak Energy

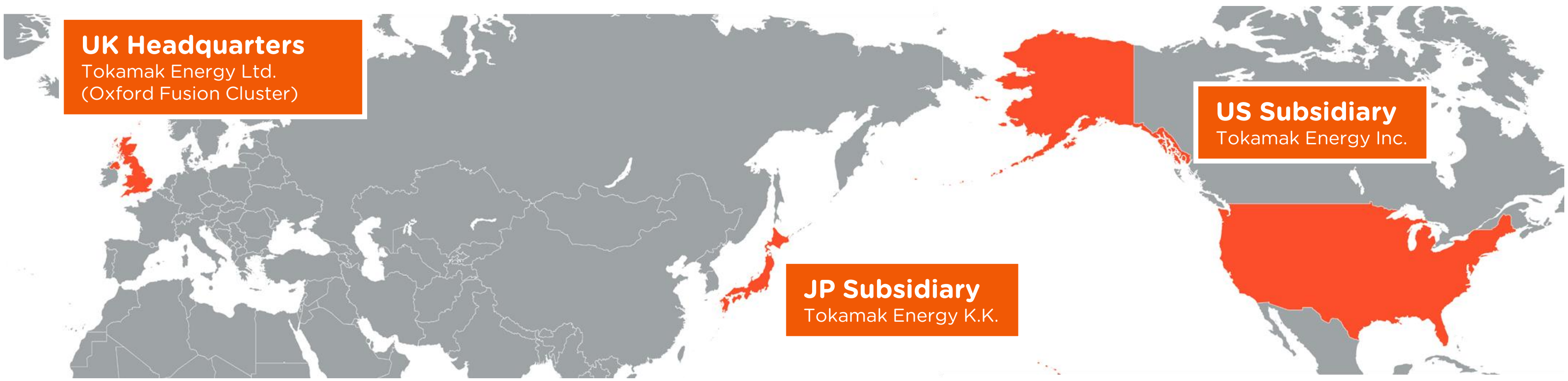
Ramp-up and Sustainment Scenarios for Tokamak Energy's Fusion Pilot Plant

Y. Takase, J. Astbury, S. McNamara,
C. Wilson, A. Alieva, N. Lopez, X. Zhang

25th Topical Conference on
Radio-Frequency Power in Plasmas

Schloss Hohenkammer
19-22 May 2025

Tokamak Energy: The leading global fusion company



260 people

- World-class scientists, engineers and commercial specialists
- 70 PhD, 100 MSc

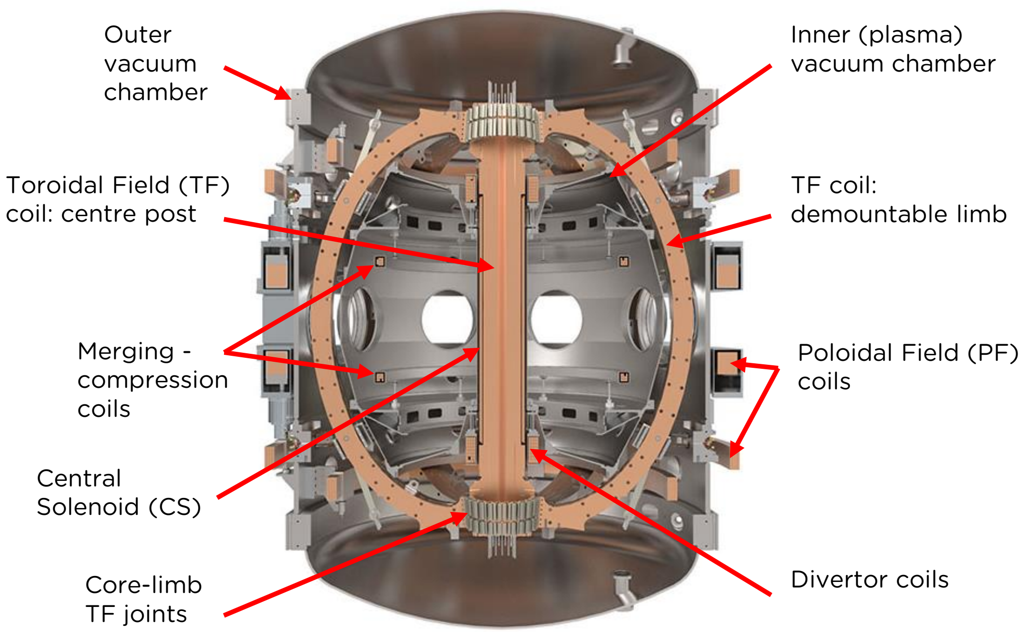
$335M raised to date

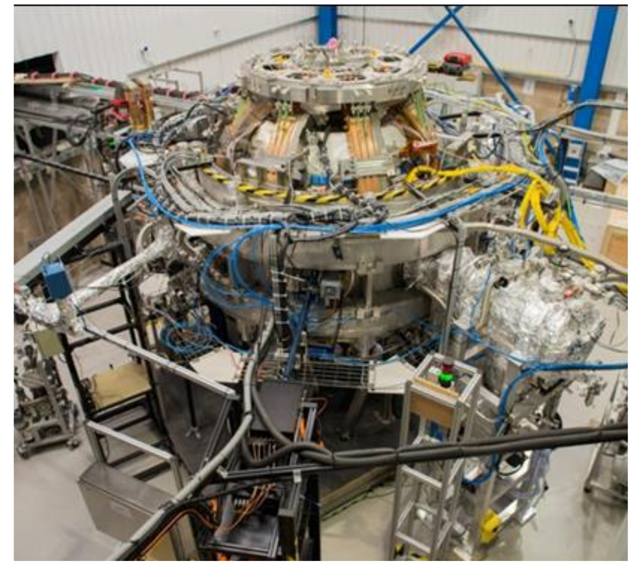
- Financial backing from private capital and government grants

Collaboration

- Industrial and R&D collaborations throughout the world

ST40: Spherical Tokamak (ST) with high magnetic field (Cu coils)





Parameter	Range
B_T [T]	0.9 – 2.1 (2.4)
I_p [MA]	0.3 – 0.8
R_{Geo} [m]	0.4 – 0.5
A	1.6 – 1.9
P_{NB}/E_{NB} [MW/kV] P_{EC}/f_{EC} [MW/GHz]	0.8 / 24, 1.0 / 55 (1.0 / 137, 104)
ψ_{sol} [Wb]	0.2
start-up	merging-compression
fuel	hydrogen/deuterium

$52M ST40 upgrade (with Li PFCs) funded jointly by US DOE and UK DESNZ (2026-2028)

Tokamak Energy:
unique experience and pedigree in commercial fusion

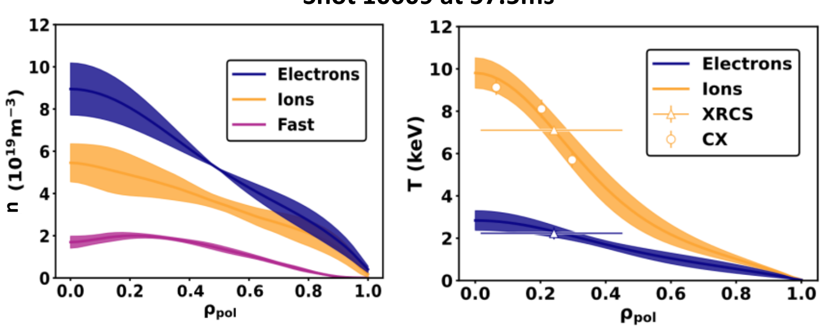
Device / Operational Pedigree:

10 years of designing, building, operating tokamaks and HTS magnets

Performance Pedigree:

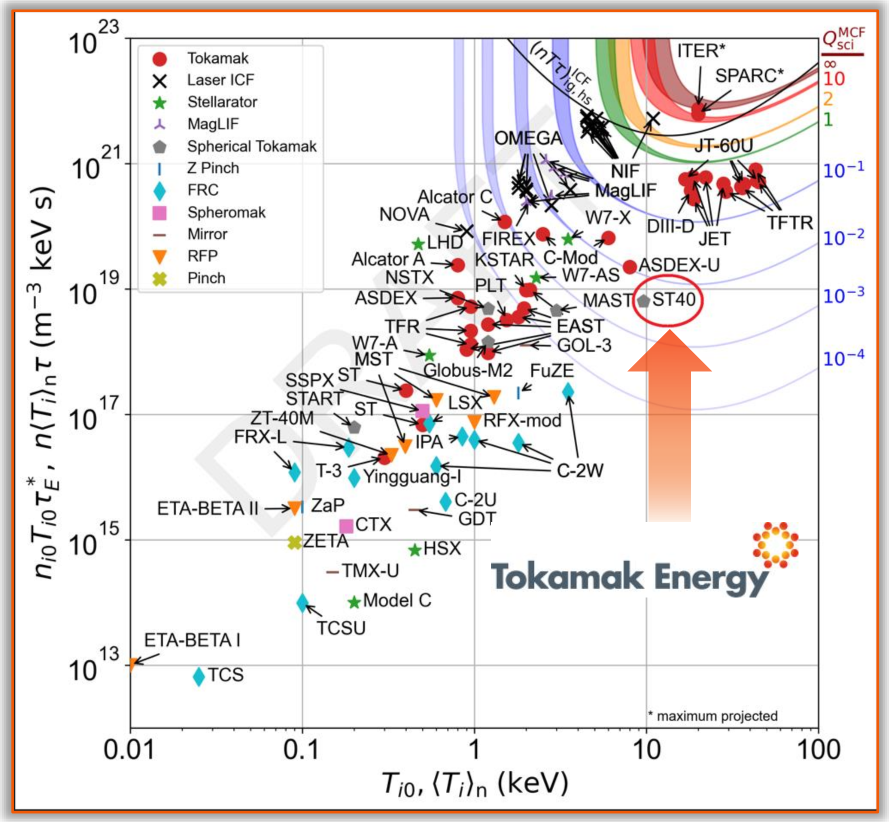
achieved the highest ion temperature T_{i0} and fusion triple product $n_{i0}T_{i0}\tau_E$ of any private company[1]

Shot 10009 at 57.5ms

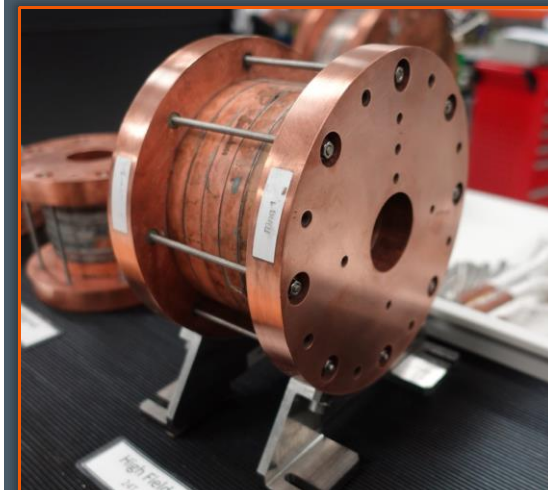


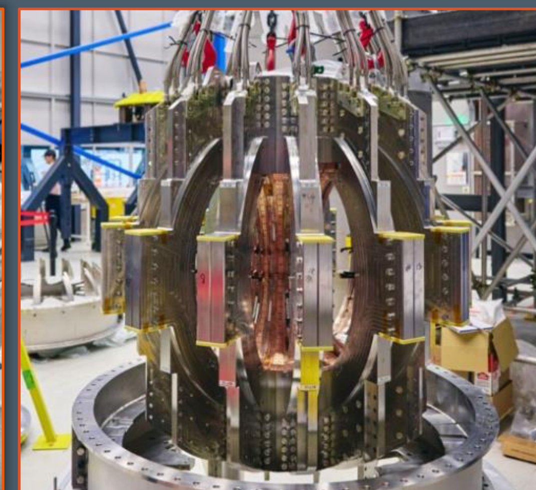
[1]S.A.M. McNamara et al. *Nucl. Fusion* **63** 054002 (2023)
DOI: 10.1088/1741-4326/acbec8

Achieved ion temperature of 100 million degrees C (requirement for DT fusion reaction).



Tokamak Energy's unique pedigree in HTS magnet development





Ultra-High Field:

World-record 24 T field at 20 K with patented high-temperature superconductor (HTS) magnet technology

Quench-Safe Magnets:

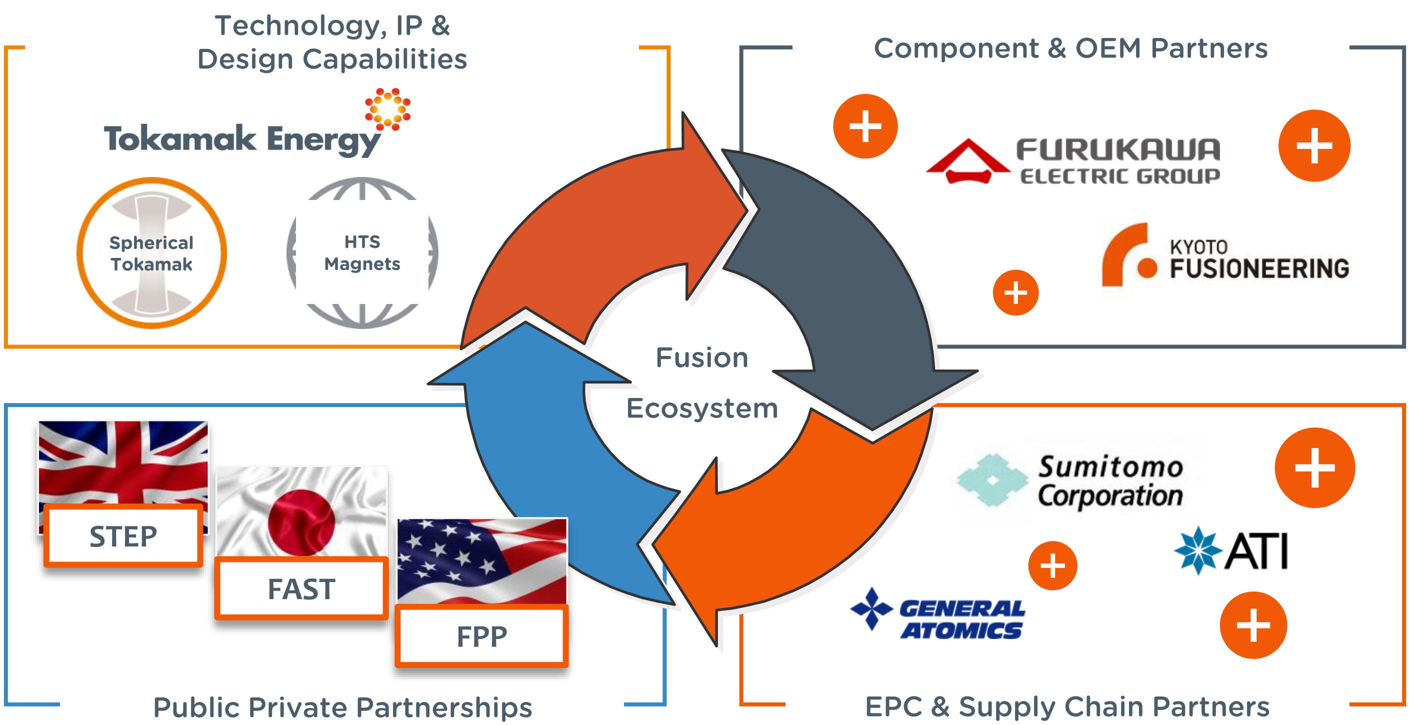
Design and manufacture robust, quench-safe HTS magnets validating our simulations

Fusion-Ready Magnets:

Demo4 is the all-HTS (TF+PF) tokamak system testbed

We know that bringing fusion to market will take more than one company

We want to bring together a diverse ecosystem of partners to accelerate the deployment of commercial fusion energy technology in the UK, US, Japan and across the world.



THE MILESTONE PROGRAM

In 2022 the U.S. Department of Energy (DOE) launched the Milestone Based Fusion Development Program.

U.S. DEPARTMENT OF ENERGY

"Milestone Program" is a federally-funded competitive technology development program that tasks awardees with maturing the technology and design readiness of a Fusion Pilot Plant (FPP).

By end of the 5-year program, awardees must reach Preliminary Design for a FPP that can...

- Demonstrate 50 MWe net output (or thermal eq.)
- Built with an overnight Capex of < $6 billion
- Start operations by 2034
- Built and operated in the United States

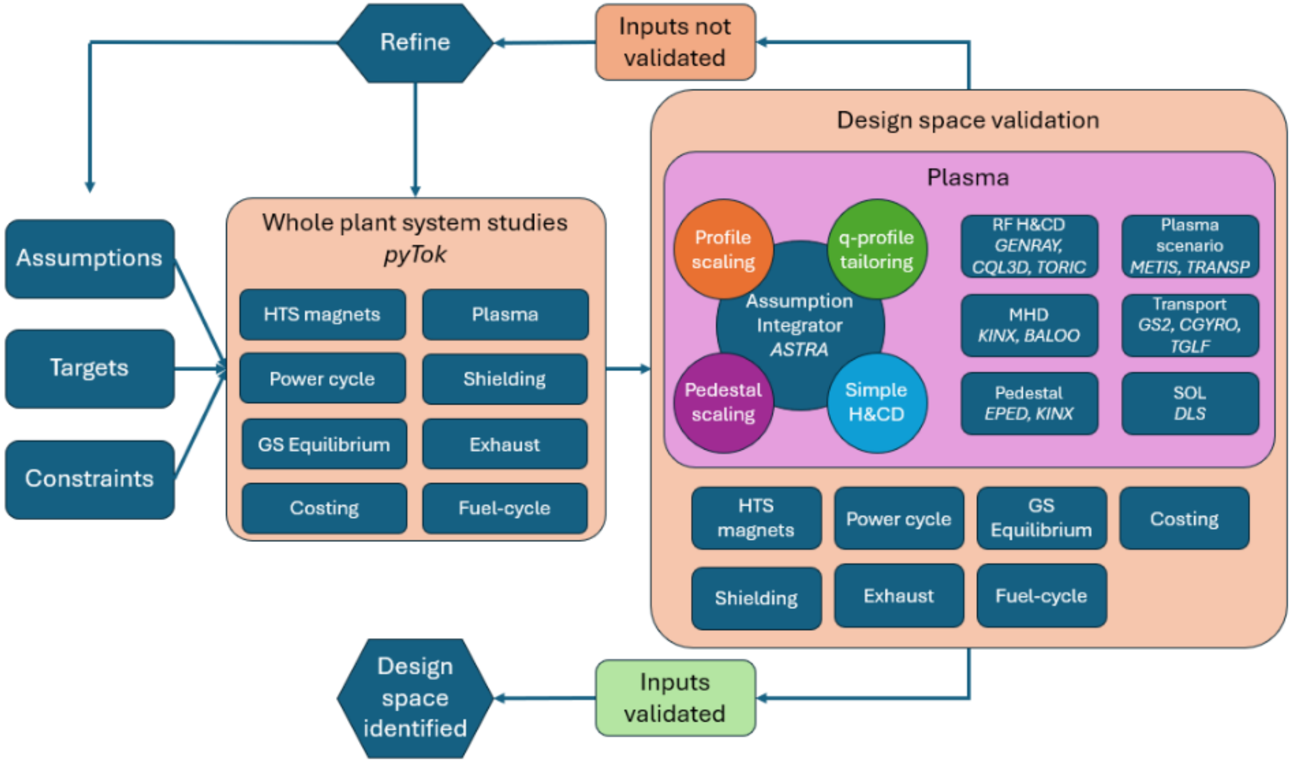
TE is among the 8 companies selected for the Milestone Program

Tokamak Energy INC.

Pre-concept design approach

- The global race to commercial fusion and **decadal timescale** motivates pragmatic design choices and the need to balance risk across, physics, engineering, R&D, schedule, cost, etc.
- Our design approach is focused on **system integration** and **phased fidelity**.
- **Iterative approach** with each cycle lasting up to 2 months.

Note: still early days so all parameters, assumptions, approaches, etc. **subject to change!**



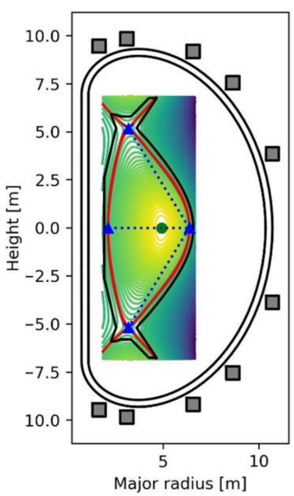
Plasma and technology assumptions

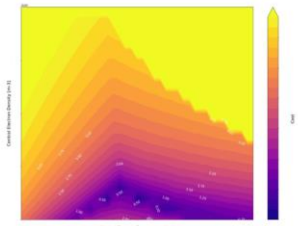
Plasma stability	$\beta_N \leq \beta_{N,\text{ no-wall}}$, $f_{GW} \leq 1.0$, $\kappa \leq 0.9$ x controllable limit, $q_{min} \geq 2.2$
Plasma scenario	Fully non-inductive operation and pulsed operation considered.
Inductive current drive	$\geq$ 50% of flux required for I_P ramp-up.
Plasma confinement	H_{ITER98} = 1.2 – 1.6 (*radiation corrected)
Magnets	Full HTS magnet set (TF + PF + CS). Continuously wound TF coils (no remountable joints), PFs outside TF.
TF coil lifetime	$\geq$ 5 full power years
Breeding blanket	Natural, slow flowing, liquid lithium breeder with helium coolant.
Tritium breeding ratio	$\geq$ 1.1
Plasma facing components	Solid as baseline with liquid lithium option also being developed.
H&CD	ECCD for flat-top and ramp-up. ICRH also being considered for ramp-up.
Centre column shielding	Graded tungsten carbide and boron carbide with helium coolant.

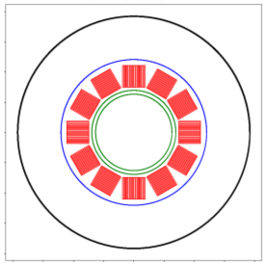
Potentially attractive design space identified by whole plant system modelling (pyTok)

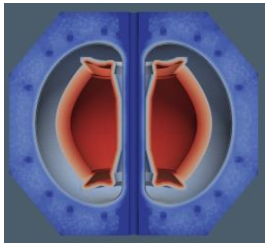
pyTok includes:

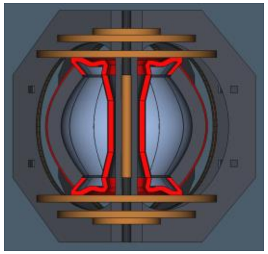
- simplified models for all major plant systems
- parametric CAD generation for cost modelling and neutronics
- large parameter space optimisation and sensitivity studies
- free-boundary equilibrium generation using FreeGS
- continuous validation against higher-fidelity codes









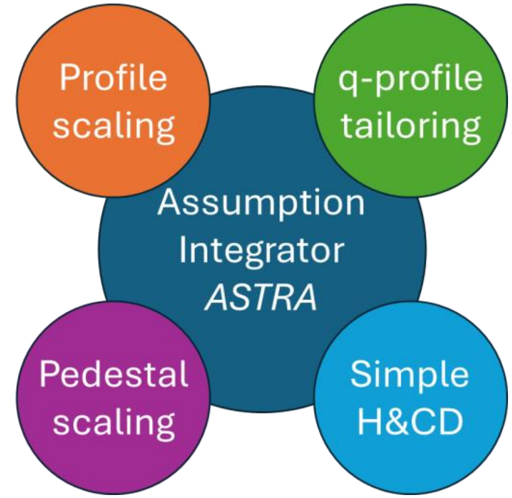


R_{Geo} [m]	4.25			
A (aspect ratio)	2.0			
B_T [T]	4.5			
δ / κ	0.5 / 2.45			
Φ_{SOL} [Vs]	45 (bipolar)			
H*(98)	**1.2**	**1.3**	**1.4**	**1.5**
I_p [MA]	**15.9**	**14.9**	**14.2**	**13.6**
P_{fus} [MW]	930	891	842	804
$P_{elec,net}$ [MWe]	**68**	**90**	**98**	**106**
$P_{H\&CD-EC}$ [MW]	141	119	103	90
$P_{rad,core}$ [MW]	230	195	180	165
Q	6.6	7.5	8.2	9
β_N	2.81	2.93	2.93	3.09
f_{bs}	0.58	0.64	0.69	0.74
P_{sep}/R [MW/m]	21.9	22.4	19.7	18.4
OCC [$Bn]	5.750	5.54	5.38	5.24
f_{NI} [-]	**0.8**	**0.85**	**0.9**	**0.95**
$t_{flat-top}$ [hr:min]	**00:25**	**00:40**	**01:05**	**02:25**

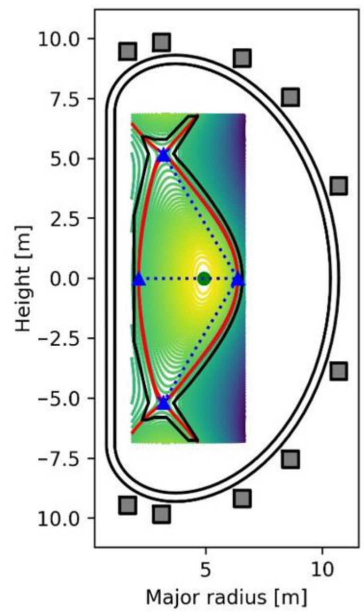
Assumption integrator

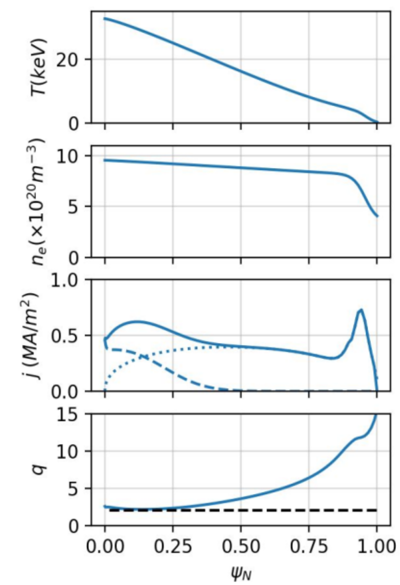
ASTRA-SPIDER 1.5D transport code used as an *assumption integrator* to develop consistent flat-top plasma operating points by combining several low fidelity models.

Turns 0D systems code output into 1D profiles + 2D equilibrium, which can be used for further analysis.



	ST-FPP
R_{Geo} [m]	4.25
Aspect ratio	2.0
B_T [T]	4.25
δ / κ	0.5 / 2.45
I_P [MA]	12.5
P_{fus} [MW]	825
$P_{elec,net}$ [MWe]	85
$P_{H\&CD\text{-}EC}$ [MW]	40
$P_{rad,core}$ [MW]	100
Fusion gain	20.6
β_N	3.6
β_T	4.95
f_{bs}	0.88
f_{GW}	1.0
P_{sep}/R [MW/m]	24.0
TF FPL [yr]	7
OCC [$Bn]	5.3

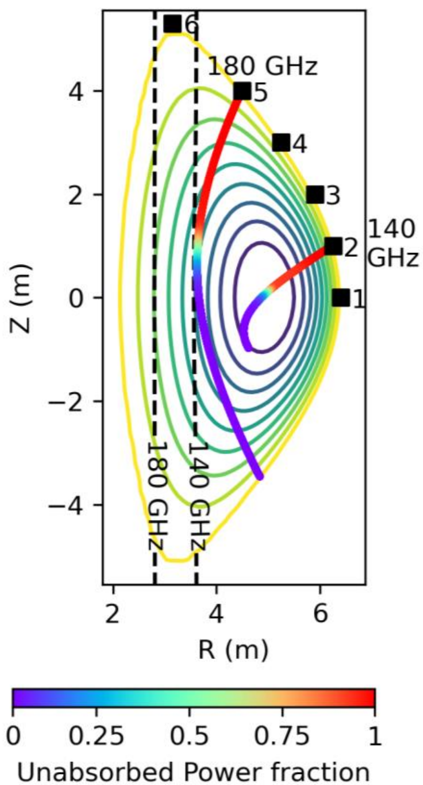


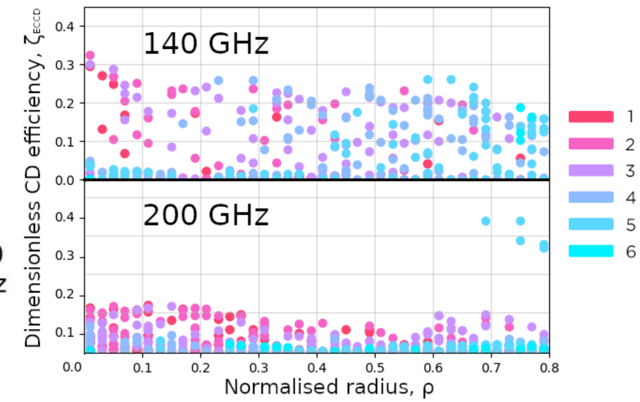


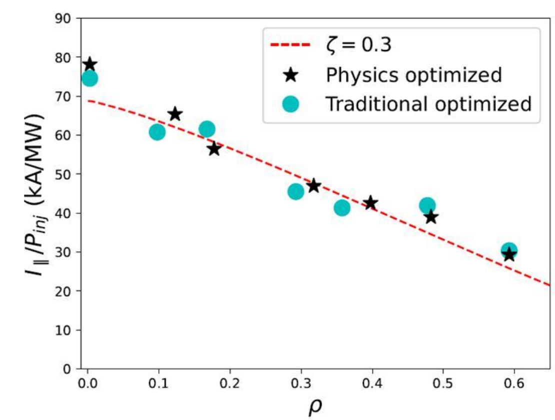
work in progress

Optimising ECCD for flat-top current drive

- Normalised current drive efficiencies of $\zeta \approx 0.3$ achievable over entire minor radius for LFS O-mode launch (f = 140 – 200 GHz).
- Due to field and density, EBW is not accessible at flat-top for either O-X-B or direct X-B.
- Physics-based optimisation[3] produces similar results with significantly fewer runs (~ 100 vs. ~ 130,000).







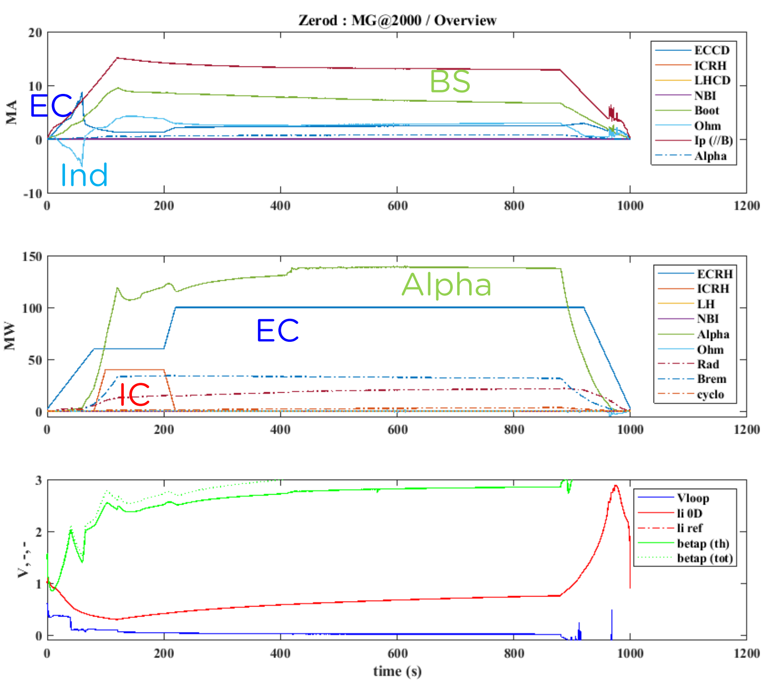
see poster presentations:
A. Alieva (Monday-15)
N. Lopez (Monday-30)

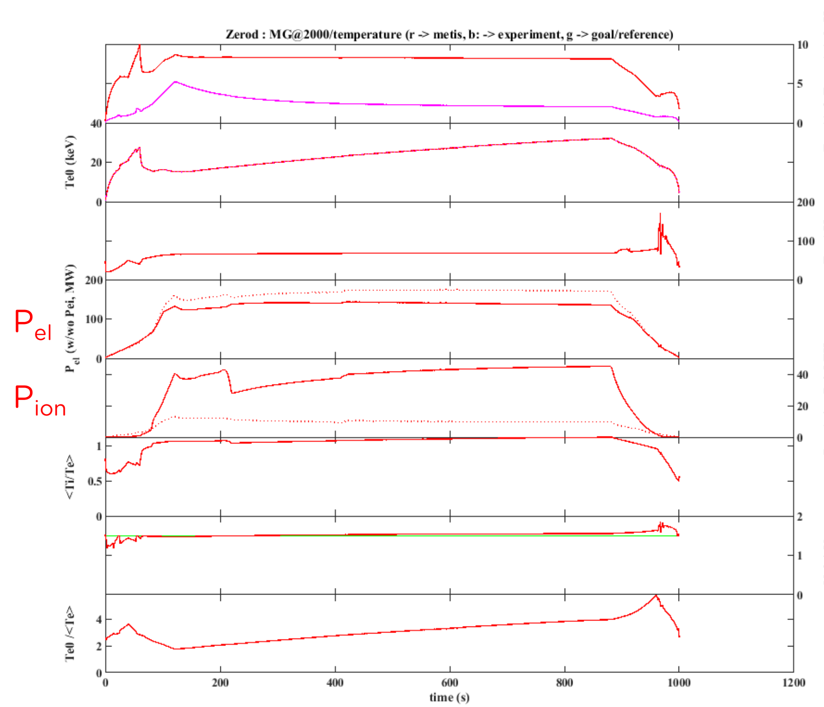
[3]N. Lopez et al. *Plasma Phys. Control. Fusion* **67** 055012 (2025).

Ramp-up modelling by (METIS[4])
IC heating used to accelerate start of fusion burn

- Densification and ion heating during the second half of I_p ramp-up
  - ➢ Increased I_{BS}
  - ➢ Additional inductive drive by B_v ramp-up during entry to fusion burn

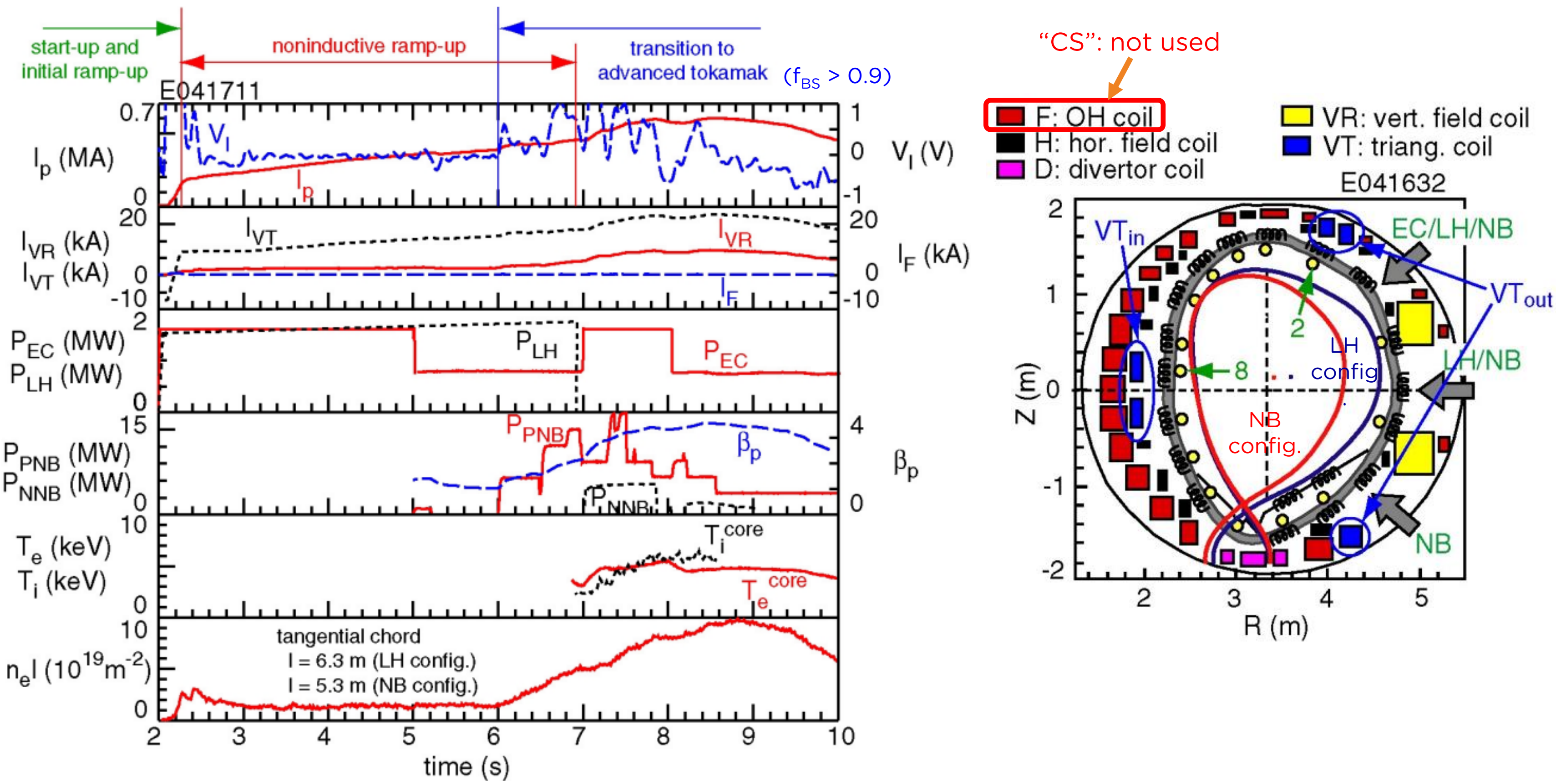
FPP with
R_0 = 4.25 m, A = 2.15,
B_t (R_0) = 4.0 T, f = 32 MHz
DT plasma + n_{He3}/n_e = 2%





[4]J. F. Artaud et al. *Nucl. Fusion* **58** 105001 (2018).

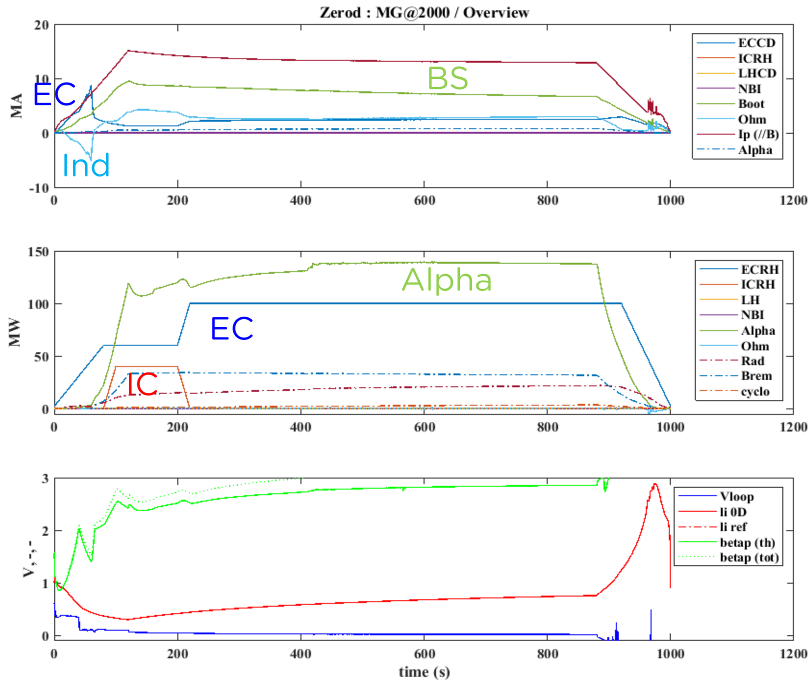
Ramp-up to "advanced tokamak" with f_{BS} ~ 0.9 was demonstrated on JT-60U[5]

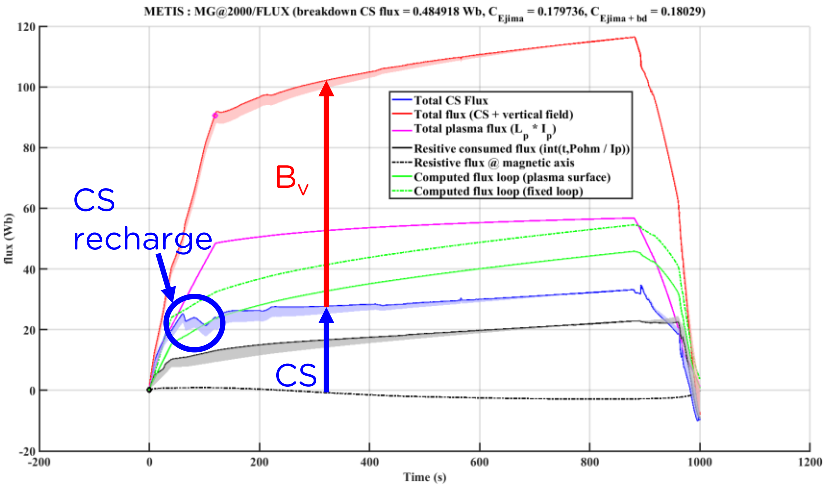


[5]Y. Takase et al. 18th IAEA FEC (Lyon 2002) PD/T-2
https://www-pub.iaea.org/MTCD/Publications/PDF/csp_019c/pdf/pdt_2.pdf ;
S. Shiraiwa et al. *Phys. Rev. Lett.* **92** 035001 (2004)
doi: 10.1103/PhysRevLett.92.035001.

Use of CS and B_v ramp-up to accelerate I_p ramp-up and assist sustainment

- Initial I_p (pressure driven) can be driven by ECH with trapped particle configuration[6]
- Fully non-inductive I_p ramp-up is slow and inefficient due to the induced negative E field
  → Add positive E field by CS to accelerate I_p ramp-up (ramp up to full I_p in 2 minutes)
- CS can be partially recharged by I_{EC} + I_{BS} + B_v ramp-up





[6]C.B. Forest et al. *Phys. Rev. Lett.* **68** 3559 (1992).

Further analyses (ongoing / planned)

- How much T_e and T_i increases can be achieved with specified P_{el} and P_{ion} depends on what happens to plasma turbulence (and therefore energy transport). METIS assumes Bohm-gyroBohm transport coefficients with the stored energy normalized by a given H factor (e.g. ITER98), but a more sophisticated modelling (TGLF, etc.) is needed. This work is in progress.

- Specification of the pedestal width/height (affects the total stored energy and fusion power significantly).

- Compatibility with heat and particle exhaust.

- There is experimental evidence that electron heating enhances turbulence, and ion heating (and associated fast ion generation) suppresses turbulence. This is the main reason for adding IC.

Summary

- Tokamak Energy is designing a Fusion Pilot Plant (FPP) based on low A tokamak. It will consist of an operationally-relevant fusion environment, and will demonstrate scalable net power in a fully-integrated system.

- The low A tokamak geometry enabled by HTS magnets offers key advantages over alternative approaches. The potential to access and sustain high confinement, high beta, and high f_{BS} is attractive. Present parameters used in the modelling are: R_0 = 4.25 m, A = 2.15, B_{t0} = 4.0 T, and I_p = 13.6 MA.

- Because of low A, the flux swing capability of the CS is limited. Novel I_p ramp-up and sustainment scenarios must be developed.

- METIS was used for scenario development. The plasma is initiated with EC power injected to the “trapped particle configuration”. Following initial ECCD at low density (assisted by CS induction), plasma is densified and ion heating by IC is applied to initiate fusion burn. The increased plasma stored energy increases I_{BS}, and additional I_p ramp-up is achieved with inductive assist from the increasing B_v.

- Optimisation of CS usage is essential to achieve a quick I_p ramp-up with minimal use of the CS flux.

Tokamak Energy acknowledges U.S. Federal support for this work under TIA DE-SC0024889

Tokamak Energy

Thank You

CONTACT US:

Yuichi Takase
yuichi.takase@tokamakenergy.com

www.tokamakenergy.com

Tokamak Energy Ltd
173 Brook Drive
Milton Park
Oxfordshire, OX14 4SD
United Kingdom