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ABSTRACT
The UKAEA-Euratom Association has now operated JET for the European Fusion Development Agreement for nearly five years, as the world-leading facility for ITER studies and featuring beryllium and tritium experience, fusion product detection and real time control of a reactor relevant plant. Developments continue in diagnostics, control, plasma heating systems and remote handling and many new systems are being installed as part of the Enhanced Performance up-grade. Facility operation requires machine protection systems based on sophisticated component analyses, to constrain operation within boundaries consistent with the desired plant life. Analyses show that life consumption of the key plant is $\simeq$10% at present, with operational limits of 4T, 5MA and 850t Vertical Disruption Event force.

INTRODUCTION
Since January 2000, JET, the most sophisticated large tokamak facility in the world, has been collectively utilised under the European Fusion Development Agreement by the European fusion research laboratories. The UKAEA-Euratom Association has been operating JET throughout this period in support of these scientific activities, primarily carried out in preparation for ITER. The torus assembly and supporting systems have been upgraded in very many ways since the original design was elaborated in the 70's, with another set of enhancements presently being implemented. Key developments include the introduction of ITER-like plasma shaping including high triangularity, divertor modifications to suit plasma currents up to 4.3MA, neutral beam heating energy and power up-grades, increasingly detailed virtual reality systems in support of remote handling and proof of principle (1MW) conjugate-tee ICRH studies. Additional new features have been developed, such as on the JET NB injectors where now one PINI on an injector box can be operated in helium simultaneously with deuterium beam operation on other PINIs on the same box. This capability enables several new experimental possibilities including beam emission spectroscopy; double charge exchange processes from helium neutrals, centrally deposited high energy helium ions for alpha particle simulation etc., all in plasmas heated by high-power deuterium beams. As a result of all these and other modifications, JET continues to offer the world the most advanced capability for fusion research in the tokamak field, with key ITER-relevant features of plasma geometry, pumped poloidal divertor, beryllium in the torus and a complete tritium fuel re-cycling system.

Thorough QA in procurements, installation and maintenance have preserved the reliability of the machine such that in 2003 the experimental campaigns achieved more operating days and higher power plasma heating than any previous operational year [1]. As the rest of this paper will elaborate, JET is extremely well placed to continue its lead role in fusion research until ITER is ready to take on this role.

2. TRITIUM SYSTEMS
JET has two truly unique capabilities in the present world of fusion research, namely a tritium facility and (as a result of the tritium itself and the radioactivity induced in the load assembly by the

DT neutrons) a very comprehensive remote handling system, described more fully below. Tritium can be injected as gas puffed into the torus or as high energy neutral beams at energies up to 102keV (so far). Using an original site inventory of 20.5g, efficient recycling allowed 99.5g of tritium to be injected during the 1997 DTE1 campaign. Scrupulous accounting and comprehensive active gas handling systems have always been used to monitor, recover and retain the tritium, so that of the 13.8g surviving radioactive decay since delivery, 9.9g are still available for use while 2.6g have been committed back to the supplier, as tritiated water for recycling. Only 0.04g has been released to the environment, mainly as gaseous stack discharges at less than 5% of the site authorisation limit, and less than a gram is held up in components such as divertor tiles in the vacuum vessel. The remaining 0.5g lies in an earlier set of divertor tiles and associated carbonaceous flakes held in controlled storage.

Figure 1 schematically depicts the plant comprising the tritium fuel cycle of the machine, for which a full description can be found in ref [2]. The Trace Tritium Experiments of 2003 featured injection of 0.4g of tritium as puffed gas and 5g as 100kV (1.1MW) neutral injection. This 5.4g total was subsequently fully recovered and is again held on the uranium powder beds (called "Product Store" in fig.1) awaiting future experimental demand. Tritium monitoring developments continue, eg some of the new diagnostic systems due to be installed in the current shutdown include Quartz MicroBalance mass variation detectors and limiter tiles with embedded tracer layers, to improve the measurements of material erosion and re-deposition, and hence of tritium retention in the torus [3].

3. REMOTE HANDLING

After any significant tritium operation of the machine, the gamma radiation field inside the torus (largely from activation products such as Co^{58}, Co^{57} and Co^{60} with half-lives of 72 days, 270 days and 5 years respectively) can be many mSv/hr, far above the ALARP target of 350µSv/hr used at Culham for brief manned entry. Such high levels would take many months to decay sufficiently (see fig. 2) and so a remote handling capability is mandated for both planned in-vessel modifications and remedial maintenance.

Successful remote handling in any application requires very thorough involvement of the remote handling system operations and tooling experts with the designers of all the equipment in the remote handling zone. This in turn necessitates a firm management policy and rigorous updating of the design configuration files. These files are used for virtual reality training of the operations, complementing actual full-scale trials in the mock-up facility, which mimics the whole interior of the torus albeit with varying degrees of detail from one octant to another. The detailed configuration files are also used for piloting the transporter (a ten metre long, 19-joint boom) in the actual machine, using highly detailed images similar to that of fig.3. The boom is driven in a sequence of pre-selected steps using standardised taught files to control all the joints etc. Warning of possible clashes arising during the boom motion is achieved using alarm trigger levels on the errors in the sensed boom position and velocity. The boom places the servo-manipulator at the task site, and then the task is carried out live using many cameras and a man-in-the-loop control system.

The JET RH system [4] continues to demonstrate many ITER-relevant features, including welding, cutting, manipulation and transfer of heavy loads (≤300kgs) making use of the newly developed force feedback system, cleaning, detailed 3D metrology (to further update the files, e.g. to account for actual torus surface shapes with their manufacturing tolerances) and of course installations and disassemblies including many bolted components. In the 2004-2005 JET EP shutdown, many enhancements are being installed using essentially all these capabilities, extending over about ten months of Remote Handling operations. The considerable success of the JET RH group can be attributed to seeking the maximum possible adaptability of solutions, use of proven technologies and components, avoidance of more than two simultaneous RH actions, provision of tool and component racking structures to facilitate the tasks and above all, exhaustive preparatory work especially in all the associated design and planning phases.

4. OPERATIONS MANAGEMENT

The UKAEA is responsible for operating JET in the sense of providing a facility for experimental direction by scientists seconded from the various European Laboratories. While the management structure has changed considerably since the days of the Joint Undertaking, the overall reliability and availability of the machine has no discernible trend with time, as shown in fig.4. An important aspect of the operational responsibility is to put in place control systems (hardwired, software-based and managerial) to inhibit any unnecessarily risky operation of the machine. Particularly stringent limits apply to thermomechanical stressing of the toroidal field coils (related to pulse duration and peak current) and electromechanical stressing of the vacuum vessel as a result of Vertical Displacement Events (VDE). The maximum vertical force, created by a VDE, ever measured in JET was ~500t, although scenarios with the potential for much larger forces (>800t) were run before 2000. The potential for high vertical forces is reached at lower plasma currents with the present generation of ITER-like high-triangularity configurations, which exploit the poloidal field system flexibility conferred by the four divertor coils inside the vessel.

The JET vacuum vessel is supported and constrained in such a way that vertical forces upon it produce a poloidal rolling motion (e.g. towards the major axis at the top and away from it at the bottom) which in turn induces radial forces on the vertical ports. In addition, any toroidal asymmetry in the disrupting plasma generates a bulk lateral movement of the vessel. The net result is relative movements of the vertical ports which create strains in the port roots, estimated to be up to 0.35% for a movement of 22mm, which would correspond to a ≈1000 cycle lifetime for an 800t VDE in a modern configuration [5]. Fortunately operation so far has not been observed to create displacements above 3.5mm rolling and 7mm sideways motion, asynchronously and therefore with negligible implications for the fatigue life of the machine.

Software protection is used for much of the plant, e.g. the safe operating envelope of the OH solenoid coils, which have a complex thermomechanical interaction with the toroidal field coils. Hard-wired trips act as a back-stop protecting, for instance, individual coils from being overheated by I^2t. In addition, experimental planning requires management agreement before extreme plasma conditions affecting the toroidal field coils or vacuum vessel (or neutron production) can be explored. JET was designed for 3.45T toroidal field operation but has a present operational maximum of 4.0T. At high toroidal field, the feeder bar region of the TF coils is highly stressed by both differential heating and self-field stresses. Modelling has produced a safe working domain for combinations of these two parameters, in terms of fatigue life consumption. Management agreement is required when the projected TF life consumption of a proposed experimental sequence is significant.

Like any other tokamak of conventional aspect ratio, JET suffers from major disruptions, those occurring in the programmed plasma current flat-top amounting to about 8% of shots. Fortunately the statistics are sparse, but the incidence of minor vacuum vessel leaks can be roughly correlated with the number and severity of disruptions accrued, leading to developments in experimental planning guidance aimed at avoiding too many losses of operational days for vessel repairs. Figure 5 shows the integrated leak probability for two different fitting functions versus shot number in JET. Actual leak occurrences are also shown in this figure.

As a result of these management controls, software interlocks etc, and of course a robust original design, the fatigue life consumption of the principle components of JET is reckoned to be less than 10%, despite over 20 years of highly productive use, much of it well beyond the original design intent. The operational limits currently in force are 4.0T, 5MA and 850t potential vertical force (this force being very strongly dependent on the plasma shape).

5. PLASMA DIAGNOSTICS AND CONTROL

Collaboration with the European fusion laboratories and others beyond the European Union, in particular the USA and the Russian Federation, has yielded many valuable improvements to the JET diagnostic and control systems, some unique to JET as a result of the thermonuclear capability and large plasma current, ie the ability to confine charged fusion products. An elegant example of this class includes the dual-array (2D) neutron and γ detector camera, used in the Trace Tritium Experiment to image γ-emission arising from the $^{9}Be(\alpha, n\_)C^{12}$ knock-on reaction and hence the source distribution of high energy alpha particles [6]. Figure 6 shows the result, derived using MHD equilibrium constraints on the 2D tomography, for a case where ICRH and NBH were combined. Here the alpha source distribution is consistent with fast ion localisation predominantly in trapped (banana) orbits. The spatial distribution of the 14MeV neutrons was also accurately determined during TTE, yielding valuable data on the effects of the various heating schemes. Still on fusion product diagnostics, the Magnetic Proton Recoil system now provides an absolutely calibrated neutron flux measurement with an energy resolution allowing the ion temperature of the reacting species to be calculated [6].

Recent JET developments in plasma equilibrium control include an eXtreme Shape Controller (XSC) and Real-Time Control (RTC) [7,8]. The XSC is a valuable addition to the feedback control of the plasma boundary, particularly important for maintaining accurately the desired evolution of elongation, triangularity and divertor strike points throughout a plasma shot, as illustrated in Fig.7. RTC makes use of a wide range of real-time and signal processing physics analysis outputs characterising the plasma with typically only a few ms of delay. Amongst other things it provides a means of achieving simultaneous control of the plasma current density profile in three zones across the minor cross-section, vital for the successful creation and sustainment of Internal Transport Barrier (reversed magnetic shear) configurations. Polarimetry and magnetic diagnostic signals are combined to yield the q(r) profile every 20ms, which is compared to a reference profile and the resulting error signals are used to control the LHCD, NBCD and ICRH.

6. NEW ENHANCEMENTS AND THE FUTURE

JET is presently in the midst of a one-year shutdown to install a number of new plasma diagnostics, enhance the power handling capability of the divertor tile structure, effect some refurbishments of supporting plant and prepare for the installation in autumn 2005 of an ITER-like ICRH antenna. The new plasma diagnostics are described comprehensively in references [9,10] and include upgrades in Magnetic Proton Recoil and Time of Flight neutron spectrometry, two new Lost Alpha detectors, a variety of Tritium Retention Diagnostics, Bolometry and IR thermography, Electron Cyclotron Emission, High Resolution (edge region) Thomson Scattering, Charge Exchange spectroscopy and MHD spectroscopy using a multi-mode Toroidicity-induced Alfvén Eigenmode antenna. Innovative data acquisition technology is also being installed to allow storage of up to more than 10 Gbytes of data per shot during the next experimental campaigns.

The Mk II Gas Box divertor assembly is being modified to the “Mark II High Delta” (i.e. high triangularity, ≈ 0.56 at the lower x-point) version, which involves introducing new High Field [-side] Gap Closure tiles and a new thermal-Load Bearing Septum Replacement Plate and associated tiles. Altogether 196 new tiles will be installed, with carefully shadowed edges so that the divertor can cope with the convected divertor power and energy loadings associated with 10-second, 40-MW plasma heating, extending to 50MW with some allowance for alpha power.

Some of the general plant refurbishments amount to replacements of dated and relatively unreliable control circuitry (e.g. for poloidal field supplies including Ohmic Heating) while others aim to improve the behaviour of the plant. These include boosted back-electron suppression supplies and cooling plates in the neutralisers (to raise the gas density and hence the neutralisation efficiency) in the neutral injectors, following recent proving trials in the test-bed [11]. Still other refurbishments have become necessary for environmental reasons, e.g. replacing the halon gas fire extinguisher system with one based on FM-200, a more benign fire-suppressant.

The most technically demanding new enhancement is the ITER-like ICRH antenna. When this is installed, with its remotely adjustable antenna circuit tuning capacitors immediately adjacent to the antenna straps, following the design pioneered in Tore Supra [12], it is expected to provide a 7MW, 30 – 55MHz capability. The optional conjugate-tee inter-strap phasing should confer a significant immunity to the rapid variations in coupling otherwise caused by Edge Localised Modes in H-mode operation, hopefully demonstrating improved reliability and power delivery capability of ICRH in highly-shaped poloidal divertor tokamaks.

CONCLUSION

JET is well poised to continue its lead role in the world effort on magnetic confinement fusion research, exploiting the flexibility and robustness of the original design, a great number of major modifications since then and a well-managed, QA-controlled installation and maintenance programme to provide highly relevant support to ITER for the foreseeable future. The present capability matches the requirements for testing a uniquely wide range of ITER-relevant systems and scenarios and after the enhancements now in hand have been implemented, it will be even better suited to such tasks.

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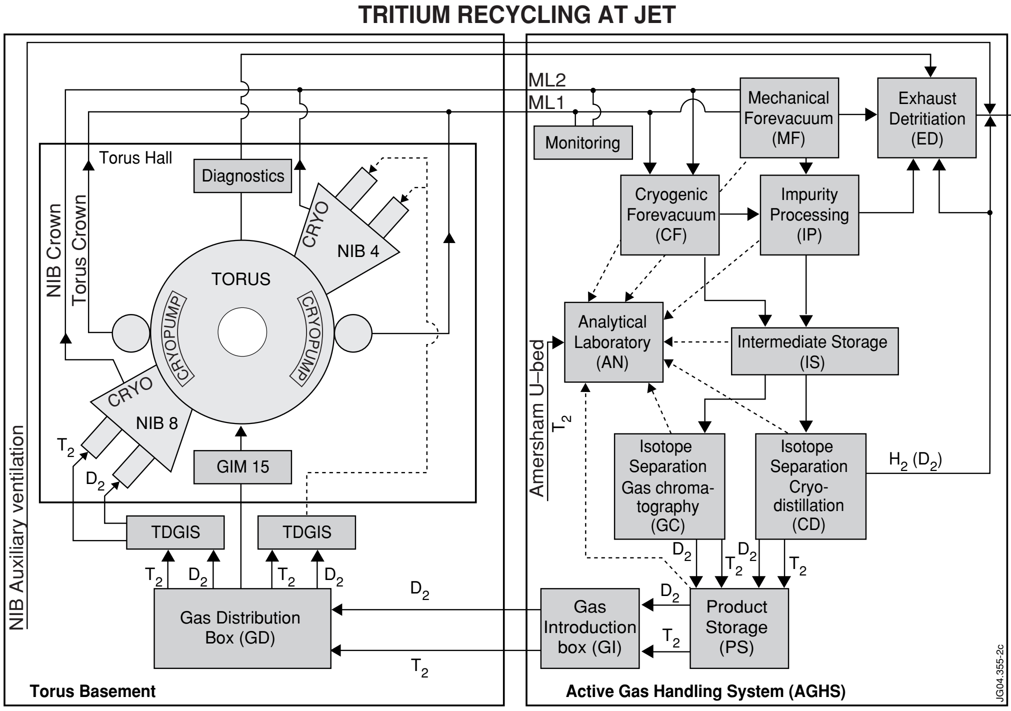

Figure 1: Schematic layout of JET tritium recycling plant.

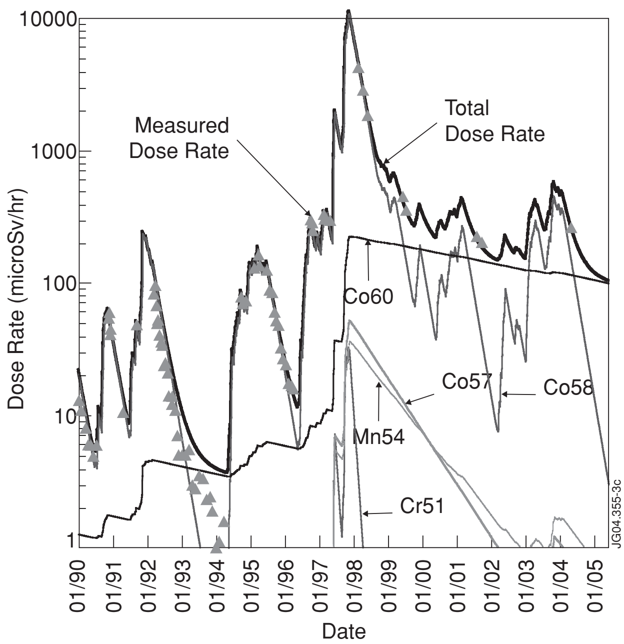

Figure 2: JET torus radioactivity levels versus time, showing the principle isotopes involved.

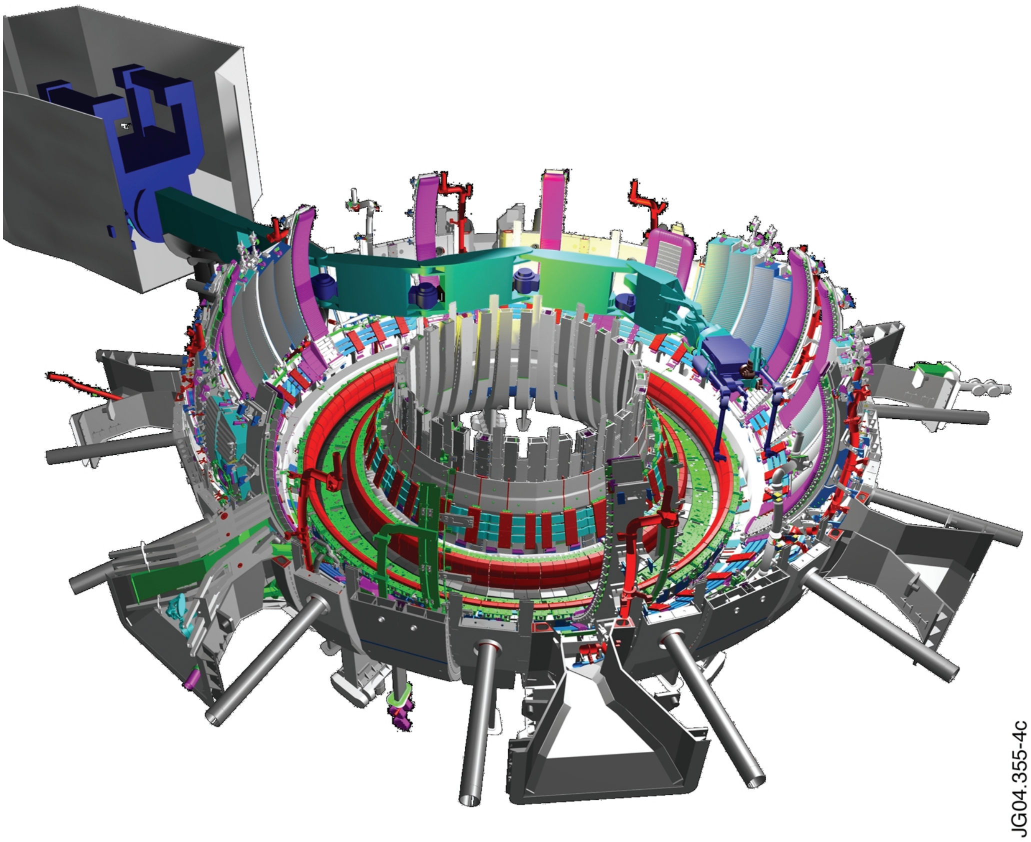

Figure 3: Virtual Reality cut-away image of JET in-vessel remote handling transporter boom with "Mascot" servo-manipulator.

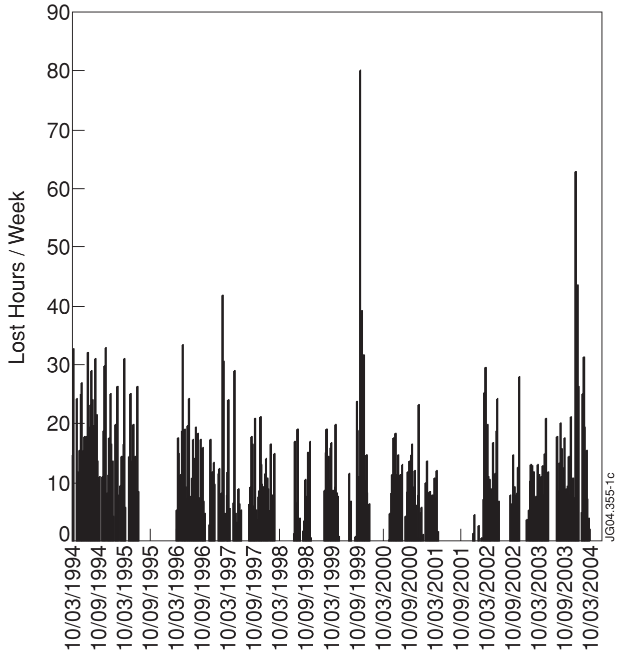

Figure 4: JET operations outages 1994 – 2004.

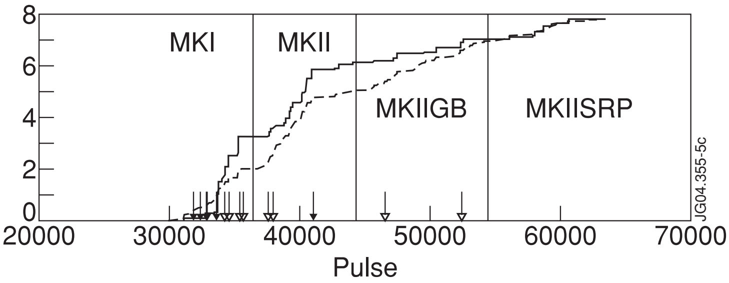

Figure 5: Integrated leak probability function since 1994 versus shot number in JET. The dashed curve is a power-law model of the disruption force, the solid curve is a threshold model. The different poloidal divertor arrangements are labelled by mark number, arrows indicate leaks and other in-vessel events.

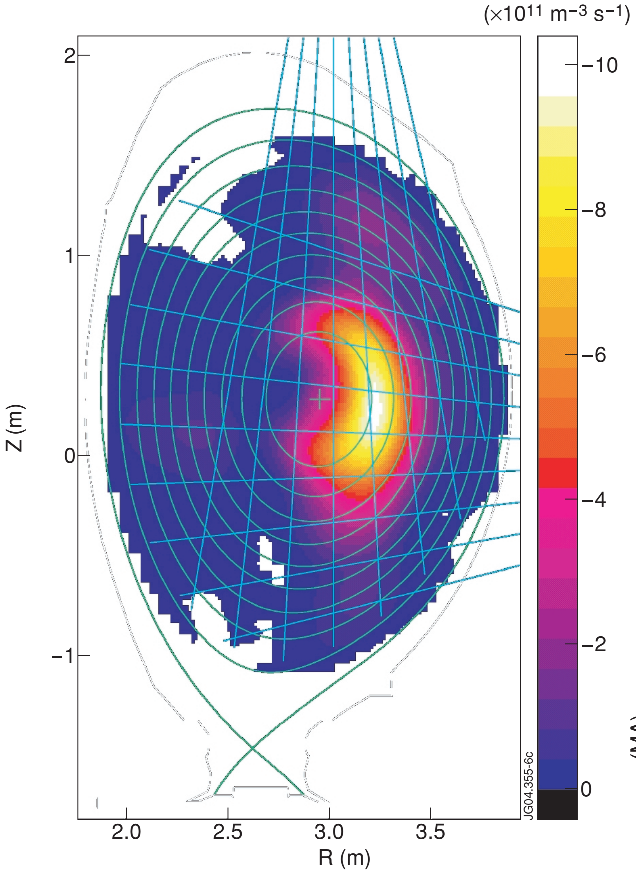

Figure 6: Tomographic reconstruction of g emissivity from $^{9}Be(a,ng)^{12}C$ reaction revealing a-particle localisation in a TTE discharge.

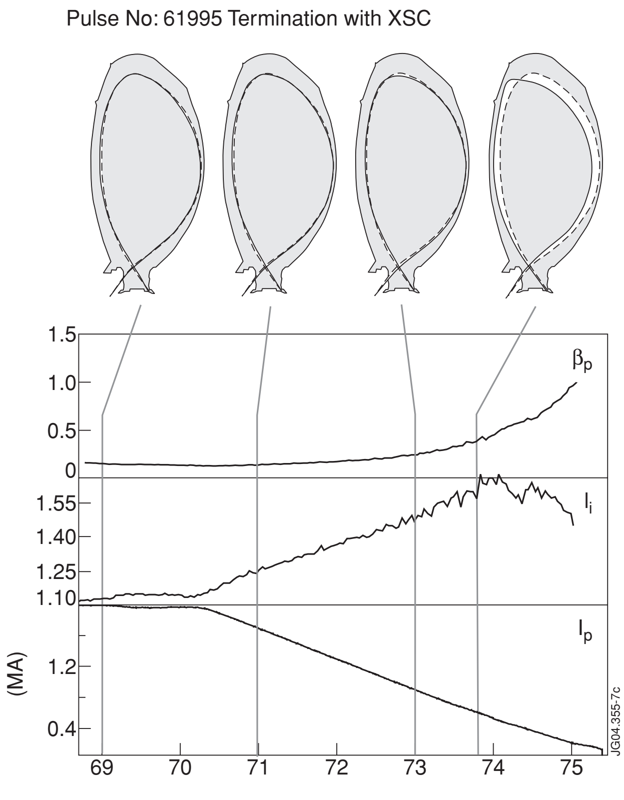

Figure 7: Demonstration of control of the plasma boundary during plasma current ramp-down, using the JET eXtreme Shape Controller.