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Abstract

Blankets are key elements of a future fusion power reactor, as they breed the fusion fuel tritium, extract the heat from the reactor for power generation and contribute to the nuclear shielding of the plasma confining magnetic field coils. On the way to the engineering implementation of fusion, in particular the blanket design approach has changed substantially. Novel blanket designs require, already from the beginning, incorporating close coupling of plasma physics with engineering physics to develop robust solutions coping with thermal, mechanical and also electrodynamic loads – not only during the stationary operating phase, but also during transients. Simultaneously, nuclear licensing feasibility as well as component failure safety must be part of the design approach. Additional key elements of the blanket design are reactor integration, the compatibility of interface functions, as well as reliable maintenance and feasibility of disassembly and recycling.

This article describes advanced blanket design approaches undertaken in the past years by the example of the helium cooled pebble bed blanket (HCPB), aiming at an efficient blanket engineering design, starting from the development of modular integral reactor analysis tools, via design analysis and engineering validation of fabrication and interface performance, towards safety analysis on the reactor level.

(German title of publication) – Blankets- Schlüsselelemente eines Fusionsreaktors - Funktionalität, Design und aktuelle Entwicklungsverfahren

1 Introduction- Blanket functionality

The realization of nuclear fusion as a future source of electric power currently undergoes the transition from a purely physics based science towards the engineering challenge. Naturally, this requires interlinking many engineering disciplines and coupling them to the physics basis. One of the largest challenges is the design of the plasma facing components (PFC), i.e., the blanket and the divertor as depicted in Figure 1, since both of them have to match several functions simultaneously at severe boundary conditions.

The central element of a fusion reactor is the blanket, which has to fulfill three primary major functions. First, it has to breed the scarce part of the fuel of a future fusion reactor – tritium. This is realized by a nuclear reaction of neutrons with lithium, of which the lithium isotope ^{6}Li is most preferred due to its high reactivity cross-section also for low neutron energies. By the ^{6}Li-reaction, not only tritium is bred, but also heat is generated volumetrically within the breeder material and the structure. Hence, the second function of the blanket is to extract the heat originating both from the plasma radiation towards the first wall and from the volumetric heat generation caused by neutron energy deposition and the nuclear reaction inside the blanket by means of a heat transfer fluid to the power conversion system (PCS). The third primary function is to provide a sufficient nuclear shielding for the plasma confining magnetic field coils, in order to reduce the neutron flux by about six to seven orders of magnitude from the first wall to the coil structures.

Particularly the first two functions allow several technical design options, each however limited by engineering constraints. Independent of the blanket design a tritium breeding ratio (TBR), defined as

$$\text{TBR} = \frac{\text{number of tritons produced per second in blanket}}{\text{number of fusion neutrons produced per second in plasma}},$$

larger than unity must be ensured. Since the blanket covers only about 82-85% of the plasma facing surface and some of the fusion neutrons are either absorbed in the structure or leaking out, a neutron (n) multiplication in form of (n, $2n$)-reactions is indispensable. Neutron multiplication can be achieved by beryllium (*Be*) or lead (*Pb*) translating to two different blanket families: the homogeneous blanket types using liquid lead-lithium alloys as breeder (and in some concepts also partially as coolant), and the heterogeneous blanket types, where neutron multiplication and breeding is realized by alternating stacked pebble beds consisting of *Be* spheres and *Li*-containing ceramic pebbles.

The power balance within such a reactor is quite complex. The core radiation of the plasma P_{rad}, the neutron heating of the structures P_n, the power associated with energy and particle diffusive and convective transport loss mechanisms P_{par} and the heating and current drive power fed into the plasma ($P_{H\&CD}$) for its operation and stabilization all have to be taken into account. Trying to quantify these energies for a future fusion power plant of 3.2 GW fusion power, we obtain a P_n of 2.6 GW that can be translated in average neutron wall fluxes at the plasma surface of about 1.9 MW/m^2 (assuming here a 1800 m^2 first wall surface). If we take a steady state plasma operation with the additional contribution of $P_{H\&CD}$ of about 130 MW, we can assume, for our example, P_{rad} ~630MW and P_{par} of ~150 MW [1]. The power P_{tr} will be responsible of large heat fluxes on the divertor targets that can potentially exceed 10 MW/m^2 [2].

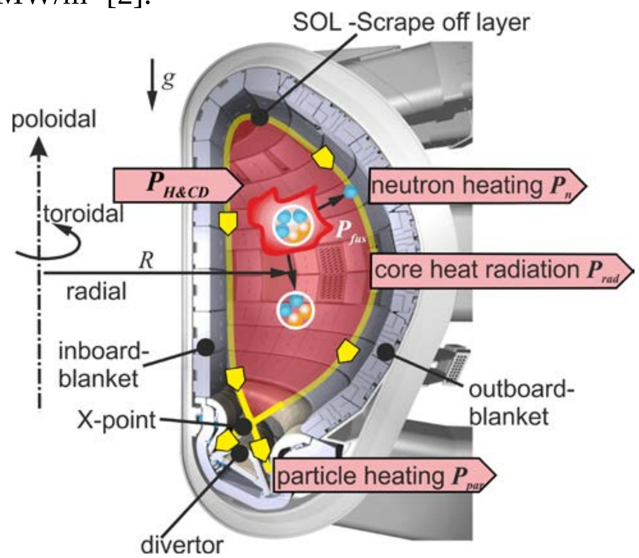

Figure 1: Sketch of a cross-section of a tokamak with presentation of blanket, divertor and contributors to the power balance of a fusion reactor within the "thermo-nuclear core".

Hence, a blanket experiences heat loads similar to other power engineering components such as receivers of concentrating solar power stations. The maximum volumetric power released in the first wall of a blanket is 25 MW/m^3 at maximum decaying exponential moving to the rear part. Its mean value for the whole blanket is only of the order of 3 MW/m^3. However, it has to be noted that the heat load originates from fast neutrons with neutron fluxes one order of magnitude larger than in a LWR, and from particle and heat radiation which even are far of being constant in time. The most severe limiting factor of all loads, however, is the neutron damage that reduces the lifetime of the blanket necessitating a full replacement at regular time. In our example (with a peak neutron wall load of ~2.5 MW/m^2) after about only 3 full power (FP) years of operation the damage to the structure will reach 75 displacements per atom (dpa) in the structure material associated to the FW with a helium production of about 750appm. Due to the high activation of the structures all maintenance and replacement procedures have to be conducted by remote handling operation, which holds for all connection/disconnection or joining operations as well. Furthermore, all plasma diagnostic instrumentation and access ports for the plasma heating systems (ECRH/ICRH and neutral beam injection) have to be routed through the blanket without deteriorating its performance.

Since the blanket is the largest power source of a fusion power plant absorbing more than 80% of the fusion power its coolant in- and outlet temperatures affect significantly the Balance of Plant (BoP) and the thermodynamic efficiency. On the other hand due to its large plasma coverage and highest material activation, the blanket is the core element for the safety demonstration, which is a pre-requisite for the licensing procedure of a future fusion plant.

In this article the entire blanket design procedure is described by the example of the helium cooled pebble bed blanket concept (HCPB) currently developed in the frame of the EUROfusion project. At first, the general trends in the lay-out approach to define blanket dimensions is addressed, while in the next step the engineering interfaces to the power conversion systems and tritium plant are discussed. Any nuclear blanket concept requires design verification in terms of its thermal integrity and manufacturability, which are described afterwards. Further on, integration concepts of blankets into the thermonuclear core are sketched before aspects of fusion power plant safety are briefly outlined.

2 General blanket design to match reactor targets

2.1 Dimensioning of the blanket

The realisation of the ITER fusion experiment in Cadarache, France, and the preparation of a future DEMO fusion reactor within the framework of the EUROfusion consortium, requires a substantially higher degree of integration of the central plasma-facing components, such as the blanket and the divertor, than required in previous more reactor design study oriented fusion power plant projects based on the Tokamak principle. Therefore, advanced blanket design processes, taking into account the plasma physics as well as the plasma confining magnetic fields, the heat and particle loads and their interaction with the plasma facing components are needed. In former conceptual studies, the iteration process towards a reactor model was realized via so-called system codes. Within those codes, zero or one-dimensional simplified multi-physics models validated by numerous experiments are applied to interrelate relevant reactor parameters such as the major radius of the tokamak R, the mean toroidal field strength B_ϕ, the plasma radius a, etc. Prominent examples of such code types are PROCESS, SYCOMORE or ARIES [4-6]. Due to their fast execution time, a large parameter regime can be rapidly screened relaying on a robust physics basis. However, engineering constraints are taken into account only marginally if at all. Hence, physics solutions outputs may be obtained which are hardly feasible from a technological point of view. The main result of system code computations is a plasma facing component geometry configuration in conjunction with a magnetic field set-up allowing with sufficient margin matching the reactor target requirements formulated. From these results then a generic CAD reactor model is deduced, allowing for engineering physics studies, which finally have to be analysed with respect to the reactor operation margins and technological feasibility. This closes the inner loop (LOOP1) of the engineering physics studies of the plasma facing components, as depicted in Figure 2.

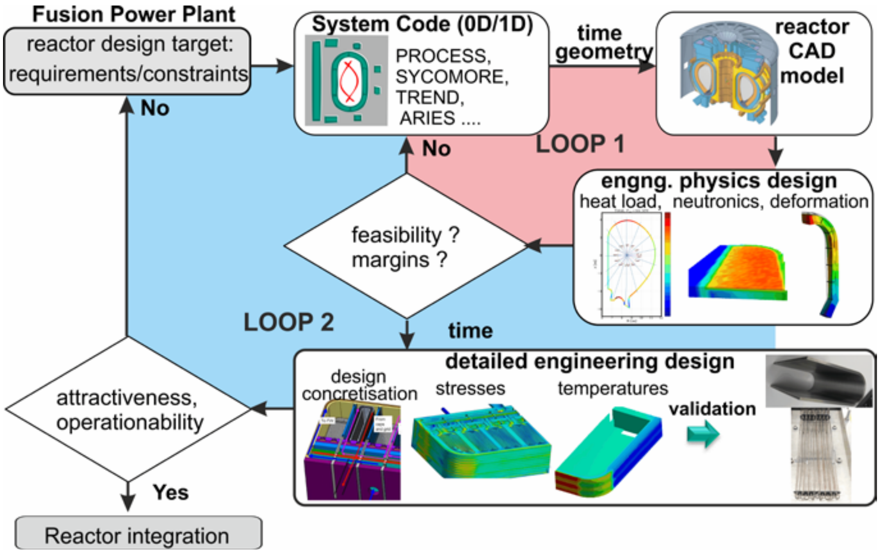

Figure 2: Current design procedure to dimension fusion blankets in the context of a system code based reactor design.

Finally, if a robust design has been obtained, those fundamental data are fed to detailed engineering design studies, now also incorporating (ramp-up and –down, transients, degradations - LOOP2). Therein, besides from design concretization, detailed stress and thermal-hydraulic/thermo-mechanic investigations are performed aiming to arrive at a validated blanket concept.

One of the major drawbacks of this approach is the absence of engineering constraints in conjunction with the time information upon arrival to the CAD model. As a result, existing engineering limitations may be ignored at this point, which are difficult to correct in a later phase of the detailed engineering design process, e.g. during each pulse strong gradients occur as depicted in Figure 3 leading to substantial loads on the structural components.

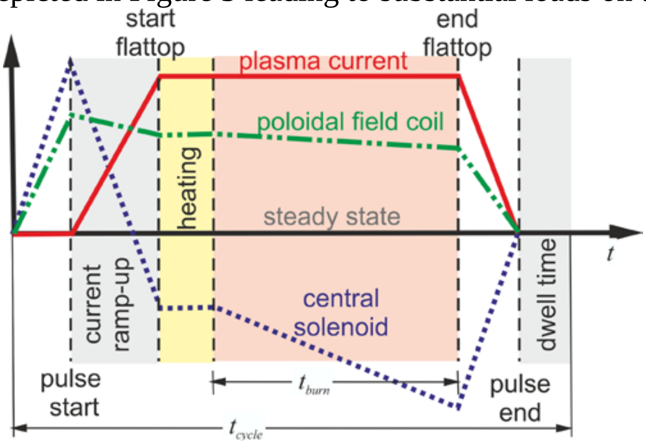

Figure 3: Sketch of the temporal evolution of plasma current and coil currents during a plasma discharge in a tokamak from [4]

In order to achieve a stronger coupling of the physics domain with the engineering design a modular integral reactor analysis tool (MIRA) is currently being developed at KIT. It contains a full time-resolved description of the plasma magnetic configuration (including the poloidal field coils / central solenoid currents), a simplified core plasma physics model accounting for density, temperature, pressure and confinement properties, and thus allow to extract the two-dimensional poloidal neutron and photon distribution as well as the charged particle flow towards the divertor. The 2D neutronics solution can be directly coupled to the technology domain, so that the tritium breeding performance in the blanket, the nuclear heating of the structures, the remaining neutron flux towards the magnetic field coils and the associated material damage can be evaluated. The electro-magnetic module integrated in the MIRA code is capable of computing the magnetic field distribution (necessary to describe the plasma position and shape), the Lorentz-Forces (acting on the blanket structure), the stored magnetic energy (required for safety calculations) and the inductance (determining the time constants in case of transients). Also the toroidal field ripple is calculated in this context. The Figure 4 shows schematically the set-up of the MIRA code and illustrates some of the outputs, more details can be found in [7, 8]. AS the next step, a power flow model will be integrated in order to extend the capabilities for assessing the steady state power balance and hence the power flows towards the primary heat transfer system (PHTS) and the power conversion system (PCS). This feature then will provide a seamless interface to Balance of Plant (BoP) studies and to the tritium plant models, where the tritium balance in the entire plant can be assessed.

Of course, due to the complexity of the multi-physics coupling of the MIRA code, it cannot fully replace the currently available 0D/1D system codes. However, once a more or less robust plasma configuration has been evaluated, it allows for more credible and refined sensitivity studies of the impact of marginal changes of blanket design and/or plasma configuration than any system code. Additionally, it provides the capability to execute uncertainty analyses of independent input parameters and thereby studying how those propagate through the system in space and time, which is an indispensable ingredient for future safety analysis.

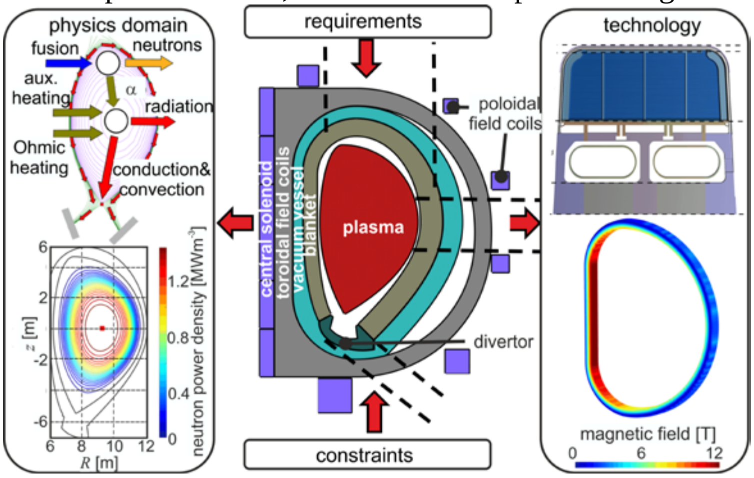

Figure 4: Sketch of the modular integral reactor system analysis code MIRA currently being developed at KIT for multi- physics studies to design fusion blankets.

2.2 Blanket design verification

The engineering physics provides conceptual design requirements for a breeding blanket, however, this far from engineering realization matching all secondary functions. Hence, a self-consistent blanket design demands concretion in terms of a technical set-up, which must be supported by available functional and structural materials and the related manufacturing technologies defining a design and safety analysis. Beyond design verification by computational means, validation through building and testing of mock-ups and prototypes is indispensable.

Any blanket design has to aim at a compact radial build, firstly reducing the reactor dimensions, then permitting for easy reactor integration without violating shielding requirements. Simultaneously, the weight should be low and the tritium breeding ratio adjustable in order to allow for margins to cope with missing plasma facing blanket surface coverage in case ports are required for vital reactor equipment (heating systems, plasma diagnostics, etc.). Furthermore, the coolant pressure losses in the blanket should be as small as possible to minimize pumping requirements. With respect to the HCPB blanket design, the architecture has been substantially simplified in the recent years, allowing a potential maximum TBR of up to 1.26 [9], a more compact radial build and optimized structure geometries, thus lowering fabrication costs and simultaneously enhancing reliability. The coolant pressure losses have been substantially reduced increasing the performance and thus the plant thermal efficiency. The

Figure 5 shows the schematic built-up of the HCPB blanket and its internal structure composed of a stack of alternatingly arranged breeder (*$LiSiO_4$*-ceramics) and *Be*-neutron multiplier beds. Cooling plates (CP) containing parallel, neighbouring channels are integrated between the stacks. Within the FW and CP´s Helium is flowing in counter-current flow pattern to homogenize the temperature. This feature potentially allows an independent feeding of the two symmetric loops introducing a partial cooling redundancy with improvement of safety. More details can be taken from [10].

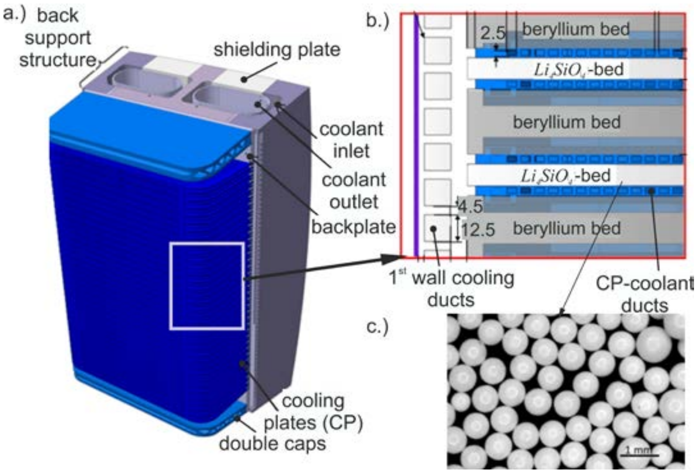

Figure 5: a.) Architecture of the HCPB Blanket, (b) cooling and breeder/multiplier arrangement and (c) breeder ceramics composed of *Li_4SiO_4* and *Li_2TiO_3*.

Any blanket design requires sophisticated studies w.r.t. the material behaviour of both functional and structural materials under irradiation and at extreme temperatures to demonstrate its functional performance at all operational load conditions and to allow for licensing.

In case of the HCPB advanced ceramic tritium breeding pebbles consisting of lithium orthosilicate (*Li_4SiO_4* - see Figure 5c) and 15-35mol% lithium metatitanate (*Li_2TiO_3* Figure 5c) were developed and exhaustively characterized in several experiments for their long-term stability, and their behaviour under irradiation. A similar approach has been made for beryllium.

Similarly also the reference structural material EUROFER which is a reduced activation ferritic martensitic steel obtained by intelligent exchange of steel alloying constituents has been improved by optimizing its composition and applying sophisticated thermomechanical treatment procedures. This so-called *advanced EUROFER* steel enables the HCPB to increase the coolant outlet temperature to a range of 600-650°C, which is desirable in terms of thermal plant efficiency. Moreover, it permits increased coolant inlet temperature (350°C) to circumvent the window of EUROFER irradiation induced embrittlement.

2.2.1 Thermal-hydraulic, thermo- mechanic & electro-dynamic performance

A challenge for any blanket is the safe removal of heat, without exceeding material design limits. The efficient heat transfer without excessive pressure losses is a real engineering challenge. To this end, an advanced cooling technology based on rib structure inside the channels has been developed for the most highly loaded first wall (FW). Vortex flow patterns are induced by the ribs located on the first wall, transferring heat from the fluid-wall interface towards the mean flow in the bulk. Experiments [11] have shown that at steady state conditions, a heat removal capability at the FW of about 1 MW/m² is achieved. Simultaneously, the required pumping power could be reduced by about 20%, due to the substantially increased wall-normal heat transfer. Currently, computational models are being developed to describe the turbulent heat transfer accurately and to allow experimental validation. Demonstrator-scale tests are planned to subsequently confirm the results on a larger scale. The manufacturing procedures to generate those rather complex structures were demonstrated as illustrated in

Figure 6. Progress in Hot Isostatic Pressing (HIP), Electrical Discharge Machining (EDM) and die-sink fabrication allows the production of prototype sample sizes [12].

The detailed engineering analysis of the thermal-hydraulic and thermo-mechanic performance follows the classical route of nuclear engineering, i.e., employing validated tools and experimental qualification through mock-ups. Thus, the functional performance for nominal operation conditions is ensured. The engineering analysis of a fusion blanket includes life-time aspects related to thermal-cycling and safety relevant failure mechanisms, etc. The physics involved, however, is completely different from that in light water reactors (LWRs). For illustration, three examples are given below.

The first one is the heat transfer validation in pebble beds at prototypical operation conditions and geometries. Because a significant fraction of the heat is volumetrically released in the pebble bed by nuclear reactions, the heat transfer in pebble beds is of vital importance. Since for this type of heat transfer of a gas flow through a sparsely packed bed (packing factor of ~63%) validated models are missing or in an early development phase, demonstration experiments are required in prototypical conditions, where the volumetric heat load is mimicked by electric resistance heaters. Since tokamaks are intrinsically pulsed reactors, the ensemble of the pebble bed is subjected to cyclic thermal stresses, producing residual strain in the bed depending on the temperature range and the stresses. This can be compensated by the swelling of the pebbles induced by neutron irradiation. Still, large gaps may appear which would lead to a reduced cooling capability and/or the occurrence of hotspots. Dedicated models have been developed and experimentally validated, see e.g. [13, 14].

Moreover, the pulsed operation of a tokamak induces large electric currents circulating in the structures during the ramp-up and shut-down of the plasma. This leads to mechanical stresses and deformation, which have to be kept within the design limits of the material. Evaluation of the loads caused by these effects requires the incorporation of all temporally changing magnetic fields from different sources (field coils, plasma), as well as the ferromagnetic nature of the EUROFER steel, in dedicated models within a computer code. A detailed description can be found in [15]. The loads which are most demanding for the structural integrity of the blanket support structure were obtained close to the end of a plasma disruption, which is an instability phenomenon likely to occur in tokamaks. In this type of event, being part of the regular design base events for safety demonstration of a fusion reactor, loads can be obtained close to the structural mechanic limits of a blanket.

Besides regular design base events, a failure of the structures separating the purge gas system (transporting the bred tritium and operating at 0.2MPa) from the coolant gas (8MPa) can lead to a so-called in-box Loss Of Coolant Accident (LOCA), which by design should lead to save operation at reduced power or controlled reactor shut-down. To this end, extensive calculations are being conducted to evaluate the maximum appearing stresses and deformations in such a type of event, for more details [16].

2.2.2 Fabrication

Taking into account the manifold mechanical, thermal and electro-magnetic loads a blanket is exposed to, which occur on times scales ranging from milliseconds to hours, manufacturing and joining technologies qualified for nuclear grade materials are a pre-requisite for nuclear licensing. For nuclear installations, all components have to comply with the codes and standards of the nuclear regulators. As shown in the previous section, fusion blanket architectures involve significantly more complex geometries and joints than those contained in any nuclear fission reactor presently in operation.

Almost all blanket concepts of first generation consist of a steel box made of a martensitic-ferritic steel (EUROFER97 in EU) with an internal stiffening grid which provides mechanical resistance. In case of the HCPB blanket, the stiffening grid separates the inner volume in several compartments containing, alternatingly, breeder or multiplier materials. The plates of the stiffening grid are designed as cooling plates (CPs) for heat extraction, while in the purge gas is circulating inside the pebble beds. The challenge is to manufacture EUROFER97 steel structures of various thicknesses in standardized, well qualified procedures complying with nuclear codes and standards (as, e.g., RCC-MRx).

For the fabrication of subcomponents for inside of the massive steel box, all joining processes are based on the use of diffusion welding (DW) or conventional welding technologies (e.g., laser welding, electron beam welding - EB, …) taking into account the specificities of the EUROFER97 steel. To attain a well described welding procedure suitable for nuclear licensing, specifications (WPS) have to be qualified to establish a closed manufacturing chain and enable quality control. Hence, mass fabrication has to rely on the production of simple parts in an automatized manner making as much as possible use of industrially available nuclear qualification procedures [17]. To illustrate this approach, Figure 6b depicts the individual parts for an HCPB breeder unit mock-up, following the aforementioned pre-requisites.

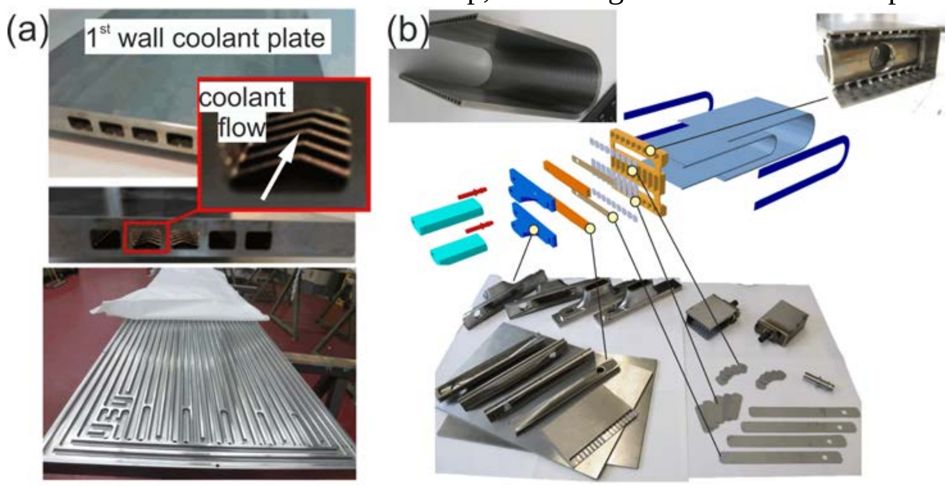

Figure 6: (a) manufacturing sample of 1st wall coolant channel with turbulence promoters [11] and half-plate of a first wall mock-up before hot isostatic pressing (bottom); (b) explosion drawing of a HCPB blanket breeder unit and different fabrication samples cooling plate with integrated channels (top left), (c) manifold, (d.) different connecting parts.

The manufacturing of the massive steel frame housing, i.e., the "First Wall (FW)", with its coolant channels which has to be high-pressure resistant, is even more challenging. Two different fabrication schemes have been developed in collaboration with industry. The first one is based on wire erosion to generate the coolant channels within the FW, and subsequent bending. In the second scheme, half plates are generated, diffusion bonded and afterwards bent as well. Novel additive manufacturing techniques like selective laser sintering (SLS) offer new options particularly for the complex structures inside the massive steel frame of the fusion blanket. Thus, complex integrated manifold distributor elements or even electrically insulated channels can be produced seamlessly within one manufacturing step. While this technology is still in its infancy in the nuclear sector, and still lacks a closed nuclear licensing framework, first manufacturing samples developed at KIT show promising results, matching the functional requirements and exhibiting sufficient material strength.

2.3 Major functional interfaces

As illustrated in Figure 1 the blanket has to match several interface functions. The three fundamental blanket interfaces are the:

- coolant transfer to the power conversion system (PCS),
- fuel extraction (tritium) from the coolant or, in case of the HCPB, from the solid breeder by means of the purge gas (helium),
- capability to allow maintenance/(dis-)assembling operations,
- balance of plant (BoP).

Below, these aspects are addressed from the blanket point of view.

2.3.1 Power Conversion System (PCS)

In order to achieve a high thermodynamic efficiency η_{th}, the temperature of the coolant when exiting the blanket should be as high as technically possible. In principle two PCS types are feasible for the HCPB blanket concept; Clausius-Rankine (steam turbine) or a Joule-Brayton (gas turbine) cycle.

The Clausius-Rankine process operates with a lower average temperature and multi-stage pressure levels. The advantage is that the technology has been extensively qualified in nuclear LWRs and components are available. However, due to the thermo-physical properties of water, it is naturally limited in temperature and it requires high water pressures (>15MPa) at high flow rates. Thereby, only moderate thermodynamic efficiencies η_{th} of the order of 40% are attainable.

The use of the Joule-Brayton process can be considered, too; this can become an option if the He coolant is available at temperatures higher than 700°C translating to a higher η_{th}. However, for such a temperature levels the material challenges of high temperature components are not yet solved, in particular w.r.t. neutron irradiation effects.

2.3.2 Tritium extraction and control

The blanket is the source to produce the tritium required for the fusion reaction. For a $3GW_{fus}$ fusion power plant the tritium consumption is about 460g/FP-day, while the radiation protection limit allows only a loss in the environment of less than 270 mg/day, which is a factor of more than 1000times less. This requirement postulates dedicated measures to keep the tritium content as low possible in the helium primary heat transfer system, due to the permeation of tritium through the structure material EUROFER from the breeder zone. Since this permeation cannot be prevented completely a coolant purification system (CPS) is intrinsically necessary to remove Tritium from the coolant. Of course, permeation barriers to inhibit tritium migration into the coolant would be desirable. However, the high neutron flux and the associated material damage prevented by now a suitable, validated technical solution. Another option is maintaining a certain hydrogen partial pressure in the main helium coolant loop to largely eliminate tritium diffusion through the structures by a counter-stream of hydrogen. While this is already widely used in fission reactors, it is still a topic of research in fusion.

Extraction of the bred tritium from the purge gas is located outside the tokamak. From there, the tritium is transferred to the tritium plant, where also the tritium streams from the CPS and the divertor pumping system arrive.

2.3.3 Reactor integration and maintenance

As already mentioned, due to the high flux of high energetic neutrons and the associated material damage and helium generation in the structural material, the life-time of blanket is limited and blanket systems require several replacement during the plant lifetime (in a FPP we can expect regular replacement every 5 calendar years). By then, the blankets have accumulated shut-down dose rates of several Sieverts per hour, and need to be extracted by means of remote handling (RH) procedures. This in turn requires supply and discharge piping schemes allowing for disconnecting / extraction by dedicated tools. For DEMO, insertion and removal of the blanket into / from the thermo-nuclear core through the upper ports is currently foreseen as preferred option. This requires space reservation to be derived from the RH necessities. The dimensions of the inlet and outlet piping are 200/250 mm for the inboard modules, while for the larger outboard blanket modules the dimensions are 250/300mm. Potentially, the inter-pipe distances and hence the space required for the pipe routing scheme could be minimised, but it is constraint by re-welding procedures. The piping scheme is illustrated in Figure 7a. Remote handling tools can access the reactor core through the upper port. The integration of the different blanket segments in the reactor via fixation at the back support structure is illustrated in Figure 7b; more details can be found in [18].

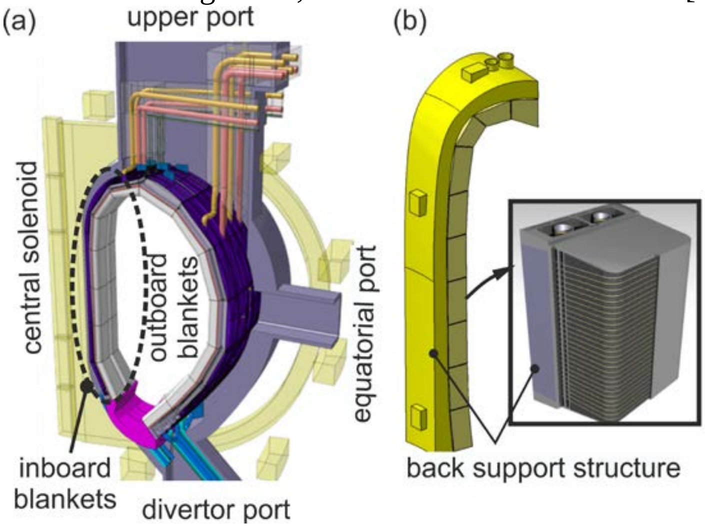

Figure 7: (a) piping scheme for the in-/outboard blankets and the divertor. (b) segmented built up of the blanket for integration in the core.

2.3.4 Balance of plant (BoP)

The Balance of Plant (BoP) of a fusion power plant (FPP) describes the ensemble of heat transfer and power conversion related systems outside the tokamak thermo-nuclear core, and includes the entire power conversion train, which is composed of the primary heat transfer system (PHTS), the intermediate heat transfer system (IHTS) and the power conversion system (PCS). Additionally, the auxiliary systems (cooling, water supply, etc.) contribute to the BoP as well as the on-site power supply. As shown in Figure 8, the main power source is the blanket. In contrast to conventional power stations, a fusion power plant requires a large set of auxiliary systems with high power consumption, such as heating & current drive, cryoplant, tritium plant etc., which are all impacting the BoP.

Due to the intermitting operation of a tokamak, a simple power conversion system without any energy storage unit doesn’t fit into any commercial grid, because this would mean that a fusion power has to be fed with considerable electrical power by the grid during the dwell time. In addition it is questionable if a turbine can survive a pulsed operation. Hence, in the current European DEMO approach an energy storage system is foreseen. Likely, this would be realized via thermal energy storage (TES) operated with solar salt, since solar salts match the output temperature of the blanket with their operation temperature regime.

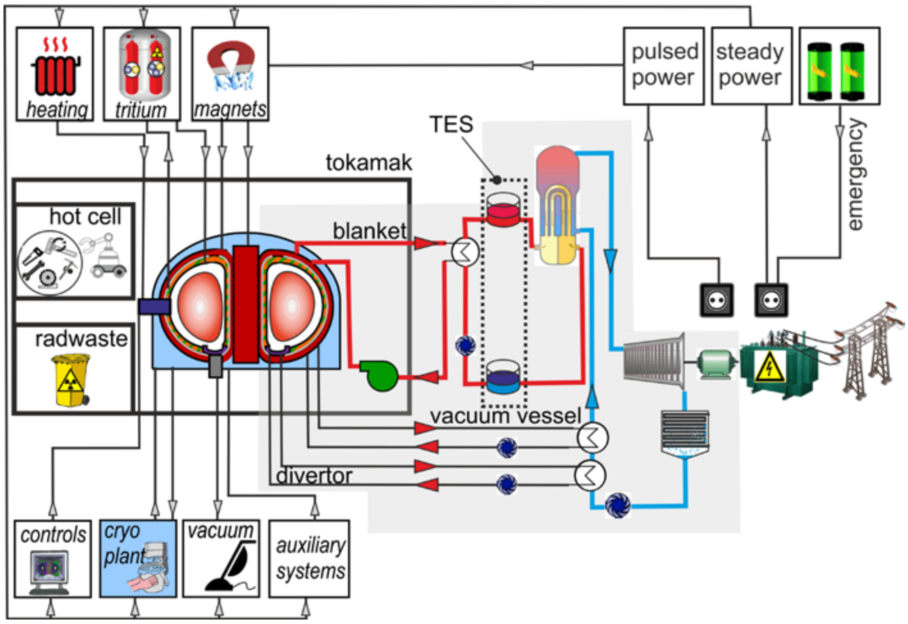

Figure 8: Power sources and sinks contributing to the Balance of Plant in a fusion reactor.

In order to provide a closed BoP analysis, all other power sources besides the blanket, as well as intermediate and low temperature heat sources (e.g., waste heat of auxiliary systems) need to be integrated in order to maximize the plant efficiency. Of course, this is still subject of intense research due to a number of uncertainties from the tokamak physics basis. E.g., it is still unclear what pulse durations can be achieved in tokamaks or which minimum dwell times are required. However, both are essential ingredients to design a robust and compact power storage system, respecting the thermal inertia constraints of the individual plant components.

3 Safety demonstration and licensing

The general safety objectives of a future fusion power plant will have to follow the rules of any other nuclear power station, which are

- the prevention of radiological hazards to the general public and environment,
- prevention of hazards to the workers following the "*as low as reasonable achievable* "principle (ALARA), and the
- minimisation of the radioactive waste disposal volume.

The nuclear safety is a prerequisite for any nuclear licensing of a facility, and although the thermal energy stored in the blanket constitutes only less than half of the energy stored in a fusion power plant [19], the blankets safety performance is of key importance towards any nuclear fusion power plant licensing. First, the blanket covers more than 80% of the plasma facing surface, i.e., it contains the vast majority of the material activated by neutron irradiation. Additionally, it contains tritium. Despite the fact that the tritium in the blanket is mostly bound in the breeder material or in the structures, it represents one of the most significant source terms for accidents in a DEMO reactor [20]. Finally, the residual heat generation in the FPP is not stopped completely after shutdown, but will continue at a few percent level (e.g., it can be less than 1 % of the fusion power after a day after shut-down [21, 22]) and decrease exponentially for the time after. Hence, the structures require a cooling after shut-down.

Therefore, any safety-adapted fusion power plant architecture requires a set of primary and secondary confinement barriers, as well as the assignment of safety functions to the individual components or installations. Here, substantial progress has been made in the recent years due to the activities for ITER as well as in the context of the DEMO reactor development within EUROfusion, to define design and licensing requirements and to classify components and systems into safety importance classes (SIC), see e.g. [23, 24].

The core of the safety demonstration is an integrated safety analyses with a complete identification of source terms. To this end, postulated accident scenarios and their consequences are studied. In the absence of a detailed design, this is conducted by means of a Functional Failure Modes and Effects Analysis (FFMEA), which has led to a set of so-called Postulated Initiating Events (PIEs), which are considered as relevant reference events [20, 24]. Within the safety analysis for each of the reference events, an accidental sequence is computed deterministically using numerical tools. This in turn requires the development, verification and validation of fusion-adapted models and tool packages. Often, well qualified packages used for the safety assessment of LWRs or other nuclear applications can be used as a basis, however, a fusion power plant substantially deviates from those not only w.r.t. the nuclear source terms involved, but also concerning other energetic source terms like the magnets and the cryo-plant, and also in geometry and multi-physics interactions. Hence, the current focus of the safety analysis is on the development and qualification of fusion power plant adapted numerical tools.

As an example, one sequence for the development and qualification of safety systems is briefly explained by the example of a "loss of flow accident" (LOFA) in the first wall of a breeding blanket. At first, a detailed computational fluid dynamics (CFD) simulation is conducted for the isothermal flow in the structure. The results of the simulation are compared to those of a system code using simplified models as shown in Figure 9b. In the next step, measurement data of the pressure drop are compared to the computed system code data (Figure 9c). Finally, an experiment is set up to study a loss of flow accident at prototypical first wall fusion reactor conditions (Figure 9d).

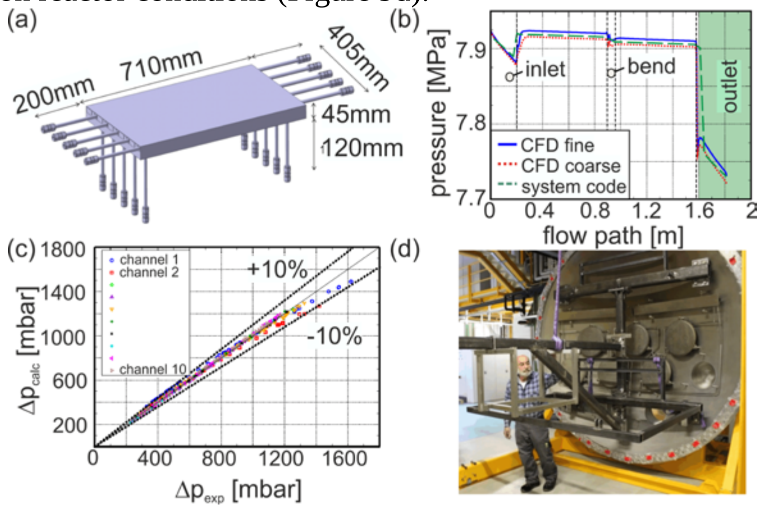

Figure 9: Approach for development and validation of models for safety assessment in fusion plants. (a) hydraulic mock-up of the 1st wall; (b) comparison of computed pressure drops of CFD with system codes; (c) comparison of measured and computed pressure losses; (d) experimental-set-up of piping to simulate a LOFA at KIT HELOKA.

Once the models are verified and validated on the component scale, the simplified models are transferred to the reactor scale. On this level, the piping as well as potential component interactions are modelled on a nodal basis similar as in conventional nuclear reactor safety computational tools. Of course, this implies the loss of the detailed geometric interactions, however, the dynamics of the system is retained. With such approach, positions of potential component failures as well as possible release paths can be identified.

Figure 10a illustrates the fusion reactor set-up for the HCPB blanket. It is divided into 18 sectors, each composed of an inboard (IB) and an outboard (OB) part. Accordingly, the primary heat transfer system (PHTS) design consists of 6 loops for the outboard blankets and 3 loops for the inboard blankets. I.e., one OB loop scopes 3 sectors and one IB-loop 6 sectors. In one sector there are three OB segments and two IB segments. The highest loaded blanket is the OB4 blanket; the location is shown in Figure 10b. For this blanket type, a logical nodal set-up has been created by means of a system code; see [25]. This model contains the pipes to the primary heat transfer system as well as links to the vacuum vessel (VV) and the vacuum vessel pressure suppression system (VVPSS). Thus, potential failures within the blanket can be identified, and the related characteristic time scales can be determined. Once this single segment analysis is validated, the blanket model can be embedded into a full scale plant model. On this basis the safety performance on a full plant scale can be studied.

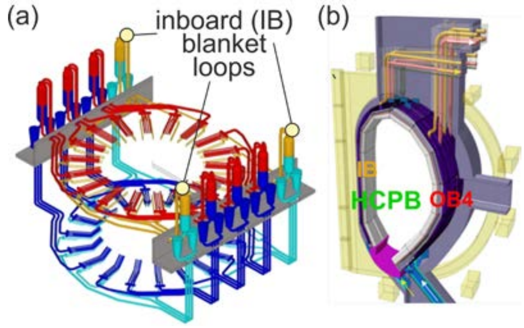

Figure 10: (a) out of vessel coolant piping of a DEMO reactor with the HCPB blanket concept. (b) Piping scheme of one sector of the DEMO reactor with HCPB blankets with the highest loaded blanket segment OB4.

4 Summary

This article describes recent advances in blanket engineering towards realization of fusion devices by the example of the helium cooled pebble blanket (HCPB).

Already in the early design phase, the use of multi-physics and multi-scale modular integral reactor analysis tools as the MIRA code developed at KIT allows a coherent blanket development, respecting not only engineering constraints, but also enabling the study of time dependent phenomena.

The engineering design derived from the basic design necessitates multi-dimensional verification of the primary and secondary blanket functions. Here, substantial progress has been made, e.g., detailed models to evaluate the heat transfer in pebble beds, advances in the description of the thermo-mechanics of pebble beds, and assessment of the loads caused by electro-dynamic forces in case of rapid plasma transients. Beyond the progress in model development and validation, advances were made in terms of heat transfer enhancement via the introduction of rib structures, increasing the heat transfer of the first wall coolant ducts, and particularly for the manufacturing methods. Here, improved procedures for wire erosion and additive manufacturing (e.g., selective laser sintering) developed in cooperation with industry open up new perspectives in blanket design, although the nuclear licensing procedures for these are still at the beginning.

With respect to the major functional interfaces, i.e., the power conversion system, the tritium plant and the balance of plant, as well as to the entire reactor integration, the collaboration of the European fusion laboratories in the frame of the EUROfusion consortium has led to a more coherent reactor development as described here for selected areas.

Finally, aspects for nuclear safety and licensing were described, including the development of models and codes and their validation by experiments especially for fusion relevant scenarios. Due to the complex multi-physics and multi-scale challenges in fusion which are different from those in fission reactor engineering, there is still substantial progress to be made in terms of verification and validation, in order to arrive at a coherent fusion reactor safety demonstration.

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