

Fully Non-Inductive scenarios: FAST case modelling

ENER

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Acknowledgments

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Topics covered in this talk

Aspects of fully non-inductive scenarios

with RF H&CD system

 Aspects of simulation capability for RF H&CD systems in fusion plasmas Predicting scenarios and performance in future fusion devices beyond the level of OD scaling laws is a challenging task.

On one hand we do not have at disposal yet a fully validated core and edge predictive transport models,...

Main aim of my talk

... on the other hand assuming 1D profile conservation starting from data in existing machine and using dimensionless parameter scaling is at least partially hindered by expected differences in the parameters between present and future machines, such as in plasma rotation, amount of electron heating and impurity concentrations.

Main aim of my talk

In this situation, the predictive activity must wisely combine both...

a) theory based

simulations

b) and empirically

based considerations



c) with the strongest possible link to

experimental results in

existing devices

Designing a scenario: Fully NICD



OUTLINE

- Introduction
- Brief overview on Fully NICD Operations in existing devices and LHCD aspects
- FAST: Fusion Advanced Torus Studies
- Fully NICD simulations and role of rotation
- Interplay between ITB and rotation

A fusion reactor for electric power generation will likely need to operate in a steady state manner [Zohm FS&T 58 2010]. The tokamak, which is presently the most advanced magnetic confinement fusion concept, requires a toroidal plasma current to provide confinement.

Introduction

This toroidal current is conventionally driven by mutual induction with the plasma acting as a secondary coil to the primary ohmic transformer solenoid.

Although inductive current drive is simple and robust, the length of the discharge is ultimately limited by the maximum flux the ohmic transformer can provide.

Non-inductive current drive is critical to

advance the tokamak towards an on-the-grid

fusion reactor.

Introduction

If auxiliary heating systems (ICRH, LH, ECRH) can be used on tokamak experiments as a means of generating toroidal current without the use of transformer action

the bulk of the non-inductive current drive in future steady state reactors will likely be provided by the bootstrap current (internally driven non-inductive current associated to enhanced pressure gradient) [Najmabadi FE&D 2006]

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Fully NICD in existing machines

Fully non-inductive scenarios in existing devices, some examples:

- Alcator C-Mod
- Tore Supra
- TCV
- DIII-D
- FTU

...

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Fully NICD in Alcator C-Mod*

Lower hybrid (LH) waves can be used on tokamak experiments as a means of generating toroidal current without the use of transformer action [Fish 1980, Fish 1987, Fish Lecture L1 6th IIS 2012].



LH waves are preferentially launched along magnetic field lines in the counter-current direction.

*[Wallace IAEA - TM SSO 2010]

Aspects of LHCD

Adjusting the phase between adjacent elements of the structure sets the parallel index of refraction of the wave:



c = the speed of light in vacuum k_{11} = the parallel wave number ω = the angular wave frequency

[See also H Tuong Lecture S1 6th IIS 2012]

These LH waves Landau damp on electrons moving ~ in phase with wave

The momentum and energy given to these electrons by the wave creates an asymmetric electron distribution function \rightarrow net flux of electrons in the counter-current direction \rightarrow therefore a net current in the co-current direction Lower hybrid current drive (LHCD) is a particularly attractive method of driving non-inductive current due to its high current drive efficiency and ability to drive current off-axis

$$\eta = \frac{n_e I_P R_0}{P_{LH}} \quad \left[10^{20} \, A W^{-1} m^{-2} \right]$$

Experimental results in C-Mod (1)



- Diverted plasma configuration
- Zero loop voltage sustained for full length of LH pulse (0.5 s) at $n_e \sim 0.5 \times 10^{20} m^{-3}$, Ip $\sim 500 kA$, B=5.4T,
- $P_{LH} \sim 750 kW$, $n_{||} = 1.6$
- Current relaxion time ~ 0.2s
- Some transformer recharge (i.e.

transient negative loop voltage)

Experimental results in C-Mod (2)

$$\eta_{C-Mod} = \frac{\prod_{P=1}^{n} R_0}{P_{LH}} \approx 2 - 2.5 \times 10^{19} \, AW^{-1} m^{-2} \quad \eta_{ITER} = 2.3 \times 10^{19} \, AW^{-1} m^{-2}$$

MSE-constrained kinetic EFIT reconstructions predict a broad current profile with q_0 ~2 and flat or slightly reversed shear during non-inductive operation (V_{loop}=0)



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Although the bulk of the non-inductive current drive in future steady state reactors will likely be provided by the bootstrap current...

... auxiliary current drive off-axis will be required to control the current profile to maximize plasma stability and confinement. **Tore Supra** is the largest superconducting tokamak in the world (a = 0.72m, R = 2.4m, $I_P < 1.7 \text{ MA}$, $B_T < 4.5 \text{ T}$, circular plasma configuration).

Fully NI plasma discharges* have been sustained in a steady-state regime for up to 6 min with an injectedextracted energy exceeding 1 GJ in 2003

*[D. van Houtte NF 44 (2004)]

Fully NICD in Tore Supra (2)

- $I_P = 0.5 0.7 MA$, $B_T = 3.4 T$, $\langle n_e \rangle = 1.5 \times 10^{19} m^{-3}$
- Zero loop voltage regime

• Plasma current generated by injecting up to 3MW LH waves at 3.7 GHz, delivered by two launchers with $n_{||}$ between 1.5 and 2.5 by choosing the phase shift between adjacent klystrons

 \cdot q₀ between 1.5 and 2, reverse shear configuration

$$\eta_{Tore-Supra} = \frac{n_e I_P R_0}{P_{LH}} \approx 0.73 \times 10^{19} A W^{-1} m^{-2}$$

Fully NICD in TCV*

First demonstration of fully non-inductive steady-state operation with electron cyclotron current drive in a tokamak ($I_P = 210kA$, ECRH = 2.7MW)

- R = 0.88 m
- a = 0.25 m
- $\bullet I_p < 1 \text{ MA}$
- $B_T < 1.43 T$



• Elongated vacuum vessel for shaping

*[Coda 27th EPS 2000]

Fully NICD in TCV* (2)

210 kA sustained in steady state by 2.7 MW co-ECCD



Fully NICD in DIII-D*

Progress toward Fully NICD using off-axis new Neutral Beam Injection (5MW) that

 broadens the current density profile by changing the distribution of beam-driven current; broadens the pressure profile; and extends the tearing mode stable duration

• a decrease in the normalized confinement results from the off-axis injection, with a further decrease when the minimum in q is above 2

*[Ferron 54th APS 2012]

Fully NICD in FTU*

Full current drive at $n_e \approx 0.5 \times 10^{20} \text{ m}^{-3}$, $B_T = 6 \text{ T}$



*[Pericoli NF 45 2005]

Fully NICD in FTU*

Current drive efficiency



*[Pericoli NF 45 2005]

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FAST: a satellite in view of ITER and DEMO

A Satellite proposal finalized to study, in an integrated experiment:

- Plasma Operation \rightarrow ELMs, Plasma Control, Heating coupling...
- Plasma Wall Interaction \rightarrow P/R~22, Full material wall,...
- Burning Plasma → Energetic particles induced collective behaviours and transport; cross-scale couplings between meso- and micro-scale fluctuations and their impact on the long time-scale plasma dynamics
- Complementary (with some Overlap) with JT60-SA

- FAST may have different and more relevant roles for preparation of ITER scenarios (even in parallel with ITER)

- FAST addresses a variety of integrated physics and technology issues (e.g. power exhaust) and may help early DEMO design

FAST main parameters

The prerequisites to be satisfied, in order to reproduce the physics of ITER relevant plasmas, yield the following set of FAST parameters [Pizzuto NF10]:



Plasma Current (MA)	<u>≤</u> 8
B _T (Τ)	≤ 8.5
Major Radius (m)	1.82
Minor Radius (m)	0.64
Elongation k ₉₅	1.7
Triangularity δ_{95}	0.4
Safety Factor q ₉₅	3
V_p (m ³)	23
<n>(m⁻³)</n>	≤ 5.5×10 ²⁰
Flat-top B _T (s)	15 -> 170
H&CD power (MW)	40
ICRH	30 (->15)
ECRH	4 (->15)
LH	6
NNBI	10 (20)?
P/R (MW/m)	22

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FAST scenarios

FAST	H-mode reference	H-mode extreme	Hybrid	d H-mode ECRH	AT	AT2	AT Full NICD
I _p (MA)	6.5	8	5	6	3.5	3	2
q 95	3	2.6	4	2.8	5	3	5
$\mathbf{B}_{\mathrm{T}}(\mathbf{T})$	7.5	8.5	7.5	6.5	6	3.5	3.5
H_{98}	1	1	1.3	1	1.5	1.5	1.5
$< n_{20} > (m^{-3})$	2	5	3	2	1.4	1,1	1
P_{th_H} (MW)	14 ÷18	22 ÷ 35	18 ÷ 23	14 ÷ 18	8.5 ÷ 12	8.5 ÷ 12	5 ÷ 7
β_{N}	1.3	1.7	2.0	1.4	2	3.2	3.4
$\tau_{\rm E}$ (s)	0.4	0.65	0.5	0.38	0.25	0.18	0.13
τ_{res} (s)	5.5	5	3	5	3	5 ÷ 6	2 ÷ 5
T ₀ (keV)	13.0	9.0	8.5	11.0	12	13	7.5
Q	0.65	1.5	0.9	0.5	0.32	0.14	0.06
t _{discharge} (s)	20	13	20	26	55	170	170
t _{flat-top} (s)	13	2	15	17	45	160	160
$\mathbf{I}_{\mathrm{NI}}/\mathbf{I}_{\mathrm{p}}$ (%)	15	15	30	20	60	80	>100
P _{ADD} (MW)	30	40	30	15+15	40	40	40
	[Calabrò, Mantica et al. IAEA 2010]	[Crisanti et al IAEA 2010 Cardinali et a al. IAEA 2010	, 11]	[Crisanti et al, IAEA 2010 Calabrò,Mantica et al IAEA 2010]	[Calabrò Vienna 2	0 6 th IAEA-TA 2010]	NSSO,

FAST AT scenarios \rightarrow physics

FAST	AT	AT2	
			Full INICD
I _p (MA)	3.5	3	2
q 95	5	3	5
B _T (T)	6	3.5	3.5
H_{98}	1.5	1.5	1.5
$< n_{20} > (m^{-3})$	1.4	1.1	1
P _{th H} (MW)	8.5 ÷ 12	8.5 ÷ 12	5 ÷ 7
β_{N}	2	3.2	3.4
$\tau_{\rm E}$ (s)	0.25	0.18	0.13
τ_{res} (s)	3	5 ÷ 6	2 ÷ 5
T ₀ (keV)	12	13	7.5
Q	0.32	0.14	0.06
t _{discharge} (s)	55	170	170
t _{flat-top} (s)	45	160	160
I_{NI}/I_{p} (%)	60	80	>100
P _{ADD} (MW)	40	40	40

ICRH	30MW
LH	6MW
ECRH	4MW

In all of them "assumed":

- same plasma shape of H-ref
 scenario
- an improved H₉₈ (=1.5)
- a slightly reverse q profile



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FAST AT scenarios \rightarrow engineering

FAST	AT	AT2	AT
газі	AI	ATZ	Full NICD
I _p (MA)	3.5	3	2
q 95	5	3	5
B _T (T)	6	3.5	3.5
H_{98}	1.5	1.5	1.5
$< n_{20} > (m^{-3})$	1.4	1.1	1
$P_{th H}$ (MW)	8.5 ÷ 12	8.5 ÷ 12	5 ÷ 7
β_{N}	2	3.2	3.4
$ au_{\mathrm{E}}\left(\mathbf{s} ight)$	0.25	0.18	0.13
τ_{res} (s)	3	5 ÷ 6	2 ÷ 5
T ₀ (keV)	12	13	7.5
Q	0.32	0.14	0.06
t _{discharge} (s)	55	170	170
t _{flat-top} (s)	45	160	160
I_{NI}/I_{p} (%)	60	80	>100
P _{ADD} (MW)	40	40	40

AT plasma configurations are consistent with the power removal and particle pumping in the divertor [Madduluno NF 09, Crisanti SOFT 2010]

In all of them "assumed":

 a residual loop voltage of about 0÷100mV

→ PFCs can easily sustain the discharges for a long time

Time limit given by copper TFCs
 adiabatic heating





Predicting performance and scenarios in future fusion devices beyond the level of OD scaling laws is a challenging task...

...predictive activity **must** wisely combine both theory based simulations and empirically based considerations, with the strongest possible link to **experimental results**: FAST scenarios <u>studied</u>, specifically here the **NICD** one, by considering both the predictions of several physics based or semi-empirical transport models and the recent transport experimental results on devices such as **JET**, **DIII-D** and **C-MOD**,

... in particular, with respect to the attainable rotation profile and its effect on thermal transport [Mantica et al. IAEA 2010]

Simulation set-up

FAST transport simulations carried out using **JETTO** code (JAMS JET suite of integrated codes):

- **TORIC*** code used for ICRH heating profiles
- FRTC code used to calculate LH heating and CD profiles
- ASCOT* code used for NBI power and torque profiles (the possibility to replace 10MW ICRH with 10MW NNBI has been also studied)
 - * M. Brambilla, Plasma Phys. Control. Fusion 41
 - * HEIKKINEN, J.A. and SIPILÄ , S.K., Phys. Plasmas 2 (1995) 3724



JAMS - JETTO modelling suite*



JAMS - JETTO modelling suite*

° Jetto	Heating
File Data Help	sustems
	Systems
🚹 N B I 🚹 R F 🚹 ECRH 🚹 I	Н
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	Number Of Particles 100000
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	Help
	Exit Submit

* JINTRAC - JET modelling suite - 2008 - S.Wiesen



Briefly on FRTC (Fast Ray Tracing Code)

- FRTC is a standard ray-tracing code coupled to a 1-D Fokker Planck model aimed at calculating the LH power deposition and current density (A. R. Esterkin and A. D. Piliya 1996 Nucl. Fusion **38** 1501, E. Barbato and A. Saveliev 2004 Plasma Phys. Contr. Fusion **46** 1283)
- In FRTC besides Landau resonant absorption, collisional absorption is also included (*E. Barbato 2011 Nucl. Fusion* **51** *103032* doi:10.1088/0029-5515/51/10/103032)
- FRTC is a module of the JETTO code, so that LHCD can be calculated in time and self consistently with the current diffusion equation

Designing a scenario: Fully NICD

FAST



Plasma with reversed q profile + sufficient rotational shear

Internal Transport Barrier ⇒ Core region of enhanced pressure gradient and associated bootstrap current, improved confinement

Role of rotation \Rightarrow

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Recent experimental results show the necessity of a toroidal rotation to get an Internal Transport Barrier (ITB) [Mantica PRL 2009, Mantica IAEA 10, De Vries PPFC 08, Politzer NF 08]

Role of rotation - Experiments

JET discharges: #52281 (NBI dominant), #69414 (ICRH dominant)



FAST will have only a small input of external momentum (like ITER) and only in the framework of the NNBI scenarios

ITB in Alcator C-mod

- 1) They do not take place in L-mode even with reverse q
- 2 They take place in H-mode where there is significant intrinsic rotation
- Because the source of intrinsic rotation is at the edge, ITB onset causes a hollow rotation profile, whose gradient helps sustaining the



The rotation has been included in the FAST transport simulations by self-consistently modelling also the momentum transport. Two sources of rotation have been considered →

First source: intrinsic rotation

- Intrinsic rotation is well known observation in tokamaks [Rice NF07, Nave EPS' 10], but a quantitative undestanding of phenomenon is still lacking...
- Using the C_MOD [Rice NF07] proposed scaling (...FAST is compact and



Intrinsic rotation

 Using the most recent theory results for the inward pinch, confirmed on JET tokamak [Mantica PRL2009], (i.e. Prandtl number Pr=1 and Pinch number ~4, derived in A.G. Peeter PRL 2007) we can evaluate the toroidal rotation profile

Second source of rotation considered: the core torque due to 10 MW NNBI

Fully NICD scenario simulation

A complete simulation of a 3.5T fully NICD scenario with reversed q has been performed using the semi-empirical mixed Bohm-gyroBohm (BgB) model and heating power:

• 30 MW, either provided fully by the ICRH system at 33 MHz in (³He)-D minority (resonant layer r/a \approx 0.1, close to the plasma centre) and power spectrum centred at parallel wave-number n_{||} =9.5, or by 20 MW ICRH and 10 MW NNBI

• 4 MW provided by the LH at 5 GHz, and $n_{||,peak}$ =2.3

- 1. Hypothise an intrinsic rotation according to Rice's scaling and calculate v_{tor} profile using Pr=1 and pinch from Peeters et al.
- 2. Use such rotation in transport simulations with BgB transport model activating the term with rotation

Designing a scenario: Fully NICD

FAST





Flat-top phase



 \rightarrow q profile features are from the LH and the BS current; LH effective and viable tool to shape this profile

Designing a scenario: Fully NICD



 Role of rotation is essential for ITB formation and sustainment q profile alone is not sufficient to provide ITB formation [De Vries NF 49 2009]

 Turbulent pinch provides peaked rotation in presence of edge intrinsic rotation to allow ITB formation [Mantica Phys. Plasma 2010, Tala IAEA 2010]



...which requires a core torque source to maintain some rotational shear and the ITB

Simulations with intrinsic rotation, in which the ITB criterion embedded in the BgB model is switched on, yielding ITB formation



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However the likely presence of an ICRH core counter torque [Nave EPS 2010, Rice NF 2003] makes rotation <u>hollow</u>, establishing sufficient rotational shear to maintain the ITB



This dynamic has been experimentally observed in C-Mod ITB plasmas (J.Rice, NF 2003)

Results

• A fully non-inductive pulse can be achieved with the intrinsic edge rotation, reversed q profile, and sustained ITB at ρ ~0.6

• T_{i0}~20 keV, T_{e0}~15 keV with n_{e0}~2 10²⁰ m⁻³ and a confinement time t~0.2 s \rightarrow n_{e0}T_{i0} t ~8 10²⁰ keVs/m³

•These parameters have to be **regarded** as overestimated due to the simplistic assumptions of the BgB model

•Attempts to use first principle GFL23 are on-going

Substituting 10MW ICRH with 10MW NNBI

• NO significant difference to the performance

within the assumptions of the BgB model

• NNBI provides a more reliable source for

maintaining rotation shear, which is an essential

ingredient for achieving ITB formation.

Conclusions on FAST fully NICD simulations

- A fully NICD pulse can be achieved in FAST with an ITB at rho~0.6 formed due intrinsic edge rotation, momentum pinch and reversed q profile
- **ITB sustainment**, due to momentum pinch vanishing in ITB region, relies on the presence of a core torque source, either in co-direction from NNBI to maintain a peaked rotation or in counter-direction from RF to make rotation hollow, establishing sufficient rotational shear to maintain the ITB

Conclusions I

- Presented an overview on fully NICD operations in exisisting machines and aspects of LHCD
- Presented FAST (ITER satellite tokamak proposal) Fully

NICD simulations (bootstrap + LHCD):

 \Rightarrow Rotation gradient \rightarrow ITB \rightarrow enhanced confinement

Such ion core confinement improvement is an essential ingredient for obtaining steadystate scenarios with a core region of enhanced pressure gradient and associated bootstrap current.



Fully Non-Inductive scenarios: FAST case modelling

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Back-up

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slides

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On LHCD

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LH waves transfer energy and momentum to fast electrons to drive current

 $\square \omega \sim 3 - 5 \times \omega_{LH}$

- LH waves launched preferentially in countercurrent direction
- LH waves Landau damp on electrons moving ~ in phase with wave
- Asymmetric distribution function yields current





On MSE - q profile in C-Mod

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MSE-constrained EFIT reconstruction indicates weakly reversed shear profile when $V_{loop} = 0$



- Equilibrium reconstruction constrained by 9 MSE measurement points and pressure profile
- Shear reversal within error bars due to the statistical error of pressure and MSE
- MSE pitch angle becomes stationary after current diffusion timescale (~ 0.2 s)

Fusion Advanced Studies Torus





Predicted toroidal rotation profile

- 100 krad/s in the centre!
- Similar to best high power JET NBI shots

Note: Not included:

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ICRH counter-torque -Is an estimate at least of the known part available from TORIC?

Ripple loss of ICRH fast ions

Neoclassical Toroidal Viscosity effects

rhotor

Result highly uncertain, but it shows that toroidal momentum dynamics may be very important in FAST even without NBI



Briefly on Toric Code*

- Numerical simulation of Ion Cyclotron Waves in tokamak plasmas
- The code TORIC solves the finite Larmor radius wave equations in the ion cyclotron range of frequencies in arbitrary axisymmetric toroidal geometry
- The model used, based on the finite Larmor radius approximation, describes the compressional and torsional Alfven waves and the ion Bernstein waves excited by linear mode conversion
- The numerical solution is based on the spectral representation of the wave fields in the poloidal angle "theta", and cubic finite elements in the radial variable

*M. Brambilla, Plasma Phys. Control. Fusion 41 (1999) 1-34.

Thanks to A. Cardinali, ENEA

Briefly on Toric Code*

Several iterations between the JETTO and TORIC codes, in combination with the SSQLFP code (which solves the guasi linear Fokker- Planck equation in 2D velocity space), to deal with the coupled problem of propagation and quasi-linear absorption of ICRH, have allowed consistent evaluation of the dynamic scenario evolution.