

Recent Progress in Reduced Activation Ferritic Steels R&D in Japan

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Abstract. The Japanese RAFSs R&D road map toward DEMO is shown. Important steps include high-dose irradiation by fission reactors, such as HFIR in ORNL, irradiation tests by 14 MeV neutrons in IFMIF and application to ITER test blanket modules to provide an adequate database of RAFS for the design of DEMO. Current status of RAFS development is also introduced. The major properties of concern are well within our knowledge and process technologies are mostly ready for fusion application. The RAFSs are now certainly ready to proceed to the next stage. Material database is already in hand and further progress is anticipated for the design of ITER test blanket. Oxide Dispersion strengthening (ODS) steels are quite promising for high temperature operation of the blanket system with potential improvements in radiation resistance of mechanical performances and of corrosion.

1. Material development road map to DEMO and beyond

The development of the structural material for the blanket is one of the most crucial issues in the fusion technology R&D. Reduced activation ferritic steels (RAFSs) have been recognized to be the prime candidate structural material for the fast realization of fusion DEMO reactor and beyond [1-3], while SiC/SiC composite materials and vanadium alloys might be attractive for their potential as high temperature structural materials. The austenitic stainless steel will be employed in ITER core components, but higher wall load makes them not suitable for DEMO because of their high swelling rate and high susceptibility to He embrittlement, and their low thermal conductivity. The Japanese RAFSs R&D road map toward DEMO (Fig. 1) is consistent with the scheduled operation of ITER test blanket and DEMO. Important steps include investigation of high-dose irradiation experiments by utilizing fission reactors, such as HFIR in ORNL, and irradiation tests by 14 MeV neutrons in IFMIF and application to ITER test blanket modules to provide an adequate database of RAFS for the design of DEMO, together with R&D of oxide dispersion strengthening (ODS) steels for high efficiency.

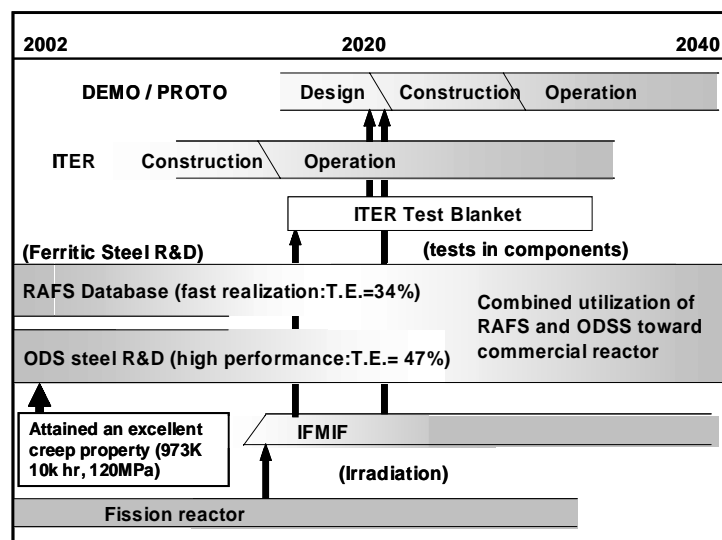


FIG. 1 RAFSs R&D road map toward DEMO.

2. Current status of RAFS development

2.1 Fabrication technologies and ferromagnetism

Japanese major candidate RAFSs, F82H [4] and JLF-1 [5], have been developed from the well-known 9Cr-1Mo heat resisting steel by replacing several alloy elements to reduce residual activity induced by fusion neutrons. Fabrication of RAFSs in an industrial large quantity has little difficulty. The requirement of reduced activation is exceptionally strict about the impurity control of the steels, and two 5-ton ingots of F82H [2] were successfully produced with the harmful elements such as Mo and Nb well below the required levels. Other fabrication processes, such as heat treatments, shaping and joining, of this kind of material are also well established. A mock-up of blanket structure was successfully fabricated with hot-isostatic pressing (HIP) process, as shown in Fig. 2.

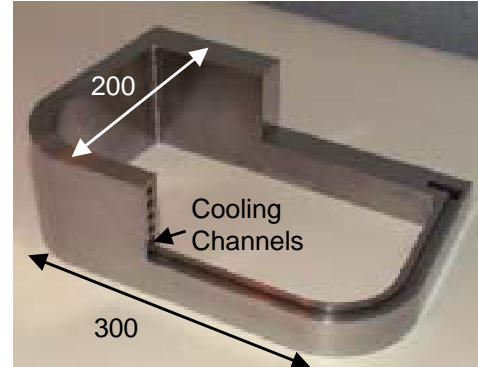


FIG. 2 A mockup with cooling channels in a shape of box with corners were successfully fabricated with F82H.

The influence of ferromagnetism on plasma operation has often been worried about. Recent operation tests of the JFT-2M with F82H plates in it show no deteriorating effects on plasma but reduction of toroidal field ripple [6].

2.2 Performance under irradiation: Design window

Ferritic/martensitic steels have been successfully used in FBR core components to a damage level of 100 displacement per atom (dpa) that corresponds to a wall loading of 10MW(annual)/m^2 . This is very encouraging for the application of RAFSs to the fusion core components. Radiation response of the Japanese RAFSs has been studied using fission reactors and ion accelerators in the domestic programs and international collaborations, which include the international collaboration of IEA RAFS working group and the Japan-US bilateral collaborations.

The domestic neutron irradiation facilities have been used to study irradiation embrittlement at low neutron dose at lower irradiation temperature ($<573\text{K}$) using relatively larger test pieces, where specimen size effect on mechanical properties has also been investigated. The Japan-US collaborations utilizing HFIR and FFTF have generated high dose data, while the IEA collaboration has promoted various property tests.

The Japan-US collaborative research for JLF-1 steel with use of FFTF/MOTA contributes irradiation database that demonstrates higher feasibility of the RAFS as a fusion structural material than 9Cr-1Mo [7] (Fig. 3). The ductile-brittle transition temperature (DBTT) significantly depends on irradiation temperature: the DBTT shift becomes large below 693K , and tends to saturate at 10 dpa. Above 693K , the shift is relatively small but softening becomes significant. These results suggest that radiation may limit operating temperature of RAFS in two ways. One is the softening at high temperature and the other is the

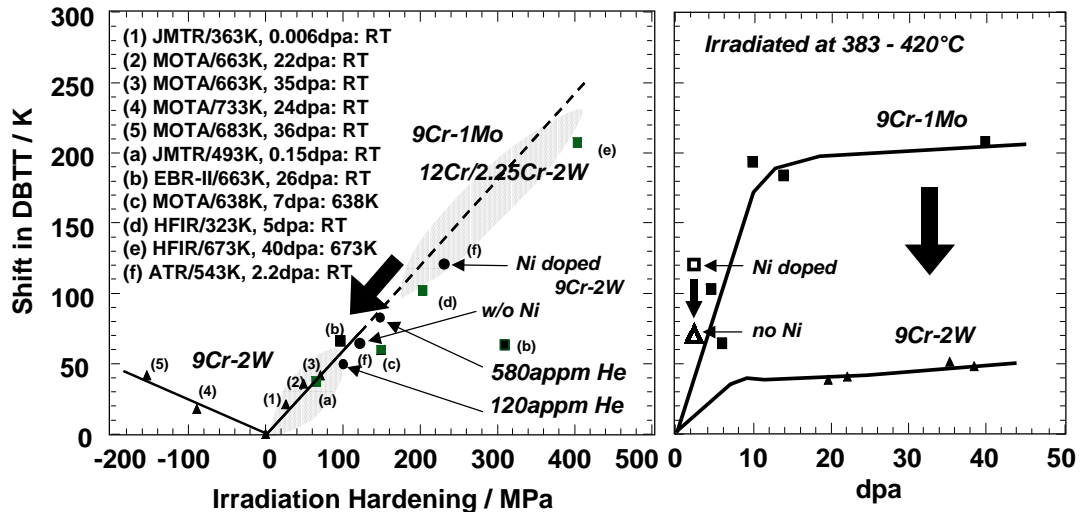


FIG. 3. Improvement of response to neutron irradiation of ferrite/martensitic steel by replacement of alloy elements for reduced activation.

embrittlement at low temperature. The former directly affects the energy conversion efficiency, and the latter does the integrity of structural component through the DBTT shift and the concern is the shift into the reactor maintenance temperature range (RT-100°C).

Another concern is transmutation helium (He) effects on the DBTT shift. Although helium might cause further degradation, no difference was observed in the DBTT shift between fast neutron irradiation and He ion implantation, as shown in Fig.3 [8]. The on-going Japan-US collaborative program will give the answer to the He-related concerns. The material response to dual and triple ion beam irradiation has also been investigated to simulate the effects of fusion environment, that is, displacement damage with existence of helium and hydrogen, although the damage rate is much higher than fusion environment. Microstructure examinations after ion beam irradiation indicated that the irradiation damage structure was influenced by the presence of H and He and also by the microstructure before irradiation [9].

Radiation-induced softening that limits the highest operation temperature is critical for RAFSs. Many efforts to improve high temperature strength have been made: 1) alloying control [10], 2) thermo-mechanical treatment, 3) ODS steel application [11]. The formation of He bubbles at elevated temperatures will be another concern, although the previous ion-irradiation database suggests that RAFSs have much better resistance to He bubble-induced embrittlement than austenitic steels.

For the IEA collaborative work, plates including TIG- and electron-beam- (EB-) welded ones were prepared from the two 5-ton F82H ingots, and distributed to the participating organizations. Distribution of the material with a good traceability enabled to compare obtained data in different organizations. The IEA heat F82H has become a reference material in many organizations. The JLF-1 steel has also been tested in the Japanese universities for demonstrating the superiority of martensitic structure as a fusion structural material. Performance data of Japanese RAFSs before and after irradiation are thus accumulated by the effort of many organizations in the world, which are now being compiled into a database [12]. The design window (Fig. 4) clearly shows the advantages of the RAFS over 9Cr-1Mo steel for fusion application.

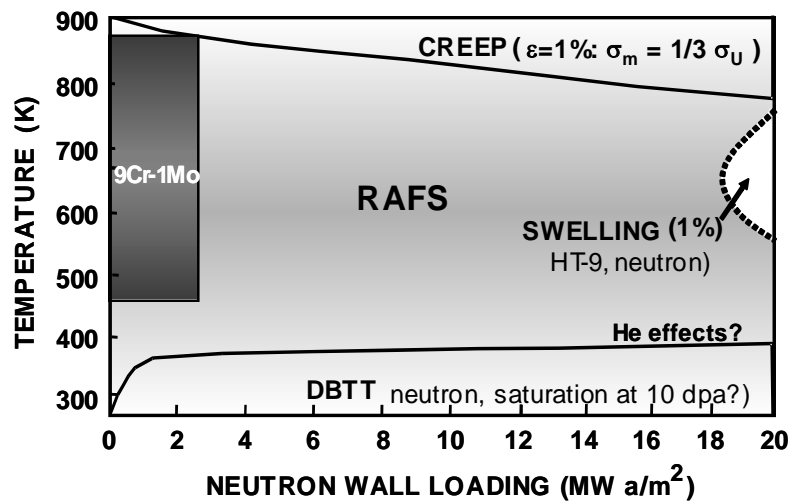


FIG. 4 Design window of RAFS (JLF-1) and 9Cr-1Mo steel: The replacement of Mo, Ni and Nb with W, Mn and V resulted in much wider window under neutron irradiation.

3. Toward the ITER test blanket module

The RAFSs are now certainly ready to proceed to the next stage. Material database is already in hand as mentioned above. HFIR irradiation in the Japan-US collaboration will yield information about the fabrication tolerances of the RAFSs in terms of composition and heat treatments. Radiation effects on welded or HIPed material, and on corrosion resistance will be also examined. Performance of welded F82H after the irradiation in HFIR to 5 dpa at 573K [13] indicated that the weld joint type specimen including weld metal and heat affected zone in the gauge section shows less radiation hardening.

Development of temperature controlled irradiation rigs in HFIR enables to investigate the varying temperature irradiation, and the results shows the varying temperature irradiation never causes acceleration of degradation in tensile properties (Fig. 5).

Designing and licensing of ITER test blanket modules using RAFS are expected to be highly affirmative but still require some irradiation tests using 14 MeV neutrons from International Fusion Material Irradiation Facility (IFMIF) to confirm several key issues. IFMIF provides the best simulation of fusion environment for the material tests and also for the component tests. Component tests in IFMIF will be followed by the operation of the ITER test blanket module, where the blanket system performances will be verified as an integral system containing coolant, tritium breeder and neutron multiplier under real fusion irradiation environments.

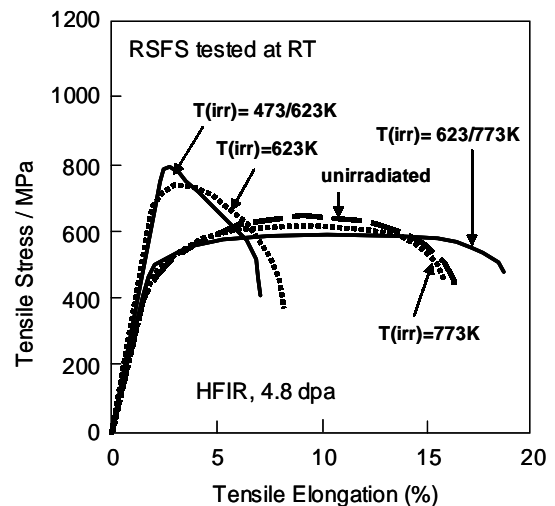


FIG. 5 Stress-strain behaviors of RAFS after constant and varying temperature irradiation in HFIR, indicating that varying temperature irradiation has no influence on tensile properties..

4. Expected future progress in RAFS toward and beyond DEMO

Judging from the accumulated knowledge so far obtained, RAFS can be employed up to a fluence level of the peak DEMO wall loading (10-15 MWa/m²) at temperatures below 773K [2]. DEMO should demonstrate, however, that fusion is an attractive option in many aspects, such as energy conversion efficiency and environmental safety. Higher operating temperature and less exchange frequency of blanket lead to higher efficiency and reduction of nuclear disposal.

The ODS steels [11] are quite promising for high temperature operation of the blanket system with potential improvements in radiation resistance of mechanical performances and of corrosion. Figure 6 demonstrates their predominance in high temperature strength among ferritic/martensitic steels. The higher operating temperature enabled by ODS steels is also beneficial in super critical water-cooled systems. Application of ODS steel for blanket structural component provides a larger design tolerance and higher energy conversion efficiency. For this purpose, development of joining technology of these materials is inevitable.

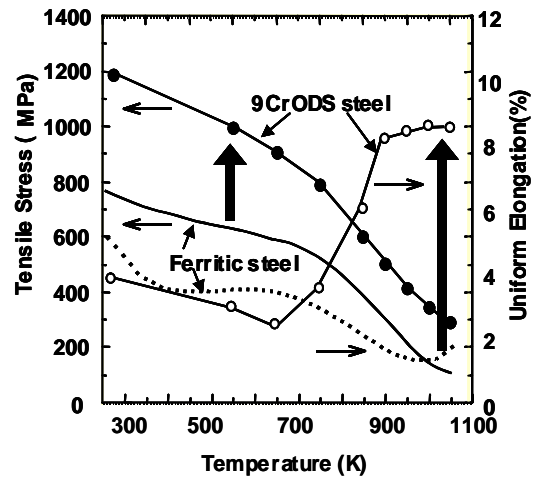


FIG. 6 Dependence of tensile properties of a ferritic steel and a ODS steel, indicating that the ODS steel is significantly superior to the ferritic steel in high temperature performance.

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