

A detailed 3D cutaway rendering of a tokamak fusion reactor, showing the complex internal structure including the central solenoid, toroidal field coils, and various diagnostic and support systems. The image is rendered in a light gray color scheme.

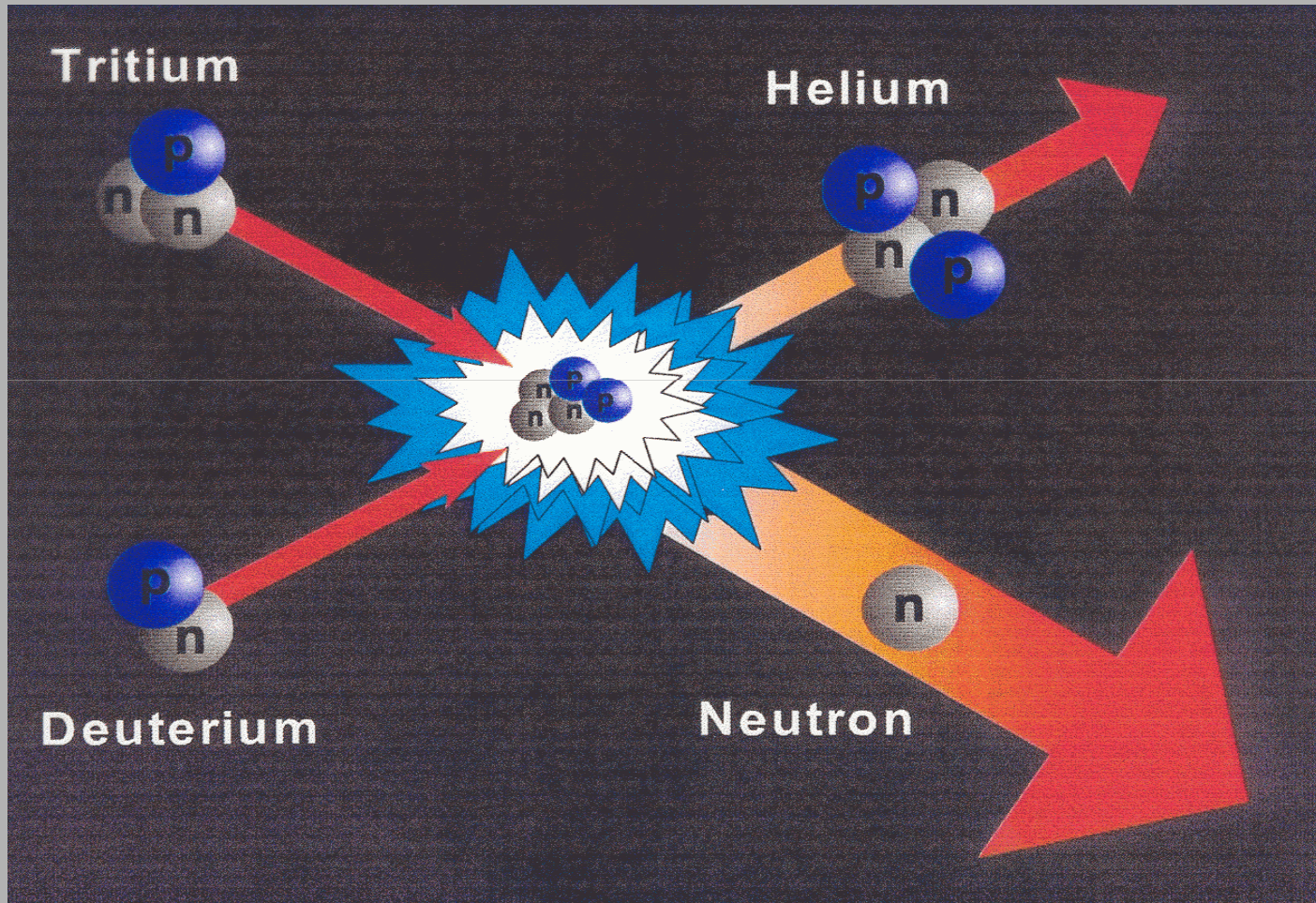
Tokamak-Konfiguration und ITER

Martin Droba

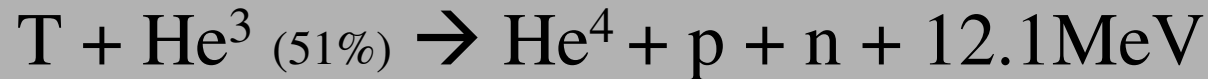
Inhalt

- Fusion
- Magnetischer Einschluss
- Stellarator
- Tokamaks
- ITER

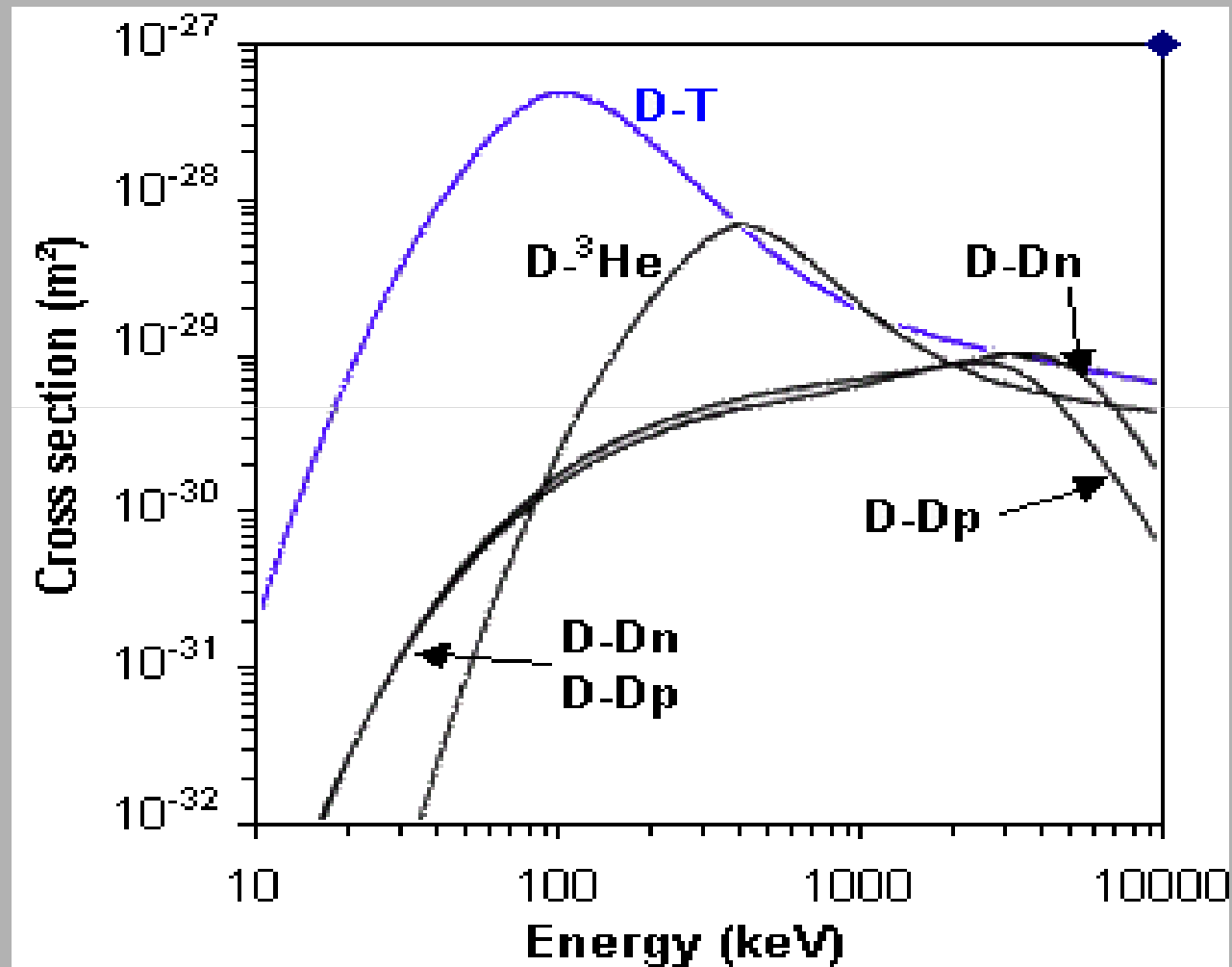
Fusion



Reaktionen



Wirkungsquerschnitte



Lawson-Kriterium

$$n_{DT} \cdot T_i \cdot \tau_E > 6 \cdot 10^{21} \text{ m}^{-3} \text{ keV s}$$

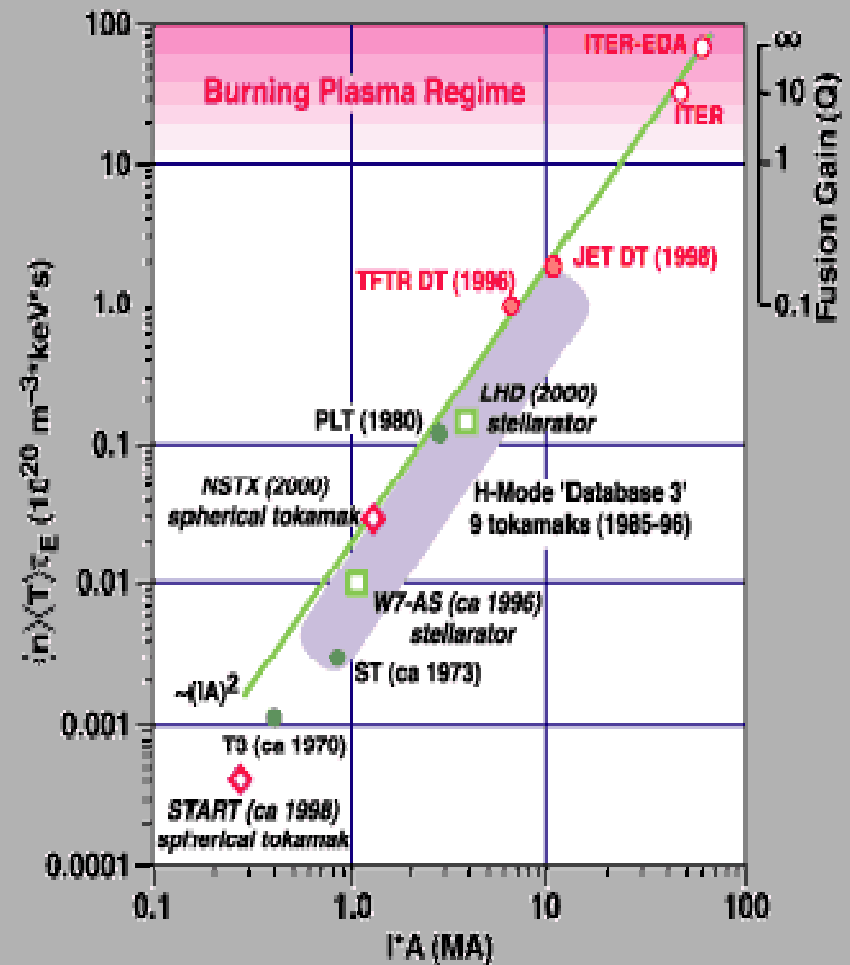
τ_E – Energieeinschlusszeit

$$\tau_E = W/P_{\text{loss}}$$

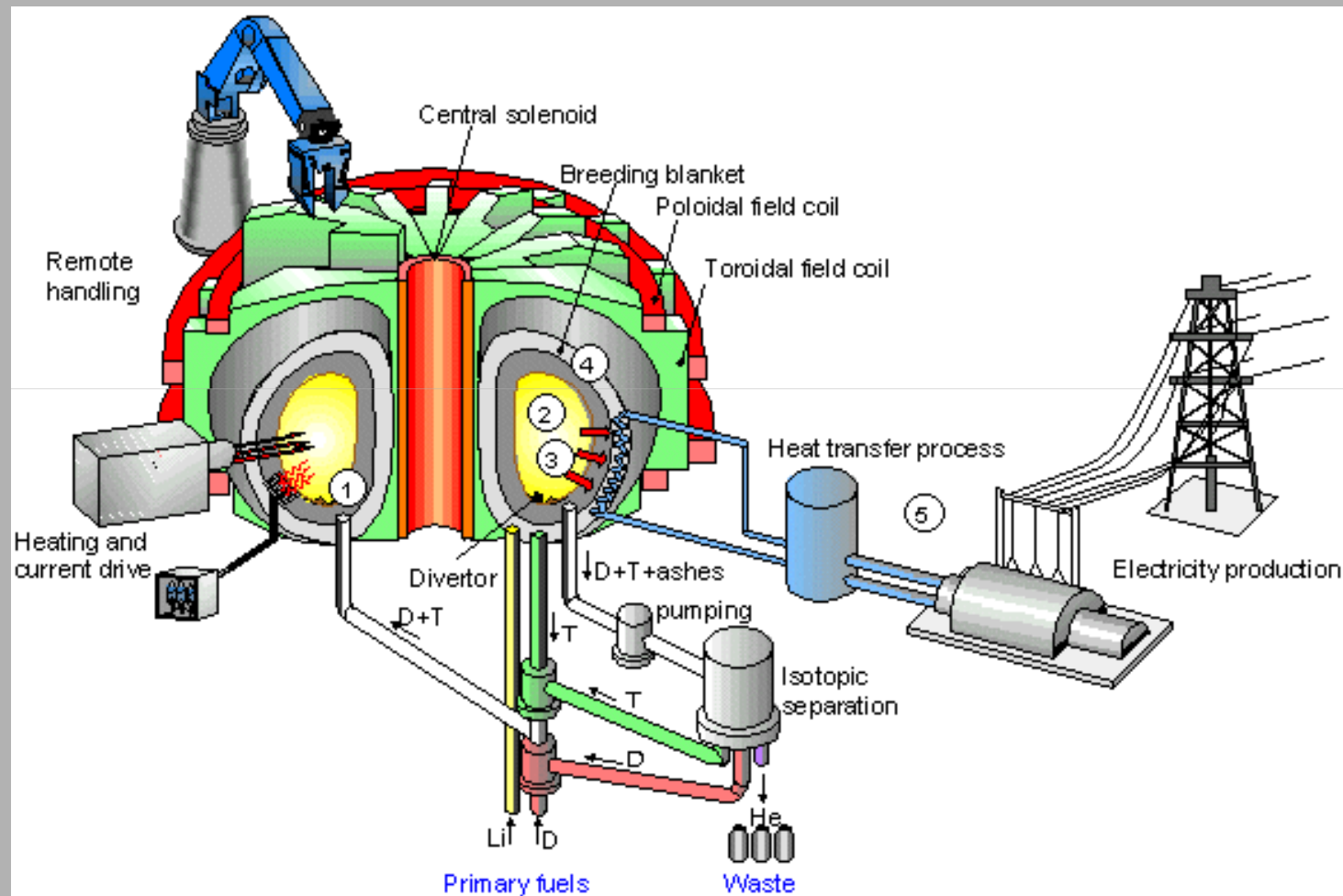
(W – Eingeschlossene Energie)

Fusionsleistung $P_F = n_D n_T \langle \sigma v \rangle \Delta E$

Q = Fusionsleistung/Heizleistung



Reaktor

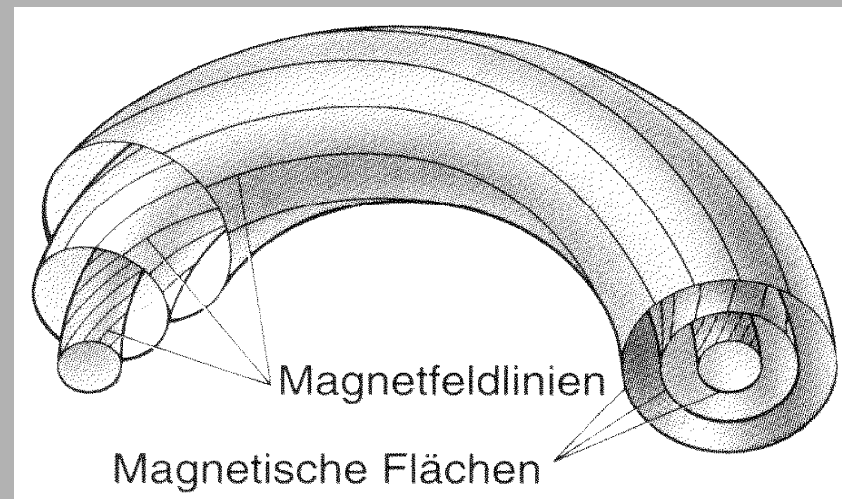
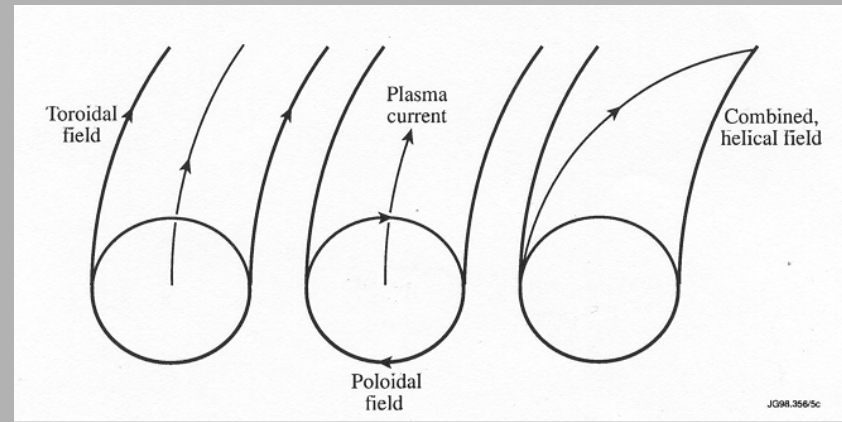


Möglichkeiten

- Gravitation (Sterne)
- Inertial Fusion (Trägheitsfusion)
- Magnetischer Einschluss
 - Tokamaks
 - Stellarators
 - RFP(Reversed Field Pinch)
 - Spheromak
 - Spiegel-Anordnung

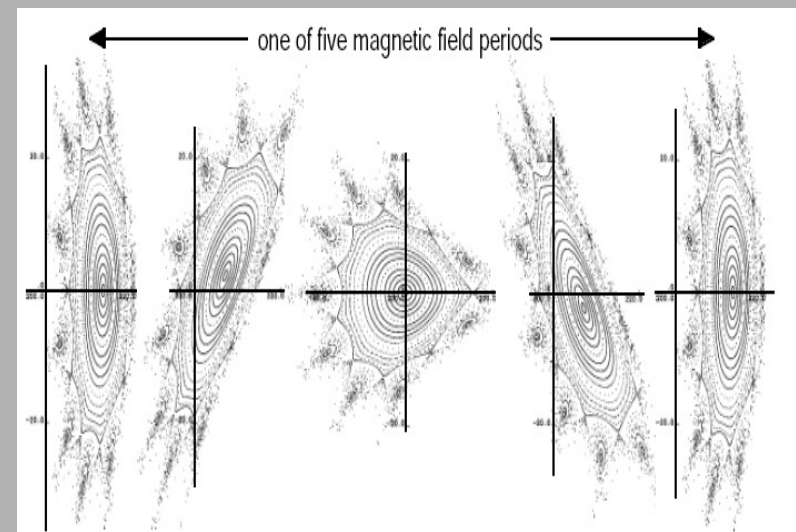
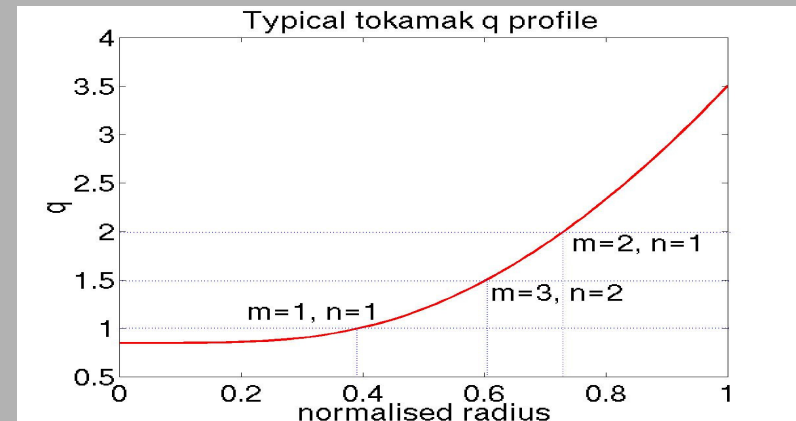
Magnetischer Einschluss

- Toroidales Feld
- Poloidales Feld
(Driftenkompensation)
- Ineinander
geschachtelte toroidale
Flächen (Koordinaten
 ψ , θ , φ)



Rotationstransformation Iota 1

- $q = a/R \cdot B_t/B_\theta$
- $q = 1/\iota$ Sicherheitsfaktor
= m/n (toroidale/poloidale Umläufe)
- Bei Tokamaks $q = 1$
($m=1, n=1$) =>
periodische
Sägezahninstabilität
- Magnetische Inseln
Größe $\sim (B_{nm}/(m \cdot d\iota/d\psi))^{0.5}$



Sägezahninstabilität

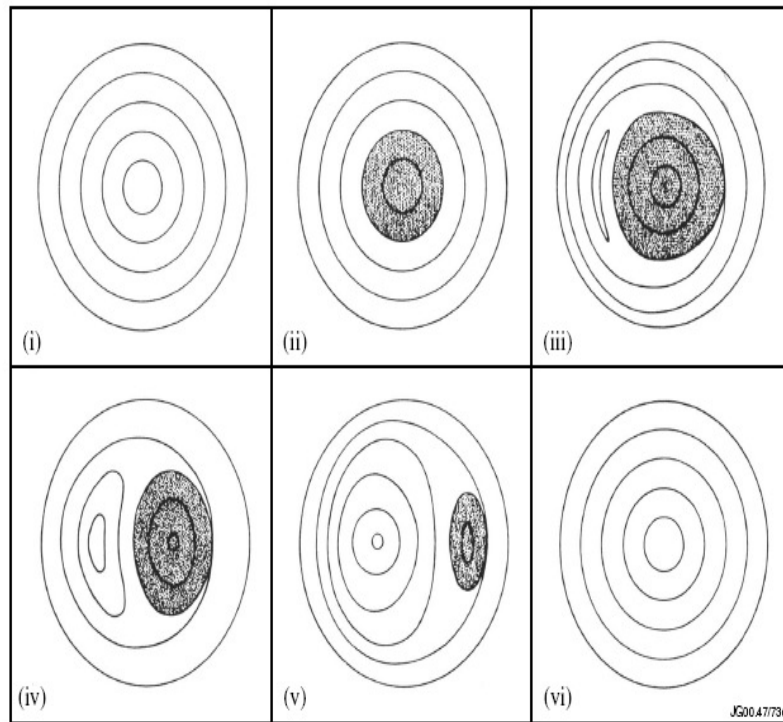


Figure 10.11. In Kadomtsev's model of the sawtooth instability magnetic reconnection in the centre of the plasma produces a magnetic island which grows to replace the original magnetic surfaces with $q < 1$ (shown shaded).

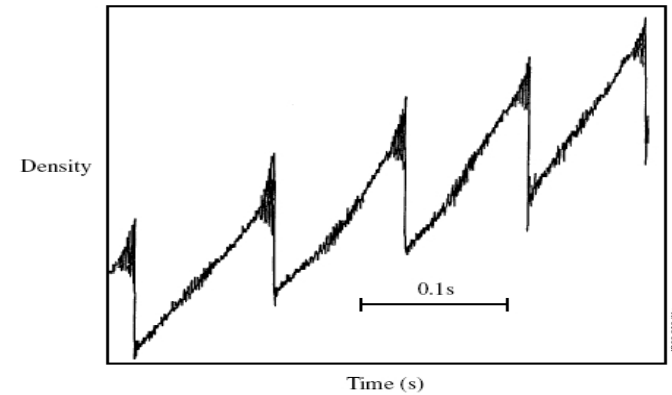


Figure 10.10. Sawtooth oscillations on the electron density.

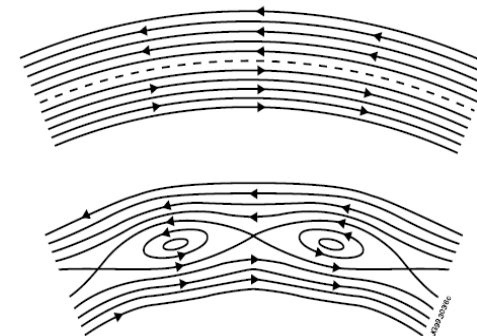
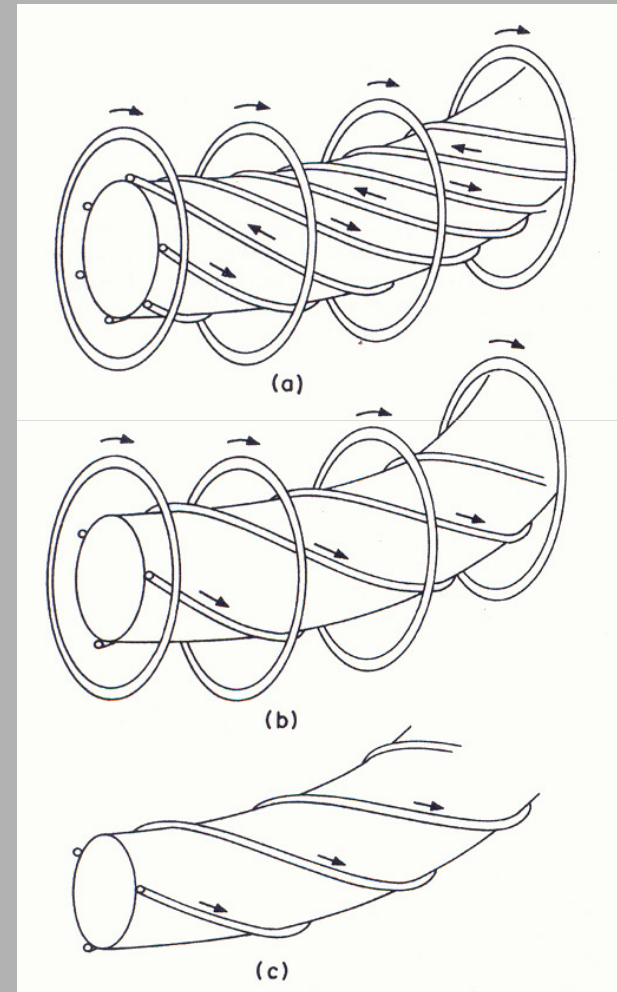


Figure 10.5. Illustrating how, in a tearing instability, the magnetic surfaces around a rational q surface reconnect to form a magnetic island.

Stellarator

- Poloidales Feld durch externe Spulen
 $di/dr > 0$
- Skalierung ISS95 (International Stellarator Scaling)

$$\tau_E = 0.079 \cdot a^{2.21} \cdot B^{0.53} \cdot P^{-0.59} \cdot R^{0.65} \cdot n^{-0.51} \cdot \iota^{0.4}$$



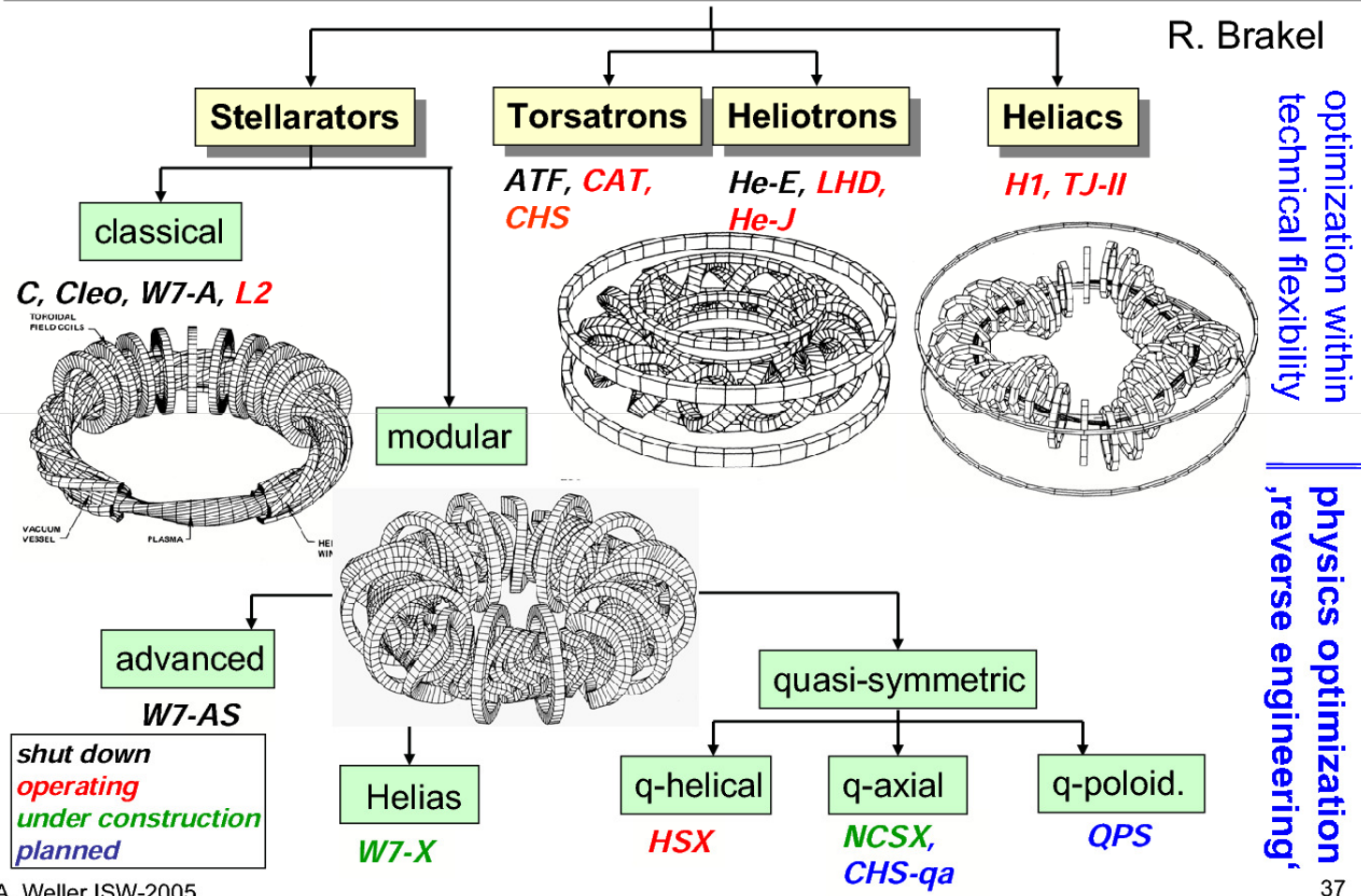
Stellarator - Gliederung

- Einfache geometrische Spulen
 - *ATF, CHS, Heliotron-E, LHD, TJ-IU, Wendelstein bis 7-A*
- Einfache geometrische Spulen mit helischen Achsenverschiebung
 - *H-1, Sheila, TJ-II*
- Modulare 3D Spulen
 - *CHS-qa, HSX, NCSX, QPS, MHH, W7-AS, W7-X*

International Stellarator Research

- Stellarators / Helical Devices -

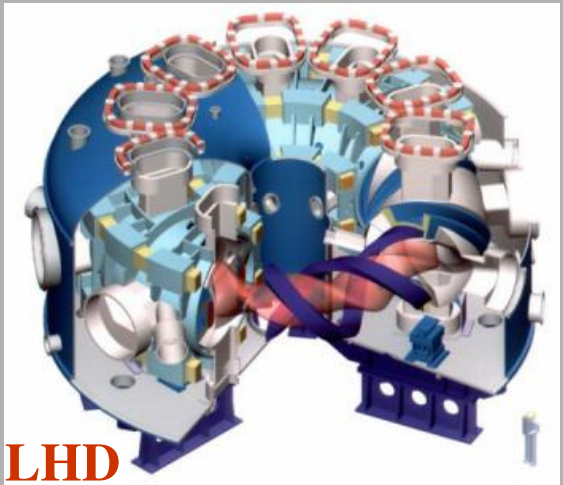
R. Brakel



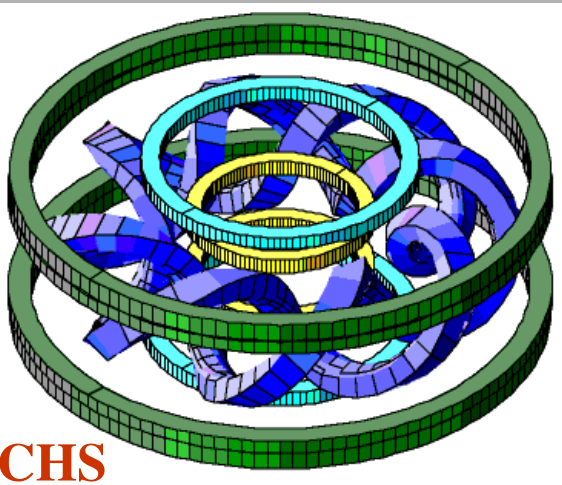
optimization within technical flexibility
 physics optimization, reverse engineering

A. Weller ISW-2005

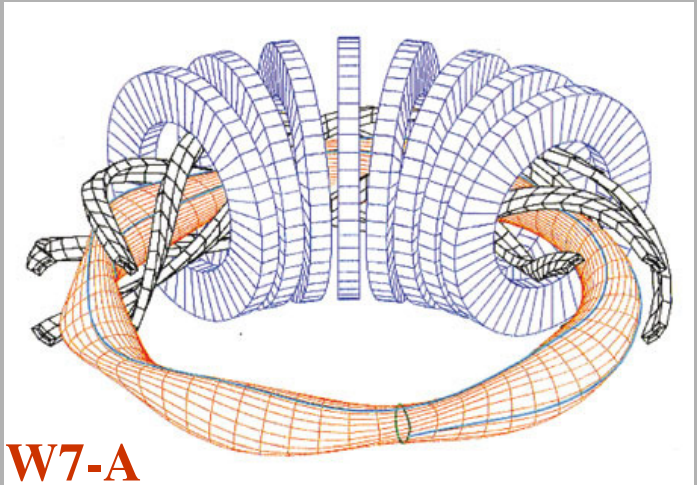
37



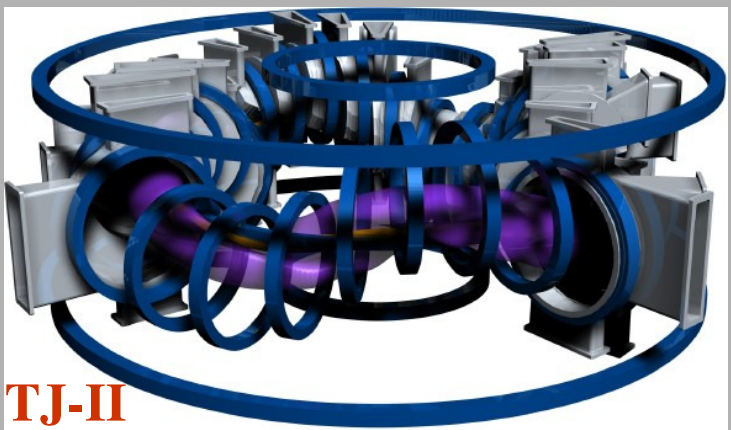
LHD



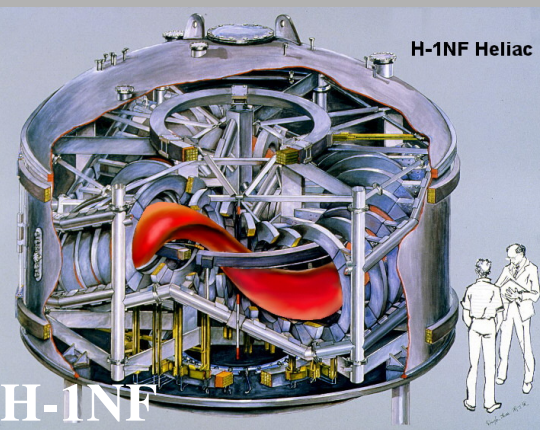
CHS



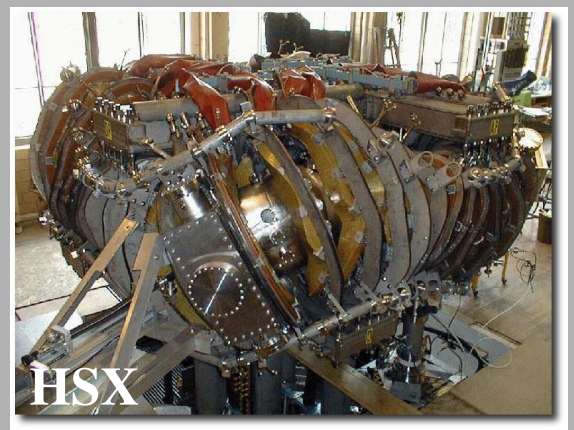
W7-A



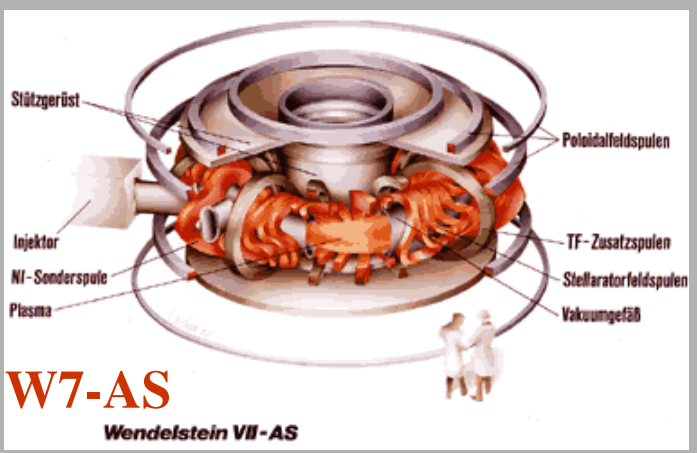
TJ-II



H-1NF

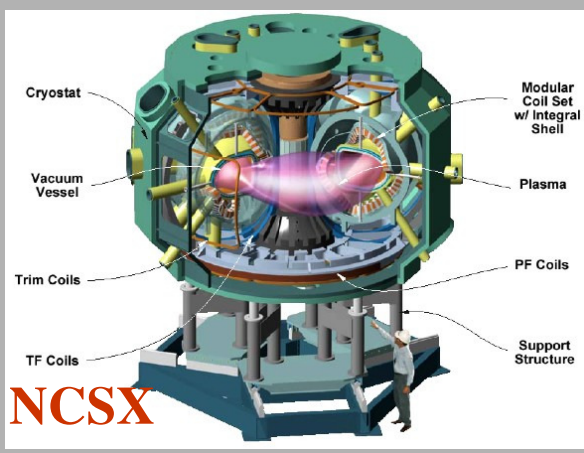


HSX

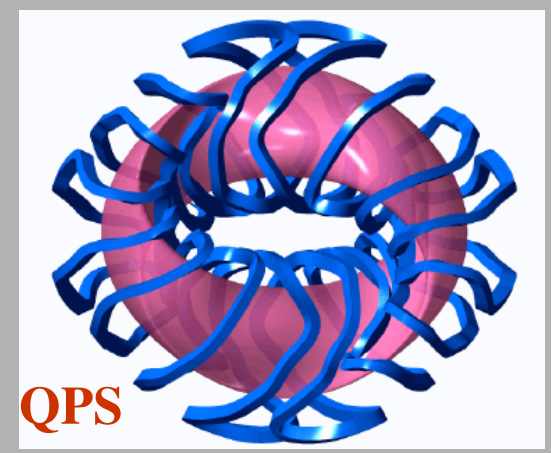


W7-AS

Wendelstein VII-AS



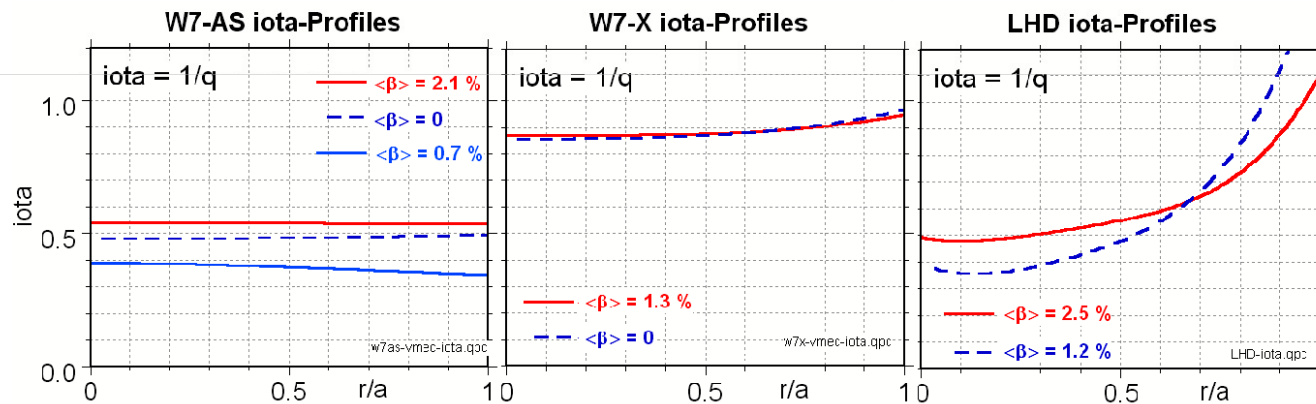
NCSX



QPS

Comparison of W7 and LHD Configurations

D.A. Spong, ORNL



W7- strategy: avoid islands by low shear, **LHD:** high shear, small islands

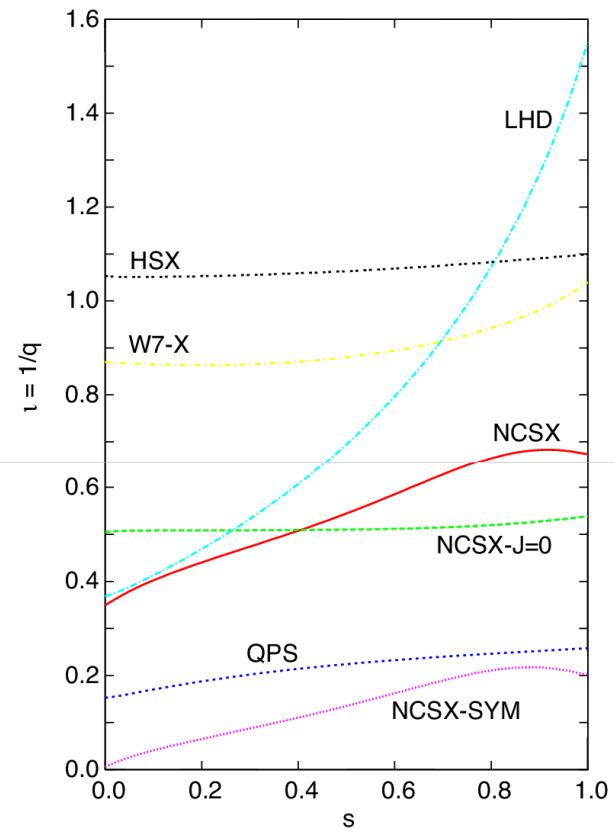
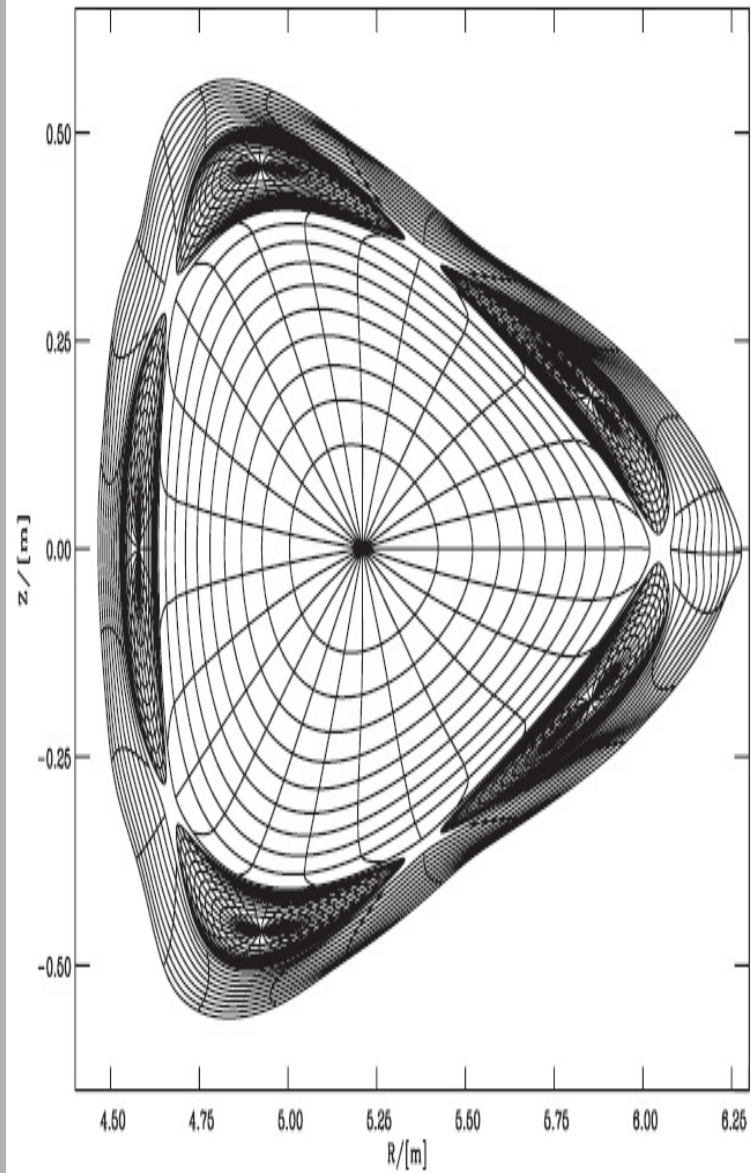


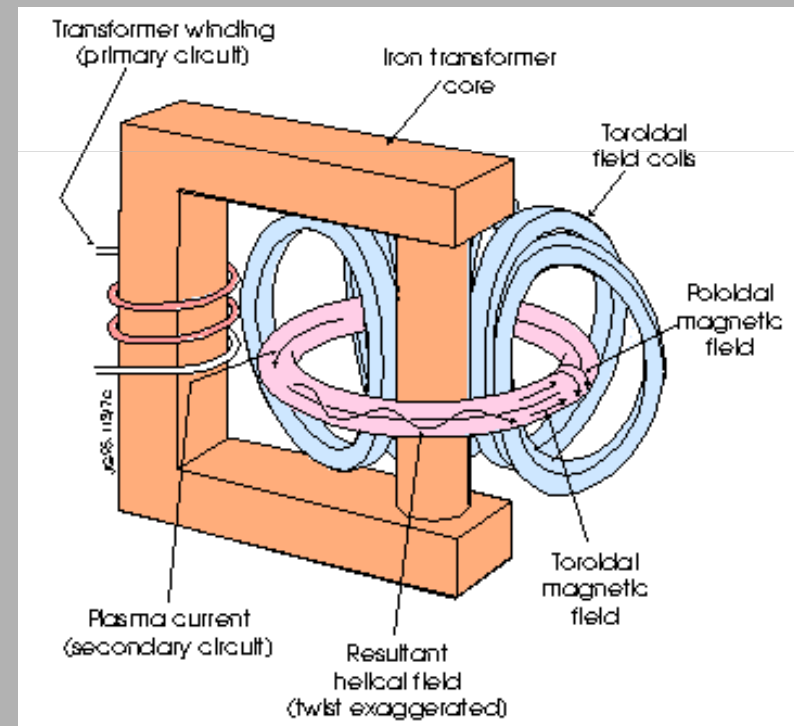
Fig.2

Токмакс

ТОРОИДАЛЬНАЯ КАМЕРА С МАГНИТНЫМИ КАТУШКАМИ

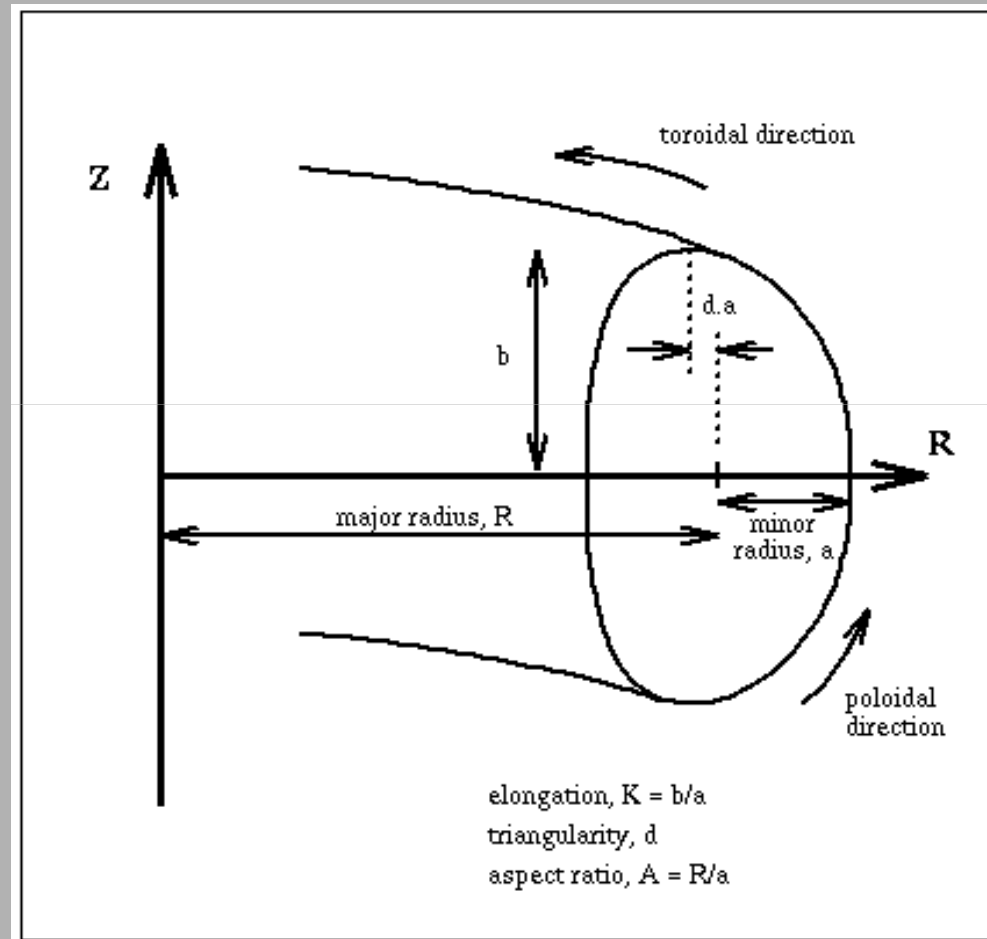
Toroidales Gefäß mit Magnetspulen

- toroidales Magnetfeld
- toroidaler Strom im Plasma (Transformator)
- Strom generiert poloidales Magnetfeld = Magnetflächen $dt/dr < 0$



Tokamaks - Form

- Quadrupolfeld →
Elliptizität
- Hexapolfeld →
Triangularität
=> D-Form
- Vertikalfeld →
Positionkontrolle



Tokamaks - Heizen

- OH (*Ohmic heating*)

- bis 1 MW
- $T < 1 \text{ keV}$

Zusätzliches Heizen (JET bis 30 MW)

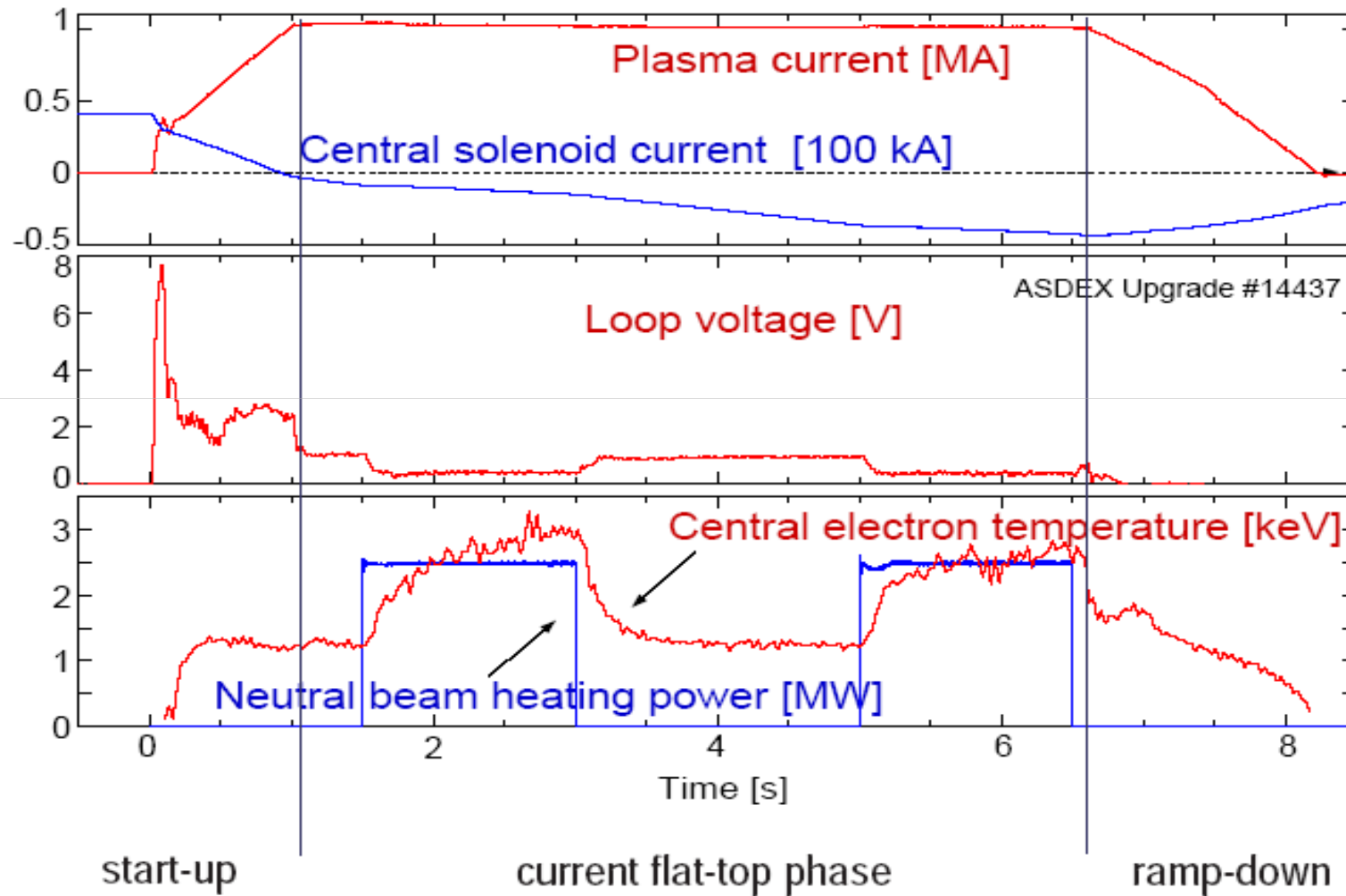
- NBH (*Neutral Beam heating*)

- $T < 30 \text{ keV}$

- RF (*Radio-frequency heating*)

- ICRH ($f \sim 50 \text{ MHz}$, $T_i \sim 14 \text{ keV}$)
- ECRH ($f \sim 100 \text{ GHz}$, $T_e \sim 10 \text{ keV}$)
- LHH (lower-hybrid heating, Landau-damping, $f > 1 \text{ GHz}$, $T_e \sim 100 \text{ keV}$)

Time history of a tokamak plasma ("shot")



Tokamaks - Moden

- L Mode (Low Confinement) – *schwach zusatzgeheizte Plasmen, turbulent*
- H Mode (High Confinement) – *Divertor (ASDEX, 1981), ITB (Internal Transport Barriers) am Rand, starke Zusatzheizung*
- ELMy Mode (Edge Localised Modes) – *wiederholende MHD Instabilität – Unreinheiten werden aus Plasma hinausgestossen*
- Skalierung

$$\tau_E = 3.3 \cdot 10^{-11} \cdot I_p^{0.93} \cdot B^{0.15} \cdot P^{-0.69} \cdot R^{1.97} \cdot n^{0.41} \cdot M^{0.19} \cdot \epsilon^{0.58} \cdot \kappa^{0.78}$$

Tokamak – Limiter und Divertor

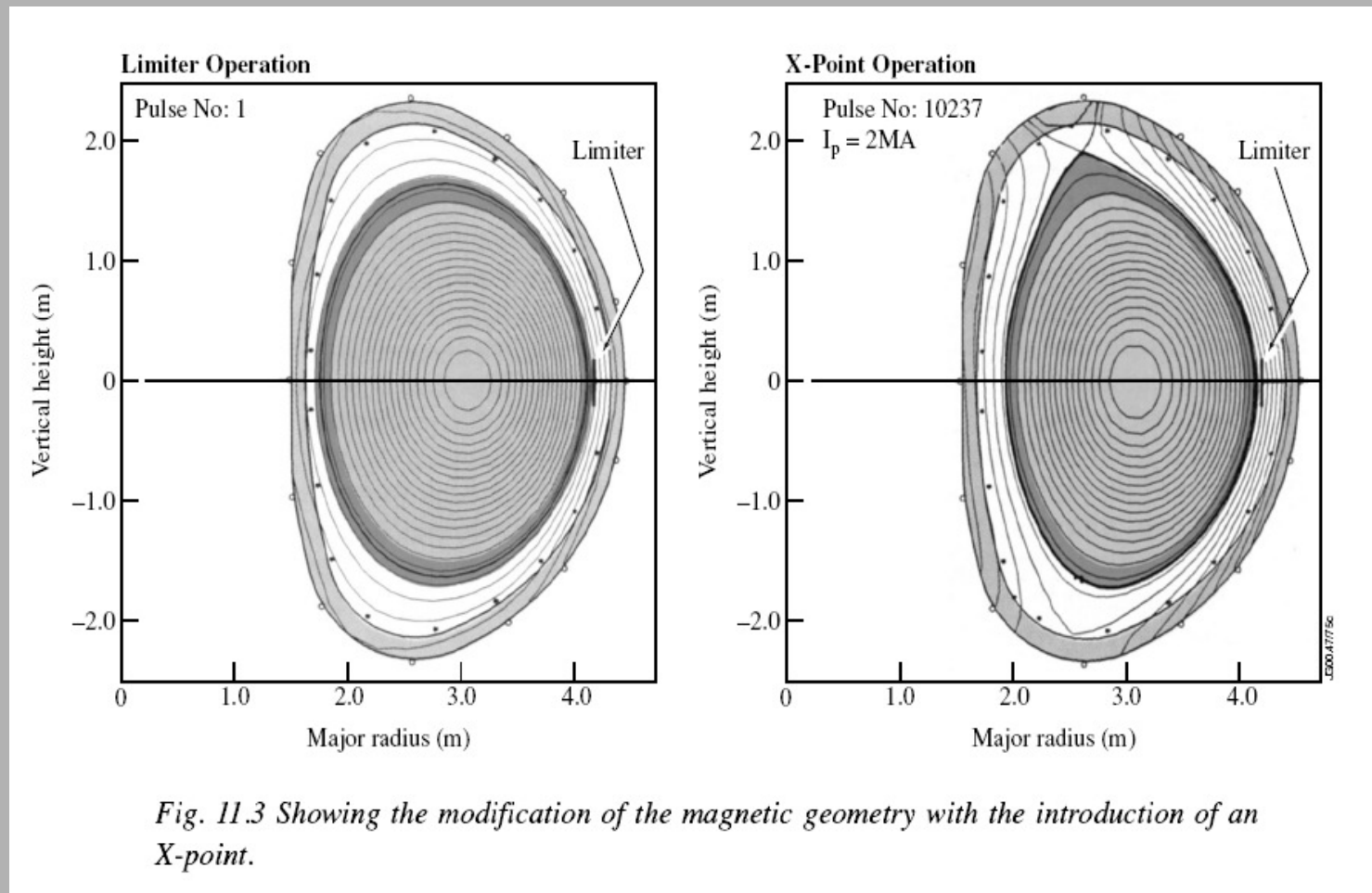
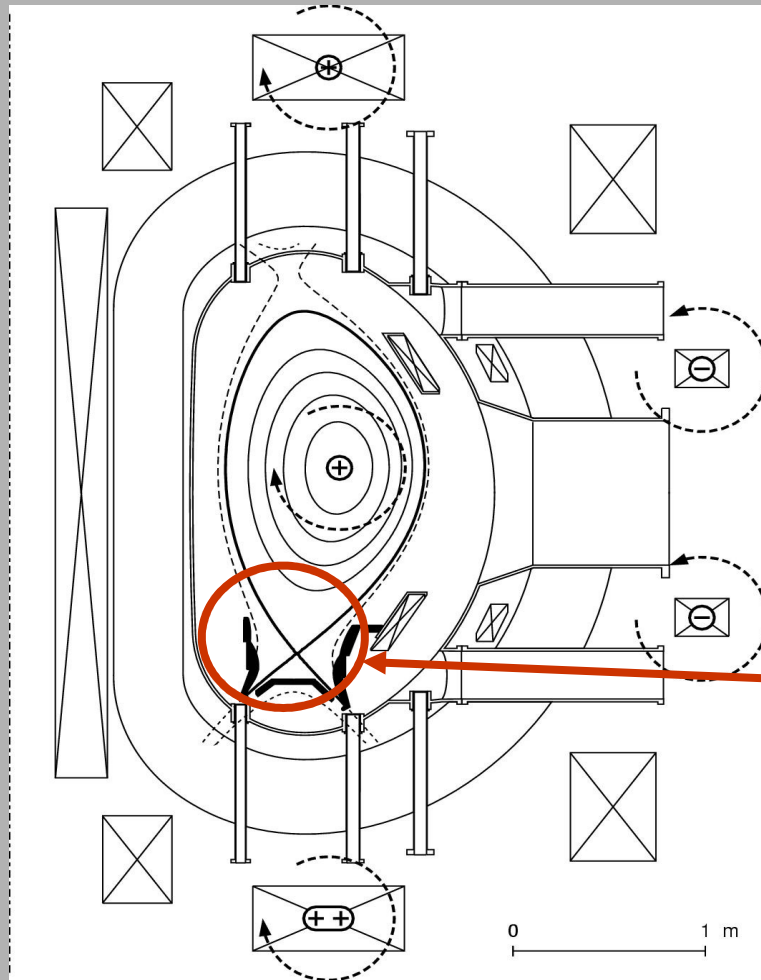


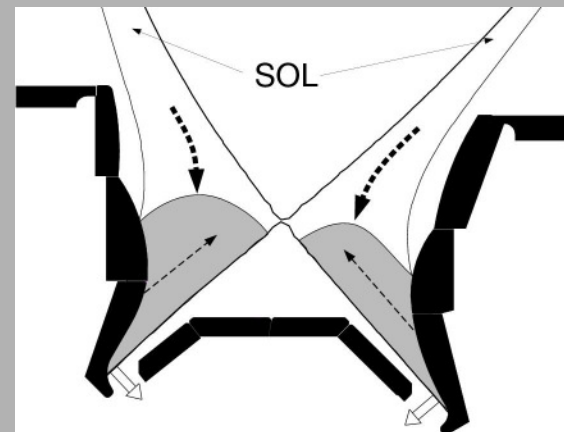
Fig. 11.3 Showing the modification of the magnetic geometry with the introduction of an X-point.

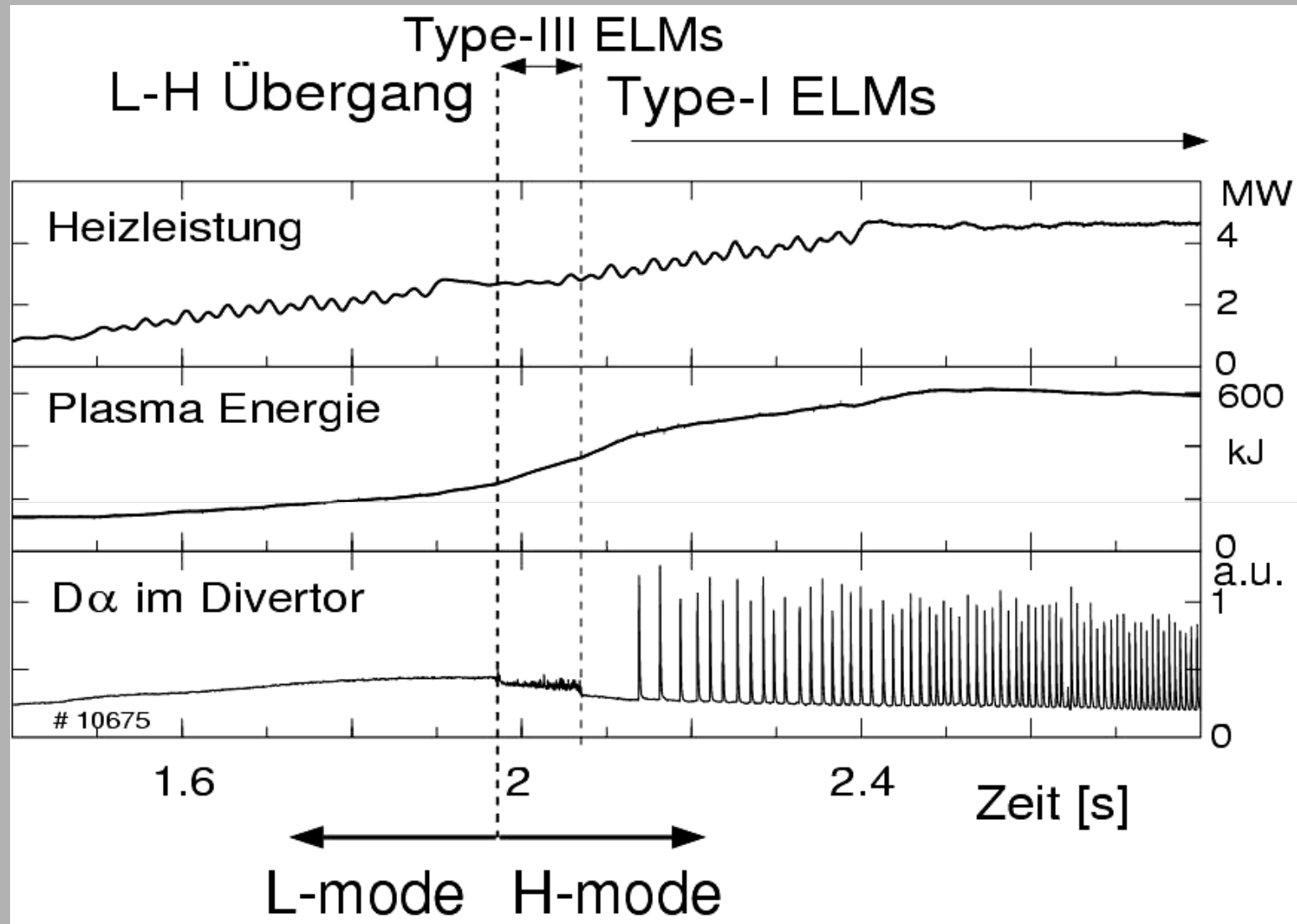
Tokamaks - Divertor



ASDEX

Separatrix trennt
Bereiche
geschlossener und
offener Feldlinien





1.2 Long Pulse Operation: Excellent Heat Removal



JET:

20s RS, 326MJ

JT-60U:

30s ELMy-H, 350MJ

LHD:

2min, 115MJ

HT-7:

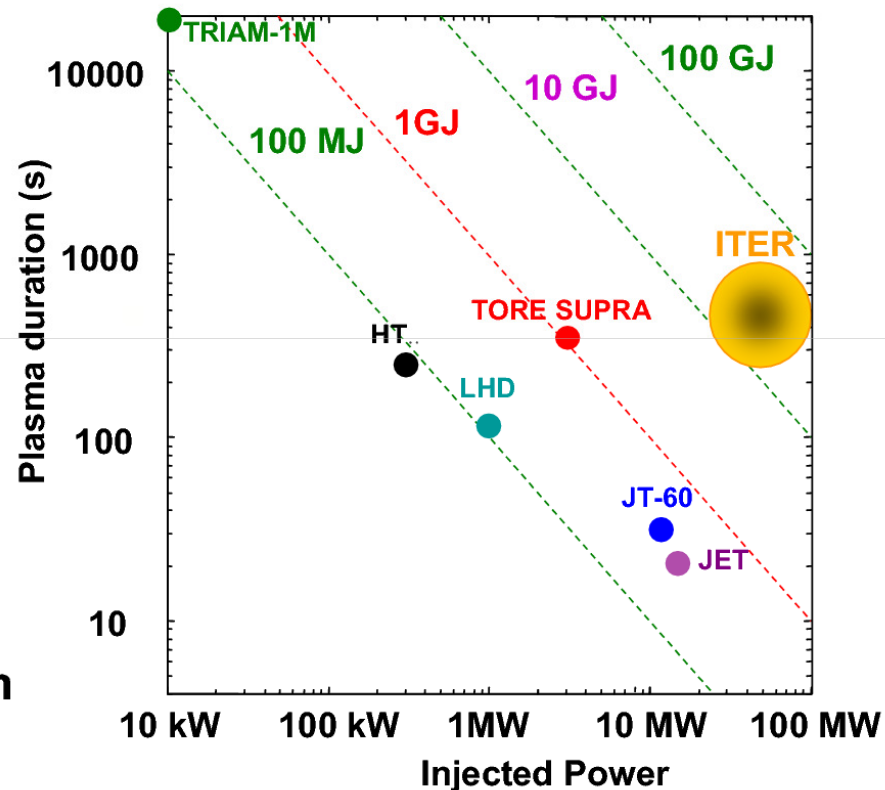
4min, T_{limiter} still rising

TORE-SUPRA:

6min, **1GJ**

TRIAM-1M:

5 hrs, No wall saturation

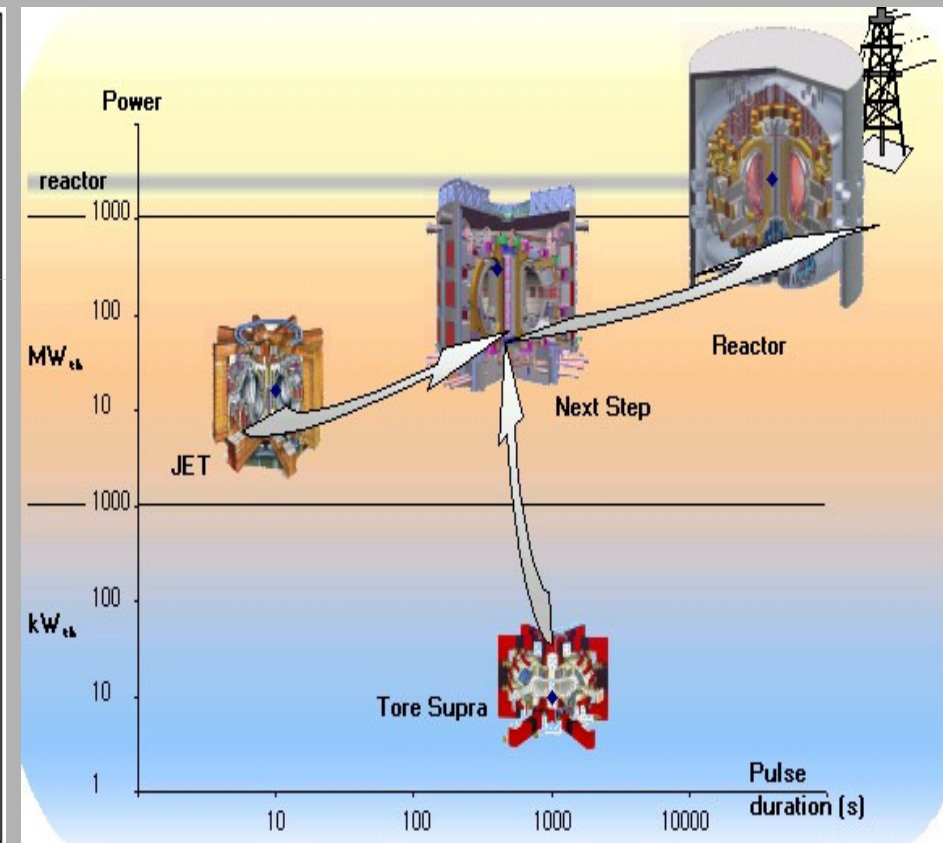
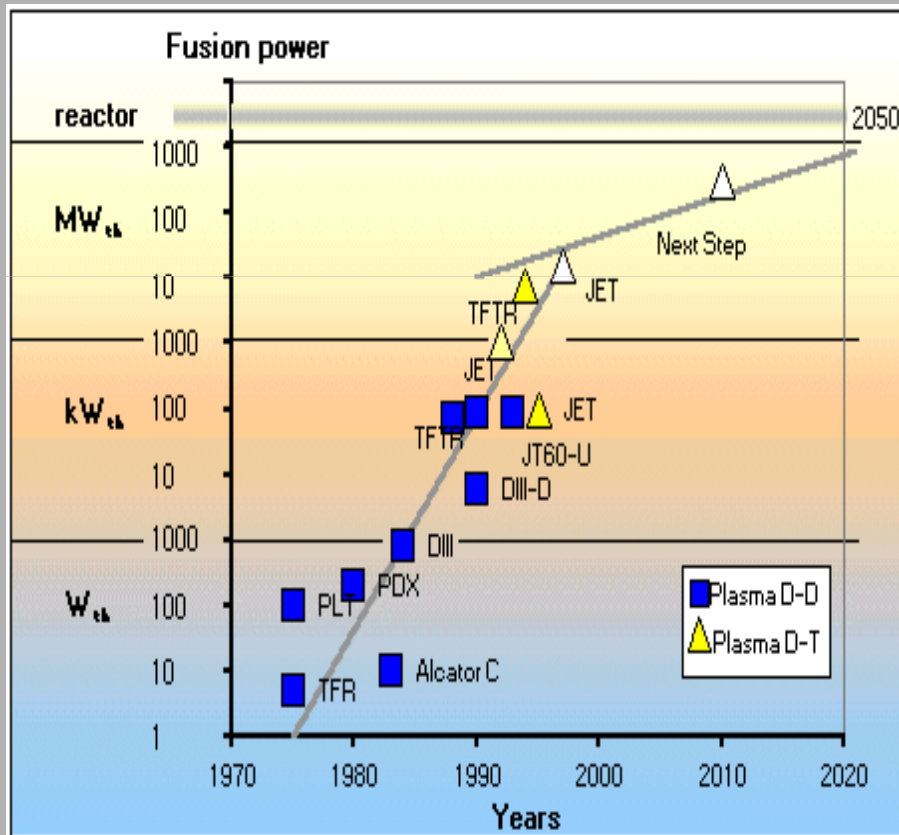


Tokamaks

- ASDEX, JET, TFTR, JT-60, Tore-Supra, TRIAM-1A, DIII-D
- Nahe Zukunft – SST-1 India ($I_p \sim 0.25\text{MA}$)
 - EAST China ($I_p \sim 2\text{MA}$)
 - KSTAR Korea ($I_p \sim 0.25\text{MA}$)
- IFMIF – Vanadium, SiC + Kompositen

ITER (International Thermonuclear Experimental Reactor)

- Iter = der Weg lat.



ITER - Ziele

- Erreichen von $Q > 10$ (induktiver Mod)
- Steady-state Operation bei $Q > 5$
- Möglichkeit einer kontrollierten Zündung
- Demonstration von Verfügbarkeit und Integrierbarkeit der Komponenten
- Komponententest für Fusionsreaktoren
- Modultest für Tritiumproduktion

ITER - Schema

Zentraler Solenoid
Nb₃Sn, 6 Module

Poloidale Spulen
Nb-Ti

Toroidale Spulen
Nb₃Sn

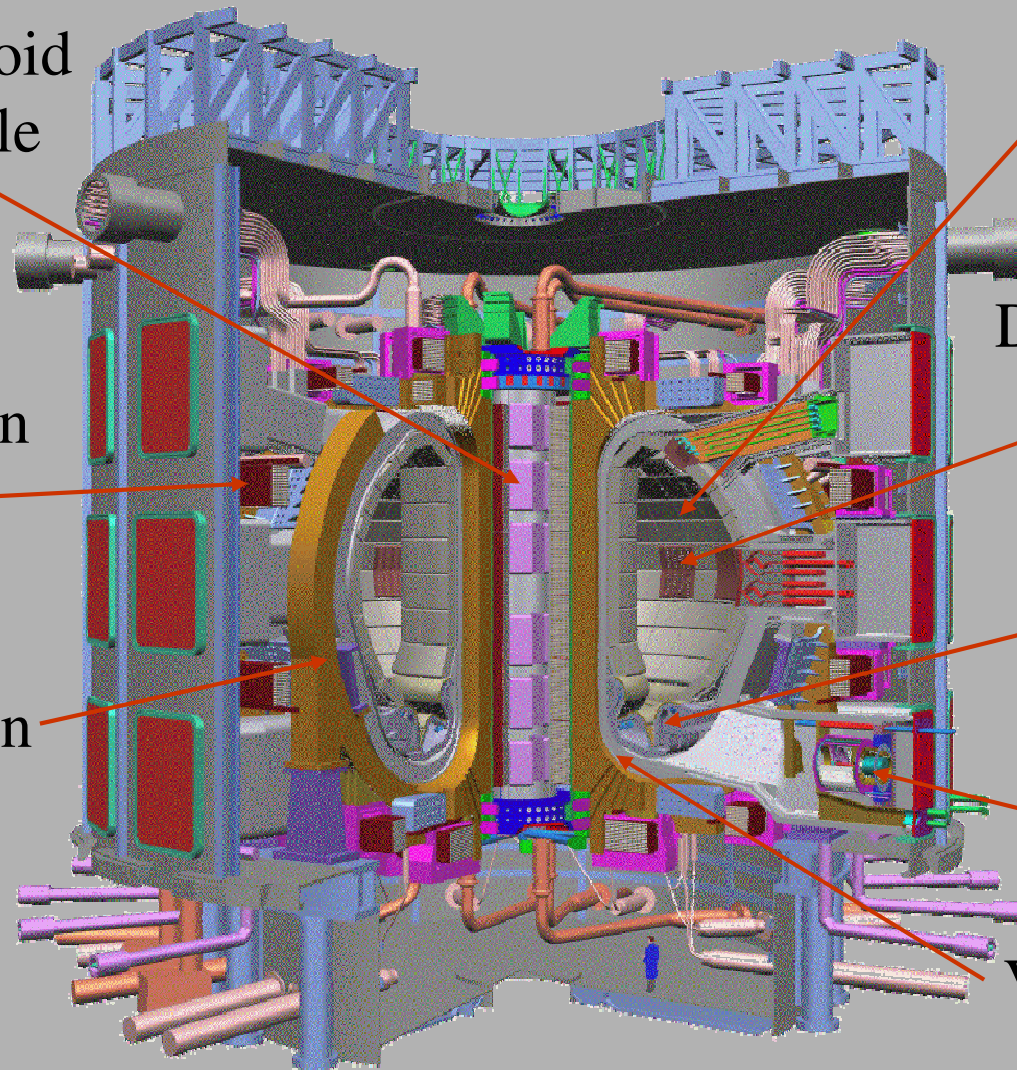
Blanket

Diagnose, Heizung,
Limiters

Divertor

Kryopumpen, 8

Vakuummkammer



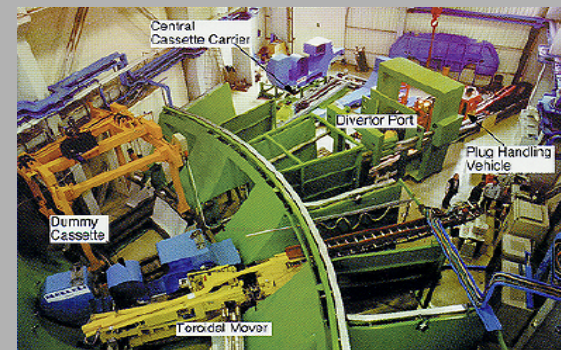
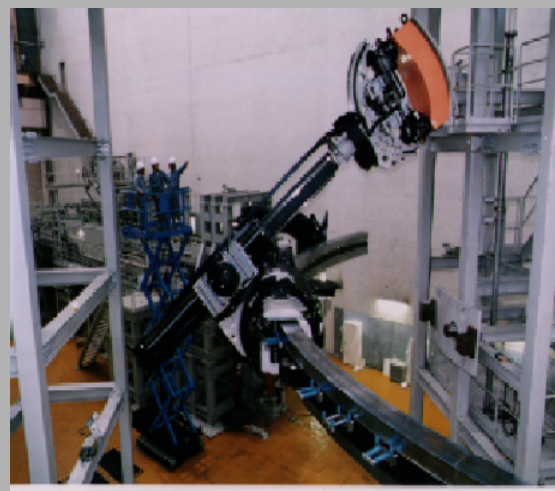
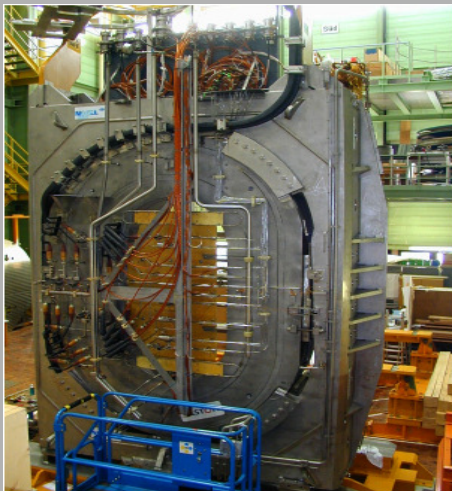
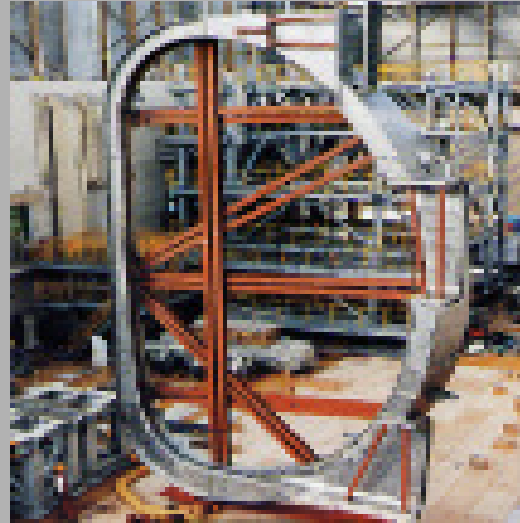
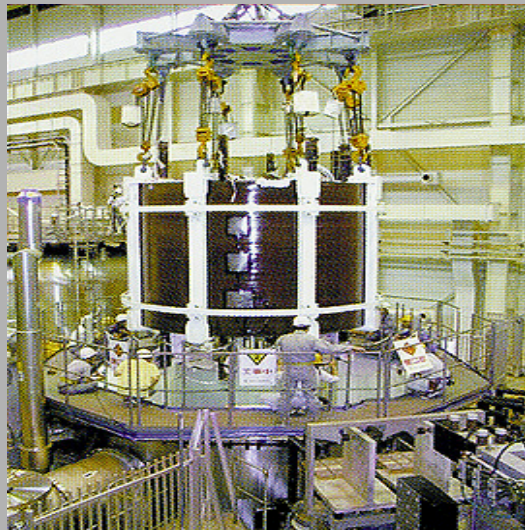
ITER - Data

Total Fusionsleistung	500 MW
Q = Fusion/ Heizung	≥ 10
Average neutron wall loading	0.57 MW/m ²
Entladungszeit	≥ 300 s
großer Radius	6.2 m
Plasmaradius	2.0 m
Plasmastrom (I_p)	15 MA
Vertical elongation @95% flux surface/separatrix	1.70/1.85
Triangularity @95% flux surface/separatrix	0.33/0.49
Safety factor @95% flux surface	3.0
Toroidalfeld	5.3 T
Plasmavolumen	837 m ³
Plasmafläche	678 m ²
Installierte Heizung/current drive power	73 MW (100 MW)

ITER – 7 Vorprojekte (Modelle)

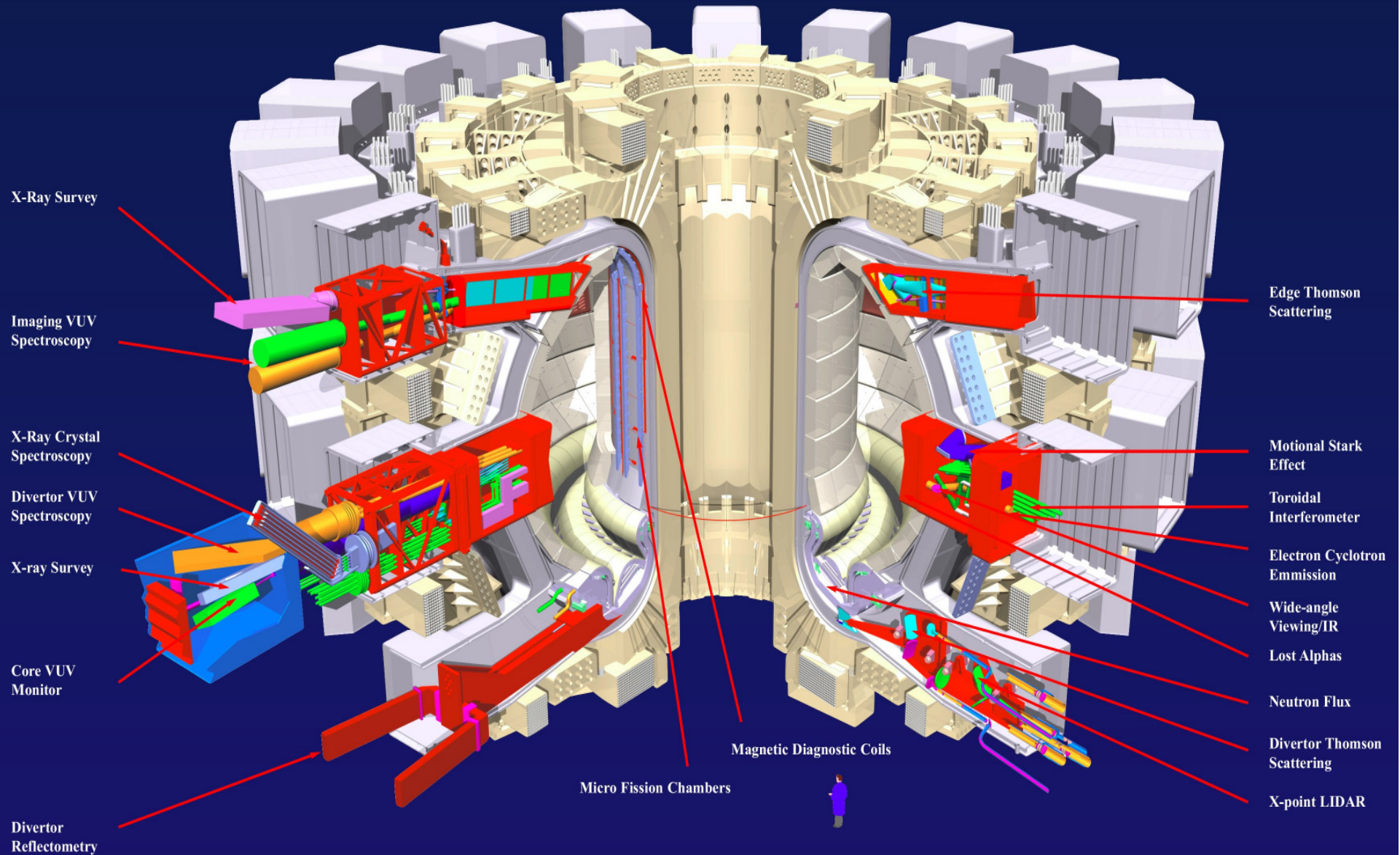
- Zentraler Solenoid (6 Module) L-1
- Toroidale Spulen (18 Spulen) L-2
- Vakuumkammer (9 Sektoren) L-3
- Blanketmodul (421 Modulen, Be, Cu, Stahl) L-4
- Divertorkasette (54 Kassetten, W, C, Cu, Stahl) L-5
- Austauschen von einem Blanketmodul (Automatisiert) L-6
- Austauschen von Divertorkasette (Automatisiert) L-7

ITER – 7 Vorprojekte (2001)



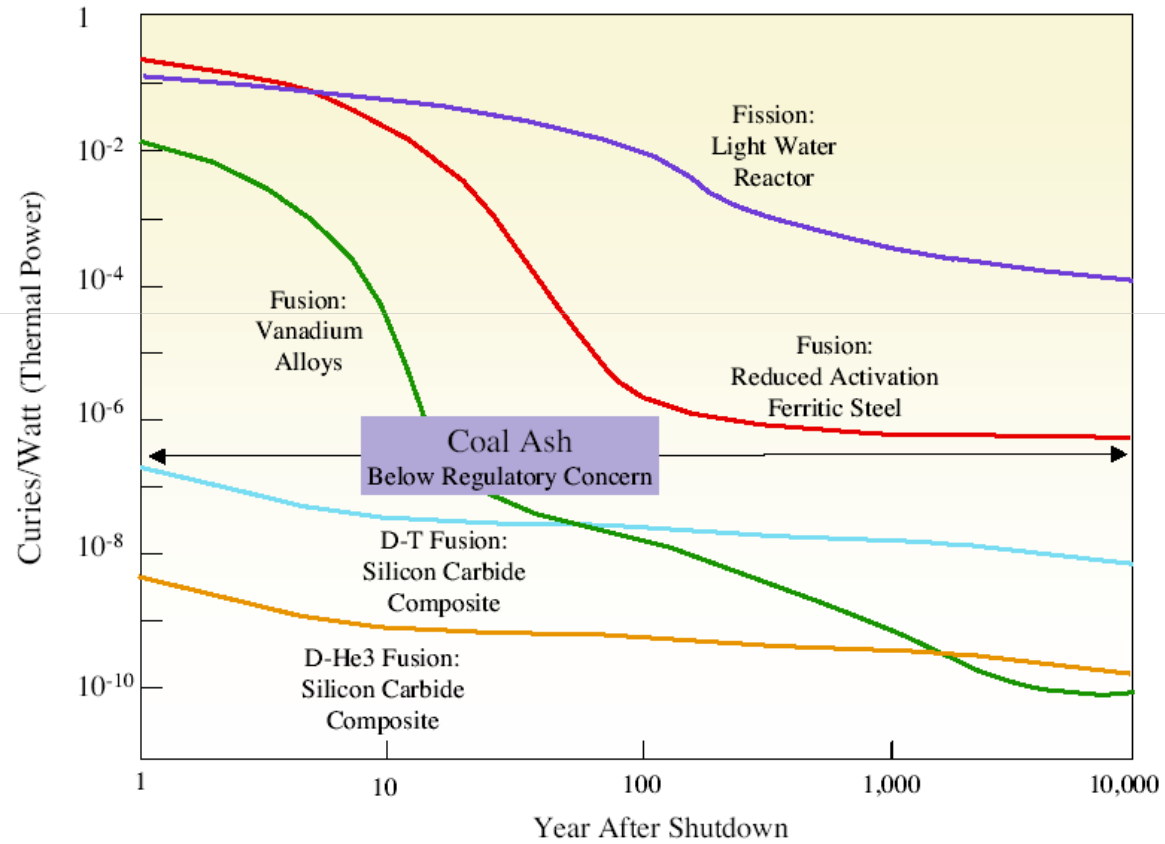
13.01.2006 Frankfurt am Main

ITER - Diagnose

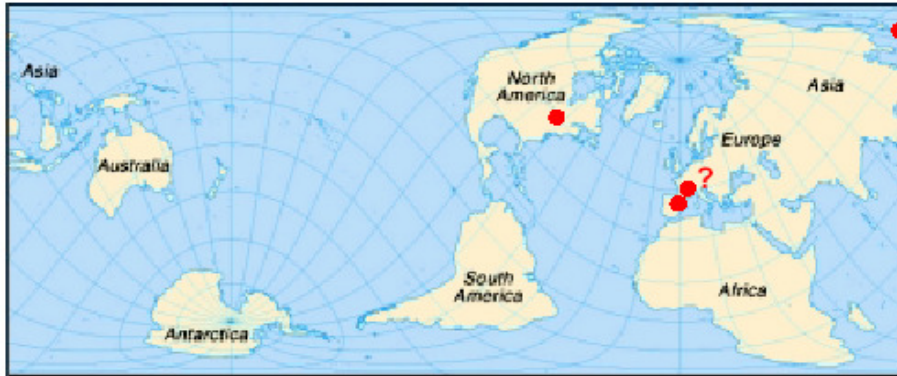


Radioaktivität (IFMIF)

*Comparison of Fission and Fusion
Radioactivity After Shutdown*



ITER Site Candidates



Japan: Rokkasho

<http://www.japantimes.co.jp/cgi-bin/getarticle.pl5?nn20020530b4.htm>
<http://apti.net.pref.aomori.jp/info-en/01/02/010201050.html>



Canada: Clarington

<http://www.itercanada.com/index.cfm>
<http://www.municipality.clarington.on.ca/>



Spain: Vandellos

<http://www-fusion.ciemat.es/fusion/iter/ITER-eng.html>



France: Cadarache

!! No official offer yet !!

<http://www-cad.cea.fr/default.htm>

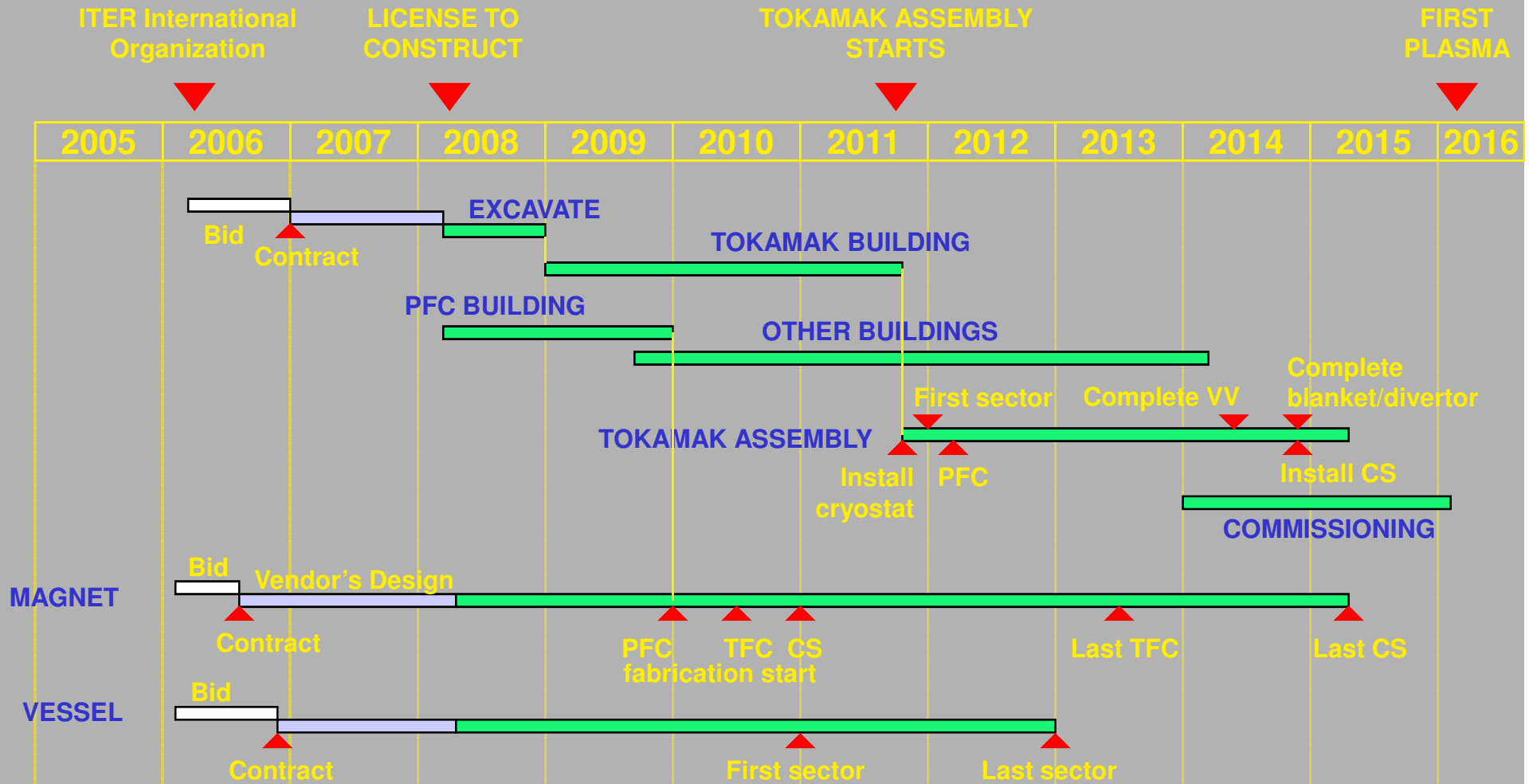
Cadarache



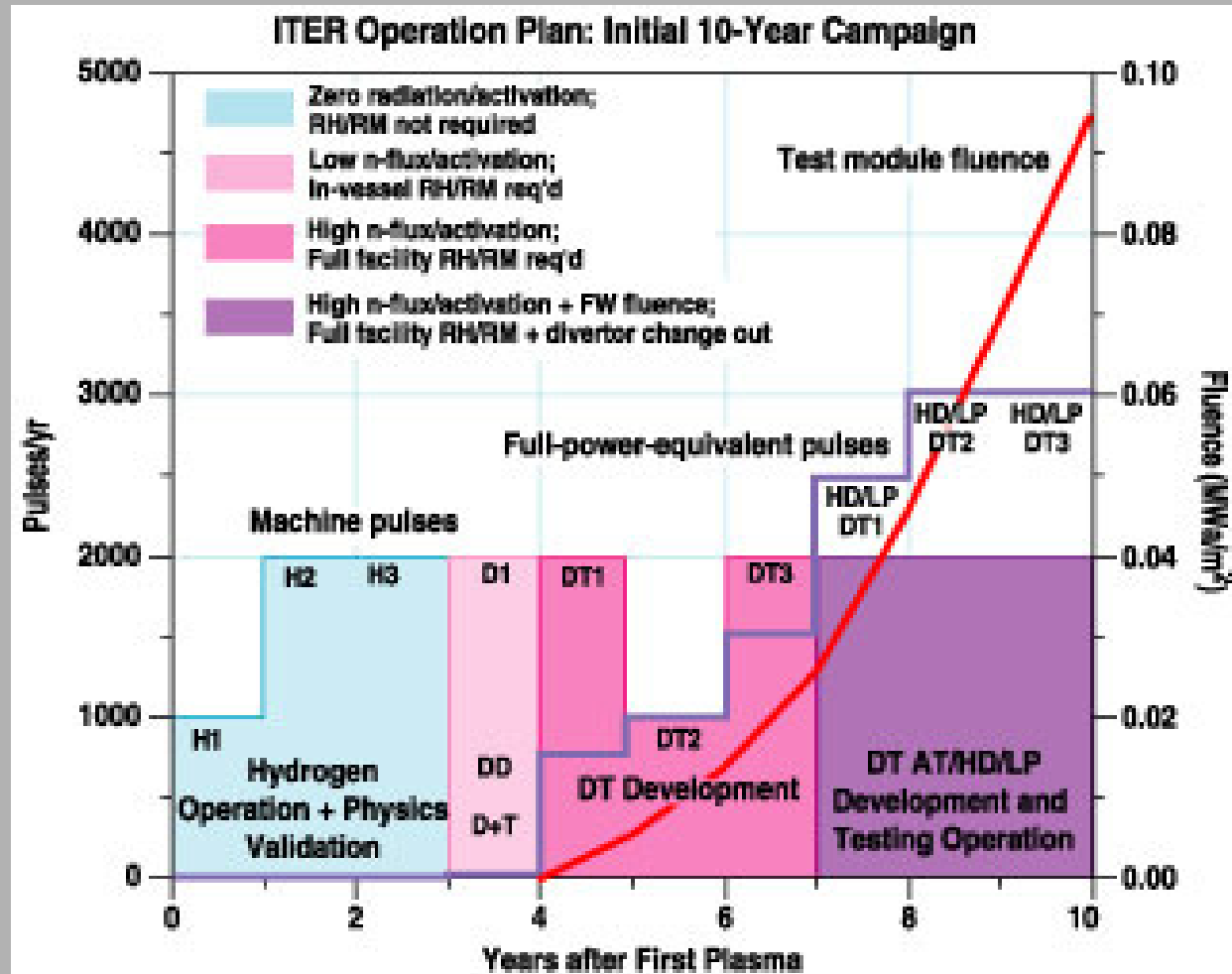
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Aufbauplan



Operationsplan



Zusammenfassung

- Steady-State Zustand für ITER und Q
- ITER = D-T Reactor
- Skalierung ? Neue Effekte ?

- J. Sanchez, V. Tribaldos, „*Keeping the options open: concept improvements and stellarator physics*“, Plasma Phys. and Controlled Fusion 47(2005) B349-B361
- A. H. Boozer, „*Physics of magnetically confined plasmas*“, Rev. of Mod. Phys 76(2004) 1071-1141
- IFRC(International Fusion Research Council), „*Status report on fusion research*“, Nucl. Fusion 45(2005) A1-A28