Report on FY07 DOE Theory Milestone: Improve the simulation resolution of linear stability properties of Toroidal Alfven Eigenmodes driven by energetic particles and neutral beams in ITER by increasing the numbers of toroidal modes used to 15

Personnel involved in milestone activities:

R. V. Budny, E.D. Fredrickson, G.-Y. Fu, N. N. Gorelenkov, C.E. Kessel, G.J. Kramer, D. McCune, J. Manickam, R. Nazikian, R. White
Princeton Plasma Physics Laboratory, Princeton University

H. L. Berk
Institute for Fusion Studies, Unviersity of Texas, Austin

J. Snipes
Massachusetts Institute of Technology Plasma Science and Fusion Center

W.W. Heidbrink, L. Chen
University of California, Irvine

A. Polevoi
ITER team & Kurchatov institute, Moscow*

Quarterly milestones:

Q1 Develop fiducial ITER numerical equilibria, using TRANSP, to determine the alpha-particle slowing down distributions and neutral beam ions for a range of operating regimes.

Q2 Analyze the normal shear discharges, performing a parameter scan to determine the linear stability of toroidal mode number n = 1-15 TAE modes.

Q3 Analyze the hybrid shear discharges, performing a parameter scan to determine the linear stability of toroidal mode number n = 1-15 TAE modes.

Q4 Analyze the reversed shear discharges, performing a parameter scan to determine the linear stability of toroidal mode number n = 1-15 TAE modes, and prepare a comprehensive review of the TAE stability of ITER discharges in the three operating regimes.

Introduction

In a thermonuclear deuterium-tritium (D-T) tokamak plasma the $3.5MeV$ alpha particles must be trapped by the magnetic field so that their energy can be transferred, primarily through electron drag, to the background plasma. One purpose of burning plasma (BP) experiments is to demonstrate that this method of self-heating will be the dominant method of heating of a plasma that is producing fusion energy. However, when the alpha particle partial pressure is significant, a physics issue arises as to whether this pressure is capable of inducing collective behavior that may cause the premature loss of alpha particles. Should this be the case, two major problems may arise: (i) it may become difficult to sustain the plasma parameters close to those required for ignition and (ii) the flux of energetic alpha particles ($\sim 3.5MeV$) to the first wall of the experiment can cause severe wall damage.

Indeed it has been demonstrated in present day (PD) experiments that the collective effects induced from energetic particles can result in premature energetic particle loss. However, it is difficult to extrapolate the results of PD experiments to BP experiments, for the following reasons. The fast particle distribution functions are often quite different. In PD experiments the energetic particle distribution are anisotropic whereas in a BP experiment the

*Electronic address: ngorelen@pppl.gov

distribution function of fusion alpha particles would be isotropic. In addition, in a BP experiment the machine size to orbit width will be significantly larger, and the spectrum (and number) of unstable modes is likely to be broader in a BP compared with PD experiments. Thus even with continued study in PD experiments, extrapolation to reliable predictions for BP experiments may remain uncertain. However, theoretical modeling and simulation can provide predictions of the likely effects of the driven modes.

It is generally believed that the Toroidal Alfvèn Eigenmodes (TAE's) [1–3] destabilized by fast ions, are the plasma waves most likely to cause significant difficulties for the containment of energetic alpha particles in fusion energy generating tokamak experiments. It has been experimentally established that in the presence of a strong enough energetic particle energy density, these modes will induce large losses of fast particles. It is also known that there exists a variety of conditions where these modes are stable or when unstable, do not induce anomalous loss.

This quarterly report describes the work performed to extend the preliminary work [4] required to address the stability issue. We will perform a systematic study of various plasma scenarios, planned for ITER, in order to determine whether linear instability to the TAE's is expected under specific burning plasma conditions. Specifically, we will study TAE stability for the three proposed scenarios; elmy H-mode, hybrid and advanced tokamak conditions. TAEs with the toroidal mode numbers up to high 15 will be investigated. With the use of analytic estimates, some extrapolation is possible to other temperature regimes of operation.

The range of toroidal mode numbers of interest is determined by applying simple analytical theory and estimating the mode number dependence of the damping and driving rates [5]. It should be noted that radiation damping becomes a significant damping mechanism when finite Larmor radius (FLR) effects increase for core ions as well as electrons, and becomes a strongly stabilizing effect at $k_\perp \rho_i \sim \sqrt{r/R}$. Here $k_\perp$ is the characteristic radial wavenumber of a TAE mode and ρ_i is the bulk ion Larmor radius calculated for ions with thermal velocity $v_T = \sqrt{2T/m}$. This damping mechanism may then compete with the alpha particle drive at moderately high toroidal mode numbers n. The fast particle drive reaches a maximum for n-numbers near

$$nq^2 \rho_h / r \simeq 1, \tag{1}$$

where ρ_h is the fast ion Larmor radius, and then beyond this value decreases with increasing n. Depending on detailed parameters, radiation damping may be a significant damping mechanism near the peak of the alpha particle drive. Application of this stability analysis to ITER has shown that one can expect the most unstable mode number to be around $n = 10$ (for both fusion alphas and beam ions) and thus it is important to be able to extend the analysis beyond this number in a systematic way. This is a challenging computational problem as the number of possible eigenmodes increases with n number. At higher n, $n > 10$, the number of grid points has to be increased in both the radial and poloidal directions to resolve the high-m, where $m = nq$, poloidal harmonic structure at the edge. Typical grid sizes range up to 400 radial points and 512 poloidal points.

Note, that from Eq.(1), the most unstable toroidal mode number depends on the safety factor value, at the point of the strongest pressure gradient, which is typically close to half of the minor radius. This means that the most unstable n numbers in elmy H-mode and in hybrid scenario (see relevant parameters in the next section) are expected to be similar, whereas in the advanced scenario with $q_{min} \sim 2$ the most unstable mode number is expected to be lower. This important property of the TAE stability in ITER remains to be confirmed, numerically.

1st Quarter report (12/29/2006)

N. N. Gorelenkov, R. V. Budny, C.E. Kessel, D. McCune, J. Manickam
Princeton Plasma Physics Laboratory, Princeton University

Quarterly Milestone: Develop fiducial ITER numerical equilibria, using TRANSP, to determine the alpha-particle slowing down distributions and neutral beam ions for a range of operating regimes.

Executive summary. *The first quarter milestone was achieved. In all, equilibria representing the evolution of eleven distinct plasma conditions were developed. The Tokamak Startup Code, (TSC), and the plasma transport simulation code, TRANSP, were used to develop fiducial equilibria for the three main ITER scenarios; the elmy H-mode, hybrid and advanced plasma regimes. The simulations addressed the evolution from start-up to steady-state, for a period of thousand seconds. TSC was used to simulate the startup and control of the plasma boundary, and TRANSP was used to obtain accurate particle distribution functions for the slowing down negative neutral beam injected, (NNBI), ions as well as the thermonuclear alpha-particles. In addition to the three fiducial ITER scenarios, eight additional scenarios, with varying NNBI injection angle, were developed for the elmy H-mode and hybrid scenarios. This will enable an evaluation of the potential of controlling the excitation of TAE modes by varying the injection angle.*

In this part of the report we concentrate on the numerical simulations of three nominal ITER plasma scenarios: elmy-H mode [6], hybrid and advanced tokamak plasma (AT, reversed magnetic shear) [7]. Even though the elmy H-mode regime was proposed first for the ITER-FEAT project, more recently the Hybrid plasma regime has attracted greater attention. It is likely that the hybrid and advanced plasmas will be the main regimes for ITER as they hold the promise of steady state plasma operations, which is essential for the next step, a power plant reactor [7]. This plasma regime was observed in Tokamak experiments [8–10] and has enhanced core confinement compared with the standard H-mode scaling. It offers the potential to operate ITER with high fusion yield at higher β_n and reduced requirements for inductive current drive. Several papers have documented encouraging predictions of Hybrid plasma performance in ITER based on various predictive models such as the GLF23 [11] and Weiland [12] models.

Methods and Modeling Techniques. The TRANSP/TSC combination is used to model ELMY H-mode and advanced plasmas for ITER. The Tokamak Startup Code, (TSC), [13] is used to simulate the startup and control of the plasma boundary adjusting the shaping and control coils. Several heating and current drive models can be used along with several prediction models (such as GLF23) to derive the evolution of the plasma temperature profiles. The output; time-dependent boundary and plasma profiles, is input to TRANSP for more detailed analysis. Since, TRANSP [14] has more comprehensive and self-consistent methods for computing the equilibrium, heating, and current drive. The TRANSP results for heating, current drive, if needed, and rotation profiles can be put back into TSC for further iterations to converge on a more accurate model.

TRANSP uses the NUBEAM Monte Carlo package [15] to model alpha heating and neutral beam heating, torque, and current drive. The RF heating and current drive are modeled using SPRUCE [16] and TORIC [17], full-wave, reduced order codes for minority ICRH.

In addition to the standard H-mode case two classes of advanced plasmas are considered: the Hybrid scenario with reduced inductive current and q_{MHD} profile maintained close to, or above unity, and the Steady State AT scenario (see section I C) with near zero inductive current. Details of the classes of ITER plasmas studied are summarized in Table I. The H-mode plasma regime is considered to be conservative for achieving $Q_{DT} \equiv P_{DT}/P_{aux} = 10$. The Hybrid regime is considered to be a path to similar Q_{DT} but requiring less inductive current, and the Steady State regime aims at longer pulse durations with close to zero inductive current drive. The equilibrium profiles have been submitted to the International Tokamak Physics Activity (ITPA) profile database maintained by the Core Modeling and Database Working Group and the Transport Working Group. The intended uses of the submissions are for code benchmarking and for inputs for down stream analysis. In normal shear H-mode plasma the evolution of the q_{MHD} profile is calculated in TRANSP. To model effects of sawteeth, sawteeth crash times are assumed, and the TRANSP sawtooth model is used to helically-mix the plasma current and fast ion profiles at the crash time if $q_{MHD}(0) < 1.0$. Otherwise, poloidal field diffusion is calculated assuming neo-classical resistivity and bootstrap current, and driven currents in the case of NBI. The sawteeth simulations resulting from this analysis generally agree well with experimental observations in plasmas, such as L-mode, H-mode, and supershots with monotonic or mildly reversed q_{MHD} profiles. The profile for ITER-FEAT would be affected by 1 MeV NNBI. If the sawtooth model is not invoked, the central values for q_{MHD} are predicted to evolve in time to ≈ 0.7.

One of the uncertainties of the viability of Hybrid plasmas in ITER is whether suitable q profiles can be created and maintained. The q profiles in present Hybrid plasmas have minimum values close to, but often above, unity. Either no or small sawteeth are observed. Often benign Neoclassical Tearing Mode (NTM) activity is observed. There is a speculation that NTM or dynamo effects create special q profiles required for Hybrid plasmas. This raises concern

units	I_p MA	I_{boot} MA	I_{nnbi} MA	I_{Oh}/I_p	$n_e(0)$ $10^{20}/m^3$	f_{GW}	T_e keV	P_{dt} MW	$\beta_\alpha(0)$ per cent
ELMy	15	2.7	1.1	0.70	1.1	0.80	22	403	0.6
Hybrid	12	2.8	4.5	0.32	0.6	0.47	33	305	1.3
AT	9	4.3	4.3	0.0	0.6	0.63	33	305	1.3

Table I: *Typical plasma parameters for three ITER scenario.*

that the required q profiles might not be accessible in ITER unless special current drive such as Electron Cyclotron (ECCD) or Lower Hybrid (LHCD) is used to control q. An alternative is to use off-axis beam-driven current (NBCD).

It was pointed out recently [18] that NBCD resulting from below-midplane NNBI aiming can maintain q above unity. The ability to alter the aiming in ITER from shot to shot is being planned. The TSC and TRANSP codes are used for time-dependent integrated predictive modeling of ITER plasmas within the region extending from the core out to the top of the edge pedestal. The modeled plasmas have reduced plasma current and high β_n relative to the baseline H-mode plasmas [6] (I_p=12 vs 15 MA and β_n near 3 vs 1.8). The reduced I_p and increased β_n imply reduced Ohmic current and the increased bootstrap current. These allow the NBCD to have a significant effect on the central q-profile. Below-axis aiming into hybrid plasmas is predicted to sustain q above unity for long ($> 800s$) durations. The indication that NBCD can maintain q above unity might obviate the need for alternatives such as ECCD or LHCD and benign NTM's to affect the q profile.

We also studied the effects of variation of the NNBI aiming into standard H-mode plasmas in ITER (with I_p = 15MA and β_{norm}=1.8, see section I A) and found that there is little effect on the central q unless the sawtooth period is long (much greater than 10s).

The auxiliary heating power for the H-mode and Hybrid plasmas are assumed to be 16.5 or 33MW of D-NNBI (with one or two beam lines), and up to 20MW of ICRH at 53 MHz (tuned to the He^3 minority resonance near the plasma center). The ITER design for the NNBI sources allows for a rotation in the vertical plane allowing the footprint of the beam in the plasma to vary by approximately 50 cm vertically from shot to shot. Injection of NNBI is shown in Figure 1.

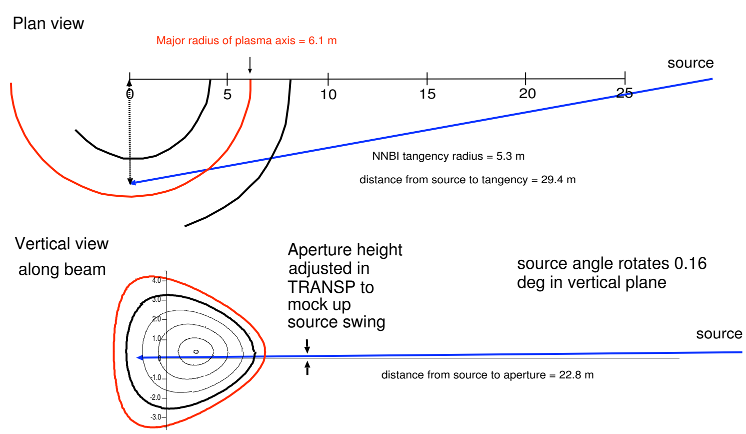

Figure 1: Schematic of NNBI in ITER. a) top view, b) side view.

The following sections contain illustrations of the main plasma profiles relevant to the TAE stability problem we are analyzing, including the effect of off-axis NBCD.

I. PLASMA PROFILES FOR THE THREE SCENARIOS

We have found that for the elmy H-mode plasma there is little effect on the central q from the change of the injection angle unless the sawtooth period is long (much greater than 10s) as can be seen from figure 7(right).

The distribution function is simulated in TRANSP using the Monte-Carlo model for each equilibrium (see example in figure 6 for elmy plasma) and is to be fitted to a special parametric dependence developed for NOVA code [5]. Its velocity space dependence is not much sensitive to the injection geometry.

TRANSP has reversed shear plasma run 60000T02, which will be extended by changing the injection angle and will be used in the TAE stability calculations in forth quarter.

A. Standard, elmy H-mode plasmas

The following table gives a list of TRANSP run numbers and a number of their representative profiles shown in the figures hereafter.

TRANSP id	# in figure legend	Y(cm)
20000T03	1	-50
20100T02	2	-38
20000T02	3	-20
20100T03	4	-10
20000T01	5	0

Table II: A list of TRANSP runs along with the number of the profile shown in the next figures. Vertical displacement of the beam line at its nearest point to its tangential radius.

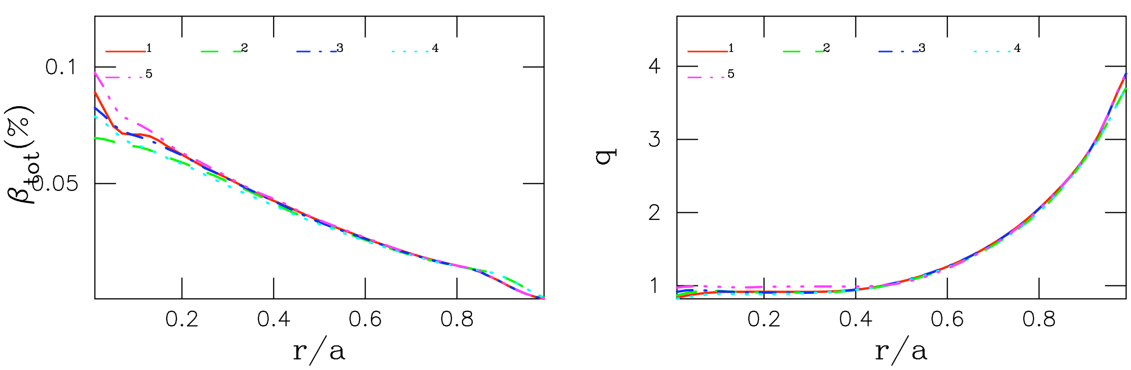

Figure 2: Total plasma beta (left) and safety factor profile (right) for elmy H-mode ITER plasmas.

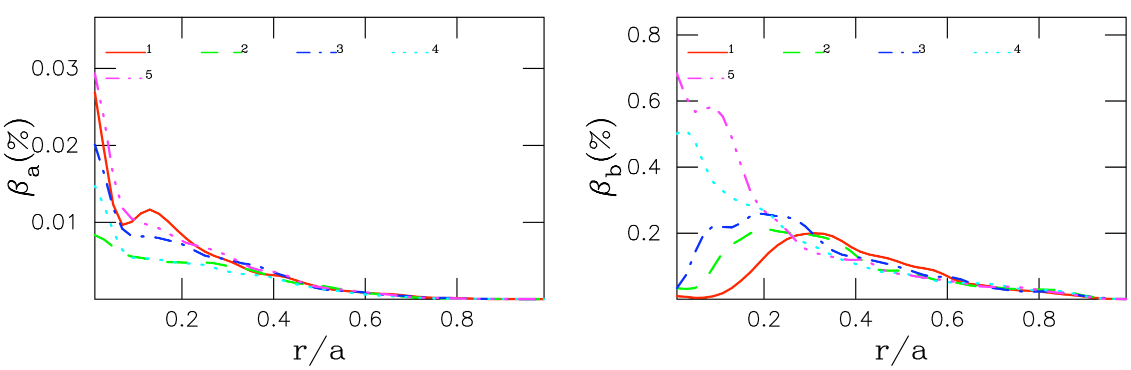

Figure 3: Fusion alphas and NNBI confined ion betas.

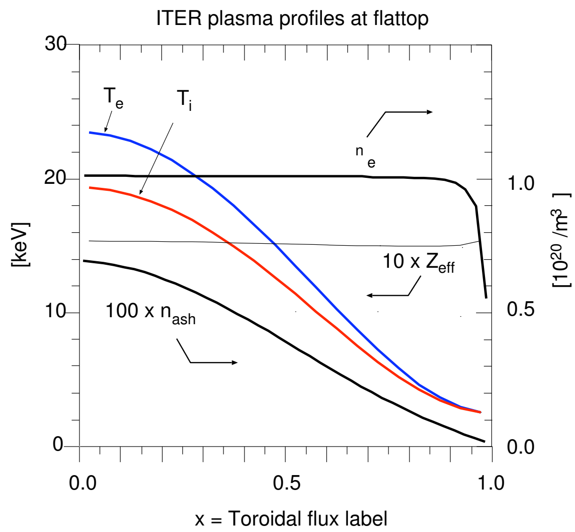

Figure 4: Profiles of the ITER H-mode plasma. The He^4 ash density times 100 is computed from the fast alpha thermalization assuming $R_{ash} = 20\%$ and $D_{ash} = 0.8$ $[m^2/s]$, and is in steady state at the time shown.

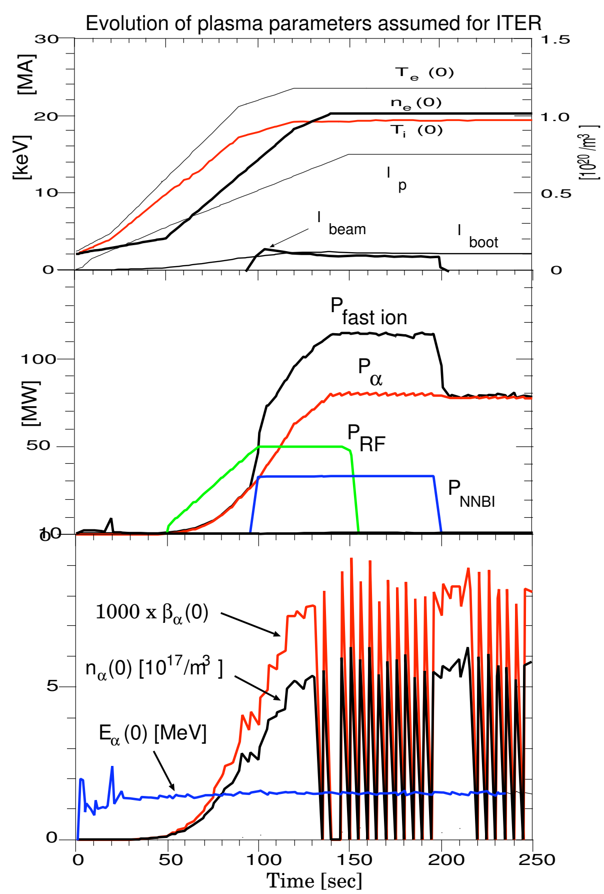

Figure 5: Time evolution of parameters in the ITER plasma. The alpha parameters in c) are volume-averaged out to the $x = 0.1$ flux surface to reduce Monte Carlo fluctuations.

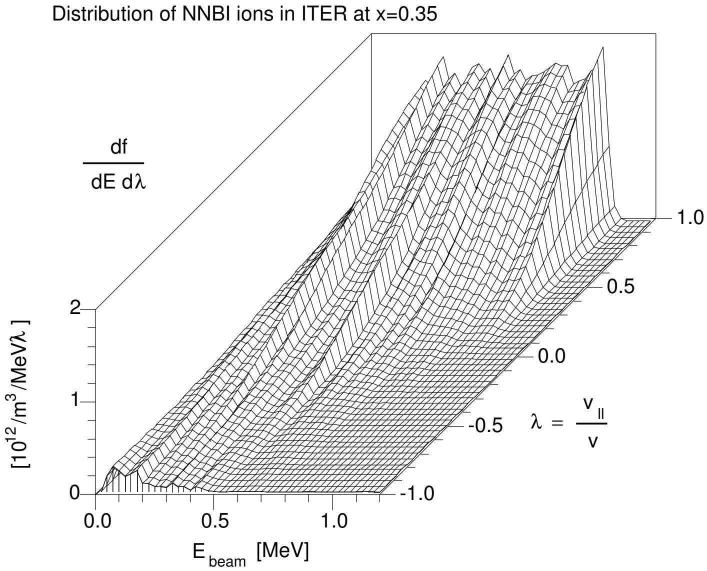

Figure 6: Distribution of the ITER NNBI beam ions in energy and pitch angle at $x = 0.35$, averaged over poloidal angle, computed by the TRANSP Monte Carlo model. The neutrals are injected at 1MeV with $v_{\|}/v \simeq 1$, and the beam ions become more isotropic as they slow down.

B. Hybrid scenario plasma

The table of TRANSP identification numbers of the simulation runs to be used in the analysis of TAE stability follows.

TRANSP id	# in figure legend	Y(cm)
40500T04	1	30
40500T03	2	25
40500A06	3	0
40500T02	4	-10

Table III: A list of TRANSP runs along with the value of NNBI vertical displacement , Y, of its injection line at its nearest point to its tangential radius.

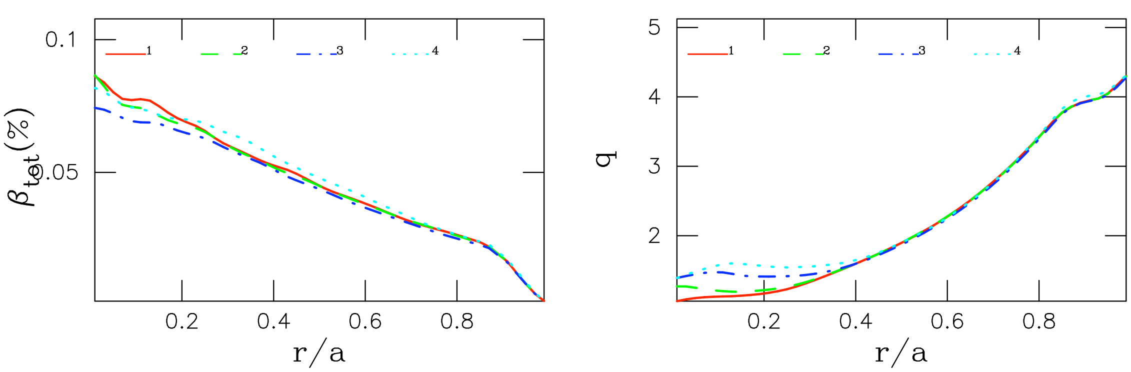

Figure 7: Total plasma beta (left) and safety factor profile (right) for Hybrid scenario ITER plasmas.

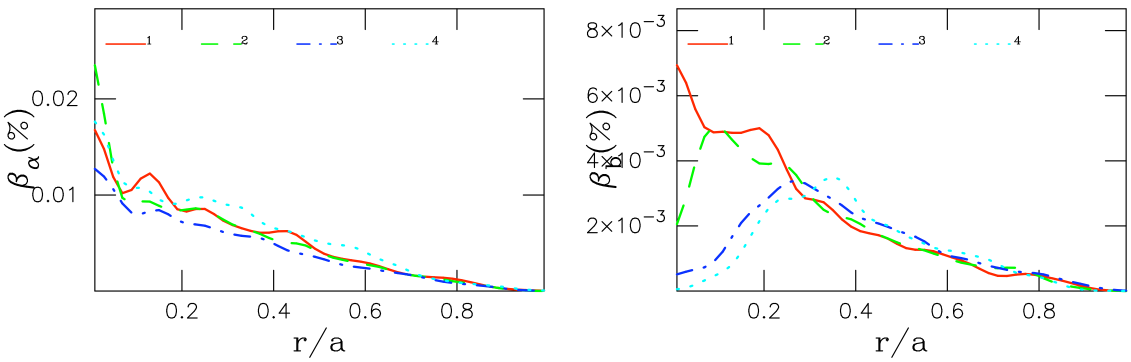

Figure 8: Fusion alphas and NNBI confined ion betas for Hybrid scenario ITER plasmas.

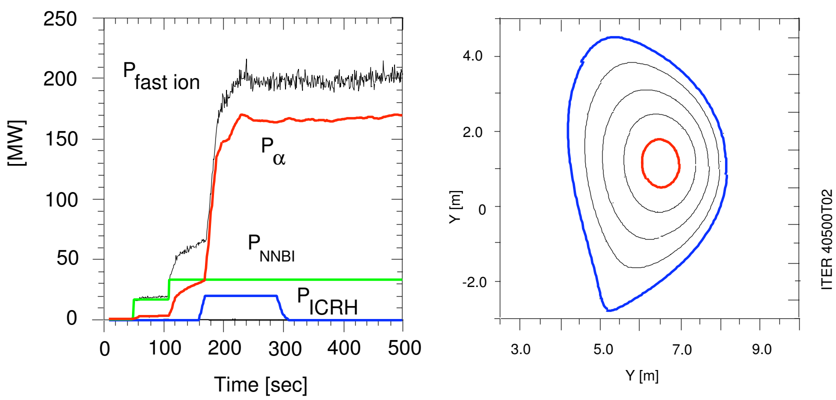

Figure 9: a) Heating powers from fusion alpha particles, NNBI, and ICRH, b) Poloidal section showing surfaces of constant toroidal flux.

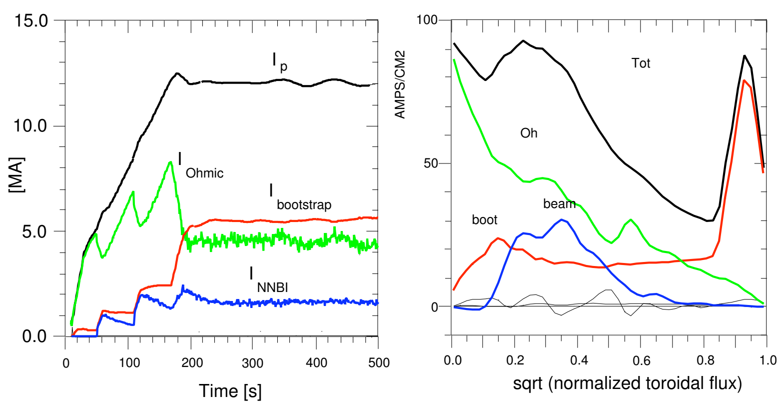

Figure 10: a) Plasma currents and b) profiles in an ITER Hybrid plasma.

C. Advanced tokamak (reversed shear) plasma scenario

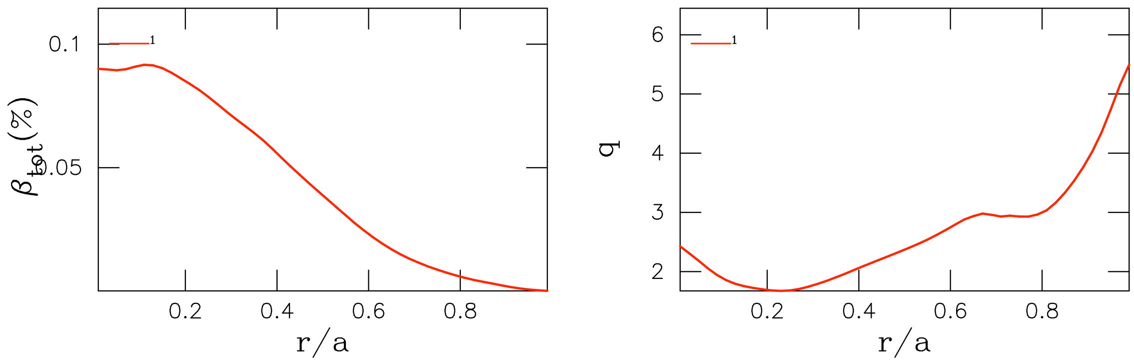

Figure 11: Total plasma beta (left) and safety factor profile (right) for elmy H-mode ITER plasmas.

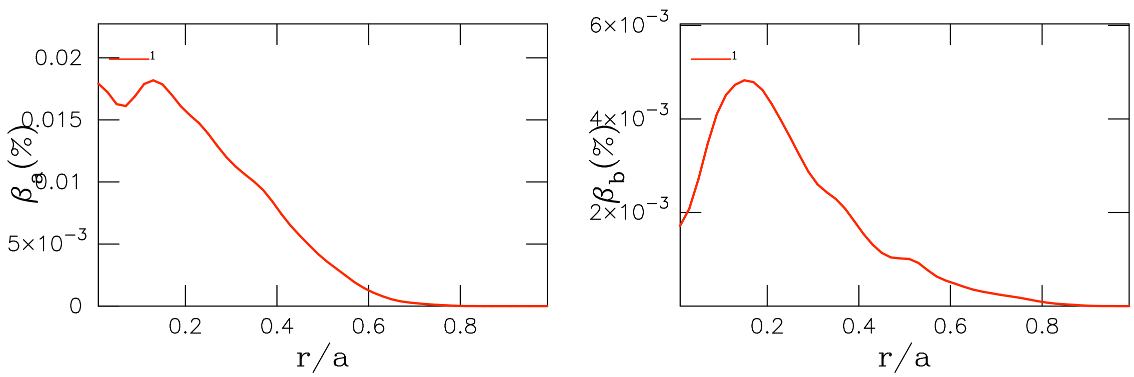

Figure 12: Fusion alphas and NNBI confined ion betas.

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