

New developments, plasma physics regimes and issues for the Ignitor experiment

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Abstract

The scientific goal of the Ignitor experiment is to approach, for the first time, the ignition conditions of a magnetically confined D–T plasma. The IGNIR collaboration between Italy and Russia is centred on the construction of the core of the Ignitor machine in Italy and its installation and operation within the Trinita site (Troitsk). A parallel initiative has developed that integrates this programme, involving the study of plasmas in which high-energy populations are present, with ongoing research in high-energy astrophysics, with a theory effort involving the National Institute for High Mathematics, and with INFN and the University of Pisa for the development of relevant nuclear and optical diagnostics. The construction of the main components of the machine core has been fully funded by the Italian Government. Therefore, considerable attention has been devoted towards identifying the industrial groups having the facilities necessary to build these components. An important step for the Ignitor programme is the adoption of the superconducting MgB₂ material for the largest poloidal field coils (P14) that is compatible with the He-gas cooling system designed for the entire machine. The progress made in the construction of these coils is described. An important advance has been made in the reconfiguration of the cooling channels of the toroidal magnet that can double the machine duty cycle. A facility has been constructed to test the most important components of the ICRH system at full scale, and the main results of the tests carried out are presented. The main physics issues that the Ignitor experiment is expected to face are analysed considering the most recent developments in both experimental observations and theory for weakly collisional plasma regimes. Of special interest is the I-regime that has been investigated in depth only recently and combines advanced confinement properties with a high degree of plasma purity. This is a promising alternative to the high-density L-regime that had been observed by the Alcator experiment and whose features motivated the Ignitor project. The provisions that are incorporated in the machine design, and in that of the plasma chamber in particular, in order to withstand or prevent the development of macroscopic instabilities with deleterious amplitudes are presented together with relevant analyses.

(Some figures may appear in colour only in the online journal)

1. Introduction

The main scientific objective of the Ignitor experiment is that of approaching and investigating for the first

time the ignition conditions of magnetically confined deuterium–tritium plasmas [1]. Advances made in the discovery of new confinement regimes and of relevant physical processes, as well as innovations in the technologies and

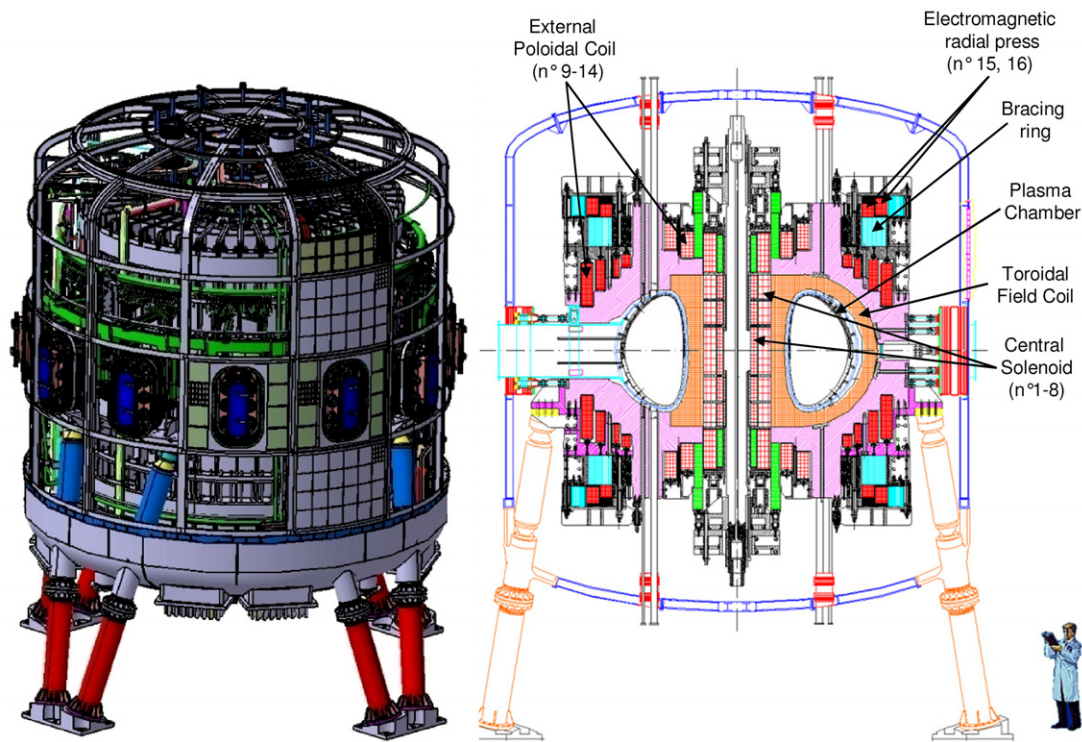


Figure 1. Three-dimensional (3D) model and cross-section of the Ignitor load assembly.

materials that have an impact on the machine concept have led to an evolution of the machine design as described in the following sections. Feasibility tests of key machine components have continued.

An important development is the start and the initial funding of the IGNIR (Ignition Goal Nearing by Italy and Russia) collaboration between Italy and Russia that involves the construction of the core of the Ignitor machine in Italy, and its installation and operation within the Trinita site (Troitsk). In particular, the construction of the main components of the core of Ignitor has been fully funded by the Italian Government. Therefore, considerable effort has been devoted towards identifying the industrial groups having the structures necessary to build these components. To this end, a complete set of ‘manufacturing’ drawings for the relevant components is being developed with the additional purpose of updating the existing detailed design of the machine (main dimensions of the plasma column: major radius $R_0 = 1.32$ m, minor radius $a = 0.47$ m, elongation $\kappa \cong 1.83$, triangularity $\delta \cong 0.4$), presented in figure 1. A joint analysis of the electrical connection of the machine to the electrical power supply system existing in the Trinita site in addition to the cryogenic system, etc is planned. This paper describes some of the numerous advances made regarding the cooling of the toroidal field magnets, the control system, the adoption of superconducting material for the largest coils of the poloidal field system, the design criteria for the plasma chamber, and the construction and test of a prototype ion cyclotron resonance frequency (ICRF) transmission line. For the physics, a summary of the simulation efforts carried out for plasma configurations with a double null and reduced parameters are reported together with an assessment of the possible control

of the thermonuclear instability by means of ion cyclotron resonance heating (ICRH) power deposition.

A parallel initiative to that for the construction of the Ignitor machine has developed that integrates the physics programme involving the study of plasmas in which a high-energy population is present, with ongoing research in high-energy astrophysics under the coordination of CIFS (Inter-university Consortium for Space Physics), with a theory effort involving the National Institute for High Mathematics and with the development of relevant nuclear and optical diagnostics involving the National Institute for Nuclear Physics (INFN) and the University of Pisa.

2. New developments on observed plasma regimes and physics issues

An important development in the physics of magnetically confined plasmas has been, even recently, the continuing discovery of new confinement regimes with different characteristics, signatures and level of powers for their access. Thus, a unified theoretical description of the modes excited at the edge of the plasma column characterizing a variety of ‘advanced’ regimes (the improved I-regime [2], the enhanced D-alpha EDA H-regime [3] and the ELMy H-regime [4]) has been proposed [5].

Given the primary goal of Ignitor, it is important for it to produce plasmas with relatively high values of the confinement parameter $n_0\tau_E$ and, at the same time, high degrees of plasma purity. This is the reason for having chosen the high-density L-regime as the reference for the most advanced regime of operation, following its original discovery by the Alcator experiments as it combines both requirements. In particular,

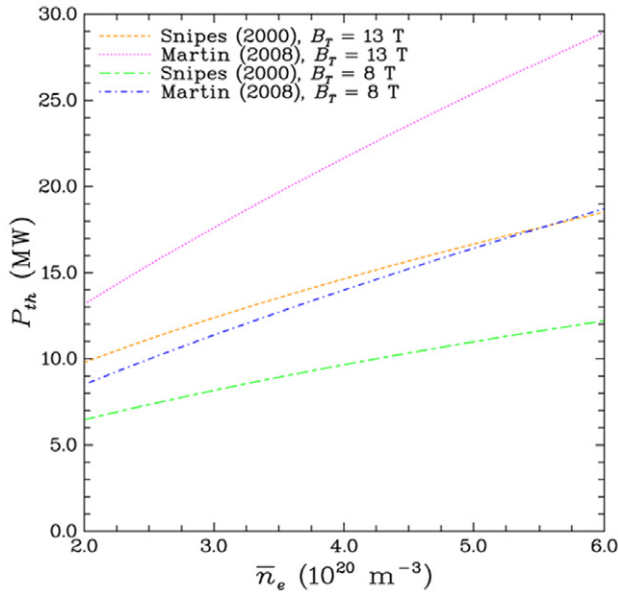


Figure 2. Total heating power threshold to access H regimes as a function of line-averaged density, for two values of the toroidal field B_T , according to two widely adopted empirical scalings, Snipes *et al* [7] and Martin *et al* [8].

the absence of particle transport barriers is a key feature to prevent impurity and ash accumulation in the plasma core. The recently investigated I-regime, besides featuring relatively high energy confinement times, involves the experimental observation, supported by the theory of its signature mode [6], that impurities do not accumulate in the main body of the plasma column [2]. On the other hand, the procedure to access this regime on different machines with different plasma configurations remains to be determined. The estimated heating power thresholds to access the H-mode [7, 8] indicate that improved confinement regimes can be attained in Ignitor over a considerable range of plasma and machine parameters with the available power (figure 2).

In fusion burning experiments such as Ignitor, where the dominant source of heating is provided by high-energy α -particles, novel phenomena are expected, including those relevant to transport processes. An important reminder of this is the recent interpretation in [9] of the 1997 JET α -particle heating experiment [10], where the observed surprising increase in the central ion temperature is attributed to a stabilization of ion-drift-wave turbulence by the energetic α -particles.

One of the key design criteria for Ignitor comes from the observation that in the temperature range 10–20 keV the D–T volumetric reaction rate $\langle \sigma_F v \rangle \propto T^2$ is a good approximation. Therefore, the plasma reactivity (reactions $\text{m}^{-3} \text{s}^{-1}$) is related to the poloidal field by

$$P_\alpha = \varepsilon_\alpha n_D n_T \langle \sigma_F v \rangle \propto n^2 T^2 \propto \beta_p^2 \bar{B}_p^4$$

as shown originally by one of us (BC). Here $\bar{B}_p$ is the average poloidal field and β_p takes values that are limited by stability considerations in practice within a relatively small range. Consequently, Ignitor is designed to produce record high values of $\bar{B}_p$. For this reason, as well as for reasons of confinement of both the thermal plasma and of the produced

α -particles, it is appropriate to operate with the highest plasma currents I_p possible ($I_p \cong 10 - 11$ MA). This involves having relatively low edge safety factors (e.g. $q_\psi \cong 3.6$) with considerable elongations ($\kappa \cong 1.83$). Thus, the possible effects of ‘internal’ modes with $n^0 = 1$ and prevalent poloidal harmonic $m^0 = 1$, that involve the most reactive region of the plasma column and produce sawtooth oscillations of the local plasma pressure, continue to receive considerable attention within the Ignitor programme. While we are following the numerical analyses that are being carried out (e.g. by Sugiyama [11]) on the non-linear evolution of the ‘ $m^0 = 1$ modes’ in different regimes, the adopted criterion for protection against their deleterious effects is that, at the highest plasma currents, the volume within the surface $q_\psi = 1$ should not exceed 10% of the total volume, while the values of β_p , representing the mode driving factor, do not exceed 0.25. Experimental evidence supports the expectation that the amplitudes and radial extensions of the emerging sawteeth are limited under these conditions [12]. In addition, for relevant simulations during most of the current ramp towards its maximum, the central value of q_0 is kept above 1.

The ‘fatal’ fraction can be defined as the amount of impurity, relative to the fuel ions, for which it is no longer possible to reach ignition, i.e. the fusion α -particle power cannot overcome the power lost by line radiation and bremsstrahlung, even in the absence of conduction losses. In practice, since the latter cannot be ignored, this limit will be lower. Before reaching the level of preventing ignition, the presence of impurities will make it more difficult to achieve, not only because radiation increases, but also as they have an effect of dilution of the fuel ions. The combined effect of $Z_{\text{eff}} > 1$ and increasing $\langle Z \rangle$ is that of shifting the ignition curve $n\tau$ versus T towards higher requirements. The effect is more severe at lower temperatures and for lower $\langle Z \rangle$. For example, at the reference ignition peak temperature of Ignitor, $T_0 \cong 11$ keV, it is virtually impossible to reach ignition if $Z_{\text{eff}} > 1.6$. This limit would occur for lower values of Z_{eff} if the dominant impurities were low- Z elements, such as He, C or O, but since the plasma chamber will be covered with a first wall made of molybdenum tiles, this is the element expected to give the most significant contribution to the plasma effective charge. The ‘fatal’ ratio of Mo to hydrogenic ions can be estimated between 0.75 and 1.0×10^{-3} .

An important aspect of the machine design is the absence of a separate divertor region. This choice was made on the basis of the observation that, at a high density, impurities are effectively screened from the main plasma, the lower edge temperature causes less sputtering from the wall, and radiation, ionization and charge exchange are all important in spreading and reducing the energy of the particles impacting on the walls. Most of the plasma energy is lost by radiation by intrinsic impurities in the outer region of the plasma column. As a result, pressure profiles tend to be more peaked relative to when low- Z elements are dominant, thus improving plasma performances. In the reference plasma equilibrium, the magnetic field topology is everywhere almost conformal to the first wall, which acts as an ‘extended limiter’ offering a large surface to spread the heat loads. The expected peak thermal power loads on the first wall in this configuration do not exceed 1.8 MW m^{-2} [13]. Thermal loads in the ‘double X-point’

configuration are expected to be locally higher but still within acceptable limits for the Mo first wall tiles. This is preferable to the presence of physical divertor coils in high-field regions where they would be subjected to severe mechanical stresses.

3. Structural philosophy for the Ignitor machine design

The adopted structural philosophy for the design of the machine magnets is being maintained. This aims at having a stress distribution as close as possible to an isotropic distribution in the most critical areas. In the framework of the ANSALDO industrial contribution to the detailed engineering of the Ignitor machine, extensive mechanical design and structural analyses for all the main components of the machine core, the so-called load assembly, are carried out. The complexity and extreme compactness of the Ignitor load assembly (see figure 1) require a 3D modelling of all the components that mechanically interact with each other. The adopted 3D model has led (i) to verify the assembly sequences of all machine components; (ii) to point out mechanical clearances and/or inconsistencies at the interfaces; (iii) to identify the best layout of the electric and fluidic lines and their supports; (iv) to position the in-vessel electromagnetic diagnostics; (v) to simulate the operation of the in-vessel remote handling system with virtual reality techniques of the remote maintenance of the first wall tile carriers, ICRH and Faraday shield components, and machine diagnostics. For this purpose, in parallel with the mechanical design activities, two-dimensional (2D)/3D drawings of each individual component are produced using the Dassault Systems CATIA V software; finally, they are assembled in a single 3D CATIA model of the load assembly. At this stage, the electro-fluidic and fluidic lines, which supply electric current and helium gas to the coils, are also added and mechanically integrated with the previous components.

The machine central post, the central solenoid, the shape and equilibrium coils, the plasma chamber and the first wall system supported by the surrounding mechanical structures, the vacuum cryostat and the supported polyethylene–boron composite sheets for neutron shielding are structurally analysed and mechanically verified to confirm that they can withstand, with proper margins, both the normal and off-normal operating loads (described in section 5) for the maximum current scenarios at 11 MA/13 T (see, for example, the results concerning the stress distribution in the C-clamps in figure 3), the reduced scenarios at 6 MA/9 T and 7 MA/9 T, the baking operations of the plasma chamber and the first wall. The scenarios that involve ‘extended limiter’ plasma configurations are up–down symmetric and the outer magnetic surfaces closely follow the cavity of the toroidal magnet over most of the vertical cross-section. This provision minimizes the out-of-plane forces produced by the plasma current and acting on the toroidal magnet.

The structural analysis of two other scenarios concerning plasma equilibrium configurations of the double X-point type, which may extend the operation of the machine into improved confinement regimes, is carried out. In these cases, the out-of-plane forces are expected to increase, raising concerns about the structural resistance of the load assembly. The

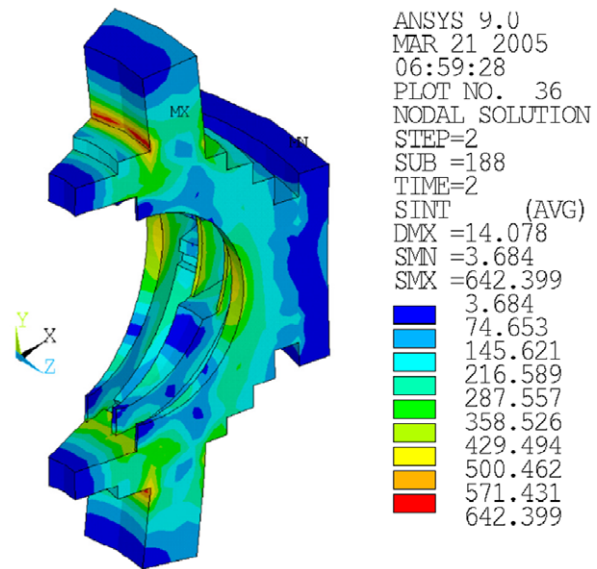


Figure 3. Tresca stress intensity (MPa) in the C-clamp at 4.65 s of the toroidal field flat-top for the 11 MA/13 T scenario under normal operating conditions.

maximum plasma current is reduced to 9–10 MA while the on-axis magnetic field is maintained at $B_T \approx 13$ T (other scenarios at even lower parameters are discussed below). The results of the structural analysis that has been performed demonstrate their feasibility from the structural point of view, as well as a better torsional stability of the machine, essential to resist the out-of-plane loads on the toroidal magnets, relative to the maximum extended limiter scenario, that derives from lower centrifugal forces produced by the central solenoid coils and to the extended operation at maximum current of the magnetic press, up to the end of the pulse flat-top.

A global seismic analysis of the load assembly with SSE/OBE is also performed in order to verify that the machine is able to withstand such external events.

4. Duty cycles and new cooling channel configuration

The operating temperature of the OFHC copper coils and the mechanical structures of the machine at the start of the current pulse should be 30 K, and 293 K for the plasma chamber and the first wall. After the pulse, the toroidal field coils reach temperatures up to 230 K, and the ohmic (OH) transformer coils reach temperatures close to 220 K for the most advanced parameter scenario and 120 K for the reduced scenarios, the external coils being always under 100 K; therefore a helium gas cooling system is required. A feasibility study of the refrigeration plant is positively completed by LINDE Kryotechnik AG for the most advanced operation scenario. Additional calculations are carried out to verify the adequacy of the plant also for the reduced scenarios with more stringent cooling times between shots.

Taking into account that the temperature propagates through a diffusion process, a new configuration for the cooling channels in the toroidal magnet is designed that minimizes the distance between the region (adjacent to the central solenoid) where the maximum temperature is produced and the region

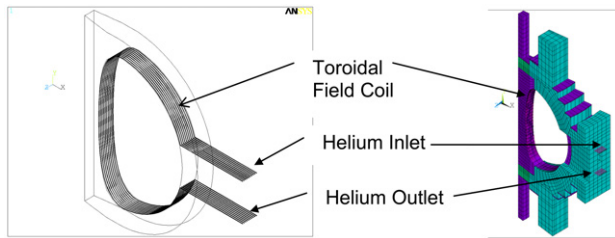


Figure 4. Layout of the TFC-modified cooling system and the relevant finite element model.

that is reached by the coolant. With the new distribution of the cooling cavities (figure 4), the structural integrity of the magnet plates is maintained and the stress distribution in the C-clamps that contain the toroidal magnets is improved. Accordingly, the cooling channels are machined on the 240 copper turns, on the plasma side of the plates and all around their poloidal length; the helium inlet is above the equatorial plane, the helium outlet under the equatorial plane, and no more cavities are required to go through the C-clamp structure. This is a positive secondary result of the present solution. The 10×4 mm grooves of the helium flow channels are machined at the inboard wedged equatorial sector and at the outboard flat equatorial sector, and are closed by electron beam welded cover plates.

Detailed mechanical verifications are made to demonstrate the structural integrity of the coil turns. Finite element models of some insulated turns for the inboard and outboard regions complete with the grooves and the cover plate are set up and run with the reference scenario loads; the channels cause limited localized stresses not too significant for the structural integrity of the turns. Just outside the grooves, the stress distribution remains essentially unchanged relative to the previous, ungrooved copper turns.

For the most extreme operation parameters of the toroidal coils (13 T) considered, the cooling time of 5 h is reduced by a factor of two, for the reduced scenarios (6–7 MA/9 T) cooling times under 1 h are permitted by the new solution.

5. Design of the plasma chamber and disruption loads

The machine component that receives the main attention is the plasma chamber system that includes the protective provisions against forces generated by disruptions, a first wall capable of restoring high thermal loadings, the incorporated ICRH system and electrical diagnostics, and the various access ports with their complexities.

The plasma chamber is a key component that has to sustain the electromagnetic and thermal loads due to plasma off-normal events, without losing the required degree of high vacuum. Moreover, the chamber plays a role in an effective stabilization of magnetohydrodynamic (MHD) modes near the plasma boundary, given its closeness to the plasma. In particular, according to the present design, the plasma chamber is made of Inconel 625 for its excellent mechanical properties; in addition, its relatively low electrical resistivity allows for a reduction of the poloidal flux consumption during the plasma start-up. Due to the pressing requirements on the mechanical and geometric integrity of the chamber, a complete revision

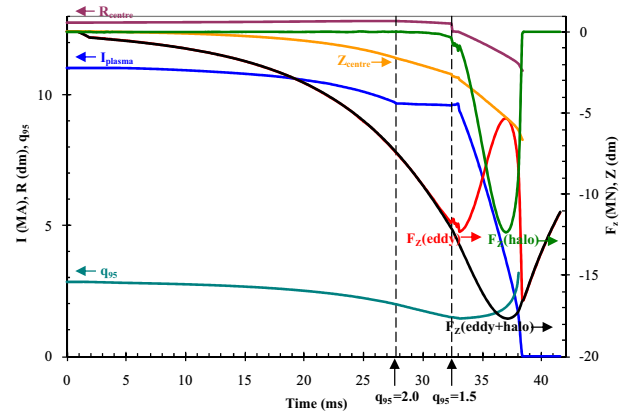


Figure 5. Time evolution of a VDE disruption in Ignitor at 11 MA/13 T.

of the first wall, plasma chamber and their supporting systems is performed, in collaboration with ANSALDO Nucleare, to account for the loads arising from plasma vertical disruption events (VDEs) and associated halo currents (HCs), on the basis of data obtained by existing machines. A full current VDE is, in fact, the worst expected event for the plasma chamber integrity, due to the fast (a few ms) transfer to the chamber of the whole plasma thermal and magnetic energy with an intrinsic poloidal (and possibly toroidal) asymmetry.

The worst expected VDE is simulated with the 2D finite element MHD code MAXFEA [14] (figure 5); the EM loads and induced currents obtained from these simulations are applied to the mechanical analysis performed with ANSYS. This fast VDE begins with a loss of vertical stabilization capability (due, for instance, to a failure in the vertical control system or to the appearance of unstable, uncontrollable MHD modes) that results in a slowly downward (or upward) moving plasma. The plasma movement is partially stabilized by the eddy currents induced on the plasma chamber, with a characteristic vertical instability growth time (about 15 ms) that depends on the vessel geometry and resistance. The plasma current slowly decreases while the plasma column moves and the edge q_ψ ('safety factor') decreases due to the plasma shrinking. When q_ψ at the edge goes below 2, the probabilities that microscopically unstable modes are excited increases and eventually the plasma suddenly loses all its thermal energy in the so-called thermal quench (TQ). Since the vertical forces on the plasma chamber are proportional to the asymmetry of the eddy current distribution, the vertical forces are greater when the plasma displacement just before the TQ is greater.

To model this worst-case scenario in the MAXFEA code, the TQ is forced when q_{95} , the safety factor at 95% of the flux, goes below 1.5. After the TQ, the plasma is relatively cold, its resistivity is high, and therefore the available poloidal flux is no longer sufficient to sustain the energy dissipated as the Joule effect in the plasma column: the plasma begins to lose its current in the current quench (CQ). The CQ duration is variable; its minimum, evaluated from the ITER disruption database, involves the plasma poloidal area, which is indicative of the plasma resistivity and hence of the post-TQ plasma temperature. This value corresponds, in Ignitor, to a linear

rate equal to about 2 MA ms^{-1} (i.e. 3 ms duration from 80% to 20% of the plasma current).

When the plasma boundary comes in contact with the wall during a disruption, the poloidal HCs flow from the plasma into the vessel and then again to the plasma along open field lines that are in thermal contact with the wall. The HCs can also be toroidally asymmetric (like the modes in the plasma boundary), applying relevant radial forces to the vessel, as well as axisymmetric pressures. The ratio of halo current to the total plasma current is inversely proportional to the CQ rate: fast CQ induces large eddy currents and small HCs; conversely, slow CQs with large plasma displacement at near full current are associated with larger HCs [15]. The toroidal distribution of the EM loads due to HCs was evaluated with MAXFEA and distributed toroidally as $(1+\cos\alpha)$, where α is the toroidal angle, assuming a toroidal peaking factor (TPF) equal to 2. A simple $I_p B_T R$ scaling to Ignitor from the values observed in JET disruptions [16] confirmed the value of about 10 MN for the net horizontal force.

A revision of the design has led to a significant increase in the vessel thickness up to 26 mm at the inboard region, 36 mm at the top/bottom regions where the HCs flow during a VDE, and 52 mm at the outboard region where the 12 equatorial ports are attached. The two narrow poloidal strips of the chamber connecting each 30° module to the adjacent one are maintained at a constant 26 mm thickness for best welding execution, and an appropriate toroidal tapering from the variable thickness to the 26 mm welding chamber is used. The increase in thickness minimizes the vertical displacement of the plasma chamber to avoid collisions with the toroidal coils, and the high stresses on the shell due to the HC; a preliminary manufacturing route was also established to verify the feasibility of the design, taking into account also the possibility of new developments in disruption mitigation techniques.

The plasma chamber is designed to move freely, to allow for thermal expansion, inside the volume defined by the toroidal field coils and their supporting C-clamps. Vertical and radial supports (acting in both centripetal and centrifugal directions with high stiffness) at the end of the radial ports are connected to the C-clamp structure and provide mechanical support to the chamber. During the plasma current pulse the connections will be locked by clamping hydraulically operated sleeves.

Non-linear dynamic transient analyses with elasto-plastic material [17] are performed by means of the ABAQUS code [18] on the variable thickness plasma chamber with eddy, halo and net horizontal load conditions. The maximum Von Mises stresses produce a plastic deformation in the top and bottom area and on the attachments between the vessel and ports. The peak residual plastic deformation located in a discrete area of the welding zone is found to be about 0.83% and the maximum residual plastic deformation located in the plate wall 0.23%. Limits for inelastic strains, according to the ASME's Boiler and Pressure Vessel Code, are satisfied.

5.1. Disruption avoidance and mitigation strategy

The active control strategy aimed at avoiding/mitigating disruptions is based on the design of a high performance plasma parameter control (H3PC) to operate with plasma

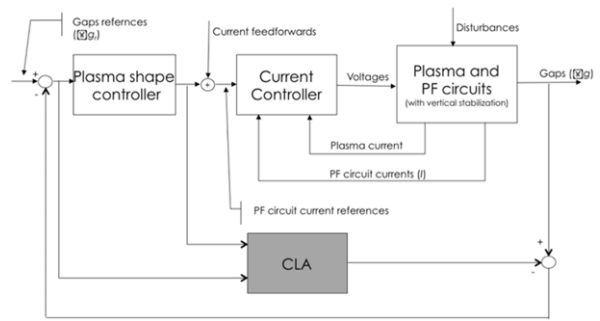


Figure 6. The overall plasma shape controller in Ignitor with the CLA system.

parameters close to their limits. Moreover, if a set of disruption predictors is available then the control system should be able to actively modify some plasma parameters in the presence of disruption warnings.

The plasma control system must simultaneously regulate a number of plasma parameters, giving rise to a complex integrated control system. In addition, this requires the continuous monitoring of various plasma performance indicators (e.g. a relevant safety factor, the internal inductance, the plasma density relative to the density limit, the status of MHD activity), which in these cases play the role of disruption precursors. The possibility of implementing this strategy is strongly dependent on the precursor advance time, and on the availability of sufficiently fast actuators.

In this phase of our study we do not consider the availability of robust disruption precursors, so we focus our attention on the design of a H3PC system. In this framework, the first step is the investigation of a current limit avoidance (CLA) scheme [19, 20].

The CLA exploits the redundancy of the PF coil system to obtain ‘almost the same plasma shape’ (at steady state) with different combinations of PF coil currents. Hence, in the presence of disturbances, it aims at avoiding that the currents in the PF coils reach their limit values, by using an optimal allocation procedure, which changes the PF coil current distribution without affecting the plasma shape substantially. This, in principle, broadens the Ignitor safe operating region, by allowing the overall control system to counteract larger disturbances (specified in terms of poloidal beta and internal inductance variations).

The current allocation is implemented by inserting the CLA (which corresponds to the grey shaded box in figure 6) between the plasma shape controller and the current controller. The output of the CLA modifies the PF current requests made by the shape controller. It also hides the plasma shape change to the shape controller in order to prevent the controller from reacting to these changes.

6. ICRH heating system and antenna prototype testing

The ICRH system is an important component of the Ignitor experiment and all efforts have been made to ensure that its design takes into account the construction experience gained in the most advanced laboratories. The system is conceived with a modular and flexible configuration to optimize the

coupling of the radio frequency (RF) power to the plasma in different regimes and over a wide range of machine parameters. Six of the twelve large horizontal ports are allocated to auxiliary heating systems, but only four ICRH antennas will be installed initially (one antenna per port). This configuration will be powered by two modules, each serving 2 four-strap antennas. Each ICRF power module consists of four high power generators providing a total power of 8 MW at the central frequency of 80 MHz (2 MW per generator). This value is reduced to 4 MW when the operating frequency is increased at values close to the upper limit of the generator band (120 MHz). The vacuum transmission system is capable of transferring 0.4 MW per strap at 115 MHz, the optimal frequency at $B_T = 13$ T, that is 1.6 MW per port-antenna, keeping the electric field in the vacuum region below 5 kV cm^{-1} . The RF configuration of the modules allows a full phase control (toroidal and poloidal) of the straps through a phase-locked loop (PLL) phase control.

In particular, for the most advanced performance scenario in D-T, the ICRH pulse is applied at a time when the magnetic field is increased from 9 to 13 T during the initial 4 s of the plasma current ramp-up to 11 MA. This requires the possibility of tuning different frequencies during the discharge in order to maintain the absorption layer in the central region of the plasma column for the control of the plasma temperature, and to accelerate the achievement of ignition in the ‘extended first wall configuration’. Since low levels of power are required, different antennas tuned to different frequencies can be utilized, up to 115 MHz that are optimal at 13 T. In the scenario at reduced parameters $B_T = 9$ T in the X-point configuration, in contrast, full power coupling in the range $f_{\text{RF}} \approx 80\text{--}95$ MHz is required to help the transition to the H-mode regime. The positions ρ/a (in the normalized plasma radius) of the main resonant harmonics of hydrogen, deuterium, tritium and ^3He , as functions of the magnetic field, are obtained by the expression

$$\rho/a = A \times \left[15.2 \times \frac{N_{\text{harm}} Z_i m_p B_0(\text{T})}{m_i f_{\text{RF}}(\text{MHz})} - 1 \right], \quad (1)$$

where A is the machine aspect ratio $R/a = 2.8$, N_{harm} is the harmonic number, Z_i is the charge of the species, m_p/m_i is the proton/ion mass ratio, B_0 is the on-axis magnetic field and f_{RF} is the applied frequency.

In general, the best ICRH scenario to achieve an efficient absorption rate is the minority heating: in the case of Ignitor, when a small fraction of ^3He (1–2%) is added to the D-T mixture, the first pass absorption on the ions near the centre of the plasma column is very efficient. The remaining coupled power is damped on the electrons over a broad radial interval of the plasma column. The fundamental harmonic of ^3He is located in a radial interval $-0.5 < \rho/a < +0.5$ when the magnetic field is varied from 9 to 13 T for $f_{\text{RF}} = 115$ MHz. This is the optimal choice for the ‘ignited scenario’ at full field because it satisfies both the requirement to remain in the frequency domain ≤ 120 MHz of the RF power generators, and the not-too-bad dislocation of the harmonic positions for the various ion species to avoid spurious absorptions at the plasma edge, as summarized in figure 7 [21].

A full size prototype of the 25Ω (actual value) vacuum transmission line (the part of the antenna which transfers

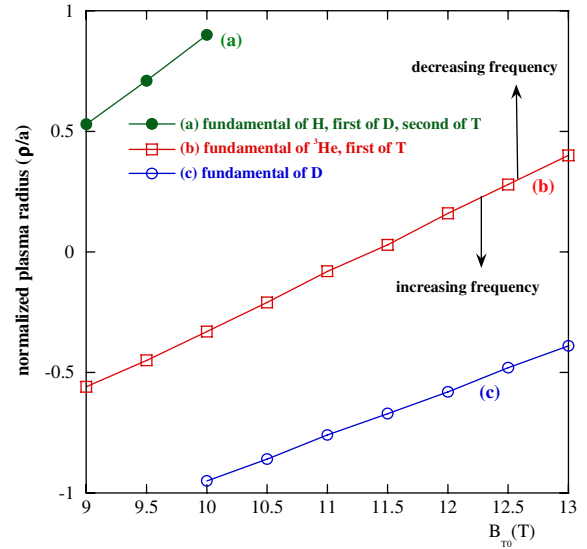


Figure 7. Resonance layer position as a function of the toroidal magnetic field for (a) fundamental, second and third harmonic of hydrogen, deuterium and tritium, respectively, (b) fundamental of ^3He and second of tritium, and (c) fundamental of deuterium. The frequency is 115 MHz; for higher frequencies the three lines (a), (b), (c) in the figure move downwards, the blue line tends to move outside the plasma boundary; for higher frequencies, the curves move upwards and, in this case, it is the green curve to move outside the margins.

the power into the plasma and separates the vacuum of the plasma chamber from the outside port flange), between the port flange and the antenna straps, with the external support and precision guiding system, is manufactured and tested to verify the antenna design and the associated installation procedures (figure 8).

The innovative quick latching system located at the end of the coaxial cables was successfully tested, providing perfect fitting with the Be-Cu spring electrical contacts. The mechanical stability and the smooth movement of the VTL coaxial system on its mechanical supporting structure were verified. A fully equipped VTL was pumped down to 6×10^{-7} bar, detecting a total leakage lower than $2 \times 10^{-8} \text{ mbar l s}^{-1}$.

High voltage tests, under different vacuum and pressure conditions, were carried out to verify the capability of the system to withstand the high voltages (about 12 kV) and electric fields (about 5 kV cm^{-1}) associated with the maximum RF power in the presence of a strong mismatch with the plasma load. The tests were started at room pressure, increasing the voltage up to 16 kV and continued under high vacuum (2×10^{-7} bar), up to 12 kV. In both cases no discharges were detected. A final test to check the electrical performances of such critical component was successfully carried out at a low pressure (3×10^{-5} bar) and 12 kV, to simulate the plasma neutral pressure in the region of the feedthrough, without discharges.

7. Superconducting poloidal field coils (magnesium diboride)

An important step for the Ignitor programme is the adoption of the superconducting material MgB_2 for the largest poloidal

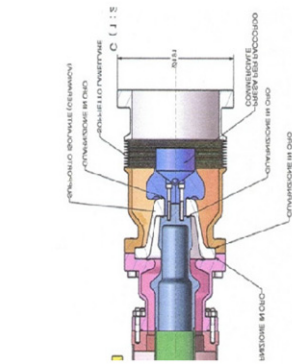
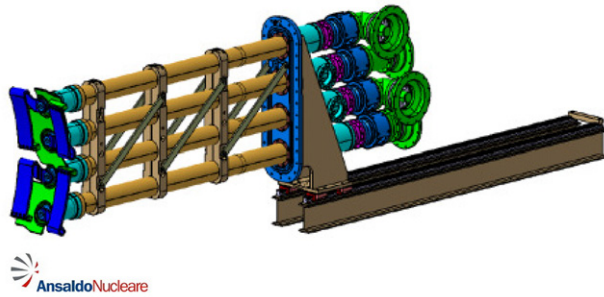


Figure 8. Top: vacuum transmission line and sliding support (actual prototype and conceptual drawing) for a single four-strap antenna; bottom: front-end ceramic feedthrough.

field coils (P14 coil, diameter of about 5 m). These coils will be cryocooled at the operating temperature of 10–15 K, a value compatible with the He-gas cryogenic cooling system of the machine. The P14 coils are designed to carry 34.7 kA, with a magnetic field, at the coils' location, in the range 4–5 T. These requirements can be met by commercially available MgB_2 strands. Different cable configurations are analysed in order to find the best compromise among an efficient cooling of the MgB_2 conductors, the feasibility of the manufacturing process and the cable cost.

The present cable design, with a diameter of 31.55 mm, includes 336 MgB_2 multifilamentary strands wrapped around a copper tube for the He-gas flow. The elementary strand is round in shape, with a diameter of 1.13 mm, and it is composed of 37 filaments inside a Ni matrix (figure 9).

The technical feasibility of these coils, as well as their stability and protection in the unlikely event of a quench, has led to decide that the multifilamentary strand will be electroplated with a $30\text{ }\mu\text{m}$ copper layer to ensure

proper stability. The strands are characterized by measuring the transport critical current as a function of the applied magnetic field at different temperatures (figure 10), meeting the requirements for the PF14 coil.

The adoption of an MgB_2 busbar system similar to the one under production for CERN is envisioned. The construction of a scaled version (1:3) of the PF14 coil is planned, with the same (superconducting) current density in order to verify the manufacturing process of the strands, cabling and coil assembly, and to test it in the same magnetic field in which the coil is expected to be imbedded in Ignitor.

8. Plasma parameters and relevant simulations

Numerical simulations of the plasma that Ignitor can produce are carried out both for maximum values of the plasma current and for lower values at which the duty cycle of the machine is considerably higher than that possible for the most extreme plasma parameters. The thermal energy transport coefficients adopted for the simulation of the plasmas that Ignitor can produce are based on experimentally observed and investigated regimes and comply with the principle of 'profile consistency' [22] implying the existence of self-organization processes, as shown by existing experiments, leading to realistic profiles. Typically, these coefficients are monotonically increasing functions of the plasma radius.

A special version of the JETTO transport code [23] was developed and is employed. Two kinds of equilibrium plasma configurations for the most advanced operations are considered: (i) an 'extended limiter' configuration ($I_p \lesssim 11\text{ MA}$) with the outermost magnetic surface adapting to that of the first wall (Mo tiles); (ii) a 'double null' (DN) configuration ($I_p \lesssim 10\text{ MA}$) with two up-down symmetric X-points located just outside the plasma chamber. The relevant simulations show that ignition achievement depends on specific combinations of density, temperature and energy confinement. The first column of table 1 reports the plasma parameters obtained when the optimal density to reach ignition with the adopted heating processes [24] is chosen and a transport model with two components is used: the so-called 'Coppi–Mazzucato–Gruber' diffusion coefficient that is relevant to the OH heating phase [25] and a coefficient representing the effect of the ubiquitous mode (also known as collisionless trapped electron mode) found originally by Coppi and Rewoldt [26]. More recently, the so-called 'Coppi–Tang' transport model that has similar characteristics has gained attention as it is based on the gyro-Bohm scaling [27] and the principle of profile consistency. In particular, this result has been validated with data from the TFTR experiment [28] before being used for ITER start-up scenario predictions [29]. The other columns include results obtained for the 10 MA configuration, and for some 'reduced' parameter scenarios (e.g. $I_p = 5$ and 6 MA, and $B_T = 8\text{--}9\text{ T}$) in a DN configuration. In the first two columns 'ignition' (i.e. the condition when the α -particle heating equals the power lost by the plasma) is achieved just at the end of the current ramp [30], in the other ones the data refer to the end of the current pulse flat-top. At ignition, the plasma temperature is on the rise, and the rate of increase of the stored energy $\partial W_{pl}/\partial t$ roughly equals the OH heating power. In the case of DN configurations, a reduction in

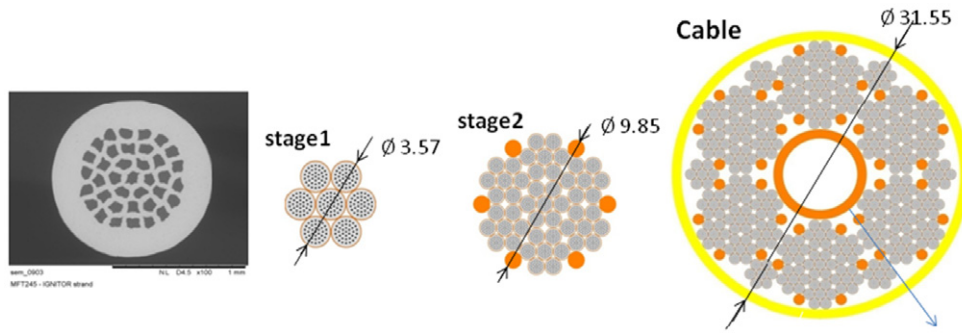


Figure 9. Cross-section of the Ignitor multifilamentary single strand and their assembly.

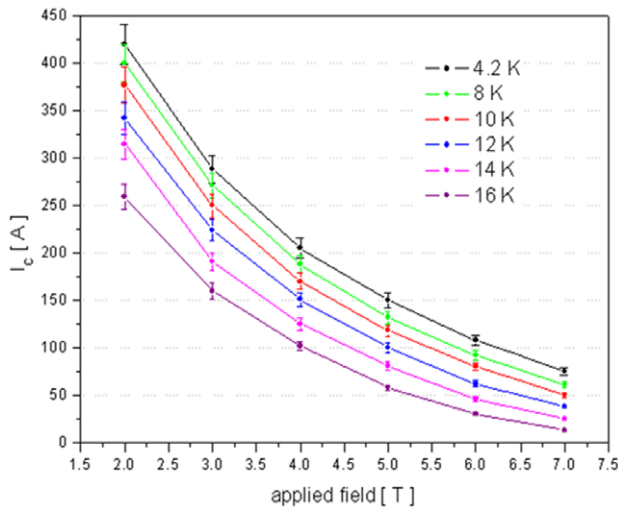


Figure 10. Superconducting critical current measurements of MgB₂ strand as a function of the applied magnetic field at different temperatures.

the thermal conductivity coefficient is applied when the total heating power (including OH, ICRH and α -particle) exceeds the threshold value required to access the H-mode regime. We have also analysed the possible sawtooth activity triggered when the peaking factor of the plasma pressure, defined by $p(0)/\langle p \rangle$, and the fractional area enclosed by the $q = 1$ surface exceed certain considered critical values.

9. Thermonuclear instability control via ICRH

As a complement to the transport analyses, the pertinent physics of the ICRH is analysed for all the operating scenarios of interest. In particular, ICRH is planned as a tool to control the plasma temperature, to accelerate the achievement of ignition in the extended first wall configuration at $B_T = 13$ T/ $I_p = 11$ MA, and to facilitate the H-mode transition in the double X-point plasma configuration ($B_T = 9$ –13 T, $I_p = 6$ –10 MA). In the fully ignited scenario, the main goal concerns essentially the possibility of controlling the thermonuclear instability, which develops after reaching ignition. The expected ability of Ignitor to achieve ignition with high peak densities ($n_0 \cong 10^{21}$ m⁻³) and relatively low temperatures makes it possible to investigate the thermonuclear instability as self-heating of the plasma by the fusion-produced

particles can lead to a significant rise in the plasma temperature and, with this, to an increase in its pressure. Then, internal plasma modes may be excited and saturate the thermonuclear instability at acceptable levels without external intervention. In the case where an internal process may not be effective, a scenario is considered whereby Ignitor is led to operate in a slightly sub-critical regime, i.e. the plasma parameters are so chosen that the thermonuclear heating power is slightly less than the power lost, and a small fraction of ³He is added to the optimal deuterium–tritium mixture. The difference between power lost and α -heating is compensated by additional ICRH that should be able to energize the minority species (minority ion heating scheme) directly, which can transfer the power to the main plasma species by collisions. In order to study this problem for Ignitor an energy balance equation is solved that includes the α -heating power, the additional ICRH and the radiation losses (bremsstrahlung) [31]. In this calculation we consider electrons and ions to be strongly coupled and, consequently, the plasma is characterized by a single plasma temperature ($T_e = T_i$). This hypothesis is very well justified for the Ignitor plasma.

The ICRH term is evaluated by calculating the power density absorbed by the plasma by numerically solving the full wave equation in 2D geometry coupled with a 2D (in velocity space) Fokker–Planck solver. TORIC [32] and SSFPQL [33, 34] are the numerical tools used for such kind of analysis.

As stated before, the power deposition profiles on ions and electrons are obtained, in a scenario where a small fraction of ³He (1–2%) is added to the D–T mixture, which improves the wave absorption on ions near the centre of the plasma column, while a substantial fraction of the coupled power is damped on the electrons in a broad radial interval of the plasma column. The distribution of power absorption among ionic species and electrons depends on the plasma temperature and it has been numerically derived by means of the full wave code. The best frequency to operate at in this ‘ignited scenario’ where the magnetic field is ramped up from 9 to 11 T is 95 MHz and from 11 to 13 T is 115 MHz. For example, calculations of the power absorption level for three different external magnetic fields (11 T, 12 T and 13 T), corresponding to three different times of the discharge evolution, for a plasma formed by D (50%), T (50%) and a small fraction of ³He (<2%), density and temperature between 5×10^{20} m⁻³ and 9×10^{20} m⁻³, and from 4 keV to 6 keV, respectively, show that the peak of absorption is located at the fundamental harmonic of ³He and second harmonic of tritium. In figure 11, the RF power

Table 1. Example of plasma parameters for several operational scenarios (JETTO Code).

Toroidal plasma current I_p (MA)	11	10	6	5
Toroidal field B_T (T)	13	13	9	8
Major radius R_0 (m)	1.32	1.35	1.35	1.36
Minor radius $a \times b$ (m ²)	0.47×0.86	0.46×0.80	0.44×0.76	0.45×0.74
Elongation k	1.83	1.74	1.73	1.64
Triangularity δ	0.4	0.44	0.35	0.30
Plasma volume V_0 (m ³)	10	8.7	8.2	8.1
Central electron temperature T_{e0} (keV)	11.5	15.8	10.3	11.2
Central ion temperature T_{i0} (keV)	10.5	13.6	9.8	10.3
Central electron density n_{e0} (m ⁻³)	9.5×10^{20}	8.7×10^{20}	5.5×10^{20}	5.8×10^{20}
Average electron density $\langle n_e \rangle$ (m ⁻³)	4.8×10^{20}	2.9×10^{20}	1.9×10^{20}	1.9×10^{20}
Central plasma pressure p_0 (MPa)	3.3	4.3	1.7	1.94
Alpha density parameter n_α^* (m ⁻³)	1.2×10^{18}	2.3×10^{18}	4.8×10^{17}	5.8×10^{17}
Average alpha density $\langle n_\alpha \rangle$ (m ⁻³)	1.1×10^{17}	4.1×10^{17}	6.1×10^{16}	9.7×10^{16}
Fusion alpha power P_α (MW)	19.2	19.8	3.5	4.9
Plasma stored energy W (MJ)	11.9	11.5	4.9	5.5
OH power P_{OH} (MW)	11.2	3.5	2.1	1.1
ICRF power P_{ICRH} (MW)	0	2.2	5.0	6.0
Bremsstrahlung power loss P_{brem} (MW)	3.9	2.3	0.7	0.77
Poloidal beta (β_p)	0.20	0.22	0.26	0.40
Toroidal beta (β_T) (%)	1.2	1.2	1.2	1.7
Edge safety factor $q_\psi = q_\psi(a)$	3.5	3.5	3.3	3.4
Bootstrap current I_{bs} (MA)	0.86	1.1	0.62	0.98
Poloidal plasma current (MA)	$\cong 8.4$	$\cong 7.9$	$\cong 4.2$	$\cong 3.0$
Energy confinement time τ_E (s)	0.62	0.58	0.46	0.45
Alpha slowing-down time $\tau_{\alpha, sd}$ (s)	0.05	0.09	0.07	0.07
Average effective charge $\langle Z_{eff} \rangle$	1.2	1.5	1.7	1.5

Note: $n_\alpha^* \equiv n_D n_T (\sigma v) \tau_{\alpha, sd}$ and $\tau_{\alpha, sd} \equiv 0.012 T_{e0}^{3/2} (keV) / n_{e0} (10^{20} m^{-3})$.

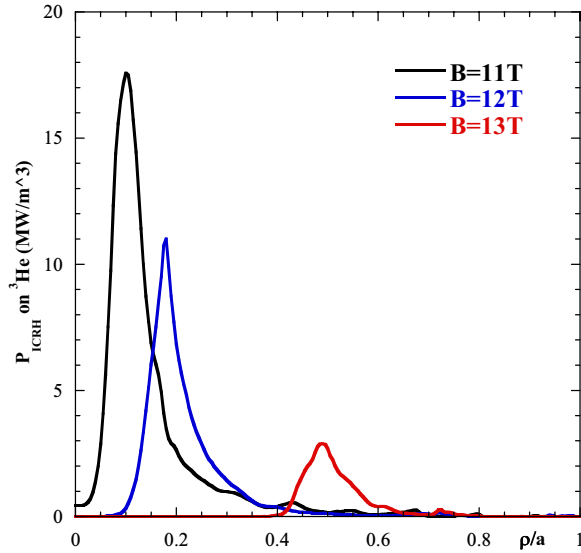


Figure 11. RF power deposition (on the minority ^3He (2%)) in MW m^{-3} versus ρ/a when the magnetic field is ramped up from 11 T (black line) to 13 T (red line) for Ignitor plasma parameters in the ‘ignited scenario’. The applied frequency is 115 MHz.

deposition is shown as a function of the normalized radius ρ/a when the magnetic field is ramped up from 11 to 13 T. It is possible to observe that the deposition is mainly concentrated at the fundamental harmonic of ^3He and a small fraction is given to the electrons via Landau damping; moreover, when the field increases, the resonance layer moves towards the periphery, but still remaining in the bulk of the plasma at 13 T. The power absorbed by ^3He (minority heating) is quasi-linearly redistributed on the collisional time essentially to the D and T bulk ions, with a fraction to the electrons.

The consequence is that the plasma temperature increases accelerating the attainment of ignition. By turning off the ICRH the plasma temperature will decrease rapidly, lowering the reaction rate and re-establishing the sub-critical ignition conditions. By operating in this way a marginal level of α -power can be maintained, and the thermonuclear instability avoided.

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