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The JET Programme in Support of ITER

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ABSTRACT
The mid-term experimental programme of the JET tokamak exploits its recently enhanced scientific capabilities (e.g., new divertor allowing high triangularity ITER- relevant scenarios, several new diagnostics) to address critical issues potentially impacting the detailed design of ITER components (e.g., first wall, heating and current drive systems, diagnostics) and, in parallel, further develop ITER operating scenarios and address specific physics issues of direct relevance to ITER (e.g. transport physics, burning plasma physics). For the longer term, activities on a "*JET programme in support of ITER*" have been launched, aiming at making optimal use of JET's unique features: large plasma size and capability to handle beryllium and tritium. A full replacement of the first wall materials is planned (beryllium in the main wall and tungsten in the divertor). This should deliver answers to urgent plasma surface interaction questions, such as tritium retention, and provide operational experience in steady and transient conditions with ITER wall materials under relevant geometry and relevant plasma conditions. In addition, the JET auxiliary heating power will be upgraded to ~ 45 MW, allowing access to ITER-relevant disruption and edge localised modes energy loss densities. This will open access to conditions of melt layer formation both on the beryllium first wall and the tungsten divertor and, on the other hand, help progressing, in particular, hybrid and advanced scenarios for ITER, which require full or partial current profile control, thereby making use of new dedicated diagnostics.

1. INTRODUCTION
With the construction of the International Tokamak Experimental Reactor (ITER) close to being started, the magnetic confinement fusion community is focusing its research efforts on scientific and technological issues related to the detailed design of ITER components, ITER operation and ITER licensing. In this respect, the JET tokamak, with its unique scientific and technical features, provides a well-equipped platform to address a substantial number of these issues. Indeed, the size, plasma current and magnetic field of JET offer access to the most ITER-relevant plasma parameter range, enabling key physics aspects to be addressed such as confinement, alpha particle physics (reduced first orbit losses of 3.5 MeV alpha particles), disruptions, edge localised modes (ELMs), power fluxes, etc...

The objectives of the 2006 and 2007 JET experimental campaigns are to study issues which could impact on the detailed design of ITER components, such as the first wall, heating & current drive systems and diagnostics; to optimise ITER operating scenarios, including the ELMy H-Mode, the Advanced Scenario and the Hybrid Scenario; and to study ITER-relevant physics such as burning plasma physics, transport and MHD and the effects of toroidal field ripple. In order to conduct these studies with the most ITER-like plasmas, the auxiliary power available on JET has been increased somewhat, and the Mk-II-GB (Gas Box) divertor has been upgraded to Mk-II-HD (High Delta, see Fig. 1) to allow high power operation with a more ITER-like, high-triangularity plasma shape (Fig. 2). In addition, a set of 25 new or upgraded diagnostics have been installed to allow observations necessary to achieve the goals of the programme [1, 2] and to test ITER-relevant diagnostic techniques.

Moreover, a new ITER-like ion cyclotron resonance heating (ICRH) antenna, expected to be 3 resilient to fast varying loads due to ELMs and to deliver 8 MW/m^2 for 10s, is foreseen to be installed for operation in 2007 [3].

The JET activities in support of ITER include also technology R&D. Recently, a particular focus has been on activities which aim at characterising dust and erosion/deposition areas in JET, improving in-situ Tritium removal techniques and providing support in topics relevant to the licensing of ITER. Recent erosion/deposition studies included laser induced break-down spectroscopy for co-deposited layer studies. Preliminary results indicate that this technique could be viable in a tokamak environment to allow the evaluation of the thickness of a co-deposited layer formed on a Tungsten surface, besides its primary application as an in-situ detritiation method [5]. Further in-situ detritiation techniques have been investigated with progress, in particular, in the development of a plasma torch system [6]. Furthermore, an activity to develop a dose rate prediction capability has been launched with a computational tool based on the MCNP Monte Carlo N-particle code [7]. Finally, JET is providing support to safety analysis in investigations related to the human factors approach in ITER licensing [8].

In the longer term, JET's unique capability to handle tritium (T) and beryllium (Be) will be fully exploited in a multi-annual "JET programme in support of ITER" which has been elaborated in a coherent approach along the following 3 main axes (see Fig. 3):

(i) experimentation with an ITER-like wall
(ii) development of plasma configurations and parameters towards the most ITER-relevant conditions achievable today
(iii) integrated experimentation in Deuterium-Tritium

2. OVERVIEW OF THE JET PROGRAMME IN SUPPORT OF ITER

2.1. SCIENTIFIC OBJECTIVES

The JET programme in support of ITER has been designed to provide significant advances and solutions in the following key areas:

- The consolidation of the plasma scenario for ITER Q=10 Operation: foreseen is the integrated optimization of the ELMy H-mode, fully compatible with ITER-like wall conditions and D-T operation.
- The development of plasma scenarios for addressing the Q=5 steady state operation and/or component testing on ITER: foreseen are the development of controlled improved H-modes (also named hybrid modes) and advanced scenarios toward long pulse/steady state high performance operation, fully compatible with ITER wall conditions and D-T operation.
- The development and testing of ITER technologies, that need not be frozen early in the ITER construction: this includes
- wall and divertor materials,
- heating, fuelling and current drive technologies, diagnostics and auxiliary systems,

- the handling and processing technologies for activated, tritiated, or beryllium contaminated components.
- ITER licensing issues: with the goal of minimising the T inventory.
- The maximisation of performance and component lifetime in ITER: including
- the development of further techniques to control plasma scenarios and transient events;
- the resolution of key physics issues and scalings to predict ITER performance and control requirements.
- The preparation for burning plasmas physics on ITER: consisting of
- an in-depth preparation for the physics arising from burning plasmas (experiments and modelling);
- the development of burning plasma diagnostics.
- The development of Integrated Tokamak Modelling tools for ITER: including support of the EU Task Force on Integrated Tokamak Modelling by providing physics data for benchmarking in ITER-relevant conditions.

2.2. ACCOMPANYING ENHANCEMENT PROGRAMME

In order to consistently, effectively and efficiently address these programmatic issues, a set of enhancements of the JET Facilities is planned (see also Fig. 3) to complement the recently enhanced scientific capabilities aforementioned (new divertor, power upgrade, new diagnostics, ITER-like ICRH antenna). The following items will be implemented in JET:

1. An ITER-like wall and divertor with the combination of metallic plasma facing materials foreseen for ITER: beryllium wall and tungsten divertor.
2. An upgrade of the neutral beam heating system, increasing the overall auxiliary heating power to ~45MW and the pulse length capabilities to up to 40s at moderate heating power (20-25MW).
3. The installation of a high frequency pellet injector system to mitigate the impact of ELM power loads on first wall components and divertor and to enable deep fuelling.
4. Diagnostics required by the change of wall materials, diagnostics needed to support further developments of ITER plasma scenarios, as well as ITER diagnostics that need to be tested on JET, such as fast wave reflectometer, radiation hard Hall probes and diagnostics for the fusion products.
5. An upgrade of the plasma control system.

A modification of the JET wall materials will bring operational experience in steady and transient conditions with the ITER first wall and divertor metallic materials (Be and tungsten), and could provide input to optimise the choice of first wall materials on ITER to balance tritium retention, to control the wall and divertor power loads below the technically acceptable limits and achieve a sufficient lifetime for Plasma Facing Components (PFCs) and, finally, to optimise configurational flexibility. JET is the only device in the world capable of testing the combination of ITER wall materials, thanks to its beryllium capability. Furthermore, the size, magnetic field and plasma current of JET make it possible to conduct such a test at the most ITER-relevant parameters. In this regard, an upgrade of the heating power, fuelling, current drive and diagnostic capabilities of JET will allow significant progress towards ITER conditions in all plasma scenarios foreseen for ITER operation. An increase of the heating power and heating pulse length would be essential for further development/optimisation of these scenarios and to progress ITER control schemes. In particular, studies would include investigating the attractive possibility of porting the ELMy H-mode and improved H-modes to higher β_N (normalised ratio of plasma pressure over magnetic pressure), testing plasma resilience to impurity influxes from ELMs, developing disruption mitigation, ELM control and Neoclassical Tearing Mode (NTM) control strategies for ITER, and progressing the less mature advanced scenario to ITER- 7 with full current drive and steady state (sustained for a full current diffusion time). At present, the study of scenarios with large ELMs (energy drop ΔW_{ELM}~1-2MJ) is limited by high force disruptions which occur when the response to the ELM produces a radial field current excursion larger than the Fast Radial Field Amplifier (FRFA) current limit (ELM triggered vertical displacement event). The relative FRFA current excursion (I_{FRFA}/I_p) increases with $\Delta\leq$or equivalently $\Delta W_{ELM}/W_{PED}$ (where W_{PED} is the pedestal energy and Irelevant $\Delta\beta_{POL}$ the plasma current) and is larger for highly shaped plasmas. Since $\Delta W^{ELM}/W_{PED}$ is found to scale with $1/\nu^*$ (inverse collisionality) [9] the existing Vertical Stabilisation capability restricts operation at low collisionality, high current and high plasma shape. The importance of JET operation in this domain with physics of direct relevance to ITER gives a strong motivation for a major enhancement of the Vertical Stabilisation system with the objective of doubling the survivable ELM size. ITER technology would also benefit significantly from upgrades of JET auxiliaries, such as a High Frequency Pellet Injector, detritiation techniques, tritium-technologies, advanced control systems and ITER-relevant diagnostics.

3. THE ITER-LIKE WALL EXPERIMENT IN JET

3.1. OBJECTIVES

As shown on Fig. 4(a), ITER is designed to have an all beryllium-clad first wall, tungsten (W) brushes over most of the divertor region and Carbon Fibre Composites (CFC) only at the target plates where the highest power fluxes are expected (divertor strike points) [10]. A main concern with the ITER materials choice is the long term 8

Tritium retention behaviour [11, 12, 13]. In carbon-clad devices this is mainly caused by erosion of carbon and its migration to plasma shadowed areas and co-deposition with hydrogen isotopes. Present estimates show that ITER would reach its permitted 350g tritium inventory in relatively few pulses (probably ~100 full performance pulses). On the other hand, current T removal techniques are not elaborated enough to provide a solid basis on which ITER could operate. This is the main reason why an all W divertor has been considered as the alternative option for the deuterium-tritium phase of ITER. However, the use of W has other serious implications for ITER, mainly with respect to melting, mixed W-Be material issues, and W plasma core contamination. The ITER reference materials have been tested in isolation in tokamaks, plasma simulators, ion beams and high heat flux test beds. However, an integrated test demonstrating both acceptable T retention and the ability to operate ITER-relevant plasmas with high power input within the limits set by these materials has not yet been demonstrated. Within the "JET programme in support of ITER", the objective of the ITER-like Wall Project is to install in JET a Be wall and an all W divertor (see Fig. 4(b)). This choice is more demanding than the ITER-reference combination but offers a cleaner comparison between an all-metal and an all-CFC JET wall; it will also provide relevant information for the preparation of an all-W divertor option on ITER.

The scientific objectives of the ITER-like wall experiment in JET can be summarised as follows:

- Demonstrate that a beryllium wall and an all-W divertor have sufficiently low fuel retention to meet ITER requirements.
- Demonstrate ITER-relevant tritium retention mitigation and detritiation techniques in a Be/W machine – including in particular the effect of trace oxygen and carbon impurities on fuel retention.
- Show how beryllium migration onto a full tungsten divertor influences W erosion and subsequently the main plasma W density in ITER-relevant operating scenarios.
- Characterize the effect of transients (ELMs and disruptions) on a Be first wall and W divertor.
- Develop control strategies applicable to ITER for detecting and limiting damage to Be and W plasma-facing components, such as relevant disruption mitigation and ELM power loss control systems. The power upgrades allow extension of this work to energy densities in transients comparable to those in ITER.
- Study melt layer behaviour in ELM and disruption energy losses and implications for subsequent plasma operation. Possible alloying of Be with W which may reduce the melting point for the mixed Be-W near surface layer.
- Develop integrated ITER compatible scenarios for an all-metal machine including impurity seeding strategies to replace the intrinsic carbon radiation which is essential for the achievement of acceptable divertor power loads in the ITER baseline edge scenario.
- Investigate special heating system related effects such as the interaction of fast ions generated by the new JET ITER-like ICRH antenna with beryllium wall components and the W divertor baffle.

3.2. BERYLLIUM WALL

The main constraints on the ITER-like Wall project are the needs to preserve the power, energy handling and force limits (due to disruptions) set for the CFC wall whilst providing a scientifically relevant materials configuration for ITER. These constraints have led to the design of solid Be tiles which are inertially cooled, segmented (to minimise eddy forces) and castellated (to avoid thermal stress cracking) with hidden bolts and tile shaping to maximise the power handling. These aims have had to be compromised with the technical capabilities of the existing supporting wall and limiter structures since a redesigning of these structures is beyond the scope and resources of the project. As shown on Fig. 5, the technical implementation of the JET main chamber tiles foresees the use of bulk Be for all the inner and outer wall guard limiters but excluding parts of 2 inner wall limiters that are hit directly by Neutral Beam Injection (NBI) shinethrough. Bulk Be is also foreseen for the upper dump plates, the saddle coil protection tiles (upper and lower), the Lower Hybrid (LH) and the Ion Cyclotron Resonance Heating (ICRH) antenna protections. The Neutral Beam (NB) shinethrough protection tiles will be made of W coated CFC but will be 3 cm recessed with respect to the Be guard limiters. In addition, the protection tiles at the upper divertor baffle areas are made from W-coated CFC, not because of power handling reasons but due to the effort required to design solid W tiles and the high associated cost. A total of 4404 main wall tiles currently installed in JET will be replaced by 1700 solid Be tiles machined from 4 tonnes of solid Be in addition to the Be-coated inconel tiles [14] which will be placed on the inner wall areas between the limiters and the W-coated CFC tiles. All Be tiles are sliced and castellated toroidally and poloidally in order to reduce electromagnetic forces and increase heat shock resistance [15, 16]. This design will serve also as an excellent test for ITER in view of the mechanical integrity and the possible material migration into gaps and the associated fuel retention.

3.3. TUNGSTEN DIVERTOR / R&D ON TUNGSTEN COATING AND BULK TUNGSTEN TILES

The technical solution proposed for the JET full W divertor, as shown on Fig. 6, includes bulk W for the load bearing septum replacement plate (LBSRP) and W coating on CFC for the remaining tiles.

In view of the need to select the most reliable W coating technique, a coordinated R&D programme was launched in early 2006. As a result, 14 different types of W coatings, including chemical vapour deposition (CVD, 4, 10, 200μm), physical vapour deposition (PVD, 4, 10μm) and vacuum plasma spraying (VPS, 200μm), have been produced on CFC tiles and tested under cyclic heat loading under coordination of IPP Garching in cooperation with CEA Cadarache, ENEA Frascati, TEKES Finnland, MEdC Romania and FZJ Germany. All coatings were subjected to a thermal screening at the GLADIS test facility (IPP-Garching) with stepwise power loads increasing from 4 MW/m^2 in 6s up to 22 MW/m^2 in 1.5s with the surviving 9 out of 14 coating types exposed afterwards to cycling tests for 200 high heat flux pulses. The different and anisotropic thermal expansion between CFC and W often led to cracking perpendicular to the fibres, inducing delamination by buckling along the fibres, melting and partial loss of coating. A 200μm VPS coating and a thin 10μm ion assisted magnetron sputtered layer behaved best. These coatings were additionally exposed to 1000 typical JET-like ELMs (0.35GW/m^2 for 1ms) in the electron beam facility JUDITH in FZJ, demonstrating their robustness for this number of cycles (note that more than 100 ELMs of this energy can occur in one high performance JET pulse).

For the most heavily power-loaded tiles, a bulk W tile concept has been developed under the leadership of FZJ Germany [17, 14]. The design was mostly determined by the strong constraint of minimising electromagnetic forces during disruptions and optimising mechanical stability [18, 19]. A design was made of 6mm W lamellae, packed together in 4 poloidal stacks and bolted together in toroidal direction but with electrical isolating spacers to reduce eddy currents [17]. Each lamella has dedicated electrical contact points to the support structure to reduce halo forces and avoid arcing (Fig. 7). A prototype of this concept has survived cycling heat flux tests (200) with 7 MW for 10s and failure test with 10 MW for 14s with temperatures exceeding 3000 degrees C.

3.4. ACCOMPANYING DIAGNOSTICS

A set of new or upgraded diagnostics will be dedicated to support the ITER-like wall experiment. A poloidal array of thermocouples will be installed for machine protection and interpretation of infra-red data from the recent wide angle viewing infra-red system. An additional fast infra-red camera is to be installed in 2007 for power deposition studies in the divertor. JET is already equipped with a unique set of diagnostics to detect erosion and deposition in the main chamber and the divertor, such as marker tiles, deposition monitors, rotating collectors and quartz micro- 12 balances. These will be re-installed in the new wall experiment to compare the present CFC wall with the metallic wall. Finally, spectroscopic diagnostics for impurity sources and core concentrations for Be and W will be improved in the visible, XUV and VUV range.

4. OTHER ENHANCEMENTS FORMING PART OF THE JET PROGRAMME IN SUPPORT OF ITER

In order to draw the maximum benefit from the ITER-like wall experiment in JET and to ensure the coherence of the JET programme in support of ITER, a number of further enhancements of the JET Facilities will allow significant progress towards ITER conditions.

4.1. NEUTRAL BEAM ENHANCEMENT

4.1.1. Objectives

In order to increase the ITER-relevance of the plasma scenarios, additional NB heating power, up to > 34MW for 20s (compared to 25MW for 10s at present) will be provided by upgrading the existing NB boxes and power supplies for operation at higher NB current [20]. Phasing the two NB boxes will make it possible to deliver about 17MW of NB power for up to 40s, for full exploitation of the pulse length capability of the JET machine. This will be essential to progress, in particular, hybrid and advanced scenarios for ITER, which require full or partial current profile control. Consequently, the overall auxiliary heating power of JET will be increased to ~ 45MW, allowing access to ITER-relevant steady-state conditions, disruption and ELM 13 loss power densities, thereby opening up a range of new possibilities for scenario development.

First, the power increase will allow stable H-mode operation at 3.4T and 85% of the Greenwald density with margins for exploitation. Second, pedestal scaling studies will resolve the uncertainties (up to a factor of 1.8) in the predictions of the pedestal energy for ITER from JET data, associated to with the edge transport barrier being limited by either MHD or thermal conduction [21]. This would have major consequences for ITER, since it should be possible to distinguish the contribution of the pedestal to the total plasma energy, W_{dia}^{ITER}. Present estimates are in the range 28-50 % of W_{dia}^{ITER}, since it is unclear which of the two pedestal scalings apply. Furthermore, dedicated confinement scaling studies will be able to clarify whether confinement scales approximately as $1/\beta$ (as assumed in the ITER design with the ITER98(y,2) scaling) or independent of β (as indicated by recent dedicated scans on JET and DIII-D [22, 23]). This ambiguity arises because the difference between the two scalings is within the uncertainty in the present stored energy measurement (~10%). These two scalings diverge with increasing power and with 45MW they would be clearly distinguishable. With a weaker dependence of confinement on β, ITER would have the potential to operate at Q ~ 20 with β_N in the range 2.5-2.8 (with $\beta_N \sim 2.8$ the fusion power on ITER would rise to 1000MW) [24]. On JET, access to $\beta_N \sim 2.8$ at low normalized Larmor radius, ρ^*, (3.4T) will be possible with heating power in the range of ~ 45MW. Thus, the verification of a new operational point for ITER will open up the possibility of new and high impact ITER scenario development on JET.

Even stronger arguments regarding the scaling of confinement apply to the hybrid regime. With an heating power of ~45MW JET will be able to operate significantly control (to ensure stationarity and stability at maximum β closer to ITER conditions (high β and low ρ^*) and provide an improved scaling for extrapolation to ITER. ITER-relevant conditions with $T_{i0} = T_{e0}$ and low plasma rotation will also be accessible with high density at 2.3-2.8MA, B > 2.6T and heating power ~45MW. JET results in this domain will be crucial since the confinement enhancement observed in smaller experiments with $T_{i0} > T_{e0}$, and high plasma rotation may give a too optimistic extrapolation of the hybrid scenario to ITER in terms of Q (>30) and pulse length (>1500s) [25]. Moreover, long pulse hybrid operation at reduced plasma current and magnetic field will allow the study of the influence of NTMs on the evolution of the current density profile, to define the requirements for the control of central safety factor, q_0, and to optimise the stability of the scenario by varying $q_{0.}$ Present assessments of the control needs for hybrid scenarios in ITER range from no control required (because of the influence of small amplitude NTMs) to strict requirements for q_0). With regard to the advanced tokamak regime, 0D calculations [26] indicate that the targeted increase in power and current drive could dramatically extend the operational space in JET in terms of current (>2.5MA) and density ($n > 6\times10^{19}m^{-3}$), with high β_N (< 3.4) and bootstrap current fraction in the range of 70% at high toroidal field (~3.5T). In particular, the calculations show that a power level of 45MW gives access to a regime where, as in a future steady state reactor, the bootstrap current is maximized together with the fusion yield and not at the expense of fusion yield as in present day experiments (i.e. by lowering the plasma current and/or toroidal magnetic field).

4.1.2. Description

The upgrade of the neutral beam system will be achieved by (i) changing the magnetic configuration of the ion sources from the present super-cusp configuration to a pure chequer-board configuration, and re-optimising the accelerators for 125kV/60A (in deuterium), (ii) replacing a number of beamline components with upgraded designs, e.g., Actively Cooled Duct Liners (see Fig.8), (iii) replacing the high voltage power supplies on the beamline on Octant 4 of JET with four new 130kV/130A modules. The main increase in NB power comes from the conversion of the ion sources. Chequer-board ion sources produce larger fractions of molecular ions (D_2^+ and D_3^+), leading to an increase of NB power as a consequence of the larger neutralisation cross-section for slower molecular ions. The plasma in the extraction region of the chequer-board ion sources is highly uniform, resulting in better optical properties and higher NB transmission, which also contributes to the higher power. The new power supplies would also improve the reliability and availability of the NB system, while allowing increased pulse length and operating voltage. An increase in operating current will also contribute to the NB power increase.

4.1.3. Accompanying diagnostics

The increased NB power will result in a significant increase in neutron fluxes. Several diagnostics will therefore have to be enhanced in order to reduce the impact of a high neutron background under these more ITER-relevant conditions. In particular, new γ-ray detectors will be implemented in order to improve energy and time resolution and a series of water and LiH attenuators will be installed on the ≥-ray cameras and spectrometers to allow the measurement of slowing down fast particles over a much wider range of plasma parameters and neutron fluxes. In addition, new solid-state detectors based on Silicon-on-Insulator technology are also foreseen for the neutral particle analysers, whose measurements are particularly affected by the background neutron noise. With regard to specific developments for ITER, radiation hard Hall probes will be tested and compact neutron spectrometers based on NE213 technology will be further developed. It is also planned to verify the potential of radiation hard CVD diamond detectors to measure UV and soft SXR radiation [27]. The performance of CVD monocrystaline detectors as compact neutron spectrometers will also be assessed.

4.2. HIGH FREQUENCY PELLET INJECTOR

Techniques for moderating large ELMs will be tested in JET with the installation of an ice pellet injector with a high repetition rate (up to 60Hz) [28], for ELM pace making, as demonstrated on ASDEX Upgrade [29]. This system will also provide a deep fuelling capability and give access to ELMy H-modes and hybrid modes, at higher β_N, under conditions of ELM control. The new pellet injector (see Fig.9) is based on the screw extruder technology developed by PELIN (Russia) and pneumatic acceleration, allowing the injection of an unlimited number of pellets with a very high level of reliability. The required performance both for ELM control (variable pellet volume 1-2mm^3, pellet speed 50-200 m/s and frequency up to 60Hz) and for plasma fuelling (variable pellet volume 35-70mm^3, pellet speed 100-500m/s and frequency up to 15Hz) are summarised in Table 1. The scientific exploitation of the high frequency pellet injector will be re-inforced with the installation of a new fast CCD camera explicitly devoted to imaging the pellet trajectory and investigating the pellet ablation rate in various magnetic configurations.

4.3. ENHANCEMENT OF THE RADIAL FIELD AMPLIFIER

Enhancements of the plasma configurational control will allow the enhanced JET capability to be exploited with ITER-relevant plasma shapes at high plasma current. For the present plasma control system, Figure 10 shows, as a function of plasma current and input power, the domain of safe operation (below either curve) with respect to detrimental disruptions induced by vertical displacement events for three ranges of plasma collisionality (ν^*). From this figure it is clear that the access to ITER-relevant collisionalities ($\nu^* \sim 0.1$) requires an upgrade of the plasma control system. The main element of this enhancement is an upgrade of the Fast Radial Field Amplifier [30]. It is completed by an upgrade of the Vertical Stabilization controller and of sensor measurements. The aim is to double the survivable ELM size, thereby permitting the study of Type I ELMy H-modes with lower, more ITER-relevant collisionality, and stronger plasma shaping. The conceptual design of the new Enhanced Radial Field Amplifier (ERFA) is aimed at increasing the output power with respect to the FRFA, providing a 12kV/5kA capability, thus doubling the current swing and enhancing the voltage capability. The structure of ERFA is similar to FRFA: four distinct units are connected in series at the output to achieve the desired voltage level. However, each unit of EFRA is rather different from FRFA, comprising a step-down transformer, a single-quadrant ac/dc thyristor converter, a capacitor bank, a dc chopper to control the dc link voltage and a four-quadrant inverter. This unit uses state-of-the-art power semiconductor components, which are able to provide faster switch-on and switch-off times and higher current capability than Gate Turn-Off thyristors. The ERFA unit was simulated in order to verify that the conceptual design is able to meet the specifications and demonstrate the adequacy of the circuit topology and the suitability of the design.

4.4. OTHER DIAGNOSTICS

The “JET programme in support of ITER” also includes new diagnostics for edge plasma parameters. The spatial and time resolution of both the Li beam measurement of edge density and the edge LIDAR will be improved significantly. The ECE radiometer will be upgraded to use the second harmonic, and thus achieve better spatial resolution, at higher frequencies. Burning plasma diagnostics will be reinforced by mostly upgrading the existing systems (see, e.g., [31]), together with a Fast Wave Reflectometer to determine the isotopic composition profile.

SUMMARY

The JET programme is strongly focused to help prepare ITER operation. With the most recent enhancements of the JET Facilities, ITER high priority issues are being addressed, such as optimizing ITER operating scenarios and qualifying the performance of heating & current drive and diagnostics in ITER-relevant conditions. Within the “JET programme in support of ITER” activities are underway to completely exchange the present wall materials (CFC) to a metal-dominated ITER-like material mix with mainly solid Be tiles in the main chamber and a full W divertor.

In addition, the overall heating power will be increased to ~ 45 MW by upgrading the NB heating system and an high frequency pellet injection system will be installed for ELM mitigation and fuelling. The plasma control system will be upgraded with a newly designed enhanced radial field amplifier. Finally, a significant number of new or upgraded diagnostics will be re-inforce the scientific exploitation of the new enhancements or allow tests in view of ITER needs. This large effort is motivated by a number of outstanding questions for which answers are required for ITER construction and operation. The most important objectives are to:

- Demonstrate that a beryllium wall plus an all-W divertor has sufficiently low fuel retention to meet ITER requirements and, if not, to develop appropriate mitigation and removal schemes.
- Characterize the main plasma wall interaction during, in between ELMs and in disruptions and explore plasma scenarios compatible with a low power handling Be wall which can be subject to melting.
- Study the formation of Be-W mixed layers and their impact on erosion and material migration.
- Study melt layer stability and melt damage evolution.
- Develop control strategies for detecting and limiting damage to Be and W plasma-facing components, such as relevant disruption mitigation and ELM power loss control systems.
- Develop integrated scenarios for an all-metal machine including impurity seeding strategies to replace the intrinsic carbon radiation to achieve acceptable divertor power loads in the current ITER baseline edge scenario.
- Demonstrate the compatibility of the foreseen ITER scenarios with a full W divertor and Be wall

CONCLUSION

"JET programme in support of ITER" will allow, on one hand side, an integrated development of plasma scenarios fully compatible with ITER-relevant wall materials and tested in D-T and in long pulses, which will save significant experimental time on ITER and, on the other hand, the optimisation of a number of auxiliary systems on JET rather than on ITER, to avoid significantly higher expenses"20 and down-time on ITER. The benefit for ITER can be significant and is likely to be both programmatic and financial:

- ITER objectives could be attained faster;
- savings on ITER capital or operating costs could result from full optimisation of several auxiliary components and control systems;
- savings on ITER operation costs should result from shorter development of scenarios

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Parameter	Required performance
Nb pellet/pulse	unlimited
Pellet Volume	
Vol. 1	Adjustable 1 to 2mm^3
Vol. 2	Adjustable 35 to 70mm^3
Injection frequency	10 to 60 Hz for Vol.1
	up to 15 Hz for Vol.2
Pellet material	Hydrogen, deuterium
Pellet velocity	Adjustable 50 to 200m/s for Vol.1
	Adjustable 100 to 500m/s for Vol.2
Reliability	98%
LHe consumption	< 40L/h
Availability	> 1 million

Table 1: Required parameters and performance of the high frequency pellet injector system

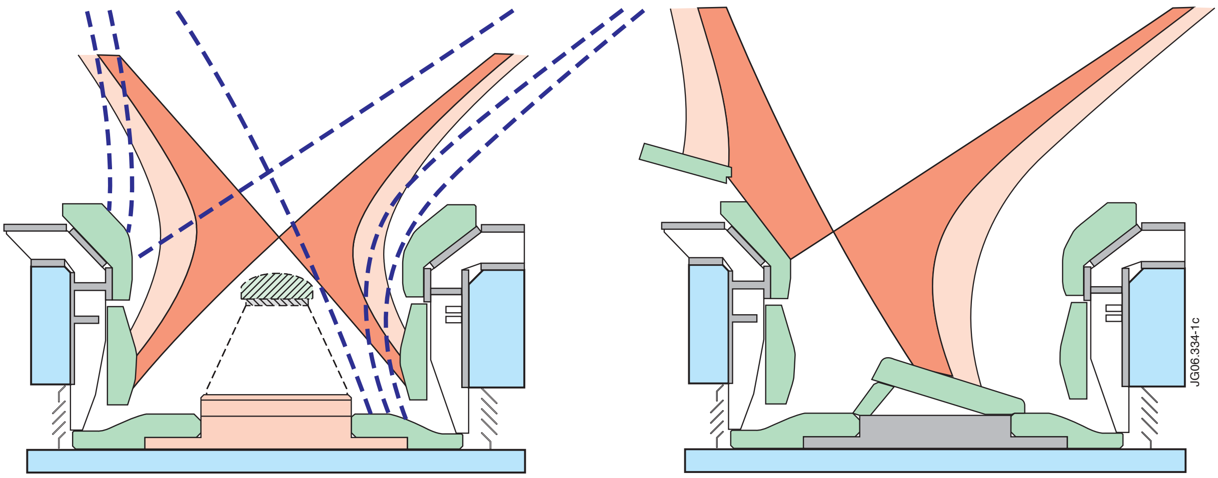

Figure 1: Magnetic configurations in previous (left) and present (right) JET divertors. The present divertor allows the strike points to be shifted inwards, to form configurations with high triangularity

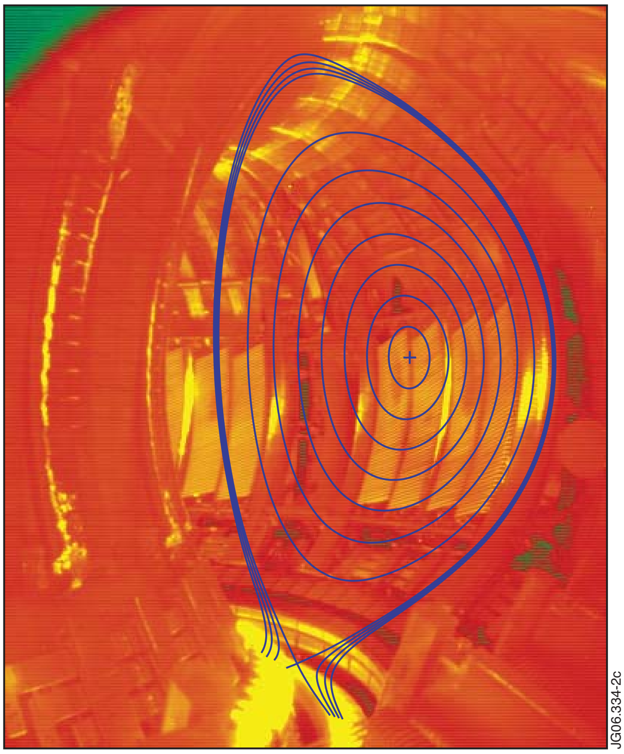

Figure 2: Example of an ITER-like high triangularity configuration made possible by the new divertor, viewed here with the new wide angle infra-red viewing system.

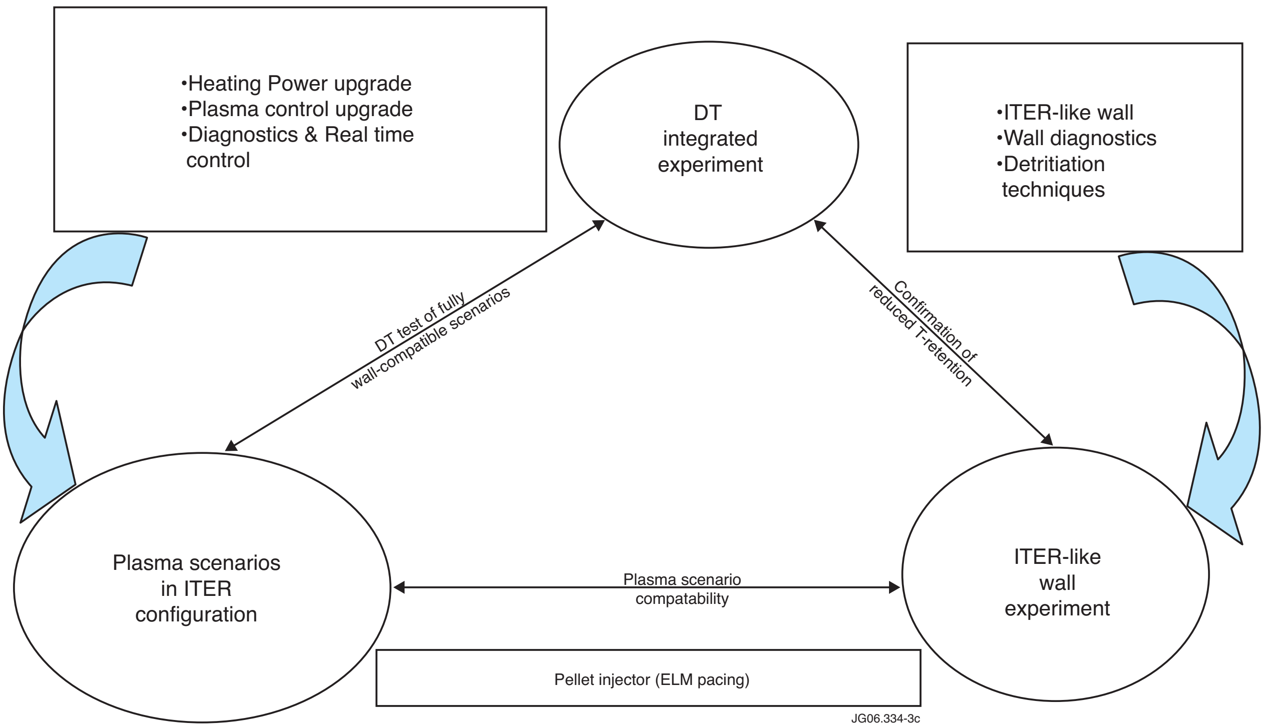

Figure 3: Synoptic diagram of the structure of the JET programme in support of ITER.

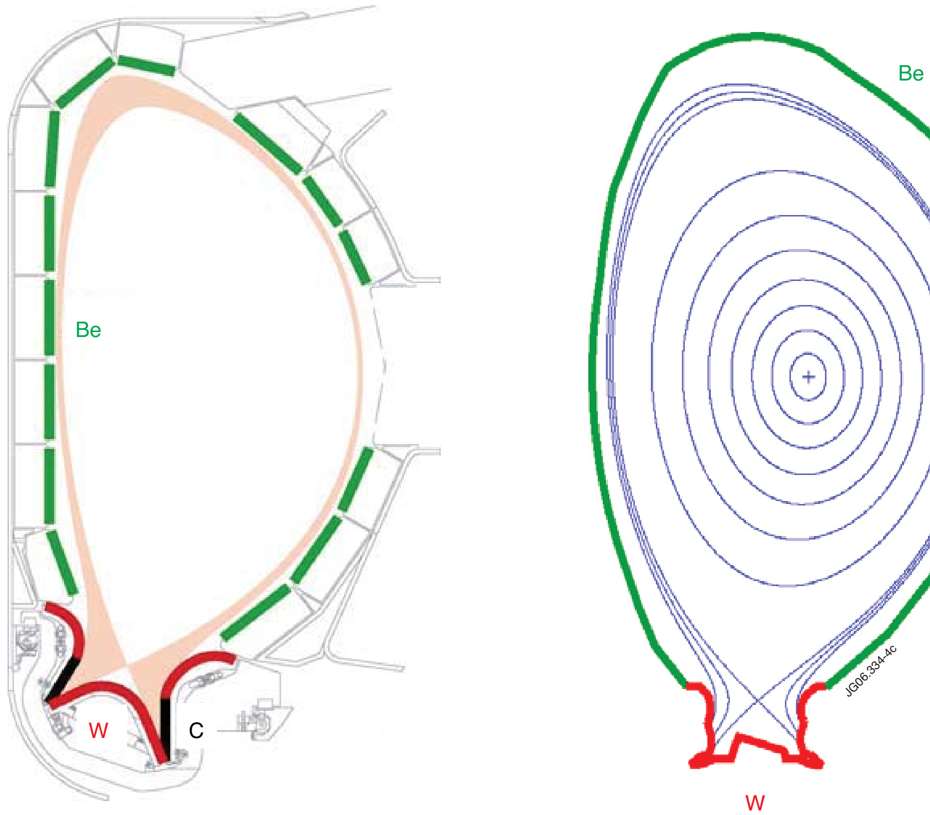

Figure 4(a): ITER primary materials choice – backup solution is all tungsten.
Figure 4(b): JET first priority is to test an all W divertor matches ITER fall back or second phase option.

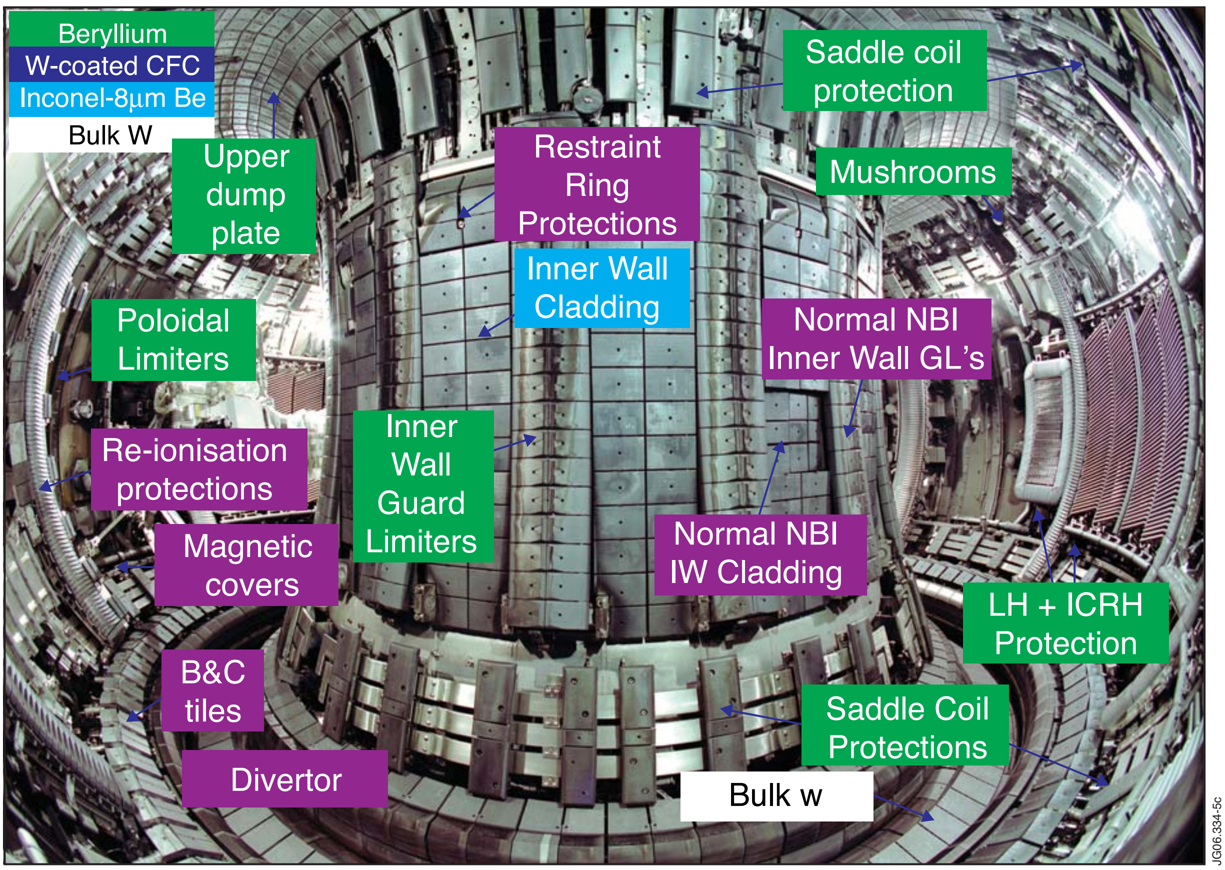

Figure 5: View of interior of JET. JET has 4404 main wall tiles to be replaced with 1700 beryllium tiles.

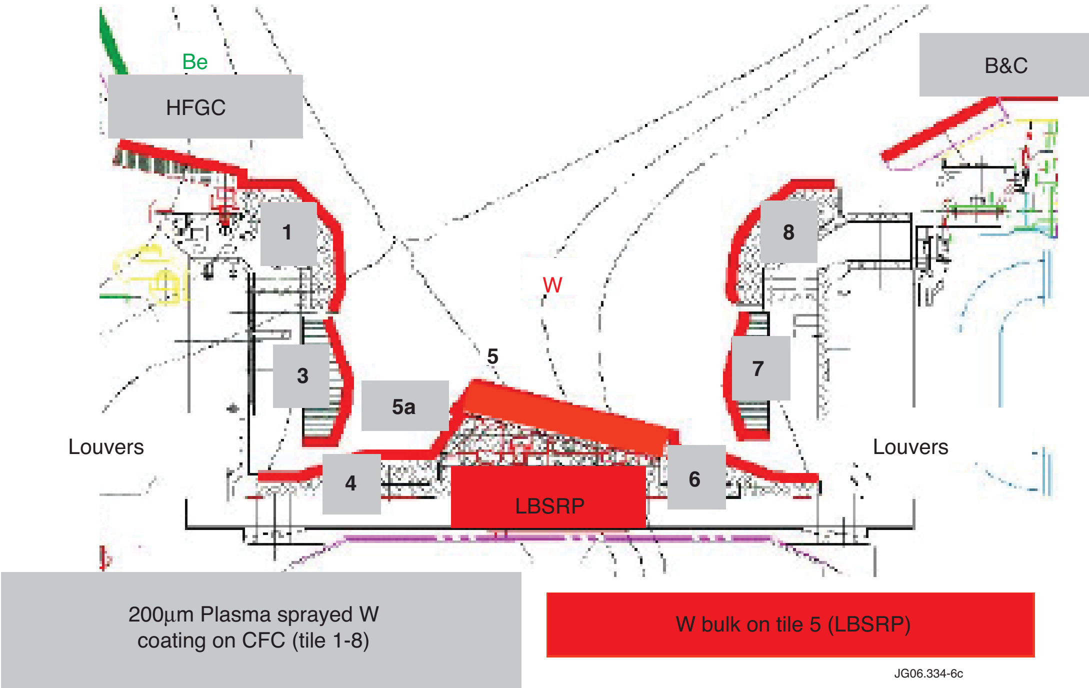

Figure 6: The JET full tungsten divertor.

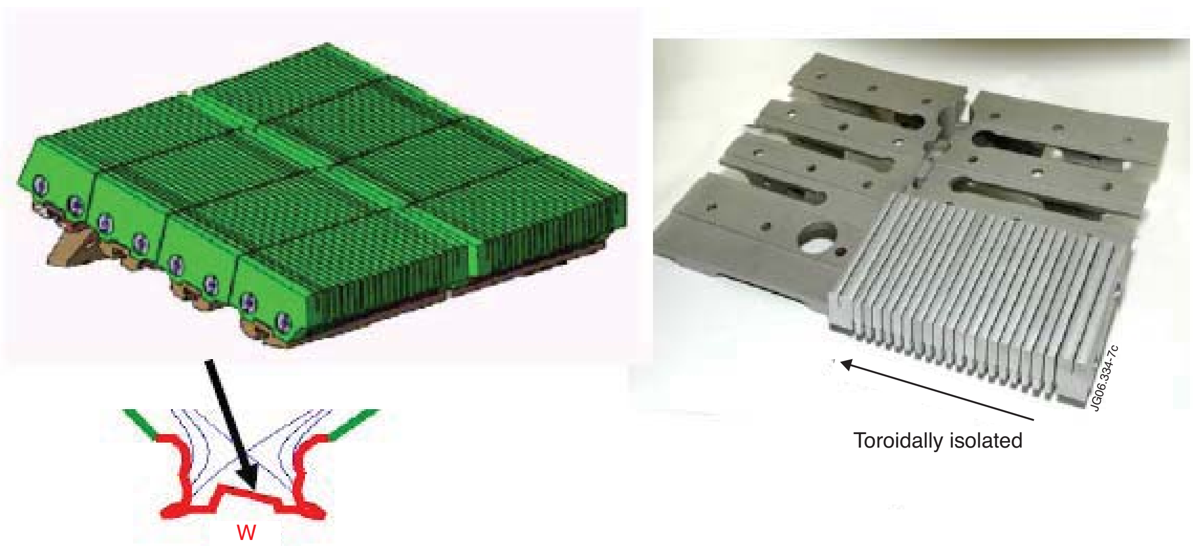

Figure 7: Bulk tungsten tile design for the load bearing plate in the future JET divertor.

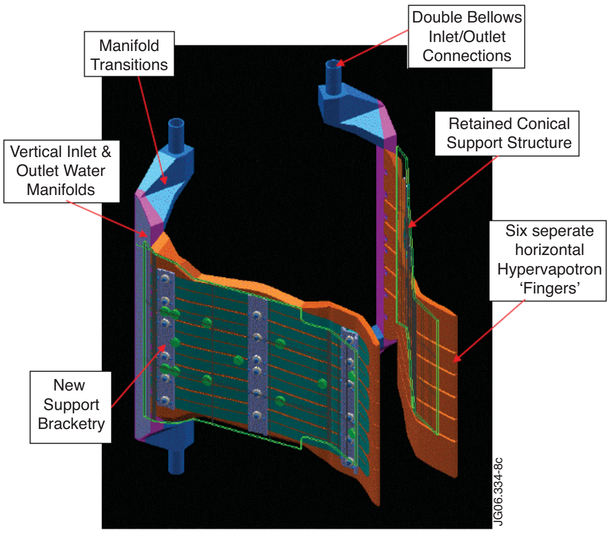

Figure 8: Catia Model of new MkIII Actively-Cooled Duct Scraper Panels, replacing original Hot and Cold side panel.

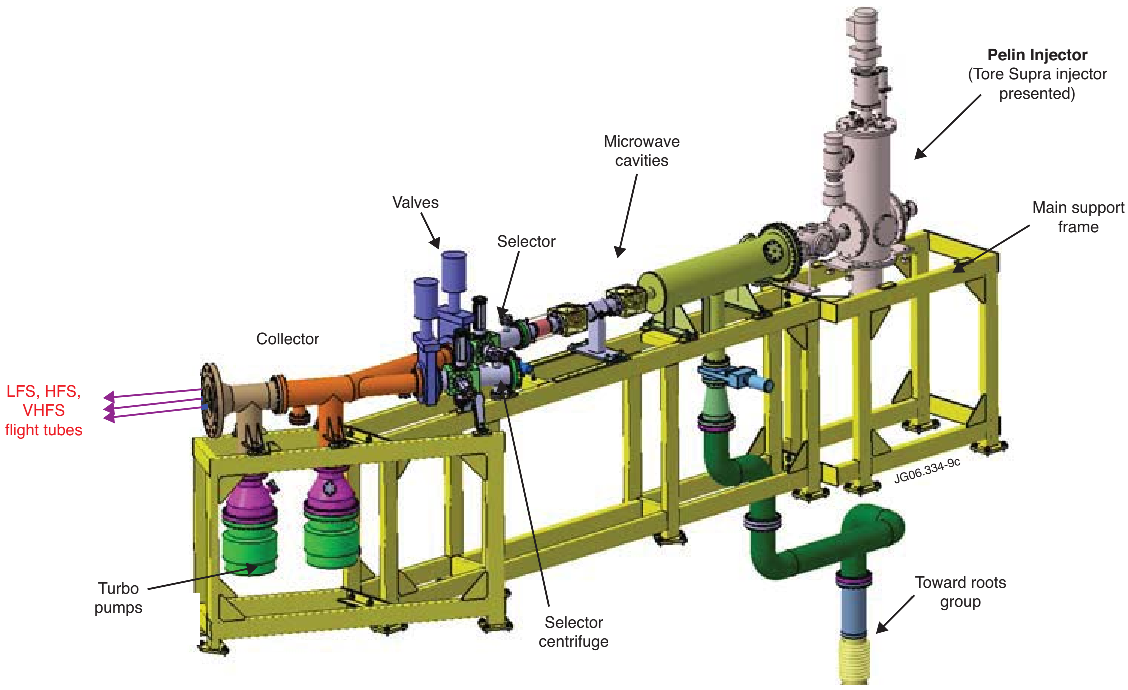

Figure 9: Overall drawing of the new high frequency pellet injector - injection line assembly. LFS, HFS and VHFS stand for low field side, high field side and vertical high field side, respectively.

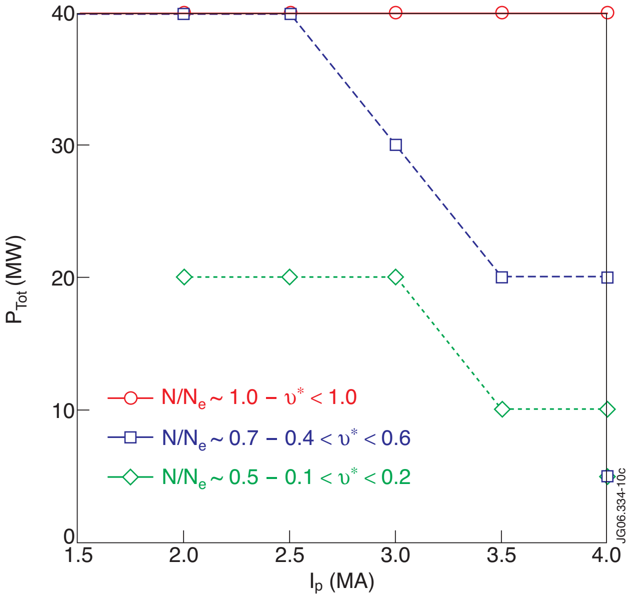

Figure 10: Domains of safe operation (below either curve) with respect to detrimental disruptions induced by vertical displacement events for three different ranges of collisionality (v^*) as a function of input power, P_{Tot} and plasma current I_p 23.