European Material Assessment Meeting

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by

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EXECUTIVE SUMMARY

A. Introduction

- The long term development towards fusion power plants (DEMO and a PROTOTYPE reactor) aims for materials which can withstand high neutron wall loading and heat fluxes and coolant pressure conditions at temperature attractive for efficient thermodynamic working cycles. In addition the materials should be compatible with high neutron fluences (10-15 MWy/m² in a power reactor) to limit the necessary replacement of the in vessel components to a minimum and should be of “low-activation” type to maintain one of the most attractive features of fusion.

- The main objective of the Material Assessment Meeting (Karlsruhe, July 2001) was the evaluation of the attractiveness and technological critical issues of the structural materials for fusion power reactor. Functional materials for high heat flux components (limiter, divertor) and for increasing the operation temperature in breeding blankets have been also considered. With respect to the advanced materials (SiC/SiC, Cr, V, Ti, W alloys) their potentialities and technical risks were analysed in order to guide the definition of the priorities in the R&D technology programme. The assessment was performed by comparing the point of view of material experts and design experts of the presently considered nuclear reactor components (breeding blankets and divertors).

- Environmental and safety aspects are crucial for fusion acceptance. The main features to be considered in defining fusion material area are: low long term activation to minimise the waste through clearance and recycling; low decay heat and low short term activation to ease maintenance and cooling and good compatibility to avoid energetic chemical reactions.

B. Materials – Present Status

- Three major structural materials which can fulfil the “low activation” requirement have been considered in different first wall and breeding blanket concepts designs: ferritic-martensitic steels, vanadium alloys and SiC/SiC ceramic composites. These three materials groups are pursued in national and international materials R&D programme in EU, Japan, USA and Russia Federation. The ferritic-martensitic steels are the furthest in development and show the fewest areas of concern. They are the reference structural materials for the in vessel components of DEMO. This type of materials will be used in the two EU test Blanket Modules (TBMs) to be tested in ITER: the Water cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB).

- Irradiation devices are used to assess the irradiation performance of the materials. In particular:
  - Thermal Reactor provide parametric studies on irradiation damage behaviour;
  - Fast Breeder Reactor allow to achieve high fluence and high dpa rates at T>300 C;
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- Irradiation devices are used to assess the irradiation performance of the materials. In particular:
  - Thermal Reactors provide parametric studies on irradiation damage behaviour;
  - Fast Breeder Reactors allow to achieve high fluence and high dpa rates at T>300 °C;
- SINQ (a spallation neutron source) leads to a very high He/dpa ratio (about 60 appm/dpa with respect to 11 foreseen in DEMO) combined with a relatively high fluence rate (10dpa/y);
- accelerator facilities allow for proton irradiation and helium implantation.

In order to complete the qualification of materials up to the full lifetime of a blanket for DEMO and a PROTOTYPE reactor (about 80 and 150 dpa respectively) an intense neutron source with relevant fusion neutron spectra will be essential. The accelerator based D-Li stripping source IFMIF (International Fusion Materials Irradiation Facility) developed under the framework of the Fusion Materials Implementing Agreement of the International Energy Agency appears to be the best choice with the potential to fulfil the demanding requirements within the required time scale.

- The selection and development of Ferritic-Martensitic 9-12% Cr steels with Reduced Activation (RAFM steels) for nuclear fusion is based on the excellent experience with commercial 9-12% Cr steels. The possibility to achieve a reduced neutron-induced long-term activation alloy, by an appropriate chemical modification of major alloying elements and a consequent limitation of harmful impurities which worsen the long-term radioactivity properties as well as the potential for recycling, were further reasons for the selection of this material group.

- As a conclusion of a development carried out in several EU Associations, an industrial batch of a 9% CrWVTa called EUROFER was specified in 1997 and produced by Böhler Austria with the presently available steel manufacturing technology in 1999. Different semi-finished products (bars, plates and tubes) with a sufficiently low content of radiologically unwanted impurities like Nb, Mo and Ni were delivered. The material is presently extensively tested in different European laboratories. First results of the characterisation campaign are promising. Equal (tensile, creep) or higher (toughness) mechanical properties have been found compared with those of the Japanese heat F82H prior and after irradiation up to 2.7 dpa.

- At present it can be stated that this first European industrial batch of RAFM steel has fulfilled the expectations. The planned irradiation experiments in material test and fast breeder reactors will show whether or not his positive finding will also hold under higher fluence levels.

- An obvious disadvantage of RAFM alloys is the limitation of the maximum operation temperature to about 550 °C for creep strength reasons, which is expected to be overcome by the development of Oxide Dispersion Strengthened (ODS) variants. In Europe the general approach is to use the EUROFER 97 alloy composition as matrix and to add Y2O3 dispersion in the range of 0.2 to 1 wt-%. It is expected that the upper temperature limit of application can be extended to 650 °C or even higher. The present R&D in EU laboratories aims at investigating different production routes and at assessing the potential of such alloys.
• The Vanadium based alloys have high potentiality for reactor concepts based on liquid lithium as breeder and coolant at high temperature. The development of self-healing, corrosion-protective and at the same time, insulating coatings (to mitigate the Magneto-Hydro-Dynamics effects) is the key issue for the future use of such a type of structural material.

• SiC/SiC ceramic composite materials have potentially high pay offs in terms of very low radioactivity and decay heat at short and intermediate decay times and offer high operating temperatures (i.e. power plants with high efficiency). The development of neutron irradiation resistant composites is a key issue. The progress made relies mainly on the availability of almost stoechiometric fibres with higher thermal conductivity, higher thermal stability and higher expected neutron irradiation resistance. Using these fibres, composites with thermal conductivity as high as 85W/m.K at Room Temperature have been produced. An advanced SiC/SiC composite using stoechiometric fibres, SiC in the fibre/matrix interface and ß-SiC matrix will be manufactured in EU during 2002.

• Exploratory studies have been carried out in EU on Chromium and Titanium alloys with the aim to improve the ductility by refining the microstructure (Cr) and define irradiation resistant alloys (Ti).

C. Breeding Blanket Design – Present Status

• Breeding blankets have been largely studied in the last decade. The EU strategy is to test in ITER some blanket mock-ups of the two EU reference blanket concepts presently developed for DEMO: the Water cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB). These mock-ups should be equipped with all DEMO-relevant technology for which a significant R&D programme is in progress.

• Taking into account all the requirements for a Fusion Power Reactor, “attractiveness” of a given concept can be qualitatively defined in terms of availability, safety, waste management, efficiency, cost of electricity, duality of the development with other research fields. The corresponding risk of development can be also identified. In the materials assessment, together with the two reference medium-term blanket concepts associated with relatively low development risk, some more advanced concepts with an overall improved attractiveness but increased development risk have been considered.

• The considered concepts, listed in the order of increased attractiveness but increased development risk are the following:
  - the two EU reference DEMO concepts, both using RAFM steel as structural material, i.e. WCLL and HCPB blankets;
  - the Dual Coolant (DC) blanket which uses RAFM steel as structural material and Pb-17Li and He coolants (with SiC/SiC flow channel inserts acting as electrical insulators);
- the self cooled Lithium blanket using V-alloy as structural material with electrical insulating coatings;
- three blankets using SiC/SiC as structural material: two self-cooled Pb-17Li blankets (TAURO and ARIES-AT) and one He-cooled ceramic breeder/Be concept proposed in Japan (DREAM);
- a self-cooled Lithium blanket where the W-structures are cooled by Li-evaporation (EVOLVE).

D. Divertor Design – Present Status

• Compared to blankets, the divertor is submitted to a much higher surface heat flux (5-10MW/m²) but to a lower neutron wall loading (about 1/3 of that of the first wall) and it can be envisaged a more frequent replacement. The lifetime is mainly dictated by erosion and neutron irradiation damage. A sacrificial layer of armour material (3 mm of W as a minimum) is required.

• Very limited and preliminary studies of divertor designs are discussed in the literature beyond those required for ITER. Five concepts have been evaluated in EU in the framework of the reactor study activities:
  - high pressure high temperature (Pressurised Water Reactor conditions) water-coolant, with RAFM (ODS) steel as structural material and W in monoblock geometry as armour;
  - a He-cooled W structure assuming conventional heat exchange laws and monoblock geometry;
  - a second He-cooled W-structure in a similar monoblock geometry, but assuming enhanced heat exchange permitted by the use of porous walls;
  - Pb-17Li coolant flowing in a SiC/SiC structure to reduce MHD effects and W tiles as amour;
  - Liquid Metal (Li, Na, NaK) cooling, based on evaporation principles associated with W structure and armour (monoblock geometry).

In all these concepts the maximum acceptable heat load is ranging between 5 and 7MW/m². The only concepts where more than 10 MW/m² can be envisaged are based on the ITER design with a Cu alloys structure and a low cooling water temperature.

E. Material Developments - Findings

• RAFM steels are the primary choice in EU for First Wall/Blanket structural applications. The use of this material should be restricted to zones where high temperature strength and high irradiation performances are required. Appropriate shielding materials optimising the requirements of low activation, low after heat and low cost should be selected.

• A new EUROFER 97 batch will be ordered to provide materials for TBM mock-ups and prototypes, to check reproducibility of steel making and to improve the quality of
the semi-finished products (tubes in particular). After results from running irradiation experiments a new heat “EUROFER 2” is foreseen.

- Design criteria and rules will be established in order to provide the designers of TBM and DEMO with an appropriate database.

- Irradiation experiments within the temperature range 250-400 °C are of primary importance in order to determine the DBTT and the fracture toughness properties. The lower design temperature limit (250 °C) should be verified. The upper design temperature limit for EUROFER is 550 °C considering 20000 hours of operation with maximum stress of 100 MPa. Creep strength of highly irradiated EUROFER should be determined. Creep fatigue has to be analysed for TBM testing in ITER.

- Corrosion behaviour of activated materials especially in water-loops and H-embrittlement under irradiation are areas to be investigated in more detail. Additional effort should be also put in welds and joints development.

- Modelling and theory should be focussed mainly in two areas of interest (i) combined effects of irradiation damage, H and He production, plastic flow and embrittlement and (ii) small specimen test technology to be qualified towards accepted standards.

- ODS steels have some potential for higher performance in HCPB and DC concepts. Priorities in development and qualification should be given for near term applications of ODS in selected FW/blanket zones with the objective to increase the maximum temperature from 550 °C (Eurofer limit) to 650 °C (reasonable extrapolation from the present data). Improvement of fabrication processes (joining in particular) for the production of components is needed.

- A limited development of SiC/SiC composite is needed for application in the DC breeding blanket concept. Mainly SiC/Pb-17Li compatibility and effectiveness of SiC/SiC as electrical insulator to mitigate MHD effects should be investigated. Advanced application of SiC/SiC composite requires to improve the material itself and to develop a material engineering database. In particular critical issues to be investigated are:
  - thermal, thermomechanical behaviour and dimensional stability under irradiation;
  - hermetic sealing of the SiC/SiC surfaces to avoid contact with coolant, breeder and neutron multiplier materials;
  - new approach in the design procedure.

- Europe is not considering at present lithium as breeder/coolant in a breeding blanket design; therefore no specific R&D is planned for the development of Vanadium alloys.

- Chromium and Titanium alloys could be used in some specific parts of the fusion reactor. No interesting conceptual designs have been proposed up to now for in vessel
structural applications. The major drawbacks are: brittleness at low temperature for Cr and hydrogen embrittlement for Ti.

- Tungsten as structural material is employed in the EVOLVE design blanket concept which claims exceptional high performance. The feasibility of fabrication and irradiation behaviour of a tungsten structure is completely unproven and its development and qualification would require a substantial increase in the material budget.

- The divertor is a critical component for Tokamak reactors and a strong effort is recommended (i) in the conceptual design development of a whole divertor system in relation to each selected blanket concept and (ii) in the development and characterisation of the required material.

- With respect to the armour materials of a divertor:
  - Tantalum is the best from the sputtering point of view, but considering the hydrogen effects (embrittlement and high tritium inventory) was not selected;
  - Tungsten remains the most promising option although it is very brittle;
  - Molybdenum alloys offer an easier fabrication procedure but present a high long-term activation.

The R&D activity should focus on W alloy as first choice and Mo alloys as back up solution with the aim to improve the ductility under irradiation and define the operating temperature window.

- Materials for high heat flux/high efficiency divertors is a key issue. The ITER divertor is not attractive from the point of view of the reactor efficiency (CuCrZr alloy is used at low temperature < 350 °C). Improvement can be obtained developing new high conductivity Cu-based materials and tungsten alloys. Cu-W alloys and Cu-composites with CFC or SiC fibres have been manufactured and their stability up to 500 °C were demonstrated in non-irradiated materials. Irradiation effects are unknown. These materials could be used in divertors with water as cooling medium at PWR condition (265-325 °C; 15.5 MPa). Tungsten alloys are the best potential candidates in the design of helium cooled divertor; advanced concepts could withstand a heat flux of up to ~10MW/m². Characterisation after irradiation, improvement of ductility, compatibility and fabrication processes (including joinings) are the main areas of investigation.

- Qualified nuclear databases and codes are required for material development and for the design and licensing of fusion reactors internal components. European nuclear data and activation libraries have been developed and extended to neutron energies of 20 MeV for these purposes. Validation will be achieved through benchmark experiments. Extrapolation of the database to even higher neutron energy in order to cover the IFMIF domain is also needed.
F. Conclusions

- EUROFER is the reference structural material for the two DEMO blanket concepts presently developed in EU (HCPB and WCLL) to be tested in ITER. The Dual Coolant concept studied in EU is also based on RAFM steel as structural material. Priorities in the R&D activities should be given to:
  - characterisation of EUROFER 97 for TBM design and licensing and development of material processing (joints, Hot Isostatic Pressing, etc...);
  - characterisation of EUROFER 97 at high fluence up to 80 dpa in Fast Breeder Reactors and up to at least 15dpa in Materials Testing Reactors;
  - possible further improvement or optimisation of EUROFER 97 composition after availability of results from high fluence irradiation experiments.

- ODS can be used as functional materials in order to improve the performance of the present HCPB and WCLL concepts with limited effort in R&D.

- As part of the long term programme at least one additional structural material for advanced blanket concepts should be evaluated and the critical issues addressed. SiC/SiC ceramic composites have been considered as a promising alternative in the EU blankets conceptual studies for its potential to increase the thermal efficiency.

- In addition to the present irradiation facilities for material characterisation (MTR, FBR, Accelerator facilities) an intense neutron source with relevant neutron fusion spectra is absolutely needed. Considering technical development risks, performance (neutron spectra, fluence, available test volume) and cost, IFMIF has been proposed at international level as the best solution to meet fusion requirements.

- The choice of a specific divertor concept can only be made in relation with the selected blanket system. A strong effort in the divertor R&D activities is required and the following priorities are recommended:
  - development of the conceptual design of a whole divertor system in relation to each selected blanket concept;
  - development and characterisation of the required materials;
  - design and fabrication of relevant mock-ups;
  - in-pile and out-of-pile testing.
1. Introduction

The objective of the European Fusion Programme is the demonstration of the feasibility of a prototype reactor as a source of electric energy that meets the needs of society: abundant fuel supply, operational safety, environmental compatibility and economic viability.

The long term development towards a DEMO and a prototype fusion power reactor aims for materials which can withstand high neutron wall loading and fluxes under temperature and coolant pressure conditions attractive for efficient thermodynamic working cycles. In addition the materials should be compatible with high neutron fluences (10-15 MWy/m² in a power reactor) to limit the necessary replacement of in vessel components to a minimum and should be of “low-activation” type to maintain one of the most attractive features of fusion [1,2].

Three major structural materials which can fulfil the requirement for “low activation” have been considered in different designs of first wall and breeding blanket concepts [3]: ferritic-martensitic steels, vanadium alloys and SiC/SiC ceramic composites. These three materials groups are pursued in national and international materials R&D programmes in EU, Japan, USA and Russia Federation. The ferritic-martensitic steels are the furthest in development and show the fewest areas of concern. They are the reference structural material for the in vessel components of DEMO. The Vanadium bases alloys have high potential for reactor concepts based on liquid lithium as coolant. The development of self-healing, corrosion-protective and electrical insulating coatings (to mitigate the MHD effects) is the key element for the future use of such a type of structural material. SiC/SiC ceramic composite materials have potentially high pay offs in terms of low radioactivity and decay heat at short and intermediate decay times and offer high operating temperatures. The development of neutron irradiation resistance composites is a key issue, which at present is pursued by the use of quasi-stoichiometric SiC fibres with properties nearly identical to the SiC matrix.

With the objective to prepare the guidelines for the future materials development activities in Europe, particularly for the next Framework Programme (FP6) starting in 2003, a Material Assessment Meeting has been organised in Karlsruhe in June 2001. About 70 experts from 15 Associations and industry (EFET) were participating and 19 invited talks were given by distinguished speakers on well defined subjects (see annex 1).

Assuming a fusion plant lifetime of 30 years and an average neutron wall load of 2.5 MW/m² the integrated exposure is about 75 MWy/m² which corresponds in steel to 750 dpa. This value is extremely far beyond any existing experience with materials in nuclear environment and it is reasonable to suspect that in the much harder neutron spectrum of a fusion reactor with respect to fission one, the material performance should be further reduced [2]. Therefore, it seems reasonable to adopt the following strategy for the R&D programme of structural materials for in vessel components of fusion reactors:

- **short term target**: development and characterisation of a material which can withstand about 70-80 dpa to be used in DEMO;
- **intermediate target**: development of a material for a prototype reactor up to a neutron fluence of 15 MWy/m² which corresponds for most of the considered materials to about 150 dpa;
- **final target**: extension of such performance limits as far as possible in order to reduce the number of replacements to a minimum during the plant lifetime.

The EU programme for the short-term target is focussed in developing the Reduced Activation Ferritic Martensitic (RAFM) steel. This material will be used as structural material in the two EU Test Blanket Modules (TBMs) to be tested in ITER-: the Water Cooled Lithium Lead (WCLL) and the Helium Cooled Pebble Bed (HCPB). The blanket solution to be chosen for DEMO will be, with high probability, one of those developed in the frame of the DEMO-test blanket programme for ITER-.

For the intermediate target further improvements in power plant efficiency/economics and in waste management appear possible by use of low activation materials that can operate at higher temperature with respect to RAFM steels. At present in Europe R&D is mainly focussed on the Reduced Activation Ferritic with Oxide Dispersion strengthened alloys and SiC/ SiC composites. A limited effort during the 5th Framework Programme (FP5) is also devoted to investigate the potentiality of Ti and Cr alloys for fusion application.

During the next Framework Programme (FP6) an important decision must be made: the construction of the tokamak machine ITER-. Assuming a positive decision, the development programme has to be finalised with the aim to be ready for the installation of the TBMs in ITER- before the starting of the operation phase in hydrogen scheduled for the year 2015. Consequently materials and fabrication processes must be frozen not later than 2005-2006 to allow the fabrication and tests of the final prototypes of TBMs in due time.
In order to complete the qualification of materials up to the full lifetime of a DEMO reactor (70-80 dpa for the RAFM steel) in an intense neutron source with neutron spectra peaked at 14MeV is needed. The acceleration based D-Li stripping source IFMIF appears the only choice with the potential to fulfil the demanded requirements within the required time scale. In Fig. 1 the main R&D areas of activities related to materials and breeding blankets development and their inter-link with fusion devices (ITER) and necessary irradiation facilities are shown.

3. **Aim of the Material Assessment Meeting**

The material assessment meeting was held in Karlsruhe, 5-9 June 2001. The main objective was the evaluation of the attractiveness and technological critical issues of the structural materials for fusion power reactor. Functional materials for high heat flux components (limiter, divertor) and for increasing the operation temperature in breeding blankets have been also considered. Breeding materials, neutron multipliers, coating insulating materials, coolants were not included in this assessment.

The assessment was performed by comparing the point of view of material experts and design experts of the presently considered nuclear reactor components (breeding blankets and divertors). Boundary conditions dictated by safety aspects, waste and recycling and utility requirement have been also presented.

The RAFM steel class is at present the reference structural materials for DEMO nuclear components. In the Material Assessment Meeting priorities in the R&D for the design and fabrication of TBMs and IFMIF and possibility of improvement (including ODS) have been the main subjects of discussion.

With respect to the advanced materials (SiC/SiC, Cr, V, Ti, W alloys) identification of attractiveness and technical risk for successful development were analysed during the assessment in order to guide the definition of the priority in the FP6. Other aspects, which are important part of the materials development programme such as nuclear database, modelling, materials requirement for high heat flux components, irradiation devices have been considered.

During the first three sessions (agenda in annex 1) presentations from experts were given. In session 4 expert groups were formed in the following subjects: RAFM & ODS; Advanced Materials; First Wall (FW)/Blanket and Shielding; Divertor designs; armour and plasma interaction. Lists of questions to be addressed to the Expert Group have been prepared (annex
II). The last plenary session was devoted to the reporting of the expert groups and to the final discussion.

4. **Background**

4.1. **General Requirements**

At present the experience with materials in an environment representative for a fusion power plant is limited. Fission reactors provide materials data, but the radiation damage and transmutation products mix deviate considerably from that in a fusion power plant. The materials choices made for ITER are not likely to provide solutions for fusion power plants either. An example is the adoption of austenitic steel as structural material. In this case it is expected that ferritic steels with low activation compositions will provide better properties for economic operation. ITER will provide the short-term test bed for ferritic steel to be used in blankets foreseen in a power plant. ITER as test bed for the irradiation behaviour of structural materials has a limited value because of the low neutron fluence. Nevertheless the integral material behaviour in TBM (magnetic effects, influence on neutronics, activation and fusion-fission correlation on low fluence level) is of interest.

Power plant owners will opt for long lifetimes in excess of 50 years, plant availability over 75% and preferable recycling of materials. This implies low after heat and low medium-term activation for the larger heavier components in order to reduce the risk related accidents, facilitate the maintenance operation and simplify the decommissioning and waste management. High operation temperatures will provide the efficiency to cope with the competing sources together in the future energy market. Using ceramics composites and refractory alloys will enhance the potential for high efficiency of fusion power.

Near the plasma, 14 MeV neutrons control the displacement damage and production of hydrogen and helium. Over a distance of about 400 mm from the first wall, the neutron spectrum changes rapidly and the production of hydrogen and helium has decreased with an order of magnitude. This shell of about half a meter thickness around the plasma must be resistant to the typical fusion damage. At further distances the damage levels are less demanding. The design and related materials choice should reflect this damage gradient in structural materials.
4.2. **Major Structural materials**

The RAFM steels are presently the reference steel for operation in near plasma locations such as blankets. Several heats have been produced at industrial scale with low activation compositions. Complex shapes can be achieved by welding (GTA, EB or laser) and HIP processing. Radiation embrittlement should be reduced.

Oxide dispersion strengthened steel would allow a 100 °C higher operation temperature. This option is being investigated now on laboratory scale batches of steel. Toughness improvement and qualification of manufacturing processes are major issues.

Vanadium alloys have been produced at semi-industrial scale. Their main drawback is that the alloys are only sufficiently compatible with liquid lithium coolant in order to avoid embrittlement by oxygen.

Chromium alloys have attractive activation properties, but suffer from embrittlement below 300 °C. Controlling interstitial levels and developing nano-structured alloy might improve the ductility and the radiation resistance of this alloy to acceptable levels.

Conventional titanium alloys do not have a reduced activation feature. Finding the right alloy for fusion applications will take time; the materials have the potential to replace steel. Hydrogen sensitivity might be a problem if suitable coating development fails.

SiC ceramic composites also have the potential to operate at high temperature, 1200 °C, but their strength and physical properties stability under radiation have to be improved before that can be realised in practice.

4.3. **Design and Materials Choice**

Material choices will not solve all design problems. The design is complex by nature and it will have to be used to overcome material limitations. Fabrication and processing are major keys to increasing the degrees of freedom for design. Alloys lend themselves to a variety of shaping technologies such as forging, extrusion, rolling, fusion and diffusion welding, powder and solid HIP processing and dissimilar welding. With surface treatments such as coating and cladding inherent compatibility weaknesses can be adjusted.

With ceramics the use of half products is limited. In the contrary near or net shaping is possible with the integrated design of the part and the material used.

To guide the materials development of structural materials several design concepts are presently being available from a solid lithium ceramic breeder blanket concept with RAFM
steel and helium coolant at 550 C to a liquid lithium concept with a tungsten 5Re alloy operating at 1300 C.

Of course these concepts have different levels of maturity, but they have an important function in prospecting the paths to economic, reliable and safe solutions for blankets.

Though the cost of a structural material is usually a small fraction of the component, it is not negligible. It is foreseen that the cost of vanadium alloys in 2025 will be a factor 4 to 7 higher than that of ODS RAFM steel or plain RAFM steel respectively. SiC ceramic composite is estimated to cost 20 to 30 times more than the steels. Then the structural materials cost does matter.

4.4. Materials Improvements

Improvements mainly aim at increasing the material lifetime and expanding its range of application under irradiation. Different routes and materials are under investigation:

• nano-microstructures obtained from severe plastic deformation and/or ultra-fine metal powders (to reduce the low temperature embrittlement). However this might reduce the high T strength to unacceptable values.

• composites and nano-particles re-enforced materials (to expand the range of temperature application to higher values and increase the creep resistance).

• HIPed parts using powder with different compositions and powder properties, allowing gradient in properties. This could be exploited through a collaboration between designers and materials experts to improve the stress/strain fields and therefore the component lifetime.

5. Present Status R&D in Europe

5.1. RAFM-Steel Development

The selection and development of ferritic-martensitic 9-12%Cr steels with reduced activation (RAFM-steels) for nuclear fusion is based on the excellent experience with commercial 9-12%Cr steels in conventional power plants and on a promising irradiation performance under high neutron dose exposure in Fast Breeder Reactors. The possibility to achieve a reduced neutron-induced long term activation by an appropriate chemical modification of major alloying elements and a consequent limitation of harmful impurities, which worsen the long
term radioactivity properties as well as the potential for recycling were further reasons for the selection of this material group [4.].

In several research associations of the EU a series of developmental RAFM 8-10% CrWVTa steels (OPTIFER/OPTIMAX, BATMAN with addition of titanium and LA 12 alloys) and an industrial batch of an 8% CrWVTa steel (F82H mod.), provided by JAERI/NKK, were produced and tested with the result that, compared with the former conventional 9-12%CrMoVNb steels like MANET II or T91 in general similar metallurgical and mechanical properties were achieved. An essential improvement was observed for the impact and fracture toughness properties, where for the unirradiated materials the ductile-to-brittle transition temperature (DBTT) could be lowered and the upper shelf energy (USE) be further increased. Experiments under low fluence neutron irradiation (≤3dpa) confirmed this advantage versus conventional F/M steels [5,6]. This finding is of special importance, since the low temperature hardening and embrittlement is one of the major open issues for this material group, especially under the complex 14 MeV neutron irradiation with high helium and hydrogen production rates, typical for the fusion environment. The experience with these newly developed materials in the fields of welding and hipping were also promising.

These results were in principle confirmed by parallel investigations of research groups in Japan, USA and Russia [7] and it should be mentioned that in the field of RAFM steels an extensive and efficient international collaboration exists since years through the IEA Implementing Agreement on Fusion Material Development.

As a conclusion of this development, an industrial batch of a 9%CrWVTa called EUROFER 97 was specified in 1997 and produced by Böhler Austria with the presently available steel manufacturing technology in 1999. Different semi-finished products (bars, plates and tubes) with a sufficiently low content of radiological unwanted impurities like Nb Mo and Ni were delivered. The material is presently extensively tested in different European laboratories [8]. First results regarding the physical metallurgy and mechanical properties have been presented in this Material Assessment Meeting. The data show, as expected, a very similar transformation, hardening and tempering behaviour as the precursor alloys. Tensile and creep strength data are in the scatter band of formerly investigated RAFM alloys and in comparison to F82H mod. a further improved impact behaviour has been found. These findings also confirm the assumption that a limitation of the W content to 1 wt % is sufficient to maintain good mechanical properties and that a minimum of several hundred wt-ppm Ta is necessary to stabilise the fine grain size during austenitisation process. This is a prerequisite for a good
impact and fracture toughness behaviour. Whereas strain-controlled isothermal fatigue data are also improved versus F 82H mod. the first thermal-mechanical fatigue experiments show a slight reduction of lifetime.

TIG, Electron Beam (EB) and diffusion weldability of EUROFER has been demonstrated; the welding parameters have been defined. TIG and EB welds have to be post-heat-treated at about 730°C for tempering. This additional treatment has to be considered in the definition of the fabrication processes of components. As an overall conclusion one can state that this first European industrial batch of RAFM steel has fulfilled the expectations. The planned irradiation experiments in material test and fast breeder reactors will show whether or not this positive finding will also hold under higher fluence levels.

5.2. RAFM-ODS Steels

An obvious disadvantage of RAFM alloys is the limitation of creep strength to about 550°C, which is expected to be overcome by the development of oxide-dispersion strengthened variants. In Europe the general approach is to use the EUROFER 97 alloy composition as matrix and to add Y₂O₃ dispersion in the range of 0.2 to 1wt-%. The activation calculations showed that the Y₂O₃ dispersions do not worsen the radiological behaviour of EUROFER alloys and it is expected that by an improved creep rupture strength the upper temperature limit of application can be extended to 650 or even 750 °C. The present work in European laboratories is concentrated on the optimisation of the production route via powder metallurgy mixing of the constituents, mechanical alloying and hot isostatic pressing (Hipping), whereas the fabrication of semi-finished products via various forming techniques has to come at a later stage. The first results of hipped EUROFER-ODS variants are encouraging with respect to tensile, creep and fatigue data, as can be seen in the Proceedings of the Material Assessment Meeting. They indicate the expected potential in high temperature mechanical properties. Not yet satisfying are the observed impact properties, where for this production route the ductile-to-brittle transition temperature increases and the upper shelf energy decreases with increasing Y₂O₃-content. This result is in agreement with parallel observations of 8%Cr-based ODS steels developed by JAERI. A positive result is the nearly isotropic, i.e. orientation-independent fracture toughness and mechanical property behaviour. This is a strong argument to further follow the path of ferritic-martensitic steels, in which the γ→α transformation provides isotropic grain structures, a prerequisite for isotropic properties. The class of competitive ferritic steels with higher Cr contents still suffers from extremely non-isotropic
properties, though they have the potential for even higher creep strength. No information is yet available about the irradiation behaviour of the RAFM-ODS steels. However, it has been reported that commercial ferritic Fe-based ODS steels of type MA957 show low swelling and irradiation creep in the temperature range between 400 and 600°C up to 100 dpa [9]. The first attempt to develop a RAFM-ODS steel is promising. Future work has to be concentrated in the adjustment of the grain size and the chemical composition in order to retain a fully martensitic structure, to achieve a finer and homogenous dispersion of yttria and to eventually improve the corrosion resistance. A further aspect of development is to retain a fine grain size by a proper stabilisation with TaC precipitates and to reduce grain boundary impurities/precipitates introduced by the powder metallurgy production route. Further work should be directed towards the optimisation of the present production route and in long terms on the development of alternate fabrication routes for semi finished products. Preliminary attempts to join ODS to RAFM steel were not successful; different routes are investigated. Keep-in touch activities in the field of high-chromium-containing ferritic steels should be arranged in the frame of the IEA-Implementing Agreement. They might provide a reasonable alternative if the present difficulties of non-isotropic properties can be overcome.

5.3. **SiC/SiC Composite**

A significant progress has been made in the last decade to achieve SiC/SmC ceramic composite to be used as structural material for Fusion application. The progress relies mainly on the availability of almost stoechiometric fibres with higher thermal conductivity, higher thermal stability and higher expected neutron irradiation resistance. Using these fibres, composites with thermal conductivity as high as 85 W/m.K at RT have been produced. For composite manufacturing densification processes leading to β-SiC matrix are currently assessed in EU. Based on these processes and using the most advanced fibres, a specification for a EU reference SiCf-SmC material has been defined. The inherent brittleness of SiCf-SmC composite requires a specific design approach that is currently developed. A strong interaction with the manufacturer is requested as the properties of the composite can be tailored to the specific application by choosing the appropriate fibres architecture, fibres to matrix interface and densification processing route.
Under neutron irradiation, amorphisation of the SiC below 150˚C gives the lower intrinsic limit of its possible application. The upper temperature limit is due to the fibre performance. Investigation of composite stability under irradiation at 1200C gives promising results. Additional irradiation effect are the results of interaction with and between fibres, interphase and matrix, however:

- the densification and the decrease of the thermal conductivity observed on the previous fibres leading to fibres/matrix debonding has been strongly improved with advanced fibres;
- the possible use of porous SiC or multilayer SiC as interphase may lead to much better radiation resistant composites.

He effect is a major issue due its high production in operation (120 appm He/dpa in the First Wall) and its limited diffusivity in SiC, which may induce swelling and high residual stresses. Decrease of the mechanical properties has been observed after high energy implantation by cyclotrons in the range 400-1000˚C. However Fusion relevant He effect will only be assessed by means of an intense 14MeV neutron source.

SiC is compatible with Li ceramics up to 1000˚C. Its chemical reactivity with static Pb17Li at 800˚C is negligible: additional tests at higher temperature and under flowing conditions are still requested.

Reliable brazed joints have been also achieved in EU. Infiltration of the braze within the composite is now well controlled.

R&D activities on SiCf-SiC are performed in EU, Japan and US. Exchange of information is held through an IEA agreement. However a stronger integration of this R&D remains necessary to address the effort to the fundamental issues of SiCf-SiC composites.

5.4. Materials for Functional Applications

High heat flux components are being tested in large plasma experiments, but the duration and intensity in a power plant demand much more from them. ITER will provide an extended testing environment for the materials and composites used in divertors, but the power plant will become a much more demanding environment.

Special processing such as fibre reinforcement, advanced welding and hot pressing of solid parts and powders has allowed the production of parts and components impossible along conventional routes. Here the progress is considerable and contributes strongly to the potentially attractive long-term solutions for fusion power plants.
Molybdenum and tungsten alloys are contenders in the choice for armour applications. Their low temperature embrittlement and difficult manufacturing will remain a drawback for large scale application. Molybdenum is not a reduced activation material, whereas tungsten is acceptable to a certain degree. Tritium extraction and waste might become a problem for both. Manufacturing has several limitations due to their high melting point.

Tantalum is in principle an attractive material. It has high temperature strength and reduced long term activation properties; H effects could be a major issue. Complex large components have already been demonstrated in the chemical industry. Irradiation effects are not well known. With the present mining information the tantalum cost is expected to remain high. The material is not abundant distributed in the earth crust.

The present experience with this class of new materials is not yet most promising. Few refractory alloys have reduced activation properties and their low temperature toughness leaves a lot to improve. Ceramic composites have several drawbacks of which low toughness is prominent.

Advanced industrial applications may provide new material solutions for high heat flux components for a fusion power plant. Therefore the fusion plant developers, be they designers or materials experts, must continuously scout the industry and academia for promising materials application advances. This will provide the economic approach basis to develop new materials for attractive fusion applications.

6. Material qualification

6.1. Available irradiation devices

The majority of irradiation experiments for the European Fusion Technology programme is presently being performed in Material Test Reactors (MTRs) which in most cases provide rigs with a reasonable volume capacity for temperature controlled irradiations and post-irradiation tests of structural material in the temperature region relevant for the reference TBM-breeding concepts. However, in-situ test facilities like creep or fatigue testing devices are not or no longer available. The class of MTRs possesses a mixed thermal-fast neutron spectrum and can under ideal conditions accumulate up to 10 dpa (Fe) per year, which limits their use to moderate fluence levels. To a certain extent the He/dpa ratio can be adapted to fusion conditions by compositional or isotopic tailoring of materials investigated.
Fast Breeder Reactors (FBRs) are no longer available for irradiation experiments in Europe. Irradiation experiments are, therefore, presently performed in foreign FBRs like BR 60 or BN 600 in Russia. Major technical drawback is at present a limited temperature range available for experiments (between about 300 and 700 C) and a modest possibility for instrumentation and temperature control of existing irradiation rigs. Improved irradiation capsules have, therefore, to be developed. From a radiation damage point of view, FBR irradiation deviates by more than an order of magnitude from gas production rates (H, He) under fusion conditions, but a possible damage accumulation up to 30 dpa (Fe) per year makes these reactors the only realistic source to achieve DEMO relevant neutron fluence levels in the range of 100 to 150 dpa within the next decade.

For specific application, accelerator-driven devices like Van de Graafs and Cyclotrons are being used in the European Fusion Technology Programme. With them it is possible to simulate with loaded particles such as electrons, protons, He-ions and heavier elements specific aspects like swelling, He-embrittlement and other phenomena. In general these experiments provide a much better control of irradiation parameters, have a great variability to adapt damage parameters like flux, fluence, gas production rates and allow sophisticated in-situ tests like creep and fatigue experiments. A major limit of these simulation techniques is the short range of loaded particles which minimises the available volume and leads often to very inhomogeneous damage profiles. With the exception of low-energetic injections they are also quite expensive. Nevertheless these experiments are of great value for the fundamental understanding of phenomena and specially effective in combination with modelling.

An interesting facility is the SINQ device, which delivers a mixed neutron-proton field with a wide-spread spallation neutron spectrum, containing an extensive high energy neutron tail in combination with extremely high-energetic protons (570MeV). This facility will provide in its final version 5-10 dpa (Fe) per year and gas/dpa ratios exceeding those in the fusion environment.

When considering the future availability of these irradiation devices one has to reckon with a continuing closure of existing reactors and other irradiation facilities within the next years. Therefore, it is very important for the Fusion Material Community to maintain at least a few of these facilities for the next decade until an intense high neutron irradiation facility will be available.
6.2. IFMIF

None of the presently existing irradiation facilities can provide a reliable data basis for the material behaviour under fusion specific neutron irradiation of high intensity. Therefore, from the beginning a need for a powerful test bed for fusion material studies has been stated in the past at many occasions. In several investigations, initiated by the IEA-Implementing Agreement on Fusion Materials, an accelerator driven d-Li neutron source, the International Fusion Materials Irradiation Facility, IFMIF, has been selected as the most appropriate and most realistic option. In more recent Concept Studies [10,11] its feasibility has been investigated and it has been shown that it can fulfil essential users requirements [12]. For instance it can adapt for structural materials the physically based damage parameters like dpa and PKA spectra as well as the transmutation production rates of relevant elements like H and He reasonably well to the fusion environment. IFMIF can also provide—with the given test volumes in the high, medium and low flux test zones—a sufficient test capacity to perform high-sophisticated and well instrumented, partly in-situ tests in the DEMO- and PROTO-relevant temperature, flux and fluence range. The limited test volume in the high flux test zone needs, however, a consequent development of the Small Specimen Test Technology including its qualification for design and licensing purposes.

In comparison to d-t based 14 MeV proposals for an intense neutron source (e.g. the Gas Dynamic Trap mirror-type or the tight-aspect-ratio Spherical Tokamak machines) the IFMIF facility is a realistic and reliable conceptual design based on proven technology with very moderate extrapolation. Key elements of this machine development like the Li-target, powerful ion sources and specific irradiation and test modules are presently investigated in a Key Element Phase and provided that an Engineering Design Activity and a consecutive positive decision for its construction will follow soon, this test bed could be available within the next decade.

It is evident that this intense neutron source is well suitable for the screening and qualification of materials and the development of an appropriate materials database for DEMO. In the high-flux region of IFMIF—which is limited in volume capacity—the potential of reference and alternative materials for prototypic application can be tested due to achievable high annual displacement rates of up to 40-50dpa. This exploratory work is very important since it can guide the final selection of most promising alternatives for commercial fusion reactors. Finally the calibration of results from simulation irradiations in presently used fission reactors and accelerators is an additional task for this facility.
6.3. Nuclear data

Specific data files and codes have been developed to cover Fusion applications over the energy range up to 20MeV. Additional data over in the range 20 to 55MeV should be requested to cover the IFMIF domain.

The data have been collected in the European Fusion File (EFF) and in the European Activation file (EAF) that enable accurate neutronics and activation calculations to be carried out. In addition the SAFEPAQ-II windows application has been developed to provide an interactive evaluation and processing tool for the nuclear data.

For the major elements considered in the fusion device internal components the nuclear data uncertainties have been significantly reduced and are typically no longer critical. It is judged that only limited improvement of the dose rate can be achieved by isotopic tailoring at reasonable cost.

Impurities play a major role on the dose rate at long times (N and Ni for Eurofer): some additional measurements are needed to investigate the effects of impurities with higher accuracy.

Benchmarking of the data for complex geometries is needed: use of existing steels and SiC blocks will be possible for further benchmark measurements with 14 MeV neutrons.

6.4. Modelling

Modelling of materials subjected to neutron irradiation has been developed to assess the effect of the radiation damage on their microstructure, mechanical properties and to extrapolate the change to higher dose and neutron energy.

To explain the irradiation damage, modelling from the atomic scale is necessary: this requests powerful computer and codes able to simulate events within a very short time domain and on a reasonable volume of matter. It does not only make predictions or analyses of results at macroscopic scale but it also allows the guidance of R&D in effective directions and cut off unattractive paths.

This approach has been helped by the CPU development (x 10^6 during the last 30 years). The trend of increasingly fast analyses tools for the next decade will promote the importance and usefulness of modelling even more.

At present the interaction of dislocations and clusters and the phase stability with binary alloys can be modelled well, explaining some observed mechanical behaviour. Several phenomena modelled at atomic-scale and meso-scale like the evolution of the irradiation defects and the
effect of He in BCC pure metal have been confirmed experimentally by neutron scattering, TEM and positron annihilation.

In the near future, multi-scale modelling development to cope with real materials and effort on non-metals is needed.

Over 20-30 keV the cascades will stabilise, thus displacement damage due to 14 MeV neutrons can be extrapolated from Fission.

However the introduction of He and H at the same time scale and the nucleation are more difficult to model. In addition, the Fission reactors can only provide a part of the answer.

This is one area of immediate and major interest for the Fusion community and the main effort has to be devoted on it.

In general the need for more theory support in the fusion materials development is stressed:

- It could be most helpful in understanding the mechanisms and provide the boundary conditions for the phenomena.
- It would help to settle the validity of extrapolating the ranges of experimental data and allow to limit the costly experimental work in the irradiation devices to the essentials.

With the help of the Fission community, which has extensively developed, modelling of radiation damage phenomena, appropriate Fission-Fusion correlation in combination with sophisticated irradiation experiments should be developed before IFMIF is available.

7. Design Requirements

7.1. Introduction

Breeding blankets have been largely studied in the last decade. Present EU strategy is to test in ITER some blanket mock-ups of the two EU reference blanket conceptual designs performed for DEMO. These mock-ups should be equipped with all DEMO-relevant technology for which a significant R&D program is in progress.

In recent years, within EU, an evaluation has started in order to understand the potential of these medium-term concepts when used in a Fusion Power Reactor (FPR).

Taking into account the additional requirement for a FPR, one can qualitatively define the “attractiveness” of a given concept in term of availability, safety, waste management, efficiency, cost of electricity, duality of the development with other research field, and identify the corresponding risk of development.
Therefore, the evaluation has to include, together with the two reference medium-term concepts associated with relatively low development risk, some more advanced concepts with an overall improved attractiveness but increased development risk.

This chapter presents the most relevant breeding blankets for FPR which has to be considered for a possible exhaustive evaluation. It must be stressed that requirements on structural material strongly depend on assumed reactor specifications, in particular on the assumed surface heat flux and neutron wall loading. Therefore the coherence and the importance of the design requirements can only be ensured by a complete reactor study.

A limited number of combinations of coolant/breeder/structural materials present a real interest for FPR breeding blankets. The considered concepts, listed in the order of increased attractiveness but increased development risk, are the following:

- the two EU reference DEMO concepts, both using RAFM steel as structural material, e.g., the Water-Cooled Lithium-Lead (WCCL) blanket and the He-Cooled Pebble-Bed (HCPB) blanket;
- the Dual Coolant (DC) blanket which use RAFM steel as structural material and Pb-17Li and He as coolants (with SiC/SiC flow channel inserts acting as electrical insulators);
- the self-cooled Lithium blanket using V-alloy as structural material and need an electrical insulating coatings;
- three blankets using SiC/SiC as structural material: two self-cooled Pb-17Li blanket (TAURO and ARIES-AT) and an He-cooled ceramic/Be concept proposed in Japan (DREAM);
- a self-cooled Lithium blanket, cooling the W-structures by Li-evaporation (EVOLVE).

The main characteristics of the corresponding concepts are given in Table 1. Some corresponding details are summarized in chapter 7.2.

Typical assumptions taken for design activities of FPR blankets are: i) no disruption; ii) continuous operation, iii) neutron fluence of about 12.5 Mwa/m² which means about 150 (Fe) dpa in FW (70 in DEMO); iv) no FW protection.

7.2. Breeding Blanket Concepts

7.2.1. Water-cooled Lithium Lead (WCCL) Blanket [13,14]

The blanket segments are essentially formed by steel boxes, which confine the slowly flowing Pb-17Li. Fig. 2 shows a view of a DEMO outboard segment. Grids of stiffener plates ensure
the resistance of the segments against accidental pressurization by the coolant, which is pressurized water. Poloidal bundles of hairpin shaped Double-Wall Tubes (DWT) remove the heat from the Pb-17Li pool. Both tube walls individually withstand the coolant pressure. The joint between the tube walls ensures thermal contact but prohibits crack propagation. The segment-box (SB) walls are directly cooled with an independent circuit ($T_{in}/T_{out} = 265/325^\circ C$). The SB cooling tubes are joined with the box structure with a technique similar to the one used for the DWTs thus achieving double confinement of the coolant. The feeding pipes for all segments enter from the top of the vacuum vessel. A corrugated geometry on the first wall minimizes thermal stress. The SB coolant is collected in the rear of the segment in poloidally oriented headers. Tritium permeation barriers (TPBs) are currently applied to surfaces in contact with Pb-17Li to minimize the effort for coolant detritiation.

Fabrication of DWTs and TPBs, and, in particular, the compatibility of the corresponding fabrication process with RAFM steel are the specific WCLL requirements on RAFM steel specifications.

The data presented in Table 1 correspond to the case where the surface heat flux is maximized. The Pb-17Li/RAFM interface temperature is above the maximum acceptable value of 480°C because the use of anti-corrosion barriers is assumed. Without this protection, the maximum acceptable heat flux would decrease to 0.8 MW/m$^2$. In all cases, the benefit of using ODS for WCLL blanket is marginal.

### 7.2.2. Helium Cooled Pebble Bed (HCPB) Blanket [15]

The Helium Cooled Pebble Bed (HCPB) DEMO blanket under development in the frame of the European Union Blanket Development Programme has been selected as a typical example of a ceramic breeder blanket. It is based on the use of ternary lithium-ceramics as breeder material, beryllium as neutron multiplier, and the low activation ferritic steel EUROFER as structural material. Breeder material and multiplier are arranged as pebble beds between flat cooling plates and cooled with helium. The basic layout of this concept is shown in Fig. 3

Typical parameters of HCPB blankets are presented in Table 1. These values correspond to the case where ODS steel is used as structural material and the surface heat flux is maximized. Use of RAFM steel would reduce the coolant exit temperature by about 50 K and the allowable power density (neutron wall load and surface heat flux) by about 20%.
The requirements on structural materials are:

- Range of operating temperatures (C): 250 (300)*- 550 (650)*.
- Interface temperatures:
  - between structure and breeder (C): up to 550 (650)*
  - between structure and Beryllium (C): up to 550 (650)*.
- Yield strength, ultimate strength, creep strength: at least as high as EUROFER 97.
- Creep elongation < 1%.
- Irradiation induced swelling < 2%.
- Ductility of new material: sufficient to allow bending during fabrication of FW box.
- Ductility at end of life: sufficient to survive coolant pressure pulses, vibrations, and handling operations.
- Fabrication of large modules (~8m×2m×1m) possible with milling, bending, diffusion welding, TIG welding.
- Fracture toughness of welds: > 70 % of base material.
- Achievable neutron fluence at FW: >10 MWa/m², resulting in 100dpa (*) values for ODS steel.

### 7.2.3. Dual Coolant (DC) Blanket [16]

An interesting variant of a self-cooled Pb-17Li blanket is the DC blanket concept, which is characterised by a helium-cooled first wall and a self-cooled Pb-17Li breeding zone. There are flow channel inserts made of SiC/\text{SiC} composite arranged in the large liquid metal ducts serving as electrical insulator and, at the same time, as thermal insulator between the helium-cooled steel walls and the flowing Pb-17Li. The liquid metal is flowing in two passes through the large poloidal ducts and is heated up there volumetrically to a temperature about 200 C higher than the maximum interface temperature (steel/LM), and about 100 C higher than the maximum steel temperature. In this design, shown in Fig. 4, the blanket structure is made of a low activation ferritic-martensitic steel, and the SiC/\text{SiC} flow channel inserts have no mechanical loads, require low thermal and electrical conductivity and are relatively easy to fabricate.

The performance of this concept is limited by the maximum allowable FW temperature and by the compatibility of the structural material with Pb-17Li, limiting the allowable interface temperature to about 500 C. Use of ODS-steels with higher strength-based temperature limit would increase the load capabilities but welding requirements would make the fabrication
more difficult. An interesting compromise is therefore a version where the entire structure is made of ferritic steel, but the FW is plated with a few mm thick layer of ODS steel. This restricts the use of ODS-steel to zones where structural temperatures > 550 C are desired. At all other places the temperature is limited to values < 550 C for compatibility reasons. The main parameters for this concept are summarized in Table 1. From these parameters it can be concluded that the DC blanket concept combines in an interesting way high performance with a limited extrapolation of the required technology since it is based on ferritic steel as structural material and SiC has no structural functions. Degradation of thermal conductivity of SiC-composite by neutron irradiation is not a problem since this material serves here as a thermal insulator.

The main requirements on the structural material are identical to the ones listed under 7.2.2 for the HCPB blankets with the exception of the following temperatures:

- Maximum temperature at the steel/ Pb-17 Li interface 450-500 C
- Maximum temperature at the SiC/ Pb-17 Li interface 650-700 C

The requirements on SiC flow channel inserts are:

- Longitudinal slots in the inserts allow for pressure equalisation between flowing liquid metal and the stagnant Pb-17 Li in the gap, avoiding primary stresses in the flow channel inserts;
- No need for high thermal conductivity of SiC since the inserts serve as thermal insulation. (assumed conductivity 2-4 W/mK);
- Required electrical resistivity (> 10^3 Ω m);
- SiC temperature and steel/Pb-17Li interface temperature as higher as possible but within the above limits;
- No major seal problem since the same liquid metal at both sides;
- Either porous SiC or the most simple 2D-woven composite can be used.

7.2.4. Self-cooled Lithium Blanket with Vanadium Alloy as Structural Material [17]

Pure lithium is an attractive liquid metal breeder especially if it is used as breeder and coolant („self-cooled blanket“). Its advantages are the high thermal conductivity and heat capacity, and compared to Pb-17Li, the better compatibility with steel and vanadium alloys. The main disadvantage of lithium is the very high chemical reactivity with water, air, and - to a lesser degree - with other gases which is an important safety concern and rules out water-cooling.
Lithium works best with vanadium alloys as structural material because the high affinity of lithium to oxygen, hydrogen, nitrogen and carbon keeps these impurities away from the vanadium alloy. This material combination allows interface temperatures up to about 700°C. Self-cooled blankets are in principle the most simple design because the heat exchanger is outside the irradiation environment. However, their need for an electrical insulator between the structure and the flowing liquid metal in order to reduce the impact of the magnetic field on flow and heat transfer is still a feasibility issue. Candidate materials for these insulating coatings are CaO and AlN. The idea is to generate and to renew the coating in situ by adding small amounts of the required elements to the flowing lithium.

As a typical example the self-cooled lithium/vanadium blanket proposed in the ARIES-RS power plant study is shown in Fig. 5. The main parameters of the blanket are listed in Tab. 1. The requirements on the structural material are:

- Range of operating temperatures: 300 - 650°C.
- Interface temperatures between structure and lithium: 300 - 650°C.
- Coating must be self-healing since not more than one electrically conductive crack is allowed in a channel.
- Achievable neutron fluence at FW: >10 MWa/m², resulting in 120 dpa.

7.2.5. Blankets with SiC/SiC structures [18]

The use of silicon carbide composites (SiC/SiC), because of their very low short- and medium-term activation and afterheat, appears as one of the most viable solution to obtain high safety standards in future fusion power reactors. Moreover, they offer the possibility of working at very high temperatures (~1200°C) which leaves the potential of very high energy conversion efficiency (50% or more). SiCf/SiC composite has been used as structural material for the first wall and blanket in several conceptual design studies. The most recent proposals are TAURO in the European Union, ARIES-AT in the United States, and DREAM in Japan. The first two designs are Pb-17Li self-cooled blankets, while DREAM is cooled by 10 MPa Helium.

Both TAURO and ARIES-AT blankets are essentially formed by a SiC/SiC box with indirectly-cooled FW that acts as container for the Pb-17Li which has the simultaneous functions of coolant, tritium breeder, neutron multiplier and, finally, tritium carrier. Because
of the relatively low SiCf/SiC electrical conductivity, high Pb-17Li velocity is allowed without needing large pressures (< 1.5 MPa).

TAURO blanket is characterized by 2m-high single modules, which are reinforced by SiCf-SiC stiffeners (see Fig. 6).

The ARIES-AT concept is characterized by a coaxial Pb-17Li flow, which occur in two 8m-high boxes inserted one into the other (see Fig. 7).

Comparable SiC/SiC database has been used for both designs, in particular a thermal conductivity as high as 20 W/mK at 1000°C has been assumed both in plane and through the thickness.

ARIES-AT is a more recent design which, starting from the TAURO design proposal, it takes advantage from the selected sector maintenance scheme of ARIES. This maintenance scheme can accommodate a design based on a coolant co-axial flow, which permits to have a coolant outlet temperature larger than the maximum SiC/SiC temperature, thus maximizing the thermal efficiency of the concept. It has been designed in order to maximize the coolant outlet temperature.

The DREAM blanket is characterized by small modules (0.5 m of height), each divided in 3 zones, FW, breeding zone and shield (see Fig. 8). Neutron multiplier material (Be), tritium breeding material (Li₂O or other lithium ceramics) and shielding material (SiC) are packed in the module as small size pebbles of diameter 1 mm for Be and Li₂O, and 10 mm for SiC. The He coolant path includes a flow through the pebble beds and a porous partition wall.

Main design issues (besides the material issues) for this kind of blanket are the development of an appropriate brazing technique, the development of appropriate models and design rules, and define methods for improving hermeticity especially against high pressure He.

7.2.6. Self-cooled Li-evaporation blanket (EVOLVE) with W-structures [19]

This blanket is based on tungsten as structural material and heat removal by the large heat of evaporation of lithium. Figure 9 shows schematically the EVOLVE first wall tubes and the lithium filled trays of the breeding zone. A wick structure at the first wall is employed to keep the surface wetted, and an overflow system keeps the lithium-level in the trays constant. Due to the excellent heat transfer in boiling liquid metal and the large thermal conductivity of tungsten, the entire structure is kept at nearly uniform temperature.
Typical Parameters of EVOLVE-Blanket are:

- Maximum neutron wall load: > 10 MW/m$^2$.
- Maximum surface heat load: > 2 MW/m$^2$.
- Lithium inlet temperature: 1100°C.
- Lithium outlet temperature: 1200°C.
- Lithium pressure: 0.03 MPa.
- Maximum tungsten temperature: 1400°C.
- Minimum tungsten temperature: 1100°C.
- Maximum interface temperature W/Li: 1300°C.

The requirements on the structural material are:

- Range of operating temperatures: 1100 to 1400°C.
- Temperatures at the W/Li interfaces: 1100 to 1300°C.
- Maximum stress in structural material: < 100 MPa.
- Achievable neutron fluence at FW: >20 MWa/m$^2$, resulting in 70 dpa.

### 7.2.7. Main conclusions

- There is a large variety of blanket concepts thinkable, employing RAFM steels, ODS steels, Vanadium alloys, SiC-composites, or tungsten alloys as structural material.
- In general, increasing attractiveness of a concept goes together with larger extrapolation of material properties and technology, including increased risk that the development could fail.
- The three blanket concepts studied in EU and based on RAFM steel as structural material are: water-cooled lead-lithium blanket, helium-cooled ceramic breeder blanket, self-cooled lead-lithium blanket with He-cooled steel structure (Dual Coolant blanket).
- These three concepts have modest requirements on the structural material. Open issues are mainly related to the behaviour of the steel in typical fusion neutron environment.
- The best use of the improved high temperature properties of ODS-steel is made with the Dual Coolant concept. Plating the FW with a layer of ODS allows to increase power density and efficiency without going into fabrication difficulties (especially welding).
- Using vanadium alloys is reasonable only if lithium metal is used as breeder and coolant. Such a self-cooled blanket, however, requires electrically insulating coatings to avoid MHD problems. The feasibility of such a coating remains still to be seen.
SiC-composites are interesting candidate structural materials, allowing power plant concepts with up to 60% efficiency in the power conversion system. However, the fabrication of such a structure, the achievement of sufficiently high thermal conductivity, and the behaviour under neutron irradiation, are still open issues.

The issue of hermeticity to high pressure Helium in the DREAM blanket design is considered extremely critical because, due to the low leakage tolerance of this system for safety reasons, the possibility of developing an acceptable coating appears doubtful and anyway much more risky than those required in TAURO and ARIES-AT. It is therefore suggested to concentrate the design and R&D effort on self-cooled Pb-17Li.

The most “exotic” of the concepts described here is the EVOLVE concept, employing tungsten as structural material and boiling lithium as coolant. This concept promises exceptional high performance compared with simple geometry and low mechanical stresses. However, this concept is a typical example of the fact, that attractive features are mostly connected with a high development risk, and the feasibility of fabrication and irradiation behaviour of a tungsten structure is completely unproven.

7.3. Divertor concepts

7.3.1. Introduction and Assumptions

Compared to blankets, the divertor is submitted to much higher surface heat flux (5-10 MW/m²) but lower neutron wall loading (about 1/3 of maximum value at first wall) and it can be envisaged a more frequent replacement. The latter clearly depends on the adopted maintenance scheme which can vary from independent maintenance such as in ITER (allowing replacement frequency at least twice as larger as for blanket) to common maintenance such as in the case of a whole sector replacement proposed in ARIES studies [20].

The lifetime is mainly dictated by erosion and neutron fluence. The consequence is the need of a sacrificial layer of armour material (e.g., 3 a minimum of 3 mm for W) and the need of testing the impact of irradiation effects up to a fluence level ranging between 1/6 and 1/3 of that to be tested for blankets (i.e., about 15 (Fe) dpa for DEMO and up to 50 (Fe) dpa for FPR). In all available studies for FPR disruptions and ELMs are excluded and continuous operations are assumed.
Very limited and preliminary studies of divertor designs are discussed in the literature beyond those required for ITER. In EU, a very preliminary evaluation has been performed in the last few years with the main objectives of checking the maximum surface heat capability of the most promising concepts. All concepts assume W as armour material and are characterized by the selected coolant which dictates the choice of structural material and its operating temperature. In all cases, the corresponding thermo-mechanical analyses refer the most critical part of the vertical target and do not investigate full system and integration issues.

### 7.3.2. Available concepts in literature

With reference to chapter 8.4.1, it can be said that “short term” concepts could use either low T water (200 - 250°C) with Cu-alloys or Cu-composite structures [20], while “medium term” concepts could use pressurized water at PWR conditions with RAFM (ODS) steel. More advanced “long term” concepts could use either He-coolant with W-structures or Pb-17Li coolant with SiC/SiC structures. Additional proposals are W-structures cooled by liquid metal evaporation (Li, Na, or NaK) or liquid surface concepts based on the use of Li, Sn, or Ga. The latter has been proposed in US studies and has the potential of coping with very high heat flux (>10MW/m²) but these concepts have been disregarded within EU because of expected severe safety issues associated with the use of liquid metals highly reactive with air.

Table 2 gives a sketch and the main characteristics of the five concepts recently evaluated within EU in the framework of the reactor study activities [21,22]. They correspond to the following materials combinations:

- high pressure high T (PWR conditions) water-coolant, with RAFM (ODS) steel as structural material and W in monoblock geometry as armour; maximum acceptable heat load is about 7 MW/m²;

- He-cooled W-structures assuming conventional heat exchange laws monoblock geometry is assumed; maximum heat load is about 5 MW/m²; the W-operating temperatures are probably too low;

- a second He-cooled W-structures in a similar monoblock geometry, but assuming enhanced heat exchange permitted by the use of porous walls; maximum acceptable heat load is about 5.5 MW/m² (similar to the previous concept) but an higher W operating temperature is allowed; more recently, an He-cooled slot concept made of W able to cope with heat flux as high as 10 MW/m² has been proposed;
• Pb-17Li coolant flowing in SiC/SiC structures to reduce MHD effects and W-tiles as armour [23]; originally this concept was proposed with W-structure and SiC/SiC flow-channel inserts as electrical insulators but because of similar performances, the simpler SiC/SiC design is preferred; maximum acceptable heat load is about 5 MW/m²;

• LM (Li, Na, NaK) cooling based on evaporation principles associated with W structures and armour (monoblock geometry); maximum acceptable heat load is about 5 MW/m².

Moreover, the following two other concepts have to be considered:

• Pb-17Li coolant flowing in combined SiC-SiC/W structures assuming high velocity toroidal flow which should considerably reduce the MHD pressure losses (not yet fully evaluated in the previous SiC/SiC concept (Proposed for ARIES-AT [20]; also in this case, maximum acceptable heat load is about 5 MW/m²);

• Low T water cooling (200°C-250°C with W armour and Cu-based structural materials (CuCrZr or Cu-composites); this relatively short-term concepts has been derived from ITER design and associated R&D results [24]; it has been recently shown that these types of concepts can allow up to 15 MW/m²; if the presence of a steel liner is required to improve lifetime (issue of electrolytic erosion) the maximum acceptable heat flux could decrease of about 20%. It must be stressed that the effects of neutron irradiation of Cu-alloys have been completely neglected.

7.3.3. Conclusions

The used temperature-window for materials are not always in coherence with the suggestions recently given by materials experts and recalled in chapter 8.4. Fabrication processes are in most cases very far from present day technology. Irradiations effects on materials properties have been neglected. Despite these favourable assumptions, it can be seen that the maximum surface heat capability of most concepts is relatively limited, of the order of about 5 MW/m² for advanced concepts to about 10 MW/m² for short term concepts. However, if extremely favourable assumptions about W properties are made, He-cooled W-structure concepts could theoretically allow up to 10 MW/m². On the other end, if n-irradiation and corrosion aspects are neglected, short term concepts based on the use of low T water and CuCrZr-structure could withstand up to about 15 MW/m².

Anyway it is clear that these limits are real technological limits for divertor targets and that future physics work should address with high priority this issue (e.g., methods to decrease the heat loads).
7.4. Safety, environment aspects, waste management [25, 26]

Requirements on materials from safety point of view:

- Allow a safe and reliable design, which means to have a material allowing appropriate modeling with well known thermal and mechanical properties, with agreed design codes, standards and rules, and with predictable ageing and irradiation effects. These requirements imply significant R&D works especially for the most advanced materials.
- Have a good compatibility with coolant and compounds without high energy chemical reactions with air (and water when present in the plant).
- Have a low decay heat and low short term activation doses for allowing an easy maintenance and cooling. Typical targets are: individual doses <5 mSv/yr, collective doses < 0.7 person Sv/yr.

As a consequence, from safety point of view, the R&D program at some stages (may be in the far future for very advanced materials) should include:

- Definition of design margin (corresponding to uncertainties of properties and loadings).
- Qualification of properties and joining technique in representatives conditions (including irradiation).
- Control of properties by sampling and surveillance program.
- Definition of inspection in service program.
- Analyses of incidental and accidental situations and definition of prevention, protection and mitigation measures target: design should be compatible with all “in situ” available energies).
- Development of inspection, cutting and joining techniques by remote handling;
- Control of impurities in materials.
- Control of activation products.

Objectives from waste management point of view are:

- Minimize volume of radioactive waste through clearance and recycling. For most materials, both clearance and recycling depends on impurity levels. Attention must be paid to all the material used in the reactor and not only FW and BB (e.g., shield, VV). Recycling may not be applicable to SiC/SiC and ODS steel. Development of detritiation methods is required.
- Minimize individual doses from repository; this may require similar actions as for clearance because impurities are likely to dominate very long term activation. Doses need
to be estimated from an appropriate fusion repository (and not from present fission repository for which the requirements are different).

- Minimize long-term global doses; specific requirement to minimize C14 production. For steel this means to minimize N contents.
- Re-use scarce materials; Be and Ta are the most critical material from this point of view.

It is clear that even the best material has some weaknesses in at least one of the above items. Only a complete strategy a reactor plant level could help to reach the best compromise.

8. Conclusions of the expert groups

8.1. RAFM-Steels

The expert group discussed the status of development and necessary future activities along the lines of general requirements for Test Blanket Modules for DEMO, material optimisation and other items.

Material qualification for TBMs.

- The lowest achievable temperature is limited by the development of hardening and the embrittlement under irradiation below 350 C. Therefore, irradiation experiments which cover the temperature range between 250 and 400 C and give insight into the complex influence of dpa, He and hydrogen on embrittlement and toughness degradation are of utmost importance. Planned experiments in MTRs in this temperature range up to 15dpa and in FBRs at the lowest possible temperature around 330 C up to 30 and 70 dpa respectively have the highest priority.
- The group considers 250 C as the low temperature limit. Considering, however, the present uncertainties in combination with high He and H transmutation rates under 14 MeV neutron irradiation and the unresolved question of a possible saturation of embrittlement under high fluence irradiation, it recommends at present a minimum temperature of about 300 C for the design.
- For the WCLL concept the upper temperature of EUROFER in liquid Pb-Li is estimated to be around 500 C. To achieve 550 C as upper limit an anti-corrosion coating with Fe/Al-Al₂O₃ seems indispensable.
- For both TBMs (HCPB and WCLL) the upper temperature limit for the given requirements of creep rupture strength of 100 MPa at 20 000 hs is at around 550 C. Though no specific data on the creep strength of highly irradiated EUROFER are presently
available, a remarkable reduction of this property is not expected from experience with this material group in FBR technology.

- For TBM testing in ITER the problem of creep-fatigue has to be analysed by designers and-if necessary- a new type of experiment with a constant primary load and an additional fluctuating secondary stress component has to be initiated. Isothermal fatigue testing has to be extended to 550 C.
- For near term application the use of EUROFER-ODS is foreseen as plasma facing part of the integrated first wall with no contact to cooling or breeding media. In this case an upper temperature limit of 650 C seems to be possible, based on present preliminary creep data.

Material development and qualification.

- A new EUROFER 97/2 batch with identical specification of alloying elements and further reduced impurities has to be ordered to check the reproducibility of steel making, improve the production of semi-finished components and provide additional material for further technology tests.
- The specification of a new heat EUROFER 200X should be foreseen only after results from running irradiation experiments (15 and 30 dpa respectively.) are available. No detailed recommendations regarding compositional changes were given. The total elimination of nitrogen, as a possible source for radioactive carbon seems to be artificial, but could be discussed with steel makers.
- Regarding the further development of EUROFER-ODS a screening with laboratory batches, in which variations of the carbon-, titanium- and chromium contents are necessary to retain a martensitic structure and to achieve an optimised size distribution of the yttria dispersions should be foreseen. Also, a broadening and improvement of fabrication processes for the production of components by powder metallurgy, mechanical alloying, hipping, extrusion techniques, welding and optimisation of final thermomechanical treatments are of great importance.
- A general drawback of present ODS development is that the basis for the powder metallurgy development is very small in the European associations. Therefore, the collaboration with universities, other research institutes and commercial companies who have this expertise should be intensified.

Materials data evaluation and property design equations.

- It is of greatest importance to provide the designers of the TBMs with assured and evaluated data in due time. This means that the collection, evaluation and the input of all
laboratory data for each task into a common data bank is within the responsibility of each individual laboratory, whereas the development of adequate property design equations is a common duty for all laboratories. The actual schedule for the delivery of data has to be periodically re-discussed with the designers.

**Other topics.**

- The development of materials with reduced activity and optimised radiological properties needs a steady state effort for improvement of the nuclear data libraries to reduce the uncertainty of activation calculations and to allow better sensitivity studies. This includes data re-evaluations of important elements and nuclear reactions and in critical cases additional benchmark experiments.

- A high priority topic is the support of the nuclear data group for the development of the International Fusion Material Irradiation Facility-IFMIF. The improvement of the Li (d,xn) source term and the necessary nuclear data development for neutron energies above 20 MeV for neutron transport calculations and the characterisation of the typical damage parameters in this irradiation facility is of great relevance. The integration of this work into the Fusion Technology Programme and a close collaboration with other projects, which use very high-energetic neutrons (e.g. European Spallation Source and Accelerator-Driven Systems for Incineration of Long term Fission Products) is strongly recommended.

- For the modelling activities two areas are of immediate interest for the structural materials community: i) The effect of superimposed He-, H- and dpa effects on the plastic flow and fracture properties of Fe-based ferritic-martensitic steels and ii) Small specimen size optimisation based on FEM-calculations, micro-toughness modelling and qualification of the Small Specimen Test Technology towards accepted standards. In general the initiation of modelling efforts for the different areas of radiation damage phenomena and the support for appropriate fission-fusion correlation in combination with sophisticated irradiation experiments is highly appreciated. A new established IEA-Working Group within the Implementing Agreement for the Development of Fusion Materials could be helpful for common arrangements.

### 8.2 Advanced Materials

**Development Paths.**

The potential of SiC ceramic composites and chromium-, vanadium-, titanium- and tungsten alloys was discussed for application in advanced fusion power plant components such as DC
and EVOLVE type blankets and divertors. The minimum development stage needed to consider the application potential in detail should be a well developed materials engineering database. These materials have not reached such a stage. The judgement on their value for fusion power plants is therefore speculative.

In general it was observed that the industry will not invest in the development of such materials for fusion applications, unless there are short term benefits on conventional applications. Therefore synergy with other advanced users in aerospace and in specialised chemical plant applications could be rewarding. In the earlier scientific stage of the development university support is crucial. Personal ties are here most effective in continuing investigations useful for fusion components.

**Applications.**

Besides the choice of materials it was also expressed that the advanced components should be designed and analysed for the possibility to exploit geometrical materials related advantages. The types of solutions that belong to this domain are the utilisation of:

- Fibre reinforcement in line with the major stress pattern.
- Honeycomb stiffness where necessary
- Graded properties by HIP processing of powder with composition and size gradients.
- Multiplexed layered structures, allowing fast out-gassing of H and He in the material.

The expert group on advanced materials rated the five major contenders on 14 major screening criteria, see Table 3. The early stage of development did not always allow a fundamental judgement. SiC ceramic composites, vanadium and tungsten alloys are presently entering the fusion materials engineering stage, but will have to go a long way to satisfy the whole range of engineering developments. Titanium and chromium alloys for fusion application are in the scientific stage and much remains to be established, before stable judgements can be issued.

**Conclusions.**

For the SiC ceramic composite the first application is expected to be in the DC blanket together with reduced activation steel for structural support. Limited development is needed in such a case for high temperature blanket, advanced SiC ceramic composites are envisaged. The SiC ceramic composite development is thought to be of a high risk nature, but most rewarding when successful.

Chromium and tungsten might be applied in the second blanket generation optimised for high temperature operation such as EVOLVE. Their properties might enhance application in high
heat flux components, but the divertor design for example will have to adapt to their limitations. Geometric materials related advantages will be a necessity.

Titanium alloys could replace steel, with a minor temperature advantage and an activation penalty. Low activation elements should replace the present day alloying elements in titanium. Hydrogen embrittlement is another threat for the use of titanium alloys. Without solving these two problems the titanium alloys will not be applied in fusion power plants. Vanadium is considered to be strongly tied to lithium as a coolant for its successful application in fusion power plants. This might be an option outside the EU. In the EU lithium is not considered an attractive coolant.

8.3 First Wall, Blanket, and Shielding

8.3.1 General discussion

- A blanket development strategy based on the use of DEMO as test bed for blankets enabling a transition from near term solutions already tested in ITER to more attractive concepts for Power Plant has been discussed. It was clear that the first set of blankets in DEMO is based on RAFM-steel as structural material which must be irradiated at least up to the fluence anticipated for DEMO (70dpa) before the DEMO-blankets are designed and build.

Table 4 shows the combination of different blanket concepts with candidate structural materials.

- RAFM-steels and especially their ODS versions are interesting candidate materials not only for DEMO but also for commercial power plants. ODS-steels have some potential for higher performance in the HCPB-concept and especially in the DC-concept. Considering the time scale and the limited fundings anticipated for material development, it is obvious that the further qualification of EUROFER should have the highest priority. A substantial part of the fundings should be used for the development and qualification of the ODS-version for use in selected FW/blanket zones.

- As a part of the long term programme, at least one additional structural material for advanced concepts should be evaluated and critical points of this material addressed by suitable experiments. The discussion on these points led to the following suggestions:
- No need for the development of Vanadium alloys in Europe, since these alloys work only with lithium as breeder/coolant, a concept developed in the USA, and such a concept does not promise better performance than the DC-concept.

- As long as there is no idea available to make an interesting conceptual design of FW/blanket based on the use of Cr- or Ti-alloys, it is not recommended to spend substantial fundings on these alloys.

- W and Mo may be in the very long run interesting materials for FW/blankets and especially for divertors. Their development and qualification would, however, require a substantial increase in the material budget and is therefore not recommended for the near future.

- The group recommends to concentrate the work on alternative materials on the further development of SiC-composites, since concepts based on this structural material have the potential for exceptionally high thermal efficiencies and this material can serve already in the DC-concept, a step between near term solutions and really advanced concepts, in a less ambitious form.

### 8.3.2. Discussions to specific points

**Requirements on EUROFER.**

- The structural material should allow for a FW-fluence of at least 10 MWa/m².

- Mechanical and thermal properties measured for EUROFER-97 are sufficient for reasonable FW/blanket designs. Lower temperature limit of 280 C in fusion spectrum needs to be confirmed for WCLL-concept.

- Material transport of activated elements, especially in water-loops, needs to be investigated.

- H-embrittlement under irradiation should be investigated.

**Use of different materials in different zones.**

- The use of EUROFER should be restricted to zones where high temperature strength and low irradiation damage is required.

- Optimum shielding materials, compromising the requirements low activation, low afterheat, and low cost should be selected with neutronics calculations in co-operation of material people and designers. One possible solution are balls of any suitable material in water-filled canisters (40% water is close to optimum).
• Different materials are used outside the blankets for HX, pipings, and so on. Transition pieces are needed.

**Particular issues of SiC-composites.**

• More work is necessary to develop and qualify suitable codes for stress/strain calculations in such an anisotropic and brittle material. This requires strong interaction between modelling, fabrication, and testing.

• The function of SiC-composites in blankets requires hermetic sealing of the surfaces in contact with coolant, breeder, and multiplier. Therefore the compatibility of the sealing layer with these materials has to be investigated up to the maximum temperature anticipated.

• A crucial point is the integrity of such a layer over the entire lifetime. Therefore the impact of cracks on hermeticity and MHD-issues has to be investigated.

### 8.4. Divertor design and materials

#### 8.4.1. Need of a HHFC development strategy

The divertor target has been recognized to be a very critical component for tokamak-based reactors. Quite successful R&D has been performed for present-day machines and for ITER. Concepts for these machines can withstand very high heat loads (up to 20 MW/m2, which correspond to the values dictated by the today knowledge of divertor physics and plasma control) and can be manufactured with sufficient reliability.

However, these concepts are not relevant for a Fusion Power Reactor (FPR) because of the associated additional requirements such as high efficiency, high reliability & availability, large lifetime associated with high neutron fluence. As described in section 7.3, the concepts having the potential to satisfy these requirements are able to cope with heat loads in the range of 5-10 MW/m2 as maximum.

It has been recognized that the step between ITER and a FPR is too large; it is therefore suggested to take advantage of the expected intermediate step, such as DEMO, in order to develop an evolutive solution, in a similar way of that foreseen in the blanket development program.

The proposed strategy is therefore the following:

• Develop “near-term” concepts to be tested in the final stage of ITER operation and to be installed in DEMO reactor. These concepts could be based on water-cooling at relatively
low T (200 C-250 C), on W or TZM as armor material, and on Cu-based structural material (such as CuCrZr or Cu-based composites with reinforcement). If a steel liner is needed, the heat load capability of these concepts will be limited to about 10 MW/m² (up to 15 MW/m² without steel liner). The use of martensitic steel as structural material and water at PWR conditions (as for the DEMO blanket) could also be envisaged if a heat load limit of 7-8 MW/m² could be acceptable; in parallel, this step can be used to improve the knowledge on divertor plasma physics and enable to define measure to reduce the heat load requirements (down to 5 MW/m²).

- Develop “long term” concepts for FPR to be tested in DEMO which could allow very high coolant temperatures. SiC/SiC and W are candidate structural materials, together with W armor and Helium or Liquid Metals coolant. Typically, these concepts are able to cope with maximum heat flux of about 5 MW/m² (it has recently been shown that up to 10 MW/m² can be reached under certain favorable assumption and extrapolations).

8.4.2 Suggested plasma boundary conditions to be used for divertor design development.

The proposed conditions are coherent with those used in the blanket development:

- Cold plasma (<5eV) in order to allow easy He exhaust;
- No off-normal events such as ELMs or disruptions;
- Continuous operation;
- Peak heat load as low as 5 MW/m² for the ultimate FPR and up to 10 MW/m² for DEMO. This assumption has to be considered as a very large extrapolation from present-day knowledge. Potential methods for achieving this goal are: i) to introduce impurities (e.g., noble gases) in the divertor area for enhancing radiation; ii) to sweep the x-point by means of special coils installed behind the divertor.
- Remain below self-sputtering threshold in order to minimize sputtering. W, Ta and Mo are the best suited materials. Because redeposition process is unknown, the thickness of the required sacrificial layer can only be given with large uncertainties. Suggested value for W is 3 mm.
- Charge exchange erosion can be neglected in the divertor target region. However, it must be taken into account in the baffle region or close to the gas injection.
8.4.3 Divertor materials

Armour materials.
Tantalum is the best performing material from sputtering point of view but it cannot be used because of hydrogen issue and availability problems.
Tungsten remains the most promising option although it is a very brittle material which worsens under irradiation. For this reason, the use of small tiles is preferable to lamella monoblocks in order to reduce cracks initiators. Maximum temperature should be 1800°C for tiles and 1300°C for monoblocks; the minimum temperature should be over 500°C/600°C.
Mo-alloys offer an easier fabrication with similar temperature-window but present the additional problem of high long-term activation.

Structural materials.
In accordance with the definition given in chapter 7.4.1, structural materials (SM) can be divided in two categories:

- short/medium term SM associated with water-cooling to be installed in DEMO.
  - CuCrZr: it could be used for taking advantage of ITER experience. Typical T-window is 200 C-350 C, a good behavior has been shown up to 10dpa. However, because of the longer lifetime requirement expected in DEMO the use of a steel liner may be necessary (in order to avoid electrolytic erosion).
  - New high conductivity Cu-based materials such as Cu-W alloy or Cu-composites with SiC fibers. These materials have already been manufactured and they are stable up to 500°C, but irradiation effects are unknown. The use of a steel liner is probably required.
  - RAFM steel associated with water at PWR conditions. It has the advantages of being already under development for DEMO blankets. It has the potential to be used in a FPR. T-window 250°C-650°C in the ODS version.

- long term SM to be used in FPR.
  - W: it is the best potential candidate. The suggested T-window is 800 C-1300 C. Irradiation effects are unknown. Moreover, even for conceptual design activity, specific design rules have to be developed because the 3Sm rule used so far is not valid for such a brittle material.
  - SiC/SiC can be considered associated with Pb-17Li cooling especially when this material combination is already used for the blanket. Development of design rules and investigations on irradiation effects are already on-going for blankets. Specific need will
be the development of SiC-SiC/W joints and the fabrication of thin walls (e.g., 1 mm-thick).

### 8.4.4 Comments and R&D priorities

It has been stressed that the choice of a specific divertor concept can only be made in relation with the selected blanket system because of significant integration and safety constraints. Moreover, it is noted that, despite the divertor target is by far the most challenging sub-component, some thoughts have to be given to the concept of the whole divertor structures because a large amount of the divertor power (~50%) is deposited outside the target. R&D priorities should be given to: i) conceptual design of a whole divertor system to derive representative and critical mock-ups; ii) fabrication of these mock-ups corresponding out-of-pile thermo-mechanical tests for screening (especially for joints technology); iii) in-pile tests, in particular for the “near/medium term” concepts.

### 9. Findings

- The potential for application of materials in Fusion Power Plant Blankets and Divertors as well as their main critical issues are summarised in Table 5 and 6.
- The primary choice for FW and blanket in DEMO is based on RAFM-steel as structural material. The use of EUROFER should be restricted to zones where high temperature strength and low irradiation damage is required. Optimum shielding materials compromising the requirements of low activation, low after heat and low cost should be selected.
- In general it was observed that the industry will not invest in the development of “advanced materials” for fusion application. Therefore synergy with other advanced applications in aerospace and/or in specialised chemical plant could be rewarding. In the earlier scientific stage of the development, university support is crucial.
- Qualified nuclear databases and codes are required for material development and the design and licensing of fusion internal components. European nuclear data and activation libraries have been developed and extended to neutron energies of 20 MeV for these purposes. Validation will be achieved through benchmark experiments. Extrapolation of the database to even higher neutron energy in order to cover the IFMIF domain is also needed.
• The importance of the collaboration for material development and qualification in the frame of IEA Fusion Materials Implementing Agreement has been recognised. This collaboration should be pursued.

9.1 Blanket Structural Materials

9.1.1 RAFM Steels

EUROFER.
• A new EUROFER 97 batch with identical specification of alloying elements and reduced impurities is needed to provide materials for TBM mock-ups and prototypes, to check reproducibility of steel making and to improve the semi-finished products (tubes in particular).
• After results from running irradiation experiments a new heat “EUROFER 2” is foreseen. The elimination of nitrogen as source for radioactive carbon should be discussed with steel makers.
• The collection and elaboration of experimental data (in-pile and out-of-pile) to provide the designers of TBM with an appropriate data base and the extension of existing design rules for the condition of fusion components are of primary importance.
• Irradiation experiments with temperature range 200-400 C are of primary importance in order to determine the DBTT and the fracture toughness properties. In particular irradiation up to 15 dpa in MTR in this range of temperature and up to 70 ÷ 80 dpa in FBR at the lowest possible temperature should be performed.
• 250 C is the lower temperature limit in TBM design. This value should be verified but, considering the present uncertainties, 280-300 C is at present the recommended lower design value.
• The upper temperature limits for EUROFER is 550 C, considering 20000 hrs of operation with maximum stress of 100 MPa. Creep strength of highly irradiated EUROFER should be verified.
• Creep fatigue has to be analysed for TBM testing in ITER; if necessary meaningful experiments should be launched.
• Material transport of activated elements especially in water-loops and H-embrittlement under irradiation are areas to be investigated in more detail.
• Modelling and theory should be focussed mainly in two areas of interest: combined effects of irradiation damage, H and He production on plastic flow and embrittlement; and the small specimen test technology (SSTT) to be qualified towards accepted standards.

• Additional effort should be put in welds and joints development.

**RAFM-ODS.**

• ODS-steels have some potential for higher performance in HCPB and DC concept. Priorities in development and qualification should be given for near term applications of ODS in selected FW/blanket zones with the objective to increase the maximum temperature from 550 C (Eurofer limit) to 650 C (reasonable extrapolation from the present data).

• A screening among different routes for production of EUROFER-ODS reinforced with Y$_2$O$_3$ particles has been launched with the aim to produce an European ODS reference material. Improvement of fabrication processes (joining in particular) for the production of components is needed.

### 9.1.2. Advanced Materials

**SiC/SiC ceramic composites.**

• From the point of view of the designer the alternative material to be developed is SiC$_f$/SiC composites since concepts based on this structural material have the potential for exceptionally high thermal efficiencies.

• A limited development of this composite is needed to apply SiC/SiC in the DC breeding blanket concept. Mainly SiC/Pb-17Li compatibility and effectiveness of SiC/SiC as electrical insulator to mitigate MHD effects should be investigated.

• Advanced application of SiC$_f$/SiC composite requires to improve the material itself and to develop a material engineering database. In particular critical issues to be investigated are:
  - thermal, thermomechanical behaviour and dimensional stability under irradiation;
  - hermetic sealing of the SiC$_f$/SiC surfaces to avoid contact with coolant, breeder and neutron multiplier materials.

This material needs, compared with conventional ones, a new approach in the design procedure.

• A first EU reference SiC/SiC will be produced in 2002 using stoechiometric fibres, SiC fibre/matrix interface and β-SiC matrix in order to improve the material properties under irradiation.
Vanadium Alloys.
- Europe is not considering at present lithium as breeder/coolant in a breeding blanket design; therefore no specific R&D is planned for the development of Vanadium alloys.

Chromium and Titanium Alloys.
- This class of materials could be used in specific parts of the fusion reactor. No interesting conceptual design has been proposed up to now for in vessel structural applications. The major drawbacks are brittleness at low temperature for Cr and hydrogen embrittlement for Ti.

Tungsten and Molybdenum Alloys.
- The application of tungsten and molybdenum alloys is envisaged mainly in the divertor (see 8.3). For structural application in a fusion reactor (EVOLVE concept) the development and qualification would require a substantial increase in the material budget.

9.2. Divertor: Armour and heat sink materials

- It has been recognised that the divertor is a critical component for Tokamak based reactors. Additional effort should be put in the Power Plant Conceptual Study to identify the type of divertor to be considered for each blanket concept.
- For the armour materials:
  - Tantalum is the best from the sputtering point of view; main drawbacks are limited availability and hydrogen effects causing embrittlement, and a high tritium inventory;
  - Tungsten remains the most promising option although it is very brittle;
  - Molybdenum alloys offer an easier fabrication procedure but present a high long-term activation.

The R&D activity should be focussed in W (Mo) alloys with the aim to improve the ductility of this type of materials under irradiation and define the operating temperature window.

- Material for high heat flux/high efficiency divertors is a key issue. The following scenarios can be considered on the basis of technical difficulties/attractiveness:
  - Short term solution. CuCrZr can be used in a water cooled divertor concept taking advantage of ITER experience. Typical operating temperature window is 200-350 C with good behaviour tested up to 10 dpa. The thermal efficiency of a Power Plant based on this type of divertor is relatively modest.
Medium term solution. New high conductivity Cu-based materials such as Cu-W alloy or Cu-composites with CFC or SiC fibres can be considered. These types of materials have been manufactured and their stability up to 500°C were demonstrated in out-of-pile experiments. Irradiation effects are unknown. The development of such Cu-based materials can allow high heat flux (>10 MW/m²) with water as cooling media at PWR conditions (inlet and outlet temperature 265-325°C respectively, pressure 15.5 MPa).

Medium/long term solution. Tungsten alloys are the best potential candidates in the design of helium cooled divertor if heat flux up to about 10MW/m² is required. These alloys should be developed and characterised. Mainly the improvement of ductility, their irradiation behaviour, compatibility and fabrication processes including joinings should be investigated.

9.3. Advanced development

• Improvement in the component design can be achieved with appropriate combination of materials in relation to the design requirements. Typical examples are:
  - material reinforcement with thermally stable nano particles to increase strength and hence the upper temperature limit;
  - materials with graded properties to better comply with the stress/strain field;
  - multilayer structure for multifunction purposes.
  Progress made in such area should be considered by the designer of advanced power plant concept.

10. Conclusions

• EUROFER is the reference structural material for the two DEMO blanket concepts developed in EU (HCPB and WCLL) to be tested in ITER. The Dual Coolant concept studied in EU is also based on RAFM steel as structural material.
  Priorities in the R&D activities should be given to:
  - characterisation of EUROFER 97 for TBMs design and licensing and development of material processing (joints, HIP etc…);
  - characterisation of EUROFER 97 at high fluence up to 80 dpa in FBR and up to at least 15dpa in MTR;
- possible further improvement or optimisation of EUROFER 97 composition after availability of results from high fluence irradiation experiments.

- ODS can be used as functional materials in order to improve the performance of the present HCPB and WCLL concepts with limited effort in R&D.

- As part of the long term programme at least one additional structural material for advanced blanket concepts should be evaluated and critical points of this material addressed. SiC/SiC ceramic composites have been considered as a promising alternative in the EU blankets conceptual studies for its potential to increase the thermal efficiency.

- In addition to the present irradiation facilities for material characterisation (MTR, FBR, Accelerator facilities) an intense neutron source with relevant neutron fusion spectra is absolutely needed. Considering technical development risks, performance (neutron spectra, fluence, available test volume) and cost, IFMIF has been proposed at international level as the best solution to meet fusion requirements.

- The choice of a specific divertor concept can only be made in relation with the selected blanket system.

A strong effort in the divertor R&D activities is required and the following priorities are recommended:

- development of conceptual design of a whole divertor system in relation to each selected blanket concept;
- development and characterisation of the required materials;
- design and fabrication of relevant mock-ups;
- in-pile and out-of-pile testing.
References


[8] G. Le Marois, R. Lindau, C. Fazio, Structural Materials for the EU Tests Blankets; the EUROFER 97, ITER Materials Assessment Report, chap. 4.1.4


Tables and Figures.

Table 1: Main characteristics and potential performances of considered blanket concepts.
Table 2: Typical parameters of some divertor concepts proposed for EU power plant studies
Table 3: Screening of SiCcc, vanadium-, titanium-, chromium- and tungsten alloys for blanket applications.
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Figure 3: Improved HCPB blanket.
Figure 4: Dual-Coolant Blanket Concept.
Figure 5: Self-cooled Lithium/Vanadium Blanket.
Figure 6: TAURO Blanket Concept.
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Figure 8: DRAM Blanket Concept.
Figure 9: Schematic of EVOLVE First Wall Tubes and Blanket Trays
<table>
<thead>
<tr>
<th></th>
<th>WCCL</th>
<th>HCPB</th>
<th>DC</th>
<th>V-alloy/Li (US)</th>
<th>TAURO</th>
<th>ARIES-AT (US)</th>
<th>DREAM (Japan)</th>
<th>EVOLVE (US)</th>
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<tbody>
<tr>
<td>Assumed surface heat flux (peak) (MW/m²)</td>
<td>1.1</td>
<td>0.8</td>
<td>0.9</td>
<td>0.8</td>
<td>0.5</td>
<td>0.34</td>
<td>0.5</td>
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<td>Neutron wall load (peak) (MW/m²)</td>
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<td>4.4</td>
<td>5.0</td>
<td>6.0</td>
<td>3.5</td>
<td>4.8</td>
<td>3</td>
<td>10</td>
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<td>Structural Material</td>
<td>RAFM</td>
<td>ODS FM</td>
<td>ODS</td>
<td>RAFM V-alloy</td>
<td>SiC/SiC</td>
<td>SiC/SiC</td>
<td>SiC/SiC</td>
<td>W</td>
</tr>
<tr>
<td>Use of coating</td>
<td>Anti- corrosion</td>
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<td>SiC/SiC FC1</td>
<td>Electr. Insulator</td>
<td>SiC (LM infiltration)</td>
<td>SiC (LM infiltration)</td>
<td>SiC (He-leaks)</td>
<td>None</td>
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<td>Breeder /multiplier</td>
<td>Pb-17Li /none</td>
<td>Li-ceramic /Be</td>
<td>Pb-17Li /none</td>
<td>Li /none</td>
<td>Pb-17Li /none</td>
<td>Pb-17Li /none</td>
<td>Li-ceramic /Be</td>
<td>Li / none</td>
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<td>Breeder max Temp. [°C]</td>
<td>~ 550</td>
<td>880</td>
<td>~750</td>
<td>610</td>
<td>950</td>
<td>1100</td>
<td>920</td>
<td>1250</td>
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<td>Type of coolant / purge gas</td>
<td>water /none</td>
<td>Helium /Helium</td>
<td>Pb-17Li and He /none</td>
<td>Li/none</td>
<td>Pb-17Li /none</td>
<td>Pb-17Li /none</td>
<td>Helium /Helium</td>
<td>Li</td>
</tr>
<tr>
<td>Coolant in/out Temp. [°C]</td>
<td>265 / 325</td>
<td>250 / 550</td>
<td>Pb-17Li ~460/700</td>
<td>330/610</td>
<td>764/1100</td>
<td>600/900</td>
<td>1100/1200</td>
<td></td>
</tr>
<tr>
<td>Coolant press. [MPa]</td>
<td>15.5</td>
<td>8</td>
<td>14 (He)</td>
<td>0.5</td>
<td>1.5</td>
<td>1.5</td>
<td>10</td>
<td>0.035</td>
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<td>Max coolant velocity [m/s]</td>
<td>13.5</td>
<td>70</td>
<td>Pb-17Li ~ 0.5</td>
<td>1.3</td>
<td>4.2</td>
<td>?</td>
<td>2</td>
<td></td>
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<tr>
<td>Purge gas press.[MPa]</td>
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<td>0.1</td>
<td>none</td>
<td>none</td>
<td>none</td>
<td>none</td>
<td>0.1</td>
<td>nonr</td>
</tr>
<tr>
<td>Min/max Structures T [°C]</td>
<td>265 / 550</td>
<td>650</td>
<td>620</td>
<td>650</td>
<td>1090</td>
<td>764/1000</td>
<td>600/954</td>
<td>1100/1400</td>
</tr>
<tr>
<td>Max interface T (SM/breeder) [°C]</td>
<td>520</td>
<td>650</td>
<td>480 (steel)</td>
<td>630</td>
<td>970</td>
<td>994</td>
<td>?</td>
<td>1300</td>
</tr>
<tr>
<td>Potential net efficiency</td>
<td>~ 33%</td>
<td>~ 37%</td>
<td>~ 44%</td>
<td>46</td>
<td>~48%</td>
<td>~58%</td>
<td>~ 45%</td>
<td>~58%</td>
</tr>
<tr>
<td>Probable lifetime limits (irradiation)</td>
<td>RAFM</td>
<td>ODS, Breeder</td>
<td>ODS RAFM</td>
<td>V-alloy coating</td>
<td>SiC/SiC</td>
<td>SiC/SiC Breeder</td>
<td>W-alloy</td>
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Table 1 Main characteristics and potential performances of considered blanket concepts
<table>
<thead>
<tr>
<th>Coolant</th>
<th>Water</th>
<th>He (classic)</th>
<th>He (porous)</th>
<th>Pb-17Li</th>
<th>LM (Na)</th>
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<tbody>
<tr>
<td>Structural Mat</td>
<td>RAFM</td>
<td>W</td>
<td>W or TZM</td>
<td>SiC/SiC</td>
<td>W</td>
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<td>Elementary cell design</td>
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<td><img src="image3.jpg" alt="Image" /></td>
<td><img src="image4.jpg" alt="Image" /></td>
<td><img src="image5.jpg" alt="Image" /></td>
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<tr>
<td>Width/Inner D (mm)</td>
<td>15 / 10</td>
<td>40 / 30</td>
<td>36 / 28</td>
<td>25 / (20x10)</td>
<td>40 (30x30)</td>
</tr>
<tr>
<td>Armour mat &amp; thk</td>
<td>W / 6.5 mm</td>
<td>W / 3 mm</td>
<td>W / 3 mm</td>
<td>W / 5.5 mm</td>
<td>W / 6 mm</td>
</tr>
<tr>
<td>Details</td>
<td>Swirl, monoblock</td>
<td>Bi-directional, multi-section</td>
<td>Porous medium</td>
<td>Joint SiC-SiC / W tiles</td>
<td>Evaporation, capillarity</td>
</tr>
<tr>
<td>SM thickness</td>
<td>0.5 mm</td>
<td>2 + 3 mm</td>
<td>5 mm</td>
<td>1 mm</td>
<td>3 mm</td>
</tr>
<tr>
<td>Coolant P &amp; v</td>
<td>15.5 MPa - 15 m/s</td>
<td>14 MPa - 280 m/s</td>
<td>8 MPa - 140 m/s</td>
<td>0.3 MPa – 2.5 m/s</td>
<td>0.15 MPa – 230 m/s</td>
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<tr>
<td>Coolant in/out T (°C)</td>
<td>300 / 315</td>
<td>500 / 551</td>
<td>632 / 800</td>
<td>300 / 810</td>
<td>830/850 (Na)</td>
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<tr>
<td>Max heat flux</td>
<td>~7 MW/m²</td>
<td>~5 MW/m²</td>
<td>~5.5 MW/m²</td>
<td>~5 MW/m²</td>
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<td>Max T armor (°C)</td>
<td>1061</td>
<td>968</td>
<td>1300</td>
<td>1126</td>
<td>1400</td>
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<tr>
<td>SM T&lt;sub&gt;max/min&lt;/sub&gt; (°C)</td>
<td>340/494</td>
<td>556/755</td>
<td>630/1160</td>
<td>304/1035</td>
<td>950/1200</td>
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<tr>
<td>Limiting Criteria</td>
<td>3 S&lt;sub&gt;m&lt;/sub&gt;</td>
<td>3 S&lt;sub&gt;m&lt;/sub&gt;</td>
<td>3 S&lt;sub&gt;m&lt;/sub&gt;</td>
<td>Tauro criteria</td>
<td>3 S&lt;sub&gt;m&lt;/sub&gt;</td>
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Table 2 Typical parameters of some divertor concepts proposed for EU power plant studies
<table>
<thead>
<tr>
<th>Criterion</th>
<th>SiC Cer. Comp.</th>
<th>Vanadium</th>
<th>Titanium</th>
<th>Chromium</th>
<th>Tungsten</th>
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<tr>
<td>Temperature</td>
<td>1000 °C</td>
<td>700 °C</td>
<td>600 °C</td>
<td>1000 °C</td>
<td>1300 °C</td>
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<tr>
<td>Lifetime</td>
<td>6 y</td>
<td>?</td>
<td>?</td>
<td>?</td>
<td>?</td>
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<tr>
<td>Compatibility</td>
<td>R&amp;D</td>
<td>Li only</td>
<td>coating</td>
<td>+</td>
<td>+</td>
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<tr>
<td>Processing</td>
<td>R&amp;D</td>
<td>0</td>
<td>+</td>
<td>0</td>
<td>-</td>
</tr>
<tr>
<td>Joining</td>
<td>+</td>
<td>-</td>
<td>+</td>
<td>0</td>
<td>0</td>
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<tr>
<td>IRR &amp; PIE</td>
<td>5 dpa (*)</td>
<td>70 dpa</td>
<td>2.5 dpa</td>
<td>5 dpa</td>
<td>40 dpa</td>
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<td>Design rules</td>
<td>Special</td>
<td>conventional</td>
<td>conventional</td>
<td>Special</td>
<td>Special</td>
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<td>Nuclear Data</td>
<td>+/-</td>
<td>+/o</td>
<td>+/o</td>
<td>+/-</td>
<td>+/-</td>
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<tr>
<td>Recycling</td>
<td>once</td>
<td>?</td>
<td>+</td>
<td>+</td>
<td>+</td>
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<td>Synergy</td>
<td>Aerospace</td>
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<td>Chem. ind.</td>
<td>NO</td>
<td>NO</td>
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<td>Y</td>
<td>Y</td>
<td>Y</td>
<td>Y</td>
<td>Y</td>
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<td>Industry</td>
<td>To be paid</td>
<td>To be paid</td>
<td>To be paid</td>
<td>To be pad</td>
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<tr>
<td>IEA co-op.</td>
<td>YES</td>
<td>YES</td>
<td>NO</td>
<td>NO</td>
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(*) Higher dpa levels have been recently achieved.

Table 3 Screening of SiCcc, vanadium-, titanium-, chromium- and tungsten alloys for blanket applications
<table>
<thead>
<tr>
<th>Blanket Concept</th>
<th>RAFM Steel</th>
<th>ODS Steel</th>
<th>Vanadium Alloy</th>
<th>SiC Composite</th>
<th>W</th>
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<tr>
<td>WCLL</td>
<td>X</td>
<td>?</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HCPB</td>
<td>X</td>
<td>X</td>
<td></td>
<td>?</td>
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<tr>
<td>Dual Coolant</td>
<td>X</td>
<td>X (FW plated)</td>
<td>(Flow Channel Insert)</td>
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<tr>
<td>Self-cooled Li</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Self-cooled Pb-17Li</td>
<td></td>
<td></td>
<td></td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>Evaporation Cooling</td>
<td></td>
<td></td>
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Table 4  Blanket Concepts and related structural materials
<table>
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<tr>
<th>Criteria</th>
<th>FM-steels</th>
<th>ODS-FM steels</th>
<th>Cr alloys</th>
<th>α Ti alloys</th>
<th>V alloys</th>
<th>W alloys</th>
<th>SiC-SiC</th>
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<td>Range of T (C) at present</td>
<td>300-550</td>
<td>350-650</td>
<td>400-700</td>
<td>RT-600</td>
<td>400-600</td>
<td>900-1300</td>
<td>800-1000 (FW)</td>
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<tr>
<td>Possible Improvement?</td>
<td>250-550</td>
<td>300-750</td>
<td>300-800</td>
<td>RT-700</td>
<td>300-700</td>
<td>800-1400</td>
<td>550-1200</td>
</tr>
<tr>
<td>T (C) required for attractive power plant</td>
<td>WCLL</td>
<td>H2O liq</td>
<td>good</td>
<td>good</td>
<td>good</td>
<td>good</td>
<td>good</td>
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<tr>
<td>Compatibility with coolant, breeder...</td>
<td>H2O liq</td>
<td>PbLiq</td>
<td>Li ceramic</td>
<td>Li ceramic</td>
<td>Li ceramic</td>
<td>Li ceramic</td>
<td>Li ceramic</td>
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<tr>
<td>Main critical issues</td>
<td>DBTT after irrad</td>
<td>* Welds, connection</td>
<td>DBTT after irrad</td>
<td>DBTT after irrad</td>
<td>DBTT after irrad</td>
<td>DBTT after irrad</td>
<td>DBTT after irrad</td>
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Table 5 Potential of Materials for Fusion Power Plant Blanket
<table>
<thead>
<tr>
<th>Criteria</th>
<th>W-base alloys</th>
<th>Mo-base alloys</th>
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<tr>
<td>Range of Temp at present (C)</td>
<td>$T_{\text{min}} = 300^\circ$ to $900^\circ$</td>
<td>$T_{\text{min}} = 300^\circ$ to $800^\circ$</td>
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<td></td>
<td>$T_{\text{max}} = 2000$</td>
<td>$T_{\text{max}} = 1800$</td>
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<td>Possible Improvement?</td>
<td>$T_{\text{min}} = 300^\circ$ to $800^\circ$</td>
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<tr>
<td></td>
<td>$T_{\text{max}} = 2000$</td>
<td>$T_{\text{max}} = 1800$</td>
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<tr>
<td>Compatibility with plasma</td>
<td>$\Phi$ Erosion $&lt; eV$?</td>
<td>$&lt; eV$?</td>
</tr>
<tr>
<td></td>
<td>$\chi$ erosion (O) $&lt; 800^\circ$ C</td>
<td>$?</td>
</tr>
<tr>
<td>T (C) required for attractive power plant</td>
<td>H2O/RAFM $-$</td>
<td>$-$</td>
</tr>
<tr>
<td></td>
<td>H2O/Cu/W(Mo) $?$</td>
<td>$?$</td>
</tr>
<tr>
<td></td>
<td>He/W(Mo) $?$</td>
<td>$?$</td>
</tr>
<tr>
<td></td>
<td>LM/W(Mo) $?$</td>
<td>$?$</td>
</tr>
<tr>
<td></td>
<td>LM/SiC/W(Mo) $300$-1035</td>
<td>$?</td>
</tr>
<tr>
<td>Critical issues</td>
<td>* design, fabrication, joining</td>
<td>* design, fabrication, joining</td>
</tr>
<tr>
<td></td>
<td>* DBTT after irradiation</td>
<td>* DBTT after irradiation</td>
</tr>
<tr>
<td></td>
<td>* plasma effect, oxidation</td>
<td>* plasma effect, oxidation</td>
</tr>
<tr>
<td></td>
<td>* after heat</td>
<td>* long-term waste</td>
</tr>
<tr>
<td></td>
<td>* erosion (lifetime)</td>
<td>* erosion (lifetime)</td>
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<tr>
<td>Application</td>
<td>1/ armour material</td>
<td>1/ armour material</td>
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<tr>
<td></td>
<td>2/ structure?</td>
<td>2/ structure?</td>
</tr>
</tbody>
</table>

* armour application
** structural application

Table 6  Potential of Materials for Fusion Power Plant Divertor
Fig. 1: Materials and Breeding Blankets R&D development and their inter-link with fusion devices and irradiation facilities.
Fig 2 3D-view of a WCLL blanket segment
Fig. 3  Improved HCPB Blanket
Fig. 4  Dual-Coolant Blanket Concept
Fig. 5  Self-cooled Lithium/Vanadium Blanket
Fig. 6  TAURO Blanket Concept

Fig. 7  ARIES-AT Blanket Concept
Fig. 8  DREAM Blanket Concept
Fig. 9  Schematic of EVOLVE First Wall Tubes and Blanket Trays
## ANNEX I

### Materials Assessment Meeting

Karlsruhe, 5-8 June 2001

**AGENDA**

<table>
<thead>
<tr>
<th>TUESDAY 5 JUNE 2001</th>
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<tr>
<td>13:00-13:05</td>
<td>Welcome</td>
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<td>13:05-13:15</td>
<td>Agenda, objectives of the meeting</td>
<td>M. Gasparotto</td>
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**Session 1: Status Report**

*Chairman: B. van der Schaaf*

<table>
<thead>
<tr>
<th>Time</th>
<th>Topic</th>
<th>Presenter</th>
</tr>
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<tr>
<td>13:15-14:15</td>
<td>RAFM &amp; ODS</td>
<td>K. Ehrlich</td>
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<tr>
<td>14:15-15:00</td>
<td>SiCf-SiC</td>
<td>B. Riccardi</td>
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<tr>
<td>15:00-15:30</td>
<td>Cr, Ti, V</td>
<td>R. Pippan</td>
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<tr>
<td>15:30-15:45</td>
<td><strong>Coffee break</strong></td>
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<tr>
<td>15:45-16:15</td>
<td>PFM: refractory materials</td>
<td>M. Victoria</td>
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<tr>
<td>16:15-16:45</td>
<td>Plasma effects</td>
<td>A. Loarte</td>
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<tr>
<td>16:45-17:15</td>
<td>Nuclear database</td>
<td>R. Forrest</td>
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<tr>
<td>17:15-18:00</td>
<td>Discussion on session 1</td>
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<tr>
<th>WEDNESDAY 6 JUNE 2001</th>
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<tbody>
<tr>
<td>8:30-9:30</td>
<td>Materials present limits, estimated margins for improvement</td>
<td>F. Tavassoli</td>
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<tr>
<td>9:30-10:00</td>
<td>Codes and standards for materials</td>
<td>E. Diegele</td>
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<tr>
<td>10:00-10:15</td>
<td><strong>Coffee break</strong></td>
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<tr>
<td>10:15-11:00</td>
<td>Basic Modelling</td>
<td>G. Martin</td>
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<tr>
<td>11:00-11:45</td>
<td>Screening of new materials</td>
<td>B. van der Schaaf</td>
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<tr>
<td>11:45-12:30</td>
<td>Discussion on session 2</td>
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<tr>
<td>12:30-14:00</td>
<td><strong>Lunch</strong></td>
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**Session 3.1: Power reactor requirements**

*Chairman: D. Maisonnier*

<table>
<thead>
<tr>
<th>Time</th>
<th>Topic</th>
<th>Presenter</th>
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<tbody>
<tr>
<td>14:00-14:45</td>
<td>Power plant studies</td>
<td>R. Andreani</td>
</tr>
<tr>
<td>14:45-16:45</td>
<td>Materials requirements for FW/BB</td>
<td>L. Giancarli, S. Malang</td>
</tr>
<tr>
<td>16:45-17:00</td>
<td><strong>Coffee break</strong></td>
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<tr>
<td>17:00-17:45</td>
<td>Materials requirements for HHF</td>
<td>K. Kleefeldt</td>
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### THURSDAY 7 JUNE 2001

<table>
<thead>
<tr>
<th>Time</th>
<th>Session 3.2: Power reactor requirements</th>
<th>Chair</th>
</tr>
</thead>
<tbody>
<tr>
<td>8:30-8:45</td>
<td>Organisation of the parallel session</td>
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<tr>
<td>8:45-9:45</td>
<td>Irradiation devices and testing</td>
<td>A. Moeslang</td>
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<tr>
<td>9:45-10:15</td>
<td>Safety requirements</td>
<td>G. Marbach</td>
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<td>10:15-10:30</td>
<td>Coffee break</td>
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<tr>
<td>10:30-11:00</td>
<td>Industry and Utility requirements for</td>
<td>F. Alonso</td>
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<td>Power Plant</td>
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<tr>
<td>11:00-11:30</td>
<td>Waste and recycling</td>
<td>I. Cook</td>
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<tr>
<td>11:30-12:30</td>
<td>Discussion on session 3</td>
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<tr>
<td>12:30-14:00</td>
<td>Lunch</td>
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### Session 4: Parallel session

<table>
<thead>
<tr>
<th>Time</th>
<th>Experts Groups *)  **)</th>
<th>Chair</th>
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</thead>
<tbody>
<tr>
<td>14:00-16:15</td>
<td>Experts Groups *)  **)</td>
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</tr>
<tr>
<td>1 st group</td>
<td>RAFM &amp; ODS</td>
<td>K. Ehrlich</td>
</tr>
<tr>
<td>2 nd group</td>
<td>Advanced materials</td>
<td>B. van der Schaaf</td>
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<tr>
<td>3 rd group</td>
<td>FW/Blanket incl. shielding</td>
<td>S. Malang</td>
</tr>
<tr>
<td>4 th group</td>
<td>Divertor designs, armour and plasma interaction</td>
<td>L. Giancarli</td>
</tr>
<tr>
<td>16:15-16:30</td>
<td>Coffee break</td>
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<tr>
<td>16:30-17:30</td>
<td>Preparation of the reporting</td>
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</table>

### FRIDAY 8 JUNE 2001

<table>
<thead>
<tr>
<th>Time</th>
<th>Session 5: Plenary session</th>
<th>Chair</th>
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</thead>
<tbody>
<tr>
<td>9:00-12:00</td>
<td>Reporting of the experts groups</td>
<td>M. Gasparotto</td>
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<tr>
<td></td>
<td>Discussion</td>
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<tr>
<td></td>
<td>Conclusion</td>
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</table>

*) The members of the experts groups will be proposed by EFDA.

**) A list of questions/topics to be discussed will be addressed by EFDA to the different experts groups. Each of them will report during the plenary session.
Considered materials

- Eurofer
- ODS-RAFM steel
- ODS-ferritic steel

Discussion

*TBM and IFMIF design requirements*

The R&D activities needed in the RAFM programme in order to design and fabricate the TBM’s to be tested in ITER-FEAT and the IFMIF facility should be identified. The requirements for the preparation of the safety report should be also considered.

- RAFM irradiation and PIE, R&D priorities:
  1- irradiation conditions,
  2- PIE: type and testing conditions,
- Additional characterisation and testing, R&D priorities:
  3- mechanical testing prior to irradiation
  4- H and corrosion effects
- RAFM specification and procurement, R&D priorities:
  5- how to specify in order to improve the performances and homogeneity of the material’s properties.
- RAFM manufacturing and implementation issues, R&D priorities for:
  6- welds and joints
  7- tubing,
  8- powder HIP material qualification.
- Rules for design and licensing, R&D priorities for:
  9- material qualification,
  10- design code.
- Nuclear data needs for:
  11- TBM’s design and licensing
  12- IFMIF (high energy neutrons)

*DEMO and Power plant design requirements*

- Application of RAFM and ODS incl. Ferritic:
13- Range of temperature: limiting factors and mechanisms,
14- Possible lifetime: limiting factors (including irradiation) and mechanisms,
15- Others limitations: interactions with the coolant/breeder/multiplier,
16- R&D priorities to improve the limits (base material, coating...)

- RAFM and ODS specification, R&D priorities to fulfil the basic design goals:
  17- Irradiation tests needs
  18- Modelling to assess and interpret the irradiation damage and the H/He effect.
  19- Economic decommissioning: possible improvements to decrease the level of impurities.
- ODS manufacturing issues
  20- How to improve the reinforcement distribution and density.
  21- Joining possibilities (weldings, diffusion bonding, powder metallurgy, others?)
- Licensing and decommissioning
  22- How to reduce the uncertainties on the activity level.
  23- Management of activated materials: potential of recycling?
- Collaboration
  24- Synergies and spin-offs with others fields and technologies.
2nd group: Advanced materials
Chairman: B. van der Schaaf

Considered structural and (functional)* materials
- SiCf/SiC
- Cr alloys
- V alloys
- Ti alloys
- W alloys
- New materials?

*) with the exception of the armour materials, to be discussed within the 4th group

Discussion
1- Minimum development needed in order to consider it for Fusion application
   - Application of advanced materials:
     2- Range of temperature: limiting factors and mechanisms,
     3- Possible lifetime: limiting factors (incl irradiation) and mechanisms,
     4- Others limitations: interactions with the coolant/breeder/multiplier,
     5- Possible application of Cr and Ti alloys (structural/functional)
   - Manufacturing and implementation issues:
     6- Processing to improve the limits
     7- Joining, coating, sealing...
     8- Availability and cost of materials
   - Characterisation of advanced materials:
     9- Irradiation testing conditions
     10- PIE: type and testing conditions
     11- Additional characterisation
   - Licensing and decommissioning
     12- Specific rules for design (in particular for SiC/SiC) and licensing
     13- Nuclear data needs
     14- Management of activated materials: potential of recycling, implication on materials and R&D.
   - Collaboration
     15- Synergies and spin-offs with others fields and technologies.
     16- Contribution (from universities, industry) to the development of new materials, how to develop?
Considered power plant concepts

- WCLL
- HCPB
- Dual coolant
- Advanced self-cooled
- ARIES RS
- EVOLVE

Discussion
- Performances and lifetime
  1- Minimum materials requirements to justify the proposed concepts.
  2- For the proposed concepts, which are the parameters limiting the performances (in terms of temperature application, compatibility).
  3- For the proposed concepts, which are the parameters limiting the lifetime and the availability (material performance under irradiation, compatibilities, activation, welds...).
- Tritium issues
  4- TBR limitation (impact on the choices of the materials),
  5- retention and detritiation (needs for coatings?, impact on performance)
- Manufacturing issues
  6- Implication on the development of fabrication technologies (welds and joints, coatings, powder...)
- Design issues
  7- which materials for shielding and vacuum vessel
  8- needs for improved design (i.e. radial segmentation) in order to optimise the waste sorting into different classes.
- Licensing and decommissioning
  9- testing (incl. irradiation) and design rules needs.
  10- additional nuclear data?
  11- safety requirements, implication on materials (decay heat, chemical reactivity...).
  12- additional industry and utility requirements, implication on materials.
Additional discussion for concepts using SiC/SiC as structural material
13- For a given mechanical load, which are the limiting composite properties (fibre, matrix, interface strength).
14- Needs for specific models and codes in order to perform design analysis?
15- Coating (needs for sealing, to avoid interaction with fibres, reliability?)

Additional discussion for concepts using V alloys as structural material
16- Which possible coolant and breeder associated with?
17- Which are the coating function and requirement?
18- Which level of impurities in the breeder/coolant can be reasonably achieved in operation?

Additional questions for concepts using W as structural material
19- Which possible coolant and breeder associated with?
20- Fabrication feasibility incl. connections with the heat exchanger?
21- Can we cope with the high after heat?

Additional questions for concepts using ODS as functional material
22- What is the main incentive to go to ODS (creep rate, YS at high T?)
23- Which application for ferritic ODS steel (up to 1050K)?
24- Joining possibilities (weldings, diffusion bonding, powder metallurgy, others?)

Additional questions for concepts using SiC as flow channels insulators
25- Which type of materials (SiC composites, porous SiC, others?)
26- Which are the requirements (electrical/thermal insulation, leak tightness, compatibility with Pb-17Li, mechanical strength).

27- Ideas for new materials, technologies and concepts in order to improve the efficiency, availability at low cost and economic decommissioning.
Considered family of concepts and armour for Divertor

- H2O/Cu/W
- H2O/RAFM
- He/W
- LM/W
- LM/SiCSiC/W
- Others ?
- W vs Mo

Discussion

- Performances and lifetime
  1- Minimum materials requirements to justify the proposed concepts.
  2- Acceptable operating temperature windows for armour materials.
  3- For the proposed concepts, which are the parameters limiting the performances (in terms of heat loads, temperature application, compatibility).
  4- For the proposed concepts, which are the parameters limiting the lifetime and the availability (material performance under irradiation, activation, joints...).
  5- Choice of the armour material: W vs Mo comparison, others materials?
- Boundary conditions
  6- Minimum outlet coolant temperature for acceptable reactor efficiency.
  7- Possible plasma boundary conditions, implication on armour materials.
- Manufacturing issues
  8- R&D priorities in the development of the armour materials.
  9- R&D priorities in the development of fabrication technologies (welds and joints, coatings, castellation...).
- Design issues
  10- Ideas for new materials, technologies and concepts in order to improve the performances (thermal loads) and lifetime.
- Licensing and decommissioning
  11- Testing (incl. irradiation) and design rules needs.
  12- Activation uncertainties and additional nuclear data,
  13- Safety requirements, implication on materials (decay heat, chemical reactivity...).
  14- Additional industry and utility requirements, implication on materials.