SPARC- EXPERIMENTAL FUSION DEVICE

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SPARC is a new fusion energy experiment, for which expectations are set pretty high in the science community. It will be using the most modern technology, from high temperature superconductors to ion cyclotron heating. It will be operating using the most promising fusion fuel, deuterium-tritium. SPARC will have 10 times smaller volume than ITER, and still manage to have a gain factor higher than 2, thanks to its strong magnetic field, high plasma currents and the sole option for heating - ion cyclotron heating. It's building is projected for this year, 2021.

SPARC- EKSPERIMENTALNA FUZIJSKA NAPRAVA

SPARC je eden najnovejših fuzijskih eksperimentov za katerega so pričakovanja v svetu znanosti postavljena zelo visoko. Z uporabo najmodernejše tehnologije, kot so na primer visoko temperaturni super prevodniki, ki zagotavljajo izredno veliko magnetno polje z manj hlajenja in ionsko ciklotronsko gretje. Deloval bo na principu najobetavnejšega fuzijskega goriva, devterij-tritij. SPARC bo 10 krat manjši kot ITER, a bo vseeno proizvedel 2 krat več energije kot jo bomo vložili- na račun visokega magnetnega polja, visokih tokov in edine možnosti za segrevanje goriva ionskega ciklotronskega segrevanja. Gradnja SPARCA se bo začela letos, 2021.

1. Introduction

Energy use correlates with many quality-of-life metrics but so does CO2 emissions. The reality is, that every single nation that has industrialised and created a prosperous life for its citizens has done so at the expense of the climate. One way to make a change in the future may be fusion power plants. The world would benefit from nuclear fusion by extracting heat to produce electricity. Fusion is a process that was known as far back as the 1930s, when Robert d'Escourt Atkinson and Fritz Houtermans provided the first calculations for the rate of nuclear fusion in stars [1]. Since then, scientists have been reproducing the fusion process on Earth. The basic idea behind it is to have 2 light elements bombard each other to form one different atomic nuclei and subatomic particles. The described reaction would be self-sustained just by simply providing a continuous supply of this light element, which should be readily and inexpensively available. Studies of the nuclear properties of light elements have indicated that three reactions may be advantageous for the production of nuclear energy: (1) deuterium-deuterium (D-D), (2) deuterium-helium(D-He3) and (3) deuterium-tritium (D-T) [2]. In Figure 1, we can see obvious reasons as to why D-T reaction is the most promising. It has a large cross section at relatively low energies compared to the other two reactions. In the spirit of that, the first tokamaks, and the first energy from D-D plasma was achieved by the late 50 s. The first prototype of modern tokamaks, JET, and the first D-T plasmas were achieved in 1991 [1]. It was built with a lot of knowledge gained from previous experiments. Since then, it seems that we are heading in the right direction. The next big experimental facility after JET will be ITER, which is already being constructed, and then later its successor will be DEMO (demonstrational

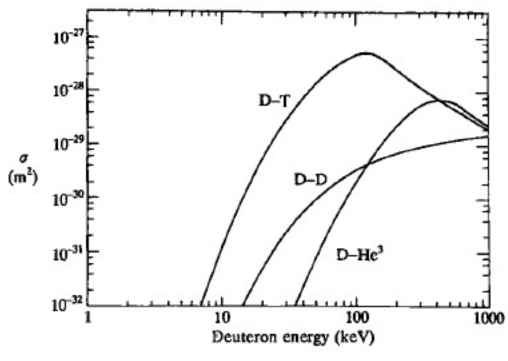

Figure 1. Cross section and deuteron energy, for three most promising fusion reactions: D-D, D-T, D-He. D-T is the most promising one, with large cross section at low energies [4].

power plant). SPARC belongs somewhere in between JET and DEMO and stands for the smallest possible affordable robust tokamak. To contain fusion in these nuclear facilities, a large magnetic field needs to be provided, using superconducting magnets. The main difference between ITER and SPARC will be in their different superconducting magnets, which will enable smaller fusion devices and higher magnetic fields.

2. Plasma and different confinement methods

The subject of fusion was briefly touched on previously but now we will build on that. The biggest difficulty with fusion is the repulsive Coulomb force between the nuclei. It's the biggest obstacle to getting enough fusion reactions. It turns out that the probability of particles just bouncing off each other is always significantly higher than the probability of fusion. If we want to overcome Coulomb scattering, we have to keep the particles in a good enough confinement. When we confine these particles at the energies required for fusion, the fuel will be in a state called plasma. The latter is an ionised gas; its behaviour is described by its collective behaviour since the long-range electromagnetic interactions dominate. If we go even further in our description of plasma, we know that it has charged particles, which create electric fields. Let's take a look at the relationship between magnetic and electric fields. From Maxwell's equations we have the following curl identities:

$$\nabla \times \overrightarrow{B} = \mu_0 \overrightarrow{J} + \frac{1}{c^2}\frac{\partial \overrightarrow{E}}{\partial t}, \nabla \times \overrightarrow{E} = -\frac{\partial \overrightarrow{B}}{\partial t} \tag{1}$$

Differentiating on both sides of Ampere's circuital law,

$$\frac{\partial}{\partial t}[\nabla \times \overrightarrow{B}] = \nabla \times \frac{\partial \overrightarrow{B}}{\partial t} = \mu_0 \frac{\partial \overrightarrow{J}}{\partial t} + \frac{1}{c^2}\frac{\partial^2 \overrightarrow{E}}{\partial t^2} \tag{2}$$

Substituting $\frac{\partial \overrightarrow{B}}{\partial t}$ from the electric field curl equation and using double curl identity $\nabla \times \nabla \times \overrightarrow{E} = -\nabla^2 \overrightarrow{E} + \nabla(\nabla \cdot \overrightarrow{E})$,

$$\nabla(\nabla \cdot \overrightarrow{E}) - \nabla^2 \overrightarrow{E} = -\mu_0 \frac{\partial \overrightarrow{J}}{\partial t} + \frac{1}{c^2}\frac{\partial^2 \overrightarrow{E}}{\partial t^2} \tag{3}$$

Defining current generated by a species in plasma as: $\overrightarrow{J} = \rho_s \nu_s = \sum_s n_s e_s \nu_s$, where n stands for number of charged particles per unit volume, ν is velocitiy of particle, e represents a charge of particle and s represents different species. A solved balance equation is then used to solve for ν_s. Using cold plasma approximation there are no net particle flows, since electrons scatter off ions and travel around them without imparting significant momentum, we consider there to be no pressure gradients. Since the plasma in unmagnetized, there is no Lorentz force. If we assume waves of form $\exp[i(k \cdot r - \omega t)]$, the Helmholtz equation can be used to specify $\nabla^2 \overrightarrow{E} = -k^2 \overrightarrow{E}$ and $-\nabla(\nabla \cdot \overrightarrow{E}) = -k(k \cdot \overrightarrow{E})$. The general form of plasma dispersion relation is found to be,

$$-k\left(k \cdot \overrightarrow{E}\right) + k^2 \overrightarrow{E} = \frac{\omega^2}{c^2}\overrightarrow{E} - \mu_0 \sum_s \frac{n_s e_s^2}{m_s}\overrightarrow{E} \tag{4}$$

Simplifying the dispersion relation for a longitudinal wave,$k \| \overrightarrow{E}$,we get

$$\omega^2 = \frac{1}{\epsilon_0}\sum_s \frac{n_s e_s^2}{m_s} \equiv \omega_p^2 \tag{5}$$

A longitudinal plasma wave oscillates at the fundamental electron plasma frequency and propagates by the electrostatic interactions of the plasma electrons. The longitudinal wave is carried by electron density fluctuations and is at a higher frequency than the ions sound wave. In this limit, the ions appear virtually stationary to the electrons, and no pressure gradients are induced. The longitudinal plasma wave has no resonances or cut-offs. Meanwhile in the simplification for the transverse wave, $k \perp E$, we have $\omega^2 = c^2k^2 + \omega_p^2$. This wave has no resonances but has a cutoff when the wave frequency is below the plasma frequency. Plasma differs from a regular neutral gas, as it is good at conducting electric current [3]. For better understanding of plasma physics, let's derive the Lawson criteria for the ignition conditions of plasma [4]. The three conditions required are: the right plasma density *(n)*, the temperature *(T)* and we need to confine this dense hot plasma for a long enough time (τ_E), for enough fusion reactions. We use temperature as the energy of the plasma system in a tokamak; it is best described with temperature, since it is a thermonuclear device. In fusion the temperature is conveniently measured in electronvolts (eV), which express the average kinetic energy of the plasma particles. Electronvolts divided by the Boltzmann constant (k_B), converts to the Kelvin scale. For the purposes of this paper, temperature (k_BT), will be described in keV. We will first take the thermonuclear power per unit's volume for D-T plasma fuel equation,

$$S_{tn} = n_d n_t \langle \sigma \nu \rangle E_\alpha \tag{6}$$

Where S stands for power density of the thermo nuclear reaction, n_d and n_t are the deuterium and tritium densities, $\langle \sigma \nu \rangle$ stands for the reaction rate, and E_α is the energy of the alpha particle. We know that the total ion density is $n = n_d + n_t$, therefore we can rewrite equation (6) as:

$$S_{tn} = n_d(n - n_d) \langle \sigma \nu \rangle E_\alpha \tag{7}$$

For maximisation of the power, we say that the densities of deuterium and tritium are the same, therefore $n_d = \frac{1}{2}n$. Now that we know what are the so called gains, let's look at the losses. In tokamaks there is a continuous loss of energy from the plasma. Energy can be lost from the hot gas in a few ways, by radiation, cyclotron emission, by conduction or by convection [5]. These losses have to be replaced by plasma heating. Due to the complicated and turbulent transport, it is hard to describe these losses in a correct mathematical way, so we usually define them by introducing a characteristic confinement time, defined by the relation, $S_L = \frac{w}{\tau_E}$, where τ_E stands for confinement time (usually in the order of a few seconds), w for power density and S again for energy density. In tokamaks the energy loss is balanced by externally supplied heating, therefore $P_H = P_L$, where P_H is the power which is supplied. The thermonuclear power equation (equation (6)) represents energy carried by neutrons, and by α particles. The neutrons leave the plasma without interaction, but the α particle is the one that transfers the energy to plasma. Thus, the total α particle heating is,

$$S_\alpha = \frac{1}{4}\overline{n^2 \langle \sigma \nu \rangle} E_\alpha \tag{8}$$

In the overall power balance, the power loss is balanced by externally supplied power plus the α particle power. That is, $S_H + S_\alpha = S_L$, or that supplied power is $S_H = S_L - S_\alpha$,

$$S_H = \left(\frac{3\overline{nkT}}{\tau_E} - \frac{1}{4}\overline{n^2 \langle \sigma \nu \rangle} E_\alpha \right) V \tag{9}$$

We want $S_L - S_\alpha = 0$, so that α heating is sufficent. So, the ignition condition for plasma is then:

$$n\tau_E > \frac{12}{E_\alpha} \frac{kT}{\langle \sigma \nu \rangle} \tag{10}$$

The right-hand side of inequality in equation (10) is a function of temperature only, from research we know that the solution has a minimum close to $T = 30$ keV, and the requirement for ignition at this temperature is: $n\tau_E > 1,5 \cdot 10^{20}\ m^{-3}$ s. However, τ_E is itself a function of temperature and the temperature at the minimum is not to be taken as an optimum condition. If we use $E_\alpha = 3,5$ MeV, the final ignition condition becomes:

$$nT\tau_E > 3 \cdot 10^{21} m^{-3} keV s \tag{11}$$

This is a very convenient form for the ignition condition, since it clearly highlights the ignition requirements. The condition would be reached for example by $n = 10^{20}\ m^{-3}$, $T = 10$ keV and $\tau_E = 3$ s. The product of these three quantities is called the Lawson criteria, with the triple product, and it sets up the conditions for burning plasma. Visualising the Lawson criterion is a powerful way to assess how close a particular fusion concept is to achieving the necessary conditions [4].

Through the decades we have learned that energy confinement is always worse than originally predicted, due to the so-called anomalous transport. As the plasma does not seem to want to cooperate, physicists have come up with scenarios where we create a transport barrier in the plasma, to retain the particles in its centre, and obtain a more effective confinement. Many experimental studies on confinement have been carried out in the major machines worldwide. This has led to a database of experiments being compiled, from which empiric laws have been determined, expressing confinement time as a function of the main machine and plasma parameters. These experiments were achievable in two operating modes, High-confinement (H-mode) and Low-confinement (L-mode) mode. First, all tokamaks operated in L-mode, later it was noticed that under certain conditions, confinement was abruptly improved above a heating power threshold, this enhanced confinement was called the H-mode, which has a factor of up to 2 times higher confinement times. A standard power law regression gives the fit (Figure 2) to the thermal confinement data in H-mode:

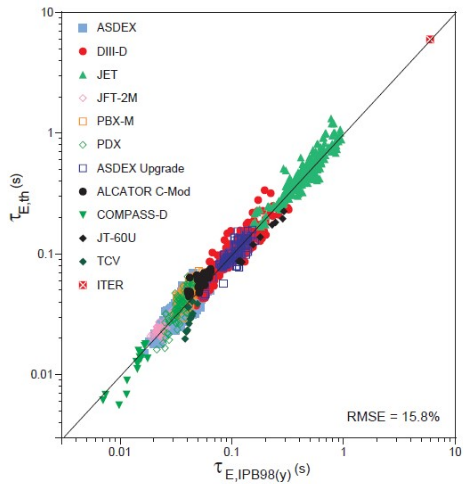

Figure 2. Comparison of H-mode thermal energy confinement time with the scaling equation (12) in the ITER H-mode. [8]

$$\tau^H_{E,th} = 0.0365 \cdot I^{0,97} B^{0,08} p^{-0,63} n^{0,41} \cdot M^{0,2} R^{1,93} \epsilon^{0,23} \kappa^{0,67} \tag{12}$$

Where κ, stands for MHD stability shaping limit, R is a wall loading constraint (in order of meters), M represents molar mass, n is ion density (in order of 10^{20}), P stands for power (in order of mega wats), I is current (in order of mega amperes), and B represents magnetic field (in the order of a couple of teslas) [6]. Also, worth mentioning is capital q ("Q"). It's one of the most important parameters when talking about producing energy from fusion as it represents the fusion energy output divided by the energy input:

$$Q = \frac{P_{out}}{P_{in}} \tag{13}$$

So if we want to make net energy, we want our Q to be at least 1 or as high as possible. One of the three most important conditions is, as mentioned, confinement. To improve this condition, scientists have developed two main approaches, inertial and magnetic confinement. We shall talk about the latter. [7, 8]

2.1 Tokamak

A tokamak is an experimental machine designed to harness the energy of fusion. Inside a tokamak, the energy produced through the fusion of nuclei is absorbed as heat in the walls of the vessel. A fusion power plant will use this heat to produce steam and then electricity by way of turbines and generators.

A strong magnetic field (several teslas) is used, to confine the very hot plasma. The strength of the magnetic field is a critical parameter in achieving these conditions. The magnetic field in tokamaks is made by powerful electromagnets where large electric currents flow (from a few to tens of kiloamperes) in loops surrounding the plasma [9]. For a fusion reactor, these currents must flow in a "superconductor" that provides zero resistance to the current, enabling the magnetic field to be produced indefinitely without having to supply electrical power to the magnets. The high-field approach to fusion has two perspectives: the confinement and stability of the plasma at the high field using short pulse copper electromagnets and advance superconductor magnet technology to obtain the highest possible magnetic field. When talking about superconductors we have two options. The first superconductors were low temperature superconductors (LTS, used by ITER), whereas the newer concept depends on high-temperature superconductors (HTS, which will be used in SPARC). The ITER magnets will be manufactured from niobium-tin (Nb3Sn) or niobium-titanium (Nb-Ti) and are predicted to achieve 5.3 T in the centre of a plasma. These magnets become superconducting when cooled with supercritical helium (critical temperature at 5.2 K) in the range of 4 K, therefore supercritical pressure is needed to cool the magnets in ITER. This means that, unlike the more familiar conductors such as copper or steel, these magnets have way higher efficiency. However, in recent years, new HTS materials have emerged as viable alternatives to LTS. One material of this kind is made of yttrium barium copper oxide or YBCO, (in some articles this is described as rare-earth barium copper oxide (REBCO)) [10].

2.2 MHD stability and beta factor

As mentioned, plasma is described with collective behaviour and it is easiest to describe it with a fluid model. Therefore, the identities of individual particles are neglected, and only the motion of fluid elements is observed. By treating plasma as a single conductive fluid, we can construct a magnetohydrodynamic, or shortened, "MHD" model. The general conditions for validity of multi-species transport equations (which are beyond this paper) are satisfied in magnetically confined fusion with one exception - high collisionality. The idea behind the MHD model is to isolate the physical processes that actually might require this exception and eliminate those degrees of freedom for modelling the system. We are using this MHD model, just so we can describe what is the yield factor of this type of tokamak, or how powerful a magnetic field we will need for our product of density, temperature and pressure. First, we have to write the independent version of the general stationary equilibrium solution:

$$\rho \overrightarrow{u} \cdot \bigtriangledown \overrightarrow{u} = - \bigtriangledown p + \overrightarrow{J} \times \overrightarrow{B} \tag{14}$$

J provides an expression for current, $\overrightarrow{J} = e(n_i Z_i u_i - n_e u_e)$, u represents the average (fluid) velocity of species and p is the plasma pressure known as $p = 2nkT$, as we have 2 fluids (electrons and ions). If we are talking about plasma with a steady flow, then $\overrightarrow{u} \neq 0$ as a class potentially having equilibrium solutions. However, for purposes of this derivation we will be talking about static equilibrium, in which case we have $\overrightarrow{u} = 0$, everywhere in addition to $\frac{\partial}{\partial t} = 0$. We then have:

$$\bigtriangledown p = \overrightarrow{J} \times \overrightarrow{B}, \tag{15}$$

which is the most basic equation of MHD theory. We can further say that $\bigtriangledown \times \overrightarrow{B} = \mu_0 \overrightarrow{J}$, so we can also rewrite equation (15) as:

$$\bigtriangledown p = \frac{\bigtriangledown \times \overrightarrow{B} \times \overrightarrow{B}}{\mu_0}. \tag{16}$$

By invoking a vector identity, we get:

$$\bigtriangledown \times \overrightarrow{B} \times \overrightarrow{B} = - \bigtriangledown \left(\frac{B^2}{2} \right) + B \cdot \bigtriangledown B \tag{17}$$

The most right term in equation (17) is the one we are interested in:

$$\overrightarrow{B} \cdot \bigtriangledown \overrightarrow{B} = B\widehat{b} \cdot (B\widehat{b}) = B^2\widehat{b} \cdot \bigtriangledown b + B\widehat{\widehat{b}b} \tag{18}$$

By using the equation (18) we can construct only perpendicular components:

$$\bigtriangledown_{\perp} \left[p + \frac{B^2}{2\mu_0} \right] = \frac{B^2}{2\mu_0}\kappa, \tag{19}$$

where $\kappa = \widehat{b} \cdot \bigtriangledown \widehat{b}$ and is perpendicular to $\widehat{b}$. So, the above equation implies that in a simple equilibrium where the magnetic field lines are straight, one has a simple relationship:

$$\bigtriangledown \left[p + \frac{B^2}{2\mu_0} \right] = 0 \tag{20}$$

The term $\frac{B^2}{2\mu_0}$ is the magnetic pressure. For better representation of magnetic pressure, we can give an example: 4 T of magnetic field has a magnetic pressure equal to 10^6 Pa, or on the other hand plasma with a density of 10^{20} m^{-3} and a temperature 15 keV has a kinetic pressure of around 10^6 Pa. We can now define the dimensionless parameter β :

$$\beta = \frac{p}{\left(\frac{B^2}{2\mu_0}\right)}. \tag{21}$$

Where B is:

$$B = B_{max} \left(1 - \frac{a+b}{R_0} \right) \tag{22}$$

Where R is a major radius, a is a minor radius and b is the thickness of the blanket and shield [3]. Parameter b have to be taken into account, because due to the functionality of blanket and shied, they are located within the magnets. This increases the radius of magnets and lowers the magnetic field density, in respect to current. Because magnets can't be exposed to neutron and heat flux, this parameter b has to be considered when calculating magnetic field profile. In tokamaks $\overrightarrow{B}$ is taken as the toroidal field at the plasma's magnetic centre. The defined parameter β can also mean a local value or a mean value for the plasma, and it can be quoted as a pressure ratio for only one component of $\overrightarrow{B}$. So, it is customary to also introduce the poloidal beta β_{pol}, and toroidal beta, β_{tor}- in which total $\overrightarrow{B}$ is replaced with toroidal and poloidal magnetic field components, respectively. β would then be:

$$\frac{1}{\beta} = \frac{1}{\beta_{pol}} + \frac{1}{\beta_{tor}} \tag{23}$$

Also β is often expressed in terms of normalised beta. This then describes how close β is to establishing major MHD activity. This operational parameter is then defined for tokamaks as:

$$\beta_N = \beta \cdot \frac{aB_T}{I_p}, \tag{24}$$

where, B_T is the toroidal magnetic field, a stands for minor radius and I_p represents plasma current in units of MA. The operational parameter can be briefly described as a fraction of plasma pressure and magnetic field pressure. This parameter could theoretically be 1, but in reality, it's around 5%.

Higherβ leads to MHD instabilities, for example the ballooning mode. Some typical values are seen in Figure 3 [12, 13].

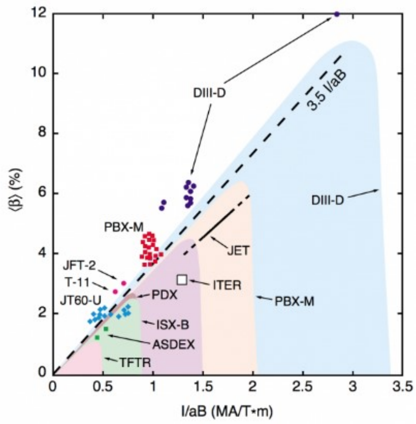

Figure 3. Typical values of beta parameter, for different fusion devices [13].

2.3 Plasma heating

After the start-up of plasma, a combination of ohmic and auxiliary heating must raise the plasma temperature to about $5-7$ keV. During this evolution, alpha power is negligible, and the heating power must overcome the losses due to thermal conduction. When we reach above $5-7$ keV, alpha power becomes dominant, heating the plasma to its final ignition temperature of $T = 15$keV. The most complicated is the first stage of heating [4]. In tokamaks we can consider four main ways of heating:

1. First let's consider ohmic heating. The ohmically-induced current is normally used for providing both a poloidal component of the magnetic field, which is a necessary component for a stable equilibrium, and for plasma heating. The toroidal current is induced by the central solenoid and the plasma acts as the secondary winding through the transformer principal. The latter is also the reason why the basic tokamak operational scenario is pulsed and not continuous. We call this the ohmic heating. We run several (even tens of) MA of current through the plasma. This is the simplest method in terms of the technology. However, the resistivity of a plasma decreases with temperature $\eta \propto \frac{1}{T^{3/2}}$. This is one reason why we need to apply different methods of heating, since when the temperature of plasma goes up, the resistivity decreases, so ohmic heating becomes less and less efficient. Analysis shows that for typical parameters in tokamak reactors, the maximum temperature achievable by ohmic heating is about $T < 3$ keV [2].

2. An additional source of heating is neutral beam injection (NBI). First, deuterium gas is ionised in a dedicated plasma chamber, these ions are then accelerated by an electrostatic field. After reaching the targeted kinetic energy, the ions are neutralised. The neutralisation is only partial and the remaining ions are deflected magnetically and sent to a dump. The neutralisation is mandatory both to respect the plasma neutrality and to be able to inject particles through the magnetic confinement wall. The neutrals can cross the magnetic field and reach the core plasma where they get ionised, transferring their energy to the bulk plasma by Coulomb collisions. The ionisation occurs either through ion or electron impact or through charge-exchange collisions. Most NBI are based on positive ion (H+, D+, ...) technology, while ITER will be operating with the negative species. For experimental purposes we have sufficient technology; the neutral beam systems are driven by positive ion sources, which have good efficiency up to 100 keV. ITER on the other hand requires 1MeV beams for better penetration, due to the high density and large size, and such beams are still being developed [2, 14, 15]

3. Another option for auxiliary heating is the use of radio frequency waves. We will describe this in detail as it will be applied in SPARC as the sole option for heating. High-frequency electromagnetic waves are launched into the plasma from an external source. There are several natural resonant frequencies of interest in a plasma: the cyclotron frequencies of the electrons and ions, and their cyclotron harmonics. Heating at the resonant frequencies of the electrons

is known as electron cyclotron heating (ECH) and for ions as ion cyclotron heating (ICH). We will focus on the working of ICH as it will be applied in SPARC. One of the difficulties of ICH is that antennas must be placed very close to the surface of the plasma to insure good coupling of the wave energy to the plasma. Therefore, the main problem is the proximity of the metal structure of the antenna to the plasma. High voltages are required to launch large amounts of power and these voltages can cause arcing and plasma breakdown near the antenna, both highly undesirable effects. A generator, transmission lines and an antenna are necessary for ion cyclotron heating. Each charged particle surrounded by a magnetic field B is circling around field lines with a gyromagnetic resonance (qB/m), where m represents the mass of the particle and q its charge. For a magnetic field of a few teslas, the gyromagnetic resonance is in the range of a few tens of MHz for light ions such as hydrogen isotopes, and between 100 and 200 GHz for electrons. ICH involves both ions and electrons being heated up. Electromagnetic waves with frequencies matching the gyromagnetic resonance of ions show in theory a high coupling with the plasma. But as these waves are far below the plasma cut-off frequencies (90 GHz), only evanescent waves should enter the plasma. Due to the strong toroidal magnetic field, the electrical conductivity is highly anisotropic and under specific polarisation conditions, electromagnetic waves can propagate radially. Such radial waves are called "Fast Alfvén magneto-static Waves (FW)". By putting the ICH antennas very close to the plasma, it is possible to excite fast radial waves, which will reach the plasma core where they will propagate. But ironically, they cannot couple directly with ions, even at the cyclotron frequency. This is because at the gyromagnetic resonance, FW show a circular right polarisation, opposite to the ions which are turning left. No transfer of energy is possible. The solution can take advantage of the fact that there are more than one ion species in the plasma. If one of the species is rare (for example a minority specie composed of traces of 3He), one can demonstrate the existence of a hybrid resonance, which is very close to the cyclotron frequency of the minority specie. Hybrid waves are combinations of longitudinal and transverse waves with $kB\perp$. They are a hybrid of two frequencies, one relating to the ion cyclotron frequency and one relating to the fundamental plasma frequency (equation (5)). The derivation of hybrid waves won't be discussed, but let's just examine the final solution:

$$c^2k^2 = \omega^2 - \sum_i \frac{\omega_p^2\omega^2}{\omega^2 - \Omega_i^2} - \frac{\left(\sum_i \frac{\omega_p^2\Omega_i}{\omega^2-\Omega_i^2}\right)^2}{1 - \sum_i \frac{\omega_p^2}{\omega^2-\Omega_i^2}} = 0 \tag{25}$$

Which has resonances at $1 - \sum_i \frac{\omega_p^2}{\omega^2-\Omega_i^2} = 0$. The lower hybrid resonances are used and are given by, $\omega^2 = \frac{\omega_p^2 i\Omega_i}{\omega_p e^2+\Omega_e^2}$,and the lower hybrid resonance is approximately, $\omega_L H^2 \sim |\Omega_e\Omega_i|$,where now i stands for ion, e for electron and Ω is e/m. Both resonances are coupled, and by exciting the hybrid resonance, which is possible, this reactive energy will transfer itself to the minority specie's cyclotron resonance, giving birth to supra-thermal particles, which give up their energy to the plasma by multiple collisions. This very efficient mode is called the minority species mode (MH) [15]. Therefore, the basic working process is: generator produces high-power radio frequency waves that are carried along a transmission line to an antenna located in the vacuum vessel, sending the waves into the plasma. This frequency from the waves can be absorbed in the plasma depending on the detailed geometry and launch conditions. The kinetic energy of the resonant ions increases and is transferred to the other plasma particles via collisions such that the plasma heats up.

4. There is one more way of heating, which we already briefly mentioned - alpha particle heating. It must balance the losses due to thermal conduction as described by the condition.

2.4 State-of-the-art-tokamaks

Let me present some of the current and future tokamak designs. The first one is the **JET** (Joint European Torus) fusion reactor. Currently JET is the largest operating tokamak in the world and the only one capable of operating with DT fuel. At the core, the fusion plasma is confined by means of strong magnetic fields and plasma currents, up to 3.5 T and 5 MA. The major diameter of the plasma torus is 3 meters, and the total plasma volume is 90 m^3. For now, it holds the world record for $Q = 0.65$ (D-T fuel), which still means they didn't gain more energy than they invested [16, 17]. The next interesting project is ITER. It's the most ambitious energy project considering fusion. Its located in southern France. The previously described JET, produced 16MW of fusion power from a total input of 24 MW (Q=0.67) in 1997. **ITER** is designed to produce a ten-fold return of energy (Q=10), or 500MW of fusion power. It will not produce electricity, but it will prepare the way for machines that can. 10 thousand tons of magnets, with a combined stored magnetic energy of 51 GJ, will produce the magnetic field of 5.3 T. The volume of the plasma contained in the centre of the vessel will be 840 m^3, which will be by far the largest to date [9]. The most promising facility after ITER is **DEMO**. DEMO stands for demonstration power plant, and it will be ITER's successor. With this device, the fusion will go from a science-driven lab to an industry-driven and technology-driven program. A key criterion for DEMO is the production of electricity. It will generate between 300 and 500 MW net electricity to the grid and operate with a closed fuel-cycle, meaning the tritium fuel will be self-sufficient.

3. SPARC

The MIT plasma science and fusion centre in collaboration with private fusion start-up Commonwealth Fusion Systems is developing a conceptual design for SPARC, a compact, high-field, net fusion energy experiment. SPARC will be at the size of existing mid-sized fusion devices, but with a much stronger magnetic field (around 2 or 3 times stronger than existing ones) [18]. Based on established physics, the device is predicted to produce around 25MW of power available for heating and $Q > 2$. SPARC could attain Q larger than 10 (Figure 4) and up to 140 MW, based on recent calculations. SPARC fits into an overall strategy of speeding up fusion development by using new high-field, high-temperature superconducting (HTS) magnets. Preliminary analysis has led to a conceptual design with a $1,65$m major radius and $0,5$ m minor radius operating at a toroidal field of 12 T and with a plasma current of 8.7 MA. Some of SPARC's predicted qualities are: energy confinement time $\tau_E \sim 0.77$ s, density of $n_e \sim 3 \cdot 10^{20}\ m^{-3}$, temperature as high as $T \sim 7$ keV and high-power density of around 7 MWm^{-3}, which is relevant to fusion reactors [18]. Its long-term goal is to introduce fusion power into the energy market in time to help mitigate global warming. The SPARC project aims to also show that the high magnetic fields possible with REBCO-based magnets allow one to reduce the size of fusion devices and facilitate rapid and lower-cost progress [18]. For instance, a minimum engineering current density of 700 A/mm2 at 20 K, 20 T is essential for the magnet system in SPARC [10].

As described, SPARC will be much smaller than ITER (approximately 10 times), hence the name SPARC- Smallest Possible Affordable Robust Compact (tokamak). Already at the time of ITER's design, it was recognised that without the constraint in magnetic field, it would be possible to build a high-gain device that was much smaller.

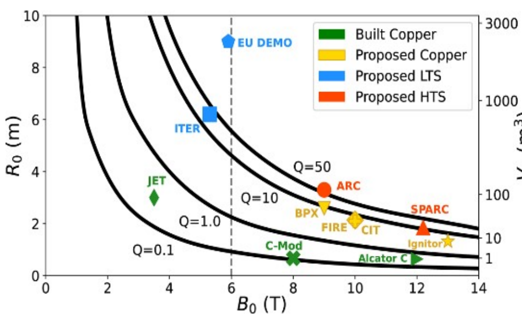

Figure 4. Modern fusion experiment/reactor facilities in Lawson criteria. With having bigger magnetic field, the radius gets smaller and so does the volume of plasma. DEMO, ITER, JET and SPARC are the ones which are in our interest [18]

A lot of designs were made based on higher magnetic field-smaller device, but these designs were deserted and never built. However, given what has been discovered since then, it seems likely that these machines would have achieved break-even (Q=1), or even a higher gain if they had been constructed. SPARC is the result of combining HTS technology with the same proven physics that led to the ITER design, but with only 2% of ITER's volume. Of course, HTS conductors have shown higher operating temperatures and higher critical currents. Commonwealth Fusion System (CFS) is using these new high temperature superconductors to build these kinds of new magnets. CFS is building a 20 T, large-bore magnet (for demonstration in 2021). Although the magnet will be producing 20 T, its operational temperature will be ranging between 10 and 30 K. Due to the increase in range of operating conditions made available through the HTS-based magnets, a new set of cryogenic fluids is being considered for forced flow cooling. The thermophysical properties of liquid helium, hydrogen, and neon are being analysed. With a target operating point defined, an experimental cryogenic flow loop has been designed with the purpose of verifying the high heat transfer rates required for the high-pressure, supercritical helium flow in the SPARC reactor. The flow loop uses a pressure differential to drive flow at a target mass flow rate of 46 g/s. To simulate a plasma pulse, the fluid flow is subject to heat fluxes up to 45 kW/m^2 for a minimum duration of ten seconds [11]. Moreover, all the above applications require large wire volumes, the SPARC device for example needing approximately 10.000 km of 4 mm wide wire. For most commercial applications of HTS, the wire cost is still too high for commercial viability and only robust, high-volume manufacturing can lower the price to a level compatible with the widespread commercial adoption of HTS [10].

Previously when we talked about heating plasma, we mentioned 2 different auxiliary ways of heating. SPARC is being designed to have up to 25 MW of 120 MHz ion cyclotron heating (ICRH) coupled into the plasma as the sole source of auxiliary heating. Its proposed as the sole auxiliary heating method based on its success in a few D-T operations. Among different methods, ICRH heating seems the most appropriate, practical and proven method that can heat high density and high field plasma in SPARC for both the pre-tritium operation and D-T operation. The pre-conceptual design has 12 four-strap antennas. Each antenna is powered by 2 MW transmitters. It will be the largest ever deployed system in a fusion experiment and must operate reliably and robustly as it will be, as stated, the only auxiliary heating method for SPARC plasma [15, 19, 20].

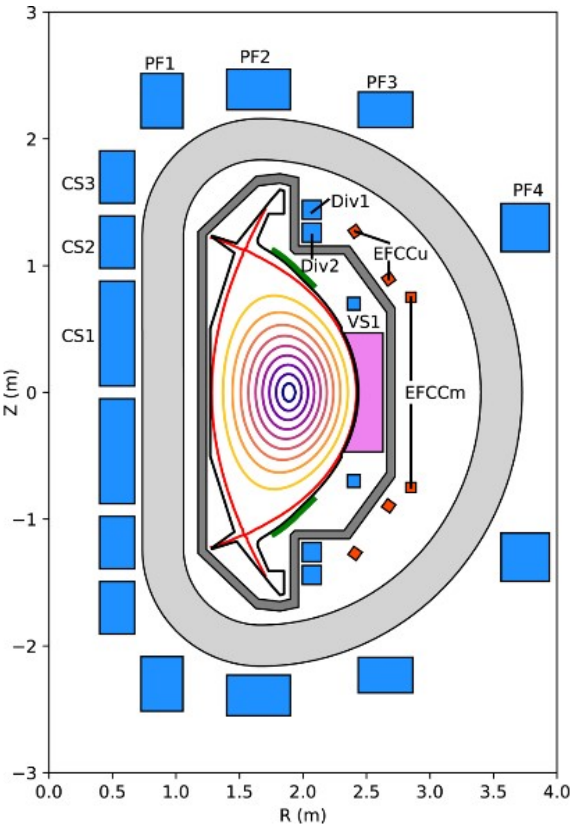

Figure 5. SPARC chamber structure including: central solenoid (blue), poloidal field (blue), toroidal coils (light gray), error field correction coils (orange-red), vacuum vessel (dark grey) and ICRH antenna (pink) [18].

The vacuum vessel in SPARC is double-walled, with space in between the walls for gas heating and cooling of the vessel. Almost half of the space between the vacuum vessel walls, as well as the space between the vacuum vessel and the toroidal field coils, is filled with neutron shielding material in order to reduce the nuclear heating of the superconducting magnets. The vacuum vessel has three ports at each toroidal location, one midplane port and a symmetric pair of off-midplane ports above and below. Situated at the bottom of the vacuum vessel, the divertor (Figure 5 - black surface represents divertor and first limiting surfaces) extracts heat and ash produced by the fusion reaction, minimises plasma contamination and protects the surrounding walls from thermal and neutronic loads. In SPARC the divertor is toroidally continuous and tightly baffled to contain neutral particles in the divertor volume. Both carbon and tungsten are currently under consideration as the material for plasma-facing components. The entire SPARC device is up-down symmetric to the maximum degree possible, enabling tests of symmetric double-null operation. SPARC's central solenoid consists of a total of three pairs of HTS upper and lower coils, labelled on figure as CS1, CS2 and CS3. The machine has four upper/lower pairs of HTS poloidal field coils outside of the toroidal field coils, also labelled as PF1-PF4, moving outward in the major radius. In addition, there are two pairs of copper coils that are internal to the toroidal field coils but external to the vacuum vessel. Since these coils are primarily used to actuate the divertor magnetic field, they are labelled Div1 and Div2 upper and lower. There is also a pair of vertical stability coils inside of the vacuum vessel, labelled VS1 upper and lower. Finally, there are three sets of picture-frame error-field correction coils, one upper, one lower and one midplane. On the Figure 5, are the

toroidal field coils coloured in light grey, the central solenoid and poloidal field coils are blue, the error field correction coils are orange-red, the vacuum vessel is dark grey and the ICRH antenna is pink. The vertical stability plates are green and the plasma separatrix is red. SPARC has 18 toroidal field coils in an attempt to balance the competing constraints of minimising magnetic field ripple and maximising vacuum vessel port width. It is being designed to have up to 25 MW of 120 MHz ion cyclotron resonance heating (ICRH) coupled into the plasma as the sole source of auxiliary heating. The central solenoid and poloidal field coil set will be capable of producing 42 Wb of magnetic flux to initiate and drive the plasma current. As the toroidal

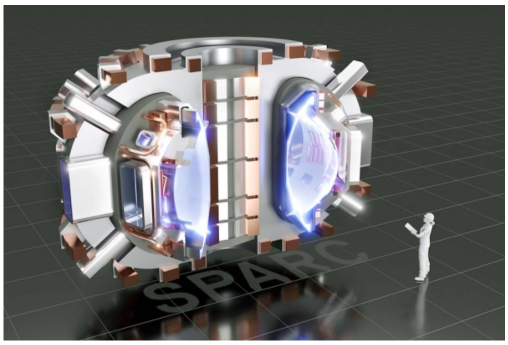

Figure 6. Virtual representation of SPARC's size with comparison to human [21]

field coils are perhaps the most novel aspect of the SPARC design, considerable effort has been dedicated to their structural, thermal and electromagnetic analysis. Similar efforts have been made for the central solenoid and poloidal field coils. Since ICRH is the only external heating source, its design has also progressed considerably in order to ensure that it will be able to reliably couple the necessary amount of power to the plasma. In addition, the effects of neutron heating have been modelled for the entire device, including determining the requirements for cooling of the superconducting magnets during D-T operation. The high volume-averaged fusion power density and tight radial build of SPARC mean that volumetric neutron heating of various components is of particular importance [13,18,19].

4. Conclusion

SPARC will be a compact, $Q > 2$ tokamak and is the logical next step on the path to timely and economical fusion energy. Conceptual designs are well on track, but construction work is yet to start - it is planned for 2021. SPARC still has a long way to go, but it has started on the right foot. A lot of pressure is on the HTS magnets. If this system doesn't perform as it should, the future of SPARC will be questionable. For now, all the experiments and development is going in favour of SPARC. It will be interesting to see the evolution of SPARC and its performance. The expectations are high, scientists are working hard, knowledge is there, and the current private-public business model seem to work - so it has all the right components to succeed and bring us one step closer to fusion power plants.

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