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Blankets - key element of a fusion reactor - functions, design and present state of development

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 a design incorporating a closed coupling plasma physics with engineering physics to develop robust designs coping with thermal, mechanical and also electrodynamic loads not only during the stationary operating phase but also during transients. Simultaneously nuclear licensing capability as well as component failure safety must be part of the design. Additional key elements of the blanket design are the ease of reactor integration, the compatibility of interface functionality, as well as reliable maintenance and disassembly, and recyclability.

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

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1 Introduction- Blanket functionality

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

Thereby, the central element of a fusion reactor is the blanket, which has to fulfil three primary major functions. First, it has to breed the fuel of a future fusion reactor –tritium. This is realized by a nuclear reaction with lithium, of which the lithium isotope Li^6 is most preferred due to its high nuclear cross-section also for lower neutron energies. By the Li^6-reaction not only tritium is bred but also volumetrically heat is generated within the breeder material and the structure. The second function of the blanket is to extract heat originating both from the plasma radiation towards the first wall and the volumetric heat generation caused by 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 in particular for magnetic field coils in order to reduce the neutron flux by about six to seven orders of magnitude from the 1st wall to the coil structures.

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

$$\mathrm{TBR} = \frac{\text{number of tritons produced per second in blanket}}{\text{number of fusion neutrons produced per second in plasma}}, \quad (1.1)$$

being 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. Potentially neutron multiplication can be achieved by beryllium (Be) or lead (Pb), which translates to two different blanket families; the homogeneous blanket types using liquid lead-lithium alloys as breeder (and in some concepts also partially as coolant), or the heterogeneous blanket class, in which neutron multiplication and breeding 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 and composed not only of the core radiation of the plasma P_{rad} but also neutron heating of the structures P_n, the power associated with energy and particle diffusive and convective transport loss mechanisms P_{par} and finally by the heating and current drive feeding power into the plasma ($P_{H\&CD}$) for its operation and stabilization. Trying to quantify these energies for a future fusion power plant of 3.2 GW fusion power [1], we obtain a P_n of 2.6GW 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 [Wenninger, 2017]. 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 assignment 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. Also the maximum volumetric power released in the first wall of a blanket is at maximum 25 MW/m^3 and its mean value for the blanket is only of the order of 3 MW/m^3, which is compared to a light water reactor (LWR~100MW/m^3) marginal. However, one has to notice that its origin is due to fast neutrons with neutron fluxes being an order of magnitude larger than in a LWR plus particle and heat radiation, which are additionally far of being constant in time. The most limiting factor of all loads is the neutron damage that limits the lifetime of the blanket necessitating a full replacement at regular time. In this example (with a peak neutron wall load of ~2.5 MW/m^2) after about only 3 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 means, which holds for all connection/disconnection or joining operations as well. Likewise 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 core element of the safety demonstration being a pre-requisite of a licensing procedure of a future fusion plant. Hence, the entire context of the blanket design with its primary functions, its interface requirements and the corresponding aspects in the view of reactor performance and feasibility can be illustrated as sketched in Figure 2.

Figure 2: Context of the blanket design in terms primary and secondary functions associated with superior integration targets to be met.

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 lay-out approach to dimension blankets 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 being 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 implementation of the ITER fusion experiment, built in Cadarache, France, and the concretization of a future DEMO fusion reactor, developed within the framework of the European project EUROfusion, requires a substantially higher degree of integration of the central plasma-facing components, such as the blanket and the divertor, than in the previous more reactor design study oriented fusion power plant projects based on the Tokamak principle. Therefore, advanced blanket design processes are required taking into account the plasma physics in conjunction with the plasma facing components as well as the plasma confining magnetic fields. In former times the iteration process towards a reactor model has been realized via so-called system codes. Within those codes zero or one-dimensional simplified multiphysics 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 exploited based on a robust physics basis, however, engineering constraints are if even only marginally depicted. Hence, by the computed code outputs physics solutions may be obtained which are from a technological point of view hardly feasible. The result of the system code computations is mainly 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 subsequently a generic CAD reactor model is deduced. This in turn allows for engineering physics studies, which have to be analyzed with respect to the reactor operation margins and their technological feasibility. This closes the inner loop (LOOP1) of the engineering physics studies of the plasma facing components, as depicted in Figure 3.

Figure 3: Design procedure to dimension fusion blankets in the context of a reactor design.

Finally, if a robust design is obtained those fundamental data are transferred to detailed engineering design studies, in which also the time information gets a part of the solution (LOOP2). Therein, aside 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, engineering limitations may be exceeded that are difficult to correct in later phase of the detailed engineering design phase. During each pulse strong gradients occur as depicted in Figure 4 leading to substantial loads for the structural components.

Figure 4: 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 developed at KIT. It contains a full temporal description of the plasma magnetic configuration (including the poloidal/ central solenoid currents), a simplified core plasma physics accounting for density, temperature, pressure and confinement properties from which a two-dimensional poloidal neutron and photon distribution as well as the charged particle flow towards the divertor can be extracted. The 2D neutronics solutions allows an immediate coupling to the technology domain, for which 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 to compute 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 calculation) and the inductance (determining the time constants in case of transients). Also the toroidal field ripple is calculated in this context. The Figure 5 shows schematically the set-up of the MIRA code and illustrates some of the outputs, more details can be found in [7, 8]. In the near future a power flow model is planned to be integrated 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). The latter feature provides then a seamless interface to Balance of Plant (BoP) studies and to the tritium plant models, in which the tritium balance in 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 a 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 to study how those propagate through the system in space and time, which is an indispensable ingredient for future safety analysis.

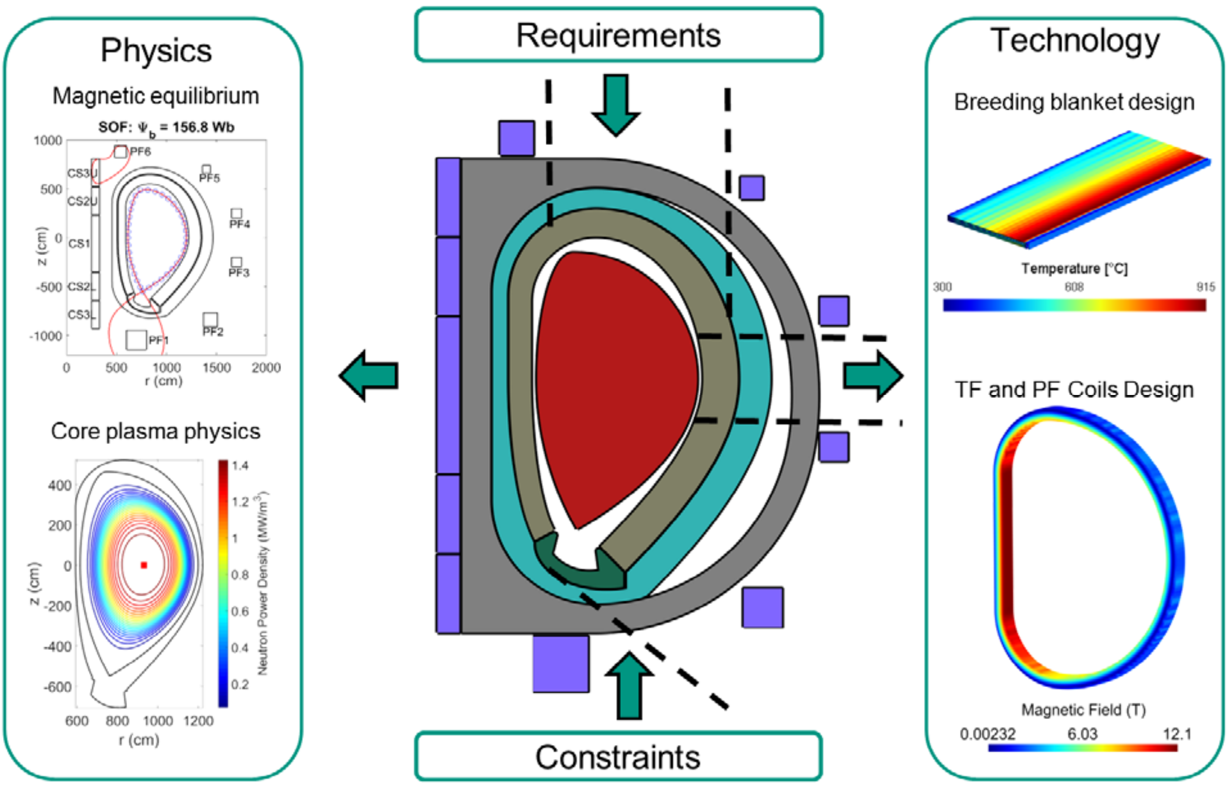

Figure 5: Sketch of the modular integral reactor system analysis code MIRA currently developed at KIT for multi-physics studies to design fusion blankets.

2.2 Blanket design verification

As mentioned the engineering physics provides conceptual design requirements for a breeding blanket, however, this far from engineering realization matching all secondary functions. Hence, a closed blanket design demands a concretization in terms of a technical set-up, which must be supported from the development of functional and structural materials via the related manufacturing technologies and finally a design and safety analysis. Aside from the verification of the design by computational means a validation by means by building and testing of mock-ups and prototypes is indispensable.

Any blanket design has to aim at compact radial build, firstly to reduce the reactor dimensions, then to allow 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 other vital reactor equipment (heating systems, plasma diagnostics, etc.). Also 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 max TBR of up to 1.26 [9], a more compact radial build and optimized structure geometries to lower fabrication costs and simultaneously enhancing reliability. Also the coolant pressure losses have been substantially reduced increasing the performance thus the plant thermal efficiency. The Figure 6 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. Between the stacks cooling plates (CP) containing parallel neighbouring channels are integrated. 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 may be taken from [10].

Figure 6: 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 with respect to 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 a licensing.

In case of the HCPB advanced ceramic tritium breeding pebbles consisting of lithium orthosilicate (Li_4SiO_4-see Figure 6c) and 15-35 mol% lithium metatitanate (Li_2TiO_3 -Figure 6c) were developed and exhaustively characterized in several experiments for their long-term stability, the compatibility with EUROFER steel, and their behaviour under irradiation. A similar approach has been made for beryllium.

Also the structural material for the blanket, a reduced activation ferritic martensitic steel, 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 enables 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 a safe heat removal not exceeding material sustainable limits. The efficient heat transfer without excessive pressure losses poses an engineering challenge; for the most highly loaded first wall (FW) channels an advanced coolant technology has been developed based on rib structures. Thereby, wall normal flow vortex structures are induced by the ribs located on the first wall transferring heat from the fluid wall interface towards the mean flow in the bulk. The functional principle is illustrated in Figure 7a. Experiments [11] conducted have shown that at steady state a heat removal capability of 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 developed to describe the turbulent heat transfer accurately and to validate them by experiments. Further demonstrator-scale tests are planned to substantiate the results on a large scale. Also the fabrication procedures to obtain those quite complex structures were demonstrated as Figure 7c illustrates. Progresses in hot isostatic pressing (HIP), Electrical Discharge Machining (EDM) and die-sink fabrication allows also the production of prototype sample sizes [12, Figure 7c].

Figure 7: (a) functional principle of turbulence enhancement by rib like structures; (b) potential arrangement of turbulence promoters at the FW; (c) fabrication sample of 1st wall coolant channel with turbulence promoters [11].

The detailed engineering analysis of the thermal-hydraulic and thermo-mechanic performance follows the classical route of nuclear engineering mainly through validated tools and experimental qualification through mock-ups as already indicated in Figure 2. Thereby, the functional performance is ensured for nominal operation conditions. The engineering analysis of a fusion blanket scopes life-time aspects by thermal-cycling or safety relevant failure mechanisms etc. The physics involved is completely different than the one in LWR's. To sketch some examples here only three examples are mentioned.

The first is the heat transfer validation in pebble beds at prototypical operation conditions and geometries. Since 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 largely absent or in an early development phase demonstration experiments are required in prototypical conditions, in which the volumetric heat load is mimiced by electric resistance heaters as depicted in Figure 8a. The data measured in the test section (Figure 8b) shown are shown in Figure 8c.

Figure 8: (a) Experimental set-up to mimic volumetric power release in pebble bed, which is integrated in prototypical dimensioned breeder zone (b). (c) Measured isothermal distribution in the mid-plane of the bed [13]. Comparison of stress-strain measurements and computations at different temperatures for a cooled pebble bed [13, 14].

Since tokamaks are intrinsically pulsed reactors, also the pebble bed configuration is submerged to cyclic thermal stresses, which lead to the occurrence of residual strain in the bed depending on the temperature range and the stresses. This is to some extend compensated by the swelling of the bed particles by neutron irradiation, however, large gaps may appear which would either lead to a reduced cooling capability or the appearance of hotspots. Dedicated models have been developed and validated through experiments [13, 14] as a comparison of computation and experimental data shown in Figure 8d illustrate.

Moreover, the pulsed operation of a tokamak induces due to the ramp-up and shutdown of the plasma large electric currents circulating in the structures leading to mechanical stresses and to elastic deformation, which have to be kept within material sustainable limits. To evaluate the loads caused by these effects demands the incorporation of all temporal changing magnetic field sources (different field coils, plasma) as well as accounting for the ferromagnetic nature of the EUROFER steel in dedicated models within a computer code; a more detailed description may be taken from [15]. The most demanding loads 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 based events of a fusion reactor, loads can be obtained close to the structural mechanic limits of a blanket.

Aside from regular design based events also a failure of the structures separating purge gas system (transporting the bred tritium operating at 0.2MPa) and coolant gas (8MPa) can lead to a so-called in-box Loss Of Coolant Accident (LOCA), which should allow for a safe reduced power operation or a controlled reactor shut-down. Hence, extensive calculations are conducted to evaluate maximum appearing stresses and deformations in such a type of event. Figure 9 shows a typical result of such a computation.

Figure 9: Computed deformation in metres for the most stressed region of the highest loaded blanket (outboard sector 4) of the HCPB blanket after an in-box LOCA at 9MPa from [16].

2.2.2 Fabrication

Due to the various different mechanical, thermal, electro-magnetic loads a blanket experiences, which also occur on different time scales ranging from milliseconds to hours, aside from qualified materials also nuclear grade accepted fabrication technologies and joining technologies are mandatory to allow for a nuclear licensing. For nuclear installation all components have to comply with so-called codes and standards of the nuclear regulators. As shown in the previous section a blanket architecture exhibits significantly more complex structural geometries, joints than appearing in any presently operating nuclear light water reactor.

Almost all blanket concepts consist of a steel box made of EUROFER97 with an internal stiffening grid which provides mechanical resistance. In case of the HCPB blanket it separates the volume in several compartments containing either breeder or multiplier materials. The stiffening plates themselves are designed as cooling plates (CP's) to enable heat extraction, while in the beds the purge gas is circulating. The fabrication challenge is to manufacture EUROFER97 steel structures of various thicknesses in standardized well qualified procedures complying with professional codes and standards (as e.g. RCC-MRx).

For the subcomponents fabrication being inside the massive steel box, all joining processes are based on the use of diffusion welding (DW) and or conventional welding technologies (e.g. laser welding, electron beam welding-EB,…) taking into account the specificities of the EUROFER97 steel. To attain a nuclear licensing dedicated well described welding procedure specifications (WPS) have to be qualified to provide a closed manufacturing chain and ensure quality control. Hence, a mass fabrication has to rely on the production of simple parts in an automatized manner taking as much as possible use of industrially available nuclear qualification procedures [17]. To illustrate this procedure Figure 10 depicts the individual parts for a HCPB breeder unit mock-up following the aforementioned pre-requisites.

Figure 10: (a) explosion drawing of a HCPB blanket breeder unit, (b) cooling plate with integrated channels, (c) manifold, (d.) different connection parts.

Even more challenging is the fabrication of the massive steel frame housing the first wall with its coolant channel, which has to be pressure resistant. Here, in collaboration with industry two fabrications schemes have been developed. The first one is based on a wire erosion technique to generate the coolant channels within the 1st wall and a subsequent bending (see Figure 11a), while in the second scheme half plates are generated, diffusion bonded and afterwards bend as well (see Figure 11b).

Figure 11: (a) First wall mock-up with integrated coolant channels generated by wire erosion. (b) half-plate of a first wall mock-up before hot isostatic pressing. (c) Manifold with integrated coolant channels fabricated by selective laser sintering.

Novel manufacturing techniques based on additive manufacturing such as selective laser sintering (SLS) offers especially for the complex structures inside the massive steel frame of the fusion blanket new options. Thereby, within one manufacturing step complex integrated manifold distributor elements or even electrically insulated channels can be seamless produced. Although this technology is in the nuclear sector still in its infancies and presently lacking of a closed nuclear licensing framework, first manufacturing samples developed at KIT (see Figure 11c) show promising results matching the functional requirements and exhibit sufficient material strength values.

2.3 Major functional interfaces

As already 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 coolant or in case of the HCPB tritium transported by the purge gas helium,
- capability for maintenance/(dis-)assembling operations.

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

2.3.1 Power Conversion System (PCS)

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

The Clausius- Rankine process exhibits a lower mean average temperature and a multi-stage pressure level, however, the technology is highly qualified in nuclear LWR's 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 corresponding temperature-entropy (*T*-*s*) diagram and the corresponding simplified piping logics are depicted in Figure 12a, c.

In case of the HCPB also the use of the Joule-Brayton process can be considered; this can become an option if coolant is available at temperatures higher than 700°C translating to a higher η_{th}. However, for those temperature levels the material challenges for the helium cooled components at simultaneously high neutron fluxes are by now not solved. An example of application with corresponding *T*-*s* diagram and the piping scheme are illustrated in Figure 12b, d.

Figure 12: T-s- diagram of the Joule-Brayton-Process (a) and the Clausius-Rankine-cycle and the corresponding simplified piping schemes (c, d)

2.3.2 Tritium plant

The blanket is the vital 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 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 entirely a coolant purification system (CPS) is intrinsically necessary to remove T from the coolant. Of course, permeation barriers to prevent tritium migration into the coolant would be desirable, the high neutron flux and the associated material damage did not allow to provide a simple technical solution. Another option is to provide a higher hydrogen partial pressure in the main helium coolant loop to prevent to a large extend the diffusion through the structures, is it is conducted in nuclear reactors, but this is currently still subject of research in fusion.

A tritium extraction is located outside the tokamak to extract the bred tritium from the purge gas and transfer it to the tritium plant, where also the tritium streams from the CPS and the divertor pumping system merge. The Figure 13 shows the functional logics for the HCPB blanket concept assuming a Clausius Rankine cycle based power conversion system.

Figure 13: Functional logics of the tritium management streams for a HCPB operated fusion plant using a Clausius Rankine cycle.

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).. Then the blankets exhibiting shut-down dose rates of several multiples of Sieverts per hour need to be extracted by means of remote handling (RH) procedures. This in turn requires supply and discharge piping schemes allowing for an extraction by dedicated tools. Already for DEMO an integration of the blanket into the thermo-nuclear core is foreseen through the upper port. This requires space reservation, which derives from the RH capabilities. The dimensions of the inlet and outlet piping are for the inboard modules 200/250mm, while for the larger outboard blankets the dimensions are 250/300mm. Potentially, the space between the pipe routing scheme could be minimised, but it is constraint by re-welding procedures. The piping scheme is illustrated in Figure 14a.

Through the upper port remote handling tools can access the core as indicated in Figure 14b. The integration of the different blanket types is carried segment wise through the back support structure as depicted in Figure 14c, more details can be found in [18].

Figure 14: (a) piping scheme for the in-/outboard blankets and the divertor. (b) Sketch of the remote handling through the upper port. (c) segmented built up of the blanket for integration in the core.

2.3.4 Balance of plant

The Balance of Plant (BoP) of a fusion power plant (FPP) describes the ensemble of heat transfer and power conversion related systems outside the tokamaks thermo-nuclear core, and include 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, to the BoP also contribute the auxiliary systems (cooling, water supply, etc.) as well as the on-site power supply.

As shown in Figure 15 the main power source is the blanket. In contrast to conventional power station a fusion power plant requires a set of high power consuming systems as heating & current drive, the cryoplant, tritium plant etc., which are all impacting the BoP.

Figure 15: Power sources and sinks contributing to the Balance of Plant in a fusion reactor.

It is clear that due to the intermitted operation of a tokamak a simple power conversion system without any energy storage unit is not fitting 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 DEMO development an energy storage system is foreseen. Likely, an energy storage is realized via thermal energy storage (TES) operated with solar salt, since solar salts match in their temperature regime the output temperature of the blanket.

In order to provide a closed BoP analysis also all other power sources as well as intermediate and low temperature heat sources (e.g. given by waste heat of auxiliary systems) need to be integrated to maximize the plant efficiency. An option, how such a power train incorporating the different heat sources of the Fusion power plant could look like is shown in Figure 16. But, how the BoP of a future fusion power plant will look is still subject of research, for which mainly uncertainties of the tokamak physics are responsible. Today 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.

Figure 16: Power train of DEMO reactor incorporating a thermal storage and the utilization various reactor internal heat sources.

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 a nuclear fusion power plant licensing. First, the blanket covers more than 80% of the plasma facing surface so that 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, like in a LWR, the residual heat generation in the FPP is not stopped completely after shutdown, but will continue at a few percent level (e.g. less than 2 % level of the gross thermal fusion plant power) at shutdown and decrease exponentially for the time after. Hence, the structures require a cooling after shut-down.

Therefore, 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. In this context in the recent years due to the activities for ITER as well in the context of the DEMO reactor development within EUROfusion a substantial progress has been made to define design and licensing requirements and to classify components and internals into safety importance classes (SIC, [21, 22]).

The core of the safety demonstration is an integrated safety analyses with a complete identification of source terms. In this context 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 led to a set of so-called Postulated Initiating Events (PIEs) which are considered as relevant reference events [20, 22]. Within the safety analysis for each of the reference events an accidental sequence is computed deterministic using numerical tools. This in turn requires the development, verification and validation of fusion adapted models and tool packages. Of course, often the basis are well qualified packages used for the safety assessment of LWR's or other nuclear applications, however, a fusion power plant substantially deviates from those not only by the nuclear source terms involved but other energetic source terms as magnetic, cryoplant and also in geometry and multi-physics interactions. Hence, the current focus of the safety analysis is directed towards development and qualification of fusion power plant adapted numerical tools.

In this context one sequence for the development and qualification of safety systems is briefly explained. For this a loss of flow accident (LOFA) in the first wall of a breeding blanket is assumed. 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 17b. In the next step measured data of the pressure drop are compared to the computed system code data (Figure 17c). Finally, an experiment is erected in which a loss of flow accident is studied at prototypical first wall fusion reactor conditions (Figure 17d)

Figure 17: Approach for the development and validation of models for the safety assessment in fusion plants. (a) hydraulic mock-up of the 1st wall; (b) comparison of computed pressure drops of CFD codes with system codes; (c) comparison of measured and computed pressure losses; (d) experimental-set-up of piping to simulate a loss of low accident at the HELOKA facility of KIT.

Once the models on the component scale are verified and validated the simplified models are transferred to the reactor scale. On the reactor scale the piping as well as potential component interactions are modelled on a nodal basis. Of course, the detailed geometric interactions gets thereby lost, however, the dynamics of the system is retained. With such an approach positions of potential component failures as well as possible release paths can be identified.

Figure 18a illustrates the DEMO reactor set-up for the HCPB blanket. It is split in 18 sectors for the primary heat transfer system (PHTS) design and consists of 6 loops for the outboard blankets (OB) and 3 loops inboard blankets (IB). 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, which location is shown in Figure 18b. For this blanket type a logical nodal set-up is created by means of a system code, for more details see [23] and Figure 18c. This model contains already 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). With this model potential failure within the blanket can be identified as well as characteristic time scales determined. Succeeding in this single segment analysis allows embedding the blanket model into a full scale plant model, which is shown for two sectors in Figure 18d. Finally, this enables to study the safety performance on a full plant scale.

Figure 18: (a) out of vessel coolant piping of a DEMO reactor with the HCPB blanket concept. (b) Piping scheme of one segment of the DEMO reactor with HCPB blankets with the highest loaded blanket segment OB4 (c) functional logics of the HCPB –OB4 blanket for a LOCA. (d) Cut of two sectors of the PHTS of a helium cooled DEMO fusion reactor.

4 Summary

This article describes recent advances in blanket engineering evolving in the transition towards realization of fusion devices by the example of helium cooled pebble blankets (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 allow for a more consistent blanket development not only by respecting engineering constraints but also enabling to study time dependent phenomena.

The engineering design derived from the basic design demands a multi-dimensional verification of the primary and secondary blankets functions, for which substantial progress has been made not only by 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. Aside from the advances in model and model validation novel developments were made in terms of heat transfer enhancement by the introduction of rib structures to increase the heat transfer of the first wall coolant ducts and especially for the fabrication methods to manufacture blanket modules. For the latter improved procedures in wire erosion, additive manufacturing as selective laser sintering developed in cooperation with industry opens new perspectives in blanket design although the nuclear licensing procedures for these are still at the beginning.

With respect to the major functional interfaces as the power conversion system, the tritium plant, the reactor integration and the balance of plant the collaboration of the European fusion laboratories in the frame of the EUROfusion project led to a more coherent reactor development as described for selected cases.

Finally, aspects for nuclear safety and licensing are described. Also in this context models and codes are developed, which are validated by experiments. Due to the complex multiphysics and multi-scale challenges in fusion deviating from those in fission reactor engineering there is still a substantial way to go in terms of verification and validation in order to arrive at a closed fusion reactor safety demonstration.

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