Relevant Developments for the Ignitor Program and Burning Plasma Regimes of Special Interest*

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Abstract Significant developments in the Ignitor program concerning both physics and technology are reported. These include i) the confirmation of the validity of the path to ignition based on the high density regimes (n_0 ~ 10^{21} m^{-3}) following relevant experiments by LHD (Japan); ii) the adoption of intermediate temperature superconducting material (MgB_2) for the largest poloidal field coils and the relevant R&D effort; iii) the identification and the analysis of the optimal double-X point configurations with plasma currents I_p ~ 9-10 MA that can be adopted to achieve ignition in the H-regime; iv) the completion of the analysis of the ICRH strategy, optimizing the time necessary to reach ignition; v) the completion of the detailed engineering design of all the components of the machine core (e.g., toroidal and poloidal field magnets, supporting mechanical structures, plasma chamber, first wall system, remote handling system, cryogenic cooling system, tritium system); vi) the construction and initial tests of the first injector capable of launching pellets with almost 4 km/s as required by the high plasma densities and temperatures at which ignition can be achieved in Ignitor; vii) the development of new diagnostic systems that can complement the electromagnetic diagnostics to provide the input for the control system of both the position and the shape of the plasma column; viii) the development of prototype coils of the electromagnetic diagnostics with ceramic insulators that are resilient to neutron and gamma radiation; ix) the initial conceptual studies of power producing reactors based on the "reactor physics" that Ignitor is designed to investigate.

1. Introduction

The optimal path to ignition, in the relevant plasma parameter space, that can be followed by experiments using existing technologies and knowledge of plasma physics, relies on the high plasma density regimes (with peak values $n_0 \cong 10^{21}$ m^{-3}) discovered by the line of high magnetic field experiments. These regimes have both outstanding confinement characteristics and degree of purity and are at the basis of the Ignitor design [1]. Their value has been rediscovered recently following the experiments by the helical LHD facility that has systematically produced plasmas with $n_0 \cong 10^{21}$ m^{-3}. Consequently, conceptual power producing reactors named HDR (Helical Demo Reactor) [2] and ARIES-CS (Conceptual Stellarator) [3], have been envisioned by the LHD (Japan) and the NCSX (Princeton) teams that would operate with the plasma parameters close to those of Ignitor when reaching ignition. While the stellarator solution for a power reactor can solve the problem of the current drive for fusion burning plasmas that is connected to the tokamak concept, near term realistic experiments to investigate ignition regimes are, in fact, possible only by the Ignitor approach at present.

Clearly, the main purpose of the Ignitor experiment is that of establishing the "reactor physics" (i.e. the physics of power producing reactors) in regimes close to ignition, where the "thermonuclear instability" can set in with all its associated non linear effects. The driving factor for the machine design ($R_0 \cong 1.32$ m, $a \times b \cong 0.47 \times 0.83$ m^2, triangularity $\delta \cong 0.4$) is the poloidal field pressure $[B_p^2/(2\mu_0)]$ that can contain, under macroscopically stable conditions, the peak plasma pressures ($p_0 \cong 3$–3.5 MPa) corresponding to ignition. The maximum magnetic field on axis, excluding the paramagnetic contribution, is $\lesssim 13$ T and, when the "extended first wall" configuration is adopted, the plasma current can reach 11 MA, with a magnetic safety factor $q_a \cong 3.5$.

2. Intermediate Temperature Superconducting Coils

One of the most significant development of the Ignitor design is the adoption of a recently developed superconducting material, magnesium diboride, for the largest poloidal field coils, having a diameter of about 5 m and producing a vertical field component of about 4 T, that operate at about 15 K and, like all other magnets of Ignitor, is cooled by He gas. The adoption of normal superconductors for the external coils that operate at relatively low magnetic fields has been considered in the past but was not pursued as it would have required a separate liquid-He cryogenic system, given that the cooling system chosen for the optimal operation of the high field coils for which Copper is the best option, uses gas-He at 30 K. Another attractive feature of MgB_2 is that it lends itself to be developed for magnets capable of reaching high fields and it opens new perspectives for the design of new kinds of fusion burning devices. Therefore a collaboration has been undertaken with the industrial Columbus group (Genoa) for the design and construction of the vertical field coils, given the expertise that the group has acquired in the construction of MgB_2 magnets with considerable dimensions as well as of normal superconducting magnets such as those for Tore Supra. At the same time a broader collaboration with material science laboratories (including Edison, INFM, Frascati and Cambridge University) to pursue further fusion relevant developments is being initiated.

3. Double X-point Configurations

In order to investigate H-regimes under fusion burn conditions and close to ignition, a series of double X-point configurations have been analyzed and a pair of optimal configurations has been identified. One of this adopts the highest toroidal field on axis ($B_T \cong 13$ T) and a toroidal current of 9 MA when the X-points lay on the outer surface of the plasma chamber but still relatively close to the edge of the plasma column. The relevant magnetic safety factors are in the range $3.45 \leq q_{95} \leq 4$. Transport analyses using the JETTO code have been carried out [4]. The consistency of the current density evolution for the required magnetic configurations with their macroscopic stability, and the possibility of accessing the high confinement (H-mode) regime have been verified. The H-mode threshold power has been estimated on the basis of the most recent multi-machines scalings and found to be consistent with the available total heating power that includes the Ion Cyclotron Frequency, the Ohmic and the α-particle heating. For the last numerical simulations about 2 MW of ICRH power absorbed by the plasma have been considered. The H-regime is modeled by a global reduction of the thermal transport coefficient used for the L-regime. When no sawtooth activity is included in the simulation, ignition conditions and plasma parameters that are similar to those expected for the 11 MA scenarios with the extended first wall configuration are attained. A quasi-stationary condition can be obtained when a process for the re-distribution of temperature/pressure profiles, such as sawteeth, is included in the analysis.

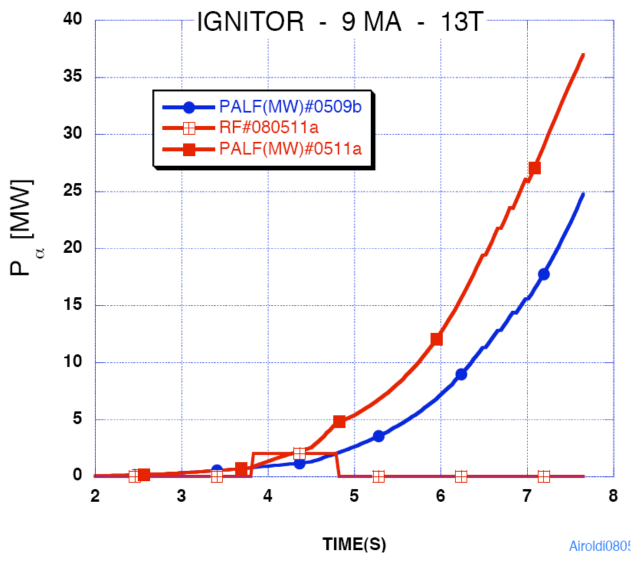

FIG.1 H-mode ignition with and without RF heating, in the absence of sawtooth activity.

The performance of Ignitor in the H regime has been analyzed also using a 0-D power balance code, assuming the energy confinement time to scale as IPB98(y,2) and adopting different scaling expressions for P_{LH}, the power threshold required to access the H regime. We find that the P_{LH}-scaling significantly affects the operating range and that recently proposed scalings [5] lead to more attractive regimes, as the required power may then be reduced considerably (by at least 30%). For moderately peaked pressure profiles ($p_0/\langle p\rangle$ = 2.9) and rather broad density profiles ($n_0/\langle n\rangle$ = 1.25), the operating space for a fusion gain parameter Q=10 is relatively ample. Improved performance (Q > 50) is possible when more peaked density profiles are considered, with peaking factors $n_0/\langle n\rangle$ = 1.60 in agreement with that resulting from the scaling proposed in Ref. [6].

4. Modelling of ICRF heated plasmas

As a complement to the transport analyses, the pertinent physics of the ICRH has been analyzed for all the operating scenarios of interest. For the Ignitor experiment, in fact, heating by means of Ion Cyclotron Resonant waves (ICRH) is planned to be used as a tool to control the plasma temperature, in particular to accelerate the achievement of ignition in the extended first wall configuration, and to facilitate the H-mode transition in the double X-point plasma configuration (B_T = 9 T, I_p= 6 MA). The analysis of the ICRH absorption is given by a numerical solution of the full wave equation in 2D geometry in the magnetic flux coordinate system coupled with a 2D (in velocity space) Fokker-Planck solver. The power deposition profiles on ions and electrons are obtained and used as input data for the transport analysis. In particular, wave equation calculations show that a small fraction of ^{3}He (1-2%) improves the wave absorption on ions near the center of the plasma column, while a substantial fraction of the coupled power, given the $n_{\|}$-spectrum radiated by the antenna, is deposited on the electrons in a broad radial interval of the plasma column. Moreover, the transport analysis, , has demonstrated that a small amount of power (few MW) has a considerable effect on easing the attainment of the main objectives of the Ignitor experiment. In particular, in the "ignition reference scenario" (B_T = 13 T, I_p=11 MA), the ICRH power is delivered power essentially for two purposes. The first one is to assist and accelerate the ignition that, with Ohmic heating only, a nominal 50% D-T plasma mixture with the same spatial profiles and Z_{eff} < 1.5, should happen at the end of the current ramp. In practice, an unbalanced fraction of tritium and deuterium and or different spatial profiles, the presence of an impurity content producing a value of Z_{eff} greater than 1.5, may be delayed. In this case the ICRH would be injected during the field ramp-up and the pulse duration should last over the flattop time. The second purpose concerns essentially the possibility to control the thermonuclear instability, which develops after reaching ignition conditions. In this case the fusion power tends to diverge and reache values that are unsustainable for the machine structure. A possible way to control the thermonuclear instability is to operate at sub-ignited conditions, using

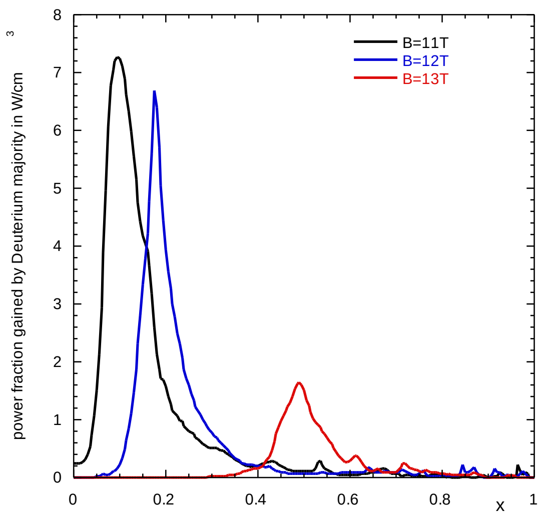

FIG. 2 Power deposition profile (in W/cm^3) on the deuterium thermal ions after collisional transfer (from energetic ^{3}He minority to D) for B_T=11 T (black line), 12 T(blue line), and 13 T (red line) in a Deuterium(50%)-Tritium(50%) plasma mixture.

the ion cyclotron heating to boost the temperature in order to reach plasma burn conditions with K_f (i.e. the ratio of the α-heating power to the rate of energy loss from the plasma) close to unity. Reducing fuelling when the averaged electron temperature exceeds an appropriate value can re-establish the sub-ignited conditions. The best frequency to operate in the "ignited scenario" where the magnetic field is ramping-up is f=115 MHz when the ICRH power is applied at t_{ICRH}>3sec, corresponding to magnetic fields >11 T. For timing less than <3 sec and fields < 11 T, a good choice of frequency could be f=95 MHz. Fig. 2 shows the power deposition profile (in W/cm^3) as a function of the radial normalized coordinate x when the magnetic field is ramping-up from 11 to 13 Tesla at a fixed frequency f=115 MHz, and a power level of about 5 MW. The plasma composition is Deuterium (50%), Tritium (50%) and a small fraction of ^{3}He (<2%). The density and temperature increases from about $5\times10^{20}m^{-3}$ to $9\times10^{20}m^{-3}$, and from 4 to 6 keV respectively. It is possible to recognize the first harmonic of ^{3}He and the second harmonic of tritium resonating at the same location, and when the field increases the resonance layer moves toward the periphery. The power absorbed by the ^{3}He (minority heating) is redistributed on the collisional time scale essentially among the deuterium and tritium bulk ions. As a consequence the plasma temperature increases accelerating the attainment of ignition.

5. Advanced Diagnostics Systems and Plasma Control

The Ignitor program has included a significant effort for the development of new diagnostics and essential auxiliary systems for the thermonuclear plasmas that the machine is expected to produce. For example, an advanced neutron spectrometer employing a detector proposed by H. Enge at M.I.T. was initially designed for Ignitor and later constructed and operated for the JET facility. Since the experiment will produce, in high performance discharges, a neutron flux at the first wall that is comparable to that expected in future power producing reactors, some diagnostics, such as the electromagnetic coils, are expected to suffer an appreciable, although reversible, degradation that can cause the measurement of some fundamental plasma parameters, such as current and position, to become problematic. At the same time, the integrated fluence at the first wall and on most other machine component is modest, and long term radiation damage is not an immediate concern.

The issue of RIC (Radiation Induced Conductivity) and of other similar effects is being addressed with an R&D program aimed at the development of electromagnetic diagnostics with higher damage threshold. At the same time, supplementary methods for plasma position measurements are explored to provide an appropriate degree of redundancy [7]. One of these is based on the diffraction and detection of soft X-ray radiation emitted near the top or bottom of the plasma column, where the distance of the LCMS from the wall is only few millimeters and gradients in the X-ray emission are more prominent. For the purpose of real-time plasma control, the system should provide an output signal to the control system without additional inputs from other diagnostics, and needs to be sufficiently fast. Therefore a crucial component of the system is the detector. Despite tremendous progress in solid state detectors technology, extremely fast read-outs are still more readily achievable with Gas Electron Multiplier devices. In particular, the new GEM detectors, developed at CERN and already tested on fusion devices [8] provide an optimal combination of design flexibility, high counting rates (>1 MHz) and low sensitivity to background radiation. The initial studies were aimed at assessing the possibility of monitoring the vertical position of the plasma column, since VDE's are the main concern for the integrity of machines with elongated plasmas, and to develop a strategy for interfacing this different type of signal with the control system.

The vertical position and shape controller for Ignitor [9] has been designed on the basis of the CREATE_L linearized plasma response model [10], which assumes an axisymmetric system and describes the electromagnetic interaction of the plasma with the surrounding structures by a small number of global parameters (i.e., β_{pol}, l_i, I_p). In particular, the vertical stabilization system has been designed assuming that the vertical plasma centroid position can be estimated by a suitable linear combination of the available magnetic measurements. A possible partial failure of these magnetic diagnostics has already been taken into account, showing a good resilience to such events. To take the new diagnostics into account, we make the assumption that the vertical position measurement provided by the spectrometer can be obtained from the real position of the centroid after application of a sample-and-hold (holding time t_H) and a quantizer (quantization level z_Q). Also a gaussian multiplicative noise with mean 1 and standard deviation 5% has been considered. Here, t_H can be an estimation of the time needed to get a measurement and z_Q of the minimum vertical position variation that can be detected. Fig. 3 shows the simulated time behavior of the vertical position after an initial displacement of 5 mm along the unstable eigenmode, due to a perturbation. The reference flat-top 11 MA configuration has been considered. Here, we assume t_H=1 ms and z_Q =2.5mm; evidently, with such parameters the system can be stabilized. A sensitivity analysis shows that with the same quantization level, the system tolerates measurement times up to 2.5 ms, while larger values of z_Q would require a faster measurement system.

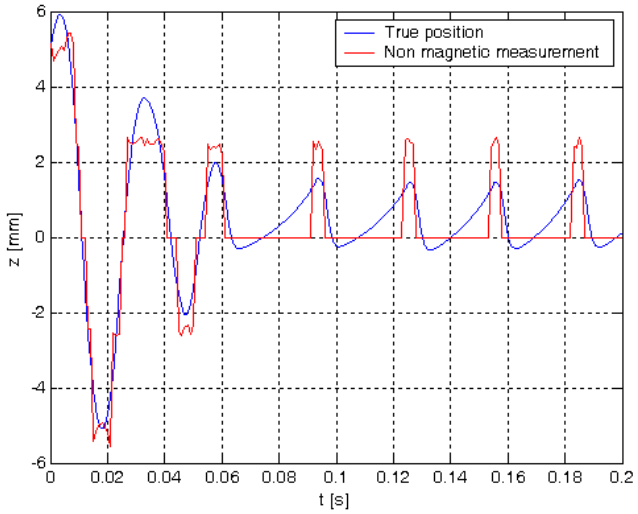

FIG. 3. Time behavior of the vertical position

6. The Fast Ignitor Pellet Injector (IPI)

The process of achieving ignition conditions requires the plasma density rise to be properly controlled. With particular reference to the ideal ignition conditions, for reasonable density and temperature profiles, ignition corresponds to peak temperatures $T_{e0} \cong 6$ keV. For this reason, a pellet injector program has always been included in the Ignitor design, for the purpose of controlling both the central density and the density profile evolution. ENEA and ORNL are collaborating on the construction of a fast pellet injector [11, 12] for the Ignitor experiment, featuring two innovative concepts: (i) the proper shaping of the propellant pressure pulse to improve pellet acceleration, and (ii) the use of fast closing (< 10 ms) valves to drastically reduce the expansion volumes of the propellant gas removal system (Fig. 4).

Fig. 4 The IPI gas removal system at ORNL

The Ignitor Pellet Injector (IPI) consists of two independent sub-systems, built by ENEA and ORNL separately. The ORNL sub-system includes the cryostat and pellet diagnostics, with related control and data acquisition system. New light gates, microwave cavity mass detector and control software have been developed specifically for this application. The ENEA sub-system, including four independent two-stage gun and pulse shaping valve, the (patent pending) gas removal system, and the associated controls and diagnostics, has been thoroughly characterized at CRIOTEC. In particular it was shown that the pressure rise in the downstream expansion volume could be completely cut-off by reducing to 1.6 ms the delay (relative to the pressure pulse time) with which the fast gate valve starts to close. Such a delay allows pellets traveling at speeds of 2 km/s or more to safely cross the gate, which is placed about 3 m downstream of the gun muzzle. The ENEA sub-system has been finally shipped to Oak Ridge, to be coupled with the ORNL facility for joint experiments. A preliminary short experimental campaign was jointly carried out. Testing of D_2 pellet formation and launch with all four barrels, using ORNL single-stage propellant valves was completed and a preliminary attempt to couple the two sub-systems, with the secondary aim of testing their information exchange protocol, has led to excellent results, and pellet speeds around 2 km/s were readily achieved. A new campaign is planned to start soon.

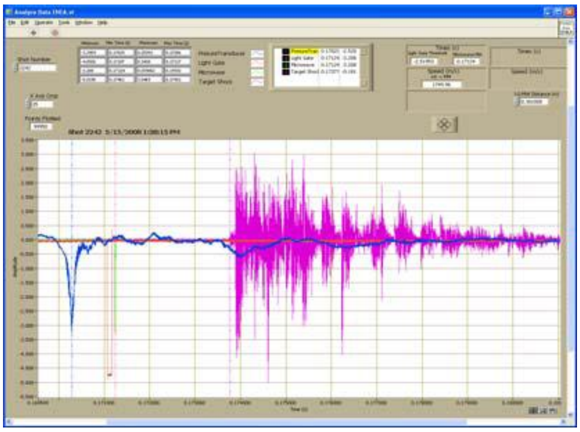

FIG.. 5 Time traces of a pellet at 2 km/s

7. Mechanical Design, Structural Analysis, and Tritium System

The ANSALDO industrial contribution to the Ignitor project has included the detailed mechanical design and structural analysis of all the main components of the machine core, the so-called Load Assembly. The machine Central Post, the Central Solenoid, the Shape and Equilibrium coils, the Vacuum Vessel and First Wall system, the surrounding mechanical structures, the Vacuum Cryostat and the supported Polyethylene Boron composite sheets for neutron shielding have been structurally analysed and mechanically verified to confirm that they can withstand with a proper margin both the normal and off-normal operating loads. In particular, the maximum current scenario at 11MA/13T, the reduced scenarios at 6MA/9T and 7MA/9T, and the Vacuum Vessel and First Wall baking operations have been considered. More recently, the structural analysis of the Load Assembly has been extended to the scenarios with double X-points configurations for the H regimes, with the maximum toroidal field on axis but a reduced plasma current of 9 MA in one case and 10 MA in the other, depending on the position of the X-points relative to the plasma chamber and first wall. Preliminary results of this analysis, still underway, for the case at 9 MA/13 T, show the feasibility of this scenario from the structural point of view. In fact, compared to the most advanced operating scenario, a better torsional stability of the machine, essential to resist the out-of-plane loads on the toroidal magnets, is assured due to the lower centrifugal forces coming from the central solenoid coils. Similar analyses with in-plane and out-of-plane loads will be soon performed also for these two new scenarios. These, together with the activity concerning the mechanical integration of the ICRH system inside the vacuum vessel equatorial ports will complete the detailed design of the machine core.

In parallel to the mechanical design activity, 3D/2D drawings of each individual component have been produced using the Dassault Systems CATIA V software. Once their design was assessed and accepted by ENEA, they have been all assembled in a single 3D CATIA model of the Load Assembly (Fig. 6). The assembly sequences of the components can be verified by using the 3D model just mentioned, pointing out mechanical clearances/inconsistencies at the interfaces between them. The model is of help in order to identify the best layout of the electric and fluidic lines and their supports on the Load Assembly, as well as that of the electromagnetic diagnostics. The design of the in-vessel Remote Handling System is verified by simulating, with virtual reality techniques, the remote maintenance of the first wall tile carriers, of the ICRH and Faraday Shields components, and of the machine diagnostics. The feasibility of the design/assembly/in-vessel remote handling operations of the machine has thus been demonstrated.

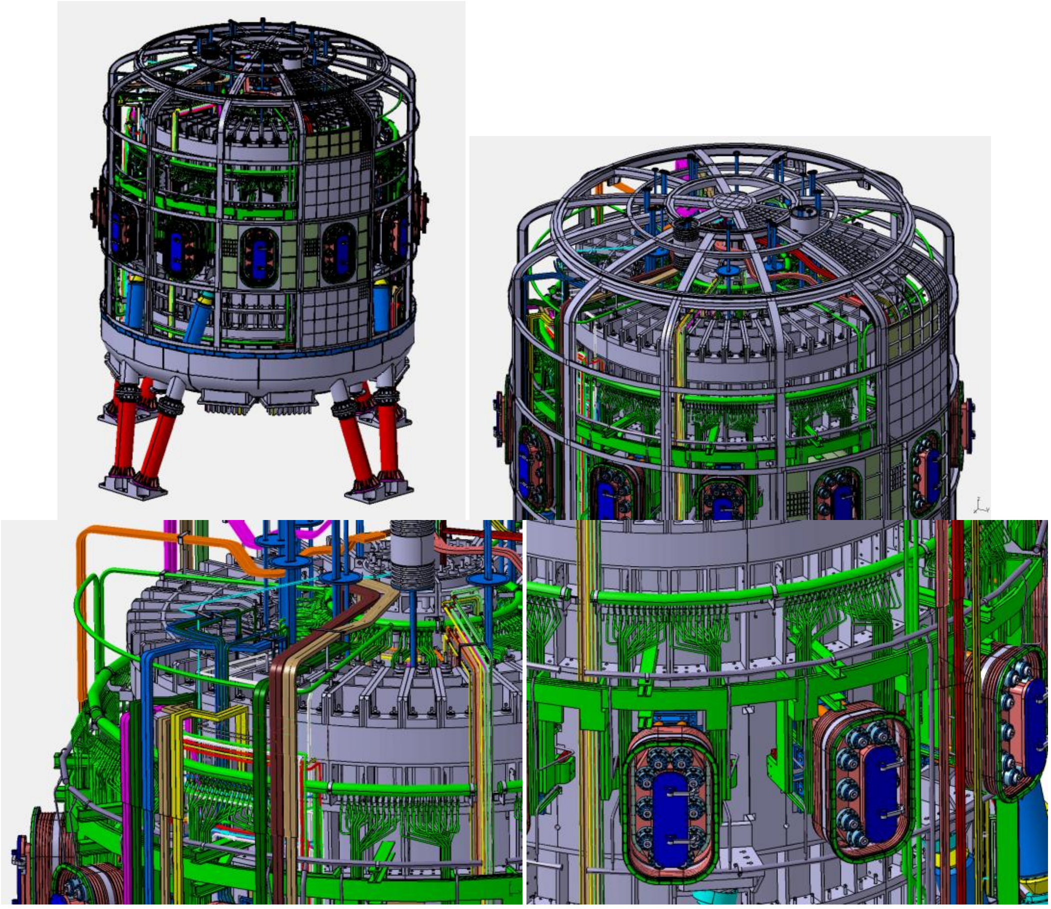

FIG. 6 3D model of the Ignitor Load Assembly

To complete the mechanical design, the analysis of the stresses produced by the out-of-plane loads has also been performed at significant times for all the operating scenarios in normal conditions and during the worst expected vertical plasma disruption events (VDE), at the end of the toroidal field coil flat-top. The non linear analysis takes into account both in-plane and out-of-plane loads. The finite element model set-up now is completely 3D and includes a full 30° machine sector (Fig.7), as the out-of-plane loads during VDE’s are not symmetrical relative to the equatorial plane. At 11 MA/13 T, the analysis shows that the assumed friction coefficient on the wedging surfaces during normal operating conditions is adequate to assure the structural stability of the Load Assembly. Furthermore, once unloaded, the structure comes back without any permanent distortion. The safety factors of averaged shear stresses against the insulation shear rupture strength is always greater than 3 at the beginning of life, while it is reduced to about 2 at the end of life because of the degradation of mechanical properties due to the neutron irradiation. The plasma vertical disruptions (VDE) cause the largest out-of-plane loads, the moment around the machine axis being 1.4 to 2.5 times higher than in normal operating conditions. The safety factors of the interlaminar shear stresses on toroidal field coils are also lower than in the normal operating conditions, but remain around 2. Keys of proper dimensions between the 30° extension C-Clamps modules were adopted to ensure the machine structural stability. The stresses calculated in the TFC are below the yield stress of the copper material at the relevant temperature; the stresses in the C-Clamps and Central Post are within the ASME allowable limits at the operating temperature.

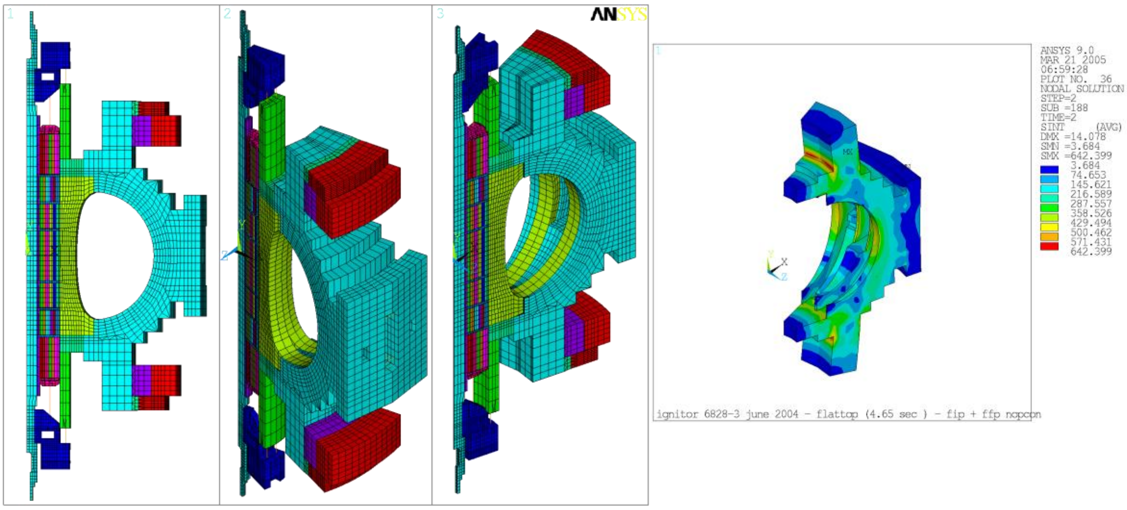

FIG. 7 3D Finite element model of a 30° sector of the Load Assembly – in plane and out of plane loads and C-Clamp Tresca stress intensity (MPa) at 4.65 sec of TFC flattop during normal

Among other recent design activities, the analysis of the tritium system has been carried out [13] with the aim of identifying its main components and the operations needed to supply the deuterium-tritium mixtures to the vacuum chamber and to recover the plasma exhaust. At the maximum performances, the Ignitor duty cycle is relatively low, and thanks to the compact machine dimensions, the amounts of tritium to be injected are very small. Tritium purification and recycling from the plasma exhaust is found to be uneconomical. Considering relevant safety requirements, the plasma exhaust will be recovered in gaseous form, avoiding oxidation and formation of hazardous tritiated water. Then, hydrogen isotopes are separated from exhaust gases by using palladium membranes and then stored in metal getters.

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[1] B. Coppi, A. Airoldi, F. Bombarda, et al., *Nucl. Fusion* **41(9)** (2001) 1253.
[2] O. Mitarai, et al., *Proceed. 21st Fusion Energy Conference* FT/P5-24, Chengdu, China (2006).
[3] M. Zarnstorff, 49th APS – DPP, Orlando, FL (2007)
[4]B. Coppi, et al., *Proceed. 35th EPS Conf. on Plasma Phys.*, Hersonissos, Greece, Paper P2.101 (2008)
[5] D. C. McDonald et al., *Plasma Phys. Control. Fusion* **48**, A349 (2006).
[6] C. Angioni et al., *Nucl. Fusion* **47**, 1326 (2007).
[7] F. Bombarda, et al., *Proceed. 35th EPS Conf. on Plasma Phys.*, Hersonissos, Greece, P2.101 (2008)
[8]D. Pacella, R. Bellazzini, A. Brez, et al., *Nucl. Instrum. & Methods A* **508,** 414 (2003)
[9] F. Villone, R. Albanese, G. Ambrosino, et al., *Bull Am. Phys. Soc.* **52(16)**, 45 (2007).
[10] R. Albanese, F. Villone, *Nucl. Fusion* **38,** 723 (1998)
[11] A. Frattolillo et al. *Proc. of the 33rd EPS Conf. on Plasma Phys.*, Rome, Italy, Paper P-4.185 (2006)
[12] A. Frattolillo et al. *Proc. 22nd IEEE/NPSS SOFE07*, Albuquerque (NM), U.S.A., 2007
[13] C. Rizzello and S. Tosti, *to be published in Fus. Eng. & Des.* (2008).