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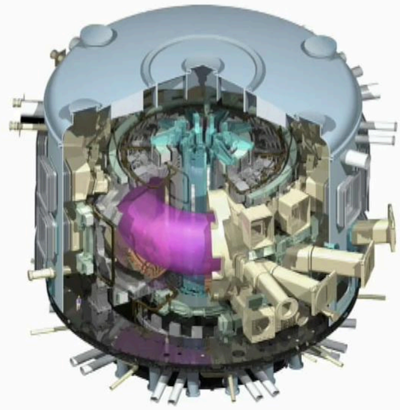
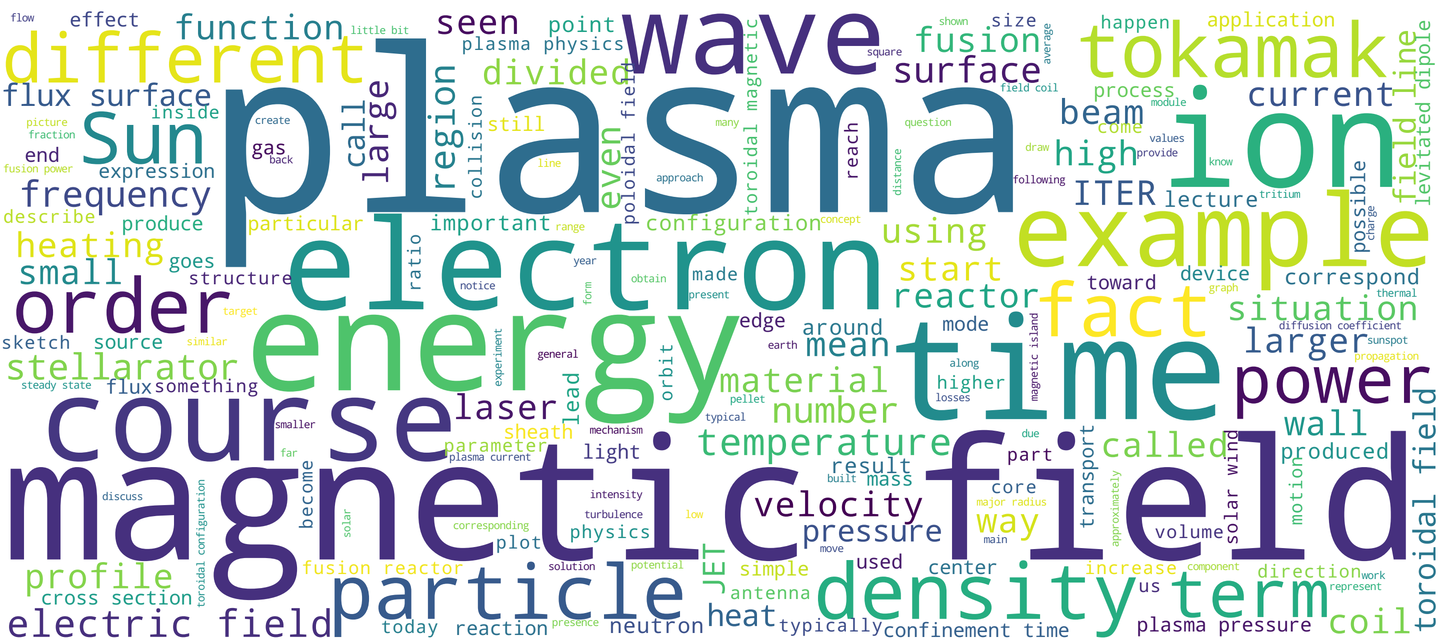


The stellarator and other confinement concepts

Plasma Physics and Application to Fusion Energy, Astrophysics and Industry

Lecture 7b

Duccio Testa



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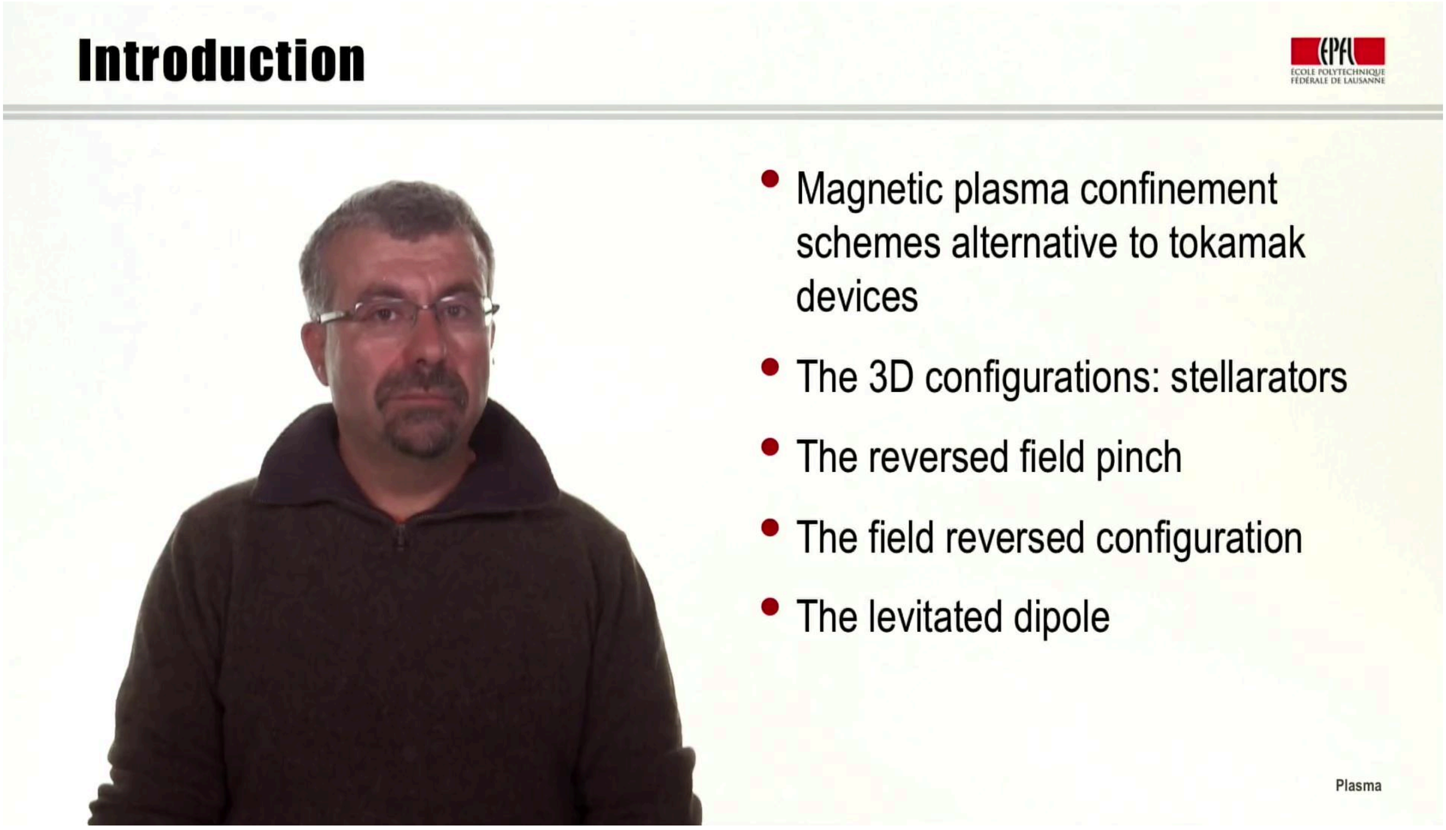


Welcome to the course Plasma Physics and Applications to Fusion Energy, Astrophysics and Industry. My name is Duccio Testa, and in this lecture we will discuss the stellarator and other confinement concepts. We will discuss magnetic plasma confinement schemes which are alternative to tokamak devices. We will start with 3D configurations: this is the stellarator, and then we will look at other concepts: the reversed field pinch, the field reversed configuration, and the levitated dipole.

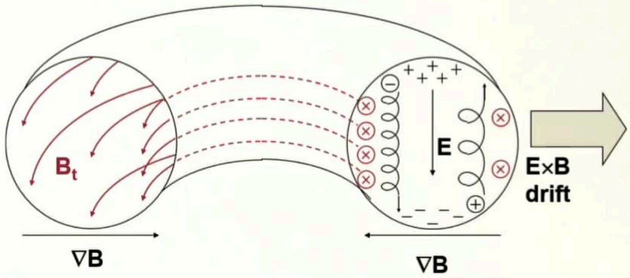
Notes

Summary

0m 05s

2D and 3D magnetic toroidal configurations

- How can charge accumulation be short-circuited in toroidal device?
- By using a toroidal current in a 2D axisymmetric configuration (tokamak)
- By breaking the axial symmetry (3D configurations)
  - Torsion of the magnetic axis
  - 3D modulation of the flux surfaces



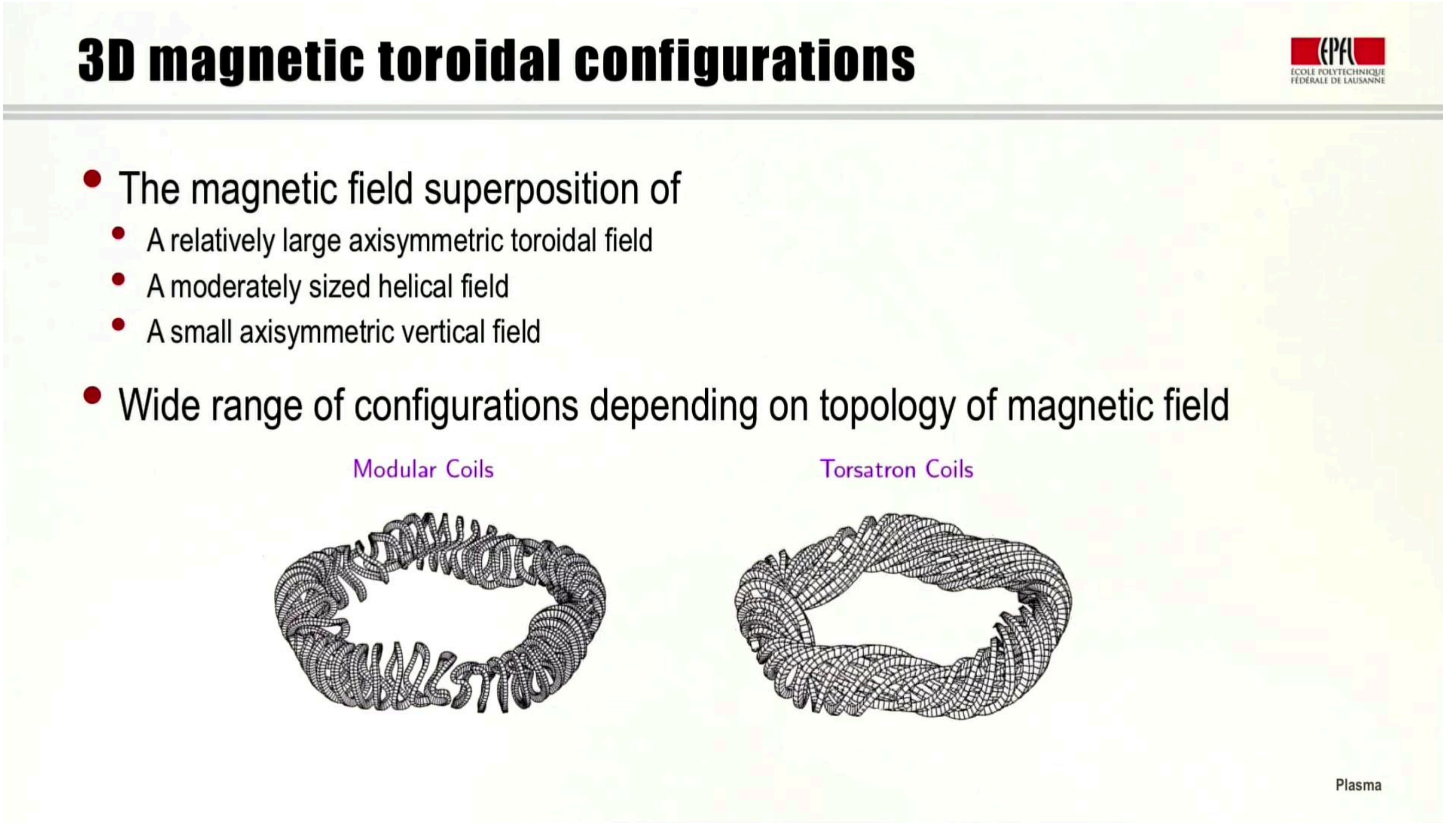
Plasma

Let's look at the confinement in 2D and 3D magnetic toroidal configuration. We've seen already that in the presence of a toroidal magnetic field, Bt, we have a ∇B drift of particles that goes in the opposite direction for particles with a different charge and this creates an electric field. So the question is how can charge accumulation be short-circuited in a toroidal device, because this electric field cross the magnetic field produces an E×B drift that pushes the whole plasma out. In fact, we have two methods to short-circuit this charge accumulation. The first is using a toroidal current in a 2D axisymmetric configuration. This leads to the tokamak concept. The second method is by breaking the axial symmetry — so we keep the 3D configuration so we can short-circuit this charge accumulation by torsion of the magnetic axis or by 3D modulation of the flux surfaces.

Notes

Summary

0m 38s



3D magnetic toroidal configuration: we have a magnetic field that is the superposition of a relatively large axisymmetric toroidal field, a moderately sized helical field and a small axisymmetric vertical field. Depending on the topology of the magnetic field we have a wide range of configurations and here we can see — two of them. If you have coils that are modular coils, then we see a configuration that has this particular shape. If we have coils that are called torsatron coils, that torsion around the plasma, then we have a torsatron configuration.

Notes

Summary

1m 44s

Stellarators: helical 3D magnetic field

Must avoid large magnetic islands and extended regions in which the magnetic field is stochastic

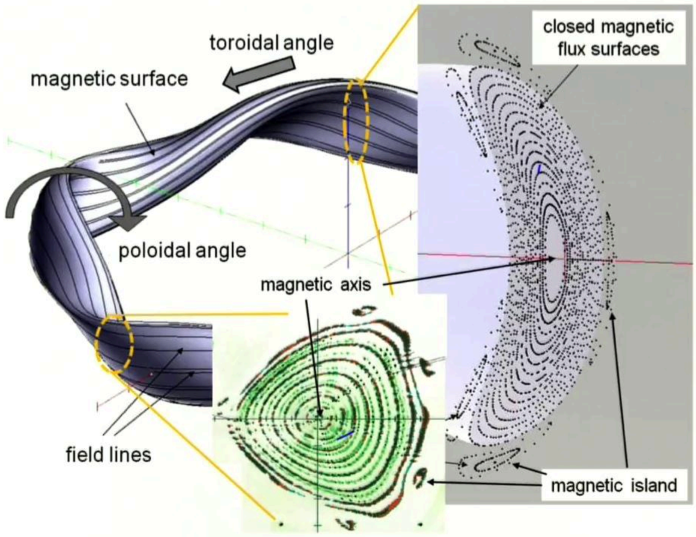


Image Credits: ORNL

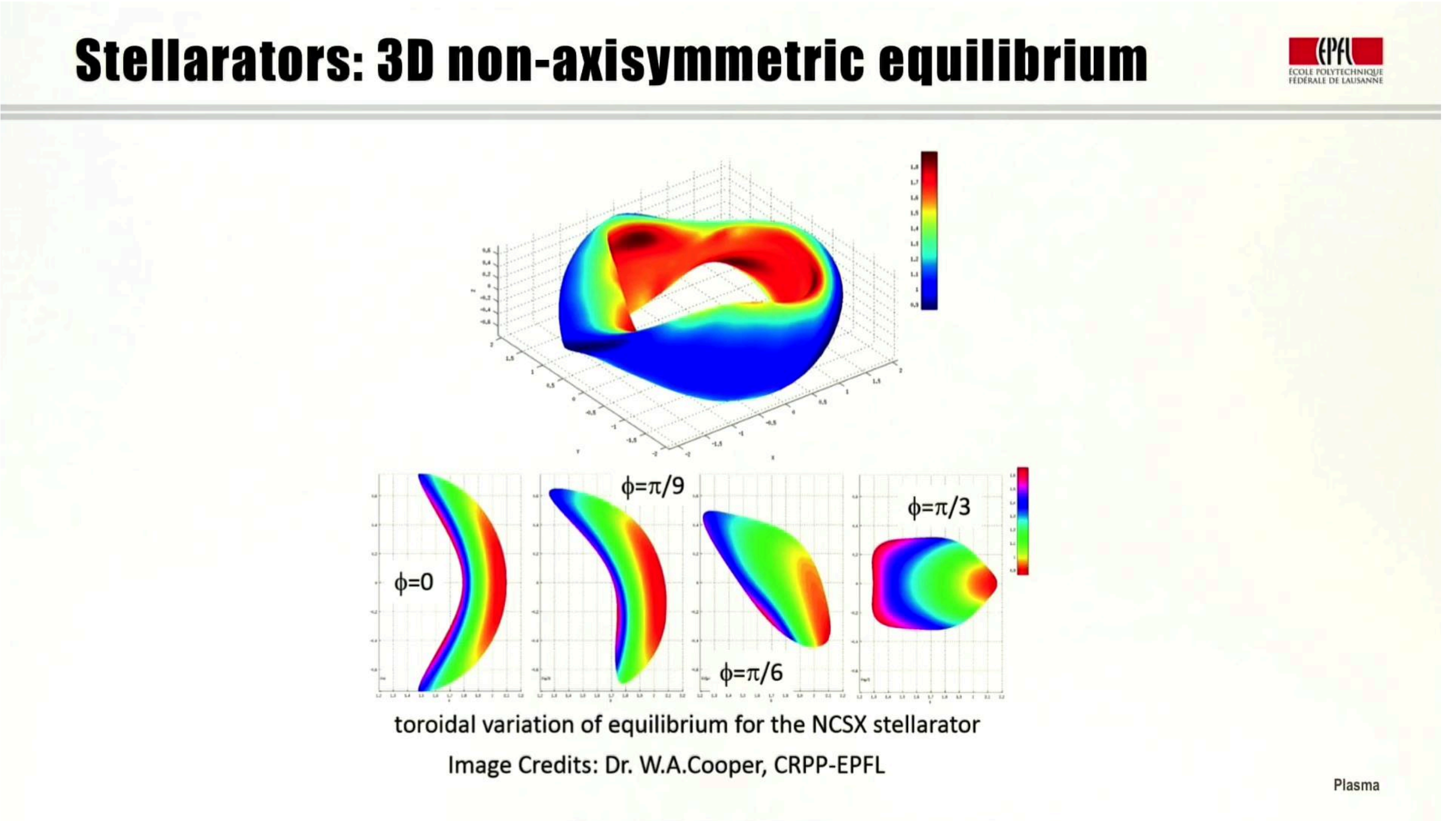
Plasma

In the stellatator, one particular example of 3D magnetic configuration, the field is 3D. What we must do in the stellarator is avoid large magnetic islands and extended regions in which the magnetic field is stochastic, because these are regions where the plasma confinement is very poor. We see our sketch of a stellarator, the toroidal angle in the magnetic surface, and we see that these surfaces are torsioned in 3D. Here we have a sketch of flux surfaces in this region of the plasma, and we see that the flux surfaces are closed upon this end, so this is a situation of good confinement. In this other region of the plasma, this is a sketch with flux surfaces, and what we see is that in the center we have well-behaved flux surfaces, but towards the edge, here, here, and here, we have magnetic islands. So we move from a region of closed magnetic flux surfaces to magnetic islands. These are regions of poor confinement. So the main purpose of the stellarator configuration is to build up coils so that we remove, as much as possible, regions where we have magnetic islands or a stochastic magnetic field.

Notes

Summary

2m 26s

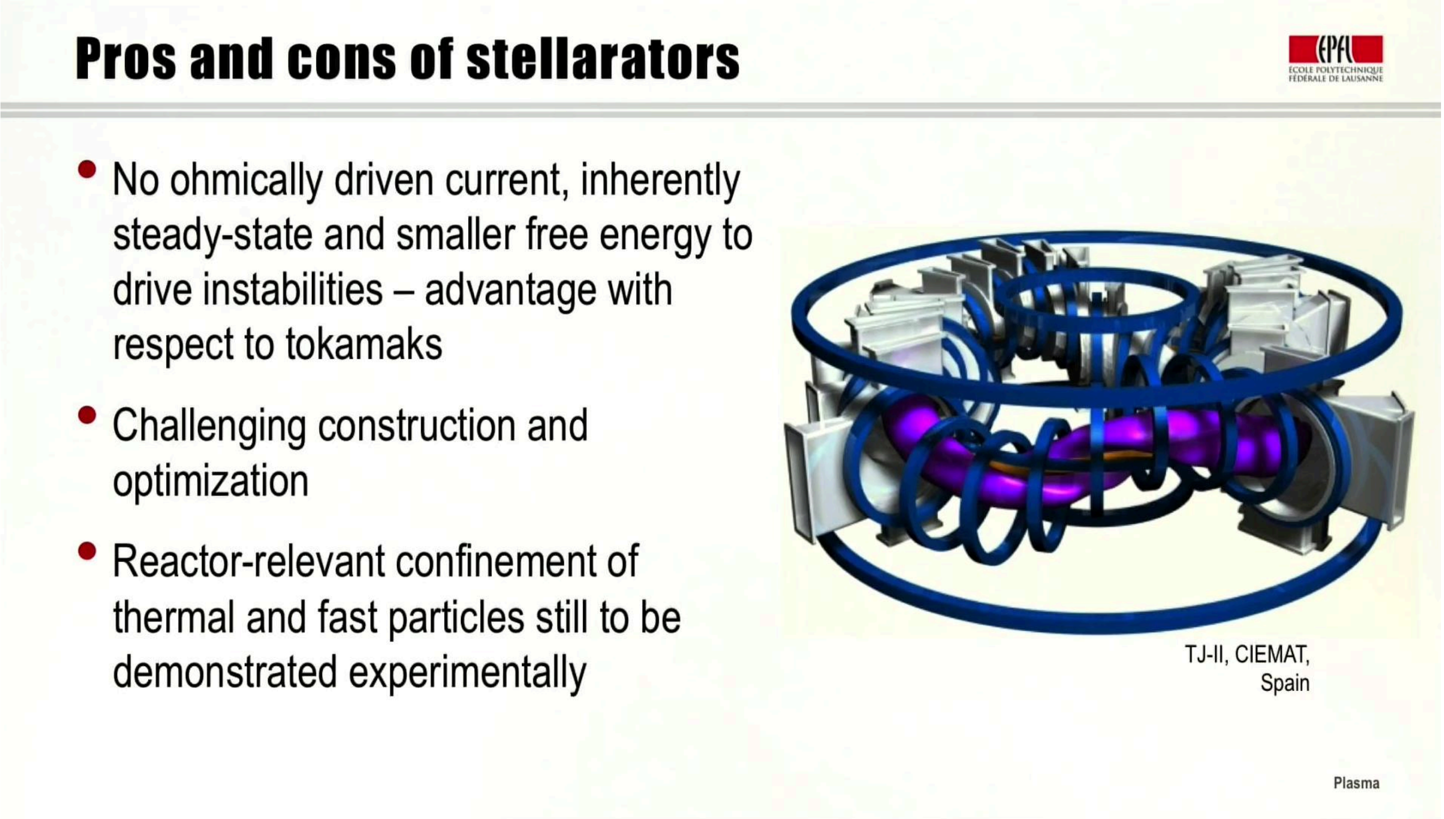


This is an example of 3D non-axisymmetric equilibrium for a stellator. What we plot here is a graph of the module of the magnetic field in 2D, in this case toroidal angle and poloidal angle, and here we see cuts of the $|B|$ graph at different toroidal positions — $\varphi = 0, \pi/9, \pi/6, \pi/3$. We see that the flux surfaces are very different. We start with a banana that is aligned, effectively, vertically and then by moving off a ninth of π in the toroidal direction, the banana shifts obliquely. Then it becomes an ellipse, and then it actually becomes a very complicated shape, a square with a triangle at the end. So here we see how the geometry of a stellarator is very complex.

Notes

Summary

3m 48s



The stellarator concept has pros and cons. The first pro is that the magnetic field is entirely driven by coils that are located outside the plasma, so there is no ohmically driven current, contrary to the tokamak. Therefore, the stellarator is an inherently steady-state device. Also, we have less free energy that can be tapped into to drive instabilities. This is a clear advantage with respect to tokamaks. However, we need to design very precisely the structure of the magnetic field — here we see an example for the TJ-II stellarator in CIEMAT in Spain, and this is a challenging construction and a challenging optimization. Furthermore, the reactor-relevant confinement of thermal and fast particles is still to be demonstrated experimentally.

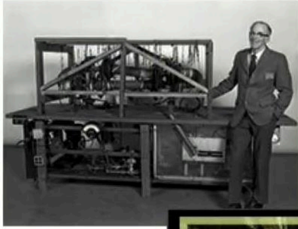
Notes

Summary

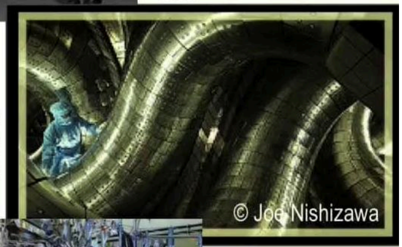
4m 45s

Stellarators: a brief history

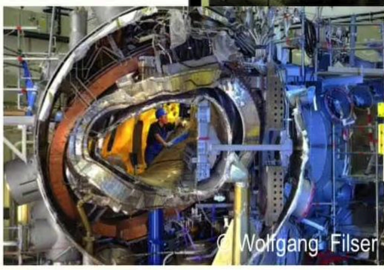
- Original idea by Lyman Spitzer (1951)
- After Geneva conference (1958), stellarator programs started in Germany, UK, Soviet Union, Japan
- Tokamaks showed better performances than stellarators, becoming mainstream
- Stellarator research was not discontinued: a number of small stellarators exist worldwide (TJ-II in Spain, HSX in Wisconsin, …)
- Largest existing stellarators: LHD in Japan and W7-X soon operational in Germany



Stellarator A, Princeton Plasma Physics Laboratory, USA



LHD, NIFS, Japan



W7-X, IPP Greifswald, Germany

Plasma

Let's look at a brief history of stellarators. The original idea was developed by Lyman Spitzer in 1951. After the Geneva conference in 1958, stellarator programs started in Germany, in the UK, in the Soviet Union, and in Japan. At the time, tokamaks showed better performances than stellarators, becoming mainstream. The stellarator research was not discontinued. In fact, a number of small stellarators exist worldwide — TJ-II in Spain, HSX in Wisconsin — the United States — and now we are moving towards very large stellarators — the LHD in Japan and the W7-X stellarator that will soon be operational in Germany.

Notes

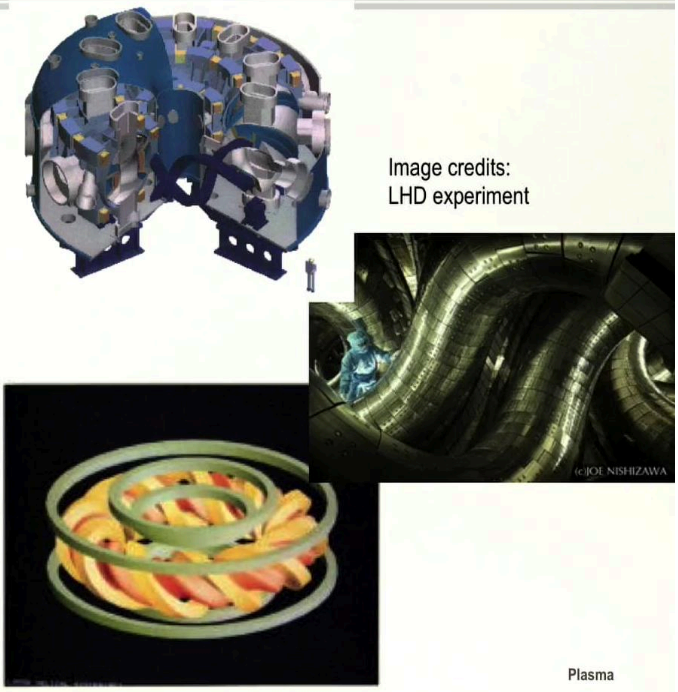
Summary

5m 37s

The LHD (Large Helical Device) stellarator

- Located in Toki, Gifu, Japan
- Largest stellarator now operating in the world
- Two superconducting helical coils, and six vertical field coils
- Initially optimized for MHD equilibrium and stability, then improved for single particle confinement

$$\beta \lesssim 5\%, \tau_E \lesssim 0.4 \text{ s},$$
$$n\tau_E T_i \simeq 2.2 \times 10^{19} \text{ keVm}^{-3}\text{s}$$

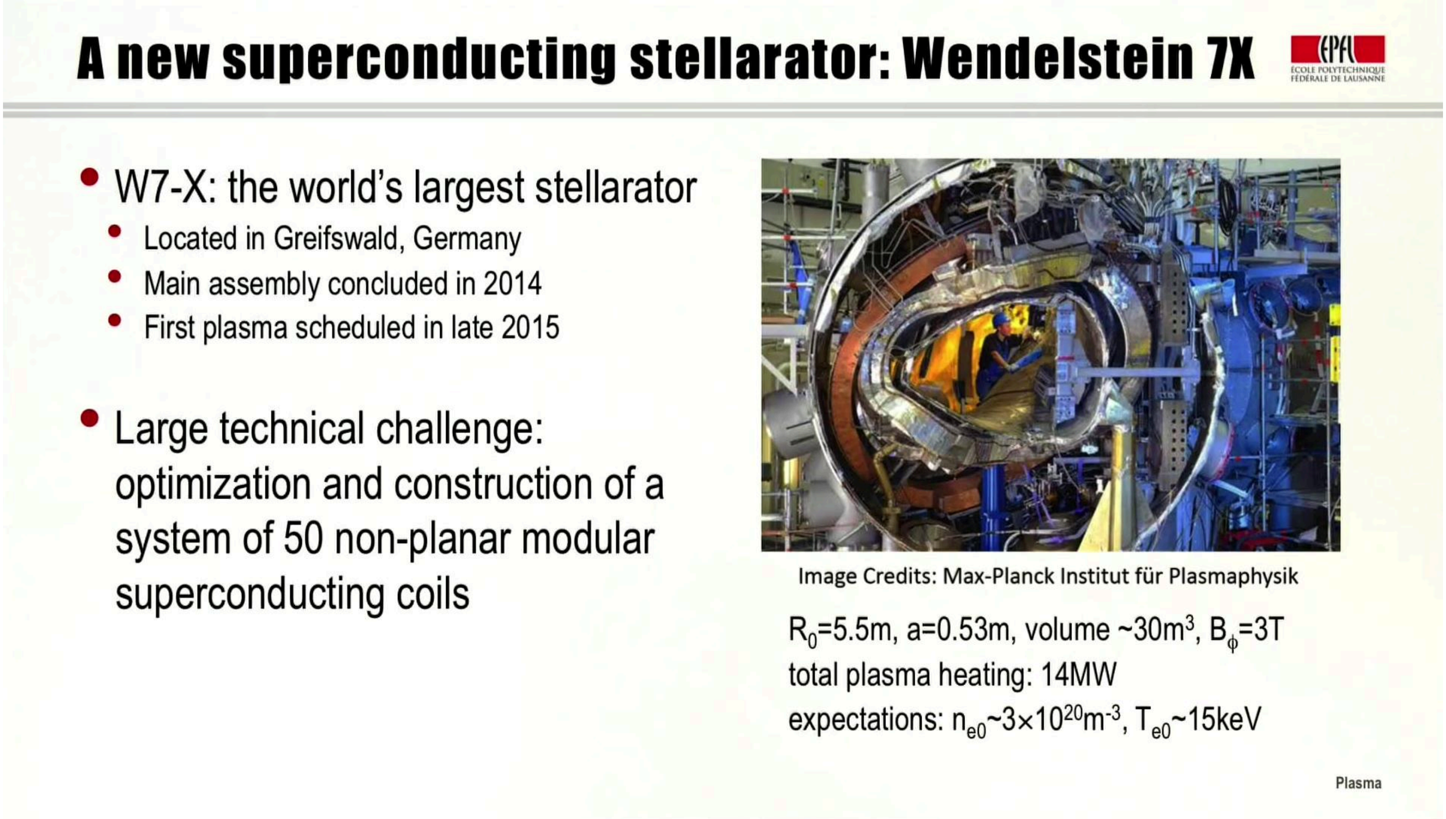


Let's look a little bit at the LHD stellarator. LHD stands for Large Helical Device. We see a few images of the LHD. The LHD is located in Toki, Japan, and is the largest stellarator now operating in the world. The magnetic field is produced by two superconducting helical coils and six vertical field coils. It was initially optimized for MHD eqilibrium and stability and then improved for single particle confinement. LHD can reach a value of the plasma β that is up to 5% and an energy confinement time τE that is up to 0.4 seconds, and then leads to a triple product n τE Ti that is of the order of 2.2×10^{19} [KeV / (m^3 s)]. So if we compare with the value of the triple product in JET that is on the order of 2.6×10^{20}, we see that there is still a factor ten of difference between the stellarator and the tokamak concept in terms of the triple product.

Notes

Summary

6m 16s



The Wendelstein 7X is a new superconducting stellarator that has been built in Germany. It is the largest stellarator. The main assembly was completed in 2014, and the first plasma scheduled for late 2015. It's a very big technical challenge. In fact, the construction relies on the optimization of a system of 50 non-planar modular superconducting coils. We see here a photo of the W7-X stellarator under construction, we see the wall of the stellarator that is a very particular shape, with a bend inside, and the parameters of the 7-X stellarator are a major radius of 5.5 m, a minus radius of 1 m, a volume of 30 m^3, and a toroidal magnetic field that is up to 3 T. The total plasma heating can be up to 14 megawatts and the expected on-axis density is 3×10^{20} m^{-3} which is a similar value to the JET tokamak, and an on-axis electron temperature of 15keV, that again is a similar value to that of the JET tokamak.

Notes

Summary

7m 22s

A new superconducting stellarator: Wendelstein 7X

- Optimized taking in consideration the orbits of all particles
- Energy and particle confinement expected to be comparable to those of a similar-sized tokamak
- Plasma discharges up to 30min to demonstrate steady-state operation

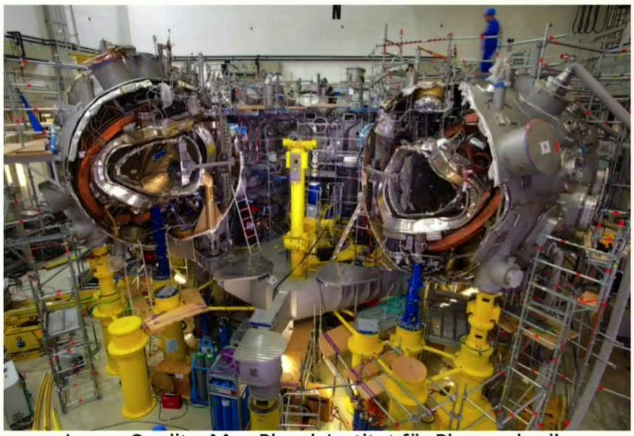


Image Credits: Max-Planck Institut für Plasmaphysik

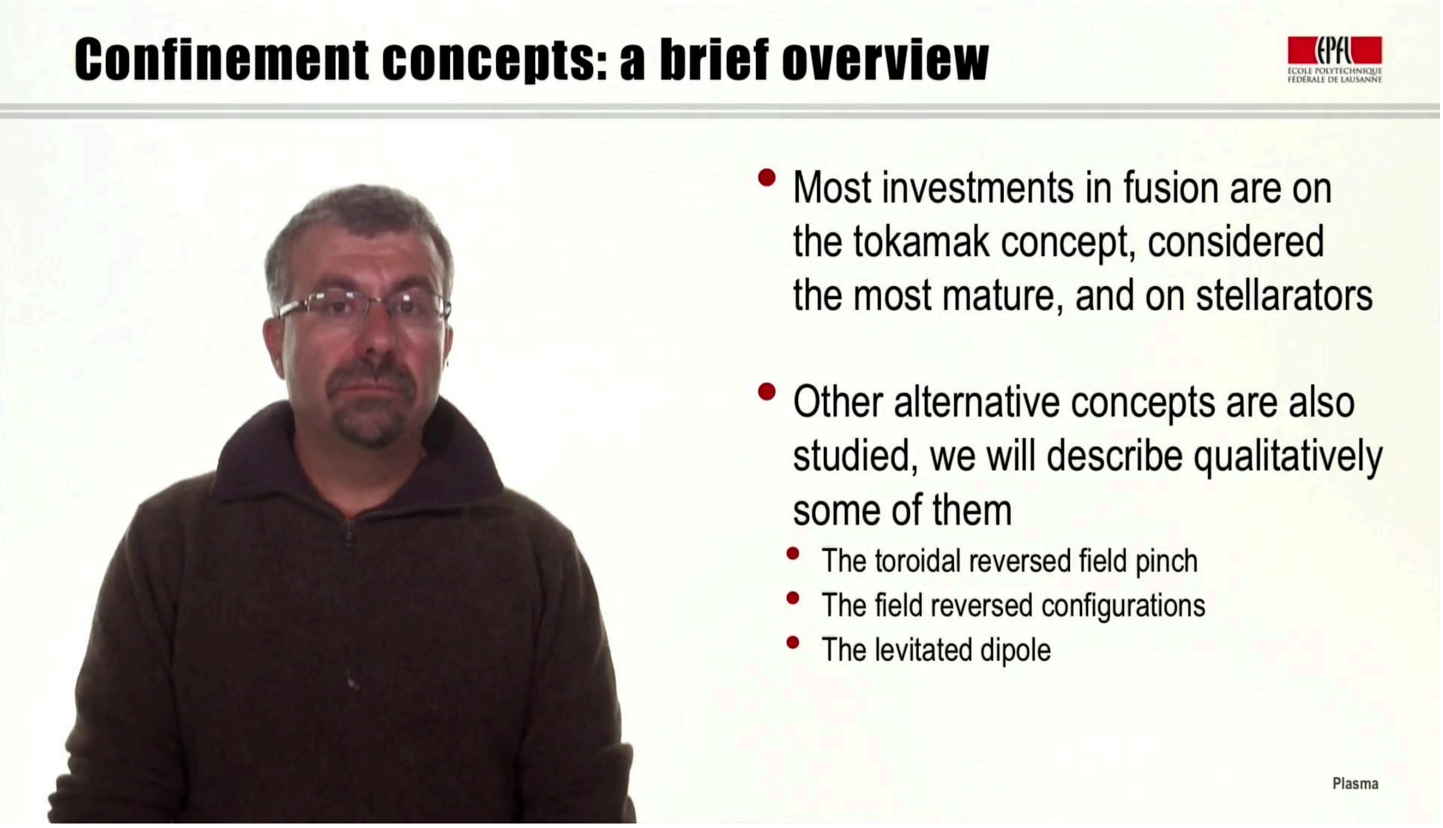
Plasma

The W7-X stellarator has been built and optimized taking into consideration the orbit of all particles. The energy and particle confinement time are expected to be comparable to those of a similar-sized tokamak. The main advantage of the W7-X stellarator is that it is a superconducting device and therefore plasma discharges can last up to 30 minutes to demonstrate the steady-state operation of the stellarator concept.

Notes

Summary

8m 39s

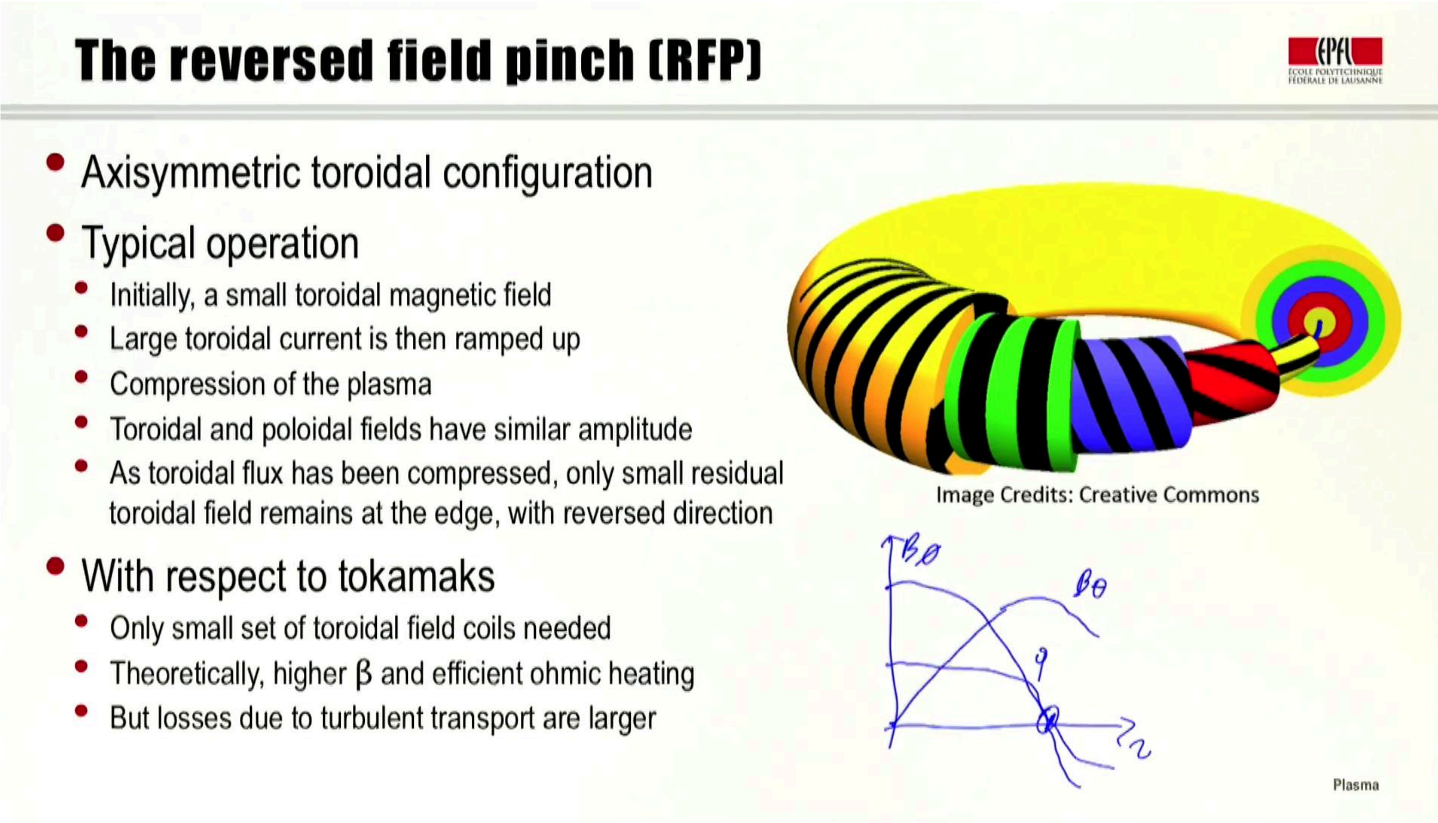


Let's have a brief overview of the confinement concepts that we have described so far. Most current investments in fusion are in the tokamak concept, that is considered the most mature, and on stellarators. However, there are many other alternative concepts that are studied, and we will describe some of them qualitatively. We will describe the toroidal reversed field pinch, the field reversed configuration, and the levitated dipole.

Notes

Summary

9m 07s

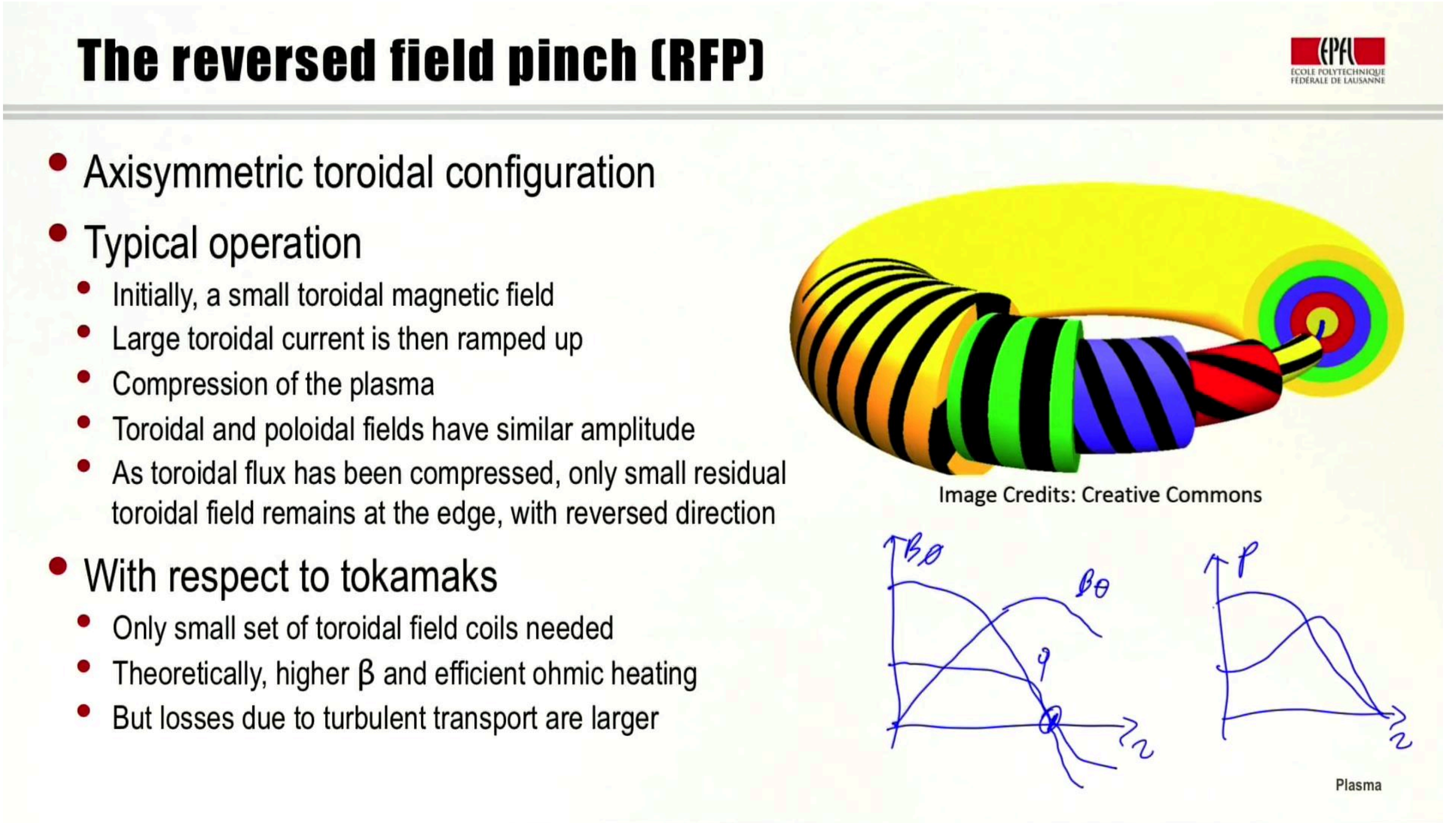


Let's look at the *reversed field pinch* that is commonly abbreviated as RFP. It is an axisymmetric toroidal configuration. Let's look at the principle of operation. Initially, we start with a small toroidal magnetic field. Then we have a large toroidal current that is ramped up, the plasma is compressed, and we end up in a situation where the toroidal and poloidal fields have very similar amplitude. As the toroidal flux has also been compressed, what remains is only a small residual toroidal field at the edge that also reverses direction. We can now draw the profiles that we can have for an RFP for the magnetic field and plasma pressure. We can look at the profiles in terms of the minor radius coordinate r. We can start with the toroidal magnetic field that is large in the plasma center, and then it crosses zero towards the plasma edge. The poloidal magnetic field is small in the plasma center, it increases towards the plasma edge, and then drops down a little bit. We can now draw the safety factor in an RFP, the q [profile] which is basically the ratio of the toroidal field to the poloidal field. So the safety factor goes to zero where the toroidal magnetic field goes to zero.

Notes

Summary

9m 34s

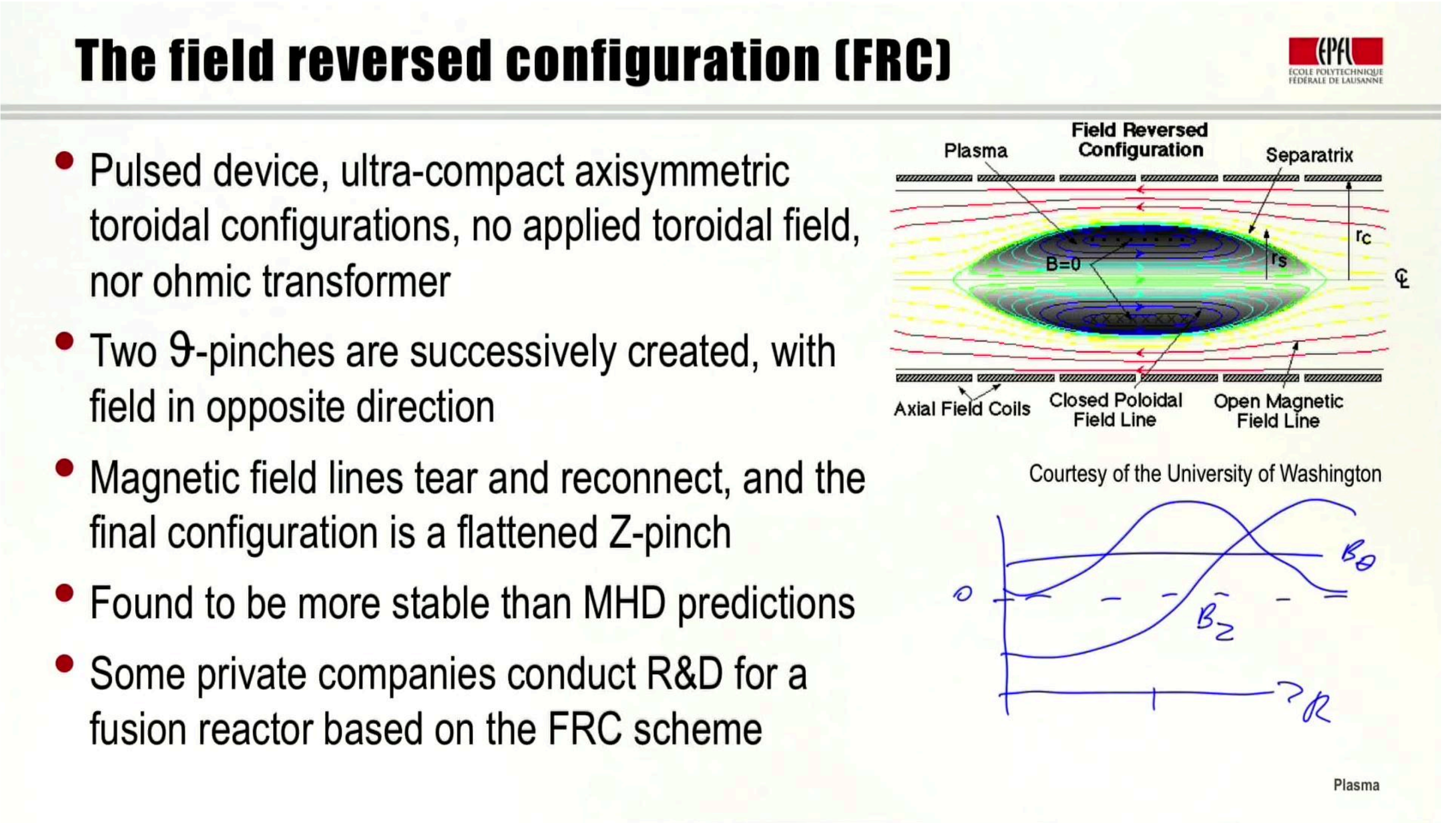


For the plasma pressure we can have a different profile, obviously. Typically, the plasma pressure peaks on the magnetic axis, but we can also have a configuration where the plasma pressure peaks off the magnetic axis. With respect to the tokamak, the RFP has one main advantage — that only a small set of toroidal field coils is needed. Theoretically, we achieve higher β and efficient ohmic heating. However, the losses due to turbulent transport are larger than those observed in a tokamak.

Notes

Summary

11m 04s



Let's now look at the *field reversed configuration* [FRC]. We see here a schematic drawing of this system. FRC's are pulsed devices, they have an ultra-compact axisymmetric toroidal configuration, there is no applied toroidal field, there is no ohmic transformer. FRC's are built by using two θ-pinches that are created successively with fields in opposite directions. The magnetic field lines first tear and then reconnect, and the final configuration is that of a flattened Z-pinch. We can now look at the profiles of the magnetic field and the plasma pressure in an FRC. The best way to draw them is as a function of the major radius coordinate R. We can look at the plasma pressure that is typically high at the center of the device. Then we can look at the axial magnetic field, Bz, that is negative on one side, goes to zero in the center of the device where the pressure peaks and becomes positive on the other side of the device. Then we have a poloidal field, that is typically small, that is relatively flat across the radial direction. The main advantage of the FRC configurations is that they are found to be more stable than what is predicted by MHD calculations. In fact, some private companies are conducting R&D studies for a fusion reactor scheme based on the FRC concept.

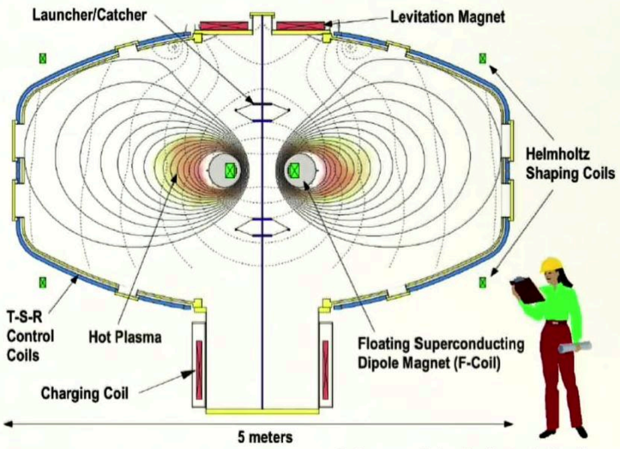
Notes

Summary

11m 39s

The levitated dipole

- Motivated by astrophysical observations (stable plasma rings in Jupiter dipolar field at very high β)
- Uses a dipolar field to confine particles produced by levitated superconducting magnet
- Unfavorable curvature everywhere limits β, but stable at high β–values
- D-T operation not possible, more challenging D-D reaction has to be used



Scheme of the levitated dipole experiment (LDX), MIT and Columbia University

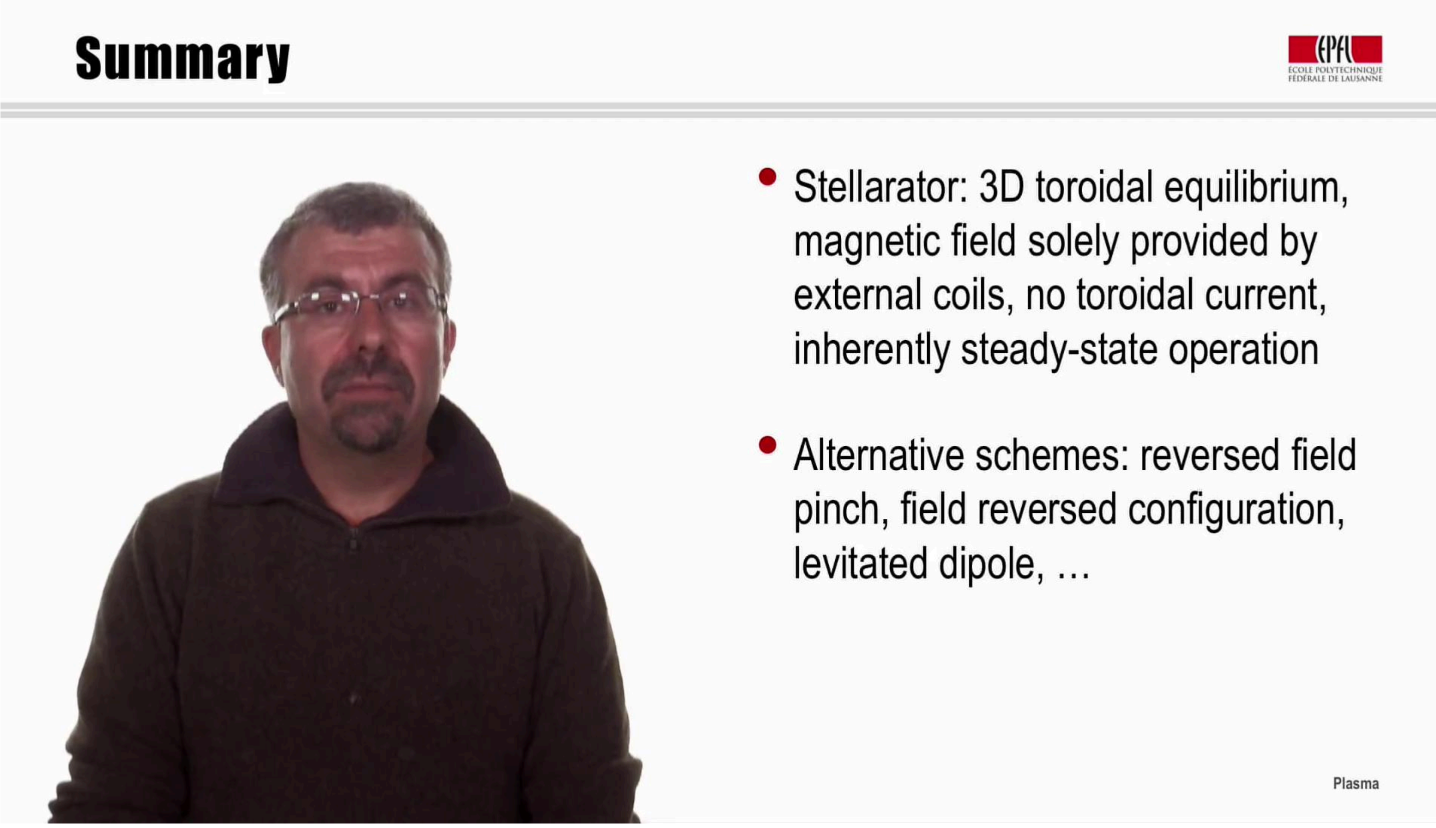
Plasma

Let's look at the *levitated dipole* experiment. We see a sketch of this experiment at MIT. This experiment is motivated by astrophysical observations of stable plasma rings in Jupiter's dipolar field at very high β. The levitated dipole experiment, LDX, uses a dipolar field to confine particles that are produced by a levitated superconducting magnet. The magnetic field has an unfavorable curvature everywhere, and that limits β in the plasma center. However, the magnetic field is that of a dipole, so it has a $1/r^3$ dependence, β, that is proportional to the B^{-2}, increases rapidly towards the plasma edge. Therefore, the levitated dipole is very stable at very high β in the outer region of the plasma. As in Jupiter — the β in Jupiter's dipolar field is of the order of 2, which is much larger than the typical β in a tokamak, that is a fraction of 1. The main drawback of our levitated dipole experiment is that D-T operation is not possible, therefore a more challenging D-D reaction has to be used.

Notes

Summary

13m 21s



We can now summarize what we have discussed in this lecture. We started with the stellarator. It's a 3D toroidal equilibrium. It's a system that allows inherently steady-state operation because there is no need for any ohmically driven toroidal plasma current, and the magnetic field is solely provided by external coils. There are many alternative confinement schemes that are studied in fusion research, and we have seen a few of them. We have looked at the reversed field pinch, at the field reversed configuration, and at the levitated dipole experiment.

Notes

Summary

14m 40s