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Predictive integrated modelling for ITER scenarios

J-F Artaud, F. Imbeaux, T.Aniel, V. Basiuk, L.-G. Eriksson, G.Giruzzi, G.T. Hoang, G. Huysmans, E. Joffrin, Y. Peysson, M. Schneider, P. Thomas

Association Euratom-CEA, CEA Cadarache, CEA-DSM-DRFC,

F-13108 St Paul lez Durance, France

jean-francois.artaud@cea.fr

Introduction Integrated modeling of ITER scenarios, e.g. inductive H-modes, steady-state and hybrid scenarios, is essential for assessing their viability. Such modeling requires an accurate description of the relevant physics involved, in particular for the heat and particle transport. Different transport models are able to reproduce the existing experiments in various devices. However, they can yield significantly different results when extrapolated to ITER, either the global performance or the profiles of plasma parameters (for example pressure and current density). In this work, the uncertainty on the prediction of ITER scenarios is evaluated. For this purpose, we use two transport models, which have been extensively validated against the multi-machine database. The first model is GLF23[1]. The second is a model in which the profile of diffusion coefficient is a gyroBohm-like analytical function, renormalized in order to get profiles consistent with a given global energy confinement scaling.

This paper reports 1-D integrated simulations of full ITER discharges using the CRONOS code[2]. The package of codes CRONOS includes modules for 2D MHD equilibrium, neoclassical transport (NClass) heat, currents and particle sources[3]; and in particular a new orbit following Monte Carlo code dedicated to the simulation of fusion-born alpha particles[4]. The CRONOS simulations give access to the dynamics of the discharge and allow for studies of the interplay of heat transport, current diffusion and sources. In addition, these results are cross with 0-D simulations.

The main motivation of this work is to study the influence of parameters such plasma current, density peaking and non-inductively driven current on current diffusion and the central safety factor in ITER hybrid scenarios.

The CRONOS code CRONOS is a full 1D1/2 time dependent transport code solving for current, heat, density, impurities and toroidal moment transport. The transport equations are solved self-consistently with magnetic equilibrium (using the module HELENA[5]). All neoclassical terms, in particular bootstrap current and resistivity, are computed with the Nclass code[3]. The sources are computed by external modules interfaced with the core. Here, the following modules have been used : SINBAD for Neutral Beam Current Drive[6,7] , PION for Ion Cyclotron Resonance Heating at the 2nd Tritium harmonic[8], Delphine for Lower Hybrid Current Drive[9]. The fusion products, alpha power deposition profile and current are computed with the module SPOT[4]: it is a Monte-Carlo code for modelling the fusion alpha particle distribution function, including finite orbit width effects. It uses the cross sections from ref. 10. Various transport coefficient modules are interfaced with CRONOS.

Transports coefficients modules In this study we used 2 modules for the transport coefficients; the first is the well-known GLF23[1]. The second, is a new transport module (called kiauto) dedicated to scenario simulations, ensuring a fast numerical convergence. It has been benchmarked against Tore Supra & JET data. The 1D spatial profile dependence, in L mode or for the core plasma in H mode, is “gyroBohm” like :

$$\chi_{e,i} = C\, f(s,\alpha,\gamma_E)\, R_{e,i}(\nabla T_i, \nabla n_e) \frac{q^2 \sqrt{T_{e,i}}}{n_e} \frac{\nabla P_{tot}}{B^2} \qquad (1)$$

where q is safety factor, $T_{e,i}$ the temperature; n_e the electronic density; P_{tot} the total thermal pressure and B the total magnetic field. $f(s,\alpha,\gamma_E)$ is a function describing the improvement of the confinement with the magnetic shear, normalized pressure gradient, radial electric field shear. $R_{e,i}$ is a weight function that adjusts the value of electron and ion diffusivity depending of the position in the Stability diagram of ITG/TEM modes[11]. The resulting simulated temperatures profiles given by kiauto are strongly dependent on the value of the C constant. The specificity of kiauto is that the C constant is computed in order to force the simulated plasma energy to follow a specific scaling law. In H mode, the transport coefficient is split into three parts (fig.1 & 2); the core coefficient is given by Eq. 1; the pedestal transport coefficient is inversely proportional to the density time a constant *P;* the intermediate part is just a spline insuring a smooth transition. The width of the pedestal can be given as a parameter or computed with the help of Sugihara model[12]. The height of the pedestal can be prescribed by a scaling law or computed using the critical pressure gradient

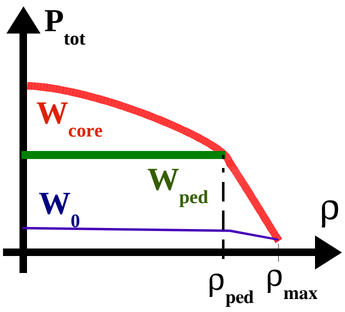

figure 1: energy decomposition in kiauto

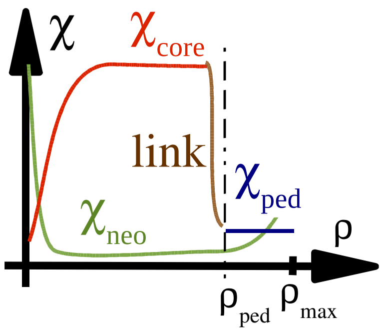

figure 2: diffusivity construction in kiauto

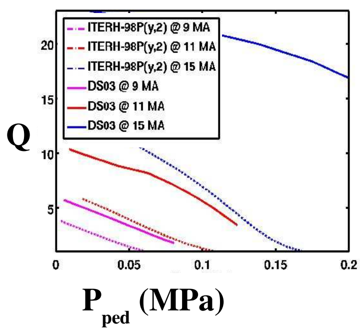

Figure 3: influence on Q of pedestal pressure in simulation made with kiauto

corresponding to the ballooning or the kink limit, as computed in the equilibrium module HELENA. The module kiauto uses various scaling laws, e.g.: 1) the standard L mode ITERH-96P(th) & standard H mode ITERH-98P(y,2); 2) the 2 terms Cordey's scaling law[13]; 3) H mode pure gyroBohm DS03 scaling[14] with scaling pedestal with low β dependence defined by Cordey[15]. We use a complete calculation of the loss power (including all radiated power) to enforce a high accuracy in the determination of the confinement time with the scaling law. The model implements a dynamical numerical mechanism equivalent to $dW/dt = -W/\tau_e + P_{loss}$. The model kiauto can also supply a pedestal model to transport models describing the plasma core only (e.g. GLF23). In the simulations performed with the model kiauto (fig. 3), the pressure at the top of the pedestal has a strong influence on the plasma performance (measured by the fusion gain $Q = P_{fus}/P_{add}$).

Simulation results All the simulations described in this paper are performed with $f_D = f_T = 37.5$ %, $f_{He} = 2$ %, $f_{Be} = 0.5$ %, $f_C = 4.5$ %; thermal transport and current diffusion only are solved for, with a prescribed electron density; flat Z_{eff} profiles; 53 MW of additional power (33 MW NBCD at 900 keV; 20 MW ICRH, 2nd harmonic of T, f = 55 Mhz). No MHD effects have been taken into account. For simplicity, the f & R effects have been turned off in the kiauto model. In order to test the parameters set and transport models used, a standard H mode ITER scenario is first simulated. In this case we have taken I_p = 15 MA, the scaling in kiauto is ITERH-98P(y,2) and the pedestal pressure is chosen in order to have Q = 10. A good agreement is found with previous simulation in the ITER-FDR plasma performance assessment documentation[16]. Here, we find Q = 9.6. We benchmarked kiauto and GLF23 (fig.4). We can observe a global agreement of the prediction but detailed profiles are quite different : the temperature peaking factors change and the ratio between electron an ion heat diffusivity change in GLF23.

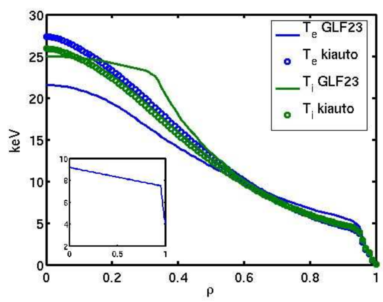

figure 4: Hybrid shot @ 11.3 MA
$Q_{GLF23} = 7$ & $Q_{kiauto} = 6$

The hybrid scenario[15,17] is primarily characterized by a low level of MHD activity and a high performance (e.g. High Q, β_N). To prevent MHD, absence of a rational surface q = 1 is required or one with a small extent. A good energy confinement is associated with the low level of MHD activity (this allows to use the scaling DS03 in kiauto, which has no β dependence). We performed a large number of simulations, varying I_p between 9 and 13 MA, the density peaking factor between 1 (ITER initial prediction) and 1.6 (assuming a dependence on the effective collisionality[18]) and we have studied the impact of LHCD up to

Row	I_p(MA)	P_{LH} (MW)	$n_e(0)/\langle n_e \rangle$	Q	β_N	P_{ped} (MPa)	V_{loop} (mV)	ρ(q=1)	t(q=1) (s)	q_0	q_{95}	neutrons/Wb	I_{ni}/I_p
1	11.3	0	1.02	4.91	2.15	0.0822	44.4	0.346	490	0.546	4.27	0.259e22	0.456
2	11.3	20	1.02	4.71	2.5	0.0984	15.5	0.0967	925	0.91	4.54	0.685e22	0.681
3	11.3	0	1.16	6.85	2.29	0.0953	30.4	0.258	296	0.676	4.6	0.415e22	0.509
4	11.3	20	1.02	2.36	1.79	0.055	99.1	0.357	395	0.498	4.25	0.129e22	0.445
5	9	20	1.02	2.01	2.11	0.0598	20.2	None	> 1200	1.22	5.72	0.502e22	0.833
6	11.3	40	1.02	4.53	2.77	0.114	0.184	None	> 3560	1.15	4.58	2.41e22	0.867
7	9	20	1.16	2.15	2.15	0.0615	10.6	None	> 3560	1.08	5.67	0.732e22	0.85
8	13	0	1.02	9.66	2.65	0.128	43.5	0.338	550	0.604	3.81	0.59e22	0.475
9	11.3	0	1.31	7.88	2.6	0.0964	43.1	0.257	575	0.688	4.34	0.573e22	0.553
10	13	20	1.02	8.76	2.97	0.15	107	0.245	760	0.701	4.01	1.36e22	0.638
11	11.3	20	1.31	7.29	2.94	0.114	7.86	0.112	1.14e+03	0.855	4.66	2.05e22	0.792
12	13	0	1.31	13	3.03	0.153	6.84	0.199	650	0.777	3.92	1.19e22	0.592
13	13	40	1.31	10.7	3.35	0.174	0.529	0.08	1.46e+03	0.893	4.11	5.5e22	0.859

table 1: synthetic results of hybrid regime simulations. Simulation row 3 is performed with GLF23, simulation row 4 used kiauto with IPB98(y,2) scaling and all others used kiauto with HDS03 scaling.

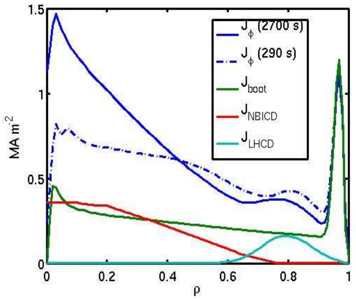

figure 5: plasma current & current sources in simulation row 2 of table1.

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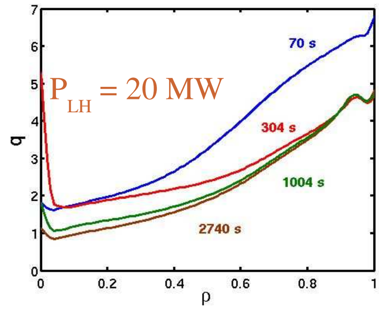

Figure 6: safety factor profiles (q) evolution in simulation row 2 of table .1

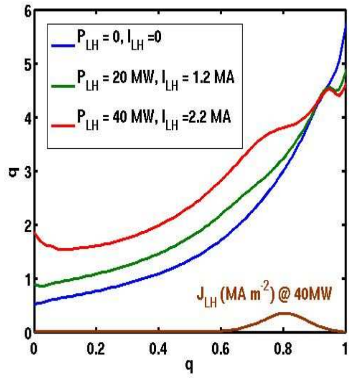

figure 7: effect of LHCD on q profiles (from simulation rows 2,4,6 ; all at 1200 s).

power level of 40MW. The pedestal pressure is chosen in order to have Q $\simeq$ 5. For most of the simulations, a fast fusion module (including orbit width effect for trapped alpha particles), benchmarked against SPOT, is used. The pulse length simulated is adapted to the current diffusion time (the duration is between 1200 s and 4000 s). Synthetic results of these simulations are presented in table 1. Figure 5 presents typical plasma and current source profiles obtained in these simulations. Figure 6 presents evolution of the q profile in the same simulation and Figure 7 shows the effect of LHCD on q profile at 1200 s. Figure 8 presents the time evolution of q(ρ=0) for each entry of table1. The parameter "neutrons/Wb" is a performance indicator of the discharge related to fluence (per Wb of consumed flux).

Conclusions The target Q = 10 can be obtained in ITER hybrid scenario at I_p = 13 MA. To achieve a stationary q-profile with q > 1 requires a large non inductive current fraction (~ 80%) that could be provided by 20 to 40 MW of LHCD. Owing to the high temperature the q-profile penetration is delayed and q=1 is reached at about 600s in ITER hybrid scenario at I_p = 13 MA, in the absence of a active q-profile control. More refined definition of the operational space for ITER hybrid scenario requires increased understanding of pedestal physics, MHD stability limits(β_N limit as a function of q-profile) and density peaking.

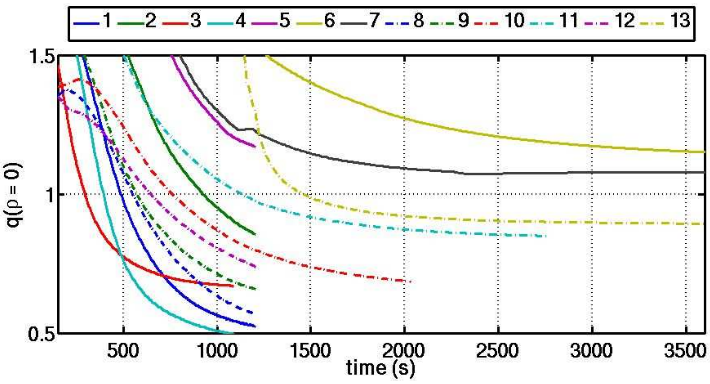

figure 8: central safety factor evolution for each row of table1.

[1] R.E. Waltz and R.L. Miller, Phys. Plasma, **6**,4265 (1999); **[2]** V. Basiuk et al., Nucl. Fus., **43,**822 (2003); **[3]** W.A. Houlberg,et al, Phys. Plasmas 4(1997); **[4]** M. Schneider et al., Proc. 12th ICPP (2004), Nice, France; **[5]** G.T.A. Huysmans et al, 1991 CP90; **[6]** Y. Feng et al, Comp. Phys. 88 (1995); **[7]** Wolle et al., PPCF (1994) 1051; **[8]** L.G. Eriksson et al, Nucl. Fus. 33 (1993) 1037-1048; **[9]** F. Imbeaux, Y. Peysson submitted to PPCF; **[10]** *H.-S. Bosch and G.M. Hale,* Nucl. Fus. Vol. 32 (1992) 611-633; **[11]** X.Garbet, 28th EPS 2004, London;**[12]** M. Sugihara, Nucl. Fus. Vol 40 (2000) 1743-1755; **[13]** J.G. Cordey, Nucl. Fus. 43 (2003), 670-674; **[14]** Petty et al, Fus. Science tech. 43 (2003) 1.; **[15]** E. Joffrin et al, 20th IAEA Fusion Energy Conference (2004) & references therein ;**[16]** ITER physics basis, Nucl. Fus. 39 (1999) vol. 12; **[17]** M.R Wade et al, Nucl. Fus. 45 (2005) & references therein; **[18]** H. Weisen, 20th IAEA FEC EX/P6-31, 2004