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Implementation of plasma diffusion models in the code TOKES

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In large tokamaks operating in the ELMy H-mode, after each ELM the divertor emits eroded ionized material into SOL and then the impurities may deteriorate the plasma confinement. The repetitive diffusion of carbon impurity into the core and its accumulation during tokamak discharge was simulated with a newly developed self-consistent two-dimensional plasma equilibrium code TOKES [1, 2]. Now in the code a simulation approach with multi–mapping behaviour of poloidal magnetic flux surfaces covering the whole vessel is developed [3]. The radial transport of the plasma species is simulated with equations of magnetic hydrodynamics. Implementation of special triangular meshes and chains of magnetic layers that contain the plasma allows natural coordination of the plasma and neutrals in the vessel.

In this work the numerical approximation scheme of TOKES developed for plasma diffusion is checked comparing numerical and corresponding analytical solutions. The diffusion equation for a plasma species density *n* is taken in a simplified form, where some processes, such as mutual diffusions of species or convective motion are omitted:

$$\frac{\partial}{\partial t}(G_3 n) = \frac{\partial}{\partial x}\left(G_3 n D_{\parallel} \frac{(2\pi q)^2}{G_1^2} \frac{4\pi \mathrm{H}}{\omega^2} \frac{\partial p}{\partial x} \right) + \oint sR\sqrt{g}dy \qquad (1)$$

Here $G_m = \oint r^{m-2}\sqrt{g}dy$, R is the major radius, $\sqrt{g}$ determines the metrics of magnetic surfaces, x and y are the 'radial' and 'poloidal' coordinates, $\mathrm{H} = \frac{G_5}{G_3} - \frac{G_3}{G_1}$, p plasma pressure, $\omega = rB_\zeta$, q is the safety factor, s a species sources, $D_{\parallel}$ longitudinal magnetic diffusivity.

Being simplified, Eq. (1) remains still too complicated for analytic solution therefore a simple toroidal plasma configuration may be implemented that allows independent verification of results. The simplest configuration that TOKES can tackle without to degenerate drastically its features is a tokamak device of circular poloidal vessel's shape (see Fig.1), with approximately circular magnetic surfaces, which implies a very large major radius. Other tokamak parameters we are keeping relevant to those of the ITER [4].

The toroidal surface minor radius is chosen to be 3 m. The "tooth" in Fig.1 having size 0.8 m simulates the limiter and determines the minor plasma radius b be equal 2.2 m. The toroidal currents corresponding to the whole current $I = 15$ MA and constant current density inside the plasma have been then produced. The major radius $R_0 = 1$ km is chosen and the magnetic layers are generated that appeared having approximately circular poloidal shapes.

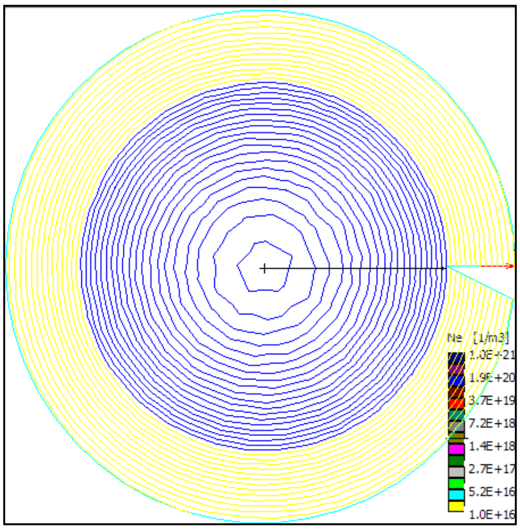

Fig. 1 The simplified tokamak configuration. The plasma occupies 20 of 34 layers

The distortions of the magnetic surfaces are due to the background triangular meshes The large aspect ratio $R_0/b \approx 455$ makes toroidal aberrations in the poloidal plane not visible and allows simplification of most important equations to be solved independently of the code. The toroidal magnetic field $B_\zeta = 5.3$ T becomes rather homogeneous over the vessel volume. The deuterium or, alternatively, helium plasma of the temperature T = 10 keV and density n = 10^{20} m^{-3} was created on the closed magnetic layers. The step of poloidal flux Δw, which is fixed across the magnetic layers, was adjusted so that 20 layers contain the plasma.

With the plasma equilibrium configuration Fig.1 all coefficients in Eq.(1) are drastically simplified. The comparison of different plasma parameters and geometrical factors, calculated analytically (lines) and those obtained numerically (the squares) shows the good agreement (Fig.2).

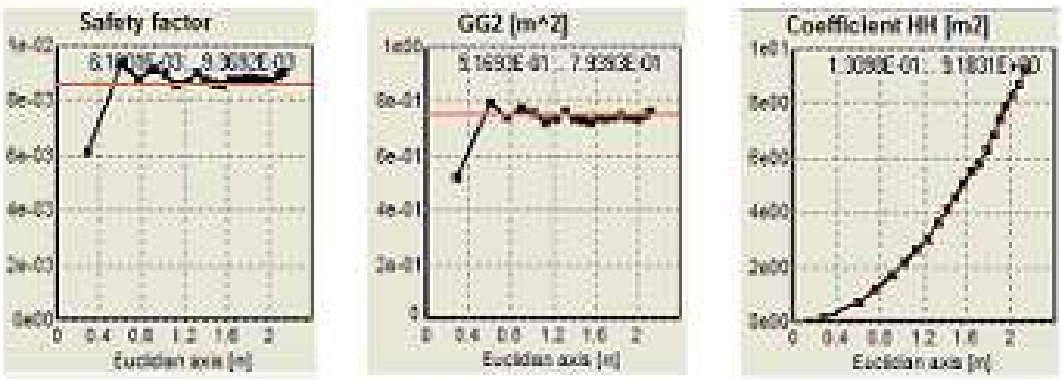

Fig.2 TOKES tests: Theory fits well the calculated functions (shown with squares).

The poloidal magnetic flux step Δw that provides a number X of magnetic layers inside the plasma follows as

$$\Delta w = \frac{I}{c}\frac{r_0}{X} \qquad \Rightarrow \qquad r = b\sqrt{\frac{x}{X}} \tag{2}$$

Here r is the minor radius. The layer index $x = 0..X{-}1$. For $I = 15$ MA, $r_0 = 1$ km and $X = 20$ it follows $\Delta w = 75$ Weber. The corresponding TOKES calculation with $\Delta w = 75$ Weber resulted in 20 layers occupied with plasma, which fits exactly to the theory (see Fig.1).

The diffusion of plasma was simulated with TOKES assuming that the T and the fuelling frequency ν_0 are constant in the plasma volume. The Eq.(1) then acquires the form:

$$\frac{\partial n}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(rD\frac{\partial n}{\partial r}\right) + \nu_0 n(0), \qquad D = \frac{1+2q^2}{4}\frac{Z+1}{Z^2}\rho_e^2 \nu_e$$

where Z is the ion charge state, ρ_e the electron gyro-radius and ν_e the ion-electron collision frequency.

Three different diffusion models were investigated both numerically and analytically: those with the classic diffusion coefficient ($D = D_c$), the Bohm diffusion ($D = D_B$) and some

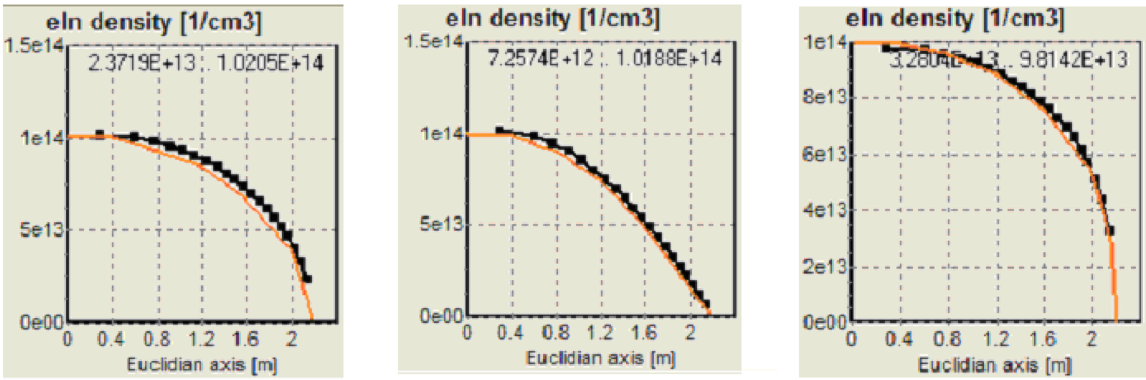

Fig.3 Stationary density for classic, Bohm and reference diffusion coefficients.

artificial reference diffusion ($D = D_r$). The D_c is proportional to n and D_B is constant for this model. As to D_r, we will assume it to be proportional to n^2: $D_r \propto n^2$, because, as we see, in this case the density profile is similar to that of H-mode. The results of comparison are presented on the Fig.3 showing a good fitting of numerical approach to the analytical one.

This internal test was additionally verified with the MHD transport code ASTRA for the same plasma configuration and diffusion coefficients. The corresponding plasma density

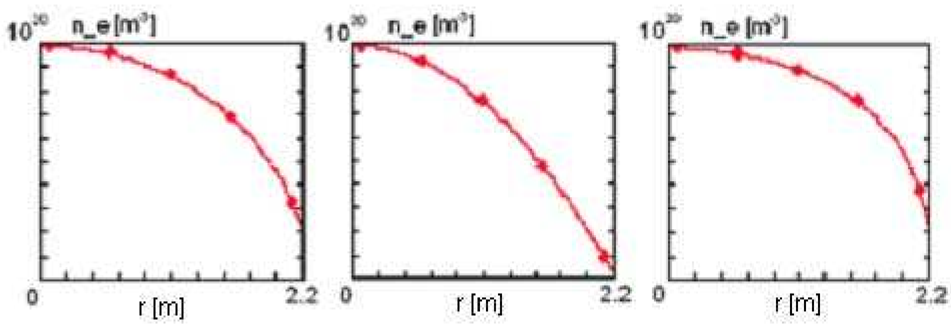

Fig.4 Stationary density produced by ASTRA for classic,Bohm and reference diffusivities

profiles calculated by ASTRA are shown on Fig.4. The agreement between TOKES and ASTRA calculations seems good.

Being developed for analysis of the transport problem, the code ASTRA is intrinsically one-dimensional and cannot describe properly the two-dimensional transport in the real tokamak plasma. Nevertheless, the previous ASTRA calculations can serve as a good benchmark case for us to introduce the non-classical IFS-PPPL transport for the ions and the RLW transport for the electrons as the core plasma transport [5]

$$D = 0.1\cdot\left(2\chi_e + \chi_i\right) \quad \left[\begin{aligned} \chi_i^{ITG} &= \chi_i^S \left|\frac{R\nabla T_i/T_i - \left(R\nabla T_i/T_i\right)_{crit}^{IFS-PPPL}}{\left(R\nabla T_i/T_i\right)_{crit}^{IFS-PPPL}}\right| \cdot H\left(\left|\frac{\nabla T_i/T_i}{\left(\nabla T_i/T_i\right)_{crit}^{IFS-PPPL}}\right| - 1\right) \\ \chi_e^{RLW} &= \chi_e^S \left|\frac{\nabla T_e - \left(\nabla T_e\right)_{crit}^{RLW}}{\left(\nabla T_e\right)_{crit}^{RLW}}\right| \cdot H\left(\left|\frac{\nabla T_e}{\left(\nabla T_e\right)_{crit}^{RLW}}\right| - 1\right) \end{aligned}\right. \tag{3}$$

where H is Heaviside function, diffusivities $\chi_i^S = 5\,m^2 s^{-1}$ and $\chi_e^S = 1\,m^2 s^{-1}$ are numerically equal to the ion and electron transport stiffness, and the critical temperature gradients are given in [5].

Reduction of plasma diffusivity due to the rotational shear and the magnetic shear is responsible for the formation of the edge transport barrier [5]

$$\chi_{E\times B} = \chi_0 s^{-1.8} / \left(1 + \omega_{E\times B}^2 / \gamma_0^2\right), \quad \omega_{E\times B} = \frac{RB_\theta}{B}\frac{\partial}{\partial r}\left(\frac{E}{RB_\theta}\right), \quad E = \frac{\nabla p}{n_i e} \tag{4}$$

where χ_0 is a transport coefficient in absence of magnetic shear s and γ_0 an estimate of ITG grows rate in the absence of stabilisation. Implementing the Alfven drift turbulence transport, which triggers the L-H transition [6] and restricting the plasma pressure gradient by the ballooning limit, we plan to obtain a two-dimensional phenomenological model for the H-mode tokamak plasma, which can be used for estimation of wall plasma loads.

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