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CCFE-PR(15)39

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 The following article appeared in Materials Science and Technology, Vol.31, Issue 8, May 2015, pp.975-980

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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. To comply with the fusion ethos of no material entering the permanent active waste disposal route, reduced activation steels have been developed and are considered the leading materials for fusion reactor's blanket structural materials. The manufacturing technologies and database for the current leading reduced activation steels have reached a maturity where basic design and implementation can be addressed. However, there remain concerns with these materials not only because the irradiation database on these alloys is incomplete, but because the current operational temperature for these materials does not enable them to be utilised in optimised plant designs. An indication of the requirements of these steels along with various proposed methods for improving reduced activation steels are critically assessed, and some indications given on future paths for progress.

Key words: Fusion, Reduced activation, Steels, Dispersion strengthening, Irradiation.

1. <u>Introduction</u>

Magnetic confinement nuclear fusion is reaching a maturity where construction of ITER (the world largest experimental tokomak fusion reactor [1]) is underway 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 to 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 ~100 years after removal from the reactor [4].

Figure 1 shows the level of radiation for several elements commonly found in steels (Fe, Cr, Ni, Mo, Nb and W) following the shutdown of a 3.6GW fusion power, fusion reactor, assuming they received anticipated blanket structural materials fusion irradiation flux of $\sim 1 \times 10^{19}$ neutron m^{-2} s^{-1} over a 5 year irradiation time [5]; marked on the graph is the ITER administrative limit at 100µSv/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 figure 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].

2. 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 highlighted the key components requiring new materials developments as the tritium breeding blanket and divertor[1] [12-13]. Figure 2 shows an artist's impression of DEMO 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, 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 different materials have been suggested, including vanadium alloys and SiC/SiC composites [15-17]; however the most technologically developed materials are reduced activation ferritic/martensitic (RAFM) steels [13].

The design criteria for the blanket structural materials on the demonstration reactors are yet to be established [18]; however, as well as conforming to the reduced activation requirements there are a range of other critical points to consider in the designing of structural steels for the blanket. An indicative, but not exhaustive, list of the material design considerations are given hereafter:

- Acceptably low neutron capture cross section to ensuring sufficient tritium breeding ratio [19], this implies limits on the quantity of material that can be used and in particular limits elements (such as W) with a high neutron capture cross section.
- Compatibility with remote handling, critically this will require compatibility with welding techniques, presently designed around laser welding [20].

[1] 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 also react with Li to produce tritium ("breeding"), essential for fuel self-sufficiency.

- Stability under cyclic operation, with $>1.5 \times 10^4$ cycles anticipated for the blanket in the current EU demonstrator reactor [13].
- Retention of mechanical properties within engineering design criteria under irradiation. The anticipated peak fusion neutron flux is of the order~1×10^{19} neutrons $m^{-2} s^{-1}$ for steels at the front of the blanket and expected component lifetimes are >1.33 full power years for DEMO and 5 full power years for an operating power plant [12, 21-22]
- Sufficient tolerance to He and H embrittlement ensuring a BDTT >20°C during operation, with anticipated levels of >100appm 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].
- Chemical compatibility with coolant (such as water or He) to ensure negligible corrosion
- Compatibility with tritium removal systems, ensuring negligible tritium retention in the material. Ensuring the total tritium inventory is kept low to meet safety regulatory requirements (tritium inventory limits set at ~3kg for ITER).
- 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 from fusion reactors [25]. Presently the operational temperatures allowable for the blanket are set by the creep life and brittle to ductile transition temperature (BDTT) of the materials and the coolant type used for the balance of plant. Water cooled primary loop coolants require operational temperatures of ~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.

3. Reduced activation ferritic martensitic (RAFM) steels

RAFM steels, developed by the fusion community, were initially designed around reduced activation versions of 9CrMo steels such as T91 steel (Fe-9Cr-1Mo-0.5Mn-0.1C-0.1Nb-0.25V) [26-27]. RAFM steels offer several advantages over austenitic steels with 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 with negligible corrosion (within the anticipated operating temperature window) [13]. RAFM steels such as Eurofer 97 (Fe-9Cr-1W-0.2V-0.07Ta-0.1C (wt%) [33]), F88H (Fe-8Cr-2W-0.14V-0.04Ta-0.1C (wt%) [33]) and CLAM (Fe-9Cr-1.5W-0.2V-0.15Ta-0.1C (wt%) [34]) are leading candidate structural materials for fusion reactor blankets. Eurofer is one of the most technologically developed RAFM steel and will be used in the EU test blanket modules in ITER [26] and is considered the baseline material choice for the EU Demonstration 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 between 350°C and 550°C after irradiation [35]. The lower temperature limit is primarily due to He embrittlement at low temperature operations, shifting the BDTT above 30°C [36-37]; and the upper limit 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 austenization 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 for improvements in the upper operational temperatures for ferritic/martensitic steels, through complex thermo-mechanical 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 >3x10^4h creep rupture at 92MPa [13]. Development of new, reduced activation variants of these 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 have been developed, no irradiation stability or long term thermal stability of the fine carbide and nitride precipitates has been determined, no long term creep performance of the steels is established and critically, these advanced steels appear to be reaching their upper limit in operational temperature before 650°C [40]. Thus despite the potential increase in the operational limits of RAFM steels offered by complex thermo-mechanical treatments, the future prospects for reduced activation steels operating above 650°C may require alternative solutions.

4. 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 for mitigation of swelling, Fe-Cr-Ni stainless steels showed reductions in volumetric swelling from ~22% to <2% through increased precipitate and dislocation densities, swelling and embrittlement remain serious concerns. 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]), and Mn and N offer the most attractive reduced activation alternatives.

Reduced activation austenitic steels utilising Mn had received some interest from the fusion community [44], however the high decay heat and potential volatilisation from Mn in loss of coolant accident conditions caused these 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 [45]. These limitations of the key reduced activation variations of austenitic steels, coupled with impaired irradiation resistance and thermo-physical properties (compared to RAFM steels), limit the prospects for austenitic steels in demanding fusion environments.

5. Reduced activation Oxide Dispersion Strengthened (ODS) steels

Another alternative area for steels development is oxide dispersion strengthened (ODS) steels. In leading ferritic ODS alloys, a fine dispersion of 2-5nm 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, 46-47]. The nano-precipitates have also been shown to be stable under irradiation [48] and are believed to improve the stability of the microstructure under irradiation and during cyclic fatigue [23, 48-50].

Figure 3 (reproduced from [51]) shows the creep properties of two grades of ODS Eurofer, an industrially produced corrosion resistant ODS alloy (PM2000) and a research grade ODS alloy (12YWT) in comparison to the creep performance of Eurofer. The improved creep performance of ODS steels in comparison to conventional RAFM steels such as Eurofer is clear. In particular, the 12YWT alloy (which is a typical modern ferritic ODS steel [23]) shows significant improvements in creep life when compared to Eurofer. The improvements in creep performance of ODS steels may enable several hundred of degrees higher operational temperatures compared to conventional RAFM steels [47].

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 thermo-mechanical treatments [23]. Although this processing method has been used to industrially produce ODS alloys in the past, for example PM2000 produced by Plansee and M957 produced by Special Metals [47], there are presently no large scale industrial manufacturers of ODS alloys. This manufacturing method is inherently more expensive than liquid metal processing and often incurs problems from batch to batch variations [13] which imposes serious concerns for the manufacture of these alloys for nuclear environments.

In addition, ODS alloys often suffer from a range of detrimental mechanical properties in comparison to conventional ferritic steels including reduced fracture toughness and ductility [52]. There are also difficulties relating to the welding of these alloys; traditional welding techniques, such as electron beam welding, are reported to maintain only 20-30% of the original strength [53]. 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 [53-56].

Overall, despite promising properties, significant work is still required to establish acceptable industrial scale production and joining of ODS steels before they can be considered as candidate materials for future fusion reactors.

6. Critical considerations for progress of fusion materials

In addition to assessing the current prospective directions for improving 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 [57]. 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 (IFMIF) has been proposed to investigate the effects of fusion irradiation on materials. However, owing to the high costs and anticipated operational timeframe for IFMIF [58], 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 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 [59]). This comparative data could enable results from these alternative methods be used in conjunction with a fusion materials irradiation facility, which may reduce the costs and timeframe for materials validation testing.

7. Summary

The prospects for reduced activation steels deployment in a demonstration fusion plants is 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 thermo-mechanical treatments (likely to be limited to ≤650°C) or through industrial development of ODS steels (potentially enabling operation above 650°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.

8. Acknowledgments

Authorship for this work was funded by the RCUK Energy Programme under grant EP/I501045.

9. References

[1] - SDC-IC, I.T.E.R. "Structural Design Criteria for ITER In-Vessel Components." *ITER Document No. G* 74 (2004).

[2] - Romanelli, F., Federici, L. H., Neu, R., Stork, D., & Zohm, H. (2013, June). A roadmap to the realization of fusion energy. In *Proc. IEEE 25th Symp. Fusion Eng* (pp. 1-4)..

[3] - Gilbert, M. R., Packer, L. W., Sublet, J. C., & Forrest, R. A. (2014). Inventory Simulations Under Neutron Irradiation: Visualization Techniques as an Aid to Materials Design. *NUCLEAR SCIENCE AND ENGINEERING*, *177*(3), 291-306

[4] - El-Guebaly, L., Massaut, V., Tobita, K., & Cadwallader, L. (2007). Evaluation of Recent Scenarios for Managing Fusion Activated Materials: Recycling and Clearance, Avoiding Disposal. *UWFDM-1333, University of Wisconsin Fusion Technology Institute~ Sep*.

[5] - Forrest, R. A., Tabasso, A., Danani, C., Jakhar, S., & Shaw, A. K. (2009). Handbook of activation data calculated using EASY-2007. *EURATOM/UKAEA Fusion Association, UKAEA FUS*, *552*.

[6] - Taylor, N., Ciattaglia, S., Cortes, P., Iseli, M., Rosanvallon, S., & Topilski, L. (2012). ITER safety and licensing update. *Fusion Engineering and Design*, *87*(5), 476-481.

[7] - Harries, D. R., Butterworth, G. J., Hishinuma, A., & Wiffen, F. W. (1992). Evaluation of reduced-activation options for fusion materials development. *Journal of nuclear materials*, *191*, 92-99.

[8] - Cheng, E. T. (1998). Concentration limits of natural elements in low activation fusion materials. *Journal of nuclear materials*, *258*, 1767-1772.

[9] - Bloom, E. E. (1998). The challenge of developing structural materials for fusion power systems. *Journal of nuclear materials*, *258*, 7-17.

[10] - Bloom, E. E., Zinkle, S. J., & Wiffen, F. W. (2004). Materials to deliver the promise of fusion power–progress and challenges. *Journal of nuclear materials*, *329*, 12-19.

[11] - Holdren, J. P., Berwald, D. H., & Budnitz, R. J. (1988). Exploring the competitive potential of magnetic fusion energy: the interaction of economics with safety and environmental characteristics.

[12] - Stork, D., Agostini, P., Boutard, J. L., Buckthorpe, D., Diegele, E., Dudarev, S. L., English, C., Federici, G., Gilbert, M. R., Gonzalez, S., Ibarra, A., Linsmeier, C., Puma, A.L., Marbach, G., Packer, L. W., Raj, B., Reith, M., Tran, M. Q., Ward, D. J., & Zinkle, S. J.

(2014). Materials R&D for a timely DEMO: Key findings and recommendations of the EU Roadmap Materials Assessment Group. *Fusion Engineering and Design*.

[13] - Stork, D., Agostini, P., Boutard, J. L., Buckthorpe, D., Diegele, E., Dudarev, S. L., English, C., Federici, G., Gilbert, M. R., Gonzalez, S., Ibarra, A., Linsmeier, C., Puma, A.L., Marbach, G., Morris, P. F., Packer, L. W., Raj, B., Reith, M., Tran, M. Q., Ward, D. J., & Zinkle, S. J. (2014). Developing structural, high-heat flux and plasma facing materials for a near-term DEMO fusion power plant: The EU assessment. *Journal of Nuclear Materials*, *455*(1), 277-291.

[14] - Rieth, M., Dudarev, S. L., Gonzalez de Vicente, S. M., Aktaa, J., Ahlgren, T., Antusch, S., Armstrong D.E.J., Balden M., Baluc, N., Barthe, M.F., Basuki, W.W., Battabyal, M., Becquart, C.S., Blagoeva, D., Boldyryeva, H., Brinkmann, J., Celino, M., Ciupinski, L., Correia, J.B., De Backer, A., Domain, C., Gaganidze, E., García-Rosales, C., Gibson, J., Gilbert, M.R., Giusepponi, S., Gludovatz, B., Greuner, H., Heinola, K., Höschen, T., Hoffmann, A., Holstein, N., Koch, F., Krauss, W., Li, H., Lindig, S., Linke, J., Linsmeier, Ch., López-Ruiz, P., Maier, H., Matejicek, J., Mishra, T.P., Muhammed, M., Muñoz, A., Muzyk, M., Nordlund, K., Nguyen-Manh, D., Opschoor, J., Ordás, N., Palacios, T., Pintsuk, G., Pippan, R., Reiser, J., Riesch, J., Roberts, S.G., Romaner, L., Rosiński, M., Sanchez, M., Schulmeyer, W., Traxler, H., Ureña, A., van der Laan, J.G., Veleva, L., Wahlberg, S., Walter, M., Weber, T., Weitkamp, T., Wurster, S., Yar, M.A., You, & J.H., Zivelonghi A. (2013). Recent progress in research on tungsten materials for nuclear fusion applications in Europe. *Journal of Nuclear Materials*, *432*(1), 482-500.

[15] - Giancarli, L. M., Abdou, M., Campbell, D. J., Chuyanov, V. A., Ahn, M. Y., Enoeda, M., Pan, C., Poitevin, Y., Rajendra Kumar, E., Ricapito, I., Strebkov, Y., Suzuki, S., Wong. P. C., & Zmitko, M. (2012). Overview of the ITER TBM Program. *Fusion Engineering and Design*, *87*(5), 395-402.

[16] - Ihli, T., Basu, T. K., Giancarli, L. M., Konishi, S., Malang, S., Najmabadi, F., Nishio, S., Raffray, A. R., Rao, C. V. S., Sagara, A., & Wu, Y. (2008). Review of blanket designs for advanced fusion reactors. *Fusion Engineering and Design*, *83*(7), 912-919.

[17] - Zinkle, S. J., Möslang, A., Muroga, T., & Tanigawa, H. (2013). Multimodal options for materials research to advance the basis for fusion energy in the ITER era. *Nuclear Fusion*, *53*(10), 104024.

[18] - Tavassoli, A. A., Diegele, E., Lindau, R., Luzginova, N., & Tanigawa, H. (2014). Current status and recent research achievements in ferritic/martensitic steels. *Journal of Nuclear Materials*, *455*(1), 269-276.

[19] - Maki, K., & Okazaki, T. (1983). Effect of blanket structure on tritium breeding ratio in fusion reactors.

[20] - Tada, E., Kakudate, S., Oka, K., Obara, K., Nakahira, M., Taguchi, K.,Itohi, A., Fukatsu, S., Takeda, N., Takahashi, H., Akou, K., Shibanuma, K., Burgess, T., Tesini, A., Matsuhira, N., Holloway, C., & Haange, R. (1998). Development of remote maintenance equipment for ITER blankets. *Fusion engineering and design*, *42*(1), 463-471.

[21] - Gilbert, M. R., Dudarev, S. L., Zheng, S., Packer, L. W., & Sublet, J. C. (2012). An integrated model for materials in a fusion power plant: transmutation, gas production, and helium embrittlement under neutron irradiation. *Nuclear Fusion*, *52*(8), 083019.

[22] - Gilbert, M. R., Dudarev, S. L., Nguyen-Manh, D., Zheng, S., Packer, L. W., & Sublet, J. C. (2013). Neutron-induced dpa, transmutations, gas production, and helium embrittlement of fusion materials. *Journal of Nuclear Materials*, *442*(1), S755-S760.

[23] - Odette, G. R., Alinger, M. J., & Wirth, B. D. (2008). Recent developments in irradiation-resistant steels. *Annu. Rev. Mater. Res.*, *38*, 471-503.

[24] - Tanigawa, H., Shiba, K., Möslang, A., Stoller, R. E., Lindau, R., Sokolov, M. A., Odette, G. R., Kurtz, R. J., & Jitsukawa, S. (2011). Status and key issues of reduced activation ferritic/martensitic steels as the structural material for a DEMO blanket. *Journal of Nuclear Materials*, *417*(1), 9-15.

[25] - Maisonnier, D., Campbell, D., Cook, I., Di Pace, L., Giancarli, L., Hayward, J., Puma, A. L., Medrano, M., Norajitra, P., Roccella, M., Sardain, P., Tran, M. Q., & Ward, D. (2007). Power plant conceptual studies in Europe. *Nuclear Fusion*, *47*(11), 1524.

[26] - Aiello, G., Aktaa, J., Cismondi, F., Rampal, G., Salavy, J. F., & Tavassoli, F. (2011). Assessment of design limits and criteria requirements for Eurofer structures in TBM components. *Journal of Nuclear Materials*, *414*(1), 53-68.

[27] - Gelles, D. S. (1996). Development of martensitic steels for high neutron damage applications. *Journal of nuclear materials*, *239*, 99-106.

[28] - Wallner, G., Anand, M. S., Greenwood, L. R., Kirk, M. A., Mansell, W., & Waschkowski, W. (1988). Defect production rates in metals by reactor neutron irradiation at 4.6 K. *Journal of Nuclear Materials*, *152*(2), 146-153.

[29] - Klabunde, C. E., & Coltman Jr, R. R. (1982). Fission neutron damage rates and efficiencies in several metals. *Journal of Nuclear Materials*, *108*, 183-193.

[30] - Tavassoli, A. A. (2002). Present limits and improvements of structural materials for fusion reactors–a review. *Journal of Nuclear Materials*, *302*(2), 73-88.

[31] - Zinkle, S. J., & Ghoniem, N. M. (2000). Operating temperature windows for fusion reactor structural materials. *Fusion Engineering and design*, *51*, 55-71.

[32] - Hishinuma, A., Kohyama, A., Klueh, R. L., Gelles, D. S., Dietz, W., & Ehrlich, K. (1998). Current status and future R&D for reduced-activation ferritic/martensitic steels. *Journal of Nuclear Materials*, *258*, 193-204.

[33] – Jitsukawa, S., Kimura, A., Kohyama, A., Klueh, R. L., Tavassoli, A. A., Van der Schaaf, B., Odette, G. R., Rensmen, J. W., Victoria, M., & Petersen, C. (2004). Recent results of the reduced activation ferritic/martensitic steel development. *Journal of Nuclear Materials*, *329*, 39-46.

[34] - Huang, Q., Li, C., Li, Y., Chen, M., Zhang, M., Peng, L., Zhu, Z., Song, Y., & Gao, S. (2007). Progress in development of China low activation martensitic steel for fusion application. *Journal of Nuclear Materials*, *367*, 142-146.

[35] - Gasparotto, M., Andreani, R., Boccaccini, L. V., Cardella, A., Federici, G., Giancarli, L., Marois, G. L., Maisonnier, D., Malang, S., Moeslang, A., Poitevin, Y., van der Schaaf,B., & Victoria, M. (2003). Survey of in-vessel candidate materials for fusion power plants—the European materials R&D programme. *Fusion engineering and design*, *66*, 129-137.

[36] - Gaganidze, E., & Aktaa, J. (2013). Assessment of neutron irradiation effects on RAFM steels. *Fusion Engineering and Design*, *88*(3), 118-128.

[37] - Gaganidze, E., Petersen, C., Materna-Morris, E., Dethloff, C., Weiß, O. J., Aktaa, J., Poystyanko, A., Fedoseev, A., Makarov, O., & Prokhorov, V. (2011). Mechanical properties and TEM examination of RAFM steels irradiated up to 70dpa in BOR-60. *Journal of Nuclear Materials*, *417*(1), 93-98.

[38] - Gaganidze, E., Schneider, H. C., Dafferner, B., & Aktaa, J. (2007). Embrittlement behavior of neutron irradiated RAFM steels. *Journal of Nuclear Materials*, *367*, 81-85.

[39] - Klueh, R. L., Hashimoto, N., & Maziasz, P. J. (2007). New nano-particle-strengthened ferritic/martensitic steels by conventional thermo-mechanical treatment. *Journal of Nuclear Materials*, *367*, 48-53.

[40] - Zinkle, S. J., & Ghoniem, N. M. (2011). Prospects for accelerated development of high performance structural materials. *Journal of Nuclear Materials*, *417*(1), 2-8.

[41] - Zinkle, S. J., & Was, G. S. (2013). Materials challenges in nuclear energy. *Acta Materialia*, *61*(3), 735-758.

[42] - Packan, N. H., & Farrell, K. (1979). Simulation of first wall damage: effects of the method of gas implantation. *Journal of Nuclear Materials*, *85*, 677-681.

[43] - Lee, E. H., & Packan, N. H. (1989). Swelling suppression in phosphorous-modified Fe-Cr-Ni alloys during neutron irradiation. In *Effects of Radiation on Materials: 14th International Symposium* (pp. 133-146). American Society for Testing and Materials, Philadelphia.

[44] - Klueh, R. L. (Ed.). (1990). *Reduced activation materials for fusion reactors* (No. 1047). Astm International.

[45] - Simmons, J. W. (1996). Overview: high-nitrogen alloying of stainless steels. *Materials Science and Engineering: A*, *207*(2), 159-169.

[46] - Ukai, S., & Fujiwara, M. (2002). Perspective of ODS alloys application in nuclear environments. *Journal of Nuclear Materials*, *307*, 749-757.

[47] - Klueh, R. L., Shingledecker, J. P., Swindeman, R. W., & Hoelzer, D. T. (2005). Oxide dispersion-strengthened steels: A comparison of some commercial and experimental alloys. *Journal of Nuclear Materials*, *341*(2), 103-114.

[48] - Ribis, J., & Lozano-Perez, S. (2014). Nano-cluster stability following neutron irradiation in MA957 oxide dispersion strengthened material. *Journal of Nuclear Materials*, *444*(1), 314-322.

[49] - Certain, A., Kuchibhatla, S., Shutthanandan, V., Hoelzer, D. T., & Allen, T. R. (2013). Radiation stability of nanoclusters in nano-structured oxide dispersion strengthened (ODS) steels. *Journal of Nuclear Materials*, *434*(1), 311-321.

[50] - He, P., Klimenkov, M., Möslang, A., Lindau, R., & Seifert, H. J. (2014). Correlation of microstructure and low cycle fatigue properties for 13.5 Cr1. 1W0. 3Ti ODS steel. *Journal of Nuclear Materials*.

[51] - Boutard, J. L., Dudarev, S., & Diegele, E. (2008). Radiation effects in structural materials for fusion power plants: the outcomes of the EU fusion program. In *Materials Issues for Generation IV Systems* (pp. 481-500). Springer Netherlands.

[52] - Klimiankou, M., Lindau, R., & Möslang, A. (2007). Direct correlation between morphology of (Fe, Cr)< sub> 23</sub> C< sub> 6</sub> precipitates and impact behavior of ODS steels. *Journal of Nuclear Materials*, *367*, 173-178.

[53] - Hedrich, H. D. (1989). Properties and applications of iron-base ODS alloys. *New materials by mechanical alloying techniques*, 217-230.

[54] - Hoelzer, D. T., Unocic, K. A., Sokolov, M. A., & Feng, Z. (2013). Joining of 14YWT and F82H by friction stir welding. *Journal of Nuclear Materials*, *442*(1), S529-S534.

[55] - Han, W. T., Wan, F. R., Leng, B., Ukai, S., Tang, Q. X., Hayashi, S., He, J. C., & Sugino, Y. (2011). Grain characteristic and texture evolution in friction stir welds of nanostructured oxide dispersion strengthened ferritic steel. *Science and Technology of Welding and Joining*, *16*(8), 690-696.

[56] - Farmerz, M. W., & Sanderson, S. (2013). Influence of heat input on friction stir welding for the ODS Steel MA956. *Friction Stir Welding and Processing VII*, 127.

[57] - Zinkle, S. J., & Snead, L. L. (2014). Designing Radiation Resistance in Materials for Fusion Energy. *Annual Review of Materials Research*, (0).

[58] - Pérez, M., Heidinger, R., Knaster, J., & Sugimoto, M. (2013, June). IFMIF: steps toward realization. In *Fusion Engineering (SOFE), 2013 IEEE 25th Symposium on* (pp. 1-8). IEEE.

[59] - Zinkle, S. J., & Möslang, A. (2013). Evaluation of irradiation facility options for fusion materials research and development. *Fusion Engineering and Design*, *88*(6), 472-482.

Figure 1

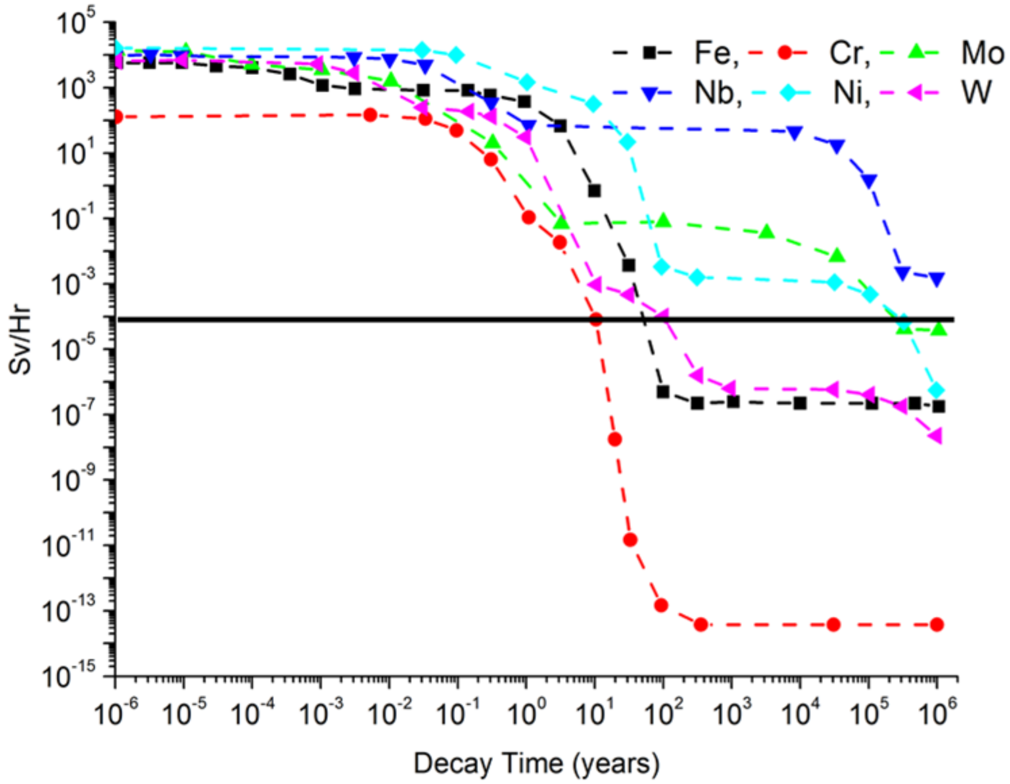

Figure 1 shows the level of irradiation from several common elements found in steels including Fe, Cr, Mo, Nb, Ni and W as a function of time after removal from a 3.6GW fusion power reactor following irradiation time of 5years; assuming they had received anticipated blanket front end irradiation doses (reconstructed from data in [5]). The black horizontal line in the image represents the ITER administrative limit for hands-on maintenance [6].

Figure 2

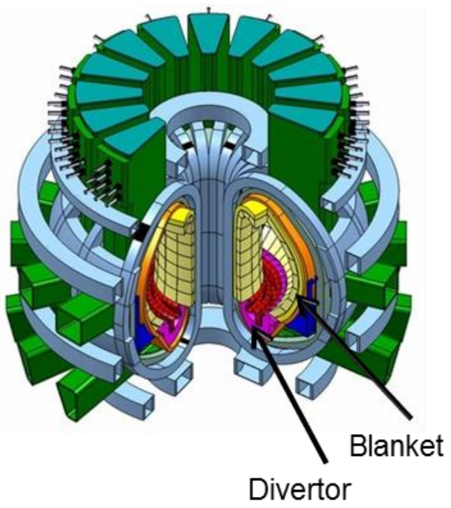

Figure 2. Artist's impression of DEMO with the locations of the blanket and divertor indicated.

Figure 3

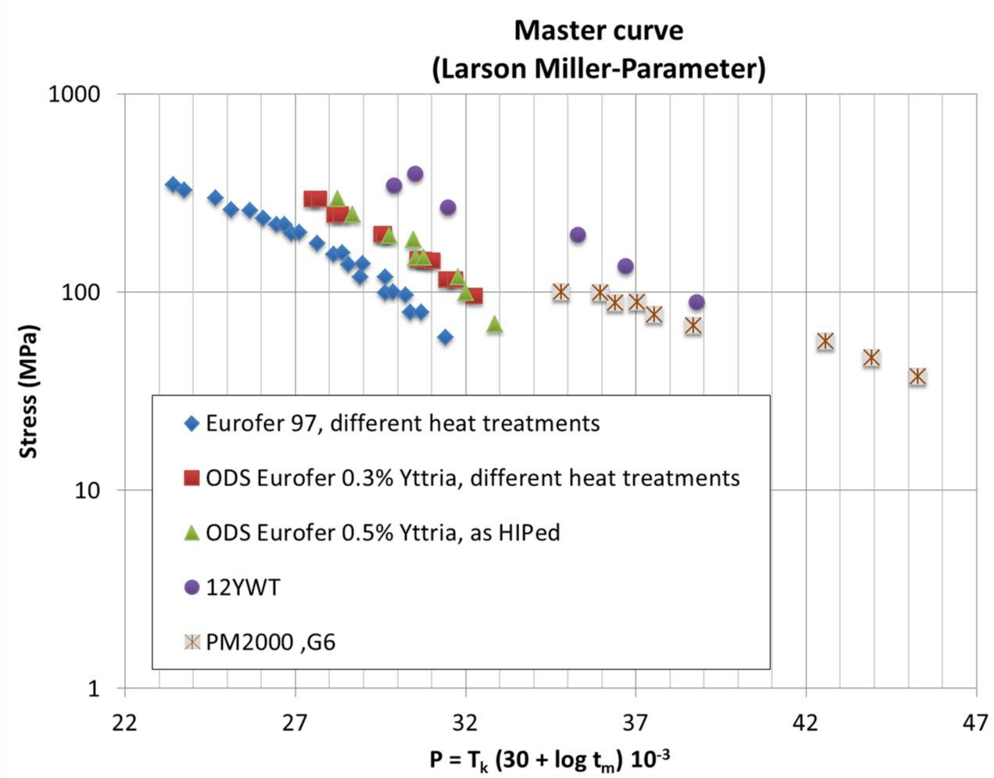

Figure 3. Creep strength of two grades of ODS Eurofer, PM2000, 12YWT (produced at Oak Ridge National Laboratory) and conventional Eurofer 97, Figure reproduced from [48].