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# International strategy for fusion materials development

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**Abstract**

In this paper, the results of an IEA-Workshop on Strategy and Planning of Fusion Materials Research and Development (R&D), held in October 1998 in Risø Denmark are summarised and further developed. Essential performance targets for materials to be used in first wall/breeding blanket components have been defined for the major materials groups under discussion: ferritic–martensitic steels, vanadium alloys and ceramic composites of the SiC/SiC-type. R&D strategies are proposed for their further development and qualification as reactor-relevant materials. The important role of existing irradiation facilities (mainly fission reactors) for materials testing within the next decade is described, and the limits for the transfer of results from such simulation experiments to fusion-relevant conditions are addressed. The importance of a fusion-relevant high-intensity neutron source for the development of structural as well as breeding and special purpose materials is elaborated and the reasons for the selection of an accelerator-driven D-Li-neutron source – the International Fusion Materials Irradiation Facility (IFMIF) – as an appropriate test bed are explained. Finally the necessity to execute the materials programme for fusion in close international collaboration, presently promoted by the International Energy Agency, IEA is emphasised. 

## 1. Introduction

One of the main technical barriers to the realization of nuclear fusion for energy supply is the development and qualification of materials for the first wall, limiters, divertors and breeding blanket components to withstand complex loading conditions including 14 MeV neutrons, neutral and charged plasma particles, and high surface heat fluxes. The potential impediment to the development of fusion power associated with materials performance has been recognised as a critical issue by the entire fusion community. To address this issue the fusion materials community has carried out R&D-activities for over two decades. From the beginning, materials R&D has profited from international collaboration. As an example, the Implementing Agreement for a Programme of Research and Development on Fusion Materials, established in 1980 by the International Energy Agency (IEA), has provided an effective frame for successful collaboration. The development of low-activation ferritic–martensitic steels, the investigations on solid breeder materials (e.g., Beatrix-II experiments) and conceptual design studies for the development of an International Fusion Materials Irradiation Facility (IFMIF) are examples. Financial constraints in national programmes as well as a partial programmatic reorientation have caused the national materials programmes to turn increasingly to international collaboration in order to accomplish their objectives. An IEA-Workshop on Strategy and Planning of Fusion Materials R&D, was held from 5–7 October 1998 in Risø, Denmark, to discuss the present status of developmental work, to identify obvious critical issues for the materials envisaged and to present the national R&D-programmes, plans and time schedules in order to seek for the possibility of a commonly agreed international material development strategy. The major results of this meeting which was attended by 20 delegates from the European Union, Japan, the People's Republic of China, Russia

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and the United States of America are summarised in this paper. Once again the need for a neutron source capable of supporting materials R&D was confirmed.

## 2. General performance goals for fusion reactors and materials requirements

The national strategies for the development of fusion power reactors differ from one country to another. Whereas in Europe, Japan and Russia a mission-oriented and hence time-driven strategy exists to develop and achieve a magnetically confined demonstration and reactor plant, a reorientation of the US programme with emphasis on the general advancement of plasma physics and fusion technology has taken place. The materials community can, however, independent from this divergence in the general research policy, agree to an envelope of common performance goals for a demonstration (DEMO) reactor or a commercial fusion power plant, which – with the exception of some inertial-confined systems – are more or less independent of the specific type of plasma device and hence can be used for the formulation of general development targets.

In the long term development, materials which can withstand high neutron wall loads under temperature and coolant pressure conditions necessary to drive efficient thermodynamic working cycles must be developed. Also the end-of-life neutron fluence (or lifetime) must be high enough to limit the necessary replacements of near-plasma components like first wall/breeding blankets to a minimum. In addition to this demand of high performance and safe operation, which is necessary in order to be competitive with conventional and nuclear power plants, the materials should be of 'low-activation' in order to achieve the ultimate environmental attractiveness of fusion power.

In Table 1 a range of performance goals presented at different occasions is compiled for magnetically confined DEMO and power reactors [1–3]. They are based on present knowledge in plasma physics and fusion technology. Key parameters that impact material development are the expected surface heat and neutron wall load in MW/m$^2$, the volume power density and radiation damage parameters like the displacement rate and transmutation reaction rates. The integrated neutron wall load in MWy/m$^2$ and derived damage parameters will determine the degradation of materials under neutron exposure and hence limit the lifetime of components. An equally important parameter is the mode of reactor operation. In the next-step machines like ITER or DEMO, one has to reckon with limited pulse lengths, whereas in power reactors a quasi-continuous or steady state operation is expected, which can eventually be achieved by the non-inductive current drive for Tokamaks or the Stellarator concept.

Whereas neutron wall load and the mode of operation are fairly independent of the materials choice, other parameters which are equally important such as the operational temperature, the primary pressure level of the cooling medium and the combination of structural with other materials are very design-dependent. General consensus exists that for integrated first wall breeding blanket concept only a limited number of combinations of structural materials with breeding/coolant media and neutron multiplier materials exist. They are compiled in Table 2 and can be classified with regard to the breeding materials into two major categories, namely solid ceramic and liquid metal breeders with the options of self-cooled or separately cooled versions. Three major structural materials, ferritic–martensitic (F/M) steels, vanadium alloys and SiC/SiC ceramic composites which can fulfil the requirement for 'low activation' have been considered in different designs. Primary system pressures of the cooling media and expected temperature ranges of operation for the structural materials are also indicated. The latter data are based on estimates of achievable maximum temperatures, limited by high temperature creep rupture strength and/or corrosion resistance and on coolant inlet temperature or possible low tempera-

Table 1
General performance goals for fusion devices

| | ITER | DEMO | REACTOR |
|---|---|---|---|
| Fusion power | 0.5–1 GW | 2–4 GW | 3–4 GW |
| Neutron wall loading (first wall) | 0.5–1 MW/m$^2$ | 2–3 MW/m$^2$ | 2–3 MW/m$^2$ |
| *Integrated wall load (first wall)* | | | |
| In MWy/m$^2$ | 0.3–1 MWy/m$^2$ | 3–8 MWy/m$^2$ | 10–15 MWy/m$^2$ |
| In displacements per atom[a] | 3–10 dpa | 30–80 dpa | 100–150 dpa |
| Operational mode | Pulsed (300–1000 s) $< 5 \times 10^4$ cycles | Quasicontinuous | |
| Plant lifetime | | | ~30 FPy |
| Net plant efficiency | | | ~30% |

[a] The following relations between neutron wall loading, neutron flux and displacements per atom have been used: 1 MW/m$^2 \simeq 3 \times 10^{14}$ $n_{tot}$/cm$^2$·s $\simeq 3 \times 10^{-7}$ dpa/s (Fe); 1 MWy/m$^2 \simeq 10$ dpa (Fe); The calculation of dpa according to the Norgett–Robinson–Torrens (NRT) model.

Table 2
Major breeding blanket concepts

| | Coolant | Breeding material | Structural material | Neutron multiplier | Operation conditions | |
|---|---|---|---|---|---|---|
| | | | | | Temperature (°C) | Pressure (MPa) |
| He/LiCe/FS/Be[a] | He | LiCe | F/M-steel | Be | 250–550 | 5–20 (8) |
| He/LiCe/SiC/SiC/Be | He | LiCe | Ceramic composite SiC/SiC | Be | 450–950 | 5–20 |
| Li/V | Li | Li | Vanadium alloy | Li | 350–750 | ~1 |
| $H_2O$/Pb–Li/FS[b] | $H_2O$ | Pb–Li | F/M-steel | Pb–Li | 250–550 | 12–15 (15.5) |

[a] HCPB – Helium-cooled pebble-bed blanket/EU; Pressure data in brackets.
[b] WCLL – Water-colled Lithium–Lead blanket/EU; Pressure data in brackets. LiCe – Lithium ceramic breeder materials: $Li_2O$, $Li_4SiO_4$,$Li_2ZrO_3$ or $Li_2TiO_3$.

ture limits set by irradiation hardening and embrittlement.

Whereas, the above mentioned structural and solid ceramic breeder materials will be limited by their thermophysical properties to neutron wall loads in the range of 2–3 MW/m$^2$, values of up to 10 MW/m$^2$ could be managed by the use of liquid lithium (or Pb–17% Li) as breeder- and refractory alloys like W–Re as first wall/structural material, provided such high plasma power densities can be realised under economic conditions. The reintroduction of materials like tungsten, which possess less attractive radiological properties under neutron exposure, has to be carefully balanced against obvious advantages like higher power density capability, high melting point and superior high-temperature creep strength.

A sound technical comparison of the above compiled design concepts for breeding blanket components is at present not yet possible. For this to be accomplished, materials development must be coupled closely with engineering design activities as discussed in Section 4.

## 3. Present status of material development, critical issues and near-term R&D

At present three material groups are pursued in national and international materials programmes [4,5]. These are advanced ferritic/martensitic steels, vanadium-based alloys and SiC-fibre reinforced ceramic SiC composites. Their selection is mainly based on favourable conventional properties and/or technical maturity, a potential for low activation and /or promising results under fission neutron irradiation. Opportunities for the consideration of other structural materials could arise as a result of major advances in the broad field of materials science, e.g., through exploratory studies on chromium-based alloys or TiAl intermetallics, or through evolving new design concepts which need specific capabilities. The above mentioned W–Re-alloys for use in high power density concepts is an actual example. The present status, critical issues and major near-term research activities for these materials are summarised in Table 3 and will now be briefly described.

Table 3
Material issues and major areas for near-term R&D

| | |
|---|---|
| RA-F/M steels | • Data base development of PCA<br>• *Fracture toughness degradation/Embrittlement by irradiation*<br>• *Ferromagnetic effects*<br>• Development of ODS-nanocomposite ferritics for high temperature application (650–750°C) |
| Vanadium alloys | • *Development of insular coatings (MHD-effects)*<br>• *Impurity (O, N, C) pick-up from environment/Embrittlement*<br>• *Fracture toughness degradation by irradiation/Embrittlement* |
| SIC-composites | • Design of composite structures for improved performance<br>• *Fundamental property response to irradiation*<br>• Development of a technology base for fabrication/Joining |
| Tungsten alloys | • Fabrication and joining technologies<br>• Radiological properties – low activation/Waste disposal/Safety<br>• *Radiation embrittlement and compatibility* |

*Ferritic–martensitic steels* are furthest along the development path in that there exists a well developed technology and a broad industrial experience with such alloys in fossil and nuclear energy technology. They show reasonably good thermophysical and mechanical properties, good compatibility with major cooling and breeding materials and a low sensitivity to swelling and helium embrittlement. In recent years low activation versions of these commercially-deployed materials have been developed with equivalent or even improved properties [6]. A major issue, according to Table 3, is the observed radiation-induced degradation of flow and fracture properties below about 350°C, though newer results indicate that the recently developed low-activation Fe–7–9%CrWVTa alloys are less sensitive to radiation hardening and embrittlement than the commercial ones. A possible influence of ferromagnetism on plasma stability and the effect of magnetically-induced Lorenz forces on structural components is under investigation and should soon be clarified. Current research is focusing in collaborative international experiments on these issues. For a further improvement of creep rupture properties the development of alloys strengthened with nano-scaled oxide dispersions (ODS-alloys) and precipitates has started to expand the application to 650°C or even 750°C.

*Vanadium based alloys*. Vanadium alloys based on V–Cr–Ti constituents have a favourable combination of physical properties and high creep strength and hence the greatest potential of the three material groups for high temperature operation in liquid lithium [7]. This alloy group has by far the fastest decay of radioactivity for interim and long decay times, especially if the concentration of radiologically unwanted impurities can be controlled. Major results of irradiation experiments regarding swelling and high-temperature embrittlement are also promising. Similar to the situation in ferritic–martensitic steels, the lower operational temperature is limited by the propensity for brittle failure which is induced by radiation hardening. A major drawback and a possible feasibility issue is the high solubility and permeability of tritium and solubility of interstitial elements like O, C and N, which can lead to catastrophic embrittlement. The development of self-healing, corrosion-protective and at the same time insulating coatings which help to mitigate the magnetohydrodynamic (MHD) effects in liquid-metal cooled breeding blanket concepts has, therefore, to be a focus of current research efforts.

The development of *SiC/SiC composite materials* presents the most difficult challenge of the three groups of materials [8]. They have potentially high payoffs in terms of very low radioactivity and decay heat at short and intermediate decay times and offer high operating temperatures. The primary feasibility issues involved in the development of these materials are a principal understanding of the effects of neutron irradiation on the behaviour of the complex fiber/interface/matrix structure under the aggravating conditions of high cross-sections for elastic displacement events and inelastic processes (e.g., the formation of helium via n,$\alpha$-processes) and the adverse impact of radiation-induced loss of thermal conductivity on allowable heat fluxes. A key issue for a reasonable application of fibre-reinforced ceramic composites of this type lies, therefore, in the development of a radiation-resistant material. This can eventually be achieved by developing quasi-stoichiometric SiC-fibres with properties nearly identical to the SiC matrix. Further drawbacks are the very limited technology base for production and joining and insufficient hermetic sealing capabilities. Finally the development of appropriate design rules for the use of these innovative materials as structural parts in fusion technology is necessary.

## 4. Strategies for material development

The pre selection of these materials for first wall (FW) breeding blankets is either based on already existing conventional data and/or on available experience with such materials in conventional and nuclear technology or simply on the expectation of the designers to achieve maximum efficiency and performance. A short assessment in Section 3 has, however, shown, that the level of knowledge about and the status of development differs remarkably for these alternatives, so that the next step research activities are specific for each material.

For the *ferritic–martensitic steels* which are furthest in development and show the fewest areas of concern, a broad data base needs to be generated in existing test and irradiation facilities within the next decade to qualify this material for DEMO-relevant breeding blankets. A major objective is to explore the technical feasibility of these blanket concepts using a reliable engineering database (concept exploration phase). This means that an *integrated research and test programme* including studies on structural and functional materials (breeder and neutron-multiplier materials) has to be executed in parallel to develop innovative fabrication technologies for sub-modules and mock-ups. Fig. 1 summarises as an example the necessary R&D work for the so-called helium-cooled pebble-bed breeding blanket (HCPB) in the EU. In a previous Alloy Screening and Development Phase I, a series of laboratory ferritic–martensitic–7–10% CrWVTa melts with reduced long term activation had been investigated; and as a result a primary candidate 9% CrWVTa alloy EUROFER has in the meantime been specified and produced. Its qualification as a structural material for DEMO test blanket modules includes the determination of conventional properties and the testing of the irradiation behaviour

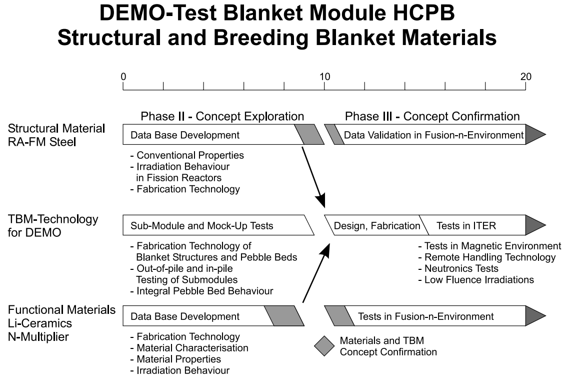


Fig. 1. The parallel development of structural and functional materials for a DEMO-test blanket module (helium cooled pebble bed concept in EU).

mainly in fission reactors up to damage levels of 10, 30 and 70 dpa. Also innovative technologies for the fabrication of sub-modules and mock-ups by hot isostatic pressing (HIPPING) and for non-conventional joining-techniques like diffusion or resistance welding have to be investigated. In parallel the development, fabrication, characterisation and testing of ceramic breeders of $Li_4SiO_4 \cdot LiO_2$ and of beryllium as a neutron multiplier material has to be pursued. This has to be accompanied by technology-oriented tests of sub-modules and mock-ups, where the integral function and behaviour of selected material combinations can be studied in out of pile- and in-pile experiments. Questions on compatibility, corrosion, mechanical interaction, radiation-induced swelling, creep and dimensional instabilities in steep thermal and neutron gradients and the effective tritium release mechanisms have to be resolved. At the end of such a Concept Exploration Phase (denoted as Phase II), a more reliable engineering concept of a breeding blanket component will be available.

A final concept confirmation (proof of principle) of the selected designs needs, however, the testing under real fusion neutron irradiation in a follow-on development Phase III. This very late confirmation of a concept reveals *the weak point* in the presently adapted R&D strategy. It is caused by the lack of an appropriate high-intensity, high energetic neutron source and relies on existing data from irradiations in fission reactors and other simulation test beds assuming that they can easily be extrapolated to fusion reactor conditions. To what extent this assumption is solid will be addressed in more detail under Section 5.

The present material programmes include, as mentioned above, also other promising alternatives like vanadium-based alloys, ceramic composites of type SiC/SiC and W–Re alloys in many breeding blanket designs. It is, however, unlikely that all these options can or should be developed in the same depth and parallel to the above 'reference solution' with ferritic–martensitic steel. This is not only because of limited manpower and resources all over the world, but mainly because of potential feasibility issues which already have been identified in Section 3. A *selective development strategy*, in which potential key issues are resolved with priority before a broad development programme is launched, could for example be applied for the qualification of vanadium alloys. For this material group the development of stable, protective and insulating coatings, which mitigate MHD effects and prevent detrimental tritium and interstitial (C,O,N) pick-up, is of utmost importance. For SiC/SiC the performance of basic studies to reduce the sensitivity to radiation damage and the broadening of the technology base for production is another example for such a selective and efficient approach.

The *consolidation of design proposals* by an intense collaboration between designers and materials scientists can also positively influence the general strategy for materials development. Whereas designers primarily 'select' the materials to achieve optimised performance targets like high wall loading and thermal efficiency or extended lifetime without knowing their real critical issues, materials scientists are sometimes too fixed to a certain material group with which they are familiar. Under these circumstances sometimes the impression prevails that the materials community is not flexible enough or too conservative to follow new and 'creative' design proposals with new or even exotic materials. From what has been explained above it is clear that only a very limited number of materials can be studied in detail. Therefore, an intensified collaboration between designers and materials community is of utmost importance. To give an example on the flexibility between design and structural materials choice, in Table 4 the achievable averaged neutron wall loads and thermal efficiencies of several breeding blanket concepts, all using ferritic–martensitic steels as structural material, are compiled. Whereas initial breeding blanket concepts with lithium ceramics like the Japanese SSTR [9] or the European HCPB [10] blankets have a relatively moderate thermal efficiency of about 35%, the development of a dispersion-strengthened ODS-ferritic alloys with an improved creep rupture strength would permit increasing the upper temperature from 550°C to 650°C. This would increase the thermal efficiency for such improved concepts as the I-HCPB to 40% at an outlet temperature of 650°C [11]. Even higher temperature might be achieved, if creep strength could be further increased, making it competitive with alternative breeding blanket concepts which operate at much higher temperature levels. Similarly, for the dual-cooled lead-lithium concepts like DCL and ARIES, design studies show [12–14] that the use of a SiC liner in the ferritic coolant channels could increase the thermal efficiency by more than 10%,

Table 4
Breeding blanket concepts and influence of materials choice on expected performance data[a]

| Blanket concept | Coolant/breeder/structural and neutron-multiplier materials | Av. neutron wall load[b] ($MW/m^2$) | Thermal efficiency (%) | Ref. |
|---|---|---|---|---|
| SSTR-JA | H2O/LiCe/*RA-FM*/Be | 2–3 | 34.5 | [9] |
| HCPB-EU | He/LiCe/*RA-FM*/Be | 3 | 35 | [10] |
| I – HCPB-EU | He/LiCe/*RA-ODS-FM*/Be | 4 | 40 | [11] |
| DCL-EU | Pb-17Li/He/*RA-FM* | 3 | 34 | [12] |
| ARIES-ST-USA | PB-17Li/He/*RA-FM + SiC Liner* | 4 | 46 | [13] |
| A-DCL-EU | PB-17Li/He/*RA-ODS-FM + SiC Liner* | 6 | 46 | [14] |
| EVOLVE-USA | Li-evaporate/Li/*W–Re* | 10 | 60 | [15] |

[a] HCPB – Helium-cooled pebble bed blanket; DCL – Dual-cooled lead–lithium blanket; EVOLVE – Evaporation of Li and vapour extraction blanket; LiCe – Lithium ceramic.
[b] The surface heat load is in the range of 8–25% of the neutron wall load.

without the necessity of replacing the ferritic steel as a structural material. The function of SiC could be restricted to thermal insulation without any structural duty. Both examples show the merits of an interactive collaboration between the materials community and designers for consolidating new concepts.

For comparative reasons the corresponding data of an innovative design EVOLVE [15] with tungsten as a structural material and liquid lithium as the coolant/breeding medium are added.

## 5. The role of an intense neutron source for materials research

*The critical issue* in material development for fusion technology is the irradiation behaviour under fusion-specific conditions. Since no appropriate 14 MeV neutron source exists at present, irradiation performance is mainly studied in irradiation facilities like fission reactors and ion accelerators. It is, however, recognised that with such 'simulation tests' either important physical damage parameters are only partially adaptable, or technical limits exist which do not allow the accumulation of radiation damage to levels necessary to test materials for high performance application. To illustrate this, in Fig. 2 the achievable He-, dpa- and relative He/dpa-rates in Fe are plotted for different fission reactors (Material Test and Fast Breeder Reactors MTR FBR) and other facilities like the RTNS II and LAMPF. For all fission reactors the relevant He/dpa rate is about one order of magnitude lower than expected in a typical fusion device (DEMO). The situation is similar for other important inelastic transmutation reactions of interest like the formation of hydrogen by (n,p)-reactions. They generally increase with neutron energy, and deviations are the largest for light elements. Attempts to adapt the relation of inelastic transmutation rates to the displacement rate, e.g., by direct implantation of additional

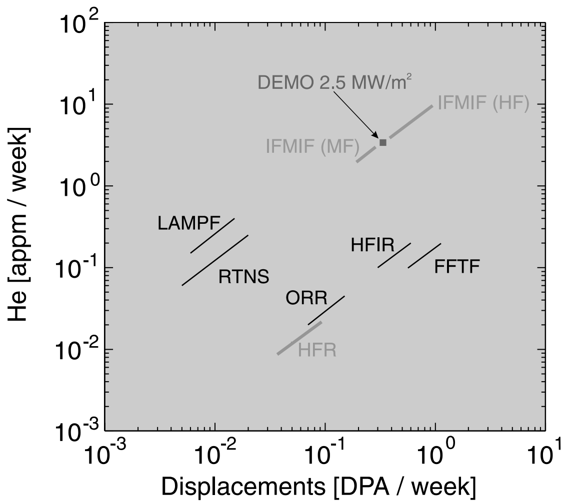


Fig. 2. Helium – and displacement rates for Fe achievable in different irradiation facilities in relation to DEMO with an average wall load of 2.5 $MW/m^2$.

helium or by adding elements which possess high (n,$\alpha$)-cross-sections in fission reactors have also their limits. In addition in most MTRs the actual achievable dpa-rates are too low to reach DEMO- or reactor-relevant dpa levels in due time. FBR's on the other side have limited possibilities for fully instrumented rigs and are in most cases restricted to the temperature window dictated by the cooling inlet and outlet temperature.

Since an adequate DT fusion environment for the qualification of materials cannot be experimentally achieved at this time, the use of theoretical models is critical to developing an understanding of how materials will respond to this unique environment. Unfortunately, the complete range of well-developed, multiscale models required for a fully predictive capability is not yet available. Simulation of primary damage formation is very robust. Extensive investigations of displacement cascade evolution have been carried out using molecular dynamics, and cascade aging studies using Monte Carlo methods are similarly well advanced. Only a limited number of materials have been investigated to date, but the techniques are well established. Kinetic models, such as the well-known rate theory, have achieved a reasonable level of success at describing radiation damage phenomena, such as microstructural evolution, which occur over longer times. However, these models are currently somewhat limited, and are hampered by uncertainties in material parameters. Thus, model calibration requires the use of experimental data that can only be obtained by post-irradiation microstructural analyses. The semi-empirical models used to correlate observed microstructural and mechanical property changes also need further verification, while more basic models of defect–dislocation interaction are showing promise. Although good progress is being made in the area of modelling, the need to validate existing data from simulation experiments and extrapolate such results to the D-T fusion reactor operating conditions make a powerful test bed for fusion materials studies indispensable.

Many concepts for such a facility have been proposed all over the world, but only the rotating target neutron source (RTNS) was realised in the USA. This D-T 14 MeV neutron source was very useful for investigating fundamental radiation damage processes and matched the relevant damage parameters like the He/dpa relation as can be seen in Fig. 2. However, as can also be deduced, RTNS was limited by its low flux intensity. The history of international activities to develop an appropriate intense neutron source is shown in Fig. 3. Plans to realise a D-Li stripping neutron source, called FMIT, were unfortunately cancelled in USA in 1984. This project had already reached a significant level of technical maturity, just at a time, when the Cottrell blue ribbon panel [16] had strongly recommended the construction of a neutron source. After a further review on fusion materials R&D by the Amelincks Senior Advisory Committee [17] a new approach through the International Energy Agency was started in 1989. Again this effort compared all relevant concepts like accelerator-driven spallation and stripping sources, as well as D-T-beam plasma and gas dynamic trap concepts. A thorough and comparative analysis of feasibility and suitability issues finally resulted in the recommendation that an accelerator-based D-Li stripping source is the most advanced and most suitable alternative. In a conceptual design activity in 1995/1996 and a follow-on conceptual design evaluation phase from 1997 to 1999, essential features of such a facility, named the International Fusion Materials Irradiation Facility (IFMIF), including construction and maintenance costs, were elaborated. In this concept which is described elsewhere in more detail [18,19], two parallel operating 125 mA deuteron beams of 35–40 MeV are focused onto a common liquid lithium target and produce neutrons at high intensity via a stripping reaction with a suitable energy spectrum peaking at around 14 MeV. This facility can fulfil essential users requirements; for structural materials the physically based damage parameters like dpa, transmutations and PKA-spectra reasonably well approach the fusion environment. This is demonstrated for example in Fig. 2 and Table 5, which show that based on three-dimensional MCNP code calculations, DEMO and reactor relevant conditions regarding neutron flux, dpa-, hydrogen-, and helium-rates can be achieved in the high flux test module (HFTM) of IFMIF. In addition accelerated testing in a limited irradiation volume is also possible. IFMIF also has – with the given test volumes in the high, medium and low flux test zones – a sufficient capacity to perform necessary types of experiments for structural, breeding and other materials in the appropriate temperature-, flux- and fluence-regimes. Such extended matrices can for the limited irradiation volume of about half a liter in the high flux test module be investigated only if miniaturized test specimens are used. For example by using a small specimen test technique (SSTT) about 1400 specimen including about 320 specimen for fatigue and fracture toughness investigations can be placed in the HFTM for one common irradiation. Fig. 4 shows a total of 27 separately instrumentable sub-capsules in which different types of miniaturized specimen can be arranged for well controlled experiments. Recommendations for miniaturized specimens to be used in mechanical tests have already been elaborated [18,20]. The small specimen test technology is, therefore, a necessary test technique for IFMIF and has to be qualified to ensure that the results can be accepted for engineering design and licensing procedures.

In comparison to many other proposals this facility is based on proven technology with very moderate

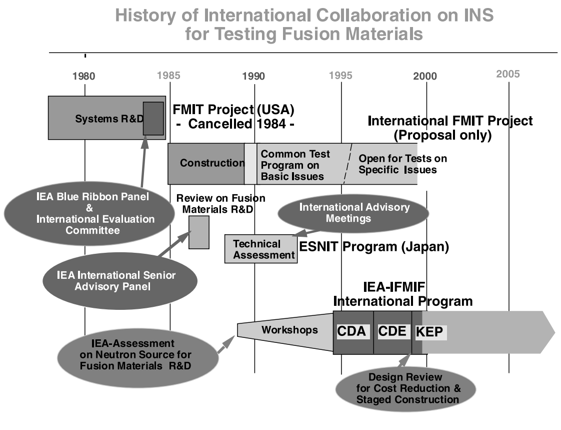


Fig. 3. History of international activities for the development of an intense neutron source for testing fusion materials.

Table 5
IFMIF High flux test module – key values[a] (3D MCNP code calculations based on: (i) Collided neutrons in Fe; (ii) Extended nuclear data libraries; (iii) Detailed 3D geometrical models)

| Irradiation parameter | | ITER[b] | DEMO[b] | IFMIF[c] |
|---|---|---|---|---|
| Total neutron flux | (n/s $cm^2$) | $4\times10^{14}$ | $7.1\times10^{14}$ | $4\times10^{14}$–$10^{15}$ |
| Hydrogen production | (appm/FPY) | 445 | 780 | 1000–2500 |
| Helium production | (appm/FPY) | 114 | 198 | 250–600 |
| Damage production | (dpa/FPY) | 10 | 19 | 20–55 |
| H/dpa ratio | (appm/dpa) | 44.5 | 41 | 35–50 |
| He/dpa heating | ($W/cm^3$) | 11.4 | 10.4 | 9.5–12.5 |
| Nuclear heating | ($W/cm^3$) | 10 | 22 | 30–55 |
| Wall load | ([$MW/m^2$) | 1.0 | 2.2 | 3–8 |

[a] Correct scaling of H, He and dpa production; accelerated irradiation in limited volume.
[b] Outbard blankets.
[c] Dependent on the position in the HFTM.

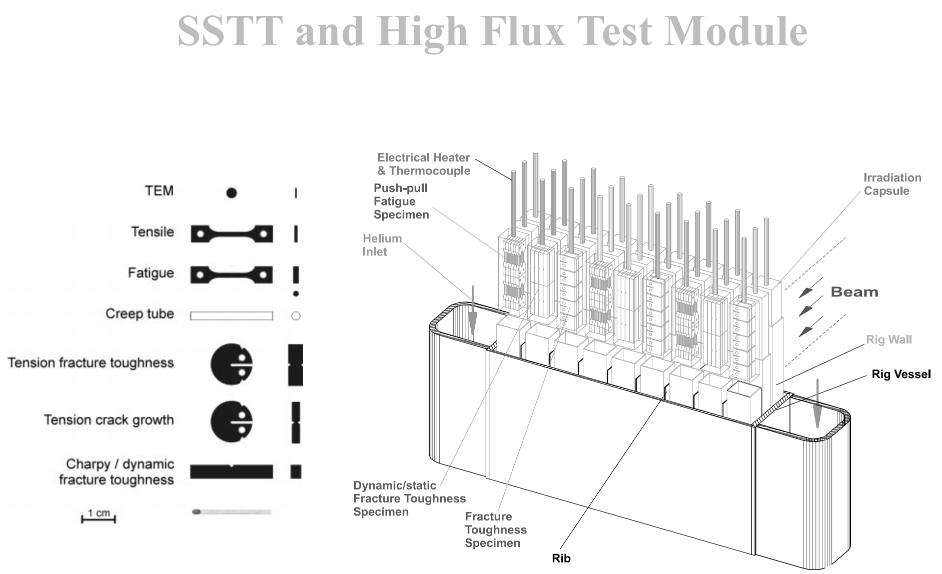


Fig. 4. The high-flux test module of IFMIF consists of 27 instrumented capsules which can independently be operated over a wide range of temperature (RT – 950°C).

extrapolations and could hence be designed and constructed in a foreseeable time. A technically realistic time schedule for its development is sketched in Fig. 5 provided that positive decisions for its realization come soon. In the initial conceptual design activity (CDA) proposal of 1995–1996 a reference design concept was elaborated that covered all the design criteria required to fulfil the stated mission guided by the users. According to Fig. 5 the engineering design and construction of this reference design with two accelerator lines (250 mA) and an option for upgrading ($4\times125$ mA and increased irradiation volume) could technically be finished within about 8 yr. The total cost estimate for this project was 797.2 million $ (values January 1996) [21].

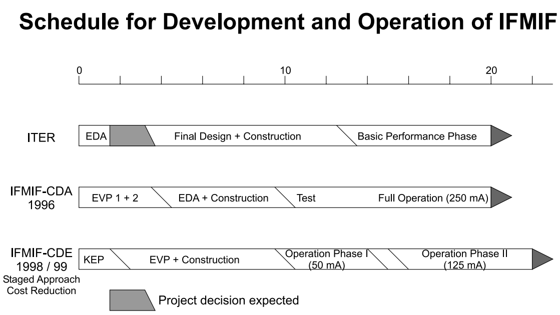


Fig. 5. Comparative schedules for development and operation of IFMIF in the initial and staged approach.

In the following conceptual design evaluation phase (CDE) phase (from 1997–1998) a ‘self-critical’ evaluation for validating the practical feasibility of this reference concept was continued and yielded significant upgrading in the technology of some of the key elements [22].

In 1999 on request of the IEA-Fusion Power Coordinating Committee, the activities focused on the question of a cost reduction – keeping the full mission of the facility – and to find a possible alternative time path, a so-called staged approach with reduced annual financial load over a longer time period. One result of this most recent evaluation [23] is a substantial reduction of the total cost from about 800 down to 500 million \$. It has been achieved by the elimination of a previously planned facility upgrade to four accelerators and further reductions of some components (one instead of two Li targets) related to the initial reference solution. It should be said that these reductions do not endanger the full mission of the initial concept but could result in a reduced system reliability and availability.

The concept of a staged facility deployment consists of three stages, one accelerator with two operation phases at 50 and 125 mA (Stages 1 and 2) with reduced test capacity, and thereafter the installation of a second accelerator with another 125 mA to achieve the full test capability of $2 \times 125$ mA in Stage 3. The consequences of such a strategy on available test capacities and research activities are given in Figs. 5 and 6. In essence, the research activities which would allow an aggressive search for high performance materials have – in comparison with the initial planning – to be postponed by roughly 10 yr into Stage 3. Whereas in the initial Stage 1, questions like fusion–fission data correlation, fundamental radiation damage studies and the above discussed material concept confirmation for DEMO test breeding blankets could be addressed. On the other side the expanded schedule would reduce the annual investment costs, relieve the financial burden and give more time to solve some of the technical risks during the development of the facility.

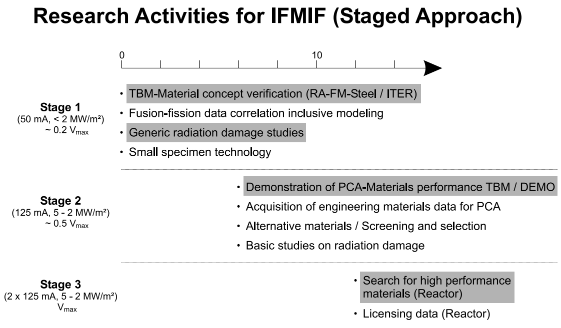


Fig. 6. The consequences of a staged approach on the research capacity and R&D activities.

Most important for the near future is the endorsement of the next step, the ‘Key Element Phase’ where key technical questions like the Li-target, the ion source of the accelerators, and specific irradiation and test modules are addressed in depth and a decision to initiate an Engineering Validation Phase for the construction of this facility.

## 6. Summary and conclusions

1. The strategy for the development of structural materials depends very much on design and concepts for first wall and breeding blanket components. One of the important performance targets is the integrated wall loading to be expected. For the European test blanket modules 70 dpa is envisaged, which is well in accordance with a range of 80–100 dpa in other programmes. The goals for prototype or commercial fusion reactors are less defined. A reasonable intermediate target lies in the range of 150 dpa.
2. Structural alloys for combined first wall/breeding blankets are also dependent upon the appropriate choice of breeding-, coolant- and neutron-multiplying materials. An assessment of different combinations leads to four major categories with three groups of structural materials: Ferritic–martensitic steels, vanadium alloys and ceramic SiC/SiC composites.
3. A mission-oriented R&D schedule for development and qualification of DEMO-relevant test blanket modules with a ferritic–martensitic steel as a ‘reference’ structural material is proposed. It describes the major activities and development phases.
4. A more selective strategy is proposed for the development of alternatives like vanadium alloys and ceramic composites of SiC/SiC-type. In a first phase, R&D work should be concentrated on identified high-risk issues, whereas a comprehensive qualification programme should be started after elimination of possible knockout factors.
5. There is an important role of existing irradiation facilities (mainly fission reactors) for the next R&D-phase and their availability is mandatory for experiments in the next decade. The limits for the transfer of results from such simulation experiments to fusion-relevant conditions have also been addressed.
6. The importance of a fusion-relevant, high-energy and high-flux neutron source for the development and qualification of materials for DEMO and reactor-relevant fluence conditions has been shown. Such a facility would also provide an opportunity to validate data generated in fission reactor and accelerator irradiations. The materials community believes that an accelerator-driven D-Li-neutron source, denoted IFMIF, can – from its technical capabilities – fulfil

the user requirements and can provide a useful test bed for material screening and selection up to reactor-relevant wall loads in a reasonable time. A prerequisite is, however, that a small specimen test technology be developed and approved in parallel. IFMIF is presently the only option which can be realised in due time.

7. The demands for a cost reduction of IFMIF and for a staged facility deployment have been investigated. One result is a substantial cut of construction cost from 800 to about 500 million $ by keeping the full mission of the initial reference concept, but elimination of a previously planned upgrade and partial reduction of the system reliability. A staged approach would reduce the annual investment cost and the risks of the technical development during construction. The consequences for the material development strategy are a general delay of research activities, especially exploratory activities for high performance materials.
8. The endorsement of the next phase for IFMIF, the so-called 'Key Element Phase' and a decision to initiate the Engineering Validation Phase for the construction of this facility are of utmost importance for the materials development.
9. The execution of the materials development programme and the installation of appropriate irradiation/test facilities enforces a close international collaboration, which is at present promoted by the International Energy Agency for the materials and nuclear technology areas.

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