Material: Ferritic/Martensitic: F82H steel
Property: Engineering Stress vs. Engineering Strain
Condition: Preirradiated and Irradiated
Data: Experimental

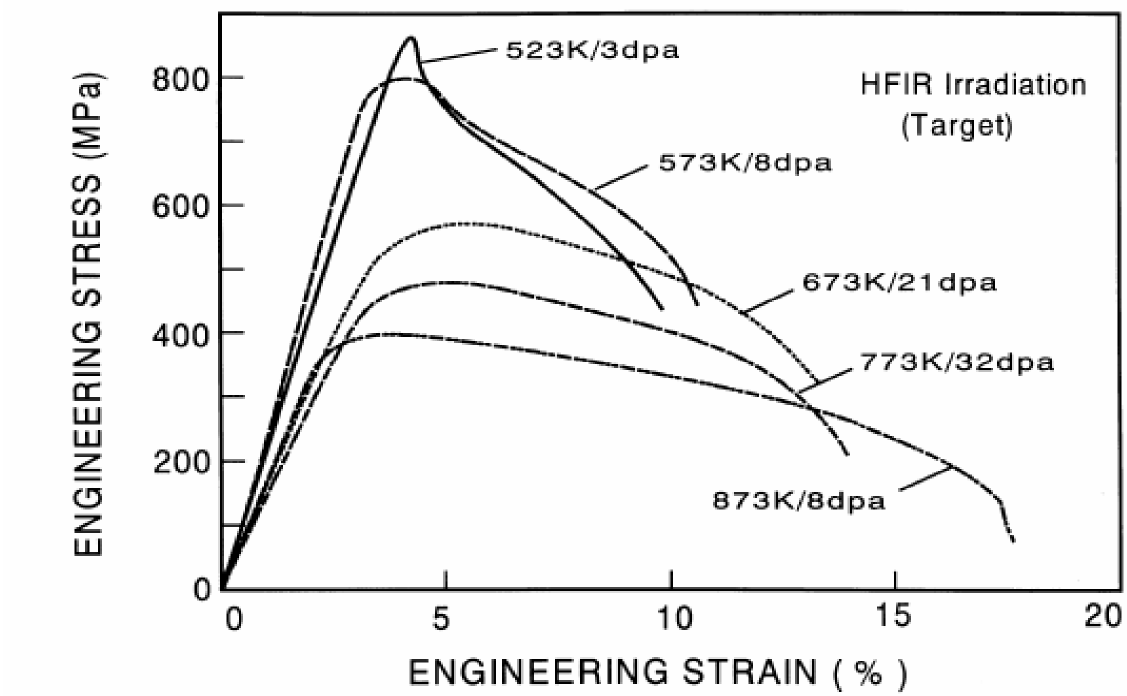

Fig. 5. Typical engineering stress–strain curves for the F82H steels before and after neutron-irradiation [8, 15].

Source:
Journal of Nuclear Materials, 1998, Volume 258-263, Page 193-204

Title of paper (or report) this figure appeared in:

Current status and future R&D for reduced-activation ferritic/martensitic steels

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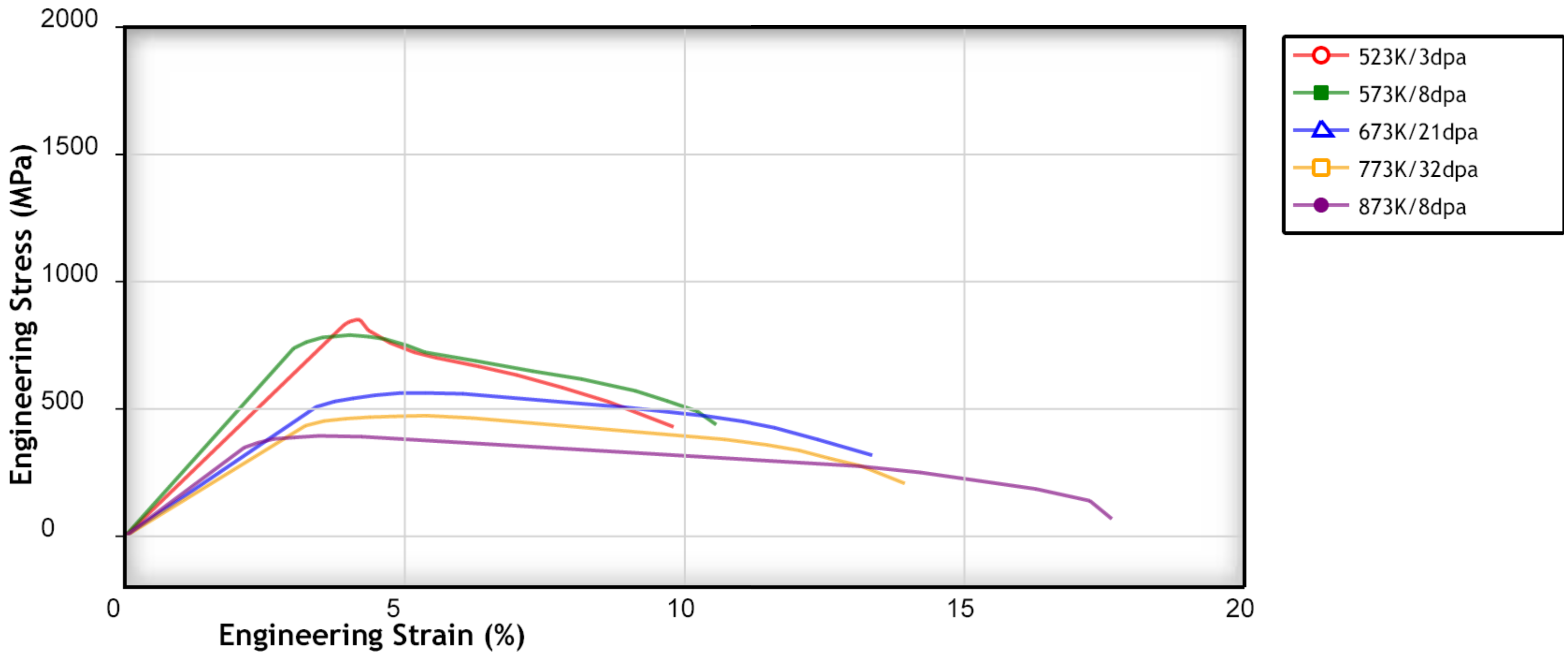
A. Hishinuma, A. Kohyama, R.L. Klueh, D.S. Gelles, W. Dietz, K. Ehrlich



Typical engineering stress-strain curves for the F82H steels before and after neutron-irradiation

Reference:

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Title: *Current status and future R&D for reduced-activation ferritic/martensitic steels*
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Plot Format:

Y-Scale: ◉ linear ○ log ○ ln
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Current status and future R&D for reduced-activation ferritic/martensitic steels

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Abstract

International research and development programs on reduced-activation ferritic/martensitic steels, the primary candidate-alloys for a DEMO fusion reactor and beyond, are briefly summarized, along with some information on conventional steels. An International Energy Agency (IEA) collaborative test program to determine the feasibility of reduced-activation ferritic/martensitic steels for fusion is in progress and will be completed within this century. Baseline properties including typical irradiation behavior for Fe–(7–9)%Cr reduced-activation ferritic steels are shown. Most of the data are for a heat of modified F82H steel, purchased for the IEA program. Experimental plans to explore possible problems and solutions for fusion devices using ferromagnetic materials are introduced. The preliminary results show that it should be possible to use a ferromagnetic vacuum vessel in tokamak devices.

1. Introduction

Reduced-activation ferritic/martensitic (RAM) steel is a primary candidate alloy as first wall and breeder blanket structural materials for the DEMO fusion reactor, because of better thermophysical properties when compared with the austenitic stainless steels currently considered as structural material for the ITER (International Thermonuclear Experimental Reactor [1,2]. The Fe–(7–9)Cr ferritic/martensitic steels are the most promising alloys from the point of irradiation resistance, especially the resistance to the phase instability and to the shift of the ductile–brittle transition temperature (DBTT) during or after irradiation as shown in Fig. 1 [3,4]. These alloys will be used for testing DEMO-relevant breeding blanket modules in the ITER [2]. To determine the feasibility of the reduced-activation ferritic/martensitic steels for fusion, an international collaborative test program, the so-called IEA round robin test [5–9], is underway under the auspices of the International Energy Agency (IEA) Implementing Agreement for a program of R&D on Fusion Materials.

Along with the collaborative international activities of the IEA, each of the participating parties in the IEA program has its own program for development of reduced-activation ferritic/martensitic steels. However, the domestic programs are not so different from each other, because the fusion reactor development itself is proceeding under international collaboration. In addition, bilateral collaborations, for example between the United States and Japan are also actively executed in the field of common irradiation experiments to effectively use fission reactors.

In this paper, activities of the IEA international collaboration on reduced-activation ferritic/martensitic steels, mainly the IEA modified F82H steel, are

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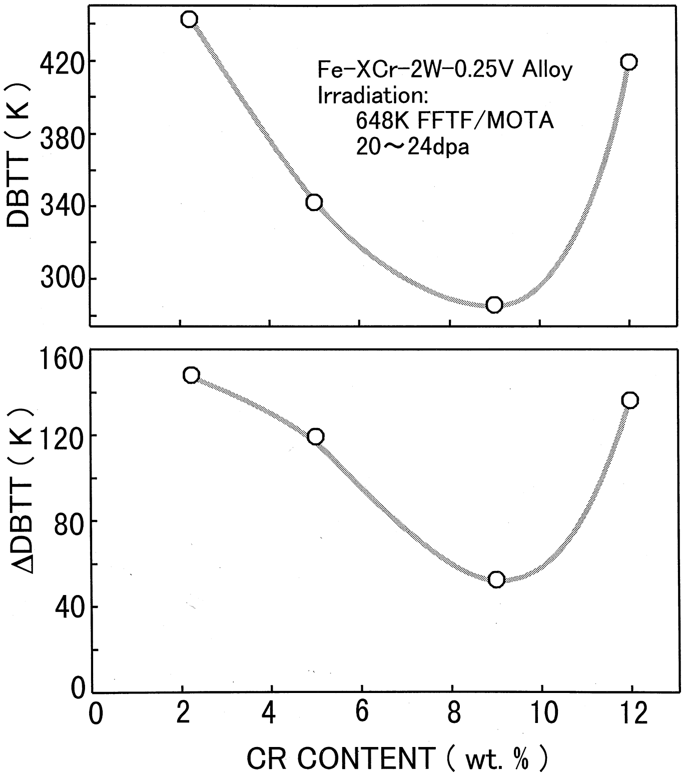

Fig. 1. Chromium-content dependence of the DBTT and the DBTT shift (ΔDBTT) for Fe–xCr–2W–0.25V alloys after neutron irradiation.

reported. Baseline properties including irradiation response are summarized to give a general view of RAM steels for fusion application. Recent progress for utilization technology and present and future work on steel development are also presented.

2. IEA collaborative program [4–9]

The IEA collaboration program grew out of an IEA workshop on ferritic/martensitic steels in Tokyo in 1992 [5]. The participants came from Japan, the European Union, the United States, and Switzerland. At the workshop, a combined test program on two large heats of reduced-activation steels was proposed. Subsequent meetings of a working group planned the composition of the large heats to be used to develop a comprehensive database on representative reduced-activation steels. The compositions proposed were similar to the Japanese F82H, an 8Cr–2WVTa steel jointly developed by the Japan Atomic Energy Research Institute (JAERI) and NKK, and to the JLF-1 steel, a 9Cr–2WVTa steel developed in the Japanese Universities fusion program, which is similar to a steel developed in the United States.

At a working group meeting at Oak Ridge in 1993 [6], specifications of the first IEA steel, a modified F82H composition, were discussed, and a tentative test program for this steel was identified. It was also decided that instead of obtaining one large heat of the former JLF-1, several smaller heats would be obtained to study the effect of elemental variations on properties. The Oak Ridge meeting was followed by a working group meeting in Stresa, Italy in 1993, where details on processing the first IEA reduced-activation steel, a 5-ton heat of a modified F82H composition were finalized. The composition, nominally Fe–8Cr–2W–0.2V–0.04Ta–0.1C, is shown in Table 1 as a typical example of RAM steels with those of the two 5-ton heats distributed.

The IEA modified F82H was produced and processed into plate material, and the plates were distributed to the participating parties. Other heats were also obtained: a 1 ton heat, a nominally Fe–9Cr–2W–0.2V–0.09Ta–0.1C steel (JLF-1 composition) and three 150-kg heats with the same chemical composition, except that one contains 20 ppm B, one contains 1% Mn, and one contains 2% Mn. In 1994, a second 5-ton heat of the IEA modified F82H was produced and processed into more plate material for base metal tests and material to

Table 1
Proposed chemical composition of the IEA modified F82H steel and those of the two 5-ton heats distributed. (Revision 3, September 30, 1993)

Elements	Compositions (wt.%)			
	Typical	Range	Heat 9741 [a]	Heat 9753 [a]
Cr	8.0	7.5–8.5	7.66	7.89
C	0.10	0.08–0.12	0.09	0.09
Mn	0.10	0.05–0.20	0.16	0.1
P	0.005	<0.01	0.002	0.003
S	0.002	<0.01	0.002	0.001
V	0.20	0.15–0.25	0.16	0.19
B	0.0003	<0.001	0.0002	0.0002
N	0.005	<0.02	0.007	0.007
W	2.0	1.8–2.2	1.96	1.98
Ta	0.04	0.01–0.06	0.02	0.04
Si	0.10	0.05–0.20	0.11	0.07
Cu	0.01	Lap [b].<0.05	0.01	0.01
Ni	0.03	Lap.<0.10	0.02	0.02
Mo	0.001	Lap.<0.05	0.003	0.003
Ti	0.05	0.004–0.012	0.01	0.004
Nb	0.00005	Lap.<0.0002	0.0001	0.0001
Sol.Al	0.01	Lap.<0.1	0.003	0.001
Fe	Bal.	Bal.	Bal.	Bal.
Co	0.005	Lap.<0.01	0.005	0.003
Ag	0.002	Lap.<0.05		
Sn	0.001	Lap.<0.004		
As	0.002	Lap.<0.005		
Sb	0.0005	Lap.<0.004		
O	0.005	Lap.<0.01		

[a] Average values.
[b] Lap.: Low as possible.
Other elements needed: Tb, Ho, Zr, Bi, Eu.

make weldments. TIG and electron-beam weldments were produced and distributed to the participants in 1995. In Europe a series of eight experimental RAM''s (OPTIFER, OPTIMAX, BATMAN and LA12 alloys) with a broad compositional range have been developed and tested [10].

The properties and studies planned for the large heats cover both irradiated and unirradiated materials and a main irradiation experiment is scheduled to be completed within this century. The experiments have proceeded from homogeneity tests in mechanical properties and chemical composition in location and rolling orientation for the distributed IEA modified F82H [8,9,11] and now have progressed to mechanical characterization and irradiation experiments.

3. Results on baseline properties

3.1. Thermophysical properties

Ferritic steels have advantages in thermal conductivity, swelling resistance, and reduced activation compared with those of the austenitic stainless steel. The thermal stress factor for ferritic steel is approximately three times larger than that of austenitic stainless steel at temperatures less than 600 K [12]. The difference results in a large difference in thermal stress generated in a first wall made of these steels. For the ITER conditions, where the maximum heat flux is estimated to be 0.6 MW/m^2, the possible wall thickness is estimated to be only about 2 mm for austenitic steel, while it is more than 10 mm thick for ferritic steels [13]. The limit is 5 mm when the maximum flux is 1 MW/m^2 for the SSTR (Steady State Tokamak Reactor), which is an example of a design for a DEMO reactor [14]. In this estimation, irradiation effect is not considered, so actually the wall thickness may be limited more severely.

3.2. Low activation properties

Another advantage of ferritic steels is that it is relatively easy to modify the composition to low-activation materials from a point of waste management considerations. The main constituent elements, Fe and Cr, meet the criteria for shallow land disposal, because small amounts of undesirable alloying elements like Mo and Nb can be relatively easily replaced by other desirable ones, for example, W, V and Ta [15]. Fig. 2 shows radioactivity decay curves of activation for reduced activation ferritic and conventional steels when they are used as first wall and blanket materials to an integrated wall loading 12 MWy/m^2 [16]. The dose rate of the RAM steels decays about eight orders and more in magnitude from 5×10^4 Sv/h immediately after the shut down of the reactor to near the hands-on level after 100 year cooling. The level of hands-on limit (2.5×10^{-5} Sv/h) allows definitely recycling and waste disposal without further protection [17], though the criteria may vary from country to country. This gives confidence that the more stringent conditions regarding very low allowable concentrations of unwanted elements can in future be achieved by advanced steel making technologies. The values above the low level waste (LLW) limit after 100 year cooling come from ^{60}Co converted from Ni and Co. Uncertainty may require further reduction of the contents of N, Nb, and Mo. Usually ^{60}Co is a main isotope to produce high radioactivity when fluence is less than 7 MWy/m^2, but at higher fluences than 20 MWy/m^2, ^{94}Nb from Nb, ^{93}Mo from Mo, and ^{193}Pt and ^{192}Ir from W produced by multi-step nuclear reactions become the main isotopes [18].

3.3. Magnetization

Ferromagnetism is one of the intrinsic features of ferritic steels, and it is a concern because it could produce to give large effects on plasma generation and control due to generating errors in the magnetic field and limiting the magnetic sensors. Fig. 3 shows the hysteresis curves at room temperature for the IEA modified F82H steel and the saturation magnetic flux

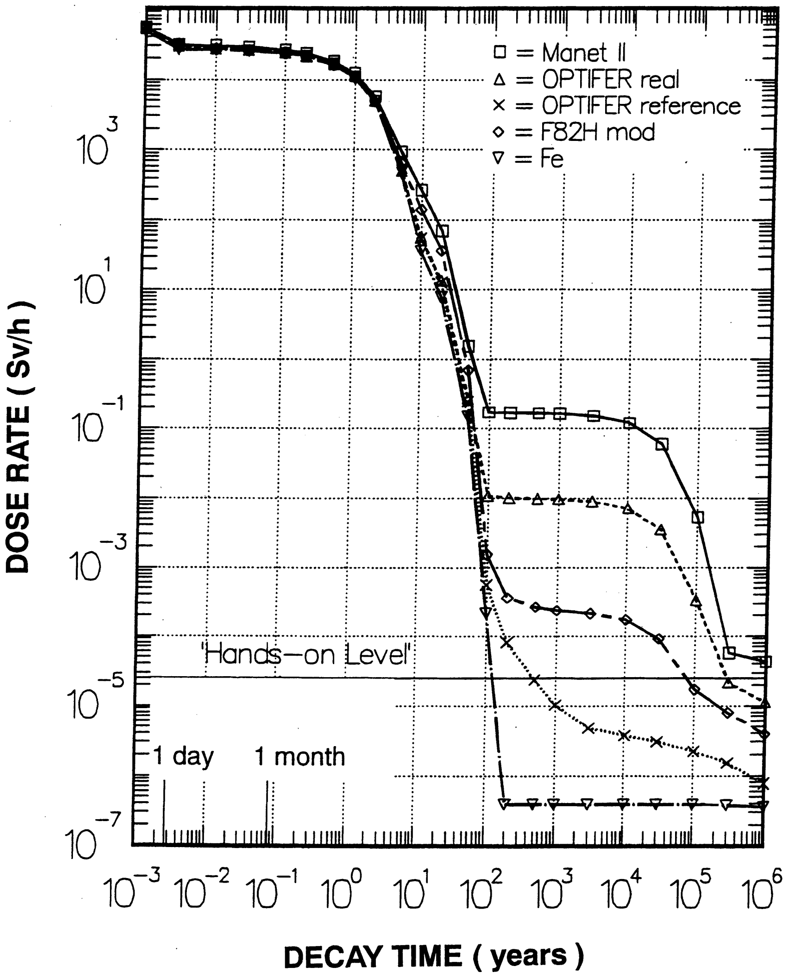

Fig. 2. Post-shutdown radioactivity for various ferritic steels and iron following exposure at the first-wall region to a wall loading of 12 MWy/m^2 [16].

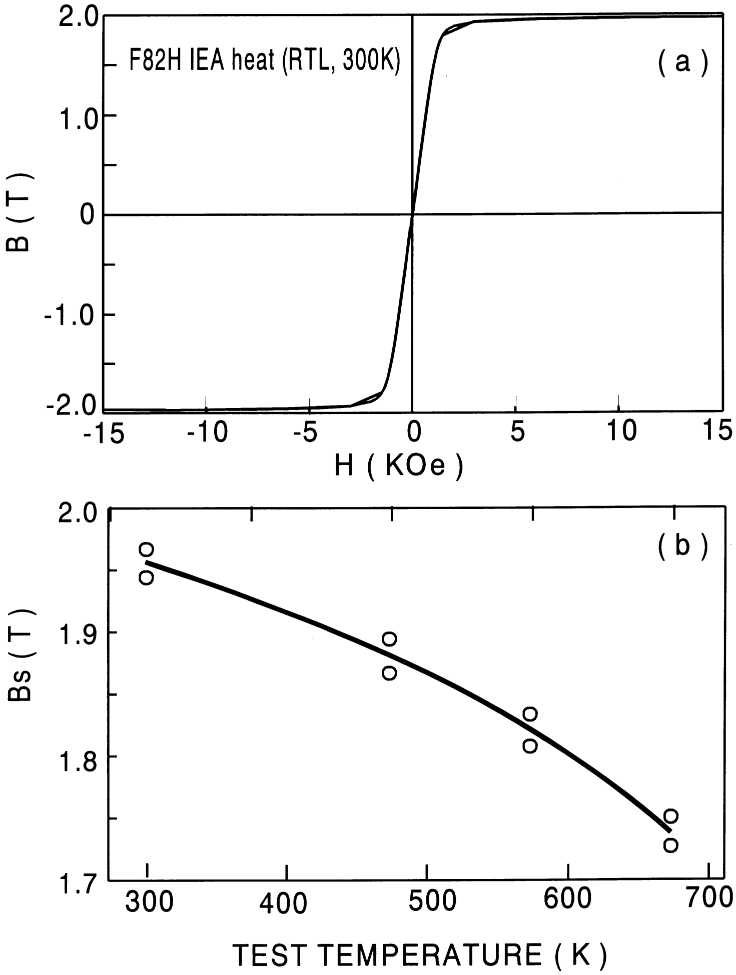

Fig. 3. Hysteresis curve of the IEA modified F82H steel at 300 K (a) and temperature dependence of saturated magnetic flux density (b) [12].

density as a function of test temperature [12]. The saturation magnetic flux density of about 1.95 T at room temperature decreases with increasing temperature and it is about 1.75 T at 673 K. Similarly, the residual flux density decreases with temperature from 0.21 T at room temperature to 0.17 T at 673 K. The discussion of this subject will be expanded in a later session.

3.4. Irradiation response

Tensile properties, yield stress and total elongation of the F82H steel after neutron-irradiation are summarized in Fig. 4 as a function of irradiation and test temperature, and they are compared with unirradiated properties [19]. Irradiation was carried out below 873 K up to doses of 3–34 dpa in the Oak Ridge Research Reactor (ORR) or the High Flux Isotope Reactor (HFIR). Increase in yield strength after irradiation was observed below 650 K and the maximum increase was around 523 K. At temperatures above 673 K, no hardening was observed. Softening due to high temperature irradiation generally occurred above 773 K [4].

Corresponding to this yield stress behavior, total elongation decreases after irradiation at temperatures below 650 K, and little degradation was observed at higher temperatures, as seen in Fig. 4(b). Even at the minimum around 523 K, a relatively large total elongation, about 8%, was observed, however the uniform elongation was very small. The dose dependence of the hardening and softening is ambiguous, but the changes seem to be saturated at less than 10 dpa. Typical engineering stress–strain curves for F82H steel irradiated to 3–32 dpa at 523–873 K are shown in Fig. 5 [19]. Irradiation at temperatures less than 573 K results in a greater than one-and-one-half fold increase in yield stress, and after yielding, the characteristic yield drop is

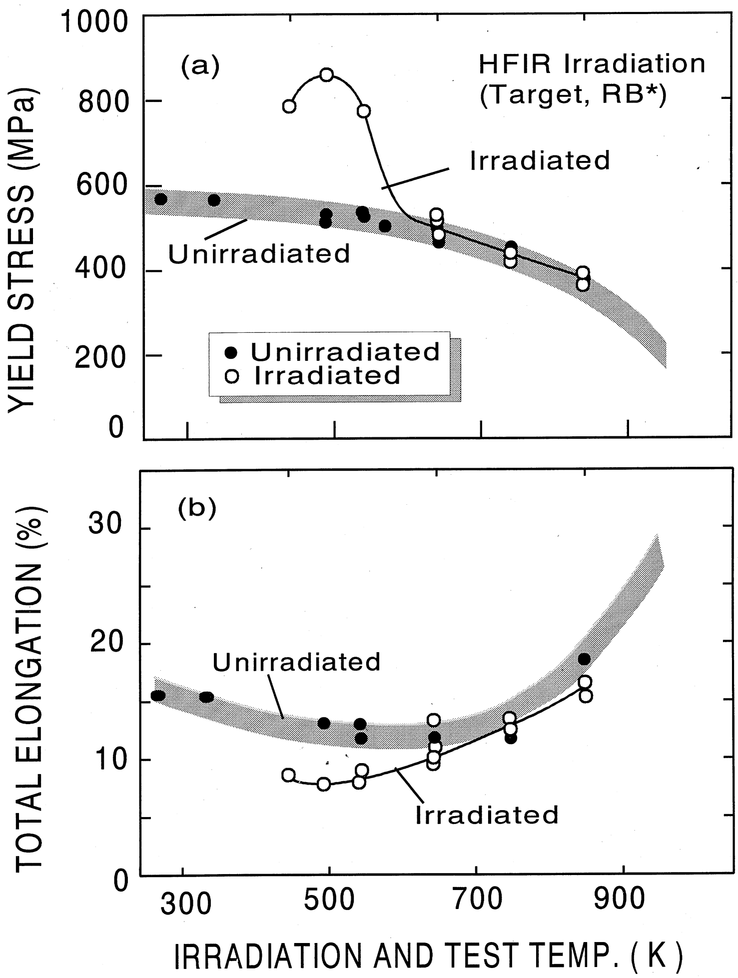

Fig. 4. Tensile properties of the F82H steel after irradiation to 3–34 dpa plotted against irradiation temperature [15].

apparent with small work hardening capability. The applied load falls rapidly and failure occurs after about 10% total elongation. The strain to necking is very small. At irradiation temperatures between 673 and 773 K, little irradiation effects are seen; the similar deformation behavior and work hardening capability to those of unirradiated materials are exhibited.

Ductile–brittle transition temperature (DBTT) has been a major concern because many studies have shown that the DBTT for ferritic steels increases rapidly with irradiation to above room temperature, even though (7–9) Cr ferritic steels have the lowest DBTT before irradiation. Fig. 6 shows change in Charpy energy to rupture for reduced-activation ferritic steel 9Cr–2WVTa before irradiation and after irradiation at 638 K up to 7, 13 and 28 dpa in HFIR [20]. The DBTT shift due to irradiation was relatively small and the dose dependence was not so large. Similar results were observed in ferritic steels irradiated in HFR, indicating that saturation in the DBTT shift with fluence appeared in the reduced activation steels F82H and 9Cr–2WVTa irradiated at 573 K to 1 dpa, while in some conventional ferritic steels with Mo and without Ta, the DBTT still increased with fluence up to 5 dpa [9,10]. It can be generally said that the reduced-activation ferritic steels with Ta appear to have better resistance to the DBTT shift due to irradiation than do the conventional ferritic steels without Ta [9,20]. One reason for this may be that the Ta addition decreases the prior austenite grain size [21]. Based on TEM and atom probe studies, it was concluded that Ta in solution leaded improved properties, and the DBTT would increase with fluence if Ta is removed from solution during irradiation. However, the mechanistic effect of Ta in solution is not fully understood.

Irradiation conditions are also very important, especially irradiation temperature [22,23] and the neutron energy spectrum [24,25]. The DBTT shift due to irradiation depends strongly on irradiation temperature and is

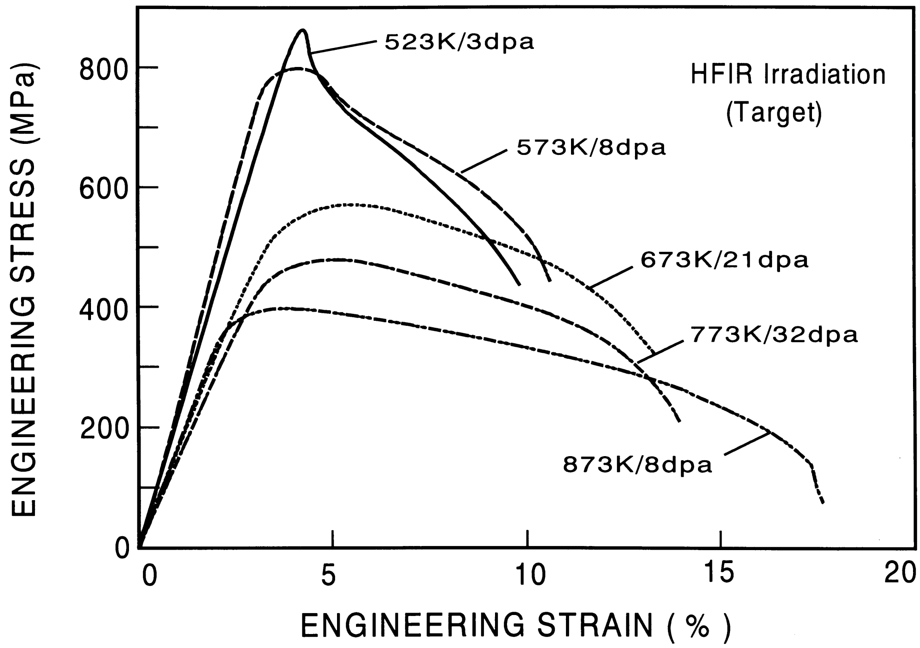

Fig. 5. Typical engineering stress–strain curves for the F82H steels before and after neutron-irradiation [8,15].

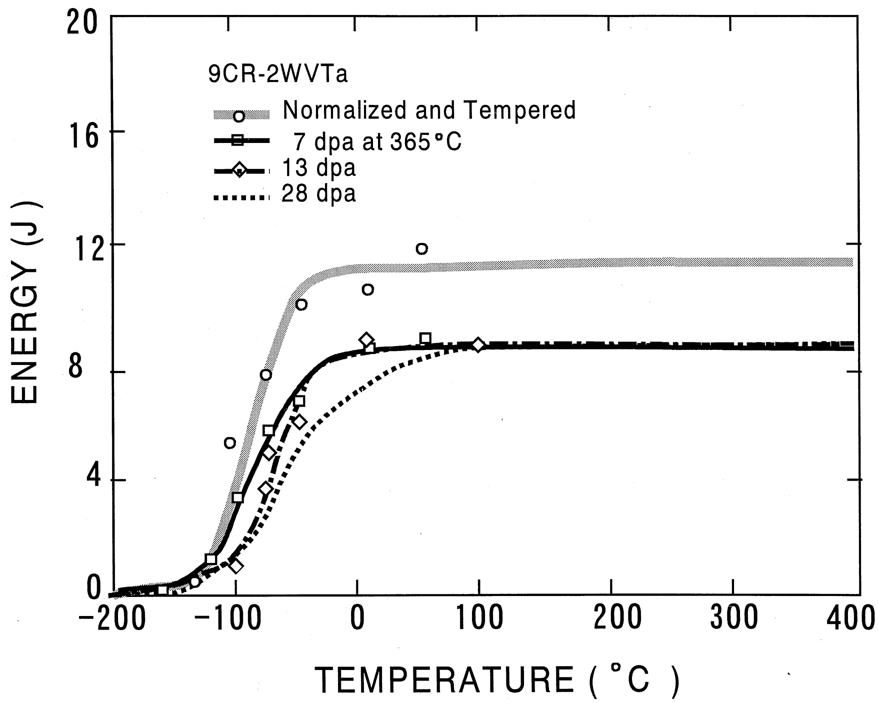

Fig. 6. Change in Charpy energy to rupture for 9Cr–2WVTa after neutron-irradiation at 638 K to 7, 13 and 28 dpa [20].

also dependent on alloy composition. An effect of helium on the DBTT shift has been demonstrated in Ni-doped and undoped 9Cr–1MoVNb steel irradiated in EBR II and HFIR [24]. The saturation in DBTT shift for the undoped 9Cr–1MoVNb steel after irradiation near 673 K in EBR II, where there was no effective helium production, appeared at the 55 K level after 13 dpa. While for the steel irradiated in HFIR to 40 dpa with 40 appm helium, a shift in DBTT of about 205 K was observed [25]. A similar helium effect was also observed in reduced activation ferritic steels in helium-ion irradiation experiments [26] and in B-doped specimens neutron-irradiated to around 0.5 dpa (^{10}B is transmuted to helium) [27]. Simultaneous helium and displacement effects have been investigated by comparison between RAM steels irradiated in HFR and in alpha-particle cyclotron [28], in which a much higher DBTT shift has been measured after helium implantation as shown in Fig. 7. The helium shows a significant contribution of microstructural development of loops and precipitates to hardening. However, this effect could not yet be translated quantitatively into DBTT shift for neutron-irradiated materials. More experimental and theoretical studies are needed to fully understand how helium affects the DBTT shift, and how the helium produced in these simulation studies in fission reactors apply to a ferritic steel in a fusion environment.

The transition temperature shifts measured in Charpy tests, however, are not to be directly applied to the first wall component of a fusion reactor, as pointed out by Enmark et al. [29], because such tests and fracture mechanism methods have been developed for light water reactors (LWR) pressure vessel integrity. The linear elastic fracture mechanism is justified by the relatively large thickness of the LWR components. On the other hand, in ferritic/martensitic steels for fusion reactor application, the first wall is relatively thin, and detectable defects are expected to be relatively shallow surface cracks. The combination of small thickness and shallow surface cracks can considerably decrease the constraint on the crack tips relative to large deeply cracked specimens. From these aspects, the use of CVN specimens should be reconsidered. It may be necessary to develop new evaluation methods [29]

Fracture toughness can indicate crack initiation and propagation resistance of the ferritic/martensitic steels. An example of the change in fracture toughness of F82H and HT9 ferritic steels after neutron-irradiation at 363 and 523 K to 3 dpa is shown in Fig. 8, compared with that of austenitic stainless steels [19]. A relatively high fracture toughness around 200 MPa $m^{1/2}$ after irradiation at 523 K to 3 dpa was observed, although this temperature generally results in large irradiation hardening. The value is nearly in the same range as the austenitic steels.

Irradiation creep in some cases could result in larger creep deformations in the lower temperature regime applicable to such fusion devices as the SSTR with a water coolant system. Fig. 9 shows the temperature dependence of creep strain (% $MPa^{-1}dpa^{-1}$) for ferritic steel HT-9 irradiated at 473, 503 and 673 K in the ORR to 7 dpa, followed by irradiation in the HFIR to 19 dpa [30,31]. The data show that the irradiation creep rate is linear in fluence and stress without any strong irradiation temperature dependence, although swelling-enhanced nonlinear creep was observed in high

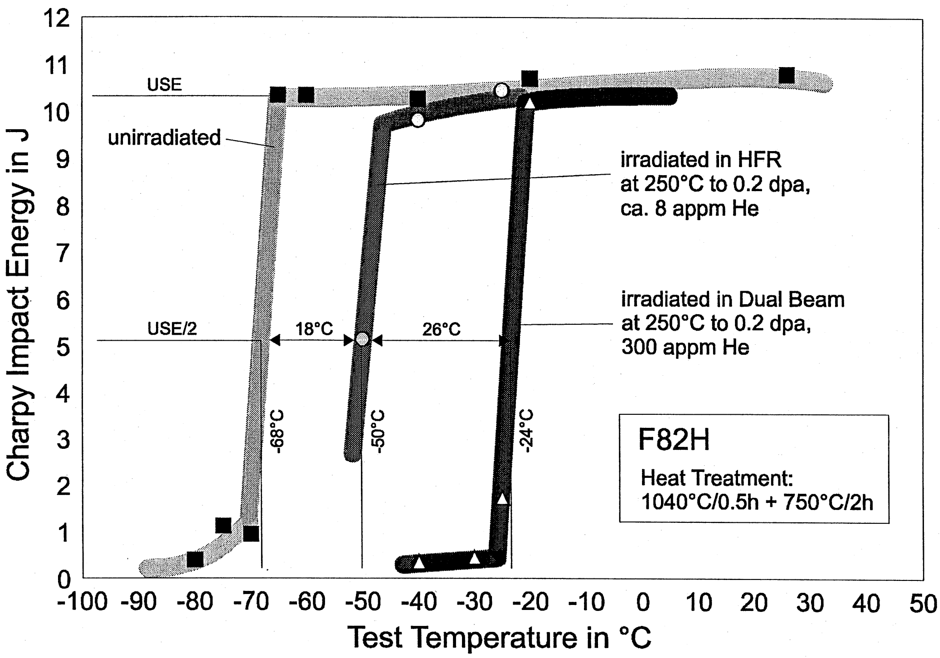

Fig. 7. Changes in Charpy energy to rupture for F82H irradiated in HFR and in alpha-particle cyclotron [28].

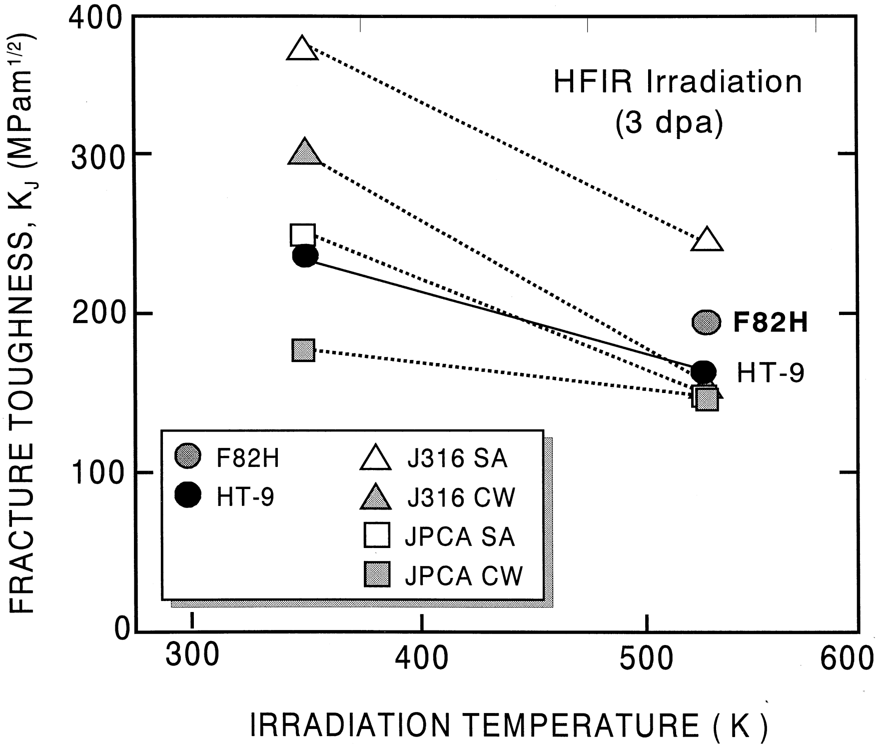

Fig. 8. Fracture toughness of the F82H steel after neutron-irradiation at 523 K to 3 dpa compared with that of HT-9 and the austenitic stainless steels J316 and JPCA irradiated at 363 and 533 K to 3 dpa [15].

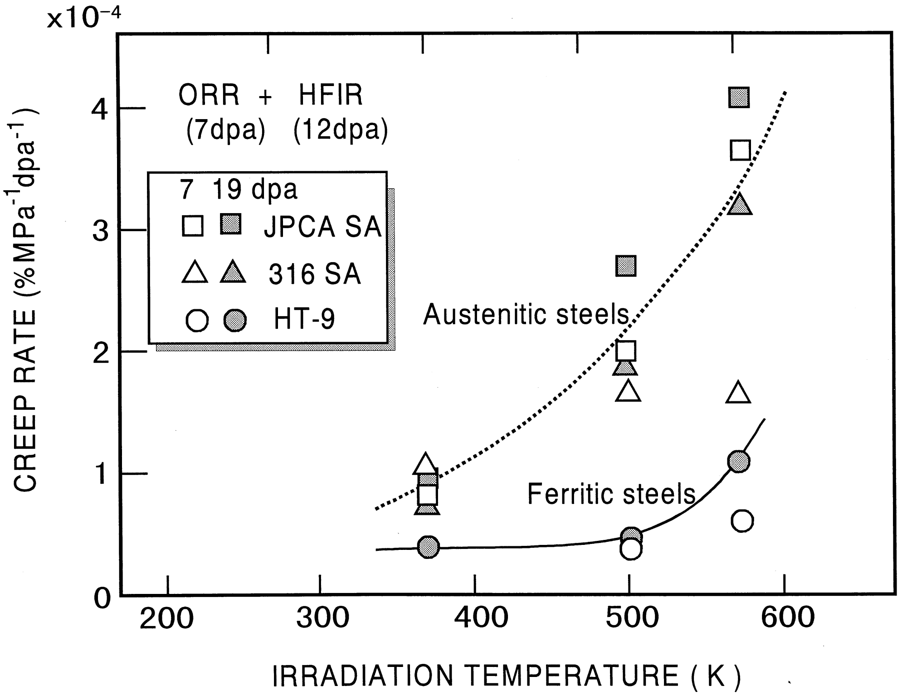

Fig. 9. Temperature dependence of creep strain for ferritic steel HT-9 compared with that for austenitic stainless steels.

temperature irradiation [32]. Ferritic steels have lower irradiation creep rates by about an order of magnitude compared with that of austenitic stainless steels, and the creep rate is given as 0.5×10^{-4}% $MPa^{-1}dpa^{-1}$. The value is not so large, but the strain would become considerable after higher doses, especially at lower temperatures if the linear dependence still exists.

Fatigue properties under irradiation may be one of the key parameters to determine the feasibility of reduced activation ferritic/martensitic steels for fusion, although very limited amounts of data are presently available [33,34]. Fatigue could introduce microstructural changes, which differ from those induced by irradiation. This may affect creep, swelling, and segregation phenomena during irradiation. However, such phenomena have not yet been investigated.

4. Steel modification and improvement

Based on data and knowledge about properties of candidate reduced activation ferritic/martensitic steels, some of which are estimated and others are anticipated, the design window for a water coolant system can be drawn roughly as shown in Fig. 10. The lower temperature may be limited by irradiation-induced embrittlement, like the DBTT shift due to irradiation. At higher temperatures, softening or loss of creep strength during irradiation, are key factors in determining the upper temperature limit. Or in the case where the steady stress on the wall materials is not so large, thermal fatigue becomes a very important factor, although not indicated in Fig. 10. In the intermediate temperature region between 373 and 723 K, irradiation hardening occurs, but there appear to be no effective softening and no effective creep strain, and the window may be open to higher fluence with some limitation due to a helium effect on the DBTT shift.

According to the SSTR design study, high strength at elevated temperatures is required from safety and economical points of view [35]. Elevated-temperature strength is also required for application with liquid–metal and helium–gas cooled systems according to design study conducted in European Union [2]. To meet these requirements, modification and improvement studies of the ferritic/martensitic steels are required to increase the strength at elevated temperature and increase the corrosion resistance. The modification study has focussed on effects of minor elements such as C, W, Ta, Ti and others, including heat-treatment effects based on the Fe–(7–9)Cr alloys [20]. Oxide dispersion strengthened ferritic/martensitic steels (ODS) can also be expected to have high-strength at elevated temperature. The ODS-type steels were basically developed for FBR

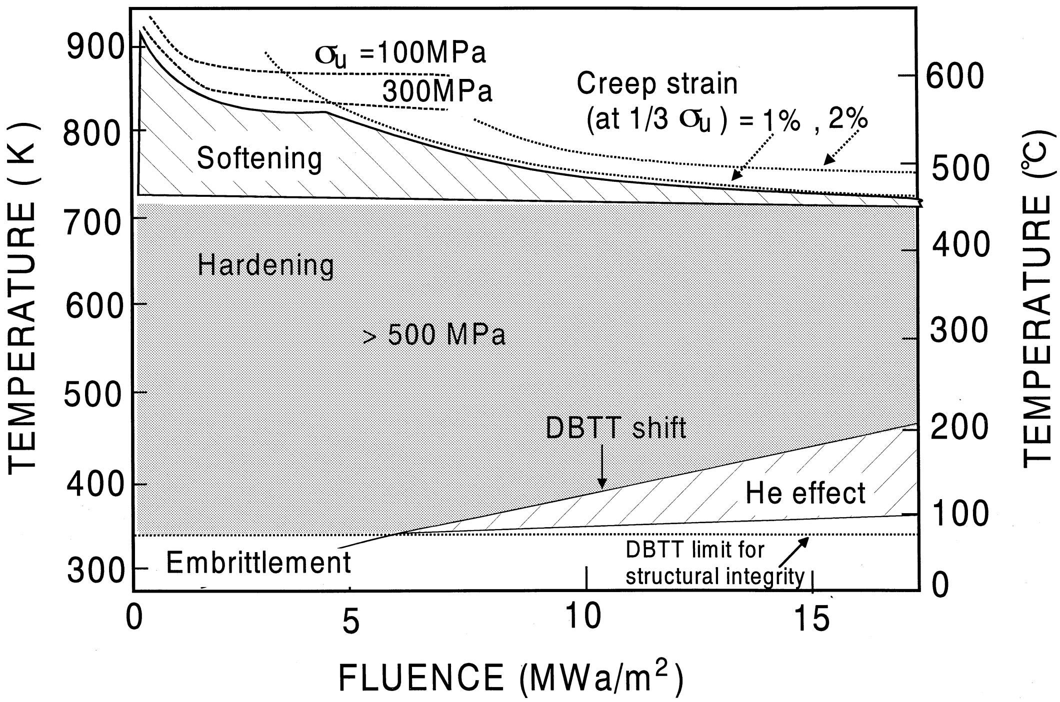

Fig. 10. Schematic design window for a tokamak type fusion reactor with water-cooling system using reduced-activation ferritic/martensitic steels.

fuel cladding materials are expected to be swelling resistance and have superior strength at elevated temperature [36–39]. Recently it has been shown that the ODS-type steel has relatively good ductility and also good resistance to DBTT shift due to irradiation; in fact, recent data show that the DBTT shifts to a lower temperature after irradiation [40]. Moreover, the fine microstructure of this type steel is expected to have a good resistance to irradiation-induced embrittlement including the helium effect. Based on the current material technology, the ODS-type steel may be used as armor tiles on the ferritic/martensitic steels.

5. Specific problems of utilizing ferritic/martensitic steels

It has long been a concern whether the ferritic/martensitic steels can be used for structural components of fusion devices concerned with the influence of magnetic materials on plasma behavior, because little knowledge existed on this problem. The main problems involved the generation of undesirable magnetic fields and magnetization forces. However, no tokamak experiment has been done using ferromagnetic materials. Recently material–plasma compatibility experiments using JFT-2M (JAERI Fusion Torus-2M) tokamak reactor, to explore the problems and find a solution for fusion reactors using ferritic steel as a structural material were planned [41]. Also, preliminary studies using HT-2 (Hitachi tokamak) small tokamak without an NBI system have been started [42,43]. By installing F82H steel inside the HT-2 vacuum vessel surrounding the plasma discharge area, unintentional magnetic fields were produced due to the F82H first wall. However, it was shown that the magnetic fields were not strong and that they did not disturb the plasma discharge. In addition, the magnetized materials can compensate the field ripple of discrete toroidal field coils. It was also shown that the vacuum pressure was roughly identical with that of a stainless steel first wall. The outgassing rate from the F82H plates was small and the vacuum condition was good enough to get normal plasma discharge [44]. The study concluded that plasma discharge should be possible with a ferromagnetic vacuum vessel in a tokamak device. It is expected that this conclusion will also be applicable to larger tokamak devices, such as JFT-2M and ITER, because the effects on the plasma due to the magnetic field produced by the ferritic wall will be small.

Another specific concern for reduced activation ferritic/martensitic steels for fusion application as structural materials is for techniques of using a quasi-brittle materials. This is a common concern for several materials. Both ferritic and austenitic steels degrade to low ductility, with very small uniform elongation and work hardening capability. In order to use such low ductility materials in structural application, it is important to limit the maximum strain required under multiaxial stress. This requires more precise design by analysis of dominating deformation and fracture mechanism.

6. Summary

The reduced activation ferritic/martensitic steels Fe–(7–9)%Cr have been considered as candidate alloys for the DEMO fusion reactors, providing good thermal conductivity and reduced activation. The irradiation response of the steels has been investigated under both international collaboration and domestic fusion programs. The results of the exploratory work for the development of reduced activation ferritic/martensitic steels are very promising, especially where the improvement in fracture toughness and impact properties and the elimination of detrimental elements for the long-term activation are concerned. The forthcoming data on the irradiation behavior may confirm this positive trend. In addition, preliminary experiments on the effect of ferromagnetic properties of ferritic steels, show that the plasma discharge should be possible with a ferromagentic vacuum vessel in a tokamak devices including effects of surface and outgas properties of F82H steel.

However, small uniform elongation and work-hardening capability are a problem as is the case with austenitic stainless steels and vanadium alloys. Fracture toughness of small specimens in thickness for fusion application needs to be carefully evaluated, and must include helium effects on toughness. In order to enlarge the design window for tokamak fusion devices to allow for safe and economical reactor designs, further modification and development of the steels is required, especially to improve elevated-temperature strength including creep and fatigue resistance.

Acknowledgements

The authors thank colleagues participating in the IEA collaboration program of reduced activation ferritic/martensitic steels for their input to this review.

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