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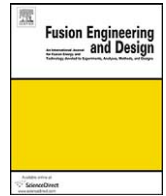
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Feasibility study of an actively cooled tungsten divertor in Tore Supra for ITER technology testing

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ABSTRACT

In order to reduce the risks for ITER Plasma Facing Components (PFCs), it is proposed to equip Tore Supra with a full tungsten divertor, benefitting from the unique long pulse capabilities, the high installed RF power and the long experience with actively cooled high heat flux components of the Tore Supra platform. The transformation from the current circular limiter geometry to the required X-point configuration will be achieved by installing a set of copper poloidal coils inside the vacuum vessel. The new configuration will allow for H-mode access, providing relevant plasma conditions for PFC technology validation. Furthermore, attractive steady-state regimes are expected to be achievable. The lower divertor target design will be closely based on that currently envisaged for ITER (W monoblocks), while the upper divertor region will be used to qualify the main first wall heat sink technology adopted for the ITER blanket modules (CuCrZr copper/stainless steel) with a tungsten coating (in place of the Be tiles which ITER will use). Extended plasma exposure will provide access to ITER critical issues such as PFC lifetime (melting, cracking, etc.), tokamak operation on damaged metallic surfaces, real time heat flux control through PFC monitoring, fuel retention and dust production.

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1. Introduction

Since its inception, Tore Supra has played a unique role in the development of reliable actively cooled plasma-facing components (PFCs) based on carbon [1], enabling long pulse operation with multi-megawatt heating and current drive input powers [2]. The current ITER baseline foresees the use of a full tungsten (W) divertor for the nuclear phase [3]. However, the required high heat flux technology has never been tested in the demanding environment of a tokamak under the steady state plasma heat fluxes ($\sim 10 \text{ MW m}^{-2}$) expected in ITER. Moreover, the fabrication time of a full ITER

divertor is extremely long (of order 8 years) such that it will not be possible to learn much from the first CFC/W divertor, to be installed in the non-active phases, before the full-W variant must be installed.

In order to reduce the risks to be faced in the design, fabrication and operation phases associated with such wide scale deployment for the first time on ITER, a feasibility study was initiated to study the relevance of equipping Tore Supra with a full-W divertor, benefitting from the unique long pulse capabilities of the Tore Supra platform, the high installed power and the long history of experience with actively cooled high heat flux components at the IRFM.

The divertor configuration should allow access to H-mode, providing relevant plasma conditions for PFC technology validation. Furthermore, with the installed RF power (15 MW), attractive steady-state regimes are expected to be achievable.

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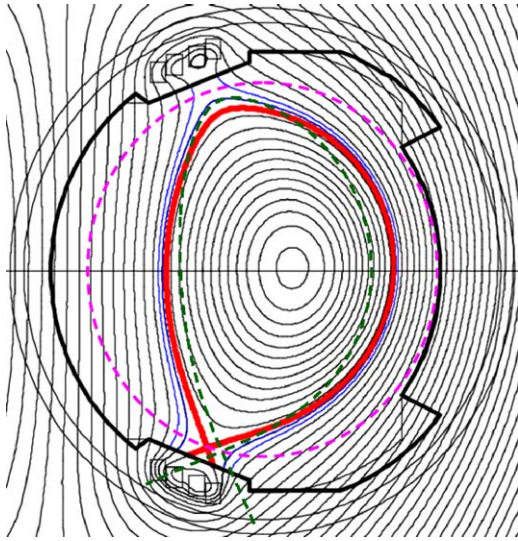


Fig. 1. Plasma equilibrium in divertor configuration; also shown limiter (CIEL in pink dashed line) and ITER (rescaled in green dashed line) last closed flux surfaces. PFC positions are only indicative. (For interpretation of the references to color in this figure legend, the reader is referred to the web version of the article.)

2. New magnetic configuration

The transformation from the current circular limiter geometry of Tore Supra (TS) to the required X-point configuration will be achieved by installing a set of copper poloidal coils inside the lower and upper parts of the vacuum vessel (Fig. 1).

A wide range of plasma equilibria will become accessible, from lower single null to upper single null passing through double null geometries. The coil symmetry also allows an adequate matching between plasma and RF antennas shapes to be maintained. Plasma volume will be reduced by 40% compared to the CIEL limiter configuration (from 25 m³ to 15 m³), while the maximum plasma current will be I_p ~ 1 MA (q₉₅ ~ 2.5). The main parameters of the divertor configuration are given in Table 1.

The high aspect ratio (A = R/a) is complementary to those of existing tokamaks and should therefore allow an interesting expansion of the current tokamak physics databases.

3. Plasma scenarios

3.1. H-mode access

The divertor configuration will allow access to the H-mode regime. According to the scaling laws used for ITER [4], at B_T = 3.8 T and n_e = 10²⁰ m⁻³, the predicted H-mode power threshold is around

Table 1
Tore Supra main parameters with tungsten divertor^a possible upgrade of the ECRH system (110 GHz gyrotrons).

| Parameter | Attributes |
|--|--|
| R/a = A | 2.5 m/0.5 m = 5 |
| κ | 1.3–1.6 |
| δ | 0.5–0.6 |
| V _p | 15 m ³ |
| B _T | 3.9 T |
| I _p | 1 MA |
| q ₉₅ at I _p of 1 MA | 2.5 |
| n _{CW} at I _p of 1 MA | 1.5 × 10 ²⁰ m ⁻³ |
| P _{LH} | 6 MW |
| P _{ICRH} | 9 MW |
| P _{ECRH} | 0.6 MW (6 MW) ^a |
| t _{nat-top} at I _p of 0.7 MA | 1000 s |

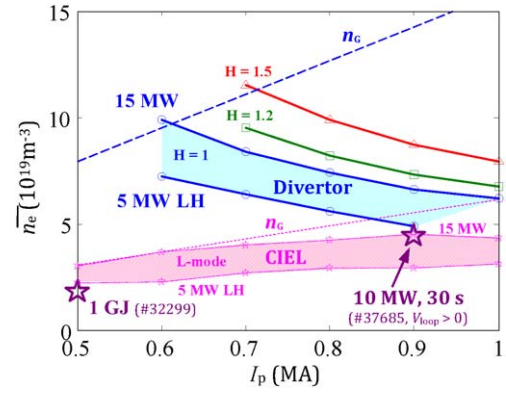


Fig. 2. Tore Supra non-inductive (V_{loop} = 0) operational diagram {I_p, n_e} as a function of the injected power (LH + ICRH) and the H-factor in the present limiter (CIEL) configuration (in pink) and the foreseen divertor configuration (in blue). 1 GJ (3 MW LH, 6 min) and 10 MW, 30 s (V_{loop} ~ 0.27 V) discharges are shown (star). (For interpretation of the references to color in this figure legend, the reader is referred to the web version of the article.)

5 MW. Given the RF power already installed on TS (15 MW), a Type I ELMing regime with β_N > 2 should be achievable with comfortable margins.

3.2. Non-inductive operation

The main asset of Tore Supra is its long pulse capability. As illustrated in Fig. 2, installation of a divertor will enhance its fully non-inductive operational domain. This diagram has been obtained using the fast integrated simulation code METIS, mixing 0D scaling laws and 1.5D equations (fast current diffusion solver). Compared to L-mode operation in limiter configuration, operation in H-mode results in higher plasma electron temperature and thus higher LH current drive efficiency. Moreover, the improved confinement and the existence of the edge pedestal in H-mode provide significant bootstrap fractions. High densities are now achievable which is essential for operation with a tungsten divertor. According to these simulations, I_p = 600 kA could be fully non-inductively driven at the Greenwald density ~ 10²⁰ m⁻³, i.e. ITER reference density (assuming H₉₈ = 1).

More detailed studies have been performed using the CRONOS suite of codes [5]. An example is given in Fig. 3 at I_p = 650 kA and 9 MW of ICRH. The transport model used was the Bohm-gyroBohm model with a prescribed density profile (f_{GW} ~ 0.55). The pedestal temperature and width were fixed at 1.1 keV and at ρ = 0.94 in order to obtain H₉₈ ~ 1 (compatible with pedestal MHD stability calculations). Particular attention was placed on the LH current profile, which has been computed using the C3PO/LUKE package (fast ray-tracing code/3D Fokker-Plank solver). Off-axis LHCD, which is an effective tool for optimizing the current profile to access Advanced Tokamak (AT) operating modes, can be obtained. In the simulation shown in Fig. 3, a fully non-inductive regime is reached with 3.7 MW of LH current drive localized at half minor radius (r/a ~ 0.5). This allows the safety factor to be maintained close to 2 with a very low or reverse magnetic shear up to r/a ~ 0.7 with q₉₅ ~ 4.2, favourable for the triggering of internal transport barriers. In this simulation, the bootstrap current fraction approaches 50%. Further simulations at higher plasma current indicate that the LH current drive profile could be located in the outer part of the plasma: r/a = 0.7 at a density of 7 × 10¹⁹ m⁻³. All these features are particularly relevant for access to AT regimes.

The new configuration opens a wide range of steady-state scenario studies, including their real time control over long duration.

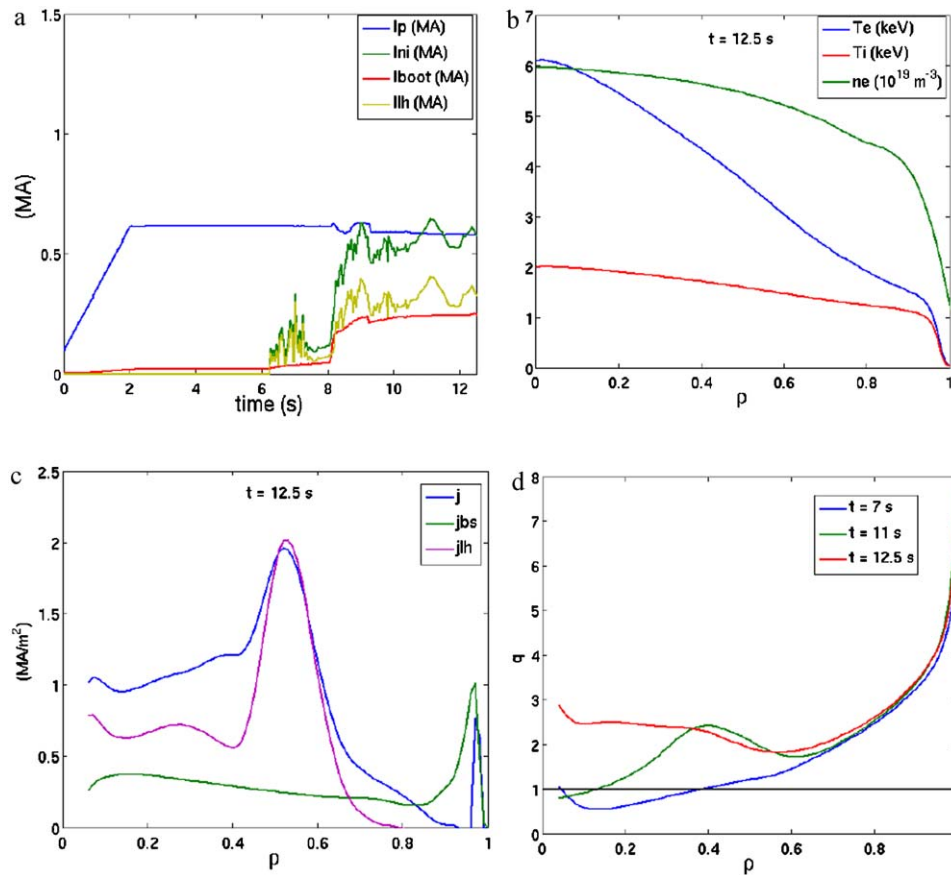


Fig. 3. CRONOS simulation of non-inductive scenario. A steady-state regime is sustained with 3.7 MW of LHCD and 9 MW of ICRH; (a) Time evolution of total, non-inductive, bootstrap and LH currents; (b) Electron temperature, ion temperature and electron density profiles at 12.5 s; (c) Total, bootstrap and LH current profiles at 12.5 s; (d) Safety factor profile at 7, 11 and 12.5 s.

4. Divertor physics issues

4.1. Tungsten contamination

The introduction of a tungsten divertor raises the issue of plasma contamination. Due to the requirement to access high plasma currents, requiring a given minimum plasma volume, the physical space available for the divertor is restricted. As a consequence, the X-point to target poloidal distances are restricted to just a few cms.

The plasma temperature at the target is of crucial importance for the tungsten erosion in the divertor. Preliminary SOLPS [6] simulations have been performed to obtain an estimate of the range of target temperatures to be expected for given power into the scrape-off layer (SOL). The predicted target plasma temperatures at the lowest power into the SOL so far simulated ($P_{\text{SOL}} = 4$ MW) are only marginally higher than the W physical sputtering threshold for D^+ ions (Fig. 4).

These SOLPS background plasmas have been used as input to the impurity Monte-Carlo transport code DIVIMP, which is employed to track W release and eventual penetration into the core plasma. At the lowest P_{SOL} , ~50% of the W released is promptly redeposited and W self sputtering enhances the background plasma (ions and charge-exchange neutrals) erosion rate by ~10%. At higher P_{SOL} , the contribution of W self-sputtering doubles, but a higher prompt redeposition fraction somewhat compensates, although the predicted core W concentration, c_W does increase. Overall, the simulation results for c_W are in reasonable qualitative agreement with earlier observations from the ASDEX Upgrade W divertor experiment [7], in which, with similar divertor geometry ("Div I") to that proposed here in terms of the outer target and the

X-point to outer target distance, $C_W < 2 \times 10^{-5}$ was found for 90% of the discharges. Given that one of the main parameters controlling the target erosion rate is the upstream density and that the densities foreseen for the diverted TS match those used in these AUG experiments, the issue of W contamination should not be a strongly limiting factor. It is also clear that extrinsic radiating impurities can also be injected to cool the divertor plasma if necessary (such scenarios will be mandatory on ITER and have been recently successful on the all-W AUG [8]).

Furthermore, the installed ICRH power could be used to avoid central accumulation in a similar way to the use of ECRH on AUG [9].

4.2. Particle exhaust

The exhaust capability of the open divertor configuration using the present pumping system through 10 bottom vertical ports connected to 10 turbopumps has been estimated with the EIRENE code and fixed plasma background. The pumping efficiency was found to be low, of the order of 0.2% of the recycling flux (to be compared to 0.9% with the present toroidal limiter installed in the lower part of the torus). A baffle has therefore been introduced on the outer side of the divertor to enhance the particle exhaust (see Section 6). The pumping efficiency of this arrangement is expected to rise to 2–4% and should be acceptable for particle control according to the present L-mode long pulse operation experience in circular geometry.

The option of adding a cryo-pump on the outer and/or inner side of the divertor to double the pumping capability is also being considered.

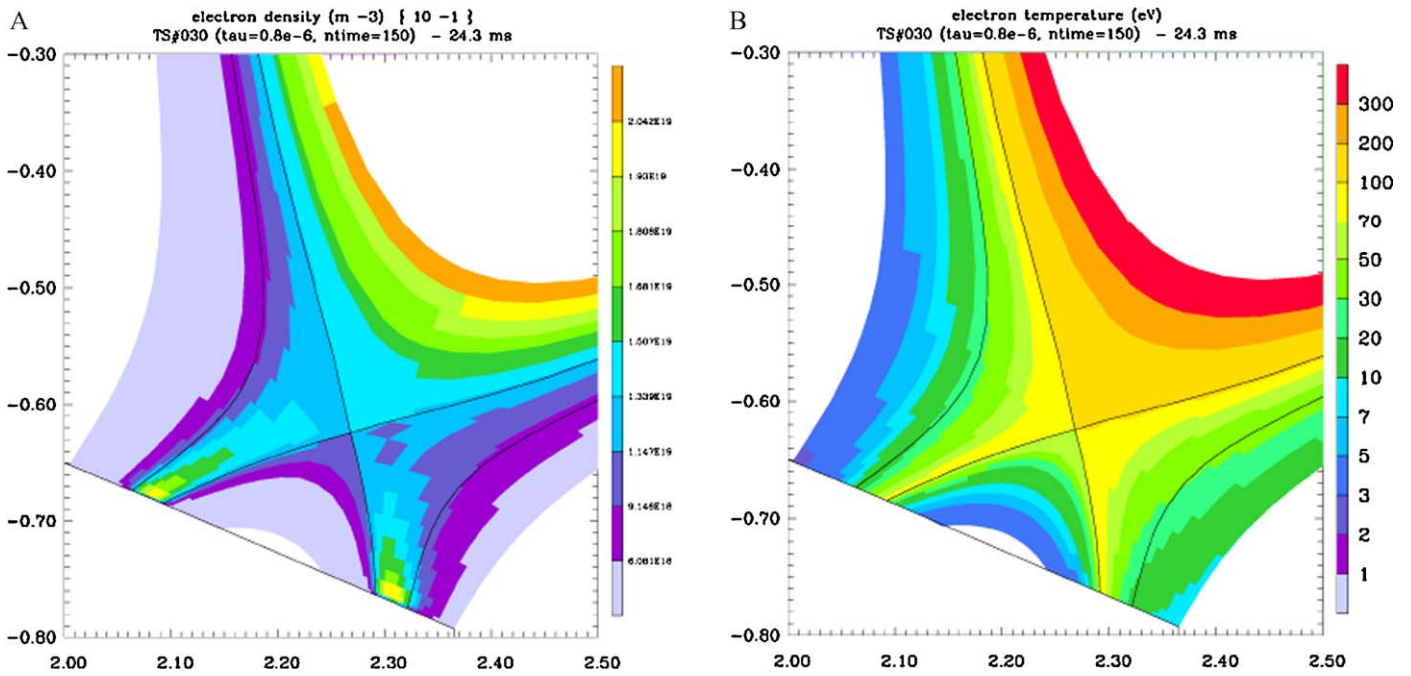


Fig. 4. SOLPS 4.0 simulation of the divertor for 4 MW of power into the SOL. (a) Electron density; (b) electron temperature ($n_{e,sep} = 1.4 \times 10^{19} \text{ m}^{-3}$ and $T_{e,sep} = 120 \text{ eV}$ at the outboard mid-plane separatrix).

5. Heat fluxes on the divertor targets

The main objective of the new configuration is to provide relevant heat loads on the divertor targets over long duration to test PFC technology and develop PFC safety systems.

Expected heat flux patterns have been computed using the poloidal field map, input power conducted in the SOL and ad hoc power decay lengths at the outer midplane. The results are shown in Fig. 5 for two equilibria with different X-point-target distances, 10 MW of power conducted through the separatrix and different power decay lengths: $\lambda_q = 5, 10$ and 15 mm. The power flux density profile is very peaked at the strike points when the X-point is far away from the target and, spreading markedly as a result of poloidal flux expansion as the X-point approaches the target. For the same SOL power (10 MW) and the same λ_q (5 mm), the maximum power flux can range from 20 MW/m² down to 5 MW/m². This

is a very interesting feature for PFC testing purposes, especially for long pulse operation, as it should allow with distant X-point high power flux densities on the targets with moderate input power.

In both configurations, the field line incidence angle at the maximum power flux density position is around 2–3°, close to values on ITER for the baseline inductive H-mode scenario. Gap, castellation, manufacturing tolerance and misalignment issues could be investigated.

6. Engineering studies

The engineering studies performed so far are detailed in a companion paper [10]. The lower divertor target design will be closely based on that currently envisaged for ITER (W monoblocks), while the upper divertor region will be used to qualify the main chamber wall heat sink technology adopted for the ITER blanket modules

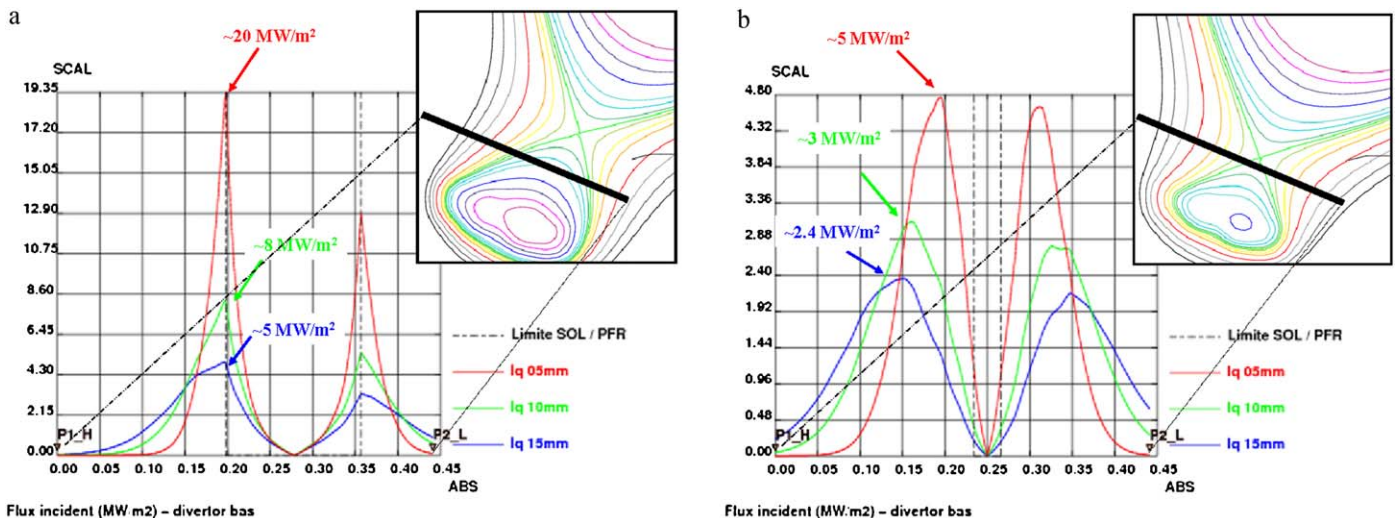


Fig. 5. Power flux density profiles on the divertor targets for two different magnetic configurations, distant (a) and close (b) X-point (a) and 3 different power decay lengths in the SOL ($\lambda_q = 5, 10, 15 \text{ mm}$). The power conducted in the SOL is 10 MW.

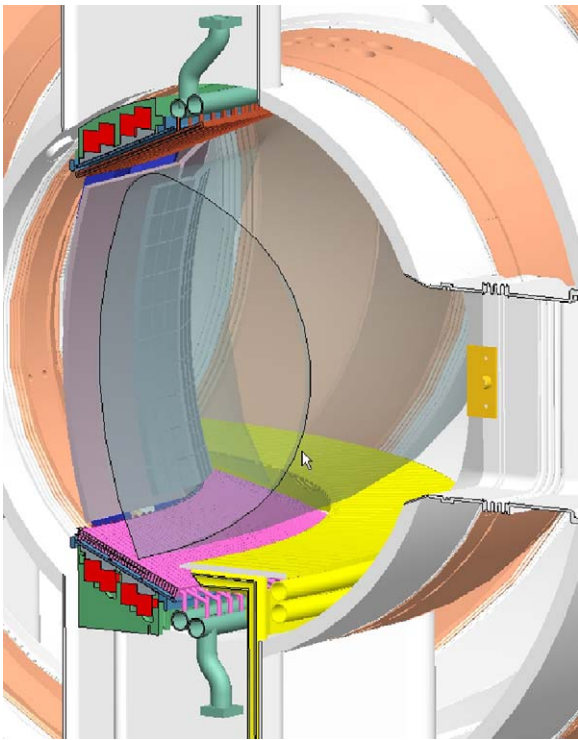


Fig. 6. 3D view of the present design showing the coils and their casing, the PFCs and the pumping baffle.

(CuCrZr copper/stainless steel) with a tungsten coating in place of the Be tiles which ITER will use (Fig. 6). Steady state performances under high heat flux (up to 20 MW m^{-2}) of W monoblock components have been studied in details in [11–14].

Five years are expected to be required for manufacture of the W divertor elements. This industrial-scale manufacturing and the associated quality assurance testing will alone be of great use to ITER.

7. Conclusions

The W divertor in Tore Supra will complement existing W divertor experiments in Europe (e.g. JET and ASDEX Upgrade) by adding the important component of long pulse capability and actively cooled surfaces to the operational experience being gathered elsewhere. Extended plasma exposure in tokamak environment on a wide scale will provide access to ITER-critical issues such as PFC lifetime (melting, cracking, etc.), tokamak operation on damaged metallic surfaces, real time heat flux control through PFC monitoring, fuel retention and dust production. Therefore, a shortening of the learning curve of ITER could be expected, especially in the fields of optimized and controlled PFC operation plus anticipated knowledge which could prove to be useful both for safety files and design refinements.

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