A full discharge tokamak flight simulator

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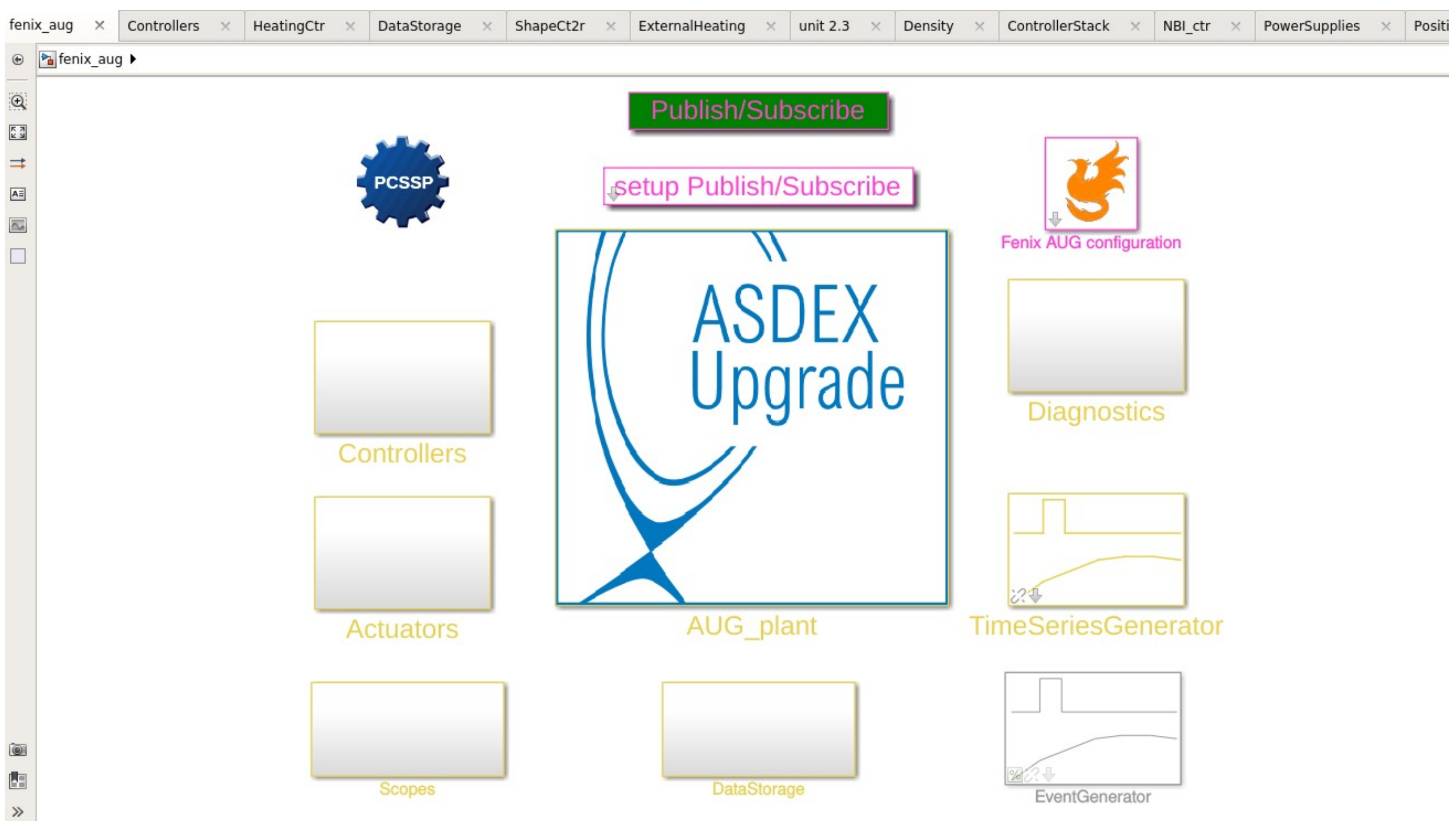
KEY POINTS:
- **Discharge preparation and prediction useful to improve pulse reliability (crucial @ ITER)**
- **Use as input only Pulse Schedule (no pre-existent experimental data needed)**
- **Kinetic and magnetic control as in the real-life tokamak plasma.**
- **Reduced physics models from 0D to 1D kinetic and 2D equilibrium to perform fast simulations**
- **Use state-of-the art codes and control framework**

Flight simulator Fenix @ IPP

› First ever fully pulse-schedule based tokamak flight-simulator Fenix [1,2] built at IPP for ASDEX Upgrade
› Built incorporating a plasma model (ASTRA+SPIDER transport-equilibrium solvers [3,4]) into the PCSSP Simulink framework [5]
› PCSSP is the framework chosen for ITER to embed the control system and its modules



› TimeSeriesGenerator converts a Pulse Schedule into waveforms for the various actuators and requests for fueling, heating, shaping, etc.
› Controllers, Actuators and sensors ↔ models for heating, fueling, coil system, and diagnostics
› Computational time for 1 AUG full discharge ~ 1-3 minutes depending on resolution and models employed
› Developed for both ASDEX Upgrade and for DEMO tokamaks

Reduced physics models in ASTRA-SPIDER

- Globally:
* **quasi-stationary MHD equilibrium model (Grad-Shafranov equation)**

- Locally:
* **Core transport model for Te, Ti, ne, nZ, j||**

* **Edge/pedestal models, L-I-H H-I-L transition models**

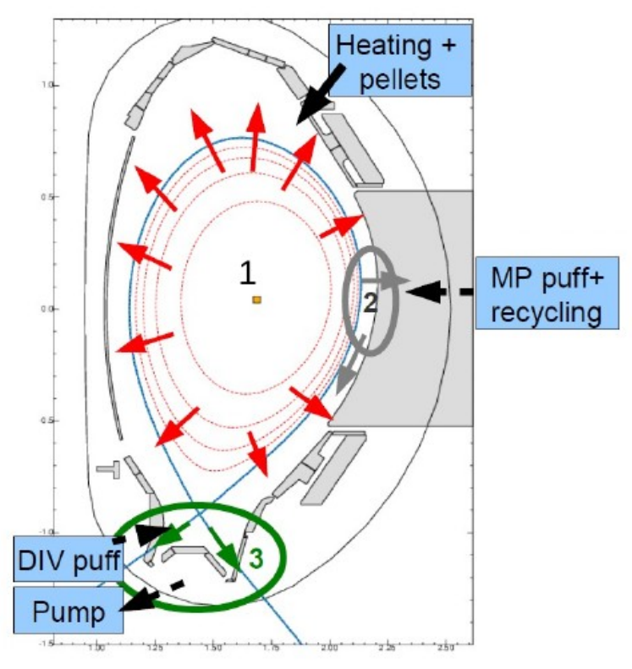
* **SOL transport model for exhaust (heat flux) and for particle balance (fueling)**

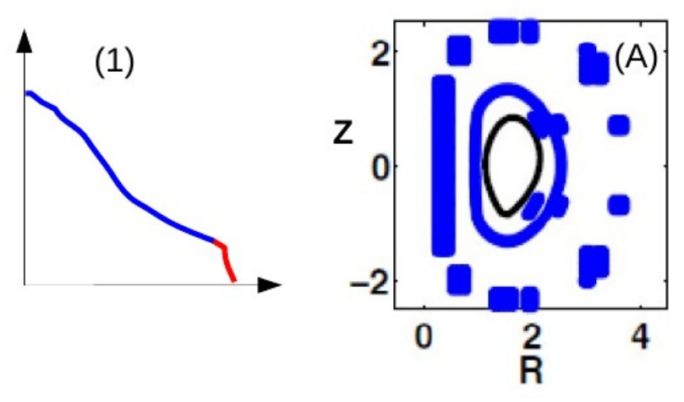
* **SOL/divertor model for exhaust and particle balance**

* **Plasma-divertor and plasma-wall interaction models**

Predictability
~ 100%
~ 70%
~ 30%
~ 10%

* (1) **Core** and pedestal plasma modeled smoothly but with different transport models

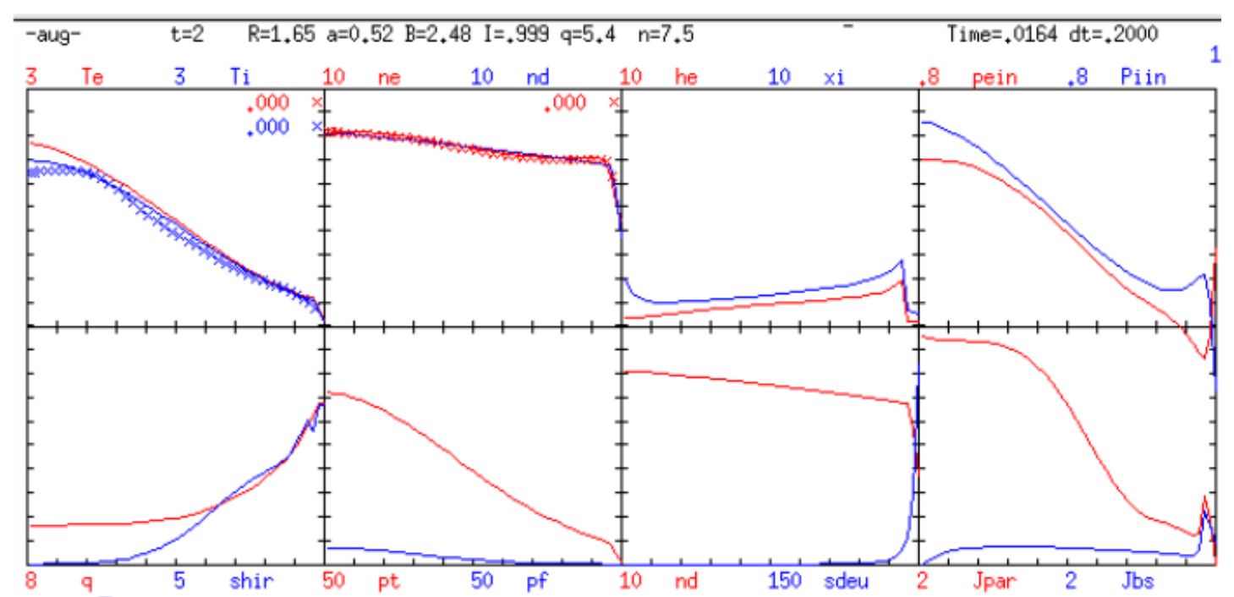




* (2-3) SOL exhaust / particle balance models

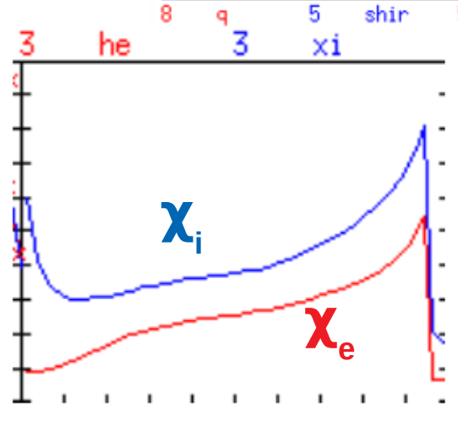
* Heating and pellet models

* (A) Equilibrium model



* The core region, defined from magnetic axis to pedestal top radius, is determined by assigning diffusivities with some models

* For T_e and T_i:
- simple gyro-Bohm model scaled to give H ~ 0.4-0.5 in L-mode, H ~ 1 in H-mode
- $\chi_e^{turb} \sim 0.6\, \chi_i^{turb}$ → this is usually observed in standard scenarios, not so much in strongly electron heated or advanced scenarios



› Electron density follows from quasi-neutrality. Solve for deuterium/tritium and impurity particle transport in the core: diffusion scaled from heat conductivity, pinch follows simple theoretical arguments
› Simple Kadomtsev-based sawtooth model
› Analytical neoclassical transport formulas
› L-H transition model based on $P_{sep,ions} > P_{LH,Martin}$ / 2 (extrapolation to low density linear)
› Pedestal model based on "average-ELM" model, with pedestal top pressure clamped at a value given by a scaling (result of EPED calculations or empirical scaling)

$$\chi \sim \left(\beta_{ped,top}/\beta_{crit}\right)^4$$

› SOL/divertor model for particles: global balance divided into "zones". Each zone connects via "diffusion-like" terms. Gas puff acts as a source, pump as a sink
› SOL/divertor model for energy: 0D scaling for plate temperature as a function of power entering the SOL at mid-plane and impurity content. Can lead to detachment when radiated power exceeds a certain threshold. Not yet tested!

›

$$\frac{dN_{j,sol}}{dt} = \Gamma_{j,plasma} + \alpha_j \Gamma_{j,midp} + D_j\left(N_{j,divr}/\varepsilon_j - N_{j,sol}\right) + D_{j,w}\left(-1+\alpha_j(1-R_j)\right)N_{j,sol}$$

$$\frac{dN_{j,divr}}{dt} = \Gamma_{j,divr} - \Gamma_{j,pump} - D_j\left(N_{j,divr} - N_{j,sol}\varepsilon_j\right)$$

* neutrals in the plasma: $n_{0,j} \sim (1-\alpha_j)\left(\Gamma_{j,midp} + (1-R_j)D_{j,w}N_{j,sol}\right)$

* $n_{j,sep} = N_{j,sol}/V_{sol}$; $n_{j,divr} = N_{j,divr}/V_{divr}$

* All Ns and Γs are in 10^{19} [p/s], Ds are in 1/s
* α_j : ionization fraction in the SOL, parametrization from Teobaldo L.
* R_j : recycling coefficient
* ε_j : enrichment coefficient ~ $n_{divr}^{0.67}$ → results in $n_{sep} \sim \Gamma_{divr}^{0.3}$

- Heating models:
  - * RABBIT [M. Weiland et al 2018 Nucl. Fusion 58 082032] for NBI
  - * TORBEAM [M. Reich et al., Fus. Eng. & Des., Vol 100, 2015, 73] for ECRF
  - * no reduced model for ICRF yet!

- Fueling models:
  - * Parametric regression obtained running HPI2 code [B. Pegourie, et al. Nucl. Fusion, 47 (2007), p. 44] on a set of plasmas
  - * NEUT for incoming neutrals

* SPIDER solves the free-boundary Grad-Shafranov equation and the circuit equations in this way:

- GS solution: computed every $dt^{(eq)}$ time step

$$\Delta^*\psi = 2\pi R \mu_0 j_\phi + \Sigma_k I_{k,coils} + (wall\ currents)$$

- Circuit equations: computed every dt time step

$$V_k = R_k I_k + L_k \frac{dI_k}{dt} + (mutual\ inductances\ between\ coils) + \frac{d\psi_{pl-coil,k}}{dt^{(eq)}}$$

* where dt is common between ASTRA and the circuit equations (and it is a integer fraction of or equal to the Simulink one)

* Since the bottleneck is the GS solver, this scheme allows much faster runs without too much loss in accuracy (unless $dt^{(eq)} >> dt$)

* Here are the inputs and outputs:

* Inputs:
- plasma pressure, current profile
- coil/wall/limiter description
- coil voltages

* Outputs:
- coil/wall currents
- poloidal flux map
- plasma boundary/shape/internal equilibrium

* employed resolution: 65x65 or 129x129 in R/Z, with 31 radial and 35 or 80 poloidal points in the plasma

* explicit time stepping (not so accurate but very fast): 1 s of real time done in 10-25 s of simulation time, depending on time step chosen for GS equation solver

* A rather hard problem (numerically speaking) is to get I_p consistent with the OH transformer and the mutual inductances

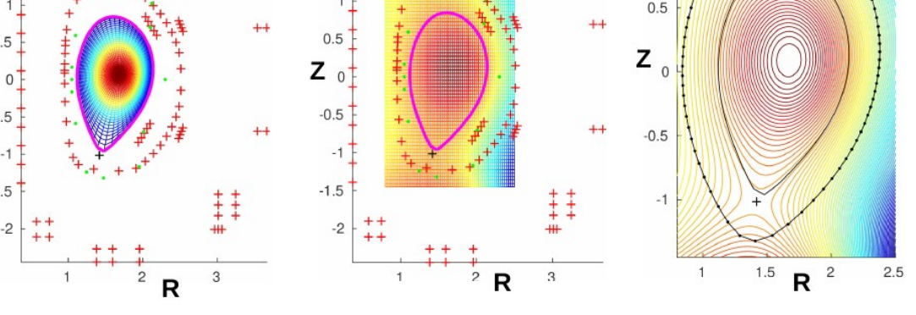
* Done in SPIDER by solving this equation implicitly:

$$\boxed{\psi_b + G_b\left[\frac{d\psi}{d\rho}\right]_b} = \boxed{\psi_{ext}}$$

Unknown
Source term

* Where G_b is a geometrical parameter coming from SPIDER, and Ψ_{ext} is the flux created by the coils

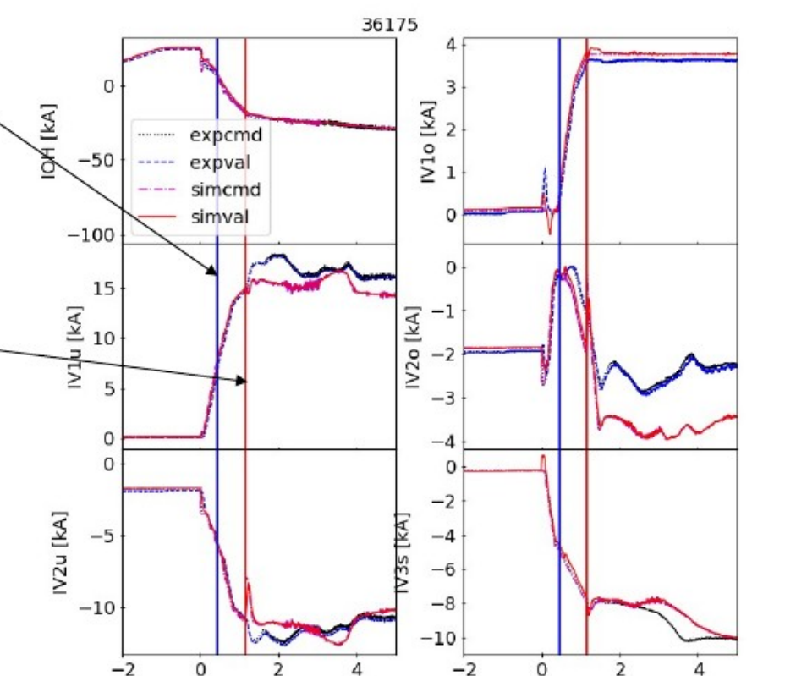
* This equation makes the system stable and allows to relate the OH swing to the actual plasma current via plasma resistance (since ASTRA solves for current diffusion in the plasma)



Full discharge prediction for ASDEX Upgrade

› Here several full discharge runs are presented, detailing the various phases and peculiarities

- **Pre-plasma, plasma ramp-up and shaping**

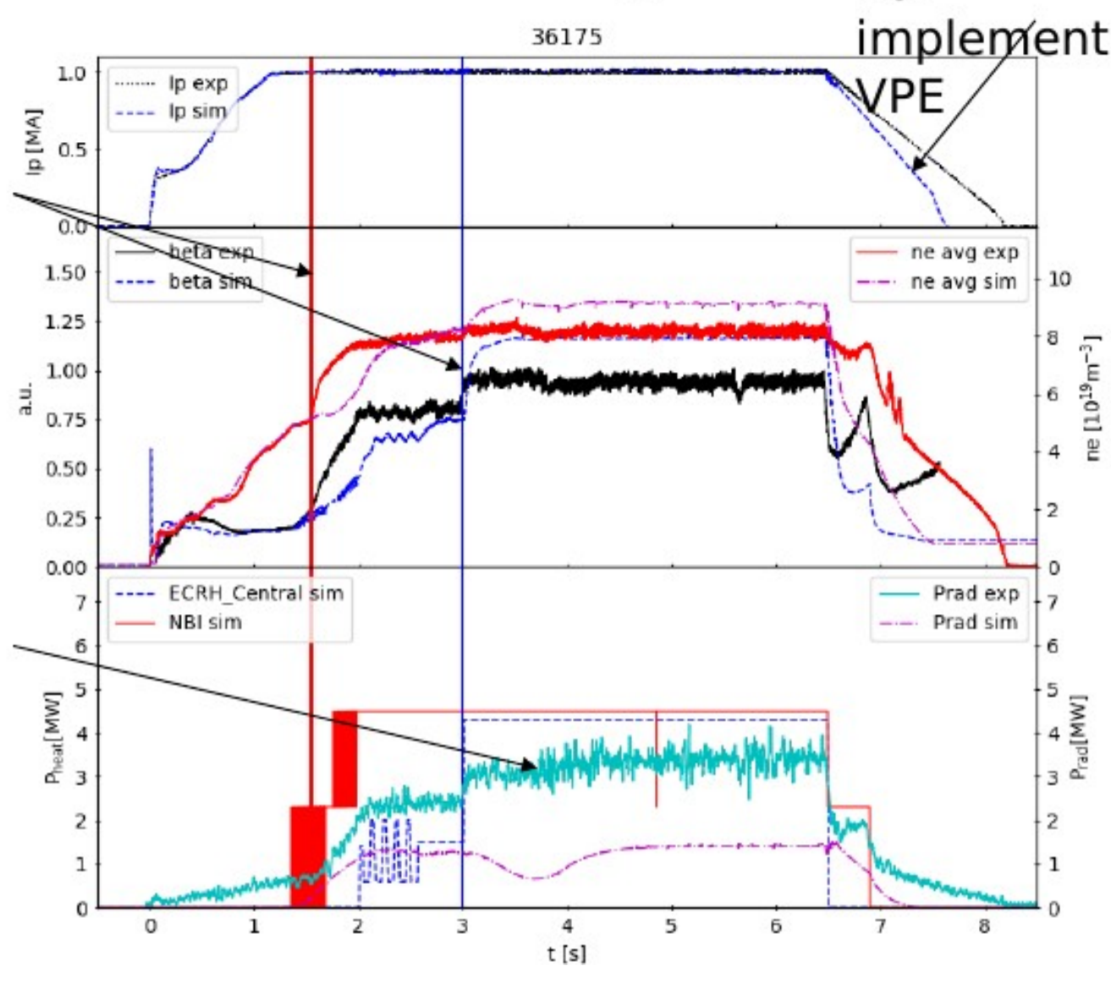


X-point formation: different edge current

Feedback shape mode switching

> simple breakdown model used, need better one
> edge current prediction important to determine current when strike point position control is employed

- **Simulating the entire discharge**



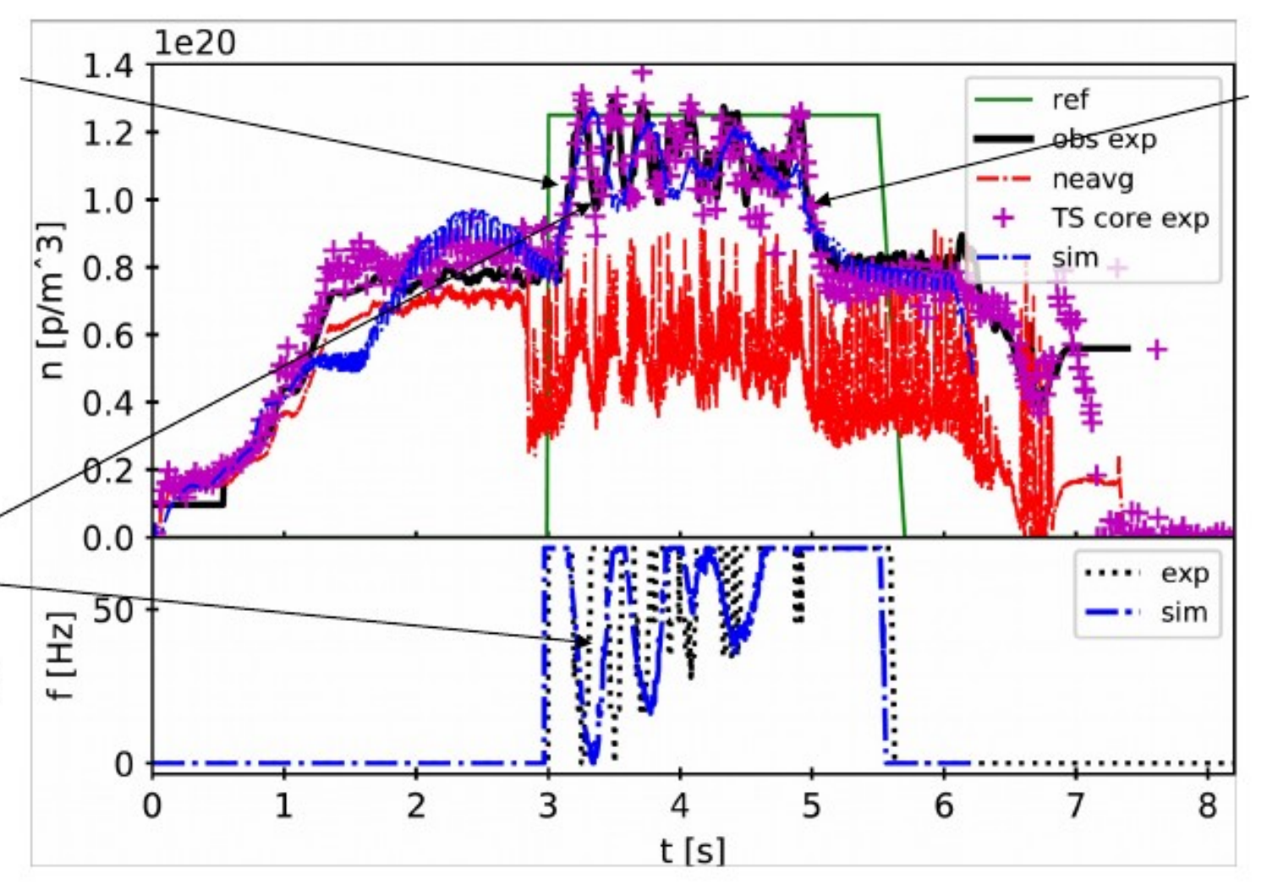
H-mode starts earlier in the experiment. Solution: tune the transport coefficients

Radiation discrepancy:

Ramp-down: not implemented VPE

Pellet fuelling #34182 with the realistic AUG pellet actuator model

Pellet transport model fitted according to plasma density response to one pellet

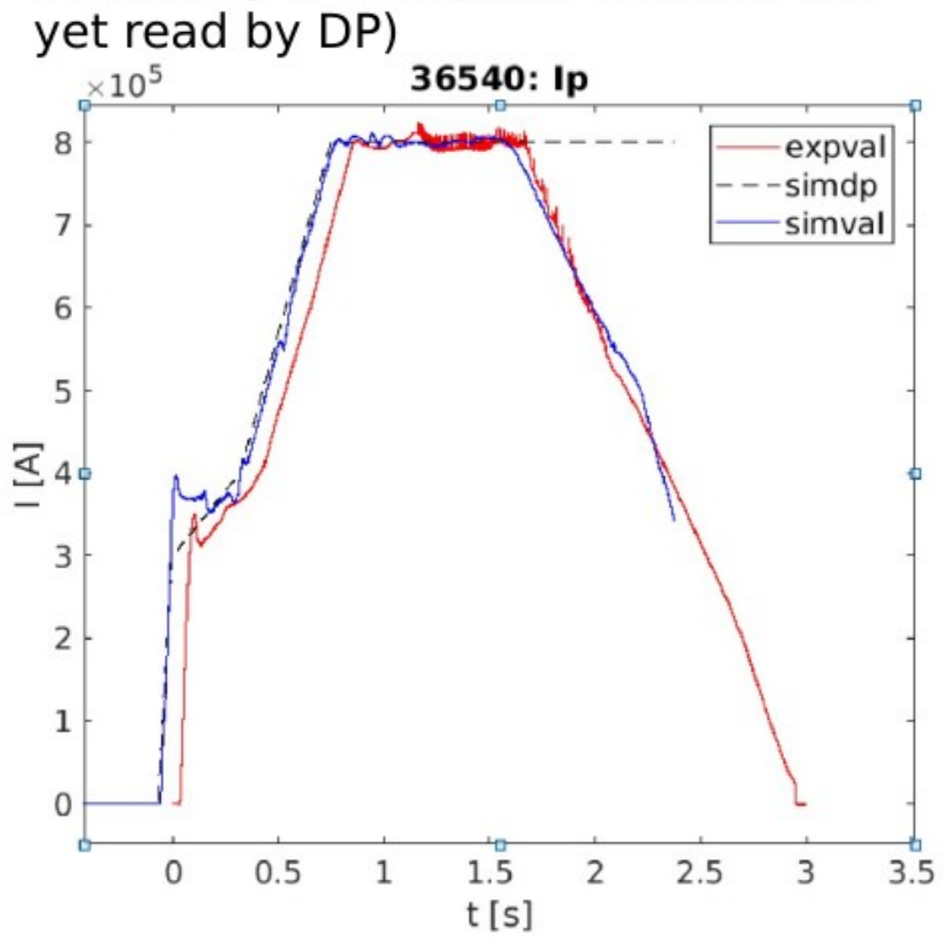


Pellet flight time

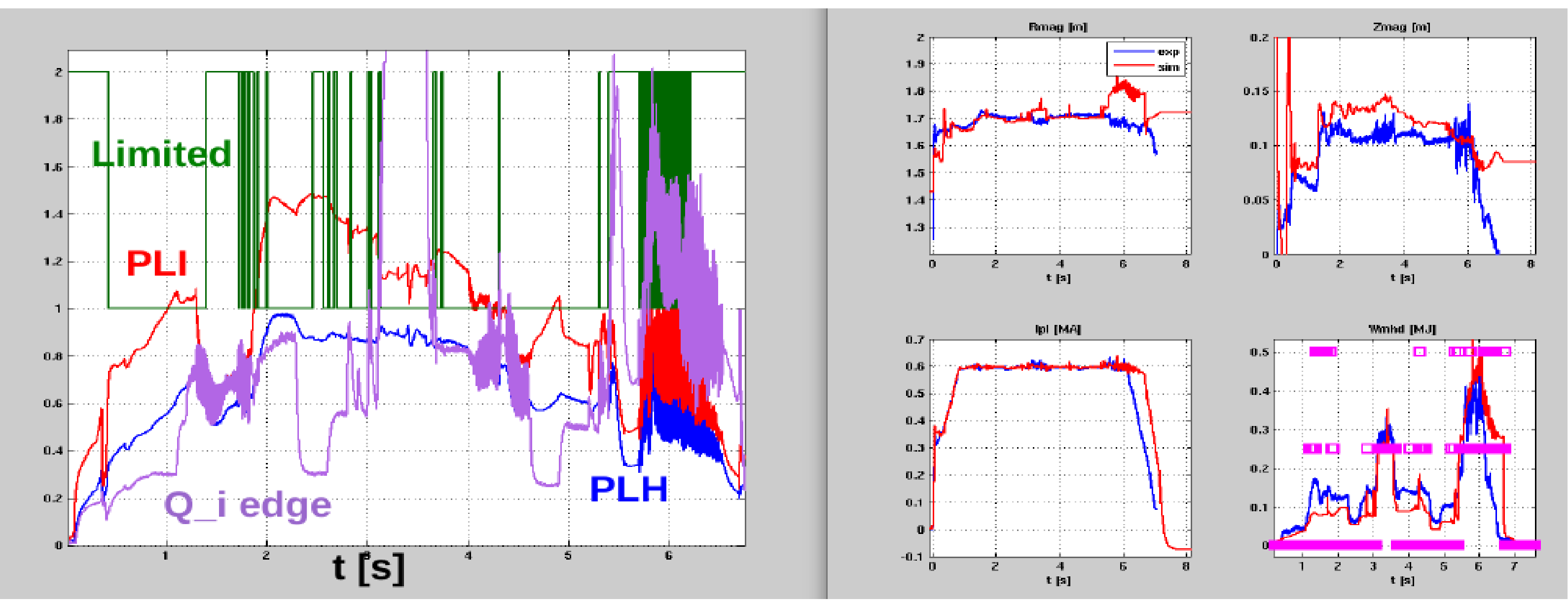
96 pellets

Similar oscillations

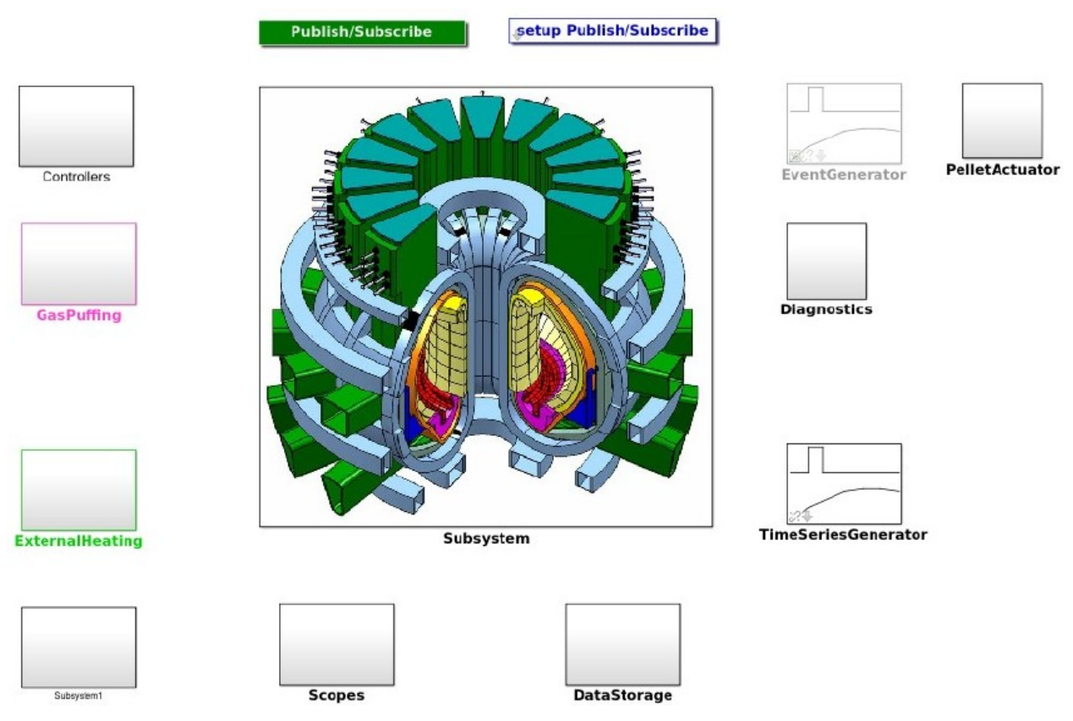
Fenix simulated discharge with wrong limit setting (offset in time is caused by 0-time point which is not yet read by DP)



- Negative triangularity discharge #36026
> Capture several phase transitions (from L-mode to dithering L-H transitions to H-mode) → interaction between shaping and edge power



DEMO studies



Kinetic control scheme

- Fusion power P_{fus}
  - Target: 2 GW
  - Actuators: NBI, pellet D/T ratio, pellet frequency
  - Diagnostics: $P_{fus} = 5 \times P_\gamma \sim n_{neutron}$
- Pedestal top electron density Greenwald fraction, n_e/n_{GW}
  - Target: 0.8 – 0.95 (given by density limit – operational limit)
  - Actuators: pellet frequency
  - Diagnostics: *electron density at r/a = 0.94*
- Divertor temperature
  - Target: < 5 eV
  - Actuators: Ar or Kr gas puff to divertor
  - Diagnostics: *divertor temperature*
- Separatrix power P_{sep}
  - Target: P_{sep} > 170 : 200 MW (about 1.2 P_{LH})
  - Actuator: Xe gas puff to midplane
  - Diagnostics: $P_{sep} = P_\alpha + P_{NBI} + P_{ECRH} - P_{rad}$
- NTM control
  - Target: small or no island
  - Actuator: ECCD at rational surface
  - Diagnostics: *electron temperature profile*

- DEMO parameters (scenario 2019):
  - **R = 8.94 m**
  - **a = 2.88 m**
  - **B_T = 5.74 T**
  - **Ip = 18.21 MA**
  - **n_{avg} ~ 8 * 10^{19} m^{-3}**
  - **k = 1.7**
  - **triang = 0.33**

- H_{98} = 1 (based on the ITER98(y,2) scaling for type-I ELMy H-modes)

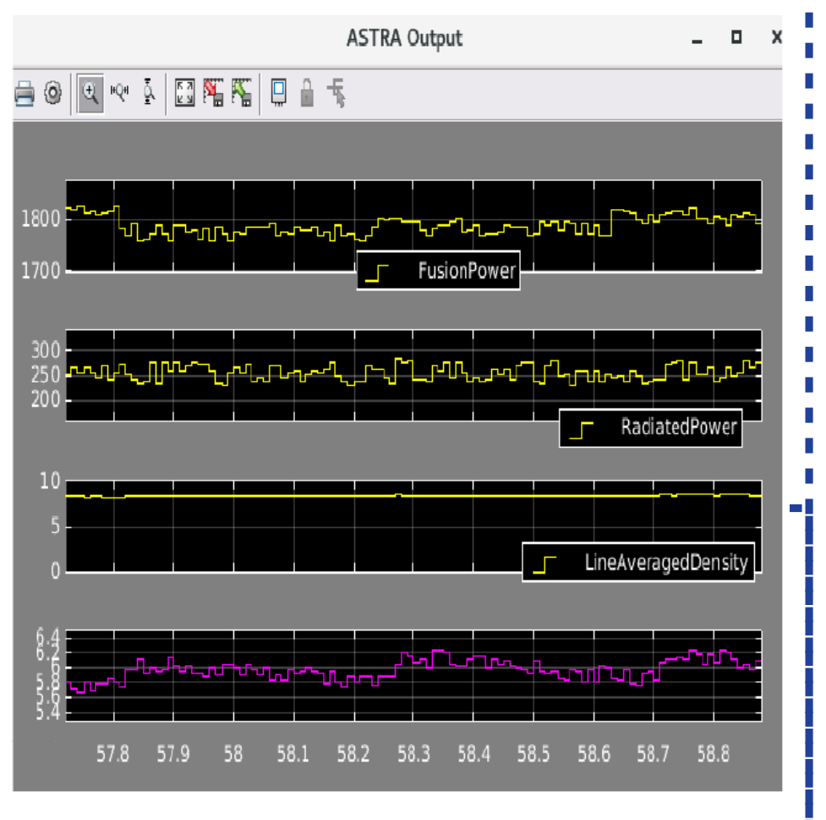
- How sensitive is sytem to noise & errors in sensors? Explore P_{sep} quantity
- P_{rad} imposes strong constraints on signal quality due to requirements to be above LH transition and below a certain value for divertor protection

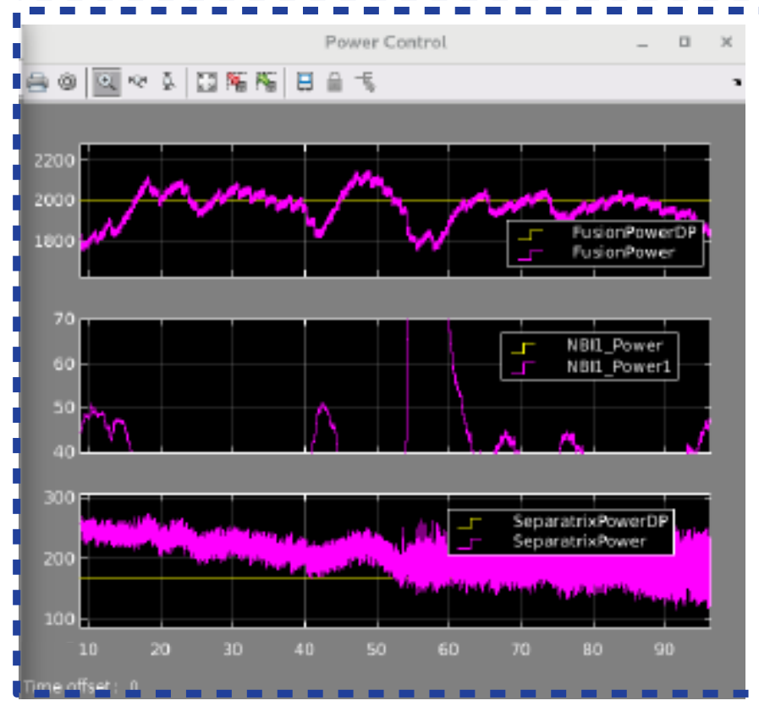
* Palpha: 1% error, 0.4 s integration time

* Prad: 10% error, 0.5 s integration time

* ne profile: 2% error, 1 ms integration time

- but we would also need:
> betapol
> temperature profile
> detachment parameters





- Oscillations of fusion power due to fueling (intermittent pellets)

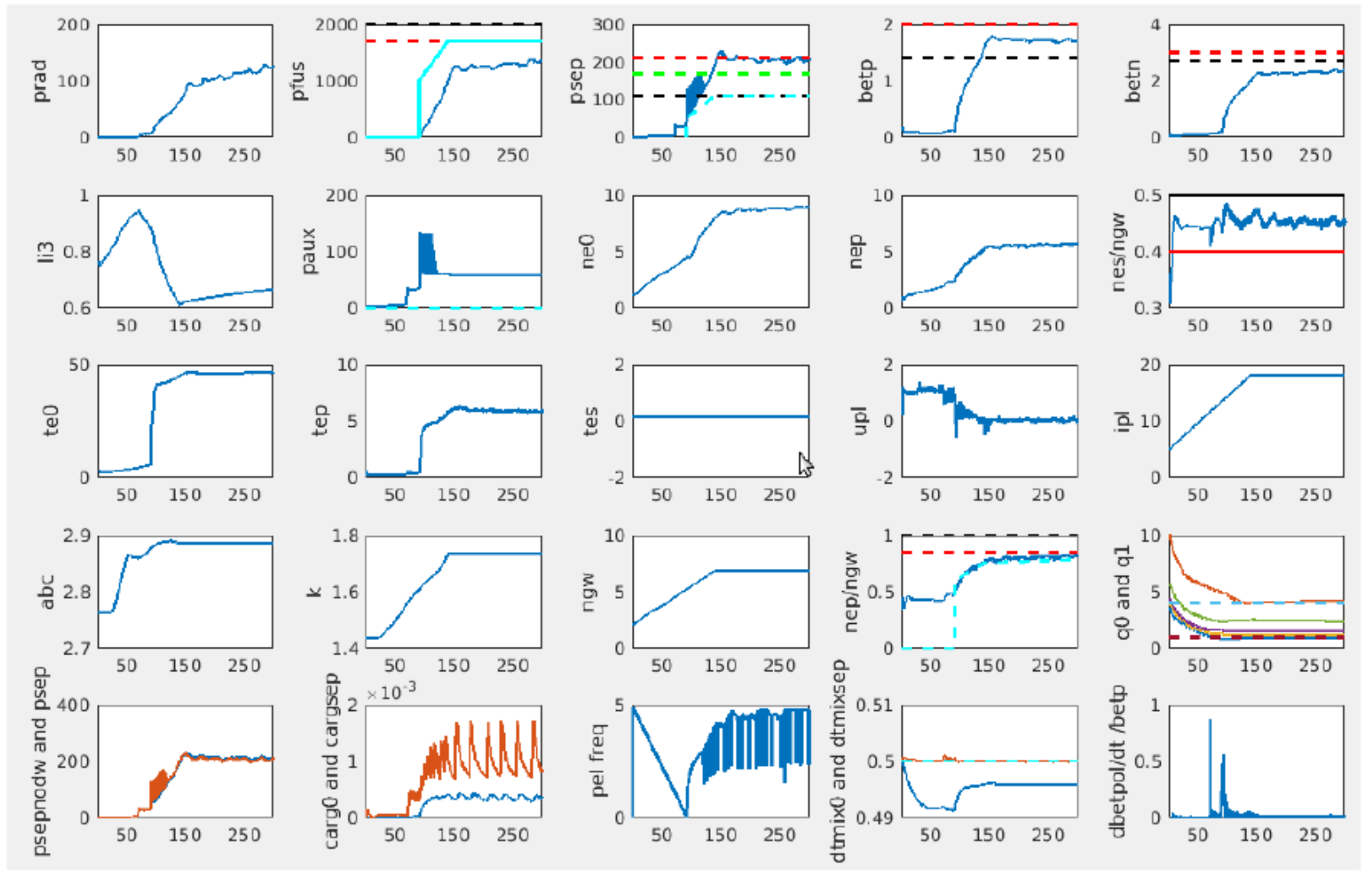
- Separatrix power (bottom subplot) oscillates strongly due to noise on Prad → overcomes lower threshold set as 1.1 P_{LH}

- NBI (but could be ECH as well) power intervenes from control command to avoid plasma dropping into L-mode
However, this command is due to the spurious P_{rad} noise

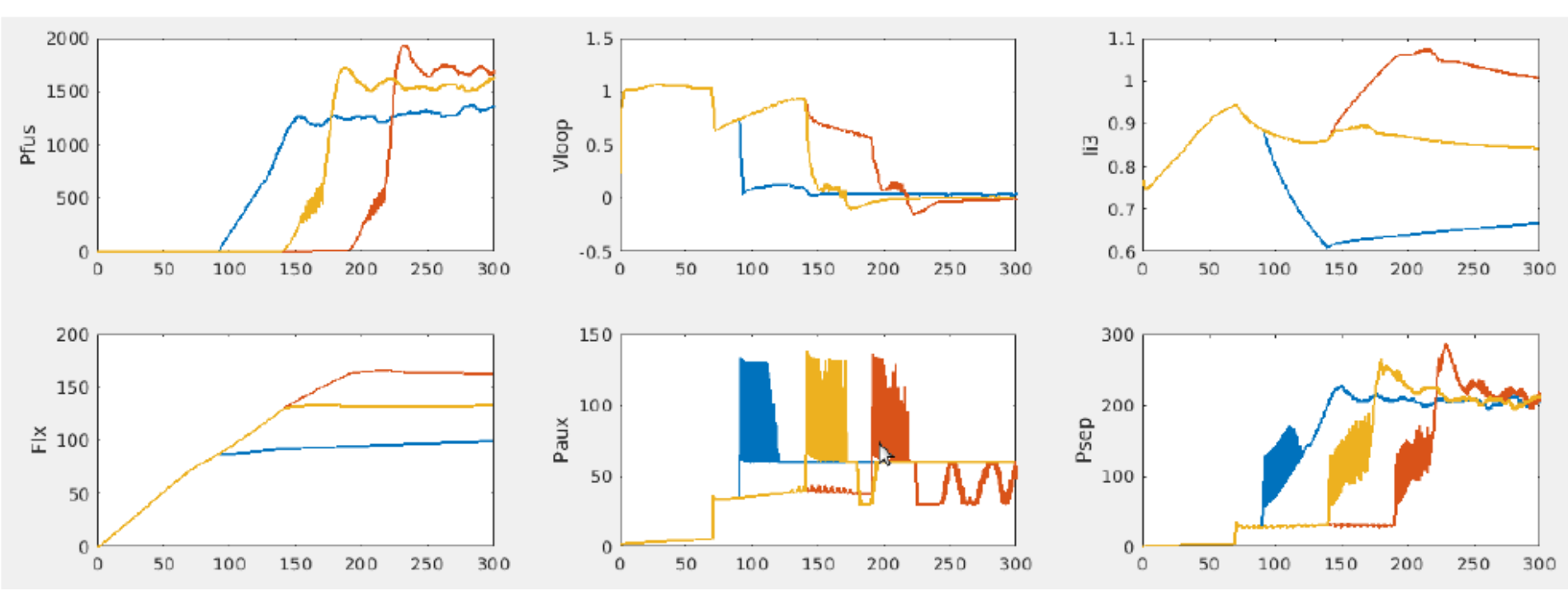
- Avoidance of this: calibrate P_{rad} sensors to have tolerable noise and error

- Latencies and integration times not so much of a problem for the kinetic control modes presented here → system has much longer timescales

- Ramp-up / entry into burn phase also simulated



- Sensitivity of burn phase performance on entry time → found first for ITER in [7]
- Later entry → lower core q profile → better confinement → higher fusion throughput
  - → but also more consumed flux and different li
  - → needs integrated modeling to find optimal trajectories
- Smooth entry into H-mode feasible by tuning density/power trajectories



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