Review of blanket designs for advanced fusion reactors

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The dominating fraction of the power generated by fusion in the reactor is captured by neutron moderation in the blanket surrounding the plasma. From this, the efficiency of the fusion plant is predominated by the technologies applied to make electricity or hydrogen from the neutrons. The main blanket concepts addressed in this paper are advanced ceramic breeder concepts, dual coolant blankets as well as self-cooled liquid metal and Flibe blankets. Two important questions that are addressed are: (i) Can we draw a bottom line conclusion on the most promising concept(s)? (ii) What are the common issues to be resolved independently from individual design and layout proposals to define a feasible route towards advanced fusion reactors? For ceramic breeder concepts, a key issue in the long term could be the limitation of beryllium as the considered multiplier in terms of world sources and achievable temperature levels. For liquid metal blankets, attractive long-term visions have been developed but major technological challenges also exist for the in-vessel blanket technology and the corresponding sub-systems. The paper proposes a strategic conclusion derived from the review of blanket designs for advanced fusion reactors.

1. Introduction

In regard to future fusion reactors for electrical power or hydrogen production the hot power core components play a key role, especially the breeding blanket where about 80% of the fusion power is collected and tritium production takes place. Due to its exceptional relevance, the question on the right breeding blanket concept has frequently been discussed in literature (e.g. see [1]). Next to the blanket technology itself, the compatibility with advanced power conversion systems and with suitable remote maintenance strategies as well as the coolant cycles have to be considered in detail. In general the different blanket concepts have to be compared in the fields of (i) overall system and development costs and (ii) risks including all related sub-systems outside the reactor like tritium extraction, heat exchangers and power conversion, (iii) tritium control, (iv) technical feasibility, (v) reliability, (vi) ease of manufacture, (vii) maintainability, (viii) compatibility with advanced reactor layouts (integration, neutron wall and surface heat loads) and (ix) safety. Different blanket options are summarized in the paper and discussed in light of the above considerations.

First, ceramic breeder blanket concepts and lithium lead concepts will be considered. Possibilities for further improvements that could possibly exist are addressed and the question whether the concepts are likely to be able to compete on the very long-term based on their attractiveness for advanced reactor options is discussed. In addition the possibilities for future molten salt (Flibe) concepts and the Helium concept using structural parts made of tungsten are addressed for a wider view on breeding blanket concepts for future reactors. As some pre-selection was necessary, other less-known advanced blanket concepts have not been included in this paper.

A suitable breeding blanket technology will have to be proven to work reliable in a demonstration or prototype fusion reactor with satisfying performance and overall parameters and at the same
time should show a clear route towards further improvements in the ongoing technological development. If the systems applied to a demonstration or prototype fusion reactor would not allow the latter, a different blanket technology has to be brought to the market at a certain point, which most likely would require additional demonstration and prototype reactors. Therefore, when concluding on advanced concepts we have to consider three scopes which we could define as follows:

1.) Would it be attractive in the long term? This could be defined as the “advanced reactor attractiveness parameter (AAP)

2.) Is a corresponding near term variant available and attractive for DEMO? This could be defined as the “DEMO attractiveness parameter (DAP)

3.) Are most technical issues solved today and is the system relatively simple (is the R&D demand tremendous)? This could be defined as the “technological simplicity parameter (TSP)

2. Advanced ceramic breeder concept

The European helium cooled pebble bed breeding (HCPB) blanket concept is based on the use of blanket modules made from low activation martensitic ferritic steel (Eurofer), which is cooled by helium. Beryllium or beryllide in pebbles with a diameter in the range of 1 mm is used as neutron multiplier material and a lithium ceramic such as Li$_4$SiO$_4$ in pebbles with a typical diameter of 0.4–0.6 mm is used as breeding material. In the reference design the box made from first wall, top and bottom plates is stiffened by vertical and horizontal stiffening plates which are welded to both, first wall, top/bottom plates and the manifold plates at the back of the modules. In the space available between the stiffening plates the breeding units are placed, which consist of additional cooling plates and the embedded lithium ceramic pebble beds as well as the space to be filled by beryllium pebbles after the module is assembled and heat treated. The advantage of this reference design is its robustness; even a sudden pressure load with the full operational pressure of 80 bar in the box could be withstood without a break of the box. During the last years the attractiveness of the completely helium cooled design was further improved by introducing more efficient heat transfer mechanism and attractive remote handling concepts based on the so-called multi-module segment approach [2,3]. The HCPB concept allows for a relatively simple integration scheme as there is no need for draining and connection of liquid metal flow channels with internal flow channel inserts (FCI).

For advanced HCPB designs several additional modifications will be necessary. It is very likely that the construction of very robust sub-modules will be given up in completely helium cooled reactors to allow simplifying design and fabrication. Already for the advanced HCPB blanket based on a SiC$_x$/SiC-composite as structural material [4] a rupture disk was proposed to handle the accidental situation of high-pressure coolant leakage, which may cause pressurization of blanket modules. In this concept two SiC$_x$/SiC components were suggested, cooling plates formed by long meanders separating the breeder ceramic pebbles from the beryllium pebbles and a first wall made from a series of parallel tubes from SiC$_x$/SiC. This design could allow outlet temperatures in the range of 700°C.

Without relying on SiC$_x$/SiC based structural components, the main principle to be applied for high helium temperature output of HCPB blankets is the use of concentric arrangements of flow channels where cold helium is flowing between the outer wall and the outside of the inner wall. The hot helium gas flowing inside the inner wall should be separated thermally from it, e.g., by flow channel inserts made of SiC$_x$/SiC. Most likely this measure is combined with the use of beryllide rather than beryllium as this allows (i) higher operational temperatures and (ii) mixed beds with ceramics allowing thicker beds when compared to beds made from ceramic pebbles only.

For the heat transfer from the ceramic and multiplier pebbles in high temperature HCPB concepts there are two possibilities to be considered. In the first case the high pressure coolant itself is directed through the pebbles and the bred tritium will be collected within the main helium coolant flow. This principle was suggested for A-SSTR-2 [5], where it was combined with a structure made from SiC$_x$/SiC. In general this arrangement combined with a SiC$_x$/SiC structure could allow helium outlet temperatures as high as 900°C, when combined with a First wall made from advanced ferritic steel the inlet temperature has to be reduced and the outlet temperatures achievable will be reduced as well, but 800°C could be still possible.

In the second case the breeder material is placed in tubes. This principle was investigated already in the past, e.g., in Europe known as breeder inside tubes (BIT) concept [6]. It is proposed here to place the breeder pebbles in small packed tube arrays (Fig. 1) which at their outside are cooled by the main helium coolant flow. Inside the small tubes containing the ceramic pebbles the tritium is collected by a separate helium purge flow as it is the case in the EU HCPB DEMO reference design. The small circular breeder pebble containing tubes are made from an advanced high temperature steel
allowing operational temperatures up to 800 °C such as ferritic ODS (oxide disperse strengthened) steels. Therefore, for this concept the helium outlet temperature could be very close to 800 °C. While the achievable helium outlet temperature could be very attractive the large number of pebble containing tubes is the disadvantage of this configuration.

Another HCPB option for outlet temperatures around 700 °C is illustrated in Figs. 2 and 3. In this concept concentric pipes are arranged in U-form inside the blanket boxes (Fig. 2) and fed from the manifolds at the back of the boxes. The diameter of the pipes is varied to accommodate for the reduction of the volumetric heat generation in radial direction. The concentric pipes consist of three thin steel tubes and a thermal insulation inside the second tube. The helium enters the pipes in the annular gap between the outermost tubes. The volume around the concentric pipes is filled with a mixed bed containing mostly beryllide and a small fraction of ceramic pebbles. The volume between the insulation layer and the innermost tube is filled with a mixture of ceramic and beryllium pebbles operated at a temperature up to 900 °C, while the coolant reaches an outlet temperature of 700 °C. As the thermal conductivity in mixed beds is better than in pure ceramic beds, the number of single tubes can be kept relatively low. The layout of the system is quite flexible and could be optimized for specific reactor conditions.

In Japan, a concept of the high temperature fusion plant was developed with ceramic breeder pebbles and SiC/SiC composite blanket structure. In the design of the advanced steady-state Tokamak reactor 2 (A-SSTR2) [5] by Nishio et al., high power density reactor with 20-T strong magnetic field with He cooled blanket is studied. Gas turbines based on the high temperature gas cooled reactor (HTGR) study pursued in Japan could be combined with advanced ceramic breeder blankets (Fig. 4).

2.1. Issues and R&D demand of advanced ceramic breeder concepts

In the case of the directly cooled bed scenario the tritium extraction from the hot high pressure helium main stream and tritium inventories in the tritium recovery system are severe issues. It has to be kept in mind that for tritium extraction reasons from the ceramics a reducing atmosphere is needed, but for tritium permeation reduction an oxidizing atmosphere is needed. Another matter of concern is the mass flow distribution and the corresponding temperature distribution in the pebble beds as well as the pressure drop in the pebble beds. From this, a blanket option offering separated main coolant and tritium extraction atmosphere seem to offer much less risks.

For mixed beds alpha-particles from the ceramics could cause a porous layer on pure beryllium pebbles under irradiation. In these porous layers high tritium inventories could possibly occur. R&D is also necessary to address the compatibility between steel and ceramic breeder at strongly enhanced temperatures up to 800 °C.

The control of the tritium breeding ratio during operation in case of ceramic breeder concepts was not considered sufficiently up to now and the question whether it would be possible to use...
dummy blankets in dedicated outboard ports that could be replaced if necessary by adjusted breeding modules needs to be analyzed in the future. One general question in regard on beryllium containing blankets concerns the availability of sufficient beryllium on the long term. While the amount of beryllium world sources is estimated to be about 80,000 tons, the current worldwide demand is not more than 100 tons per year, less than the amount needed for a DEMO fusion machine. An alternative high temperature compatible solid neutron multiplier option would be highly preferable on the very long term and therefore should be part of the research programs. As beryllium allows for attractive blanket energy gains its replacement by other multipliers might reduce the performance of ceramic breeder concepts in terms of their power production efficiency.

For the versions using SiC/SiC structures, additional issues are all issues related with the required R&D still necessary for this material as discussed in more details in Section 4.

3. Dual coolant concept

A large variety of lead lithium breeding concepts have been proposed up to now. As today Pb-15.7Li is considered the best choice but various studies are based on Pb-17Li we use the common abbreviation PbLi. A completely helium-cooled PbLi blanket option, the HCLL concept, has been assessed in the PPCS/DEMO studies as one of the blanket options for DEMO and is proposed by EU for the ITER test blanket module (TBM) program [7]. The HCLL concept is the short-term variant of the PbLi concepts and from this is considered suitable for DEMO. In contrast, dual coolant concepts utilize helium only to cool a ferritic steel structure (including the first wall) and slowly flowing PbLi acts as self-cooled breeder in the inner channels [8]. The type of dual coolant concepts considered in this paper allows for operating the breeder at a higher temperature than the structural blanket walls by means of flow channel inserts (FCI from SiC, Fig. 5) to maximize the power cycle efficiency [9]. Use of a liquid breeder also provides the possibility of active tritium breeding control during operation by adjusting the $^6$Li enrichment. Use of He coolant as a separate coolant for the first wall/structure also facilitates the pre-heating of the blankets before the liquid breeder is filled in, serves as guard heating in case the liquid breeder can not be circulated, and provides independent and redundant decay-heat removal in case the liquid metal loop is not operational. In addition, cooling the first wall region of the blanket (where the heat load is highest) with helium (instead of a liquid metal) avoids the need for an electrically insulating coating in this high velocity region to prevent the large MHD pressure drop associated with liquid metal flow.

Such a dual coolant concept with FCI was originally developed as part of the ARIES-ST study [9,10] and then at FZK in Germany [11]. It has been further developed as part of the ARIES-CS study [12], and is also now being considered as a US test module candidate for ITER [13]. As an example, the modular concept adapted for the ARIES-CS compact stellarator geometry is described here; the design procedures included consideration of credible fabrication and maintenance procedures as well as optimization of the design geometry and parameters (of the blanket and coupled Brayton cycle) within the compact stellarator power plant [12].

The blanket module concept is illustrated in Figs. 6 and 7, with both helium and PbLi fed to the back of the blanket module through concentric pipes. The helium coolant (assumed at 10 MPa) is routed to first cool the first wall in a single pass with an alternating toroidal flow configuration to create a more uniform temperature (and reduce thermal stresses); it is then routed in a combination of series
Fig. 8. The HTL blanket module structure (left) and PbLi flow scheme inside module (right).

The PbLi flows slowly (∼10 cm/s or less) in the large inner channels in a two-pass poloidal configuration. The PbLi channels are lined with a SiC FCI (with no structural function as the thin Pb-17Li layer and the bulk PbLi are pressure-balanced through a thin slot or holes in the SiC), as illustrated in Fig. 5. This FCI plays a key thermal insulation function to allow high temperature (∼700 °C) PbLi in the channel while the PbLi/steel interface temperature is maintained below its compatibility limit. It also provides the electrical insulation needed to minimize the MHD effects on the PbLi pressure drop and flow distribution. To help increase the coolant temperature at the first wall (and thermal efficiency) while maintaining the reduced activation ferritic steel (RAFS) temperature within its 550 °C limit, the first wall (FW) is made up of a layer of ODS ferritic steel diffusion bonded to the thin RAFS module wall.

Several variations of the lithium-lead-based blankets with dual-coolants have been heliumed for a series of fusion power plants (named FDS series) in China [14,15]. A He/PbLi Dual-cooled lithium-lead (called DLL) blanket for FDS-II [16,17] has been designed in details as a “multi-modules” structure by means of FCI to allow PbLi use at high temperature of ∼700 °C. It features a big rectangular steel box which is enclosed by U-shape FW, covers, and back plates (BPs). The structural material is made of the reduced activation ferritic/martensitic (RAFM) steel, e.g. CLAM—China low activation martensitic steel [18]. A concept of dual functional lithium lead (DFLL) testing blanket module has been developed to flexibly validate and demonstrate both the technologies of single coolant and dual coolant lithium blankets for ITER [19,20].

An advanced dual coolant blanket concept named HTL (high temperature liquid) blanket has been developed by the FDS Team to obtain a high PbLi outlet temperature of about 1000 °C with the RAFM steel as main structural material and using multi-layer SiCf/SiC flow channel inserts (MFCI) in the PbLi flow channels [21,22]. Fig. 8 shows the HTL blanket structure and PbLi flow scheme. The breeder zone is configured three concentric PbLi channels by MFCIs. The directions provide restrictions in toroidal and radial between layers of FCI forming PbLi channel. He cools the structural components to keep them below the RAFM steel limit of 550 °C and PbLi is heated to high outlet temperature when meandering through the MFCIs, as no additional cooling is foreseen for the breeding zones. The 400 °C PbLi feeds into module from outside pipe of manifold, then flows in series in outside, middle and inside FCI channel, and finally, about 1000 °C PbLi flows out of module into inside pipe of manifold.

Recently in Japan, a series of experimental efforts for the DCLL concept has started on the technological development of the PbLi-SiC-RAFM combination with high temperature PbLi loop at Kyoto University. This study is not based on a specific blanket design, but intends to provide basic data for DCLL and self-cooled PbLi concept. The loop contains test section made of NiTE SiC composite tube and operated in the range of 500–900 °C. Compatibility of materials with PbLi, hydrogen permeability, solubility and diffusivity, MHD pressure drop in an insulating SiC tube are being measured. Technology for utilization of high temperature fusion heat is also developed on intermediate heat exchanger and

Fig. 9. 3D image of the TNT loop, which is now operated using heat transfer salt (HTS) as a Filbe simulant and has the total capacity of 0.1 m³ and can be operated at 600 °C.
hydrogen production process from biomass, those would be an essential part of the future fusion energy systems.

In India a baseline design for the IN-DEMO reactor has been carried out. A dual coolant lead–lithium ceramic breeder (LLCB) concept for the blanket is being considered with low activation ferritic steel as the structural material. The ceramic breeder material is lithium titanate and multiplier/coolant is lead lithium. A graphite reflector is used at the end with tungsten carbide and stainless steel as shielding material. Also a LLCB-TBM is proposed by India to be tested in the ITER environment. Since this concept is a blend of solid and liquid breeder concepts, it would inherit pros and cons of both the concepts and the challenge is to demonstrate its novelty, engineering feasibility and superiority of design. The neutronic model (using Monte Carlo code) is undergoing refinement to update the various nuclear analyses, including activation, dose-rate decay, heat, and shielding.

3.1. Main issues and R&D demand of dual coolant concepts

Key R&D issues include material characterization, development and fabrication (such as compatibility limits of FS/PbLi and of SiC/PbLi under prototypical conditions and development of ODS FS), as well as a better understanding of MHD effects. Other main challenges exist in the area of sub-systems like heat exchangers and tritium extraction systems. A tritium extraction system is proposed based on tritium permeation through tube walls made of a refractory metal from the flowing liquid breeder into a vacuum chamber. This method seems to have the potential to reduce the maximum tritium partial pressure in the PbLi to values <1 Pa, facilitating in this way tritium control decisively. However, the concept needs experimental verification and the use of refractory metals for large components could not be realized without strong R&D programs. Outside the Tritium extraction system, of course, Tritium permeation has to be reduced by reliable permeation barriers. The power conversion system should be a closed gas turbine cycle, whereby the gas–water heat exchangers (pre-coolers) as part of the gas turbine cycles have to be considered in regard to the possibility of tritium being captured in water.

4. Self-cooled PbLi

Fully self-cooled blankets using SiCf/SiC as structural material have been proposed for TAURO [23] and the ARIES-AT study [24]. The self-cooled lithium lead (SCLL) blanket mainly addressed in this section has been studied as model D in the framework of the European Power Plant Conceptual Study (PPCS) [25]. The blanket is formed by only two materials: the SiCf/SiC structure and the PbLi which acts as breeder/multiplier and coolant. Within PPCS, this reactor model is associated with the largest attractiveness and at the same time with the largest development risk. In fact, the SiCf/SiC structures [26] allow high coolant temperature, and show very low short term activation and afterheat levels. Associated with the use of PbLi (that is a material easily recyclable) as breeder, coolant, neutron multiplier and tritium carrier, this system achieves high plant efficiency and has the potential for reaching good safety standards. This choice leads to have low afterheat, low operating pressures and low chemical reactivity with water and air in the in-vessel components and, therefore, permits to minimize the energy stored in the vacuum vessel.

The design is based on the principle of coaxial flow which allows having a maximum PbLi outlet temperature of 1100 °C without exceeding the limit of 1000 °C for SiC/SiC. The 6-mm-thick first wall is protected by a 2-mm-thick layer of tungsten. Each outboard segment is formed by five modules attached on a thick back plate that gives sufficient strength for segment transport during replacement and allows its attachment to the back components. The PbLi enters from the bottom in the thin external layer, turns down at the top and flows down at low velocity in the central region.

The analyses have been performed assuming a SiCf/SiC thermal conductivity after irradiation of 20 W/m/K that is significantly larger than the values achievable in present-day composites. Performed MHD analyses have shown that, assuming a magnetic field of 4 T in the outboard blanket and 8 T in the inboard ones, the corresponding pressure drops are respectively 0.44 and 0.85 MPa. These pressure drop values are obtained assuming fully established flow in the blanket which seems to be justified for the very elongated blanket geometry. Three-dimensional effects and pressure drops in the supplying lines will give additional contributions which should be evaluated in future analyses.

The high coolant temperature allows to theoretically reaching reactor efficiency as high as 61%. Such a high temperature could also allow the use of such a system for direct H-production.

Because of the required high PbLi velocity, the PbLi in the blanket is renewed more than 1000 times a day. Therefore, the T-partial pressure in a single pass remains very low and it is neither necessary nor efficient to try to extract on-line all the produced tritium. It is therefore proposed to derive a fraction (say 1%) of the PbLi flow after having passed through the heat-exchanger in order to avoid too large heat losses. The extractor could be characterized by He-bubbles in counter-flow to the PbLi at 700 °C and achieve T-extraction efficiency as high as 80%. Material issues have not yet been assessed.

PbLi purification may be necessary for extracting radioisotopes characterized by high ingestion and inhalation hazard potential in case of accidental release. The purification may occur in-line or by batches depending on the chosen criteria.

4.1. Main issues and R&D demand for the SCLL blanket

R&D requirements concern issues directly related to the blanket designs such as the SiCf/SiC properties, the in-vessel components fabrication, the pipes connections and attachment systems, the development of the out of vessel components, in particular the high-temperature heat exchanger. In terms of priorities, thermal conductivity through the thickness, effects of irradiation, and brazing/joining techniques are clearly the most important items to be addressed. At present, in EU a limited R&D effort is devoted to this advanced SCLL blanket. Most R&D is focused on the improvement of SiCf/SiC composite characteristics and on SiCf/SiC compatibility with lithium–lead. Main on-going activities in EU are the manufacturing of advanced 2D/3D composites featuring improved thermal conductivity of about 20 W/m/K through the thickness (before irradiation), their mechanical and thermal characterization, and corresponding irradiation testing up to a dose of 8 dpa (C).

5. Self-cooled Flibe

The molten salt Flibe, especially the composition (LiF)2(BeF2), offers good tritium breeding characteristics, low MHD pressure drop due to its low electrical conductivity, and in general better safety features than liquid metal breeders due to the low chemical reactivity with air and water, and the exceptional low tritium inventory. For these reasons, it became the reference blanket in the Force Free Helical Reactor (FFHR) [27].

Major drawbacks for the use of Flibe in the breeding blankets of attractive power plants are the extremely low thermal conductivity (about 1 W/m/K) at 500 °C, compared to 15 W/m/K for PbLi), an extremely high viscosity (100 times higher than of the water in a
pressurized water reactor!), and a melting point of 459 °C, requiring advanced DDS steels with an operating temperature of > 650 °C as structural material. To mitigate the corrosion of the structural material, FT has to be reduced to T3 through contact of the Flibe with Be which serves at the same time as neutron multiplier.

A novel Flibe blanket concept has been proposed in the US APEX study [28], named recirculating flow concept [27,29]. In this concept, only about 25% of the entire FW flow is send to the power conversion system, and the larger part (−75%) of the flow is recirculated directly to the blanket in order to combine efficient cooling of the FW with a high exit temperature for an efficient Brayton cycle power conversion system. By this measure, it was possible to design a self-cooled Flibe blanket with a maximum neutron wall load of 5.4 MW/m² and a maximum FW surface heat flux of 1 MW/m², leading to an efficiency of ∼45% in the power conversion system.

Related to the self-cooled Flibe blanket in FFHR design activities [30], many R&D’s have recently made great progress in the Flibe issues. At the level of 1 MW/m², the Tohoku-NIFS-thermofluid (TNT, Fig. 9) loop experiments have revealed a four times enhancement (steel structures) (SiC/SiC structures) of heat transfer by adoption of sphere-packed pipe (SPP) flows at a low flow velocity range around 4 m/s, resulting in large merits on reducing the total pressure loss and MHD effects [31]. Regarding materials compatibility, the Japan-US joint program JUPITER-II has revealed remarkable and stable suppression of HF in Flibe due to dissolved Be, confirming the quite effective Redox control [32]. Concerning tritium permeation from the point of view of safety, double walled tube concepts have been evaluated and the He gas or pure Flibe with low flow rates are shown to be sufficient and feasible as a permeation barrier [33], where the tritium leakage at the heat exchanger is remaining as a severe issue. Regarding the operation temperature window limited by Flibe and structural materials, a Flibe/V alloy blanket concept has been proposed [34], where, in spite of Be Redox, a W coating using WFe mixed in Flibe is a candidate and disengagement of HF from Flibe becomes a new key issue.

6. Evaporation of lithium and vapor extraction (EVOLVE) blanket concept

In the frame of the US APEX study [28] a very ambiguous goal of a maximum neutron wall load of 10 MW/m² together with a FW surface heat flux of 2 MW/m² had been set as criteria for the selection of blanket concepts.

This was the starting point for a novel concept, based on the use of the exceptional high heat of vaporization of lithium (about 10 times higher than water) for an effective heat extraction from the first wall as well as the breeding zone [35,36]. A reasonable range of boiling temperature of this alkali metal is 1200–1400 °C, corresponding with a saturation pressure of 0.035–0.2 MPa. For this temperature range, tungsten alloys are suitable structural materials due to the high strength, the high thermal conductivity, and the excellent compatibility with boiling lithium. Calculations indicate that such an evaporating system with Li at ∼ 1200 °C as breeder/coolant can remove a neutron wall load of > 10 MW/m² with an accompanying first wall surface heat flux of >2 MW/m². Thermal-hydraulics and neutronics calculations have shown, that the system provides adequate tritium breeding and shielding, very high efficiency of the Brayton cycle power conversion system (>55%), and low stresses in the blanket structure due to the low coolant pressure.

The main issues are the fabricability of the tungsten alloy structure, and the performance under high fluence neutron irradiation. The minimum operational temperature of > 1200 °C is well above the ductile-brittle transition temperature, and the temperature variations inside the entire structure are < 100 K due to the high thermal conductivity and the excellent heat transfer under boiling conditions. Because of the demanding operating conditions and the level of material R&D required, this concept is considered more as a longer term, later generation concept.

7. Conclusions

As highlighted in the introduction, next to the advanced scenario attractiveness also the attractiveness for DEMO and the technical simplicity are important criteria to be taken into account when deriving a strategic conclusion from the review on advanced blanket concepts. Based on today’s knowledge the ratings shown in Table 1 seem to be possible without falling into the trap of pseudo-accurate schemes with large numbers of parameters.

As the ceramic breeder class of concepts is one of the good candidates for an ultra fast track scenario and reliable DEMO application, their further development and tests in ITER are necessary. If tritium...
production in ITER would be required to allow for extending its operation or for starting DEMO machines. ITER application with low-temperature water-cooled ceramic breeder blankets could also be envisaged. For increasing the long-term perspectives new efforts should be undertaken to allow for replacing Be-based multipliers. Visions for outlet temperatures in the range between 700 and 900 °C have been developed, while the R&D level reached for these concepts is still low. Only when overcoming the open issues, advanced HCPB concepts would be attractive for hydrogen production. Development of a credible in-situ tritium control system is also needed.

Also PbLi concepts are good candidates for an ultra fast track scenario and reliable DEMO application, in particular with the short term variant HCLL. Dual coolant blankets could become the best choice for DEMO application if open issues can be solved in time by adding adequate resources to R&D programs. Once the reliable operation and economical attractiveness of dual coolant blankets is reached, ceramic breeder and exclusively helium-cooled PbLi blankets could become dispensable. At this point the next logical step would be the focus on advanced DCLL and SCLL concepts.

The important question is whether DCLL will allow for reliable operation of a DEMO machine after ITER. If this is not the case, DEMO could be based on HCPB or HCLL blanket concepts and DCLL could be included in form of a test program. In the case of a HCLL-based DEMO the efficiency would be slightly reduced in comparison with the HCPB blankets, while its commonality with the advanced PbLi concepts could be an advantage. Some lessons for the DCLL concept could be derived from the HCLL system, which in addition to the HCPB/TBM is being prepared by EU for tests in ITER. However, to address the different issues caused by use of flow channel inserts, much higher PbLi mass flow, higher breeder temperature, lower tritium partial pressure and different MHD-effects a straightforward DCLL test already in ITER – as proposed by the US – would be reasonable to keep the option of a DCLL-based DEMO reactor.

Filile concepts could be an alternative, especially when unsolvable problems with PbLi blankets would occur. EVOLVE concept might be a possibility to be considered in the far future.

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