Supplement to
"ASDEX Upgrade, Definition of a Tokamak Experiment with a Reactor-Compatible Poloidal Divertor"
(IPP-Report 1/197, March 1982)

by

ASDEX Upgrade Design Team
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IPP 1/211 November 1982

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Abstract

Since March 1982 the better understanding of the divertor physics, both by theory and experiments, and the development of the ASDEX Upgrade concept have considerably improved and simplified the ASDEX Upgrade design. Single null poloidal divertor configurations were calculated, which can well compete with elongated limiter configurations in reduced poloidal field effort. The role of recycling and its limitation set by the available energy flux, observed experimentally and explained by a plasma boundary flow model, led to a refined formulation of the line density requirements.

Finally, a discussion of the attainable temperature and densities allowed clearly to distinguish between ASDEX and ASDEX Upgrade and pointed out the dominant role of the plasma current.

The ASDEX Upgrade basic data are summarized as presented to the EURATOM advisory board.

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1. Introduction

In March 1982 the ASDEX UG report IPP 1/197 was written as basis for application for EURATOM preferential support phase I. Since then the development and understanding of the ASDEX UG concept has made progress in physics and technical area as well. This was caused to a part by experimental results of ASDEX and other tokamaks and supporting theoretical work to a part by the reactor and questions of experts to the ASDEX UG proposal and to a good part by the progressing analysis of the components of the tokamak machine.

Experimental results, primarily of ASDEX, and outcome of continued theoretical investigations in particular lead to

a) increased confidence into the poloidal divertor concept as a viable solution for the energy transfer, impurity control and helium pumping problem for INTOR and fusion reactors and into the choice of machine parameters for ASDEX Upgrade;
b) emphasis on the (INTOR-type) single null divertor configuration for the experimental investigations of ASDEX Upgrade;
c) incorporation of a pumping system capable of Helium pumping as a design criterium;
d) realization of the potential of a divertor-tokamak to make unique contributions also to general tokamak problems like plasma confinement β_p-limits and current drive.

The increased effort in the ASDEX UG tokamak system design in conjunction with investigations on the physical configurations and parameter regimes realizable and available by the conceived techniqual concept has finally led to an rigorously simple tokamak design. It uses water cooled copper coils for the TF magnet and the poloidal field system. All PF-coils are located outside the TF-magnet. Each coil system uses its own force support structure such that the function of each system is clearly seperated. The emphasis has been set on single null divertor and high plasma current (I_p = 2 MA) operation. Double null divertor configurations can still be produced with reduced parameters or reduced discharge duration. The IPP-Garching fly wheel generator power supply suffices to carry through the ASDEX UG programme with the aims described below.

This solution urged the IPP Garching to dispense with a superconducting ASDEX UG concept which was conceptually worked out in co-operation with the Kernforschungszentrum Karlsruhe. The superconducting coils had increased size and costs of the system considerably - in costs by more than a factor of two - and had introduced uncertainties with respect to construction time, e.g. because of the required development of poloidal field coils for time varying currents.

A critical discussion of the physical parameter regimes attainable by ASDEX itself including the possibility of modifying the ASDEX multipole coil system had shown that ASDEX can not nearly meet the aims set. Furthermore has ASDEX a tightly scheduled programme up to the year 1986 a year in which the construction of ASDEX UG should reach completion.

The results of these careful investigations persuaded the IPP Garching to propose the ASDEX UG as described in /1/ and summarized below.

2. The Aims of ASDEX Upgrade

The aims of ASDEX UG cover essentially all plasma boundary and first wall problems which can be investigated by discharges without thermonuclear heating. In particular they encompass:

a. the study of plasma boundary problems of a fusion reactor associated with the
   - transfer of the energy flux to the walls
   - control of wall produced impurities
   - helium pumping

   in a poloidal divertor and a pump limiter configuration.

b. the utilization of the improved impurity and plasma density control of divertor configurations
   - to study the confinement properties of clean, separatrix-bound plasmas in the high current regime
   - to study long pulses and current drive.

c. the provision of a sound basis for the final decision on the NET concept, in particular whether a pump limiter solution suffices or a divertor is needed.

Plasma boundary layer studies should, in particular, compare different promising solutions for the energy transfer and impurity balance problem like

- high density scrape-off layer with protection of the target plates by a zone of intensive localized recycling
- a hot boundary layer in front of a low-Z coated pump limiter taking the major fraction ot the energy flux, with nearly unatenuated sputtering but sufficiently small impurity confinement time
- a radiation zone formed by a saturated concentration of medium-Z impurities forming a protective cold plasma mantle in front of limiter and walls

and examine whether a high enough neutral helium density can be achieved in the pumping chamber of a poloidal divertor and a toroidal pump limiter for efficient ash removal.

These boundary layer studies should be carried out with the poloidal divertor configuration (a single null divertor with an elongated plasma cross-section) and a pump limiter configuration with similar shape of interiour flux surfaces. Both plasma configurations have high reactor potential but are - unassisted - unstable against vertical displacements.

2.1 Requirements for achieving the aims

Passively stabilizing elements inside the main field coils and active feedback on PF coils outside the toroidal field coils will be provided to test and demonstrate the control needs of a divertor and toroidal pump limiter.

Reactor similarity requirements on the scrape-off needed for either

a) the formation of a photosphere radiating a dominant fraction of the input power in a limiter, or
b) the formation of a thick, screening, scrape-off layer even in case of an open divertor configuration

have to be satisfied.

The favourable properties of a poloidal divertor demonstrated by ASDEX shall be utilized also for a study of plasma confinement, heating and current drive.

In particular, we will pursue the studies of the so-called H regime found in ASDEX into the regime of plasma currents, densities and temperatures allowing a comparison with the results of non-divertor devices like JET, TFTR and TORE-SUPRA. We will further explore this parameter space for confinement regimes inaccessible or difficult to reach for limiter-bound devices. These studies may hopefully benefit also the pump limiter scenario through understanding of the basic features of these regimes, and indicate a way for reaching it.

Theories and experiments for extremely long pulse or current drive scenarioes indicate the importance of a control of plasma and impurity density like can best be achieved in a divertor tokamak. By having (from power supply and cooling of TF and PF coils) the potential for pulse lengths of several minutes for plasma currents in the range of present ASDEX-values ($<$ 500 kA) ASDEX Upgrade should serve a test bed for further such studies.

3. Physics basis update

In this section we summarize recent physics developments which have had a significant impact on either the layout of ASDEX Upgrade, the assessment of its importance, or the viability of the divertor solution for a tokamak reactor.

3.1 Energy limits on recycling

Measurements of the line-integrated scrape-off density in ASDEX /2/ show a strongly nonlinear increase with bulk plasma density to a saturation value. This saturation value is an approximately linear function of input power suggesting its interpretation (fig. 1) as an energy limit on divertor recycling, in good qualitative agreement with predictions of a one-dimensional parallel flow model (fig. 2) /3/.

In the saturated regime the plasma temperature in the target plate vicinity stays below 10 ev, i.e. in a regime where the ionization rate starts decaying rapidly, and the impurity production is very low. An increase in power input results in an increase of scrape-off plasma density, but not of temperature. This is accompanied by an increase in neutral gas density (fig. 3); applied to reactor requirements these results imply that pumping (of helium) should become the easier, the higher the energy flow in the scrape-off region entering the pumping chamber.

These results show that even at high power flows, a divertor tokamak can be expected to operate with low divertor plasma temperatures and small impurity production provided it can be brought into the high recycling regime. At the same time they underline that the divertor will have a significantly higher probability of solving the helium pumping problem than the pump limiter, as in the latter only flux surfaces with strongly reduced power flow will enter the target chamber.

3.2 Discovery of a regime of improved confinement and high β_{pol} in neutral beam heated divertor discharges in ASDEX

High power ($P_{INJ} \gtrsim 2$ MW) neutral injection heated discharges in ASDEX /4/ have shown the existence of a new confinement regime (labelled H) distinguished by significantly increased values of τ_E, τ_p and β_p over the results of other tokamaks and of limiter discharges in ASDEX. The failure to reproduce H-type results so far in limiter discharges is not theoretically understood. It might be connected to the very broad temperature and density profiles and the resulting high edge temperatures associated with the H-regime, which may not be compatible with the impurity househould of a limiter device.

Obviously a divertor tokamak can realize with relative ease discharge regimes not immediately accessible to limiter tokamaks. By reaching them and allowing to study their basic necessities it hopefully will indicate a way for realizing them also in other tokamaks.

ASDEX Upgrade should pursue these studies into a regime of n, T and I_p allowing a direct comparison with results of JET, TORE-SUPRA, TFTR and other next generation devices.

3.3 The role of the plasma current in determining energy confinement and density limit

Tokamak experiments with additional heating described at the Baltimore conference concur in suggesting an approximately linear dependence of energy confinement time on plasma current - a dependence holding also for the ASDEX results in the H regime. PDX and Doublet III have verified this dependence down to q-values of order 2, and Doublet III has demonstrated also the beneficial effect of the increase in plasma current made possible by noncircular plasma cross-sections.

The achievable density in Ohmically heated devices has long been known to depend also on q in the form of the so-called Hugill diagram. Recent experiments in Doublet III have increased the range of the $n_{crit} \sim \frac{1}{q}$ dependence down to values below 2. This implies that devices designed with more conventional assumptions about the achievable q will be limited in density by their current carrying rather than by their B_t/R capability. An increase in density by neutral injection by approximately a further factor of 2 beyond this limit has frequently been obtained, but results are still far off from indicating an equivalence between Ohmic and neutral injection heating.

For ASDEX Upgrade this implies that its high current-carrying capability (a factor of 4 larger than in ASDEX) has become an essential design feature rather than only a by-product of its B_t/R capability and the noncircular cross-sections. The prime limiting components on I_p in a divertor tokamak are the large required multipole currents; the design of an optimized single null configuration with a concurrent reduction in the sum of the needed multipole currents by a factor of order 3 as compared to the double null lay-out has made it possible to raise $I_{p,max}$ to 2 MA.

3.4 The importance of ß-limits on ASDEX Upgrade parameters

For sufficiently high heating powers the possible plasma parameters of a device are limited by the achievable ß-values. For plasma edge experiments, in particular the parameter β_p/A is of importance as it describes poloidal unsymmetries of the configuration and is expected to influence the observed unsymmetries in transport across the boundary. Average plasma parameters are then limited by geometry and plasma current through the expression

$$\langle nT \rangle = 2 \cdot 10^8 \frac{\beta_p I_p^2}{a^2} \frac{1}{(1+b/a)^2} \qquad (\text{mKs} - \text{Kev})$$

High values of n and T are of importance for the reactor-relevance of bulk plasma confinement studies where the discovery of the H-regime in ASDEX has underlined the potential of a divertor tokamak to make unique contributions. The ß-limited plasma parameters are however also important for edge physics studies, as they will determine the average temperature gradients in the high heating power regime. In this situation the temperature gradient $\langle T \rangle/a$ will scale like

$$\frac{\langle T \rangle}{a} \sim \frac{\beta_p}{\langle n \rangle} \frac{I_p^2}{a^3} \frac{1}{(1+b/a)^2} \sim I_p^2 \frac{A}{a^3} \frac{1}{\langle n \rangle}$$

Because of the high plasma current capability (a factor of 3 - 4 higher than ASDEX) the proposed ASDEX Upgrade design offers a nearly order of magnitude improvement in these parameters and approaches with $\overline{n}_e a \simeq 10^{20} m^{-2}$ and the average temperature gradient the absolute reactor (INTOR) needs. The β_{pol} limited bulk plasma parameters of ASDEX and ASDEX Upgrade are compared in fig. 4, which gives the obtainable average temperatures as a function of line density $\overline{n} \cdot a$, under the assumption that the Hugill critical density curve ($\overline{n}_{crit} \sim B_t/Rq$) can be arbitrarily surpassed. The permitted plasma temperatures at a given value of na will then be approximately a factor of 10 larger in ASDEX Upgrade than in ASDEX, where in the line-density regime $\overline{n}a \gtrsim 7.5 \times 10^{19}\ m^{-2}$ $\langle T \rangle$ would be limited to below 0.5 keV.

These average values imply of course also that the temperature gradients will be dramatically different. Evaluating, in fact the figure of merit $I_p^2 \cdot A / a^3$ describing the absolute average temperature gradient under the assumptions of equal density, β_{pol}/A and ellipticity we obtain a ratio of 10.5 : 10.4 : 1.6 between INTOR, ASDEX Upgrade and a (noncircular) ASDEX.

3.5 Comparison of the poloidal field requirements of divertor and pump limiter tokamak designs

Poloidal field design studies for INTOR carried out in parallel by Japanese, USA and European teams showed that external currents needed for producing a D-shaped (JET-like) plasma cross-section for use in a pump limiter device are only about 30 % less in total mega-ampere-turns [MAT] of the shaping and equilibrium currents than those required for an optimized, single null divertor configuration. An analysis of the reasons for this surprisingly small difference is given in appendix A.

The experience gained during our INTOR studies applied to ASDEX Upgrade allowed to design pump limiter and single null divertor configurations with essentially equal total MAT in the external windings. Compared to the double null option the reduction in MAT amounts to a factor ~3. The studies also showed such a single null configuration to be indeed by far optimum from the point of view of poloidal field requirements. This justifies to concentrate in ASDEX Upgrade on it and the corresponding pump limiter configuration. The relatively minor difference in the MAT requirements between divertor and pump limiter configurations also underline the reactor potential of the former.

3.6 Requirements for the suppression of CX-produced impurities in limiter and divertor tokamaks

Previously it had been shown /5/ from an analysis of ZEPHYR and JET simulation calculations that the suppression of the CX-impurity production and the conversion of conductive into radiative heat flow require a line-integrated electron density $2 \times 10^{15} cm^{-2}$ between the wall and the 300 ev isotherm for iron as wall material. From a more detailed analysis resulted that most of the sputtering originates at temperatures $\lesssim 50$ ev, with a line density of $4 \times 10^{14}\ cm^{-2}$ required between this isotherm and the wall for its effective suppression, approximately in agreement also with the findings of /6/.

This combination of parameters at the 50 ev isotherm implies an extremely low temperature gradient, which would be compatible with conductive throughput of the energy flux envisaged for INTOR ($\sim 30\ W/cm^2$) only for an unconceivably high value of $_e$ in this region ($600\,000\ cm^{+2}s^{-1}$ for $n_e = 5 \times 10^{13}\ cm^{-3}$). In order to allow such a flat temperature profile the major part of the energy flux has therefore to be passed through another channel. In the classic cold plasma mantle situation this is achieved by conversion into radiation in a layer (photosphere) which for medium Z impurities like iron will extend inward approximately to the 300 ev isotherm and - for radiating a power flux like given above - comprise a region of $\approx 2 \times 10^{15}\ cm^{-2}$ line density.

In the divertor case the energy flux to the walls is attenuated by the parallel heat flow into the divertor chamber, and a sufficiently thick cold plasma layer might form in the scrape-off. First calculations suggested a similar situation to apply to the shadow of a protruding limiter; it has only recently been recognized that in the case of a large-area limiter, <u>CX-sputtering will predominantly happen on the exposed surface of the limiter</u>, as the recycling will be concentrated in this region and the probability of fast CX neutrals to hit again the limiter will be large. Thus - contrary to the situation for narrow poloidal limiters and divertors - the line-density in the shade of a large area limiter does not contribute significantly towards a reduction of charge exchange sputtering, and formation of a cold plasma mantle in front of the limiter by radiation still seems the only chance for the suppression.

For the pump limiter configuration in ASDEX Upgrade a boundary layer line-density of $\int ndl \approx 2 \times 10^{15}$ cm^{-2} between the limiter surface and the 300 ev isotherm remains therefore a necessary condition for suppression of CX-sputtering. Expected to originate mainly from the exposed limiter edge rather than the vessel walls, the produced impurities will also be more dangerous, as there is no possibility for their ionization and removal by the limiter scrape-off.

In the divertor case, suppression of the CX-production of impurities will primarily depend on the possibility of establishing a sufficiently thick scrape-off layer between the separatrix (which in the absence of radiative losses is expected to have temperatures in the range of 100 ev) and the vessel wall. The plasma requirements originating from this condition are further discussed below.

3.7 Plasma parameter requirements for an INTOR-similar divertor scrape-off

Experimental investigations on ASDEX and Doublet III, and theoretical studies in the INTOR framework have identified a scrape-off with high recycling in the divertor chamber as the most promising one from the point of view of energy removal, impurity control and helium pumping. Assuming the probability for a neutral particle born at the target plate to be re-ionized before leaving the target chamber to be R_{DIV}, localized recycling will enhance the charged particle flux to the target plates by a factor of $1/(1-R_{DIV})$. As the plasma temperature at the sheath entrance will be determined by the condition: energy flow = $\gamma kT \cdot$ ion flow (γ typically 8), this flux amplification will reduce the particle temperature by the same factor. For high enough recycling - when γkT starts to fall into the neighbourhood of 30 eV - hydrogen ionization and radiation will moreover become significant energy loss channels. For the divertor temperatures observed in the high density, high injection power regime in ASDEX ($\lesssim$10 eV) the sputtering and self-sputtering production of impurities becomes very small while the increased hydrogen flux reduces furthermore their chance of reaching the bulk plasma.

In a divertor with close coils - like ASDEX - the strong field line compression in the divertor throat allows to use very narrow entrance slits. This enforces a long neutral particle residence time in the divertor chamber, irrespective of the scrape-off plasma parameters. Reaching a regime of high recycling will be significantly more difficult in an open divertor chamber, like necessitated by a design with distant coils where neutrals born at the target plates have a high probability of reaching the bulk plasma after having transversed once the scrape-off. For an effective flux amplification (desired are about two orders of magnitude in INTOR) the scrape-off in the target plate vicinity has to be of sufficient line density to stop the major fraction of neutrals within one transit. Obviously once initiated, there will be a strong avalanching effect, as the re-ionized neutrals will add to the plasma line density and thereby - at least as long as the temperature stays above 10 eV - increase further the re-ionization probability.

A figure of merit for the probability of an open divertor scrape-off layer to develop into this "opaque" solution will be the scrape-off thickness expected <u>in the absence</u> of divertor recycling. This can be estimated by equating the charged particle flux across the separatrix

$$\phi_s = 2\pi^2 (a+b) R \cdot D \cdot \frac{n_b}{\lambda}$$

with the flow onto the target plates

$$\phi_t = 2\pi R \int n\,dl \cdot v_{\parallel} \frac{B_p}{B_t} \cdot \nu$$

where D is the diffusion coefficient near the plasma edge, λ the average deposition depth of neutrals replenishing the plasma, v the plasma flow velocity in the scrape-off, ν the number of active divertor throats (experimentally $\nu \approx 2$ for both single and double null divertor) and B_p/B_t the pitch of the field lines near the target plates. For n_b one should insert the density of the bulk plasma at the surface where most of the neutrals have been stopped; n_b will therefore be, depending on the existence of an inward drift term, approximately equal or smaller than the average plasma density.

At the target plates the flow will be sonic, and the power flow onto the target plates

$$P_H(1-f) = p_H \cdot 2\pi^2 R\ (a+b) \cdot (1-f)$$

(P_H the heating power and f the radiated fraction) will be related to the particle flow by

$$\gamma \cdot kT \cdot \phi_t = P_H\ (1-f)$$

These solutions can be combined to

$$\int n\,dl \approx n_b^3 \frac{R}{q} \frac{\pi D^{3/2} m_i^{1/2} \gamma^{1/2}}{4 N_L^{3/2} [p_H(1-f)]^{1/2}}$$

(for $1/2\ n_b = N_L$ - the numerical value of the Lehnert criterium - and $B_p/B_t \sim a/(qR)$).

Applying the standard ASDEX-diffusion model ($D = 0.4\ m^2/s$ - density independent - and an inward drift term leading to $n_b \sim 2\ \bar{n}$) to the ASDEX UP device as proposed here leads to a value of $\int ndl = 2 \times 10^{17} m^{-2}$ for $\bar{n} = 2 \times 10^{20}$, and the same numerical value for INTOR with $\bar{n} = 1.4 \times 10^{20}$. This should be adequate to cause flipping over to the dense solution. More important probably than the numbers given there is the observation that the expected scrape-off line-density should be a much stronger function of density than of geometrical dimensions vindicating the direction of parameter changes between ASDEX and ASDEX Upgrade.

Recycling in the divertor chamber also raises the midplane scrape-off by reducing the Mach number of the flow in front of the recycling zone. One-dimensional (radial) simulation caclculations for devices with large enough dimensions and densities (INTOR, JET, ASDEX Upgrade) including this effect in a simplified form then tend to develop into a regime where the refuelling flux from gas-puffing becomes nearly completely absorbed in the scrape-off layer. The density of the bulk plasma in this case is maintained exclusively by diffusion or drift-type inwards-flow from this region. The resulting thick midplane scrape-off then also tends to suppress the CX production of impurities from the walls, in spite of separatrix temperatures in the range of $\gtrsim 100$ eV. Obviously this is another extremely important question to study in ASDEX Upgrade: although divertor tokamaks could probably live with CX production of impurities as already scrape-off layers with $\int ndl \ll 10^{18}\ m^{-2}$ will suffice to ionize and guide them into the divertor chamber, their suppression would certainly facilitate design of the first wall.

3.8 The role of helium pumping in determine possible scrape-off-regimes of a reactor

The need for helium-ash removal in INTOR and the reactor introduces a new criterium for the pumping requirements, which otherwise have been only determined by the desired degree of plasma density control. The helium concentration establishing itself in a stationary, ignited state depends on

- helium vs. energy transport in the bulk plasma

- helium and hydrogen ion flow in the scrape-off
- the neutral particle dynamics including charge exchange and ionization processes for helium and hydrogen in the target chamber
- and the available pumping speed for helium and hydrogen neutrals.

A simple estimate, based on similarity between energy and the transport of thermalized α-particle transport and the respective source distributions gives the average concentration in a stationary burning plasma as

$$\frac{\langle n_{He} \rangle}{\langle n_e \rangle} = \frac{\chi_e}{100\ D} + \frac{n_{He}(a)}{\langle n_e \rangle}$$

yielding - for the INTOR transport convention - 4 % even in case of a vanishing helium boundary density $n_{He}(a)$. The situation would be aggravated if models giving a better fit to experimental hydrogen profiles (lower values of D or inward drift terms) were found to apply also to helium. Differences between helium and hydrogen particle flow in the scrape-off may occur in strong recycling situations where non-monotonic potential distributions along the field lines may arise.

The available pumping speed and the helium production rate will determine the neutral helium density in the divertor chamber. The actual density of neutral hydrogen isotopes will depend on the form of refuelling, recycling out of and outside the divertor chamber and eventual de-mixing processes in the scrape-off.

A crude estimate can be given however by dividing the helium density by its tolerable concentration at the plasma edge, where account has to be taken of the fact that $\langle n_{He} \rangle / \langle n_e \rangle$ will always be larger than $n_{He}(a)/n_e(a)$ because of the different birth profile for helium and hydrogen ions. Estimates for INTOR conditions give - for assumed pumping speeds of 2×10^5 l/s for helium and D-T and a production of 2×10^{20} α-particles/sec - hydrogen molecule densities of 2 x 10^{19} m^{-3} for a permitted $n_{He}(a)/n_e(a) = 10^{-2}$ (correspon-ding - for an INTOR transport model - to $\langle n_{He}(a)\rangle / \langle n_e \rangle = 5 \times 10^{-2}$).

The compatibility of such values with pump limiter and divertor solutions have to be examined, particularly in view of the statements made above concerning the observations of energy flux limits on the achievable neutral densities.

The importance and the uncertainties of helium transport in the plasma and its dynamics in the scrape-off region and divertor chamber have also convinced us to include a pumping system with helium capability into the design of ASDEX Upgrade, and to consider the possibility of injection of helium atoms by a modified neutral injection system (a few 100 kW at 25 keV) at a later stage.

4. Technical Concept Update

Untill recently all conceivable options were still investigated in comparison which could possible meet the aims, or part of the aims, described in Sect. 2, although with different effort. These options were:

- the normal conducting reference system (ASDEX UG)
- a superconducting concept (ASDEX UG SL)
- an ASDEX reconstruction.

All these different concepts were worked out at least to such a degree that they could be valuated with respect to the following criteria:

- are poloidal field configurations and the attainable parameter regime suited to reach the aims
- is the technical concept simple and compatible with reactor requirements
- is the concept assured by the present technology
- are the costs sufficiently low
- can the project be constructed compatible with the IPP personnel, budget and time schedule.

4.1 The ASDEX UG Reference System

The reference system was from onset designed in order to meet the aims by using proven technology. However,
the necessary effort in the poloidal field system produced by the double null divertor requirements in conjunction with the power supply limitations of the IPP forced to consider compromises like internal equilibrium field coils and a two step experimental programme, the second step employing a new 1.5 GJ fly wheel generator.

Fortunately the finding of single null divertor configurations with strongly reduced poloidal currents relieved from this situation and allowed to decide the following simplifications:

- to put all poloidal field coils outside the TF-magnet
- to use coils symmetric in position and cross-section above and below the tokamak midplane. The coil cross-sections are designed for full current (1.2 MA) double null divertor operation and 20 MA/m^2 in order to get the full benefit of ohmic power loss reduction for single null operation

- to adjust the ASDEX UG programme to the IPP-Garching energy supply facilities.

The chosen poloidal field coil arrangement allows now to produce single null and double null divertor configurations and equally elongated limiter controlled configurations (fig. 5). The attainable combinations of plasma current and toroidal field is determined by the combined stresses in the TF coil structure.

The operational regime thus available is described in the next section. Placing all PF coils outside the TF-magnet and retaining the shape of the TF-coils allowed to enlarge the vacuum vessel radially by about 0.1 m at the outer contour. This brought more space for installations e.g. ICRH antenna and pumping divertor and limiter structure. Fig.6 shows the ASDEX UG reference system design in a vertical cross section.

4.2. The Superconducting ASDEX UG SL Concept

This concept was investigated in cooperation with experts from the Kernforschungszentrum Karlsruhe. The superconductor technology determined current densities necessary support structure dimensions and cryostat requirements, data which were used together with the ASDEX UG dimensional and parametrical requirements as imput for the optimization code TOBASCO.

Two superconducting options were originally persued characterized by their current density in the conductor windings:

- 35 MA/m^2 for advanced superconductor technology (Nb_3Sn), advanced cooling technique (superfluid He) or adiabathical stabilization
- 25 MA/m^2 concentional superconductor (NbTi) cryogenically fully stabilized.

The systems which resulted from optimization calcualtions are described by the data of Table I . For comparison those of the ASDEX UG reference system and the LCT project are also shown. Both superconducting systems are larger in size and magnetic energy because of the reduced average current densities in the magnet throat and the increased OH flux swing requirements due to $\frac{dI_p}{dt}$ limitations by the superconductors.

The advanced concept (35 MA/m^2) was descarded mainly be the uncertainty in the development time necessary for the quality assured superconductor. The conventional (25 MA/m^2) concept was dispensed with because of high investment costs ($\approx$ 2 x ASDEX UG) and increased complexity compared with the reference system. The current density averaged over the magnet throat is more descriptive than the superconductor values. They amount to 15 MA/m^2 for the advanced and 13 MA/m^2 for the conventional concept. The fact that the technological development is anyway assured by the LCT project contributed to this decision.

However, the development of superconducting poloidal field coils was regarded as a challenging task. Since the poloidal field coils of the reference system are exchangeably designed the option of using superconducting PF coils in a late phase is kept open.

4.3 ASDEX Modification

The ASDEX reconstruction using the ASDEX TF coils but exchanging vacuum vessel and OH and PF coil systems was already discussed in the ASDEX UG report /1/.

The high costs, the reduced operation time of ASDEX the technical risk of overloading the ASDEX TF coils by 20 % and the reduced parameter regime compared with the reference system proved this solution as not attractive enough.

Still the question was asked what ASDEX itself could achieve with respect to the aims of ASDEX UG by retaining vacuum vessel and OH coil system but altering the multipol coil system inside the vessel. It is easy to show that such a concept fails to meet the ASDEX UG requirements in essentially all respects. The poloidal divertor geometry is not sufficiently similar to those of INTOR or a reactor. The attainable plasma parameter regime determined by the attainable plasma current can not reach sufficient line density values and plasma temperatures even if the Murakhami or Hugill density limits are surpassed. The discussion in the next Section explains this in greater detail.

4.4 The heating of the ASDEX UG and helium injection

The heating method of ASDEX is not subject to this application for preferential support. The ICR heating system is under design in cooperation with the ICRH project group. The generator development contract for ASDEX and WVII covers also the ASDEX UG development needs at the uper limit of 115 MHz. The antennas and their protective structure are presently conceptually designed and calculated in order to define and initiate the necessary development programme.

The presently discussed fusion technology programme of EURATOM foresees an ECRH generator development up to 100 GHz. This very frequency equals the first harmonic electron cyclotron frequency at B_o = 3.6 T. The ECR heating scenarios are presently discussed in order to provide for their use in case the clystron development offers generators in due course.

Helium can be injected for the purpose of proving the He pumping capability through the divertor. Since the trapping cross-section of He varies weakly with energy only an acceptable central deposition can be obtained with an ASDEX beam line. The required He flux of 2×10^{20} sec^{-1} can be obtained by one beam line with 3 periplasmatron sources operating at 55 keV ($P_{He} \simeq$ 1 MA). The only major investment needed are He pumps with 10^5 l/sec pumping speed.

For heating and injection systems full power operation for up to 10 sec is required which is equivalent to a steady state layout.

5. The Operational Regime of ASDEX UG

The dominant role of the plasma current for reaching high temperatures and densities was discussed in Section 3. The plasma current I_p is thereafter the most important design parameter. A sufficiently large toroidal field B_o has to provide safe tokamak operation ($q_a > 2$).

The tokamak machine parameter combination of I_p and B_o are usually limited by mechanical stresses in the coil structure. In ASDEX UG bending stresses produced by the divertor vertical field components in the TF coils lead in conjunction with the tensile stress to an highly nonlinear dependence of the admissible plasma current as function of the toroidal field. In fig. 7 the stress limiting curve in the I_p, B_o plane is shown for single null divertor operation. Also shown are curves for constant discharge duration as determined by the IPP energy supply. Single null divertor discharges with I_p = 2 MA and B_o = 3.25 T (q_a = 2.11) can be sustained for 5 sec. Double null divertor discharges can still reach I_p = 1.5 MA for 5 sec.

A comparison of the attainable plasma parameters as determined by the plasma current and the permissible poloidal beta ($\beta_p = \frac{A}{2}$ or $\sqrt{A}$, whatever is larger) is shown in fig. 4 for ASDEX UG and ASDEX. The solid lines give the attainable temperature for varying discharge current at the critical line density (Hugill). The dashed lines give the attainable temperatures at maximum plasma current (2 MA for ASDEX UG, 0.5 MA for ASDEX) for varying density, i.e. deviation from the critical density. Even if ASDEX exceeds the Hugill density limit and achieves line densities as required for a dense plasma boundary the attainable temperature becomes too low ($\langle T_e + T_i \rangle < 0.5$ keV).

Different tokamaks may be compared again by their attainable temperature at the β_p limit plotted versus the line density $\bar{n}_M \cdot a$ at the Murakhami limit. The latter is the more suited density parameter for this case because of the different dimensions of the experiments. fig. 8 shows all major tokamaks as taken from ref. / 7 /. Although ASDEX UG differs in dimensions little from ASDEX it nevertheless is a large tokamak with respect to the attainable parameters and more comparable to TORE SUPRA, TFTR, JT60 and even JET. ASDEX itself can not nearly reach the parameter regime in order to investigate reactor relevant plasma boundary physics even not if the current is raised to 0.7 MA which the vacuum vessel can eventually stand in case of a disruption. It was the object of the ASDEX UG design to reach with minimized effort sufficiently high I_p, T, n in a reactor relevant configuration in order to investigate the above defined aims.

6. Summary of the ASDEX UG Design Data

6.1 Basic Data

Type of device: axisymmetric divertor tokamak with normal conducting water cooled coil systems. All poloidal field coils positioned outside the TF magnet coils.

Plasma current	I_p	$\leq$ 2 MA
Major plasma radius	R_o	1.6 m
Horizontal plasma diameter	2a	1.0 m
Vertical plasma dimension	2b	1.6 m
Aspect ratio	A	3.25 m
Toroidal field strength	B_o	$\leq$ 3.9 T
OH-flux swing	$\Delta\phi$	9.5 Vs
Pulse length	t_{FT}	$\geq$ 5s
Break between pulses	t_p	$\leq$ 10 min
Heating power (ICRH)	P_H	12 MW
Average wall load	P_w	0.3 MW/m^2

6.2 Modes of operation

Single null divertor:

I_p = 2 MA /1.5 MA
B_o = 3.25 T /3.9 T
t_{FT} = 5 s

I_p = 1.7 MA
B_o = 2.5 T
t_{FT} = 10 s

Double Null divertor:

I_p = 1.2 MA
B_o = 3.25 MA
t_{FT} = 5 s

Pumping limiter ($b/a \simeq 1.6$):

I_p = 2 MA
B_o = 3.25 T
t_{FT} = 5 s

Long pulse current drive:

$$I_p = 0.5 \text{ MA}$$
$$B_o = 1 \text{ T}$$
$$t_{FT} \simeq 2 \text{ min}$$

Design requirements:

full load pulses	N_{100}	$= 10^4$
half load pulses	N_{50}	$= 10^5$
hard full load disruptions	N_D	$= 10^3$

References:

/1/ ASDEX UG, Definition of a tokamak experiment with reactor compatible poloidal divertor, IPP-Garching report 1/197, March 1982

/2/ M.Keilhacker et al., 9th Conf. Plasma phys. and Contrl. Nucl. Fusion Res., Baltimore (1982), paper R-2

/3/ R.Chodura et al., ibid, paper D-3-1

/4/ F.Wagner et al., ibid, paper A-3

/5/ K.Lackner, J.Neuhauser in M.Keilhacker, U.Daybelge (editors) Proceedings IAEA Technical Committee, Meeting on Divertors and Impurity Control (Garching 1981) pg. 58

/6/ F.Engelmann and M.Tessarotto, INTOR Phase One, European Contributions, vol. 2, pg. 354 (EUR FU BRU/XII-132/82/EDV2)

/7/ World Survey of Major Facilities in Controlled Fusion Research, 1982 edition, IAEA Vienna 1982

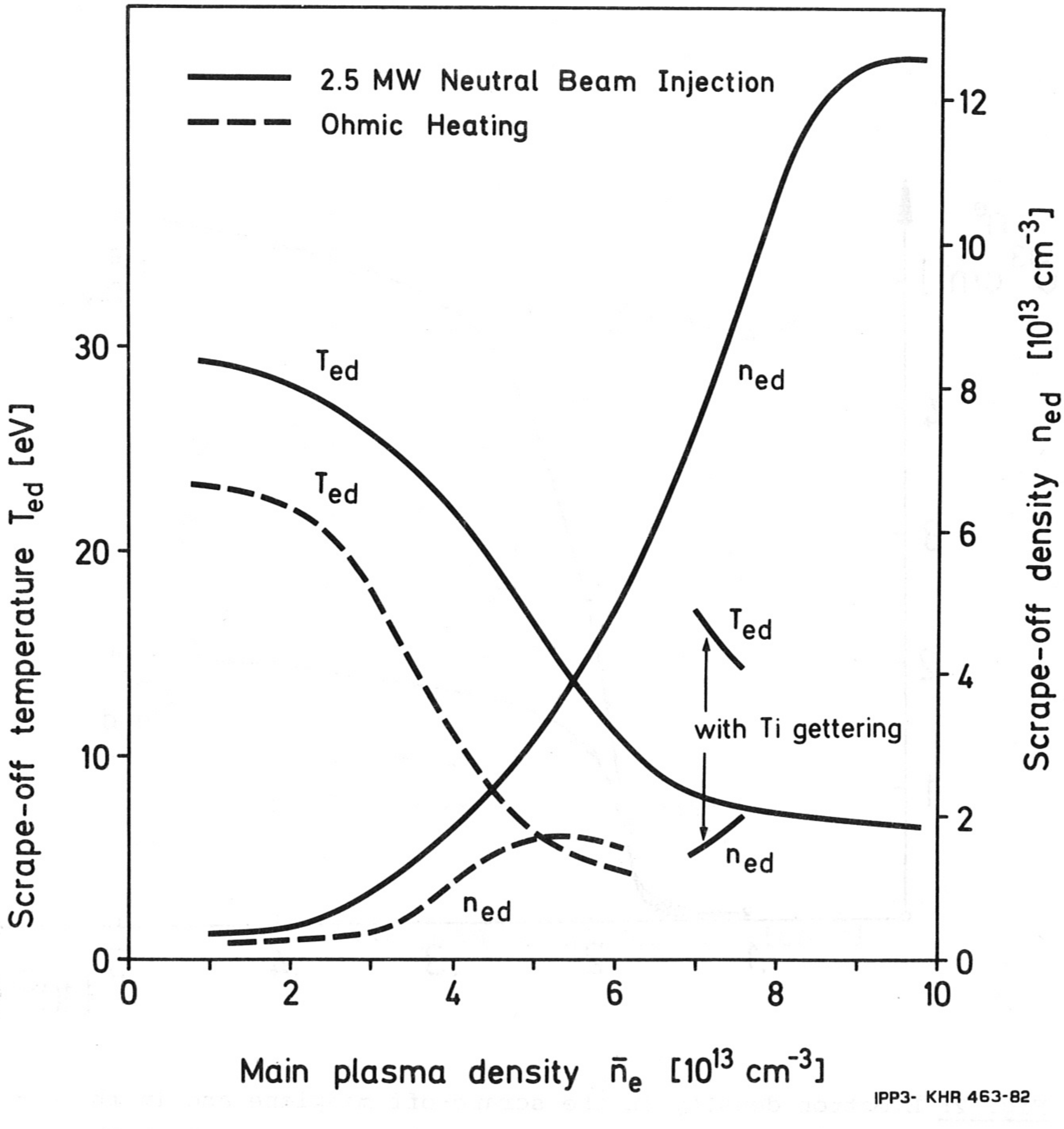


Fig. 1: Saturation of scrape-off density in dependence on the bulk plasma density and heating power.

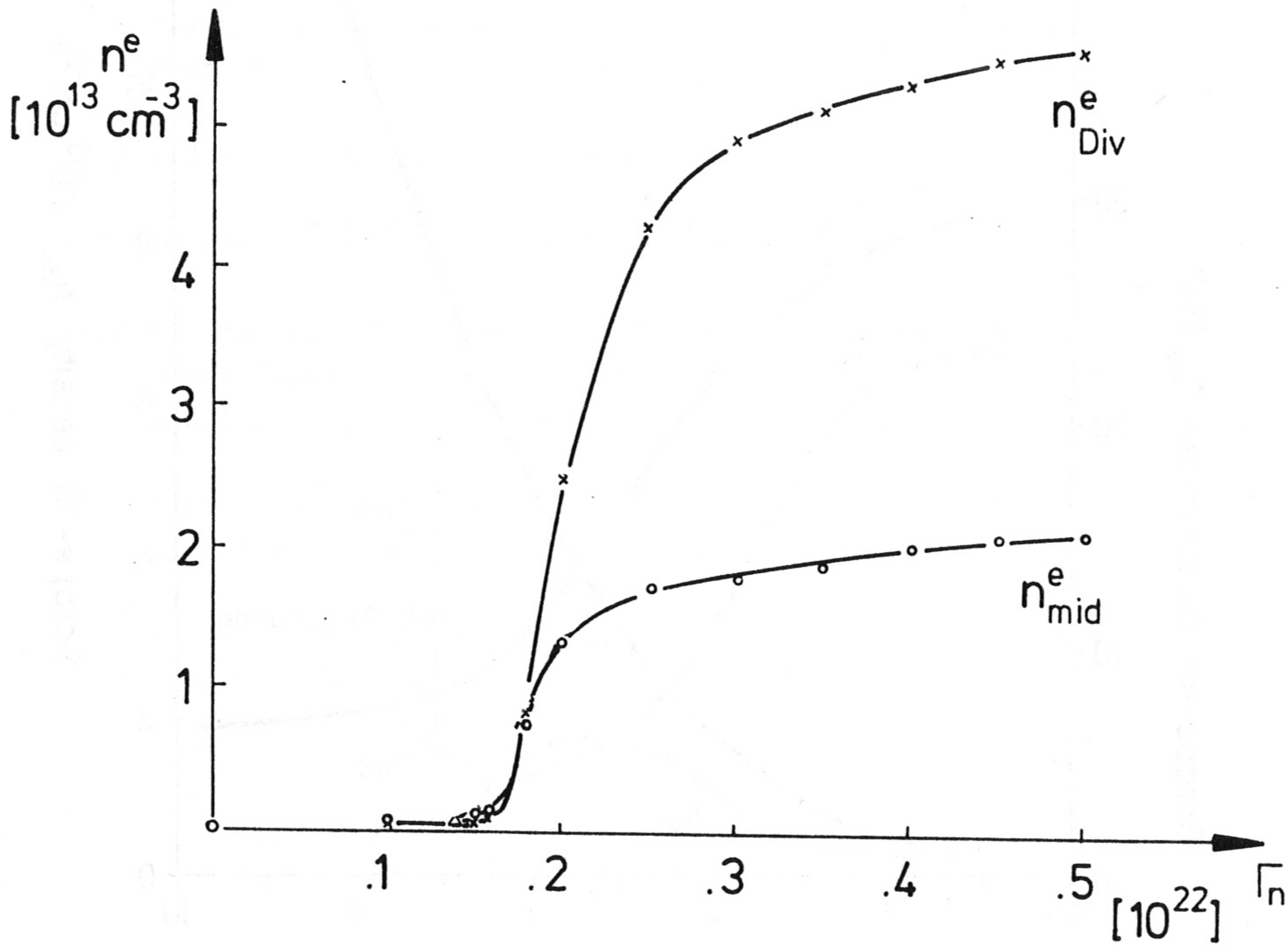


Fig. 2: Electron density in the scrape-off midplane and in the divertor as function of the particle flux from the bulk plasma Γ_n, calculated with a one dimensional flow model.

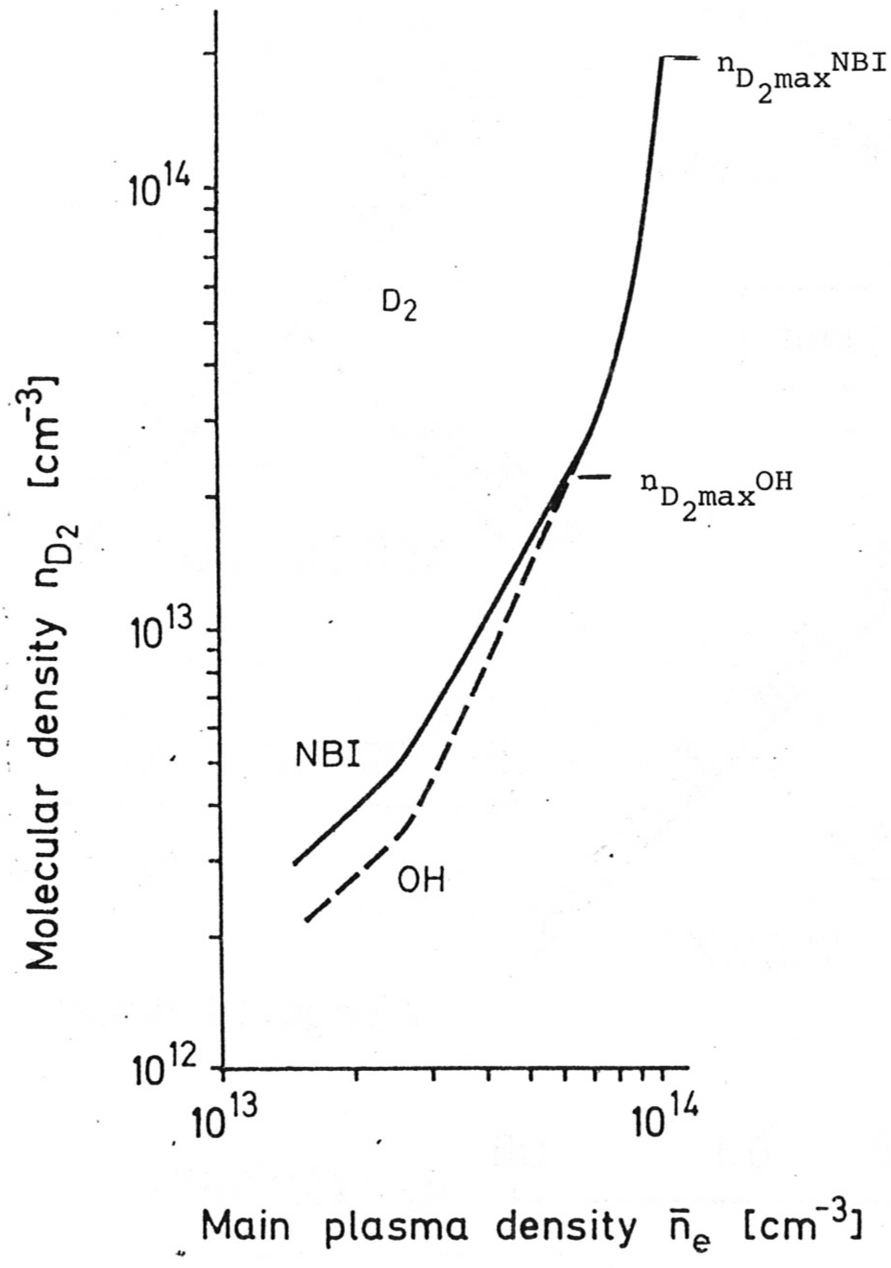
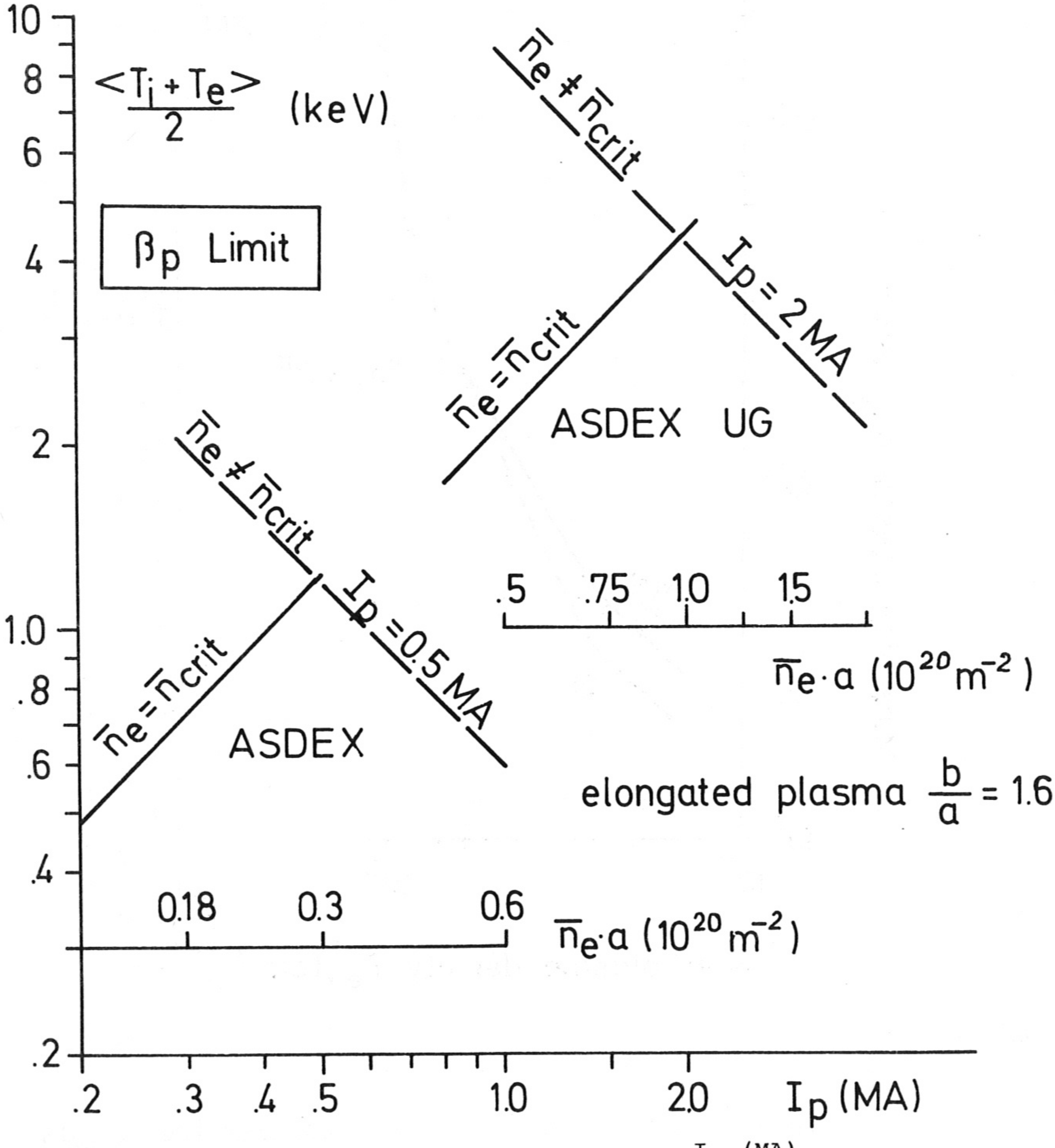
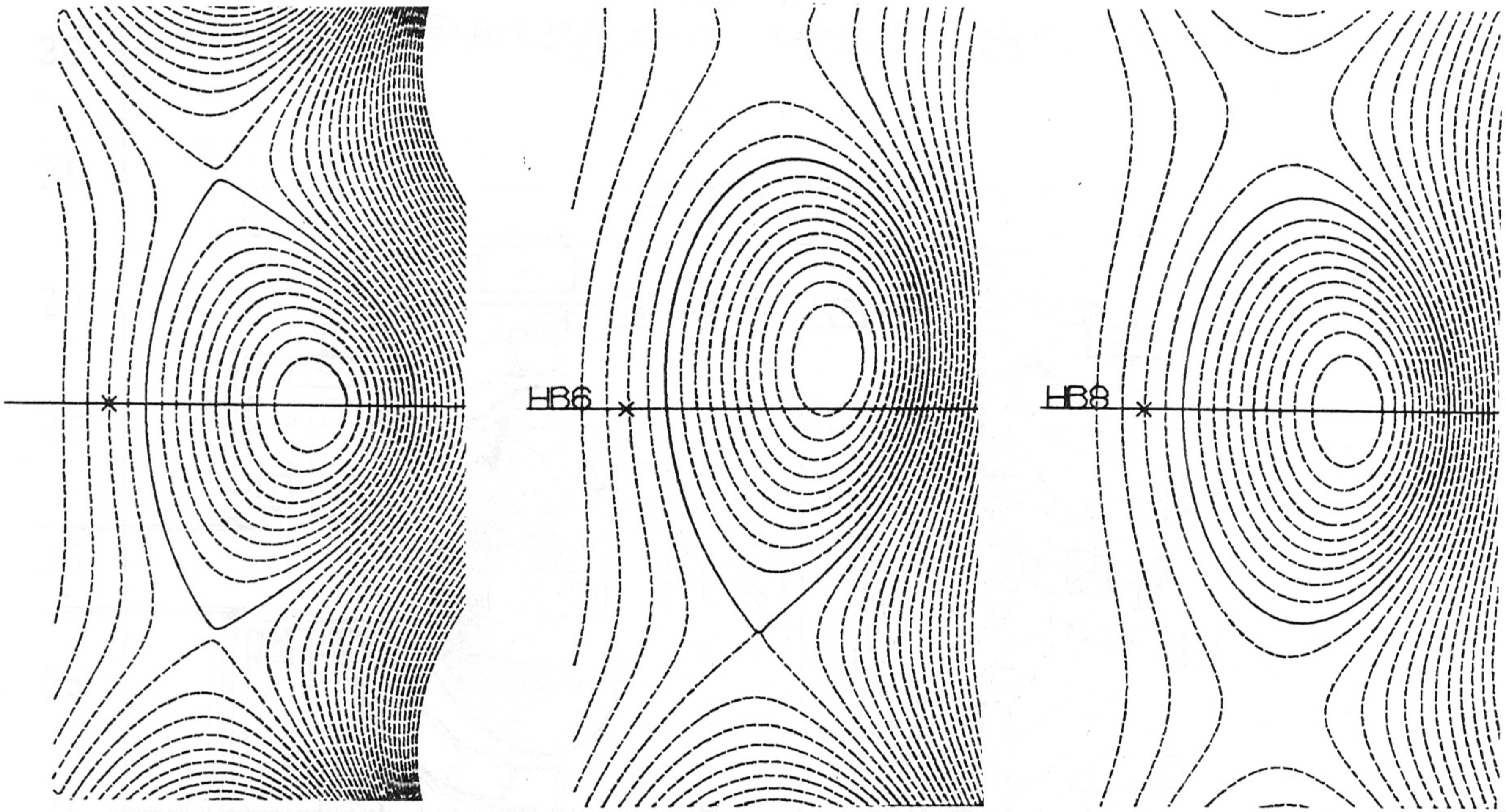


Fig. 3: Neutral gas density increase in divertor chamber with pulk plasma density.



$$\bar{n}_{crit.} = 1.65 \cdot 10^{20} \frac{I_p \text{ (MA)}}{a^2 (1+b/a)^2}$$

Fig. 4: Attainable temperatures at the β_p-limit ($\beta_p = \sqrt{A}$) for ASDEX and ASDEX UG at the Hugill density $n_{crit.}$ limit (solid line) for different plasma currents. The dashed lines give the temperature for fixed $I_{p\ max}$ but $n_e \neq n_{crit}$.



$\Sigma|I_m| = 14\,MA$ $\Sigma|I_m| = 5\,MA$ $\Sigma|I_m| = 3.6\,MA$

Fig. 5: Double null, single null divertor and limiter controlled configurations for ASDEX UG produced by PF-coils located outside the TF-magnet. The sum of PF-coil currents for I_p = 1 MA is also given for each configuration.

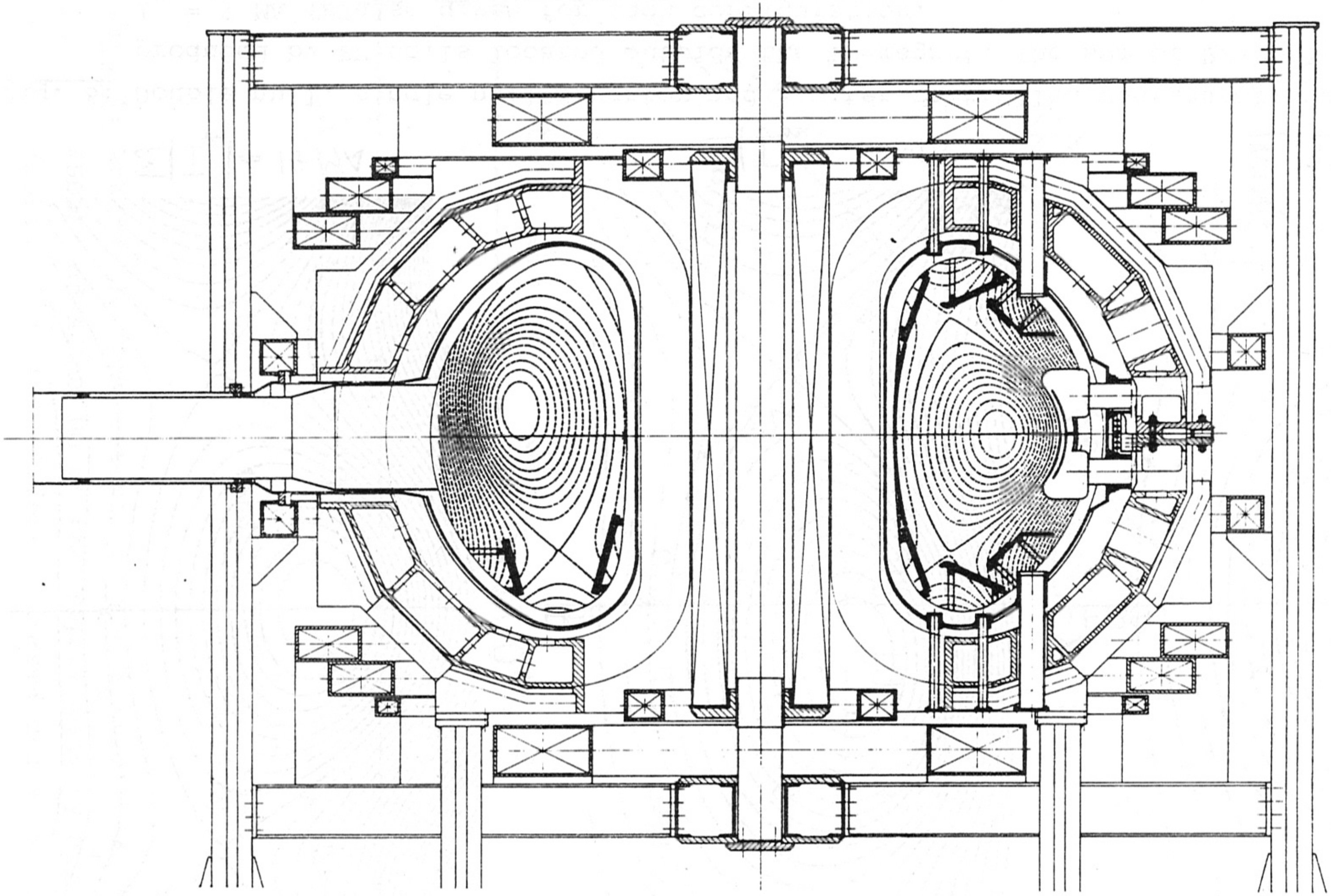


Fig. 6: The ASDEX UG tokamak system with a double null (right) and a single null divertor (left) configuration.

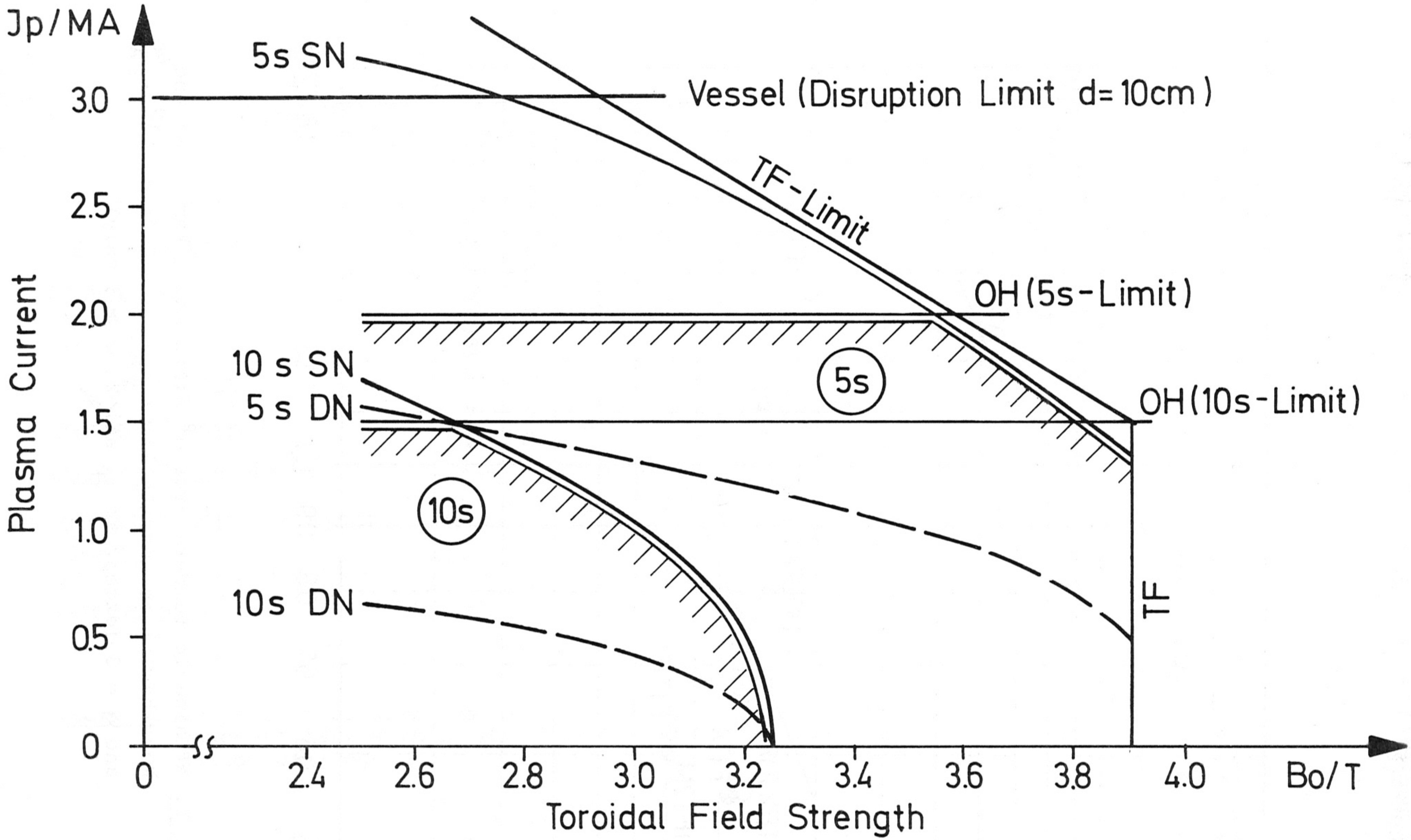


Fig. 7: The operational regime of ASDEX UG limited by coil stresses (TF-limit) and OH flux limit. Curves for 5 and 10 s discharge duration for single null (SN) and double null (DN) divertor are also shown.

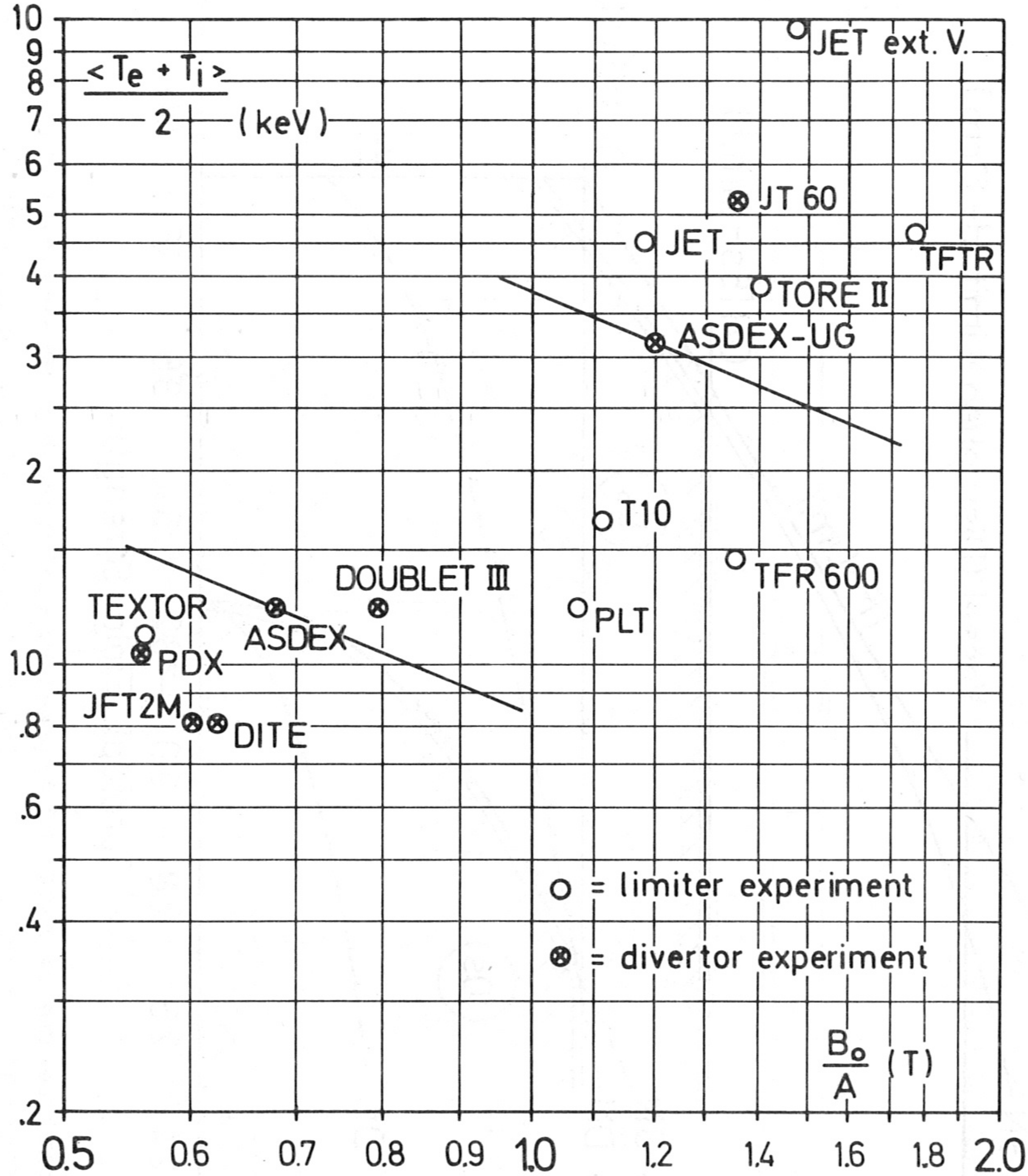


Fig. 8: Attainable temperatures of different tokamaks at the Murakhami line density limit $n_M \cdot a = 6.25 \cdot 10^{19} \cdot \frac{B_o}{A}$ and q = 3 (except if I_p allows $q > 3$ only).

ASDEX UG SL

		NL	SL (25 MA/m^2)	SL (35 MA/m^2)	LCT (EURATOM)
Small plasma radius	(m)	0.5	0.5	0.5	--
Aspect ratio		3.25	3.98	3.7	--
Field on axis	(T)	3.9	4.78	4.44	--
B_{max} at conductor	(T)	7.33	8.41	8.3	--
Stored TF energy	(MJ)	415	880	650	800
Amount of conductor	(MA·m)	293	478	397	410
Current density in winding	(MA/m^2)	28	26	35	28
Current density in throat	(MA/m^2)	20	13	15	14
Heating power	(MW)	12	15	14	--

Table 1: Comparison between the reference system (NL) and two superconducting concepts (SL) with 25 MA/m^2 and 35 MA/m^2 current density in the conductor windings.