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**Corso di Dottorato di Ricerca in Ecosistemi e Sistemi Produttivi, Ingegneria dei Biosistemi -
Ciclo XXXI**

**Design and realization of prototype of Plasma Facing Unit and Thermal Shields of fusion
reactor**
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Introduction

Nuclear Fusion is one of the most promising and appealing research fields that could help to answer the increasing needs of clean energy. The energy producing process that powers the stars is based on nuclear fusion reactions. This process is characterized by the union of multiple charged light atomic nuclei to form an heavier nucleus, with the release of energy. In the sun the main nuclear reaction converts Hydrogen into Helium under the extreme pressure of gravity. This condition is called gravitational confinement and is not reproducible out of the stars.

The operation of fusion power reactors will impose very demanding operation conditions on the Plasma Facing Components (PFCs). The PFCs will be operated in an environment which comprises incident particles and heat flux from the plasma. The surface of the plasma-facing material (PFM) is subjected to erosion by energetic ions and neutral atoms escaping from the plasma [Bolt, 2001].

PFC in DEMO consists in: an armour material facing the plasma in beryllium or tungsten; heat sink in copper to transfer the heat from the armour to the cooling channels that should have adequate thermal conductivity and good resistance to water corrosion.

This thesis work aims to contribute for the optimization of the monoblock thickness of the Plasma Facing Component (PFC) in DEMO Divertor.

The thickness of the monoblocks chosen for the DEMO PFU is currently 4mm, while the ITER thickness is 12mm. To carry out this work it was decided to start from the geometry proposed by ENEA (ITER-like) with a thickness of 12 mm monoblocks.

A reduced thickness in the direction of the tube axis allows to reduce junction stresses due to thermal loads. The supports welded on a small surface may not be sufficient to withstand the electromagnetic loads that tend to tear the PFUs from the supports.

The model applied is therefore appropriate to describe the actual system of constraints of PFU when local effects of load and heat exchange are neglected. In order to investigate whether the monoblock thickness or gap between monoblocks is dominant for the fatigue life of CuCrZr tube, two different configurations with 4mm thick monoblocks and different gap, and a configuration with 12mm thick monoblocks have been studied.

FEM (Finite Element Method) analysis consists of a CuCrZr tube 60mm long, corresponding to the distance between two successive support attachment to the cassette as well in the current ITER divertor design. Between the W monoblocks and the tube 1mm of pure copper interlayer is present. Appropriate symmetry conditions allow to simulate the entire PFU. The

pipe and interlayer materials have elasto-plastic behavior, while the monoblock Tungsten is considered to have linear elastic properties. The aim is to verify the differences in terms of stress induced in the pipe by a nominal thermal load.

Another part of this thesis work focused on the HRP joining technique carried out by the ENEA research group with which I collaborated.

The primary objective of this work was to verify the possibility to obtain, after the Hot Radial Pressing (HRP) process, the PFUs within final shape and tolerances. In this case no additional machining or finishing of the PFUs would be requested to meet technical specifications.

Unlike HRP, hot Isostatic Pressing (HIP) is a process that has the following characteristics: subjects a component to both elevated temperature and isostatic gas pressure in a high pressure containment vessel; the pressurizing gas most widely used is argon; the chamber is also heated and deformed; pressure is applied to the material from all directions (hence the term "isostatic"). It is usually used to reduce the powder porosity (sintering) but in Fusion manufacturing is used to join the heat sink (Cu alloy) to Cu interlayer. It follows that:

- The joining is obtained by solid molecular diffusion at the required parameters
 - The bonding temperature is lower than other processes (brazing, AMC, cast ecc)
 - The joint has an high strength but lower than brazing
 - The residual stresses are usually lower
 - High deformation of the components and of the canister has to be expected due to the required high pressure
 - Machining is required for the extraction of the component from the canister after the process
- The last part of the thesis work consists of a preliminary thermo-hydraulic analysis, in order to estimate the power requested by the cryogenic system.

The Divertor Tokamak Test (DTT) facility has been proposed by ENEA, in collaboration with other Italian institutions, to investigate power exhaust solutions with a machine capable to integrate all DEMO relevant physics and technology issues.

One of the main challenges in the roadmap is to develop a heat and power exhaust system able to withstand the large loads expected in the divertor of a fusion power plant. Therefore, in parallel with the programme to optimise the operation with a conventional divertor based on detached conditions to be tested on ITER, EUROfusion has launched a dedicated project to investigate alternative power exhaust solutions for DEMO, and the design of a new machine named "Divertor Tokamak Test facility" (DTT), capable of eventually integrating all relevant physics and technology issues. The set of possible alternative solutions to be assessed includes advanced magnetic configurations and liquid metal divertors.

My innovative contribution is a preliminary thermo-hydraulic analysis necessary to size the power of the cryogenic system by FEM models.

Thermal analysis was performed using a two-dimensional model in order to optimize the position and number of cooling pipes as a function of distance between the ribs in each of the components (VVTs, PTS and CTS), and for different poloidal sections. To minimize the heat transferred to the superconductive coils, the vacuum vessel (VV), heated by water at 100°C, is shielded by a Thermal Shield (TS) cooled with helium at 70K. The thermal shield design is cooled by Helium which flows inside pipes welded on one side, internally to a box structure similar to the one of JT-60SA.

1. Nuclear fusion and magnetic control

1.1. The fusion power

The fusion of nuclei of Hydrogen isotopes Deuterium and Tritium is one of the most promising fusion reaction reproducible on Earth yielding the following nuclear reaction:



In the previous equation, at the right hand side we can see as a product of the reaction one neutron and one nucleus of Helium, also known as alpha-particle, with a total energy production of 17,6 MeV. In order to obtain an overall positive energetic balance it is necessary to heat the ionized gases, called plasma, up to fusion temperature keeping them confined in a limited space for a time sufficient to obtain more energy power then the one necessary to compensate the losses and the energy necessary to reach this condition. This is known as Lawson's criterion and can be written as:

$$N \cdot \tau > L(T)$$

where N is the charged particle density, τ is the energy confinement time and $L(T)$ represents the limit of the ignition domain. The effective temperatures requested are of the order of hundred millions of centigrade degrees. In those conditions the plasma must not be allowed to come into contact with the walls of the reactor container, since a part of the wall surface would evaporate and the plasma would be immediately polluted by impurities and cooled down, loosing the conditions needed for the fusion and damaging the reactor itself. The aim of the international research in thermonuclear control is the realization of a device able to confine the plasma satisfying Lawson's criterion.

The first possible approach is the *inertial confinement* that aims to confine the plasma for a very short time τ but at very high density N . This is obtained by focusing high power laser beams (or ions or electrons beams) on a target (pellet) of few mm diameter filled with D-T fuel causing it to reach a very high pressure and temperature. If the fuel is dense and hot enough, the fusion reaction rate will be high enough to burn a significant fraction of the fuel through a micro explosion before it dissipates. The current technological challenge in this method is achieving a powerful and homogenous pellet irradiation at a high repetition rate.

The second approach is the *magnetic confinement* where the density N is lower but the confinement time is kept for longer period confining the ionized plasma particle using

magnetic fields. This approach is based on the principle that ionized electrically charged particles, positively for the nuclei of the Hydrogen isotopes and negative for the electrons, orbit around the magnetic lines.

To confine those particles long enough, the most promising approach is to maintain the field lines on a closed surface. For this reason, the best results have been achieved using a vessel with toroidal shaped geometry. In fact, so far, the *tokamak* machines have proven to be the most successful devices even if the Lawson criterion has not yet been achieved in any reactor. Actually, it is worth observing that the latest generations of machines have come close to reach this goal.

A fundamental parameter to the physics of the problem is the fusion energy gain factor Q , that represents the ratio between fusion power produced in a nuclear fusion reactor and the power required to maintain the plasma in steady state. The *Joint European Torus* experiment (JET) has the current record that has been achieved with a value equal to 0,65 (note that $Q \geq 1$ is the condition for self-sustained plasma) reached in 1997 [M. Keilhacker and the JET], by using microwaves and neutral particle beams as additional heating input power, for a total of 16 MW of thermal power.

From November 2006 an international agreement formally agreed to fund *ITER* (original acronym of International Thermonuclear Experimental Reactor, now used with the Latin meaning), a new large tokamak device. ITER has the objective to demonstrate the scientific and technological feasibility of nuclear fusion allowing achieving and study the fusion plasmas under conditions very similar to those expected in a fusion power plant. It should produce 500 MW of fusion thermal power sustained for up to 1000 seconds with a Q factor ≥ 10 .

1.2. Tokamak devices

The so-called “tokamak”, a Russian acronym for *TO*roidalnaya *k*amera ee *MA*gnitnaya *K*atushka (“тороидальная камера с магнитными катушками”), which translated literally means “Toroidal Chamber with Magnetic Coils”, is nowadays the most used configuration for the plasma confinement. In this experimental device, the current flowing in the turns of the toroidal field coils surrounding the reactor generates a toroidal magnetic field. Unlike what happens in the case of magnetic mirrors, this toroidal magnetic field is inversely proportional to the distance from the axis of symmetry of the torus. In fact, considering a torus (which can be thought as a solenoid bent into a donut Shape), like the one in Figure 1.1:

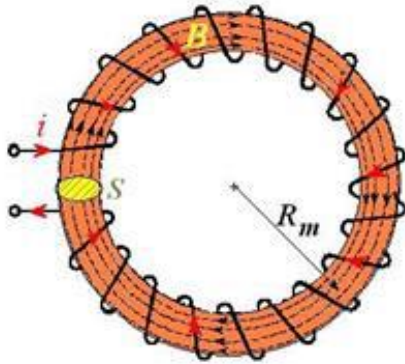


Figure 1.1: Solenoid wound on toroidal core

on which N turns are wound, if i is the current flowing through them, by applying the Ampère's law to the generic concentric circumference with R radius, chosen as a closed line, we can write:

$$(B_{tor})(2\pi R) = \mu_0 i N$$

and then the toroidal magnetic field B_{tor} is given by:

$$B_{tor} = \frac{\mu_0 i N}{2\pi R}$$

B_{tor} is inversely proportional to the distance R from the symmetry axis of the system.

Since a charged particle describes a helical trajectory around magnetic field lines (Figure 1.2):

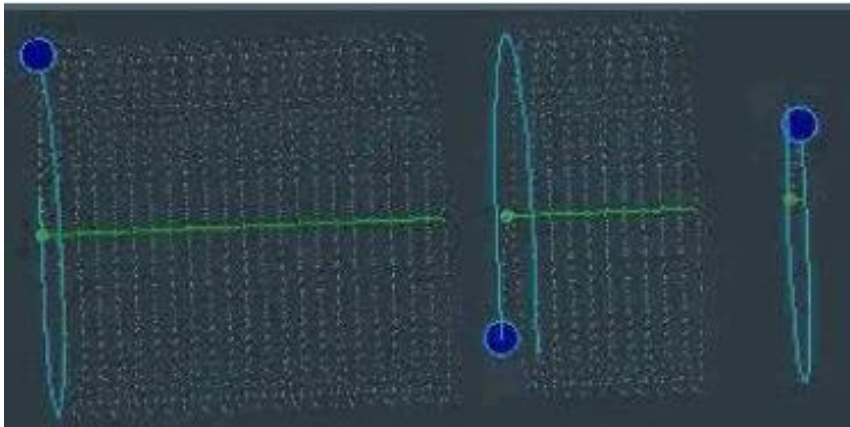


Figure 1.2: Trajectory of a charged particle around a magnetic field line.

on a simple circular trajectory of this type, the particle undergoes a slow cross drift, due to the drift gradient of the magnetic field and centrifugal force, depending on the sign of its charge. For example, the ions will drift up (as illustrated on the diagram opposite) and the electrons will drift down.

This suggests that, by itself, the toroidal field is not sufficient to allow the plasma confinement. Therefore, to compensate this effect, the idea is to stabilise the configuration by adding a poloidal component to the toroidal magnetic field (Figure 1.3). Inside a tokamak this field, with closed force lines lying in planes perpendicular to the toroidal direction, is mainly produced by a current flowing in the plasma (in the toroidal direction, as shown in Figure 2.4).

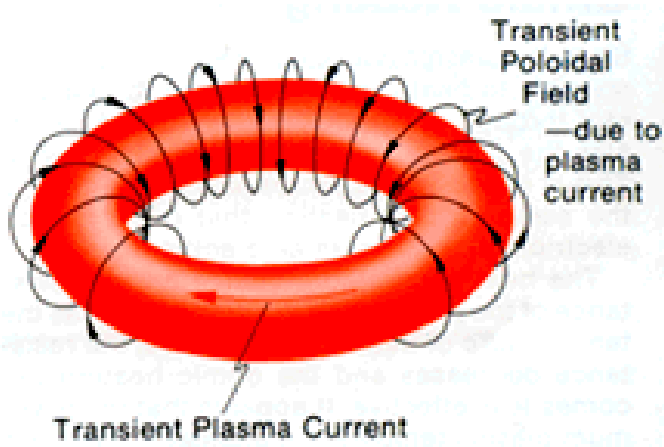


Figure 1.3: Poloidal magnetic field due to the plasma current.

The superposition of toroidal and poloidal magnetic fields produces magnetic field lines that are helices twisted round stacked toroidal surfaces having a helical path around the torus (Figure 1.4).

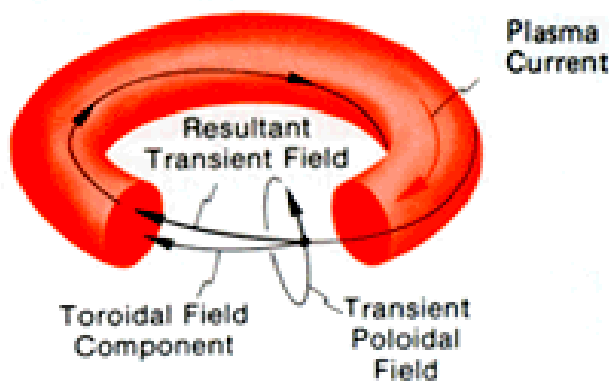


Figure 1.4: Magnetic field line trajectories in the plasma.

A family of nested toroids formed by the magnetic field lines arise. The inner toroid, degenerated into a single closed curve, is defined *magnetic axis*.

Even in this case, the particle undergoes a drift, because of its charge, which distances it from the field line. The particle then spends half its time head upwards, where the vertical drift

moves it away from the magnetic surface, and the other half head down, where the vertical drift pulls it back to the magnetic surface. The drift effect is thus on average compensated.

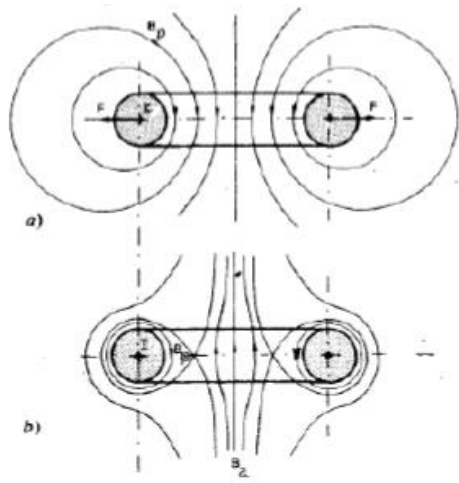


Figure 1.5: Configuration of the poloidal magnetic field a) before and b) after the application of a vertical field.

A third, vertical magnetic field B_z fixes the position of the current in the plasma and prevents a drifting of the plasma due to the magnetic field gradient from the region of high magnetic field on the inside of the toroid to the region of lower magnetic field on its outside (Figure 1.5). The coils generating the vertical field are defined “poloidal field coils” (see Figure 1.6). In a tokamak, toroidal and vertical magnetic fields are produced by external coils, whereas the poloidal magnetic field is induced by a current flowing toroidally in the plasma. This plasma current I_p is generated by the transformer effect, from a primary circuit of which the secondary, in a single turn, is the plasma. In a tokamak, the “central solenoid” is the magnet that drives the current in the plasma.

In Figure 1.6 it is possible to see the whole structure of a tokamak.

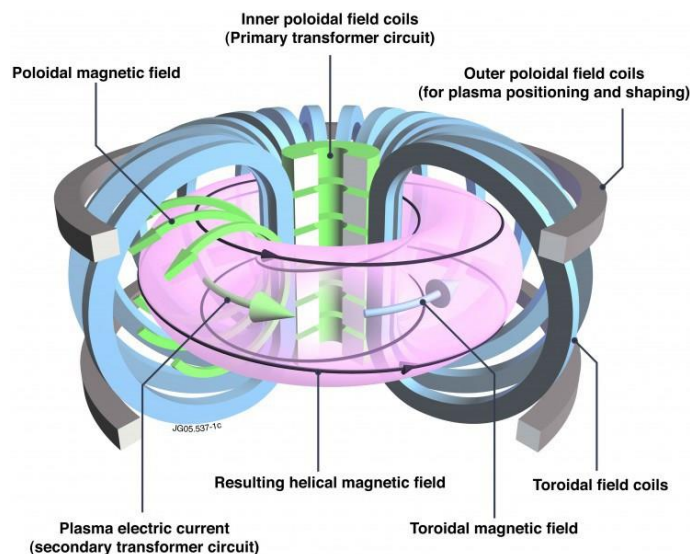


Figure 1.6: Coils and magnetic fields in a Tokamak.

Another magnetic confinement configuration is the “stellarator”, in which the magnetic field is provided completely by external toroidal as well as poloidal coils. In this experimental device, in addition to the standard toroidal coils, helical windings around the vacuum chamber that contains the plasma are employed. These additional windings (Figure 1.7) create a helical magnetic field in the toroidal chamber.

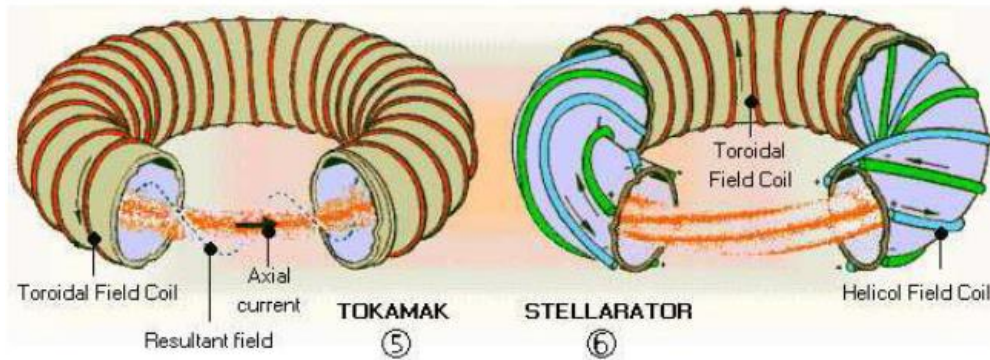


Figure 1.7: Comparison between tokamak and stellarator.

The fact of not having an intense current flowing in the plasma is an advantage in the event of *plasma disruptions*. A disruption is a violent event that terminates a magnetically confined plasma, and it is usually the consequence of a rapidly growing instability, often of the MHD type. In a disruption, the kinetic and magnetic energy of the plasma drop drastically and heat and particles are released on a short timescale and dumped on the first wall of the containment chamber, causing damage in proportion to the stored energy. The loss of confinement may be also associated with the production of runaway electrons, which are electrons accelerated close to relativistic velocity by the rapidly changing electric field induced by the disruption, which may also produce damages, and it is one of the open issues in the ITER machine project. Moreover, typically a tokamak operates in a pulsed mode since the plasma current is generated by the transformer effect; the pulse duration is thus limited by the capacity of the primary circuit generating the plasma current. An alternative is the use of the current drive induced by the additional heating systems, which could in principle make the pulse steady state. A demonstration of fully non inductive current was achieved in several European and international experiments [FuseNet, <https://www.fusenet.eu/node/1210>]. Conversely, a stellarator can operate continuously, since the fields come entirely from external coils and there is no plasma current.

1.2.1. Plasma configurations

In the magnetic confinement devices, in order to reduce the content of impurities into the plasma, which usually give rise to radiation losses and also dilute the fuel, the plasma is confined inside closed magnetic flux surfaces. Two techniques are used:

- the first is to define an outer boundary of the plasma with a material limiter;
- the second is to keep the particles away from the vacuum vessel by means of a modification of the magnetic field.

When the plasma is directly in contact with the wall, the first point of contact with a solid object, which thus “limits” the plasma, defines the so-called “Last Closed Flux Surface” (LCFS) which determines the plasma boundary. The name “*limiter*” defines both this configuration and the components that are in direct contact with the plasma. It is also possible to produce a “*divertor*” configuration in which the contact between plasma and wall takes place on the divertor plates that consequently have to be designed to withstand a large power flux. Typically, this configuration is generated by producing a null point in the poloidal magnetic field within the chamber (Figure 1.8).

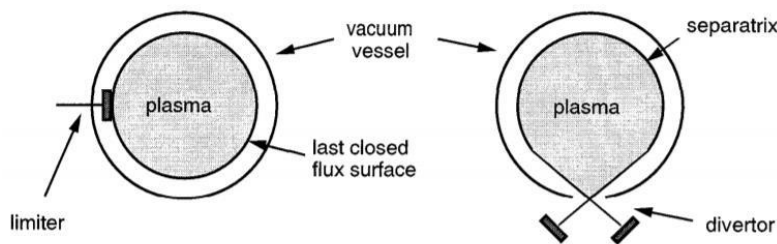


Figure 1.8: Limiter (left) and Divertor (right) plasma configurations.

Limiter configuration

In the *limiter configuration*, the charged particles close to the outer edge of the plasma, orbit around to the field lines impinging on the limiter: this is defined “plasma-wall interaction”. When the plasma particle hits a plasma facing component, it is neutralized (it becomes atom or molecule recovering electrons), becoming unaffected by the magnetic field and thus free to move, colliding with the walls of the chamber or with other particles, until it will be ionized again. At this point, if it has been ionized in the plasma core, it begins once more to orbit around the field lines continuing to feed the plasma, otherwise it impinges on the wall again (Figure 1.9).

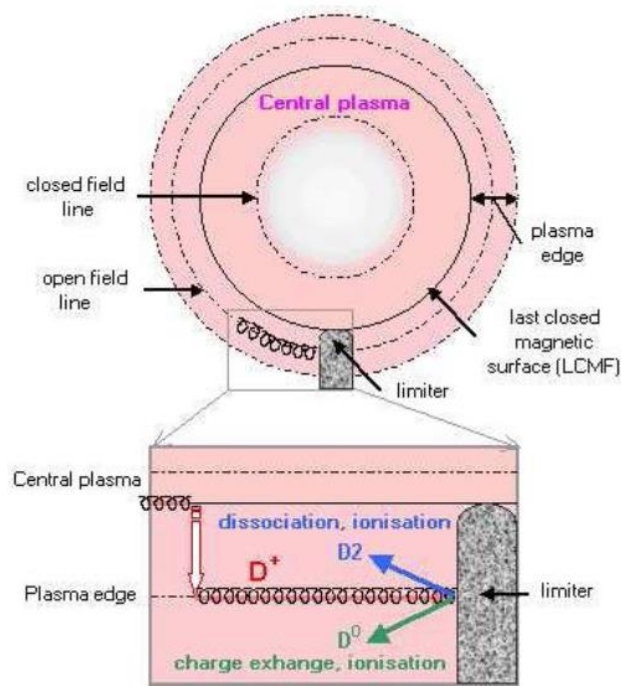


Figure 1.9: Plasma-wall interaction

This process continues until the particle leaves the system, absorbed from the wall or from an external pumping system. In a tokamak a very wide range of temperatures is covered within few meters of distance: from tens of million-degrees in the plasma core, to tens of thousands of degrees at the plasma edge, to hundreds on the components surfaces up to temperatures close to the absolute zero of superconductors (e.g., in ITER superconducting magnets will be employed). With the development of plasmas at increasingly higher performance, the importance of plasma-wall interactions is becoming more important. With the increase of the power coupled to the plasma it has been observed that the wall, under the bombardment of high energy particles, emit impurities. These impurities give rise to radiation losses and also dilute the fuel reducing the overall performance of the machine.

A first solution to this drawback, in addition to an accurate limiter design, has been the development of new materials.

Divertor configuration

The reduction of impurities entering the plasma plays a significant role in the successful operation of tokamaks. This requires a separation of the plasma from the vacuum vessel. Therefore, after the optimization of limiter materials, the second idea was to keep the particles away from the plasma core by means of a modification of the magnetic field to produce a magnetic divertor. This idea led to the *divertor axisymmetric configuration* (Figure 1.10), where the LCFS is no longer defined as the point of contact with a solid surface, as in the

limiter configuration, but rather it coincides with a magnetic boundary created by adding a suitable coil around the tokamak.

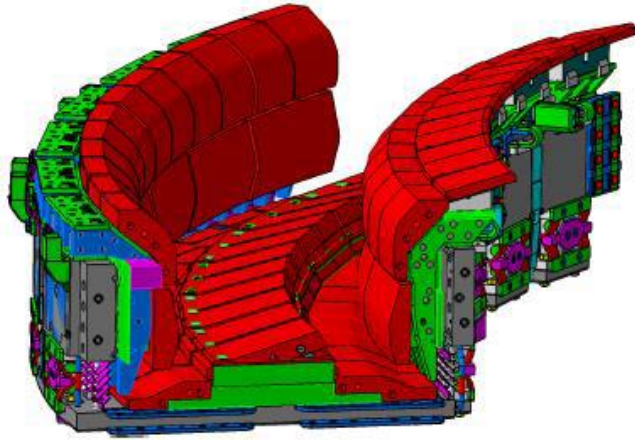


Figure 1.10: Example of a typical divertor in a tokamak.

The aim of the divertor is to lead the outgoing particles to a “target” surface well separated from the plasma, and to restrict the impurity back-flow. A difficult problem associated with the divertor is that of limiting the power density flowing to the target surface. This is necessary to avoid high surface temperatures which can lead to surface melting or catastrophic impurity release by evaporation or other processes.

The required magnetic field is produced by toroidal conductors that create a null (also called “X-point”) in the poloidal field and a separation of open and closed magnetic surfaces. These divertors have the advantage of preserving the essential axisymmetry of the tokamak and can be combined with D-shaped or elliptical cross-sections (Figure 1.11).

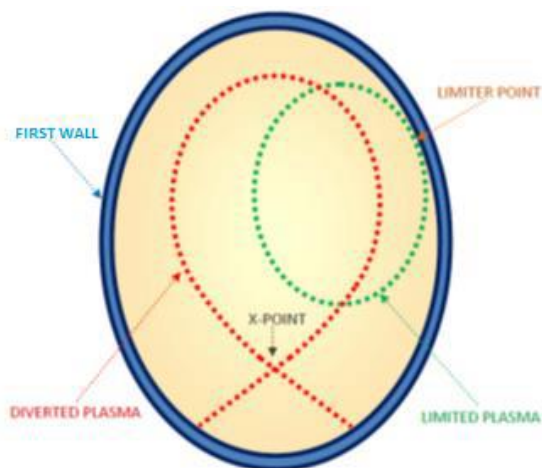


Figure 1.11: Limiter and divertor plasma configurations.

In Figure Figure 1.12, the particle flow leaving the plasma by radial diffusion is represented by the white arrow. In the limiter configuration (left), the charged particles escaped from the

plasma central region follow the field lines impinging on the limiter. The impact can neutralize these charged particles, and on the other hand can remove neutral impurities from the wall. The neutral particles do not follow the magnetic field lines until they are ionized, most likely in the plasma core.

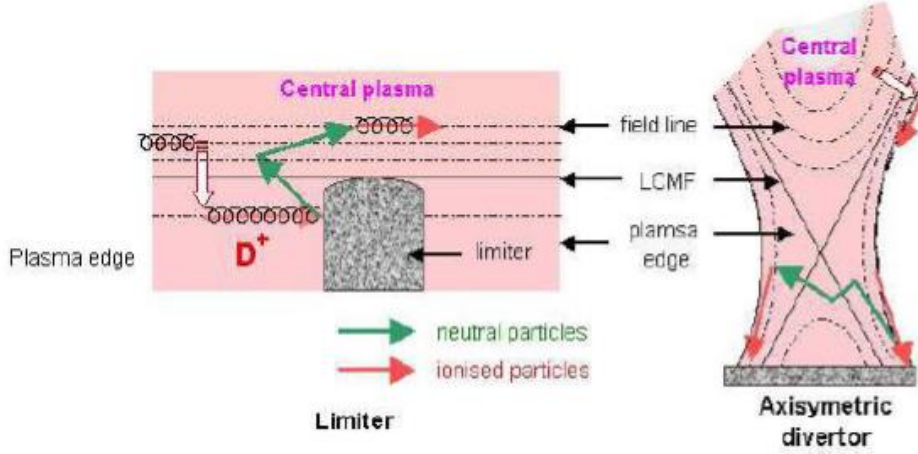


Figure 1.12: Differences between the limiter and divertor plasma configurations.

In the divertor configuration (right), the particle flow moving further away from the plasma is guided by the field lines up to the neutralization regions, far away from the plasma core. More likely, the impurities are ionized again in this marginal region and in this way, following the field lines, are intercepted again by neutralization divertor plates. Therefore, they remain closed in a circuit that prevents them from interfering with the central plasma region and can be effectively removed through cryocondensation pumps.

1.3. Plasma heating

In order to heat the plasma up to 100 millions degrees, namely the temperature needed to trigger the fusion reactions, different techniques are used.

The plasma is self-heated by ohmic effect (“ohmic heating”), thanks to the (toroidal) plasma current I_p induced by the central solenoid.

At low temperatures, the energy transferred to the plasma is considerable and, in large tokamaks, easily produces temperatures of a few keV. This energy, expressed for convenience in terms of specific power dissipation, is given by:

$$P_{\Omega} = \eta J^2$$

Where η is the plasma resistivity and J is the current density. However, as the temperature increases, the frequency of collisions and the resistivity drop. Consequently, at the temperatures required for ignition the ohmic heating is greatly reduced requiring additional heating.

1.4. Energy production in a thermonuclear power plant

A tokamak reactor is considerably more complex than the present experimental devices. Figure 1.13 illustrates conceptually the structure of such a thermonuclear fusion device. A blanket surrounds the plasma. The blanket has mainly three roles. First, it absorbs the 14 MeV neutrons, carrying 80% of the energy produced in the fusion reaction, transforming their energy (essentially kinetic) into heat then removed by a suitable cooling liquid. Secondly, by absorbing neutrons, the blanket protects the superconducting coils and the other external components.

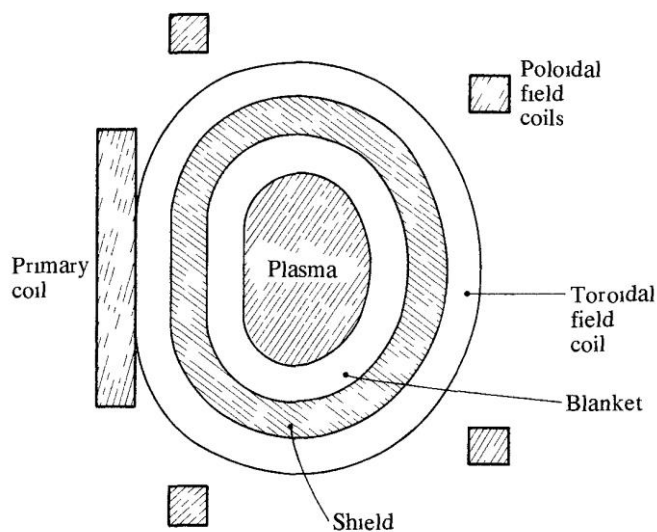
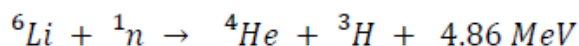
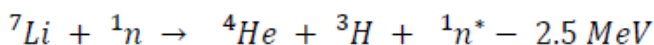


Figure 1.13: Layout of the main components in a conceptual fusion reactor device. [Wesson, 2004]

Finally, the blanket allows the breeding of tritium, which is practically non-existent in nature, feeding the fusion reaction. The blanket is therefore made of light metal lithium (Li, which abounds in the rocks of the Earth's crust and is also present, even though in low concentrations, in the oceans), which is transformed, once the neutron produced in the fusion reaction reaches the blanket, into tritium and helium according to the reactions:



Note that n^* is a slow neutron. The sign '-' in the first equation expresses the need to provide energy from the outside.

A tritium atom is produced in each reaction neutron-lithium, as described in the previous expressions, but it is not possible to design the blanket so that all neutrons undergo such a

reaction. In order to overcome this lack and create a overall breeding ratio higher than one, a neutron multiplier, as for instance beryllium, has to be used. The neutron flux from the plasma decays in the blanket; a blanket thickness between 0.6 and 1.0 m is usually sufficient to absorb most of the neutrons. The flow of neutron energy passing through the outer wall of the blanket in the form of heat must be reduced by a factor of $10^6 \div 10^7$ before reaching the superconducting coils to prevent both the radiation damage and the heating of such coils. This protection is obtained by placing a shield of about 1 m thick of material, such as the steel, between the blanket and the coils.

In an experimental tokamak, such as JET, the surface surrounding the plasma is rather defined “first wall”. The direct contact between the plasma and this first wall is limited to the divertor region during the main stages of a discharge whereas the main plasma touches the “limiter”, only in the early formation stages of the plasma.

In a nuclear power plant, capable of producing energy by nuclear fusion, the power load that reaches the solid surfaces and must be dissipated will be very large. The consequent need to minimize the flow of impurities in the plasma along with a greater design flexibility, favours the divertor configuration.

The idea of a tokamak reactor operating with a current generated by an induced electric field, and therefore as a pulsed device, has substantial drawbacks related to the fatigue stress produced by the thermal cycling and the interruption of the power output. These problems could be removed by a non-inductive current drive system that would make continuous operation feasible. Part of plasma current can be sustained from the self-generated bootstrap current. Such operating scenarios bring the efficiency demanded of the non-inductive current drive system within the range of technical feasibility. Current drive in the plasma centre can be provided by fast waves, high energy neutral beam injection or electron cyclotron waves, whereas efficient current drive at large minor radius is more difficult because of the lower electron temperature and the deleterious effects of trapped electrons.

The heat leaving the plasma and the one produced in the blanket should be removed through a suitable liquid or gaseous coolant and then transformed into electricity by conventional means, as shown in Figure 1.14 and Figure 1.15.

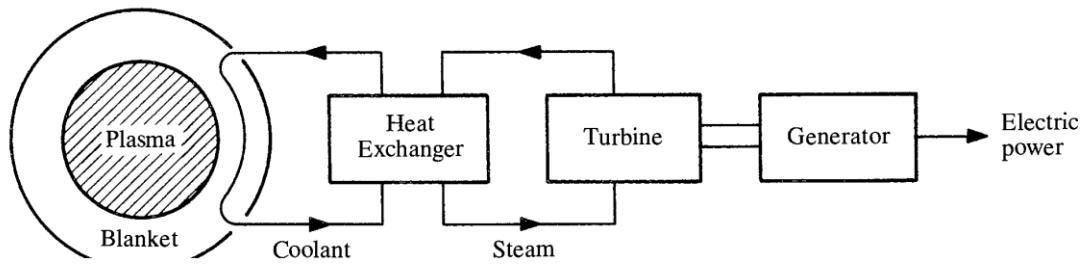


Figure 1.14: How thermonuclear power absorbed by the blanket could be converted into electrical power by conventional means (in the example, a steam turbine). [Wesson,2009]

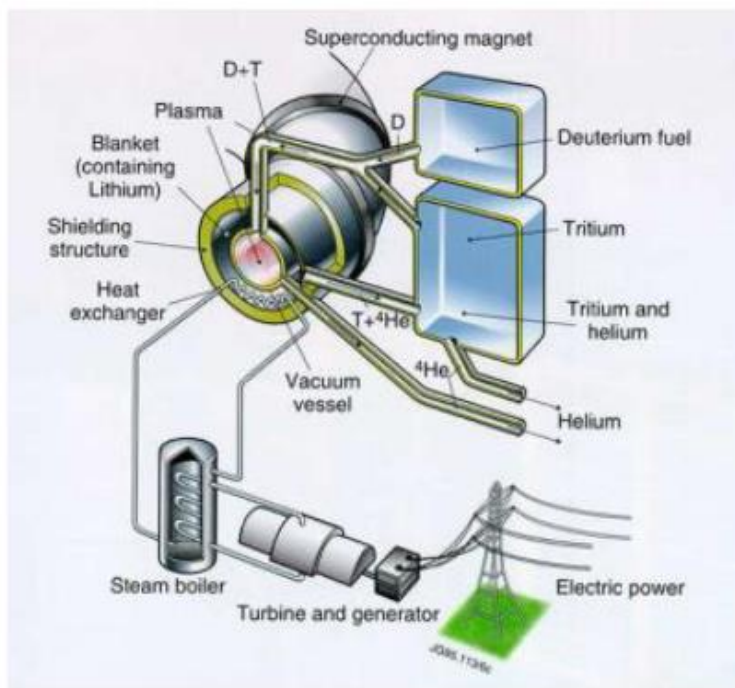


Figure 1.15: Scheme of a fusion power plant.

1.5. The present and future of the fusion: from JET to DEMO through ITER and DTT

JET (Joint European Torus), operating since 1983, is currently the largest tokamak in the world and the only operational fusion experiment capable of using the Deuterium-Tritium reaction. Since 2000, the scientific and technical programme has been conducted under the European Fusion Development Agreement (EFDA) and the device has been operated on behalf of EFDA by the United Kingdom Atomic Energy Authority. The principal aims of the experiment are the investigation of heating and confinement in reactor relevant plasma

conditions, the investigation of plasma-wall interactions and the study of α -particle production, confinement, and consequent plasma heating.

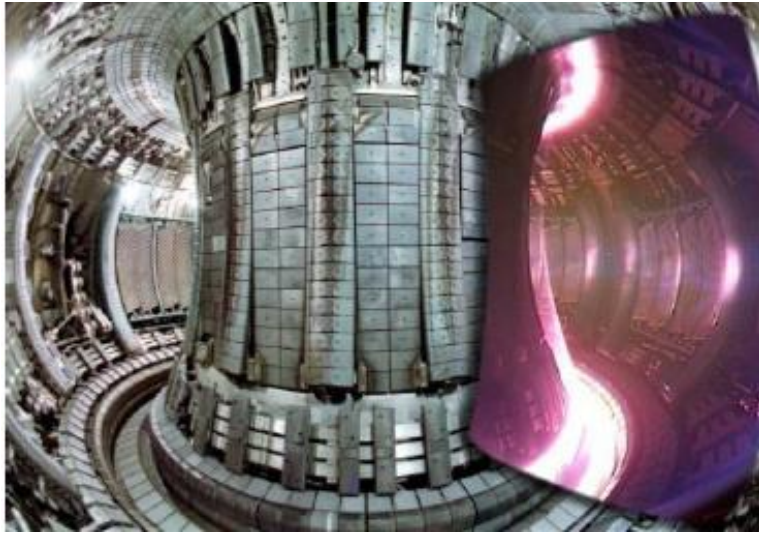


Figure 1.16: The JET vacuum chamber

The heart of the structure is the Vacuum Vessel (Figure 1.16) in which the pressure could reach up to a millionth of a millionth of the density of air.

During experiments, the interaction between currents and magnetic fields produces very intense forces. A D-shaped poloidal cross-section tends to overcome these forces minimizing stresses in the toroidal field coils and maximizing at the same time the plasma volume.

The experiments in JET are carried out with pulsed approach. Usually, during the experimental campaign every twenty minutes there is a shot lasting up to 40 seconds. For each shot, gas puffing pushes a tenth of a gram of matter in the chamber.

An extensive diagnostic suite of around 100 individual instruments capturing up to 18 gigabytes of raw data per plasma pulse is available. Despite continuous progress achieved by JET and other fusion experiments, a larger and more powerful device would be necessary in order to demonstrate the feasibility of nuclear fusion. This is the purpose of the ITER project . ITER (International Thermonuclear Experimental Reactor Figure 1.17) will be the world's largest magnetic confinement plasma physics experiment. The ITER project aims to make the transition from experimental studies of plasma physics to full-scale electricity-producing fusion power stations. The ITER fusion reactor has been designed to produce 500 megawatts of output thermal power for several seconds while needing 50 megawatts to operate. Thereby the machine aims to demonstrate the principle of producing more energy from the fusion process than is used to initiate it ($Q \geq 1$), something that has not yet been achieved in any fusion reactor.

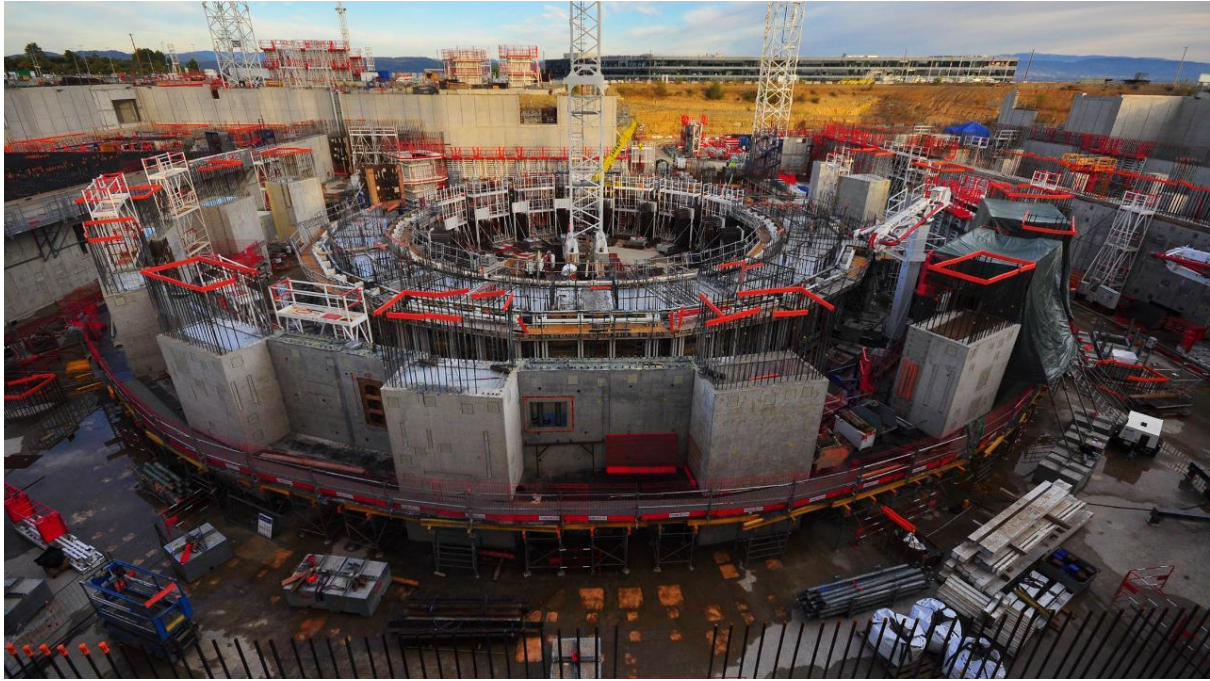


Figure 1.17: ITER (France)[ITER site].

The amount of fusion energy a tokamak is capable of producing is a direct result of the number of fusion reactions taking place in its core. The larger is the vessel, the larger is the volume of the plasma and therefore the greater is the potential for fusion energy.

With ten times the plasma volume of the largest machine operating today, the ITER Tokamak will be a unique experimental tool, capable of longer plasmas and better confinement. ITER has been designed specifically to:

1) Produce 500 MW of fusion power

ITER will not capture the energy it produces as electricity, but - as first of all fusion experiments in history to produce net energy gain - it will prepare the way for the machine that can.

2) Demonstrate the integrated operation of technologies for a fusion power plant

ITER will bridge the gap between today's smaller-scale experimental fusion devices and the demonstration fusion power plants of the future. Scientists will be able to study plasmas under conditions similar to those expected in a future power plant and test technologies such as heating, control, diagnostics, cryogenics and remote maintenance.

3) Achieve a deuterium-tritium plasma in which the reaction is sustained through internal heating

Fusion research today is at the threshold of exploring a "burning plasma" - one in which the heat from the fusion reaction is confined within the plasma efficiently enough for the reaction to be sustained for a long duration. Scientists are confident that the plasmas in ITER will not

only produce much more fusion energy, but will remain stable for longer periods of time [ITER site].

4) Test tritium breeding

One of the missions for the later stages of ITER operation is to demonstrate the feasibility of producing tritium within the vacuum vessel. The world supply of tritium (used with deuterium to fuel the fusion reaction) is not sufficient to cover the needs of future power plants. ITER will provide a unique opportunity to test mockup in-vessel tritium breeding blankets in a real fusion environment.

5) Demonstrate the safety characteristics of a fusion device

ITER achieved an important landmark in fusion history when, in 2012, the ITER Organization was licensed as a nuclear operator in France based on the rigorous and impartial examination of its safety files. One of the primary goals of ITER operation is to demonstrate the control of the plasma and the fusion reactions with negligible consequences to the environment.

Despite the size, the aim and the cost of ITER EUROfusion is already designing its successor. DEMO (DEMONstration Power Station Figure 1.18) is a proposed nuclear fusion power station. The objectives of DEMO are usually understood to lie somewhere between those of ITER and a “first of a kind” commercial station. The following parameters are often used as a baseline for design studies: DEMO should produce at least 2 gigawatts of fusion power on a continuous basis, and it should produce 25 times as much power as required for breakeven. DEMO’s design will be on the scale of a modern electric power station.

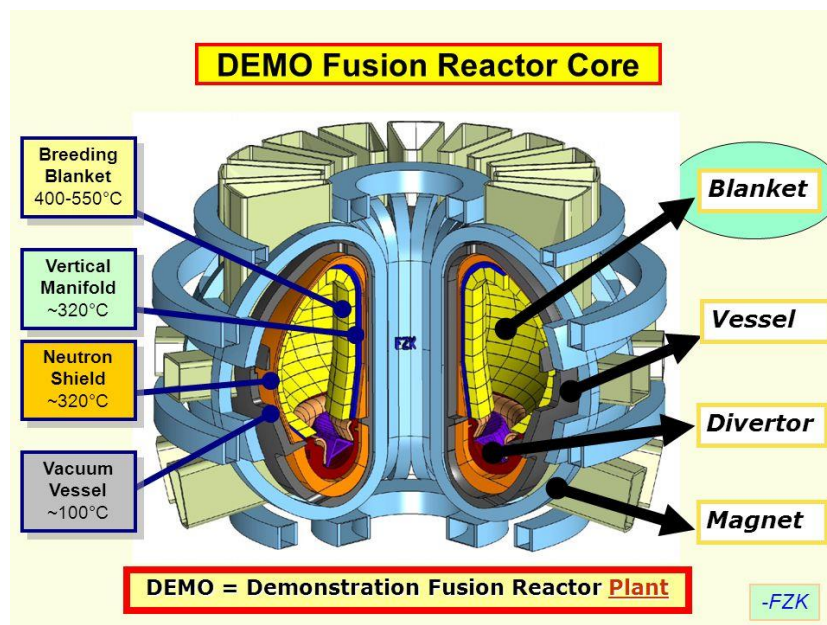


Figure 1.18: DEMONstration Power Station

To achieve its goals, DEMO must have linear dimensions about 15% larger than ITER, and a plasma density about 30% greater than ITER.

At the beginning of 2012, the European Commission requested EFDA to prepare a *technical roadmap* to fusion electricity by 2050. The realisation of fusion energy has to face a number of technical challenges. For all of them, candidate solutions have been developed and the goal of the programme is to demonstrate that they will also work at the scale of a reactor. *Eight different roadmap missions* have been defined and assessed. They will be addressed by universities, research laboratories and industries through a goal-oriented programme during the Horizon 2020 period. According to the present roadmap, a demonstration fusion power plant (DEMO), producing net electricity for the grid at the level of a few hundred Megawatts is foreseen to start operation in the early 2050s. Following ITER, it will be the single step to a commercial fusion power plant.

The mission number two, related to the development of an adequate solution for the enormous plasma heat exhaust of DEMO, is one of the most challenging among the roadmap missions. The design of a new machine named “Divertor Tokamak Test facility” (DTT), capable of integrating all relevant physics and technology issues is included in the EUROfusion roadmap. The DTT project proposal is presently being realized by the Italian Research agency ENEA (Italian National Agency for New Technologies, Energy and Sustainable Economic Development) [DTT Divertor Tokamak Test Facility, ISBN 978-88-8286-318-0, July 2015]. DTT facility will assess possible alternative solutions to the power exhaust issue, including advanced magnetic configurations and liquid metal divertors. DTT should operate integrating various aspects, with significant power loads, flexible divertors, plasma edge and bulk conditions approaching as much as possible those planned for DEMO, at least in terms of dimensionless parameters. The machine will have the possibility to test several different magnetic divertor topologies, in reactor relevant regimes. Different plasma facing materials will be tested (tungsten, liquid metals) up to a power flow of the order of 20 MW/m^2 . The final target of this experiment is the realization of an integrated solution (bulk and edge plasma) for the power exhaust in view of DEMO. The related studies and experiments will allow a valuable development of innovative technologies in several different fields, with relevant spin off for the industries of all European Countries. The present plan on the realization of the device will cover a time of around 7 years, starting from the first tender (during 2019) up to full commissioning and the first plasma (during 2026). The operations

should then cover a period of more than 20 years, up to the initial phases of the DEMO realization. Figure 1.19 Shows a DTT design.

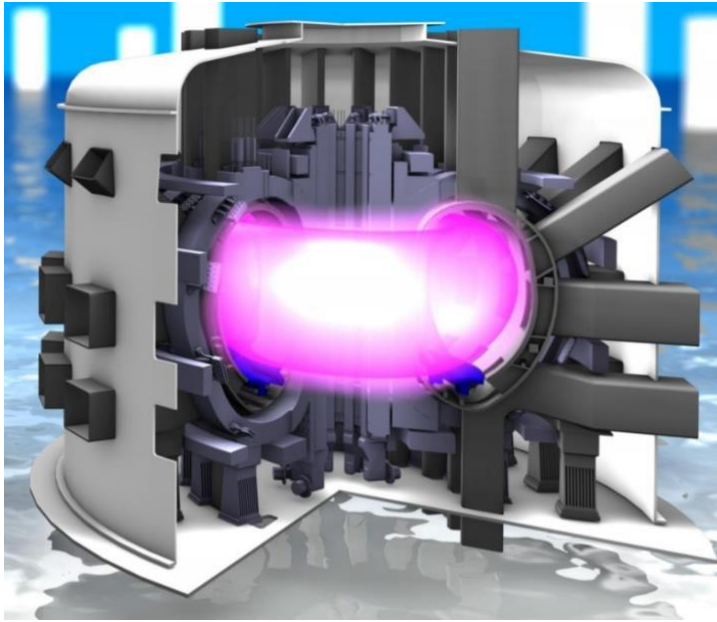


Figure 1.19: DTT design

1.6. The alternative magnetic divertor configurations

Conventionally, the “alternative” or “advanced” divertors are defined as magnetic geometries where a second X-point is added in the divertor region to address the serious challenges of burning plasma power exhaust. The position of the second X-point allows distinguishing among the alternative solutions. In the following, the history and the main characteristics of the fundamental magnetic alternative configurations will be described.

1.6.1. Double-Null (DN) configuration

The DN (Albanese, 2019) is chronologically the first alternative (to the Single Null) magnetic configuration introduced in fusion plasmas. Double null configuration produces a second first order null point in the poloidal magnetic field in the upper part of the main chamber

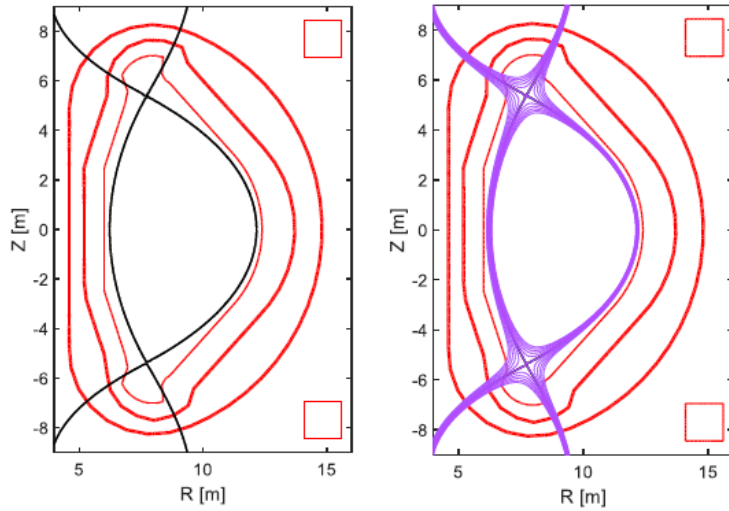


Figure 1.20: Plasma separatrix in a DEMO Double Null configuration.

The presence of a second X-point involves the need of a second divertor in the upper part of the chamber, allowing the wall interaction area to be doubled and halving the heat load reaching the divertor (compared to the SN case) [Wesson, 2004] but also halving the connection length to the target. It also allows a more triangularly shaped plasma to be obtained, which is helpful in obtaining high β and high energy confinement.

1.6.2. X-Divertor (XD) configuration

The XD was introduced in 2004 [Wesson, 2004]. In Figure 1.21, a few of the representative 2004 XD configurations from the original paper are displayed; (a) NSTX and (b) ITER.

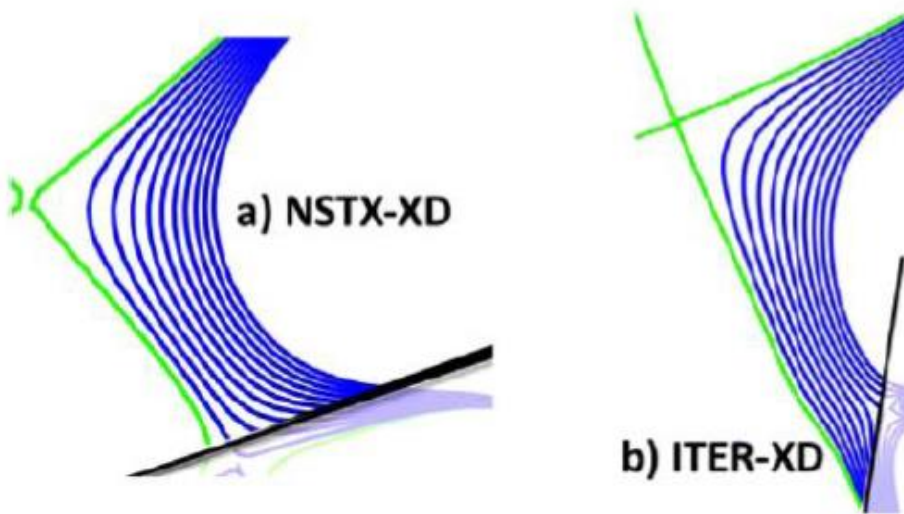


Figure 1.21: Flux plots of the 2004 XD equilibria, regenerated using CORSICA for (a) NSTX and (b) ITER. To emphasize the physically relevant regions where power is exhausted, only SOL field lines are shown in bold.

The most important characteristics of the XD are listed below:

- a) The XD configuration is created by “inducing a second axisymmetric X-point downstream of the main plasma X-point.”
- b) The beneficial result is that “field line lengths from the core X-point to the wall can be increased and flux expansion can be increased.” The physical consequence of these characteristics was predicted to be a reduced heat flux on the divertor plate.
- c) The new X-point produced a new geometry in the SOL flux surfaces. The flux surfaces flared outward, rather than contract inward as in a Standard Divertor (see Figure 1.22).
- d) The name X-Divertor succinctly describes the physical essence of the configuration - the downstream SOL interacting with a new X-point.

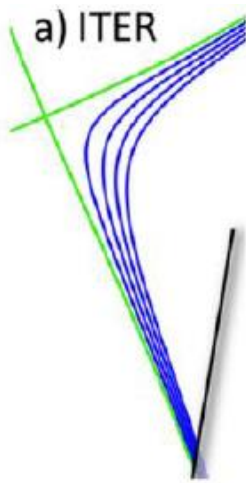


Figure 1.22: Standard Divertor flux surfaces, for ITER equilibria generated with the free boundary code CORSICA. Standard Divertors have convergent flux surfaces

1.6.3. Snowflake Divertor (SFD)

The SFD family, built around the basic configuration (to be called pure Snowflake) that creates a second order null at the main plasma X-point, was introduced in 2007. Such a second order null in the core X-point produces a six-fold symmetry in the magnetic field within the divertor region, leading to the succinctly descriptive name “Snowflake.” The pure Snowflake was complemented by two variants “plus” and “minus” in the first publications on the subject [Ryutov, 2007] and [Ryutov, 2008] (see Figure 1.23).

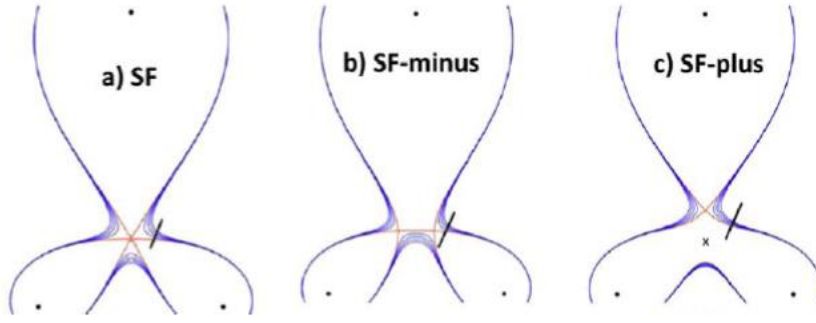


Figure 1.23: The exact Snowflake (SF), the Snowflake minus (SFm), and the Snowflake plus (SFp) configurations from 2007–2008 [Ryutov, 2007] and [Ryutov, 2008]. The flux expansion is greatly increased in the region near the core X-point, but rapidly decreases downstream so that the SOL is strongly convergent (even more than for a Standard Divertor).

These variants do not have one second order null; instead, they have two first order nulls (like the XD). The two nulls are placed relative to each other so that the advantages of the pure Snowflake, as the enormous flux expansion and increase in the line length in the vicinity of the two nearby X-points, may be maintained while its major problem, namely the instability to small perturbations, is overcome.

Unlike the X-Divertor, in the Snowflake Divertors the flux surfaces in the power exhausting SOL region are all convergent in Figure 1.23.

1.6.4. Super-X Divertor (SXD)

In the same year as the Snowflake (2007), a more advanced variant of the XD, named the “Super-X” divertor (SXD, see Figure 1.24), was presented; the latter reaches even a greater flux expansion by superposing toroidal expansion (by placing the divertor plate at the maximum possible major radius) on the poloidal flux expansion [Kotschenreuther, 2004]-[Valanju, 2008]. The name reflects the intended function: modification of the X-Divertor concept for “superior” flux expansion and line length.

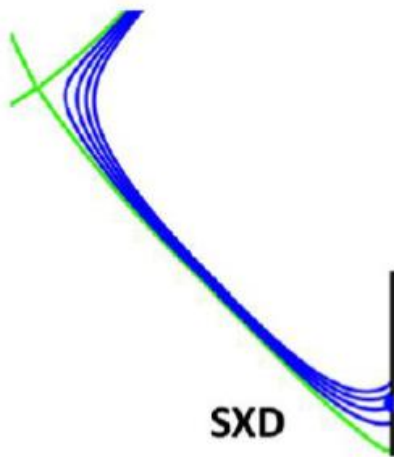


Figure 1.24: SXD with both poloidal and toroidal flux expansions near divertor plate at large major radius R .

2. Plasma Facing Component (PFC) in Solid Divertors

Thermal power density absorbed by the PFCs could get up to hundreds of MW/m^2 , which is definitely an unbearable value for any material. In order to prevent this drawback it is necessary to increase as much as possible the surface exposed to the heat flux by means of the targets' inclination, expansion of magnetic field lines and strike points sweeping.

PFCs consist in: an armour material facing the plasma (where beryllium and tungsten are the most attractive armour materials); heat sink (high thermal conductivity and adequate mechanical strength) to transfer the heat from the armour to the cooling channel that should have adequate thermal conductivity and good resistance to water corrosion (see Figure 2.1). A full rationale for the use of the different materials is dealt with in [Zinkle, 2015]

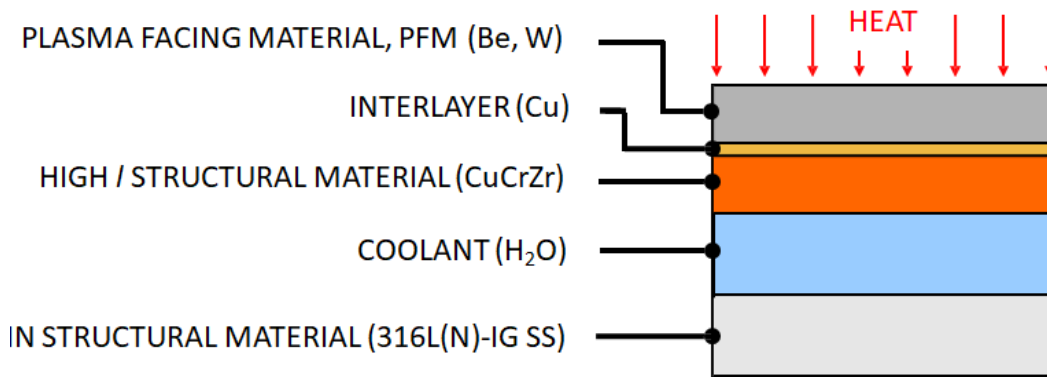


Figure 2.1: Basic concept of PFC

In the Figure 2.1 a typical choice of materials for this component is shown. The materials that have to be used to realize the PFC's need to withstand high heat fluxes and to absorb the heat without damage nor melting. Indeed, in this way no undesired impurity will be introduced in the plasma by their erosion, which would cause the decreasing of plasma purity and would increase the risk of sudden losses of confinement (disruptions).

Therefore, the surfaces directly exposed to the PFCs plasma, the armour materials, need to be realised respecting the following:

- Heat and erosion resistant materials;
- Materials exhibiting a high thermal diffusivity, in order to transmit the heat to the cooling fluid;
- Materials which are not easily damaged by neutrons and low activation;
- In the shape of little tiles (monoblocks), to minimise thermal stress ;

- Accurately polished and aligned surfaces to avoid leading edges to be directly exposed to the plasma;

Basically tungsten and beryllium are the materials used because satisfy enough these conditions. In the past, also Carbon Fiber Composite (Figure 2.2) materials reinforced with carbon fibers belong to this category apart from the fact that they cannot be used with tritium because they tend to retain it.



Figure 2.2: Small scale prototype of plasma facing component with straight part in Carbon Fiber Composite

2.1. ITER Divertor

The inner and outer vertical targets consist of monoblocks with swirl tape in the coolant channel and their lower parts are defined as the high heat flux handling areas, whereas, the dome consist of flat tiles with hypervapotron cooling channel and is exposed to lower heat flux.

The inner and outer vertical targets (IVT and OVT) are positioned at the intersection of magnetic field lines where particle bombardment will be particularly intense in ITER. As the high-energy plasma particles strike the vertical targets, their kinetic energy is transferred to heat and the heat is removed by active water cooling.

The heat flux expected at the strike point of the ITER divertor vertical targets is estimated at 10 MWm^2 (steady state) and 20 MWm^2 (slow transients). Tungsten, with the highest melting point of all the metals, has been chosen as armour material following an international R&D effort, encouraging experimental results, and successful prototype testing.

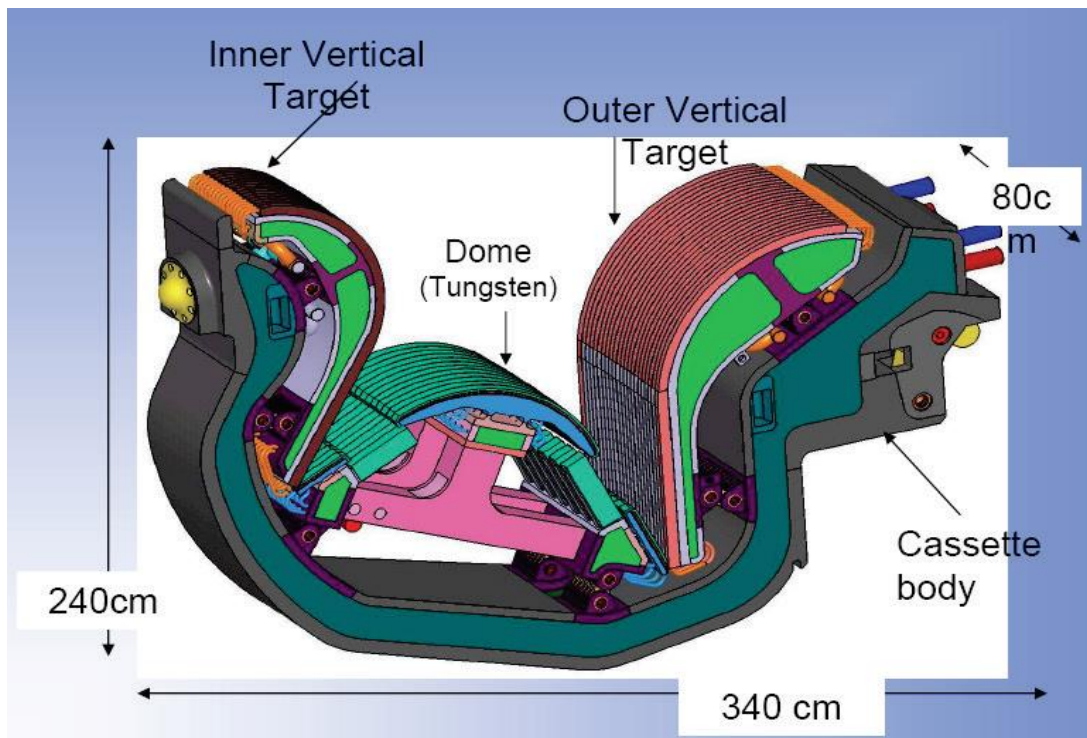


Figure 2.3: Model of ITER Divertor

The most delicate components are those on the vertical target surfaces, which in principal receive the highest heat flux of all PFCs. Features are also included for exhaust of neutral gas impurities to the vacuum pumping system.

In Figure 2.4 a model of monoblock for ITER divertor are showed.

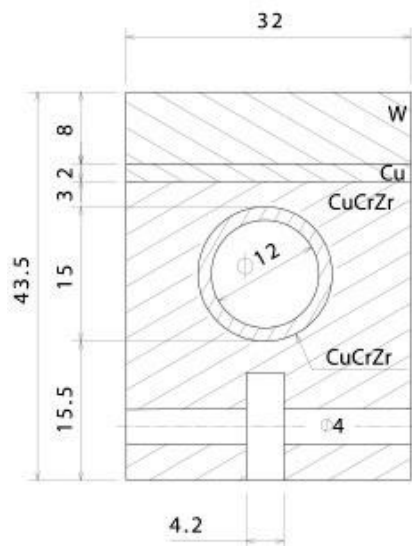


Figure 2.4: Section of a W flat tile joined to the CuCrZr heat sink with a Cu OFHC interlayer between them

The joining technologies used are the following:

- Brazing** is a process for joining similar or dissimilar materials using a filler metal

- The filler metal is melted so it is chosen according to process and working temperatures, capability to wet the two materials, mechanical strength of the final joint;

- A properly-made joint will be in many cases as strong or stronger than the based materials being joined;

- The residual stresses are usually rather high due to the CTE mismatching and to the brazing temperatures

- Casting**: Pouring molten metal into a mold to produce an object of desired shape.

- The pure metal is cast directly on the surface to join;

- The final joint interface is located between the solidified metal and the material to be joined;

- The residual stresses are usually very high due to the CTE mismatching and to the casting temperatures;

- Active Metal Casting (AMC Plansee)**: To realize copper to Carbon/Carbon joints PLANSEE SE has developed the active metal casting (AMC) process. Following this process, after laser micro-machining of the interface, copper is cast onto a laser structured C/C surface in the presence of activating elements (i.e. Ti, Si), which results in a characteristic sound joint;

- Solve the wetting problem of copper onto CFC by Ti activation;

- The joint has an high strength and in the CFC case is as strong or stronger than the based materials being joined;

- The residual stresses are usually rather high due to the CTE mismatching between Cu and CFC.

- Pre-brazed casting (PBC ENEA)**: To realize copper to Carbon/Carbon joints ENEA has developed this process. A multithreading machining of the CFC blocks is performed before the treating of the ceramic material (CFC) by 'brazing' with a commercial alloy. The Cu-OFHC is cast later in the hole to obtain the Cu//CFC joint

- Solve the wetting problem of copper onto CFC by a commercial alloy

- The joint has an high strength and in the CFC cases is as strong or stronger than the based materials being joined

- The residual stresses are usually rather high due to the CTE mismatching between Cu and CFC.

- Electron beam welding (EBW) is a melting welding process in which a beam of high-velocity electrons is applied to the materials being joined. The workpieces melt as the kinetic energy of the electrons is transformed into heat upon impact, and the filler metal, if used, also melts to form part of the weld. The welding is often done in conditions of a vacuum to prevent dispersion of the electron beam;
- Can be used only for metallic materials;
- Almost all metals can be welded by the process
- Total heat input is actually much lower than that of any arc welding process
- Distortion is slight.

2.2. DEMO Divertor

PFC of the DEMO divertor is composed by tungsten armour monoblocks surrounding a soft copper interlayer, needed to mitigate the different thermal expansion and facilitate the diffusion bonding, and finally a structural copper alloy (CuCrZr) cooling pipe.

The divertor PFUs are cooled using water at 150 °C, 5 MPa and velocity between 12 and 16 m/s. Water cooling in this regime gives excellent heat transfer performance, as shown in many text in literature such as [Federici, 1997], where different correlations are used up to the fully developed subcooled boiling regime, such as the one from [Thom, 1965]. An additional turbulence enhancer is added inside the cooling pipe in the form of a twisted tape, in the regions where the maximum heat flux capability of the divertor target is needed, i.e. where the plasma intercepts the PFC, named strike points. For very high performances, low temperature water is used, due to the need to sustain steady state heat fluxes of the order of 10-20 MW/m². This is done to avoid reaching the Critical Heat Flux (CHF), with the departure from nucleate boiling (DNB) in which steam bubbles no longer break away from the solid surface of the hottest part of the pipe, and bubbles dominate, and, as a consequence, the heat flux dramatically decreases. Vapor essentially insulates the bulk liquid from the hot surface, leading to an extremely reduced capability of the cooling liquid to reduce the heat flux, which brings as a likely consequence the damage of the PFC system. While the coolant temperature in the divertor is too low for efficient power conversion, it is anticipated that the heat exhausted in the divertor will be useful to the power cycle via a pre-heater stage [Barucca, 2018].

CuCrZr is selected as it is unrivalled in its combination of thermal conductivity and strength. However, under irradiation it suffers embrittlement at ~150-200 °C while softening and creep

occur at $\sim 300\text{-}350\text{ }^{\circ}\text{C}$, thus it may only be useful for doses up to 5 dpa [Fabritsiev, 1997]. However, as the divertor is a low fluence region (2-3 dpa/fpy), 2 fpy operation of the divertor is thought to be feasible. Tungsten is considered the most suitable armour material due to its refractory nature, good conductivity, very low sputter yield and low tritium retention. At least 5 mm of tungsten thickness is thought to be required at the plasma facing surface, to resist an erosion of $\sim 2\text{ mm/fpy}$.

Although low activation is desired, at present the need for a viable solution for the divertor overwhelms this. Copper would not be low activation, but the relatively small volume of waste arising from the target plates is taken to be acceptably low [Visca, 2016].

An integrated program of conceptual design activities for a European DEMO reactor was launched in the framework of the EUROfusion Consortium, in which the capability of reliable power handling was identified as one of the most serious technological and scientific challenges for commercially viable fusion power.

A dedicated R&D work package ‘Divertor’ (WPDIV) has been set up in the EUROfusion Consortium in order to develop the conceptual design solution for the DEMO divertor system together with the related technology. The objective of the WPDIV is to assure sufficient power exhaust capability and structural reliability of the DEMO divertor against the stationary and transient operational scenarios under cumulative neutron dose of up to 13 dpa. The envisaged lifetime of a DEMO divertor cassette was defined to be two full power years (fpy). The final conceptual design report is due by end of 2020. Fig 2.5 illustrates the work breakdown structure of the WPDIV. The WPDIV consists of two subproject areas: ‘cassette design & integration’ and ‘target development’.

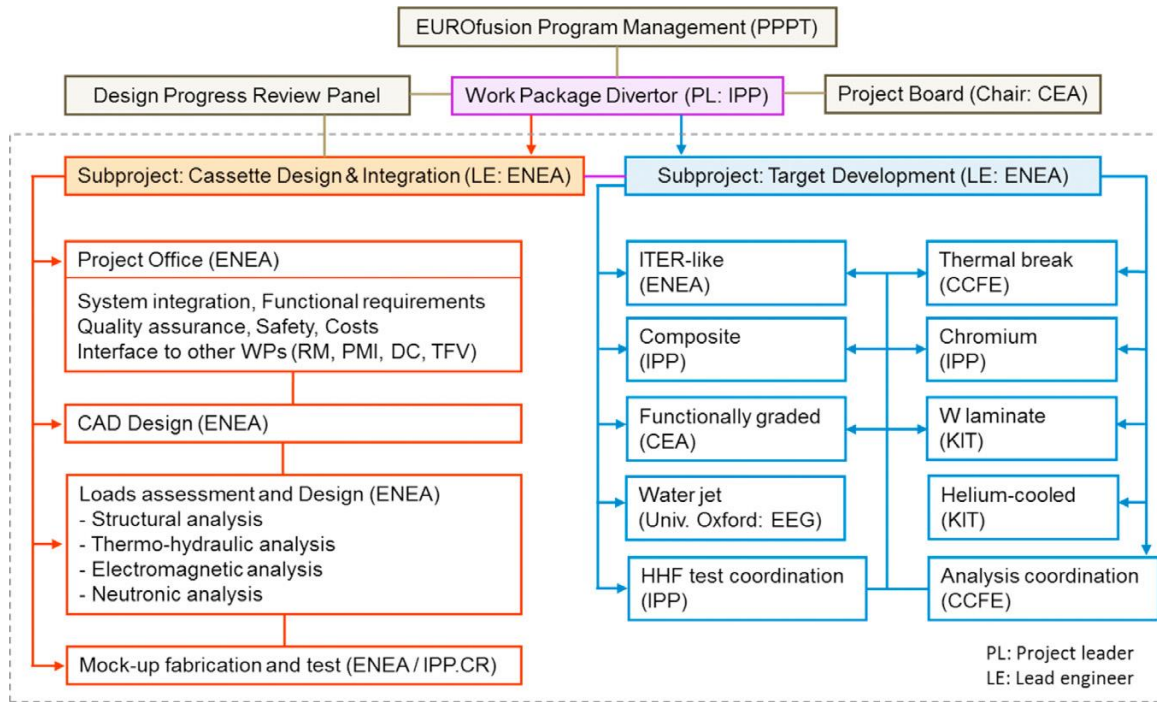


Figure 2.5: Schematic project breakdown structure of the EUROfusion work package 'Divertor' [You, 2018]

Target Development is EUROFusion activity related to the design and optimization of these components. Advanced configurations and different materials have been proposed by different research centers. The purpose of the activity is to compare the different configurations under high heat flux. A comprehensive overview of the methodology and the facility used withing the European program is reported in [You, 2018].

The CEA proposes a monoblock design equipped with an interlayer consisting of a few microns of functional gradient material that gradually turns from copper to tungsten.

The CCFE proposes a "thermal break" model in which small holes are drilled in the pure copper interlayer with the aim of distributing the heat flow entering the refrigerant fluid over a larger surface of the pipe.

ENEA proposes an ITER-like model constituted as for the divertor of the ITER machine. A PFU consists of a copper alloy tube (CuCrZr) armoured with tungsten and a pure copper interlayer interposed among them. The pure copper applied inside the hole of the monoblock is obtained by casting, while the interlayer-tube joint occurs in the solid state through a welding process properly set and patented in ENEA laboratory.

All research centers participating in this activity have produced six small-scale mock-ups representative of the proposed project. All mock-ups are subject to non-destructive tests before being tested for thermal fatigue. During this fatigue test, the mock-up is cyclically

loaded by an electron+ gun under high heat fluxes. This test is carried out under vacuum and active cooling is ensured by water flowing inside the pipe of the PFC.

2.2.1. Monoblock with a Thermal break interlayer

One such alternative divertor concept is the Thermal Break [Barrett, 2015]. The main benefit of this concept is to help mitigate the heat flux concentration at the upper part of the tube by use of a thermal barrier interlayer inserted between the W monoblock and the Cu alloy tube. In the presence of the thermal break layer, the heat flux is diverted from the upper part and bypasses around the tube through the armor. The peak heat flux and the maximum temperature at the tube is reduced while the W monoblock is heated more uniformly.



Figure 2.6: Thermal break concept

For the thermal break layer a porous Cu material made of felt or foam can be used as a variant of drilled holes. A significant side effect is that the porous thermal break interlayer has such a low stiffness that it interrupts the load transfer at the bond interface reducing the thermal stress concentrated around the tube. This mechanical break gives a beneficial effect.

2.2.2. Monoblock with a wire-reinforced composite tube

The high tube temperature values under slow transient loads suggest that loss of strength by high temperature and irradiation will be a critical design issue in terms of structural reliability. The composite heat sink utilize the powerful strengthening effect by use of hard tungsten reinforcements in form of either wires or particles. It is well known that W wire reinforced Cu composites exhibit much higher yield stress and rupture strength than matrix [You, 2015].

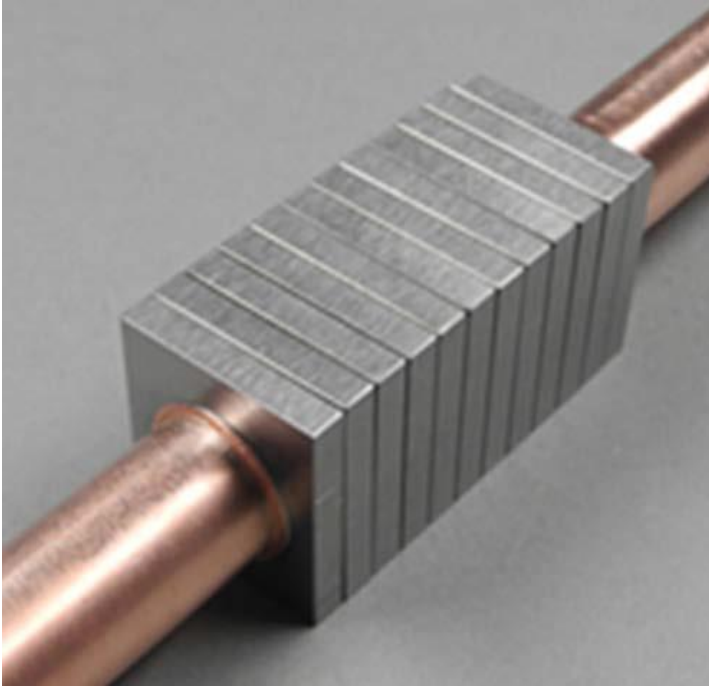


Figure 2.7: Cooling pipe composite

2.2.3. Monoblock with a functionally graded interlayer

In the conventional ITER-type target design the armor and the tube are joined via a soft Cu interlayer. The role of the Cu interlayer is to reduce thermal stress by plastic yield enhancing structural integrity. However, there is a concern for the Cu interlayer due to the deteriorative irradiation effects such as swelling and embrittlement by transmutation [Fabritsiev, 1997]. Furthermore, the Cu interlayer was shown to suffer from plastic fatigue under HHF loads [Pintsuk, 2015]. A target design without thick Cu interlayer was investigated. In this concept, W monoblocks are joined to the CuCrZr tube via a thin bond coat made of compositionally graded W/Cu layer. The thickness of the bond coat was set at first 17 μm , but variants with larger thickness will be also tested.

A picture of a coated tungsten block is shown in Fig 3.8. Test mock-ups can be successfully manufactured by an industrial process using hot isostatic pressing technique.



Figure 2.8: tungsten monoblock with a hole before joining to a tube. The inner wall was coated with a graded W/Cu bond coat

2.3. Divertor prototype Plasma facing unit

Below are the images related to the manufacturing of prototypes for the divertor parts (Fig. 2.9-2.12).

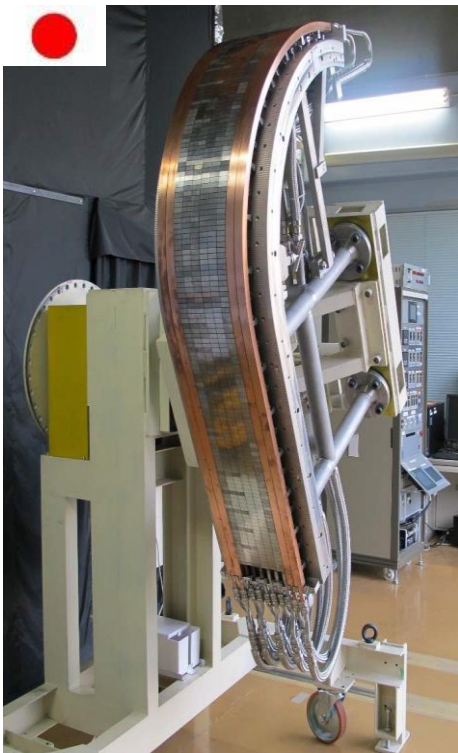


Figure 2.9: Full-scale prototype Tungsten Plasma-Facing Units (PFUs)



Figure 2.10: Full-scale tungsten IVT PFUs



Figure 2.11: Full-scale Dome Umbrella



Figure 2.12: Full-scale Dome particle reflector plate

3. Hot Radial Pressing (HRP) facility for fabrication of ITER vertical target divertor

The ITER Divertor Inner Vertical Target (IVT) plasma-facing components have been manufactured in a European environment which has seen ENEA and Ansaldo Nucleare S.p.A. deeply involved. [Visca, 2005].

3.1. Manufacturing process

The divertor targets and dome consist of a steel support structure on which the so-called plasma facing units (PFUs) are mounted. The PFUs represent the plasma facing unit. They are protected by the armor exposed to plasma and alpha particles. The armor is made out of W monoblocks with a drilled hole. Into this hole the heat sink is inserted, which is a precipitation hardened copper alloy (IG-CuCrZr) cooling tube (Figure 3.1).



Figure 3.1: W monoblock with Cu interlayer

A copper-OFE interlayer is interposed between the armor and the tube in order to mitigate stresses during operation and testing phases. The properties that encouraged considering W as the armor material for ITER are:

- its low sputter yield;
- low tritium retention;
- high melting temperature;
- good thermal conductivity.

During normal operations a heat flux of 5–10 MW/m² is deposited at the bottom segment of the IVT. However the capability to remove up to 20 MW/m² during the transient events (“slow transient” phase) of 10 s needs to be provided. The manufacturing process of these

crucial units is mainly divided into two phases: firstly the monoblocks have to be prepared, casting the Cu-OFE interlayer inside the machined hole in the W monoblock that will host the tube, followed by the non-destructive (ND) control and machining of the hole to the final diameter of 15 mm. Afterwards, the monoblocks have to be joined to the CuCrZr tube. The final joining between monoblocks and CuCrZr tube is performed by Hot Radial Pressing (HRP) [Visca, 2005],[Visca, 2007]. The increase of object size to be manufactured, from medium scale Qualification Prototype (QP) to full scale, requested the modification of tools and equipment used for the HRP.

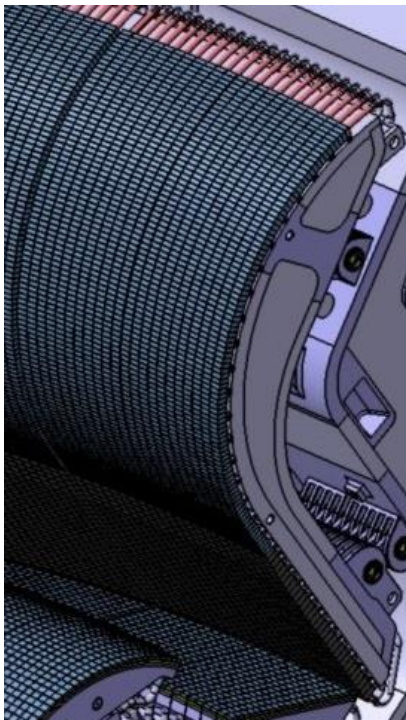


Figure 3.2: 3D view of the full-W full-scale IVT prototypes.

ENEA in collaboration with the equipment supplier (T.A.V. S.p.A.-Tecnologie Alto Vuoto, Caravaggio IT) designed a new facility, able also to host the full-scale prototype of a PFU and to control the high stresses during the pressurizing phase of HRP (about 60 MPa).

The primary objective of this work was to verify the possibility to obtain, after the Hot Radial Pressing (HRP) process, the PFUs within final shape and tolerances (17 Micron). In this case no additional machining or finishing of the PFUs would be requested to meet technical specifications. To this aim, a new HRP equipment, with internal heating, and a particular internal steel structure (named “holder”) keeping in place the PFU, was designed and installed in the ENEA lab. The basic principle adopted during the design was to leave the PFU as much as possible in a “stress free” state during the heating and cooling phase of HRP.

Considering the complex geometry of the PFU and also its elongated dimension, the temperature control is regarded as crucial. Therefore the heating system was divided in 16 different zones individually controlled and monitored. The internal pressure was applied only after the required temperature uniformity was reached. At this point, the PFU and its holder were frozen in the reached position. Special pneumatic stoppers avoided any radial translation, but also the axial elongation of the tube was not allowed by external pistons acting on both tube ends. In Figure 3.3 is shown a PFU after HRP process.



Figure 3.3: The full-scale full W PFU after HRP

3.2. UT controls and dimensional results

In order to check the joining interface between the tube and the monoblocks a non-destructive test has been carried out in the ENEA lab by a dedicated UT system developed internally and widely tested based on the water gap ultrasonic pulse-echo technique. In the final control of the PFU no defect was detected and a report was issued. A specific dimensional check procedure has been supplied for a proper geometrical check of the PFUs according to reference drawings. After the visual inspection, the PFU was placed onto a measurement jig, realized for the purpose.

The PFU was supported in different points by means of ball transfer units that allowed the PFU to remain in its “stress free” position by minimizing the friction on the horizontal directions. Then, in order to proceed with the measurement operations, the extremities of the PFU were locked. A laser tracker system was used for the measurements of the radius of the external surface of the curved zone, straightness of the straight part and planarity of the monoblocks on lateral surfaces.

The most relevant dimensions are the radius of the external surface of the curved zone and the planarity of the lateral surfaces by monoblocks width families. They both were verified comparing the points of the surfaces acquired by laser tracker with the reference model, by means of best fit data management techniques.

The UT system is composed of a helical flexible shaft of the probe with resolution and dimensioning of defects up to 0.5 mm.



Figure 3.4: UT-System by ENEA

The products of the ultrasound scan are visible in the A-scan and in the C-scan.

The A-scan is the representation of the amplitude of the rectified signal in terms of percentage of the screen as a function of time in a specific point.

The c-scan is a 2d map (Fig 3.5) of a 'slice' of the material cut along orthogonal surfaces to the beam direction. The color of the points in the map represents the maximum value of the signal amplitude in the selected time interval for each point.

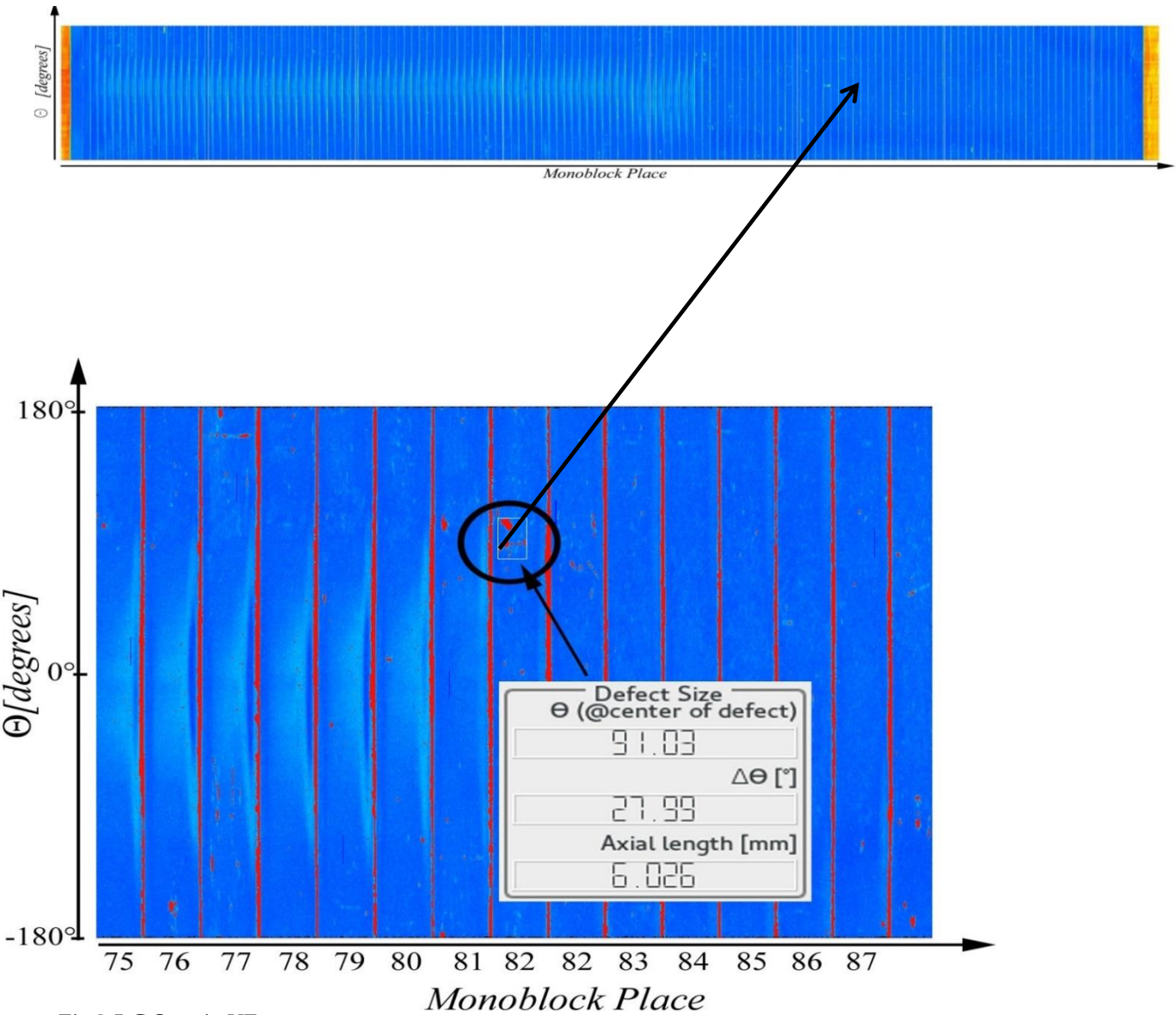


Fig 3.5 C-Scan in UT system

4. Thermal-mechanical analyses in DEMO divertor

During tokamak operations, different types of high energy ionized particles are produced and leave the plasma after the confinement time. The particles escaping from the plasma enter the scrape off layer (SOL) region and reach the Divertor region. The particles reaching the Divertor region cause a localized thermal load around the strike-points, i.e. the intersections of the separatrix with the divertor. In order to spread on a larger region this thermal load, it is convenient to resort to a periodical movement of the strike-points [Ambrosino, 2008]. The different techniques providing this movement can be classified in:

- “*strike point sweeping*”, if it generates a movement of the only region of plasma boundary close to the X-point keeping the rest of the shape practically unchanged;
- “*wobbling*”, in which the whole plasma is displaced periodically in a rigid way.

In this Chapter the possibilities to implement a 3D model of the DEMO divertor target tiles are explored and the results of a thermal analysis carried out are presented.

Main European-DEMO design parameters and requirements that impact the divertor design have been defined from the ‘Work Package Divertor’ (WPDIV) project [Ambrosino, 2008]. Extensive design-to-code study has been then conducted at ENEA for an ITER-like target design under DEMO-relevant loading and cooling conditions. The current consolidated materials, concept design and fabrication process of ITER target divertor was assumed: W as armour material, Cu-OFHC as stress-relieving interlayer joined to W by casting process, precipitation-hardened CuCrZr-IG (ITER Grade) alloy (solution annealed aged cold worked) as heat sink material joined to the monoblocks by HRP (Hot Radial Pressing) process. The hydraulic parameters used in FEA consist of a water coolant pressure of 5 MPa and operating temperature of 150 °C, in order to work with a reasonable safety margin (50–80 °C) with respect to the max working temperature before reaching the critical heat flux. The coolant velocity is 16 m/s to enhance the heat transfer coefficient, despite the consistent pressure drop. A swirl tape (thickness 0.8 mm and twist ratio 2) is also involved to enhance the power exhaust capabilities. Steady state heat flux of 10 MW/m² during normal operation and up to 20 MW/m² for repeated slow transients have been applied

The parameters taken into account for the optimization were: the coolant tube inner diameter from 10 to 14mm; the tube thickness from 1.5mm to 2mm. The pure copper interlayer was varied from 0.5mm to 1.5mm while the W block side dimension from 3 to 5mm. The armor thickness from the surface to the interlayer was fixed at 5 mm. The influence of the varying

armor thickness on the thermal and structural performance was studied elsewhere. The optimized dimensions resulting from the FEA analysis are the following: tube inner diameter and thickness: 12mm and 1.5mm respectively; interlayer thickness and armor side thickness: 1mm and 3mm respectively.

Particular attention was spent in modelling boundary conditions reproducing the attachment system of plasma facing unit to cassette. By means of a 3D FEM analysis, the geometry of the mock-up was optimized in terms of thermo-mechanical behavior by minimizing maximum von Mises stress in tube and W surface temperature [You, 2016, Li, 2016]. Optimized mock-up geometry was identified and assessed in a 3D CAD model by ‘Monoblock elastic analysis procedure’ (MEAP) [Furdson, 2016].

This work showed that the ITER-like target with an optimized geometry could pass the elastic structural design criteria of the ITER SDC-IC code. Indeed, it turned out that the cooling tube of the ITER-like target could just pass the 3Sm criterion of the C type (cyclic) load category indicating that progressive plastic deformation (ratchetting) might be the most critical failure mode. On the other hand, the low cycle fatigue (LCF) lifetime of the tube, which was predicted using S-N curve data, was estimated to reach at least 40000 load cycles.

4.1. Impact of the Monoblock Thickness in an ITER-Like Configuration of DEMO Divertor

In the present chapter, main findings of our study, recently published in FED [Maviglia, 2018] and related to thermo-mechanical analyses on DEMO divertor, will be described in detail. The thermal-mechanical analyses carried out on the DEMO Divertor tungsten monoblocks showed that the fatigue lifetime of the CuCrZr tube and copper interlayer, under cyclic thermal loads, depends on the axial thickness of the W. One of the monoblocks thickness currently considered for plasma facing component of the DEMO divertor is 4mm. This configuration is compared with ITER design, where the W thickness is 12 mm. Both configurations have been simulated in ANSYS Workbench environment. In this work, rather than a single monoblock, an entire plasma facing unit (PFU) is analyzed with its constraints to the cassette. The FEM model consists of a CuCrZr tube 60mm long, corresponding to the distance between two successive support attachment to the cassette as well in the current ITER divertor design. Between the W monoblocks and the tube 1 mm of pure copper interlayer is present. Appropriate symmetry conditions allow to simulate the entire Mock-Up. The pipe and interlayer materials have elasto-plastic behavior, while the monoblock W is

considered to have linear elastic properties. In this study the plastic fatigue lifetime of the CuCrZr tube and the copper interlayer has been estimated by applying a cyclic heat flux (20MW/m^2) to the surface of the tungsten armor.

On the basis of the most recent assessment [You, 2016], fatigue life under cyclic thermal load is an extremely relevant parameter for vertical target design of the future fusion reactors divertor. An analysis of Mock-Up fatigue behavior has been carried out.

DEMO target design, considered in this work, is ITER-like design: the targets are CuCrZr copper alloy cooling pipes, where pressurized water circulates, with W monoblock as armor.

The thermal-mechanical analyses carried out on the DEMO Divertor tungsten monoblocks have shown that the fatigue lifetime of the CuCrZr tube, under cyclic thermal loads, depends on the axial thickness of the tungsten [Li, 2016]. One of the monoblock thicknesses currently considered for plasma facing component of the DEMO divertor target, is 4mm. In this work, this configuration is compared with ITER design, where the W thickness is 12 mm. This kind of analysis has already been carried out in earlier works [Li, 2016], which consider only one block. The aim of this work is to test the results with a model that simulates the entire PFU bound to the support structure. The geometry of the monoblock in the plane orthogonal to the tube axis has been optimized [Crescenzi, 2015] in order to minimize the stresses in the structural material (CuCrZr tube), and thus increase fatigue life of the component. The results of these optimizations were used to choose the dimensions of the monoblock.

Furthermore, the results of thermal fatigue tests on target ITER-like mock-ups highlighted the occurrence of cracks in the tube axial direction of tungsten monoblocks when the load reached 20MW/m^2 . [Crescenzi, 2015],[Li, 2014]. The cause of these cracks is related to the recrystallization of tungsten subjected to high temperatures. The recrystallization temperature of the pure tungsten might be around $1300\text{ }^\circ\text{C}$ ([Nogami, 2017],[Zhang, 2015]). Based on the finite element analysis described in the following section, for monoblocks geometry, hydraulic condition and load considered, the recrystallized region could have a thickness between about 1.5mm, in the center and 2.5mm at the external part of the monoblocks. The mechanical characteristics of the tungsten subjected to a recrystallization worsen and the yield strength considerably decreases inducing plastic behavior in the upper part of the tungsten monoblock under high heat flux test (HHFT). Based on the HHFT results at 20MW/m^2 , the recrystallization of the upper part of monoblock and the cyclic load seem correlated with the onset of the crack. In any case, during qualification program of the tungsten ITER divertor vertical targets where 250 tungsten monoblocks were tested, about 30% were showed macro cracks in the tungsten and this appears due also to the tungsten manufacture process

Monoblocks from forged tungsten bars or rolled tungsten plates seem to have different behavior [Merola, 2015]. These aspects were not been faced in present work, but a recrystallized tungsten layer has been considered only for the purpose of analyzing the impact of the monoblock axial thickness in the plastic behavior of the W in terms of trend and not of absolute value of stress and strain. In this work, it has been considered a tungsten with physical and mechanical isotropic properties deducible from SDC-IC and monoblock upper layer with plastic behavior. For simplicity and comparison with other works (e.g. [Nogami, 2017]), a uniform layer of 3mm on top monoblock surface has been placed in FEM model.

4.2. Geometry and material

For the purpose of identifying the actual advantage of decreasing the monoblock thickness, different PFU models with same monoblock plan geometries, but different thickness, have been considered: one with 4 mm thick monoblocks and the other one with 12 mm thick monoblocks. The monoblock plan dimensions, that have been chosen, are the following and shown in Figure 4.1. Width of monoblock: 23 mm; armor thickness: 5 mm; side thicknesses of the armor: 3 mm; cooling tube inner diameter: 12 mm; copper interlayer thickness: 1 mm; copper alloy tube thicknes: 1.5 mm.

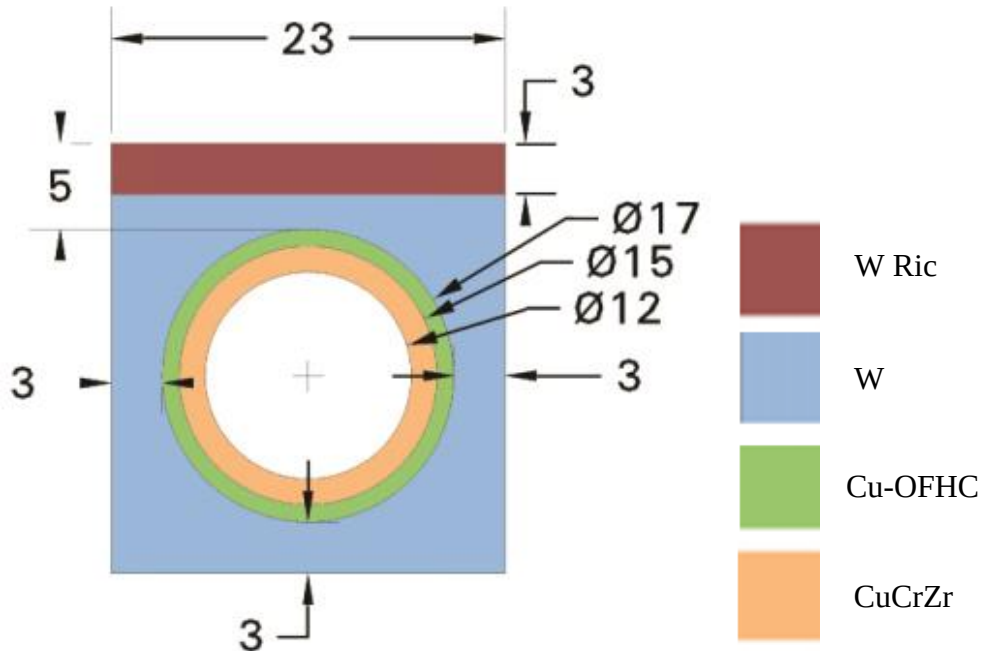


Figure 4.1. Monoblock dimensions and materials.

In this work, it has been chosen to model a PFU part of 62.5mm length, corresponding to distance between adjacent attachments to the cassettes in the current ITER divertor design. In

particular, the gap between 12 mm thick monoblocks are placed 0.5 mm as present ITER configuration; while in the model with 4mm thick monoblock a gap of about 0.2 mm are chosen. In this manner the loaded surface is the same and also the total power adsorbed form PFU. In advance, to check the gap dimension influence on the results, a model whit 4mm thick blocks and about 0.5mm gaps is considered (Figure 4.2). In this last model, load surface is less than for others, then thermal load has been increased up to 21.4 MW/m² to maintain the same total power adsorbed. The design of the ITER-like PFU consists of an array of tungsten monoblocks as armor, a CuCrZr alloy tube as heat sink and a thick soft copper interlayer between the cooling tube and the monoblocks.

Tungsten is considered a brittle material with linear behavior of the monoblock except for a 3 mm layer at the plasma facing surface which is considered ideally plastic. Bilinear isotropic hardening model are used for recrystallized tungsten and the stress/strain curve is showed in Figure 4.2. The other material properties are the same used for pure tungsten and come from the SDC-IC v3 [Structural Design Criteria For ITER In-Vessel Components SDC-IC, 2011].

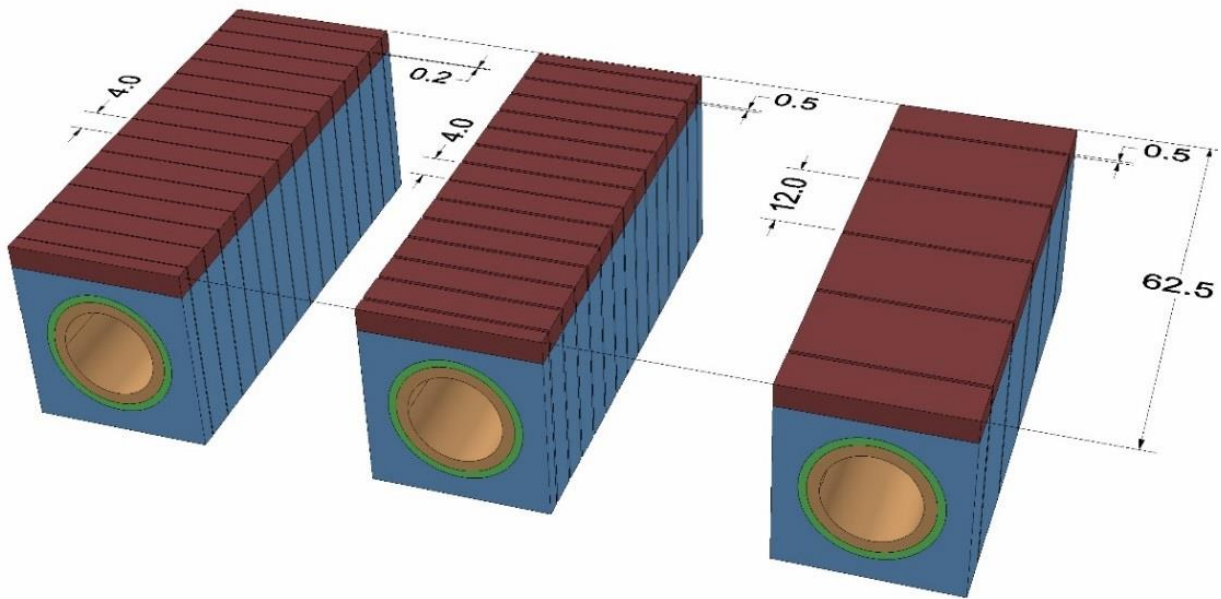


Figure 4.2: Model geometry: configuration with 4 and 12 mm thick monoblocks of axial length 62.5mm

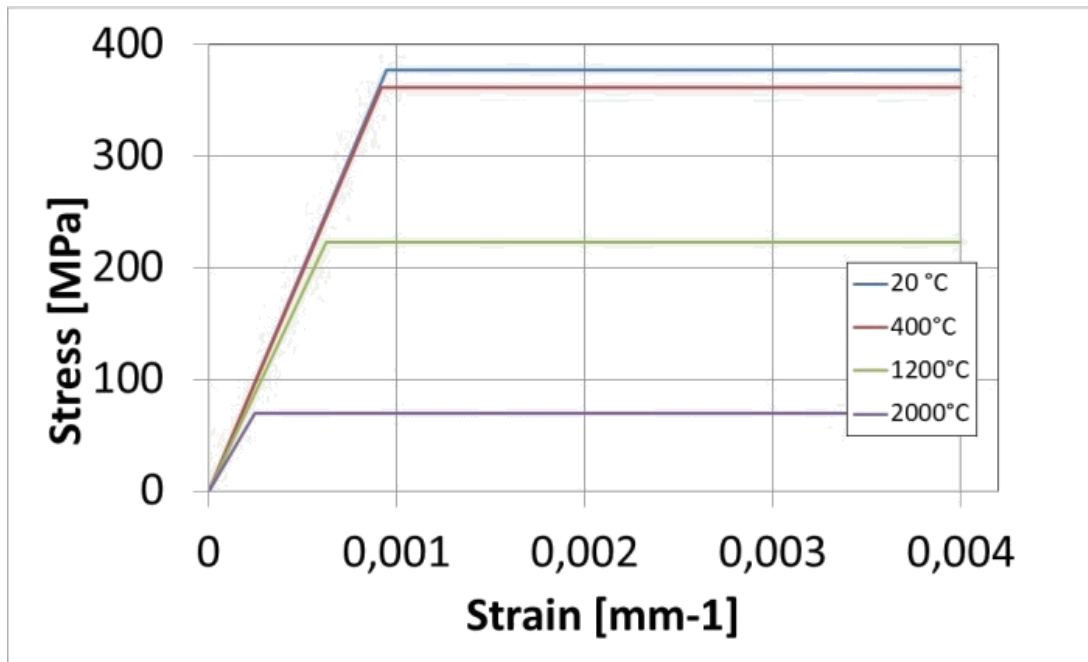


Figure 4.3: Ideally plastic recrystallized tungsten behavior

The tube CuCrZr and interlayer Cu OFHC are considered to be nonlinear materials. To describe their plastic behaviour a cinematic hardening Chaboche model was used. In particular, for the CuCrZr copper alloy a copper alloy in solution annealed and aged condition called treatment B was considered. The mechanical properties of the CuCrZr and Cu OFHC come from SDC-IC v3, while the Chaboche model coefficients come from [Fursdon, 2016] and are reported in table 4.1 and 4.2.

Table 4.1. Cu Chaboche $N=3$

Yield Stress MPa	Material Constant C1 MPa	Material Constant γ_1	Temperature C	Material Constant C2 MPa	Material Constant γ_2	Material Constant C3 MPa	Material Constant γ_3
45	40000	7500	20	11000	846	1000	55
18	10000	4000	400	2500	450	250	55

Table 4.2- CuCrZr- treatment B Chaboche N=3

Yield Stress MPa	Material Constant C1 MPa	Material Constant γ_1	Temperature C	Material Constant C2 MPa	Material Constant γ_2	Material Constant C3 MPa	Material Constant γ_3
180	3.e+005	4000	20	30000	825	6000	45
152	2.e+005	5000	350	30000	1000	6000	48

4.3. Boundary conditions and load

In order to reduce the number of finite elements, the symmetries of the PFU has been exploited with respect to the orthogonal plane to the surface facing the plasma, passing through the axis of the tube. In addition, all nodes of the surfaces at the extremes have been constrained to have the same displacement in the direction of the axis of the tube. This implies that the outer faces of the monoblocks at the ends cannot rotate, but may move in the axial direction. In this way an infinitely long tube has been simulated (Figure 4.4)

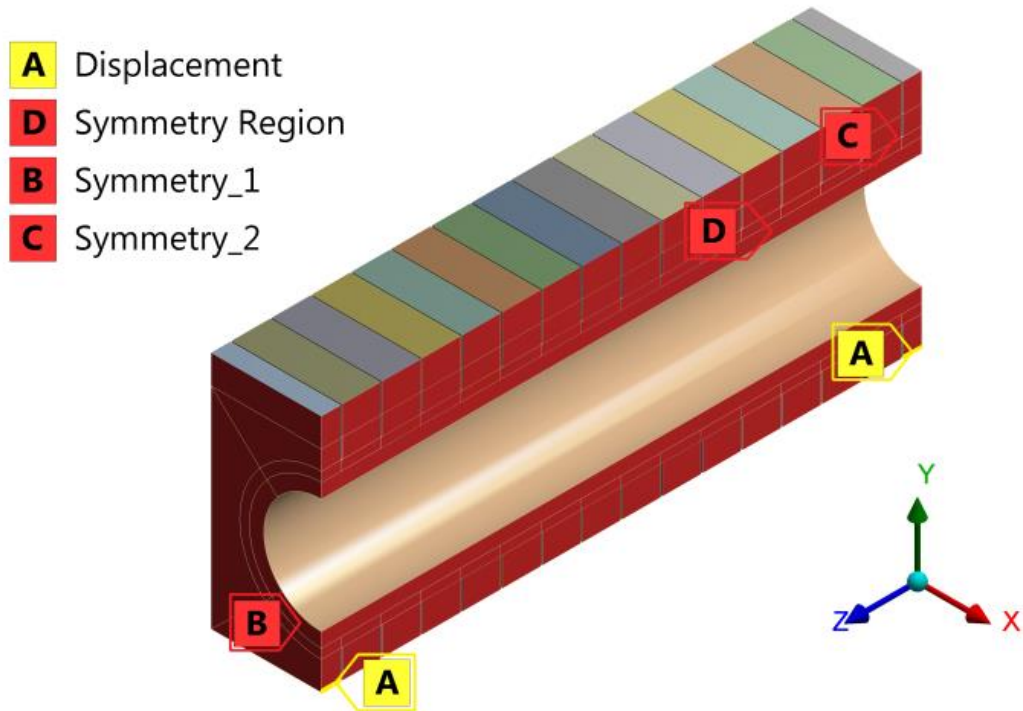


Figure 4.4: Symmetry and constraints of the model; A- the edges are constrained to move in z-direction only; B and C surfaces cannot rotate ; on the D-surface symmetry condition with respect to the orthogonal plane to the surface facing the plasma is imposed.

In the ITER PFUs, a monoblock with a brazed steel support, through which the PFU is constrained to the supporting structure, is placed between sequences of 4 monoblocks of 12 mm. The constraint is actualized through a cylindrical pin inserted in a slot that allows the PFU to move in the axial tube direction, but prevents the component to move towards the plasma. To simulate the presence of this constraint, the central edges of the extreme monoblocks have been constrained to have null displacements in the plasma direction (y-direction in Figure 4.4). The stationary analysis consists of 40 steps. In the first step, the component was cooled by the stress free temperature of 580 °C to ambient temperature, thus taking into account the residual stress due to the manufacturing process of the component. In particular, the HRP process [Visca, 2005] has been simulated, where the junction between the CuCrZr tube and the monoblocks occurs at 580°C. In the subsequent steps, the temperature field obtained from the solution of a coupled thermal analysis was cyclically applied. 20 load cycles have been simulated. For the thermal analysis it was considered: heat flux is 20 MW/m² (or 21.4 MW/m²), cooling water temperature 150°C and pressure of 4 Mpa, velocity 14 m/s, heat transfer coefficient of 160000 W/m²K.

Figure 4.5 shows the temperature obtained as a result of the analysis for the considered configurations. The temperature range for the configurations with the same load area are the same, while the configuration with 4mm thick monoblocks and 0.5 mm gap loaded with 21.4MW/m² on the top surface, exhibits higher temperature.

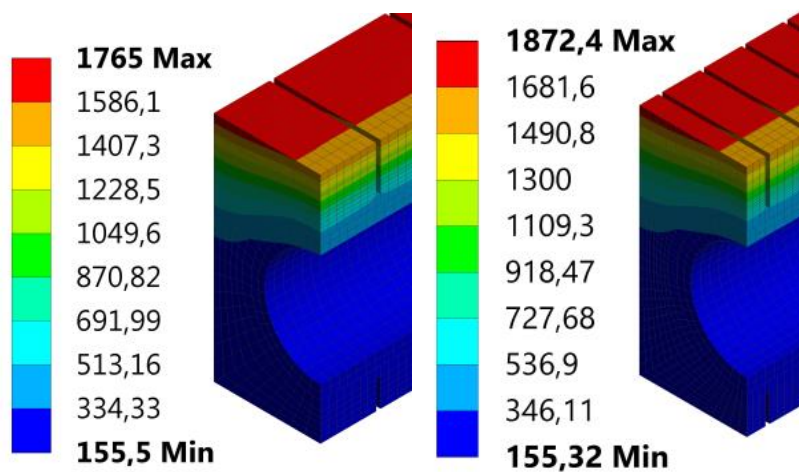


Figure 4.5: Temperature field (°C) in the of PFC assessed for 12mm thick monoblock and 0.5mm gap (left, same for 4mm thick monoblock and 0.2mm gap) and for 4mm thick monoblock and 0.5mm gap (right)

By imposing on this last model the same load of 20 MW/m², maximum temperature decrease to 1743 °C. However, such decrease does not have to be ascribed to the monoblock thickness, but to the reduced loaded surface with respect to exchange surface.

4.3.1. Mesh

Convergence analysis in terms of accumulated plastic strain in the tube has been conducted. The element number has been progressively increased with particular attention to the number of divisions in the tube and interlayer thickness. Furthermore, at least two elements in the copper interlayer between the monoblocks were inserted in the axial direction. A noticeable difference was appreciated in the transition from a mesh with 2 to 3 divisions. The difference decreases considerably, when passing from 3 to 4. Figure 4.7 shows some of the obtained results. Blue lines refer to the 12 mm thick monoblock: continuous line refers to the model with 3 thickness divisions of the tube and interlayer. The dot line refers to the model with 4 divisions. The node number in the first case is 260000, while in the second case it becomes 340000, nevertheless, the difference in results is negligible. Element and node number for each model are reported in table 4.3. The element type is SOLID186.

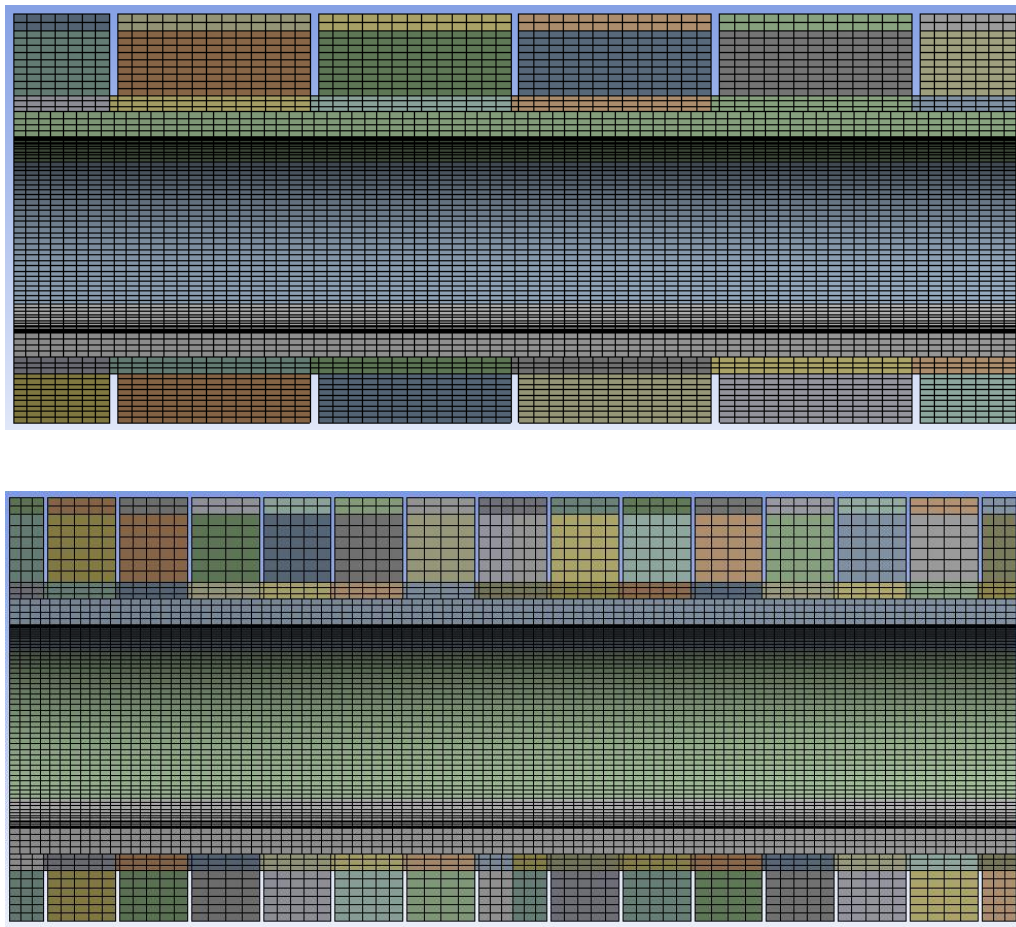


Figure 4.6: Finite element mesh for both configurations

Table 4.3 FEM mesh

	Element	Nodes
12mm	74740	342589
4mm, 15 blocks	103000	485362
4mm, 14 blocks	30680	150824

In Figure 4.7 are showed convergence analisys in term of equivalent plastic strain at the CuCrZr tube. Red and green lines refer to the 4 mm monoblocks and gap size respectively of 0.2 and 0.5. In particular, red lines show an increase of accumulated plastic strain with only 2 division in the tube thickness. For the calculation a mesh with 4 divisions inside the tube and interlayer thickness has been selected.

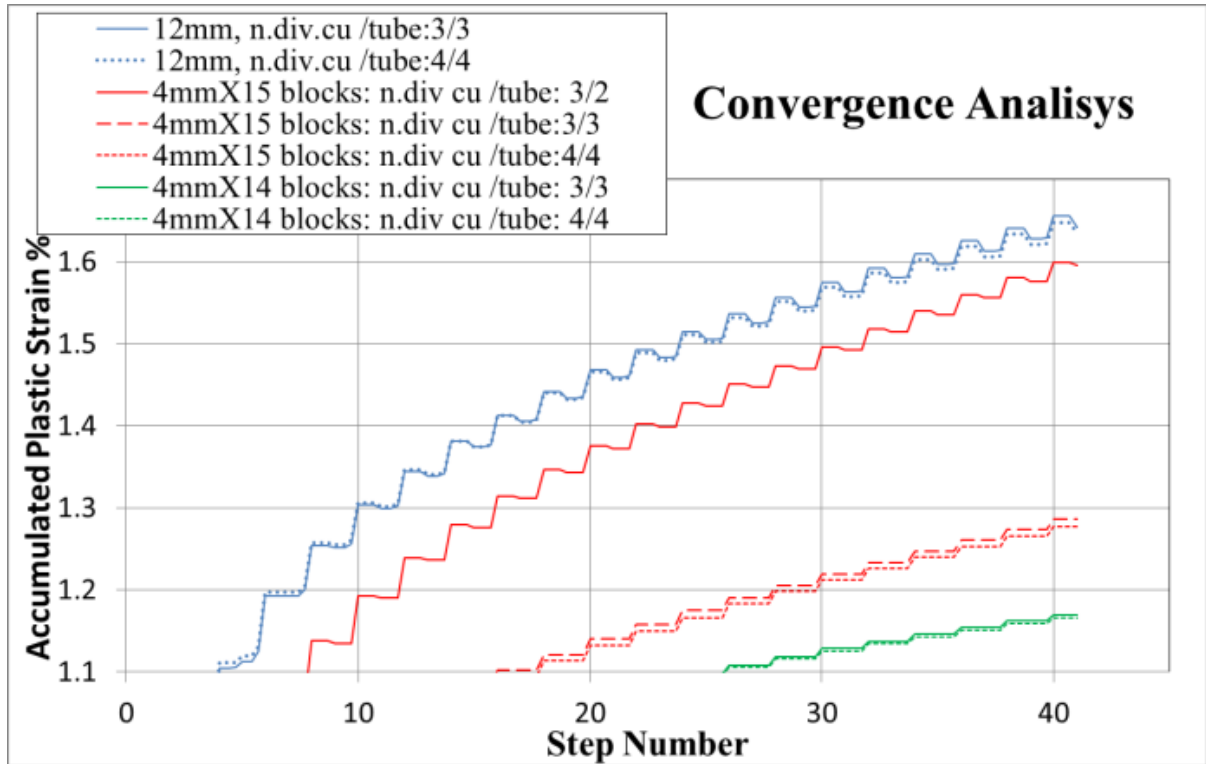


Figure 4.7: Convergence analisys in term of equivalent plastic strain at the CuCrZr Tube.

4.4. Results and conclusion

In the analysis thermal load and water cooling heat transfer coefficient are uniform: variation along axis tube are not considered. The results obtained with these axial uniformity are in tight agreement with those obtained with the model of a single monoblock when the symmetries and constraint are those used in [You, 2016]. Moreover, the cross section of tube and interlayer are constrained to remain in plan. This simplified model is therefore

appropriate to describe the actual system of constraints of PFU when local effects of load and heat exchange are neglected.

With respect to the tube, the Figure 4.8 shows accumulated equivalent plastic strain. It can be noticed that strain with selected mesh no longer show significant increases after 20 loading cycles. In addition, the greatest influence on accumulations of plastic deformation is given by the size of the block and not by the size of the gap. In this paragraph, the results for the model with 12 mm monoblocks and 0.5 mm gap and those of the model with 4 mm monoblocks and 0.2 mm gap will be described. For these models, since the thermal load and the temperature fields are the same (20 MW/m^2), any result differences can be attributed exclusively to the width of the monoblock.

The maximum equivalent and total plastic strain are reached at the centre of the monoblocks in the inner surface of the tube where the temperature is lower, for both the configurations. The maximum temperature reached on the tube is 465°C on the upper part of the external surface.

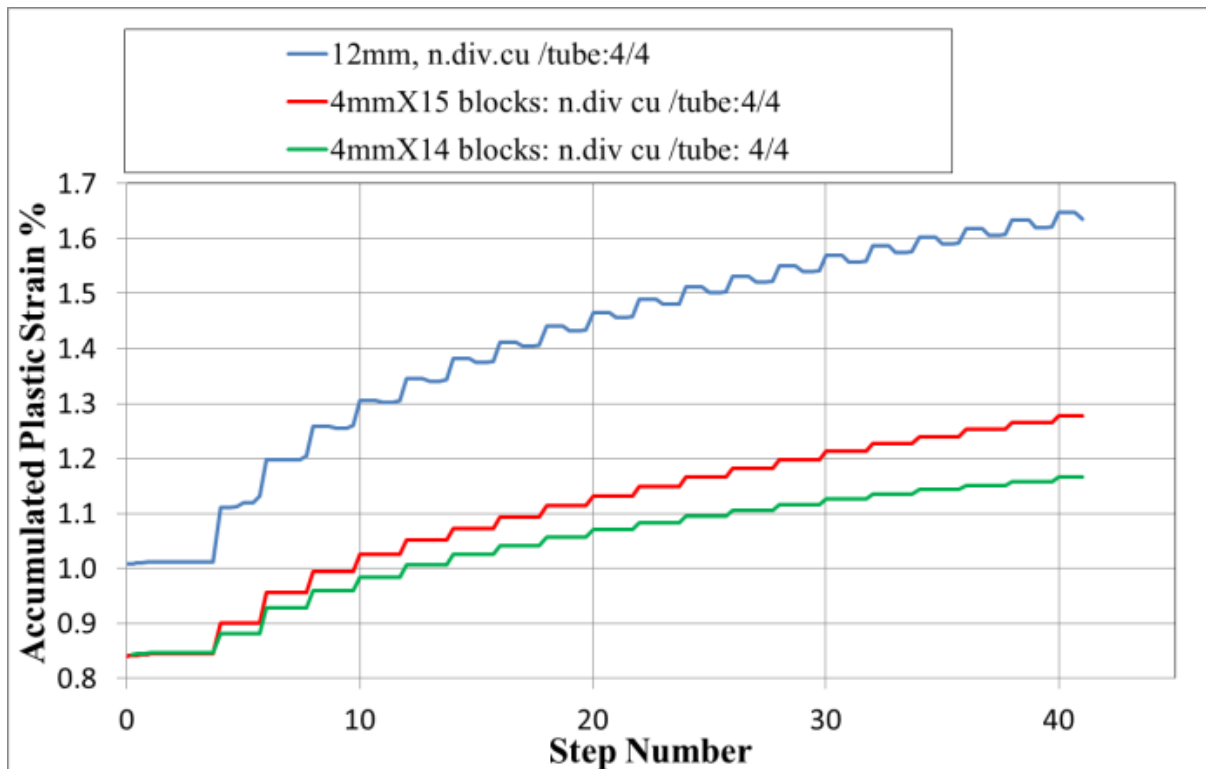


Figure 4.8: Accumulated Plastic Strain for mesh selected

Since the plastic strain does not increase significantly after 20 cycles, the total strain reported in Tab. 5.4 referred to the 20th cycles may be assumed as the highest and it can be used to estimate the fatigue life of the tube by means of Low Cycle Fatigue Curves (LCFC).

LCFC in Figure 4.9 come from ITER Material Properties Handbook [ITER Material Properties Handbook, 2016] and refers to a test temperature between RT and 350 °C. Dashed line used for fatigue life estimation is a fit curve of test data.

Tab. 4.4 Equivalent Total strain in CuCrZr Tube

CuCrZr Equivalent Total Strain %					
Cycle Number	4	8	12	16	20
12 mm	1.53	1.68	1.77	1.85	1.91
4 mm X 15 blocks	1.29	1.39	1.47	1.53	1.58
4mm X 14 blocks	1.25	1.34	1.39	1.43	1.46

The maximum total strain of CuCrZr tube is 1.91% in the configuration with 12 mm monoblocks, corresponding to about 370 cycles, while for the configuration with 4 mm monoblocks, the total strain of 1.58% corresponds to a number of cycles of about 480 (Tab 4.4).

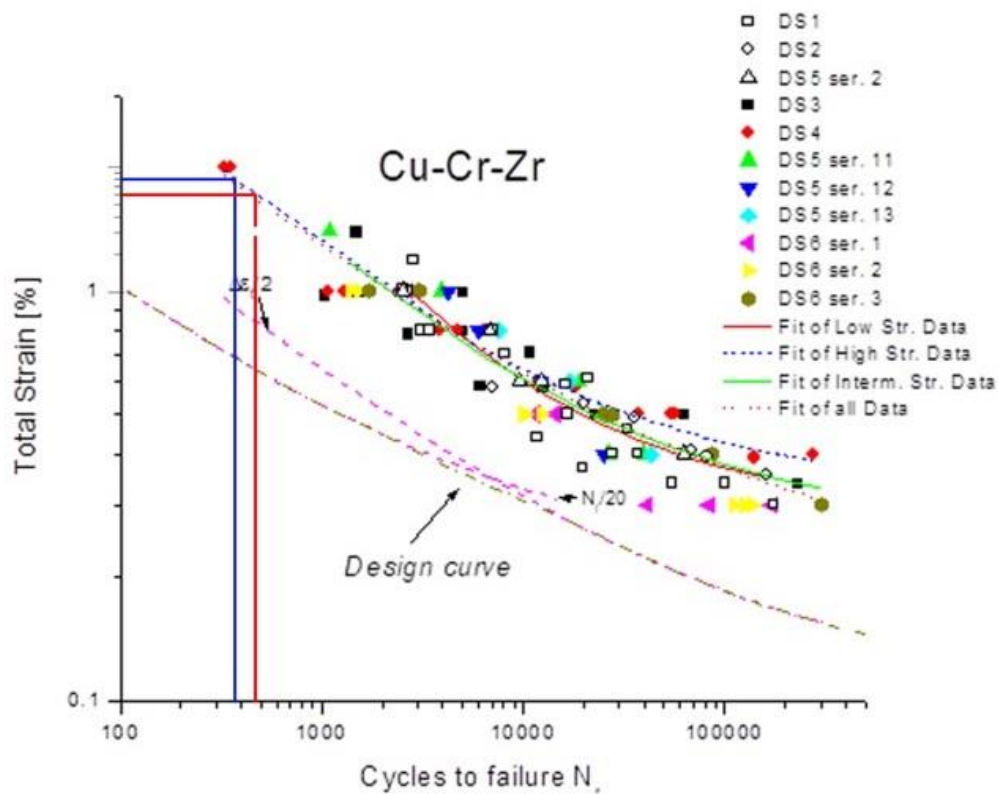


Figure 4.9: CuCrZr number of cycles using the LCFC between RT and 350 °C

Detachments at interface between tube and interlayer can be related to hoop stresses on the tube. The maximum value is about 400 MPa for both the configurations. Such value is reached during the cooling phase to 150°C. Moreover, it is very close to the residual stresses of the manufacturing process and decreases slowly with the increase of the number of cycles, due to the progressive hardening of the material. However, the position of the maximum

changes with the increase of the number of cycles. In the configuration with 12 mm in the thermalization phase a 150°C after the fabrication, the maximum is in the centre of the monoblock. After 5 cycles the maximum moves at the edge of the monoblock. In the configuration with 4 mm the maximum is slightly lower, and also remains localized under the monoblocks, thus decreasing the probability of the arising of detachments in the junction between the tube and the interlayer. In Figure 4.10 the circumferential stress inside the CuCrZr tube are showed.

As stated before, the monoblock surface generally exhibits axial cracks on the centre of the monoblock when it is subjected to cyclic loads at 20 MW/m².

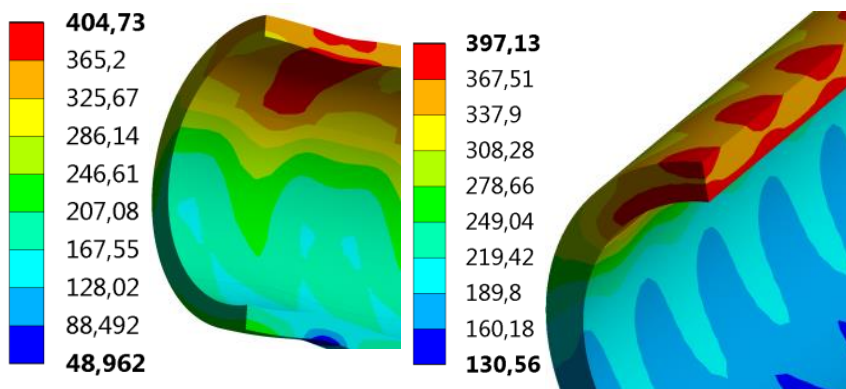


Figure 4.10: Hoop stress (MPa) in CuCrZr Tube 12 mm thick monoblocks (left) and 4 mm thick monoblocks the last cooling step (right)

Figure 4.11 shows the stress field in the Cu OFHC Interlayer. The stress fields in the tube and in the interlayer do not show great differences for the two configurations. However, the plastic deformations for the two configurations lead to a considerably different behavior in term of life expectancy of the component.

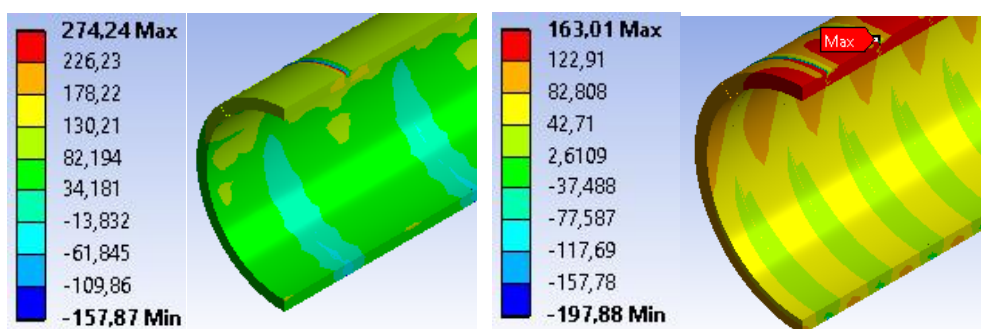


Figure 4.11: Hoop stress in Cu OFHC Interlayer (12 and 4 mm)

Hoop stresses are responsible for the onset of tungsten cracks along the axis tube direction. The maximum hoop stresses are at the interface with the copper interlayer. The Figure 4.12

show that the 4 mm configuration exhibits a lower maximum stress value and its position is not in the high temperature zone. This implies a lower probability of occurrence of this type of fracture in PFU with 4 mm monoblocks.

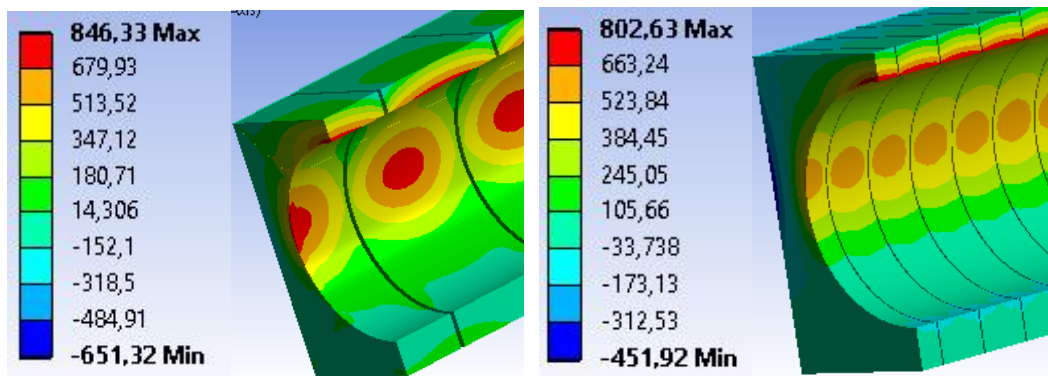


Figure 4.12: Hoop stress in Tungsten Monoblock (12 and 4 mm)

Hoop stresses are responsible for the onset of tungsten cracks along the axis tube direction. The maximum hoop stresses are at the interface with the copper interlayer. This implies a lower probability of occurrence of this type of fracture in PFU with 4 mm monoblocks.

Figure 4.13 shows the maximum stress normal to crack plan (marked with red in the sketch). It turns out to be greater than the yield both under compression (heat loading step) and under traction (cooling at 150 °C).

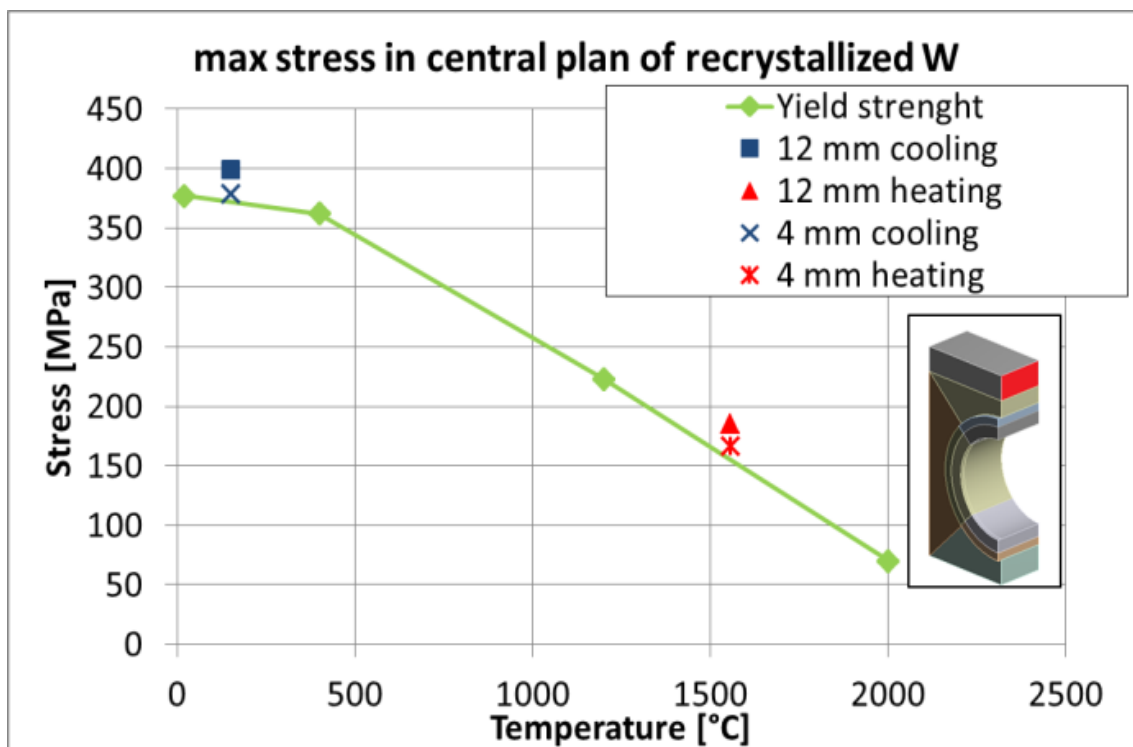


Figure 4.13: Maximum stress in recrystallized tungsten at central plan of HHF perpendicular surface at tube axial direction

Figure 4.14 shows the plastic deformations in the recrystallized tungsten layer for the two configurations. Only small plastic strain is produced by the initial load, and the armor surface readily enters into a purely elastic state due to the shakedown effect.

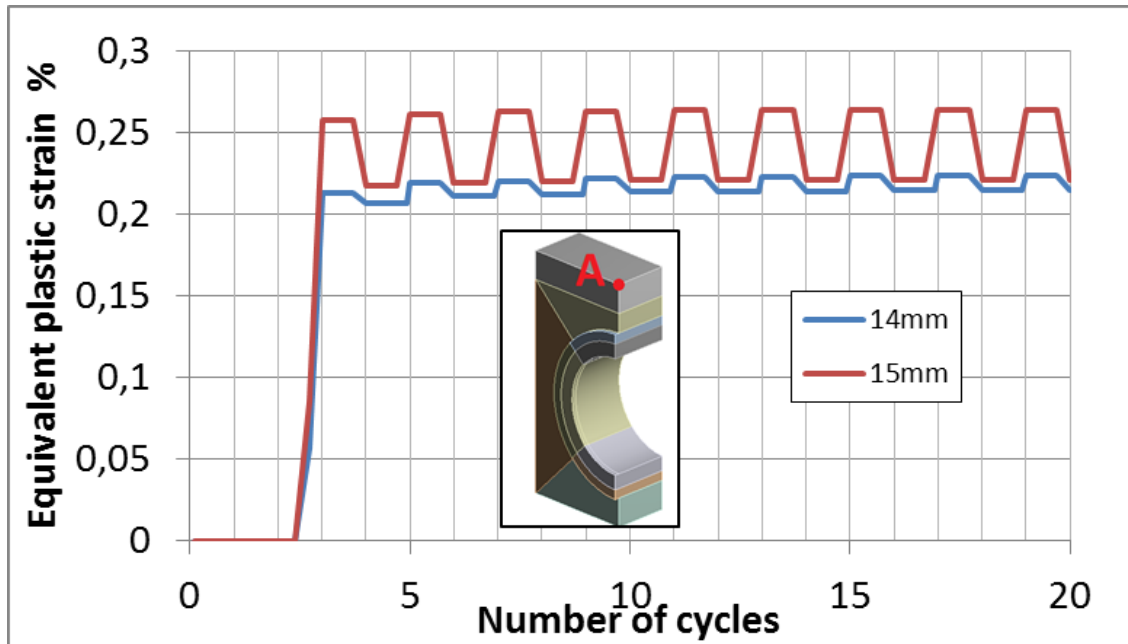


Figure 4.14: Evolution of accumulated equivalent plastic strain at armor surface positions A

The present study addresses a comparison between different design configurations of PFU of the DEMO divertor. Plastic material behaviour has been introduced to investigate the fatigue lifetime of the CuCrZr tube. This study has been carried out using a model consisting of a length of PFU included between two adjacent attachment to the cassette. The results obtained are very close to those obtained with model of single monoblock when only a quarter of monoblock is modelled, symmetry is imposed on the sectioned faces and, moreover, the cross section of tube and interlayer are constrained to remain in plan. This simplified model is therefore appropriate to describe the actual system of constraints of PFU when local effects of load and heat exchange are neglected. In order to investigate whether the monoblock thickness or gap between monoblocks is dominant for the fatigue life of CuCrZr tube, two different configurations with 4 mm thick monoblocks and different gap, and a configuration with 12 mm thick monoblocks have been studied. Numerical results have shown that the gap does not considerably influence the fatigue life of CuCrZr tube. The configuration with 4 mm thick monoblocks showed an improved component behaviour both in terms of tube fatigue life and in terms of hoop stress and plasticity strain in recrystallized tungsten. In any case, the advantages found of the 4 mm configuration compared to that of 12mm were not so

significant as to justify the considerable costs increase due to new design of the PFU support system respect to that already developed for the ITER divertor.

5. Thermal-Hydraulic Analysis for Thermal Shield (TS) of Divertor Tokamak Test Facility (DTT)

This chapter reports the results of a second paper [Maviglia, 2019] , on the thermal analyses of Thermal Shield (TS) carried out to define a preliminar design of this component, to estimate the heat load for the cryogenic plant and to give an indication of the number of the pipes and distance from the ribs and pipes. These indications should be used for first attempt placement of the pipes in a 3D model loads on refrigerator systems in DTT (Figure 5.1).

Over the last decades in the context of international nuclear fusion research, Italy has played a crucial role, supported by education and training of universities and research institutions as well as successful involvement of the national industries, which led to a virtuous circle oriented towards innovation that enriches the country [DTT Team, Blue book 2015].

The following are the characteristics of the DTT machine (DTT Team, 2016 and 2019).

Table 5.1 Main dtt parameters for a reference standard x point scenario

R (m)	2.15	β_N	1.5
a (m)	0,7	T_{Res} (sec)	8
I_P (MA)	6	V_{Loop} (V)	0.17
B_T (T)	6	Z_{eff}	1.7
V (m ³)	33.0	P_{Rad} (MW)	13
P_{ADD} (MW)	45	P_{Sep} (MW)	32
H98	1	T_{Ped} (KeV)	3.1
$\langle n_e \rangle$ (10 ²⁰ m ⁻³)	1.7	n_{Ped} (10 ²⁰ m ⁻³)	1.4
n_e/n_{eG}	0.45	β_p	0.5
$\langle T_e \rangle$ (KeV)	6.2	P_{Div} (MW/m ²) (No Rad)	~ 55
T (sec)	0.47	P_{Sep}/R (MW/m)	15
$n_e(0)$ (10 ²⁰ m ⁻³)	2.2	$P_{Tot}B/R$ (MW T/m)	125
$T_e(0)$ (KeV)	10.2	λ_q (mm)	~ 2.0

The DTT device then be devoted to tests of alternative divertor solutions, including advanced magnetic configurations and liquid metal targets. The set of small internal coils allows to locally modify the topology of a given magnetic configuration, so as to produce a very large set of quite different topologies.

Albeit it is expected that the experience gained by the operation of ITER will allow to operate a fusion reactor under quiescent and non-disruptive plasma conditions, a very limited number of disruptions may still occur. Such off-normal events can cause ablation of material from the heated surface and progressive damage. Giving a very stringent limit on the allowable number of disruptions in a reactor the sputtering erosion of the PFM during normal operation will determine the component lifetime. Experiments with high-Z materials, especially tungsten (W), are being carried out in the ASDEX Upgrade tokamak, since the erosion of this material under normal operation conditions is considerably lower than the plasma induced erosion of low Z materials like carbon (C) or beryllium (Be). The disadvantage of high Z materials, however, is the far lower tolerable impurity concentration within the plasma which poses strong demands on plasma control. Further to sputtering, metallic materials might show additional erosion by forming blisters under incidence of ions from the plasma. Tritium from the plasma is absorbed by the PFMs and might enter the coolant of the PFCs (Zhou, 2003). The extend to which tritium is being taken up by the PFCs is highly dependent on the material used, and the state of the internal nano- and microstructure of the material. Mechanical stress and strain within the materials is caused by the superposition of residual stress stemming from the manufacturing process and the additional thermal stress during operation. Neutron damage can lead to the degradation of the mechanical properties and in some cases to a decrease of the thermal conductivity and dimensional changes of the PFMs. These boundary conditions indicate that the selection of a suitable PFM is difficult, since many of the processes mentioned above can at present only be quantified by making largely simplifying assumptions.

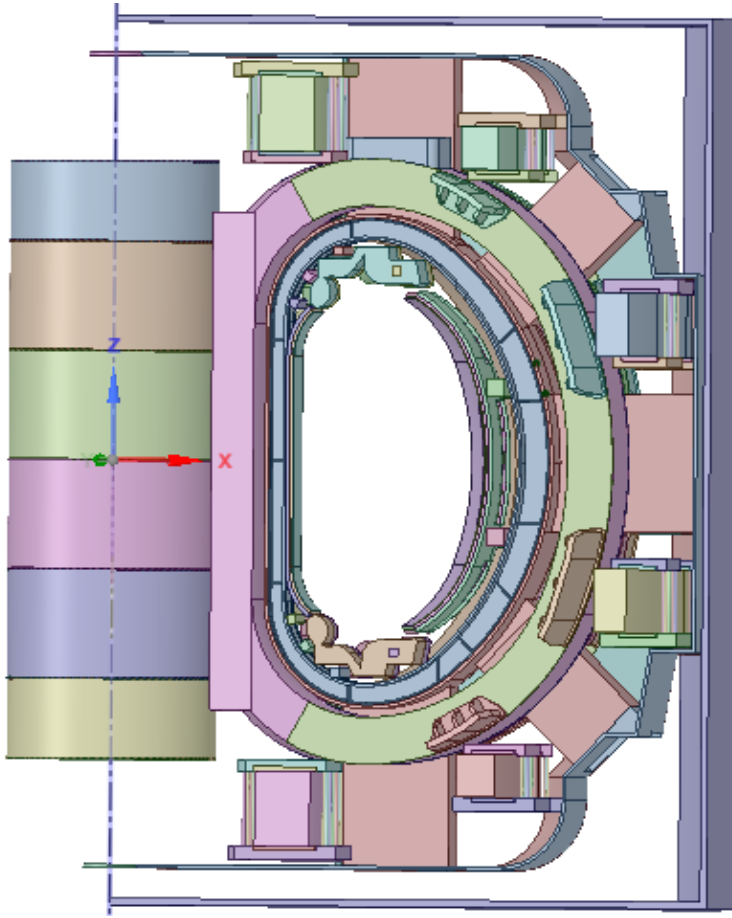


Figure 5.1 Section of DTT

5.1. Thermal Shield in DTT

The thermal shield component is cooled by Helium which flows inside pipes welded on one side, internally to a box structure similar to that of the new Japanese machine, JT-60SA ([Yoshida, 2014], [Nuclear Fusion Research and Development, 9.5 Thermal Shields for a Large Superconducting Magnet, JAEA R&D Review 2015]). Helium extracted from the cryogenic system at 70K, cools first the shield around the Vacuum Vessel (VVTS) and then the cryostat thermal shield (CTS) passing from the ports (PTS). Thermal analysis was performed in order to determine the better position and number of cooling pipes as a function of distance between the ribs in each component (VVTS, PTS and CTS), and for different poloidal sections.

5.1.1. Vacuum Vessel thermal shield (VVTS)

The vacuum vessel thermal shield (VVTS) is closely shaped along the gap from TF coils and vacuum vessel. The VVTS consists of an Inboard VVTS and the Outboard-VVTS and is manufactured by 10-degree unit. An assembled 18 degree-sector VVTS divided in two different categories are shown in Figure 5.2.

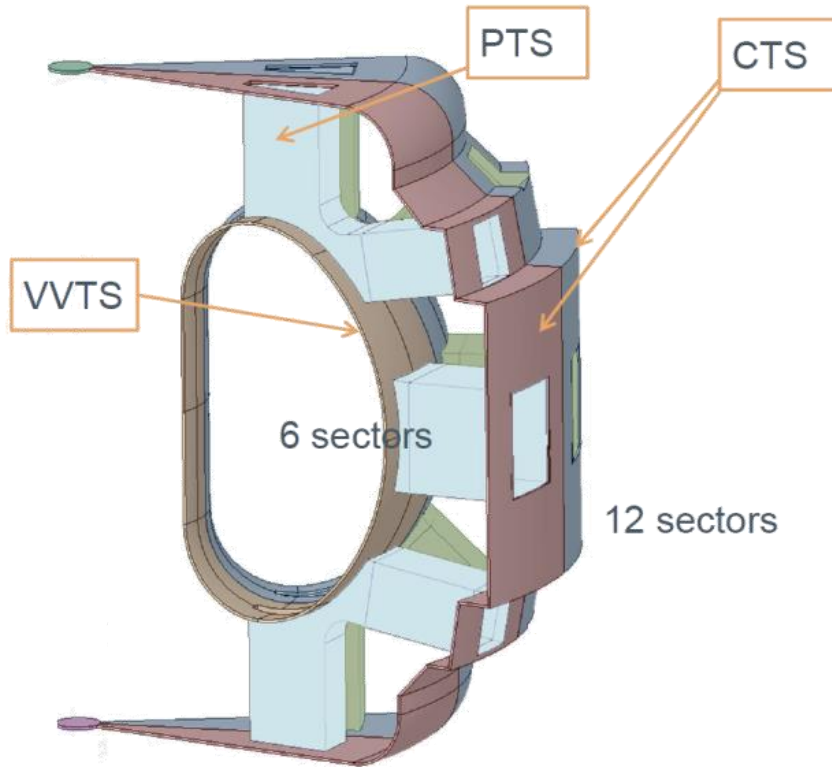


Figure 5.2 Thermal shield schema in DTT facility

The Thermal shield is a box structure having a total thickness of 25 mm. It is composed of two 3-mm thick stainless-steel sheets on the upper and lower part, ribs along the poloidal direction of 10 mm thickness, cooling pipes having an inner diameter of 10.9 mm, welded on one sheet. The pressure inside the cooling pipes is 18 bar Figure 5.3.

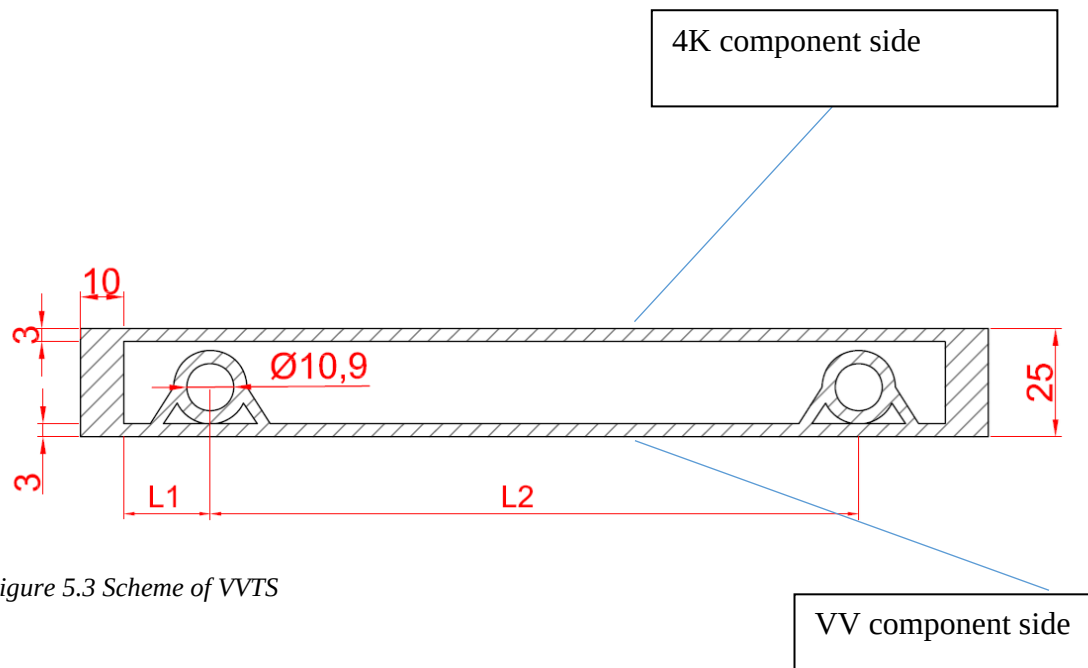


Figure 5.3 Scheme of VVTS

5.1.2. Port thermal shield

PTS consists of cooling pipes and two SUS316L plates. The PTS covers each corresponding port. The PTS is categorized into 2 types depending on the individual port shape; Upper-PTS (Figure 5.4), Central-PTS (Figure 5.5), and Oblique-PTS (Figure 5.5).

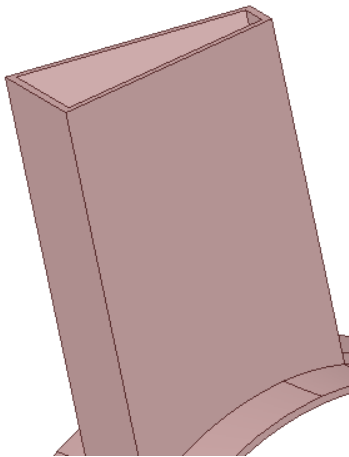


Figure 5.4 Upper Port

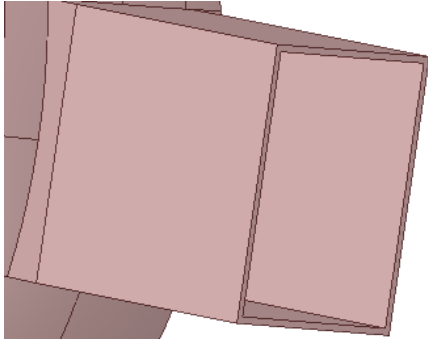


Figure 5.5 Central Port

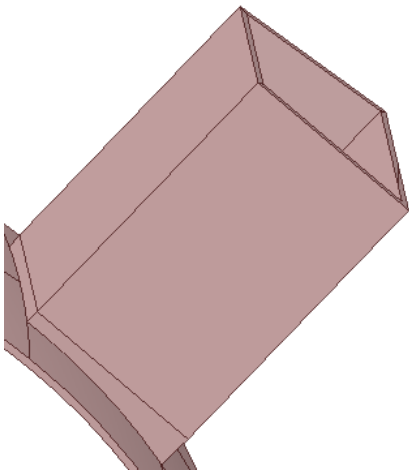


Figure 5.6 Oblique Port

5.1.3. Cryostat thermal shield

The CTS consists of cooling pipes and two SUS316L plates with a thickness of 3 mm. Multilayered Insulator (MLI) is located on ambient side. The CTS is classified into 3 types: Bottom-CTS, Middle-CTS and Top-CTS. The CTS is manufactured in 10-degree unit and electrically segmented into 18 sectors in the toroidal direction. Insulation couplers and mechanical couplers are used for 10-degree.

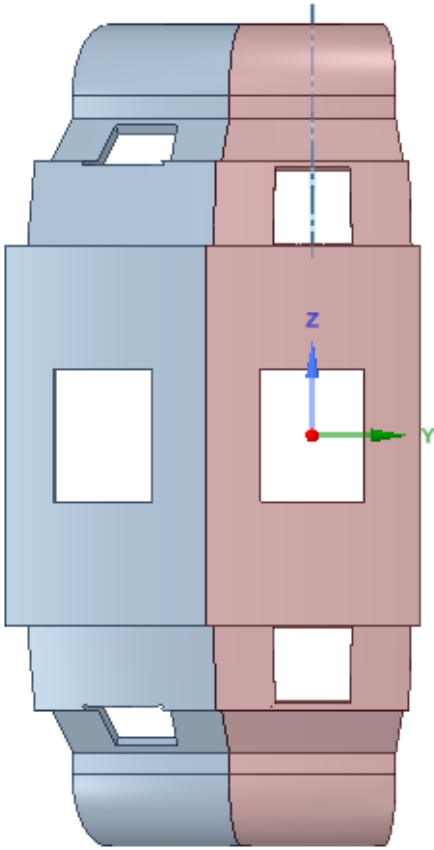


Figure 5.7 Cryostat thermal shield

5.2. Analysis of Thermal Shields

2-D analyses have been carried out for different configurations with two or four tubes between two consecutive ribs (Fig 5.8). The number of tubes has been chosen in the VVTS, PTS, and CTS at different poloidal positions depending on the distance from the ribs.

The optimal distance between two consecutive ribs and the nearest pipe (L1 in figure 5.8) has been chosen such that the average temperature on the panel surface facing the coils does not exceed by three degrees the inlet Helium temperature. Distance between two consecutive ribs depends on the poloidal position (because 10 toroidal degrees between ribs are fixed), then also the distance between pipe and pipe (L2) changes, but, in the 4 pipes model, it is forced to be the same.

Helium flows inside the tubes and enters first in the VVTS at 70K. In the analysis, Helium bulk temperature has been considered equal to 70K for VVTS, 80K for PTS, and 90K for CTS. The temperature of the VV is 100 °C, while the cryostat temperature is 22 °C. Coils have been assumed to be at 4.5K. The pressure inside the cooling pipes is 18 bar. The

software used for these analyses is Ansys Workbench 18.2. The emissivity of the internal surfaces of TS is assumed equal to as 0.3, while the emissivity of the external surfaces of TS 0.15; the VV emissivity is 0.15 and the coils emissivity is 1.



Figure 5.8 Configuration with 2-tube and 4-tube

The optimal distance between two consecutive ribs has been chosen such that the average temperature on the panel surface facing the coils does not exceed by three degrees the inlet Helium temperature.

Helium heat exchange coefficient has been evaluated assuming a velocity of 22 m/s (Tab. 5.1.)

<i>Table 5.1. Heat transfer coefficient</i>		
	Temperature (K)	Heat transfer coefficient (W*m-2*K-1)
VVTS	70	3543
PTS	80	3399
CTS	90	3267

5.3. Results Thermal shield

As an example of optimization result one can refer to the case of the inboard part of the VVTS. VVTS sees VV at 100°C on one side, and coils at 4.5K from the other side (Figure 5.9). The internal distance between two ribs is 180 mm, which corresponds to a length of half a sector in the toroidal plane. It can be noticed that a distance between pipes and ribs equal to 20 mm verify the established criterion.

In Figure 5.9 are showed the models used to analyze inboard and outboard of TS.

Model for internal VVTS

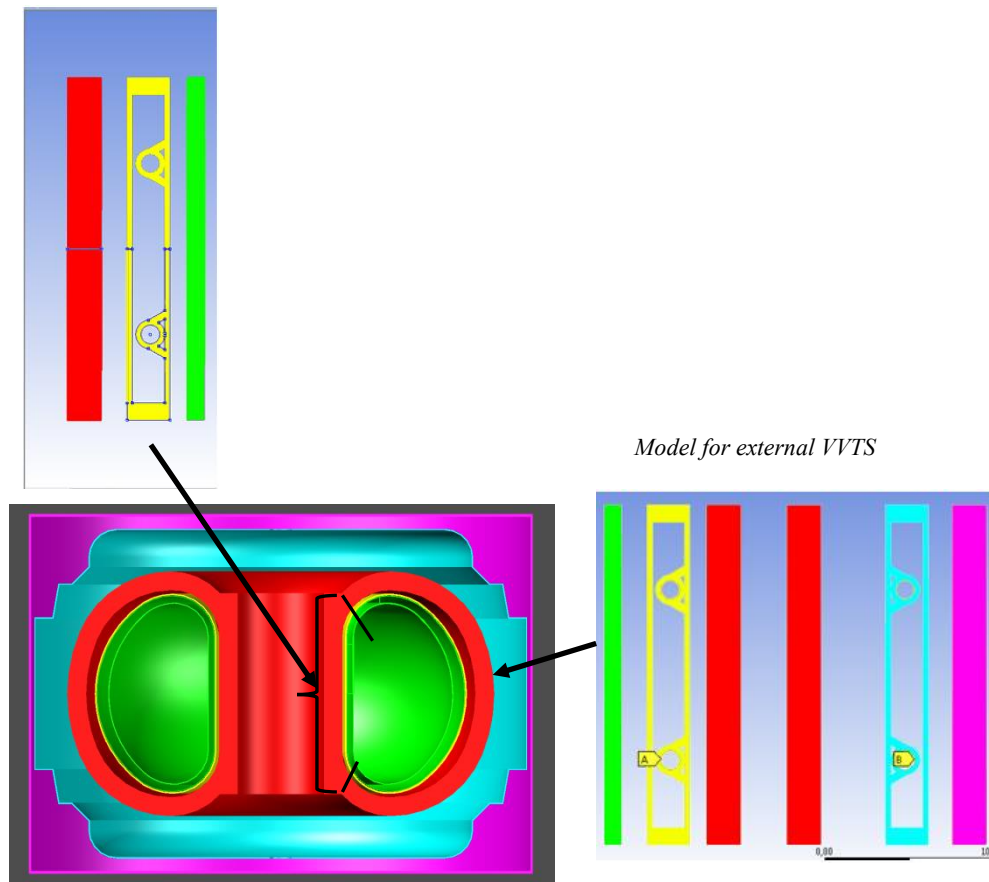


Figure 5.9 Sketch of the model used

In the outboard zone, the distance between ribs is such that 4 pipes are necessary between the ports, while only 2 pipes should be placed laterally to the ports. From the 2D analysis the power per surface unit can be calculated. Then, to evaluate the total power, the results obtained for the 2 or 4 pipes models are multiplied by proper areas. Considering the inboard, power density from the 2 pipes model is multiplied by the inboard area. For the outboard, the power density from 4 pipes model is multiplied only by the area between ports, while that from the 2 pipes model is multiplied by the area of the VVTS laterally to the ports (Figure 5.10 and Table 5.2).

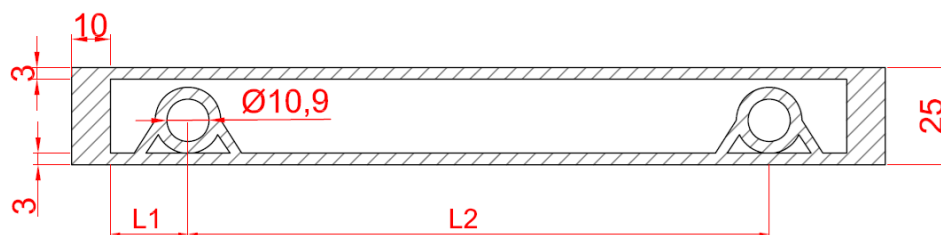


Figure 5.10 2-tube model

Table 5.2 2D model with 2 pipes: influence of the distance from the ribs on the average temperature of the cryogenic components side face

	Distance btw pipe and rib	Pipe pitch	Average temperature	Evaluation
Case 1	20mm	140mm	71,09K	Good
Case 2	40mm	100mm	73,50K	Bad

The same procedure was adopted for the 4-pipe configuration (Figure 5.11 and Tab 5.3).

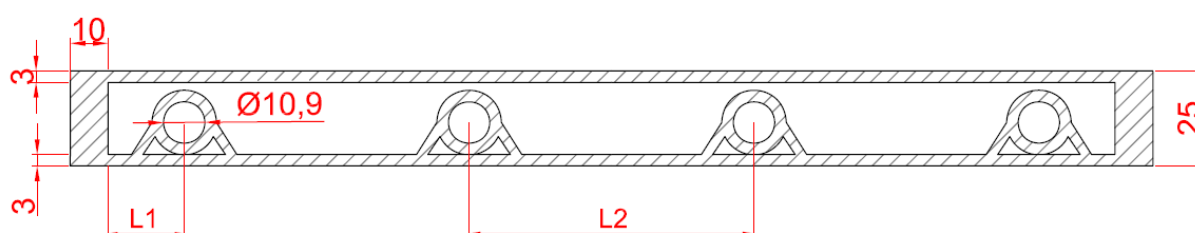


Figure 5.11 4-tube model

Table 5.3 2D model with 4 pipes: influence of the distance from the ribs on the average temperature of the cryogenic components side face

	Distance btw pipe and rib	Pipe pitch	Average temperature	Evaluation
Case 1	20mm	310mm	82,90K	Good
Case 2	40mm	310mm	84,23K	Bad

With the different optimized models, the thermal loads on coils and TS have been evaluated in term of power per surface unit. Different poloidal sectors have been selected and the corresponding areas measured and multiplied by power per surface unit calculated with the model related to the sector considered. For example, inside the inboard part of VVTS, only 2 pipes with distance of 20mm from the ribs proved to be enough to satisfy the criterion. The power per surface unit adsorbed by the Helium was multiplied per inboard area only.

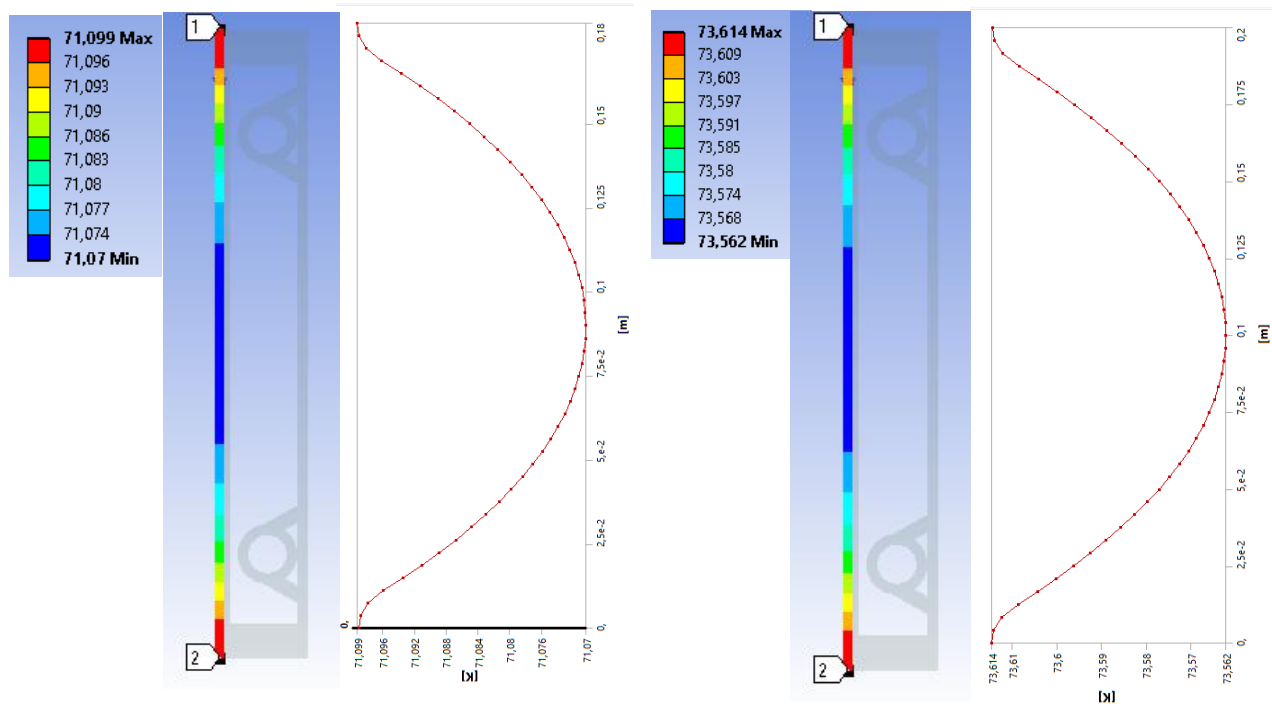


Figure 5.12 Temperature in two different model

To this aim, the specific power of the 4 pipes models is multiplied by the outboard VVTS surface should be added, to obtain the total power absorbed by the VVTS. Same procedure was used to estimate the power on PTS and CTS. Tab 5.4 shows the results obtained for the TS. In the first column the sum of the area of each poloidal sector area, and in the second the

total power for each component is reported. In the last column equivalent power at 4 K is obtained using the standard efficiency considered for cryoplants.

Tab.5.4 Heat load in Thermal shield

Heat load in TS W/m ²						
	4.5K	70K	80K	100 °C	90K	22 °C
2 Tube VVTS	-0,25	-88,68		88,9		
2 Tube PTS	-0,41		-88,44	88,8		
4 Tube VVTS	-0,27	-88,54		88,7		
4 Tube PTS	0,43		-88,25	88,6		
4 Tube CTS	-0,57				-33,94	34,5

The power needed to refrigerate the TS was calculated as follows:

$$\text{VVTS Heat Load} = 88.68 \times 22.3 + 88.54 \times 73 = 8.4 \text{ kW}$$

$$\text{PTS Heat Load} = 88.25 \times 170 + 88.44 \times 120 = 25.6 \text{ kW}$$

$$\text{CTS Heat Load} = 33.94 \times 221.5 = 7.5 \text{ kW}$$

$$P_{4K} = P_{70K} \frac{T_2}{(T_1 - T_2)};$$

$$T_1 = 70 \text{ K } T_2 = 4.2 \text{ K } P_{4K} = P_{70K} / 16$$

Tab 5.5 Area of the Thermal shield

	Surface 4 tube [m ²]	Surface 2 tube [m ²]
WTS	73,1	22,3
PTS	170	120
CTS	221.5	

TS Total Power= 41.5 kW

TS total power equivalent to 4K =2.6kW

Tab. 5.6 Heat load in Coils

Heat load in Coils W/m ²						
	4.5K	70K	80K	100 °C	90K	22 °C
2 Tube VVTS	-0,25	-88,68		88,9		
2 Tube PTS	-0,41		-88,44	88,8		
4 Tube VVTS	-0,27	-88,54		88,7		
4 Tube PTS	0,43		-88,25	88,6		
4 Tube CTS	-0,57				-33,94	34,5

The power needed to refrigerate the coils was calculated as follows:

TF Total Power=0.25X22.3+ 0.27X39.4+0.57X62.1+0.43x89=0.09 kW

PF Total Power= 0.41X72+0.57X72=0.07kW

Coil total power =0.16kW

Same procedure was used to estimate the power on PTS and CTS.

Table 5.8 shows the results obtained for the TS. In the first column the area, and in the second the total power for each component are reported. In the last column equivalent power at 4K is obtained using the standard efficiency considered for cryoplants ($P_{4K}=P_{70K} T_2/(T_1 - T_2)$ with $T_1=70$ K and $T_2=4.2$ K because the Helium for the TS cooling is extracted from the cryoplant at 70 K)

Tab 5.7 Area of the thermal shields

	Surface model 2 tube [m ²]	Surface model 4 tube VVTS/CTS [m ²]	Surface 4 Tube PTS [m ²]
TF	22.3 int inboard	39.4 int outboard 62.1 ext outboard	89
PF		72	72

Table 5.8. Power on Thermal Shield

	Total Areas (m ²)	Total Power (kW)	Equivalent Power at 4K (kW)
VVTS	96.47	8.56	0.535
PTS	272.48	24.06	1.504
CTS	266.26	10.92	0.682
<i>Total Power</i>			<i>2.7</i>

Therefore, 2.7 kW is the total power required by the cryogenic system for the TS when the neutron heating is not considered. The same procedure was applied to determine the total power on the coil system and the results are summarized in Table 5.9.

Table 5.9. Power on Coils

	Total Areas (m ²)	Total Power (KW)
TF	146.91	0.053
PF	102.52	0.040
<i>Total Power</i>		<i>0.093</i>

In this work, thermo-hydraulic analyses were carried out for a preliminary design of the DTT thermal shield. In the TS, analysis results showed that the maximum acceptable distance between pipes and ribs is 20 mm. Therefore, analysis allowed to estimate the 4K equivalent power required to the cryogenic system for the TS and for the coil system, equal to 2.7KW and 0.093KW respectively.

It is important to remark that neutron heating during discharge was not considered and should be added for the correct definition of the requirements of the cryogenic system.

6. Conclusions

The work carried out for the present thesis has allowed me to enhance my expertise on the thermal-machanical finite element analyses and led to two papers on the topic.

The results reached out about the optimization of the monoblocks' thickness have been considered relevant for the realization of DEMO divertor, which is expected to be finished on 2050 and have been published on FED.

In order to realize this work, we decided to start from the geometry proposed by ENEA (ITER-like). The monoblock thickness of 12 mm was compared with the 4 mm one, currently hypothesized for DEMO.

The present study copes with a comparison between different design configurations of PFU of the DEMO divertor. In detail, plastic material behaviour has been introduced to investigate fatigue lifetime of the CuCrZr tube. This study has been carried out using a model consisting of a length of PFU between two adjacent attachment to the cassette. The results obtained are in agreement with those obtained with model of single monoblock when only a quarter of monoblock is modelled, symmetry is imposed on the sectioned faces and, moreover, the cross section of tube and interlayer are constrained to remain in plan. This simplified model is therefore appropriate to describe the actual system of constraints of PFU when local effects of load and heat exchange are neglected. In order to investigate whether the monoblock thickness or gap between monoblocks is dominant for the fatigue life of CuCrZr tube, two different configurations with 4 mm thick monoblocks and different gap, and a configuration with 12 mm thick monoblocks have been studied. Numerical results have shown that the gap does not considerably influence the fatigue life of CuCrZr tube. The configuration with 4 mm thick monoblocks led to an improved component behaviour both in terms of tube fatigue life and in terms of hoop stress and plasticity strain in recrystallized tungsten. In any case, the advantages of the 4 mm configuration, if compared to that of 12mm, were found not so

significant to justify the considerable costs increase due to new design of the PFU support system respect to that already developed for the ITER divertor.

The configuration with 4mm thick monoblocks shown an improved component behaviour both in terms of tube fatigue life and in terms of hoop stress and plasticity strain in recrystallized tungsten. In any case, the advantages found of the 4mm configuration compared to that of 12mm were not so significant as to justify the considerable costs increase due to new design of the PFU support system respect to that already developed for the ITER divertor.

Regarding DTT's VVTS I made the first 2D analyzes of this component. The Divertor Tokamak Test (DTT) facility has been proposed by ENEA, in collaboration with other Italian institutions, to investigate power exhaust solutions with a machine capable of integrating all DEMO relevant physics and technology issues. The DTT machine will be able to host, in different phases of its life-time, advanced divertor magnetic configurations and liquid metal solutions, and will be able to withstand the large loads expected in the DEMO fusion power plant. To minimize the heat transferred to the superconductive coils, the vacuum vessel (VV), heated by water at 100°C, is shielded by a Thermal Shield (TS) cooled with helium at 70K. By means of a preliminary thermo-hydraulic analysis, the power requested to the cryogenic system has been estimated.

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