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Materials R&D for a timely DEMO: Key findings and recommendations of the EU Roadmap Materials Assessment Group

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ABSTRACT

The findings of the EU Fusion Programme's 'Materials Assessment Group' (MAG), assessing readiness of Structural, Plasma Facing (PF) and High Heat Flux (HHF) materials for DEMO, are discussed. These are incorporated into the EU Fusion Power Roadmap [1], with a decision to construct DEMO in the early 2030s.

The methodology uses project-based and systems-engineering approaches, the concept of Technology Readiness Levels, and considers lessons learned from Fission reactor material development. 'Baseline' materials are identified for each DEMO role, and the DEMO mission risks analysed from the known limitations, or unknown properties, associated with each baseline material. R&D programmes to address these risks are developed. The DEMO assessed has a phase I with a 'starter blanket': the blanket must withstand $\geq$2 MW yr m^{-2} fusion neutron flux (equivalent to ~20 dpa front-wall steel damage). The baseline materials all have significant associated risks, so development of 'Risk Mitigation Materials' (RMM) is recommended. The R&D programme has parallel development of the baseline and RMM, up to 'down-selection' points to align with decisions on the DEMO blanket and divertor engineering definition. ITER licensing experience is used to refine the issues for materials nuclear testing, and arguments are developed to optimise scope of materials tests with fusion neutron ('14 MeV') spectra before DEMO design finalisation. Some 14 MeV testing is still essential, and the Roadmap requires deployment of a $\geq$30 dpa (steels) testing capability by 2026. Programme optimisation by the pre-testing with fission neutrons on isotopically- or chemically-doped steels and with ion-beams is discussed along with the minimum 14 MeV testing programme, and the key role which fundamental and mission-oriented modelling can play in orienting the research.

1. Introduction

Within current fusion power development programmes, the concept of a Demonstration Fusion Power Reactor (DEMO), as the last step before the deployment of true commercial Fusion

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Power Plants, is widely supported. A key objective of the EU fusion Roadmap for Horizon 2020 [1] is to lay the foundation of a DEMO to follow ITER, foreseen as a device capable of generating several hundred MW of nett electricity to the grid and operating with a closed fuel-cycle by 2050.

Many technologies must be mastered to realise a DEMO design. ITER is foreseen as the major 'test-bed' step for DEMO, but there will need to be parallel development programmes for a number of technologies, as ITER will not reach several of the required testing conditions, the principal short-fall being in the total fluence of high-energy neutrons produced in the relatively-low duty cycle of ITER high performance plasma burn periods. Thus in-vessel irradiation of materials used in constructing the tokamak first-wall and plasma-facing components will require a testing programme outside ITER of some complexity. The performance of the in-vessel materials is centrally-important to the success of the DEMO mission. Suitably robust and durable materials determine not only the fundamental 'existence' of a workable DEMO design, but also play a key role in demonstrating the economics of fusion. The latter comes from the lifetime of components, which strongly affects productive operating duty cycle, and from the materials' temperature operating ranges, which affect the plant's thermodynamic efficiency.

As part of the EU Fusion Roadmap process, a 'Materials Assessment Group' (MAG) was established: to review the structural, high-heat flux (HHF) and plasma facing (PF) materials development programmes for a fusion power reactor; to identify the major gaps in knowledge; to establish a coherent strategy and road map; and to define a resource-loaded plan. The MAG findings are contained in a report [2], and have been incorporated into the Roadmap document. This paper reviews the MAG methodologies, key findings, proposed development plans and recommendations.

2. DEMO concept used in the assessment

2.1. Characteristics of the EU Roadmap DEMO

The Roadmap foresees an 'early' decision to construct at DEMO machine in the early 2030s [1], following the important results from the ITER D-T programme and particularly the ITER Test Blanket Module (TBM) programme. This near-term DEMO would probably be a long-pulse (several hours) device, with the realistic extrapolation from ITER parameters consistent with the goal of several hundred MW of electricity generation to the grid and a fully-closed tritium fuel cycle. The system code studies fix the near-term DEMO as a relatively large machine (major radius ~ 9 m), with relatively low plasma normalized pressure ($\beta_N \sim 2.5$) and a fusion power output ~ 1.8 GW_{th}. The details of such a machine are given in a companion paper [3].

2.2. Blanket and divertor philosophy

The Roadmap DEMO is foreseen as having two phases of test programme operation. In 'Phase 1′ the breeding blanket and divertor concepts will be refined with the possibility of installing modified versions in the subsequent 'Phase 2′, where the final, reactor, concepts will be proven. Thus between the phases it is envisaged that the technologies of the Breeding Blanket (BB) and divertor might change, including possibly the structural and functional materials. The only constraint imposed on the changes would be that the *coolants* for the BB and divertor could not be changed between phases. This constraint arises from the need to avoid changing the *Balance of Plant* (BoP), as such a change would be unacceptably high in cost, both in capital and, equally importantly in a 'fast track' approach, in delays to completion of the programme.

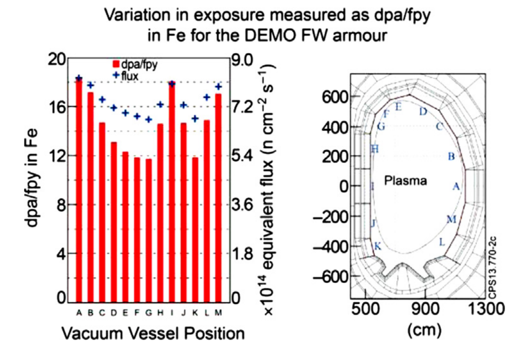

Fig. 1. Poloidal variation of neutron flux and damage in breeding blanket first-wall iron for a 2.7 GW_{th} fusion reactor of similar size to the Roadmap DEMO (data from [4,5]).

The concept of a 'starter' blanket and divertor has been developed, to define DEMO Phase 1 goals. This will be discussed further below, but in brief, the DEMO starter BB should have a first wall structure capable of withstanding ≥2 MW yr m^{-2} fusion neutron fluence (equivalent to ~20 dpa front-wall steel damage). The starter divertor materials should match this in terms of their own neutron flux at the front face, and withstand the accompanying erosion damage from the plasma's scrape-off layer particle flux. For Phase 2, the BB first wall structure should withstand ≥50 dpa steel damage.

2.3. Neutron damage levels in DEMO

The levels of neutron flux and material damage in DEMO and a fusion reactor have been simulated in many papers. For the MAG exercise, we took recent simulations from Gilbert et al. [4,5] using the most up-to-date cross-sections for the evaluation of damage levels and helium/hydrogen transmutation production. The simulations use a model 2.7 GW_{th} fusion reactor, similar to the concept 'Model B' plant from the EU Power Plant Conceptual Studies (PPCS) work of 2005 [6] with a helium-cooled 'Pebble Bed' (lithium salt and beryllium multiplier) BB and a tungsten-surfaced divertor. The simulation (shown in Fig. 1) shows poloidal variation of ±20% in the neutron flux through the first wall surface, with a peak of $\sim 8.25 \times 10^{18}$ n m^{-2} s^{-1}, corresponding to damage ~ 18 displacements per atom per full power year of operation (dpa/fpy) in a first wall steel. Scaling this down to the DEMO concept of Section 2.1 (1.8 GW_{th}) we find a damage level of ~ 12 dpa/fpy in the first wall steel of the DEMO BB. The simulations [4] also show a production ratio for transmutation helium of ~12 appm/dpa.

3. Methodologies in the assessment

Focussing on a near-term DEMO concept motivates the use of a *Project-based approach* to the development of materials, with evaluation of risks and mitigating strategies, and linkage to decision points in the concept, system and detailed design processes of the relevant in-vessel systems. Additionally we adopt a *systems engineering approach* in order to: concretise the problems of constructing the DEMO sub-systems as assemblies of materials; highlight trade-offs and constraints; and prioritise R&D needs. This mirrors the systems engineering approach adopted in the EU DEMO design process [3,6]. The MAG has used the concept of Technology Readiness Levels (TRLs) applied in an approximate manner to the relevant materials. This has already been applied to fusion materials elsewhere [7].

3.1. Elements of the project-oriented approach

For each of the materials categories (Structural, HHF and PF), the likely loading conditions in service were identified. Then a survey was made of current R&D knowledge, including materials development *per se*, but also joining and fabrication technologies, mechanical and thermo-mechanical properties, interaction with relevant coolants, and design code status. Particular attention was paid to irradiation effects, and also to the performance requirements for the in-vessel systems linkages to other sub-systems of DEMO, especially BoP. From the available data, *Baseline Material(s)* was (were) identified for the immediate pursuance of the DEMO concept design. These materials were selected as the best available which had progressed at least beyond TRL3 (roughly 'proof of concept', with basic properties and performance known over the relevant operational range, joining techniques investigated, coolant compatibility demonstrated and irradiation effects measured), and had at least some of the elements of TRL4-5 demonstrated (roughly 'relevant multi-effects in integrated environments', with joining techniques demonstrated, large quantity fabrication, prototypes built and operated in simulated integrated environment, data and models existing on irradiation and coolant corrosion effects, and relevant nuclear use codes in existence) [7].

For the Baseline Material a *Risk Log* was developed listing risks to the DEMO mission from the known limitations, or unknown properties of the material. Risks were categorized in the usual format from 'Low' to 'Very High', using the definition of Risk Level = Impact × Likelihood. For each risk above 'Moderate' level a *Mitigation Strategy* was identified. The mitigations could be in the form of a constraint on the design, an operational 'work-around', more basic R&D, or the development of a new material. Risks were evaluated post-mitigation on the assumption of a success. If residual risks with the Baseline Material still included risks above 'Moderate', the evaluation moved to the identification of the development of substitute materials, categorized as *Risk Mitigation Materials (RMM)*.

The inclusion of a RMM into the development programme was restricted to those passing the threshold of at least TRL1, i.e. applied research on the material existed in basic form, and conceptual studies of the application of the material had begun. This process can be recursive, in that risks exist in the adoption of RMMs, themselves, but to avoid an analysis, with nested Risk Logs, the MAG fixed on the known *Issues* with particular RMMs, and, for RMMs which had progressed to the TRL3 stage, established '*Issues tables*'. Addressing these issues, added to the other risk mitigation measures aimed at improving the baseline, then formed the R&D programme for a material category. These programmes were developed with the RMMs being researched alongside the improved Baseline, until appropriate down-selection points are reached, at which, based on the stage of the DEMO in-vessel components' design, the Baseline Material would be confirmed as sufficiently low risk, or, alternatively, the RMM would replace the Baseline Material if the latter remained too risky, or the RMM represented a quantifiable improvement. After these selection points the balance of R&D resources would thus be altered. If the programme resources were sufficient, and promise looked attractive, a non-selected RMM could still continue in development under the Roadmap's 'Advanced Concepts' programme [1], but without the near-term DEMO project time constraints.

For the HHF and PF categories of materials, the existing RMMs are at a very early stage of development (TRL1-2). At these early stages it is not possible to establish accurate Issues Tables, so for these RMMs a generic materials development programme was identified, based on timely achievement of the TRL4-5 level.

3.2. Lessons from fission material development

DEMO environments for key components, such as blanket and divertor, will be markedly different from those in fission reactors, but there are key aspects from fission materials developments that may contribute to defining a DEMO R&D materials programme and provide some experience to optimise development. In particular, fission programmes include development of materials used in an irradiation environment where mechanical and dimensional properties may suffer significant degradation. The implications of fission practice to materials R&D issues relevant to the design and licensing of DEMO and the lessons learnt were examined in the MAG process.

Fission experience underlines that the development and selection of structural materials for plants and components operating in irradiated environments requires *inter alia* that: materials must be suitable and commercially available; a trusted supply chain for component design and manufacture must exist; joining techniques, especially welding, must be developed under rigorous scrutiny of regulators and operators; design, including repair and replacement must be covered by design codes, with a 'Data Handbook' available at an early stage; as components must demonstrably retain function and integrity during their design life, degradation mechanism and failure probabilities (including stress-corrosion cracking and coolant interface effects) must be assessed, with accurate operational conditions modelling and experimental validation. Achieving these requirements takes considerable time, and for the 'Generation IV' advanced fission systems, new materials developments are estimated as 10–15 years duration [8], to take a material from TRL1 to the threshold of reactor deployment. Thus any fusion materials required for an early-2030s DEMO decision should essentially already be at least at TRL1-2 level. No completely new developments are relevant on the Roadmap's timescale.

Important specific lessons come from *Regulatory Factors* controlling materials selection and development approaches, and *In-service inspection*.

3.3. Regulatory factors in materials development

There are no established fusion design codes or practices to assure nuclear safety and a consistent engineering approach, but fission experience shows that greatest weight will be given to real plant experience and precedence. For ITER the French licensing authorities have accepted Safety Case proposals as the essential safety arguments have been met. These are based largely on extension of established methodologies such as passive-safety containment and defence in depth, developed within the nuclear industry [9]. The likely regulatory perspective of the DEMO safety case should be established at an early stage, and should include as much use of ITER precedent as possible.

The 'Fusion Materials Data Handbook' available in the early stages of the design process will not include data for true 'fusion neutron spectrum' irradiations (see Section 6), and the question arises regarding the use of 'degradation formulae' as existed in the early fission programme. The relevant example involves degradation of reactor pressure vessels (RPVs) from fission light water reactors (LWRs) and the prediction of in-service and end-of-life properties of RPVs in LWRs. The RPV is the main safety-related component in an LWR, representing the primary containment barrier against design basis accidents (DBAs), and is thus equivalent to the vacuum vessel in ITER. For the early LWRs, standard irradiation degradation factors were allowed in design substantiation, but tests showed that embrittlement would be much higher, thus surveillance programmes began to monitor the change of RPV properties. This fission experience will prohibit use of such formulaic degradation factors at the Licensing Stage of DEMO.

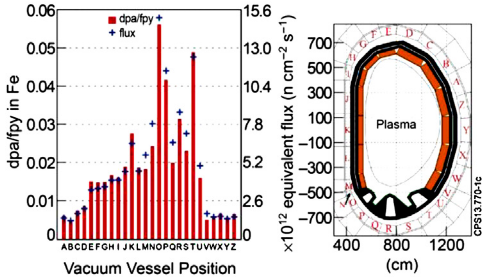

Fig. 2. Poloidal variation of neutron flux and damage in vacuum vessel iron for the 2.7 GW_{th} fusion reactor simulated in [4,5]).

It is probable that a decision will be required on any threshold in irradiation dose below which un-irradiated data can be used. In fission systems this can be very low in Safety Related (SR) components [10]. The high levels of first wall neutron damage in DEMO rule out the use of un-irradiated data for BB and divertor structures, but scope exists for reducing the requirement for fusion-specific (high-energy) neutron data. For ITER, the vacuum vessel (VV) has been established as the primary safety barrier. This is a natural boundary and can be easily connected to expansion volumes. Moreover in DEMO the VV is shielded by the BB and divertor structure. The simulations show [5] that the neutron flux to which it is subjected is very much reduced (by more than two orders of magnitude over most of the poloidal cross-section) and much softer (negligible fluence above 1 MeV), than at the in-vessel first wall. Its irradiation embrittlement should be low, with damage levels < 1–2 dpa, even after 30 full-power years' service (see Fig. 2), with negligible helium and hydrogen production by transmutation. Of course the detailed design of in-vessel components in the BB and shield will have to be implemented carefully to avoid neutron-streaming weak points, The MAG recommends that ITER practice in this respect is carried over to DEMO, with Safety Analysis to establish the primary safety boundary for DEMO at an early stage in design. With such low fluence, the DEMO VV could be constructed of similar material to ITER (Austenitic 316L) and the radioactive waste at end-of-life could still be lower than that in from a first wall blanket steel constructed of Reduced Activation steels developed for fusion.

3.3.1. In-service inspection and maintenance

Fission experience shows that in-service inspection (ISI), surveillance programmes and maintenance requirements provide the template needed to maintain the original or sufficient margins in nuclear plant and components and to return it to service in a safe manner following plant outages. Fission programmes have featured post-commissioning surveillance to build up the confidence in the design, and to help establish the (mechanistically) based dose damage relationships (DDRs) for a wide range of material and irradiation variables. The Safety Cases for early classes of fission reactors (e.g., the UK 'Magnox' reactors) relied on these DDRs, underpinned by mechanistic understanding and critical experimental data.

MAG therefore recommends that it is essential to develop at an early stage the remote ISI, material surveillance, maintenance and repair procedures to be used in-vessel on DEMO and to factor these into the design concept.

4. Structural materials

4.1. Loadings

The structural materials for the BB and the divertor substructure should provide stable and robust designs for the load-bearing surfaces and the coolant channels. The neutron irradiation load will vary strongly through the BB as the high energy flux is degraded through damage and transmutation, breeding and multiplication reactions. For the concept DEMO we take 15 dpa/fpy (~150 appm/fpy helium concentration) as the damage level at the front face steel (a 25% safety margin above the simulations of [4,5]), but the simulations show that the neutron flux will have dropped by an order of magnitude after traversing the first 0.25 m of a composite helium-cooled blanket (an even larger drop would occur for water-cooled concepts). As a result the radiation damage will drop to the equivalent of a few dpa, and the risks associated with any particular structural material will change through the blanket structure (the rear of the Divertor will be similarly shielded). For the 20 dpa lifetime of the concept DEMO's BB, this corresponds to $\geq$ 1.33 full power years of operation, and, with a 2–2.5 h pulse time [3], the number of fatigue cycles will be $\geq 6 \times 10^3$, rising to $\geq 15 \times 10^3$ in the Phase 2 concept. The mechanical loads on in-vessel components in a near-term DEMO concept will have a cyclic component typical of a pulsed device. Structural material will require lifetime against creep-rupture of $\geq 12 \times 10^3$ h at maximum stress and typical operating temperature for Phase 1 ($\geq 30 \times 10^3$ h for Phase 2).

4.2. Baseline structural steels

The MAG analysis confirms reduced activation ferritic martensitic (RAFM) steels the Baseline Material choice for the BB structure. The EU developed version is EUROFER [11]. This Fusion programme development has a good overall balance of mechanical properties required (strength ductility, fracture toughness, creep resistance, fatigue resistance). It has sufficient corrosion resistance to liquid (LiPb) metals in breeder concepts at the relevant low flow conditions and for interface temperatures <~475 °C, and sufficient compatibility with He-gas cooling, making it compatible with the He-cooled EU reference blanket concepts [3]. Its development is in the TRL4-5 range, and significant quantities have been fabricated industrially, welding developed and fission-irradiated properties measured to an extent. Very importantly, it has the relatively good stability of a body-centred cubic (bcc) latticed material under fission neutron irradiation, with very low swelling.

The Risk Log for EUROFER shows many issues however, the most serious, with very high impact on any DEMO design, relate to their limited temperature operating window, being suited to particular BB designs with FW temperature ranges between 350 and 550 °C [11]]. The most serious risk comes from low-temperature embrittlement under fission neutron irradiation. The exact temperature limits are uncertain because of the un-quantified added effect of the fusion-neutron produced helium embrittlement, but safe operation of blanket should only be guaranteed if irradiated above 325–350 °C operating temperature, as this avoids a shift in the ductile–brittle transition temperature (DBTT) to unusable levels (ie. avoids a DBTT ≈ 0 °C or above) [12,13].

Helium effects are not sufficiently understood due to the absence of an irradiation facility with a fusion-relevant neutron spectrum. The issues surrounding fusion spectrum testing are discussed in Section 6. Presently, surrogate simulation techniques exist such as fission reactor irradiation of ^{10}B-doped steels, or He ion-beam implantation data. The former may overestimate the helium embrittlement effects, as the production rate for helium is much accelerated compared to the displacement damage [13], whilst the latter are restricted to the surface region. As an example, data exists [14] showing that at ~30 dpa the properties begin to worsen as He concentration exceeds ~ 500 appm. This limit is only ~ 50% above the simulated He concentration from [5], and indicates the need for better data and understanding. There is evidence that EUROFER's irradiation damage can be annealed-out by post-irradiation high temperature (550 °C) cycling [15], but this could

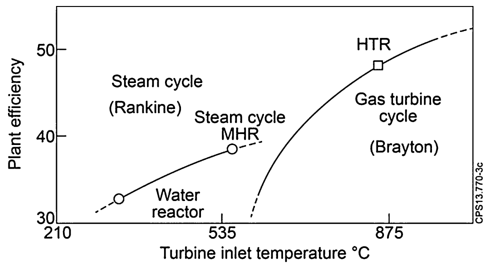

Fig. 3. Plant efficiency as a function of turbine inlet temperature (source Ref. [18]).

be difficult to achieve in a realistic design, and the effect on the mobility of transmutation helium is unknown.

In addition, high impact risks come from the decline in EUROFER's strength above the 500–550 °C range, where the creep-rupture failure also drops below 10^4 h, for stress levels of ~ 100 MPa similar to those proposed as maximum primary stress levels for fusion reactor developments of fission codes such as RCC-MRx [16].

The low-temperature embrittlement, coupled with the decline in strength gives a difficult, relatively narrow, temperature operating window for a Breeding Blanket and make it difficult to envisage a high-temperature coolant loop (with high thermal efficiency) with a EUROFER blanket as its 'front end'. This runs counter to the use of helium cooling, as the coolant gas cannot be used at a temperature range where the potentially higher thermodynamic efficiency of a Brayton cycle can be employed, or even reach the upper limits of a Rankine cycle, as shown in Fig. 3. Studies of BoP using the helium temperature limits arising from EUROFER indicate [17] that the thermodynamic efficiency of Rankine or Brayton cycles limited to < 480–500 °C is insufficient to reach the nett electricity generation target for the DEMO concept when the high installed pumping power for the helium circulation (~150 MW) is subtracted. This entails a further high-impact risk to the DEMO mission.

The overall risks to the DEMO mission from the use of helium-cooling technology, where circulators and compact heat exchangers for high-pressure helium are immature technologies [17], motivate a strategy of parallel blanket concept development [3]. Water-cooled systems have the attraction of mature, low-risk BoP options essentially similar to those of fission pressurised water reactors (PWR), and BoP studies [17] show that even simple systems come close to the required overall efficiency. EUROFER's embrittlement temperature makes its use problematic in a water-cooled blanket such as the water-cooled lithium lead concept, with 290–320 °C operating temperature. Indications exist that some melts of RAFM steels have had superior embrittlement properties around 300 °C, especially the EUROFER 97WB [19], and the Japanese F82Hmod3 [20,21]. For risk mitigation the MAG recommends a development programme to push RAFM to lower embrittlement temperature. As with all 8–9% Cr FM steels, corrosion under irradiation would be an issue if water were to be used in a BB, and coating and coolant chemistry mitigations will be required.

4.3. Risk mitigation—high temperature steels

The high mission-risks associated with the RAFM baseline, and the central importance of the blanket structure, make an active risk mitigation programme for the structural steels necessary, either as complete replacements for EUROFER, or to complement the use of EUROFER in zones of high irradiation. Two existing lines for RMM are recommended:

- adapting developments outside fusion for high-temperature ferritic–martensitic (HT FM) steels with improved high temperature creep strength (up to ~ 600–650 °C) achieved using thermo-mechanical treatment (TMT), or thermal treatment to improve the microstructure and density of radiation defect recombination centres; and
- pursuing the development of oxide dispersion strengthened (ODS) alloys, subject of fusion R&D for a decade, but still without industrialisation. These have good high temperature tensile strength and creep resistance.

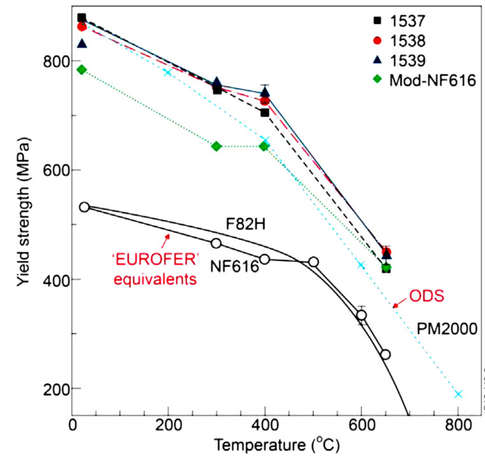

Fig. 4. Temperature variation of yield strength of laboratory heats of US HT RAFM steels (1537-1539, and Mod-NF616), compared to US developed ODS steel (PM2000) and US and Japanese RAFM steels (NF616 and F82H) (data from [24]).

The use of solution annealed 316L-(N) stainless steel as RMM for BB structural steel was rejected. It has been studied as a potential candidate blanket structural material for fusion reactors, but its power exhaust capability is limited and might be an issue for a DEMO BB. Crucially, these steels' resistance to radiation effects is limited. Their ductility and fracture toughness are severely degraded during irradiation around 300 °C and damage of ~10 dpa [22,23]. At mid-range temperatures of 400–500 °C they are susceptible to unacceptable volumetric void swelling for doses >20 dpa, and they have been shown to suffer from severe He-embrittlement at high temperature (>550 °C), in slow strain rate testing, for He contents above 10–100 appm (well below the ~ 240 appm DEMO 'Starter Blanket' FW conditions).

The chosen Structural RMM types have reached reasonable levels of development (TRL3-5), and their advantages and drawback issues can be identified. The composite Fig. 4 shows how the US developments along these lines have demonstrated much higher tensile strength than conventional RAFM variants (and their similar strength to ODS). The current 'industrial standard' FM steel in the power generation field is Type 92 (9Cr, 0.5Mo, 2W), a development of the Type 91 from which the EUROFER-type RAFM steels were developed. This has a creep-rupture performance of 10^5 h at 100 MPa and 620 °C, but this drops below 10^4 at 650 °C. Even development of this for reduced activation would yield benefits. The goal for DEMO Phase 1 BB is >1.2×10^4 h at temperature, and for a reactor the EU PPCS fusion reactor studies [25] identified a 5 year blanket lifetime as an economic goal (~5×10^4 h).

Amongst the new HT generation of TMT steels, the creep-rupture performance still needs proving beyond ~ 3×10^3 h at 650 °C[26], however *thermally treated* type 92 variants[27] have reached > 3×10^4 h at this temperature and 92 MPa[28], indicating the promise of these steels The TMT or thermally-modified FM steels have the drawbacks of:.

- very limited development of reduced activation variants [24,29];
- lack of fission irradiation data and data on helium transmutation embrittlement.

They are expected to exhibit low-temperature embrittlement problems, but their high density of nanoscale precipitates and microstructures are predicted to lead to superior performance to EUROFER on low temperature and helium embrittlement [21]. As 'classical' steels their industrial fabrication and welding development will be relatively straightforward, already reaching a mature stage

The ODS programmes intend to *complement* RAFM steels with a structural material aiming at both higher temperature and improved irradiation resistance, and with dispersed oxides intended to provide precipitate sites to 'fix' the helium gas bubbles generated by high-energy fusion neutrons thus preventing movement to grain boundaries and enhanced embrittlement. Although tubes and sheets have been successfully produced, ODS steels currently have the following drawbacks:

- the experimental batches produced typically have low fracture toughness at room temperature, implying further basic development, perhaps via impurity control;
- fabrication of components will be difficult as the current materials have anisotropic mechanical properties unless a complex TMT is followed;
- the quality of the experimental heats is highly variable.

Moreover, the steels are only available in small (kg size), laboratory made quantities, and the process for manufacture, with much powder metallurgy and intensive complex TMT, will have to be scaled-up and industrialised. Other serious issues for ODS are the lack of welding techniques available, and the poor knowledge of the non-irradiated engineering parameter database in the case of 12–14Cr versions of these steels.

The ODS steels also share the low-temperature radiation embrittlement problem with EUROFER. There are early experimental indications however, that ODS steels exhibit less radiation-induced hardening than conventional RAFM steels [30], as oxide dispersion is increasing effective point defects sinks. Thus less severe low-temperature radiation embrittlement is expected. This more hopeful picture, expected also with the HT FM steels, is attributed to the higher density of precipitate nanoclusters, improving the dislocation density and providing sinks for radiation defects [21]. This effect is already seen in the improved radiation hardening for conventional RAFM steels, when compared to the coarser-grained RPV (bainitic) steels in the range up to a few dpa, and the accompanying lower change in DBTT as radiation hardening takes place [31].

The MAG recommends that the development of Reduced Activation HT FM steels be pursued in close co-operation with industry, as common goals exist in the development and industrialisation of these steels. The highest priorities for this development are the production of reduced activation versions of the TMT and thermally-treated steels, and obtaining data on the irradiated properties of the existing varieties. For ODS it is also of crucial importance to engage with industry to discuss the industrialisation of production processes, and establish robust welding techniques. This effort needs to be pursued in parallel with the more fundamental experimental and modelling research, as there are considerable hurdles to overcome.

5. HHF/PFC materials

5.1. Loadings

The loading for DEMO plasma facing (PF) materials is severe, especially in the Divertor. Analysis of DEMO options reveals a complex situation [3], and power loadings on the Divertor can be as high as 50 MW m^{-2} in the absence of shielding radiation. The highest fraction of plasma power radiated, achieved by impurity seeding, is ~90% [32]. Erosion by plasma ions is severe in the Divertor region, where the fluxes rise by more than two orders of magnitude to >5 × 10^{22} m^{-2} s^{-1} [33]. With the impurity seeded plasmas, moreover, the erosion is dominated by introduced impurities, as discussed below. We have taken a power loading of ≤20 MW m^{-2} as a guideline for the Divertor. This requires some thickness of armour to avoid 'punch-through' erosion of the PF surface, and hence for the Divertor, the PF armour must also have a high thermal conductivity, essentially acting as a HHF material too. The loadings for the first wall armour are much lower, ~1 MW m^{-2} –5 MW m^{-2} on the basis of ITER estimates [34]. The neutronics simulations [4,5] scaled to the DEMO concept power level show that the neutron damage level in the DEMO PFC tungsten will be ≤ 9 dpa/fpy for first wall armour, and ≤6–7 dpa/fpy for the divertor, whilst for tungsten in the shielded plasma 'strike-zones' of the divertor structural tungsten the requirement may be as low as 2–3 dpa/fpy. The helium transmutation is much lower than in steel, with predicted levels ≤4 appm/fpy in the first wall, and ≤2 appm/fpy in the divertor armour. As with the steel values, we have taken a 25% margin. This is particularly important for tungsten, as the recent cross sections used in the simulations [35,36] have led to an increase ~ threefold in the tungsten damage predictions [5].

5.2. Selection of water-cooled divertor as baseline

As indicated in [3], the choice of concepts for Divertors lies between low-temperature (100–200 °C coolant) water-cooled concepts, of the type used for ITER [34], and high-temperature (600 °C coolant) helium-cooled concepts [37]. The latter are at a much lower level of development (< TRL3), and limited to <10 MW m^{-2}, whereas the former are at ~TRL4, and tested to ~20 MW m^{-2} [34]. In general water-cooled systems can reach much higher heat transfer values than gas-cooled systems, and hence the near-term concept DEMO selects a water-cooled Divertor for the Baseline [1,3].

5.3. Baseline PFC and HHF materials

Tungsten is regarded as the baseline material for state-of-the-art plasma-facing component technology, and this is confirmed by the MAG assessment. The key advantages of tungsten are: the high threshold energy for sputtering, around 100–200 eV by hydrogen isotopes; and the low retention of tritium in the material, important both for the feasibility of the D-T fuel cycle and the Safety Case for fusion reactors, which, on ITER precedent, is keenly concerned with the inventory of mobilisable tritium in case of any loss of vacuum accident (LOVA). The highest risks relating to tungsten as a PFC material are:

- uncertainty in the erosion effect of neutron irradiated tungsten materials (effects of neutron-induced defects);
- uncertainty in the H isotope retention behavior following neutron irradiation.

For both these effects there is an urgent need for plasma stream experiments on neutron-irradiated tungsten.

Other high impact risks are the intrinsic brittleness of tungsten, the lack of radiation embrittlement data and the un-irradiated materials database which has some serious shortcomings (e.g. thermal fatigue data) if tungsten were used structurally, as opposed to PFC armour. R&D programmes must resolve these last two situations and engineering workarounds are required to address the first. For tungsten used as a Blanket PFC covering, there is an urgent need to develop self-passivating tungsten alloys to guard against oxidation/deflagration in an up-to-air accident scenario—note that

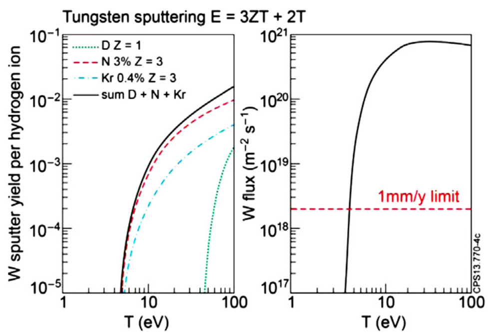

Fig. 5. Effective sputtering yields for tungsten as a function of temperature for different species, assuming $T_e = T_i$. The black curve gives the total yield for the species mix indicated. The right picture shows the corresponding tungsten influx for the left graph conditions. A 1 mm/fpy DEMO divertor erosion limit is shown corresponding to divertor target temperature ≤5 eV.

although mitigating this risk will be necessary for DEMO, licensing, it may not feature as a problem in ITER because of the much lower level of transmutation products, with the resultant lower nuclear afterheat.

Serious risks for tungsten come from the damage by plasma edge instabilities ('ELMs') where energy densities ~GW m^{-2} impinge on the divertor surface for ~ms periods. The damage threshold for tungsten is ~0.2 GW m^{-2} [38], and one of the major risk mitigation measures identified by MAG, and the subject of worldwide research effort, is the development of high-performance plasma regimes where ELMs are minimized or eliminated [39,40]. Many of the plasma regimes feature significant impurity injection to limit the power flux to the divertor and to control ELMs, and these are now combined in machines such as ASDEX-U and JET [32,41] with the need to control tungsten influx into the plasma. These impurities impinging on the divertor lead to greatly enhanced sputtering rate (as shown in Fig. 5) and the requirement to keep the plasma temperature to ~ 5 eV in front of the divertor, a very difficult target [32]. Moreover, it is to be noted that the 5 eV limit corresponds to only 5 MW m^{-2} power loading, thus running counter to one of the main arguments for adopting the water-cooled divertor. Consideration of Fig. 5 shows that this limit could be lifted if nitrogen seeding were not used. As the improvement in performance of the 'tungsten-walled' plasmas which comes from nitrogen seeding does not yet extrapolate without caveats from ASDEX-Upgrade to JET, the tokamak R&D is far from finished for extrapolation to DEMO. One major risk mitigation measure for DEMO, identified in the Roadmap [1] is to develop plasma regimes consistently integrated with divertor technology for DEMO. This will require not only experiments in ITER, but in other machines, possibly with a dedicated 'Divertor Test Tokamak' being required [1]. The avoidance of other problems with tungsten, such as re-crystallisation (occurs above ~ 1100 °C), or the growth of 'surface fuzz' during bombardment by the (~10% concentration) fusion plasma helium ions [42] (occurs above ~900 °C), there are likely to be very stringent levels of coolant temperature (less than around 150 °C) for water-cooled divertor concepts. Also in the range around 800–900 °C the most serious irradiation-induced swelling (~1–1.6%) takes place in tungsten at 10 dpa levels [43]. Such levels could be reached in the Roadmap DEMO Phase 1, and could cause problems in later fusion developments.

Copper alloys are recommended as the material for the HHF heat sinks in the water-cooled Divertor design, due to the unrivalled heat conduction of copper, and the existing tungsten–copper alloy design for ITER [34], which aims to prove a working design tolerant of the brittle nature of tungsten. The most serious issues for copper alloys relate to the rapid loss of ductility under irradiation at temperatures <180 °C, and the alloys' operating temperature for DEMO should thus be kept above 200 °C. In addition for some alloys, i.e. Cu–Cr–Zr, a combination of normal over-ageing and the decrease in strength with irradiation, limits the upper temperature for engineering structural applications to 350 °C for doses up to ~5 dpa. Design use of copper alloys (without better composites) may have to be restricted to substructure and coolant systems in the *immediate Divertor strike zone vicinity*, where neutronics simulations [4] show that damage is reduced by a factor ~ 3 compared to the outer edges of the divertor (figures ~ 3 dpa/fpy are predicted).

5.4. Advanced HHF/PFC materials

Even with tungsten and copper alloy R&D programmes recommended urgently to mitigate risks**:** an urgent start of irradiation campaigns to close gaps in the tungsten and copper alloys database, an early completion of tungsten materials data base to define design operating parameters enabling design concepts to be selected, the completion of R&D to establish joining of tungsten armour to RAFM, and the development of self-passivating alloys, there remain residual high-impact risks. The clear 'show-stopper' nature of the divertor development, and the known risks with tungsten and copper alloys make it essential to develop an RMM programme for HHF and PFC Armour. Many promising candidates have been identified for possible divertor/first wall applications:

- fibre and foil reinforced composites of copper and tungsten [44,45];
- tungsten laminates [46];
- tungsten–copper functionally-graded materials.

Notwithstanding promising results [47], these are at a very early stage of development (TRL 1-2), and there is a complete lack of data on irradiation results. The MAG proposes a generic materials R&D programme to develop towards a down-selection before the end of Horizon 2020. We also note the importance of tritium permeation barrier development, associated with an integration of the barrier layers in a first wall component, is required. Ceramic barrier layers must be developed both as an integral part of a first wall component.

6. Materials irradiation testing issues

As they will suffer from helium-transmutation production, it is clear that data from irradiation under a 'Fusion neutron spectrum' is essential as a precursor to final engineering decision on DEMO materials. The extent of this data can fall short of the full 'qualification' of materials envisaged in the full IFMIF project [48]. If the relatively-lightly irradiated Vacuum Vessel is chosen as the primary boundary, with passive safety and engineered 'defence in depth' provisions, then the He-content [4,5] can be limited to ~10^{-4} appm/fpy and normal fission-spectrum testing is sufficient. This reduces the licencing burden of the '14 MeV' neutron test data, but development of suitable engineering codes for design, and stakeholder-funding and regulatory requirements in general will still require the 14 MeV tests. The Roadmap milestone for provision of this data to match the Early DEMO construction decision in the early 2030s is to achieve at least 30dpa damage in steel samples by 2026. At this level, as previously noted, the helium-transmutation effects are beginning to be measurable. The MAG recommends that the Codes and Standards milestone for Early DEMO should be 2028 in the Roadmap for DEMO, but the exercise for the relevant materials should be started in Horizon 2020. Fusion spectrum irradiations will dominantly be from small size samples, and the use of these within standards such as RCC-MRx and ASME needs endorsement and adoption as soon as possible.

Long-term, it is clear that fusion materials development must continue, and therefore the full-IFMIF realisation should be kept as an ultimate goal, but the MAG, and Roadmap note that the full-IFMIF development may be too ambitious to be accelerated to meet the above milestones. The Group therefore believes that an early start on a more basic 14 MeV accelerator should be made. It would be optimum if this basic accelerator could be developed into the full-IFMIF at an appropriate stage (i.e. during the DEMO Phase 1 construction). Proposals exist for lower power, reduced irradiation volume (50–100 cm^3), accelerator-based 14 MeV sources which would provide the 30dpa data levels by 2025 [49,50], and the recommendation is for a technical assessment of alternatives [2], including the evaluation of the irradiation volume for the test materials. The IFMIF high-flux irradiation volume (500 cm^3) is based on a complete evaluation of all relevant parameters, so the evaluation needs to prioritise amongst these, in a risk-based manner.

To optimise a 14 MeV testing programme, extensive precursor R&D with isotopically- and chemically-tailored steels (featuring ^{54}Fe or ^{10}B-doping to produce helium nuclei under fission irradiation), and high-energy ion implantation experiments for structural and PFC/HHF materials is required. Data from these has been available in preliminary experiments (for a review see [51]), and there are problems of interpretation and extrapolation of the results. The ion-implanted data generally is restricted to near-surface volumes, whilst the tailored steels are either limited to a few dpa upper limit, or the helium-transmutation production rates occur on a much more rapid timescale than the displacement damage, leading to defect-migration issues. The technical resolution of these issues and the role of modelling must be evaluated.

7. Role of modelling

Materials modelling is important for understanding the processes involved in the early irradiations at low dpa, and with surrogate helium transmutation production, and can aid the optimisation of R&D and provide deep insights to produce a more targeted approach, thus conserving R&D funds. The progress and prospects of the EU Materials Modelling programme have been reviewed many times [52,53]. The priorities for Horizon 2020 have been identified:

- the development of multiscale models for the accumulation of radiation defects and transmutation products, in complex microstructures and complex alloys, in order to develop scaling laws for the relevant baseline and RM materials;
- the investigation of fundamentals of radiation and helium embrittlement effects, including modelling of mechanical response of irradiated materials at variable strain rates;
- the development of models for high-temperature phase stability and microstructural stability of materials, the determination of factors limiting the compatibility of materials under high-dose irradiation, and the correlation of models with ion-beam and fission irradiation experiments and nanoscale and small sample testing;
- addressing the problem of highly heterogeneous swelling, resulting from highly spatially heterogeneous neutron flux distributions in fusion systems;
- the integration of models for microstructural evolution with neutron transport calculations, the development of capabilities for the computer-model-based assessment of the end-of-life conditions for fusion power plant components, and the interfacing of such assessments with the planning of irradiation campaigns and materials testing programmes. In this respect, the MAG exercise has underlined that predictions of dpa rates are subject to significant uncertainty. One of the key requirements in future fusion materials modelling is a more reliable description of radiation damage, which may, or may not, include the concept of dpa.

8. Conclusions

The EU Fusion Roadmap exercise, with its emphasis on a DEMO concept for which a construction decision would be made in the early-2030s allows a sharp focus to be applied to the issues of materials development. The adoption of the project-based methodology allows risk analysis to prioritise the R&D programme needed to produce materials to maximise the success of the DEMO mission. This is aided by the treatment of systems engineering issues, and analysis of materials systems, and applying lessons learnt in the fission programme, especially in the development of safety cases, codes and standards.

A set of *Baseline Materials* has been identified for blanket structural applications (RAFM steels), plasma-facing components (tungsten), and high-heat flux materials (tungsten and copper alloys). The risks attendant in using these can be addressed via identified R&D programmes, but to be more certain of an optimum portfolio of materials, the development of *Risk Mitigation Materials* has also been identified for initial parallel development: high temperature FM and ODS steels for the structure, and composite tungsten and copper materials (especially laminate, fibre-reinforced materials and functionally-graded materials) for the PF and HHF applications.

Materials testing with a fusion neutron spectrum remains a high priority, but using the ITER precedent and fission experience, the vacuum-vessel of DEMO can be identified as the primary safety barrier, and this can be tested with a fission spectrum in existing machines, thus reducing the burden on fusion spectrum testing. The acceleration and optimisation of a fusion spectrum test programme is recommended via the early deployment of a less powerful 14 MeV neutron source compared to IFMIF, and by the pursuit of precursor programmes with isotopically- and chemically-tailored steels and helium ion implantation. Fundamental modelling is important for understanding the extrapolation of these results to a true fusion neutron irradiation condition, and has many other key roles, identified in the recommendations.

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