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Critical Assessment 12: Prospects for reduced activation steel for fusion plant

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CRITICAL ASSESSMENT

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M. J. Gorley*

The development of new, high performance reduced activation materials is increasingly recognised as one of the key enabling technologies required for the advancement of civil fusion power. Reduced activation steels are considered the leading materials for fusion reactor blanket structural materials. The manufacturing technologies and database for the current leading reduced activation steels have reached a state of maturity where basic design and implementation can be addressed. However, there remain concerns with these materials due to an incomplete irradiation database and because of their limited operational temperature window. The requirements of these steels along with various proposed methods to improve reduced activation steels are critically assessed, and some indications given on future paths for progress.

Keywords: Fusion, Reduced activation, Steels, Dispersion strengthening, Irradiation, Critical assessment

Introduction

Magnetic confinement nuclear fusion is reaching a state of maturity. Construction of ITER (the world largest experimental tokomak fusion reactor)[1] is under way and conceptual designs for demonstrator reactors (designed to provide net electricity to the grid) are already progressing with construction anticipated in the early 2030s.[2]

Whilst there is no radioactive core in a fusion plant, under the high energy fusion neutron irradiation within a fusion reactor, materials undergo changes in nuclide composition (transmutations) and some of the new nuclides may be radioactive, activating the materials. The irradiation levels and decay rates of these activated materials are dependent upon the elements (or more precisely the isotopes) used in the material.[3] To reduce the radioactive waste footprint from fusion the materials used in the reactors need to meet the criteria of low/reduced activation. These criteria require all materials used in a fusion reactor to be suitable for recycling or disposal in non-active landfills approximately 100 years after removal from the reactor.[4]

Figure 1 shows the level of radioactivity for several elements commonly found in steels (Fe, Cr, Ni, Mo, Nb and W) following the shutdown of a 3·6 GW fusion power, fusion reactor, assuming an anticipated blanket structural materials fusion irradiation flux of $\sim 1\times10^{19}$ neutron $m^{-2}\ s^{-1}$ over a 5 year irradiation time;[5] marked on the graph is the ITER administrative limit at 100 μSv/Hr for items available for hands-on maintenance.[6] Although a full calculation of each alloy is required to determine if it will meet the reduced activation requirement, it is clear from Fig. 1 that many elements commonly used in steels such as Ni, Nb and Mo will be significantly detrimental to the activation of the steels and thus must be removed or replaced by elements such as W or V.[3,6–8]

Critical to the future of the fusion programme is the development of reduced activation materials that can operate within the severe environment present in a fusion reactor. These reduced activation materials must enable safe, prolonged operation, at temperatures that can promote a high thermodynamic efficiency of the plant.[9–12]

Fusion materials requirements

Presently no detailed engineering designs or operational conditions exist for the demonstration reactors. However, evaluations by the EU fusion community's materials assessments group have identified the key components requiring new materials developments as the tritium breeding blanket and divertor.**[12,13] Figure 2 shows an artist's impression of EU DEMO (the EU DEM Onstration Power Plant, a proposed nuclear fusion power plant that is intended to be the next step after the ITER experimental nuclear fusion reactor) with the locations of the divertor and blanket indicated.

The plasma-facing surface of the divertor and blanket will likely be produced from W owing to its high sputter resistance and high melting temperature, and because W is a reduced activation element.[14] The structural material choice for the blanket is less certain[12,13] and a range of materials have been suggested, including vanadium alloys and SiC/SiC composites;[15–17] however the most techno-

**The divertor is situated along the bottom of the interior of the reactor structure and is the only point where the plasma is in direct contact with the reactor. Its primary function is to extract the helium produced by the fusion reaction and other impurities from the plasma. The blanket covers the remaining surfaces of the interior reactor structure, providing shielding to the vessel from the heat and neutron fluxes of the fusion reaction. The neutrons are slowed down in the blanket, where their kinetic energy is transformed into heat for electrical power production, and they also react with Li to produce the tritium ('breeding'), essential for fuel self-sufficiency.

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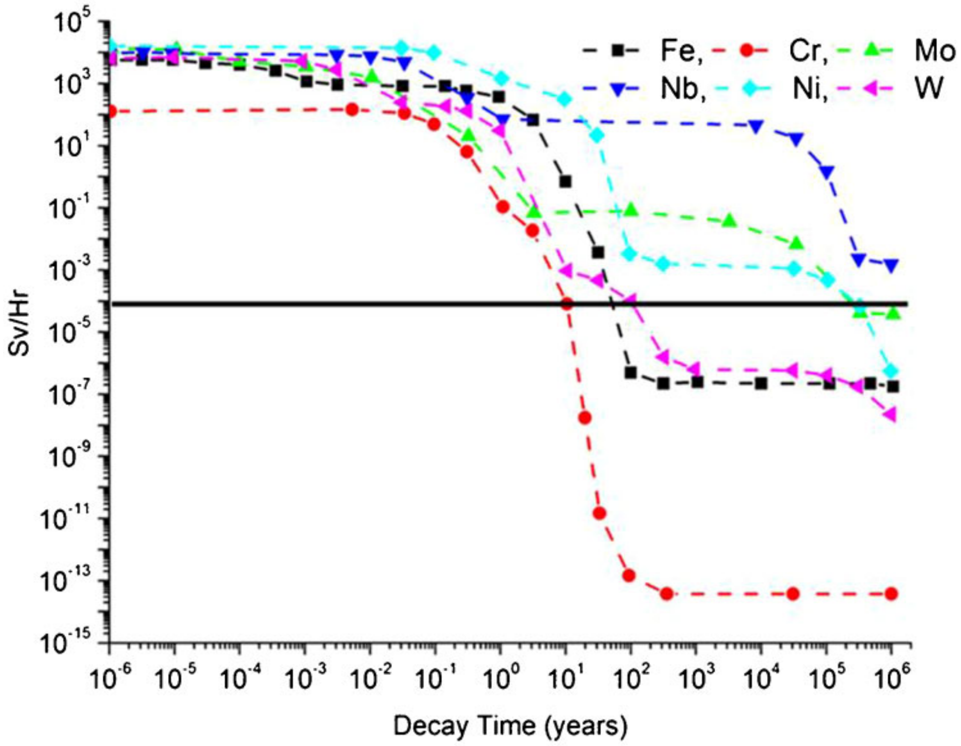
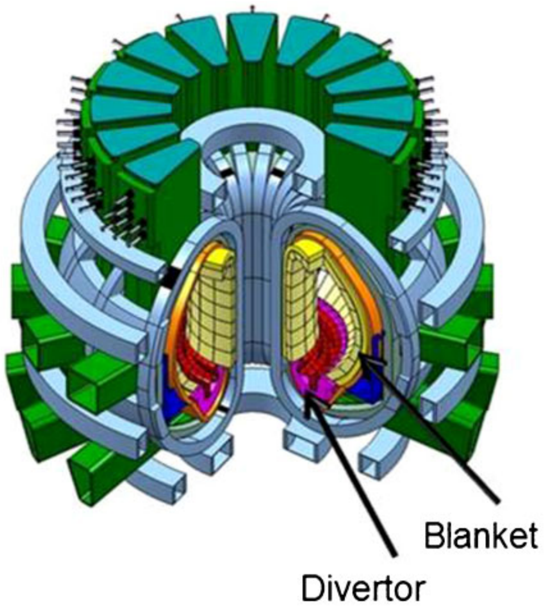


1 Level of irradiation from several common elements found in steels as function of time after removal from 3·6 GW fusion power reactor following irradiation time of 5 years, assuming they had received anticipated blanket front end irradiation doses (reconstructed from data in Ref. 5): black horizontal line in image represents ITER administrative limit for hands-on maintenance[6]

logically developed materials are reduced activation ferritic/martensitic steels.[13]

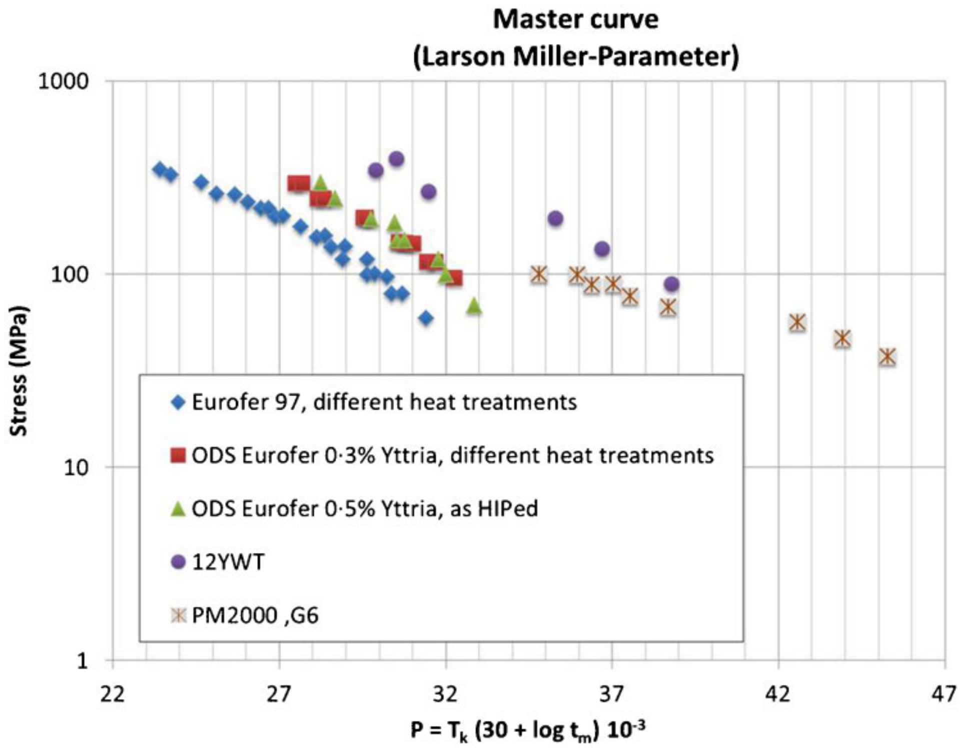
The design criteria for the blanket structural materials on the demonstration reactors are yet to be established, hence few quantitative values for the required properties can be provided with any confidence.[18] However, as well as conforming to the reduced activation requirements, a range of other critical points must be considered in the designing of structural steels for the blanket. An indicative,



2 Artist's impression of DEMO with locations of blanket and divertor indicated

but not exhaustive, list of the material design considerations is given hereafter:

(i) an acceptably low neutron capture cross-section to ensure a sufficient tritium breeding ratio;[19] this implies limits on the quantity of material that can be used, and in particular limits on elements (such as W) with a high neutron capture cross-section
(ii) compatibility with remote handling: critically this will require compatibility with welding techniques, presently designed around laser welding[20]
(iii) stability under cyclic operation, with $>1{\cdot}5\times10^4$ cycles anticipated for the blanket in the current EU demonstrator reactor[13]
(iv) retention of mechanical properties within engineering design criteria under irradiation (the anticipated peak fusion neutron flux is of the order $\sim1\times10^{19}$ neutrons $m^{-2}\,s^{-1}$ for steels at the front of the blanket and expected component lifetimes are $>1{\cdot}33$ full power years for DEMO and 5 full power years for an operating power plant)[12,21,22]
(v) sufficient tolerance to He and H embrittlement to ensure a brittle to ductile transition temperature (BDTT) $>20^\circ C$ during operation, with anticipated levels of >100 at.-ppm of He and H produced per full power year in steels at the front of the blanket due to (n, α) and (n, p) reactions[9,10,23,24]
(vi) chemical compatibility with coolant (such as water or He) to ensure negligible corrosion
(vii) compatibility with tritium removal systems, to ensure negligible tritium retention in the material



3 Creep strength of two grades of ODS Eurofer, PM2000, 12YWT (produced at Oak Ridge National Laboratory) and conventional Eurofer 97 (reproduced from Ref. 52)

and to meet safety regulatory requirements for the total tritium inventory (tritium inventory limits set at ~3 kg for ITER)

(viii) dimensional (<1% total swelling) and structural integrity at the operational temperature.[12]

The operating temperature of the blanket plays an important role in the thermodynamic efficiency and hence the anticipated cost of electricity generation from fusion reactors.[25] The operational temperatures allowable for the blanket are currently set by the creep life and BDTT of the materials and the coolant type used for the balance of plant. Water cooled primary loop coolants require operational temperatures of about 290–320°C whereas more advanced systems, such as He cooled blankets, potentially enabling higher plant efficiencies, typically would require operating temperatures of 650°C or above.[13] Presently the blanket materials are considered a key limiting factor in the utilisation of high operating temperatures and the development of new reduced activations steels for this environment is a key driver in fusion materials research.

Reduced activation ferritic/martensitic (RAFM) steels

Reduced activation ferritic/martensitic steels, developed by the fusion community, were initially designed around reduced activation versions of 9CrMo steels such as T91 steel (Fe–0·1C–0·5Mn–9Cr–1Mo–0·1Nb–0·25V, wt-%).[26,27] Reduced activation ferritic/martensitic steels offer several advantages over austenitic steels such as improved dimensional stability (reduced creep and swelling) under neutron irradiation[27–29] and improved thermal conductivity and expansion (approximately 2·5 times better at 500°C).[30–32]

Overall, RAFM steels typically offer a good balance of the required mechanical properties for use in fusion reactors, including: good fracture toughness, high strength, high cycle fatigue tolerance and ductility.[2] In addition they are typically compatible with He gas coolants and water coolants, demonstrating negligible corrosion (within the anticipated operating temperature window).[13] Reduced activation ferritic/martensitic steels such as Eurofer 97 (Fe–0·1C–9Cr–0·07Ta–0·2V–1W),[33] F82H (Fe–0·1C–8Cr–0·04Ta–0·14V–2W)[33] and CLAM (Fe–0·1C–9Cr–0·15Ta–0·2V–1·5W)[34] (all wt-%) are leading candidate structural materials for fusion reactor blankets. Eurofer, one of the most technologically developed RAFM steels, will be used in the EU test blanket modules in ITER[26] and is considered the baseline material choice for the EU DEMO reactor design.[12,13,18]

However, there remain serious concerns for the development and use of these steels in fusion reactors. The most critical of these concerns relates to the limited operational temperature window, typically 350–550°C after irradiation.[35] The lower limit is primarily due to He embrittlement in low temperature operations, which can shift the BDTT to above 30°C;[36,37] the upper limit is due to loss of strength, limiting the creep life.[13]

The lower temperature limit for RAFM steels can be improved by optimising the processing conditions, with some modified batches of Eurofer and F82H showing superior resistance to irradiation embrittlement around 300°C due to alternative heat treatments at higher austenisation temperatures.[38]

The more challenging issue for the prospects of utilising RAFM steels in fusion reactors relates to the upper temperature limit. There is evidence from outside the fusion community that improved upper operational temperatures can be achieved for ferritic/martensitic steels, through complex thermomechanical heat treatments that increase the number density of nitride and carbide precipitates.[39] These advanced ferritic/martensitic steels have shown some promise, with Fe–9Cr–2W–0·5Mo type 92 steels reaching $>3\times10^4$ h creep rupture life at 92 MPa.[13] Development of new, reduced activation variants of these grades offers one of the most promising methodologies to enhance the upper operational temperature for RAFM steels. However, as yet, no reduced activation version of these advanced steels has been developed, no data on the irradiation stability or long term thermal stability of the fine carbide and nitride precipitates have been determined, no long term creep performance data for the steels have been established and – critically – these advanced steels appear to be reaching their upper limit in operational temperature below 650°C.[40] Thus, despite the potential increase in the operational limits of RAFM steels offered by complex thermomechanical treatments, future advances for reduced activation steels capable of operating above 650°C may require alternative solutions.

Reduced activation austenitic steels

Austenitic steels under neutron irradiation exhibit excessive swelling and He embrittlement (far worse than that observed in RAFM steels).[41] Despite some evidence[42,43] for methodologies to mitigate swelling (Fe–Cr–Ni stainless steels have shown reductions in volumetric swelling from $\sim$22% to <2% through increased precipitate and dislocation densities), swelling and embrittlement remain serious concerns.[42,43] In addition, reduced activation austenitic steels need alternative austenitic stabilising elements to replace Ni (which, along with Cu and Co, is not a low activation element);[5] Mn and N offer the most attractive reduced activation alternatives.

Reduced activation austenitic steels utilising Mn have received some interest from the fusion community;[44,45] however, concerns over the high decay heat and potential volatilisation of Mn in loss of coolant accident conditions caused these steels to be abandoned for use in fusion reactors.[7] High N containing austenitic steels suffer from a lack of stability at the temperatures required for operation due to the formation of Fe and Cr nitrides.[46] These limitations of the key reduced activation variations of austenitic steels, coupled with impaired irradiation resistance and thermophysical properties (compared to RAFM steels), limit the prospects for austenitic steels in demanding fusion environments.

Reduced activation oxide dispersion strengthened steels

Another alternative area for steels development is oxide dispersion strengthened (ODS) steels. In leading ferritic ODS alloys, a fine dispersion of 2–5 nm diameter thermodynamically stable Y, Ti and O rich precipitates are uniformly distributed throughout a ferrite matrix. These 'nano-precipitates' act as pinning points for He, potentially delaying the onset of He swelling and embrittlement;[23] in addition they can reduce the average grain size of the steel and impede dislocation motion, which can increase the high temperature creep properties.[23,47,48] The nano-precipitates have also been shown to be stable under irradiation[49] and are believed to improve the stability of the microstructure under irradiation and during cyclic fatigue.[23,49–51]

In Fig. 3 the creep performance of two grades of ODS Eurofer, an industrially produced corrosion resistant ODS alloy (PM2000) and a research grade ODS alloy (12YWT) are compared with that of Eurofer.[52] The superior performance of the ODS steels relative to the conventional RAFM Eurofer is clear. In particular, the 12YWT alloy (which is a typical modern ferritic ODS steel)[23] shows significant improvement in creep life over Eurofer. The improvements in creep performance of ODS steels may enable operational temperatures to be raised by several hundreds of degrees compared with those of conventional RAFM steels.[48]

Presently the only proven means of mass producing these modern ferritic ODS steels is via mechanical alloying of steel powder and yttrium containing oxide/intermetallic powders, followed by hot isostatic pressing/extrusion and thermomechanical treatments.[23] Although this processing method has been used for the industrial production of ODS alloys in the past (for example PM2000 produced by Plansee and M957 produced by Special Metals),[48] there are presently no large scale industrial manufacturers of ODS steels. This manufacturing method is inherently more expensive than liquid metal processing and often incurs problems from batch to batch variations[13] that raise serious concerns for the manufacture of these alloys for nuclear environments.

In addition, ODS alloys often suffer from detrimental mechanical properties in comparison to conventional ferritic steels, including reduced fracture toughness and ductility.[53] There are also difficulties relating to the welding of these alloys; traditional welding techniques, such as electron beam welding, are reported to retain only 20–30% of the original strength.[54] Alternative (non-molten) welding techniques such as friction stir welding appear better suited for joining ferritic ODS alloys and strengths of 50–60% of the original base material have been reported in these joints.[54–57]

Overall, despite promising properties, significant work is still required to establish acceptable industrial scale techniques for production and joining of ODS steels before they can be considered as candidate materials for future fusion reactors. Economical and reproducible production of ODS alloys on an industrial scale may require a step change in thinking. Areas such as direct inclusion of the yttrium during gas atomisation (either in the melt or sprayed into droplets as a powder during the gas atomisation step) or rolling together alternating layers of metal/oxide/metal to mass produce ODS alloys need to be considered.

Critical considerations for progress of fusion materials

In addition to assessing the current prospective directions to improve reduced activation steels, there are critical factors that must be considered when reviewing fusion materials, including the lack of fusion relevant irradiation spectrum data and the timeframe for validation testing of new materials.

The effects of a true fusion neutron spectrum on materials are still largely unknown.[12] Critical to the future development and validation of any materials to be utilised in a fusion reactor will be assessments of the effects of fusion irradiation on materials properties.[58] Access to this vital information has been impaired by a lack of any materials testing facilities that can produce a representative fusion neutron spectrum. The International Fusion Materials Irradiation Facility has been proposed to investigate the effects of fusion irradiation on materials. However, owing to the high costs and anticipated operational timeframe for the International Fusion Materials Irradiation Facility,[59] it was recognised that an early fusion neutron source is required.[13] The future development of materials for fusion will likely require rapid commissioning and intelligent utilisation of an early neutron source. These initial evaluations should ensure that the materials degradation mechanisms from a fusion neutron spectrum are readily evaluated and compared with modelling predictions and fission/heavy ion irradiations (which are cheaper and easier to perform).[60] These comparative data could enable results from these alternative methods to be used in conjunction with a fusion materials irradiation facility, which may reduce the costs and timeframe for materials validation testing.

Summary

The prospects for reduced activation steels deployment in a demonstration fusion plant are assisted by the significant technological development and understanding of current leading RAFM steels. However, to promote higher efficiencies in the demonstration reactors and enable utilisation of steels in future commercial reactors, the operational temperature window will need to be broadened. The most promising route for extending the operating temperature of reduced activation steel appears to lie with complex thermomechanical treatments (likely to be limited to $\leq 650^{\circ}C$) or through industrial development of ODS steels (potentially enabling operation above $650^{\circ}C$). These advanced steels will require accelerated validation testing, including investigation of fusion neutron spectrum irradiation stability, if they are to be safely implemented into future fusion reactors.

Overall, reduced activation steels are standing at a point where their adoption into future commercial fusion reactors is uncertain, and the anticipated demanding environmental conditions of these future fusion reactors are forcing them beyond their current limits. The future prospect of reduced activation steels appears dependent upon the successful development of new advanced steels that can push beyond the current state of the art.

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