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## **An overview of dual coolant Pb-17Li breeder first wall and blanket concept development for the US ITER-TBM design**

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**Abstract.** An attractive blanket concept for the fusion reactor is the dual coolant Pb-17Li liquid (DCLL) breeder design. Reduced activation ferritic steel (RAFS) is used as the structural material. Helium is used to cool the first wall and blanket structure, and the self-cooled breeder Pb-17Li is circulated for power conversion and for tritium breeding. A SiC<sub>f</sub>/SiC composite insert is used as the magnetohydrodynamic (MHD) insulation to reduce the impact from the MHD pressure drop of the circulating Pb-17Li and as the thermal insulator to separate the high temperature Pb-17Li from the helium cooled RAFS structure.

For the reference tokamak power reactor design, this blanket concept has the potential of satisfying the design limits of RAFS while allowing the feasibility of having a high Pb-17Li outlet temperature of 700°C. We have identified critical issues for the concept, some of which include the first wall design, the assessment of MHD effects with the SiC-composite flow coolant insert, and the extraction and control of the bred tritium from the Pb-17Li breeder. R&D programs have been proposed to address these issues. At the same time we have proposed a test plan for the DCLL ITER-Test Blanket Module program.

**Key words:** blanket, Pb-17Li breeder, dual-coolant, helium-cooled, ITER-TBM

## **1. Introduction**

In support of the ITER Test Blanket Module (TBM) program, we have been focusing on the dual coolant Pb-17Li liquid breeder (DCLL) blanket design, a concept that has been explored extensively in the US [1,2] and by the European Union [3]. With the use of reduced activation ferritic steel (RAFS) as the structural material we are limited to a maximum steel structure temperature of  $<550^{\circ}\text{C}$ . At the same time we have to remove the first wall heat flux, breed adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency. After a period of assessment, we have selected the DCLL concept [4] to achieve these design requirements. The basic approach of the DCLL concept is shown in Fig. 1. A key element in the approach is the use of the  $\text{SiC}_f/\text{SiC}$  composite (SiC-composite) flow channel insert (FCI) [5]. This FCI element performs the key functions of reducing the magnetohydrodynamic (MHD) effect of the circulating self-cooled Pb-17Li breeder and thermally isolating the high temperature Pb-17Li in the main channel from the low temperature RAFS structure, which is cooled by helium in the smaller channels. The Pb-17Li is flowing in the larger channel at low velocity in order to achieve high outlet coolant temperature. This paper presents the development of our DCLL design, with focus on summarizing the following topics: applying the concept to our reference reactor design, design of the ITER-TBM, the first wall design, the MHD effect with the FCI and the handling of tritium extraction and control at high temperature. Our initial TBM test plan is also summarized.

## **2. DCLL for Reference Reactor Design**

To select an ITER-TBM first wall and blanket design, we have to make sure that the concept can provide adequate performance when it is extrapolated to a high performance

tokamak reactor design. We performed such an assessment and the corresponding reference reactor design parameters are given in Table 1 [4].

An outboard poloidally segmented blanket module for the reference reactor design is shown in Fig. 2. It has a two-pass poloidal Pb-17Li flow. The Pb-17Li inlet and outlet temperatures are 460°C and 700°C, respectively. The 8 MPa helium cools the first wall, and all the RAFS structure has helium inlet and outlet temperatures of 300°C and 480°C, respectively. Concentric pipes are used for both Pb-17Li and helium coolants circulating in and out of the blanket module. Neutronics calculations have been performed to determine the important nuclear performance parameters. The overall tritium breeding ratio is estimated to be 1.15 with  $\text{Li}^6$  enrichment at 90% and excluding any breeding in the divertor region. With the use of multiple-reheat Brayton closed cycle gas turbine power conversion system [4,6,7] a gross power conversion efficiency of >40% can be projected.

### **3. DCLL ITER-TBM Design**

The DCLL ITER-TBM design is shown in Fig. 3. We have selected a similar blanket configuration as for the reference tokamak reactor design shown in Sec. 2. Design parameters for the ITER-TBM are presented in Table 2.

### **4. TBM First Wall Design**

Figure 4 shows details of the counterflow first wall design. Power loading density distributions in the blanket components were generated by neutronics calculations. As shown, the helium coolant is designed to pass the first wall five times before leaving the first wall panel. The temperature increase for each pass is indicated. In order to maintain the first wall maximum allowable temperature of <550°C, the first wall cooling channel surface on the plasma side will have to be roughened in order to enhance the heat transfer

coefficient. We are continuing to assess other helium flow characteristics like system pressure drop, flow distribution and stability.

## **5. TBM Neutronics Results**

Initial 1-D neutronics calculations for the TBM were performed. With the limited geometry of the TBM in the ITER machine, the local tritium breeding ratio is only 0.664 with 30 cm module depth. Tritium is generated at the rate of  $1.43 \times 10^{-6}$  g/s in the TBM. Nuclear heating in different components of the TBM was evaluated and used for the thermal-hydraulics assessment as shown in Sec. 4.

## **6. MHD Effect of the FCI**

A key element in the DCLL is the SiC-composite FCI. With the possibility of tailoring the electrical and thermal properties of the SiC-composite material by controlling the manufacturing process of the composite, we performed a detailed MHD analysis to determine the design window of critical FCI material properties [8]. The analysis was performed for the front poloidal channel under the reference reactor design conditions. Flow equations coupled with the induction equation were solved for the FCI design assuming fully developed flow conditions with MHD code specially developed for such flows [9]. Two possibilities of equalizing the Pb-17Li fluid pressure between the main fluid channel and the gap between the SiC-composite insert and the RAFS wall were tested. These are the cases for the use of pressure equalization slots (PESs) and pressure equalization holes (PEH). Velocity profiles for these two cases are shown in Fig. 5 as a function of the SiC-composite electrical conductivity. As shown, in the core the fluid motion is suppressed, yet jet flows can be noticed on both sides of the channel parallel to the magnetic field and in the adjacent gap sections. We can see that higher electrical

conductivity will lead to higher jet velocity on the edges and therefore enhance the heat transfer. For the PES case, a significant amount of induced electric current will flow through the slot. As a result, inverse flow appears in the slot area. In comparison, from the flow stability point of view, the PEH design is preferred.

Similar calculations were performed to assess the temperature distribution as a function of the thermal conductivity of the FCI. Results show that due to the volumetric power generation of the SiC-composite, when the thermal conductivity reaches a value of  $k = 2 \text{ W/m.K}$ , the energy balance between the SiC-composite and the Pb-17Li reaches a no heat leakage condition across the SiC-composite. For the MHD pressure drop of the slowly moving Pb-17Li, based on the MHD modeling and with the use of SiC-composite FCI, the MHD pressure drop is reduced by 50-100 times when compared to the case without the use of FCI. This factor would become smaller when 3-D MHD effect is taken into consideration. In summary the recommended property ranges for SiC-composite are:  $\sigma = 20\text{-}50 \text{ 1}/\Omega\text{-m}$  and  $k = 2\text{-}5 \text{ W/m-k}$ . Vendors have been contacted to demonstrate the feasibility of manufacturing SiC-composite with the specific range of property.

## **7. Tritium Extraction and Control at High Temperature**

Pb-17Li has low tritium solubility; therefore the tritium mobility in the DCLL blanket system is high. For the DCLL design it becomes critical to have an efficient method of extracting tritium from the Pb-17Li in order to minimize the amount of tritium release to the public. Figure 6 shows the proposed Pb-17Li loop schematic, starting from the Pb-17Li concentric tube to the vacuum permeator tritium extraction station, then to the heat exchanger before the Pb-17Li is returned to the blanket. For the reference reactor design, the outlet PbLi has a maximum temperature of  $700^\circ\text{C}$ . The critical element of this loop is the vacuum permeator tritium extraction system, with the Pb-17Li flowing in thin wall

tubes and the tritium being extracted via high vacuum outside of the tube bank. Single tube vacuum permeator performance and the corresponding tritium loop equilibrium calculation, utilizing the tritium migration analysis program (TMAP) [10], were performed. The focus was on the required length of the permeator tube in order to reduce the tritium concentration in the PbLi to an acceptable level. In summary, it was found that with a Nb tube (1 cm diameter and wall thickness of 0.5 mm) and length of 4 m, an exit tritium pressure of 0.2 Pa can be achieved for a mass transfer coefficient of 5.6 m/s. An acceptable tritium inventory of less than 100 gm per quadrant of the reference tokamak reactor could be achieved. These results show the potential feasibility of the vacuum permeator extraction approach, but many basic R&D issues remain, including the control of the reaction between Nb and oxygen impurities in the vacuum side. Similarly, we will have to assess the suitable heat exchanger material for high temperature Pb-17Li and helium. Nb and other refractory alloys are candidate materials to be evaluated.

## **8. Test Plan**

For the ITER-TBM, the initial test plan is shown in Fig. 7. As envisioned, we are planning to have four test modules with progressive integrated testing of different elements of the DCLL design. The first MHD module will be inserted at day one of ITER operation. The last module would be similar to the integrated test module discussed in this paper.

## **9. Conclusions**

We have completed our assessment and selection of the DCLL as the first wall and blanket concept for the ITER-TBM. For the reference tokamak power reactor design, this blanket design concept has the potential of satisfying the design limits of RAFS while allowing the feasibility of having high Pb-17Li outlet temperature of 700°C. Using the

Brayton cycle power conversion system, the project gross thermal efficiency is >40%. We have begun the design of the DCLL ITER-TBM design including the assessment of critical issues. Some of these include the first wall design, the assessment of MHD effects with the SiC-composite FCI design, and the extraction and control of the bred tritium from the Pb-17Li breeder. The suitable range of electrical and thermal conductivity for the SiC-composite are  $\sigma = 20-50 \text{ 1}/\Omega\text{-m}$ , and  $k = 2-5 \text{ W/m-k}$ , respectively. The vacuum permeator tritium extraction system seems to be credible, but basic R&D issues need to be addressed. At the same time we have formulated the DCLL ITER-TBM test program. We are considering four progressively more integrated test modules during the first 10 year phase of ITER operation. The first module will be tested during day one of ITER operation.

### **Acknowledgment**

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## **List of Figure Captions**

Fig. 1. Cross section of the Pb-17Li breeder unit cell of DCLL (dimensions are in mm).

Fig. 2. DCLL configuration for the Reference Tokamak Reactor Design.

Fig. 3. DCLL ITER-TBM module design schematic.

Fig. 4. Counterflow helium cooled TBM first wall and corresponding input and output parameters.

Fig. 5. Pb-17Li flow distribution in the flow channel as a function of electrical conductivity of SiC-composite FCI for the cases of PES and PEH. (Magnetic field is applied in the z-direction.)

Fig. 6. The Pb-17Li loop showing the concentric tubes, the vacuum permeator for tritium extraction and the PbLi to helium heat exchange.

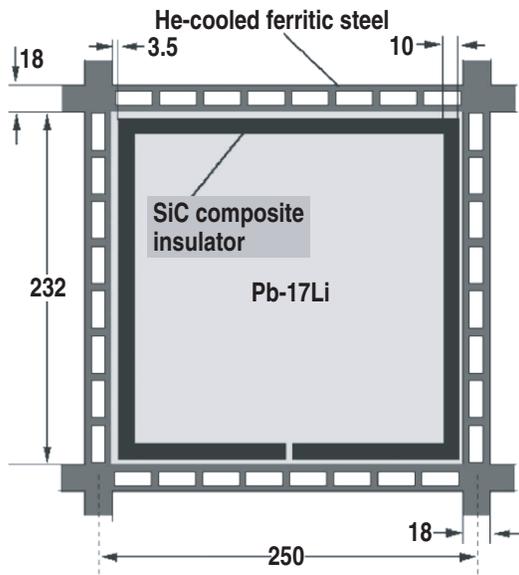
Fig. 7. DCLL TBM testing during the first phase of ITER.

Table 1 Reference Tokamak Reactor Design Parameters

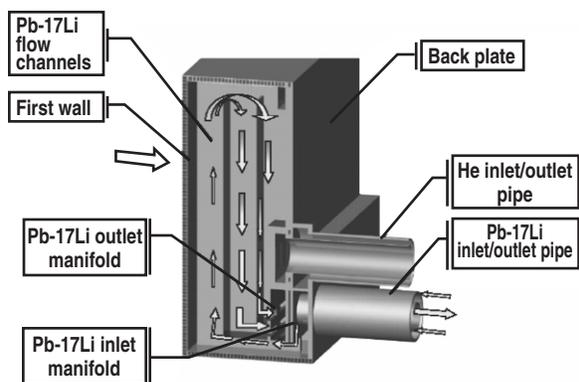
Major radius	5.8 m
Aspect ratio	2.6
Plasma elongation	1.9
Fusion power	2116 MW
Peak outboard neutron wall loading	3.72 MW/m <sup>2</sup>
Average chamber neutron wall loading	2.13 MW/m <sup>2</sup>
Peak surface heat flux at outboard midplane	0.5 MW/m <sup>2</sup>

Table 2 ITER-TBM Design Parameters

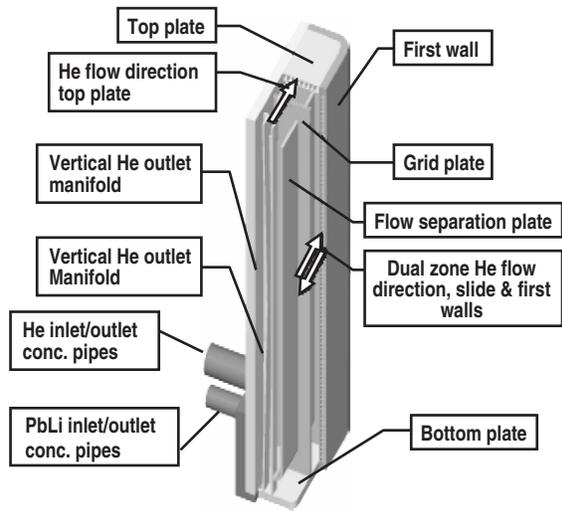
Fusion power	500 MW
Design heat flux	0.5 MW/m <sup>2</sup> for 10% of the area 0.3 MW/m <sup>2</sup> for 90% of the area
Design neutron wall loading	0.78 MW/m <sup>2</sup>
Disruption heat load	0.55 MJ/m <sup>2</sup>
Plasma burn time	400 s
Time between shots	2000 s
Duty factor	0.25
Half module frontal dimensions	64.5 cm (width) by 194.0 cm (height)
Radial thickness	~0.3 m



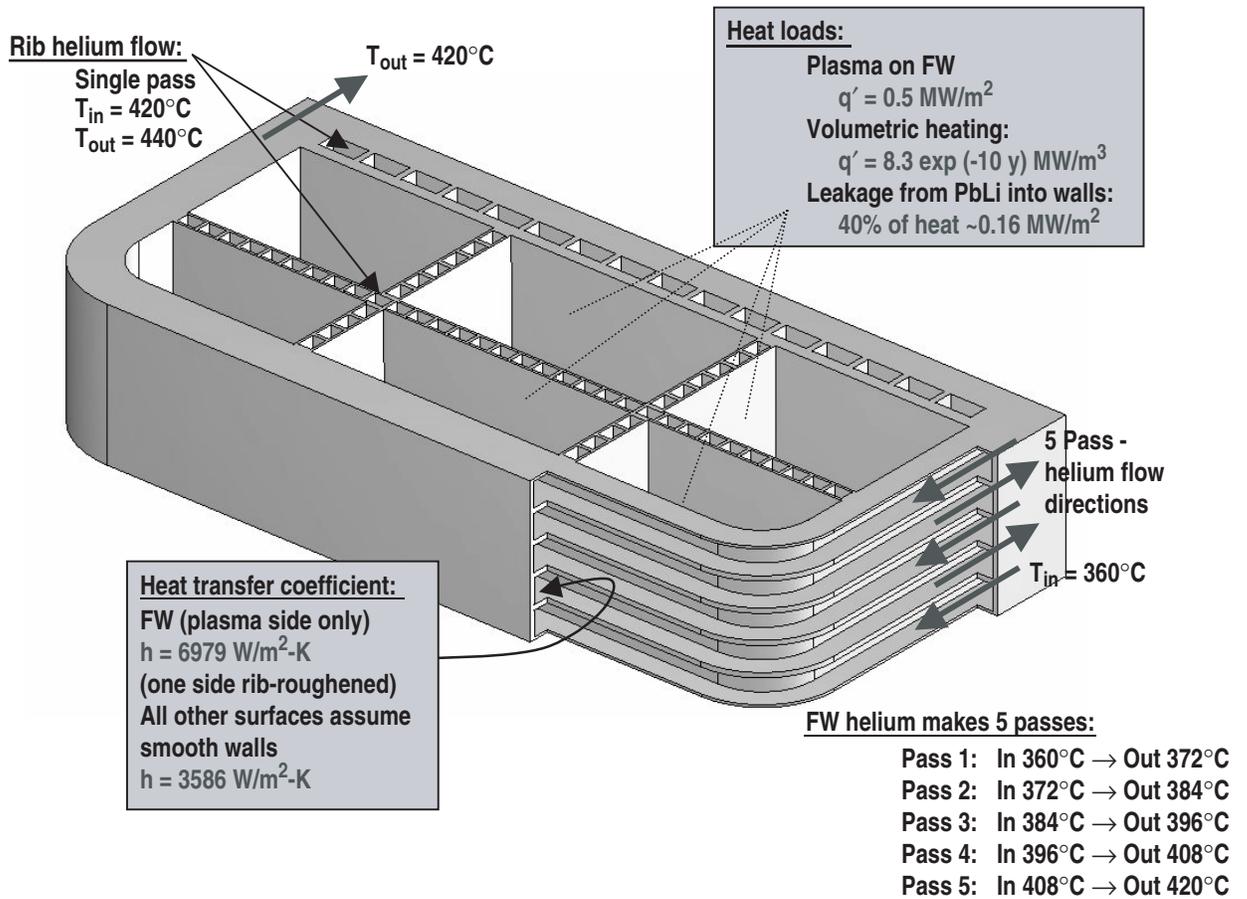
C.P.C. Wong Figure 1



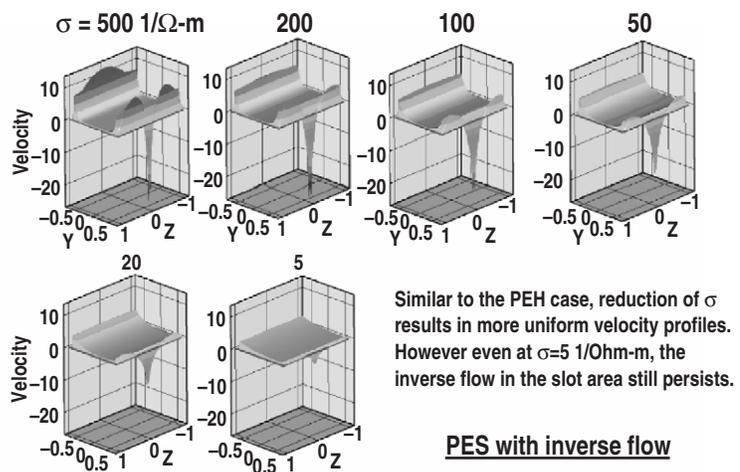
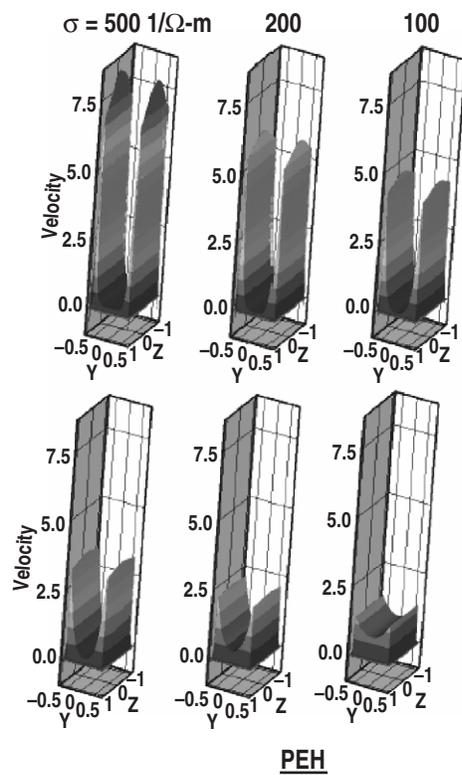
C.P.C. Wong Figure 2



C.P.C. Wong Figure 3

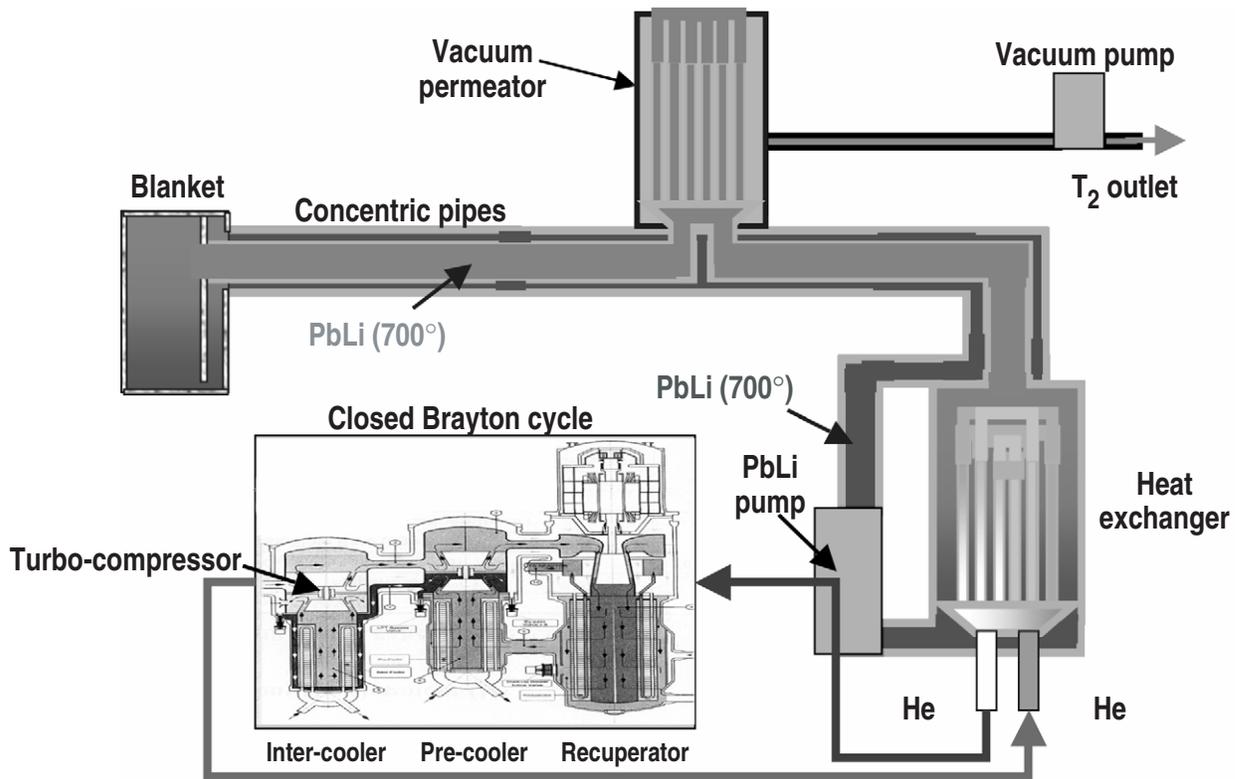


C.P.C. Wong Figure 4



$\sigma$ (1/Ω-m)	PEH: dP/dx (MPa/m)	PES: dP/dx (MPa/m)
500	0.0054	0.0055
200	0.0026	0.0030
100	0.0014	0.0019
50	0.0008	0.0011
20	0.0003	0.0006
5	0.0001	0.0002

dP/dx at different  $\sigma$  for PEH and PES



C.P.C. Wong Figure 6

## DCLL TBM Testing During 1st Phase of ITER

Year 0	Year 1	Year 2	Year 3	Year 4	Year 5	Year 6	Year 7	Year 8	Year 9	Year 10
<b>MHD TBM</b> - Ancillary system and TBM checkout - Initial fluid flow and MHD tests - FW and EM tests	<b>Full B-field and surface heat flux</b>			<b>Full nuclear heating</b>			<b>Fluence accumulation</b>			
	<b>EM/Struct/Neutronics TBM</b> - Magnetic field part. due to FS - Structure deformation and reliability in FW heat flux and EM loads under tokamak operation - Ferromagnetic effects on main field - Some MHD flow tests - Tritium prod. and neutron flux									
	<b>Thermofluid-MHD TBM</b> - FCI thermal performance and scaled FCI stress - Initial flow corrosion - Tritium permeation									
<b>First Plasma</b>		<b>Down time?</b>			<b>Integrated TBM</b> - Corrosion and compatibility in nuclear field - Tritium control - Tritium breeding and neutronics - Integrated performance - Preliminary reliability				<b>Integrated TBM</b> - Radiation damage - ~1 dpa in the_FCI and structure	

C.P.C. Wong Figure 7