CEFDA05-1285

Task Title: TW5-TRP-003: SEGMENTATION AND MAINTENANCE ASSESSMENT AND COMPARISON OF CONCEPTS

INTRODUCTION

In the roadmap of the development of fusion energy, ITER is the following major step. The main objective of ITER is to demonstrate prolonged fusion power production in deuterium-tritium plasma. Half way through the scientific exploitation of ITER, it is planned to begin the design of the demonstration fusion reactor, DEMO. Construction of DEMO should start when ITER is fully exploited.
The development of the technologies which will be required for this demonstrator have 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 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 availability of the reactor. This task focuses on the aspects concerning the Remote Handling of the blanket elements during the maintenance phases.
As it is impossible to refurbish the worn blanket inside the torus, it is necessary to carry it fully to the hot cells. This operation implies that the first wall is segmented in smaller parts called blanket modules. Each module needs to be connected to the coolant circuit, so, without considering the efficiency of the heat extraction, the units size have an effect on the number of modules, the number of manifolds and connections, the size of the ports of the reactor, the masses to be carried out, the size of the transfer casks… This is the reason why, the maintenance operations are very closely linked with the choice of the segmentation of the blanket.

The objectives of this task started in october 2005 are to review the different possible segmentations to allow the selection of a reference concept for the forthcoming DEMO conceptual design for three reference design:

- Large modules
- Banana segments
- Multi-module-segments (MMS)

2005 ACTIVITIES

CONCEPTS DESCRIPTION

Large modules (figure 1)

The goal is to limit the total number of blanket modules in order to reduce the replacement time and to maximize the plant availability. The maximum size of the modules is limited by the opening of the equatorial ports. The size of the port is itself limited by the TF and PF magnets arrangement. Only basic concepts for the handling devices have been developed during the PPCS Study and there are limited details on the hydraulic and mechanical connections of the modules.

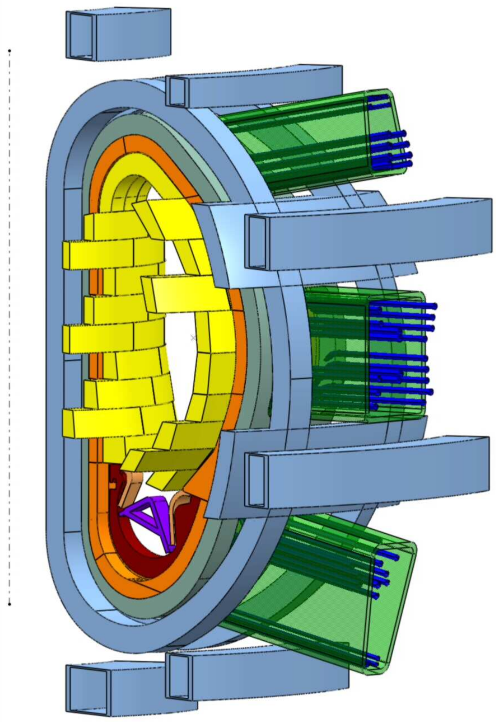

Figure 1: Illustration of the Large Modules concept proposed by FZK

Vertical segments ("bananas")

The "banana" segments were extensively analysed in the NET design and in the ITER CDA design. The width of the vertical port is limited by the TF coils and its radial dimension by the upper PF coils. One of the main features of this segmentation is the location of all hydraulic connections in a secondary vacuum enclosure accessible from the top of the machine. Detailed design of blanket handling devices were developed, but the complete logistics implications were not fully assessed.

Multi-module-segments (MMS)-(figure 2)

The MMS concept consists in a banana back-plate supporting a number of smaller modules. It can be handled through the vertical ports as a banana segment whilst individual modules could be replaced through the equatorial port like in ITER.

This concept has never been analysed in detail.

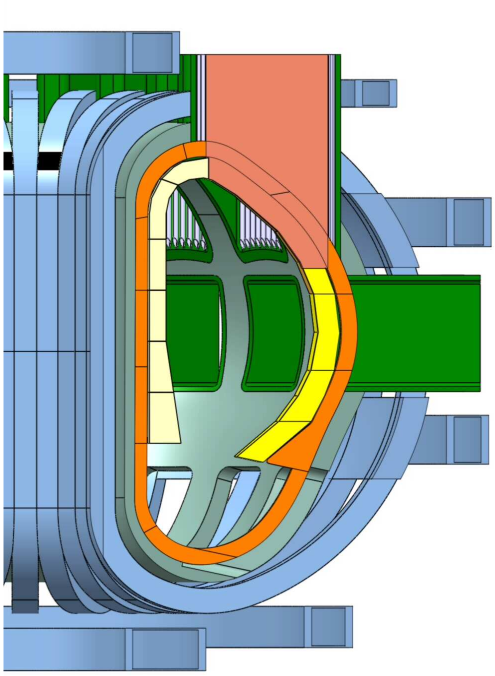

Figure 2: Illustration of the Multi Modules Segment (MMS) concept proposed by FZK

SCOPE OF WORK

The scope of the whole task is to review these segmentations considering the following main criteria:

- manifolds connection and disconnection
- mechanical attachment system
- handling procedures and handling devices (including intervention time)
- logistics
- operational safety

This work is done in close collaboration with FZK and ENEA for the segmentation designs and maintenance concepts and CIEMAT investigates the possible attachments of the modules.
The CEA will focus on the assessment and comparison of concepts, which means:

- Review the logistics associated with each concept: number of transfer casks, contamination control, radiation shine-through during handling, hot cell requirements, etc. and review the possible and required logistics equipments
- Assess the duration of the blanket replacement assuming the parallel work of between 2 and 4 blanket handling devices and analyse the scheme principles
- Assess the 3 possible blanket segmentations from the maintenance and logistics standpoints.
- Recommend a segmentation for the DEMO conceptual study to be started after the completion of this task

REVIEW OF POSSIBLE & REQUIRED LOGISTICS EQUIPMENTS

The ITER main Remote Handling Equipments concerning the current assessment on DEMO are the following:

- In- Vessel Transporters system, effectors & tools
- Divertor Cassette Handling system
- Port/Plug handling system
- Maintenance Cask

These equipments are the references which will guide the first analysis of the proposed concepts. The first comparison criteria we have defined are inspired of these solutions. The other existing reference we take is the JET RHE principally the BOOM and it's MASCOTT. Considering, theses reference, we have started to build a list of comparison criteria on the RHE:

- **Handling Directions**: Radial, Vertical, Toroïdal, Poloïdal, (+/-) (figure 3)
- **Guidance principle** (guided or not):
    - "Guidance"
        - Trajectory guidance of the RHE
        - Trajectory guidance of the MODULES
        - Full or partial load support
    - No "Guidance"
        - Cantilever handler
        - Vertical handling
- **Vacuum Vessel Interface**
    - In Vessel permanent structure**:** Permanent rail or support installation
    - No In Vessel permanent structure
        - No specific In VV support
        - Cantilever handling
        - Vertical Handling
        - Temporary support for RHE or MODULES
- **Strategy of Blanket Modules Transport**
    - Single Manipulator: The manipulator transports the blanket module from the VV to the transfer cask alone. Possible to use mechanism to assist the handling task
    - Several Manipulators: Use of different systems to handle the module from the in vessel to the transfer cask

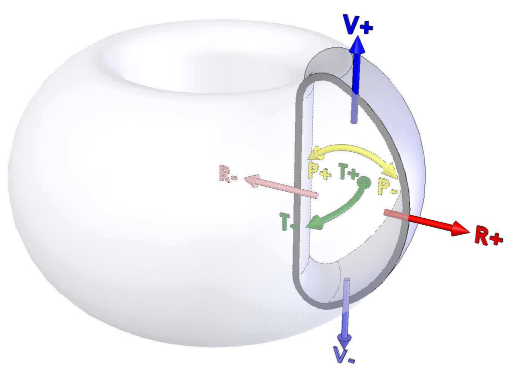

Figure 3: Symbolic representation of Handling Directions

This non-exhaustive list of criteria will be enriched during the following work in 2006.

PRÉLIMINARY ANALYSIS ON FZK LARGE MODULES RHE

FZK started to work on the Remote Handling for the Large Modules Concept (figure 4), and this is the only entry we have yet. This solution is an In-Vessel Transporter-like machine.

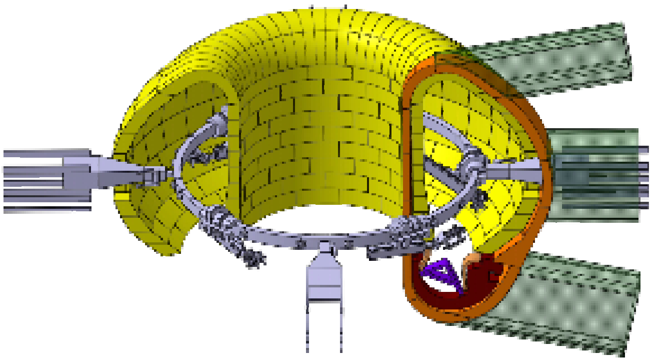

Figure 4: RHE design for Large Modules concept (FZK)

For this design, the first analysis we can make is that this system will benefit of the advantages of the ITER blanket RHE concept:

- Limited interface with Vacuum Vessel
- No specific In-VV support
- Able to remove any single blanket

And we can notice that the payload has been increased to 10 T (4 T for the ITER's IVT). The mass of the whole system is near 200 tons, and the ratio between the payload on the mass of the manipulators is about 2.5 which is very a very advanced value compared to industrials robots. So it seems that this kind of system won't be stiff enough to ensure the accuracy needed. And it will probably be difficult to handle 4 modules at the same time because of the disturbances due to the deformations of the beam which supports the vehicules. That makes the work in parallel quite impossible in this concept.

INFLUENCE OF BLANKET SEGMENTATION ON RHE

One of the important parameters to study is how the segmentation of the blanket influences the design of the RHE. The start of the analysis begins by the criteria given in the table below:

Number of modules	→	Number of manipulations
Geometrical disparity of the modules	→	Complexity of the handling devices
Mass of the modules	→	Difficulty to afford gravity load
Number of equipments on each modules	→	Simplicity of the handler
Size of each modules	→	Accuracy of the position

These simple elements must be guidelines for all choices of strategy for designing the associated RHE to the associated segmentation.

CONCLUSIONS

In the first phase of the work, we have reviewed the different Remote Handling Equipments usable in the fusion context. The main bases are the studies made for ITER and the solutions used in JET.

One of the goals is to perform a complete maintenance of the blanket modules at given periods in DEMO (1 time per 5 full power years), and it is not the case of the ITER reactor. Replacement of shield blanket modules is likely to be required a few times during the lifetime of the ITER machine due to the following normal operation and fault conditions. It means that ITER-relevant RHE for the blanket modules are not obligatorily good solutions for DEMO. However, for ITER, it is needed to proceed the full replacement of the cassette modules at least 4 times. These conditions are nearer the need for DEMO-relevant RHE design.

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TW4-TRP-002-D02b

Task Title: CONCEPTUAL DESIGN OF A HCLL REACTOR TRITIUM CONTROL & MANAGEMENT ANALYSIS, THERMO-HYDRAULIC AND THERMO-MECHANICAL ANALYSES

INTRODUCTION

Within the framework of the European Power Plant Conceptual Studies (PPCS), one of the reactor models, the model AB, is based on a Helium-Cooled Lithium-Lead (HCLL) blanket [1]. A view of the corresponding blanket module and of the detail of a Breeder Unit (BU) is shown in figure. 1. The integration and the design of the HCLL blanket and associated circuits and components within the model AB reactor plant has been addressed and performed in another parallel subtask [2], [3]. The mechanical, thermo-mechanical and thermo-hydraulic analyses for the HCLL have been based on the similar analyses performed for the HCLL DEMO blanket modules [4] taking into account the larger size of the PPCS reactor.
The objective of this task was to check validity of the analyses performed for DEMO when extrapolated to the PPCS specifications and to assess the T-management and control in the HCLL blanket and associated systems. The maximum authorized T-release to the environment is assumed to be 27 Curies/day, equivalent to 1 g/year. However, this allowance will be for the whole plant, including other sources of tritium like reactor refuelling, divertor pumping etc. Even taking the whole allowance for the breeding blanket system, the tritium isolation ratio J1/J5 is to be as high as 200 000, with the assumption of a reactor availability of 82 % (300 operating days per year).

2005 ACTIVITIES

The 2005 performed activities focused on the importance of taking into account the T transport in the Pb-17Li.

In fact, the preliminary tritium management assessment [5], based on an analytic model, evaluated the tritium permeation rat for different Permeation Reduction Factor and different re-circulation rates. It is recalled that, for 10 re-circulations per day, a lithium lead extractor efficiency of 0.8 and a PRF=50, the computed tritium mean concentration is $C_m = 0.065$ g.m^{-3}, and consequently for the 585 m^3 of Pb-17Li present in the HCLL blanket the tritium inventory is 38.1 g (not taking into account the Pb-17Li outside the module: e.g. pipes, pumps, de-tritiation units).
With a tritium partial pressure in helium of $P_{T_2\text{-He}} = 5.47\times10^{-3}$ Pa (that is maximum allowable pressure when considering the T-permeation through conventional Incalloy-pipes in SG [6]), the tritium mass fraction is 1.03 10^{-9}. Considering the whole He inventory equal to 12 400 kg, the mass of tritium present in helium is only 0.013 g.

Conversely, if T-permeation in the steam generator could be reduced to zero, the acceptable tritium partial pressure taking into account only the helium leakage would be: $P_{T_2\text{-He}} = 11.1$ Pa. That leads to a tritium inventory in helium of 25.8 g.

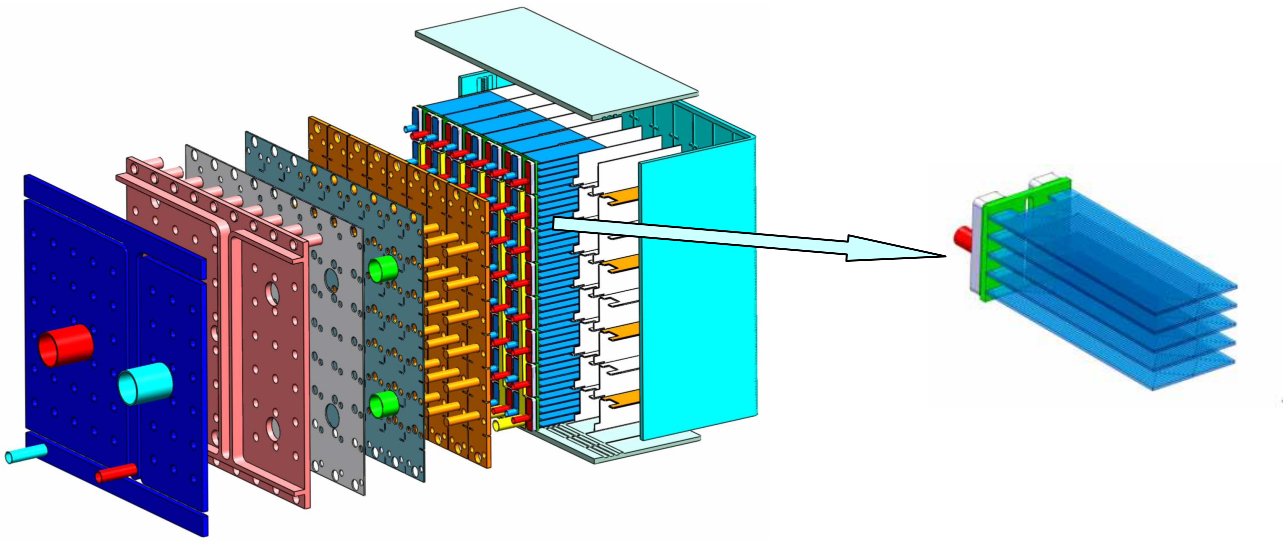

Figure 1: Detailed views of a HCLL blanket module and of a Breeding Unit

These results highlight that in order to have a He-purification plant of a reasonable size, the tritium permeation in the steam generator has to be drastically reduced.

However, the analytical computations described above make the assumption that all the produced tritium inside the LiPb bulk is immediately available for permeation through the Eurofer walls. We can suppose that not taking into account T transport in the LiPb leads to a very pessimistic result (nearly all the produced T escapes into the He coolant, if no permeation barriers are used).
Actually, the tritium has to cross the LiPb bulk before reaching the walls and permeating across them. Moreover, the T transport through the LiPb is not only due to diffusion, but also to the advective movement of the LiPb itself. This movement is the result of the LiPb imposed circulation (to ensure a given recycling rate) and to the buoyancy forces due to thermal gradient and concentration gradient. Furthermore, the velocity field is complex since MHD effects are significant in an electrical conductive material submitted to high magnetic field.
In order to study these effects, we have developed a FEM using the CEA code Cast3M, using a simplified 2D representation of the BU [6].

The tritium concentration c in the LiPb is driven by the mass equation:

$$\frac{\partial c}{\partial t} + V \cdot \nabla c - D_{LiPb} \nabla^2 c = S$$

where V is the LiPb velocity ($m.s^{-1}$)

- D_{LiPb} : the T diffusivity in LiPb ($m^2.s^{-1}$)
- S: the source term (mol $m^{-3}.s^{-1}$)

The source term S is function of the distance r from the first wall, its expression derived using a neutronic Monte-Carlo simulation shows that the tritium production is mainly localised very close to the FW. The boudary condition at the LiPb / Wall interface expresses that T flows are equal in both side of the interface, meanwhile the T concentration at the Wall / He interface is imposed to be zero (negligible T-partial pressure in the He coolant).
Velocity profiles are imposed according to various assumptions in order to test the sensitivity of the results to the flow profile.

We have tested various assumptions, all with velocity oriented along r axis (no turbulence):

- parabolic velocity profile as it would be obtained if no MHD effects were not involved (hydrodynamic profile)

- flat velocity profile with very thin Hartmann layer as it would be obtained on Hartmann wall in MHD flow.

The main restriction of that 2D laminar velocity profiles is that they do not take into account the possible mixing of tritium concentration due to transverse velocity. These non-laminar velocity profiles will occur during expansion and contraction of the LiPb flow, or will be generated by thermally induced rolls or turbulence close to the walls. However, considering the contracted flow across the thin slot close to the FW that ensures the communication between BUs, we can make the assumption that, at this level, a very efficient LiPb mixing occurs. Consequently, the model in composed of two LiPb branches, the input concentration of the backward branch being the mean output concentration of the forward branch.

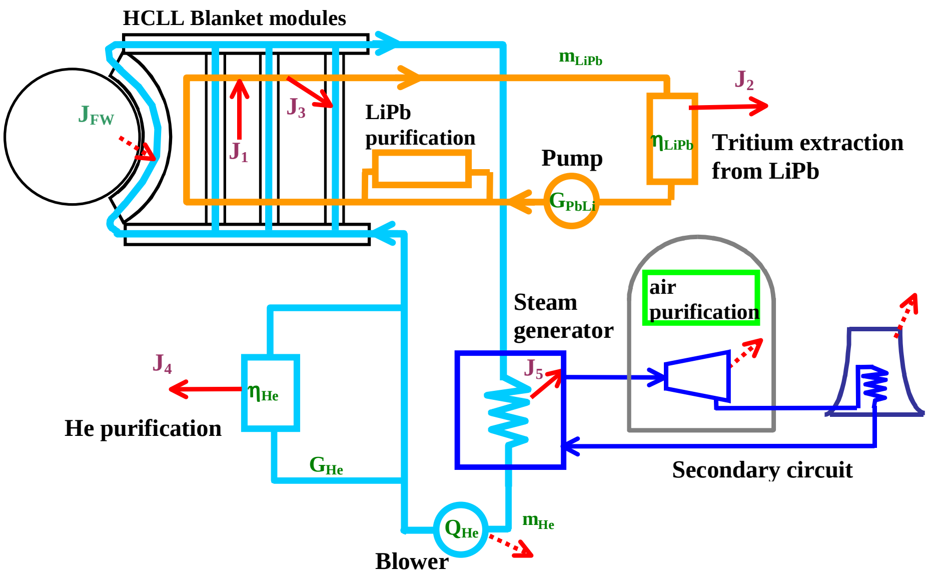

Figure 2: Scheme of the main tritium-related system and corresponding Tritium flow in a HCLL fusion reactor

To take into consideration the efficiency (η_{LiPb}) of the LiPb detritiation unit, the input concentration of the forward branch is made equal to (1 - (η_{LiPb}) times the output concentration of the backward branch.

The interest of FEM comes when we observe the T-concentration stratification in the LiPb bulk and the tritium depleted layer close to the permeating walls that leads to a reduced permeation rate vs. the above results. We can see (figure 3.) that in the middle of the LiPb flow, the T concentration is lower than on the sides, due to a faster velocity. However, even closer to the walls, the concentration falls down due to the permeation. For a flat velocity profile (MHD type), the T concentration increase on the side is no more present, but the tritium depletion close to the walls remains.

The permeation rate vs. the number of recycling per day (figure 4) shows a dramatic reduction of permeation due to the LiPb stratification that can be compared to the benefit of a permeation barrier with a PRF equal to 50 when the LiPb recycling rate is around 10

The sensitivity to velocity profile is here damped due to the concentration mixing between the forward and backward branches. When this mixing is not applied, flat velocity profile experiences lower permeation than parabolic one. The model seems to be more sensitive to the flow rate than to the velocity profile

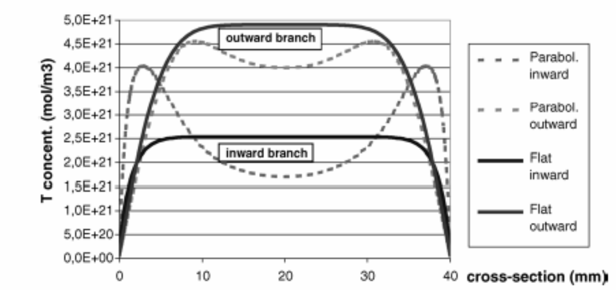

Figure 3: Tritium concentration profiles at outputs across Pb-17Li flow for a parabolic and flat velocity profiles.

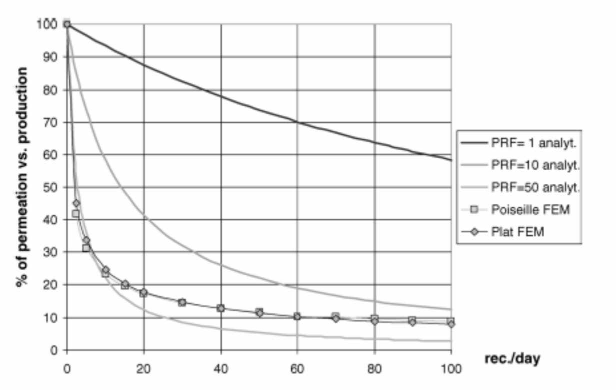

Figure 4: Permeation vs Pb-17Li recycling rate for parabolic (Poiseuille) and flat velocity profiles.

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TW5-TRP-002-D03a

Task Title: ANALYSIS OF CURRENT PROFILE CONTROL IN TOKAMAK REACTOR SCENARIOS

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 establishement and the control of an non-inductively driven current density profile. This task is a first attempt to analyse this problem by means of the integrated modelling code CRONOS. CRONOS 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. The deliverable of this task is a CRONOS simulation of a DEMO steady-state discharge, in which the various current sources combine to yield an MHD-stable current density profile.

2005 ACTIVITIES

D calculations

In parallel to the detailed 1D simulations for DEMO and in order to cross-check the results obtained in the frame of the PPCS study with the PROCESS code, we have used a simple modeling for a commercial power plant design. The model, implemented in the HELIOS code, includes the plasma core, the scrape-off layer and divertor and the power plant energy balance.

- The plasma core is described using a 0D model including the effect of the density and temperature profiles (generalized parabolic profiles $X(\rho) \propto (1-\rho^2)^{\alpha_X}$). The transport losses are described by a global energy confinement time normalized to the IPB98(y,2) scaling law (with an H_H factor). The non-bootstrap part of the plasma current is supposed to be entirely driven by the neutral beam injected power with an efficiency $\gamma_{CD} = 0.35\times10^{19}\langle T_e\rangle$. The bootstrap fraction is supposed to be given by the Hoang fit and the synchrotron radiation losses by the Fidone-Meyer fit with a reflection coefficient of 0.7. The fast alpha pressure is calculated using the Uckan et al formulation. The conducted power P_{net} and the power P_{sep} crossing the separatrix are calculated and compared to the two different ITER scalings for the L-H transition power ($P^{(1)}_{\text{L-H}}$ and $P^{(2)}_{\text{L-H}}$).
- The Helium ash content is calculated supposing $\frac{\tau^*_{He}}{\tau_E} = 5$ where τ^*_{He} is the apparent helium confinement time.
- The total line radiation in the plasma mantle and divertor is supposed to be given by the Matthews law. The repartition of the line radiation power between the plasma mantle and the divertor is taken from JET discharges with high triangularity, high density, good confinement and high fractional line radiation, i.e. 61 % in the plasma mantle and 39 % in the divertor. The part of the power crossing the separatrix which is not radiated in the divertor contributes to the time averaged peak thermal load on the divertor plates according to the Pacher et al fit. This fit is calibrated in order to get the results of a B2-Eirene simulation for ITER with injected Argon.
- The electrical power going to the network is calculated from the exact power plant energy balance. In the special case of the choice of the model C technology, the following parameters are taken:

  $M_B = 1.17$, $\eta_{th} = 42.6$ %, $f_{aux} = 5.13$ %, $\eta_{inj} = 70$ %

  where M_B is the blanket amplification factor, η_{th} is the efficiency of the power plant thermal cycle, f_{aux} is the fraction of the gross electrical power which is used for auxiliary devices (pumps and cryogeny), and η_{inj} is the plug efficiency of the neutral beam injector.

The above model has been first tested by applying it to ITER. Then, the model was applied to PPCS-A, PPCS-C and DEMO-C assuming only He (ashes) and Ar (injected) impurities. Discrepancies with respect to the results of the PROCESS code have been analysed in detail.

Considering our model to be a good preliminary tool for the design of a reactor, taking the assumption of only He and Ar impurities, and taking the power plant efficiencies of the model C technology, we have performed two 1000 MWe DEMO designs. The first one supposes H mode operation with parameters taken from the JET high triangularity discharges described above (this regime will be referred to as JET-like H mode). The second one supposes Advance mode operation with parameters taken from the reference ITER Advanced scenario (this regime will be referred to as ITER-like Advanced mode).

For each of the above physical mode of operation, we consider two different values of the maximum magnetic field in the superconductor: B_{tmax} = 13.6 T which is the value taken presently for PPCS model C and B_{tmax} = 16 T which we consider to be accessible in the future, either with new generation Nb_3Sn at 1.8 K or with HT_c superconductors at 4.2 K. The inner separation between the plasma and the external part of the superconductor is taken to the constant value Δ_{int} = 1.669 m.

It was first shown that, in a tokamak with given A, R, density and temperature profiles (α_n and α_T), τ^*_{He}/τ_E , H_H, $\bar{n}_e/n_G$, q_{peak}, the amplification factor Q as well as the electrical power Pen are decreasing functions of q_{95}. This means that, with these constraints, the best thermonuclear performances in a tokamak are obtained for the maximum allowed plasma current for the regime considered. As a consequence, a q_{95} = 3 safety factor will be chosen for H mode operation reactor design.
For the H mode DEMO, we take the following design assumptions:

$A = 3,\ q_{95} = 3,\ H_H = 1,\ \frac{\bar{\eta}_e}{\eta_G} = 1,\ \alpha_n = 0.01,\ \alpha_T = 1.5,\ q_{peak} = 10\ \mathrm{MW/m^2}$

For each value of the major radius R, the operating point corresponding to the above boundary conditions can be calculated, resulting in a given value of the electrical power Pen delivered to the network. P_{en} = 1000 MW is obtained for R ≅ 8.15 m in the case B_{tmax} = 13.6 T and for R ≅ 6.92 m in the case B_{tmax} = 16 T. The corresponding thermonuclear plasmas exhibit relatively low values of the amplification factor Q (Q = 10.6 for both cases). The corresponding power plant efficiency $\eta_{PP} = P_{en}/P_{nucl}$ is strongly affected by the required circulated power corresponding to these values of Q; we get $\eta_{PP} \cong 31.9$ % to be compared with ηth = 42.6 %. As an illustration, the mains parameters of the 13.6 T, 8.15 m JET-like H mode DEMO are given in the following table:

For the Advanced mode DEMO, we take the following design assumptions:

$A = 3,\ q_{95} = 4{,}5,\ H_H = 1.57,\ \alpha_n = 0.1,\ \alpha_T = 1.5,\ q_{peak} = 10\ \mathrm{MW/m^2}$

It can be shown that, for a given B_{tmax}, the P_{en}(R) curve has a maximum. This phenomenon is due to the strong enhancement of the apparent Helium confinement time when R is increased, leading to very large values of the He fraction and consequent dilution. For $\bar{n}_e/n_G = 0.85$ the maximum is below 1 000 MW for all Bt_{max}. This means than there is no ITER-like Advanced mode DEMO with 1 000 MWe for such a fractional Greenwald density.

For $\bar{n}_e/n_G = 0.95$, the problem has a solution. We get R = 8.78 m for B_{tmax} = 13.6 T and R = 7.33 m for B_{tmax} = 16 T.

The corresponding amplification factor is 20.8 and 21.8 respectively, giving reasonable values of the power plant efficiency, 36.1 % and 36.3 % respectively. The mains parameters of the 16 T, 7.33 m ITER-like Advanced DEMO are given in the following table:

In conclusion, within the frame of our model, a purely non inductive 1 000 MWe DEMO with C technology may be designed, as well in JET-like H mode as in ITER-like Advanced mode. In H-mode, the power plant efficiency is marginally acceptable (Q ~ 11) whereas it is correct in Advanced mode (Q ~ 22).

Table 1:

R (m) / A	8.15 / 3	f_{He} (%)	6.90
V (m^3)	2 000	P_{fus} (MW)	2 760
S (m^2) / L (m)	1 220 / 24.7	P_{add} (MW)	260
I_p (MA)	25.1	Q	10.6
B_t (T)	6.28	f_{BS} (%)	28.1 (α_j = 0.5)
κ_{95} / δ_{95}	1.7 / 0.33	β_N (β_{Nth})	2.67 (2.35)
q_{95} (fit)	3	P_B / P_S (MW)	181 / 102
α_n	0.01	P_{net} / P_{sep} (MW)	535 / 286
α_T	1.5	$P^{(1)}_{L\text{-}H}$ / $P^{(2)}_{L\text{-}H}$ (MW)	80.3 / 147
$\bar{n}_e/n_G$	1	P_{line} (MW)	408
$\langle n_e \rangle$ (10^{20} m^{-3})	1.08	P_{line}/P_{net} (%)	76.3
τ^*_{He}/τ_E	5	$P_{non\text{-}rad}$ (MW)	127
f_{Ar} (%)	0.870	q_{peak} (MW/m^2)	10
Z_{eff}	3.80	P_{en} (MW)	1 000
$\langle T_e \rangle$ (keV)	17.5	η_{PP} (%)	31.9

Table 2:

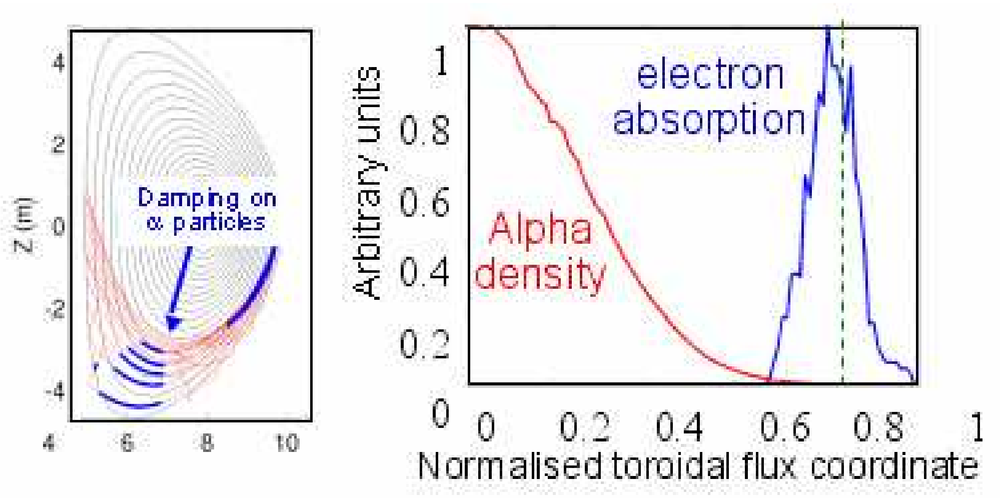
R (m) / A	7.33 / 3	f_{He} (%)	13.9
V (m^3)	1 660	P_{fus} (MW)	2 430
S (m^2) / L (m)	1 080 / 24.8	P_{add} (MW)	111
I_p (MA)	21.5	Q	21.8
B_t (T)	7.03	f_{BS} (%)	49.6 (α_j = 0.5)
κ_{95} / δ_{95}	1.87 / 0.47	β_N (β_{Nth})	3.13 (2.78)
q_{95} (fit)	4.5	P_B / P_S (MW)	112 / 188
α_n	0.1	P_{net} / P_{sep} (MW)	303 / 182
α_T	1.5	$P^{(1)}_{L\text{-}H}$ / $P^{(2)}_{L\text{-}H}$ (MW)	80.2 / 142
$\bar{n}_e / n_G$	0.95	P_{line} (MW)	199
$\langle n_e \rangle$ (10^{20} m^{-3})	1.05	P_{line}/P_{net} (%)	65.5
τ^*_{He}/τ_E	5	$P_{non\text{-}rad}$ (MW)	104
f_{Ar} (%)	0.404	q_{peak} (MW/m^2)	10
Z_{eff}	2.51	P_{en} (MW)	1 000
$\langle T_e \rangle$ (keV)	21.4	η_{PP} (%)	36.3

D calculations

Predictive simulations of DEMO scenarios by means of the CRONOS code have been prepared by the development of a new module for the computation of the synchrotron radiation loss profile, an essential ingredient for the physics of a high-temperature fusion reactor. This work has been done in collaboration with the Polytechnic University of Barcelona (Spain), where the EXATEC module has been developed. This module has now fully coupled to CRONOS and yields the detailed radial profile of the loss term associated with synchrotron radiation.

A second preparatory study consists in the calculation of the fraction of Lower-Hybrid power absorbed by alpha particle in a typical (advanced) DEMO scenario, in which LH waves are expected to be one of the main tools to generate a transport barrier. This is important for the choice of the wave frequency to be used.

The calculation has been done by CRONOS, with a DEMO 2-D equilibrium computed by the magnetic equilibrium code Helena, an alpha particle distribution function computed by the Monte Carlo code SPOT, and LH wave propagation computed by the ray-tracing code Delphine. It has been found that for the same frequency chosen for ITER (5 GHz), the alpha particle absorption is less than 1%, which implies that a very similar technology can be employed in ITER and DEMO. The ray trajectories and the profiles of alpha distribution and electron absorption are shown in the following figure.



CONCLUSIONS

The preliminary part of this task, consisting of the 0-D design and the development of important modules of the CRONOS suite of codes, has been completed in 2005. The code is now ready to be exploited to find a viable scenario of steady-state operation for DEMO. 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.

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