Materials for the plasma-facing components of fusion reactors

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Abstract

According to current knowledge and understanding, nuclear fusion can be developed to a sustainable energy technology. Fuel is abundant and key points as fusion power production and alpha particle heating have already been demonstrated. The next-step device ITER (International Thermonuclear Experimental Reactor) is designed to demonstrate net power production and to address most of the technological issues on the way to a power reactor. There is, however, a series of materials problems related to the plasma-facing components and to the structural materials which cannot be fully addressed by ITER. These developments are covered by long-term development of radiation resistant low activation materials, heat sink materials, plasma-facing protection materials as well as functional tritium barrier materials.
A brief survey of the current status of materials development for plasma-facing applications is given in the present paper. To provide materials which can sustain the severe loading conditions in a fusion environment is a key issue in developing fusion as an economic energy technology without long-lived radioactive waste.

Keywords: nuclear fusion, low activation materials, plasma-facing materials, high heat flux materials, ExtreMat Integrated Project

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1. Introduction

Controlled nuclear fusion of hydrogen isotopes has been a research topic in the world since around 1950. Among many attractive features of fusion as an energy source are the abundance of fuel for the reaction, the high specific energy release per reaction, and - in contrast to nuclear fission techniques - the inherent safety of the process due to its thermonuclear nature. To achieve controlled nuclear fusion on earth various schemes have been proposed, among which the magnetic confinement in the so-called tokamak arrangement has been the most successful one up to now.
The principal problem to overcome in the fusion of two atomic nuclei is the repulsive electrostatic force, which acts in between them. The highest probability to achieve the fusion process, i. e. the highest reaction rate can be met when the two hydrogen isotopes deuterium (D) and tritium (T) are employed at a temperature in the range of 10...20 keV, one to two hundred million K.[1]

$$D + T \longrightarrow {}^{4}He\ (3.5\ MeV) + n\ (14.1\ MeV)$$

The released energy of 17.6 MeV is distributed between a neutron and a charged ^{4}He nucleus, an alpha particle. In a magnetic confinement scheme, the neutron is not confined and escapes from the plasma. The charged ^{4}He particle, however, is confined by the magnetic field. Its energy must be thermalised to sustain the high plasma temperature.
The availability of these two fuel components on earth is very different: Deuterium is abundant in nature, its concentration in natural hydrogen is about 0.015%. Therefore it is practically inexhaustibly available from sea water. Tritium, however, does not occur naturally in sufficient quantities because it is radioactive with a half life of about 12 years. The concept for a fusion power plant is that the fusion reactor produces its own supplies of tritium in a self-sustained fashion. The neutrons produced by the fusion reaction are employed for producing tritium from the two naturally occurring lithium isotopes:

$$^{6}Li + n \longrightarrow {}^{4}He + T + 4.8\ MeV$$
$$^{7}Li + n \longrightarrow {}^{4}He + T + n' - 2.5\ MeV$$

This is achieved by surrounding the inside of the plasma vessel with a so-called blanket, a structure that absorbs the neutrons and contains the tritium breeding fuel. It is this blanket which also absorbs the energy released with the neutron. Therefore it must contain coolant channels. The coolant carries the produced tritium to an extraction stage and the tritium is finally injected into the plasma as new fuel. In this way radioactive tritium is handled only inside a closed system of a fusion reactor facility. The first wall covers the blanket surface towards the plasma and has the task to absorb the heat and particle fluxes from the plasma. The divertor is the component which is in most intense interaction with the plasma. At the divertor surface the impinging ions are neutralized which allows to exhaust the neutral atoms from the plasma by pumping. As a result, the plasma particle and heat fluxes are highest on the divertor surface, Table 1.

2. The status of fusion research

Magnetic confinement of fusion plasmas has been widely investigated in various magnetic field configurations. Nowadays two toroidally closed configurations are mainly pursued: The tokamak and the stellarator. Since a purely toroidal magnetic field configuration does not yield a stable plasma confinement, an additional twist of the field lines is necessary. In the tokamak configuration this is achieved by using an external toroidal field and additionally driving a current through the plasma in toroidal direction. The stellarator concept, on the other hand, does not employ current drive inside the plasma, but instead uses external magnetic fields of complex shapes. Tokamaks nowadays produce the better results concerning the so-called fusion product of plasma density, temperature, and energy confinement time. Stellarators have no necessity for current drive and therefore have a general advantage concerning stability and continuous operation. Figure 1 shows the progress of the fusion product, that has been made over the last decades. Today's large devices are close to the breakeven condition Q=1, which means fusion power equals input heating power. Reactor conditions are reached in plasmas that are dominantly heated by α-particles from fusion reactions. Since the fusion product also scales with the volume of the plasma, reactor conditions can only be met in a larger next step device. ITER will reach this regime.

Present day fusion research facilities generally do not operate with deuterium/tritium mixed plasmas. The breakeven condition of a D-T plasma, however, has already been reached. Only a few experiments on realistic mixtures producing fusion power have been done. In its 1997 deuterium/tritium experiment the Joint European Torus tokamak JET has achieved the substantial amount of 16 MW of fusion power corresponding to Q=0.62 (see Figure 2).[2] Significant alpha particle heating of the confined plasma has been demonstrated in the Princeton Tokamak Fusion Test Reactor TFTR[3] as well as in JET[4].

3. The ITER project, studies for DEMO and fusion reactors

Recognizing the importance of the next step to a rector-scale fusion device, in 1987 the European Union, Japan, the Russian Federation, and the United States initiated the so-called Conceptual Design Activities for ITER, the International Thermonuclear Experimental Reactor. It was agreed that the overall objective of ITER would be "to demonstrate the scientific and technical feasibility of fusion energy for peaceful purposes".[5]

The scientific and technical performance specifications were reformulated in 1998. Now the ITER design envisages the demonstration of extended burn of deuterium-tritium plasmas with a fusion power of 500 MW and a ratio of fusion power to input heating power Q of at least ten.[6]

Another target is the demonstration of technologies essential to a reactor in an integrated fusion technology system. ITER will enable the science of self-heated plasmas in relevant power plant regimes to be studied. Prototypes of key components have already been manufactured and tested: Central Solenoid for inductive current drive, superconducting toroidal field coils, complete blanket module mock-ups as well as full scale modules of the vacuum vessel.[6]

In 2005 the decision was made to build ITER in Cadarache (France) as a joint project of the European Community, Japan, Russia, US, Rep. Korea, China and India.

In conjunction with further long-term materials development, modelling and neutron irradiation experiments of these materials, ITER is supposed to provide a sufficient scientific and technological basis for a "first of its kind" fusion reactor.[7] A study for such a Demonstration Reactor DEMO which will lead to a conceptual design is under way in the EU. The Power Plant Conceptual Study has evaluated different options for a commercial fusion power plant based on varying degrees of technological extrapolation from the present state.[8] These variants include a water-cooled reactor design, and several Helium cooled reactor designs. The availability of suitable materials and the achievable progress in plasma engineering dictates which of these variants will become a realistic base for the design of a fusion reactor.

4. Material issues for plasma-facing components

The plasma-facing components of a fusion reactor consist mainly of the first wall and the divertor. A cutaway view of ITER showing the first wall and the divertor is given in fig. 3. The first wall, a structure with coolant channels is covered with a plasma-facing material that provides protection against the particle load from the plasma as well as protection against damage from transient heat loads. The stationary heat loads from the plasma to the first wall are moderate and may reach up to 1 MW/m^2 in a reactor. Erosion of the plasma-facing wall material is regarded as the dominating lifetime issue in a reactor.
The divertor is the component with most intense plasma contact. The plasma-facing material of the divertor has to be bonded to a high thermal conductivity heat sink material in order to remove very high stationary heat fluxes of up to 20 MW/m^2.
Tritium which is used as fuel in fusion reactors should not enter the structural materials or the coolant of the plasma-facing components and the blanket. This requires the development of a specific barrier technology which has to be integrated into the plasma-facing components and the blanket of fusion reactors.
Main differences in the operation conditions of the plasma-facing components in ITER and in a reactor are the number of cycles, since a reactor should be operated in steady state, the reduced transient loading conditions required for economic reactor operation and the neutron damage which is much high in a reactor compared to ITER. The loading conditions of the materials for the plasma-facing components in ITER and in a first generation (DEMO-like) fusion reactor are listed in Table 1.[9]

4.1 Structural and heat sink materials

One of the main fusion reactor material issues is the intense irradiation with neutrons. This immediately brings about two of the key problems in the development of structural materials:
- The materials must not suffer strong radiation-induced embrittlement or other degradation
- Radioactive waste must be minimized: The half-life of neutron-induced activation products should be as short as possible

In order to meet these requirements, three lines of development are being pursued:[10,11] Reduced activation ferritic-martensitic steels are currently the furthest developed technology. Vanadium alloys based on the V-Cr-Ti system have a great potential for high-temperature operation in combination with a liquid lithium breeder system. Finally, SiC-fiber-reinforced SiC composite ceramic materials (SiC-SiC) would have a great potential for very low radioactivity and for very high operating temperature, corresponding to a high degree of efficiency. This system, however, surely constitutes the most difficult technological challenge of the three groups. The aim is to arrive at materials which allow operation at reactor relevant coolant temperatures and, at the end of the life, to be recycled or disposed of under strongly relaxed conditions.

The mainstream of this work has resulted in a ferritic-martensitic steel with 9% Cr and substitution of the elements which cause the strongest activation like Mo and Nb. Data regarding the properties after neutron irradiation hint that embrittlement can be reduced when this material is being irradiated above 250°C to 300°C. Of highest importance in this respect is the irradiation induced shift of the ductile to brittle transition temperature (DBTT). Fusion specific grades like F82H and the European EUROFER grade show only a moderate shift of the ductile to brittle transition temperature DBTT.[10,11] In a recent irradiation experiment to a high dose of 32 dpa at 300°C irradiation temperature EUROFER showed an increase of the DBTT by 200 degrees from -100°C to +100°C.[12] However, this shift will be enhanced by irradiation with high energy neutrons typical for the fusion neutron energy spectrum. Under fusion conditions the high energy neutrons will cause He production from transmutation reactions on the order of 10 appm/dpa in steel. The He will interact with the displacement induced damage and increase the embrittlement at low temperatures.

Further development of steel-based materials is being carried out on oxide dispersion strengthened 9%Cr and 12/14% Cr steels to increase the upper temperature limit for operation from 550°C for EUROFER to higher values.[12]

In addition to this the properties of SiC-SiC especially in terms of thermal conductivity and resistance to neutron irradiation are being improved. The use of SiC-SiC either as structural material or as heat insulation liner material would allow He-coolant temperatures in a range up to 1000°C.[13]

In order to remove heat fluxes of up to 20 MW/m^2 from the divertor, heat sink materials with very high thermal conductivity have to be applied. In the ITER divertor CuCrZr alloy will be used.[14] The joining technology of the plasma-facing materials to CuCrZr has been developed to a standard allowing the removal of stationary surface heat loads on the order of 20 MW/m^2. Joining methods for dissimilar bonds of Be-Cu and W-Cu are high temperature brazing and hot isostatic pressing. CFC (carbon fibre reinforced carbon)-Cu bonds can be also joined by active metal casting (AMC). The heat removal capability under thermal cycling conditions of CFC-CuCrZr model components with active cooling has been subject of reviews.[15,16] Under intense neutron irradiation, CuCrZr can be used only up to 350°C, and thus it is not compatible with the high coolant temperatures required in a fusion reactor.

In order to make use of the very high thermal conductivity of copper also in a fusion reactor, long fibre reinforced Cu MMCs (metal-matrix composites) are being developed. The reinforcement of Cu with crystalline long SiC fibres should result in creep resistant Cu-based materials for high temperature applications. Development of Ti-containing interfaces on nm-scale between the fibre surface and the Cu matrix resulted in interfacial strength values which are of the order of the values obtained with developmental grades of SiC_f reinforced Ti-alloys for aeroturbine applications.[17,18] Fibre push-out experiments show an interfacial shear strength value of 60 MPa for SiC_f-Cu. In scanning electron micrographs the plastic deformation of the Cu matrix after a push-out test is clearly visible, figure 4.

4.2 Plasma-facing materials and barrier films

The main plasma-material interaction processes and the selection of the plasma-facing materials for ITER have been subject of reviews.[19,20] For ITER a detailed Materials Assessment Report has been issued which compiles the fusion relevant aspects and data on Be, W and CFC materials.[14]

In ITER, the components facing the plasma will consist of carbon-fiber reinforced carbon, tungsten and beryllium as plasma-facing material and CuCrZr as high thermal conductivity heat sink. The selection of these materials has been driven mainly by the need to furnish the ITER first wall with a plasma-facing material of low atomic number (Z) in order to provide maximum operational flexibility which coincides with the high tolerance of the plasma towards influxes of low Z impurities. The use of carbon on the first wall, as applied in most present day experiments, has been avoided, since the first wall is subject to erosion processes. The eroded carbon would eventually redeposit in the form of amorphous hydrocarbon films with high fractions of tritium and thus lead to a rapid build-up of a high tritium inventory. This is undesirable from a radiological point of view for ITER, but it must also be avoided in a fusion reactor because tritium must be produced on site and must not get permanently bound in hydrocarbon films.
In addition to the very high heat loads to the divertor under steady-state operation the effect of heat flux transients has major impact on the materials selection. Experimental investigations on the materials damage resulting from thermal transients have been carried out in plasma, electron and ion beam facilities.[16] Regarding the divertor, the materials selection for ITER has been mainly driven by highly peaked transient heat fluxes which would lead to excessive erosion of most materials during e.g. disruptions. In divertor target areas with highest transient heat loads, carbon-fibre reinforced carbon material (CFC) with 3-d fiber reinforcement and very high thermal conductivity will be applied. In order to minimize the amount of carbon used as plasma facing material, the less highly loaded regions of the divertor will be furnished with tungsten. Concerning a fusion reactor, it is expected that the effect of thermal transients has to reduced very strongly to allow the operation of components with reasonable lifetime, since the erosion and fatigue of the plasma-facing materials would be excessively large.

Beryllium to cover the major part of the plasma-facing wall, as foreseen in ITER, will not be a suitable option for an economic fusion reactor concept. Therefore an alternative must be identified. Plasma-material interaction studies in the tokamak ASDEX Upgrade indicate that tungsten may be a plasma-facing material which can provide sufficient lifetime of the first wall components.[21,22] The use of tungsten on large surface areas in this tokamak resulted in low plasma impurities, well below the tolerable limit of $2 \cdot 10^{-5}$ and very little tungsten erosion from energetic neutral atoms escaping from the plasma. It was shown that plasma-sprayed W coatings with a thickness of 2mm on Eurofer steel substrates can withstand 1000 thermal cycles with heat loads of 2 MW/m^2, which is sufficient for the first wall application in a fusion reactor[23]. Further plasma-facing materials issues related to the application in a reactor are described in [24].
Hydrogen isotopes as the fuel of the fusion reaction constitutes a very general problem for metallic materials, especially for ferritic-martensitic steels because of the low solubility and for vanadium alloys because of exothermic hydride formation, leading to embrittlement.[25,26]
Furthermore, the transport of tritium through the heat sink materials into the reactor coolant has to be limited. Thin film technology is employed to deposit hydrogen permeation barrier coatings with film thickness of one micrometer and below. These coatings are to be applied as a

sandwiched barrier between the plasma-facing material and the metallic heat sink/wall material of the plasma-facing components. α-crystalline alumina (Al_2O_3) films which were deposited by low temperature physical vapour deposition (PVD) techniques on EUROFER steel showed that permeation reduction factors of 1000 can be reached for deuterium.[27] Further application of permeation barriers with electrical insulation and corrosion resistant functions are required for specific breeding blanket designs. Some design concepts, which employ pure lithium or lithium containing liquid metals/molten salts, must rely on coatings for the purpose of corrosion protection, ellctrical insulation as well as their hydrogen barrier function.[28,29] Here, work on thin (1 μm) erbia (Er_2O_3) coatings has demonstrated that at the same time a good corrosion resistance and a permeation reduction by a factor of approx. 400 can be obtained, fig. 5.[30]

6. Concluding remarks

The physics and technology of hot fusion plasmas have already reached a very elaborate stage. Key features of energy production by magnetic confinement thermonuclear fusion have already been demonstrated: Alpha particle heating, necessary for self-sustained burn, has been demonstrated and a substantial amount of fusion power has been produced. Solutions for the materials problems are currently being worked out, some being already at hand. The preparations for ITER, for example, have brought the issue of high heat flux components a big step forward. Further reactor-oriented development of materials for the plasma-facing components is under way. The topics of structural materials and of high temperature heat sink materials are being addressed as well as the further development of plasma-facing materials and tritium permeation barriers for fusion reactors.

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	ITER		reactor	
	First Wall	**Divertor target**	**First Wall**	**Divertor target**
component replacements	none	up to 3	5 year cycle	5 year cycle
av. neutron fluence (MWa/m^2)	0.3	max.0.15*	10	5
displ./transmut. product. dpa / appm (He); (dpa / %Re for W)	Be 1 / 1000 Cu 3 / 30 SS 3 / 30	CFC 0.7 / 230 W 0.7 / 0.15% Re Cu 1.7 / 16 SS 1.6 / 16	W 30 / 6% Re RAFM steel 120 / 1200	W 15 / 3% Re Cu 60 / 600 RAFM st. 60 / 600
normal operation				
No. of cycles	30000	10000	< 1000	< 1000
Peak particle flux ($10^{23}/m^2s$)	0.01	~10	0.02	~10
Surface heat flux (MW/m^2)	< 0.5	~10** / 3	< 1	...10...
PFM operational temp. (°C)	Be: 200...300	W: 200...1000 CFC: 200...1500	W: 550...700	W: 350...500
ELM energy density (MJ/m^2)	-	< 1	–	reduced
ELM duration (ms)/ {Frequency}	-	0.2/ {few Hz}	–	"grassy"?
off-normal operation				
Peak energy density (MJ/m^2)	60 (VDEs)	30 (Disr.)	-	?
Duration (ms)/ {Frequ. (%)}	300 {1%} (VDEs)	1-10 {<10%} (Disr.)	-	1-10, max. 10 events

* without replacement;
** slow transients 20 MW/m^2 lasting 10 s (10% frequency)

Table 1: Operation conditions for the plasma-facing components of ITER and a of DEMO-like fusion reactor[9]

List of figure captions

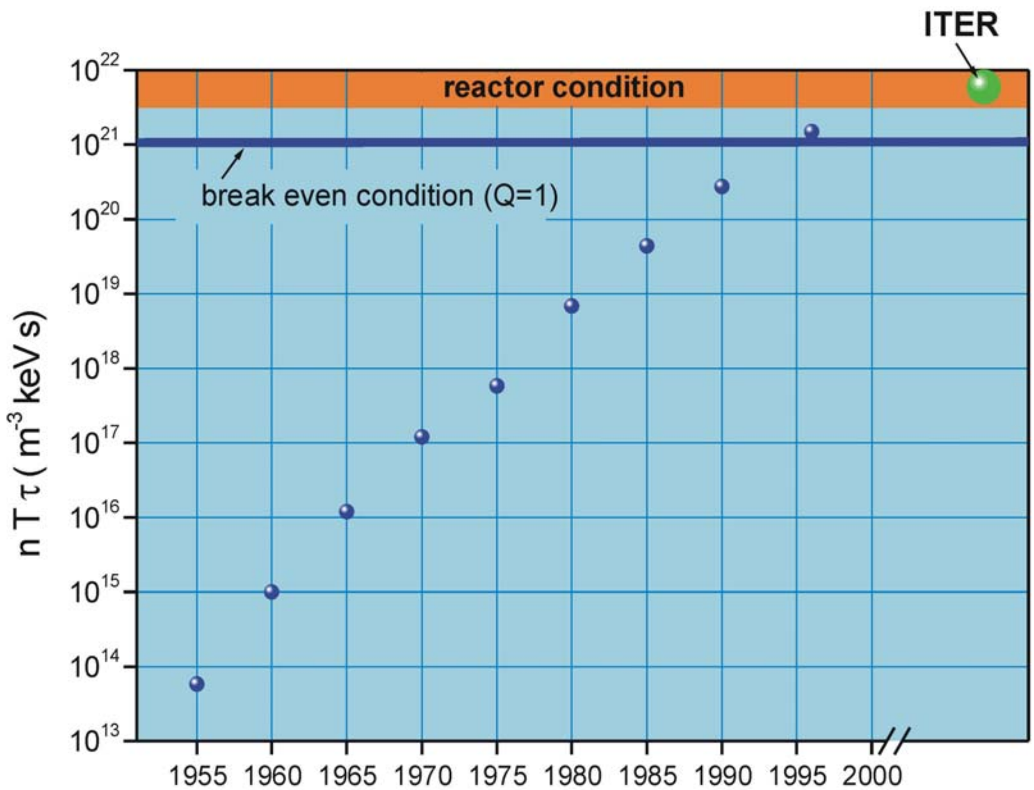
Figure 1: Development of the fusion product of plasma density (n), plasma temperature (T) and energy confinement time(τ) (after [1]).

Figure 2: Achieved fusion power in the JET tokamak during the 1997 deuterium/tritium experiment (after [2]). The peaked profiles refer to special transient discharge scenarios.

Figure 3: Cutaway view of ITER vessel half showing the first wall and the divertor (source: www.iter.org)

Figure 4: Scanning electron micrographs of a SiC_f-Cu MMC after push-out test. The interfacial strength between fibre and matrix has been adjusted by a nm-scale Ti interlayer. Intense plastic deformation of the Cu in the push-out region is visible.

Figure 5: Permeability of deuterium through Eurofer steel at 600°C without and with a 1μm Er_2O_3 coating.



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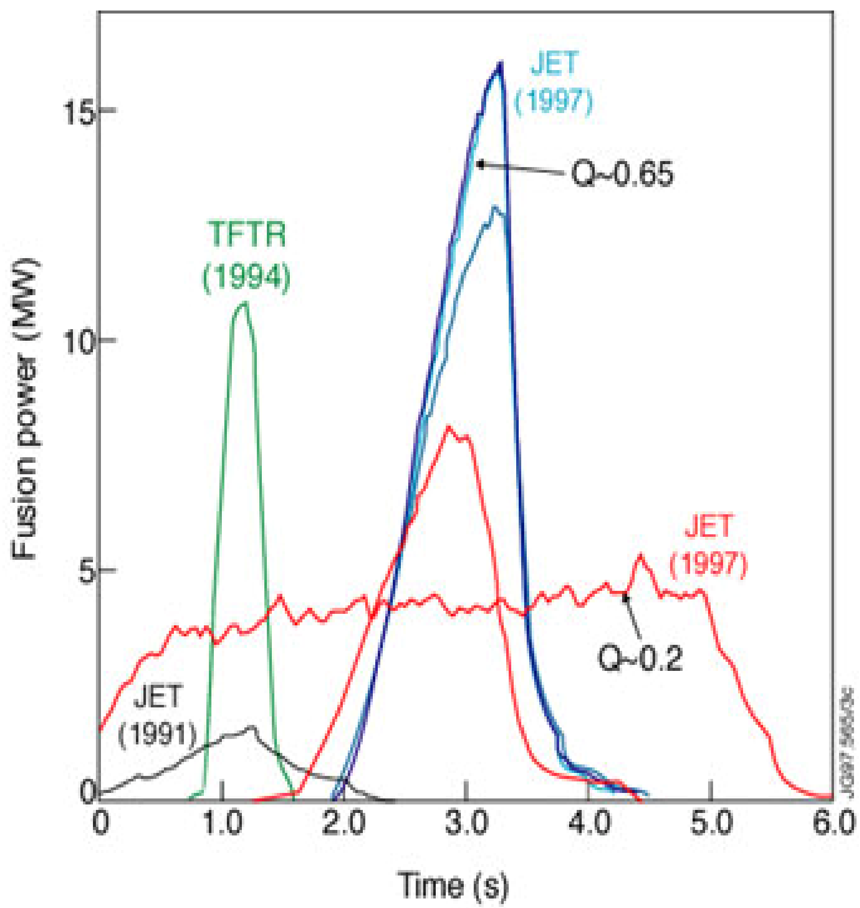

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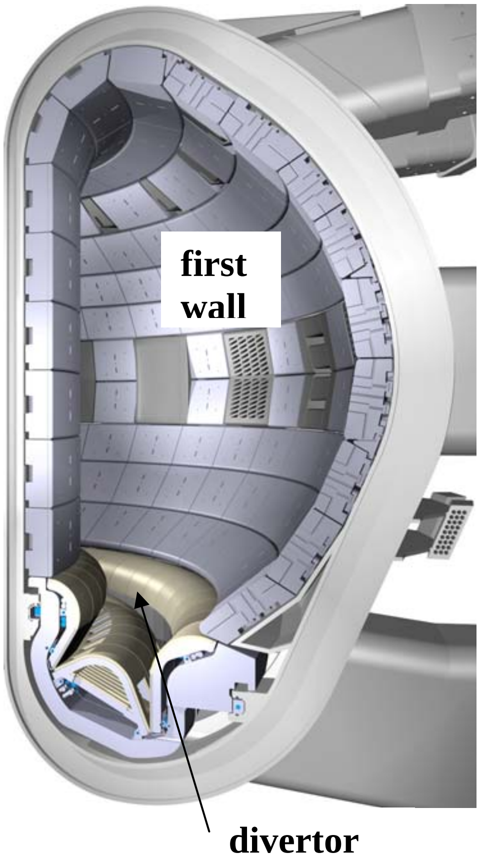

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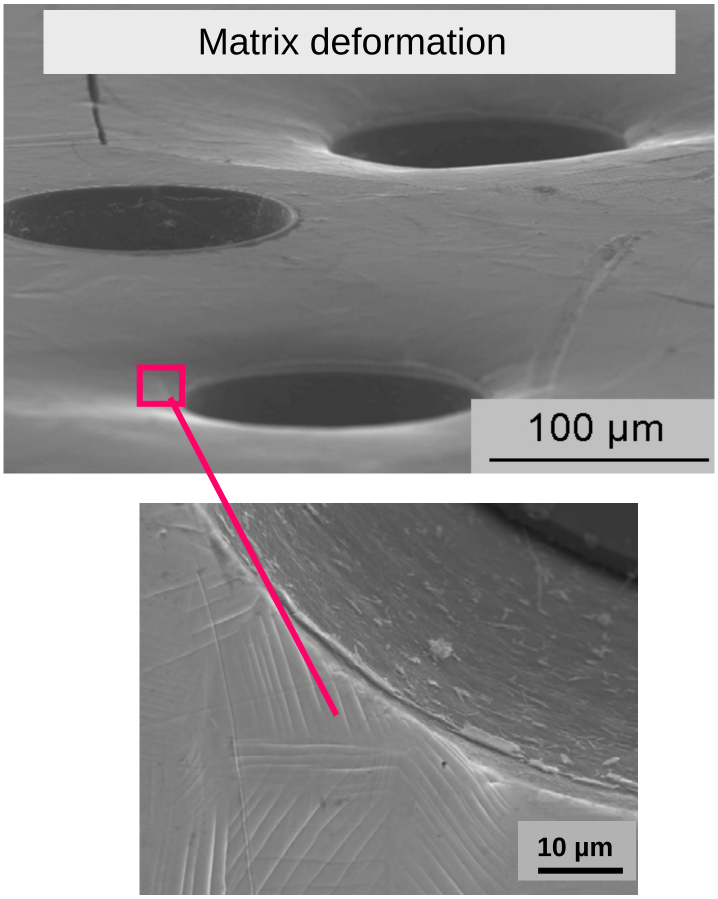

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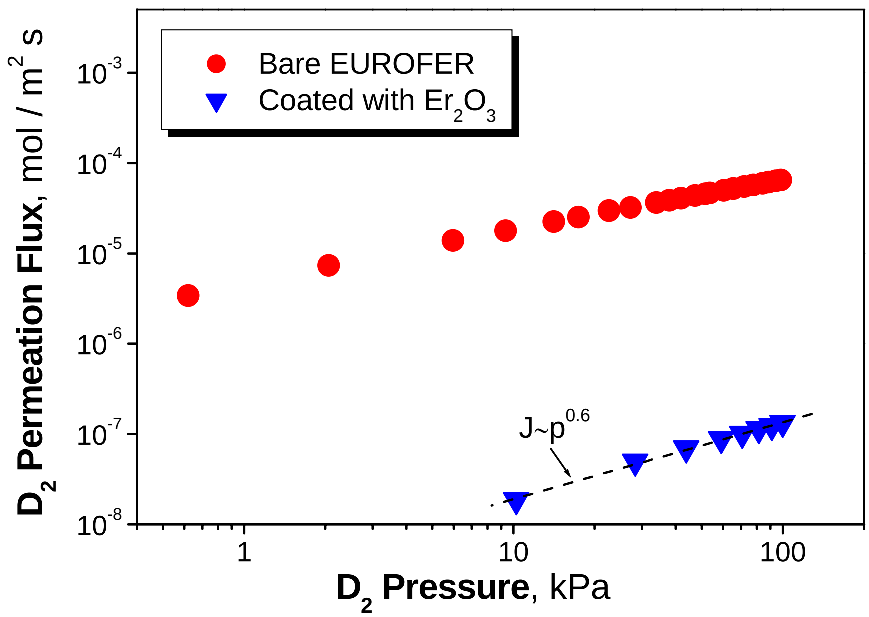

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