A volte non sono Aeroplani … la Roadmap europea verso il Reattore a Fusione Nucleare

R.Albanese e M.Mattei

Parte II
Il progetto ITER e il problema del controllo magnetico

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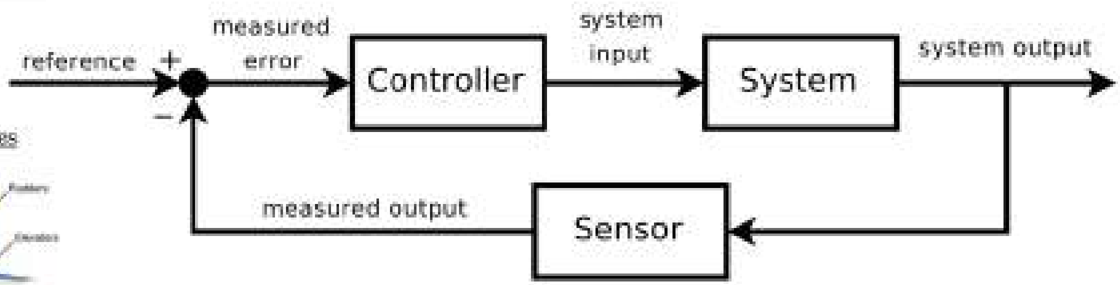
Iniziamo dall'ITER personale di Ingegnere Aeronautico…

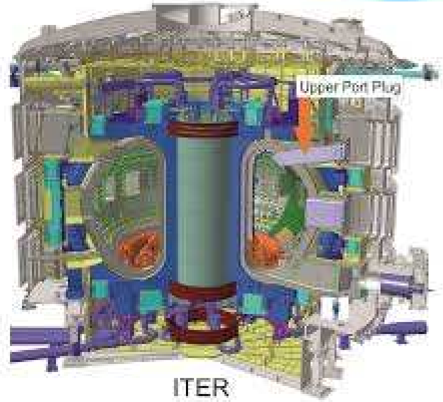
1993

1995

1997

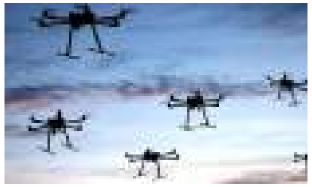
2001





2003

2019



Università
degli Studi
della Campania
Luigi Vanvitelli

2008

… e dalla storia di della cooperazione internazionale sulla fusione

18 Aprile 1956 Igor V. Kurchatov accompagna Khrushchev in UK e visita Harwell

Physicists the world over are attracted by the extraordinarily interesting and very difficult task of controlling the thermonuclear reaction…"

"…reach an agreement on cooperation in the peaceful use of nuclear energy."

35 years ago - ITER

1985 - Summit di Ginevra – accordo di collaborazione promosso da Reagan e Gorbachev (Euratom, Giappone, USA, USSR)

1988 – inizia il Conceptual design

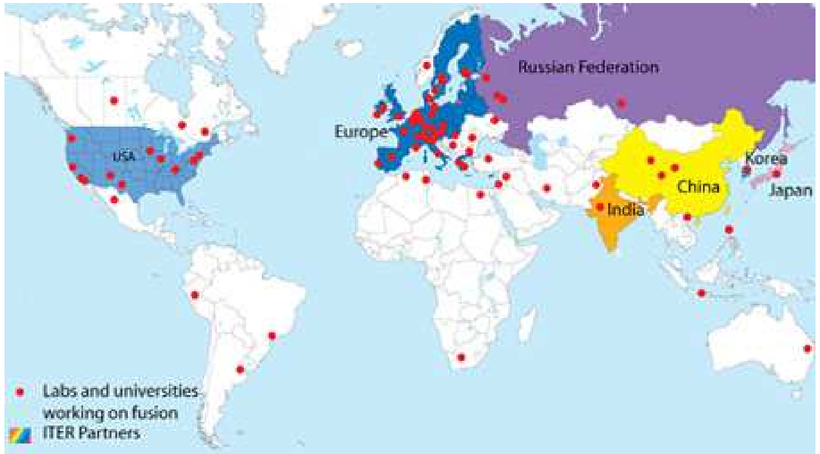
2001 – Final design approvato dagli stati membri

2003 – Cina e Korea si uniscono al progetto

2005 – Si unisce l'India

2005 – Si stabilisce il sito per la costruzione (Cadarache)

2006 – Chirac e Barroso promuovono la firma dell' ***ITER Agreement***



Nazioni coinvolte

ITER timeline

2005 Decision to site the project in France
2006 Signature of the ITER Agreement
2007 Formal creation of the ITER Organization
2010-2014 Ground support structure and seismic foundations for the Tokamak
2012 Nuclear licensing milestone: ITER becomes a Basic Nuclear Installation under French law

2014-2021 Construction of the Tokamak Building (access for assembly activities in 2019)
2010-2021 Construction of the ITER plant and auxiliary buildings for First Plasma
2008-2021 Manufacturing of principal First Plasma components
2015-2023 Largest components are transported along the ITER Itinerary

2020-2025 Main assembly phase I
2022 Torus completion
2024 Cryostat closure
2024-2025 Integrated commissioning phase
Dec 2025 First Plasma
2026 Begin installation of in-vessel components
2035 Deuterium-Tritium Operation begins

Vita operativa attesa di ITER: 20 anni!

ITER funding

Poposed Costs
2005 €5 billion for the construction and €5 billion for maintenance and the research connected with it during its lifetime.
2009 +€9.6 billion for the construction
2010 +€4.6 billion for the construction
2016 > €20 billion in total

Funding share
- 45% European Union (5/11)
- 55% China, India, Japan, South Korea, Russian Federation, USA (1/11 each).

Euratom will contribute to 34% of the total costs during the operation and deactivation phases

Research staff at Cadarache and construction contracts
- European Union (4/11)
- Japan (2/11) (special status)
- Others (1/11)

Obiettivi di ITER

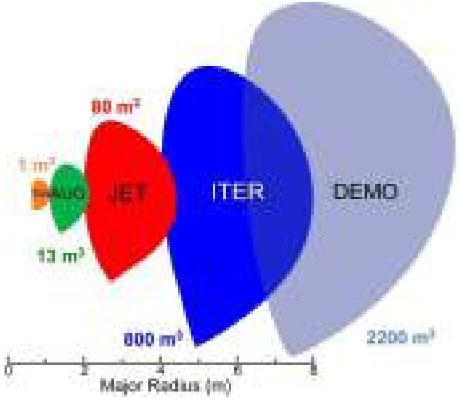
Q ≥ 10

1) Produce 500 MW of fusion power for pulses of 400 s
1997 at JET - 16 MW of fusion power from 24 MW of power injected into its heating systems (Q=0.67).
ITER designed for Q ≥ 10. 50 MW of injected heating power → 500 MW of fusion power for long pulses.
ITER will not capture the power it produces as electricity

2) Demonstrate the integrated operation of technologies for a fusion power plant
study plasmas under conditions similar to those expected in a future power plant and test technologies such as heating, control, diagnostics, cryogenics and remote maintenance in an integrated way.

3) Achieve a deuterium-tritium plasma in which the reaction is sustained through internal heating
In a *burning plasma*, the energy of the helium nuclei produced when hydrogen isotopes fuse becomes large enough—because of the large number of reactions—to exceed the plasma heating that is injected from external sources. First burning plasma device in the world.

4) Test tritium breeding blankets
Demonstrate the feasibility of producing tritium within the vacuum vessel. The world supply of tritium is not sufficient to cover the needs of future power plants.



5) Demonstrate the safety characteristics of a fusion device
first in the world to have successfully undergone the rigorous examination of its safety case. Demonstrate control of the plasma and fusion reactions with negligible consequences to the environment.

Fusion4Energy

Fusion for Energy (F4E) is the European Union’s Joint Undertaking for ITER and the Development of Fusion Energy. The organisation has three objectives:

- F4E is responsible for providing Europe’s contribution to ITER
- F4E also supports fusion research and development initiatives through the Broader Approach Agreement, signed with Japan – a fusion energy partnership which will last for 10 years.
- F4E will contribute towards the construction of demonstration fusion reactors.

https://f4e.europa.eu

F4E is established for a period of 35 years from 19 April 2007 and is located in Barcelona, Spain.

Filmato su ITER

https://www.youtube.com/watch?v=IP7Vuqz-MAE

ITER layout


china eu india japan korea russia usa
iter
TOKAMAK
& PLANT SYSTEMS

ITER – Magnet System

Largest and most integrated superconducting magnet system ever built.
Produce magnetic fields to initiate, confine, shape and control the ITER plasma

10.000 Tonnes, combined stored magnetic energy of 51 Gigajoules (GJ),
Manufactured from niobium-tin (Nb3Sn) or niobium-titanium (Nb-Ti),
Superconducting when cooled with supercritical helium in the range of 4K

Superconducting magnets are able to carry higher current and produce stronger magnetic field than conventional counterparts. They also consume less power and are cheaper to operate

Internally cooled superconductors called "cable-in-conduit conductors," in which bundled superconducting strands—mixed with copper—are cabled together and contained in a structural steel jacket.

Nb3Sn strands used in ITER's toroidal field and central solenoid magnet systems - more than 100,000 km - were produced by nine suppliers in a procurement effort that lasted from 2008 to 2015

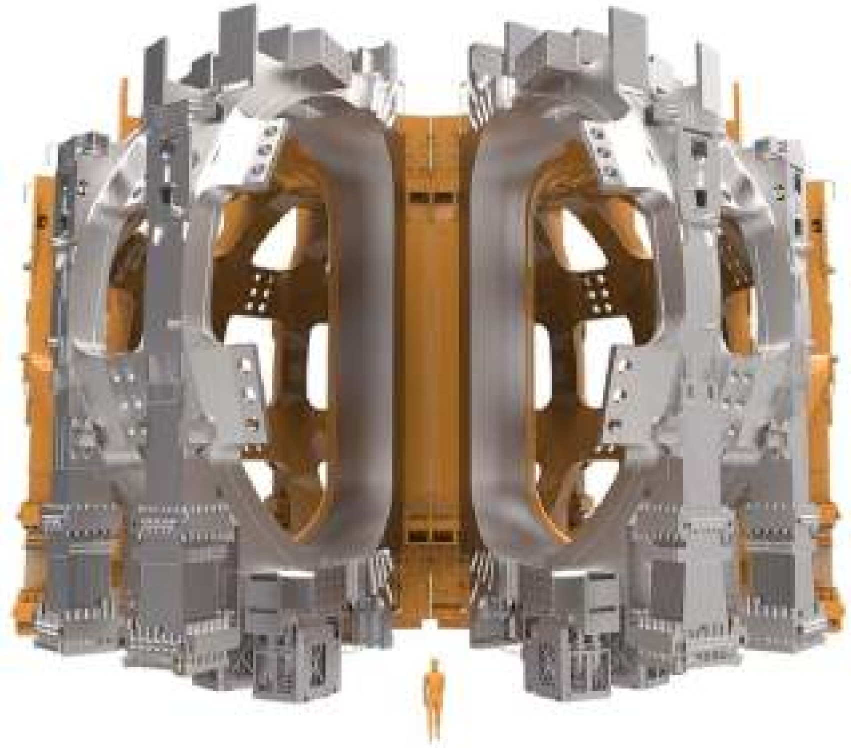
Toroidal Field System

18 "D"-shaped toroidal field magnets placed around the vacuum vessel produce a magnetic field whose primary function is to confine the plasma particles.

Total magnetic energy of 41 gigajoules and a maximum magnetic field of 11.8 Tesla.

One TF coil: 310 tonnes, 9 x 17 m

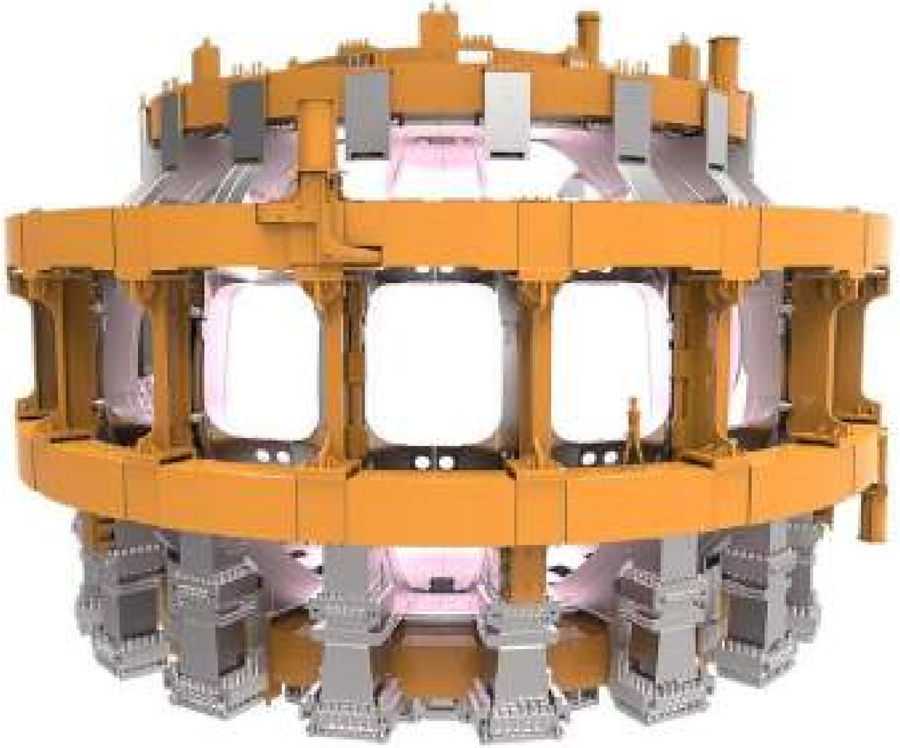
Toroidal field coils are wound in "double pancakes"—layers of spiralled conductor embedded in radial plates and encased in large stainless steel structures.



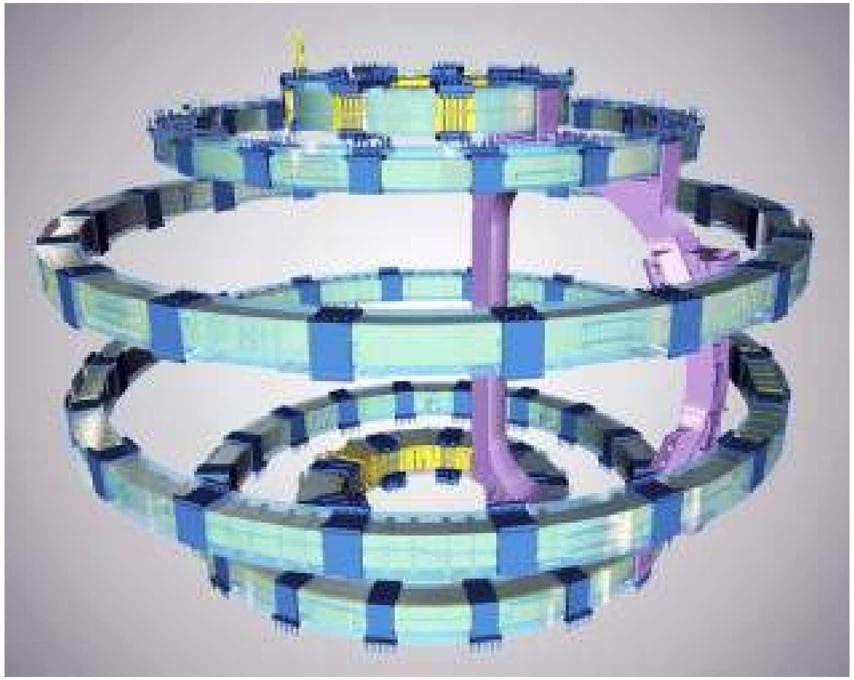
The "superstructure" of toroidal field coils pushes the limits of manufacturability. State-of-the-art welding techniques will be necessary to reach high quality requirements.

Poloidal Field System

Six ring-shaped poloidal field coils are situated outside of the toroidal field magnet structure to shape the plasma and contribute to its stability.
The largest coil has a diameter of 24 metres; the heaviest is 400 tonnes. The poloidal field coils are designed to produce a total magnetic energy of 4 gigajoules and a maximum magnetic field of 6 tesla.
The four largest poloidal field coils will be produced from niobium-titanium superconductor in an on-site facility.



PF Coils on site fabrication



Six poloidal field coils positioned horizontally around the ITER vacuum vessel and D-shaped toroidal field coils will help shape the plasma and keep it in suspension away from the walls. The top poloidal field coil (PF1) will be supplied by Russia; the five lower ring coils are under the procurement responsibility of Europe. Four of these will be produced on site. (PF6 will be produced by China under contract with Europe)*

Coils manufactured on site: PF2, PF3, PF4, PF5
Diameter of coils: from 17 metres (PF2, PF5) to 24 metres (PF3, PF4)
Amount of NbTi cable-in-conduit superconductor per coil: from 6 km (PF2) to 14 km (PF3)
Weight of final assemblies: 200 to 400 tonnes
First completed coil: 2019
Procurement responsibility: Europe
Contractors: ASG Superconductors, Italy (engineering integration); Sea Alp, Italy (winding tooling), Dalkia-Veolia, France (site and infrastructure); Elytt Energy, Spain and Alsyom/SEIV, France (handling and impregnation); CNIM, France (manufacturing and cold testing)

Central Solenoid

Allows a powerful current to be induced in the ITER plasma and maintained during long plasma pulses. 13 m tall (18 m with structure), 4m wide and 1ktonnes weight, Six independent coil packs wound from niobium-tin superconducting cable.

Stored magnetic energy of 6.4 GJ in the central solenoid will initiate and sustain a plasma current of 15 MA for durations of 300-500 seconds. Maximum field of 13 T in the centre of the stacked modules.

The independently operating coil packs will create large electromagnetic forces that pull in different directions. Ir order to maintain the structural integrity of the central solenoid assembly, a special support structure has been designed. The support structure will have to withstand forces in the range of 100 MN, or over 10,000 tonnes of force.

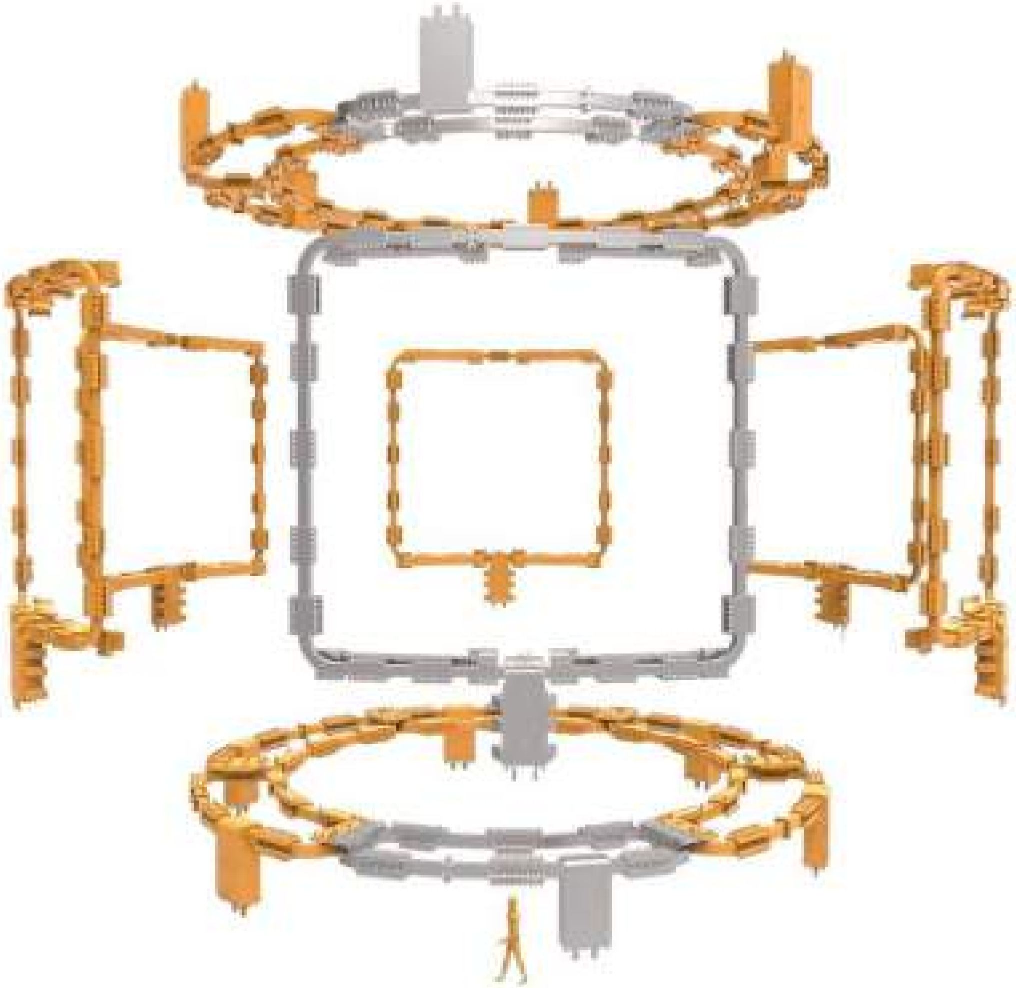
100 MN

30 MN

Correction Coils

Eighteen superconducting correction coils inserted between the toroidal and poloidal field coils will compensate for field errors caused by geometrical deviations due to manufacturing and assembly tolerances. Although much lighter and thinner than the toroidal and poloidal field coils—and running a smaller current (10 kA)—the correction coils measure up to 8 metres in width and present particular challenges for assembly and installation.

The correction coils will be arranged in groups of six around the toroidal circumference above, at and below the mid-plane of the vacuum vessel.



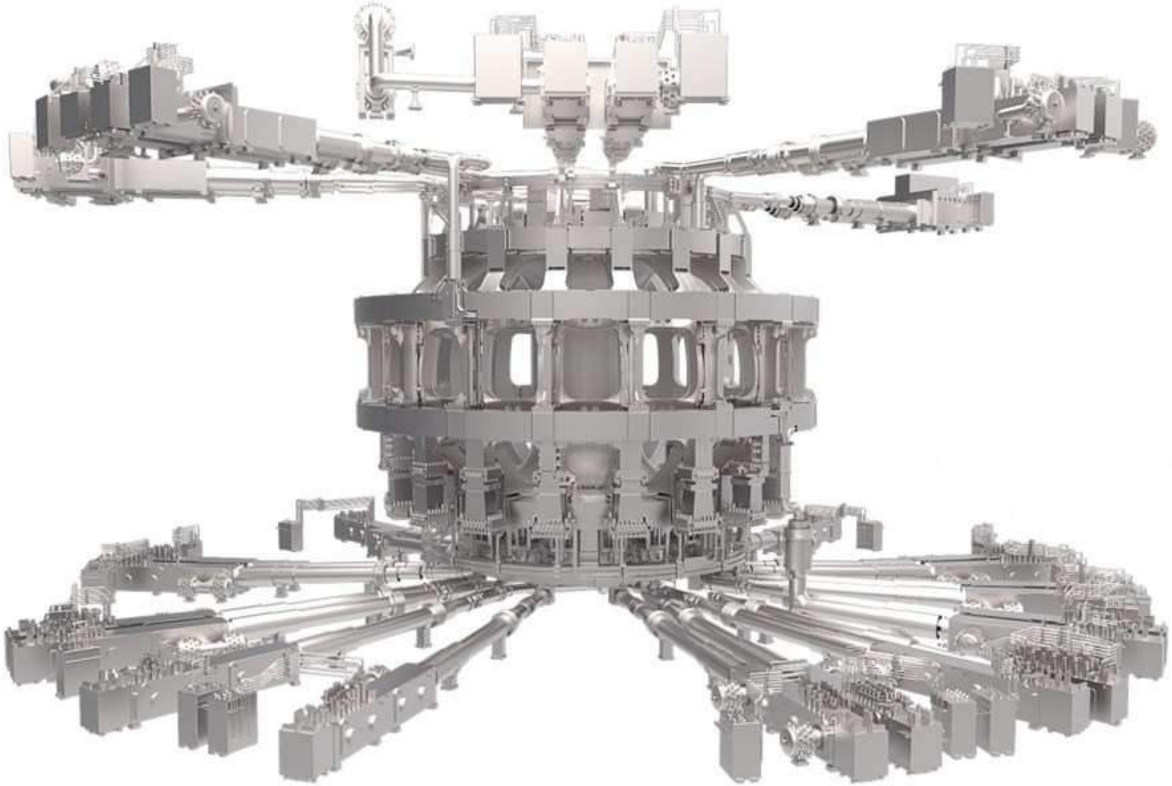
Magnet Feeders

Magnet feeders convey and regulate the cryogenic liquids to cool and control the temperature of the magnets and connecting the magnets to their power supplies.

31 superconducting feeders will relay electrical power and cryogens through the warm-cold barrier to the ITER magnets.

The design operating current of the feeders is 68kA. High temperature superconductor (HTS) current leads transmit the high-power currents from the room-temperature power supplies to the low-temperature superconducting coils 4K with minimum heat load.

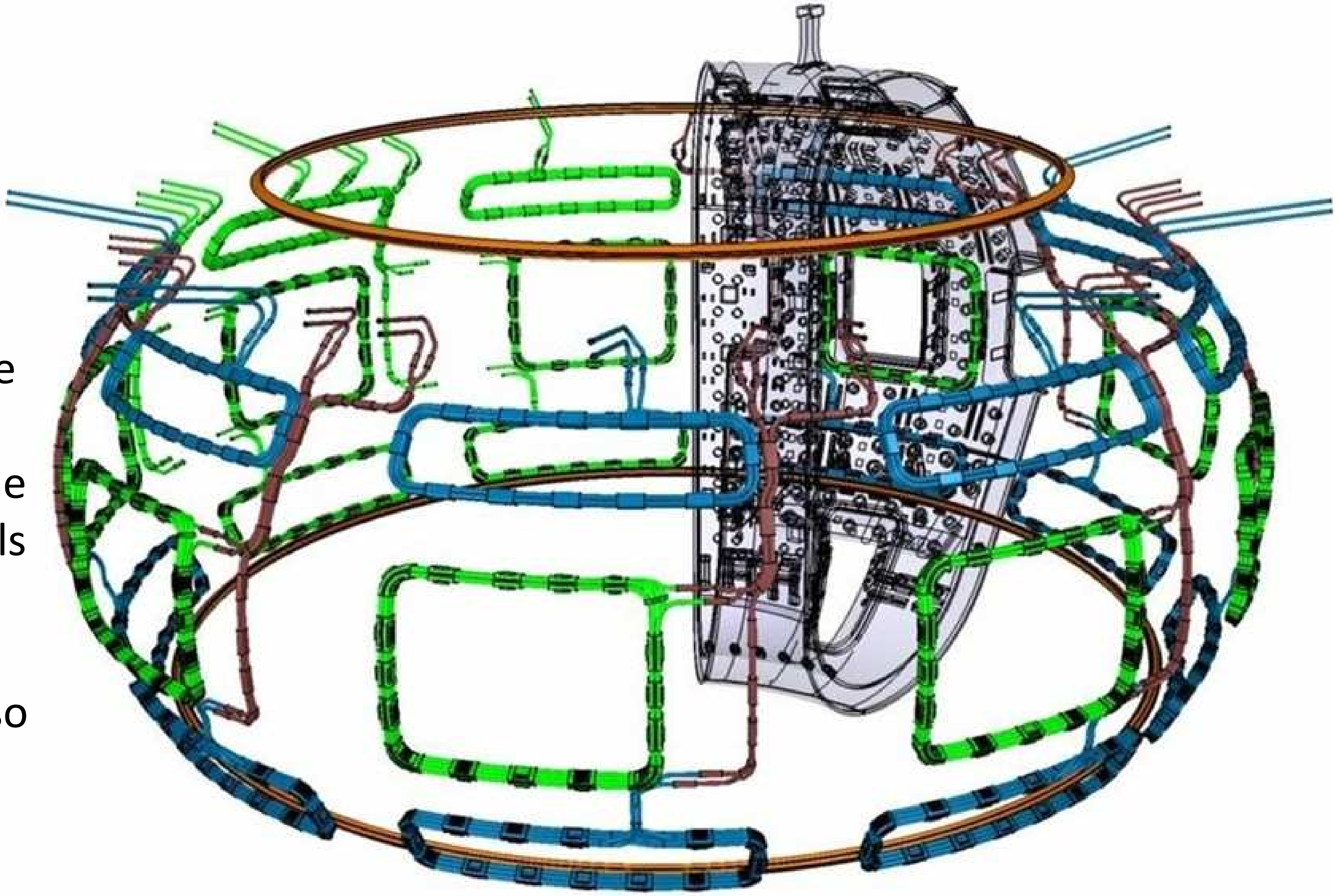
Superconducting busbars, made out of steel conduit containing niobium-titanium superconductor cable, are designed to absorb the large temperature variations during the cool-down of the machine.



In Vessel Coils

Two non-superconducting coil systems inside of the ITER vacuum vessel provide additional plasma control capabilities.

Two vertical stability coils installed above and below the machine's mid-plane provide fast vertical stabilization of the plasma; another set of 27 coils fixed to the wall of the vessel create resonant magnetic perturbations in the plasma so that certain types of plasma instabilities, called Edge-Localized Modes (ELMs), are avoided.



Vacuum Vessel

The ITER experiments will take place inside the vacuum vessel, a hermetically sealed steel container that houses the fusion reactions and acts as a first safety containment barrier. The vacuum vessel provides a high-vacuum environment for the plasma, improves radiation shielding and plasma stability, acts as the primary confinement barrier for radioactivity, and provides support for in-vessel components such as the blanket and the divertor. Cooling water circulating through the vessel's double steel walls will remove the heat generated during operation.

44 openings, or ports, in the vacuum vessel provide access for remote handling operations, diagnostics, heating, and vacuum systems.

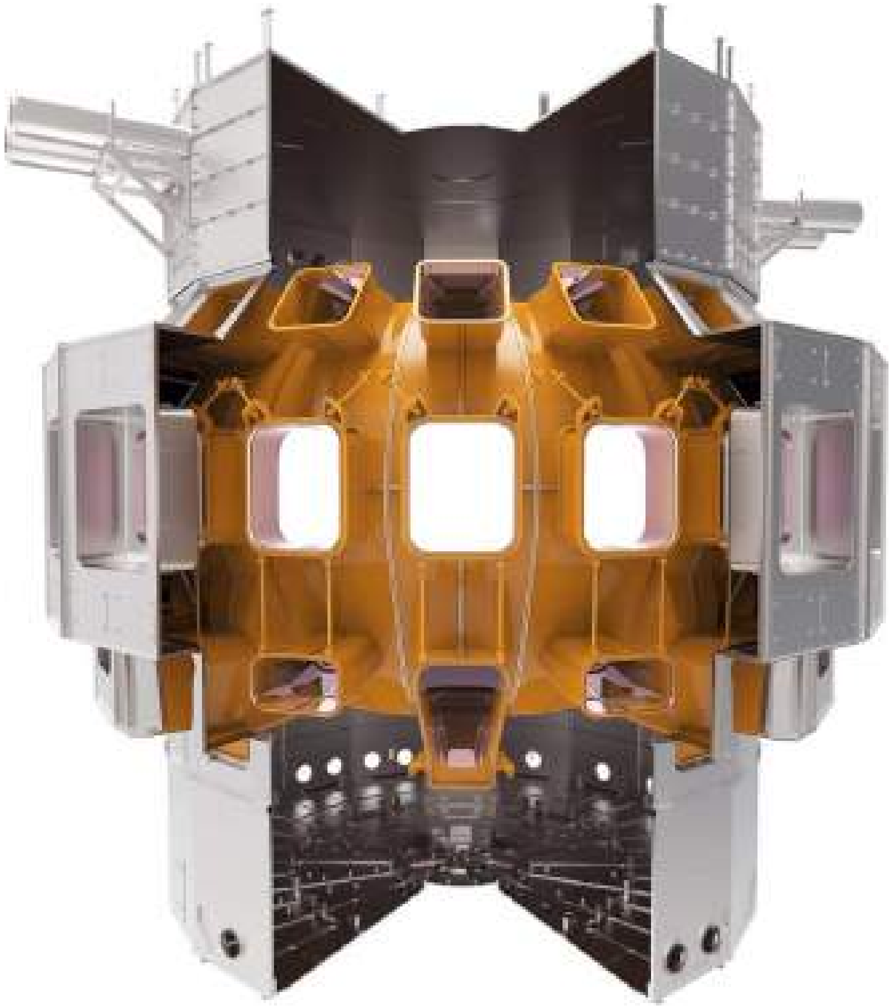
Blanket modules will provide shielding from the high-energy neutrons produced by the fusion reactions.

The ITER vacuum vessel, with an interior volume of 1,400 m^3: the volume of the plasma contained in the centre of the vessel (840 m^3) is fully ten times larger than that of the largest operating tokamak in the world today.

Thermal Shielding

The ITER vacuum vessel will measure 19.4 metres across (outer diameter), 11.4 metres high, and weigh approximately 5,200 tonnes.
With the installation of the blanket and the divertor, the vacuum vessel will weigh 8,500 tonnes

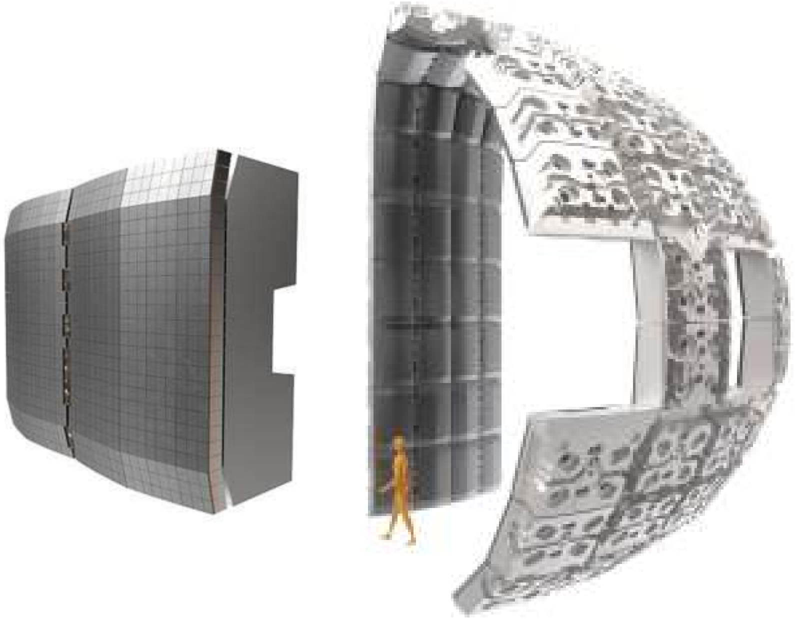
Two layers of thermal shielding are interposed between the vacuum vessel and the cryostat to minimize heat loads transferred by thermal radiation and conduction from warm components to the components and structures that operate at 4.5K (such as the magnets).



Blanket Modules

The 440 blanket modules completely cover the inner walls of the VV to protect the steel structure and the superconducting TF magnets from the heat and high-energy neutrons produced by the fusion reactions.
As the neutrons are slowed in the blanket, their kinetic energy is transformed into heat energy and collected by the water coolant. In a fusion power plant, this energy will be used for electrical power production.

Due to its unique physical properties (low plasma contamination) beryllium has been chosen as the element to cover the first wall. The rest of the blanket modules will be made of high-strength copper and stainless steel.
First fusion device to operate with an actively cooled blanket. The cooling water—injected at 4 MPa and 70 °C—is designed to remove up to 736 MW of thermal power.
During later stages of ITER operation, some of the blanket modules will be replaced with specialized modules to test materials for tritium breeding concepts.



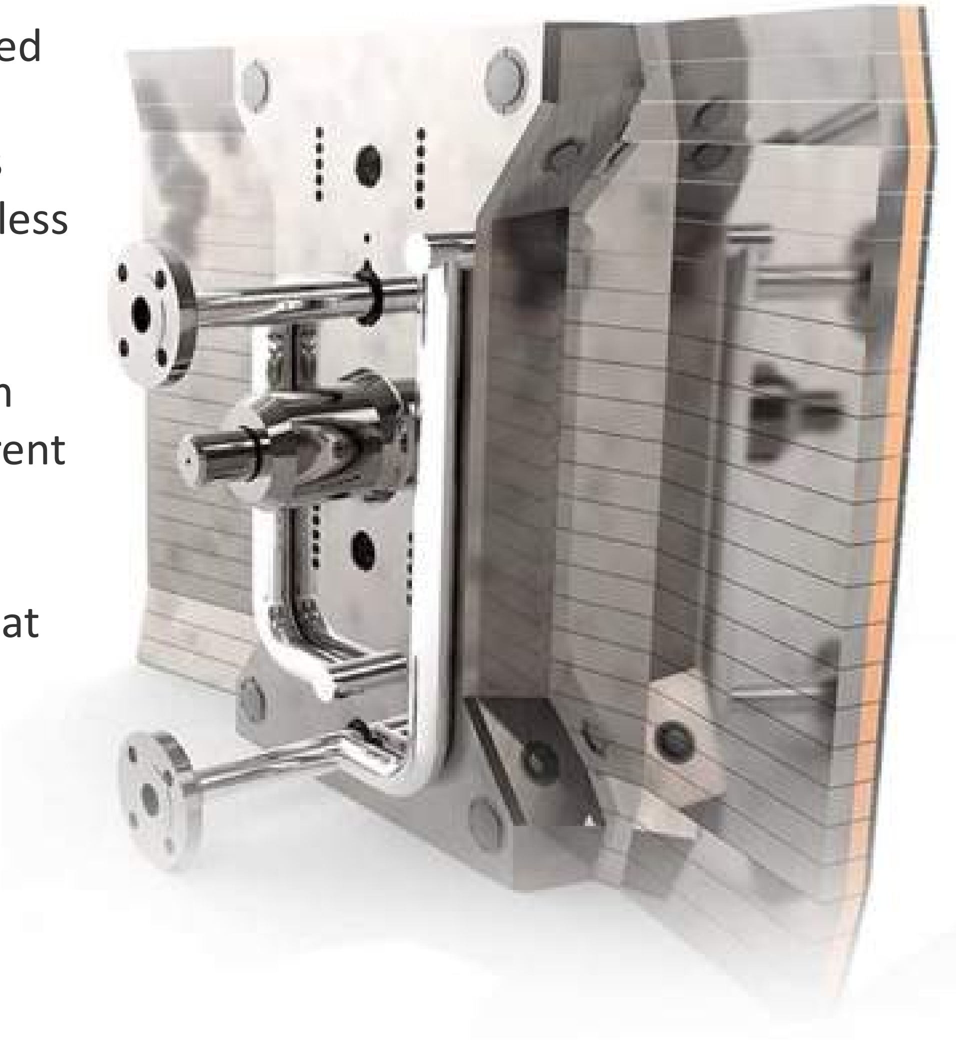
Covers a surface of 600 m²

First Wall

The first wall panels are the detachable, front-facing elements of the blanket that are designed to withstand the heat flux from the plasma. These components are made of beryllium tiles bonded with a copper alloy and 316L (N) stainless steel.

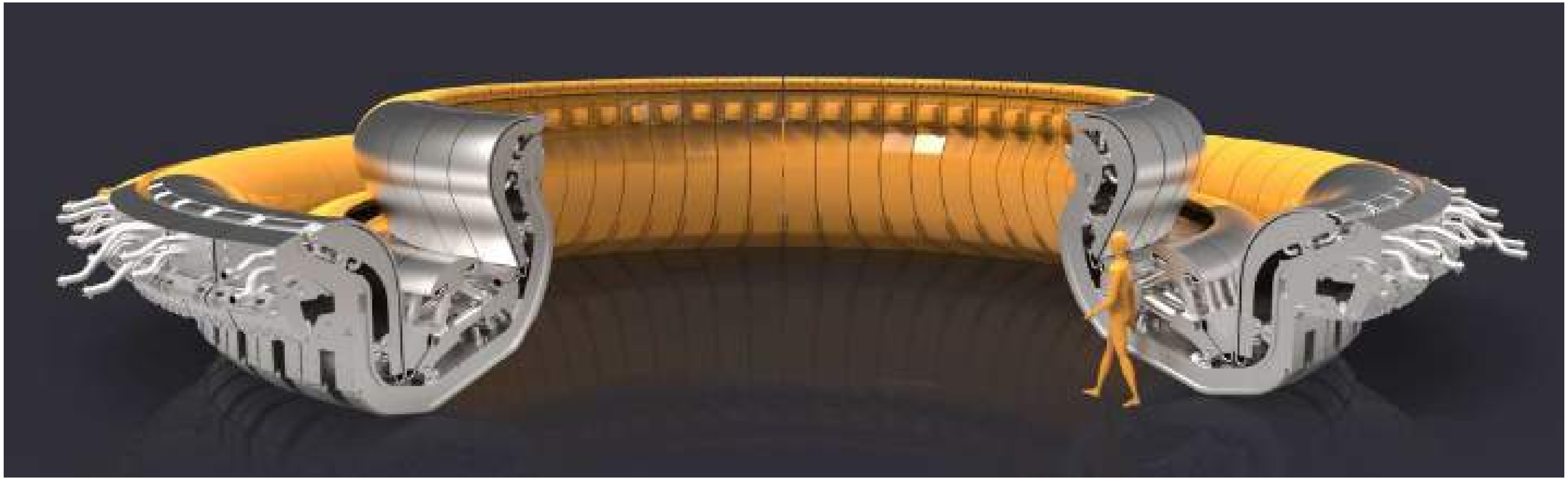
Depending on their position inside the vacuum vessel the first wall panels are subject to different heat fluxes.
Two different kinds of panels have been the object of a multi-year qualification program that has included the fabrication of semi- and full-scale mockups and testing:

- a normal heat flux panel designed for heat fluxes of up to 2 MW/m^2
- an enhanced heat flux panel that can withstand heat fluxes of up to 4.7 MW/m^2.



Divertor

Situated at the bottom of the vacuum vessel, the divertor extracts heat and ash produced by the fusion reaction, minimizes plasma contamination, and protects the surrounding walls from thermal and neutronic loads.
54 "cassette assemblies" with supporting structure in stainless steel and three plasma-facing components: the inner and outer vertical targets and the dome. The cassette assemblies also host a number of diagnostic components for plasma control and physics evaluation and optimization.



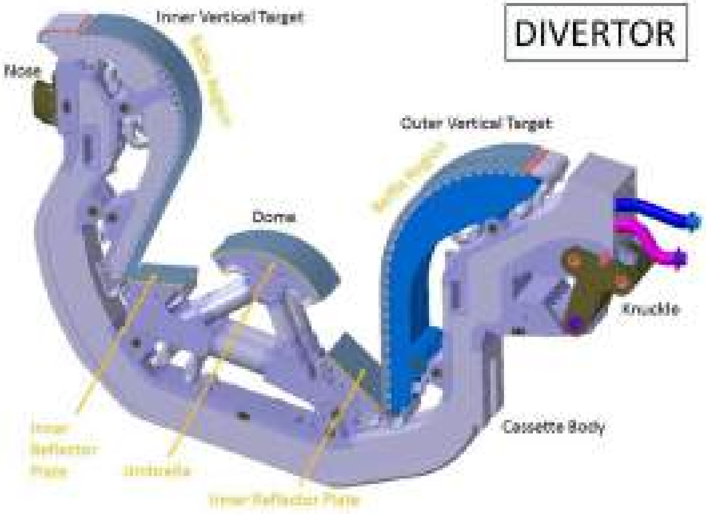
Divertor Targets

The heat flux sustained by the ITER divertor vertical targets is estimated at 10 MWm² (steady state) and 20 MWm² (transients). Tungsten, with the highest melting point of all the metals, has been chosen as the armour material following an international R&D effort, encouraging experimental results, and successful prototype testing.

The 54 ten-tonne cassette assemblies of the ITER divertor will be installed—and also replaced at least once during the machine's lifetime—by remote handling.

2 MWm²

20 MWm²



Cryostat

The ITER cryostat—the largest stainless steel high-vacuum pressure chamber ever built (16,000 m³)—provides the high vacuum, ultra-cool environment for the ITER vacuum vessel and the superconducting magnets.

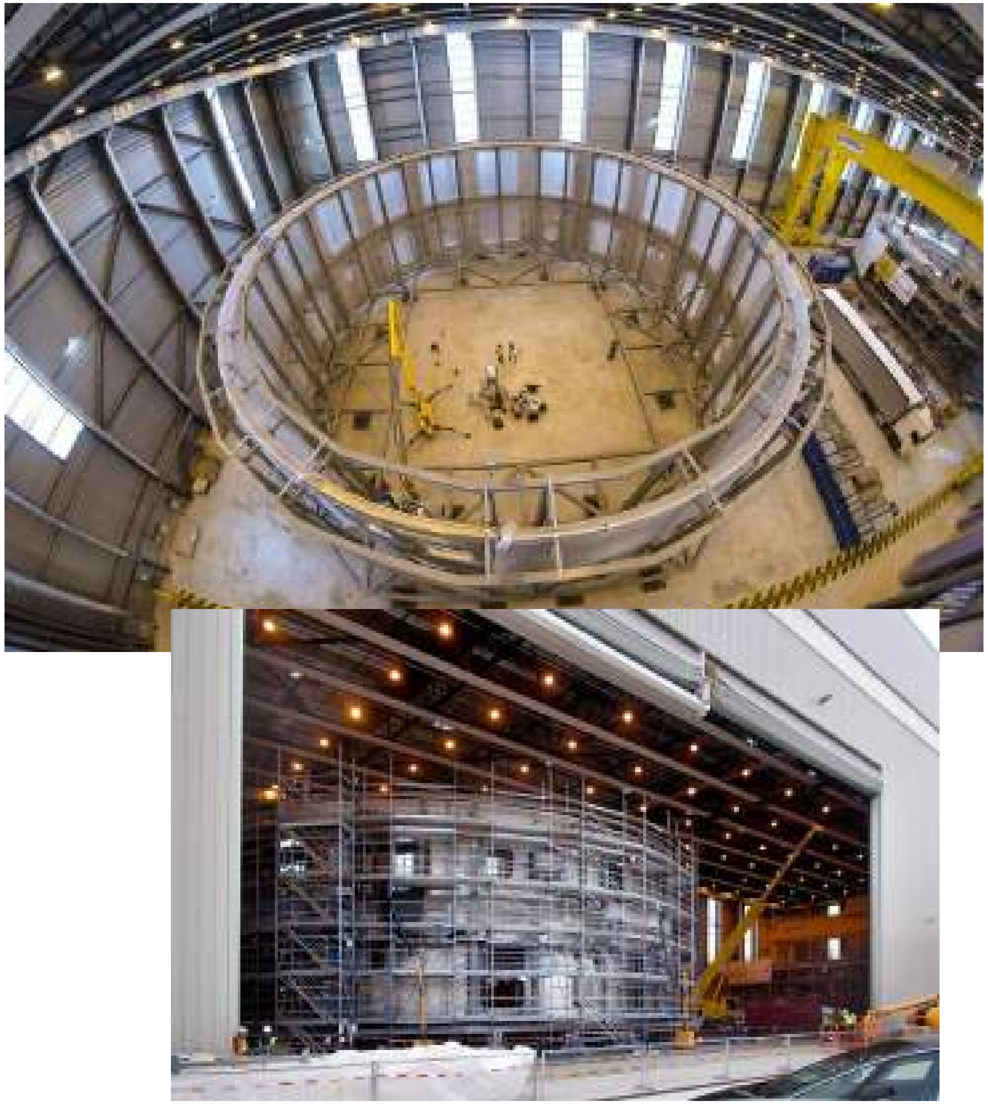
Nearly 30 metres wide and as many in height

Manufactured from stainless steel, the cryostat weighs 3,850 tonnes. Its base section—1,250 tonnes—will be the single largest load of ITER Tokamak assembly.

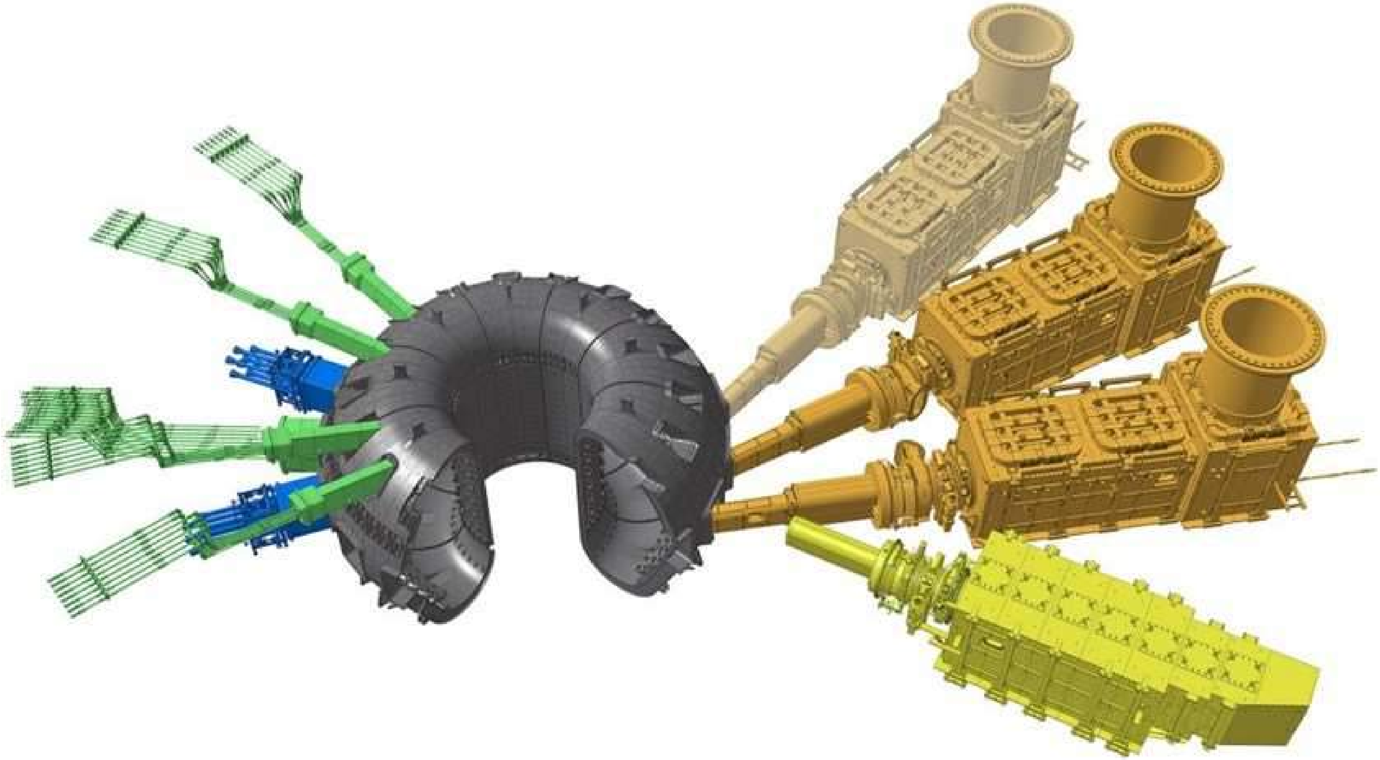
Cryostat

The cryostat has 23 penetrations to allow access for maintenance as well as over 200 penetrations—some as large as four metres in size—that provide access for cooling systems, magnet feeders, auxiliary heating, diagnostics, and the removal of blanket sections and parts of the divertor.

Large bellows situated between the cryostat and the vacuum vessel will allow for thermal contraction and expansion in the structures during operation. The structure will have to withstand a vacuum pressure of 1 x 10 -4 Pa; the pump volume is designed for 8,500 m^3



External Heating Systems

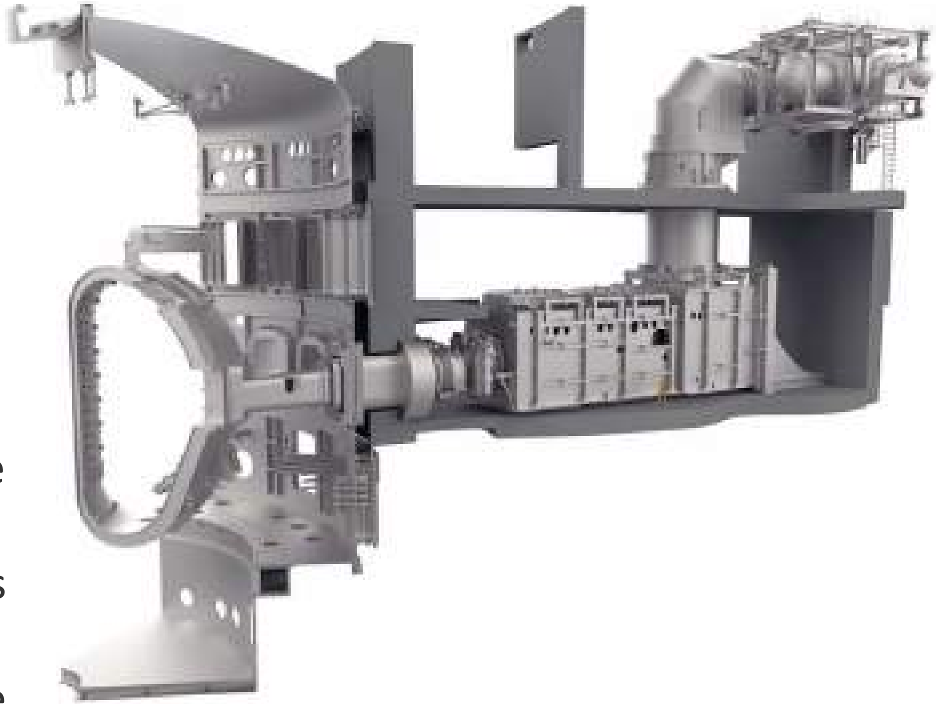


The ITER Tokamak will rely on three sources of external heating to bring the plasma to the temperature necessary for fusion: neutral beam injection (right) and two sources of high-frequency electromagnetic waves—ion and electron cyclotron heating (left, blue and green launchers).

Neutral Beam Injection

NBI are used to shoot uncharged high-energy particles into the plasma where, by way of collision, they transfer their energy to the plasma particles.

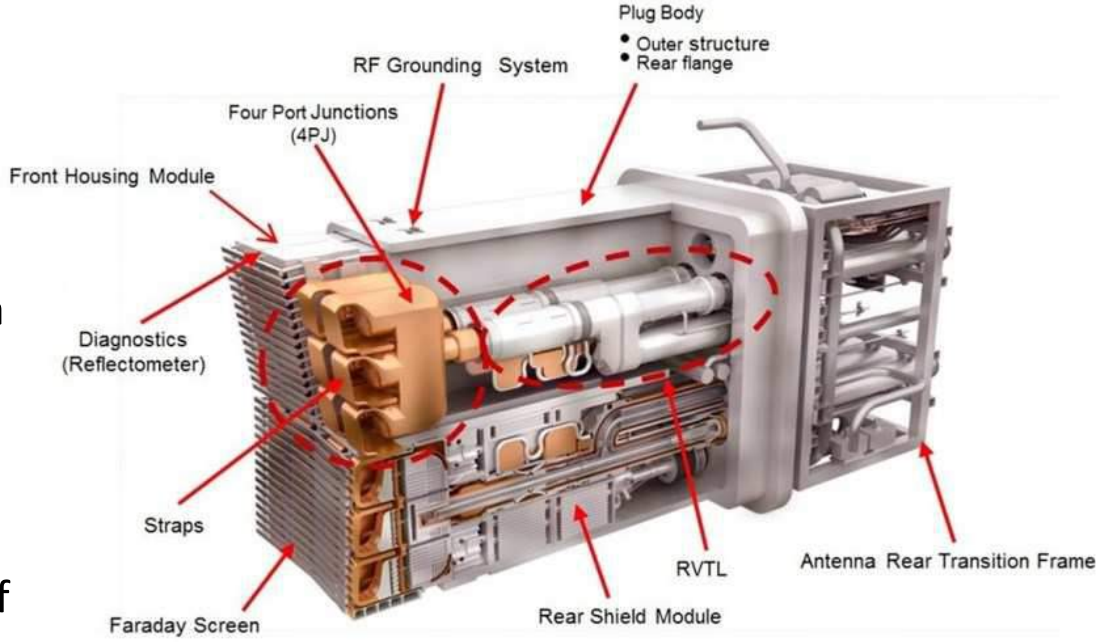
Before injection, deuterium atoms must be accelerated outside of the tokamak to a kinetic energy of 1 Mega electron Volt (MeV).
Only atoms with a positive or a negative charge can be accelerated by electric field; for this, electrons must be removed from neutral atoms to create a positively-charged ion. The process must then be reversed before injection into the fusion plasma; otherwise the electrically-charged ion would be deflected by the magnetic field of the plasma cage.
In neutral beam injection systems, the ions pass through a cell containing gas where they recover their missing electron and can be injected as fast neutrals into the plasma.



A test program is underway now at the Neutral Beam Test Facility in Padua, Italy to investigate challenging physics and technology issues in advance of the installation of the neutral beam equipment at ITER. 16.5MW

Ion cyclotron

Ion and electron cyclotron heating methods use radio waves at different frequencies to bring additional heat to the plasma, much in the same way that a microwave oven transfers heat to food through microwaves.
In ion cyclotron resonance heating (ICRH), energy is transferred to the ions in the plasma by a high-intensity beam of electromagnetic radiation with a frequency of 40 to 55 MHz.



Two 45-ton ion cyclotron resonant heating antennas will deliver 10 MW of heating power each into the ITER machine.

Electron Cyclotron Resonance Heating

ECRH heats the electrons in the plasma with a high-intensity beam of electromagnetic radiation at a frequency of 170 GHz, the resonant frequency of electrons. The electrons in turn transfer the absorbed energy to the ions by collision.

The electron cyclotron heating system is also used to deposit heat in very specific places in the plasma, as a mechanism to minimize the build-up of certain instabilities that lead to cooling of the plasma.

In comparison to the ICRH system, the ECRH has the advantage that the beam can be transmitted through air which simplifies the design and allows the source to be far from the plasma, simplifying maintenance.

Each of ITER's 24 gyrotrons will generate a microwave beam over a thousand times more powerful than a traditional microwave oven. Gyrotrons for ITER's electron cyclotron system are under development in Europe, India, Japan and Russia.

CODAC

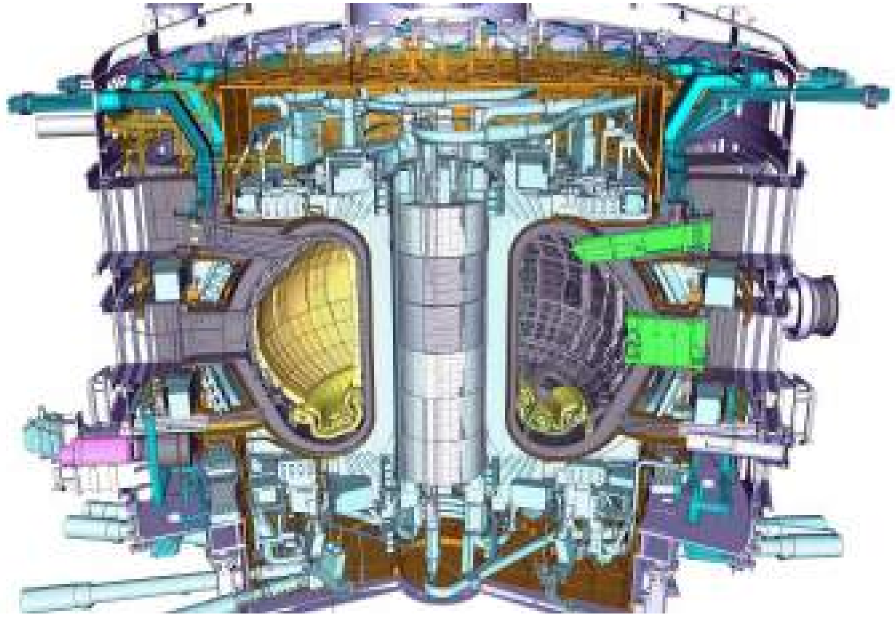
The ITER **CODAC** (Control, Data Access and Communication) system can be thought of as the brain and central nervous system of ITER. It physically connects all plant systems using computer networks and makes sure they speak the same language.

CODAC interfaces to more than 30 different plant systems which are developed and procured by the ITER partners. These plant systems generate and exchange an estimated total number of about 1,000,000 signals.

In addition to enable communication and integration, CODAC provides a number of other services essential to ITER operation. These include applications for the central supervision and orchestration, the **Plasma control system** and the central data archiving.

Controllare un tokamak come un velivolo

Multivariabilità
Nonlinearità
Incertezze
Presenza di disturbi
Safety critical application

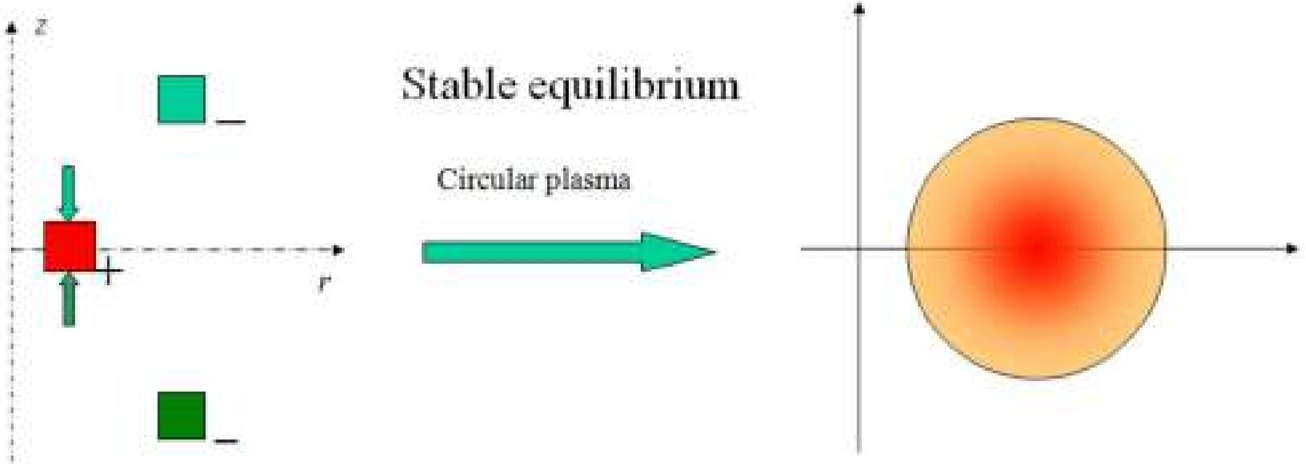


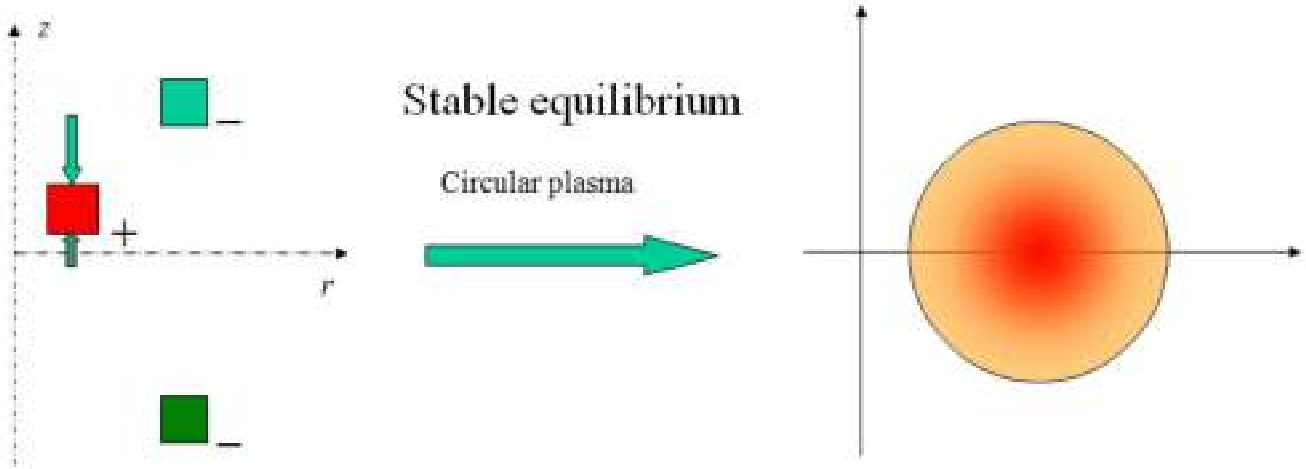
Stabilizzazione
Controllo gerarchico
Inseguimento di traiettoria
Problemi di stima dello stato
Disaccoppiamento delle dinamiche

Principi di controllo magnetico

Simplified electromechanical model with three conductive rings, two rings are kept fixed and in symmetric position with respect to the r axis, while the third can freely move vertically.

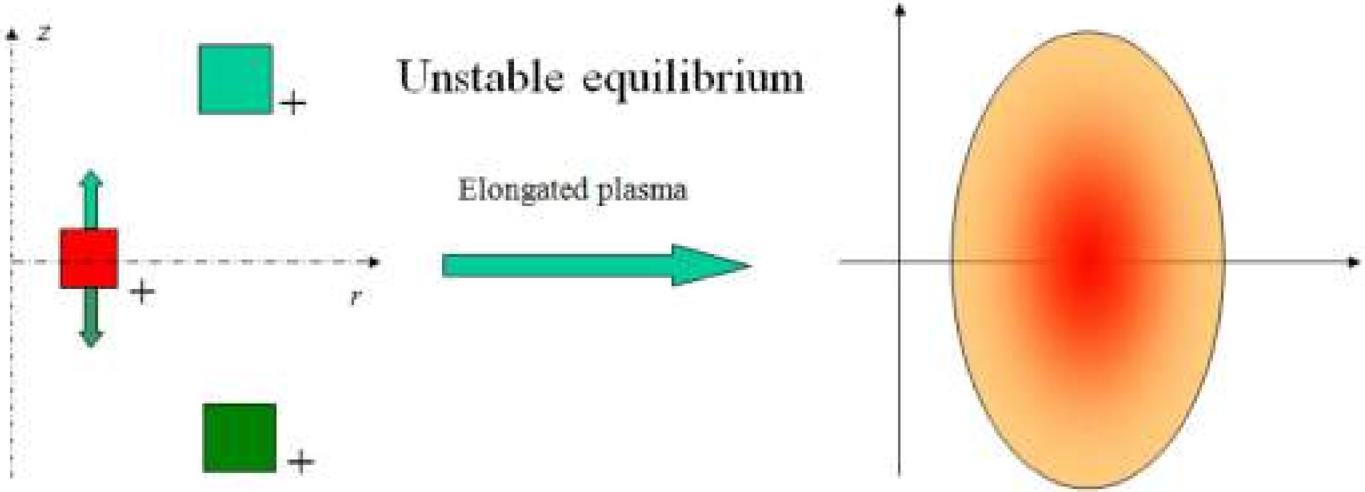
If the currents in the two fixed rings are equal, the vertical position z = 0 is an equilibrium point for the system.

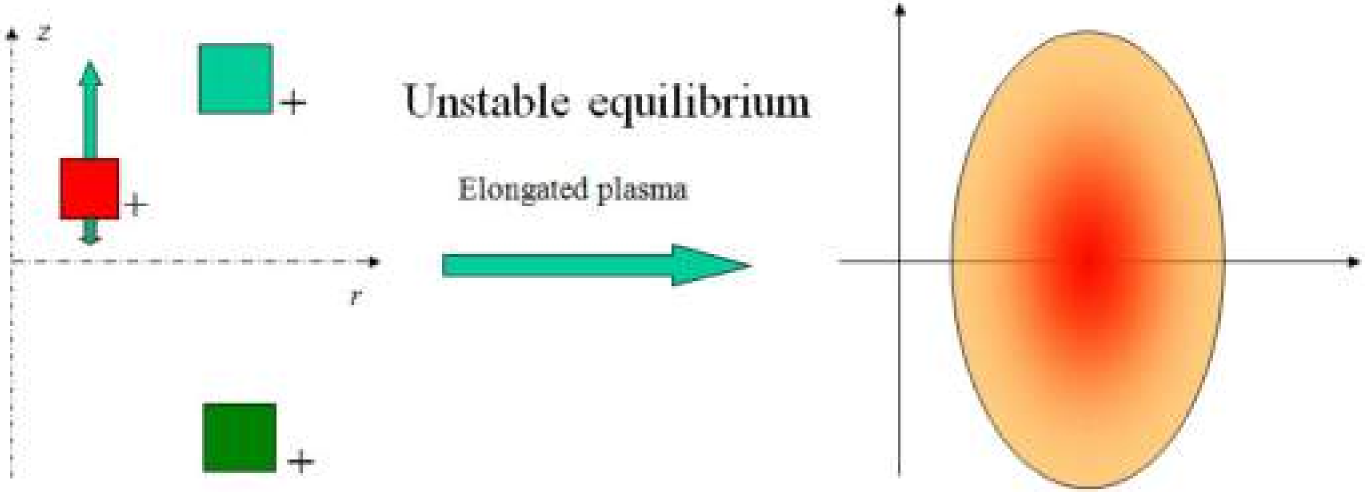




Principi di controllo magnetico

On the contrary… the vertical stability problem

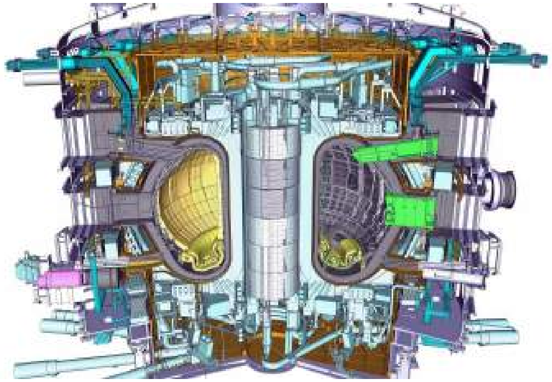


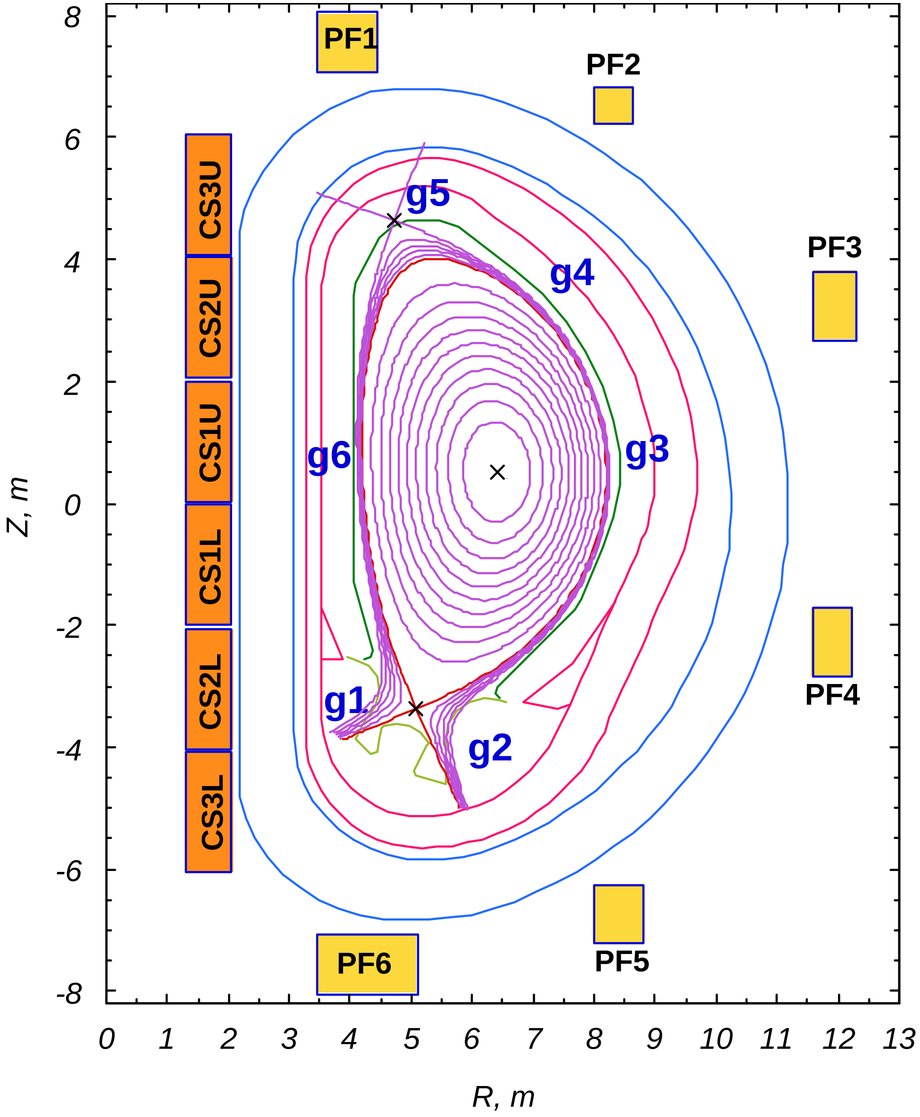


Attuatori

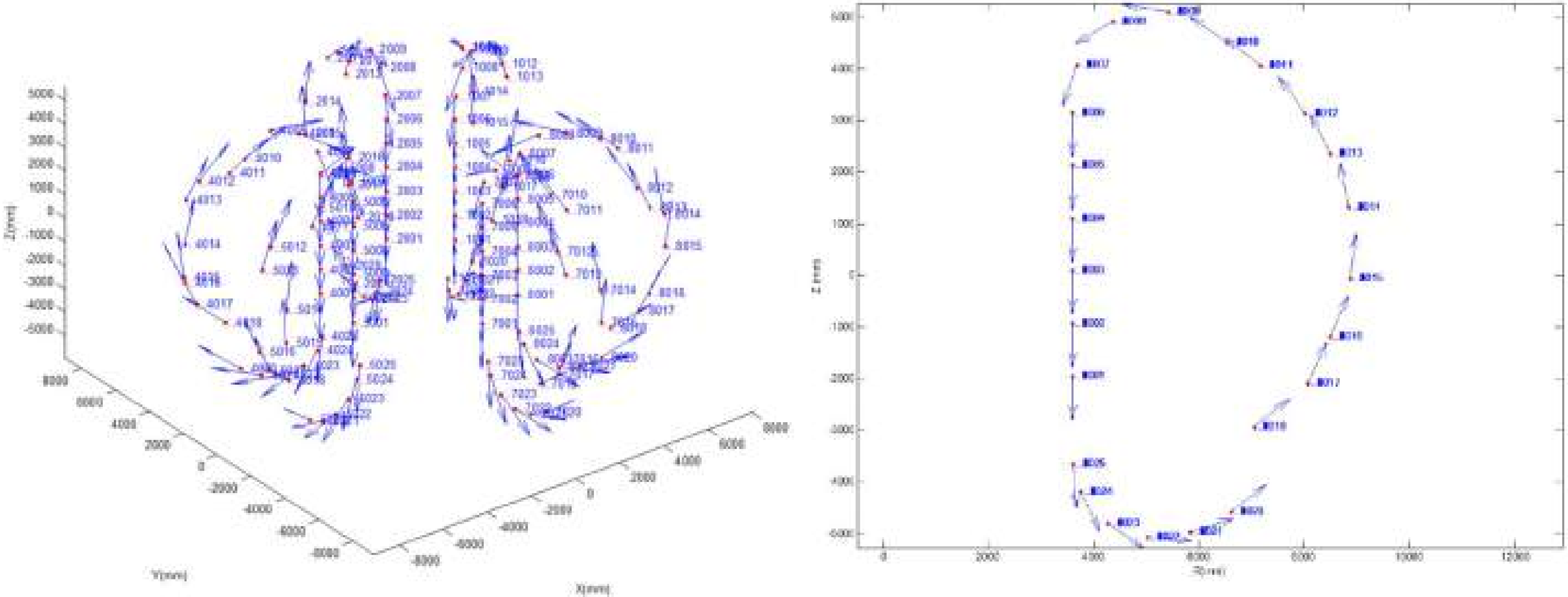
Alcuni problemi del controllo magnetico assisimmetrico

- Stabilizzazione verticale
- Controllo di forma
- Controllo di Ip
- Ricostruzione delle grandezze da controllare



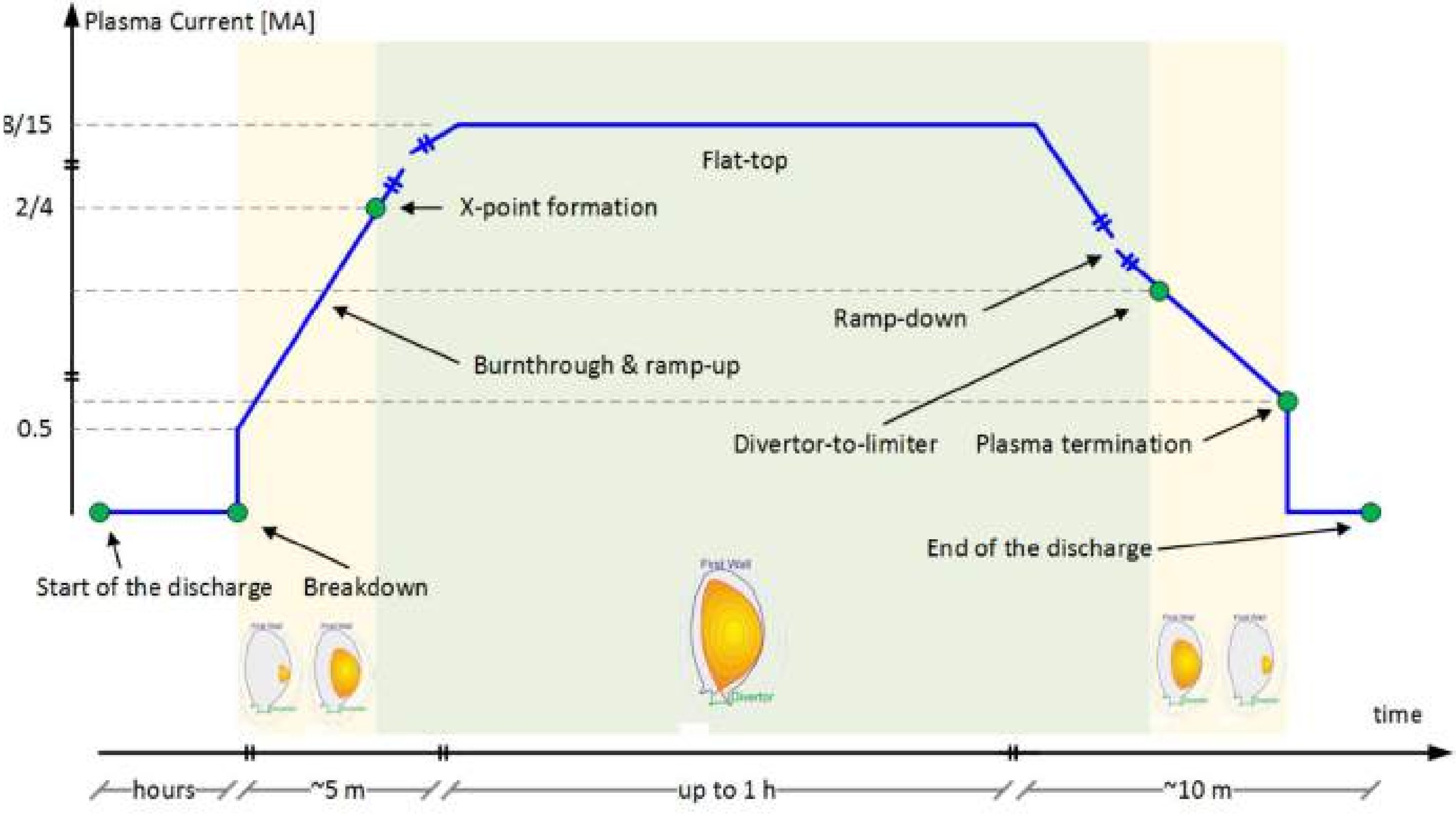


Sensori



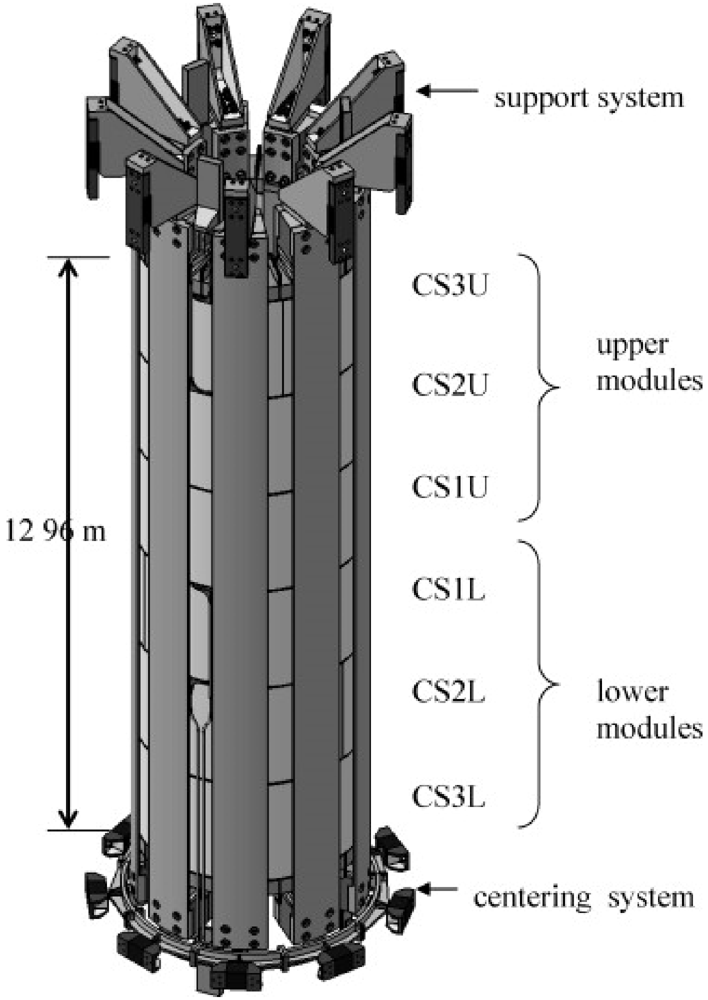
Positions of the AA probes in the ITER tokamak, according to the current design.
The AA probes measure the magnetic field in six different sectors.
On the right side the arrangement of the probes on one sector is shown.

La missione di volo di un tokamak

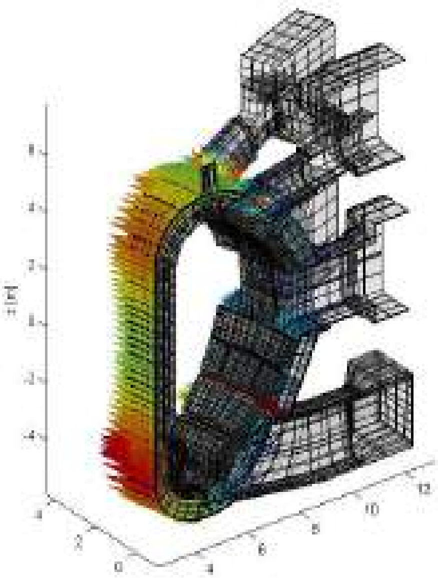


Carica il solenoid centrale

Current Control per immagazzinare Energia



Start of Discharge – Take Off

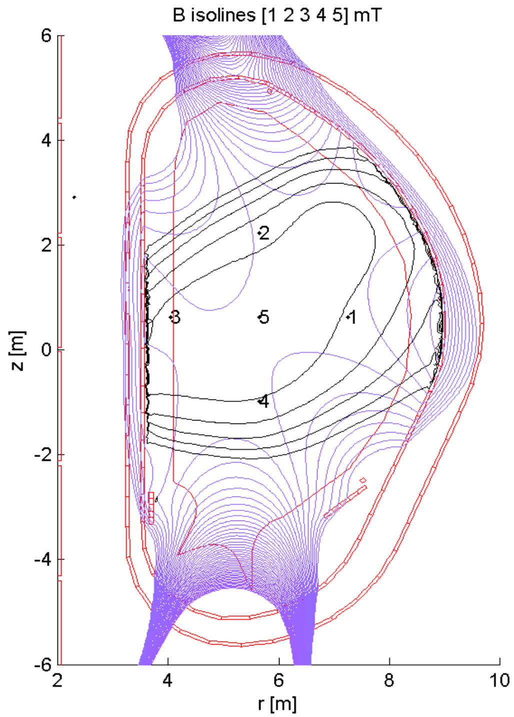


Utilizzo di Switching Networks

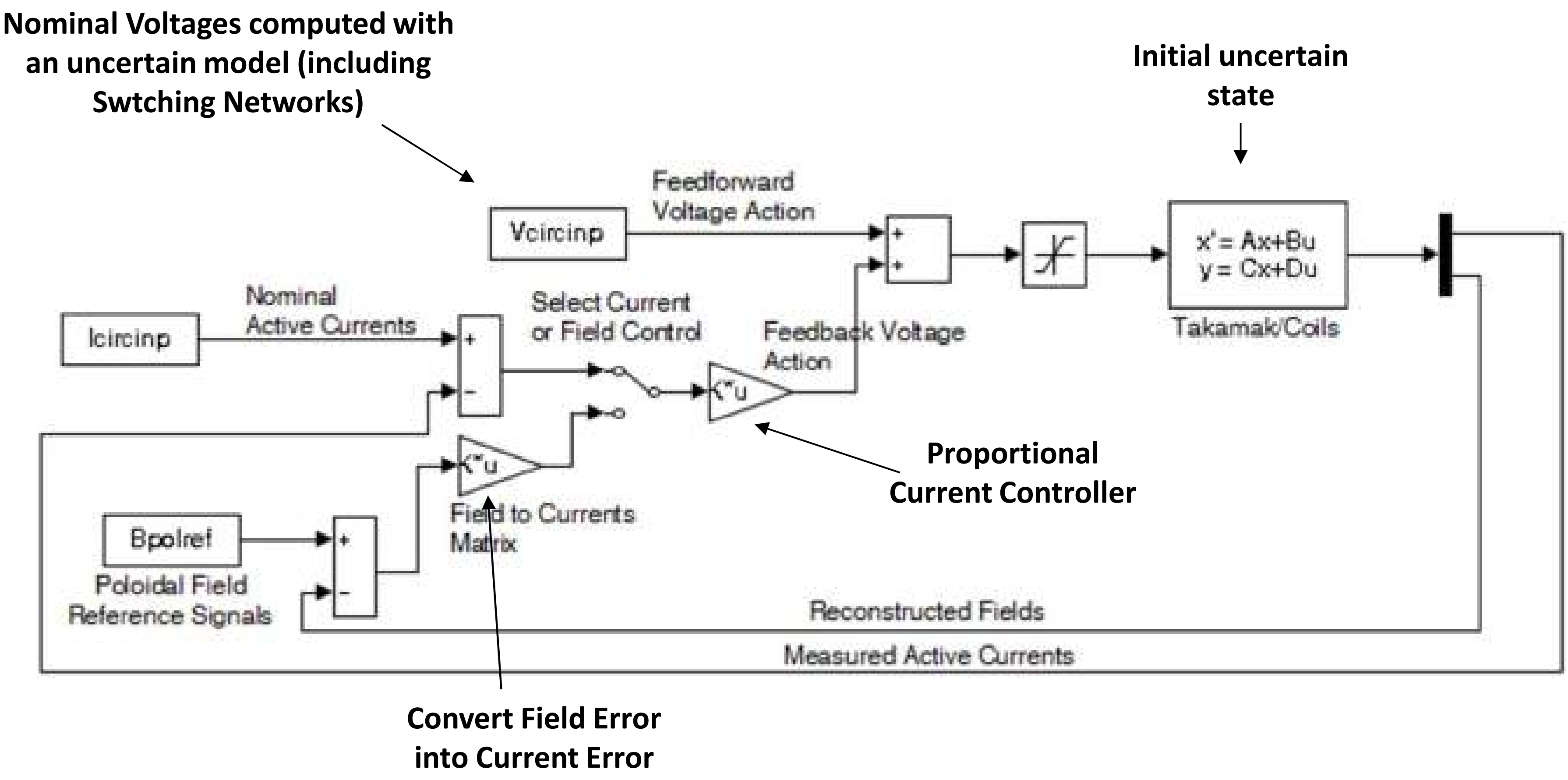
Elevate correnti nei passivi

Basso campo magnetico ed elevato campo elettrico

Utilizzo di sistemi in radiofrequenza ECRH



Controllo del BD in feedback


Nominal Voltages computed with an uncertain model (including Swtching Networks)

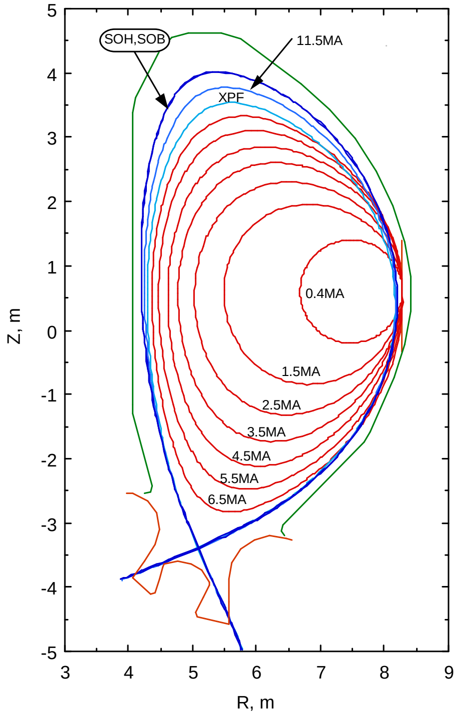
Initial uncertain state

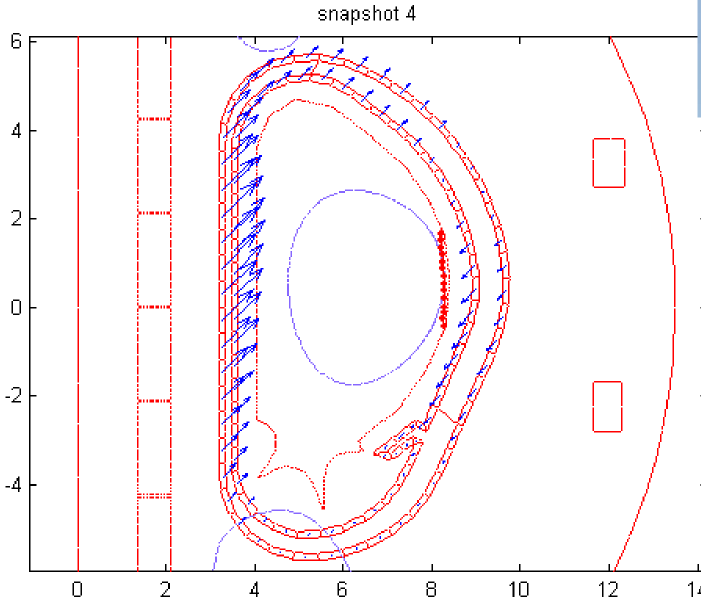
Proportional Current Controller

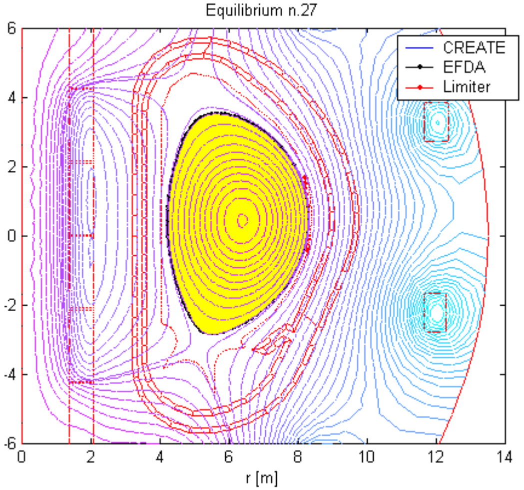
Convert Field Error into Current Error

Salita Corrente e XPF

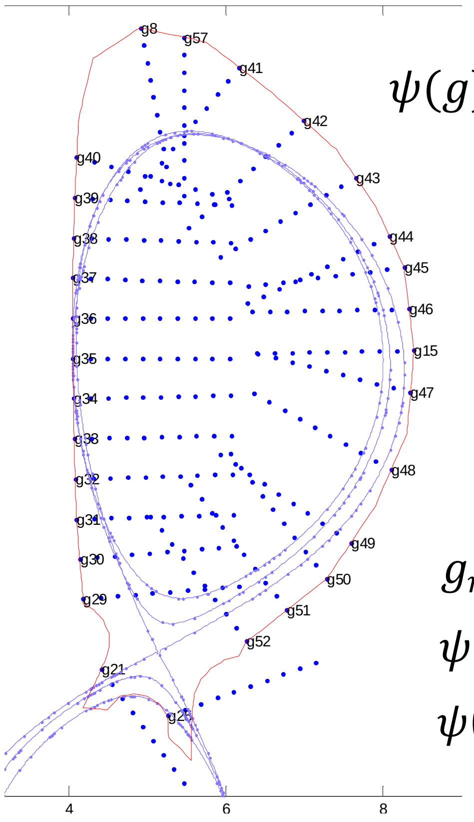
- rampa di corrente di trasformatore per dIp/dt
- aumento del campo verticale per sostenere l'equilibrio
- campo di quadrupolo per l'elongazione e l'Xpoint formation
- configurazione variabile







Flat Top - CROCIERA

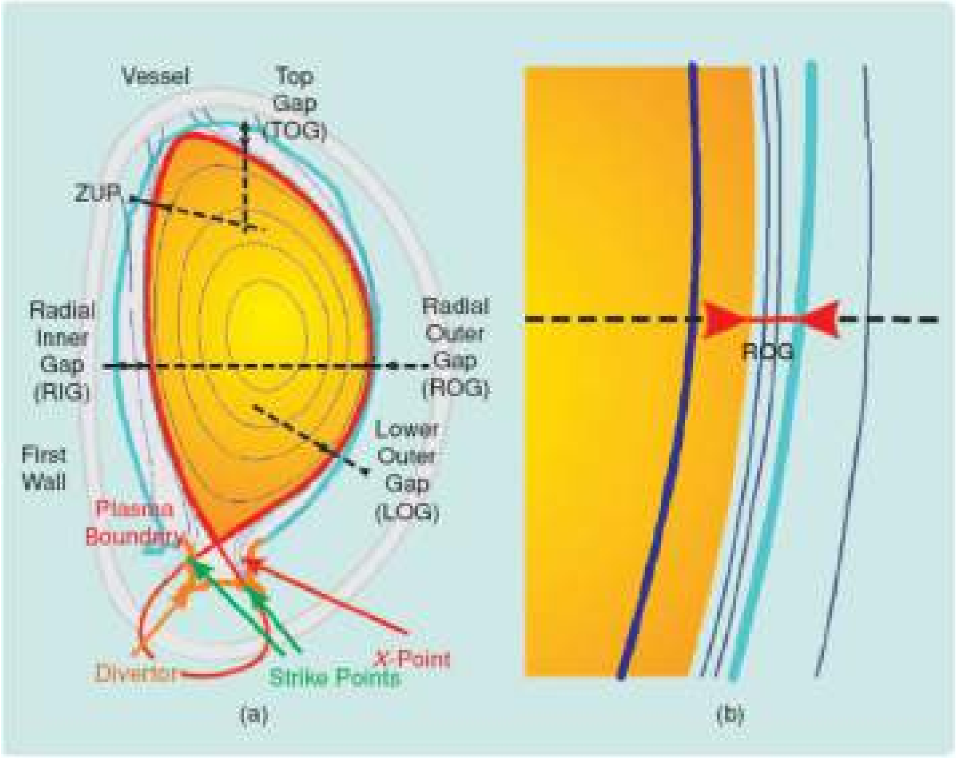


$$\psi(g) = \psi_B$$

$$g_{ref} - g \quad = 0$$

$$\psi(g_{ref}) - \psi_L = 0$$

$$\psi(g_{ref}) - \psi_X = 0$$

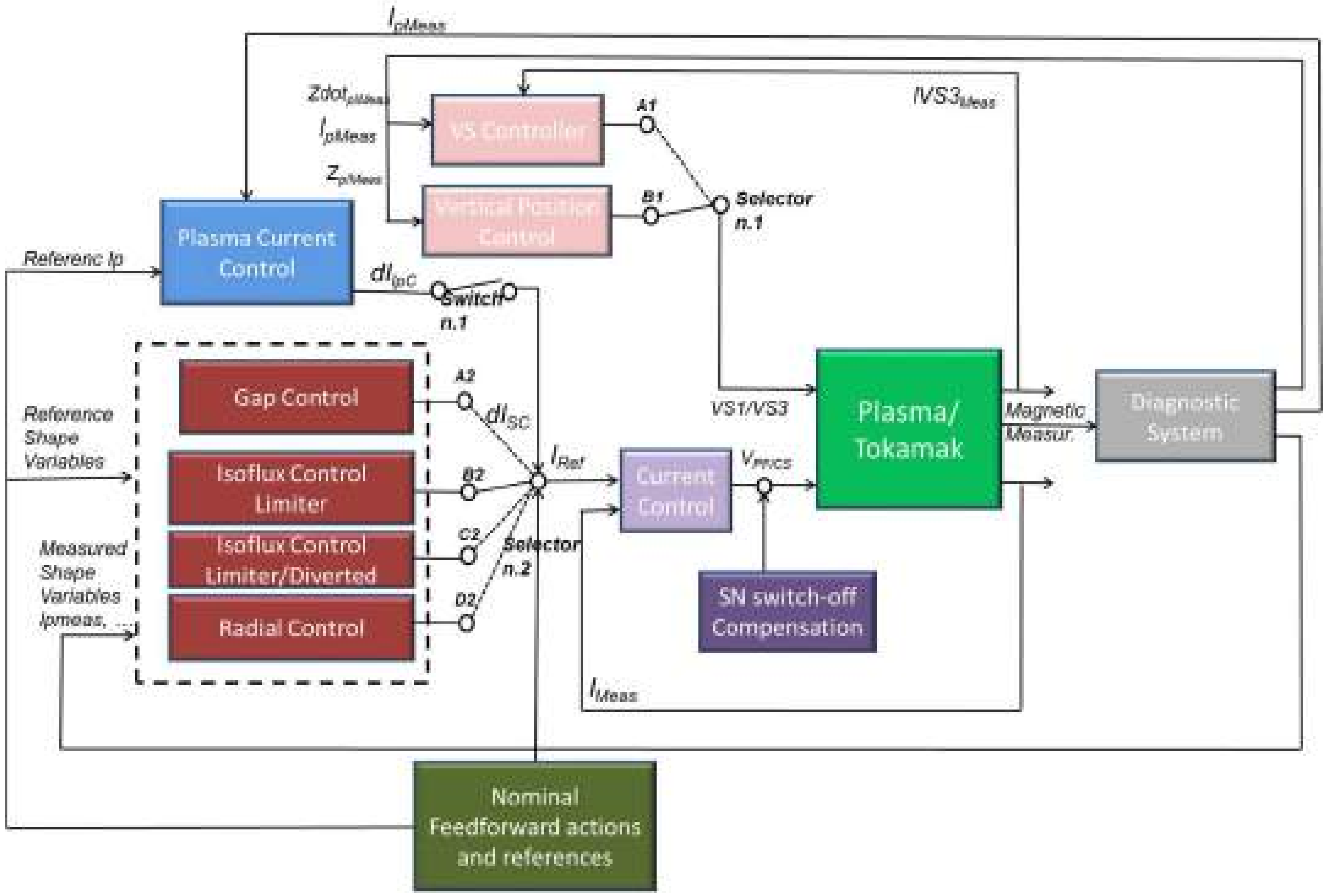


A set of plasma shape descriptors
on 29 control segments

Disturbances and strong
parametric variations

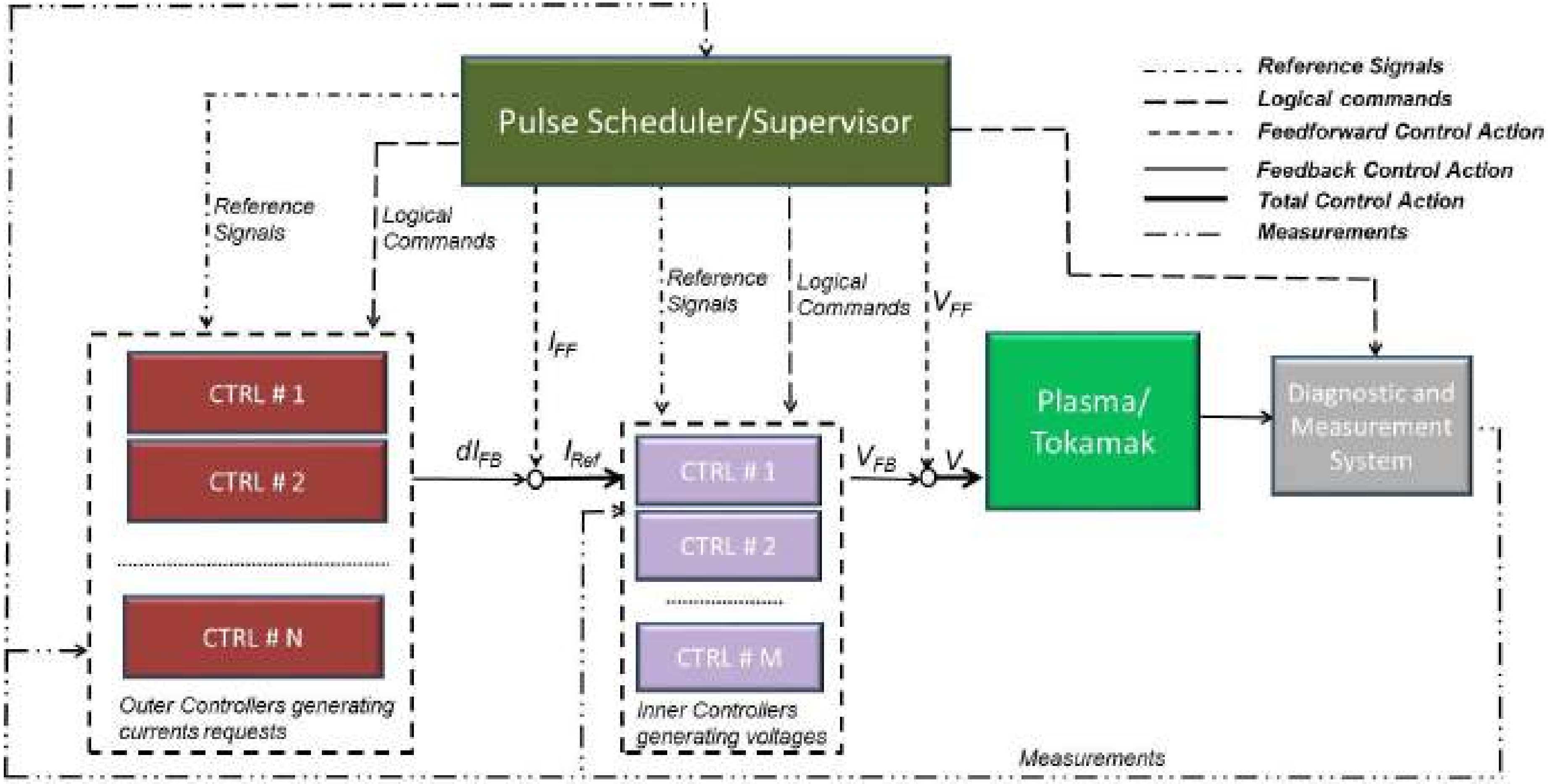
Schema di controllo magnetico ITER

General Magnetic Control Scheme for Ramp up



Schema di controllo magnetico ITER

Magnetic Control general architectural scheme

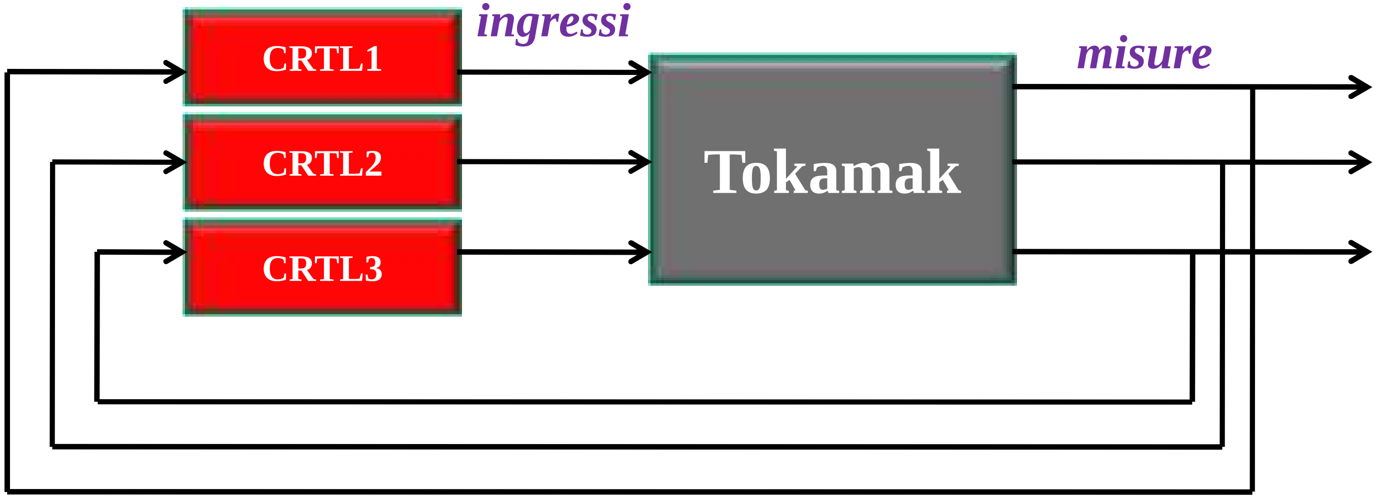


Controllo Classico (PID – azioni ritardo e anticipo SISO)

☺ **Semplicità di progettazione e di implementazione**

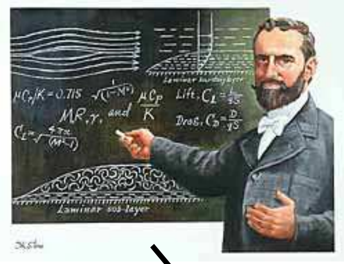
☹ **Necessita di disaccoppiamento e scheduling**

☹ **Difficoltà di portare in conto in fase di progetto ottimalità, robustezza, reiezione ai disturbi**

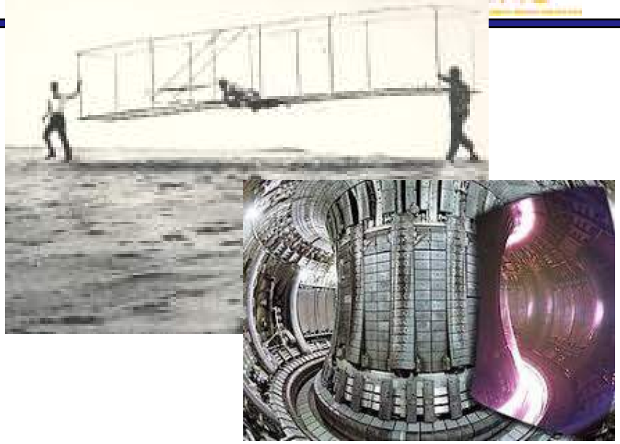


Approccio Model Based

Leggi Fisiche

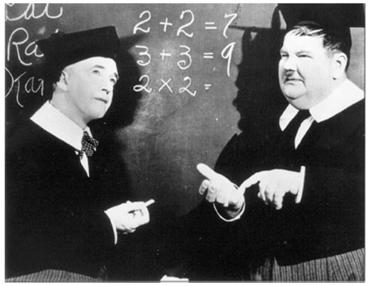


Esperimenti



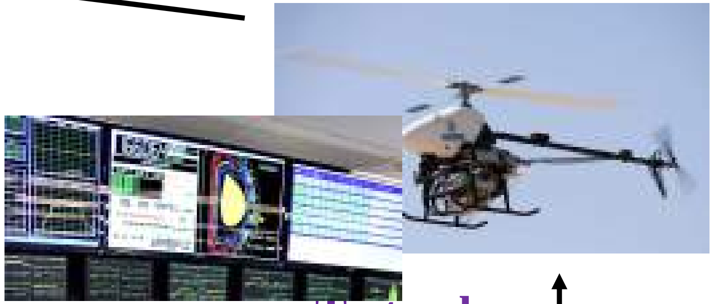
Equazioni

Parametri



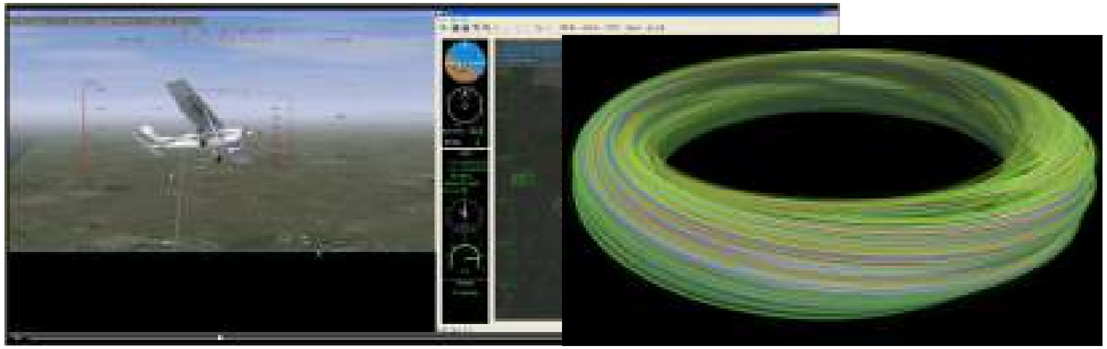
Modello Matematico

Soluzione del problema di controllo

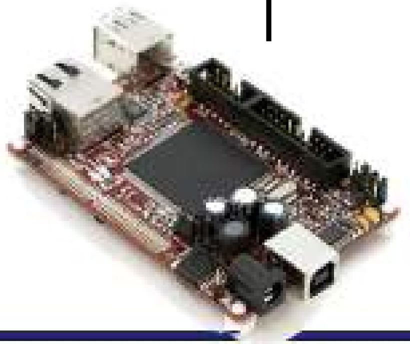


Test volo

Verifica in Simulazione



Prototipazione



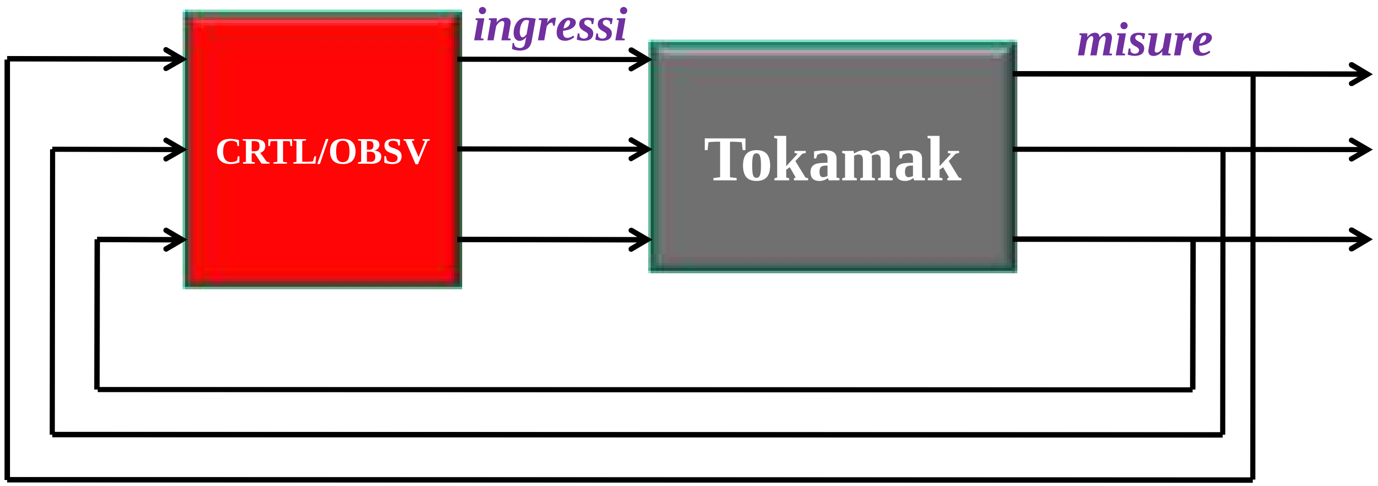
Possibili approcci moderni per il Controllo

Controllo ottimo

☺ Intrinsecamente multivariabile e ottimale (basato sull'ottimizzazione di un funzionale di costo). Porta in conto lo sforzo di controllo.

☹ Necessità di un modello affidabile e di un osservatore. Ridotta leggibilità del controllore. Possibile necessità di scheduling. Procedura trial and error.

$$J = \tfrac{1}{2}\mathbf{x}^{\mathrm{T}}(t_f)\mathbf{S}_f\mathbf{x}(t_f) + \tfrac{1}{2}\int_{t_0}^{t_f} \left(\mathbf{x}^{\mathrm{T}}(t)\mathbf{Q}(t)\mathbf{x}(t) + \mathbf{u}^{\mathrm{T}}(t)\mathbf{R}(t)\mathbf{u}(t)\right)\mathrm{d}t$$

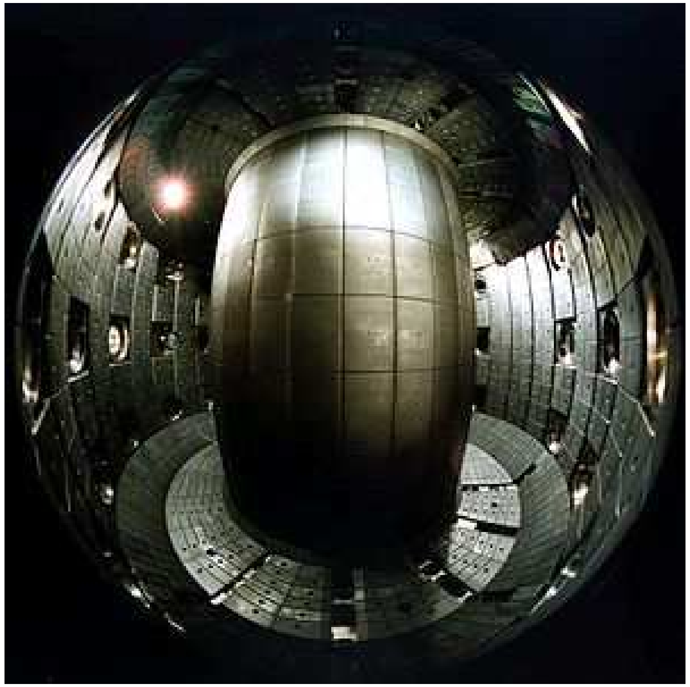


Possibili approcci moderni per il Controllo

Controllo H-infinito / mu-synthesis

☺Intrinsecamente multivariabile e ottimale (worst case analysis). Porta in conto incertezze strutturate in frequenza. Unificazione di incertezze e prestazioni. Specifiche frequenziali. Soluzioni numeriche affidabili per il caso lineare.

☹ Necessità di un modello affidabile (struttura controllore/osservatore), ridotta leggibilità del controllore. Conservatitivà a volte elevata.

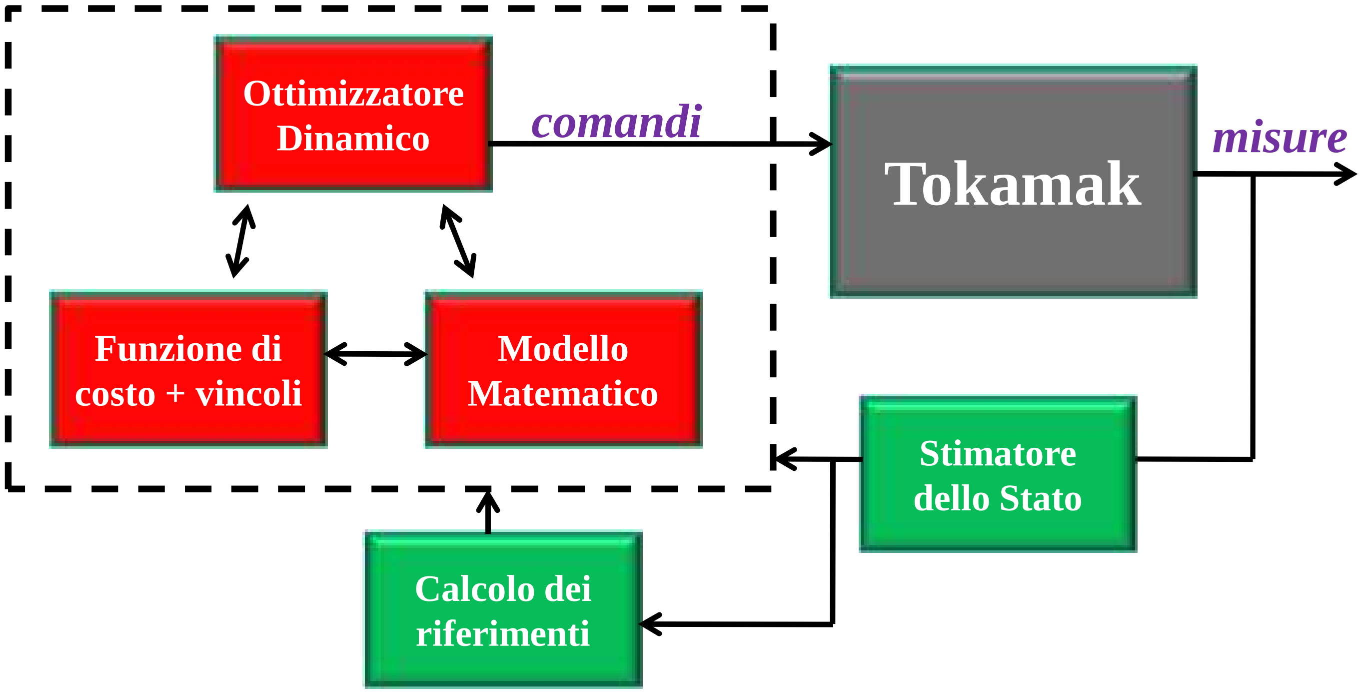


Approcci LPV per il gain scheduling

Possibili approcci moderni per il Controllo

Controllo predittivo

☺ Alte prestazioni e ampio inviluppo di funzionamento.
☺ Possibilità di portare in conto i vincoli
☹ Necessità di un modello affidabile.
☹ Onere computazionale.


Ottimizzatore Dinamico

comandi

Tokamak

misure

Funzione di costo + vincoli

Modello Matematico

Stimatore dello Stato

Calcolo dei riferimenti

Trattamento dei Vincoli nel Controllo

Main Problems

- Limits avoidance can be managed if enough degree of freedom are available. (e.g. one current reaches its limit → can be replaced by a combination of other currents having almost the same effect on shape
- In some cases the dependence of the constrained variables from the state/input is non-linear (e.g. in the case of vertical forces)
- The limits avoidance at one time instant may compromise performance later in a certain time window

ITER lavorerà con margini limitati

Principali vincoli per il controllo

Power Supplies

- Voltages

Coil System

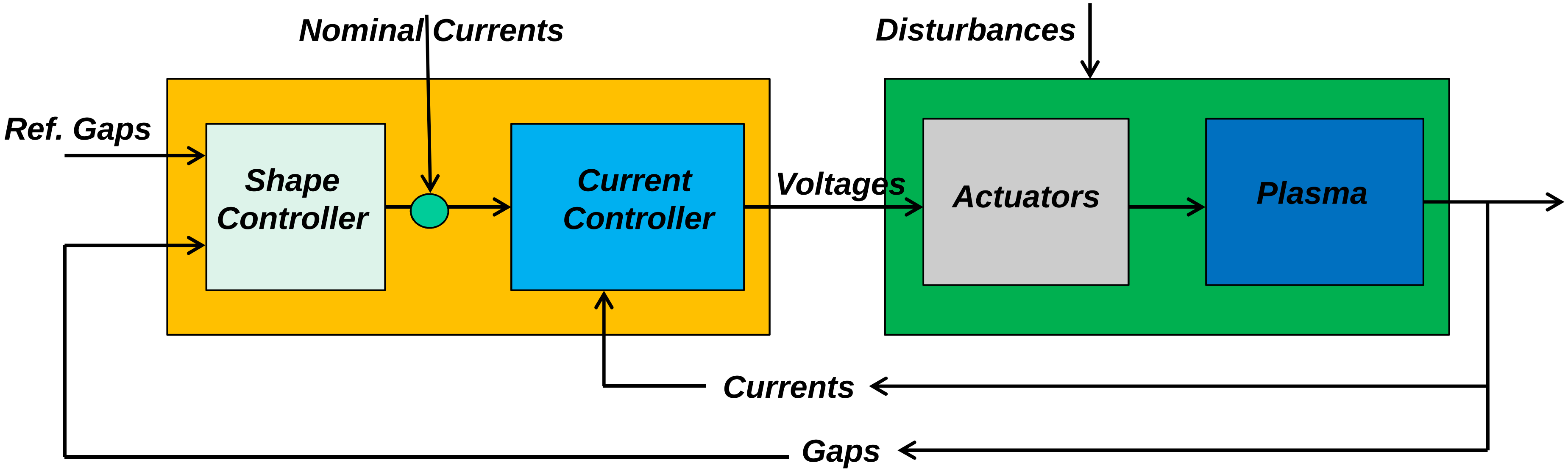
- Currents
- Fields
- Forces
- Temperature

Plasma and first wall

- Plasma-wall distance
- Strike points position
- First and second X-point

Possibili approcci per il limit avoidance

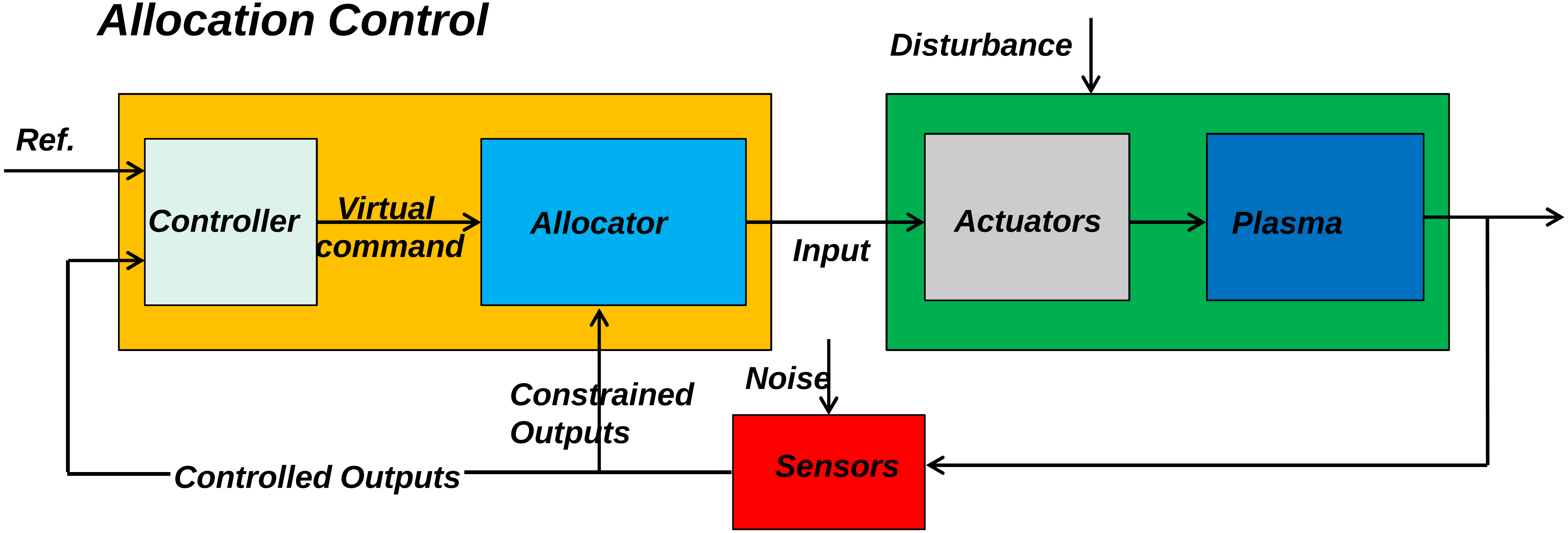
Feedback+Feedforward control strategy with current control



- ☺ With the off-line optimization of nominal currents the optimal solution is achieved in a large time window and margins can be left for the feedback control action;
- ☺ It could possible to schedule the feedback controller so as to avoid the use of those variables close to their limits

Possibili approcci per il limit avoidance

Allocation Control



- Some redundancy is needed to design a control allocator
- The allocator decides at each step how to command the actuators to achieve the virtual commands decided by the controller
- The allocation is based on an optimization which can take into account constraints
- CREATE team implemented on JET a first nonlinear dynamic allocator with the XSC

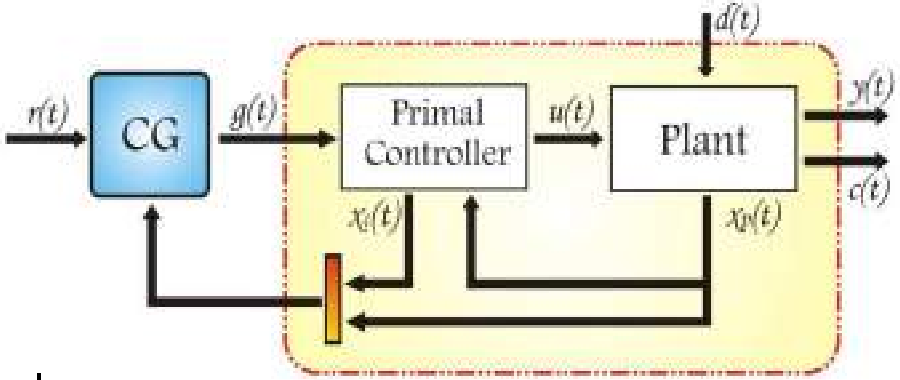
Model Based Predictive Control

☺ Optimization on a receding horizon time window

☺ In the linear model validity hypotheses, closed form solutions are available guaranteeing stability and performance also in the presence of input/state constraints

☹ Increased computational cost

☹ Possible conservatism



Two possibile ways to cope with constraints

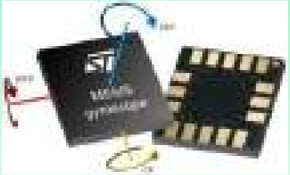
- Direct constrained Model Predictive control
- Reference Governor approach: CG mitigates the reference commands generating a sort of best approximation compatible with the prescribed constraints

Cenno ai problemi di stima

Determinazione di posizione, assetto e velocità a partire da misure di sensori eterogenei per
- **grandezza misurata;**
- **rumore di misura;**
- **frequenza di acquisizione;**
- **continuità del segnale e possibile fault di sensore**

Pitot

Giroscopi



Accelerometri

Magnetometro

GPS

⇩ ***Velocità Relativa***
(bassa frequenza, disturbo di processo rumore atmosferico)

⇩ ***Velocità Angolare***
(alta frequenza, Rumore di misura, bias)

⇩ ***Accel lineare***
(alta frequenza, Rumore di misura, bias)

⇩ ***Direzione***
(bassa frequenza, Rumore di misura, bias)

⇩ ***Posizione Assoluta***
(bassa frequenza, Rumore di misura, Segnale non continuo)

Stimatore dello Stato

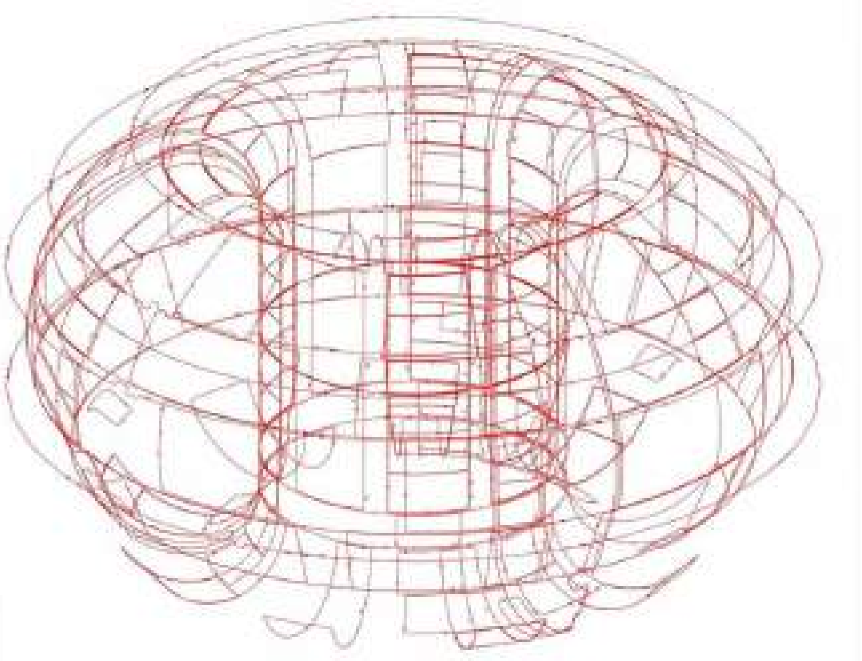
⇩

Stima posizione, assetto, velocità

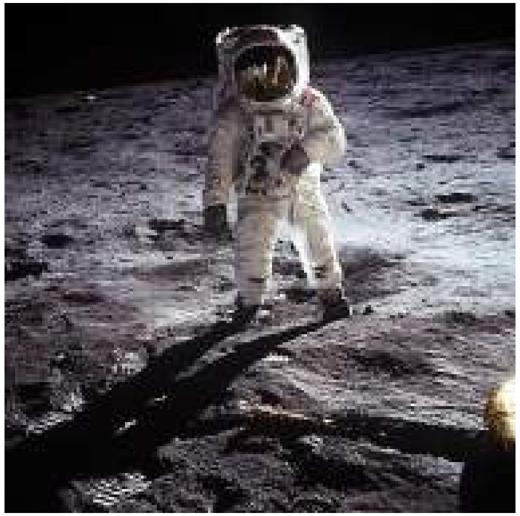
Cenno ai problemi di stima

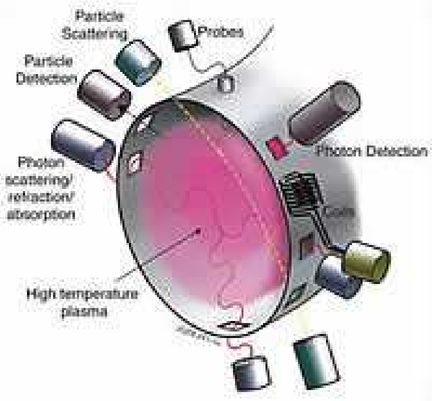
Ricostruzione della forma, posizione e corrente di plasma attraverso sensori magnetici

- Utilizzo di sensori di campo e di flusso
- Stime basate su modelli inversi basati su codici di equilibrio
- Stime basate sui principi della fisica
- Stime statiche e stime dinamiche (Kalman Filtering)



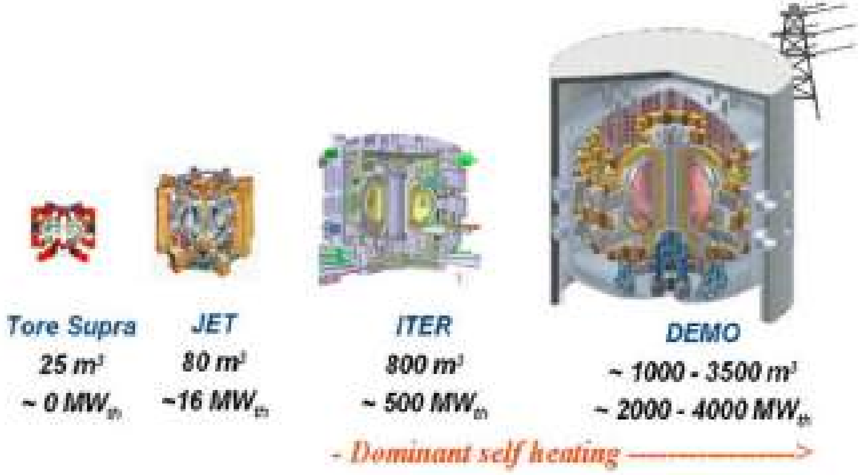
Vacuum vessel flux loops.
Distribuiti sulla superficie del VV con 234 sensori singoli.



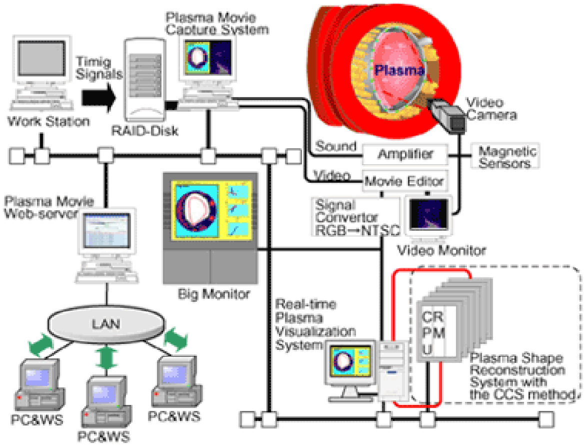


L'impegno personale attuale …

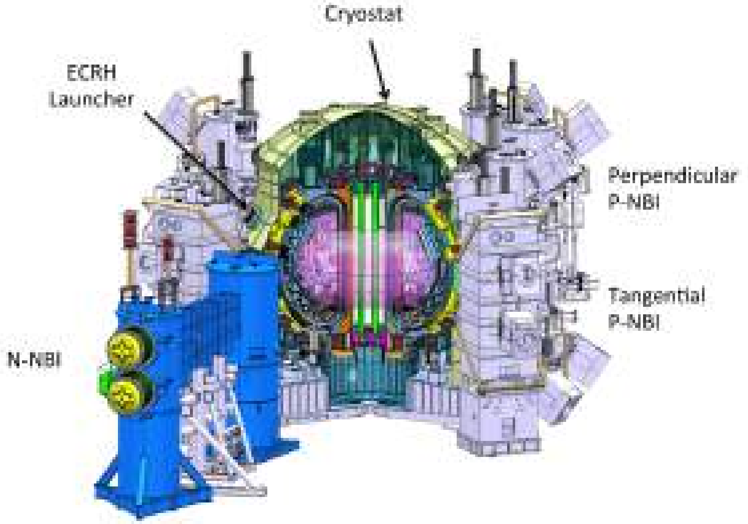
Breakdown experiments on TCV



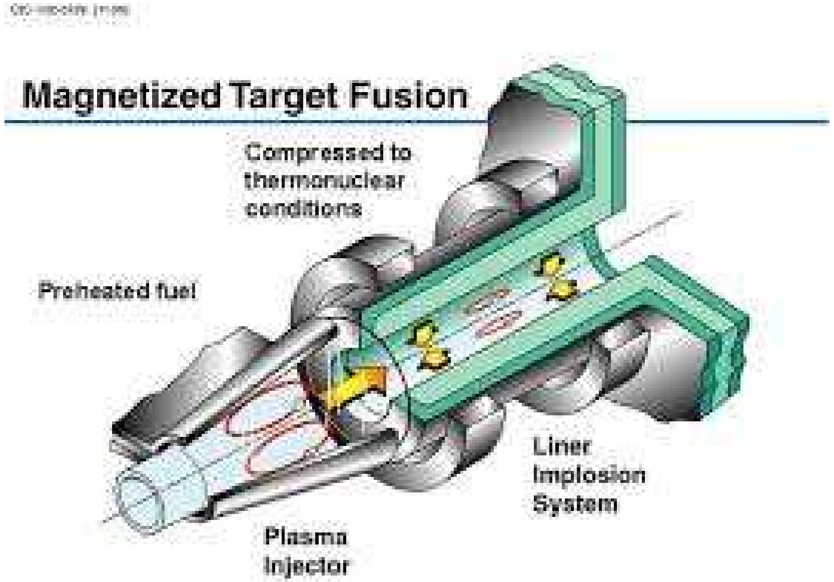
Plasma shape reconstruction on ITER

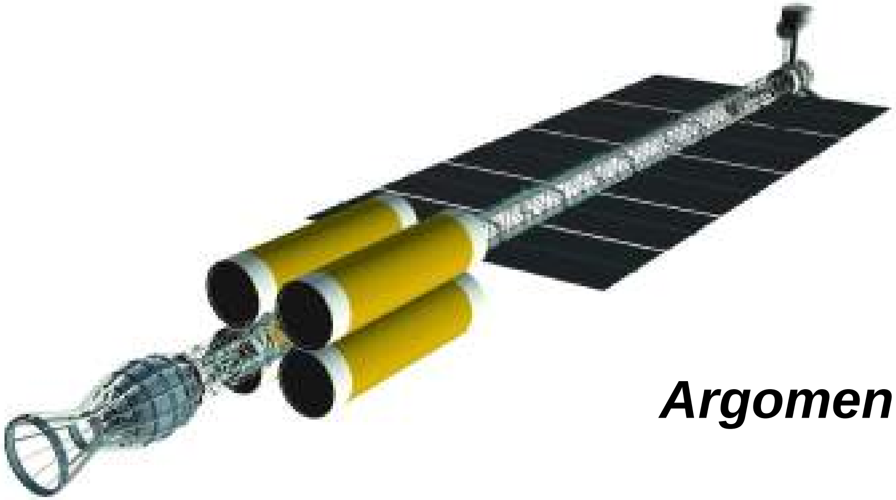


Modelling and Control on JT60SA, ITER, DEMO



e il possible impegno futuro … ☺





Argomenti di sintesi aerospazio-fusione

Grazie per l'attenzione !