Transport codes for magnetic fusion: ASTRA (overview of applications)

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• Objectives and key elements of integrated scenario modelling for tokamaks

• Core transport modelling: transport equations and numerical tools

• Physics applications: ASTRA
  - interpretative analysis
  - validation of transport models
  - scenario development
  - plasma control
  - beyond core modelling: integrated core-SOL-divertor simulations

• Summary, perspectives and open issues
Objectives of transport modelling

- Interpretation of existing experiments
- Development of empirical models based on experimental observation
- Validation of theory-based models – link between experiments and theory
- Prediction of future experiments on existing tokamaks and optimisation of operational scenarios in modelling
- Prediction for future devices (ITER, DEMO, JT60-SA …)
Integrated scenario modelling: key physics processes

- Energy, particle and momentum transport: interaction of charged particles with micro-turbulence, test particles in stochastic magnetic fields, ...
- MHD events: sawteeth, NTMs, fishbones, ELMs
- RF heating and current drive: plasma-wave interaction, fast particle physics
- Neutral beam injection, gas puff, pellets: plasma-neutral interaction, atomic physics, fast ion physics
- Plasma equilibrium and shape control
- Divertor and SOL physics
- Impurity and radiation

Multi-scale time and space, highly non-linear coupling between different processes
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• Summary, perspectives and open questions
Braginskii (reduced) transport equations

- Based on integration of kinetic equation
- Maxwellian distribution functions with small perturbations

\[
\frac{1}{V_i} \left( \frac{\partial}{\partial t} - \frac{\dot{B}_0}{2B_0} \frac{\partial}{\partial \rho} \right) (V_i n_e) + \frac{1}{V_i} \frac{\partial}{\partial \rho} \Gamma_e = \text{particle sources and sinks}
\]

\[
\frac{3}{2} (V_i)^{-5/3} \left( \frac{\partial}{\partial t} - \frac{\dot{B}_0}{2B_0} \frac{\partial}{\partial \rho} \right) \left[ (V_i)^{5/3} n_e T_e \right] + \frac{1}{V_i} \frac{\partial}{\partial \rho} \left( q_e + \frac{5}{2} T_e \Gamma_e \right) = \text{electron heating (including waves) & heat losses}
\]

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\]

\[
\sigma_{||} \left( \frac{\partial \psi}{\partial t} - \frac{\rho \dot{B}_0}{2B_0} \frac{\partial \psi}{\partial \rho} \right) = \frac{J^2 R_0}{\mu_0 \rho} \frac{\partial}{\partial \rho} \left( \frac{G_2}{J} \frac{\partial \psi}{\partial \rho} \right) - \frac{V_i}{2\pi \rho} \times \text{bootstrap and externally driven current densities (including HF wave driven)}
\]

- System closure via fluxes (\(\Gamma_e, q_e, q_i, j_{BS}, j_{CD}\)) expressed as a functions of \(n_e, T_e, T_i\) and their gradients – either from theory or from experimental observations (empirical)
Presently in use: ASTRA, CORSICA, CRONOS, FASTRAN, JETTO, ONE-TWO, (P)TRANSP, TOPICS, TSC
ASTRA specific characteristics:

- Interactive mode
- Code compiler
- Platform for coupling rather than the code: different combination of modules can be used (w/o transport solver)
- 10 diffusion-type equations built-in (turbulence amplitude, toroidal and poloidal velocity, different plasma species, ...)
- Local deployment worldwide:
  - tokamaks: ASDEX Upgrade, CDX-U, COMPASS, DIII-D, FTU, Globus-M, JET, JT-60U, KSTAR, MAST, T-10, TCV, TFTR, Tore Supra, ITER
  - W7-X stellarator
  - RFX
  - State university of Mexico (UNAM), Imperial College (London, UK)
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Estimation of effective diffusivities: $\chi \sim \left[ \int (P_{\text{heat}} - P_{\text{loss}}) \, dv - (3/2) \frac{dW}{dt} \right] / (n \nabla T)$

Nishijima et al, PPCF 2005 (AUG)

- similar heating power (17 MW), toroidal field (2.7 T), plasma current (2.5 MA), initial density in discharge with and without Ar seeding
- thermal ion diffusivity reduces with Ar seeding
Validation of core theory-based transport models: current ramp up

NBI assisted Ipl ramp up at JET: accurate prediction at low NBI power, but GLF23 builds an ITB at high power. Even larger Ti over-prediction with 10 MW [Voitsekhovitch et al PPCF 2010]

\#72516, 4 MW of NBI

Bohm-gyroBohm, GLF23, re-scaled Coppi-Tang, data

Te

Ti

\#72511, 7 MW of NBI

Te

Ti

OH Ipl ramp up at AUG: satisfactory GLF23 prediction later during the Ipl ramp, while the discrepancy at lower currents [Fable et al, NF 2011]
Validation of theory-based transport models: L-mode

Theory-based core transport models (GLF23, MMM07) in combination with DRIBM model for edge transport

JET L-mode #79575

GLF23+DRBM/ASTRA
MMM08+DRBM/PTRANS

More L-mode examples for DIII-D, JET and TFTR are in Rafiq et al IAEA 2012
Validation of core theory-based transport models: H-mode

- AUG: scan in $n_e \sim (3.85-6.2) \times 10^{19} \text{ m}^{-3}$, $I_{pl} (0.4-1.2 \text{MA})$, $P_{NBI} (2.5-12.5 \text{MW})$

- $T_i$ profiles are stiff (similar $L_{Ti,cr} = - (T_i/\nabla T_i)_{cr} \sim 4$ in all shots) while $T_e$ change shape at low $n_e$

- $\partial \chi / \partial (R/L_Ti)$: 20 m$^2$/s for IFS-PPPL, 2.5 m$^2$/s for Weiland model

- Accurate predictions with ITG-based models: Weiland, IFS-PPPL

- CDBM model failed to predict linear relations between core and edge $T_i$

Similar study of stiffness at JT60-U and validation of MMM and RLW models: Mikkelsen et al. NF 2003

Figure 5. Simulated core to edge temperatures; blue diamonds correspond to the Weiland model, red stars to IFS/PPPL, green crosses to GLF23 and black triangles to CDBM: (a) $T_i$, (b) $T_e$. In (a) the line reproduces the experimental points, being the same as that in Fig. 1(a). In (b) the lines reproduce the experimental points at low and high $T_i(0.8)$, as in Fig. 1(b).
Validation of core theory-based transport models: impact of stiffness on plasma performance in H-mode

Voitsekhovitch et al, ISM Working Session, November 2010

JET H-mode (79698):
- 3.6 T / 4.5 MA
- DD / C wall
- 23 MW of NBI, 2.5 MW of ICRH
- projection to DT (same density and rotation velocity)

Prediction for DT plasma:
- Increase of NBI power up to 35 MW is envisaged
- increase of $\chi_s$ with power $\rightarrow$ little/no effect on temperature and Q
AUG: weak \(ExB\) shear effect on core confinement, but improved pedestal [Na et al, NF 2006]

JET: \(ExB\) shear stabilisation is important for core confinement [Voitsekhovitch et al, NF 2009]
Validation of core transport models:
Internal Transport Barrier

ITB with monotonic $q$ in AUG18695: GLF23 (solid), Weiland model (dashed). Measurements are shown by symbols [Tardini et al NF 2007]

ITB with semi-empirical ExB and magnetic shear stabilisation model for TFTR, DIII-D and JET [Voitsekhovitch et al, Phys. Plasmas 1999; Czech J. Phys. 1999]


- simulations of ITB dynamics is not always successful
- estimation / measurements of ExB shear?
- anomalous poloidal rotation [K Crombe et al PRL 2005]?
- more complicated physics of turbulence stabilisation?
Non-linear ExB shear quench rule in JET hybrids?

- $ExB$ shear quench rule in GLF23: $\gamma_{net} = \gamma_{max} - \alpha_E \gamma_{ExB}$ (0.5 < $\alpha_E$ < 1.5) [Waltz et al, PoP 1997]

- $\alpha_E$ determined in gyrofluid & gyrokinetic turbulence simulations (large $\alpha_E$ range depending on physics assumptions and plasma conditions)

- here $\alpha_E$ is adjusted in self-consistent modelling of $Te$, $Ti$, $ni$ and $Vtor$ for each of 7 JET hybrid shots performed under different conditions

- $\chi_\phi = Pr\chi_i$, Pr=0.3 for shots with strong $ExB$ shear stabilisation, otherwise larger Pr uncertainty

- non-linear ExB shear quench rule ($\alpha_E$ increases with rotation)?

- or other hidden effects? turbulence stabilisation by fast ion pressure?
Development of advanced scenario with RF heating and current drive (Tore Supra)

- fully non-inductive operation
- improved confinement (flat or reversed q-profile)
- validated transport models (strong coupling between transport, q, pressure, heating and current drive)

- start after breakdown
- LHCD at low density to form slightly reversed q at low $n_e$,
- increase density and heating: LHCD efficiency reduces, but BS current replaces LHCD
- LHCD timing is important: $q_0$ below 1 with late LHCD
- steady-state: $I_{BS}/I_{tot} = 0.4$, $I_{LH}/I_{tot} = 0.6$
- scenario optimised manually, plasma control is desirable
Plasma control algorithms

Two point current profile control

[Moreau et al Nucl. Fusion 1999]

Actuators:
- Loop voltage
- LHCD and FWEH power

(1) Control of $q$ at reference radius $r_{ref}$ using loop voltage

$$U_{ext}(t) = C_U \left( \frac{1}{q_{ref}} - \frac{1}{q(r_{ref}, t)} \right)$$

if $q(r_{ref}) > q_{ref}$ then $U_{ext} > 0$
if $q(r_{ref}) < q_{ref}$ then $U_{ext} < 0$

(2) Replacement of OH current with a NI current at reference radius, i.e. $(E_{//}(r_{ref}) = 0)$ and control of $q_0$

$$\delta P_{central} = C_0 E_{//} (r = 0, t)$$

(4) Control of LH deposition via LH spectrum

(5) Burn control for ITER via density
- feedback control starts at 15 s, $q_{0\text{ref}} = 4.5$, $q(r_{\text{ref}=0.5}) = 1.7$
- stationary reversed shear configuration and ITB achieved
- same reference q values are achieved with different transport models, but different power is needed
Control algorithms allow to achieve the prescribed q values, but:

- Long relaxation time and large transient deviations from reference: gains adjustment for smooth and relatively fast evolution?

- q profile is not controlled apart from two reference points (MHD stable?)

Further developments for control of kinetic and magnetic profiles:

D. Moreau et al, Nucl. Fusion 51 (2011) 063009
Beyond core modelling: core-SOL-divertor simulations including impurity

Core / pedestal: impurity and main species, but simplified sources and transport (H98(y,2)), no equilibrium

**COREDIV:**
- impurity simulations (ionization, CX, recombination, transport)
- self-consistent particle source from divertor (sputtering cross-sections)
- parallel losses in SOL
- core and SOL radiation

**Core transport codes:**
- fixed/free boundary equilibrium
- H&CD: NBI (NUBEAM, FP), ICRH (TORIC)
- theory-based transport: GLF23, TGLF, MMM, NCLASS

**Compromise between physics complexity and simulation speed**

**JETTO (ASTRA, CRONOS): sophisticated equilibrium, transport, H&CD modules, current diffusion, but no impurities/SOL/divertor**

\[
\begin{align*}
\Gamma_i, P_{\text{rad}}, P_{\text{ion}}, P_{\text{Brem}}, \\
P_{\alpha}, Z_{\text{eff}}, P_{j}^{\text{rad,k}}, n_{j}^{k}, \\
\Gamma_{j,\text{divertor}}, Q_{\text{divertor}}' \\
\text{Helium, impurity} \\
\end{align*}
\]

\[
\begin{align*}
R, a, B_{\theta}, \varepsilon, \kappa, I_{p}, \\
H_{98}, P_{\text{aux}}, P_{\text{aux}}^{e}, \\
\langle n_{e} \rangle, n_{\text{sep}}, \\
T_{e(i),\text{sep}}, T_{e(i),\text{ped}}, P_{\text{ped}} \\
\end{align*}
\]
- Without impurity seeding, the radiation is 33% and $P_{\text{sep}} > P_{\text{LH}}$ (H-mode), but power to plate is too large (76 MW)

- Neon seeding reduces the power to plate, but $W$ production & radiation increases ($W$ self- and sputtering by D is replaced with sputtering by N) $\rightarrow$ power through separatrix is below L-H power threshold

- model for $W$ diffusion and pinch?
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Summary, perspectives and open issues

• Broad application domain for transport codes (multi-physics, multi-machine) has been illustrated

• Good predictive capabilities of theory-based models achieved in a number of cases illustrate applicability of transport theories

• Still more work needs to be done (mechanisms of suppression of anomalous transport, impurity transport, …)

• Improvement of numerics for stiff transport

• Integrated modelling tools are needed (coupled transport, free boundary, MHD, core-SOL-divertor)
Future numerical tools

- ASTRA as a prototype of European Transport Solver (ETS_A):

European Transport Solver: a schema of the workflow [Kalupin et al IAEA 2012]

- Next step: IMAS - planning, execution and analysis of ITER pulses
Grigory Pereverzev developed and maintained a unique, flexible, user-friendly, multi-machine transport code used for a number of interesting and important physics studies - a very valuable contribution, highly appreciated by fusion transport community!