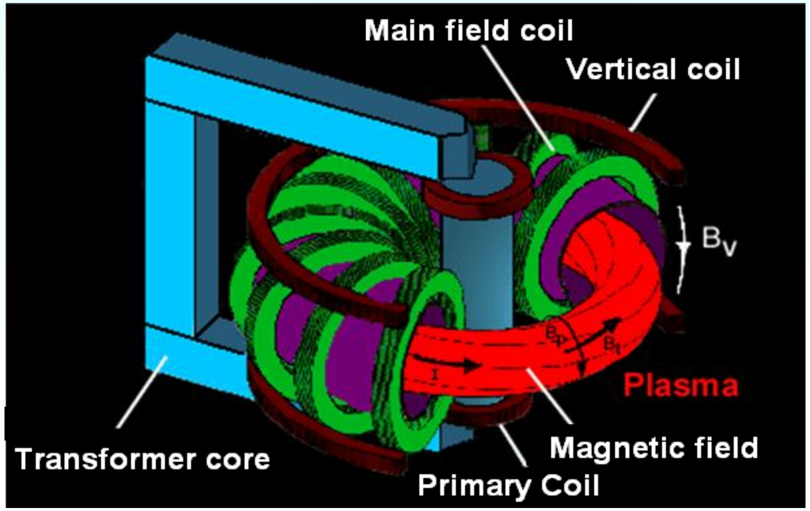
New technologies of electric energy converters and actuators

Contents

Source: Internet



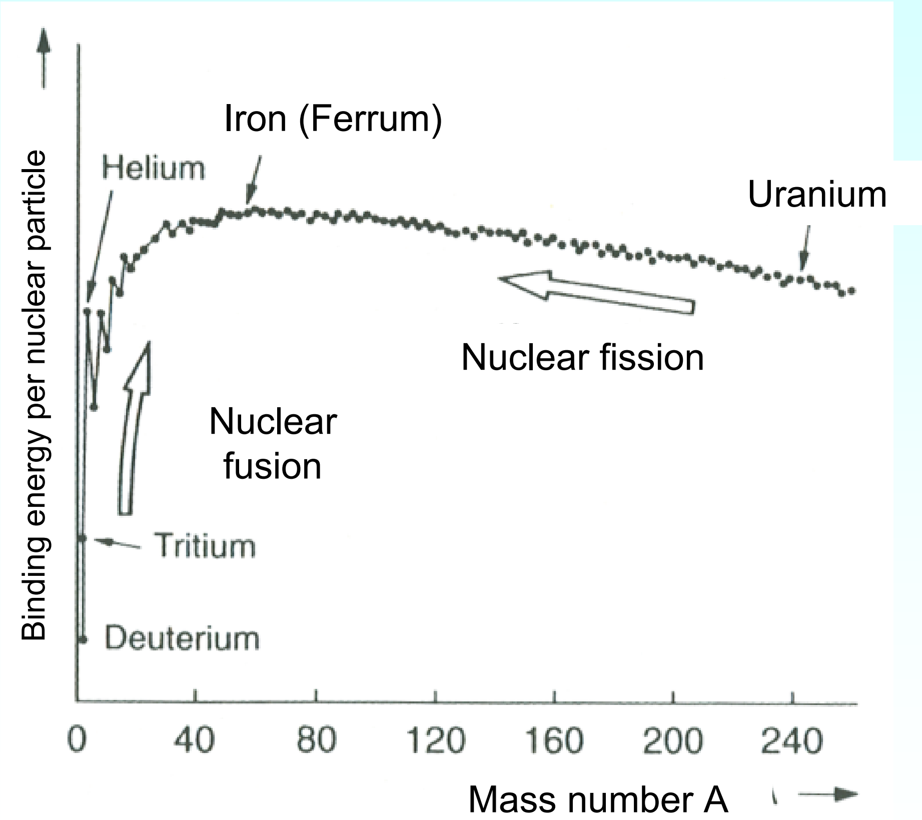
New technologies of electric energy converters and actuators

5. Fusion research

5.1 Fusion reaction

Linkage energy of nucleus elements

- Stable atomic nuclei (= highest linkage energy) are Fe, Co, Ni, Cu.
- Linkage energy is set free at fusion of lighter nuclei and fission of heavier nuclei.
- High gain of linkage energy at fusion from H-nuclei to He.



Source: FSZ Karlsruhe, Germany

Fusion reaction with H-ions

- **H-ions** with mass number 2 **(Deuterium D)** and mass number 3 **(Tritium T)** as reaction partners.

- Released linkage energy W_f per fusion is shared as kinetic energy on fusion particles.

$${}_1^2D + {}_1^3T \rightarrow {}_2^4He + {}_0^1n + 17.62 MeV$$, He: 3.52 MeV and n: 14.1 MeV

$${}_1^2D + {}_1^2D \rightarrow {}_2^3He + {}_0^1n + 3.27 MeV$$, He: 0.82 MeV and n: 2.54 MeV

$${}_1^2D + {}_1^2D \rightarrow {}_1^3T + {}_1^1p + 4.03 MeV$$, T: 1.01 MeV and p: 3.02 MeV

$${}_1^2D + {}_2^3He \rightarrow {}_2^4He + {}_1^1p + 18.4 MeV$$, He: 3.7 MeV and p: 14.7 MeV

- Rate of reaction R^* = Number of fusion reactions per volume V and time t

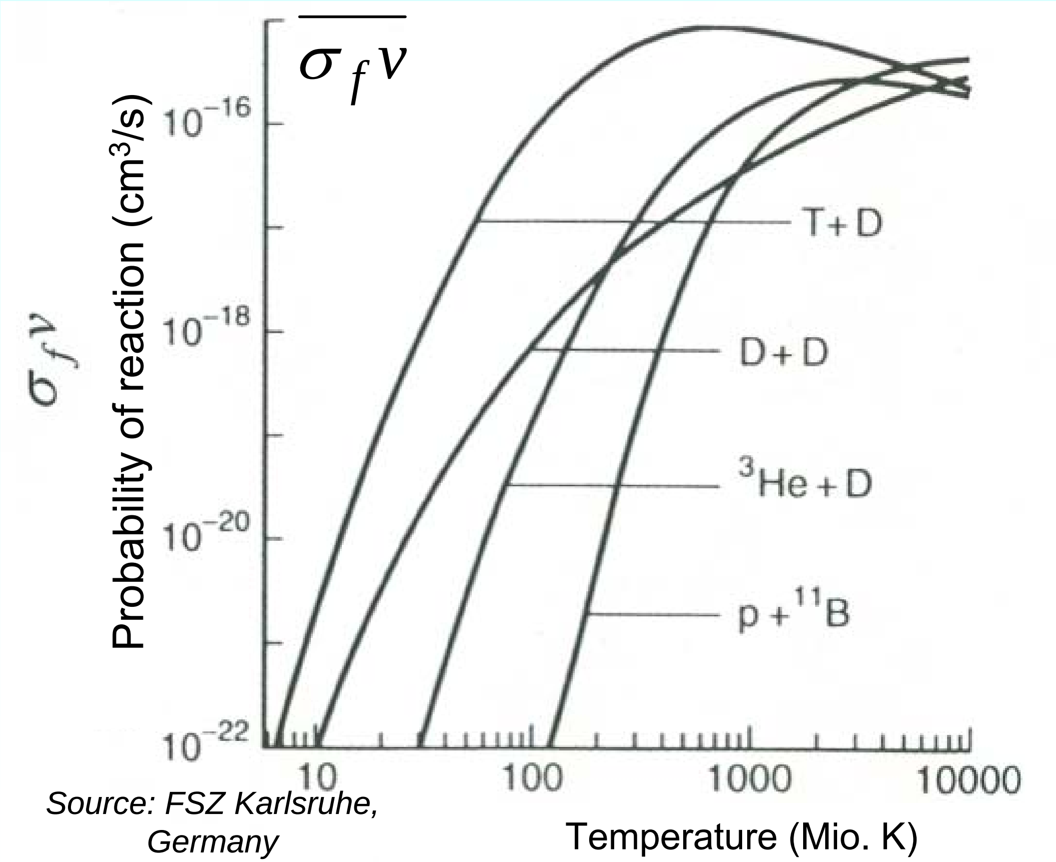
$$R^* = n_1 \cdot n_2 \cdot \overline{(\sigma_f \cdot v)}$$

n_1, n_2: Number of particles/volume of reaction partner
σ_f : „effective“ impact cross section
v : particle velocity

Example: $n_1 = n_2 = 10^{20}/m^3$, $\overline{\sigma_f v} = 10^{-22} m^3/s$: $R^* = 10^{18}$ reactions per m^3 and s

Necessary temperature for fusion reaction

- The graph below shows, that D-T-reactions have the highest reaction rate *R** at „lowest“ temperatures of „only“ 100 ... 200 Mio. K. This high *T* must be created artificially, because it does not exist on earth naturally.



Source: FSZ Karlsruhe, Germany

- Nuclei are charged positive: They have repulsive *Coulomb*-forces F_C
- High thermal nucleus velocity is needed to overcome F_C for an impact = high temperature *T* needed

$$m \cdot v^2 / 2 = (3/2) \cdot k \cdot T$$

- At $T = 10^8$ K every gas is completely ionized in (+) ions and (-) electrons = This is called a *„plasma“*

Radiation losses and heat losses at fusion

- Radiation loss density $p_b = P_b/V$: Orbits of charged particles are deflected due to collisions and experience acceleration & deceleration. Hence they radiate electro-magnetic waves. This radiation leaves the plasma = loss of plasma energy.

- Diffusion- and heat loss density $p_d = P_d/V$: Particles are leaving the plasma due to collisions = loss of kinetic energy = loss of heat

- Time of inclusion τ_E: During this time a hot plasma loses its complete thermal energy W_{th} by P_d-losses. $W_{th} = P_d \cdot \tau_E$

- *Example:* $n_i = n_e = n, \; T_i = T_e = T:$

Kinetic energy per particle: $m \cdot (v_x^2 + v_y^2 + v_z^2)/2 = (3/2) \cdot kT$

Thermal energy density: $W_{th}/V = 2n \cdot (3/2) \cdot kT$

Loss density of diffusion / heat conductivity: $p_d = P_d/V = n \cdot (3 \cdot kT)/\tau_E$

Conclusion:

For limited losses p_d, the time of inclusion should be very long.

New technologies of electric energy converters and actuators

Summary:
Fusion reaction

- Nuclei with low mass number result in free energy, when fused to heavier nuclei
- Deuterium-Tritium fusion to Helium and free neutrons deliver maximum energy per fusion reaction
- 150 … 200 million K necessary to generate stripped nuclei ("plasma")
- Radiation and diffusion losses during the fusion must be supplied by the fusion energy itself

New technologies of electric energy converters and actuators

5. Fusion research

5.2 Stable fusion operation

Stable fusion operation

1. Plasma is heated with an external heat power source density $P_h/V = p_h$.

2. After fusion, the fusion energy W_f is distributed on the α-particles (He- or T-nuclei) as kinetic energy W_α and on the neutrons as kinetic energy W_n:
   $W_f = W_\alpha + W_n$
   Thermo-nuclear power density of fusion: $p_t = R^* \cdot W_f = p_\alpha + p_n$
   with $p_\alpha = R^* \cdot W_\alpha,\ p_n = R^* \cdot W_n$

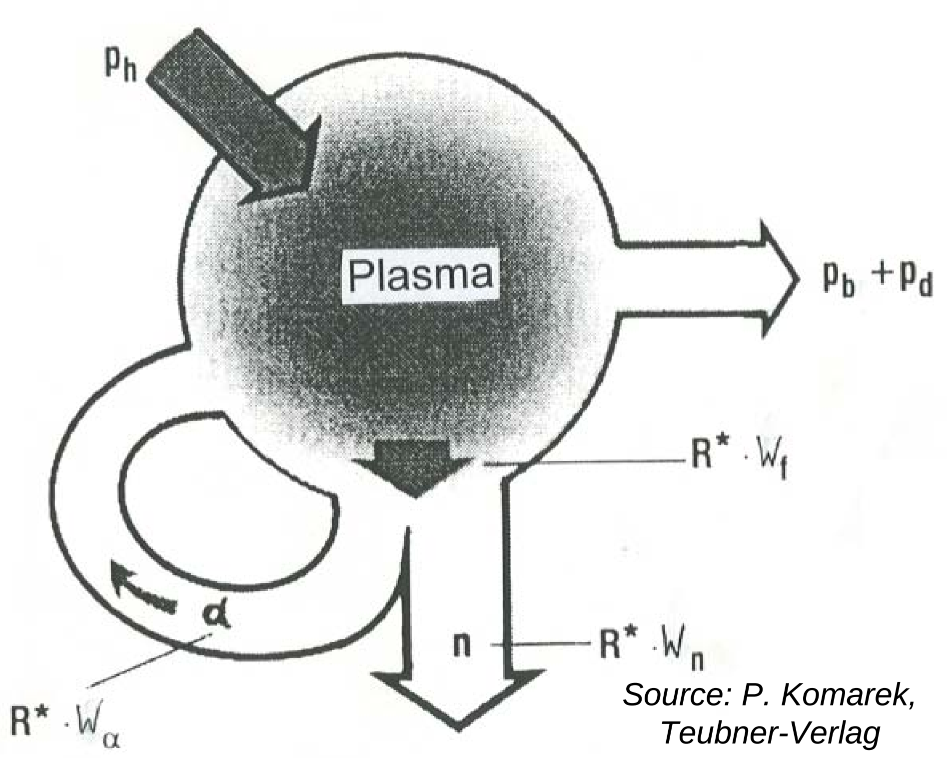
3. Stable fusion operation: No further heat input is needed: $p_h = 0$. The losses are covered by the kinetic energy of the plasma α-particles: $p_\alpha = p_b + p_d$

4. Neutrons leave the plasma. Their kinetic energy is used to convert water via a heat exchanger into steam: $p_{steam} = p_n$.

This stable fusion operation with steam generation could not be reached till now!

JET-Experiment/*Culham, UK*, reached only the „zero reactor condition“:
(= no-load operating fusion reactor without losses). $p_t = p_\alpha + p_n = p_h$

Energy flow at stable fusion operation



Source: P. Komarek, Teubner-Verlag

- Estimate of ignition condition:
  a) Particle density *n*: $n_D = n_T = n/2$
  b) Neglecting radiation losses

$$p_b = 0: \; p_\alpha = p_d$$
$$p_\alpha = R^* \cdot W_\alpha = (n/2)^2 \cdot \overline{(\sigma_f v)} \cdot W_\alpha$$
$$p_d = 3 \cdot n \cdot kT / \tau_E$$
$$\Rightarrow n \cdot \tau_E = \frac{12 \cdot kT}{W_\alpha \cdot \overline{(\sigma_f v)}}$$

c) D-T-reaction: $T = 2 \cdot 10^8 K$
$\overline{\sigma_f v} = 10^{-15} cm^3 / s$

with $k = 1.38 \cdot 10^{-23} J/K, n = 10^{20} / m^3$
$W_\alpha = 3.52 MeV = 5.6 \cdot 10^{-13} J$ we get:

$$\tau_E = 0.6 s$$

- *Ignition:* Particle density: $10^{20}/m^3$,
  Temperature: 100 ... 200 Mio. K,
  Inclusion time: 1 ... 2 s
  (LAWSON-criterion)

New technologies of electric energy converters and actuators

Summary:
Stable fusion operation

- Radiation and diffusion losses must be supplied by the kinetic energy of the generated Helium nuclei
- Free neutrons heat up the vessel wall with the heat exchanging fluid
- Minimum time of ca. 1 … 2 s with a particle density $10^{20}/m^3$ at a temperature of 100 ... 200 million K necessary for stable operation **(*LAWSON-criterion*)**

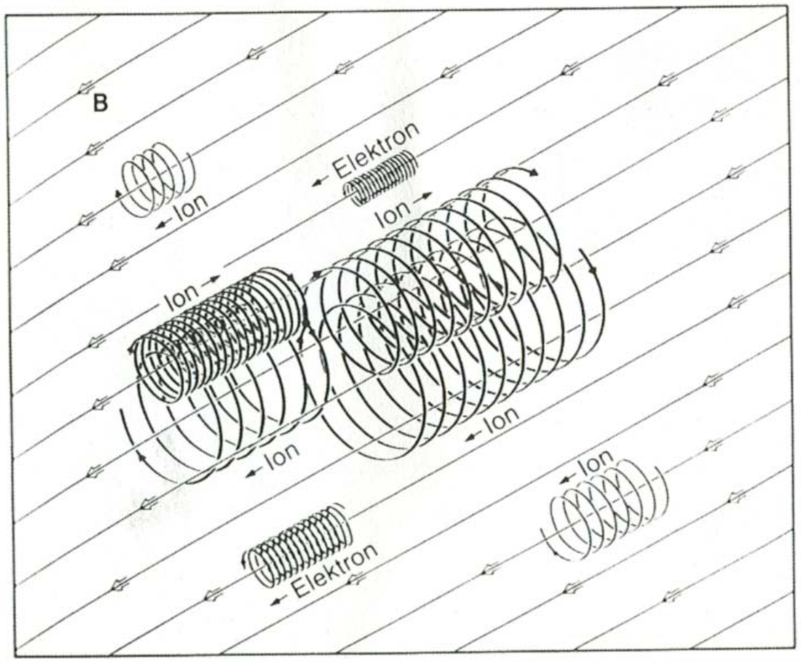
New technologies of electric energy converters and actuators

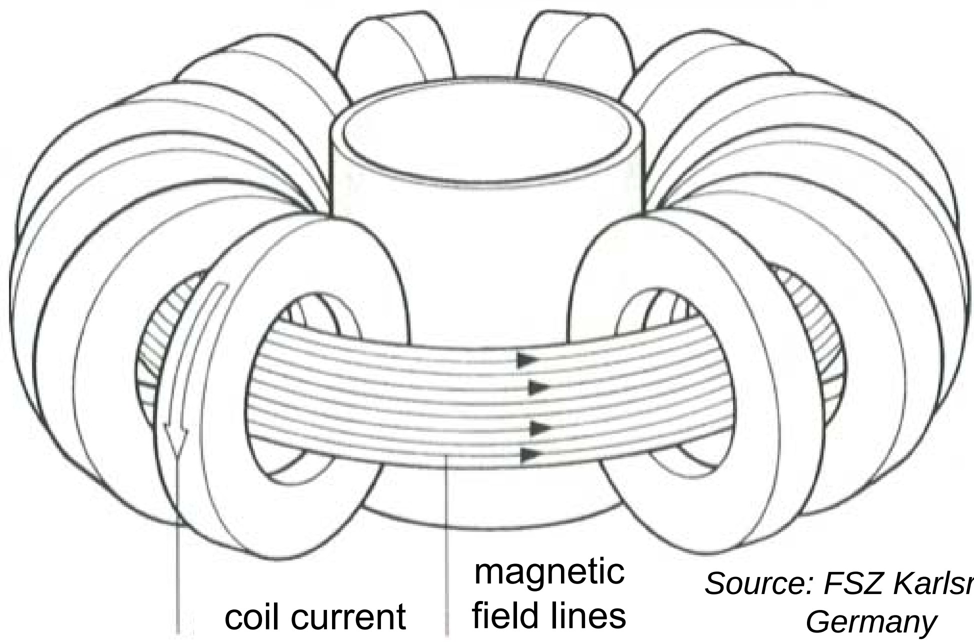
5. Fusion research

5.3 Magnetic field layout for contactless plasma-inclusion

Magnetic inclusion of the plasma

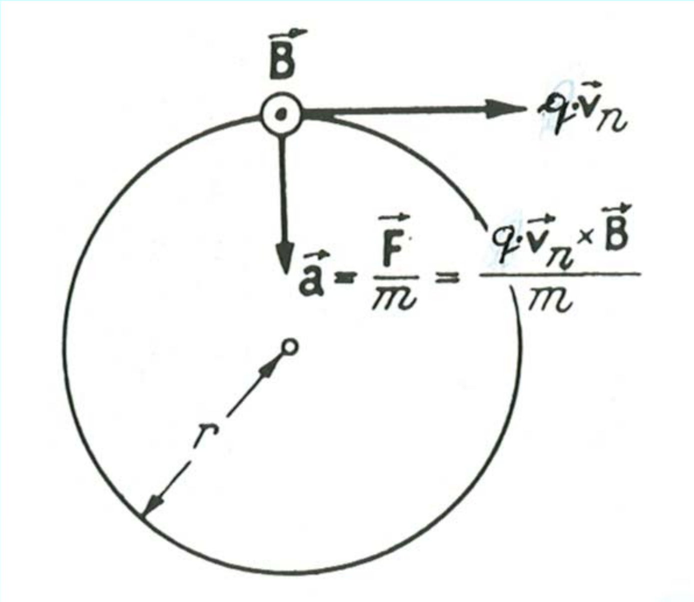
- No solid material stands temperatures of 100 ... 200 Mio. K. So the plasma should be included contact-free by magnetic forces.
- Positive & negative charged particles move on spiral lines with contrary senses of rotation in / against *B*-field direction = no lateral leakage of particles possible!
- Ring coils excite a torus *B*-field = The closed field lines do not allow a frontal leakage of particles!

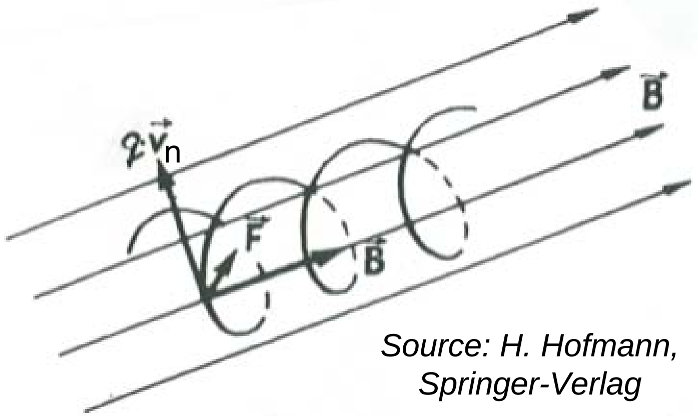




Source: FSZ Karlsruhe, Germany

Circular orbit of a charged particle in *B*-field





Source: H. Hofmann, Springer-Verlag

- Magnetic field *B*, particle velocity *v* :
  v decomposed into $v_t \uparrow\uparrow B$ and $v_n \perp B$.
- *LORENTZ-force* on electric charge *q*:

$$\vec{F} = q(\vec{v} \times \vec{B}) = q \cdot v_n B \cdot \vec{e}_{vB}$$

Particle is accelerated normal to *B* and *v* :
$\vec{a} = \vec{F} / m$: This leads to circular orbit with radius r.

- Centrifugal force on orbiting particle: $F_F = m \cdot v_n^2 / r$
- Equilibrium determines radius: $F = F_F$

$$r = m \cdot v_n / (q \cdot B)$$

- Angular frequency: $\omega = v_n / r = (q \cdot B) / m = 2\pi \cdot f$

Example: Tritium nucleus: $q = |e| = 1.6 \cdot 10^{-19} As$

m = m_0 = mass at rest: $5.1 \cdot 10^{-27} kg$

At *B* = 5 T: *f* = 25 MHz

Screwing up of torus *B*-field lines

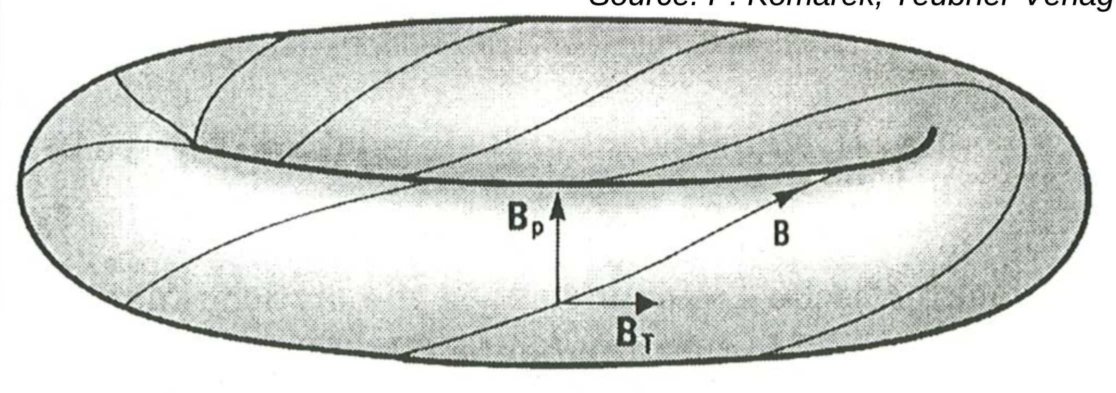
- Torus *B*-field is bigger inside than outside: $\oint_C \vec{H}\cdot d\vec{s} = Q\cdot N_c\cdot I_c = 2\pi\cdot r\cdot H_t$

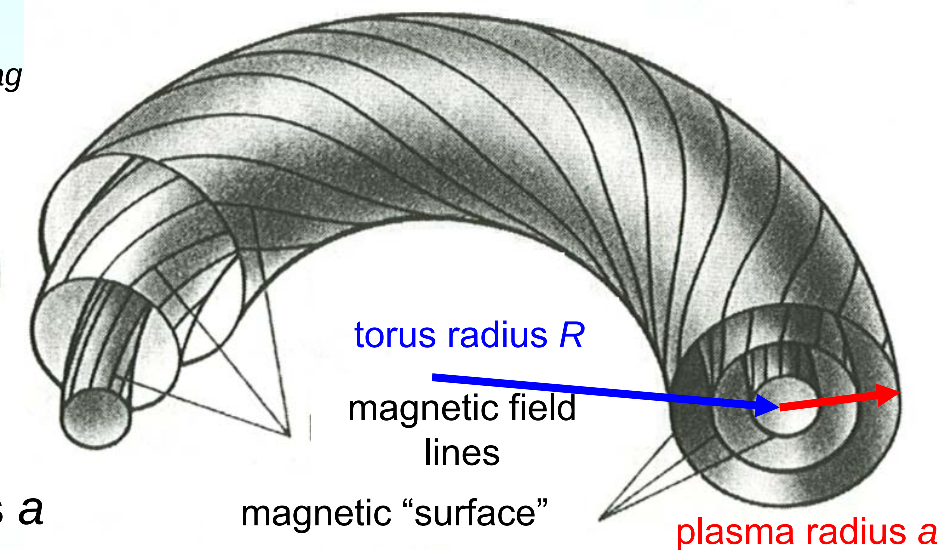
$$B(r) = \mu_0\cdot Q\cdot N_c\cdot I_c/(2\pi\cdot r)$$

Q: number of ring coils, N_c: turns per ring coil, I_c: ring coil current

- Due to $B \sim 1/r$ the charged particles are drifting from inside to outside = the plasma gets UNSTABLE. *Help:* Poloidal field B_p in direction of the ring coil currents screws up the B_t-field lines: $\vec{B} = \vec{B}_t + \vec{B}_p$. So the particles stay inside of the „torus *B*-surface" and do not drift laterally.

Source: P. Komarek, Teubner-Verlag





„Aspect ratio" *A* = Torus radius *R* / Plasma radius *a*

New technologies of electric energy converters and actuators

Summary:
Magnetic field layout for contactless plasma-inclusion

- Torus-like field for magnetic trapping via *Lorentz* forces on charged particles
- Transversal drift of particles due to field gradiemt must be compensated
- Screwed field lines as drift compensator
- TOKAMAK and STELLARATOR arrangements for screwing the field lines

New technologies of electric energy converters and actuators

5. Fusion research

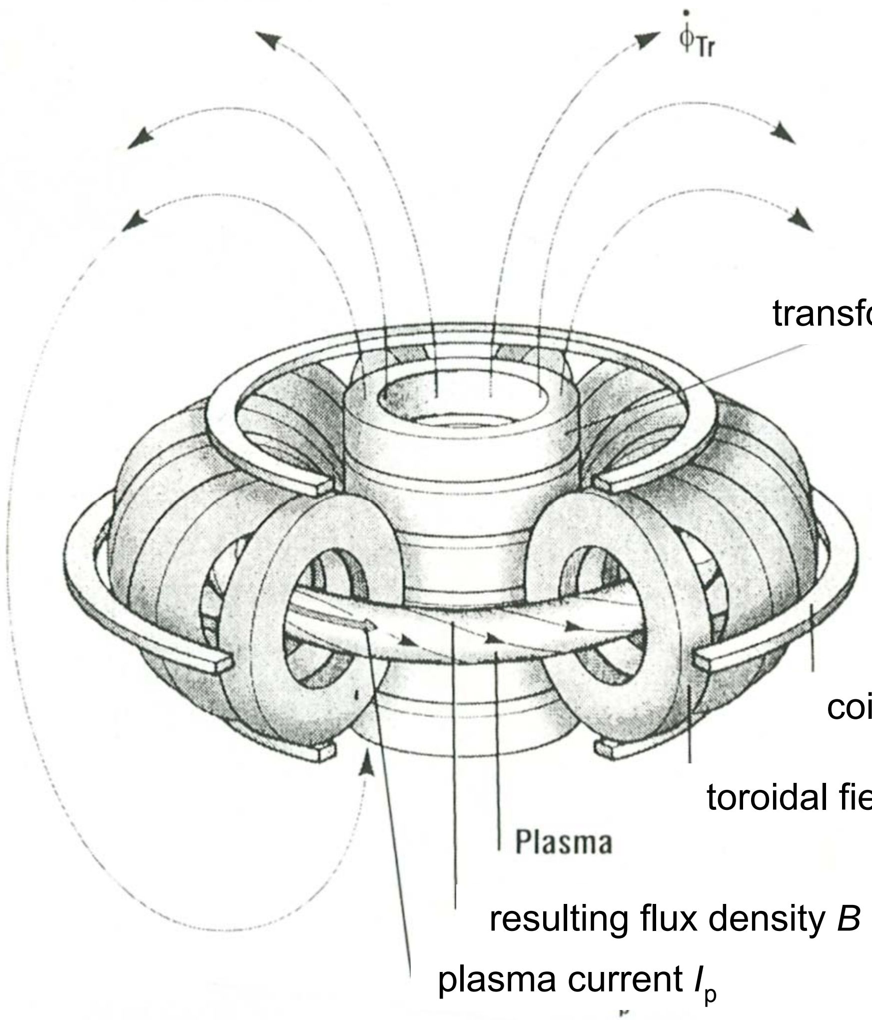
5.4 TOKAMAK and STELLARATOR

Creation of poloidal field B_p

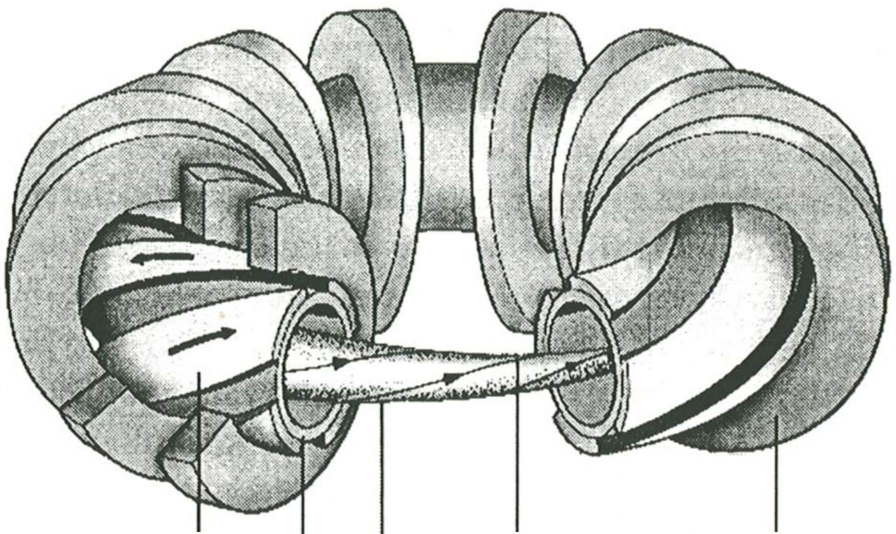
TOKAMAK	STELLARATOR
Adjustable current in a central transformer coil induces a circular plasma current I_p, its self-field B_p screws up the field lines B_t .	Helical coils (= screw-shaped conductors) with alternating DC current direction create a field B_p, that screws up the field lines B_t.
Aspect ratio: A = 3.5 ... 4	A = 10 ... 15
Plasma volume is shaped as a torus ring	Plasma ring is deformed and screwed with varying cross-sectional areas
Heating of plasma is done by I_p: Current heating	Heating of plasma via external source (e. g. HF-heating with an ele. mag. wave)
Pulsed operation, because transformer current must change. Continuous operation possible, if plasma current is kept in hot plasma via an external heating.	Continuous operation possible
Large scale experiments already done	Helical coils difficult to manufacture. *German* approach with non-planar coils instead.

5.4 TOKAMAK and STELLARATOR

Tokamak



Stellarator

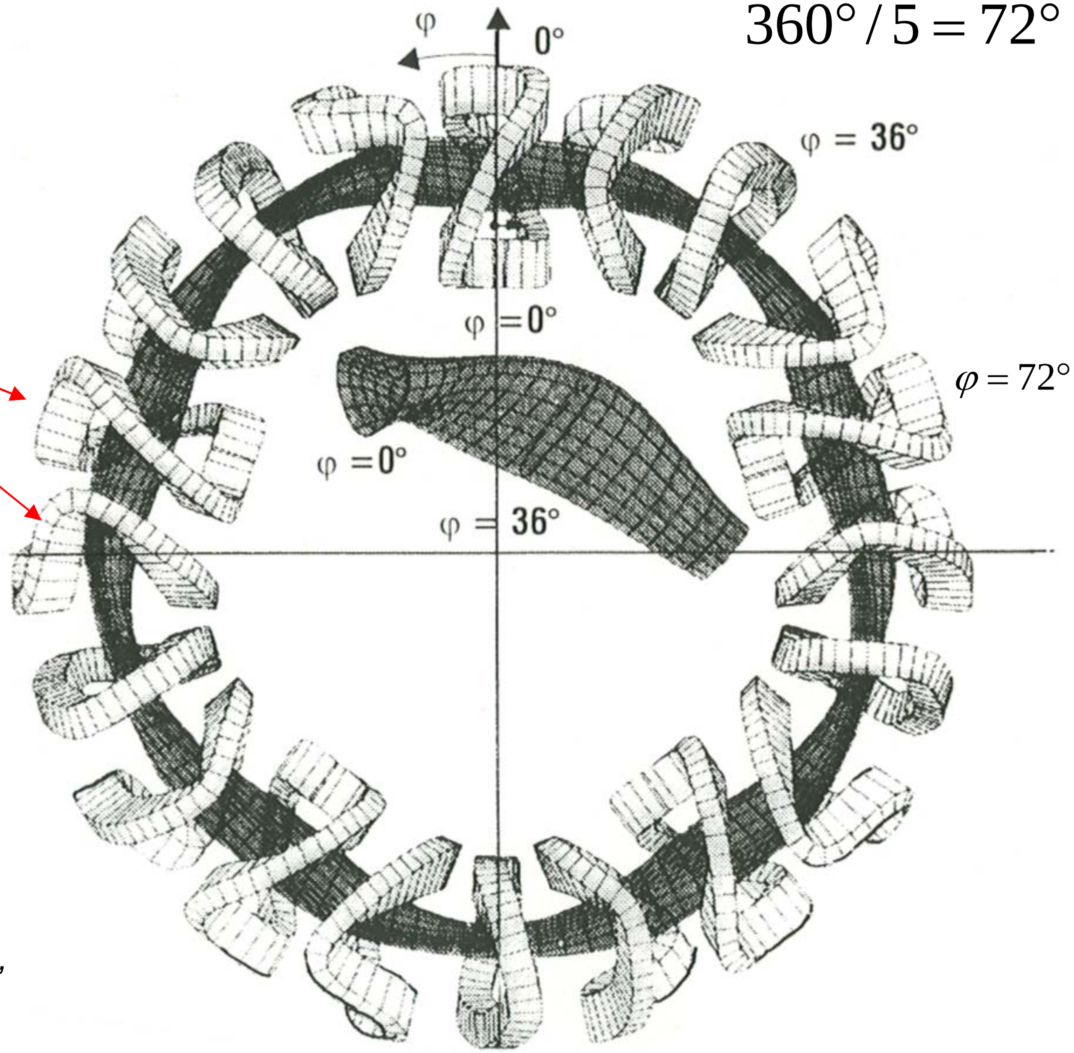


Source: P. Komarek, Teubner-Verlag

Non-planar coils in Stellarator

- Helical coils inside of the torus ring coils are difficult to manufacture
- *Help:* Three-dimensional bent coils („non-planar“) can be manufactured more easily (Idea of *Garching* research centre)
- These coils replace the ring coils and helical coils and excite the same resulting magnetic screwed-up torus magnetic field

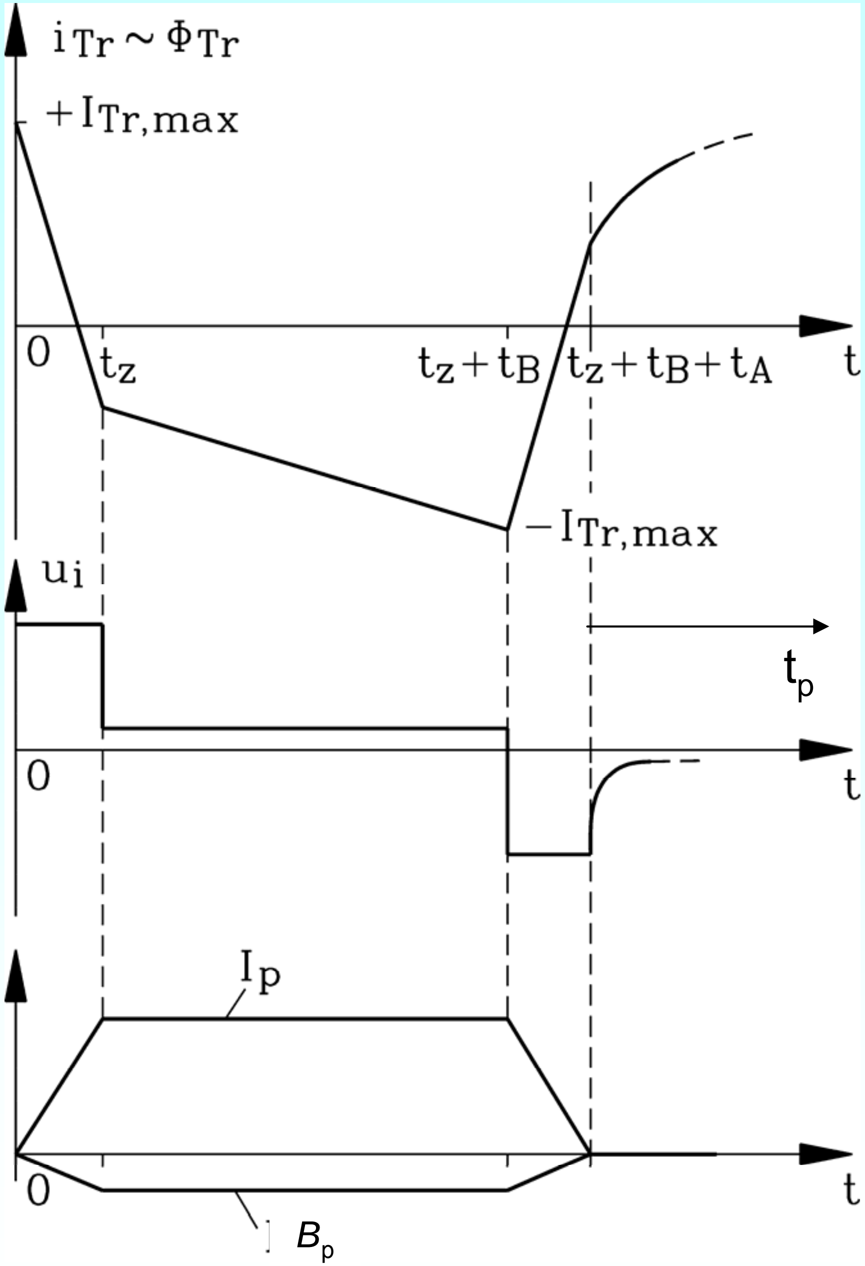
Wendelstein W7-AS, *Garching*,
4 different coils, 5 x 4 = 20 coils



Source: P. Komarek, Teubner-Verlag

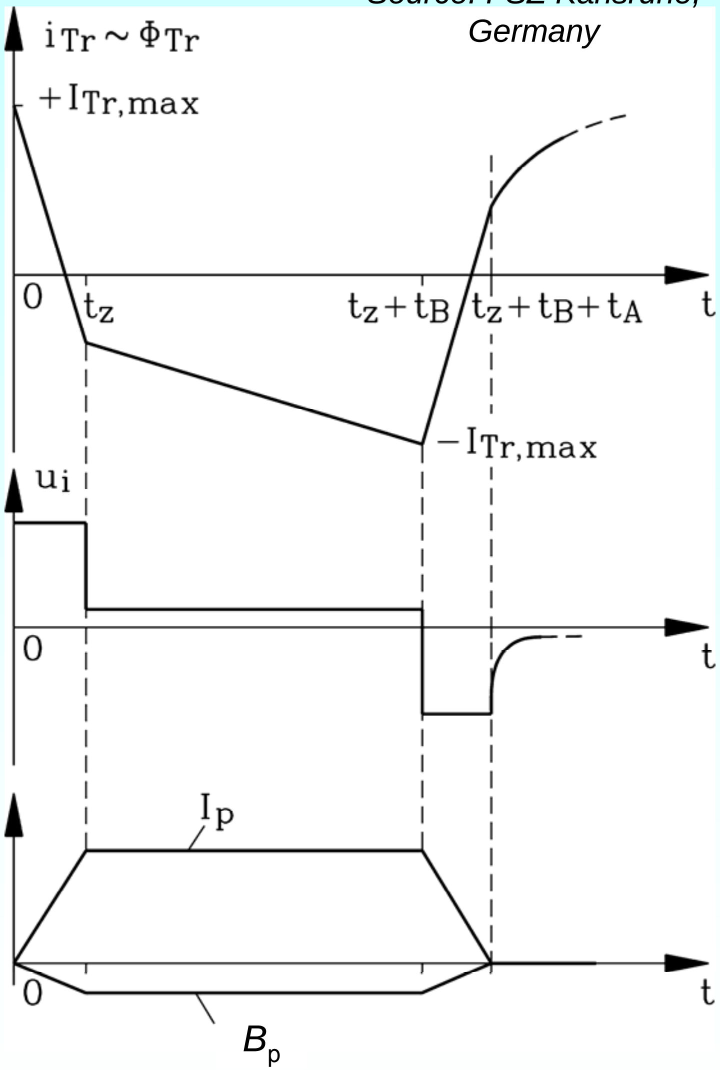
Tokamak: Variation of operational parameters with time

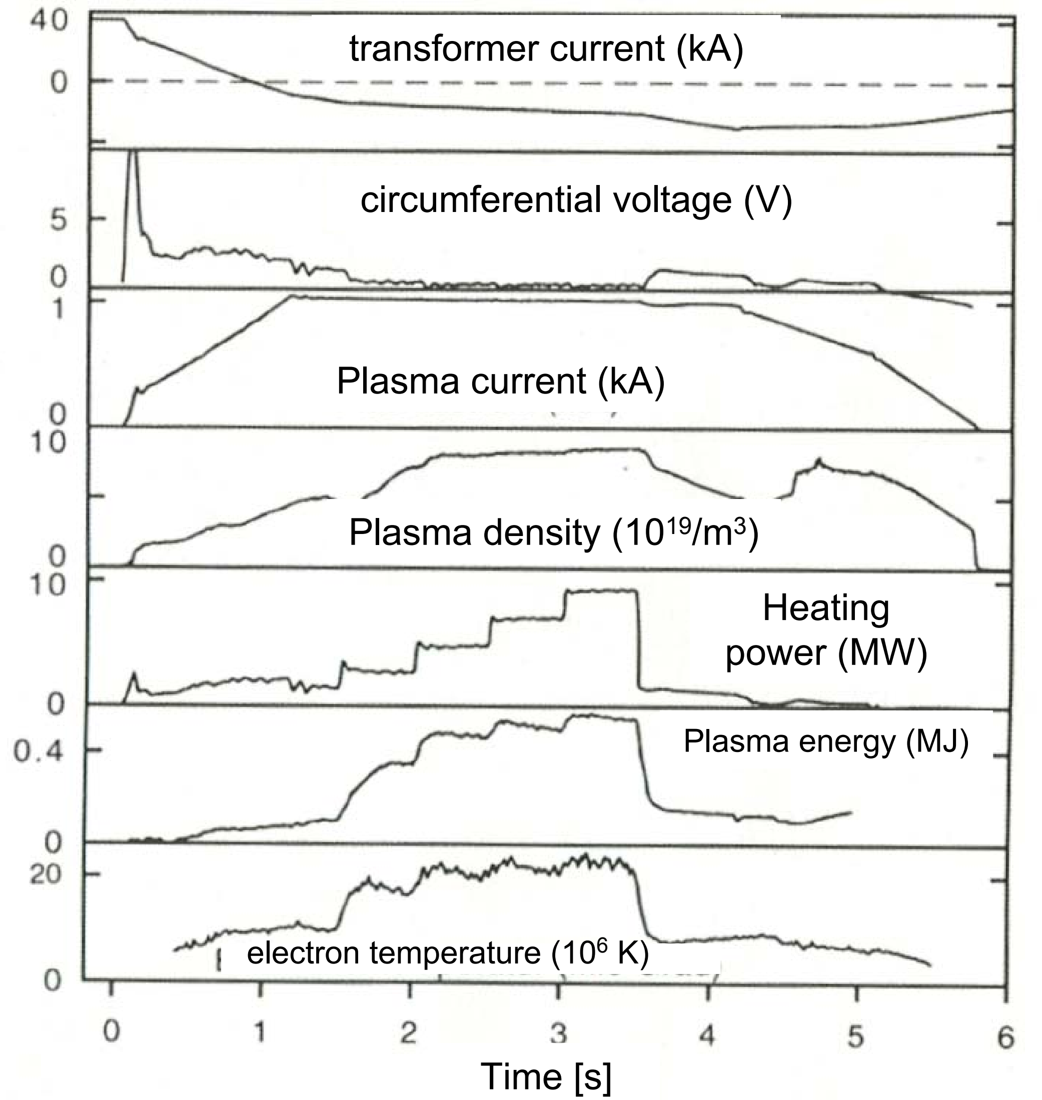
- t_z: Ignition of plasma at 10^{-5} mbar by induced voltage u_i
- t_B: Ohmic heating of plasma via the plasma ring current I_p („burning phase") and operation for fusion
- t_A: Switch-off after reaching the max. negative transformer current
- t_p: Change of transformer current to positive value, „charging" of short-circuit generators



Tokamak: Measured time variation during operation

Source: FSZ Karlsruhe, Germany





Feeding of *Tokamak* central transformer coil via a homopolar generator

- A solution for feeding the central transformer coil with low voltage, high current is with homo-polar generators, having large rotor inertia J due to an additional fly-wheel.

- The homo-polar generator is driven by a small motor up to rated speed n at no-load operation. The stator field winding is excited with DC field current I_f and induces in the rotor winding the no-load DC voltage U_{a0}.

- The driving motor is switched off. Via a DC current breaker the *Tokamak* coil is switched to the homo-polar generator. With variable I_f a variable voltage U_a drives a variable transformer coil current. With a high peak current the induced plasma ring voltage ignites the fusion via ohmic heating.

- At a low rate of change of I_f from $I_{f,max}$ to $-I_{f,max}$ during the operating time of e.g. 10 s the transformer field coil reverses from $U_{a,max}$ to $-U_{a,max}$, so that a constant plasma ring current is obtained. During that time the speed of the homo-polar generator drops, as its kinetic energy $W_{kin} = J \cdot (2\pi \cdot n)^2 / 2$ is consumed by the braking effect of the electrical losses.

- Afterwards the transformer coil is switched off. The driving motor has to spin up the generator again ("Charging"), and the pulsed operation starts again.

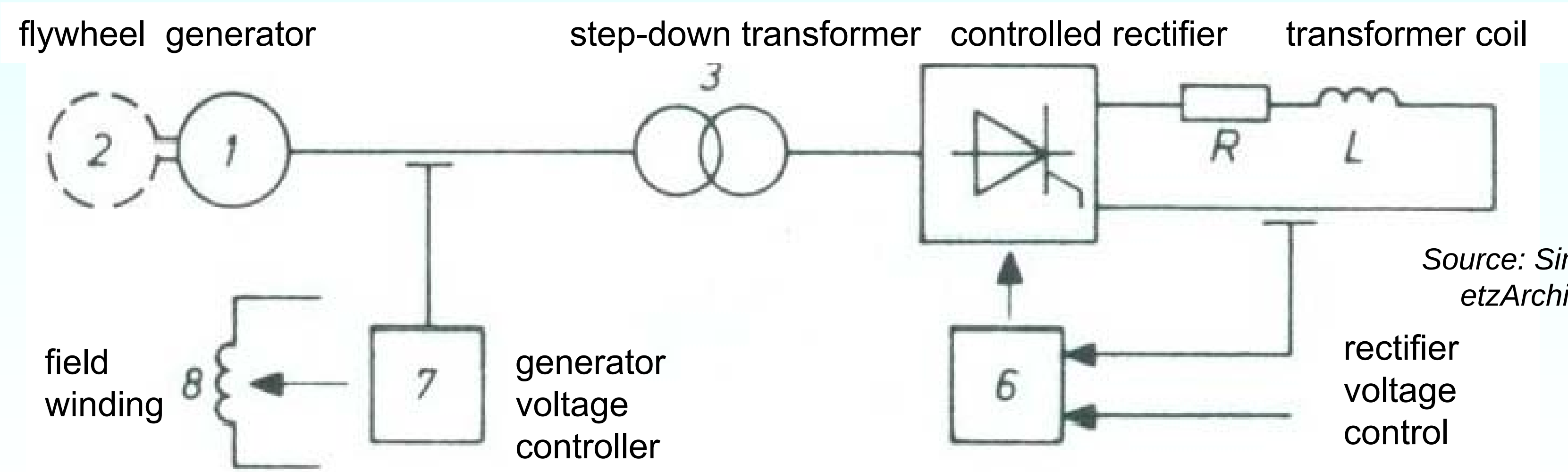
Feeding of *Tokamak* central transformer coil via a synchronous generator

- The feeding with a time-varying transformer coil current is done from a special synchronous turbine generator (1) with a thyristor-controlled rectifier (4)

- The generator is spun up to rated speed by a small motor over long time. Then the kinetic energy in the rotating mass is transformed into electrical energy, fed via the rectifier to the transformer coil.

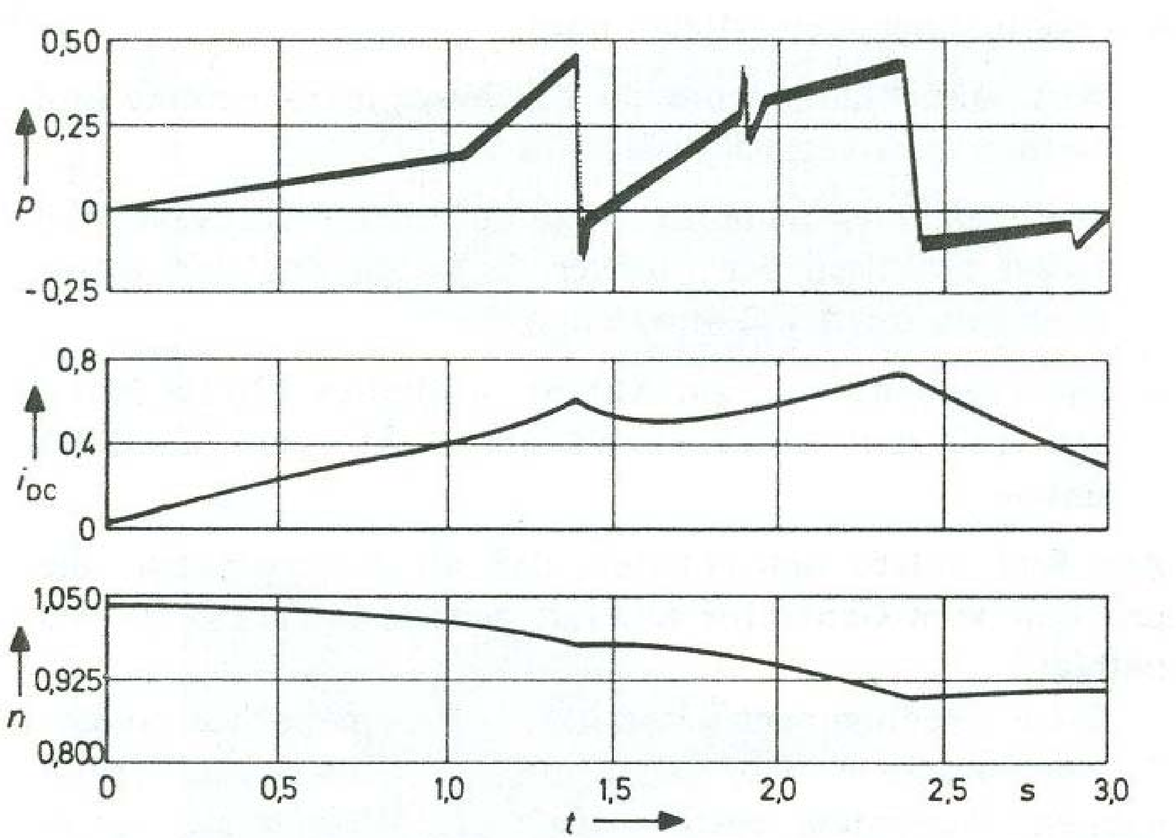
- The necessary coil current variation is achieved by a time-variation of the thyristor control angle, which leads to time-varying DC output voltage.

- During the transformer coil feeding the generator speed decreases. It must be sped up again after the plasma power cycle. Hence only a pulsed plasma operation is possible. The grid must only deliver the small motor power for generator acceleration and is not loaded by the big plasma pulse power.



Source: Simond, Canay: etzArchiv 10, 1988

Tokamak power cycle at *Ecole Polytechnique de Lausanne*



P/S_N
S_N = 220 MVA
Real power fed to the central transformer coil
Max. energy per pulse 100 MJ

I_{DC}/S_N
Varying DC current as transformer coil excitation current

n/n_N
n_N = 3600/min
Declining generator speed during the plasma power pulse

$$W_{kin}(t=0) = J\cdot(2\pi n_N)^2/2 = 389.4MJ$$

$$W_{kin}(t=3s) = J\cdot(2\pi\cdot 0.9 n_N)^2/2 = 315.4MJ$$

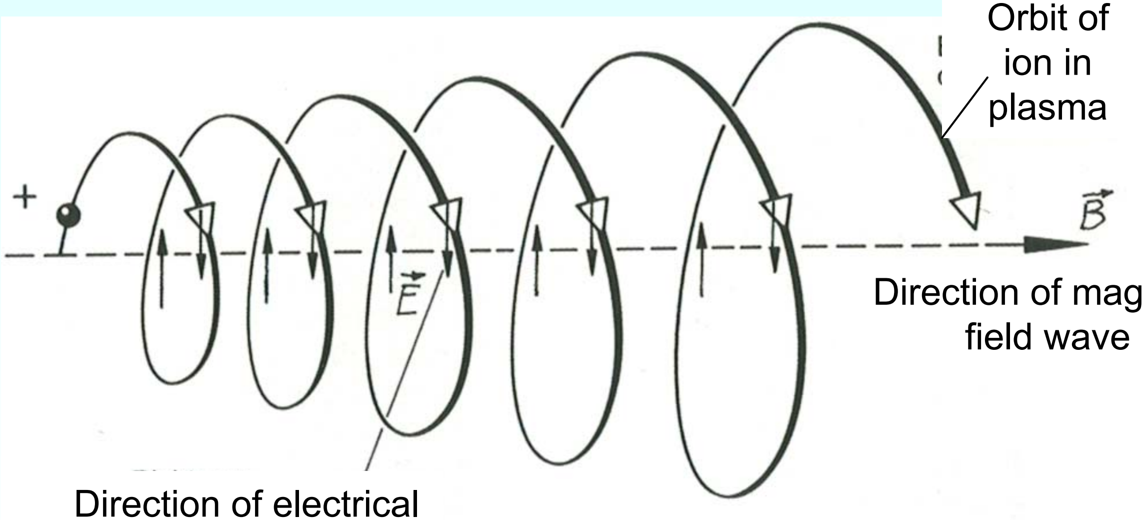
$$\Delta W_{kin} = 389.4 - 315.4 = 74MJ$$

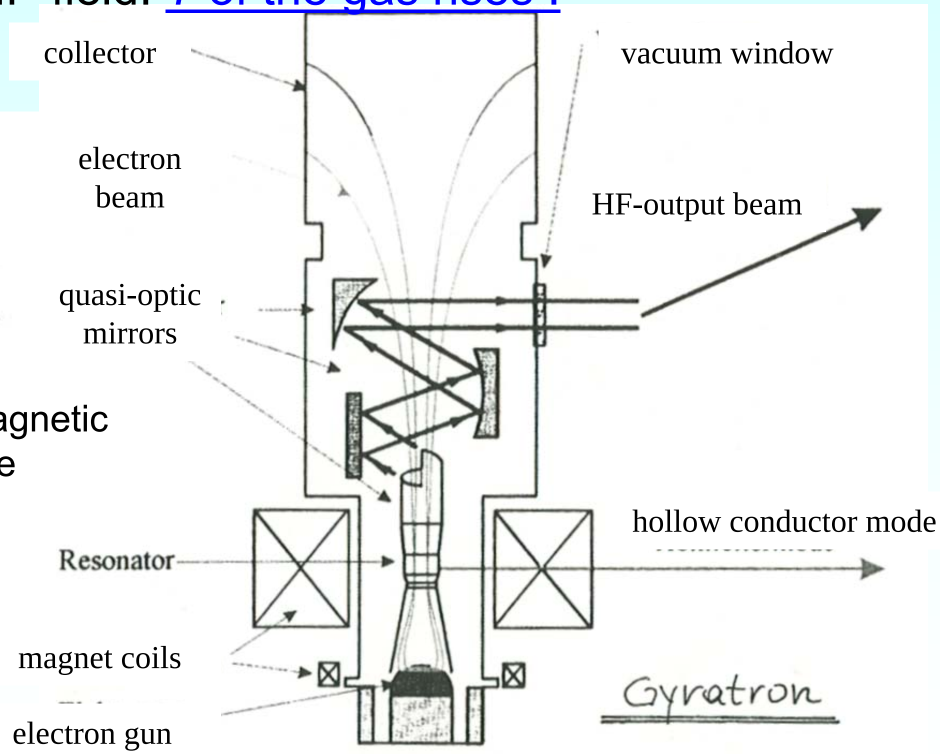
Four-pole generator:
U_N = 10 kV, S_N = 220 MVA,
f_N = 120 Hz, 3600/min, J = 5480 kg·m²

Source: Simond, Canay: etzArchiv 10, 1988

Stellarator: Principle of plasma heating

- Heating of plasma with HF electromagnetic wave (HF-Radiation)
- *Gyratron:* Radio tube for circular polarized waves (travelling field tube = interaction of a modulated electron beam with an electromagnetic field wave)
- Circular polarized field-vector rotates with cyclotron frequency of the magnetic field. Charged plasma particles also rotate (e. g. Tritium at B = 5 T with 25 MHz). These charged particles take over via $F = qE$ the kinetic energy from the HF-field: *T of the gas rises !*





Source: FSZ Karlsruhe, Germany

5.4 TOKAMAK and STELLARATOR

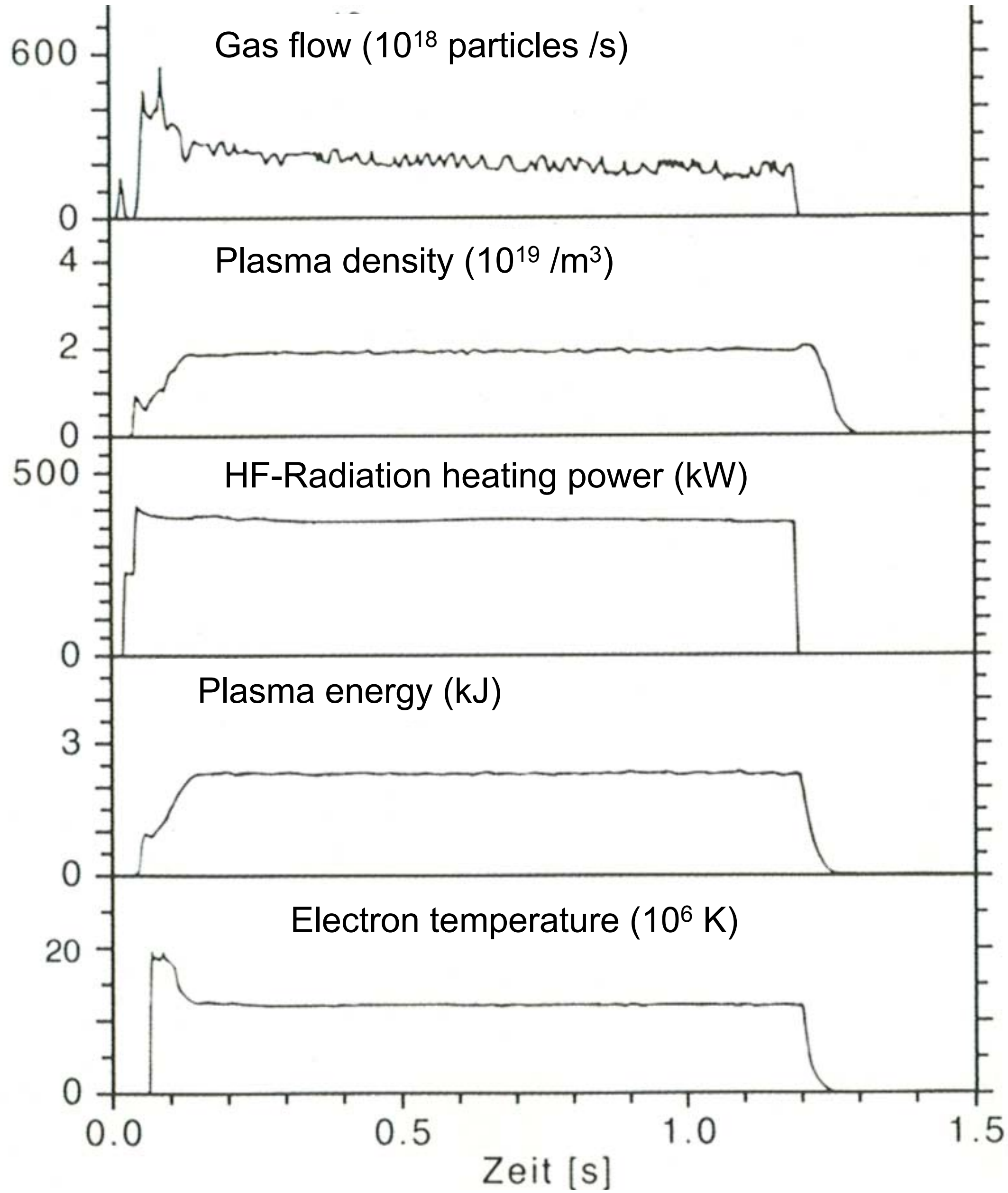
Stellarator: Variation of operational parameters with time

1. Switch-on of magnetic field (It can include charged particles without a plasma ignition)

2. Introduction of hydrogen gas for being heated with HF-waves (10 ... 50 MW-power range of the Gyratron)

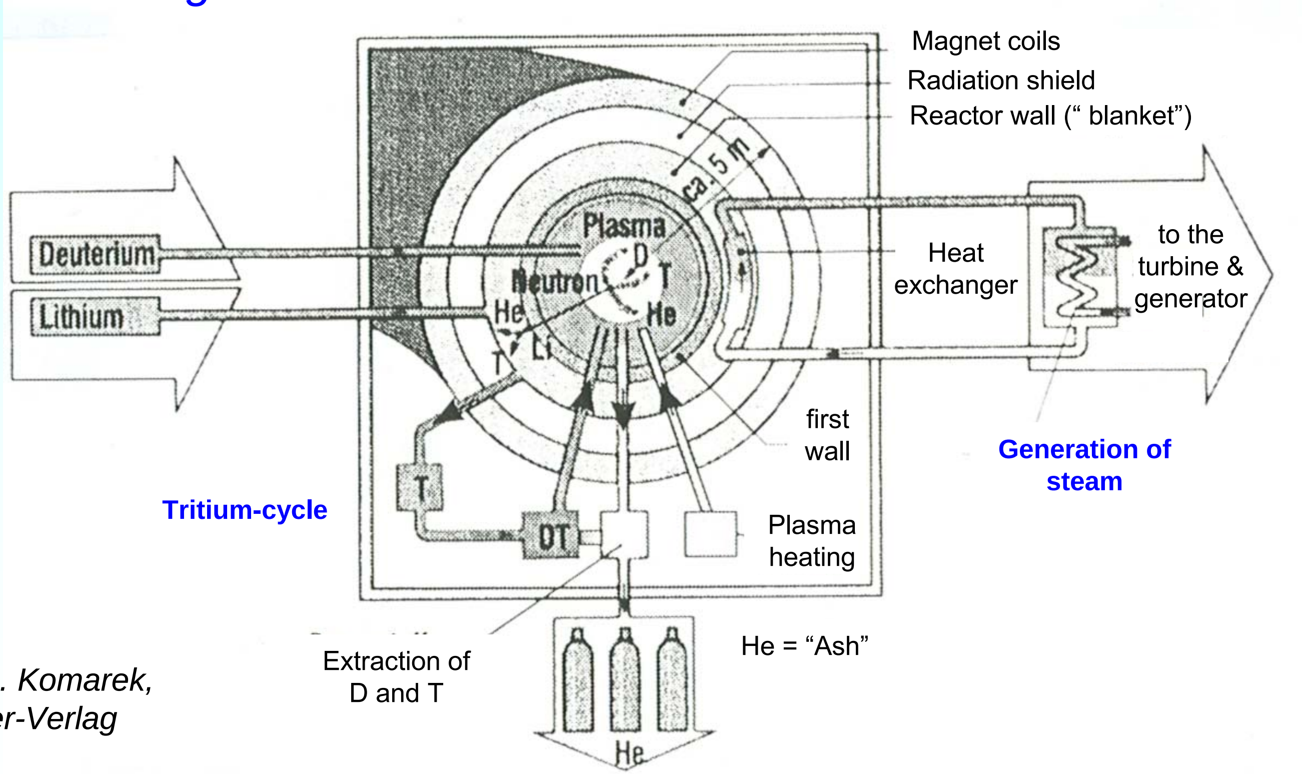
3. Plasma operation: Theoretically this period is unlimited (no pulsed operation as in *Tokamak*)

Source: FSZ Karlsruhe, Germany



Set-up of a fusion reactor

- For D-T-Fusion T must be „created" from Li: ${}^{6}_{3}Li + {}^{1}_{0}n \rightarrow {}^{4}_{2}He + {}^{3}_{1}T + 4.78 MeV$
- Li-"blanket" e. g. from Li-ceramic is bombarded with neutrons
- Sudden deceleration of neutrons in the Li-blanket: Braking energy heats up the blanket; blanket cooling with He-gas ⇒ heat exchanger ⇒ steam generation ⇒ steam turbine ⇒ *synchronous generator*

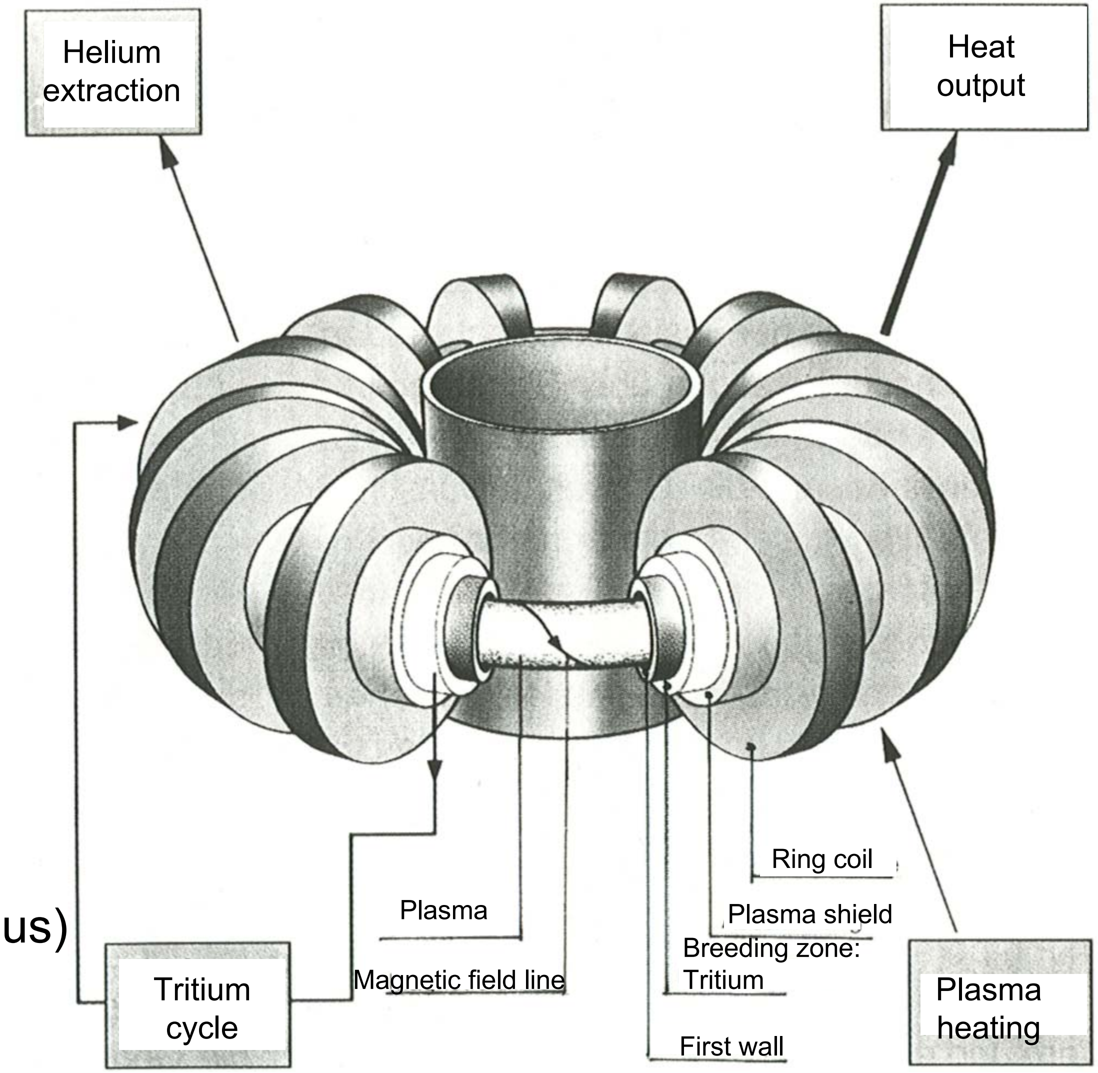


Source: P. Komarek, Teubner-Verlag

- Radiation shield necessary to absorb electromagnetic γ-radiation from the decelerated charged particles and to reduce the neutron flow

Data of a future *Tokamak*-fusion reactor

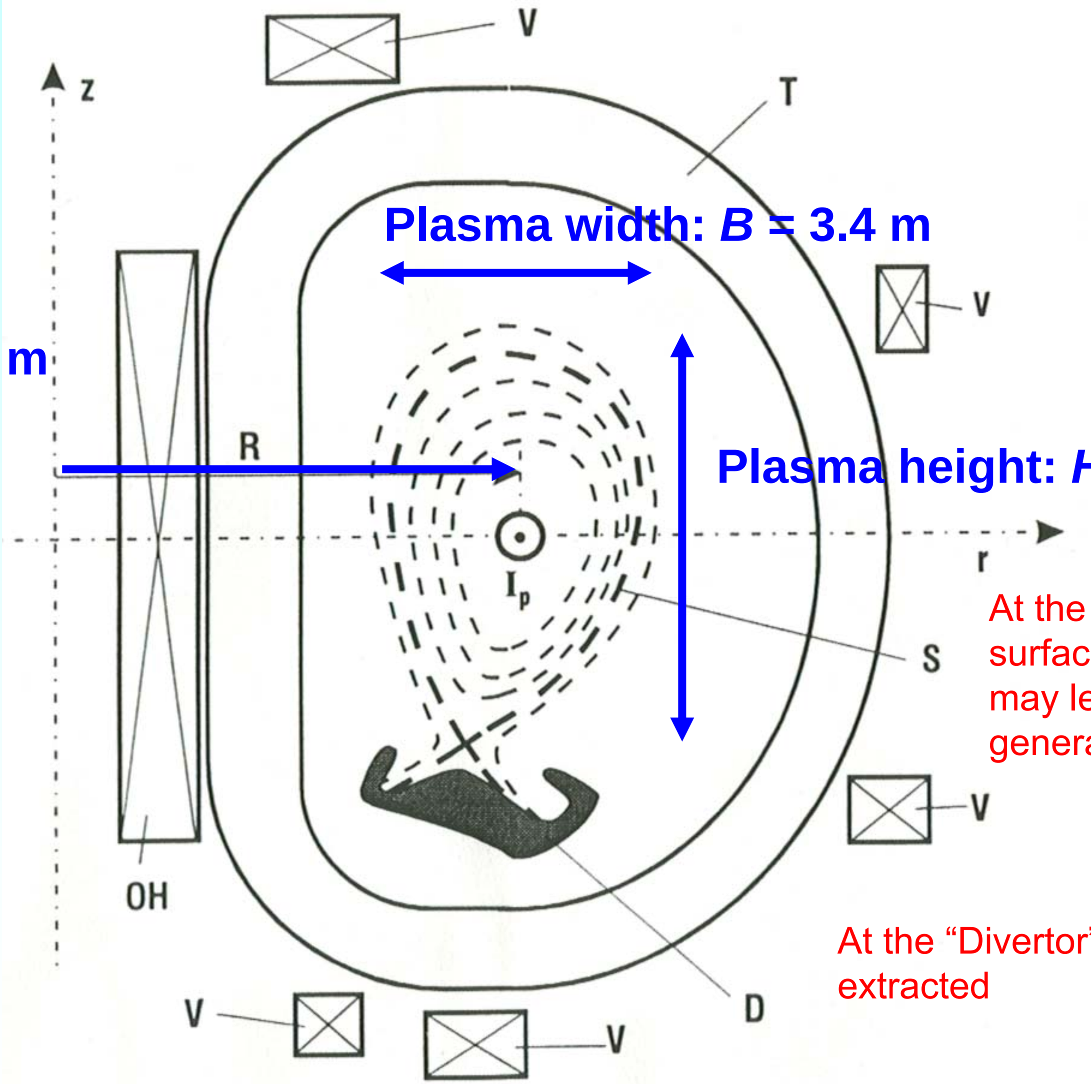
Radius:	17 m
Height:	23 m
Plasma radius:	7 m
Plasma height:	6 m
Plasma width:	3.4 m
Plasma volume:	760 m^3
Magnetic field:	9 T
Max. plasma current:	12 MA
Ignition heating:	60 MW
Wall load by neutron flow:	3 MW/m^2
Fusion power:	3 GW (continuous)



Source: FSZ Karlsruhe, Germany

Plasma dimensions of a *Tokamak* rating 3 GW

Source: P. Komarek, Teubner-Verlag



Plasma width: *B* = 3.4 m

Plasma radius: *R* = 7 m

Plasma height: *H* = 6 m

V: vertical field coils (poloidal field coils)

T: Torus “ring” coil as “D-Coil”

D: Divertor

S: Separatrix

I_p: Plasma current

OH: Central transformer winding

At the “Separatrix” magnetic field surface *S* the charged particles may leave the magnetic trap. S is generated by the V-coils.

At the “Divertor” D the He-nuclei are extracted

Energy of a future fusion reactor

- ***Ignition condition:*** Particle density: $n = 10^{20}/m^3$, temperature: T = 200 Mio. K
- **D-T-fusion:** particle density: $n_D = n_T = n/2, \quad n = 10^{20}/m^3$
- At **temperature**: T = 200 Mio. K is $\overline{\sigma_f v} =$ 10^{-21} m³/s for D-T-fusion
- **Reaction rate *R**:** $R^* = n_D \cdot n_T \cdot \overline{(\sigma_f \cdot v)} = (10^{20}/2) \cdot (10^{20}/2) \cdot 10^{-21} = 2.5 \cdot 10^{18} /(m^3 \cdot s)$
- **Energy balance per fusion reaction:** ${}^2_1D + {}^3_1T \rightarrow {}^4_2He + {}^1_0n + 17.62 MeV$

  with He: 3.52 MeV and n: 14.1 MeV for STABLE operation:

  W_α = 3.52 MeV covers losses, W_n = 14.1 MeV used as energy for steam generation

- **Fusion power** inside the reactor with V = 760 m³ plasma volume:

$$P_F = R^* \cdot W_n \cdot V = 2.5 \cdot 10^{18} \cdot 14.1 \cdot 10^6 \cdot 1.6 \cdot 10^{-19} \cdot 760 = \underline{\underline{4.29 \cdot 10^9}}\,\mathrm{W}$$

  Fusion reactor can create over 4.3 GW of power (theoretically).

- Real **fusion power** inside the reactor with V = 760 m³ plasma volume:
  - Due to the losses in the plasma only **3 GW of fusion power is realistic!**

Radioactivity in fusion power plants

- The steel construction is hit by the neutron flow and is activated.
- A special „low-activation"-steel is used. At the life end of a fusion power plant the scrap steel has to be kept in enclosures for about 200 years, and can then be used again.
- In general the waste in a fusion plant is by the factor 1000 less active than in a fission plant.
- Tritium is active with a half life of 12 years. Due to the Tritium-Lithium-cycle the amount of Tritium in a power plant is small (about 1 kg per reactor).
- Due to this „rather" small amount of activated material within a fusion power plant there is no need to evacuate anybody beyond the site boundary in the event of a worst case scenario.
- The contained "plasma" fuel density is so low, that never a self-ignition like a H-bomb might happen.

Standards for magnetic coils

*: *Stellarator* values are the lower ones

**: Alternating field only at *Tokamak*

- Torus-ring coils due to the high coil current in the **future only super-conducting (NbTi, 4.3 K)**

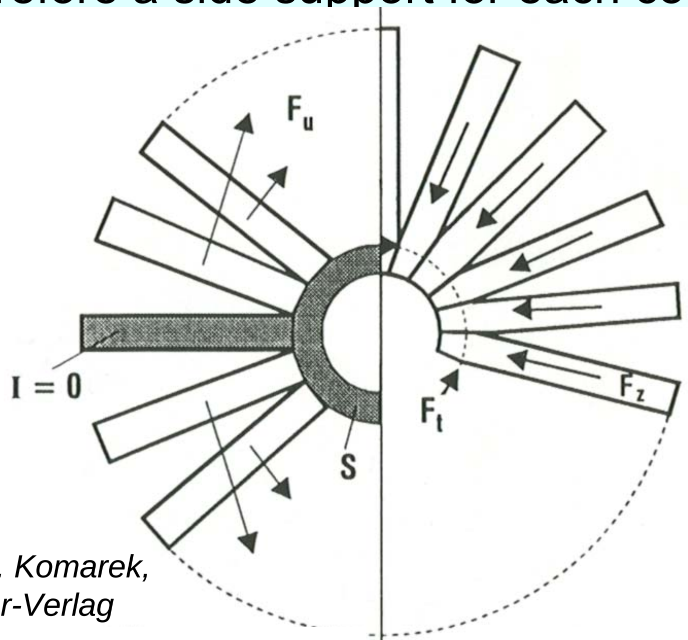
Max. magnetic flux density	12 - 15 T *
Winding current density	2 - 3 kA/cm²
Stored magnetic energy: - Per coil - In the total system	 1 - 2 GWs * 40 - 100 GWs
Main dimensions: - Torus ring coils - Poloidal field coils - Coil volume	 18 m x 10 m * bis ca. 30 m ∅ 10 m³ /coil
Magnetic AC fields: ** Transformer coil: pk-to-pk Torus ring coils: - Transients due to poloidal field coils - Transients due to plasma stop	 ± 13 T, t ≤ 10 T/s ± 2 T, t ca. 1 T/s ≤ 40 T/s ca. 20 ms
Discharge voltages	10 - 20 kV
Radiation: Neutron flux γ-radiation	 10^{23} m^{-2} 10^{8} Gy
Life span	> 20 years at full load

Source: P. Komarek, Teubner-Verlag

G_γ : Gamma radiation constant (Unit: J·m²/kg)

Demands for the coil system of the *Tokamak*

- Radial forces („Wheel forces“) at ring coils bigger at the inner part, because of $B_T \sim 1/r$
- Straight inner coil side („D-form“): Is lowering the coil bending stress
- Inner bearing cylinder takes over the inner pressure forces
- With one coil failing (quenching = no coil current) the torus arrangement tends to straighten. Therefore a side support for each coil is necessary.



Failure of one coil: I = 0 | Normal operation

F_u: circumference force

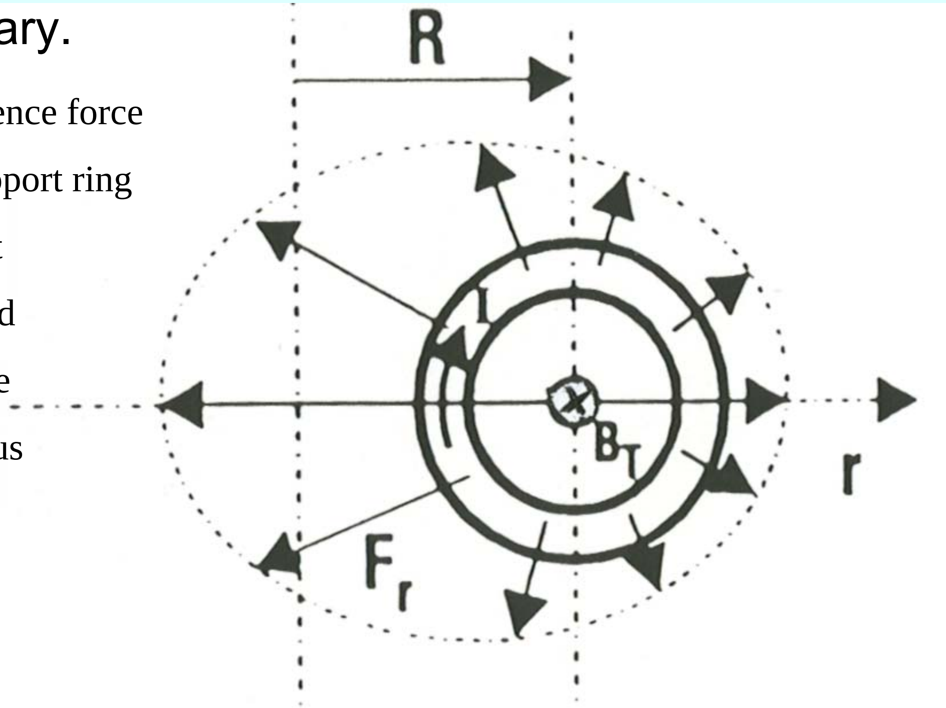
S: Central support ring

I: Coil current

B_T: Torus field

F_r: radial force

R: Torus radius



Source: P. Komarek, Teubner-Verlag

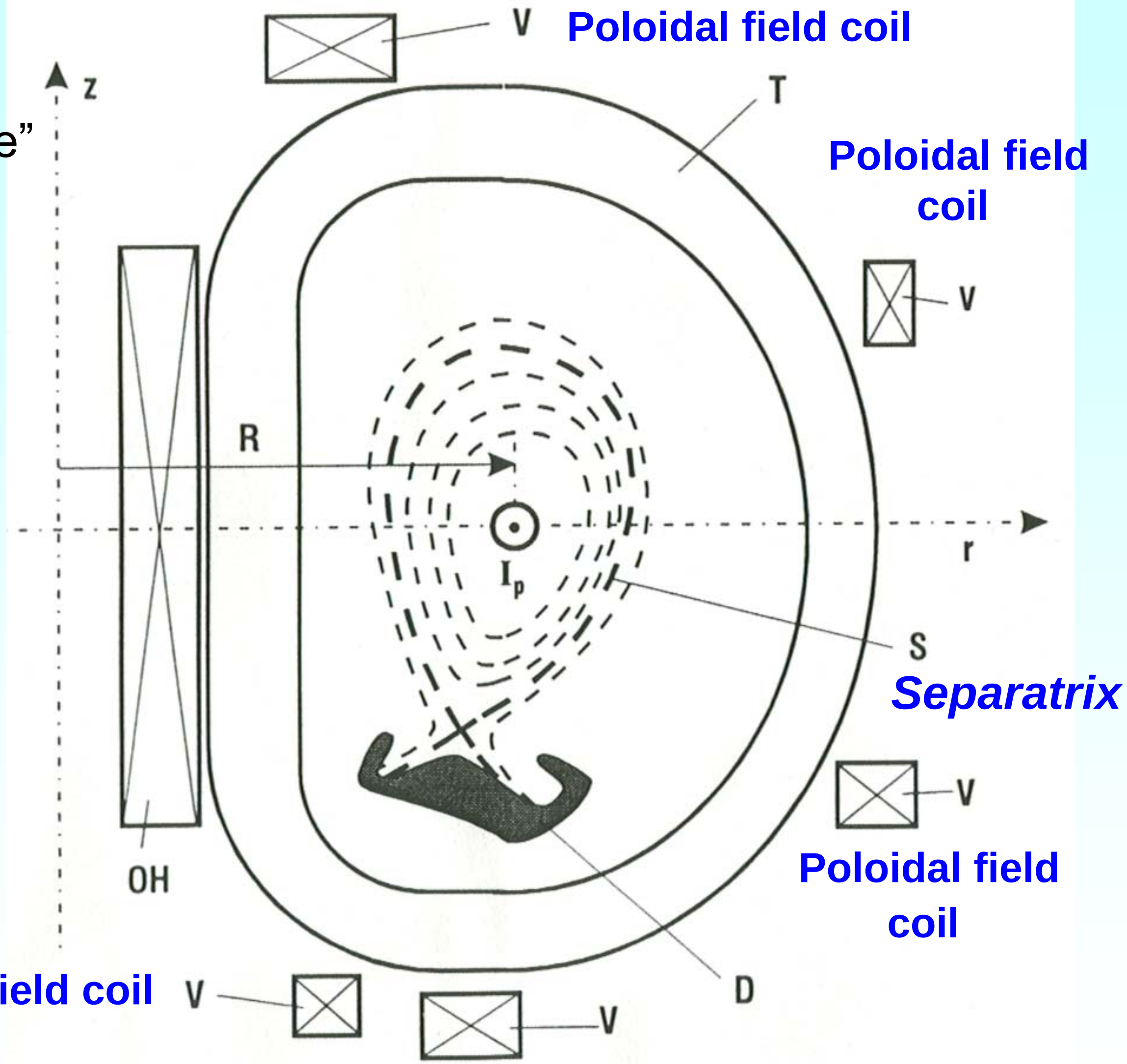
Operation of poloidal field coils in *Tokamak*

Poloidal field coils V excite a vertical magnetic field, that opens the “surface” of the screwed-up *B*-field lines (“Separatrix” surface). This is necessary to:

a) Remove of the low energy He-ions from the plasma („ash removal“)

b) Remove the impurities from the plasma into the divertor chamber D.

V: vertical field coils (poloidal field coils)
T: Torus “ring” coil
D: Divertor
S: Separatrix
I_p: Plasma current
OH: Central transformer winding



Source: P. Komarek, Teubner-Verlag

Summary:
TOKAMAK and STELLARATOR

- TOKAMAK needs central transformer for voltage induction
- Screwing of field lines by induced plasma current
- Plasma current losses for heating up the plasma
- Pulsed operation of the transformer coil for voltage induction
- STELLARATOR helical coils or especially shaped torus coils for screwing of field lines
- Heating of the plasma by HF high power Gyratron
- TOKAMAK prototypes are far more developed than the STELLARATOR
- Both are still prototypes

5. Fusion research

5.5 Plasma experiments – Status of research

Plasma experiments with *Tokamak* arrangements

Copper coils:

- ASDEX (Axial symmetrical divertor experiment) *Garching, D*, since 1990, investigation of outer plasma layers, especially for the divertor, Cu ring coils used
- JET (Joint European Torus) *Culham, UK*, since 1983, Plasma operated near ignition, Cu ring coils used with 300 MW excitation, 25s-pulses, I_p = 7 MA

  1 MW fusion power 1991, „Zero reactor condition“
  16 MW fusion power 1999 .
- TFTR (Tokamak Fusion Test Reactor) *Princeton, USA*, 12 MW fusion power 1995, Cu ring coils used
- JT60 (Japanese Torus) *JAERI-Institute, Naka, J,* Cu ring coils used

Superconducting coils:

- TORE SUPRA, *CEA-Institute, Cadarache, F,* since 1988, testing of superconducting ring coils in continuous duty
- T15 (Torus 15), *Kurchatov-Institute, Moscow, R,* 1988, testing of superconducting coils
- TRIAM-1M, *Kyushu-University, J,* testing of superconducting coils

Superconducting torus coils at TORE SUPRA and T15

	TORE SUPRA	T 15
Superconductor	NbTi	Nb_3Sn
Matrix	Cu/CuNi	Cu
Dimensions [mm]	2,8 x 5,6	18 x 5
Ratio Superconductor vs. Copper α	2 : 1	4 : 1
LHe-channel cross section in % of conductor cross section	28,6	13
Current at 9 / 9.3 T [kA]	1,4	5,6
Critical current [kA] / Field [T] / Temperature [T]	1,4/9/4,2	10/8,5/4,2
Transient field at plasma stop: ΔB [T] during time τ [s]	0,6 10 - 20	0,7 20
Conductor cross section		

Source:

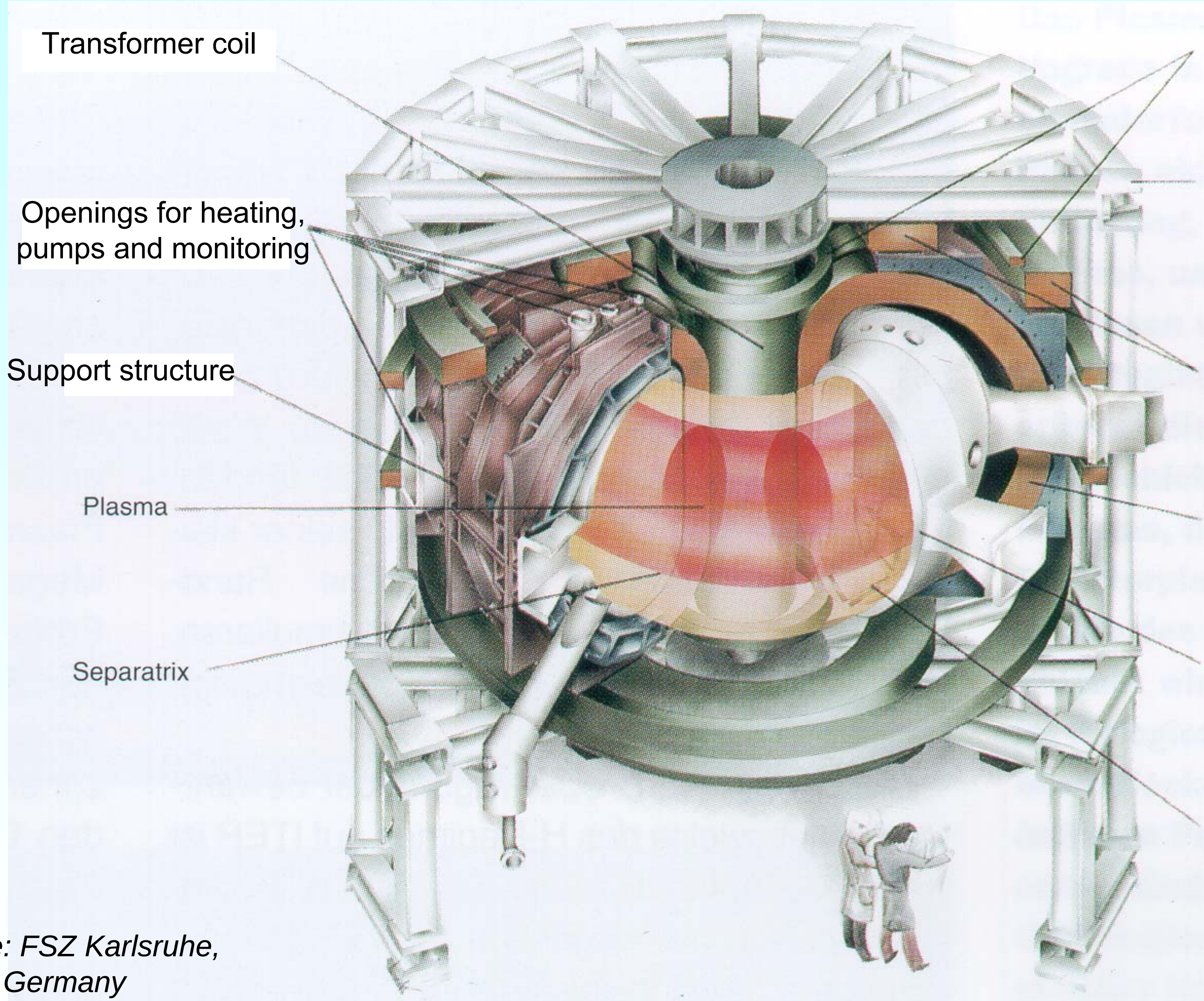
P. Komarek, Teubner-Verlag

Superconducting torus coils at TORE SUPRA and T15

	TORE SUPRA	T-15
Torus radius	2,25	2,43
Plasma radius *a* [m] a [m]	0,70	0,70
Plasma current I_p [MA]	1,7	1,4 - 2,3
Plasma active time [s]	30 - 120	5
Magnetic flux density in the plasma centre B_T [T]	4,5	3,5-5,0
Max. magnetic flux density at the superconductor B_m [T]	9,0	6,5-9,3
Average coil current density [MA/m^2]	40	25-36
Torus ring coil shape	circle	circle
Average torus ring coil diameter [m]	2,6	2,59
Number of torus ring coils	18	24
Stored magnetic energy in the coils [MJ]	600	380-790

Source: P. Komarek, Teubner-Verlag

Schematics of Tokamak ASDEX Upgrade-fusion experiment



Source: FSZ Karlsruhe, Germany

Tokamak ASDEX Upgrade-Fusion experiment

Radius/Height: 5 m / 9 m

Plasma radius: 1.65 m

Plasma volume: 14 m^3

16 Torus copper coils

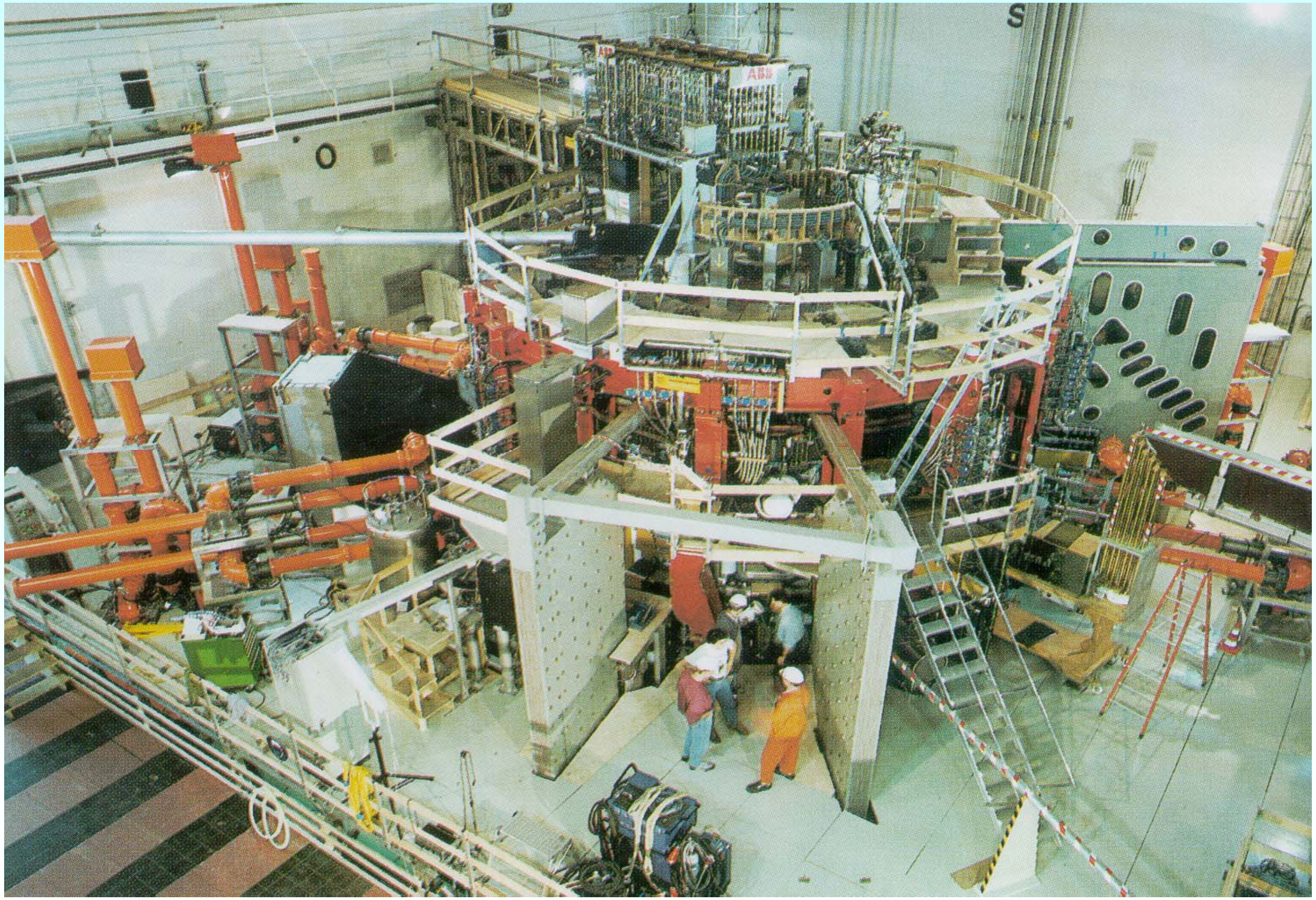
Coil current 84 kA

B_T = 3.9 T

Plasma current: 2 MA

Plasma operation time: 10 s

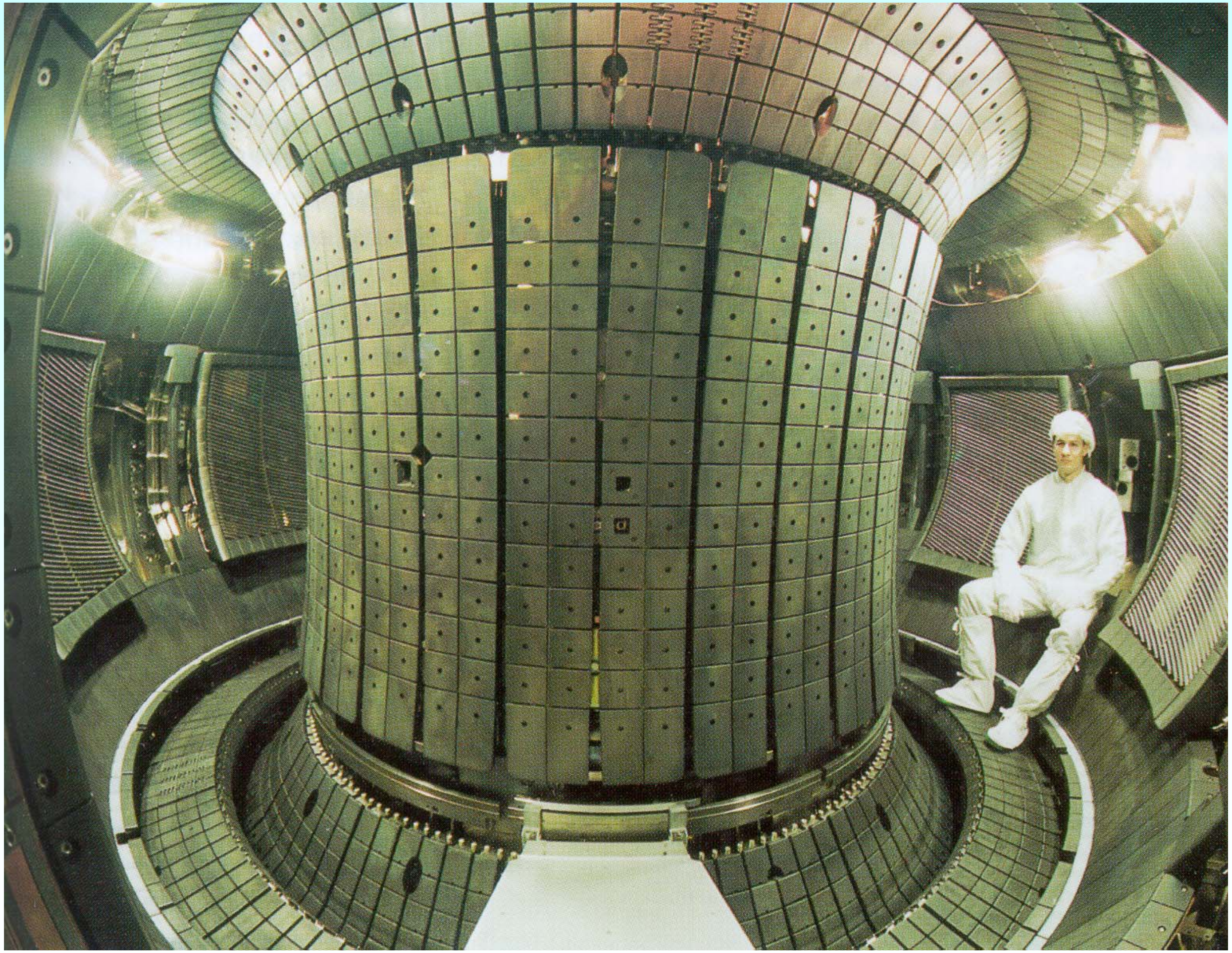
Heat power: 10 MW



Source: FSZ Karlsruhe, Germany

Plasma vessel of ASDEX Upgrade-experiment

Graphite tiles as casing protect the vessel wall from radiation of the hot plasma and the plasma from impurities (particles from the vessel wall)



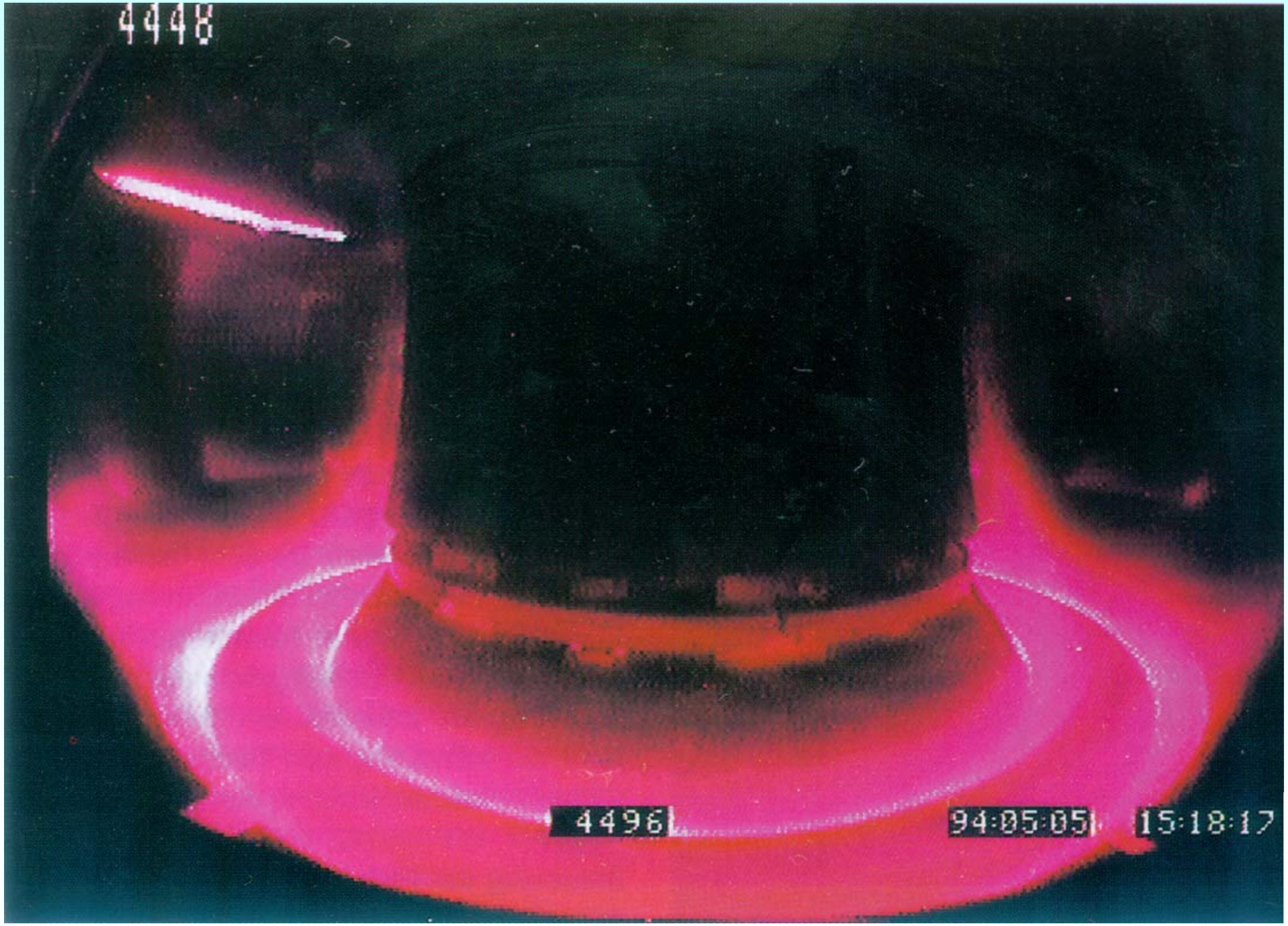
Source: FSZ Karlsruhe, Germany

"Burning" (hot) plasma in ASDEX Upgrade-experiment

Left side we have a pellet (frozen Deuterium) shot into the plasma with about 1.2 km/s, where it evaporates.

Shooting rate: max. 80 pellets/second

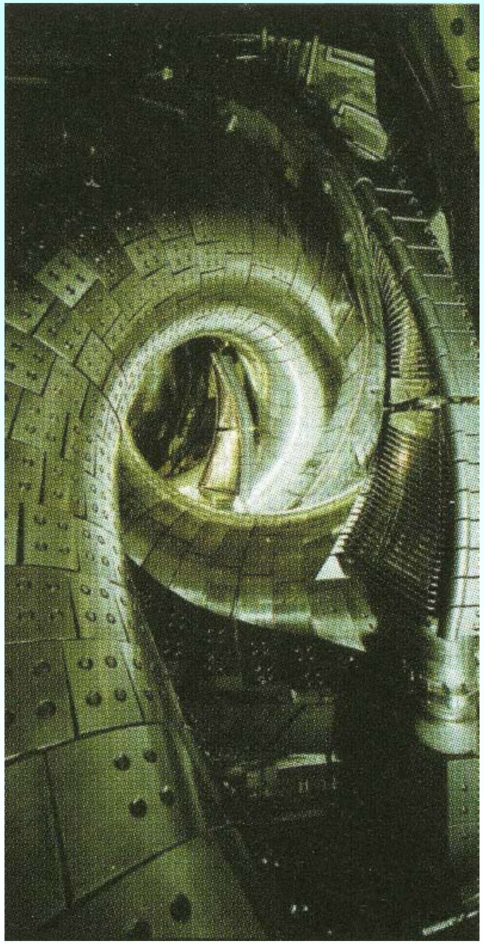
Pellet flow rate controls the plasma density.

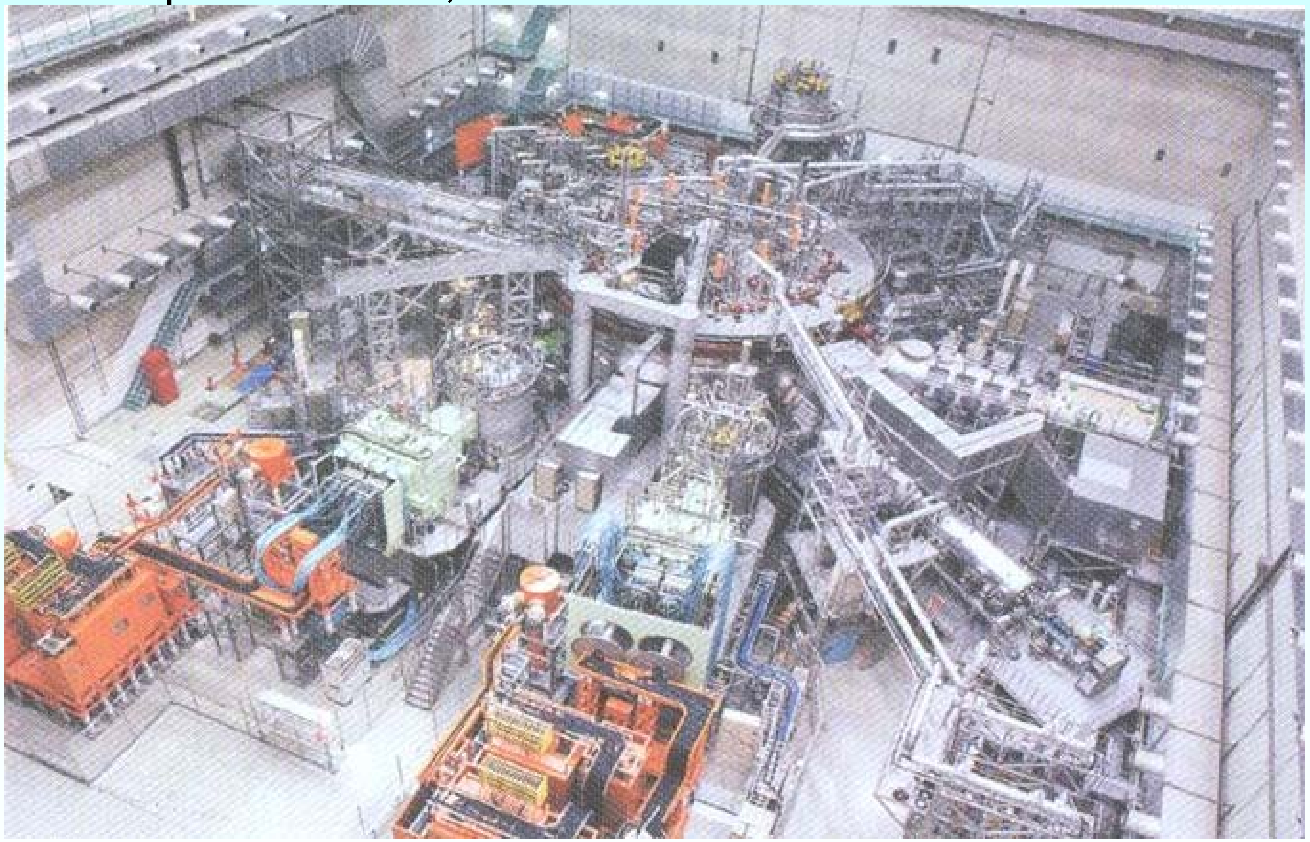


Source: FSZ Karlsruhe, Germany

Plasma experiment (*Stellarator*)

- LHD (Large helical device), *Nat. Institute for Fusion Science, Nagoya/Toki, J*, two helical coils, ALL coils made of NbTi-superconductor, since 1997





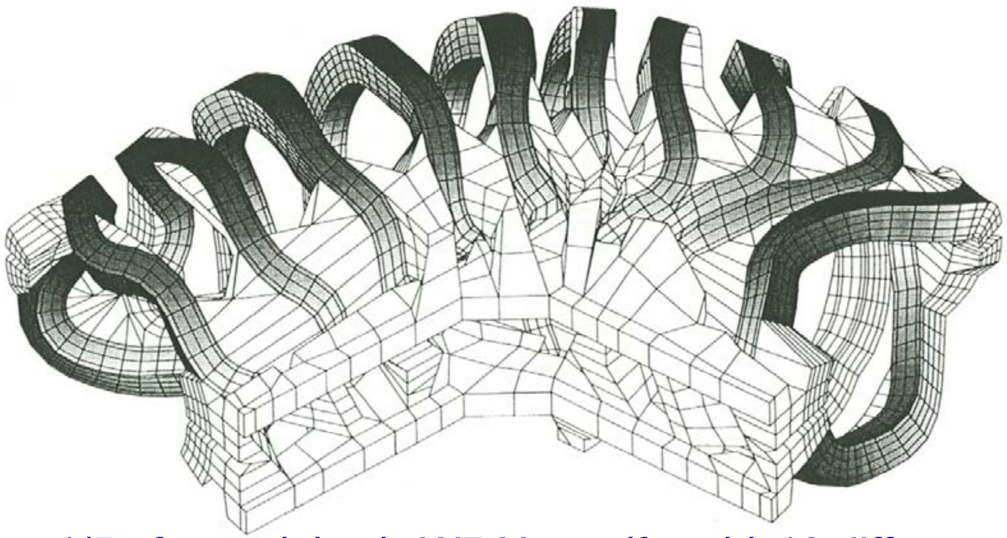
Arrangement of the two SC helical coil systems

Source: Hitachi, Company-folder, Japan

Plasma experiment (*Stellarator*)

- *Wendelstein* 7-AS (Advanced Stellarator), *Garching, D*, since 1988, Cu coils
- LHD (Large helical device), *Nat. Institute for Fusion Science, Nagoya/Toki, J*, two helical coils, ALL coils made of NbTi-superconductor, since 1997
- under construction: *Wendelstein* W7-X, *Greifswald, D*, NbTi-coils (6 T in coil, 3 T in plasma)

Average torus radius	$R = 5{,}5$ m	
Average coil radius	$r_c = 1{,}5$ m	
Average plasma radius	$a = 0{,}5$ m	
Flux density in the plasma centre	$B_T = 3$ T	
Flux density at the coil conductor	$B_{max} = 6$ T	
Turns per ring coil x coil number	$w = 120 \times 50$	
Rated coil current	$I = 14{,}8$ kA	
Stored magnetic energy	$Q_s = 620$ MWs	
Coil inductance	$L = 5{,}66$ H	
Current change rate	$dI/dt \leq 50$ A/s	
Time constant for rapid de-excitation	$\tau = 3$ bis 5 s	

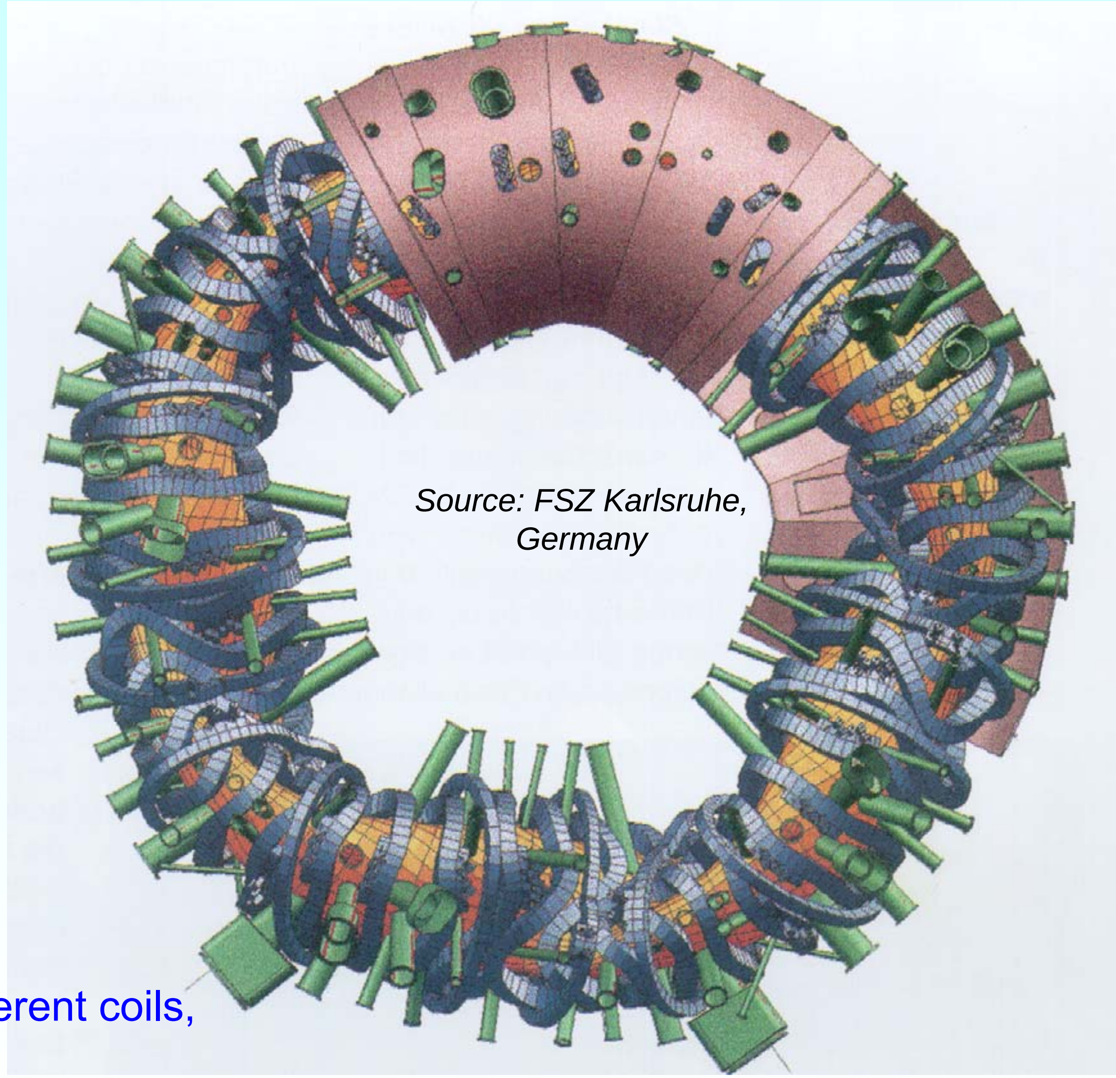


1/5 of *Wendelstein* W7-X, *Greifswald*, 10 different coils, 5 x 10 = 50 coils in total

Source: P. Komarek, Teubner-Verlag

Coil arrangement for WENDELSTEIN 7-X (Stellarator)

- Non-planar coil arrangement repeats 5 times at the circumference
- 50 non-planar NbTi coils
- Additional 20 planar NbTi ring coils for experimental use
- One common cryostat for all 70 coil (4 K, LHe I)
- Inner wall of cryostat (yellow) is at the same time the wall of the plasma vessel
- Different connections for measurement, heating and pumps lead through the thermally insulated cold coil area.



Wendelstein W7-X, *Greifswald*, 10 different coils,
5 x 10 = 50 coils in total

Welding of the plasma vessel for WENDELSTEIN 7-X (Stellarator)

2015:

First He-plasma operation for 0.1 s at 10^6 K:
- 1 mg He-gas,

-1.8 MW HF ignition power

Next step: 2016: H_2-plasma

Source: VDI nachrichten 18.12.2015

Source: energiewirtschaft, 105 (2006), no.5, p. 22

In construction: Fusion experiment ITER (*Tokamak*)

- Preparation: LCT (Large Coil Task): 1976 – 1988: 6 superconducting D-coils (3 from USA, one from *Japan, Switzerland, EU*), tested in *Oak Ridge Nat. Lab*., USA.
- ITER (International Tokamak Experiment Reactor): All coils superconducting
  Goal: fusion power / heating power = 10 („near ignition“ operation)

18 D-coils 8.0 x 12.5 m excite a torus field 5.7 T in the plasma, 11.8 T at the coil surface, so Nb_3Sn is needed.

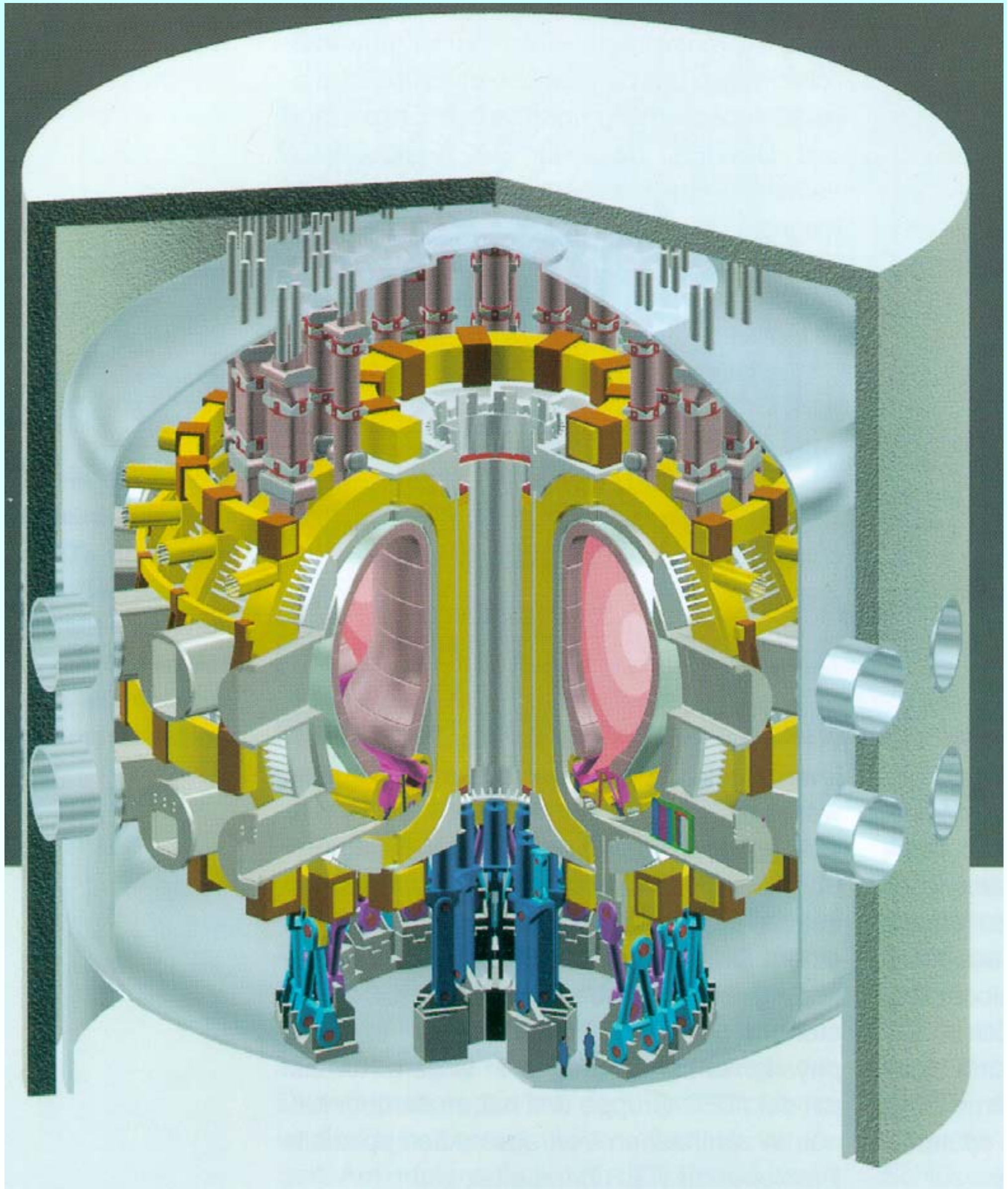
- Central transformer coil (∅ 4 m, H = 10 m): 13.5 T (J_c = 700 A/mm^2,12 T), Nb_3Sn
- 6 poloidal field coils (max. 24 m diameter) excite 5 ... 6 T, so NbTi possible
- Reactor: Ignition time: 60 s, burning time: 1000 s, switch-off time: 60 s

1. Phase:

a) Building of a central transformer coil (Central Solenoid CS): *Japan* + *USA*
b) Testing of CS: in *JAERI / Naka, J*, 45 kA nominal current (2000), 0.5 T/s = dB/dt
c) Building of prototype-torus coil (63 tons !): *Russia* + *EU*, 63 kA nominal current
d) Testing in *Karlsruhe*: Result: Nominal current 80 kA operation OK (2001). External field testing by the transformer coil (2002)

2. Phase: 2003/2006: Location *Cadarache/F* negotiated and engineering phase for building of the reactor (estimated 3 Mrd. Euro)

Layout of ITER-*Tokamak*-Fusion experiment



Plasma: in profile (right side): pink

Blanket: Outside of plasma: red

Plasma vessel with mounted divertor plates: violet

Magnets (superconducting): yellow

Cryostat for magnets and electromagnetic shielding: grey

Source: FSZ Karlsruhe, Germany

Roadmap for industrial use of fusion

Experiment ITER: Data

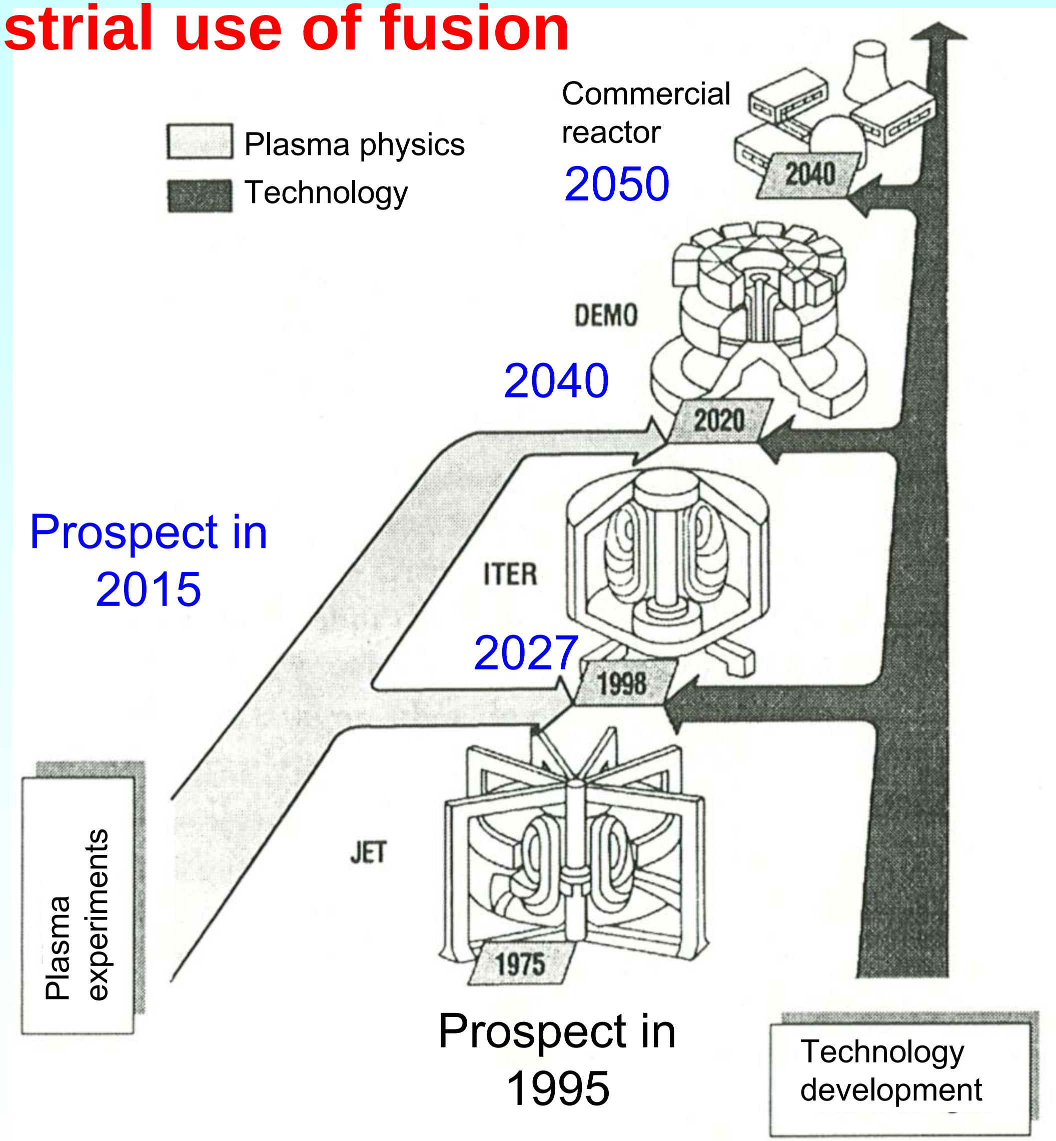
Radius:	18 m
Height:	36 m
Plasma radius:	8.14 m
Plasma height:	9.4 m
Plasma width:	5.6 m
Plasma volume:	2000 m^3
Magnetic field:	5.7 T
Max. Plasma current:	21 MA
Ignition heating:	100 MW
Aimed temperature:	170 Mio. K

Load by neutrons *): 10^{25} neutrons/m^2

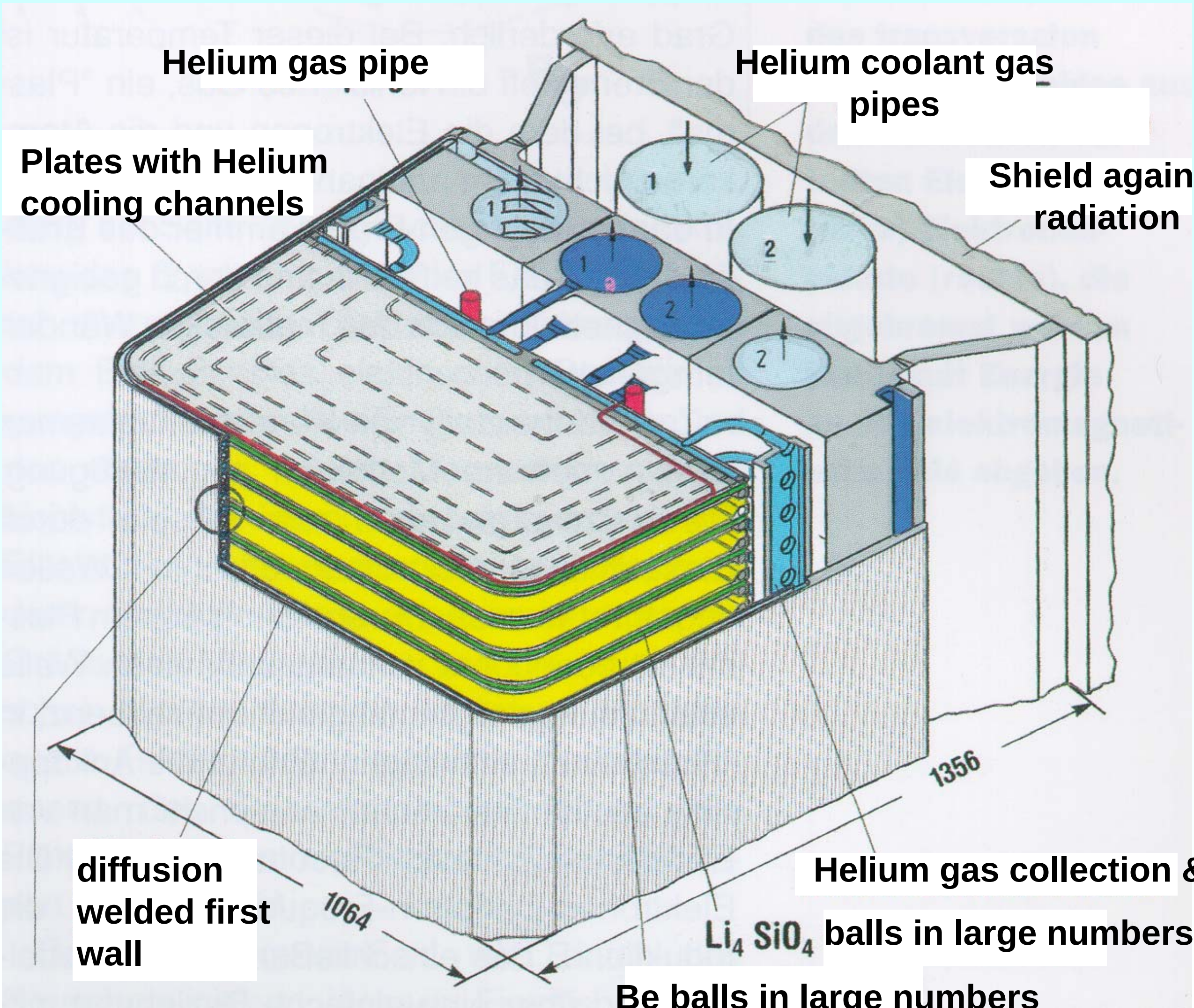
Fusion power: 500 MW (for 400 … 1000 s)

*) future reactors will have about 10-times more load

Source: P. Komarek, Teubner-Verlag

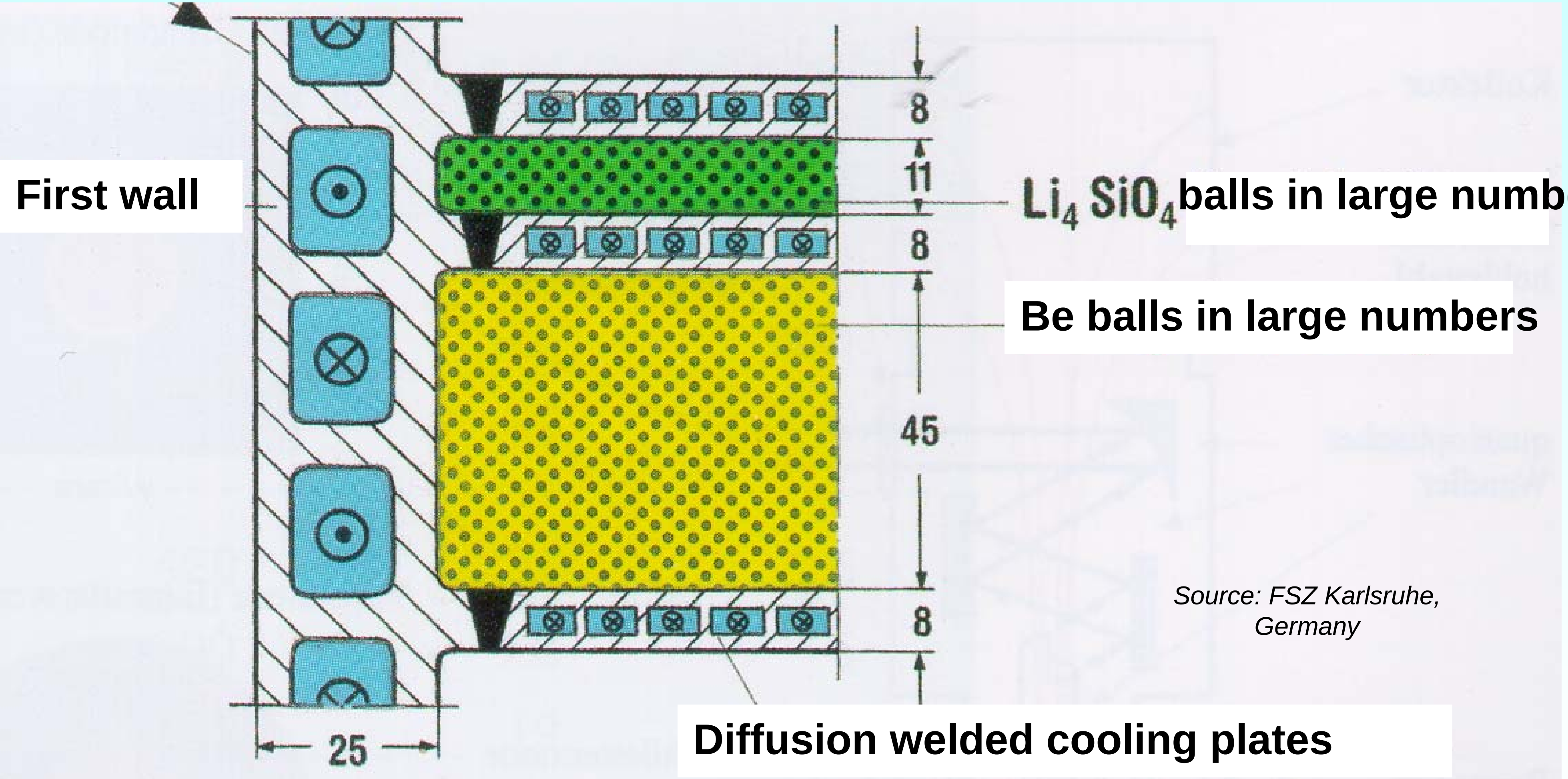


Outside segment of Li-blanket and He-cooling



Source: FSZ Karlsruhe, Germany

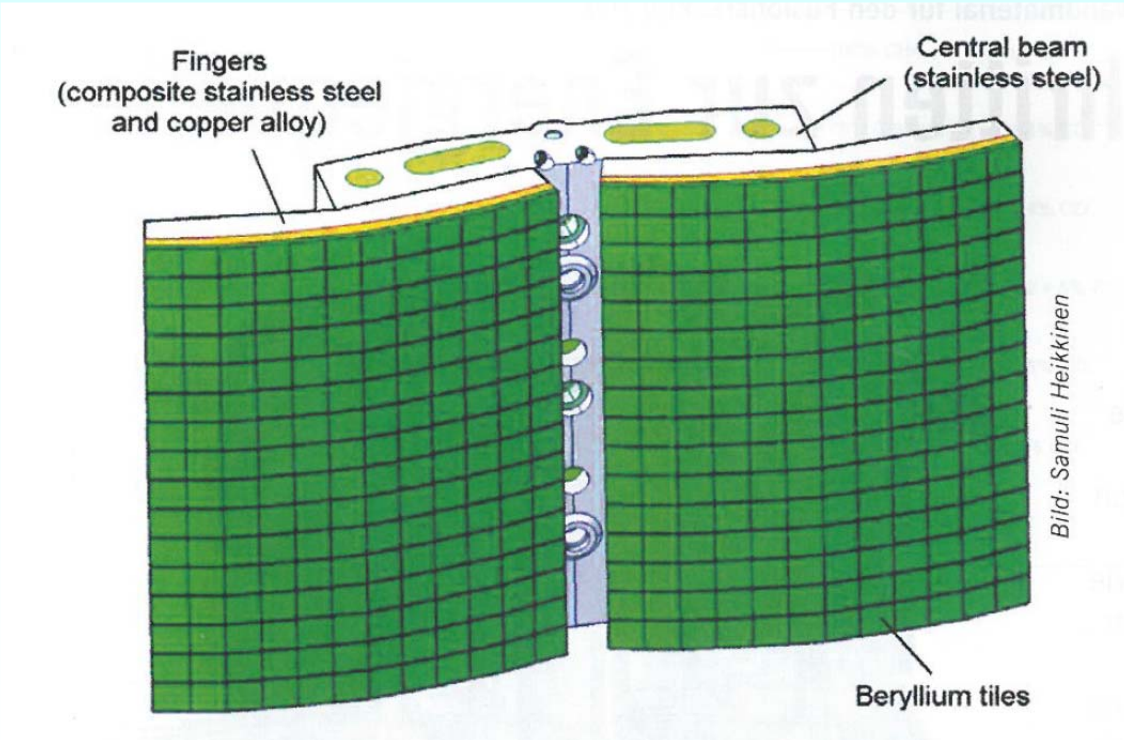
Cross section of Blanket-outside segment



Source: FSZ Karlsruhe, Germany

Fist wall design for ITER

First wall: Expected peak power input: 4.7 MW/m^2
Stainless steel plates are combined with Cu-Cr-Zr-Alloy layer and Be-tiles at max. 1040°C and high pressure (Hot Isostatic Pressing HIP) as a composite



Cu-Cr-Zr-Alloy:
- High thermal conductivity
- High ductility
- Resistence against neutron bombardement:
  - material gets less brittle than others
  - less surface wall debris dust generated

Pulsed Tokamak operation:
- Big number of temperature cycles
- Big temperature differences within the wall

Aim: No cracks
No material creeping
No reduction of elasticity

to cope with cycling thermal expansion and shrinking

Source: BWK 67 (2015), no. 12, p. 26

New technologies of electric energy converters and actuators

Summary:
Plasma experiments – Status of research

- National projects now focused as international projects to save money
- TOKAMAK project ITER at *Cadarache*, *France*, under (slow) progress
- STELLARATOR national project at *Greifswald*, *Germany*
- Zero-load fusion condition was already reached at *Culham*, *UK*
- Huge efforts in the past to develop the superconducting coils
- Heat exchanger and wall construction still open issues
- ITER shall allow zero-reactor condition with superconducting coils in 2027
- Further progress needed for a commercial power plant
- Commercial use not before 2050 expected

That´s all, folks !