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Technical aspects of the joint JET-ISX-B beryllium limiter experiment*

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ABSTRACT

An experiment has been performed on the Impurity Study Experiment (ISX-B) tokamak to test beryllium as a limiter material. Beryllium is an attractive candidate for a limiter and has been proposed for use in the Joint European Torus (JET) experiment. A temperature-controlled, segmented, beryllium top-rail limiter was located inside the plasma radius described by the existing titanium limiters. An extended set of diagnostics was added for measurement of scrapeoff and limiter parameters. These included visible and infrared monitoring systems, probes, and surface analysis experiments. Tokamak experiments included parameter surveys of both ohmically heated and neutral-beam-heated plasmas and an extended fluence test of the limiter. The most

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significant effect of operation with a beryllium limiter was the reduction in low-Z_{eff} impurities caused by gettering action of beryllium deposited on the liner walls. The experiment required the design and implementation of contamination control apparatus and work procedures to prevent the accidental dispersion of beryllium dust.

INTRODUCTION

A continuing problem with impurity control in tokamaks is the interaction between the wall materials and the plasma. This problem is particularly important with nondivertor tokamaks in which there is no implicit mechanism for preventing evaporated or sputtered high-Z materials from being transported into the plasma. As tokamaks become larger and the energy of the neutrals incident on the walls becomes higher, this problem is expected to grow in magnitude. A possible palliative to this impurity source is to make the first surface from a low-Z material. Although the edge plasma will be cooled by radiation, the ions will be fully stripped before they reach the central confinement regions of the plasma and so will not add significantly to the power loss by radiation. For this mechanism to be effective, the limiter would have to be made of similarly low-Z material. There are only two candidate materials: carbon and beryllium. Extensive experiments have been made with carbon limiters; the primary drawbacks to their use are the chemical erosion of the material in a hydrogenic plasma and the hydrogen absorption of the active surface. The latter is important for two reasons: the impact on density and particle control and the increased tritium inventory required for a deuterium-tritium (D-T) experiment. The drawbacks to the use of beryllium are that beryllium dust is highly toxic and that beryllium has a low melting point compared to carbon. The toxicity requires precautionary measures that in general will complicate tokamak operation, and the low melting point renders the limiter more vulnerable to surface damage during transient heat loads caused by, for instance, disruptions.

It has been suggested that beryllium limiters should be installed in the Joint European Torus (JET) tokamak,[1] and two experiments have been performed in order to gain some insight into this application. The first pioneering experiment was performed on the Unitor experiment by Hackmann and Uhlenbusch; it indicated a sensible improvement in the plasma performance over that obtained with stainless steel and carbon limiters.[2] The second experiment was performed in the Impurity Study Experiment (ISX-B) tokamak at Oak Ridge National Laboratory (ORNL) and is the subject of this paper.

EXPERIMENT DESCRIPTION

The ISX-B tokamak is a medium-size, research-oriented tokamak with neutral beam heating. The primary mission of the device has been the exploration of confinement scaling and impurity transport with auxiliary heating.[3] Following the end of the main program, a joint experiment by JET and the ISX-B team was initiated to install a beryllium limiter and to study the survivability of the limiter in an operating environment that would be as transferable as possible to the planned JET experiments. In addition, extensive measurements were made to identify the effect on the plasma parameters of using beryllium as a limiter material.

A top-rail limiter was fabricated from 12 solid S-65-B beryllium tiles, which were installed on a temperature-controlled base plate. Figure 1 shows a cutaway of the top-rail beryllium limiter assembly, installed on the tokamak. This figure includes a toroidal projection of the titanium-carbide-coated limiters previously used for the main experimental program and a representation of the plasma cross section. Figure 2 shows the details of the limiter tiles, which are made from solid beryllium. Alternate tiles were sliced into 10- by 13-mm tessellations to test whether this treatment resulted in any significant reduction in surface stress. Figure 3 shows the results of a theoretical simulation of the thermal surface stresses and indicates the considerable improvement expected.[4] The tiles were bolted down to a stainless steel base plate, which was maintained at elevated temperature by passing a 200°C fluid (Dowtherm LF[5]) through single-loop stainless steel lines attached to the base plate. The material was maintainted at an elevated temperature to improve the ductility of the beryllium and hence its ability to resist stresses. Figure 4 shows the rather dramatic increase in ductility of S-65-B beryllium even at modest elevated temperatures.[4]

A number of diagnostics were added to the machine for this experiment, including spectroscopic and infrared viewing of the limiter, Langmuir probes to investigate the scrapeoff layer, and a surface analysis station to study material and surface effects.[6] In addition, extensive precautions were taken to prevent the accidental escape of beryllium dust into the area surrounding the experiment and to suitably monitor the experimental area for any possible contamination with beryllium. Figure 5 shows a plan view of the apparatus after modification for the beryllium limiter experiment.

EXPERIMENTAL PROGRAM

The primary objective of the experimental program was to evaluate the suitability of beryllium for installation in the JET experiment. The two areas of greatest concern are the mechanical survivability of the limiter under thermal and mechanical stresses and the hydrogen retentivity, the latter because of information to be gained on expected tritium inventories. To meet the requirements for data relevant to these concerns, a set of specifications was established for the thermal cycle and particle fluences to which the limiter would be exposed. These were that the surface temperature would be cycled from base temperature to 800°C for at least 3000 beam-heated plasma shots and that the accumulated surface fluence to the limiter would be at least 10^{22} ions•cm^2. The surface fluence was defined as $0.4n_e(kT_e/m_i)^{1/2}$ in the plasma edge, with the plasma parameters calculated from Thomson scattering measurements at the edge. Figure 6 shows the time history of this "score sheet" during the lifetime of the experiment.

In addition to these limiter qualifying experiments, a set of experiments was defined to compare the plasma and edge properties for a reference set of measurements taken with a titanium carbide limiter at the same radius (22 x 24 cm) as the beryllium limiter experiments. These experiments were initially performed with plasma parameters that resulted in power loads to the beryllium limiter that did not produce limiter melting. Results with the two limiters were quite similar. Materials released from the wall, low-Z (O, C) and medium-Z (Cr, Fe, Ti) impurities, rather than material from the limiter dominated the plasma behavior. It was found that the power to the limiter was a sensitive function of the plasma current for constant density and beam power. This effect is displayed in Fig. 7, which shows the temperature of the different tiles for different plasma currents. It was decided to increase the power to the limiter, by use of the effect illustrated in Fig. 7, in order to melt the surface, with the expectation that some beryllium would be evaporated into the plasma and redeposited onto the liner walls, where it would act as a getter for the recycling impurities. This is exactly what happened. Figure 8 shows the intensity of an oxygen and a beryllium line before and after the limiter melting experiments. As can be seen, the oxygen line is much reduced in intensity following the limiter melting. Further evidence of this gettering can be seen from measurements of the plasma radiation. A typical measure of machine cleanliness is the ratio $P_{rad}/\bar{n}_e$. Figure 9 shows this ratio as a function of shot number during a single day of operation. Initially the ratio is high and characteristic of ungettered discharges. Then after several plasma shots this ratio falls towards values typical of "clean" gettered discharges.

A consequence of the high limiter temperature was that significant amounts of beryllium were transported into the plasma. This effect can also be seen in Fig. 8, where the intensity of the beryllium line is drastically increased for the case of getter levels of power to the limiter. Spectroscopic estimates of the total plasma beryllium content, n_{Be}/n_e, were as high as 0.05. The beryllium is fully ionized in the main plasma volume, so this value is very sensitive to the model assumed for the radial profile of the beryllium ions. These high levels of beryllium did not have any serious impact on the plasma parameters or operating envelope, which remained those typical of ISX-B operation with the usual getter material, titanium.

Close to 2 g of beryllium are missing from the limiter tiles, with the center tiles showing a weight loss of about 0.75 g each. Photographs of the surface indicate that most of this was probably lost as droplets formed by the melting beryllium and falling to the floor of the vacuum vessel. Visual inspection of the floor shows beryllium droplets and splatter in the regions toroidally near the limiter, and a window underneath the limiter shows considerable spallation from molten droplets.

Work is still under way to quantitatively determine the amount of beryllium plated out on the interior of the plasma vessel. These deposits result from a cycling of beryllium through the plasma, and the amounts deposited locally are expected to vary considerably with the distance of the sampled area from the plasma edge and from the limiter itself. To facilitate the measurements, a number of sample plates were distributed around the vessel before the experiment began. Some of these were placed on the bottom of the vessel and were therefore at maximum distance from the core plasma. These have been analyzed by a nuclear reaction; the beryllium coatings are about one monolayer on samples underneath the limiter, falling to about 0.25 monolayer 180° away (on the other side of the torus). The beryllium inventory suggested by this distribution is only about 2 mg. As expected, though, there are much thicker deposits elsewhere. An Auger analysis of sample plates from the midplane toroidally near the limiter indicates about one monolayer deposited per high-current (limiter-melting) discharge.

These deposition measurements continue. However, on the basis of present evidence, the bulk of the limiter material was lost in the form of molten droplets, and only a small fraction was cycled through the plasma and deposited on the walls.

CONCLUSIONS

Under ungettered conditions the plasma parameters are dominated by the wall materials when limiters that do not contribute significantly to the impurity source are used. For gettered discharges the primary contaminant is the getter material, if produced by limiter melting. Using beryllium as the getter material does not appreciably increase the Z_{eff} of the discharge. The main effect of the gettering was to reduce the source of recycling gases such as oxygen and nitrogen. This in turn reduces the sputter contribution of the uncoated wall surfaces.

Beryllium is a satisfactory limiter material if run at temperatures below its melting point. This requires large surface areas and close matching to the power scrapeoff length, which can be only approximately predicted.

For short-pulse tokamaks the thermal properties of beryllium are acceptable for limiter operation. This may not be true for long-pulse experiments in which the stress scale lengths may be many centimeters.

ACKNOWLEDGMENTS

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[5]Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute its endorsement, recommendation, or favoring by the United States Government or any agency thereof.

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FIGURE CAPTIONS

FIG. 1. Cross section of the ISX-B vacuum vessel at the beryllium limiter location, showing the plasma outline and the toroidal projection of the titanium carbide limiters.

FIG. 2. Drawing of tessellated beryllium limiter tile, showing mounting details.

FIG. 3. Effect of slot spacing on surface stress (model assumes 2.5 kW/cm^2 for 0.3 s).

FIG. 4. Measured values of ductility as a function of temperature for two grades of beryllium.

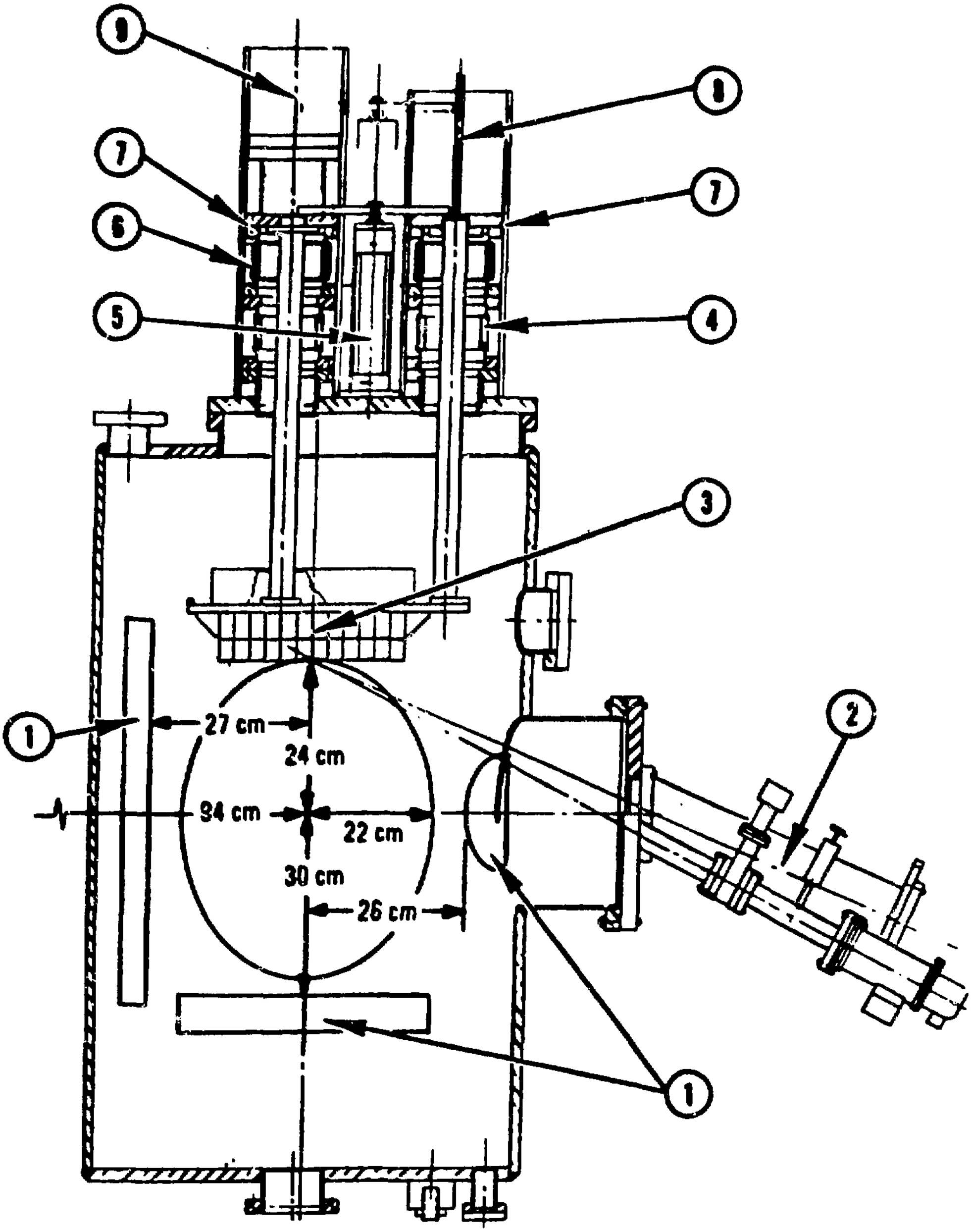
FIG. 5. Plan view of ISX-B tokamak, showing configuration used during the beryllium limiter experiment.

FIG. 6. Graph of beam-heated shots (stepped curve) and estimated limiter fluence (smooth curve) during the beryllium limiter experiment.

FIG. 7. Limiter temperature as a function of plasma current for neutral-beam-heated discharges (P_B = 0.85 MW).

FIG. 8. Time history of oxygen and beryllium lines.
Solid line: 115 kA, no limiter melting. Dashed line: 155 kA, limiter melting during discharge.

FIG. 9. Plasma current and $P_{rad}/\bar{n}_e$ as a function of shot number during one day of operation.



1–TOROIDAL PROJECTION OF TiC LIMITERS
2–LIMITER SPECTROSCOPY
3–LIMITER TILES
4–CERAMIC BREAK
5–PNEUMATIC CYLINDER
6–BELLOWS
7–FINGER STOCK
8–COOLING LINES
9–THERMOCOUPLES

Fig. 1. Cross section of the ISX-B vacuum vessel at the beryllium limiter location, showing the plasma outline and the toroidal projection of the titanium carbide limiters.

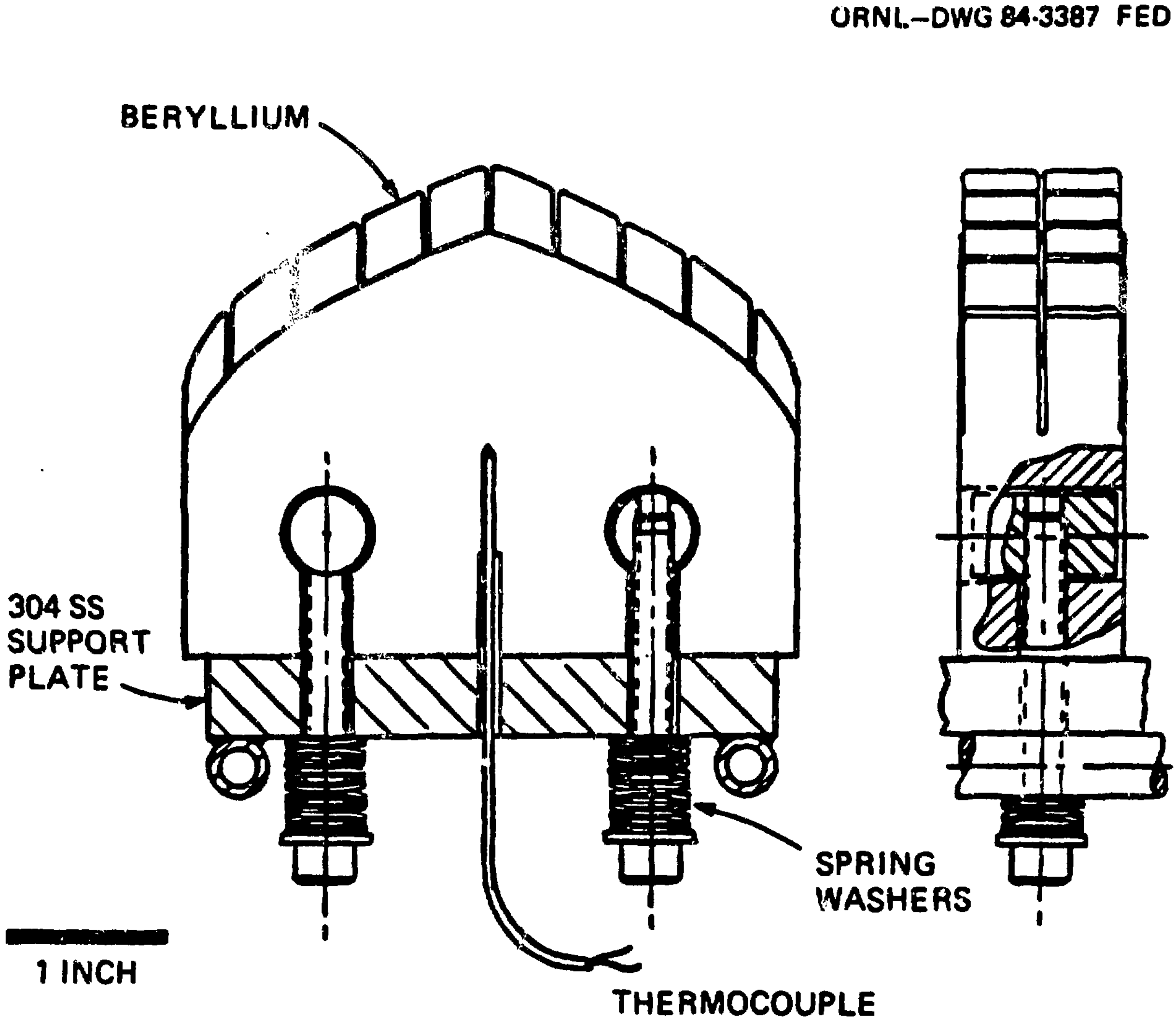


Fig. 2. Drawing of tessellated beryllium limiter tile showing mounting details.

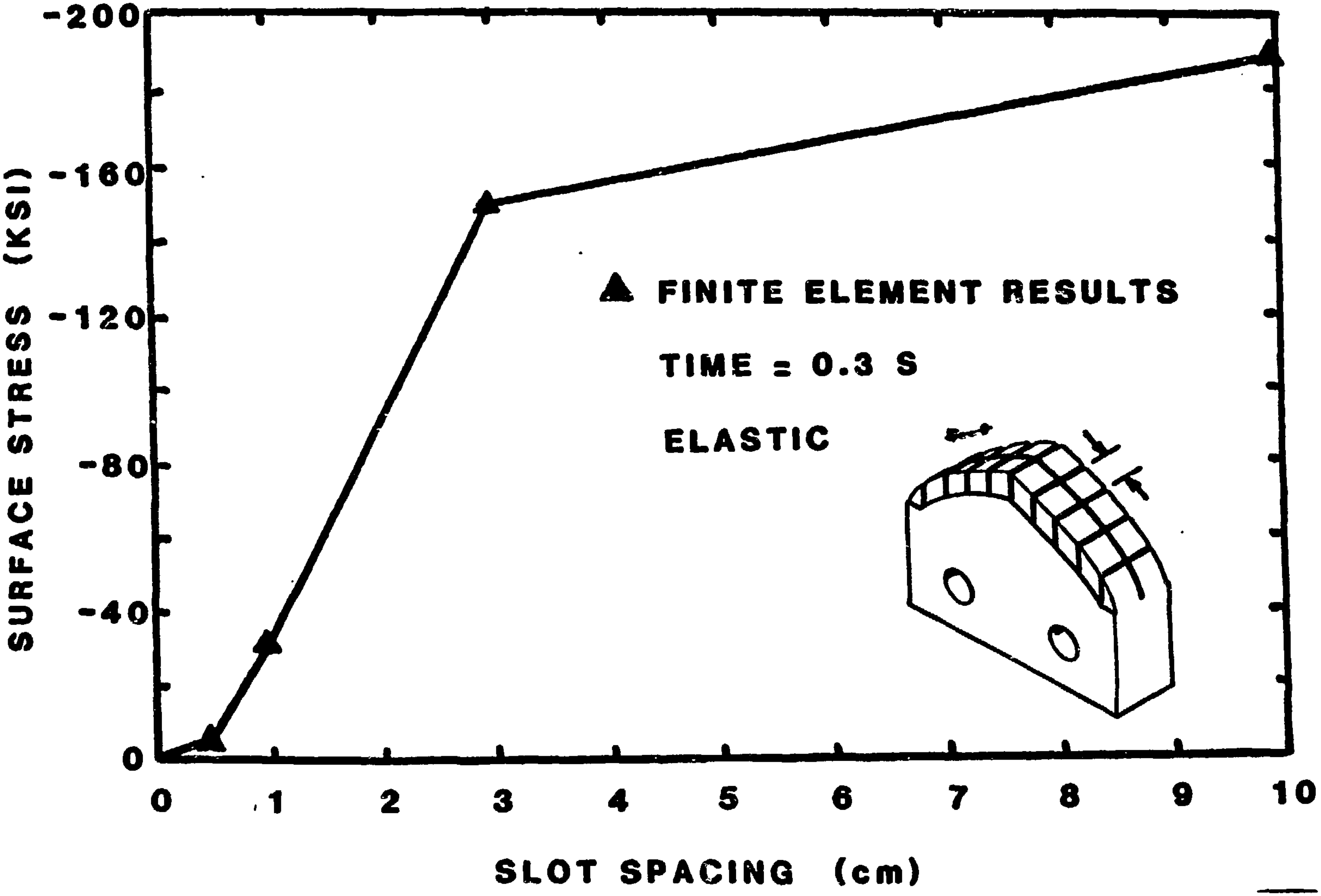


Fig. 2. Effect on surface stress and changing slot spacing (model assumes 2.5 kW/cm2 for 0.3 secs.)

DUCTILITY

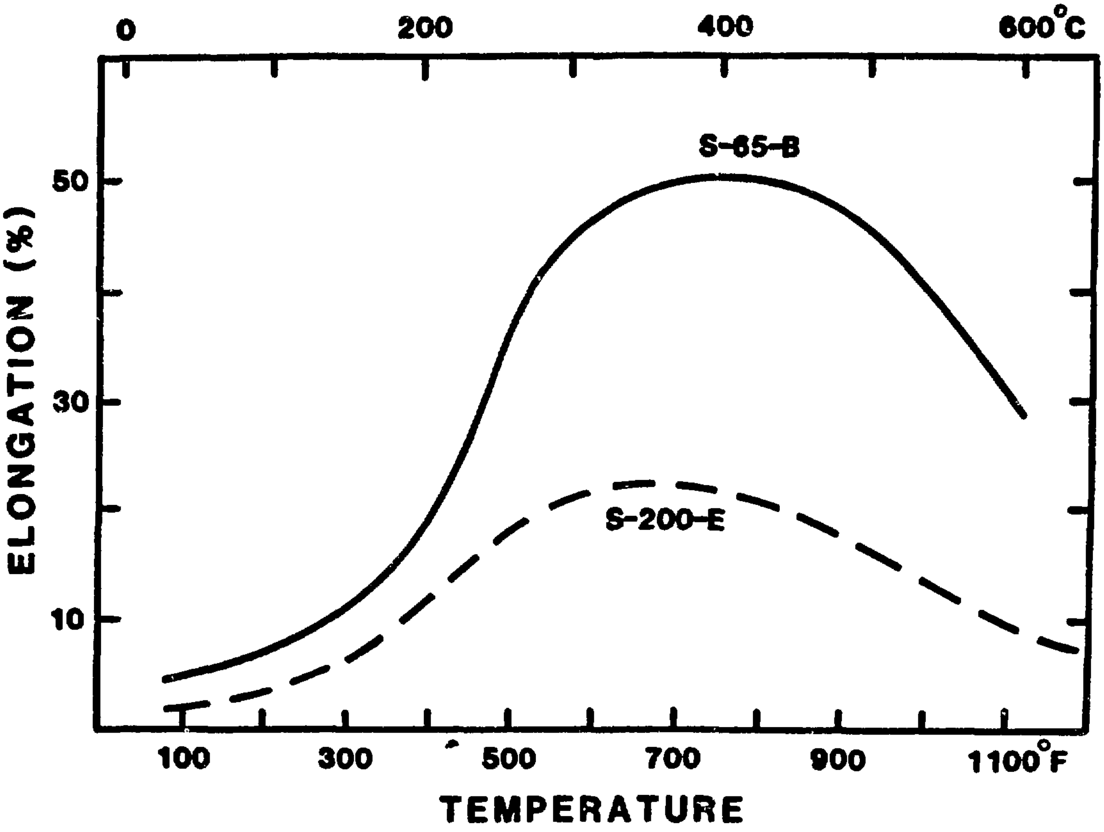


Fig. 4. Measured values of ductility as a function of temperature for two grades of beryllium.

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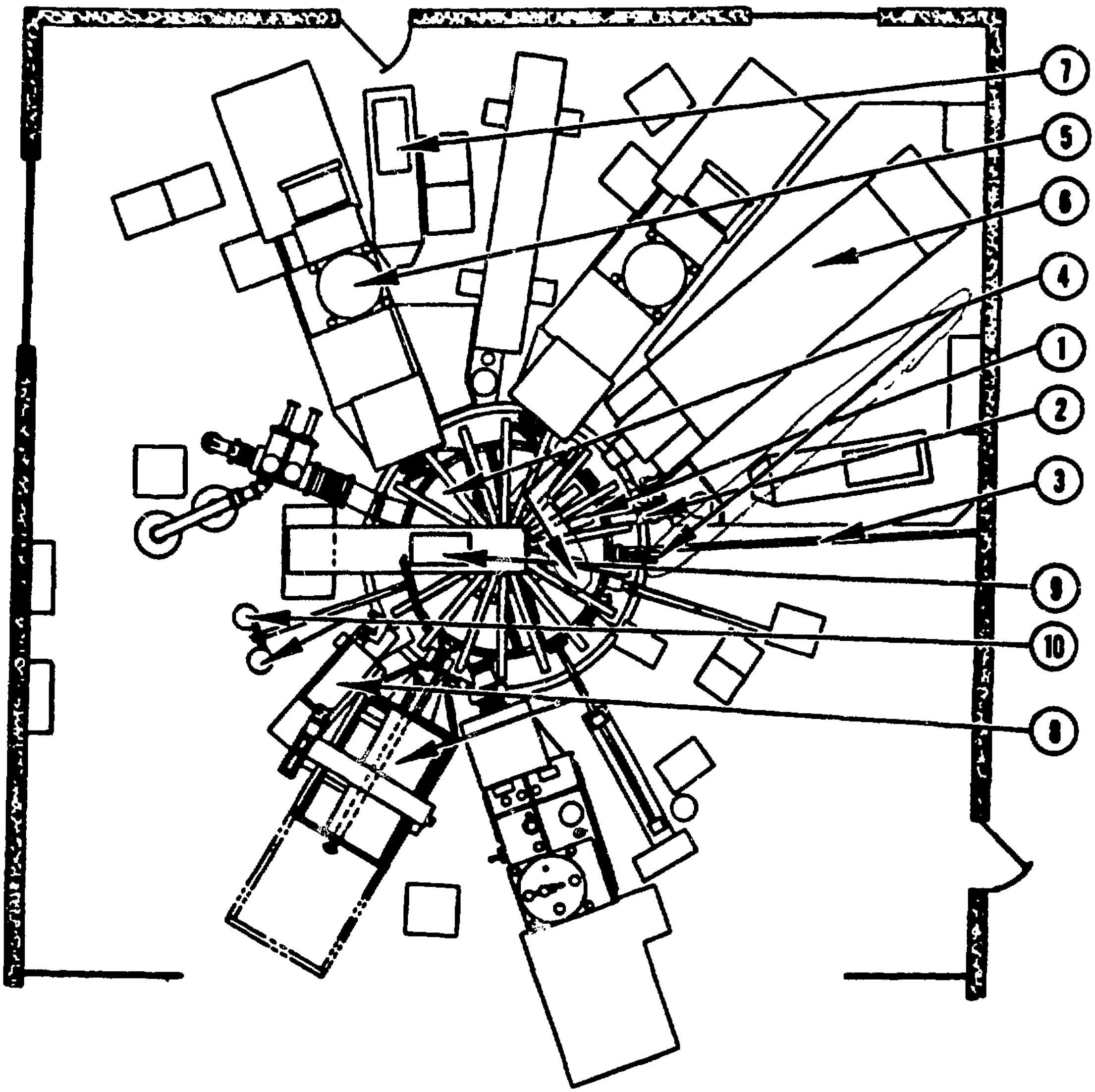


1–BERYLLIUM LIMITER
2–LIMITER SPECTROSCOPY
3–SURFACE ANALYSIS TRANSFER STATION
4–LANGMUIR PROBES
5–BEAM LINE USED DURING EXPERIMENT
6–THOMSON SCATTERING APPARATUS
7–VISIBLE
8–V.U.V. SPECTROMETERS
9–TiC LIMITERS
10–LINER VENTILATION APPARATUS

Fig. 5. Plan view of ISX-B tokamak showing configuration used during the beryllium limiter experiment.

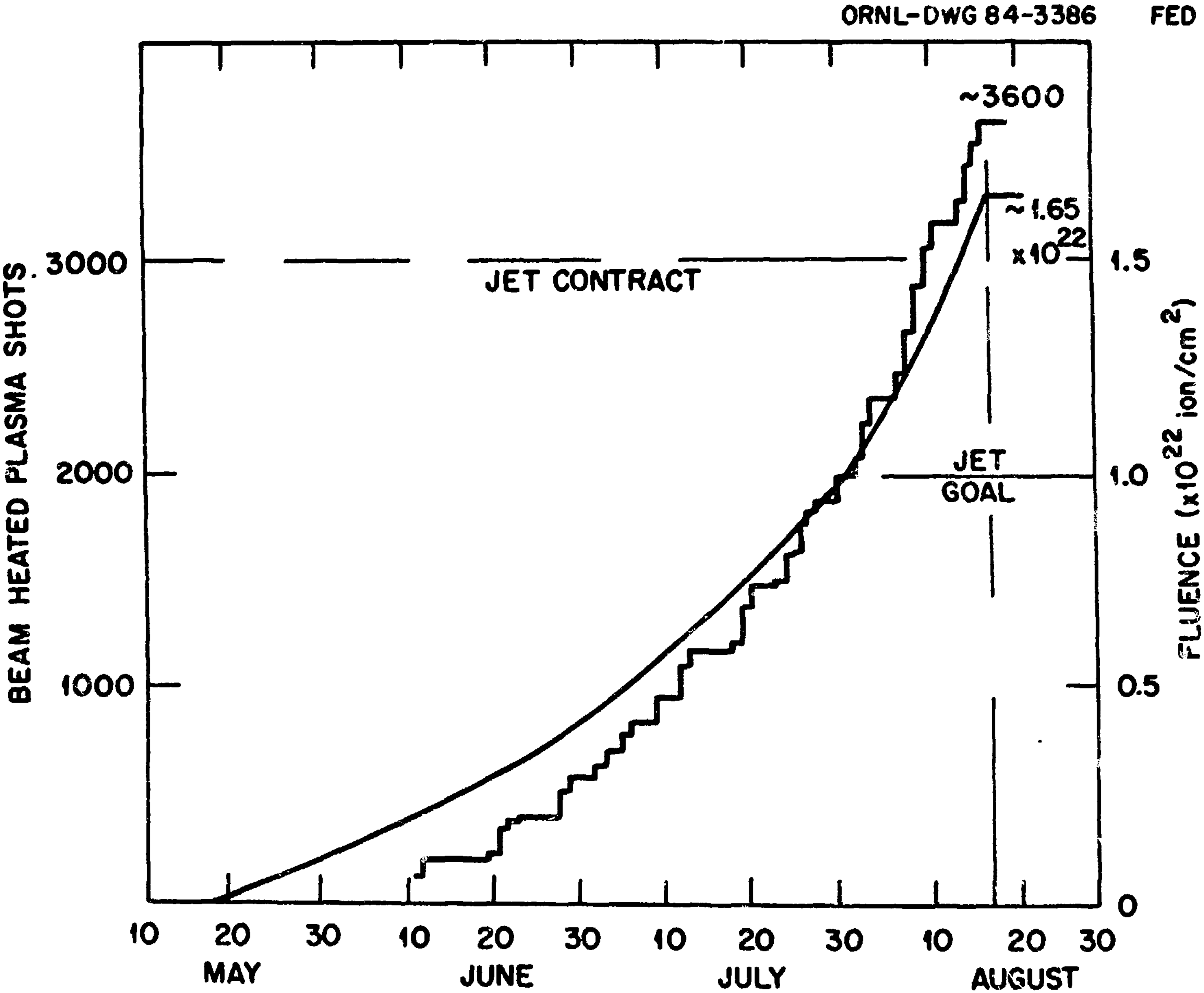


Fig. 6. Graph of beam heated shots and estimated limiter fluence during beryllium limiter experiment.

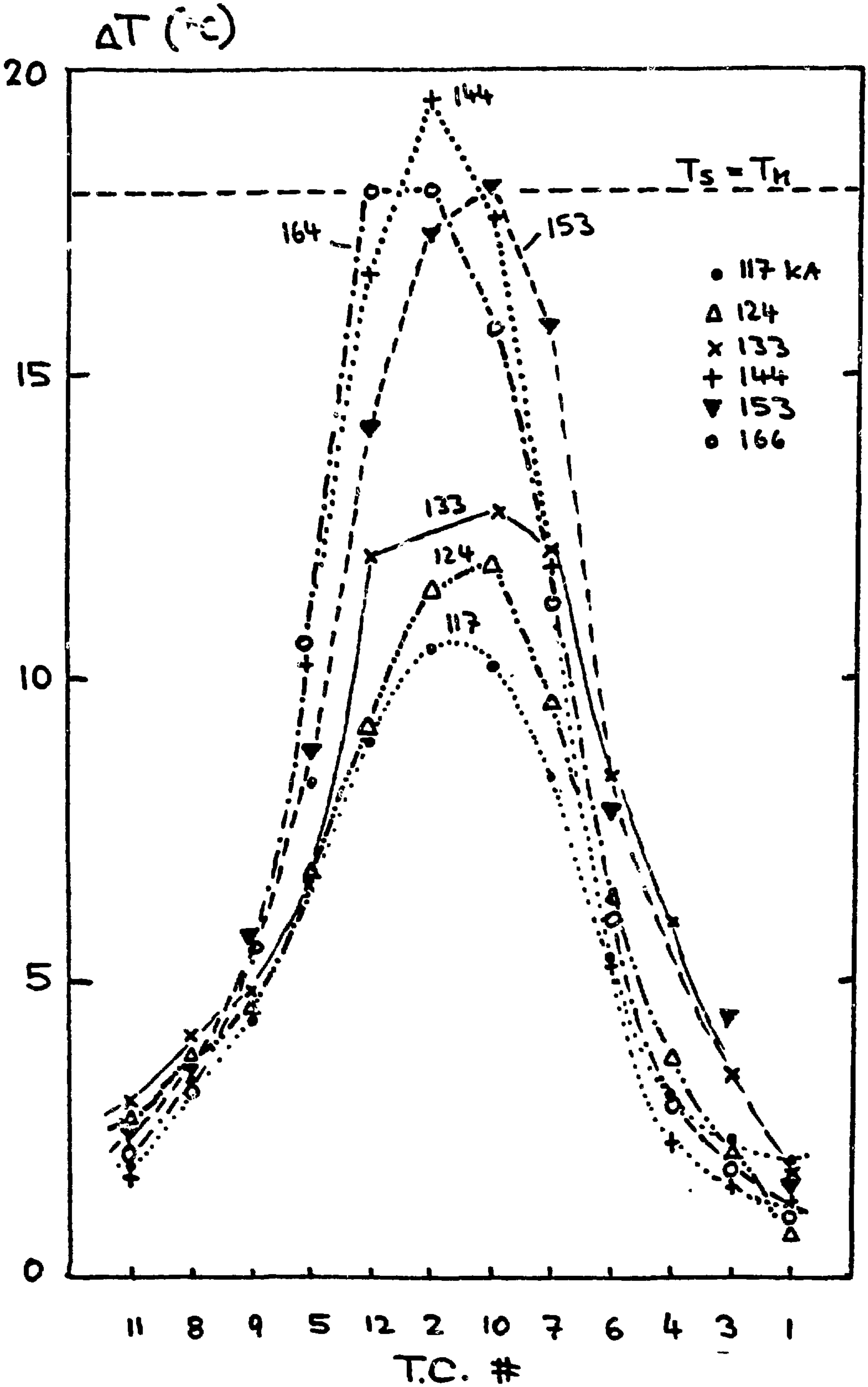
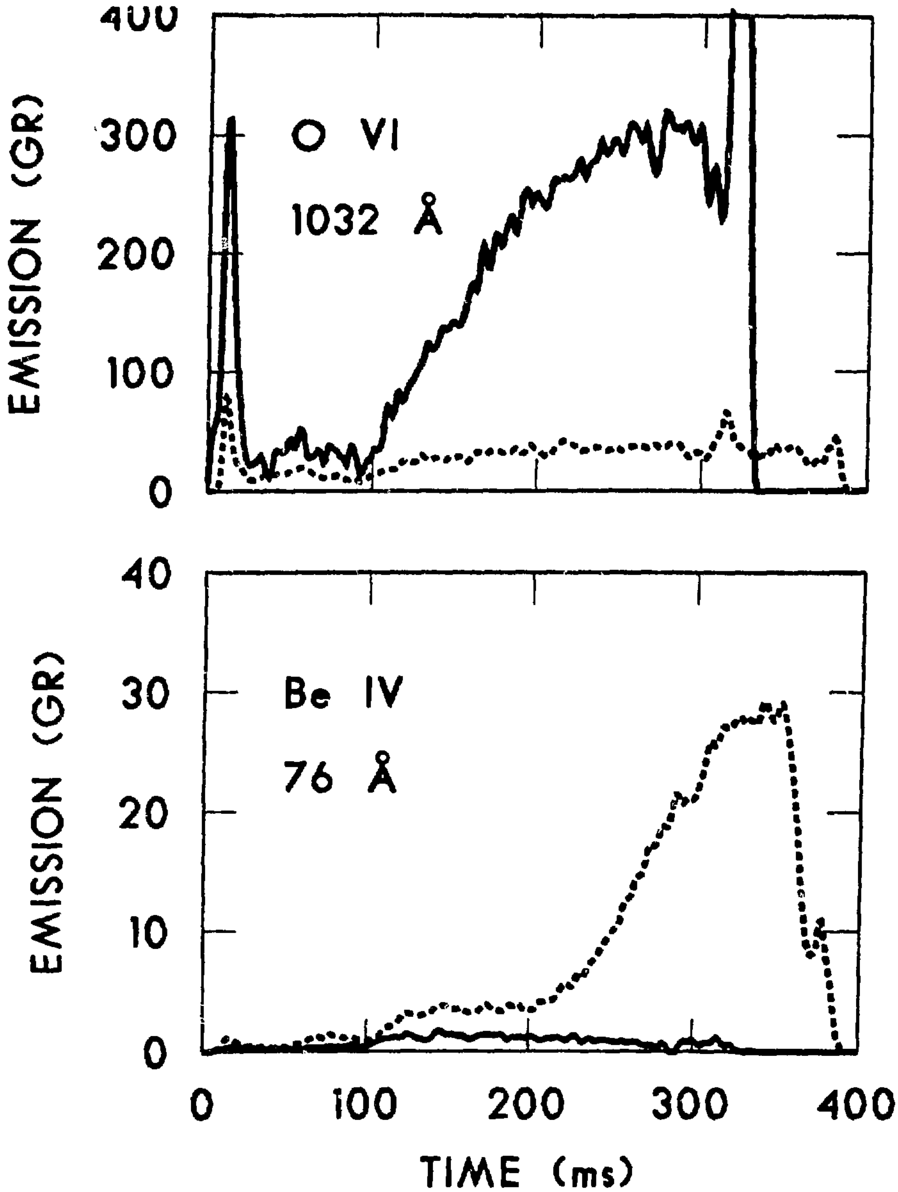


Fig. 7. Limiter temperature and energy distribution as function of plasma current for NBI heated discharges (PB=0.85 MW)



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