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and the JET Team

Results of Recent Deuterium/ Tritium Experiments in JET

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Results of Recent Deuterium/ Tritium Experiments in JET

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ABSTRACT

During 1997, JET carried out a campaign of operation in deuterium/tritium. A total of 99 grams of tritium was admitted to the torus using gas puffing and neutral beam injection. With a site inventory of 20 grams of tritium, this required repeated re-processing of the gas recovered from the torus using the JET active gas handling plant. Around 220 tokamak pulses were carried out with tritium concentrations above 40%, during which a total of $2.5 \cdot 10^{20}$ 14 MeV neutrons were produced. Emphasis was placed on re-producing conditions close to those anticipated in the ITER experimental fusion reactor, in particular maintaining dimensionless parameters important in the physics of confinement. The experimental program included high fusion yield hot-ion and optimized shear scenarios in particular for the study of alpha particle physics. Achievements included a maximum fusion power of 16 MW in hot-ion H-mode at a Q of 0.6; first production of DT power (8 MW) in optimized shear; a Q of 0.2 for 5 seconds in an ITER relevant steady state ELMy H-mode at a fusion power of 4 MW; a Q of 0.22 in RF only discharges; and observation of alpha particle heating. Tritium was found to give a marked reduction in the H-mode threshold and an improvement in edge pedestal stability but no change in global confinement. The optimized shear scenario required re-optimization in tritium, only partially achieved. The results are generally consistent with ignition in ITER. Retention of tritium in the torus is much higher than anticipated and tritium recovery during the clean-up campaign was modest. The divertor tiles have since been replaced remotely with no personnel access to the torus. Tritium release and the dose to personnel have been well within the low approved levels.

JET has successfully completed this tritium campaign, producing both physics and technical data invaluable to the design of next step devices. The results in particular demonstrate the importance of operations in tritium in reliably predicting the performance of future machines.

I. INTRODUCTION

During 1997, JET has carried out the so-called DTE1 series of experiments in deuterium-tritium mixtures up to 90% tritium.¹ This experiment follows on from the preliminary tritium experiment (PTE) in 1991, in which just two pulses at 10% tritium fraction were carried out. The full tritium experiment (DTE2) for which JET has been designed will produce an order of magnitude more neutrons than DTE1.

The DTE1 experiment has been carried out with strict constraints on the number of neutrons to be produced. A limit of $2.5 \cdot 10^{20}$ was set in order to limit the activation of the machine and allow subsequent man access to the torus within 18 months. A site inventory limit of 20 grams of tritium was set. These limits, whilst constraining the extent of the experimental program, have allowed a wide range of experiments to be completed and much data critical to the performance of ITER to be obtained.

The key objectives of the program were threefold:

- to achieve high fusion performance in the hot-ion H-mode and optimized shear scenarios and observe alpha particle heating
- to obtain ITER relevant data in DT operations, notably confinement scaling and H-mode threshold scaling with tritium and the study of ICRH Physics in ELMy H-mode plasmas.
- to study technical aspects of DT operations, including operation of an industrial scale tritium processing facility, and remote handling technology in exchange of the divertor tiles subsequent to DTE1 operations.

These objectives have all been met with results which are presented below. The implications of this data for ITER, which are generally favorable, are discussed and important areas requiring further study are identified.

II. PREPARATIONS FOR THE TRITIUM EXPERIMENT

A. Regulatory background

The JET experiment has, since its inception, had operation in deuterium - tritium as a key objective. The initial approval of the project, planning permission for the site and technical design of the machine took this fully into account, with a design limit of 10^{24} neutrons being adopted. A maximum tritium inventory on site of 90 grams was set. The DTE1 experiment is, however, an interim campaign using DT which does not aim, by a wide margin, to achieve these original targets.

The limit to the number of 14 MeV neutrons to be produced during DTE1 was set at $2.5 \cdot 10^{20}$, including tritium clean-up operations. This limit is the maximum production consistent with subsequent manned entry into the torus within an 18 month period from the end of DTE1, allowing for some limited DD operation in this time. Neutron irradiation of the torus, which is 70% nickel, produces both cobalt 58 and cobalt 60 isotopes. The former has a half life of 70 days and dominates the in-vessel dose rate for a period of a year. There is however, a steady accumulation of Cobalt 60 with a 5 year half life which becomes increasingly important. The in-vessel dose rate after DTE1 has a cobalt 60 component of around 140 $\mu\text{Sv/hr}$.

This neutron limit, equivalent to about 1gigaJoule of fusion energy production, is two orders of magnitude greater than produced during the PTE experiment in 1991, and allows for substantial period of operation with optimized DT ratio, typically 60 seconds at full fusion power. Nonetheless, this budget was required to be strictly conserved during operations in order to allow for optimization of the various regimes, and sophisticated machine control systems were employed to this end. Such optimization was in practice not always achieved.

JET is obliged to apply the safety standards of the host organization. Approval for operation in DT was obtained from the Safety Directorate of the UK Atomic Energy Authority. This approval was based on the so-called Pre-Commissioning Safety Report, a substantial document

which included analyses of the various failures modes which may arise (loss of cooling, loss of vacuum, etc.), and which defined the essential safety requirements under which the operations could proceed. These requirements were enforced within the project by administrative and physical controls which included the appointment of two named individuals as holders of an ‘authority to operate’ for the torus as for the tritium plant respectively, with responsibility to the Director for the safety of operations. Several hundreds of documents covering local rules, commissioning procedures, operating procedures, alarm responses, emergency responses and torus operational limits embedded the safety case requirements in JET operations. Formal procedures were applied to the design approval, commissioning, and subsequent intervention in safety critical protection systems.

Whilst the site approval allows up to 90 grams of tritium on site, for the purposes of DTE1, the inventory was limited to 20 grams. This was sufficient to allow the planned program whilst ensuring that the worst case fault (release of the total inventory) would not lead to unacceptable consequences outside the site boundary.

The UK Environmental Agency regulates waste discharges to the environment. JET site limits during DTE1 are summarized in table 1.

Authorised waste discharge	Monthly TBq	Annual TBq
HTO	20	90
HT+ others	25	110
Aqueous	2	10
ATO authorised release	<200 GBq/day	
Site Management objective	40 GBq/day	

Table 1. Approved site discharge limits

Site management limits were set on daily release as tabulated. These limits are used to maintain best practice in the control of discharges to ensure compliance with the approved limits. The holders of the ‘Authority to Operate’ may authorize daily releases in excess of the management objective, up to the limit in the table. Higher daily releases require prior justification and notification of the Environmental Agency. Site releases are constantly monitored and alarms raised if unexpected increases are detected. At no time during the experiment were site limits exceeded. The highest monthly level recorded has been less than 10% of the limit.

B The Active Gas Handling Facility

JET have implemented an on-site tritium processing facility, the so-called active gas handling plant (AGHS). This facility has been described in detail elsewhere.² The AGHS receives the

tritium in uranium oxide beds, and extracts and supplies tritium to the torus as required. The AGHS also provides the backing pumps (normally liquid helium cryopumps but with mechanical pumps available if required) for the main torus and neutral beam injector turbo-molecular pumps. All gas from the torus is thus recovered into the AGHS, which uses cryo-distillation or gas chromatography to recover the tritium with a purity of typically 99.5%. This tritium can then be re-cycled to the torus.

The AGHS also provides an exhaust gas detritiation facility (the EDS). This has a throughput of $500 \text{ m}^3/\text{hr}$. Catalytic re-combiners are used to oxidize tritiated hydrogen gas to tritiated water. Molecular sieve dryers scrub this active water vapor with high efficiency, to achieve an overall reduction in tritium content by a factor 1500. This facility is a key feature of the safety case as it maintains torus underpressure during loss of vacuum (LOVA) events. In addition, many diagnostic backing lines are pumped by the EDS.

C. Tritium Supply to the Torus

Tritium gas could be admitted to the torus by puffing gas on the mid-plane or in the divertor. In addition, the octant 8 neutral beam system was modified to operate on tritium.³ The energy of the injected tritons was 160 kV, and the system allowed combined operation with deuterium or tritium on different pairs of ion sources within the same injector. This allowed precise tritium fraction scaling experiments. The neutral beam injectors also incorporate large cryopumps, and most of the tritium supplied by the AGHS was accumulated on these cryopumps, the inventory of which could also reach several grams. The octant 4 injector, which operated at 80 kV in deuterium only, also accumulated small quantities of tritium by pumping the torus.

The divertor in use during DTE1 was the so-called Mk IIAP version, with CFC tiles. This incorporated liquid helium in-vessel cryopumps of about $100 \text{ m}^3/\text{s}$ pumping speed. Most of the tritium admitted to the torus was accumulated on these cryopumps, the inventory of which could reach several grams. The only beryllium in the torus was the screens of the ICRF antennae.

JET policy was to re-generate the cryopumps each night during tritium operation to limit the in-vessel inventory. During tritium operations, the torus hall was maintained under a depression of a few mbar. The oxygen concentration was reduced below 15% by addition of nitrogen gas as a fire suppressing measure. This also prevented man access to the torus hall, except in emergency wearing breathing apparatus, and contributed strongly to the reliability of the system.

III. PHYSICS RESULTS DURING TRITIUM OPERATIONS

A. High Fusion Performance

The highest fusion yields in DD operation at JET have been obtained with the hot-ion H-mode and the optimized shear scenario, with strong NB heating. In preparation for the DT experiment,

the toroidal field coils were assessed in close detail and confirmed to be able to support operation at 3.8 Tesla, compared to the previous 3.4 Tesla limit.⁴ The hot-ion scenario is limited by coil stresses at the high plasma current required to 3.6T. Operation in DTE1 at this TF field has given a record fusion yield of 16.1 MW fusion power.⁵ The main parameters of this pulse are shown in fig 1.

The fusion yield is some 30% higher than projected from the 3.4T DD rates, an improvement which is in part attributed to the higher NB power, and in part to the enhanced TF field. The uncorrected Q (power out/power in) is 0.6, which increases to 0.9 if adjusted for the rate of change of stored energy. The pulse is terminated and the fusion power limited by a giant ELM, which occurs a little earlier than in the reference 3.4 Tesla DD pulse.

B. Alpha particle heating

Even at the highest fusion power achieved in DTE1, the alpha particle heating is only 3 MW compared to a total heating power of 25 MW. Unambiguous observation of alpha particle heating has required a scan of similar pulses with tritium fraction varying from 0 to 90%. A highly reproducible hot-ion H-mode scenario with 10 MW of NB heating was chosen for this experiment.⁶ The variation in stored energy and energy confinement time with tritium fraction is shown in fig 2.

The energy confinement time shows little or no isotopic dependence, while there is a clear increase in stored energy at mid range consistent with the increased alpha particle power in this region. In fig 3, the observed electron temperature is plotted against the power in the alpha particles.

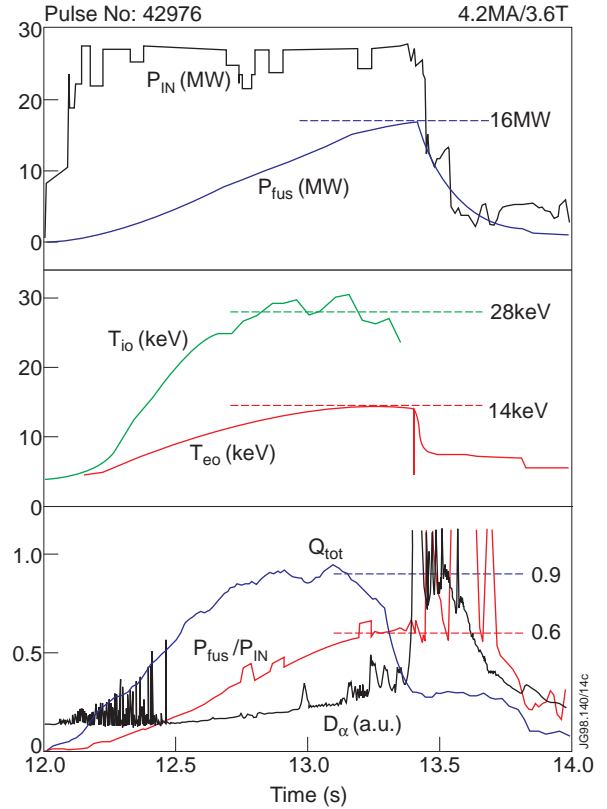


Fig. 1 The Hot-ion H-mode pulse giving the record fusion yield of 16.1 MW

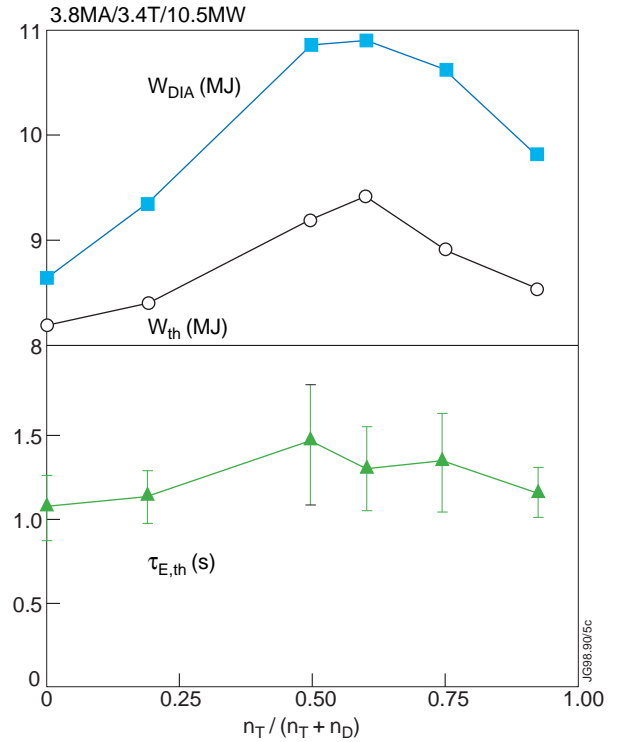


Fig. 2 Variation of stored energy and energy confinement time with tritium fraction for similar pulses.

It is noted that the electron temperature is proportional to the alpha power. The slope of the curve is the same as obtained with central ICRF electron heating indicating similar efficiency.

C. Optimized shear scenario

The optimized shear scenario has been extensively developed at JET in DD operation at 3.4 T, and has produced the record JET DD yield. This scenario relies on precise control of the current ramp in order to achieve favorable current and q profiles. LHCD is used in the early stages, and ICRF and neutral beam later in the ramp, as tools to optimize the profile. Under optimum conditions, internal transport barriers are formed at typically mid radius which lead to a substantial increase in energy confinement time; ‘H-factors’ of nearly 2 are achieved whilst still in L-mode. The transition to H-mode is delayed as long as possible, but when it occurs an edge pedestal and further increase in confinement is obtained, followed however by a large ELM which terminates the high performance phase. The current ramp is a delicate balance between MHD instabilities and early H-mode transition. Since the latter is mass dependent, it turned out to be necessary to re-optimize the scenario in DT. Constraints on the neutron budget did not allow this to be completed, nonetheless, a fusion yield of 8.3 MW was obtained.⁷ Internal transport barriers were successfully formed in DT plasmas with similar power threshold to DD plasmas. Fig 4 shows the variation in thermal diffusivity with radius obtained using the TRANSP code⁷.

The diffusivity clearly falls dramatically inside the transport barrier at r/a of 0.6. Projections of performance potentially to be achieved in this scenario using existing JET heating capability indicate that there is much yet to be achieved, which is likely to be one of the key objectives of the coming DD and future DT campaigns.

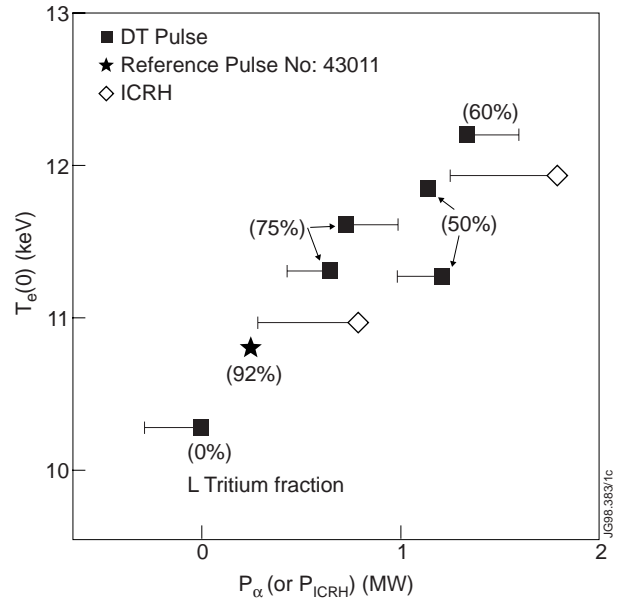


Fig. 3 Variation of electron temperature with power appearing in alpha particles. The bars indicate corrections arising from differences in NB heating power between pulses.

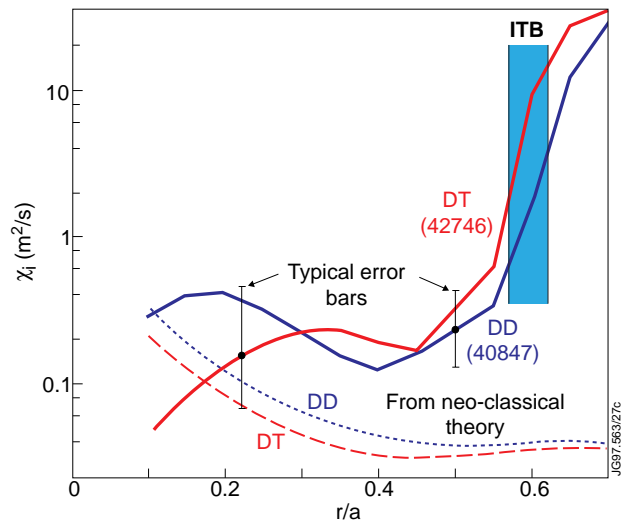


Fig. 4 Thermal diffusivity obtained from the TRANSP code showing sharp decrease inside the transport barrier at r/a=0.6.

D. 'Steady state' ELMy H-modes

The ELMy H-mode scenario reaches near steady state performance which is characteristic of the operational regime anticipated in ITER. Consequently, intensive study of ELMy H-Modes has been carried out during DTE1. A record fusion energy of 22 MJ at $Q=0.2$ has been obtained in this scenario⁸, as shown in fig 5.

All ICRF heating scenarios for ITER have been validated in ELMy H-modes⁸. Strong central ion heating has been observed in both D minority and He³ minority experiments. Second harmonic tritium heating in JET produces high energy triton tails, which damp on the electrons; in ITER, the reduced power density will lead to direct ion heating⁹. A steady Q of 0.22 has been produced with 6MW of RF only heating⁹.

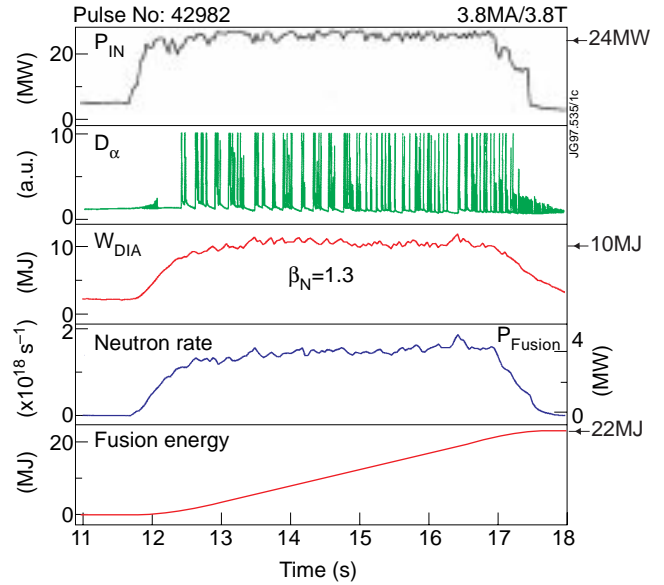


Fig. 5 Record fusion energy of 22 MJ achieved in 4.5 sec steady state ELMy H-mode at $Q=0.2$.

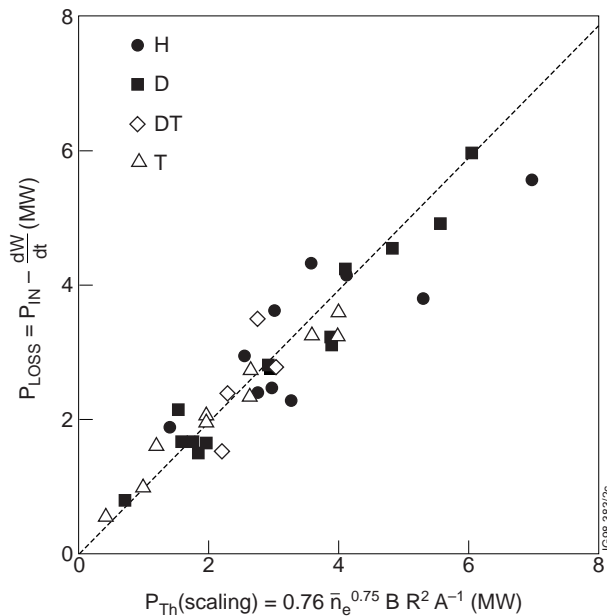


Fig. 6 Threshold power for L to H mode transition.

The threshold power required to induce a transition from L to H-mode in DT operation has been measured and is shown in fig 6, together with data from D and H experiments.

It is a clear that the scaling with isotopic mass (A) as A^{-1} well represents the data set. This scaling in threshold power is favorable to ITER, reducing the additional heating power requirement during start-up in pure T plasma by 30%. There appears to be no hysteresis in the H to L reverse transition.

The stability of the edge pedestal has also been found to improve with isotopic mass for NB heated plasmas, but not with ICRF heated plasmas. These results are consistent with a model in which the width of the pedestal is related to the local ion gyroradius. This is beneficial for confinement but may results in larger ELMs, which can lead to power handling

problems on first wall components. Localized central heating may combine good confinement with a low edge pedestal and weak ELMs. The modest ELM's observed with ICRH heating suggests that alpha heating will also give small ELM's.

The measured global energy confinement time in DT plasmas is consistent with EPS97 scaling as $A^{0.2}$, as shown in fig 7, but a scaling independent of isotopic mass is also a good fit to the experimental data.

On detailed analysis, the experimental data is consistent with a difference in confinement scaling between the core and edge plasmas. The former varies about as $A^{-0.2}$ as expected from gyro-Bohm scaling, whilst the latter varies as $A^{0.5}$. The two effects compensate to a first approximation to give the observed weak overall dependence on A.

The EPS97 scaling is consistent with ignition in ITER and is supported by the JET results. A series of JET pulses have been run varying the so-called ρ^* parameter (minor radius over central ion gyroradius). Extrapolation of two such pulses to ITER have been carried out assuming both gyro-Bohm and Bohm scaling⁸. The results are shown in table 2.

The high q scenario predicts full ITER fusion power, but requires operation at a density a factor 1.5 above the Greenwald limit. The reduced q scenario ignites at the Greenwald limit with gyro-Bohm scaling (which is supported by the JET data), but not Bohm.

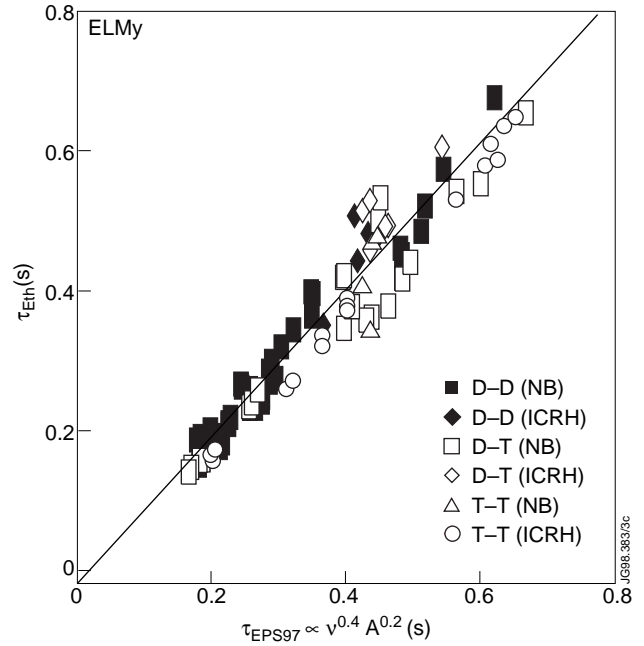


Fig. 7 Comparison of measured energy confinement time with EPS 97 scaling, with $A^{0.2}$ mass dependence.

Scenario	gyro-Bohm	Bohm
q=3.4, n=1.5n _{GW}	ignition , 1.5 GW	Q=5
q=2.7, n= n _{GW}	ignition, 1.2 GW	Q=10

Table 2. Projected performance of ITER in two scenarios extrapolated from JET DT pulses; n_{gw} is the Greenwald density limit.

IV. TECHNICAL RESULTS FROM THE TRITIUM EXPERIMENT

A. Tritium handling

During DTE1, a total of 99.3 grams has been supplied by the AGHS, with more than 220 pulses being run above 40% tritium. By the end of the campaign, some 11 grams was retained in the torus after regeneration of the cryopumps. A typical operating cycle comprised four days of tokamak operation, consuming 2-3 grams per day, followed by four days of re-processing. A total of 8 re-processing cycles were carried out by AGHS. This primarily utilized the gas chromatography system to produce tritium with 99.5% isotopic purity. The cryo-distillation system has a high residual inventory and was not appropriate to operation with limited site inventory, but was used to concentrate the tritium to typically 30%.

Around 65 grams of the total was supplied to the octant 8 neutral beam system, nearly all of which was trapped on the cryopumps and returned to the AGHS. Around 34.5 grams were supplied directly to the torus via the gas inlet system. Of this, only about 23 grams was recovered to the AGHS, the balance of 11 grams being retained in the torus at the end of DT operation.

The AGHS operated with an excellent reliability and availability throughout the campaign.

B. Tritium Retention

The PTE experiment¹⁰ yielded data on both tritium fraction and tritium inventory during and after the experiment. The tritium fraction during DTE1 was measured by various methods. Diagnostics within the divertor gave the tritium fraction in the plasma during the pulse. Diagnostics on the pumping line in the AGHS yielded the tritium fraction in the recovered gas. These measurements may be expected to give systematically different results depending, for example, on the gas loading of the first wall. The results of these measurements are shown in fig 8. Also shown in fig 8 is the tritium fraction predicted on the basis of the tritium inlet using the PTE data. There is good agreement with observation.¹¹

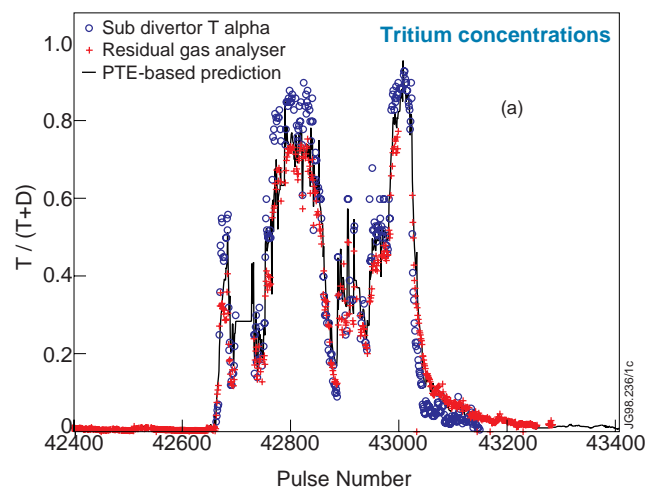


Fig. 8 Variation of the tritium fraction during DTE1 compared to predicted fraction using PTE data.

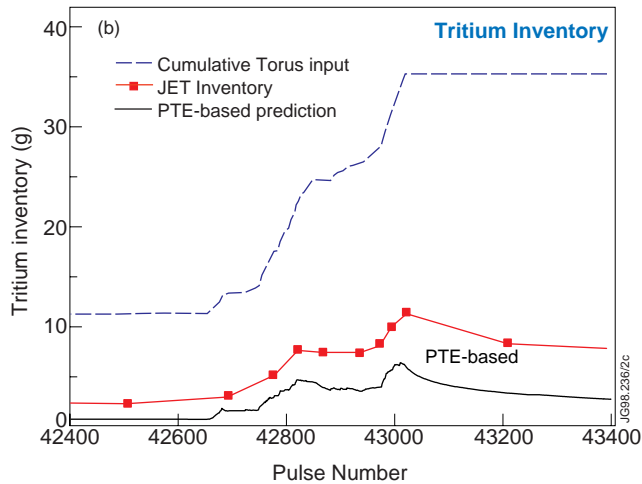


Fig. 9 Total in-vessel inventory of tritium compared to the tritium supply to the torus and predictions using PTE1 data.

The accumulation of tritium in the torus during the campaign is shown in fig 9, together with total tritium supplied to the torus and the inventory to be expected using the PTE data. It is apparent that the retained tritium is a factor of three higher than expected. Co-deposition of tritium with carbon sputtered from the target plates has been identified as the main process contributing to this effect. Carbon films removed from the torus during DD operation have been analyzed and shown to have a deuterium-to-carbon atomic fraction varying from 0.25 to 0.1 as the temperature of the film during deposition increases from 20 to 300 C.

Around 5 grams of such material has been removed from the torus during the present shutdown. Whilst detailed analysis of the tritium content has not yet been done, indications are that this contains c.1 gram of tritium.

The tritium retention in ITER can be estimated from the yield of sputtered carbon from the divertor plates.¹¹ Assuming an entrapment of 1.7% of the tritium entering the divertor, as observed in JET, and scaling to ITER size, a total of 13 grams of tritium is retained after each full performance pulse. The allowable in-vessel inventory of 1 kg would be achieved after less than 100 pulses. Detailed modifications to the ITER divertor design to eliminate carbon deposition on cold surfaces are being assessed as a result of this data.

C. Tritium Clean-up

Various methods of recovery of the tritium retained in the torus were investigated. Tokamak pulsing in deuterium initially recovered meaningful quantities and over a period of 8 weeks, halved the tritium inventory. However, by the end of this time, recovery was down to 10 mg per day. Glow discharge cleaning in deuterium could only be tested for a short period as the gas throughput exceeded the AGHS capacity, but the effect appears to be small. Heating of in-vessel components had little effect, as did soaking in deuterium at 0.1 mbar, 320C. The benefits of this apparently observed during PTE are now attributed to contamination of the tritium monitor. Soaking in 1 bar nitrogen at 150°C recovered negligible tritium. Soaking in humid air (c. 7°C dew point) at 150°C recovered significant tritium, but was limited in time due to deleterious effects on other in-vessel components. Continuous flushing in room air for four months at 20C during the shutdown has recovered c.10 mg per week.

D. Tritium management

An accounting for the original 20 gram site tritium inventory is given in Table 3.

Of the original 20 grams, all but 0.02 grams remains on site. Gaseous discharge accounts for the bulk of the remaining 0.02 grams. The exhaust de-tritiation system (EDS) has generated around 12 m³ of tritiated water which is presently held on site. A total of 13.6 grams remains available for torus operation.

During the DTE1 experiment and the post DTE1 shutdown, there have been no reportable releases of tritium or exposure of personnel. The monthly discharges have never exceeded 10% of the authorized levels, and with one exception, have been below 5%.

	grams
In AGHS	13.6
Tritiated water from EDS (12m ³ at 60 TBq/m ³)	2
Gaseous Discharge (HT+HTO)	0.02
Aqueous Discharge (10000 m ³ at 0.02 GBq/m ³)	0.0005
Removed from torus:	
c. 5g carbon flakes	c. 1
c. 1 tonne tile assemblies	c. 1
Total	c. 18
Site inventory	20
Balance in torus	2±1

Table 3. Accounting for the site 20 gram inventory

Pending a detailed assay of the tiles and flakes removed from the torus, it is estimated that 1-3 grams will remain in the torus at the start of the next campaign in June 1998.

E. Torus activation

The measured in-vessel dose rate 3 months after DTE1 was 4.2 mSv/hr. This is a little higher than anticipated, taking account of the extended period of neutron production. The margin of error is within the tolerances of the original estimates, taken together the tolerances in measurement of neutron flux and in-vessel activation. The underlying cobalt 60 will contribute c.140 µSv/hr to the in-vessel dose 18 months after DTE1 when manned access to the torus will again be feasible, depending on the intervening DD neutron production.

F. Remote handling

During the present 4 month shutdown, the divertor tiles have been removed and the new gas box divertor tiles installed, entirely by remote handling. This work is described in detail elsewhere at this conference.¹² The work has now been completed on schedule and fulfills one of the key JET objectives to implement effective remote handling technology. This is an important demonstration of the feasibility of carrying out complex in-vessel interventions in a Tokamak after substantial DT operation, which is critical to the success of ITER. The cumulative site dose during 1998, which covers the last month of torus operation and the shutdown, is less than 0.008 man.Sv. This is a remarkably low figure and indicates what can be achieved notwithstanding tritium operations given appropriate preparation.

V. CONCLUSIONS

The JET tritium experiment has been successfully completed, much of the in-vessel tritium has been recovered, and the divertor tiles have been subsequently replaced with the gas box divertor tiles using full remote handling techniques.

JET has achieved 16 MW fusion power at an uncorrected Q of 0.6 in the hot-ion H-mode, and 5 MW for four seconds at a Q of 0.2 in 'steady-state' ELMy H-mode.

Key results for ITER include the following:

- the H-mode threshold has a strong favorable mass dependence
- confinement in ELMy H modes has little or no mass dependence, but is nonetheless consistent with ignition of ITER
- the performance of the optimized shear scenario is sensitive to isotopic mass and requires careful optimization in tritium to achieve the full potential; internal transport barriers were clearly established in DT plasmas.
- alpha particle heating has been observed and is equivalent to ICRF electron heating of the same power
- direct ICRF ion heating has been clearly demonstrated using both D and He³ minorities in DT plasmas, and a Q of 0.2 obtained in ICRF only discharges; second harmonic tritium electron heating has also been demonstrated

Important technical objectives have also been met:

- the tritium processing plant has operated reliably and effectively
- tritium retention has been quantified and identified as an important factor for ITER
- the technology of tritium accounting, and control of waste discharge and personnel exposures have been demonstrated to work effectively.
- remote handling technology has been demonstrated in the post DTE1 tile exchange

The DTE1 campaign has fully met the objectives set, has produced unique results which are invaluable for the design of ITER, has demonstrated the importance of experiments in tritium, and shown that Tokomaks can combine meaningful tritium operation with major in-vessel interventions.

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