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Abstract. **Within the framework of the EU Power Plant Conceptual Study (PPCS), also the design of the Dual-coolant Blanket (the so-called model C) was assessed technologically. A brief summary of the results shall be given below, starting with some very important design requirements to be fulfilled and a general description of the concept. This work is performed under the coordination of Forschungszentrum Karlsruhe in co-operation with CEA, EFET/IBERTEF, UKAEA, VR, and VTT.**

I. INTRODUCTION

As a strategy of the EU PPCS, four different power plant models are investigated: Model A: Based on a water-cooled lead-lithium (WCLL) blanket using a ferritic/martensitic steel EUROFER as structural material and a water-cooled divertor with a CuCrZr alloy as structural material; Model B: Based on a helium-cooled ceramics/beryllium pebble bed (HCPB) blanket using EUROFER as structural material and a He-cooled divertor with W alloys as structural material; Model C: Based on a dual-coolant (DC) blanket (bulk self-cooled lead/lithium + He-cooled structures) using EUROFER as structural material and SiC$_f$/SiC as thermal and electrical insulators, and a He-cooled divertor made of W alloys and EUROFER as structural materials; Model D: Based on a self-cooled lead/lithium (SCLL) blanket and lead-lithium-cooled divertor, both made of SiC$_f$/SiC structural material.

Model C [1] is a compromise of the "near-term" models A and B with their limited attractiveness and the "very advanced" model D which is characterized by very attractive features, but considerable development risks. It offers the following advantages related to the DC design: a) Simple construction, i.e. less complexity, smaller heat transfer area, leading to a high reliability, b) a higher coolant temperature than that of the structure is attainable by circulating the liquid metal coolant with internal heat sources and by the use of SiC$_f$/SiC flow channel inserts (FCIs), c) less permeation safety issues due to the possibility of using the gas turbine cycle. Further advantages which are typically related to the use of liquid-metal breeders are: a) No nuclear damage of the liquid-metal breeder, b) replenishment of the Li-6 breeder material can be done easily online, c) only the structures and inserts have to be exchanged or recycled, d) relatively small thickness of shielding for vacuum vessel (VV) and magnet coils is required when using a water-cooled low-temperature shield.

The PPCS refers to a standardized commercial power plant with a unit size of 1500 MWe.

II. THE CONCEPTUAL DESIGN OF MODEL C

A. Physics of the Reactors and Design Requirements

Models C and D are based on advanced physical assumptions characterized by: a) high β and high confinement, with realistic plasma pressure gradients, b) MHD stabilization by strong plasma shaping, c) high bootstrap current fraction, d) low divertor power loads and low Zeff, no ELMs are foreseen in reactor operation. Analysis with the PROCESS code [2], [3] shows that, indeed, the above assumptions lead to a high Q, reduced-size reactor, high bootstrap current fraction, and reduced plasma current when compared to models A and B, with nuclear loads limited to < 2.5 MW/m². Also, the heat load to the divertor could be reduced to 5 MW/m² (model D). In all cases, the net power plant output to the grid is 1500 MWe and the D-T fuel mix is 50-50.

The following overall design requirements and criteria are imposed: a) Easy replacement of blanket and divertor modules, b) sufficient shielding of VV (for reasons of reweldability) and magnets to make them to life-time components, c) low volumetric fraction of steel to enhance the breeding ratio (TBR>1), d) use of oxide dispersion-strengthened (ODS) steel limited to the plasma-facing zone of the first wall with highest temperature, e) sufficiently high coolant inlet temperature to avoid material embrittlement (DBTT) under irradiation, and sufficiently high coolant outlet temperature to allow the use of a BRAYTON cycle power conversion system, f) to minimize the tritium permeation loss, purification and tritium extraction systems for liquid metal have to be foreseen.
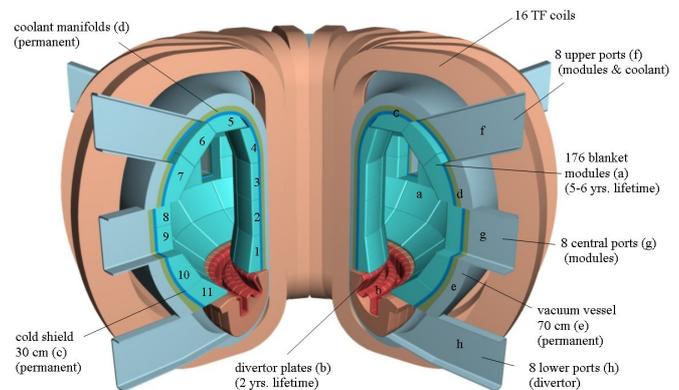

Fig. 1. Fusion reactor with DC blanket modules (model C).

B. Brief Design Descriptions of the Blanket and Divertor

Blanket: The DC concept which is based on the study in [4] and the ARIES-ST study [5] is characterized by the use of self-cooled breeding zones with the eutectic lead-lithium alloy Pb-17Li serving as breeder and coolant at the same time, the use of a helium-cooled reduced-activation ferritic/martensitic (RAFM) steel structure like EUROFER, and the use of SiC$_f$/SiC FCIs serving as electrical and thermal insulators. The plasma-facing surface of the first wall is plated with a small layer of ODS. Instead of the "banana segments" adopted in earlier studies [6], [7], the blanket segmentation now consists of "large modules" [8], which helps to reduce thermal stresses and to better cope with the forces caused by disruptions, while

maintenance is facilitated. A total of 11 modules, five of them at the inboard and six at the outboard (Fig. 1), form a 7.5° segment of the torus. They are large, stiff boxes with a helium-cooled grid structure inside forming flow channels for the Pb-17Li (Fig. 2). The modules are divided into a lifetime part (cold shield, coolant manifold, and vacuum vessel) and a breeding and hot shield part which will be exchanged at 6-years' intervals. For cooling the entire blanket structure, high-pressure (8 MPa) helium gas is used. Two separate He systems provide for a redundancy of decay heat removal. Counter-flow manifolds ensure a uniform temperature distribution to minimize thermal stresses. The inlet temperature of the helium amounts to 300 °C, the outlet temperature to 480 °C. Moreover, the liquid-metal breeder Pb-17Li also serves as a coolant. It enters the modules at 460 °C and leaves them at 700 °C, a value that is maximized for efficiency reasons. Primary coolant loop and manifold are provided with concentric tubes, with the "hot outlet coolants" being in the inner tube (for the Pb-17Li coolant in particular, this tube is thermally insulated with SiC_f/SiC inserts), and the "cold inlet coolants" flowing through the annular channels.
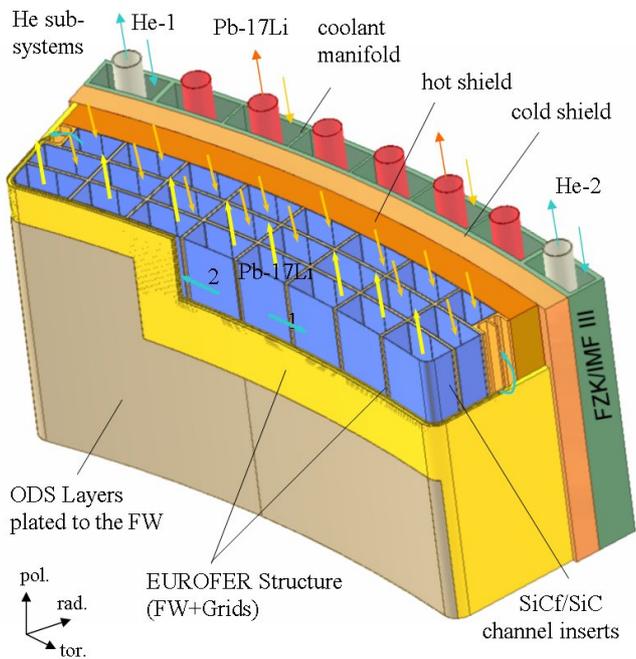

Fig. 2. DC equatorial outboard blanket module (radxtorxpol: 1.5x3.0x1.6 m³).

Divertor: About 15% of the heat energy are released into the divertor which also serves as a trap for plasma impurities. A high heat flux of 10 MW/m² at least is assumed to hit the divertor target plates. For divertor cooling, helium gas is preferred, because it is compatible with other materials, which is of major relevance with respect to safety, and, therefore, ensures a good integration of the divertor in the power conversion system. A high helium outlet temperature is favorable to increase thermal efficiency. A modular design and small temperature gradients are recommended to reduce thermal stresses. The main design criteria are: a) The lower boundary of the divertor operation temperature window has to be higher than the ductile-brittle transition temperature (DBTT), and b) the temperature at the upper boundary has to be lower than the recrystallization limit of the structural components made of refractory alloys under irradiation, c)

high heat transfer coefficients must be ensured, while the coolant mass flow rate and, therefore, the pressure loss as well as the pumping power have to be as low as possible. The proposed design of a He-cooled modular divertor with integrated pin array (HEMP) consists of target plates divided into small modules. Underneath each tile of tungsten used as thermal shield, a finger-like heat transfer module (thimble) is brazed on. A pin array as flow promoter is integrated at the bottom of the thimble to increase the cooling surface. Helium at 10 MPa and 700 °C enters the pin array at high velocity to cool the target plates and is heated up to about 800 °C.

Main neutronic and thermohaulic data as well as the power balance of the DC blanket concept are summarized in Tab. I.

C. Neutronic Analysis and Shielding Efficiency

Distribution of neutron wall loading was calculated with the MCNP code. More than 90% of the fusion neutron power are loaded on the blanket modules, while the remainder flows through the divertor opening. Concerning volumetric heating, a major fraction of $\cong$ 80% of the volumetric heat power is generated in the blanket segments, including the first wall. With the DC reference design, $\approx$ 4% are generated in the water-cooled low-temperature (LT) shield. This power may not be utilized for electricity production and, therefore, must be minimized, e.g. by enhancing the shielding capacity of the high-temperature shield. Regarding the shielding efficiency, there are two essential requirements that must be fulfilled: a) Reweldability of lifetime components made of steel and b) sufficient protection of the superconducting toroidal field (TF) coils. Based on existing data, the current assumption is that rewelding of stainless steel should be successful at helium concentrations below 1 appm. Calculations to estimate the helium production in EUROFER steel show that even after a lifetime cycle of 40 years, reweldability is achieved. Hence, the LT shield may be designed as a lifetime component, if welded joints are placed on its rear. The TF coil, on the other hand, is protected from the penetrating radiation by the blanket, the shield, and the vacuum vessel. An efficient neutron moderator (water or a hydride) is required to this end, combined with a good neutron absorber (steel, tungsten, etc.). The radiation loads of the TF coils were calculated for the inboard mid-plane, where the shielding requirements are highest due to the minimum space available between the plasma and the TF coil. It is noted that the design limits are met by the DC reference design.

D. Thermohaulic, Thermomechanic, and MHD Analyses

The layout of the blanket and the divertor requires iterations between system code analysis and blanket layout concerning neutronic, thermohaulic, thermomechanical, MHD, and velocity field calculations to determine a set of reactor parameters. For the ODS layer on the first wall, the maximum temperature should stay below 650 °C due to creep rupture, while the interface temperature of the EUROFER structure and Pb-17Li should be below 500 °C due to corrosion. For thermomechanical calculations, data of T91 were used instead of ODS data. The results show that the requirements can be fulfilled.

Also for the divertor, an assessment of temperatures and stresses was undertaken. Structural design criteria as required by the ITER structural design code are met, i.e. mechanical stresses do not exceed design limits under any operation

condition. From these values, it is expected that fatigue of some anticipated 100-1000 cycles of reactor shutdown with cooling down from operation conditions to RT is permissible. In a first thermohydraulic assessment of the divertor, a heat transfer coefficient of approx. 60 kW/m²K related to the surface area of the basis plate was determined with a corresponding pumping power of about 5.5% related to the heat removal.

MHD analysis shows that SiC_f/SiC inserts with a wall thickness of 5 mm already ensure a sufficient electric insulation, resulting in a small pressure drop in the Pb-17Li channels and, thus, a small pumping power for the liquid-metal coolant.

E. Power Conversion System

For safety reasons (chemical reaction between water and liquid metal) and to attain a high thermal efficiency, a Brayton cycle (closed-cycle helium gas turbine) is considered as the reference concept. Thus, tritium permeation losses to the environment can be minimized. Four parallel closed three-compression-stage Brayton cycles are used leading to a thermal net efficiency of about 43%.

F. Tritium Recovery and Purification Systems for Pb-*17Li* and He Cooling Loops

The requirements on the tritium removal and recovery system are to keep the tritium inventory low in the total blanket system and to limit the tritium losses to the environment to an acceptable value. This mainly refers to the tritium which permeates through the walls of the heat exchanger and intercoolers into the water. These losses may easily be limited to acceptable values due to the low temperatures (maximum helium temperature ≈ 300 °C, water temperature ≈ 30 °C) in these components.

Several methods were proposed and assessed for tritium removal from Pb-17Li. During the breeding process, also helium is produced. Due to its low solubility in Pb-17Li, bubbles will be formed. It might be straightforward to combine helium bubble removal with the tritium removal system discussed, since some of the methods might be also efficient for helium bubble removal. Finally, liquid-metal purification systems are also required in general to control the oxygen content of the system and remove corrosion products. For irradiated Pb-17Li, additional removal of heavy metal isotopes (Po, Hg, Ti) will be necessary.

Several helium loops cool the different systems of the plant. Helium gas has to be cleaned regularly so as to remove gaseous and radioactive impurities (especially tritium). The coolant purification system will also serve as a means of pressure control.

G. Balance of Plant and Fusion Power Plant Layout

This section deals with additional components to set up an entire operational and functioning plant for generating electric power. The following main systems were considered: a) Primary heat transport system, b) power conversion cycle, c) service water system for cooling auxiliary systems, d) component cooling water system to supply selected auxiliary components with cooling water. The component cooling water system acts as an intermediate barrier between the circulating water system and potentially radioactive cooling loads. Thus, the possibility of radioactive leakage to the environment is

reduced, e) circulating water system: The circulating water system provides for a continuous supply of cooling water to the heat rejection heat exchanger, the intercoolers, the component cooling water heat exchanger, and the service water system, f) water treatment plant, g) compressed-air system, h) fire protection, i) electrical power, j) HVAC system: The HVAC system provides for the ventilation and air conditioning of different plant buildings.

The whole site of the power plant (Fig. 3) consists of several buildings to house the reactor, auxiliary systems, the power supply, and the turbines, but also workshops and offices. The design of the tokamak building (Fig. 4) and the hot cell building will be further evaluated, based on the design of the ITER site.
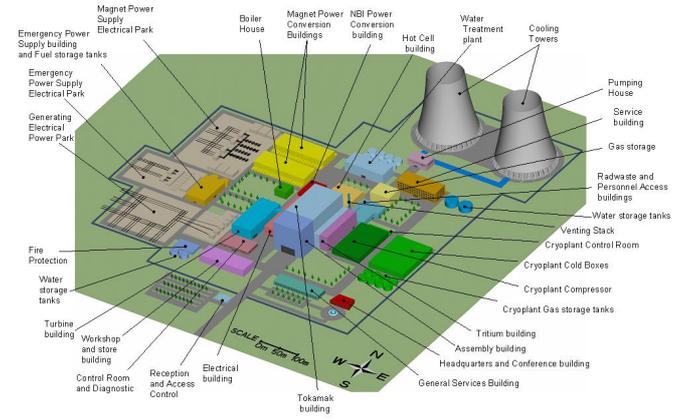

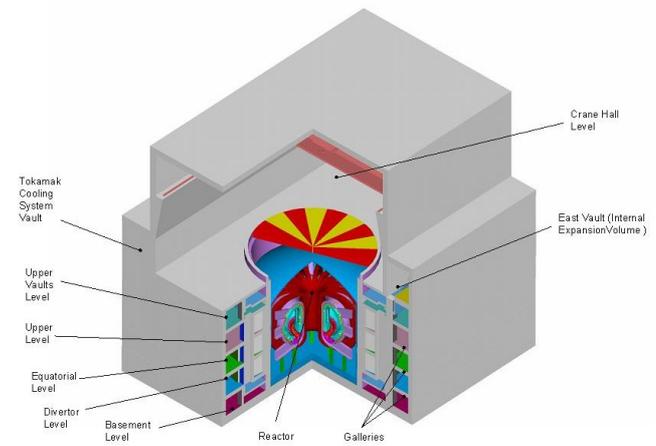Fig. 3. Fusion power plant, general layout.

Fig. 4. Tokamak building, general view.

III. Conclusions and Outlook

The technologies and plasma physics models employed for model C represent a compromise of the "near-term" models A and B with their limited attractiveness and the "very advanced" model D with its very attractive features, but considerable development risks. Model C is based on a self-cooled lead-lithium breeding zone and a helium-cooled structure made of the reduced-activation ferritic steel EUROFER as well as on a helium-cooled divertor. Flow channel inserts made of SiC_f/SiC composite in the lead-lithium channels serve as thermal and electrical insulators in

order to minimize magneto-hydrodynamic (MHD) pressure loss and obtain high coolant exit temperatures and, thus, a high efficiency of the power conversion system, for which the BRAYTON cycle (closed-cycle helium gas turbine) is used. The modular designs of the DC and the He-cooled divertor are addressed. Within the framework of this study, a self-cooled liquid-metal breeder blanket is assessed for a standardized commercial power plant with a typical unit size of 1500 MWe. The overall results of this study lead to the conclusion that the plant model C has a high potential of meeting the goal of fusion research, i.e. to develop an economically and environmentally attractive energy source.

TABLE I
MAIN DATA OF THE DC BLANKET CONCEPT

	Overall plant	
Electrical output [MW]	1500	
Fusion power [MW]	3410	
	Blanket	Divertor
Average neutron wall load [MW/m^2]	2.27	1.7
Max. neutron wall load [MW/m^2]	3.0	
Average surface heat load [MW/m^2]	0.45	0.67
Max. surface heat load [MW/m^2]	0.59	10
Alpha-particle surface power [MW]	546	136
Heating power [MW]		112
Neutron power [MW]	2445	283
Energy multiplication	1.17	1.17
Thermal power [MW]	3408	583
Surface area [m^2]	1077	69.3 (target)
Helium coolant:		
- Inlet temperature [°C]	300	700 (target)
- Outlet temperature [°C]	480	800 (target)
- Pressure [MPa]	8	10 (target)
- Mass flow rate [kg/s]	1528	473 (bulk)
		477 (target)
- Pumping power, η = 0.8 [MW]	30	3.4
Pb-17Li coolant:		
- Inlet temperature [°C]	480	
- Outlet temperature [°C]	700	
- Mass flow rate [kg/s]	46053	
Secondary helium loop:		
- Inlet temperature [°C]		285
- Outlet temperature [°C]		700
- Pressure [MPa]		15
Net efficiency (blanket/divertor cycle)		0.43

Key Issues and R&D Needs

Blanket: a) MHD - modeling and computations of 3D inertial flows in expansions, b) tritium recovery: The experience gained so far from the use of components for tritium recovery is not sufficient to reliably design such a system. More work on liquid/gas contactors is recommended, which should also include other volatile radioactive isotopes, c) Pb-17Li purification: Uncontrolled precipitation of corrosion products in liquid-metal loops must be avoided by using efficient purification systems. The aim should be to keep

these products in solution and to trap them in cold traps, thus preventing them from depositing, especially on the surfaces of the heat exchangers. Much more work is required for removing radioactive isotopes from the liquid metal. In particular, the radiotoxic α-emitter Po-210 formed from Bi-209 is considered to be a critical issue. Techniques to remove thallium and mercury are not yet available, d) SiC$_f$/SiC FCIs: Fabrication routes, compatibility with Pb-17Li flow at high temperatures above 700 °C, irradiation experiments, e) investigation of electromagnetic forces caused by disruptions.

Divertor: a) Material issues: In the long term, a development of W alloys is needed, which broadens the operational temperature window from the today's range of 800 – 1200 °C to 600 – 1300 °C by increasing the recrystallization temperature and simultaneously lowers the DBTT. Potential use of graded materials should be considered, b) choice of appropriate materials to reduce activation and widen the design options: Replacement of TZM as thimble material by tungsten or tungsten alloy and use of ODS EUROFER as structural material for the plate structure instead of TZM, c) development of fabrication routes and joining technology, in particular joining of steel to W, surviving frequent temperature cycles between RT and the operating temperature of about 600 °C, d) alternative: Development of transition pieces. The large mismatch in the thermal expansion coefficients of steel and refractory alloys, which are 10 - 14 x 10^{-6}/K and 4 – 6 x 10^{-6}/K, respectively, will cause very high local plastic strains at edges and corners.

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