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The Role of JET for the Preparation of the ITER Exploitation

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ABSTRACT
The JET programme is devoted to the consolidation of ITER design choices and the qualification of ITER integrated regimes of operation. During the experimental campaigns carried out in 2008 and 2009 attention focussed on the test of the ITER-like ICRH antenna, the ITER scenario preparation, the verification of the adequacy of the ITER poloidal field coil design and the test of disruption mitigation methods such as massive gas injection. From 2011 the new ITER-like wall with all beryllium and tungsten plasma facing components, the neutral beam power upgrade and the enhanced control and diagnostic capability will allow key questions on plasma-wall interactions, fuel retention and plasma impurity control with the foreseen ITER wall materials to be addressed. Finally, feasibility studies have confirmed the option of installing an ITER-technology based 170GHz/10MW Electron Cyclotron Resonance Heating system for the control of MHD activity and the development of advanced tokamak scenarios, and 32 in-vessel coils for ELM control capable of producing magnetic perturbation spectra with a Chirikov parameter above unity for plasma currents up to 5MA. During the ITER construction phase, JET will be the only device of its class in operation and therefore the best machine to prepare ITER operation - saving time and reducing risk from the ITER programme.

1. INTRODUCTION

The efficient exploitation of ITER will depend crucially on the scientific and technical preparatory work performed in present day devices to optimize those aspects of the ITER design that are not yet frozen and to develop safe and effective operating scenarios. The Joint European Torus, JET, is ideally positioned to advance the state of ITER preparations by virtue of its size, ITER-like geometry, large plasma current and unique capability to operate with tritium fuel and beryllium plasma facing components. In the last couple of years, these features have allowed significant progress in the understanding of key ITER physics issues and in testing and validating the performance of new technologies. As examples of this work, this paper presents results from studies on the ITER integrated scenario preparation and ITER-like Ion Cyclotron Resonance Heating (ICRH) antenna and on the effect and mitigation of disruptions.

To take full advantage of the scientific opportunities offered by the JET facilities a series of strategic upgrades are an integral part of the ongoing JET programme in support of ITER [1] and will also be reported on here. The JET ITER-Like Wall (ILW) [2] will provide the first experience of tokamak operation with a Beryllium (Be) first wall and tungsten (W) divertor, as is planned for the activated phase of ITER. Compared to carbon, the new wall materials should demonstrate a major beneficial impact on fuel retention and on the lifetime of Plasma Facing Components (PFCs). The Neutral Beam Injection (NBI) system is being upgraded with the routinely available total injected power in deuterium being increased from the present 20MW to 30MW, while the maximum NBI pulse length is doubled to 20 s, or 40 s at half power [3]. Finally, to address the limitations of the JET Vertical Stabilisation (VS) system in controlling high current plasmas at the ITER-relevant collisionalities (ν^*) that will be achievable with the increased heating power [4] the JET plasma control system has recently been upgraded and successfully commissioned on plasma [5], using a model-based approach to minimise the time and risk to the main JET programme. A number of new diagnostics have also come to fruition.

With these enhancements, the JET programme in the coming years will provide an unequalled opportunity for advancing ITER preparations in conditions as close to those foreseen in ITER as is possible in any present fusion device. The main objectives of the JET programme now being elaborated for the 2011-2014 period are to:

Demonstrate sufficiently low fuel retention for a Be/W wall to meet ITER requirements.
Study the formation of Be-W mixed layers, their impact on W erosion, material migration and resistance to melting damage.

Develop control strategies for detecting and limiting damage to Be and W plasma facing components by steady state and transient heat loads.

Develop fully integrated scenarios for an all-metal machine, demonstrating the required confinement for ITER with impurity seeding strategies to replace the intrinsic carbon radiation and active mitigation of Edge Localised Modes (ELMs) for acceptable wall and divertor power loads.

The possibility of further improving the JET capability of preparing ITER operation has also been investigated. Specifically, two feasibility studies have been conducted for a 170GHz/10MW Electron Cyclotron Resonant Heating (ECRH) system and a set of Resonant Magnetic Perturbation (RMP) coils.

2. ITER INTEGRATED SCENARIO PREPARATION

The ITER Poloidal Field (PF) coils must be able to safely control the plasma during the current ramp up to 15MA, current flat top, and current ramp down phases of the discharge. The plasma internal inductance should be kept between $0.7 < l_i(3) < 1.0$ during the current ramp up phase and excessive increases of the inductance must be avoided during the current ramp down to maintain vertical stability while avoiding additional flux consumption from the central solenoid.

JET has performed ITER scenario demonstration discharges in deuterium in 2008 [6]. These results contributed to the modification of the ITER coil design and have been used as a reference for the helium discharges carried out in 2009 for the assessment of the ITER non-active phase performance [7]. For the helium campaign the neutral beam system was fully converted to He injection, using the technique of argon frosting to ensure He pumping.

Good control of the internal inductance is achieved with both gasses during the current ramp-up using a full bore plasma shape with early X-point formation at 0.8MA, equivalent to forming a diverted plasma at 4.5MA in ITER. In this scenario early heating is required to keep l_i below 0.85 when using the fastest current ramp rate available (0.36MA/s), still maintaining an MHD stable plasma up to $q_{95} = 3$ with a transition to H-mode that in deuterium JET discharges occurs at 7-9MW of input power and in helium at similar power levels (8-11MW).

During the current ramp-down the plasma inductance can be maintained within the ITER limits by remaining in H-mode, figure 1. If heating is not available simultaneous control of the internal inductance and avoidance of flux consumption can be achieved by combining an appropriate ramp-down rate with a strong reduction in plasma elongation to reduce the vertical instability growth rate.

Apart from higher flux consumption for helium discharges during plasma initiation deuterium and helium discharges are very similar with respect to key requirements for ITER plasma control. Studies in helium during the ITER non-active phase should therefore provide a good test of the operational space available in the current ramp-up and ramp-down phase of the discharge.

3. DISRUPTION STUDIES

Plasma disruptions are a key issue for all large tokamaks due to the electromechanical and thermal loads they can place on the tokamak structure. For ITER, and to a lesser degree also for JET with the ITER-like wall, a thorough understanding of disruptions and how they can be avoided or mitigated is critical. The JET disruption studies in support of ITER make use of a comprehensive set of recently upgraded diagnostics, including enhanced magnetic diagnostics capable of measuring asymmetries in the poloidal halo- and toroidal plasma current in four positions 90° apart [8] and improved infrared thermal imaging for power load studies [9]. Additionally, a fast Disruption Mitigation Valve (DMV) has been installed for disruption studies by Massive Gas Injection (MGI) [10].

3.1 ELECTROMAGNETIC LOADS

Disruptions cause electromagnetic loads by forces from halo currents, currents with a composite path, partially in the plasma column and partially in the plasma facing conductive structure and the vessel wall, and forces from eddy currents induced by the rapidly varying magnetic fields during the current quench.

During asymmetric Vertical Displacement Event (VDE) disruptions the plasma current and vertical current moment are n=1 toroidally asymmetric, leading to sideways forces of up to 4MN in JET [11]. Scaled with Noll's formula the equivalent forces are an order of magnitude larger in ITER [12,13], which is designed for 48 MN. The possibility of a rotating asymmetry at the 8 Hz ITER fundamental mechanical vessel resonance frequency is a concern. Results from JET [13] however indicate that large asymmetries (~10%) are observed only for short to moderate current quench times (up to 50ms, corresponding to ~250ms in ITER if scaled with the plasma cross section area [14]) and that the asymmetries are significantly smaller for longer quench times. This implies that, at the ITER vessel resonance frequency, large asymmetries will only be able to complete a small number of rotations, limiting the force amplification.

The halo currents can be reduced by up to 60% by massive gas injection if the Thermal Quench (TQ) is initiated before a significant vertical movement has taken place. Figure 2 shows the halo current fraction multiplied by the toroidal peaking factor as function of the delay between the thermal quench and a vertical displacement of 10 cm. A fast reaction time is essential for the successful

In order to keep forces from eddy currents tolerable, the current decay time must stay above the lower bound of $\tau_{CQ}/S \approx 1.7$ ms/m^2 for ITER (with S the pre-disruption plasma cross-section area). This limit can be reached with pure Ar MGI in JET, whereas the deuterium mixtures show a slower current decay.

3.2 THERMAL LOADS AND RUNAWAY ELECTRONS

The peak heat loads during the thermal quench can be reduced by enhancing the radiation with MGI. In the cooling phase up to 50% of the thermal energy stored in the plasma before the DMV is activated is lost, predominantly by radiation before the TQ. About 40% of the remaining energy is radiated during the TQ. Thus, only 30% of the initial energy is lost by convection to plasma facing components during the TQ, only a small fraction of which is found in the divertor. The radiated power shows a poloidal peaking factor below 1.5 during the thermal and current quench. In contrast, a peaking factor of 3.5 is found during the thermal quench in an unmitigated VDE, which in ITER could cause Be melting by radiation [15].

A survey of about 1500 JET disruptions shows that signs of runaway generation can be found in around 15% of all disruptions [16]. Runaway generation is successfully avoided by the injection of Ar/D2 or Ne/D2 mixtures, due to the suppression of the Dreicer mechanism. In contrast, injection of pure Ar leads to runaway generation even at low toroidal magnetic fields. Although runaways can be safely avoided by MGI in JET disruptions the density reached is still a factor 50 below the critical density for avalanche suppression which is essential in ITER where runaway currents of up to 10MA are expected due to the strong avalanche amplification.

4. ITER-LIKE ICRH ANTENNA

4.1. PERFORMANCE ON JET PLASMA

Plasma operation with the JET ITER-like Antenna (ILA) [17-21], based on a similar design concept as the ITER ICRH antenna [22] with a closely packed array of short low inductance straps, has demonstrated operation at ITER-relevant power densities (up to 6.2 MW/m^2 on L-mode, 4.1 MW/m^2 on H-mode) and RF voltages (42 kV, also on ELMy H-mode plasmas). No evidence of increased impurity production has been found at these power densities that are up to 6 times higher than hitherto achieved on JET [20].

In the ILA, the straps are mounted as four Resonant Double Loops (RDLs) arranged in a 2 toroidal by 2 poloidal array [18]. Each RDL consists of two poloidally adjacent straps fed through matching capacitors from a T-junction. Tolerance to plasma load variations is achieved by adjusting the matching capacitors such that the two branches of an RDL have approximately complex conjugate impedances. The impedance, Z_{CT}, at their connection point, referred to as Conjugate-T (CT), is then typically between 3 and 6Ω with optionally a small imaginary part to equalize the voltages between the straps of an RDL.

Two key issues with such an antenna are the design of a robust arc detection system to replace the standard Voltage Standing Wave Ratio (VSWR) protection in ELM tolerant operation and the control of the antenna matching elements in the presence of high mutual coupling between straps. Two different arc detection systems have been validated on JET H-mode plasmas; the Scattering Matrix Arc Detection system (SMAD) [23, 24, 25] and the Sub-Harmonic Arc Detection system (SHAD) [26]. SMAD, which can identify arcs also in low-impedance areas of the antenna by comparing measured probe voltages and directional coupler signals to those calculated using an RF model, was successfully commissioned and proved capable of protecting the antenna from arcs also during ELMy H-mode operation [25]. After tuning the system sensitivity SHAD also proved to be efficient and should offer an extra level of protection. Simultaneous matching of the four separate conjugate-T systems proved feasible, and confirmed the ITER-relevant use of a high power density conjugate-T antenna.

4.2 VALIDATION OF ITER COUPLING PREDICTIONS

The main issue of concern for ITER was the low coupling (0.8Ω/m) measured for the ILA on H-mode plasma with 5 cm strap to separatrix distance, lower than the originally anticipated 1.5Ω/m. To assess the implications of the measured coupling for the coupling predictions made for ITER using RF codes such as TOPICA [27] a strap-separatrix distance scan with well-diagnosed L-mode density profiles was carried out and the coupling compared to TOPICA modelling. Good agreement for the effective strap resistance per unit length, R'_{eff}, and the effective conductance at the RF probe, G_{eff}, within the error bars was found, figure 3 [21]. This is in agreement with earlier TOPICA validation on Tore Supra [28], DIII-D [29] and Alcator-CMod [27] and, provided the edge density profiles used are realistic, gives confidence in the predictive capability of the code for ITER.

5. ENHANCEMENT PROGRAMME (EP2)

5.1 ITER-LIKE WALL PROJECT

The JET ITER-like wall [2,30] is the principal experiment on plasma-wall interactions and plasma-compatibility with the material combination foreseen for the activated phase of ITER. The replacement of all Carbon Fibre Composite (CFC) tiles in JET is scheduled to be completed by the end of 2010, with plasma operation starting in 2011. In preparation for the new wall an extensive set of experiments was carried out in 2008 - 2009 to (i) develop techniques that ensure the safe operation with the new wall (eg. development of N_2 seeding techniques to reduce physical sputtering) and (ii) provide reference plasmas with carbon PFCs that will be repeated with the ILW (eg. to quantify fuel retention) [31].

In the new wall, the main elements in direct contact with the plasma (upper dump plates, inner and outer guard limiters) are made from solid Be tiles, segmented to minimise eddy currents and castellated to avoid thermal stress cracking. Be-coated (8μm) Inconel is used for recessed areas of the inner wall subject to lower heat and particle fluxes and W-coated CFC tiles are used in a few areas of the first wall exposed to high power loads, such as the neutral beam shinethrough area.

W-coated CFC is also used for most of the divertor tiles and an extensive R&D programme has been undertaken to develop and qualify suitable techniques for W-coating of CFC. The mismatch in CFC- and W thermal expansion is a key issue, which for thick W-coatings can lead to delamination and subsequent melting damage. The technique chosen, Combined Magnetron Sputtering and Ion Implantation (CMSII), combines a thin (14μm) W-coating with a molybdenum interlayer and provides good adhesion to the CFC substrate and, for all but the highest load areas, an acceptable expected life-time [32].

For the Load Bearing Septum Replacement Plate (LBSRP), which is the most heavily power-loaded divertor tile during operation with ITER-like configurations, a bulk-W tile has been developed [33]. To minimise electromagnetic forces during disruptions each tile consists of four poloidal stacks of 6 mm thick and 40 mm high W-lamellas, toroidally electrically insulated by Al_2O_3-coated molybdenum spacers.

5.2. NEUTRAL BEAM ENHANCEMENT PROJECT

The aims of the JET neutral beam enhancement project are to increase the deuterium neutral beam power to at least 34 MW, increase the beam pulse length at full power to 20s (40 s at half power if the two beam boxes are used in sequence) and to improve the reliability and availability of the system [3]. The power upgrade will allow stable H-mode operation at higher plasma currents and magnetic fields to push the development of ITER scenarios to lower ρ^* and ν^* and higher β_N and the increased pulse length will be essential to progress the hybrid and steady state scenarios for ITER on plasma pulse lengths greatly exceeding the current profile relaxation time.

The ion sources of the JET Positive Ion Neutral Injectors (PINI) are being changed from supercusp to chequerboard configuration, producing more molecular ions with higher neutralisation cross-sections, and the accelerators are being re-optimised for 125kV/65A deuterium beam operation with an increased extraction aperture (from 11 mm to 11.5mm) and reduced acceleration gap (from 16mm to 15mm). The four-fold increase in fractional and molecular residual ion power and the increased pulse length further requires a number of beamline components to be replaced by upgraded designs, the most important being the actively cooled duct scrapers to cope with the beam re-ionization in the port.

Two reconfigured PINIs were installed on JET and conditioned to operating voltages above 120kV in 2009, producing a record JET power on plasma for a single PINI of 2.08 MW. This gives confidence that the whole system, when commissioned, should be capable of delivering a maximum power to the plasma in excess of 34MW (30MW routinely) [34].

5.3. PLASMA CONTROL UPGRADE / PCU-2

A project dedicated to the enhancement of the JET Vertical Stabilization system was launched in 2006, including an upgrade of the power supply of the Radial Field Amplifier (ERFA) and of hardware and software of the VS control system (VS5). The main aim was to double the JET capability in stabilising high current plasmas when subject to perturbations, in particular large ELMs. ERFA was delivered to JET in early 2009, tested on dummy loads [35] and was ready for plasma commissioning by the summer of 2009 [5].

Together with the ERFA amplifier, the JET VS system has been equipped with a new digital real-time controller, VS5, based on Advanced Telecommunication Computing Architecture (ATCA®) hardware. The design of VS5 has been driven by the requirements of speed, reliability, flexibility and robustness against interrupts during communications. The choice of the new controller was guided by electromagnetic modelling of the interaction between the plasma and surrounding conductors and by simulations of closed-loop controller behaviour. All essential features of the VS5 controller were commissioned in advance of the installation of the new radial field amplifier. Following the installation and connection of ERFA to the JET radial field coils, a campaign of plasma experiments was dedicated to the basic commissioning of ERFA, the exploration of the operating range of ERFA/VS5 and the physics optimisation of the new system. Experiments started with quiescent plasmas, with increasing values of plasma current and instability growth rate, $\gamma \sim 100 - 1400\text{s}^{-1}$, progressing to test the ERFA/VS5 response to controlled perturbations, then to small ELMs and, finally, large ELM regimes. A special effort was devoted to assessing different options for connection of ERFA to the radial field coils, and a low inductance configuration was selected. The new ERFA/VS5 system behaved well in challenging ELM conditions, with more than 1MJ of rapid energy loss, and in high current operation up to 4.5MA. Extrapolations based on these results indicate that the ERFA/VS5 system amply met its design objectives and will allow safe JET operation in a significantly expanded parameter space.

The Plasma Control Upgrade (PCU) project and the integrated ERFA/VS5 commissioning have provided a full size test of the application of a model-based approach to design and implement an essential subsystem in a tokamak environment. It has demonstrated clearly the benefits of an approach based on phased commissioning, modelling and offline algorithm validation in terms of machine safety and optimisation of the use of experimental time. This has proved to be a very powerful method, which could be extended to ITER for the design of both the vertical stabilization and the current and shape controller.

5.4. DIAGNOSTICS

About twenty different projects have been launched in the framework of the EP2 programme to improve the diagnostic capability of JET, especially in relation with the ILW test. To date, five projects have been successfully closed (sometimes exceeding the original goals) and the others are expected to be closed immediately after the completion of the EP2 shutdown. The main improvements are in the following areas: spectroscopy and other edge diagnostics for the ILW (eight projects), upgrade of fast particle diagnostics (neutron and gamma ray cameras and spectrometers, neutral particle analyser), test of new ITER-relevant diagnostic concepts and technologies (seven projects), improvements of real time measurements and control (two projects) and improvements of the time and spatial resolution of profile measurements (ECE and multi band sweeping reflectometer). Significant efforts are devoted to increase the protection capability for the ILW. To that end, twelve robust near-IR cameras and eight pyrometers are being installed. Data from these will be integrated into a new global Vessel Thermal Map that will be used to determine the appropriate protection response from various actuators (plasma shape, heating systems, gas fuelling) depending on where overheating occurs. Most of the existing JET control software is being updated in the process of this project.

Finally, the installation of the ILW presents a unique opportunity to perform an in-situ calibration of neutron diagnostics in an environment with the ITER-relevant materials, particularly Be which is a neutron multiplier.

6. FEASIBILITY STUDIES

With the EP2 JET will be able to qualify ITER scenarios with Be/W PFCs. The JET capabilities of preparing ITER would be further improved by the installation of an ECRH system and a set of RMP coils.

The main functions of an ECRH system on JET would be to ensure core electron heating, to control MHD activity (such as Neoclassical Tearing Modes and sawteeth) and to control the safety factor profile in Advanced Tokamak (AT) scenarios by localized current drive. This method has been developed on medium size devices but only a machine like JET can access specific ITER relevant conditions. For example, the very long period “monster” sawteeth generated by a population of energetic particles that lead to very dangerous behaviour can only be produced on machines of the JET class.

The ECRH feasibility study [36], carried out in collaboration with the Russian Federation, has demonstrated that the choice of 170GHz (2^{nd} harmonic) is optimal for operation around a toroidal magnetic field of 3T. This is the value mostly used on JET for the development of advanced regimes of operation. The choice of 170GHz clearly makes the best use of the R&D carried out for the ITER ECRH system. Simulations performed with state-of-the-art codes, in particular the beam-tracing code GRAY [37] and the integrated modelling suite CRONOS [38] show that a power level of 10MW is adequate for NTM/sawtooth control and current profile control in AT scenarios.

The mitigation and suppression of ELMs is one of the main issues for reliable ITER operation. The use of RMP is at the moment the only method that has shown the possibility of ELM suppression [39]. While the ELM losses are usually expected to increase as the pedestal collisionality is decreased, JET results [40] indicate a clear correlation with the normalized Larmor radius (i.e. with machine size). Lacking a model for the ELM suppression, the only possibility to progress in this field in the near term is to test the capability of a RMP system on JET.

The RMP feasibility study [41], made in collaboration with the US DOE, has demonstrated the feasibility of a RMP system designed using the same assumptions as the ITER system. Two rows of 24 and 8 internal coils respectively are positioned above the outer midplane. The current in the coils is limited by structural considerations to 60kAt, which is enough to produce a Chirikov parameter (evaluated in the absence of plasma response) for n = 3 and n = 4 above unity up to a plasma current of 5 MA. The RMP spectrum produced in this way has a large flexibility and allows to reproduce both the spectra produced in the existing experiments and that envisaged for the ITER RMP system.

CONCLUSION AND OUTLOOK

By virtue of its main characteristics (large size, ITER-like geometry, large plasma current capability) and new enhancements nearing completion (ITER-like wall, neutral beam and diagnostics enhancements) JET is the ideal machine to advance the state of ITER preparations. In the near term, these capabilities will be fully exploited in characterising the plasma-wall interactions and plasma compatibility of the Be/W first wall and in developing and optimising safe operational scenarios for all-metal walls in ITER-relevant conditions. This work is planned to lead up to a full deuterium-tritium campaign in the 2015 time frame for fully integrated tests of the Q = 10 ITER baseline scenario, including the required active techniques for plasma-wall compatibility (impurity seeding, active ELM mitigation) in a metallic machine.

In the longer term, recent feasibility studies have confirmed the option of installing a 170GHz/10MW Electron Cyclotron Resonance Heating system based on ITER technology and 32 in-vessel RMP coils for ELM control. If realized these upgrades would allow JET to significantly advance preparations for ITER operations - saving time and reducing risk from the ITER programme.

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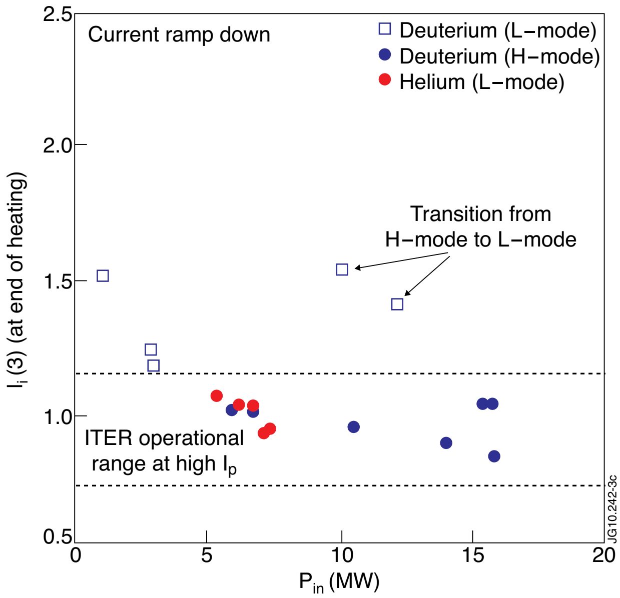

Figure 1: Plasma internal inductance at end of heating phase during current ramp down at 0.14MA/s.

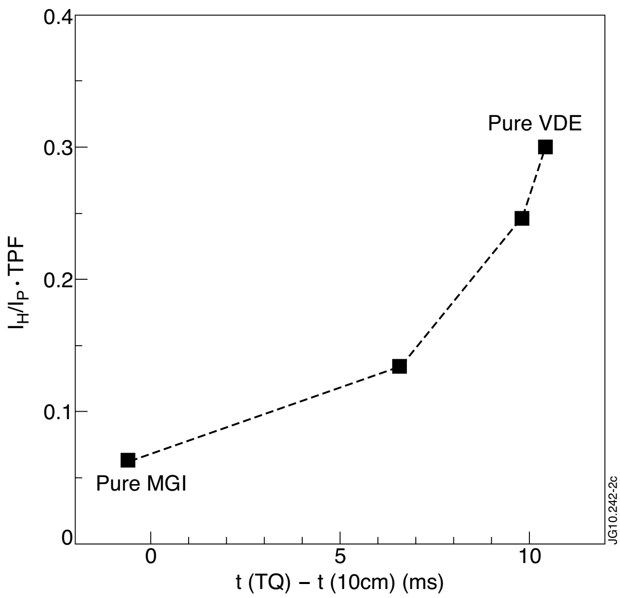

Figure 2: Halo current fraction times toroidal peaking factor as function of delay between the thermal quench and a vertical displacement of 10 cm for fast ($\tau_{growth} \approx 5ms$) VDEs.

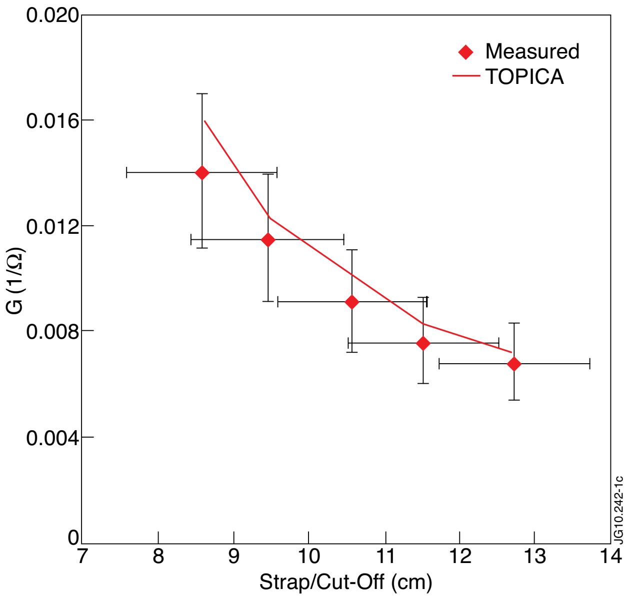

Figure 3: The coupling in terms of effective conductance calculated from the TOPICA data. Representative error bars are shown of ±1cm on position and ±21% on power.