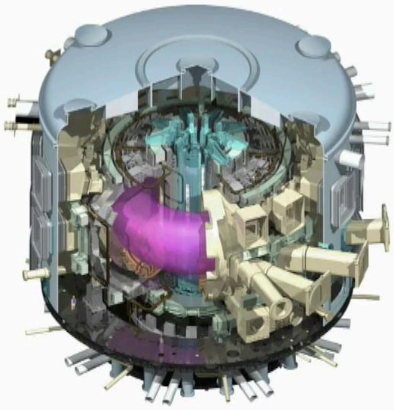
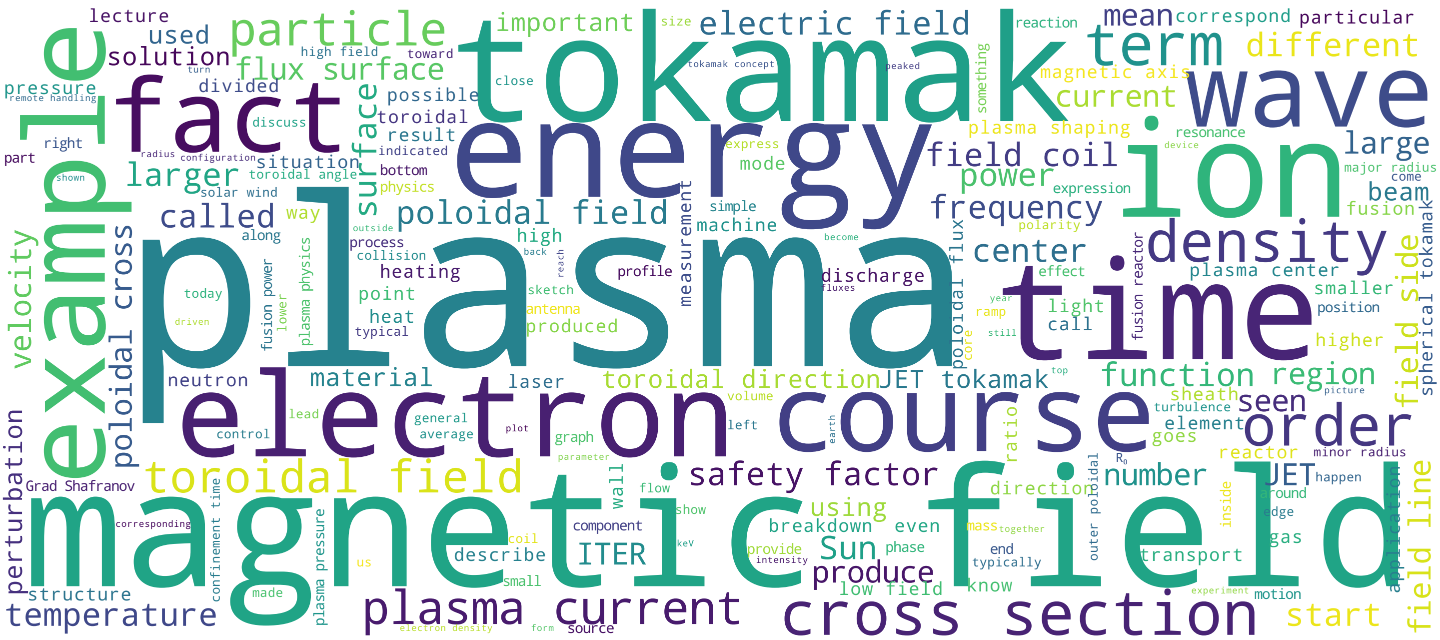


The tokamak concept

Plasma Physics and Application to Fusion Energy, Astrophysics and Industry

Lecture 7a

Duccio Testa



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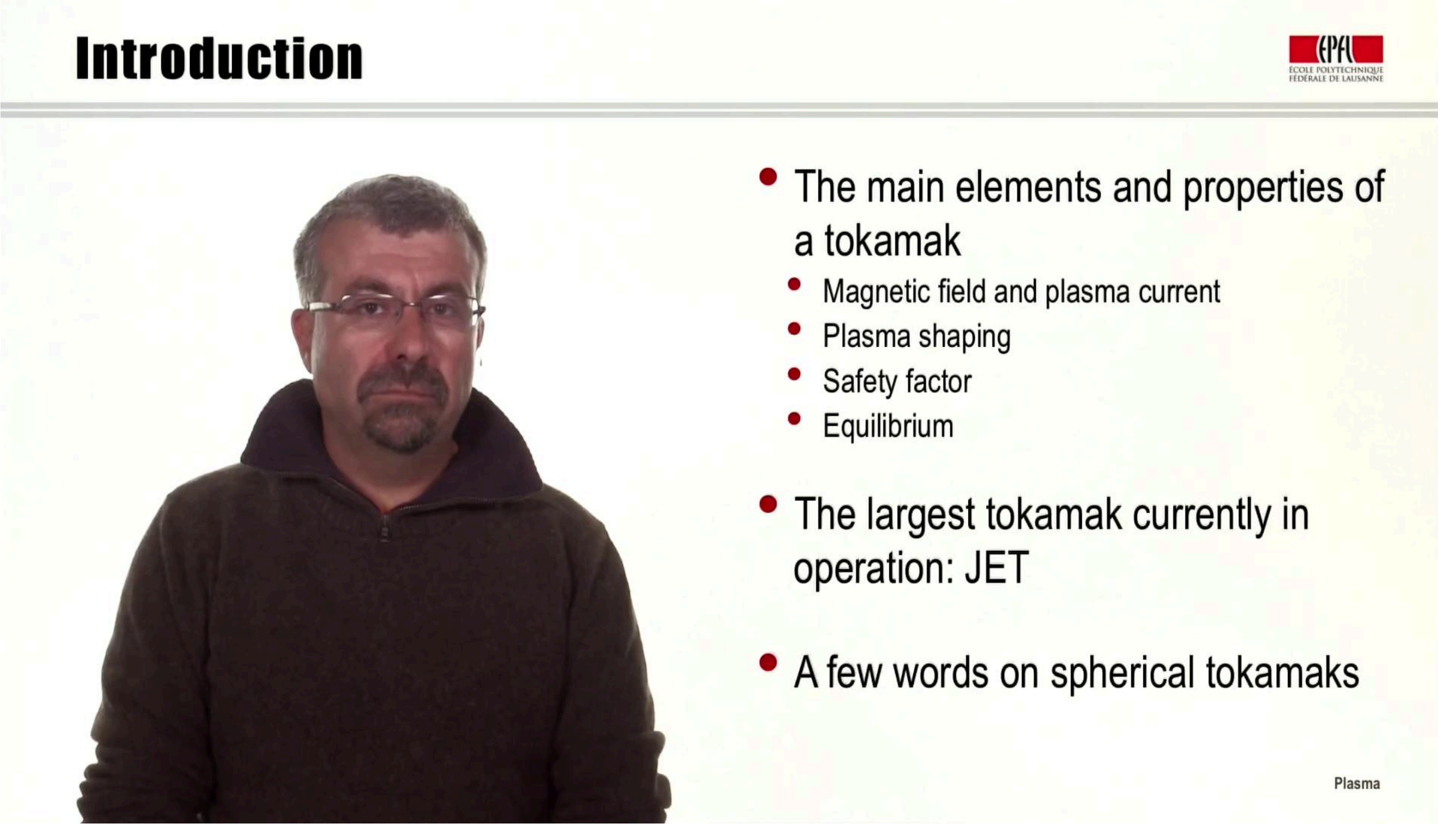
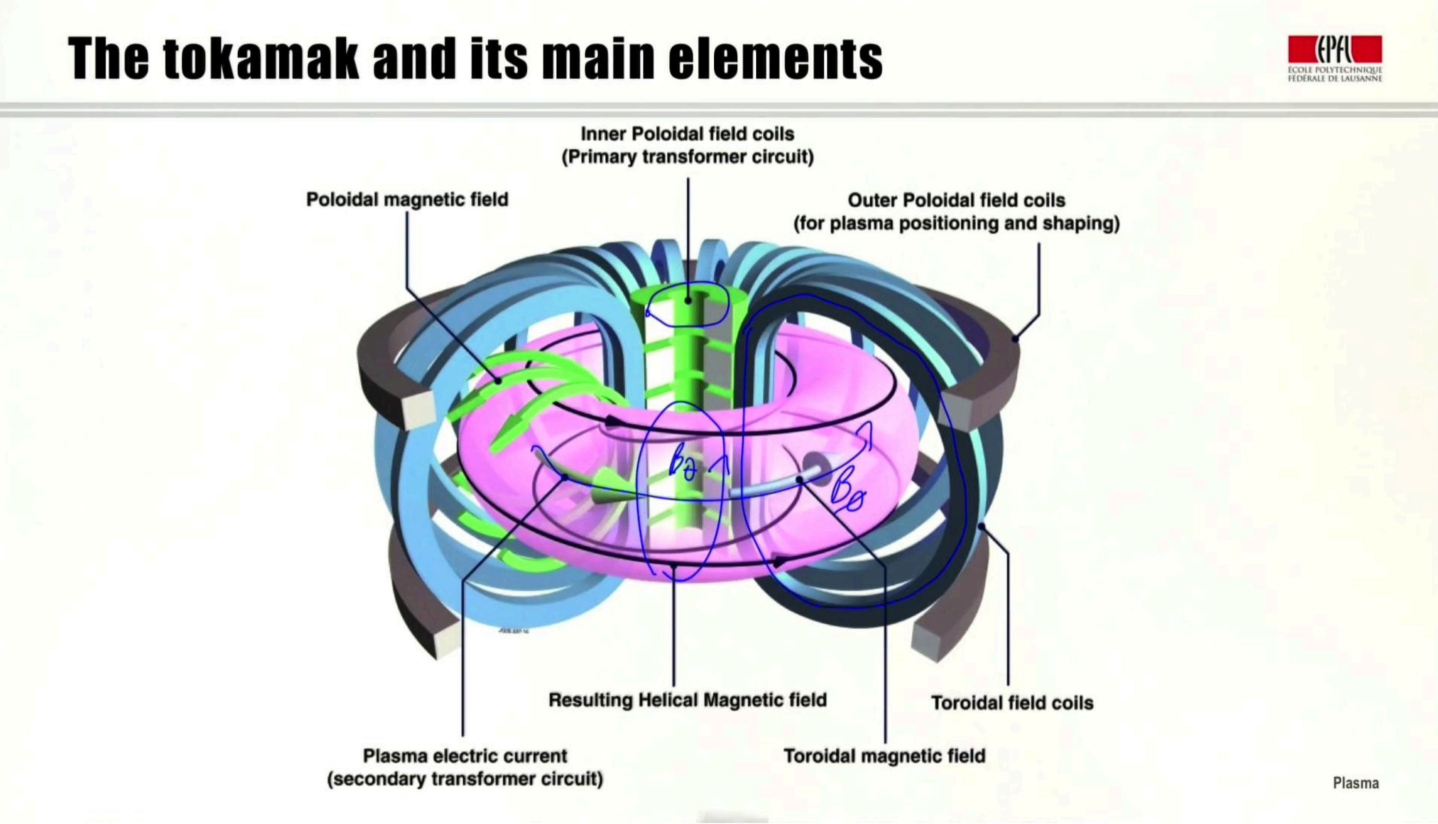


Welcome to the course on plasma physics and applications to fusion energy, astrophysics, and industry. My name is Duccio Testa and in this lecture, we will discuss the tokamak concept. We will start with the main elements and properties of a tokamak. We'll look first at the magnetic field and plasma current, then at plasma shaping, after the safety factor, and then at the equilibrium in a tokamak. We'll then describe briefly the largest tokamak currently operating in the world, which is the JET tokamak in England and then we'll conclude this lecture with a few words on the spherical tokamaks.

Notes

Summary

0m 05s



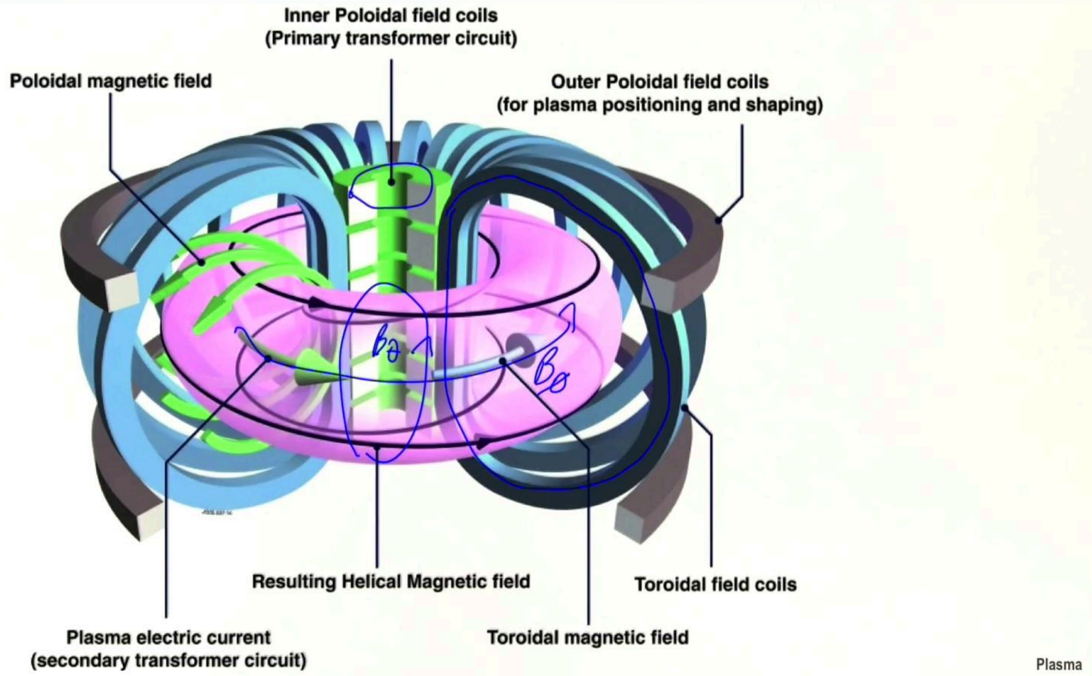
In this drawing here, we see a sketch of the tokamak and of it's main elements. The first element we need to talk about is the toroidal field coil. It's this structure here and as it's name indicates the toroidal field coils produce the toroidal field that goes in the toroidal direction. There is a current that flows in the coil so the field is along the toroidal angle. The second element that we need to discuss is the method to produce the plasma current. We do this by a transformer action. What we do, we drive a current in the inner poloidal field coils. These are these green structures here. This is the primary of the transformer circuit. The plasma itself, this pink element here, is in fact the secondary of the transformer itself. So by producing a flux ring to the inner poloidal field coils changing the current, ramping up and down the current, in the inner poloidal field coils, an electric field is driven in the toroidal direction, which produces the plasma current. The plasma current produces the poloidal magnetic field. The resulting magnetic field is helical, because we have a toroidal field in the toroidal direction, and the poloidal field. And we see here in black the helical path of the field line.

Notes

Summary

0m 44s

The tokamak and its main elements

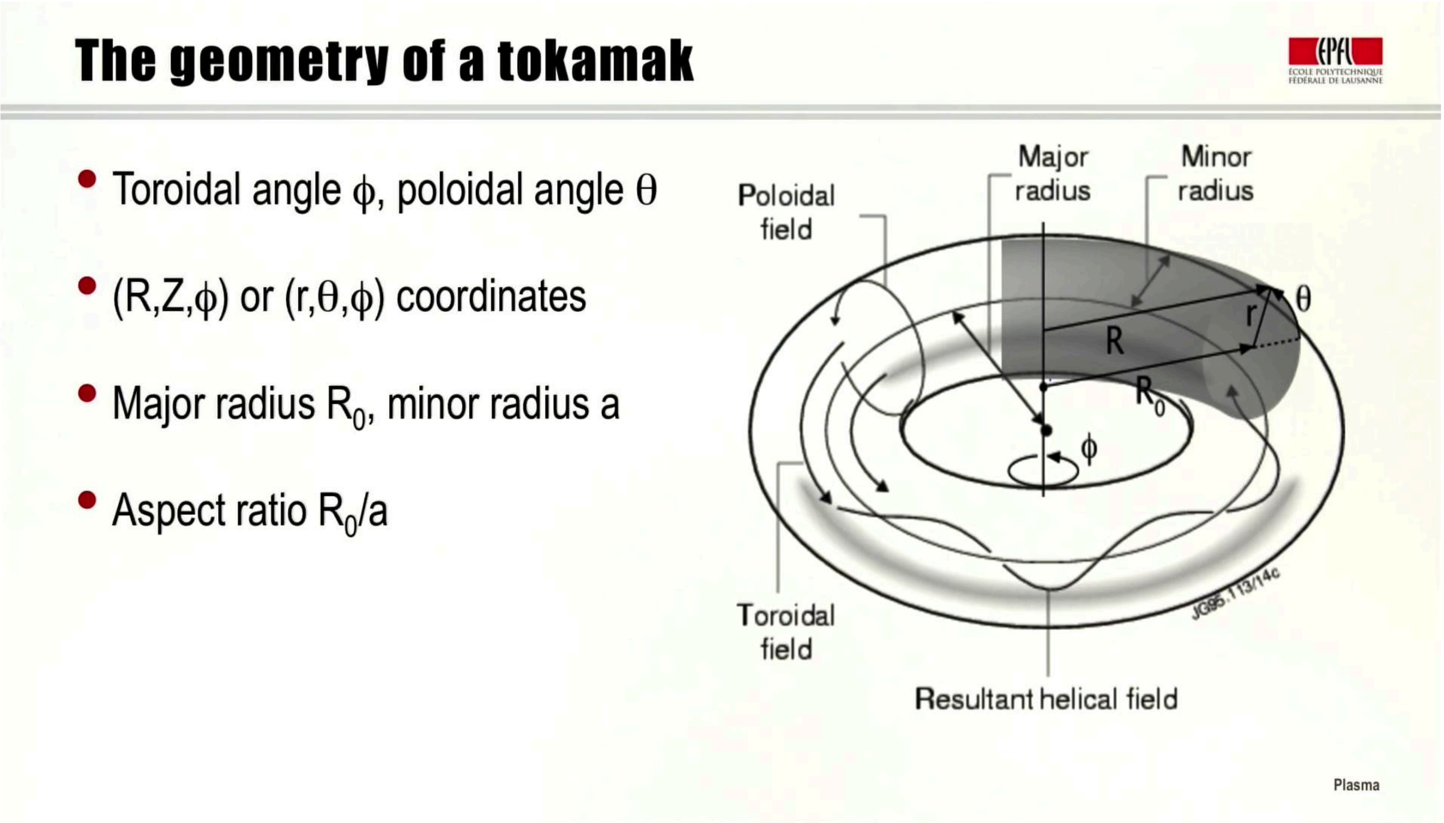


We have then the third element is this grey structure that are wound around the tokamak: these are the outer poloidal field coils and these are used to control the plasma position and the plasma shaping.

Notes

Summary

2m 15s

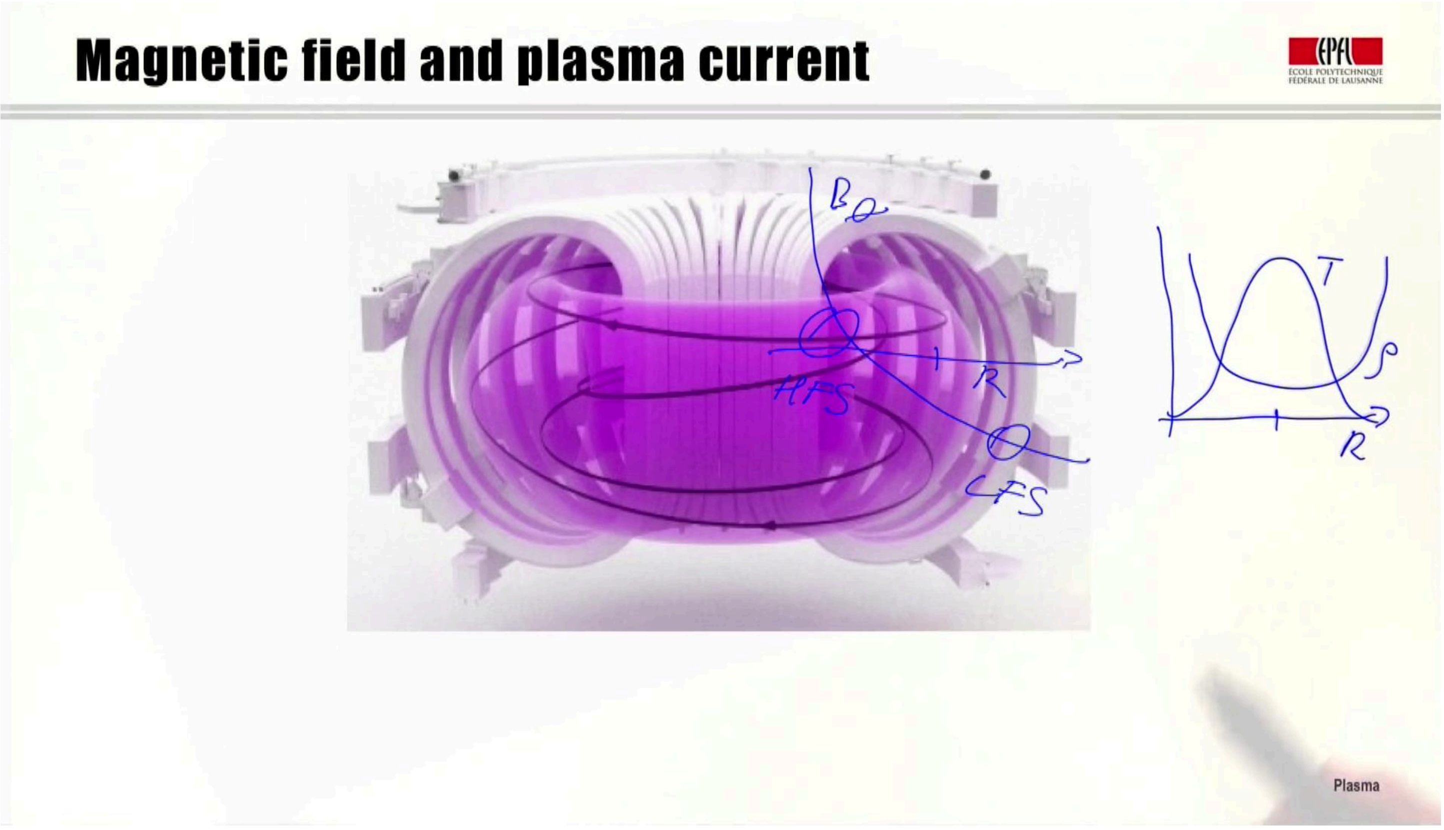


Now let's explain in more details the geometry of a tokamak and we will use this drawing here. A tokamak is in fact a cylinder with its ends folded up together. So we have two angles: the first is the toroidal angle Φ that goes in the plane on the cylinder and then we have the poloidal angle, θ that describes the plane perpendicular to the toroidal angle Φ. Then we can describe the tokamak geometry using effectively two systems of coordinates. The first one is a quasi-Cartesian coordinate system (R, Z, Φ), are the coordinates along the radial direction [R], Z the vertical coordinate, Φ the toroidal angle or we can use quasi-cylindrical coordinate (r,θ,Φ), r is the radius [measured form the center] of the poloidal cross-section, θ is the poloidal angle and Φ is again the toroidal angle. You have two main quantities, R_0 is the *major radius* of the tokamak, and then if we consider the last surface in the poloidal cross-section it has a radius a, this is the *minor radius* and the ratio between R_0 and a is called the *aspect ratio* of the tokamak.

Notes

Summary

2m 29s



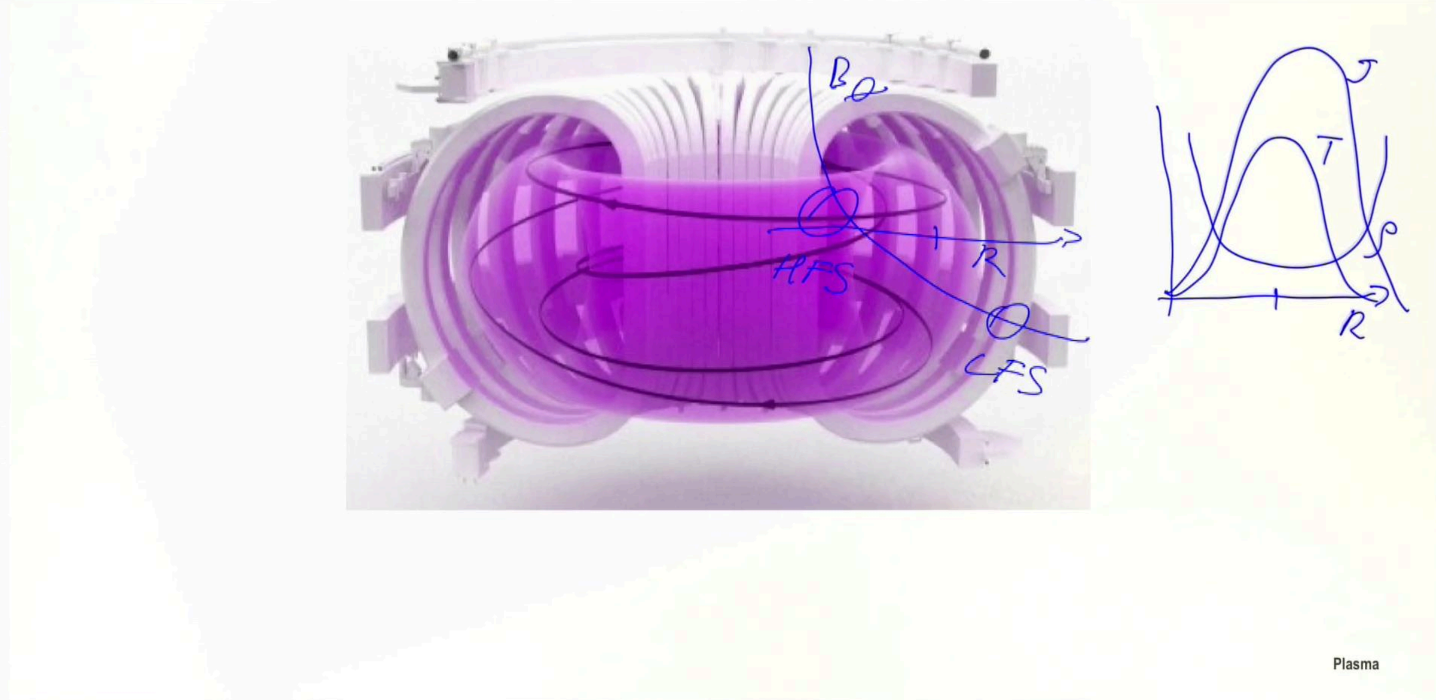
Now here again is a sketch of a tokamak. Let's look at the radial dependence of the toroidal field and of the plasma current. Remember, the toroidal field is produced by the toroidal field coil, this structure that is wound around the poloidal cross-section. Current flows in this structure and therefore the toroidal field has 1/R dependence. The toroidal field is higher at the center of the tokamak and this is lower towards the outer boundary of the tokamak. This allows us to define the high-field side and the low-field side of the plasma cross-section Obviously, the toroidal field is higher in this region of the plasma: this is the *high-field side*, and is lower in the outer region of the plasma: this is the *low-field side*. Now what about the plasma current? In a typical tokamak, the plasma temperature is peaked at the center of the poloidal cross-section. So if we describe-- if you look at the radial coordinate R, goes in this direction, this would be the center of the poloidal cross-section. The plasma temperature T is peaked in the center. We know that the resistivity has an inverse dependence on the plasma temperature, so the resistivity is smaller at the center of the poloidal cross-section.

Notes

Summary

4m 00s

Magnetic field and plasma current



And this tells us that the plasma current is larger where the temperature is larger because this is where the plasma resistivity is smaller. This obviously is the typical tokamak situation. There are different methods to produce the plasma current and therefore we may end up with a slightly different profile of J.

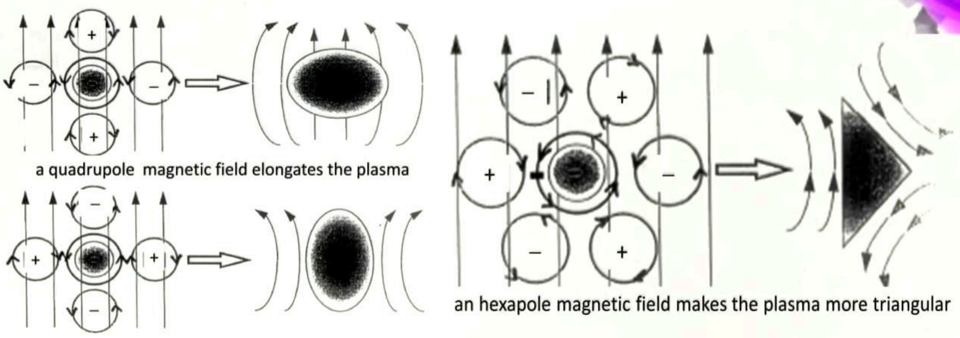
Notes

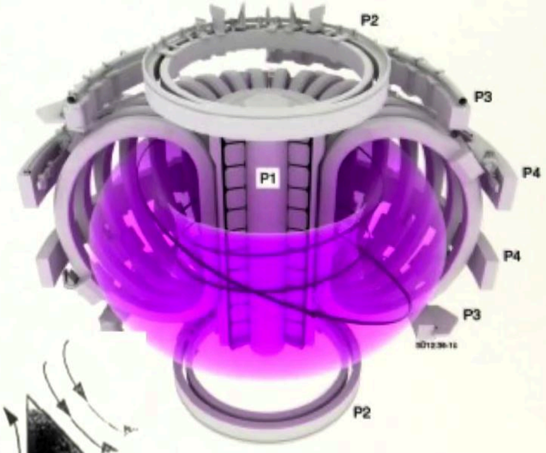
Summary

5m 34s

Plasma shaping

- Shaping from superposition of B-fields produced by coils and by current in plasma
- Poloidal field coils also control radial and vertical plasma position





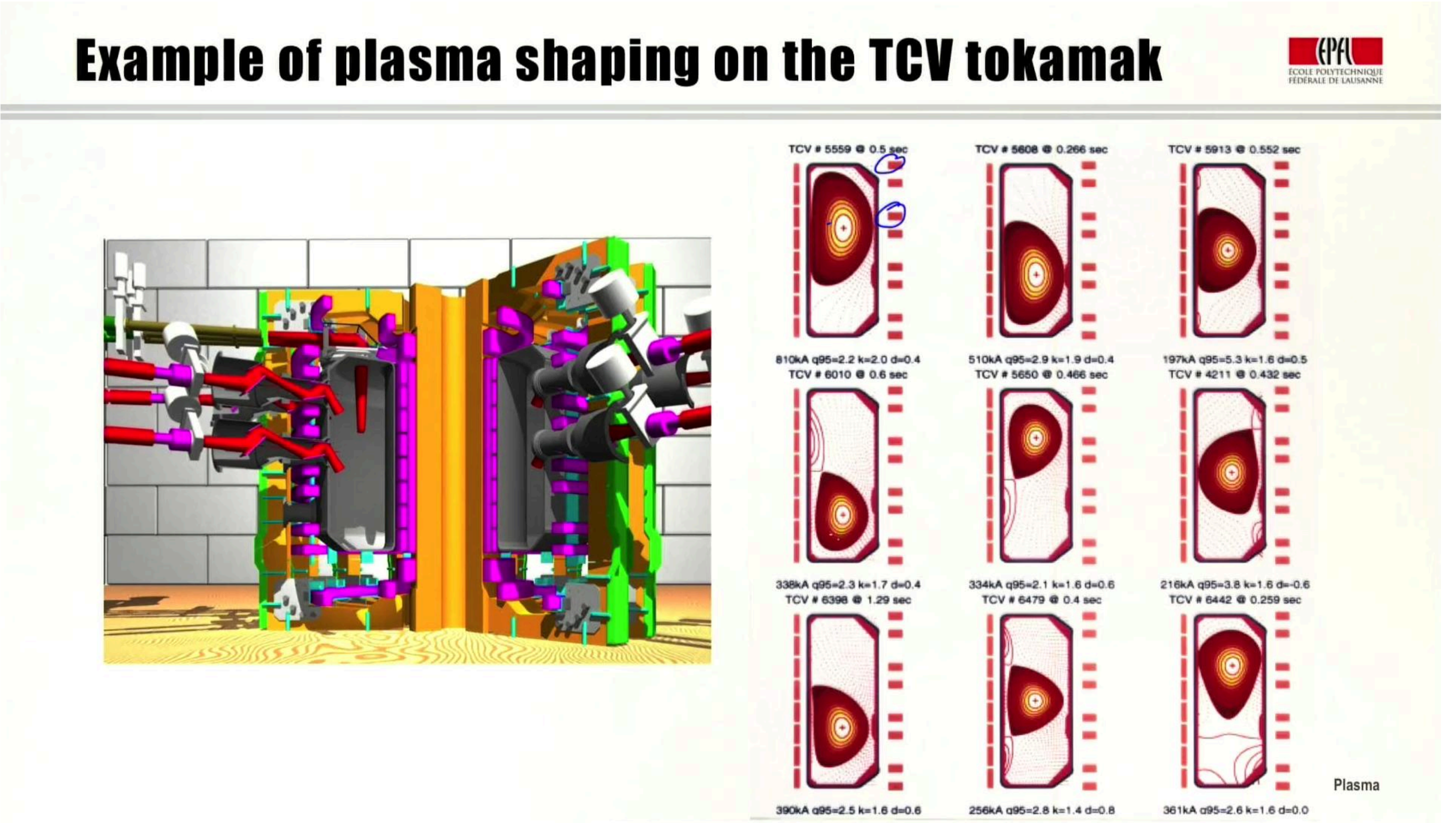
Plasma

Now let's look at plasma shaping. In particular, plasma shaping is produced by the outer poloidal field coils, so we start from a superposition of magnetic fields produced by coils and by the plasma current. These outer poloidal field coils that are labeled P2, P3, P4, in fact are used to control the radial position of the plasma, the vertical position of the plasma and produce plasma shaping. And here we see two examples of possible plasma shaping. The first one: if we produce a quadropole field using the outer poloidal, the coil, then the quadrupole field elongates the plasma and depending on the polarity plus on top and bottom, minus minus, left and right, then we have a plasma that is elongated in the horizontal direction. If we invert the polarity minus on top and bottom, plus left and right, then we elongate our plasma in the vertical direction. If we have a hexapole magnetic field then we have a plasma that tends to have a more triangular cross-section and this is an example taking this polarity of the fields produced by the outer poloidal field coils. We have a plasma that it triangular toward the right.

Notes

Summary

5m 56s



Plasma shaping is very important because it is one of the possible methods that we have to control instabilities and to optimize the performance of a fusion reactor. Here in Lausanne, we have a tokamak, the TCV tokamak. Its acronym stands for *Tokamak à Configuration Variable*. It's a tokamak that is optimized for the study of plasma shaping. Here we see a sketch of the TCV tokamak and here are some examples of the plasma shapes that we have obtained on TCV. TCV is optimized for the study of plasma shaping because we have a large number our outer poloidal field coils as indicated by these pink structures on the left and right of our plasma and we see that by changing the current and the polarity of these coils then we realize an incredible number of plasma shapes. We start from plasmas that are rather elongated and sort of a quasi-triangular at top of the machine, -this configuration here- to plasmas that are really triangular towards the bottom, again at top of the machine, plasmas that are basically like a droplet at top of the machine, or like a droplet at bottom of the machine. So here there is really a huge range of plasma shapes that can be obtained.

Notes

Summary

7m 17s

MHD equilibrium of a tokamak

- Reminder: force balance

  $$\mathrm{j} \times \mathbf{B} = \nabla p$$

  - Nested flux surfaces, coinciding with isobaric surfaces, on which current flows
  - Surfaces labeled by value of magnetic flux ψ
- Confinement efficiency

  $$\beta = nT/(B^2/2\mu_0)$$

- MHD equilibrium condition averaged over flux surfaces and expressed in terms of poloidal flux

  Grad-Shafranov equation

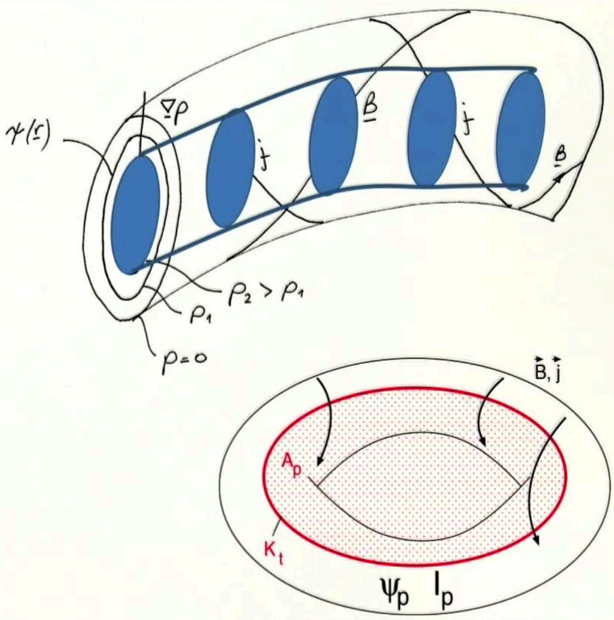


Image Credits: Mathias Groth, Aalto University

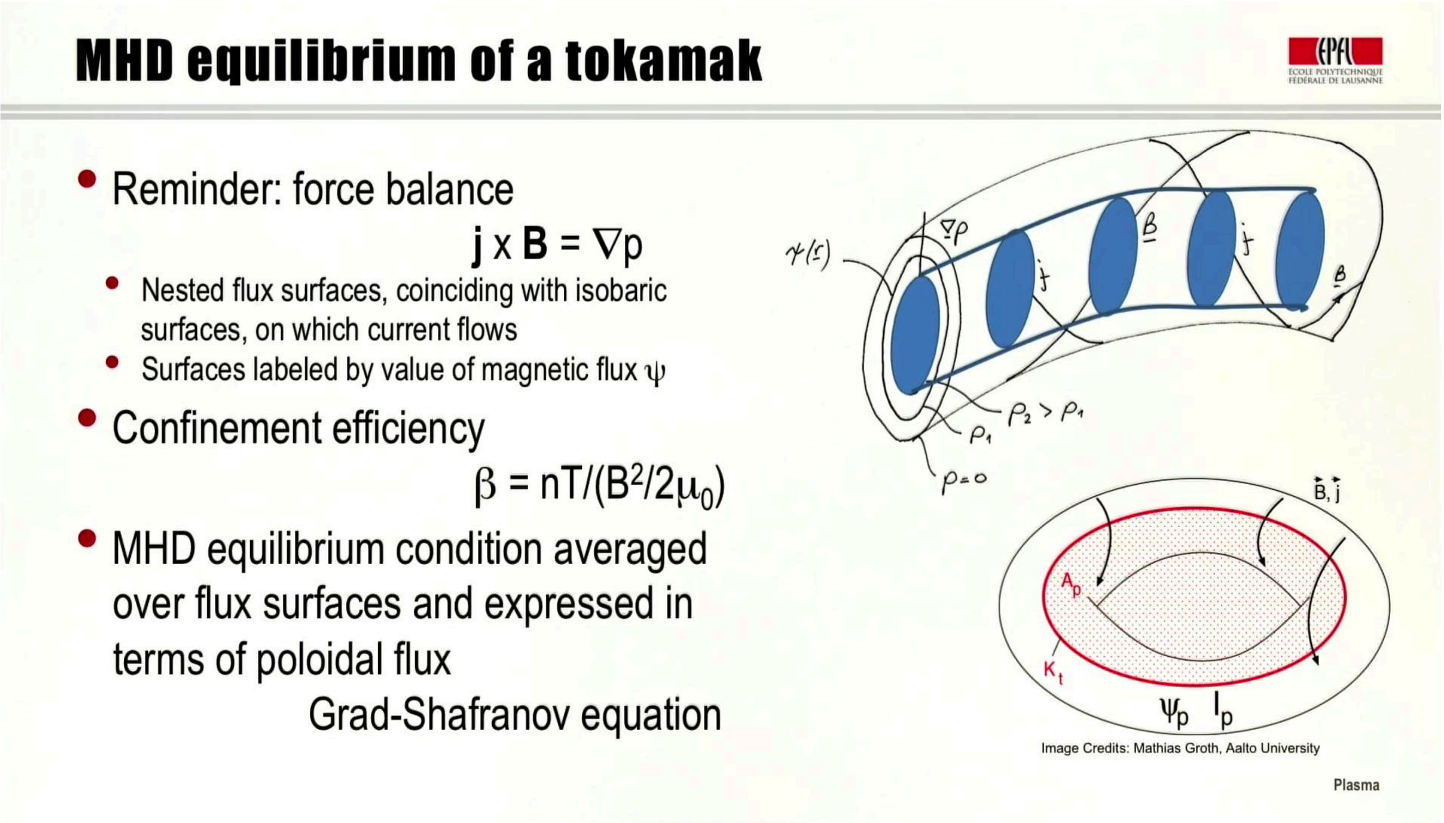
Plasma

Now let's look at the MHD equilibrium in a tokamak. Just as a reminder the MHD equilibrium is basically given by the force balance condition, j x B = ∇p. What this condition tells us is that we have nested flux surfaces that coincide with isobaric surfaces on which the current flows. Here we have a sketch of these flux surfaces that are indicated in blue along the toroidal direction and we can label these flux surfaces by the value of the magnetic flux ψ that is captured by these surfaces. We have an element that tells us the confinement efficiency: this is β. β is the ratio between the plasma pressure nT and the magnetic field pressure or $B^2 / 2\ \mu_0$ We can express the MDH equilibrium condition as an average over flux surfaces and express it in terms of the poloidal flux. This leads us to the *Grad-Shafranov* equation that we will see in the next slide. So let's look again at these two figures to understand the poloidal fluxes. In the top figure, the flux that you are looking at goes along the toroidal direction so is the flux of the magnetic field along the toroidal direction that is captured by a surface in the poloidal plane and this would be the toroidal flux.

Notes

Summary

8m 40s



The poloidal flux is actually the opposite. The poloidal flux ψp is the flux that is captured on a poloidal surface AP that is limited by a boundary Kt that it's in the toroidal direction. We express the Grad-Shafranov equation in terms of the poloidal flux ψp because for the solution of the Grad-Shafranov equation we use measurements of magnetic fields and fluxes at the plasma edge and what these measurements are are measurements of poloidal fluxes, so this is why we express the Grad-Shafranov equation in terms of the poloidal flux.

Notes

Summary

10m 09s

The Grad-Shafranov equation

EPFL ECOLE POLYTECHNIQUE FEDERALE DE LAUSANNE

j x B = ∇p → $\boxed{\Delta^*\psi(R,Z) = -\mu_0 R^2 p'(\psi(R,Z)) - \mu_0^2 F(\psi(R,Z)) F'(\psi(R,Z))}$

$$\Delta^* = R\left\{\frac{\partial}{\partial R}\left(\frac{1}{R}\frac{\partial}{\partial R}\right) + \frac{1}{R}\frac{\partial^2}{\partial Z^2}\right\}, F(\psi(R,Z)) = \frac{1}{\mu_0} R B_\phi(R,Z)$$

$$p'(\psi) = \frac{\partial p(\psi(R,Z))}{\partial \psi}, F'(\psi) = \frac{\partial F(\psi(R,Z))}{\partial \psi}$$

- Boundary conditions
  - Measurements of poloidal and radial magnetic field and fluxes at the plasma edge
- Initial guess for p(ψ(R,Z)) and F(ψ(R,Z)) and constraints on the solution
  - Measurements of the plasma pressure profile
  - Measurements of the profile pitch of the magnetic field lines
  - All measurements must be expressed in terms of ψ (i.e. fitted to flux-functions)

Plasma

Here we have the Grad-Shafranov equation. Basically again it starts by the usually equilibrium condition j x B = ∇p the equation is here on the top line. We see that it is a very complicated equation that has an operator Δ*. This is an equation in terms of the poloidal flux ψ(R,Z) [subscript omitted for simplicity] which contains two functions: the derivative p' of a pressure p(ψ(R,Z)) and a slightly more complicated function F(ψ(R,Z)) and its derivative F', F we can say is a functional of the toroidal field, BΦ and F is a function of the flux. This operator Δ*, we see contains second order derivatives with respect to the R and Z coordinates. So this turns out to be an elliptical equation in the flux of two functions p and p', F and F' that are both function of the magnetic flux, specifically of the poloidal flux. Since we have a derivative, we need boundary conditions and these boundary conditions are provided by measurements of the poloidal and radial magnetic field and the fluxes at the plasma edge. For the solution, we start with an initial guess for the pressure [p], and the F function that we provide as a function of the flux, and using constraints on this solution.

Notes

Summary

10m 53s

The Grad-Shafranov equation

j x B = ∇p

$$\boxed{\Delta^*\psi(R,Z) = -\mu_0 R^2 p'(\psi(R,Z)) - \mu_0^2 F(\psi(R,Z)) F'(\psi(R,Z))}$$

$$\Delta^* = R\left\{\frac{\partial}{\partial R}\left(\frac{1}{R}\frac{\partial}{\partial R}\right) + \frac{1}{R}\frac{\partial^2}{\partial Z^2}\right\},\ F(\psi(R,Z)) = \frac{1}{\mu_0} R B_\phi(R,Z)$$

$$p'(\psi) = \frac{\partial p(\psi(R,Z))}{\partial \psi},\ F'(\psi) = \frac{\partial F(\psi(R,Z))}{\partial \psi}$$

- Boundary conditions
  - Measurements of poloidal and radial magnetic field and fluxes at the plasma edge
- Initial guess for p(ψ(R,Z)) and F(ψ(R,Z)) and constraints on the solution
  - Measurements of the plasma pressure profile
  - Measurements of the profile pitch of the magnetic field lines
  - All measurements must be expressed in terms of ψ (i.e. fitted to flux-functions)

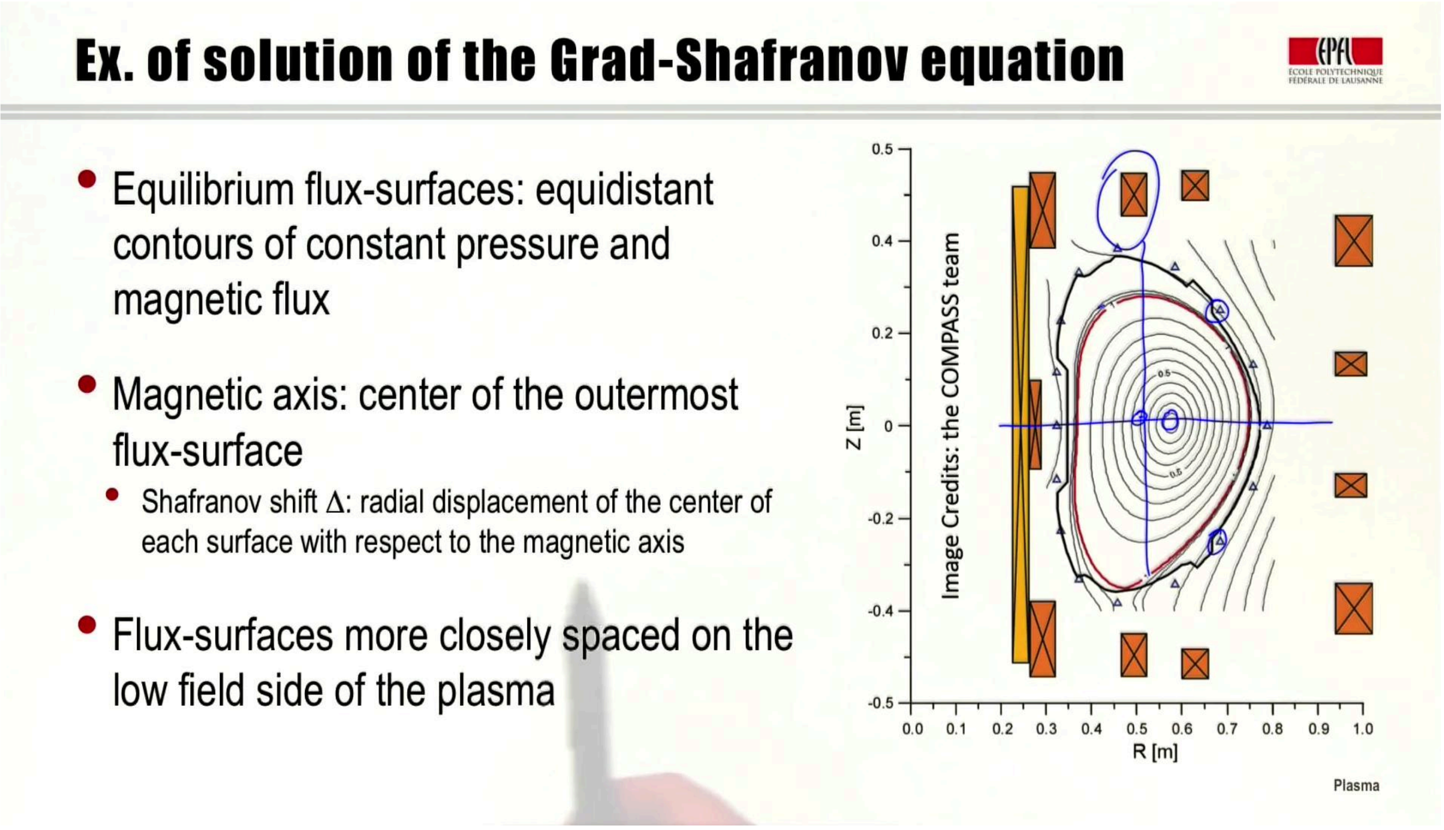
Plasma

The constraints are provided by measurements of the plasma pressure profile, by measurements of the profile of the pitch of the magnetic field lines, and those measurements must be expressed in terms of ψ, i.e. fitted to flux functions.

Notes

Summary

12m 27s

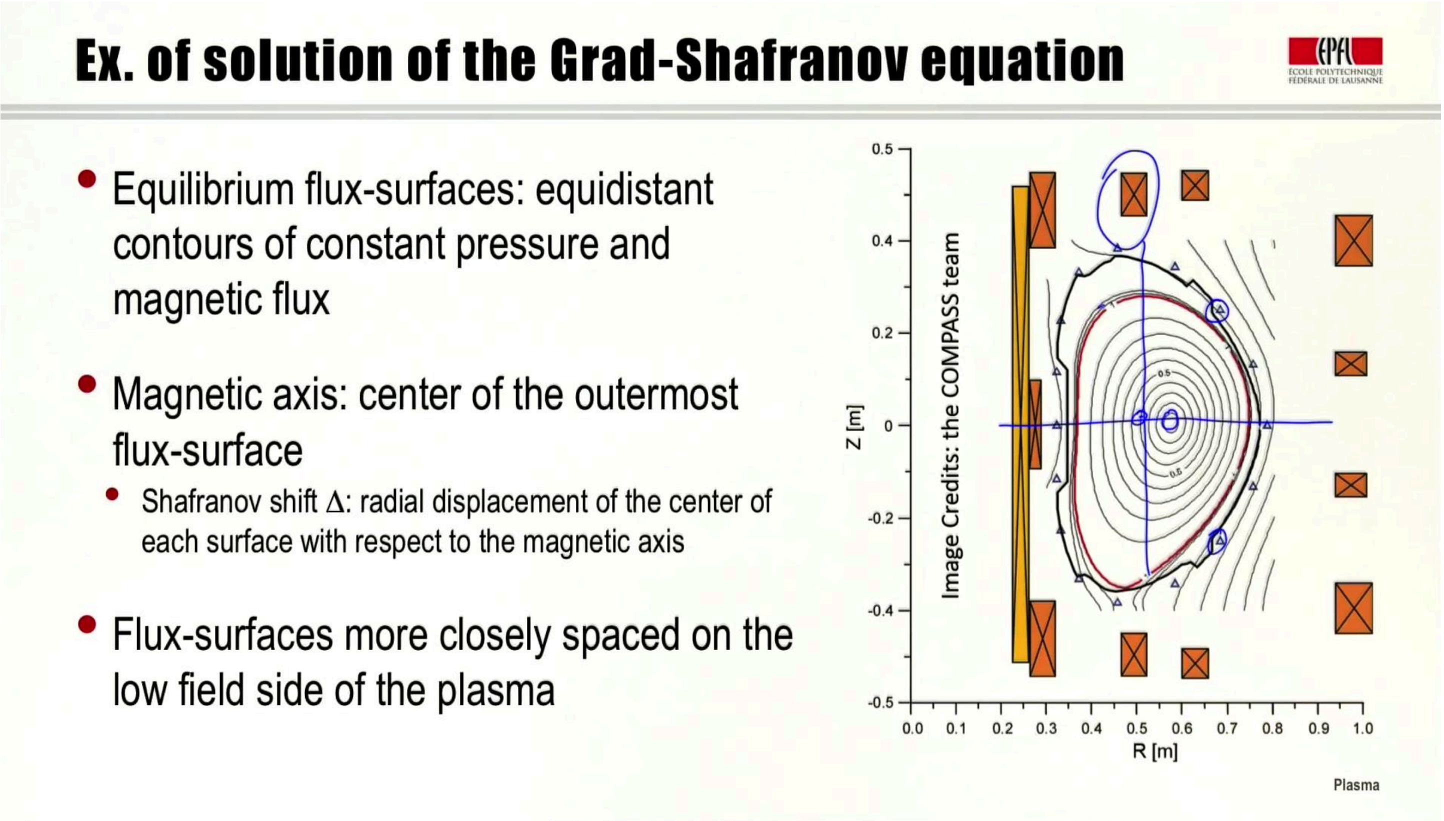


What we show here in this graph is an example of a solution of the Grad-Shafranov equation. Again, this orange structure here are the cross-sections of the coils that are used to produce the fields and shape the plasma and control it's position. These black triangles here,- we see we have many around the poloidal cross-section of the plasma- indicate the positions of the coils and flux loops that are used to measure the magnetic field fluxes that provide the constraints to the solution of the Grad-Shafranov equation. The solution, as we see here, is expressed in terms of equilibrium flux surfaces. These are equidistant contours of constant pressure and magnetic flux. Using this graph we can define a few quantities. First, we can look at the center of the outermost flux surface. This surface is described by this red line and we see that the center is here. The position of the center of the outermost flux surface is the magnetic axis. What we see is that if we look, for instance, at the innermost flux surface, the innermost circle here, the center of the innermost flux surface does not coincide with the center of the outermost flux surface. There is a shift and this is called the *Shafranov shift*.

Notes

Summary

12m 42s



It is the radial displacement of the center of each surface, with respect to the magnetic axis. What we also note is that flux surfaces are more closely spaced on the low field side of the plasma.

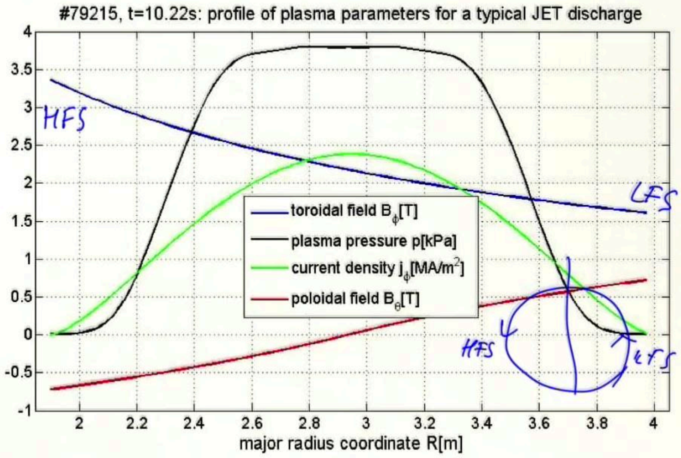
Notes

Summary

14m 08s

Typical equilibrium profiles in a tokamak

- 2D axi-symmetric equilibrium
- p(R) and toroidal current $j_\phi(R)$ typically peak on magnetic axis
- $B_\phi(R) \propto 1/R$, and $B_\theta(R)/B_\phi(R)$ varies from ~0.1 to ~0.25 from high field side to low field side ($B_\theta(R_0) \approx 0$)



Plasma

Now we can look at the typical equilibrium profiles in tokamak and we take a real case is a discharge from the JET tokamak and we look at the profile at one particular time point. First, in a tokamak, the equilibrium is 2D axisymmetric and for usual operating conditions the pressure, p, the toroidal current jΦ, typically peak on the magnetic axis. The toroidal field, the BΦ, we've seen already as 1/R dependence and the ratio between the poloidal and the toroidal field varies from 0.1 to 0.25 by going from the high field side to the low field side. The poloidal field is zero on the magnetic axis. So here we have a graph on these quantities. In blue is the toroidal field, larger on high field side of the plasma, smaller on the low field side of the plasma. The plasma pressure is indicated with the black line. We see it's peaked at the plasma center and this one has a really a wide profile. This is the toroidal current density J in green, again peaked at the plasma center and the poloidal field, Bθ, is indicated in red. It changes sign by going through the magnetic axis because the poloidal field, if you move around the poloidal cross-section, obviously goes up on the low-field side and then goes down on the high-field side, so it changes sign by going through the magnetic axis.

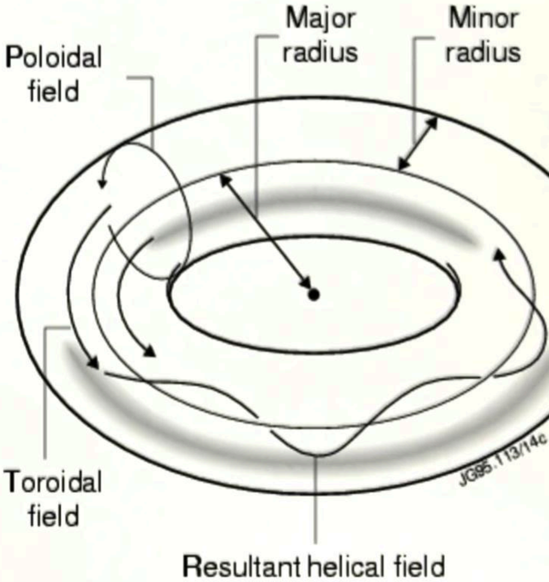
Notes

Summary

14m 21s

Tokamak safety factor – link with plasma stability

- Instabilities are generally associated with rational values of the safety factor
- Toroidal and poloidal mode numbers (m/n) = (1,1), (2/1), (3/2), …



Plasma

Now there's just one more element that allows us to fully describe the tokamak: it is the *safety factor*. We remember that the field lines follow a helical path around the toroidal direction and we see this line here describes the field line. So we can find the pitch angle of the magnetic field line: it is the ratio between the poloidal field and the toroidal field. Then by using the pitch angle we can define the safety factor and we can take an approximation for a large aspect ratio tokamak, it means that R_0, the major radius is much larger than a, the minor radius, and we can take this condition as averaged over flux surfaces. The safety factor, q, becomes only a function of r, the minor radius of the poloidal cross-section and we see that it depends on r/R_0, the major radius, and is the ratio between the toroidal and the poloidal field. The importance of the safety factor is that it is a measure of the plasma current contained within the radius r and in particular at the plasma edge when r = a, the minor radius, the safety factor is inversely proportional to the total plasma current, Ip. The safety factor is very important in tokamaks because there is a link with the plasma stability.

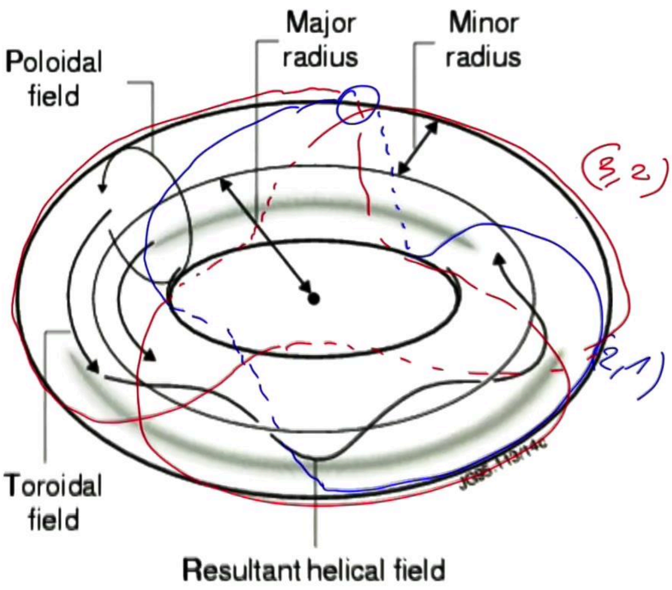
Notes

Summary

16m 02s

Tokamak safety factor – link with plasma stability

- Instabilities are generally associated with rational values of the safety factor
- Toroidal and poloidal mode numbers (m/n) = (1,1), (2/1), (3/2), …



Plasma

In fact, we find that in tokamak instabilities are generally associated with rational values of the safety factor. It's a resonance condition between the perturbation of the magnetic field and the pitch angle of the equilibrium magnetic field. We can usually decompose our perturbation into toroidal, n, and poloidal, m, mode numbers and the rational values of m/n are therefore described by elements, 1/1, 2/1, and 3/2. So if you look here at this tokamak, what would be a (2,1) m=2, n=1 perturbation? It's a perturbation that goes around twice in the poloidal direction, m while only going around once in the toroidal direction. So the m=2, n=1 component would be a component that, as I said, goes around twice in the poloidal direction, once in the toroidal direction. So let's look at the perturbation. Let's say we start at this point here. We want to go around twice in the poloidal cross-section, once in the toroidal cross-section, so we start with this. So this is our (2,1) perturbation. A (3,2) perturbation would be one that goes three times around the poloidal cross-section, while going twice along the toroidal angle. So we start again from here and there it is. So this would be the (3,2) perturbation, three turns in the poloidal direction, two turns in the toroidal direction.

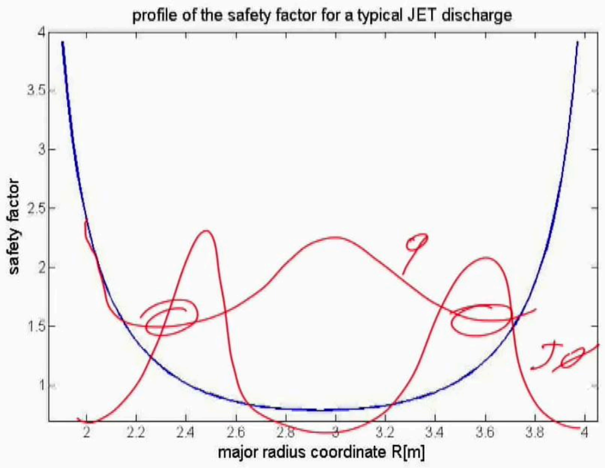
Notes

Summary

17m 25s

Typical safety factor profile in a tokamak

- Safety factor increases from plasma center, $q(R_0) \approx 1$, towards edge
- Safety factor can be non-monotonic depending on the current profile



Plasma

Now we can look at a typical safety factor profile in a tokamak and again, we'll take an example from the JET tokamak, the same discharge that we've seen before. The safety factor increases from the plasma center where q is typically around 1, towards the edge. This is typical situation for the safety factor of a monotonic safety factor profile that is linked to a current profile, current density profile Jφ that is peaked on the magnetic axis. We can have slightly different current profiles. There, for instance, can be produced by waves or by pressure gradient and then the safety factor can actually change. So if you have a current profile that is peaked slightly off axis, like this one here, then this will produce a safety factor profile that is above one in the plasma center and that has two minima. We have two minima here of the safety factor profile and the value at the plasma center is larger than one. And one advantage is that since the safety factor is above one at the plasma center, we remove all instabilities that are associated with the (1,1) mode from the plasma center.

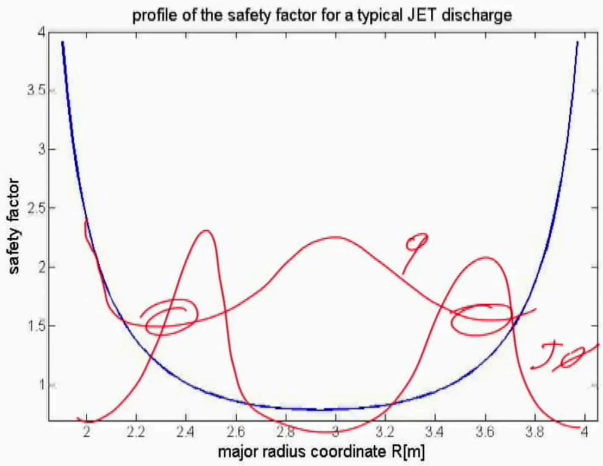
Notes

Summary

19m 09s

Typical safety factor profile in a tokamak

- Safety factor increases from plasma center, $q(R_0) \approx 1$, towards edge
- Safety factor can be non-monotonic depending on the current profile



Plasma

However, we have resonant values of the q-profile that can appear twice, here and here, this would be a (3/2) resonance so that is: the resonance condition is met twice and therefore if the perturbation grows to occupy all this region of the plasma between two resonant surfaces then it's a huge perturbation and this can have significant consequences on the performance of the discharge.

Notes

Summary

20m 35s

Typical sequence of events in a tokamak discharge

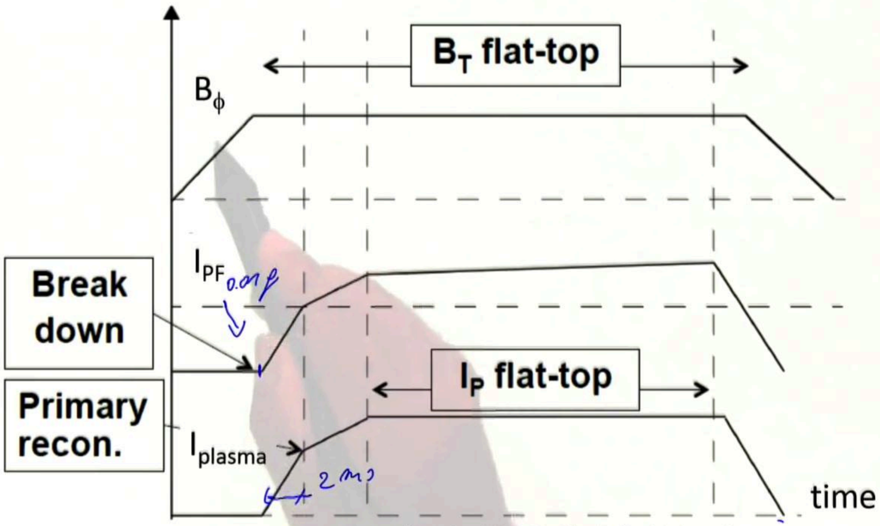


Image Credits: Mathias Groth, Aalto University

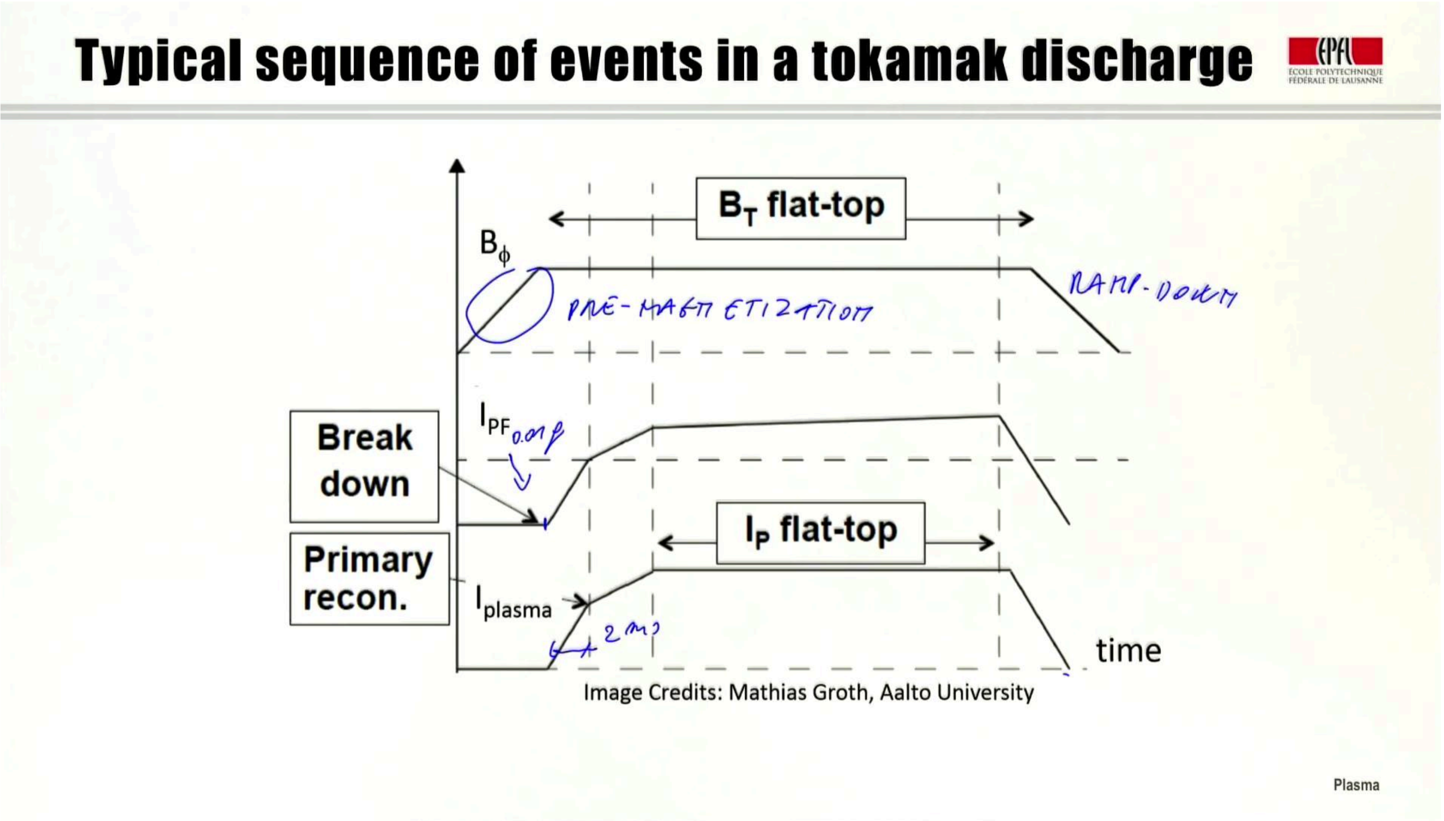
Plasma

Now we can look at the typical sequence of events in a tokamak discharge. We start with the current in the primary field coils, Ipf. It is first ramped down to a negative value, then at some point we start ramping up towards zero. At this moment we inject some gas in the plasma, not a lot: around 0.01 gram or there about. The current in the primary field coils ramps up, and we see that this is where the plasma current also starts to ramp up. This is the point where we start having a breakdown. So when the current in the primary field coils goes towards zero, this is what this is called the primary reconnection current in the inner poloidal field coils and at this moment we know if the breakdown has been sustained or not. This time scale is very short, typically of the order of a few milliseconds, 1 to 3 ms. If the breakdown has been sustained, we keep up ramping the current in the poloidal field coil, the plasma current ramps up, and then we get to a flat-top and at the end of the discharge we ramp it down. For the magnetic field, we usually start with a pre-magnetization. In fact, if you see the toroidal field starts to ramp up before we ramp up the current in the poloidal field coils.

Notes

Summary

21m 09s



This is the phase that is called the pre-magnetization of the plasma and this is useful to facilitate the breakdown. We then again have a flat-top for the toroidal field and then the ramp down phase at the end of the discharge.

Notes

Summary

22m 41s

A discharge in the JET tokamak

Plasma

We can now look at a video of a real JET discharge, a real plasma of JET and we will see flashes of light. This is where and when the plasma is formed. The patches of light indicate where there is radiation from the plasma to the wall. So we see the plasma is formed here at the bottom, (there are flashes of light), it fills then the poloidal cross-section and at this moment the plasma is terminated, this is the ramp down phase, again, at the bottom of the tokamak.

Notes

Summary

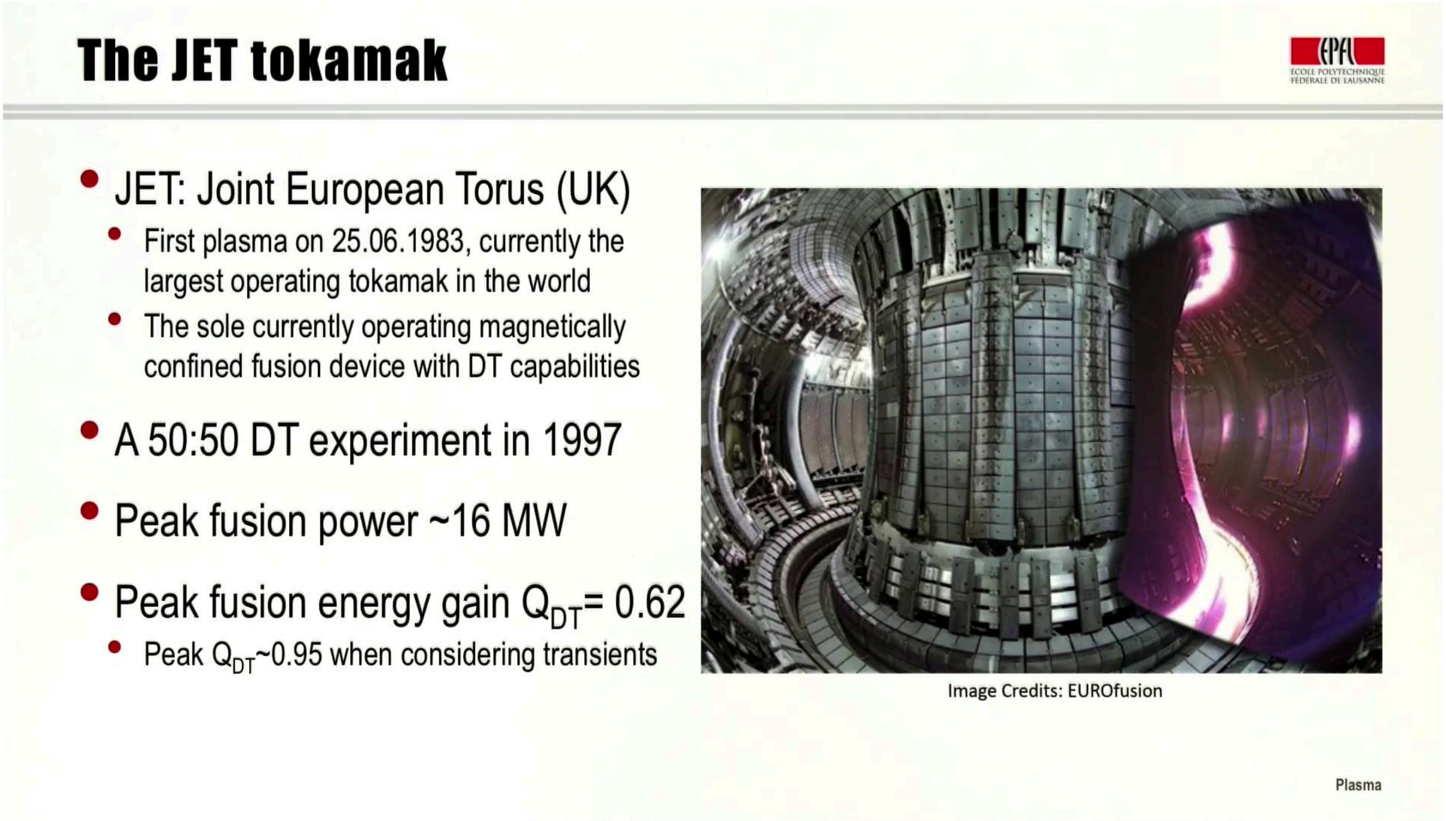
23m 01s

We can now talk very briefly about the history of the tokamaks. Tokamaks have been used in fusion research since the late 50s. In fact, the first tokamak that started operation was the T-1 tokamak in 1957 at the Kurchatov Institute in Moscow. Now, there are around 40 tokamaks that are currently operating worldwide. We have tokamaks in the USA, in Europe, in India, Iran, Brazil, South Korea, China, Japan, and Russia.

Notes

Summary

23m 48s



We will talk a little bit more about the JET tokamak. JET is an acronym that stands for Joint European Torus. It is in England, close to Oxford. The first JET plasma was in June 1983 and currently it is the largest operating tokamak in the world. We see here is a photo of the inside of the JET tokamak. We superimpose a typical plasma, we see these lights here where the plasma radiates to the wall, top, bottom, and a little bit on the low field side. The JET tokamak is currently the sole operating magnetically confined fusion device which has the capability of deuterium-tritium operation. One of the main highlights of the JET tokamak was a 50:50 DT experiment that was carried out in 1997. During this DT experiment we achieved a peak fusion power of around 16 megawatt, this is currently the world record and the peak fusion energy gain Q_DT of order 0.62 raised up to 0.95, so quite close to 1 when we consider transients.

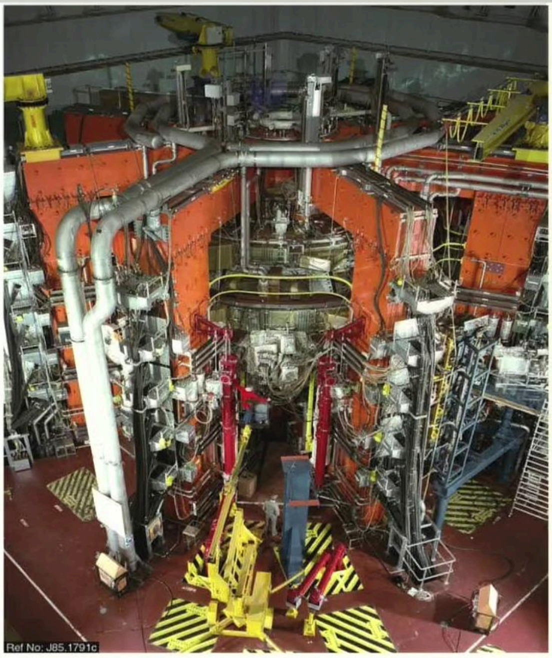
Notes

Summary

24m 17s

The JET tokamak: design & operational parameters

- Major radius R_0=2.98m; minor radius a=1.25m
- $B_{\phi 0}$=3.5T & I_p=3.5MA, up to $B_{\phi 0}$=4T & I_p=6MA
- Additional heating power up to 40MW
- Typically n_{e0}~2-5×10^{19}m^{-3}, T_{e0}~5-15keV~T_{i0}
- Typical energy confinement time τ_E~500msec
- Max($n\tau_E T$)~2.6×10^{20}keVm^{-3}s
- Flat-top pulse length 20-60s
- Wall materials: beryllium and tungsten



Plasma

We have here a photo of the JET tokamak. On the outside we see lots of elements and let me just tell you about a few design and operational parameters. The major radius of the JET tokamak is 3m, the minor radius is larger than one meter, 1.25 meter. The typical magnetic field and plasma current 3.5 tesla and 3.5 MA but operation is possible up to a value of the magnetic field on the magnetic axis $B\Phi_0$ of 4 Tesla, a plasma current up to 6 MA. We have additional heating power up to 40 MW. Typical value of the electron density and temperature in the plasma center are ne_0, electron density at the plasma center between 2 and 5 x 10^{19} particles per meter cubed, peak electron temperature and ion temperature at the plasma center between 5 and 15 keV. The typical energy confinement time τE is on the order of half a second, and this gives a maximum triple product n τE T, which is of the order of 2.6 x 10^{20} keV per meter cubed per second. We have a flat-top pulse length that is between 20 and 60 seconds and the materials of which the wall of JET is built are beryllium and tungsten.

Notes

Summary

25m 30s

An example of fusion technology on JET

- Remote handling enables work inside JET vessel without human presence
- Remote handling carried out by a dexterous, force-reflecting master-slave servo-manipulator
  - The slave unit is transported inside the torus on the end of a 10m long articulated robot
  - The master unit is driven by experienced operators situated in the control room

Plasma

Now JET is an experiment about the physics of fusion but also about technologies, and one particular technology we'd like to focus on is the remote handling technology.

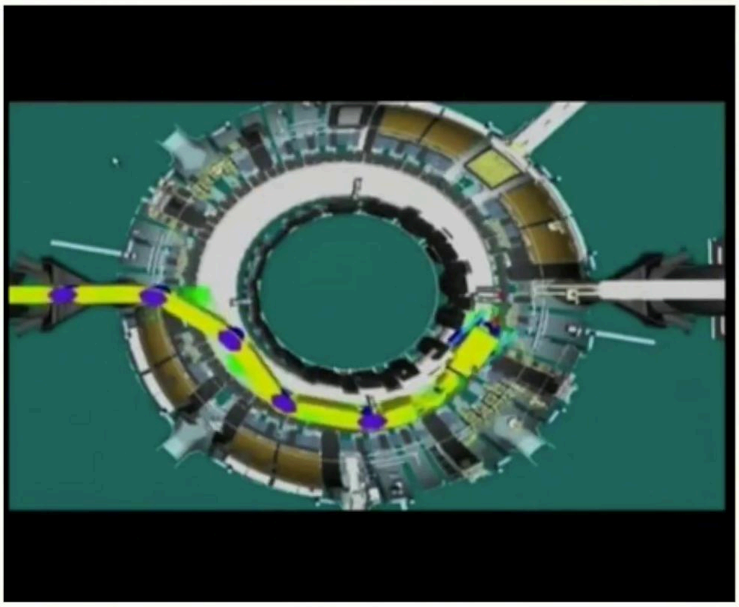
Notes

Summary

26m 52s

An example of fusion technology on JET

- Remote handling enables work inside JET vessel without human presence
- Remote handling carried out by a dexterous, force-reflecting master-slave servo-manipulator
  - The slave unit is transported inside the torus on the end of a 10m long articulated robot
  - The master unit is driven by experienced operators situated in the control room



Plasma

Remote handling enables work inside the JET vessel, without human presence. Remote handling is carried out by a dexterous force-reflecting master-slave servo manipulator.

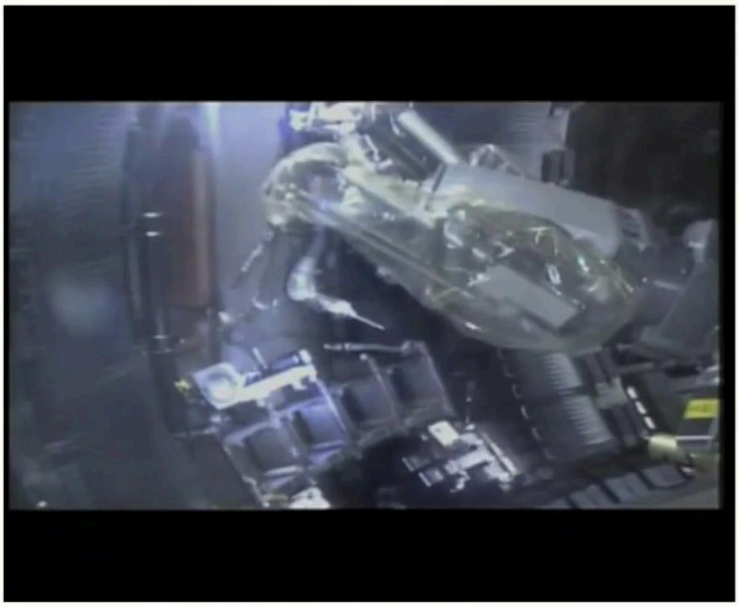
Notes

Summary

27m 01s

An example of fusion technology on JET

- Remote handling enables work inside JET vessel without human presence
- Remote handling carried out by a dexterous, force-reflecting master-slave servo-manipulator
  - The slave unit is transported inside the torus on the end of a 10m long articulated robot
  - The master unit is driven by experienced operators situated in the control room



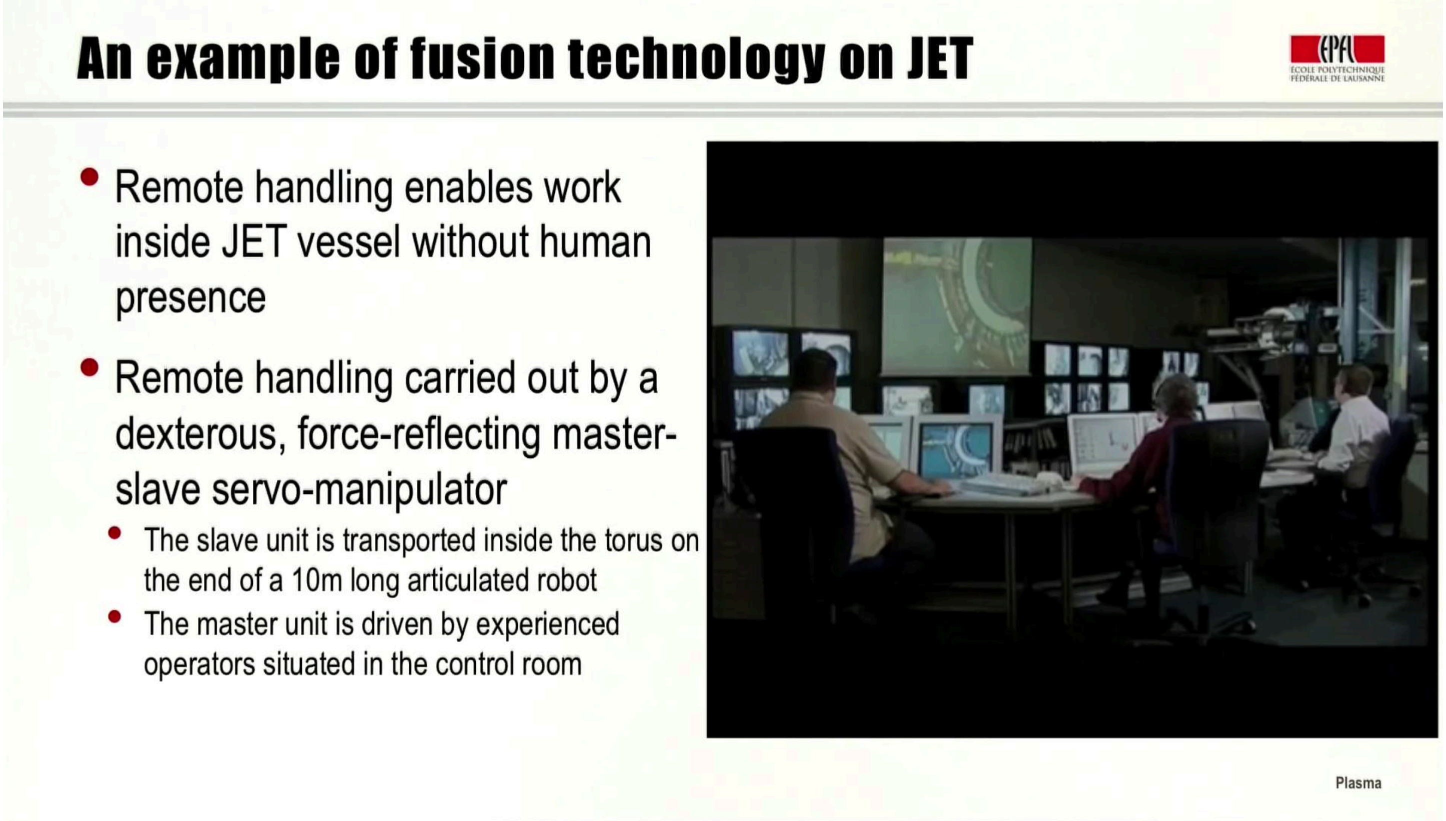
Plasma

The slave unit is transported inside the torus on the end of a ten meter long articulated robot and the master unit is driven by experienced operators situated in the control room.

Notes

Summary

27m 21s



We have here a video that shows a few examples of remote handling operation in JET and in particular there is a focus on the installation of antennae, the TA antennae that are used to drive and detect *Alfvén eigenmodes* on JET. We are particularly keen on this video because the TA antenna is something that we designed and built here at CRPP in Lausanne.

Notes

Summary

27m 30s

The spherical tokamak

- Conventional tokamak concept: large aspect ratio $R_0/a \approx 3.5$
- Spherical tokamak concept: aspect ratio $R_0/a \approx 1$, much smaller than a conventional tokamak
  - Lower plasma pressure, but higher value of $\beta = p/(B^2/2\mu_0)$: more compact magnetic field structure and smaller B needed for confinement
  - Central solenoid exposed to the plasma

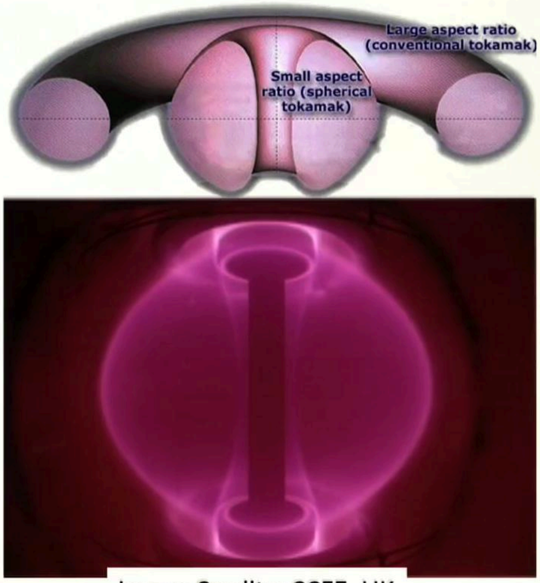


Image Credits: CCFE, UK

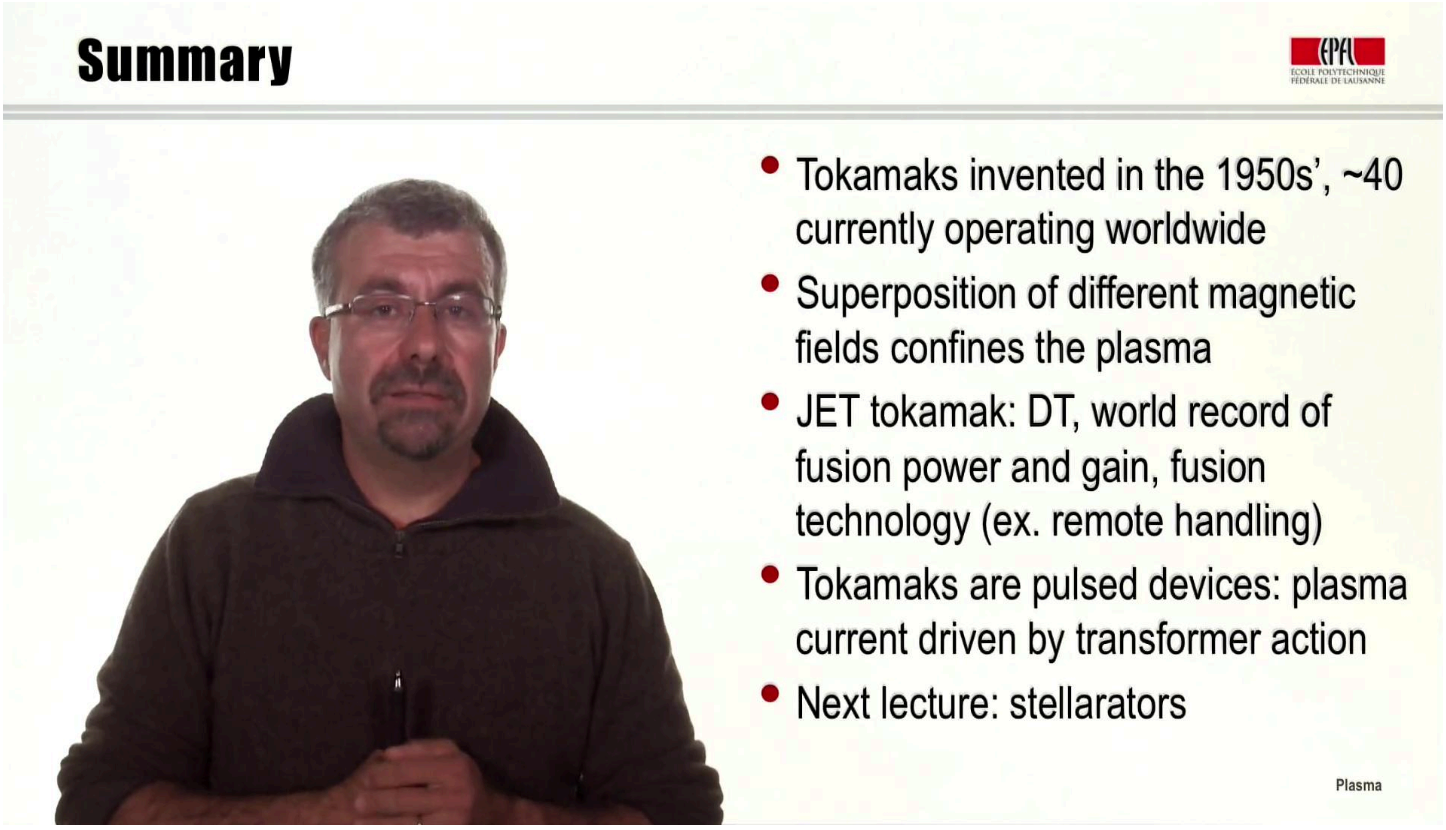
Plasma

Now so far we have described what we normally call the "conventional" tokamak concept. We see an illustration here. The conventional tokamak concept is a tokamak which has a large aspect ratio, R_0/a typically in excess of 3, between 3 and 4, 3.5 for instance, in the case of JET. The spherical tokamak concept we see here in this graph has an aspect ratio of R_0/a that is close to 1, much smaller than a conventional tokamak. We see here in fact the relative scale of the spherical tokamak and the conventional tokamak. Here we have an example of a spherical tokamak, again in the UK. The main advantage of a spherical tokamak is that despite the fact that in general we achieve a lower plasma pressure, there is a higher value of β, which is the ratio of the plasma pressure to the magnetic field pressure and this is because the spherical tokamak is a more compact structure, and so a smaller magnetic field is needed for confinement, which basically increases β. However, as we see here in the photo the discharge of a spherical tokamak, the plasma can get really close to this column here, that's called the "central column" in the spherical tokamak. This central solenoid is then exposed to the plasma and therefore care must be taken to avoid the plasma touching the center solenoid.

Notes

Summary

27m 54s



Now let's summarize our lecture. Tokamaks have been invented in the 50s and there are around 40 currently operating tokamaks worldwide. In a tokamak, the plasma is confined, thanks to the superposition of different magnetic fields. For the JET tokamak -it's a DT tokamak- holding the world record of fusion power and fusion gain and it is a great example of fusion technologies, for instance, remote handling. It's important to know that tokamak are pulsed devices, the plasma current is driven by transformer action. In the next lecture, we will look at the *stellarator* that is an example of confinement that in principle can be steady state.

Notes

Summary

29m 30s