SUMMARY OF THE SIXTH INTERNATIONAL CONFERENCE ON FUSION REACTOR MATERIALS, STRESA, ITALY, SEPTEMBER 27–OCTOBER 1, 1993

The International Conference on Fusion Reactor Materials (ICFRM) addresses open material questions in thermonuclear fusion devices. The ICFRM meetings traditionally offer a worldwide forum for the presentation of the latest results as well as the opportunity for international material scientists to collect information in this challenging field of research.

The conference was begun in 1984 by Professor Hasiguti in Tokyo and is organized every 2 yr. The previous conferences were held in Chicago, Illinois (1986); Karlsruhe, Germany (1987); Kyoto, Japan (1989); and Clearwater, Florida (1991). The Sixth ICFRM (ICFRM-6) took place in the Congress Centre of Stresa, Italy. Stresa is located on the western shore of Lago Maggiore in a panoramic and picturesque setting facing the Borromean Islands. The town has about 5000 inhabitants and is an internationally renowned resort area. ICFRM-6 was organized by the Institute for Advanced Materials of the Commission of the European Communities and was chaired by Dr. P. Schiller. The conference included contributed and invited papers, which were presented in plenary, oral, and poster sessions. All the manuscripts will be refereed and after acceptance will be published in a special volume of the *Journal of Nuclear Materials*.

More than 600 abstracts were submitted to ICFRM-6, which represents an increase of almost 50% from ICFRM-5. Finally, 431 contributed papers were scheduled (55 orals and 376 posters) from 21 countries. Figure 1 shows the geographic distribution of the presenting authors. Most of the papers (93%) were submitted by the four partners of the International Thermonuclear Experimental Reactor (ITER) project, i.e., Japan, the European Communities, the United States, and the Russian Federation.

Contributions from Japan came mainly from Japan Atomic Energy Research Institute (JAERI), the University of Tokyo, Kyushu University, and Tohoku University.

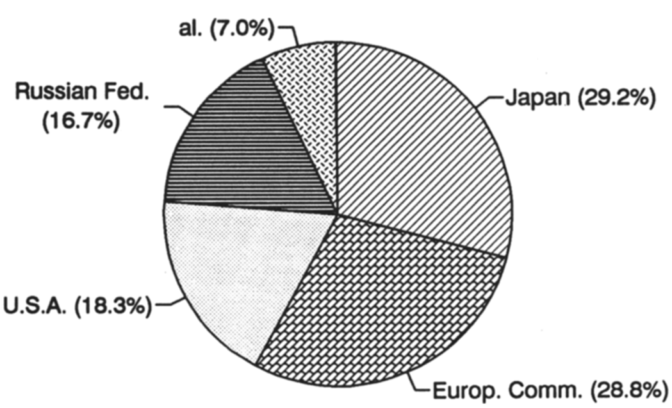
The European Community contributions came mainly from Germany, Italy, and The Netherlands, which sent 37, 28, and 14%, respectively, of the European Community total. Forschungszentrum Jülich (KFA) and Kernforschungszentrum Karlsruhe (KfK) equally represented most of the German papers. Contributions from Italy came mainly from ENEA and the Institute for Advanced Materials, Ispra site. The Netherlands was represented mainly by ECN, Petten and the Institute for Advanced Materials, Petten site.

Most of the contributions from the United States came from Pacific Northwest Laboratory (PNL) (23% of the United States total), Argonne National Laboratory (19%), Oak Ridge National Laboratory (ORNL) (14%), and the University of California at Los Angeles and at Santa Barbara (10%).

Of the papers that came from the former Soviet Union, 85% came from the Russian Federation. The remainder came equally from Ukraine and Latvia. The D. V. Efremov Institute and the A. A. Baikov Institute of Metallurgy of the Russian Academy of Science sent most of the papers.

Apart from the ITER partners, China was the best represented country at ICFRM-6 with nine contributed papers.

Figure 2 shows the classification according to the topic of the contributed papers. One of the main subjects was radiation damage, which also includes radiation effects, radiation facilities, and simulation of radiation effects. Steels were equally divided between austenitic and ferritic/martensitic. High-heat-flux materials included the divertor and the limiter materials, mainly beryllium, carbon, and molybdenum alloys. The effects of neutron irradiation and of plasma disruption were the most widely investigated subjects of this class of materials. Studies on tritium/hydrogen effects as well as tritium release were presented for carbon and doped carbon materials, beryllium, austenitic and martensitic/ferritic stainless steels, molybdenum alloys, vanadium alloys, tungsten, copper, titanium, and nickel as well as gallium liquid metal and some materials used as a tritium storage getter. Tritium release was also studied on ceramic breeders and liquid lithium. Breeding and neutron multiplier materials included lithium-lead eutectic and lithium ceramics (zirconate, orthosilicate, aluminate, and lithium oxide). Most of the papers on low-activation materials concerned austenitic and martensitic/ferritic steels, vanadium alloys, and silicon carbide (SiC) composites. Presentations on ceramic insulators and windows dealt mainly with the change of mechanical, physical, and dielectric properties of irradiated ceramic materials. Joining included papers on brazing of AISI 316L stainless steel with itself and with vanadium and a carbon-fiber composite (CFC)



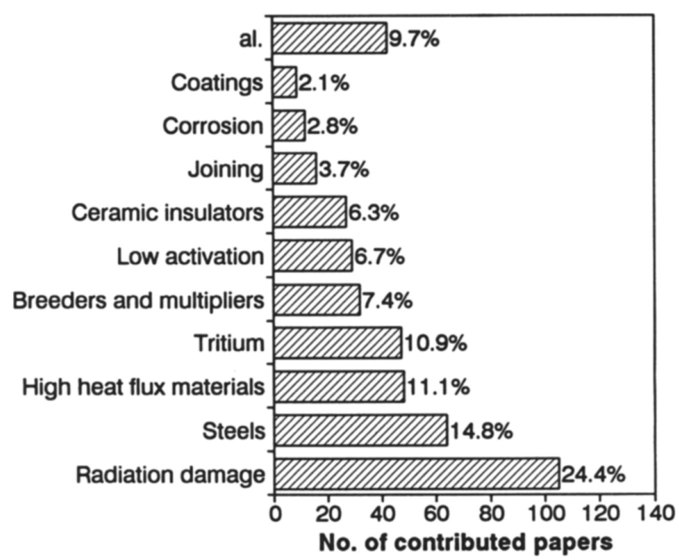


Fig. 2. Main subjects of the contributed papers.

and on brazing of a copper alloy. Diffusion bonding was investigated to join beryllium with copper, AISI 316L stainless steel, a molybdenum alloy, and CFC. The same technique was used to join dispersion-strengthened copper to AISI 316L stainless steel. Results of irradiated tungsten inert gas and electron beam welding on austenitic steels were also presented. The possible joining of a carbon-based material and a SiC composite was shown. Most of the papers on corrosion dealt with the compatibility of lithium-lead or lithium with different structural materials and insulating layers. Fewer contributed papers were submitted on the following topics: coatings and barriers, advanced materials (i.e., vanadium and titanium alloys and SiC composites), component fabrication and testing, thermal fatigue, sputtering, material requirements and engineering for near-term devices, and shielding materials.

During the conference, 18 invited papers were presented in plenary sessions during the 5 days of the conference.

After the opening talks of J. P. Contzen (Director General, Joint Research Centre), E. D. Hondros (Head, Institute for Advanced Materials), R. Andreani (Director, ENEA Fusion Department), and P. Schiller (Conference Chair, Institute for Advanced Materials), W. B. Gauster (ITER Joint Work Site) discussed the impact of material selection on the ITER design. He reviewed the ITER engineering design activity design principles aimed at giving an attractive image of fusion as a possible source of energy. Particular attention was therefore paid to design flexibility and reliability, reactor relevance, safety, cost, and waste disposal. These requirements should be pursued by a conservative extrapolation of the current understanding of tokamaks. Currently, the main design parameters are the following.

Nominal major radius (m)	7.75
Minor radius at midplane (m)	2.8
Toroidal field at nominal radius (T)	6
Maximum toroidal field ripple at plasma edge (%)	Less than ±2
Number of superconducting toroidal field coils	24
Divertor configuration	Single null
Burn time	1000 s at 25 MA
Fusion power	
Normal operation (MW)	≈1500
Extended operation (MW)	≈3000

The average neutron wall loading on the first wall is anticipated to be 1.5 MW/m^2, and the integrated wall loading at the end of the extended operation will be 3 MW·yr/m^2. Although the power sharing between the first wall and the divertor is still uncertain, an estimate of the heat flux on the first wall leads to 0.4 and 0.5 MW/m^2 for the average and the peak values, respectively. For the divertor, those figures become 1.5 and 3 to 5 MW/m^2.

The first wall/blanket design consists of canisters made of vanadium alloy whose plasma-facing surface is covered by a beryllium coating. They are protected by poloidal limiters located between each pair of canisters and made of beryllium. Liquid lithium acts as both breeder and coolant. This concept is very attractive because of its simplicity and the favorable thermomechanical properties of vanadium. One of the main issues to be addressed is the effectiveness of the self-healing AlN insulating coatings on the inner surface of the canisters aimed at reducing the pressure drops in the liquid-metal circulation system. If the insulating coating failed to be successful, lithium would become stagnant, and helium cooling would be necessary. Other questions to be investigated are the behavior of vanadium under irradiation, chemical compatibility, and tritium permeation. During the basic performance phase, there is no need for a breeding blanket; therefore, an alternative solution is also studied. This consists of canisters made of steel except the front side, which is made of copper. The coolant is water.

The divertor concept relies on the possibility of spreading the energy entering the divertor chamber onto the divertor side walls by means of a combination of charge-exchange and radiation. To obtain this, the divertor will utilize about one-fourth of the volume of the plasma chamber. Significant work is needed to study the physics of the concept, to investigate the erosion due to sputtering of the divertor materials, and to evaluate the thermal and mechanical stress limits. The divertor structure consists of a series of tubes shaped to form baffles. Several concepts are being studied. The coolant should be water if a shielding blanket is considered in the project or liquid metal or helium if a breeding blanket is foreseen.

The entire plasma-facing surface will be covered by a coating of beryllium. Its thermal conductivity does not decrease rapidly with neutron irradiation, and it has oxygen gettering properties. Moreover, this type of coating allows a lower tritium inventory and better plasma performance [no disruptions occur because of density limit; magnetohydrodynamic (MHD) disruptions are much slower; runaway electrons are strongly reduced].

The optimization of the ITER design to meet the requirements of performance, simplicity, safety, and cost is a major challenge of the whole project.

T. Shikama (Tohoku University) gave an overview of the irradiation effects on the electrical conductivity of ceramic insulators. Studies are carried out by means of X or gamma rays, electrons, and ions. The importance of performing neutron irradiation to have more reactor-relevant conditions for the investigation of the radiation-induced electrical degradation effect was pointed out.

D. S. Gelles (PNL) discussed the radiation effects in beryllium. Two detrimental mechanisms occur: displacement damage and transmutation. The former leads to hardening and embrittlement, and the latter generates helium thus causing swelling and embrittlement. The presentation reviewed the existing data on swelling of beryllium at different temperatures and different fast neutron fluences. At low temperature and low fluence, swelling increases linearly with fluence. But above 6.4×10^{22} n/cm^2, $E > 1$ MeV, the law becomes bilinear or quadratic. At a higher temperature (above 400°C), swelling dramatically increases with fluence and temperature. Data on changes in mechanical properties due to irradiation were also presented. At low temperature, strength increases with fluence until ductility approaches zero; then strength decreases. At higher irradiation temperatures, hardening reduces, and the strength behavior becomes more complex. All the tests performed so far have shown severe degradation in the mechanical properties with irradiation. However, a strong dependence on the beryllium fabrication process was found, and therefore, the results obtained so far may be unduly pessimistic if applied to future manufacturing procedures.

J. P. Quian (Southwestern Institute of Physics) presented fusion reactor material research in China. Austenitic and ferritic steels, with some modified grades to improve radiation resistance and induced activation, constitute the main options for structural materials. A certain interest, but with limited financial support, is also given to vanadium alloys thanks to their attractive properties. Ceramic materials could be considered in the future. Both ceramic (mainly lithium oxide and lithium aluminate) and liquid (i.e., liquid lithium and lithium-lead) breeders are studied. An experimental facility is under operation to investigate tritium production and release from lithium aluminate. Regarding liquid breeders, most efforts go into corrosion tests with structural materials and the study of MHD pressure drops. Neutron multiplication in beryllium and lead was studied experimentally, and results were compared with numerical calculations. The general trend is that a multiplication factor lower than that predicted is found in the experiments. Research on plasma-facing materials includes graphite, CFC composite, boron-doped carbon, and some refractory high-Z materials such as molybdenum alloys and tungsten. Simulation tests on carbon materials and coatings are in progress to investigate the erosion rate under heat loads due to plasma disruptions.

P. Fenici (Institute for Advanced Materials, Ispra Site) stressed the importance of ceramic matrix composites, i.e., SiC/SiC, in the development of materials for the demonstration reactor. The most attractive features of SiC/SiC are rapid decrease of the induced radioactivity and low afterheat, high operation temperature and thermal shock resistance, low weight of the structure, and relatively low atomic number. SiC composites are already used in the aerospace industry, and one can benefit from this existing experience. Recent results were reviewed. Preliminary tests have shown that the chemical vapor infiltration technique is effective in improving the leak tightness of liquid breeders. Lithium-lead has good compatibility with SiC/SiC, whereas ceramic breeders, especially orthosilicate, show higher reactivity. Data on irradiated materials are still scarce. Degradation of the fibers was observed with a reduction of their strength. However, helium implantation experiments have not shown dramatic degradation of SiC/SiC properties. The most interesting point that Fenici made was that the current industrial production of SiC/SiC composite already fulfills all the low-activation limits expected for a blanket structural material. Regarding the first wall, only the hands-on limit (25 μSv/h after 100 yr of decay) is exceeded; nevertheless, this limit can almost be achieved if isotopic tailoring of silicon is performed (^{28}Si replaced with ^{29}Si and ^{30}Si).

M. Akiba (JAERI) discussed the effects of plasma disruptions on structural and plasma-facing materials. His results concerning erosion depth are noteworthy. Simulation tests performed by means of an electron beam or laser on CFC materials have obtained erosion two to five times higher than that predicted. On the other hand, disruption simulations carried out by means of a plasma gun, and therefore much more reactor relevant, have shown erosion depths 15 to 100 times lower for the CFC and 2 to 10 times lower for steels as compared with similar electron beam/laser tests.

L. Boulanger (Commissariat à l'Energie Atomique) presented the microstructure and micromechanics of brazings under irradiation by ions. Two plates made of Type 316 stainless steel were joined using BNi7 as a brazing filler (76% nickel, 14% chromium, 10% phosphorus, and 0.2% iron, by weight). They were irradiated by nickel, chromium, helium, and neon ions accelerated by a Van de Graaff machine up to a damage level of 10 dpa. The microstructural evolution of phosphides (M_2P and M_3P, M = nickel, chromium, iron) was analyzed by transmission electron microscopy. Mechanical tests were also performed by a quadratic pyramid indenter (Vickers hardness test).

N. Yoshida (Kyushu University) discussed the role of transmission electron microscopy in the study of radiation damage. In particular, electron irradiation in a high-voltage electron microscope has the great advantage in that it can continuously follow the evolution of the damage during irradiation. Results of electron and neutron irradiated austenitic steels were reported at both low and high doses.

H. Trinkaus (KFA) discussed defect accumulation under cascade damage conditions. Two types of defect were described: self-interstitial atoms and vacancies. The former have a certain thermal stability, whereas the latter decay quickly. Because of this different thermal stability, an asymmetry occurs in their production (production bias). The basic physics of the production, stability, mobility, and interactions with other defects of self-interstitial atoms and vacancies was presented.

G. R. Odette (University of California, Santa Barbara) presented the mechanisms and the models of the ductile-to-brittle transition and the irradiation embrittlement of body-centered cubic metals such as ferritic/martensitic steels and vanadium alloys. These materials have a number of attractive properties for fusion applications, but under neutron irradiation, they become brittle. Fundamental theoretical and experimental approaches were described to investigate the open questions on ductile-to-brittle transition (e.g., identity of initiation sites; microcrack nucleation/propagation criteria; quasi-cleavage mechanisms; dependence of the "weakest link statistics" properties from temperature, stress, and strain; and intergranular brittle fracture modes).

V. K. Shamardin (Research Institute of Atomic Reactors) presented experimental results on swelling and mechanical properties of austenitic steels, based on the iron-chromium-manganese system, neutron irradiated in a temperature range of 320 to 800°C. The maximum damage level was in the range of 30 to 60 dpa according to different steel grades. Swelling proved to be appreciably lower than in iron-chromium-nickel steels: At 60 dpa, it never exceeded a few percent for temperatures between 400 and 530°C. A comparison of the mechanical properties of manganese steels with Type 316 stainless steel was also presented as well as an investigation on the change of the microstructure under irradiation.

I. M. Neklyudov (Kharkov Institute of Physics and Technology) presented the features of structure-phase transformations and the segregation process of austenitic and ferritic/martensitic steels irradiated with heavy ions and with neutrons in fast fission reactors. A detailed analysis was given of the influence of different factors on structure-phase transformation processes.

C. García-Rosales (Max-Planck-Institut für Plasmaphysik) described the erosion processes in plasma wall interaction. Both low-Z materials, like carbon, and high-Z, like tungsten, were considered taking into account the different erosion mechanisms according to the material temperature and the plasma edge temperature. Because of the high particle and energy fluxes on the limiter and the divertor, physical and chemical sputtering occurs even at low target temperatures for the carbon material. At higher temperatures (say above 1200°C), radiation-enhanced sublimation dominates. Boron or titanium doping of carbon practically suppresses chemical sputtering and shifts the onset of radiation sublimation toward higher temperatures. If the plasma edge temperature can be reliably controlled (below 15 eV), the use of tungsten would be preferable because of its higher threshold for sputtering and better redeposition factor. Regarding the first wall, the lower particle flux could allow the use of a coating of boron carbide, which, besides having a low atomic number and reduced chemical sputtering, has the potential of being able to be repaired *in situ* by plasma spray.

R. Matera (Institute for Advanced Materials, Ispra site) reported the results of thermal fatigue experiments carried out on a variety of specimens and prototypical first-wall mock-ups made of AISI 316L stainless steel. They were tested in the Thermal Fatigue Test Laboratory, Ispra, which was designed, built, and is now being used for the purpose of simulating as closely as possible the real thermal cycles of the next-step fusion reactor. A bare first wall without welded and brazed joints proved to be able to withstand without substantial damage the normal and off-normal heat loads expected for ITER. The maximum allowable number of fatigue cycles given by nuclear standard codes offers a wide safety margin. A reversed situation occurs in first-wall mock-ups with welded or brazed joints. Here, a premature failure of components was observed, and the design by analysis procedures may lead to a nonconservative estimate of the thermal fatigue lifetime. Therefore, in this case, the design by experiment seems to be mandatory.

E. A. Kenik (ORNL) discussed irradiation-assisted crack corrosion cracking for austenitic alloys in aqueous environments. He stressed the importance of radiation-induced segregation as one of the major factors influencing corrosion behavior. It causes chromium depletion at grain boundaries thus increasing the susceptibility of the material to localized corrosion. Also, other elements segregate at grain boundaries, but their effective role is not yet well understood. Another important factor in irradiation-assisted crack corrosion cracking is matrix hardening and loss of ductility. Water radiolysis and the subsequent production of aggressive compounds like H_2O_2 further contribute to increasing the corrosion under irradiation.

G. M. Kalinin (Research and Development Institute of Power Engineering) presented the combined action of neutron irradiation and hydrogen effects on austenitic steels. Samples were irradiated in a fission reactor, and then the hydrogen permeability and solubility were measured. An appreciable increase of permeability was found, and the higher it was the lower was the irradiation temperature. Hydrogen solubility decreases up to one order of magnitude at low irradiation temperature. Results of tritium production and release from lithium-lead eutectic under irradiation were also presented.

P. L. Andrew [Joint European Torus (JET) Joint Undertaking] reviewed tritium retention data for first-wall materials. Regarding stainless steels and Inconel, the maximum bulk hydrogen concentration in the tokamak is very low, i.e., $<10^{-4}$ hydrogen/metal. On the other hand, diffusivity is rather high and increases rapidly with temperature. This is an important concern as tritium can permeate through the containment material and escape into the coolant or into the external environment. Completely different behavior is found in graphite where the diffusivity is several orders of magnitude lower than all the relevant metals in all the temperature ranges of interest for fusion applications. Unfortunately, concentrations of hydrogen may reach values as high as 0.5 hydrogen/carbon. One of the main issues related to the use of graphite is codeposition of carbon and hydrogen atoms, which is significant especially for temperatures lower than a few hundred degrees. This phenomenon can generate an unacceptably high tritium inventory in the case of an extensive use of graphite. Beryllium shows a lower diffusivity of hydrogen when compared with most relevant metals, such as steels, tungsten, and vanadium. Data on hydrogen solubility are scarce but nevertheless show values from three to six orders of magnitude lower than steels. Refractory metals, like molybdenum and tungsten, show a very low hydrogen solubility, whereas diffusivity is higher than in steels. Hydrogen retention in vanadium alloys is extremely high and strongly depends on temperature and pressure. Below 180°C, hydride formation occurs, which causes severe embrittlement. The diffusivity is the highest for any materials of fusion interest.

M. Dalle Donne (KfK) presented the material problems and requirements related to the development of fusion blankets. At first, the main issues related to structural materials were discussed. High-temperature swelling for austenitic steels, low-temperature embrittlement and radiation hardening for martensitic/ferritic steel, and irradiation behavior and scarce technological experience on large-scale structures for vanadium alloys are the main concerns. Regarding the solid breeder blanket, the use of lithiated ceramics (lithium oxide, aluminate, zirconate, and orthosilicate) and beryllium as the neutron multiplier is envisaged. Most efforts investigate the irradiation effects on these materials. From this point of view, irradiation at much higher fluences is required to give relevant indications for a DEMO reactor. The liquid-metal blanket envisages the use of lithium or lithium-lead eutectic and coolants like water (only for lithium-lead), helium, or liquid metals. The main fields of research are the development of tritium barriers to decrease permeation, the development of insulators to reduce MHD pressure drop, and studies on compatibility and irradiation damage effects.

On October 1, ICFRM-6 closed. Almost 400 participants attended the conference. The appointment for all fusion material researchers for 1995 is in Moscow, where ICFRM-7 will be held.

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