**CEFDA05-1281**

# Task Title: TW5-TRP-005: HCLL BLANKET DESIGN FROM PPCS-MODEL AB

## INTRODUCTION

Within the framework of the European Power Plant Conceptual Study (PPCS), the relatively near-term reactor model AB based on the use of Helium-Cooled Lithium-Lead blanket (HCLL) has been developed and assessed. The HCLL blanket is based on the use of EUROFER as structural material, of Pb-Li (Li at 90% in 6Li) as breeder, neutron multiplier and tritium carrier, and of helium as coolant with inlet/outlet temperature of 300/500°C and 8 MPa pressure. The initial study has been focused on the "large module" maintenance scheme. The conclusion of such a study indicated that the use of multi-module maintenance could allow improvement in the blanket design by reducing the manifold complexity, in decreasing the He pressure drop and in locating pipes rewelding in a low neutron flux region.

## 2006 ACTIVITIES

The year 2006 was devoted to the conceptual design of the model AB HCLL blanket based on the Multi-Module Segment (MMS) maintenance scheme.

The basic principle (figure 2) is to have relatively small modules, welded on a strong poloidal back structure, in order to form a blanket segment which can be removed from the top in a similar manner as a banana-shaped segment. The MMS are removed as a single component and, during segment replacement, the feeding pipes to be cut and rewelded are only those close to the top port for helium and bottom port for PbLi in order to allow most of LiPb to drain by gravity, that are regions submitted to very low neutron flux. This back structure is a common collector which will allow the feed and collection of the fluids (He and PbLi) in parallel for each module.

It is assumed that the reactor features (tables 1 & 2) a major radius of 8.4 m and 16 toroidal coils defining a vacuum vessel with 16 sectors (figure 1); the maintenance will be performed vertically through the 16 upper ports. Each sector is formed by 2 inboard and 3 outboard MMS. Therefore, each upper port will allow the extraction of 5 MMS. Each MMS has 6 modules, and the general size of these modules is about 2x1m² with the higher edge along the poloidal direction.

*Table 1: Main reactor parameters*

| PARAMETER | MODEL AB |
|---|---|
| Unit size (GWe) | 1 |
| Mass flow rate (kg/s) | 2700 |
| Aspect ratio | 3 |
| Major radius (m) | 8.4 |
| Elongation (95% flux) | 1.87 |
| Triangularity (95% flux) | 0.47 |
| Number of coils | 16 |

*Table 2: Main characteristics of radial built*

| Components | Inboard thickness (mm) | Outboard thickness (mm) | Material | Attachment | Temperature (°C) |
|---|---|---|---|---|---|
| TF Coils | 1026 | 1026 | | | |
| Vacuum vessel | 400 | 750 | | | |
| Gap for thermal expansion | 20 | 20 | | | |
| Hot Ring Shield | 200 | 400 | Eurofer | Rail + Thrust | ≈300 |
| Segment manifold | ≈400 | ≈500 | Eurofer | Top-hung : bolt + thrust | 300-500 |
| Module | ≈600 | ≈800 | Eurofer | Welded | ≈500 |

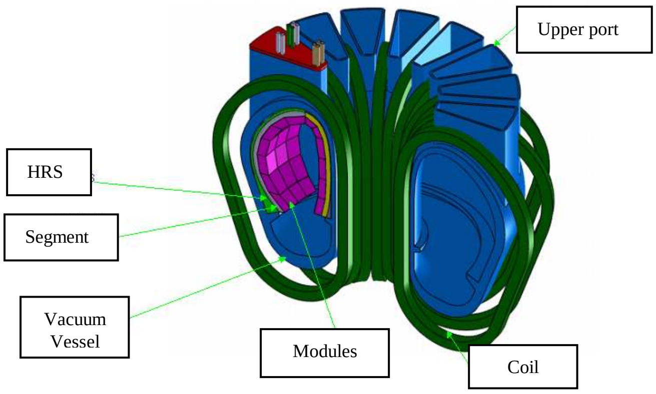


*Figure 1: General view of the reactor*

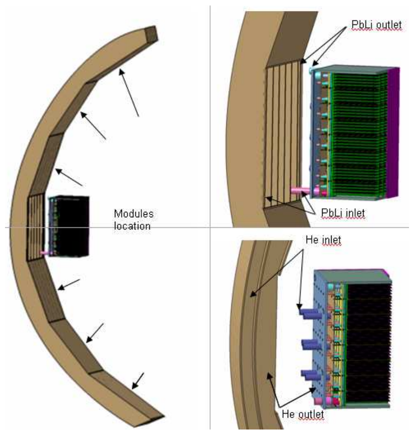


*Figure 2: Illustration of « segment + modules » shpe*

The selected manifold concept minimizes the number of back plates, and has to withstand a thermal gradient of 200°C on its thickness. So the hot part 500°C (He outlet and PbLi chambers) is on the module side, and the cold part 300°C (He inlet) on the back side.

The segments are supported on the Hot Ring Shield which is also at 300°C. However, taking into account the different thermal inertia of the different components, the segments are top hung by bolts in a local area, aligned on the gravity center of the segment with its modules.

On the inboard side, the 2 segments are top hung independently on the hot ring shield. On the outboard side, only the 2 lateral segments of a sector are top hung on the hot ring shield and the middle segment is fixed on the 2 lateral one. So it is the set of 3 segments which is top hung on 2 fixation area of the hot ring shield.

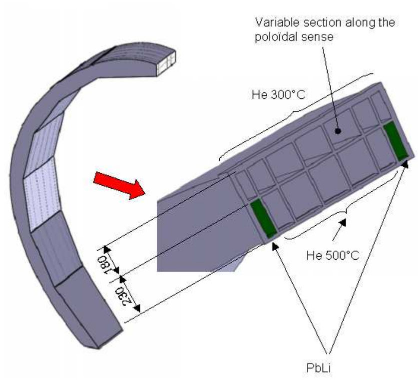


*Figure 3: Manifold cross section*

F.E. calculations showed that the thermal loads in manifold are allowable, but had a high level and induced a deformation of the segment C-shape. In order to minimize this effect, several evaluations were performed (figure 4).

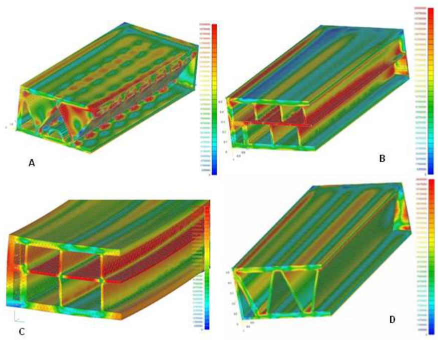


*Figure 4: Numerous cross sections proposed to reduce thermal stresses*

Pressure drop assessment has been also made. The total pressure drop of a segment with its modules, taking into account with the different simplification assumptions is about 0.3 MPa, and it is parted as follow:
- Segment manifold: 0.0289 MPa,
- Modules manifold: 0.0466 MPa (the new design decreases this pressure drop of 0.03 MPa),
- First wall / Stiffening plates: 0.1180MPa / 0.092 MPa,
- Cooling plates: 0.105 MPa.

These calculations show that the higher resistance in this loop takes place in the First Wall and Cooling Plates which represents 75% of the total pressure drop. So the effort on the design to decrease the pressure drop of the circuit has to focus on these elements. This result seems to be intrinsic to the present concept : cooling a structure which has a very high thermal flux, with helium, needs a very high velocity to have a high thermal exchange, and so a high pressure drops. One way would be to design an enhancement exchange devices, i.e. fins, in order to keep a high level of thermal exchange with a lower velocity of helium.

Assuming an inlet temperature of 300°C, an inlet pressure of 80 bar, and a pumping efficiency of 0.8, the required pumping power, for a helium flow rate of 2700 kg/s (table 1), is 148 MW so 15 % of the 1 GWe produced by the reactor.

This result takes into account only the pressure drop of the set “segment+module”. On the He loop, others components, like valves, heat exchangers, … will induce pressure drop, increasing the pumping power which could represent 20-25% of the electric power supplied by the reactor.

## CONCLUSIONS

A MMS concept was studied in 2006 with its main technical options:

- Maintenance independency of each sector, which induce 16 x 5 segments (3 on outboard side and 2 on inboard side),
- Procedure for the segments removal,
- Conceptual design of segments (optimization of fluid distribution), composed of 6-7 modules,
- Proposal ways to the segment fixations,
- Conceptual design of the equatorial module,
- Pressure drop assessment, which seems to be to key issue of the concept. The pressure drops of the He side, estimated for a MMS only, require a pumping power of about 15 % of the electric power produced by the reactor and from which 75% are due to HCLL modules.

This preliminary design study highlights several key points, on the MMS concept proposed, to be checked in more details:
- Attachment of the segments on the hot ring shield,
- Modules design on the extremity of the segment,
- Maintenance of segments,
- Thermo mechanical assessment of the segment for its optimization with taking into account EM loads.

## REPORTS AND PUBLICATIONS

[1] L. CACHON et al, " HCLL blanket concept for the DEMO conceptual study (EFDA 05-1280 / TW5-TRP-005)" NT DTN/STPA/LTCG 2006-013

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**CEFDA05-1285**

# Task Title: TW5-TRP-003: DEMO – BLANKET SEGMENTATION AND MAINTENANCE REMOTE HANDLING ISSUES

## INTRODUCTION

The development of technologies required for the DEMO demonstrator planned in the roadmap of fusion energy has already started in the aim of testing some DEMO-relevant components inside ITER. The breeding and high-grade energy extraction blanket modules are some of these elements. Indeed, these components facing the plasma are exposed to very high thermal loads, so it's necessary to replace them at given periods.
Due to the radiation level, it is necessary to make the operations fully by Remote Handling. This critical maintenance needs to be done in acceptable times in regard to the reactor availability. This task focuses on the aspects concerning the Remote Handling of the blanket elements during maintenance phases.

## 2006 ACTIVITIES

The objectives of this task are to review the different possible segmentations to allow the selection of a reference concept for the forthcoming DEMO conceptual design for three candidate designs (see figure 1):

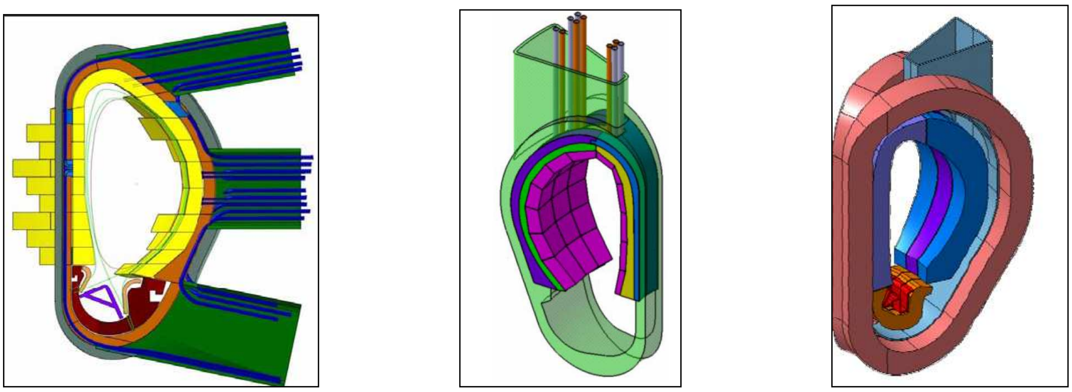
- Large Modules (LM)
  This kind of modules results from a poloidal and toroidal segmentation to allow an acceptable size to handle the modules by the equatorial ports. The size of the port is itself limited by the TF and PF magnets arrangement. This is the retained solution for ITER tokamak.
- Vertical Segments (VS or "bananas")
  The vertical segments are mainly obtained by a toroidal segmentation and sections are divided poloidaly in two major parts, the inner and outer boards.
  The maintenance of these elements should be carried out by upper vertical ports. For these ducts, the main geometrical constraint is the TF magnet arrangement.

### REVIEW OF MAIN EXPERIENCE IN HANDLING EQUIPMENT FOR FUSION

The base of this task is to analyze how the maintenance can be done in the context of DEMO. This implies to consider how it is planned in the case of ITER, even if the objectives of these two plants are not the same. In ITER, the Remote Handling Equipment is composed of the following items: (from ITER DDD – 2.3):

- In-Vessel Transporters system, effectors & tools (see figure 2)
- Divertor Cassette Handling system (see figure 3)
- Port/Plug handling system
- Maintenance Cask
- Repair/ Maintenance in hot cell
- Cryopump handling system
- In Vessel Viewing system
- In Vessel metrology system
- NBI Maintenance system

Today, the main relevant concepts applicable for DEMO concern the In-Vessel Transporter system currently in charge of the in-vessel remote maintenance of ITER blanket modules and the Divertor Remote Handling equipment in charge of the cassette exchange.



*Large modules* — *Vertical segment multi-modules* — *Vertical segment*

*Figure 1: The three candidate designs*

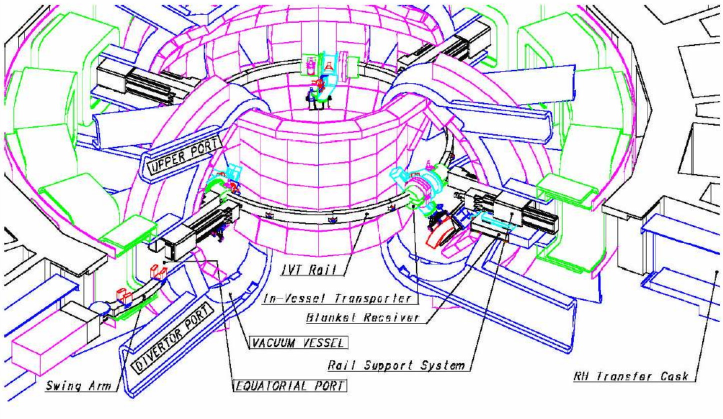



*Figure 2 ITER In-Vessel Transporter*

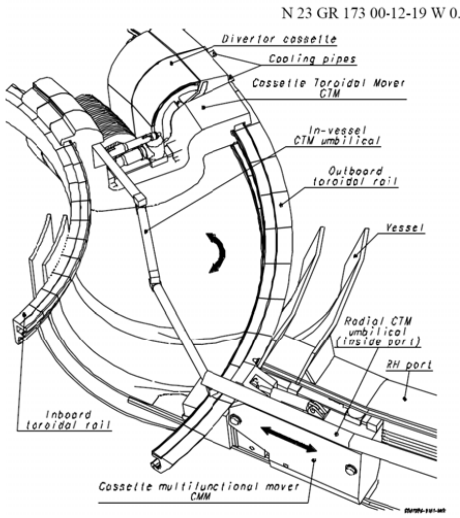



*Figure 3: Divertor Remote Handling*

The JET experience shows the benefit of use of a boom type system. Main idea of this kind of systems is to handle in-vessel components by mean of an articulated boom through a tokamak's equatorial port. Generally, the boom's end effector is made of universal force feedback manipulators able to carry limited payload with high dexterity. In the case of DEMO this concept is relevant as support for services tasks (cutting welded joints, unfastening bolted flanges and couplings, disconnecting water and gas pipes …) as shown in figure 4.

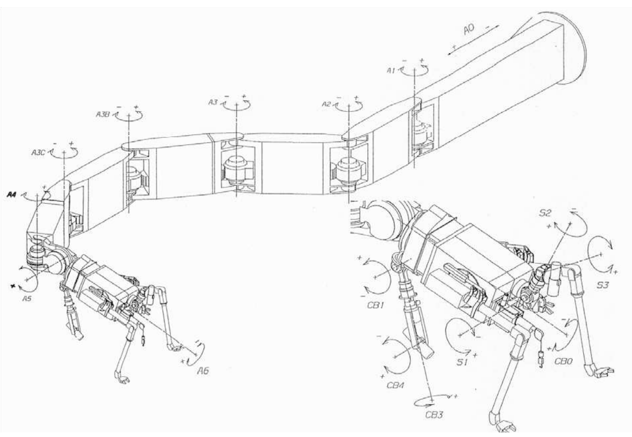


*Figure 4: JET boom type RHE*

In the Next European Torus (NET) framework, a vertical handling concept has been designed to perform the handling of vertical segments in the perspective of handling Bananas-like blanket modules (see figure 5).

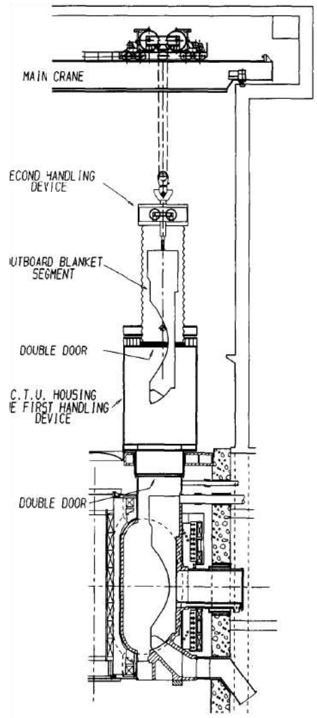



*Figure 5: NET handling Bananas-like blanket modules*

*Table 1: Synthesis of main RH families for fusion plants*

| | Blanket Modules RHE | | | Divertor Cassettes RHE | |
|---|---|---|---|---|---|
| | ROBERTINO | BOOM | IVT | CMM | CTM |
| Family | Vertical Handling | Cantilever handling | Cantilever rail + vehicle | Radial Mover handling | Toroidal Mover handling |
| Type of access | Vertical | Equatorial | Equatorial | Divertor | Divertor |
| Reactor or concept | NET | JET | ITER | ITER | ITER |

**Method of assessment**

The scope of this part is to explain the different drivers used for the assessment and to detail what kind of data is expected from the partners of the task. These criteria will be used to assess the relevance of the concepts in regard to maintenance purpose. The five items used for the analysis are these ones:

- Number of in-vessel components to replace
- Work in parallel
- Logistics
- Simplicity of handling equipment
- Time estimation

For each driver, an explanation is given with examples when possible. All the different parameters required for each concept are not already known at this state of the study, but this method has been built to be also applied on further advanced design activities.
As an example, the table 2 gives the result of the use of the parameters of evaluation on the state of the art RH concepts when considering RHE assessment for the simplicity criteria consideration.

*Table 2: Handling parameters application on the state of the art*

| | Blanket Modules RHE | | | Divertor Cassettes RHE | |
|---|---|---|---|---|---|
| | ROBERTINO | BOOM | IVT | CMM | CTM |
| Family | Vertical Handling | Cantilever handling | Cantilever rail + vehicle | Radial Mover handling | Toroidal Mover handling |
| Type of access | Vertical | Equatorial | Equatorial | Divertor | Divertor |
| Directions | V+ | R+, T+/- | R+,T+/-,P+/- | R+ | T+/- |
| Principle | No Guidance | No Guidance | Guidance | Guidance | Guidance |
| VV Interface | No | No | No permanent | Permanent | Permanent |
| Strategy | Single | Several | Several | Single | Several |
| Safety / Maturity | Concept | Proof | Concept | Concept | Proof Concept |

The time estimation will be mainly a conclusion of the union of the 4 main parameters given above. It will concern all the tasks and subtasks we will define, for example:

- Opening the needed ports
- Installation of maintenance systems
- Cutting cooling pipes
- Disconnection of the modules
- Replacing Manipulation (estimation on robotic knowledge)
- Transport to Hot Cell Building (go and back)
- Re-connection of the modules
- Re-Welding the Pipes and control of the tightness
- Withdrawal of maintenance systems
- Closing ports

Each of these steps (or more) will be considered and weighted following the parameters of the segmentation.

## CONCLUSIONS

A time effective replacement of the first wall modules of the Vacuum Vessel is necessary to make profitable a fusion machine. Time effective maintenance scheme of internal toroidal Vacuum Vessel components remains always a severe issue highly related with first wall design. This task gives an overview of the different kinds of main maintenance equipments that have been used or designed until today for the replacement of the blanket. The state of the art was used as base for building comparison criteria to assess the maintenance scheme propositions associated with their segmentations. The main criteria families were the following: number of components to replace, logistics, work in parallel, simplicity of handling equipment, and the maintenance time estimation. Issues and recommendations are given here to highlight the different problems that have no definitive answer yet. These considerations could also be used and developed as guidelines for further design activities.

## REPORTS AND PUBLICATIONS

Blanket segmentation and maintenance, Remote handling issues
DTSI/SRI/LPR/06RT027

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**TW5-TRP-002-D03a**
**TW6-TRP-002-D02**

# Task Title: ANALYSIS OF CURRENT PROFILE CONTROL IN TOKAMAK REACTOR SCENARIOS USING REALISTIC TREATMENT OF CURRENT DRIVE EFFICIENCIES

## INTRODUCTION

On the path to the development of the first commercial fusion reactor, DEMO will be the next step after ITER, with the aim of testing the main technology options at a somewhat reduced electrical power with respect to the commercial reactor (1 GW vs 1.5 GW). The DEMO reactor should work in steady-state, therefore, one of the main physics challenges will be the establishment and the control of a non-inductively driven current density profile. This task will consist in the analysis of this problem by means of the integrated modelling code CRONOS. CRONOS [1] is a suite of numerical codes for the predictive/interpretative simulation of a full tokamak discharge. It integrates, in a modular structure, a 1-D transport solver with general 2-D magnetic equilibria, several heat, particle and impurities transport models, as well as heat, particle, current and momentum sources. This task will provide a set of CRONOS simulations of a DEMO steady-state discharge, in which the various current sources combine to yield an MHD-stable current density profile. It should be stressed that the existence of such a scenario is not a priori guaranteed. Therefore, the 1-D modelling is an essential step for the first design phase of DEMO and of course also of the following commercial reactor.

To this end, a substantial improvement of the CRONOS code was necessary, i.e., the upgrade of the NBCD (Neutral Beam Current Drive) package. In order to allow for more realistic calculations of the NBCD for a beam energy of 2 MeV, the evolution of the fast ion distribution which results from the NB injection has to be computed by an orbit following Monte-Carlo code. This was done by coupling the particle source computation of the module SINBAD with the Monte-Carlo code SPOT [2].

CRONOS can in principle solve coupled evolution equations for electron and ion temperatures, plasma current, density, rotation and impurities. Nevertheless, even for ITER, this type of fully integrated simulations are still a very difficult task and will also be of limited practical interest in the absence of a well developed pedestal model and reliable edge-core coupling. Therefore, for this task we concentrate on simulations of coupled heat and current transport equations, at prescribed plasma density and impurity content, with no evolution of the rotation velocity profile.

This work is expected to deliver a set of CRONOS simulations (time evolutions and profiles of the most relevant plasma quantities) for at least one reference scenario, to be adjusted in connection with the evolution of the PROCESS and HELIOS 0-D simulations.

## 2006 ACTIVITIES

**CRONOS simulation results**

With the aim of analyzing the performance of DEMO for different plasma regimes, two configurations have been chosen as representative, respectively, of the full inductive scenario with low bootstrap fraction, and of one more advanced which could be close to the steady state regime with lower inductive current. The global characteristics of the operation scenarios considered for DEMO as well as some of the main global parameters obtained in the simulations are shown in table 1. In fact, the full inductive scenario is just an extrapolation of the expected ITER inductive regime, with a high amount of external current, large major and minor radii and small elongation and triangularity. Unlike the inductive case, the advanced scenario tends to decrease the inductive current, the toroidal vacuum magnetic field and major and minor radius, whereas the bootstrap fraction increases. In this configuration longer or even steady-state discharges are expected, however, the large amounts of non-inductive current necessary can be a drawback. For comparison, the equilibria computed by the HELENA code self-consistently with the stationary phase of the CRONOS runs are shown in figure 1, for the inductive and the advanced DEMO, respectively.

*Table I: Global characteristics of the DEMO operation scenarios*

| Parameter | Inductive | Advanced |
|---|---|---|
| Major radius R (m) | 9.55 | 7.5 |
| Minor radius a (m) | 3.15 | 3.0 |
| Elongation/Triangularity | 1.7/0.25 | 1.9/0.47 |
| $B_t$ (T) | 7.0 | 6.0 |
| I (MA) | 30.5 | 19 |
| $n_{e,0}$/ $<n_e>$ ($10^{19}$ $m^{-3}$) | 12/10.3 | 13/10 |
| $n_e/n_G$ | 1 | 1.25 |
| $T_{e,0}$ (keV) | 42 | 35 |
| $T_{i,0}$ (keV) | 60 | 40 |
| $P_{fus}$/ $P_{add}$ (MW) | 4300/246 | 2500/135 |
| $P_{ES}$/ $P_{bremms}$ (MW) | 120/156 | 30/100 |
| $f_{BS}$ (%)/ Q | 28/17.5 | 48/18.5 |
| $q_0/q_{95}$ | 0.81/3.4 | 1.4/4.5 |

The density and the electron and ion temperature profiles as well as the heating power profiles obtained for the advanced DEMO when t=2500s (very close to a steady state) are shown in figure 2(a) and 2(b) respectively.

The central ion temperature, $T_{i,0} \approx 40$ keV, is slightly higher than the central electron temperature $T_{e,0} \approx 35$ keV, although the pedestal is similar $T_{ped} \approx 6$ keV. Heating by off-axis NBI (95 MW) and by LH waves (40 MW) is applied. Ad hoc, but reasonable, values for the power deposition profiles and for the CD efficiencies have been used in these preliminary simulations (the new NBI module and a full ray-tracing/Fokker-Planck module ofr LHCD, which require long CPU times, will be used in a later phase of the project). The current density profiles in steady state and the q profiles at three different times are shown in figure 3. These plots display good properties: an inverted and marginally higher than unity q profile. The fusion power obtained is 2.5 GW, in reasonable agreement with the 0-D analysis. In this scenario, long pulse operation is made possible by a large amount of injected power (= 195 MW), which implies a rather low Q = 18.5. Still, as shown in figure 5, the non-inductive current fraction is of the order of 75%, the pressure limit ($\beta_N = 4l_i$) is attained, as well as the Greenwald limit. Improvements could come from the use of a transport model allowing for the formation of an ITB (which is not the case for the GLF23 model used in these simulations).

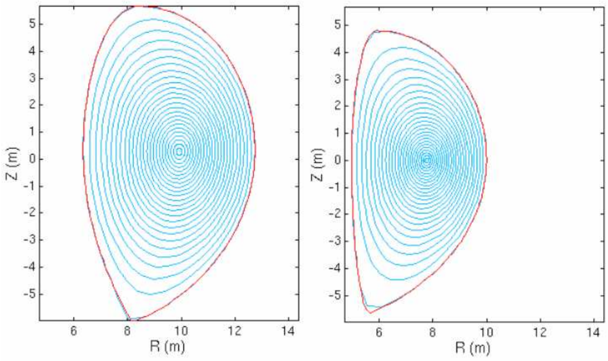


*Figure 1: Computed equilibria for the inductive (left) and the advanced (right) DEMO, respectively*

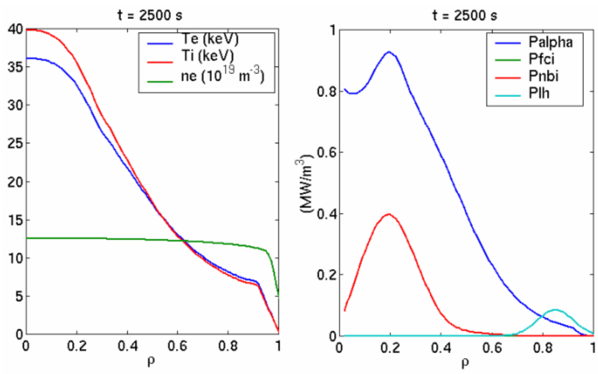


*Figure 2: Computed temperature and density profiles (left) and heating power profiles (right) for the advanced DEMO parameters*

## CONCLUSIONS

The results obtained in these first series of CRONOS simulations are encouraging since they show that a regime close to the hybrid one can be possibly attained in DEMO with a reasonable amount of CD power. Nevertheless, the main aim of obtaining a fully non-inductive regime (at a value of Q ~ 20) has not been attained. Since the total current drive power cannot be further increased without lowering the Q, the basic ingredient that has to be exploited is the development of a transport barrier. If a pronounced reversal of the q profile can be obtained on a large part of the plasma cross section by a suitable combination of off-axis current drive sources, the bootstrap current will increase at the barrier location, enhancing the effect in a positive non-linear loop. The next problem will of course be the control of such a process. Finally, the use of full NBCD and LHCD computations by advanced modules (Monte-Carlo and ray-tracing/Fokker-Planck respectively) will be necessary in order to validate the scenario.

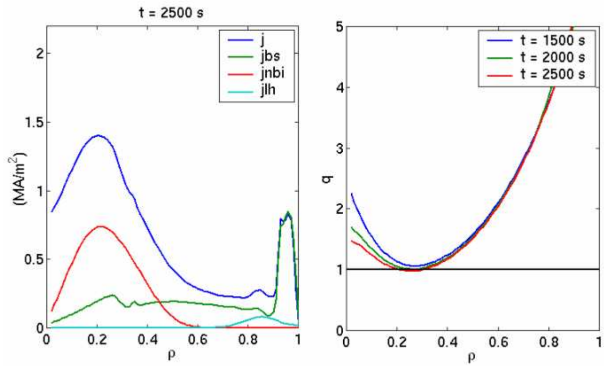


*Figure 3: Computed current density profiles at steady state (left) and safety factor profiles at three different times (right) for the advanced DEMO parameters*

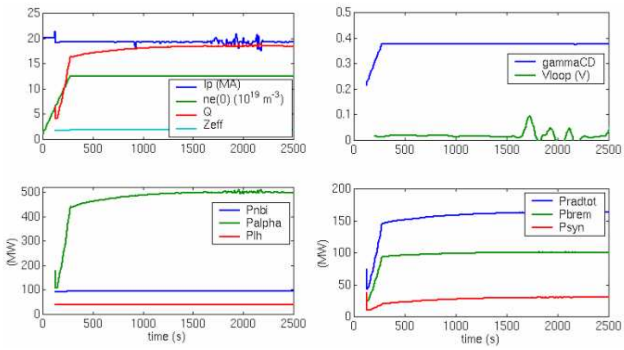


*Figure 4: Computed time evolutions of several plasma quantities for the advanced DEMO parameters*

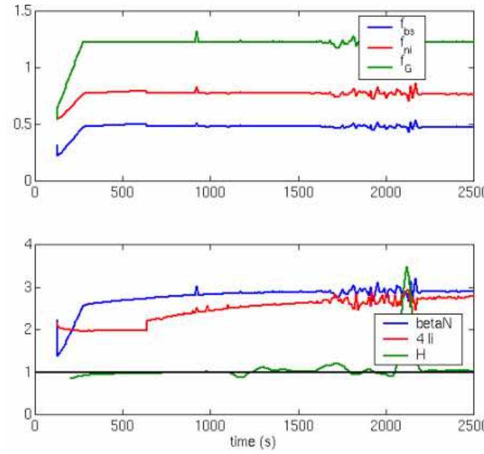


*Figure 5: Computed bootstrap, non-inductive, Greenwald fractions (top) and $\beta_N$, 4li, H-factor time evolutions (bottom) for the advanced DEMO parameters*

## REFERENCES

[1] V. Basiuk, Nucl. Fusion 43 (2003) 822

[2] M. Schneider et al., PPCF 47 (2005) 2087

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**TW6-TRP-006**

# Task Title: DELIVERY OF THE MAGNET SYSTEM OUTLINE FOR A DEMO REACTOR

## INTRODUCTION

In the framework of DEMO conceptual studies, the TF magnet system has to be integrated at an early stage in the overall design development.

As a matter of fact, located at the very heart of the tokamak, the TF system interact with many aspects of the machine.

It is therefore important to rapidly identify and simply parameterise the mains aspects of this interaction in order to optimally adapt the machine design.

## 2006 ACTIVITIES

### MAIN ASPECTS OF THE MAGNET INTERACTION WITH THE TOKAMAK DESIGN AND COMPONENTS

The different aspects of this interaction have been presented in three meetings which took place:

- One at Garching corresponding to the kick-off meeting,
- Two at CEA-Cadarache dedicated to the presentation of the ESCORT code and to the modules maintenance.

These aspects were presented on the basis of the work performed in a previous TW5-TMSC-HTSMAG EFDA task [1], [2], [3], dedicated to scooping studies for HTS fusion magnets.

The DEMO magnet system will probably represent about 30 % of DEMO investment cost, a substantial part of the cost is the TF system and it is therefore crucial to select the appropriate technologies.
Several technologies are presently envisaged for the DEMO TF system. 5 K temperature operation and A15 materials such as $Nb_3Sn$ constitute the main option, which should be confirmed in relation with the recent developments on the ITER conductor program. HTS material associated with higher operation temperature: 20 K for Bi2212 material and 50 K for YBCO. The selection has to be made, not only as a function of the cost and of the industrial availability of the materials, but also as a function of the impact on the overall efficiency of the machine.

The stainless steel structures represent 90 % of the magnet section for a tokamak like DEMO, at given toroidal magnetic field. The size of the Tokamak, characterized by the major radius must be sufficiently large to accommodate the so-called radial built of the magnet system in the central region of the TF magnet inner legs. The couple (toroidal magnetic field – major radius) has to be adequately selected such as to satisfy this requirement which gives a very important impact on the machine design.

The major radius should be also sufficiently large, taking into account the TF inner leg radial built, to leave some place in the center of the machine for the central solenoid. The required flux and the associated radius of the central solenoid must therefore be defined in DEMO according to the objectives and according to the most probable scenario. This is not the case at the present time.

Regarding the maintenance of the machine and the extraction through the upper port of the blanket modules, it has to be checked that the available space between the two outer legs of the neighboring magnets is sufficient. This also can impact the overall design.

### ESCORT: A TOOL TO HELP FOR THE MAGNET SYSTEM DESIGN AND FOR ITS INTEGRATION INSIDE THE MACHINE

The ESCORT code is written in FORTRAN 77 (3500 lines), which allows to take into account subroutines developed in the framework of ITER and other projects such as for:
- Conductor dimensioning;
- Field calculations.

ESCORT can be considered as a tool for a first dimensioning of magnetic systems for Tokamaks. This can be illustrated through ITER:
**ITER GOAL**: 400 MW fusion power, Q=10, 500 s of plasma discharge in inductive mode.
The selection of the triplet (R, a, $B_t$) (major radius, minor radius, magnetic toroidal field) is made using ESCORT. It provides:
- The TF magnetic system preliminary design including the space for the blankets and a realistic section of structural material;
- The central solenoid design able to deliver the required flux to achieve a 500s inductive plasma discharge.

As it was seen during ITER design, for a given aspect ratio, several couples (R, $B_t$) are possible:
The investment cost or access through ports can contribute to the final selection.
For a first design approach the factor of merit $\xi = R^2 B_t^3$ is very representative of the machine performances (fusion power and amplification factor).

Several output graphics are available with ESCORT. Figure 1 shows a cross section of a DEMO version in the equatorial plane, illustrating the space available between two outer TF legs for blanket modules extraction during maintenance of the machine.

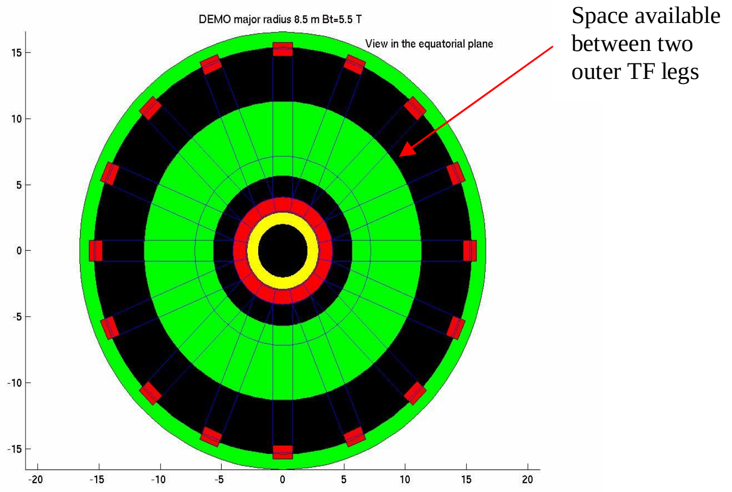


*Figure 1: Output graphic of ESCORT showing a cross section of the TF magnet system in the equatorial plane for a version of DEMO*

## CONCLUSIONS

In the realisation of the task objectives, the ESCORT code appears as a very useful tool able to deliver preliminary designs of the TF system, during the DEMO conceptual studies.

It is now foreseen to deliver the main characteristics of three DEMO TF versions at three different temperatures of operation: 5 K, 20 K and 50 K.

In relation with the CEA team of Cadarache, a criterion is being established to ensure that there is sufficient place in DEMO preliminary design to extract the blanket modules through the upper port between two TF outer legs.

## REFERENCES

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