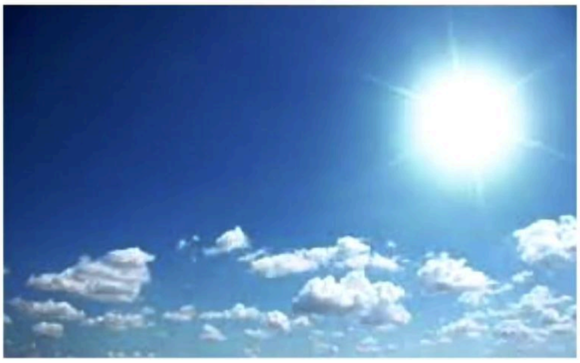
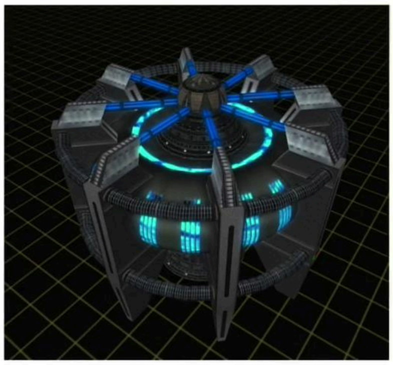


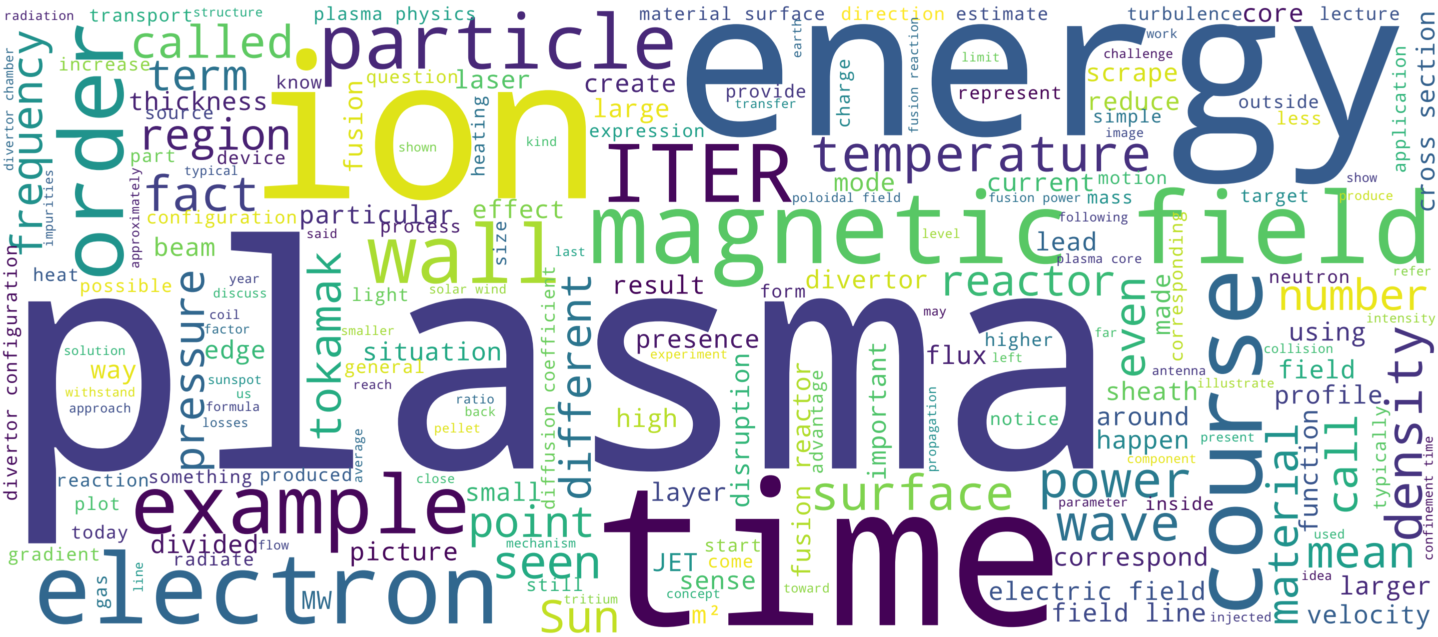


Fusion: plasma wall interactions

Plasma Physics and Application to Fusion Energy, Astrophysics and Industry

Lecture 6e

Ambrogio Fasoli



Search MOOC

Video

Introduction

Welcome to the course on Plasma Physics and Applications. In the last lecture, we have derived together a very simple design of a fusion reactor based on the magnetic confinement approach. We have seen that, in our minimization of the cost and maximization of the output power, one key parameter was standing out, and that was the wall loading that we can tolerate. That is, the maximum power that the plasma-facing walls can withstand before being damaged. That's no wonder because, of course, what we're trying to do in a fusion reactor is to trap matter that's hotter than at the surface of the sun into a material object, which is the reactor itself. Today we will explore, a little bit, this question of the interaction between the plasma and the surrounding walls in the reactor by looking at the requirements of the reactor first wall, by looking at the concepts of limiters and divertors, by defining the scrape off layer and seeing what its characteristics are, by looking at the advantages of the divertor concept, and why we are actually pursuing that as a possible solution. We will also explore what the plasma facing materials are that are foreseen for ITER, and what the challenges are that remain for the divertors to be optimized. And in doing that, we'll illustrate a few ideas on how to go forward, and propose innovative divertor configurations.

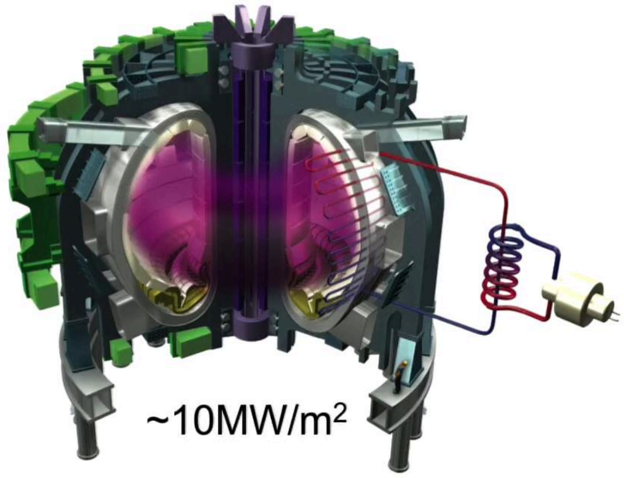
Notes

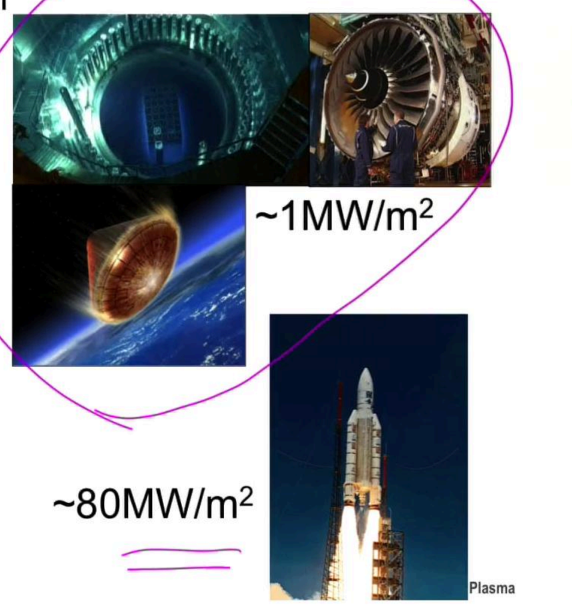
Summary

0m 06s

Requirements for reactor first wall

- Withstand very large heat fluxes on the material
  - Limit erosion, melting





First of all, the reactor first wall has to withstand very large heat fluxes on the material. Has to withstand, in a sense, that erosion has to be very limited and there has to be no melting. These pictures give you an idea of what the fusion reactor compares to. We have seen that we to have about 5, say, over to 10 MW per m^2 in the reactor. What object can we compare that to? In the pictures here, we see objects that are normally subject to power fluxes of the order of a megawatt per m^2: so much less than the fusion reactor. That is the case of a fission reactor; that is the case of a large turbine; and that is the case of a spacecraft re-entering the atmosphere. We go to a different, kind of, order of magnitude, and that's even more difficult than the fusion reactor, when we go to the surfaces that are exposed to fluxes of the order of 80 MW per m^2 when a rocket departs.

Notes

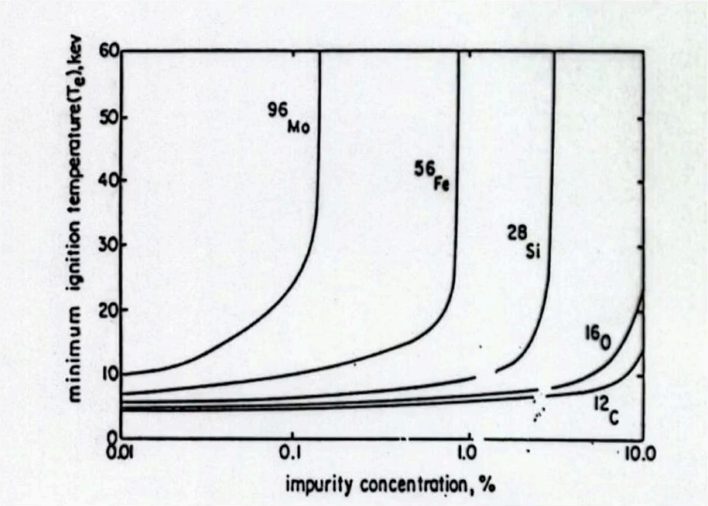
Summary

1m 34s

Requirements for reactor first wall

- Keep the plasma pure

$$\frac{P_b}{\text{volume}} \simeq A n^2 Z_{\text{eff}} T_e^{1/2}$$



Minimum ignition temperature goes up with impurity concentration

Plasma

The reactor first wall has to keep the plasma pure. That is very important, for two reasons. If we inject impurities into the plasma volume, we have dilution of the plasma fuels. That is, we replace the ions with which we want to achieve fusion by ions that don't achieve any result for us. And also, if we inject impurities in the plasma, they radiate mainly by line radiation, if impurities are not fully ionized, or by bremsstrahlung radiation. And as we have seen together, the bremsstrahlung radiation depends on the charge and atomic number of the impurity. So if we look at the curve that shows the minimum ignition temperature as a function of the impurity concentration, we can see that that temperature, which is calculated here just using the bremsstrahlung as a loss channel -- we know this is an optimistic view -- but, even with that, the impurity concentration influences that minimum temperature, of course making it- going up, as the concentration goes up. But even more importantly, we notice here that that curve changes dramatically as you go from low Z impurities -- this is the case of carbon -- to high Z, such as iron and molybdenum. So you have to avoid the injection of impurities in the core, in particular, of high Z impurities.

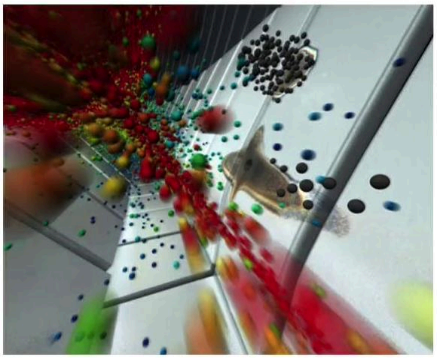
Notes

Summary

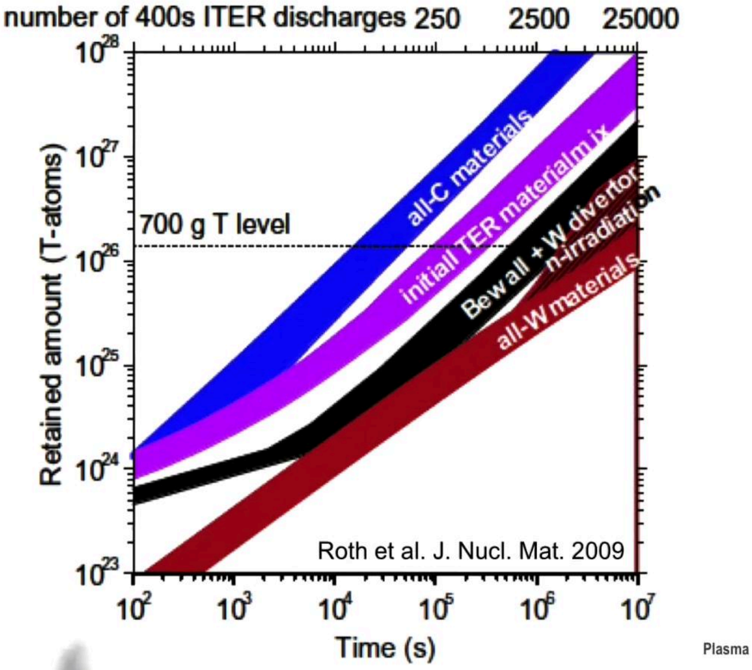
2m 40s

Requirements for reactor first wall

- Minimise retention of Tritium (co-deposition with Carbon)



Courtesy of Leena Aho-Mantila and Jyrki Hokkanen (CSC – IT Center for Science Ltd).



The reactor first wall has to, also, minimize the retention of tritium. Tritium is co-deposited with carbon, if we are in the presence of a carbon-based wall, and stays there. So that's, of course, unacceptable for a fusion reactor. In this pictorial view on the left, we see the mechanism with which the tritium is trapped on the surfaces, and, more importantly, on the right-hand side here, we see the retained amount of tritium in a wall as a function of time of plasma, in a sense. This is the case of ITER. Maybe you represent it as a function of time for the plasma, or as a function of these charges in ITER taken of a duration of 400 seconds. And what we see here is that we have a level which we cannot overcome, which is 700 grams. That's the level that we can afford having of tritium, of course, trapped in the wall of the reactor; we cannot go above that. But, actually, if we have an all-carbon-based wall, we do go above that, and we do go above that quite quickly, only after a few hundreds of ITER discharges. So that's not possible. So we need to go to the other options here. In particular, if we look at the red stripe here, that indicates the situation in which we have materials that all are tungsten-based.

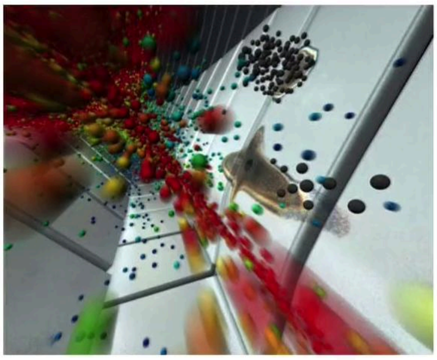
Notes

Summary

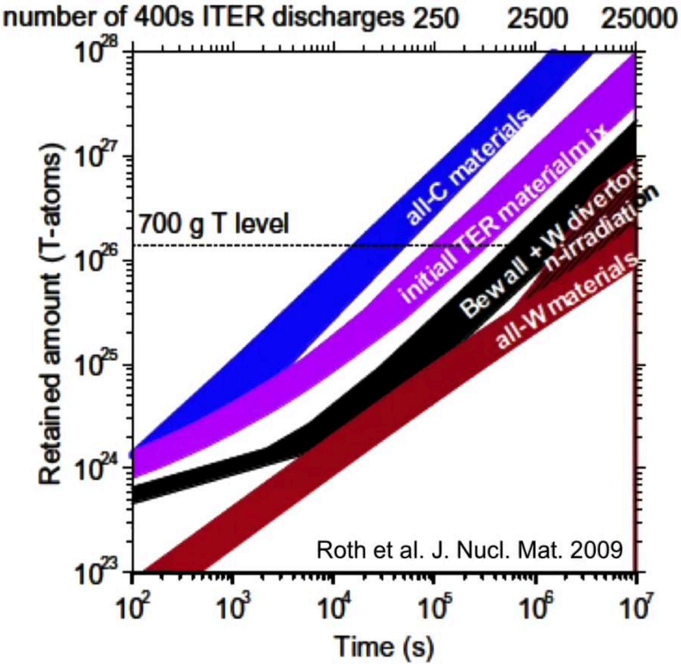
4m 07s

Requirements for reactor first wall

- Minimise retention of Tritium (co-deposition with Carbon)



Courtesy of Leena Aho-Mantila and Jyrki Hokkanen (CSC – IT Center for Science Ltd).



Plasma

And with these materials, we never go above that level, even if we have 25,000 discharges of ITER, at 400 seconds. So that's a situation that's compatible with the requirements for the inventory of tritium in the plasma walls.

Notes

Summary

5m 36s

Requirements for reactor first wall

- Withstand large heat fluxes, exhaust fusion and external heating power
- Keep the plasma pure
- Minimise retention of Tritium
- Minimise dust production
- Provide vacuum containment
- Fuel the plasma
- Remove Helium ashes (pumping)

Plasma

So to briefly recap: we have seen that the wall has to withstand large heat fluxes; and, of course, to exhaust both the fusion produced and the external heating power; it has to keep the plasma pure, and it has to minimize tritium retention. There are other constraints: the walls must minimize production of dust, which can create a very large perturbation at the edge of the plasma, and therefore affect the fusion performance. It has to be compatible with providing the containment for vacuum, and it has to be compatible with fueling the plasma -- that is, injecting the particles that need to be forming the plasma fuels, once they're ionized. And finally, it has to, also, contribute to pumping the helium ashes. This is the helium that has been produced by the fusion reaction. It has given, in the form of α particles, of course, its energy to the fuel ions themselves. So this is something that we need to take out, once it has given its energy.

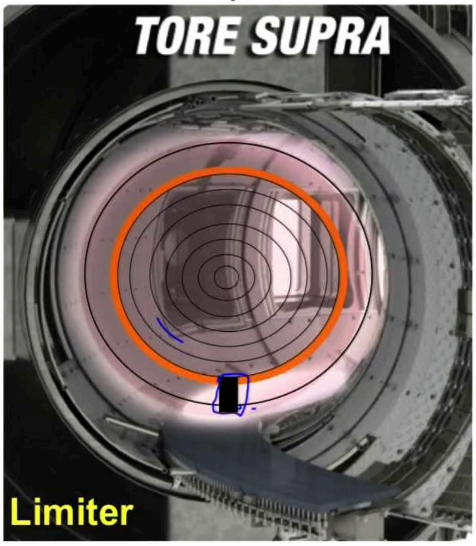
Notes

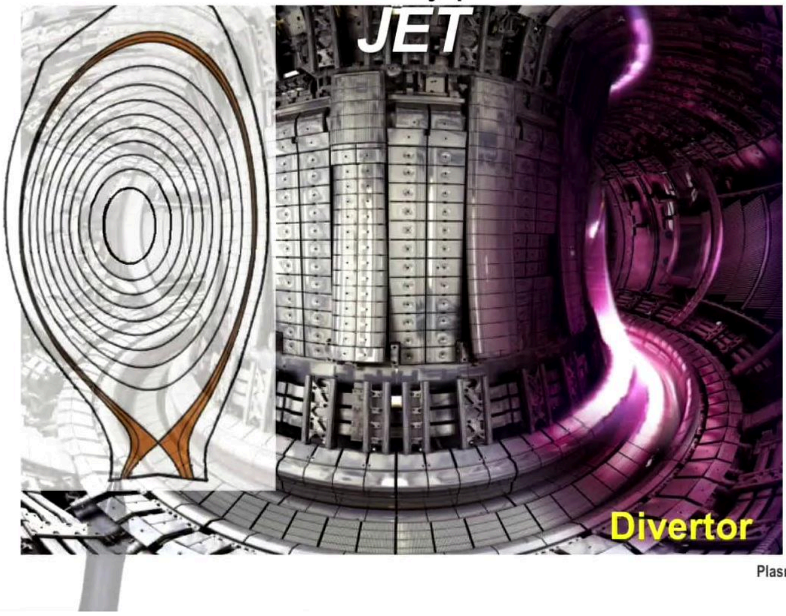
Summary

5m 52s

Limiter and divertor configurations

Direct contact of plasma with vessel wall must be limited to well-defined areas, which take the power carried by particles and not radiated by plasma





Plasma

Currently, there are two possible configurations for the part of the plasma that is close to the wall: what we call the *limiter*, and what we call the *divertor*. The question is that you have to define very well a specific area of the walls that is in direct contact with the plasma. That area is the one that will have to take the power that's carried by the particles issued from the plasma, and that's not being radiated before it arrives at the wall. So the limiter configuration is shown on the left. We have a set of, in this case, concentric field lines. They represent the flux surfaces in the plasma. And the outer part of them intercepts a material surface. This is just, of course, an abstract representation, but it's superimposed on an actual realization of a limiter configuration, which is in the Tore Supra tokamak in France. The limiter is quite large, in this case, so the portion of the plasma that's all the way to the outside actually directly touches this material surface. So the plasma is *limited*, we say, on that material surface. On the right-hand side, we have a picture of JET, which is an example of a tokamak that has a divertor.

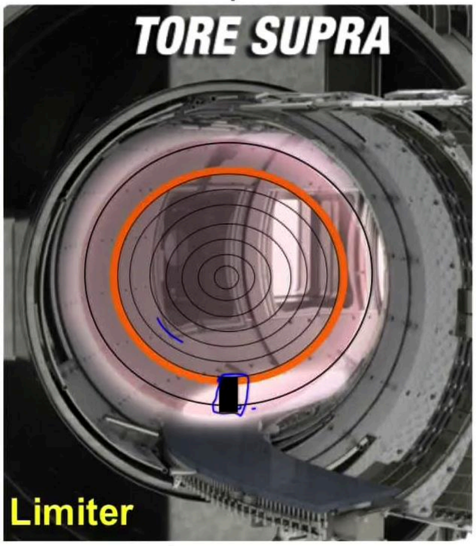
Notes

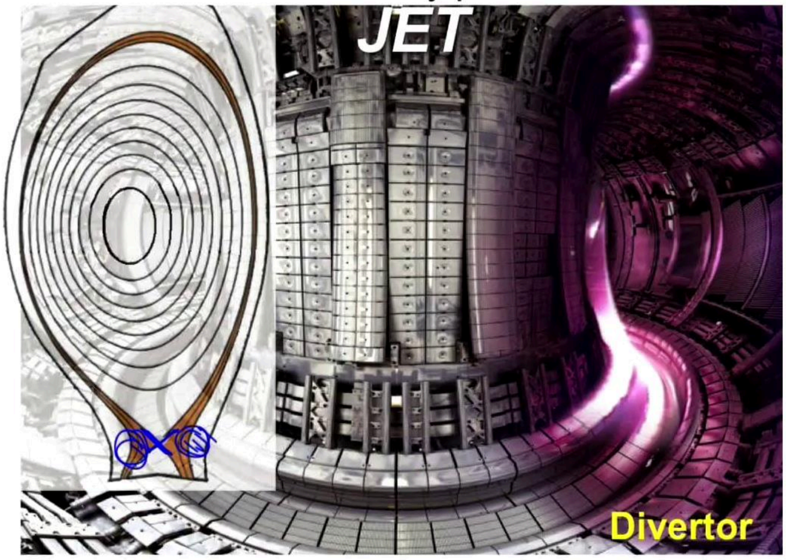
Summary

6m 48s

Limiter and divertor configurations

Direct contact of plasma with vessel wall must be limited to well-defined areas, which take the power carried by particles and not radiated by plasma





Plasma

A divertor is a magnetic configuration that provides a null point here for the poloidal field. We call this the *X-point*, because, of course, it can be seen as an X figure. And by doing that, we have, basically, a separation between the flux surfaces inside that are not touching any material surface and the lines that go around, that now are the only ones that actually touch a material surface. So we really have a much clearer separation between confined core and edge of the plasma. The X-point, of course, needs to be achieved with specific magnetic coils. In the case of JET, they are inside the vessel; in the case of ITER, they will be outside the vessel. And they have to provide, again, a null point for the poloidal field which is the X-point. In this image, we also see the light emitted by the plasma when it's run in the configuration of an X-point. And you can see, at the bottom here, we have a so-called divertor chamber which is corresponding to this region, these two regions here, on the picture.

Notes

Summary

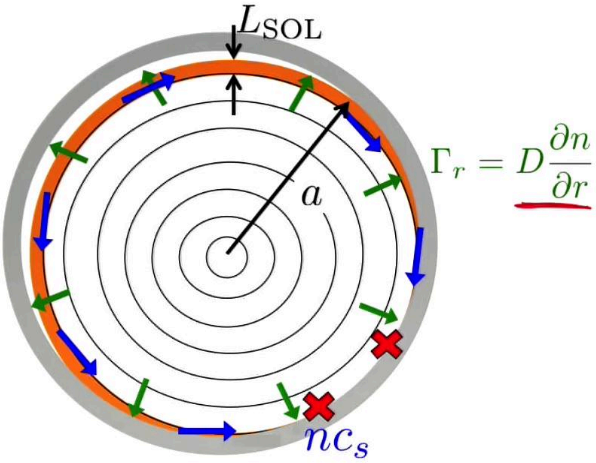
8m 11s

The scrape off layer - SOL

Outer layer of plasma in direct contact with the material wall

SOL thickness results from balance between cross-field and parallel dynamics

Ex. of estimate for limiter case



Plasma

Let's consider, now, what we call the *scrape off layer*, or SOL. This is the outer layer of the plasma that's in direct contact with the material wall. It plays a very important role, because it determines, really, the effect of the plasma on the wall itself, and it also has influences on the plasma core behavior, in some circumstances. The thickness of the SOL results from a balance between cross-field and parallel dynamics. And we can estimate that for a simple case of a limiter configuration, in a circular plasma. So this is the sketch I'd like to concentrate on. This orange stripe is my scrape off layer. It will touch the material surface, and that's where the parallel flow will play a role in the sense that the flow along the poloidal field lines will actually impinge on the limiter there. The thickness of that channel, if you like, is *L*sol, so my thickness of the SOL- of this scrape off layer that I would like to estimate. And that has to be compensated -- that has to be fed, in a sense -- by the perpendicular flux coming from the simple Fick's law. So that's the cross-field transport due to the gradient of the density times a diffusion coefficient, *D*.

Notes

Summary

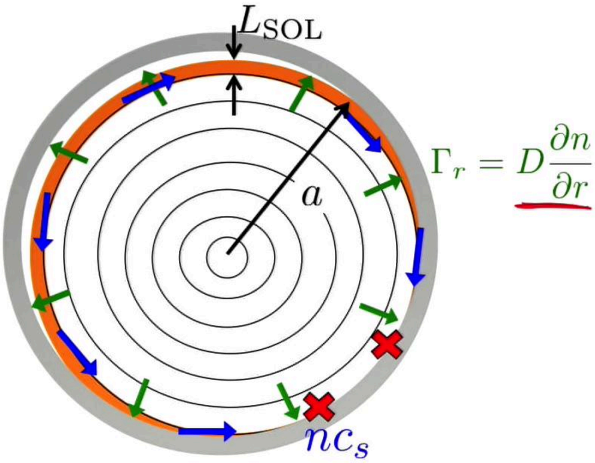
9m 23s

The scrape off layer - SOL

Outer layer of plasma in direct contact with the material wall

SOL thickness results from balance between cross-field and parallel dynamics

Ex. of estimate for limiter case



$$\Gamma_r \overbrace{2\pi a \; 2\pi R_0}^{\text{surface}} = 2\, n\, c_s \overbrace{L_{SOL}\, 2\pi R_0}^{\text{surface}}$$

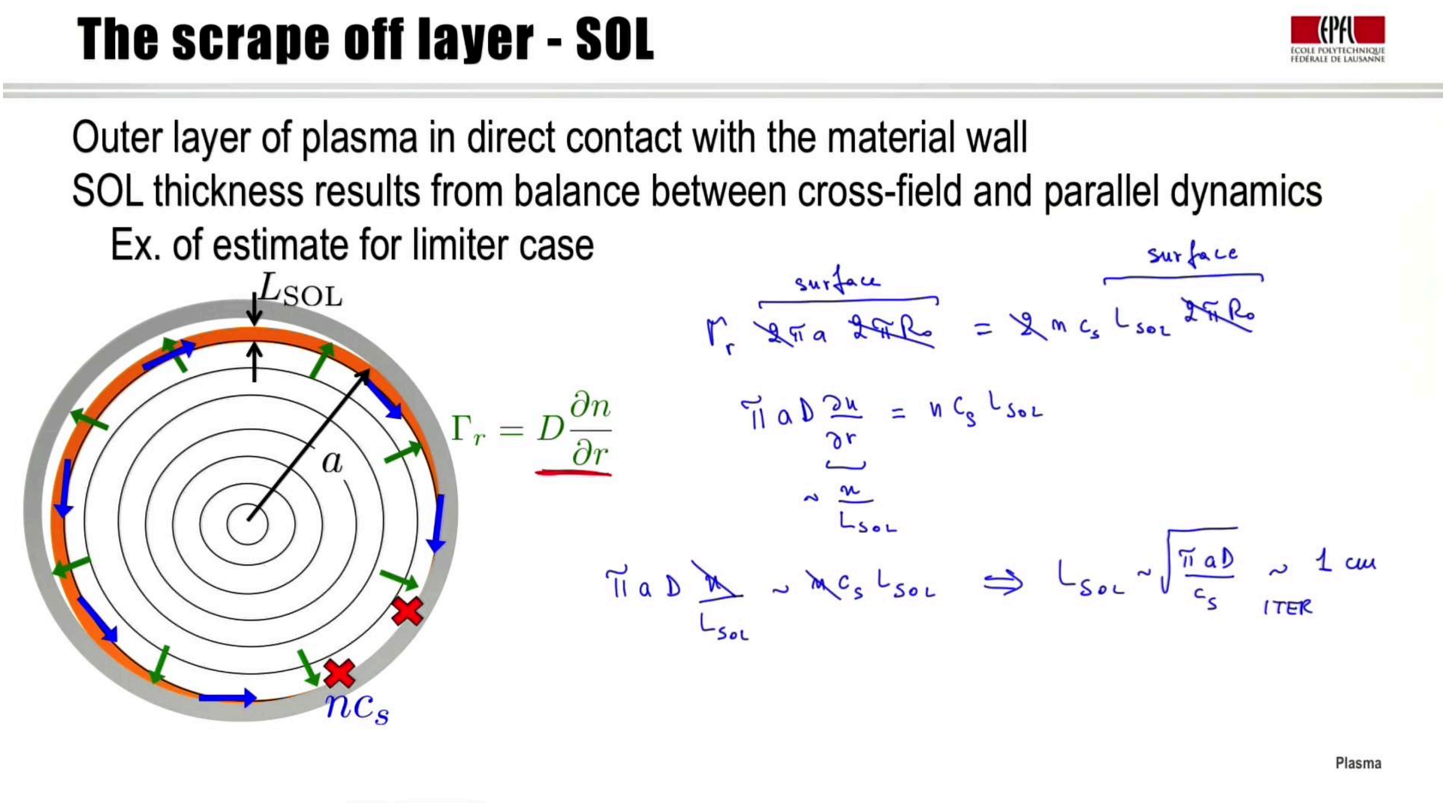
Plasma

So let me do the simple calculations. So I consider the flux, $\mathbf{\Gamma}_r$, times the surface over which it takes place, which is $2\pi a\ 2\pi R_0$ That has to be equal to 2 times... -- and the factor of 2 comes from the fact that we're coming from both sides -- 2 n cs Lsol $2\pi R_0$ because (Lsol $2\pi R_0$) is my surface for the calculation of this parallel flux. I'd like to make a couple of comments here. First, I remind you that cs is the ion sound speed. As you have seen in a previous lecture, this is the characteristic speed with which the ions move to a material surface that faces the plasma. They do that under the influence of a nonlinear electrostatic potential structure, which we call the *sheath*, that the plasma itself develops to maintain quasi-neutrality. Second, I draw your attention to the fact that I have significantly simplified this formula. I say that the plasma is outflowing at the velocity of the order of cs to the vessel wall. In reality, the plasma is flowing at cs in a direction along the magnetic field. So in the calculation, in principle, while we need to introduce the angle of a magnetic field that goes back to the wall, say α, and write that the outflow velocity is cs sin α.

Notes

Summary

10m 46s



However, in this estimate of the order of magnitude of the SOL width, we neglect this, taking a value of the sin α order 1. So I simplify the factors that are identical on the left and on the right, and I write πaD, times the gradient of the density in the radial direction is equal to (n cs) times the thickness of the scrape off layer. But the gradient in the radial direction can be estimated simply just like the density over the scale length over the density variation: Lsol So the density is assumed vary over a scale length that corresponds to the thickness of the scrape off layer itself. So we can write... πa D (n / Lsol) and that's approximately equal to n cs Lsol So the density goes away, and we can estimate the thickness of the scrape off layer. which is the square root of πa D/cs. We can take typical numbers, for example, for ITER, with an empirical value for D, the particle diffusion coefficient, typically of the order of 1 m² per second, and come up with something of the order of a centimeter. So that's very small. That means that we have a very thin layer around the plasma that is in direct contact with the material wall, and that thin layer will carry a lot of power to a very small surface.

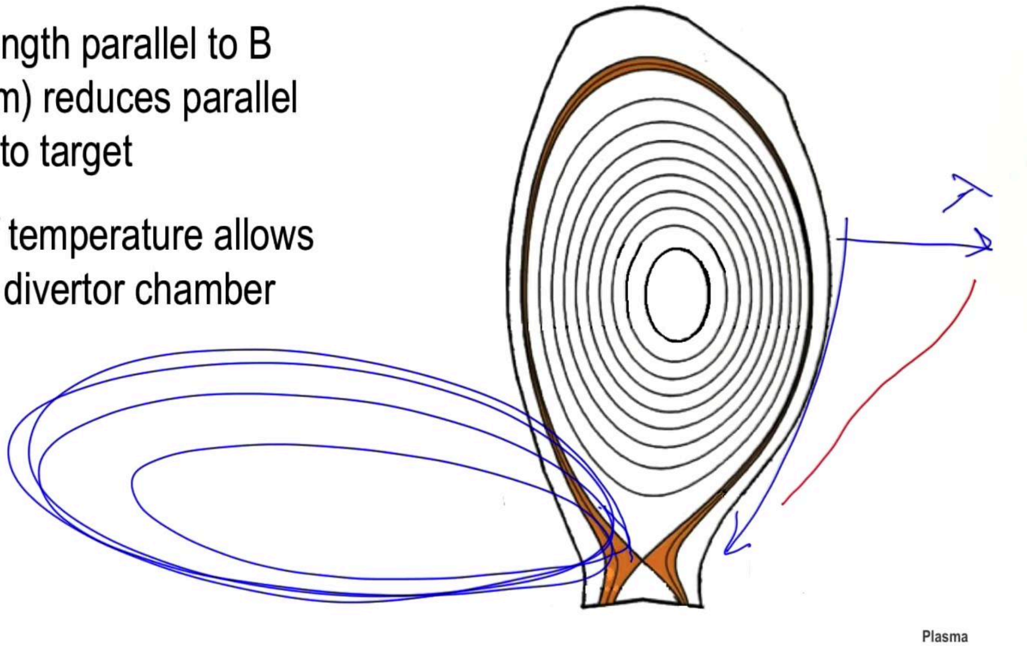
Notes

Summary

12m 14s

Advantages of divertor concept

- Long connection length parallel to B (e.g. in ITER ~150m) reduces parallel power flux arriving to target
- Parallel gradient of temperature allows low temperature in divertor chamber (~5eV)



Let's look specifically at the advantages of divertor concept, which is the concept that we are now following and using for all present and future devices we have in mind. First of all, there's a very long connection length parallel to B, and that is the length, the effective length, of the field lines, that the particles see as they approach the X-point. So if we represent the X-point, as we have done before in a 2-D configuration, that's poloidal cut, I need to try and draw the 3-D field lines that correspond to that. The field lines will start to go around, and around, and around, for a long distance, because of the toroidal field, before they can actually reach the target. So for ITER, for example, that length is about 150 meters. The fact that it is so long gives us, in a sense, a lot of time to reduce the power flux that arrives to the target. That also enables me to have gradient of temperature in the parallel direction. So if I go down this way, say, and I plot, say, the temperature in this axis, the temperature will not be uniform, but it actually will go down. So that means, as I approach the target place, I can have a colder and colder plasma.

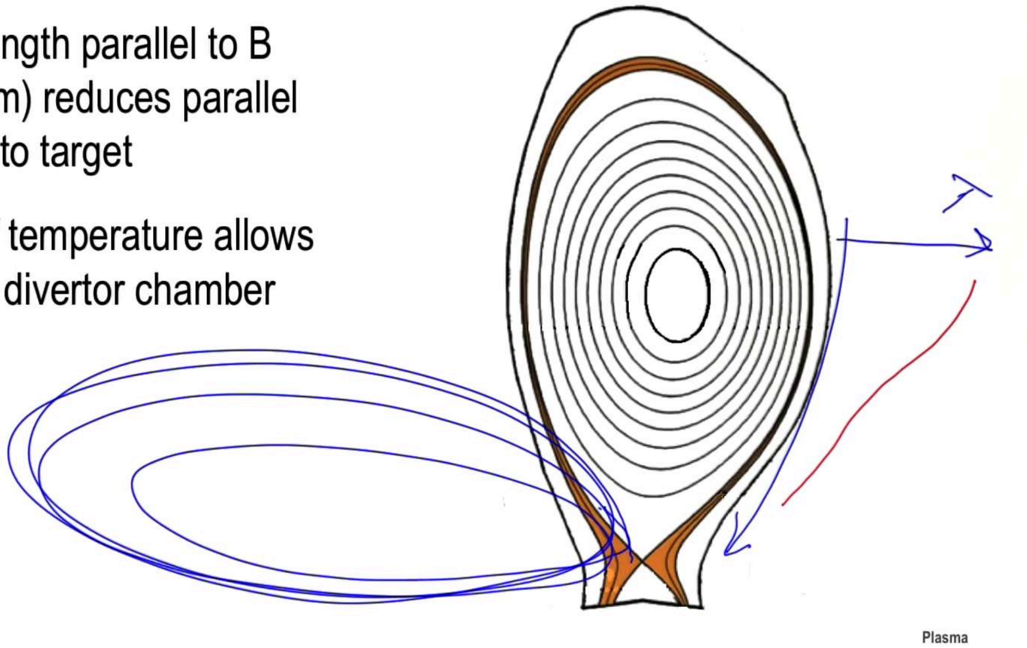
Notes

Summary

13m 54s

Advantages of divertor concept

- Long connection length parallel to B (e.g. in ITER ~150m) reduces parallel power flux arriving to target
- Parallel gradient of temperature allows low temperature in divertor chamber (~5eV)



And I will be able, in a well-designed configuration, to have pretty low temperature in the divertor chamber, of the order of only a few eVs, although I have thermonuclear temperatures in the core of the plasma, and much higher temperatures, of course, upstream of that.

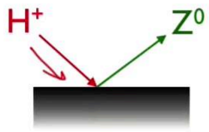
Notes

Summary

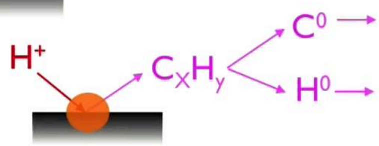
15m 22s

Advantages of divertor concept

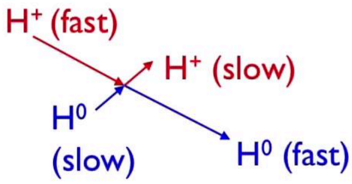
- Reduce erosion and impurity production by
  - Physical sputtering by ions



  - Chemical sputtering by ions



  - Neutral impact (charge exchange collisions)



- Reduce impurity transport back to main chamber

Plasma

Following up on the advantages of divertor concept, because we're able to reduce the temperature, we're also able to reduce the erosion and impurity production by a number of effects, including the physical sputtering by the ions -- so the fact that the ions impinge on the surface with their energy, and extract atoms from the surface. We can also reduce the chemical sputtering by the ions -- the ions that impinge on the surface can create a chemical composite. It can also reduce the neutral impact -- that is, the number of charge exchange collisions -- with this reduced temperature. The configuration will enable us, also, to reduce the transfer of impurities back to the main chamber. That is a key element in the need for plasma to be maintained pure.

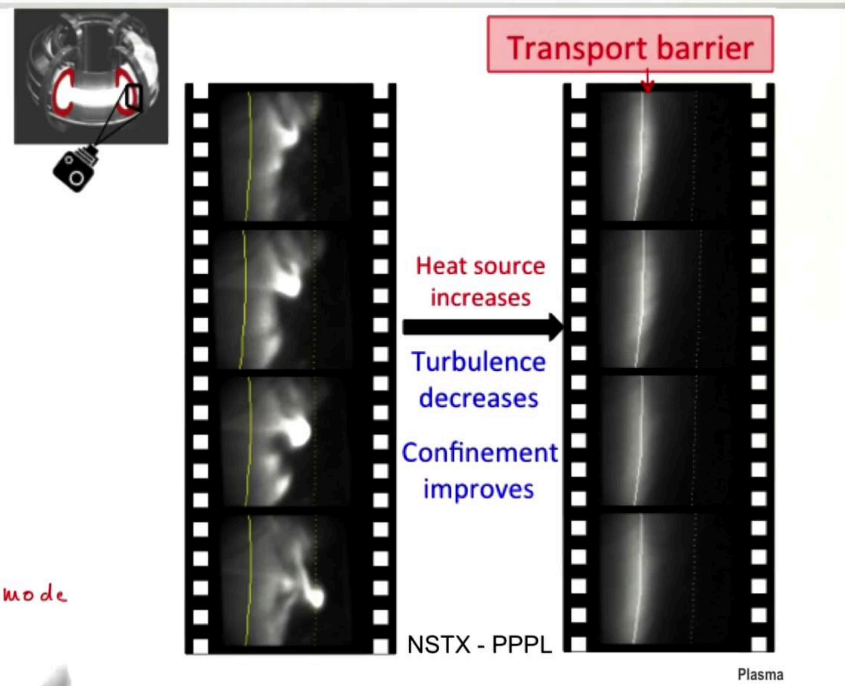
Notes

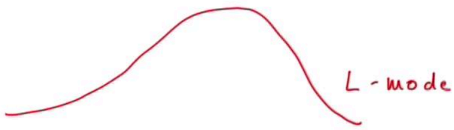
Summary

15m 39s

Advantages of divertor concept

- Easier access to high confinement regimes





It has also been noticed experimentally, that in the presence of an X-point -- that is, in the presence of a divertor configuration -- it's easier to access what we call 'high confinement regimes.' These are regimes in which the edge of the plasma, we notice that the turbulence decreases, and therefore the confinement improves. The decrease of the turbulence is represented here, in this sequence of images taken at the very edge of the plasma, in which you see that, in some circumstances, there are bubbles -- sometimes we will refer to them as *blobs* -- that take particles and energies out quite effectively. But in the presence of an X-point, we can create what's called a *transport barrier* -- that is, a region of very quiescent plasma -- that, in fact, makes it very difficult to particles and energy to go across it. The profiles that result are significantly improved, in the sense of fusion. Let me just draw them in a completely qualitative way. This is a situation in which we have what we call the *L-mode* so still a turbulent edge.

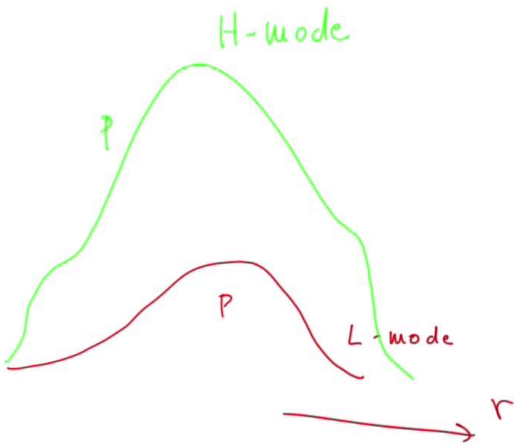
Notes

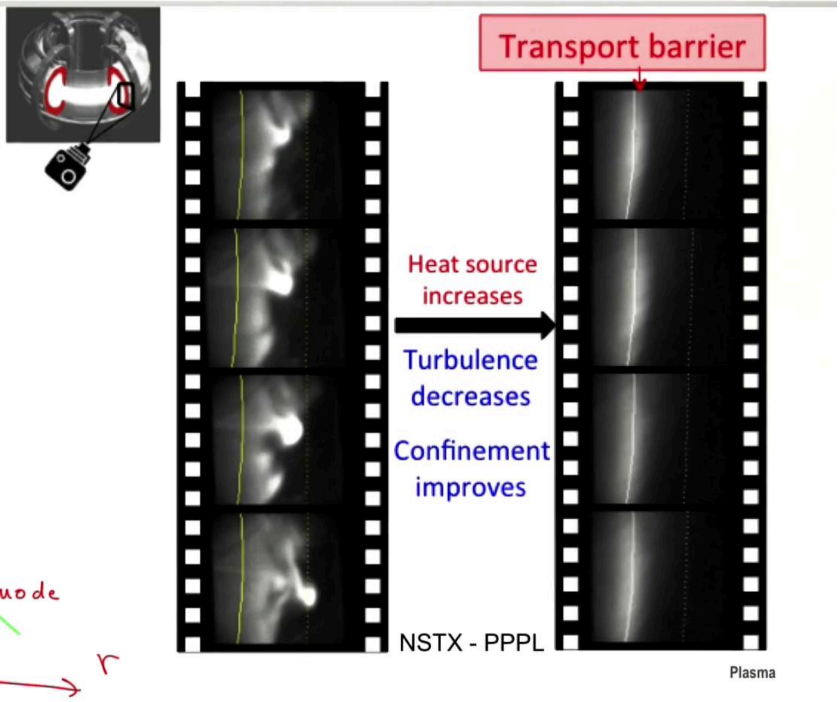
Summary

16m 26s

Advantages of divertor concept

- Easier access to high confinement regimes





And if I have a situation in which I manage to create a divertor configuration *and* to create this transport barrier, which again is possible, and actually easy, relatively easy, in the presence of a divertor configuration, then I jump up in my profile, I have a very steep gradient, and I have a much higher value of pressure. This is a profile of a pressure, and that's called *H-mode*, for *high confinement mode*, as opposed to L-mode, for low confinement mode, that we have seen before. Still, in both cases, we had the pressure represented as a function of the radial direction. So, divertor configuration makes it easier to access the high confinement regime, which makes us gain, significantly, performance in the plasma, in the sense of fusion power that can be produced.

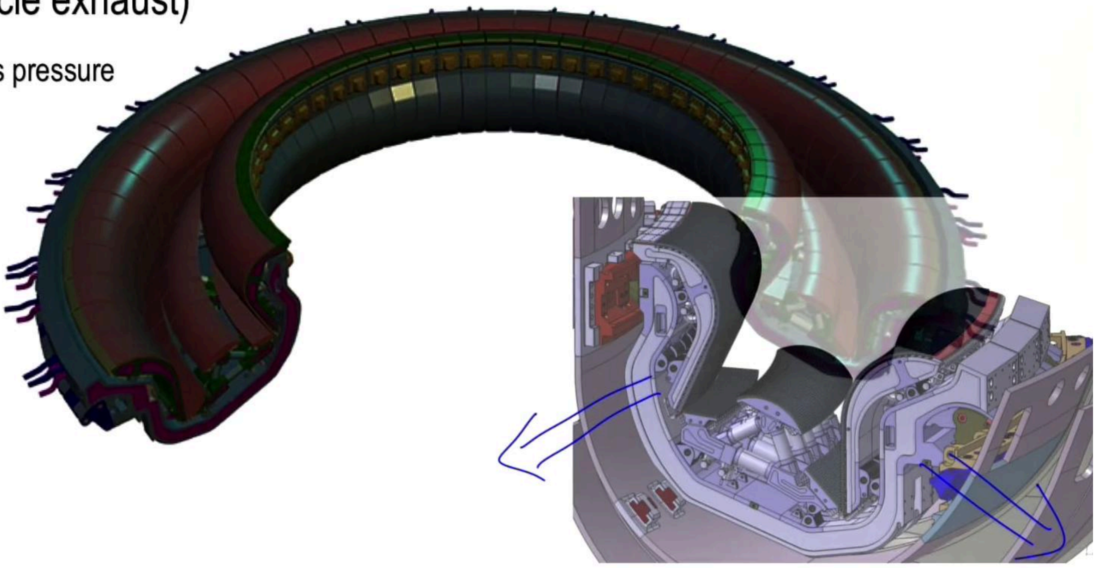
Notes

Summary

17m 33s

Advantages of divertor concept

- Pumping (particle exhaust)
  - Higher neutral gas pressure
  - Cryopumps



The divertor creates a region of relatively large pressure, and that makes it easy, or easier, for the pumps to function in that region, and therefore to exhaust the particles that need to be taken out of the plasma volume. I can also install cryopumps around the diverter chamber and extract these particles effectively, on the other side of the divertor target plates. This is the design of the ITER divertor, and the ITER, specifically, set up for cryopumping.

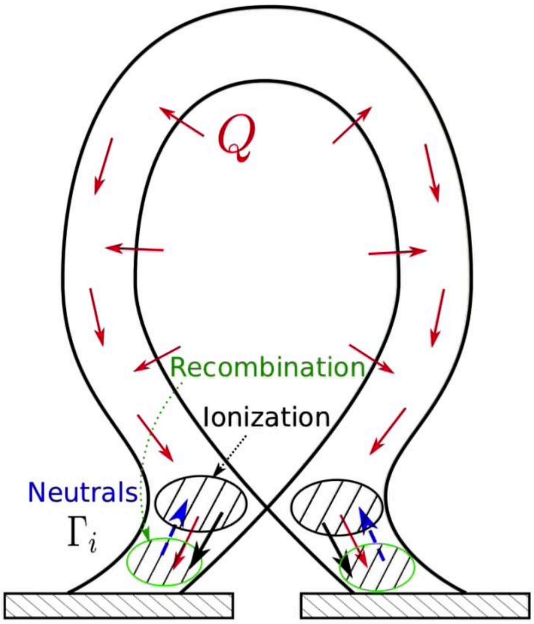
Notes

Summary

18m 30s

Advantages of divertor – plasma detachment

- At ~5eV: $\sigma_{ionisation} < \sigma_{charge\ exchange}$
- Energy is transferred from ions to neutrals, which spread power deposition (neutral cushion)
- T is further reduced and e-i volumetric recombination occurs close to the targets
- Low energy flux to the target as most of power is dissipated in radiation



Plasma

So, divertor allows plasma to be cold close to the walls, in that specific area. And if the plasma is cold -- say it's of the order of a few eVs, in terms of temperature; say 5eV -- that means that the ionization cross-section becomes smaller than the charge exchange cross-section, which makes it easy for energy to be transferred from ions to neutrals. And the neutrals help a lot, because they spread the power deposition, forming what we call a neutral cushion. That reduces further the temperature of the plasma in the divertor chamber, and the recombination between electron and ions can happen over an entire volume that is close to the target, but not on the surface, necessarily. That reduces the flux of energy to the target even further, because most of the power will, therefore, be dissipated by radiation before the plasma reaches the target. So this plasma detachment is a situation in which we would like, really, to be for ITER and for the fusion reactors, because that's where we limit the power that directly goes to the target as we radiate it before the plasma arrives to the target.

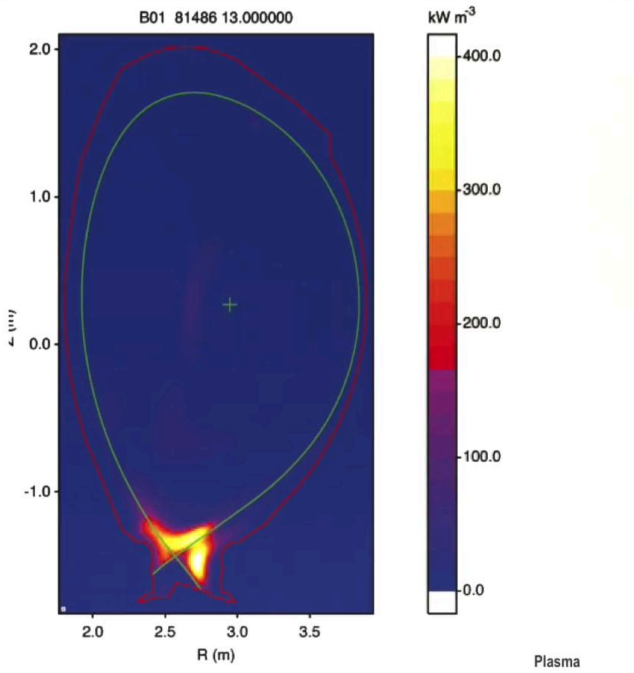
Notes

Summary

19m 12s

Advantages of divertor – plasma detachment

- At ~5eV: $\sigma_{\text{ionisation}} < \sigma_{\text{charge exchange}}$
- Energy is transferred from ions to neutrals, which spread power deposition (neutral cushion)
- T is further reduced and e-i volumetric recombination occurs close to the targets
- Low energy flux to the target as most of power is dissipated in radiation



And that radiation is visualized in this image, and that is really concentrated around the X-point, in our divertor configuration.

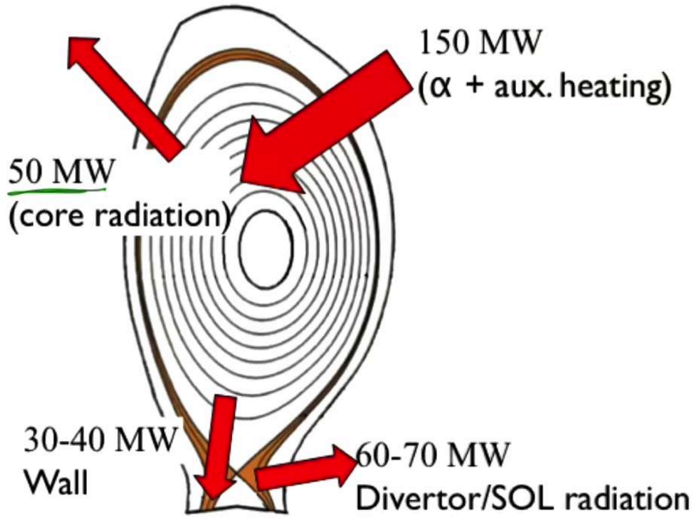
Notes

Summary

20m 27s

Advantages of divertor concept

- Heat exhaust by volumetric loss processes (radiation, charge exchange)
  - Estimate of various contributions in ITER



With 3.5 m^2 wetted area, P_{wall} < 10 MW m^{-2}

Plasma

And sort of to give numbers to this power balance, here I take the case of ITER. As we said, we like to exhaust, effectively, heating power and particles. And in this case, there's a lot of the heat exhaust that takes place via radiation. So say I inject 150 MW, both from α-particle heating and from auxiliary heating from outside. Of course, we're not at ignition, so we still have some external heating of the plasma. And in a typical ITER case that we foresee, we will radiate from the core about 50 MW of those 150, and in a divertor, if we manage to detach it, will radiate, say, about 70 MW of the remaining 100. And only about 30 MW will go directly to the wall. So if we have about three, four m² of plasma wall area that can be, as we call it, *wetted* by that power, that means that the power to the wall will be giving a flux that will be no more than 10 MW per m². So we are, really, in a region that can be managed. If we don't radiate this amount, if we don't radiate this 60 or 70 MW in a divertor on the scrape off layer, well, we will exceed this amount, and therefore we will damage the divertor target plates very, very quickly.

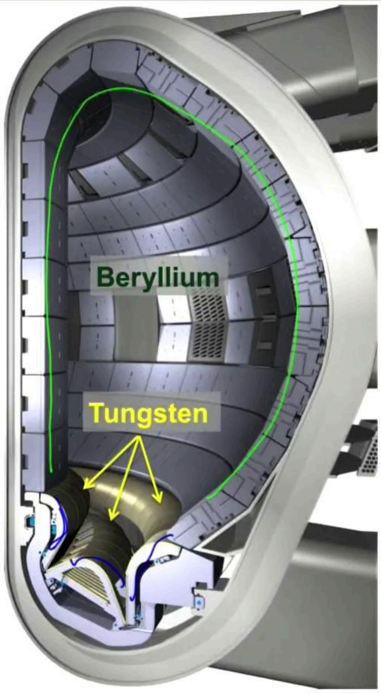
Notes

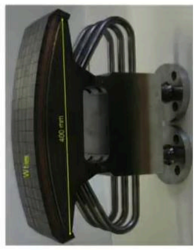
Summary

20m 37s

The choice of first wall materials for ITER

- ITER divertor will entirely be made of tungsten
  - Low tritium retention, high threshold for sputtering
- Walls will be made of beryllium
  - Low-Z, low tritium retention, good oxygen getter
- Materials chosen also to minimise deterioration of thermo-mechanical properties under neutron irradiation





Plasma

So let me briefly discuss the choice of the materials for the first wall for ITER. This choice now is definitive. The ITER divertor will be entirely made- be made of tungsten. Tungsten is high-Z material, but it has a high threshold for sputtering, so only very, very, very small amounts will be actually injected into the plasma core. And it has a big advantage of having very low tritium retention. So that's the material for the divertor. The rest of the walls -- all the walls outside the divertor chamber -- will be made of beryllium. Beryllium is a metal with relatively low Z, so even if some atoms are injected into the plasma core, they'll radiate much less than the tungsten, and dilute the plasma much less, as well. It also has a relatively low tritium retention, and in addition, it's a very good oxygen getter to pump the remaining oxygen from the chamber. This combination of materials is chosen also to minimize the deterioration of the thermo-mechanical properties under the radiation of the fusion-generated neutrons.

Notes

Summary

22m 10s

Further challenges for divertors - transients

Edge Localised Modes, ELMs

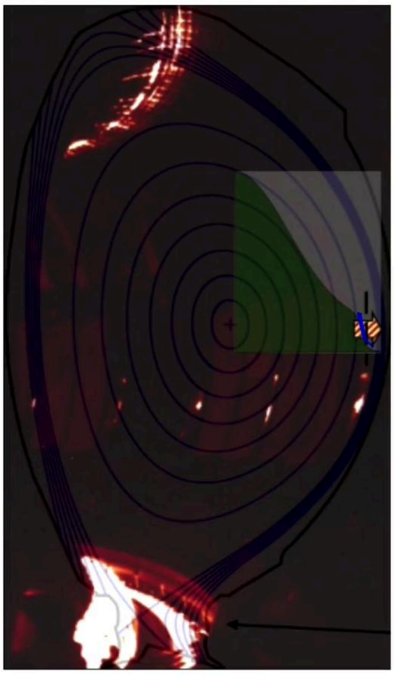
- Large edge gradients give rise to instabilities that generate outwards bursts of energy and particles → large thermal loads

- Ex. ELMs in ITER

  ~ 15MJ, t ~ 0.2ms, over ~ 6m^2 → ~ 10GW/m^2

  surface temperature ~ 6000°C, penetration ~ 0.15mm

  → melting of metals



Plasma

So all of this seems to work well. Nevertheless, there are challenges that we're still facing for the divertor, in particular, in view of ITER and of the other following burning plasma experiments and demonstration reactors. And perhaps the most important element to discuss is that of *transients*. These are events that are, so-called, *off-normal*, so that are coming on top of the steady-state load to the wall. First, are what we call the *Edge Localized Modes*, or *ELM*s. These are instabilities that are generated by the very large edge gradients which we obtained in the high-confinement regime. This is a picture of a pressure profile. You can see that at the edge, you have a very steep gradient. Very steep gradients is equivalent to very large thermodynamic potential for driving instabilities, and this instability will happen, and happen in a very short timescale, generating violent bursts of energy and particles to the wall -- that is, large thermal loads. And if we extrapolate what we know of ELMs today to ITER, we expect something like 15MJ of energy expelled in a single event -- in an event that lasts a fraction of a millisecond, and that is going to wet an area of a few m^2.

Notes

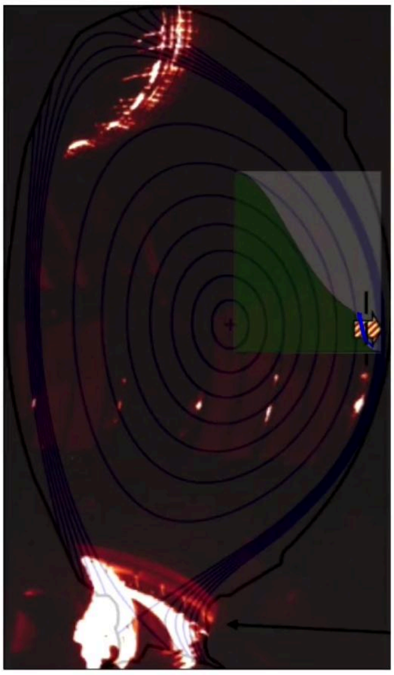
Summary

23m 32s

Further challenges for divertors - transients

Edge Localised Modes, ELMs

- Large edge gradients give rise to instabilities that generate outwards bursts of energy and particles → large thermal loads
- Ex. ELMs in ITER
  ~ 15MJ, t ~ 0.2ms, over ~ 6m² → ~ 10GW/m²
  surface temperature ~ 6000°C, penetration ~ 0.15mm
  → melting of metals



So that is an enormous amount of power per m². It's about 10GW per m². And even if you evaluate materials like tungsten, the surface temperature will go up very significantly, to about 6000°C, over a layer that's not infinitely thin, but is about 0.1, 0.2mm. That means the metal will melt. So this is something we, of course, must avoid.

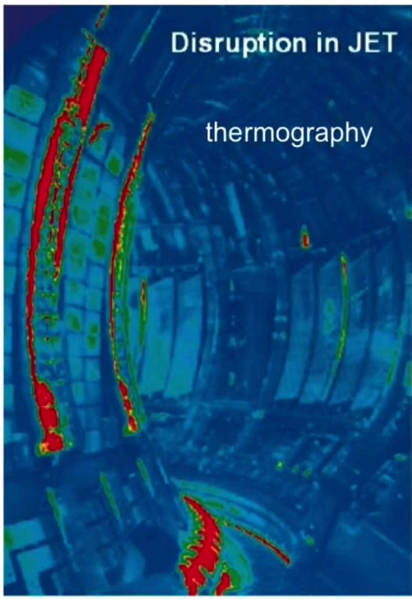
Notes

Summary

24m 54s

Further challenges for divertors - transients

Disruptions
- Sudden loss of plasma leading to large deposition of energy on walls
  - Ex. ITER full energy disruptions: peak energy densities on divertor of 5 - 20 MJ/m^2 over ~ 1.5 - 3ms
  - W divertor lifetime exceeded in ~300 disruptions



Plasma

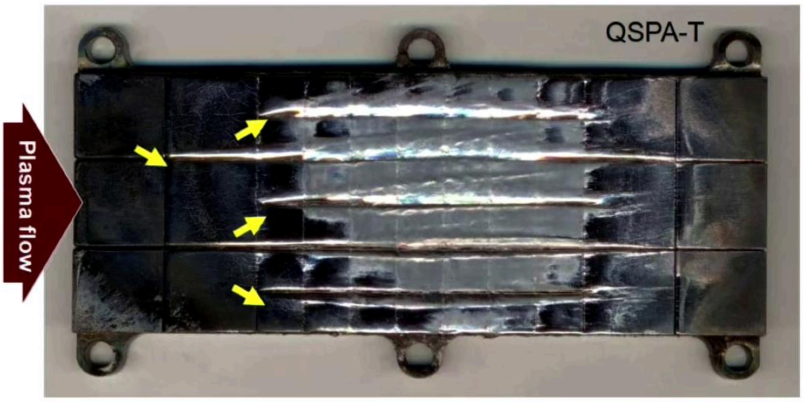
Second example of these transient events is what we call a *disruption*. A disruption is a sudden loss of plasma control that leads to the plasma being terminated over a very short timescale, and sort of impinging on the wall. And that generates a very large deposition of energy on the walls, of course. The image here is a thermographic image of the JET vessel during a disruption, and you can see that there are specific areas inside the vessel that are made very hot by the energy deposited by the disruption. And you can also see an image here that visualizes the consequences that the disruption can have. This is the melting of a very thick piece of Inconel, which is a very resistant, special steel. In ITER, we anticipate that a disruption can lead to peak energy densities in a divertor region of 5-20MJ per m^2, over short timescales: again, 1-3 milliseconds. And if we have these, the lifetime of the divertor will be exceeded in a few hundreds of these disruptions. So we must avoid having these events.

Notes

Summary

25m 26s

Further challenges for divertors - transients



A. Zhitlukhin, et al., J. Nucl. Mater. 363–365 (2007) 301

Ex. W exposed to 100 plasma pulses simulating ELMs of 1.5 MJ/m^2

We don't have materials that withstand for sufficiently long time these thermal loads, therefore we need to act on plasma to avoid or mitigate these violent transient events

Plasma

We simulate in plasma devices that are simpler than tokamaks what happens to materials that are facing these kinds of thermal loads from plasma. This is an example from the QSPA facility in which the tungsten was exposed to about a hundred plasma pulses that are simulating ELMs of 1.5 MJ per m^2. You can see, of course, very clearly the damage that this tungsten feels- this tungsten has to withstand. In fact, we don't have materials that can last for a sufficiently long time under these thermal loads. So that means that, from the material side, we need to evolve. But we also need to evolve in our capability of avoiding these transients, or mitigating the effects of them.

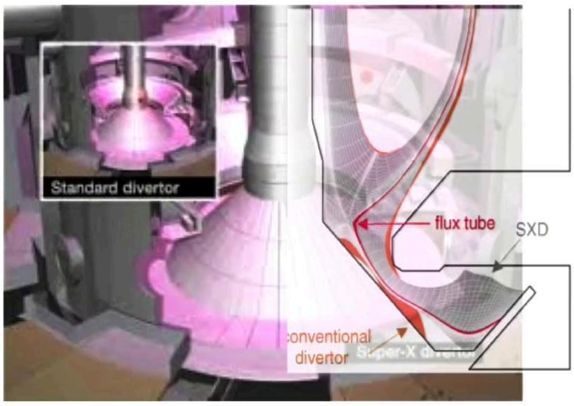
Notes

Summary

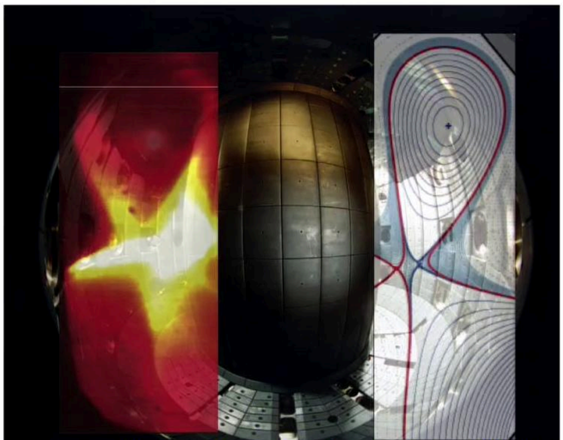
26m 44s

Innovative divertor configurations

- New divertor configurations are explored for DEMO and reactors
  - Limit material erosion, increase radiated power with detached plasma, keep core plasma pure
- Ex. of alternative concepts: liquid metal walls, super-X, snowflake, …



Super-X divertor design for MAST-upgrade



Snowflake divertor tested on TCV

Plasma

I'd like to conclude the discussion by just highlighting briefly, that there are also new ideas, as far as the configuration of the divertor, ideas that are explored not for ITER, which is already designed, but for the steps beyond ITER, in particular, for DEMO, which is a step that will demonstrate the economical feasibility of fusion, not just the scientific and technological feasibility. The requirements for the divertor of ITER, and for the plasma wall interaction of ITER in general, are, of course, valid, and even stronger for the steps following ITER. Again, we have to limit the material erosion; we have to increase the relative power as much as we can. That means we have to have a detached plasma, and we have to maintain the plasma core very pure. In order to do that in more and more stringent situations, such as that of a DEMO reactor, we are exploring, essentially, three possibilities, or three kinds of possibilities. We may have a liquid metal wall. So there would be a thin layer of wall that will circulate around the plasma, -- for example, lithium -- that will take away particles and heat. Or, we can have special magnetic configurations that are issued from the conventional divertor, the configuration we have explored together in this lecture.

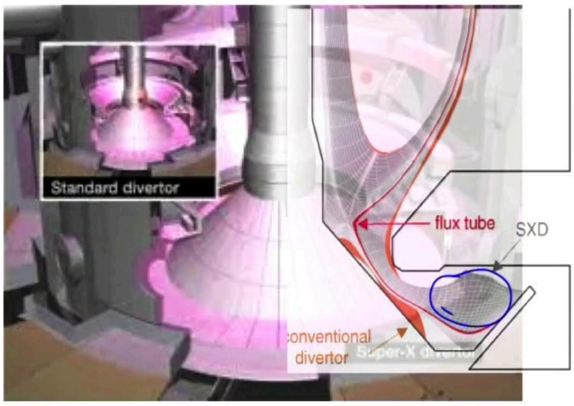
Notes

Summary

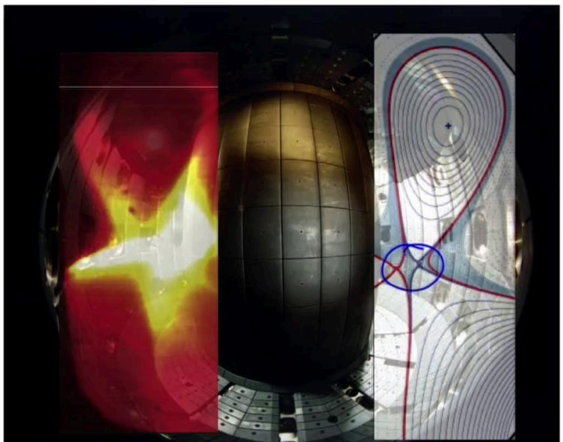
27m 39s

Innovative divertor configurations

- New divertor configurations are explored for DEMO and reactors
  - Limit material erosion, increase radiated power with detached plasma, keep core plasma pure
- Ex. of alternative concepts: liquid metal walls, super-X, snowflake, …



Super-X divertor design for MAST-upgrade



Snowflake divertor tested on TCV

Plasma

One is the so-called *Super-X*. A picture of it is here. This is the design for the MAST-upgrade facility, in Culham, UK. Super-X consists of an expansion of the divertor target chamber, in a sense. So it's a clever way of designing the detail of the divertor chamber so that, in fact, one of the legs of the divertor is effectively not only longer, but also of much higher volume. And much higher volume means that you can radiate more easily the power before the plasma reaches the target. Another option is so-called *snowflake*, which we have pioneered in Lausanne on the tokamak TCV, where it was first demonstrated. The snowflake is, perhaps, even a simpler idea. So the divertor, instead of having two legs to carry the power away from the core, it has four legs, which means that you have more possibilities for plasma exhaust. You also have a longer connection length, and the region around the X-point, which is more complex, and it may radiate more, and may also lead to a different stability property for the plasma, with respect in particular to the edge, violent instabilities we have seen before. So these are concepts that are being explored.

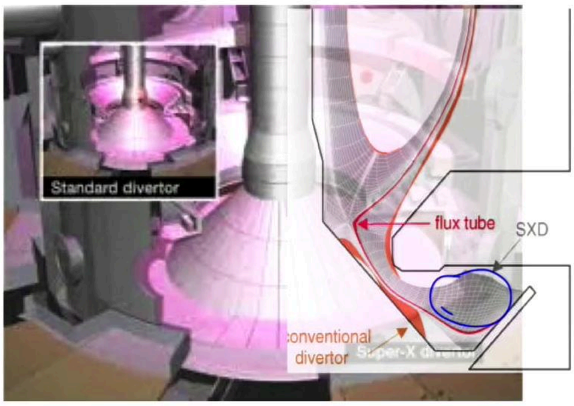
Notes

Summary

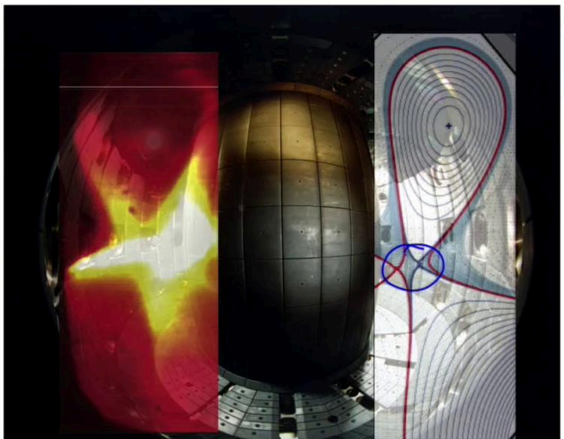
29m 03s

Innovative divertor configurations

- New divertor configurations are explored for DEMO and reactors
  - Limit material erosion, increase radiated power with detached plasma, keep core plasma pure
- Ex. of alternative concepts: liquid metal walls, super-X, snowflake, …



Super-X divertor design for MAST-upgrade



Snowflake divertor tested on TCV

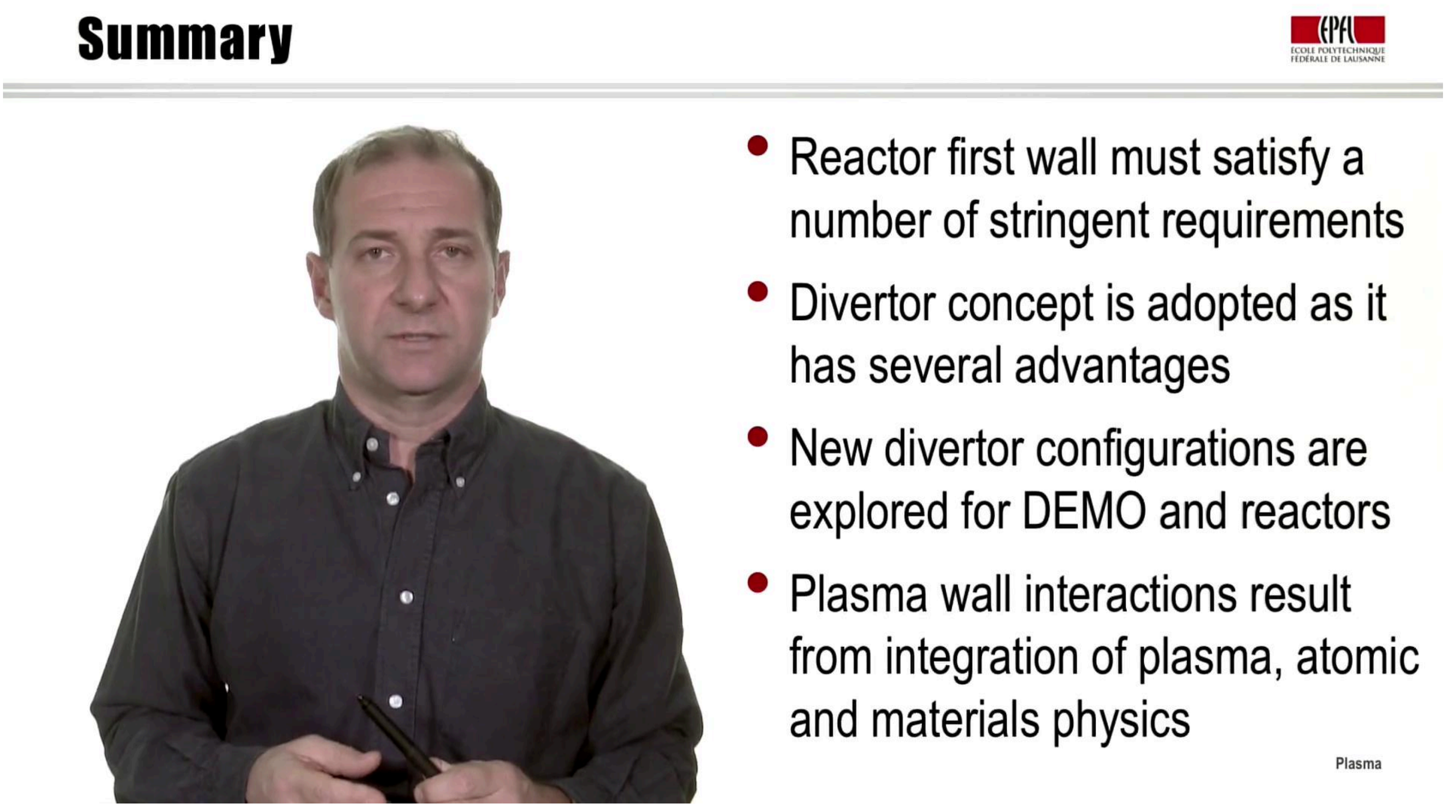
Plasma

It's not clear that they will be advantageous over what we know today, which I can refer to as a conventional divertor, but we need to explore these, because the challenges we have seen together are really, very, very demanding for the future devices.

Notes

Summary

30m 26s



To summarize, we have seen that the reactor first wall must satisfy a number of very stringent requirements. We have seen that the divertor concept is the one adopted commonly, because it has several advantages over the limited concept. New divertor configurations and innovative schemes for constructing a divertor are being explored, in particular, for the steps beyond ITER, that is DEMO and the reactors. The plasma wall interaction is the result of a very intricate combination of plasma physics, atomic physics, and materials physics. In the next module we will look, in fact, at some aspects of this materials physics, and the issues associated with the functional and structural materials for reactors.

Notes

Summary

30m 43s