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** See Annex*

VACUUM PROBLEMS AND SOLUTIONS RELATED TO THE TRITIUM EXPERIMENT AT JET

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ABSTRACT

A mixture of deuterium and tritium as fuel was introduced for the first time into a tokamak during the Preliminary Tritium Experiment, carried out successfully at JET in November 1991. The main aims were to produce in excess of 1 MW of fusion power, to validate plasma codes under these conditions, to determine tritium retention in plasma facing components, to establish procedures for tritium removal from components in contact with the plasma and to demonstrate the technology related to the safe usage of tritium. The experiments were undertaken within limits imposed by restrictions on vessel activation and tritium usage resulting in a tritium inventory of 0.2 g and a maximum number of neutrons of 1.5×10^{18}. The paper describes the preparations for this experiment with respect to modifications to the vacuum and gas handling system and gives details of materials facing the plasma and conditioning techniques employed. A summary of experimental results is presented together with estimates of the tritium retention in the walls.

INTRODUCTION

The aims of JET were already defined as early as 1975 [1], and one of the main objectives is to study α-particle production, confinement and subsequent heating of plasmas in conditions and dimensions approaching those needed in a thermonuclear rector. Such an investigation can only start once operational scenarios are established which allow JET to generate sufficient fusion power to undertake meaningful exploration of the experiments. As a prerequisite the reliability of all machine components has to be maintained at such a level that tritium can be safely used as fuel without creating safety problems. Tokamak operation with a one to one ratio of deuterium to tritium (DT) is planned for the active phase of JET which is scheduled to take place during 1995/1996. The Active Gas Handling System [2], which is presently under commissioning, will then be available. Its main tasks are to prepare the gas mixtures to be used during tokamak operation, to clean and to recirculate the exhaust gases from the torus, neutral and pellet injection and to perform isotope separation.

At the end of 1991 plasma performance was at such a level that in the best cases neutron yields were obtained in deuterium plasmas, which showed that for discharges using DT mixtures breakeven would have been achieved, i.e. the heating power would have been balanced by the α-particle and neutron power. Secondly the machine components proved to be reliable enough to predict operation over some months without any major failure and thirdly the licensing procedures for having up to 90 g of tritium on site were well advanced and much

of the safety preparation for a preliminary tritium experiment could be based on this [3].

With all the main preconditions fulfilled, it was decided to perform a tritium experiment with about 10% of tritium in the fuel with the following objectives:

-to produce in a controlled way fusion power in excess of 1 MW
-to validate plasma codes and to provide a basis for predicting the performance of D-T plasmas
-to determine tritium retention in the vacuum vessel, in the neutral beam heating system and the vacuum system
-to establish procedures for tritium removal
-to demonstrate the technology related to tritium usage
-to validate procedures for handling tritium in compliance with regulatory requirements

In order to perform a DT-experiment at this stage in the JET programme it was necessary to limit the total neutron production to about 1.5×10^{18} neutrons so that the resulting vessel activation would be compatible with the necessity to carry out extensive work inside the vessel three months after the experiment. At this time the activation was to be below 100 $\mu Sv h^{-1}$. In addition, the total amount of tritium available was restricted to ~0.2 g as the JET tritium reprocessing plant is not scheduled to be operational before 1993. Taken together, these limitations restricted the number of high performance discharges to a few. As a consequence no optimization of plasma performance could be carried out.

The preparation of this Preliminary Tritium Experiment included a number of discharges using a mixture of 1% tritium (tracer quantities) in deuterium in lieu of pure deuterium These trials mainly had the purpose to act as calibration facility for the various neutron diagnostics that had to operate with 14 MeV neutrons instead of 2.5 MeV ones from the deuterium-deuterium reactions. The whole experiment took place within three weeks. After that time operation without tritium resumed for three months.

TRITIUM INTRODUCTION

Tritium was introduced into the torus by using neutral beam injection. The utilisation of a neutral beam system is an effective way of bringing tritium into the type of discharge selected for the DT experiment by ensuring that it reaches the hot, dense centre of the discharge. Tritium was supplied from an uranium storage bed and buffer reservoir through a pressure regulator and needle valve and introduced into the neutralisers of two of the sixteen Positive Ion Neutral Injection sources (PINI 5 and PINI 6) as shown in Fig 1a and 1b. A second uranium bed loaded with deuterium was also available for commissioning discharges before the experiment and decontamination shots afterwards.

The tritium gas introduction system was enclosed in a secondary ventilated containment system (glove box) and installed inside the torus hall close to the neutral injector.The ventilation is exhausted via a monitored stack

For the experiment with the one percent tritium one uranium bed was loaded with 0.019 barl (50 Ci) of tritium and 2.08 barl of deuterium. Although the gas mixture loaded into the bed is well known, there is however an uncertainty in the DT ratio evolving from the bed. For the full experiment, an uranium bed was loaded with 0.9 barl (2400 Ci) of tritium. The uranium bed was heated to a temperature of typically 450 C to deliver gas at a pressure of 0.9 bar and the gas was directly sent to the neutral injector.

The gas is injected (see Fig. 1b) at ground potential between the last grid of the neutral injector and the neutralizer tube. It should be noted that since each neutral injector has its own neutralizer tube, it has been possible to operate two of them with tritium and the remaining six in the same injector box with deuterium. A detailed description of the neutral beam system is given in [4].

For the full experiment only about 6% of the tritium taken from the storage bed was injected into the plasma as energetic ions, the remaining 94% were required to operate the injectors and they were collected on the cryopanels and ion beam dumps in the neutral injection vacuum system. The neutrals were injected at an equivalent atomic current of 12 A at a power of 0.75 MW for each PINI. The tritium gas required for one pulse of one second duration was 22.5 mbarl per PINI (1 g tritium ~ 10^4 Ci ~$2x10^{23}$ atoms ~ $3.7x10^{14}$ Bq ~ 3730 mbarl STP). The tritium was injected through two PINIS the remaining fourteen neutral injectors were operated in deuterium. The tritium fuelling rate relative to the total was ~13%. Including operation at one percent of tritium and conditioning pulses a total of about 1003 ± 73 Ci was required, 54 ± 6 of which were actually injected into the torus.

VACUUM SYSTEM AND TRITIUM RECOVERY

The 200 m^3 volume of the torus is evacuated by four turbomolecular pumps providing a pumping speed for hydrogen of 9 $m^3 s^{-1}$. In addition the cryopumps of the two neutral injector boxes are connected to the torus through ports which limit the available additional pumping speed to another 5 $m^3 s^{-1}$. The torus turbopumps and the neutral injector boxes are connected by large diameter backing lines to roughing pump sets exhausting to the atmosphere. A detailed description of the JET pumping system is given in [5].

For the Preliminary Tritium Experiment the Active Gas Handling System was not yet available therefore the vacuum system had to be modified in such a way as to contain the exhaust gases in a closed volume To handle tritium the backing pumps were substituted by a closed system comprising a cryopump and uranium getter beds [6] as shown schematically in Fig. 1c. This system was enclosed in a plastic tent and the ventilation air is exhausted through a monitored stack.

Just prior to each discharge the backing line was isolated from the cryopump so that the exhaust evolving after the discharge would accumulate downstream of the turbopumps, using the backing lines as a the buffer volume. This volume has a size of 6 m^3. During this time the exhaust activity could be measured on line using an ionisation chamber and sampled by a removable sample bottle connected to the backing line. The activity of the gas bottled samples was measured on a separate system. Before the next discharge the accumulated exhaust was pumped onto the cryopump. About 2 times each day the cryopump was regenerated into a reservoir of 0.345 m^3 volume at which time the activity of the gas was measured by another ionisation chamber connected to this reservoir. After expansion into the collection tank the gases are circulated on two uranium beds for the absorption of hydrogen isotopes. The total capacity of the two uranium beds is 900 barl which is the gas throughput for 3-4 weeks of JET operation. Gas not collected on the uranium beds is exhausted through a monitored stack.

This mode of operation had to be continued for a few weeks after the tritium experiment up to a time when the release was below
5.9×10^{18} atoms per day which corresponds to the permitted daily release of 10 GBq [3]. Once values below this limit were obtained,
the exhaust from the torus was routed through a monitored stack to the atmosphere.

TRITIUM MEASUREMENTS AND MONITORING

The amount of gas injected into the neutral beam system was measured using a small reservoir which was filled with tritium before the shot. The pressure drop in this reservoir gives the amount of tritium used. The error of this measurement is about 10% including systematic errors due to uncertainties of internal volumes of uranium beds and isotope mixture supplied.

The tritium concentration in the plasma was determined by applying a short burst of deuterium neutral beams and measuring 14 MeV and 2.5 MeV neutron rates. The ratio of DT to DD neutrons gives the D:T ratio in the plasma

The amount of tritium being pumped out of the torus was evaluated using a mass spectrometer located at the high vacuum side of one of the turbopumps. Mass numbers 2, 3, 4 and 5 as well as the total pressure were monitored for 1000 s after the start of the discharge.

Tritium exhausted from the torus or the neutral beam system was measured by allowing the exhaust gases to accumulate for 560 s in the backing line with the valve to the cryopump closed. Measurements of the activity were made after each discharge using the ionisation chamber in the backing line. At the same time it was also possible to sample the exhaust and analyse the gas sample on a remote system [7].

Tritium discharged through the stack was monitored based on a flow through an ionisation chamber to measure the total tritium concentration. The instrument

takes a fraction of the air flowing through the stack and from the measured flow rate derives the integrated discharge.

Tritium retained in the vacuum vessel was estimated by analysing a collector probe present in the JET vessel throughout the entire time that the closed exhaust system was used. The probe simulated the areas in contact with the plasma and the wall. Four probe samples were removed 250 discharges after the tritium experiments and analysed. In addition, after the opening of the vacuum vessel three months later, wall samples were taken for analysis of the tritium content.

FIRST WALL MATERIALS AND VESSEL CONDITIONING

The interior of the JET vacuum vessel is shown in Fig 2. It consists of a continuous top X-point target plate which is clad with carbon fibre composite and a continuous bottom X-point target plates, clad with beryllium tiles and graphite tiles, the beryllium tiles intercepting the plasma footprint. The pair of toroidal belt limiters above and below the mid-plane carry different materials facing the plasma. The upper comprises of beryllium and the lower of carbon. All other plasma contacting surfaces such as the inner walls, Rf Antennae, protection limiters are of carbon fibre composites, fine grain graphite or beryllium. The two target plates were carefully aligned to avoid toroidal steps and in addition individual tiles were carefully shaped to minimise the effect of residual protrusions and steps which were about 1 mm.

The plasma contacting surfaces were extensively conditioned by a combination of glow discharge cleaning in deuterium and helium and tokamak discharge operation. They were coated at regular intervals with thin (50 - 300Å) beryllium layers by periodically evaporating beryllium inside the vacuum vessel [8]. A fresh layer was deposited about 12 hours before the DT experiment.

DISCHARGE TYPE AND MAGNETIC CONFIGURATION

A range of possible JET discharge types was in principle suitable for the aims of the DT experiment. However, pellet fuelling was excluded since the present pellet injector was not designed to operate with DT plasmas. Attention concentrated on discharges heated by neutral beam injection only. Of these, the highest neutral yield obtained in the hot ion mode was a discharge at low density with the ion temperature significantly higher than electron temperature. These discharges made in the H-mode regime (ie. an X-point discharge with improved confinement above an input power threshold).

Ultimately, a single null X-point discharge was chosen. In this configuration, shown in Fig. 3, ions drift away from the target towards the plasma. This had been found to lead to more equal power loading between the inner and outer branches of the X-point. Overall this configuration allows consistently higher energy input and longer duration of the high performance phase of the discharge before the introduction of impurities.

EXPERIMENTAL RESULTS

A detailed description of the experimental results is published elsewhere [9]. Here only the results from two discharges will be summarized. They are DT discharges with 100% tritium introduced into 2 neutral injectors. Only two pulses (Nos. 26147 and 26148) of this type were attempted. Both were similar and each produced fusion power in excess of 1.5 MW.

Fig. 4 shows the time development of some parameters for pulse 26148. The plasma current started to rise at the time t = 0. It was maintained at a flat top in excess of 3 MA from 5 s to 15 s and then decreased towards zero, which was reached near 25 s. All but Z_{eff} (a measure for impurity content) increased throughout the H-mode phase of the discharge which started at 12.4 s and ended with a carbon bloom at 13.3 s. At the same time, in spite of still being in the H-mode, the good plasma performance was terminated: the neutron output decreases drastically due to reduction of the ion temperature and increased dilution by graphite. The carbon bloom was caused by hot spots on the graphite target plates. The neutral beam power versus time, as shown in the figure, was chosen to maximize the neutron output as derived from code calculations and in fact the measured values are in good agreement with the computed ones.

From the discharge parameters it is for the first time possible to measure directly the fusion amplification factor (ratio of fusion power to power loss from the plasma). Previously it had to be calculated by extrapolation from DT discharges. Direct measurement for discharge 26148 gives a value of 0.15. A similar discharge with an optimized mixture of tritium-deuterium would have a Q_{DT} of 0.46. This is well below the best value obtained for the pulse 26087 which would give an extrapolated Q_{DT} of 1.14. The main reason is that with the limitations in neutron production it was not possible to optimize the discharges.

For the main tritium experiment foreseen in 1996, there will be 12.5 MW of tritium and 8 MW of deuterium neutral beam injection both at higher power per particle than used so far. The better beam penetration should give higher values of Q_{DT}. It should also be possible to couple up to 20 MW of ICRH to the plasma either alone or in combination with neutral beam heating, in which case the total fusion power should also increase. Experiments with the pumped divertor(expected to start end of 1993) are expected to control impurities and give a cleaner plasma which should lead to a further increase of the fusion amplification factor. In addition it will be possible, in a long campaign, to optimize the performance of the DT discharges to come close to that of number 26087.

CLEAN UP PROCEDURE

Approximately 40 hours after the last tritium fuelled discharge a planned sequence of clean up discharges started [10]. The first eight identical ohmic heated discharges were aimed at revealing the shot to shot evolution of the tritium release. The

same magnetic configuration as for the tritium fuelled discharges was adopted. This was followed by a discharges of alternative configuration (shot No. 26159) with the plasma successively touching the upper belt limiter, the inner wall and the lower belt limiter to probe other surfaces for tritium. To compare the cleaning efficiency of helium versus deuterium plasmas, ^{3}He was used to fuel a series of five discharges. After a brief resumption of the normal experimental programme more time was devoted to detritiation of the vessel because the tritium release level was still above the target value of 6×10^{18} tritons per day which meant that the torus exhaust still could not be routed to the atmosphere. A sequence of 24 discharges in deuterium was made, each ending in a planned high density disruptions, followed by a few hours soak of the torus in deuterium at ~2 Pa. At this stage the regular experimental programme was once again resumed. About two weeks after the initial main introduction of tritium, the level of tritium evolving from the torus was sufficiently low so that the torus exhaust was routed to the atmosphere once again. At this stage glow discharge cleaning of the torus was performed. Glow discharge cleaning was not attempted earlier because the closed exhaust system could not handle the high gas load.

TORUS CLEAN UP

Integrated tritium releases for a 560 s period following the start of the plasma is shown in Fig. 5 as a function of the shot number. The two highest values correspond to the two discharges with 100% of tritium at two neutral injectors. During the subsequent clean up the amount of tritium released decayed roughly exponentially with the shot number for the first ten discharges.

The total amount of gas recovered in a period of 560 s following the tritium fuelled discharges is about 50% of the total gas input. This is in agreement with previous gas accounting measurements [11]. The tritium recovery, however, was only about 12%. This agrees qualitatively with release of tritium atoms after DD discharges and indicates that atoms from the central plasma get trapped in the walls for some time. After the discharge with shot number 26159 about 2 times as much tritium was set free than after the preceding discharge. This may be due to the pulse termination by an unplanned disruption.

The switch from deuterium plasmas to ^{3}He plasmas resulted in a 2.5 fold reduction in the amount of tritium recovered. This suggests that surface recombination of D and T atoms and subsequent release as DT molecules is the dominant mechanism at the end of a discharge. There was, however, still a substantial amount of deuterium in the plasma because of fuelling with a deuterium helium mixture and due to outgassing of deuterium implanted during previous discharges. While about 3.5×10^{22} deuterium atoms where needed to fuel a deuterium clean up discharge only about 8×10^{21} D and 2.5×10^{21} ^{3}He atoms were required to fuel the ^{3}He discharges for identical electron density. This reflects the smaller capacity of the walls for pumping ^{3}He. For the same reason the plasma content is about 50% ^{3}He despite the larger contribution of deuterium to the fuelling.

Additional heating by deuterium neutral beams made very little difference to the quantity of tritium recovered. This indicates that the release process is largely independent of the energy of the particles hitting the wall. Therefore it appears that particle induced desorption plays a minor role compared to surface recombination. This is as well in agreement with the observations during the ^{3}He fuelled discharges.

Deuterium soaking of the torus resulted in the removal of more tritium than obtained after immediately preceding discharges. However, if the quantity of recovered tritium is normalised to the amount of deuterium used, the D:T ratio of the releases was nearly the same as for the tokamak discharges.

A twenty minute period of glow discharge cleaning in deuterium took about 5×10^{17} tritium atoms out of the vessel. That is about the same as for four tokamak discharge at this time. However, the amount of deuterium required for the twenty minutes cleaning is equivalent to 100 tokamak discharges indicating that the T:D ratio is very small compared to tokamak operation.

Figure 6 shows the release of tritium for a longer time showing the sample bottle results. Only this technique has sufficient sensitivity to measure the tritium exhaust before and long after the tritium experiment. The data before correspond to the quantity of tritium atoms released during a campaign of five high fusion yield DD discharges. This was followed by the first introduction of tritium in JET with tracer quantities of tritium (first step in Fig. 6) and then by the two discharges at about 10% concentration (peak in Fig. 6). After two months and more than 1000 tokamak discharges, the tritium release from the torus is almost down to the level which can be attributed to DD fusion triton generation.

Neutron yield measurements show that moving the plasma from one surface to another produced very little difference in the DT ratio of the plasma. This suggests that the tritium was not localised in the target plates at the start of the clean up but rather evenly distributed over the plasma facing components..

The tritium retention in the collector probe was $1.5\ 10^{18}$ cm $^{-2}$ after discharge No. 26400. The total vessel inventory at this time can be roughly estimated to be 1.8×10^{20} tritium atoms. This is about 17% of the total tritium injected into the torus and corresponds to a time about 250 discharges after the tritium introduction. The preliminary analysis of in-vessel components removed from the torus gives an estimate of about 3.8×10^{19} tritons retained in the vessel at the end of the experimental campaign.

NEUTRAL INJECTOR CLEAN UP

Two neutral injection boxes are connected to the torus, each accommodating eight PINIS. One of the systems (octant 8) was used for the tritium injection, the other (octant 4) was only operated with deuterium.

The tritium used was 1003 Ci ± 73, of that 54 ± 6 were injected into the torus. After the tritium experiment the LHe panels of the cryopumps of the octant 8 system were warmed up and 988 Ci were released and collected in the tritium recovery system. Neutral injection conditioning pulses were then carried out to remove implanted tritium from the beam dumps and calorimeters [4]. This was successful as the 14 MeV neutron count from these areas decreased by a factor of one hundred after about 100 beam-seconds.

During the LHe panel regeneration which followed the Neutral Injection conditioning pulse, 50 Ci were released, the next regeneration yielded 7 Ci and the subsequent (after a gas purge) yielded 3.5 Ci. In all cases the tritium was collected and stored in the tritium recovery system. For the gas purge the neutral injector box was brought up to ambient temperature, filled with nitrogen at a pressure of 0.3 bar. and then the gas was vented through the stack over a period of several days. The release was 1 Ci.

The total recovered tritium was apparently more than the amount used. This can be attributed to errors in the measuring systems and especially to the uncertainty in the amount of tritium used as discussed above.

Due to the possibility of pumping tritium out of the torus onto the cryopanels, the octant 4 neutral injection system was also carefully regenerated and after three cycles about 2.5 Ci were collected in the tritium recovery system.

SUMMARY

For the first time experiments in high fusion performance tokamak plasmas have been performed using a DT fuel mixture. An equivalent tritium neutral beam of 24 A was injected into a DT plasma heated by deuterium neutral beams. The integrated total neutron yield over the high power phase which lasted about 2 s was 7.2×10^{17}. Total fusion releases were 1.7 MW at peak power and 2 MJ of energy. The amount of tritium injected and the number of discharges with tritium were deliberately restricted for operational convenience.

The techniques used for introducing, tracking , monitoring and recovering tritium have been demonstrated to be highly effective. Essentially all of the tritium introduced into the neutral beam system has been recovered and the same is true for the torus. These levels were sufficiently low for the JET experimental programme in deuterium to continue normally because after a short time the amount of tritium being released from the torus had been reduced to a level suitable for atmospheric release.

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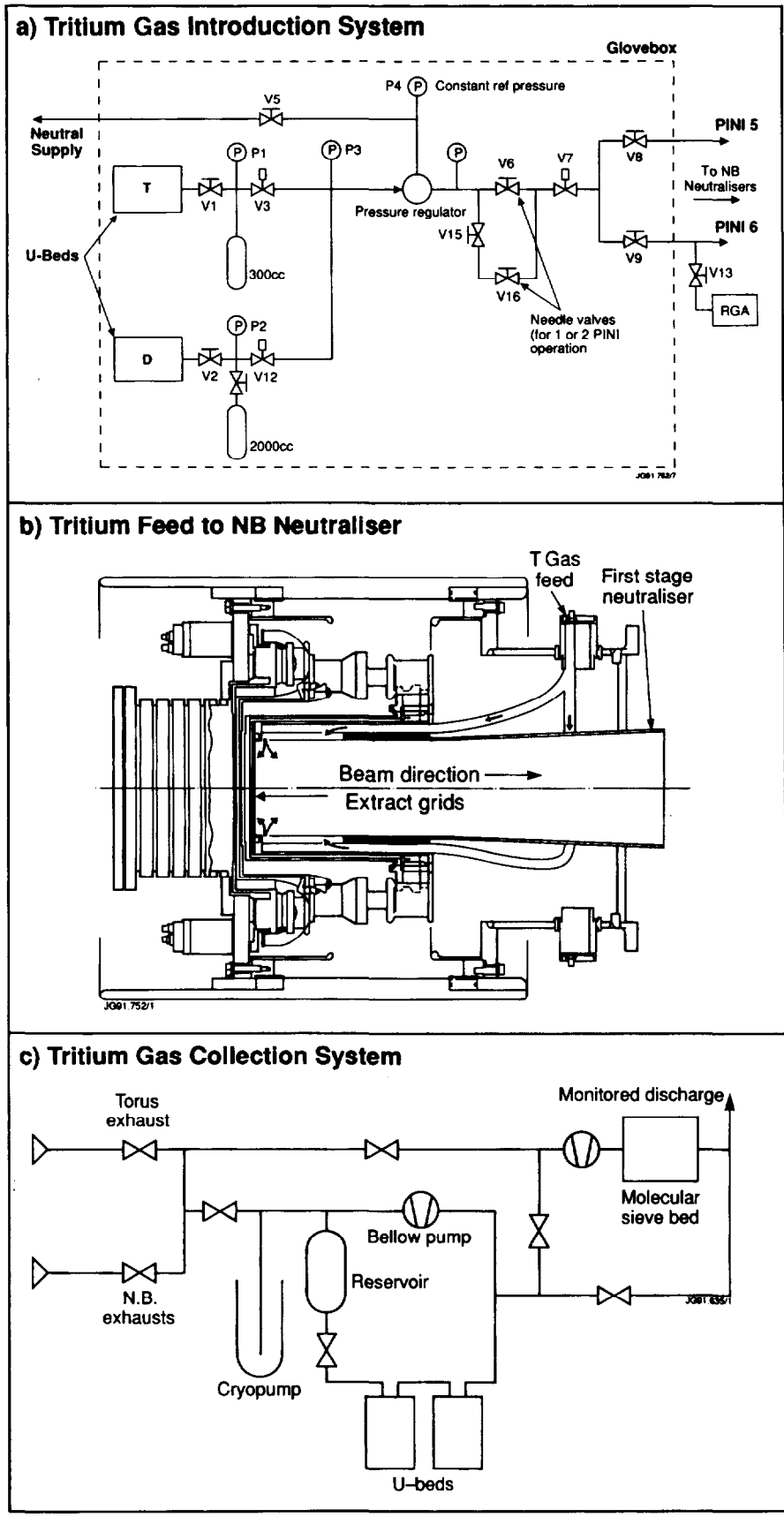


Fig. 1: (a) The tritium gas introduction system, (b) the tritium feed to the neutral beam neutralizer, (c) the tritium gas collection system.

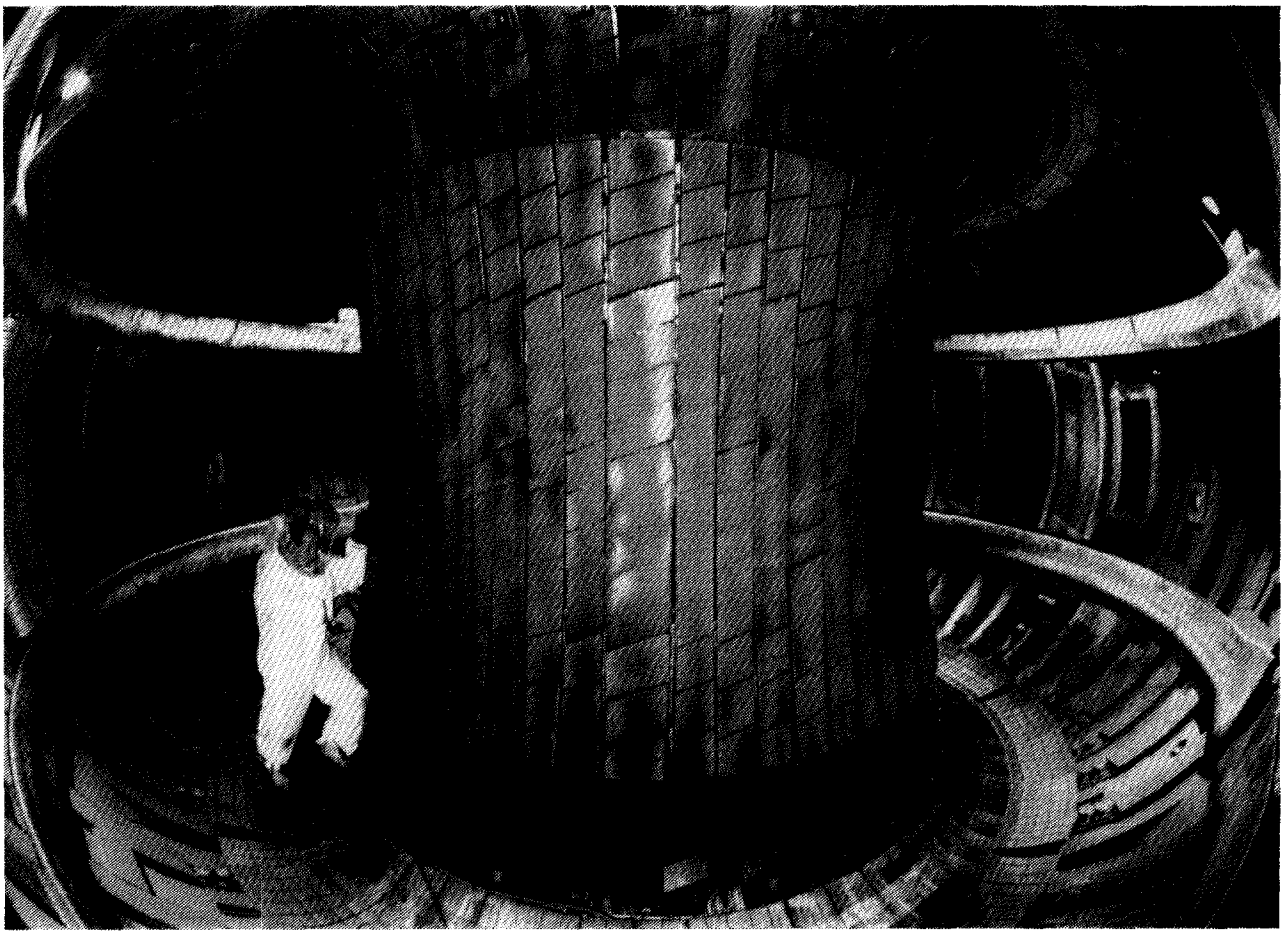


Fig. 2: The inside of the JET vacuum vessel prior to the tritium experiment

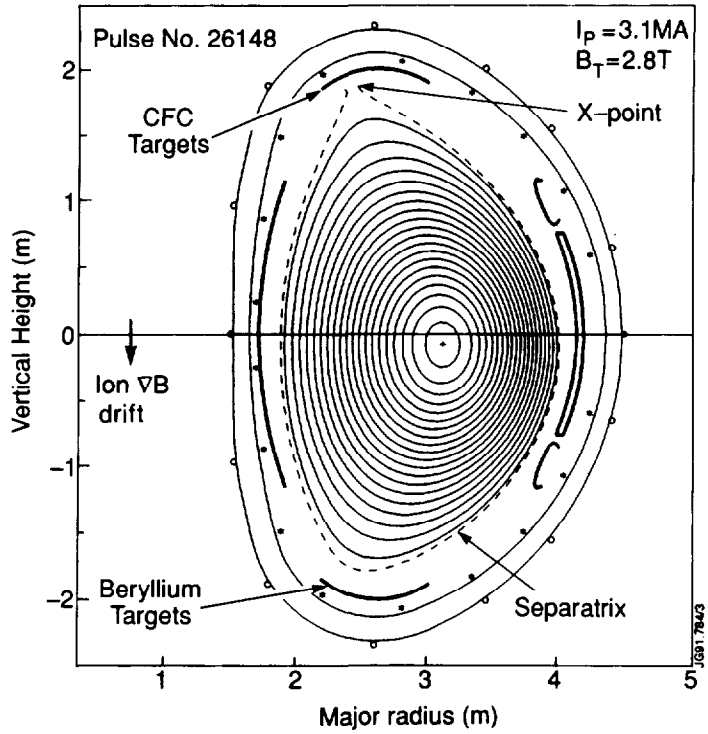


Fig. 3: Magnetic configuration of the plasma used for DT operation. The magnetic axis is at 3.15 m, the horizontal minor radius is 1 m, the elongation κ=1.6 and the safety factor q_ψ=3.8. Shown are the separatrix, the X-point, the ion ∇B drift direction and the carbon fibre composite and the beryllium targets

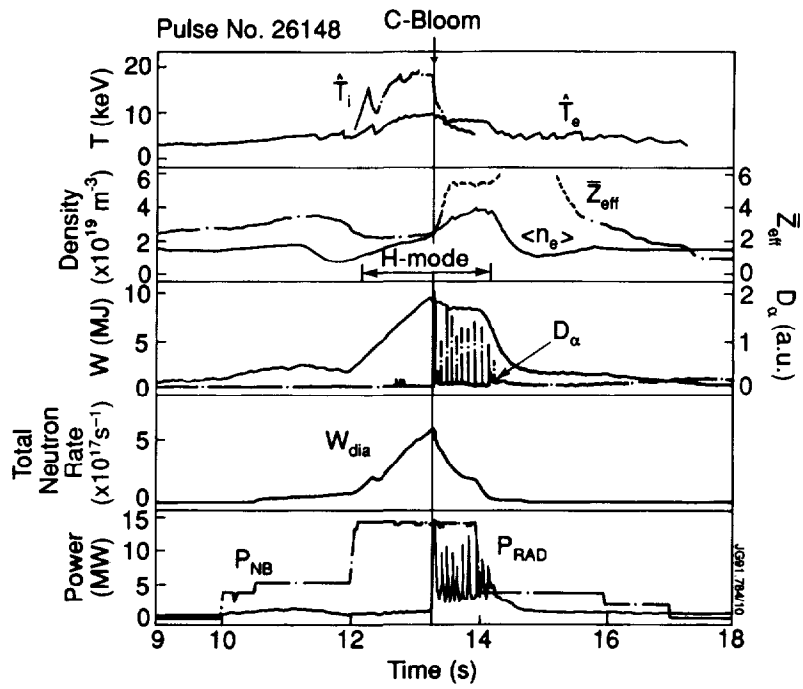


Fig. 4: Time development of the central ion and electron temperature, the volume-averaged electron density the line averaged Z_{eff}, the plasma diamagnetic energy, the D_α emission, the total neutron rate, and the neutral beam and radiated powers for pulse No. 26148. After the "carbon bloom", the Z_{eff} measurement (dotted line) is affected by black body radiation emanating from the target.

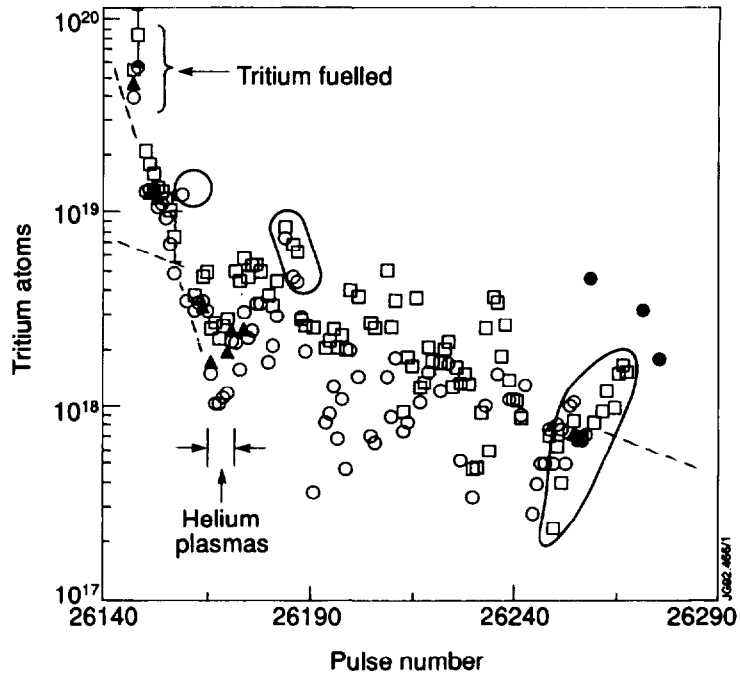


Fig. 5: Integrated release of tritium from the torus a a function of shot number. Circles indicate measurements with the ionisation chamber in the backing line (closed circles measurements following the soak), squares give results from the residual gas analyzer, and triangles data from the sample bottles. The enclosed areas indicate discharges which ending in planned disruption with the exception of discharge No. 26159 (unplanned disruption)

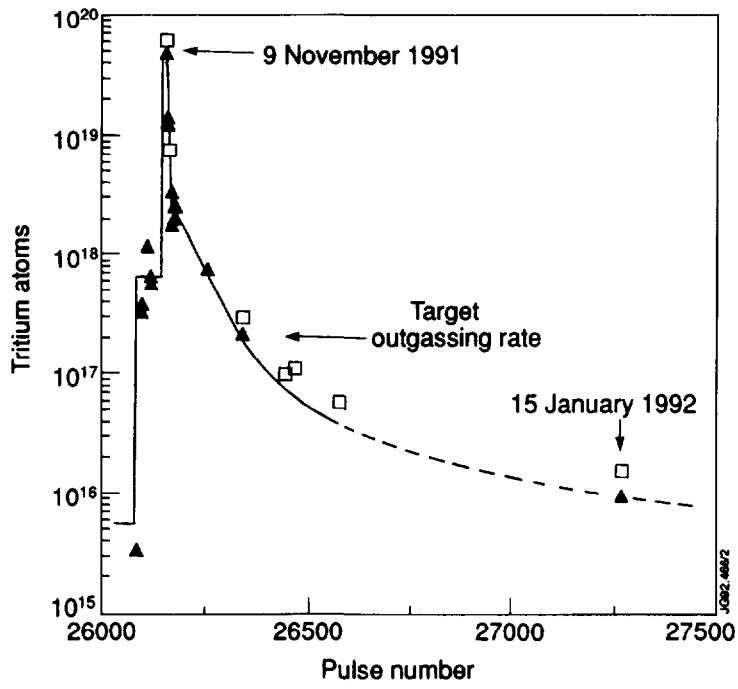
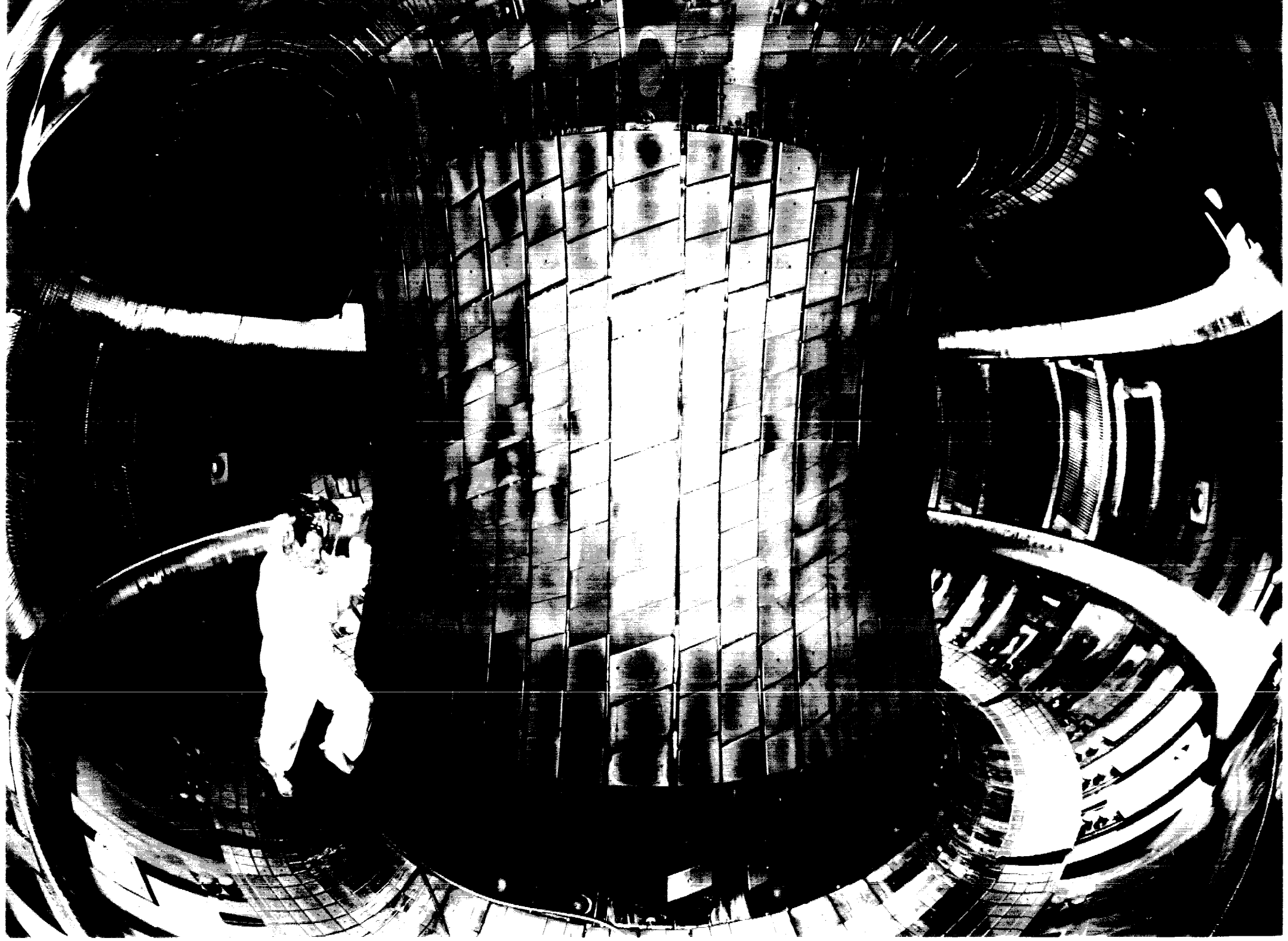


Fig 6: Long term tritium release from the vessel measured by sample bottles. Triangles indicate sampling at 560 s after a discharge, squares sampling at 1160 s



Appendix I

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