Plasma confinement in tokamaks

J. Stockel

Institute of Plasma Physics, Academy of
Association EURATOM / IPP.CR, Czech Republic

- **WHY tokamaks?**
  - Motivation of tokamak research
- **WHAT is tokamak?**
  - Pinciple
  - Basic hardware
  - Basic plasma diagnostics
- **HOW tokamak operates?**
  - Basic plasma performance
  - Advanced plasma performance
- **WHEN we expect the final results?**
  - ITER, DEMO
  - Way to fusion power plant

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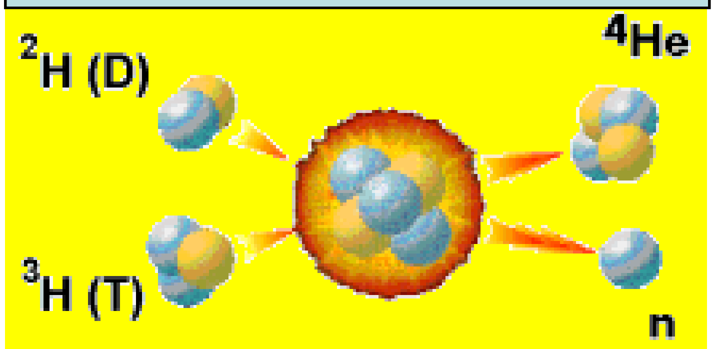
Why we study tokamaks for 50 years?

- Final goal of the tokamak research is to reach fusion of deuterium and tritium nuclei for production of electricity

Nuclear fusion



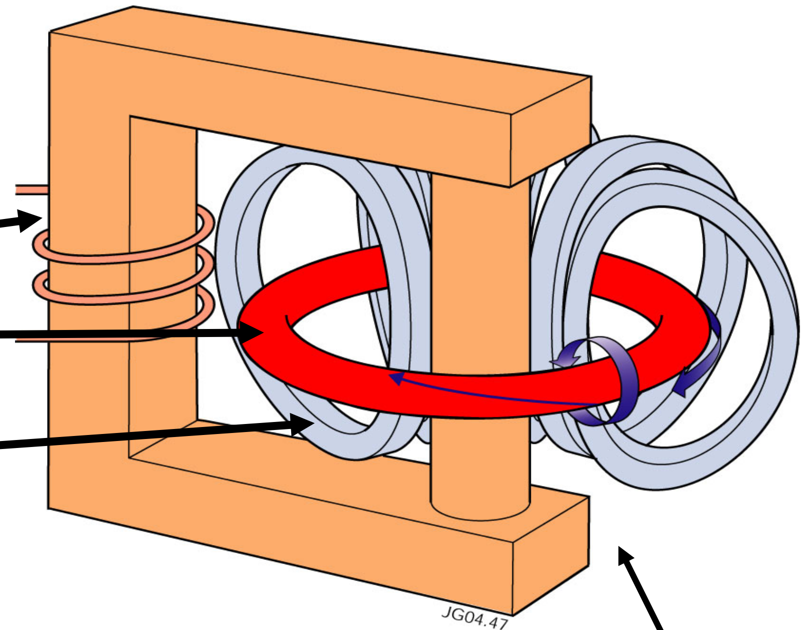
- Fusion plasmas must fulfill so called Lawson criterion:

> Plasma temperature > 200 000 000 K (~ 20 keV)
>
> Plasma density > 10^{20} m^{-3} (10^6 times less than in atmosphere)
>
> Confinement time > 1 s

<u>DEFINITION</u>; TOKAMAK is an experimental facility, in which the toroidal plasma ring can be confined for a sufficiently long time and heated up to high enough temperatures

Tokamak - basic principle

Tokamak, abbreviation from Russian: **TO**roidalnaya **KA**mera, s **MA**gnitnami **K**atushkami (means “toroidal vessel” with “magnetic coils”)



Tokamak is composed of three basic components

- Large transformer
- Plasma ring as secondary winding
- Coils for confinement of plasma ring by magnetic field (toroidal solenoid)

Electric current I generated in the plasma ring by the transformer

- delivers the ohmic power $P_{ohmic} = I^2 R_{plasma}$ to plasma (heating)
- generates the poloidal magnetic field in the plasma ring $B_{poloidal} \sim I/2\pi a$

REMEMBER! Because of the transformer, tokamak is **pulse** device

(in contrast to so called stellarators – see the next lecture)

Necessary condition for formation of stable ring of plasma

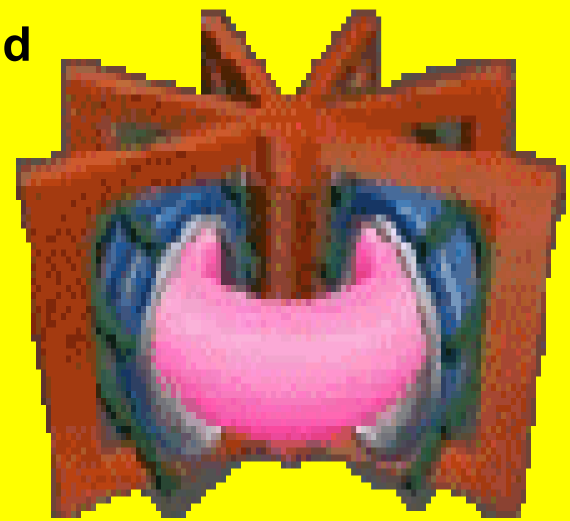
Magnetic field lines must have helical shape in any toroidal plasma!!

Magnetic field lines of toroidal solenoid

+

Poloidal magnetic field of plasma current

=



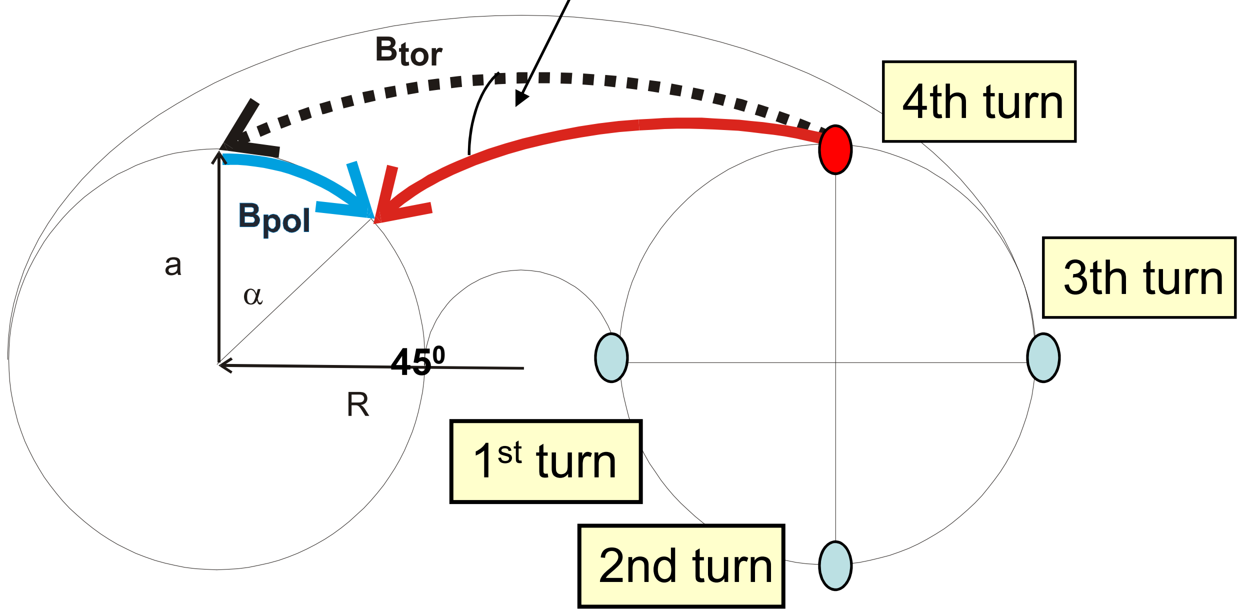
Result:

Helical magnetic field lines!!

Helicity of a magnetic field line

Theory of any toroidal facility requires that the magnetic field line must have a small pitch at the edge of plasma ring!!!

Pitch angle ~ Bpol/Btor ~ I_{plasma}/Btor



Stability of plasma in tokamaks

For quantitative analysis we define so called safety factor q:
– how many toroidal rotations are necessary for a single rotation of a magnetic field line in the poloidal direction (2π)

At the plasma edge
(*Cylindrical approximation*)

$$q(a) = \frac{aB_{TOR}}{RB_{POL}} = \frac{2\pi a^2 B_{TOR}}{\mu_0 R I_{plasma}}$$

$$B_{POL}(a) = \frac{\mu_0 I_{plasma}}{2\pi a}$$

Stability of plasma requires the safety factor at the edge
$q(a)^{MIN} > 2 - 3$

Maximum plasma current in a tokamak

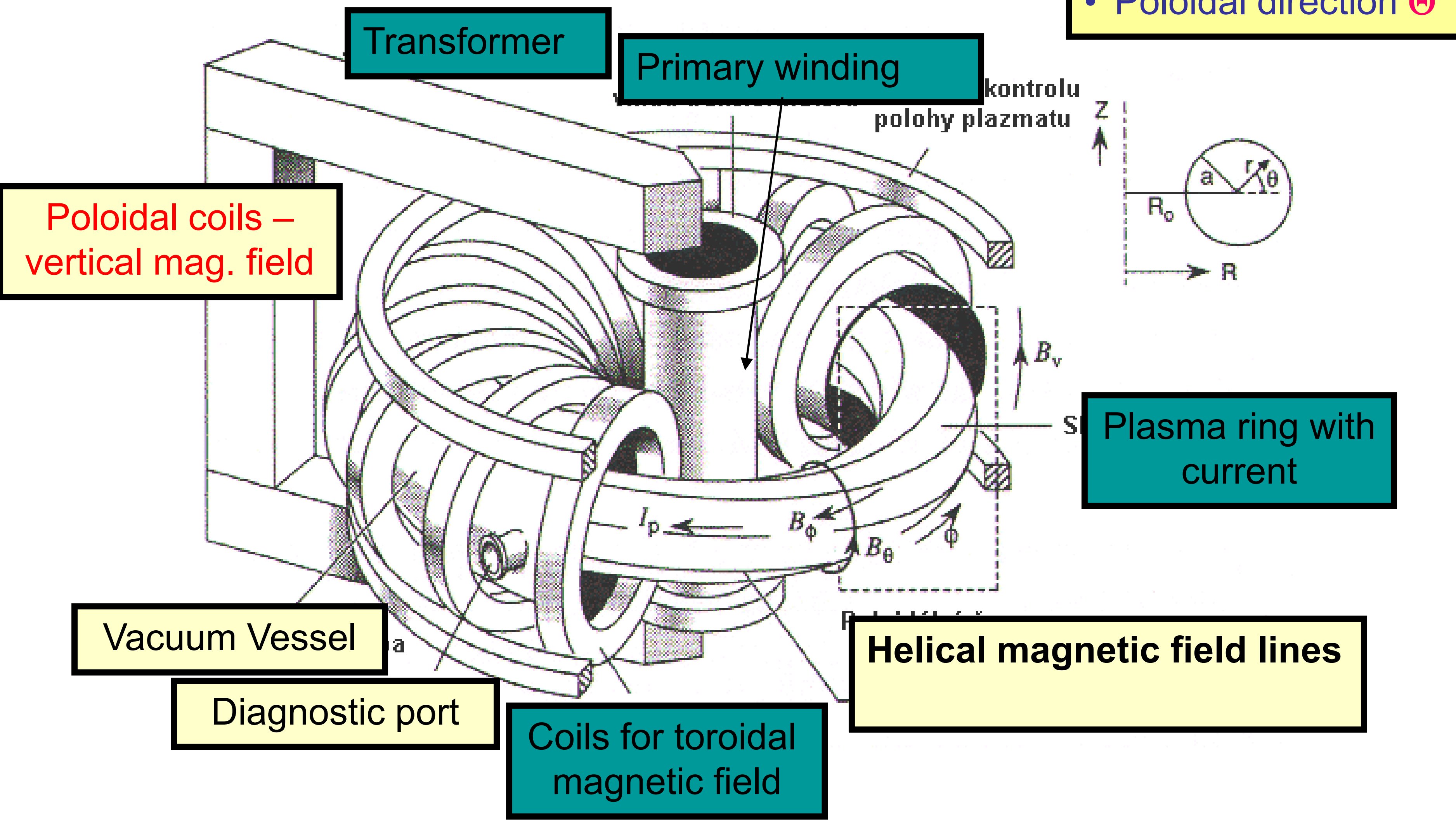
$$\mathrm{Imax} = \frac{\mathrm{aB_{TOR}}}{\mathrm{RB_{POL}}} = \frac{2\pi a^2 \mathrm{B_{TOR}}}{\mu_0 \mathrm{Rq_{min}}}$$

Dimensions of a tokamak and the value of the toroidal magnetic field determine the maximum plasma current which can be driven in a tokamak:

CASTOR	**a = 0.085 m, R = 0.40 m, Bt = 1.0 T**	**=> Ip^{Max} ~ 0.033 MA**
JET	**a = 1.250 m, R = 2.96 m, Bt = 3.45 T**	**=> Ip^{Max} ~ 3.2 MA**

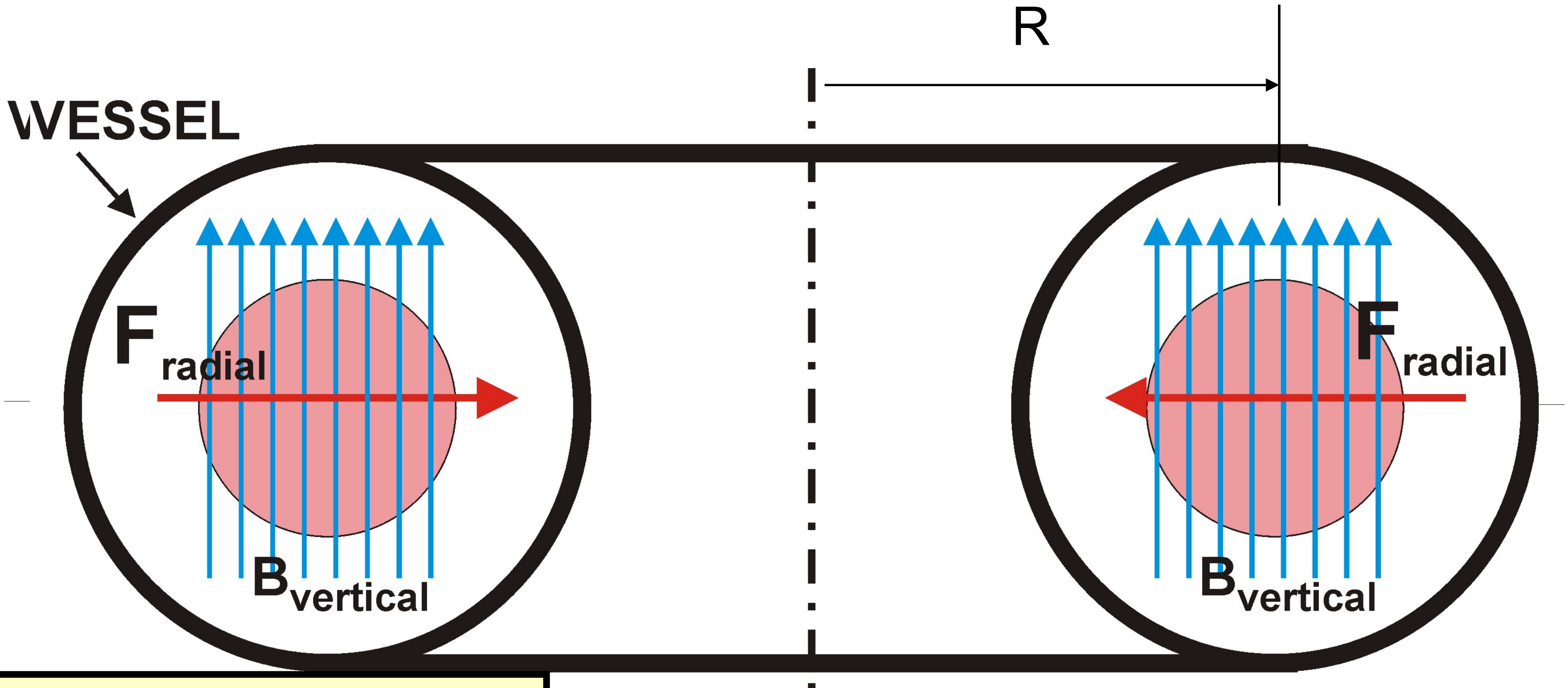
Tokamak - more details

- Major radius **R**
- Minor radius **a**
- Toroidal direction ϕ
- Poloidal direction Θ



Equilibrium position of plasma column in the vessel

Plasma ring with the electric current J_p expands in the direction of major radius



Two reasons:
Ampere force
(expansion of the current loop)

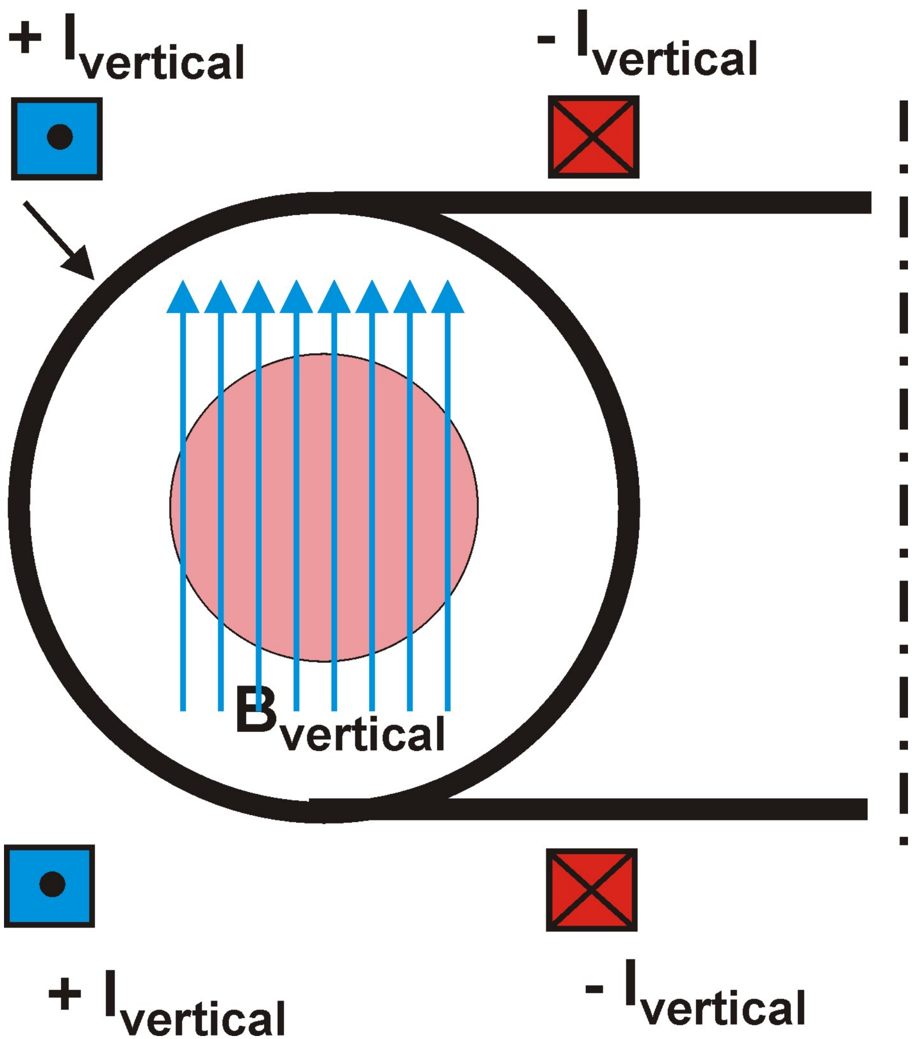
Kinetic pressure of plasma

Increase of R is compensated by the radial force

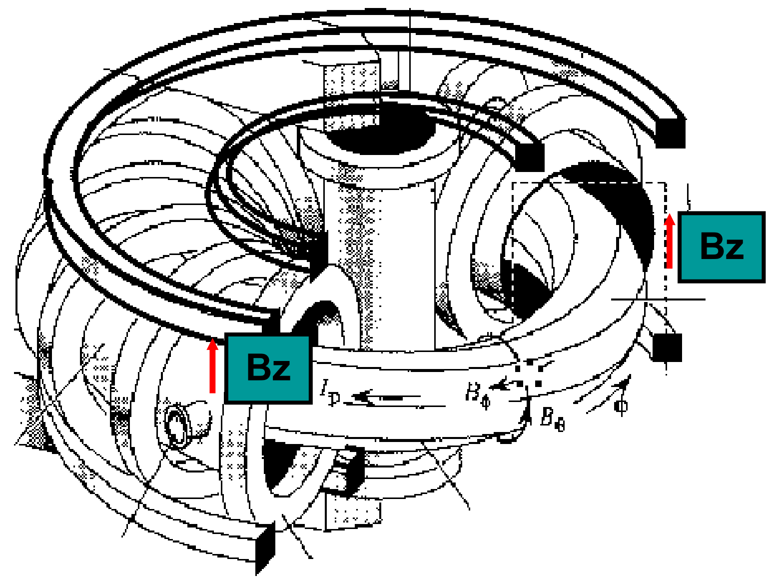
$$F_{radial} = J_{plasma} \times B_{vertical}$$

where B_z is external vertical magnetic field

How to generate vertical magnetic field?



Vertical magnetic field is generated by four loops with defined direction of current
(quadrupole configuration)



Similar approach is adopted for control of the vertical displacement of plasma column

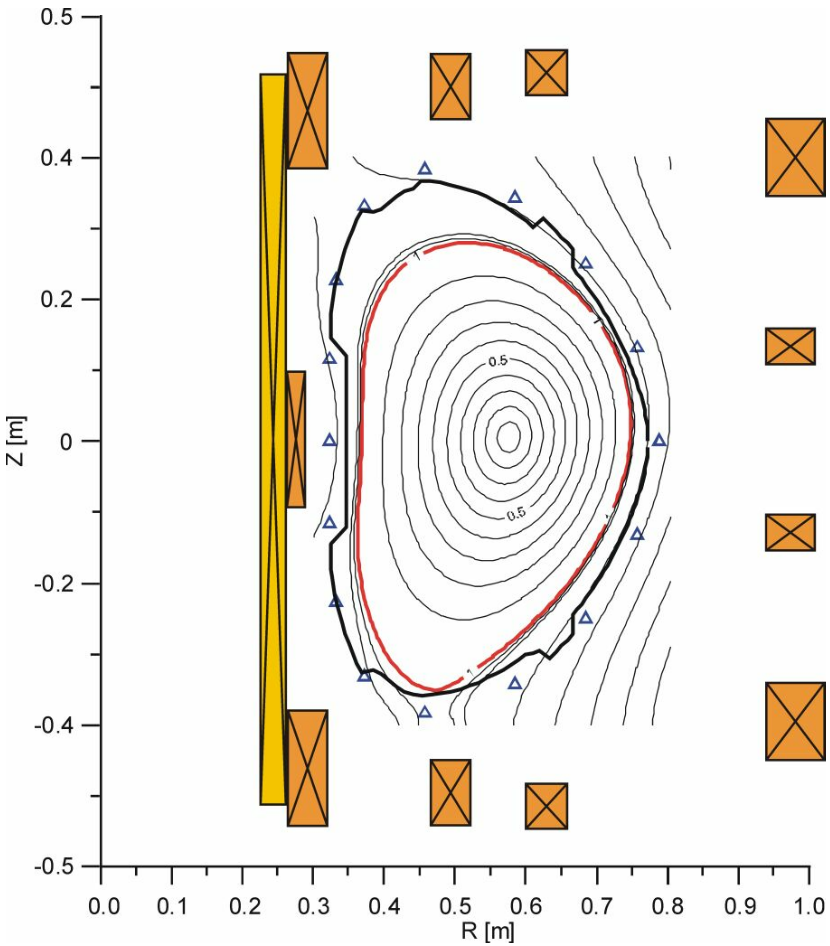
Tokamak winding - more realistic

Modern tokamaks are equipped by a more sophisticated system of windings:

Central solenoid (primary winding) to generate plasma current

Windings for keeping the equilibrium position of plasma column within the tokamak vessel

Windings for shaping of the cross section of the plasma column



Tokamaks in the world (~ 175 facilities built)

EURATOM	JET – the largest facility in the word
Germany	ASDEX U, TEXTOR 94
France	TORE – SUPRA
GB	MAST (spherical)
Italy	FT-U
Switzerland	TCV
Czech	COMPASS, GOLEM(CASTOR)
Portugal	ISTTOK

USA	D IIID, ALCATOR C-mod, HBT-EP
Japan	JT- 60
Russia	T-10, TUMAN 3, FT-2, GLOBUS (spherical), T11-M
Canada	STORM-1M
China	EAST, HT-7, J-TEXT, HL-2A
South Korea	KSTAR
India	Aditia, SINP, (SST-1 under construction)
Brazil	ETE (spherical), TCABR, NOVA-Unicamp
Iran	IR-T1

~ 32 tokamaks currently under operation differing in:

Major radius 0.4-4 m, Toroidal magnetic field 0.5-4 T, Current 0.01-4 MA

Pulse length 0.01 – ~300 sec and magnetic configuration (limiter, divertor)

Hardware required to have tokamak plasma?

- A tokamak
- Vacuum and gas handling system
- Power supplies
- Basic diagnostics
- Control and Data acquisition system

We will exploite a small tokamak CASTOR as an example for demonstration of basic features

CASTOR tokamak

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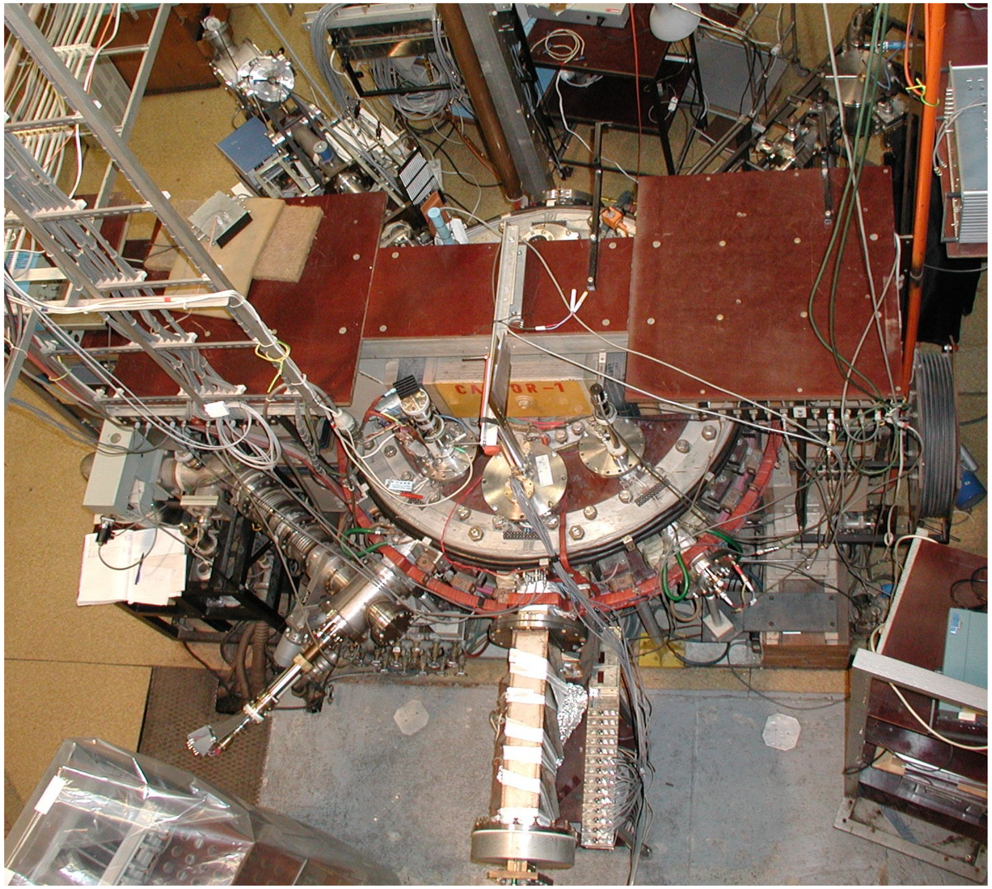
Major radius	**40 cm**
Minor radius	**8,5 cm**
Toroidal magnetic field	**< 1,5 T**
Plasma current	**5 - 20 kA**
Pulse length	**< 50 ms**

Features:

Small tokamak

Routine operation (100 shots/day)

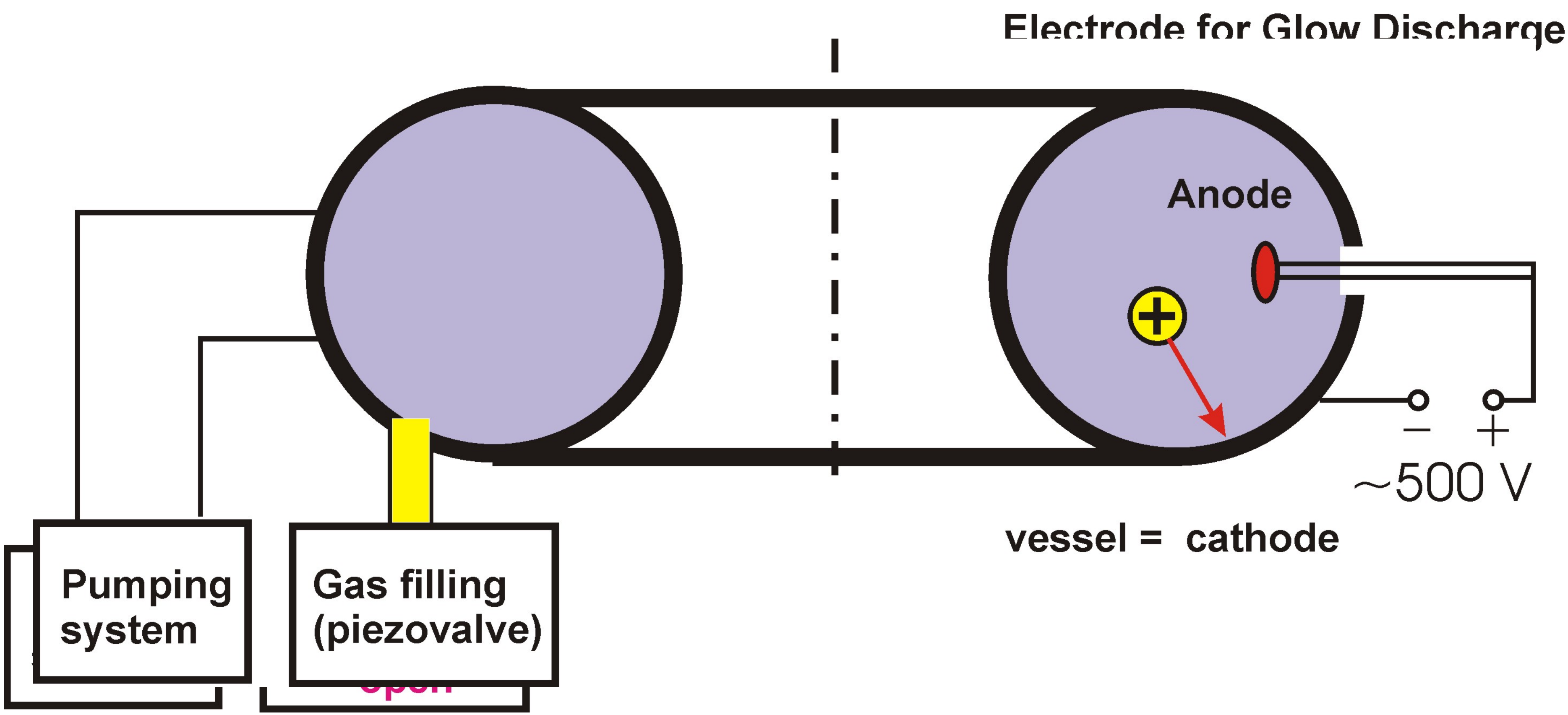
Flexible (a good plasma already 1 day after opening)



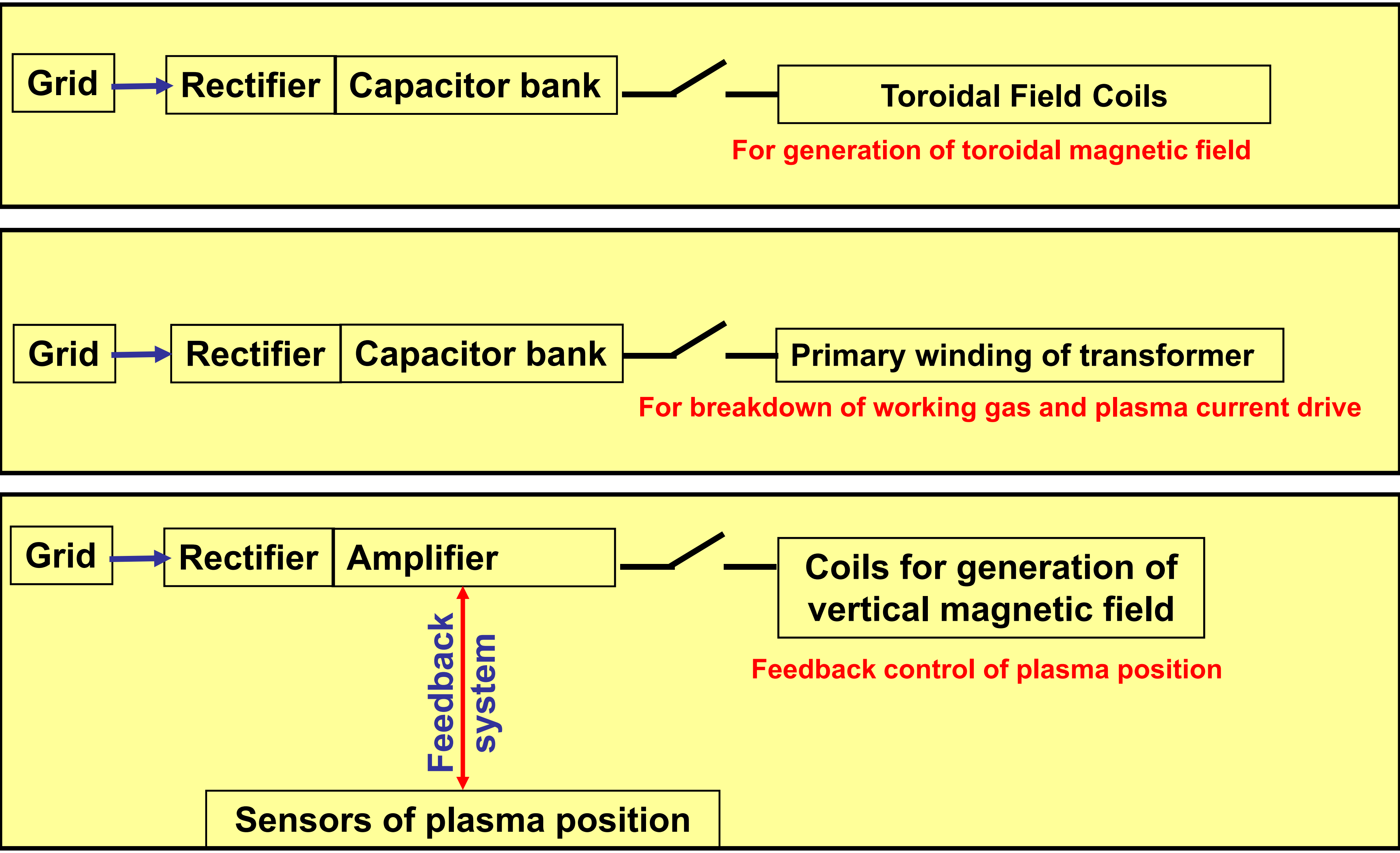
Preparation of the tokamak for operation

- **Pumping** of the vessel down to 10^{-8} mBar (10^{-6} Pa) by turbomolecular or cryo pumps
- **Baking** of the vessel to 150 - 250° C
- **Glow discharge cleaning**

Vessel is filled by Hydrogen gas (~ 1 Pa), electrode (anode) is inserted into the vessel and biased (to ~ +500 V) and the glow discharge is burning.
Inner wall (cathode) is bombarded by H ions and molecules are desorbed from the inner surface of the vacuum vessel



Basic power supplies of a simple tokamak
(schematically)



Power supplies for modern tokamaks

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- Modern tokamaks require a huge electrical power, which is usually not available from the public electrical grid (up to several hundreds MW in pulses 1-1000 sec)
- Therefore, the required energy must be accumulated before the discharge (for small facilities in condenser banks, for large tokamaks by means of flying-wheel generators)
- All power supplies must be pre-programmed to provide defined current pulses to individual windings
- Some power supplies must be controlled by a feedback systems (perpendicular fields for control of plasma position)
- Control system of PS is rather complex and expensive

Requirements for PS are significantly reduced, if superconducting windings are used in a tokamak (working at the temperature of liquid Helium).

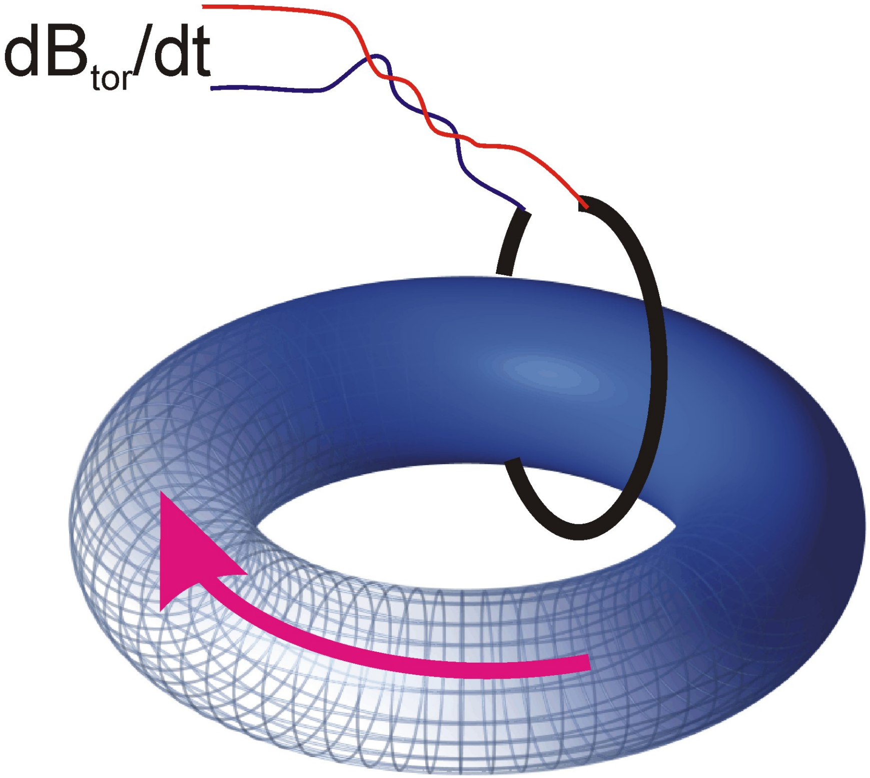
Necessary for long pulse operation of contemporary facilities like TORE Supra, EAST, KSTAR and definitely for next step tokamaks (ITER) and finally for the reactor

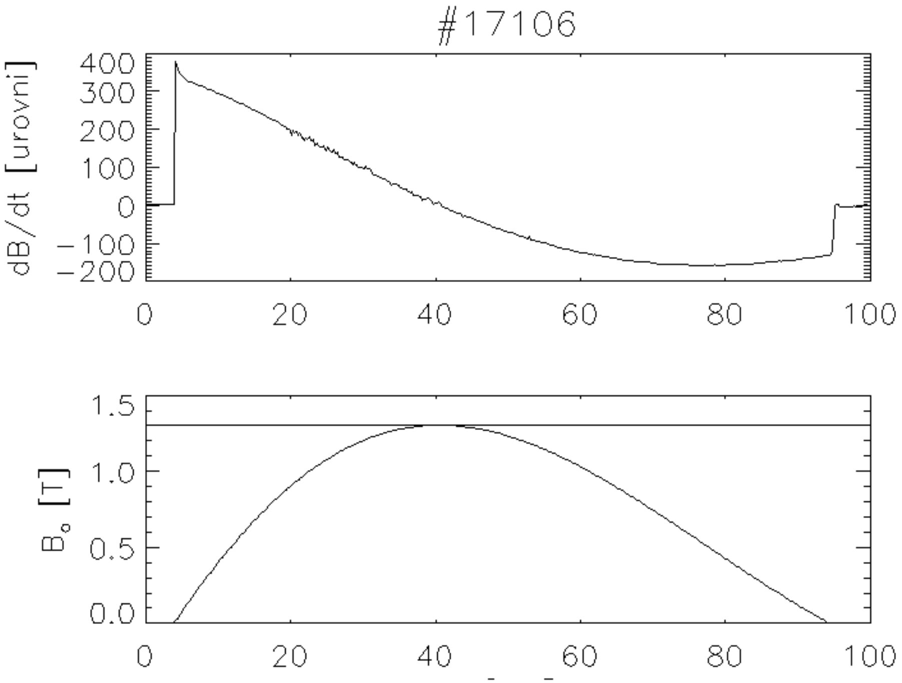
Basic diagnostics to operate a tokamak

We have to measure (at least):

- **Toroidal magnetic field**
- **Plasma current**
- **Loop voltage**
- **Position of plasma column in the vessel**
- **Plasma density**
- **Plasma radiation in visible range (H_alpha, impurity lines, soft x-ray, …..)**

Toroidal magnetic field



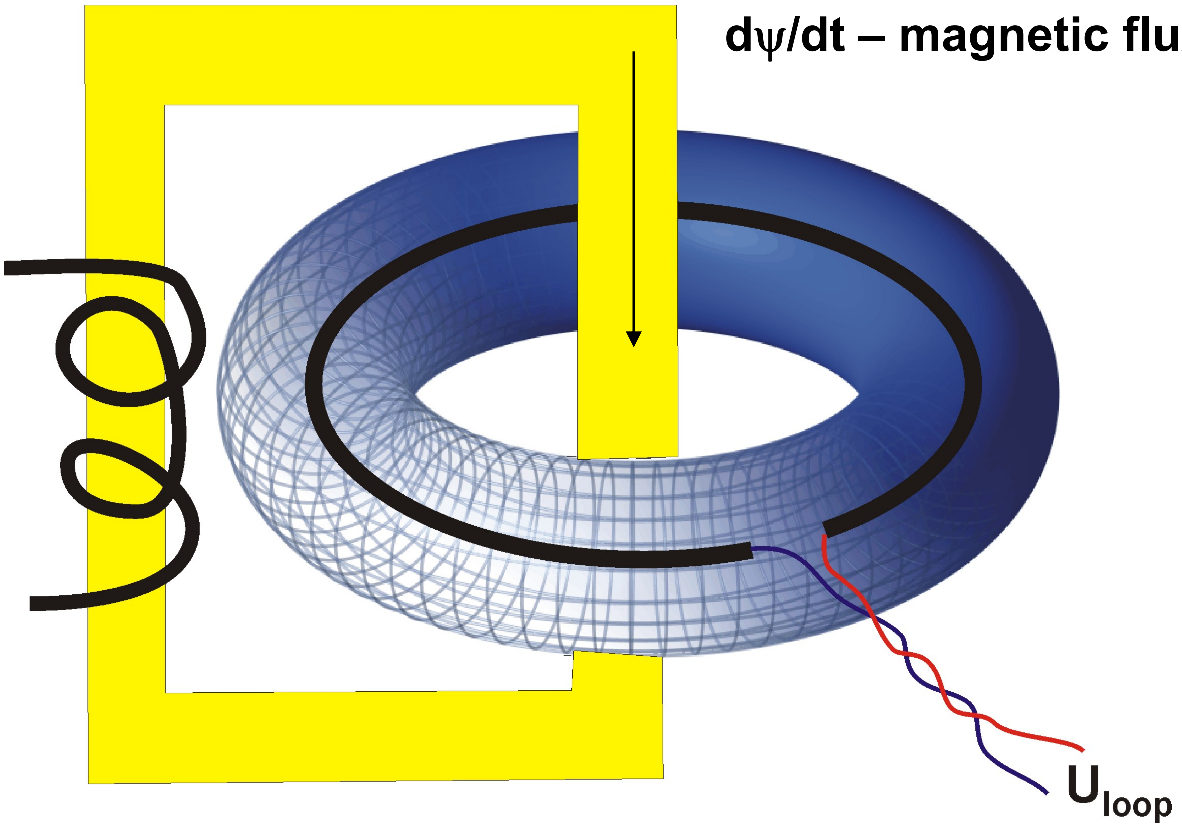


Signal of the loop has to be integrated

- **analogue integrator**
- **numerical integration**

Loop voltage

$d\psi/dt$ – magnetic flux

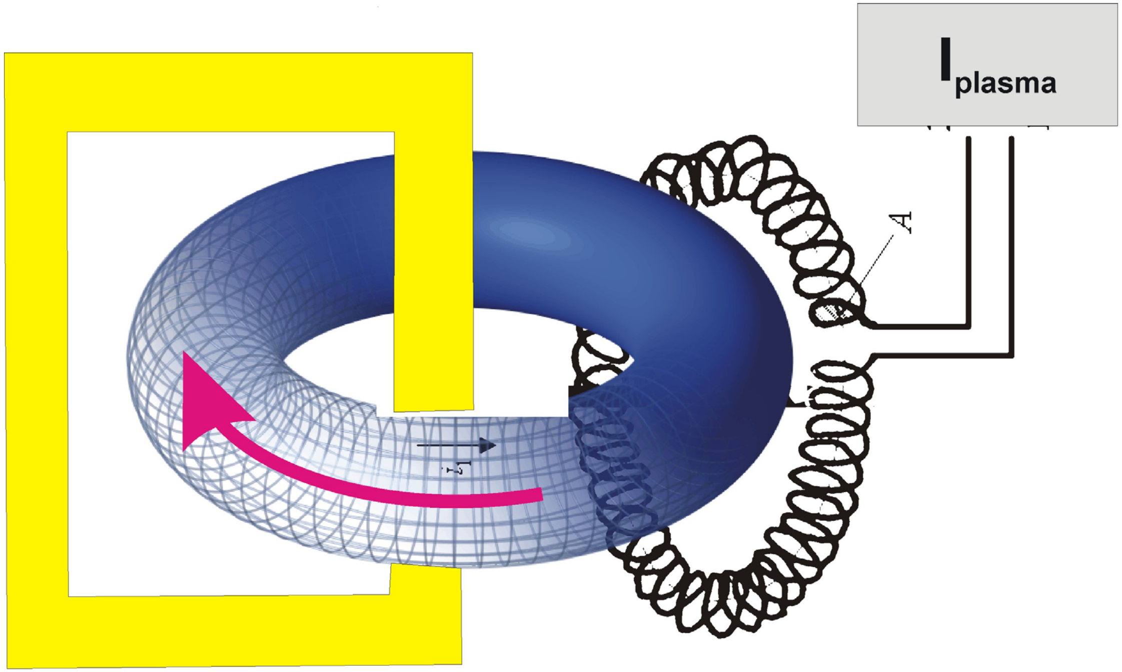


$Uloop = - d\psi/dt$

The loop voltage is measured by a loop located in the proximity of the plasma column

The toroidal electric field $E_{tor} = U_{loop}/2\pi R$ accelerates charged particles in the toroidal direction (and drives plasma current)

Plasma current

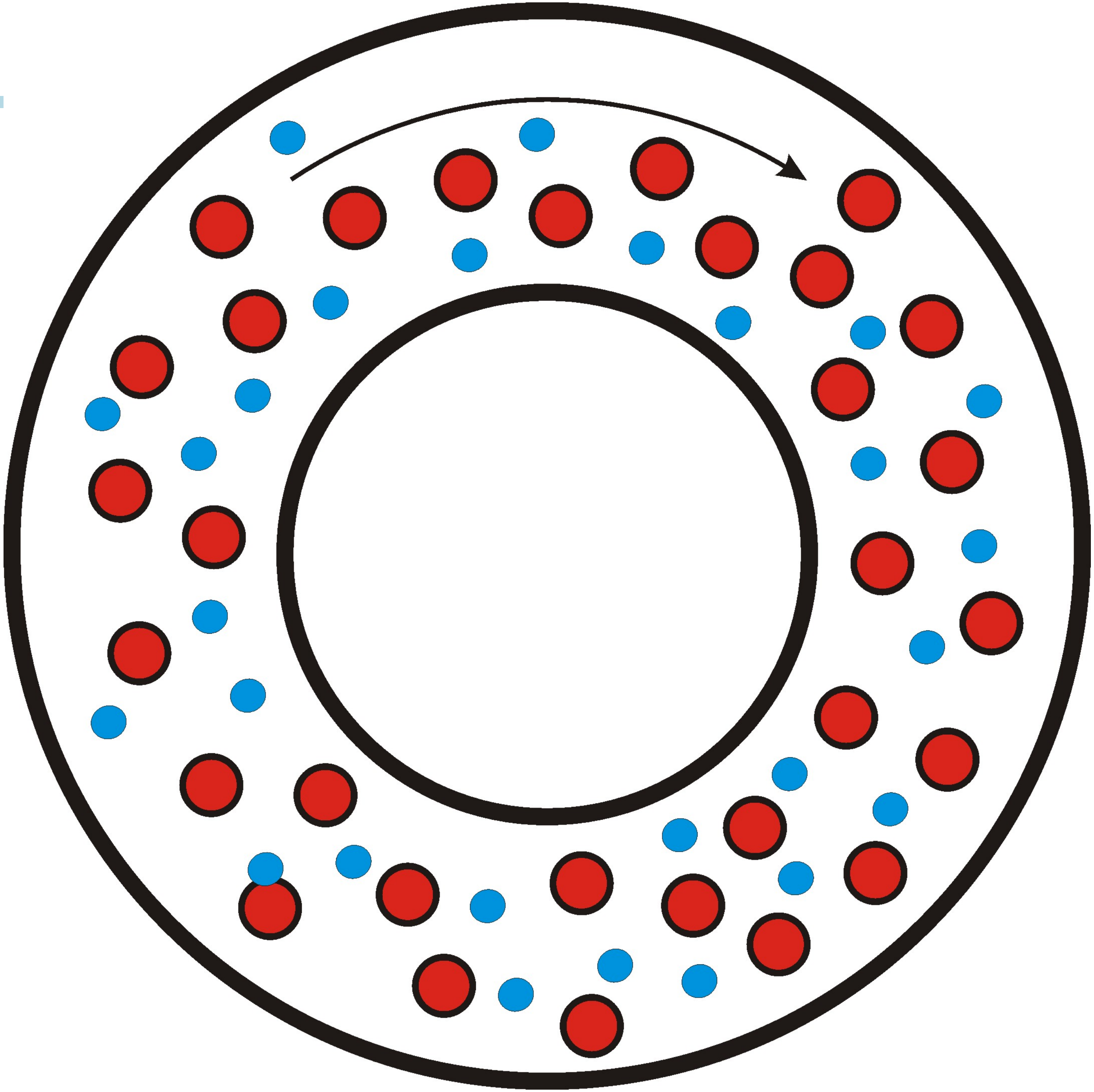


Plasma current is measured by means of the Rogowski coil
(the solenoid of a toroidal shape surrounding the plasma column)

Start-up phase of a tokamak discharge

- Tokamak vessel is filled before the discharge by the working gas – hydrogen or deuterium. Typical pressure is ~ 10-50 mPa
- Some free electrons are generated in the vessel by means of an ionization source

  – UV lamp or a small electron gun (pre-ionization)
- A trigger pulse is applied to the control system of power supplies
- A trigger pulse starts operation of the data acquisition system
- The toroidal magnetic field is generated inside the vessel
- The time-dependent current in the primary winding of the transformer generates the toroidal electric field in the vessel
- Free electron are accelerated in toroidal direction and ionize molecules of the working gas
- Density of charged particles increases exponentially in time
- Plasma current increases
- Fully ionized plasma fills the vessel (in 0.1-10 ms – depending on the size of tokamak)



Start-up phase of a discharge on CASTOR

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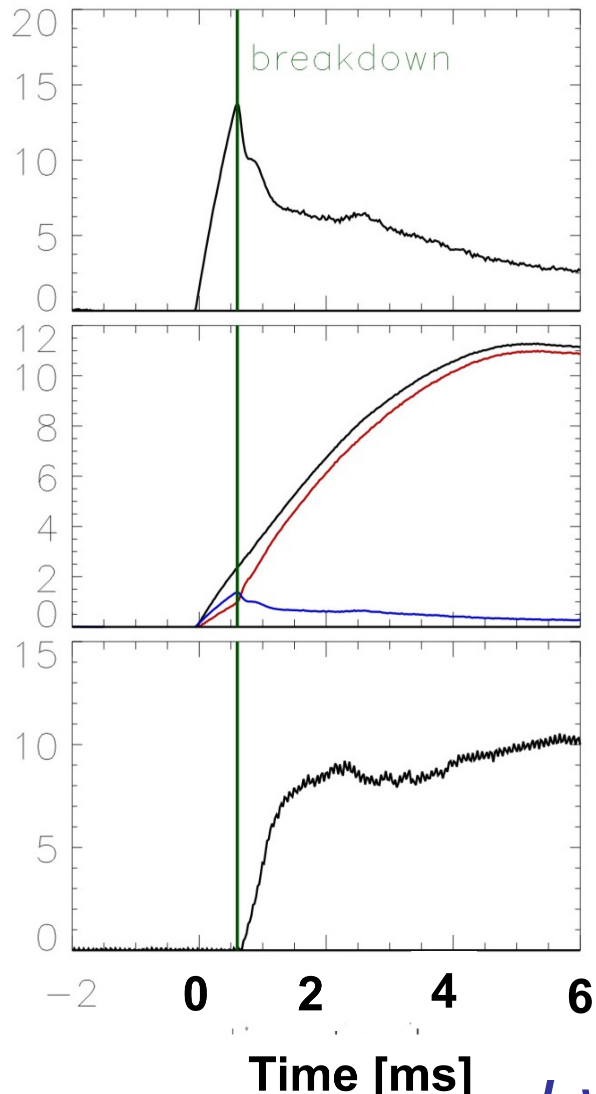
Loop voltage [V]
Uloop

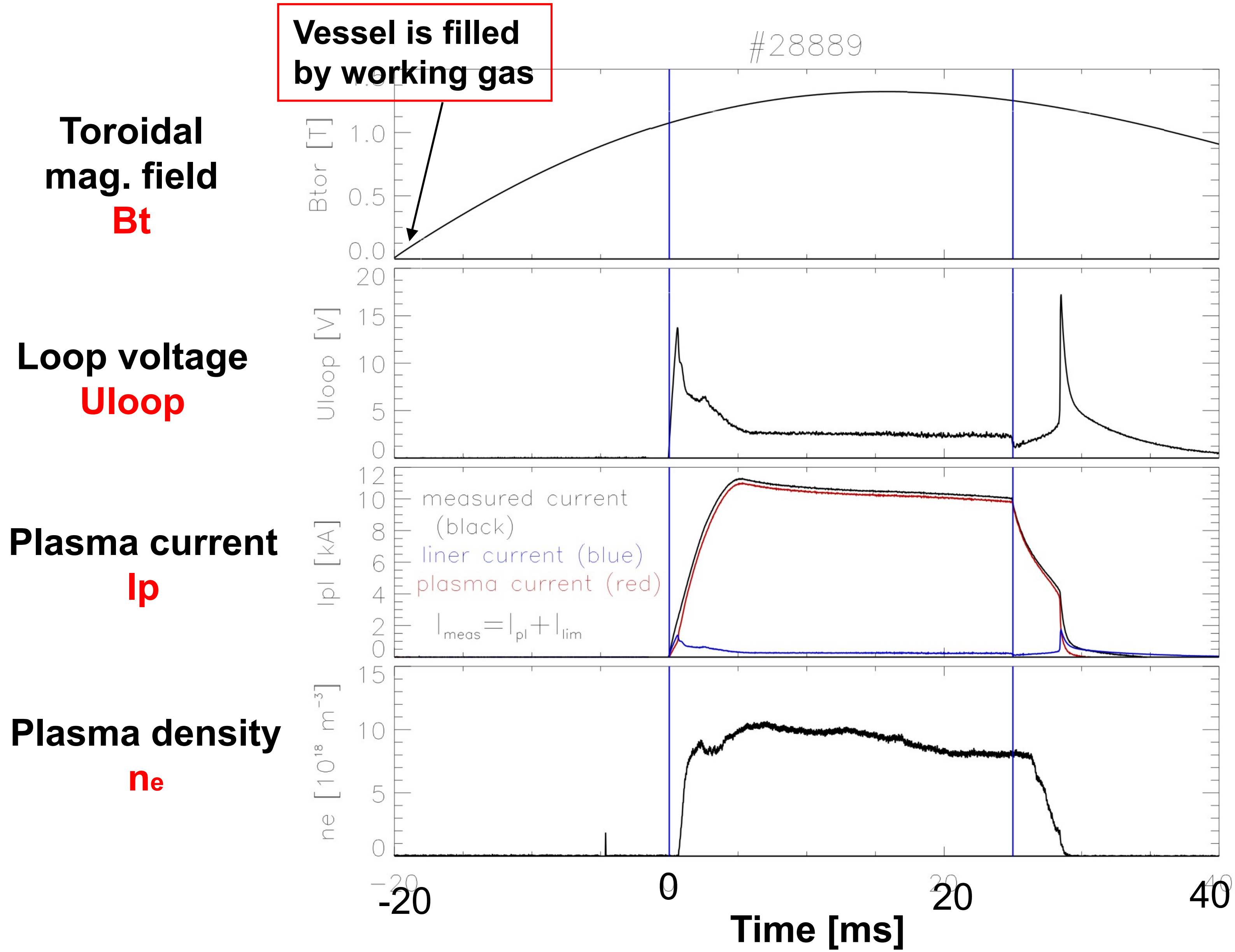
Toroidal current [kA]
I_plasma+ *I_vessel*

Plasma density
n_e [10^{19} m^{-3}]

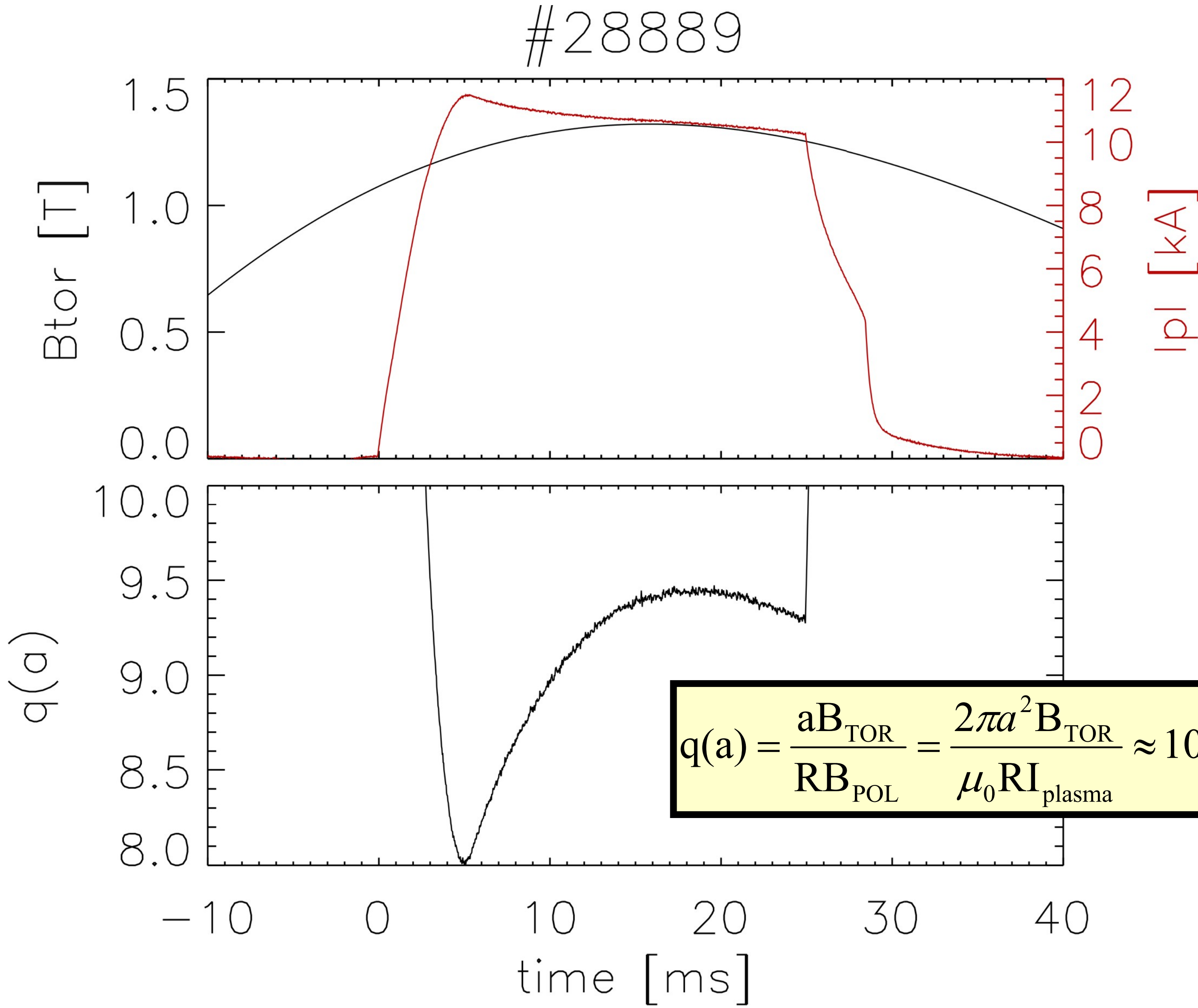


I_vessel = I_plasma – Uloop/Rvessel

Temporal evolution of a discharge on the CASTOR tokamak

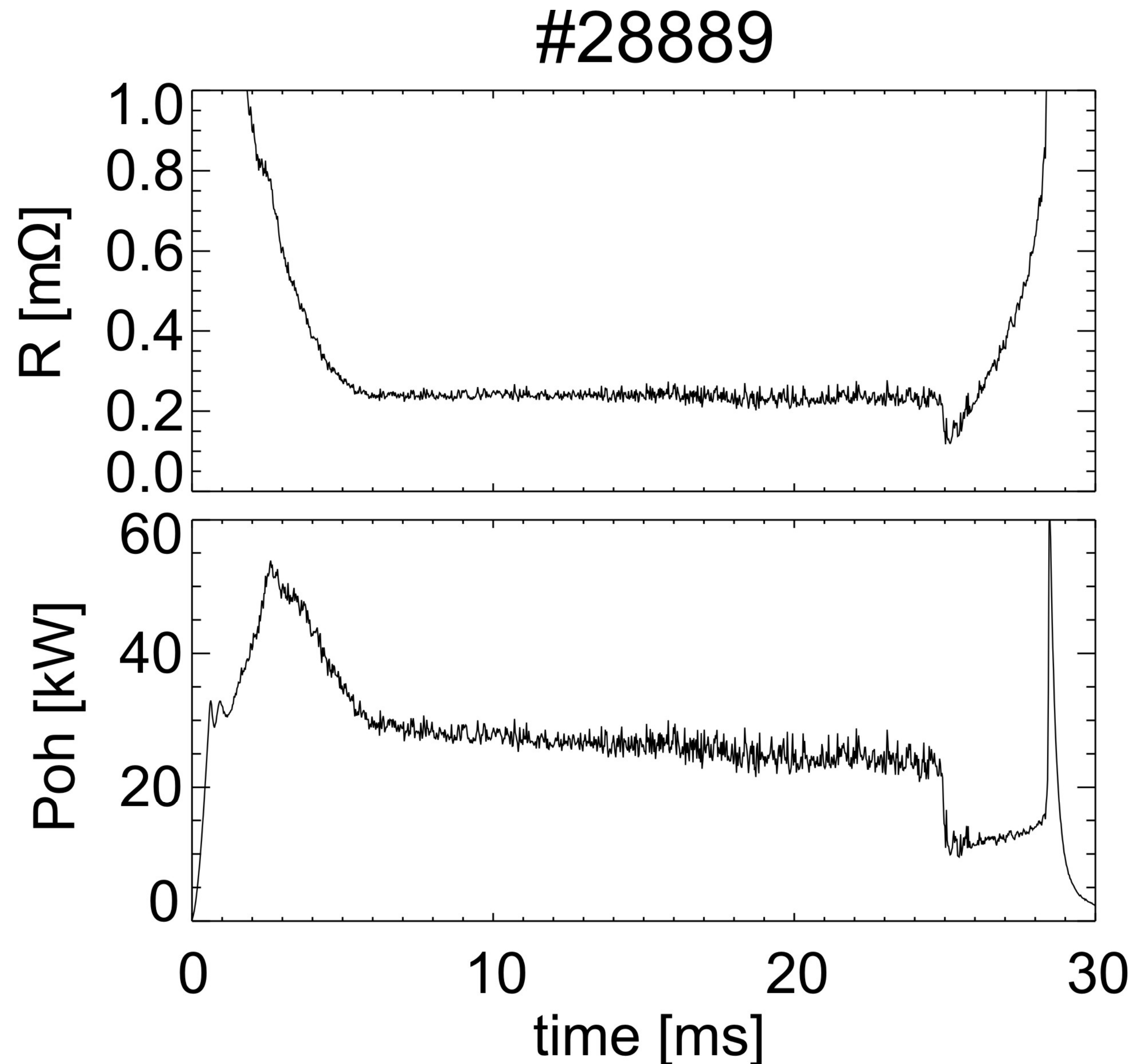


Safety factor



$$q(a) = \frac{aB_{TOR}}{RB_{POL}} = \frac{2\pi a^2 B_{TOR}}{\mu_0 R I_{plasma}} \approx 100 \frac{B_{TOR}}{I_{plasma}}$$

Resistivity of plasma column and ohmic power



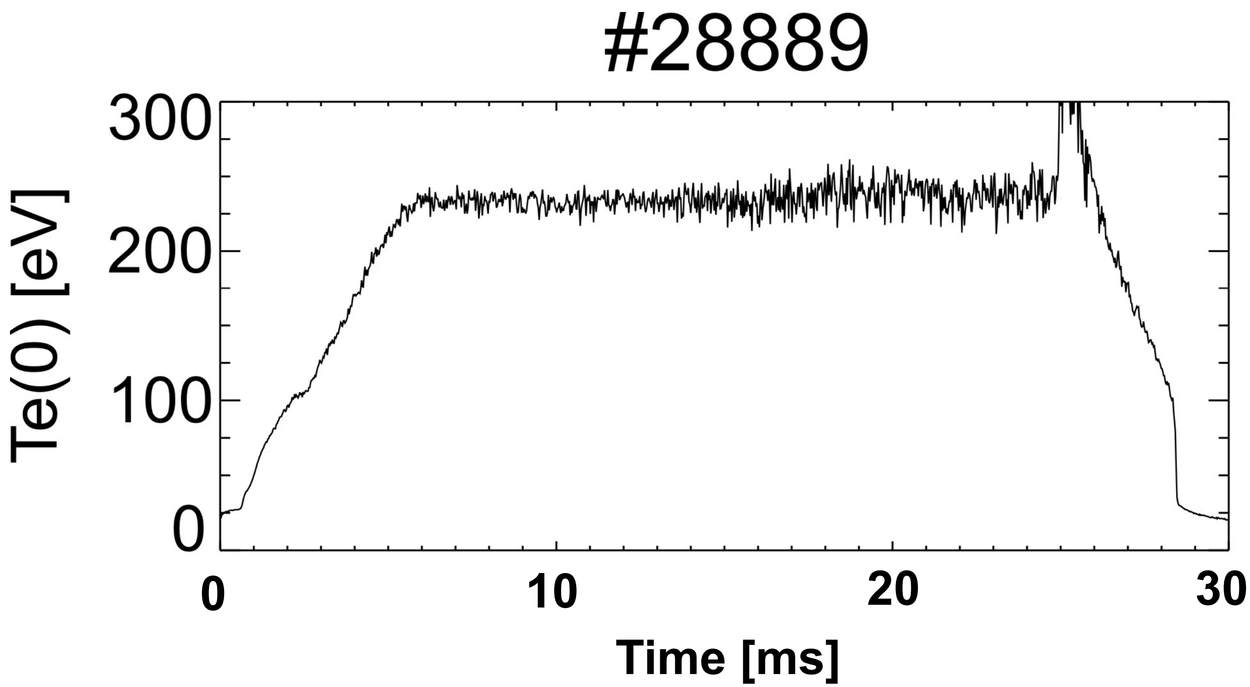
$R = U_{loop} / I_{plasma}$

$P_{oh} = U_{loop} * I_{plasma}$

Estimate of the electron temperature

Plasma resistivity depends mainly on the electron temperature. According Spitzer formula for fully ionized plasmas, higher electron temperature implies lower plasma resistivity. For a tokamak with the circular cross section

$$T_e(0) = 5{,}5\left(\frac{RI_p[\mathrm{kA}]}{a^2 U_{loop}}\right)^{2/3}$$



CASTOR
Te(0) ~ 230 eV

Global confinement of energy

$$\frac{dW}{dt} = P - \frac{W}{\tau_E}$$

Heating

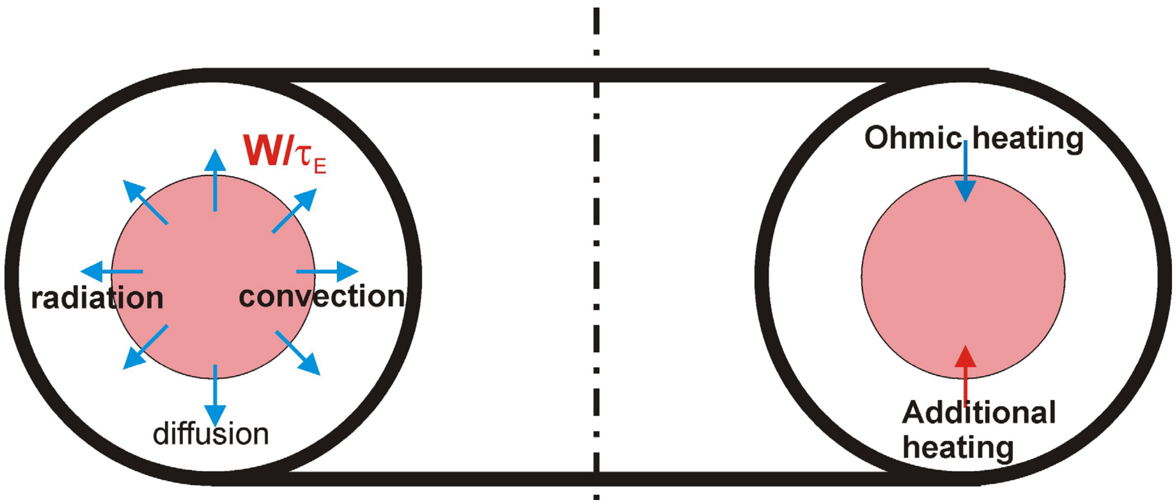
Energy losses

Balance equation for energy

τ_E **Energy confinement time !!**

characterizes global losses of energy from the plasma ring

***see Lawson criterion**



In the steady state

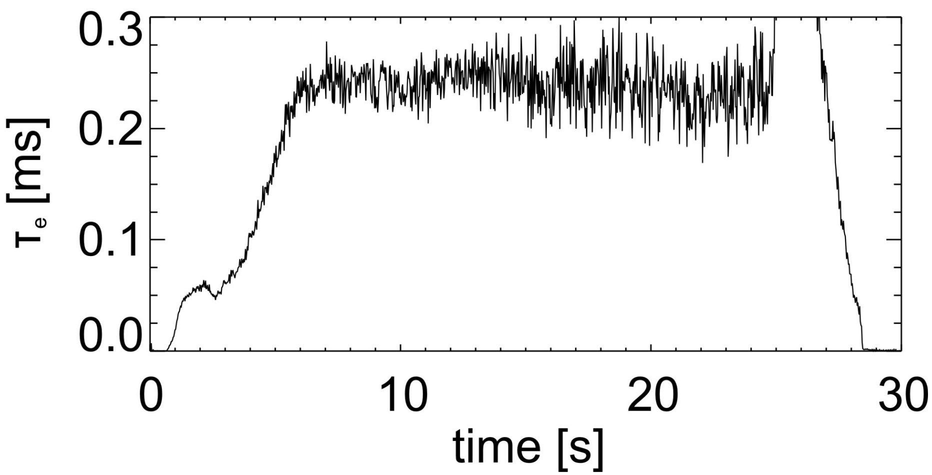
$$\tau_E = W/\ P$$

Total kinetic energy in the torus

$W = n*T * V$

is measured and divided by the heating power P

Confinement time on CASTOR



Electron confinement time

$\tau_e = (n_e * T_e * V) / P_{OH}$

We see that the plasma parameters on the small tokamak CASTOR are very far away from those required for fusion reactor!!

In particular:

- **Plasma temperature is more than 1000x less than required 20 keV**
- **Confinement time is again more than 1000x less than required 1 sec**

How to improve plasma confinement in tokamaks?

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Global energy confinement in tokamaks is mostly determined by the heat conductivity χ **- the dominant channel of energy losses**

$$\tau_E \sim a^2/\chi$$

However!!!!

- **χ were found (just from experiments) to be 100-1000x larger then that predicted by "classical" theory of transport in magnetized plasmas, because**
- **Particles and heat are transported across magnetic field lines not by collisions, but due to the plasma turbulence!**
- **Its role was recognized ~ 30 years ago, but its nature is not yet fully understood!**

=> there is not any final theoretical prediction for χ

Therefore, to improve the plasma confinement we have to:

- **Increase the size of tokamaks**

  (this is the reason why larger and larger facilities are constructed)

- **Reduce the level of plasma turbulence (within so called transport barriers)!**

  (this is why the plasma turbulence and its suppression is studied)

What to be solved to achieve a high performance plasma in a tokamak?

- Additional plasma heating
- Confinement of plasma (formation of transport barriers)
- Equilibrium and shape of plasma column
- Plasma-wall interaction
- ……….
- …$\Rightarrow$ ∞

Some of these issues will be briefly discussed now

Ohmic heating in tokamaks

Plasma – secondary winding of the transformer (a single loop) with a finite conductivity

Ohmic power dissipated in the plasma column

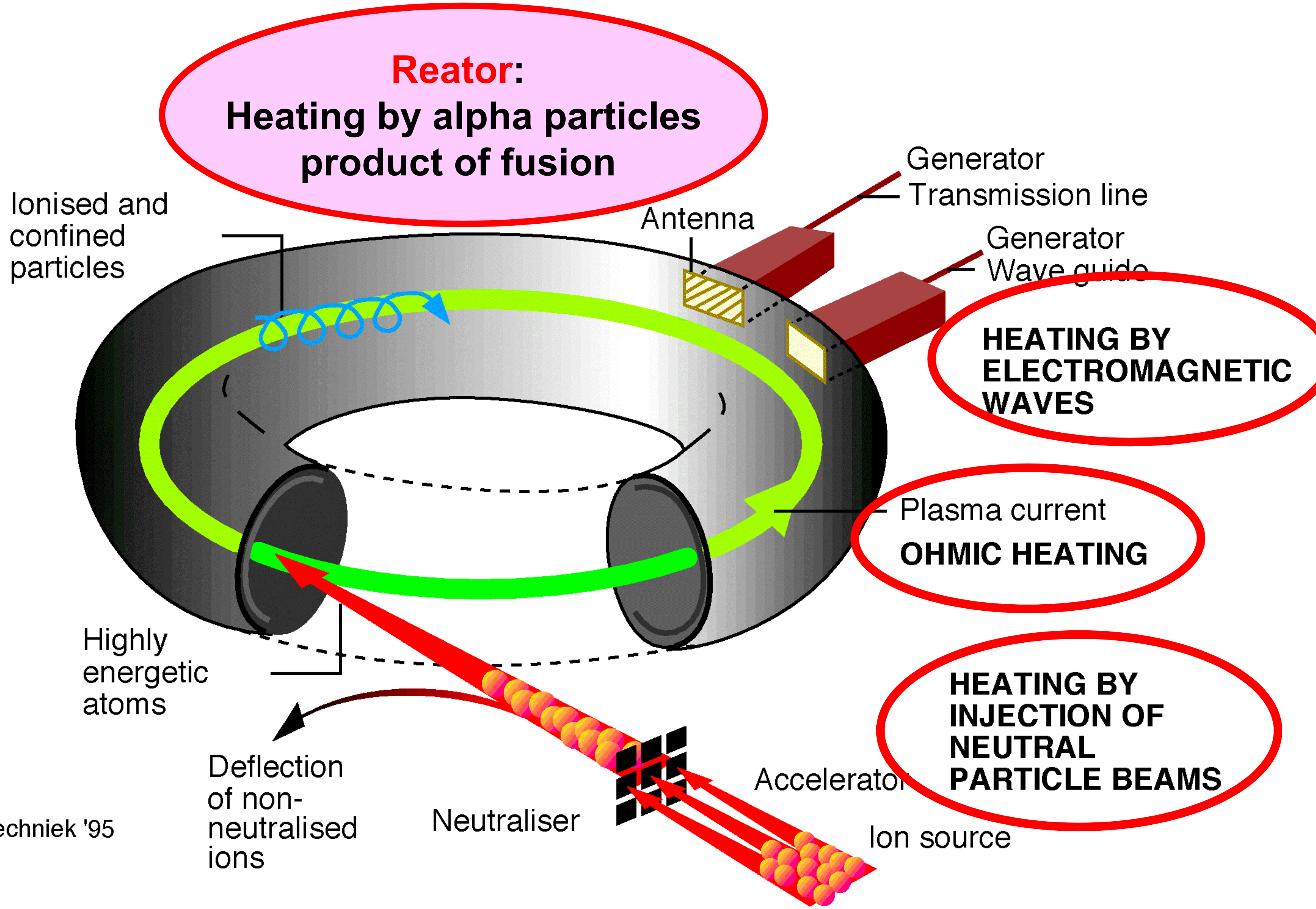
$$P_{OH} = I^2_{\ \ plazma} R_{plasma} \propto I^2_{\ \ plazma} T_e^{-3/2}$$

Ohmic power decreases with increasing temperature:

- effective only up to T ~ 1-2 keV (~10 – 20 milion degrees)
- Ohmic power is almost negligible in large tokamaks (several %)

- Additional plasma heating is required!!!

How to reach ultrahigh temperatures?



Summary of plasma heating in tokamaks

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Basic heating:

Ohmic heating (OH) – plasma of finite conductivity is heated by electric current

Alpha- particle heating - plasma is heated by products of fusion (reactor)

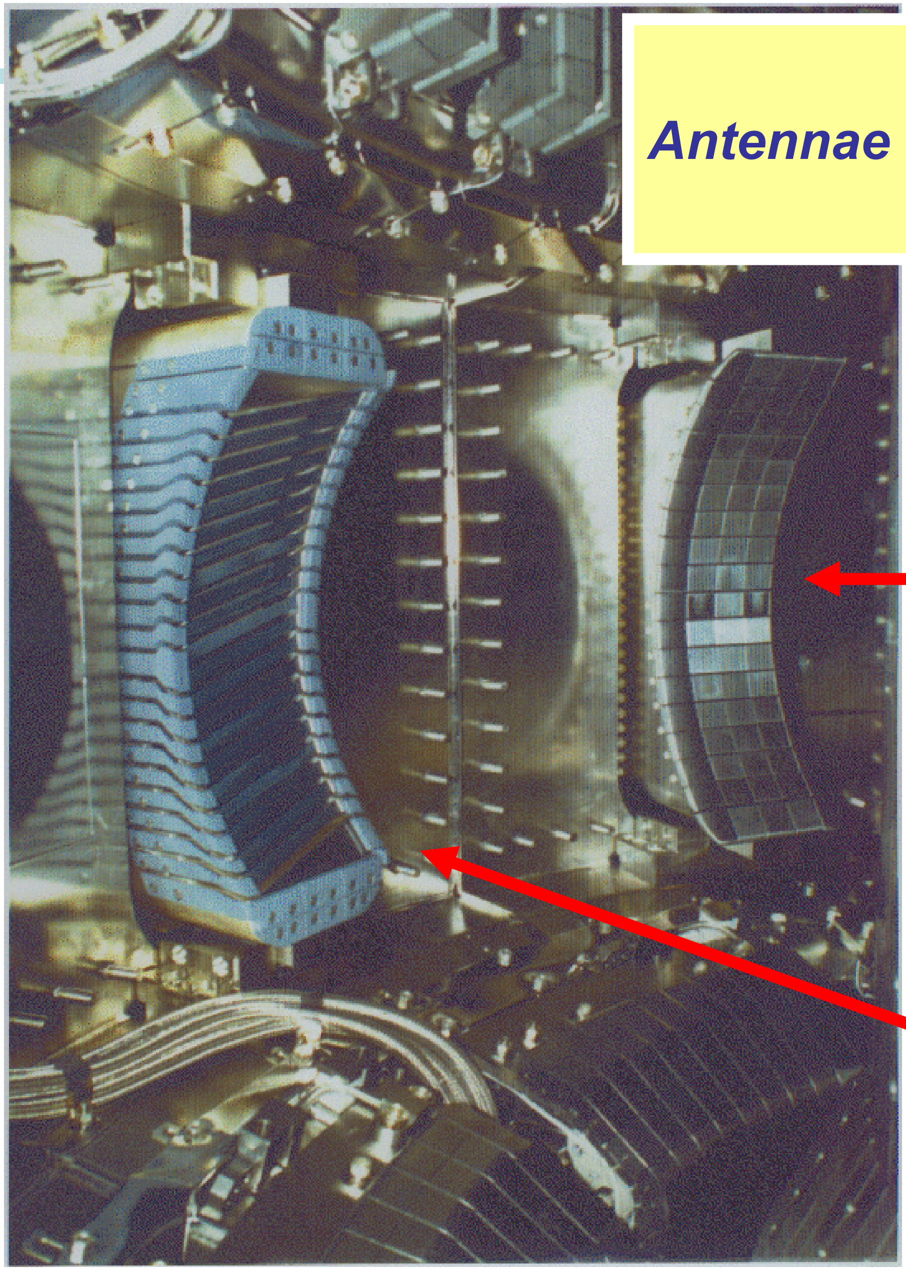
Additional heating:

Neutral Beam Injection (NBI) – (H, D, T) – beams (0,05-1 MeV) are injected into the plasma and transfer their kinetic energy to electrons or ions

Electromagnetic waves – are injected into the plasma by special antenna systems. Frequency of the wave is selected to be in the resonance with eigen-frequencies of plasma:

ECRH – **Electron Cyclotron frequency (20-200 GHz)**
ICRH - **Ion Cyclotron frequency (20- 200 MHz)**
LH - **Lower Hybrid frequency (1-10 GHz)**

Antennae for additional heating on JET



Antenna for lower hybrid waves

a system of waveguides (grill)

Antenna for heating by ion cyclotron waves

Plasma-wall interaction

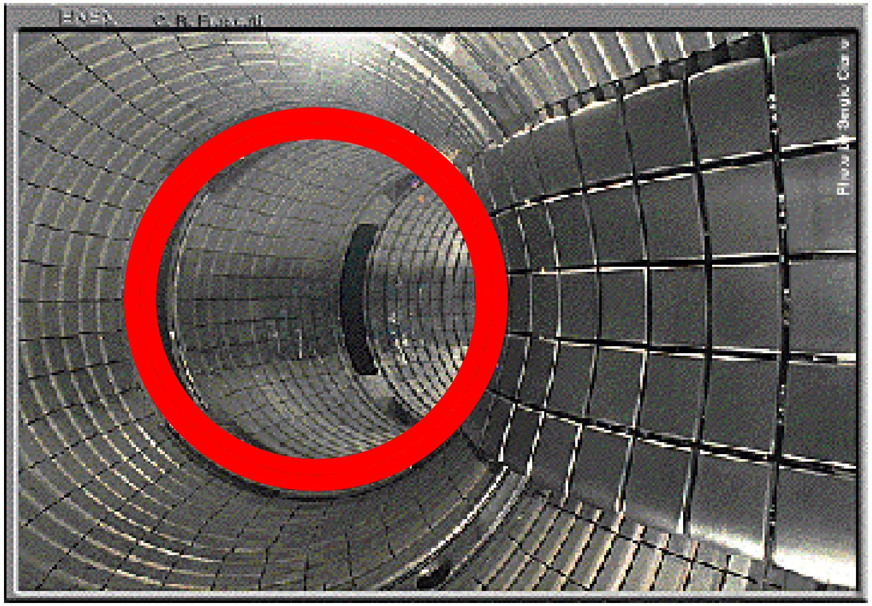
Plasma confinement in tokamaks is not ideal !!
Inner wall of the vessel has to survive harsh power flux (~ 1 - 20 MW/m^2)

- Heat flux must be diverted to specific location inside the vessel, where active cooling is possible
- First-wall elements exposed to the heat flux must be made of materials with a high melting point (graphite, tungsten)

Two concepts how to protect the first wall:
LIMITER – in small tokamaks of the first generation (old)
DIVERTOR – in large facilities with long pulse duration (novel since 1980)

Last Closed Magnetic Surface is defined by a LIMITER

Poloidal limiter –
circular diaphragm, which separates the hot plasma from the wall



Poloidal limiters were exploited in the first generation of tokamaks

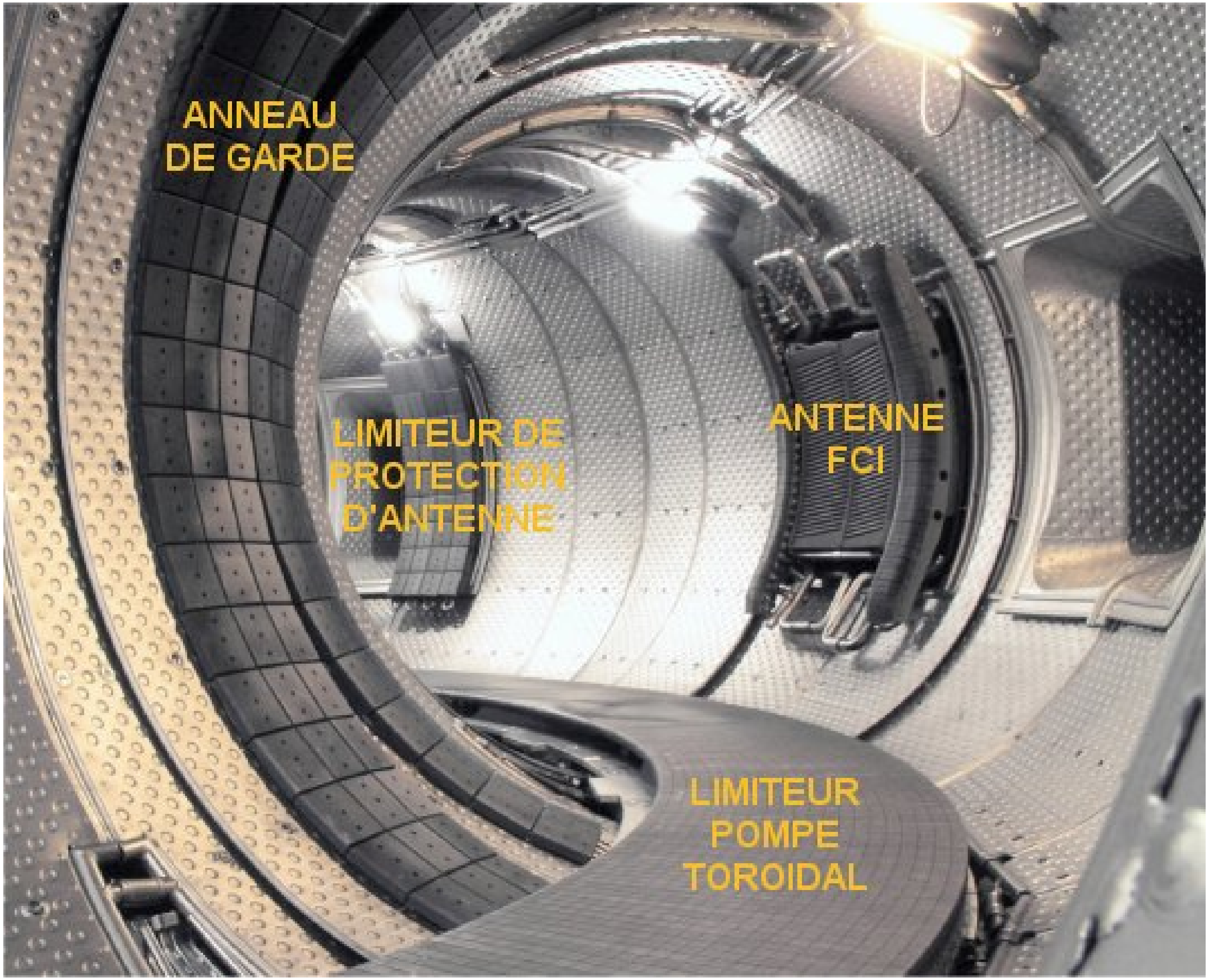
Advantage – simple construction

Problems – only s relatively small surface is exposed to the power losses, complicated cooling

Tokamak with a toroidal limiter

advantage – large surface => less power density

TORE-SUPRA, France



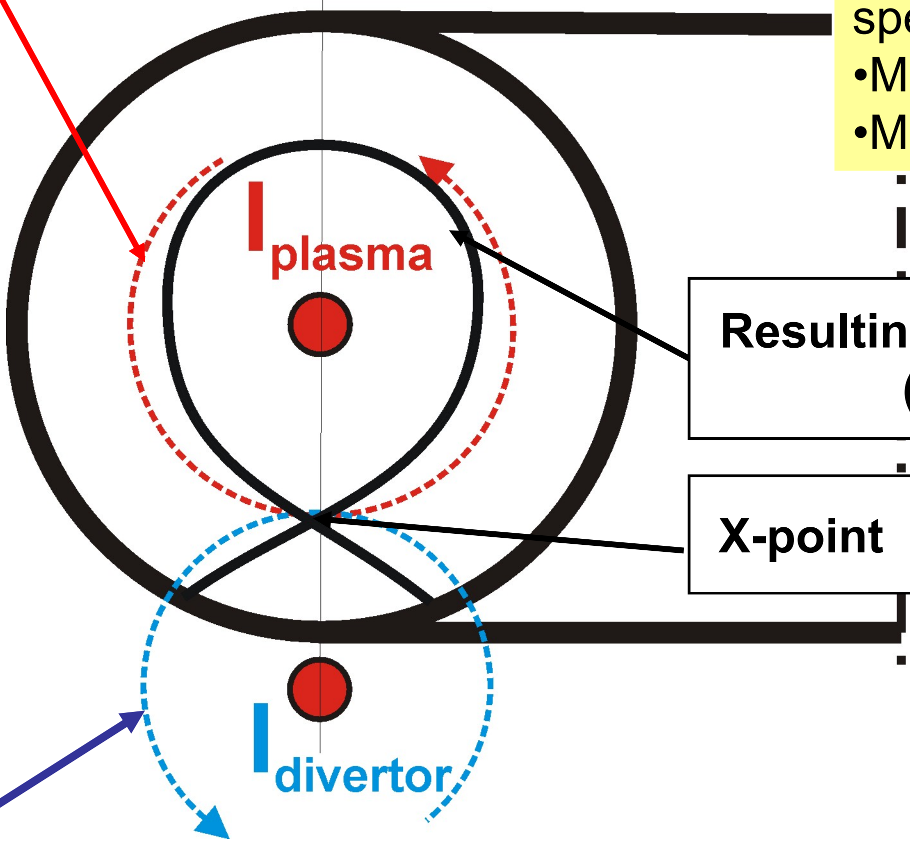
Evolution of the discharge in TORE Supra

Probe is inserted from the top of the torus

Divertor configuration

- Region at the bottom of the torus
- where the heat flux is dissipated by a special magnetic configuration
- Made of graphite or tungsten
- Massively cooled

Poloidal magnetic field of plasma current



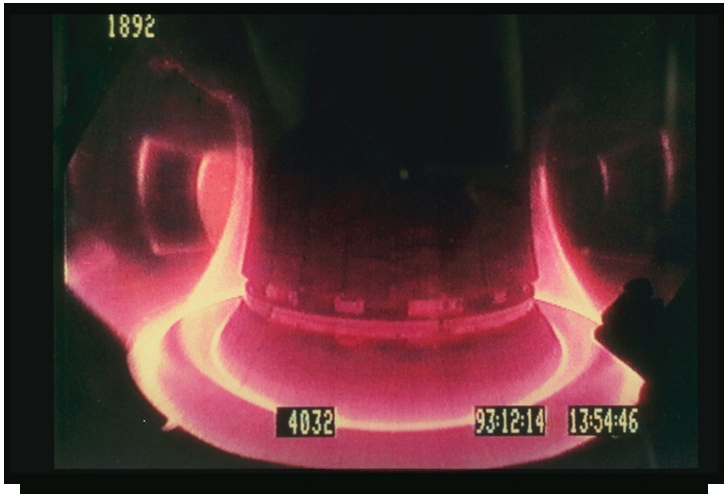
Resulting magnetic surface (separatrix)

X-point

Poloidal magnetic field of an external toroidal loop with the electric current in the same direction as the plasma current

Divertor in ASDEX-U

Garching bei Munchen



JET – *view into the discharge chamber*

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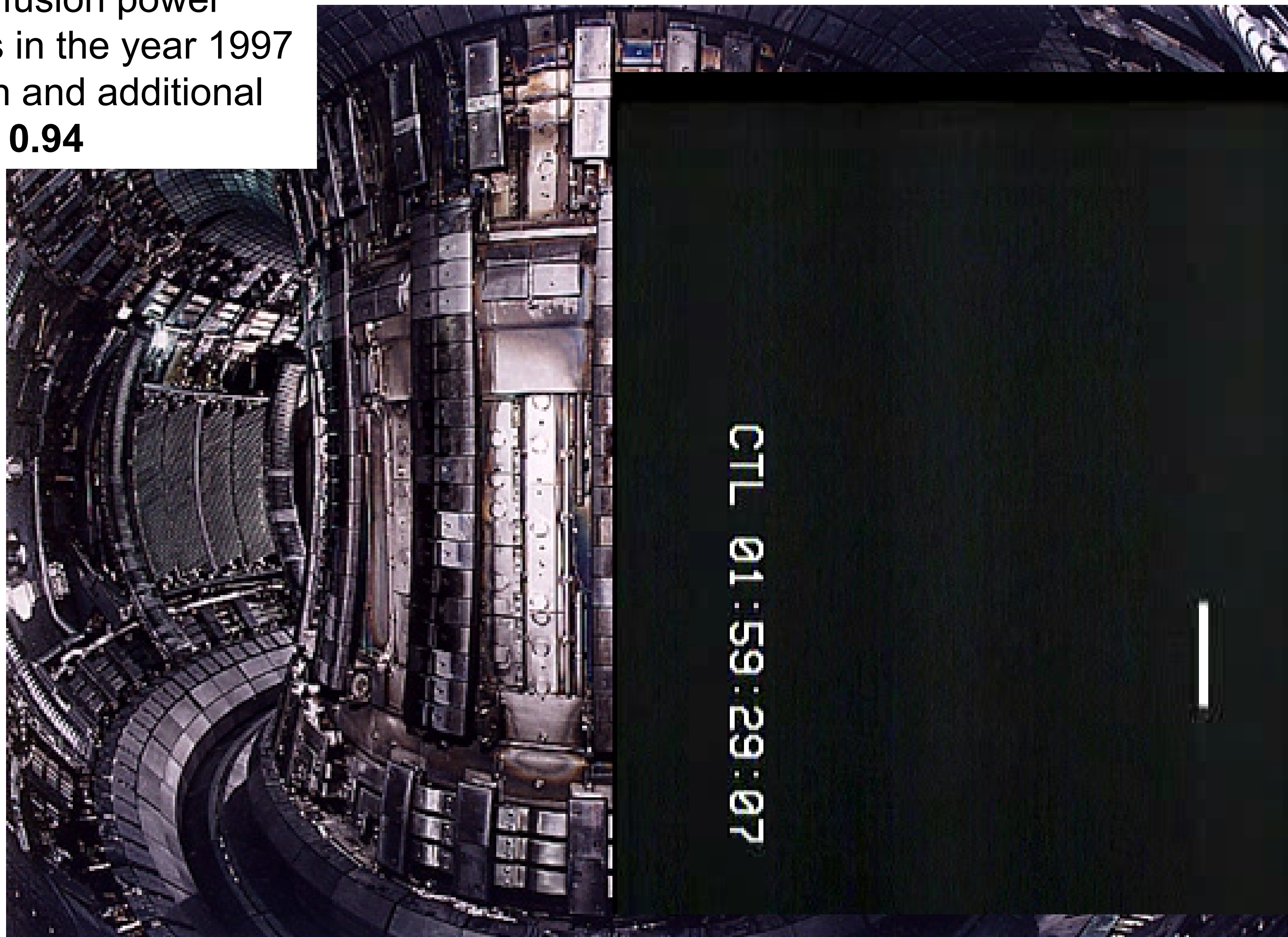
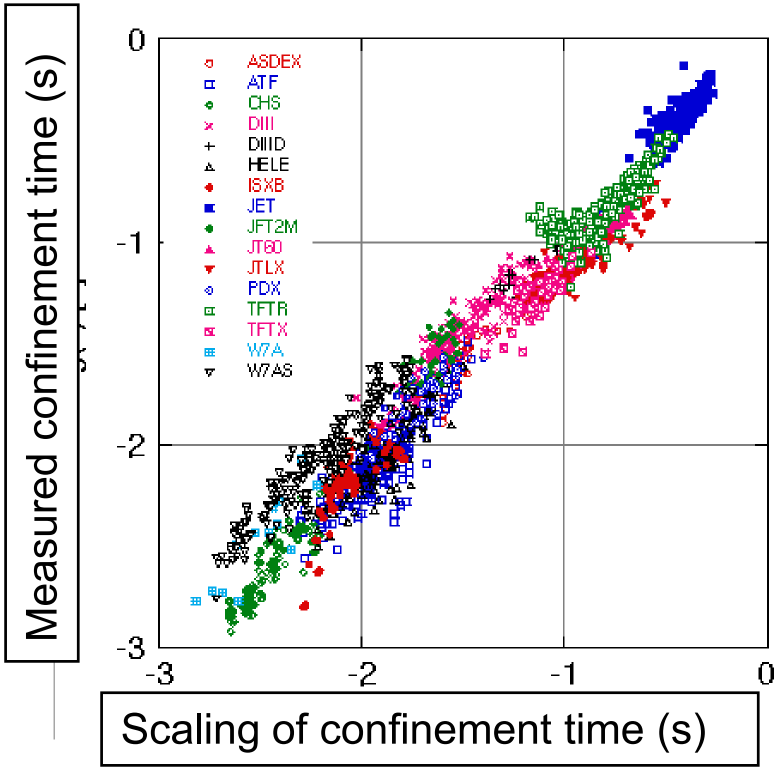
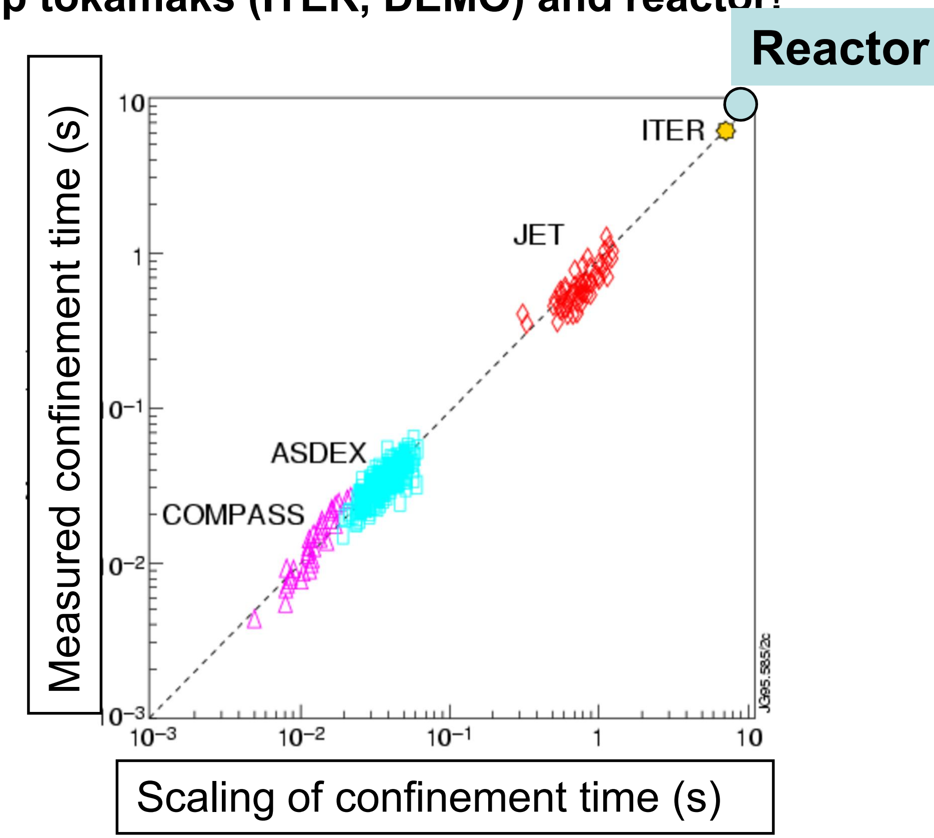
Production of fusion power **16 MW** for 1 s in the year 1997
Ratio of fusion and additional heating **Q_{tot}~ 0.94**



Extremely important for design of next-step tokamaks (ITER, DEMO) and reactor!





$$\tau_E \approx I_p^1 P_\Sigma^{-0.5} \kappa R^{1.75} a^{-0.37}$$

Confinement improves by increasing dimensions and current
Confinement deteriorates by increasing additional heating power

Next steps to fusion reactor

It is necessary:

- **Construct large tokamak (~3x larger than JET);**
- **With long pulse operation (500 – 1000 s);**
- **With fusion power 10 x larger than the heating power**

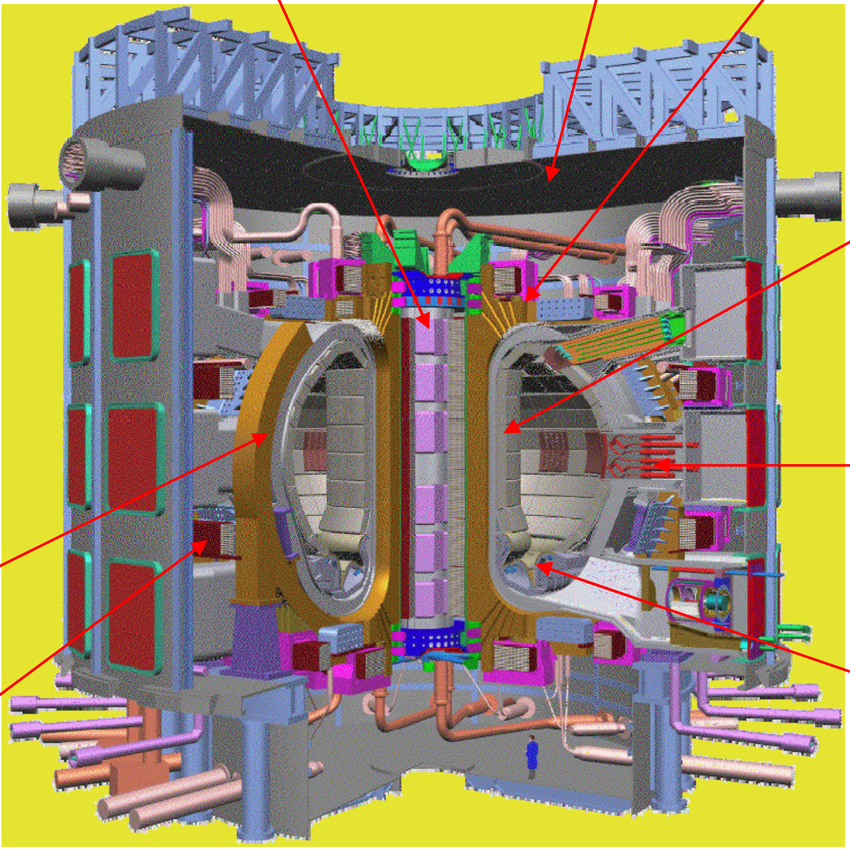
to clarify:

- **Plasma physics with dominant α particle heating (maybe new instabilities, transport barriers, ……);**
- **Technology of first wall at extreme power loads up to 20 MW/ m^2 (cooling, new materials, lifetime……);**
- **Technology blanket (separation of tritium, …..);**
- **and many other issues ………..**

International Termonuclear Experimental Reactor ITER

Plasma current	15 MA
Magnetic field (superconducting magnets)	5.3 T
Major radius	~6 m
Plasma volume	840 m^3
Pulse length	>400 s
Heating power	50 MW
Fusion power	500 MW

$Q_{fusion} > 10$



Current status of ITER project

Partners

- EURATOM (51%), Japan, USA, Russia, China, Korea, India (< 10%)
- Costs: construction ~ 10 billions Euro, operation ~ 5 billions Euro

At the moment

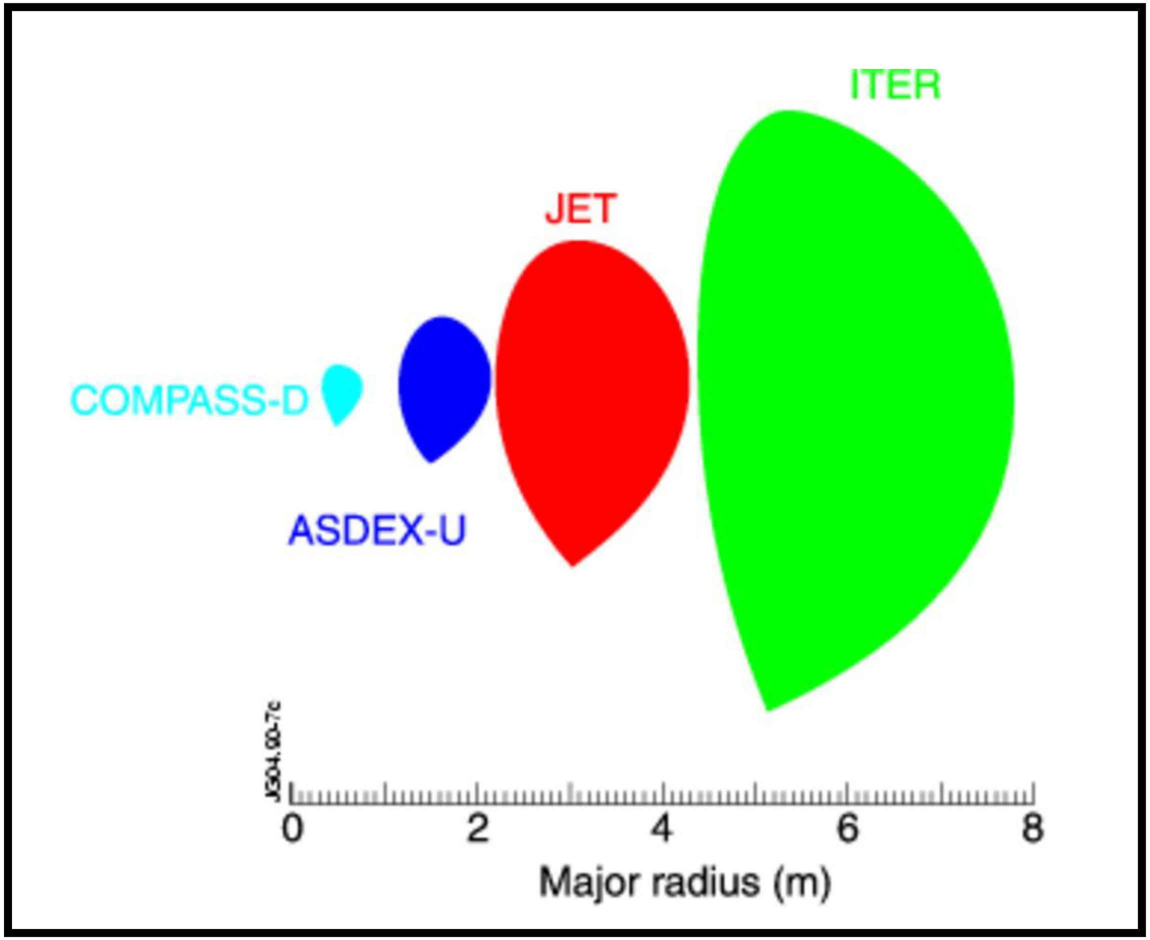
- Project is ready
- Location is fixed - Cadarache, near Marseille
- Legal entity, responsible for the project for next 40 years is formed
- Licensing process is underway in France (until mid of 2008)
- Start of construction 2007 (non-nuclear part - offices) , 2010 (experimental hall)
- Duration of construction ~ 8 years
- First plasma is expected in 2018, operational for next 25 years is envisaged

Czech contribution to fusion research - COMPASS

IPP AS CZECH REPUBLIC

COMPASS is a relatively small tokamak, but its geometry (magnetic configuration) is similar to ITER (but 10x smaller)

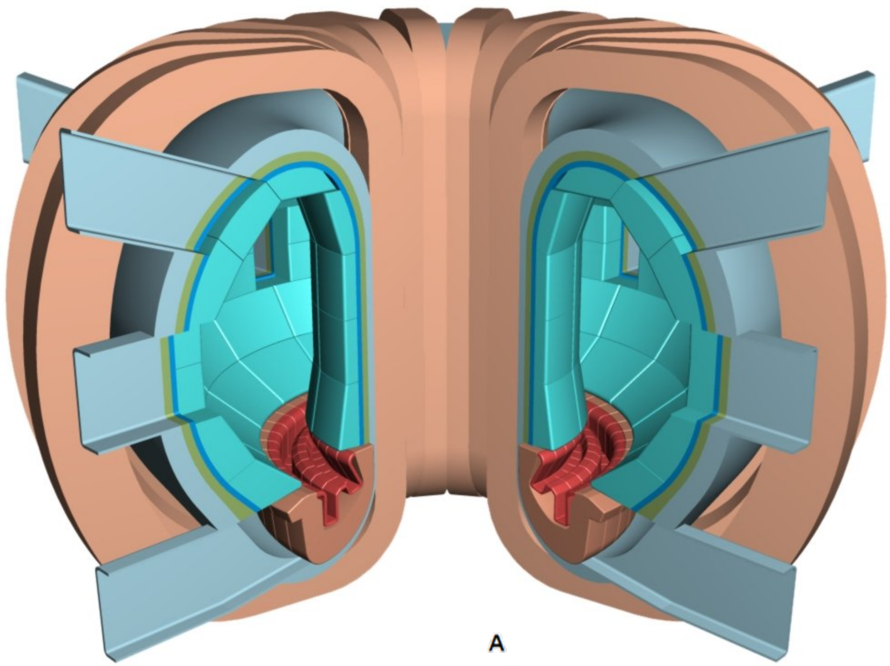
Cross section of plasma column

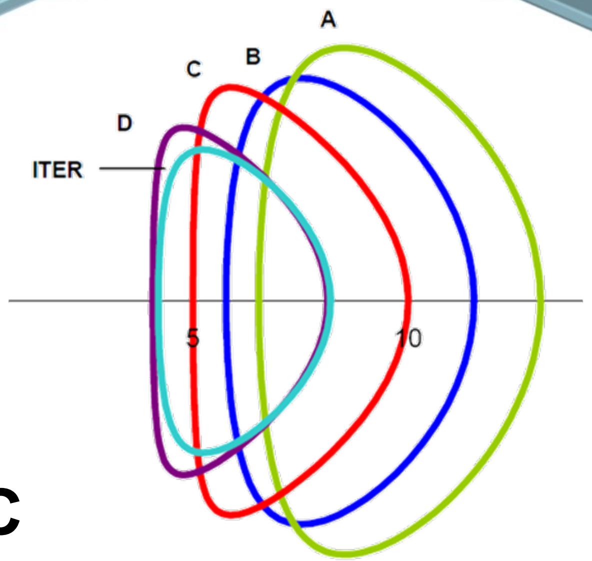


Major radius	0.56 m
Minor radius	0.2 m
Plasma current	<350 kA
Magnetic field	0.8–2.1T
Triangularity	~ 0.5
Elongation	~ 1.6
Pulse length	< 1 s

- Divertor configuration
- NBI (0.6 MW)
- Advanced diagnostics

DEMO



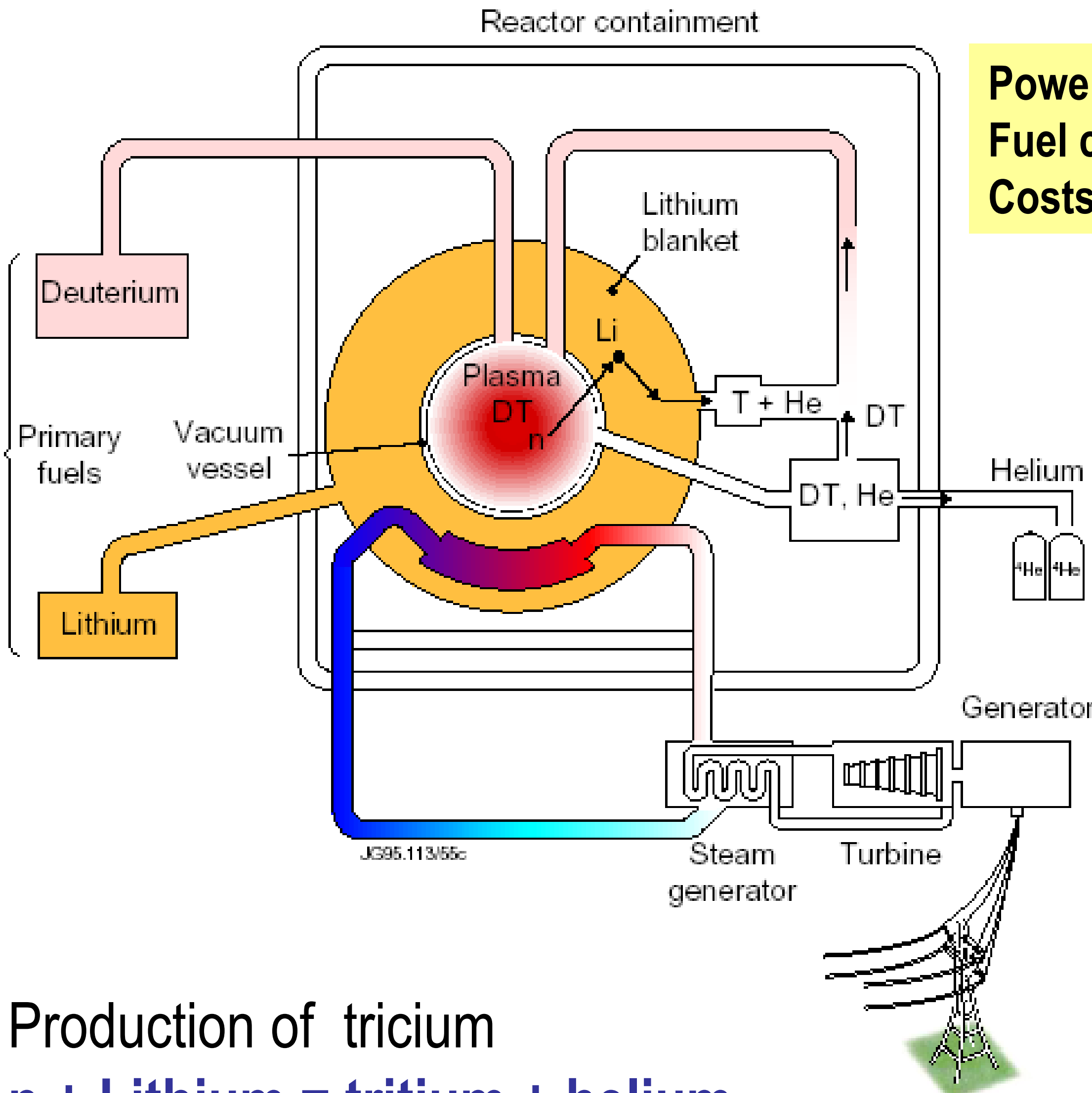


model C

- ITER-like tokamak
- long pulse operation (several hours)
- Electricity production
- First wall from tungsten
- Huge neutron flux
  (up to 80 dpa)
- Construction around 2030 (but maybe sooner – fast track)

Concept of fusion power plant

Power	**1-2 GW**
Fuel consumption	**~ 1 t D+T/rok**
Costs	**10 GigaEuro**



$$D + T \rightarrow {}^4He + n$$

Tritium production

$$^6Li + n \rightarrow {}^4He + T$$

$$^7Li + n \rightarrow {}^4He + T + n$$

Deuterium + Lithium ⟶ Helium + Energy

Production of tricium

n + Lithium = tritium + helium

Conclusion

- **Tokamak physicists are convinced that economical fusion reactor on the basis of magnetic confinement is possible to manufacture and operate around the year 2050 (tokamaks JET, TFTR, JT-60, ….).**
- **Key decision – to build ITER, where positive output of energy will be demonstrated**
- **However, many technological questions, like new materials, production of tritium, production of electricity etc, have to be solved (DEMO)**
- **Progress in fusion research depends definitelly on requirements of community (status of conventional energy resources, involvement of industry, ….)**